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

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 46
License No. NPF-63

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

   A. The application for amendment by Carolina Power & Light Company, (the licensee), dated August 27, 1993, as supplemented February 17, 1994, 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. NPF-63 is hereby amended to read as follows:

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 46 , are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

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: March 3, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 46

FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

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
2-2	2-2
2-8	2-8
2-10	2-10
B 2-1	B 2-1
B 2-1a	B 2-1a
B 2-4	B 2-4
B 2-5	B 2-5
3/4 2-14	3/4 2-14
3/4 9-2	3/4 9-2
---	3/4 9-2a
B 3/4 1-2	B 3/4 1-2
B 3/4 2-6	B 3/4 2-6
5-6	5-6

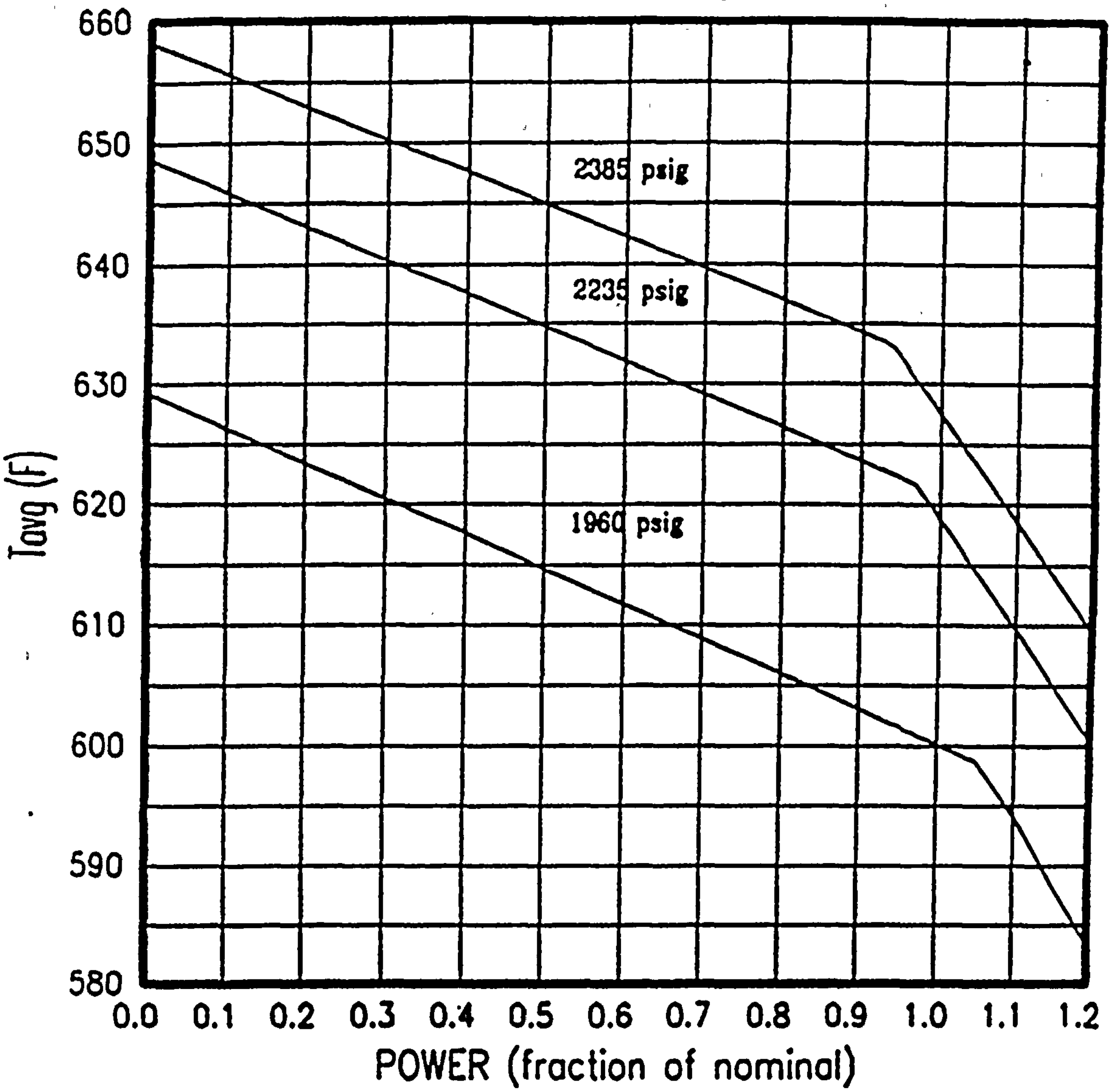

FIGURE 2.1-1
REACTOR CORE SAFETY LIMITS - THREE LOOPS IN OPERATION

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: (Continued)

T	=	Average temperature, °F;
$\frac{1}{1+\tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s;
T′	≤	580.8°F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	=	0.001072/psig;
P	=	Pressurizer pressure, psig;
P′	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $q_t - q_b$ between -21.6% and +12.0%, f_1 (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds -21.6%, the ΔT Trip Setpoint shall be automatically reduced by 2.36% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + 12.0%, the ΔT Trip Setpoint shall be automatically reduced by 1.57% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.1% ΔT span.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 3: (Continued)

K_8	=	0.002/°F for $T > T''$ and $K_8 = 0$ for $T \le T''$,
T	=	As defined in Note 1,
T''	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\le$ 580.8°F),
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.3% ΔT span.

NOTE 5: The sensor error for temperature is 1.9 and 1.1 for pressure.

