ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATIONS CHANGES

RELATED TO

SAFETY LIMIT, REACTOR CORE

AND

CONTROL ROD AND POWER DISTRIBUTION LIMITS,

SHUTDOWN REACTIVITY

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286

2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT, REACTOR CORE

Applicability

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure and coolant temperature during four-loop operation.

Objective

To maintain the integrity of the fuel cladding.

Specification

The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 2.1-1 for four-loop operation. The safety limit is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

Basis

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating the hot region of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters:

thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 DNB "L" grid geometry correlation.(3) The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.(1)

The curves of Figure 2.1-1 represent the loci of points of thermal power, coolant system pressure and average temperature for which the DNBR is no less than 1.30. The area where clad integrity is assured is below these lines.

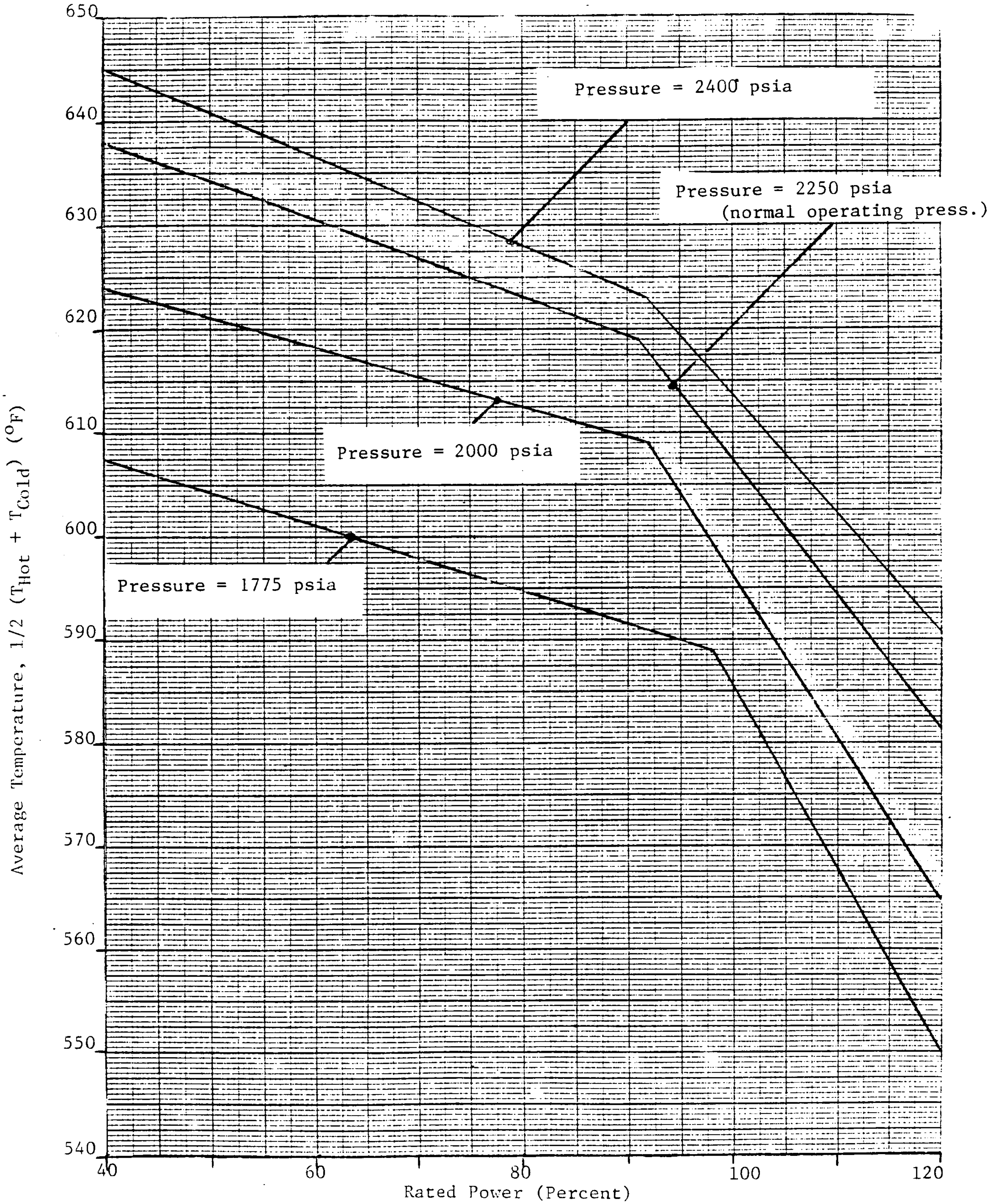
The calculation of these limits includes an $F^N_{\Delta H}$ of 1.55, DNB penalties for increased pellet eccentricity, local power spikes, 8% uncertainty in $F^N_{\Delta H}$, up to 24% steam generator tube plugging, and a reference cosine with a peak of 1.55 for axial power shape.(3)

Figure 2.1-1 includes an allowance for an increase in the enthalpy rise hot channel factor at reduced power based on the expression:

$F^N_{\Delta H} = 1.55\ [1 + 0.2\ (1-P)]$ where P is the fraction of rated power.(3)

The control rod insertion limits are covered by Specification 3.10. Higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits for four loop operation as dictated by Figure 3.10-4, insures that the DNBR is always greater at partial power than at full power.(3)

Indian Point Unit 3
Reactor Coolant Flow > 323600 GPM
24 Percent Tube Plugging



100 Percent Rated Power is Equivalent to 3025 MW_{th}

Amendment No.

Figure 2.1-1

D E L E T E D

Figure 2.1-2. Core Limits - Three Loop Operation

ΔT_o = Indicated ΔT at rated power, $\leq 63.5^oF$

T_{avg} = Average Temperature, ^oF

T' = Indicated T_{avg} at nominal conditions at rated power, 574.7^oF

P = Pressurizer pressure, psig

P' = Indicated nominal pressurizer pressure at rated power = 2235 psig

$K_1 \leq 1.135$

$K_2 = 0.0114 \pm .00057$

$K_3 = 0.00066 \pm .0000033$

K_1 is a constant which defines the over temperature ΔT trip margin during steady state operation if the temperature, pressure and f (ΔI) terms are zero.

K_2 is a constant which defines the dependence of the overtemperature ΔT set point to T_{avg}

K_3 is a constant which defines the dependence of the overtemperature ΔT set point to pressurizer pressure.

ΔI = $q_t - q_b$, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power.

$f(\Delta I)$ = a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests, where q_t and q_b are as defined above such that:

(a) for $q_t - q_b$ within -20, + 10 percent, $f(\Delta I) = 0$.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds + 10 percent, the ΔT trip set point shall be automatically reduced by an equivalent of 6.0 percent of rated power.

(c) for each percent that the magnitude of $q_t - q_b$ exceeds -20 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 1.5 percent of rated power.

(5) Overpower ΔT

$$\Delta T \leq \Delta T_o \left[K_4 - K_5 \frac{dT_{avg}}{dt} - K_6 (T_{avg} - T') - f(\Delta I)\right]$$

where

ΔT_o = indicated ΔT at rated power, (100% full power measured ΔT, no greater than 63.5° F.)

T_{avg} = average temperature, °F

T' = indicated T_{avg} at nominal conditions at rated power, $\leq$ 574.7°F

K_4 $\leq$ 1.089

K_5 = 0 for decreasing average temperature

$\geq$ 0.175 sec/°F for increasing average temperature

K_6 = 0 for $T \leq T'$

$\geq$ 0.00116 for $T > T'$

K_4 is a constant which defines the overpower ΔT trip margin during steady state operation if the temperature and the $f(\Delta I)$ terms are zero.

K_5 is a constant determined by dynamic considerations to compensate for piping delays from the core to the loop temperature detectors; it represents the combination of the equipment static gain setting and the time constant setting.

K_6 is a constant which defines the dependence of the overpower ΔT setpoint to T_{avg}.

$f(\Delta I)$ = as defined above.

$\frac{dT_{avg}}{dt}$ = rate of change of T_{avg}

(6) Low reactor coolant loop flow:

(a) $\geq$ 90% of normal indicated loop flow

(b) Low reactor coolant pump frequency — $\geq$ 55.0 cps

(7) Undervoltage - $\geq$ 70% of normal voltage

Amendment No.

3 LIMITING CONDITIONS FOR OPERATION

For the cases where no exception time is specified for inoperable components, this time is assumed to be zero.

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System; operational components; heatup, cooldown, criticality, activity, chemistry and leakage.

Objective

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

Specification

A. OPERATIONAL COMPONENTS

1. Coolant Pumps

a. At least one reactor coolant pump or one residual heat removal pump in the Residual Heat Removal System when connected to the Reactor Coolant System shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

b. When the reactor is critical and above 2% rated power, except for natural circulation tests, at least two reactor coolant pumps shall be in operation.

c. The reactor shall not be operated at power levels above 10% rated power with less than four (4) reactor coolant loops in operation.

Basis

When the boron concentration of the Reactor Coolant System is to be reduced the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one half hour. The pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant.

Heat transfer analyses show that reactor heat equivalent to 10% of rated power can be removed with natural circulation only (1); hence, the specified upper limit of 2% rated power without operating pumps provides a substantial safety factor.

The reactor shall not be operated at power levels above 10% rated power with less than four (4) reactor coolant loops in operation until safety analyses for less than four loop operation have been submitted by the licensee and approval for less than four loop operation at power levels above 10% rated power has been granted by the Commission. (See license condition 2.C.(3))

Each of the pressurizer code safety valves is designed to relieve 420,000 lbs. per hr. of saturated steam at the valve set point.

If no residual heat were removed by the Residual Heat Removal System the amount of steam which could be generated at safety valve relief pressure would be less than half the capacity of a single valve. One valve therefore provides adequate protection for overpressurization.

The combined capacity of the three pressurizer safety valves is greater than the maximum surge rate resulting from complete loss of load (2) without a direct reactor trip or any other control.

The requirement that 150 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown.

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal off possible RCS leakage paths.

References

1) FSAR Section 14.1.6

2) FSAR Section 14.1.8

Applicability:

Applies to the limits on core fission power distributions and to the limits on control rod operations.

Objectives:

To ensure:

1. Core subcriticality after reactor trip,
2. Acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation and transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and
3. Limit potential reactivity insertions caused by hypothetical control rod ejection.

Specifications:

3.10.1 Shutdown Reactivity

The shutdown margin shall be at least as great as shown in Figure 3.10-1.

3.10.2 Power Distribution Limits

3.10.2.1 At all times, except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

$$F_Q(Z) \leq (2.14/P) \times K(Z) \text{ for } P > 0.5$$
$$F_Q(Z) \leq (4.28) \times K(Z) \text{ for } P \leq 0.5$$
$$F^N_{\Delta H} \leq 1.55 \, [1 + 0.2 \, (1-P)]$$

where P is the fraction of full power at which the core is operating. K(Z) is the fraction given in Figure 3.10-2 and Z is the core height location of F_Q.

on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F_{\Delta H}^{N}$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

It should be noted that $F_{\Delta H}^{N}$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^{N}$.

An upper bound envelope of 2.14 times the normalized peaking factor axial dependence of Figure 3.10-2 has been determined consistent with Appendix K criteria and is satisfied by all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10. The results of the loss of coolant accident analyses based on this upper bound normalized envelope of Figure 3.10-2 demonstrate a peak clad temperature of 1995°F, which is below peak clad temperature limit of 2200°F. [2]

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

In the specified limit of $F_{\Delta H}^{N}$ there is a 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F_{\Delta H}^{N} \leq 1.55/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape of flux difference control and control bank insertion limits, are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation in $F^N_{\Delta H}$ allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met. In Specification 3.10.2, F_Q is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests).

The procedures for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of Flux Difference (ΔI) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = ΔI/fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control assure that F_Q upper bound envelope of 2.14 times Figure 3.10-2 is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with

the control rod bank more

than 190 steps withdrawn (i.e. normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup

D E L E T E D

Figure 3.10-5 Insertion Limits 100 Step

Overloop 3 Loop Operation

Amendment No. ~~14~~

ATTACHMENT II

SAFETY EVALUATION

RELATED TO

SAFETY LIMIT, REACTOR CORE

AND

CONTROL ROD AND POWER DISTRIBUTION LIMITS,

SHUTDOWN REACTIVITY

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286

Section I - Description of Modification to the Technical Specifications

The proposed changes to the Technical Specifications are shown in Attachment I. Sections 2.1, 2.3, 3.1 and 3.10 of the Technical Specifications have been revised. These proposed changes delete references to three loop operation and incorporate the results of a recent ECCS reanalysis which modifies the reactor core limits.

The Westinghouse ECCS reanalysis report is based on a steam generator tube equivalent plugging level of twenty-four (24) percent and is enclosed as Attachment III to this submittal.

Section II - Purpose of Modification to the Technical Specifications

The purpose of the modification is to revise the IP-3 Technical Specifications so as to comply with the ECCS reanalysis for up to 24% steam generator equivalent tube plugging. References to three loop operating have been deleted from the Technical Specifications since IP-3 is restricted to 10% power or less for less than four loop operation.

Section III - Impact of the Change to the Technical Specifications

The proposed changes to the Technical Specifications do not require modifications of any system or subsystem. This change will permit plant operations with a higher percentage of steam generator tubes plugged. This requires a lower heat flux peaking factor value to meet peak clad temberature requirements, based on the approved Westinghouse 1981 model. The modification as proposed will not impact the ALARA or Fire Protection Program at IP-3, nor will it adversely impact the environment.

Section IV - Implementation of the Modification to the Technical Specifications

The proposed changes to the Technical Specifications are included as Attachment I to this letter.

Section V - Conclusion

The incorporation of these modifications: a) will not change the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report: b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the basis for any Technical Specification; and d) does not constitute an unreviewed safety question.

Section VI - References

(a) IP-3 FSAR

(b) IP-3 SER

ATTACHMENT III

EMERGENCY CORE COOLING SYSTEM REANALYSIS

FOR STEAM GENERATOR TUBE PLUGGING

LEVEL OF TWELVE PERCENT

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286

The loss of Coolant Accident (LOCA) has been reanalyzed for Indian Point Unit III with 24% S.G. tubes plugged. The following information amends 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-8839[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, 10, 11, 12, 13 and 14. The analysis presented here was performed with the 1981 version of the evaluation model which includes modifications delineated in reference 16.

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

Containment Pressure Relief

Branch Technical Position CSB6-4 states that the evaluation of a containment pressure relief system design should include "an analysis of the reduction in containment pressure resulting from the partial loss of containment atmosphere during the accident for ECCS backpressure determination." An analysis has been performed for Indian Point Unit 3 based on the limiting FAC analysis case (DECLG break, $C_D = 0.4$) which was obtained using the 1981 Westinghouse Evaluation Model.

Valves in the containment pressure relief system will close shortly after the beginning of a postulated LOCA transient based on the response to the containment isolation signal. The containment pressure relief at Indian Point Unit 3 consists of a single 10-inch pressure relief line.

This line is conservatively represented in the analysis by the following model:

1. The frictional resistance associated with duct entrance and exit bases, filters, duct work bends and skin friction has not been considered.
2. Fan coastdown effects are ignored.
3. Steady-state flow is immediately established through the purge system ducts at the inception of the LOCA.
4. A 3.5 second valve closure time is considered. No credit is taken for the reduction in flow area with time as the valve moves towards the fully closed position.

A mixture of steam and air will pass through the containment pressure relief lines during the time that the isolation valves are assumed to remain open. The effects of varying the exhaust gas composition have been investigated by considering two extreme cases, air flow exclusively and steam flow exclusively. For the purposes of this analysis it was conservatively assumed that critical flow will be established thru the pressure relief lines at the inception of the LOCA and will be maintained until valve closure time.

Equation (4.18) in reference (17) was used to calculate the critical flow of air thru the maximum available area (10" diameter/line). Figure 14 of reference (18) was used to establish the critical flow rate of steam through the pressure relief lines. The total mass released during the time in which the valves are assumed to be open is calculated as 247.5 lbs. of air or 178.5 lbs. of steam.

The reduction in containment pressure from the calculated mass loss is less than 0.1 psi in the case of either air flow or steam flow. A containment pressure reduction of this magnitude on the calculated peak clad temperature (PCT) is expected to be minor (less than 1.0°F).

If consideration of the effects of containment pressure relief on LOCA is applied to the 24% tube plugging case (FQ=2.14), no additional reduction in peaking is necessary.

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 $2200^{o}F$ based on a total core peaking factor of 2.20

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.

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 written to Darrell G. Eisenhut of the Nuclear Regulatory Commission, letter number NS-TMA-2165, December 16, 1979.

16. NS-TMA-2448

17. Shapiro, A. H. The Dynamils and Thermodynamics of Compressible Fluid Flow, Volume 1, p. 85.

18. 1967 ASME Steam Tables, p. 301.

[Westinghouse Letter to NRC]

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Technology Division

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May 15, 1981

NS-TMA-2448

Mr. James R. Miller
Special Projects Branch
Division of Licensing
U. S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, Maryland 20014

Dear Mr. Miller:

On several occasions, representatives of the Nuclear Regulatory Staff and Westinghouse have discussed the need to close several issues that currently affect Appendix K analyses. Appendix K analyses performed by Westinghouse and submitted for NRC approval are presently appended with interim estimates of the impact of the fuel rod burst and blockage models required by the NRC in NUREG-0630. These estimates are based on generic sensitivity studies and were chosen to bound the impact. These interim estimates have been provided for every analysis performed since November, 1979. Based on additional generic sensitivity studies performed by Westinghouse, credit is being allowed by the NRC for the use of the "UHI Software Technology" models (Reference 1) to cancel most of the penalty associated with the NUREG-0630 models.

In addition to the above, Westinghouse is now providing Optimized Fuel Assemblies to many licensees. Analyses performed for the Optimized Fuel Report, WCAP-9500, and for several licensees require an adjustment to the FLECHT reflood heat transfer correlation (Reference 2). Approval of this adjustment is necessary.

Westinghouse has also informed the staff of a change that has been made to the approved Appendix K Evaluation Model to accurately model the interaction between the pumped safety injection flow and the accumulator injection flow (Reference 3). It is the understanding of Westinghouse that Part II of Appendix K to 10CFR50 requires that this change be submitted to the NRC for review and approval.

Lastly, all analyses submitted since December, 1978 incorporate changes to th[e] evaluation model described in References 4 and 5. The changes in the evaluat[ion] model have either no impact on peak clad temperature or are required to prope[rly] model the system under investigation. These two letters have been referenced for every LOCA plant application since December, 1978 and have been part of o[ur] model. These items have already been discussed with the Staff and we believe [are] also informally approved by the NRC.

It is clear that all Appendix K analyses today are being performed with changes to the Evaluation Model that have been reviewed but not yet formally approved by NRC. Westinghouse strongly believes that the resultant model is in compliance with Appendix K, however, it seems prudent to bring the formal approval process up to date. This will assure the highest degree of confidence in the determination of peaking factor being used in operating plant technical specifications. Westinghouse is also fully aware of manpower limitations at the NRC as well as at Westinghouse due to post-TMI demands and pressure to revitalize the plant licensing effort. Therefore, Westinghouse would like to propose that the above model changes be formally approved by the NRC. The changes will be explicitly made to our computer codes. The new model will be known as the "1981 version" of the Westinghouse Appendix K Evaluation Model. Westinghouse believes that the approval of the model described herein, which in essence is incorporation of either previously submitted and reviewed items or NRC mandated changes (NUREG-0630), will provide an evaluation model which is more integrated thereby removing the requirement to provide and review the related appendages to each application. Only model changes which minimize manpower required for review on the part of both NRC and Westinghouse are included. For completeness, these changes are again described in the Appendix to this letter. All previous correspondence is also attached.

If there are any questions, please call Dr. V. J. Esposito, (412) 373-4059.

Very truly yours,

T. M. Anderson, Manager
Nuclear Safety Department

RAM/ls

References

1. Letter from T. M. Anderson, Westinghouse Electric Corporation to J. R. Miller, U.S. Nuclear Regulatory Commission, NS-TMA-2311, September 15, 1980.

2. Letter from L. E. Hochreiter, Westinghouse Electric Corporation, to N. Lauben, U.S. Nuclear Regulatory Commission, SE-LEH-434, March 27, 198

3. Letter from T. M. Anderson, Westinghouse Electric Corporation, to D. Ross, U.S. Nuclear Regulatory Commission, NS-TMA-2384, December 22, 1

4. Letter from T. M. Anderson, Westinghouse Electric Corporation, to J. Stolz, U.S. Nuclear Regulatory Commission, NS-TMA-1981, November 1, 1978.

5. Letter from T. M. Anderson, Westinghouse Electric Corporation, to R. Tedesco, U.S. Nuclear Regulatory Commission, NS-TMA-2014, December 13, 1978.

6. Hardy, D. G., "High Temperature Expansion and Rupture Behavior of Zircaloy Tubing", National Topical Meeting on Water-Reactor Safety, Salt Lake City, Utah, March, 1973.

7. Bordelon, F. M., et.al., "LOCTA-IV Program: Loss-Of-Coolant Transient Analysis", WCAP-8301, June, 1974 (W Prop. 2).

1. NUREG-0630 Models

The LOCTA-IV and SATAN-VI codes will be modified to incorporate the NUREG-0630 models for calculating the burst temperature, assembly flow blockage and cladding burst strain as specified by NRC. An algorithm to calculate the cladding heat up rate and a revision to the model for calculating clad swelling prior to burst are also included.

The NRC's new burst temperature curve has been programmed in tabular form for heat up rates of 0°, 14°, and 28° centigrade per second corresponding to Figure 3 in NUREG-0630. These numbers were verified by comparison to equation 3-2 in NUREG-0630. The burst temperature at each node is determined by parabolic interpolation at the appropriate cladding hoop stress and heat up rate. The 28°C/sec burst curve is used for heat up rates greater than 28°C/sec and the isothermal burst curve is used when the clad is cooling down.

The assembly flow blockage curves, corresponding to Figures 14 and 15 in NUREG-0630 are included in the changes. For heat up rates of 10°C/sec or less Figure 14 is used and for heat up rates of 25°C/sec or greater, Figure 15 is used. The flow area reduction is determined as a function of the known burst temperature. For heat up rates between 10°C/sec and 25°C/sec the reduction in flow area is determined by linearly interpolating between the two curves.

The circumferential strain curves, shown in Figures 6 and 7 of NUREG-0630, are incorporated in the same manner as the reduction in flow area curves.

An algorithm to calculate the cladding heat up rate will be included in LOCTA-I to be used in the revised swelling and rupture models. The heat up rate is calculated for each axial node on the fuel rod, but only the heat up rate at the peak clad temperature location is used to calculate the burst temperature. The algorithm for calculating heat up rate must be meaningful for any type of clad temperature transient it may encounter. The following discussion illustrates h this is accomplished.

Figure 1 demonstrates a number of hypothetical conditions that may be encounter during a fuel rod heat up calculation. This curve is not from an actual transi For the purposes of this discussion, each lettered point represents a calculati time step.

