March 7, 1997

Mr. W. R. Robinson, Vice President
Shearon Harris Nuclear Power Plant
Carolina Power & Light Company
Post Office Box 165, Mail Code: Zone 1
New Hill, North Carolina 27562-0165

SUBJECT: ISSUANCE OF AMENDMENT NO. 68 TO FACILITY OPERATING LICENSE NO. NPF-63 REGARDING CHEMISTRY DATA CHANGES FOR TS FIGURES 3.4-2, AND 3.4-3 - SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 (TAC NO. M97542)

Dear Mr. Robinson:

The Nuclear Regulatory Commission has issued Amendment No. 68 to Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit No. 1. This amendment changes the Technical Specifications (TS) in response to your request dated December 30, 1996.

The amendment revises chemistry data shown on TS Figures 3.4-2 and 3.4-3 for the Reactor Coolant System's Pressure/Temperature Limits and the associated Bases.

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:
Ngoc B. Le, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:
1. Amendment No. 68 to NPF-63
2. Safety Evaluation

cc w/enclosures:
See next page

(* See SER memo dated 2/24/97 from KWichman to MReinhart)

FILENAME - G:\HARRIS\HAR97542.AMD

OFFICE	PM:PDII-1	LA:PDII-1	D:PDII-1	OGC	A/BC:EMCB*
NAME	NLe	Dunnington	MReinhart	C Marco	KWichman
DATE	2/27/97	2/27/97	3/7/97	2/28/97	2/24/97*
COPY	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No

CP-1

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 68
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 December 30, 1996, 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. 68, 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 and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Mark Reinhart

Mark Reinhart, Acting 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: March 7, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 68

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
3/4 4-35	3/4 4-35
3/4 4-36	3/4 4-36
B 3/4 4-6	B 3/4 4-6
B 3/4 4-8	B 3/4 4-8
B 3/4 4-11	B 3/4 4-11
B 3/4 4-12	B 3/4 4-12
B 3/4 4-14	B 3/4 4-14

