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AUTH.NAME AUTHOR AFFILIATION
UTLEY,E.E. CAROLINA POWER & LIGHT CO.
RECIP.NAME RECIPIENT AFFILIATION
SCHWENCER,A. OPERATING REACTORS BRANCH 1

SUBJECT: REQUESTS LICENSE AMEND TO ALLOW SMALL POSITIVE MODERATOR TEMP COEFFICIENT DURING REACTOR STARTUP & VARIOUS ADMINISTRATIVE CHANGES.FORWARDS PROPOSED REVISIONS TO TECH SPECS,SUPPORTING ANALYSIS & $5,200 AMEND FEE.

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April 18, 1979

FILE: NG-3514(R) SERIAL: GD-79-1038

Office of Nuclear Reactor Regulation
ATTENTION: Mr. Albert Schwencer, Chief
Operating Reactors Branch No. 1
United States Nuclear Regulatory Commission
Washington, D. C. 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261
LICENSE NO. DPR-23
LICENSE AMENDMENT TO ALLOW A SMALL POSITIVE
MODERATOR TEMPERATURE COEFFICIENT AND VARIOUS ADMINISTRATIVE CHANGES

Dear Mr. Schwencer:

In accordance with the Code of Federal Regulations, Title 10, Part 50.90 and Part 2.101, Carolina Power & Light Company requests the attached revisions to the Technical Specifications for its H. B. Robinson Steam Electric Plant. Changes to the Technical Specifications are indicated by a vertical line in the right-hand margins of the affected pages which are enclosed.

The proposed change to Section 3.1.3 would allow a small positive moderator temperature coefficient, 2.0 pcm/°F, during reactor startup. This condition could only exist at the beginning of the fuel cycle and at low power levels. After a very brief period of power operation, the coefficient would become negative due to reduced boron concentration as Xenon and fission products build into the core. As the attached plant transient analysis shows, this change causes no design, operational, or safety criteria to be violated.

The other attached changes are administrative in nature. They delete references to early cycle requirements, correct figures or clarify wording. The change to Figure 3.10-2 changes the Zero Boron Shutdown Margin to 1.77. This is the number that was actually used in the Safety Analysis, and the graph is being changed to conform with it.

In accordance with 10CFR170.12(c), we have determined that this request constitutes one Class III amendment and one Class II amendment. Our check for $5,200 is enclosed as payment for this amendment fee.

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w/check
$5,200.00

411 Fayetteville Street • P. O. Box 1551 • Raleigh, N. C. 27602

If you staff has any questions concerning the attached information, we will be glad to discuss them either by telephone or at a meeting with representatives of your staff.

Yours very truly,

E. E. Utley
Senior Vice President
Power Supply

JJS/mf
Attachments

Sworn to and subscribed before me this 18th day of April, 1979

Franklin Murray
Notary Public

My Commission Expires: October 4, 1981

FRANKLIN MURRAY
NOTARY PUBLIC
WAKE COUNTY, N. C.

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 2.0 pcm/°F.

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 3.1-1.

3.1.3.3 When the reactor coolant temperature is in a range where the moderator temperature coefficient is greater than 2.0 pcm/°F, 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 2.0 pcm/°F 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 pressure. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient[(2)] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The heatup curve of Figure 3.1-1 includes criticality limits which are required by 10 CFR Part 50, Appendix G, Paragraph IV.A.2.c. Whenever the core is critical, additional safety margins above those specified by the ASME Code Appendix G methods, are imposed. The core may be critical at temperatures equal to or above the minimum temperature for the inservice hydrostatic pressure tests as calculated by ASME Code Appendix G methods, and an additional safety margin of 40°F must be maintained above the applicable heatup curve at all times.

If the specified shutdown margin is maintained (Section 3.10), there is no possibility of an accidental criticality as a result of an increase of moderator temperature or a decrease of coolant pressure.[1]

The requirement for bubble formation in the pressurizer when the reactor has passed the threshold of one percent subcriticality will assure that the Reactor Coolant System will not be solid when criticality is achieved.