The instantaneous heat up rate is used until the cladding temperature is within of the burst temperature. When this condition is reached (Point A), the clad temperature and time are recorded to be used as a reference for the calculation As long as the clad temperature is above the reference temperature (Points B, C D), the heat up rate at each succeeding time step is determined by:

$$HUR = \frac{\sum_{i=1}^{n} \frac{T_i - T_{ref}}{\tau_i - \tau_{ref}}}{n}$$

where HUR = heat up rate

Ti = clad temperature at ti

Tref = reference clad temperature

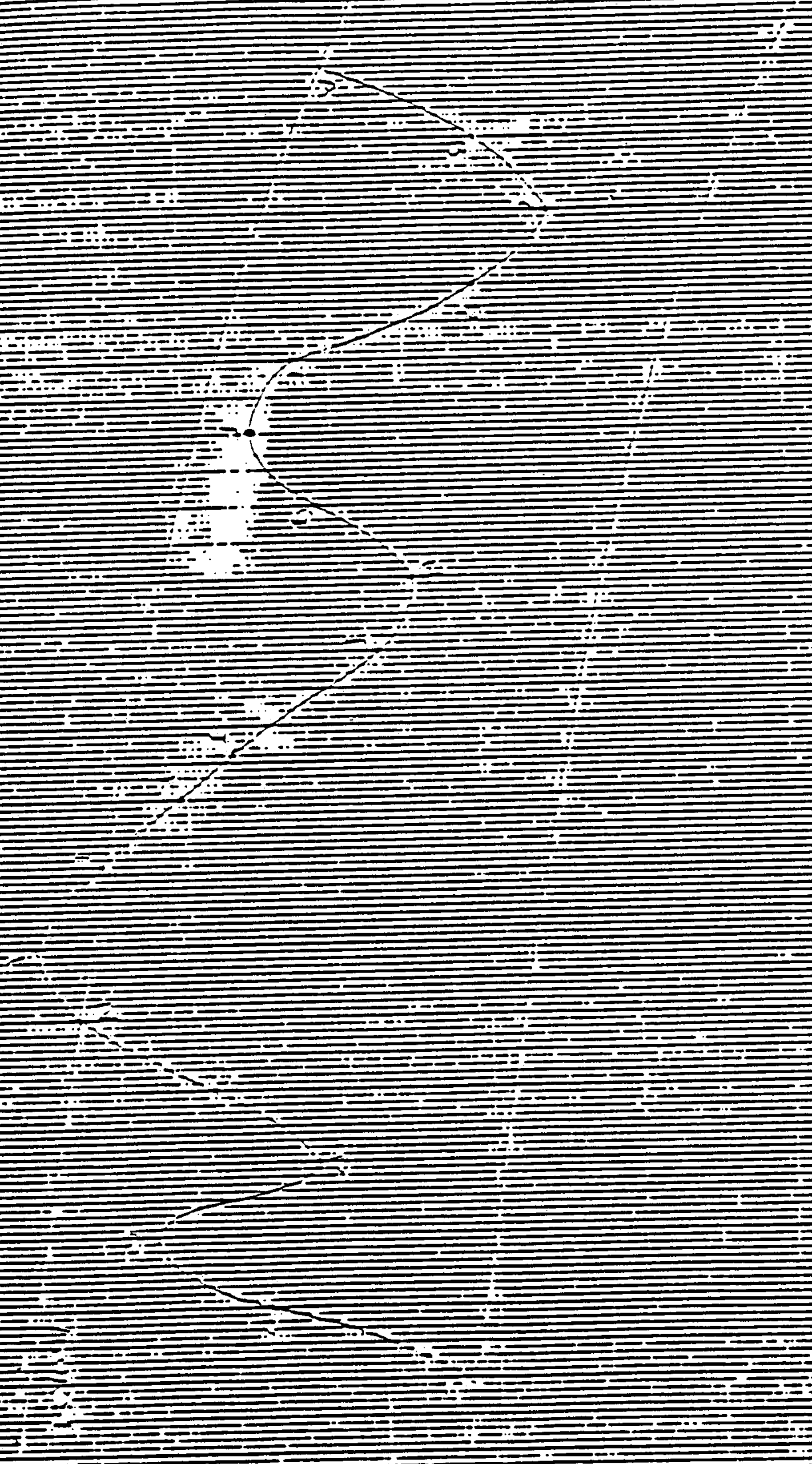
ti = transient time

tref = reference time

When the clad temperature falls below the reference clad temperature (Case E), this calculation stops and the most recent heat up rate is used until the temperature begins to rise. When the temperature reaches a new minimum (Point F) the reference temperature and time are reset and equation (1) is used from this point (Points G, H and I).

At Point J the temperature falls below the reference temperature and the heat up rate calculated at I is used. If the clad temperature falls below TBURST - 200 the instantaneous heat up rate is used (Points L and M). At Point N the reference time and temperature are reset and these are used in equation (1) until burst occurs at point S.

This average heat up rate algorithm is appropriate since instantaneous heat up rate can be very misleading. Also, the technique will give conservatively lower heat up rates than the technique used to determine heat up rate from the ORNL data for the burst curve in NUREG-0630 where "initial" heat up rate was used.



The fuel rod uniform strain model in the current LOCTA code was developed from the data that was published by Hardy.[8] The data was derived from a series of single rod burst tests on electrical resistance heated rods in a vacuum chamber. During the experiment, no provision was made for direct diametral expansion measurement; therefore, the rod diametral measurements were accomplished after the rod was ramped and allowed to cool. The Hardy data was correlated by Westinghouse to be of the form

$$\frac{d\varepsilon}{dt} = A \text{ EXP } [C\sigma(t) - B/T(t)L] \tag{6}$$

where ε is the true strain, σ is the true stress, T is the temperature, and t is the time using a least squares fit.

As previously mentioned, the LOCTA code contains a uniform fuel rod strain model. One objective of the Multirod Burst Test (MRBT) Program is to provide a data base that could be used to assess the magnitude of geometrical changes of fuel rod cladding in a multirod array during a LOCA. The proposed models in NUREG-0630 are based on a reasonably large data base. The ORNL/MRBT and REBEKA data are included in the data base. An analysis of the ORNL/MRBT data for tests B-1, B-2, and B-3 indicates that the average rod stram just prior to clad burst is approximately 20 percent.

In order to expedite the review process, however, the current 10 percent clad swelling limit is being retained in LOCTA-IV.

When the proposed NRC burst curve is used with the current Westinghouse clad strain model, clad burst occurs earlier in time and the rod strain prior to clad burst is as low as 3 percent in some cases. Thus, the clad strain rate was artificially increased to obtain a more realistic prior to burst strain. To accomplish this task, the C constant in equation 1 was increased by 50 percent to increase the strain rate. Since the constant in the strain equation was modified, the rate of strain calculated with the revised model does not agree with the experimental data of Hardy. However, the strain just prior to burst does agree with the experimental data from ORNL and REBEKA. The clad strain at burst using the NUREG-0630 models and new swelling model is consistant with the clad strain at burst when the present models are used.

In summary, the Westinghouse LOCTA strain model agrees with established experimental data. When the model is used with the NRC burst curves, clad burst occurs early in time and the rod strain prior to clad burst does not agree with the ORNL and REBEKA data. To make the calculated strain more consistent with the experimental data when using the NRC burst curves, the strain rate in the strain model was enhanced. Thus, the proposed new clad strain model agrees with the ORNL and REBEKA data and can be used with NRC burst curves.

It should be noted Westinghouse still does not agree with the use of the models in NUREG-0630. NUREG-0630 contains models for fuel rod burst temperature, burst strain and flow blockage which are overly conservative due to the use of a typical data and/or inappropriate interpretation of data. However, if the use of those models is required, it is far better to incorporate them in the Evaluation Model rather than as an appendage to the Evaluation Model.

2.7 UHI Software Technology

The NRC has reviewed and approved the models that comprise a package known as the "UHI Software Technology" for application to all Westinghouse plants equipped with upper head injection, the Westinghouse reload of Millstone, Unit 2 and, most recently, the Zion plant. Westinghouse has requested generic approval of these models (Attachment 1). Table 1 demonstrates the impact of UHI software technology on a typical 4 loop, 3250 Mwt, 15 x 15 fuel plant.

It is our understanding that minimal review is necessary before providing such generic approval. In fact, it is our understanding from conversations with the staff that the only uncertainty has to do with the use of the 2-D downcomer model.

Westinghouse has performed sensitivity studies to determine the impact of the two-dimensional downcomer. This has been done for a preliminary version of the February 78 SATAN model as well as the approved version of the February, 1978 model and the February, 1978 model modified with the UHI Software Technology. The preliminary February 78 analysis showed that there was very little change in results when the two dimensional downcomer was used. This sensitivity was performed for a 4 loop, 3411 Mwt plant with 17 x 17 standard fuel. This change resulted in a change in peak clad temperature of less than 2°F, with onl slight differences in the transients. The impact of the 2-D downcomer modeling for a 4 loop 15 x 15 fuel plant using the Feb 78 model is discussed in WCAP-952 This study also shows a small sensitivity to the downcomer modeling.

Table 2 summarizes the downcomer modeling sensitivities performed for the February, 1978 model modified with UHI Software Technology. This sensitivity w performed for a 4 loop, 3411 Mwt plant with 17 x 17 optimized fuel assemblies. The accumulator/safety injection interaction change was included in these cases These cases include a comparison of the UHI technology cases with the February 1978 model. A case using the UHI Software Technology version with a one-dimen- sional downcomer was also run with a discharge coefficient of 0.4. This case showed an end of bypass consistent with other analyses.

These cases show that the biggest change in the results are due to the UHI technology, with a peak clad temperature benefit of approximately 300°F. Chan ing from the one-dimensional to the two dimensional downcomer resulted in only an eleven oF change in PCT.

Table I

REFLOOD MODE

FQT	C_D	PCT	TIME	LOCATION
1.93	0.6	1974	104.6	7.5
1.93	0.8	2025	106.8	7.5
2.20	0.6	2048	134.6	7.5
2.20	0.8	1985	143.8	7.75

BURST MODE

PCT	TIME	LOCATION	MODEL
1859	47.13	5.75	FEB 78
2156	51.0	5.76	FEB 78
1827	50.82	6.25	UHI TECHNOLOGY
1737	50.65	6.25	UHI TECHNOLOGY

TABLE 2

	Feb 78 Analysis CD = 0.6 FQ = 2.32	UHI Technology 1-D downcomer CD = 0.6 FQ = 2.32	UHI Technology 2-D downcomer CD = 0.6 FQ = 2.32
End of ECC bypass, sec	26.5	25.1	26.6
End of blowdown, sec	28.6	25.4	26.8
Bottom of core recovery, sec	40.7	38.0	38.0
End of blowdown clad temp. @ 7.5' °F	1474	1230	1250
BOC clad temp at 7.5' °F	1699	1514	1502
Peak clad temperature, °F Location, Ft.	2089 7.5	1791 7.5	1780 7.6

3. Optimized Fuel FLECHT Heat Transfer

Westinghouse submitted WCAP-9500 for approval and Westinghouse has answ[illegible] review questions on the use of the FLECHT correlation for optimized fuel. Westinghouse has submitted supplementary information (Attachment 2) descri[illegible] the adjustment to the FLECHT correlation for optimized fuel rods. Westingh[illegible] believes that this adjustment is simple and in compliance with Appendix K and will require minimal additional NRC review. Analyses using this adjustment have already been provided to three licensees.

4. Accumulator/Safety Injection Interaction

Westinghouse has informed the NRC of a change that was made to accurately model the interaction between pumped safety injection flow and accumulator injection flow (Attachment 3). This change resulted in an increase in peak clad temperatures greater than 20°F for some plants. This change was reviewed by the Westinghouse Safety Review Committee and found not to be reportable as an Unreviewed Safety Question, Substantial Safety Hazard on Significant Deficiency based on the application of unused benefits including the 65°F reduction in initial pellet temperature (see SER on WCAP-8720 dated March 27, 1980) and taking into account the water in the accumulator surge line upstream of the check valves.

Westinghouse believes that sufficient information was provided in Attachment 3 for the NRC review. The adjustment is straight forward and necessary for compliance to Appendix K. All analyses performed since December, 1980 incorporate these adjustments.

TABLE 1

LARGE BREAK

TIME SEQUENCE OF EVENTS

	DECL C_D = 0.8 (Sec)	DECL C_D = 0.6 (Sec)	DECL C_D = 0.4 (Sec)
START	0.0	0.0	0.0
Rx Trip Signal	0.554	0.559	0.569
S. I. Signal	0.92	1.06	1.31
Acc. Injection	11.6	13.8	18.3
End of Blowdown	27.59	30.58	36.94
Bottom of Core Recovery	42.96	46.08	53.84
Acc. Empty	57.48	60.67	66.20
Pump Injection	25.92	26.06	26.31
End of Bypass	27.59	30.58	36.94

TABLE 2

LARGE BREAK

	DECLG C_D = 0.8	DECLG C_D = 0.6	DECLG C_D = 0.4
Results			
Peak Clad Temp. °F	1937	1989	2039
Peak Clad Location Ft.	7.5	7.5	7.5
Local Zr/H_2O Rxn (max)%	2.85	3.3	5.10
Local Zr/H_2O Location Ft.	7.5	7.5	5.75
Total Zr/H_2O Rxn %	< 0.3	< 0.3	< 0.3
Hot Rod Burst Time Sec.	35.2	37.6	40.2
Hot Rod Burst Location Ft.	5.75	6.0	5.75

Calculation	
NSSS Power Mwt 102% of	3025
Peak Linear Power Kw/ft 102% of	13.74
Peaking Factor (At License Rating)	2.20
Accumulator Water Volume	800 ft^3

Fuel region + cycle Analyzed	Cycle	Region
Unit 1		
Unit 2 (if applicable)		

TABLE 2a

LARGE BREAK

Plots for This Case are 1D, 2D, 3D, 7D, & 8D

	DECLG C_D=	DECLG C_D= 0.4 F_q= 2.14	DECLG C_D=
Results			
Peak Clad Temp. °F		1995	
Peak Cald Location Ft.		7.5	
Local Zr/H_2O Rxn (max)%		3.38	
Local Zr/H_2O Location Ft.		5.75	
Total Zr/H_2O Rxn %	< 0.3	< 0.3	<0.3
Hot Rod Burst Time Sec		41.6	
Hot Rod Burst Location Ft.		5.75	

Calculation	
NSSS Power Mwt 102% of	3025
Peak Linear Power kw/ft 102% of	13.36
Peaking Factor (At License Rating)	2.14
Accumulator Water Volume	800 ft^3

Fuel region + cycle analyzed	Cycle	Region
Unit 1		
Unit 2 (If applicable)		

TABLE 3

CONTAINMENT DATA (DRY CONTAINMENT)

Net Free Volume	2.61×10^6 Ft^3
Initial Conditions	
Pressure	14.7 psia
Temperature	90.0 °F
RWST Temperature	40.0 °F
Service Water Temperature	35.0 °F
Outside Temperature	-20.0 °F
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

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 0.0250	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 (con't)

PAINTED STRUCTURAL HEAT SINK DATA

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

TABLE 4

REFLOOD MASS & ENERGY RELEASES

Indian Point Unit #3

DECLG C_0 = 0.4

TIME (Sec)	M (TOTAL) (LBm/Sec)	MH(TOTAL) (BTU/Sec)
53.836	0.0	0.0
61.696	38.14	4.92 E + 4
70.696	121.67	1.49 E + 5
83.796	128.87	1.55 E + 5
99.196	316.87	2.11 E + 5
115.496	362.14	2.18 E + 5
132.996	370.32	2.13 E + 5
171.196	380.11	2.01 E + 5
213.996	389.23	1.88 E + 5
262.496	399.25	1.72 E + 5

THE BROKEN LOOP INJECTION SPILL DURING BLOWDOWN IS

TIME	MASS	ENERGY	ENTHALPY
0.000	3005.129	179165.764	59.620
1.010	2700.410	160998.419	59.620
2.010	2476.518	147650.025	59.620
3.010	2301.243	137200.107	59.620
4.010	2158.746	128704.457	59.620
5.010	2039.829	121614.621	59.620
6.010	1938.318	115562.534	59.620
7.010	1850.354	110318.104	59.620
8.010	1772.770	105692.553	59.620
9.010	1703.717	101575.597	59.620
10.010	1641.440	97862.657	59.620
11.010	1584.302	94485.876	59.620
12.010	1532.936	91393.668	59.620
13.010	1485.197	88547.475	59.620
14.010	1441.073	85916.747	59.620
15.010	1400.175	83478.432	59.620
16.010	1362.149	81211.339	59.620
17.010	1326.656	79095.216	59.620
18.010	1293.413	77113.294	59.620
19.010	1262.183	75251.355	59.620
20.010	1232.764	73497.333	59.620
21.010	1205.227	71855.606	59.620
22.010	1179.258	70307.363	59.620
23.010	1154.749	68846.165	59.620
24.010	1131.436	67456.228	59.620
25.010	1109.384	66141.482	59.620
26.010	1161.963	63453.057	54.609
27.010	1143.790	62285.219	54.455
28.010	1126.490	61176.380	54.307
29.010	1110.020	60120.675	54.162
30.010	1094.348	59115.656	54.019
31.010	1079.686	58172.960	53.880
32.010	1065.853	57281.980	53.743
33.010	1052.679	56432.833	53.609
34.010	143.886	1155.013	8.027
35.010	143.959	1155.595	8.027
36.010	144.035	1156.207	8.027

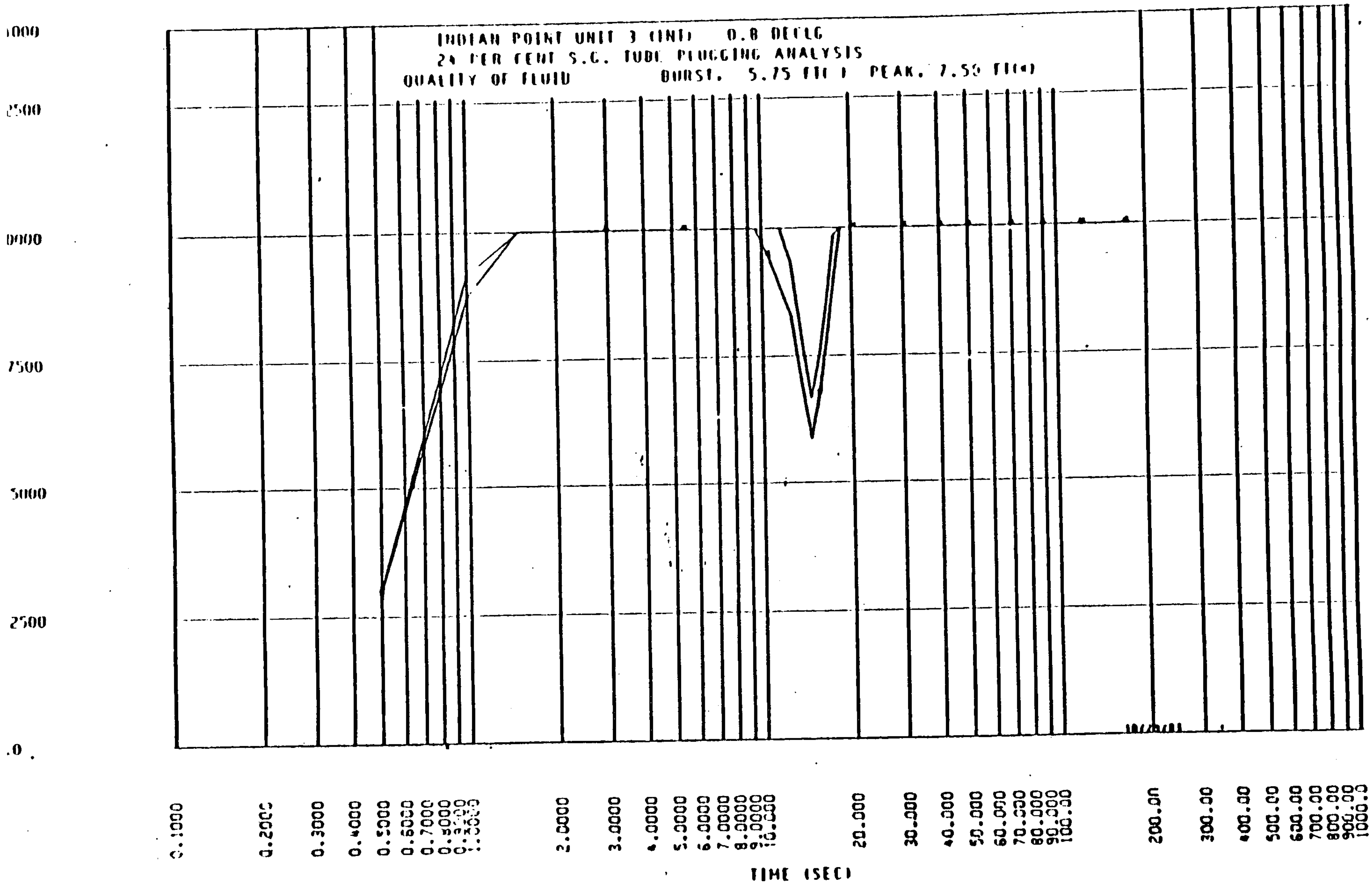


FIGURE 1A FLUID QUALITY
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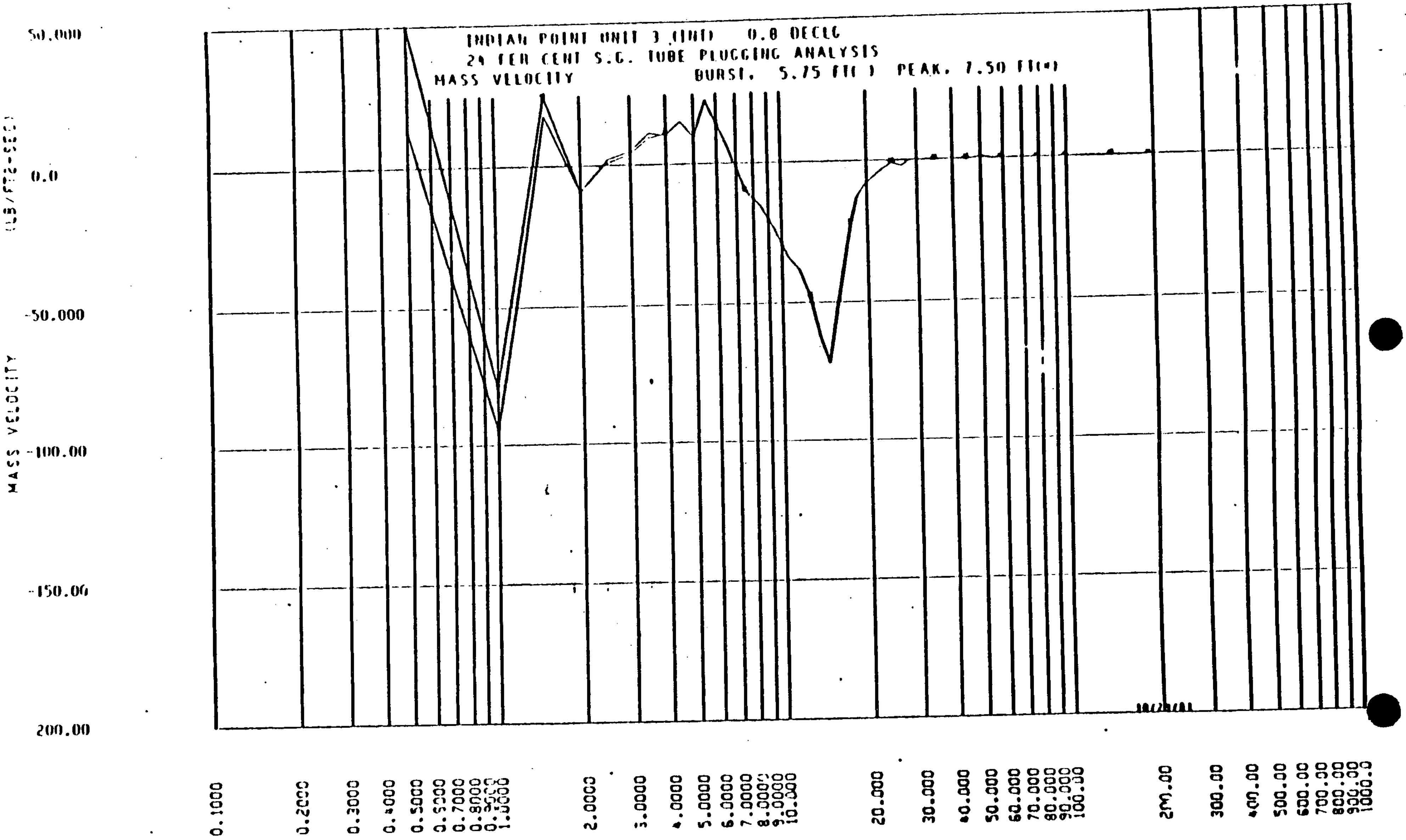

