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Docket No. 50-261

Carolina Power & Light Company
 ATTN: Mr. J. A. Jones
 Senior Vice President
 336 Fayetteville Street
 Raleigh, North Carolina 27602

Gentlemen:

The Commission has issued the enclosed Amendment No. 13 to Facility License No. DPR-23 for the H. B. Robinson Unit 2 facility. This amendment includes Change No. 38 to the Technical Specifications, and is in response to your request dated October 2, 1974 as supplemented March 14, 1975, April 18, 1975, June 20, 1975, and July 24, 1975.

This amendment (1) revises the operating limits in the Technical Specifications based upon an acceptable evaluation model that conforms to the requirements of 10 CFR § 50.46, and (2) terminates restrictions imposed on the facility by the Commission's December 27, 1974 Order for Modification of License, and imposes instead, limitations established in accordance with 10 CFR § 50.46.

The Commission's staff has evaluated the potential for environmental impact associated with operation of the facility in the proposed manner. From this evaluation, the staff has determined that there will be no change in effluent types or total amounts, no increase in authorized power level, and no significant environmental impact attributable to the proposed action. Having made this determination, the Commission has further concluded pursuant to 10 CFR Part 51, § 51.5(c)(1) that no environmental impact statement need be prepared for this action. Copies of the related Negative Declaration and supporting Environmental Impact Appraisal are enclosed. As required by Part 51, the Negative Declaration is being filed with the Office of the Federal Register for publication.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Robert W. Reid, Chief
 Operating Reactors Branch #4
 Division of Reactor Licensing

Enclosures:

| | | | | | |
|---------|-------------------------------------|---------------------|----------|-----------------|------------|
| 1. | Amendment No. 13 | | | | |
| OFFICE | 2 -> Negative Declaration | ORB#4 RIngram/dg | OELD | ORB#4 RWREid | RL/AD:OR's |
| SURNAME | 3 -> Environmental Impact Appraisal | IDNBridges/dg | | | KRGoeller |
| DATE | 4. Safety Evaluation | 10/10/75 | 10/17/75 | 10/10/75 | 10/17/75 |
| | 5 -> Federal Register Notice | | | | |

Carolina Power & Light Company - 2 -

cc: w/enclosures

OCT 17 1975

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-261

H. B. ROBINSON UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 13
License No. DPR-23

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Carolina Power & Light Company (the licensee) dated October 2, 1974, as supplemented 3/14/75, 4/18/75, 6/20/75 and 7/24/75, 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; and
 - 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.
2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Facility License No. DPR-23 is hereby amended to read as follows:



" B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 38 ."

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Reactor Licensing

Attachment:
Change No. 38 to the
Technical Specifications

Date of Issuance:

OCT 17 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 13
CHANGE NO. 38 TO THE TECHNICAL SPECIFICATIONS
FACILITY OPERATING LICENSE NO. DPR-23
DOCKET NO. 50-261

Revise Appendix A as follows:

Remove Pages

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3.3-9
3.10-1 - 3.10-11

Insert Pages

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3.3 EMERGENCY CORE COOLING SYSTEM, AUXILIARY COOLING SYSTEMS,
AIR RECIRCULATION FAN COOLERS, CONTAINMENT SPRAY, POST
ACCIDENT CONTAINMENT VENTING SYSTEM, AND ISOLATION SEAL
WATER SYSTEM

Applicability:

Applies to the operating status of the Emergency Core Cooling System, Auxiliary Cooling Systems, Air Recirculation Fan Coolers, Containment Spray, Post Accident Containment Venting System, and Isolation Seal Water System.

Objective:

To define those limiting conditions for operation that are necessary: (1) to remove decay heat from the core in emergency or normal shutdown situations, (2) to remove heat from containment and critical components in normal operating and emergency situations, and (3) to remove airborne iodine from the containment atmosphere following a postulated Design Basis Accident.

Specification

3.3.1 Safety Injection and Residual Heat Removal Systems

- 3.3.1.1 The reactor shall not be made critical, except for low temperature physics tests, unless the following conditions are met:
- a. The refueling water tank contains not less than 300,000 gal. of water with a boron concentration of at least 1950 ppm.
 - b. The boron injection tank contains not less than 900 gal. of 20,000 to 22,500 ppm boron solution at a temperature of at least 145°F. Two channels of heat tracing shall be available for the flow path.

- c. Each accumulator is pressurized to at least 600 psig and contains at least 825 ft³ and no more than 841 ft³ of water with a boron concentration of at least 1950 ppm. No accumulator may be isolated.
- d. Three safety injection pumps are operable.
- e. Two residual heat removal pumps are operable.
- f. Two residual heat exchangers are operable.
- g. All essential features including valves, interlocks, and piping associated with the above components are operable.
- h. During conditions of operation with reactor coolant pressure in excess of 1000 psig the A.C. power shall be removed from the following motor operated valves with the valve in the specified position:

| <u>Valves</u> | <u>Position</u> |
|----------------|-----------------|
| MOV 862 A&B | Open |
| MOV 864 A&B | Open |
| MOV 865 A,B,&C | Open |
| MOV 878 A&B | Open |
| MOV 863 A&B | Closed |
| MOV 866 A&B | Closed |

- i. During conditions of operation with reactor coolant pressure in excess of 1000 psig the air supply to air operated valves 605 and 758 shall be shut off with valves in the closed position.
- j. Power operation with less than three loops in service is prohibited.

3.3.1.2 During power operation, the requirements of 3.3.1.1 may be modified to allow any one of the following components to be inoperable. If the system is not restored to meet the requirements of 3.3.1.1 within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.1.1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

- a. One accumulator may be isolated for a period not to exceed 4 hours.
- b. If one safety injection pump becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours, provided the remaining two safety injection pumps are demonstrated to be operable prior to initiating repairs.
- c. If one residual heat removal pump becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours, provided the other residual heat removal pump is demonstrated to be operable prior to initiating repairs.
- d. If one residual heat exchanger becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours.
- e. If any one flow path including valves of the safety injection or residual heat removal system is found to be inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours, provided the other flow path(s) are demonstrated to be operable prior to initiating repairs. The hot leg injection paths of the Safety Injection System, including valves, are not subject to the requirements of this specification.
- f. If the boron concentration in the boron injection tank falls below 20,000 ppm, and is greater than 15,000 ppm, the reactor may remain in operation for a period not to exceed 24 hours. If the concentration is less than 15,000 ppm, the reactor will be placed in the hot shutdown condition utilizing normal operating procedures.
- g. Power or air supply may be restored to any valve referenced in 3.3.1.1.h. and 3.3.1.1.i. for the purpose of valve testing or maintenance providing no more than one valve has power restored and provided that testing and maintenance is completed and power removed within 24 hours except for accumulator isolation valves (MOV 865 A,B,&C) which will have this time period limited to 4 hours.

3.3.1.3 When the reactor is in the hot shutdown condition, the requirements of 3.3.1.1 and 3.3.1.2 shall be met. Except that the accumulators may be isolated, and in addition, any one component as defined in 3.3.1.2 may be inoperable for a period equal to the time period specified in the subparagraphs of 3.3.1.2 plus 48 hours, after which the plant shall be placed in the cold shutdown condition utilizing normal operating procedures.

3.3.2 Containment Cooling and Iodine Removal Systems

3.3.2.1 The reactor shall not be made critical, except for low temperature physics tests, unless the following conditions are met:

- a. The spray additive tank contains not less than 2505 gal. of solution with a sodium hydroxide concentration of not less than 30% by weight.
- b. Two containment spray pumps are operable.
- c. Four fan cooler units are operable.
- d. All essential features, including valves, controls, dampers, and piping associated with the above components are operable.
- e. The system which automatically initiates the sodium hydroxide addition to the containment spray simultaneously to the actuation of the containment spray is operable.

