NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

DOCKET NUMBER: 50-261

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TO:	FROM:	DATE OF DOCUMENT
Mr. Robert W. Reid	Carolina Power & Light Company Raleigh, North Carolina E. E. Utley	12/22/77
		DATE RECEIVED: 12/30/77

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Notorized 12/22/77....trans the following:	License No. DPR-23 Appl for Amend: tech specs proposed change concerning provision for operability & surveillance requirements for proposed overpressure system and imposition of other operating conditions to improve overpressure protection.........

REACTOR VESSEL OVERPRESSURIZATION
DISTRIBUTION PER G. ZECH 10-21-76

PLANT NAME: H. B. Robinson Unit No. 2
RJL 12/30/77 (1-P) (10-P)

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Carolina Power & Light Company

December 22, 1977

FILE: NG 3514 (R)

SERIAL: NG-77-1457

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

RECEIVED DEC 30 1977 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

RE: H. B. ROBINSON STEAM ELECTRIC PLANT UNIT NO. 2
DOCKET NO. 50-261
OPERATING LICENSE NO. DPR-23
REQUEST FOR LICENSE AMENDMENT -
OVERPRESSURE PROTECTION TECHNICAL SPECIFICATIONS

Dear Mr. Reid:

In accordance with the Code of Federal Regulations, Title 10, Parts 2.101 and 50.90, Carolina Power & Light Company hereby requests a revision to the Technical Specifications for H. B. Robinson Unit No. 2. The requested changes provide operability and surveillance requirements for our proposed overpressure systems and impose other operating conditions to improve overpressure protection. The requested Technical Specifications are consistent with the criteria which you sent to us on December 2, 1977, as revised on December 9, 1977.

Revised pages to implement the requested changes are attached. The changes are marked by a vertical bar in the right hand margin.

Yours very truly,

E. E. Utley
Senior Vice President
Power Supply

CSB/mf
Attachment

Sworn to and subscribed before me this 22nd day of December, 1977.

FRANKLIN MURRAY NOTARY PUBLIC WAKE COUNTY, N.C.

My Commission Expires October 4, 1981

Franklin Murray
Notary Public

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Carolina Power & Light Company

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3.0 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objective

To specify those Reactor Coolant System conditions which must be met to assure safe reactor operation.

Specification

3.1.1 Operational Components

3.1.1.1 Coolant Pumps

a. At least one reactor coolant pump or the Residual Heat Removal System shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

b. When the reactor is critical, except for special low power tests during initial start-up testing, at least one reactor coolant pump shall be in operation.

c. Reactor power shall not exceed 10% rated power unless at least two reactor coolant pumps are in operation.

d. Reactor power shall not exceed 45% of rated power with only two pumps in operation.

e. A reactor coolant pump may be started (or jogged) only if there is a steam bubble in the pressurizer or the steam generator temperature is no higher than 50°F higher than the temperature of the reactor coolant system.

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, and 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 reactor coolant system temperature is below 350°F and the RCS system is solid and not vented to the containment except that one PORV may be inoperable for seven days provided that the other PORV and the necessary logic to initiate its action are operable.

e. Operation of the overpressure protection system to relieve a pressure transient must be reported as required in Section 6.9.3.

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 heatup and cooldown rates shall not exceed $100^{o}F/hr$. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than $320^{o}F$.

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, Summer 1972 Addenda, Non-Mandatory Appendix G. 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 under certain conditions of irradiation. In pressure vessel material, the most serious mechanical property change is the reduction in the upper shelf impact strength. Accompanying the 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. The increase in the Charpy V-notch 50 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 (PORV's) connected to the station instrument air system, a backup nitrogen supply, and 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.

3.3.1.3 When the reactor is in the hot shutdown condition, the requirements of 3.3.1.1 and 3.3.1.2 shall be met. Except that the accumulators may be isolated, and in addition, any one component as defined in 3.3.1.2 may be inoperable for a period equal to the time period specified in the subparagraphs of 3.3.1.2 plus 48 hours, after which the plant shall be placed in the cold shutdown condition utilizing normal operating procedures. The safety injection pump power supply breakers must be racked out when the reactor coolant system temperature is below 350°F and the system is not vented to containment atmosphere.

3.3.2 Containment Cooling and Iodine Removal Systems

3.3.2.1 The reactor shall not be made critical, except for low temperature physics tests, unless the following conditions are met:

a. The spray additive tank contains not less than 2505 gallons of solution with a sodium hydroxide concentration of not less than 30% by weight.

b. Two containment spray pumps are operable.

c. Four fan cooler units are operable.

d. All essential features, including valves, controls, dampers, and piping associated with the above components are operable.

e. The system which automatically initiates the sodium hydroxide addition to the containment spray simultaneously to the actuation of the containment spray is operable.

TABLE 4.1-1 (Continued)

	Channel Description	Check	Calibrate	Test	Remarks
21.	Containment Sump Level	N.A.	R	N.A.	
22.	Turbine Trip Set Point**	N.A.	R	R	
23.	Accumulator Level and Pressure	S	R	N.A.	
24.	Steam Generator Pressure	S	R	M	
25.	Turbine First Stage Pressure	S	R	M	
26.	Emergency Plant Portable Survey Instruments	M	R	M	
27.	Logic Channel Testing	N.A.	N.A.	M(1)	(1) During hot shutdown and power operations. When periods of reactor cold shutdown and refueling extend this interval beyond one month, the test shall be performed prior to startup.
28.	Turbine Overspeed Protection Trip Channel (Electrical)	N.A.	R	M	
29.	4 Kv Frequency	N.A.	R	R	
30.	Control Rod Drive Trip Breakers	N.A.	N.A.	M	
31.	Overpressure Protection System	N.A.	R	M	

**Stop valve closure or low EH fluid pressure

S	- Each Shift	M	- Monthly
D	- Daily	Q	- Quarterly
W	- Weekly	P	- Prior to each startup if not done previous week
B/W	- Every two weeks	R	- Each Refueling Shutdown
A/R	- After each refueling startup	N.A.	- Not applicable

TABLE 4.1-3 (Continued)

		Check	Frequency	Maximum Time Between Tests
13.	Turbine Inspection	Visual, Magnaflux and Die Penetrant	Every five years	6 years
14.	Fans and Associated Charcoal and Absolute Filters for Control Room and Residual Heat Removal Compartments	Fans functioning. Charcoal and absolute filter efficiencies checked $\geq 99\%$ for Iodine and 0.3 Micron Particulate. DOP Test on absolute filters. Freon Test on Charcoal Filter Units	Each refueling shutdown	NA
15.	Isolation Seal Water System	Functioning	Each refueling shutdown	NA
16.	Overpressure Protection System	Functioning	Each refueling shutdown	NA

*NA - Not applicable

(4) Abnormal degradation of systems other than those specified in 6.9.2.a(3) above designed to contain radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in Technical Specifications need not be reported under this item.

6.9.3 Special Reports

Special reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

	Area	Reference	Submittal Date
a.	Containment Leak Rate Testing	4.4	Upon completion of each test
b.	Initial Containment Structural Test	4.4	Within three months following completion of test
c.	Fuel Inspection	2.1	Upon completion of the inspection at second and third refueling outages
d.	Inservice Inspection Evaluation	4.2	After five years of operation
e.	Containment Sample Tendon Surveillance	4.4	Upon completion of the inspection at 5 and 25 years of operation
f.	Post-operational Containment Structural Test	4.4	Upon completion of the test at 3 and 20 years of operation
g.	Overpressure Protection System Operation	3.1.2.1e	Within 30 days of operation