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April 11, 1979

Docket No. 50-261

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

Dear Mr. Jones:

The Commission has issued the enclosed Amendment No. 36 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2. This amendment is in response to your application dated March 6, 1979.

The amendment authorizes the removal of all port-length control rods from the reactor.

This amendment constitutes only a partial response to the changes requested in your March 6, 1979 request. The balance of your request will be the subject of later Commission action.

The NRC Safety Evaluation and Notice of Issuance of this amendment are also enclosed.

Sincerely,

/s/

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

Const cup

Enclosures:
1. Amendment No. 36 to DPR-23
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures: See next page

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DATE	~~4/ /79~~	4/11/79	4/12/79	4/4/79	4/11/79

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

April 11, 1979

Docket No. 50-261

Mr. J. A. Jones
Senior Vice President
Carolina Power and Light Company
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Raleigh, North Carolina 27602

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A. Schwencer

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Operating Reactors Branch #1
Division of Operating Reactors

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2. Safety Evaluation
3. Notice of Issuance

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345 Courtland Street, N. W.
Atlanta, Georgia 30308

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. 36
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 March 6, 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 satisified.

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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. 36 , 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: April 11, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 36

FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Revise Appendix A as follows:

Remove the following pages and insert identically numbered revised pages:

Pages

ii
3.10-2
3.10-4
3.10-8
3.10-9
3.10-11
3.10-13
3.10-14
3.10-19
5.3-2

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 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:

$F_Q\,(Z) \leq (2.20/P) \times K(Z)$ 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 the function given in Figure 3.10-3, and Z is the core height location of F_Q.

3.10.2.1.2 The predetermined power level at which APDMS initiation is required is given by the relation

$$P_{APDMS} \leq \frac{1.435}{Fxy} \times 0.94$$

3.10.2.1.3 Fxy shall be determined for the unrodded core plane regions away from fuel support grids, located between a core plane elevation 2.0 feet from the top of the core and a core plane elevation 2.0 feet from the bottom of the core with no control rod inserted more than 2.0 feet into the core. This determination shall be made from the movable incore detector maps specified in 3.10.2.3.

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^N 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.

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 Deleted

3.10.6 Inoperable Control Rods

3.10.6.1 A 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.

3.10.6.2 No more than one inoperable control rod shall be permitted during power operation.

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 percent of full power in an hour between 20 percent and 100 percent 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 percent $<P \leq 100$ percent) provided that the plant has operated at or above power level P for at least 72 cumulative hours out of any seven-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 percent of full power in an hour. Alternatively, reactor power increase can be accomplished by a single step increase less than or equal to 10 percent of full power followed by a maximum ramp rate of 3 percent of full power in an hour beginning three 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.

shutdown margin. The specified control rod insertion limits meet the design basis criteria on (1) potential ejected control rod worth and peaking factor,[4] (2) radial power peaking factors, $F_{\Delta H}$, and (3) required margin shutdown.

The various control rod banks (shutdown banks, control banks) 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 (control rod 12 feet out of 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 will ensure that design margins to core limits will be maintained under both steady state and anticipated transient conditions.

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 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.

d. $F^N_{\Delta H}$, 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 $F^N_{\Delta H}$ 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 $F^N_{\Delta H}$.

It has been determined by extensive analysis of possible operating power shapes that the design limits on peak local power density and on minimum DNBR at full power are met, provided the values of F_q and $F_{\Delta H}$ in Specification 3.10.2.1 are not exceeded.

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:

a. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position.

b. Control rod banks are sequenced with overlapping banks as shown in Figure 3.10-1.

c. The control bank insertion limits are not violated.

d. Deleted

e. 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 the limits of Specification 3.10.2.1 are not exceeded.

The permitted relaxation in $F_{\Delta H}^{N}$ 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 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 the specified diviation of ΔI is

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 of 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 percent 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

5.3.1.5 There are 45 full-length RCC assemblies in the reactor core. The full-length RCC assemblies contain 144 inch length of silver-indium-cadmium alloy clad with the stainless steel.[5]

5.3.1.6 Up to 10 grams of enriched fissionable material may be used either in the core, or available on the plant site, in the form of fabricated neutron flux detectors for the purposes of monitoring core neutron flux.

5.3.2 <u>Reactor Coolant System</u>

5.3.2.1 The design of the Reactor Coolant System complies with the Code requirements.[6]

5.3.2.2 All piping, components and supporting structures of the Reactor Coolant System are designed to Class I requirements.

5.3.2.3 The nominal liquid volume of the Reactor Coolant System, at rated operating conditions, is 9343 cubic feet.[7]

References

(1) FSAR Section 3.2.3
(2) FSAR Section 3.2.1
(3) FSAR Section 3.2.1
(4) FSAR Section 3.2.3
(5) FSAR Sections 3.2.1 and 3.2.3
(6) FSAR Table 4.1-9
(7) FSAR Table 4.1-1
(8) "Description and Evaluation of Test Assemblies Containing Gadolinia Bearing Fuel Rods" submitted with letter dated January 5, 1973, from CP&L to the Director of Licensing.
(9) "Description and Evaluation of Test Assemblies Containing Gadolinia Bearing Fuel Rods - H. B. Robinson Unit No. 2 Cycle 3" submitted with letter dated March 12, 1974, from CP&L to the Director of Licensing.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 36 TO FACILITY LICENSE NO. DPR-23

CAROLINA POWER AND LIGHT COMPANY

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

DOCKET NO. 50-261

INTRODUCTION

By letter dated March 6, 1979, Carolina Power and Light Company (the licensee) requested amendment of the Technical Specifications appended to Facility Operating License DPR-23 for H. B. Robinson Unit 2. The proposed amendment would permit removal of the part-length control rods. This has been done on other Westinghouse reactors.

