2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, and flow and pressurizer level.

Objective

To provide for automatic protective action in the event that the principal process variables approach a safety limit.

Specification

2.3.1 Protective instrumentation settings for reactor trip shall be as follows:

2.3.1.1 Startup protection

a. High flux, power range (low set point) $\leq$25% of rated power.

2.3.1.2 Core protection

a. High flux, power range (high set point) $\leq$109% of rated power.*

b. High pressurizer pressure $\leq$2385 psig.

c. Low pressurizer pressure $\geq$1835 psig.

*This setting limit shall be less than or equal to 92% of rated power when operating under the reduced temperature conditions described in the November 11, 1981 license submittal.

d. Overtemperature ΔT

$\leq \Delta T_o \{K_1 - K_2 (T - 575.4) + K_3 (P - 2235) - f(\Delta I)\}$ *

where:

ΔT_o = Indicated ΔT at rated power, °F

T = Average temperature, °F

P = Pressurizer pressure, psig

K_1 = 1.1619

K_2 = 0.01035

K_3 = 0.0007978

and $f(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $(q_t - q_b)$ within +12% and -17% where q_t and q_b are percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power, $f(\Delta I) = 0$. For every 2.4% below rated power level, the permissible positive flux difference range is extended by +1 percent. For every 2.4% below rated power level, the permissible negative flux difference range is extended by -1 percent.

(2) For each percent that the magnitude of $(q_t - q_b)$ exceeds +12% in a positive direction, the ΔT trip setpoint shall be automatically reduced by 2.4% of the value of ΔT at rated power.

(3) For each percent that the magnitude of $(q_t - q_b)$ exceeds -17%, the ΔT trip setpoint shall be automatically reduced by 2.4% of the value of ΔT at rated power.

*When operating under the reduced temperature conditions described in the November 11, 1981 license submittal, replace the number 575.4 with 537.9 in the overtemperature ΔT calculation.

e. Overpower Δ T

$$\leq \Delta T_o \left[K_4 - K_5 \frac{dT}{dt} - K_6 (T - T') - f(\Delta I) \right]$$

where:

ΔT_o = Indicated ΔT at rated power, ^0F

T = Average temperature, °F

T' = Indicated average temperature at nominal conditions and rated power, °F*

K_4 = 1.07

$K_5 = \begin{cases} 0 \text{ for decreasing average temperature} \\ 0.2 \text{ seconds per °F for increasing average temperature} \end{cases}$

K_6 = 0.002235 for $T \geq T'$; K_6 = 0 for $T < T'$
$f(\Delta T)$= as defined in d. above.

f. Low reactor coolant loop flow $\geq 90\%$ of normal indicated flow

g. Low reactor coolant pump frequency ≥ 57.5 Hz

h. Under voltage $\geq 70\%$ of normal voltage.

2.3.1.3 Other Reactor Trips

a. High pressurizer water level $\leq 92\%$ of span

b. Low-low steam generator water level $\geq 14\%$ of narrow range instrument span.

2.3.2 Protective instrumentation settings for reactor trip interlocks shall be as follows:

2.3.2.1 The low pressurizer pressure trip, high pressurizer level trip, and the low reactor coolant flow trip (for two or more loops) may be bypassed below 10% of rated power.

2.3.2.2 The single-loop-loss-of-flow trip may be bypassed below 45% of rated power.

*The value of T' for nominal conditions and rated power is 575.4°F. When operating under the reduced temperature conditions described in the November 11, 1981 license submittal, replace the number 575.4 with 537.9 in the overpower ΔT calculation.

TABLE 3.5-1

ENGINEERED SAFETY FEATURE SYSTEM INITIATION INSTRUMENT SETTING LIMITS

NO.	FUNCTIONAL UNIT	CHANNEL ACTION	SETTING LIMIT
1.	High Containment Pressure (HI Level)	Safety Injection*	$\leq$ 5 psig
2.	High Containment Pressure (HI-HI Level)	a. Containment Spray** b. Steam Line Isolation	$\leq$ 25 psig
3.	Pressurizer Low Pressure	Safety Injection*	$\geq$ 1700 psig
4.	High Differential Pressure Between any Steam Line and the Steam Line Header	Safety Injection*	$\leq$ 150 psi
5.	High Steam Flow in 2/3 Steam Lines***	a. Safety Injection* b. Steam Line Isolation	$\leq$ 40% (at zero load) of full steam flow $\leq$ 40% (at 20% load) of full steam flow $\leq$ 110% (at full load) o full steam flow
	Coincident with Low T_{avg} or Low Steam Line Pressure		$\geq$ 541°F T_{avg} **** $\geq$ 600 psig steam line pressure ****
6.	Loss of Power		
	a. 480V Emerg. Bus Undervoltage (Loss of Voltage) Time Delay	Trip Normal Supply Breaker	328 Volts $\pm$ 1 Volt .75 $\pm$ .25 sec.