References

(1) FSAR Table 3.2.1-1
(2) FSAR Figure 3.2.1-10

SUPPORTING ANALYSIS FOR H. B. ROBINSON POSITIVE MODERATOR TEMPERATURE COEFFICIENT

A plant transient analysis of H. B. Robinson Unit 2 was performed to assess the impact of a positive moderator temperature coefficient on thermal margin. The positive moderator temperature coefficient represents a predicted hot zero power value and was conservatively applied to the full power transient analyses.

The plant transients analyzed were those previously identified by ENC[1] to be the most severe with respect to DNB margin; i.e.,

(1) Locked rotor at BOC.

(2) Loss of reactor coolant flow at BOC.

The transients were analyzed with the plant parameters listed in Table 1 and with neutronics parameters as listed in Table 2. Conservative multipliers were applied to the reactivity coefficients. A value of 1.25 times the moderator coefficient and 0.8 times the Doppler coefficient which is consistent with previous ENC application. In addition, the current analysis applied measurement uncertainties as indicated in Table 1. The results of the analyses are summarized in Table 3. The minimum DNB reached was 1.58 for the locked rotor transient. This result is well above the established MDNBR limit of 1.30.* The relative effect of changing the moderator coefficient from zero to 2.0 pcm/°F is provided in Table 4.

Although previous ENC analyses had considered additional plant transients, a review of those transients for which the system response would be adversely affected by a positive moderator coefficient (power increase transients such as loss of load, rod withdrawal, etc., at BOC) shows that the fuel response and resultant thermal margin is essentially not affected. In these transients, although the positive moderator coefficient contributes to increased power, sufficient thermal lag exists such that the effective rod heat flux before reactor scram is not significantly increased. Thus, these transients were not analyzed.

Transients previously analyzed at EOC conditions are not affected by this BOC parameter change.

As a means of comparing the current analyses with the previous transient analysis, a comparison of controlling PTSPWR2 input and the resulting DNBRs is presented in Table 5. Aside from the differences in power

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* DNBR values less tha 1.30 are acceptable for the locked rotor transient as it is a Class III incident for which the 1.30 value corresponds to an accepted fuel damage point.

level and peaking factors used between the two calculations, the controlling parameter which resulted in the low transient MDNBRs in the previous calculations (XN-75-14) was the initial steady state MDNBR. In these analyses, this initial MDNBR was forced to a value of 1.87. The 1.87 value was used since this was the steady state MDNBR obtained by a previous fuel vendor analysis. This posture was taken to: (1) insure a conservative anticipated steady state MDNBR value, and (2) provide a base for comparison to "previous vendor analysis" as required in the licensing procedure at the time this report was published. The MDNBR calculations are for the high enthalpy rise subchannel. The initial MDNBR value in the current analysis was verified by an independent analysis using the procedures outlined in Reference 2.

Based on the results for the plant transients using a positive moderator temperature coefficient, it concluded that no design, operational or safety criteria, are violated.

References:

(1) XN-75-14, "Plant Transient Analysis of the H. B. Robinson Unit 2 PWR at 2300 MWt."

(2) XN-75-48, "Definition and Justification of Exxon Nuclear Company DNB Correlation for BWRs."

Table 1

OPERATING PARAMETERS USED IN PTS-PWR ANALYSIS
POSITIVE MODERATOR TEMPERATURE COEFFICIENT
OF H. B. ROBINSON UNIT 2

	Current Value
Core	
Total Core Heat Output, MWt	2346*
Heat Generated in Fuel, %	97.4
Pressure, psia	2220*
Hot Channel Factors	
Heat Flux, Total	2.62
Enthalpy Rise, F_H^N	1.58
Coolant Flow Rate, lbs/hr	101.5×10^6
Coolant Temperature, °F	
Core Inlet	550.5*
Heat Transfer	
Average Heat Flux, Btu/hr-ft^2	182,800
Steam Generators	
Steam Flow, lb/hr	3.423×10^6
Steam Temperature, °F	523
Steam Pressure, psia	850
Feedwater Temperature, °F	441.7

* Include uncertainties: +2% in power, +4.4°F in core inlet temperature, and -30 psi in primary pressure.

