August 20, 1993

Docket No. 50-400

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Mr. W. S. Orser
Executive Vice President
Nuclear Generation
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Orser:

SUBJECT: ISSUANCE OF AMENDMENT NO. 38 TO FACILITY OPERATING LICENSE NO. NPF-63 REGARDING TECHNICAL SPECIFICATION CHANGE TO PRESSURE-TEMPERATURE LIMITS - SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 (TAC NO. M85876)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 38 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 February 26, 1993, as supplemented July 27, 1993.

The amendment revises Technical Specifications 3/4.4.9, Pressure/Temperature Limits, by replacing the current 5 year heatup and cooldown limitations with revised limitations based on predicted reactor vessel neutron exposure at 11 effective full-power years of operation.

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

Enclosures:
1. Amendment No. 38 to NPF-63
2. Safety Evaluation

cc w/enclosures:
See next page

*See previous concurrence

OFC	PM:PD21:DRPE	AD:PD21:DRPE	*OGC	
NAME	NBLe:tmw	SSBajwa	EHoller	
DATE	08/20/93	08/20/93	08/12/93	

Document Name: HAR85876.AMD

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Carolina Power & Light Company

Shearon Harris Nuclear Power Plant, Unit 1

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AMENDMENT NO. 38 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. 38
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 February 26, 1993, as supplemented July 27, 1993, 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:

9308250431 930820
PDR ADOCK 05000400
P PDR

(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. 38, 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 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Bajwa

S. Singh Bajwa, 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: August 20, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 38

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 4-41	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 4-8	B 3/4 4-8
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 4-14	B 3/4 4-14
B 3/4 4-15	B 3/4 4-15