DISCUSSION AND EVALUATION

The Technical Specifications, as now written, require that these part-length rod cluster control assemblies (PLRCCAs) be withdrawn and excluded from the core at all times during reactor operation. The PLRCCAs are not needed, used or assumed to be available in any safety analysis of the facility. The proposed removal, therefore, will not cause any change in required reactivity characteristics or safety margins at full power, low power or shutdown. To the contrary, removal will eliminate the potential for part-length rod insertion into the core during operation. Such an event could cause an abnormal flux distribution or reactor shutdown.

In order to preserve the current dynamic operating characteristics of the reactor (i.e., pressure drops, coolant flow rates, etc.) which could be affected if just removal of the PLRCCAs were to be performed, the licensee proposes to install thimble plug assemblies in the spaces previously occupied by PLRCCAs. The thimble plug assembly consists of a flat base plate with short rods suspended from the bottom surface and a spring pack assembly. The twenty short rods, called thimble plugs, project into the upper ends of the guide thimbles to reduce the bypass flow area. Fuel assemblies without control rods, burnable poison rods, or source rods use identical devices. Similar short rods are also used on the source assemblies and fuel assembly guide thimbles. As installed in the core, the thimble plug assemblies interface with both the upper core plate and with the fuel assembly

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top nozzles by resting on the adapter plate. The spring pack is compressed by the upper core plate when the upper internals assembly is lowered into place. Each thimble plug is permanently attached to the base plate by a nut which is locked to the threaded end of the plug by a pin welded to the nut.

All components in the thimble plug assembly, except for the spring, are constructed from type 304 stainless steel. The springs are wound from Inconel X-750 for corrosion resistance and high strength.

The thimble plugs will effectively limit bypass flow through the rod cluster control guide thimbles in the fuel assemblies from which the PLRCCAs have been removed, just as they currently limit bypass flow in those assemblies which do not contain control rods, source rods, or burnable poison rods.

Based on the considerations that (1) the PLRCCAs are not needed for reactor operation, (2) that removal of these assemblies will remove the chance for an abnormal flux distribution or reactor shutdown and (3) that insertion of the thimble plug assemblies will preserve the current dynamic operating characteristics of the reactor, we conclude that this change is acceptable.

ENVIRONMENTAL CONSIDERATION

We have determined that this amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that this amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

CONCLUSION:

We have concluded, based on the considerations discussed above, that: (1) because the amendment involves neither a significant increase in the probability or consequences of accidents previously considered nor a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public:

Dated: April 11, 1979

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-261

CAROLINA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 36 to Facility Operating License No. DPR-23, to the Carolina Power and Light Company, (the licensee), which revised Technical Specifications for operation of the H. B. Robinson Steam Electric Plant Unit No. 2 (the facility) located in Darlington County, Hartsville, South Carolina. The amendment is effective as of the date of its issuance.

The amendment authorizes the removal of all part-length control rods from the reactor.

The application for the 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. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the licensee's submittal dated March 6, 1979, (2) Amendment No. 36 to License No. DPR-23, and (3) the Commission's related Safety Evaluation. All of these items are 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. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 11th day of April 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

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

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

Corrected Copy - Submitted 5-1-79

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 36 TO FACILITY LICENSE NO. DPR-23

CAROLINA POWER AND LIGHT COMPANY

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

DOCKET NO. 50-261

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DISCUSSION AND EVALUATION

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In order to preserve the current dynamic operating characteristics of the reactor (i.e., pressure drops, coolant flow rates, etc.) which could be affected if just removal of the PLRCCAs were to be performed, the licensee proposes to install thimble plug assemblies in the spaces previously occupied by PLRCCAs. The thimble plug assembly consists of a flat base plate with short rods suspended from the bottom surface and a spring pack assembly. The twenty short rods, called thimble plugs, project into the upper ends of the guide thimbles to reduce the bypass flow area. Fuel assemblies without control rods, burnable poison rods, or source rods use identical devices. Similar short rods are also used on the source assemblies and fuel assembly guide thimbles. As installed in the core, the thimble plug assemblies interface with both the upper core plate and with the fuel assembly top nozzles by resting on the adapter plate. The spring pack is compressed by the upper core plate when the upper internals assembly is lowered into place. Each thimble plug is permanently attached to the base plate by a nut which is locked to the threaded end of the plug by a pin welded to the nut.

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Based on the considerations that (1) the PLRCCAs are not needed for reactor operation, (2) that removal of these assemblies will remove the chance for an abnormal flux distribution or reactor shutdown and (3) that insertion of the thimble plug assemblies will preserve the current dynamic operating characteristics of the reactor, we conclude that this change is acceptable.

ENVIRONMENTAL CONSIDERATION

We have determined that this amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that this amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

CONCLUSION:

We have concluded, based on the considerations discussed above, that: (1) because the amendment involves neither a significant increase in the probability or consequences of accidents previously considered nor a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public:

Dated: April 11, 1979