3.3.2.2 During power operation, the requirements of 3.3.2.1 may be modified to allow any one of the following components to be inoperable. If the system is not restored to meet the requirements of 3.3.2.1 within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.2.1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

- a. If one fan cooler unit or the flow path for a fan cooler unit becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours, provided both containment spray pumps are demonstrated to be operable.

sodium hydroxide addition, are capable of being operated on emergency power with one diesel generator inoperable. If one diesel generator is operating or another source of emergency power is available, the other containment spray pump, with sodium hydroxide addition, can be operated to provide iodine removal in excess of the minimum requirements. Adequate power for operation of the redundant containment heat removal system (i.e. four fan-cooler units and two containment spray pumps) is also assured in this case.

The Component Cooling System is different from the other systems discussed above in that the components are so located in the auxiliary buildings as to be accessible for repair after a loss-of-coolant accident. (4)

A total of four service water pumps are installed, a minimum of two of which are required to operate during the postulated loss-of-coolant accident. (5)

A minimum of 300,000 gallons of water will be maintained in the refueling water storage tank. This requirement is based on recirculation mode operation which may start with a depth of 1.5 feet on the containment floor. This depth of water is equivalent to the amount of water in the primary system plus 60% of the refueling water storage tank, approximately 215,000 gallons of water at 263°F. (1)

Analysis have shown that the consequences of the steam line break accident are successfully mitigated with a boron injection tank boron concentration of 15,000 ppm or greater. (9) The specification of 20,000 ppm as a minimum concentration is maintained to provide additional margin in the event of such an accident.

The post accident containment venting system is designed with redundant air supply and vent paths. The valves in the system will be demonstrated to be operable prior to criticality. Testing of the air supply system is not required because of the long lead time between an accident and the required operation of the venting system. This period of time will permit maintenance effort, if required. The efficiency of the filters in each vent path was not used in this safety analysis; therefore, testing of these filters is not required. (6)

The Isolation Seal Water System provides a reliable means for injecting seal water between the seats and stem packing of the globe and double disc types of isolation valves and into the piping between closed diaphragm type isolation valves. (7)

The minimum 825 ft³ and maximum 841 ft³ of water in the accumulators correspond to an instrument reading of 61.5% and 80.4% of instrument span, respectively. | 3

References

- | | |
|---|--------------------------|
| (1) FSAR Section 6.2 | (4) FSAR Section 9.3 |
| (2) FSAR Section 6.3 | (5) FSAR Section 9.6.2 |
| (3) FSAR Section 14.3.5 | (6) FSAR - Appendix 6B |
| | (7) FSAR - Section 5.2.2 |
| (8) CP&L report and supplemental letters of September 29, November 5, December 8, 1971, and March 20, 1972. | |
| (9) CP&L letter of August 30, 1974. | |

REQUIRED SHUTDOWN MARGINS, CONTROL ROD, AND POWER DISTRIBUTION LIMITS

Applicability:

Applies to the required shutdown margins, operation of the control rods, and power distribution limits.

Objective:

To ensure (1) core subcriticality after a reactor trip and during normal shutdown conditions, (2) limited potential reactivity insertions from a hypothetical control rod ejection, and (3) an acceptable core power distribution during power operation.

Specifications:

3.10.1 Full Length Control Rod Insertion Limits

3.10.1.1 (Deleted by Change No. 21 issued 7/6/73)

3.10.1.2 When the reactor is critical, except for physics tests and full length control rod exercises, the shutdown control rods shall be fully withdrawn.

3.10.1.3 When the reactor is critical, except for physics tests and full length control rod exercises, the control rods shall be no further inserted than the limits shown by the solid lines on Figure 3.10-1 for 3 loop operation.

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3.10.1.4 After 50% of the second and subsequent cycles as defined by burnup, the limits shall be adjusted as a linear function of burnup toward the end-of-core life values as shown by the dotted lines on Figure 3.10-1.

3.10.1.5 Except for physics tests, if a part-length or 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

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c. Limit power to 70% of rated power for 3 loop operation.

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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 and part length rods fully withdrawn.

3.10.2 Power Distribution Limits

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:

$$F_Q(Z) \leq (2.30/P) \times K(Z) \text{ for } P > .5$$

$$F_Q(Z) < (4.60) \times K(Z) \text{ for } P \leq .5$$

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

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

3.10.2.2 If either measured hot channel factor exceeds these values the reactor power shall be reduced so as not to exceed a fraction of the design value equal to the ratio of the F_Q^H or $F_{\Delta H}^N$ limit to measured value, whichever is less, and the high neutron flux trip setpoint shall be reduced by the same ratio. If subsequent incore mapping cannot, within a 24 hour period, demonstrate that the hot channel factors are met, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

3.10.2.3 Following initial loading and at regular monthly intervals thereafter, power distribution maps using the movable detector system, shall be made to confirm that the hot channel factor limits of specification 3.10.2.1 are satisfied. For the purpose of this confirmation:

- a. The measurement of total peaking factor, F_Q^{Meas} , shall be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error.
- b. The measurement of enthalpy rise hot channel factor, $F_{\Delta H}^N$, shall be increased by four percent to account for measurement error.

3.10.2.4 The reference equilibrium indicated axial flux difference for each excore channel as a function of power level (called the target flux difference) shall be measured at least once per effective full power quarter. If the axial flux difference has not been measured in the last effective full power month, the target flux difference must be updated monthly by linear interpolation using the most recent measured value and the value predicted for the end of the cycle life.

3.10.2.5 The indicated axial flux difference shall be considered outside of the limits of sections 3.10.2.6 through 3.10.2.9 when more than one of the operable excore channels are indicating the axial flux difference to be outside a limit.

3.10.2.6 Except during physics tests, during excore detector calibration and except as modified by 3.10.2.7 through 3.10.2.9 below, the indicated axial flux difference shall be maintained within a $\pm 5\%$ band about the target flux difference (defines the target band on axial flux difference).

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- 3.10.2.7 At a power level greater than 90 percent of rated power, if the indicated axial flux difference deviates from its target band, the flux difference shall be returned to the target band immediately or reactor power shall be reduced to a level no greater than 90 percent of rated power.
- 3.10.2.8 At a power level no greater than 90 percent of rated power,
- a. The indicated axial flux difference may deviate from its +5% target band for a maximum of one hour (cumulative) in any 24 hour period provided the flux difference does not exceed an envelope bounded by -11 percent and +11% percent at 90% power and increasing by -1% and +1% for each 2% of rated power below 90%. If the cumulative time exceeds one hour, then the reactor power shall be reduced immediately to no greater than 50% power and the high neutron flux setpoint reduced to no greater than 55% of rated power.
 - b. A power increase to a level greater than 90% of rated power is contingent upon the indicated axial flux difference being within its target band.
- 3.10.2.9 At a power level no greater than 50 percent of rated power,
- a. The indicated axial flux difference may deviate from its target band.
 - b. A power increase to a level greater than 50% of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than two hours (cumulative) out of the preceding 24 hour period.

One half of the time the indicated axial flux difference is out of its target band up to 50% of rated power is to be counted as contributing to the one hour cumulative maximum the flux difference may deviate from its target band at a power level less than or equal to 90% of rated power.

3.10.2.10 Alarms shall normally be used to indicate non-conformance with the flux difference requirement of 3.10.2.7 or the flux difference-time requirement of 3.10.2.8.a. If the alarms are temporarily out of service, the axial flux difference shall be logged, and conformance with the limits assessed, every hour for the first 24 hours, and half-hourly thereafter. The requirement for alarms becomes effective December 1, 1975.

3.10.3 Quadrant Power Tilt Limits

3.10.3.1 Except for physics tests and during power increases below 50% of full power, whenever the indicated quadrant power tilt ratio exceeds 1.02, the tilt condition shall be eliminated within two hours or the following actions shall be taken:

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- a. Restrict core power level and reset the power range high flux setpoint to be less two percent of rated values for every percent of indicated power tilt ratio exceeding 1.0, and
- b. If the tilt condition is not eliminated after 24 hours, the power range high flux setpoint shall be reset to 55% of allowed power. Subsequent reactor operation would be permitted up to 50% power for the purpose of measurement and testing to identify the cause of the tilt condition.

