REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8206280499 DOC.DATE: 82/06/24 NOTARIZED: NO DOCKET #
FACIL:50-261 H. B. Robinson Plant, Unit 2, Carolina Power and Ligh 05000261
AUTH.NAME AUTHOR AFFILIATION
ZIMMERMAN,S.R. Carolina Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
VARGA,S.A. Operating Reactors Branch 1

SUBJECT: Formally submits addl info re Cycle 9 operation Tech Spec Table 3.5.1 adequately describes setpoint for full load.Clad swelling & rupture,rod injection analysis & updated steam generator tube rupture analysis addressed.

DISTRIBUTION CODE: A001S COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 16
TITLE: General Distribution for after Issuance of Operating License

NOTES:

	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL
	ORB #1 BC	01	7	7				
INTERNAL:	ELD/HDS1		1	0	NRR/DHFS DEPY08		1	1
	NRR/DL DIR		1	1	NRR/DL/ORAB		1	0
	NRR/DSI/RAB		1	1	~~REG FILE~~	04	1	1
	RGN2		1	1				
EXTERNAL:	ACRS	09	10	10	LPDR	03	1	1
	NRC PDR	02	1	1	NSIC	05	1	1
	NTIS		1	1				

TOTAL NUMBER OF COPIES REQUIRED: LTTR 28 ENCL 26

Carolina Power & Light Company

JUN 24 1982

Office of Nuclear Reactor Regulation
ATTN: Mr. Steven A. Varga, 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
ADDITIONAL INFORMATION - CYCLE 9 OPERATION

Dear Mr. Varga:

SUMMARY

During the NRC review of Carolina Power & Light Company's (CP&L) submittals for H. B. Robinson, Unit No. 2 (HBR2) Cycle 9 operation, several questions were raised by the NRC staff. The majority of the questions, which were received informally, were responded to either verbally via conference calls or via telecommunications. The purpose of this letter is to formally submit the information previously provided via telecopy. The respective staff questions and responses are addressed in attachments to this letter.

The NRC Staff requested that one question, responded to verbally, be addressed in this submittal. This question was regarding the HBR Technical Specifications (TS) Table 3.5-1, item 5, "High Steam Flow in 2/3 Steam Lines." The Setting Limit at 20% load is $\leq$40% of full steam flow, and at full load is $\leq$110% of full steam flow. The question was:

"... does ΔP set point increase linearly with power from 20% to 100%."

Carolina Power & Light Company informed the NRC Staff that it did increase linearly. The NRC Staff suggested that we submit a TS change request to modify the TS to state that between 20% load and full load the ΔP setpoint

A001

increases from 40% to 110% linearly. Carolina Power & Light Company believes that it is unnecessary to request this because the existing specification adequately describes the required setpoint.

Should you have any further questions regarding the information, please contact a member of my staff.

Yours very truly,

S. R. Zimmerman
Manager
Licensing & Permits

DCS/lr (024C2T3)
Attachments

cc: Mr. James P. O'Reilly (NRC-RII)
Mr. G. Requa (NRR)

ATTACHMENT 1

Clad Swelling and Rupture

The NRC Staff contacted Exxon Nuclear Company, Inc. (ENC), the HBR2 fuel supplier, and inquired whether ENC had utilized the Staff's clad swelling and rupture model as given in NUREG-0630 to perform the ECCS analysis for HBR2 under the Cycle 9 operating conditions. If it had not been done, the Staff would require that adequate assurance be provided that application of the NUREG-0630 correlation would not result in exceeding the criteria of 10CFR50.46.

The HBR2 ECCS analysis was performed with ENC's approved model, WREM-IIA, which did not include the NUREG-0630 correlation for fuel clad swelling and rupture. The results of this analysis was compared to ENC's new evaluation model, EXEM/PWR, which does incorporate the NUREG-0630 swelling and rupture model and it has been determined that application of the newer model would decrease peak clad temperature relative to the results obtained using WREM-IIA. The following details were telecopied to the NRC staff:

"Exxon Nuclear Company (ENC) has considered the effect of swelling and rupture on the LOCA-ECCS analysis performed for H. B. Robinson Unit 2. Specifically, ENC has compared the results with its currently approved evaluation model (submitted to the NRC by Carolina Power & Light Company in ENC report XN-NF-82-18) and its new evaluation model, EXEM/PWR, which incorporates the NUREG-0630 swelling and rupture model. This latter model is described in ENC reports XN-NF-82-07 and XN-NF-82-20, which have been submitted by ENC for NRC approval.

Calculated reflood rates for the H. B. Robinson Unit 2 reactor exceed 1.0 inch per second during the significant portion of the reflood transient, hence, calculated cooling is based on the FLECHT correlation and is unaffected by the NRC requirement for steam cooling and blockage flow diversion. Therefore the peak cladding temperature calculated at all points other than the rupture location will be insensitive to the swelling and rupture model used.

The only influence of the swelling and rupture model on plants such as H. B. Robinson Unit 2 is at the calculated rupture node where the cladding area is increased due to rupture (both inside and outside) which magnifies the areas available for metal-water reaction and heat transfer. Application of EXEM/PWR to H. B. Robinson Unit 2 would decrease PCT relative to the results obtained using the currently approved evaluation model."

ATTACHMENT 2

The Core Performance Branch requested information with regard to the control rod ejection analysis for Cycle 9. The question was as follows:

> "How do the consequences of a control rod ejection accident at reduced coolant temperatures compare with previous cycle results in regard to fuel damage and RCS pressure surge?"

