Babcock & Wilcox

Power Generation Group

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December 17, 1976

Mr. S. A. Varga, Chief
LWR Branch No. 4
Division of Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Varga:

Attached is additional ECCS small break analysis for B&W's 177 Fuel Assembly Lowered-Loop NSS. This analysis is in accordance with the small break model as approved in BAW-10104A, Rev. 1, "B&W's ECCS Evaluation Model," March 1976.

This analysis was requested by letters from J. F. Stolz to K. E. Suhrke, February 4, 1976, and from D. F. Ross to K. E. Suhrke, March 22, 1976 in connection with the review of BAW-10103, Rev. 2, "ECCS Analysis of B&W's 177-FA Lowered-Loop NSS." The other analysis (.5 ft^2 with large break model) requested in Mr. Stolz's letter was submitted to you on October 8, 1976. This small break analysis predicts results well below the acceptance criteria of 10CFR50.46.

This submittal completes B&W's response to all the Staff's questions on BAW-10103, Rev. 2. Following your review, these analyses will be incorporated into BAW-10103. If you have questions concerning this matter, please feel free to contact Henry A. Bailey (Extension 2678) of my staff.

Very truly yours,

Kenneth E. Suhrke
Manager, Licensing

KES:HAB:dc
Attachment

cc: R. B. Borsum (B&W)(2)
Thomas H. Novak
Zoltan R. Rosztoczy (NRC)

CONTENTS

LIST OF FIGURES

1. INTRODUCTION

Presented here are the results of an analysis of hypothetical loss-of-coolant accidents resulting from postulating small breaks in the reactor coolant system of B&W's 177FA lower loop plants. B&W's ECCS Evaluation Model as defined in BAW-10104[5] was used for the analysis. See pg 5-4-7 diagram.

Small breaks are defined as ruptures of the reactor coolant system with leak areas of 0.5 ft^2 or less. Break areas considered for this study are: (1) 0.44 ft^2 CFT line break since it has the minimum ECC system available to mitigate the LOCA, (2) 0.5 ft^2 break at the pump discharge as it represents the transition break between the small and large break models, (3) 0.04 ft^2 break at the pump suction which was shown to be the most limiting small break in BAW-10052.[1]

The current analysis of these breaks verifies the conservative nature of the cladding temperature calculations presented for the same breaks in BAW-10052 and BAW-10064[4]. Therefore, the analysis contained herein, coupled with the analyses of BAW-10052 and BAW-10064 provide an appropriate spectrum of breaks for the evaluation of the effects of small leaks and the demonstration of the ability of the ECCS to effectively control them.

BAW-10052 {
.3 ft² @ Pump Suct - 780°F PCT
.1 " " " " - 826 " "
.1 " " " Disch - 730 " "
.04 " " " Suct - 971 " "

BAW-10064 - CFT line Brk - 1200°F PCT @ 10.75 ft - for PA @ 9.5 ft.

2. SUMMARY AND CONCLUSIONS

The various breaks analyzed in Section 4 of this report show that the core remains covered throughout the transient. During the initial period when the transient is flow controlled, sufficient flow is maintained such that CHF does not occur and nucleate boiling heat transfer predominates. Since the core remains covered by a mixture, pool film boiling will be maintained during the quiescent period of the accident. This heat transfer mechanism is sufficient to maintain the cladding temperature within a few degrees of the fluid saturation temperature (Reference 4, BAW-10064). Therefore, for the maximum linear heat rate covered by BAW-10103[7], the transient cladding surface temperature will never exceed its initial value of 660F, no metal water reaction will occur, and the core geometry will remain coolable as no cladding rupture will occur. Long term cooling is established as the HPI and LPI pumped injection systems provide fluid in excess of the boiloff rate due to core decay heat. Thus the five Acceptance Criteria in 10CFR50.46 are met.

The new modeling techniques used in the CRAFT2[2] analyses for the present studies show improvement in the core performance when compared with the results of the same breaks reported in BAW-10052 and BAW-10064. Therefore, if all the breaks reported in BAW-10052 and BAW-10064 were re-analyzed with the present model, the same trend of improvement in core performance would be realized. Thus, the present analyses in conjunction with the analyses of BAW-10052 and BAW-10064 provide a suitable small break spectrum for demonstration of compliance of the ECC system with the five Acceptance Criteria in 10CFR50.46.

3. METHOD OF ANALYSIS

The analysis uses the CRAFT2[2] code to develop the history of the reactor coolant system hydrodynamics. For small leak analysis it is sufficient to use smaller models than are used for large loss-of-coolant studies because hydrodynamic responses are slow enough for simpler models to describe them.[1] The CRAFT model uses 19 nodes to simulate the reactor coolant system, two nodes for the secondary system, and one node for the reactor building. A schematic diagram of the model is shown in Figure 4.1 along with the node descriptions. Control volumes (nodes) in and around the vessel are all connected by a pair of flow paths to allow the occurrence of counter-current flow. The break is located in the cold leg piping either at the lowest point in the pipe at the pump suction, or at a point opposed to the high-pressure injection nozzle at the pump discharge, or in the core flood line joining the CF nozzle. The Wilson, Grenda and Patterson[3] average bubble rise model is used for all nodes. Within the core region, however, a multiplier of 2.38 is applied to the calculated bubble rise velocity. Section 5 of this report demonstrates that a multiplier of 2.38 in CRAFT2 gives a mixture height within $\pm$ 2% of that predicted by FOAM[4]. Thus, no FOAM analysis will be needed if the CRAFT2 mixture level remains above the core by 2% of the active length.

The following assumptions are made for conditions and system responses during the accident:

1. The reactor is operating at 102% of the steady-state power level of 2772 Mwt.
2. The leak occurs instantaneously, and a discharge coefficient of 1.0 is used for the entire analysis. Bernoulli's equation was used for the subcooled portion of the transient while Moody's correlation was used in the two phase portion.
3. No offsite power is available.
4. The reactor trips on low pressure at 1900 psia.
5. The safety rods begin entering the core after a 0.5 second delay from the time the reactor trip signal is reached.

6. The reactor coolant pumps trip and coast down coincident with reactor trip.
7. One complete train of the emergency safeguards system fails to operate, leaving two CFTs and only one high-pressure injection and one low-pressure injection system available for pumped injection to mitigate the consequence of a cold leg break. For the CFT line break, only one CFT and one high pressure injection system is assumed available for providing ECC fluid to the vessel.
8. The auxilliary feedwater system is assumed to be available during the transient. It mainly removes heat from the upper half of the steam generator during the initial stages of the transient. When the secondary side of the steam generator becomes a source of heat to the primary system, the assumption of auxiliary feedwater maximizes the energy that must be relieved.
9. ESFAS signal error band is considered in the analysis to signal the actuation of the HPI.

The CRAFT2 results obtained from the present analysis are sufficient to meet the five Acceptance Criteria of 10CFR50.46 in that the core was always covered by a two-phase mixture, hence no separate thermal analysis is necessary for cladding temperatures during the transient. If required, as in the case of uncovery to within 0.25 feet above the active core (Ref. Section 5), the cladding heatup can be calculated by the procedure outlined in Section 5.2.3 of BAW-10104A.

4. RESULTS OF SMALL LEAK TRANSIENTS

This section presents a detailed evaluation of the three breaks considered along with explanations of the phenomena involved.

4.1 Explanation of Curves

The following categorical explanations are provided to aid in understanding the parameters illustrated in the curves:

Core Power: This curve indicates the normalized thermal power as calculated by CRAFT2.

Core Flow: This curve represents the total flow rates of core paths 1 and 2 of Figure 4.1. The curve shows flow rates mainly during the flow controlled part of the transient.

Pressure: This is the pressure at the top of the core node as calculated by CRAFT. The core node, in these analyses, includes the core, upper plenum, upper head, and the core bypass.

Boil-Off Due To Decay Heat: The liquid boil-off rate is given in terms of equivalent amount of HPI or HPI + LPI injection rate needed to dissipate the core decay heat. Mathematically:

$$\text{Boil-off rate} = \text{(core decay heat rate)} (h_g - h_{in})$$

Where: h_g = enthalpy of saturated steam at core pressure

h_{in} = enthalpy of injected water

Inner Vessel Mixture Height: This curve shows the mixture height in the core node as calculated by the CRAFT code. The lines spanning the curve indicates the top of the active core, hot leg regions and the vent valve region.

Core Liquid Level: This curve, in contrast to that for the inner vessel mixture height shows the effective core liquid height and volume with the lower plenum filled with a mixture at the void fraction calculated by CRAFT2. This volume is representative of the liquid volume within the core node that would be used to calculate the mixture height within the core.

4.2 0.44 Ft^2 CFT Line Break

The break is assumed to be at the CFT line nozzle joining the reactor vessel and is limited in area to 0.44 ft^2 by the nozzle insert in the CFT line. Node 13 in Figure 4.1 is the break node, and the analysis takes credit for one CFT and one HPI pump.

Figures 4.2 and 4.3 show core power and core flow rate respectively. Rapid initial depressurization (see Figure 4.4) causes reactor trip and start of reactor coolant pump coastdown within the first second. Flashing of system liquid slows the depressurization while the steam generator continues to remove energy from the primary coolant thereby helping to decrease the pressure. The lower pressure limit of the ESFAS setpoint error band is reached by about 10 seconds which initiates main feedwater and steam line isolation procedures and signals actuation of HPI. At about 40 seconds, the RCS pressure drops below the secondary steam generator pressure and heat removal to the secondary side drops off sharply and becomes a source of heat to the primary causing a slower depressurization. System flow has degraded such that core flow is predominately due to natural circulation and quiescent period of transient begins. HPI system provides makeup starting at about 50 seconds aiding depressurization. Core flood tank flow begins at 140 seconds aiding further depressurization but the diminishing leak flow (Figure 4.5) slows the depressurization rate. The core flood tanks are emptied by about 500 seconds after which the rate of depressurization is steady but very slow. Figure 4.6 is a plot of HPI and total ECC water flow. Long term cooling is assured in that by 850 seconds the HPI injection rate exceeds the boil-off due to core decay heat. Figure 4.7 is a plot of core liquid inventory and mixture height. It shows that while much of the core liquid inventory is depleted, the mixture level predicted by CRAFT remains at a level where it is able to spill into the hot legs and, for most of the time, through the vent valves. System oscillations are observed after 1120 seconds resulting in a decrease in core liquid volume and vessel mixture height. These reductions will soon be overcome as the boil-off rate is already exceeded by the injection rate and therefore will result in an increase in the core liquid volume. No cladding temperature transient will occur since the core is always covered with a mixture and the HPI injection rate has exceeded the boil-off assuring long term cooling capability.

4.3 0.5 Ft^2 Split At Pump Discharge

The break is assumed to occur at the bottom of node 10 of Figure 4.1. The analysis takes credit for two CFTs, one HPI pump and one LPI pump.

The core power, flow rate, pressure, leak rate, ECC water flow and core fluid inventory history are shown in Figures 4.8 through 4.13 respectively. RC pumps and reactor trips occur in less than a second. The HPI actuation signal is received by about 10 seconds when the ESFAS setpoint on low pressure limit is reached. The secondary becomes a source of heat to the primary when, around 40 seconds, the SG primary pressure drops below the secondary pressure. The HPI and CFT flow begins by 45 and 109 seconds respectively. The salient features of system depressurization for this break are similar to those of CFT line break except in the present case, the initiation of LPI flow at 191 seconds results in a quicker termination of the transient. The combined HPI and LPI injection rate exceeds its boil-off due to decay heat by 195 seconds thus establishing long term cooling. No cladding temperature transient will occur since the core is always covered by a mixture.

4.4 0.04 Ft^2 Split At Pump Suction

The break is assumed to be at the bottom of node 9 of Figure 4.1. The analysis takes credit for two CFTs, one HPI and one LPI pump.

Figures 4.14 through 4.19 show core power, flow rate, pressure, leak rate, HPI flow rates and core fluid inventory respectively. No CFT or LPI flow occurred by the end of the analysis since the system pressure remained higher than the CFT actuation pressure of 590 psi and remained higher than the dead head pressure of the LPI. The reactor and R.C. pump trip occurs by about 15 seconds. The depressurization is slower in this case compared to the breaks described previously due to very small size of the break resulting in a lower leak flow. ESFAS setpoint limit on low pressure is reached by 46 seconds and the HPI flow begins by 81 seconds aiding depressurization. The system pressure is still above the secondary pressure and subcooled liquid prevails in the steam generator primary side due to the heat removal by the secondary side of the steam generator. By 120 seconds, two phase mixture is realized in the SG primary side slowing the system depressurization. The slow rate of depressurization continues even after the primary side pressure drops below the secondary side pressure at 770 seconds. The steam generator primary side in the broken loop is filled with steam by 910 seconds. The pump suction nodes in the broken loop contain very little liquid mass after 910 seconds hence the rate of depressurization increases due to the large volume of steam now discharging through the break. The HPI flow rate exceeds the boil-off rate due to decay heat by 1250 seconds, thereby establishing long term cooling. The core is always covered with a mixture hence no cladding temperature transient will occur.

In this particular transient, long term cooling is initially established by use of one HPI pump. No LPI or CFT injection took place since the system pressure was above the injection actuation pressures. Thus, it is concluded that for breaks less than or equal to 0.04 ft^2 the HPI alone is capable of matching decay heat boil-off and maintaining a liquid inventory sufficient to preclude any cladding temperature excursions.

5. PHASE DISTRIBUTION MULTIPLIER FOR THE CRAFT2 WILSON SEPARATION MODEL

The phase separation model available in CRAFT2 includes the option of placing a multiplier on the bubble rise velocity calculated by the Wilson model. Although this velicity is based on the average mixture void fraction, the use of a proper multiplier corrects it for expected non-uniformity in phase distribution within the mixture. Because the mixture swelling is closely tied to the bubble separation rate, varying this multiplier effectively permits tuning the mixture height calculations. For small breaks then, this multiplier can be adjusted so that the mixture calculated by CRAFT2 in the core node matches that produced by the more detailed mixture swelling code, FOAM. This, of course, would preclude the necessity of doing a FOAM analysis as long as the mixture is above the core by an amount corresponding to the level uncertainty.

A series of FOAM calculations were made to determine the amount of liquid necessary to just cover the core with a froth using the axial power shape given in BAW-10074.[6] Results were obtained for pressures ranging from 200 to 1200 psia with a core power of 2.13% and 4.26% of 2772 Mwt. The core parameters used were those of 177 FA plants. Steady state CRAFT2 runs were made using several different bubble rise multipliers for each combination of power, pressure and associated FOAM core liquid volume. The equilibrium core mixture heights calculated by CRAFT2 were noted. Figure 5.1 shows a plot of difference in CRAFT2 and FOAM mixture height versus bubble rise multiplier in CRAFT2. A multiplier of 2.38 as seen in Figure 5.1, produces a CRAFT2 mixture level within + 1.8% and - 2% of that calculated by FOAM over the range of parameters most likely to exist when core uncovering is a possibility during a small break accident. Using a tolerance of $\pm$ 2%, it is concluded that no FOAM analysis is need if the mixture height exceeds the active core by three inches (0.25 feet). If mixture height drops to less than 0.25 feet above the core, the cladding heatup analysis using FOAM results will be done as outlined in Section 5.2.3 of BAW-10104.[5]

6. REFERENCES

1. C.E. Parks, B.M. Dunn, and R.C. Jones, "Multinode Analysis of Small Breaks for B&W's 2568 Mwt Nuclear Plants", BAW-10052, Rev. 1, Babcock & Wilcox, Lynchburg, Va., October 1975.

2. R.A. Hedrick, J.J. Cudlin and R.C. Foltz, "CRAFT2 - Fortran Program For Digital Simulation of a Multinode Reactor Plant During Loss of Coolant," BAW-10092, Rev. 2, Babcock & Wilcox, April 1975.

3. J.F. Wilson, R.J. Grenda, and J.F. Patterson, "The Velocity of Rising Steam in a Bubbling Two-Phase Mixture," ANS Transactions, 5 (1962).

4. B.M. Dunn, C.D. Morgan, and L.R. Cartin, "Multinode Analysis of Core Flooding Line Break for B&W's 2568 Mwt Internals Vent Valve Plants, BAW-10064, Rev. 1, Babcock & Wilcox, October 1975, (FOAM code is discussed in this reference).

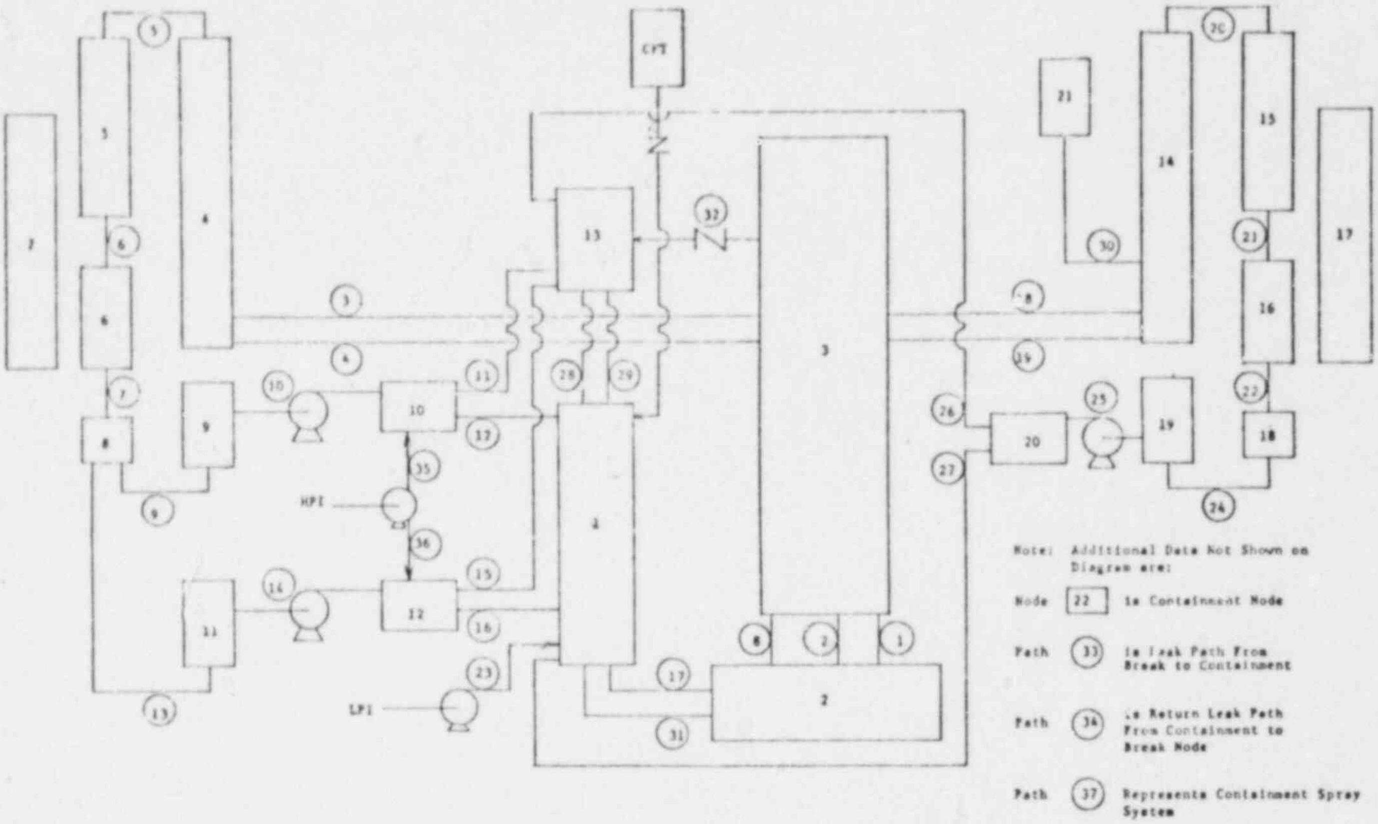
5. B.M. Dunn, R.C. Jones, L.R. Cartin, C.E. Parks, and R.J. Salm,"B&W's ECCS Evaluation Model", BAW-10104A, Rev. 1, Babcock & Wilcox, Lynchburg, Va, March 1976.

6. R.C. Jones, B.M. Dunn, and C.E. Parks, "Multinode Analysis of Small Breaks for B&W's 205 Fuel Assembly Nuclear Plants With Internals Vent Valves", BAW-10074A, Rev. 1, Babcock & Wilcox, Lynchburg, Va., March 1976.

7. R.C. Jones, J.R. Biller, and B.M. Dunn, "ECCS Analysis of B&W's 177-FA Lowered Loop NSS", BAW-10103, Rev. 2, Babcock & Wilcox, Lynchburg, Va., April 1976.

FIGURE 4-1

CRAFT Noding Diagram For Small Break



Node No.	Identification
1	Downcomer
2	Lower Plenum
3	Core, Core Bypass, Upper Plenum, Upper Head
4,14	Hot Leg Piping
5,15	Steam Generator Upper Head, SG Tubes (Upper Half)
6,16	SG Tubes (Lower Half)
8,18	SG Lower Head
9,11,19	Cold Leg Piping (Pump Suction)
10,12,20	Cold Leg Piping (Pump Discharge)
13	Upper Downcomer (Above the ℄ of Nozzle Belt)
21	Pressurizer
22	Containment

Path No.	Identification
1,2	Core
3,4,18,19	Hot Leg Piping
5,20	Hot Leg, Upper
6,21	SG Tubes
7,22	SG Lower Head
8	Core Bypass
9,13,24	Cold Leg Piping
10,14,25	Pumps
11,12,15,16,26,27	Cold Leg Piping
17,31	Downcomer
23	LPI
28,29	Upper Downcomer
30	Pressurizer
32	Vent Valve
33,34	Leak & Return Path
35,36	HPI
37	Containment Sprays

