March 7, 1986

Docket No. 50-261

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Mr. E. E. Utley, Senior Executive Vice President
Power Supply and Engineering & Construction
Carolina Power and Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Utley:

The Commission has issued the enclosed Amendment No. 97 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your request dated November 13, 1985.

The amendment deletes requirements for maintenance of a highly borated inventory in the Boron Injection Tank (BIT) and the associated heat tracing required to maintain the high boron concentration in solution.

During our review and our discussions with your staff (Reference 2 of Safety Evaluation), we required certain confirmatory information. We received your confirmatory response by your letter dated March 6, 1986, and found these responses satisfactory.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly <u>Federal Register</u> notice.

Sincerely,

/s/

Glode Requa, Project Manager
PWR Project Directorate #2
Division of PWR Licensing-A

Enclosures:
1. Amendment No. 97 to DPR-23
2. Safety Evaluation

cc: w/enclosures
See next page

LA:PAD#3	PM:PAD#2	RSB	E.B.	D:RSB	D:PAD#2	OELD
CVogan	GRequa:hc	CBerlinger	GBagchi	ERossi	LRubenstein	M. Karman
3/7/86	3/7/86	3/7/86	3/7/86	3/7/86	3/7/86	3/7/86

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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. 97
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 November 13, 1985, 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. 97, 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

L. S. Rubenstein

Lester S. Rubenstein, Director
PWR Project Directorate #2
Division of PWR Licensing-A

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 7, 1986

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 97 FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.3-2	3.3-2
3.3-3	3.3-3
3.3-4	3.3-4
3.3-14	3.3-14
4.1-6	4.1-6
4.1-10	4.1-10
4.1-11	4.1-11

b. 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.

c. Three safety injections pumps are operable.

d. Two residual heat removal pumps are operable.

e. Two residual heat exchangers are operable.

f. All essential features including valves, interlocks, and piping associated with the above components are operable.

g. 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

h. 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.

i. 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 four 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. 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 four hours.

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]

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 ft^3 and maximum 841 ft^3 of water in the accumulators correspond to an instrument reading of 61.5% and 80.4% of instrument span, respectively.

References

(1) FSAR Section 6.2
(2) FSAR Section 6.3
(3) FSAR Section 14.3.5
(4) FSAR Section 9.3
(5) FSAR Section 9.6.2
(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.

TABLE 4.1-1 (Continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
9.	Analog Rod Position	S (1,2)	R	M	(1) With step counters (2) Following rod motion in excess of six inches when the computer is out of service
10.	Rod Position Bank Counters	S (1,2)	N.A.	N.A.	(1) Following rod motion in excess of six inches when the computer is out of service
11.	Steam Generator Level	S	R	M	
12.	Charging Flow	N.A.	R	N.A.	
13.	Residual Heat Removal Pump Flow	N.A.	R	N.A.	
14.	Boric Acid Tank Level	D (1)	R	N.A.	(1) Bubbler tube rodded weekly
15.	Refueling Water Storage Tank Level	W	R	N.A.	
16.	Deleted				
17.	Volume Control Tank Level	N.A.	R	N.A.	
18.	Containment Pressure	D	R	B/W (1)	(1) Containment isolation valve signal
19.	Deleted by Amendment No. 85				
20.	Boric Acid Makeup Flow Channel	N.A.	R	N.A.	