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

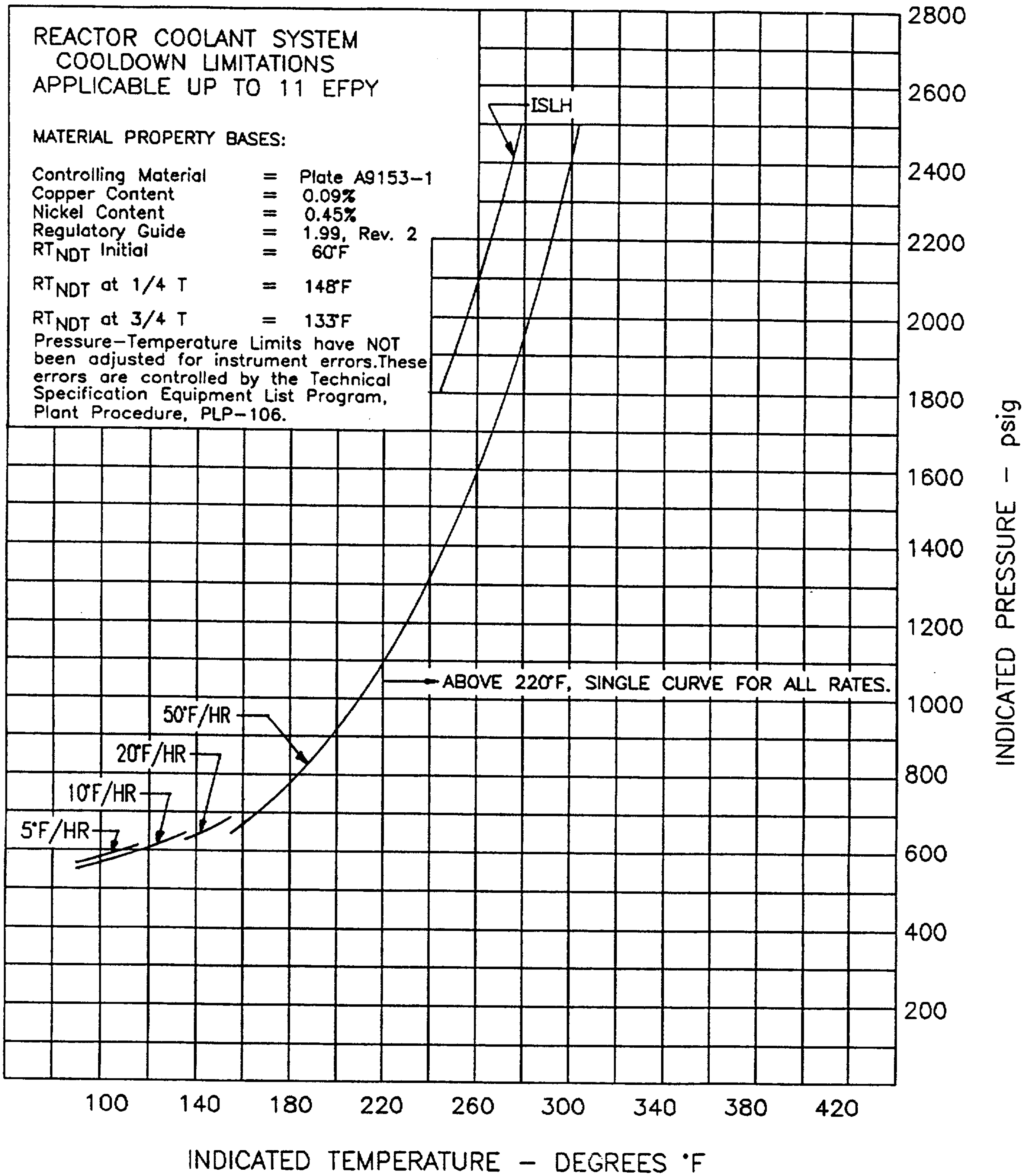


FIGURE 3.4-2
REACTOR COOLANT SYSTEM
COOLDOWN LIMITATIONS - APPLICABLE UP TO 11 EFPY