The following response was telecopied to the NRC staff:

ROD EJECTION ANALYSIS FOR H. B. ROBINSON CYCLE 9

A Control Rod Ejection Accident is defined as the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of a Rod Cluster Control Assembly (RCCA) and drive shaft. The consequence of this mechanical failure is a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel damage.

The rod ejection accident analysis presented in the document XN-75-44 is still applicable to Cycle 9 operation. The ejected rod worths and peaking factors were calculated using the XTG code. No credit was taken for the power flattening effects of Doppler or moderator feedback in the calculations of ejected rod worths or radial peaking factors. Further, the consequences of the ejected rod transients have been evaluated assuming the adverse conditions of full insertion of Bank D at 102% over power and Banks C and D at 0% power. The calculations made for Cycle 9 were two-dimensional (x-y) and the results are shown in the following table. Ejected rod worths and peaking factors, as listed in the table indicate the close resemblance between Cycles 4 through 8. For Cycle 9, the radial peak power (D Bank in) values at BOC is less than that calculated in XN-75-44. However, at EOC conditions, a peak radial power of 1.67 is calculated for Bank D fully inserted prior to ejection; about 8% above the peaking factor assumed in XN-75-44. However, this 8% increase is more than offset by a 15% reduction in reactor operating power for Cycle 9.

	Cycle 8		Cycle 9	
	BOC	EOC	BOC	EOC
Ejected Rod Worths:				
102% Power (pcm)	<380	<390	<380	<390
0% Power (pcm)	<560	<540	<560	<540
Peaking Factors Associated with Ejected Rod Analysis:				
Radial (unrodded)	1.42	1.37	1.56	1.40
Axial (Core Average)	1.20	1.09	1.17	1.08
Radial (D-Bank in)	1.54	1.48	1.55	1.67

ATTACHMENT 3

NRC Staff Request:

Provide an updated steam generator tube rupture analysis.

The Following Response Was Telecopied to the NRC Staff:

"For the steam generator tube rupture transient, the leakage rate of primary fluid to the secondary side is highest during the initial period of the transient when a high primary-to-secondary pressure differential exists. The high pressure differential exists for only a short period of time until the pressurizer empties and primary system pressure drops to the saturation pressure corresponding to the hot leg temperature.

With respect to reduced power - low T_{AVE} operation of H. B. Robinson, two factors have changed in the conservative direction. First the pressurizer level for operation at power is 2.3 feet less so that the pressurizer empties sooner, and hence the time period of high primary-to-secondary pressure differential is less. Secondly, hot leg temperatures are less, so that the primary pressure after the pressurizer empties is less for reduced power - low T_{AVE} operation.

Based on prior analyses and the reduced pressurizer level, it is estimated that about 4000 lbs. less primary-to-secondary leakage will occur during the initial high pressure differential time interval for reduced power - low T_{AVE} operation relative to former conditions.

For reduced power - low T_{AVE} operation hot leg primary coolant temperature is about 40°F less than for prior conditions. The corresponding reduction in saturation pressure is 420 psi. This primary pressure reduction is larger than the 290 psi reduction in secondary pressure for reduced power - low T_{AVE} operation. Thus primary-to-secondary leakage flow after the pressurizer empties will be less for reduced power - low T_{AVE} operation.

Since the level of secondary system contamination is directly dependent on the amount of primary coolant leaked to the secondary, it is concluded that the SGTR accident will be less severe under the reduced power - low T_{AVE} schedule than the event previously analyzed in FSAR Section 14."

ATTACHMENT 4

NRC Staff Request:

The following transients, which were considered in the HBR2 safety analysis (FSAR Chapter 14) were not analyzed by ENC in either their 1975 nor 1982 submittals:

(a) Loss of normal feedwater

(b) Loss of AC power coincident with turbine trip

(c) Chemical and volume control system malfunction

The FSAR does not provide DNBR, primary and secondary pressure plots for these transients. Provide the missing information or justify that the resulting DNBR and pressure values are in all cases bracketed by analyses.

The Following Response Was Telecopied to the NRC Staff:

"The core and system responses to a chemical and volume control system (CVCS) malfunction are essentially the same as the core and system responses to slow rod withdrawal events. Rod withdrawal events have been fully considered in XN-NF-82-18, ECCS and Plant Transient Analyses for H. B. Robinson Unit 2 Reactor Operating at Reduced Primary Temperature, and the results reported therein conservatively bound the outcome of the CVCS malfunction with respect to DNBR, RCS and secondary pressure.

The results of the 3 pump coastdown event presented in XN-NF-82-18 and the results of the recent H. B. Robinson loss of load transient analysis to evaluate the peak pressurization* bound the DNBR and pressure (RCS and Secondary) responses of the reactor to a loss of A.C. power with coincident turbine trip, respectively. Adequate decay heat removal capacity to prevent core uncovery and subsequent fuel damage is demonstrated by the loss of normal feedwater analysis considered below.

The loss of normal feedwater event results in reactor scram well before any potential degradation of primary coolant conditions. Thus, the loss of normal feedwater event is considered non-limiting with respect to DNBR. The loss of load transient results in considerably more severe RCS pressurization than does the loss of normal feedwater. The DNBR and system pressure aspects of the loss of normal feedwater evente are therefore bounded by the DNBR results of the 3 pump coastdown as reported in XN-NF-82-18, and by the pressurization results of the recent loss of load transient analysis noted above.