TABLE 4.1.2

FREQUENCIES FOR SAMPLING TESTS

		Check	Frequency	Maximum Time Between Tests
1.	Reactor Coolant Samples	- Gross Activity (1)	Minimum 1 Per 72 hrs.	3 days
		- Radiochemical (2)	Monthly	45 days
		- Radiochemical for $\bar{E}$ Determination	1 per 6 mos. (6)(7)	6 months
		- Isotopic Analysis for Dose Equivalent I-131 Concentration	1 per 14 days (7)	14 days
		- Isotopic Analysis for Iodine ##?Inlud-ing I-131, I-133 and I-135	a) Once per 4 hours (8) b) One sample (9)	
		- Tritium Activity	Weekly	10 days
		- Cl & O_2	5 day/week	3 days
2.	Reactor Coolant Boron	Boron concentration	Twice/week	5 days
3.	Refueling Water Storage Tank Water Sample	Boron concentration	Weekly	10 days
4.	Boric Acid Tank	Boron concentration	Twice/week	5 days
5.	Spray Additive Tank	NaOH concentration	Monthly	45 days
6.	Accumulator	Boron concentration	Monthly	45 days
7.	Spent Fuel Pit	Boron concentration	Prior to Refueling	NA*
8.	Secondary Coolant	Gross activity	Minimum 1 Per 72 hrs.	3 days
		Isotopic Analysis for Dose Equivalent I-131 Concentration	a) 1 per 31 days (10) b) 1 per 6 months (11)	
9.	Stack Gas Iodine & Particulate Samples	I-131 and particulate radioactivity releases	Weekly (3)	10 days
10.	Steam Generator Samples	Primary to secondary tube leakage	5 days/week	3 days

NOTES TO TABLE 4.1-2

(1) A gross activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of μCi/gram.

(2) A radiochemical analysis shall consist of the quantitative measurement of each radionuclide with half life greater than 30 minutes making up at least 95% of the total activity of the primary coolant.

(3) When iodine or particulate radioactivity levels exceed 10% of the limit in Specification 3.9.2.1, the sampling frequency shall be increased to a minimum of once each day.

(5) Deleted.

(6) Sample to be taken after a minimum of 2EFPD and 20 days of power operation have elapsed since the reactor was last subcritical for 48 hours or longer.

(7) Samples are to be taken in the power operating condition.

(8) Sample taken at all operating conditions whenever the specific activity exceed 1.0 μCi/gram DOSE EQUIVALENT I-131 or $100/\bar{E}$ μCi/gram. These samples are to be taken until the specific activity of the reactor coolant system is restored within its limits.

(9) One sample between 2 and 6 hours following a thermal power change exceeding 15 percent of the rated thermal power within a one-hour period. Samples are required when in the hot shutdown or power operating modes.

(10) Sample whenever that gross activity determination indicates iodine concentrations are greater than 10% of the allowable limit.

(11) Sample whenever the gross activity determination indicates iodine concentrations are below 10 percent of the allowable limit.

NA* - Not applicable.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 97 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

Carolina Power and Light Company (CP&L) has requested deletion of requirements regarding maintenance of a highly borated water inventory in the Boron Injection Tank (BIT) and the associated heat tracing required to maintain the high boron concentration in solution (Ref. 1). CP&L requested approval for two options:

1. Allow the BIT and associated piping to remain in place, but do not fill with highly borated water, and,
2. Remove the BIT and associated piping.

CP&L elected to initially pursue option 1, with the intention that the request for deletion of requirements pertinent to the BIT apply to both options.

CP&L noted, as partial justification:

> "Accident analysis has shown that an inventory of highly borated water in the BIT is not necessary to maintain an acceptable margin of safety to fuel failure during the postulated steamline break event. Furthermore, CP&L has evaluated the effects of this proposed change on the environmental conditions within the containment during the postulated steamline break (SLB) and determined that the equipment's environmental qualification envelope would not be exceeded. Therefore, CP&L requests removal of the TS (Technical Specification) requirement for this highly borated water supply in order to eliminate the substantial maintenance and surveillance requirements involved in maintaining the inventory. This change will also eliminate the potential to block the Safety Injection (SI) Flow Path due to boron precipitation should the heat tracing fail or become damaged."

Reference was also made to the recommendations of Generic Letter 85-16 (Ref. 2). Copies of the affected TS pages were provided (enclosure to Ref. 1), as was an Exxon Nuclear Company (ENC) (Ref. 3) report which provides the results of the supporting analyses. Additional information was provided as documented in References 4 and 5.

The BIT was originally incorporated into the H. B. Robinson design to mitigate the consequences of postulated steamline breaks. It is located downstream of the SI pumps, which in turn are located downstream of the Refueling Water Storage Tank (RWST). During SLB accidents, activation of the SI pumps would force the BIT inventory into the Reactor Coolant System (RCS), followed by water from the RWST. As identified by CP&L and Generic Letter 85-16, the BIT has been a source of problems, and the staff has approved its removal in a number of nuclear power plants.