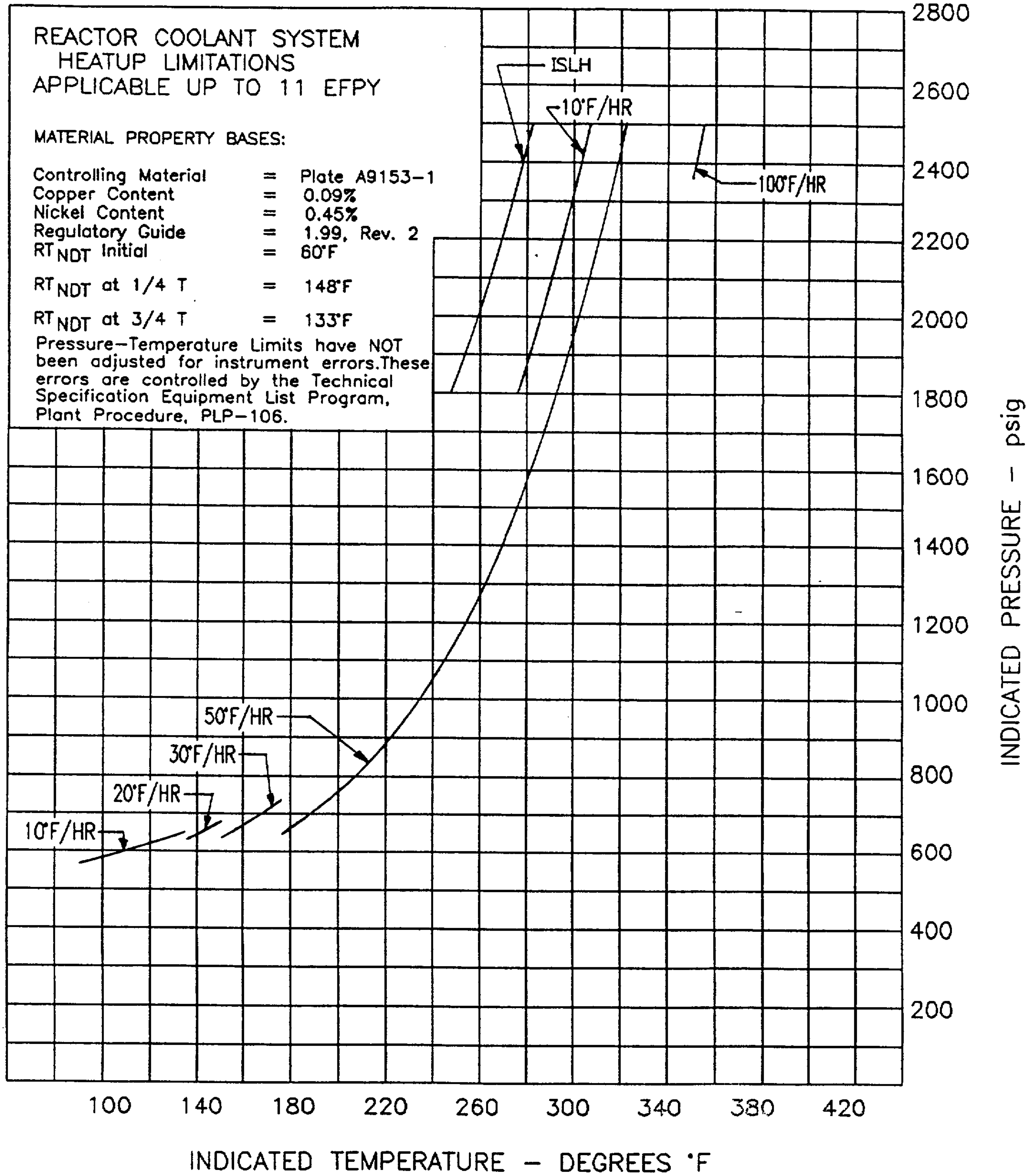


FIGURE 3.4–3

REACTOR COOLANT SYSTEM

HEATUP LIMITATIONS – APPLICABLE UP TO 11 EFPY

TABLE 4.4-6

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