NOTE 6: The sensor error for steam flow is 0.9, for feed flow is 1.5, and for steam pressure is 0.75.

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (in this application, the HTP correlation for Siemens Fuel and the WRB-1 or WRB-2 correlation for Westinghouse Fuel). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the design DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

2.1.1 REACTOR CORE (Continued)

These curves are based on an enthalpy hot channel factor, $F_{\Delta H}$, specified in the CORE OPERATING LIMITS REPORT (COLR) and a limiting axial power shape. An allowance is included for an increase in calculated $F_{\Delta H}$ at reduced power based on the expression:

$$F_{\Delta H} = F_{\Delta H}^{RTP} \ [1 + PF_{\Delta H} \ (1-P)]$$

Where P is the fraction of RATED THERMAL POWER,

$F_{\Delta H}^{RTP}$ = $F_{\Delta H}$ limit at RATED THERMAL POWER specified in the COLR, and

$PF_{\Delta H}$ = Power Factor Multiplier for $F_{\Delta H}$ specified in the COLR.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the $f_1(\Delta I)$ function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

Power Range, Neutron Flux (Continued)

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to transport to and response time of the temperature detectors (about 4 seconds), and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for transport to and response time of the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for transport to and response time of the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded.

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Reactor Coolant Flow

The Reactor Coolant Low Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90.5% of nominal full loop flow. Above P-8

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the following limits:

a. Indicated Reactor Coolant System T_{avg} ≤ 586.1°F after addition for instrument uncertainty, and

b. Indicated Pressurizer Pressure ≥ 2185 psig* after subtraction for instrument uncertainty.

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its indicated limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 Each of the parameters shown in Specification 3.2.5 shall be verified to be within its limit at least once per 12 hours.

* This limit is not applicable during either a Thermal Power Ramp in excess of ±5% Rated Thermal Power per minute or a Thermal Power step change in excess of ±10% Rated Thermal Power.

TABLE 3.9-1

ADMINISTRATIVE CONTROLS TO PREVENT DILUTION DURING REFUELING

CP&L VALVE NO. (Ebasco Valve No.)	DESCRIPTION	REQUIRED POSITION
1CS-149 (CS-D121SN)	Reactor Makeup Water to CVCS Makeup Control System	Lock closed; may be opened to permit makeup to Refueling Water Storage Tank provided valves 1CS-156 and 1CS-155 are maintained closed with their main control board control switches in "shut" position, and manual valves 1CS-274, 1CS-265 and 1CS-287 are locked closed.
1CS-510 (CS-D631SN)	Boric Acid Batch Tank Outlet	Locked closed; may be opened provided the boron concentration of the boric acid batch tank $\geq$ the greater of 2000 ppm or the boron concentration required to maintain K_{eff} less than or equal to 0.95, as specified in the COLR and valve 1CS-503 is closed.
1CS-503 (CS-D251)	Reactor Makeup Water to Boric Acid Batch Tank	Lock closed, may be opened provided valve 1CS-510 is closed.
1CS-93 (CS-D51SN)	Resin Sluice to CVCS Demineralizers	Lock closed.
1CS-320 (CS-D641SN)	Boron Recycle Evaporator Feed Pump to Charging/SI Pumps	Lock closed.

TABLE 3.9-1 (Continued)

ADMINISTRATIVE CONTROLS
TO PREVENT DILUTION DURING REFUELING

CP&L VALVE NO. (Ebasco Valve No.)	DESCRIPTION	REQUIRED POSITION
1CS-570 (CS-D575SN)	CVCS Letdown to Boron Thermal Regeneration System	Closed with main control board control switch in "shut" position, and BTRS function selector switch maintained in "off" position; no lock required.
1CS-670 (CS-D599SN)	Reactor Makeup Water to Boron Thermal Regeneration System	Lock closed.
1CS-649 (CS-D198SN)	Resin Sluice to BTRS Demineralizers	Lock closed.
1CS-98 (CS-D740SN)	Boron Thermal Regeneration System Bypass	Opened with main control board control switch maintained in "open" position; no lock required.

BASES

MODERATOR TEMPERATURE COEFFICIENT (Continued)

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging/safety injection pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide the required SHUTDOWN MARGIN as defined by Specification 3/4.1.1.2 after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at BOL

QUADRANT POWER TILT RATIO (Continued)

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The preferred sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8. If other locations must be used, a special report to NRC should be submitted within 30 days in accordance with 10CFR50.4.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR that is equal to or greater than the design DNBR value throughout each analyzed transient. The indicated T_{avg} value and the indicated pressurizer pressure value are compared to analytical limits of 586.1°F and 2185 psig, respectively, after an allowance for measurement uncertainty is included.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 45.0 psig and a peak air temperature of 380°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly normally containing 264 fuel rods clad with Zircaloy-4 except that limited substitution of fuel rods by filler rods consisting of Zircaloy-4, stainless steel, or by vacancies may be made in fuel assemblies if justified by a cycle-specific evaluation. Should more than a total of 30 fuel rods or more than 10 fuel rods in any one assembly be replaced per refueling a Special Report describing the number of rods replaced will be submitted to the Commission, pursuant to Specification 6.9.2, within 30 days after cycle startup. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of 3.5 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 5.0 weight percent U-235. Fuel with enrichments greater than 4.20 weight percent U-235 shall contain sufficient integral burnable absorbers such that the requirement of Specification 5.6.1.a.2 is met.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 52 shutdown and control rod assemblies. The shutdown and rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80% silver, 15% indium, and 5% cadmium, or 95% hafnium with the remainder zirconium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 9410 ± 100 cubic feet at a nominal T_{avg} of 580.8°F.