EVALUATION

Carolina Power and Light Company (CP&L) has requested deletion of requirements regarding maintenance of a highly borated water inventory in the Boron Injection Tank (BIT) and the associated heat tracing required to maintain the high boron concentration in solution (Ref. 1). This request was briefly reviewed by the NRC staff, and a preliminary compilation of items requiring clarification between the staff and CP&L, was prepared. This served as the basis for a telephone conference call, between the staff and CP&L, which provided the necessary clarifications. The compilation of items needing clarification and the CP&L responses are described in Reference 4. Confirmation of the Reference 4 information and further clarifications are provided in Reference 5.

The H. B. Robinson Design.

The H. B. Robinson SG's are equipped with integral flow restrictors at the SG outlet which reduce the size of the largest SLB. Each main steamline is equipped with a fast closing swing disk isolation valve which will block flow in the normal flow direction when allowed to close. Reverse flow is prevented by a second swing disk valve located downstream of the isolation valve. All large steamline piping located outside of containment is effectively open to the environment. The buildings which house the equipment do not have walls.

The H. B. Robinson containment building is classified as a "large dry" containment in that it has a large internal volume, is designed to meet high internal pressure requirements, and does not use ice as a means of pressure reduction in response to a large break loss-of-coolant accident (LOCA).

CP&L Analyses Supporting BIT Removal.

Analyses were provided in support of the BIT removal request which purported that injection of borated water from the BIT is unnecessary with respect to maintenance of an acceptable margin of safety to fuel failure. The analyses were based upon the assumption that the BIT remains in place, but is filled with unborated water. The Minimum Departure from Nucleate Boiling Ratio (MDNBR) was found to be decreased from that obtained with the BIT as presently configured, but a significant margin of safety remained. Slightly more energy was found to be released into containment for SLB's inside containment, but the containment conditions would not result in component temperatures above the established environmental qualification limits.

Proposed TS Changes.

The proposed TS changes delete requirements and surveillances regarding boron concentration in the BIT and operability of heat tracing. BIT volume specification and tank level surveillance requirements are also removed. Option 1 of the CP&L request allows the BIT to remain in place, but isolated from sources of concentrated boric acid and by closed and locked valves. The staff does not believe such action precludes inadvertent concentration of boric acid in the BIT, and therefore requires that periodic sampling of the boron concentration in the BIT be conducted (on a rate schedule to be negotiated with the NRC Project Manager) until such time as inadvertent boron concentration can be precluded. Boron concentration can be considered as precluded when all piping which could serve as a path for boron with a concentration greater than that of the RWST has been physically disconnected from the BIT. This additional option, requested by Reference 1, is summarized in Reference 4. The staff concurs with the additional option.

With the exception as described in the above paragraph, we find the TS changes acceptable.

Superheat

The SG that is affected by an SLB will be depleted in inventory and the tube bundle will become uncovered on the secondary side. Since the tubes will effectively be at the RCS temperature, which will be higher than the SG secondary side temperature, the tendency will be to superheat steam being generated in the SG. The temperature can then be substantially above the qualification temperature of equipment exposed to steam during a SLB. The licensee's calculation of mass and energy releases following a steam line break has incorporated steam superheating effects. The worst break was determined by the licensee to be a full double-ended break at hot zero power assuming no entrainment. We have reviewed the licensee's assumptions in determining level of superheating and the worst break and find them to be conservative. The calculation methodology used by the licensee for mass and energy release (Ref. 6) is under staff review and will be discussed in a separate section. The staff concludes that the mass and energy release data used by the licensee for containment analysis are acceptable. The methodology used is being reviewed as a separate item (see Pg. 4 Calculation Methodology).

The licensee indicated in a meeting on March 6, 1986, that all the safety-related equipment outside containment is not contained in a closed building and is protected by jet shields from direct exposure to the steam. Moreover, the licensee stated that the actuation of all the safety-related equipment is required within 70 seconds while the steam generator tubes would not uncover until after 255 seconds for the worst case steam line break. Based on the information stated above, the staff concludes that superheated steam will not adversely affect the safety-related equipment outside containment.

The licensee has described the analysis which addresses the effects of superheat conditions on the environmental qualification of equipment in Reference 5 and Reference 1.

