Final Report of the Results of the Reanalysis for a Large Break Loss of Coolant Accident at Indian Point Unit 3

The Loss of Coolant Accident (LOCA) has been reanalyzed for Indian Point Unit III The following information amends the Safety Analysis Report section on Major Reactor Coolant System Pipe Ruptures. The results are consistent with acceptance criteria provided in reference 1.

The description of the various aspects of the LOCA analysis is given in WCAP-8339[2]. The individual computer codes which comprise the Westinghouse Emergency Core Cooling System (ECCS) evaluation model are described in detail in separate reports [3-6] along with code modifications specified in references 7, 9 and 10. The analysis presented here was performed with the February 1978 version of the evaluation model which includes modifications delineated in references 11, 12, 13 and 14.

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Results

The analysis of the loss of coolant accident is performed at 102 percent of the licensed core power rating. The peak linear power and total core power used in the analysis are given in Table 2. Since there is margin between the value of peak linear power density used in this analysis and the value of the peak linear power density expected during plant operation, the peak clad temperature calculated in this analysis is greater than the maximum clad temperature expected to exist.

Table 1 presents the occurence time for various events throughout the accident transient.

Table 2 presents selected input values and results from the hot fuel rod thermal transient calculation. For these results, the hot spot is defined as the location of maximum peak clad temperatures. That location is specified in Table 2 for each break analyzed. The location is indicated in feet which presents elevation above the bottom of the active fuel stack.

Table 3 presents a summary of the various containment systems parameters and structural parameters which were used as input to the COCO computer code[6] used in this analysis.

Tables 4 and 5 present reflood mass and energy releases to the containment, and the broken loop accumulator mass and energy release to the containment, respectively.

The results of several sensitivity studies are reported[8]. These results are for conditions which are not limiting in nature and hence are reported on a generic basis.

Figures 1 through 17 present the transients for the principle parameters for the break sizes analyzed. The following items are noted:

Figures 1A - 3C: Quality, mass velocity and clad heat transfer coefficient for the hotspot and burst locations

Figures 4A - 6C: Core pressure, break flow, and core pressure drop. The break flow is the sum of the flowrates from both ends of the guillotine break. The core pressure drop is taken as the pressure just before the core inlet to the pressure just beyond the core outlet

Figures 7A - 9C: Clad temperature, fluid temperature and core flow. The clad and fluid temperatures are for the hot spot and burst locations

Figures 10A - 11C: Downcomer and core water level during reflood, and flooding rate

Figures 12A - 13C: Emergency core cooling system flowrates, for both accumulator and pumped safety injection

Figures 14A - 15C: Containment pressure and core power transients

Figures 16, 17: Break energy release during blowdown and the containment wall condensing heat transfer coefficient for the worst break

Conclusions - Thermal Analysis

For breaks up to and including the double ended severance of a reactor coolant pipe, the Emergency Core Cooling System will meet the Acceptance Criteria as presented in 10CFR50.46.[1] That is:

1. The calculated peak clad temperature does not exceed 2200°F based on a total core peaking factor of 2.04

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircalloy in the reactor.

3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The cladding oxidation limits of 17% are not exceeded during or after quenching.

4. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

References for Section 15.4.1

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors", 10CFR50.46 and Appendix K of 10CFR50.46. Federal Register, Volume 39, Number 3, January 4, 1974.

2. Bordelon, F. M., Massie, H. W., And Zordan, T. A., "Westinghouse ECCS Evaluation Model-Summary," WCAP-8339, July 1974.

3. Bordelon, F. M., et al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," WCAP-8302 (Proprietary Version), WCAP-8306 (Non-Proprietary Version), June 1974.

4. Bordelon, F. M., Et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8301 (Proprietary Version), WCAP-8305 (Non-Proprietary Version), June 1974.

5. Kelly, R. D., et al., "Calculational Model for Core Reflooding after a Loss-of-Coolant Accident (WREFLOOD Code)." WCAP-8170 (Proprietary Version), WCAP-8171 (Non-Proprietary Version), June 1974.

6. Bordelon, F. M., and Murphy E. T., "Containment Pressure Analysis Code (COCO)," WCAP-8327 (Proprietary Version), WCAP-8326 (Non-Proprietary Version), June 1974.

7. Bordelon F. M., et al., "The Westinghouse ECCS Evaluation Model: Supplementary Information," WCAP-8471 (Proprietary Version), WCAP-8472 (Non-Proprietary Version), January 1975.

8. Salvatori, R., "Westinghouse ECCS - Plant Sensitivity Studies," WCAP-8340 (Proprietary Version), WCAP-8356 (Non-Proprietary Version), July 1974.

9. "Westinghouse ECCS Evaluation Model, October, 1975 Versions," WCAP-8622 (Proprietary Version), WCAP-8623 (Non-Proprietary Version), November, 1975.

10. Letter from C. Eicheldinger of Westinghouse Electric Corporation to D. B. Vassalo of the Nuclear Regulatory Commission, letter number NS-CE-924, January 23, 1976.

11. Kelly, R. D., Thompson, C. M., et. al., "Westinghouse Emergency Core Cooling System Evaluation Model for Analyzing Large LOCA's During Operation With One Loop Out of Service for Plants Without Loop Isolation Valves," WCAP-9166, February, 1978.

12. Eicheldinger C., "Westinghouse ECCS Evaluation Model, February 1978 Version," WCAP-9220-P-A (Proprietary Version), WCAP-9221-A (Non-Proprietary Version), February, 1978.

13. Letter from T. M. Anderson of Westinghouse Electric Corporation to John Stolz of the Nuclear Regulatory Commission, letter number NS-TMA-1981, Nov. 1, 1978.

14. Letter from T. M. Anderson of Westinghouse Electric Corporation to John Stolz of the Nuclear Regulatory Commission, letter number NS-TMA-2014, Dec. 11, 1978.

15. Letter from T. M. Anderson of Westinghouse Electric Corporation to Darrell G. Eisenhut of the Nuclear Regulatory Commission, letter number NS-TMA-2163, November 16, 1979.

The Nuclear Regulatory Commission (NRC) issued a letter dated November 9, 1979 to operators of light water reactors regarding fuel rod models used in Loss of Coolant Accident (LOCA) ECCS evaluation models. That letter describes a meeting called by the NRC on November 1, 1979 to present draft report NUREG 0630, "Cladding Swelling and Rupture Models for LOCA Analysis." At the meeting, representatives of NSSS vendors and fuel suppliers were asked to show how plants licensed using their LOCA/ECCS evaluation model continued to conform to 10 CFR Part 50-46 in view of the new fuel rod models presented in draft NUREG 0630. Westinghouse representatives presented information on the fuel rod models used in analyses for plants licensed with the Westinghouse ECCS evaluation model and discussed the potential impact of fuel rod model changes on results of those analyses. That information was formally documented in letter NS-TMA-2147, dated November 2, 1979, and formed the basis for the Westinghouse conclusion that the information presented in draft NUREG 0630 did not constitute a safety problem for Westinghouse plants and that all plants conformed with NRC regulations. In the November 9, 1979 letter, the NRC requested that operators of light water reactors provide, within sixty (60) days, information which will enable the staff to determine, in light of the fuel rod model concerns, whether or not further action is necessary.

This letter provides information on the LOCA analysis of your plant required to respond to that request. Note, however, that a significant amount of discussion and information exchange between Westinghouse and the NRC has transpired since the November 2 letter (NS-TMA-2147) was prepared and the basis for demonstrating compliance with 10 CFR part 50 has been modified. The following is an outline of the significant events that occurred since November 2 and is provided to update you on this situation.

As a result of compiling information for letter NS-TMA-2147, Westinghouse recognized a potential discrepancy in the calculation of fuel rod burst for cases having clad heatup rates (prior to rupture) significantly lower than 25 degrees F per second. This issue was reported to the NRC staff, by telephone, on November 9, 1979, and although independent of the NRC fuel rod model concern, the combined effect of this issue and the effect of the NRC fuel rod models had to be studied. Details of the work done on this issue were presented to the NRC on November 13, 1969 and documented in letter NS-TMA-2163 dated November 16, 1979. That work included development of a procedure to determine the clad heatup rate prior to burst and a reevaluation of operating Westinghouse plants with consideration of a modified Westinghouse fuel rod burst model. As part of this reevaluation, the Westinghouse position on NUREG-0630 was reviewed and it was still concluded that the information presented in draft NUREG-0630 did not constitute a safety problem for plants licensed with the Westinghouse ECCS evaluation model.

On December 6, 1979, NRC and Westinghouse personnel discussed the information thus far presented. At the conclusion of that discussion, the NRC staff requested Westinghouse to provide further detail on the potential impact of modifications to each of the fuel rod models used in the LOCA analysis and to outline analytical model improvements in other parts of the analysis and the potential benefit associated with those

improvements. This additional information was compiled from various LOCA analysis results and documented in letter NS-TMA-2174 dated December 7, 1979.

Another meeting was held in Bethesda on December 20, 1979 where NRC and Westinghouse personnel established: 1) The currently accepted procedure for assessing the potential impact on LOCA analysis results of using the fuel rod models presented in draft NUREG-0630 and 2) Acceptable benefits resulting from analytical model improvements that would justify continued plant operation for the interim until differences between the fuel rod models of concern are resolved.

The information following on pages 3 - 6 is expected to satisfy the NRC request for information on your plant which will enable the NRC to determine whether or not further action is necessary.

Part of the Westinghouse effort provided to assist in the resolution of these LOCA fuel rod model differences is documented in letter NS-TMA-2175, dated December 10, 1979, which contains Westinghouse comments on draft NUREG-0630. As stated in that letter, Westinghouse believes the current Westinghouse models to be conservative and to be in compliance with Appendix K.

A. Evaluation of the potential impact of using fuel rod models presented in draft NUREG-0630 on the Loss of Coolant Accident (LOCA) analysis for Indian Point Unit 3 with 12% steam generator tube plugging.

This evaluation is based on the limiting break LOCA analysis identified as follows:

BREAK TYPE - DOUBLE ENDED COLD LEG GUILLOTINE

BREAK DISCHARGE COEFFICIENT $C_D = 0.8$

WESTINGHOUSE ECCS EVALUATION MODEL VERSION February '78

CORE PEAKING FACTOR 2.04

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR THE BURST REGION OF THE CLAD - 1931.1 °F = PCT_B

ELEVATION - 6.0 Feet.

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR A NON-RUPTURED REGION OF THE CLAD - 1995.47 °F = PCT_N

ELEVATION - 7.5 Feet

CLAD STRAIN DURING BLOWDOWN AT THIS ELEVATION - 6.23 Percent
MAXIMUM CLAD STRAIN AT THIS ELEVATION - 10.0 Percent

Maximum temperature for this non-burst node occurs when the core reflood rate is GREATER than 1.0 inch per second and reflood heat transfer is based on the FLECHT calculation.

AVERAGE HOT ASSEMBLY ROD BURST ELEVATION - N/A Feet

HOT ASSEMBLY BLOCKAGE CALCULATED - N/A Percent

1. BURST NODE

The maximum potential impact on the ruptured clad node is expressed in letter NS-TMA-2174 in terms of the change in the peaking factor limit (FQ) required to maintain a peak clad temperature (PCT) of 2200°F and in terms of a change in PCT at a constant FQ. Since the clad-water reaction rate increases significantly at temperatures above 2200.°F, individual effects (such as ΔPCT due to changes in several fuel rod models) indicated here may not accurately apply over large ranges,

but a simultaneous change in FQ which causes the PCT to remain in the neighborhood of 2200.°F justifies use of this evaluation procedure.

From NS-TMA-2174:
For the Burst Node of the clad:

- 0.01 ΔFQ → ~ 150°F BURST NODE ΔPCT

- Use of the NRC burst model (and the revised Westinghouse burst model) could require an FQ reduction of 0.027.

- The maximum estimated impact of using the NRC strain model is a required FQ reduction of 0.03.

Therefore, the maximum penalty for the Hot Rod burst node is:

$$\Delta PCT_1 = (0.027 + .03)\ (150^{\circ}F/.01) = 855^{\circ}F$$

Margin to the 2200°F limit is:

$$\Delta PCT_2 = 2200.^{\circ}F - PCT_B = \underline{\ 269\ }\ ^{\circ}F$$

The FQ reduction required to maintain the 2200°F clad temperature limit is:

$$\Delta FQ_B = (\Delta PCT_1 - \Delta PCT_2)\ \left(\frac{.01\ \Delta FQ}{150^{\circ}F}\right)$$

$$= (\underline{855} - \underline{269})\ \left(\frac{.01}{150}\right)$$

$$= \underline{\ 0.04\ } \quad \text{(but not less than zero).}$$

2. NON-BURST NODE

The maximum temperature calculated for a non-burst section of clad typically occurs at an elevation above the core mid-plane during the core reflood phase of the LOCA transient. The potential impact on that maximum clad temperature of using the NRC fuel rod models can be estimated by examining two aspects of the analyses. The first aspect is the change in pellet-clad gap conductance resulting from a difference in clad strain at the non-burst maximum clad temperature node elevation. Note that clad strain all along the fuel rod stops after clad burst occurs and use of a different clad burst model can change the time at which burst is calculated. Three sets of LOCA analysis results were studied to establish an acceptable sensitivity to apply generically in this evaluation. The possible PCT increase resulting from a change in strain (in the Hot Rod) is +20.°F per percent decrease in strain at the maximum clad temperature

locations. Since the clad strain calculated during the reactor coolant system blowdown phase of the accident is not changed by the use of NRC fuel rod models, the maximum decrease in clad strain that must be considered here is the difference between the "maximum clad strain" and the "clad strain at the end of RCS blowdown" indicated above.

Therefore:

$$\Delta PCT_3 = \left(\frac{20^oF}{.01 \text{ strain}}\right) (\text{MAX STRAIN} - \text{BLOWDOWN STRAIN})$$

$$= \left(\frac{20}{.01}\right) (\underline{10} - \underline{6.23})$$

$$= \underline{75^oF}$$

The second aspect of the analysis that can increase PCT is the flow blockage calculated. Since the greatest value of blockage indicated by the NRC blockage model is 75 percent, the maximum PCT increase can be estimated by assuming that the current level of blockage in the analysis (indicated above) is raised to 75 percent and then applying an appropriate sensitivity formula shown in NS-TMA-2174.

Therefore,

$$\Delta PCT_4 = 1.25^oF\ (50 - \text{PERCENT CURRENT BLOCKAGE}) + 2.36^oF\ (75-50)$$

$$= 1.25\ (50 - \underline{0}) + 2.36\ (75-50)$$

$$= \underline{121.5}\ ^oF$$

But since $V_{in} > 1.0$ at PCT time, $\Delta PCT_4 = 0.0$.
If PCT_N occurs when the core reflood rate is greater than 1.0 inch per second $\Delta PCT_4 = 0$. The total potential PCT increase for the non-burst node is then

$$\Delta PCT_5 = \Delta PCT_3 + \Delta PCT_4 = 75 + 0 = 75.0$$

Margin to the 2200°F limit is

$$\Delta PCT_6 = 2200^oF - PCT_N = 2200 - 1995 = 205$$

The FQ reduction required to maintain this 2200°F clad temperature limit is (from NS-TMA-2174)

$$\Delta FQ_N = (\Delta PCT_5 - \Delta PCT_6) \left(\frac{.01 \Delta FQ}{10^oF\ \Delta PCT}\right)$$

$\Delta FQ_N = \underline{-0.13}$ but not less than zero.

Therefore, $\Delta FQ_N = 0.0$.

The peaking factor reduction required to maintain the 2200 °F clad temperature limit is therefore the greater of ΔFQ_B and ΔFQ_N,

or; $\Delta FQ_{PENALTY}$ = 0.04

B. The effect on LOCA analysis results of using improved analytical and modeling techniques (which are currently approved for use in the Upper Head Injection plant LOCA analyses) in the reactor coolant system blowdown calculation (SATAN computer code) has been quantified via an analysis which has recently been submitted to the NRC for review. Recognizing that review of that analysis is not yet complete and that the benefits associated with those model improvements can change for other plant designs, the NRC has established a credit that is acceptable for this interim period to help offset penalties resulting from application of the NRC fuel rod models. That credit for two, three and four loop plants is an increase in the LOCA peaking factor limit of 0.12, 0.15 and 0.20 respectively.

C. The peaking factor limit adjustment required to justify plant operation for this interim period is determined as the appropriate ΔFQ credit identified in section (B) above, minus the $\Delta FQ_{PENALTY}$ calculated in section (A) above (but not greater than zero).

FQ ADJUSTMENT = 0.20 - 0.04 = 0.16

Therefore, ΔFQ Adjustment = 0.0

TABLE 1

LARGE BREAK - TIME SEQUENCE OF EVENTS

EVENT	OCCURENCE TIME (SECONDS)		
	DECLG, C_D = 1.0	DECLG, C_D = 0.8	DECLG, C_D = 0.6
Accident Initiation	0.0	0.0	0.0
Reactor Trip Signal	0.578	0.581	0.585
Safety Injection Signal	0.87	0.94	1.06
Start Accumulator Injection	13.6	14.1	16.2
End of ECC Bypass	24.985	25.984	26.320
End of Blowdown	28.315	29.236	29.539
Bottom of Core Recovery	41.863	43.103	42.806
Accumulators Empty	55.65	56.140	58.109
Start Pumped ECC Injection	25.87	25.94	26.06

TABLE 2

LARGE BREAK - ANALYSIS INPUT AND RESULTS

Quantities in the calculations:

Licensed core power rating	3025 * MWt
Total core peaking factor	2.04
Peak linear power	102% of 12.74 kw/ft
Accumulator water volume	800 cubic feet per tank
Accumulator pressure	600 PSIA
Number of Safety Injection Pumps Operating	2
Steam Generator Tube Plugging Level	12% percent (uniform)
Fuel Parameters - Cycle All	Region All

Results

	DECLG, C_D = 1.0	DECLG, C_D = 0.8	DECLG, C_D = 0.6
Peak clad temperature (°F)	1992.5	1995.5	1980.7
Location (feet)	6.0	7.5	7.75
Maximum local clad/water reaction (%)	4.72	3.99	3.71
Location (feet)	6.0	6.0	5.5
Total core clad/water reaction (%)	<0.3	<0.3	<0.3
Hot rod burst time (seconds)	31.8	34.8	34.0
Location (feet)	6.0	6.0	5.5

* 2% is added to this power level to account for calorimetric error.

TABLE 3

CONTAINMENT DATA (DRY CONTAINMENT)

Net Free Volume	2.61×10^6 Ft^3
Initial Conditions	
Pressure	14.7 PSIA
Temperature	90.0 ^oF
RWST Temperature	40.0 ^oF
Service Water Temperature	35.0 ^oF
Outside Temperature	-20.0 ^oF
Spray System	
Number of Pumps Operating	2
Runout Flow Rate	3000 GPM
Actuation Time	20 Secs.
Safeguards Fan Coolers	
Number of Fan Coolers Operating	5
Fastest Post Accident Initiation of Fan Coolers	30 Secs.

TABLE 3 (Cont'd)

STRUCTURAL HEAT SINK DATA

	Thickness (in)	Material	Area, ft^2
1)	0.0065 0.375 36.0	Paint Steel Concrete	49,838
2)	0.0065 0.500 36.0	Paint Steel Concrete	32,072
3)	12.0	Concrete	15,000
4)	0.375 12.0	Stainless Steel Concrete	10,000
5)	12.0	Concrete	61,000
6)	0.0065 0.500	Paint Steel	68,792
7)	0.0065 0.375	Paint Steel	79,904
8)	0.0065 0.250	Paint Steel	27,948
9)	0.0065 0.1875	Paint Steel	69,800
10)	0.125	Steel	3,000
11)	0.138	Steel	22,000

TABLE 3 (Cont'd)

PAINTED STRUCTURAL HEAT SINK DATA

	Structural Heat Sink Surface Area (Ft^2)	Structural Heat Sink Thickness (In)	Paint Thickness (Mils)
12)	0.0065 0.0625	Paint Steel	10,000
13)	0.0065 0.75 36.0	Paint Steel Concrete	565
14)	0.019 1.25 0.500 36.0	Stainless Steel Insulation Steel Concrete	7,634
15)	0.375	Steel	1,800

TABLE 4

DECLG BREAK (C_D = 0.8)

REFLOOD MASS AND ENERGY RELEASE

Time (Sec)	Mass Flow (LBM/Sec)	Energy Flow (BTU/Sec)
43.1	0.0	0.0
43.853	4.3	5.617×10^3
50.706	40.1	5.201×10^4
59.234	146.7	1.790×10^5
71.034	215.2	2.000×10^5
84.934	326.2	2.299×10^5
116.234	361.5	2.266×10^5
151.334	371.4	2.143×10^5
190.234	380.4	2.009×10^5
283.534	400.2	1.690×10^5

TABLE 5

DECLG BREAK (C_D = 0.8)

BROKEN LOOP ACCUMULATOR MASS AND ENERGY RELEASE

Time (Sec)	Mass Flow (LBM/Sec)	Energy Flow (BTU/Sec)
0.0	3005.	179,174.
2.0	2471.	147,292.
4.0	2149.	128,112.
6.0	1925.	114,785.
8.0	1755.	104,639.
10.0	1618.	96,493.
12.0	1506.	89,768.
14.0	1412.	84,169.
16.0	1332.	79,425.
18.0	1266.	75,497.
20.0	1210.5	72,169.
22.0	1163.	69,309.
24.0	1120.	66,777.
26.0	0.	0.

FLUID QUALITY -- DECLG (C_D = 0.8)

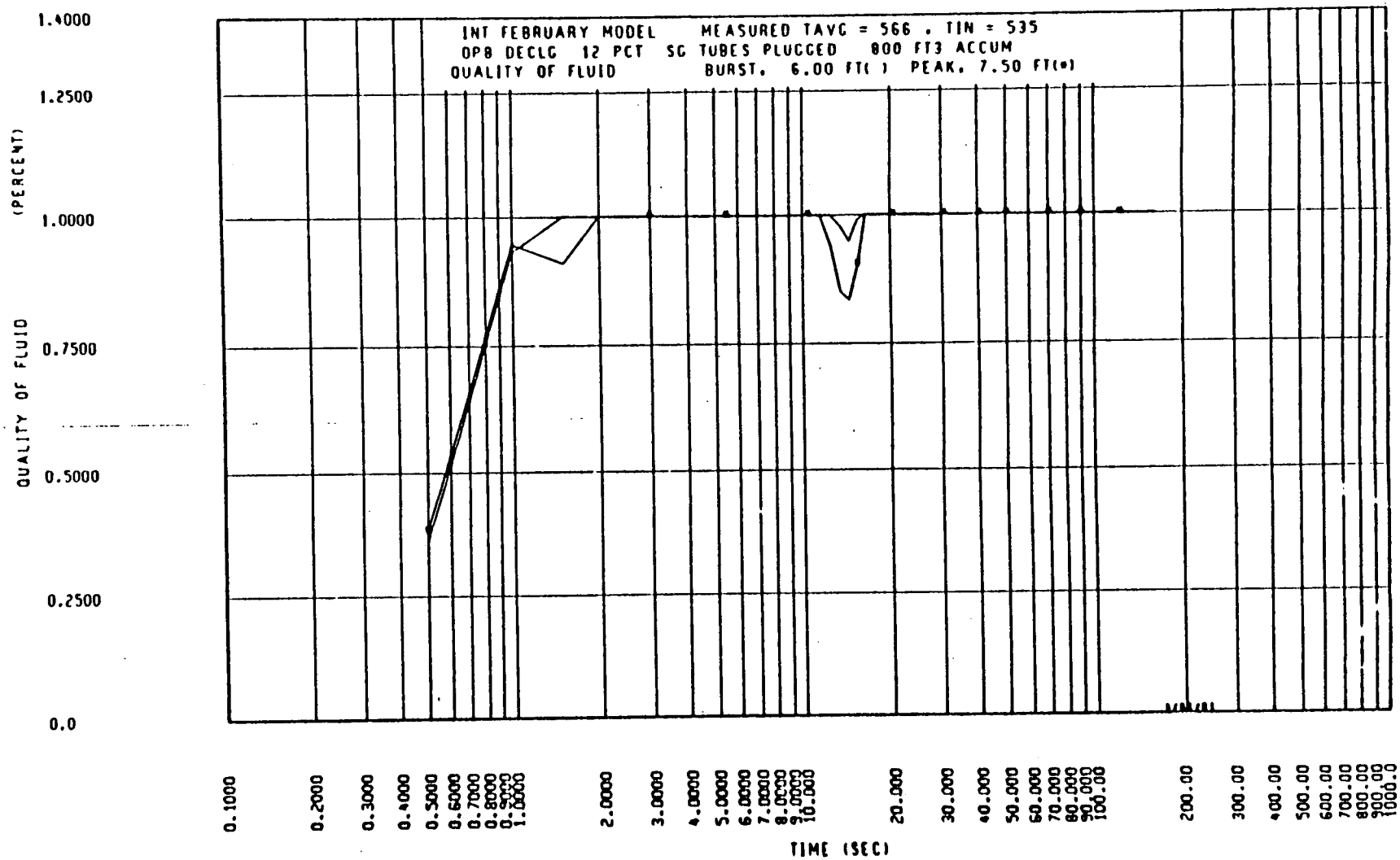


Figure 1A

MASS VELOCITY - DECLG (C_D = 0.8)