TABLE 3.5-1 (Continued)

ENGINEERED SAFETY FEATURE SYSTEM INITIATION INSTRUMENT SETTING LIMITS

NO.	FUNCTIONAL UNIT	CHANNEL ACTION	SETTING LIMIT
6. (Cont'd)	b. 480V Emerg. Bus Undervoltage (Degraded Voltage) Time Delay	Trip Normal Supply Breaker	412 Volts $\pm$ 1 Volt 10.0 Second Delay $\pm$ 0.5 sec.
7.	Containment Radioactivity High	Ventilation Isolation	$\leq$ 2 X Reading at the Time the Alarm is Set with Known Plant Conditions

* Initiates also containment isolation (Phase A), feedwater line isolation and starting of all containment fans.
** Initiates also containment isolation (Phase B).
*** Derived from equivalent ΔP measurements.
**** These setting limits shall be greater than or equal to 524°F and 450 psig when operating under the reduced temperature conditions described in the November $\boxed{11,}$ 1981 license submittal.

TABLE 3.5-3 (Continued)

INSTRUMENTATION OPERATING CONDITIONS FOR ENGINEERED SAFETY FEATURES

NO.	FUNCTIONAL UNIT	1 MINIMUM CHANNELS OPERABLE	2 MINIMUM DEGREE OF REDUNDANCY	3 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 CANNOT BE MET
2.	CONTAINMENT SPRAY			
	a. Manual*	2	0**	Cold Shutdown
	b. High Containment Pressure* (Hi-Hi Level)	2/set	1/set	Cold Shutdown
3.	LOSS OF POWER			
	a. 480V Emerg. Bus Undervoltage (Loss of Voltage)	2/bus (a)	1/bus(b)	Maintain Hot Shutdown
	b. 480V Emerg. Bus Undervoltage (Degraded Voltage)	2/bus	1/bus	Maintain Hot Shutdown(c)

* Also initiates a Phase B containment isolation.
** Must actuate two switches simultaneously.
*** When primary pressure is less than 2000 psig, channels may be blocked.
**** When primary temperature is less than 547°F, channels may be blocked. (d)
***** In this case the 2/3 high steam flow is already in the trip mode.
(a) During testing and maintenance of one channel, may be reduced to 1/bus.
(b) During testing and maintenence of one channel, may be reduced to 0/bus.
(c) The reactor may remain critical below the power operating conditions with this feature inhibited for the purpose of starting reactor coolant pumps.
(d) When operating under the reduced temperature conditions described in the November 11, 1981 license submittal, the channels may be blocked when primary temperature is less than 530°F.

3.I The following operating license condition is effective from the time the H. B. Robinson Unit 2 returns to power operation subsequent to the August 1981 steam generator inspection and to remain in effect until the next refueling outage:

a. A primary to secondary pressure test at approximately 1825 psi differential shall be performed prior to operation at power levels such that estimated corrosion is equivalent to that of 24 effective full power days operation as shown in figure 4.3.3 in Attachment B of CP&L's letter of August 27, 1981. A period of seven additional calendar days is permitted for flexibility for scheduling the necessary test. This test shall be repeated after each interval of operation such that the estimated corrosion is equivalent to that of 24 effective full power corrosion equivalent days operation until the end of cycle 8 operation.

b. At the end of core life of the present cycle, an eddy current examination shall be performed. The scope of this inspection will be submitted to the NRC for approval at least 45 calendar days prior to this end of core life inspection.

c. During the remainder of the cycle 8 operations, the following steam generator tube leakage criteria shall be in effect. Specifically, the plant shall be shut down if the verified primary to secondary leakage in one steam generator exceeds any of the following:

1. A sudden increase of 0.1 gallon per minute (gpm) if the total leakage rate in that steam generator exceeds 0.2 gpm.

2. If the leakage rate in that steam generator exceeds 0.2 gpm and an upward trend in leakage rate in excess of 0.02 gpm per day is verified. This trend will be established using at least five valid consecutive daily samples.

d. Should the plant be required to shut down to repair a steam generator tube leak as indicated in item (c) above, an inspection shall be performed as mutually agreed upon by the NRC Staff and CP&L.

