December 26, 1990

Docket No. 50-400

DISTRIBUTION
See attached list

Mr. Lynn W. Eury
Executive Vice President
Power Supply
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Eury:

SUBJECT: ISSUANCE OF AMENDMENT NO. 23 TO FACILITY OPERATING LICENSE NO. NPF-63 REGARDING PRESSURE-TEMPERATURE LIMITS - SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1, (TAC NO. 77587)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 23 to Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit 1. This amendment consists of changes to the Technical Specifications in response to your request dated September 10, 1990, as supplemented November 20, 1990.

The amendment revises the reactor coolant system (RCS) pressure/temperature limits of Technical Specifications (TS) 3.4.9.1 and 3.4.9.2 to protect the reactor pressure vessel (RPV) from the potential of brittle fracture as the RPV neutron exposure increases from three (3) effective full power years (EFPY) to five (5) EFPY. In addition, the low pressure overpressure protection (LTOP) set points are adjusted accordingly and an effective lower temperature limit for usage of the LTOP set points has been added to ensure that the set points are used only in the region where the system can provide the necessary protection.

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

Sincerely,

Original Signed By:

Richard A. Becker, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 23 to NPF-63
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	:LA:PD21:DRPR	:PM:PD21:DRPR	:D:PD21:DRPR	:	:	:	:
NAME	:PAnderson:	: RBecker:sw	: EAdensam	:	:	:	:
DATE	:12/10/90	:12/18/90	:12/21/90	:	:	:	:

OFFICIAL RECORD COPY
Harris Am

DF01
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Mr. L. W. Eury
Carolina Power & Light Company

Shearon Harris

cc:

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AMENDMENT NO. 23 TO FACILITY OPERATING LICENSE NO. NPF-63 - HARRIS, UNIT 1

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OC/LFMB

cc: Harris Service List

310047

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

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 23
License No. NPF-63

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

   A. The application for amendment by Carolina Power & Light Company, (the licensee), dated September 10, 1990, as supplemented November 20, 1990, 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 2.C.(2) of Facility Operating License No. NPF-63 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 23 , are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION
Original Signed By:

Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 26, 1990

SUBJECT TO INCLUDING STATEMENT REGARDING NOV 20 SUPP. INFO IN SAFETY EVAL OR SHOLLY MEMO.

OFC	LA:PD21:DRPR	PM:PD21:DRPR	OGC	D:PD21:DRPR			
NAME	PAnderson	RBecker:sw	E HOLLER	EAdensam			
DATE	11/27/90	11/28/90	12/17/90	12/26/90			

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ATTACHMENT TO LICENSE AMENDMENT NO. 23

FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
viii	viii
3/4 4-35	3/4 4-35
3/4 4-36	3/4 4-36
3/4 4-38	3/4 4-38
3/4 4-40	3/4 4-40
3/4 4-41	3/4 4-41
B 3/4 B-7	B 3/4 B-7