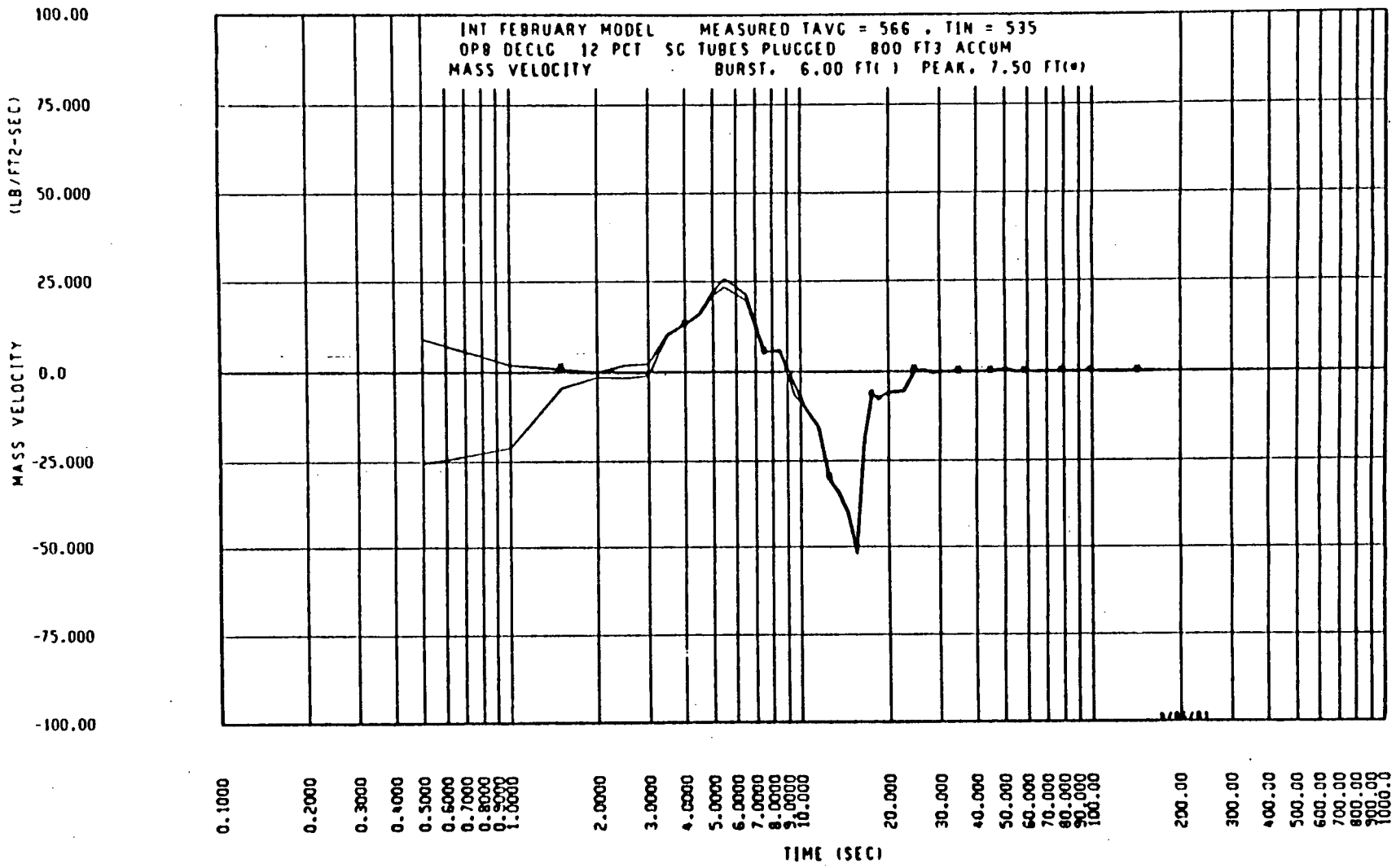


Figure 2A

HEAT TRANSFER COEFFICIENT - DECLG (C_D = 0.8)

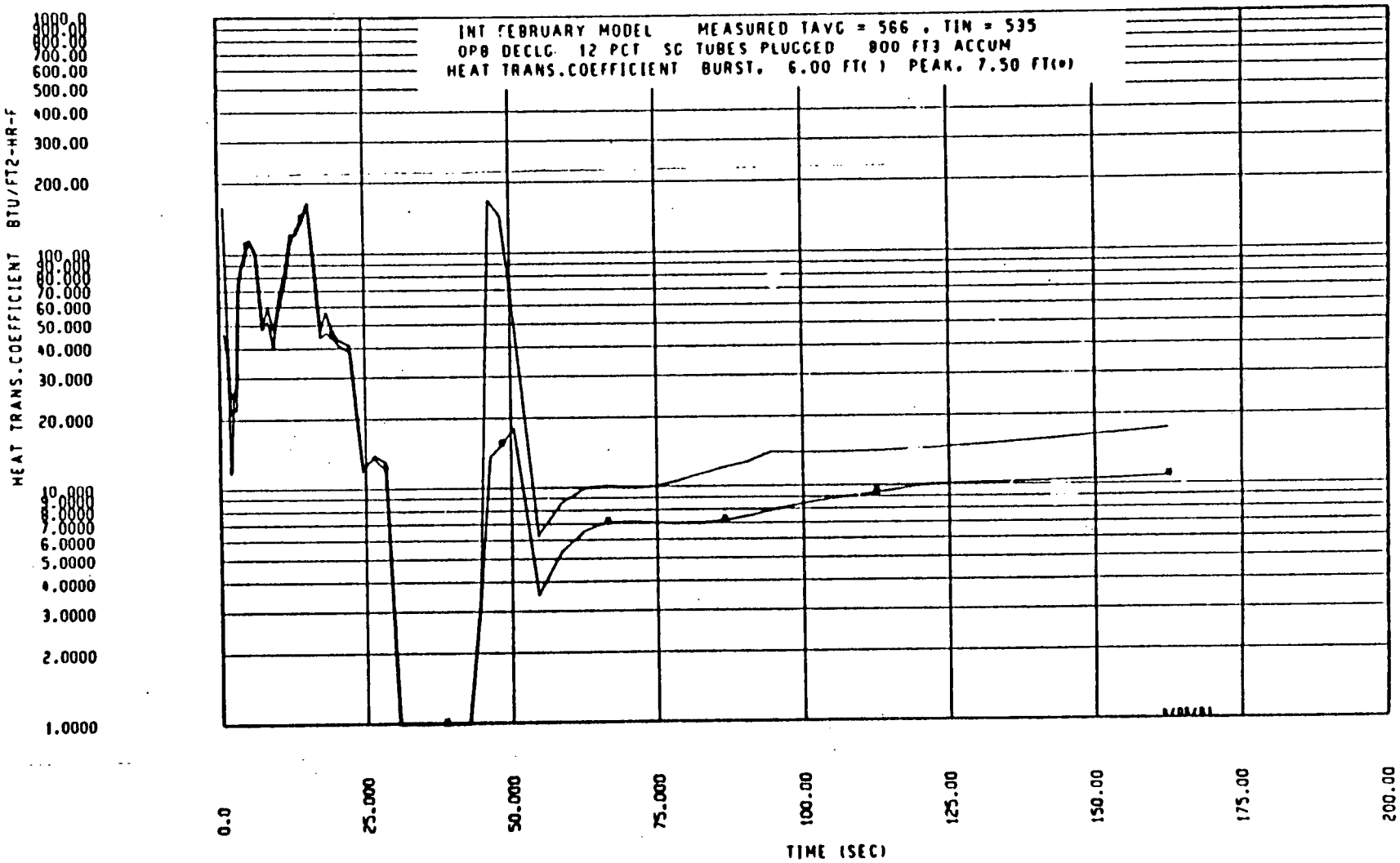


Figure 3A

CORE PRESSURE - DECLG (C_D = 0.8)

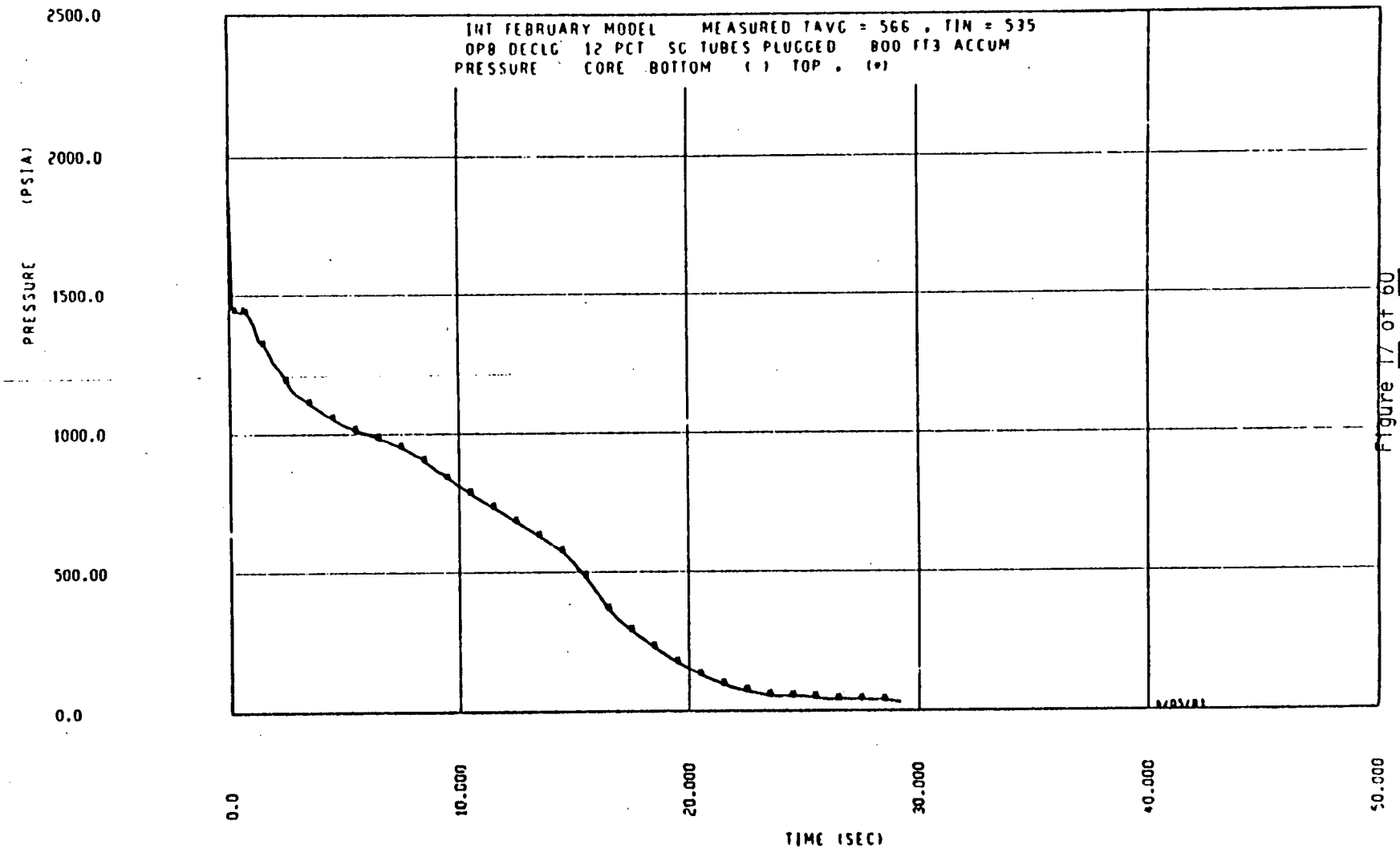


Figure 4A

Figure 17 of 60

BREAK FLOW RATE - DECLG (C_D = 0.8)

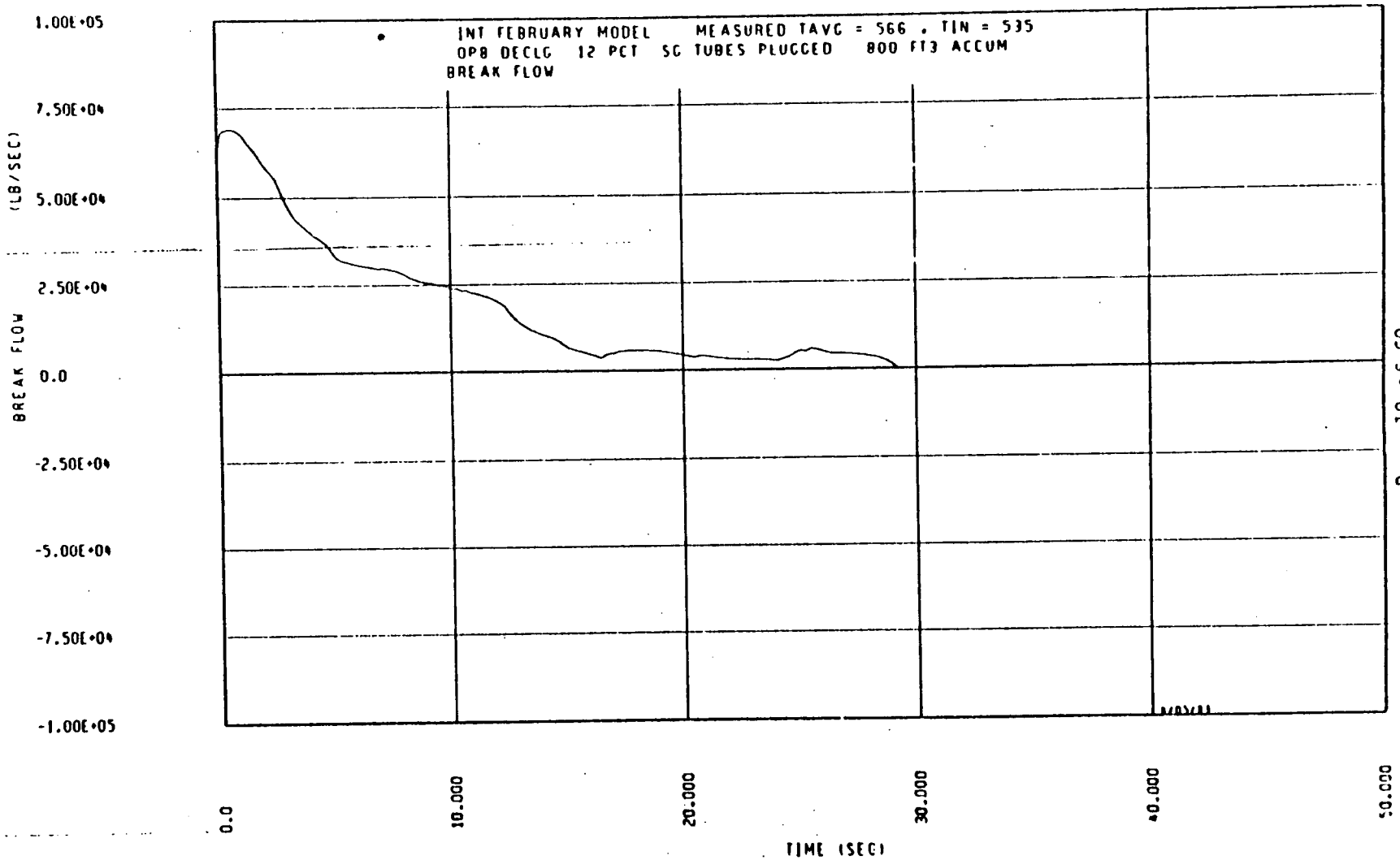


Figure 5A

CORE PRESSURE DROP - DECLG (C_D = 0.8)

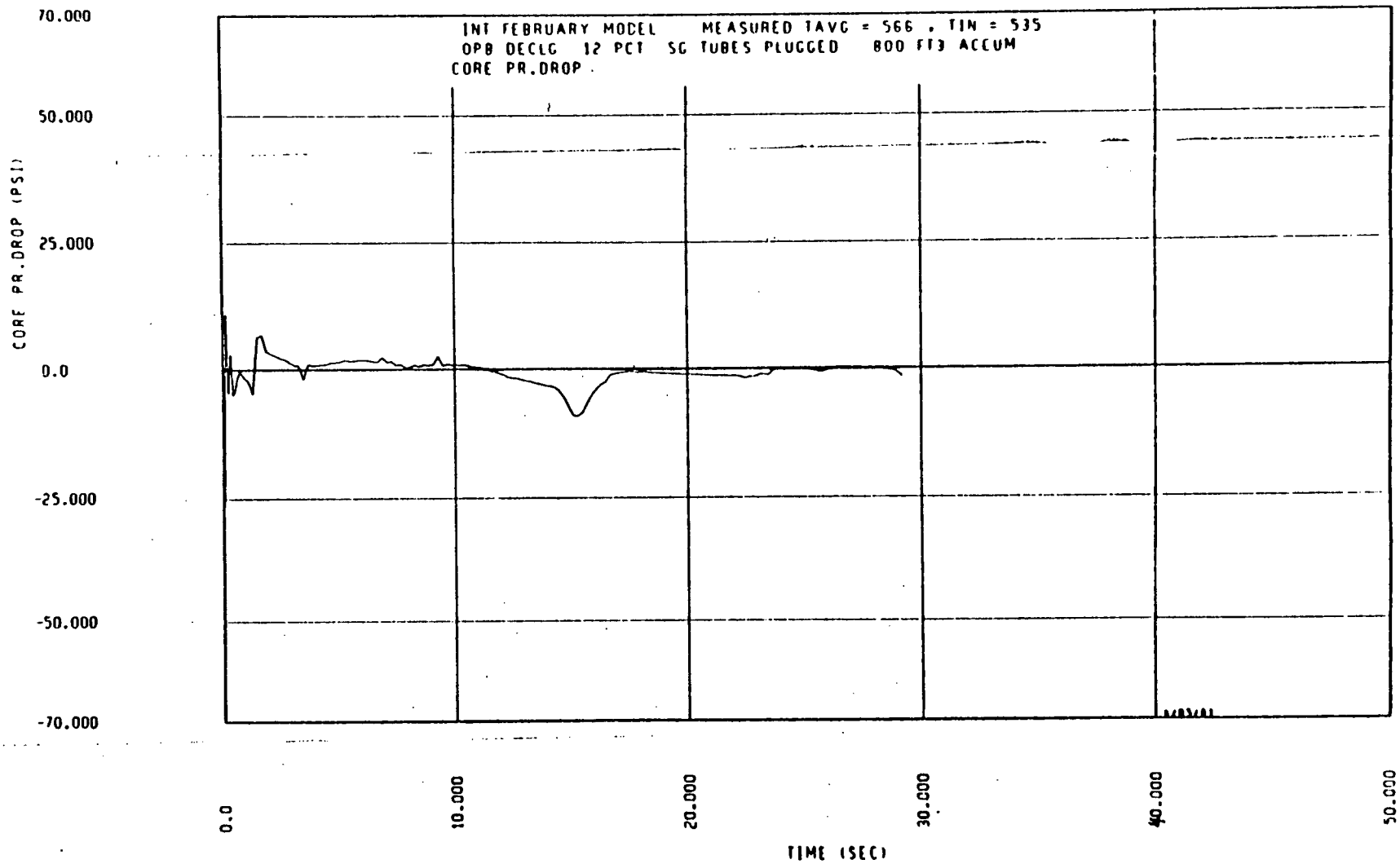


Figure 6A

PEAK CLAD TEMPERATURE - DECLG (C_D = 0.8)

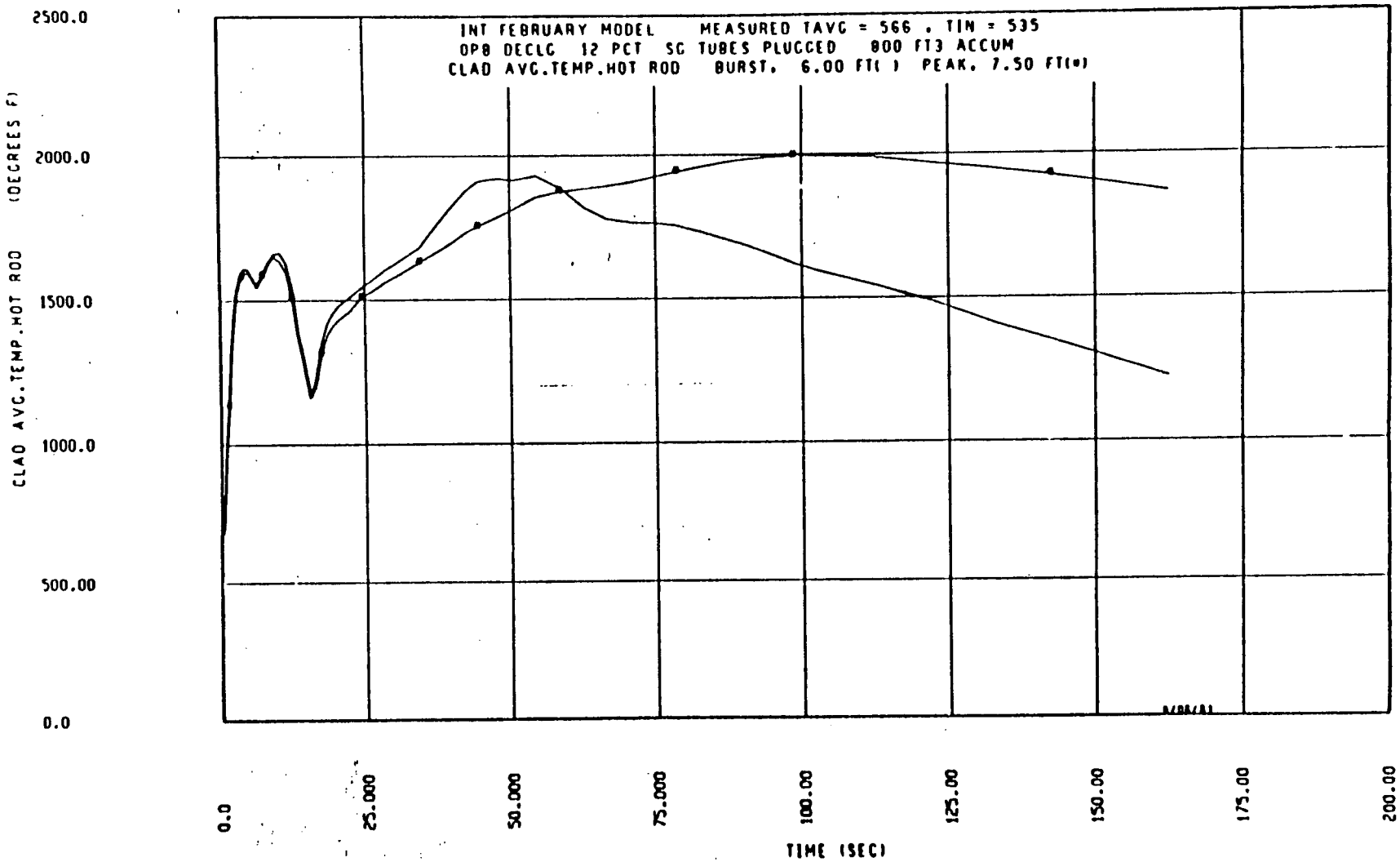


Figure 7A

FLUID TEMPERATURE - DECLG (C_D = 0.8)

