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RECIP.NAME RECIPIENT AFFILIATION
VARGA,S.A. Operating Reactors Branch 1

SUBJECT: Submits revised ECCS analysis.Unit 2 can operate at 10% tube plugging at existing limits & remain within 10CFR50.46 criteria w/significant margin.Two Exxon Nuclear Co repts re ECCS analysis encl.

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CP&L
Carolina Power & Light Company

August 5, 1980

File: NG-3514(R) Serial No.: NO-80-1156

Office of Nuclear Reactor Regulation
Attention: 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
REVISED EMERGENCY CORE COOLING SYSTEM ANALYSIS

Dear Mr. Varga:

Carolina Power & Light Company (CP&L) has reperformed its Emergency Core Cooling System (ECCS) analysis, as described in its letter of July 22, 1980. The new analysis was performed using Exxon Nuclear Company's (ENC) WREM-IIA ECCS evaluation model, assuming ten percent (10%) of the steam generator tubes were plugged. The ENC WREM-IIA model has previously been reviewed and accepted by the NRC. The old ECCS analysis was performed assuming six percent (6%) steam generator tube plugging and using ENC's WREM-I model. Using ENC's newer ECCS model, CP&L has found that H. B. Robinson Unit No. 2 can operate with 10% tube plugging at existing limits, and remain within the Nuclear Regulatory Commission's (NRC) 10CFR50.46 criteria with significant margin.

In the new ECCS analysis, the value of F_Q was held constant at 2.2 and the large guillotine breaks were calculated with coefficients of 1.0, 0.8, and 0.6 applied to the break. The results of this analysis indicated that the previously identified limiting break, the double-ended cold leg guillotine break with a coefficient of 0.8 (0.8 DECLG), was found to remain limiting, and the peak clad temperature decreased from the old value of 2152°F to a new value of 1976°F. This reduction is due to the NRC approved modeling changes which have occurred between the old WREM-I model and the new WREM-IIA model. The most notable of the changes in the new model would be the decreased hot wall delay time and credit taken for the portion of the fluid in the cold leg during the reflood calculation.

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For your information, we have included as attachments, two ENC reports on the H. B. Robinson Unit No. 2 steam generator tube plugging ECCS/PTS Analysis. The first is a report on the plant transient analysis for the fifteen percent (15%) tube plugging which bounds the 10% plugging assumed for the LOCA analysis. The second is a report on the revised LOCA/ECCS analysis with 10% of the steam generator tubes plugged.

CP&L considers the new ECCS analysis to be the valid ECCS analysis for the operation of H. B. Robinson Unit No. 2 commencing with Cycle 8. We have further determined it involves no unreviewed safety questions and will require no changes in the Unit Technical Specifications. Therefore, refueling for Cycle 8 will be performed in accordance with the authorization of 10CFR50.59(a). Tentatively, the refueling outage is scheduled to begin on August 8, 1980 and be completed within approximately seven weeks.

If you have any questions concerning these matters, please contact our staff.

Yours very truly,

E. E. Utley
Executive Vice President
Power Supply and
Engineering & Construction

JHE/jc (101-214)
Enclosures

cc: Mr. J. D. Neighbors (NRC)

H.B. ROBINSON - PTS ANALYSIS FOR 15% TUBE PLUGGING

By R. B. Macduff - Licensing & Safety Engineer

A plant transient analysis of H.B. Robinson Unit 2 was performed to assess the impact of 15% steam generator tubes plugged on thermal margin. The analysis showed that tube plugging of the magnitude considered did not significantly impact thermal margin.

The plant transients analyzed were those previously identified by ENC(1) to be the most severe with respect to thermal margin, i.e.

(1) Locked Rotor at BOC

(2) Loss of Reactor Coolant Flow at BOC

(3) Slow Rod withdrawal at BOC

(4) Fast Rod withdrawal at BOC

The transients were analyzed with the plant parameters listed in Table 1 and with neutronics parameters listed in Table 2. Conservative multipliers were applied to the reactivity coefficients. A value of 1.25 times the moderator coefficient and 0.8 times the Doppler coefficient were used. This is consistent with previous ENC application. The core flow rate shown in Table 1 was used in the analysis presented in References (1) and (3) and the current analysis. This value was less than the flow predicted by Westinghouse(2) for as much as 15% of the tubes plugged. As a consequence temperatures, flows and pressures of the primary remained the same as those used in previous Exxon(1) analyses. . Only secondary conditions for temperature and pressure were adjusted to account for a 15% reduction in tube heat transfer area.

The results of the analyses are summarized in Table 3. The minimum MDNBR was 1.58 for the locked rotor transient. This result is well

above the MDNBR limit of 1.30*.

Previous ENC analyses had considered additional plant transients. A review of these transients shows that they were not as limiting as those selected for analysis. The ones selected have shown only a small effect on thermal margin (less than 1%) due to plugging. These less severe accidents would not be expected to behave differently from those analyzed.

Based on the results of this analysis, it is concluded that for up to 15% tube plugging only a small change occurs to MDNBR with the resulting MDNBR still greatly higher than the limiting MDNBR of 1.30.

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* DNBR values less than 1.30 are acceptable for the locked rotor transient as it is a class III incident for which a small amount of fuel damage could be accepted.

REFERENCES

1. Kahn, J.D., "Plant Transient Analysis of the H.B. Robinson Unit 2 PWR for 2300 MWT, XN-75-14, July 15, 1975.

2. Letter from Mr. J.D. Martin of CP&L to Mr. T.J. Helbling of ENC dated June 10, 1980. Serial number NF-80-353, with enclosure from Mr. R.S. Longdon of W to Mr. R.B. Starkey of CP&L, presenting results of W study related to tube plugging at H.B. Robinson Unit 2.

3. Cherng, J.C. "Review of Plant Transient Analysis for Positive Moderator Temperature Reactivity Feedback for the H.B. Robinson Unit Number 2 Nuclear Plant", XN-NF-79-42, June 22, 1979.

TABLE 1

OPERATING PARAMETERS USED IN PTSPWR ANALYSES OF H.B. ROBINSON UNIT 2

PARAMETER	VALUE
CORE	
Total Core Heat Output, MWt	2346
Heat Generated in Fuel, %	97.4
Nominal System Pressure, psia	2220.
HOT CHANNEL FACTORS	
Heat Flux, Total	2.62
Enthalpy Rise, F_H^N	1.58
COOLANT FLOW RATE, lb/hr	101.5×10^6
COOLANT TEMPERATURE, °F	
Nominal Inlet	550.5
HEAT TRANSFER	
Average Heat Flux, BTU/hr ft^2 °F	182,800
STEAM GENERATORS	
Steam Flow, lb/hr	3.423×10^6
Steam Temperature, °F	517.2
Steam Pressure, psia	793
Feedwater Temperature, °f	441.7
PHYSICAL PARAMETERS	
S/G Heat Transfer Area ft^2	37765.
S/G Volume of Primary in Tubes ft^3	1741.2
Number of Tubes	2771.

TABLE 2

NEUTRONICS PARAMETERS*

PARAMETER	BOC	EOC
Moderator temperature coefficient (PCM/°F) (PCM=$10^{-5}\Delta\rho$)	+2.0	-32.0
Moderator pressure coefficient (PCM/psi)	-.02	0.0
Delayed neutron fraction (%)	0.600	0.535
Total rod worth ($)	10.01	9.68
Doppler coefficient ($\Delta\rho$ /°f - 10^5)	-1.0	-1.5

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* Analyses applied conservative multipliers to these values

TABLE 3

TRANSIENT RESULTS

Variable	Locked Rotor		Three pump Coast Down		Slow Rod Withdrawal*		Fast Rod Withdrawal	
	No plug	Plug	No plug	Plug	No plug	Plug	No plug	Plug
Peak Core Power (MWt)	2429.7	2429.7	2392.7	2392.8	2698.1	2771.6	2880.4	2880.5
Peak Primary Pressure (psia)	2282.4	2282.7	2277.1	2277.7	2343.7	2394.9	2255.5	2259.6
Max. core heatflux BTU/hr ft^2°F	182800	182800	182800	182800	206097	211596	193378	193379
MDNBR (NO BOW)	1.587	1.586	1.859	1.858	1.941	1.866	2.110	2.109
MDNBR (with BOW)	1.431	1.430	1.679	1.678	1.702	1.629	1.907	1.903

* The distinction between no tubes plugged case and tubes plugged case included using the latest overtemperature ΔT-trip function, thus the trip occurred later and MDNBR was lower. The effect of only plugging tubes was essentially no change in MDNBR.

ATTACHMENT 2

H. B. ROBINSON UNIT NO. 2 LOCA/ECCS ANALYSIS WITH

10% STEAM GENERATOR TUBES PLUGGED

Prepared by:

S. E. Jensen, W. V. Kayser, J. C. Cherng

LIST OF TABLES

List of Figures

Attachment

H.B. ROBINSON UNIT NO. 2 LOCA/ECCS ANALYSIS WITH 10% STEAM GENERATOR TUBES PLUGGED

Prepared by

S.E. Jensen, W.V. Kayser, J.C. Cherng

This attachment presents results of a revised LOCA/ECCS analysis performed for the H.B. Robinson Unit No. 2 reactor. This analysis differs from previous Exxon Nuclear Company (ENC) analyses[1], in that the steam generator tube plugging was increased from the previously assumed 6% value to a value of 10% uniform plugging. The analysis was performed using the current approved ENC WREM-IIA ECCS evaluation model[2], while the prior analysis used the older ENC WREM I model[3]. In order to expedite this calculation, no change in the input allowed linear heat generation rate or total peaking was made for this analysis. The results of the analysis show that the H.B. Robinson Unit No. 2 reactor can operate in conformance with NRC 10 CFR 50.46 criteria[4] with the existing limits of total linear heat generation rate (LHGR) of 13.43 kw/ft (102% of 13.17 kw/ft), a total peaking (F_Q) of 2.2, and up to an average uniform plugging of 10% of the steam generator tubes.

The method of performing the analysis with 10% steam generator plugging was the same as that used in the previous ENC analysis[1].

Based on data from the NSSS vendor supplied by Carolina Power and Light (CP&L)[5] the reactor vessel flow will exceed the value used in ENC analysis, even with up to 15% tube plugging. Therefore, no initial flow reduction was input. Revisions made to the input include: (1) steam generator volumes and flow areas were reduced to reflect 10% tube plugging; (2) steam generator heat transfer area was reduced corresponding to 10% tubes plugged; (3) secondary conditions were revised to reflect the 10% plugging; (4) the pump head and pressure distribution were revised to reestablish initial steady-state conditions. These changes were implemented throughout the required blowdown and reflood models.

To assure that the limiting break was identified with the ENC WREM IIA model, three large guillotine breaks were calculated with coefficients of 1.0, 0.8 and 0.6 applied to the break. Calculated event times for the three cases are shown in Table 1, and final temperature and metal-water reaction results appear in Table 2. The previously identified limiting break, the double-ended cold leg guillotine break with a coefficient of 0.8 (0.8 DECLG), was found to remain limiting with the ENC WREM IIA model.

The system and fuel rod input nodalization is identical to that of the previous analysis. The system blowdown nodalization is given in Figure 1. System blowdown results for the limiting 0.8 DECLG break are given in Figures 2-9. Calculated hot channel results are given in Figures 10-16. Figures 17 and 18 give the extended power and containment pressures, respectively. Figures 19-22 show reflood results, and Figure 23 gives the TOODEE2 heatup results.

The calculated peak cladding temperature (PCT) for the 0.8 DECLG break was 1976°F with less than 7% maximum local metal-water reaction and well below 1% core-wide metal-water reaction. Based on these results, the H.B. Robinson No. 2 reactor can operate with 10% tube plugging at existing limits, and remain within NRC 10 CFR 50.46 criteria with significant margin.

Table 1

H. B. ROBINSON

LARGE BREAK EVENTS

10% STEAM GENERATOR TUBE PLUGGING CASES

Event	Time (Seconds)		
	DECLG (1.0)	DECLG (0.8)	DECLG (0.6)
Start Start	0.0	0.0	0.0
Initiate Break	0.1	0.1	0.1
Safety Injection Signal	0.6	0.6	0.6
Accumulator Injection, Intact Loop	12.1	12.1	13.0
Accumulator Injection, Broken Loop	1.6	3.1	5.3
End-of-Bypass	22.7	22.8	23.9
Bottom of Core Recovery	44.0	44.2	45.1
Accumulator Empty	46.2	46.2	46.2
Safety Pump Injection	25.6	25.6	25.6
Peak Clad Temperature Reached	118.8	119.8	123.2

Table 2

H. B. ROBINSON

LARGE BREAK RESULTS

10% STEAM GENERATOR TUBE PLUGGING CASES

Result	DECLG (1.0)	DECLG (0.8)	DECLG (0.6)
Peak Cladding Temperature, °F	1955	1976	1962
Peak Temperature Location, ft	7.125	7.125	7.125
Local Zr/H_2O Reaction, (Max.) %	<6	<7	<6
Total Zr/H_2O	<1%	<1%	<1%
Hot Rod Burst Time, sec.	42.9	42.0	46.0
Hot Rod Burst Location, ft.	6.125	6.125	6.125
Linear Heat Generation Rate, BOCREC	.667	.666	.644

REFERENCES

(1) XN-76-54, "LOCA Analyses for H.B. Robinson Unit No. 2 Using WREM-Based ECCS Evaluation Model with Reduced LPSI Flow, Steam Generator Plugging, and Increased Upper Head Temperature", December 1976.

