May 22, 1987

Docket No. 50-261

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~~Docket No. 50-261~~
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Mr. E. E. Utley, Senior Executive Vice President
Power Supply and Engineering & Construction
Carolina Power and Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Utley:

SUBJECT: PAGES OMITTED FROM AMENDMENT NO. 113 ISSUED MARCH 31, 1987 FOR H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 (TAC NO. 64199)

On March 31, 1987, the Commission issued Amendment No. 113 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2. The amendment revised the Technical Specifications (TSs) by replacing the heatup and cooldown curves in Figures 3.1-1 and 3.1-2 with two sets of curves. The replacement curves are applicable for use up to 12.5 and 15 effective full power years, respectively.

Due to an administrative error, corresponding changes to text pages of the TSs were inadvertently omitted from the amendment when issued. Enclosed are the omitted pages for incorporation into the TSs. The changes to these TS pages in no way affect the staff's evaluation and conclusion that support the changes to the Technical Specifications for H. B. Robinson, Unit 2. TS page 3.1-5 has been changed to reflect the current TSs. This correction has been discussed with and agreed to by your staff.

Sincerely,

Kenneth T. Eccleston

Kenneth T. Eccleston, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II

Enclosure:
As stated

cc: See next page

PE:PD21:DRPR	PM:PD21:DRPR	LA:PD21:DRPR	D:PD21:DRPR
GRequa/vag	KEccleston	PAnderson	EAdensam
05/20/87	05/20/87	05/20/87	05/ /87

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ENCLOSURE TO CORRECTION LETTER FOR

AMENDMENT NO. 113 FACILITIES OPERATING

LICENSE NO. DPR-23

DOCKET NO. 261

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.1-4	3.1-4
3.1-5	3.1-5
3.1-5a	3.1-5a
3.1-11	3.1-11

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 3.1-1a and Figure 3.1-2a (for vessel exposure up to 12.5 EFPY) or Figure 3.1-1b and Figure 3.1-2b (for vessel exposure up to 15 EFPY). The 15 EFPY curves may be used for operation prior to the end of 12.5 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 3.1-2a or 3.1-2b (as appropriate). This rate shall not exceed 100°F/hr. in any one hour. The limit lines for cooling rates between those shown in Figure 3.1-2a or Figure 3.1-2b may be obtained by interpolation.

c. Primary system hydrostatic leak tests may be performed as necessary, provided the temperature limitation as noted on Figure 3.1-1a or Figure 3.1-1b (as appropriate) 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 3.1-1b and 3.1-2b 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 3.1-1b and 3.1-2b 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.

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

3.1.3 Minimum Conditions for Criticality

3.1.3.1 Except during low power physics tests, the reactor shall not be made critical at any temperature, above which the moderator temperature coefficient is greater than:

a) +5.0 pcm/°F less than 50% of rated power, or
b) +5.0 pcm/°F at 50% of rated power and linearly decreasing to 0 pcm/°F at rated power.

3.1.3.2 In no case shall the reactor be made critical above and to the left of the criticality limit shown on Figure 3.1-1a or 3.1-1b (as appropriate per 3.1.2.1).

3.1.3.3 When the reactor coolant temperature is in a range where the moderator temperature coefficient is greater than as specified in 3.1.3.1 above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least 1% until normal water level is established in the pressurizer.

Basis

During the early part of fuel cycle, the moderator temperature coefficient may be slightly positive at low power levels. The moderator coefficient at low temperatures or powers will be most positive at the beginning of the fuel cycle, when the boron concentration in the coolant is the greatest. At all times, the moderator coefficient is calculated to be negative in the high power operating range, and after a very brief period of power operation, the coefficient will be negative in all circumstances due to the reduced boron concentration as Xenon and fission products build into the core. The requirement that the reactor is not to be made critical when the moderator coefficient is more positive than as specified in 3.1.3.1 above has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase of moderator temperature or decrease of coolant