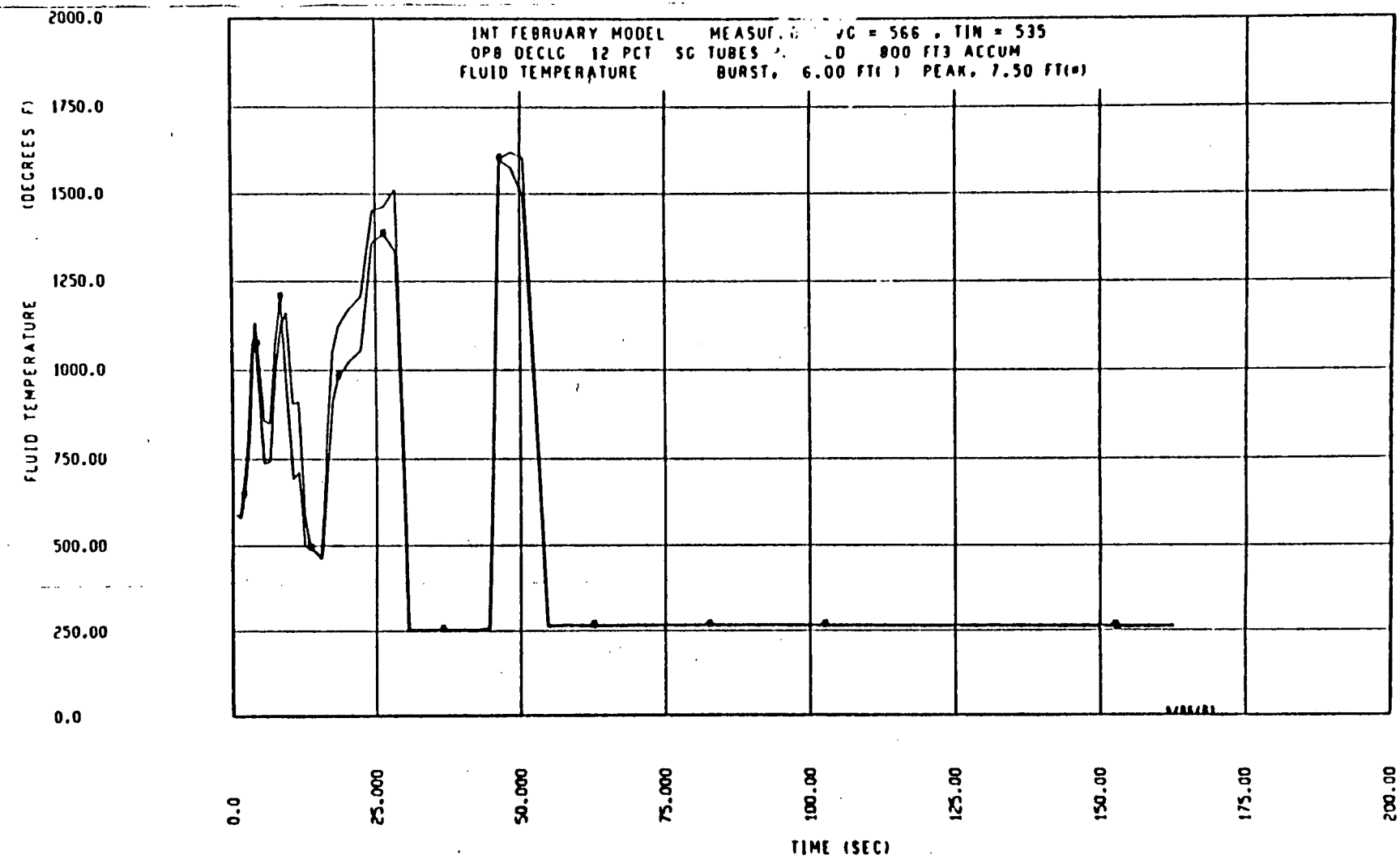


Figure 8A

CORE FLOW RATE - DECLG

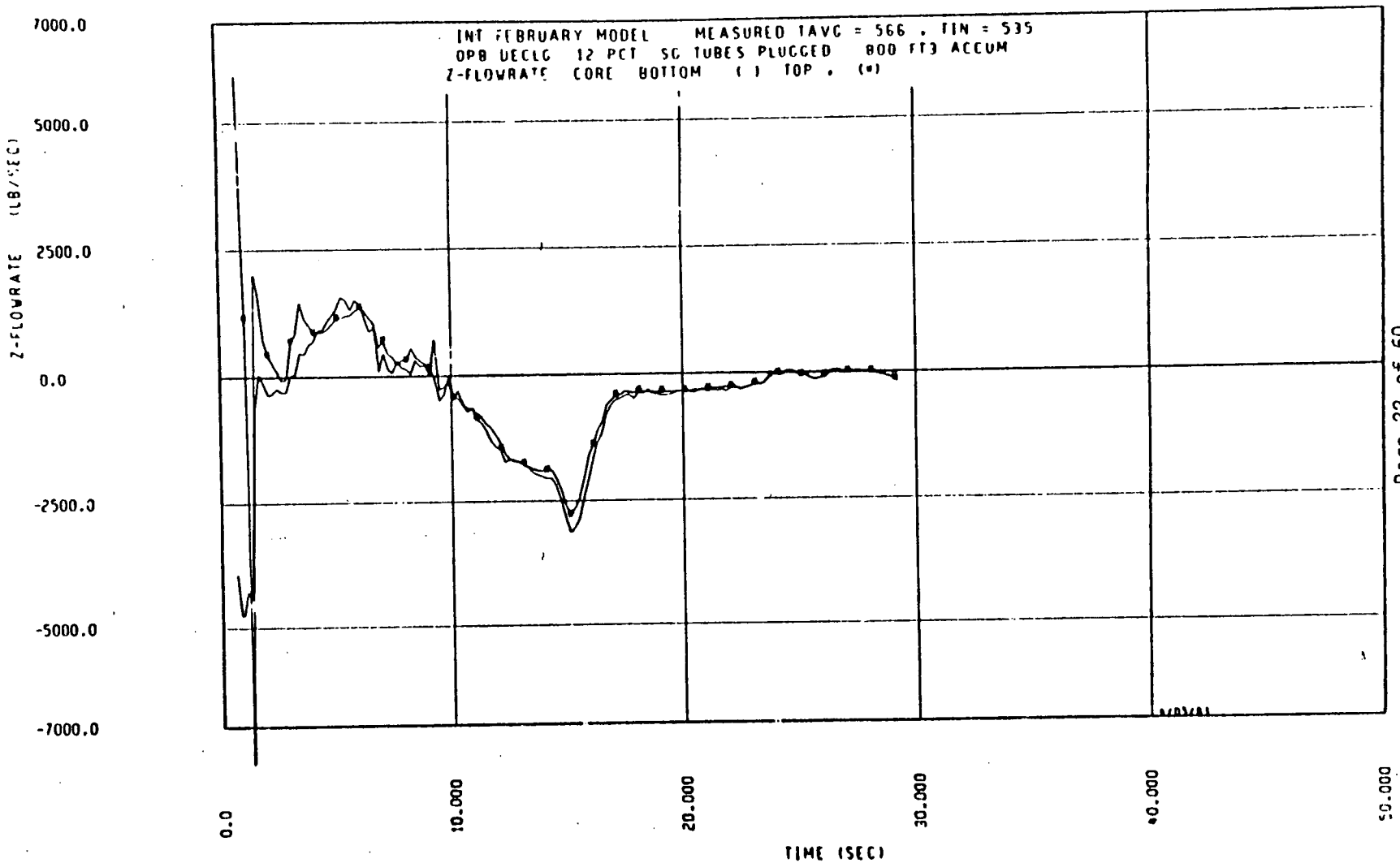


Figure 9A

REFLOOD TRANSIENT - DECLG (C_D = 0.8)

DOWNCOMER AND CORE WATER LEVELS

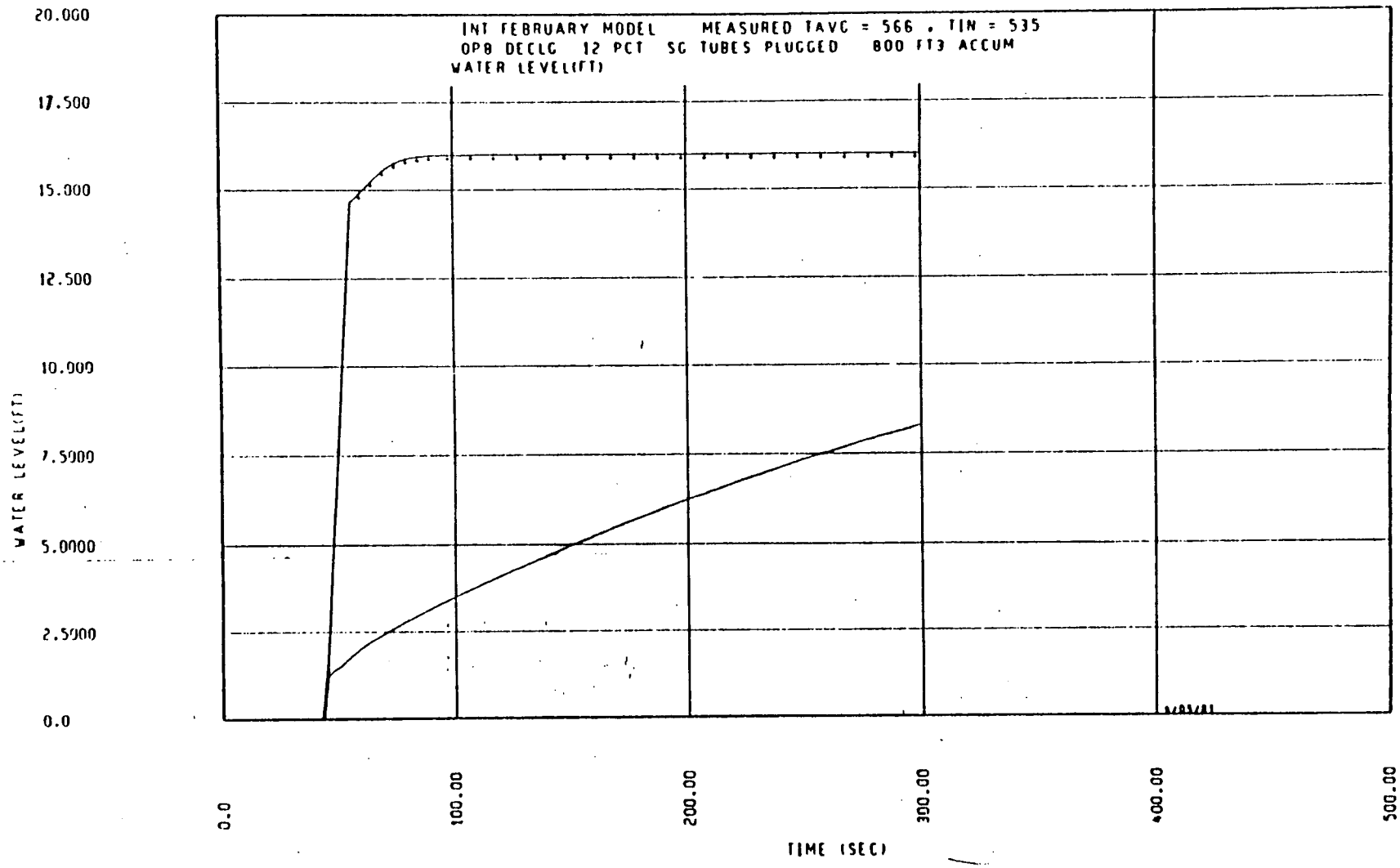


Figure 10A

REFLOOD TRANSIENT - DECLG (C_D = 0.8)

CORE INLET VELOCITY

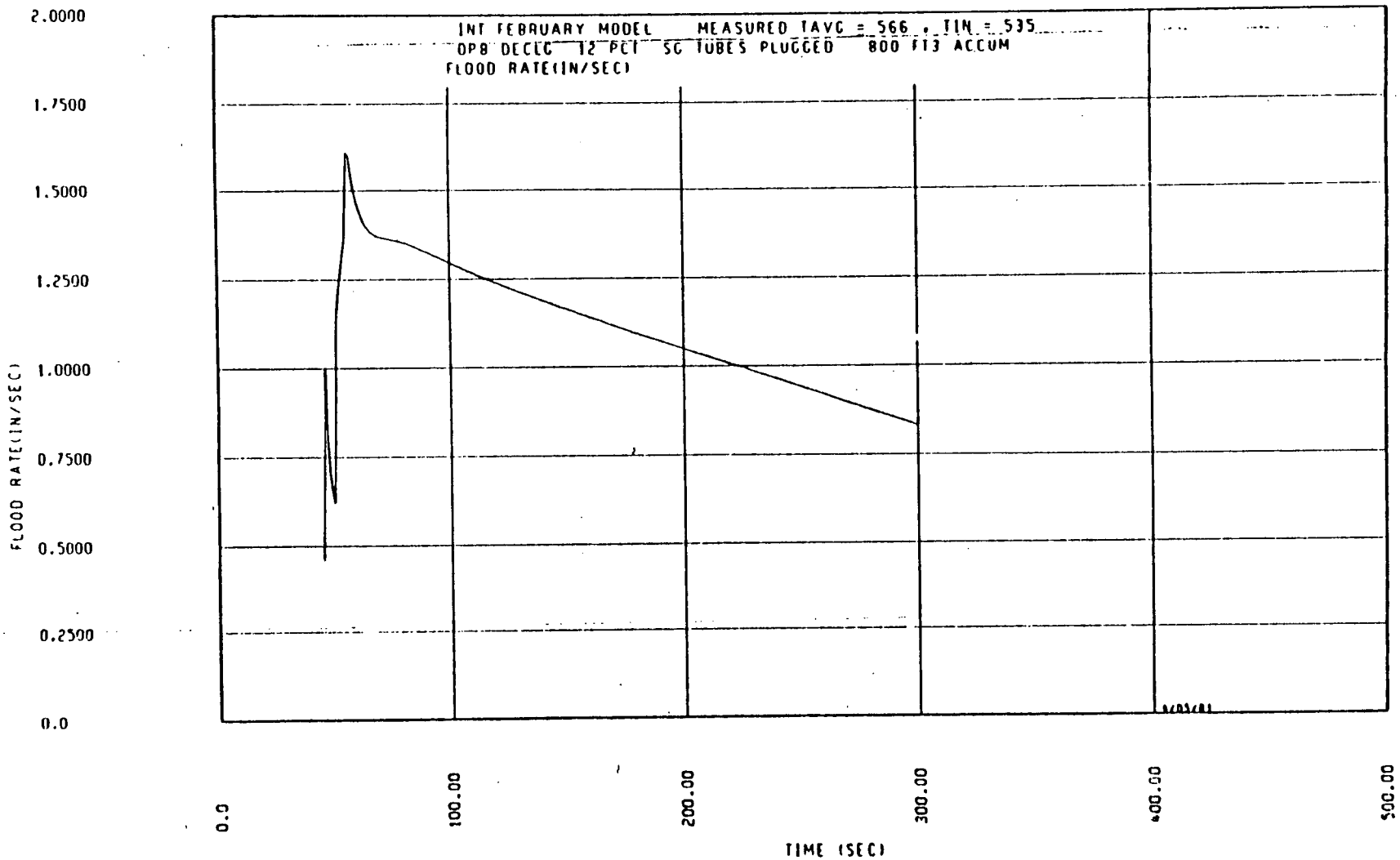


Figure 11A

ACCUMULATOR FLOW (BLOWDOWN) - DECLG (C_D = 0.8)

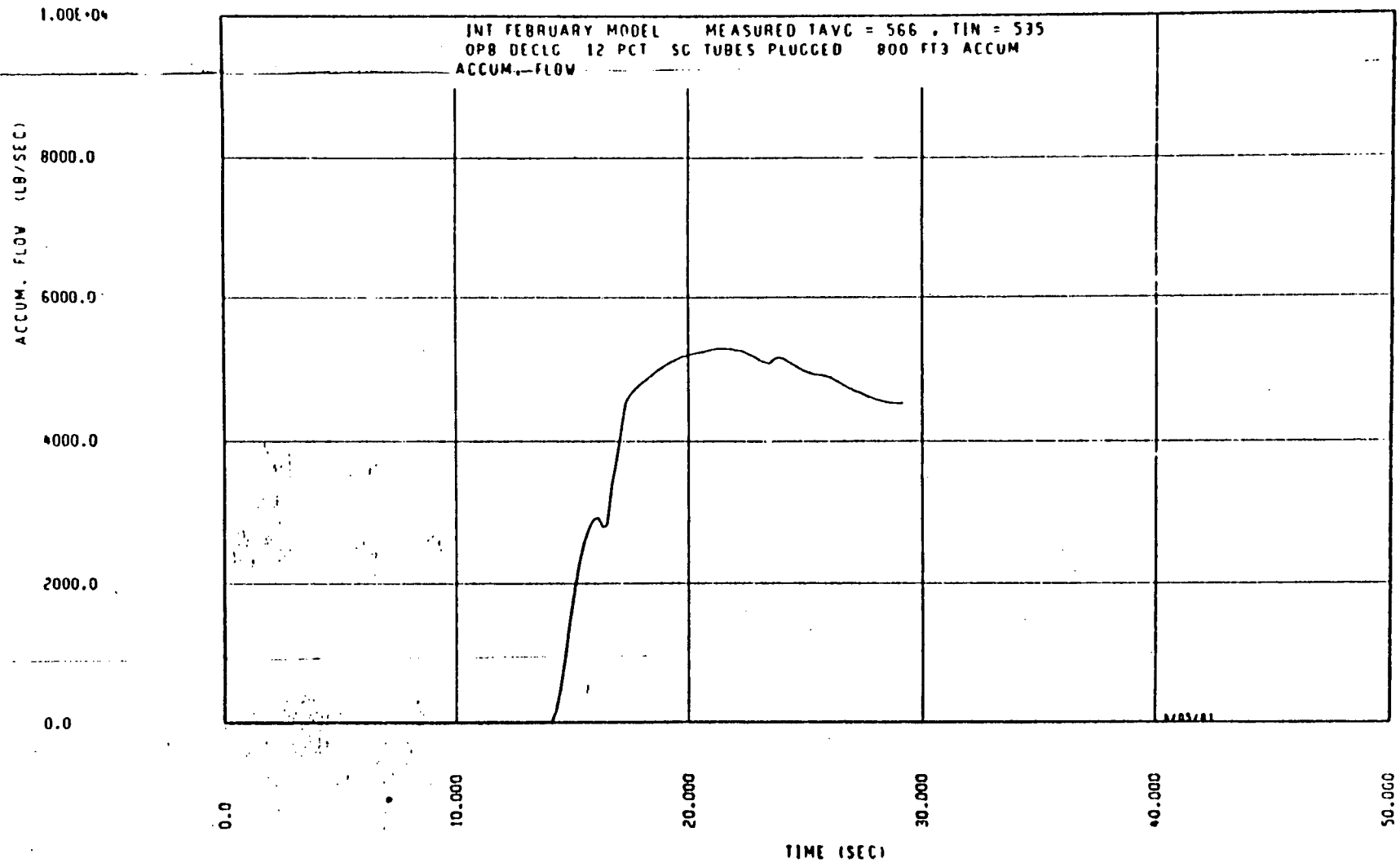


Figure 12A

K+E 10 X 10 TO THE CENTIMETER 46 1510
18 X 25 CM. MADE IN U.S.A.
KEUFFEL & ESSER CO.

$C_D = 0.8$

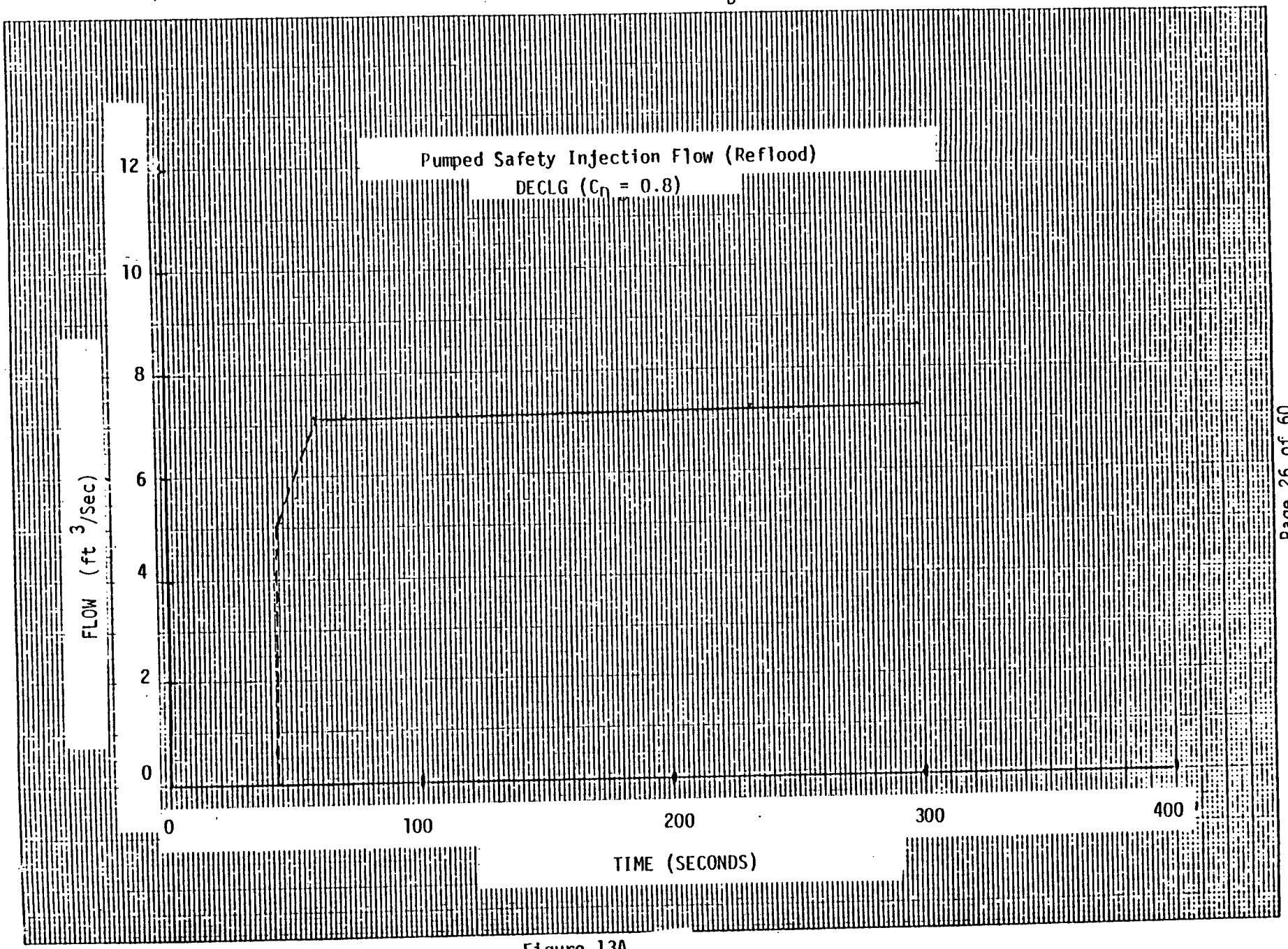


Figure 13A

SEMI-LOGARITHMIC 46 6012
4 CYCLES X 70 DIVISIONS
KEUFFEL & ESSER CO.

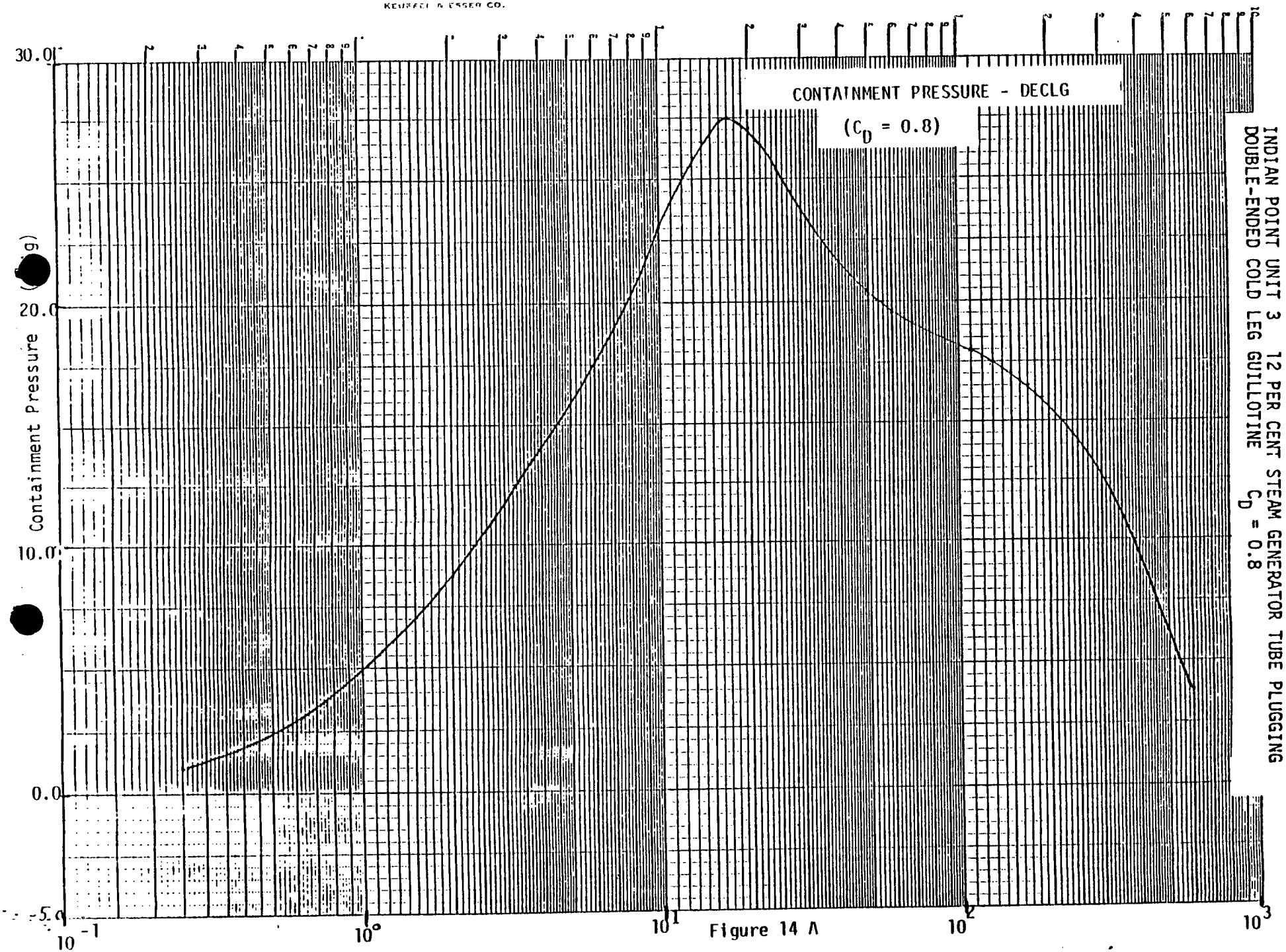


Figure 14 A

INDIAN POINT UNIT 3 12 PER CENT STEAM GENERATOR TUBE PLUGGING
DOUBLE-ENDED COLD LEG GUILLOTINE C_D = 0.8

CORE POWER TRANSIENT - DECLG (C_D = 0.8)

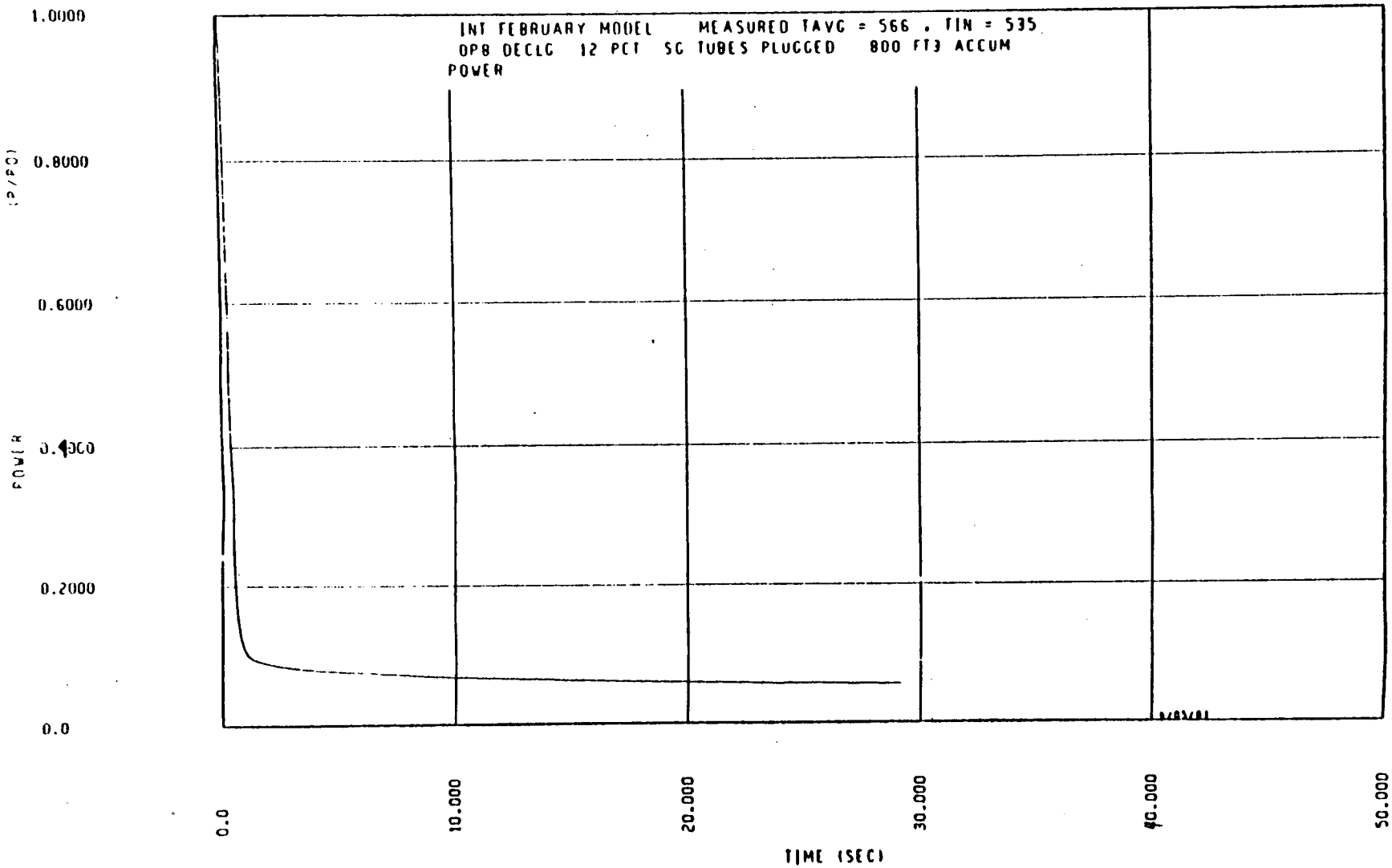


Figure 15A

FLUID QUALITY - DECLG (C_D = 1.0)