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

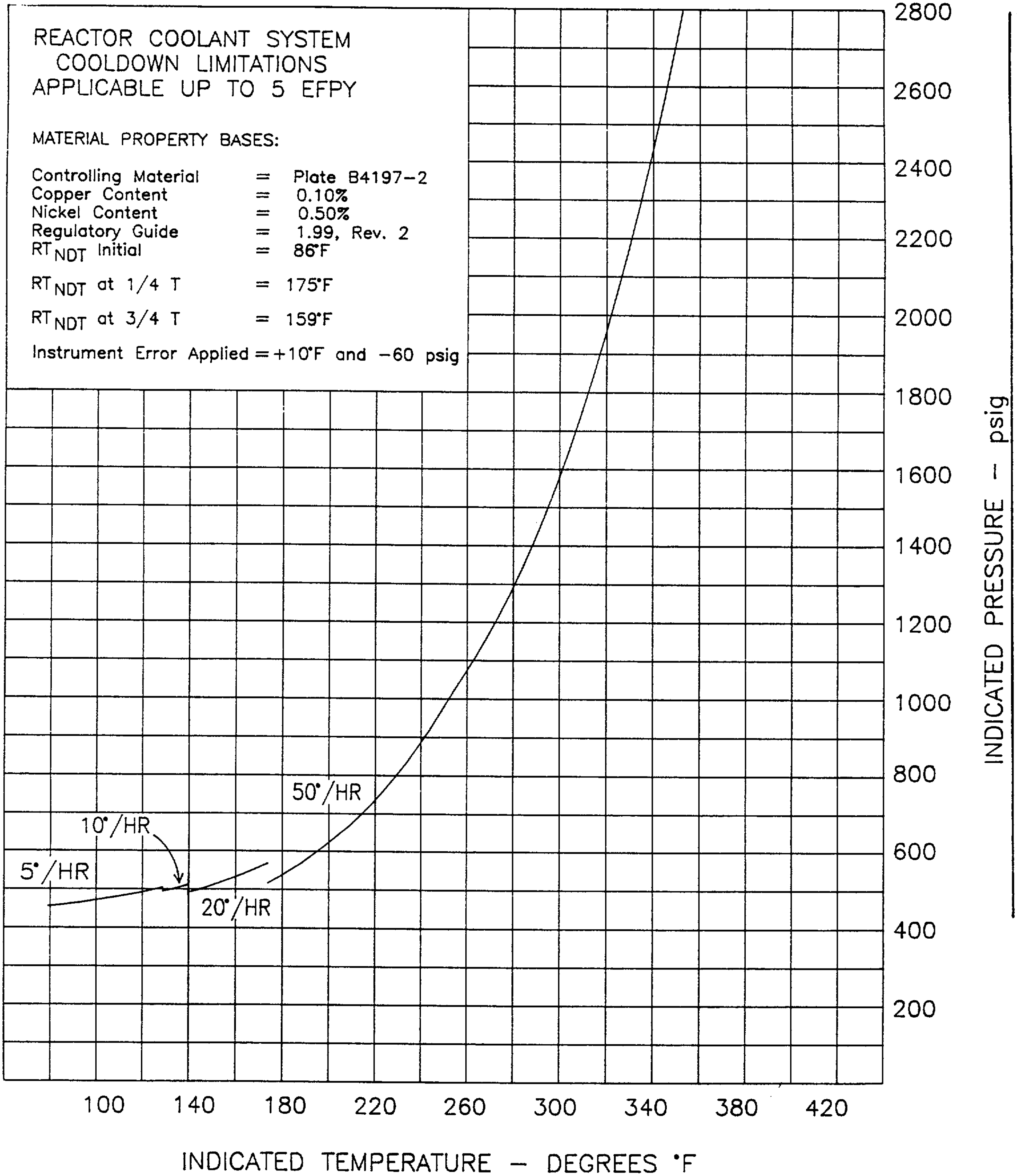


FIGURE 3.4-2

REACTOR COOLANT SYSTEM

COOLDOWN LIMITATIONS - APPLICABLE UP TO 5 EFPY

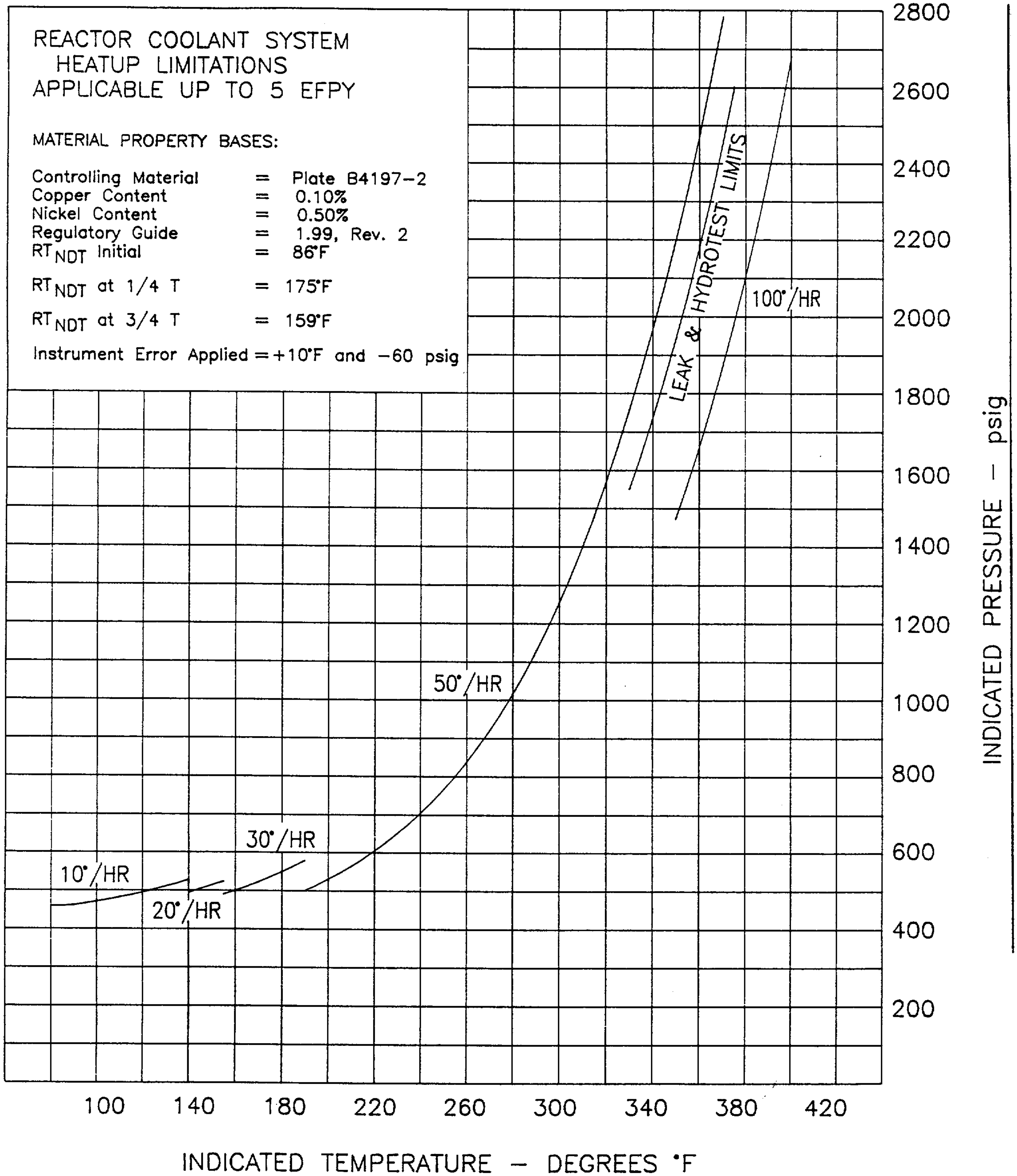


FIGURE 3.4-3
REACTOR COOLANT SYSTEM
HEATUP LIMITATIONS - APPLICABLE UP TO 5 EFPY

TABLE 4.4-6

MAXIMUM COOLDOWN AND HEATUP RATES
FOR MODES 4, 5, AND 6 (WITH REACTOR VESSEL HEAD ON)

COOLDOWN RATES

TEMPERATURE*	COOLDOWN IN ANY 1 HOUR PERIOD**
350-175°F	50°F
175-140°F	20°F
140-130°F	10°F
< 130°F	5°F

HEATUP RATES

TEMPERATURE*	HEATUP IN ANY 1 HOUR PERIOD**
< 140°F	10°F
140-155°F	20°F
155-190°F	30°F
190-350°F	50°F

*Temperature range used should be based on the lowest RCS cold leg value.

**Temperature used should be based on lowest RCS cold leg value except when no RCP is in operation; then use an operating RHR heat exchanger outlet temperature.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.4 At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.9 square inches, or

* b. Two power-operated relief valves (PORVs) with setpoints which do not exceed the limits established in Figure 3.4-4.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 325°F, MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

a. With one PORV inoperable, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.9 square inch vent within the next 8 hours.

b. With both PORVs inoperable, depressurize and vent the RCS through at least a 2.9 square inch vent within 8 hours.

c. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

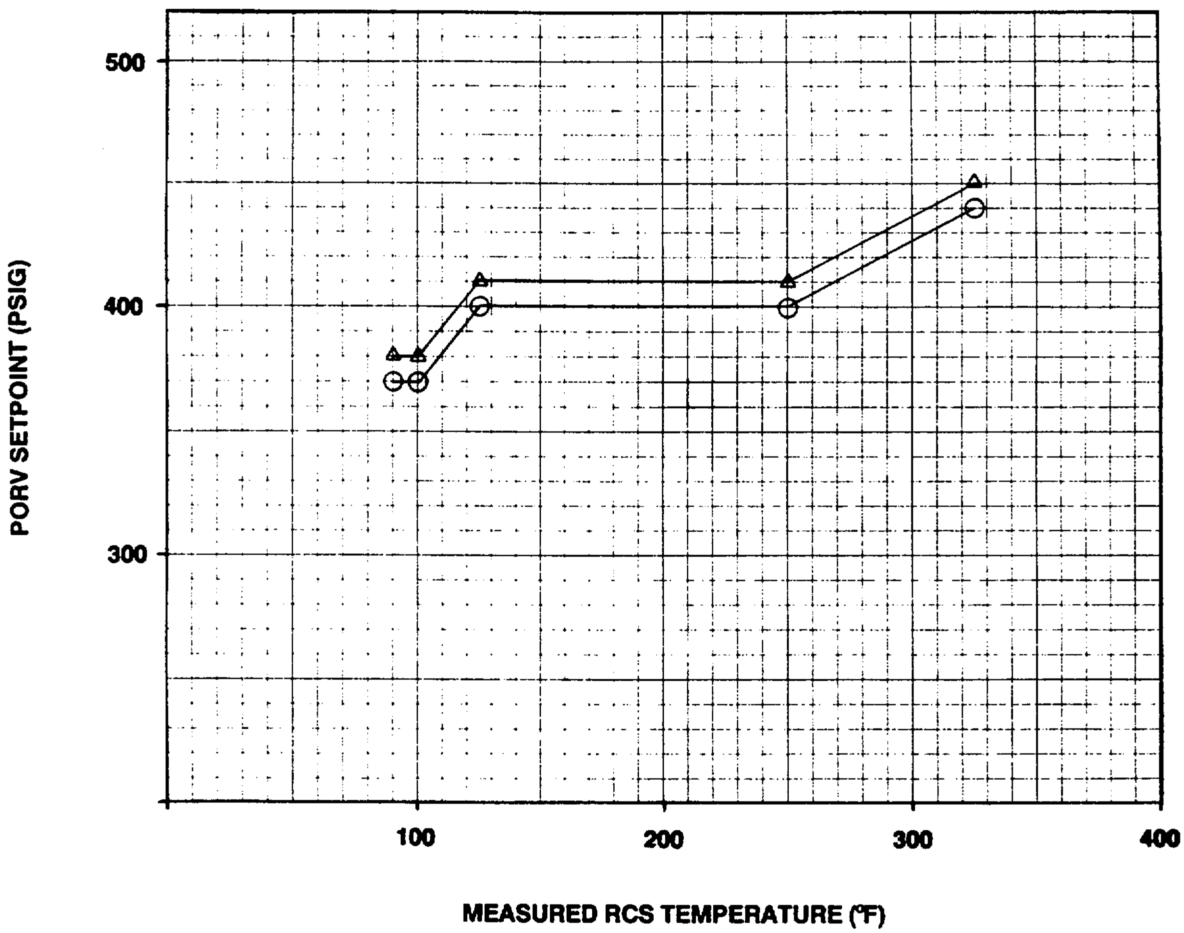
4.4.9.4.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

* Credit may only be taken for the setpoints when the RCS cold leg temperature $\geq$ 90°F.



RCS TEMP °F	LOW PORV * PSIG ○	HIGH PORV * PSIG △
90	370	380
100	370	380
125	400	410
250	400	410
300	427	437
325	440	450

* VALUES BASED ON 5 EFPY REACTOR VESSEL DATA AND CONTAINS MARGINS OF -10°F AND +60 PSIG FOR POSSIBLE INSTRUMENT ERROR

FIGURE 3.4-4

MAXIMUM ALLOWED PORV SETPOINT FOR THE LOW TEMPERATURE OVERPRESSURE SYSTEM

PRESSURE/TEMPERATURE LIMITS (Continued)

b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 625°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness testing of the ferritic materials in the reactor vessel was performed in accordance with the 1971 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code. These properties are then evaluated in accordance with the NRC Standard Review Plan.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 5 effective full power years (EFPY) of service life. The service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content, and nickel content of the material in question, can be predicted using Figure B 3/4.4-1 and the value of ΔRT_{NDT} computed by Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 23 TO FACILITY OPERATING LICENSE NO. NPF-63

CAROLINA POWER & LIGHT COMPANY

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

By letter dated September 10, 1990, as supplemented November 20, 1990, Carolina Power and Light Company (CP&L) submitted a request for changes to the pressure-temperature (P-T) limits in the Shearon Harris Nuclear Power Plant, Unit 1 (Harris), Technical Specifications (TS), Section 3.4. The November 20, 1990, letter provided clarifying information that did not change the initial determination of no significant hazards consideration as published in the Federal Register (55 FR 40462) on October 3, 1990. This revision changes the P-T limits from 3 to 5 effective full power years (EFPY). The proposed P-T limits were developed based on Regulatory Guide (RG) 1.99, Revision 2, and they provide limits for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide TS for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the TS. The P-T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear plants in the U.S. Appendices G and H to 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P-T limits. An acceptable method for constructing the P-T limits is described in Standard Review Plan (SRP) Section 5.3.2.

To evaluate the P-T limits, the staff uses the following NRC regulations and guidance: Appendices G and H to 10 CFR Part 50; the American Society of Testing Materials (ASTM) Standards and the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Revision 2; SRP Section 5.3.2; and Generic Letter 88-11.

Appendix G to 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code.