*See Attachment 5.

The following factors affect thermal expansion of the primary coolant during a loss of normal feedwater event initiated from the maximum power point of the reduced power - low T_{AVE} operating program:

(1) Relative to the case reported in Chapter 14 of the FSAR, there is a 15% decrease in decay heat level due to the reduced core power.

(2) There is a 20% reduction in primary to secondary heat transfer area, due to steam generator tube plugging.

(3) With 20% tube plugging, there is 3% less initial primary coolant.

The FSAR analysis of the loss of normal feedwater event has demonstrated that sufficient auxiliary feedwater capacity exists to remove decay heat when the reactor is operated at 2300 MWt. There is therefore adequate capacity to remove decay heat when the reactor is operated at 1955 MWt. Any decrease in primary-to-secondary heat transfer area which results from steam generator tube plugging will be manifest as an increase in primary-to-secondary ΔT. It is conservatively assumed that the secondary temperature will persist at the saturation temperature corresponding to the steam generator safety valve setpoint at 1100 psia (556°F), so that a 1°F increase in primary-to-secondary ΔT will appear as a 1° increase in primary temperature. Taking into account the 15% reduction in decay heat and the 20% reduction in heat transfer area results in a 6.25% increase in primary-to-secondary ΔT relative to the FSAR analysis. This leads to the prediction of a maximum average primary temperature of 608°F, compared to the 605°F reported in the FSAR analysis from full load - high T_{AVE} conditions.

Based on the 608° average primary temperature, thermal expansion of the initial mass of coolant will lead to the expulsion of 140 cu. ft. of liquid through the pressurizer safety valve. This coolant loss represents a 10% degradation of the margin to core uncovery at ultimate cooldown, leaving approximately 850 cu. ft. of liquid above the core. Fuel damage resulting from core uncovery will thus not occur in the loss of normal feedwater event initiated under the reduced power - low T_{AVE} operating schedule with 20% steam generator tube plugging."

ATTACHMENT 5

NRC Staff Request

"The loss of load transient analyzed in ENC report XN-NF-82-18 results in a peak primary pressure of 2460 psia and a peak secondary pressure of 1051 psia. The set points for the primary safety valves is 2485 psig. The lowest steam generator safety valve setpoint is 1085 psig. Discuss whether your analysis provides sufficient allowance for uncertainties and setpoint tolerances and whether either the primary or secondary safety valves could lift under the most adverse combinations of uncertainties and set point tolerances and what the consequences of safety valve lifting would be."

The Staff Was Sent a Copy of the Following:

H. B. ROBINSON LOSS OF LOAD TRANSIENT

"Loss of load involves plant behavior after a trip of the turbine-generator without a direct reactor trip. The present loss of load analysis has been made to evaluate peak primary system pressurization relative to the vessel integrity limit of 2750 psia. The transient responses for this event were evaluated from 1955 MWt with the most severe assumptions; namely, loss of load at BOC with a positive moderator coefficient (+2.5 pcm/°F) and no automatic reactor control. The steam dump and turbine bypass were not allowed. The feedwater pumps were assumed to trip with the reactor. The steam line power operated relief valves have been neglected. The steam line safety valves were assumed to operate. The pressurizer relief valve and pressurizer spray were assumed inoperable. Table 1 shows the safety valve setpoints and associated uncertainties used in the analysis. For conservatism, a muliplier of 0.8 was applied to the Doppler coefficient of -1.0 pcm/°F.

Figures 1 to 7 show the plant responses following a loss of load from 1955 MWt. After closure of the turbine stop valves, the pressure in both steam generators increases and reaches 1006 psia at 39.81 seconds. The steam line safety valve setpoint is not reached in 40 seconds. The reactor is tripped at 6.52 seconds on high pressure after 1.0 second delay. The pressurizer safety valve opens at 7.36 seconds. The peak primary pressure was 2585 psia, which is well below the maximum allowable value of 2750 psia. The average primary coolant temperature increases by less than 11°F. The minimum DNB ratio during the transient never gets below the initial value."

Table 1 Safety Valve Setpoints and their Uncertainties

Steam Line Safety Valve A setpoint	1111 psia (1100 + 1%)
" B "	1137 psia (1125 + 1%)
" C "	1152 psia (1140 + 1%)
" D "	1167 psia (1155 + 1%)
Pressurizer Safety Valve	2575 psia (maximum allowed in Technical Specifications)
High Pressurizer Pressure Setpoint	2430 psia (2400 + 30)