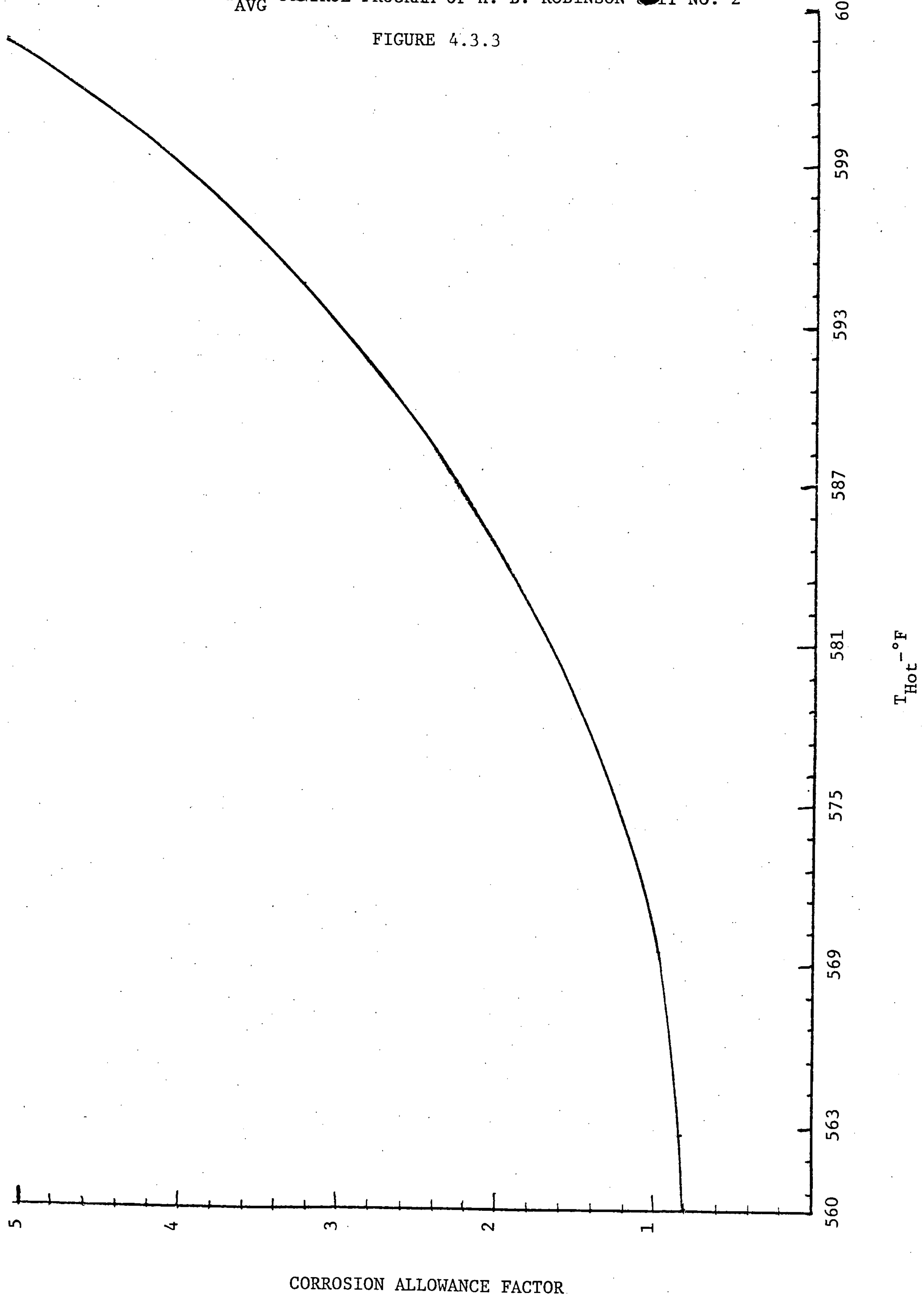
e. The NRC Staff shall be provided with a summary of the results of the eddy current examination performed under item (b) above.

4. ~~CP&L shall observe such standards and requirements for the protection of the environment as are validly imposed pursuant to authority established under Federal and State law and as are determined by the Commission to be applicable to the facility covered by this operating license. This condition does not apply to (a) radiological effects since such effects are dealt with in other provisions of this operating license or (b) matters of water quality covered by section 21(b) of the Federal Water Pollution Control Act.~~

