Docket No. 50-261

September 4, 1984

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Mr. E. E. Utley, Executive Vice President
Power Supply and Engineering & Construction
Carolina Power and Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Utley:

The Commission has issued the enclosed Amendment No. 82 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your request dated October 14, 1983.

The amendment would revise the Technical Specifications to incorporate new heatup and cooldown limitation curves.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular monthly Federal Register notice.

Sincerely,

/s/SVarga

Glode Requa, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No.82 to DPR-23
2. Safety Evaluation

cc: w/enclosures
See next page

ORB#1:DL	ORB#1:DL	ORB#1:DL	OELD	AD:OR:DL
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Mr. E. E. Utley
Carolina Power and Light Company

H. B. Robinson Steam Electric
Plant 2

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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:

(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, 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	Insert Pages
3.1-5 through 3.1-8	3.1-5 through 3.1-8
3.1-21	3.1-21
3.1-22	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 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.

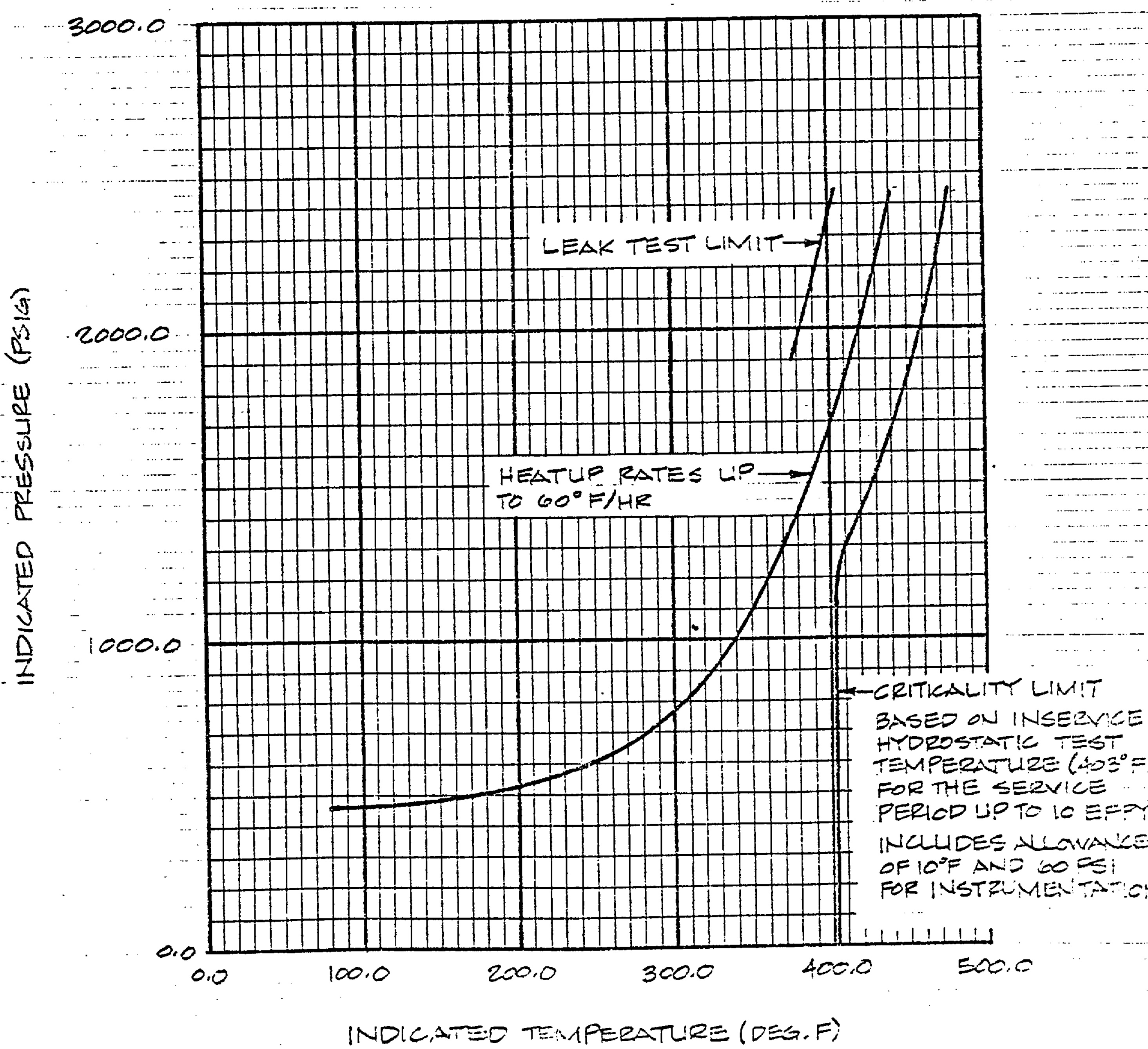
As in the case of heatup, allowable pressure temperature relations are generated for both steady state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest. Again adjustments are made to account for pressure and temperature instrumentation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