FIGURE - 4.2

CORE POWER FOR CFT LINE BREAK

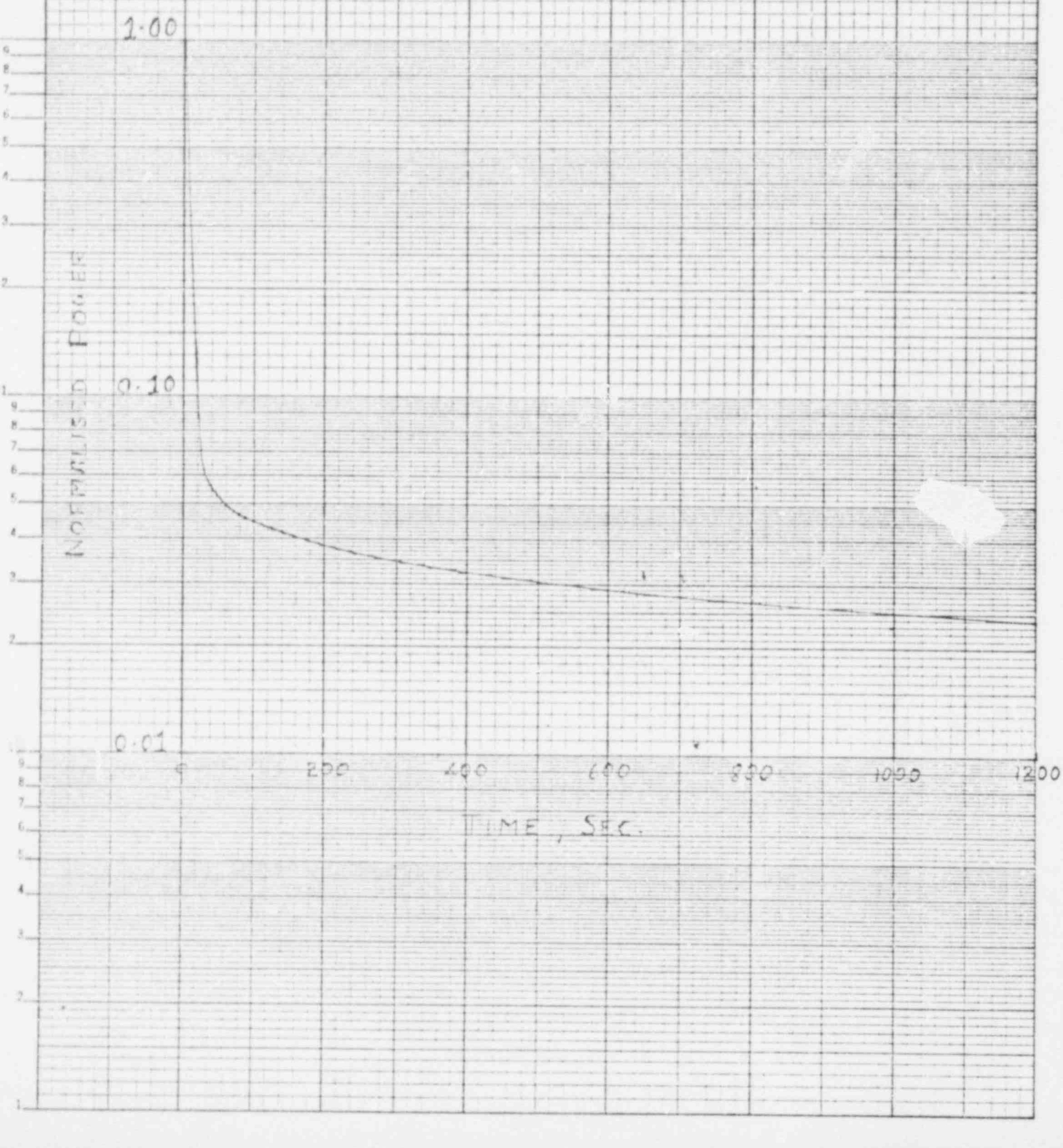


FIGURE - 4.3

CORE FLOW FOR CFT
LINE BREAK

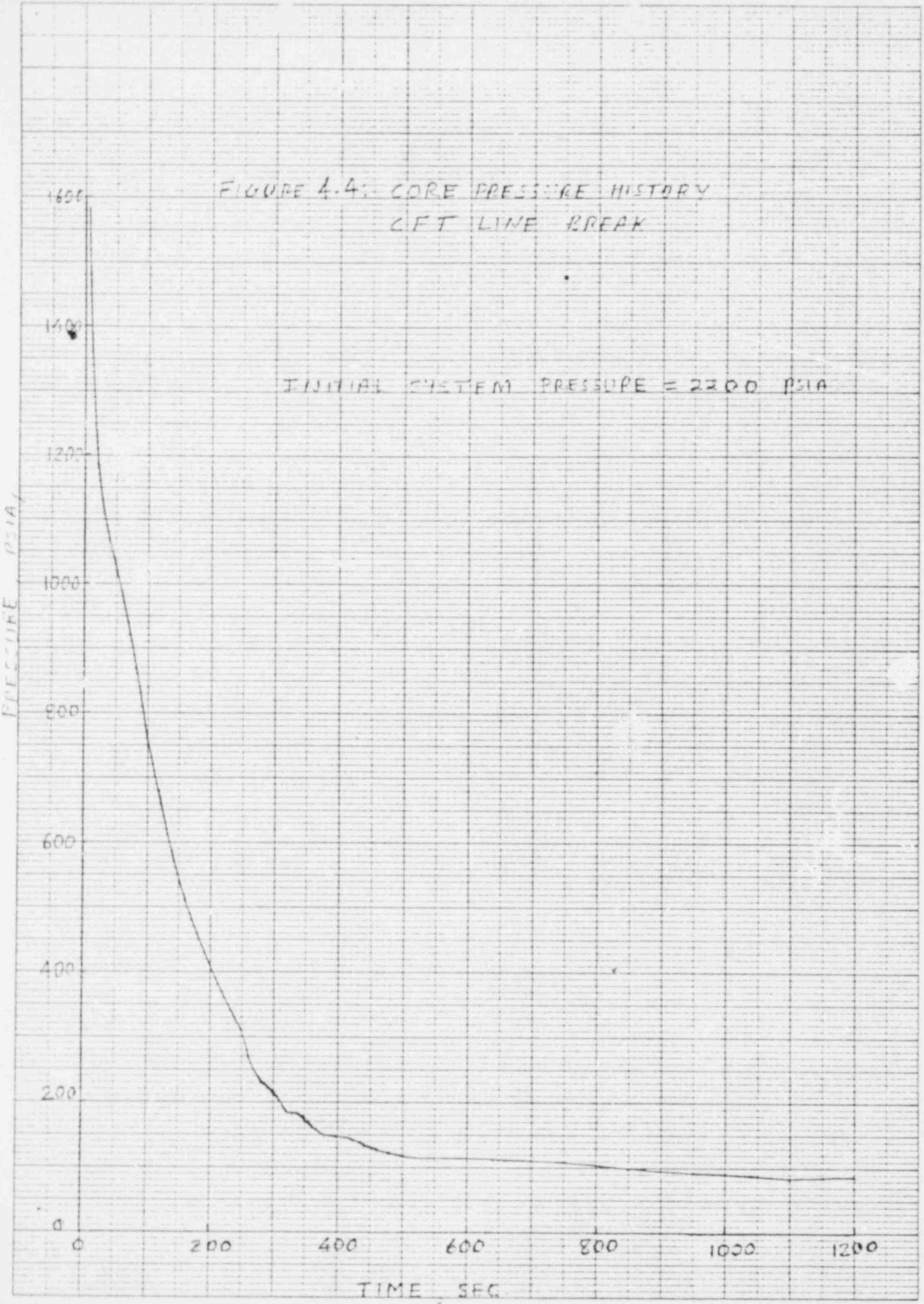


FIGURE 4.4: CORE PRESSURE HISTORY
CFT LINE BREAK

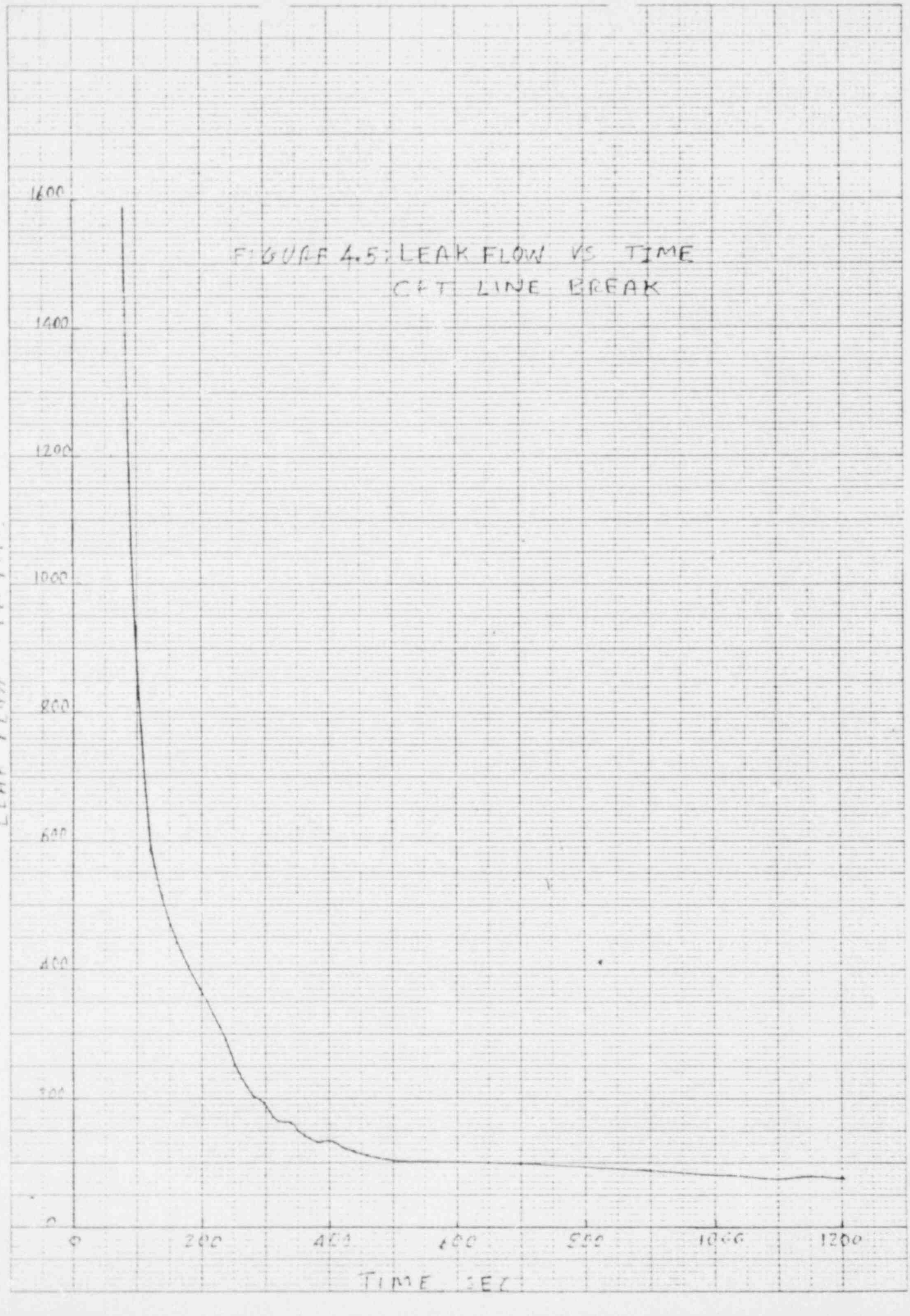
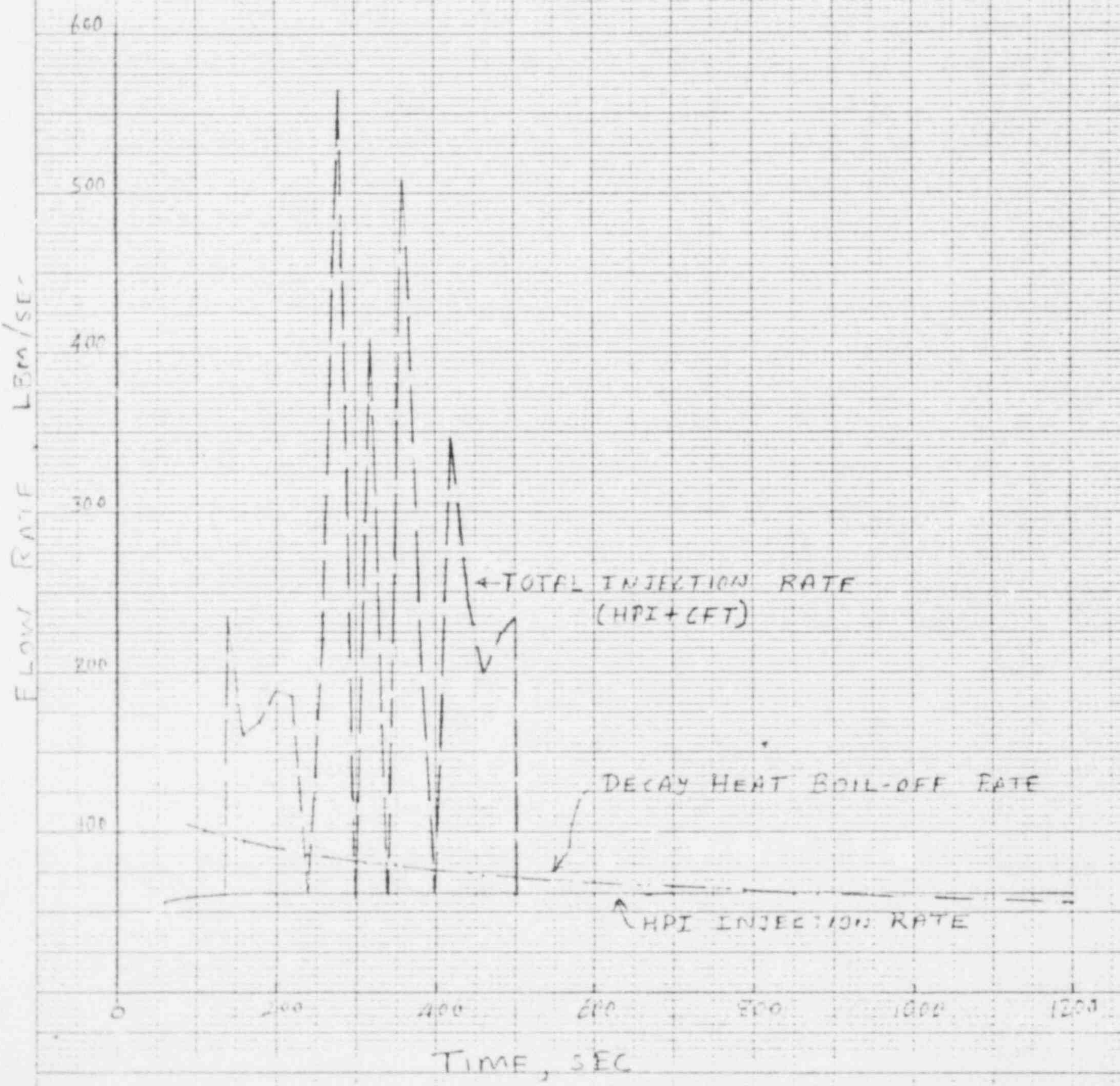
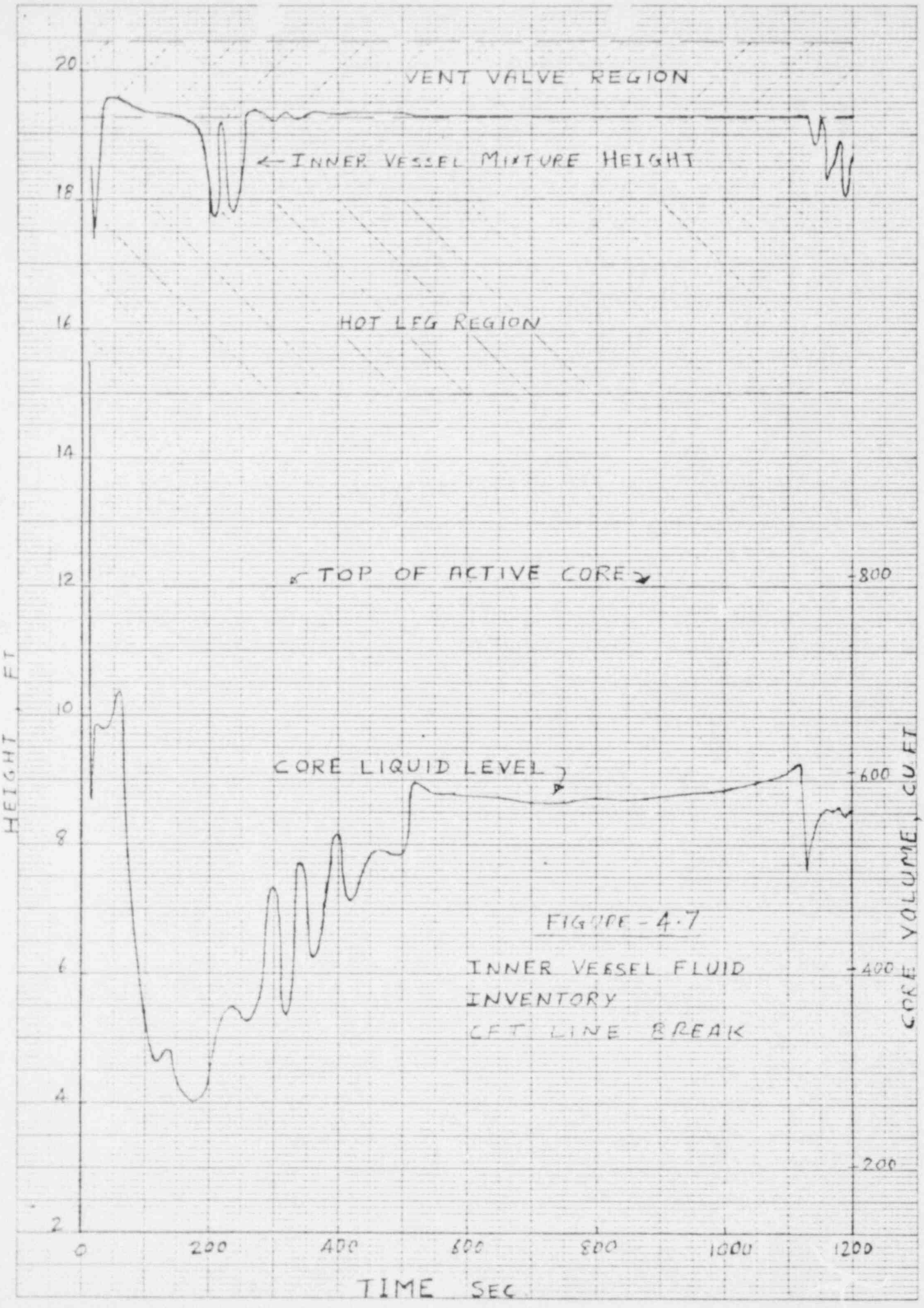


FIGURE 4.5: LEAK FLOW VS TIME
CFT LINE BREAK

FIGURE 4.6: ECC INJECTION RATE
CFT LINE BREAK





VENT VALVE REGION
INNER VESSEL MIXTURE HEIGHT
HOT LEG REGION
TOP OF ACTIVE CORE
CORE LIQUID LEVEL
FIGURE - 4.7
INNER VESSEL FLUID
INVENTORY
CFT LINE BREAK
HEIGHT, FT
CORE VOLUME, CU FT
TIME SEC
20
18
16
14
12
10
8
6
4
2
800
600
400
200
0
200
400
600
800
1000
1200

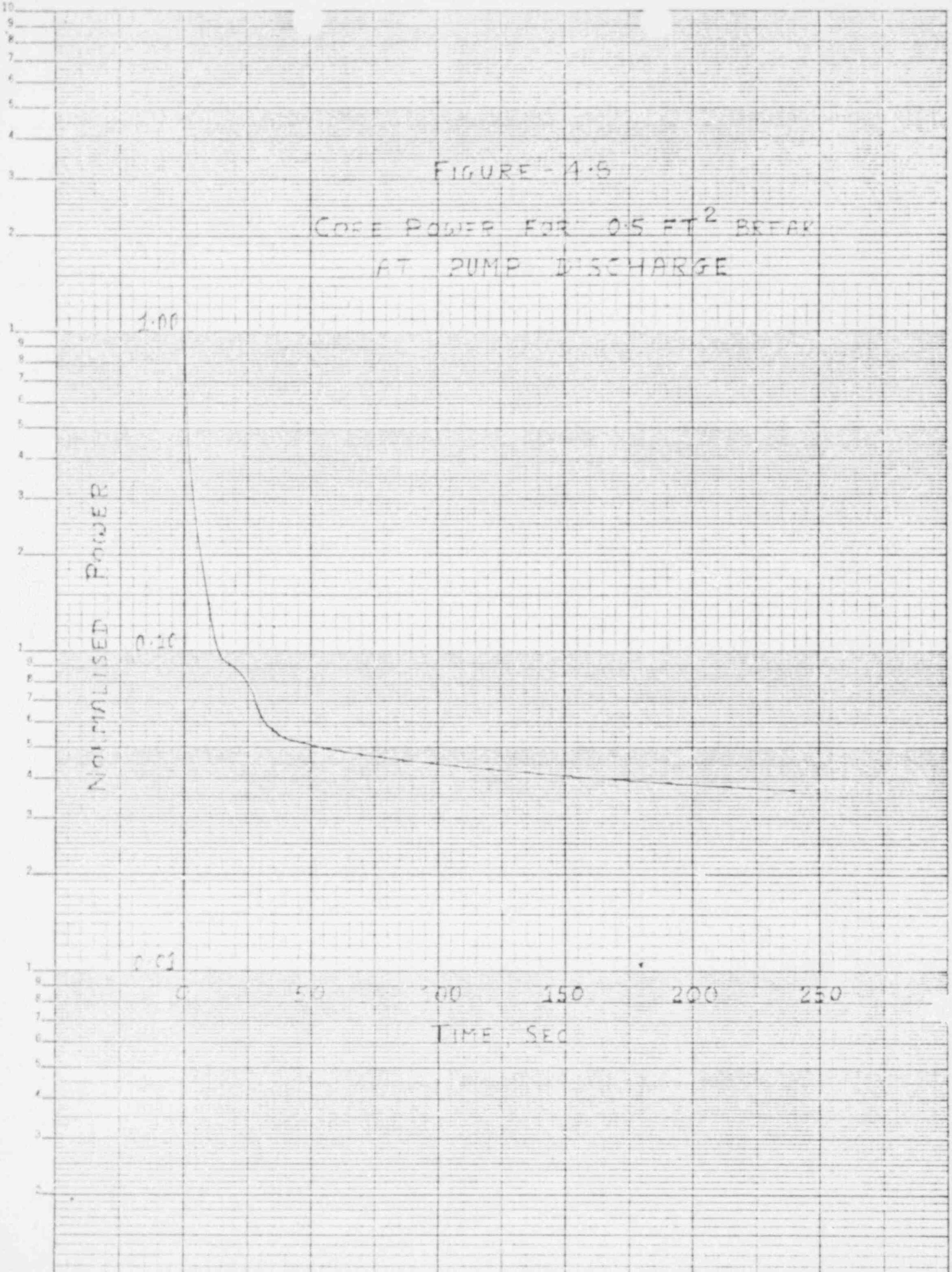


FIGURE - 4.5
CORE POWER FOR 0.5 FT² BREAK
AT PUMP DISCHARGE
NORMALISED POWER
1.00
0.10
0.01
0
50
100
150
200
250
TIME, SEC

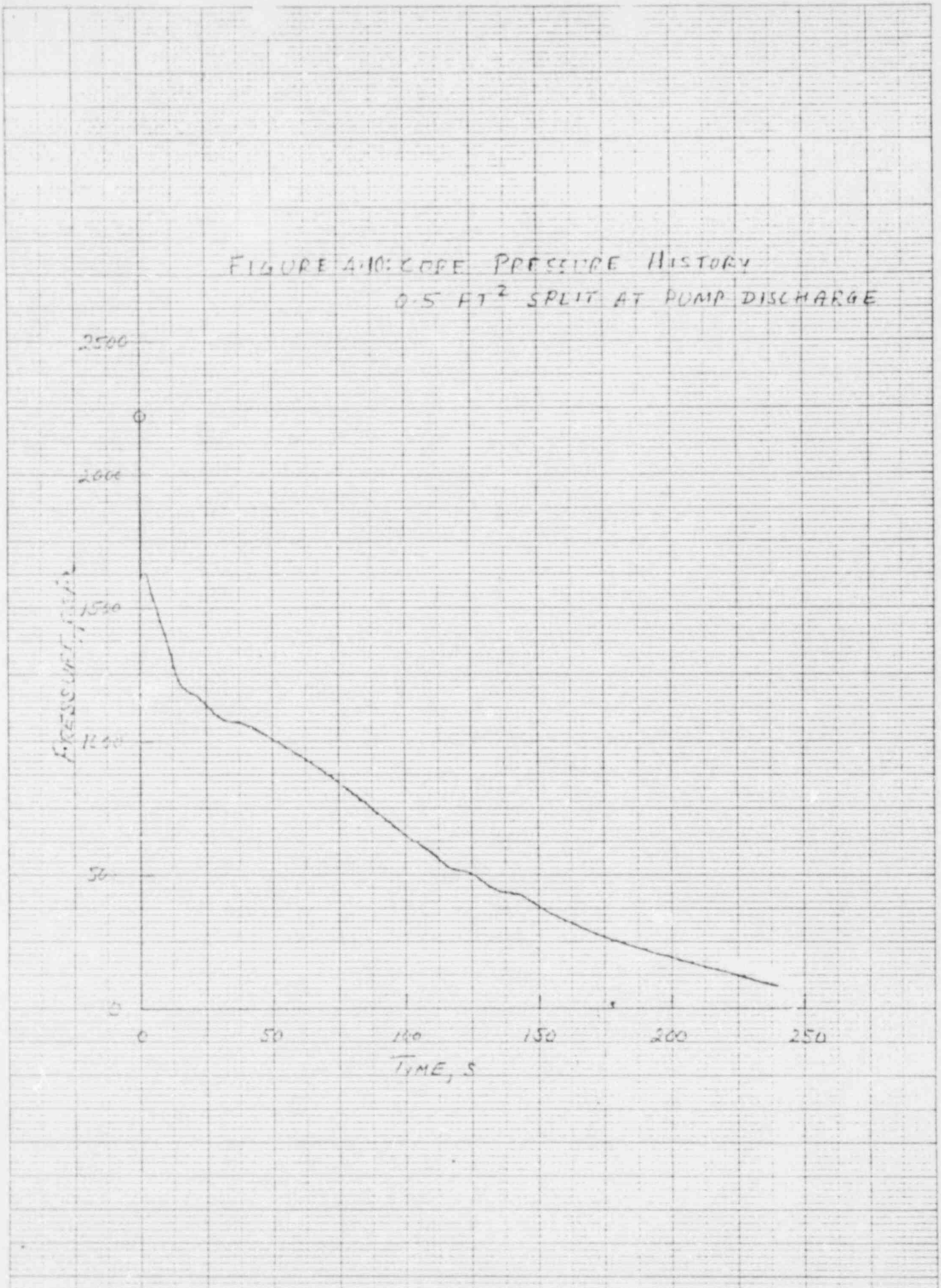


FIGURE 4.10: CORE PRESSURE HISTORY
0.5 FT^2 SPLIT AT PUMP DISCHARGE
2500
2000
1500
1000
500
0
PRESSURE, PSIA
0
50
100
150
200
250
TIME, S

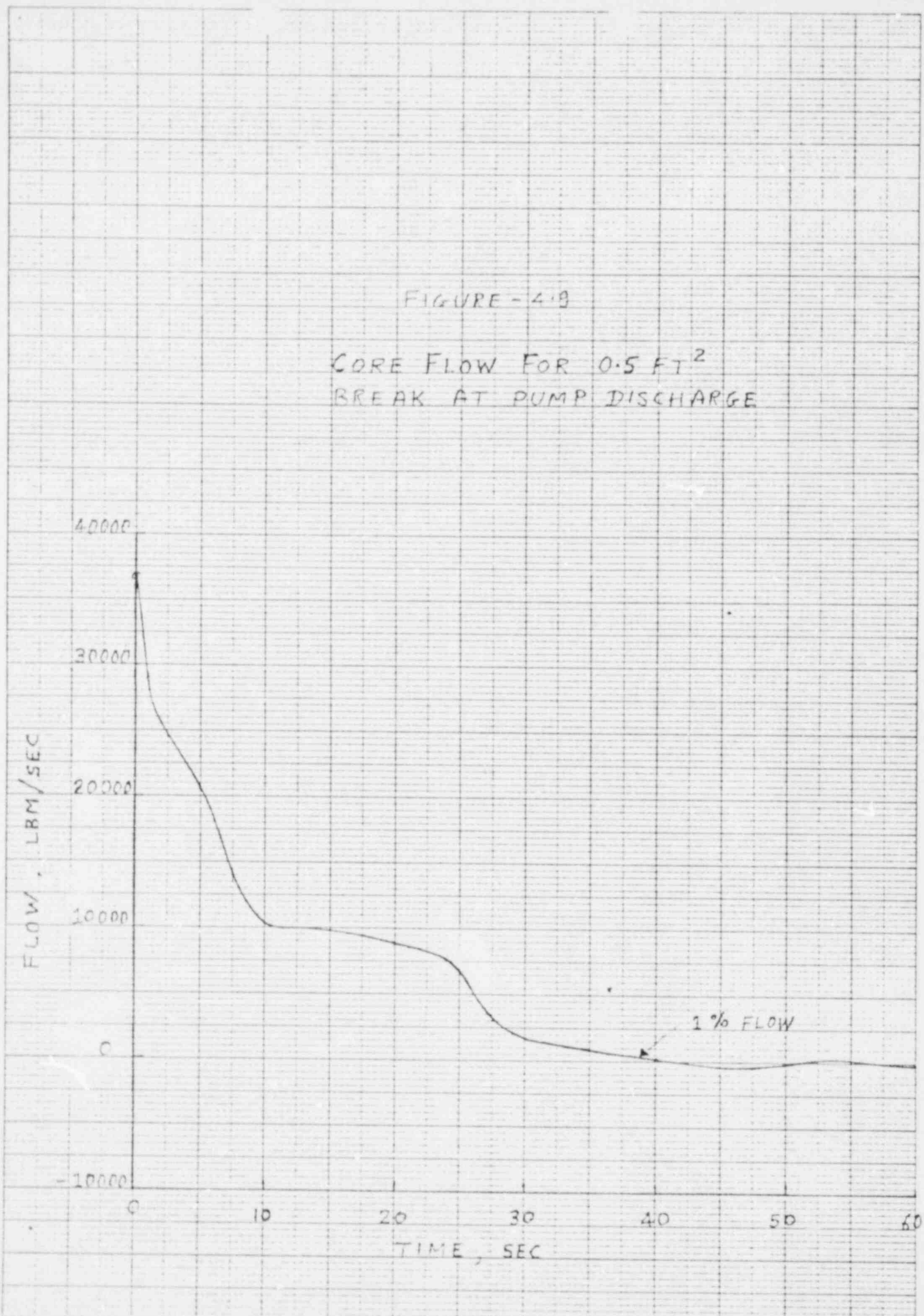


FIGURE - 4.9
CORE FLOW FOR 0.5 FT^2
BREAK AT PUMP DISCHARGE
40000
30000
20000
10000
0
-10000
FLOW, LBM/SEC
1% FLOW
0
10
20
30
40
50
60
TIME, SEC
46 1320

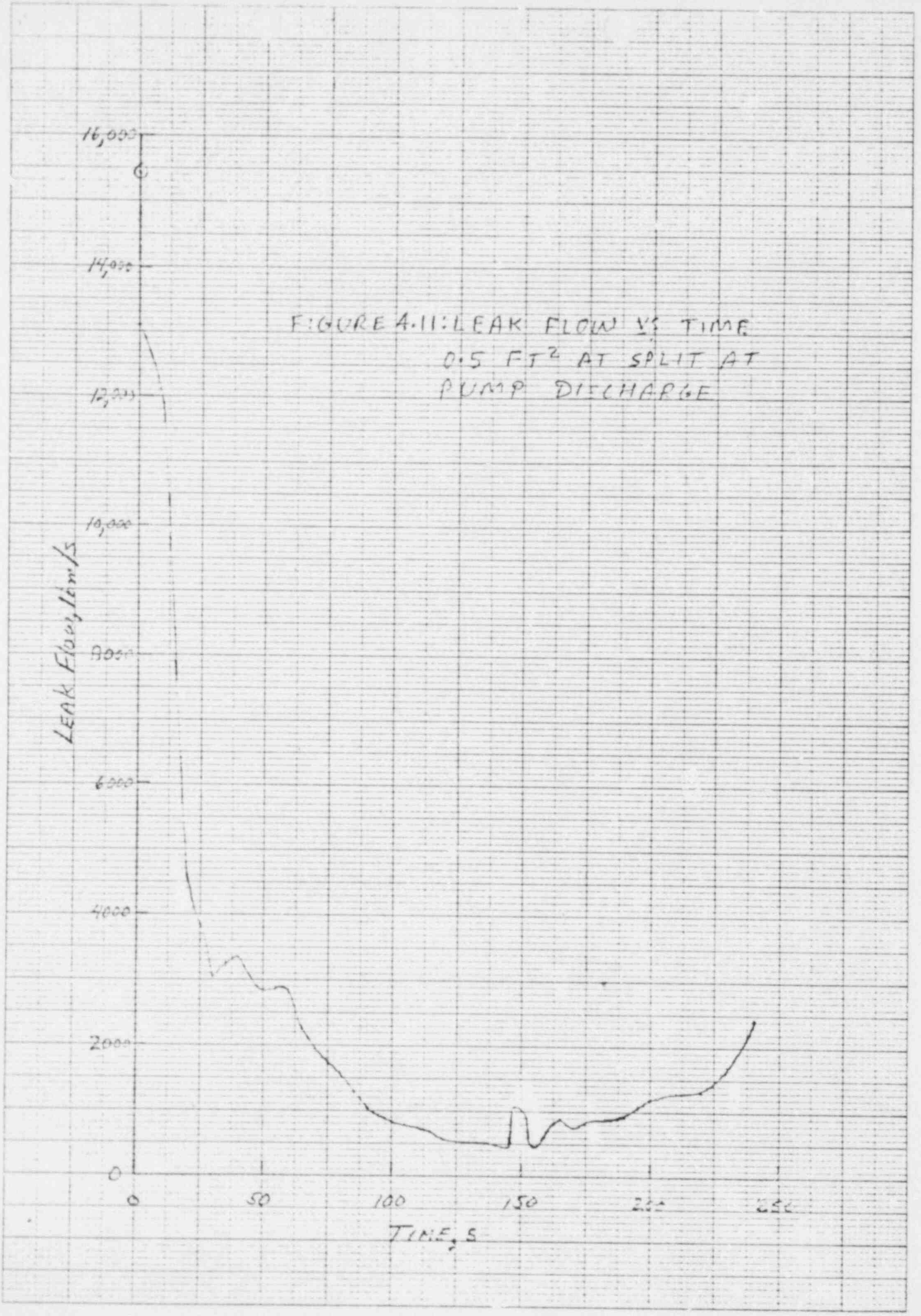
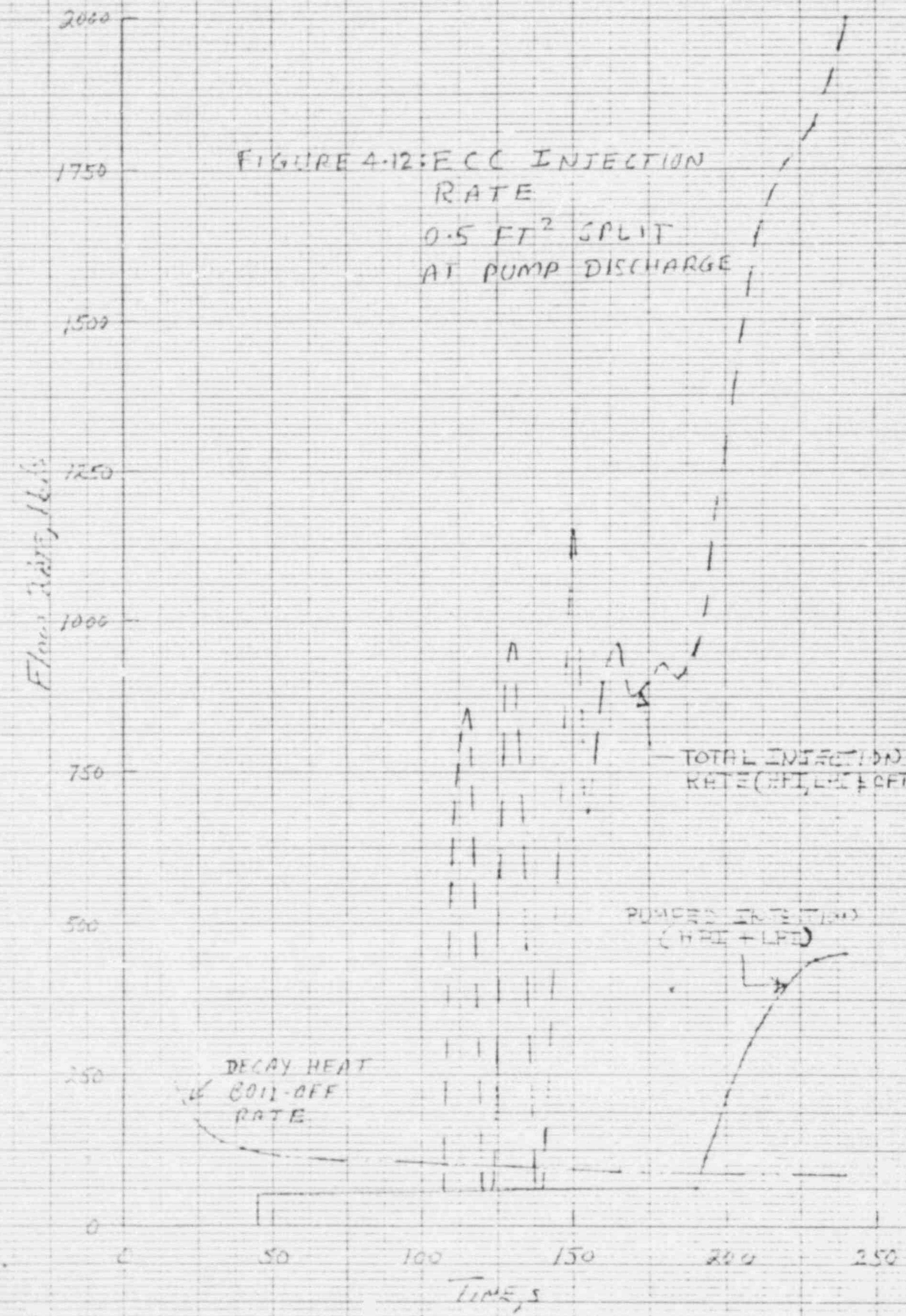
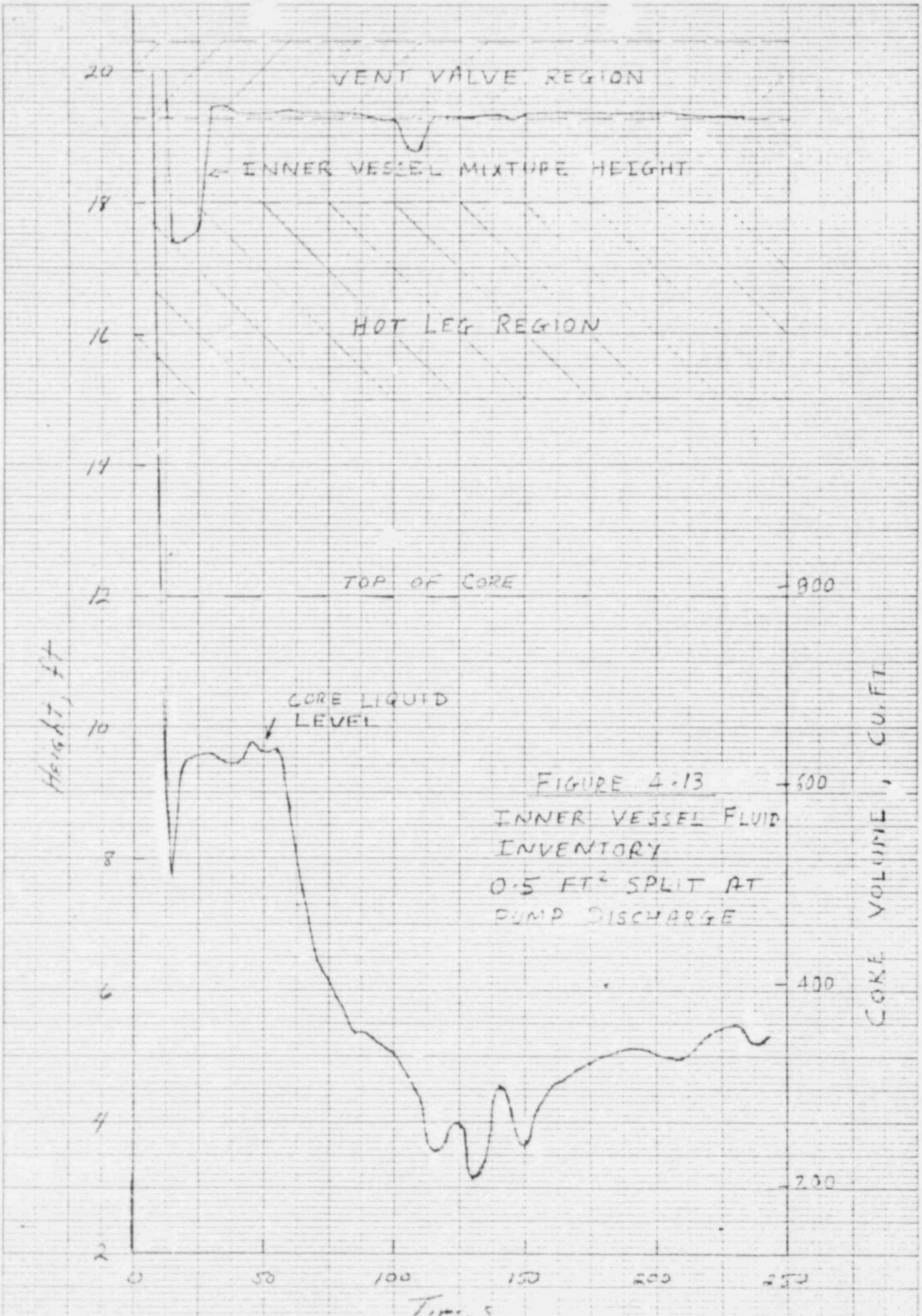