Deleted Amendment #2, dtd 11-15-72

CORROSION ALLOWANCE FACTOR AS A FUNCTION OF POWER FOR PRESENT T_{AVG} CONTROL PROGRAM OF H. B. ROBINSON UNIT NO. 2

FIGURE 4.3.3



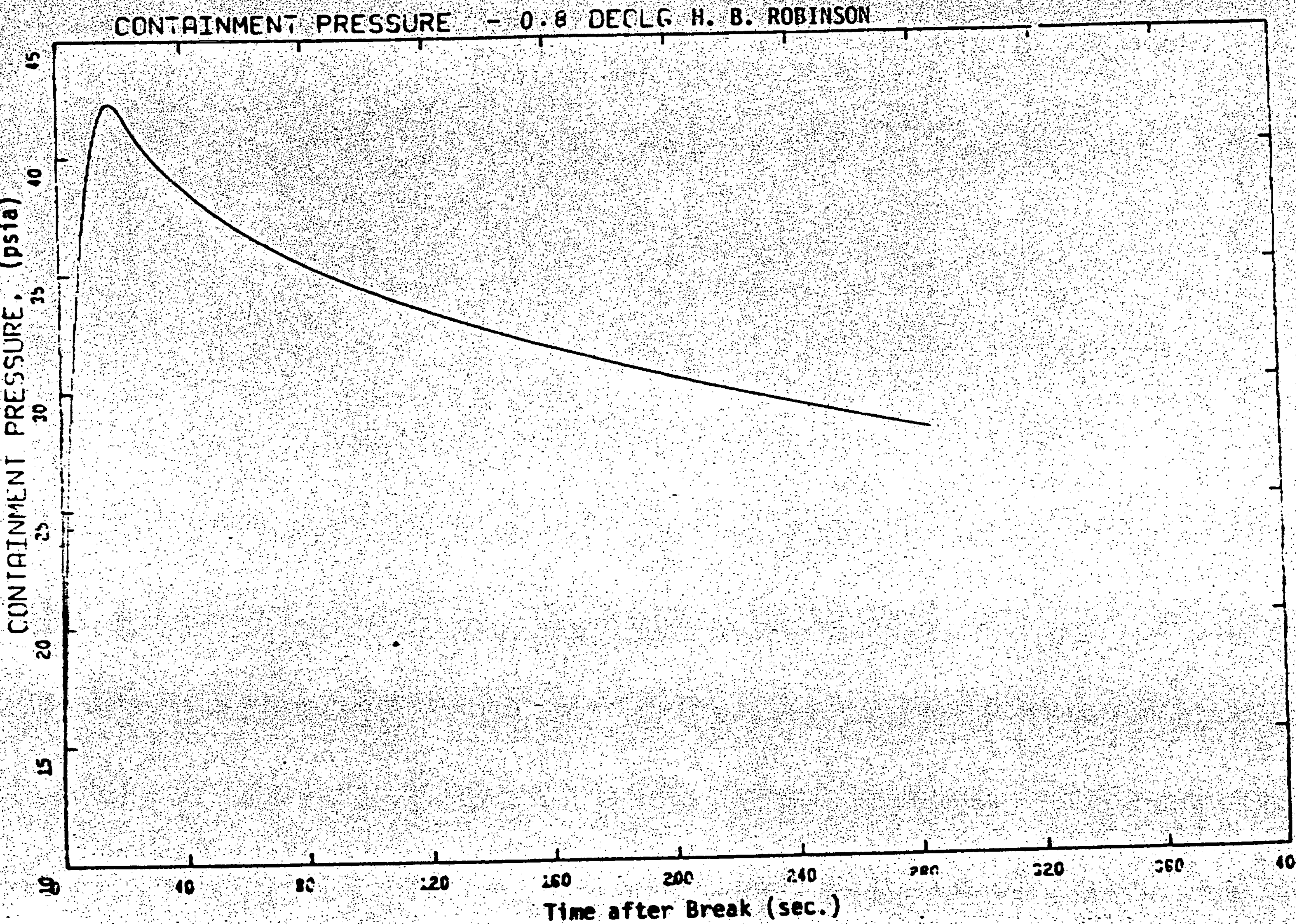
BASIS FOR ESTIMATES OF PEAK CLADDING TEMPERATURE EFFECTS FOR H. B. ROBINSON OPERATING AT REDUCED TEMPERATURE

In the ENC Qualitative Evaluation of the effects of reduced temperature and power operation for H. B. Robinson, a bounding peak cladding temperature (PCT) increase of 200°F was given due to the reduced temperature effect. This number was given based on unreported results obtained for a similar Westinghouse two-loop PWR. The reason for the PCT increase is the reduced primary system temperature or enthalpy which reduces the LOCA energy release to the containment, which in turn reduces containment pressure, reducing reflood rates, and thereby increases calculated PCT. ENC believes the two-loop PWR results, which were performed at an assumed 100% power, are suitable to bound the reduced primary temperature effects on H. B. Robinson because of the similarity of calculated results for both reactors, particularly with regard to containment pressure and reflood rate. The attached calculated results for containment pressure and reflood rate for the two reactors demonstrate this similarity.

