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ECCS AND PLANT TRANSIENT ANALYSES FOR H.B. ROBINSON UNIT 2 REACTOR OPERATING AT REDUCED PRIMARY TEMPERATURE

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Prepared by:

S.E. Jensen 3/10/82

D.R. Swope

C.H. Wu

F.T. Adams

Reviewed by : G.C. Cooke, Manager
Plant Transient Analysis

Concur : G.F. Owsley, Manager
Reload Fuel Licensing

Approve : J.N. Morgan, Manager 3/11/82
Licensing & Safety Engineering

Approve : G.A. Sofer, Manager 11 MAR 82
Fuel Engineering & Technical Services

/mb

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1.0 INTRODUCTION AND SUMMARY

Safety analyses for the H.B. Robinson Unit 2 nuclear power plant have been performed to support reduced temperature operation at reduced power. The reduced coolant temperature (T_{AVE}) was taken at 537.1^{o}F with power at 85% of full rated power. The LOCA ECCS analysis was performed in accordance with 10 CFR 50, Appendix K, for the limiting double-ended cold leg guillotine (DECLG) break (CD=0.8) at beginning-of-life conditions. The calculated peak clad temperature (PCT) for this break was 2077^{o}F which is in conformance with 10 CFR 50.46 criteria. This result corresponds to a total linear heat generation rate (LHGR) of 11.8 kw/ft, and a total peaking (F_Q^T) of 2.32 at 85% of rated power.

Plant transient analyses have also been performed to support operation of H.B. Robinson Unit 2 at reduced primary coolant temperature and reduced reactor power. The thermal margin criteria of MDNBR $\geq$1.30 is met for all the limiting anticipated operational transients and the locked rotor accident which were analyzed for the new full power (1955 MWt) conditions. Among these transient events initializing from 1955 MWt the locked rotor accident MDNBR was the lowest with a value of 2.19. In all cases, MDNBR is improved relative to prior analyses owing to the reduced coolant temperature and core power being considered.

The large steam line break accident has also been re-analyzed. Since this event is initiated from hot zero power conditions, this accident is largely unaffected by the reduced temperature and power conditions. In the present analysis, the large steam line break results in a minimum critical heat flux ratio of 1.19 using Modified Barnett Critical Heat Flux correlation. For the uncontrolled rod withdrawal transient at hot zero power conditions, MDNBR is calculated to be 2.11. The revised setpoints for overtemperature ΔT and overpower ΔT were confirmed as being adequate for the reduced primary coolant temperature and reactor power conditions of operation.

2.0 H.B. ROBINSON UNIT 2 LOCA ECCS ANALYSIS FOR REDUCED TEMPERATURE OPERATION

This section presents the results of a LOCA ECCS analysis for the H.B. Robinson Unit 2 reactor operating with reduced primary coolant temperature. Reduced temperature operation requires that the reactor operate at reduced power; therefore the analysis was performed at 85% of rated power (2300 MWt). The previously identified limiting break was recalculated with the NRC approved ENC WREM-IIA PWR ECCS evaluation model, and input appropriate for reduced temperature operation.

In addition to reduced operating temperature and power, Carolina Power & Light (CP&L) specified additional parameter changes to assure a bounding analysis for planned operation. These included: increased assumed steam generator tube plugging, reduced primary system flow, and increased radial peaking.

ENC revised the input data for the most recent previous ECCS analysis of H.B. Robinson Unit 2 to incorporate the specified reduced temperature operating conditions. Specific changes are listed in Table 2.1. The steady state pressure and energy balances were recalculated based on these changes. System nodalization and all other parameters remained the same as in previous analyses. The LOCA ECCS calculations were made for the limiting break (0.8 DECLG) and beginning-of-life fuel conditions. These calculations resulted in a peak clad temperature (PCT) of 2077°F and a predicted maximum local ZR/H_2O reaction of 6.05%. The allowed linear heat generation rate (LHGR) is 11.8 kw/ft which corresponds to an F_Q^T of 2.32 at 85% of rated power. The calculated event times are listed in Table 2.2 and calculated results summarized in Table 2.3. Plotted results are shown in Figures 2.1 through 2.27.

The ENC WREM-IIA model includes the following computer codes: RELAP4-EM/ENC28FC for the blowdown and hot channel analyses; CONTEMPT LT/22 for the containment backpressure analysis; REFLEX for the core reflood analysis; and TOODEE2/MAY79 for the heatup analysis. These code versions are identical to those used for several previous ENC analyses including H.B. Robinson Unit 2 except for RELAP4-EM/ENC28FC. This version differs from the previous versions (RELAP4-EM/ENC28FA used for blowdown and RELAP4-EM/ENC28FB used for hot channel) in that the 28FB and 28FC versions include an improved stored energy convergence procedure and a corrected calculation of the critical heat flux. PCT changes resulting from these changes are negligible. Version 28FC differs from 28FB only by adaptation to the Cyber 176 in addition to the Cyber 175 and gives identical results.

The calculated Peak Cladding Temperature (PCT) for reduced temperature operation of 2077°F at 85% power compares with a PCT of 2185°F for the full power, high temperature operation at the same peaking (F_Q^T = 2.32). This confirms the ENC position that the reduction in linear heat generation rate (LHGR) associated with the 15% reduction in power would be sufficient to offset expected detrimental effects associated with reduced temperature operation. Several detrimental effects of the reduced temperature conditions and assumptions were identified in the analysis. They are: (1) Reduced heat transfer during blowdown primarily due to decreased core flow; (2) A slower power decay in the core due to reduced voiding. The lower quality conditions in the core for the reduced temperature case affect the power decay but have little effect on the core heat transfer due to the early lockout of the quality dependent heat transfer regimes required by 10 CFR 50 Appendix K. (3) Reduced containment pressure results from the reduced energy release, and reflood rates are reduced due to the decreased steam density associated with this pressure reduction. (4) Reduced saturation pressures for the lower operating temperatures result in a longer blowdown time with lower pressure early in the blowdown. As a result, accumulators inject early and flow for a longer time during blowdown. 10 CFR 50 Appendix K requires all ECCS coolant injected during blowdown to be assumed lost. The reduced remaining accumulator inventory for low temperature operation gives a lower downcomer water level vs. time, and thus reduced flood rates. (5) The additional conservatisms (increased steam generator tube plugging, reduced system flow, and increased radial peaking) also contribute to increased calculated PCT's.

In conclusion, the analysis for H.B. Robinson Unit 2 with ENC fuel, operating under the specified reduced temperature and power conditions, with a total LHGR equal to or less than 11.8 kw/ft (F_Q^T = 2.32) shows that the maximum PCT that the plant can experience due to the limiting LOCA is 2077°F. The maximum local metal-water reaction calculated is 6.05%, well below the 17% limits of 10 CFR 50.46, and the total core-wide metal-water reaction is less than 1.0%.

Table 2.1 Input Data Changes for Reduced Temperature Operation

	Previous Analysis	This Analysis
Tube Plugging	15%	20%
Power (nominal)	2300 MW	1955 MW
$F_{\Delta H}$	1.55	1.60
F_Q^T	2.32	2.32
Primary Coolant Flow at Pumps	89965 gpm/loop	82700 gpm/loop
Vessel T_{AVE} ^oF	579.5	537.1
S.G. Tubes (per loop):		
Fluid Volume	580.4 ft^3	538.4 ft^3
Flow Area	36.316 ft^2	34.176 ft^2
Heat Transfer Area - inside, Heat Transfer Area - outside, Slab Volume	85% nominal	80% nominal
Secondary Pressure	781.2 psi	580 psi
Temperature	426.4^{o}F	416.63^{o}F
T_{SAT}	525.7^{o}F	482.57^{o}F
Flow/loop	932.2 lbm/s	777.4 lbm/s

Table 2.2 H.B. Robinson Limiting Break Event Times for Reduced Temperature Operation

Event	Time (seconds)
Start	0.0
Initiate Break	0.1
Safety Injection Signal	0.6
Accumulator Injection, Broken Loop	1.3
Accumulator Injection, Intact Loop	11.0
Pressurizer Empties	8.5
End-of-Bypass	23.9
Safety Pump Injection, HPSI	25.6
Start of Reflood	47.39
Safety Pump Injection, LPSI	48.9
Accumulators Empty	55.27
Peak Clad Temperature Reached	61.2

Table 2.3 H.B. Robinson Unit 2 Limiting Break ECCS Results for Reduced Temperature Operation

Parameter	Results
Peak Cladding Temperature, °F	2077
Peak Temperature Location, ft.	6.0
Local Zr/H_2O Reaction (max) %	<7.0
Local Zr/H_2O Location, ft.	6.0
Total Zr/H_2O Reaction, %	<1.0
Hot Rod Burst Time, sec.	38.7
Hot Rod Burst Location, ft.	6.0