FIGURE 2A MASS VELOCITY
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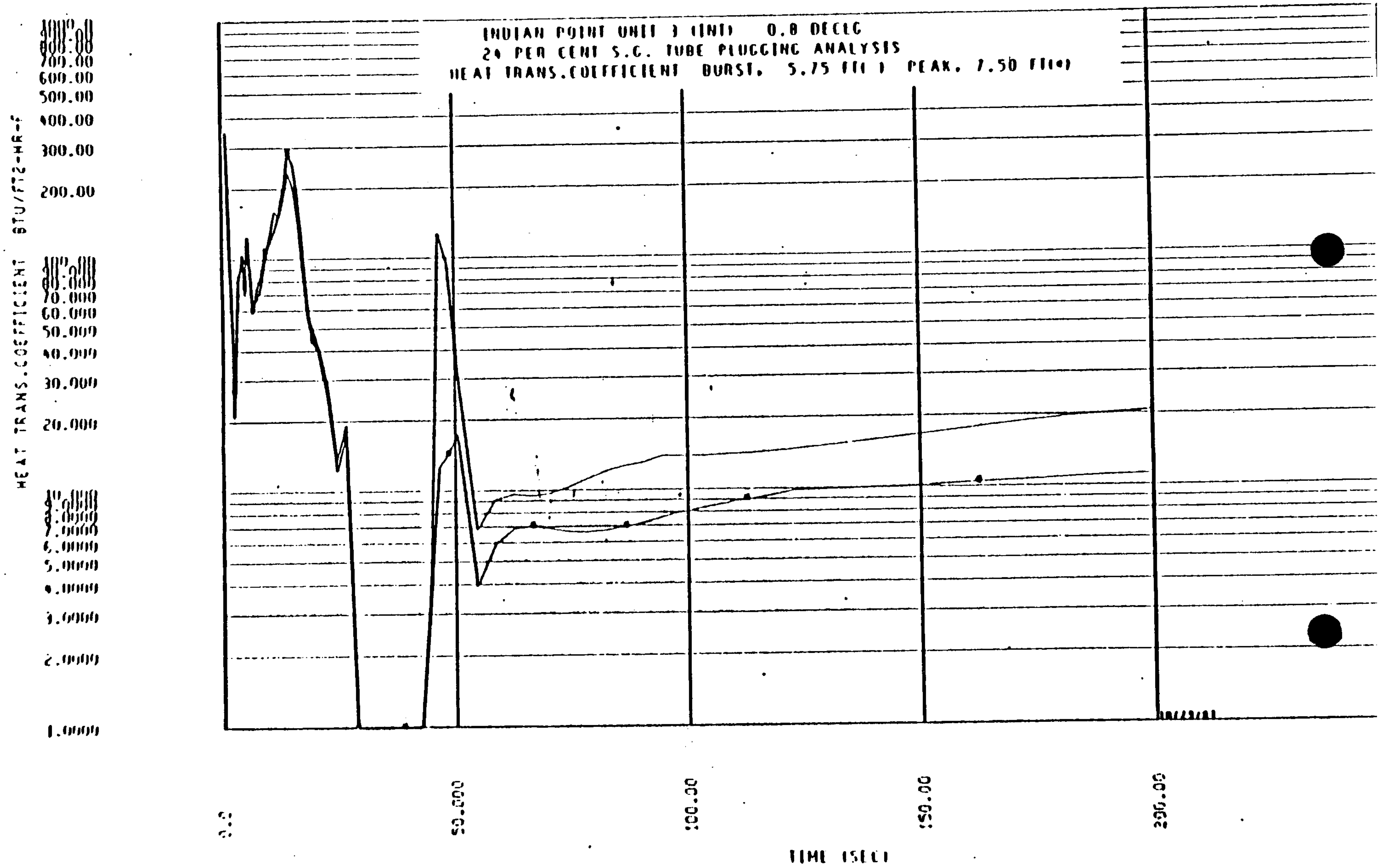

FIGURE 3A HEAT TRANSFER COEFFICIENT
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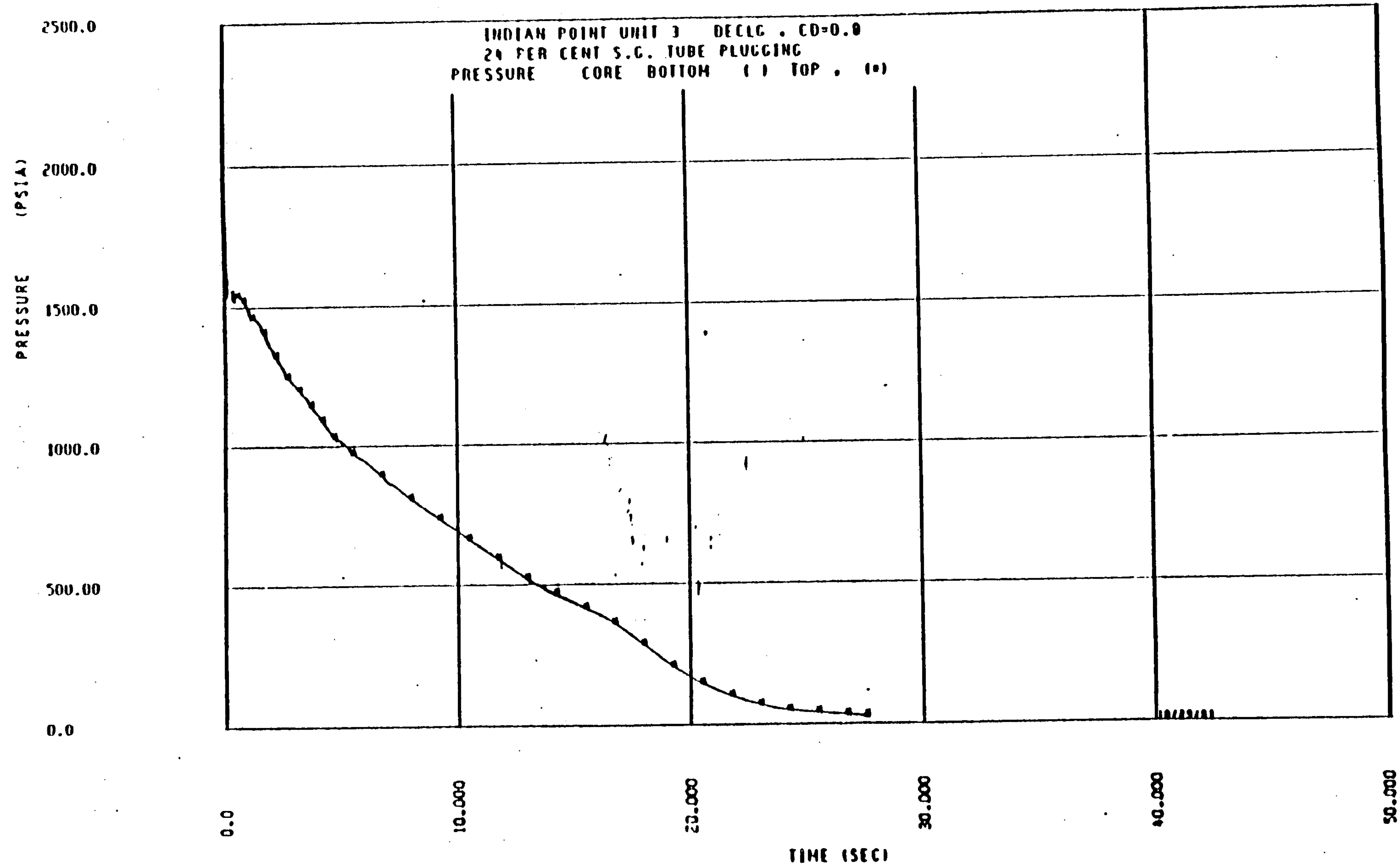

FIGURE 4A CORE PRESSURE
DECLG(CD = 0.8)

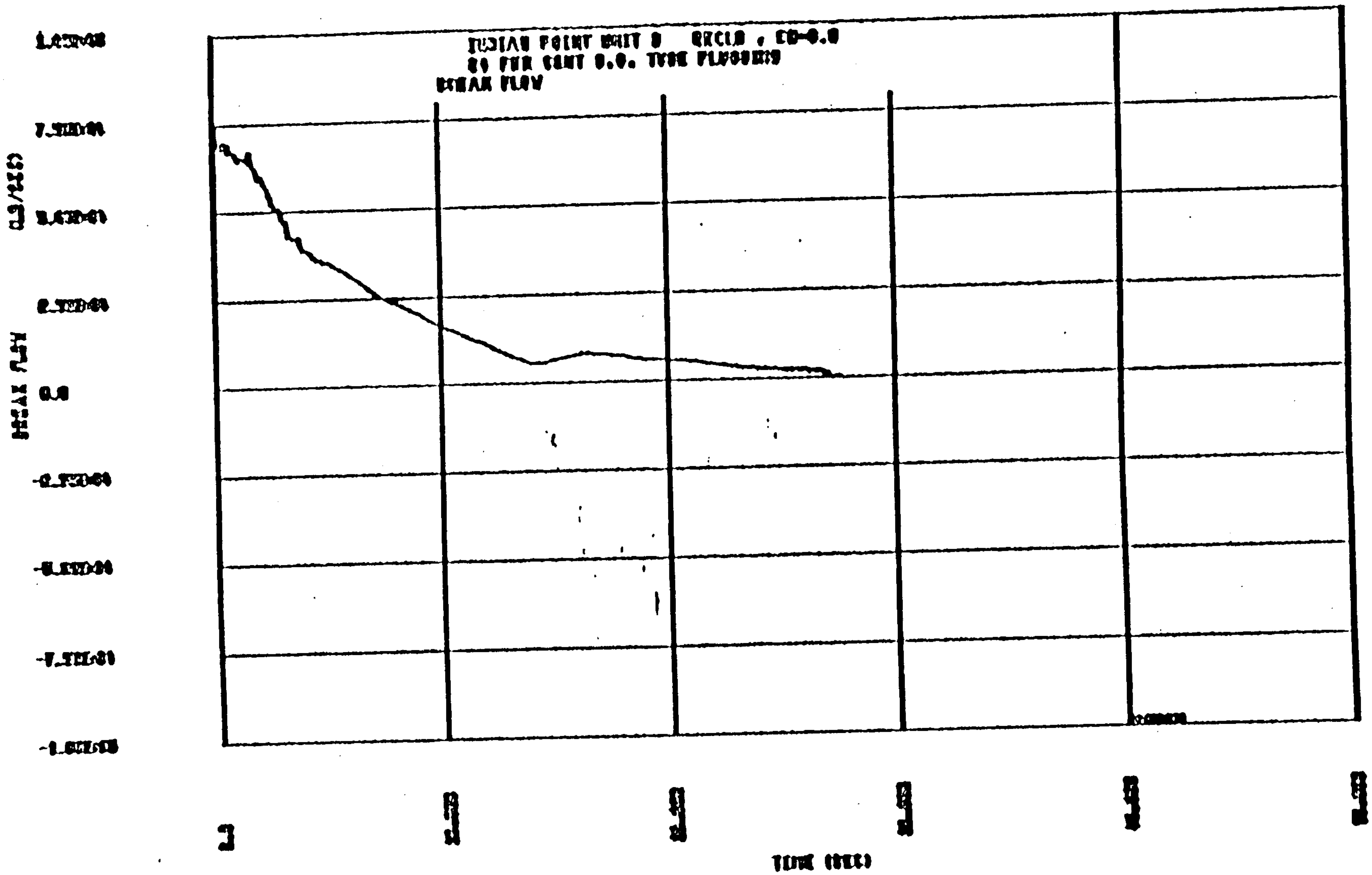

FIGURE 5A BREAK FLOW RATE
DECLG(CD = 0.8)

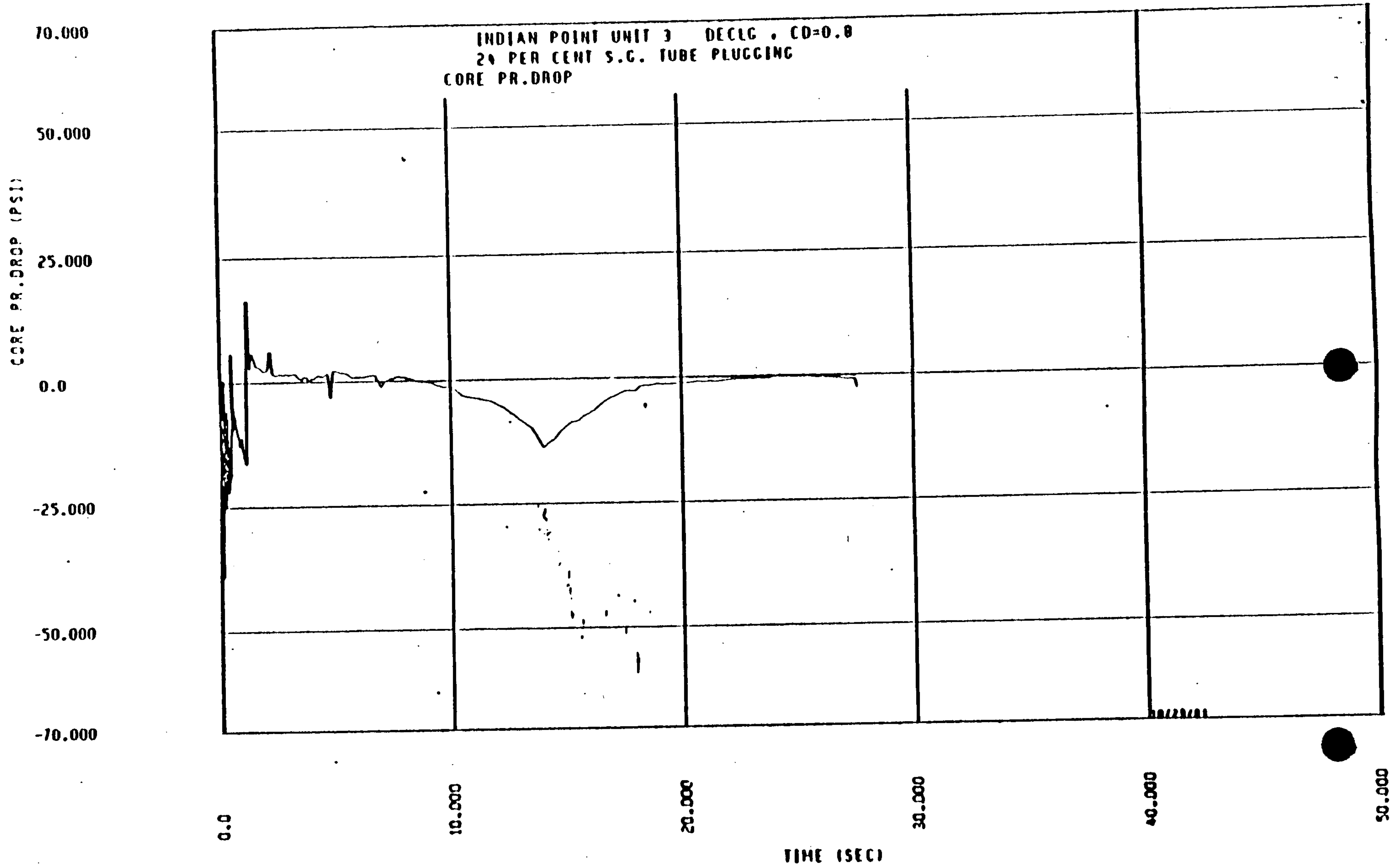

FIGURE 6A CORE PRESSURE DROP
DECLG(CD = 0.8)

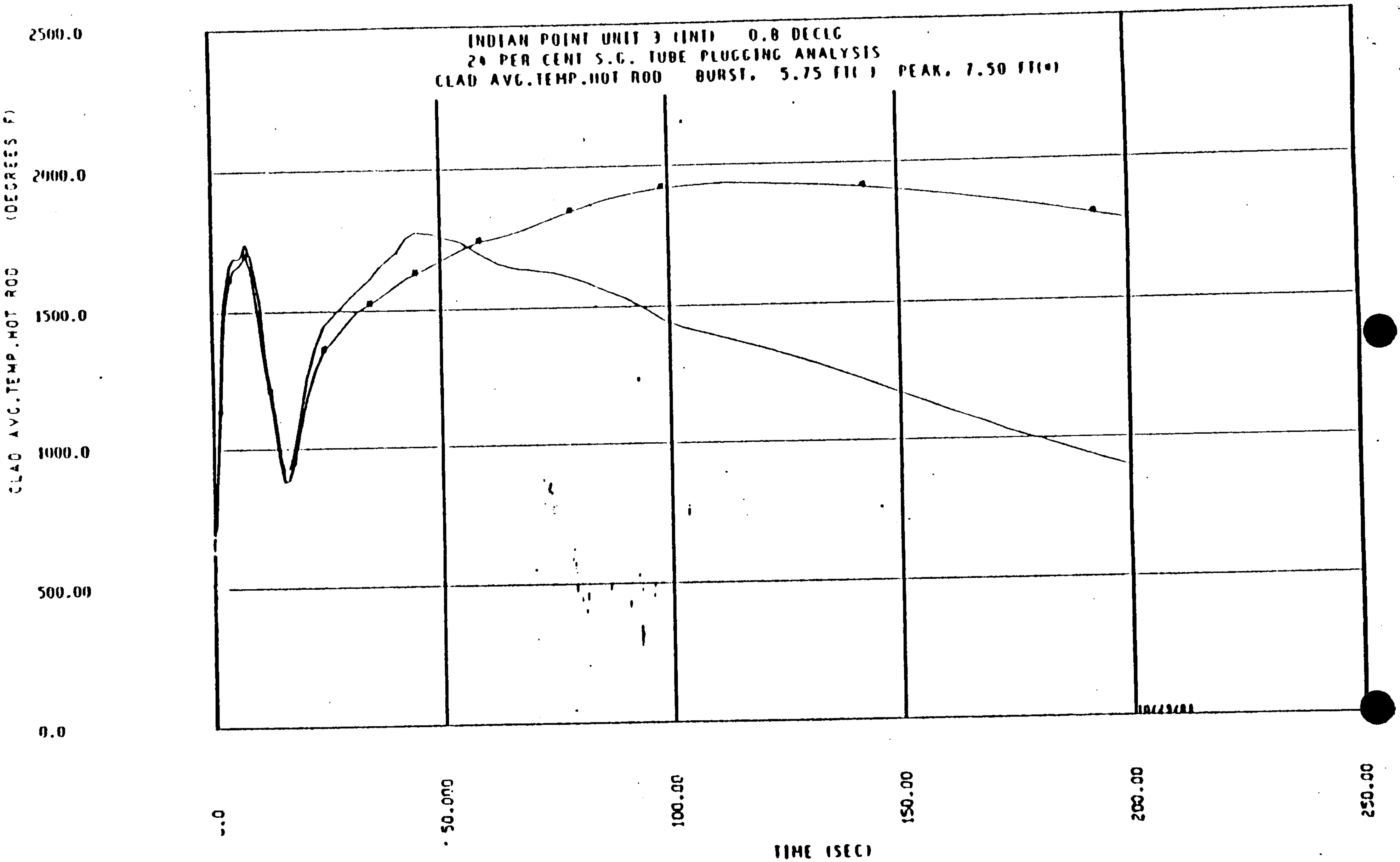

FIGURE 7A PEAK CLAD TEMPERATURE
DECLG(CD = 0.8)

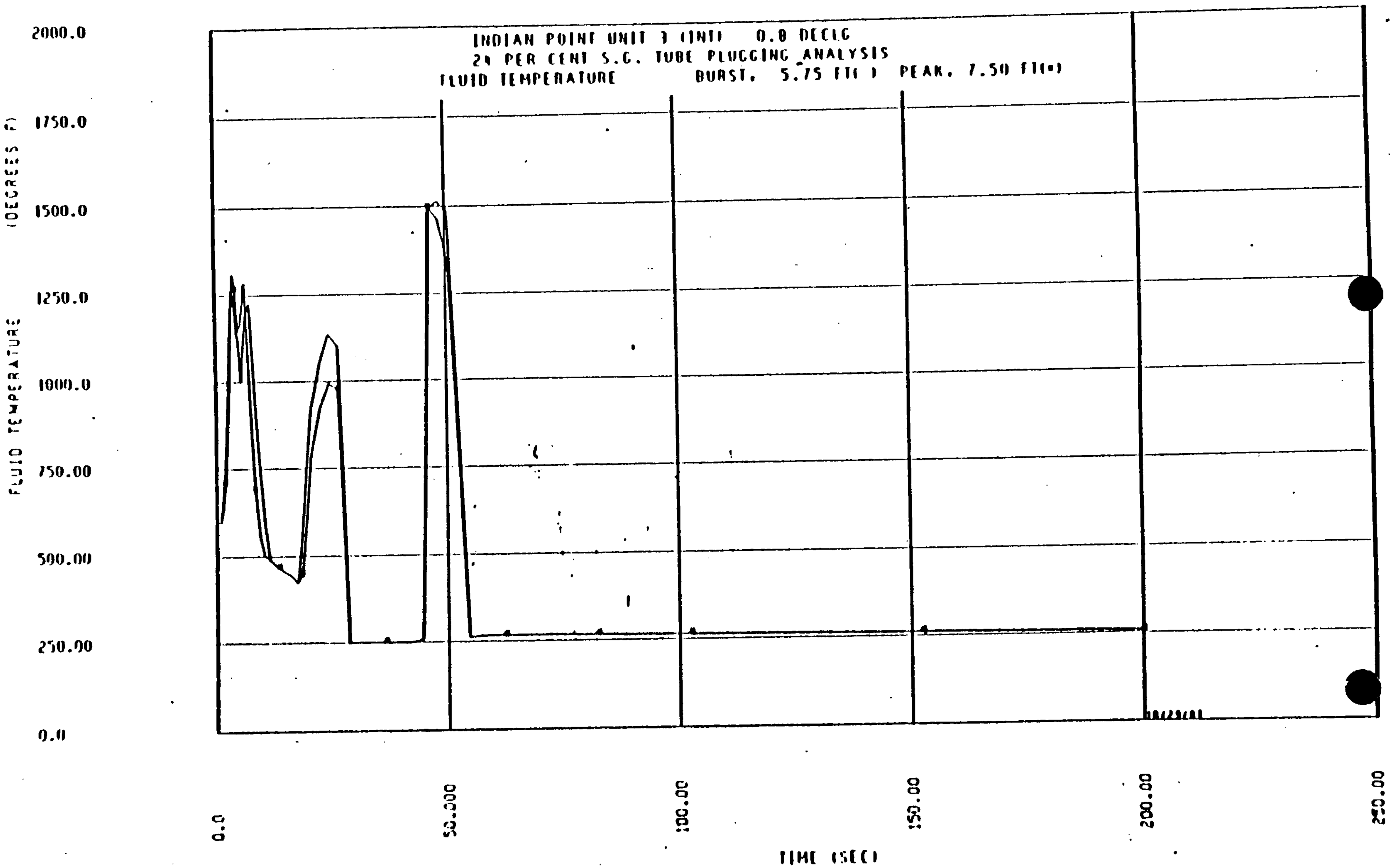

FIGURE 8A FLUID TEMPERATURE
DECLG(CD = 0.8)

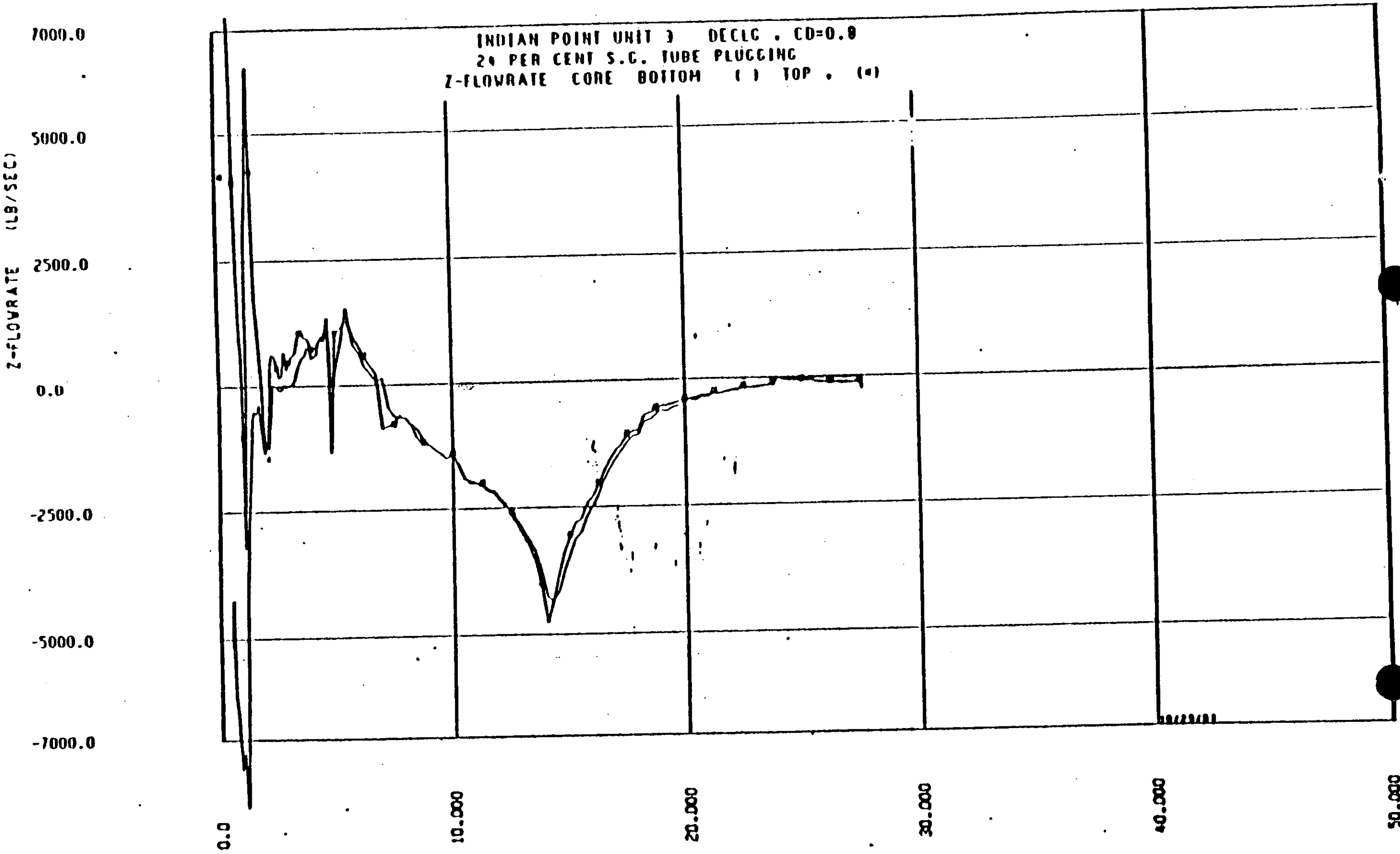

FIGURE 9A CORE FLOW (TOP AND BOTTOM)
DECLG(CD = 0.8)