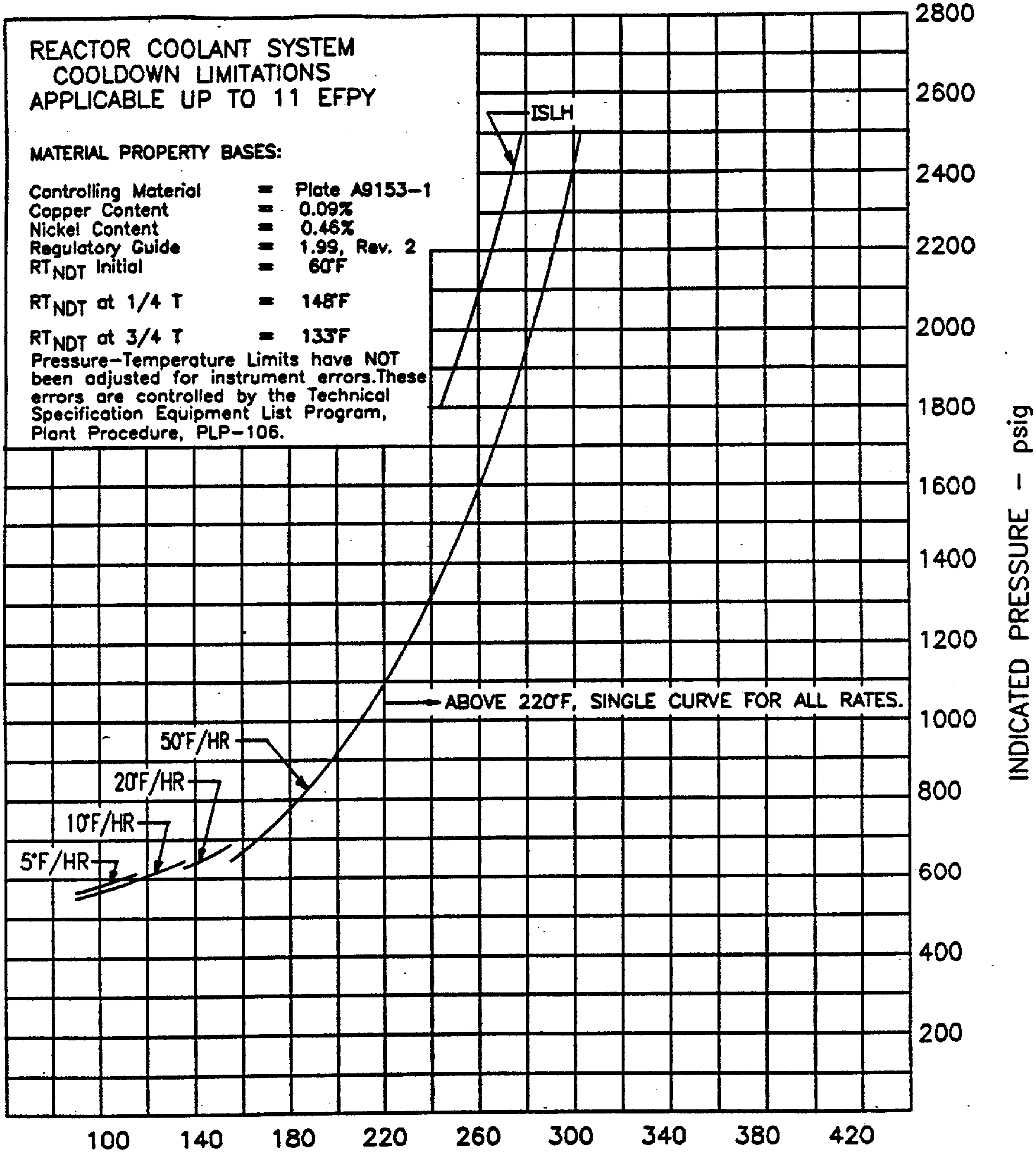

FIGURE 3.4-2

REACTOR COOLANT SYSTEM

COOLDOWN LIMITATIONS - APPLICABLE UP TO 11 EFPY

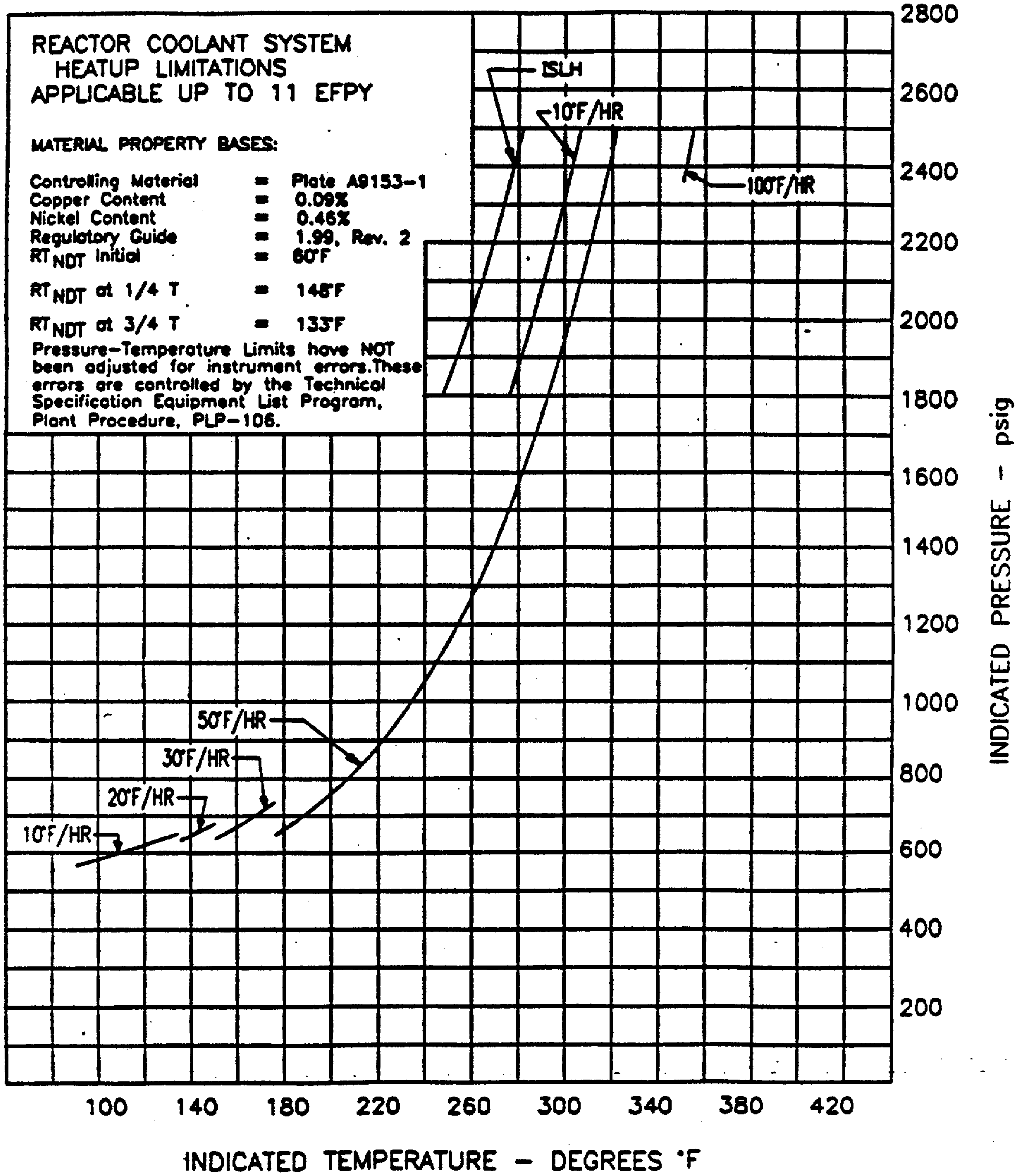

FIGURE 3.4-3

REACTOR COOLANT SYSTEM

HEATUP LIMITATIONS - APPLICABLE UP TO 11 EFPY

SPECIFIC ACTIVITY (Continued)

distinction between the radionuclides above and below a half-life of 15 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture occur, since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, and 10 CFR 50 Appendix G and H. 10 CFR 50, Appendix G also addresses the metal temperature of the closure head flange and vessel flange regions. The minimum metal temperature of the closure flange region should be at least 120°F higher than the limiting RT NDT for these regions when the pressure exceeds 20% (621 psig for Westinghouse plants) of the preservice hydrostatic test pressure. For Shearon Harris Unit 1, the minimum temperature of the closure flange and vessel flange regions is 120°F because the limiting RT NDT is 0°F (see Table B 3/4 4-1). The Shearon Harris Unit 1 cooldown and heatup limitations shown in Figures 3.4-2 and 3.4-3 and Table 4.4-6 are not impacted by the 120°F limit.

