July 29, 1994

Docket No. 50-261

Mr. C. S. Hinnant, Vice President
Carolina Power & Light Company
H. B. Robinson Steam Electric Plant,
Unit No. 2
Post Office Box 790
Hartsville, South Carolina 29551-0790

Dear Mr. Dietz:

SUBJECT: ISSUANCE OF AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NO. DPR-23 REGARDING PRESSURE TEMPERATURE LIMITS - H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 (TAC NO. M87780)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 149 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBR). The amendment changes the pressure-temperature (P-T) limits from 15 to 24 effective full power years in the HBR Technical Specifications in response to your request dated September 15, 1993.

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

Sincerely,

Original Signed by:

Brenda L. Mozafari, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 149 to DPR-23
2. Safety Evaluation

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DATE	7/6/94	7/6/94	7/6/94	7/15/94	

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AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NO. DPR-23 - H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

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

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Sincerely,

Brenda Mozafari

Brenda L. Mozafari, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 149 to DPR-23
2. Safety Evaluation

cc w/enclosures:
See next page

cc:

Mr. H. Ray Starling
Manager - Legal Department
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

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

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
H. B. Robinson Steam Electric Plant
Route 5, Box 413
Hartsville, South Carolina 29551

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta St., N.W., Ste. 2900
Atlanta, Georgia 30323

Mr. Marc P. Pearson
Plant Manager
Carolina Power & Light Company
H. B. Robinson Steam Electric Plant
Post Office Box 790
Hartsville, South Carolina 29551

Public Service Commission
State of South Carolina
Post Office Drawer 11649
Columbia, South Carolina 29211

Mr. Dayne H. Brown, Director
Department of Environmental,
Health and Natural Resources
Division of Radiation Protection
Post Office Box 27687
Raleigh, North Carolina 27611-7687

Mr. Robert P. Gruber
Executive Director
Public Staff - NCUC
Post Office Box 29520
Raleigh, North Carolina 27626-0520

Mr. Max Batavia, Chief
Bureau of Radiological Health
South Carolina Department of Health
and Environmental Control
2600 Bull Street
Columbia, South Carolina 29201

Mr. H. W. Habermeyer, Jr.
Vice President
Nuclear Services Department
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Hartsville Memorial Library
147 West College Avenue
Hartsville, South Carolina 29550

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-261

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 149
License No. DPR-23

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

   A. The application for amendment by Carolina Power & Light Company (the licensee), dated September 15, 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 3.B. of Facility Operating License No. DPR-23 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 149 , are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Pat D[illegible] for

David B. Matthews, 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: July 29, 1994

ATTACHMENT TO LICENSE AMENDMENT NO.149

FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

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
v	v
3.1-21	3.1-21
3.1-21a	--
3.1-22	3.1-22
3.1-22a	--
3.1-4	3.1-4
3.1-5	3.1-5
3.1-6	3.1-6
3.1-7	3.1-7

LIST OF FIGURES

3.1.2 Heatup and Cooldown

3.1.2.1 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2 (for vessel exposure up to 24 EFPY). These limitations are as follows:

a. Over the temperature range from cold shutdown to hot operating conditions, the heatup rate shall not exceed 60°F/hr. in any one hour.

b. Allowable combinations of pressure and temperature for a specific cooldown rate are below and to the right of the limit lines for that rate as shown on Figure 3.1-2. This rate shall not exceed 100°F/hr. in any one hour. The limit lines for cooling rates between those shown in Figure 3.1-2 may be obtained by interpolation.

c. Primary system hydrostatic leak tests may be performed as necessary, provided the temperature limitation as noted on Figure 3.1-1 is not violated. Maximum hydrostatic test pressure should remain below 2350 psia.

d. The overpressure protection system shall be operable whenever the RCS temperature is below 350°F and not vented to the containment. One PORV may be inoperable for seven days. If the inoperable PORV has not been returned to service within 7 days, or if at any time both PORVs become inoperable, then one of the following actions should be completed within 12 hours:

   1. Cooldown and depressurize the RCS or

2. Heatup the RCS to above 350°F.

e. Operation of the overpressure protection system to relieve a pressure transient must be reported as a Special Report to the NRC within 30 days of operation.

