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

July 28, 2000

Mr. James Scarola, 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: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 - ISSUANCE OF AMENDMENT REGARDING PRESSURE/TEMPERATURE LIMITS (TAC NO. MA8642)

Dear Mr. Scarola:

The Nuclear Regulatory Commission has issued Amendment No. 100 to Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit No. 1 (HNP), in response to your request dated April 12, 2000, as supplemented on June 2, 2000. This amendment revises Technical Specification (TS) 3/4.4.9.2, "Pressure/Temperature (P-T) Limits - Reactor Coolant System," and TS 3/4.4.9.4, "Overpressure Protection System," and the associated Bases. Specifically, the amendment incorporates results of the Reactor Vessel Surveillance Program capsule analysis and an exemption from 10 CFR 50.60(a), based on American Society of Mechanical Engineers Code Case N-640. The exemption to use Code Case N-640 was previously approved by separate correspondence.

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

Sincerely,

Richard J. Laufer

Richard J. Laufer, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:
1. Amendment No. 100 to NPF-63
2. Safety Evaluation

cc w/enclosures:
See next page

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AMENDMENT NO. 100 TO FACILITY OPERATING LICENSE NO. NPF-63 - HARRIS, UNIT 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. 100
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 April 12, 2000, as supplemented on June 2, 2000, 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. 100 , 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

Richard P. Correia

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: July 28, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 100

FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
viii	viii
xiv	xiv
3/4 4-34	3/4 4-34
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 4-41	3/4 4-41
B 3/4 4-6	B 3/4 4-6
B 3/4 4-7	B 3/4 4-7
B 3/4 4-9	B 3/4 4-9
B 3/4 4-11	B 3/4 4-11
B 3/4 4-12	B 3/4 4-12
B 3/4 4-13	B 3/4 4-13

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

LIMITING CONDITION FOR OPERATION

3.4.9.2 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate as shown on Table 4.4-6.

b. A maximum cooldown rate as shown on Table 4.4-6.

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: MODES 4, 5, and 6 with reactor vessel head on.

ACTION:

With any of the pressure limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; if the pressure and temperature limit lines shown on Figure 3.4-2 and 3.4-3 were exceeded, perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or maintain the RCS T_{avg} and pressure at less than 200°F and 500 psig, respectively.

SURVEILLANCE REQUIREMENTS

4.4.9.2.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.2.2 Deleted from Technical Specifications. Refer to the Technical Specification Equipment List Program, plant procedure PLP-106.