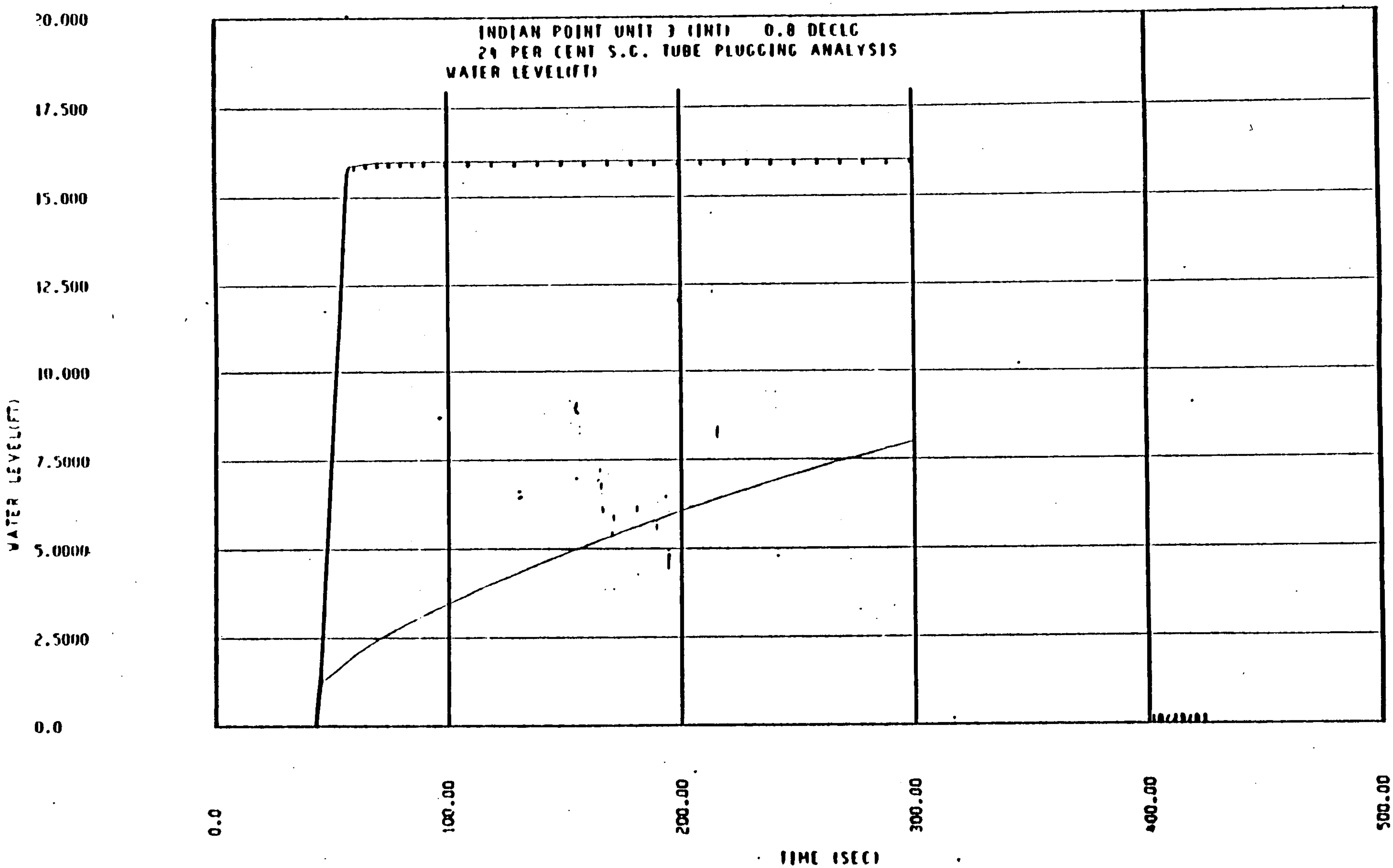

FIGURE 10A REFLOOD TRANSIENT - CORE & DOWNCOMER WATER LEVELS DECLG(CD = 0.8)

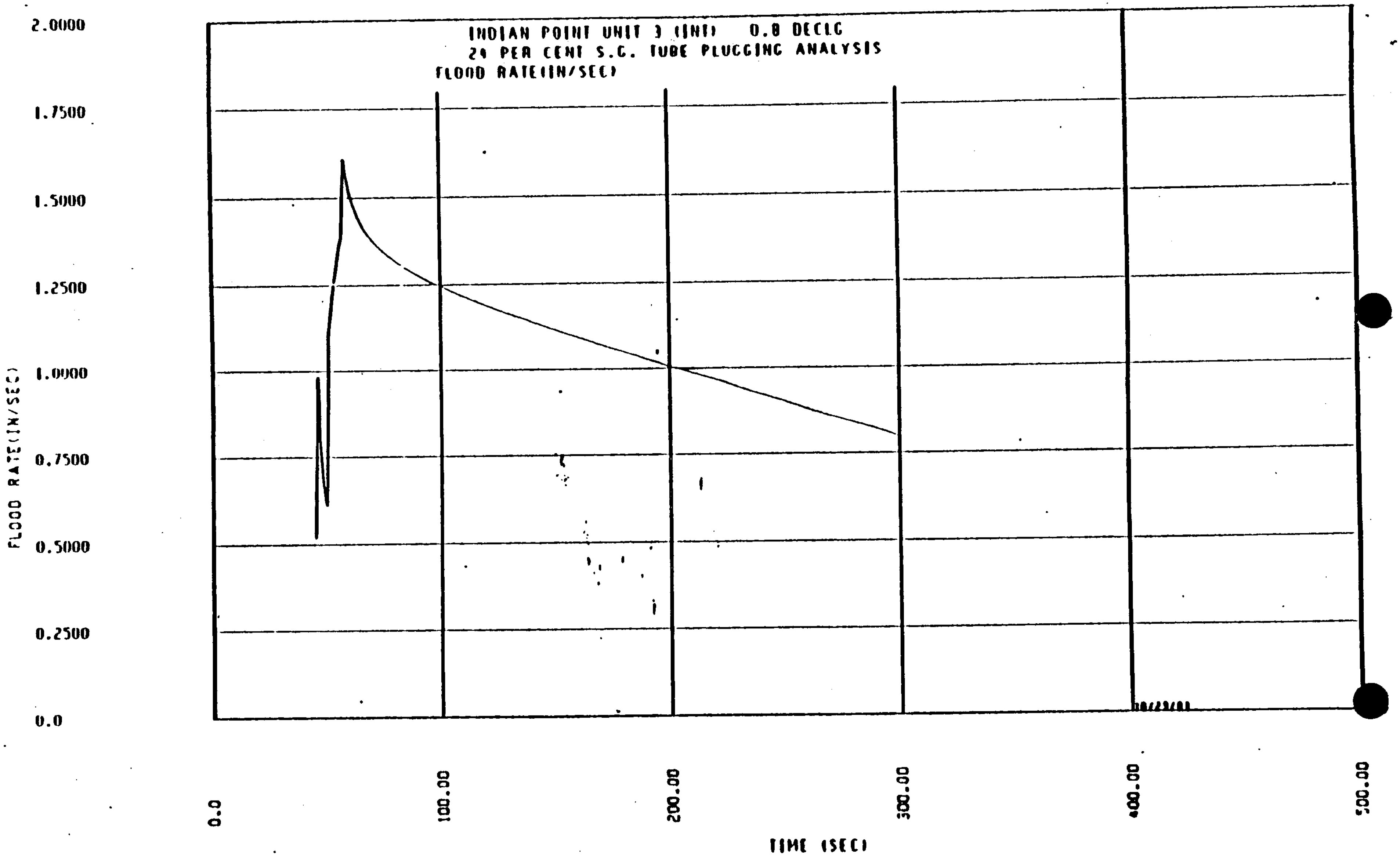

FIGURE 11A REFLOOD TRANSIENT
CORE INLET VELOCITY
DECLG(CD = 0.8)

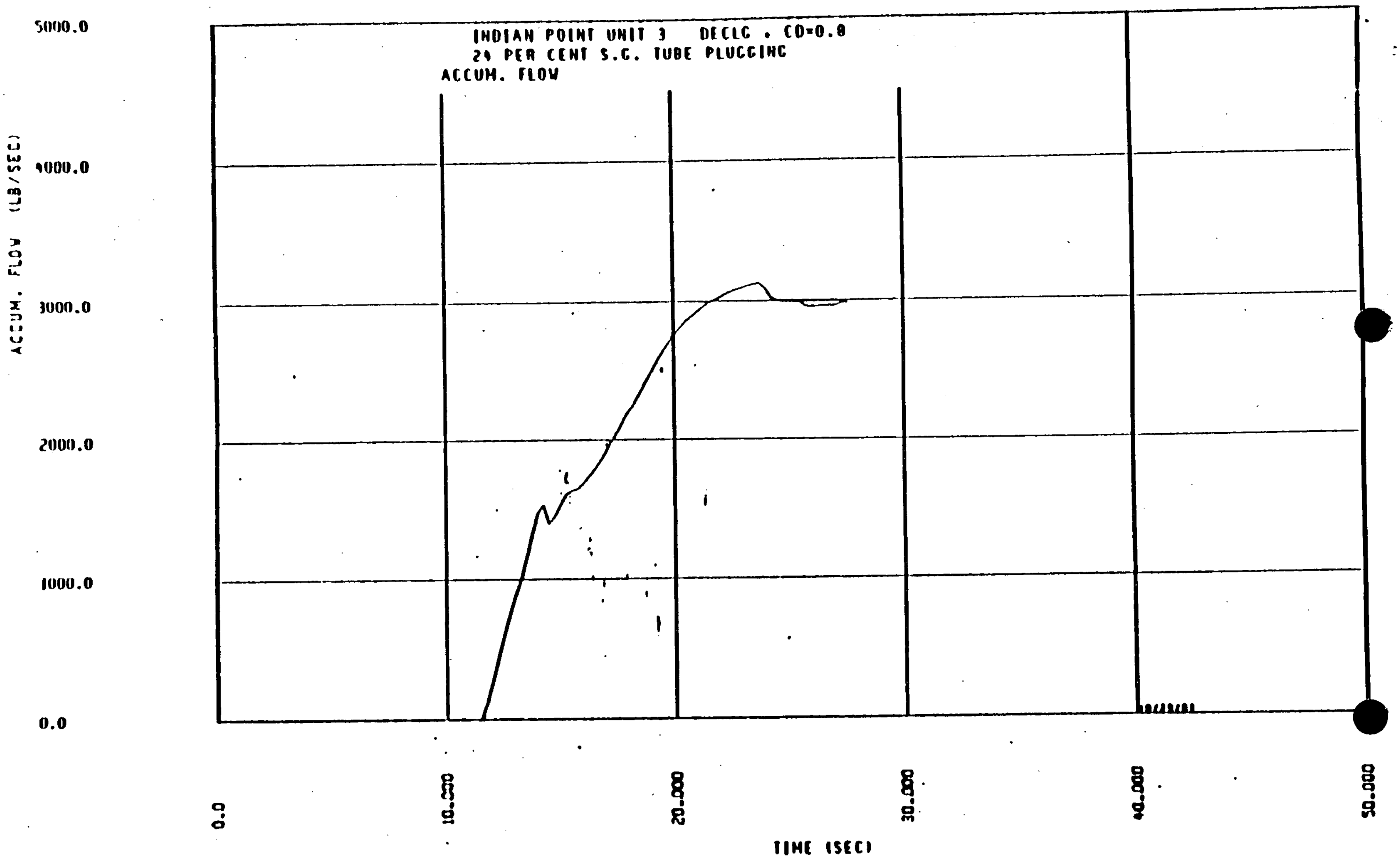

FIGURE 12A ACCUMULATOR FLOW(BLOWDOWN)
DECLG(CD = 0.8)

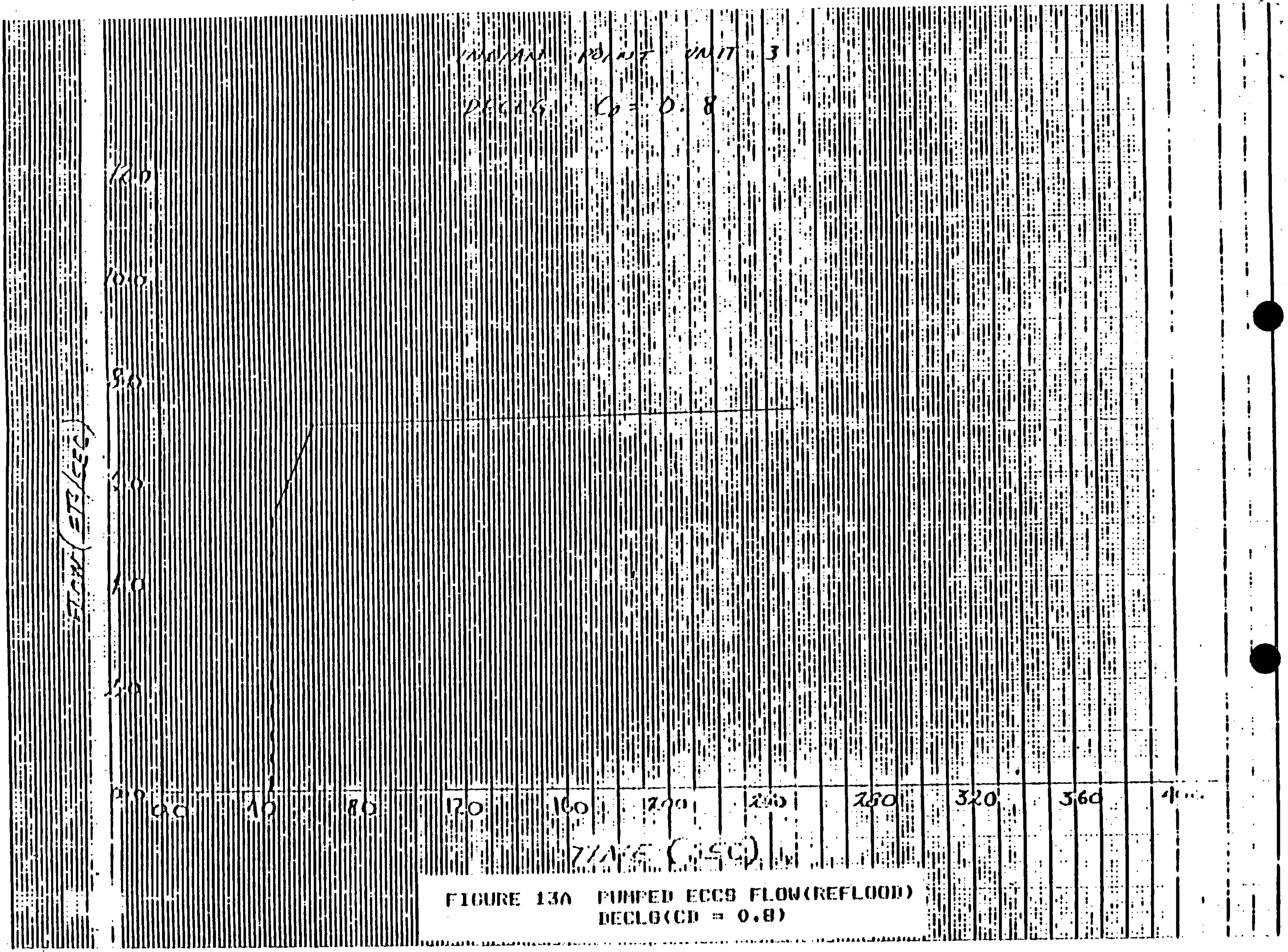

FIGURE 13A PUMPED ECCS FLOW(REFLOOD)
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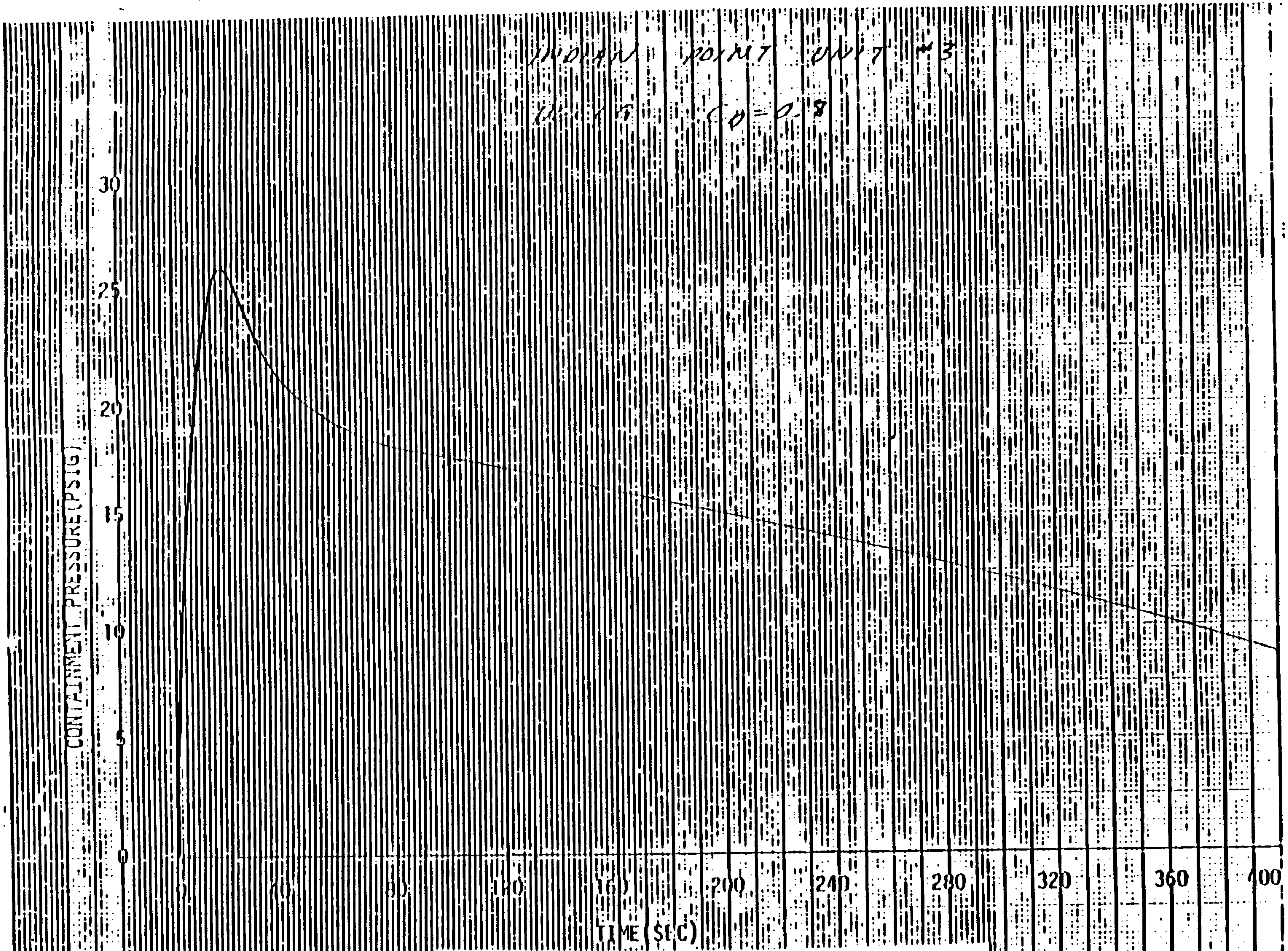

FIGURE 14A CONTAINMENT PRESSURE
DECLG(CD = 0.8)

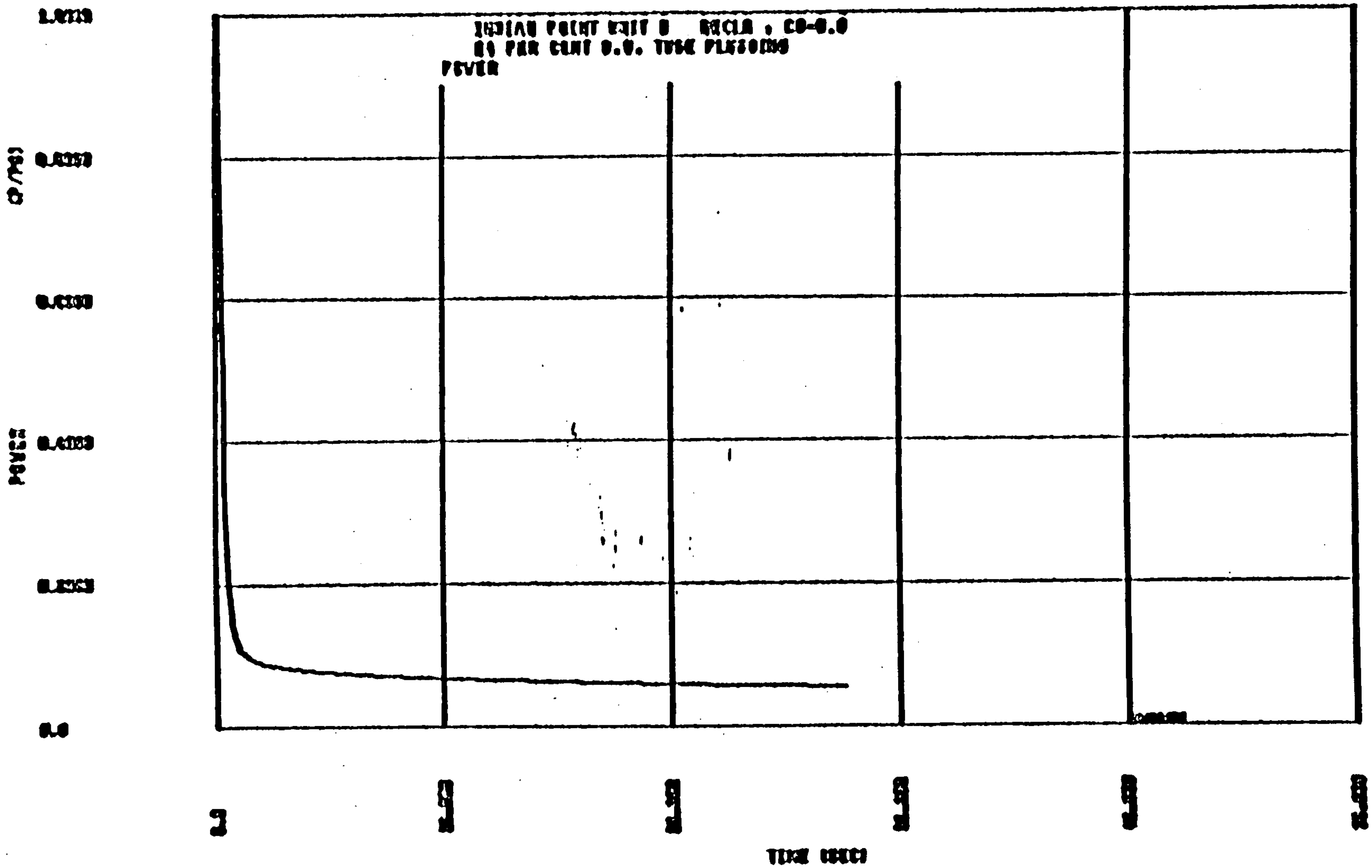

FIGURE 15A CORE POWER TRANSIENT
DECLG(CD = 0.8)