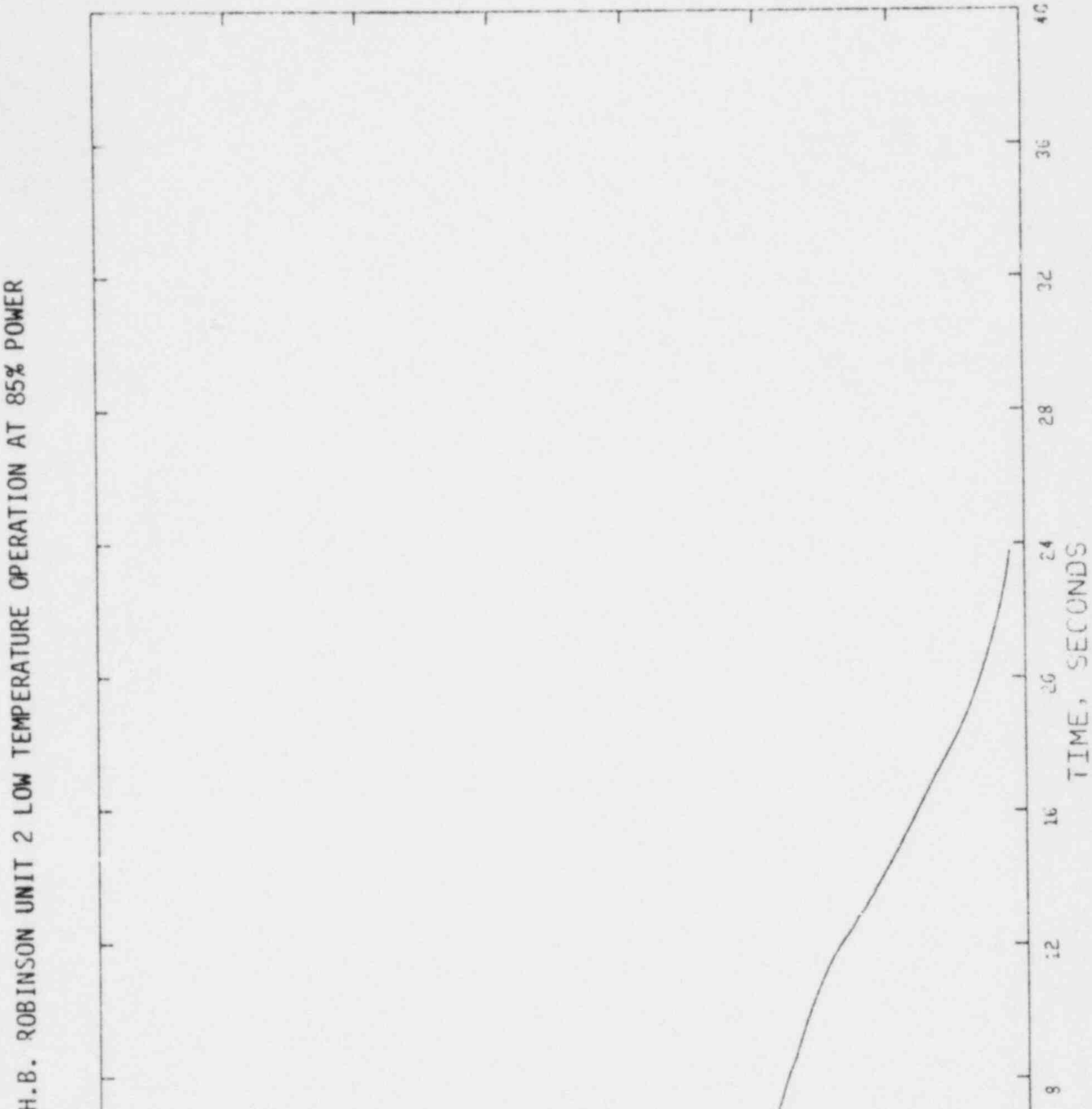

Figure 2.1 Blowdown System Pressure - 0.8 DECLG

H.B. ROBINSON UNIT 2 LOW TEMPERATURE OPERATION AT 85% POWER

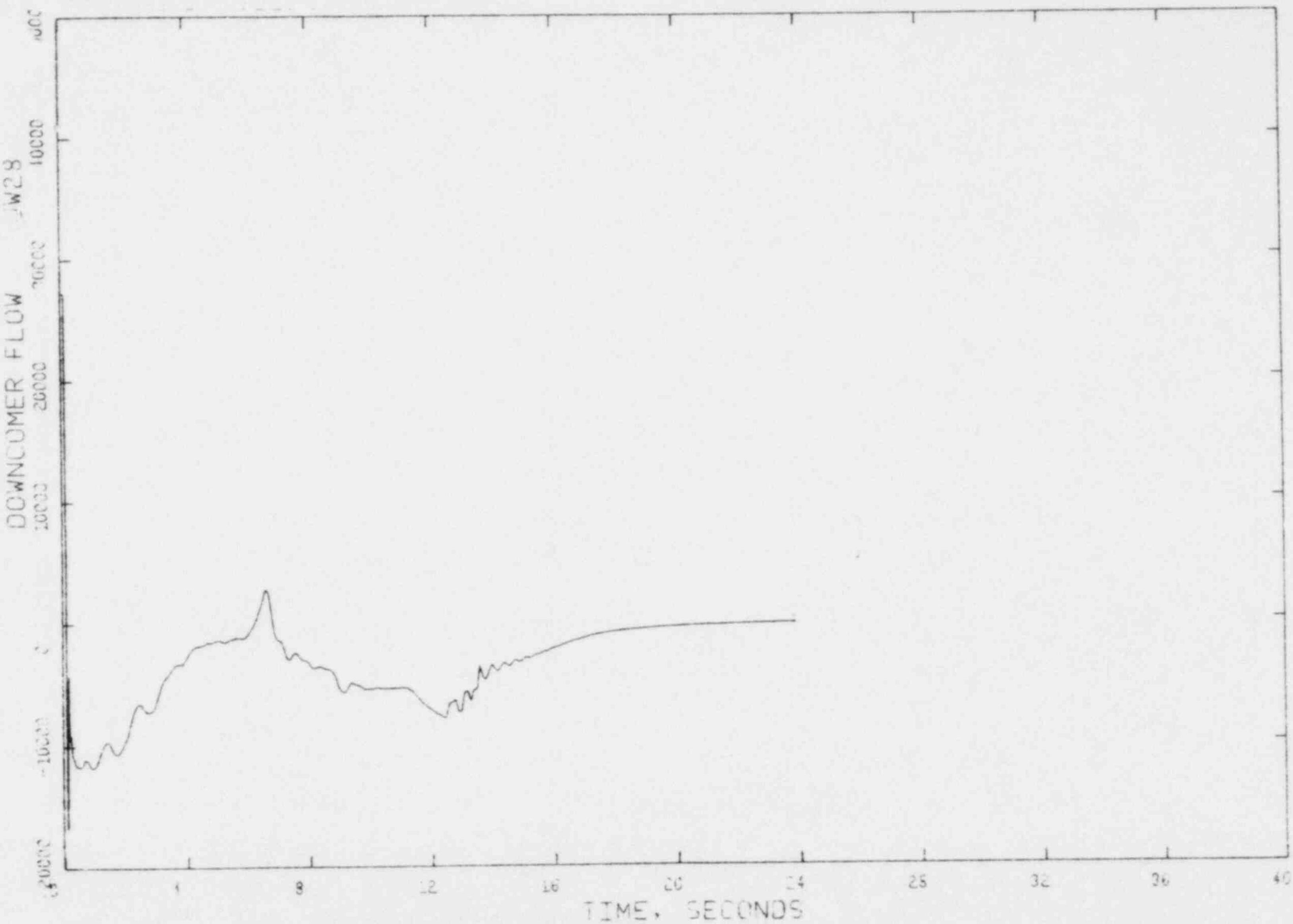

Figure 2.2 Blowdown Downcomer Flow - 0.8 DECLG

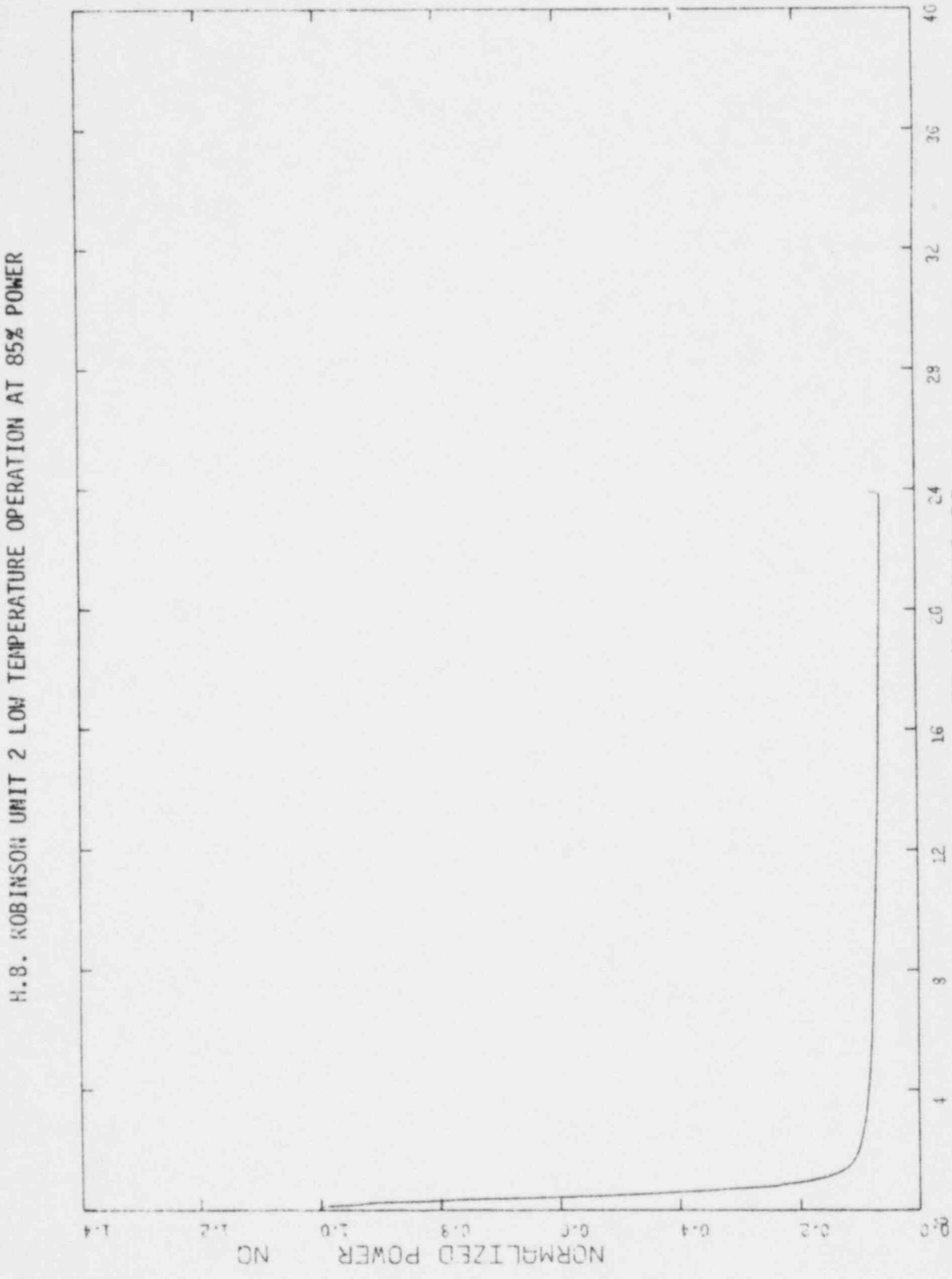

Figure 2.3 Blowdown Normalized Power - 0.8 DECLG

H.B. ROBINSON UNIT 2 LOW TEMPERATURE OPERATION AT 85% POWER

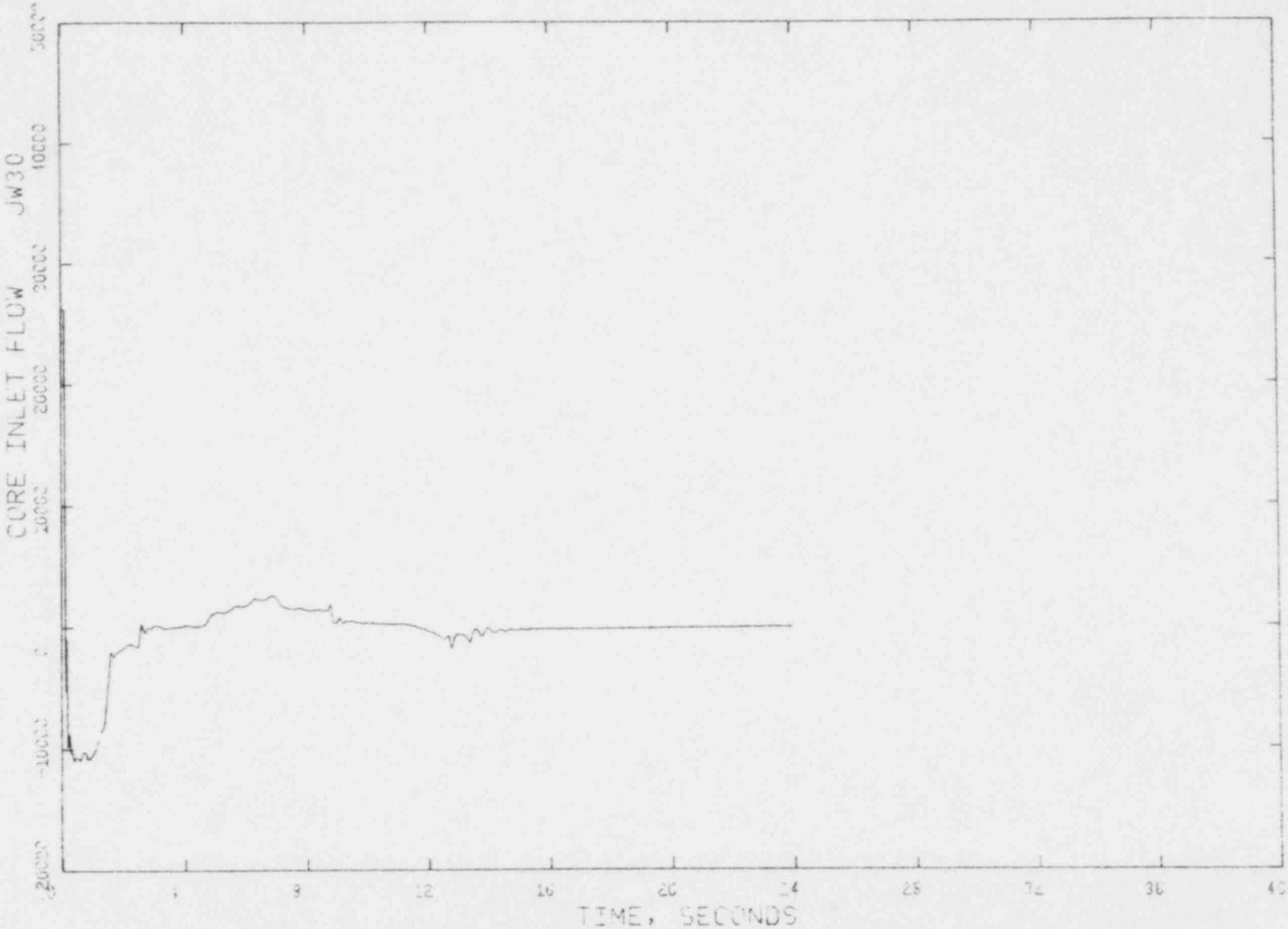

Figure 2.4 Blowdown Core Inlet Flow - 0.8 DECLG

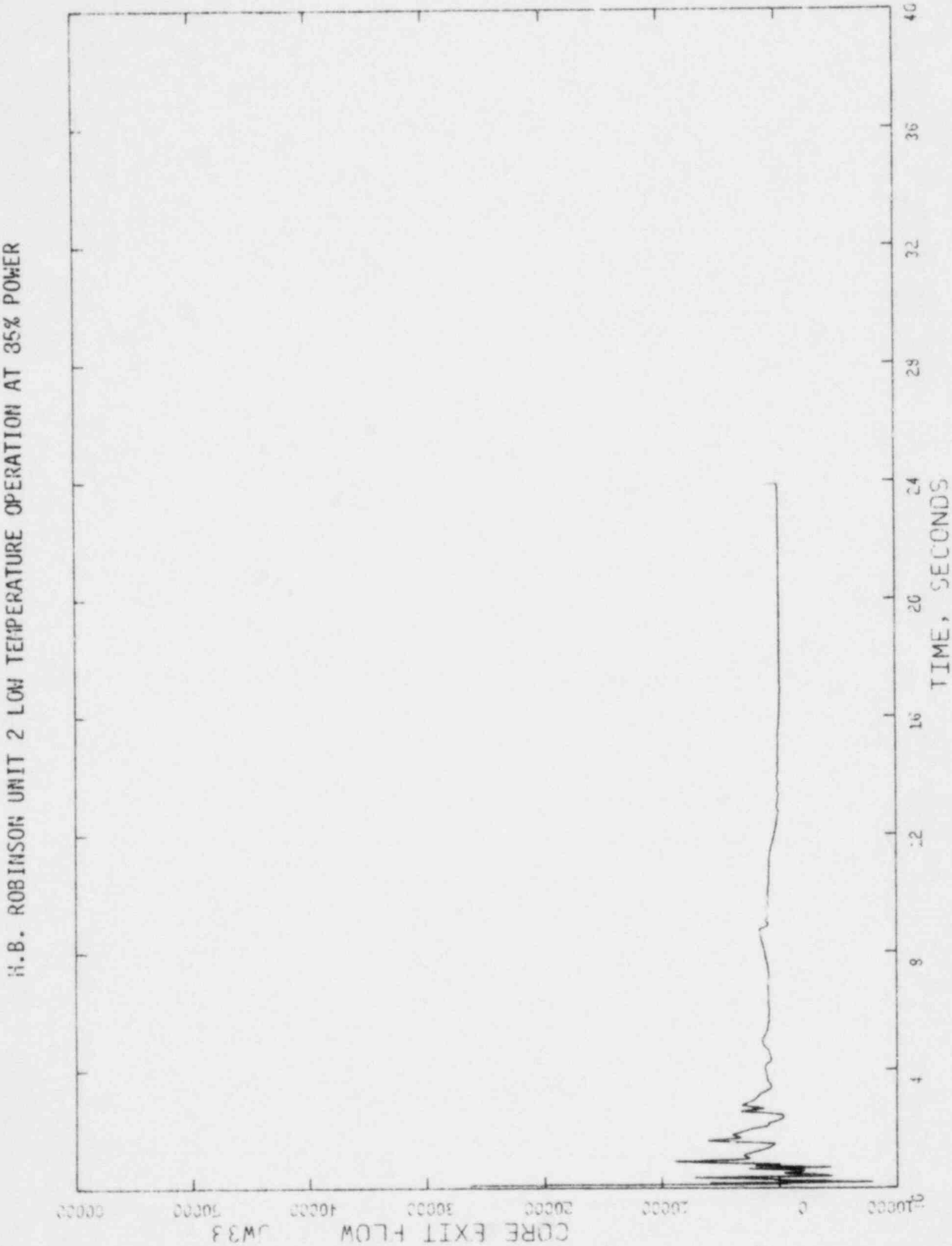

Figure 2.5 Blowdown Core Exit Flow - 0.8 DECLG

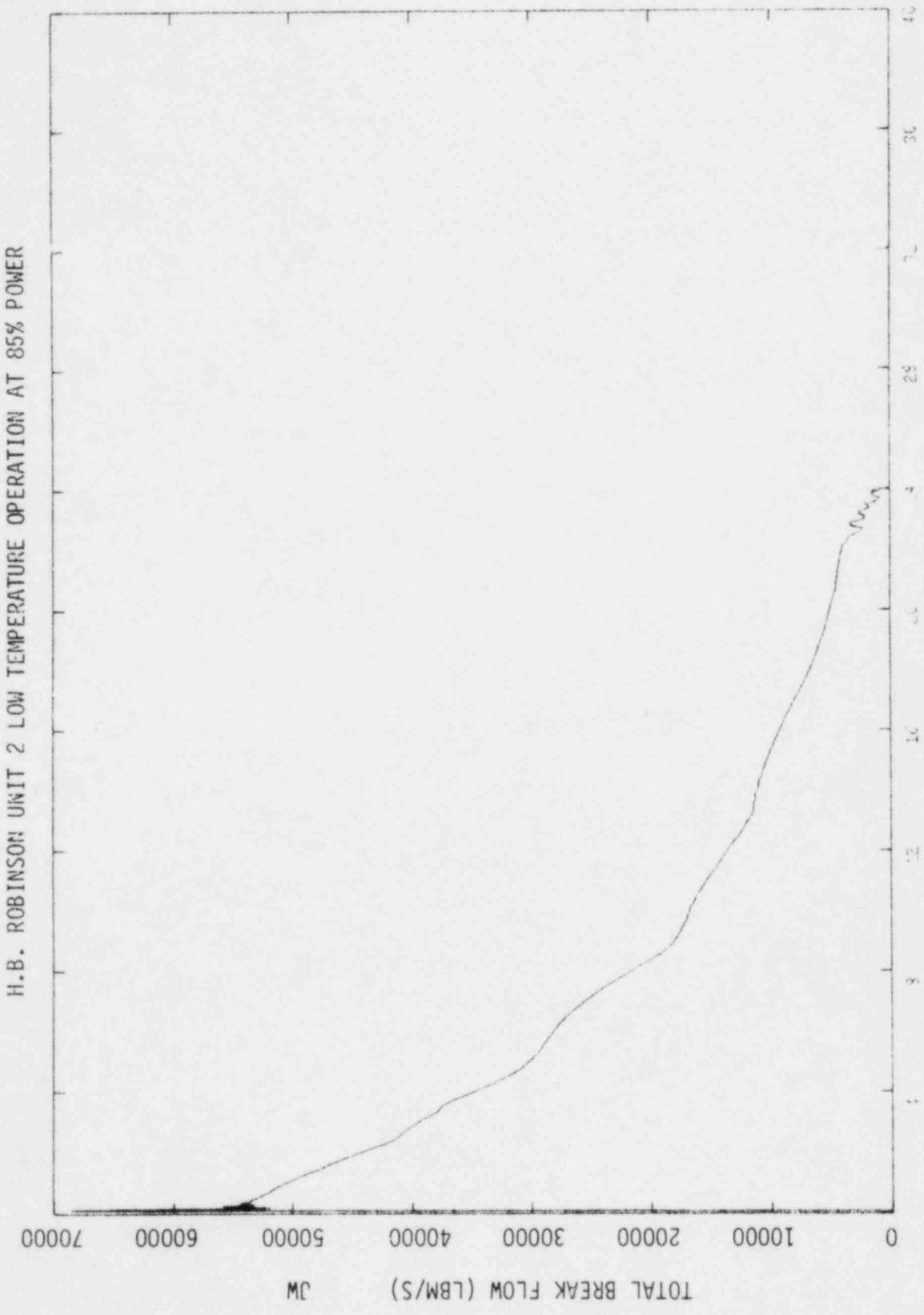

Figure 2.6 Blowdown Total Break Flow - 0.8 DECLG

H.B. ROBINSON UNIT 2 LOW TEMPERATURE OPERATION AT 85% POWER

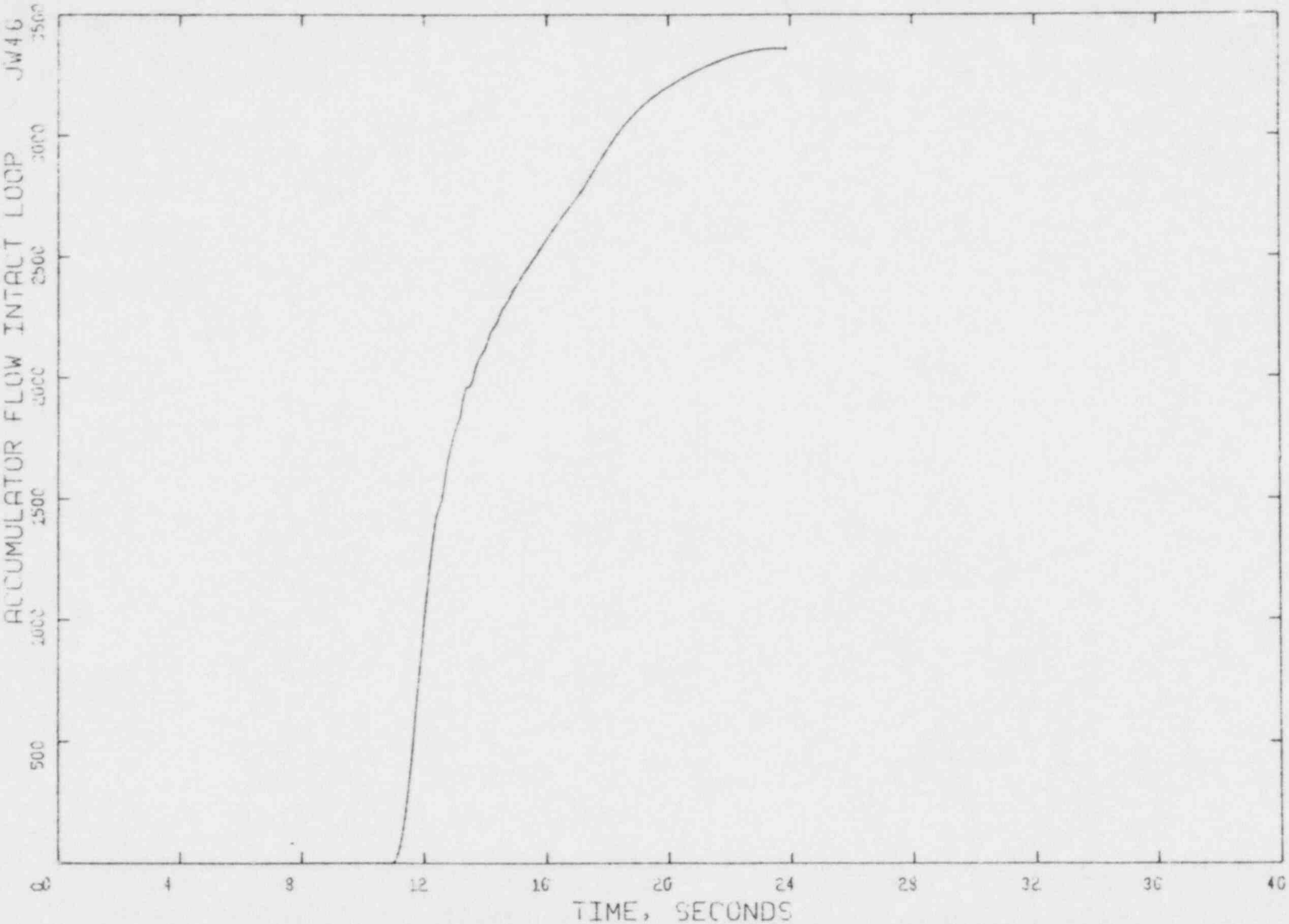

Figure 2.7 Blowdown Intact Loop Accumulator Flow - 0.8 DECLG

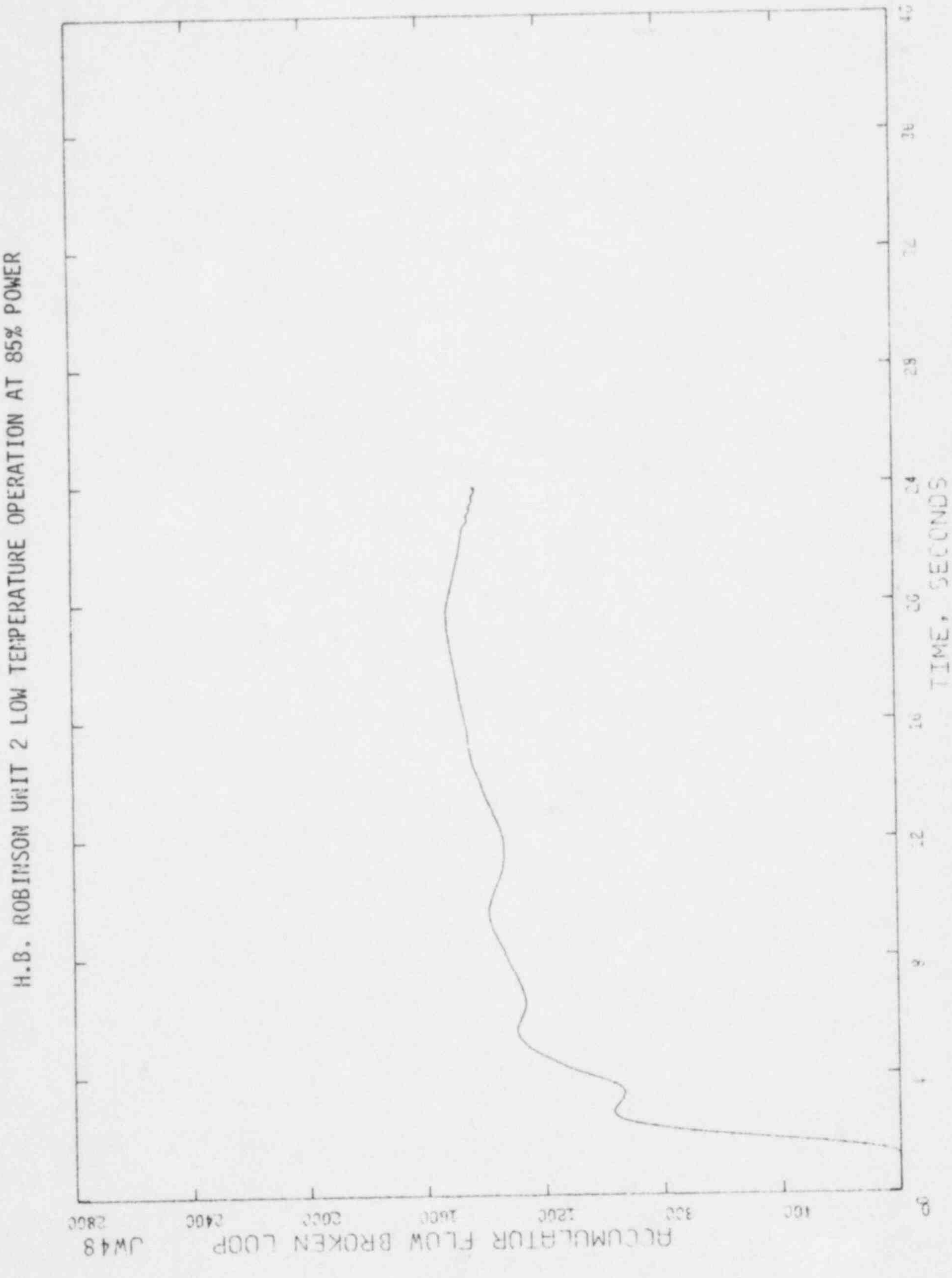

Figure 2.8 Blowdown Broken Loop Accumulator Flow - 0.8 DECLG

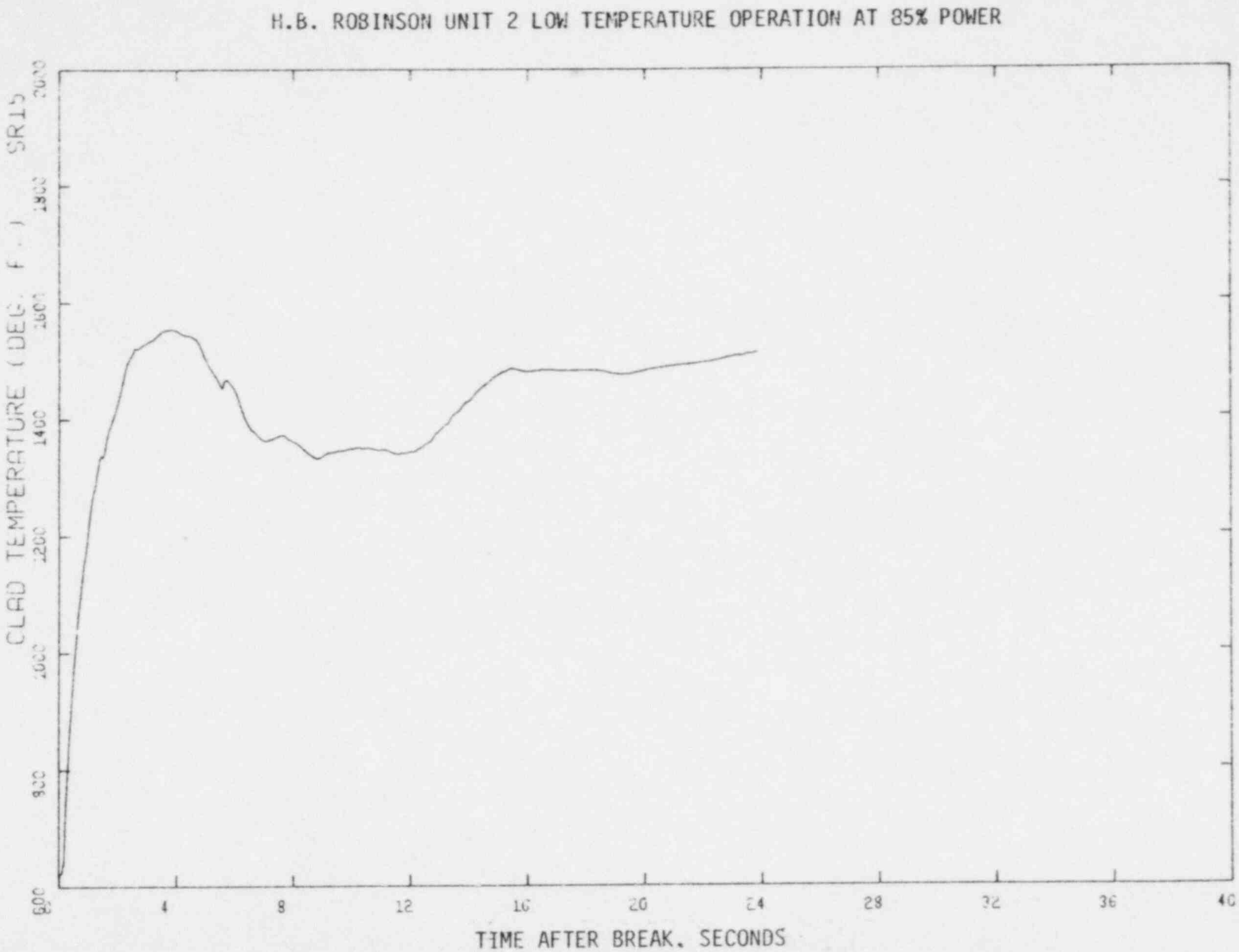

Figure 2.9 Hot Rod Clad Temperature at PCT Location - 0.8 DECLG

H.B. ROBINSON UNIT 2 LOW TEMPERATURE OPERATION AT 85% POWER

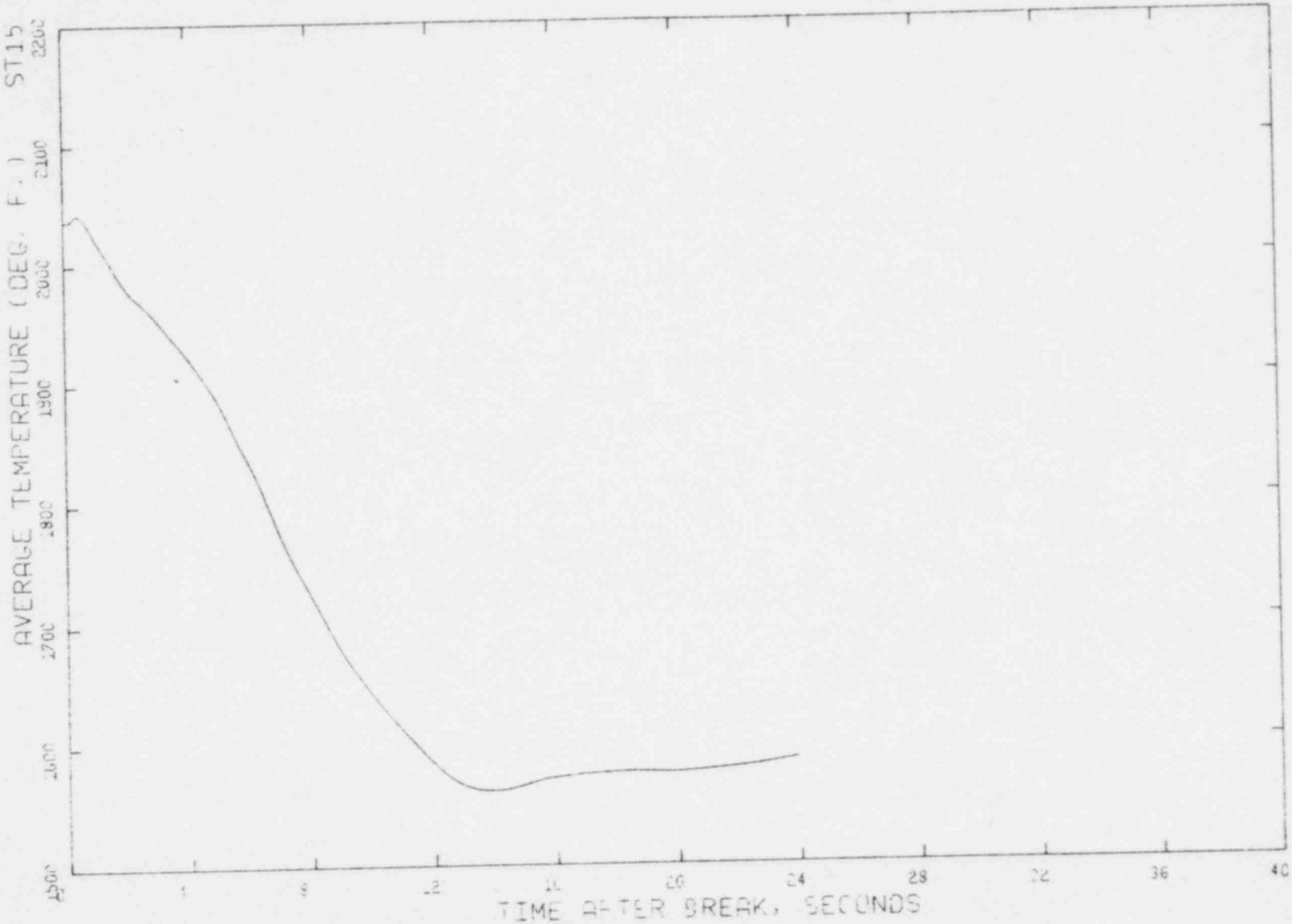

Figure 2.10 Hot Rod Average Temperature at PCT Location - 0.8 DECLG

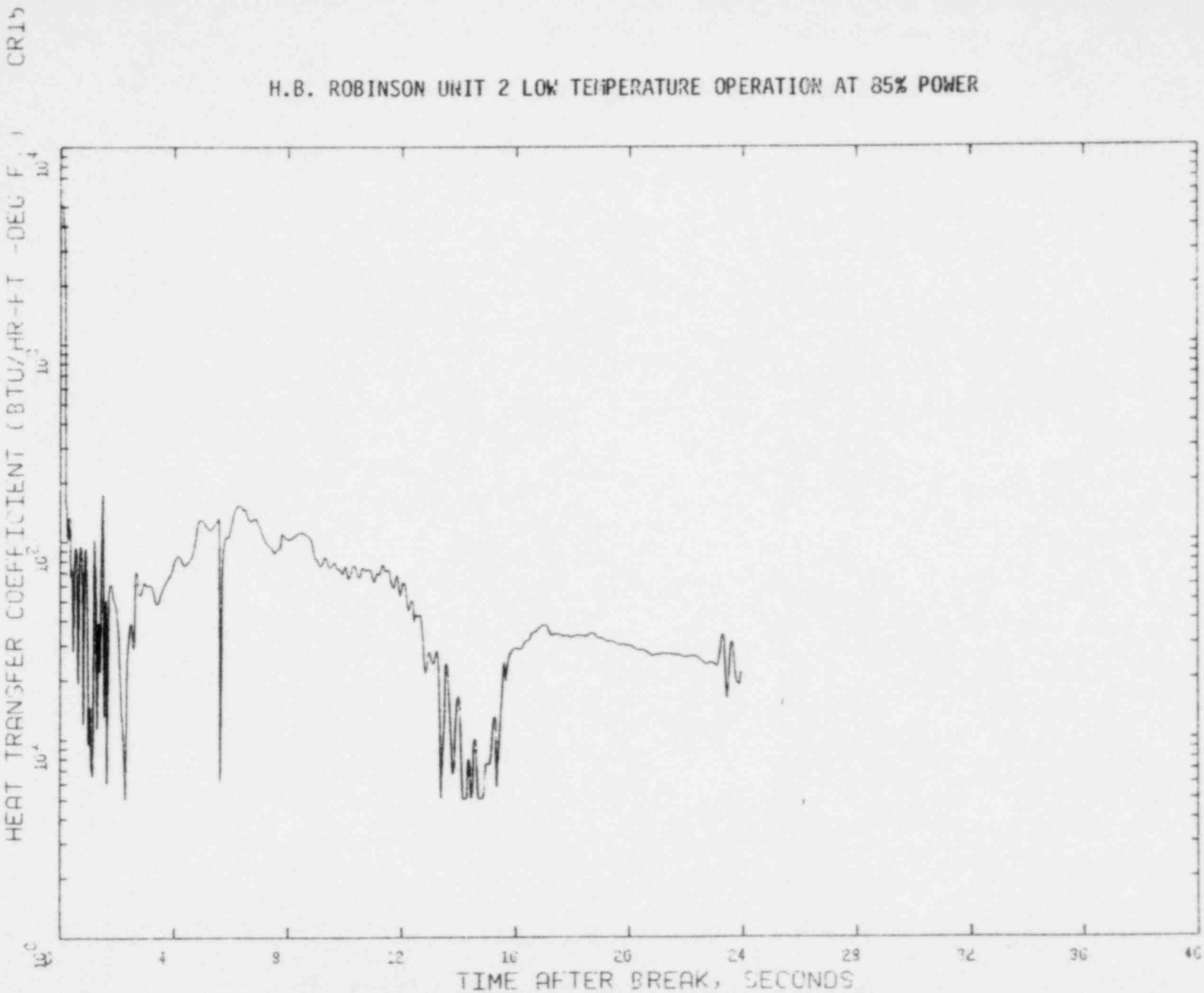

Figure 2.11 Hot Rod Heat Transfer Coefficient at PCT Location - 0.8 DECLG

H.B. ROBINSON UNIT 2 LOW TEMPERATURE OPERATION AT 85% POWER

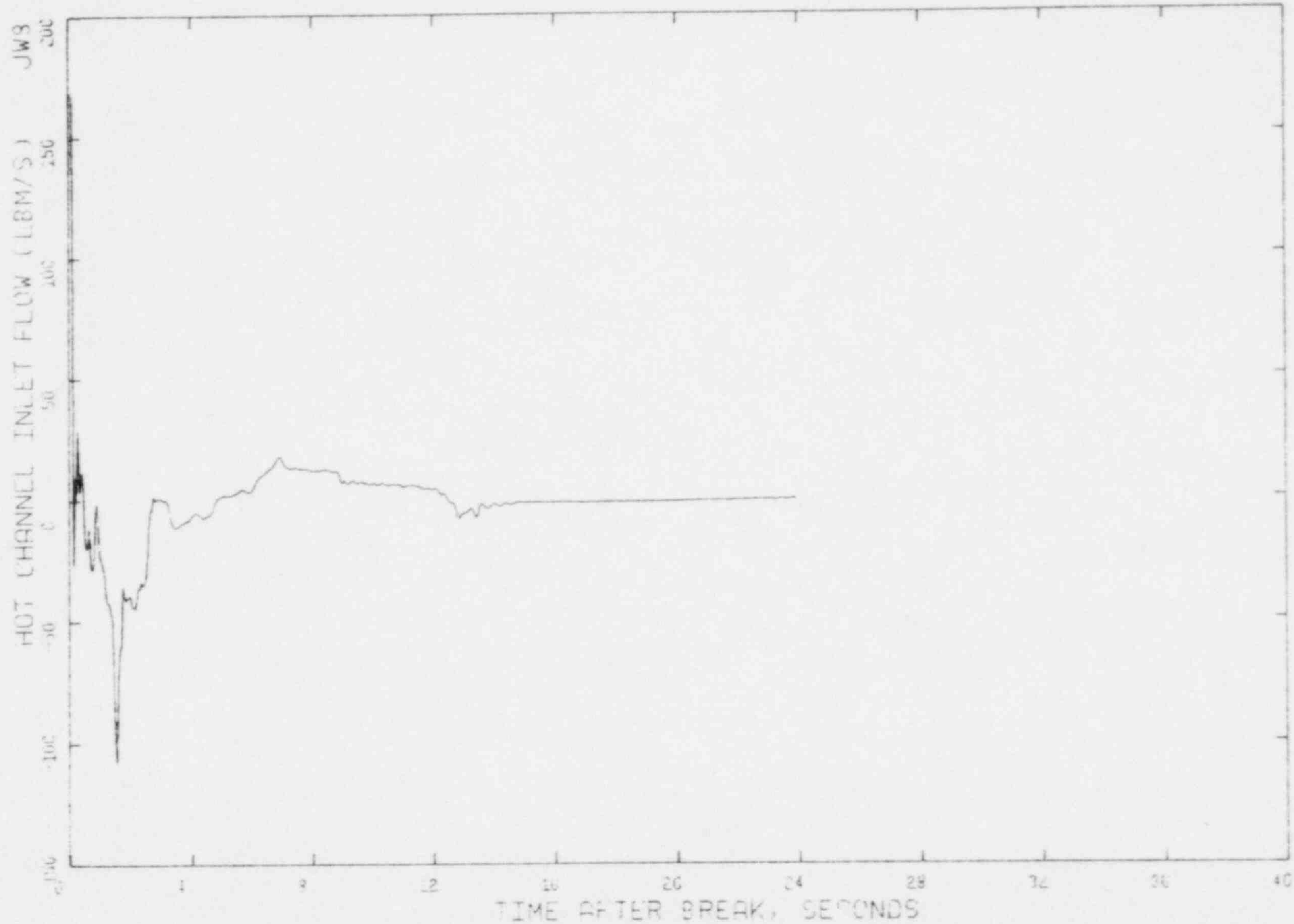

Figure 2.12 Hot Channel Inlet Flow - 0.8 DECLG

H.B. ROBINSON UNIT 2 LOW TEMPERATURE OPERATION AT 85% POWER

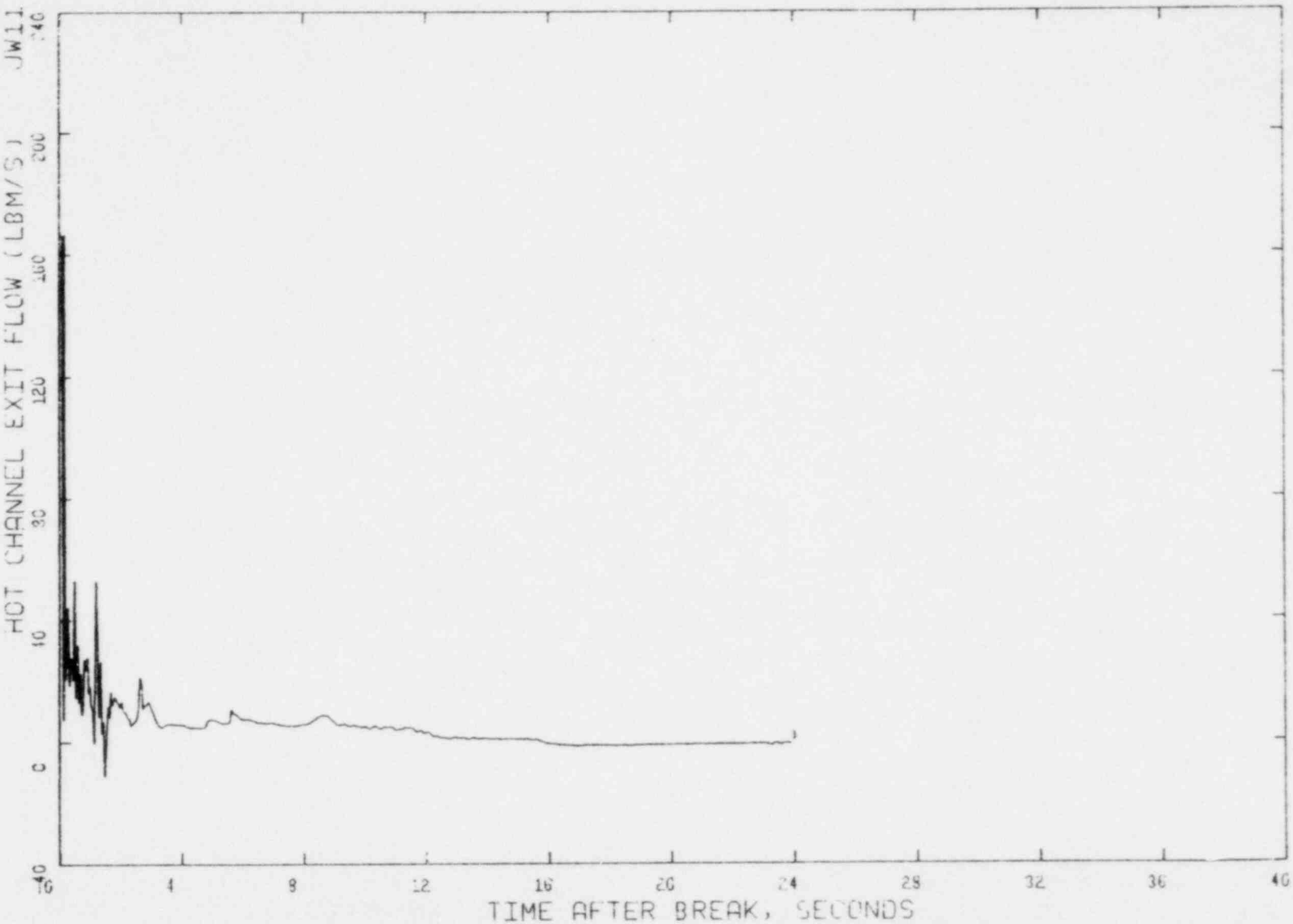

Figure 2.13 Hot Channel Exit Flow - 0.8 DECLG

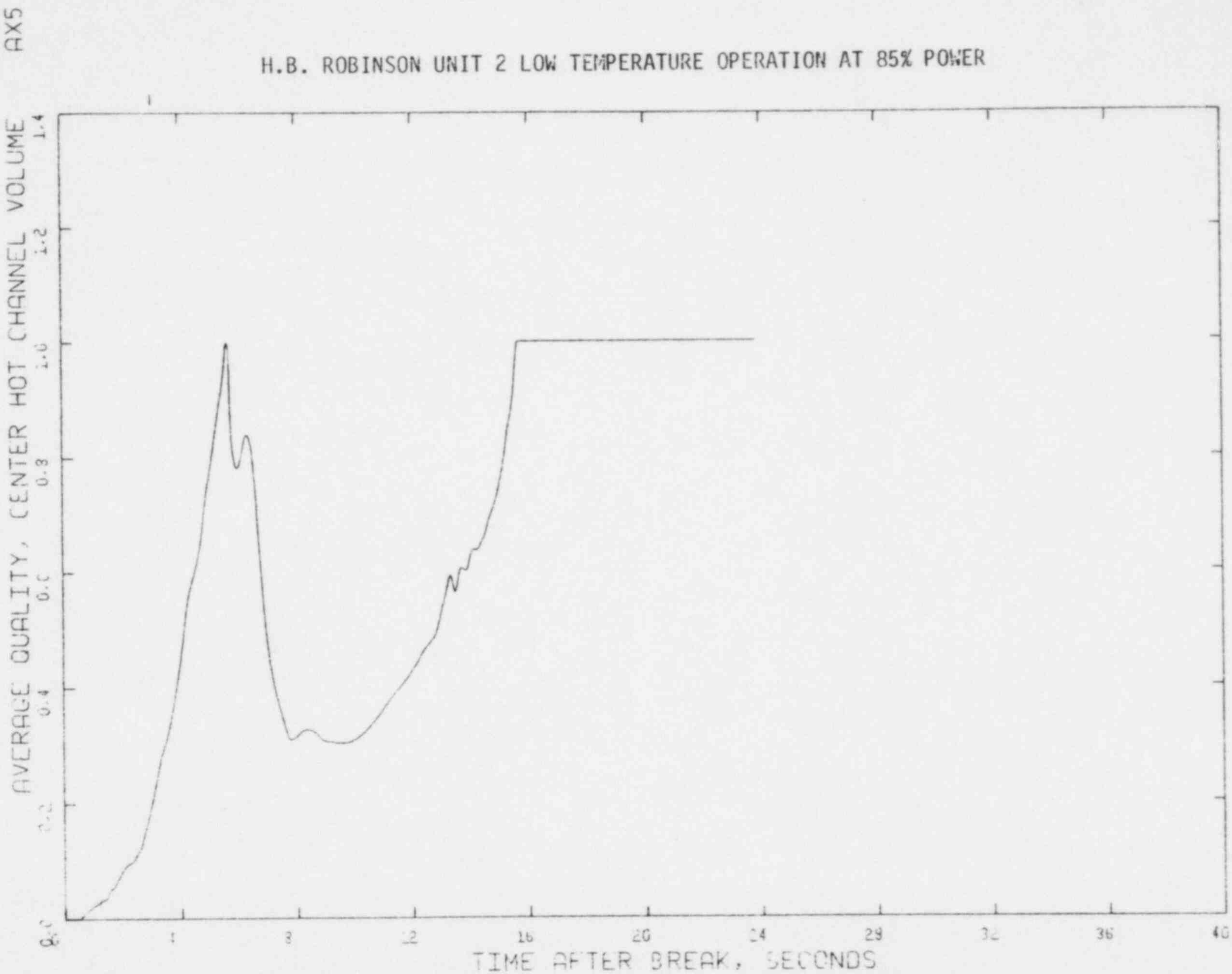

Figure 2.14 Hot Channel Center Volume Average Quality - 0.8 DECLG

H.B. ROBINSON UNIT 2 LOW TEMPERATURE OPERATION AT 85% POWER

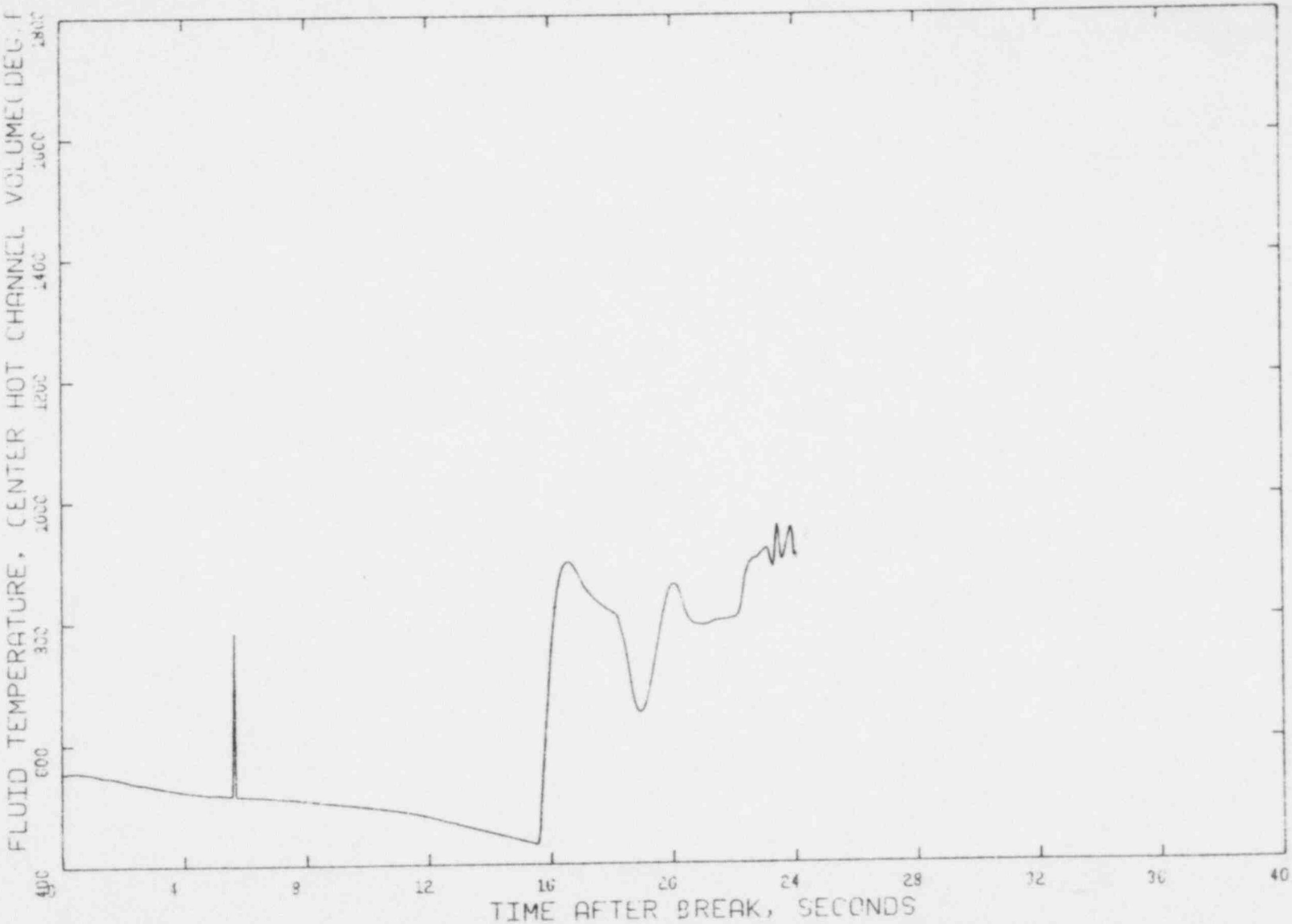

Figure 2.15 Hot Channel Center Volume Fluid Temperature - 0.8 DECLG

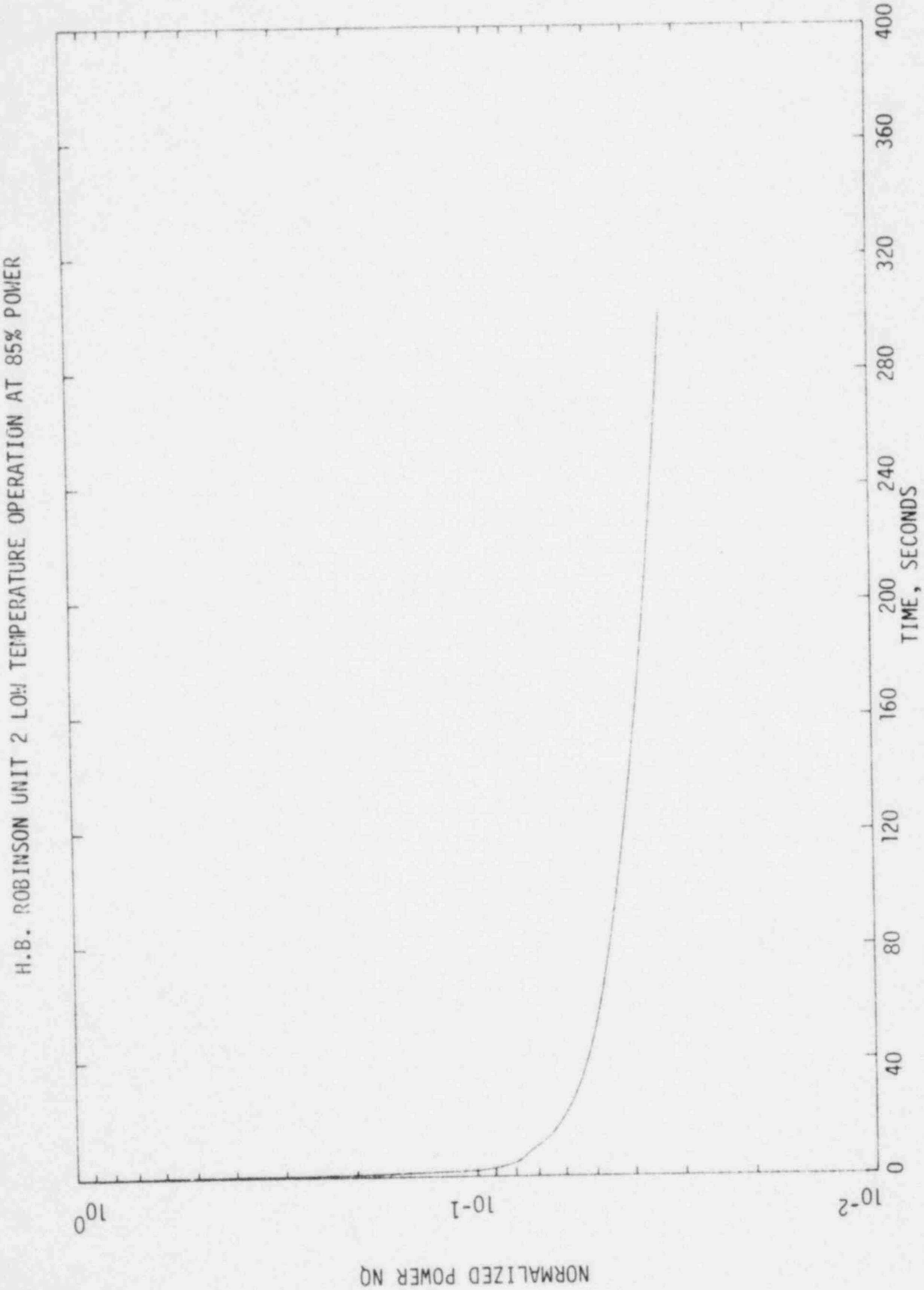

Figure 2.16 Extended Normalized Power - 0.8 DECLG

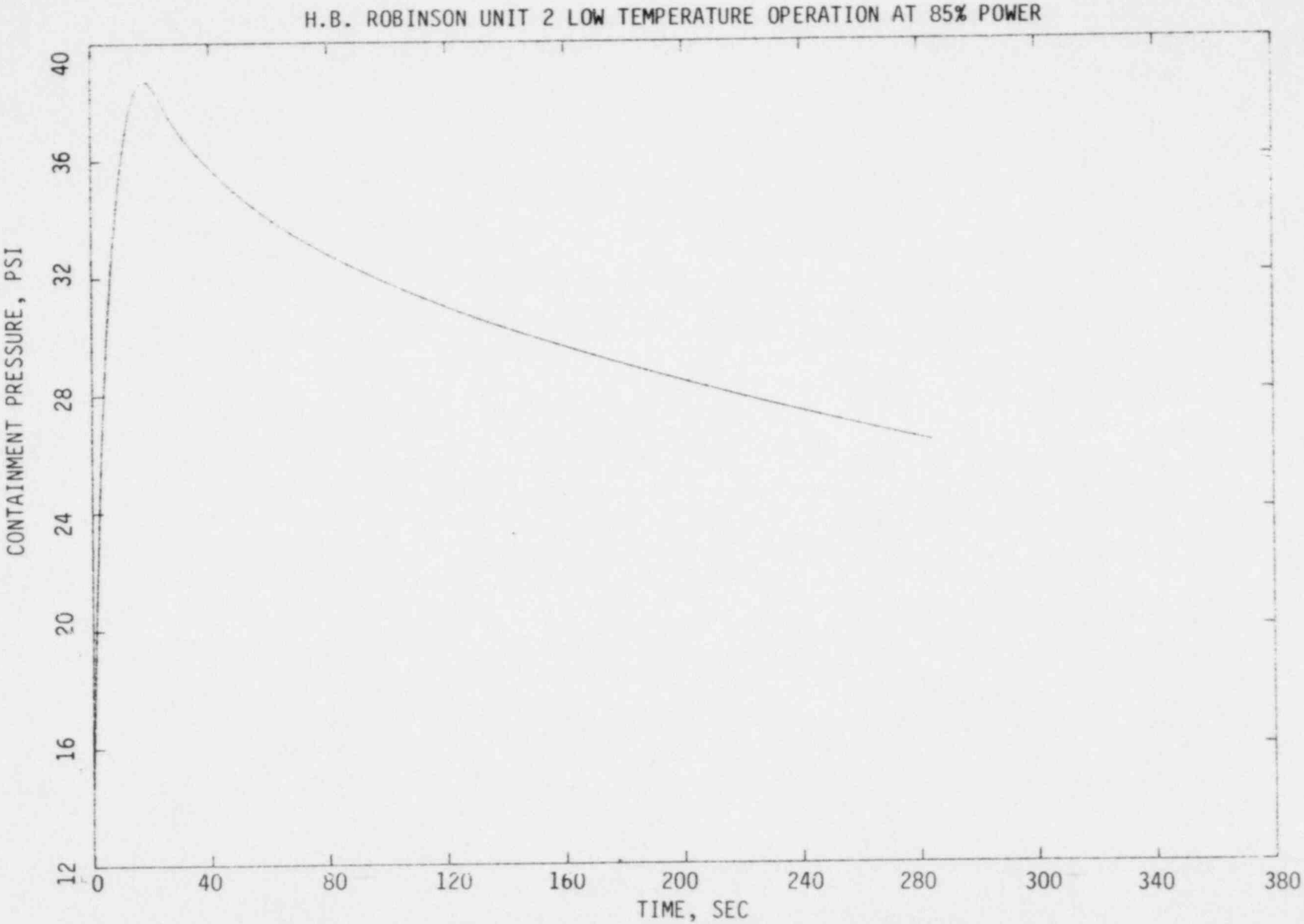

Figure 2.17 Containment Pressure - 0.8 DECLG

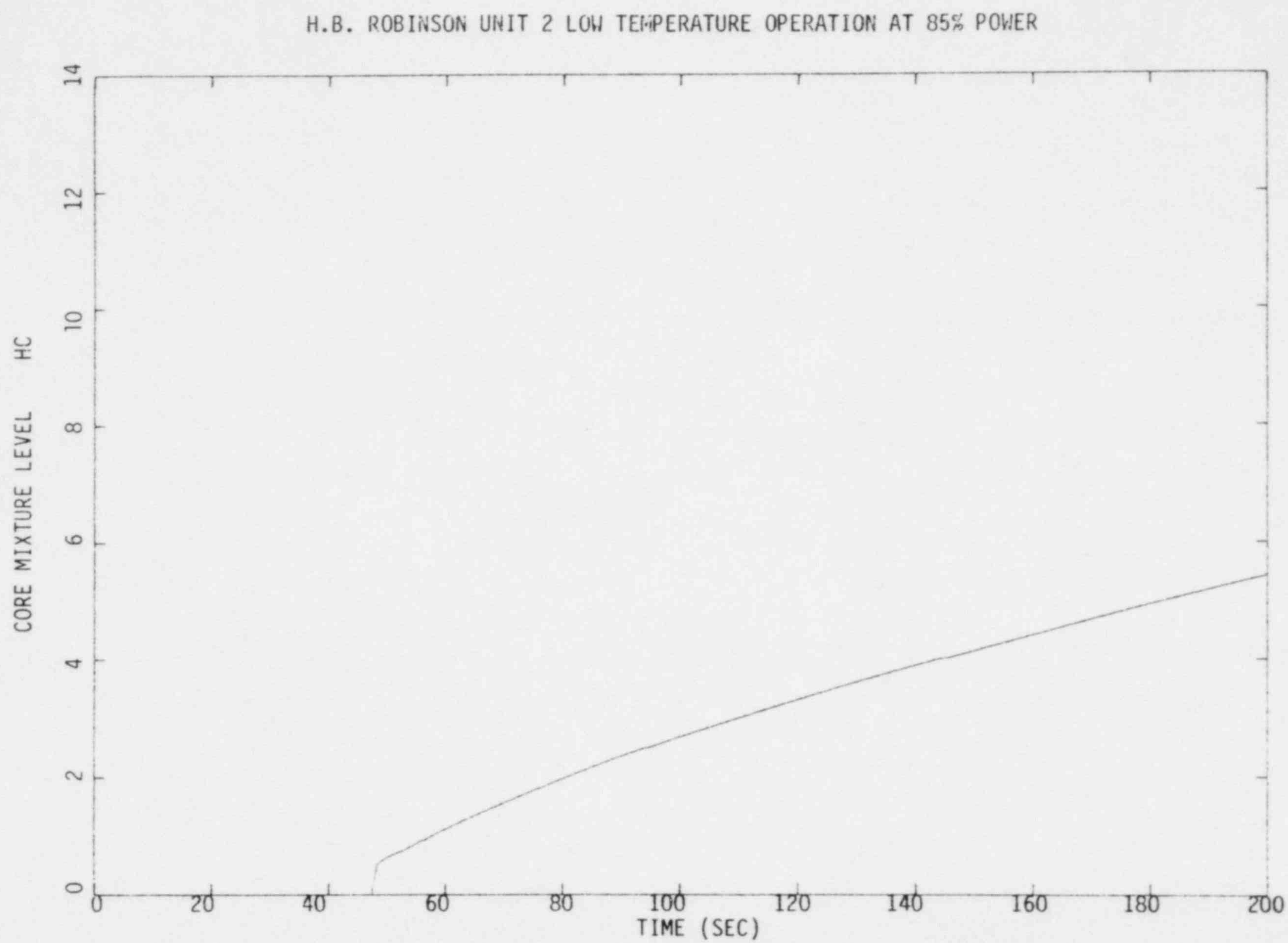

Figure 2.18 Core Mixture Level - 0.8 DECLG

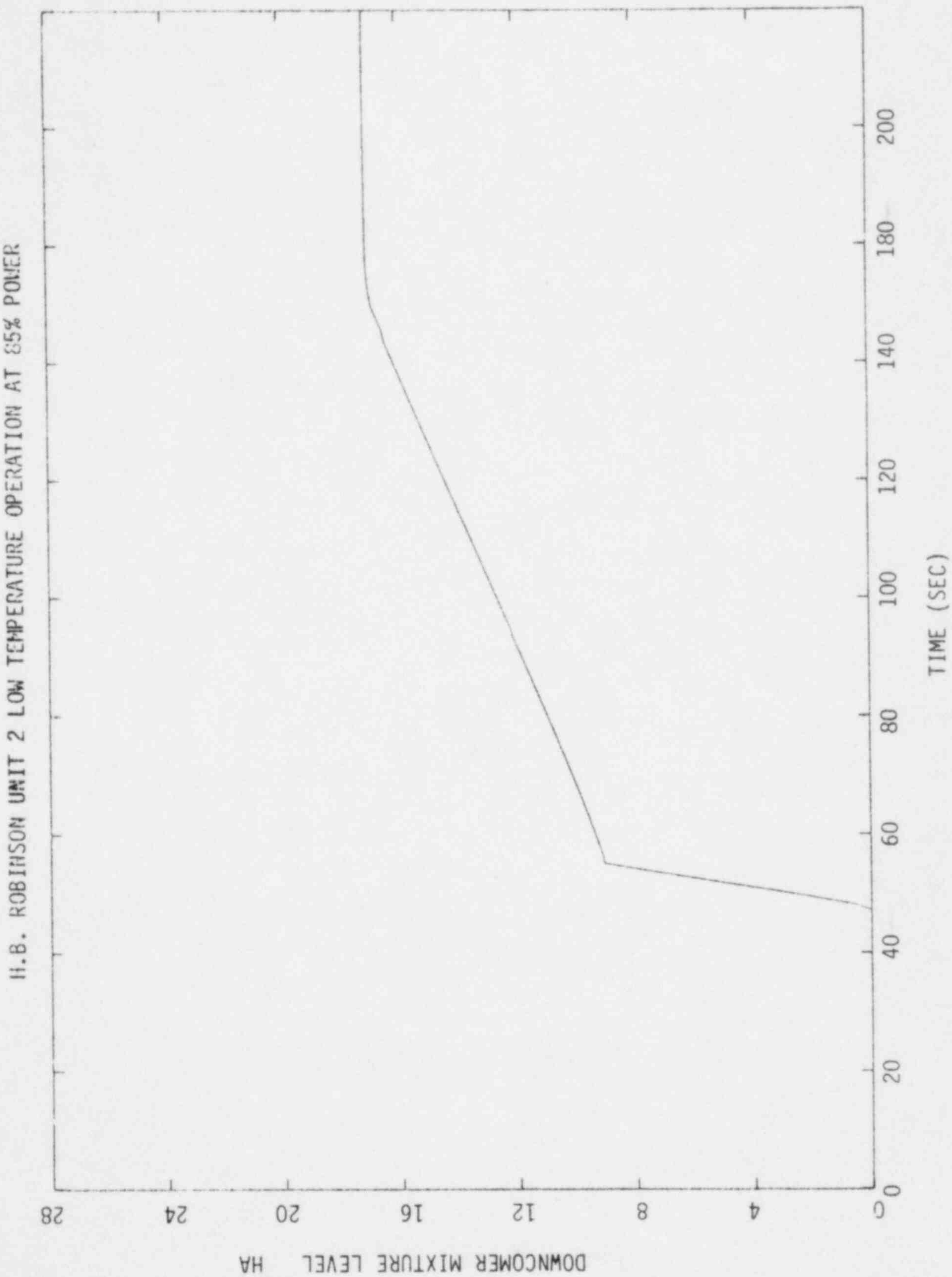

Figure 2.19 Downcomer Mixture Level - 0.8 DECLG

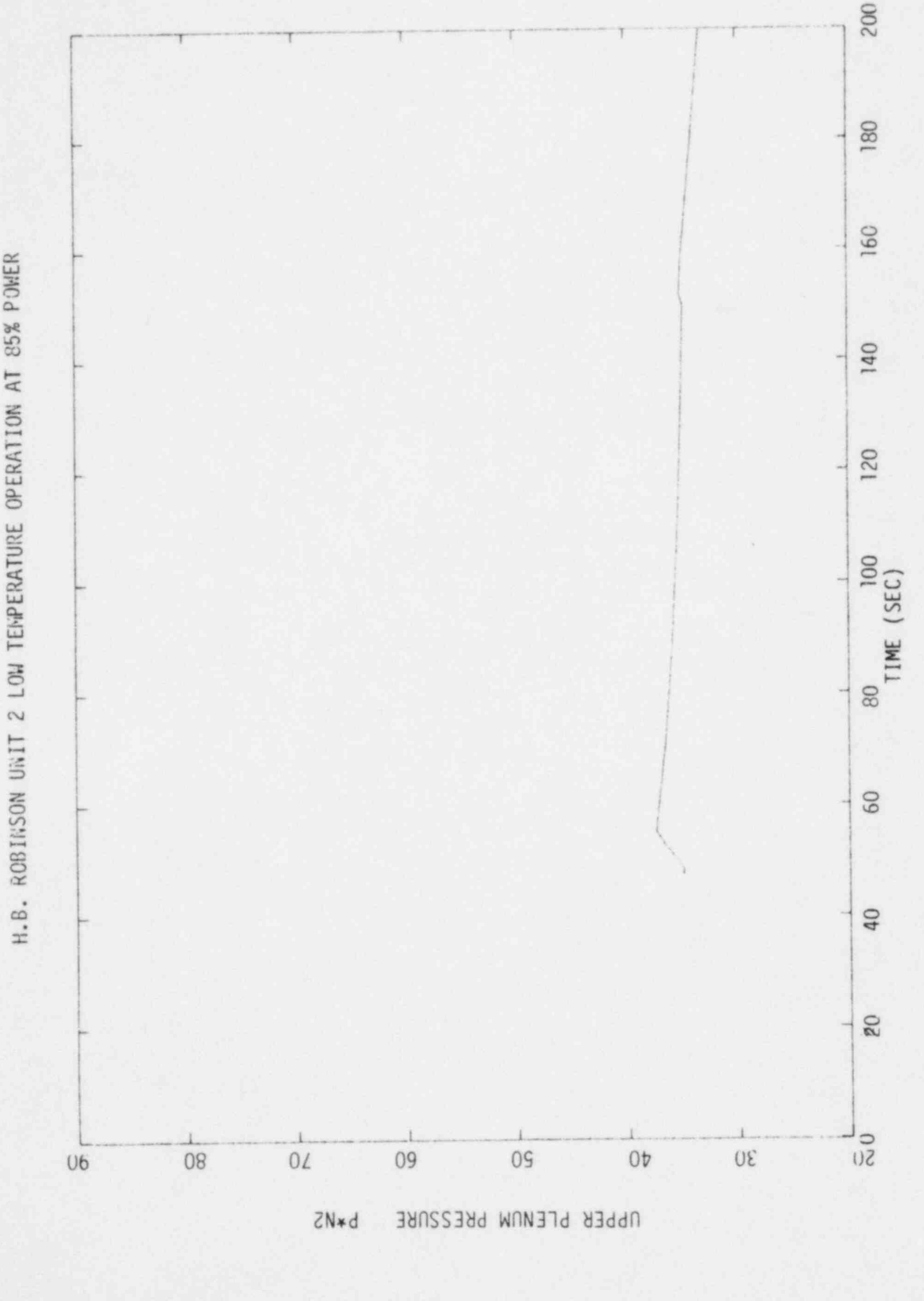

Figure 2.20 Upper Plenum Pressure - 0.8 DECLG

H.B. ROBINSON UNIT 2 LOW TEMPERATURE OPERATION AT 85% POWER

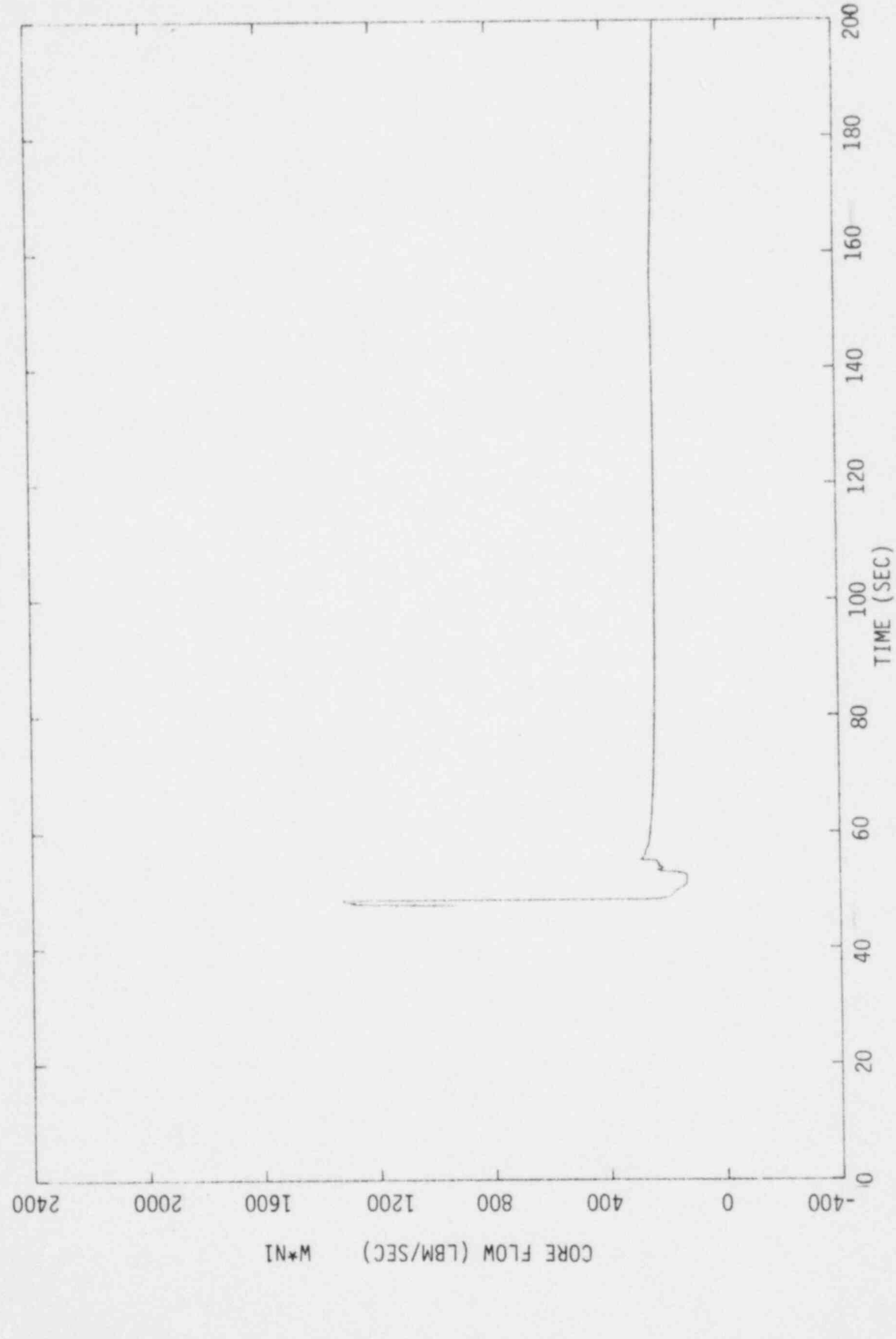

Figure 2.21 Core Flow - 0.8 DECLG

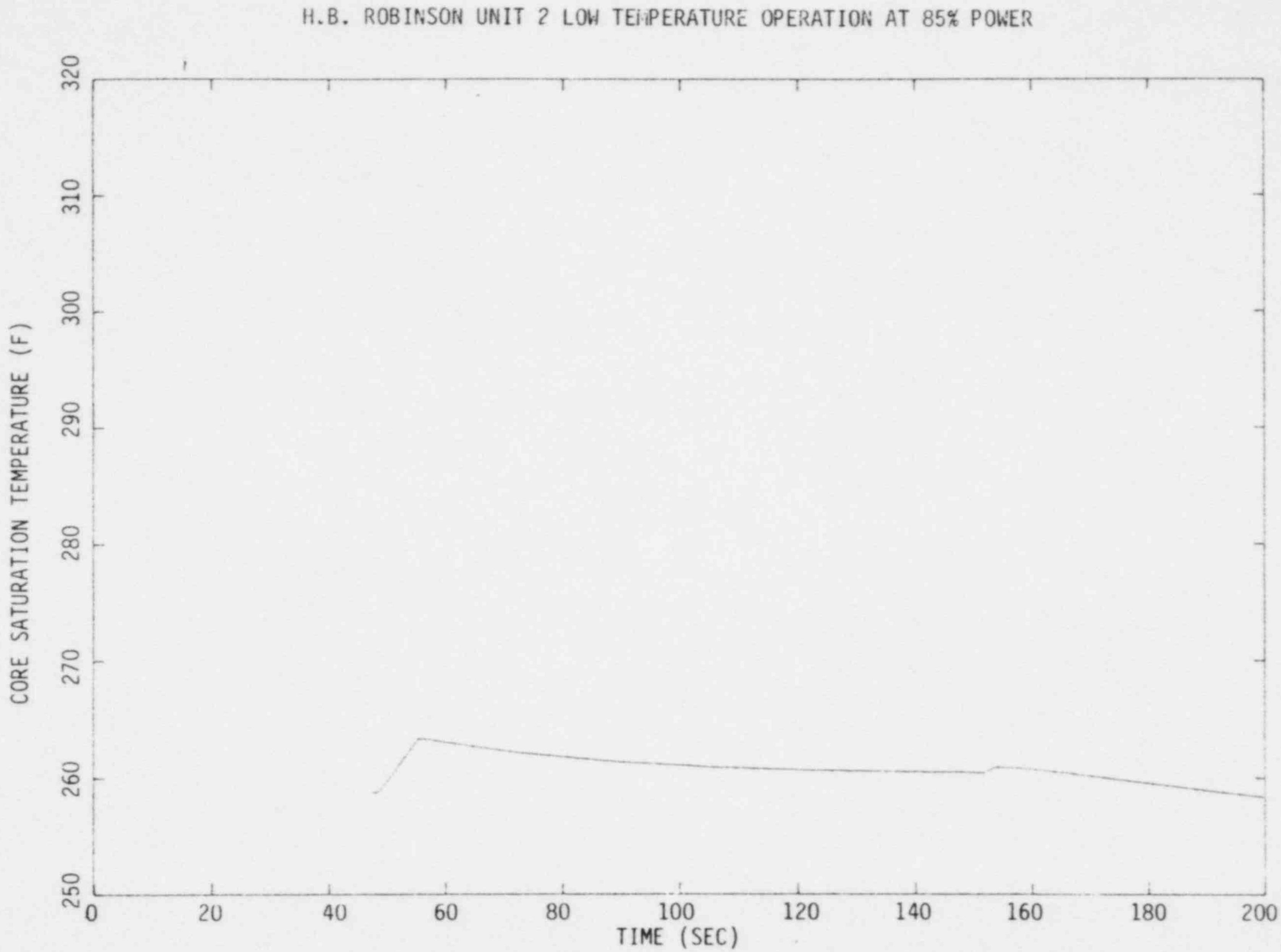

Figure 2.22 Core Saturation Temperature - 0.8 DECLG

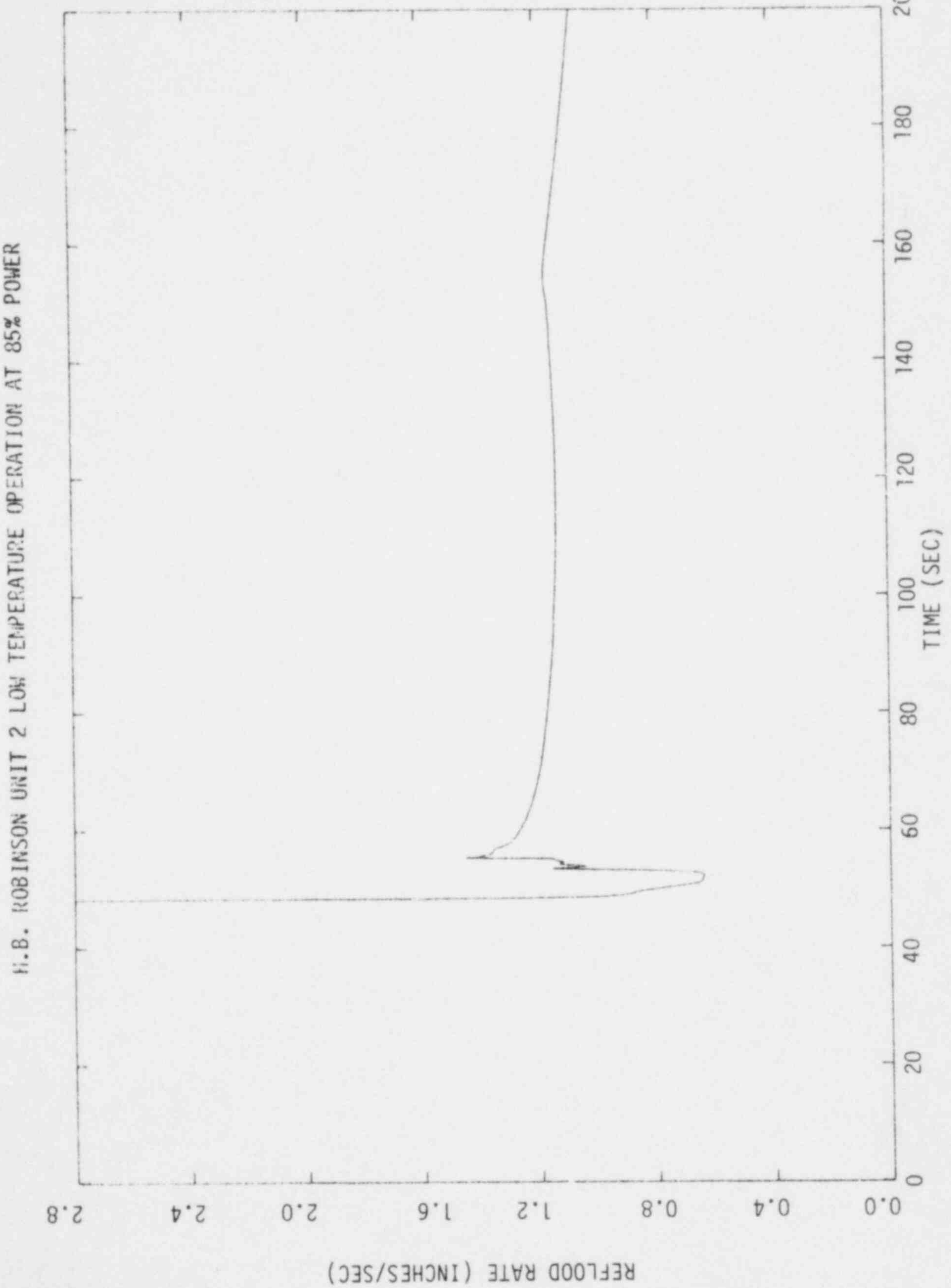

Figure 2.23 Reflood Rate - 0.8 DECLG

H.B. ROBINSON UNIT 2 LOW TEMPERATURE OPERATION AT 85% POWER

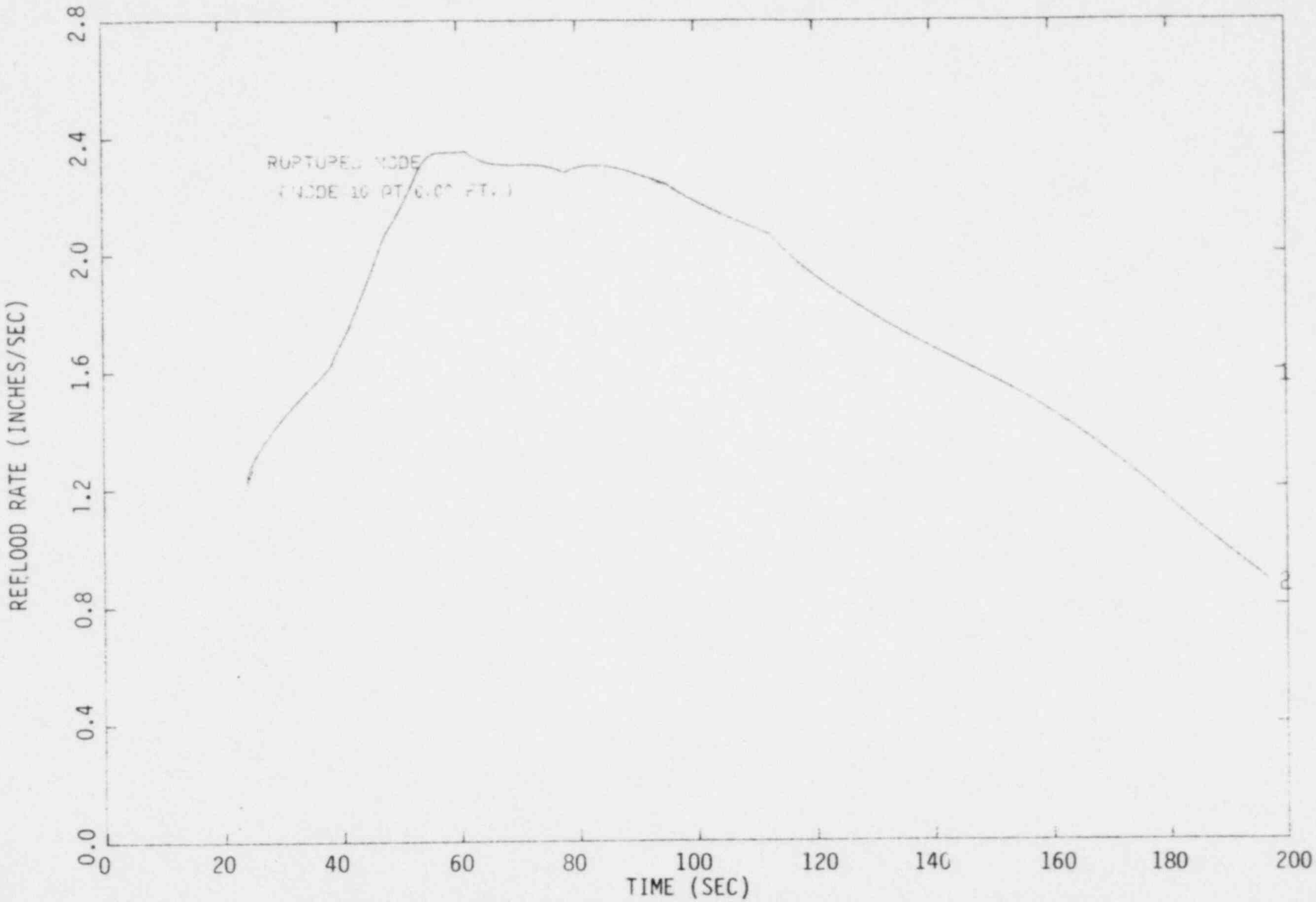

Figure 2.24 Peak Clad Temperature - 0.8 DECLG

H.B. ROBINSON UNIT 2 LOW TEMPERATURE OPERATION AT 85% POWER

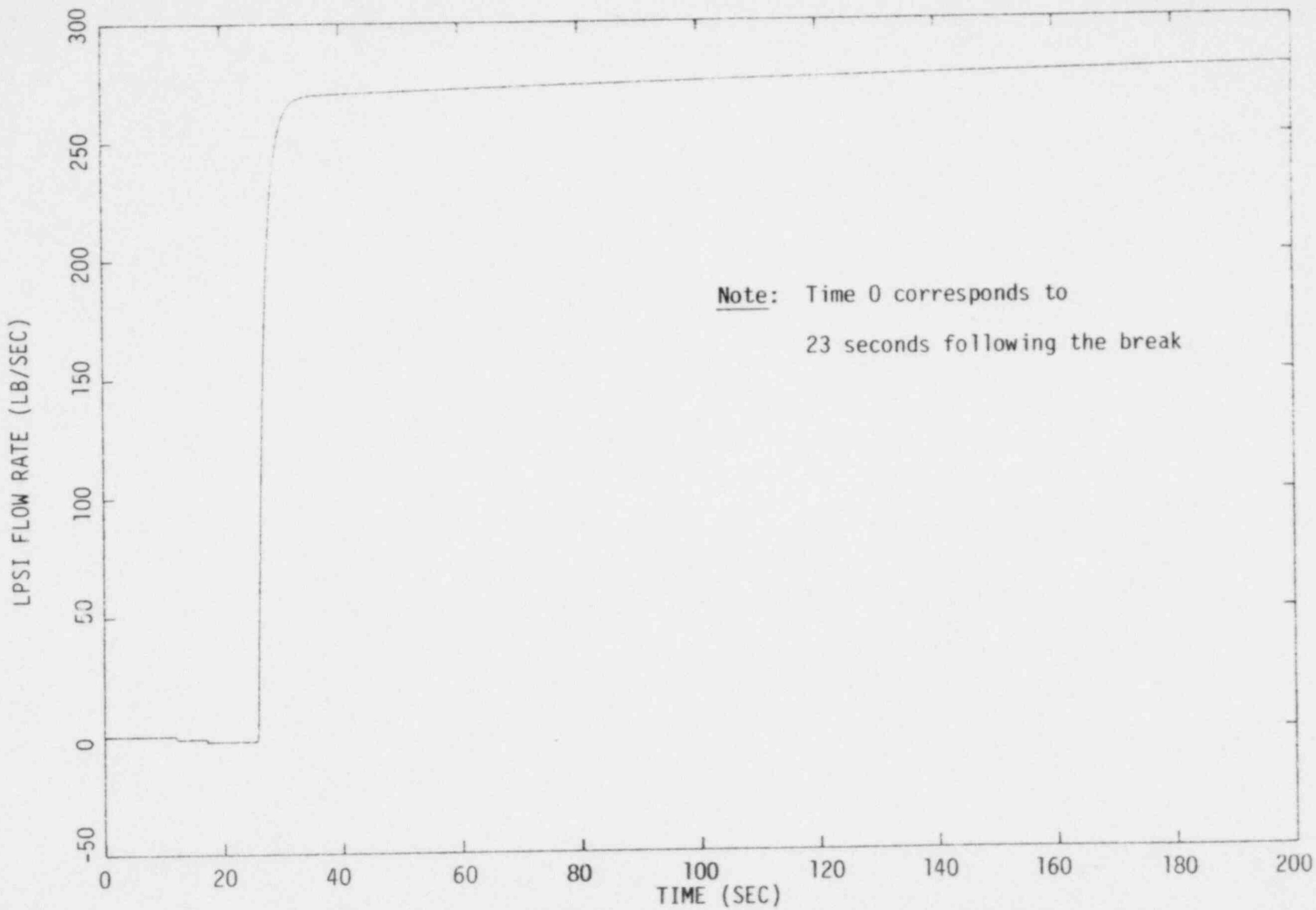

Figure 2.25 LPSI Flow - 0.8 DECLG

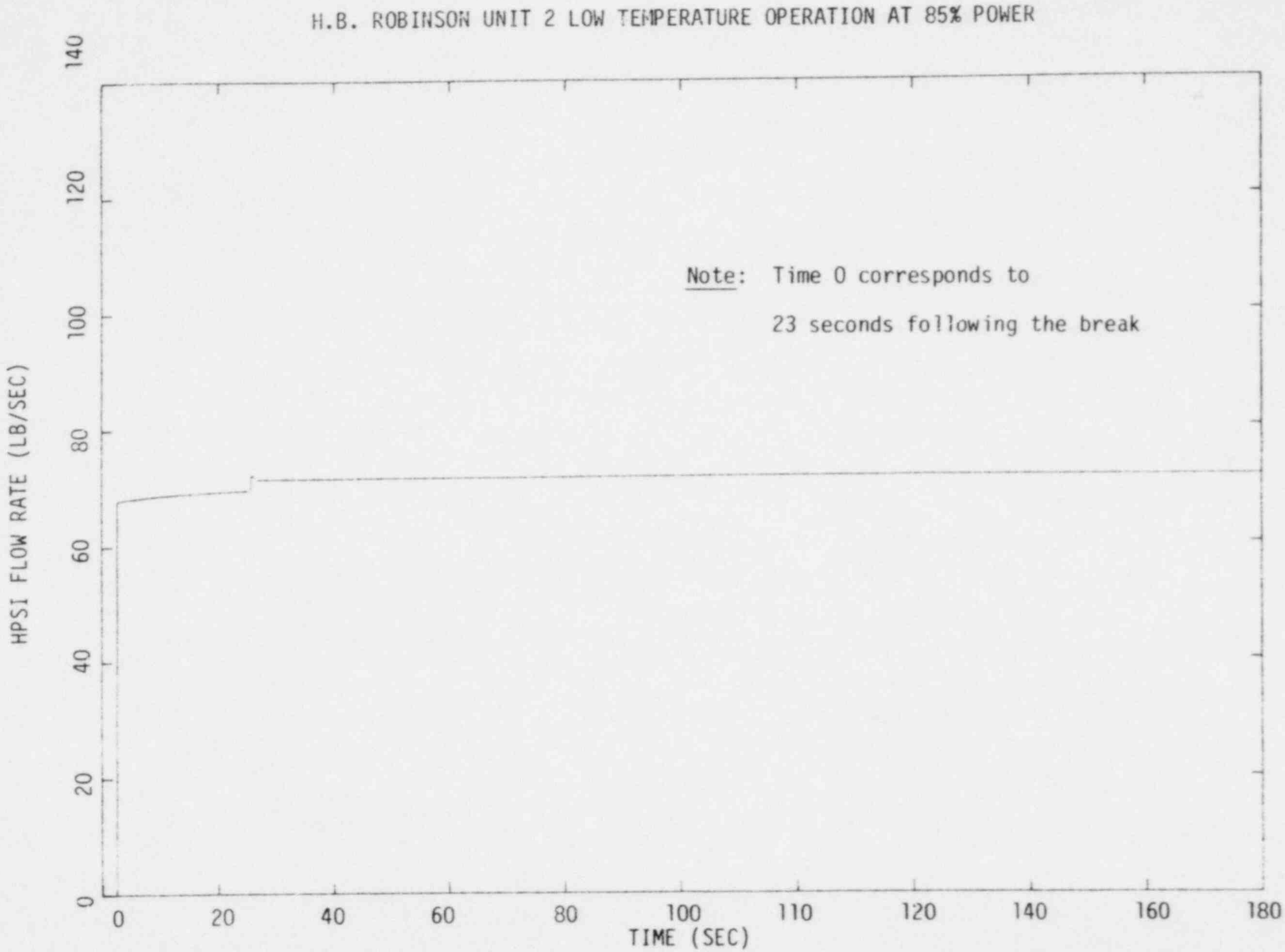

Figure 2.26 HPSI Flow - 0.8 DECLG

H.B. ROBINSON UNIT 2 LOW TEMPERATURE OPERATION AT 85% POWER

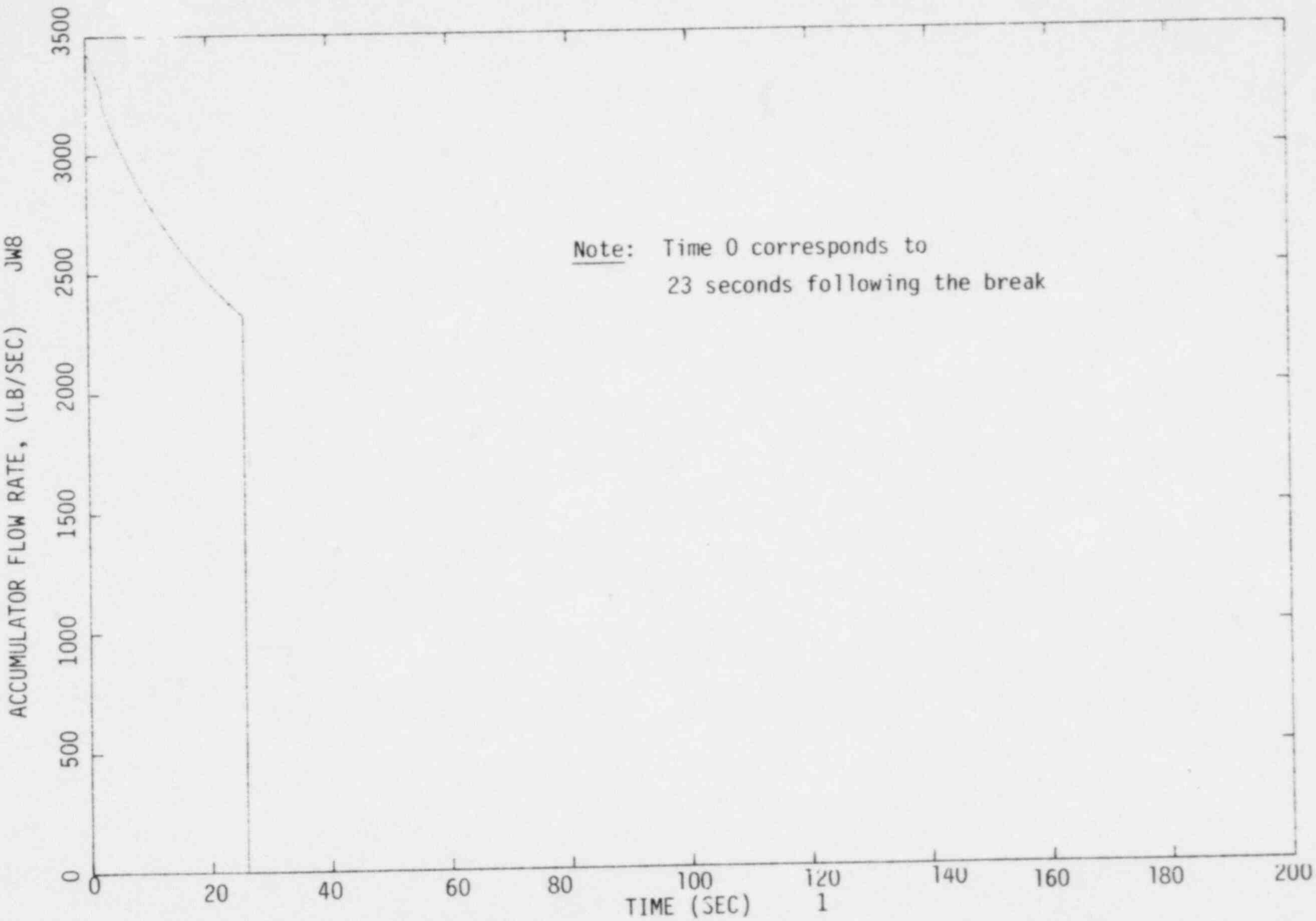

Figure 2.27 Accumulator Flow - 0.8 DECLG

3.0 PLANT TRANSIENT ANALYSIS

3.1 INTRODUCTION AND SUMMARY

This analysis considers the reactor operation of the H.B. Robinson Unit 2 Nuclear Power Plant with a reduced primary coolant temperature and power. With this new T_{AVE} schedule (Figure 3.1), thermal power at H.B. Robinson Unit 2 is limited to 85% of 2300 MWt: The present analysis is in support of H.B. Robinson operation with these new conditions. Results of the analysis will envelope steam generator tube plugging up to 20% (average for three steam generators) and a slightly reduced primary coolant flow associated with this plugging. Results of the analysis for the more limiting transients for H.B. Robinson Unit 2 demonstrate that ENC reload fuel continues to meet plant safety margin requirements during design base events. The transients were analyzed using the Exxon Nuclear plant transient simulation code PTSPWR2[1]. Supporting subchannel analysis was performed using standard ENC methodology[2]. The results of the analyses for the following design base events, as well as the input parameters used to simulate the reactor system, are reported herein.

Event	Incident Class*
1. Uncontrolled Control Rod Withdrawal	
• Fast Rod Withdrawal	II
• Slow Rod Withdrawal	II
2. 3-Pump Coastdown	III
3. Locked Rotor	IV
4. Loss of External Electric Load	II
5. Excess Load	II
6. Large Steam Line Break	IV

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* Consistent with current FSAR incident classification for PWR's.

Events 1 through 5 were initiated from 85% of 2300 MWt, while event 6 was initiated from hot standby. The thermal margin criteria for the Class II and III events is a Minimum Departure from Nucleate Boiling Ratio (MDNBR) $\geq$ 1.30 based on the W-3 correlation(2). In the case of Class IV accidents, some fuel damage is acceptable provided it is confined to a limited number of fuel rods in the core.

The analyses are based on an ENC fueled core using conservative neutronic parameters calculated for the H.B. Robinson Unit 2 core. The results of the analyses are summarized in Table 3.1. The lowest MDNBR for Class II and III events initiated at 1955 MWt was 2.48 for the uncontrolled slow rod withdrawal transient. The locked rotor accident, a Class IV event, was analyzed and the MDNBR was found to be 2.19. The large steam line break resulted in a minimum critical heat flux ratio of 1.19 which is based on the Modified Barnett Critical Heat Flux Correlation. Based on the Modified Barnett Correlation statistics and the fact that high peaking is limited to the vicinity of the stuck control rod it is concluded that the number of rods which potentially might experience boiling transition is very small (<1%).

In summary, the transients initiated from the new full power at reduced T_{AVE} showed increased thermal margins relative to prior analysis and all transients showed acceptable thermal margins.

3.2 CALCULATIONAL METHODS AND INPUT PARAMETERS

The present analysis for the H.B. Robinson plant was performed using the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR2)(1). The PTSPWR2 code is an Exxon Nuclear digital computer program developed to model the behavior of pressurized water reactors under normal and abnormal operating conditions. The model is based on the solution of the basic transient conservation equations for the primary and secondary coolant systems. The transient conduction equation is solved for the fuel rods, and a point kinetics model is used to calculate the core neutronics. The program calculates fluid conditions such as flow, pressure, mass inventory and steam quality, heat flux in the core, reactor power, and reactivity during the transient. Various control and safety system components are included as necessary to analyze postulated events. A hot channel model is included to trace the departure from nucleate boiling (DNB) during transients. The DNB evaluation is based on the hot rod heat flux in the high enthalpy rise subchannel and uses the W-3 correlation(2) to calculate the DNB heat flux. Model features of the PTSPWR2 code are described in detail in Reference 1.

A diagram of the system model used by PTSPWR2 is shown in Figure 3.2 As illustrated, the PTSPWR2 code models the reactor, two independent primary coolant loops (the second loop includes loop # 2 and #3 in the actual reactor configuration) including all major components (pressurizer, pumps) two steam generators (the second steam generator includes steam generators #2 and #3 in the actual reactor configuration), and the steam lines, including all major valves (turbine stop valves, isolation valves, pressure relief valves, etc.)

The present calculations were performed using the NOV76A version of ENC's PTSPWR2 code along with updates. The updates included an update to the pressurizer model and a correction to the mass balance on the secondary side of the steam generator. The pressurizer model update replaced the model used in the prior analysis with one which takes into account adiabatic compression of the steam when the steam is compressed by liquid insurges into the pressurizer. An equilibrium (flashing) model is maintained for liquid outsurges or whenever the pressure of the steam phase drops below saturation with liquid remaining in the pressurizer. With this model realistic increases in primary system pressurization are calculated for present plant transients that involve volumetric expansion of the primary system liquid due to temperature increases. The code correction for the steam generator secondary side mass balance provides a better representation of secondary response for the rod withdrawal, loss of primary flow and loss of load transients. The correction does not impact MDNBR for these transients since the time of MDNBR is generally before the secondary begins to impact primary system response. The mass balance correction is not a factor in the steam line break transients since this area of the code is bypassed in the steam line break calculations.

To ensure conservative predictions of system responses with resulting minimum values for the DNB ratios, conservative asumptions are applied. These assumptions can be grouped into two general categories:

1. Generic assumptions, applicable to all transients, based on steady state offsets.

2. Assumptions which conservatively encompass H.B. Robinson Unit 2 neutronic parameters.

The generic assumptions (Category 1) are applied to all full power transients to account for steady state and instrumentation errors. The initial DNBR conditions are obtained by adding the maximum steady state errors to rated values as follows:

Reactor Power	=	1955 MWt + 2% (39.1 MWt) for calorimetric error.
Inlet Coolant Temperature	=	510 + 4°F for deadband and measurement error.
Primary Coolant System Pressure	=	2250 - 30 psia for steady state fluctuation and measurement errors.

The combination of the above parameters acts to minimize the initial minimum DNB ratio. It is noted that the above steady state errors are not included in the plant system modeling but rather are used to conservatively bound the initial MDNBR. Table 3.2 shows a list of operating parameters used in the analysis.

The trip setpoints incorporated into the PTSPWR2 model for H.B. Robinson Unit 2 are based on the Technical Specification limits and have been revised for the changed system conditions. These limiting trip setpoints with their associated time delays for each trip function are listed in Table 3.3. Overtemperature and overpower ΔT trip functions are detailed in Section 3.6.

The ENC fuel design parameters for H.B. Robinson Unit 2 are summarized in Table 3.4. Table 3.5 lists the neutronics parameter values which conservatively bound the H.B. Robinson Unit 2 core for both the beginning and end of cycle. A design axial power profile with a peaking factor F_Z = 1.55 was used in the analysis. This profile is shown in Figure 3.3. The scram reactivity curve used in the analysis is shown in Figure 3.4.

The assumptions in Category 2 refer to the reactivity feedback effects from moderator temperature changes and Doppler broadening. For full power transients, a 1.25 multiplier is applied to the moderator temperature coefficient. An attenuation factor of 0.8 or a magnification factor of 1.2 has been applied to the Doppler feedback coefficient, depending on which factor results in the worst case. Table 3.6 contains the multipliers used and the resulting moderator and Doppler feedback coefficients applicable for full power transients.

3.3 OVERTEMPERATURE AND OVERPOWER ΔT SETPOINTS

A major aspect of the present study has been the verification of adequately conservative safety system setpoints for plant operation at reduced primary coolant temperature and reactor power. For reduced temperature operation, the principal setpoint changes were to the overtemperature ΔT and overpower ΔT trip functions. In both cases, the essential change has been to reduce the reference T_{AVE} parameter, T', in order that the operating margin to trip would not increase for the new reduced temperature conditions. The increased MDNBR (increased thermal margin) results for the transients analyzed in this report at reduced temperature conditions substantiate the setpoints for reduced temperature operation. A listing of the overtemperature and overpower ΔT trip functions can be found in Section 3.6.

3.4 TRANSIENT ANALYSIS

3.4.1 Initial Conditions

Evaluation of the effects of the proposed reduced temperature and reactor power operation on steady state thermal margin was performed in accordance with standard ENC thermal hydraulic calculational methodology described and referenced in (2). H.B. Robinson Unit 2 cycle 9 has an all ENC fuel core. The reduced temperature calculations have been performed for a T_{AVE} of 537.9^{o}F, and a thermal power equal to 1955 MWt (85% of rated 2300 MWt) versus prior analyses at 575.4^{o}F and 2300 MWt. System pressure remains unchanged at 2250 psia. For the reduced temperature conditions with 1955 MWt, the initial steady state DNBR is conservatively calculated to be equal to or greater than 3.13. This represents a significant increase in initial MDNBR when compared with the previous analysis at 2300 MWt[(3)] in which the initial MDNBR was 1.87. It is noted that both the reduced temperature and power contribute to the improved initial DNBR. A comparison of the most limiting transients for the reduced temperature and reactor power conditions illustrates that this gain in initial DNBR more than offsets any changes in DNBR during the transients for reduced temperature operation. The improved MDNBR for the transients supports the revised core safety system setpoints. The analysis results presented include the most limiting rod withdrawals at full power conditions, and the most limiting loss of flow accidents which have previously been shown as the worst accidents relative to thermal margin. The large steam line break and excess load transients demonstrate the characteristics of reactor coolant cooldown incidents. The loss of load completes the transients analyzed.