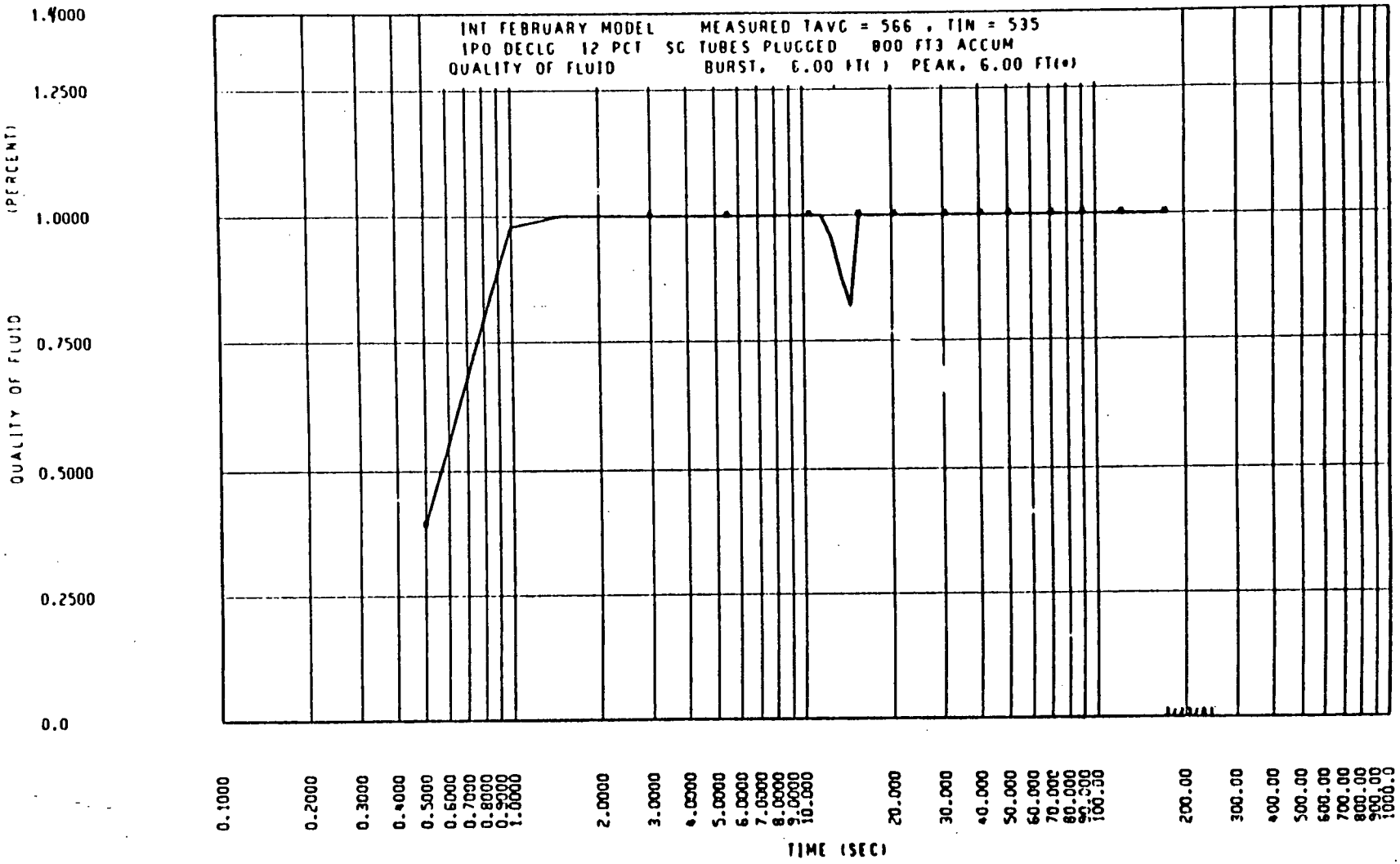


Figure 1B

MASS VELOCITY - DECLG (C_D = 1.0)

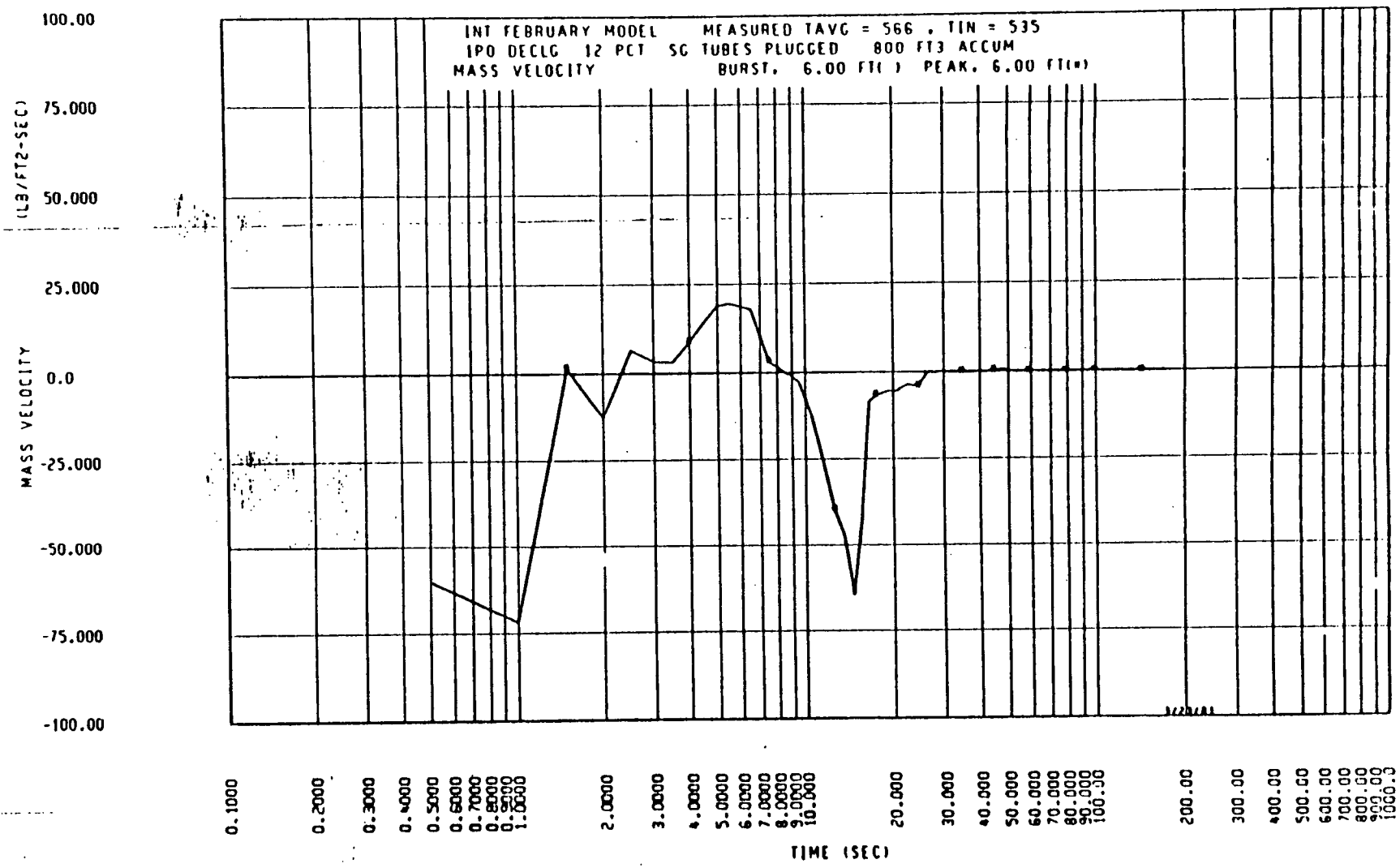


Figure 2B

HEAT TRANSFER COEFFICIENT - DECLG (C_D = 1.0)

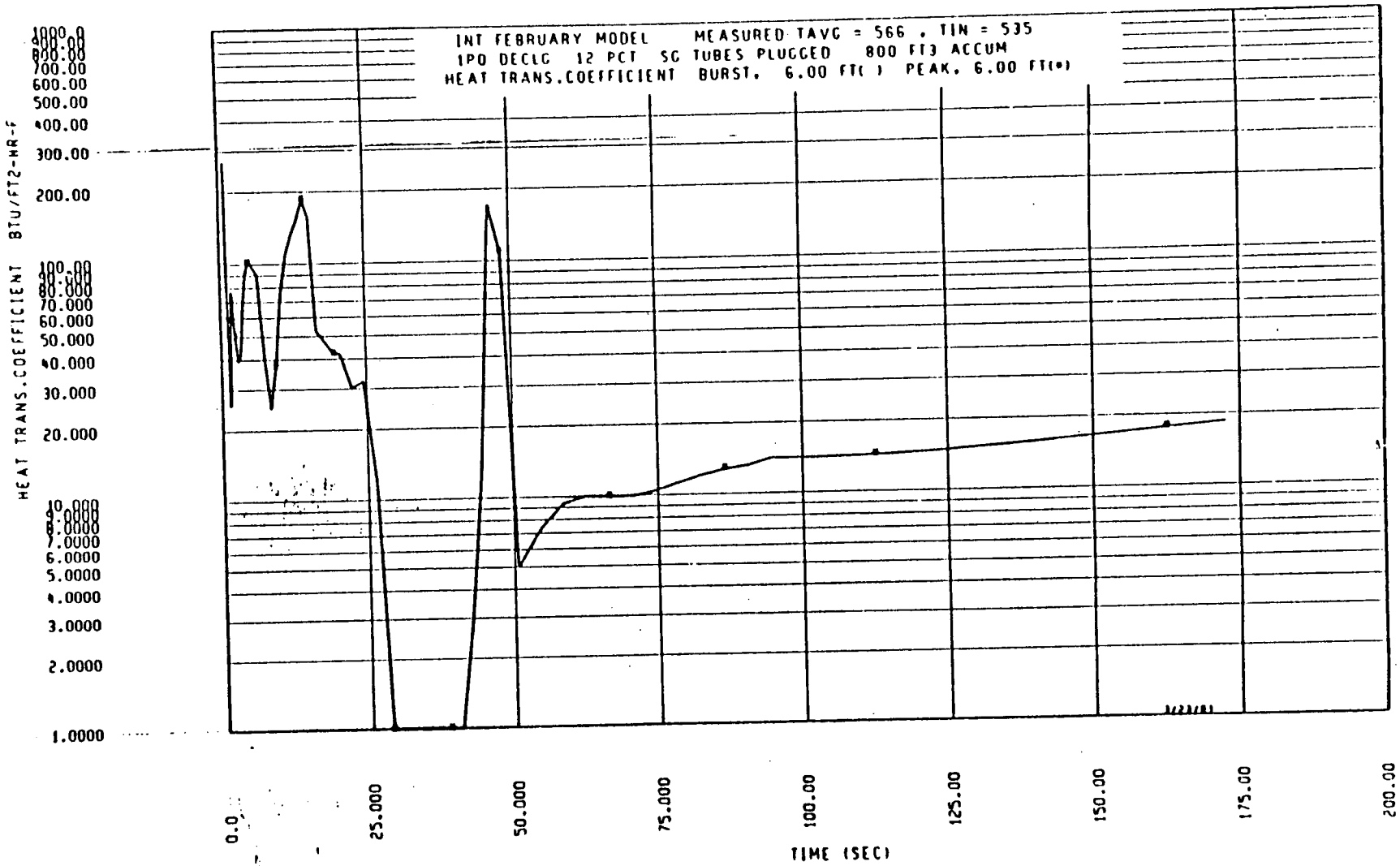


Figure 3B

CORE PRESSURE - DECLG (C_D = 1.0)

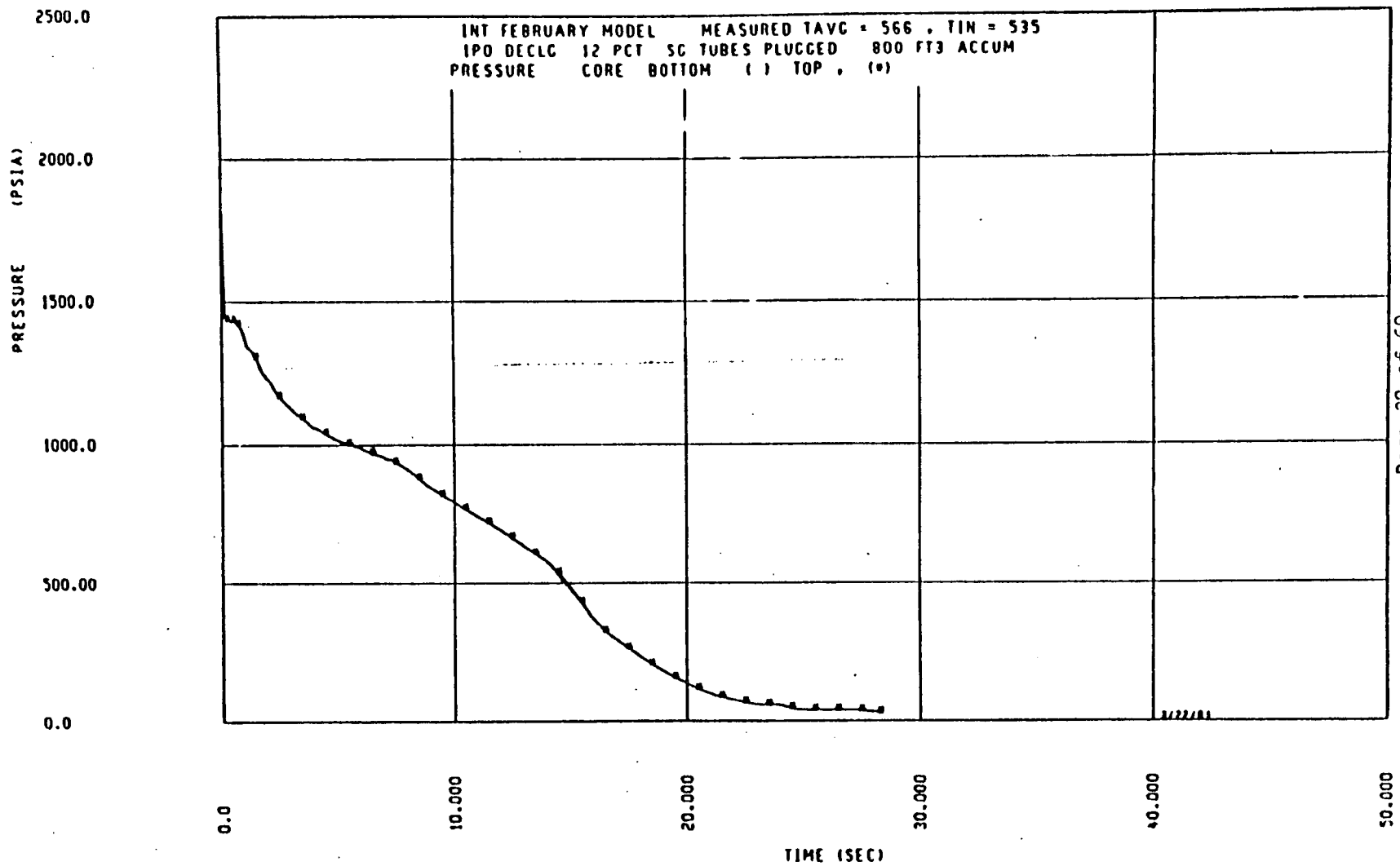


Figure 4B

BREAK FLOW RATE - DECLG (C_D = 1.0)

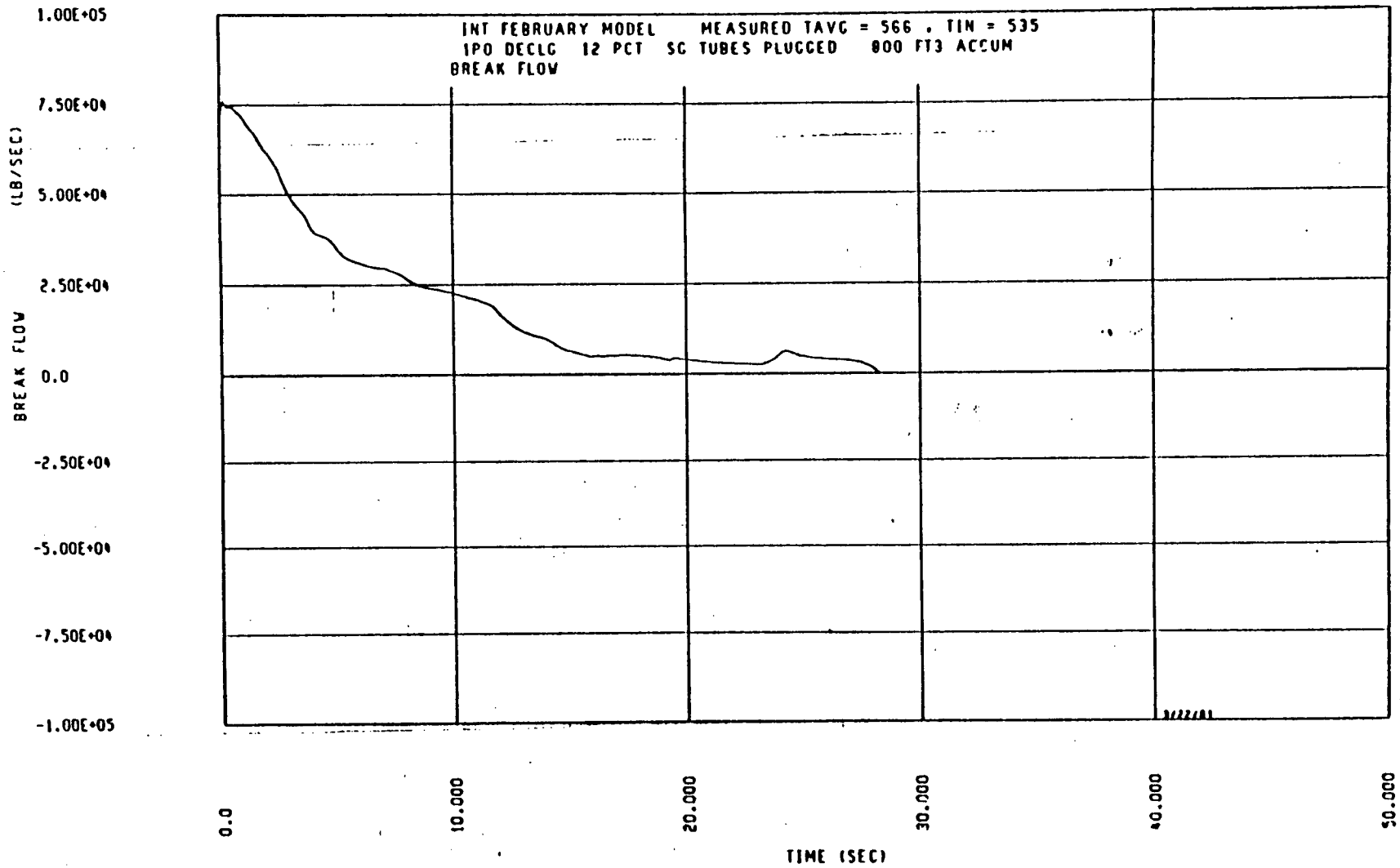


Figure 5B

CORE PRESSURE DROP - DECLG (C_D = 1.0)

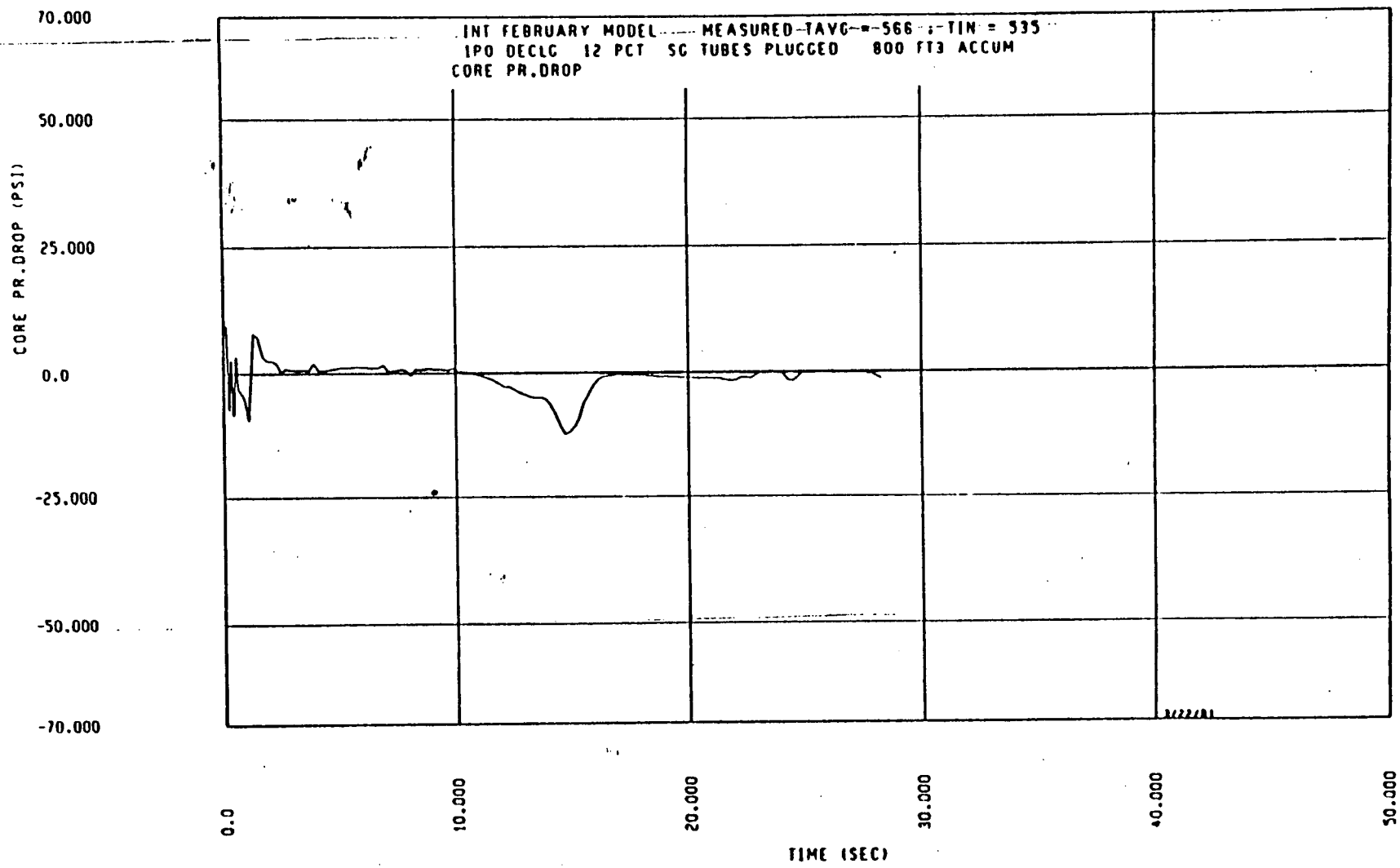


Figure 6B

PEAK CLAD TEMPERATURE - DECLG (C_D = 1.0)