FIGURE A.11: LEAK FLOW VS TIME
0.5 FT^2 AT SPLIT AT
PUMP DISCHARGE

FIGURE 4.12: ECC INJECTION RATE
0.5 FT^2 SPLIT
AT PUMP DISCHARGE




VENT VALVE REGION
INNER VESSEL MIXTURE HEIGHT
HOT LEG REGION
TOP OF CORE
CORE LIQUID LEVEL
FIGURE 4.13
INNER VESSEL FLUID INVENTORY
0.5 FT² SPLIT AT PUMP DISCHARGE
Height, Ft
Core Volume, Cu. Ft
Time, s

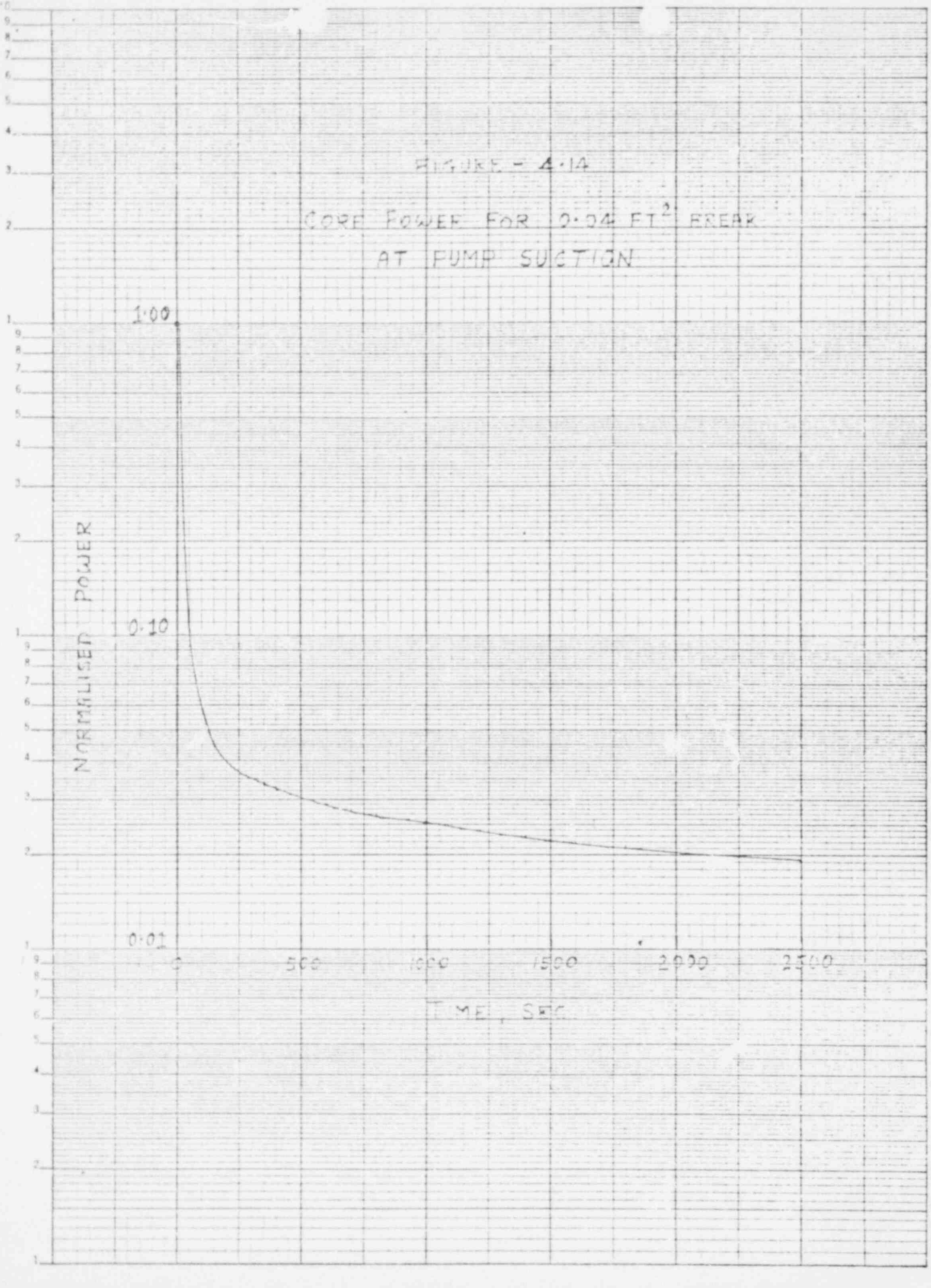
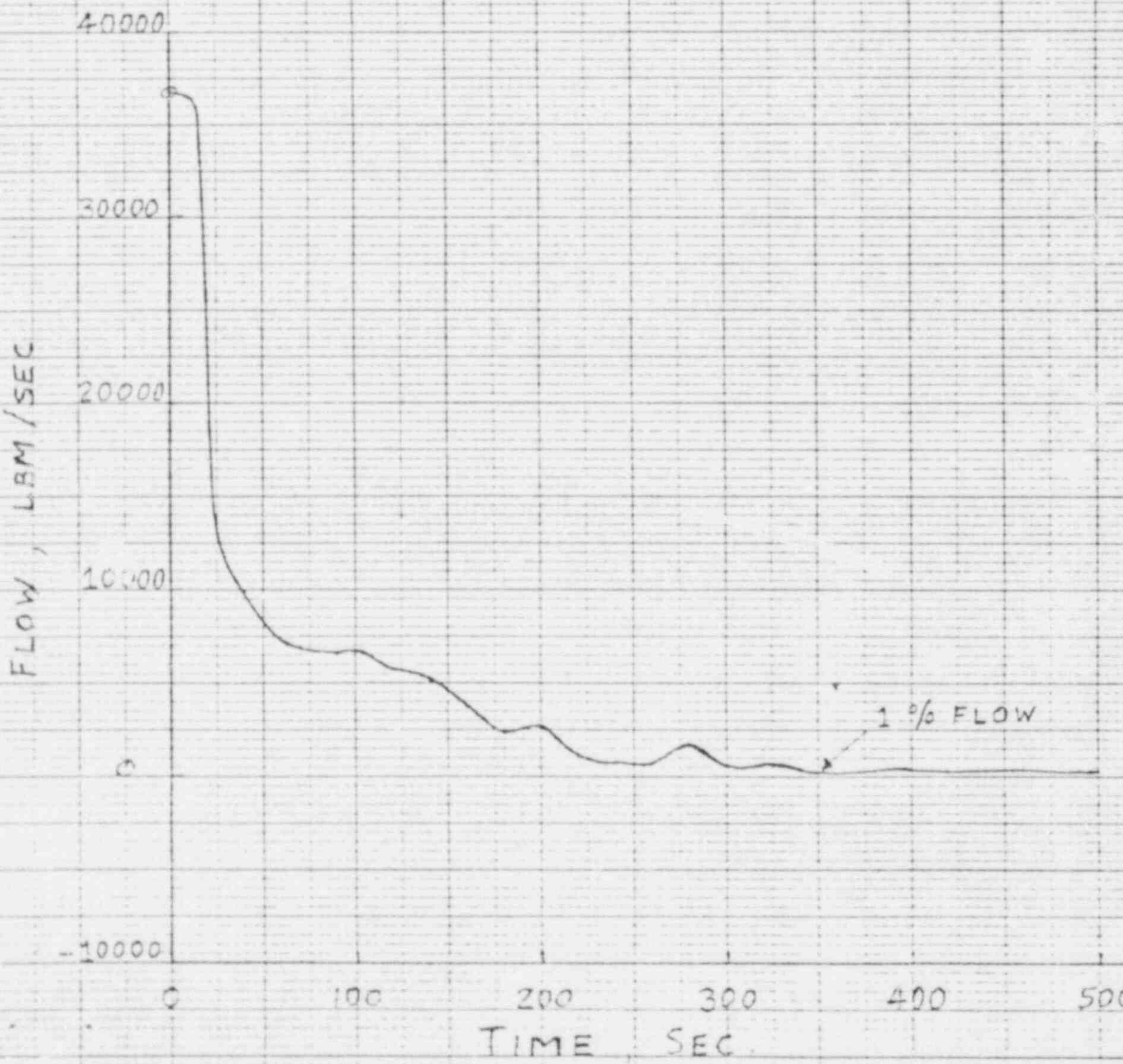
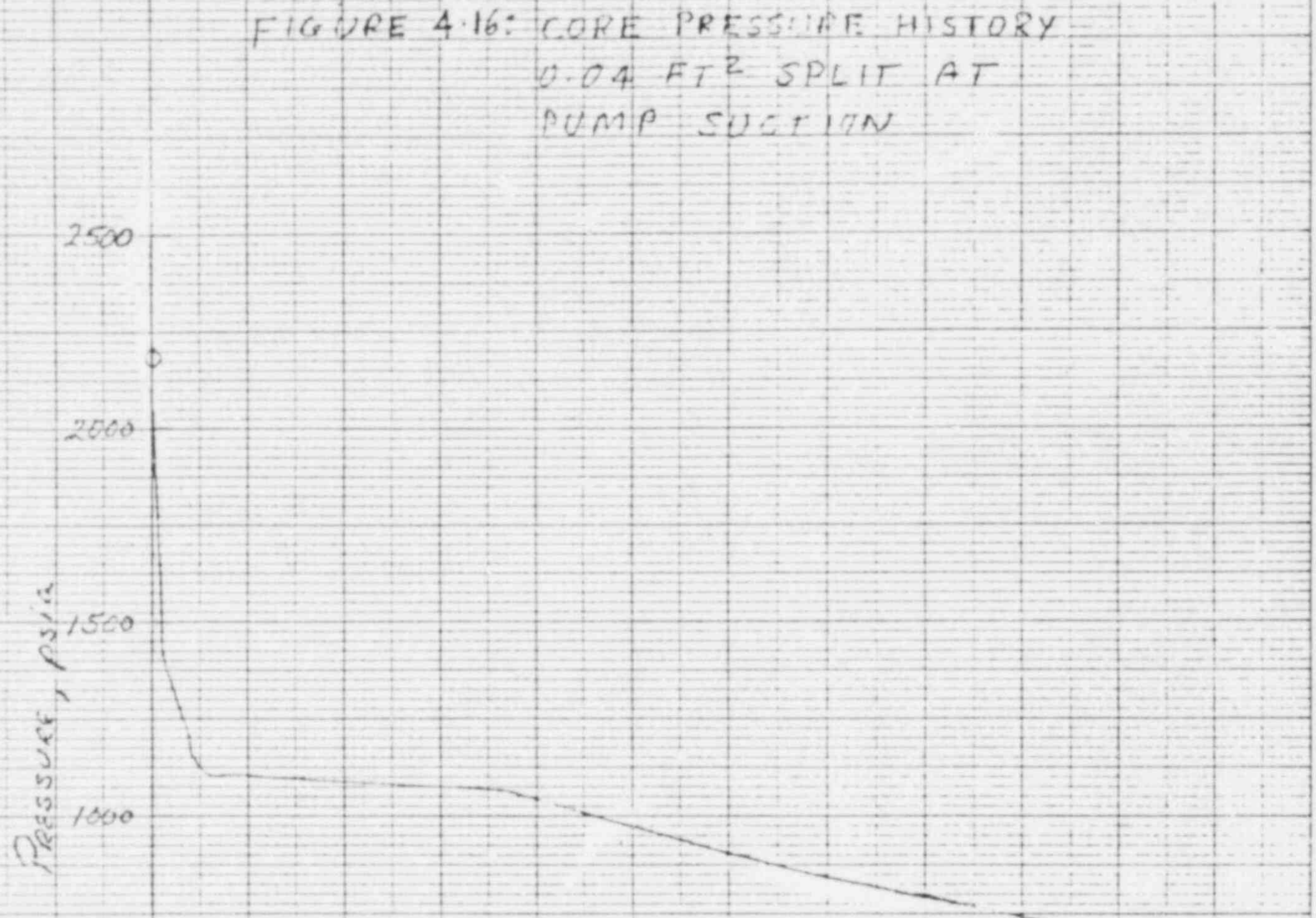


FIGURE - 4.14
CORE POWER FOR 0.04 FT² BREAK
AT PUMP SUCTION
1.00
0.10
0.01
NORMALISED POWER
0
500
1000
1500
2000
2500
TIME, SEC

FIGURE - 4.15

CORE FLOW FOR 0.04 FT^2
BREAK AT PUMP SUCTION





46 1322

K&E 10 X 10 TO 1/2 INCH 7 X 10 INCHES KEUFFEL & ESSER CO.

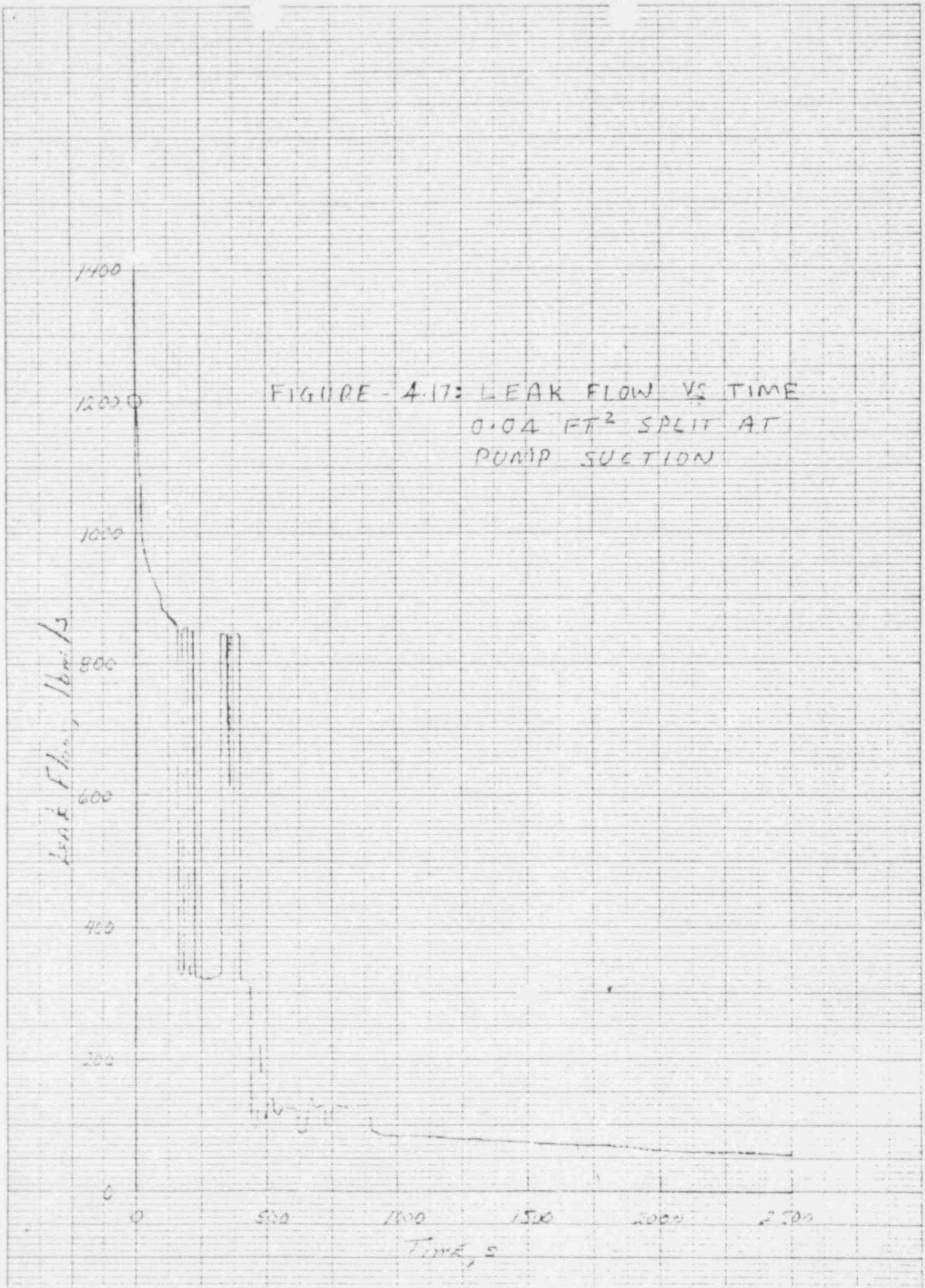
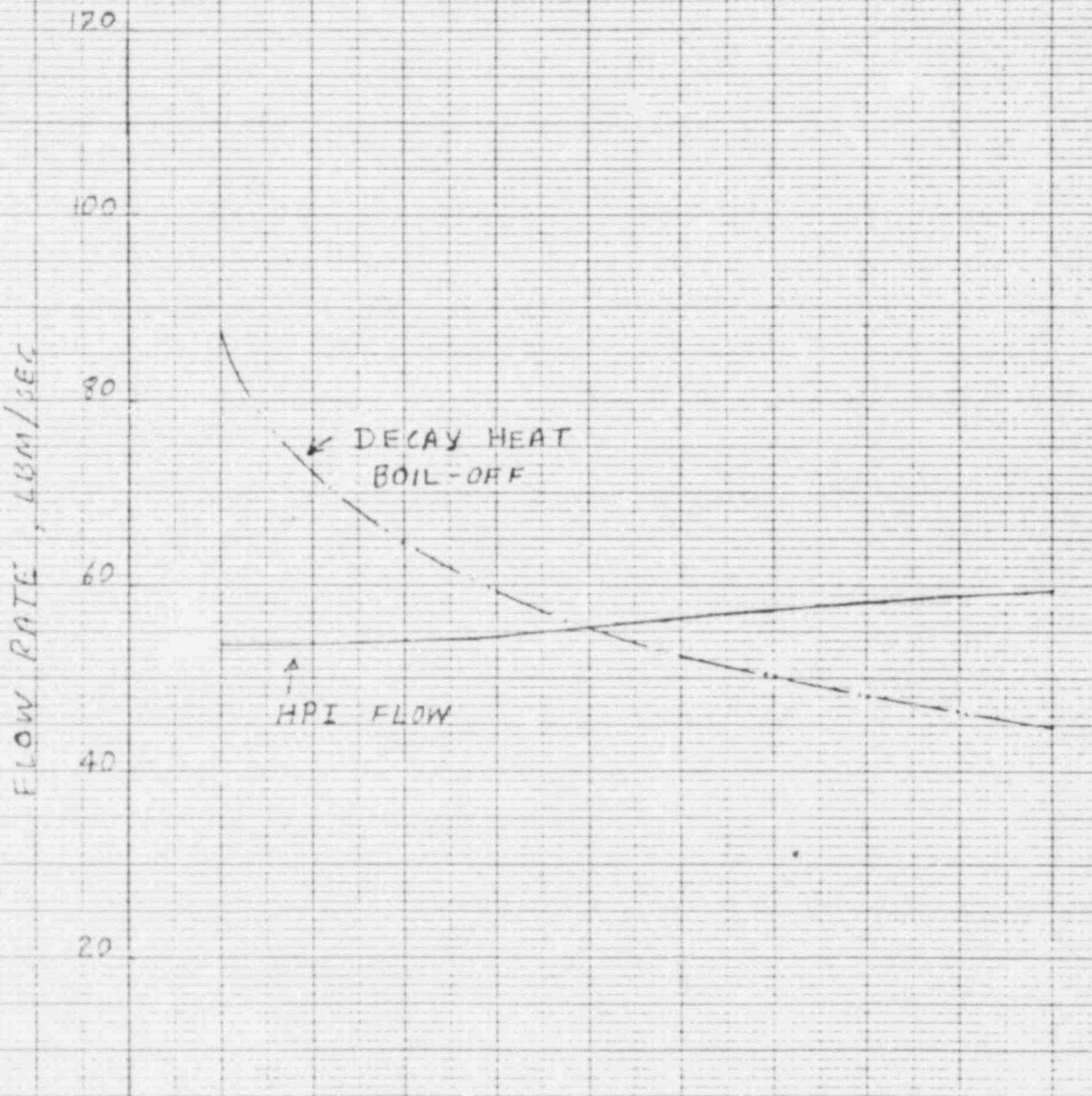


FIGURE - 4.17: LEAK FLOW VS TIME
0.04 FT² SPLIT AT
PUMP SUCTION

FIGURE - 4.18
ECC INJECTION RATE
0.04 FT^2 SPLIT AT PUMP
SUCTION



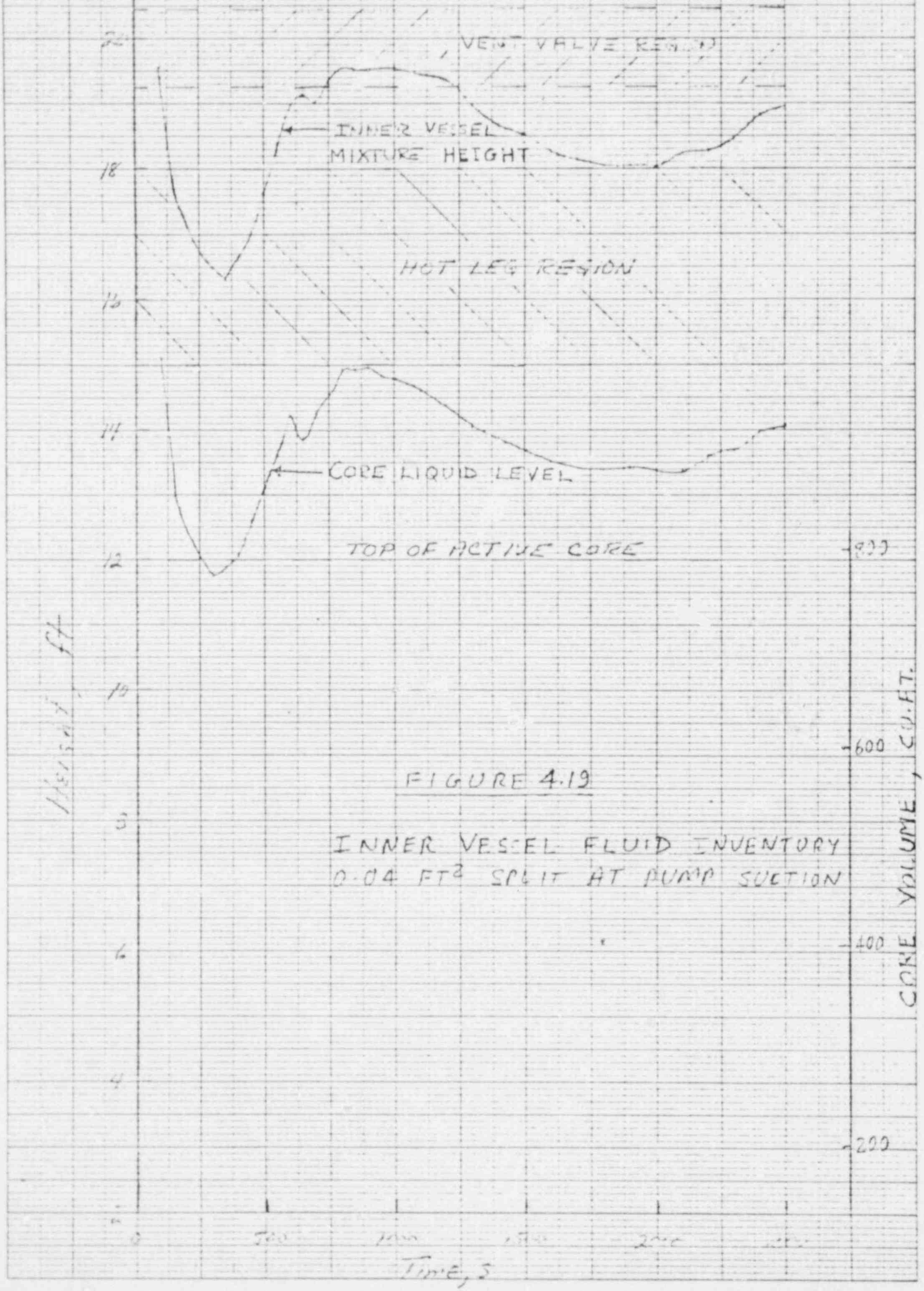


FIGURE 4.19

INNER VESSEL FLUID INVENTORY
0.04 FT^2 SPLIT AT PUMP SUCTION

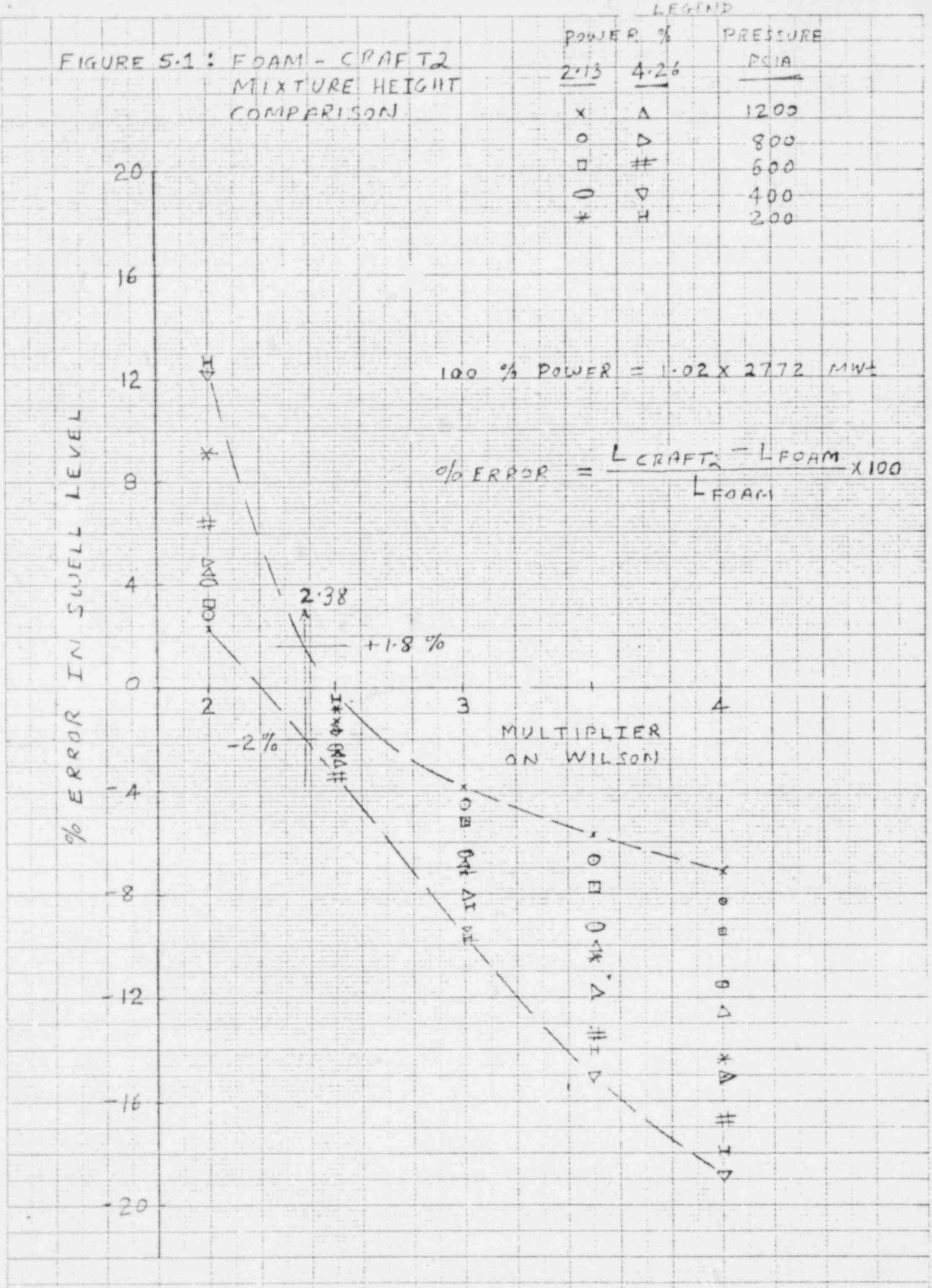


FIGURE 5.1: FOAM - CRAFT2 MIXTURE HEIGHT COMPARISON

TO: G.R. Mazetis

RCDY 12/16/76

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RENE

Here is a better writeup. Use previous figures, if you can. I'm trying to get the set that goes with this writeup.