3.4.2 Uncontrolled Rod Withdrawal

The withdrawal of a control rod bank adds reactivity to the reactor core, causing both the power level and the core heat flux to increase. Since the heat extraction from the steam generator remains relatively constant, there is an increase in primary coolant temperature. Unless terminated by manual or automatic action, this power mismatch and the resultant coolant temperature rise could eventually result in a DNB ratio of less than 1.3. While the inadvertent withdrawal of a control rod bank is unlikely, the reactor protection system is designed to terminate such a transient while maintaining an adequate margin to DNB.

In this incident, the reactor may be tripped by the overtemperature ΔT function, by the nuclear overpower function, or by other reactor protective safety system setpoints. The analysis presented here confirms the adequacy of the setpoints protecting the plant. Both a fast rod withdrawal and a slow rod withdrawal were analyzed from an initial power level of 1955 MWt. Beginning-of-cycle kinetics coefficients were used with an appropriate multiplier applied to the Doppler coefficient (see Table 3.6.)

Figures 3.5 to 3.11 show plant responses for a fast rod withdrawal (5.625×10^{-4} $\Delta\rho$/sec) from 1955 MWt. A nuclear overpower trip (121% setpoint) occurs at 2.70 seconds. The DNB ratio drops from an initial value of 3.13 to 2.82. Pressure increases to a maximum of 2310 psia with core average temperature increasing by less than $3^{o}F$.

The system responses to a slow rod withdrawal of 2.5×10^{-5} $\Delta\rho$/sec are depicted in Figures 3.12 to 3.18. The nuclear overpower trip setpoint (121%) is reached at 44.01 seconds, and the minimum DNB ratio during the transient is 2.48. The slow rod withdrawal transient is terminated by an overtemperature ΔT trip when the nuclear overpower trip mechanism is defeated purposely in order to show that the overtemperature ΔT trip will function properly. The overtemperature ΔT trip setpoint is reached at 44.35 sec, scramming the reactor after a 2.3 second delay. The minimum DNB ratio during the transient is then 2.43. At slower rates of reactivity insertion, the overtemperature ΔT trip potentially could occur prior to the high flux trip. For the rod withdrawal accidents at reduced coolant temperature condition, power peaking increases about the withdrawn rod are not expected to be more than a few percent and these are more than offset by the significantly improved MDNBR results presented here.

3.4.3 Rod Withdrawal from Hot Zero Power

This transient is analyzed in two steps. First, the PTSPWR2(1) code is used to calculate the reactor conditions during the transient. The reactor is assumed to be at a hot zero power condition (HZP, power=1.0×10^{-13} MWt at t=0) with a uniform core average temperature of 530^{o}F. A moderator temperature coefficient of $+2.5 \times 10^{-5}$ $\Delta\rho/^{o}F$, a Doppler coefficient of -1.0×10^{-5} $\Delta\rho/^{o}F$ and a reactivity insertion rate of 60×10^{-5} $\Delta\rho$/sec were used in the calculations. The peak core average heat flux from PTSPWR2 calculations was then used to establish DNBR for the limiting rod in the core. Standard ENC methodology(2) for core thermal margin analysis was used to calculate this MDNBR. The resulting MDNBR for the control rod withdrawal transient initiated from hot zero power was calculated to be 2.11 which is significantly greater than 1.30. Figures 3.19 and 3.20 show the plant responses during the transient.

3.4.4 3-Pump Coastdown

The 3-pump coastdown transient is postulated to occur as a result of a loss of electric power to the primary coolant pumps. The transient results in an increase in coolant temperature which, in combination with the reduced flow, reduces the margin to DNB. Only the most severe case has been analyzed. This case is the loss of power to all three pumps when the reactor system is operating at 1955 MWt. Beginning-of-cycle values of kinetics coefficients are assumed. For conservatism, a multiplier of 0.8 was applied to the Doppler coefficient. The loss of power to all pumps will result in a reactor trip due to either under-voltage or under-frequency at the bus. For conservatism, however, the trip was taken to be on a low flow signal. This allows a further flow reduction at full power, and a more conservative calculation of margin to DNB.

Figures 3.21 to 3.27 depict plant responses after the loss of all three pumps. A reactor trip occurs at 2.63 seconds. A minimum DNB ratio of 2.58 is reached 3.50 seconds after the beginning of coastdown. System pressure peaks at 2314 psia.

3.4.5 Locked Rotor

In the unlikely event of a seizure of a primary coolant pump, flow through the core is drastically reduced. The reactor is tripped by the resulting low flow signal. The coolant enthalpy rises, decreasing the margin to DNB. The locked rotor transient was analyzed assuming three loop operation with instantaneous seizure of one pump from 1955 MWt. The feedwater pumps were assumed to trip with the reactor. Beginning-of-cycle kinetics coefficients were used as the BOC moderator coefficient is the most adverse. A 0.8 multiplier was applied to the Doppler coefficient.

The responses for the locked rotor transient are shown in Figures 3.28 to 3.34. The reactor is scrammed at 1.03 seconds by a low flow signal. Core average temperature increases by 10.7^{o}F with system pressure reaching 2321 psia, well below peak pressure limit of 2750 psia. The DNB ratio in the analysis reaches a minimum of 2.19 after a 5% MDNBR penalty is imposed to account for future potential differences in the degree of plugging between steam generators. The 5% penalty accounts for the case in which the seized rotor loop is the one with the least tube plugging (highest flow) since the flow impact would be highest for this case. This penalty is considered more than adequate to cover a 10% difference in tube plugging between loops.

3.4.6 Loss of Load

Loss of load involves plant behavior after a trip of the turbine-generator without a direct reactor trip. This transient has been reanalyzed to ascertain the MDNBR for reduced temperature operation. Since thermal power is 15% less than in previous analyses, reanalysis relative to peak pressurization is not considered necessary. The transient responses for this event are evaluated from 1955 MWt with the most severe assumptions; namely, loss of load at BOC with a positive moderator coefficient, 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 are neglected. The steam line safety valves are assumed to operate. For conservatism, a multiplier of 0.8 was applied to the Doppler coefficient.

Figures 3.35 to 3.41 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 1051 psia at 19.89 seconds. The steam line safety valve setpoint is not reached. The reactor is tripped at 12.46 seconds on high pressure after 1.0 sec delay. The peak primary pressure was 2460 psia. The average primary coolant temperature increases by less than 23°F. The lowest value for the minimum DNB ratio during the transient is 2.91 which occurs at 12.50 seconds.

3.4.7 Excess Load

In an excess load incident, an increase in steam flow through the steam generator causes a power mismatch between reactor power and steam generator demand. For this case, a 10% step increase in rated turbine load is analyzed at 1955 MWt. Since excess load is a reactor coolant cooldown transient the excess load incident is analyzed at EOC with no automatic control assumed. Figures 3.42 to 3.48 illustrate plant responses to this transient. Core power reaches 2114.8 MWt after 41.89 seconds as power level rises due to the large negative moderator coefficient. The lowest MDNBR during the transient is 2.79 which occurs at 51 sec. Core conditions achieve a new steady state without a reactor scram.

3.4.8 Steam Line Break

The break of a steam pipe (or safety valve failure) results in a sharp reduction in steam inventory in a steam generator. The resulting pressure decrease causes an energy demand from the primary coolant which reduces coolant temperature and pressure. With a negative moderator temperature coefficient, this causes reactivity insertion into the core which could, under pessimistic circumstances, lead to criticality and core damage if unchecked.

The steam line break transient is simulated with the PTSPWR2 plant transient simulation code. As a worst case, the steam line break is assumed to occur at hot zero power conditions corresponding to a core average temperature of 530°F. At this time, the steam generator secondary side water inventory is at a maximum, prolonging the duration and increasing the magnitude of the primary loop cooldown. For conservatism, the most reactive control rod is assumed to be stuck out of the core when evaluating the shutdown capability of the control rods. The reduction in primary to secondary heat transfer area occasioned by steam generator tube plugging has been conservatively accounted for by assuming that the loop in which the steam line break occurs is that loop having the least number of plugged tubes (8% of tubes plugged). This assumption maximizes the heat transfer rate from the primary coolant to the broken loop, and thus maximizes the moderator cooldown and the magnitude of the return to power.

The reactivity as a function of core average temperature and the variation of local reactivity (near the stuck rod) as a function of core power used in this analysis are shown in Figures 3.49 and 3.50. A shutdown reactivity of 1.77% $\Delta\rho$ was assumed.

The initial steam flow is calculated from the Moody curve for critical flow of saturated steam[5]. It is assumed that the two intact steam generators also blow down to the containment until the closure of the main steam isolation valves. The initial break flow is

7229 lb/s per loop.

Figures 3.51 to 3.56 depict the transient responses for the worst steam line break: a large break inside the containment with outside power available. The core returns to power at 7.5 seconds, somewhat earlier than in the reference case(4). The larger initial break flow used in this analysis results in a faster cooldown of the moderator than observed in the reference case. The consequent higher rate of reactivity insertion causes a faster return to power. Boron reaches the core at 43. seconds, terminating the power increase. The safety injection actuation signal occurred at 10 seconds on a low pressurizer pressure signal. The time required to sweep the safety injection lines clear of residual water is implicitly accounted for in the calculation. The core average heat flux peaks shortly after the power increase is terminated. Maximum heat flux is 40% of the rated value of 179,218 BTU/hr-ft^2.†

The minimum critical heat flux ratio is calculated to be 1.19 at the time of peak core average heat flux. The Modified Barnett critical heat flux correlation(6) was employed in the calculation in conjunction with an overall hot channel factor of 13.7 and core conditions corresponding to the time of peak heat flux. This is judged to be an acceptable outcome for the steam line break event, with few if any fuel rods experiencing boiling transition.(<1%)

3.5 DISCUSSION OF TRANSIENT ANALYSIS RESULTS

The transient analyses for the H.B. Robinson Unit 2 Nuclear Power Plant for conditions of reduced primary coolant temperature and reactor

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† The rated value of 179,218 BTU/hr-ft^2 corresponds to a power rating of 2300 MWt.

power all show adequate margin to safety limits. The neutronics data used in this analysis are consistent with or conservative with respect to the previous analysis[4]. A comparison of operating parameters used in the present analyses with those used in the previous analysis[4] is shown in Table 3.7. For reduced primary coolant temperature and reactor power the limiting transient analyses reported in sub-section 3.4 all showed increased margins when compared to the previous analysis[4].

Several additional transients previously analyzed[4] were not reanalyzed. These included:

- startup of an inactive loop
- loss of feedwater
- RCCA drop
- loss of A.C. power
- chemical and volume control system malfunction
- reduction in feedwater enthalpy accident.

They are not considered to be limiting transients and should remain non-limiting for reduced temperature and reactor power because their rate of reactivity addition is enveloped by the more limiting transients of sub-section 3.4, and the steady state MDNBR increases with reduced temperature and reactor power.

3.6 REVISED SAFETY SYSTEM SETPOINTS FOR OPERATION AT REDUCED TEMPERATURE AND POWER

The current safety system protective settings for reduced temperature operation are:

(a) High flux, power range $\leq$109% (of 1955 MWt)

(b) High Pressurizer Pressurizer Pressure $\leq$2385 psig

(c) Low Pressurizer Pressure $\geq$ 1835 psig

(d) Overtemperature ΔT

$$\leq \Delta T_o [K_1 - K_2(T-T') + K_3(P-P') - f(\Delta I)]$$

where

ΔT_o = indicated ΔT at 85% of 2300 MWt, ^oF

T = average temperature, ^oF

T' = 537.9^oF

P = pressurizer pressure, psig

P' = 2235 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, where q_t and q_b are the percent power in the top and bottom half of the core respectively, and $q_t + q_b$ is the total core power in percent of rated power such that:

(i) for $q_t - q_b$ within -17, +12 percent,

$f(\Delta I) = 0$. 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.

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds +12 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.4% of rated power.

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds -17 percent, the ΔT setpoint shall be automatically reduced by an equivalent of 2.4% of rated power.

(e) Overpower ΔT

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

where

ΔT_o = indicated ΔT at 85% of 2300 MWt, ^oF

T = average temperature, ^oF

T' = 537.9^oF

K_4 = 1.07

K_5 = 0.0 for decreasing average temperature

0.2 seconds per ^oF for increasing average temperature

K_6 = 0.002235 for $T \geq T'$

0.0 for $T < T'$

$f(\Delta I)$ = as defined in (d) above

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

(g) Low reactor coolant pump frequency $\geq$ 57.5 H_z.

(h) High pressurizer water level $\leq$ 92% of span.

(i) Low low steam generator water level $\geq$ 14% of narrow range instrument span

(j) Startup protection - high flux $\leq$ 25% rated power.

The present analyses used setpoints which are consistent with or conservative with respect to the above. It is important to note that the high flux trip (item a) as a percent of power is not changed. However, the absolute trip function has been changed since full power has been reduced from 2300 MWt to 1955 MWt. Based on the analyses, the limiting accidents being presented in this report verified that the overtemperature and overpower ΔT setpoints are adequate.

With the effects of reduced coolant temperature and reactor power included, the safety system setpoints are confirmed to provide conservative protection for the plant.

Table 3.1 Summary of Results

Transient (Class)	Maximum Power Level (MWt)	Maximum Core Average Heat Flux (Btu/hr-ft^2)	Maximum Pressurizer Pressure (psia)	MDNBR (W-3)
Initial Conditions for Transients	1955.0	152,336	2250.0	3.13
Uncontrolled Rod Withdrawal (II) @ 5.625×10^{-4} $\Delta\rho$/sec	2480.6	165,871	2310	2.82
Uncontrolled Rod Withdrawal (II) @ 2.5×10^{-5} $\Delta\rho$/sec	2369.8	180,322	2353	2.48
Loss of Flow - (III) 3-Pump Coastdown	1988.3	152,335	2314	2.58
Loss of Flow - (IV) Locked Rotor	2009.1	152,335	2321	2.19*
Loss of Load (II)	2090.9	154,968	2460	2.91
Excess Load (II)	2114.8	164,742	2251	2.79
Large Steam Break (IV)	945.9	71,981	**	1.19***

* A 5% penalty was applied to account for less steam generator plugging in the locked rotor loop.
** Pressure decreases from initial value.
*** Calculated with the Modified Barnett Critical Heat Flux Correlation.

Table 3.2 Parameter Values Used in PTSPWR2 Analysis of H. B. Robinson Unit 2

	Analysis input Value
Core	
Total Core Heat Output, MWt	1955.
Heat Generated in Fuel, %	97.4
System Pressure, psia	2250.
Hot Channel Factors	
Total Peaking Factor F_Q^T	2.55
Enthalpy Rise Factor, $F_{\Delta H}^N$	1.60
Total Core Flow, 10^6 lb/hr	101.
Effective Core Flow, 10^6 lb/hr	96.4
Coolant Average (Vessel) Temperature, ^oF	537.9
Heat Transfer	
Average Heat Flux, BTU/hr-ft^2	152,336
Steam Generators	
Total Steam Flow, 10^6 lb/hr	8.16
Steam Temperature, ^oF	478.5
Feedwater Temperature, ^oF	408.0

Table 3.3 H. B. Robinson Unit 2 Trip Setpoints

	Setpoint	Used in Analysis	Delay Time
High Neutron Flux	109%	121%	0.5 sec.
Low Reactor Coolant Flow	90%	87%	1.0 sec.
High Pressurizer Pressure	2400 psia	2400 psia	1.0 sec.
Overtemperature ΔT*			2.3 sec.

* The overtemperature ΔT trip is a function of pressurizer pressure, coolant average temperature, and axial offset. The T' and P' setpoints are contained within the function relationship. This is discussed in Section 3.6.

Table 3.4 Exxon Nuclear Reload for H. B. Robinson Unit 2
Fuel Design Parameters

Fuel Radius	0.17825	Inch
Inner Clad Radius	0.1820	Inch
Outer Clad Radius	0.2120	Inch
Active Length	144.0	Inch
Number of Fuel Rods in Core	32,028	

Table 3.5 H. B. Robinson Unit 2 Kinetic Parameters

Symbol	Parameter	Value: Beginning-of-Cycle	Value: End-of-Cycle
α_M	Moderator Coefficient ($\Delta\rho/^{o}F \times 10^{-5}$)	+2.0	-35.0
α_D	Doppler Coefficient ($\Delta\rho/^{o}F \times 10^{-5}$)	-1.0	-1.7
α_p	Pressure Coefficient ($\Delta\rho$/psia $\times 10^{-6}$)	-0.2*	+4.0*
α_V	Moderator Density Coefficient ($\%\Delta\rho$)/(g/cm^3)	-1.8*	+31.5*
α_B	Boron Worth Coefficient ($\Delta\rho$/ppm $\times 10^{-5}$)	-7.0	-9.0
β_{eff}	Delayed Neutron Fraction (%)	0.700	0.510
α_{CRC}	Total Rod Worth, N-1, ($\%\Delta\rho$)	-4.0	-4.0

* For the limiting transients being analyzed, they are assumed to be zero for conservatism.

Table 3.6 Moderator and Doppler Coefficients

Transient	Moderator Feedback Multiplier	Resulting Coefficient $\Delta\rho/^{o}F \times 10^{-5}$	Doppler Feedback Multiplier	Resulting Coefficient $\Delta\rho/^{o}F \times 10^{-5}$
Fast Rod Withdrawal	1.25	+2.5	1.2	-1.2
Slow Rod Withdrawal	1.25	+2.5	0.8	-0.8
3-Pump Coastdown	1.25	+2.5	0.8	-0.8
Locked Rotor	1.25	+2.5	0.8	-0.8
Loss of Load	1.25	+2.5	0.8	-0.8
Excess Load	1.25	-43.8*	0.8	-1.36*
Large Steam Line Break	---	***	---	**

* Excess load transient is more limiting at EOC due to large negative α_M.

** See Figure 3.50

*** See Figure 3.49

Table 3.7 Comparison of Operating Parameters for H.B. Robinson Unit 2

	Nominal Values	
	This Analysis	Previous Analysis[4]
Reactor Power, MWt	1955	2300
Primary Coolant Flow Rate, 10^6 lb/hr	101.0	101.5
Reactor Coolant Pressure, psia	2250	2250
Reactor Coolant Temperature, ^oF		
Vessel Outlet	564.4	604.7
Vessel Average	537.9	575.4
ΔT	54.4	58.5
Steam Generators		
Steam Temperature, ^oF	478.5	523.0
Steam Pressure, psia	558.0	850.0
Steam Flow, 10^6 lb/hr total	8.16	10.07
Feed Temperature, ^oF	408	441.7
Zero Load Temperature, ^oF	530	547

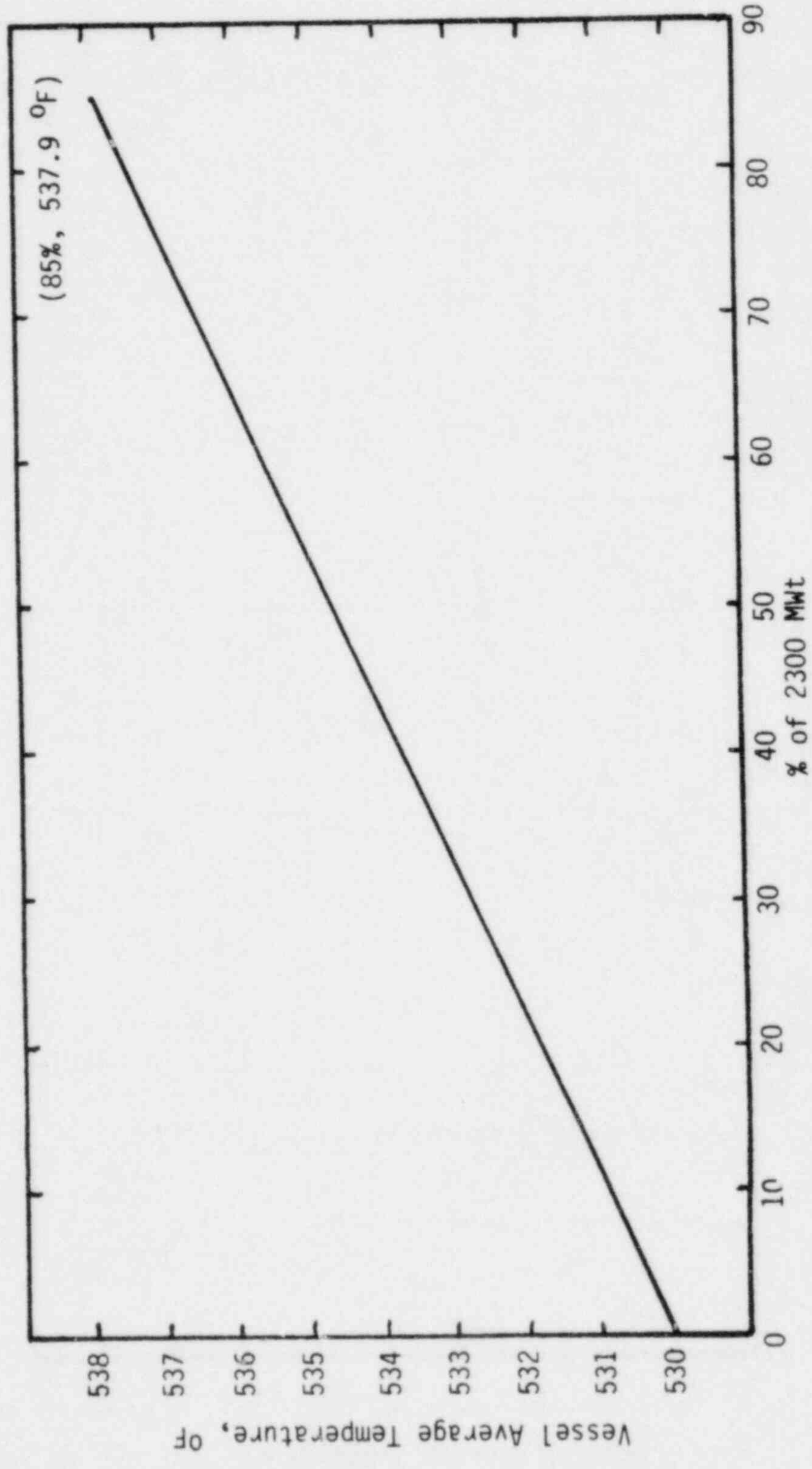

Figure 3.1 Average primary system temperature vs. power for H. B. Robinson Unit 2 reduced T_{AVE} and reduced power (85%) operation

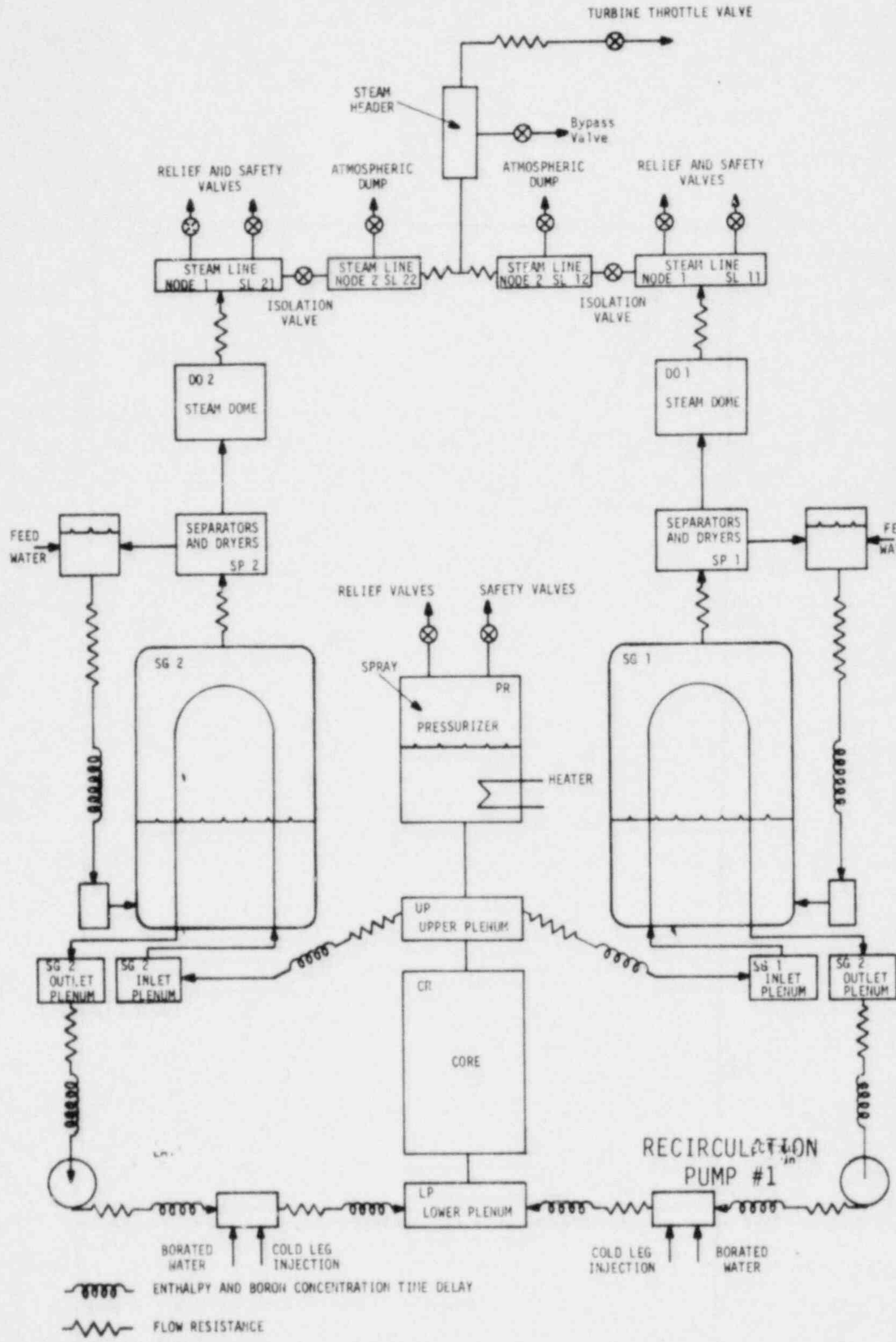

Figure 3.2 PTSPWR2 System Model

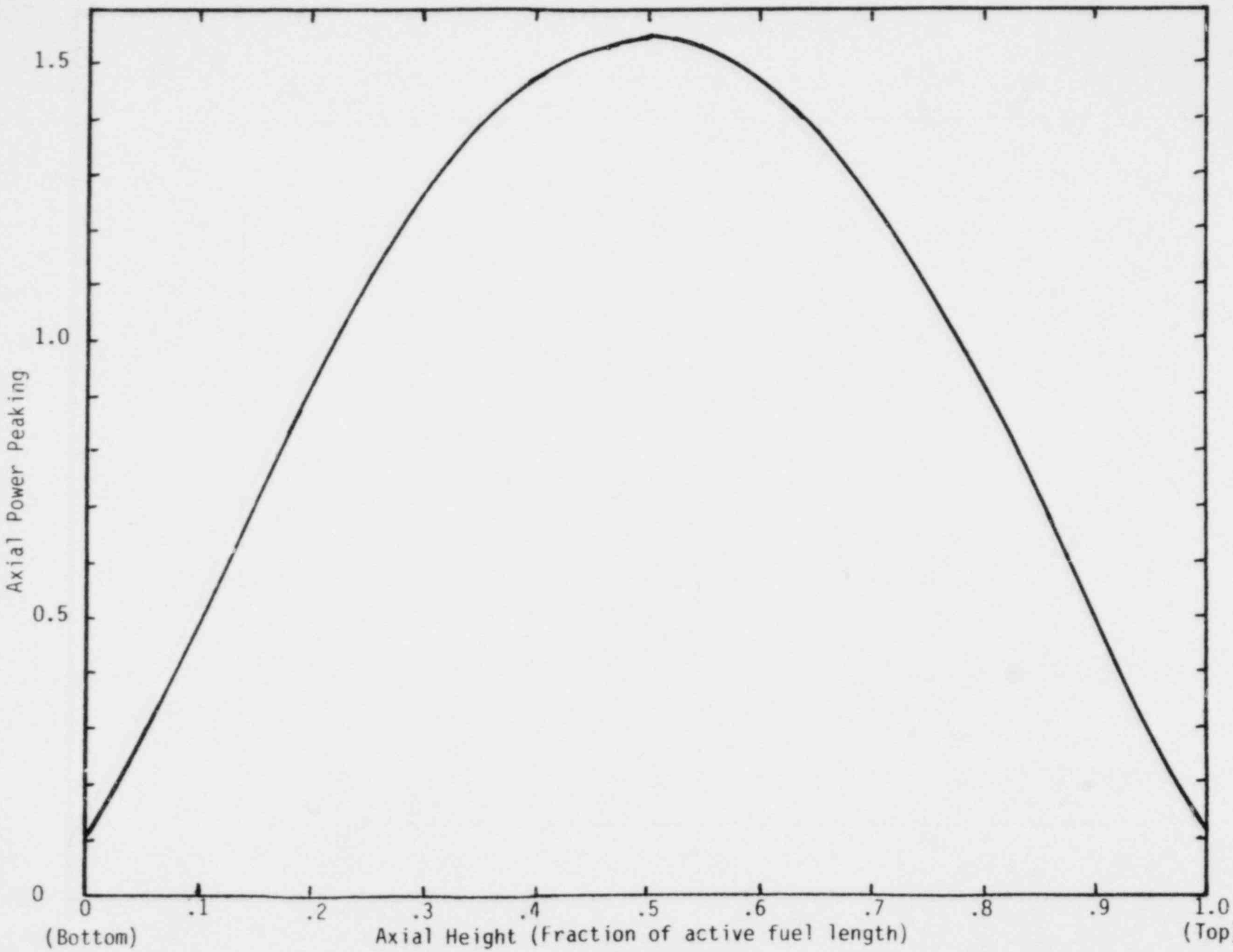

Figure 3.3 Axial power profile used in transient analysis of H.B. Robinson Unit 2

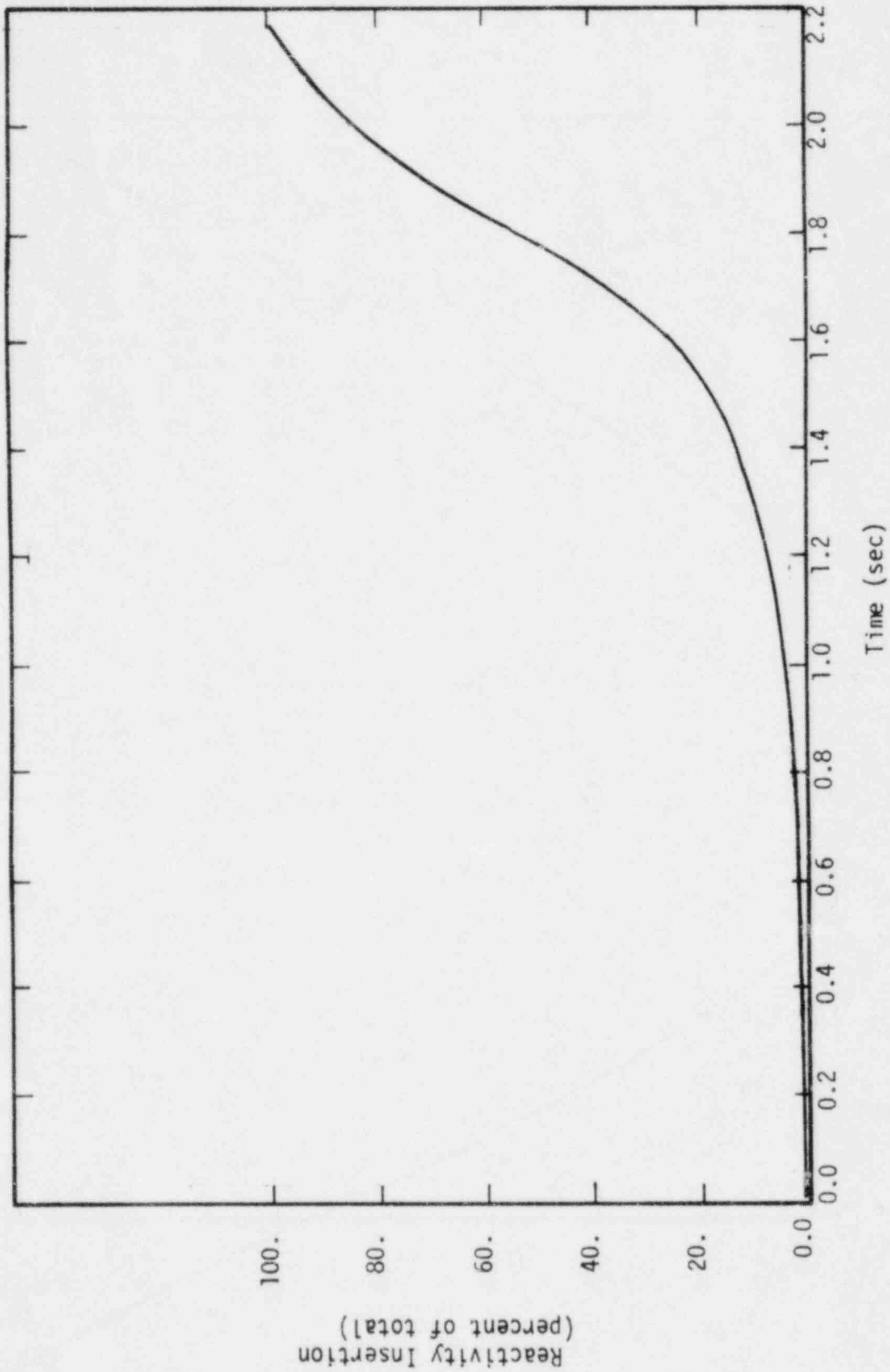

Figure 3.4 H. B. Robinson Unit 2 Scram Curve

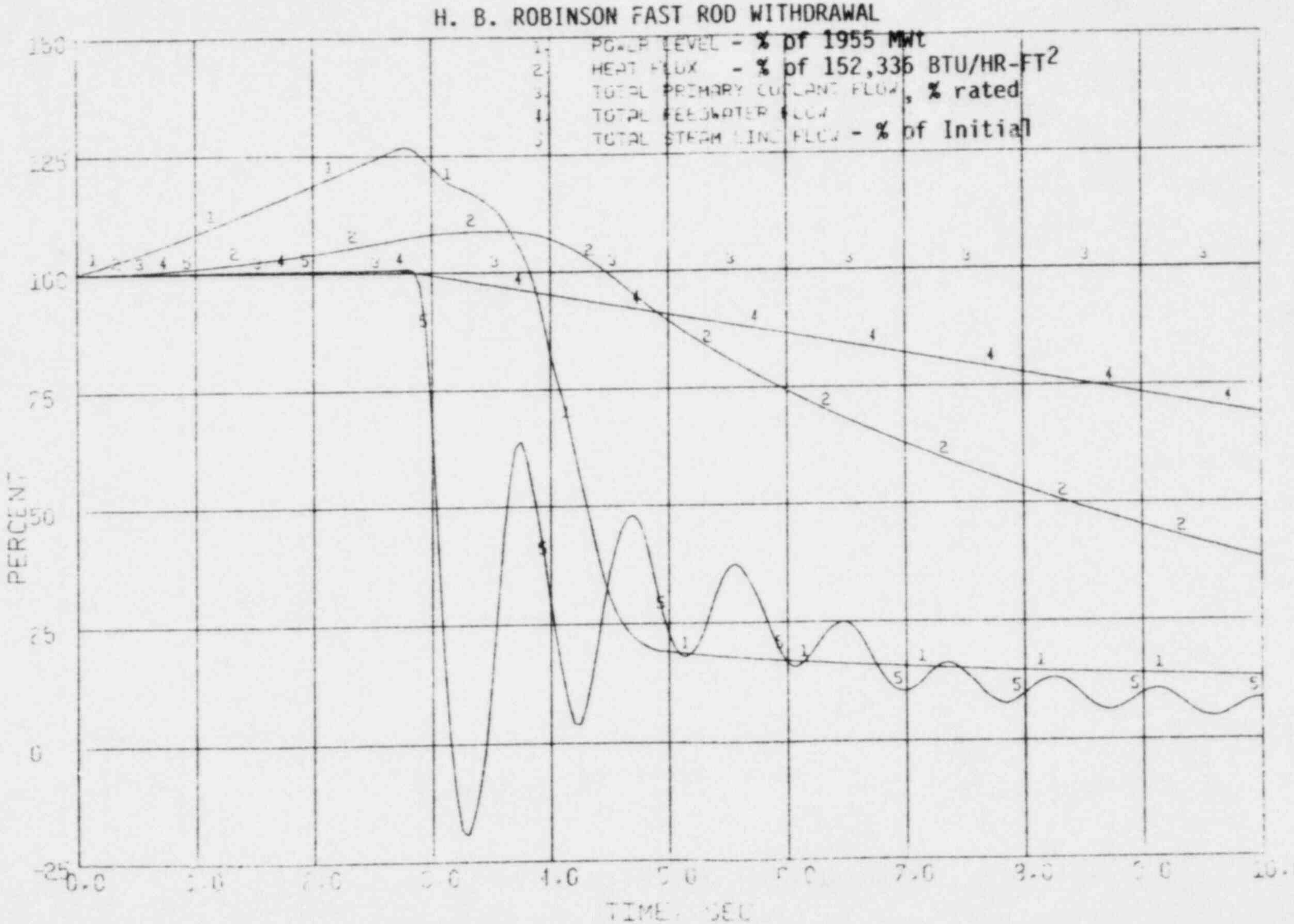

Figure 3.5 Power, heat flux and system flows for fast control rod withdrawal

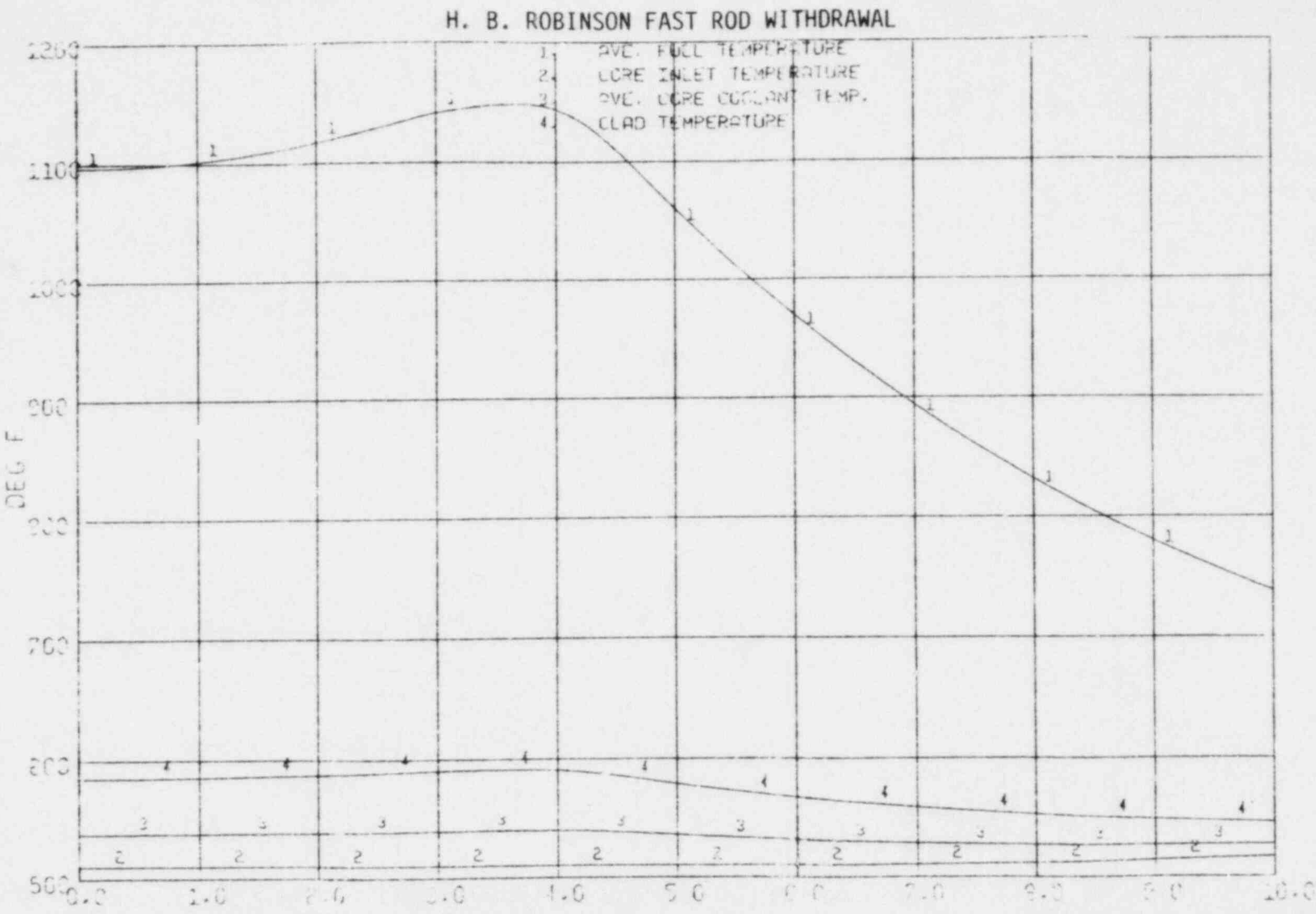

Figure 3.6 Core temperature responses for fast control rod withdrawal

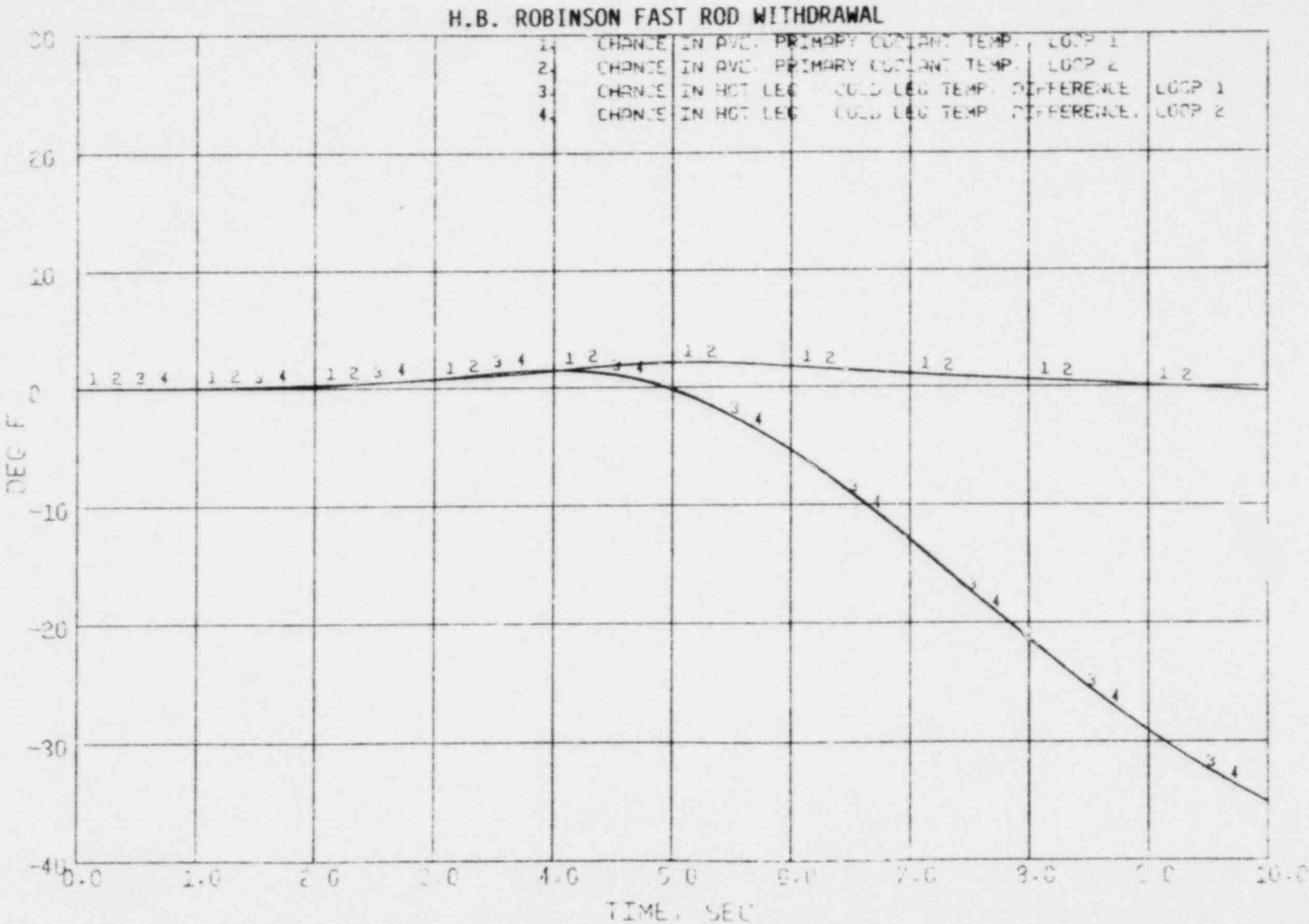

Figure 3.7 Primary loop temperature changes for fast control rod withdrawal

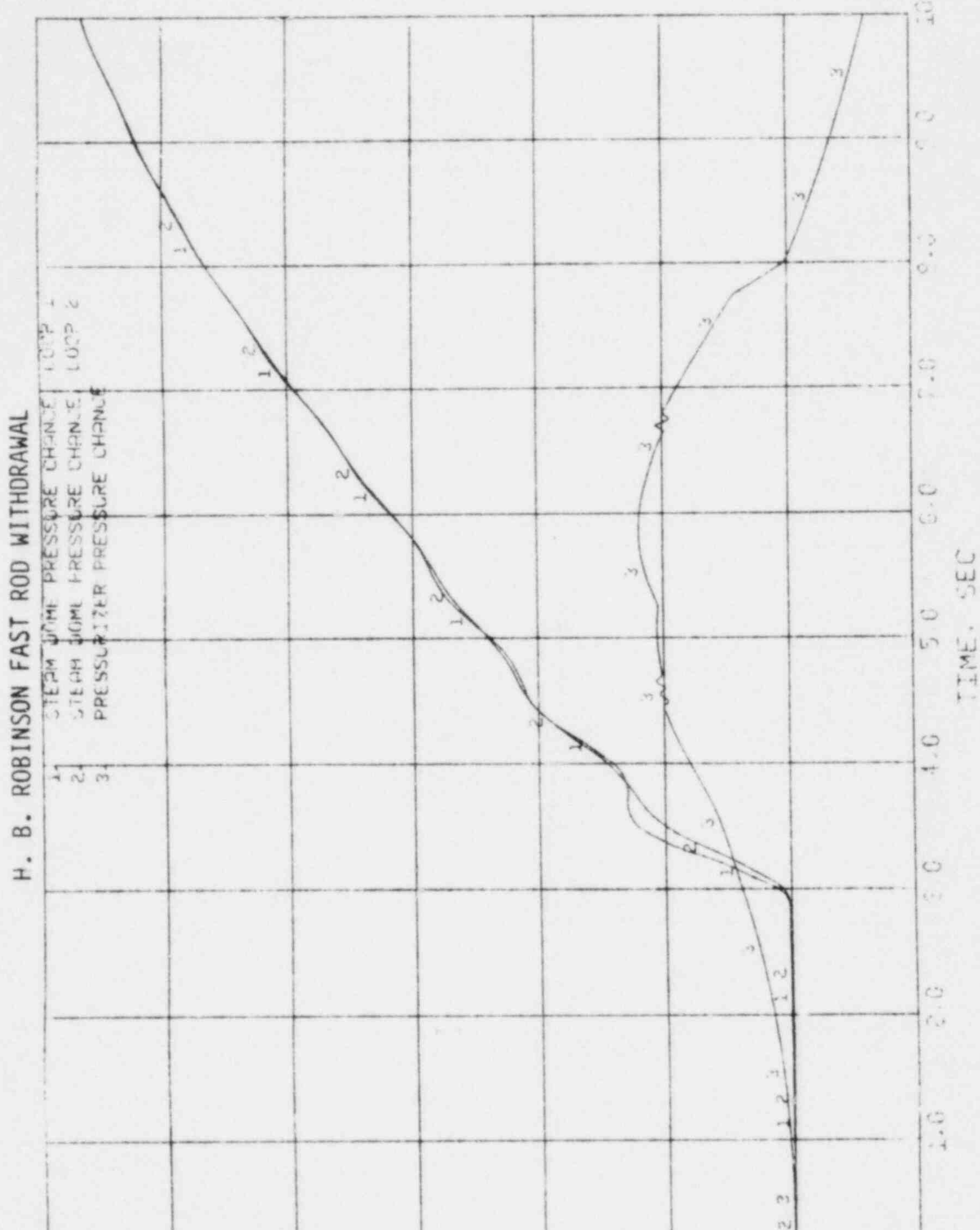

Figure 3.8 Pressure changes in pressurizer and steam generators for fast control rod withdrawal

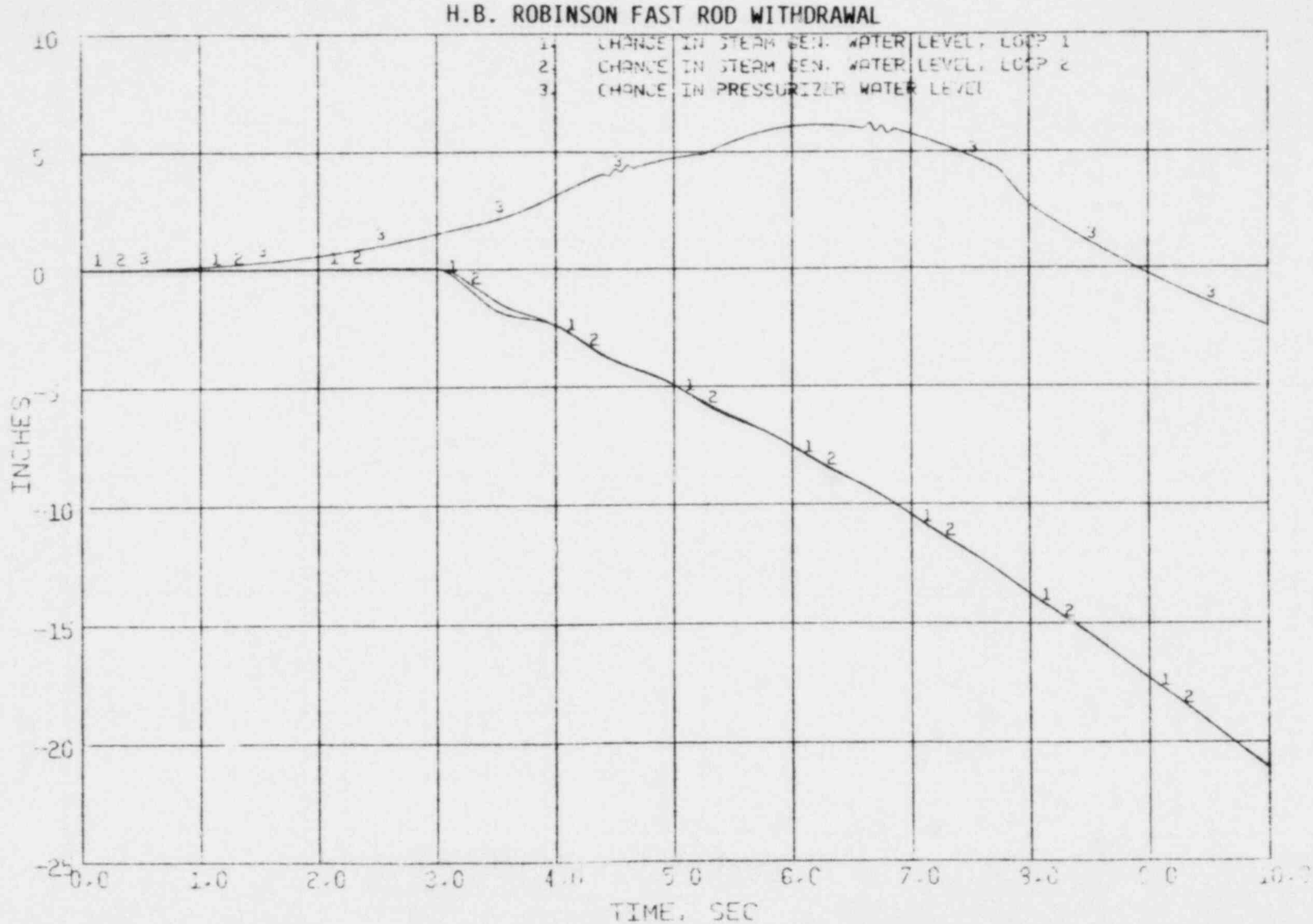

Figure 3.9 Level changes in pressurizer and steam generators for fast control rod withdrawal

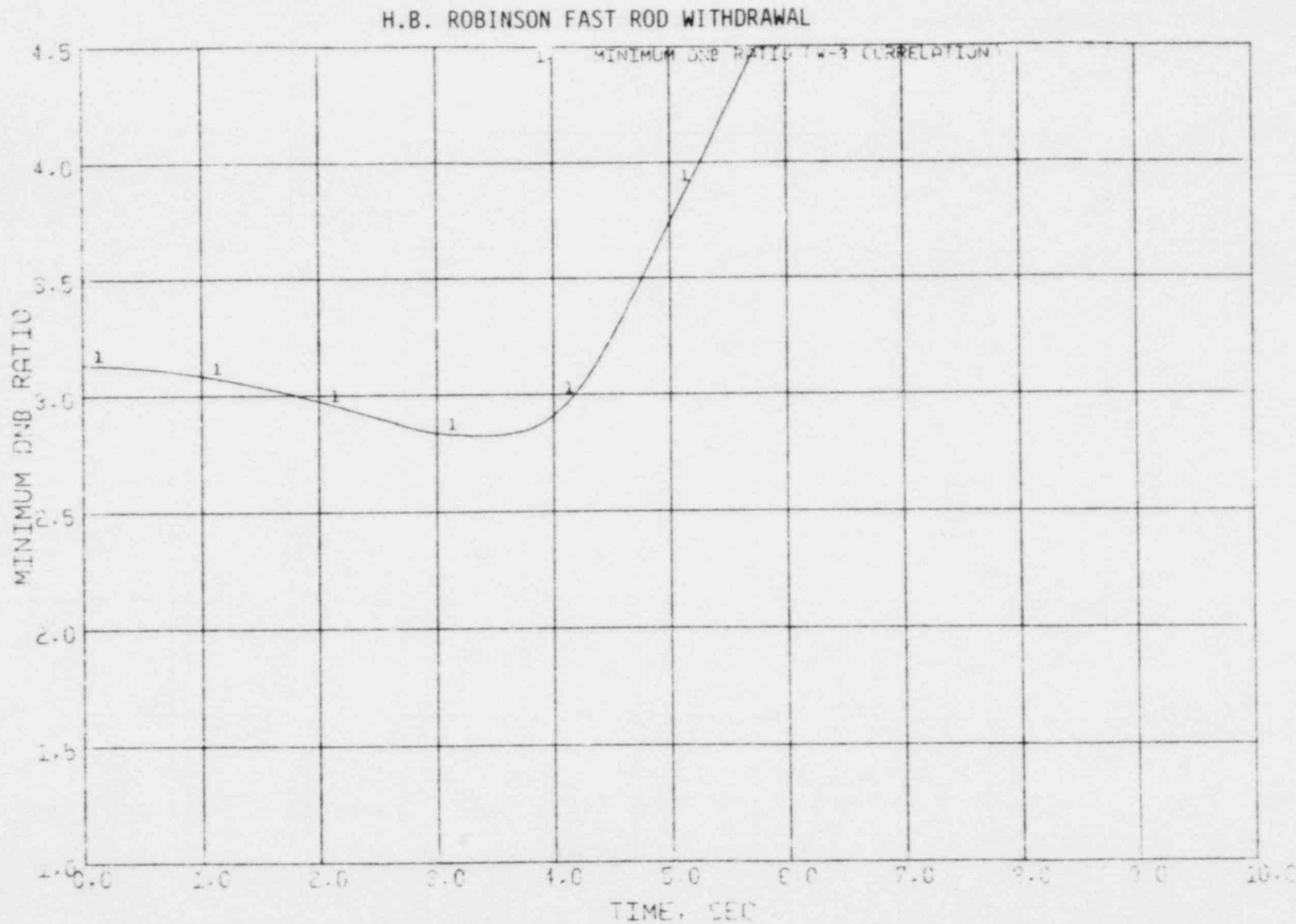

Figure 3.10 Minimum DNB ratio for fast control rod withdrawal

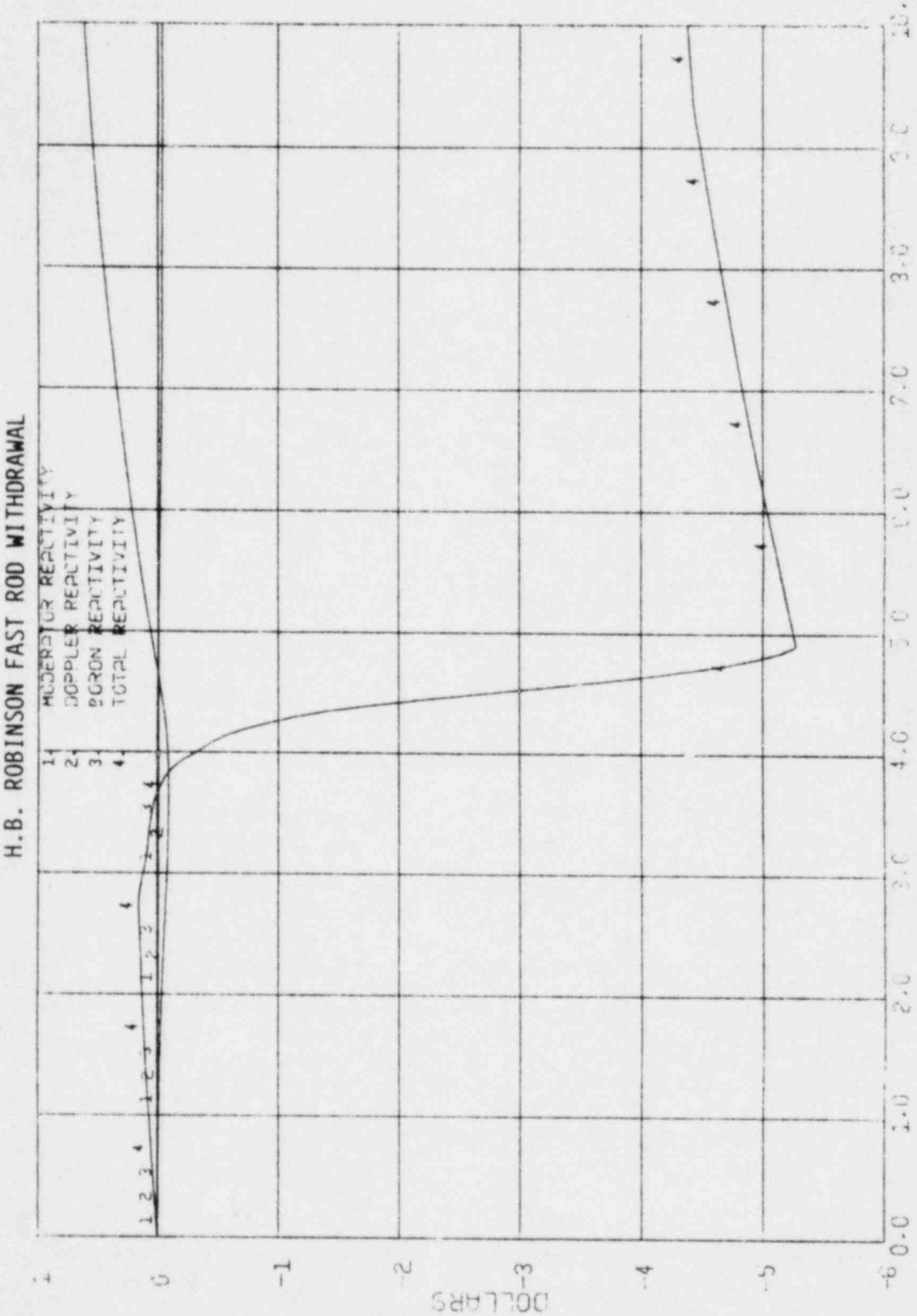

Figure 3.11 Reactivity worth for fast control rod withdrawal

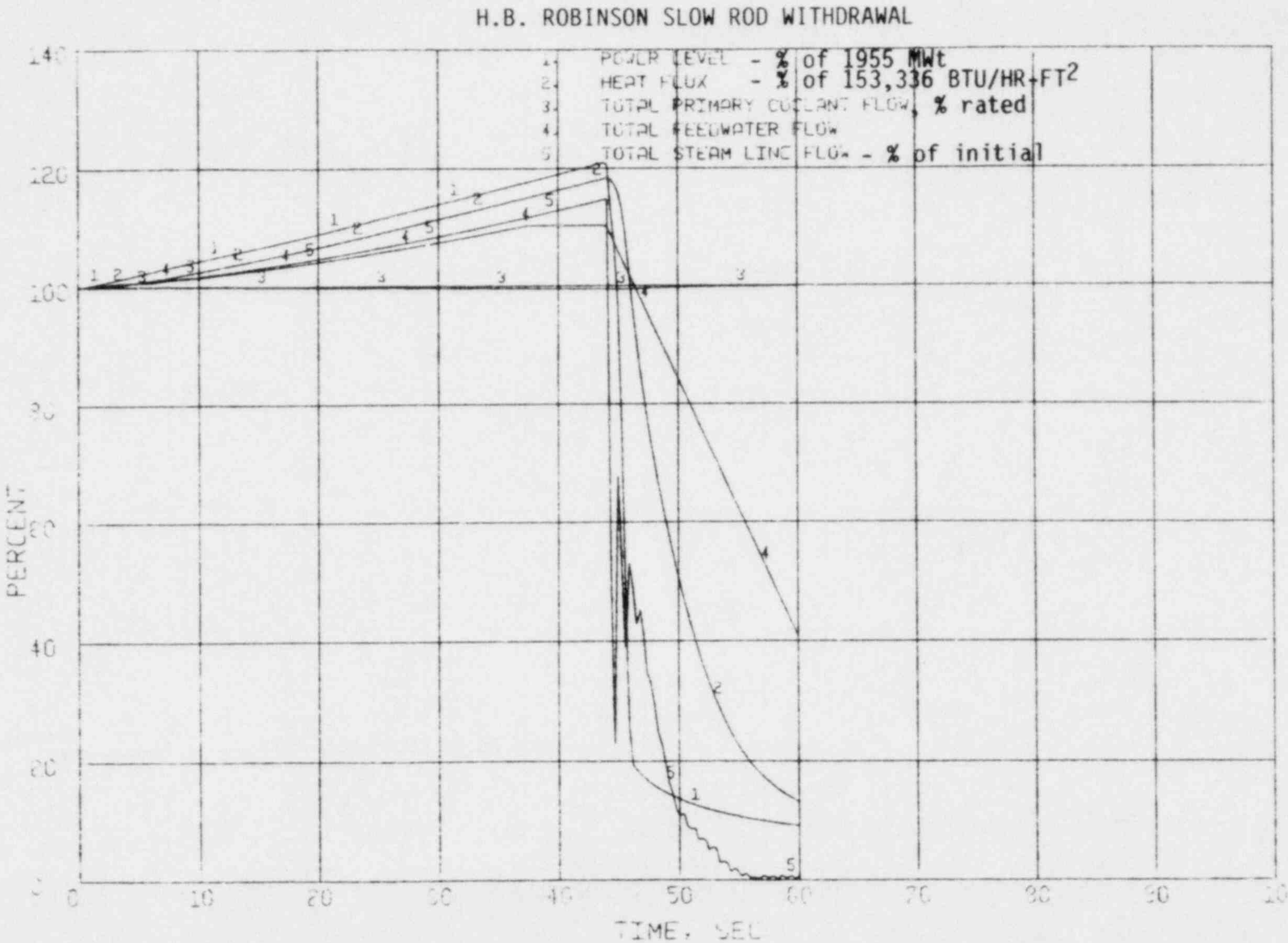

Figure 3.12 Power, heat flux and system flows for slow control rod withdrawal

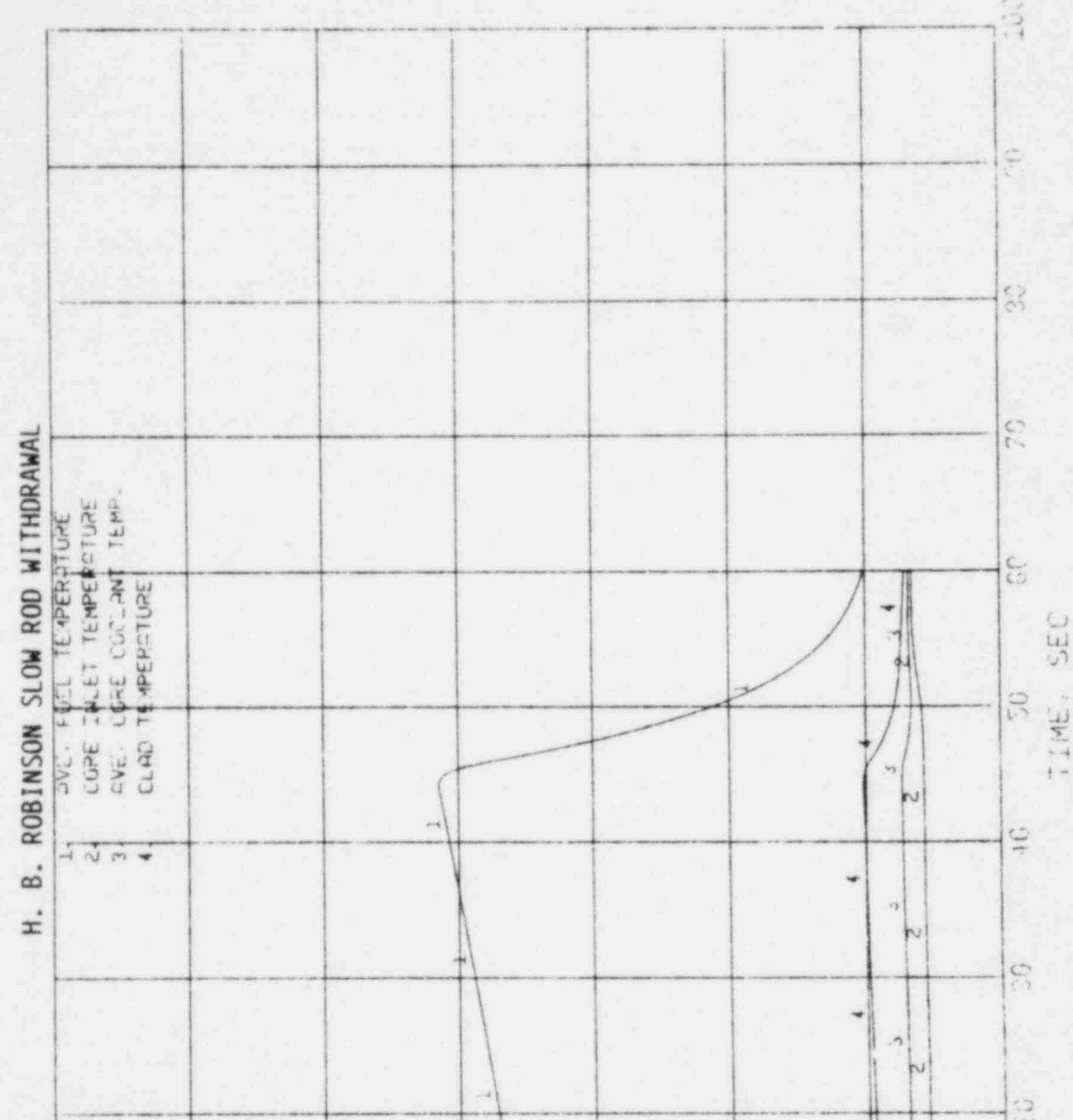

Figure 3.13 Core temperature responses for slow control rod withdrawal

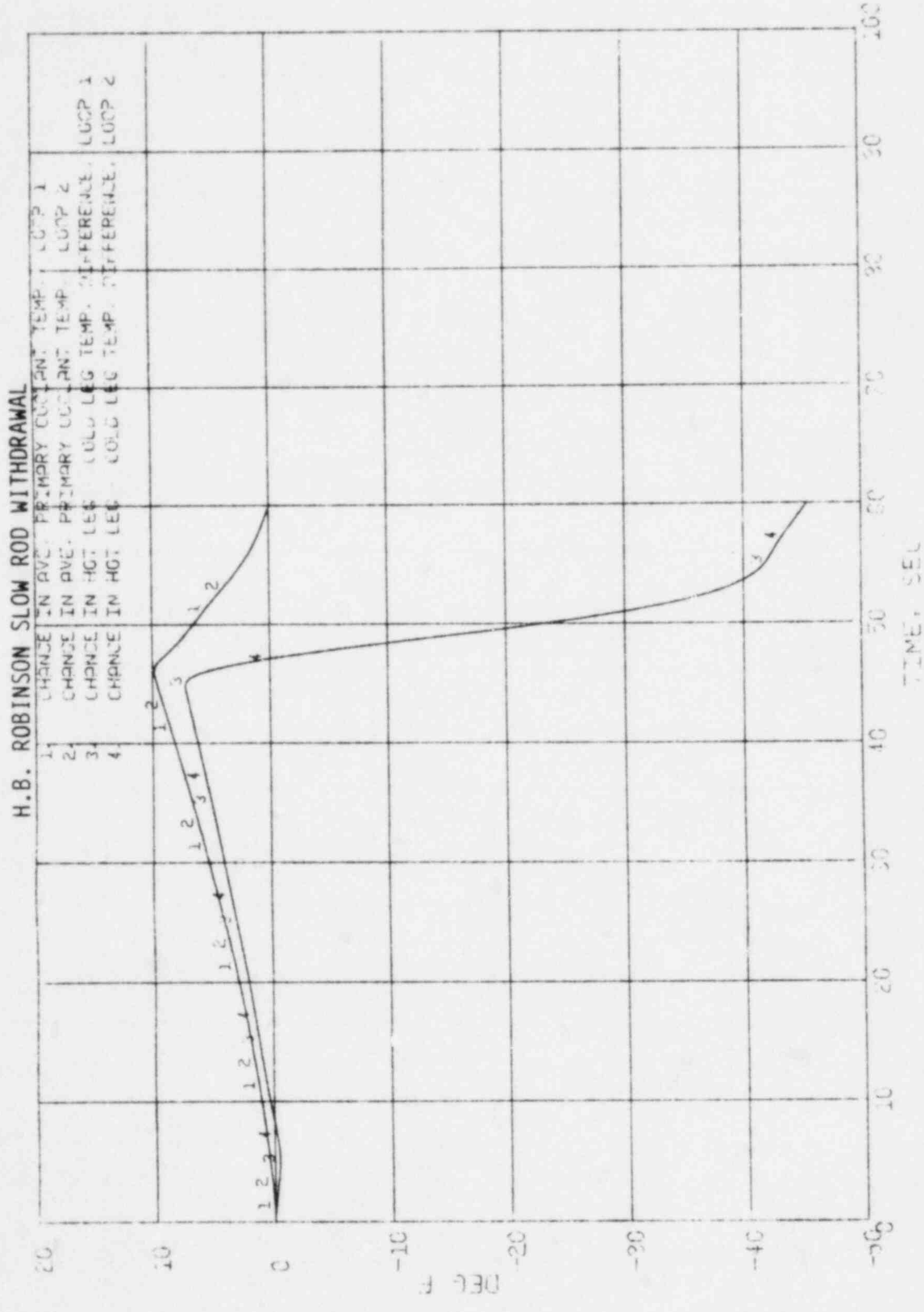

Figure 3.14 Primary Loop temperature changes for slow control rod withdrawal

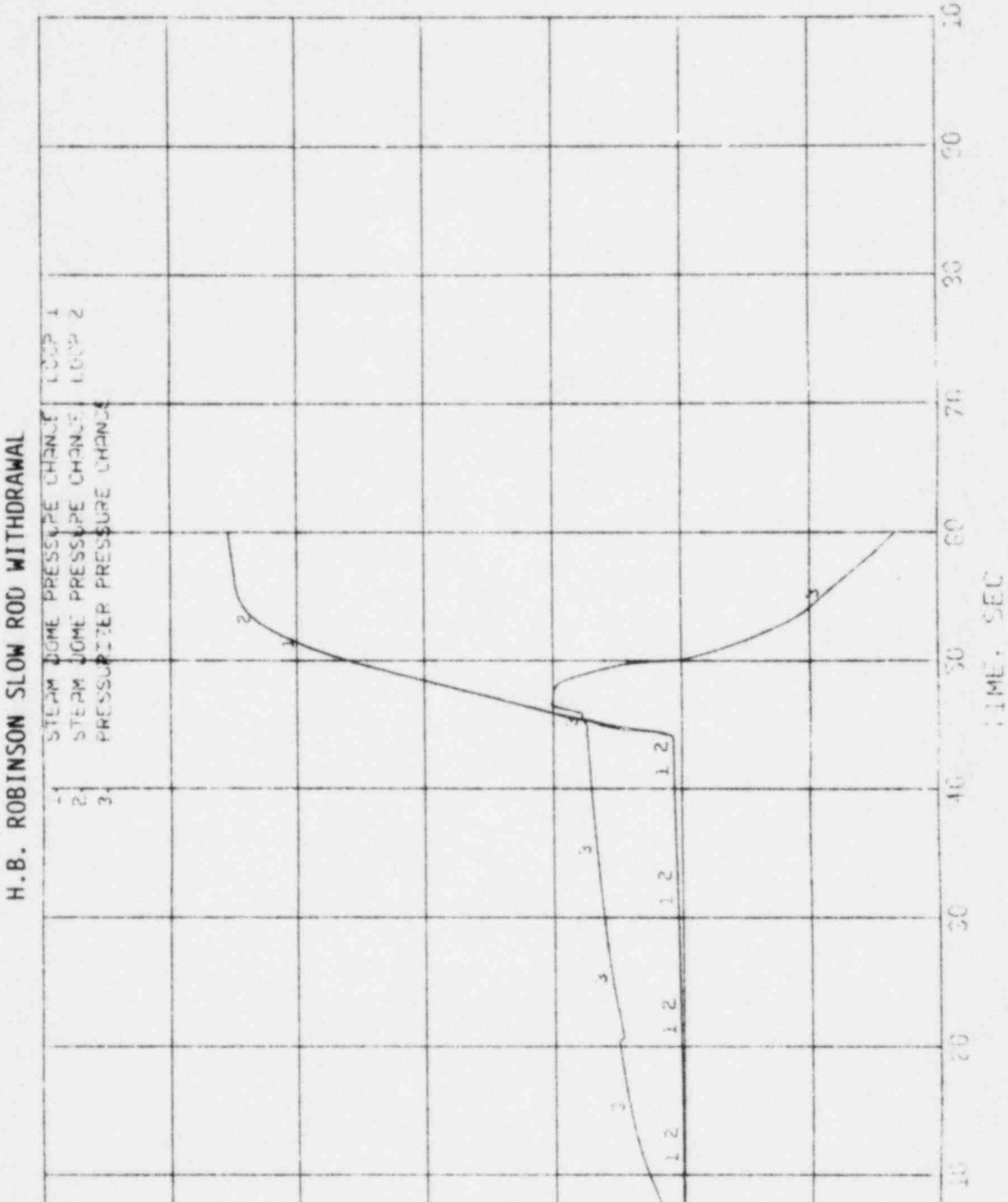

Figure 3.15 Pressure changes in pressurizer and steam generators for slow control rod withdrawal

H.B. ROBINSON SLOW ROD WITHDDRAWAL

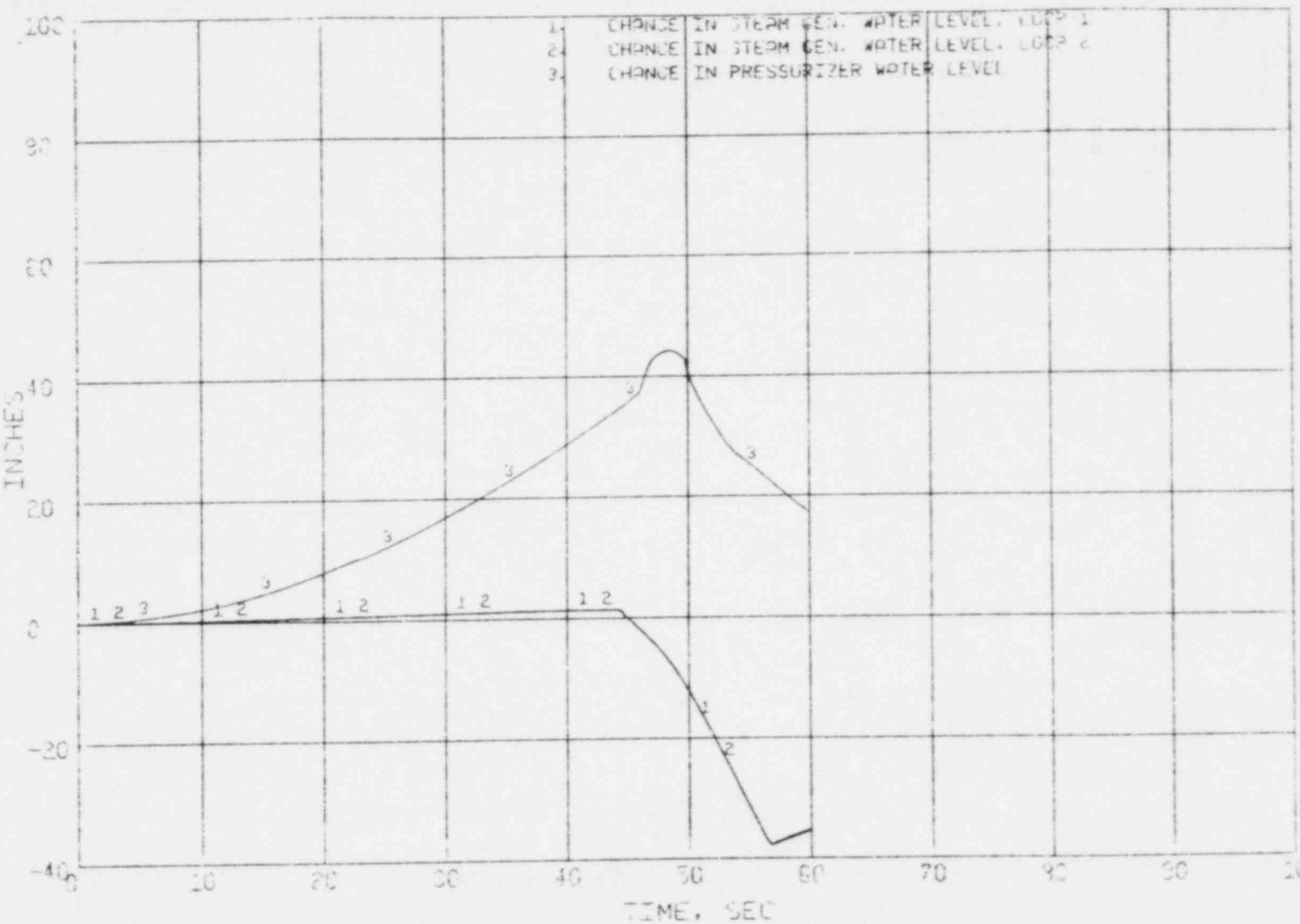

Figure 3.16 Level changes in pressurizer and steam generators for slow control rod withdrawal

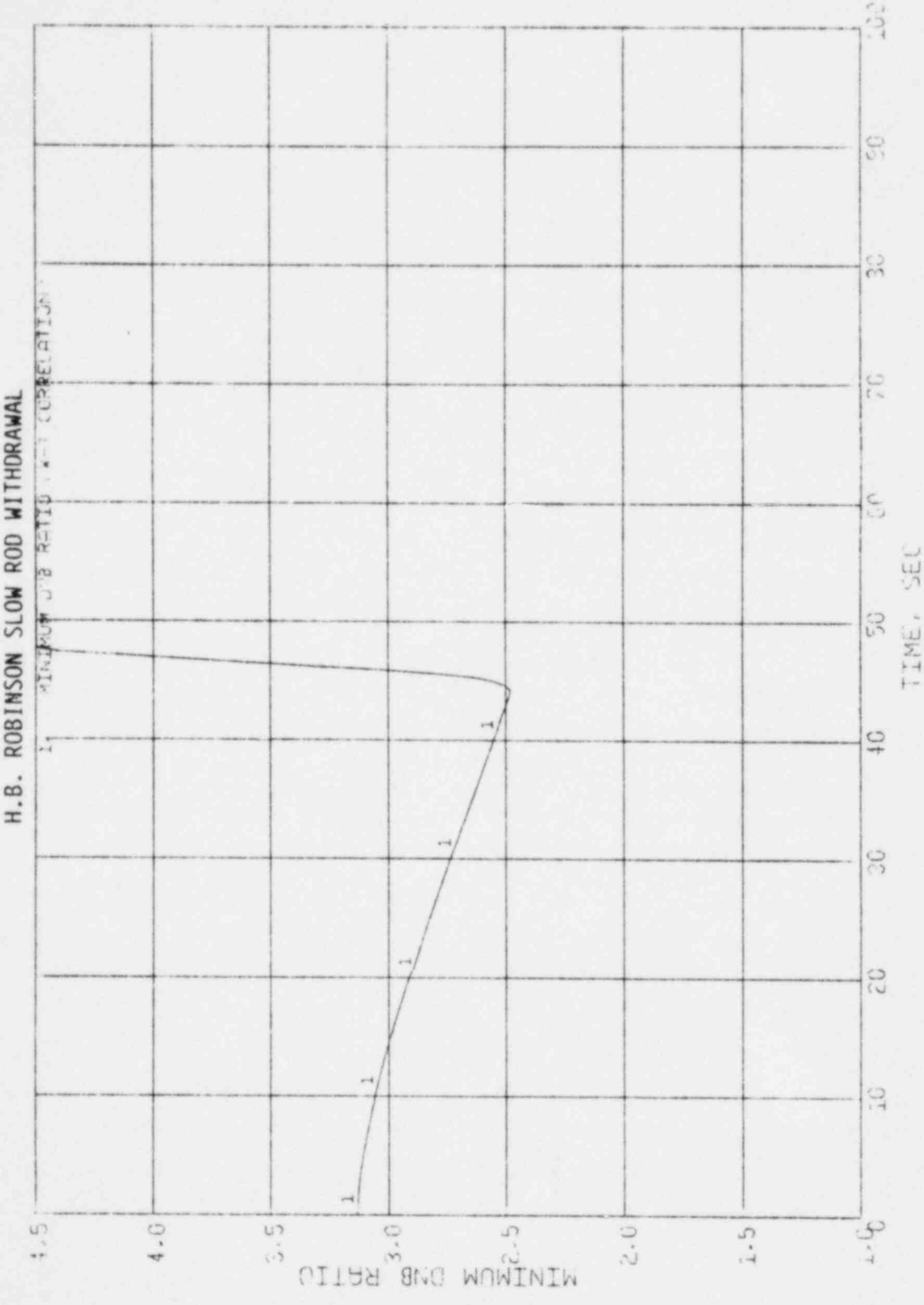

Figure 3.17 Minimum DNB ratio for slow control rod withdrawal

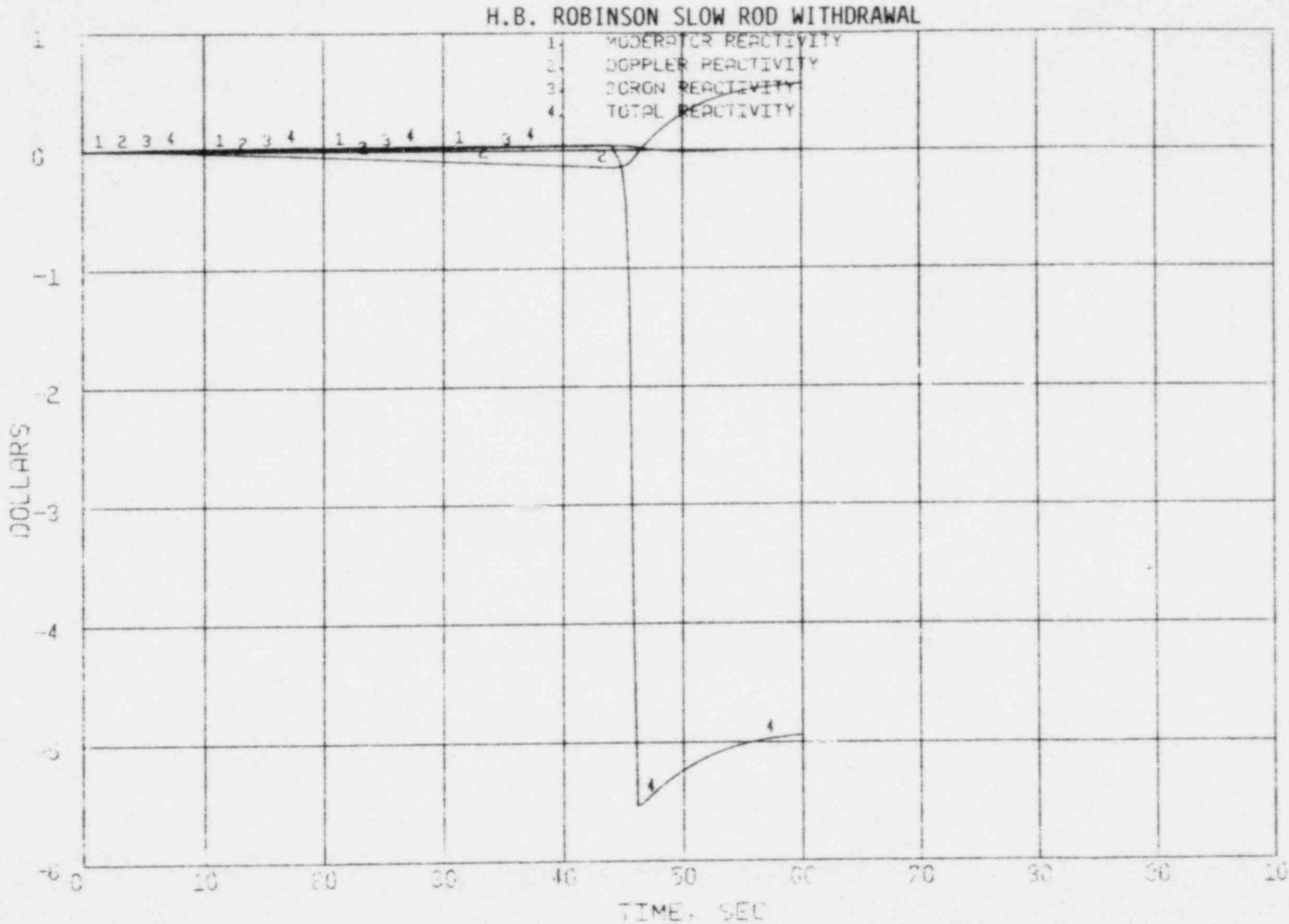

Figure 3.18 Reactivity worth for slow control rod withdrawal

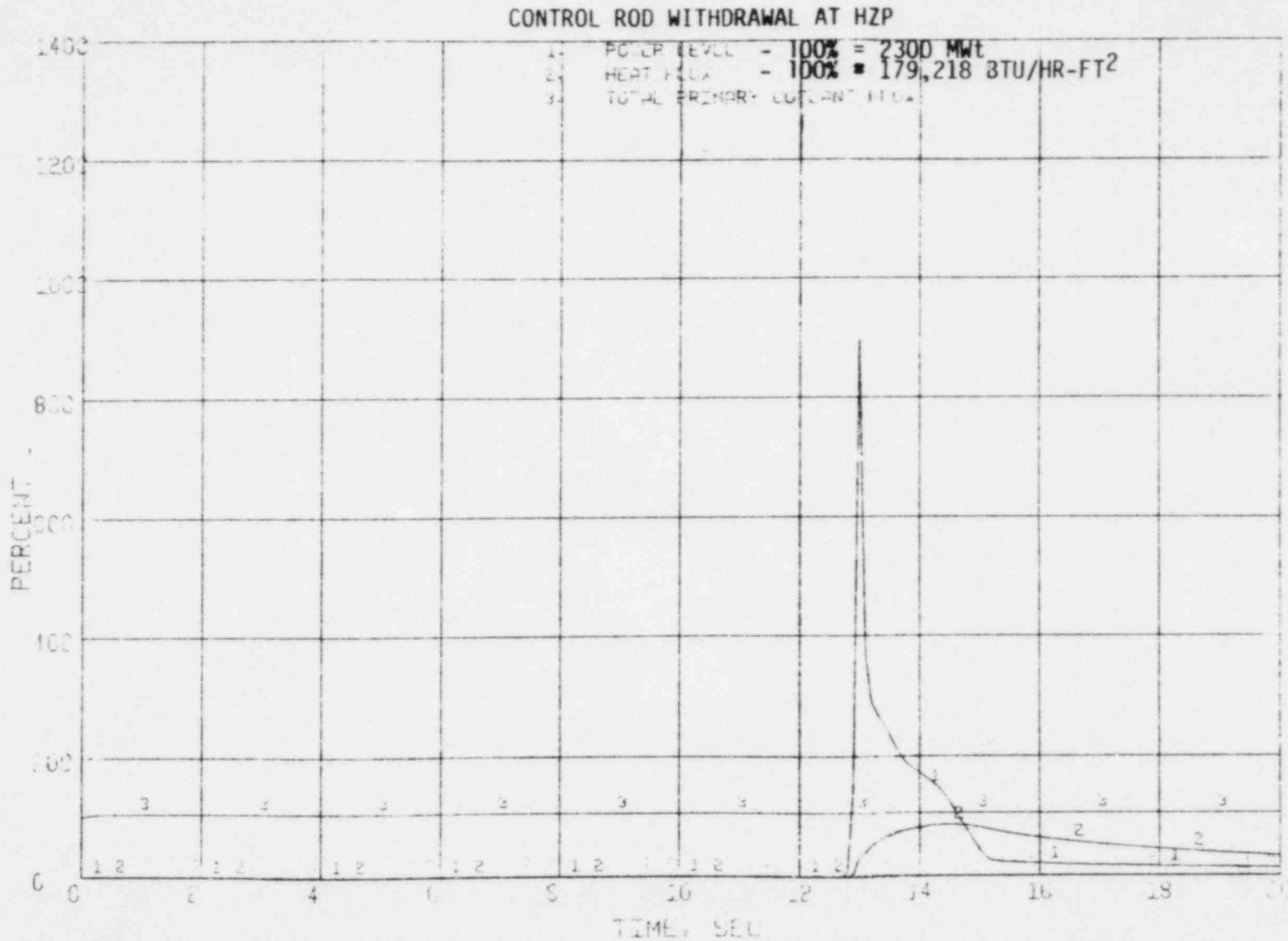

Figure 3.19 Power, heat flux and primary coolant flow for control rod withdrawal at HZP

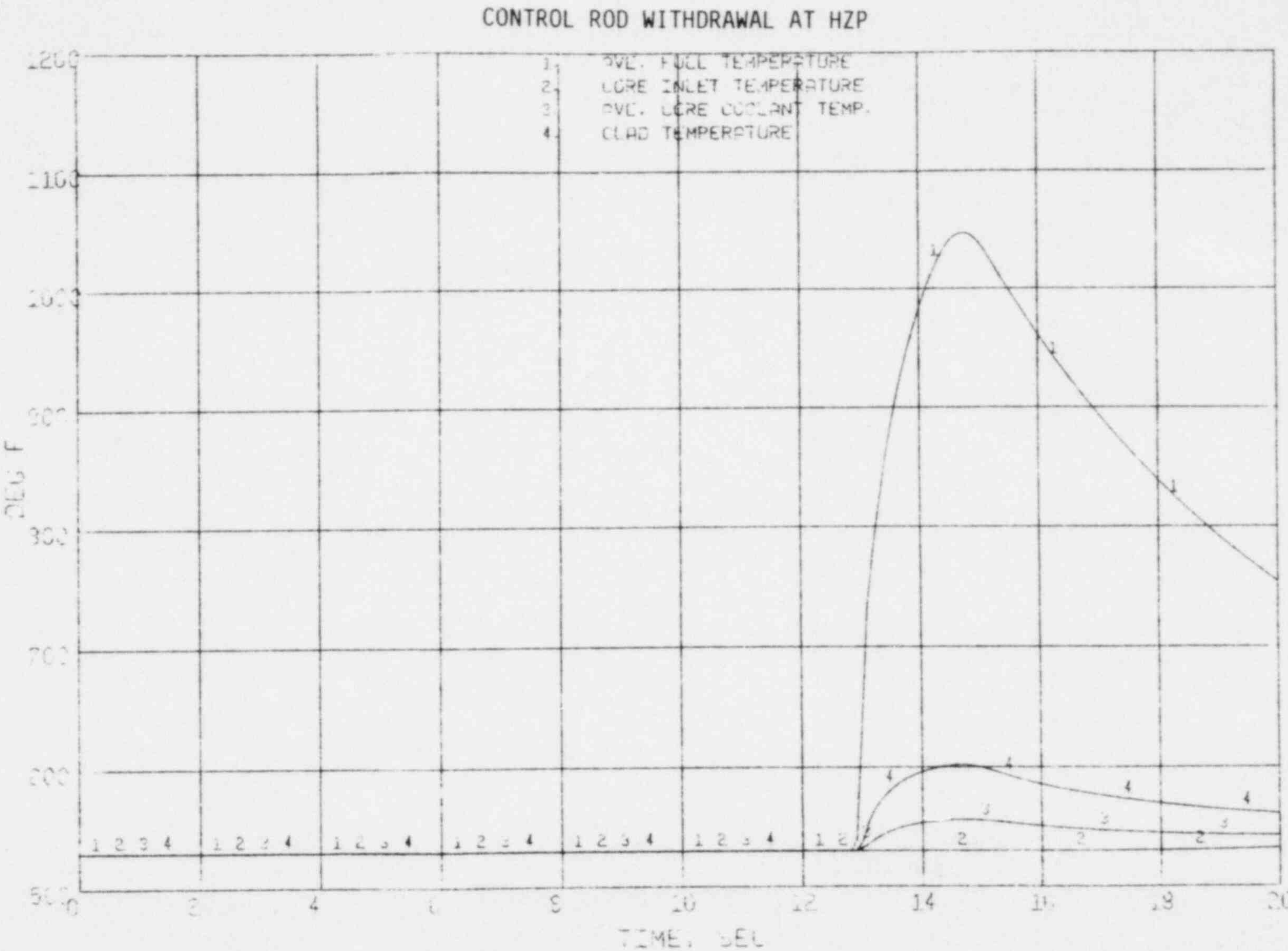

Figure 3.20 Core temperature responses for control rod withdrawal at HZP

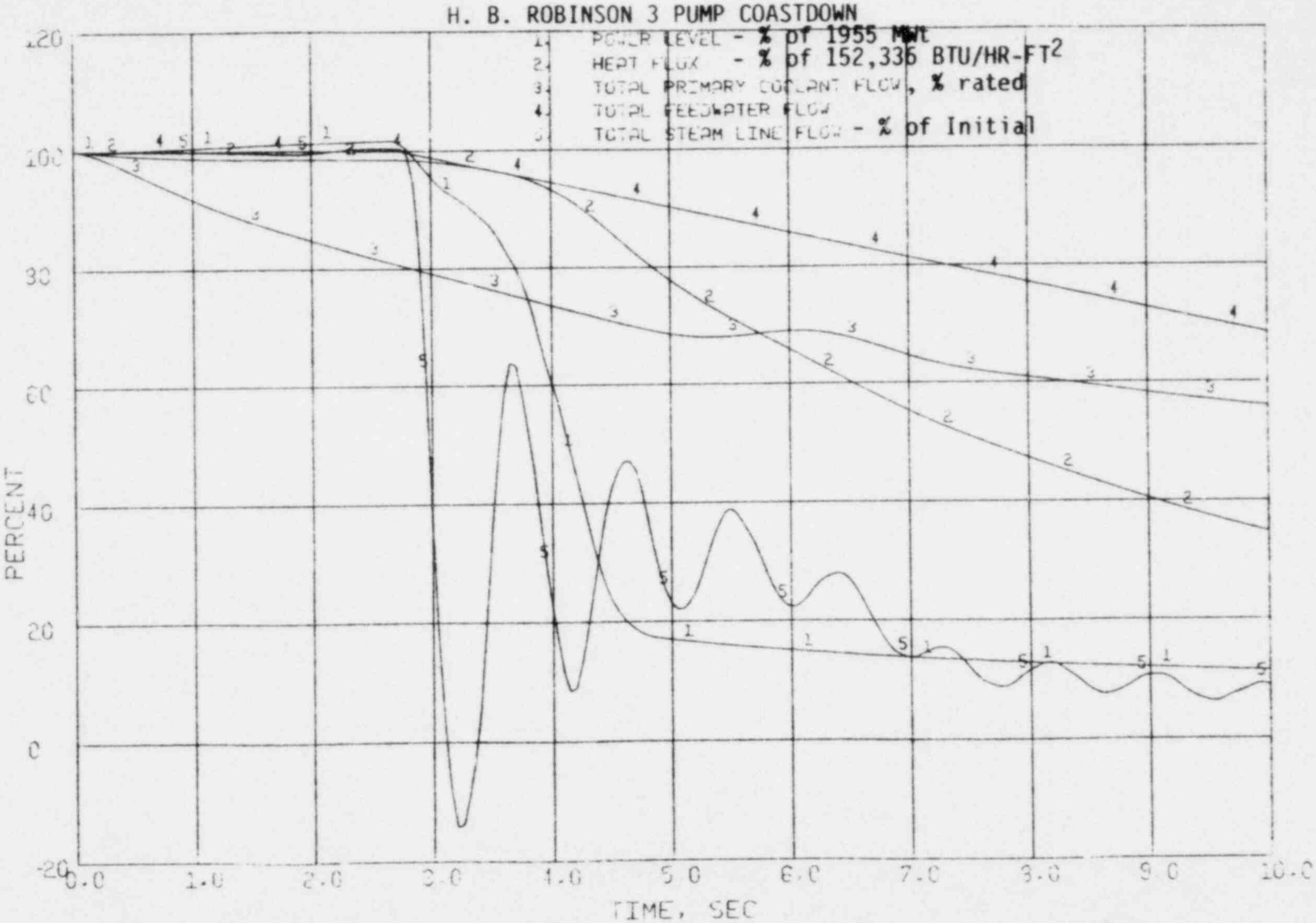

Figure 3.21 Power, heat flux and system flows for coolant pump trip.

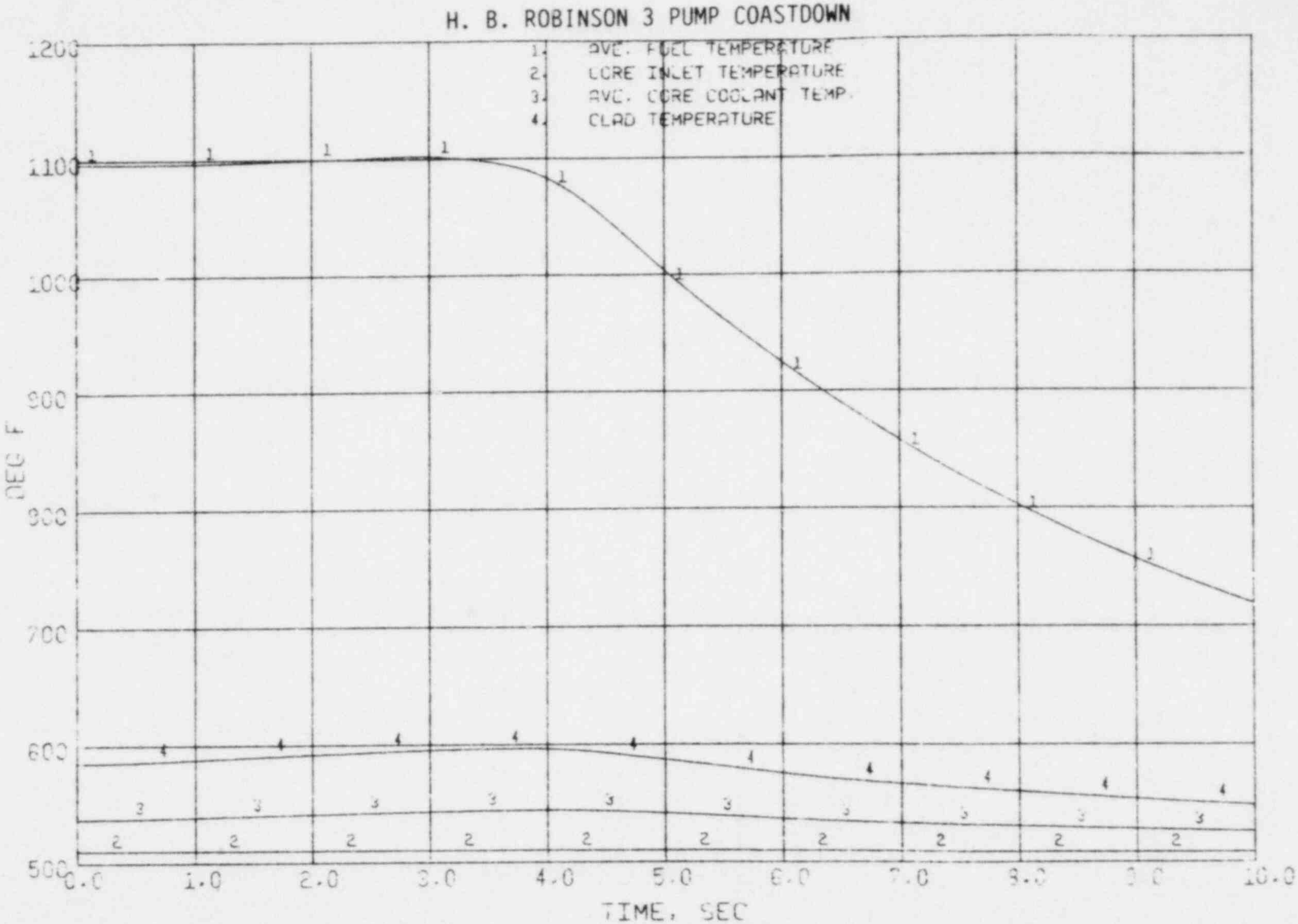

Figure 3.22 Core temperature responses for coolant pump trip.

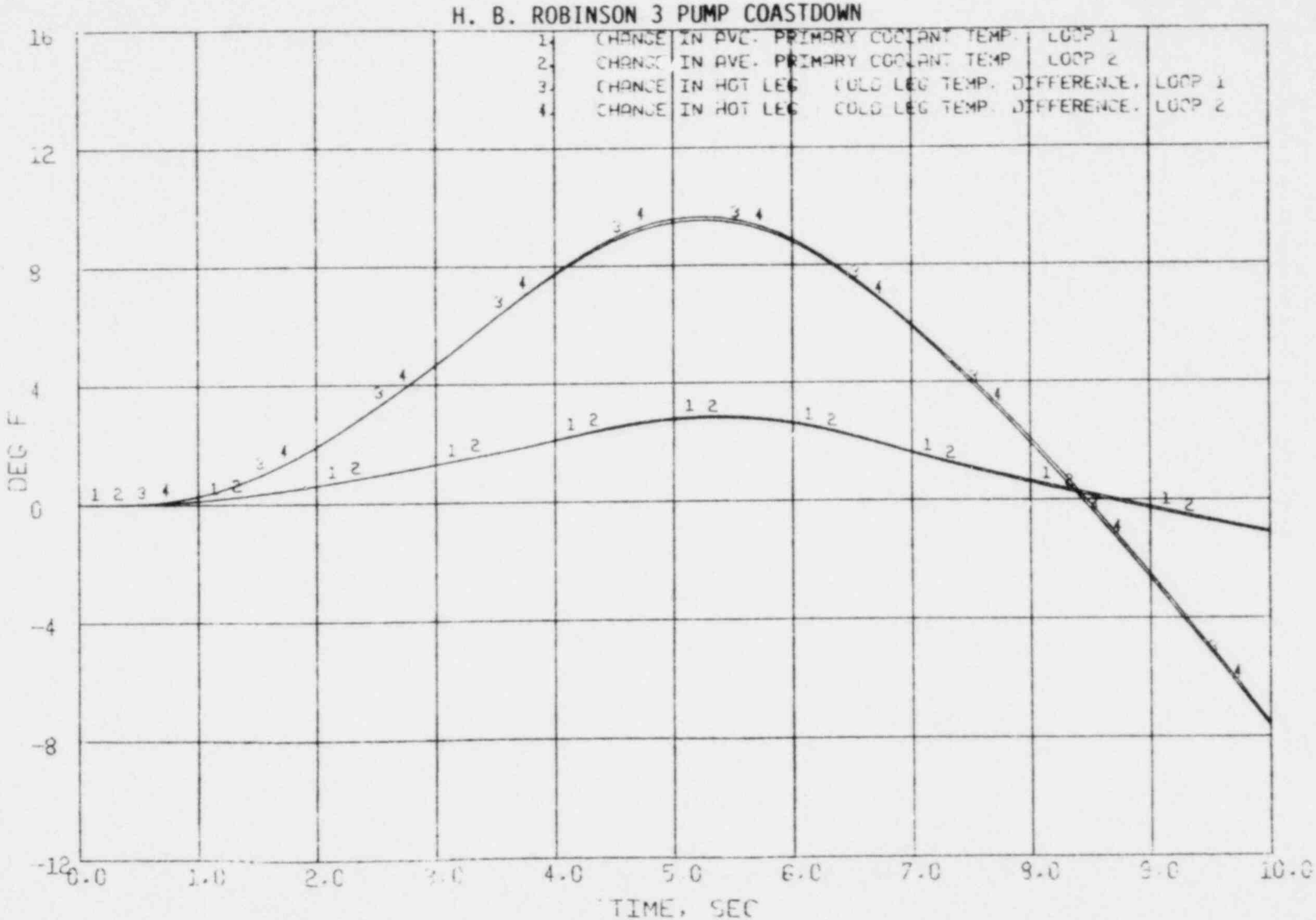

Figure 3.23 Primary loop temperature changes for coolant pump trip.

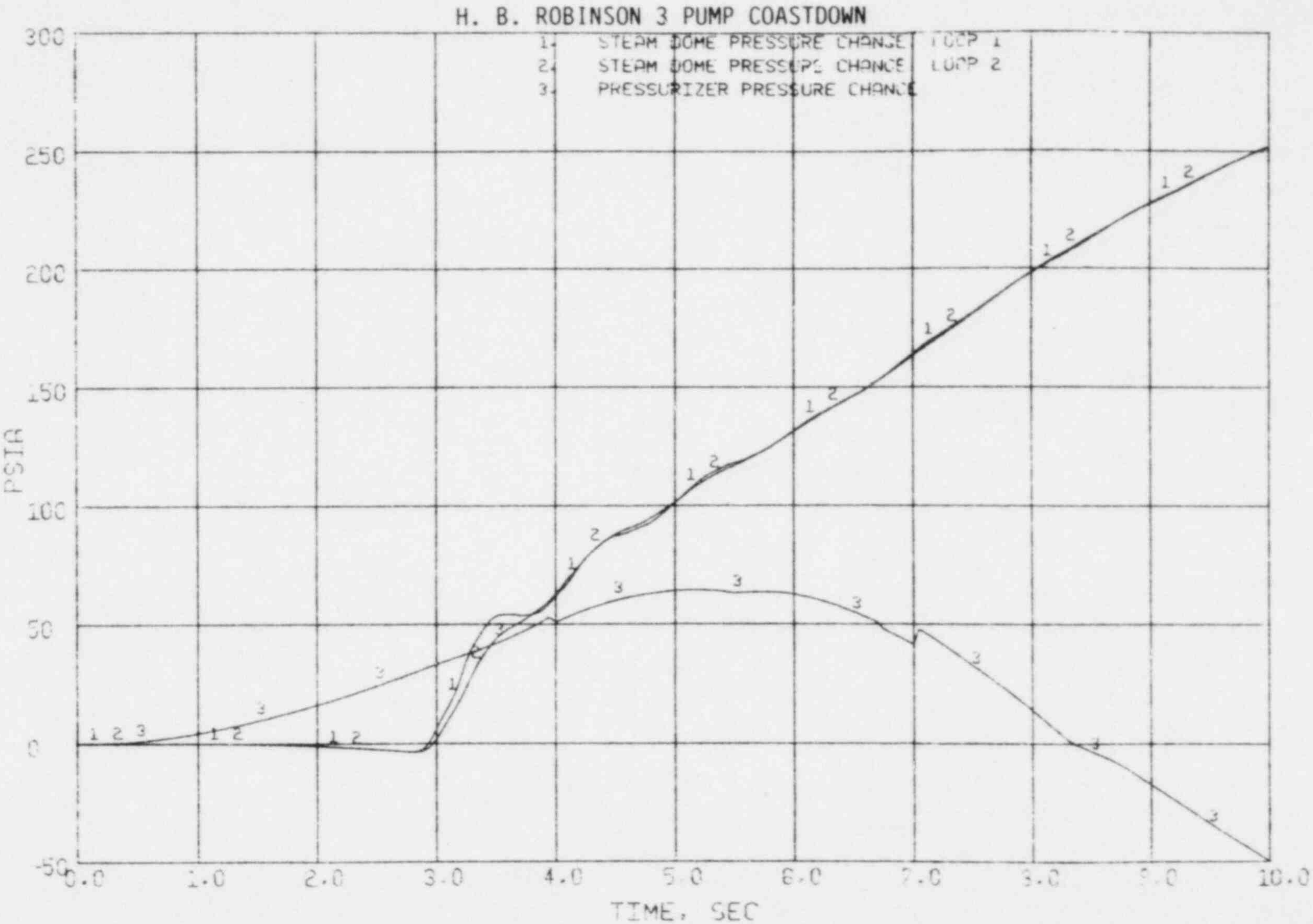

Figure 3.24 Pressure changes in pressurizer and steam generators for coolant pump trip.

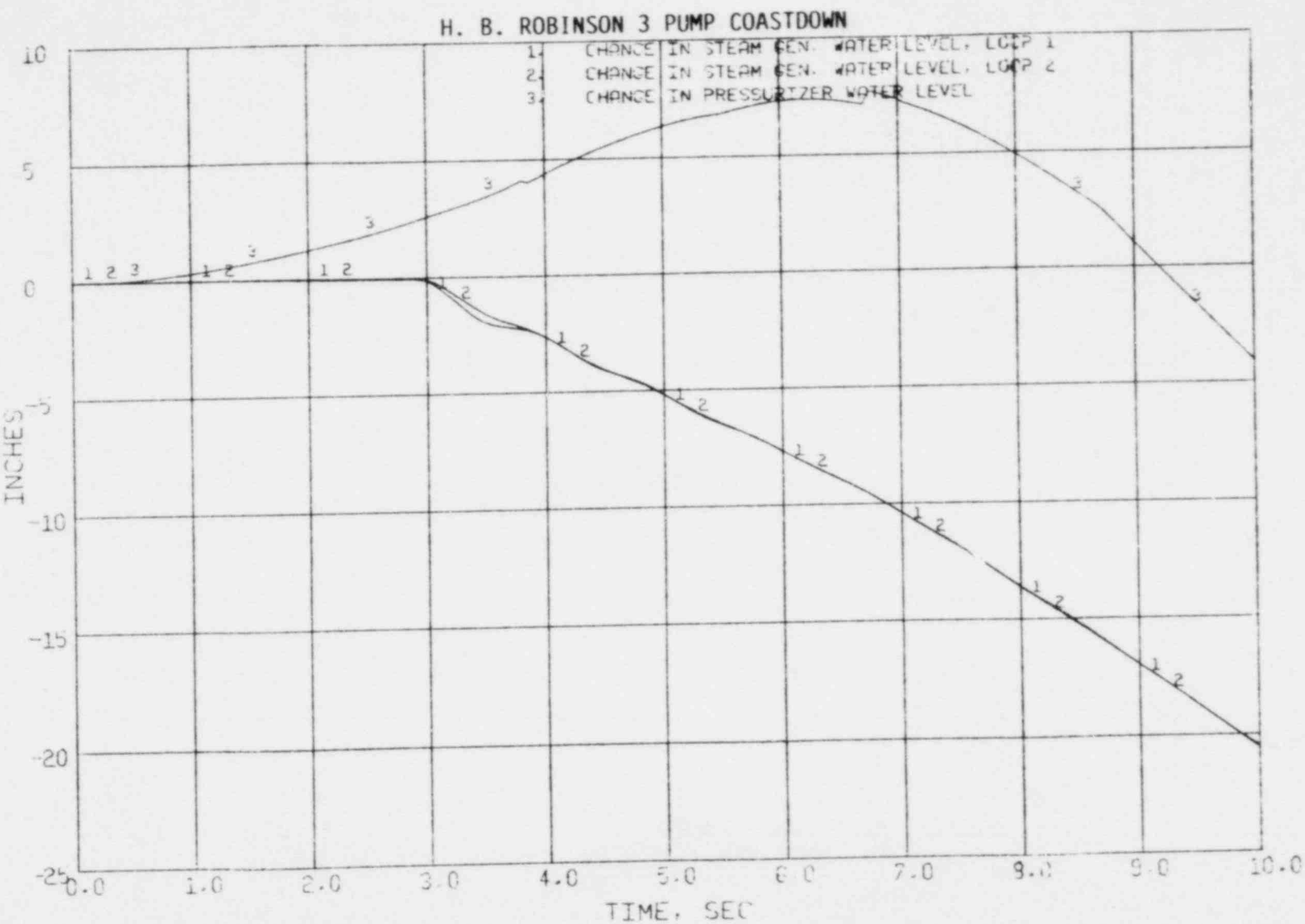

Figure 3.25 Level changes in pressurizer and steam generators for coolant pump trip.

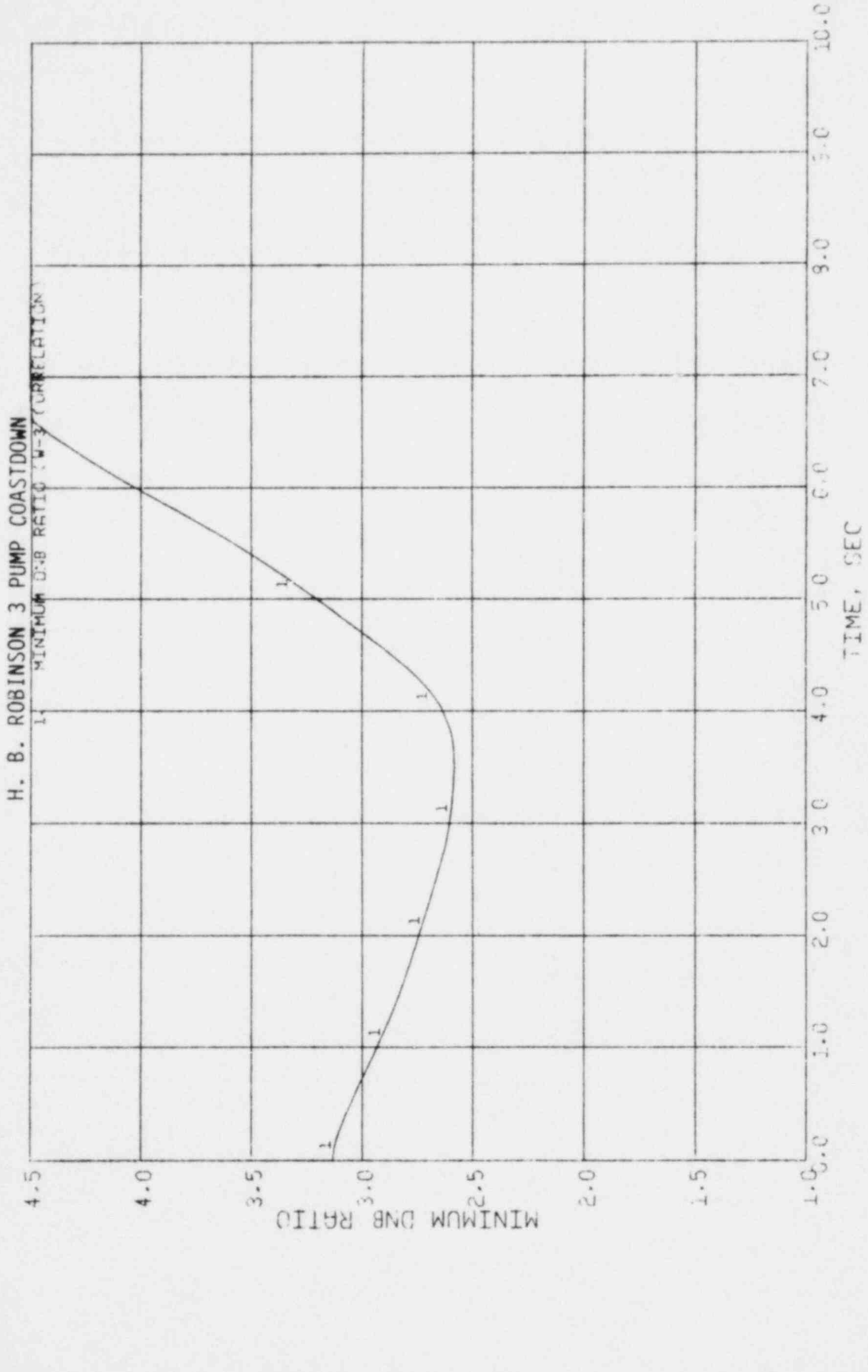

Figure 3.26 Minimum DNB ratio for coolant pump trip.

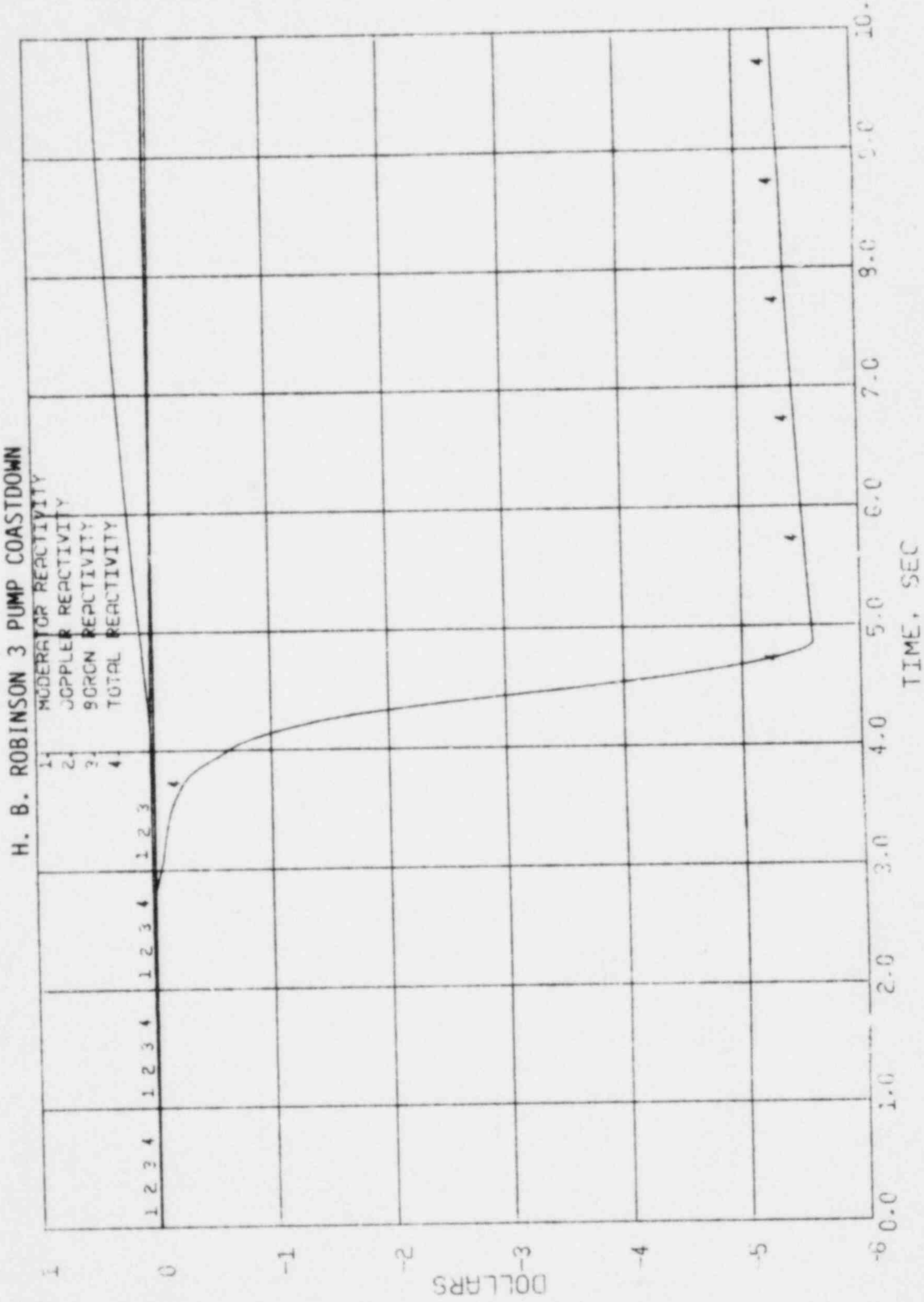

Figure 3.27 Reactivity worth for coolant pump trip.

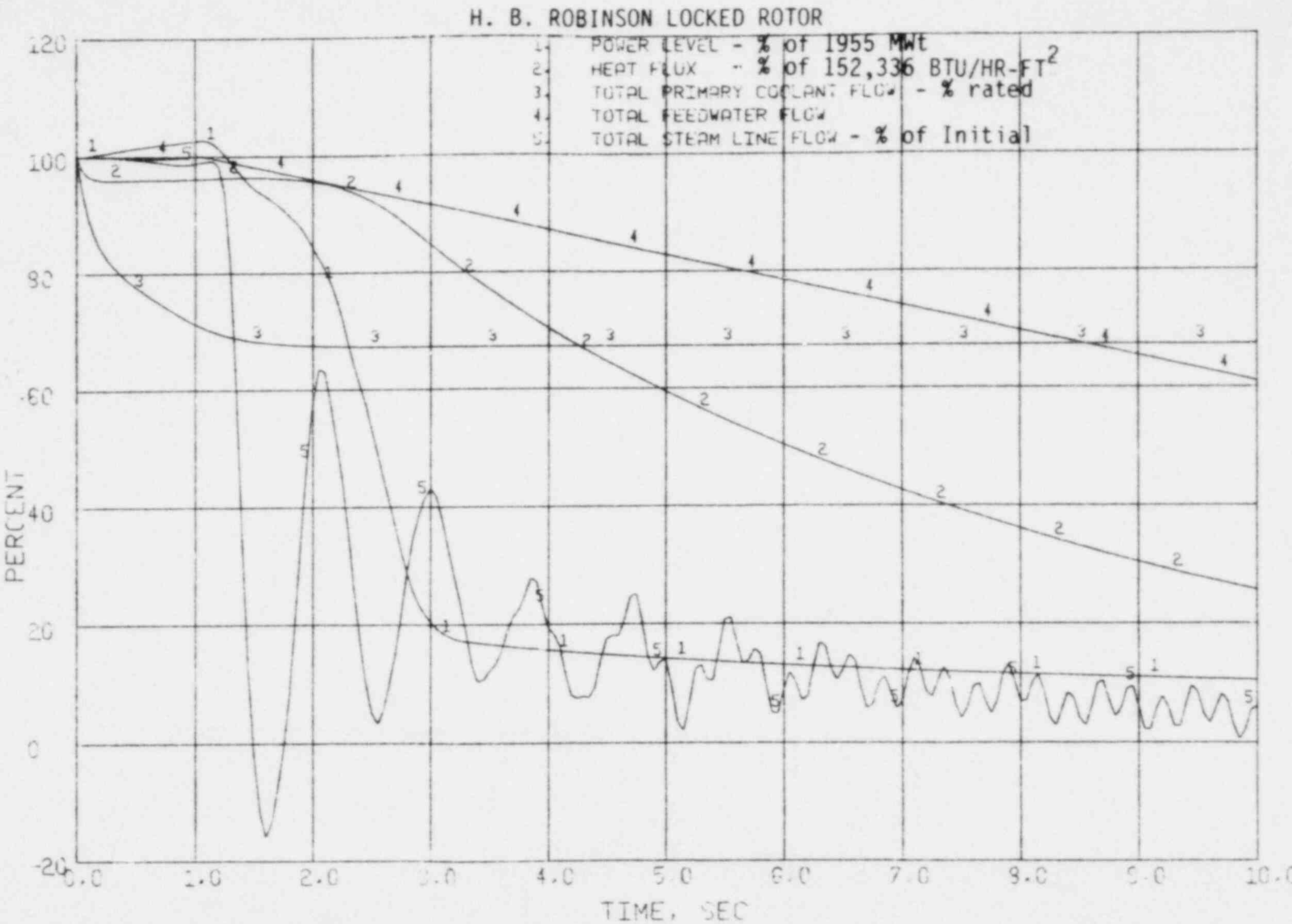

Figure 3.28 Power, heat flux and system flows for coolant pump seizure.

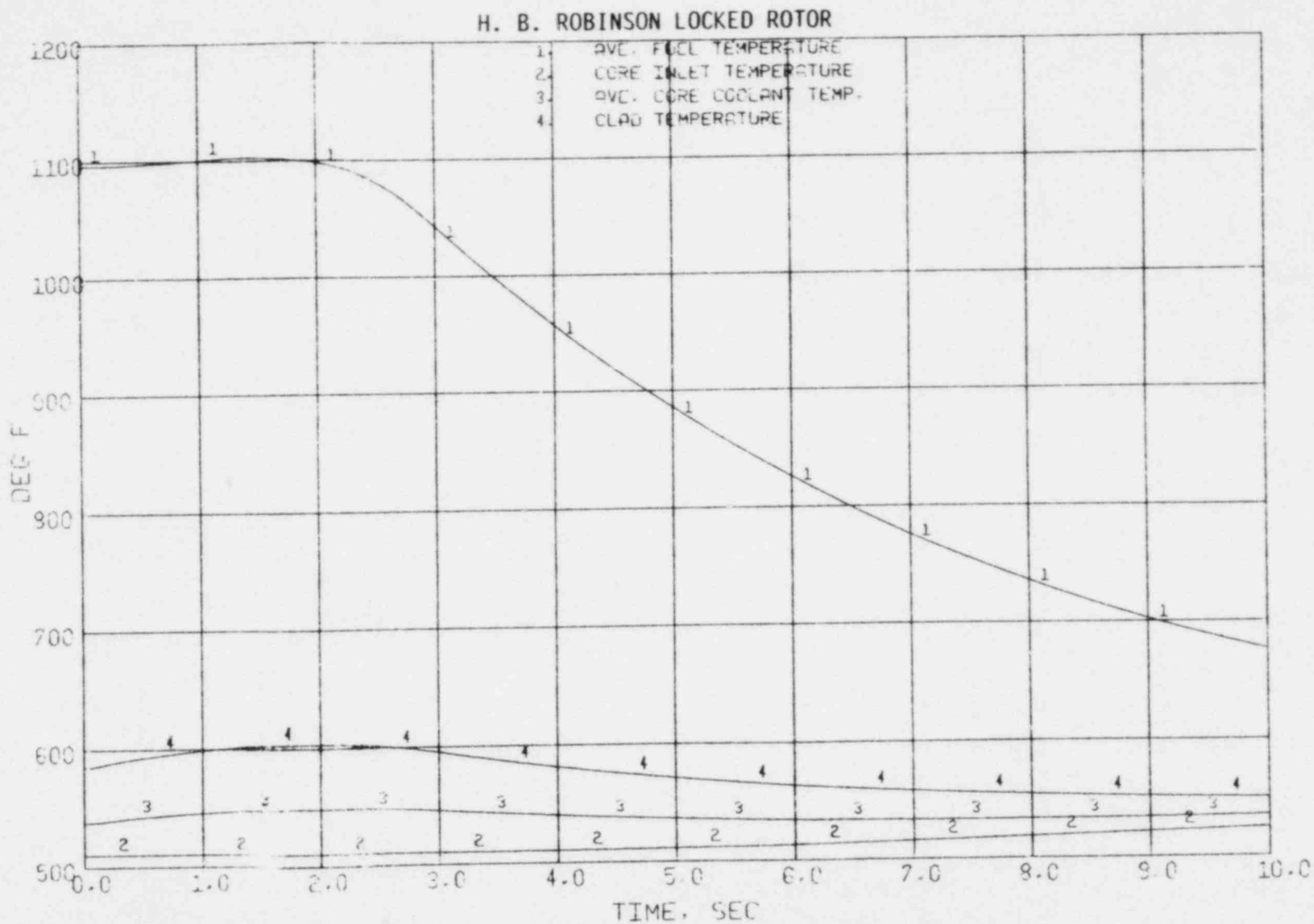

Figure 3.29 Core temperature responses for coolant pump seizure.

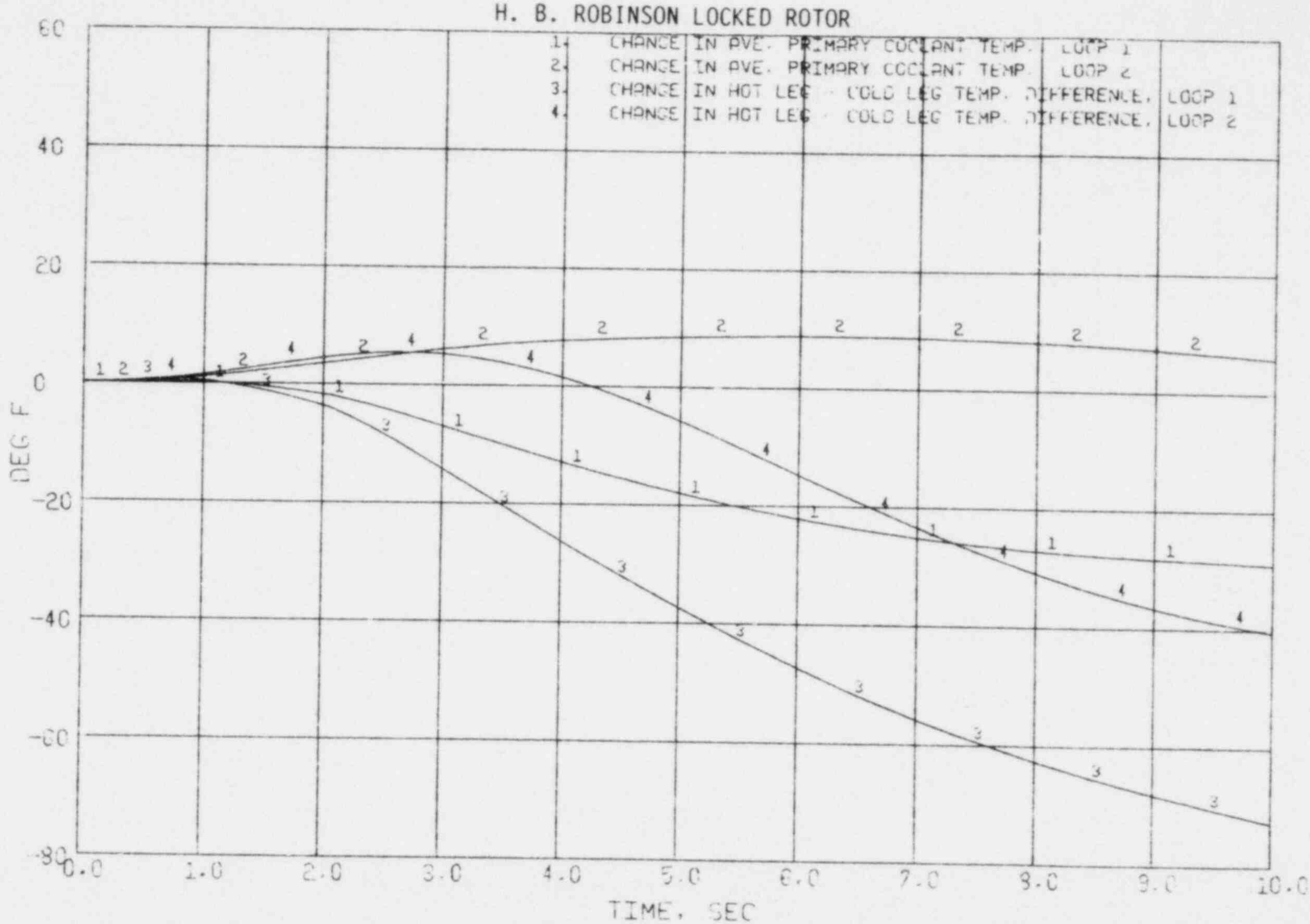

Figure 3.30 Primary loop temperature changes for coolant pump seizure.

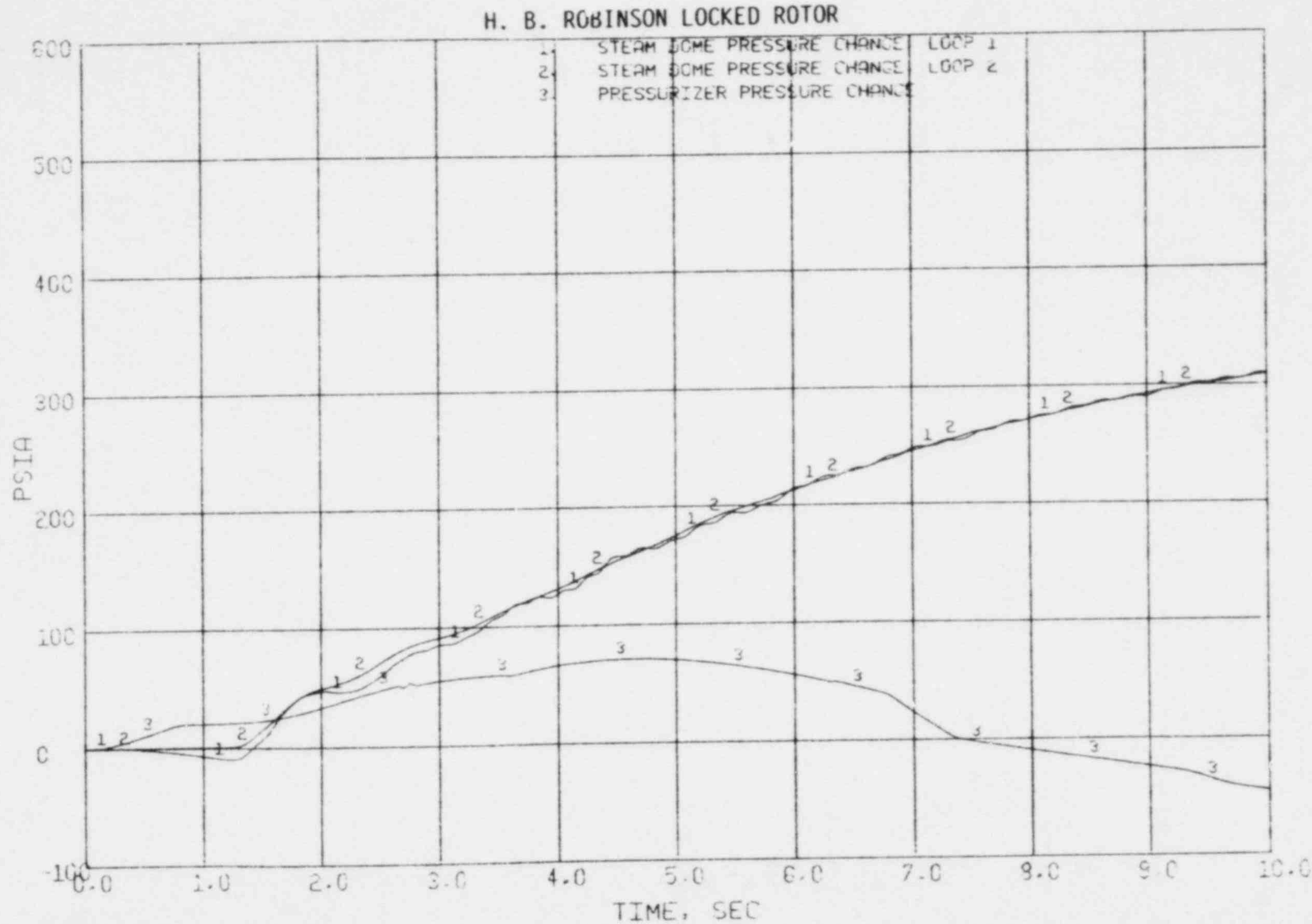

Figure 3.31 Pressure changes in pressurizer and steam generators for coolant pump seizure.

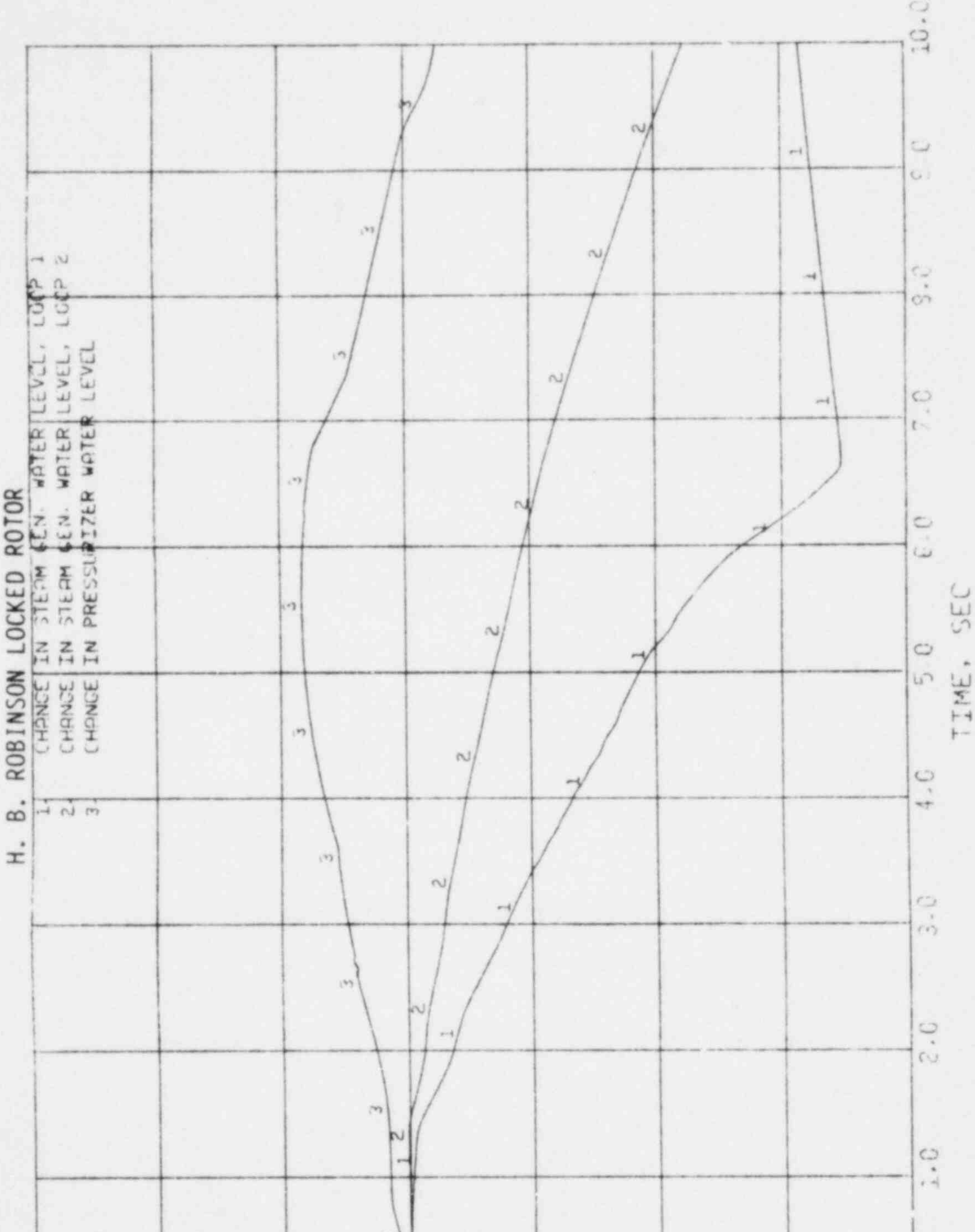

Figure 3.32 Level changes in pressurizer and steam generators for coolant pump seizure.

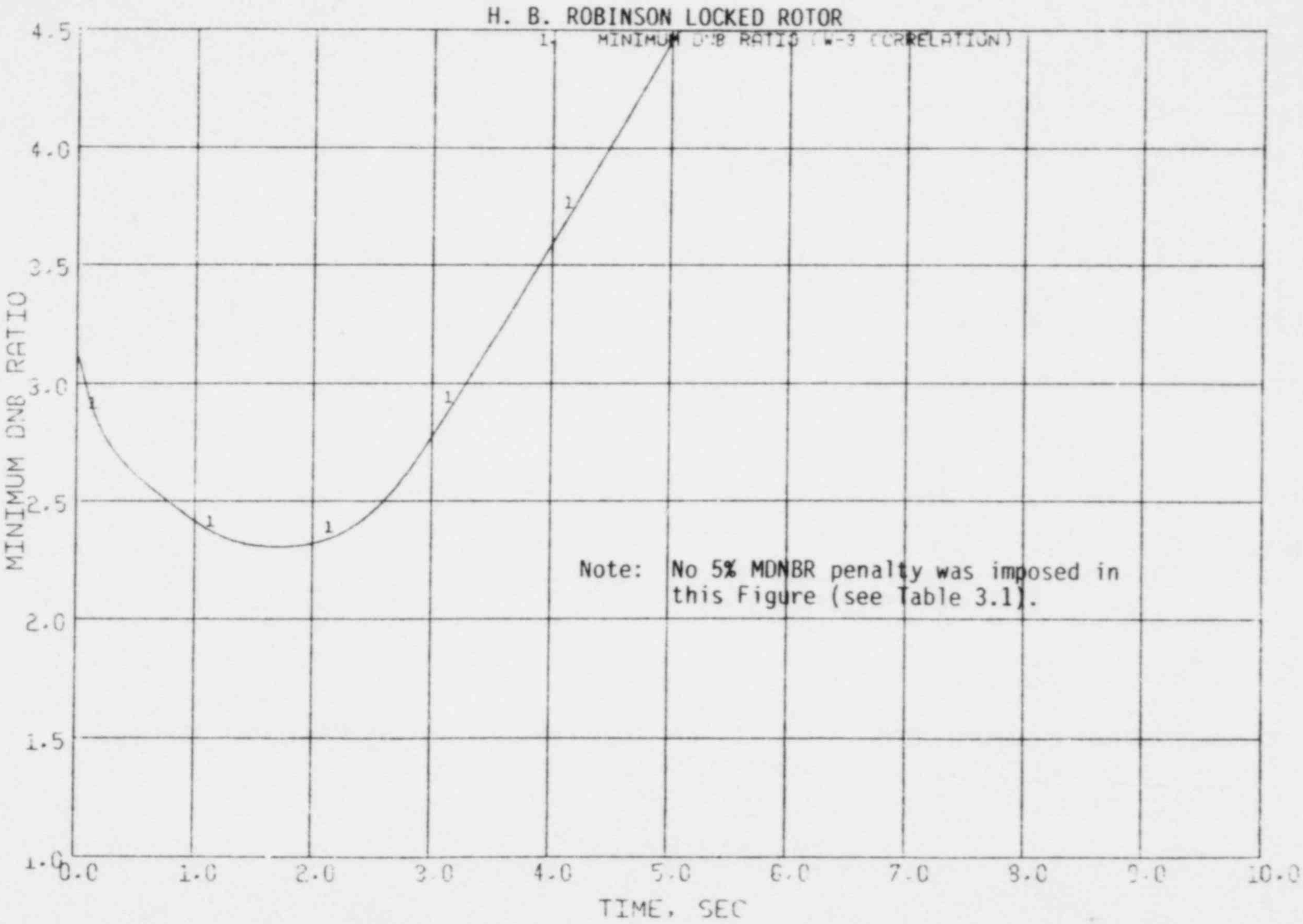

Figure 3.33 Minimum DNB ratio for coolant pump seizure.

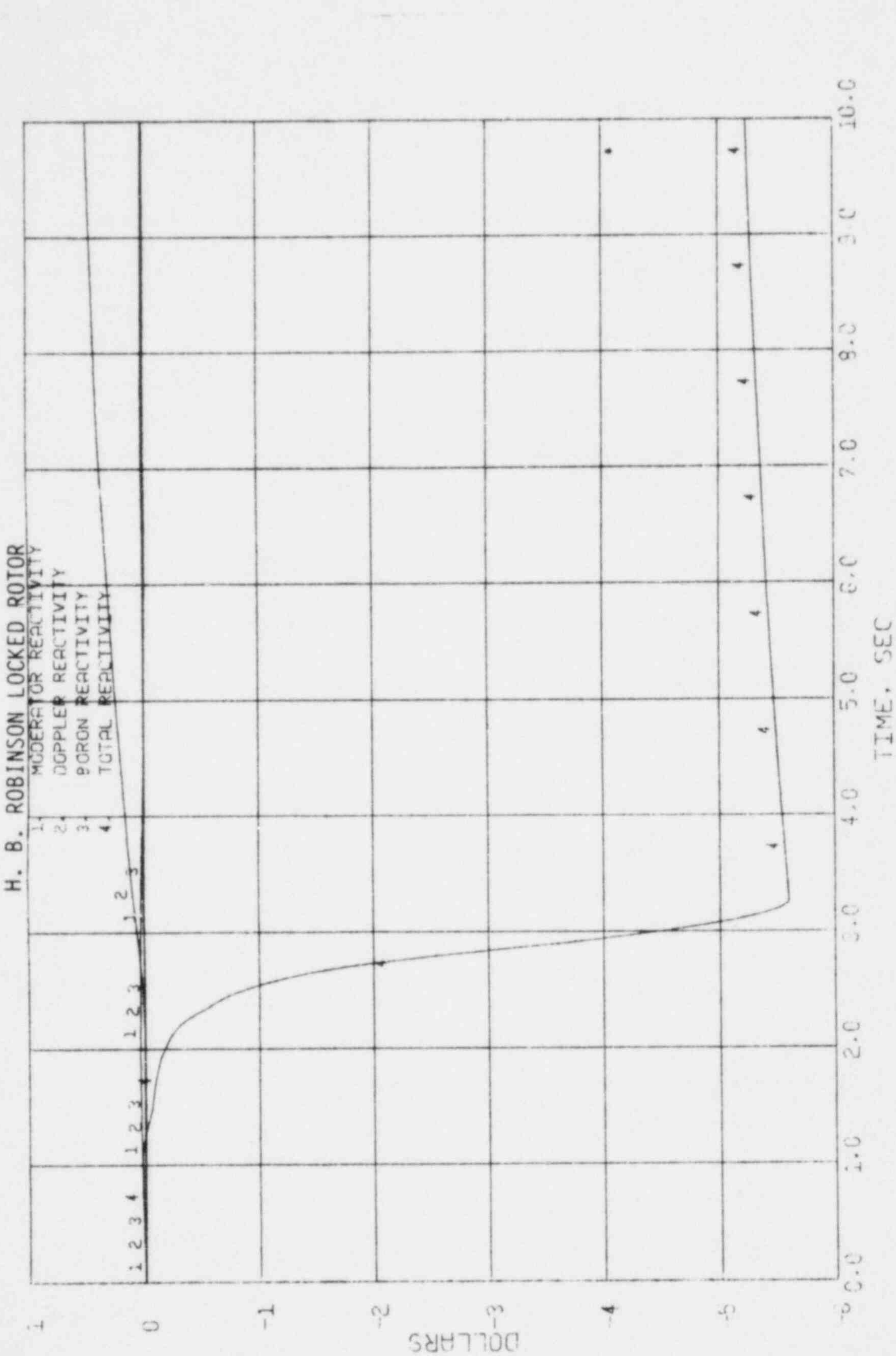

Figure 3.34 Reactivity worth for coolant pump seizure.

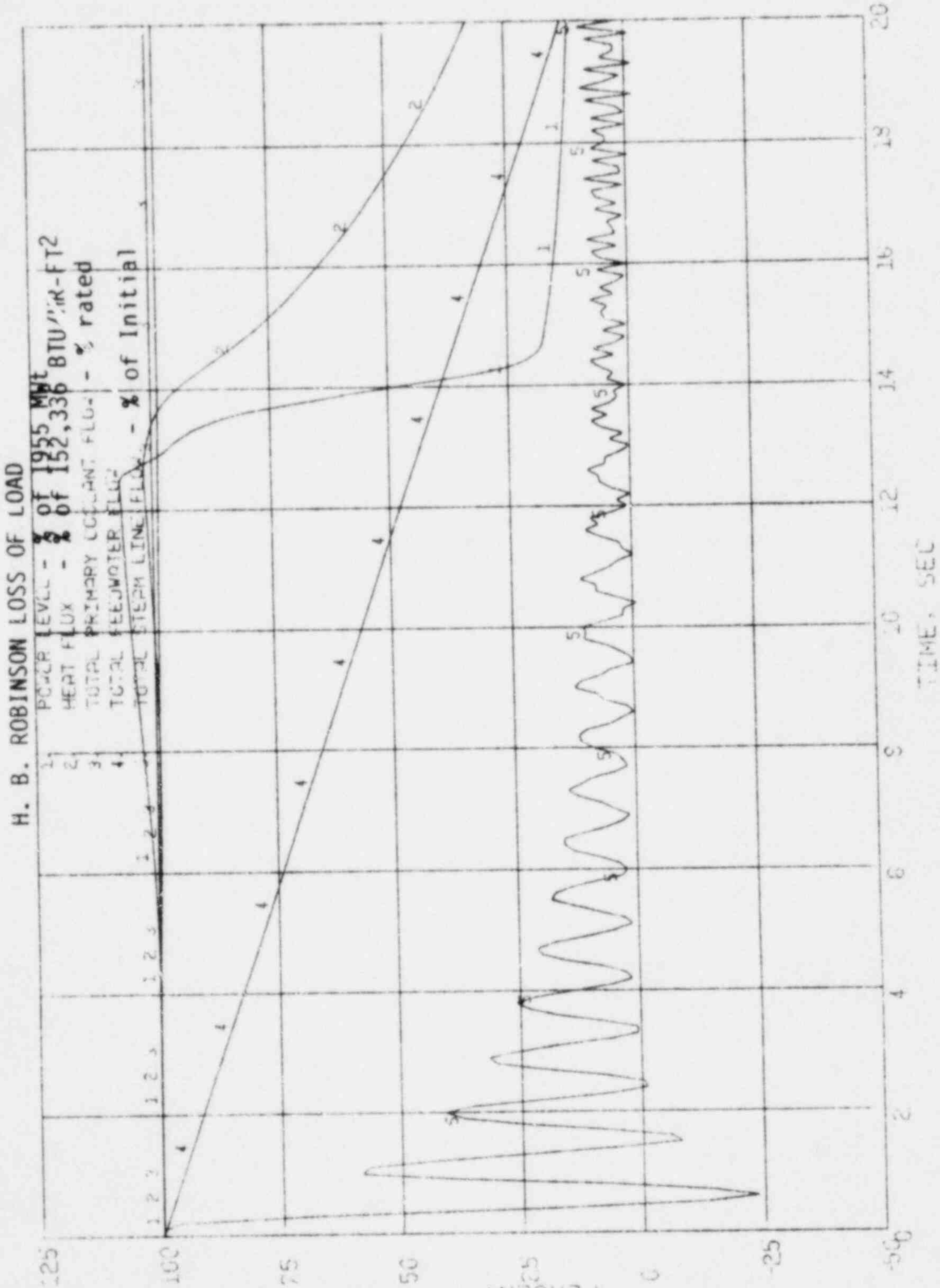

Figure 3.35 Power, heat flux and system flows for loss of load.

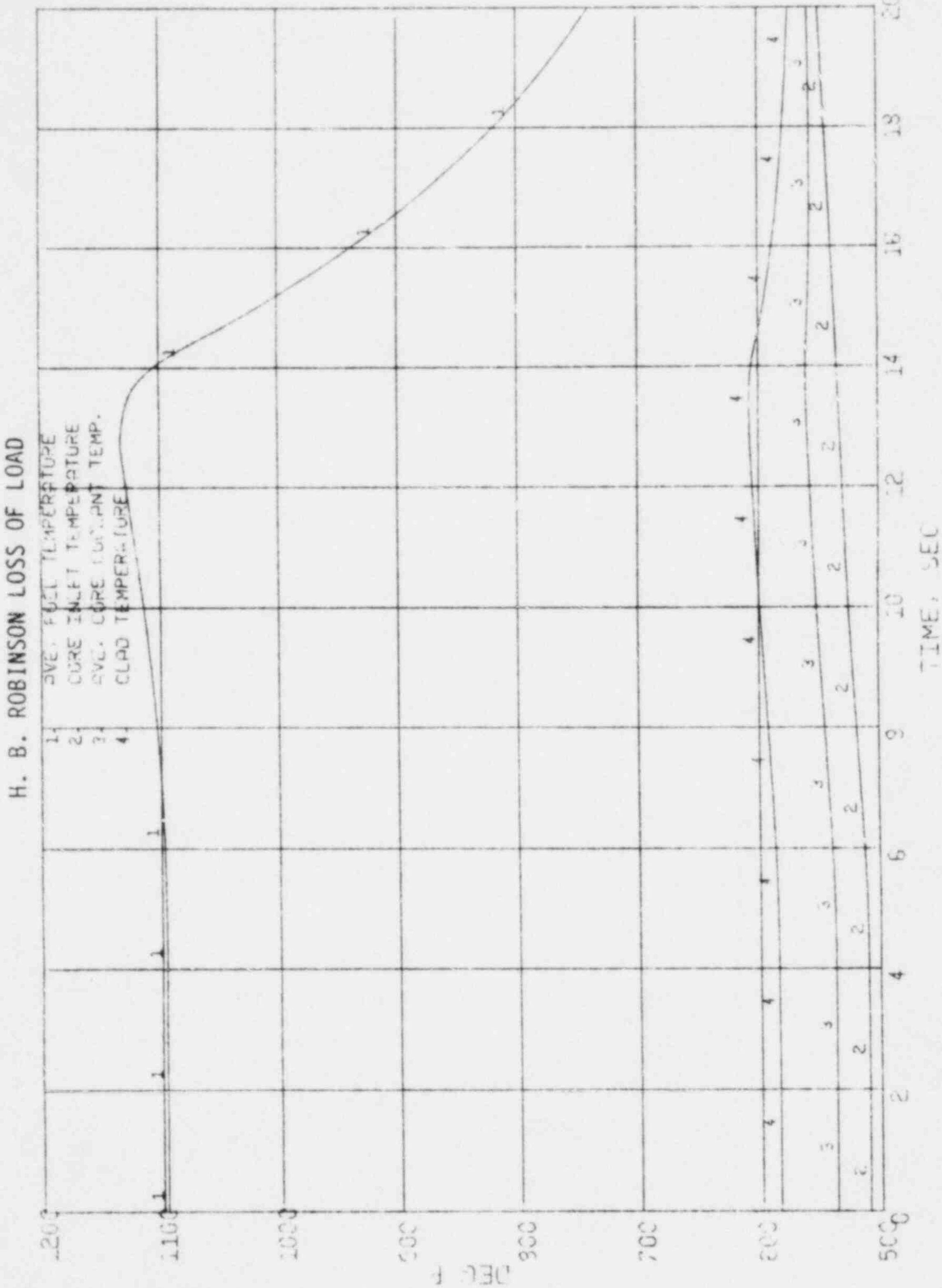

Figure 3.36 Core temperature responses for loss of load.

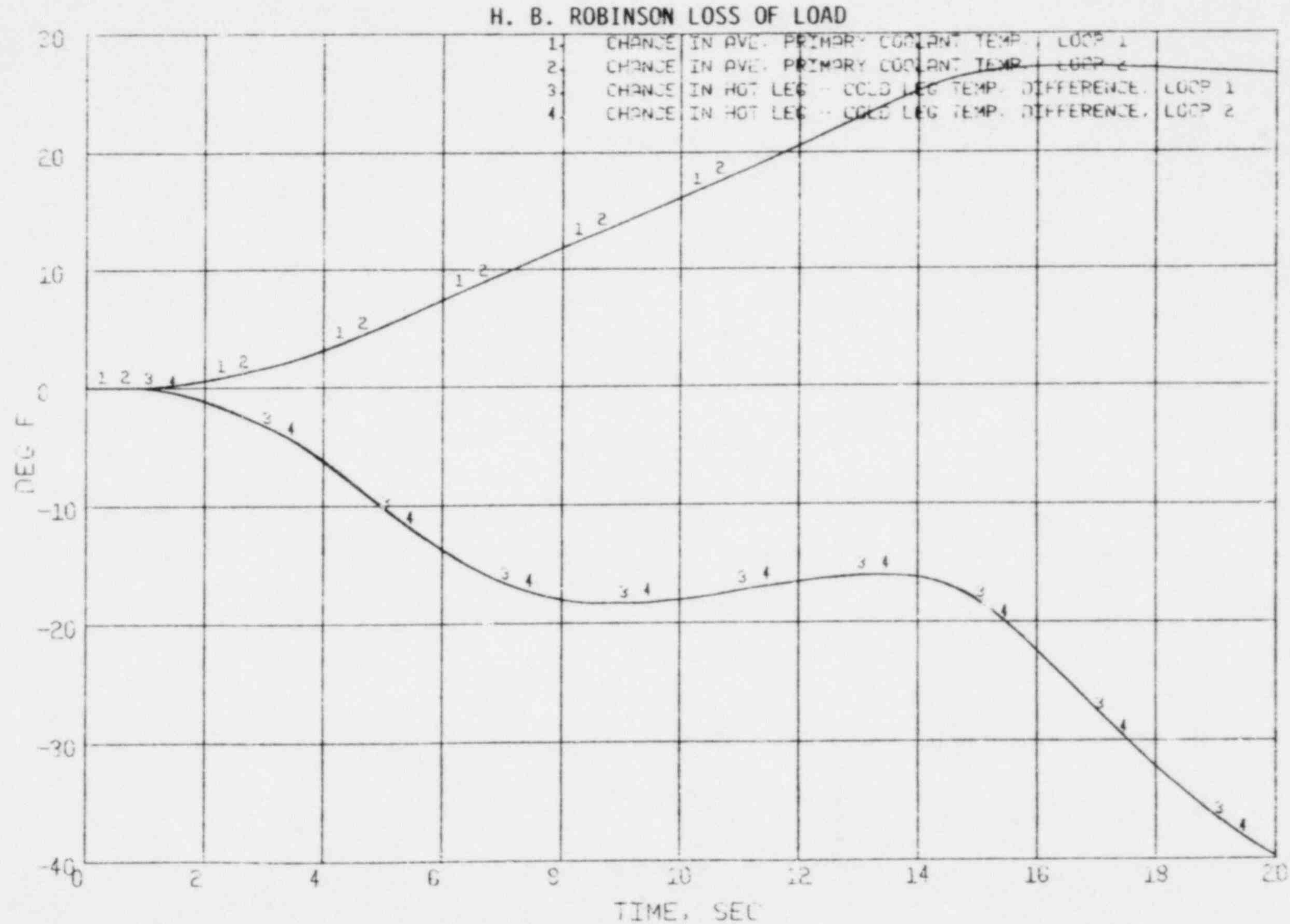

Figure 3.37 Primary loop coolant temperature changes for loss of load.

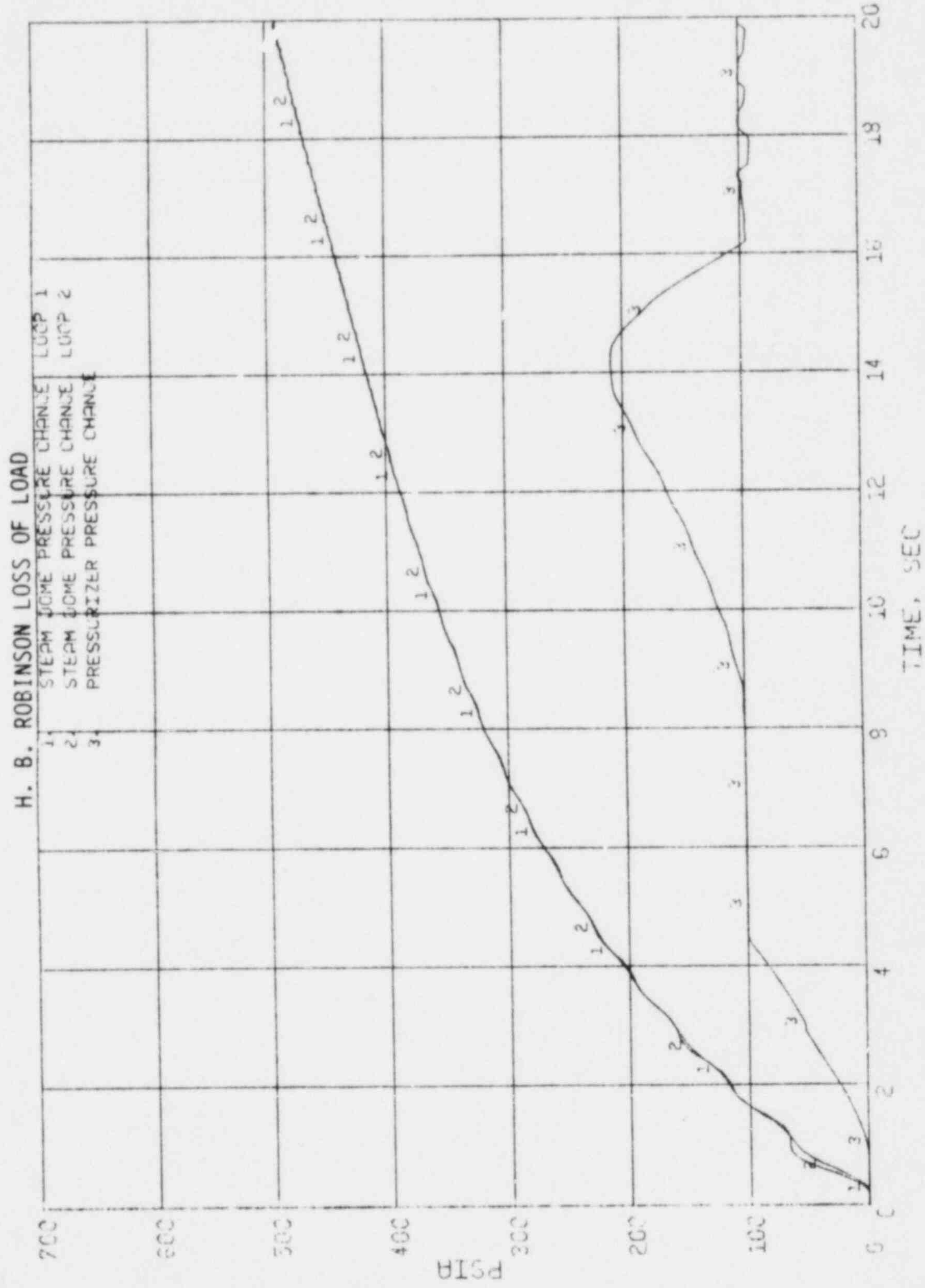

Figure 3.38 Pressure changes in pressurizer and steam generators for loss of load.

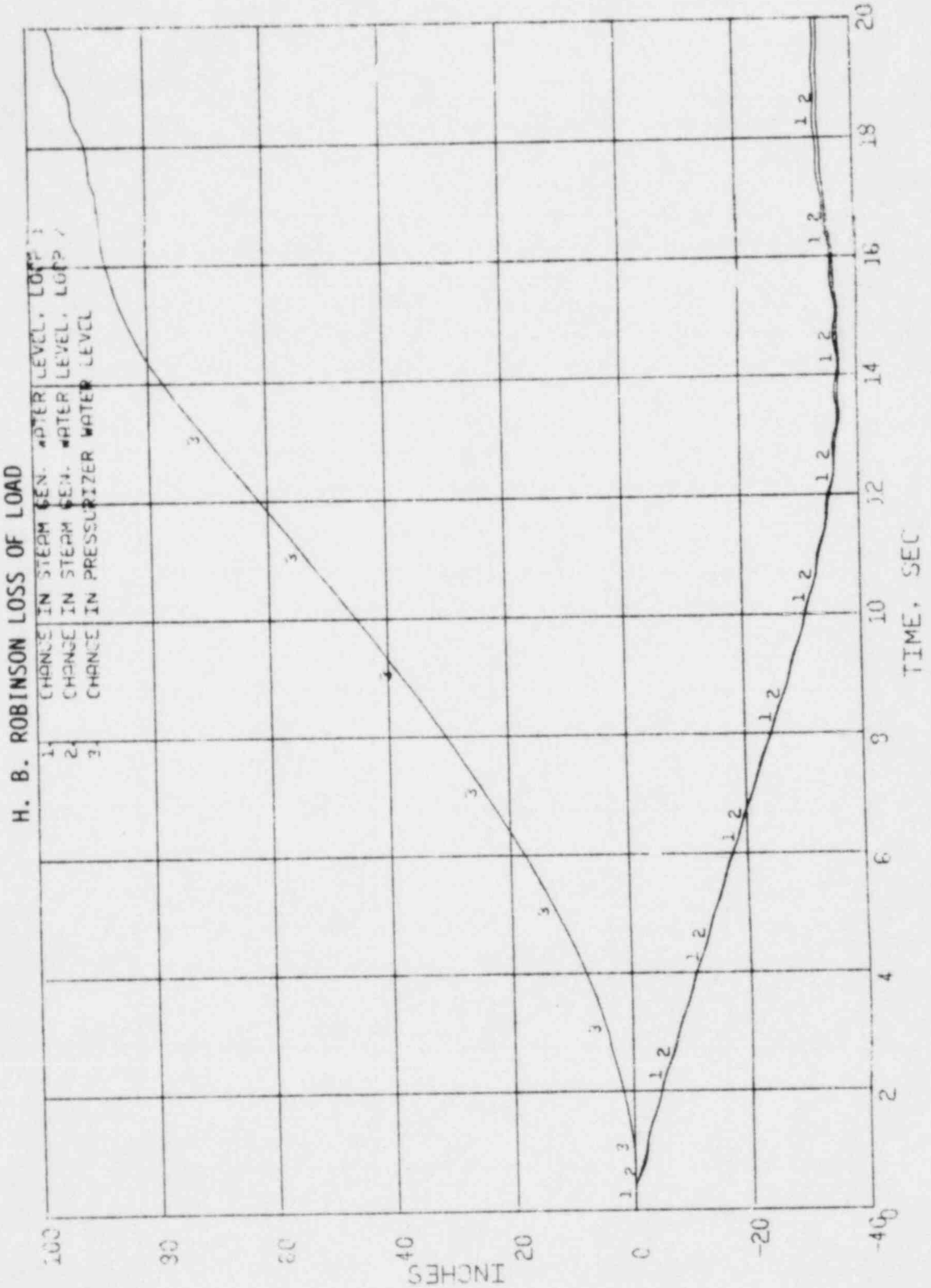

Figure 3.39 Level changes in pressurizer and steam generators for loss of load.

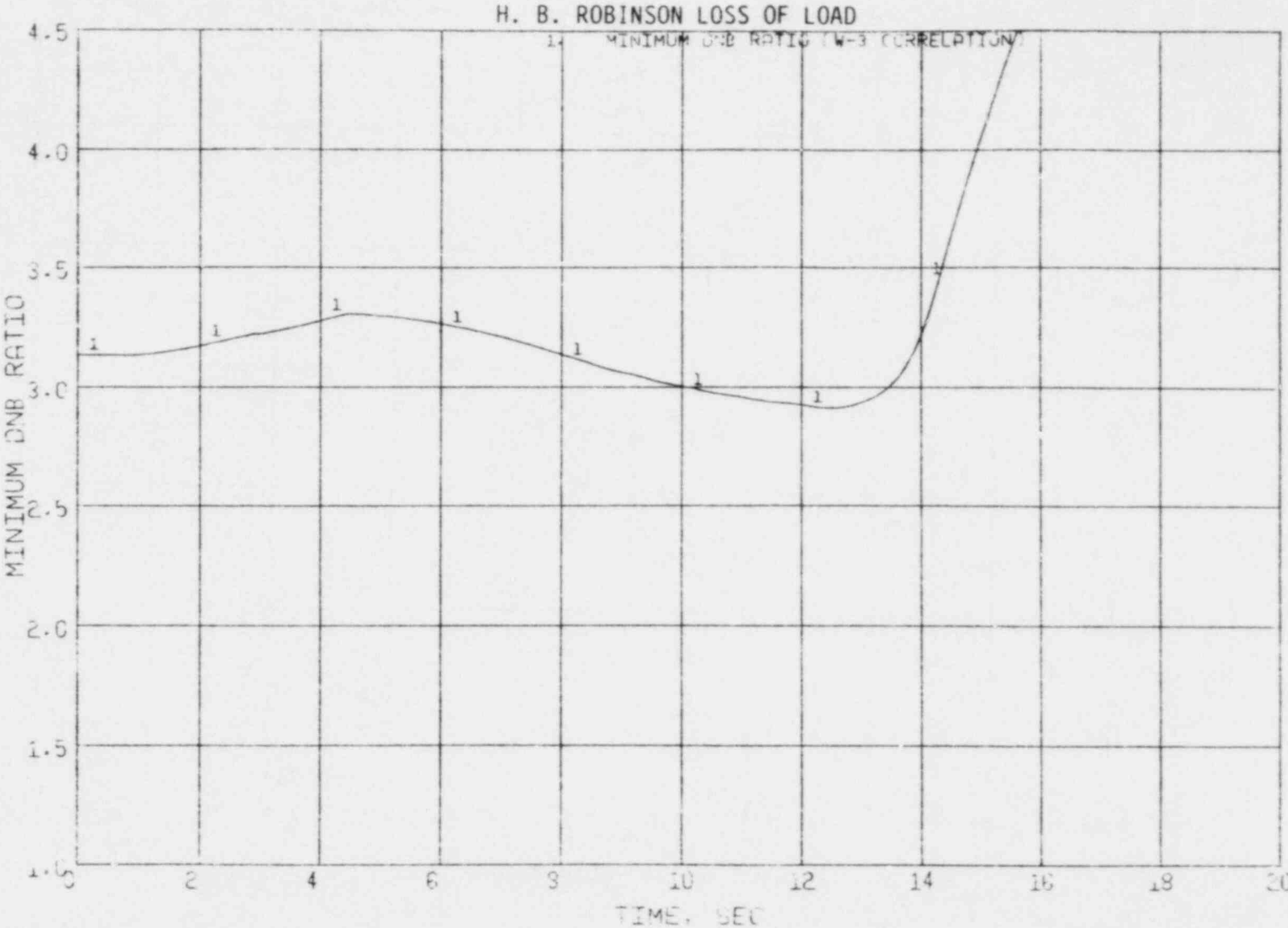

Figure 3.40 Minimum DNB ratio for loss of load.

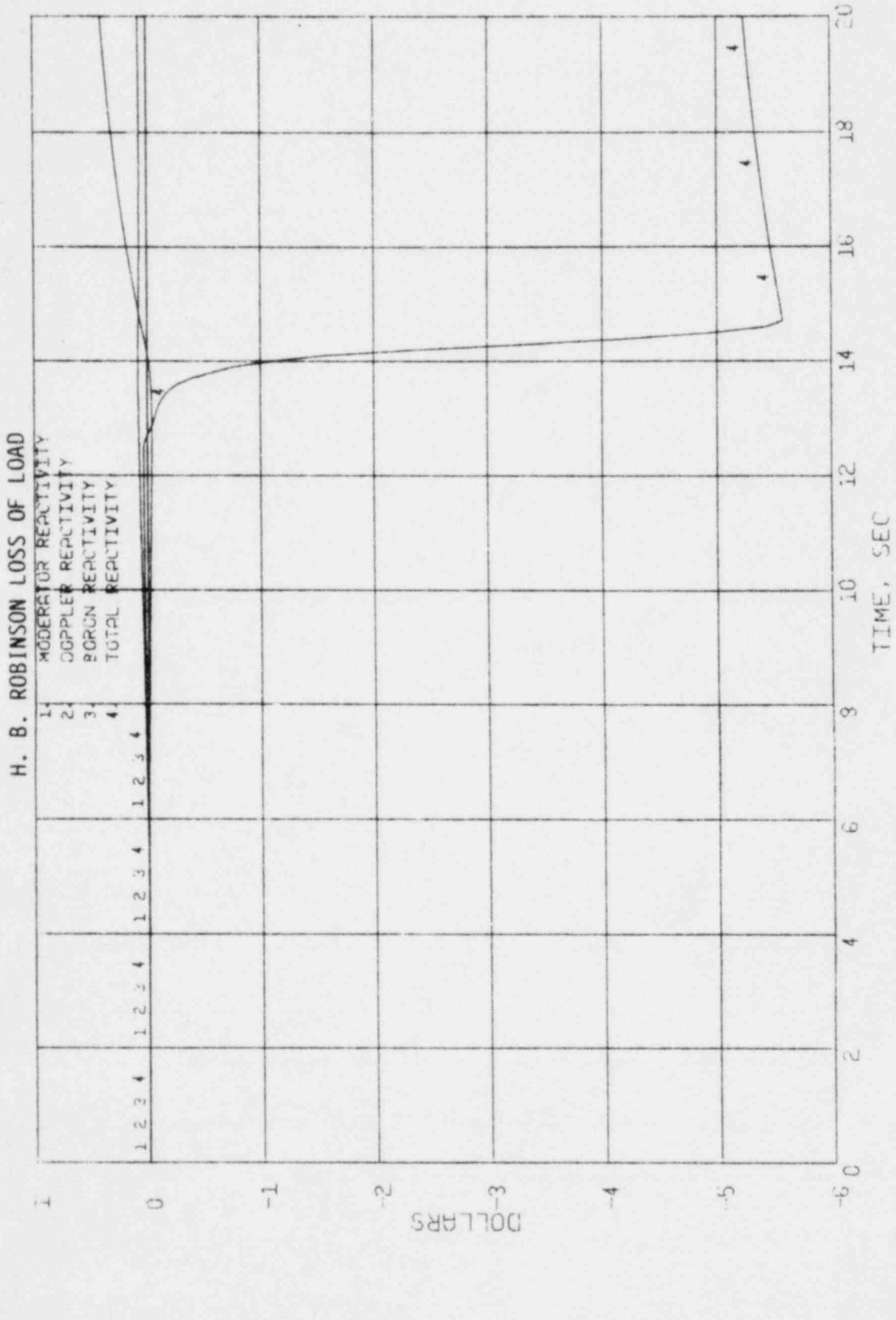

Figure 3.41 Reactivity worth for loss of load.

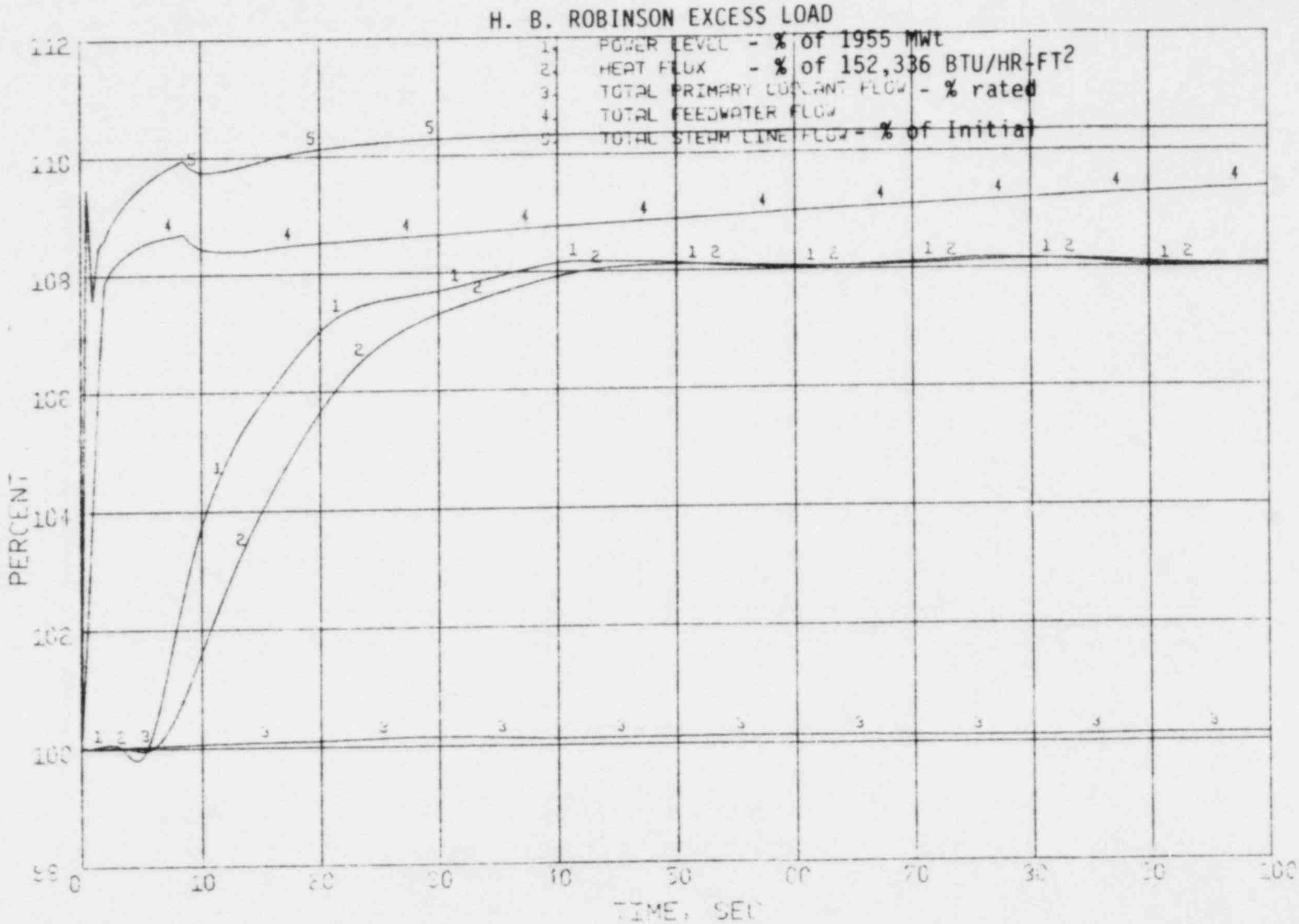

Figure 3.42 Power, heat flux and system flow for excess load.

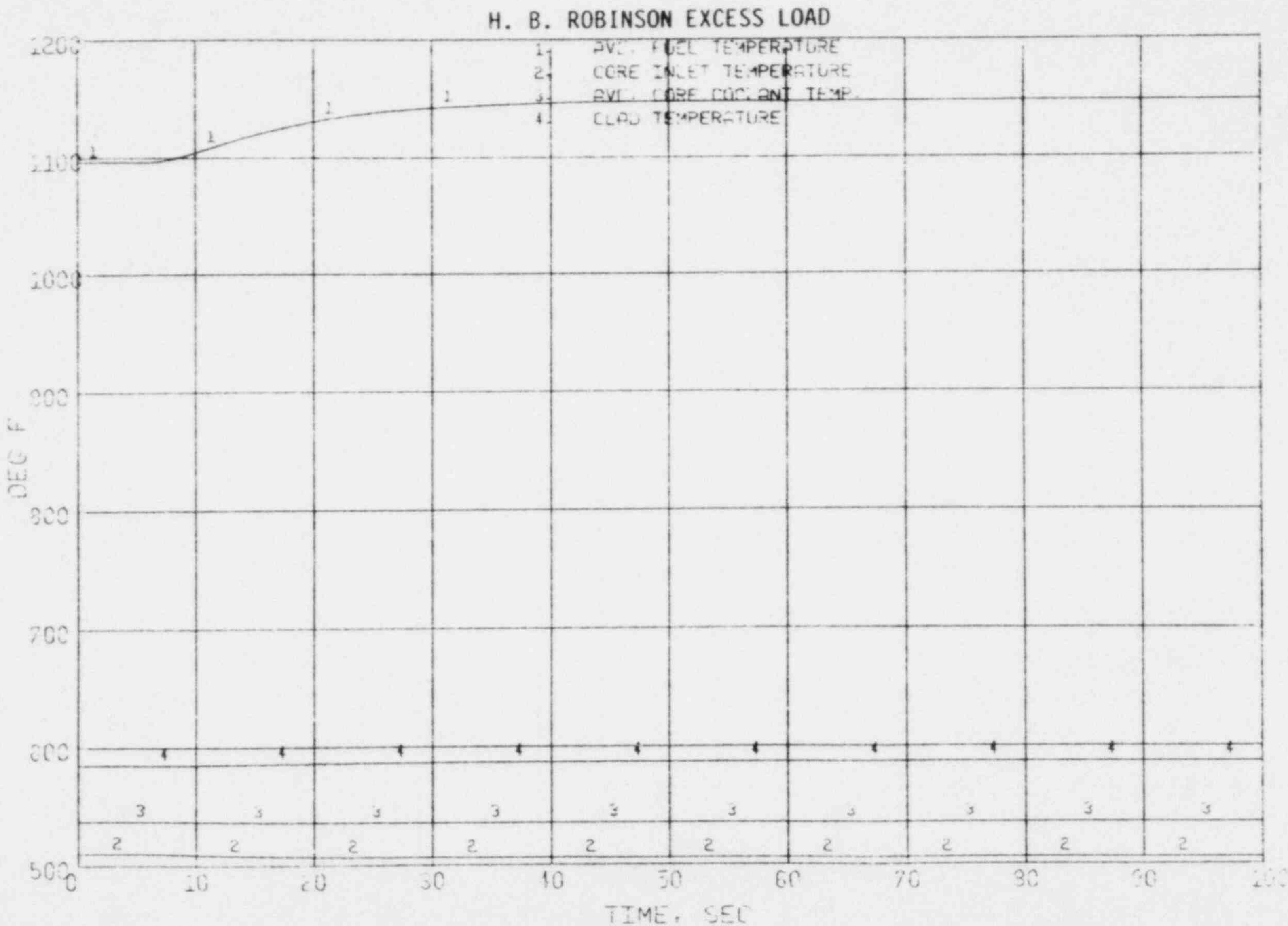

Figure 3.43 Core temperature responses for excess load.

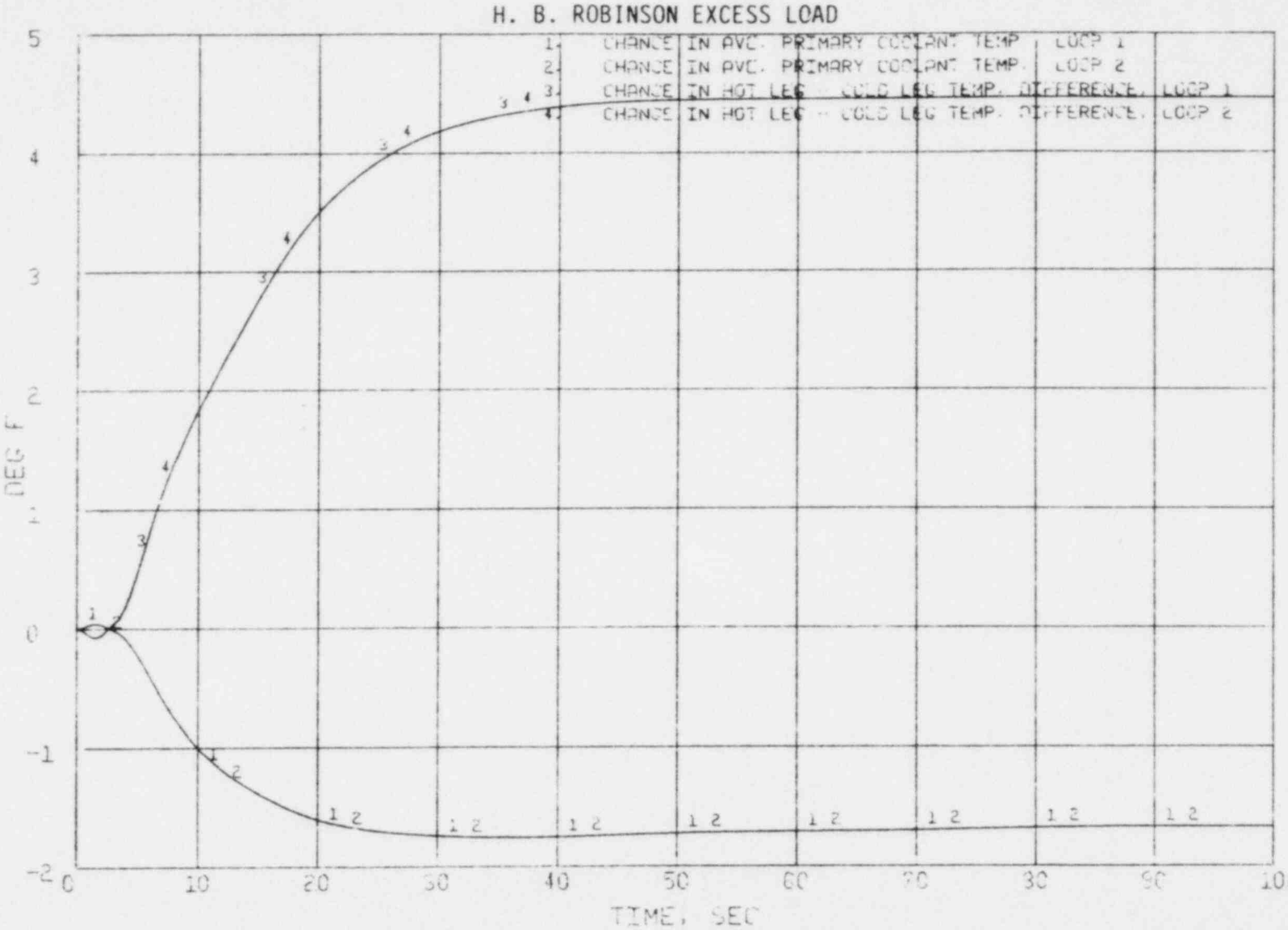

Figure 3.44 Primary loop coolant temperature changes for excess load.

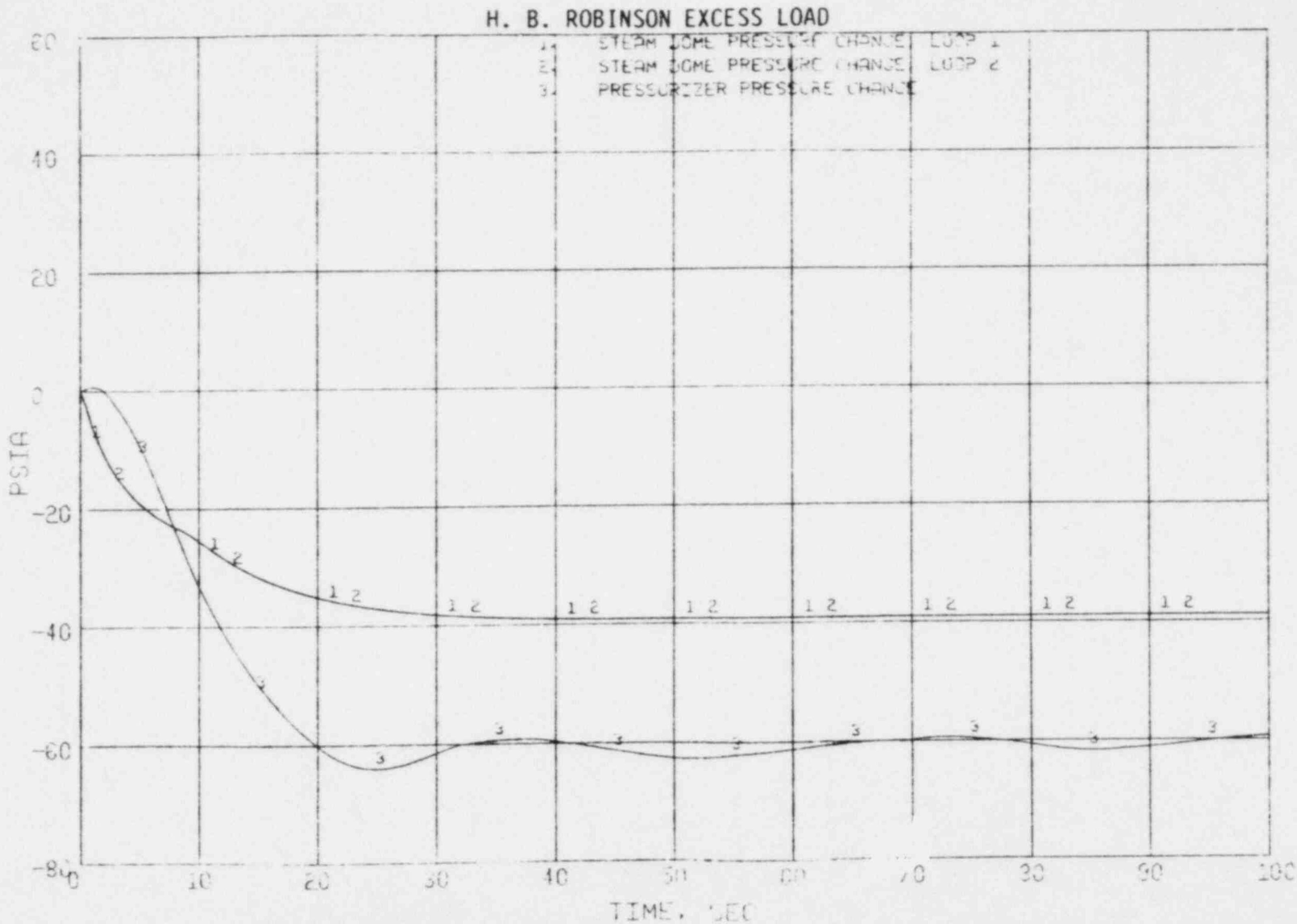

Figure 3.45 Pressure changes in pressurizer and steam generators for excess load.

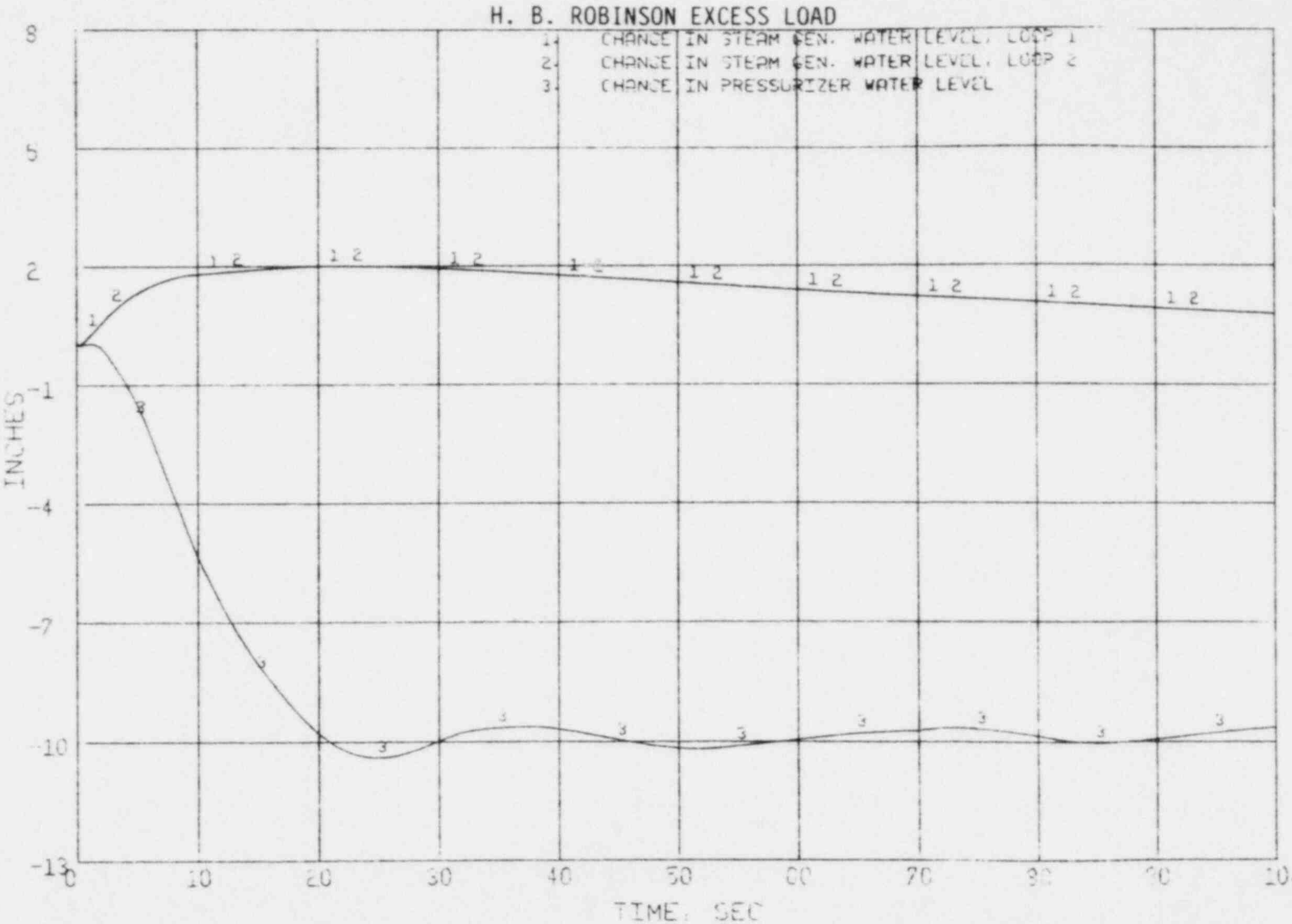

Figure 3.46 Level changes in pressurizer and steam generators for excess load.

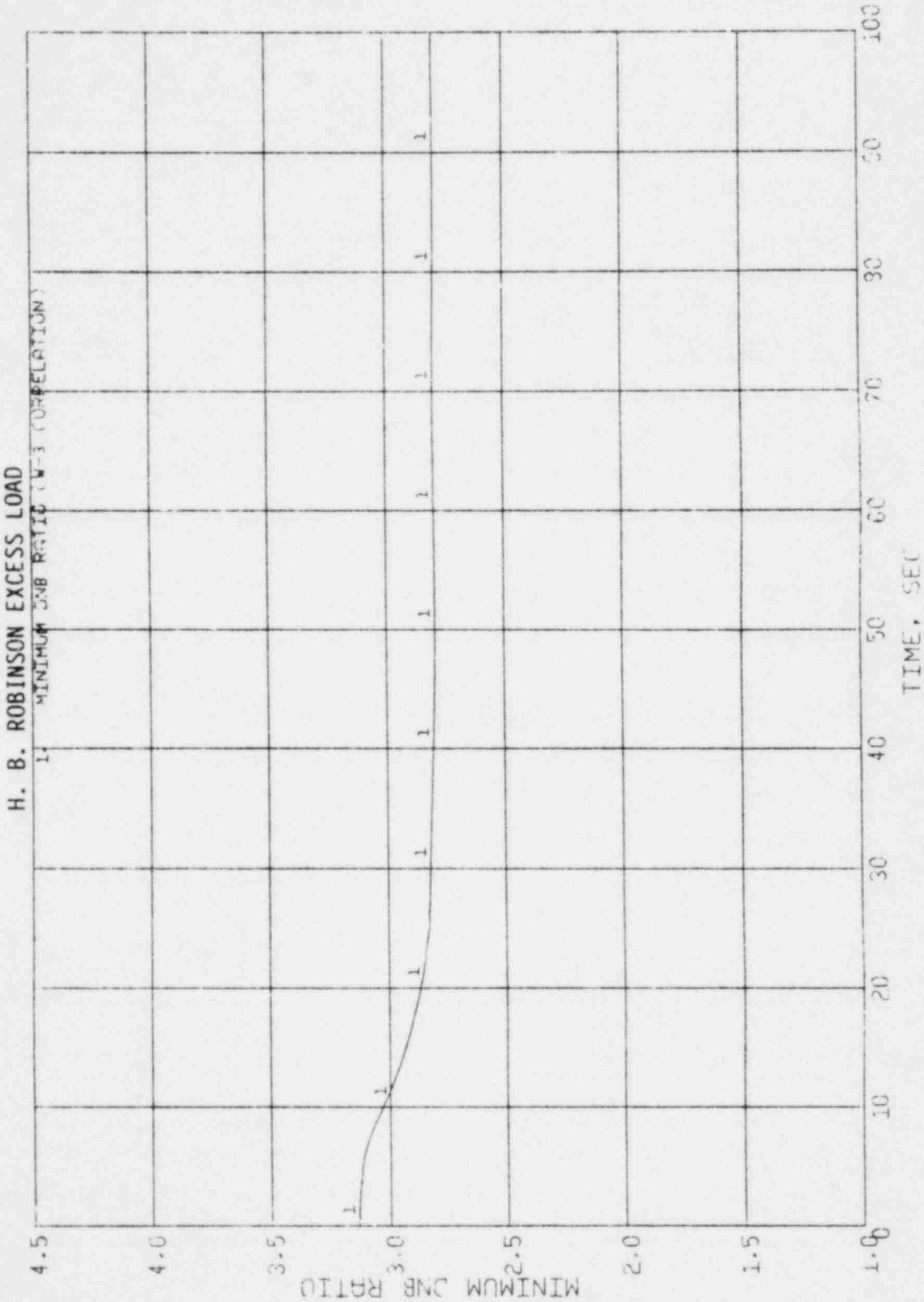

Figure 3.47 Minimum DNB ratio for excess load.

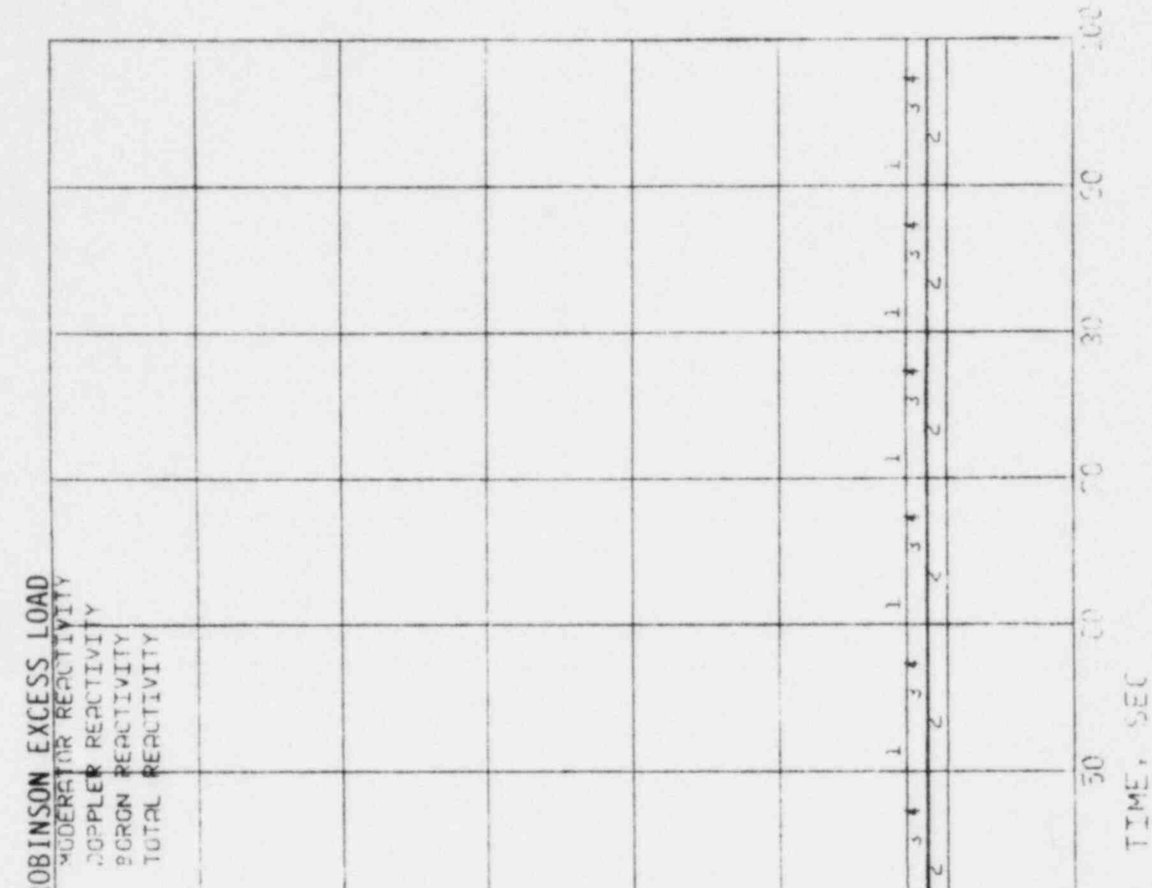

Figure 3.48 Reactivity worth for excess load.

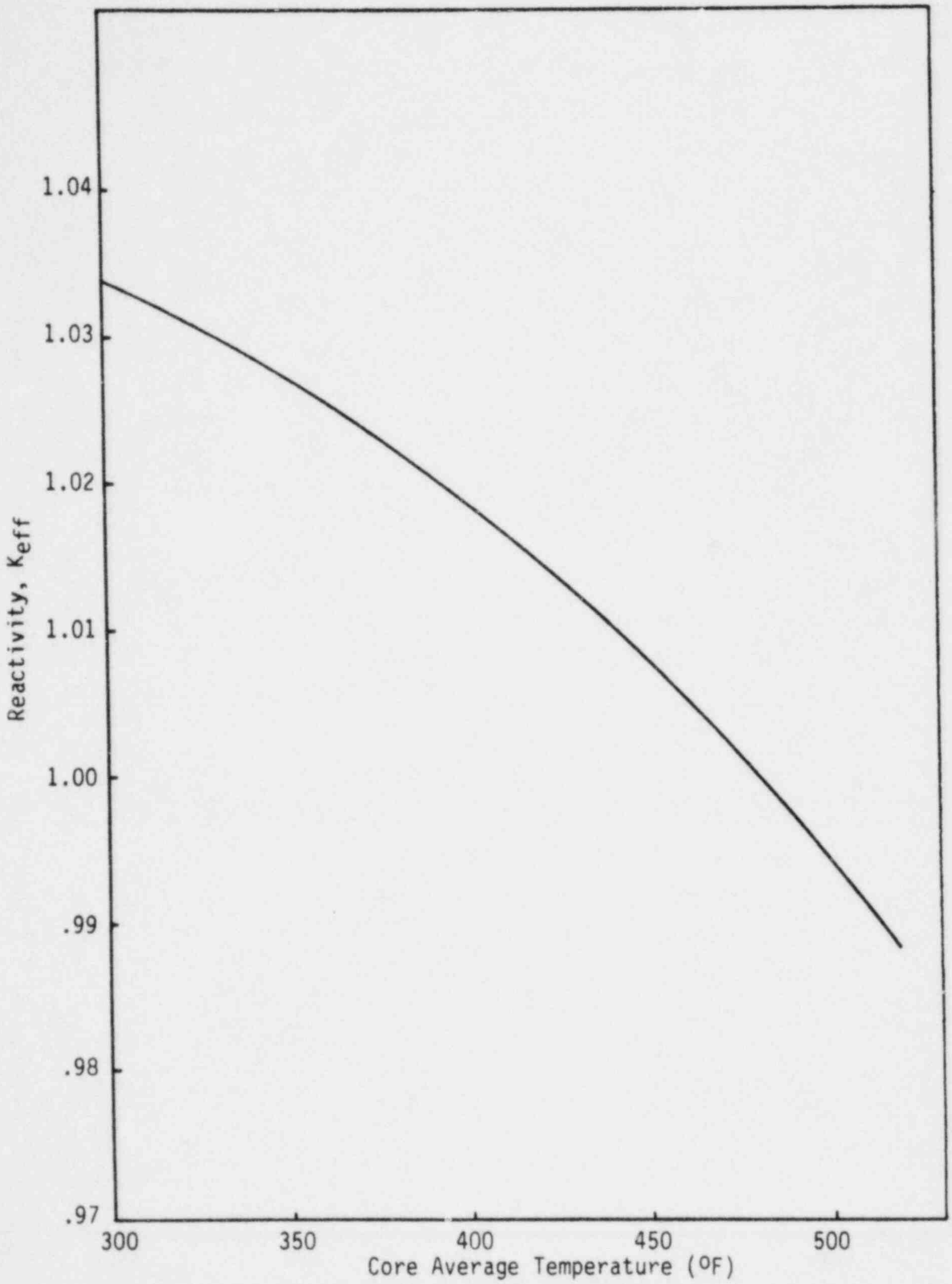

Figure 3.49 Variation of reactivity with core average temperature at EOC.

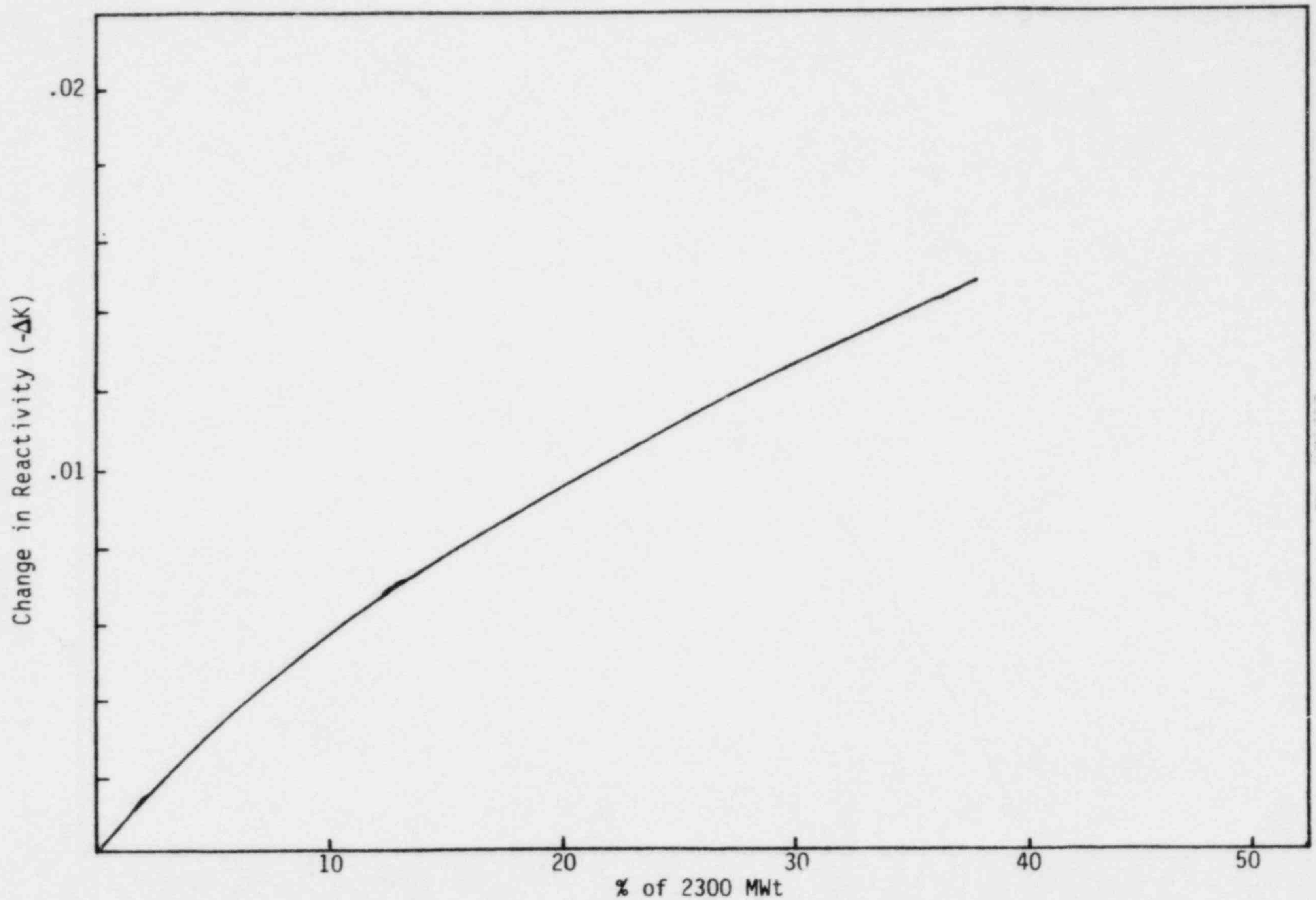

Figure 3.50 Variation of reactivity with power at constant core average temperature.

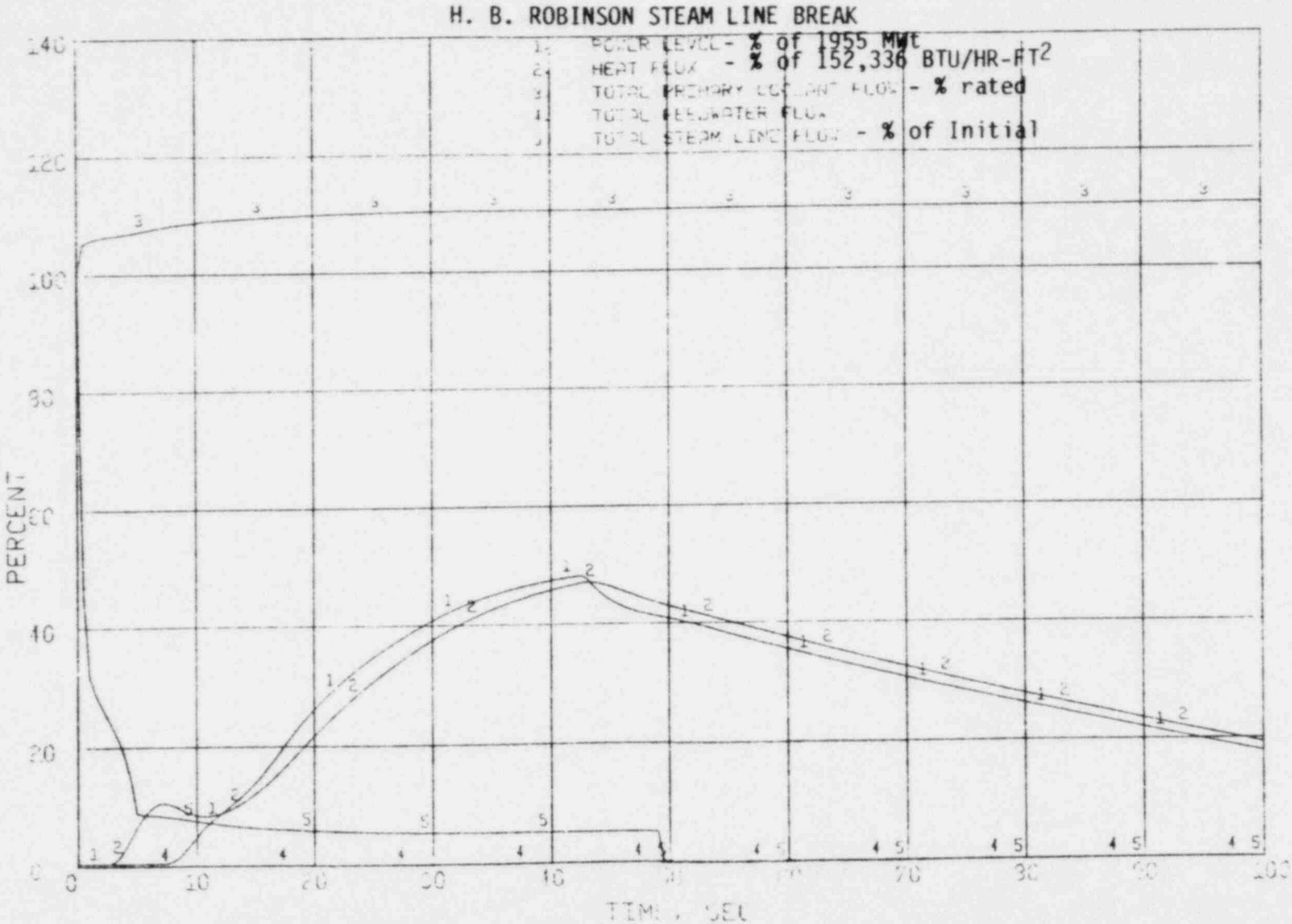

Figure 3.51 Power, heat flux, and system flows for H. B. Robinson steam line break.

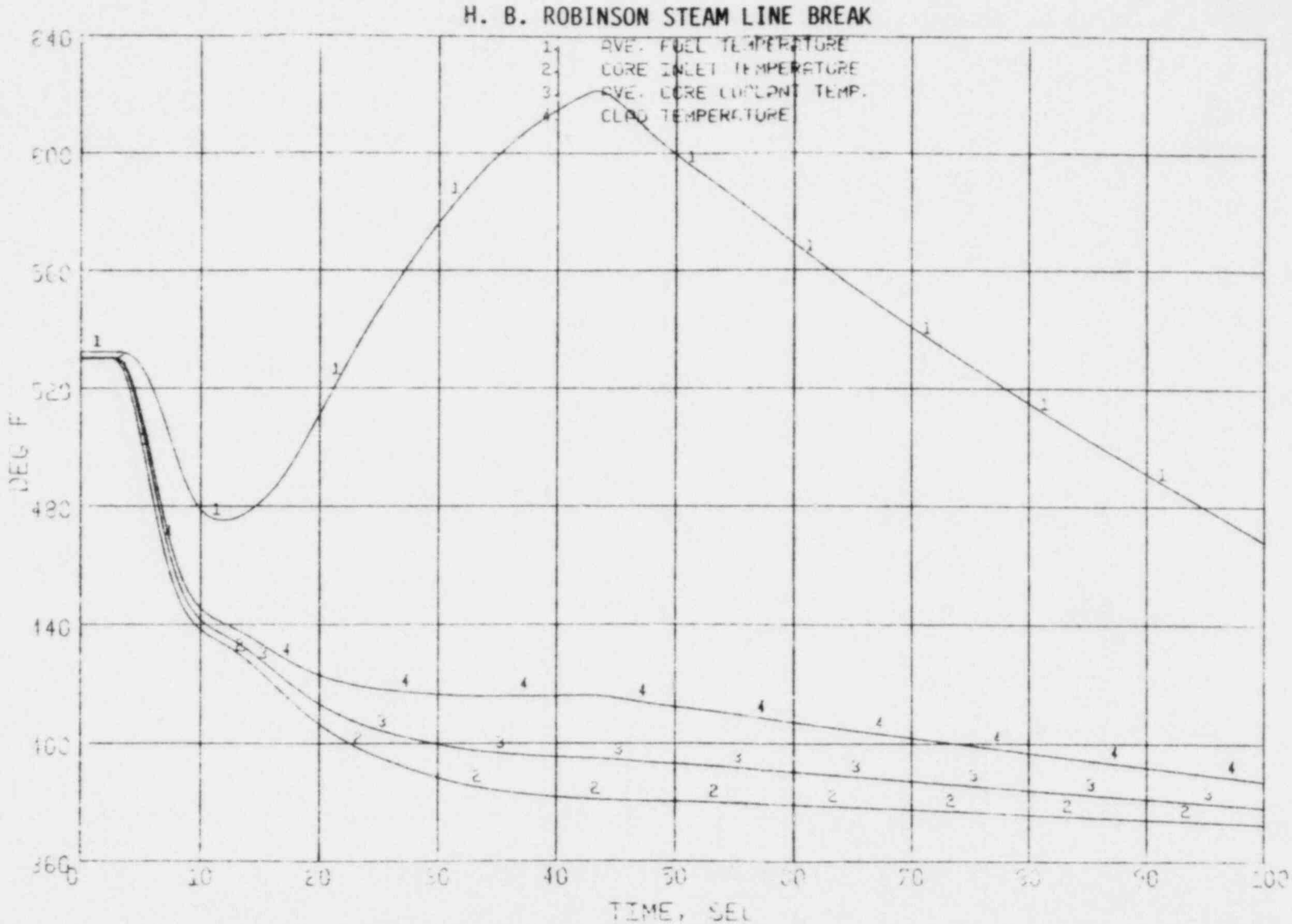

Figure 3.52 Core temperature response for H. B. Robinson steam line break.

H. B. ROBINSON STEAM LINE BREAK

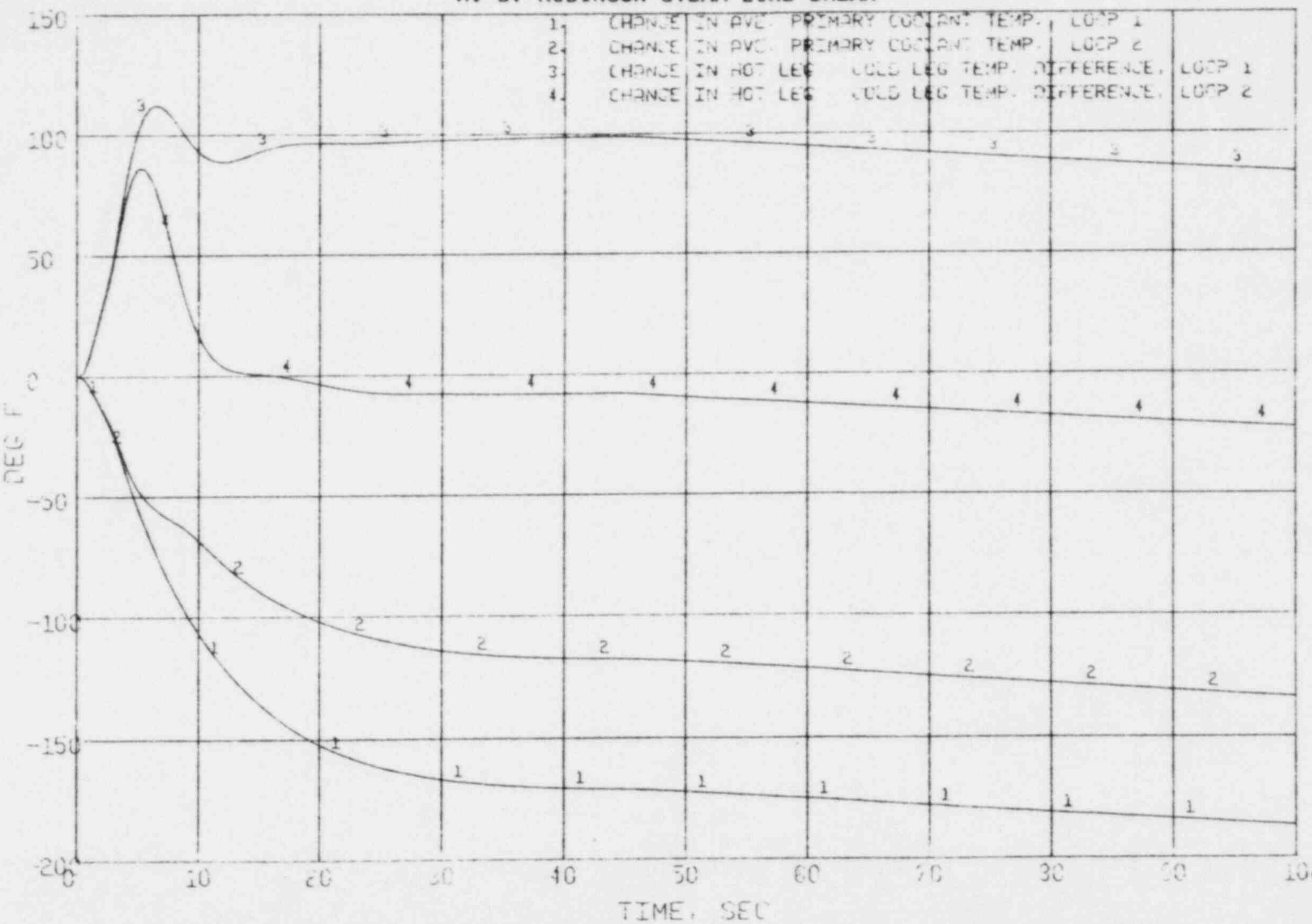

Figure 3.53 Primary system temperature changes for H. B. Robinson steam line break.

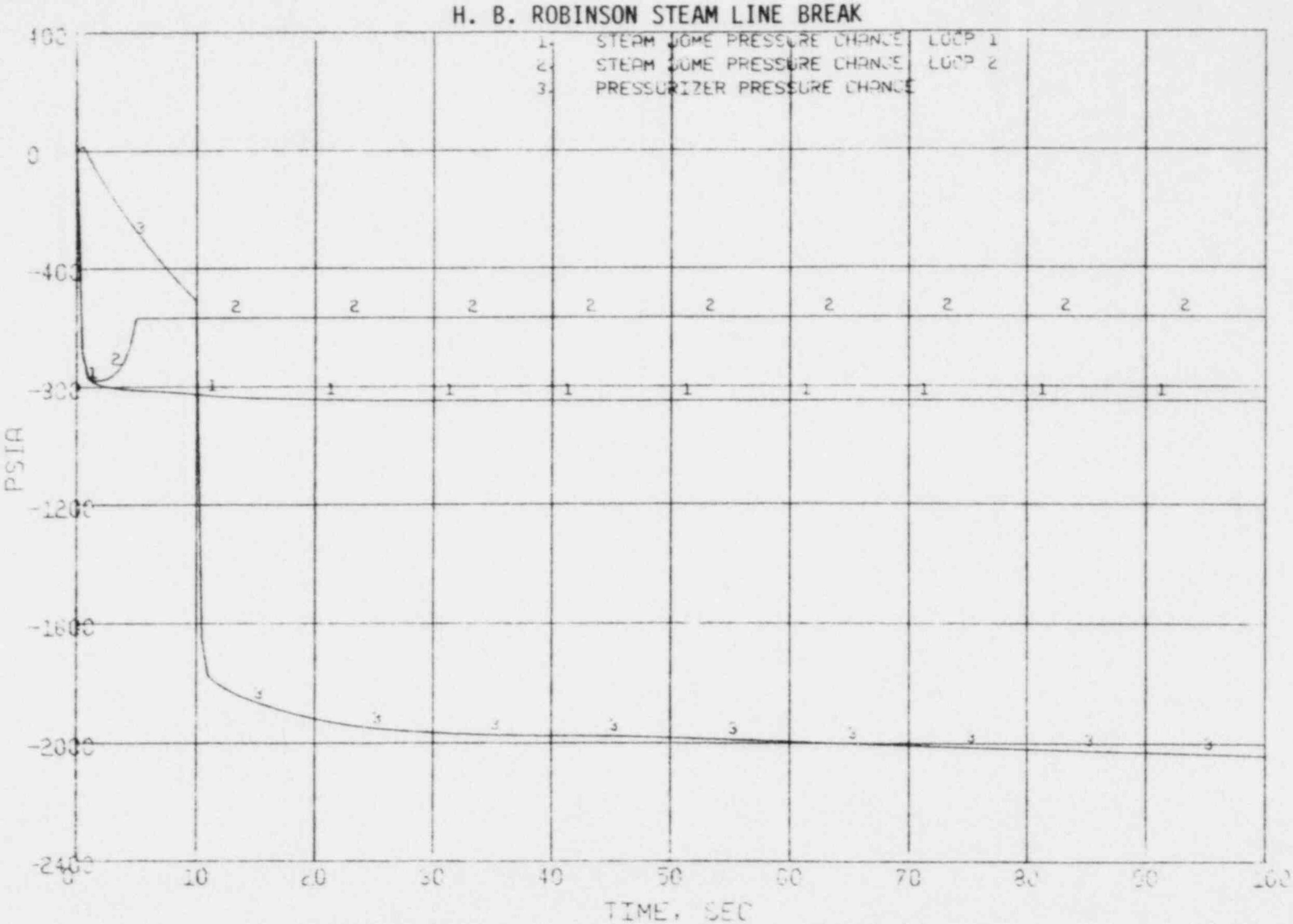

Figure 3.54 Pressure changes for H. B. Robinson steam line break.

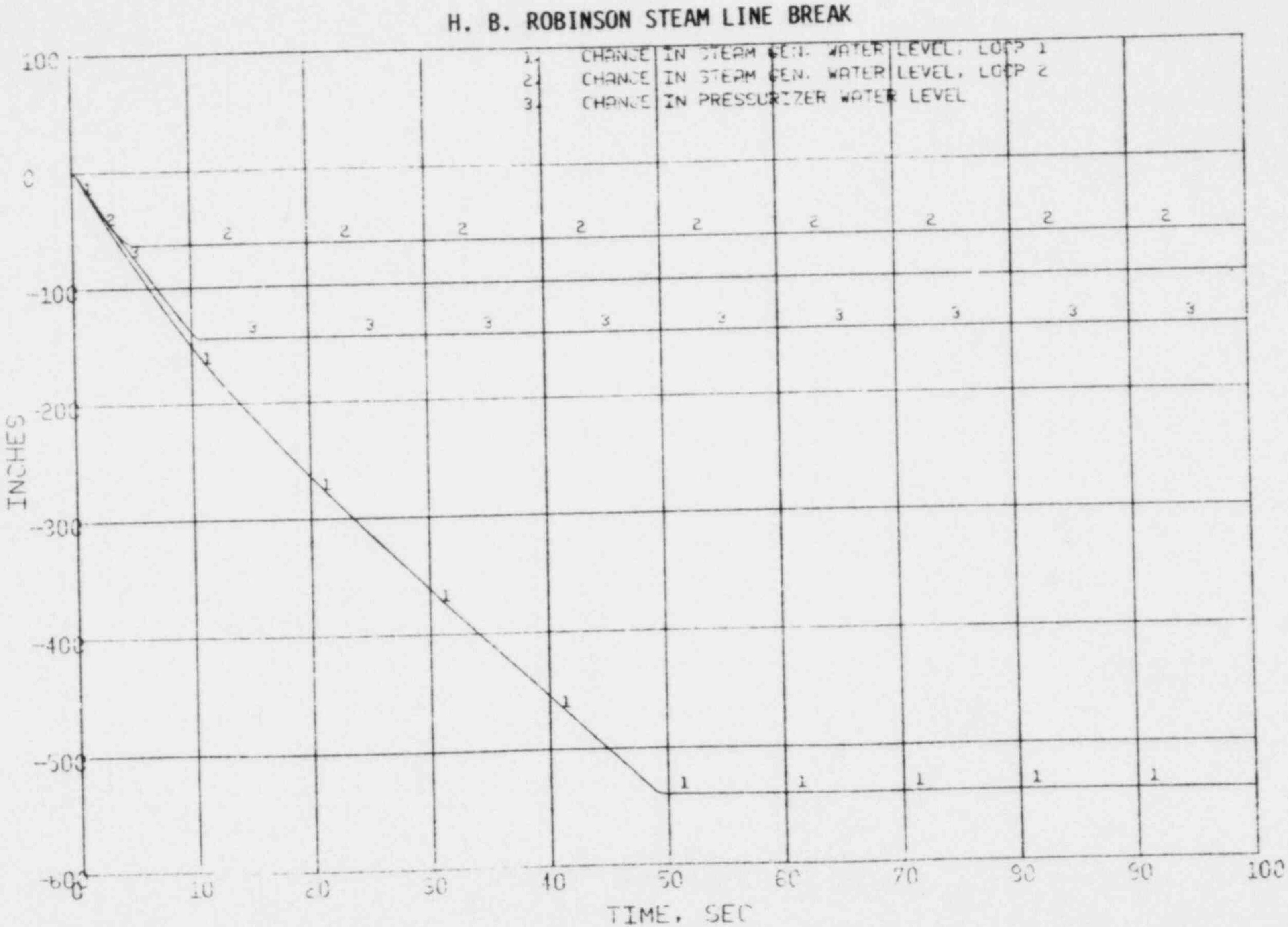

Figure 3.55 Level changes for H. B. Robinson steam line break.

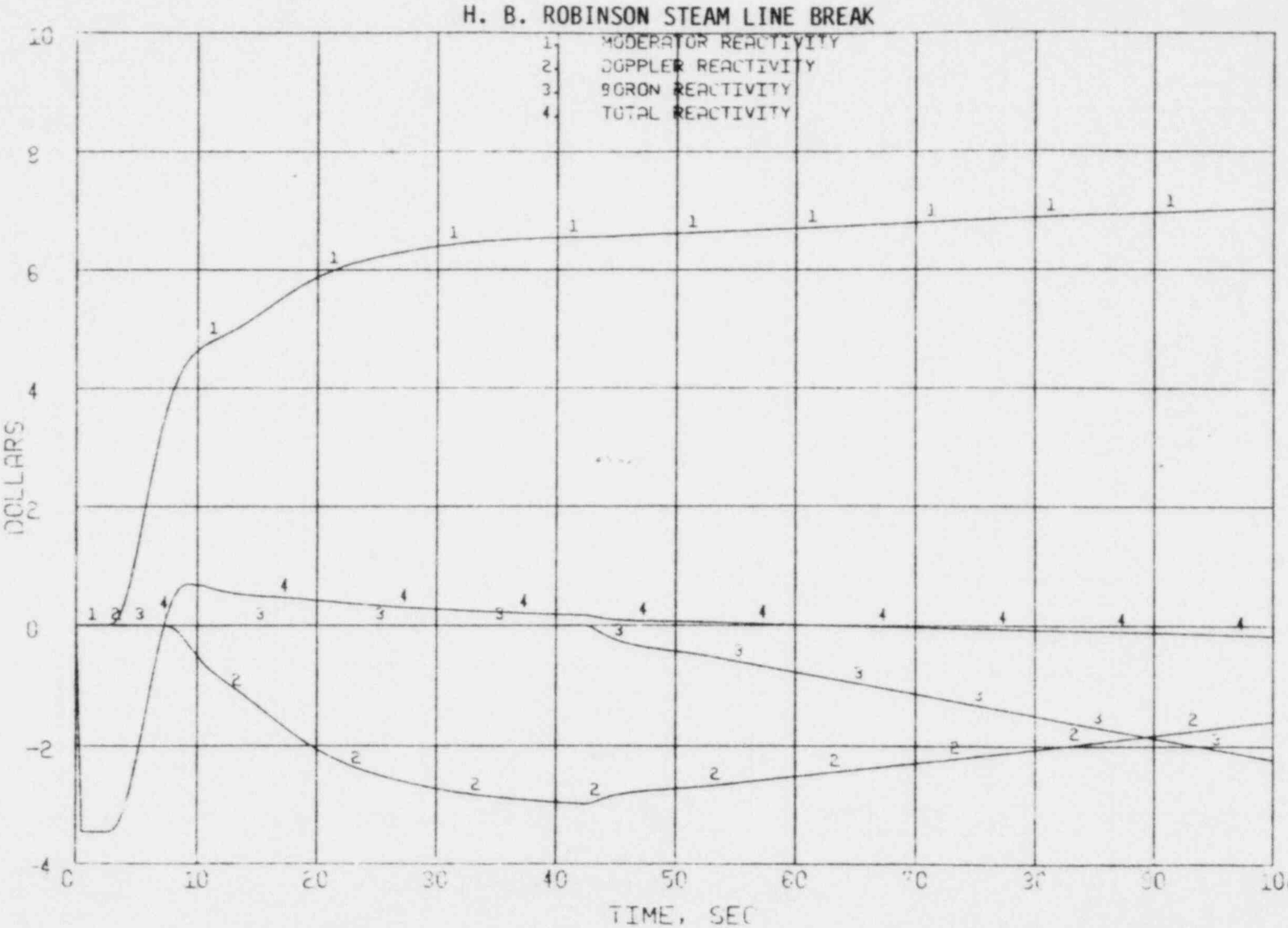

Figure 3.56 Core reactivity response for H. B. Robinson steam line break.

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6. E.D. Hughes, IN-1412, "A Correlation of Rod Bundle Critical Heat Flux for Water in the Pressure Range 150 to 725 psia".

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