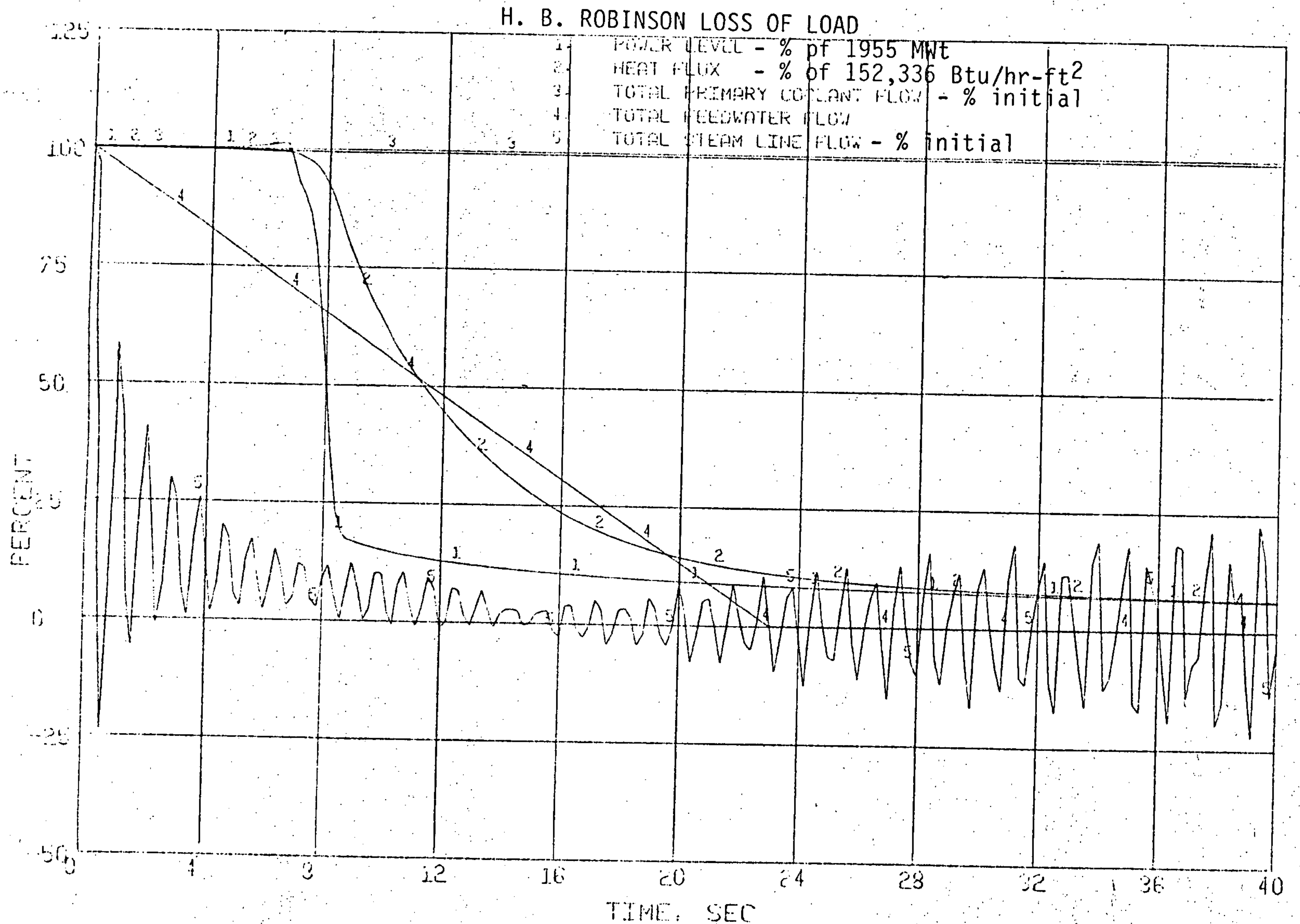

Figure 1 Power, Heat Flux and System Flows for Loss of Load

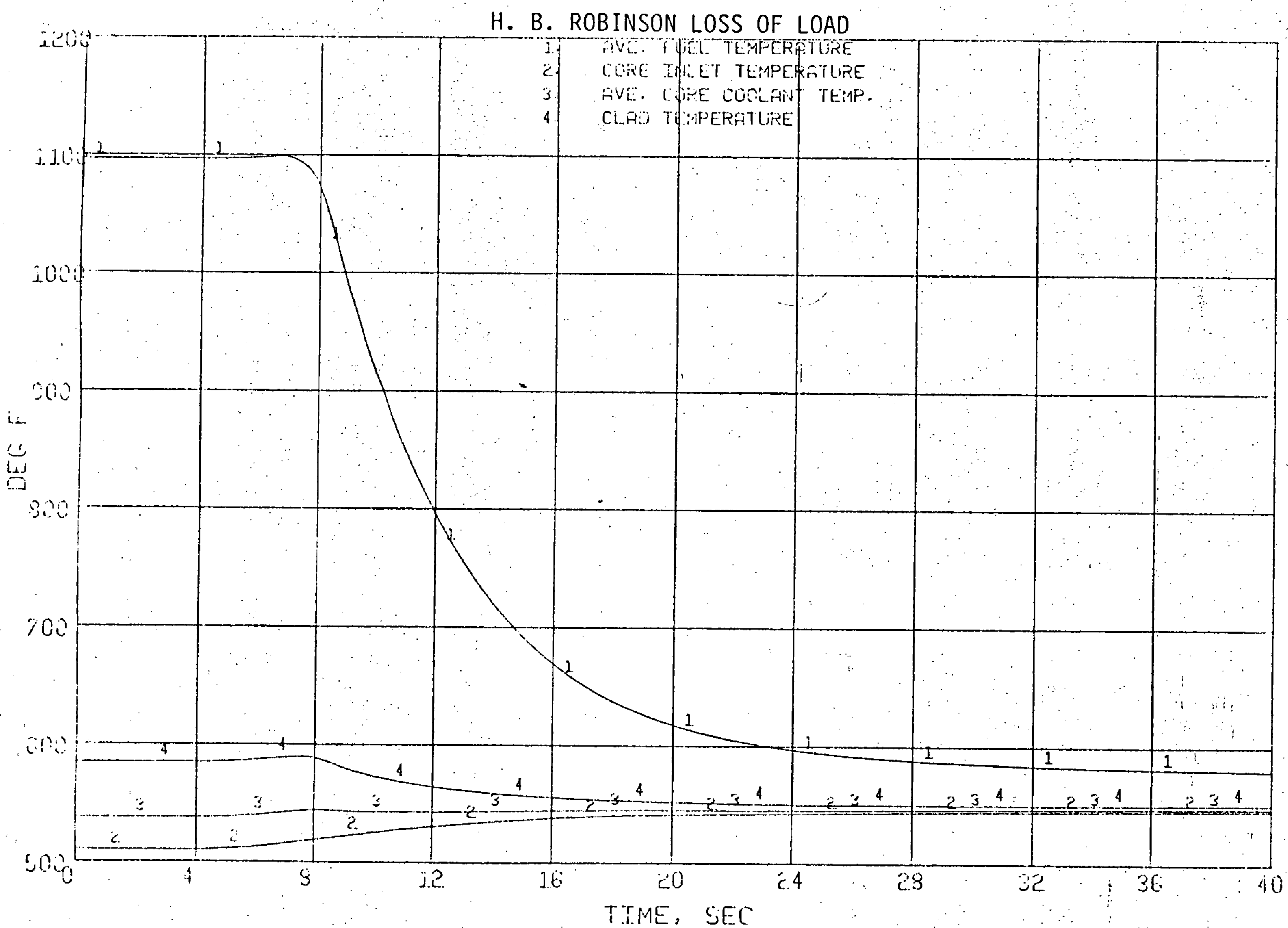