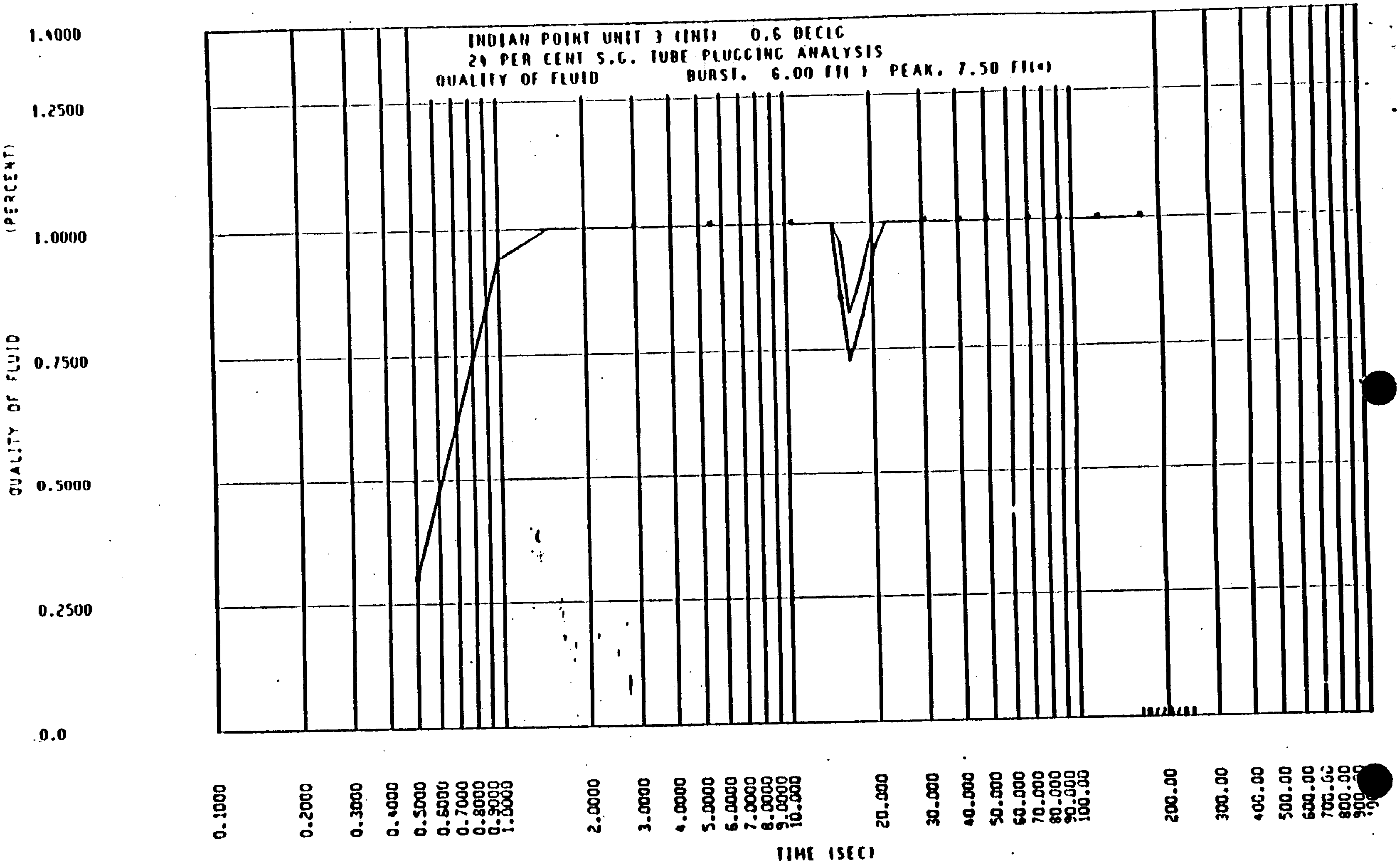

FIGURE 1B FLUID QUALITY
DECLG(CD = 0.6)

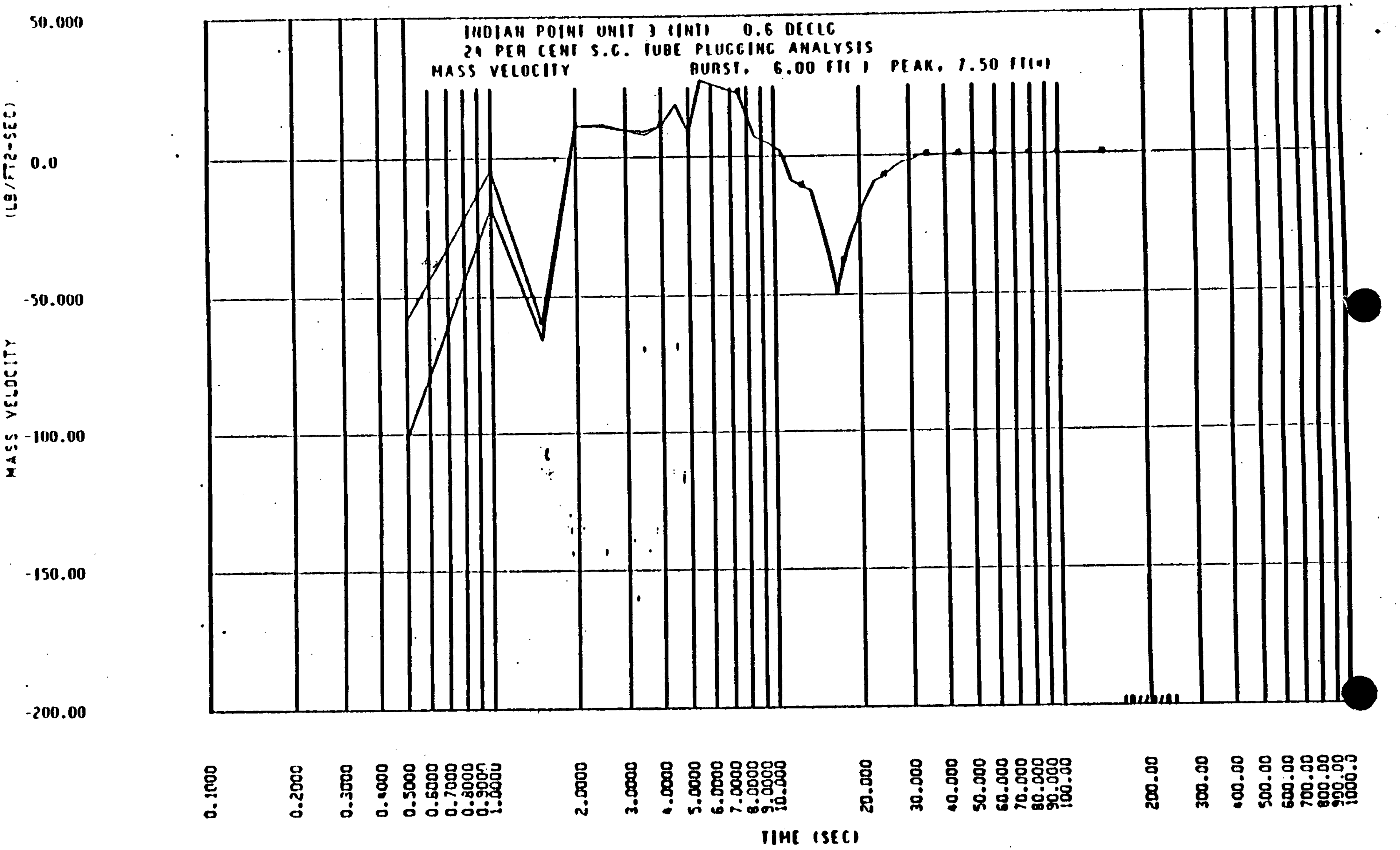

FIGURE 2D MASS VELOCITY
DECLG(CD = 0.6)

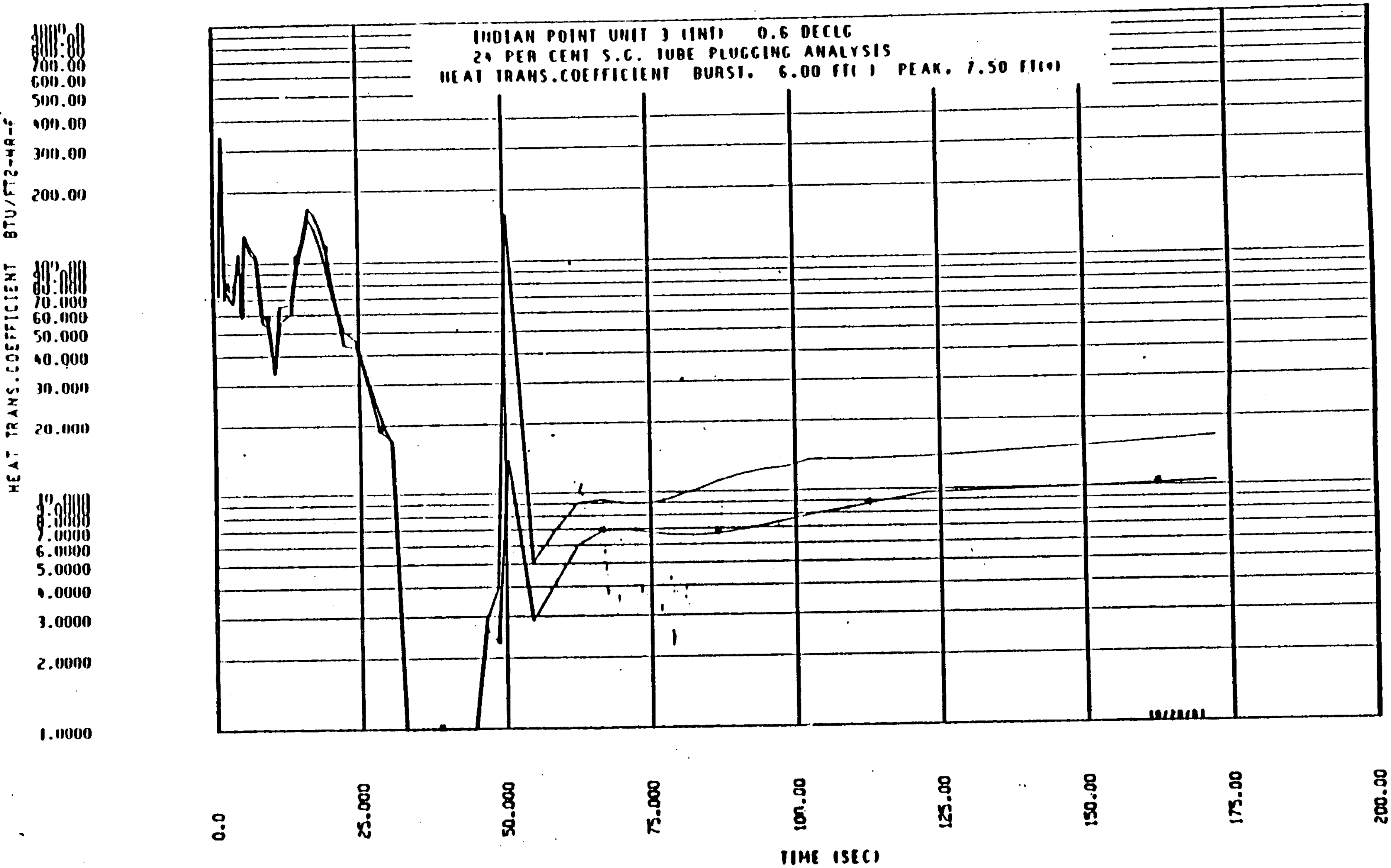

FIGURE 3D HEAT TRANSFER COEFFICIENT
DECLG(CD = 0.6)

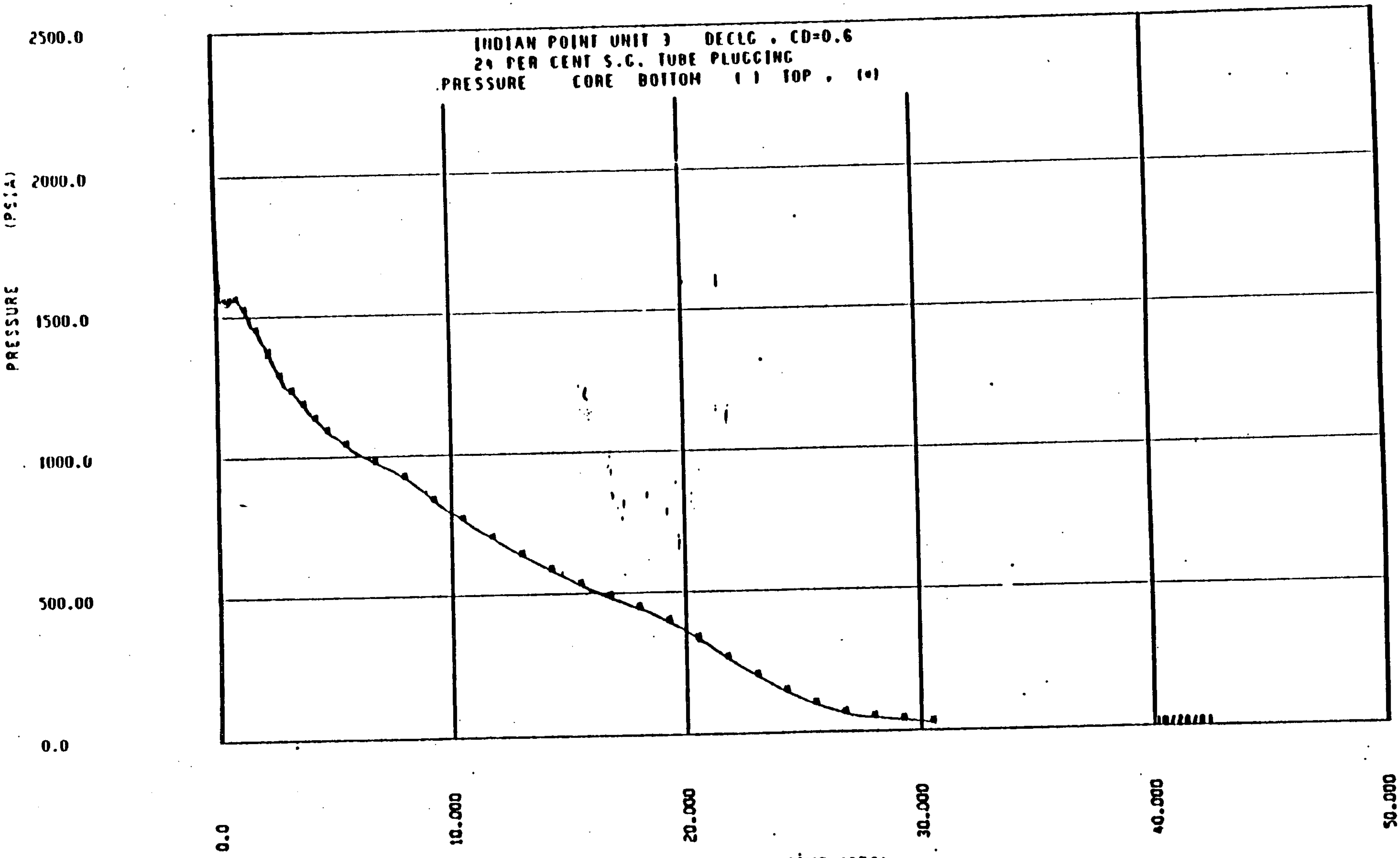

FIGURE 4B CORE PRESSURE
DECLG(CD = 0.6)

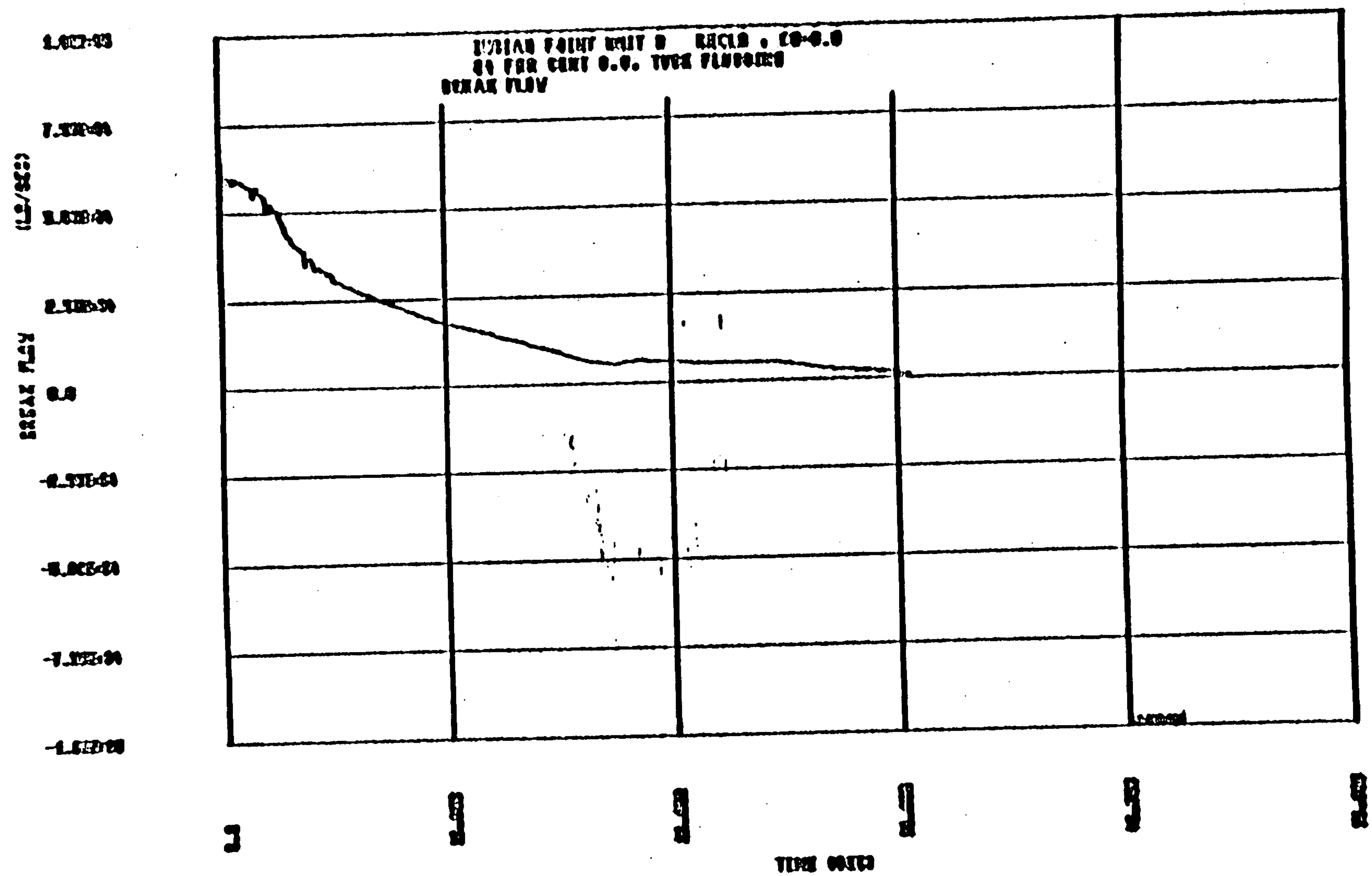

FIGURE 5D BREAK FLOW RATE
DECLO(CD = 0.6)

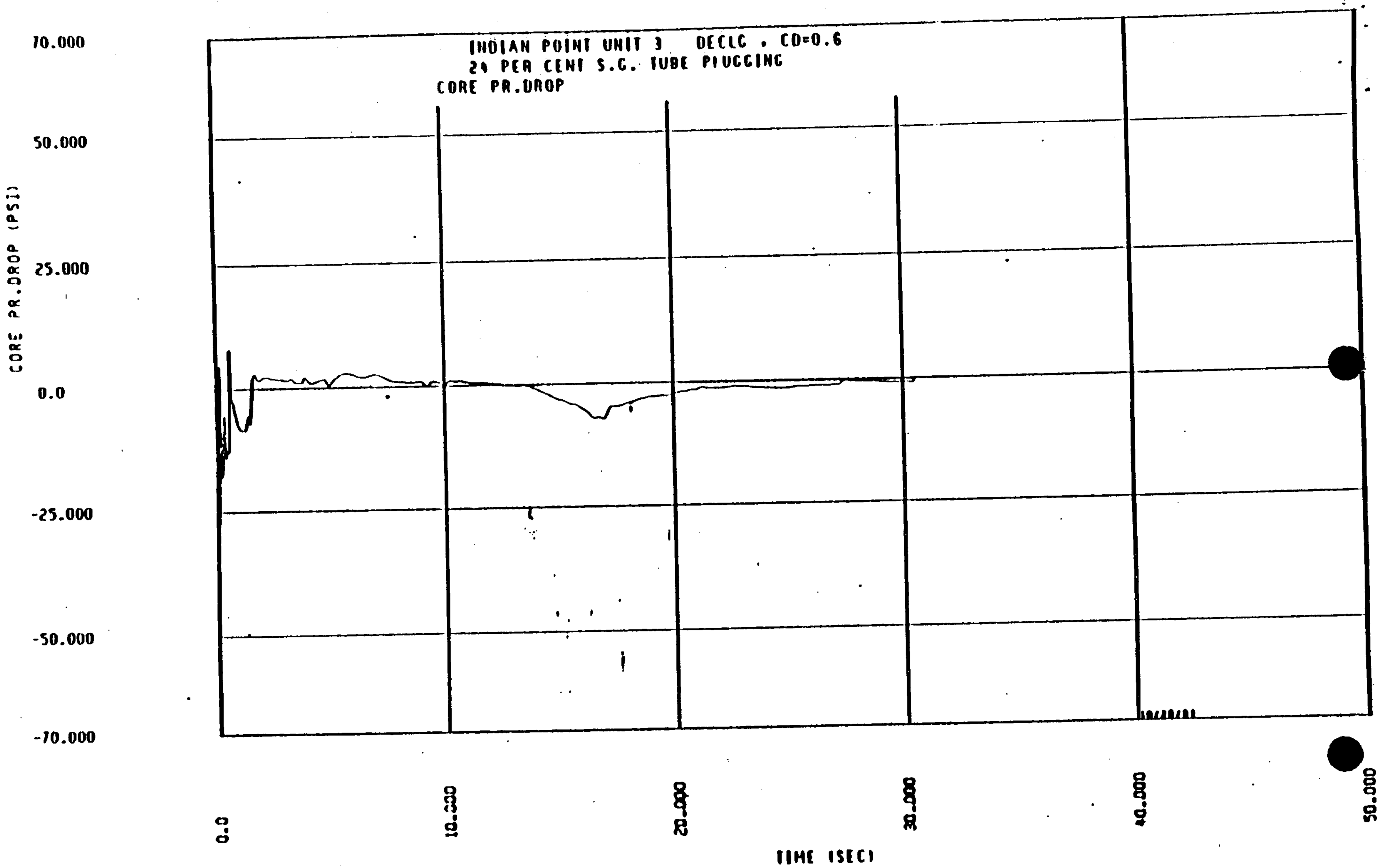

FIGURE 6D CORE PRESSURE DROP
DECLG(CD = 0.6)