3.10.3.2 Except for low power physics tests, if the indicated quadrant tilt exceeds 1.09 and there is simultaneous indication of a misaligned rod:

- a. The core power level shall be reduced by 2% of rated values for every 1% of indicated power tilt exceeding 1.0, and
- b. If the tilt condition is not eliminated within two hours, the reactor shall be brought to a hot shutdown condition.
- c. After correction of the misaligned rod, reactor operation will be permitted to 50% power until the indicated quadrant tilt falls below 1.09.

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3.10.3.3 If the indicated quadrant tilt exceeds 1.09 and there is not simultaneous indication of rod misalignment, except as stated in Specification 3.10.3.2.c, the reactor shall immediately be brought to a hot shutdown condition.

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3.10.4 Rod Drop Time

3.10.4.1 The drop time of each control rod shall be not greater than 1.8 seconds at full flow and operating temperature from the beginning of rod motion to dashpot entry.

3.10.5 Part Length Control Rod Banks

3.10.5.1 The eight (8) part length control rods shall be configured under administrative control into one of the following part length rod configurations.

- a. Four part length rod occupying core positions K-6, K-10, F-6, and F-10 shall constitute a part length control rod bank, hereafter designated bank P-1.

- b. Four part length rods occupying core positions P-3, H-2, H-14, and B-8 shall constitute a part length control bank, hereafter designated part length bank P-2.
- c. Combined Banks P-1 and P-2, hereafter designated bank P-3.

3.10.5.2 The part length control rods will not be inserted. They will remain in the fully withdrawn position except for physics tests and for axial offset calibration which will be performed at 75% of permitted power or less.

3.10.6 Inoperable Full Length and Part Length Control Rods

3.10.6.1 A full length or part length control rod shall be deemed inoperable if (a) the rod is misaligned by more than 15 inches with its bank, (b) if the rod cannot be moved by its drive mechanism, or (c) if its rod drop time is not met in the case of a full length rod.

3.10.6.2 No more than one inoperable control rod shall be permitted during power operation. This requirement does not apply to part length rods when they are fully withdrawn from the core.

3.10.6.3 If a full length control rod cannot be moved by its mechanism, boron concentration shall be changed to compensate for the withdrawn worth of the inoperable rod such that shutdown margin equal to or greater than shown on Figure 3.10-2 results.

3.10.7 Power Ramp Rate Limits

3.10.7.1 During the return to power following a shutdown where fuel assemblies have been handled (e.g. refueling, inspection), the rate of reactor power increase shall be limited to 3% of full power in an hour between 20% and 100% of full power. This ramp rate requirement applies during the initial startup and may apply during subsequent power increases depending on the maximum power level achieved and length of operation at that power level. Specifically, this requirement can be removed for reactor power levels below a power level P ($20\% < P \leq 100\%$) provided that the plant has operated at or above power level P for at least 72 cumulative hours out of any 7-day operating period following the shutdown.

3.10.7.2 The rate of reactor power increases above the highest power level sustained for at least 72 cumulative hours during the preceding 30 cumulative days of reactor power operation shall be limited to 3% of full power in an hour. Alternatively, reactor power increase can be accomplished by a single step increase less than or equal to 10% of full power followed by a maximum ramp rate of 3% of full power in an hour beginning 3 hours after the step increase.

3.10.8 Required Shutdown Margins

- 3.10.8.1 When the reactor is in the hot shutdown condition, the shutdown margin shall be at least that shown in Figure 3.10-2.
- 3.10.8.2 When the reactor is in the cold shutdown condition, the shutdown margin shall be at least 1% $\Delta k/k$.
- 3.10.8.3 When the reactor is in the refueling operation mode, the shutdown margin shall be at least 10% $\Delta k/k$.

Basis:

The reactivity control concept is that reactivity changes accompanying changes in reactor power are compensated by control rod motion. Reactivity changes associated with vapor, xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated by changes in the soluble boron concentration. During power operation, the shutdown groups are fully withdrawn and control of reactor power is by the control groups. A reactor trip occurring during power operation will put the reactor into the hot shutdown condition.

The control rod insertion limits provide for achieving hot shutdown by reactor trip at any time assuming the highest worth control rod remains fully withdrawn with sufficient margins to meet the assumptions used in the accident analysis. In addition, they provide a limit on the maximum inserted rod worth in the unlikely event of hypothetical rod ejection and provide for acceptable nuclear peaking factors. The solid lines shown in Figure 3.10-1 meet the shutdown requirement for the first 50% of Cycle 3. The end-of-cycle life limit is represented by the dotted lines. The end-of-cycle life limit may be determined on the basis of plant startup and operating data to provide a more realistic limit which will allow for more flexibility in plant operation and still assure compliance with the shutdown requirement. The maximum shutdown margin requirement occurs at end of core life and is based on the value used in analysis of the hypothetical steam break accident. Early in core life, less shutdown margin is required, and Figure 3.10-2 shows the shutdown margin equivalent to 1.75% reactivity (3) at end of life with respect to an uncontrolled cooldown. All other accident analyses are based on 1% reactivity shutdown margin. The specified control rod insertion limits have been revised for Cycle 3 in order to meet the design basis criteria on (1) potential ejected control rod worth and peaking factor (4), (2) radial power peaking factors, F_{DR} , and (3) required shutdown margin.

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The various control rod banks (shutdown banks, control banks, and part length rods) are each to be moved as a bank, that is, with all rods in the bank within one step (5/8 inch) of the bank position. Position indication is provided by two methods: a digital count of actuation pulses which shows the demand position of the banks and a linear position indicator (LVDT) which indicates the actual rod position(2). The 15-inch permissible misalignment provides an enforceable limit below which design distribution is not exceeded. In the event that an LVDT is not in service, the effects of a malpositioned control rod are observable on nuclear and process information displayed in the control room and by core thermocouples and in-core movable detectors. The determination of the hot channel factors will be performed by means of the movable in-core detectors.

The two hours in 3.10.1.5 are acceptable because complete rod misalignment (part-length or full-length control rod 12 feet out of

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alignment with its bank) does not result in exceeding core safety limits in steady state operation at rated power and is short with respect to probability of an independent accident. If the condition cannot be readily corrected, the specified reduction in power to 70% will ensure that design margins to core limits will be maintained under both steady state and anticipated transient conditions.

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The intent of the test to measure control rod worth and shutdown margin (Specification 3.10.1.6) is to measure the worth of all rods less the worth of the worst case for an assumed stuck rod; that is, the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequently over the life of the plant, to be associated primarily with determinations of special interest such as end of life cooldown, or startup of fuel cycles which deviate from normal equilibrium conditions in terms of fuel loading patterns and anticipated control bank worths. These measurements will augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

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Operation with abnormal rod configuration during low power and zero power testing is permitted because of the brief period of the test and because special precautions are taken during the test.

Two criteria have been chosen as a design basis for fuel performance related to fission gas release, pellet temperature, and cladding mechanical properties. First, the peak value of linear power density must not exceed 21.1 kW/ft. Second, the minimum DNBR in the core must not be less than 1.30 in normal operation or in short term transients.

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In addition to the above, the initial steady state conditions for the peak linear power for a loss-of-coolant accident must not exceed the values assumed in the accident evaluation. This limit is required in order for the maximum clad temperature to remain below that established by the ECCS Acceptance Criteria. To aid in specifying the limits on power distribution the following hot channel factors are defined.

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F_Q , Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F_Q^N , Nuclear Heat Flux Hot Channel Factor, is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions.

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F_Q^E , Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes through the core. Thus, the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

It has been determined by extensive analysis of possible operating power shapes that the design limits on peak local power density and on minimum DNB at full power are met, provided:

$$F_Q^N \leq 2.233 \cdot K(z) \text{ and } F_{\Delta H}^N \leq 1.55$$

$K(z)$ is the normalized peaking factor axial dependence used in the LOCA analysis and is shown in Figure 3.10-3. For normal operation, it is not necessary to measure these quantities. Instead, it has been determined that, provided certain conditions are observed, the above hot channel factor limits will be met; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position.
2. Control rod banks are sequenced with overlapping banks as shown in Figure 3.10-1.
3. The control bank insertion limits are not violated.
4. Part length control rods are not inserted.
5. Axial power distribution control procedures, which are given in terms of flux difference control, are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined on the difference in power between the top and bottom halves of the core.