Figure 2 Core Temperature Responses for Loss of Load

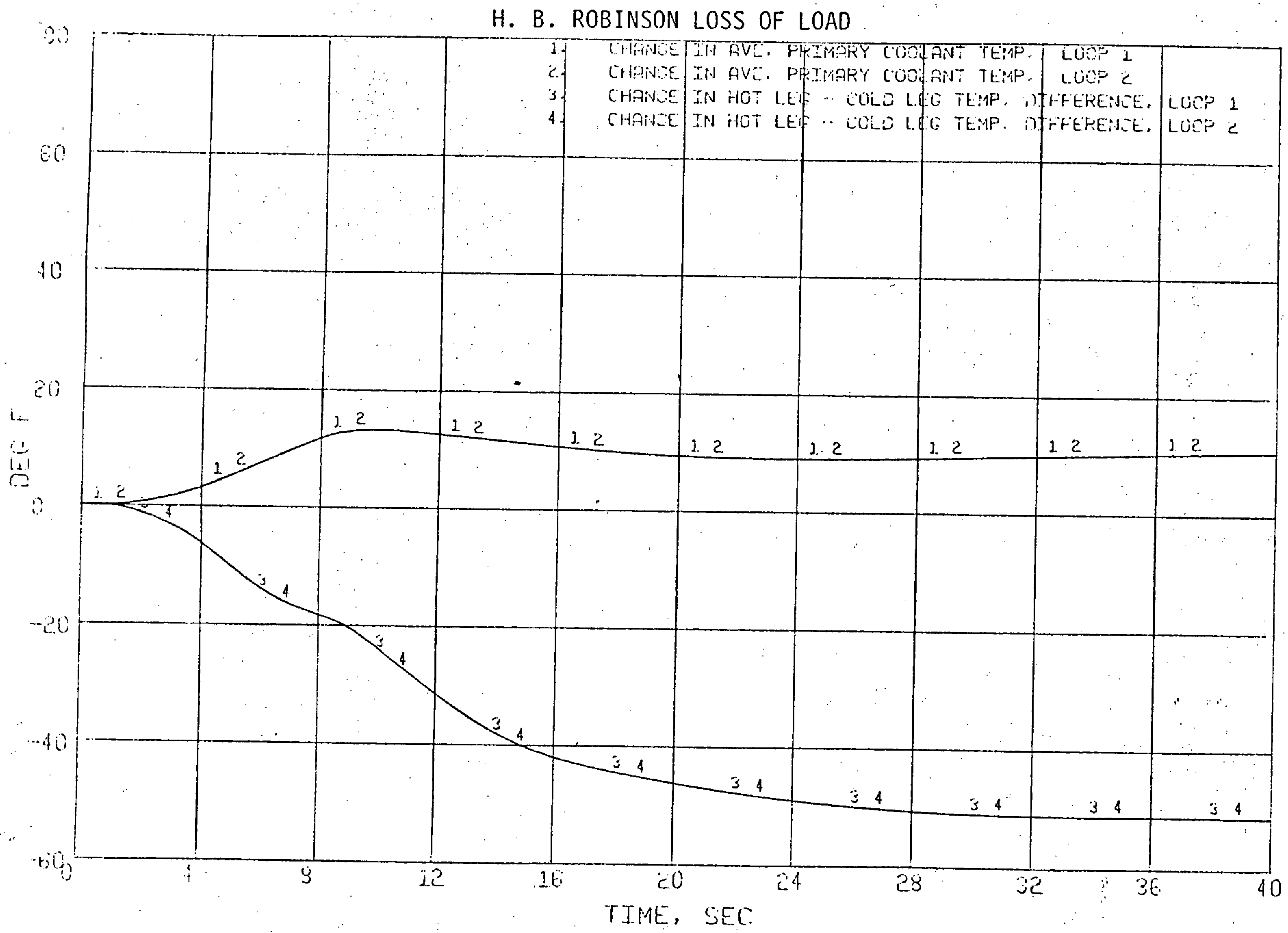

Figure 3 Primary Loop Coolant Temperature Changes for Loss of Load

