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

CAROLINA POWER AND LIGHT COMPANY

DOCKET NO. 50-261

H. B. ROBINSON STEAM ELECTRIC PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 43
License No. DPR-23

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

A. The application for amendment by Carolina Power and Light Company (the licensee) dated April 18, 1979, as supplemented August 8, 1979, 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 3.B. of Facility Operating License No. DPR-23 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 43 , 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

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 26, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 43

FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain verical lines indicating the area of change.

Remove	Replace
2.1-5	2.1-5
2.1-6	2.1-6
3.1-11	3.1-11
3.1-12	3.1-12
3.10-2	3.10-2
3.10-20	3.10-20
3.10-21	3.10-21
5.3-1	5.3-1
6.9-8	6.9-8

The safety limit curves given in Figures 2.1-1 and 2.1-2 are for constant flow conditions. These curves would not be applicable in the case of a loss of flow transient. The evaluation of such an event would be based upon the analysis presented in Section 14.1 of the FSAR.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure, and thermal power level that would result in a DNB ratio of less than 1.30[3] based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 575.4°F, and a steady state nominal operating pressure of 2235 psig. Allowances are made in initial conditions assumed for transient analyses for steady state errors of +2% in power, +4°F in Reactor Coolant System average temperature, and ±30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 percent less than the value at nominal full power operating conditions. The steady state nominal operating parameters and allowances for steady state errors given above are also applicable for two loop operation except that the steady state nominal operating power level is less than or equal to 45%.

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References

(1) FSAR Section 3.2.2
(2) FSAR Section 14.1.3
(3) FSAR Section 7.2.1
(4) WCAP-8243, "H. B. Robinson Unit 2 - Justification for Operation at 2300 MWt, December, 1973.

3.1.3 Minimum Conditions for Criticality

3.1.3.1 Except during low power physics tests, the reactor shall not be made critical at any temperature, above which the moderator temperature coefficient is greater than:

a) +2.0 pcm/°F at less than 50% of rated power, or

b) +2.0 pcm/°F at 50% of rated power and linearly decreasing to 0 pcm/°F at rated power.

3.1.3.2 In no case shall the reactor be made critical above and to the left of the criticality limit shown on Figure 3.1-1.

3.1.3.3 When the reactor coolant temperature is in a range where the moderator temperature coefficient is greater than as specificed in 3.1.3.1 above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least 1% until normal water level is established in the pressurizer.

Basis

During the early part of fuel cycle, the moderator temperature coefficient may be slightly positive at low power levels. The moderator coefficient at low temperatures or powers will be most positive at the beginning of the fuel cycle, when the boron concentration in the coolant is the greatest. At all times, the moderator coefficient is calculated to be negative in the high power operating range, and after a very brief period of power operation, the coefficient will be negative in all circumstances due to the reduced boron concentration as Xenon and fission products build into the core. The requirement that the reactor is not to be made critical when the moderator coefficient is more positive than as specified in 3.1.3.1 above has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase of moderator temperature or decrease of coolant pressure. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient[2] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The heatup curve of Figure 3.1-1 includes criticality limits which are required by 10 CFR Part 50, Appendix G, Paragraph IV.A.2.c. Whenever the core is critical, additional safety margins above those specified by the ASME Code Appendix G methods, are imposed. The core may be critical at temperatures equal to or above the minimum temperature for the inservice hydrostatic pressure tests as calculated by ASME Code Appendix G methods, and an additional safety margin of 40°F must be maintained above the applicable heatup curve at all times.

If the specified shutdown margin is maintained (Section 3.10), there is no possibility of an accidental criticality as a result of an increase of moderator temperature or a decrease of coolant pressure.(1)

The requirement for bubble formation in the pressurizer when the reactor has passed the threshold of one percent subcriticality will assure that the Reactor Coolant System will not be solid when criticality is achieved.

References

(1) FSAR Table 3.2.1-1
(2) FSAR Figure 3.2.1-10

3.10.1.5 Except for physics tests, if a full-length control rod is more than 15 inches out of alignment with its bank, then within two hours:

a. Correct the situation, or

b. Determine by measurement the hot channel factors and apply Specification 3.10.2.1, or

c. Limit power to 70 percent of rated power for three-loop operation.

3.10.1.6 Insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-2 must be maintained except for the low power physics test to measure control rod worth and shutdown margin. For this test the reactor may be critical with all but one full length control rod inserted.