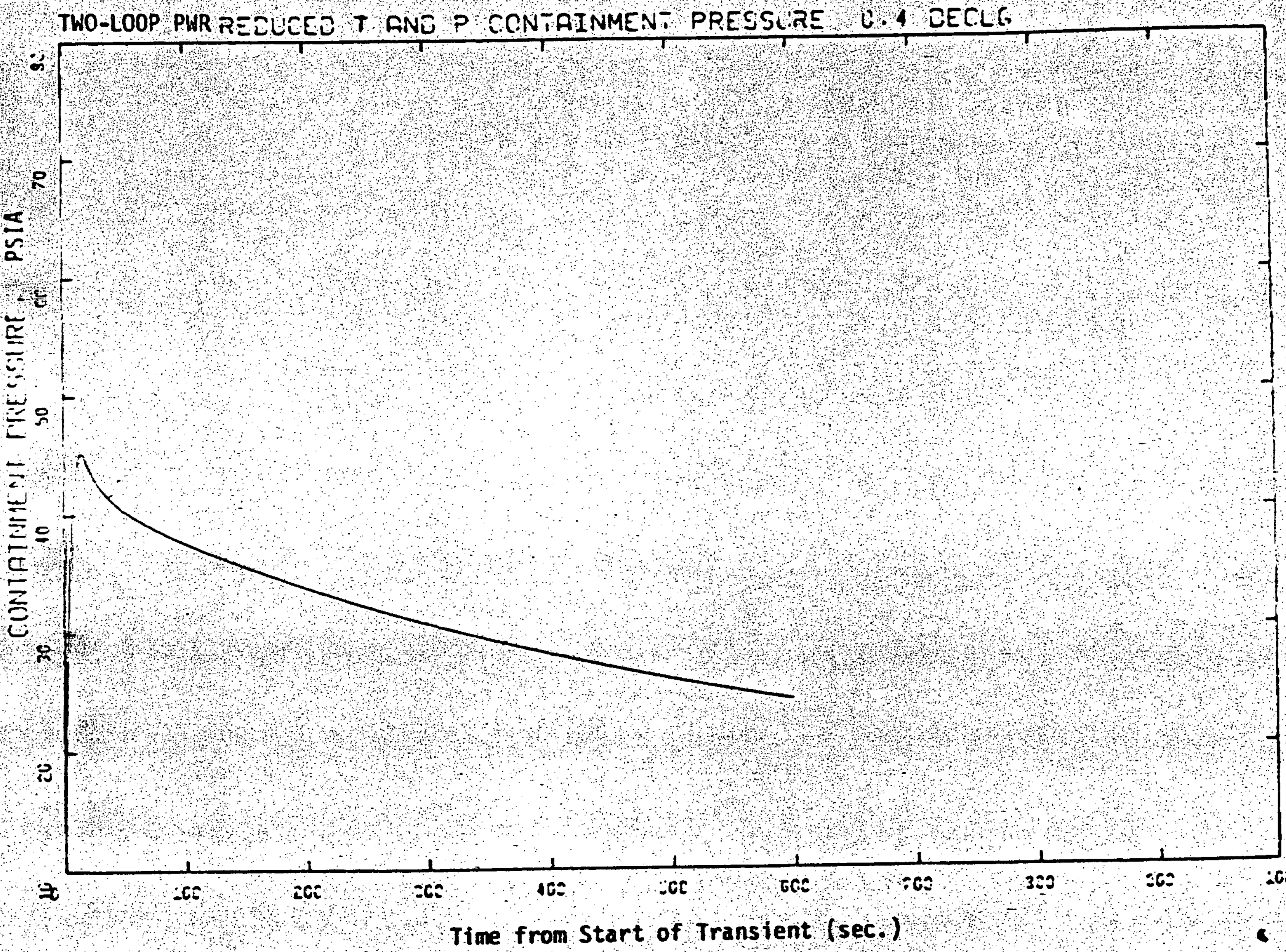
The 200°F PCT increase for the two-loop PWR is the difference between the low temperature results and the current licensing analysis. About 45°F of this increase is associated with incorporation of the ENC WREM-IIA model. The actual 155°F PCT difference between the two cases using the same analytical models was calculated to occur on a ruptured node with enhanced metal-water reaction.

The PCT increase at the node corresponding to the non-ruptured PCT location is about 80°F. A similar PCT increase (∿80°F) would be expected for H. B. Robinson, which due to reduced power, will not become rupture node limited. The more conservative 200°F difference was chosen to assure that any system dissimilarities are bounded.