Jerry

LIST OF FIGURES

1. INTRODUCTION

Presented here are the results of an analysis of hypothetical loss-of-coolant accidents resulting from postulating small breaks in the reactor coolant system of Crystal River 3. B&W's ECCS Evaluation Model as defined in BAW-10104[5] was used for the analysis.

Small breaks are defined as ruptures of the reactor coolant system with leak areas of 0.5 ft^2 or less. Break areas considered for this study are: (1) 0.44 ft^2 CFT line break since it has the minimum ECC system available to mitigate the LOCA, (2) 0.5 ft^2 break at the pump discharge as it represents the transition break between the small and large break models, (3) 0.04 ft^2 break at the pump suction which was shown to be the most limiting small break in BAW-10052.[1]

The current analysis of these breaks verifies the conservative nature of the cladding temperature calculations presented for the same breaks in BAW-10052 and BAW-10064[4]. Therefore, the analysis contained herein,coupled with the analyses of BAW-10052 and BAW-10064 provide an appropriate spectrum of breaks for the evaluation of the effects of small leaks and the demonstration of the ability of the ECCS to effectively control them.

2. SUMMARY AND CONCLUSIONS

The various breaks analyzed in Section 4 of this report show that the core remains covered throughout the transient. During the initial period when the transient is flow controlled, sufficient flow is maintained such that CHF does not occur and nucleate boiling heat transfer predominates. Since the core remains covered by a mixture, pool film boiling will be maintained during the quiescent period of the accident. This heat transfer mechanism is sufficient to maintain the cladding temperature within a few degrees of the fluid saturation temperature (Reference 4, BAW-10064). Therefore, for the maximum linear heat rate covered by BAW-10103[7] the transient cladding surface temperature will never exceed its initial value of 660F, no metal water reaction will occur, and the core geometry will remain coolable as no cladding rupture will occur. Long term cooling is established as the HPI and LPI pumped injection systems provide fluid in excess of the boiloff rate due to core decay heat. Thus the five Acceptance Criteria in 10CFR50.46 are met.

The new modeling techniques used in the CRAFT2[2] analyses for the present studies show improvement in the core performance when compared with the results of the same breaks reported in BAW-10052 and BAW-10064. Therefore, if all the breaks reported in BAW-10052 and BAW-10064 were re-analyzed with the present model, the same trend of improvement in core performance would be realized. Thus, the present analyses in conjunction with the analyses of BAW-10052 and BAW-10064 provide a suitable small break spectrum for demonstration of compliance of the ECC system with the five Acceptance Criteria in 10CFR50.46.

METHOD OF ANALYSIS

The analysis uses the CRAFT2[2] code to develop the history of the reactor coolant system hydrodynamics. For small leak analysis it is sufficient to use smaller models than are used for large loss-of-coolant studies because hydrodynamic responses are slow enough for simpler models to describe them.[1] The CRAFT model uses 19 nodes to simulate the reactor coolant system, two nodes for the secondary system, and one node for the reactor building. A schematic diagram of the model is shown in Figure 4.1 along with the node descriptions. Where? Control volumes (nodes) in and around the vessel are all connected by a pair of flow paths to allow the occurrence of counter-current flow. The break is located in the cold leg piping either at the lowest point in the pipe at the pump suction, or at a point opposed to the high-pressure injection nozzle at the pump discharge, or in the core flood line joining the CF nozzle. The Wilson, Grenda and Patterson[3] average bubble rise model is used for all nodes. ✓ Within the core region, however, a multiplier of 2.38 is applied to the calculated bubble rise velocity. ? Section 5 of this report demonstrates that a multiplier of 2.38 in CRAFT2 gives a mixture height within ± 2% of that predicted by FOAM[4]. Thus, no FOAM analysis will be needed if the CRAFT2 mixture level remains above the core by 2% of the active length.

The following assumptions are made for conditions and system responses during the accident:

1. The reactor is operating at 102% of the steady-state power level of 2772 Mwt.
2. The leak occurs instantaneously, and a discharge coefficient of 1.0 is used for the entire analysis. Bernoulli's equation was used for the subcooled portion of the transient while Moody's correlation was used in the two phase portion.
3. No offsite power is available.
4. The reactor trips on low pressure at 1900 psia.
5. The safety rods begin entering the core after a 0.5 second delay from the time the reactor trip signal is reached.

6. The reactor coolant pumps trip and coast down coincident with reactor trip.
7. One complete train of the emergency safeguards system fails to operate, leaving two CFTs and only one high-pressure injection and one low-pressure injection system available for pumped injection to mitigate the consequence of a cold leg break. For the CFT line break, only one CFT and one high pressure injection system is assumed available for providing ECC fluid to the vessel.
8. The auxilliary feedwater system is assumed to be available during the transient. It mainly removes heat from the upper half of the steam generator during the initial stages of the transient. When the secondary side of the steam generator becomes a source of heat to the primary system, the assumption of auxiliary feedwater maximizes the energy that must be relieved.
9. ESFAS signal error band is considered in the analysis to signal the actuation of the HPI.

The CRAFT2 results obtained from the present analysis are sufficient to meet the five Acceptance Criteria of 10CFR50.46 in that the core was always covered by a two-phase mixture, hence no separate thermal analysis is necessary for cladding temperatures during the transient. If required, as in the case of uncovery to within 0.25 feet above the active core (Ref. Section 5), the cladding heatup can be calculated by the procedure outlined in Section 5.2.3 of BAW-10104A.

4. RESULTS OF SMALL LEAK TRANSIENTS

This section presents a detailed evaluation of the three breaks considered along with explanations of the phenomena involved.

4.1 Explanation of Curves

The following categorical explanations are provided to aid in understanding the parameters illustrated in the curves:

Core Power: This curve indicates the normalized thermal power as calculated by CRAFT2.

Core Flow: This curve represents the total flow rates of core paths 1 and 2 of Figure 4.1. The curve shows flow rates mainly during the flow controlled part of the transient.

Pressure: This is the pressure at the top of the core node as calculated by CRAFT. The core node, in these analyses, includes the core, upper plenum, upper head, and the core bypass.

Boil-Off Due To Decay Heat: The liquid boil-off rate is given in terms of equivalent amount of HPI or HPI + LPI injection rate needed to dissipate the core decay heat. Mathematically:

$$\text{Boil-off rate} = (\text{core decay heat rate})\ (h_g - h_{in})$$

Where: h_g = enthalpy of saturated steam at core pressure

h_{in} = enthalpy of injected water

Inner Vessel Mixture Height: This curve shows the mixture height in the core node as calculated by the CRAFT code. The lines spanning the curve indicates the top of the active core, hot leg regions and the vent valve region.

Core Liquid Level: This curve, ~~in contrast to that for the inner vessel mixture height~~ shows the effective core liquid height and volume with the lower plenum filled with a mixture at the void fraction calculated by CRAFT2. This volume is representative of the liquid volume within the core node that would be used to calculate the mixture height within the core.

4.2 0.44 Ft^2 CFT Line Br.

The break is assumed to be at the CFT ... joining the reactor vessel and is limited in area to 0.44 ft^2 by the nozzle insert in the CFT line. Node 13 in Figure 4.1 is the break node, and the analysis takes credit for one CFT and one HPI pump.

Figures 4.2 and 4.3 show core power and core flow rate respectively. Rapid initial depressurization (see Figure 4.4) causes reactor trip and start of reactor coolant pump coastdown within the first second. Flashing of system liquid slows the depressurization while the steam generator continues to remove energy from the primary coolant thereby helping to decrease the pressure. The lower pressure limit of the ESFAS setpoint error band is reached by about 10 seconds which initiates main feedwater and steam line isolation procedures and signals actuation of HPI. At about 40 seconds, the RCS pressure drops below the secondary steam generator pressure and heat removal to the secondary side drops off sharply and becomes a source of heat to the primary causing a slower depressurization. System flow has degraded such that core flow is predominately due to natural circulation and quiescent period of transient begins. HPI system provides makeup starting at about 50 seconds aiding depressurization. Core flood tank flow begins at 140 seconds aiding further depressurization but the diminishing leak flow (Figure 4.5) slows the depressurization rate. The core flood tanks are emptied by about 500 seconds after which the rate of depressurization is steady but very slow. Figure 4.6 is a plot of HPI and total ECC water flow. Long term cooling is assured in that by 850 seconds the HPI injection rate exceeds the boil-off due to core decay heat. Figure 4.7 is a plot of core liquid inventory and mixture height. It shows that while much of the core liquid inventory is depleted, the mixture level predicted by CRAFT remains at a level where it is able to spill into the hot legs and, for most of the time, through the vent valves. System oscillation are observed after 1120 seconds resulting in a decrease in core liquid volume and vessel mixture height. These reductions will soon be overcome as the boil-off rate is already exceeded by the injection rate and therefore will result in an increase in the core liquid volume. No cladding temperature transient will occur since the core is always covered with a mixture and the HPI injection rate has exceeded the boil-off assuring long term cooling capability.

4.3 $0.5\ Ft^2$ Split At Pump Discharge

The break is assumed to occur at the bottom of node 10 of Figure 4.1. The analysis takes credit for two CFTs, one HPI pump and one LPI pump.

The core power, flow rate, pressure, leak rate, ECC water flow and core fluid inventory history are shown in Figures 4.8 through 4.13 respectively. RC pumps and reactor trips occur in less than a second. The HPI actuation signal is received by about 10 seconds when the ESFAS setpoint on low pressure limit is reached. The secondary becomes a source of heat to the primary when, around 40 seconds, the SG primary pressure drops below the secondary pressure. The HPI and CFT flow begins by 45 and 109 seconds respectively. The salient features of system depressurization for this break are similar to those of CFT line break except in the present case, the initiation of LPI flow at 191 seconds results in a quicker termination of the transient. The combined HPI and LPI injection rate exceeds its boil-off due to decay heat by 195 seconds thus establishing long term cooling. No cladding temperature transient will occur since the core is always covered by a mixture.

4.4 $0.04\ Ft^2$ Split At Pump Suction

The break is assumed to be at the bottom of node 9 of Figure 4.1. The analysis takes credit for two CFIs, one HPI and one LPI pump.

Figures 4.14 through 4.19 show core power, flow rate, pressure, leak rate, HPI flow rates and core fluid inventory respectively. No CFT or LPI flow occurred by the end of the analysis since the system pressure remained higher than the CFT actuation pressure of 590 psi and remained higher than the dead head pressure of the LPI. The reactor and R.C. pump trip occurs by about 15 seconds. The depressurization is slower in this case compared to the breaks described previously due to very small size of the break resulting in a lower leak flow. ESFAS setpoint limit on low pressure is reached by 46 seconds and the HPI flow begins by 81 seconds aiding depressurization. The system pressure is still above the secondary pressure and subcooled liquid prevails in the steam generator primary side due to the heat removal by the secondary side of the steam generator. By 120 seconds, two phase mixture is realized in the SG primary side slowing the system depressurization. The slow rate of depressurization continues even after the primary side pressure drops below the secondary side pressure at 770 seconds. The steam generator primary side in the broken loop is filled with steam by 910 seconds. The pump suction nodes in the broken loop contain very little liquid mass after 910 seconds hence the rate of depressurization increases due to the large volume of steam now discharging through the break. The HPI flow rate exceeds the boil-off rate due to decay heat by 1250 seconds, thereby establishing long term cooling. The core is always covered with a mixture hence no cladding temperature transient will occur.

In this particular transient, long term cooling is initially established by use of one HPI pump. No LPI or CFT injection took place since the system pressure was above the injection actuation pressures. Thus, it is concluded that for breaks less than or equal to 0.04 ft^2 the HPI alone is capable of matching decay heat boil-off and maintaining a liquid inventory sufficient to preclude any cladding temperature excursions.

5. PHASE DISTRIBUTION MULTIPLIER FOR THE CRAFT2 WILSON SEPARATION MODEL

The phase separation model available in CRAFT2 includes the option of placing a multiplier on the bubble rise velocity calculated by the Wilson model. Although this velicity is based on the average mixture void fraction, the use of a proper multiplier corrects it for expected non-uniformity in phase distribution within the mixture. Because the mixture swelling is closely tied to the bubble separation rate, varying this multiplier effectively permits tuning the mixture height calculations. For small breaks then, this multiplier can be adjusted so that the mixture calculated by CRAFT2 in the core node matches that produced by the more detailed mixture swelling code, FOAM. This, of course, would preclude the necessity of doing a FOAM analysis as long as the mixture is above the core by an amount corresponding to the level uncertainty.

A series of FOAM calculations were made to determine the amount of liquid necessary to just cover the core with a froth using the axial power shape given in BAW-10074.[6] Results were obtained for pressures ranging from 200 to 1200 psia with a core power of 2.13% and 4.26% of 2772 Mwt. The core parameters used were those of 177 FA plants. Steady state CRAFT2 runs were made using several different bubble rise multipliers for each combination of power, pressure and associated FOAM core liquid volume. The equilibrium core mixture heights calculated by CRAFT2 were noted. Figure 5.1 shows a plot of difference in CRAFT2 and FOAM mixture height versus bubble rise multiplier in CRAFT2. A multiplier of 2.38 as seen in Figure 5.1, produces a CRAFT2 mixture level within + 1.8% and - 2% of that calculated by FOAM over the range of parameters most likely to exist when core uncovering is a possibility during a small break accident. Using a tolerance of ± 2%, it is concluded that no FOAM analysis is need if the mixture height exceeds the active core by three inches (0.25 feet). If mixture height drops to less than 0.25 feet above the core, the cladding heatup analysis using FOAM results will be done as outlined in Section 5.2.3 of BAW-10104.[5]

6. REFERENCES

1. C.E. Parks, B.M. Dunn, and R.C. Jones, "Multinode Analysis of Small Breaks for B&W's 2568 Mwt Nuclear Plants", BAW-10052, Rev. 1, Babcock & Wilcox, Lynchburg, Va., October 1975.

2. R.A. Hedrick, J.J. Cudlin and R.C. Foltz, "CRAFT2 - Fortran Program For Digital Simulation of a Multinode Reactor Plant During Loss of Coolant," BAW-10092, Rev. 2, Babcock & Wilcox, April 1975.

3. J.F. Wilson, R.J. Grenda, and J.F. Patterson, "The Velocity of Rising Steam in a Bubbling Two-Phase Mixture," ANS Transactions, 5 (1962).

4. B.M. Dunn, C.D. Morgan, and L.R. Cartin, "Multinode Analysis of Core Flooding Line Break for B&W's 2568 Mwt Internals Vent Valve Plants, BAW-10064, Rev. 1, Babcock & Wilcox, October 1975, (FOAM code is discussed in this reference).

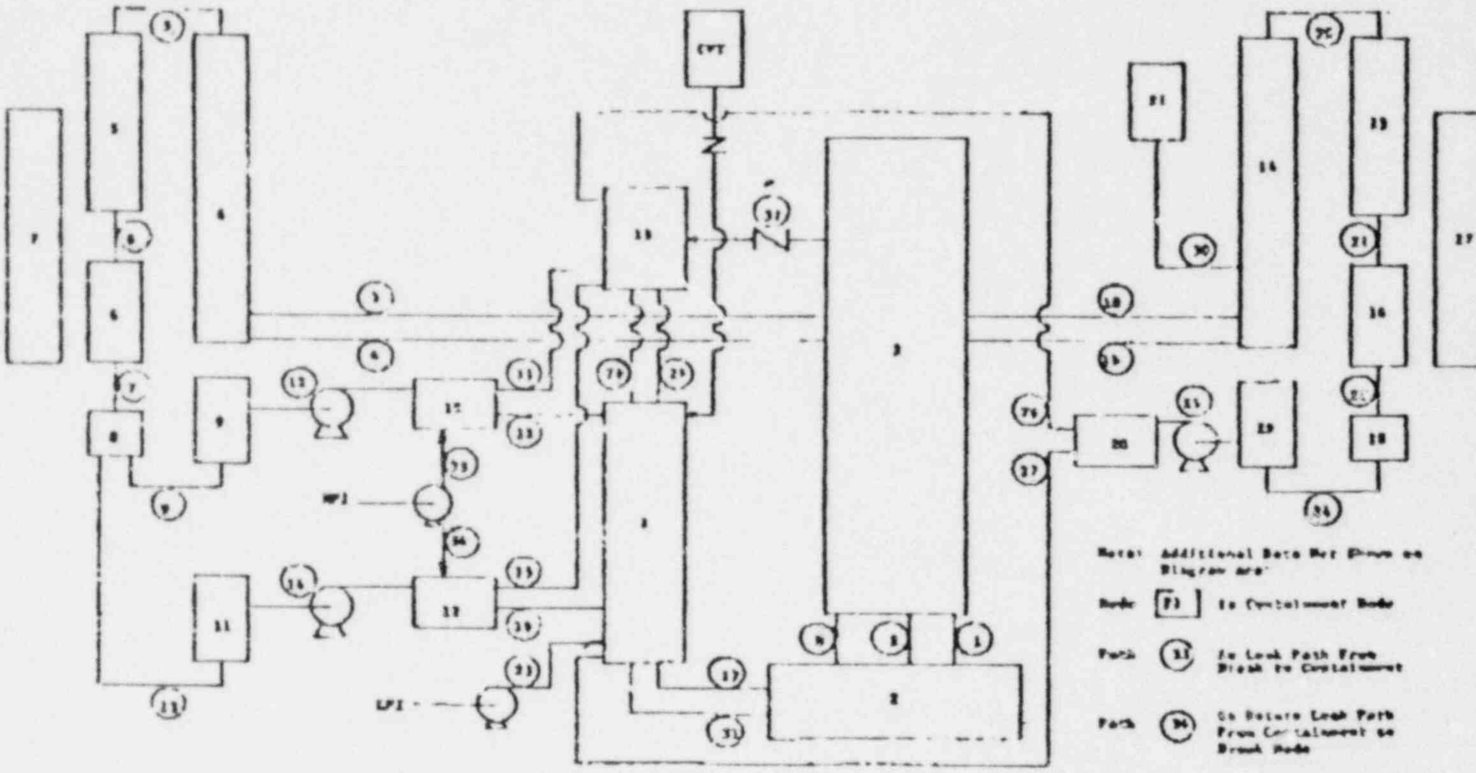
5. B.M. Dunn, R.C. Jones, L.R. Cartin, C.E. Parks, and R.J. Salm,"B&W's ECCS Evaluation Model", BAW-10104A, Rev. 1, Babcock & Wilcox, Lynchburg, Va, March 1976.

6. R.C. Jones, B.M. Dunn, and C.E. Parks, "Multinode Analysis of Small Breaks for B&W's 205 Fuel Assembly Nuclear Plants With Internals Vent Valves", BAW-10074A, Rev. 1, Babcock & Wilcox, Lynchburg, Va., March 1976.

7. R.C. Jones, J.R. Biller, and B.M. Dunn, "ECCS Analysis of B&W's 177-FA Lowered Loop NSS", BAW-10103, Rev. 2, Babcock & Wilcox, Lynchburg, Va., April 1976.

FIGURE 4-1

CRAFT Noding Diagram For Small Break



Note: Additional Data Not Shown on Diagram are:

Node [22] Is Containment Node

Path (33) Is Leak Path From Break to Containment

Path (34) Is Return Leak Path From Containment to Break Node

Path (37) Represents Containment Spray System

Node No.	Identification
1	Downcomer
2	Lower Plenum
3	Core, Core Bypass, Upper Plenum, Upper Head
4,14	Hot Leg Piping
5,15	Steam Generator Upper Head, SG Tubes (Upper Half)
6,16	SG Tubes (Lower Half)
8,18	SG Lower Head
9,11,19	Cold Leg Piping (Pump Suction)
10,12,20	Cold Leg Piping (Pump Discharge)
13	Upper Downcomer (Above the ℄ of Nozzle Belt)
21	Pressurizer
22	Containment

Path No.	Identification
1,2	Core
3,4,18,19	Hot Leg Piping
5,20	Hot Leg, Upper
6,21	SG Tubes
7,22	SG Lower Head
8	Core Bypass
9,13,24	Cold Leg Piping
10,14,25	Pumps
11,12,15,16,26,27	Cold Leg Piping
17,31	Downcomer
23	LPI
28,29	Upper Downcomer
30	Pressurizer
32	Vent Valve
33,34	Leak & Return Path
35,36	HPI
37	Containment Sprays

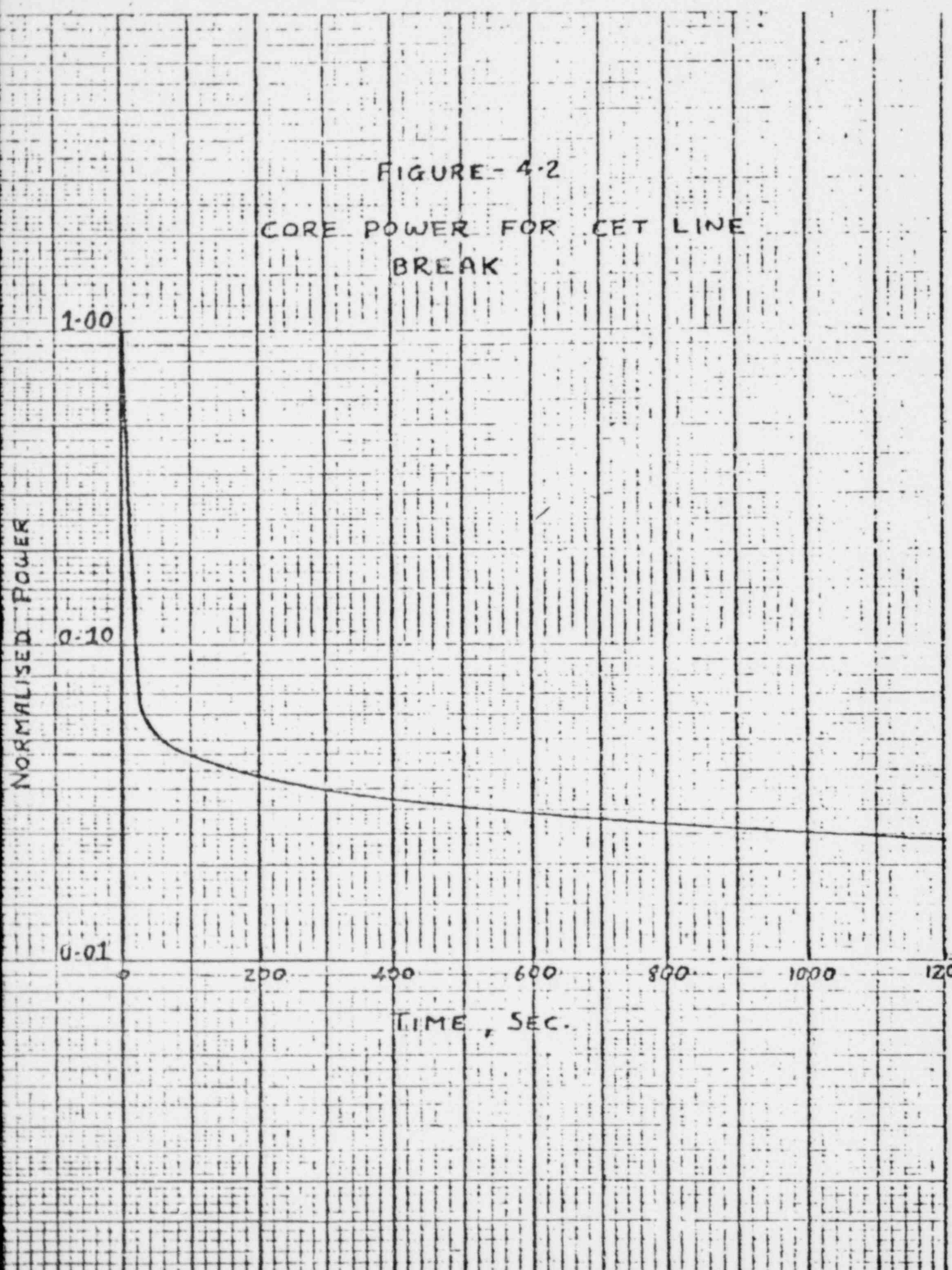


FIGURE- 4·2
CORE POWER FOR CET LINE BREAK
1·00
0·10
0·01
NORMALISED POWER
0
200
400
600
800
1000
1200
TIME, SEC.

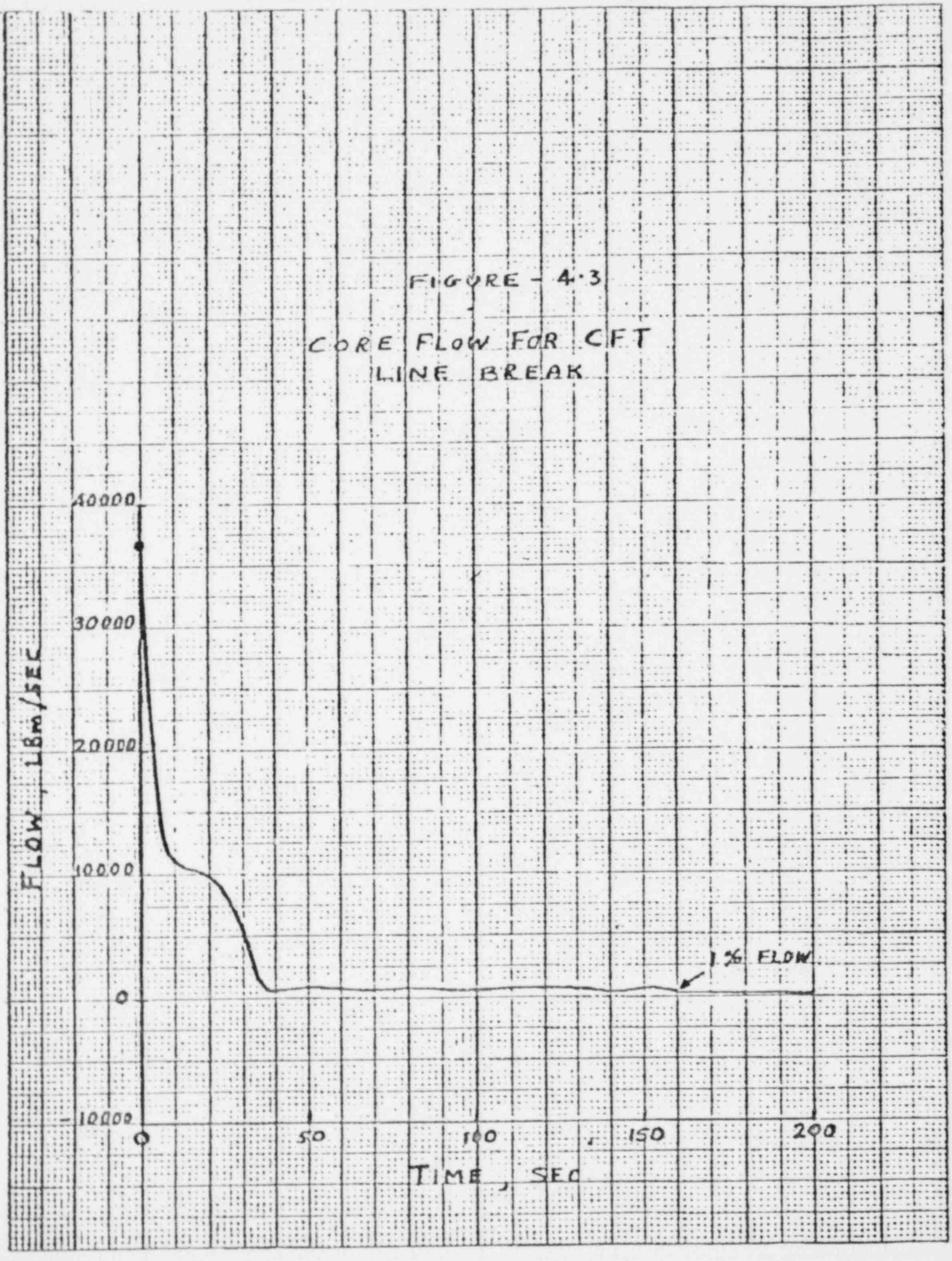


FIGURE - 4.3
CORE FLOW FOR CFT
LINE BREAK
40000
30000
20000
10000
0
-10000
FLOW, LBm/SEC
0
50
100
150
200
TIME, SEC
1% FLOW

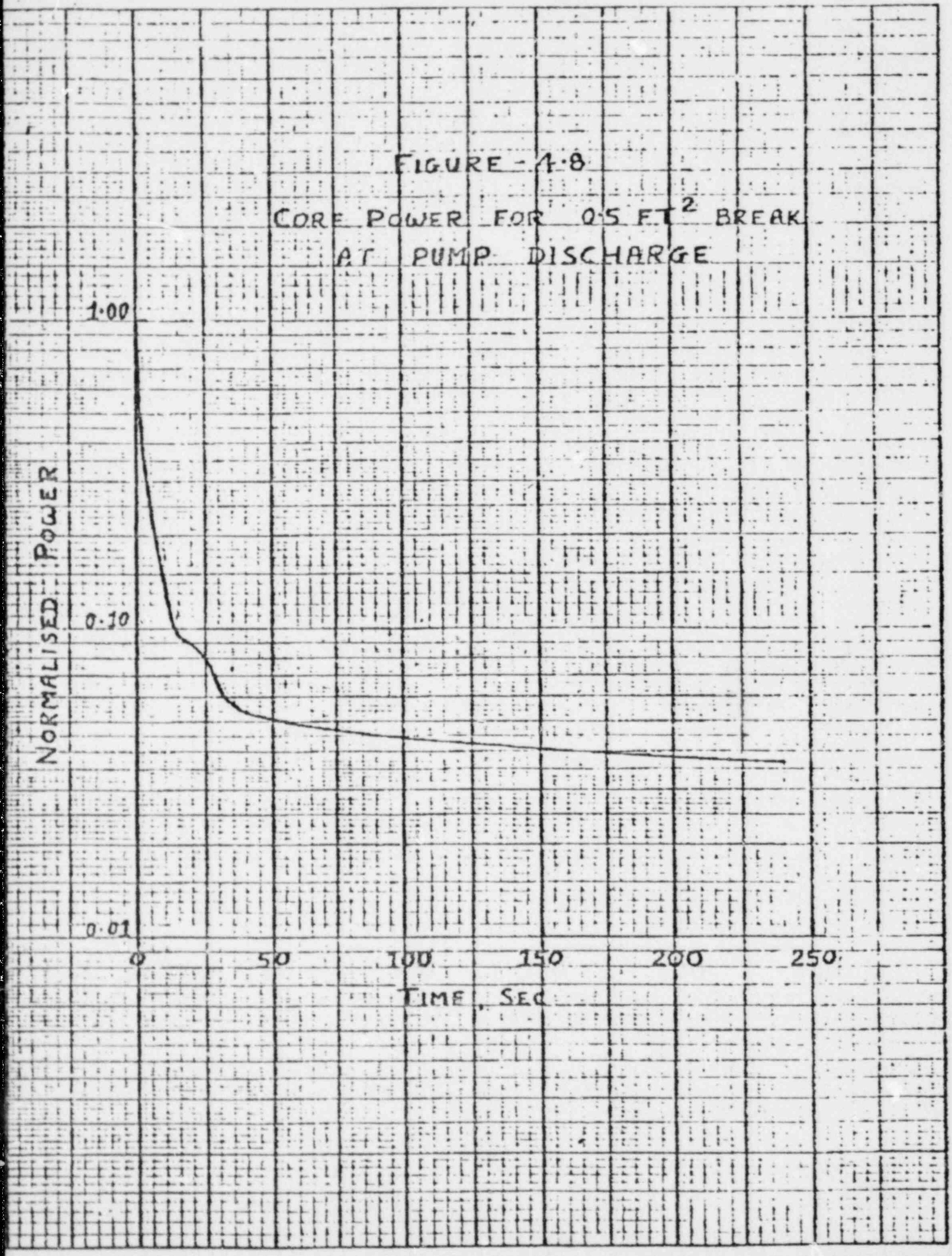


FIGURE - 1·8
CORE POWER FOR 0.5 FT^2 BREAK
AT PUMP DISCHARGE
1.00
0.10
0.01
NORMALISED POWER
0
50
100
150
200
250
TIME, SEC