For operation at a fraction P of full power the design limits are met, provided,

$$F_Q^N \leq \frac{2.233 \cdot K(z)}{P} \text{ in the flux difference range } -17 \text{ percent to } +12 \text{ percent}$$

$$\text{and } F_{\Delta H}^N \leq 1.55 \{1 + 0.2 (1-P)\}$$

where P is the fraction of full power at which the reactor is operating: $0 \leq P \leq 1.0$.

The permitted relaxation in F_{AH}^N with reduced power allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met.

The procedures for axial power distribution control referred to above include operator control of flux difference to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of Flux Difference (ΔI) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = ΔI /fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset, it varies primarily with burnup.

The target (or reference) value of flux difference is determined as follows: At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with part length rods withdrawn from the core and with control Bank D more than 190 steps withdrawn. This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of ± 5 percent ΔI is permitted from the indicated reference value. During periods where extensive load following is required, it may be impossible to establish the required core conditions for measuring the target flux difference every month. For this reason, the specification provides two methods for updating the target flux difference.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power.

Strict control of the flux difference is not possible during certain physics tests, control rod exercises, or during the required periodic excore calibration which require larger flux differences than permitted. Therefore, the specifications on power distribution are not applicable during physics tests, control rod exercises, or excore calibrations; this is acceptable due to the extremely low probability of a significant accident occurring during these operations. Excore calibration includes that period of time necessary to return to equilibrium operating conditions. In some instances of rapid plant power reduction automatic rod motion will cause the flux difference to deviate from the target bank when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band, however, to simplify the specification, a limitation of one hour in any

period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequence of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for flux difference in the range +14 percent to -14 percent (+11 percent to -11 percent indicated) increasing by +1 percent for each 2 percent decrease in rated power. Therefore, while the deviation exists the power level is limited to 90 percent or lower depending on the indicated flux difference. In all cases the (+5) percent target band is the Limiting Condition for Operation. Only when the target band is violated do the limits under specification 3.10.2.7 apply.

If, for any reason, flux difference is not controlled with the ± 5 percent band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the limits is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished by using the chemical volume control system to position the full length control rods to produce the required indication of flux difference.

An upper bound envelope of 2.30 times the normalized peaking factor axial dependence has been determined from extensive analysis considering all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10.2. The specifications on power distribution control insure that xenon distributions are not developed which, at a later time could cause greater local power peaking even though the flux difference is then within limits. The results of a loss of coolant accident analysis based on this upper bound envelope indicate that the peak clad temperature would not exceed the 2200°F limits. The nuclear analyses of credible power shapes consistent with the power distribution control procedures have shown that the F_q limit of 2.30/P is not exceeded:

For transient events the core is protected from exceeding 21.1 KW/ft locally, and from going below a minimum DNBR of 1.30, by automatic protection on power, flux difference, pressure and temperature.

Measurements of the hot channel factors are required as part of startup physics tests and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors.

In the specified limit of F_q^N there is a 5% allowance for uncertainties (1) which means that normal operation of the core within the defined conditions and procedures is expected to result in a measured $F_q^N < 2.233/1.05$; for example, at rated power even on a worst case basis. When a measurement is taken, experimental error must be allowed for and 5% is the appropriate allowance for a full core representative map taken with the movable incore detector flux mapping system.

In the specified limit of F_{AH}^N there is an 8% allowance for design prediction uncertainties which means that normal operation of the core is expected to result in $F_{AH}^N \leq 1.55/1.08$ at rated power. The uncertainty to be associated with a measurement of F_{AH}^N by the movable incore system on the other hand is 4% which means that the normal operation of the core shall result in a measured $F_{AH}^N \leq 1.55/1.04$ at rated power. The logic behind the larger design uncertainty in this case is that (a) abnormal perturbation in the radial power shape (e.g., rod misalignment) affect F_{AH}^N in most cases without necessarily affecting F_{AH}^N through movement of part length rods and can limit it to the desired value (8) while the operator has some control over F_{AH}^N through F_{AH}^Z by motion of control rods, he has no direct control over F_{AH}^N , and (c) an error in the predictions for radial power shape which may be detected during startup physics tests can be compensated for in F_{AH}^N by tighter axial control, but compensation for F_{AH}^N is less readily available. ^q

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Quadrant power tilts are based upon the following considerations. The radial power distribution within the core must satisfy the design values assumed for calculation of power capability. Radial power distributions, measured as part of the startup physics testing, are periodically measured at a monthly or greater frequency. These measurements are taken to assure that the radial power distribution with any quarter core radial power asymmetry conditions are consistent with the assumptions used in power capability analyses. It is not intended that extended reactor operation would continue with a power tilt condition which exceeds the radial power asymmetry considered in the power capability analysis.

During normal plant startup, quadrant power tilt ratio may exceed 1.02 due to instrumentation instabilities as a result of rodded configurations and low excore detector signal levels below 50% of full power. Sustained power operation below 50% of full power would require a renormalization of the calculational methods for determining power tilt to compensate for change in signal levels once equilibrium conditions are met.

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The two-hour time interval in this specification is considered ample to identify a dropped or misaligned rod and complete realignment procedures to eliminate the tilt. In the event that the tilt conditions cannot be eliminated within the two-hour time allowance, additional time would be needed to investigate the cause of the tilt condition. The measurements would include a full core physics map utilizing the movable detector system. For a tilt condition ≤ 1.09 an additional 22 hours time interval is authorized to accomplish these measurements. However, to assure that the peak core power is maintained below limiting values, a reduction of reactor power of two percent for each one percent of indicated tilt is required. Physics measurements have indicated that the core radial power peaking would not exceed a two-to-one relationship with the indicated tilt from the excore nuclear detector system for the worst rod misalignment.

the event the tilt condition of 1.09 cannot be eliminated after 24 hours, the reactor power level will be reduced to the range required for low power physics testing. To avoid reset of a large number of protection setpoints, the power range nuclear instrumentation would be reset to cause an automatic reactor trip at 55% of allowed power. A reactor trip at this power has been selected to prevent, with margin, exceeding core safety limits even with a nine percent tilt condition. If a tilt ratio greater than 1.09 occurs which is not due to a misaligned rod, the reactor power shall be brought to a hot shutdown condition for investigation.

However, if the tilt condition can be identified as due to rod misalignment, operation can continue at a reduced power (2% for each one percent the tilt ratio exceeds 1.0) for the two-hour period necessary to correct the rod misalignment.

The specified rod drop time is consistent with safety analyses that have been performed. (1)

Part length rod insertion has been limited to eliminate adverse power shapes (Section 3.10.5.2).

An inoperable rod imposes additional demands on the operator. The permissible number of inoperable control rods is limited to one in order to limit the magnitude of the operating burden, but such a failure would not prevent dropping the operable rods upon reactor trip.

Normal reactor operation causes significant pellet cracking and fragmentation. Consequently, handling of irradiated fuel assemblies can result in relocation of these fragments against the cladding. Calculations show that high cladding stresses can occur if the reactor power increase is rapid during the subsequent startup.

The 72-hour period allows for stress relaxation of the clad before the ramp rate requirement is removed, thereby, reducing the potential harmful effects of possible pellet or fragment relocation.

The 3% limit is imposed to minimize the effects of adverse cladding stresses resulting from part power operation for extended periods of time. The time period of 30 days is based upon the successful power ramp demonstrations performed on Zircaloy clad fuel in operating reactors, resulting in no cladding failures.