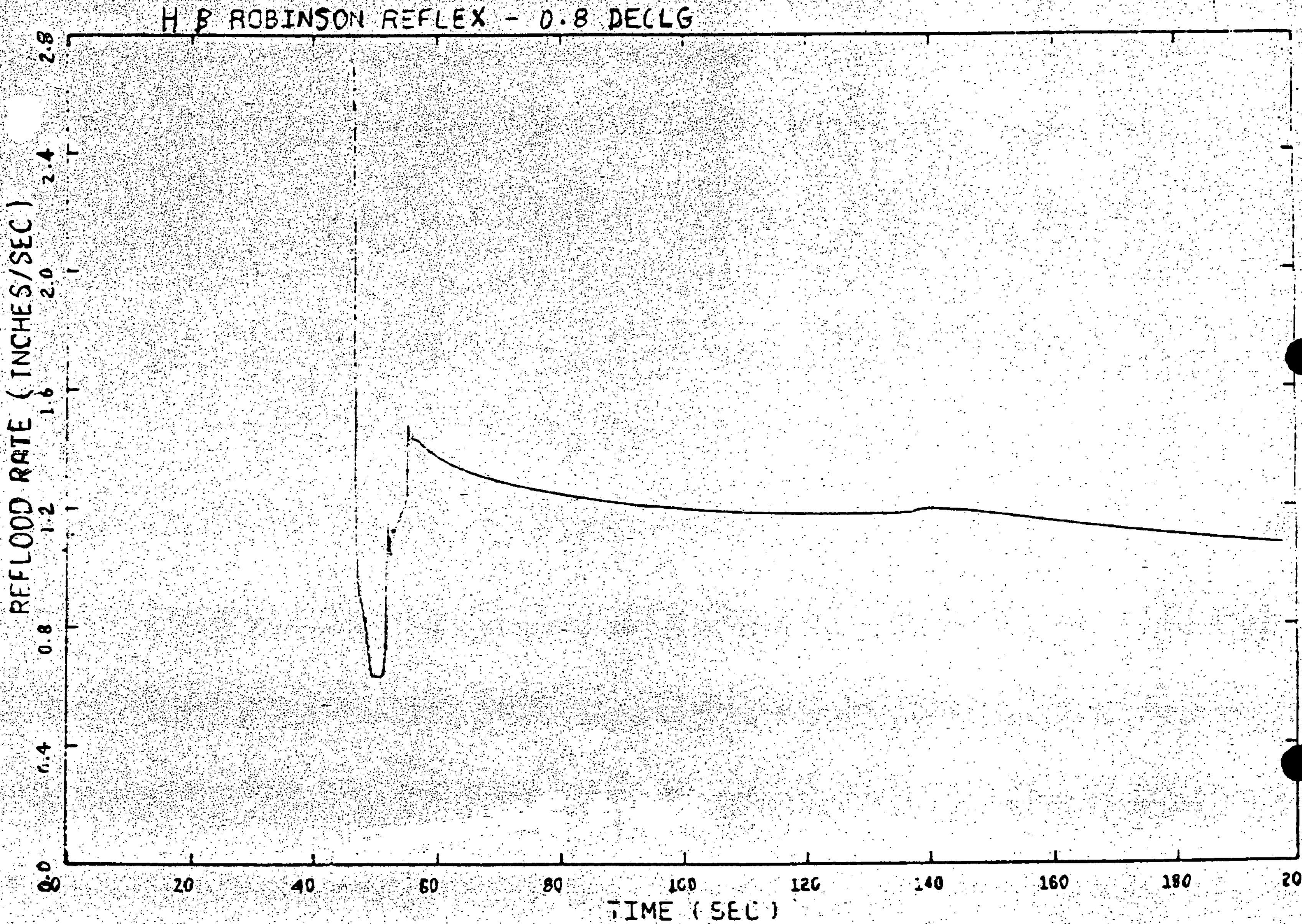
For H. B. Robinson the 300°F PCT decrease due to power reduction stems from the assumed 15% reduction in total core power while maintaining the 2.2 peaking factor which assures a 15% reduction in allowed linear heat generation rate. ENC has performed several LOCA-ECCS calculations for the H. B. Robinson reactor at various assumed power peakings and steam generator tube plugging values. The variation of PCT with local power (20-25°F per percent local power) was obtained directly from these H. B. Robinson results. The 300°F decrease is a minimum number resulting from the 15% power reduction and the minimum value of the range. On an average basis, the benefits of reduced power are four times the best estimate of reduced temperature effects and a significant reduction in PCT (1922°F based on latest ENC analysis) is expected for H. B. Robinson operating at the reduced power and temperature.



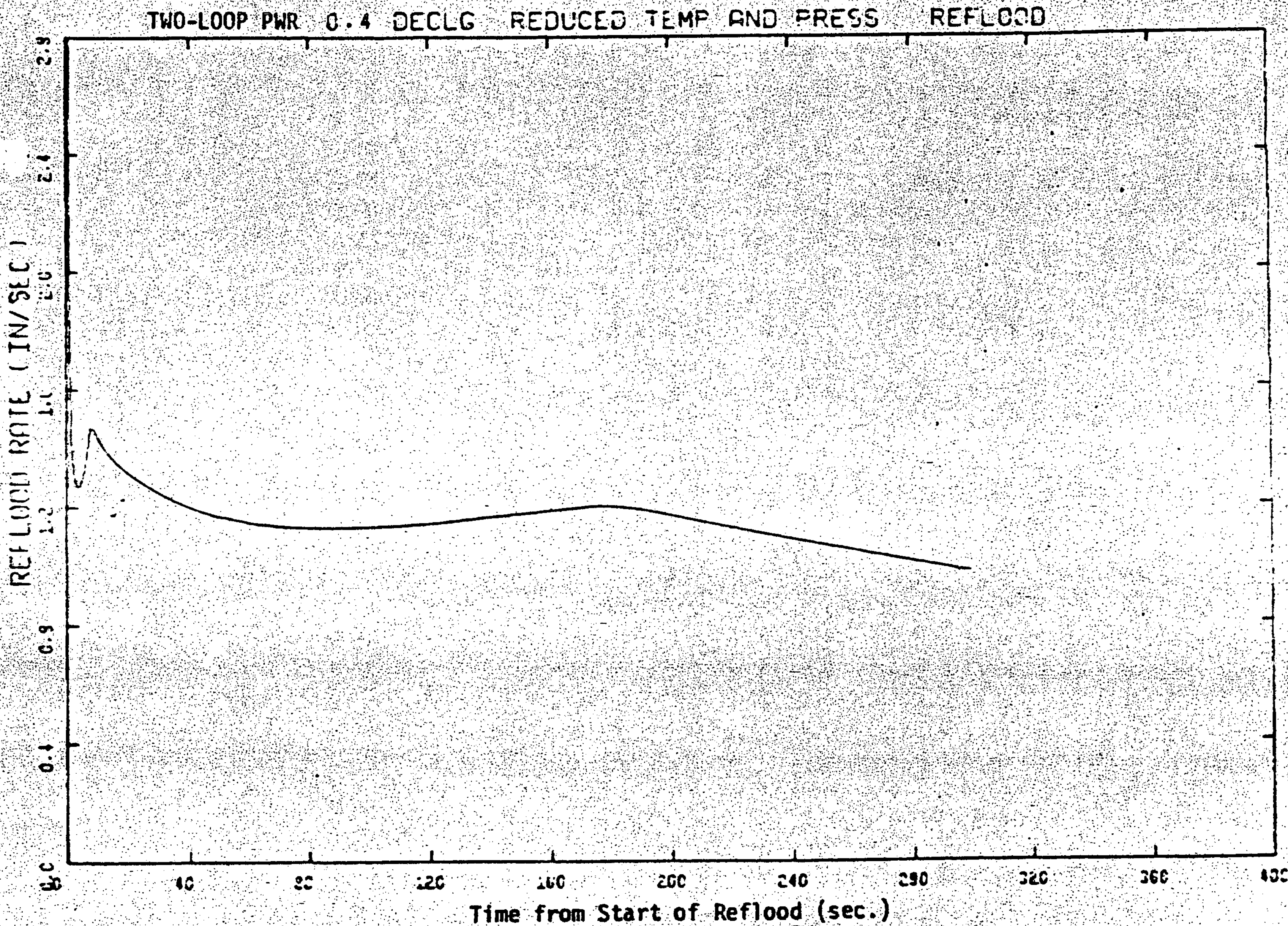
H.B. Robinson Containment Pressure - 0.8 DECLG