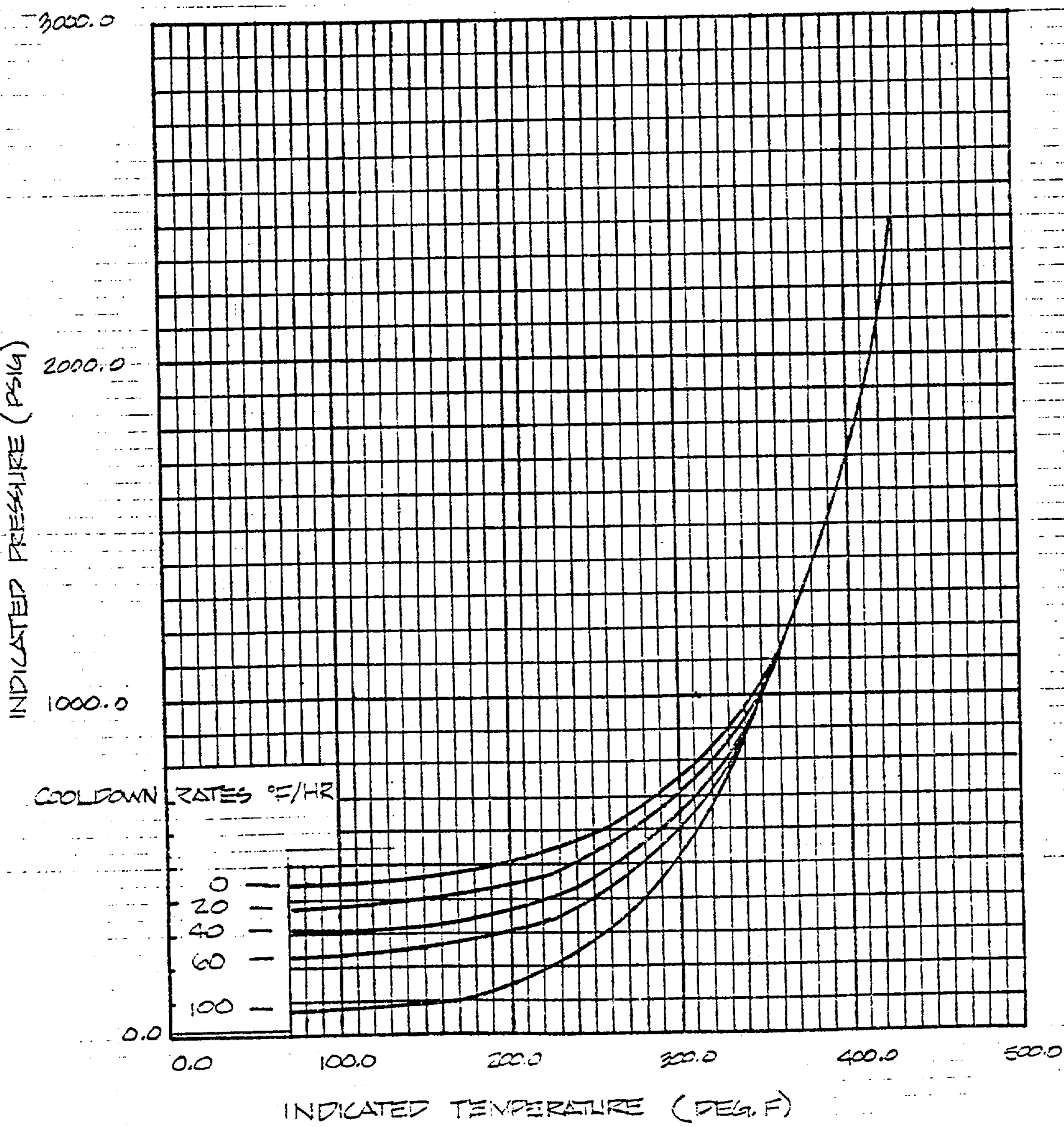
H.B. ROBINSON UNIT NO. 2 REACTOR COOLANT SYSTEM
HEATUP LIMITATIONS APPLICABLE UP TO 10 EFPY

FIGURE 3.1-1

CONTROLLING MATERIAL	:	WELD METAL
COPPER CONTENT	:	0.35 WT. %
PHOSPHORUS CONTENT	:	0.012 WT. %
RT_{NDT} INITIAL	:	0°F
RT_{NDT} AFTER 10 EFPY	:	1/4T, 274°F 3/4T, 129°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.B. ROBINSON UNIT NO. 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE UP TO 10 EFPY.

FIGURE 3.1-2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

CAROLINA POWER AND LIGHT COMPANY

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

DOCKET NO. 50-261

In letter from M. A. Mc Duffie to S. A. Varga dated October 14, 1983, the Carolina Power & Light Company (CPLC) requested a change to the H. B. Robinson Steam Electric Plant Unit No. 2 (HBR-2) reactor vessel pressure temperature limits, Figures 3.1-1 and 3.1-2 of the plants technical specifications. CPLC indicates that the pressure-temperature limits will meet the requirements of Appendix G, 10 CFR 50, for a period of time corresponding to 10 effective full power years (EFPY).