(2) XN-NF-78-30, "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-IIA", August 1979, and amendments.

(3) XN-75-41, "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model", July 1975 and Supplements and Revisions thereto.

(4) 10 CFR 50.46 and Appendix K of 10 CFR 50, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors", Federal Register, Vo. 39, Number 3, January 9, 1974.

(5) Letter, J.D. Martin (CP&L) to T.J. Helbling (ENC) dated June 10, 1980.

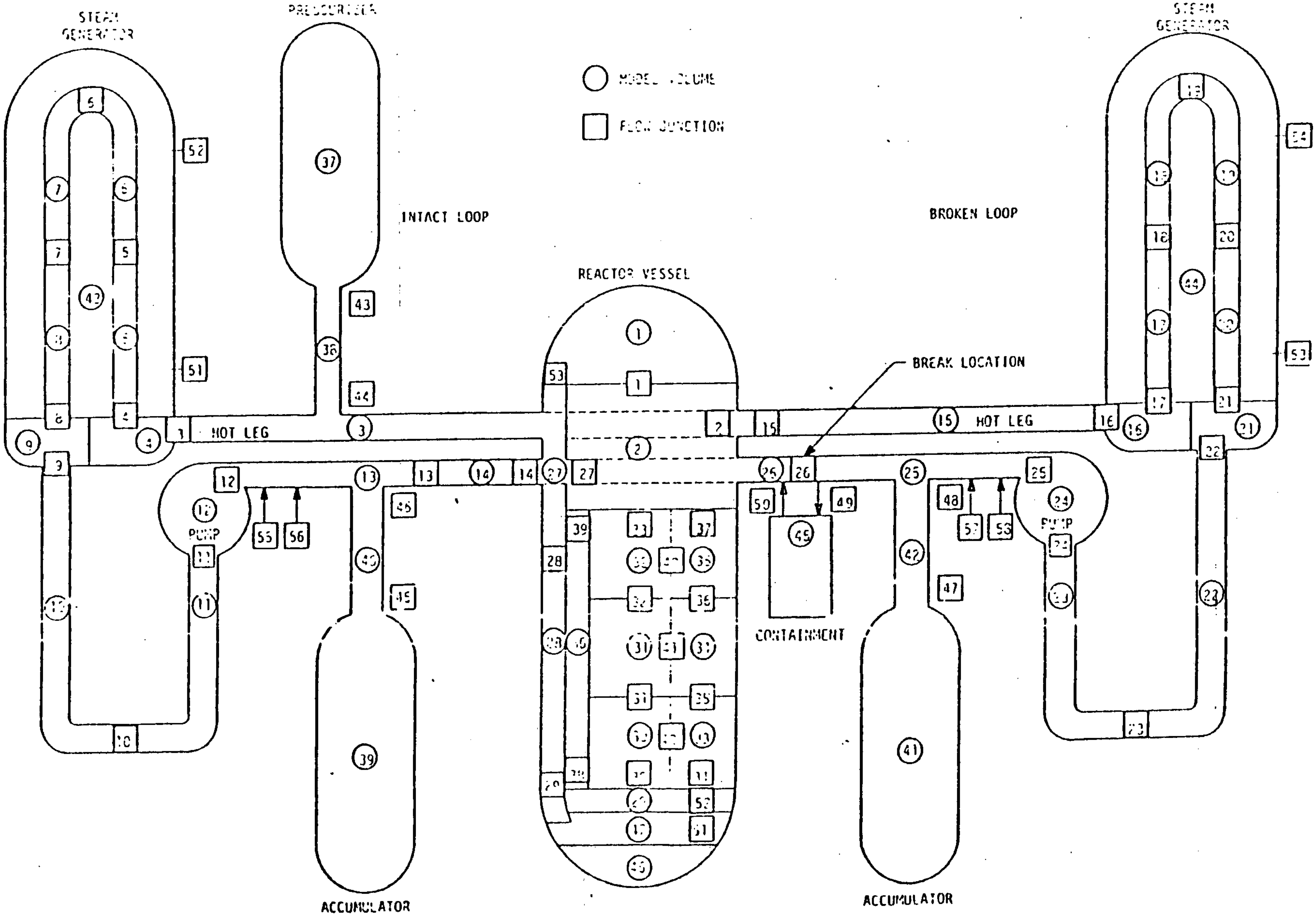

Figure 1 Blowdown System Nodalization

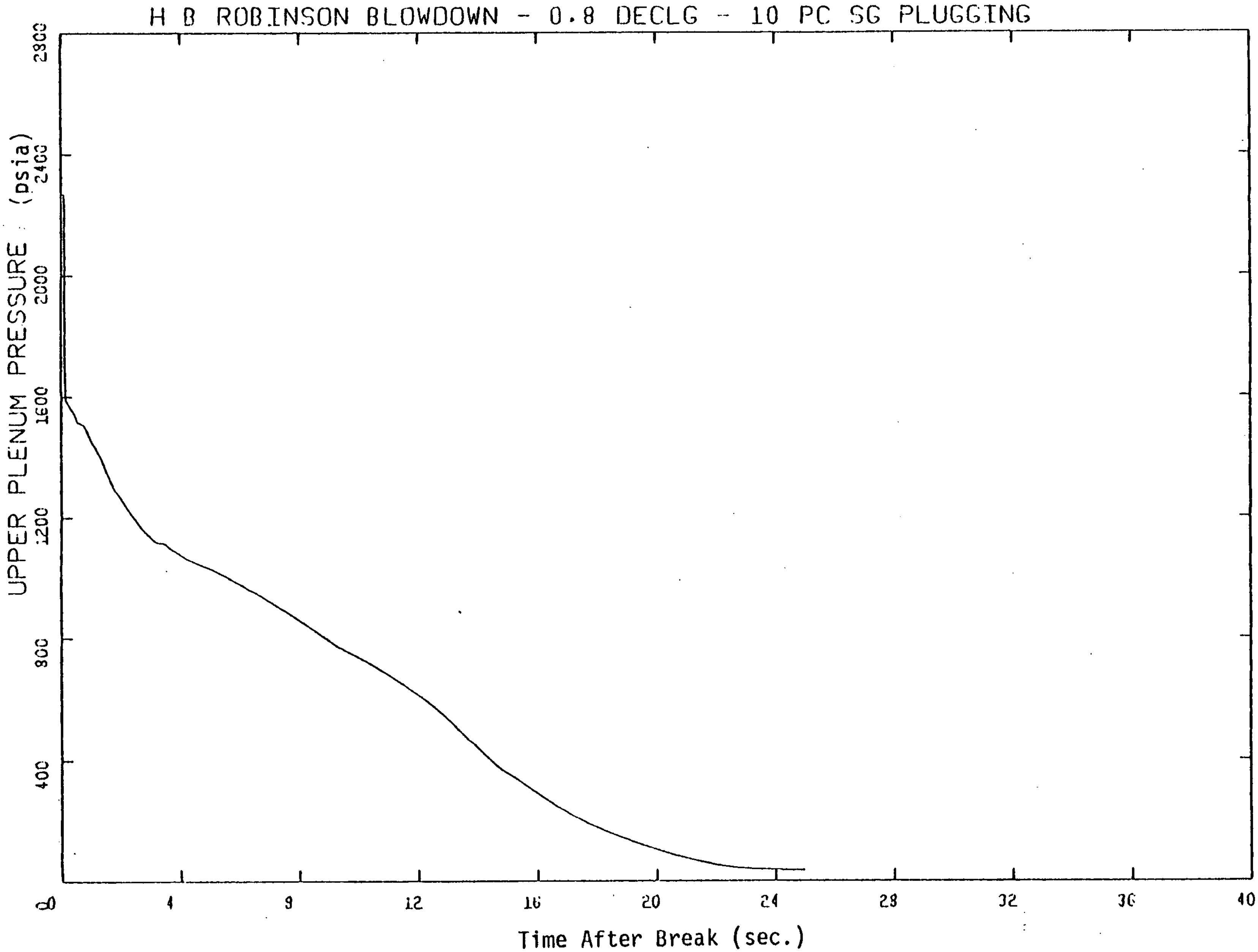

Figure 2 Blowdown System Pressure - 0.8 DECLG 10% SG Plugging

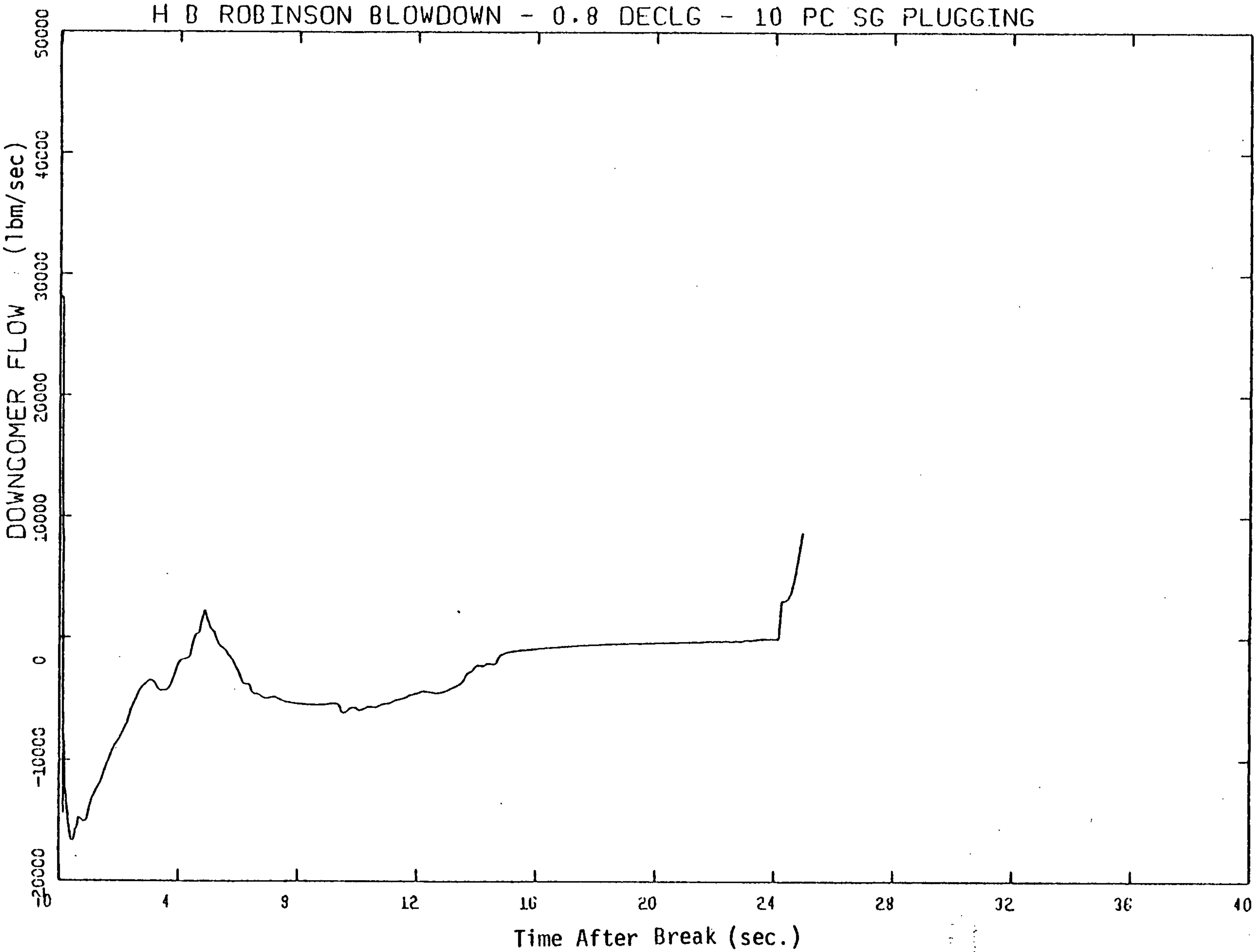

Figure 3 Blowdown Downcomer Flow - 0.8 DECLG 10% SG Plugging

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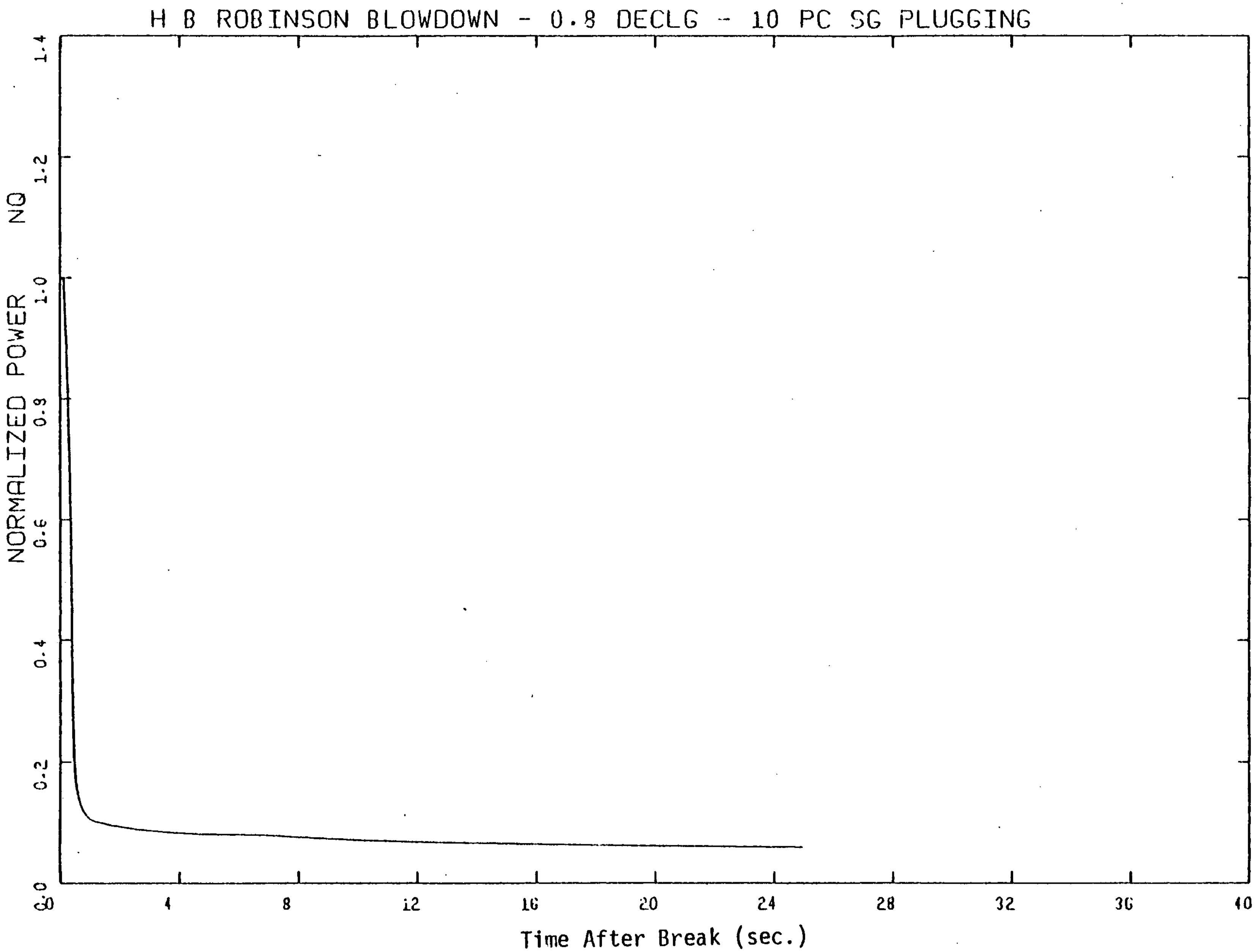