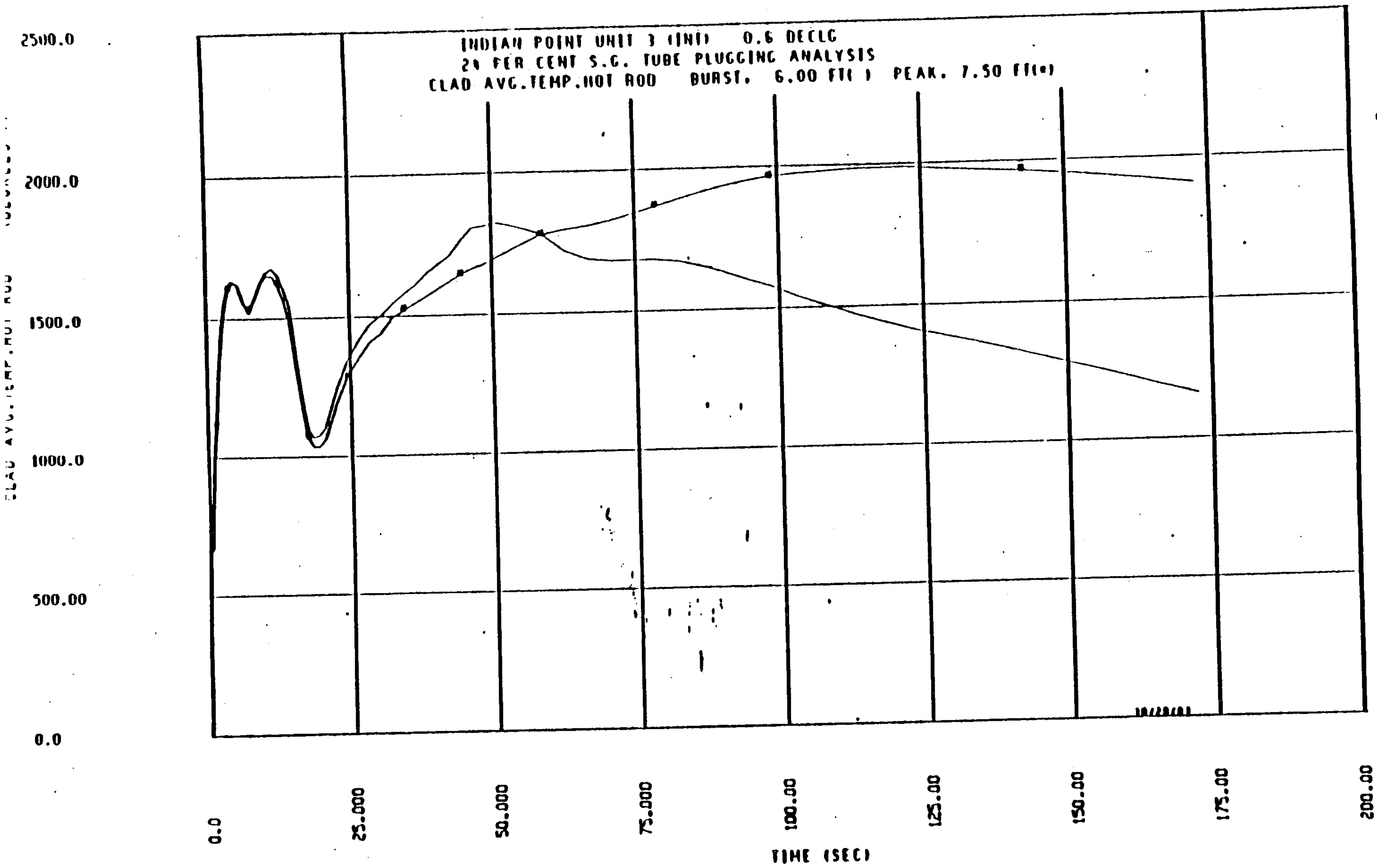

FIGURE 7B PEAK CLAD TEMPERATURE
DECLG(CD = 0.6)

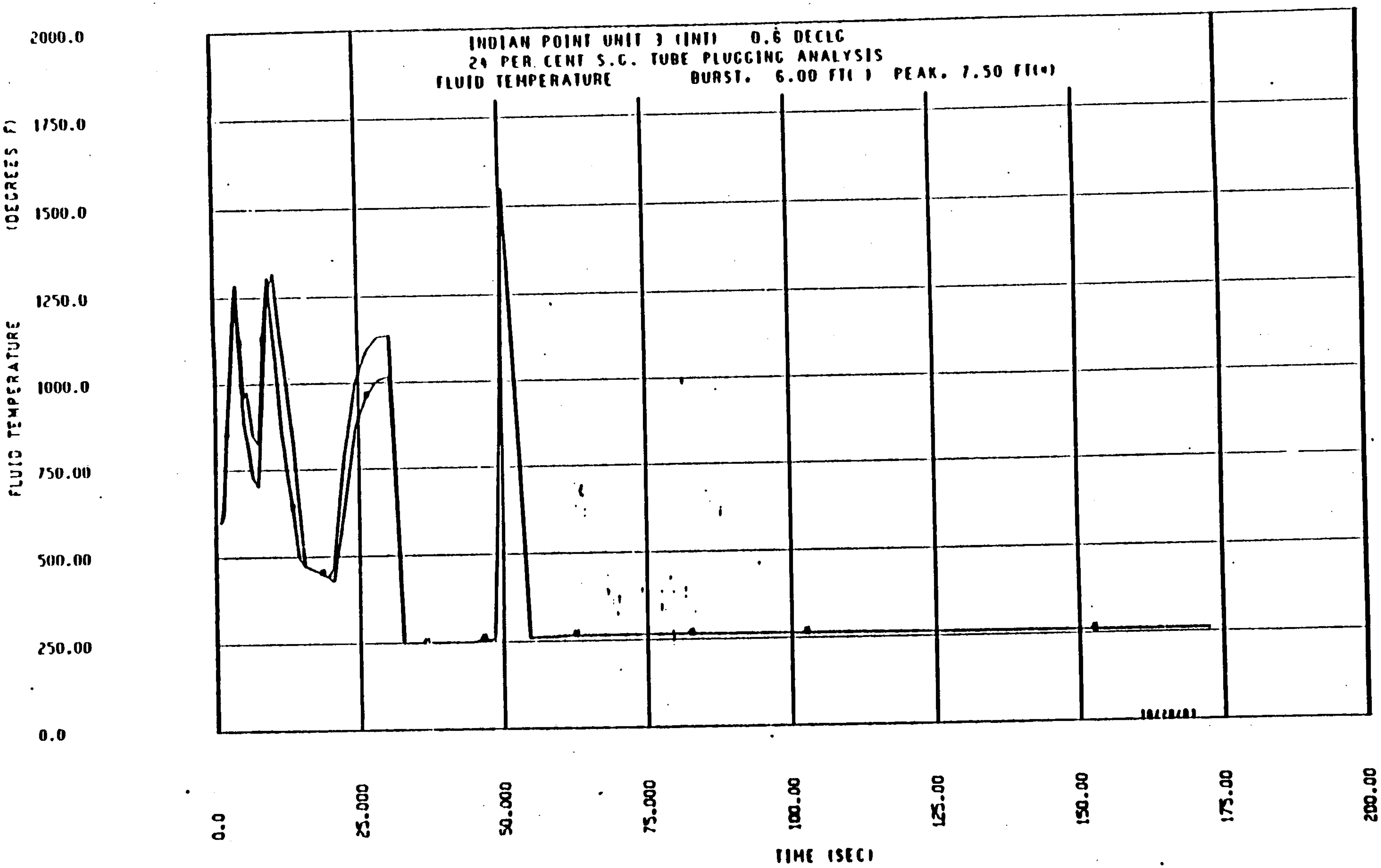

FIGURE 00 FLUID TEMPERATURE
DECLG(CD = 0.6)

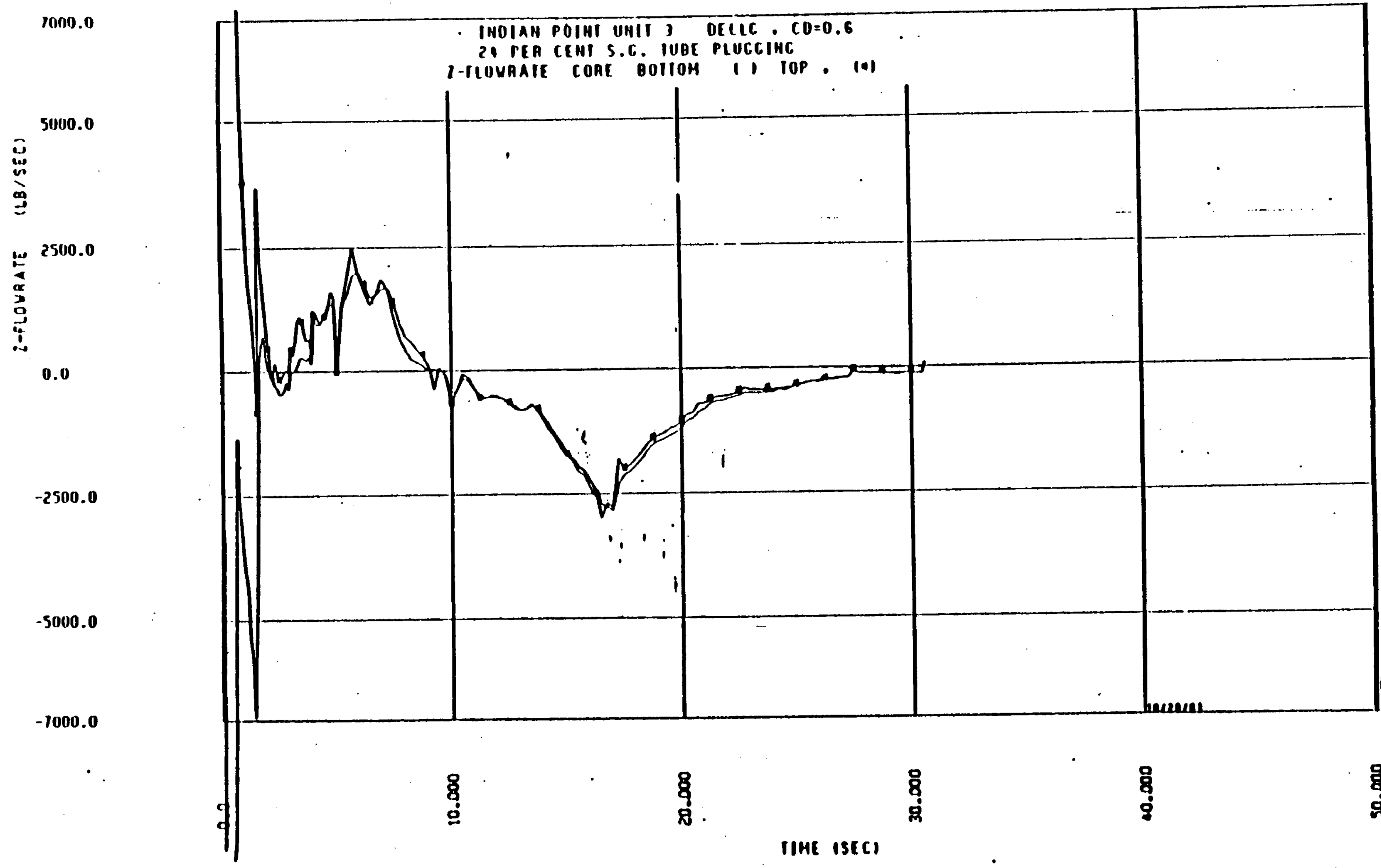

FIGURE 9D CORE FLOW (TOP AND BOTTOM)
DECLG(CD = 0.6)

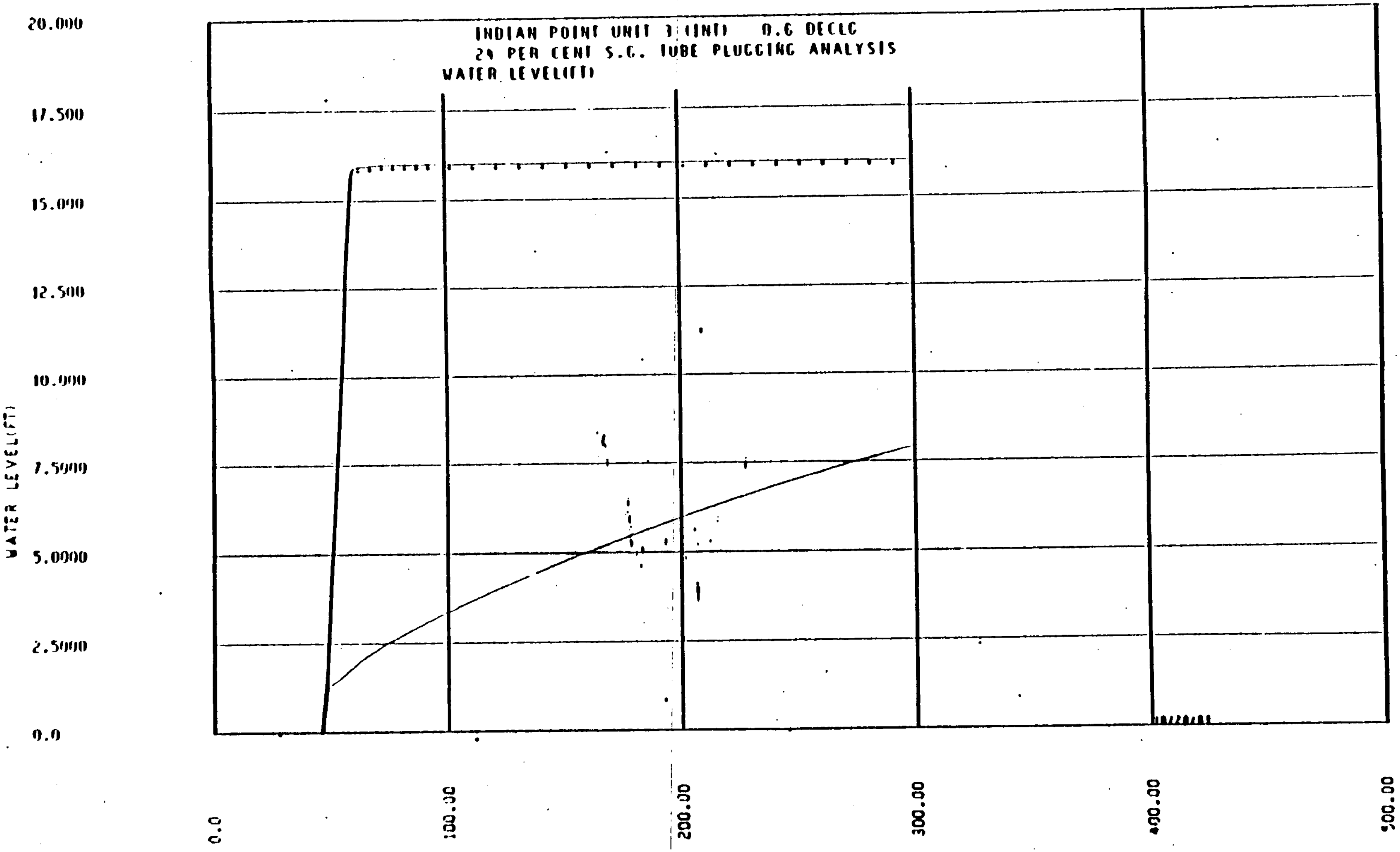

FIGURE 10B REFLOOD TRANSIENT - CORE & DOWNCOMER WATER LEVELS DECLG(CD = 0.6)

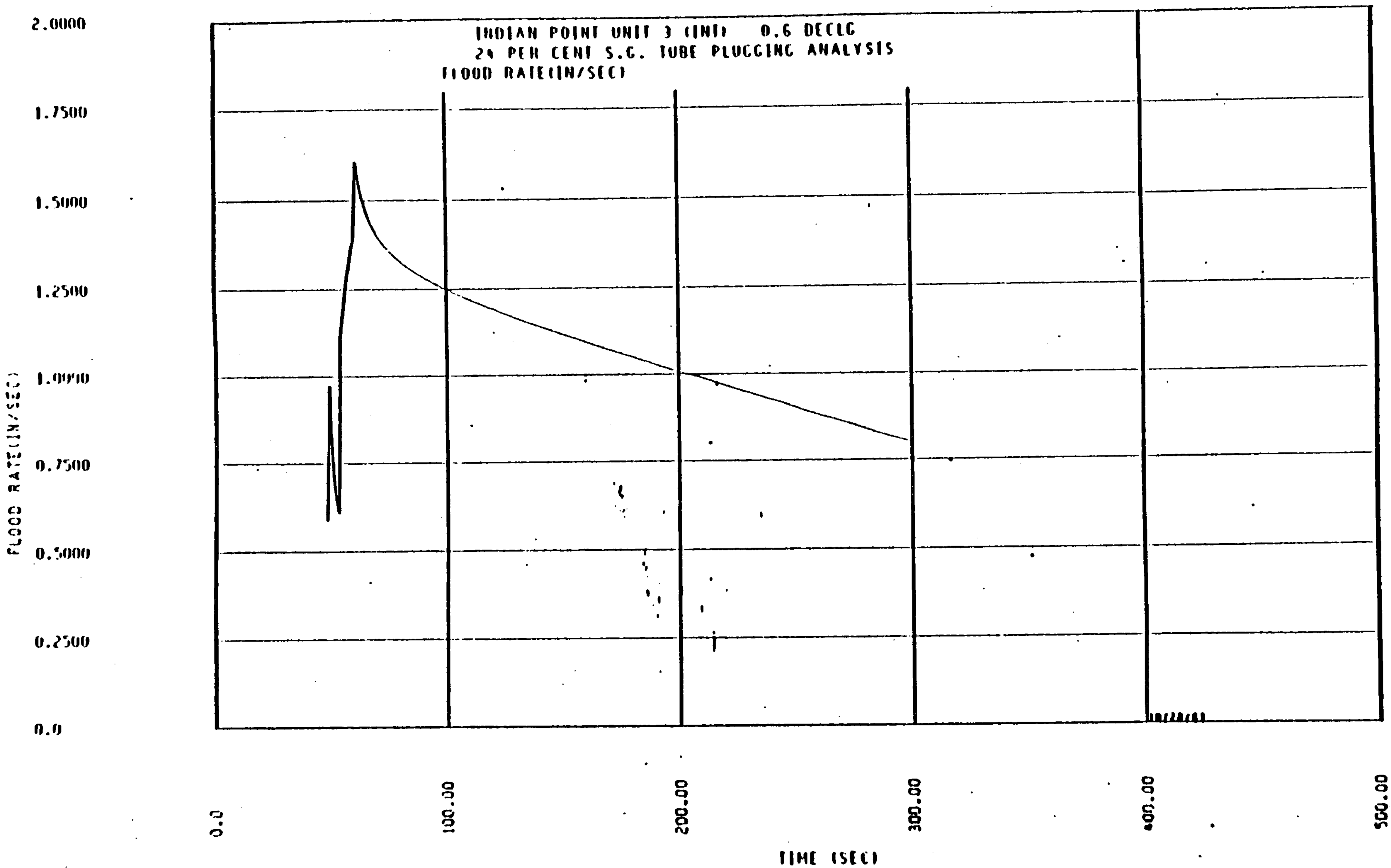

FIGURE 11D REFLOOD TRANSIENT
CORE INLET VELOCITY
DECLG(CD = 0.6)

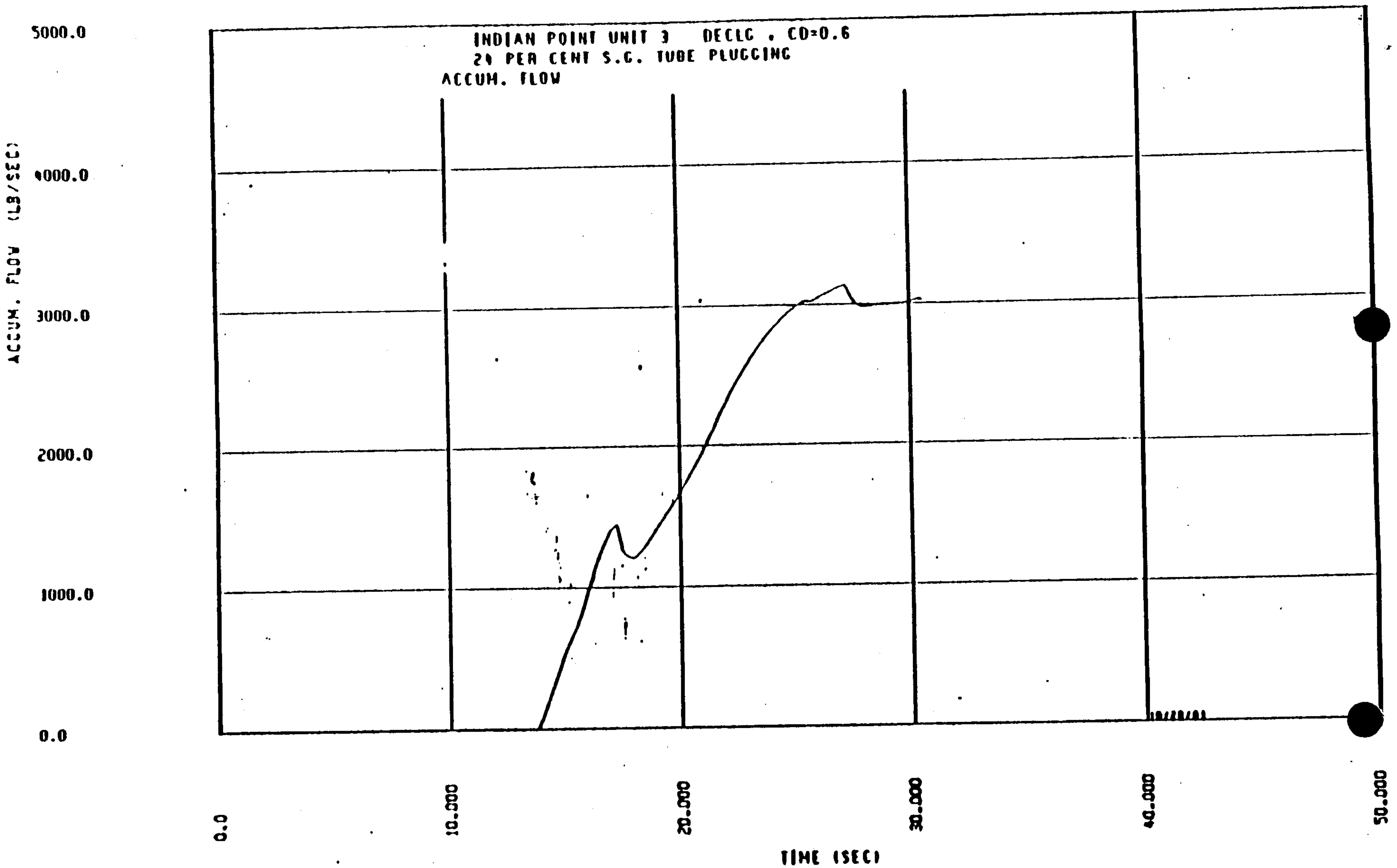

FIGURE 12D ACCUMULATOR FLOW(BLOWDOWN)
DECLG(CD = 0.6)