COOLDOWN RATES

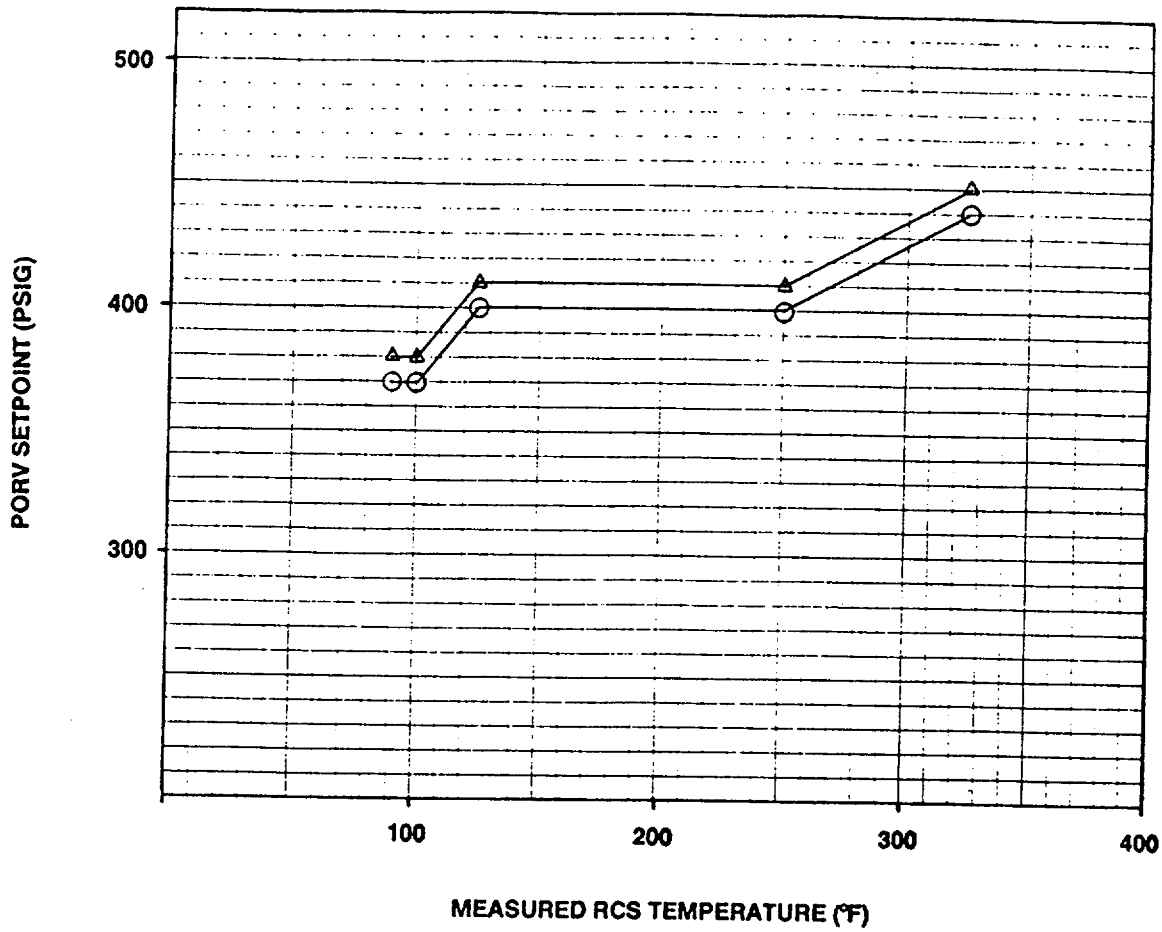
TEMPERATURE*	COOLDOWN IN ANY 1 HOUR PERIOD*
350-155°F	50°F
155-135°F	20°F
135-115°F	10°F
< 115°F	5°F/10°F**

