NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

DOCKET NUMBER 50-261

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TO:	FROM:	DATE OF DOCUMENT
Mr Reid	Carolina Power & Light Co Raleigh, NC E E Utley	1-4-77
		DATE RECEIVED 1-6-77

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DESCRIPTION	ENCLOSURE
Ltr notarized 1-4-76.....trans the following: 2p PLANT NAME: H B Robinson	Amdt to OL/Change to Tech Specs: Consisting of revisions incorporating new heatup & cool-down curves resulting from analysis of samples in the irratiation specimen measurement program ....(43 cys encl rec'd) 6p DO NOT REMOVE ACKNOWLEDGED

SAFETY — FOR ACTION/INFORMATION — ENVIRO 1-11-77 ehf

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Regulatory Docket File

CP&L

Carolina Power & Light Company

January 4, 1977

FILE: NG-3514(R)

SERIAL: NG-77-005

Director of Nuclear Reactor Regulation
ATTN: Mr. Robert W. Reid, Chief
Operating Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

H. B. ROBINSON STEAM ELECTRIC PLANT UNIT NO. 2
DOCKET 50-261
LICENSE NO. DPR-23
REQUEST FOR LICENSE AMENDMENT - REVISION OF TECHNICAL SPECIFICATION

Dear Mr. Reid:

In accordance with the Code of Federal Regulations, Title 10, Part 50.90 and Part 2.101, and Technical Specification 3.1.2.4, Carolina Power & Light Company (CP&L) hereby requests a revision to the Technical Specifications for H. B. Robinson Unit 2. This revision incorporates new heatup and cooldown curves resulting from analysis of samples in the irradiation specimen measurement program.

Capsule V was removed from H. B. Robinson Unit No. 2 during the 1975 refueling outage and was analyzed by Southwest Research Institute. The complete analysis report on this capsule is being submitted separately and in accordance with 10CFR50 Appendix H and Regulatory Guide 10.1.

The revised heatup and cooldown curves derived from this analysis are attached as replacement pages for the unit Technical Specifications. Other replacement pages simplify the bases for this section by deleting some discussion which is merely copied from the referenced reports.

An additional revision to the Technical Specifications is required as a result of incorporating the revised heatup and cooldown curves. This revision deletes the requirement for a differential pressure test across the tube walls of the steam generators, as is presently required by Specification 4.7.2. With the revised heatup and cooldown curves now being proposed, it will be impractical to conduct the differential pressure testing of steam generator tubes as specified without exceeding hydrostatic test pressures for the irradiated reactor pressure vessel or desired maximum pressures on fuel rods. Present testing now requires reactor coolant system temperatures below 427°F with primary pressure at 2250 psia to achieve the required differential pressure of 1900 psi between the primary and secondary systems. This is in conflict with the minimum temperature of 438°F presented on the heatup curve for achieving a pressure of 2250 psia. This testing has been required since late 1972 and has never resulted in the discovery of any leaking or ruptured tube. With the additional tube inspections conducted over the past four years using eddy current testing and the definition and incorporation of conservative tube plugging criteria, it is CP&L's position that removal of this unique requirement from the H. B. Robinson Technical Specifications will not result in an increased risk to the health and safety of the public and will resolve this conflict in requirements.

The changes are indicated by vertical lines in the margin of the attached pages. We request your prompt attention to this revision since the applicability of the present heatup and cooldown curves extends only to approximately February 11, 1976.

On August 13, 1976, the Company submitted a retyped version of the H. B. Robinson Technical Specifications and asked that the NRC issue the retyped version to replace the present version. CP&L would like to reemphasize the need for these retyped Technical Specifications. Several amendments have been requested by the Company and issued by the NRC since the submittal, and inconsistencies in format between the NRC and Company versions remain a problem. Your expeditious attention to the issuance of the retyped version is requested.

As required by Commission regulations, this submittal is signed under oath by a duly authorized officer of the Company.

Yours very truly,

E. E. Utley
Senior Vice President
Power Supply

MFP/dkm

Attachments

Sworn to and subscribed before me this 4th day of January, 1977.

Franklin Murray
Notary Public

FRANKLIN MURRAY
NOTARY PUBLIC
WAKE COUNTY, N. C.

My Commission Expires October 4, 1981

surveillance capsule is periodically removed from the operating nuclear reactor and the encapsulated specimens tested. The increase in the Charpy V-notch 50 ft-lb temperature (Δ RT_{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.

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.