Table 2

NEUTRONICS PARAMETERS*

Parameter	BOC	EOC
Moderator Temperature Coefficient (pcm/°F) (pcm = 10^{-5} $\Delta\rho$)	+2.0	-32.0
Moderator Pressure Coefficient (pcm/psi)	-0.02	0.0
Delayed Neutron Fraction (%)	0.600	0.535
Total Rod Worth ($)	10.01	9.68
Doppler Coefficient ($\Delta\rho$/°F x 10^5)	-1.0	-1.5

* Design conservatism multipliers as defined were applied in actual analysis.

Table 3

TRANSIENT RESULTS

Variable	Locked Rotor	Three Pump Coastdown
Peak Core Power (MWt)	2410	2384
Peak Primary Pressure (psia)	2269	2270
Maximum Core Heat Flux	182,800	182,800
MDNBR (w/o rod bow)	1.58	1.89
MDNBR (w/rod bow)	1.40	1.69

Table 4

EFFECT OF POSITIVE MODERATOR COEFFICIENT ON MDNBR FOR TRANSIENTS ANALYZED

	Moderator Coefficient pcm/°F		ΔMDNBR*
	0	+2.0	
Locked Rotor, MDNBR	1.65	1.58	0.07
Loss-of-Flow, MDNBR	1.96	1.89	0.07

* ΔMDNBR = MDNBR reduction due to positive moderator coefficient.

Table 5

COMPARISON OF H. B. ROBINSON TRANSIENT DATA

	XN-75-14	Current Analysis*
Initial Power Level (MWt)	2300	2346
F_Q^T	2.65	2.62
$F_{\Delta H}^N$	1.55	1.58
Initial MDNBR	1.87	2.29
Transient MDNBR		
Loss of Flow	1.68	1.96
Locked Rotor	1.40	1.65

* With a moderator coefficient of zero.

Other Technical
Specification
Changes

The safety limit curves given in Figures 2.1-1 and 2.1-2 are for constant flow conditions. These curves would not be applicable in the case of a loss of flow transient. The evaluation of such an event would be based upon the analysis presented in Section 14.1 of the FSAR.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure, and thermal power level that would result in a DNB ratio of less than 1.30[3] based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 574.2°F, and a steady state nominal operating pressure of 2235 psig. Allowances are made in initial conditions assumed for transient analyses for steady state errors of +2% in power, +4°F in Reactor Coolant System average temperature, and ±30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 percent less than the value at nominal full power operating conditions. The steady state nominal operating parameters and allowances for steady state errors given above are also applicable for two loop operation except that the steady state nominal operating power level is less than or equal to 45%.