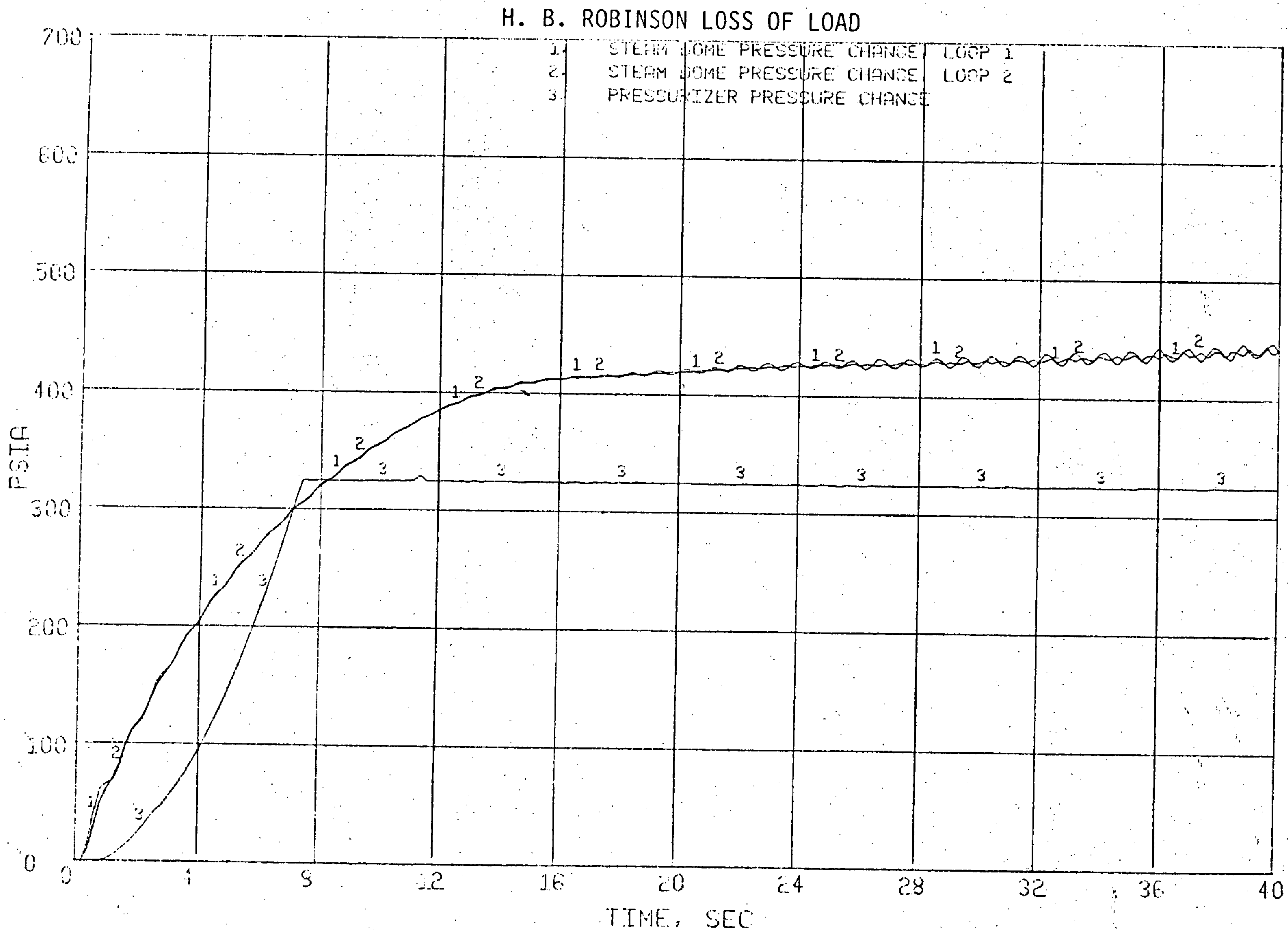

Figure 4 Pressure Changes in Pressurizer and Steam Generators for Loss of Load

H. B. ROBINSON LOSS OF LOAD