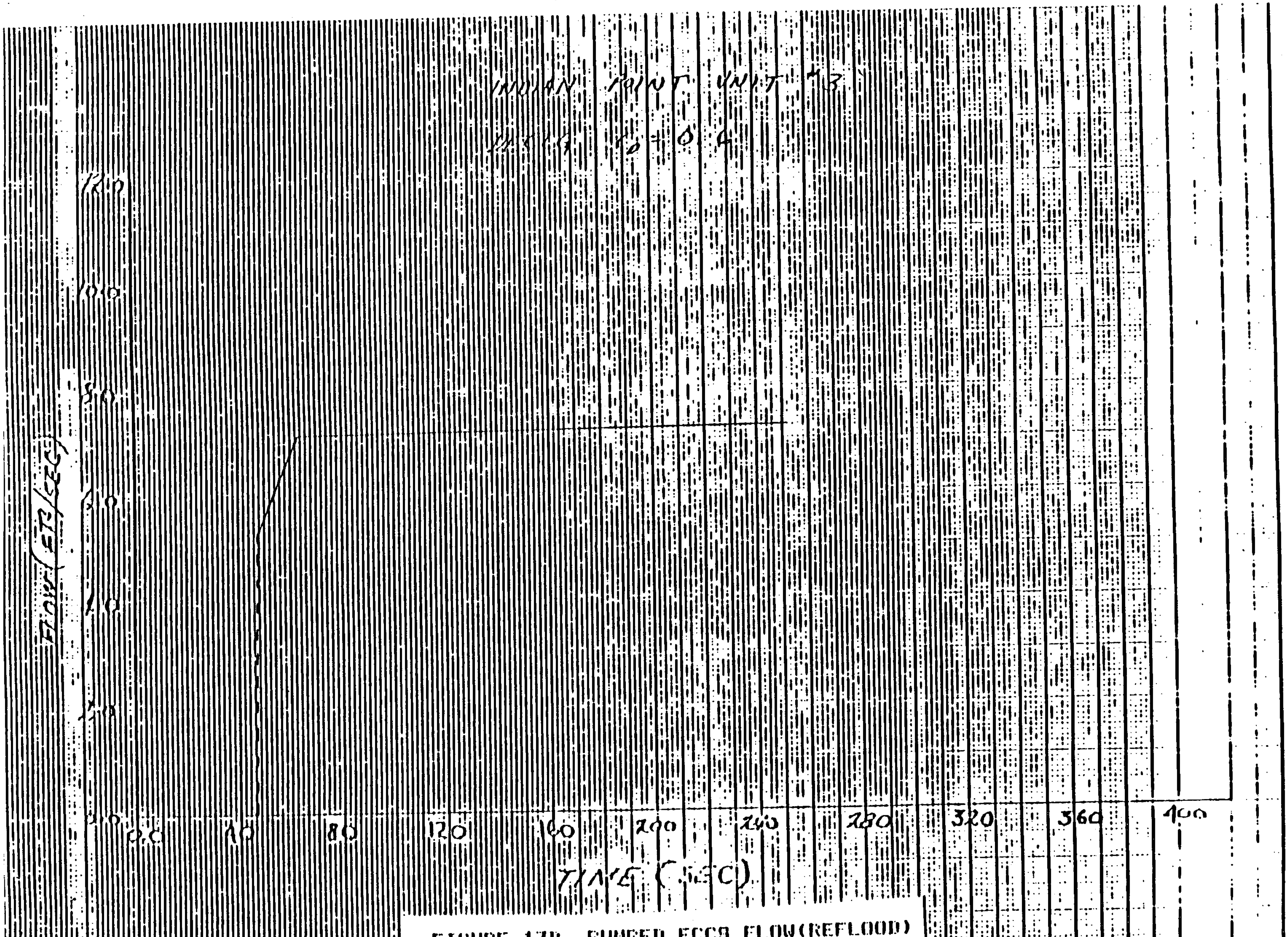

FIGURE 13D PUMPED ECCS FLOW(REFLOOD)
DECLG(CD = 0.6)

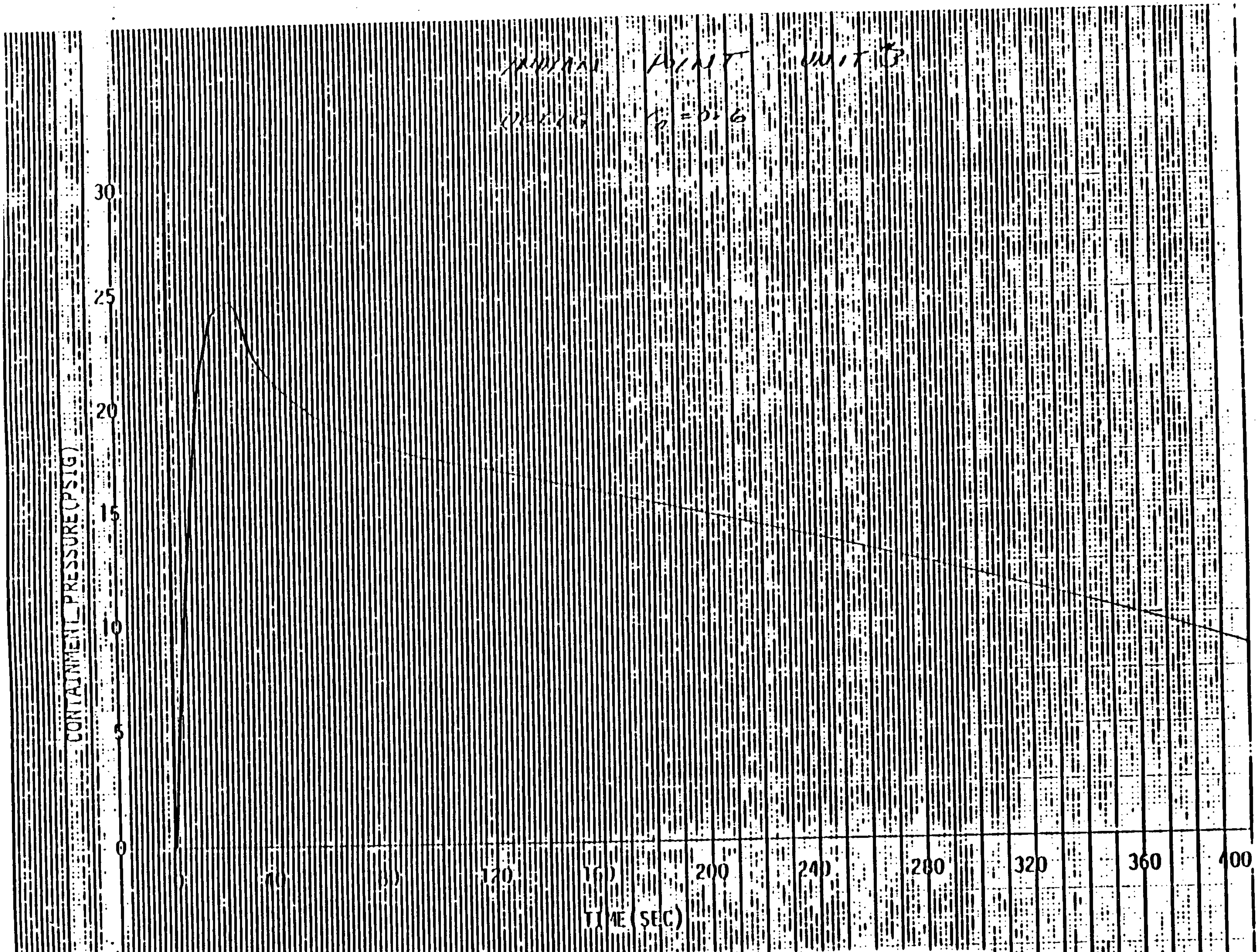

FIGURE 14B CONTAINMENT PRESSURE
DECLG(CD = 0.6)

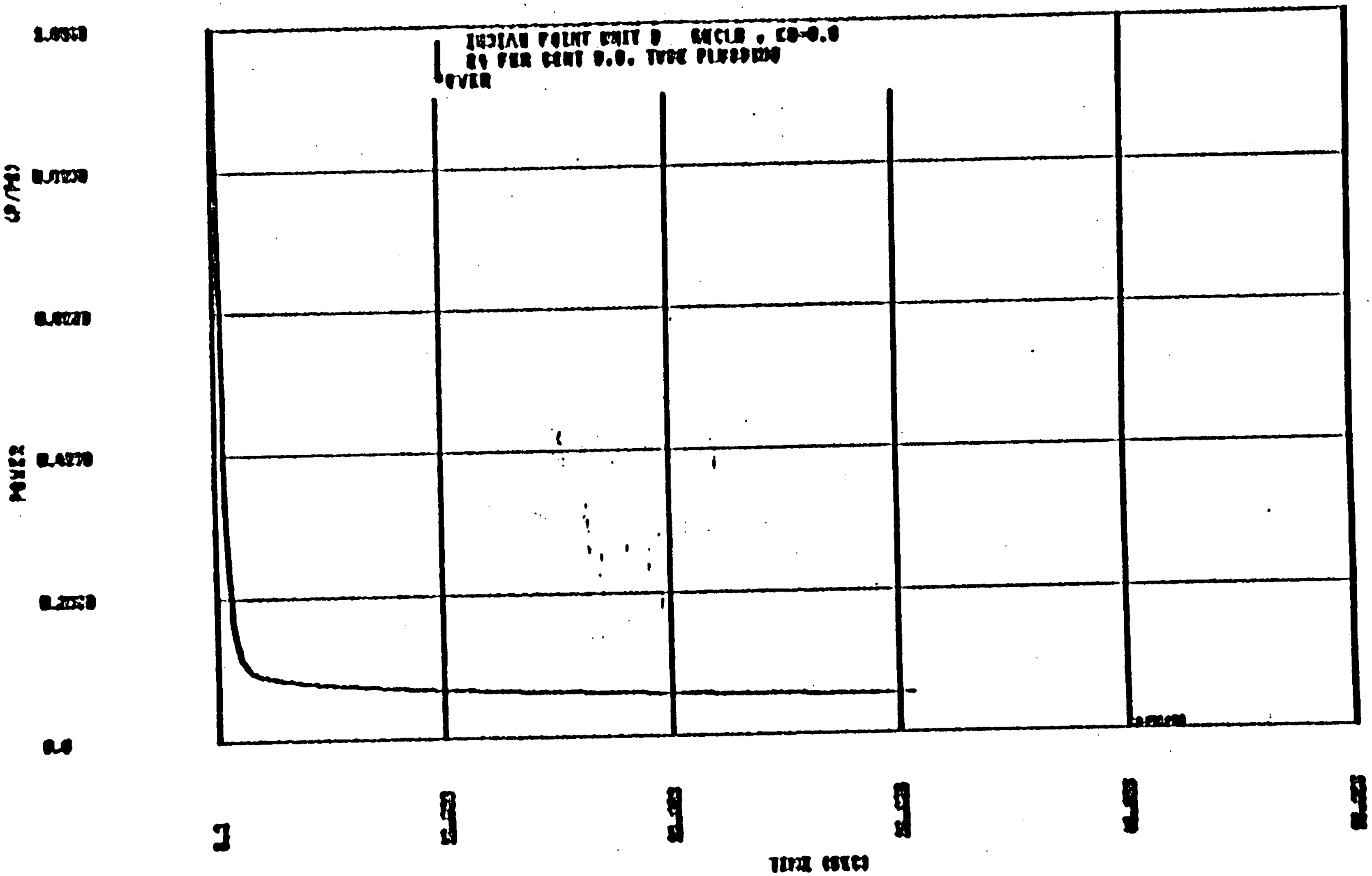

FIGURE 15B CORE POWER TRANSIENT
DECLB(CD = 0.6)

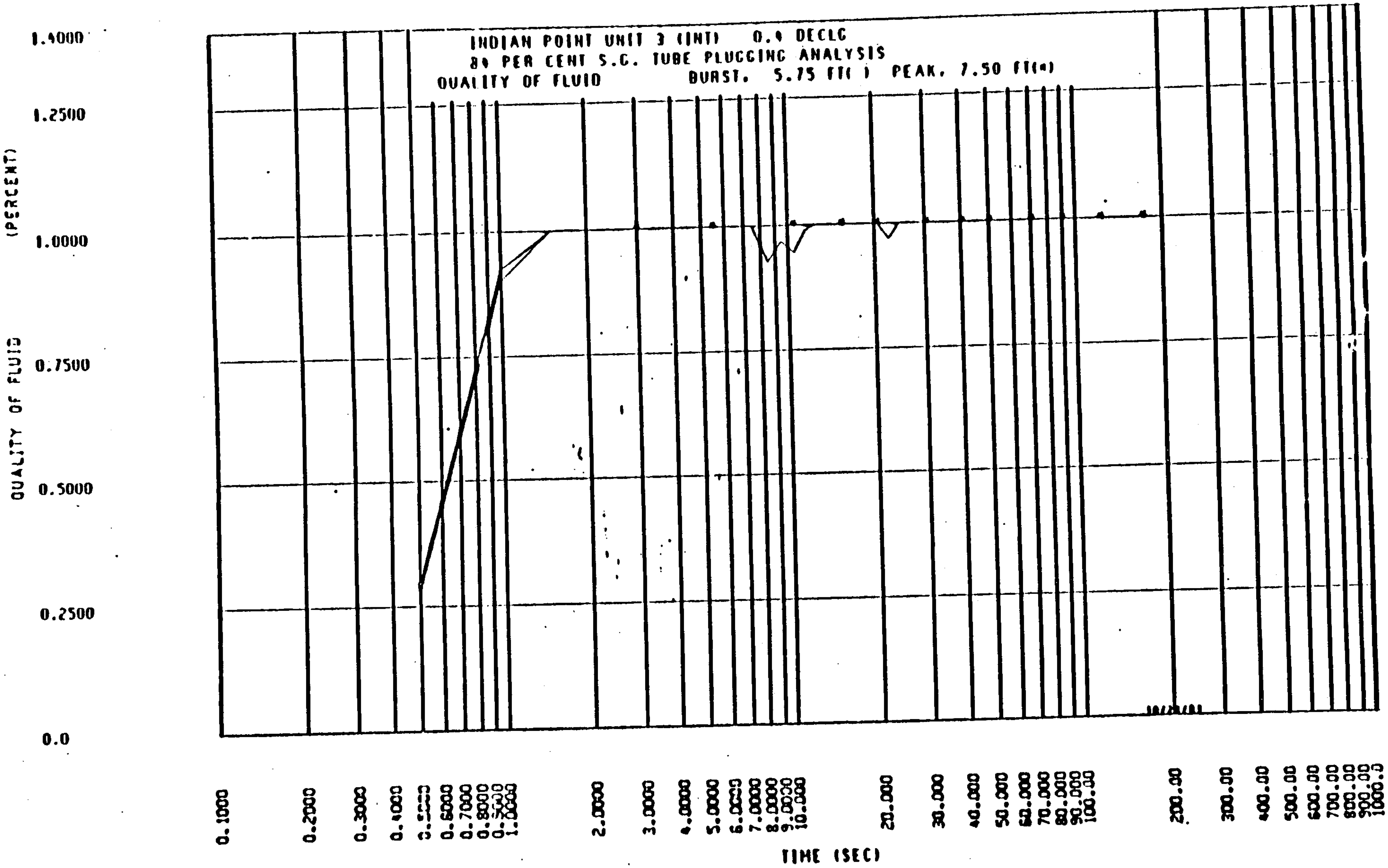

FIGURE 1C FLUID QUALITY
DECLG(CD = 0.4)

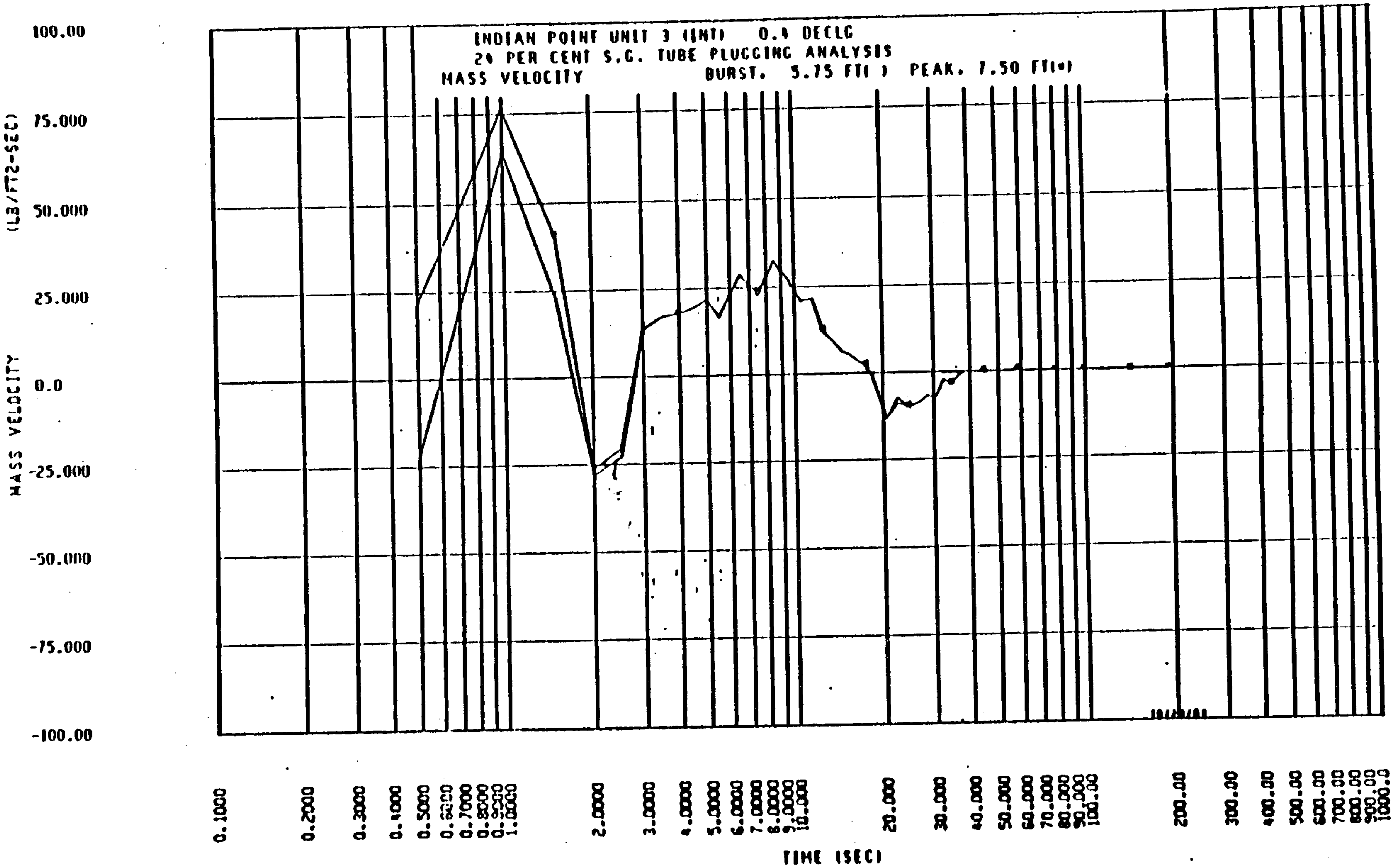

FIGURE 2C MASS VELOCITY
DECLG(CD = 0.4)

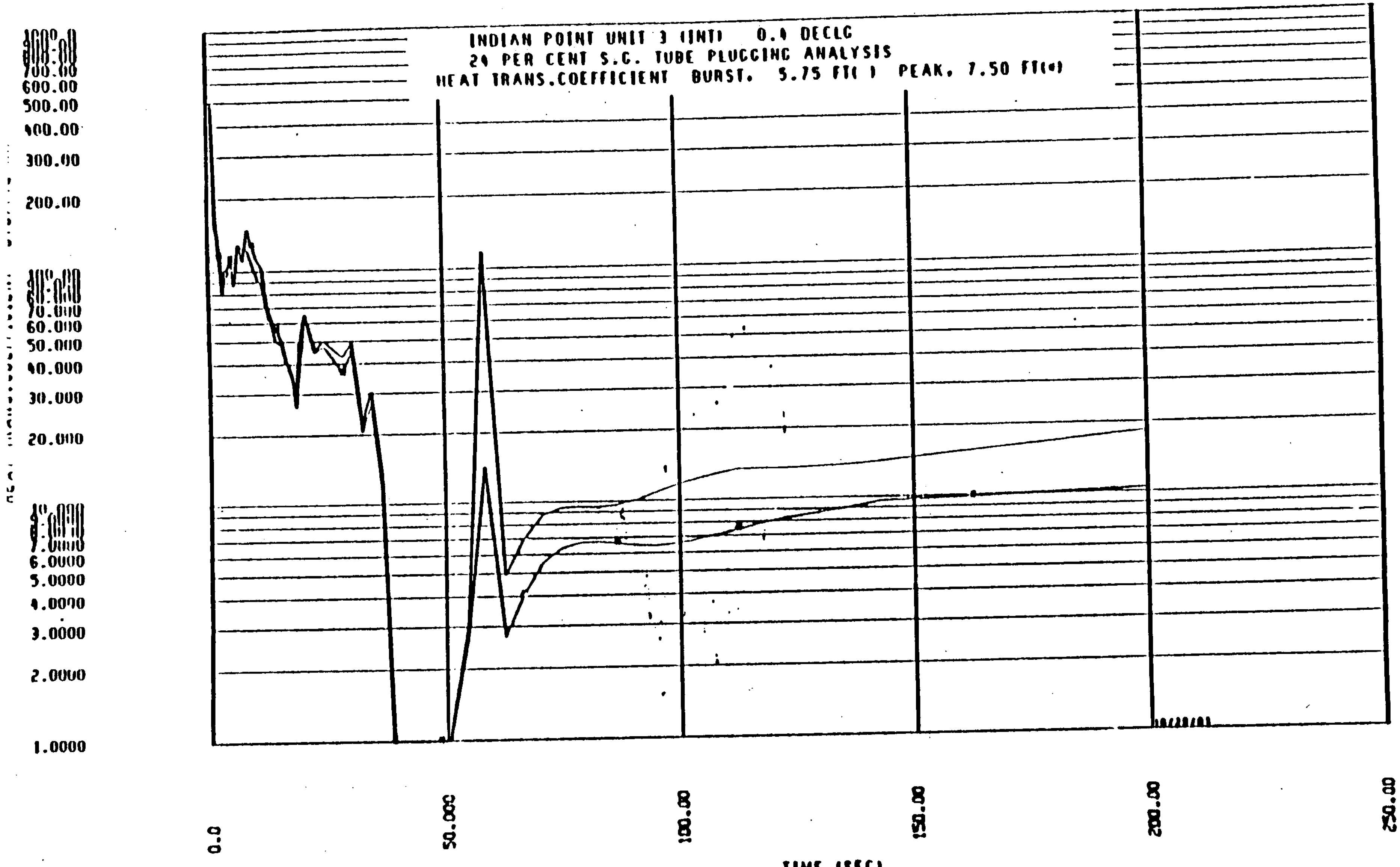

FIGURE 3C HEAT TRANSFER COEFFICIENT
DECLG(CD = 0.4)

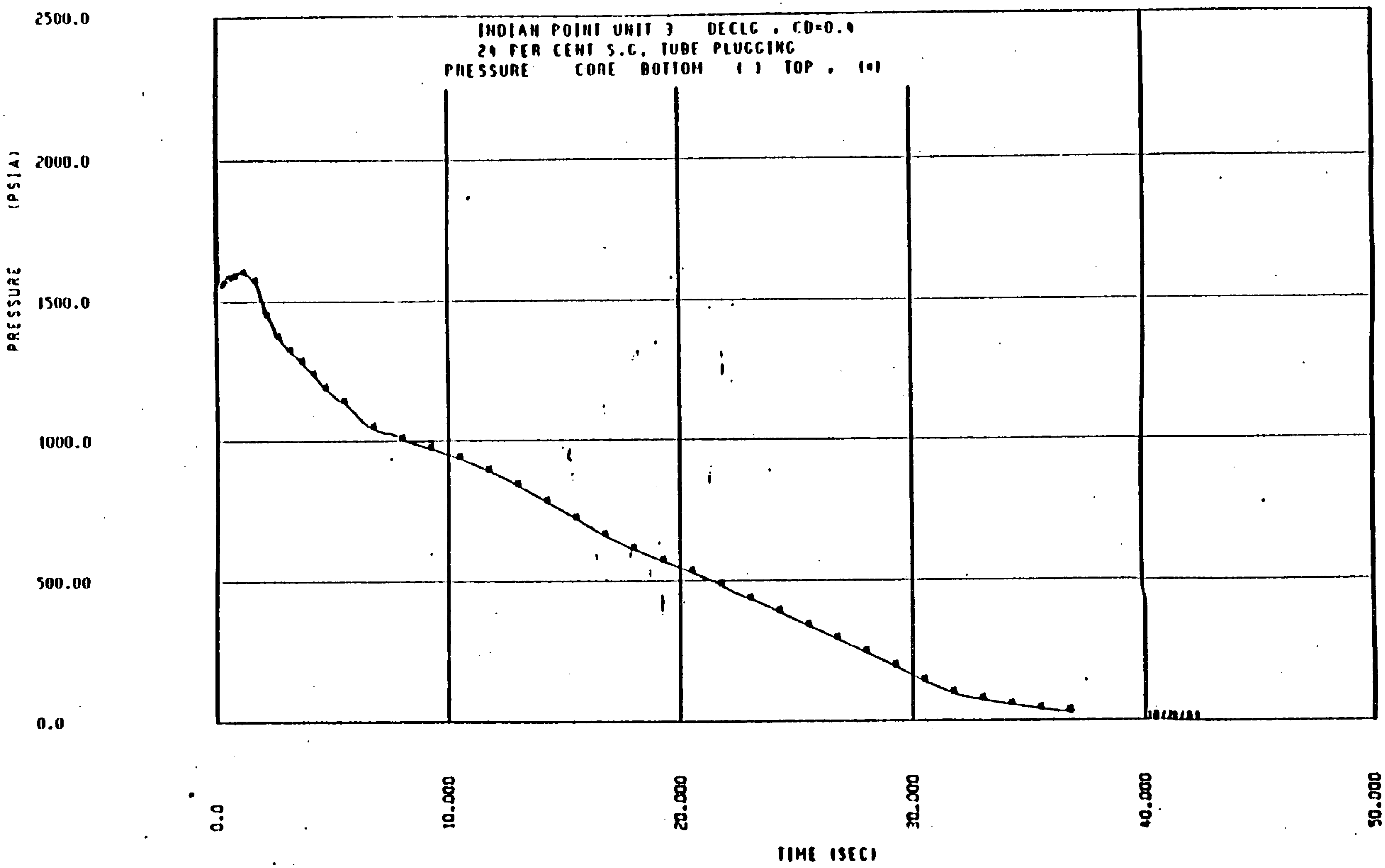

FIGURE 4C CORE PRESSURE
DECLG(CD = 0.4)