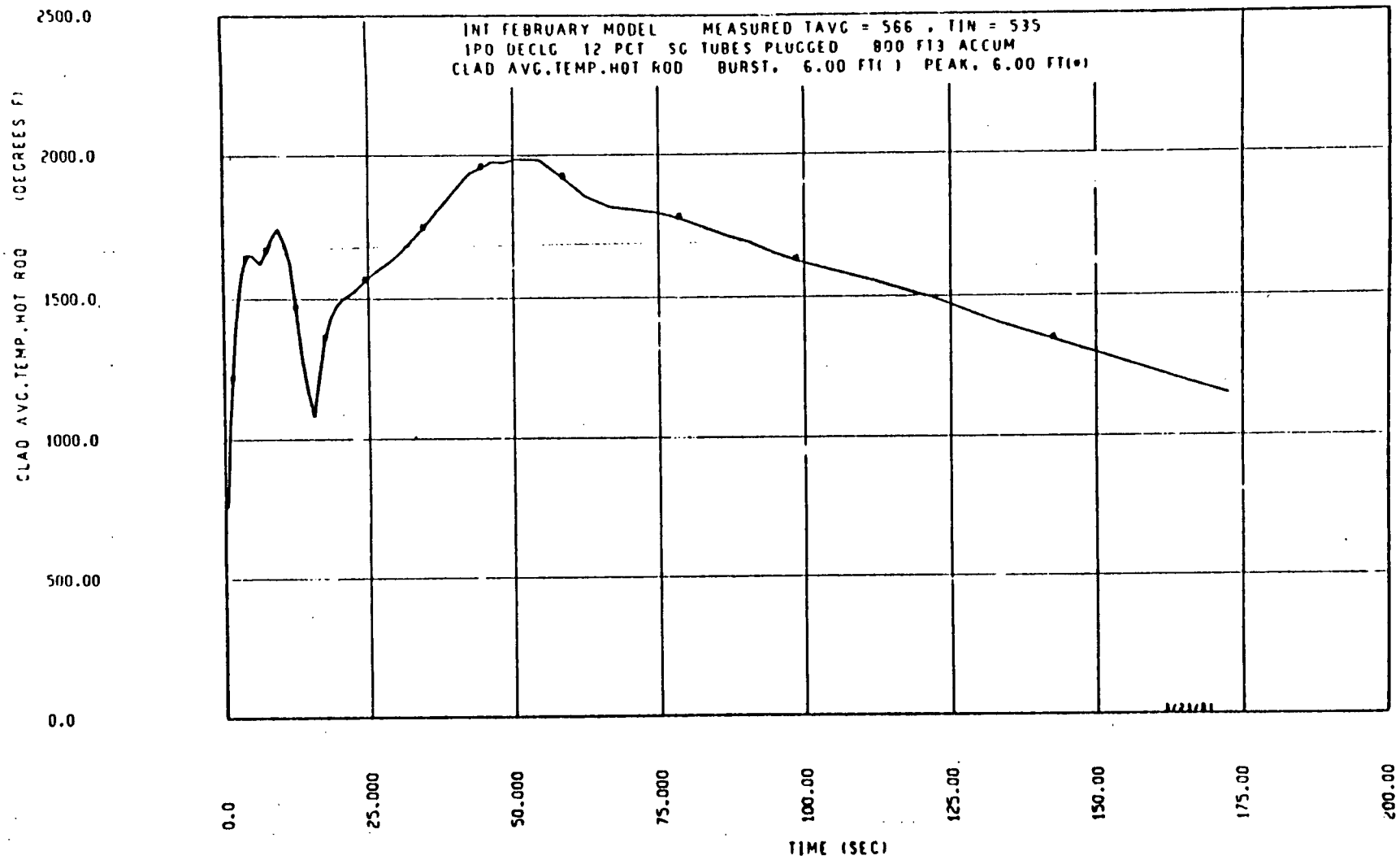


Figure 7B

FLUID TEMPERATURE - DECLG (C_D = 1.0)

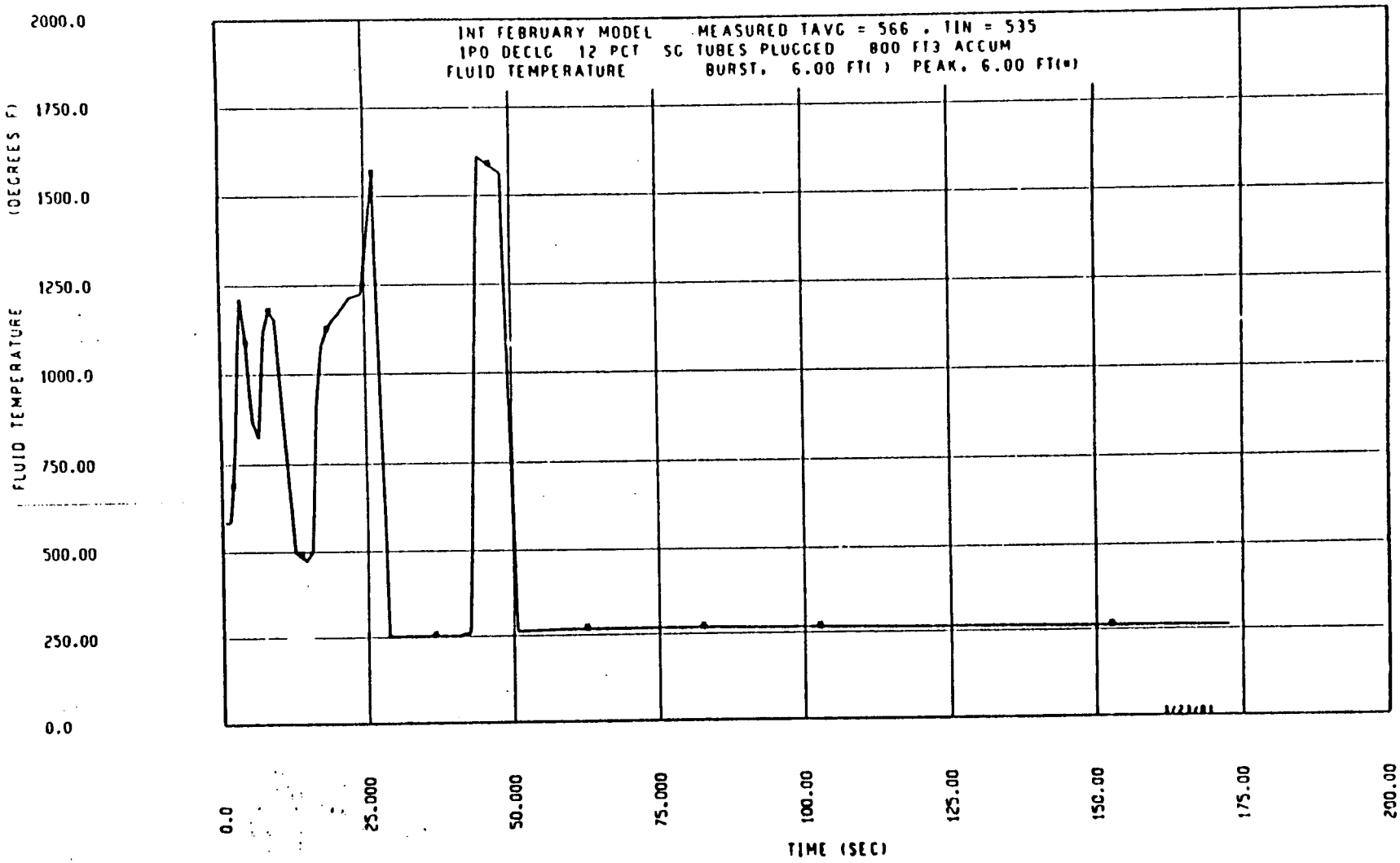


Figure 8B

CORE FLOW RATE - DECLG (C_D = 1.0)

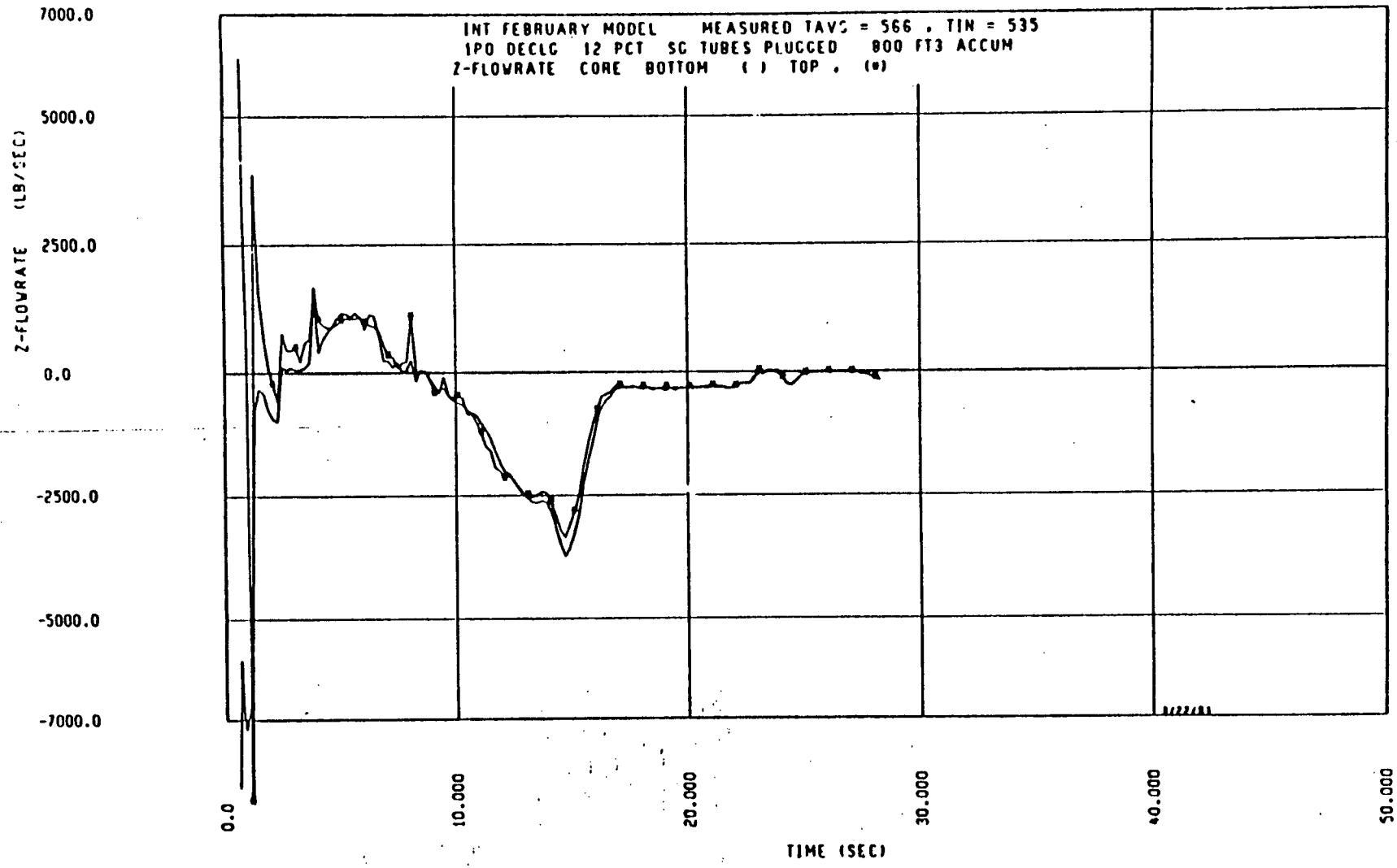


Figure 9B

REFLOOD TRANSIENT - DECLG (C_D = 1.0)

DOWNCOMER AND CORE WATER LEVELS

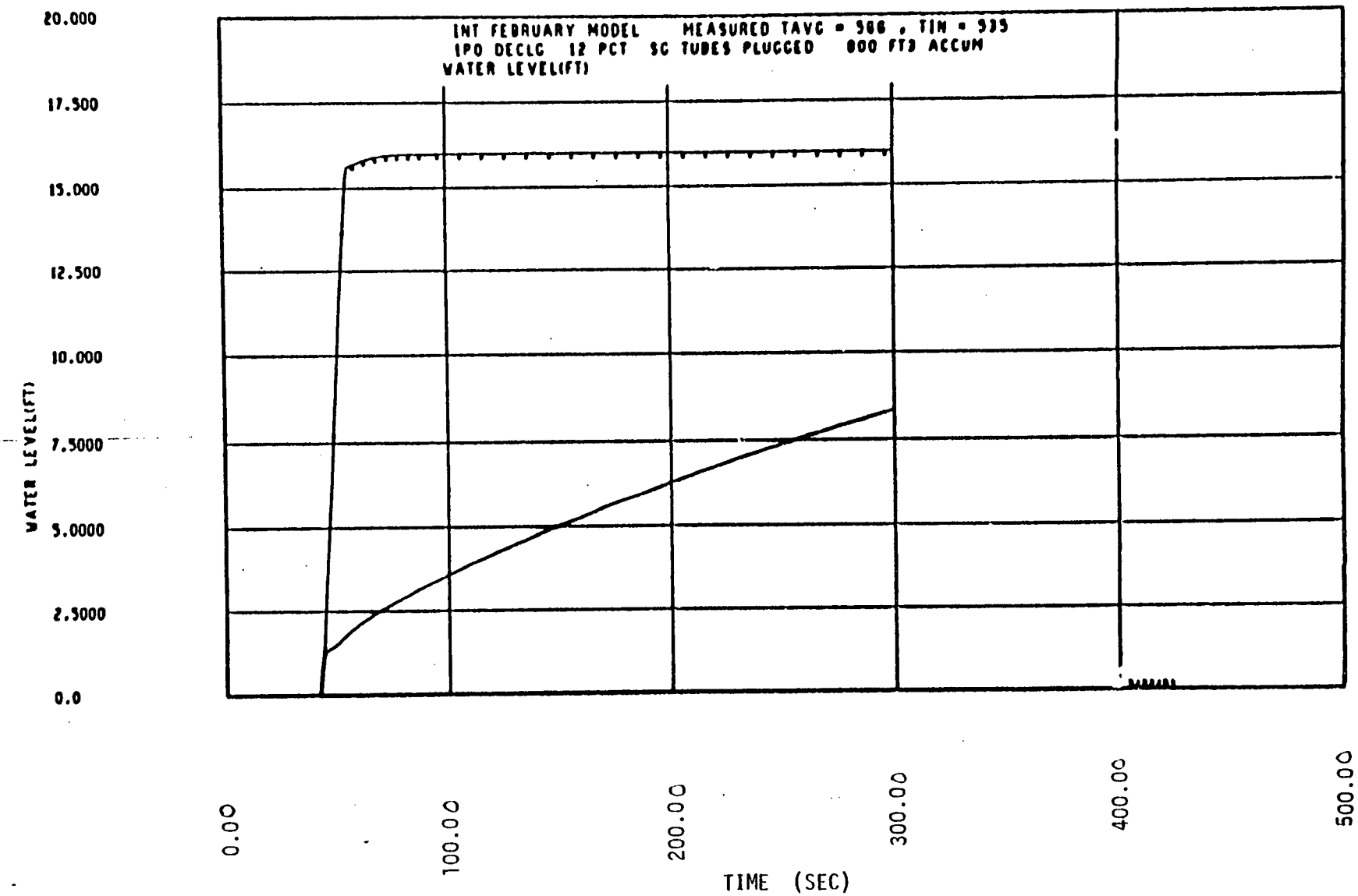


Figure 10B

REFLOOD TRANSIENT - DECLG (C_D = 1.0)

CORE INLET - VELOCITY

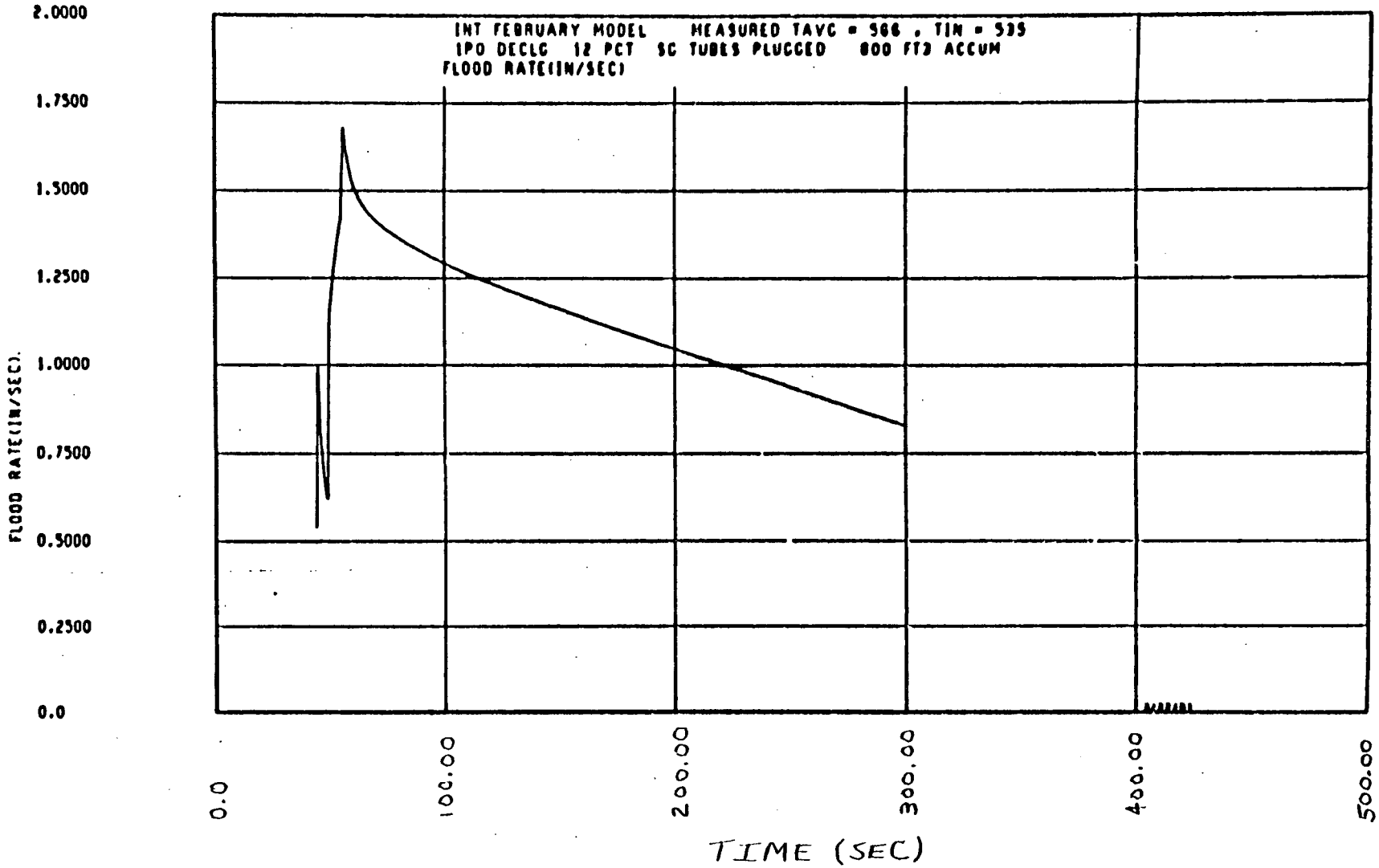


Figure 11B

ACCUMULATOR FLOW (BLOWDOWN) - DECLG (C_D = 1.0)

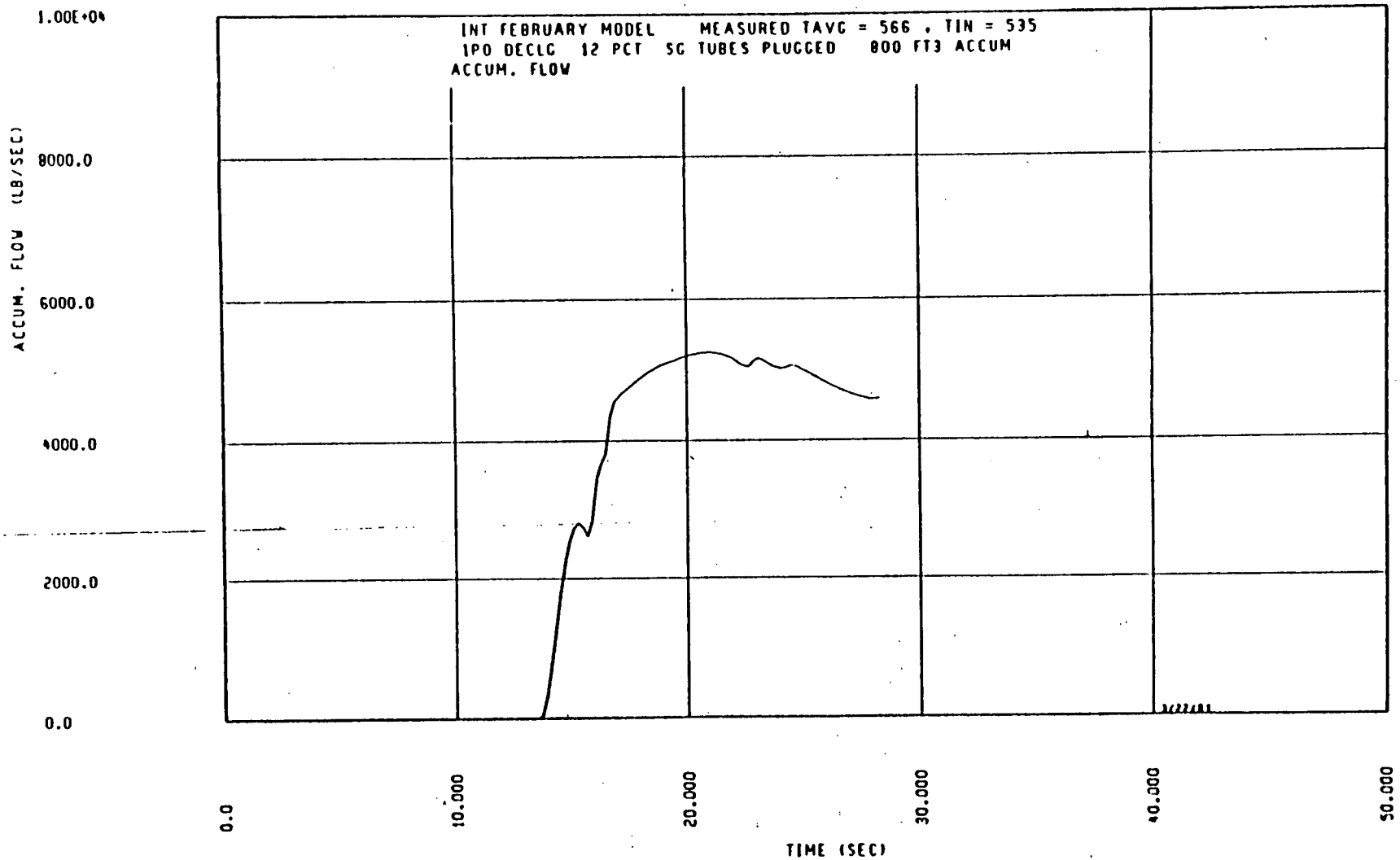


Figure 12B

K+E 10 X 10 TO THE CENTIMETER 46 1510
18 X 25 CM. MADE IN U.S.A.
KEUFFEL & ESSER CO.

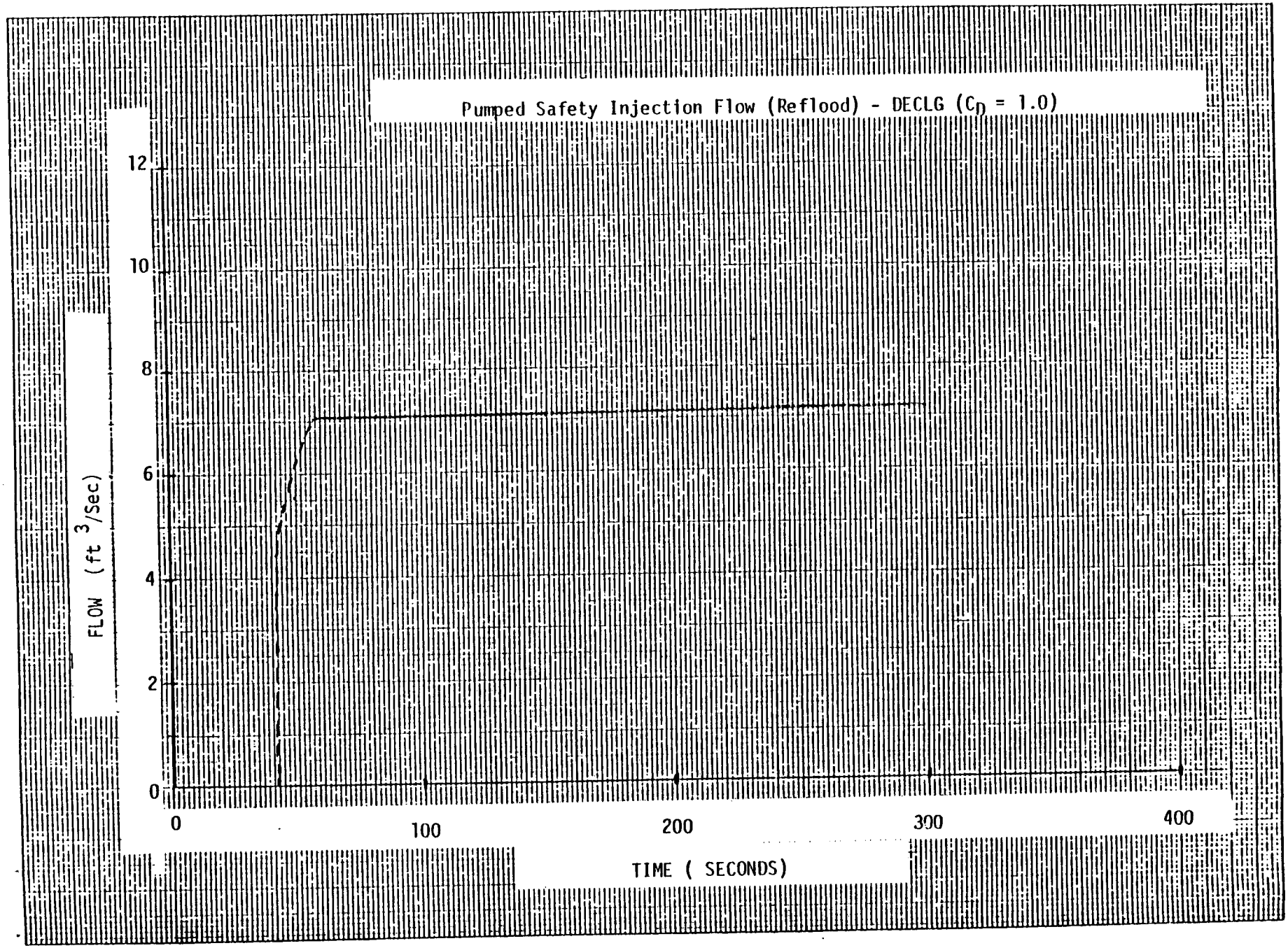
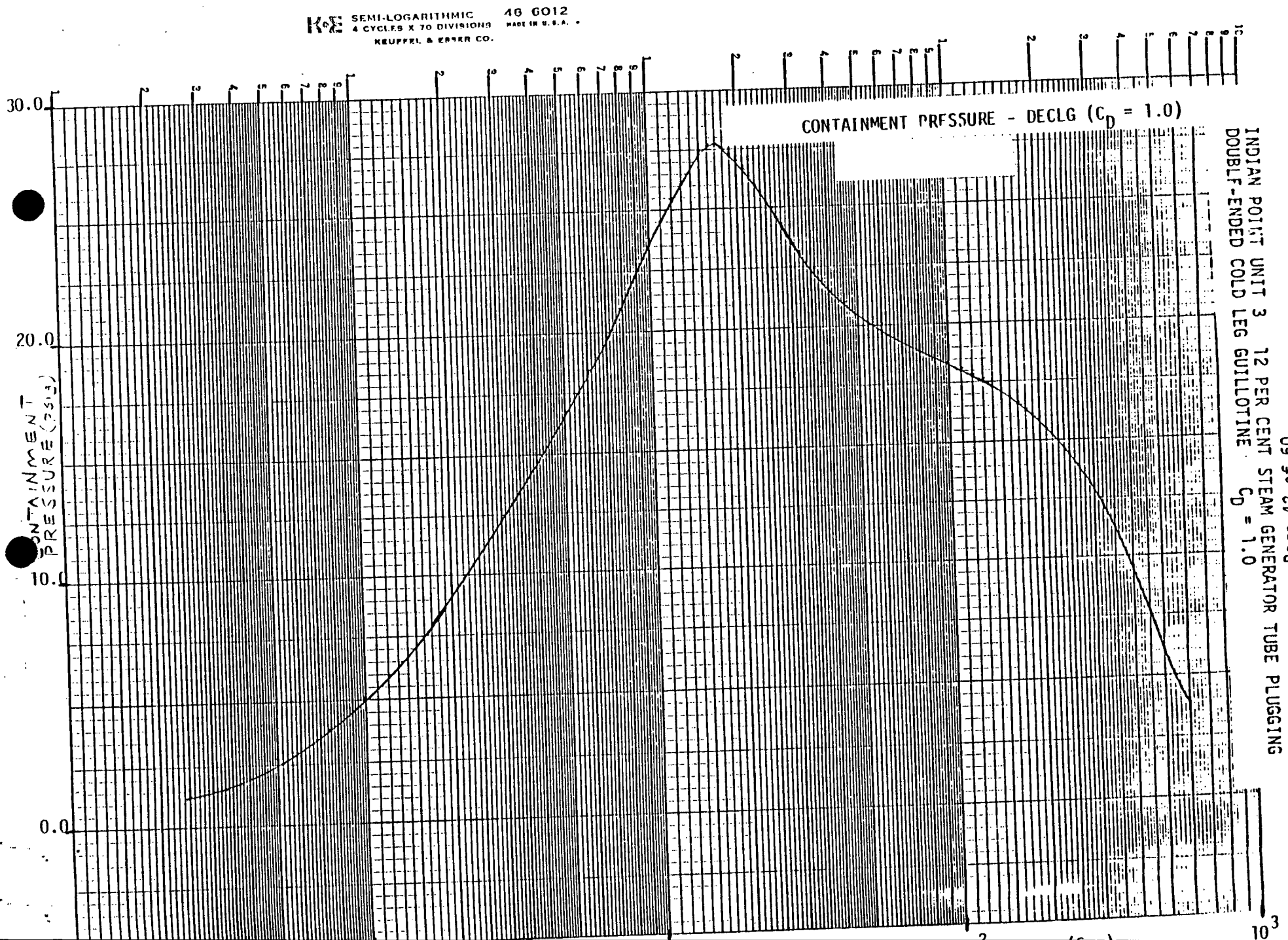