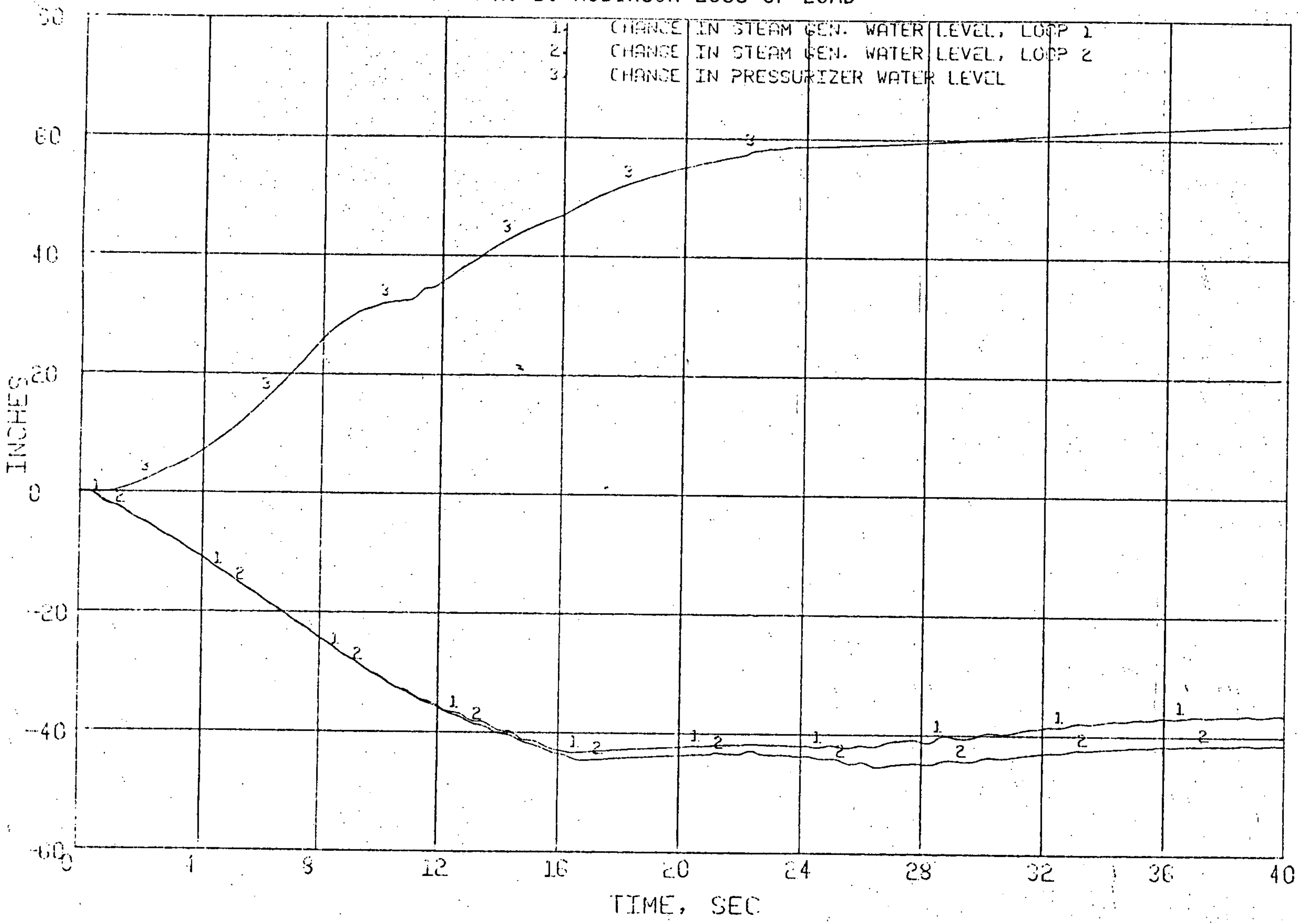

Figure 5 Level Changes in Pressurizer and Steam Generators for Loss of Load