References

- (1) FSAR, Section 14 and WCAP-8243
- (2) FSAR, Section 7.3
- (3) WCAP-8243, Section 4.4.2
- (4) WCAP-8243, Section 4.4.3

CONTROL GROUP INSERTION LIMITS FOR
THREE LOOP OPERATION

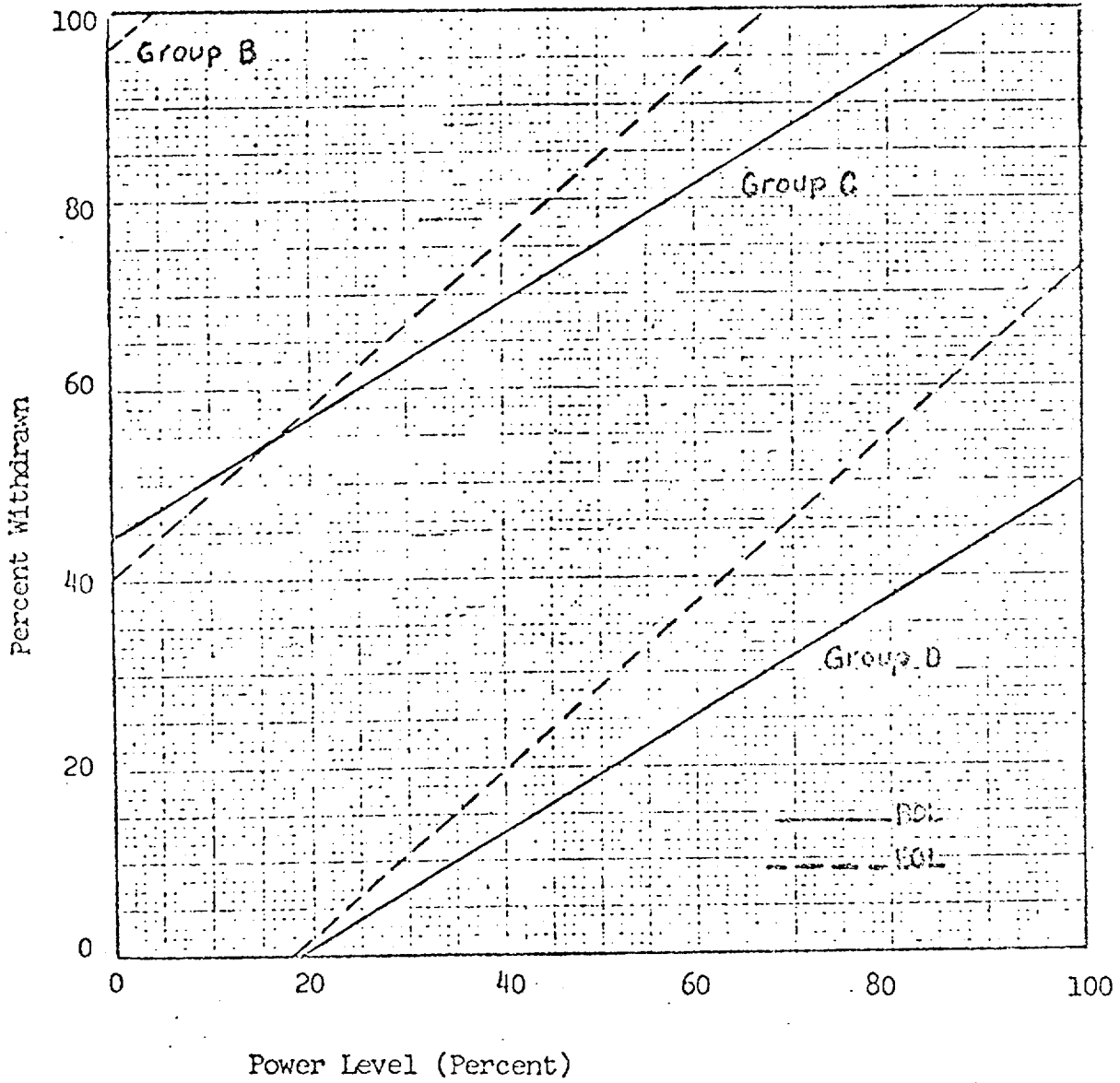


FIGURE 3.10-1

3.10-16

REQUIRED SHUTDOWN
VS. BURN CONCENTRATION
H. B. ROBINSON #2 - CYCLE 3

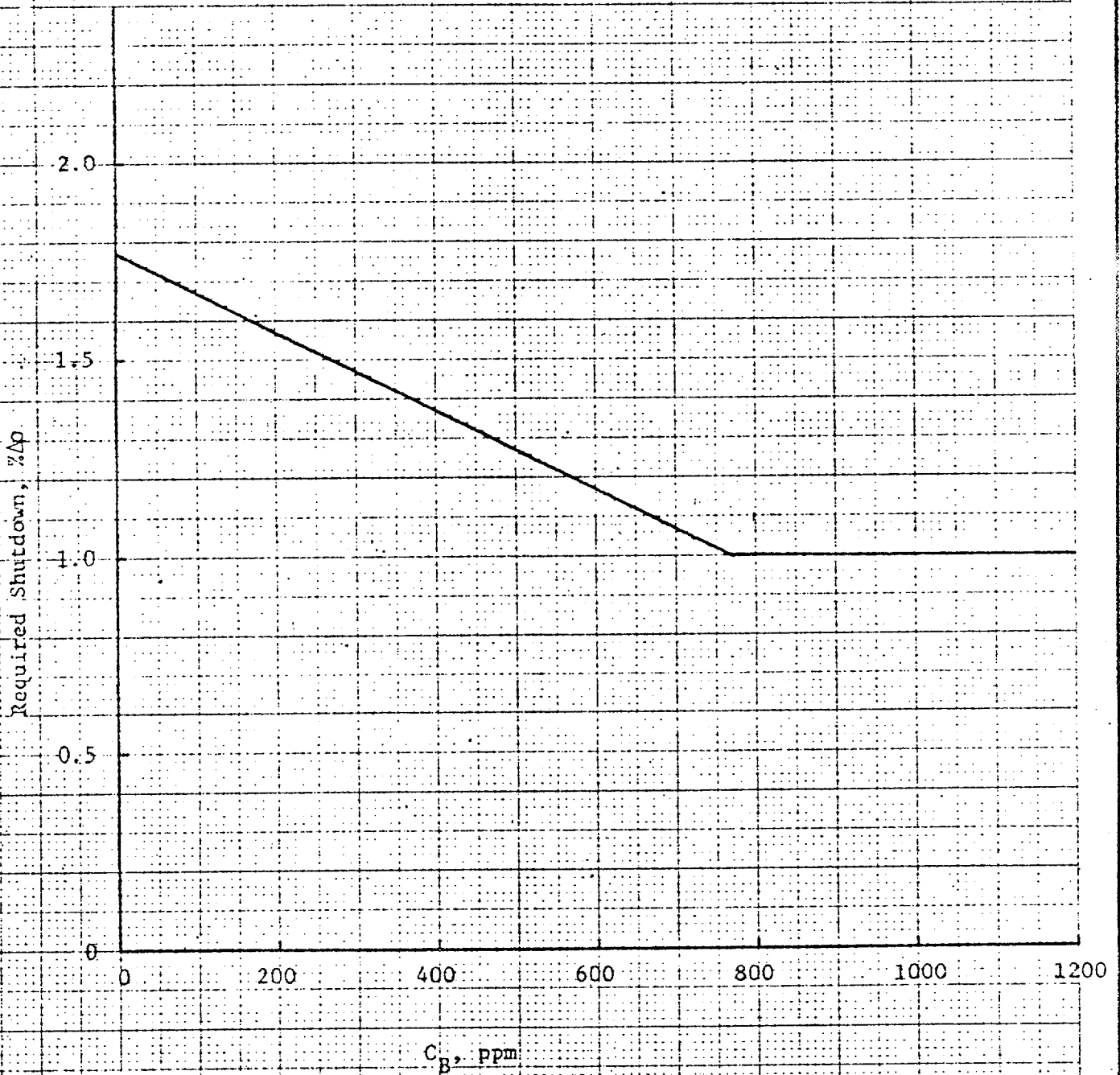


Figure 3.10-2

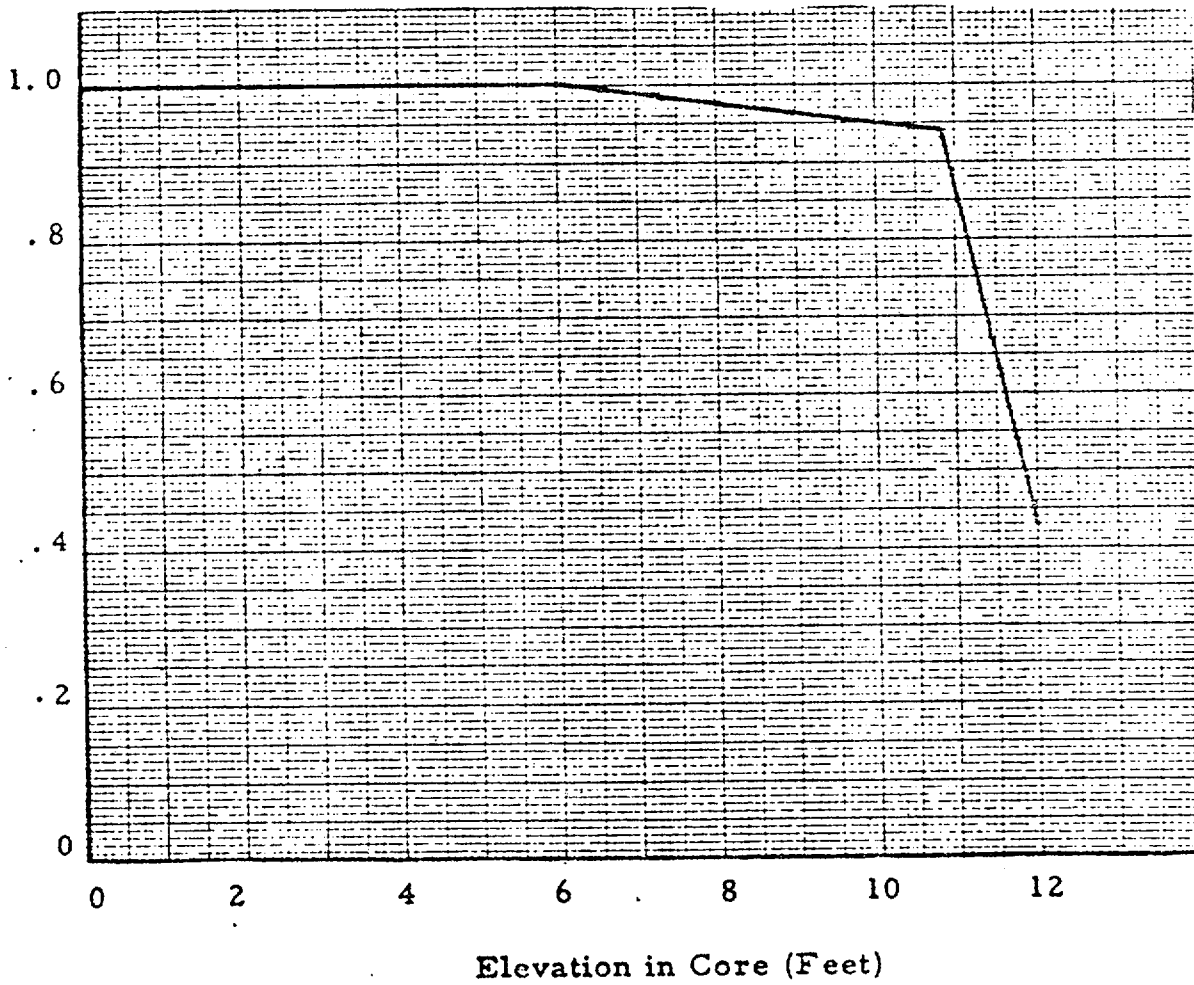


Figure 3.10-3 Normalized Axial Dependence Factor for F_Q versus Elevation

NEGATIVE DECLARATION
REGARDING PROPOSED CHANGES TO THE
TECHNICAL SPECIFICATIONS OF LICENSE DPR-23
H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT 2
DOCKET NO. 50-261


The Nuclear Regulatory Commission (the Commission) has considered the issuance of changes to the Technical Specifications of Facility Operating License No. DPR-23. These changes would authorize the Carolina Power and Light Company (the licensee) to operate the H. B. Robinson Steam Electric Plant, Unit 2 (located in Darlington County, Hartsville, South Carolina), with changes to the limiting conditions for operation associated with fuel assembly specific power resulting from application of the Acceptance Criteria for Emergency Core Cooling System (ECCS).

The U. S. Nuclear Regulatory Commission, Division of Reactor Licensing, has prepared an environmental impact appraisal for the proposed changes to the Technical Specifications of License No. DPR-23, H. B. Robinson Steam Electric Plant, Unit 2, described above. On the basis of this appraisal, the Commission has concluded that an environmental impact statement for this particular action is not warranted because there will be no environmental impact attributable to the proposed action other than that which has already been predicted and described in the Commission's Final Environmental Statement for H. B. Robinson Steam Electric Plant, Unit 2, published in April 1975.

The environmental impact appraisal is available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Hartsville Memorial Library, Home and Fifth Avenues, Hartsville, South Carolina.

Dated at Rockville, Maryland, this 25th day of September 1975.

FOR THE NUCLEAR REGULATORY COMMISSION


Wm. H. Regan, Jr., Chief
Environmental Projects Branch 4
Division of Reactor Licensing

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

ENVIRONMENTAL IMPACT APPRAISAL BY THE DIVISION OF REACTOR LICENSING

SUPPORTING AMENDMENT NO. 13 TO DPR-23

CHANGE NO. 38 TO THE TECHNICAL SPECIFICATIONS

CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT 2

ENVIRONMENTAL IMPACT APPRAISAL

1. Description of Proposed Action