Two-Loop PWR Containment Backpressure, 0.4 DECLG Break



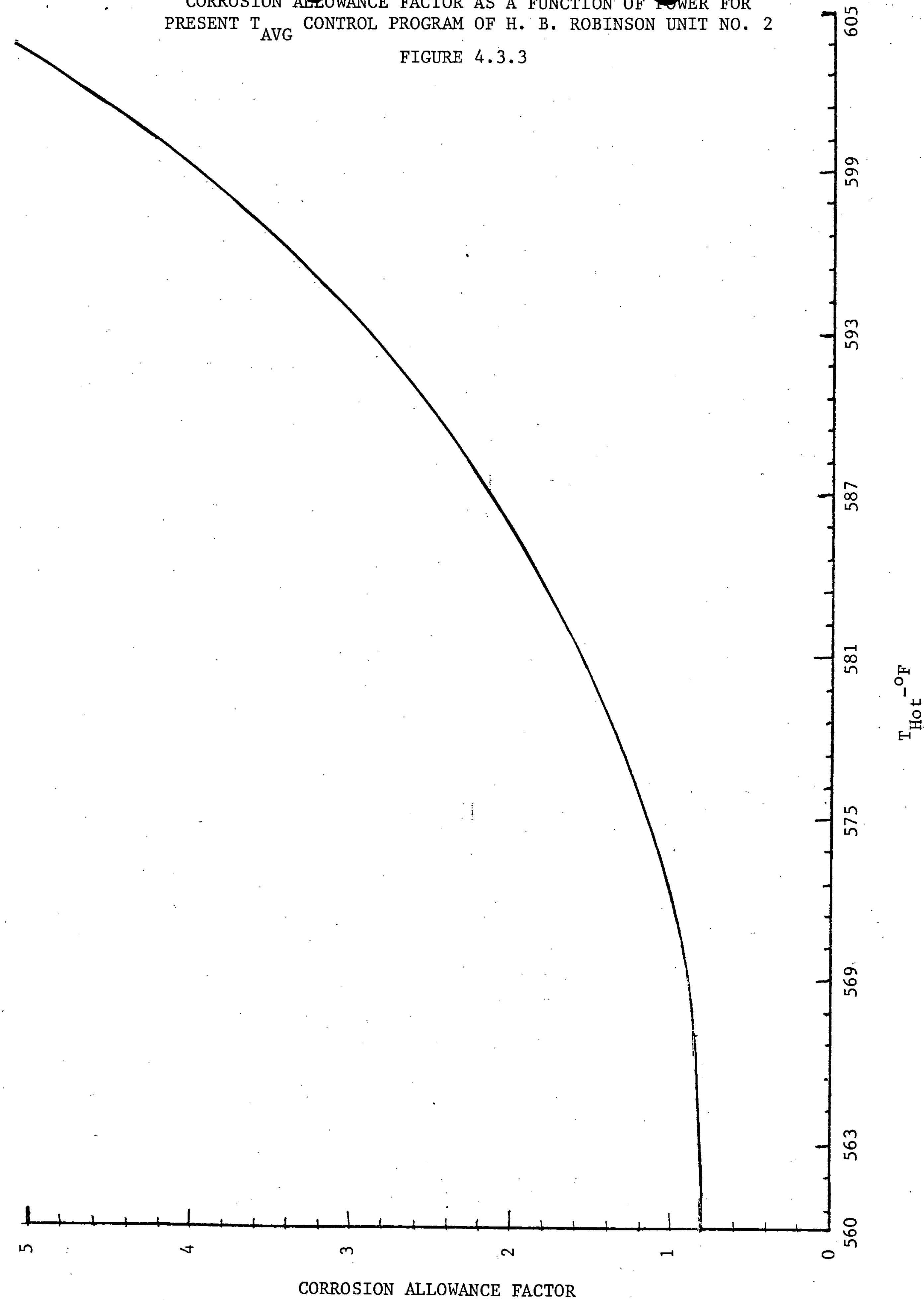
H. B. Robinson Reflood Rate for 0.8 DECLG