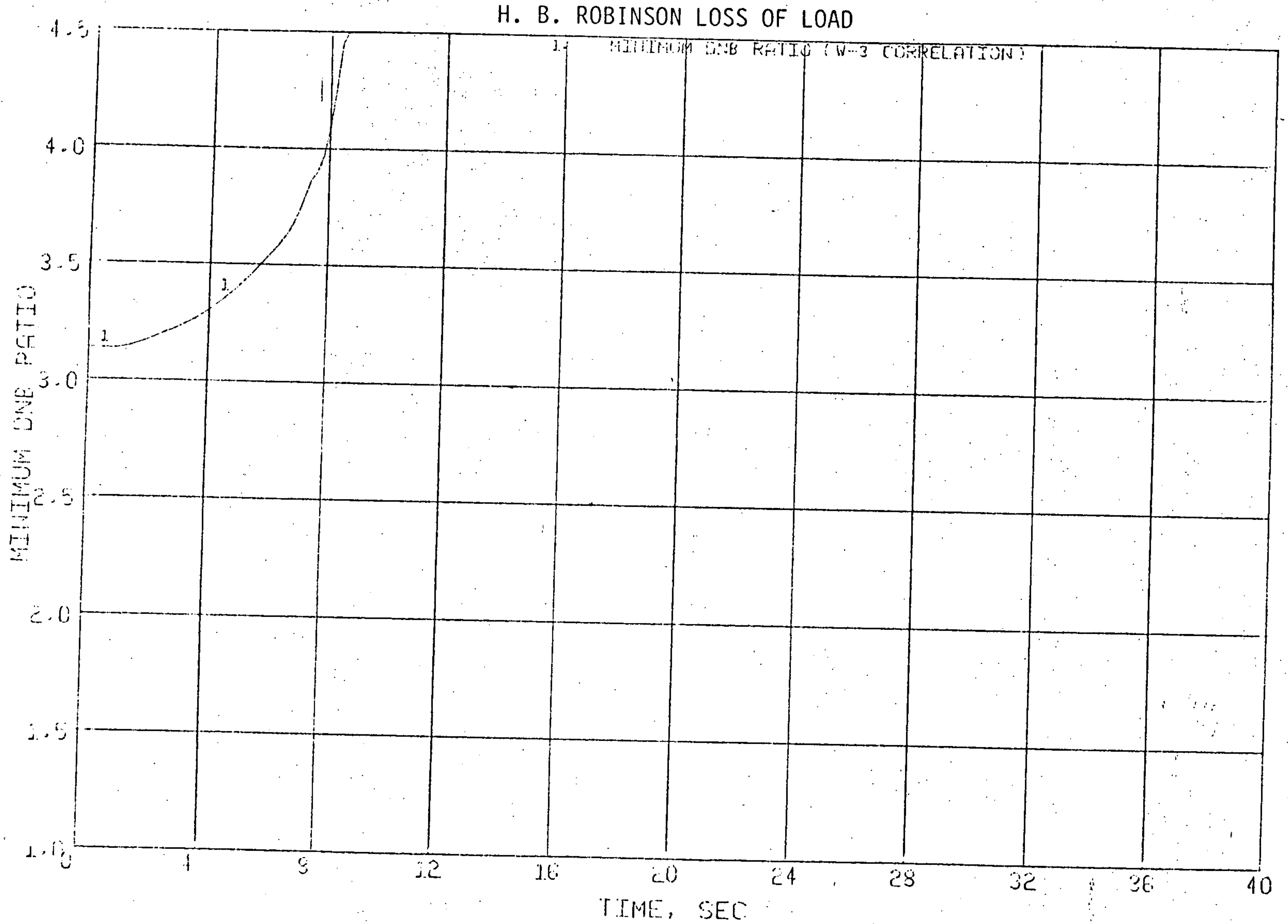

Figure 6 Minimum DNB Ratio for Loss of Load

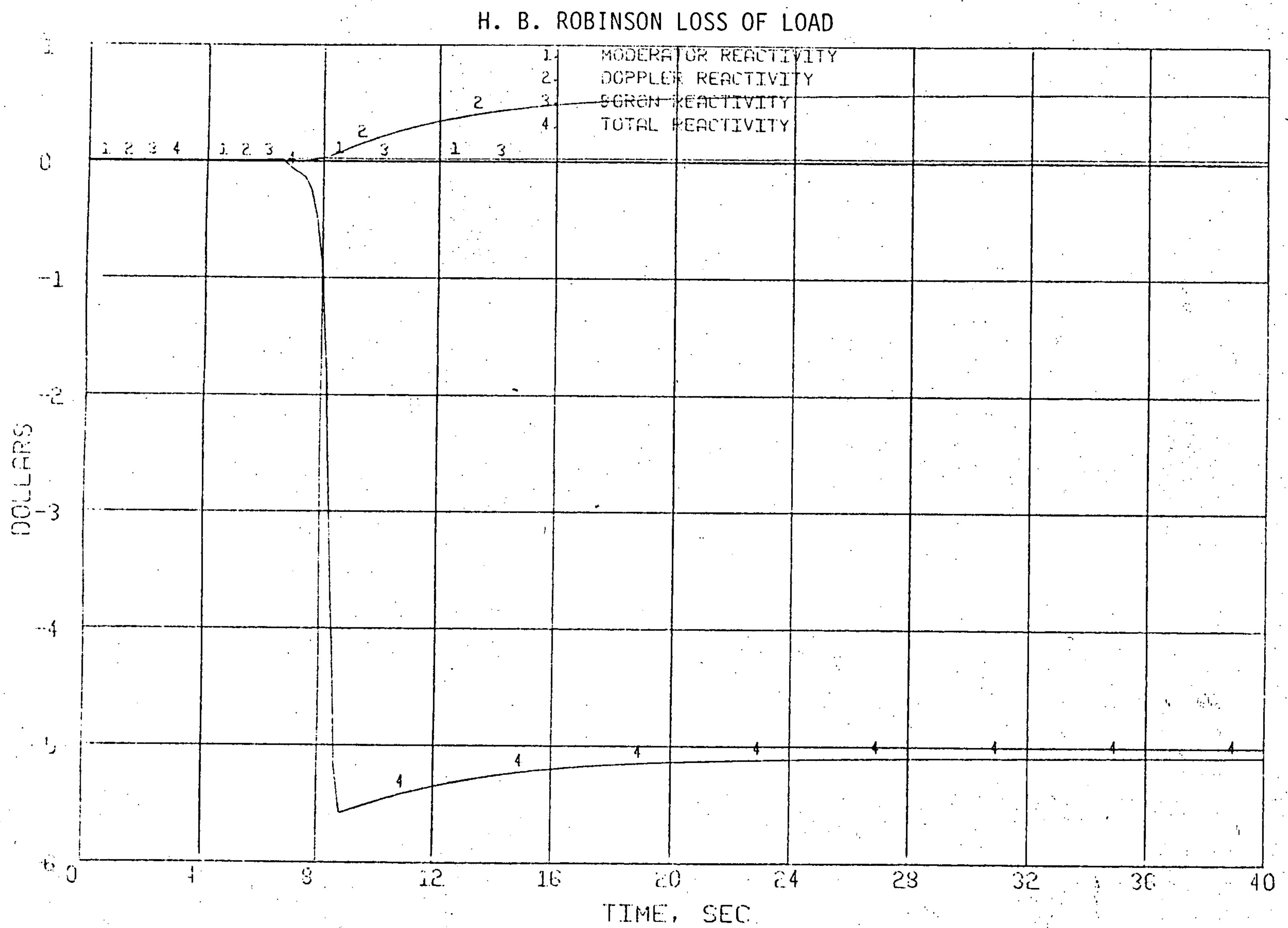

Figure 7 Reactivity Worth for Loss of Load