By letter dated October 2, 1974, and March 14, 1975, Carolina Power and Light Company (CP & L) submitted proposed changes to the Technical Specifications, Appendix A, to License DPR-23. The proposed changes resulted from the application of the Acceptance Criteria for Emergency Core Cooling System (ECCS) to the present core. Supplemental information relating to the ECCS evaluation has been supplied by CP & L in their letters of April 18, June 20, and July 24, 1975. In addition, the licensee stated that there would be no environmental impact associated with these proposed changes. The staff has independently reviewed this matter and the conclusions are set forth below.

CP & L is presently licensed to operate H. B. Robinson Steam Electric Plant, Unit 2, located in the State of South Carolina, Darlington County, at power levels up to 2,200 megawatt thermal (MWT). The proposed change to incorporate the ECCS Acceptance Criteria does not result in an increase or decrease in power levels of the unit. The restrictions on heat generation rates will require careful control of fuel operating history. However, there should be no reduction on total burnup resulting from the revised ECCS evaluation methods. Since neither power level nor fuel burnup is affected by the action, the action does not affect the benefits of electric power production considered for the captioned facility in the Commission's Final Environmental Statement (FES) for H. B. Robinson Steam Electric Plant, Unit 2, Docket No. 50-261, dated April 1975.

2. Environmental Impacts of Proposed Action

Potential environmental impacts associated with the proposed action are those which may be associated with incorporation of the ECCS Acceptance Criteria and utilization of nuclear fuel for this facility.

It is particularly noted that in the absence of any significant change in power levels, there will be no change in cooling water requirements and consequently no increase in environmental impact from radioactive effluents and thermal effluents for normal operation or post-accident conditions which in turn could not lead to significant increases in radiation doses or thermal stress to the public or to biota in the environment.

For normal operating conditions, no environmental impact other than as described in the Commission's Final Environmental Statement (FES) for H. B. Robinson Steam Electric Plant, Unit 2, Docket No. 50-261, dated April 1975, can be predicted for the proposed action. The staff's calculated releases of radioactive effluents, both gaseous and liquid, are based on expected radionuclide production and their release rates to the environment. The estimates of radionuclide production and their release rates are not significantly affected as the licensed reactor power is unchanged. No increase in the calculated release of radioactive effluents is predicted. Consequently, no increases in radiation doses to man or other biota are predicted.