3.1.2.2 The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the vessel is below 120°F.

3.1.2.3 The pressurizer shall neither exceed a maximum heatup rate of 100°F/hr nor a cooldown rate of 200°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

3.1.2.4 Figures 3.1-1 and 3.1-2 shall be updated periodically in accordance with the following criteria and procedures before the calculated exposure of the vessel exceeds the exposures for which the figures apply.

a. At least 60 days before the end of the integrated power period for which Figures 3.1-1 and 3.1-2 apply, the limit lines on the figures shall be updated for a new integrated power period utilizing methods derived from the ASME Boiler and Pressure Vessel Code, Section III, Appendix G and in accordance with applicable appendices of 10CFR50. These limit lines shall reflect any changes in predicted vessel neutron fluence over the integrated power period or changes resulting from the irradiation specimen measurement program.

b. The results of the examinations of the irradiation specimens and the updated heatup and cooldown curves shall be reported to the Commission within 90 days of completion of the examinations.

steels such as ASTM A302 Grade B parent material of the H. B. Robinson Unit No. 2 reactor pressure vessel are well documented in the literature. Generally, low alloy ferritic materials show an increase in hardness and other strength properties and a decrease in ductility and impact toughness under certain conditions of irradiation. Accompanying a decrease in impact strength is an increase in the temperature for the transition from brittle to ductile fracture.

A method for guarding against fast fracture in reactor pressure vessels has been presented in Appendix G, "Protection Against Non-Ductile Failure," to Section III of the ASME Boiler and Pressure Vessel Code. The method utilizes fracture mechanics concepts and is based on the reference nil-ductility temperature, RT_{NDT}.

RT_{NDT} is defined as the greater of: 1) the drop weight nil-ductility transition temperature (NDTT per ASTM E-208) or 2) the temperature 60°F less than the 50 ft-lb (and 35 mils lateral expansion) temperature as determined from Charpy specimens oriented in a direction normal to the major working direction of the material. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{IR} curve) which appears in Appendix G of Section III of the ASME Boiler and Pressure Vessel Code. The K_{IR} curve is a lower bound of dynamic, crack arrest, and static fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IR} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined utilizing these allowable stress intensity factors.

The Certified Material Test Reports (CMTRs) for the original steam generators provided records of Charpy V-notch tests performed at +10°F. Acceptable Charpy V-notch tests of +10°F indicate RT_{NDT} is at or below this temperature. The steam generator lower assemblies were replaced in 1984 and the material tests results indicate the highest RT_{NDT} is 60°F or below. The ASME code recommends that hydrostatic tests be performed at a temperature not lower than RT_{NDT} plus 60F°, thus the pressurizing temperature for the steam generator shell is established at 120°F to provide protection against nonductile failure at the test pressure.

V-notch 30 ft-lb temperature (ΔRT_{NDT}) due to irradiation is added to the original ΔRT_{NDT} to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} (RT_{NDT} initial + ΔRT_{NDT}) is utilized to index the material to the K_{IR} curve and in turn to set operating limits for the nuclear power plant which take into account the effects of irradiation on the reactor vessel materials. Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods (2) derived from Appendix G to Section III of the ASME Boiler and Pressure Vessel Code. The approach specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown on the K_{IR} curve in Appendix G for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by pressure gradients.

Following the generation of pressure-temperature curves for both the steady state and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point-by-point comparison of the steady state and finite heatup rate data. At any given temperture, the allowable pressure is taken to be the lesser of the two values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments.

The use of the composite curve is mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the O.D. to the I.D. location; and the pressure limit must, at all times, be based on the most conservative case. The cooldown analysis proceeds in the same fashion as that for heatup, with the exception that the controlling location is always at the I.D. position. The thermal gradients induced during cooldown tend to produce tensile stresses at the I.D. location and compressive stresses at the O.D. position. Thus, the I.D. flaw is clearly the worst case.

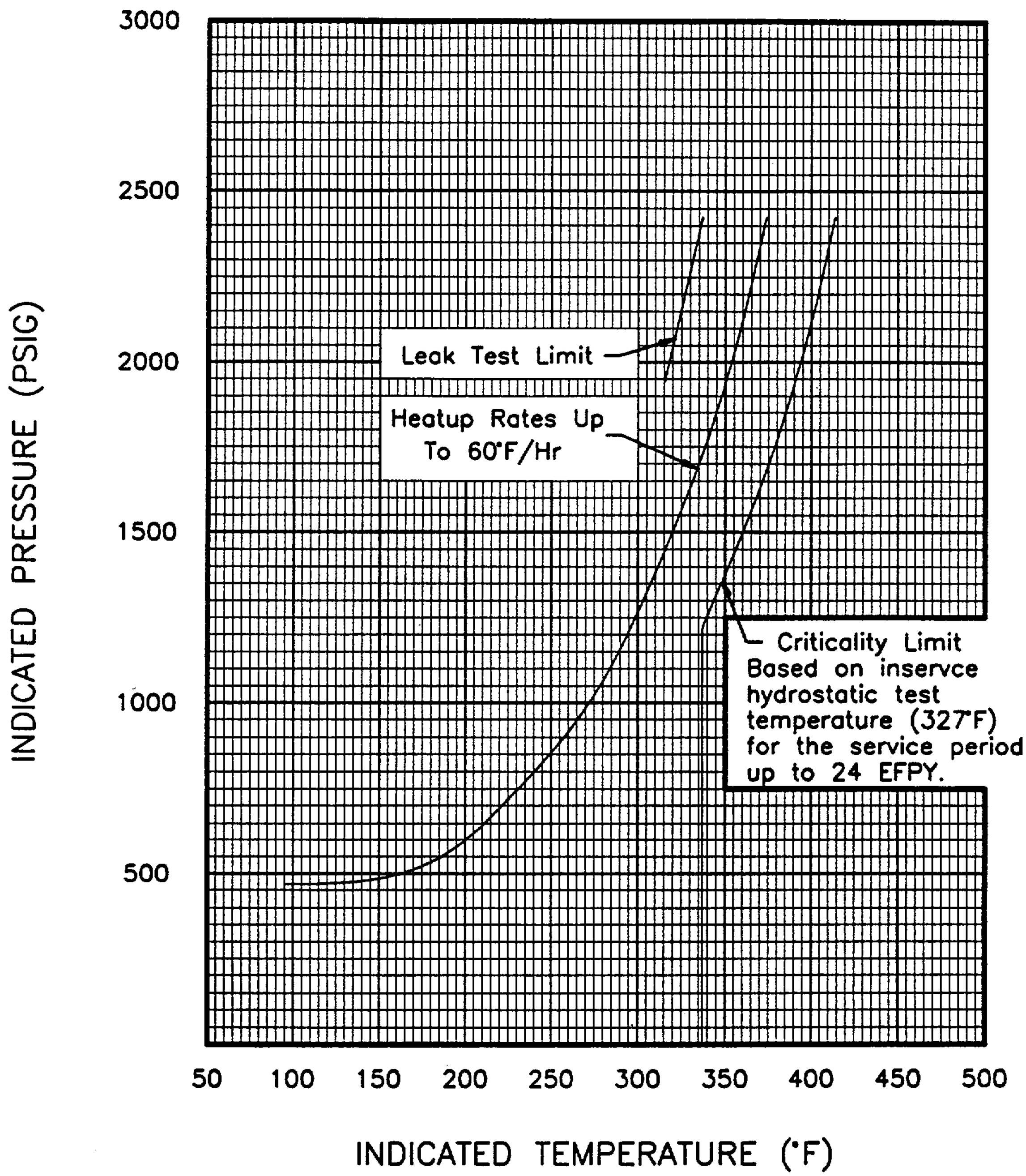
As in the case of heatup, allowable pressure temperature relations are generated for both steady state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of

MATERIALS PROPERTIES BASE

Controlling Material	:	Lower Circumferential Weld
Copper Content	:	0.20 wt.%
Nickel Content	:	1.06 wt.%
RT_{NDT} Initial	:	−80°F
RT_{NDT} After 24 EFPY	:	1/4 T. 207.83°F 3/4 T. 137.15°F

Curves applicable for heatup rates up to 60°F/Hr for the service period up to 24 EFPY.

Includes +10°F and −60 PSIG allowance for instrumentation error.



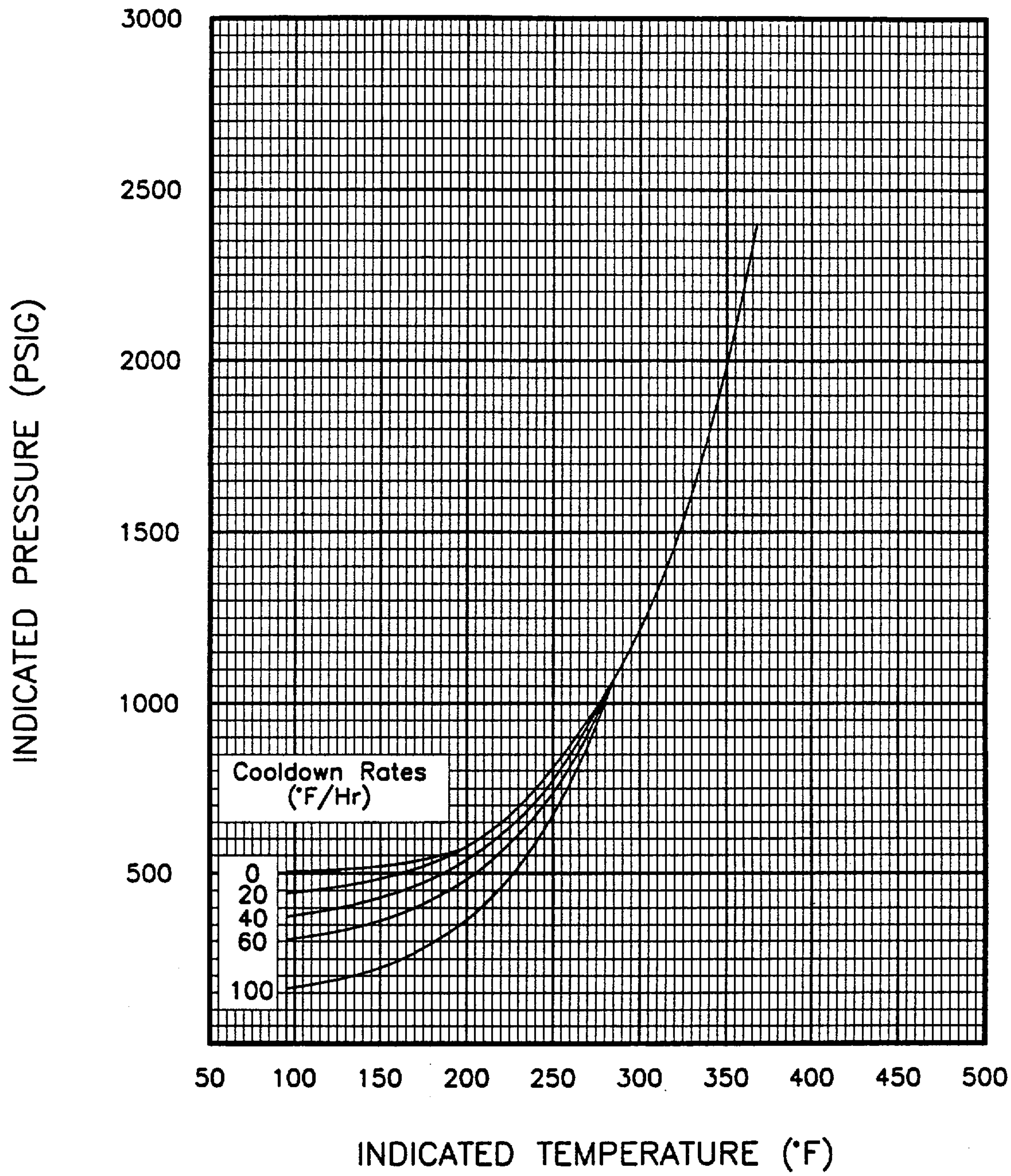
H.B. Robinson Unit #2 CP&L CAROLINA POWER & LIGHT COMPANY Technical Specifications	Reactor Coolant System Heatup Limitations Applicable Up To 24 EFPY	FIGURE 3.1−1

MATERIALS PROPERTIES BASE

Controlling Material	:	Lower Circumferential Weld
Copper Content	:	0.20 wt.%
Nickel Content	:	1.06 wt.%
RT_{NDT} Initial	:	−80°F
RT_{NDT} After 24 EFPY	:	1/4 T. 207.83°F 3/4 T. 137.15°F

Curves applicable for cooldown rates up to 100°F/Hr for the service period up to 24 EFPY.

Includes +10°F and −60 PSIG allowance for instrumentation error.



H.B. Robinson Unit #2 CP&L CAROLINA POWER & LIGHT COMPANY Technical Specifications	Reactor Coolant System Cooldown Limitations Applicable Up To 24 EFPY	FIGURE 3.1−2

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NO. DPR-23

CAROLINA POWER & LIGHT COMPANY

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-261

1.0 INTRODUCTION

By letter dated September 15, 1993, Carolina Power & Light Company (licensee) requested a change to the pressure-temperature (P-T) limits from 15 to 24 effective full power years in the H. B. Robinson Steam Electric Plant, Unit No. 2, (HBR) Technical Specifications (TS). The staff evaluated the P-T limits based on the following NRC regulations and guidance:

Regulatory Guide 1.99, Revision 2

Regulatory Guide (RG) 1.99, Revision 2, describes the methodology used by the NRC staff for evaluating all submittals regarding P-T limits and for all analyses that require an estimate of vessel beltline embrittlement (except those for pressurized thermal shock).

Appendix G to 10 CFR Part 50

Appendix G to 10 CFR Part 50 requires that the P-T limits for the reactor vessel be at least as conservative as those obtained in Appendix G of the American Society of Mechanical Engineers Code (ASME Code), Section III.

Generic Letter 88-11

Generic Letter (GL) 88-11 requests that, unless they can justify the use of different methods, the licensees use the methods in RG 1.99, Revision 2, to predict the effect of neutron irradiation by calculating adjusted reference temperature (ART) of the vessel materials. The ART is defined as the sum of unirradiated 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 and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the vessel material.

Generic Letter 92-01

Generic Letter 92-01 requests that the licensees submit the reactor vessel materials data which should be the basis for P-T limits.

Standard Review Plan (SRP) Section 5.3.2.

Section 5.3.2 of the SRP provides guidance on calculation of the P-T limits using linear elastic fracture mechanics methodology specified in Appendix G to the ASME Code, Section III.

2.0 EVALUATION

The HBR reactor vessel was fabricated by Combustion Engineering. The licensee reported that the limiting material with the highest ART was the lower circumferential weld. This beltline weld was fabricated using a submerged arc process with Linde 1092 flux and a weld wire heat number of 34B009. In the licensee's P-T limits submittal, the lower circumferential weld was reported to contain 0.20 percent of copper and 1.06 percent of nickel and to have an unirradiated RT_{ndt} of -80°F. The unirradiated RT_{ndt} was determined from tests of surveillance weld material in the Millstone Unit 1 reactor vessel which the licensee believes was fabricated from the same heat of weld wire as used in the lower circumferential weld in the HBR reactor vessel. The test results were published in a General Electric Company report, NEDC-30299. However, since the licensee has not provided traceability of the Millstone Unit 1 weld wire heat number, the unirradiated RT_{ndt} from the Millstone Unit 1 surveillance material cannot be utilized by HBR at this time.

The NRC staff performed an independent calculation and identified the same lower circumferential weld as the limiting material. For this weld, the NRC used a chemistry of 0.17 percent copper and 0.92 percent nickel that was based on the licensee's responses to GL 92-01, dated November 29, 1993. In its response to GL 92-01, the licensee determined the amount of copper as the mean value from four welds fabricated using the same heat number of weld wire as the lower circumferential weld in the HBR reactor vessel. The amount of nickel was determined as the mean value from the two welds fabricated using the same heat number of weld wire as the lower circumferential weld. Based on the chemistry, the staff determined a chemistry factor of 197.8.

The NRC staff used an unirradiated RT_{ndt} of -56°F, because this is the generic value for Combustion Engineering fabricated welds with Linde 1092 flux when plant specific values are not available. Until the licensee proves the traceability of archive weld material in the Millstone Unit 1 reactor to the lower circumferential weld in the HBR reactor vessel, a generic value of -56°F for unirradiated RT_{ndt} should be used. Using the same neutron fluence as the licensee used, the staff calculated an ART of 209.8 °F at the 1/4T location and 148.5°F at the 3/4T location.

Substituting the staff calculated ARTs, 148.5°F and 209.8°F, into the equations in SRP 5.3.2, the staff determined that the licensee's proposed P-T limits are acceptable for the requested 24 effective full power years (EFPY) because the P-T limits have enough margin to comply with the requirements in paragraphs IV.A.2 and IV.A.3 of Appendix G to 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature on P-T limits based on the RT_{ndt} of the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on a licensee reported flange RT_{ndt} of 60°F, the staff has determined that the proposed P-T limits satisfy Section IV.2 of Appendix G.

The discrepancy between the licensee's and the NRC staff's unirradiated RT_{ndt} for the lower circumferential weld will be resolved during the current staff review of the licensee's responses to GL 92-01.

3.0 SUMMARY

The NRC concludes that the proposed P-T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are acceptable for 24 EFPY because the limits conform to the requirements of Appendix G of 10 CFR Part 50. The proposed P-T limits may be incorporated into the HBR TS.

The discrepancy between the licensee's and the NRC staff's unirradiated RT_{ndt} for the lower circumferential weld will be resolved during the current staff review of the licensee's responses to GL 92-01.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of South 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, and there has been no public comment on such finding (58 FR 52980). 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.

Principal Contributor: John Tsao

Date: July 29, 1994