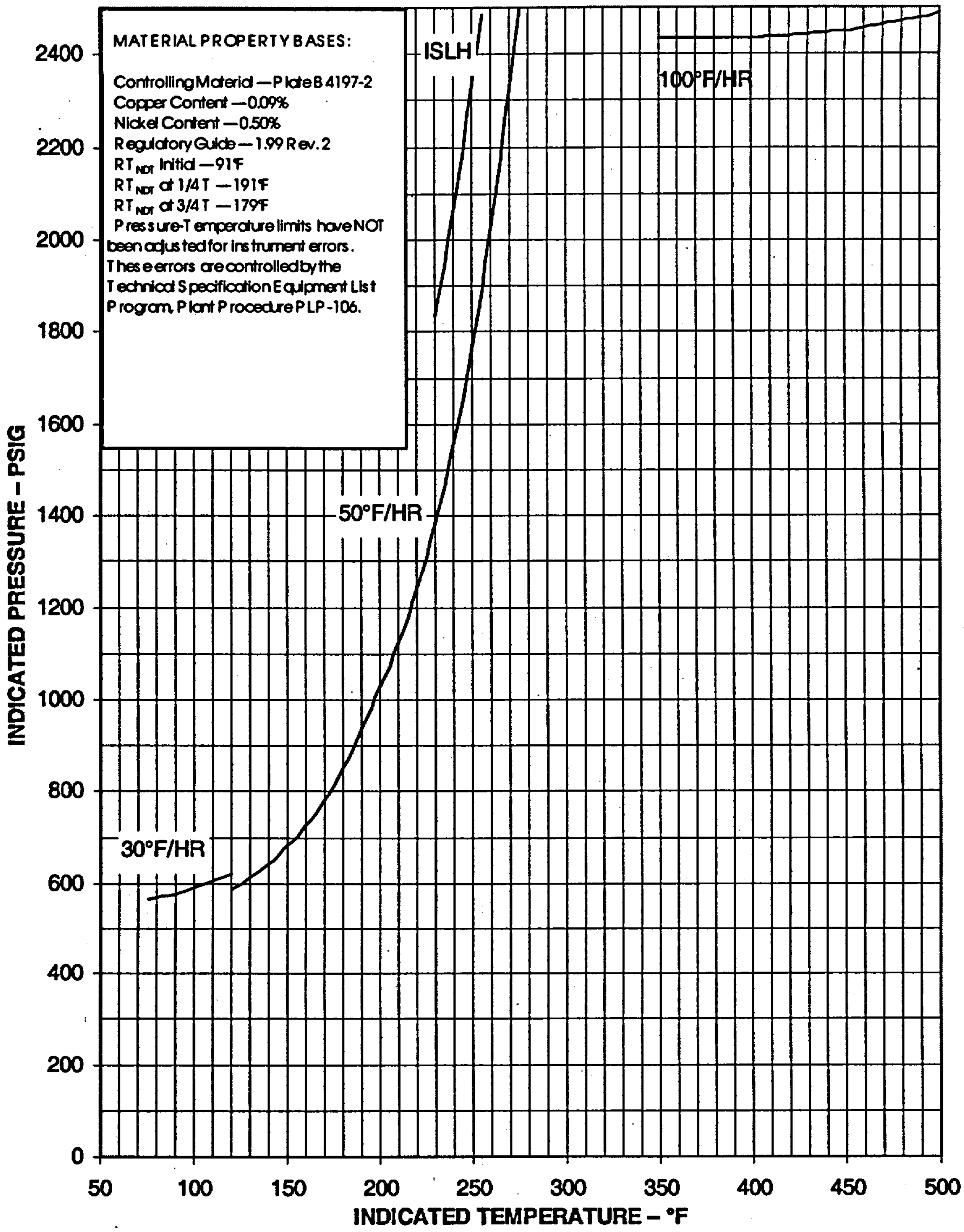


FIGURE 3.4-2
REACTOR COOLANT SYSTEM
COOLDOWN LIMITATIONS – APPLICABLE TO UP TO 36 EFPY

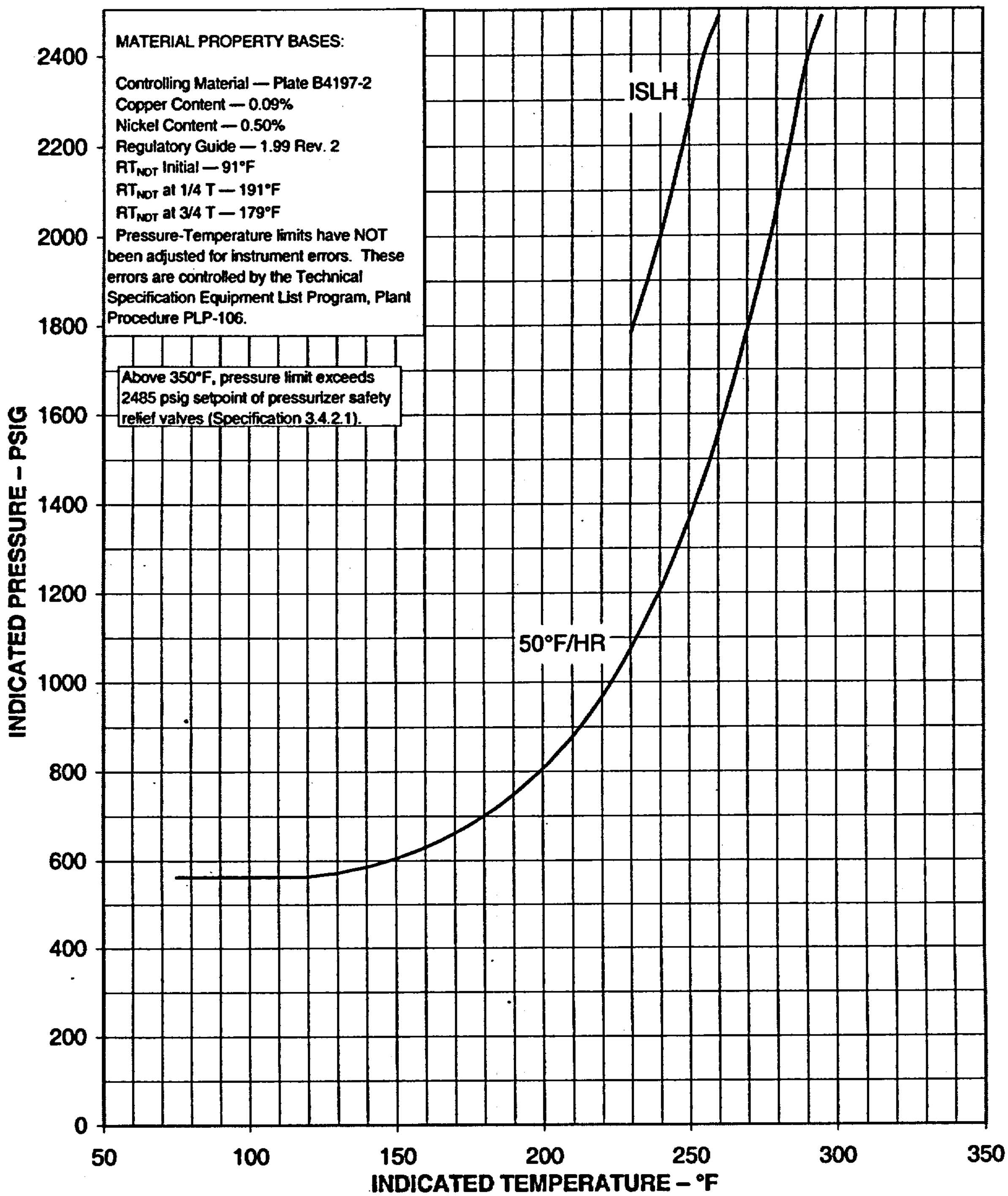


FIGURE 3.4–3
REACTOR COOLANT SYSTEM
HEATUP LIMITATIONS—APPLICABLE UP TO 36 EFPY

TABLE 4.4-6

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

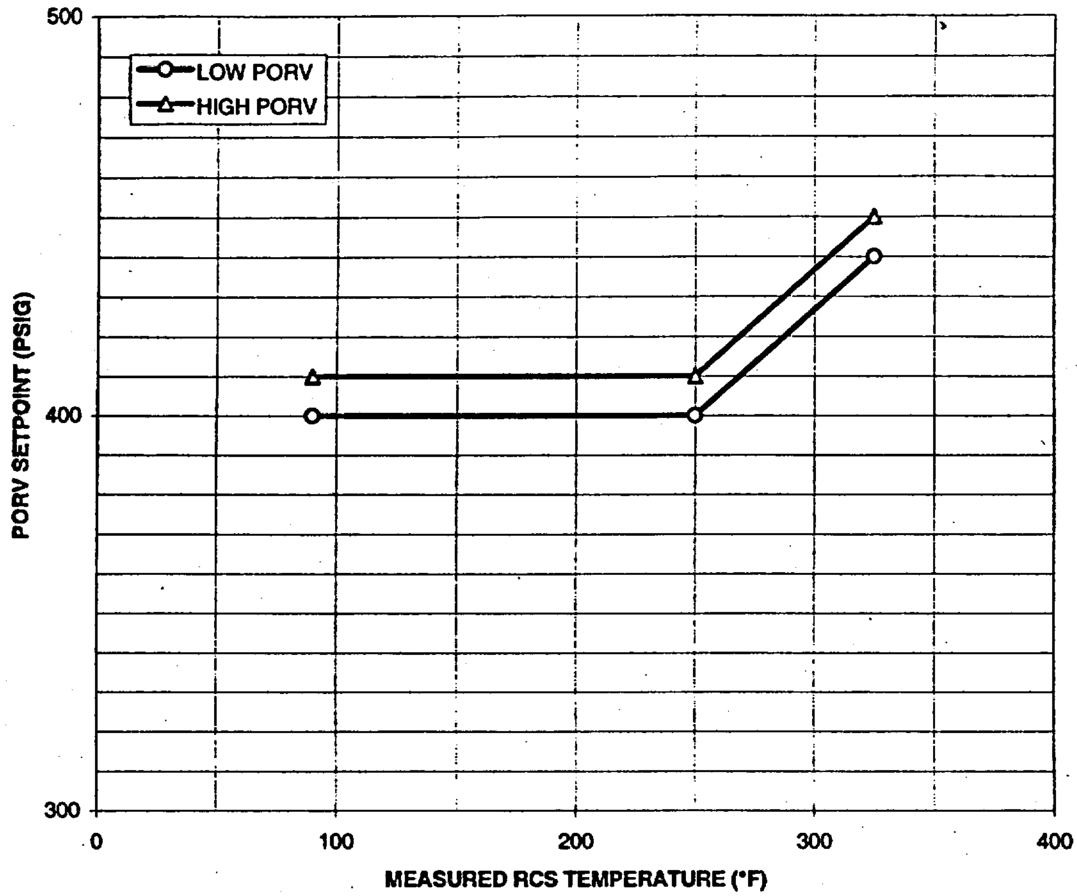
COOLDOWN RATES

TEMPERATURE*	COOLDOWN IN ANY 1 HOUR PERIOD*
350-120°F	50°F
< 120°F	30°F

HEATUP RATES

TEMPERATURE*	HEATUP IN ANY 1 HOUR PERIOD*
<350°F	50°F

*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.



RCS TEMP (°F)	LOW PORV* (psig)	HIGH PORV* (psig)
90	400	410
250	400	410
325	440	450

* VALUES BASED ON 36 EFPY REACTOR VESSEL DATA

INSTRUMENT ERRORS ARE CONTROLLED BY THE TECHNICAL SPECIFICATION EQUIPMENT LIST PROGRAM, PLANT PROCEDURE PLP-106.

FIGURE 3.4–4

MAXIMUM ALLOWED PORV SETPOINT FOR THE LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM

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, ASME Code Case N-640, 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).

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

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 36 effective full power years (EFPY) of service life.

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}, including margin, computed by Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."