Figure 13B


K&E SEMI-LOGARITHMIC 46 6012
4 CYCLES X 70 DIVISIONS MADE IN U.S.A.
KEUFFEL & ESSER CO.
CONTAINMENT PRESSURE - DECLG (C_D = 1.0)
CONTAINMENT PRESSURE (PSIA)
30.0
20.0
10.0
0.0
10^3

CORE POWER TRANSIENT - DECLG (C_D = 1.0)

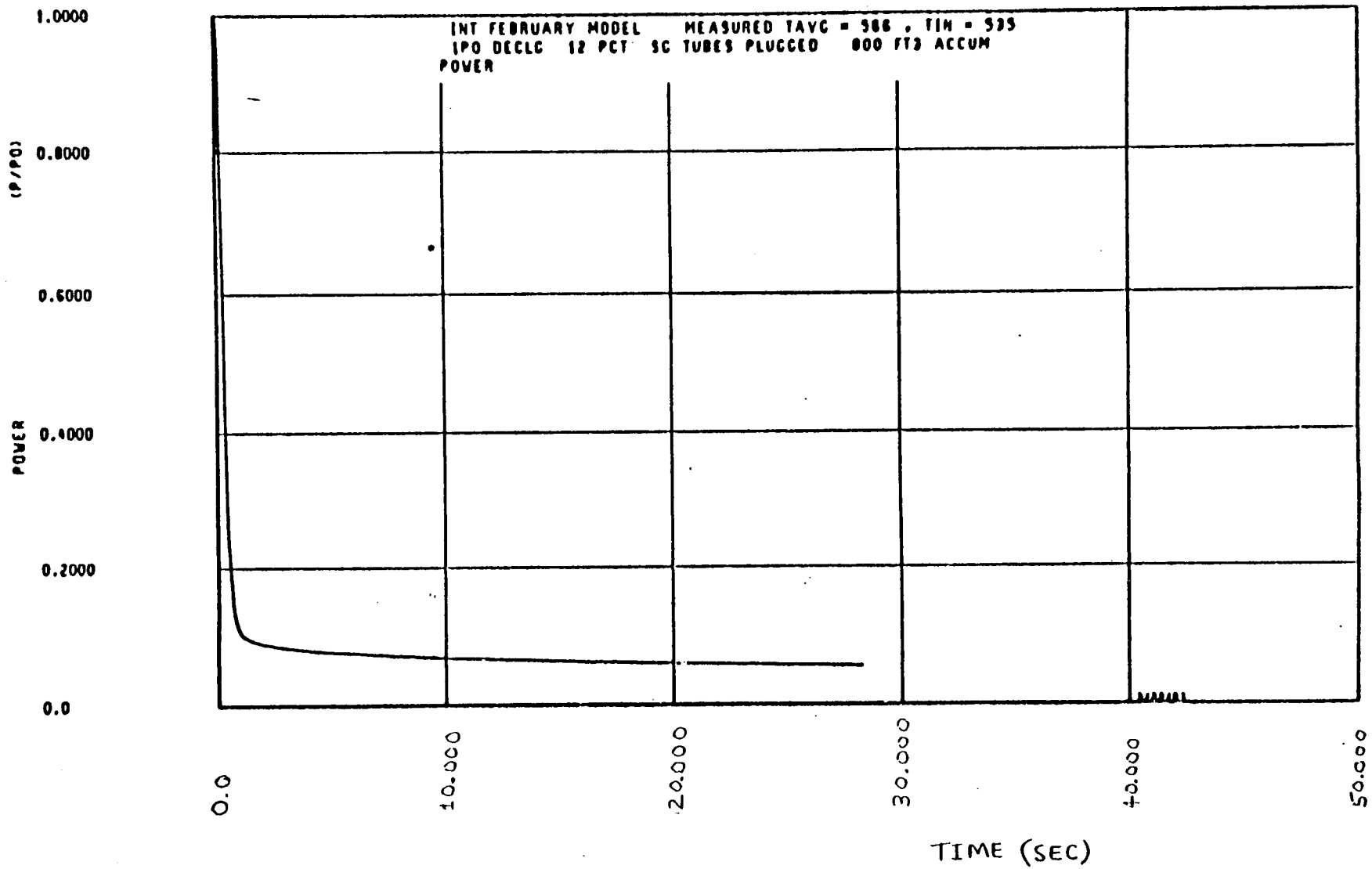


Figure 15B

FLUID QUALITY - DECLG (C_D = 0.6)

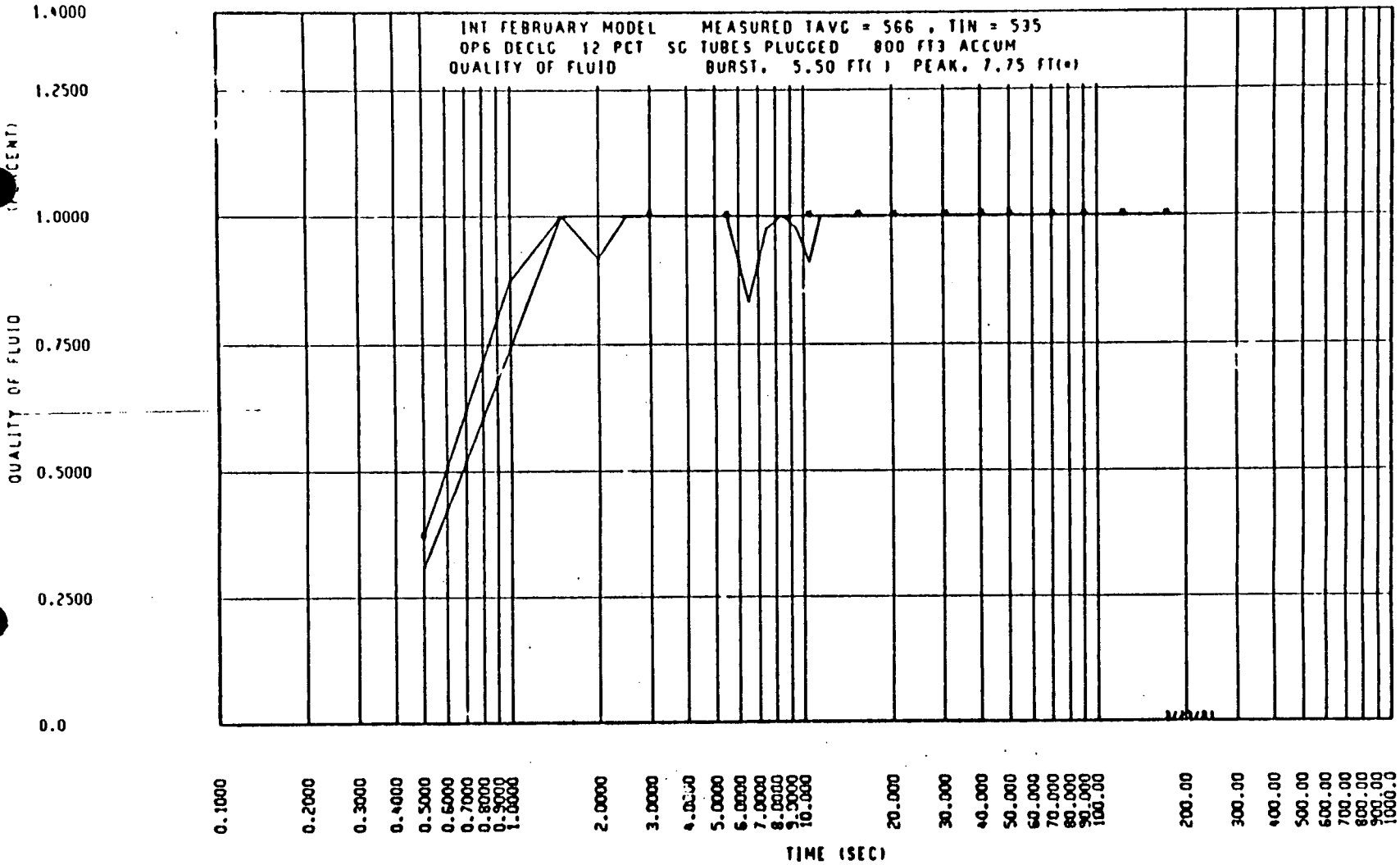


Figure 1C

MASS VELOCITY - DECLG (C_D = 0.6)

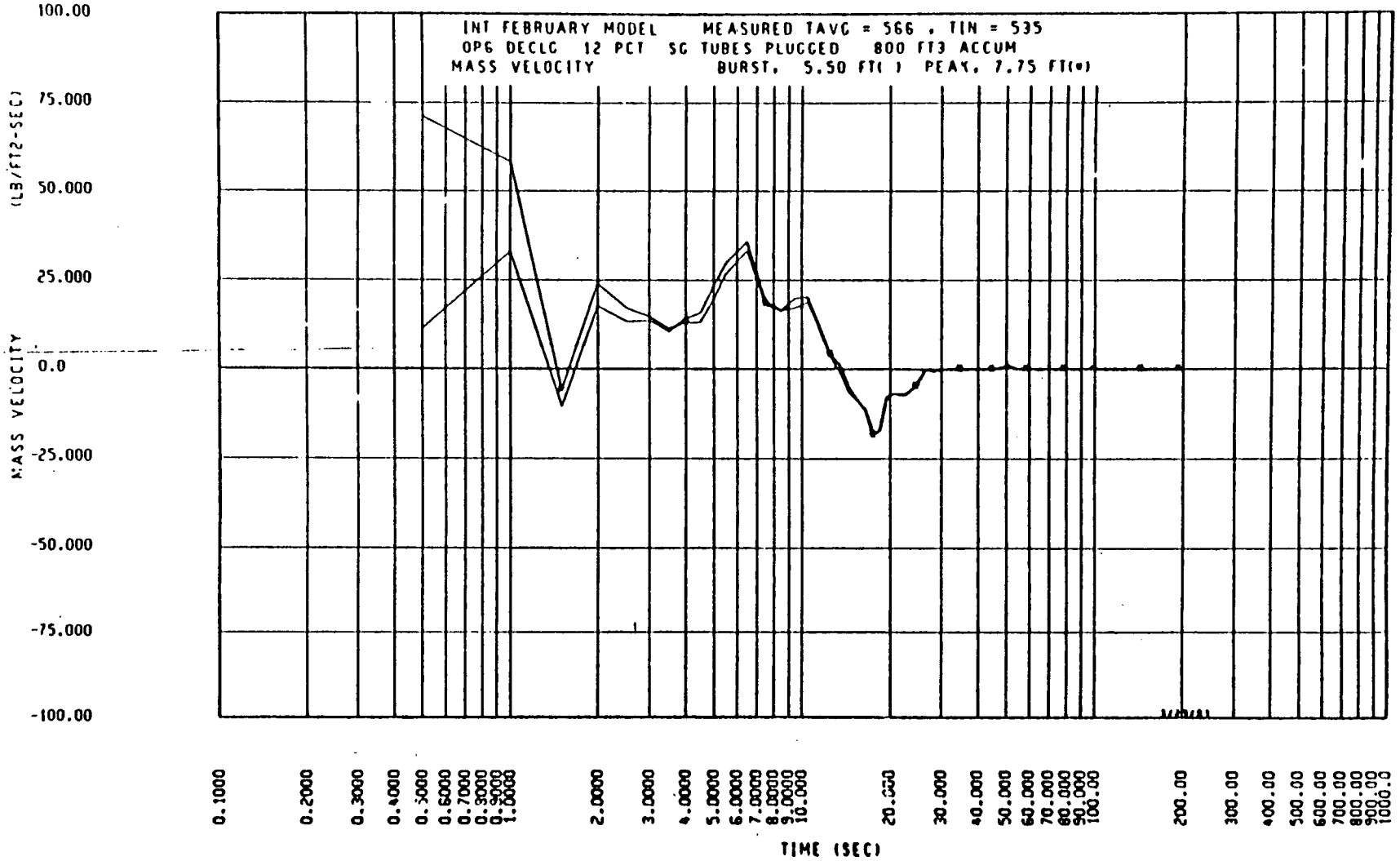


Figure 2C

HEAT TRANSFER COEFFICIENT - DECLG (C_D = 0.6)

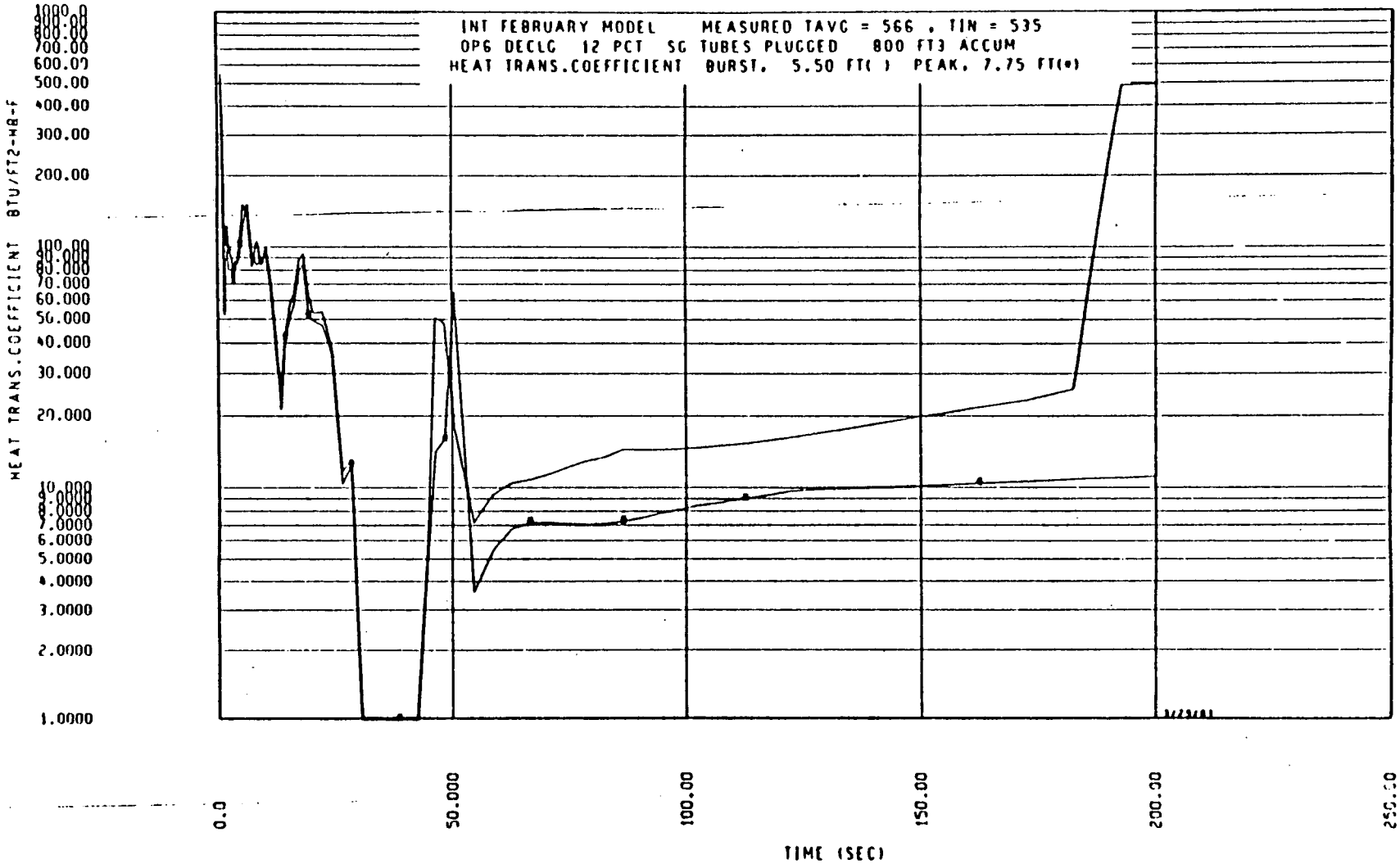


Figure 3C

CORE PRESSURE - DECLG (C_D = 0.6)

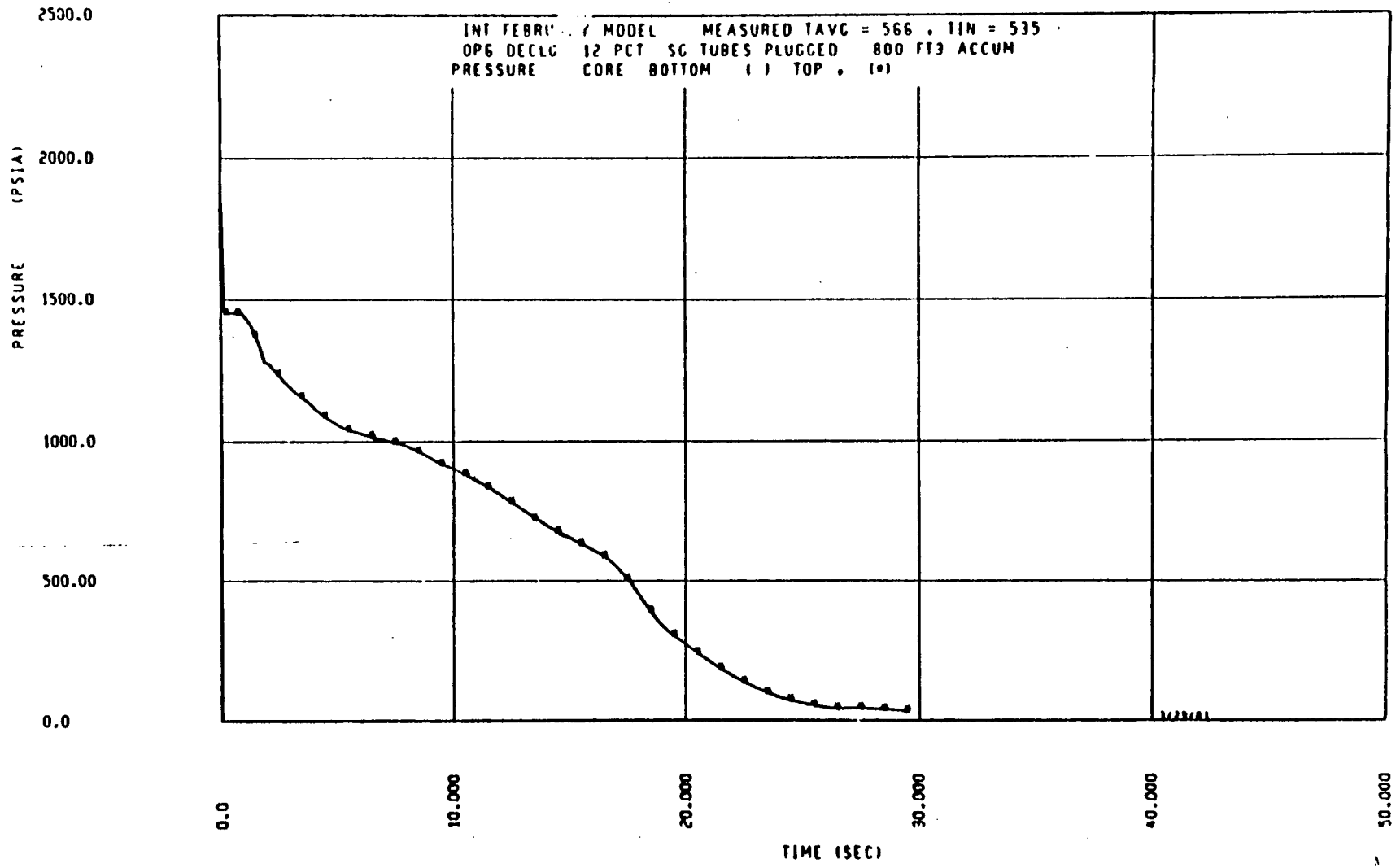


Figure 4C

BREAK FLOW RATE - DECLG (C_D = 0.6)

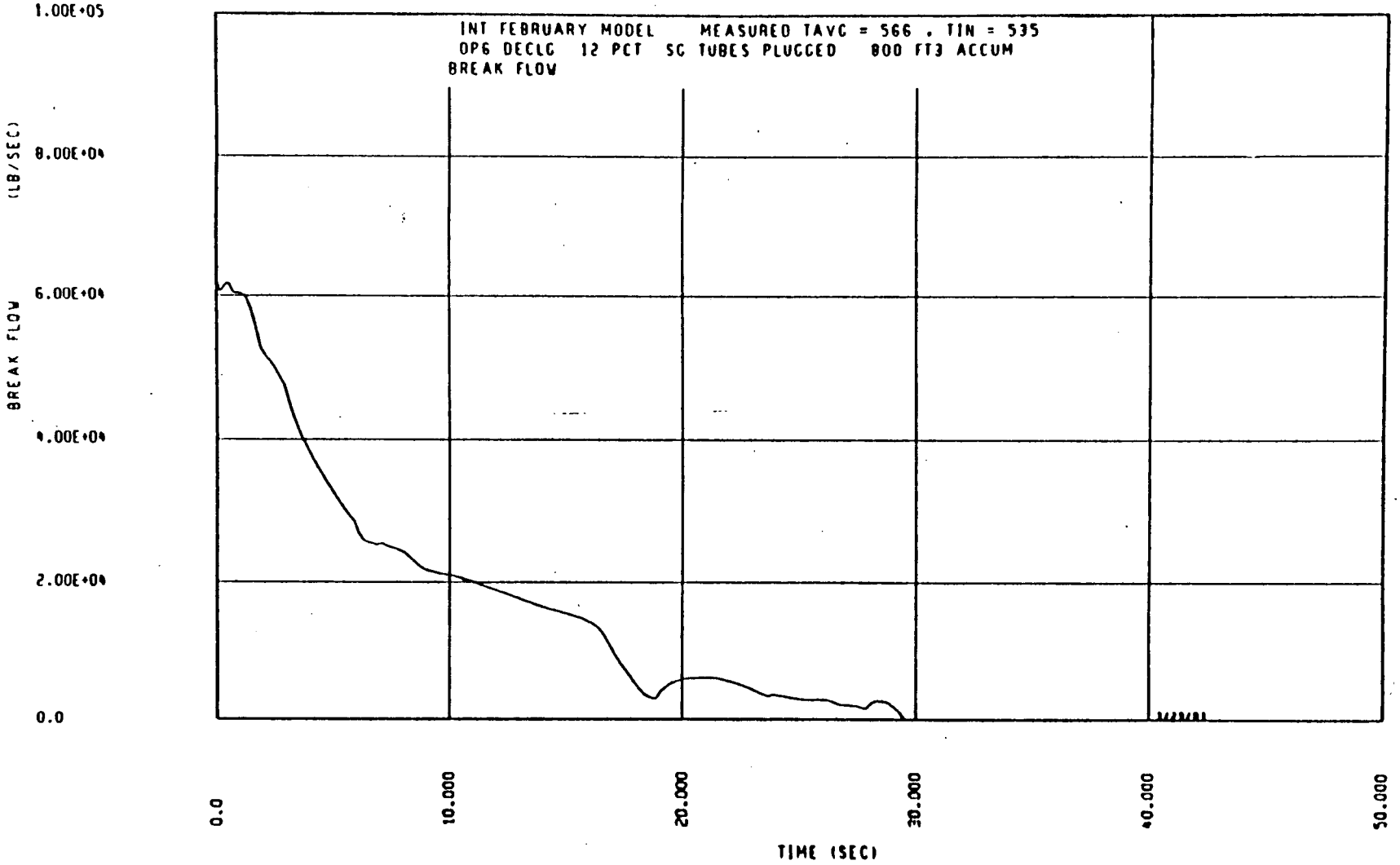


Figure 5C

CORE PRESSURE DROP - DECLG (C_D = 0.6)

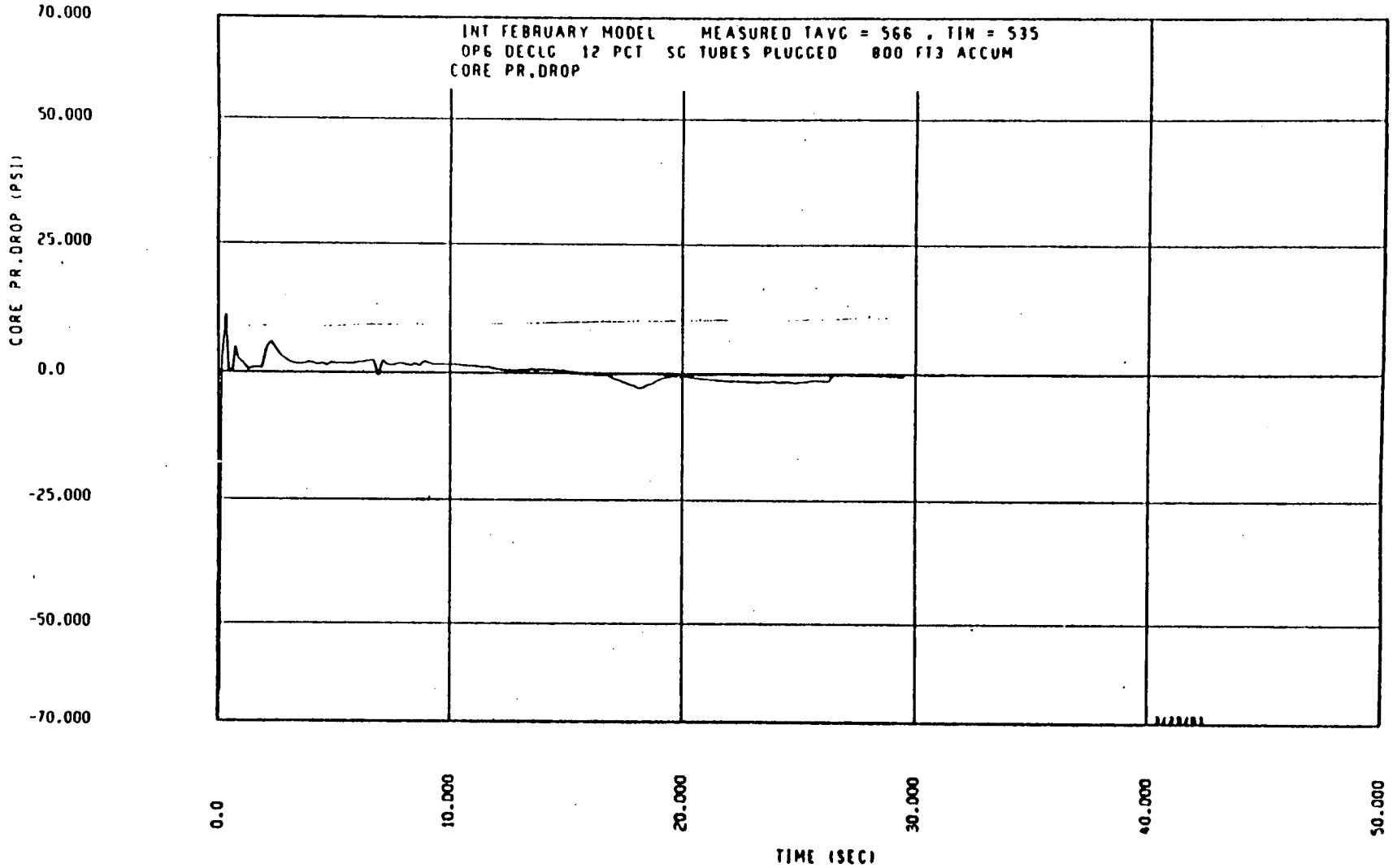


Figure 6C

PEAK CLAD TEMPERATURE - DECLG (C_D = 0.6)

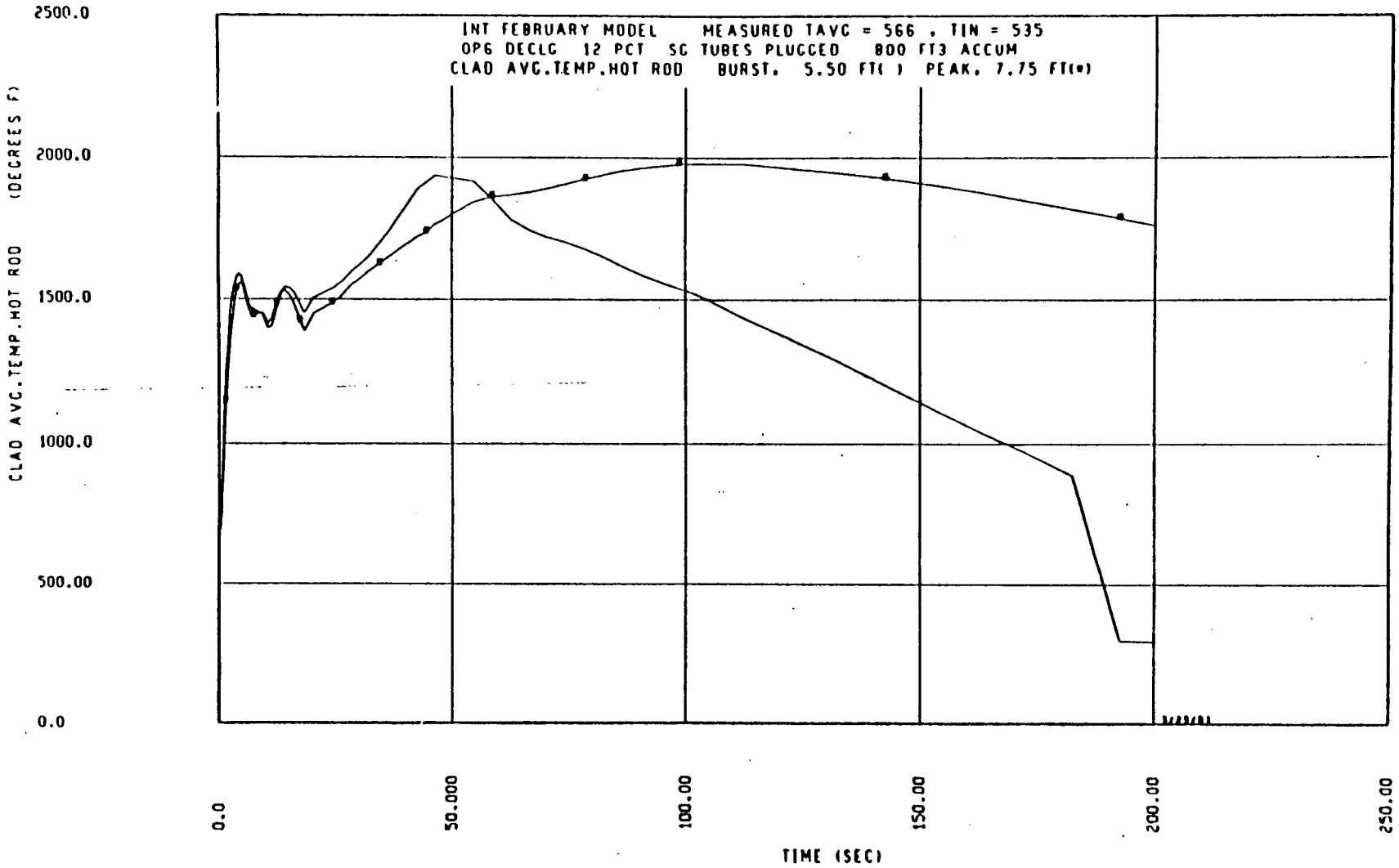


Figure 7C

FLUID TEMPERATURE - DECLG (C_D = 0.6)

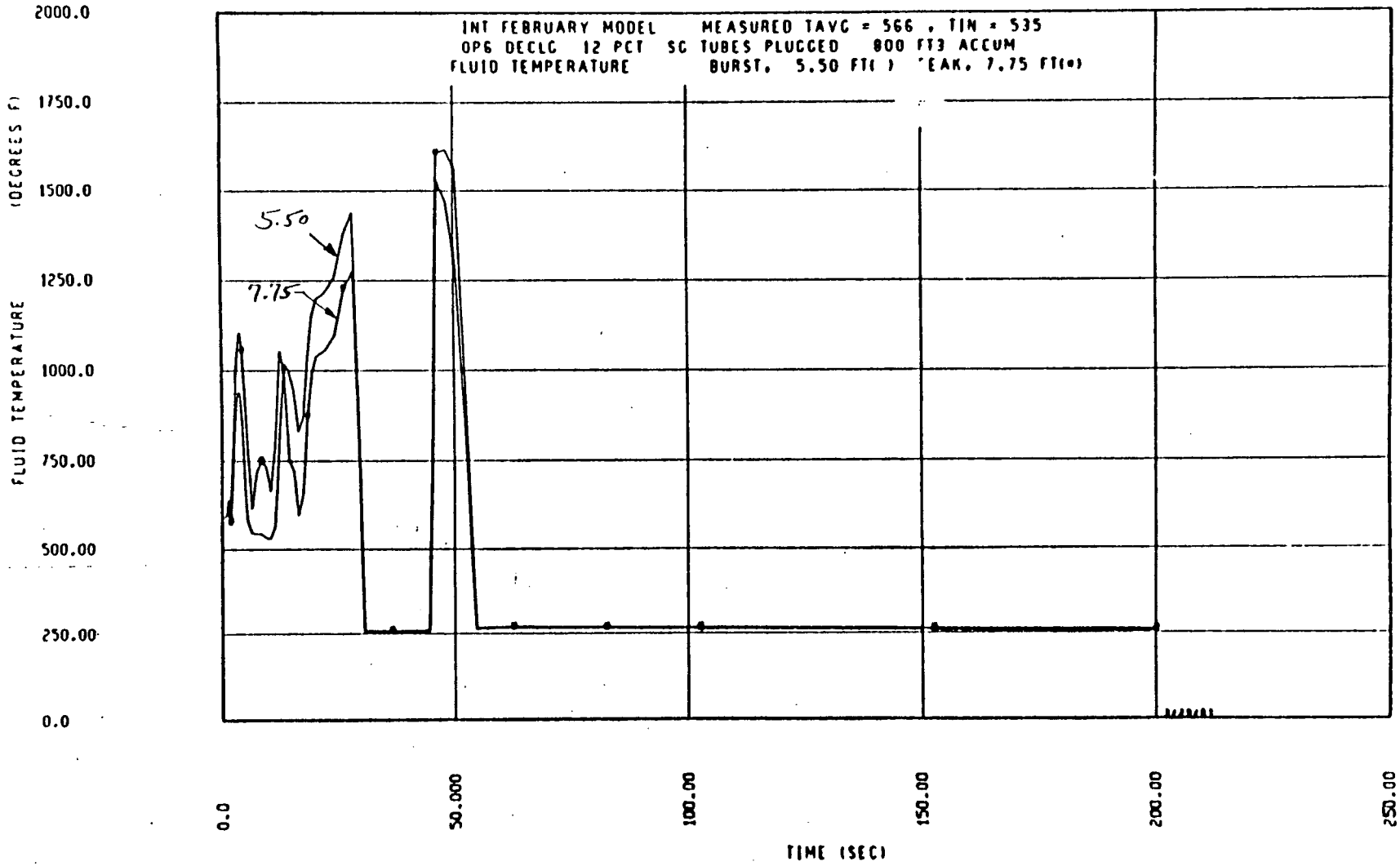


Figure 8C

CORE FLOW RATE - DECLG (C_D = 0.6)

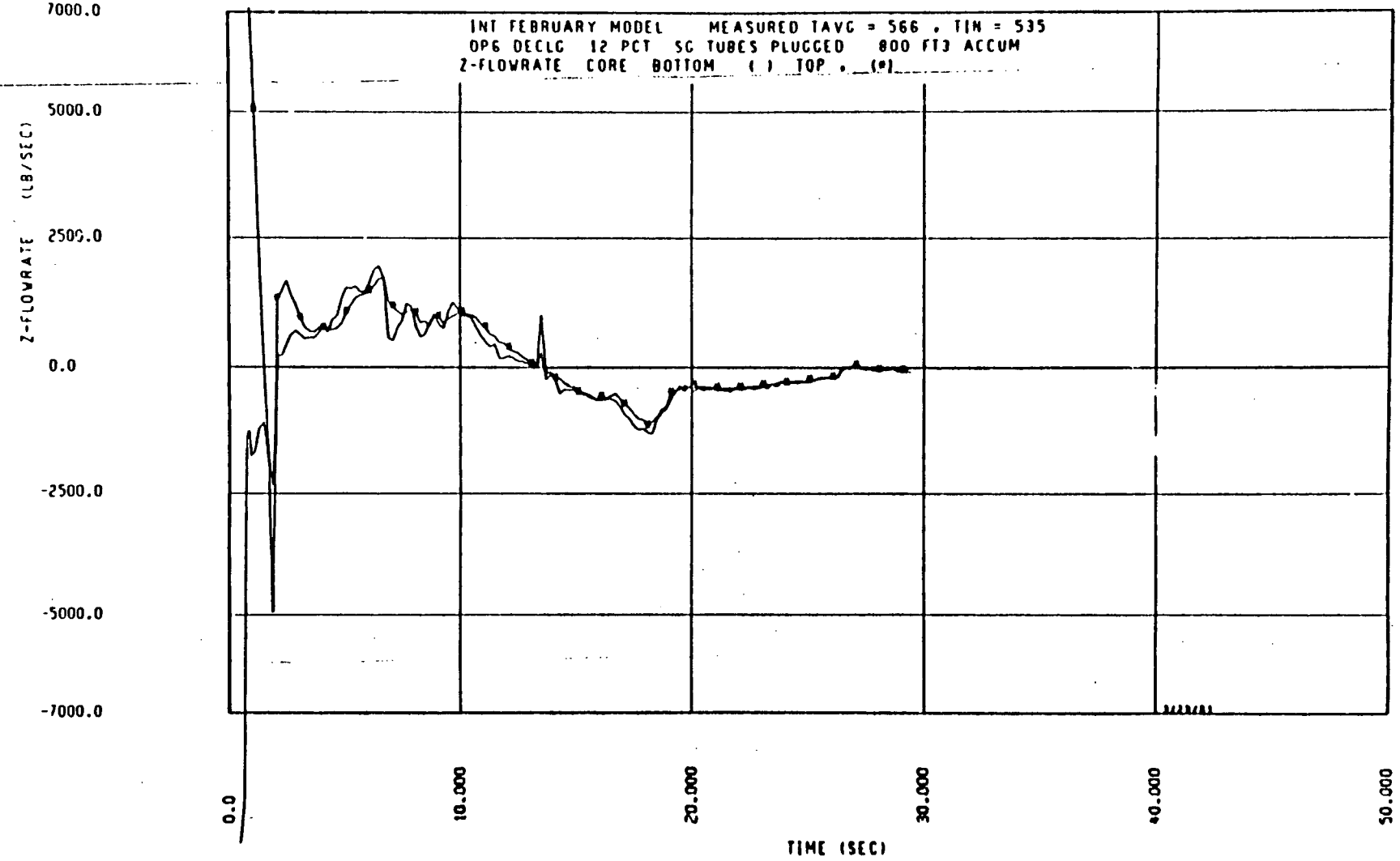


Figure 9C

CORE FLOW RATE - DECLG (C_D = 0.6)

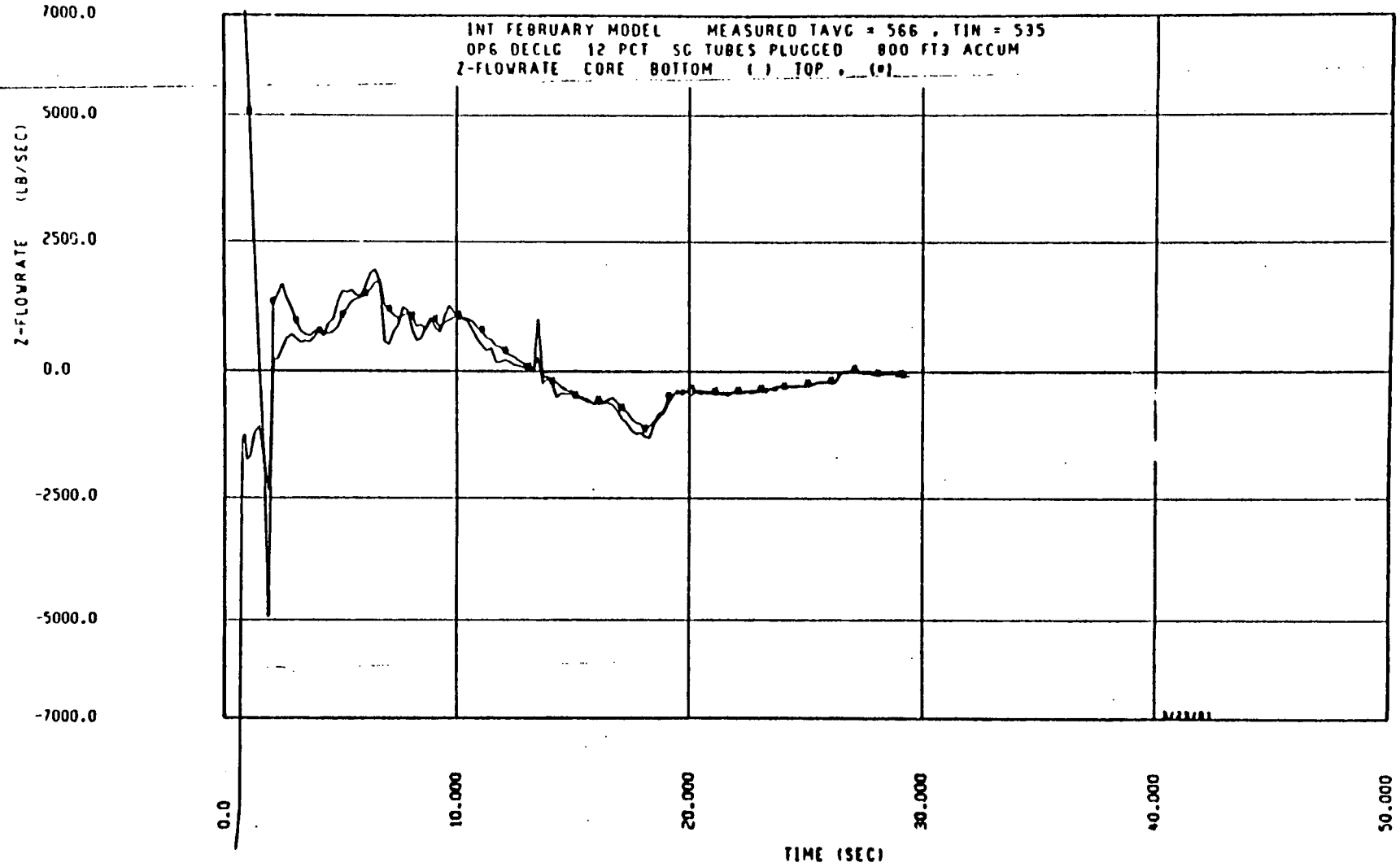


Figure 9C

REFLOOD TRANSIENT - DECLG (C_D = 0.6)

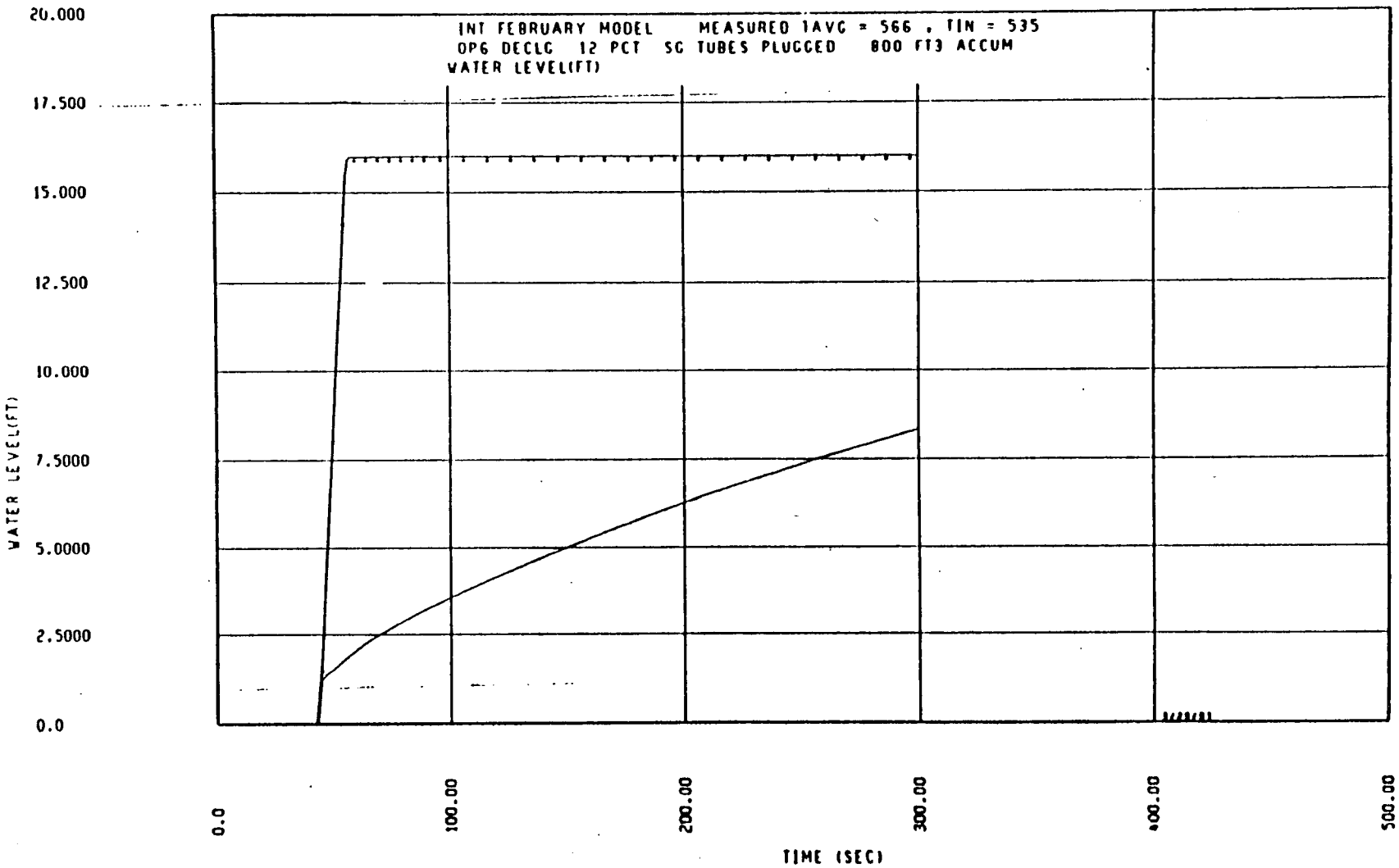


Figure 10C

REFLOOD TRANSIENT - DECLG (C_D = 0.6)