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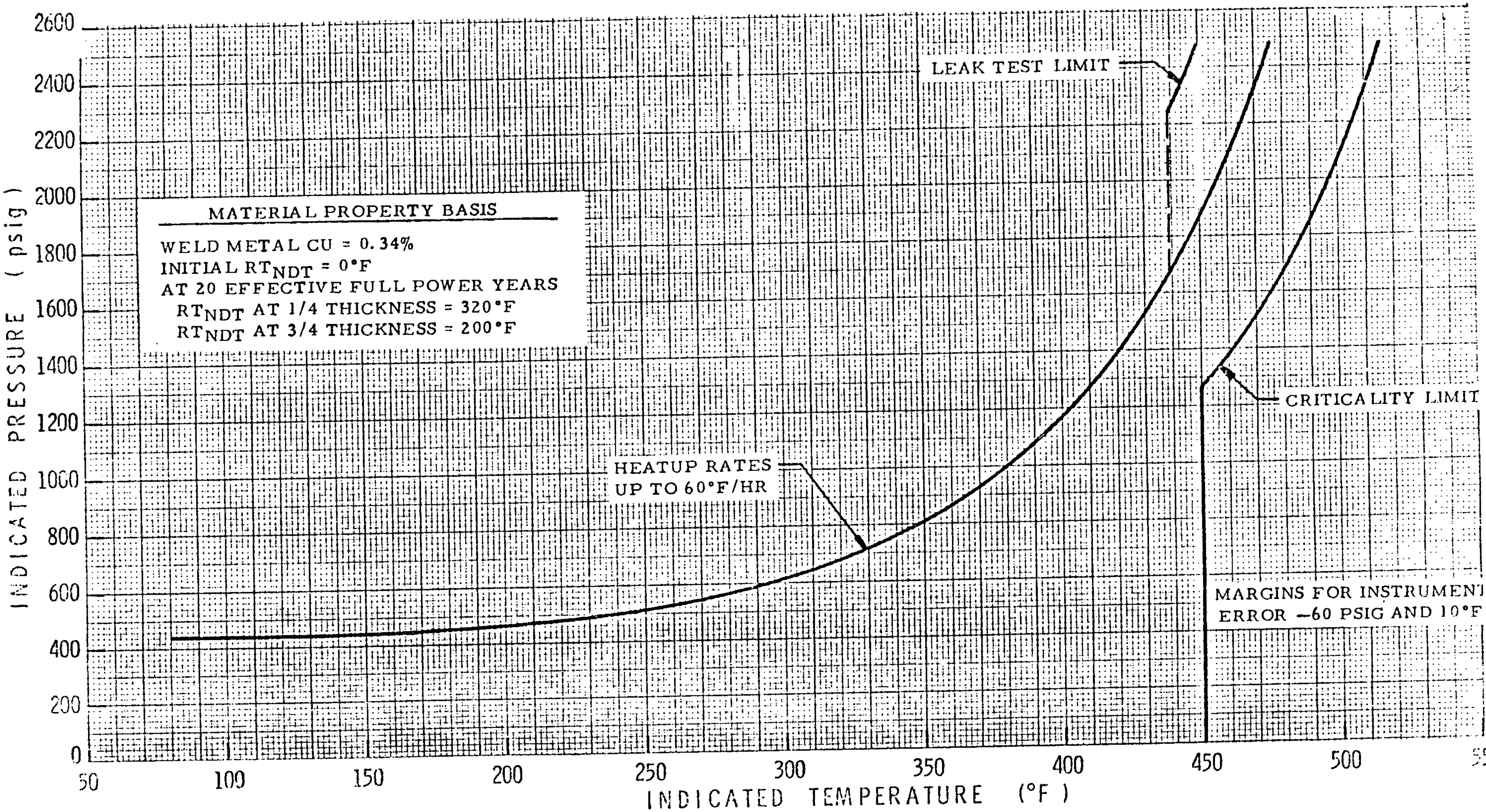

Figure 3.1-1 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE FOR PERIODS UP TO 20 EFFECTIVE FULL POWER YEARS

4.7 SECONDARY STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to periodic testing of secondary system components and surveillance of secondary coolant.

Objective

To verify the ability of secondary system components to function as required and to prevent system degradation.

Specification

4.7.1 The main steam stop valves shall be tested at each refueling interval of each 15 $\pm$ 3 months, whichever occurs first. Closure time of five seconds or less shall be verified. The valves are tested under no flow and at no load conditions.

Basis

The main steam stop valves serve to limit an excessive Reactor Coolant System cooldown rate and resultant reactivity insertion following a main steam break incident. Their ability to close upon signal should be verified at each scheduled refueling shutdown. A closure time of five seconds was selected as being consistent with expected response time for instrumentation as detailed in the steam line break incident analysis.

References

FSAR - Section 10.4
FSAR - Section 14.2.5