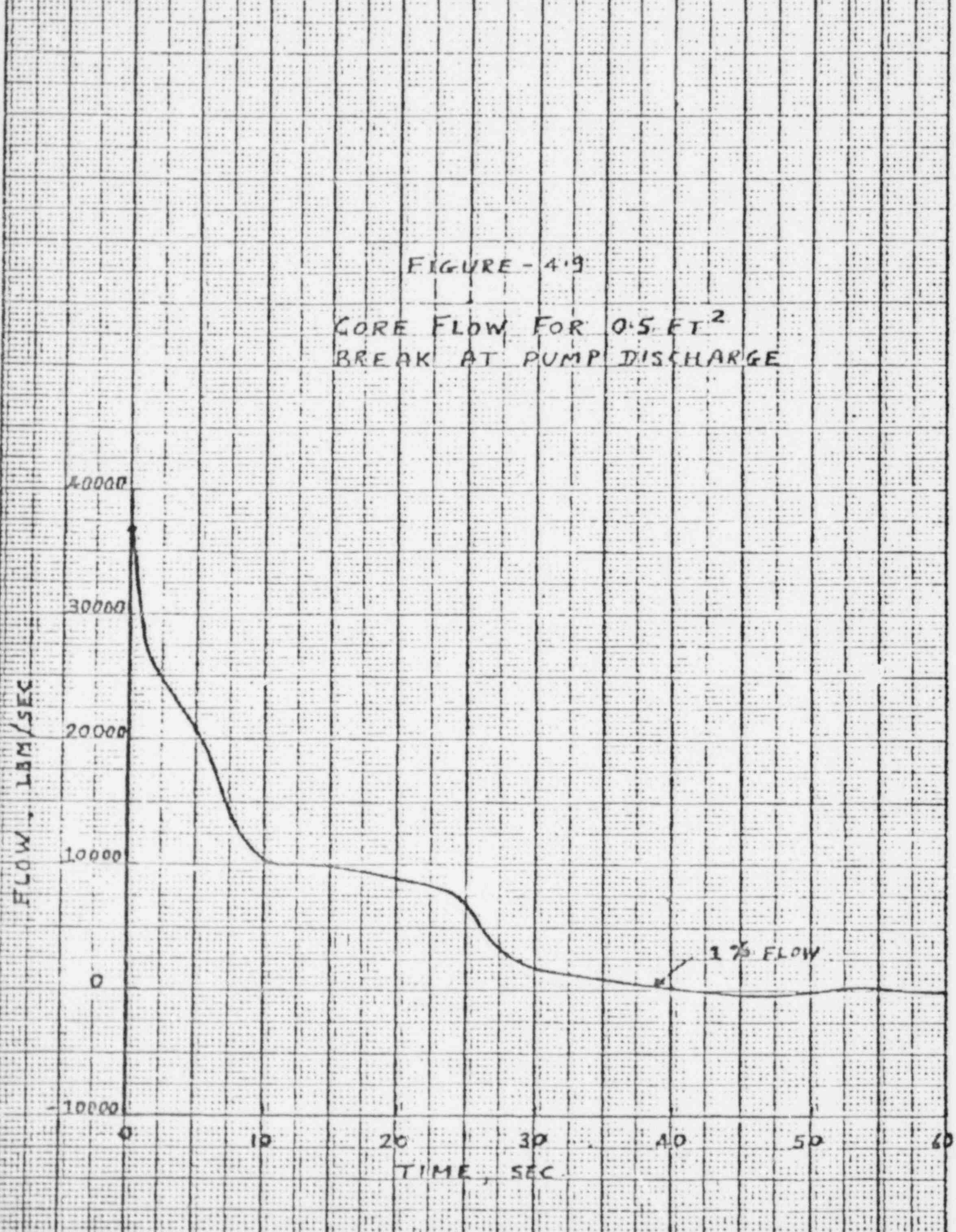


FIGURE - 4.9
CORE FLOW FOR 0.5 FT^2
BREAK AT PUMP DISCHARGE
40000
30000
20000
10000
0
-10000
FLOW, LBM/SEC
0
10
20
30
40
50
60
TIME, SEC
1% FLOW

FIGURE - 4.14

CORE POWER FOR 0.04 FT^2 BREAK AT PUMP SUCTION

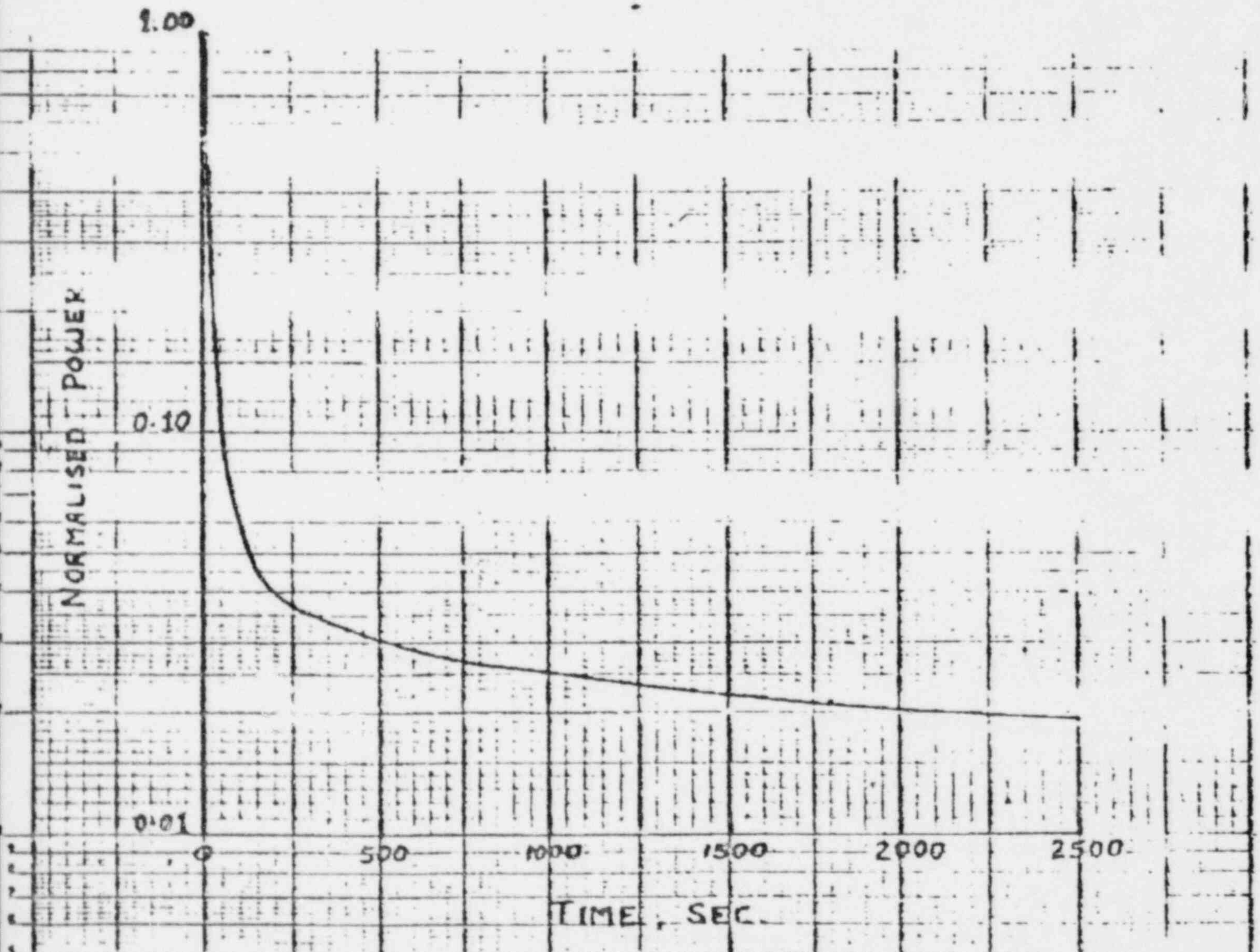


FIGURE 4.15

CORE FLOW FOR 0.04 FT^2 BREAK AT PUMP SUCTION

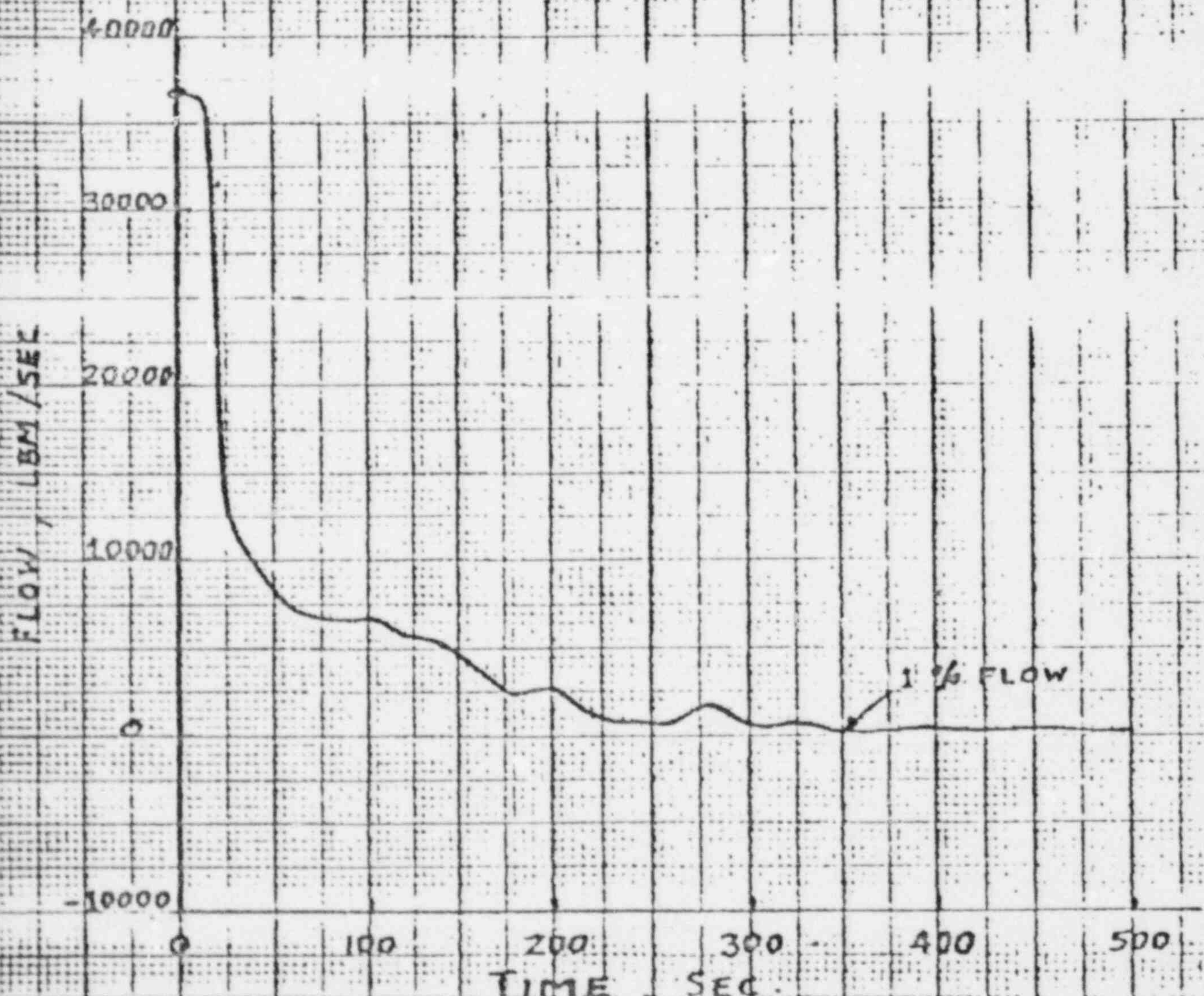
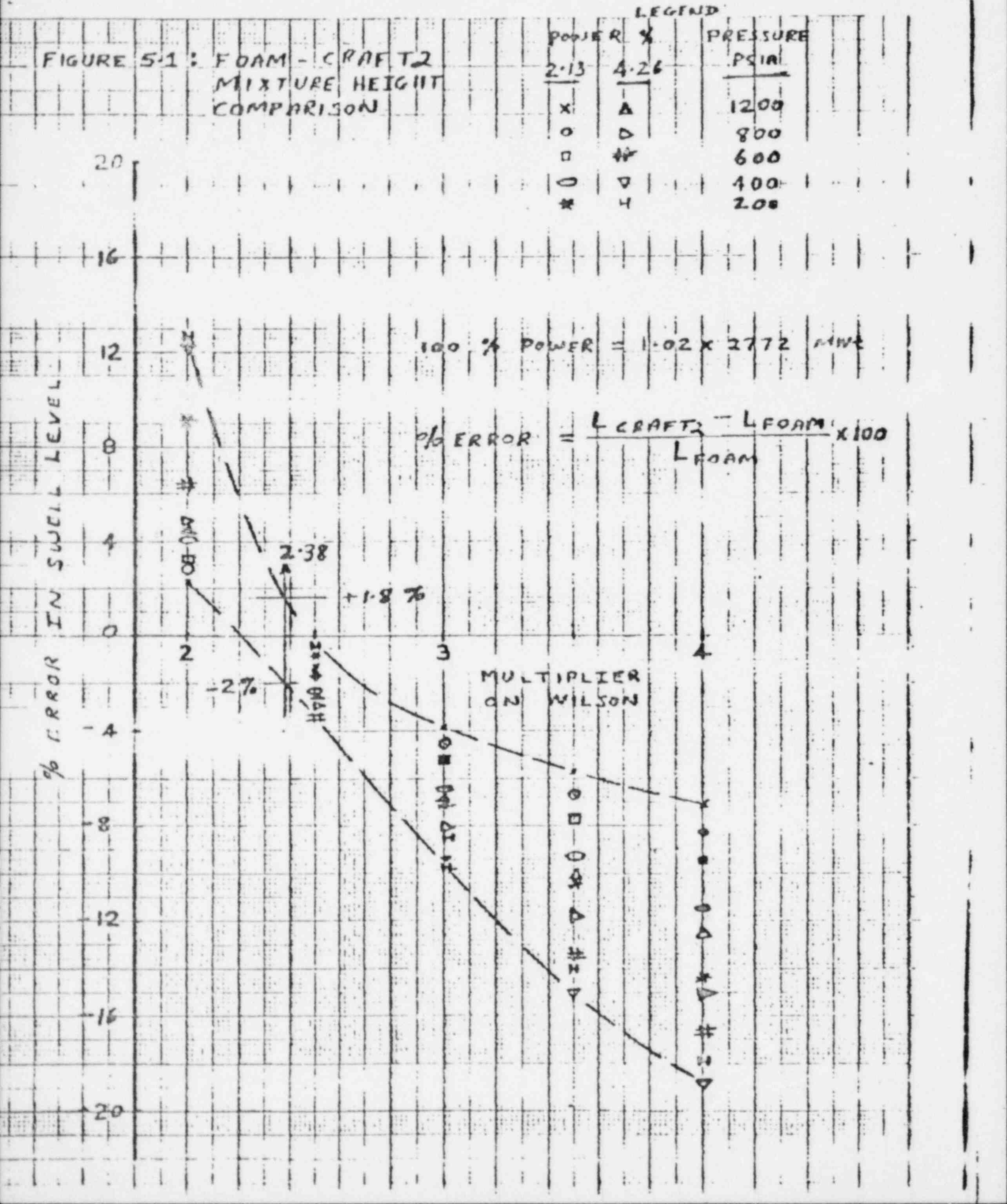


FIGURE 5-1: FOAM - CRAFT2 MIXTURE HEIGHT COMPARISON



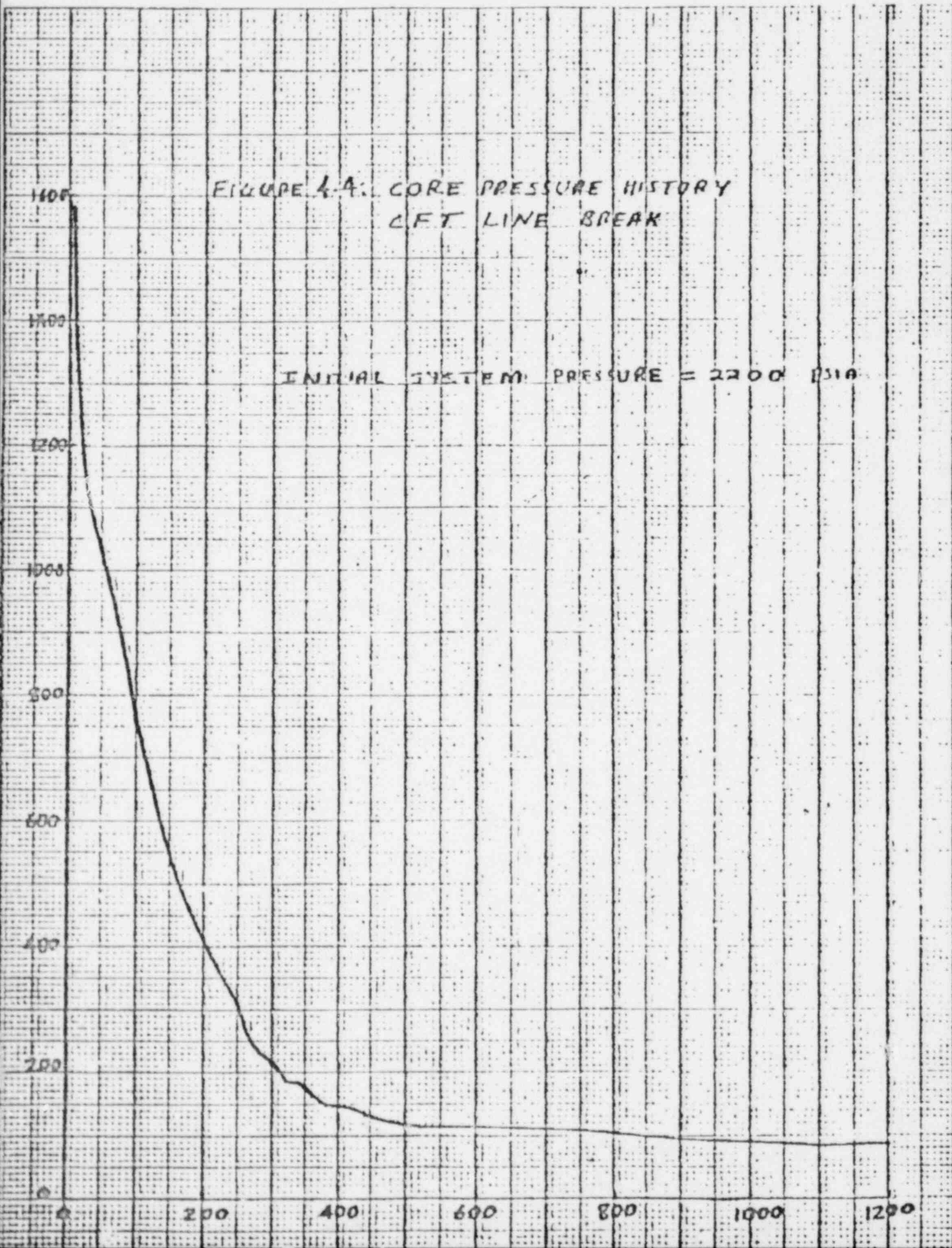


FIGURE 4.4. CORE PRESSURE HISTORY
CFT LINE BREAK
INITIAL SYSTEM PRESSURE = 2200 PSIA
1400
1200
1000
800
600
400
200
0
200
400
600
800
1000
1200