1. The reactor coolant temperature and pressure and system cooldown and heatup rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 and Table 4.4-6 for the service period specified thereon:

    a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	GRADE	HEAT NO	Cu (wt.%)	Ni (wt.%)	T_{NDT} (°F)	INITIAL RT_{NDT} (°F)	CHARPY UPPER SHELF ENERGY TRANSVERSE FT-LB
Closure Hd. Dome	A533,B,CL1	A9213-1	-	-	-10	8	114
Head Flange	A508, CL2	5302-V2	-	-	0	0	135
Vessel Flange	"	5302-V1	-	-	-10	-8	110
Inlet Nozzle	"	438B-4	-	-	-20	-20	169
" "	"	438B-5	-	-	0	0	128
" "	"	438B-6	-	-	-20	-20	149
Outlet Nozzle	"	439B-4	-	-	-10	-10	151
" "	"	439B-5	-	-	-10	-10	152
" "	"	439B-6	-	-	-10	-10	150
Nozzle Shell	A533B,CL1	C0224-1	.12	-	-20	-1	90
" "	"	C0123-1	.12	-	0	42	84
Inter. Shell*	"	A9153-1	.09	.46	-10	60	83
" "*	"	B4197-2	.09	.50	-10	91	71
Lower Shell*	"	C9924-1	.08	.47	-10	54	98
" "*	"	C9924-2	.08	.47	-20	57	88
Bottom Hd. Torus	"	A9249-2	-	-	-40	14	94
" " Dome		A9213-2	-	-	-40	-8	125
Weld (Inter & Lower Shell Vertical Weld Seams)*		4P4784	.05	.91	-20	-20	>94
Weld (Inter. to Lower Shell Girth Seam)*		5P6771	.03	.94	-80	-20	80

* For Beltline Materials, copper and nickel valves are "best estimates".