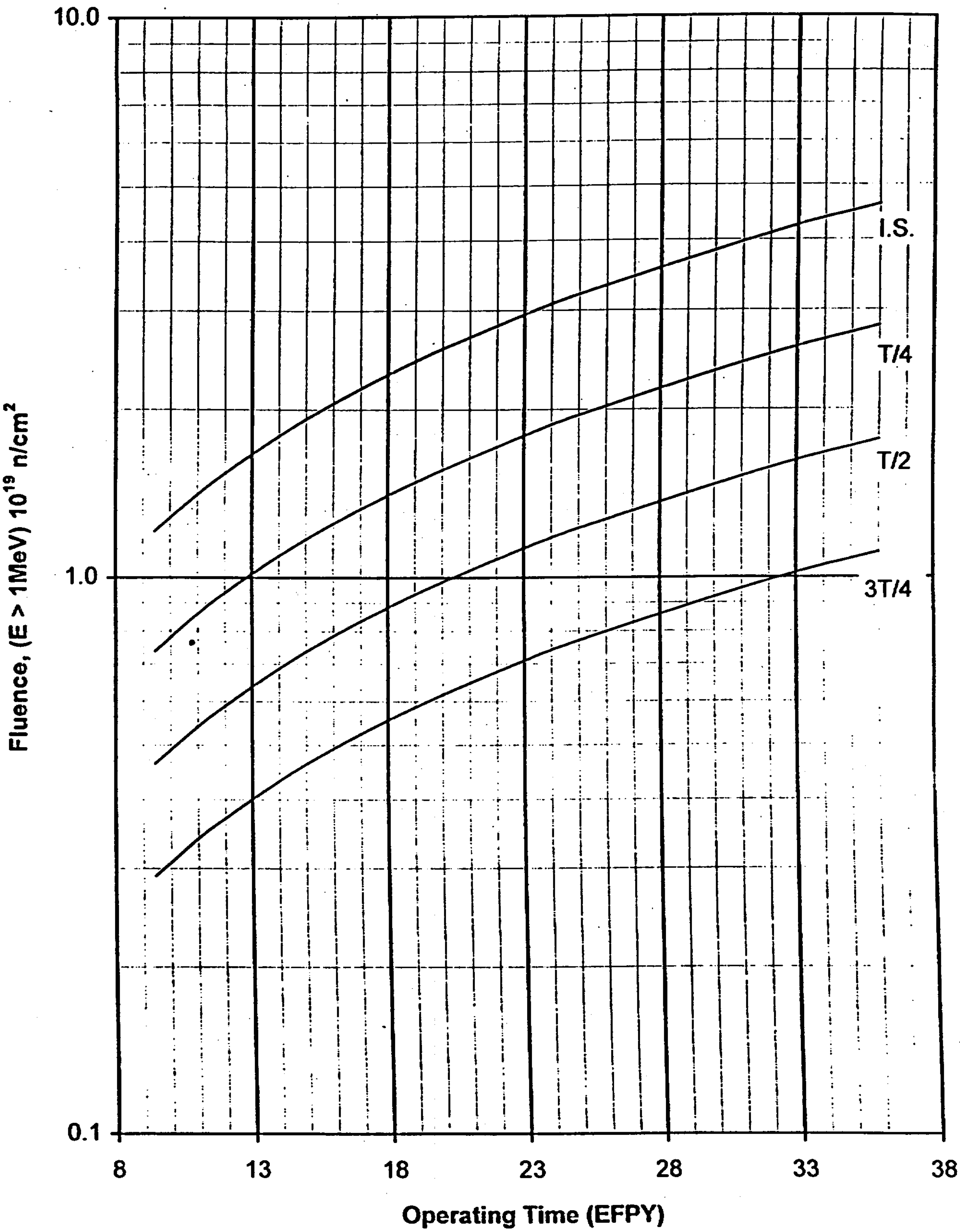


FIGURE B 3/4.4-1
FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE

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 and ASME Code Case N-640 for the reactor vessel controlling material.

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 for the beltline shell region 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. A semielliptical inside corner flaw is assumed for the nozzle regions with a depth of one-quarter of the nozzle belt wall thickness. The inlet nozzle is used in the calculation procdures since the inner radius of this tapered nozzle is larger at the corner than the inner radius of the more tapered outlet nozzle. The dimensions of these postulated cracks, referred to in Appendix G of ASME Section XI as reference flaws, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for 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 reference fracture toughness curves defined in the ASME Code. Pressure-temperature limits are developed for the vessel using the K_{IR} curve defined in Appendix A to the ASME Code, as permitted by ASME Code Case N-640. For the remaining components of the primary pressure boundary, pressure-temperature limits are based on the K_{IR} curve defined in Appendix G to the ASME Code. The K_{IR} curves are given by the equations:

Vessel regions:

$$K_{IR} = K_{Ic} = 33.2 + 2.806 \exp [0.02(T-RT_{NDT} + 100°F)] \quad (1a)$$

Remaining regions:

$$K_{IR} = K_{Ia} = 26.8 + 1.233 \exp [0.0145(T-RT_{NDT} + 160°F)] \quad (1b)$$

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:

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

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 wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. The pressure stress intensity factors are obtained and allowable pressures are calculated from equation 2.

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 and the inlet nozzle corner. During cooldown, the controlling location of the flaw is always at the inside surface 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. The composite limit curves are developed considering the controlling reactor vessel component, either the beltline shell or the inlet nozzle.