HEATUP RATES

TEMPERATURE*	HEATUP IN ANY 1 HOUR PERIOD*
<135°F	10°F
135-150°F	20°F
150-175°F	30°F
175-350°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.

**10°F/HR cooldown rate may be used if less than three RCPs are operating.



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

* VALUES BASED ON 11 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 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 III, 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

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

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	.45	-10	60	83
" "	"	B4197-2	.10	.50	-10	91	71
Lower Shell	"	C9924-1	.08	.45	-10	54	98
" "	"	C9924-2	.08	.44	-20	57	88
Bottom Hd. Torus	"	A9249-2	-	-	-40	14	94
" " Dome		A9213-2	-	-	-40	-8	125
Weld (Inter & Lower Shell Vertical Weld Seams)			.06	.91	-20	-20	>94
Weld (Inter. to Lower Shell Girth Seam)			.04	.95	-20	-20	88

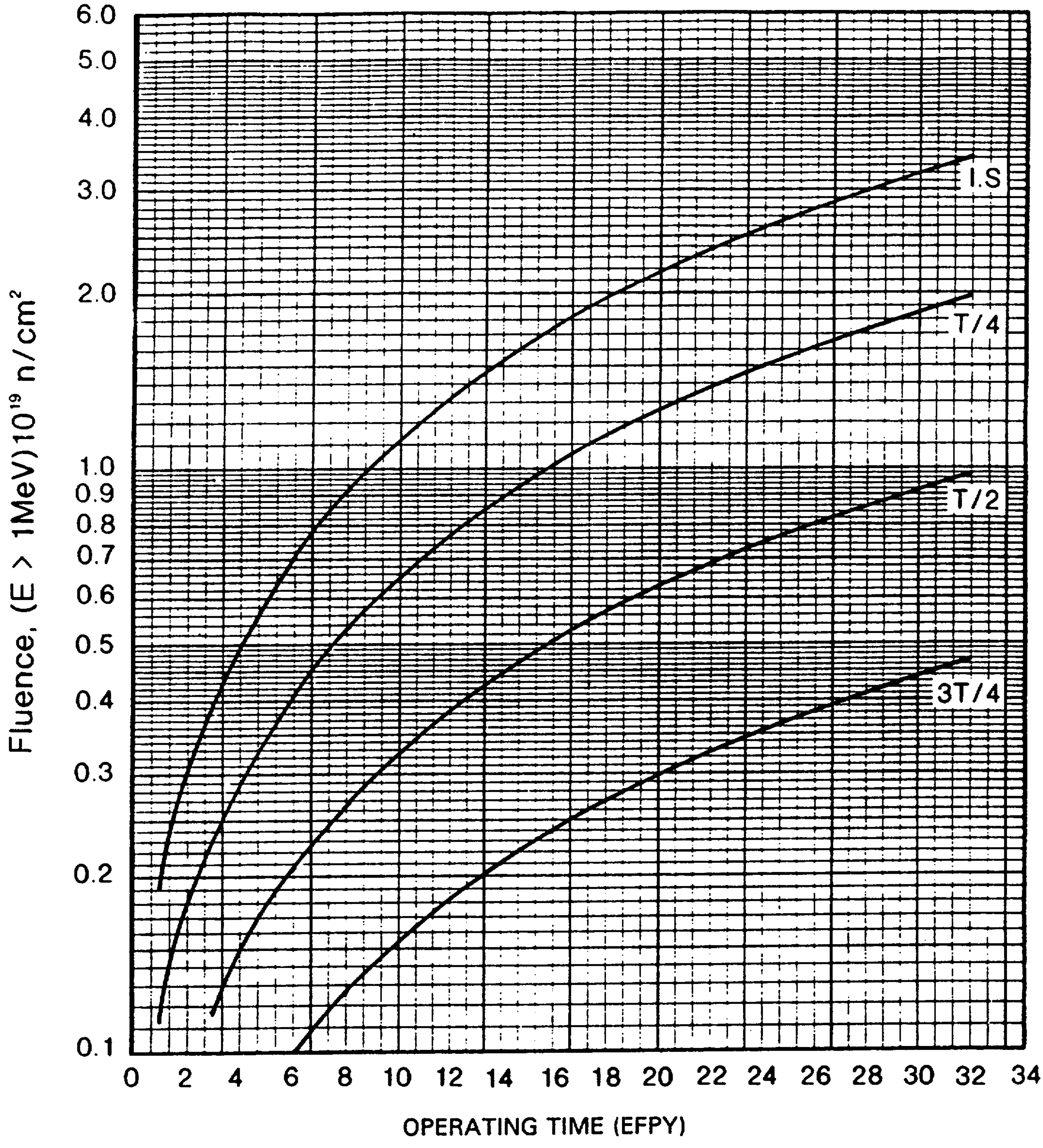


FIGURE B 3/4.4-1

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 III 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 III 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 with no fuel in the reactor vessel.

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.

LOW TEMPERATURE OVERPRESSURE PROTECTION (Continued)

of the LTOPS assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that Appendix G criteria will not be violated with consideration for a maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and single failure. LTOP instrument uncertainties are controlled by the Technical Specification Equipment List Program, Plant Procedure PLP-106. To ensure that mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout of all but one charging/safety injection pump while in MODES 4 (below 325°F), 5 and 6 with the reactor vessel head installed and disallow start of an RCP if secondary temperature is more than 50°F above primary temperature.

The maximum allowed PORV setpoint for the LTOPS will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H, and the reactor vessel service life.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1977 Edition and Addenda through Summer 1978.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer steam space ensures that the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737,"Clarification of TMI Action Plant Requirements," November 1980.

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 when no fuel is in the reactor vessel. Otherwise, normal heatup and cooldown P-T curves apply.

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 38 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 February 26, 1993, as supplemented July 27, 1993, Carolina Power & Light Company (the licensee) requested a revision to Technical Specification (TS) 3/4.4.9 for the Shearon Harris Nuclear Power Plant (SHNPP). The revision to TS 3/4.4.9 would replace the current 5 year pressure-temperature limits with limitations through 11 effective full power years (EFPY) of operation.

To determine the acceptability of the licensee's request, the staff reviewed (1) the licensee's conformance to the basis for revising the effective date of the limits with respect to the requirements of Appendix G of 10 CFR Part 50 and Regulatory Guide 1.99, Revision 2, and (2) the adequacy of the licensee's current low temperature overpressure protection system (LTOP) setpoints, to ensure that the peak pressure developed during a design basis mass input event will not exceed the revised Appendix G P/T limits for the current heatup/cooldown rates and the extended temperature ranges.