The equipment items involved have been identified and the most limiting environmental profile has been calculated. The licensee states that based on surface temperature calculations all equipment items are within their qualification profiles and are still considered qualified. The licensee has certified that the surface temperature calculations were performed in accordance with the guidance in NUREG-0588 and adequate time margin, based on the most limiting condition, has been provided in accordance with Regulatory Guide 1.89, Revision 1.

The staff has reviewed the information provided by the licensee and concludes that this issue has been adequately addressed.

Calculation Methodology

A portion of the analyses supporting the CP&L request involve application of a RELAP 5 based calculation methodology (see Ref. 3). This methodology has been submitted to the staff for review (Ref. 6), but staff review is not complete. Methodology review is a separate issue, and is independent of the CP&L BIT modification request.

We have reviewed the CP&L submitted analysis results and conclusions, and find them reasonable when compared to other similar investigations. We have progressed sufficiently far in the methodology review upon which the results are based that we are confident they will be found acceptable. Should the methodology review results be different than expected, further investigation will be required.

The Reference 6 methodology does not address superheat. CP&L has performed scoping calculations which establish that superheat is not a problem to equipment (see prior SER section) and that there are no unusual problems associated with RCS system response (including the core). They have further stated that consideration of superheat will not result in a return to power that significantly differs from calculations of the event which do not include superheat. This is sufficient confirmation for purposes of BIT modification. Further aspects of superheat analysis methodology will be addressed in staff review of Reference 6.

SUMMARY

The staff concurs in the CP&L position that BIT boric acid concentration is a source of problems. For example, staff experience is that problems with SI system reliability have been caused by failure to maintain the boric acid in solution at critical locations. Further, staff experience is that the safety margins are greater than originally thought, as evidenced by an increase in understanding of plant behavior via improved analysis methods. Although removal of the BIT does perturb SLB severity, in general there is sufficient safety margin that the design basis accidents are adequately mitigated. Therefore, no undue risk to the health and safety of the public results, and the benefits are considered to outweigh the deficits.

The situation may be clarified further by considering experience and risk analysis. The general picture is one in which SLB is a low probability, low risk event. Difficulties with high boron concentration have occurred many times, in some cases to the point of disabling SI systems, with obvious impact upon risk. Hence, the practical approach of elimination or modification of the equipment containing high boron concentration solutions is a logical result.

The staff has not identified any features in the H. B. Robinson plant that would negate any of these general conclusions. We, therefore, concur with the CP&L request, subject to the one requirement that BIT boric acid concentration be periodically monitored until such time as the BIT is physically disconnected from potential sources of concentrated boric acid.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

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: March 7, 1986

PRINCIPAL CONTRIBUTOR:

Warren Lyon

REFERENCES

1. Cutter, A. B., "H. B. Robinson Steam Electric Plant, Unit No. 2, Docket No. 50-261/License No. DPR-23, Request for License Amendment, Boron Injection Tank Dilution, Serial NLS-85-332, Letter addressed to Director of Nuclear Reactor Regulation, Carolina Power & Light Company, Nov. 13, 1985.

2. Thompson, Hugh L., "High Boron Concentrations," Letter addressed to all licensees of operating reactors and applicants for an operating license, Generic Letter 85-16, USNRC, Aug. 23, 1985.

3. "Analysis of the Steamline Break Event with Boron Injection Tank Removal or Dilution to Zero Concentration Boric Acid for H. B. Robinson Unit 2," XN-NF-85-17(P) (Proprietary), Exxon Nuclear Company, Inc., May 1985.

4. Requa, Glode, "Report of Conference Call with Carolina Power & Light Regarding H. B. Robinson - 2 BIT Removal," Docket No. 50-261, March 5, 1986.

5. Zimmerman, Sherwood R., "H. B. Robinson Steam Electric Plant, Unit No. 2, Docket No. 50-261/License No. DPR-23, Boron Injection Tank (TAC No. 60301), Letter from Carolina Power and Light to Director of Nuclear Reactor Regulation, NRC, Serial NLS-86-083," March 6, 1986.

6. "Steamline Break Methodology for PWR's," XN-NF-84-93(P) Exxon Nuclear Company, Inc., November 1984.