PRESSURE/TEMPERATURE LIMITS (Continued)

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T inside surface location is at a higher temperature than the fluid adjacent to the inside surface. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside surface. The thermal gradients during heatup produce compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of

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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 April 12, 2000, as supplemented June 2, 2000, Carolina Power & Light Company (CP&L, the licensee) requested a change to the Technical Specifications (TS) for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). The proposed amendment would revise TS 3/4.4.9.2, "Pressure/Temperature (P-T) Limits - Reactor Coolant System," and TS 3/4.4.9.4, "Overpressure Protection System," and the associated Bases. Specifically, the licensee proposed revising the P-T limits and the low temperature overpressure protection (LTOP) system setpoints to provide new limits that are valid to 36 effective full-power years (EFPY). The proposed changes incorporate the results of the Reactor Vessel Surveillance Program capsule analysis and an exemption from 10 CFR 50.60(a), based on American Society of Mechanical Engineers (ASME) Code Case N-640. The exemption to use Code Case N-640 for HNP was previously approved by separate correspondence.

The supplemental submittal dated June 2, 2000, provided clarifying information that did not change the scope of the April 12, 2000, application or the proposed no significant hazards consideration determination published in the *Federal Register* on May 3, 2000 (65 FR 25762).

2.0 BACKGROUND

The NRC has established requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The staff evaluates the P-T limit curves based on the following NRC regulations and guidance: 10 CFR 50, Appendix G; Generic Letter (GL) 88-11; GL 92-01, Revision 1; GL 92-01, Revision 1, Supplement 1; Regulatory Guide (RG) 1.99, Revision 2 (Rev. 2); and Standard Review Plan (SRP) Sections 5.2.2 and 5.3.2. GL 88-11 advised licensees that the staff would use RG 1.99, Rev. 2, to review P-T limit curves. RG 1.99, Rev. 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation. GL 92-01, Rev. 1, requested that licensees submit their reactor pressure vessel (RPV) data for their plants to the staff for review. GL 92-01, Rev. 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the staff as the basis for the staff's review of P-T limit curves and as the basis for the staff's review of pressurized thermal shock (PTS) assessments (10 CFR 50.61 assessments). Appendix G to 10 CFR Part 50 requires that P-T limit curves be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code. Appendix G to

10 CFR Part 50 also provides minimum temperature requirements that must be considered in the development of the P-T limit curves.

SRP Section 5.3.2 provides an acceptable method of determining the P-T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. Appendix G requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 for hydrostatic testing curves. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to 1/4 of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

The Appendix G ASME Code methodology requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}). The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term.

The ΔRT_{NDT} is a product of a chemistry factor (CF) and a fluence factor. The CF is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The M term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the CF was determined using the tables in RG 1.99, Rev. 2, or surveillance data. The M term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence and the calculational procedures. RG 1.99, Rev. 2, describes the methodology to be used in calculating the M term.

3.0 EVALUATION

The licensee submitted P-T limit curves and LTOP setpoints valid for operations up to 36 EFPY. As described in Framatome Technologies, Inc. report BAW-2355, "Analysis of Capsule X - CP&L Shearon Harris Nuclear Power Plant - Reactor Vessel Material Surveillance Program" (dated October 1999)(Ref. 1), the licensee determined that the limiting ART for the RPV is from the intermediate shell plate B4197-2, which is included in the HNP surveillance program required by Appendix H to 10 CFR Part 50. The results of the surveillance program, described in BAW-2355, indicate that the CF for that plate is reduced below that provided in the tables of RG 1.99, Revision 2. However, the licensee proposes use of a full M term to determine the ART of the plate because two of the data points exceed the σ_Δ value of RG 1.99, Rev. 2. For the limiting material, the licensee calculated an RT_{PTS} value of 196°F at 36 EFPY, based on an initial RT_{NDT} of 91°F, and a plant-specific M term of 34°F (σ_I = 0°F and σ_Δ = 17°F). For construction of the P-T limit curves, the licensee determined an ART of 191°F for the limiting material at the 1/4 T location at 36 EFPY, and 179°F at the 3/4 T location at 36 EFPY. The limiting reference temperature of the closure flange region is 0°F.

Based on these ART values for the limiting beltline material, the licensee used the methodology of Appendix G, as modified by Code Case N-640, to calculate the P-T limits and LTOP setpoints. The minimum temperature requirements of Table 1 of Appendix G to 10 CFR Part 50 have a minimal impact on the P-T curves for HNP. Specifically, operating procedures call for core criticality only when the unit is at normal operating temperature and pressure; thus, minimum temperature requirements 2.a and 2.b (for core not critical during normal operation) from Table 1 of Appendix G to 10 CFR Part 50 apply for the heatup and cooldown curves. These requirements only impact the cooldown curves from 118°F to 120°F, with the allowable pressure at 120°F reduced from 625 psi to 621 psi. This perturbation is barely discernable on the cooldown curve.