3. Conclusion and Basis for Negative Declaration

On the basis of the foregoing analysis, it is concluded that there will be no environmental impact attributable to the proposed action other than has already been predicted and described in the Commission's FES for H. B. Robinson Steam Electric Plant, Unit 2. Having made this conclusion, the Commission has further concluded that no environmental impact statement for the proposed action need be prepared and that a negative declaration to this effect is appropriate.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-261

CAROLINA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY
OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 13 to Facility Operating License No. DPR-23 issued to Carolina Power & Light Company which revised Technical Specifications for operation of the H. B. Robinson Unit 2, located in Darlington County, Hartsville, South Carolina. The amendment is effective as of its date of issuance.

The amendment (1) revises the operating limits in the Technical Specifications based upon an acceptable evaluation model that conforms to the requirements of 10 CFR § 50.46, and (2) terminates restrictions imposed on the facility by the Commission's December 27, 1974 Order for Modification of License, and imposes instead, limitations established in accordance with 10 CFR § 50.46.

The application for amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on July 7, 1975 (40 F.R. 28509). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO.13 TO LICENSE NO. DPR-23

(CHANGE NO.38 TO TECHNICAL SPECIFICATIONS)

CAROLINA POWER & LIGHT COMPANY

H. B. ROBINSON STEAM ELECTRIC UNIT NO. 2

DOCKET NO. 50-261

Introduction:

By letter dated October 2, 1974 as supplemented March 14, April 18, June 20, and July 24, 1975, Carolina Power & Light Company requested changes to the Technical Specifications appended to Facility Operating License DPR-23 for the H. B. Robinson Steam Electric Plant Unit No. 2. The purpose of the request is to revise portions of the Technical Specifications related to the emergency core cooling system (ECCS). These revisions are based on the licensee's reevaluation of the ECCS performance.

Discussion:

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License (1) implementing the requirements of 10 CFR 50.46, "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors". One of the requirements of the Order was that the licensee shall submit a reevaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms with the provisions of 10 CFR Part 50, Section 50.46. The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendment as may be necessary to implement the evaluation results. As required by our Order of December 27, 1974, Carolina Power & Light Company (CP&L) has submitted the ECCS reevaluation and related Technical Specifications which are applicable to the present Robinson-2 core (cycle 3).

The March 14, 1975 correspondence forwarded the ECCS reevaluation using the approved Westinghouse evaluation model of December 25, 1974 and covered the required spectrum of large pipe breaks for the Robinson plant. This reevaluation was supplemented in a letter dated July 24, 1975 in which one calculation was resubmitted using the approved March 15, 1975 change to the Westinghouse evaluation model. The latter calculation included an upgraded steam cooling model. The correspondence of April 18, 1975 forwarded the Robinson submittal addressing the effects of boron precipitation on the long-term core cooling capability following a loss-of-coolant accident (LOCA).

This submittal adopted the generic report (2) on this subject by Westinghouse and provided justification for and information on the Robinson-2 procedures to adequately initiate such cooling following a LOCA. The correspondence of June 20, 1975 proposed requirements for monitoring the incore peaking factor established by the ECCS reevaluation. The correspondence of July 24, 1975 also provided information concerning submerged valve motors within containment, single failure or operator error which could cause any manually controlled, electrically operated valve to move to a position that could adversely affect the ECCS, and justification of the limiting fuel region utilized in the LOCA analysis.

Evaluation:

The background of the NRC staff review of the Westinghouse ECCS models and their application to H. B. Robinson Unit 2 is described in the staff's Safety Evaluation Report (SER) for Robinson dated December 27, 1974 issued in connection with the Order. The bases for acceptance of the principal portions of the evaluation model are set forth in the staff's Status Report(3) of October 1974, the Supplement(4) to the Status Report of November 1974 (referenced in the December 27, 1974 SER) and a staff review(5) dated April 22, 1975 of the change to the model. The December 27, 1974 SER also describes the various changes required in the earlier evaluation model. Together, the December 27, 1974 SER, the Status Report and Supplement, and the staff review of April 22, 1975 describe an acceptable ECCS evaluation model and the basis for the staff's acceptance of the model. The Robinson ECCS evaluation conforms to the accepted model.

The licensee has submitted LOCA analyses that address both small and major reactor coolant system pipe ruptures. The small break LOCA incorporated the previously acceptable submittal of October 2, 1974. A three break spectrum, specific for H. B. Robinson, was submitted and an applicable generic plant sensitivity study was referenced in conformity with the break spectrum requirements of 10 CFR 50.46(a).

The analyses identified the worst break size as the 0.4 double-ended cold leg guillotine break with a calculated peak clad temperature of 2200°F; this is acceptable as specified in 10 CFR Part 50, Section 50.46(b). In addition, the calculated maximum local metal/water reaction of 9.6% and total core wide metal/water reaction of less than 0.3% are well below the allowable limits of 17% and 1%, respectively. These results are for region 5 (cycle 3) fuel which was identified as the limiting fuel in the core.

These analyses assumed that there was a coincident loss of offsite power at the initiation of the LOCA, which would result in pump speed coastdown. A sensitivity analysis was presented for the limiting LOCA with no loss of offsite power. The results showed that the loss of offsite power which was assumed in the analyses is more conservative. The analysis was presented for three loop operation only, hence, the reactor will not be allowed to operate with one or more idle loops.

Our review of plant-specific assumptions regarding the Robinson analysis included the areas of containment pressure, single failure criterion, long term boron concentration buildup, and power distribution control and monitoring.

A. ECCS Containment Pressure Evaluation

The ECCS containment pressure calculations for Robinson, Unit 2 were done using the Westinghouse ECCS evaluation model. We required, however, that justification of the plant-dependent input parameters used in the analysis be submitted for our review.

This information was submitted for H. B. Robinson, Unit 2 by letter dated December 4, 1974. Carolina Power & Light has reevaluated the containment net-free volume, the passive heat sinks, and operation of the containment heat removal systems with regard to the conservatism for ECCS analysis. This evaluation was based on measurements within the containment and from as-built drawings to which additional margin was added. The containment heat removal systems were assumed to operate at their maximum capacities and minimum operational values for the spray water and service water temperatures were assumed.

We have reviewed the plant-dependent information used for the ECCS containment pressure analysis for H. B. Robinson, Unit 2 and concluded that the values used are conservative. Therefore, the containment pressures were calculated in accordance with the requirements of Appendix K to 10 CFR Part 50 of the Commission's regulations.

B. Single Failure Criterion

Appendix K to 10 CFR Part 50 of the Commission's regulations requires that the combination of ECCS subsystems to be assumed operative shall be those available after the most limiting single failure of ECCS equipment has occurred. The worst single failure which would minimize the emergency core cooling available to cool the core and provide maximum containment cooling was identified by Westinghouse as the loss of a low pressure ECCS pump.

The review of the Robinson piping and instrumentation diagrams indicated that the spurious actuation of specific electrically operated valves could affect the failure modes and effects analysis. The following valves have been identified as not satisfying the single failure criteria:

| <u>Valve No.</u> | <u>Location</u> | <u>Initial ECCS Orientation</u> |
|------------------|-------------------|-------------------------------------|
| 862 A/B* | RWST Line | Open |
| 863 A/B* | RHR/SI Crossline | Closed |
| 864 A/B* | RWST Line | Open |
| 865 A/B/C | Accumulators | Open |
| 866 A/B* | Hot Leg Injection | Closed |
| 878 | SI Discharge Tie | Open |

* These valves must be actuated to change position during long term cooling.

CP&L has proposed to align the valves in their normal ECCS orientation and then remove A. C. power at the motor control centers. The NRC staff has previously concluded that deenergizing A. C. power to motor operated valves is an acceptable procedure to preclude spurious actuation. However, the valves denoted by asterisks in the above list must be actuated during the switchover from the injection to recirculation phase (approximately 25 minutes after a LOCA) or at the switchover to hot leg injection (18 hours). In order to actuate these valves at various times after the LOCA would require personnel to restore A. C. power at the motor control centers outside of the control room. We have evaluated the time available for the operator to accomplish the necessary manual actions for changeover from injection to recirculation mode of operation. There are 15 minutes available to perform the manual actuation operations involved in switchover which is in excess of the time required to perform the actions.

