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March 8, 1982

Docket No. 50-261

Mr. J. A. Jones
Senior Executive 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. 64 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated September 26, 1980.

The amendment revises the Technical Specifications related to reactor cooling, emergency core cooling, and refueling to ensure sufficient redundancy in decay heat removal capability is maintained during all modes of plant operation.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

/S/

William J. Ross, Project Manager
Operating Reactors Branch No. 1
Division of Licensing

Enclosures:
1. Amendment No. 64 to DPR-23
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

no legal objection to amendment & F.R. notice only

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Department of Justice
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Raleigh, North Carolina 27602

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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. 64
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 September 26, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-23 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 64 , 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

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 8, 1982

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 64 TO FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.1-1	3.1-1
3.1-3	3.1-3
3.1-3a	3.1-3a
3.3-2	3.3-2
3.3-5	3.3-5
-	3.3-5a
-	3.3-12a
3.8-2	3.8-2
3.8-3	3.8-3
3.8-4	3.8-4
-	3.8-4a
6.9-8	6.9-8

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objective

To specify those Reactor Coolant System conditions which must be met to assure safe reactor operation.

Specification

3.1.1 Operational Components

3.1.1.1 Coolant Pumps

a. At least one reactor coolant pump or the Residual Heat Removal System shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

b. When the reactor is critical, at least one reactor coolant pump shall be in operation.

c. Power operation with less than three loops in service is prohibited.

d. Deleted.

e. A reactor coolant pump may be started (or jogged) only if there is a steam bubble in the pressurizer or the steam generator temperature is no higher than 50°F higher than the temperature of the reactor coolant system.

Basis

When the boron concentration of the Reactor Coolant System is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is ~~running~~ while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one-half hours. The pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant.

Specification 3.1.1.1.c requires that all three reactor coolant pumps be operating during power operation to provide core cooling in the event that a loss of flow occurs. The flow provided will keep DNB well above 1.30. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. Specification 3.1.1.1.b will allow special low power physics testing to be conducted following refueling under adequately controlled conditions.

The pressurizer is necessary to maintain acceptable system pressure during normal plant operation, including surges that may result following anticipated transients.

Each of the pressurizer code safety valves is designed to relieve 288,000 lbs. per hr. of saturated steam at the valve setpoint.[2] Below 350°F and 450 psig in the Reactor Coolant System (RCS), the Residual Heat Removal System can remove decay heat and thereby control system temperature. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than the capacity of a single valve. One valve therefore provides adequate defense against over-pressurization of the RCS for primary coolant temperatures less than 350°F and two valves provide protection for any temperature.

The combined capacity of the three pressurizer safety valves is greater than the maximum surge rate resulting from the complete loss of load[3] without a direct reactor trip or any other control. ASME Section III of the Code allows a maximum variation in the setpoint of 3 percent above the design set pressure.

The requirement that 125 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency power source provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown.

One steam generator capable of performing its heat transfer function will provide sufficient heat removal capability to remove core decay heat after a normal reactor shutdown. The reactor cannot be made critical without water in all three steam generators, since the low-low steam generator water level trip prevents this mode of operation. Two operable steam generators are therefore adequate.

References

(1) FSAR Section 14.1.12
(2) FSAR Table 4.1-3
(3) FSAR Section 14.1.10

b. The boron injection tank contains not less than 900 gallons 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 ft^3 and no more than 841 ft^3 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. control power shall be removed from the following motor operated valves with the valve in the specified position:

Valves	Position
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

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. The safety injection pump power supply breakers must be racked out when the reactor coolant system temperature is below 350°F and the system is not vented to containment atmosphere.

3.3.1.4 When the reactor is in the cold shutdown condition (except refueling operation when Specification 3.8.1.e applies), both residual heat removal loops must be operable. Except that either the normal or emergency power source to both residual heat removal loops may be inoperable.

a. If one residual heat removal loop becomes inoperable during cold shutdown operation, within 24 hours verify the existance of a method to add make-up water to the reactor coolant system such as charging pumps, safety injection pumps (under adequate operator control to prevent system overpressurization), or primary water (if the reactor coolant system is open for maintenance) as back-up decay heat removal method, or prepare and submit a thirty-day written report pursuant to 6.9.2.b.

b. Restore the inoperable RHR loop to operable status within 14 days or prepare and submit a special report to the Commission pursuant to Specification 6.9.3.1 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the loop to operable status.

c. If both residual heat removal loops become inoperable during cold shutdown operation, close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere prior to the reactor coolant average temperature exceeding 200°F, restore at least one residual heat removal loop to operable status within one hour or prepare and submit a thirty-day written report pursuant to 6.9.2.b. If the reactor coolant average temperature exceeds 200°F with both RHR loops inoperable and the reactor coolant system open for maintenance, provide prompt notification to the Commission pursuant to 6.9.2.a.

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 gallons 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.