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References

(1) FSAR Section 3.2.2
(2) FSAR Section 14.1.3
(3) FSAR Section 7.2.1

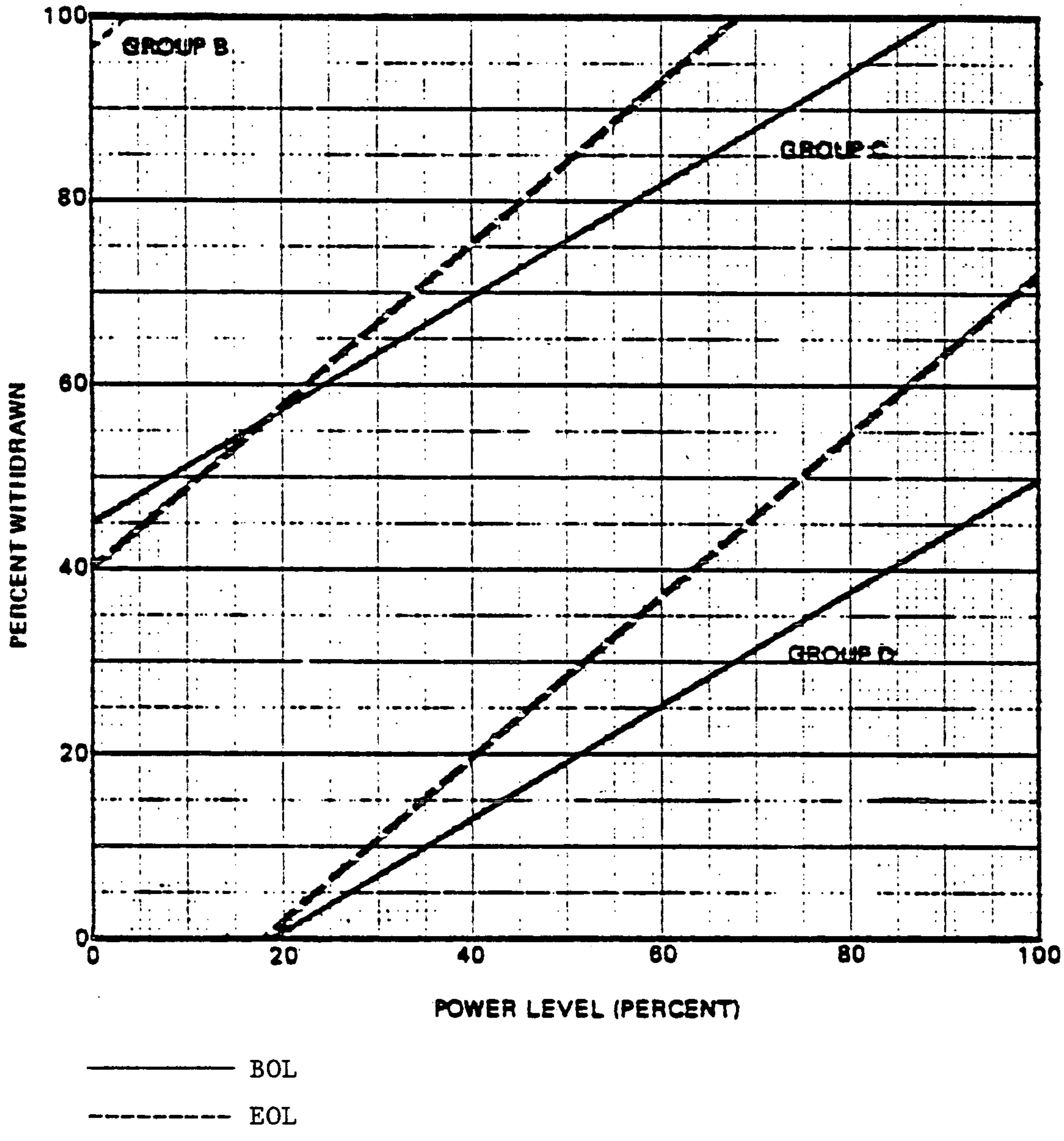

Figure 3.10–1 CONTROL GROUP INSERTION LIMITS FOR THREE LOOP OPERATION

5.3 REACTOR

5.3.1 Reactor Core

5.3.1.1 The reactor core contains approximately 71 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy - 4 tubing to form fuel rods. The fuel rods in the core are all pre-pressurized fuel rods. The reactor core is made up of 157 fuel assemblies. Each fuel assembly contains 204 fuel rods.[1]

5.3.1.2 Deleted

5.3.1.3 Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will be no more than 3.5 weight percent of U-235.

Deleted

5.3.1.4 Deleted

c.	Deleted		
d.	Inservice Inspection	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.	Fire Protection System	3.14	As specified by limiting condition for operation.

3.10.1.5 Except for physics tests, if a full-length control rod is more than 15 inches out of alignment with its bank, then within two hours:

a. Correct the situation, or

b. Determine by measurement the hot channel factors and apply Specification 3.10.2.1, or

c. Limit power to 70 percent of rated power for three-loop operation.

3.10.1.6 Insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-2 must be maintained except for the low power physics test to measure control rod worth and shutdown margin. For this test the reactor may be critical with all but one full length control rod inserted.

3.10.2 Power Distribution Limits

3.10.2.1 At all times except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

$$F_Q\ (Z) \leq (2.20/P) \times K(Z) \text{ for } P > .5$$
$$F_Q\ (Z) < (4.40) \times K(Z) \text{ for } P \leq .5$$
$$F^N_{\Delta H} < 1.55\ (1 + 0.2(1-P))$$

where P is the fraction of licensed power at which the core is operating, K(Z) is based on the function given in Figure 3.10-3, and Z is the core height location of F_Q.