3.1 P-T Limits

The staff performed an independent calculation of the ART values for the limiting material using the methodology in RG 1.99, Revision 2. Based on these calculations, the staff verified that the licensee's limiting material for the HNP reactor vessel is the intermediate shell plate B4197-2 at 36 EFPY. The staff's calculated ART values for the limiting material agree with the licensee's calculated ART values. Further, except for the 36 EFPY fluence value assumed by the licensee, the data for the beltline materials reported in this submittal is consistent with that in the NRC's reactor vessel integrity database (RVID).

The staff performed check calculations to verify the P-T limit curves using the appropriate limiting ART values for the HNP RPV beltline materials. The staff found good agreement with the submitted P-T curves, as calculations confirmed various points on the submitted P-T limit curves within a few degrees of indicated temperature. The staff also found that the minimum temperature requirements of Table 1 of Appendix G to 10 CFR Part 50 were properly implemented in the P-T limit curves.

Thus, the staff determined that the P-T limit curves satisfy the requirements in Paragraph IV.A.2 of Appendix G of 10 CFR Part 50 as modified by Code Case N-640, and hence, the requirements of 10 CFR 50.60.

The staff concludes that the proposed P-T limit curves for the reactor coolant system (RCS) for heatup and cooldown satisfy the requirements in Appendix G to Section XI of the ASME Code, as modified by Code Case N-640, and Appendix G of 10 CFR 50, for 36 EFPY. The proposed P-T limit curves also satisfy GL 88-11 because the method in RG 1.99, Rev. 2, was used to calculate the ART. Hence, the proposed P-T limit curves are acceptable for incorporation in the HNP TS.

3.2 LTOP Setpoints / Fluence

In addition to changing the P-T limit curves as described above, the licensee proposed changes to the cooldown and heatup curves in Figures 3.4-2 and 3.4-3, respectively, and to the power-operated relief valve (PORV) setpoints.

3.2.1 Background

At HNP, the rate of embrittlement in the vessel base metal and the vessel welds is very small because the copper and nickel contents are very low. The low embrittlement rate in combination with increased acceptable stresses from the application of the ASME Code Case N-640 allowed the licensee to request an extension of the currently applicable limits of 11 EFPY to 36 EFPY. The 36 EFPY are beyond the 32 EFPY that are normally assumed for a 40-calendar year license based on an average load factor of 0.80. This review assumes that the 36 EFPY will be utilized within the current license with a load factor greater than 0.80. This assumption is consistent with HNP historical data.

The results of the testing and analysis include both the Charpy tests and dosimetry and are documented in BAW-2335 (Ref. 1). This review is limited to the vessel dosimetry and the associated extrapolation of the fluence value to the end of license. The methodology used for the fluence analysis has been approved by the NRC for Framatome Technologies Incorporated (FTI) as described in the topical report BAW-2241P-A (Ref. 2); therefore, the dosimetry is acceptable.

The combined effect of the capsule X dosimetry with the application of Code Case N-640 leads to limits which are less restrictive at 36 EFPY than the existing limits at 11 EFPY. This is due to the very low copper and nickel content of the base metal and the welds in the belt region of the pressure vessel, which define a very small chemistry factor and, thus, a very low embrittlement rate.

3.2.2 TS Changes

The cooldown and heatup curves in TS Figures 3.4-2 and 3.4-3 are changed to reflect the new limits. The heatup curve is now limited to a single heatup rate of 50°F/hr. The cooldown curve is reduced to two segments of 50°F/hr and 30°F/hr, respectively.