CORE INLET VELOCITY

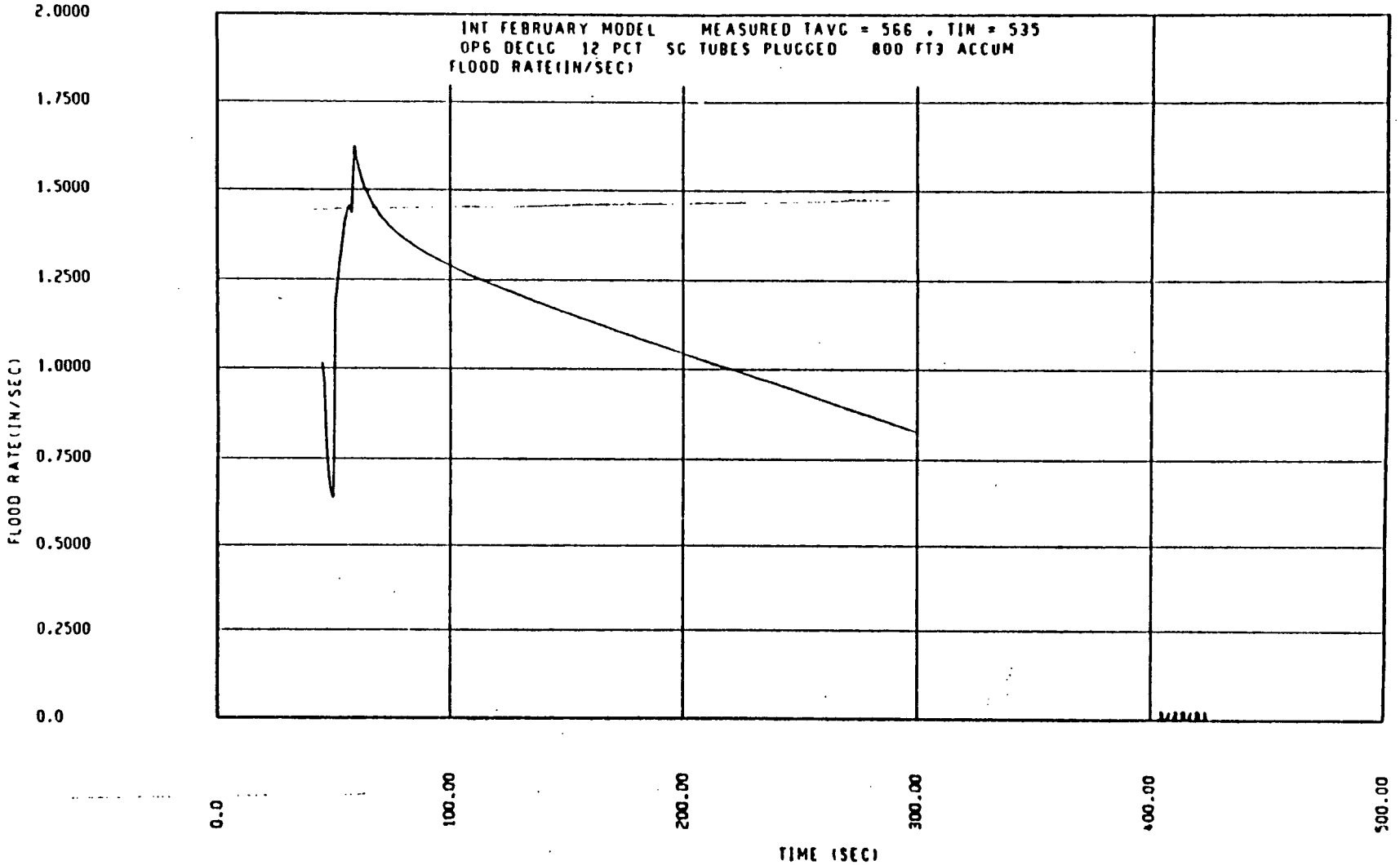


Figure 11C

ACCUMULATOR FLOW (BLOWDOWN) - DECLG (C_D = 0.6)

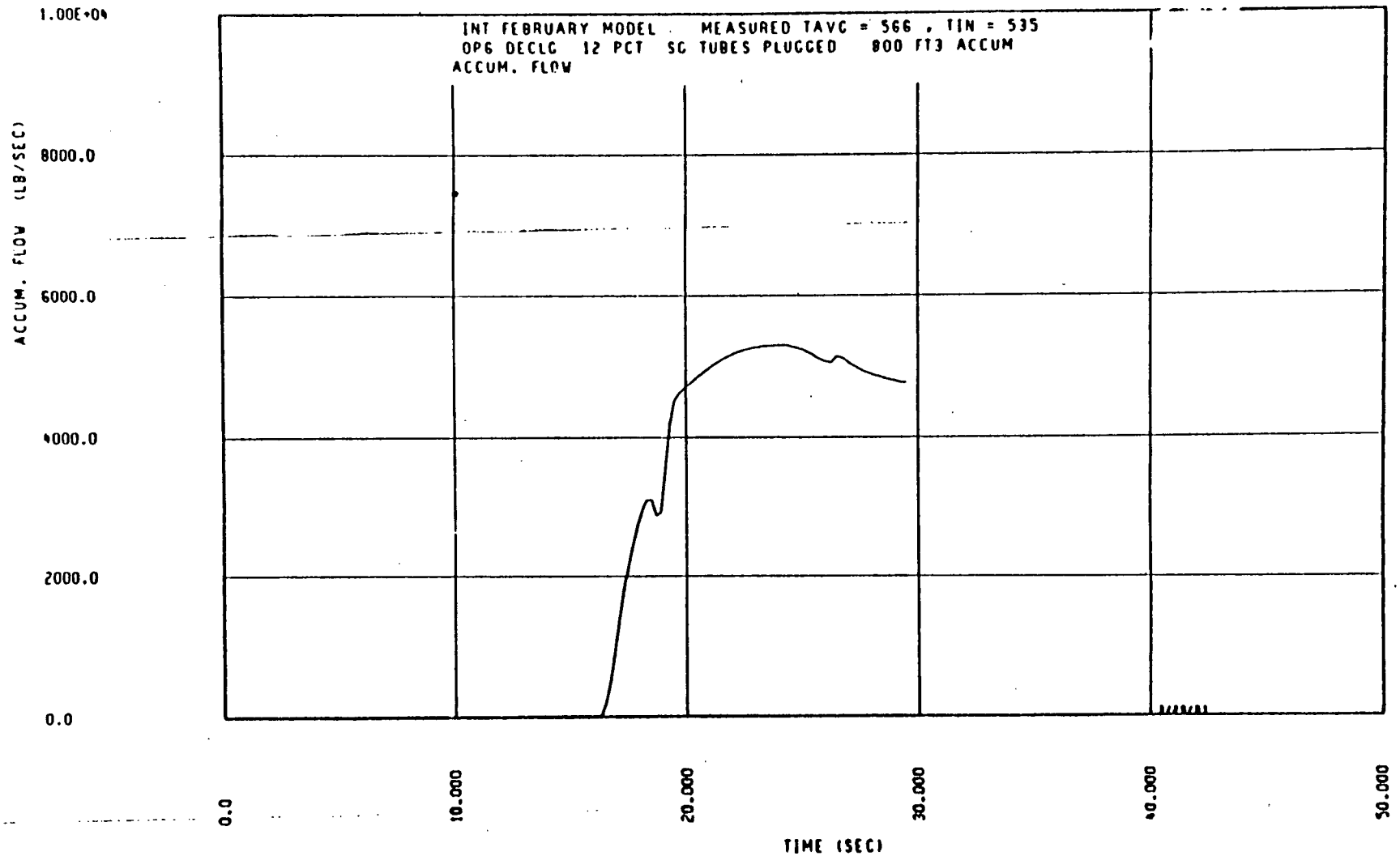


Figure 12C

K&E 10 X 10 TO THE CENTIMETER 46 1510
18 X 25 CM. MADE IN U.S.A.
KEUFFEL & ESSER CO.

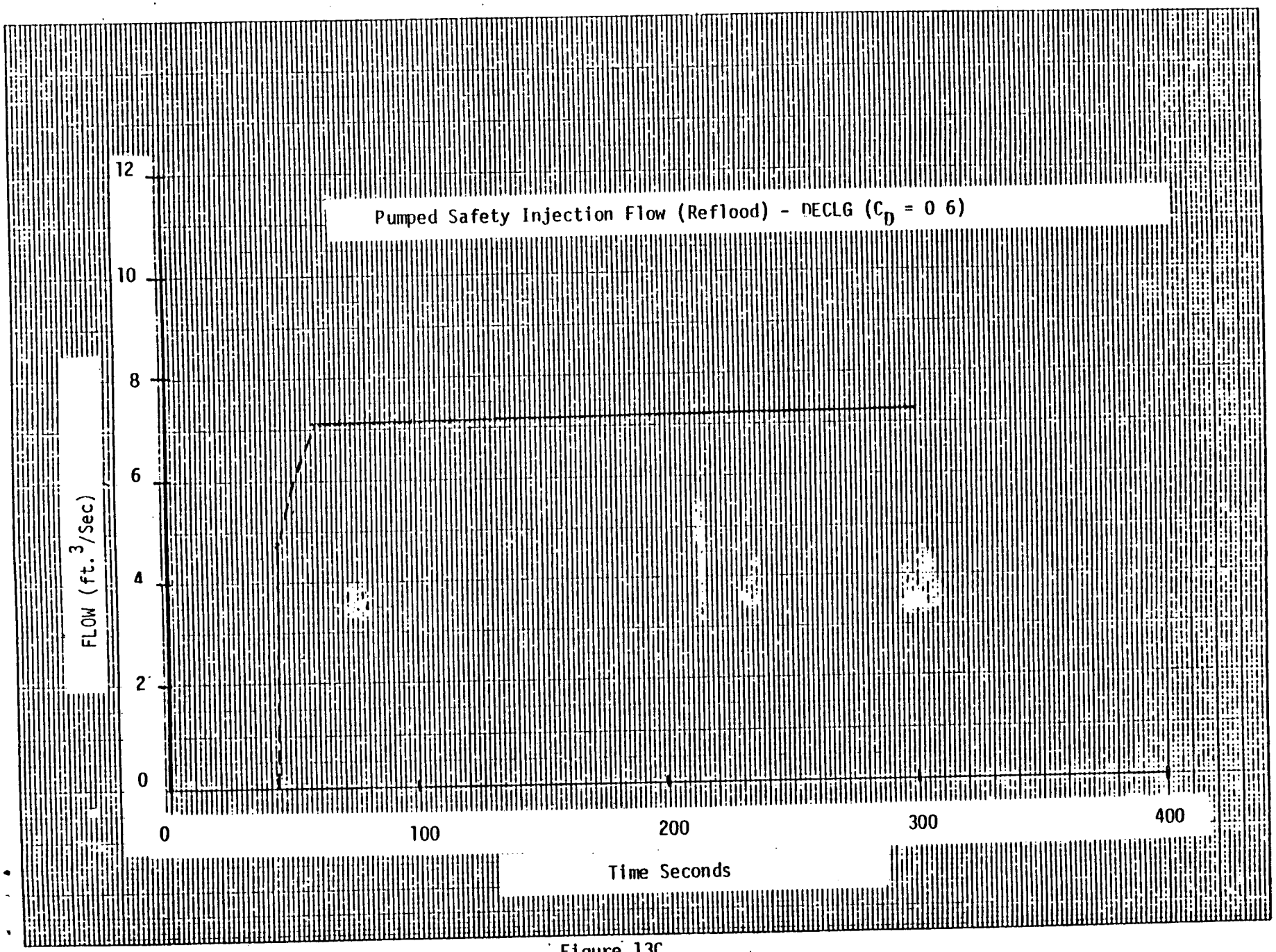


Figure 13C

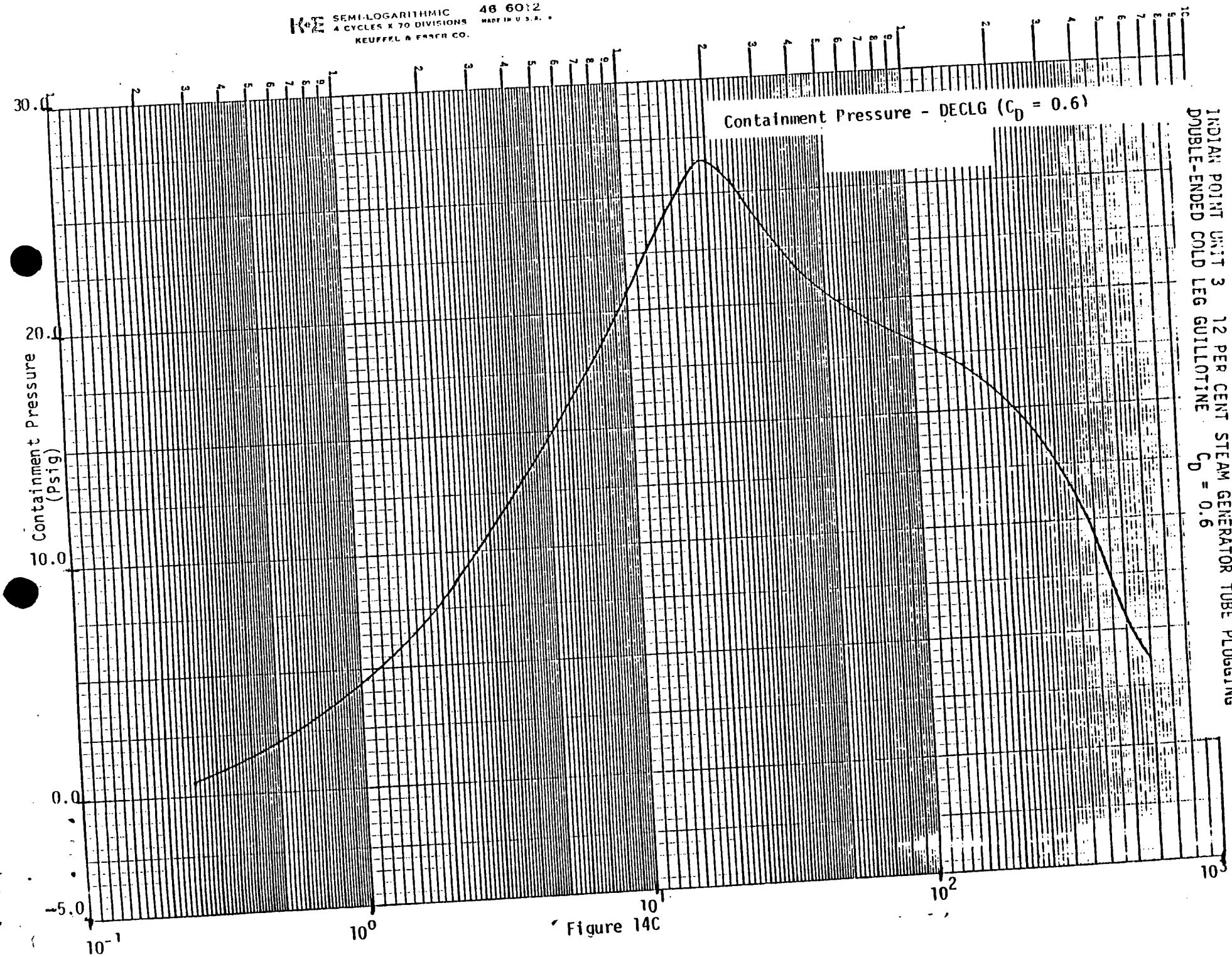

K+E SEMI-LOGARITHMIC 46 6012
4 CYCLES X 70 DIVISIONS MADE IN U.S.A.
KEUFFEL & ESSER CO.
Containment Pressure - DECLG (C_D = 0.6)
Containment Pressure (Psig)
30.0
20.0
10.0
0.0
-5.0
10^{-1}
10^{0}
10^{1}
10^{2}
10^{3}

Figure 14C

CORE POWER TRANSIENT - DECLG (C_D = 0.6)

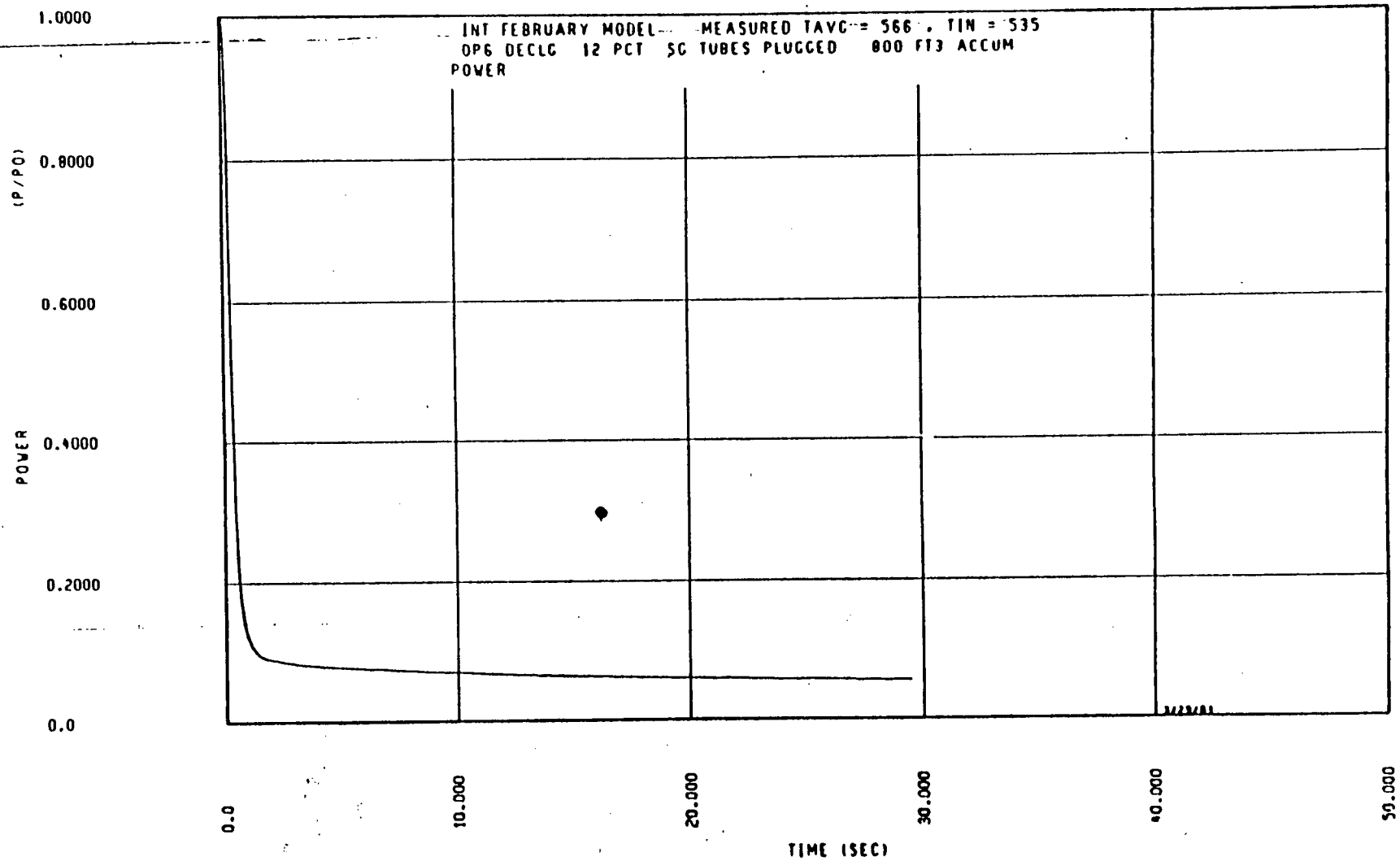


Figure 15C

BREAK ENERGY RELEASED TO CONTAINMENT - DECLG (C_D = 0.8)

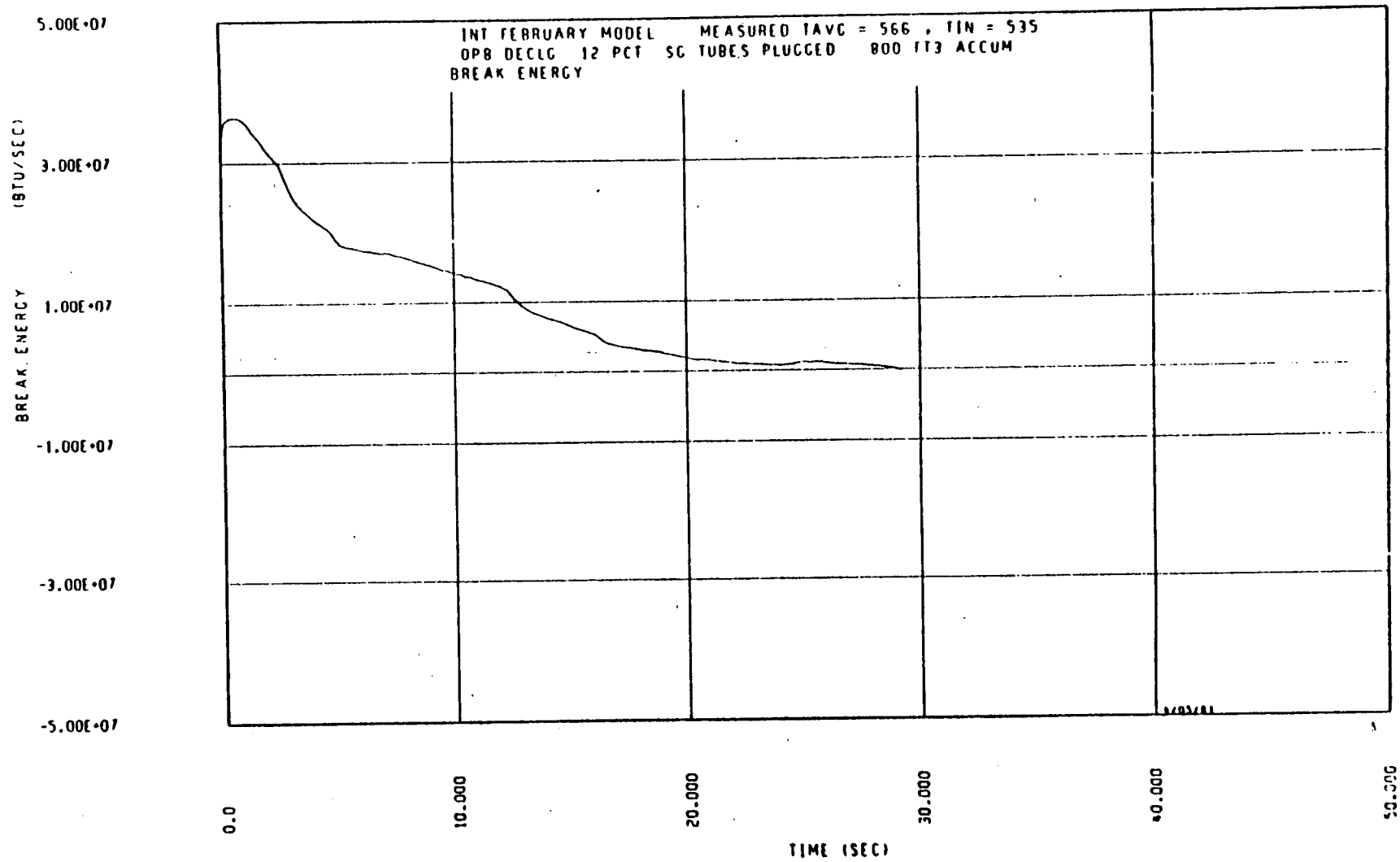


Figure 16

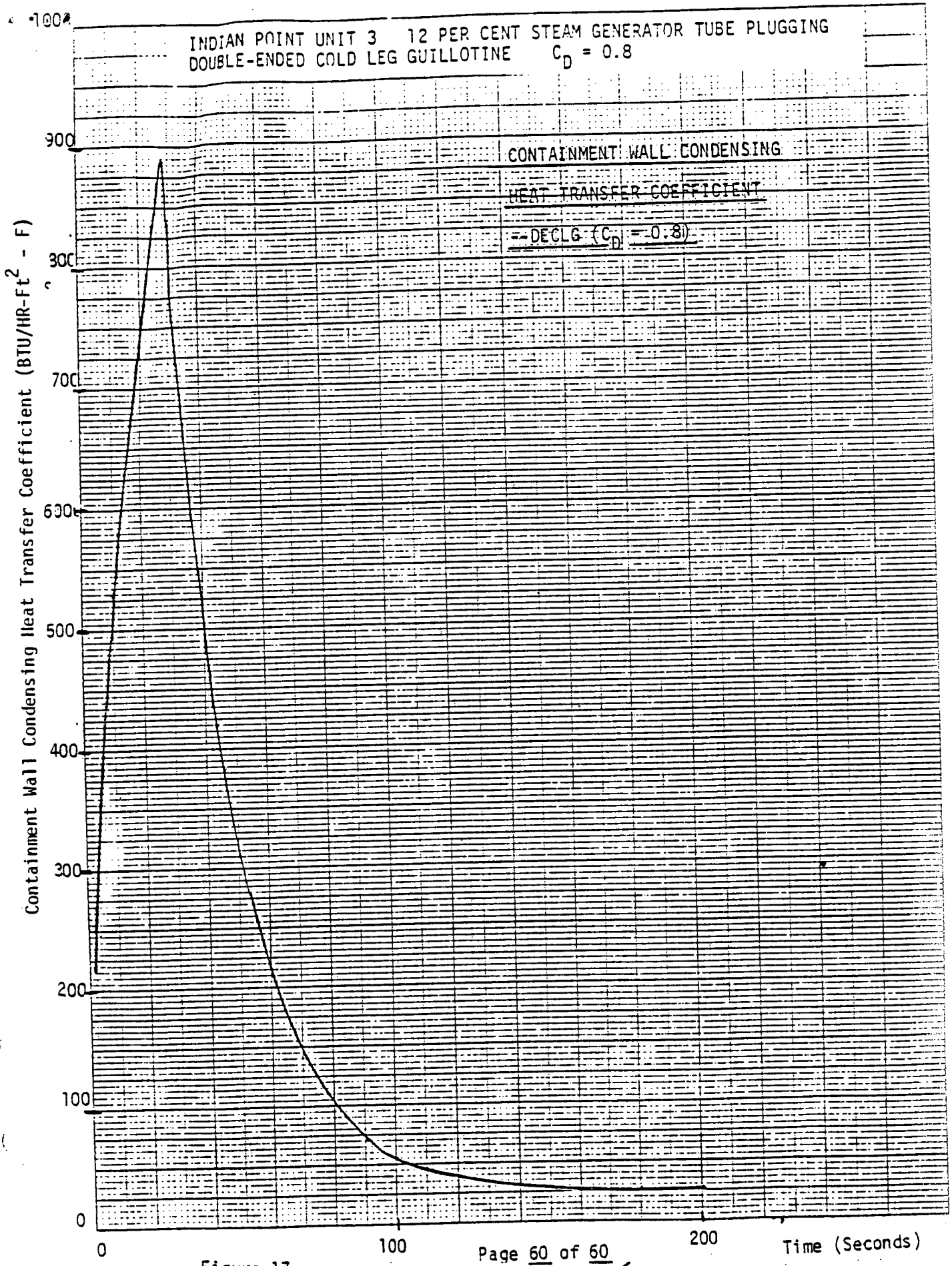

INDIAN POINT UNIT 3 12 PER CENT STEAM GENERATOR TUBE PLUGGING
DOUBLE-ENDED COLD LEG GUILLOTINE C_D = 0.8
CONTAINMENT WALL CONDENSING
HEAT TRANSFER COEFFICIENT
DECLG (C_D = 0.8)
Containment Wall Condensing Heat Transfer Coefficient (BTU/HR-Ft^2 - F)
1000
900
800
700
600
500
400
300
200
100
0
0
100
200
Time (Seconds)

Figure 17