In particular, Appendix G specifies that the beltline materials in the surveillance capsules be tested in accordance with Appendix H to 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards for surveillance testing requirements. These surveillance tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Revision 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H to 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor vessel beltline.

2.0 EVALUATION

To review the licensee's request, the staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Harris reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Revision 2. The staff determined that the material with the highest ART at 5 EFPY at 1/4T and 3/4T (T = reactor vessel beltline thickness) was intermediate shell plate B4197-2 with 0.1% copper (Cu), 0.5% nickel (Ni), and an initial RT_{ndt} of 86°F.

For the limiting beltline material, plate B4197-2, the staff calculated the ART to be 174.1°F at 1/4T and 158.3°F at 3/4T. The staff used a neutron fluence of 5.49E18 (E18=10^{18}) neutrons/square centimeter (n/cm^2) at 1/4T and 2.17E18 n/cm^2 at 3/4T.

The licensee used the method in RG 1.99, Revision 2, to calculate an ART of 175°F at 1/4T and 159°F at 3/4T for the same limiting metal. Substituting the ART of 175°F into equations in SRP 5.3.2, the staff verified that the proposed P-T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G to 10 CFR Part 50.

The licensee has not removed any surveillance capsules from the Harris reactor vessel. According to the Harris Final Safety Analysis Report (FSAR), the first surveillance capsule will be removed at about 3 EFPY, which will be the next refueling outage. The staff has determined that all surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

In addition to beltline materials, Appendix G to 10 CFR Part 50 also imposes P-T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the pre-service system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 0°F, the staff has determined that the proposed P-T limits satisfy Section IV.A.2 of Appendix G. Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. The material with the lowest initial USE is plate B4197-2 with an initial USE of 74 ft-lb. Using the method in RG 1.99, Revision 2, the predicted Charpy USE of the plate at the end of life will be greater than 50 ft-lb and, therefore, is acceptable.

The staff concludes that the proposed P-T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 5 EFPY because the limits conform to the requirements of Appendices G and H to 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Revison 2, to calculate the ART. Hence, the proposed P-T limits may be incorporated into the Harris TS.

Low temperature overpressure protection (LTOP) is provided by the pressurizer overpressure relief valves (PORVs). These PORVs are set to open at a pressure low enough to prevent violation of the Appendix G heatup and cooldown curves should an RCS pressure transient occur during low temperature operations. The licensee identified the most limiting overpressure transients analyzed to determine the PORV setpoints for LTOP in Amendment 19. The same setpoints are utilized for the five EFPY. An analysis was performed by the licensee to ensure that the pressure overshoot beyond the LTOP setpoint is such that the new Appendix G pressure-temperature curves are not exceeded during the transient. The licensee stated that the new curves, in conjunction with the associated TS changes in the heatup and cooldown ranges and the existing LTOP system setpoints, provide the required assurance that the reactor pressure vessel is protected from brittle fracture up to five EFPY of operation.

In addition, the proposed change will add an effective lower temperature limit to TS Figure 3.4-4 for use of the LTOP system to maintain protection of the RCS. This modification would extend the low and high PORVs setpoint lines from 100°F down to 90°F. The limit was added to ensure that LTOP setpoints are used only in a region where the system can provide the necessary protection. Below 90°F, overpressurization protection is provided by administrative procedure that will implement TS 3.4.9.4(a) in which the reactor coolant system is depressurized with a vent area greater than 2.9 square inches. All other LTOP setpoint design bases remain the same as those used previously for the three EFPY setpoint determination. The staff reviewed the current and the extended PORV's setpoints to the new derived pressure-temperature curves and concluded that the setpoints for the PORVs are within the allowable range and are, therefore, acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes the surveillance requirements. 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 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.

5.0 CONCLUSION

The Commission made a proposed determination that this amendment involves no significant hazards consideration which was published in the Federal Register (55 FR 40462) on October 3, 1990, and consulted with the State of North Carolina. No public comments or requests for hearing were received, and the State of North Carolina did not have any comments.

The staff has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

1. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988

2. NUREG-0800, Standard Review Plan, Section 5.3.2, Pressure-Temperature Limits

3. Shearon Harris, Unit 1, Final Safety Analysis Report

4. Shearon Harris, Unit 1, Technical Specifications, January 1987

5. Letter from A.B. Cutter, CP&L to USNRC Document Control Desk, Subject: Request for License Amendment Reactor Coolant System Pressure-Temperature Limits, September 10, 1990

6. Generic Letter 88-11, NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations, USNRC, July 12, 1988

Dated: December 26, 1990

Principal Contributors: J. Tsao
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