Pressure-temperature limits are required to meet the fracture toughness requirements of Appendix G to 10 CFR Part 50. Appendix G also requires the licensees to predict the effect of neutron irradiation on the reactor vessel beltline materials. In aiding the licensees to meet the Appendix G requirements, the staff issued Generic Letter 88-11 and requested that licensees use the methods in Regulatory Guide (RG) 1.99, Revision 2, to predict the effect of neutron irradiation on the reactor vessel beltline materials. The effective period of time for pressure-temperature limits is dependent upon the adjusted reference temperatures (ART) for the limiting beltline material. Revision 2 to RG 1.99 recommends two methods of determining the ART. In one method (Position 1.1. of RG 1.99, Revision 2), the ART are determined from the amount of copper, nickel and neutron fluency of the limiting reactor vessel beltline material. The other method (Position 2.1 of RG 1.99, Revision 2), is recommended when the licensee has credible surveillance data for the limiting reactor vessel beltline material.

In addition, Section IV.A.2 in Appendix G of 10 CFR Part 50 requires the pressure-temperature limits to meet the requirements of Appendix G of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) and establishes additional closure flange limitations. The fracture mechanics criteria in Appendix G of the ASME Code requires that the pressure-temperature limits be determined from the ART at the 1/4 wall thickness (1/4t) and the 3/4 wall thickness (3/4t) locations. Section IV.A.2 in Appendix G of 10 CFR Part 50 specifies limits when the core is critical.

Based on the results of the Reactor Vessel Capsule Surveillance Program and the requirements of Appendix G of 10 CFR Part 50, the licensee revised the pressure-temperature limits. These revised limits will replace the current five EFPY heatup and cooldown limits of TS 3.4.4.9 which expire at the end of August 1993 time frame. The new limits are based on the predicted reactor vessel neutron exposure at 11 EFPY and are less restrictive than those they replace, i.e., for the existing heatup and cooldown rates, the allowable maximum RCS pressure is higher at each temperature and the effective temperature ranges for the existing heatup. Additionally, the cooldown rates have been adjusted downwards to lower temperatures, based on the increased pressure margin. However, the low-temperature overpressure protection (LTOP) setpoints and enable temperature have not been revised and will remain at their current values, as previously discussed.

2.0 EVALUATION

2.1 Licensee conformance to Appendix G requirements:

The SHNPP reactor vessel beltline consists of four plates, four vertical weld seams and a circular girth weld seam. The limiting materials in the beltline are the plates A9153-1 and B4197-2 that are in the intermediate shell. Materials from plate B4197-2 are being irradiated in surveillance capsules in the SHNPP reactor vessel. Two capsules have been withdrawn and tested. The test results satisfy the data scatter criteria in RG 1.99, Revision 2. Hence, the ART for plate B4197-2 may be calculated using Position 2.1 of RG 1.99, Revision 2. Since plate A9153-1 is not contained in the SHNPP surveillance capsules, its ART must be calculated using Position 1.1. of RG 1.99, Revision 2. Using the methodologies recommended in RG 1.99, Revision 2, the test results from the SHNPP surveillance capsules, and the licensee's projected neutron fluences at 11 EFPY, the staff has calculated that plate A9153-1 has the highest ART at the 1/4t location and plate B4197-2 has the highest ART at the 3/4t location. The ART at 11 EFPY calculated by the staff were 146°F for the 3/4t location. The ART calculated by the licensee were 148°F and 133°F at the 1/4t and 3/4t locations, respectively.

The staff has performed calculations using the methodology in Appendix G of the ASME Code. These calculations confirm that the proposed pressure-temperature limits meet the requirements of Appendix G of 10 CFR Part 50 through 11 EFPY.

Based on the reference temperatures reported for the materials in the closure flange region of the reactor vessel, the limits according to Appendix G of the ASME Code are more restrictive than the closure flange limits in Section IV.A.2 in Appendix G of 10 CFR Part 50.