The PORV setpoints are credited for RCS temperatures at or above 90°F. The corresponding PORV settings are: for temperatures <250°F, the setpoints are 400/410 psig low/high, respectively; for temperatures between 250°F and 325°F, the setpoints change linearly from 400/410 psig low/high to 440/450 psig low/high, respectively; for temperatures above 325°F, the pressure can assume any value up to 2400 psig. For heatup between 90°F and 125°F, the new heatup rate is 50°F/hr and the corresponding pressure limit is 563 psig. For cooldown between 90°F and 125°F, the new cooldown rate is 30°F/hr and the corresponding pressure limit is 566 psig. The enable temperature did not change from 325°F.

At the above setpoints, the peak RCS pressure that occurs after the PORV has opened at the setpoint is less than the allowable pressure for the limiting heatup or cooldown transient. This accounts for instrument uncertainties, response times, and PORV flow discharge capacities. The proposed TS changes are, therefore, acceptable.

TS Figure B 3/4.4-1 has been replaced to reflect the extended period of operation to 36 EFPY and the surveillance capsule X dosimetry results. These results are also acceptable because they reflect an updated projection of the vessel fluence.

The TS bases have also been updated in line with the results of the surveillance capsule, the new P-T curves and LTOP limits.

4.0 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.

5.0 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 (65 FR 25762). 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.

6.0 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.

7.0 REFERENCES

1. BAW-2335, "Analysis of Capsule X Carolina Power & Light Company Shearon Harris Nuclear Power Plant" by M. J. Devan and S. Q. King, Framatome Technologies Incorporated, October 1999.

2. BAW-2241P-A, Revision 1, "Fluence and Uncertainty Methodologies" by J. R. Worsham III, Framatome Technologies Incorporated, Lynchburg VA, April 1999.

Principal Contributors: A. Hiser
L. Lois

Date: July 28, 2000

Mr. James Scarola
Carolina Power & Light Company

Shearon Harris Nuclear Power Plant
Unit 1

cc:

Mr. William D. Johnson
Vice President and Corporate Secretary
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Resident Inspector/Harris NPS
c/o U.S. Nuclear Regulatory Commission
5421 Shearon Harris Road
New Hill, North Carolina 27562-9998

Ms. Karen E. Long
Assistant Attorney General
State of North Carolina
Post Office Box 629
Raleigh, North Carolina 27602

Public Service Commission
State of South Carolina
Post Office Drawer
Columbia, South Carolina 29211

Mr. Mel Fry, Director
Division of Radiation Protection
N.C. Department of Environment
and Natural Resources
3825 Barrett Dr.
Raleigh, North Carolina 27609-7721

Mr. Terry C. Morton
Manager
Performance Evaluation and
Regulatory Affairs CPB 7
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602-1551

Mr. Robert J. Duncan II
Plant General Manager
Carolina Power & Light Company
Shearon Harris Nuclear Power Plant
P.O. Box 165, Mail Zone 3
New Hill, North Carolina 27562-0165

Mr. John H. O'Neill, Jr.
Shaw, Pittman, Potts & Trowbridge
2300 N Street, NW.
Washington, DC 20037-1128

Mr. Chris L. Burton
Director of Site Operations
Carolina Power & Light Company
Shearon Harris Nuclear Power Plant
Post Office Box 165, MC: Zone 1
New Hill, North Carolina 27562-0165

Mr. Robert P. Gruber
Executive Director
Public Staff NCUC
Post Office Box 29520
Raleigh, North Carolina 27626

Chairman of the North Carolina
Utilities Commission
Post Office Box 29510
Raleigh, North Carolina 27626-0510

Mr. Vernon Malone, Chairman
Board of County Commissioners
of Wake County
P. O. Box 550
Raleigh, North Carolina 27602

Mr. Richard H. Givens, Chairman
Board of County Commissioners
of Chatham County
P. O. Box 87
Pittsboro, North Carolina 27312

Ms. Donna B. Alexander, Manager
Regulatory Affairs
Carolina Power & Light Company
Shearon Harris Nuclear Power Plant
P.O. Box 165, Mail Zone 1
New Hill, NC 27562-0165

Mr. Eric A. McCartney, Supervisor
Licensing/Regulatory Programs
Carolina Power & Light Company
Shearon Harris Nuclear Power Plant
P. O. Box 165, Mail Zone 1
New Hill, NC 27562-0165