The licensee has committed to modifying his design to eliminate operator actions outside of the control room while precluding single failures that would result in loss of cooling capability. The licensee has submitted⁽⁶⁾ modifications which will be made at the end of the present fuel cycle. The proposed modifications are scheduled to be ready for cycle 4 operation which will commence on or about December 1, 1975. Until such time that the required modifications are completed, the following interim procedures⁽⁷⁾ have been instituted.

1. Operating personnel on each shift will be specifically designated to restore A. C. power at the motor control centers to MOV 862 A/B, 863 A/B, and 864 A/B. These personnel are not to be considered available for other duties following a LOCA until the successful completion of switchover to the recirculation mode of ECCS operation.
2. In the event of a LOCA, these personnel will be dispatched immediately from the control room to the motor control centers. Power will be restored to these valves at the instruction of the control room

following the actuation of the RWST low level alarm in the sequence specified in the emergency operating procedures. Communication systems will be provided between the control room and the assigned operating personnel.

In addition, the licensee has identified two air operated valves in the RHR discharge line (valves 605 and 758) which will be locked into the closed position during power operation. As a result, the air supply to these valves will be shut off during normal operation.

We have reviewed the analysis of the ECCS performance and conclude that single failure criteria has been adequately considered and that the ECCS with control procedures as adopted are acceptable.

C. Long Term Boron Concentration Buildup

The licensee has submitted the emergency operating procedures proposed for the long term post LOCA core cooling period and has stated that these procedures will prevent excessive concentrations of boron in the reactor vessel. The procedures were supported by a Westinghouse analysis(2). We have reviewed the proposed emergency procedures and the referenced analysis and concluded that the existing ECCS can be operated in a manner that will prevent excessive boric acid concentration from occurring.

The initial cold leg injection period has been modified from the originally proposed 20 hours to 18 hours. Following the initial 18 hour cold-leg injection period, the licensee has agreed to utilize the following acceptable procedures:

1. simultaneous hot and cold leg injection, or if one RHR pump is not operable,
2. alternate hot and cold leg injection, using sufficiently short time periods between change-overs to prevent excessive boric acid buildup in the core region.

CP&L has modified their original long term cooling procedures to specify that motor-operated valves 866A and 866B in the safety injection (SI) pump discharge line to the hot legs and motor-operated valves 750 and 751 in the hot leg suction line be opened within 2 hours following a LOCA even though hot leg injection will not be required for 18 hours. CP&L has changed their procedures so that these valves can be opened within 2 hours because the valve motor operators have been proven by test to be operable only for periods up to 2 hours in a post-LOCA environment. Valves 866A and 866B are locked in the closed position during normal operation so that a safety injection pump initiation signal will not initiate hot leg injection. Prior to opening valves 866A, 866B, 750 and 751, following a LOCA, valve 869 in the hot leg injection header and valve 743 in the RHR supply line will be closed. Closing valves 869 and 743 will prevent premature hot leg injection and since these valves are located outside containment the valve position can be visually verified. When hot leg injection is required (18 hours), it can be initiated through the normal path using valve 869 or through an alternate line using a backup procedure.

Operation in accordance with the above described procedures will prevent excessive boric acid concentration from occurring and is acceptable.

D. Power Distribution Control and Monitoring

The value of the total peaking factor for the Robinson Plant which maintains peak clad temperature limits in the LOCA analysis is 2.30. The licensee has proposed to employ the constant axial offset control (CAOC) method of power distribution control and monitoring to ensure that the peaking factor does not exceed 2.30 in normal operation of the power/plant. CAOC limits the peaking factor by restricting xenon redistribution during power changes. This will prevent adverse xenon distributions and resultant axial power distributions which could reduce margins to DNB limits during anticipated transients.

The licensee has been using his movable Incore instrumentation to monitor the power distribution with the axial power distribution monitoring system (APDMS). Although the APDMS is an effective method for monitoring of the peaking factor, it does not provide the control features of CAOC which also maintains DNB margins.

The generic Westinghouse CAOC analysis⁽⁸⁾ justifies a peaking factor limit of 2.32. Normally supplemental peaking factor monitoring is required to justify a lower peaking factor. However, the generic analysis supports a substantially lower peaking factor towards the end of a reactor cycle for operation without part length control rods. Since the Robinson Technical Specifications prohibit the use of part length control rods, and this is a reload cycle, the generic analysis supports a peaking factor of 2.30 for the remaining portion of Cycle 3.

The Technical Specifications submitted by CP&L have been modified to reflect NRC standardized wording for this section. CP&L has concurred with these wording changes.

We therefore conclude use of CAOC in Robinson will provide increased safety margin and is acceptable. The licensee has agreed to install two alarms to warn against (1) deviation from the required $\pm 5\%$ flux difference control band at power levels above 90% of rated power, and (2) deviation from the one hour limit that the flux difference may exceed the $\pm 5\%$ control band at power levels at or below 90% of rated power. These alarms are similar to alarms required of other utilities utilizing CAOC for power distribution control. CP&L has been advised that operation until December 1, 1975 without the alarms will be permissible after which time such alarms will be required in conjunction with the use of the CAOC method of power distribution control. This deadline has been included in the Technical Specifications.

Also included in this change are proposed modifications that add additional conservatism to the limits on power ramp rate. These limits were submitted based on discussions with NRC personnel and are intended to prevent

fuel-clad mechanical interaction failures. These ramp rates are consistent with the vendor recommendations and have been reviewed by the NRC staff and found acceptable.

Summary:

The licensee has submitted a reevaluation of the ECCS cooling performance for the H. B. Robinson Steam Electric Unit No. 2. We have reviewed the reanalysis and have concluded:

- (1) The evaluation has been performed wholly in conformance with the requirements of Appendix K to 10 CFR 50 Section 50.46.
- (2) ECCS cooling performance for H. B. Robinson Unit 2 will conform to the peak clad temperature, maximum oxidation, and hydrogen generation criteria of 10 CFR 50.46(b).

In addition, we have concluded that:

- (1) ECCS cooling performance will be adequate despite any postulated failure of any single component.
- (2) Adequate systems and procedures exist to provide reasonable assurance that boron precipitation will not occur within the reactor vessel.
- (3) Adequate procedures exist to provide reasonable assurance that the peaking factor limit of 2.3 (at 2300 MWt) will not be exceeded.

Conclusion:

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated:

BIBLIOGRAPHY

- (1) "Order for Modification of License for H. B. Robinson Steam Electric Plant Unit No. 2," dated December 27, 1974 enclosed in letter from G. Lear, NRC to J. A. Jones, CP&L.
- (2) Letter from C. Case, Westinghouse, to T. Novak, NRC, dated April 1, 1975.
- (3) "Status Report by the Directorate of Licensing in the Matter of Westinghouse Electric Co., ECCS Evaluation Model Conformance to 10 CFR 50 Appendix K," October 15, 1974.
- (4) "Supplement to the Status Report by the Directorate of Licensing in the Matter of Westinghouse Electric Co., ECCS Evaluation Model Conformance to 10 CFR 50, Appendix K", November 13, 1974.
- (5) "NRC Staff Review of the Westinghouse, ECCS Evaluation Model", report from V. Stello to R. C. DeYoung, Jr., dated April 22, 1975.
- (6) Letter from E. E. Utley, CP&L, to R. W. Reid, NRC, dated September 17, 1975.
- (7) Letter from E. E. Utley, CP&L, to G. Lear, NRC, dated July 24, 1975.
- (8) T. Marita, et.al., "Topical Report - Power Distribution Control and Load Following Procedures", WCAP-8403, September, 1974.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-261

CAROLINA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY
OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 13 to Facility Operating License No. DPR-23 issued to Carolina Power & Light Company which revised Technical Specifications for operation of the H. B. Robinson Unit 2, located in Darlington County, Hartsville, South Carolina. The amendment is effective as of its date of issuance.

The amendment (1) revises the operating limits in the Technical Specifications based upon an acceptable evaluation model that conforms to the requirements of 10 CFR § 50.46, and (2) terminates restrictions imposed on the facility by the Commission's December 27, 1974 Order for Modification of License, and imposes instead, limitations established in accordance with 10 CFR § 50.46.

The application for amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on July 7, 1975 (40 F.R. 28509). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.