PRESSURE/TEMPERATURE LIMITS (Continued)

The cooldown and heatup limits of Figures 3.4-2 and 3.4-3 are based upon an adjusted RT_{NDT} (initial RT_{NDT} plus predicted adjustments for this shift in RT_{NDT} plus margin).

In accordance with Regulatory Guide 1.99, Revision 2, the results from the material surveillance program, evaluated according to ASTM E185, may be used to determine ΔRT_{NDT} when two or more sets of credible surveillance data are available. Capsules will be removed and evaluated in accordance with the requirements of ASTM E185-82 and 10 CFR Part 50, Appendix H. The results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The cooldown and heatup curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various cooldown and heatup rates are calculated using methods derived from Appendix G in Section XI of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section XI as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which cooldown and heatup curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the

PRESSURE/TEMPERATURE LIMITS (Continued)

metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T - RT_{NDT} + 160)] \qquad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\, K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress.

K_{It} = the stress intensity factor caused by the thermal gradients.

K_{IR} = constant provided by the Code as a function of temperature relative to the RT_{NDT} of the material.

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice leak and hydrostatic (ISLH) test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

PRESSURE/TEMPERATURE LIMITS (Continued)

heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

The composite curves for the heatup rate data and the cooldown rate data in Figures 3.4-2 and 3.4-3 have not been adjusted for possible errors in the pressure and temperature sensing instruments. However, the heatup and cooldown curves in plant operating procedures have been adjusted for these instrument errors. The instrument errors are controlled by the Technical Specification Equipment List Program, Plant Procedure PLP-106.

"ISLH" pressure-temperature (P-T) curves may be used for inservice leak and hydrostatic tests with fuel in the reactor vessel. However, ISLH tests required by the ASME code must be completed before the core is critical.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

LOW TEMPERATURE OVERPRESSURE PROTECTION

The OPERABILITY of two PORVs or an RCS vent opening of at least 2.9 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 325°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than 50°F above the RCS cold leg temperatures, or (2) the start of a charging/safety injection pump and its injection into a water-solid RCS.

The maximum allowed PORV setpoint for the Low Temperature Overpressure Protection System (LTOPS) is derived by analysis which models the performance

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

CAROLINA POWER & LIGHT COMPANY

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

By letter dated December 30, 1996, the Carolina Power & Light Company (the licensee) submitted a request for changes to the Shearon Harris Nuclear Power Plant, Unit 1 (SHNPP), Technical Specifications (TS). The requested changes would revise (1) chemistry data shown on TS Figures 3.4-2 and 3.4-3 for TS 3/4.4.9, "Pressure/Temperature Limits" and (2) its associated Bases.

In the December 30, 1996, submittal, the licensee submitted slight changes to some of the best estimate chemistry values for the beltline materials. These changes resulted from cooperative data sharing activities in response to Generic Letter (GL) 92-01, Revision 1, Supplement 1. Specifically, the licensee reviewed available reactor pressure vessel (RPV) beltline material data, and identified plants that have the same weld heat identifications as those contained in the Harris vessel (sister plants). In addition to sharing data with sister plants, the licensee also shared data with the Westinghouse Owners Group (WOG) in order to determine best estimate chemistry for the beltline materials. As a result of these assessments, the nickel content of the limiting material increased from 0.45% to 0.46%. This slight change in nickel content does not affect the pressure-temperature (P-T) limits in the SHNPP TS. However, some documentation changes in TS Figures 3.4-2 and 3.4-3 are necessary. The amendment request also revises the associated Bases to reflect chemistry revisions and minor material property changes for the beltline weld fabricated from weld wire heat 5P6771. In addition, the licensee submitted revisions to the Bases in order to comply with 10 CFR 50, Appendix G rule changes.

2.0 EVALUATION

The staff evaluates the P-T limits based on the following NRC regulations and guidance: Appendix G to 10 CFR Part 50; GL 88-11 and 92-01; Regulatory Guide (RG) 1.99, Rev. 2; and Standard Review Plan (SRP) Section 5.3.2. Appendix G to 10 CFR Part 50 requires that P-T limits for the reactor vessel must be at least as conservative as those obtained by Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel

Code. GL 88-11 requires that licensees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on the adjusted reference temperature (ART) of reactor vessel materials. The ART is defined as the sum of initial nil-ductility transition reference temperature (RT_{NDT}) of the material, the increase in RT_{NDT} caused by neutron irradiation, and a margin to account for uncertainties in the prediction method. The increase in RT_{NDT} is calculated from the product of a chemistry factor (CF) and a fluence factor. The CF may be calculated using credible surveillance data, obtained by the licensee's surveillance program, as directed by Position 2 of RG 1.99, Rev. 2. If credible surveillance data are not available, the CF is calculated dependent upon the amount of copper and nickel in the vessel material as specified in Table 1 of RG 1.99, Rev. 2. GL 92-01 requires licensees to submit reactor vessel materials data, which the staff uses in the review of the P-T limits submittals.