Two-Loop PWR Reflood Core Flooding Rate, 0.4 DECLG Break

CORROSION ALLOWANCE FACTOR AS A FUNCTION OF POWER FOR
PRESENT T_{AVG} CONTROL PROGRAM OF H. B. ROBINSON UNIT NO. 2

FIGURE 4.3.3



ENCLOSURE 3

SUMMARY

In the rod withdrawal transient a reduction in MDNBR from initial conditions is calculated to occur. This reduction occurs because core power increases until reactor trip conditions are reached and because of peaking augmentation in the vicinity of the withdrawn rod. Reactor trips for the present reduced temperature conditions have been reset so as to prevent higher fractional core power change until trip for the new conditions. Based on neutronics calculations, the power peaking augmentation at the withdrawn rod due to reduction of moderator temperature is increased by only about 3%. This increase in peaking augmentation is more than offset by the large reduction in trip setpoints. Thus rod withdrawal transients will be substantially less limiting for the revised operating conditions.

QUANTIFICATION OF STEADY STATE THERMAL MARGIN

The steady state thermal margin calculations employed the conditions given in Table 1 and allowed for 12% overpower, 4.5% bypass flow, and measurement uncertainties as noted. The calculated MDNBR is 2.83, indicating very substantial thermal margin at the reduced power, temperature, and flow conditions. The reference calculation(1) employed rated full load (1.12 X 2300 MWt) operating conditions and yielded an MDNBR of 1.87. The increased margin at the reduced power conditions results from a 15% reduction in clad surface heat flux and a 25% increase in inlet subcooling relative to the rated full load condition. The margin to burnout at the reduced power condition is about twice that calculated for rated full load conditions.

EFFECT OF REDUCED MODERATOR TEMPERATURE ON HOT CHANNEL PEAKING DURING THE ROD WITHDRAWAL TRANSIENT

With respect to normal primary coolant temperature, reduced primary coolant temperature should have an insignificant impact on power peaking magnitude in the uncontrolled rod withdrawal at power transient. Because the change in coolant temperature is global (corewise) not local, and power density calculations are relative, the power peaking magnitude should remain essentially the same at different primary coolant temperatures.

Nevertheless, calculations have been performed to quantify the effect of reduced moderator temperature on hot channel peaking during the rod withdrawal transient. These calculations indicate that the reduced moderator temperature under the new temperature program results in about a 3 percent increase in peaking augmentation caused by an inadvertent control rod withdrawal. This effect is judged to be insignificant in itself and is certainly more than offset by the 15% reduction in the high nuclear flux and overtemperature ΔT reactor trip setpoints which will be in effect under the reduced operating temperature conditions.

The moderator temperature coefficient for the remainder of Cycle 8 is calculated to be between -10.pcm/°F and -32pcm/°F, well below the Technical Specification limit of +2 pcm/°F.

Of all transients involving the loss of secondary side heat removal capability the Loss of Normal Feedwater transient is the most significant. The key assumptions in the FSAR analysis of this transient are: (1) Immediate Reactor Trip, (2) Auxiliary Feedwater availability, (3) Reactor Power at 102%. The new reduced temperature program will not affect the first two assumptions, but will result in a lower Reactor Power. Therefore, the consequences of a Loss of Normal Feedwater transient under the reduced temperature program will be more conservative than analyzed in the FSAR.

Table 1 Conditions used in the Steady State Thermal Margin Calculation

CORE CONDITIONS	NOMINAL	MARGIN ANALYSIS*
Power Level, MWt	1955	2233*
Primary System Pressure, psia	2250	2220
Coolant Flow Rate, mlb/hr	98.0	93.58
Coolant Inlet Temperature, °F	510	514***
T_{avg} SCHEDULE		
T_{avg} at no load, °F		530
Linear Gain, °F/% power		0.0935
POWER PEAKING FACTORS		
$F^N_{\Delta H}$		1.55
Axial Peaking Factor		1.55
Engineering Heat Flux Factor		1.03
Total Peaking Factor		2.47
Fraction of Power Deposited in the Fuel		.974
RESULT		
MDNBR, at overpower, for reduced power and temperature		2.83
MDNBR, at overpower, for rated conditions (reference 1)		1.87

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* Steady state overpower thermal margin analysis: 12% overpower allowance, and 2% power measurement uncertainty: 1955 x 1.02 x 1.12 = 2233.

** Analysis value assumes 4.5% bypass flow.

*** Analysis allows for 4°F deadband and measurement uncertainty

REFERENCE:

(1) XN-75-25, Volume 1, "H. B. Robinson Fuel Design Report Volume 1 - Mechanical and Thermal Hydraulic Design for Cycle 4", June 1975.