The proposed pressure-temperature limits do not contain limits for core criticality because Technical Specification 3.1.1.4 contains core criticality limits that are more restrictive than the limits required by Section IV.A.3 in Appendix G of 10 CFR Part 50.

2.2 Adequacy of the licensee revised analysis:

The staff reviewed the licensee's evaluation of the available pressure margin for the existing heatup/cooldown rates. The pressure margin is defined as the difference between the allowable Appendix G limit and the transient peak pressure reached in the mass input event. The mass input event is the design basis transient for the LTOPs setpoint determination and entails a scenario such as the inadvertent startup of one charging/safety injection (SI) pump and resulting injection into a water-solid RCS. No credit is taken for letdown flow and relief through the RHR relief valve. In the analysis of this event, the initial pressure is taken as the LTOP setpoint value for a given temperature.

Because the revised Appendix G pressure-temperature limits are less restrictive than the current limits, an increase in pressure margin is obtained as a result. However, certain revisions have also been made in the thermal-hydraulic analysis of the mass input event which are used to support the proposed TS changes. Since these revisions affect pressure margin, they are discussed below.

In the existing analysis of the mass input event, the injection flow rate is taken as the runout value for an SI pump. This assumption is overly conservative because pump runout would correspond to a depressurized RCS. During startup operations, the RCS is normally pressurized to allow for reactor coolant pump operation. In the revised analysis of the mass input event, the licensee has relaxed this assumption and used a more realistic, reduced flow rate consistent with a water-solid RCS, pressurized at the LTOP setpoint. The result is a lower peak pressure for the mass input event and a corresponding increase in pressure margin.

The existing analysis of the mass input event accounts for the pressure differential due to static head between the LTOP's pressure sensor location and the reactor vessel beltline mid-plane (the principal region of concern for LTOP protection). However, the dynamic pressure drop that occurs between these points is not considered in the existing analysis and neglecting this factor will result in an underestimation of the transient peak pressure. In the revised analysis which is being used to support the proposed TS change, the licensee has accounted for dynamic pressure drop by assuming operation of all three (3) RCPs and the two (2) RHR pumps at full flow (except for temperatures below 115°F where 2 RCPs are assumed to be operating). The result of this revised analysis is an increase in transient peak pressure and thus a reduction in margin on the order of 50 psi for all three RCPs in operation (and 26 psi for two RCPs in operation).

Instrumentation uncertainties for pressure and temperature sensors have been accounted for in the current analysis of the mass input event. However, based on new data obtained from vendor, the uncertainty in the pressure sensing instrumentation has been increased. This increase has been considered in the revised licensee analysis and resulted in a 5 psi decrease in pressure margin.

The combined effects of the less restrictive Appendix G limits, the decreased injection flow for the mass input event, the consideration of the dynamic pressure drop between the vessel beltline and pressure sensor location, and the increase in uncertainty in the pressure sensoring instrumentation, have resulted in a net increase in pressure margin. This margin is variable over the temperature range of interest. To take advantage of this gain in pressure margin, the revised 10 CFR Part 50, Appendix G, pressure-temperature curves for the current heatup/cooldown rates have been shifted so each curve becomes effective at lower temperatures, and yet still can provides adequate margin.

The staff concludes that the proposed pressure-temperature limits are acceptable through 11 EFPY because (1) the licensee meets the requirements of Appendix G of 10 CFR Part 50 and the ART were as calculated in accordance with RG 1.99, Revision 2; (2) the licensee has adequately demonstrated the basis for the usage of the existing LTOPs setpoints, and that the peak pressure developed during the design basis mass input event will not exceed the revised 10 CFR Part 50, Appendix G, pressure-temperature limits for the current heatup/cooldown rates and extended temperature ranges. Hence, the staff concurs with the licensee's analyses for the above-proposed pressure-temperature limit change, and finds the proposed change to be acceptable for incorporation into the SHNPP Technical Specifications.

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 effluent 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 (58 FR 16854). 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.

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