SRP 5.3.2 provides guidance on calculation of the P-T limits using linear elastic fracture mechanics methodology specified in Appendix G to Section III of the ASME Code. The linear elastic fracture mechanics methodology postulates sharp surface defects that are normal to the direction of maximum stress and have a depth of one-fourth of the reactor vessel beltline thickness (1/4T) and a length of 1-1/2 times the beltline thickness. The critical locations in the vessel for this methodology are the 1/4T and 3/4T locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

The limiting material in the Shearon Harris reactor vessel is the intermediate shell plate that was fabricated using heat number A9153-1. The licensee determined that the slight increase in the nickel content of the limiting material from 0.45% to 0.46% did not affect the P-T limits. The staff performed independent calculations of the CF values for the beltline materials using the revised best estimate chemistry and the methodology in RG 1.99, Revision 2. Based on these calculations, the staff verified that the CF of the limiting material remained unchanged. Since the revised best estimate chemistry does not affect the limiting material, it will not affect the P-T limits. Therefore, Amendment 38 dated August 20, 1993, which approved the P-T limits curves to 11 effective full power years (EFPY), remains valid.

Some beltline materials experienced a slight reduction in copper and nickel content which resulted in a CF that remained the same or could have been reduced. For those CF values that could have been reduced, the licensee elected to maintain the current conservative CF value.

Assessment of the newly acquired data resulted in the determination of a revised dropweight temperature (T_{NDT}) for the weld fabricated from weld wire heat 5P6771. The original T_{NDT} value was -20°F, and the revised value is -80°F as determined from the unirradiated surveillance weldment test data. However, the revised T_{NDT} value does not affect the initial reference temperature since it is based on the surveillance weldment temperature for the Charpy 50 ft-lb value minus 60°F. The current initial reference temperature value of -20°F remains valid, and the weld is not the limiting reactor vessel beltline material. Therefore, the P-T limits are not affected.

Based on the above, the staff finds the proposed revisions to the applicable Bases sections for In-Service Leak & Hydrotests (ISLH), and the ASME Code Section to be used for the development of P-T limits acceptable since the revisions are consistent with the amended rule for 10 CFR 50, Appendix G, that was effective January 18, 1996.

The staff also notes that the revisions to the best estimate chemistry do not affect any significant results relative to the withdrawal schedule or testing of any surveillance capsules.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of North Carolina official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The 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. The NRC 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 the amendment involves no significant hazards consideration, and there has been no public comment on such finding (62 FR 4342). Accordingly, the 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 the amendment.

5.0 CONCLUSION

The Commission 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 the 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. Code of Federal Regulations, Title 10, Part 50, Appendix G, Fracture Toughness Requirements

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

5. ASME Boiler and Pressure Vessel Code, Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components"

6. December 30, 1996, Letter from W. R. Robinson to USNRC Document Control Desk, Subject: Shearon Harris Nuclear Power Plant Request for License Amendment RCS Pressure/Temperature Limits

7. August 20, 1993, Letter from N. Le to W. S. Orser, Subject: Issuance of Amendment No. 38 To Facility Operating License No. NPF-63 Regarding Technical Specifications Change to Pressure-Temperature Limits-Shearon Harris Nuclear Power Plant, Unit 1 (TAC NO. M85876)

8. November 16, 1995, Letter from J. W. Donahue for W. R. Robinson to USNRC Document Control Desk, Subject: Shearon Harris Nuclear Power Plant Response to Generic Letter 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity"

Principal Contributor: Andrea Lee

Date: March 7, 1997

AMENDMENT NO. 68 TO FACILITY OPERATING LICENSE NO. NPF-63 - HARRIS, UNIT 1

Docket File
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