3.10.2 <u>Power Distribution Limits</u>

3.10.2.1 At all times except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

$F_Q(Z) \leq (2.20/P) \times K(Z)$ for $P > .5$

$F_Q(Z) < (4.40) \times K(Z)$ for $P \leq .5$

$F^N_{\Delta H} < 1.55\ (1 + 0.2(1-P))$

where P is the fraction of licensed power at which the core is operating, K(Z) is based on the function given in Figure 3.10-3, and Z is the core height location of F_Q.

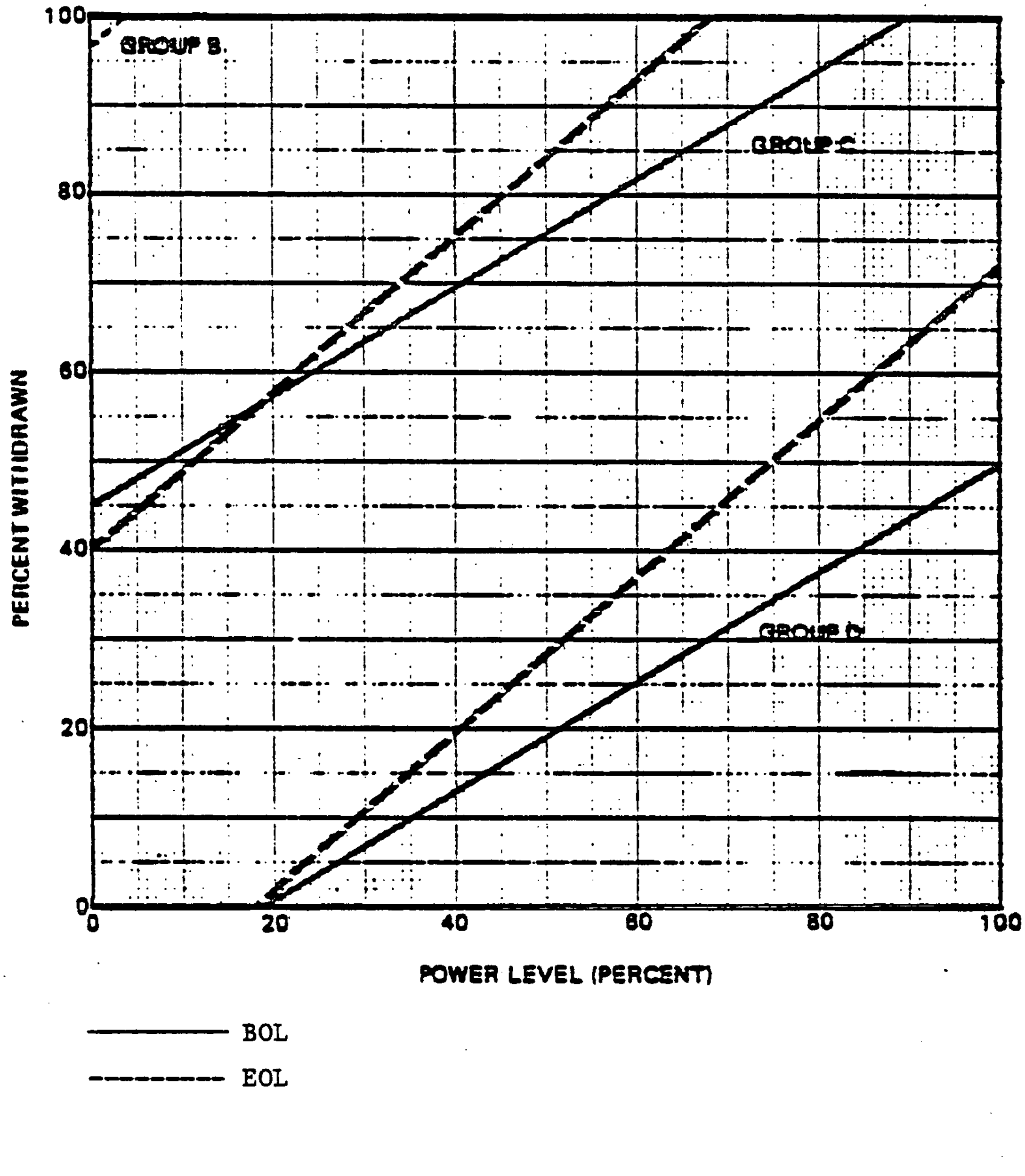

Figure 3.10-1 CONTROL GROUP INSERTION LIMITS FOR THREE LOOP OPERATION

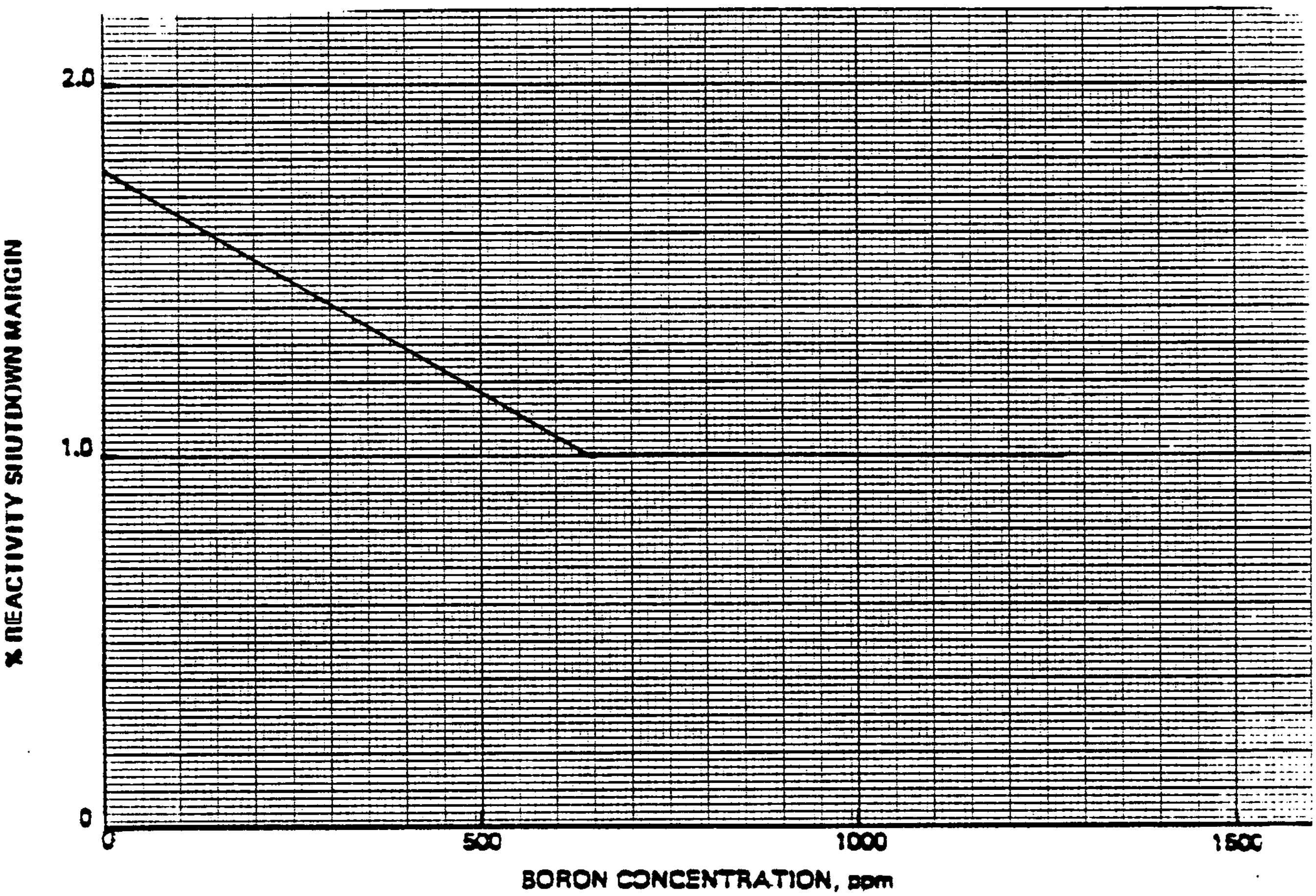

Figure 3.10—2 SHUTDOWN MARGIN VS BORON CONCENTRATION

5.3 REACTOR

5.3.1 Reactor Core

5.3.1.1 The reactor core contains approximately 71 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy - 4 tubing to form fuel rods. The fuel rods in the core are all pre-pressurized fuel rods. The reactor core is made up of 157 fuel assemblies. Each fuel assembly contains 204 fuel rods.(1)

5.3.1.2 Deleted

5.3.1.3 Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will be no more than 3.5 weight percent of U-235.

Deleted

5.3.1.4 Deleted

c.	Deleted		
d.	Inservice Inspection	4.2	After five years of operation
e.	Containment Sample Tendon Surveillance	4.4	Upon completion of the inspection at 5 and 25 years of operation
f.	Post-operational Containment Structural Test	4.4	Upon completion of the test at 3 and 20 years of operation
g.	Fire Protection System	3.14	As specified by limiting condition for operation.
h.	Overpressure Protection System Operation	3.1.2.1e	Within 30 days of operation.