Specification 3.3.1.4 requires that two residual heat removal loops be operable when the reactor is in cold shutdown. One RHR loop will normally be in operation in cold shutdown to remove decay heat, maintain the reactor coolant system temperature required for maintenance, ensure even mixing, and produce gradual reactivity changes well within the capability of the operator to control. However, single failure considerations require that two loops be operable.

The limiting conditions for operation associated with specification 3.3.1.4 will allow routine cold shutdown maintenance to be performed, ensure core cooling capability is maintained, and ensure the NRC is kept informed of significant problems with plant cold shutdown temperature control.

Specification 3.3.1.1.h requires control power be removed from the listed valves to prevent inadertent operation caused by control power wiring failures (inadvertent mispositioning of these valves could violate the single failure criteria of the accident analysis). Removal of the control power only will allow these valves to be restored to an operable status quickly when required for recovery from a loss of coolant accident.

indi[illegible]ion available in the containm[illegible]. When core geometry is not being changed at least one source range neutron flux monitor shall be in service.

e. At least one residual heat removal loop shall be operable, refueling cavity water level $\geq$ Plant elevation 272 ft. - 2 in. whenever fuel assemblies are being moved within the reactor pressure vessel, and Tave $\leq$ 140^{o}F.

f. During reactor vessel head removal and while loading and unloading fuel from the reactor, the minimum boron concentration of 1950 ppm shall be maintained in the primary coolant system and verified by sampling once each shift.

g. Direct communication between the control room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

h. Movement of fuel within the core shall not be initiated prior to 100 hours after shutdown.

i. The Spent Fuel Building ventilation system shall be operating when handling irradiated fuel in this area. Prior to moving irradiated fuel assemblies in the spent fuel pool, the ventilation system exhaust shall be aligned to discharge through HEPA and impregnated charcoal filters. When in operation, the exhaust flow of the Containment Purge System shall discharge through HEPA and impregnated charcoal filters. When the Containment Purge System is not in operation at least one automatic containment isolation valve shall be secured in each line penetrating the containment which provides a direct path from the containment atmosphere to the outside atmosphere.

j. If any of the specified limiting conditions for refueling are not met, refueling of the reactor shall cease; work shall be initiated to correct the conditions so that the specified limits are met; and no operations which may increase the reactivity of the core shall be made.

k. The reactor shall be subcritical as required by 3.10.8.3.

3.8.2 The Spent Fuel Building filter system and the Containment Purge filter system shall satisfy the following conditions:

a. The results of the in-place cold DOP and halogenated hydrocarbon tests at greater than 20 percent design flows on HEPA filters and charcoal adsorber banks shall show $\geq$ 99 percent DOP removal and $\geq$ 99 percent halogenated hydrocarbon removal.

b. Verification by way of a laboratory carbon sample analysis from the Spent Fuel Building filter system carbon and the Containment Purge filter system carbon to show $\geq$ 90 percent radioactive methyl iodine removal in accordance with test 5.b of Table 5-1 of ANSI/ASME N509 except that $\geq$ 70 percent relative humidity air is required.

c. All filter system fans shall be shown to operate within $\pm$ 10% of design flow.

d. During fuel handling operations, the relative humidity (R.H.) of the air processed by the refueling filter systems shall be $\leq$ 70 percent.

e. From and after the date that the Spent Fuel Building filter system is made or found to be inoperable for any reason, fuel handling operations in the Spent Fuel Building shall be terminated immediately.

Basis

The equipment and general procedures to be utilized during refueling are discussed in the Final Facility Description and Safety Analysis Report. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety.[1] Whenever changes are not being made in core geometry one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition.

One residual heat removal loop will normally be in operation during refueling operations to remove decay heat, maintain Tave $\leq 140^{\circ}F$, minimize the effect of a boron dilution event, and maintain a uniform boron concentration. The requirement to have one loop operable will allow the loop to be secured for brief periods of time to facilitate fuel movement. The refueling cavity water level specified ensures that there will be at least 23 feet of water above the reactor pressure vessel flange whenever fuel assemblies are being moved within the reactor pressure vessel. This prevents a fuel assembly from becoming partially uncovered while being transported over the vessel flange. This cavity level requirement also provides a large heat sink to ensure adequate time is available to initiate emergency methods to cool the core should the operable RHR loop fail.

The boron concentration of 1950 ppm will keep the core subcritical even if all control rods were withdrawn from the core. During refueling, the reactor refueling cavity is filled with approximately 285,000 gallons of borated water.

The boron concentration of this water at 1950 ppm boron is sufficient to maintain the reactor subcritical by at least 10% $\Delta k/k$ in the refueling condition with all rods inserted, and will also maintain the core subcritical even if no control rods were inserted into the reactor.[2] Weekly checks of refueling water storage tank boron concentration ensure the proper shutdown margin.[3] Direct communications allow the control room operator to inform the manipulator operator of any impending unsafe condition detected from the control board indicators during fuel movement.