Figure 4 Blowdown Normalized Power - 0.8 DECLG 10% SG Plugging

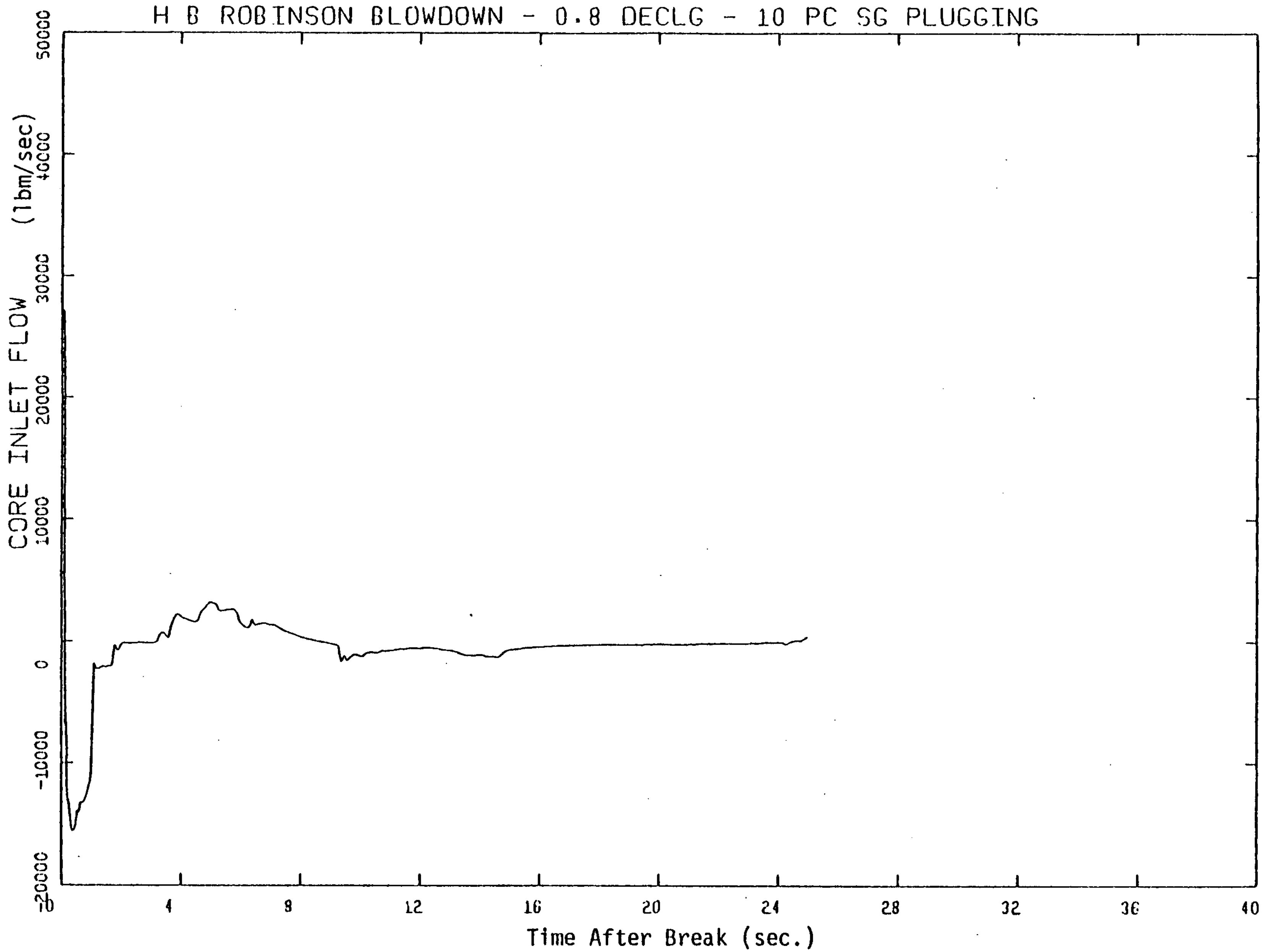

Figure 5 Blowdown Core Inlet Flow - 0.8 DECLG 10% SG Plugging

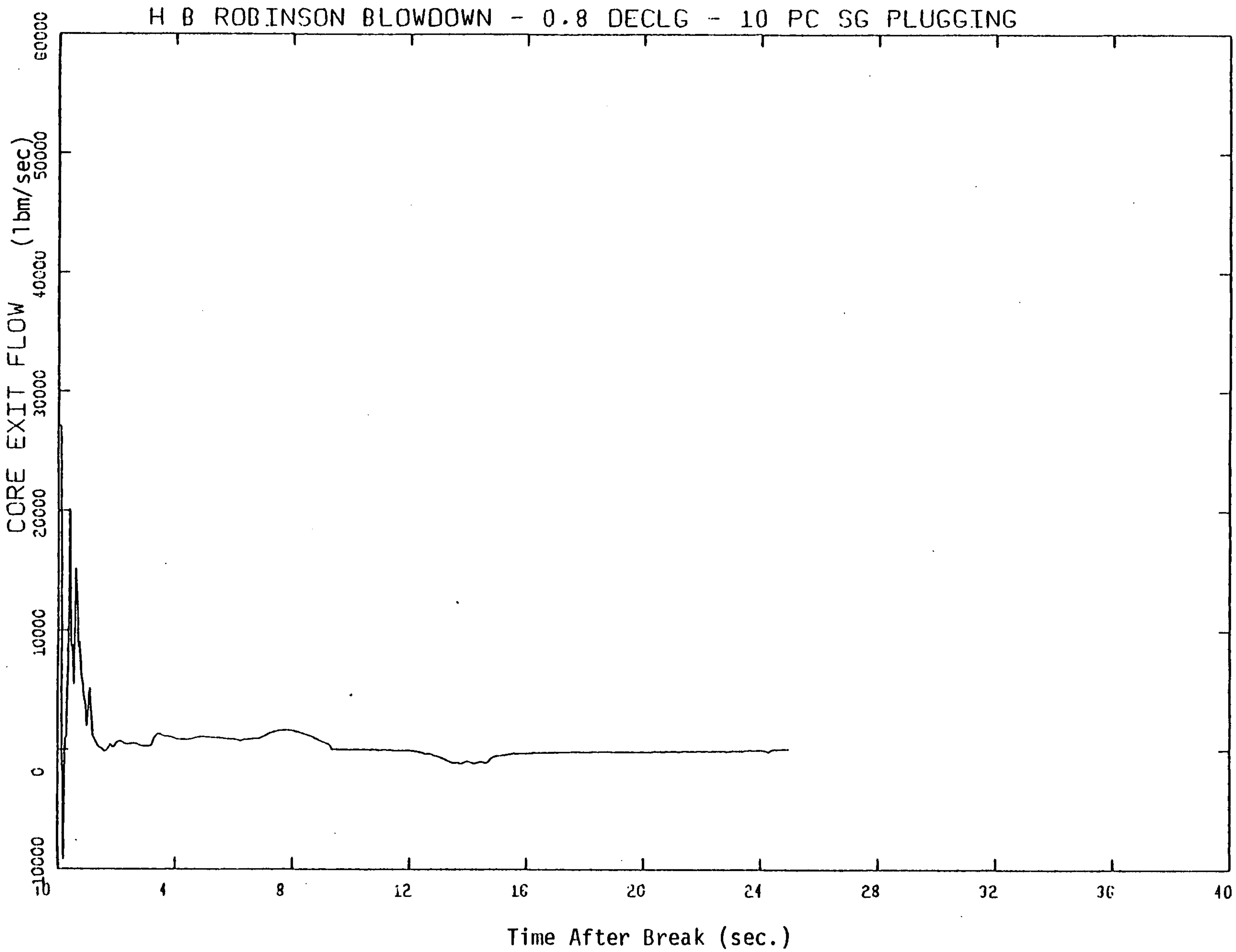

Figure 6 Blowdown Core Exit Flow - 0.8 DECLG 10% SG Plugging

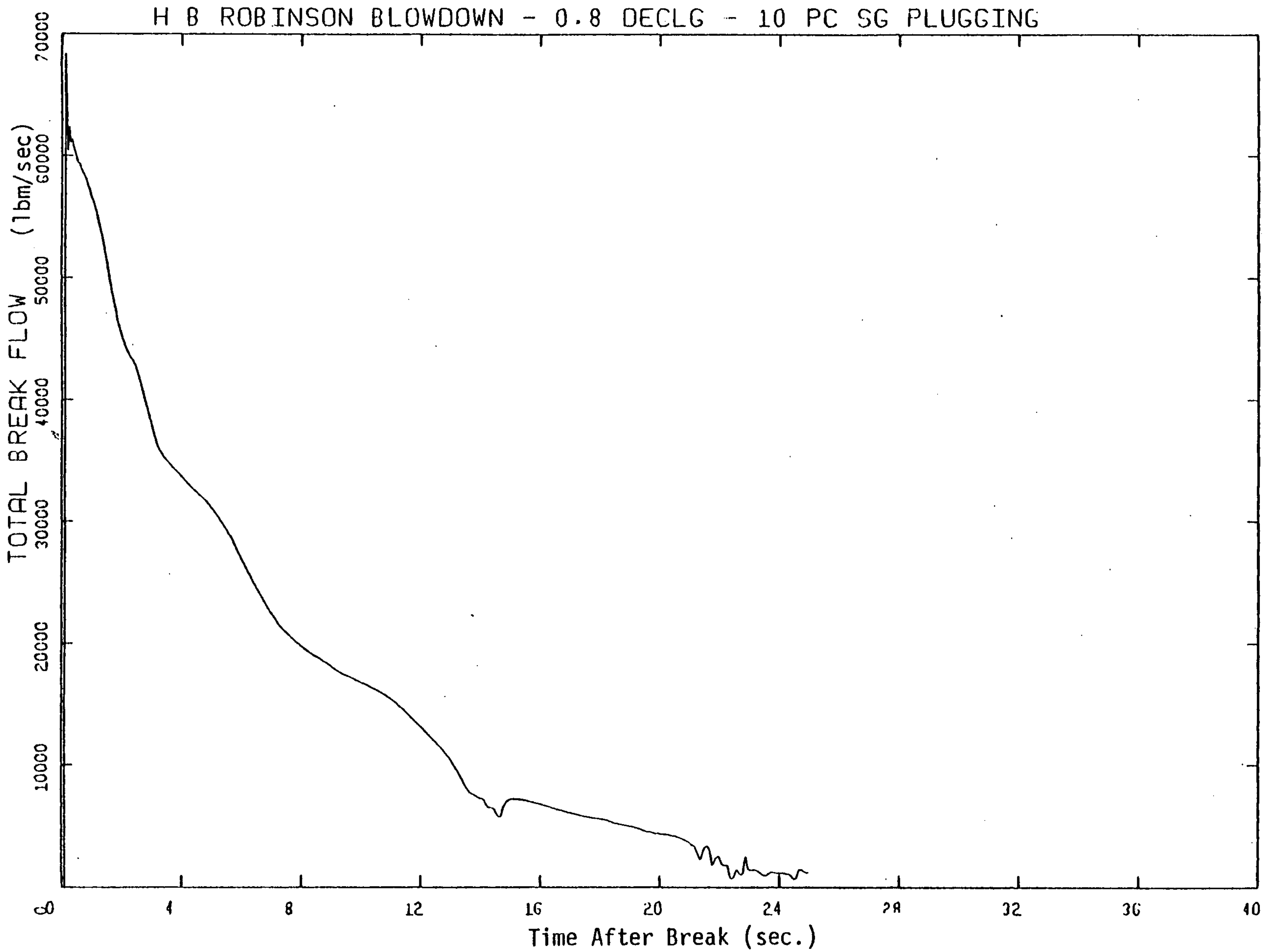

Figure 7 Blowdown Total Break Flow - 0.8 DECLG 10% SG Plugging

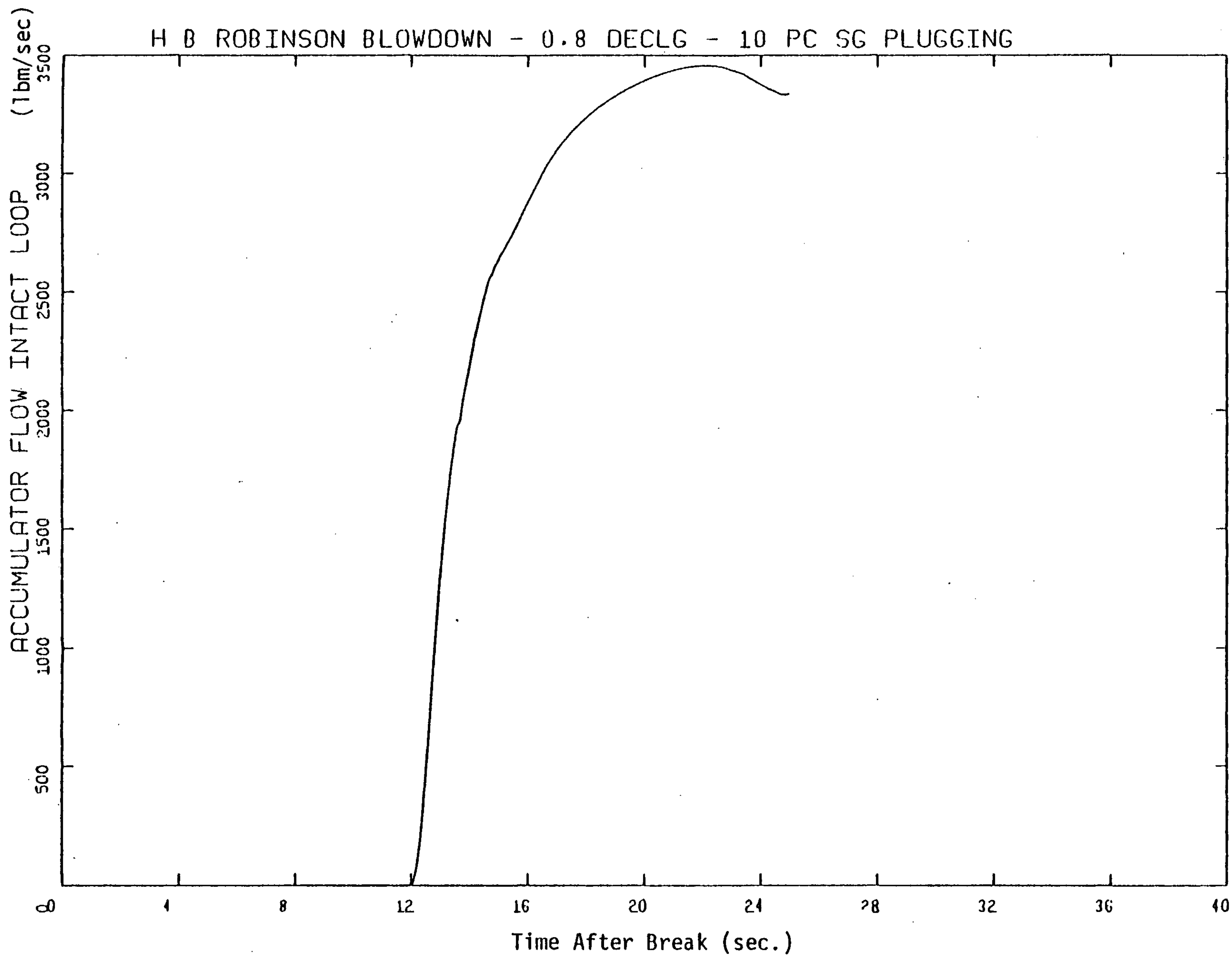

Figure 8 Blowdown Intact Loop Accumulator Flow
0.8 DECLG 10% SG Plugging

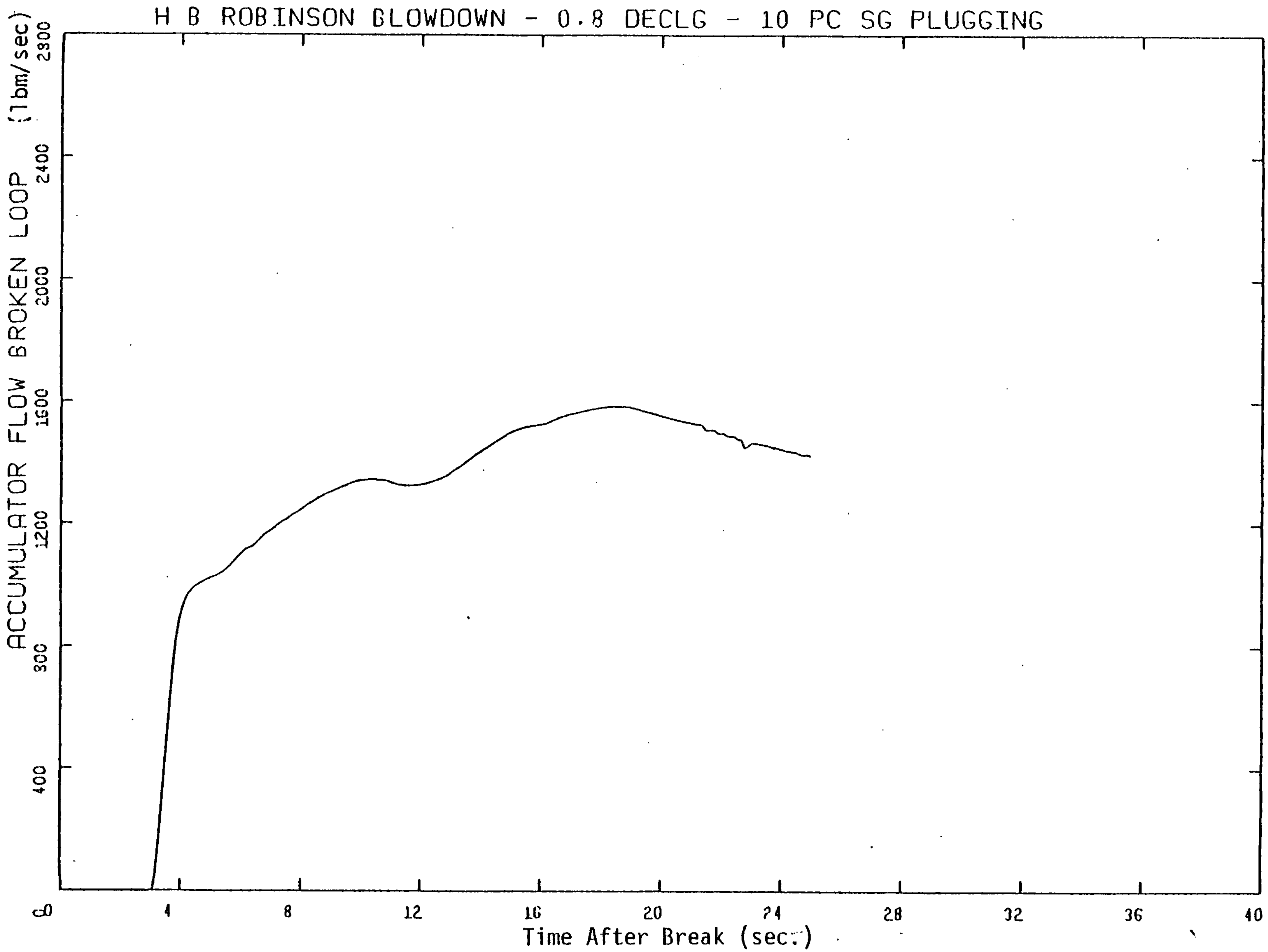

Figure 9 Blowdown Broken Loop Accumulator Flow
0.8 DECLG 10% SG Plugging

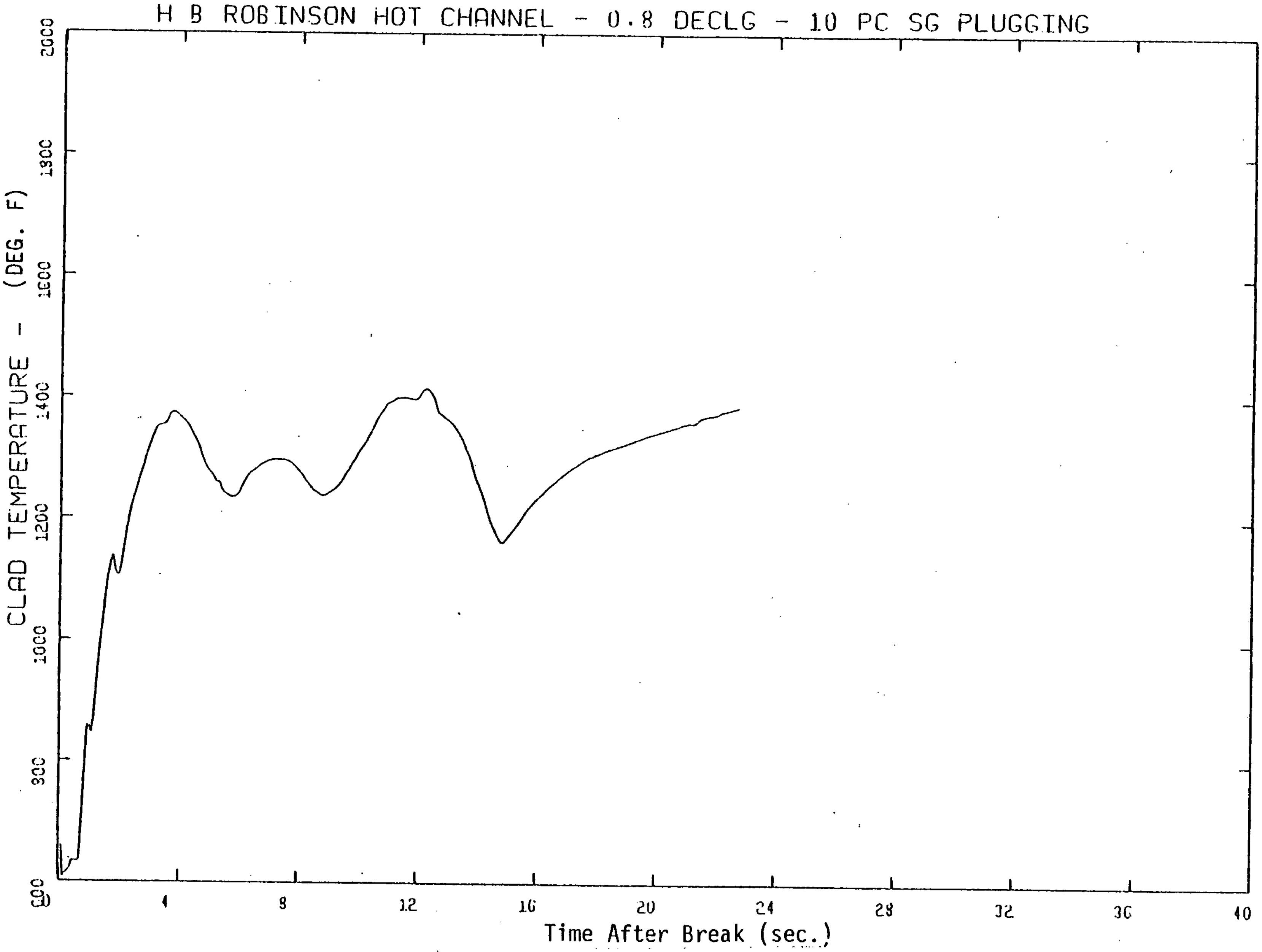

Figure 10 Hot Rod Clad Temperature at PCT Location
0.8 DECLG 10% SG Plugging

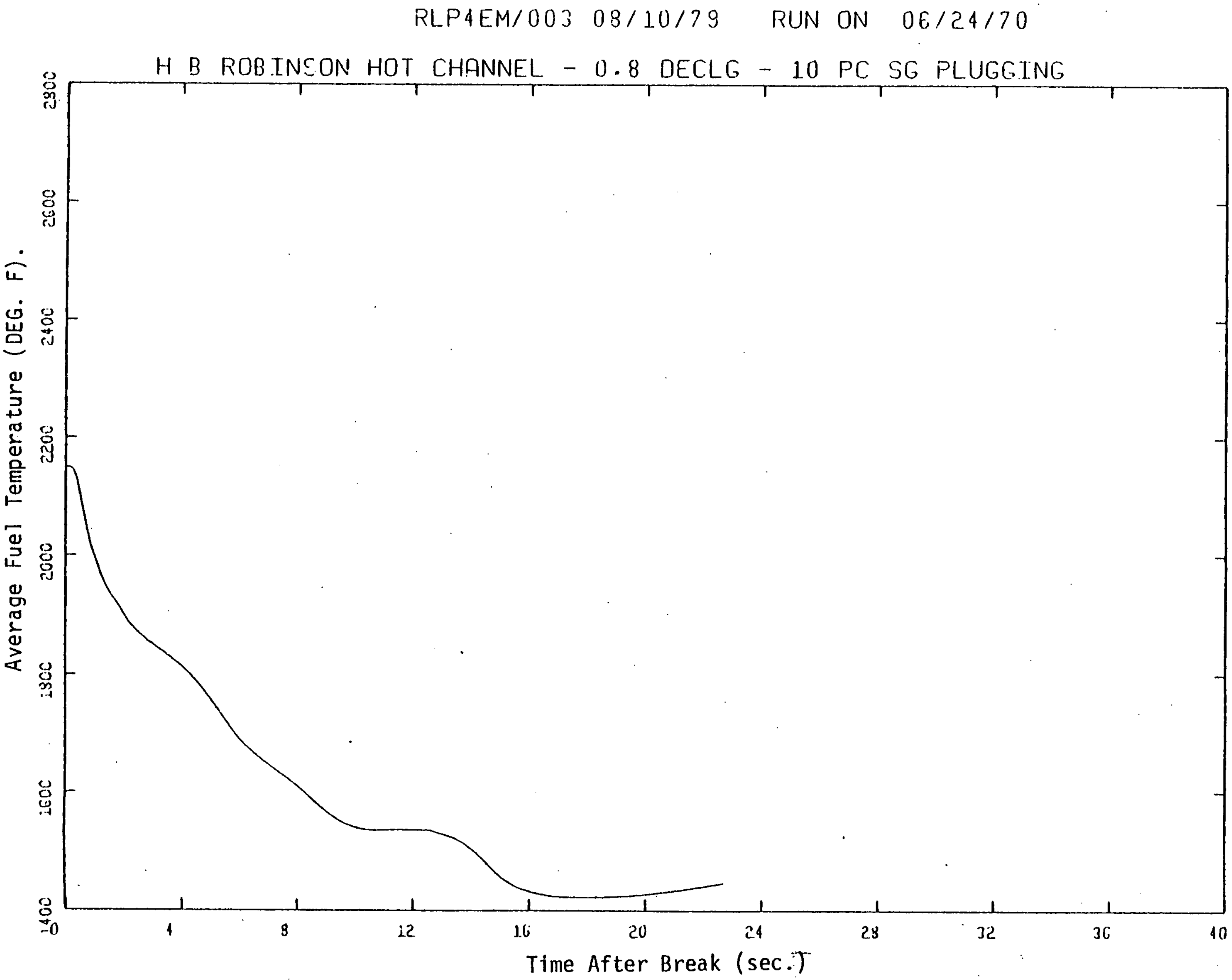

Figure 11 Hot Rod Average Temperature at PCT Location
0.8 DECLG 10% SG Plugging

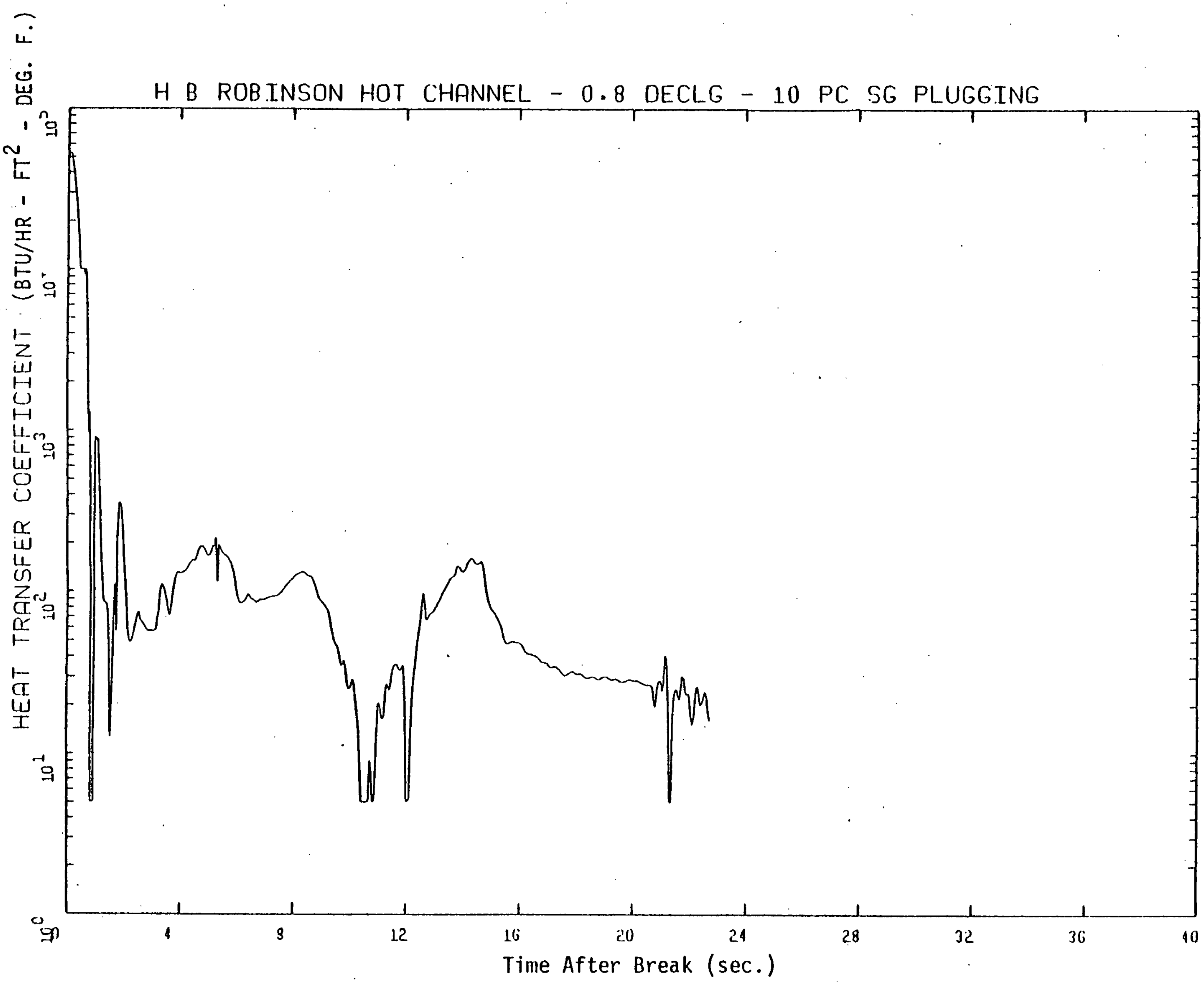

Figure 12 Hot Rod Heat Transfer Coefficient at PCT Location
0.8 DECLG 10% SG Plugging

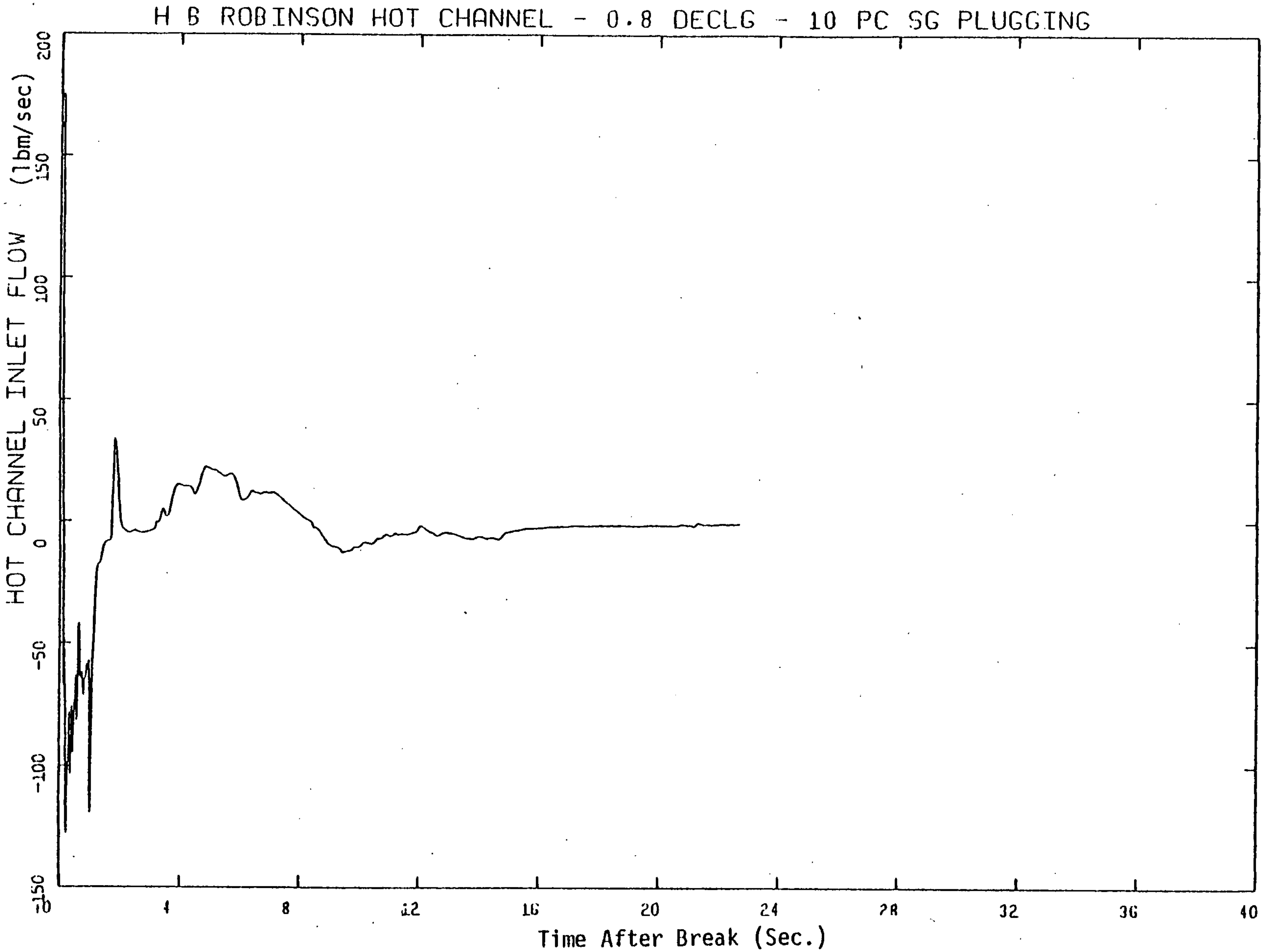

Figure 13 Hot Channel Inlet Flow - 0.8 DECLG 10% SG Plugging

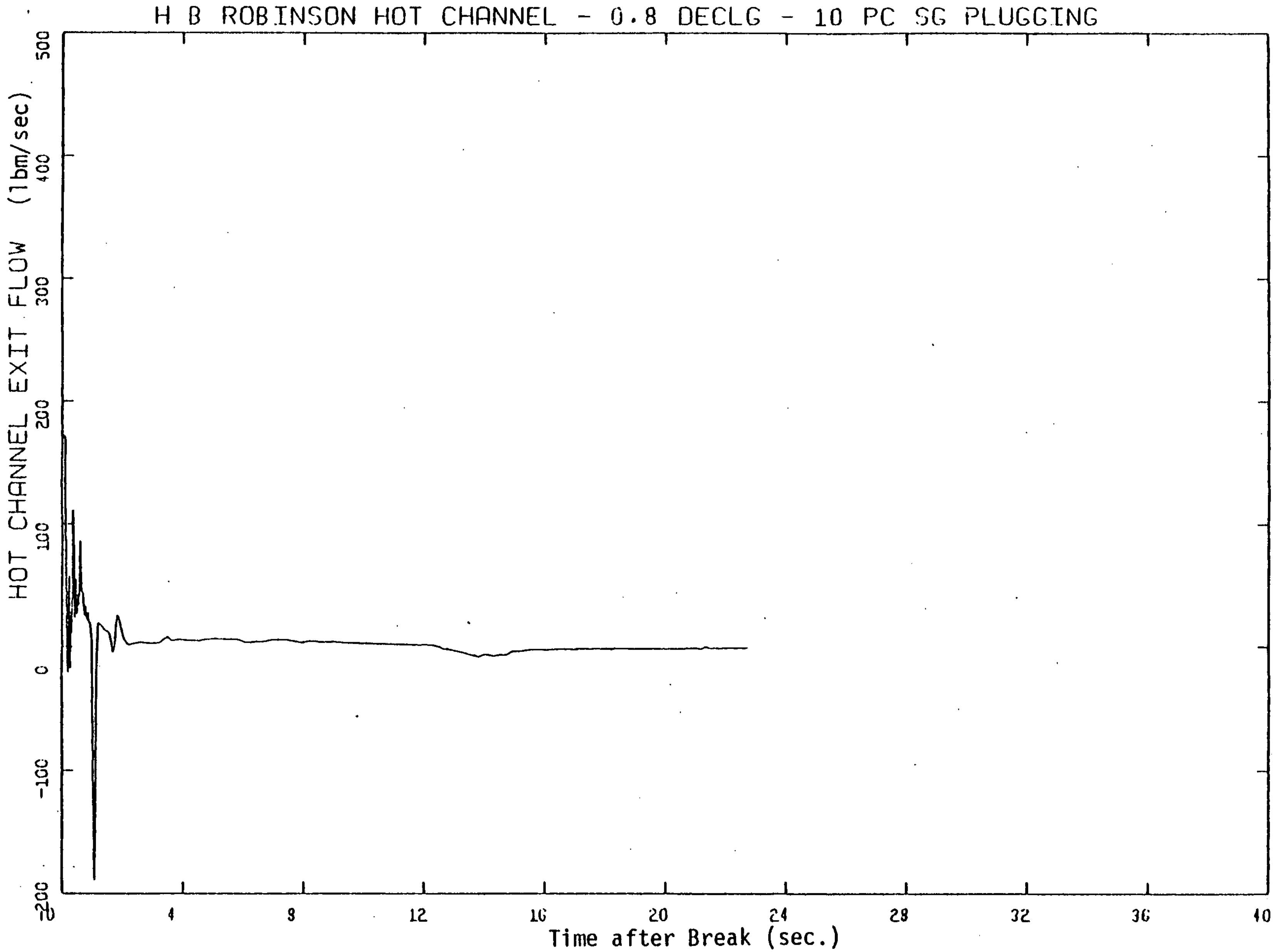

Figure 14 Hot Channel Exit Flow - 0.8 DECLG 10% SG Plugging

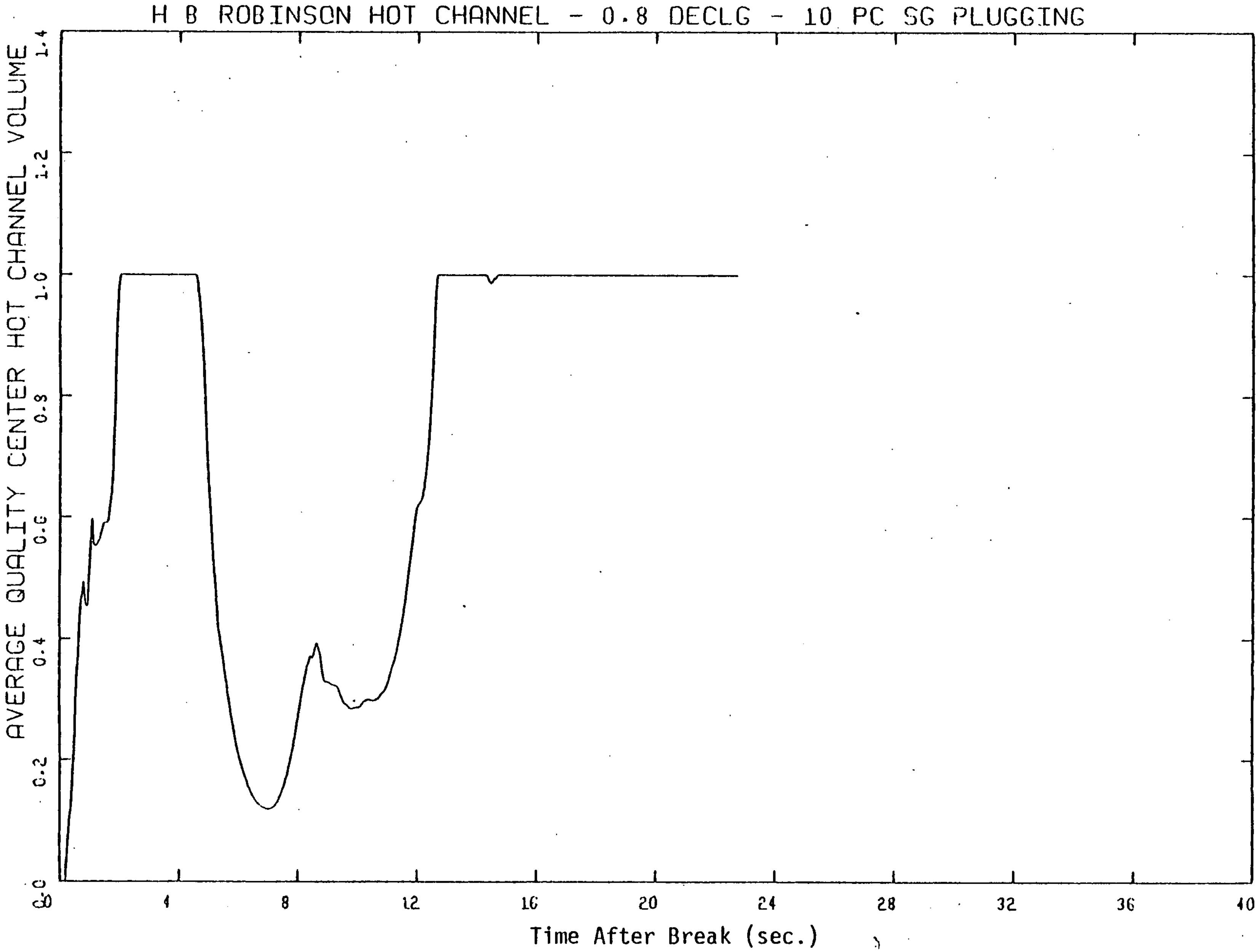

Figure 15 Hot Channel Center Volume Average Quality
0.8 DECLG 10% SG Plugging

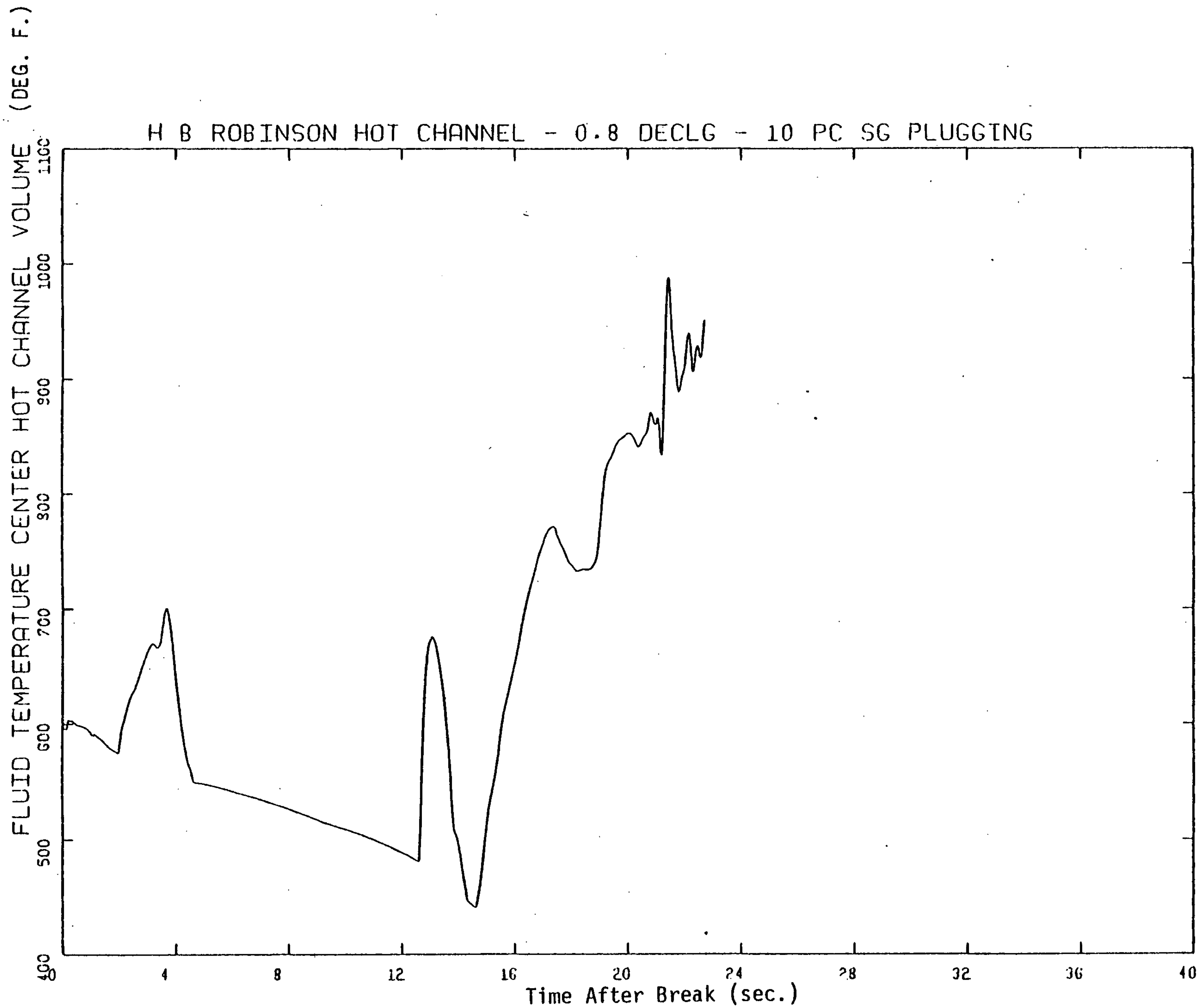

Figure 16 Hot Channel Center Volume Fluid Temperature
0.8 DECLG·10% SG Plugging

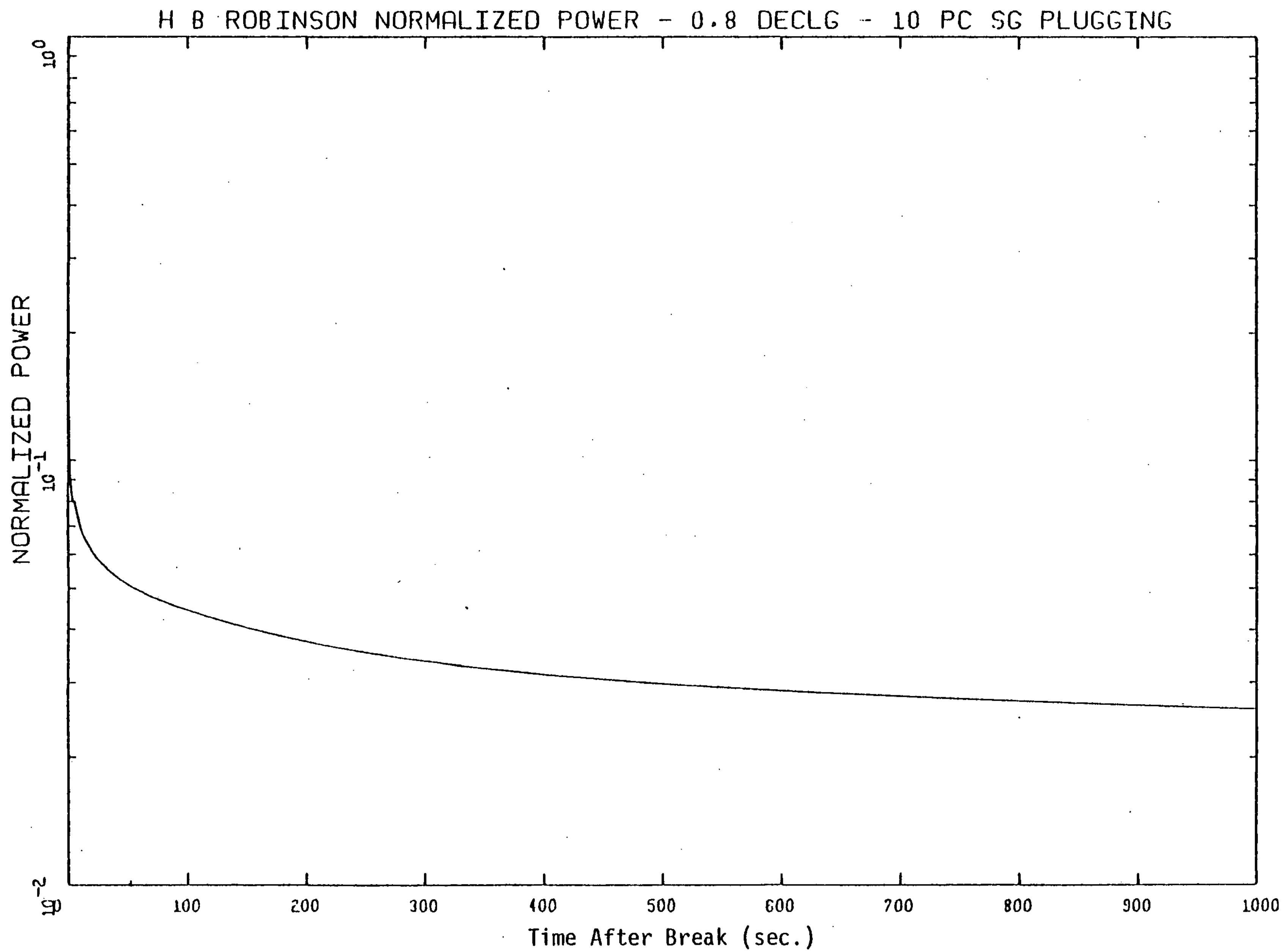

Figure 17 Extended Normalized Power - 0.8 DECLG 10% SG Plugging

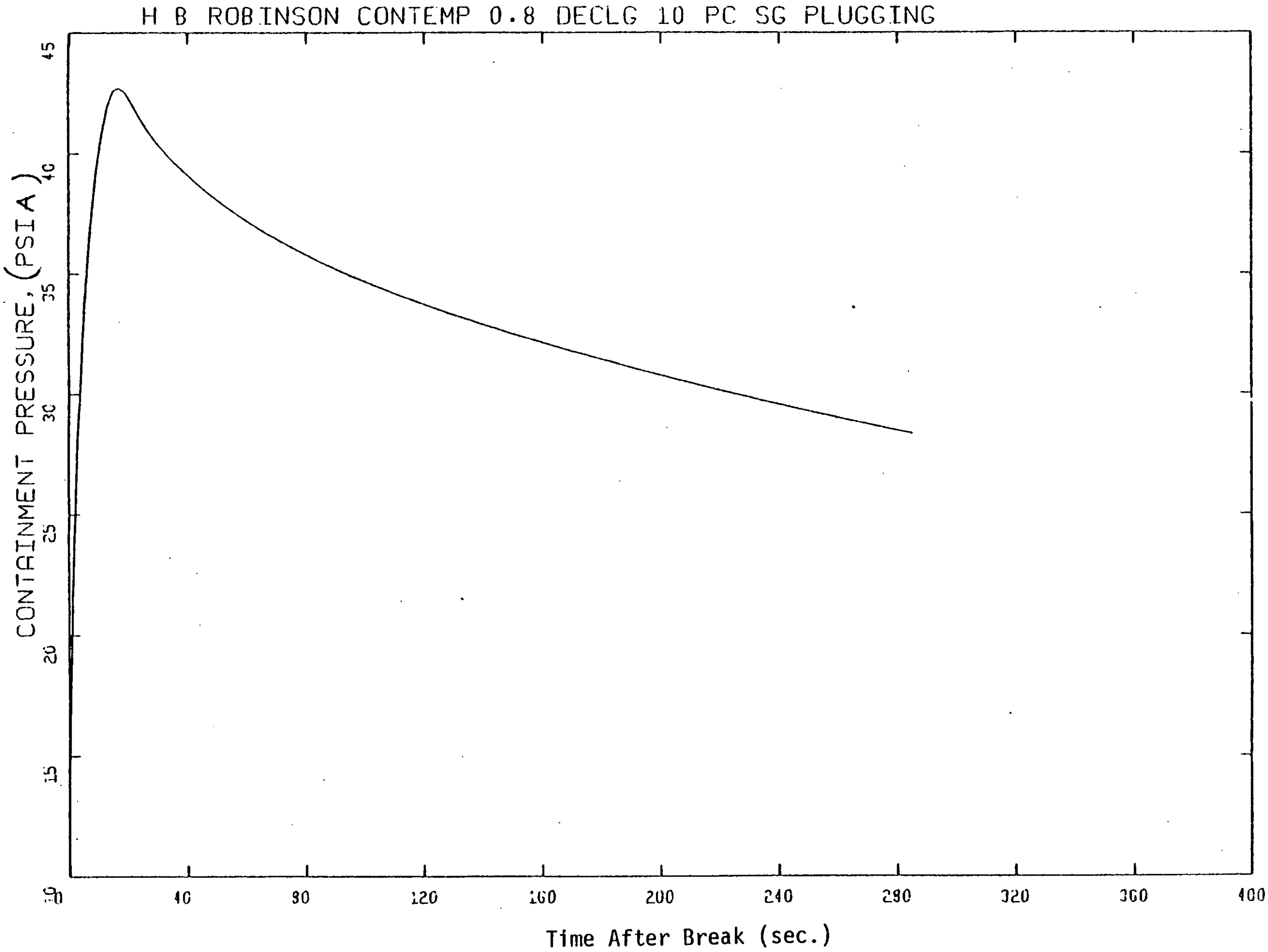

Figure 18 Containment Pressure

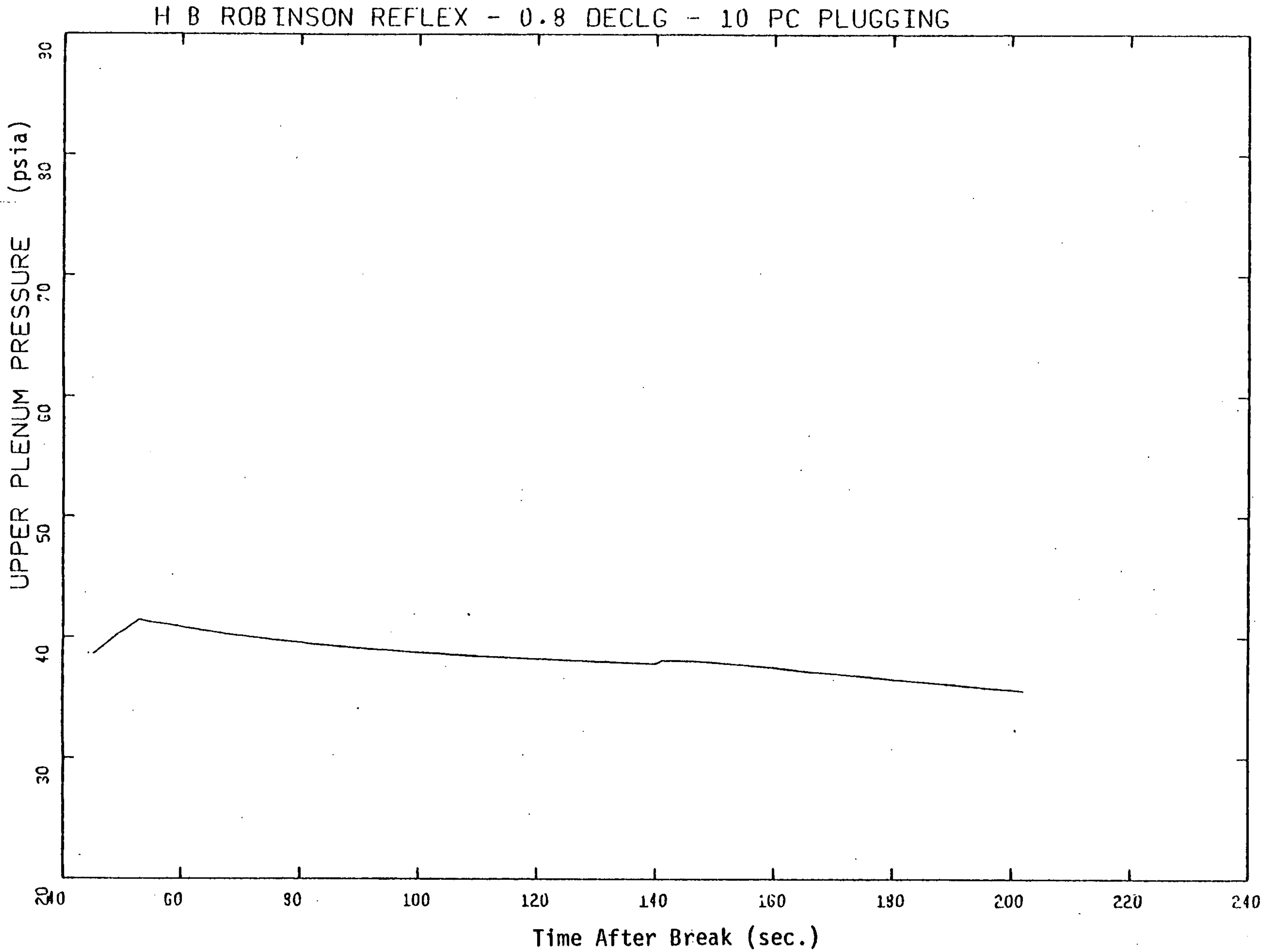

Figure 19 Reflood Upper Plenum Pressure

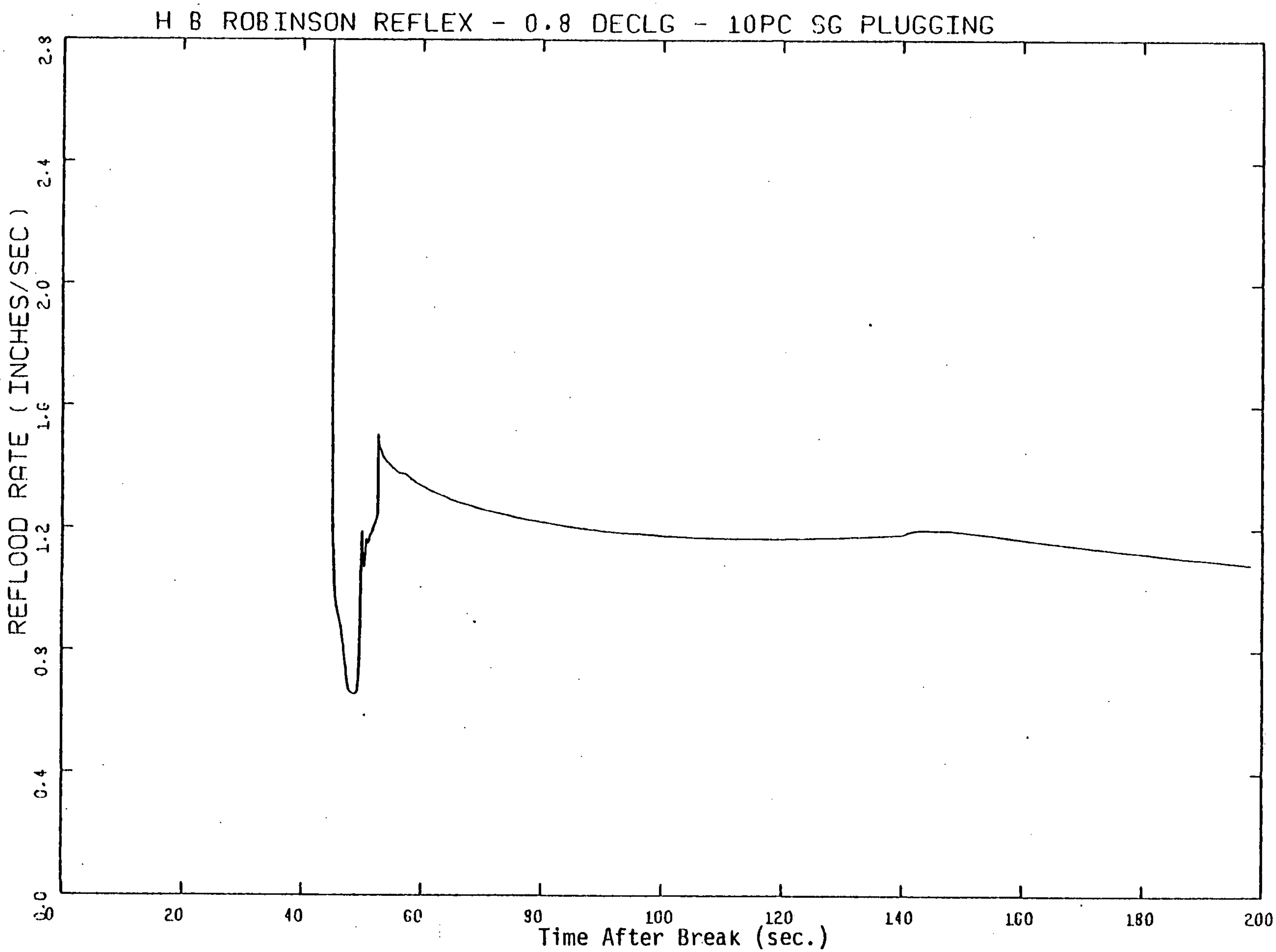

Figure 20 Reflood Core Flooding Rate - 0.8 DECLG 10% SG Plugging

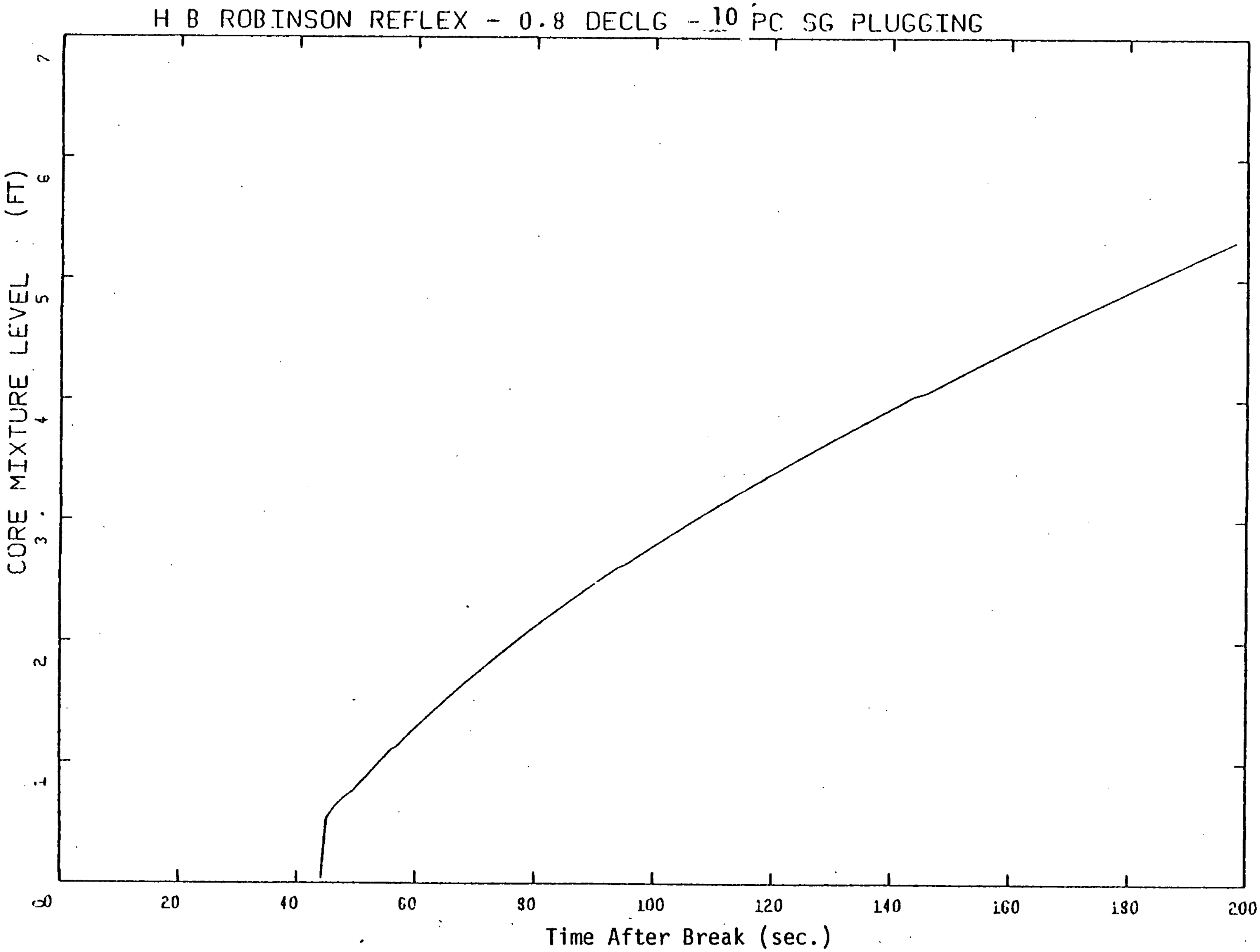

Figure 21 Reflood Core Mixture Level - 0.8 DECLG 10% SG Plugging

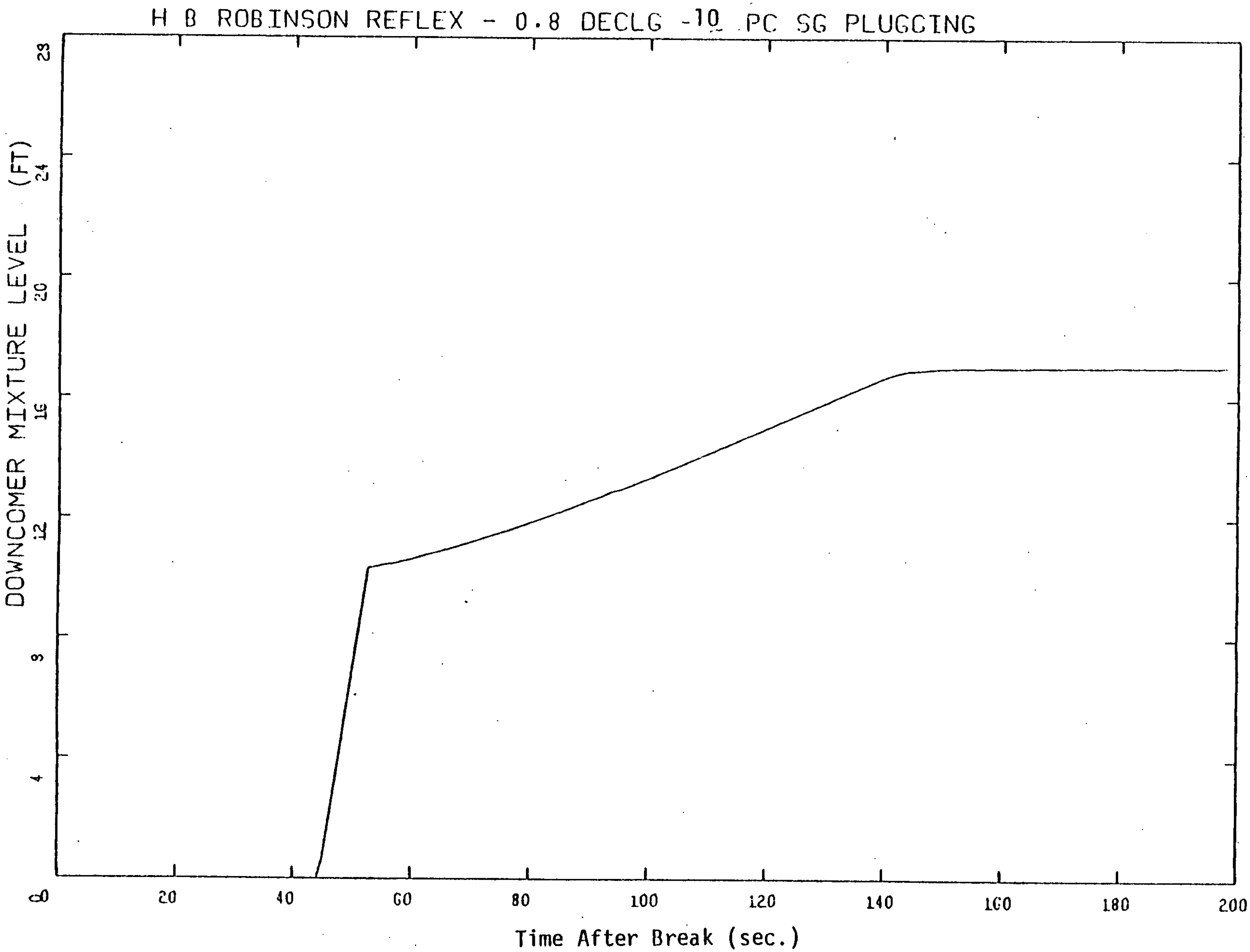

Figure 22 Reflood Downcomer Mixture Level - 0.8 DECLG 10% SG Plugging

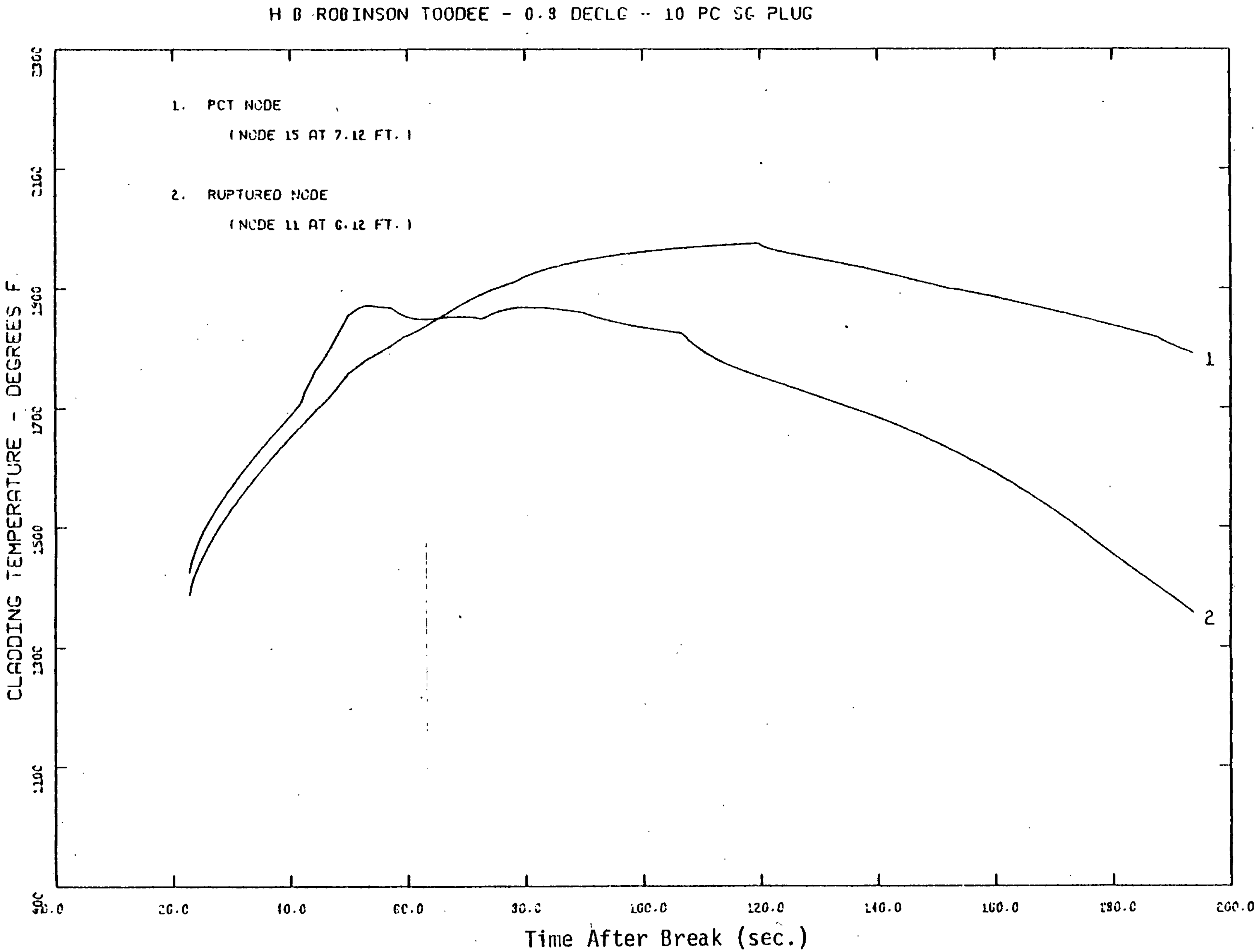

Figure 23 TOODEE2 Peak Cladding Temperature - 0.8 DECLG 10% SG Plugging