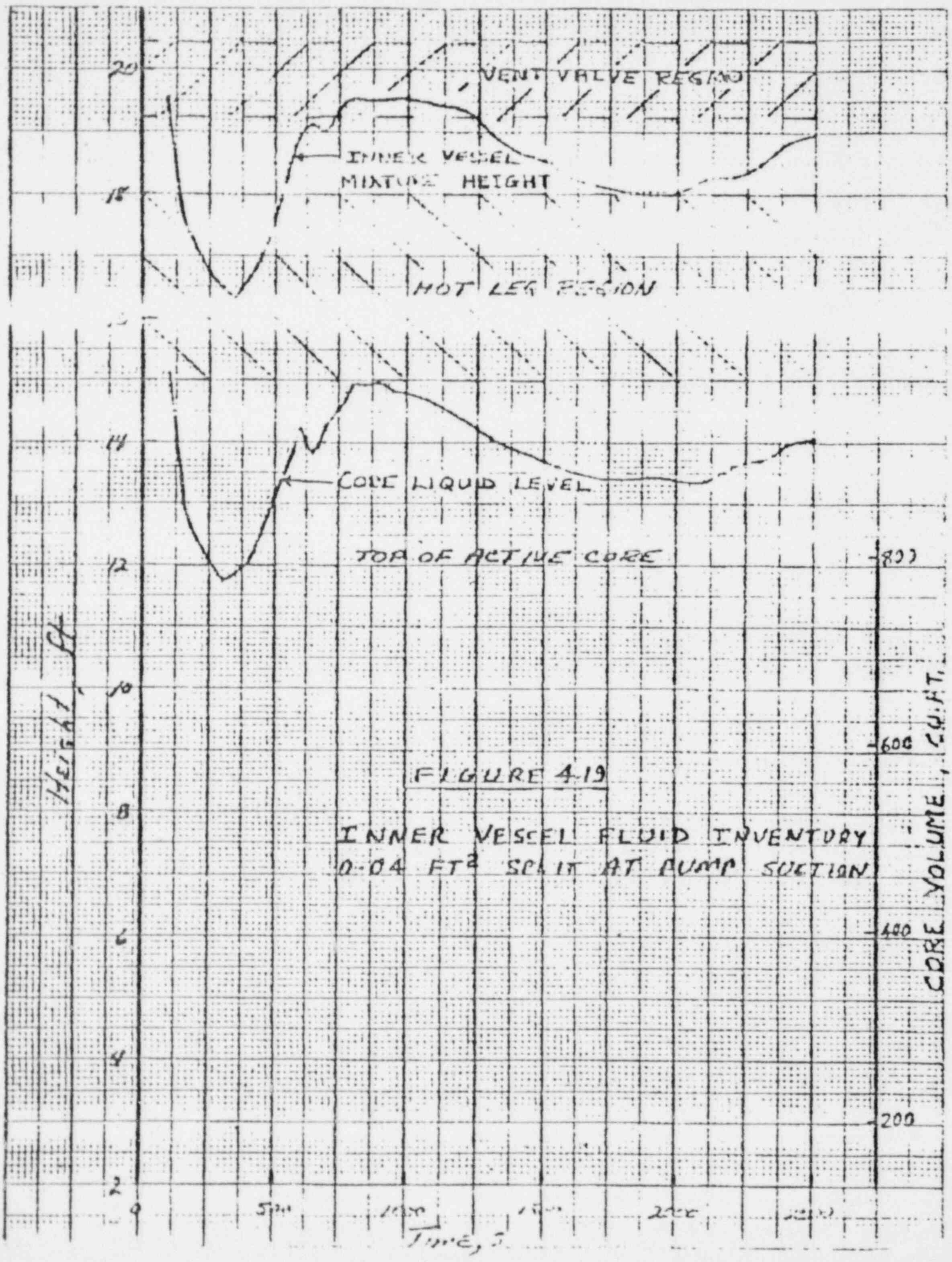


FIGURE 4.19

INNER VESSEL FLUID INVENTORY
0.04 FT² SPLIT AT PUMP SUCTION

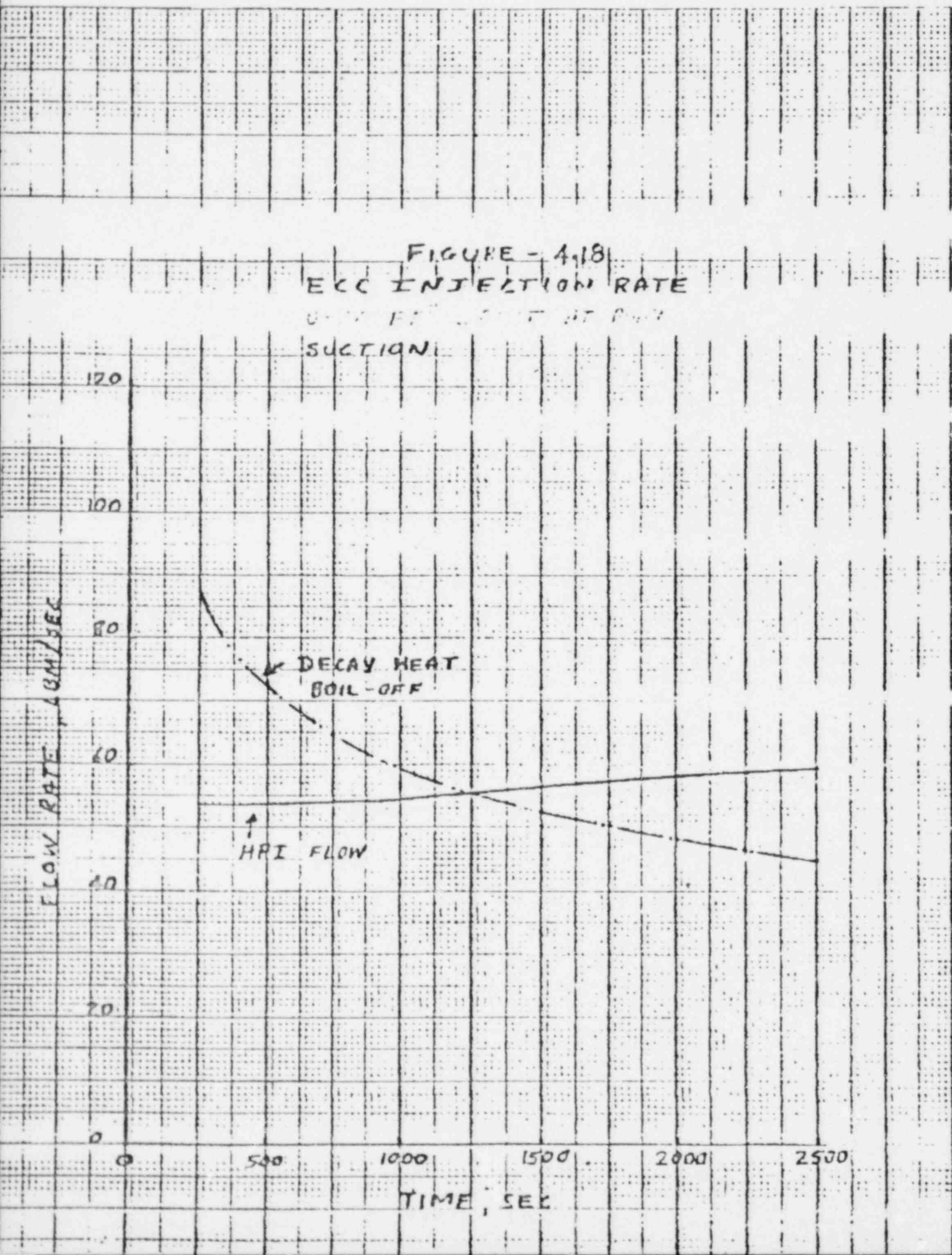


FIGURE - 4.18
ECC INJECTION RATE
SUCTION

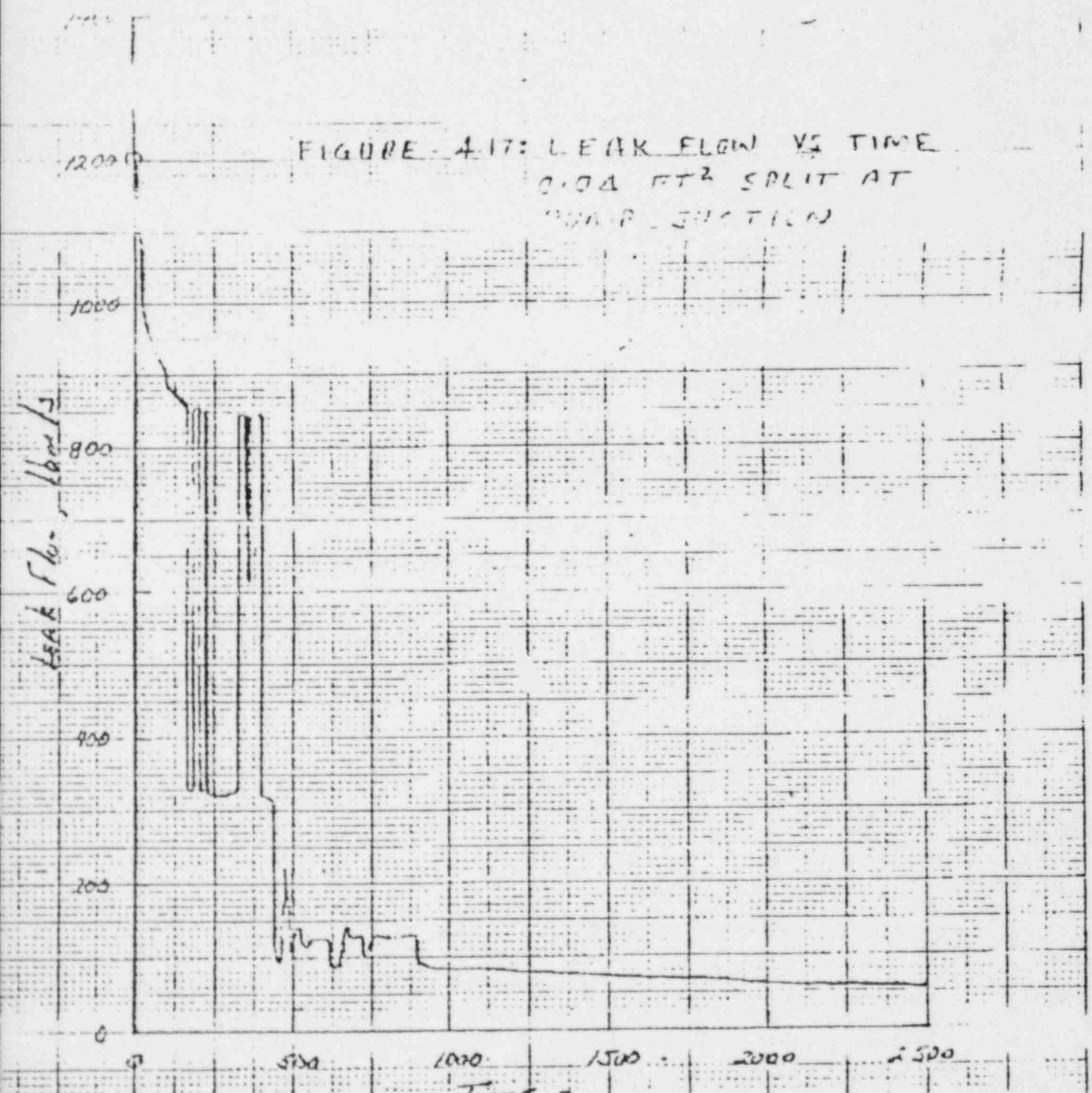
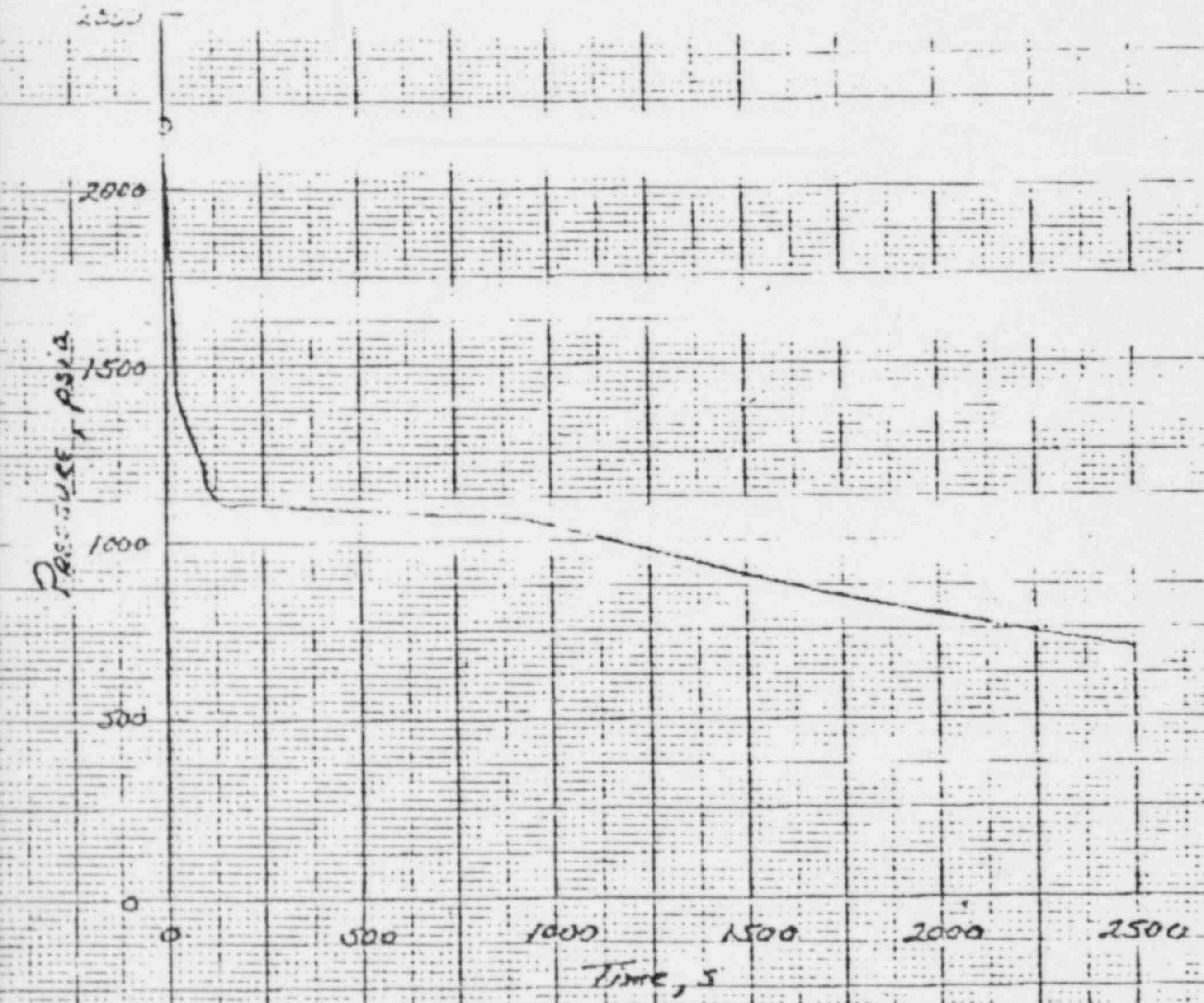


FIGURE 4.17: LEAK FLOW VS TIME
0.04 FT² SPLIT AT
PUMP SUCTION

FIGURE 4.16: CORE PRESSURE HISTORY

PUMP SUCTION



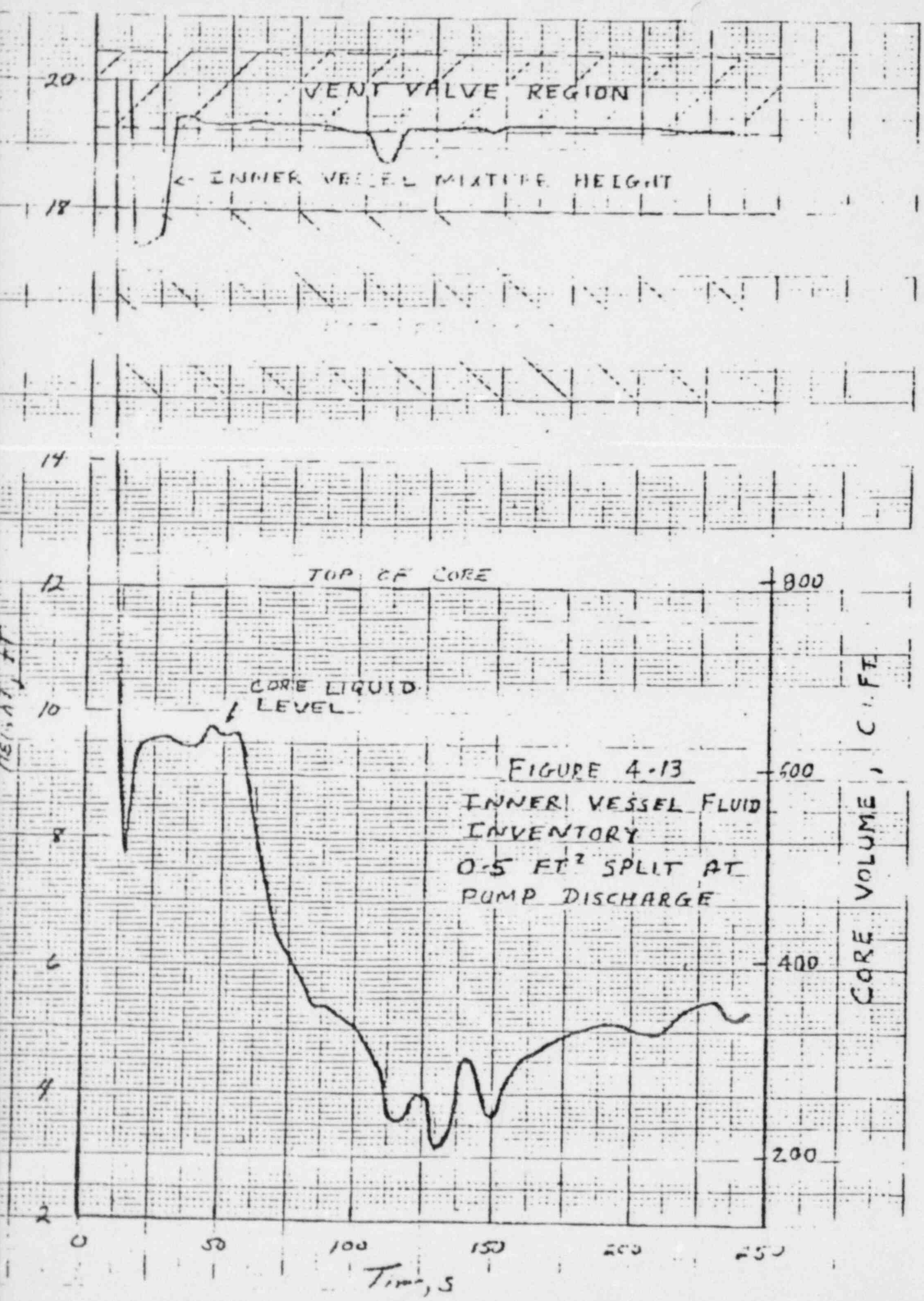


FIGURE 4-13
INNER VESSEL FLUID INVENTORY
0.5 FT² SPLIT AT PUMP DISCHARGE

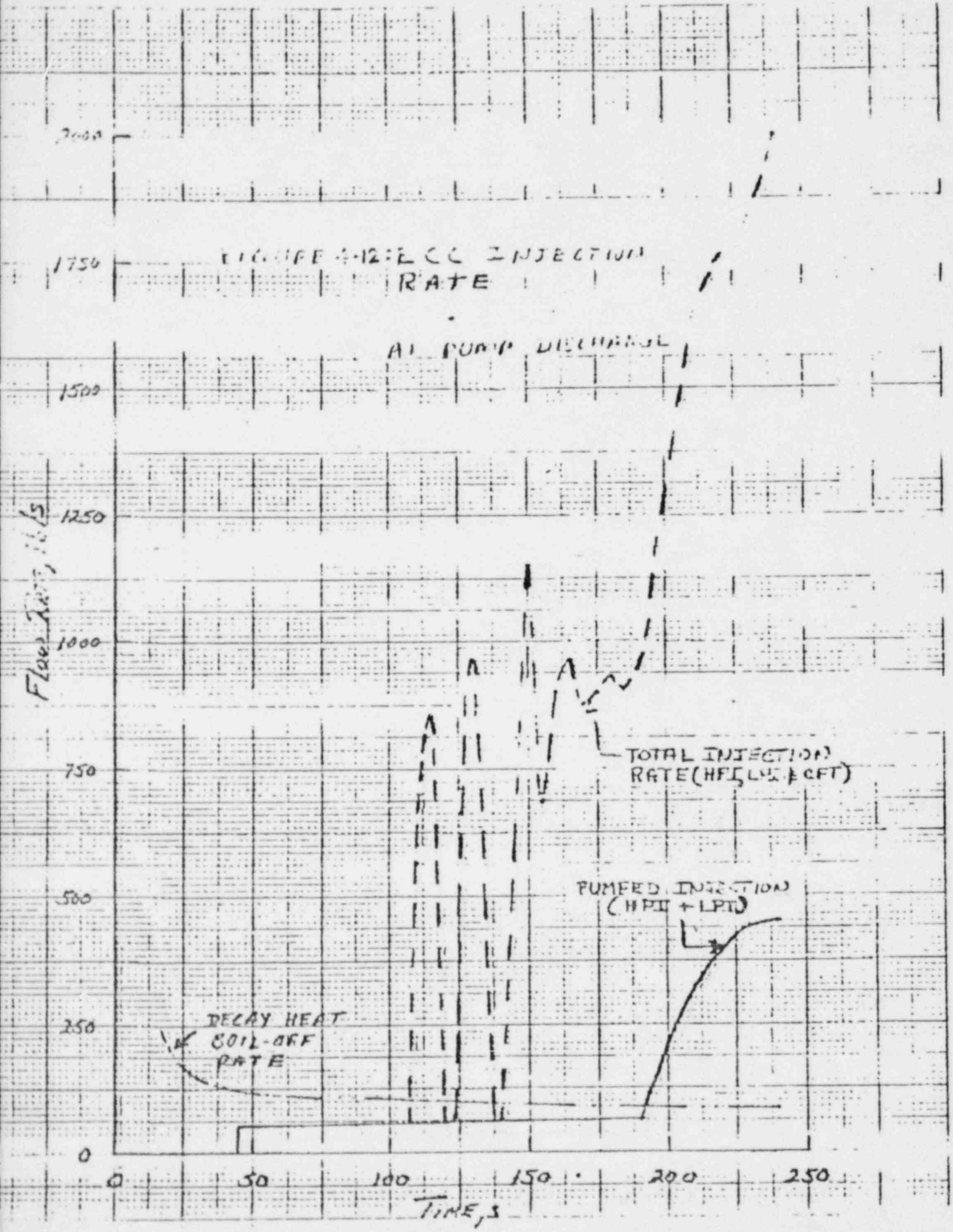


FIGURE 4-12: ECC INJECTION RATE
AT PUMP DISCHARGE
2000
1750
1500
1250
1000
750
500
250
0
0
50
100
150
200
250
TIME, s
TOTAL INJECTION RATE (HPI, LPI & CFT)
PUMPED INJECTION (HPI + LPI)
DECAY HEAT BOIL-OFF RATE

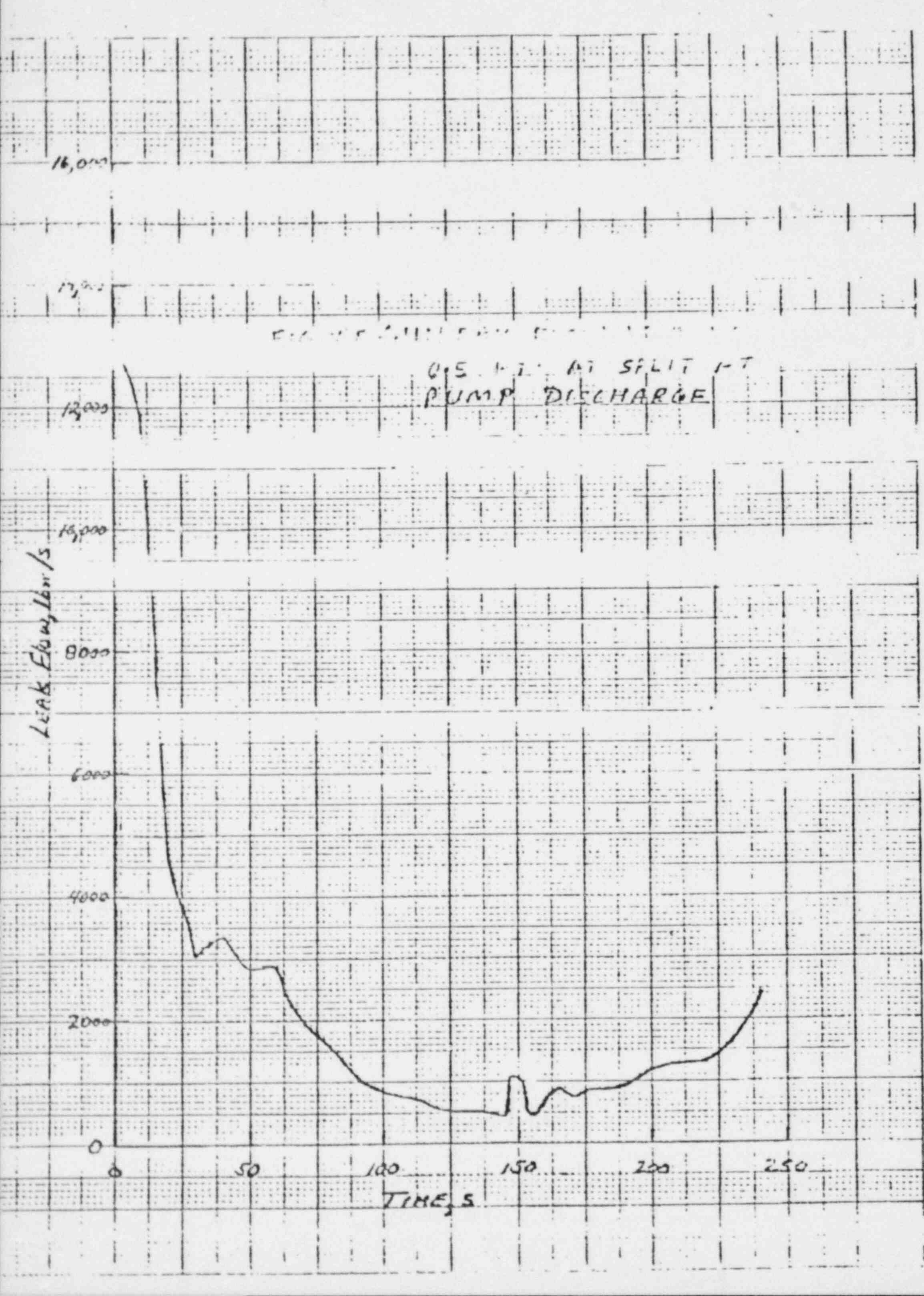
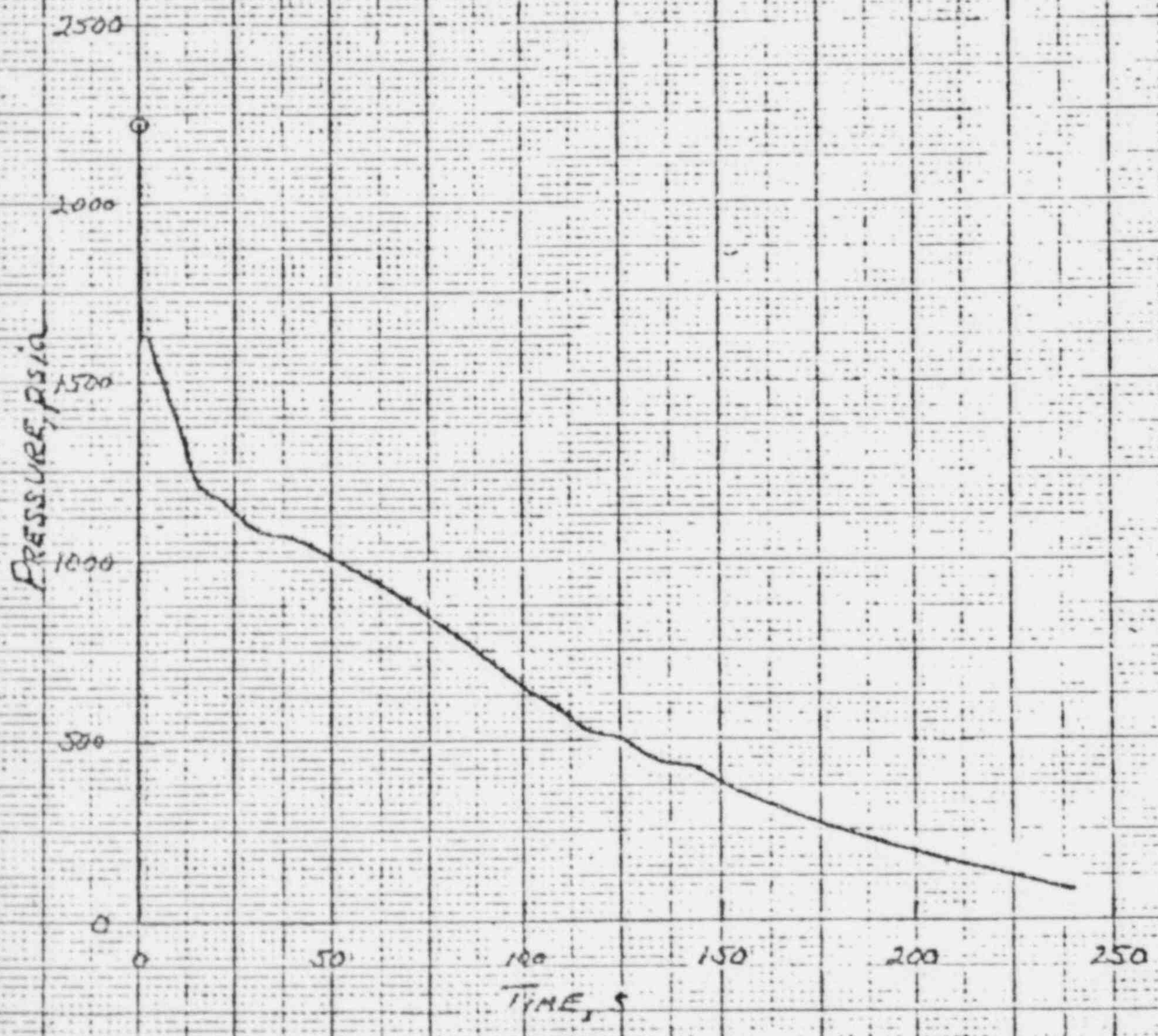

16,000
12,000
10,000
8000
6000
4000
2000
0
PUMP DISCHARGE
0
50
100
150
200
250
TIME, S

FIGURE 4.10: CORE PRESSURE HISTORY
0.5 FT^2 SPLIT AT PUMP DISCHARGE



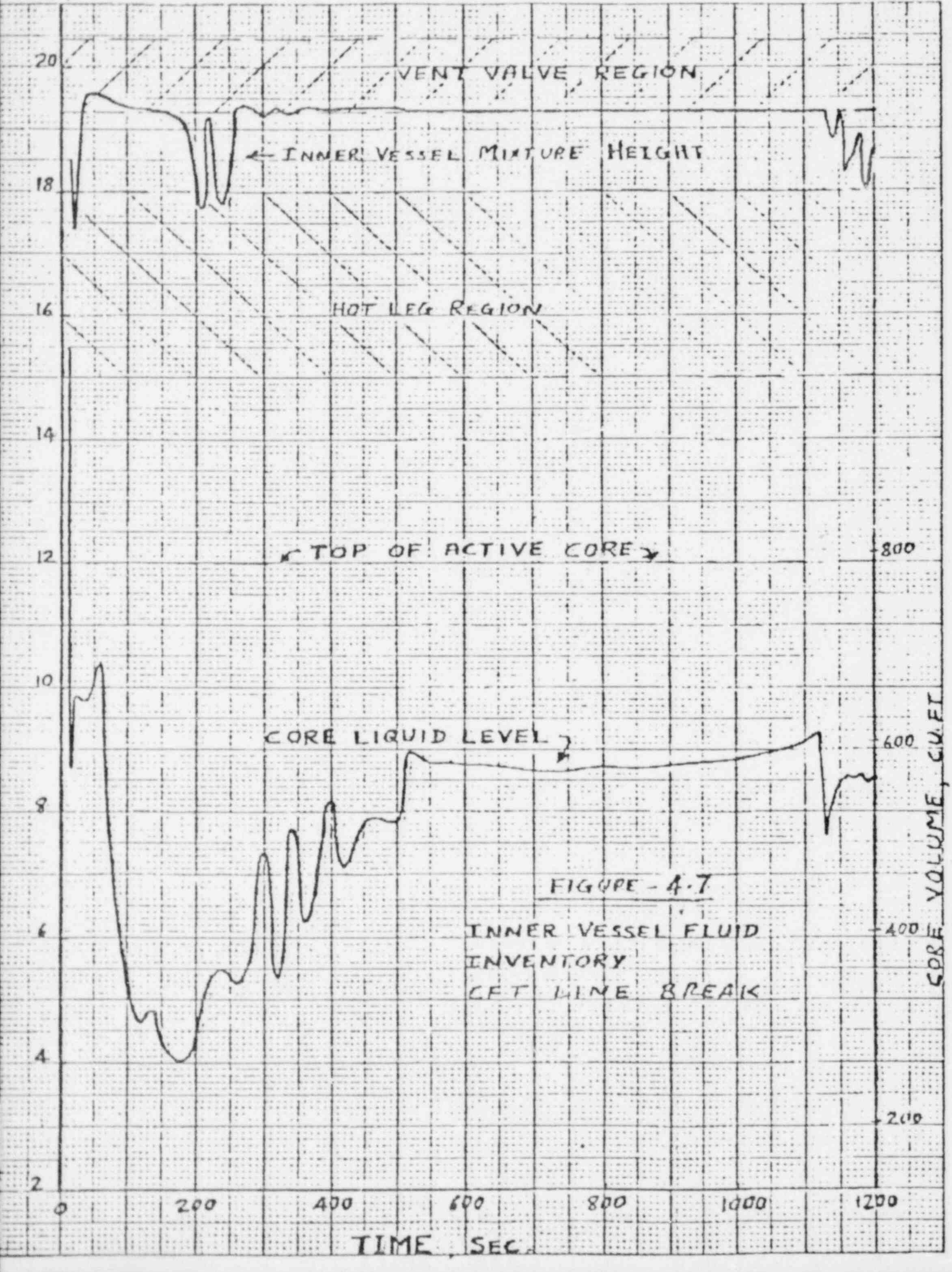


FIGURE - 4.7

INNER VESSEL FLUID INVENTORY

CFT LINE BREAK

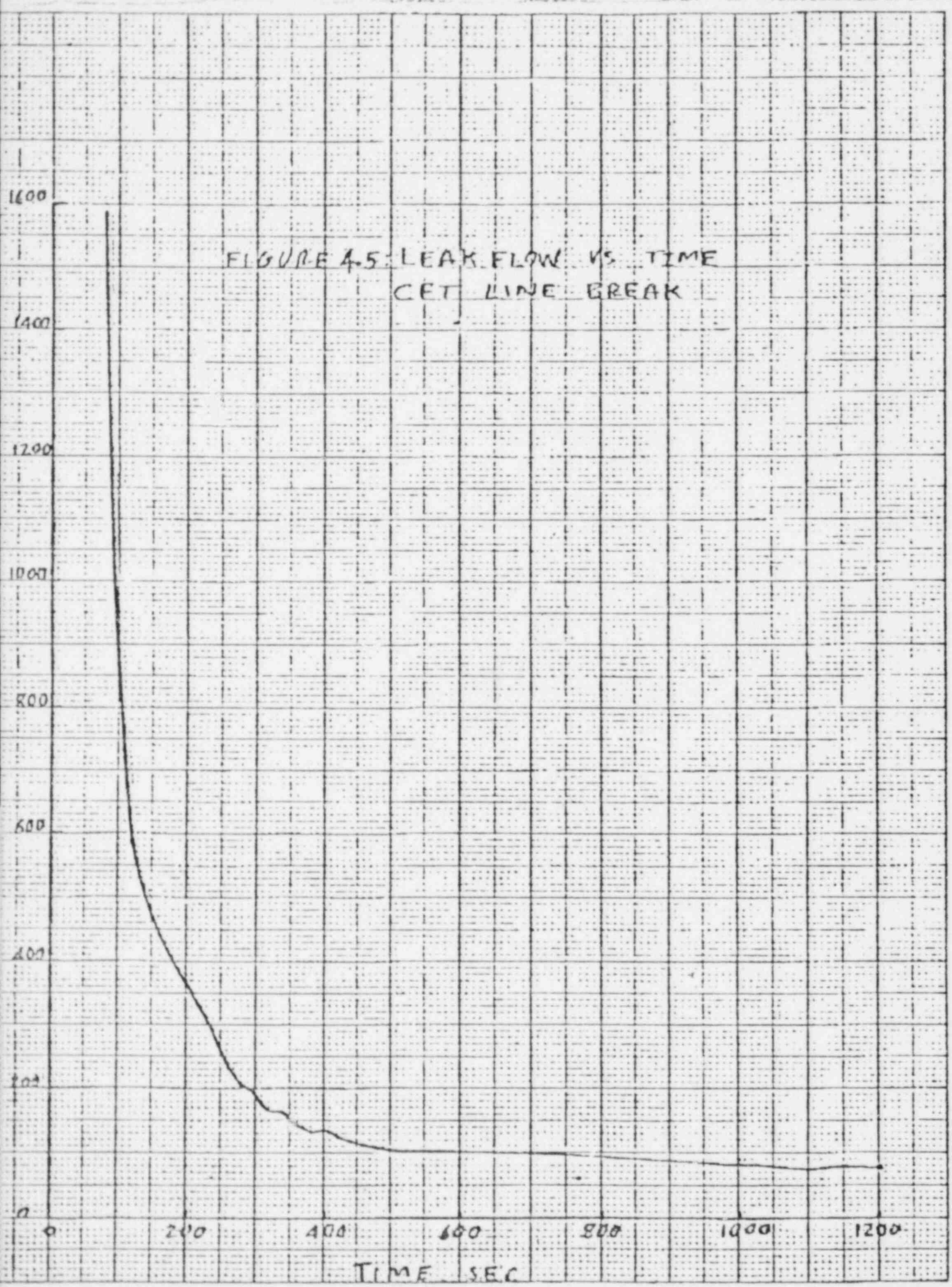


FIGURE 4.5 LEAK FLOW VS TIME
CET LINE BREAK
1600
1400
1200
1000
800
600
400
200
0
0
200
400
600
800
1000
1200
TIME SEC

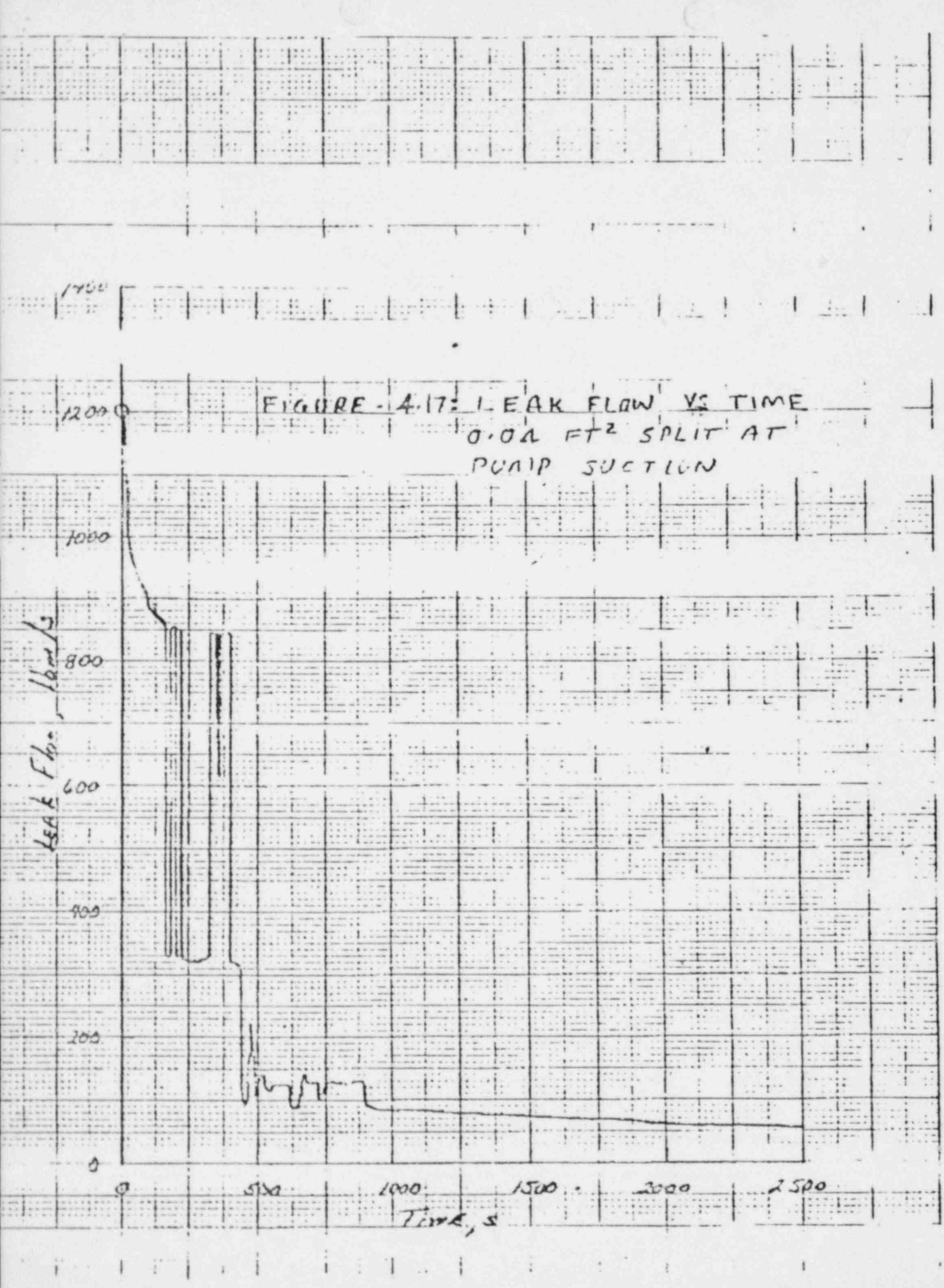


FIGURE - 4.17: LEAK FLOW VS TIME
0.02 FT² SPLIT AT
PUMP SUCTION
1400
1200
1000
800
600
400
200
0
Leak Flow, lbm/s
0
500
1000
1500
2000
2500
Time, s