In addition to the above safety features, interlocks are utilized during refueling to ensure safe handling. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time.

c.	Deleted		
d.	Inservice Inspection	4.2	After five years of operation
e.	Containment Sample Tendon Surveillance	4.4	Upon completion of the inspection at 5 and 25 years of operation
f.	Post-operational Containment Structural Test	4.4	Upon completion of the test at 3 and 20 years of operation
g.	Fire Protection System	3.14	As specified by limiting condition for operation
h.	Overpressure Protection System Operation	3.1.2.1.e	Within 30 days of operation
i.	Cold Shutdown RHR Loop Operability Requirements	3.3.1.4	As specified by limiting condition for operation

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 64 TO FACILITY OPERATING 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 of June 11, 1980 the staff requested that licensees of all operating pressurized water reactors amend their Technical Specifications with respect to reactor decay heat removal capability. The basis for this request was founded in a number of events where decay heat removal capability was seriously degraded due to inadequate administrative controls when the plants were in shutdown modes of operation. We believe that such degradation would have been prevented if redundancy in decay heat removal capability had been maintained.

In its response dated Septebmer 26, 1980, Carolina Power and Light Company (the licensee) proposed revisions to the Technical Specifications of H. B. Robinson Unit No. 2 to resolve the staff's concerns.

Evaluation

H. B. Robinson Unit No. 2 (Robinson-2) does not have Standard Technical Specifications. Consequently, the format of Technical Specification revisions provided in the staff's letter of June 11, 1980 could not be followed. Appropriate changes have been made, however, to Limiting Conditions of Operations pertaining to the Reactor Cooling System (Section 3.1), Emergency Core Cooling Systems (Section 3.3) and to Refueling (Section 3.8). A comparison of capabilities is presented in Table 1 and shows that, with the new revisions, Robinson-2 meets all of the requirements provided in the staff's guidance.

The licensee has also committed in Section 3.8 to maintain the water level in the refueling cavity to $\geq$ plant elevation 272 feet whenever fuel assemblies are being moved within the reactor pressure vessel. This commitment is in response to the staff's letter of August 15, 1980 in which we requested that Technical Specifications ensure that the core flange is covered by at least 23 feet of water when fuel is being transferred from the reactor so as to prevent excessive radiation exposure. At Robinson-2 the required depth of water is attained when the surface of the water is at plant elevation 272 feet.

The licensee has also clarified a requirement to deenergize certain pumps related to the ECCS system during plant operation. A previous review (Amendment 13 issued October 17, 1975) had identified a need to eliminate the need for operators to leave the control room to reestablish power to some of these pumps to initiate the recirculation cooldown phase associated with a LOCA. The licensee has completed the necessary modifications to permit this control power to be removed and restored from the control room.

The revisions to the decay heat removal systems provide an acceptable response to the staff's concerns related to this capability.

Environmental Consideration

We have determined that the 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 the 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 does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve 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 not 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.

Date: March 8, 1982

TABLE 1

DECAY HEAT REMOVAL CAPABILITIES

MODE OF OPERATION	STANDARD PWR TECHNICAL SPECIFICATIONS	ROBINSON-2 TECHNICAL SPECIFICATIONS
Operation and Startup	All RCP loops in operation	All RCP loop in operation 2 RHR loops operable
Hot Standby[a] (≥350°F)	2 RCP loops operable 1 RCP loop in operation	1 RCP loop in operation All RCP loops in service 2 RHR loops operable
Hot Shutdown (200-300°F)	2 RCP or 2 RHR loops operable 1 RCP or 1 RHR loop in operation	1 RHR loop in operation All RCP loops in service
Cold Shutdown (≤200°F)	2 RCP or 2 RHR loops operable 1 RCP or 1 RHR loop in operation	1 RHR loop normally in operation 2 RHR loops operable (1 RCP loop in operation)[b]
Refueling (Water > 23 ft)	1 RHR loop in operation	1 RHR loop normally in operation
Refueling (Water < 23 ft)	2 RHR loops operable	2 RHR loops operable

(a) Robinson-2 does not use Hot Standby Mode. Hot Shutdown extends from T_{avg} = 200° to 547°F with the reactor not critical. A RHR loop is used < 350°F and a RCP loop > 350°.

(b) Reactor coolant pumps are started in Cold Shutdown during plant heatup.

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. 64 to Facility Operating License No. DPR-23 issued to 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, South Carolina. The amendment is effective as of the date of issuance.

The amendment revises the Technical Specifications related to Reactor Cooling, Emergency Core Cooling, and Refueling to ensure sufficient redundancy in decay heat removal capability is maintained during all modes of plant operation.

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 this 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 that 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 application for amendment dated September 26, 1980, (2) Amendment No. 64 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 29550. 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 Licensing.

Dated at Bethesda, Maryland, this 8th day of March 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing