



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

CAROLINA POWER AND LIGHT COMPANY

DOCKET NO. 50-261

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

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 82
License No. DPR-23

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Carolina Power and Light Company (the licensee) dated October 14, 1983, 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:

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(B) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 82, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga
Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 4, 1984

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 82 FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Revise Appendix A as follows:

Remove Pages

3.1-5 through 3.1-8

3.1-21

3.1-22

Insert Pages

3.1-5 through 3.1-8

3.1-21

3.1-22

- 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 70°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 exposure 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 10 CFR 50. 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.

Basis

The ability of the large steel pressure vessel that contains the reactor core and its primary coolant to resist fracture constitutes an important factor in ensuring safety in the nuclear industry. The beltline region of the reactor pressure vessel is the most critical region of the vessel because it is subjected to neutron bombardment. The overall effects of fast neutron irradiation on the mechanical properties of low alloy ferritic pressure vessel

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 the ASME 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 value of RT_{NDT} , and in turn the operating limits of nuclear power plants, can be adjusted to account for the effects of radiation on the reactor vessel material properties. The radiation embrittlement or changes in mechanical properties of a given reactor pressure vessel still can be monitored by a surveillance program such as the Carolina Power & Light Company, H. B. Robinson Unit No. 2 Reactor Vessel Radiation Surveillance Program⁽¹⁾ where a surveillance capsule is periodically removed from the operating nuclear reactor and the encapsulated specimens tested. These data are compared to data from pertinent radiation effects studies and an increase in the Charpy

V-notch 30 ft-lb temperature (ΔRT_{NDT}) due to irradiation is added to the original RT_{NDT} to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} (RT_{NDT} initial + RT_{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 temperature, 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.

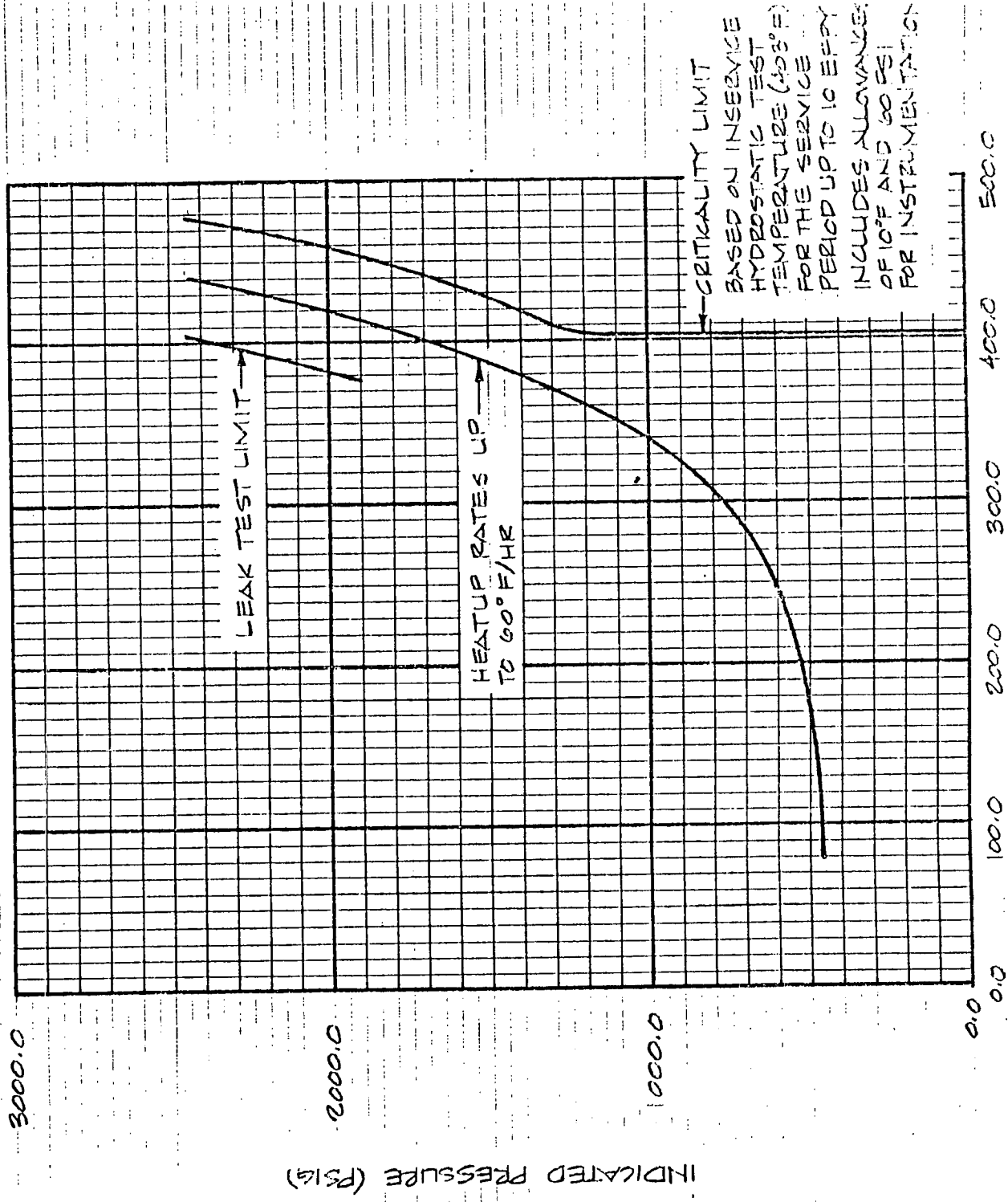
As in the case of heatup, allowable pressure temperature relations are generated for both steady state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of

interest. Again adjustments are made to account for pressure and temperature instrumentation error.

The overpressure protection system consists of two operable pressurizer Power Operated Relief Valves (PORVs) connected to the station instrument air system, a backup nitrogen supply, and the associated electronics.

References

1. S. E. Yanichko, "Carolina Power & Light Company, H. B. Robinson Unit No. 2 Reactor Vessel Radiation Surveillance Program," Westinghouse Nuclear Energy Systems - WCAP-7373 (January 1970).
2. E. B. Norris, "Reactor Vessel Material Surveillance Program for H. B. Robinson Unit No. 2, Analysis of Capsule V," Southwest Research Institute - Final Report SWRI Project No. 02-4397.



← CRITICALITY LIMIT
 BASED ON INSERVICE
 HYDROSTATIC TEST
 TEMPERATURE (400°F)
 FOR THE SERVICE
 PERIOD UP TO 10 EFPY
 INCLUDES ALLOWANCE
 OF 10°F AND 50 PSI
 FOR INSTRUMENTATION

INDICATED TEMPERATURE (DEG. F)

H.B. ROBINSON UNIT NO. 2 REACTOR COOLANT SYSTEM
 HEATUP LIMITATIONS APPLICABLE UP TO 10 EFPY

FIGURE 3.1-1

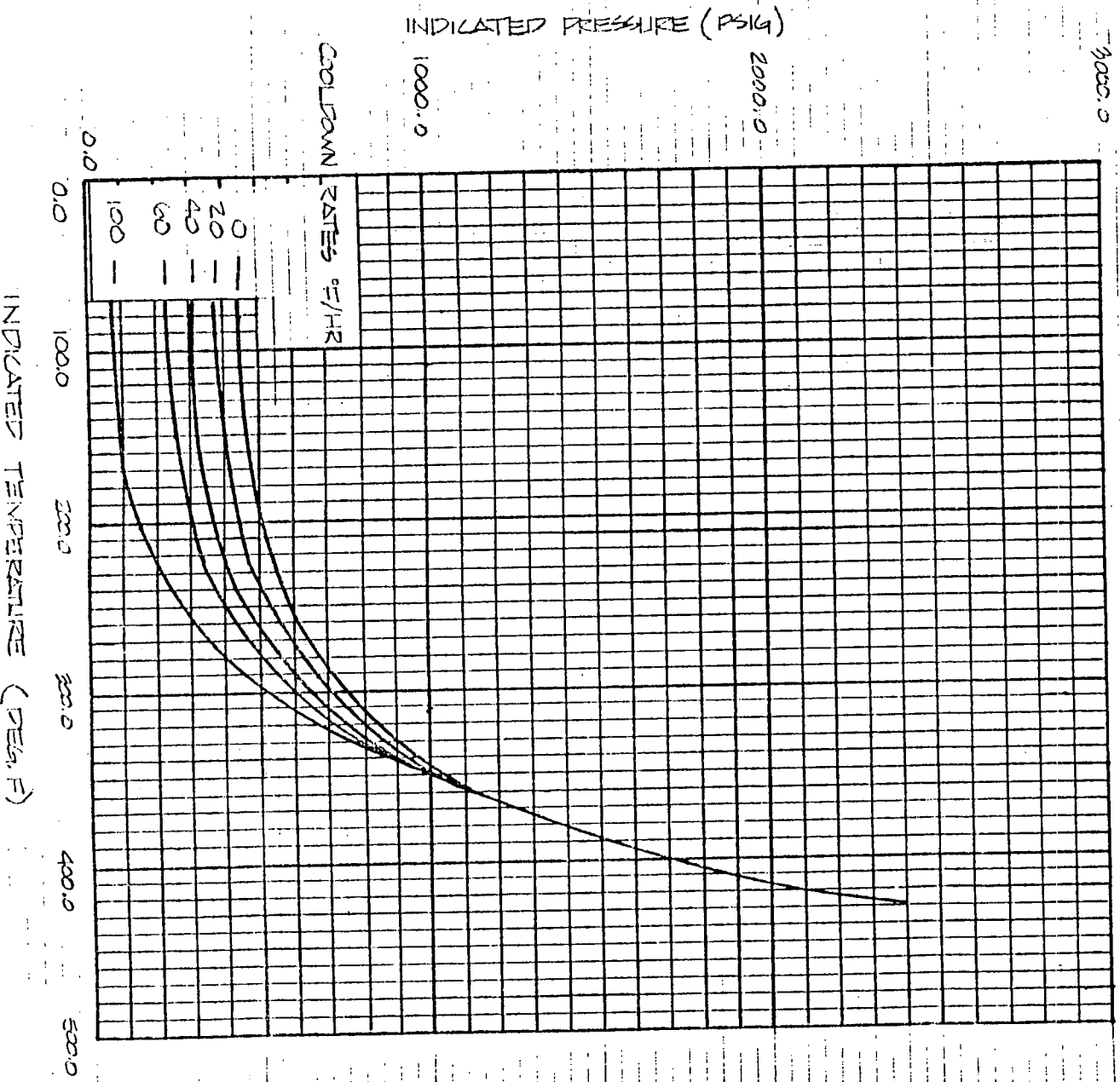
3.1-21

Amendment No. 82

CONTROLLING MATERIAL : WELD METAL
 COPPER CONTENT : 0.95 WT. %
 PHOSPHORUS CONTENT : 0.012 WT. %
 RT NOT INITIAL : 0°F
 RT NOT AFTER 10 EFPY : 1/4T, 274°F
 : 3/4T, 1290°F

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 10 EFPY.

INCLUDES 10° AND 60 PSI ALLOWANCE FOR INSTRUMENTATION.



H.E. ROBINSON UNIT NO. 2 REACTOR COOLANT SYSTEM
 COOLDOWN LIMITATIONS APPLICABLE UP TO 10 EFPY.

FIGURE 3.1-2