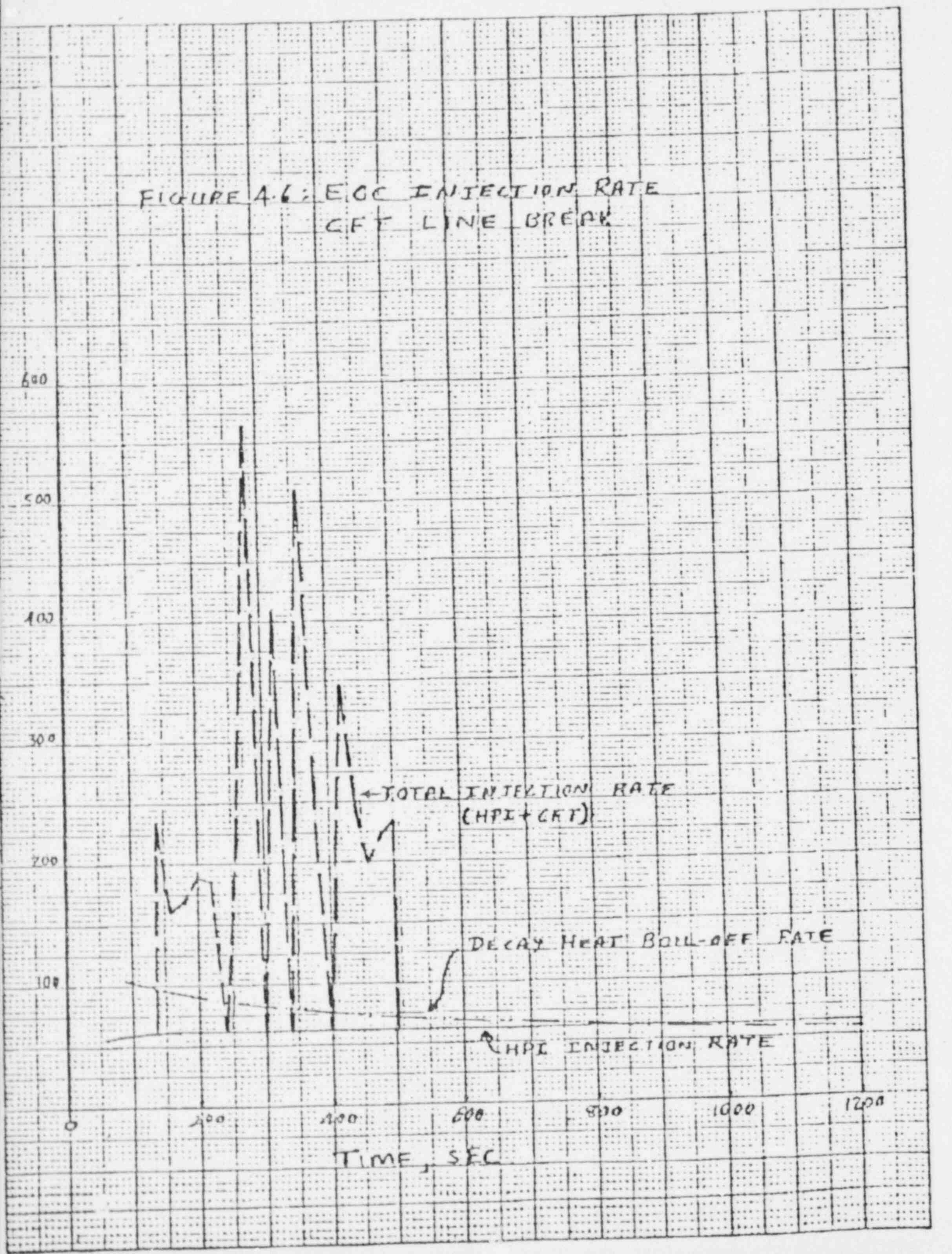


FIGURE 4.6: ECC INJECTION RATE
CFT LINE BREAK
600
500
400
300
200
100
0
200
400
600
800
1000
1200
TIME, SEC
←TOTAL INJECTION RATE
(HPI + CFT)
DECAY HEAT BOIL-OFF RATE
HPI INJECTION RATE

LONGHAND MEMORANDUM THE BABCOCK & WILCOX COMPANY

TO G.R. Mazetis - NRC

FROM Henry Bailey - B&W ext 2678

CUST.

FILE NO. OR REF.

RCVD part 11/15/76 & part 11/17/76

SUBJ. Small Break Analysis - BAW-10103

DATE 11/11/76

CRITICAL PATH ON CR-3

cc Zeltam R. 11/19/76

ICJP 11/17/76

SMALL BREAK ANALYSIS OF 177 FA LOWERED LOOP PLANTS

The CRAFT model used in these analyses is shown in Figure 1. The model is designed specifically to handle the small break LOCA. Control volumes in and around the vessel are all connected by a pair of flow paths to allow the occurrence of countercurrent flow. Phase separation within each control volume is computed using the Wilson, Grenda, and Patterson average bubble terminal velocity correlation. Within the core region, however, a multiplier of 2.38 is applied to the calculated bubble rise velocity. A comparison analysis has shown that a multiplier of 2.38 in CRAFT gives a mixture height within ± 2% of that predicted by FOAM. Thus no FOAM analysis will be needed if the CRAFT mixture level remains above the core by 2% of the active length.

where (4) (5) (6)

Calculations were performed for the following breaks:

1. CFT Line Break (0.44 ft^2): This case results in minimum ECC system available to mitigate this relatively large, small LOCA.
2. 0.5 ft^2 Split Type Break At Pump Discharge: This case was chosen as it represents the transition break between the small and the large break models.
3. 0.66 ft^2 Split Type Break At Pump Suction: This case was reported as the most limiting small break in BAW-10052.

The system response in each case have shown that during the initial period when the transient is flow controlled, sufficient flow is maintained such that CHF does not occur, and nucleate boiling heat transfer predominates.

Since the core remains covered by the mixture for the remainder of the transient, pool film boiling will be maintained. This heat transfer mechanism is sufficient to keep the cladding temperature within a few degrees of the fluid saturation temperature. Therefore, the transient clad temperature will never exceed its initial value, no metal water reaction will occur, and the core geometry will remain coolable as no cladding rupture will occur. Long term cooling is also established as the HPI injection rate is shown to eventually exceed the boil off due to core decay heat. Thus, the acceptance criteria of 10CFR50.46 are met.

A brief description of each case is now described below:

1. CFT Line Break:

The break is assumed to be at the CFT line nozzle joining the RV and is limited in area to 0.44 ft^2 by the nozzle insert in the CFT line. Node 15 in Figure 1 is the break node, and the analysis takes credit for one CFT and one HPI pump. 7

Figure 2 shows the system pressure history during the accident. Figure 3 shows leak rate thru the break, and Figure 4 is a plot of HPI and total ECC water flow. Long term cooling is assured in that the boiloff due to core decay heat is exceeded by the HPI injection rate by 850 seconds. Figure 4 is a plot of core liquid inventory and mixture height. It shows that while much of the core liquid inventory is depleted, the mixture level predicted by CRAFT remains at a level where it is able to spill into the hot legs and, for most of the time, through the vent valves. System oscillations are observed after [illegible] seconds resulting in a decrease in core liquid volume and vessel mixture height. These temporal perturbations will soon be overcome as the b[illegible]ff rate is already exceeded by the injection rate and theref[illegible] will result in an increase in the core liquid volume. No clad temperature transient will occur since the core is always covered with a mixture and the HPI injection has exceeded the boil off assuring a long term cooling.

2. 0.05 ft^2 Split At Pump Discharge:

The break is assumed to be at the bottom of node 10. The analysis takes credit for two CFT, one HPI pump and one LPI pump.

The system pressure, leak rate, ECC water flow and core fluid inventory history are shown in Figures 6,7,8 and 9 respectively. The core is always covered by a mixture and the boil off due to decay heat is exceeded by the HPI & LPI injection by 195 seconds thus establishing long term cooling.

3. 0.04 ft^2 Split At Pump Suction:

The break is assumed to be at the bottom of node 9. The analysis takes credit for two CFTs, one HPI and one LPI pump.

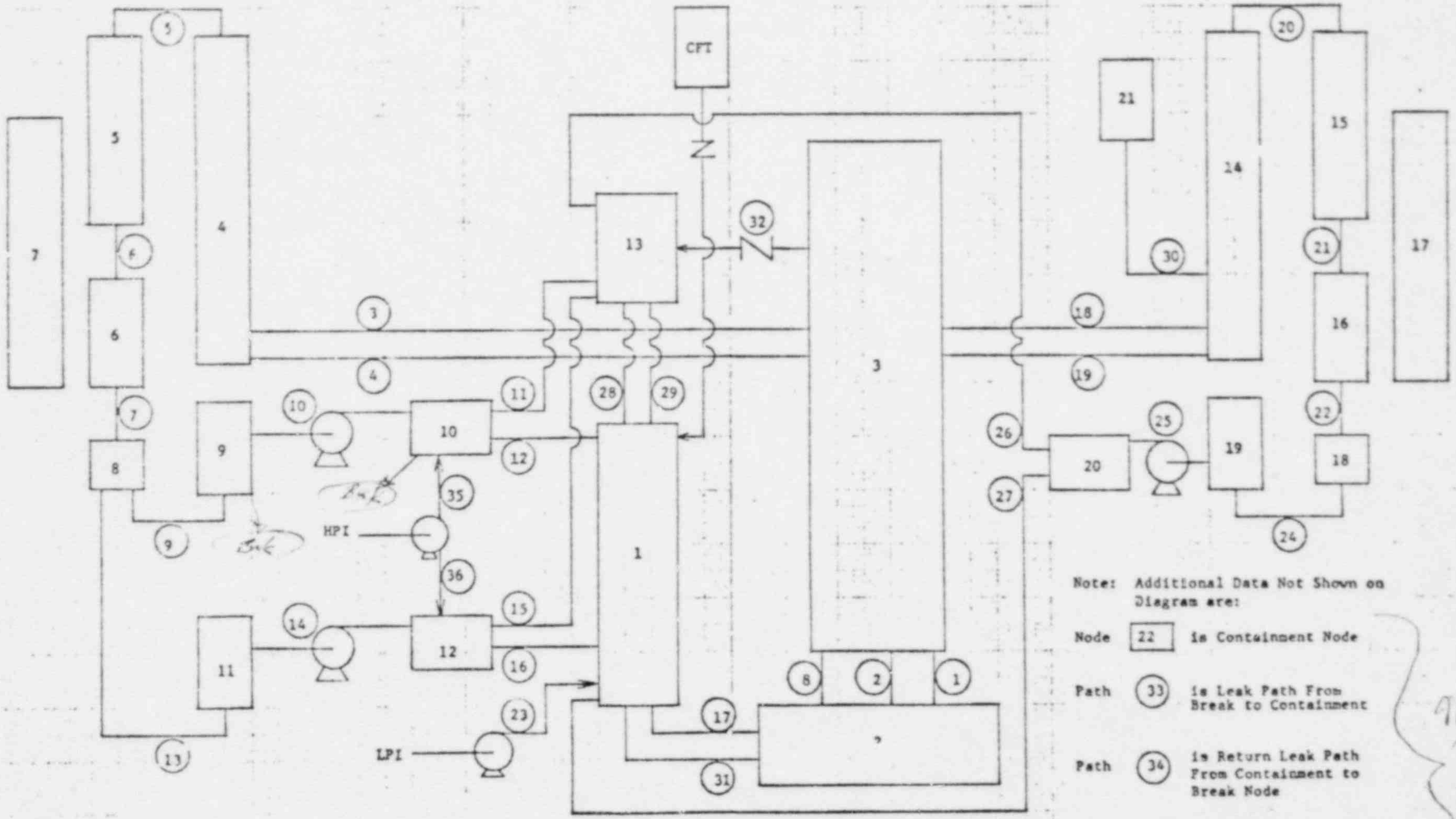
The Figure 10,11,12 & 13 shows system pressure, leak rate, core fluid inventory history and HPI water flow rates respectively. The core is always covered by a mixture and the boil off due to decay heat is exceeded by the HPI flow rate by 1250 seconds, indicating an establishment of long term cooling.

In this particular transient, long term cooling is initially established by use of one HPI pump alone. No LPI and/or CFT injection took place since the system pressure was above the injection actuation pressures. Thus for breaks less than or equal to 0.04 ft^2 the HPI system alone is capable of matching decay heat boil off and maintaining a suitable liquid inventory to preclude any cladding temperature excursions.

B+W Small Break Analysis (BAW-10103)

FIGURE-1

CRAFT Noding Diagram For Small Br-.k



Note: Additional Data Not Shown on Diagram are:

Node 22 is Containment Node

Path 33 is Leak Path From Break to Containment

Path 34 is Return Leak Path From Containment to Break Node

Path 37 Represents Containment Spray System

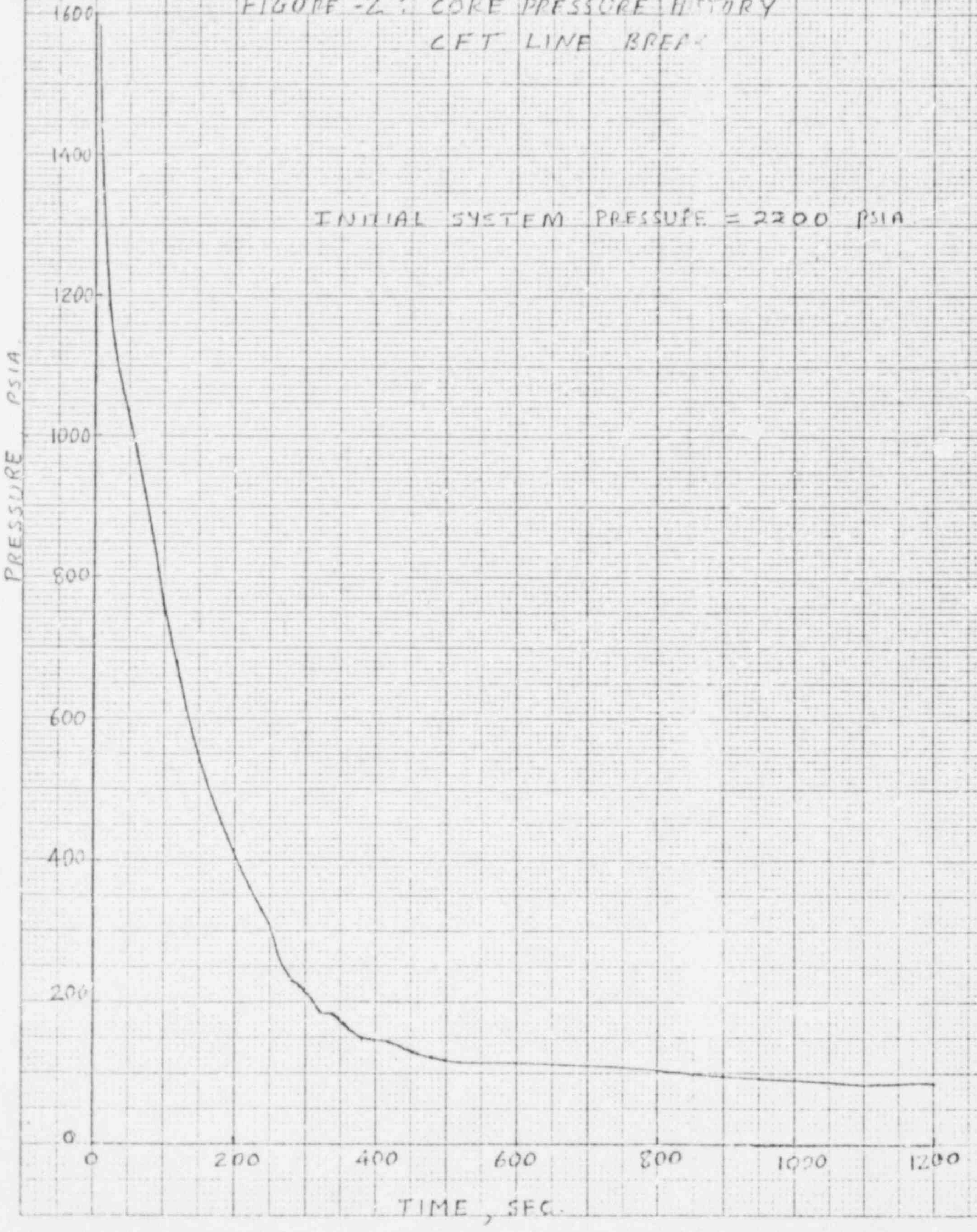
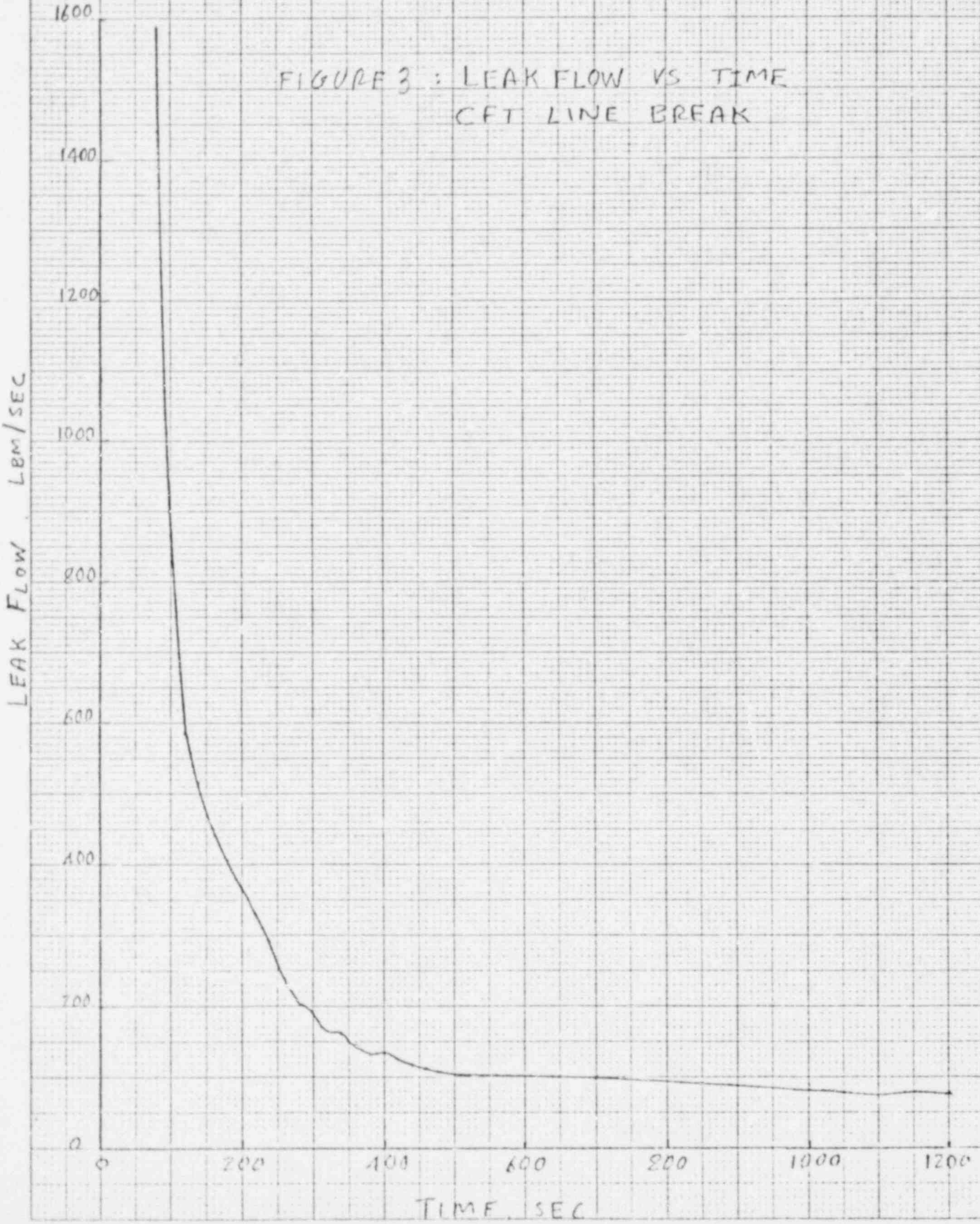
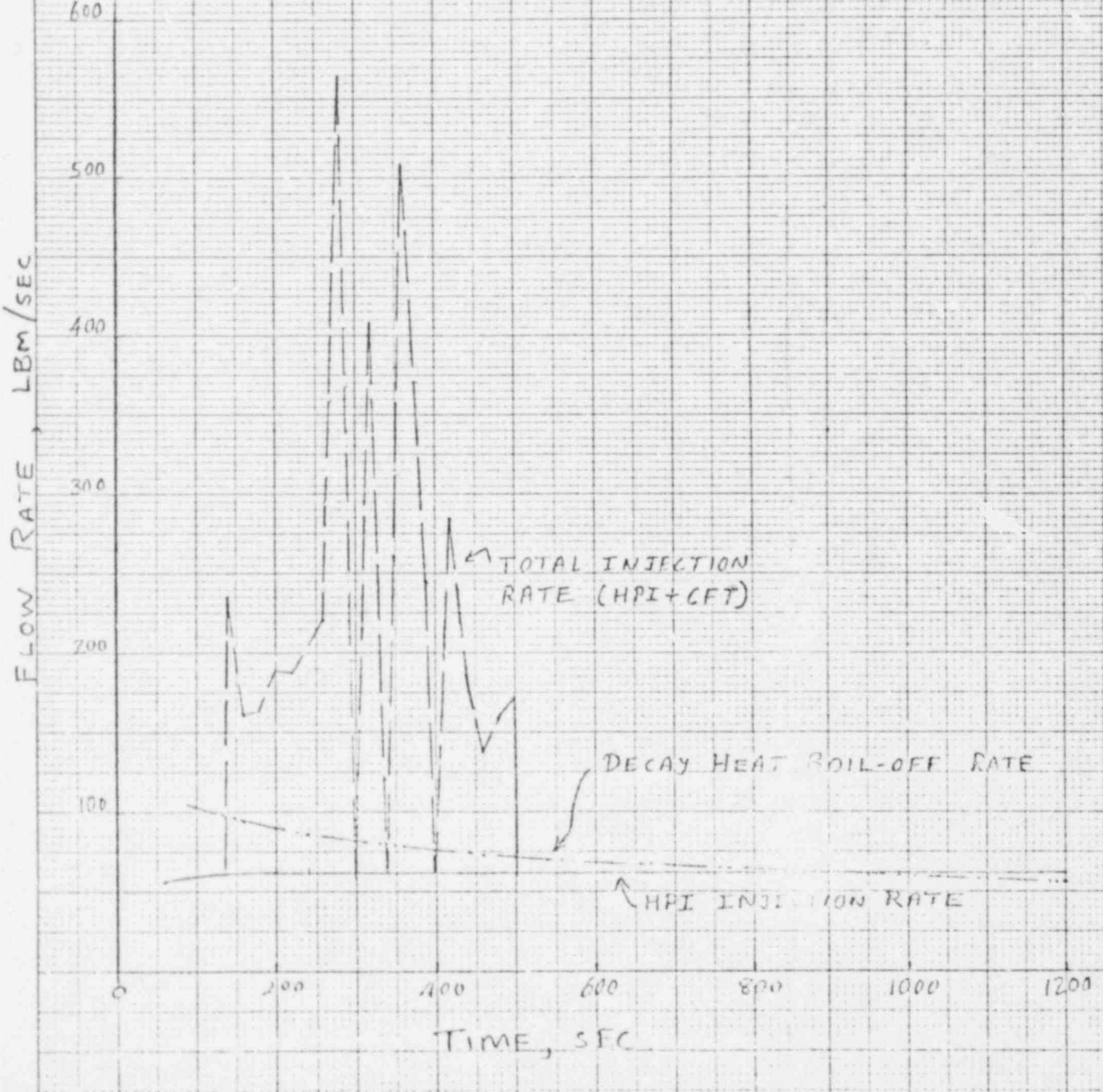


FIGURE -2 : CORE PRESSURE HISTORY
CFT LINE BREAK
INITIAL SYSTEM PRESSURE = 2200 PSIA
PRESSURE PSIA
1600
1400
1200
1000
800
600
400
200
0
0
200
400
600
800
1000
1200
TIME , SEC.



K+E 10 X 12 TO ½ INCH 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A. 46 1320

FIGURE 4: LONG TERM CORE COOLING
CFT LINE BREAK



CFT

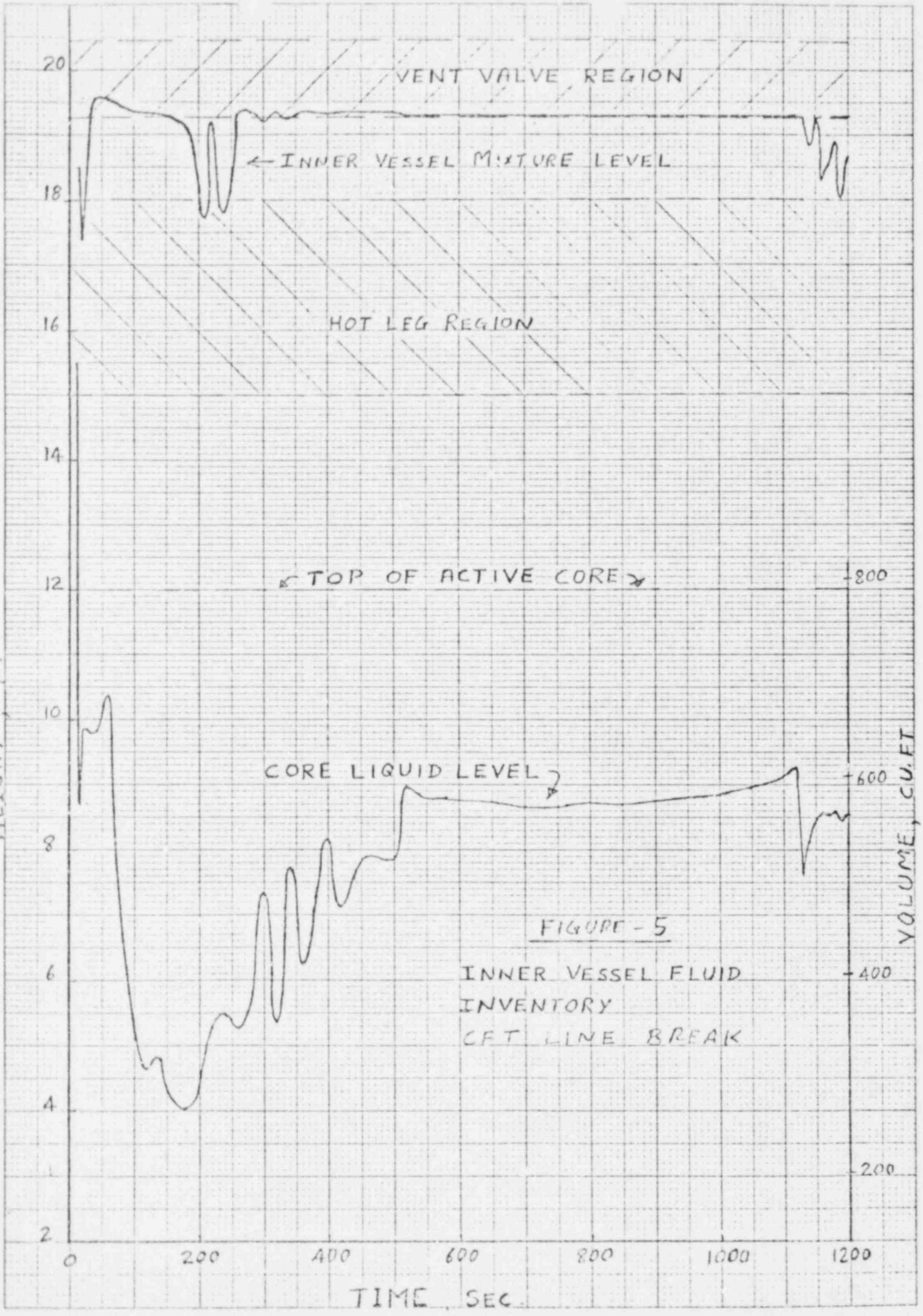
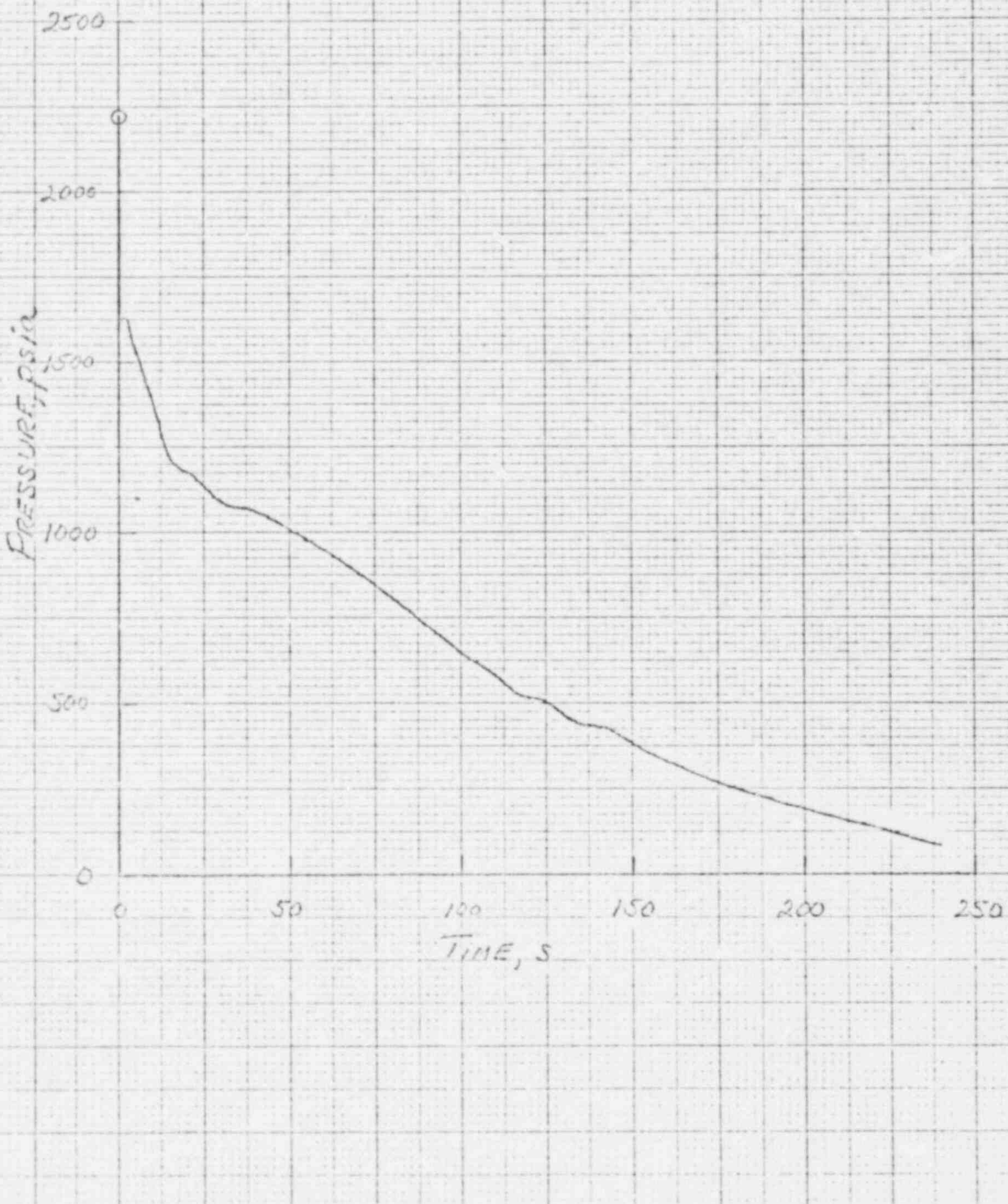