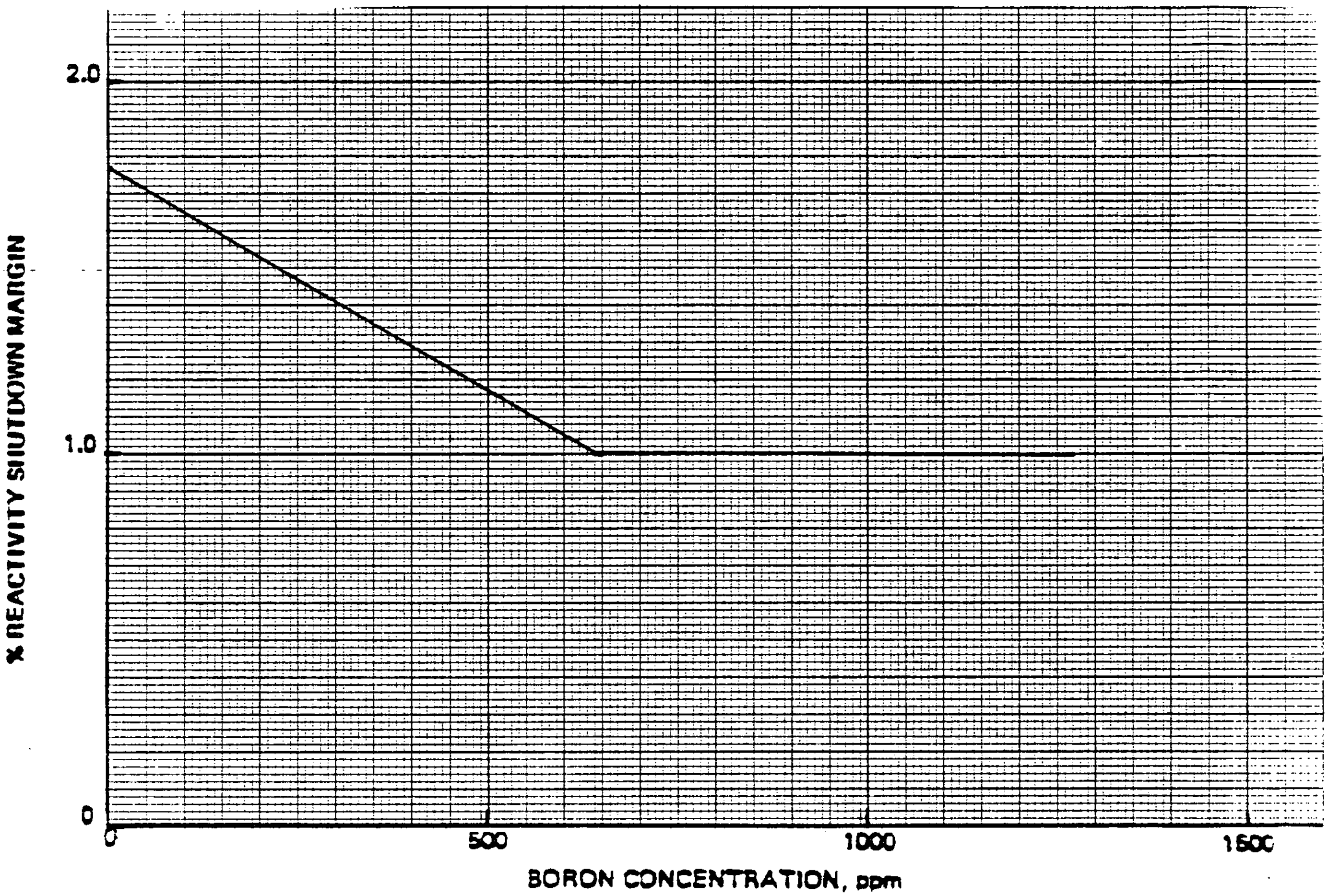

Figure 3.10—2 SHUTDOWN MARGIN VS BORON CONCENTRATION