Pressure-temperature limits must be calculated in accordance with the requirements of Appendix G, 10 CFR 50, which became effective on July 26, 1983. Pressure-temperature limits that are calculated in accordance with the reqirements of Appendix G, 10 CFR 50, are dependent upon the initial RT_{NDT} for the limiting materials in the beltline and closure flange regions of the reactor vessel and the increase in RT_{NDT} resulting from neutron irradiation damage to the limiting beltline material.

The HBR-2 reactor vessel was procured prior to the issuance of the Appendix G, 10 CFR 50 regulation. However, the HBR-2 reactor vessel materials must meet the safety margins and testing requirements of the regulation. Appendix G, 10 CFR 50, requires that samples from each reactor vessel material be fracture toughness tested to determine their initial (unirradiated) RT_{NDT}. The limiting reactor vessel materials were not fracture toughness tested to determine their initial RT_{NDT}. The initial RT_{NDT} of the limiting reactor vessel materials were determined using the criteria in Branch Technical Position - MTEB 5-2 which is documented in Standard Review Plan Section 5.3.2 of NUREG-0800.

The limiting beltline material is the weld metal which was fabricated using Linde 1092 flux and RACO 3 wire with nickel added. The licensee indicates that Branch Technical Position MTEB 5-2 results in an initial RT_{NDT} of 0°F for this material. The licensee indicates that the limiting closure flange region material is the vessel flange forging, in which the initial RT_{NDT} is estimated as 40°F.

The increase in RT_{NDT} resulting from neutron irradiation damage depends upon the predicted amount of neutron fluence and the rate of embrittlement of the limiting reactor vessel beltline material. The licensee indicates that at 7.48 EFPY the neutron fluence at the inside surface of the limiting weld was $13.5 \times 10^{18} n/cm^2$ and that the subsequent rate of increase in fluence per EFPY would be $1.05 \times 10^{18} n/cm^2$ (Reported in the meeting summary memorandum dated February 11, 1983 between CP&L and the NRC staff). This rate of increase results in a predicted neutron fluence for the limiting weld at the inside surface of $1.61 \times 10^{19} n/cm^2$ at 10 EFPY. The rate of increase in neutron fluence was reviewed and accepted by the staff.

The incease in RT_{NDT} resulting from neutron irradiation damage was estimated by the licensee using the upper limit lines in Regulatory Guide 1.99, Rev. 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." Table 1 compares the observed increase in RT_{NDT} of the surveillance weld metal to that predicted using the upper bound limit line in Regulatory Guide 1.99 Rev. 1 and the Guthrie Mean Formula in Commission Report SECY 82-465. The surveillance weld metal is not from the same heat of flux and wire as that used in the fabrication of the limiting beltline weld. However, it may be used to evaluate the effect of irradiation on the beltline weld, since it was fabricated using the same type of flux and wire as the limiting beltline weld. The surveillance material test results indicate that the increase in RT_{NDT} of surveillance weld metal is significantly less than that predicted by the upper bound limit line of Regulatory

Guide 1.99, Rev. 1. Hence, the Regulatory Guide upper bound limit line should provide a conservative estimate as to the amount of increase in RT_{NDT} resulting from neutron irradiation for the HBR-2 limiting reactor vessel beltline weld.

We have used the unirradiated RT_{NDT} for beltline and closure flange materials, which were previously discussed, the neutron fluence estimates of the licensee, the Regulatory Guide 1.99 Rev. 1 method of estimating neutron irradiation damage, and Standard Review Plan 5.3.2 method of calculating pressure-temperature limits to evaluate the applicant's proposed pressure-temperature limits. Our evaluation indicates that the proposed pressure-temperature limit curves meet the safety margins of Appendix G, 10 CFR 50, for a period of time corresponding to 10 EFPY. Hence, the proposed curves may be incorporated into the HBR-2 Technical Specification.

Environmental Consideration

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The 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 occupation radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 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 this amendment.

Conclusion

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: September 4, 1984

Principal Contributor:

B. Elliot

TABLE 1

Comparison of Observed and Calculated Increase in RT_{NDT}

of Weld Metal in Surveillance Capsules

Surveillance Capsule	Capsule Fluence (n/cm^2)	Increase in RT_{NDT} (°F) Observed	Calculated Using R.G. 1.99 Rev.1 (2)	Calculated Using Guthrie (1) Mean Formula
Capsule T (3)	4.11×10^{19}	285	320	334
Capsule V (4)	4.51×10^{18}	175	220	184

1. Guthrie Formula is identified on page E-6 of Commission Report SECY-82-465, "Pressurized Thermal Shock (PTS)"

2. Calculation using the upper limit line of Regulatory Guide 1.99 Rev. 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials"

3. Capsule T test results are reported in Westinghouse Report WCAP 10304, "Analysis of Capsule T From H. B. Robinson Unit 2 Reactor Vessel Radiation Surveillance Program

4. Capsule V test results are reported in Southwest Research Institute Report, "Reactor Vessel Material Surveillance Program For H. B. Robinson Unit No. 2 Analysis of Capsule V"