FIGURE - 5

INNER VESSEL FLUID
INVENTORY
CFT LINE BREAK

FIGURE 6: CORE PRESSURE HISTORY
0.5 FT² SPLIT AT PUMP DISCHARGE



0.5 ft²

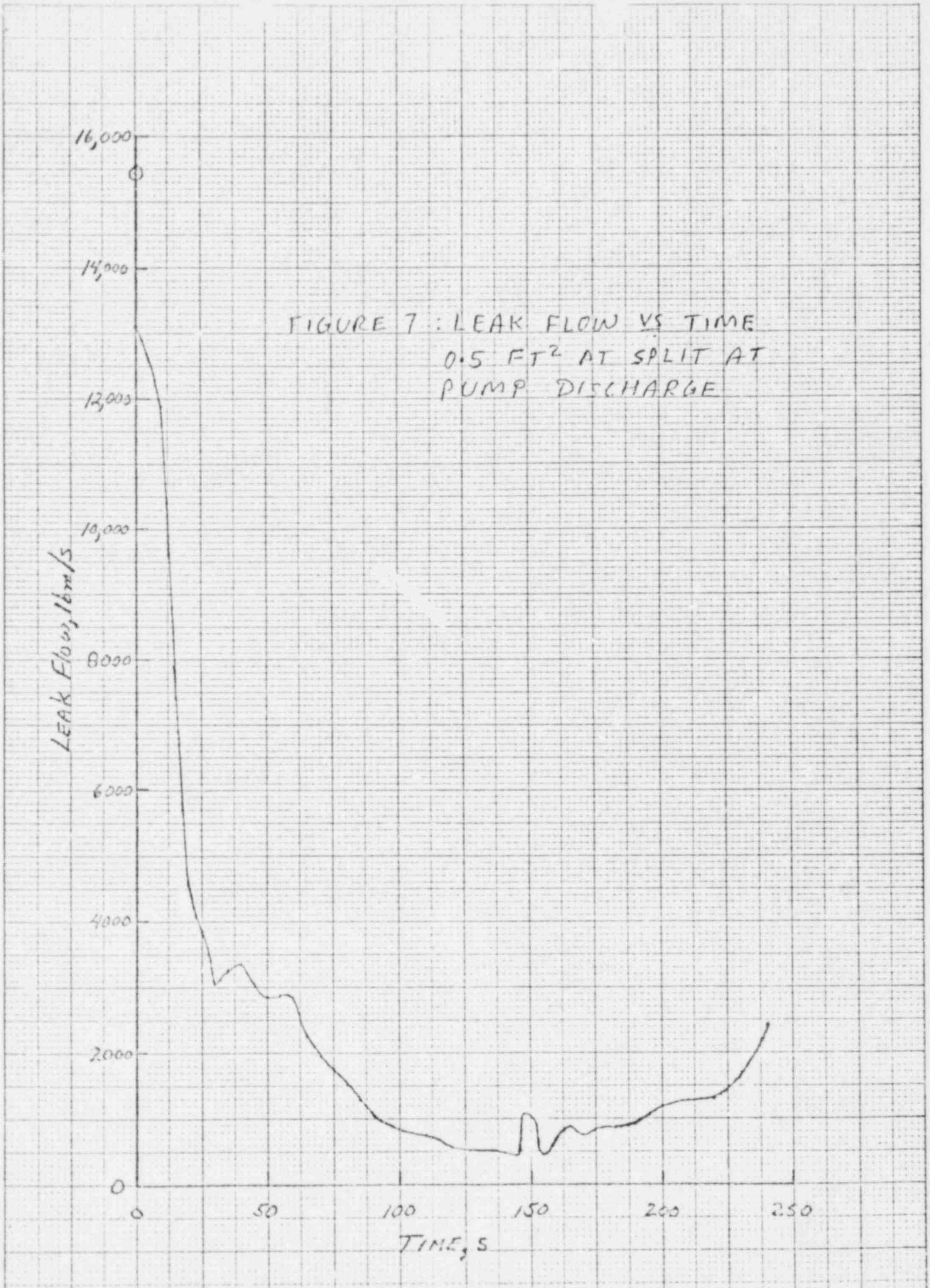


FIGURE 7 : LEAK FLOW VS TIME
0.5 FT² AT SPLIT AT
PUMP DISCHARGE
16,000
14,000
12,000
10,000
8000
6000
4000
2000
0
0
50
100
150
200
250
LEAK Flow, lbm/s
TIME, S

46 1320

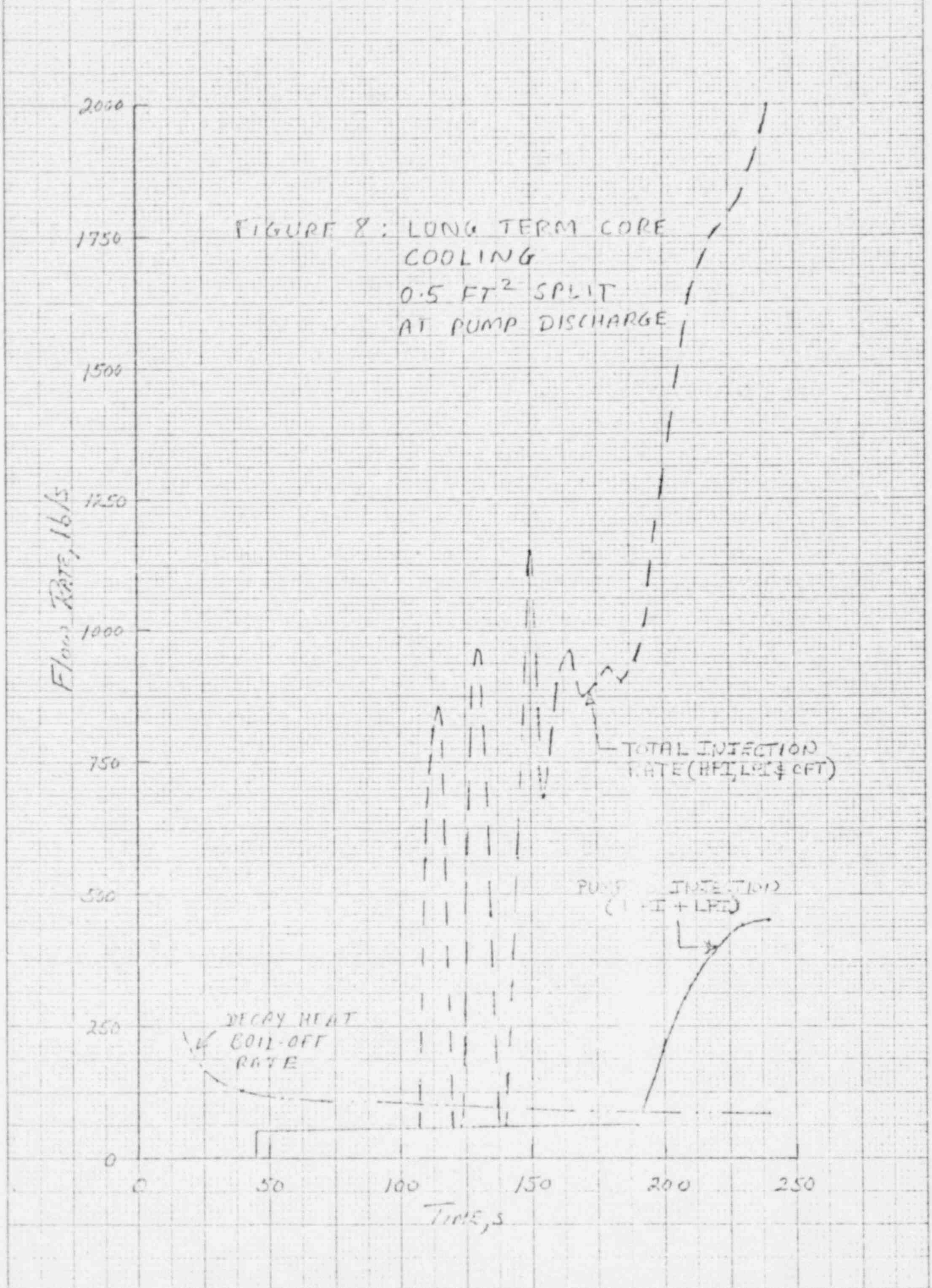


FIGURE 8: LONG TERM CORE COOLING
0.5 FT² SPLIT
AT PUMP DISCHARGE
Flow Rate, lb/s
Time, s
2000
1750
1500
1250
1000
750
500
250
0
0
50
100
150
200
250
TOTAL INJECTION RATE (HPI, LPI & CFT)
PUMP INJECTION (HPI + LPI)
DECAY HEAT BOIL-OFF RATE

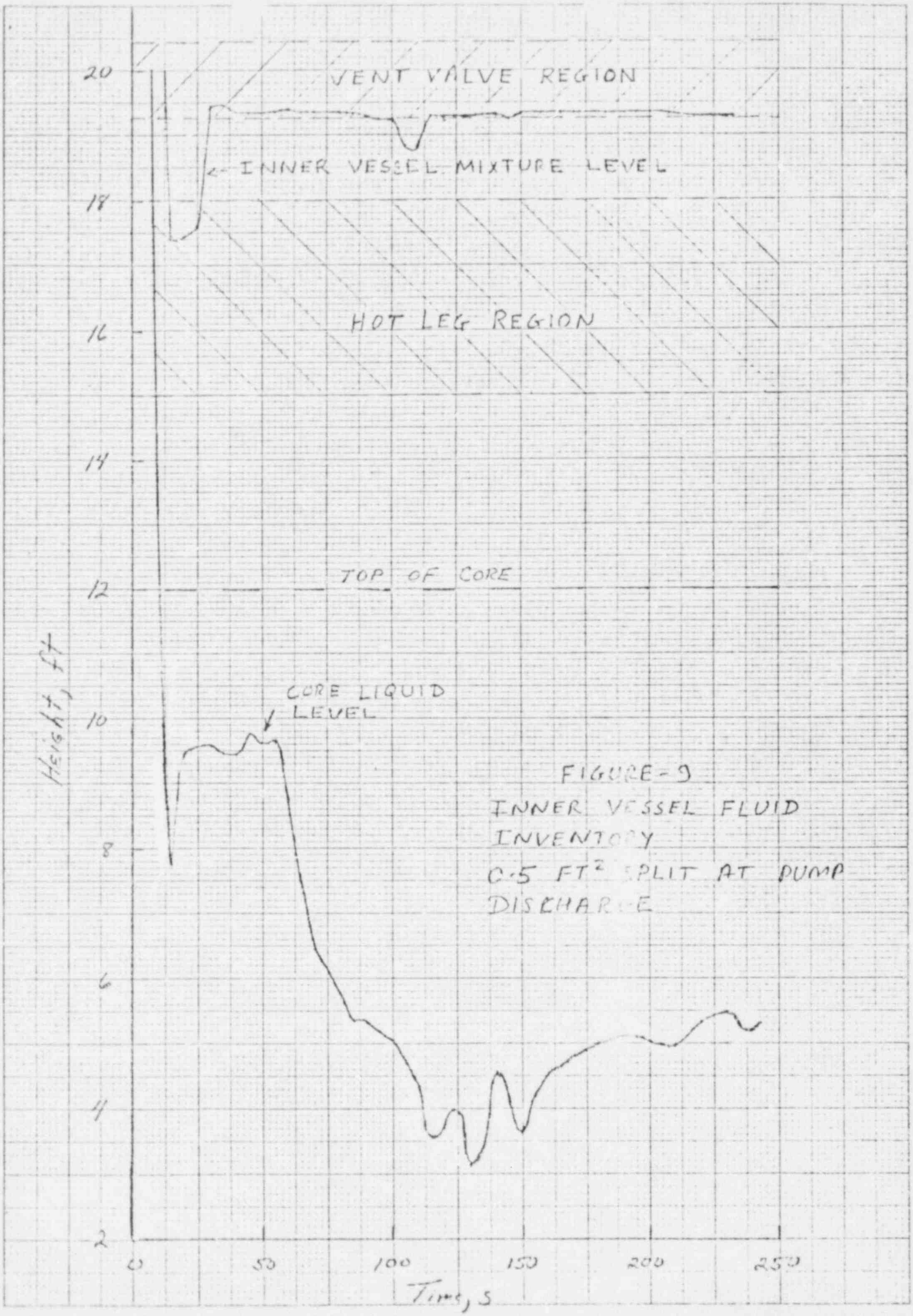
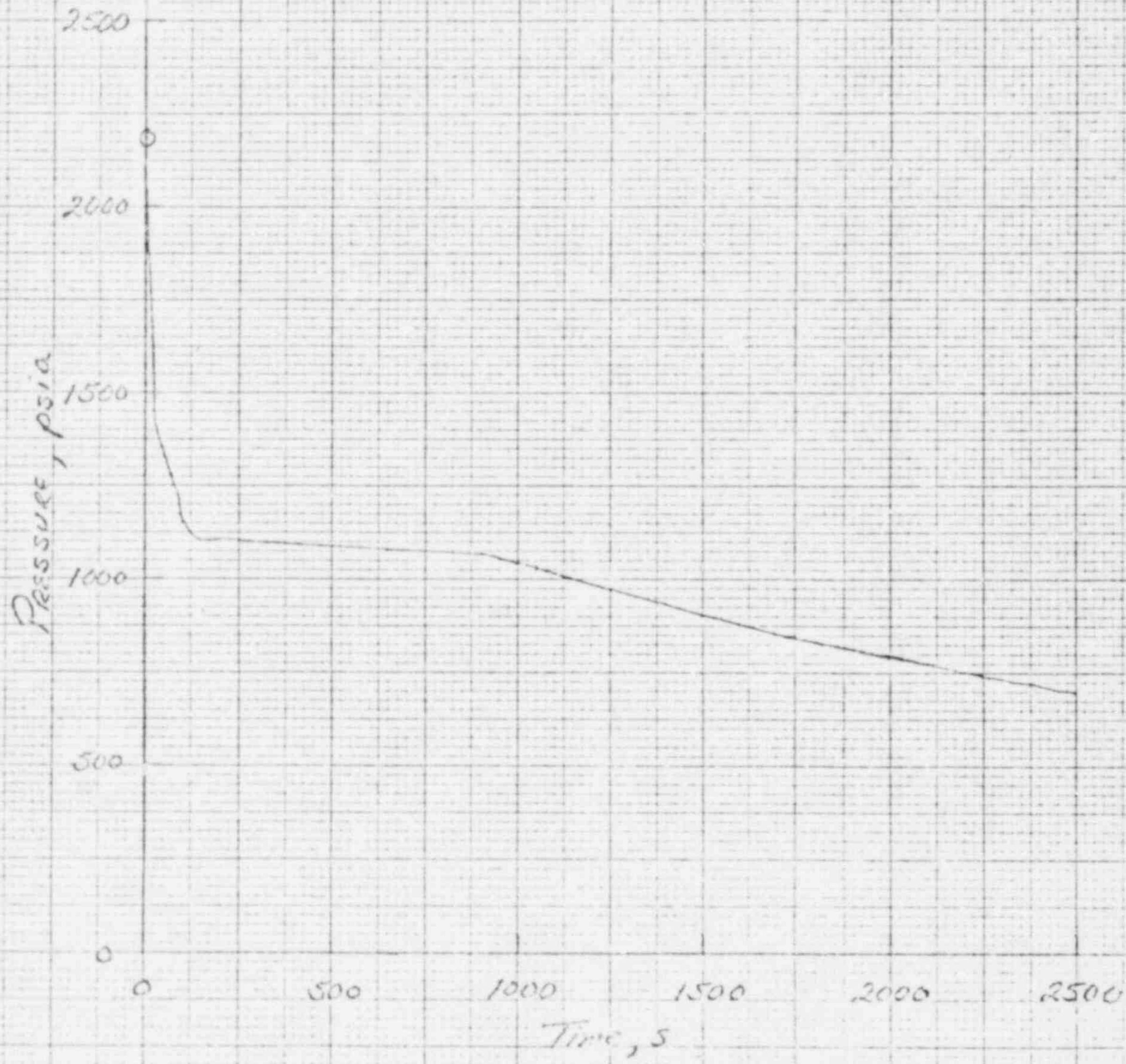
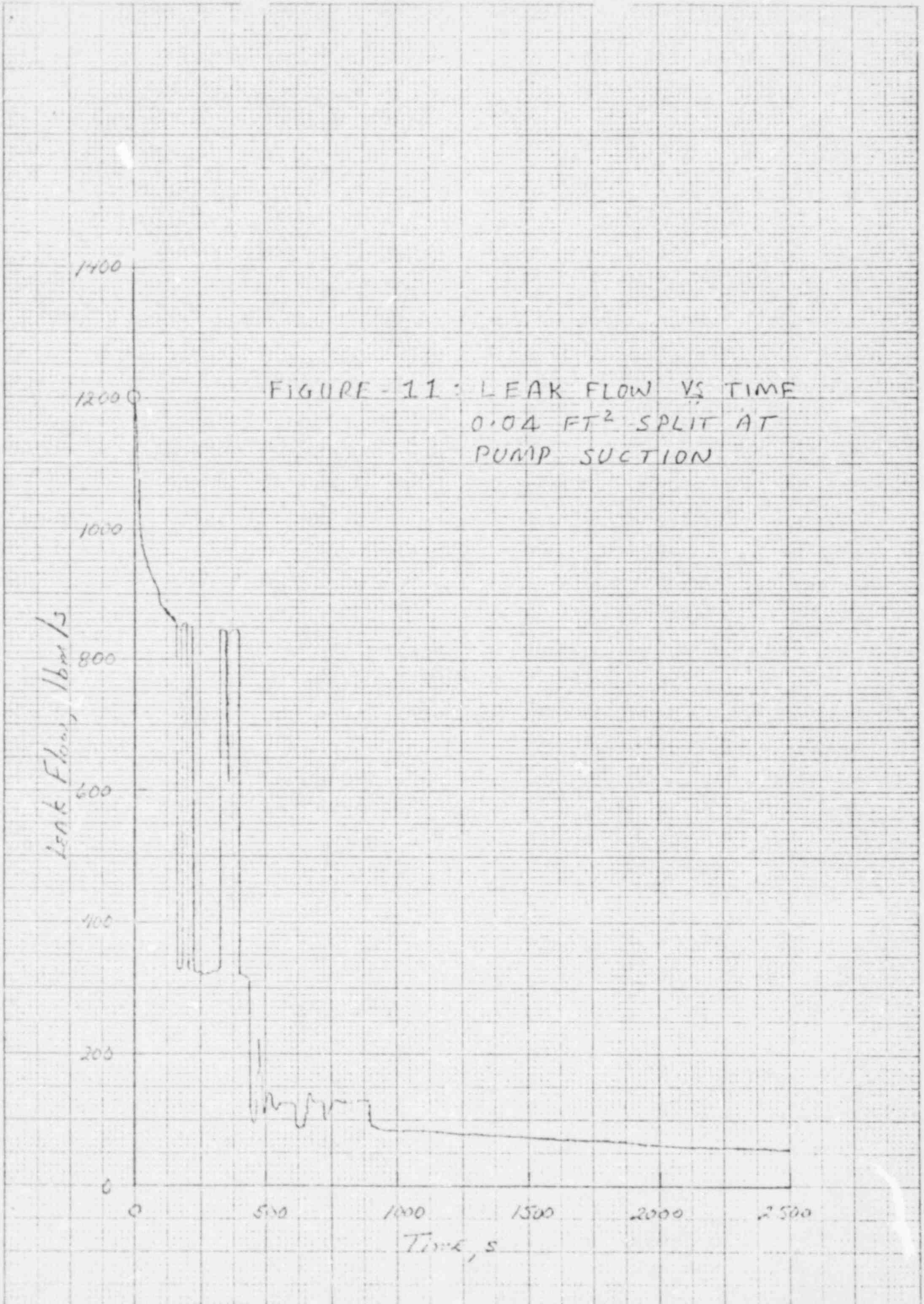


FIGURE-9
INNER VESSEL FLUID INVENTORY
0.5 FT² SPLIT AT PUMP DISCHARGE

FIGURE 10: CORE PRESSURE HISTORY
0.01 FT^2 SPLIT AT
PUMP SUCTION



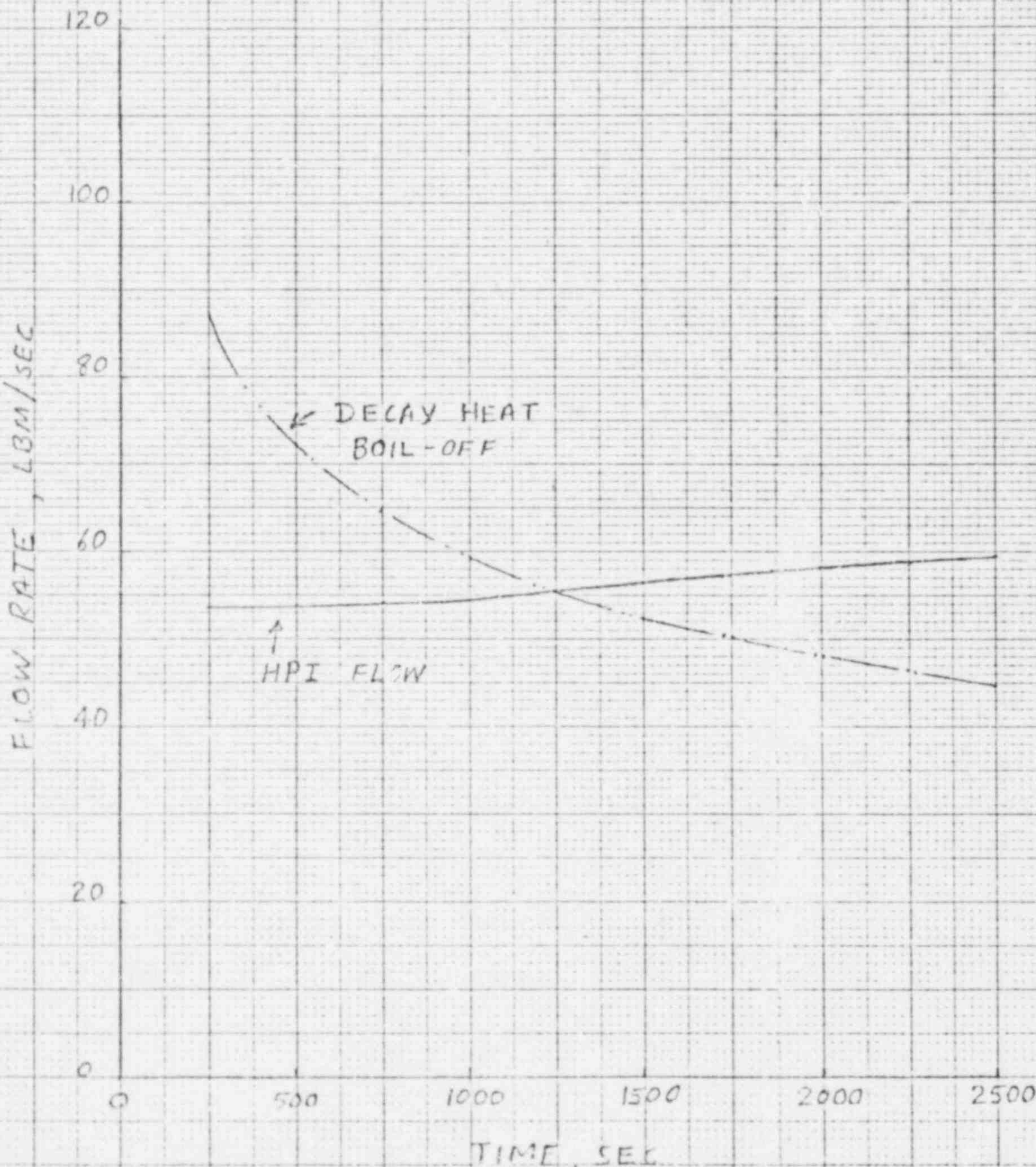


0.04 ft²

46 1322

K&E 10 X 10 TO ½ INCH 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.

FIGURE - 13
LONG TERM COOLING.
0.04 FT^2 SPLIT AT PUMP
SUCTION.



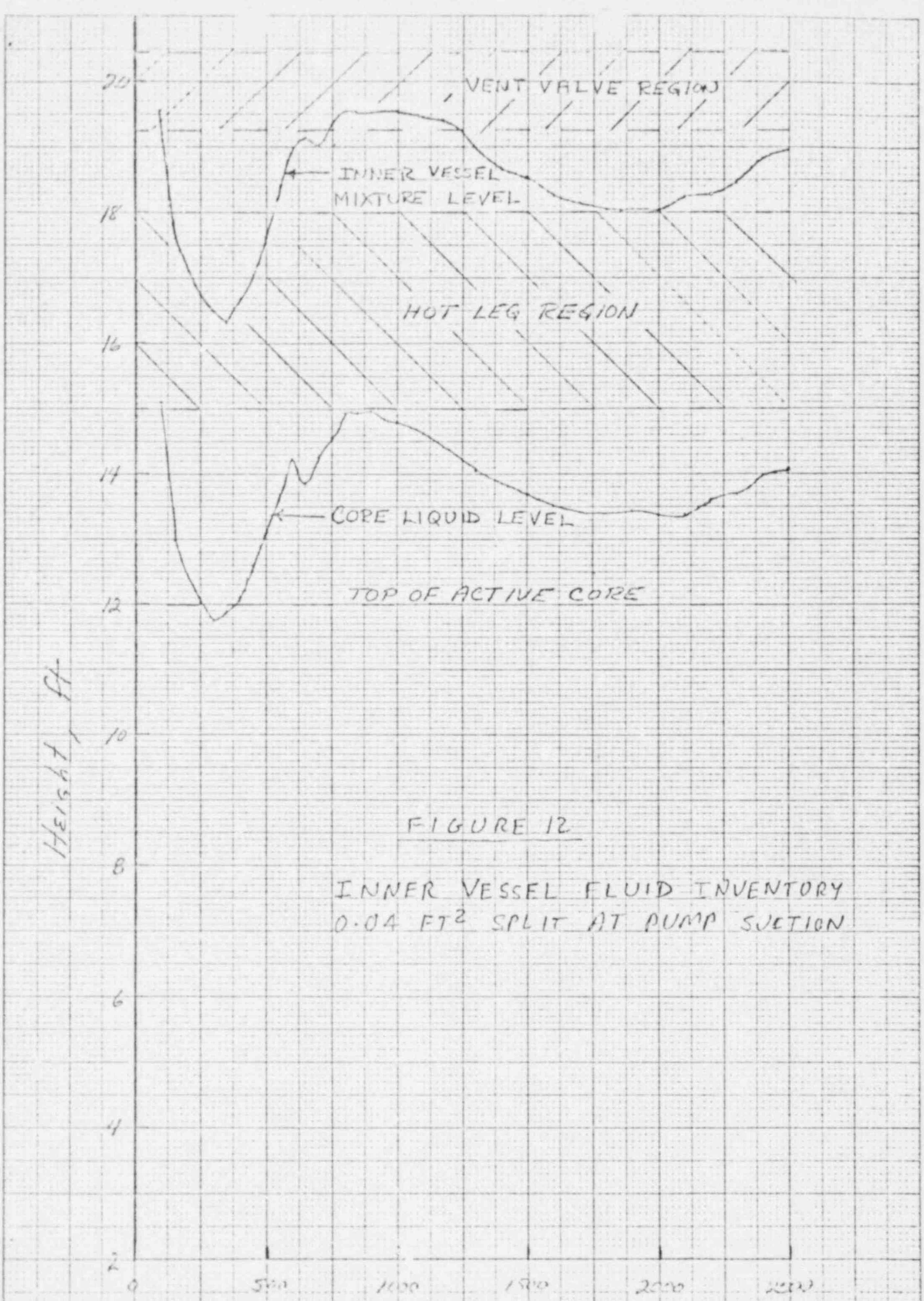


FIGURE 12

INNER VESSEL FLUID INVENTORY
0.04 FT^2 SPLIT AT PUMP SUCTION

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