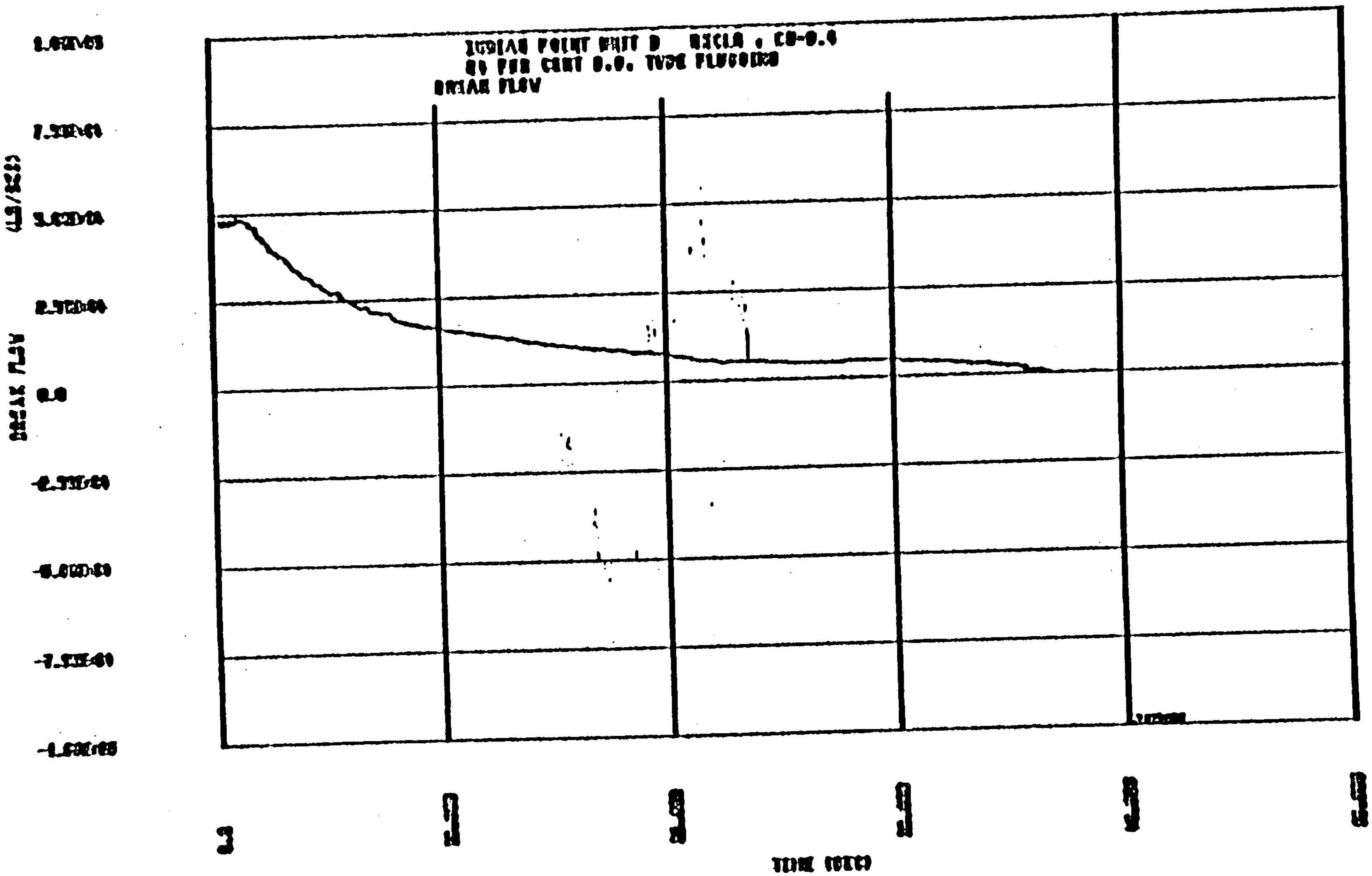

FIGURE 5C BREAK FLOW RATE
DECLG(CD = 0.4)

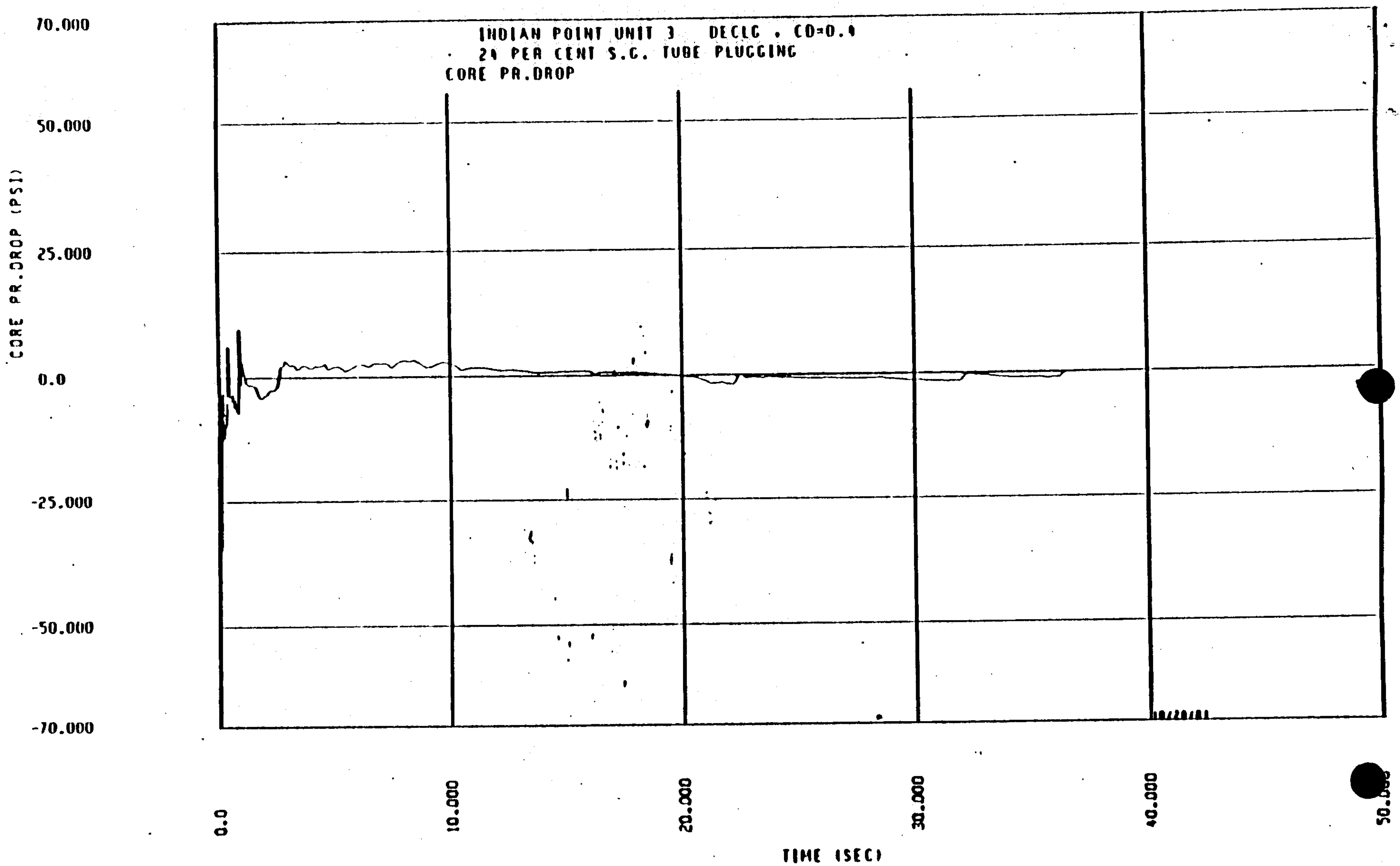

FIGURE 6C CORE PRESSURE DROP
DECLG(CD = 0.4)

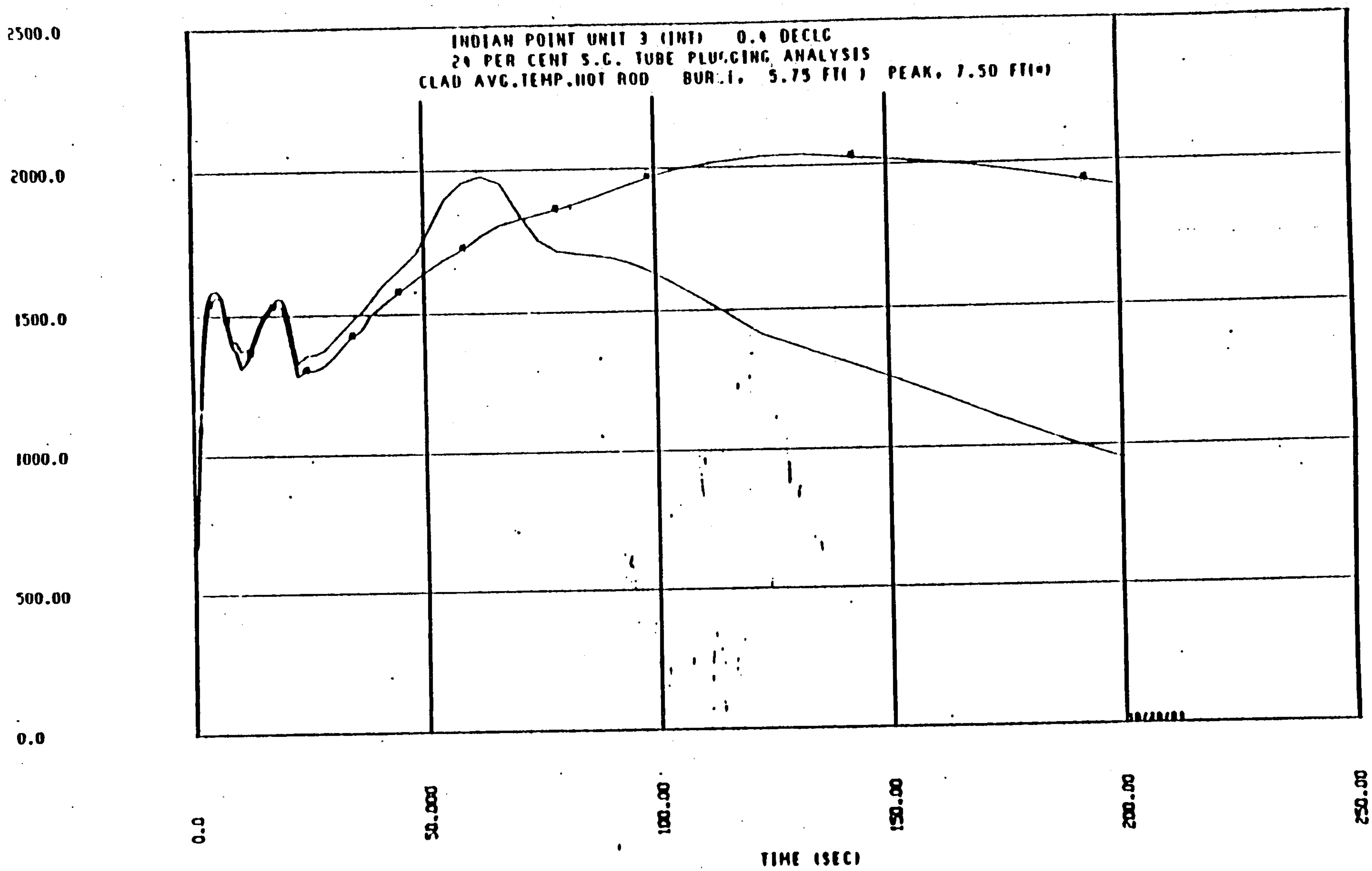

FIGURE 7C PEAK CLAD TEMPERATURE
DECLG(CD = 0.4)

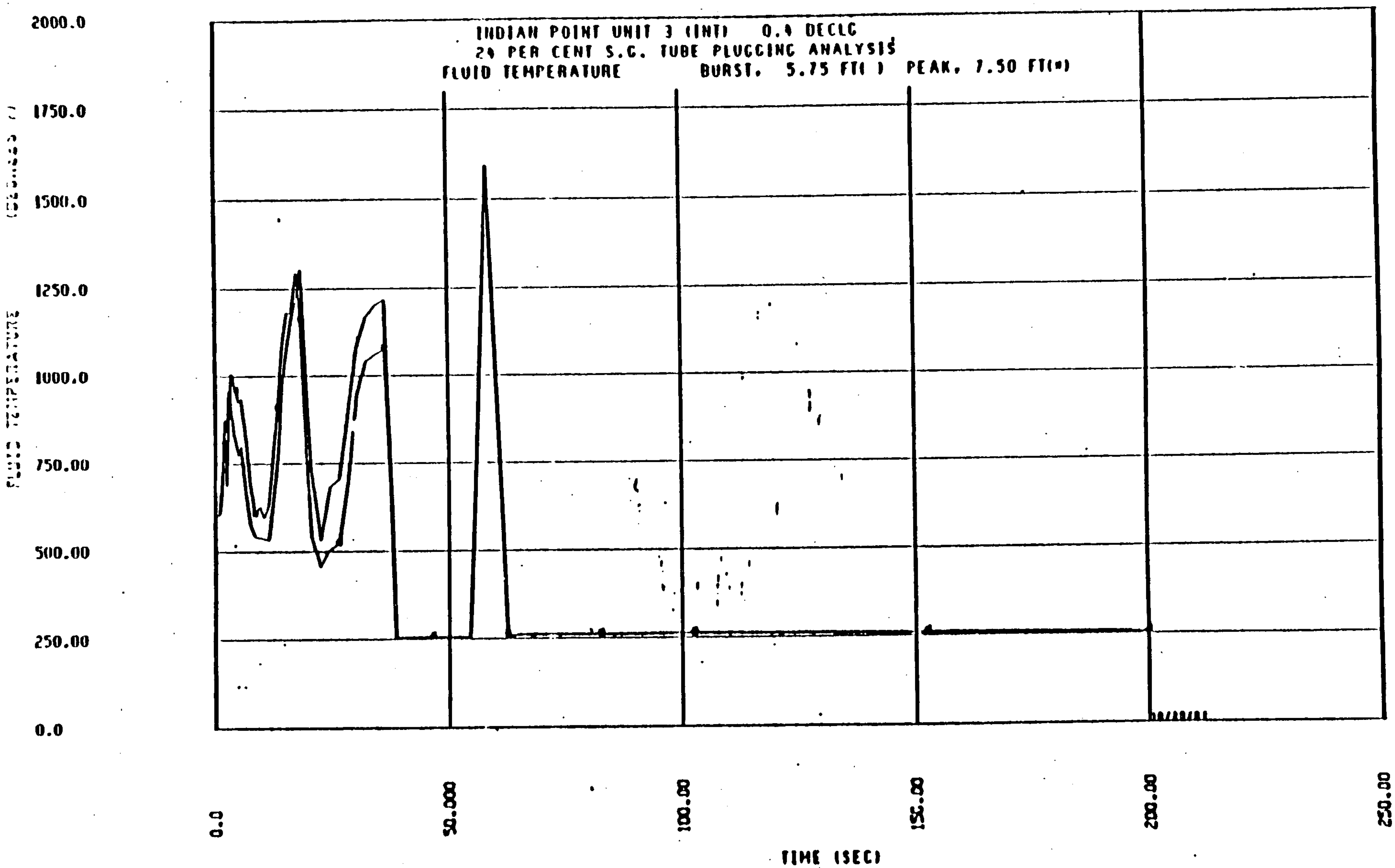

FIGURE 8C FLUID TEMPERATURE
DECLG(CD = 0.4)

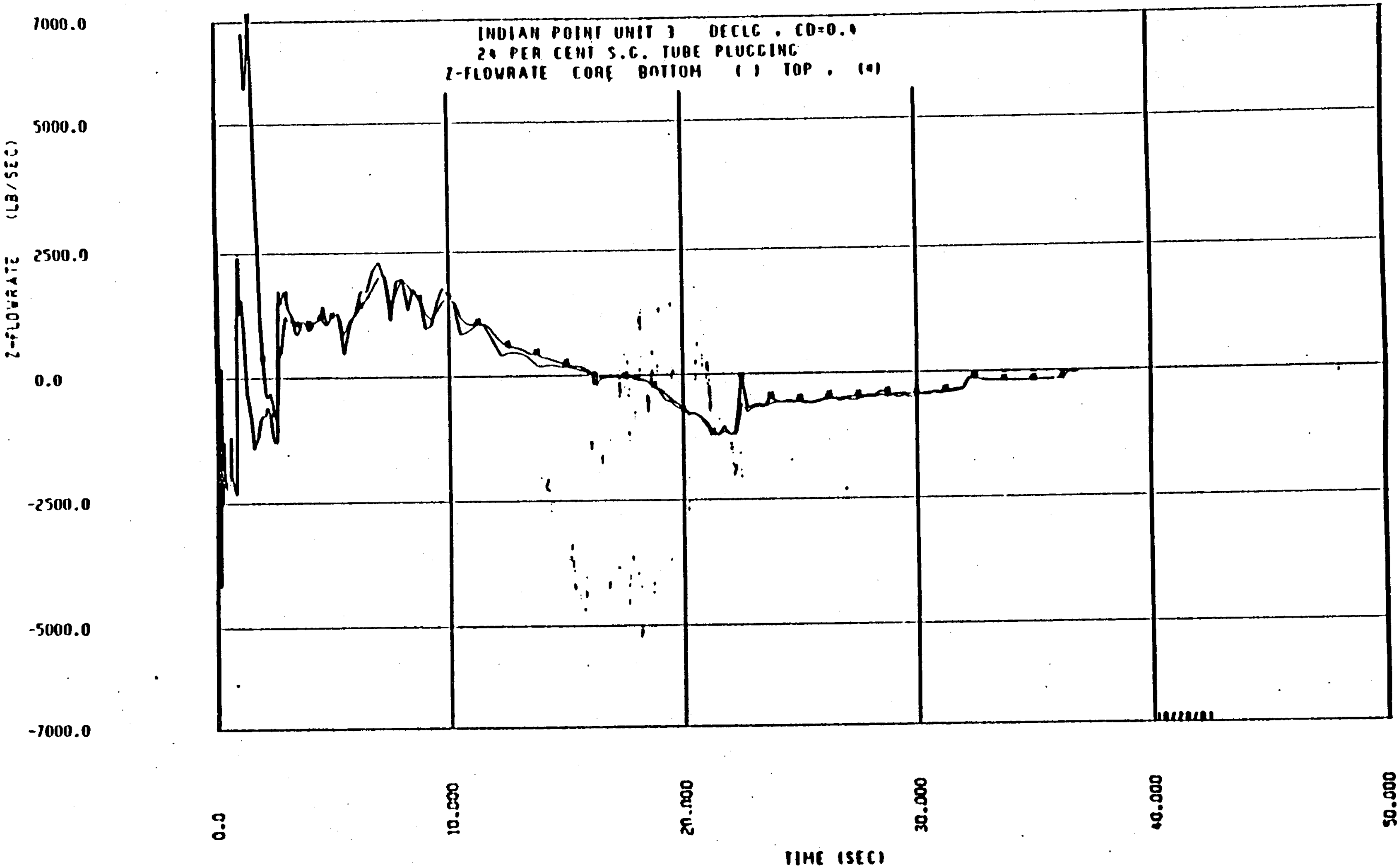

FIGURE 9C CORE FLOW (TOP AND BOTTOM)
DECLB(CD = 0.4)

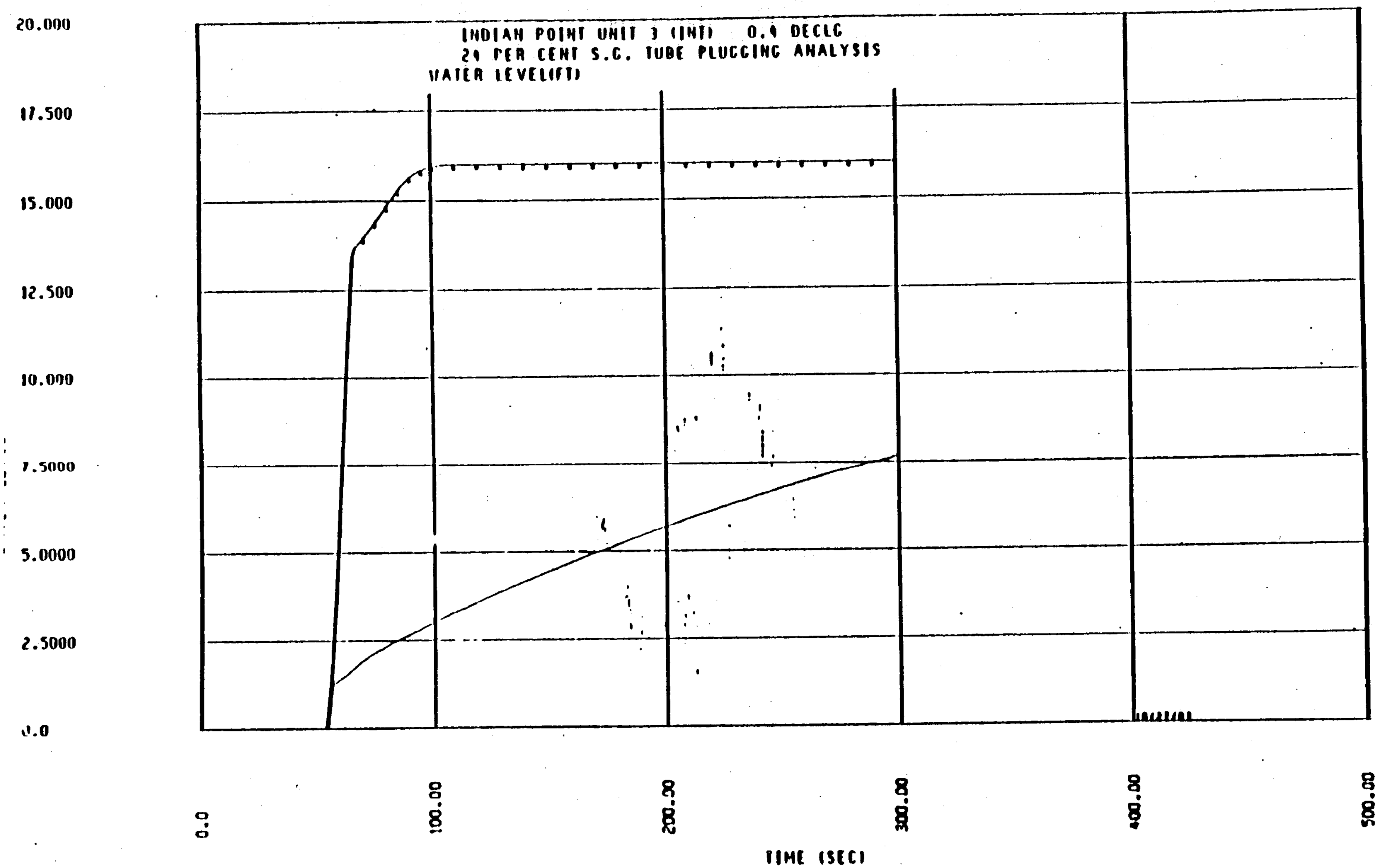

FIGURE 10C REFLOOD TRANSIENT - CORE & DOWNCOMER WATER LEVELS DECLG(CD = 0.4)

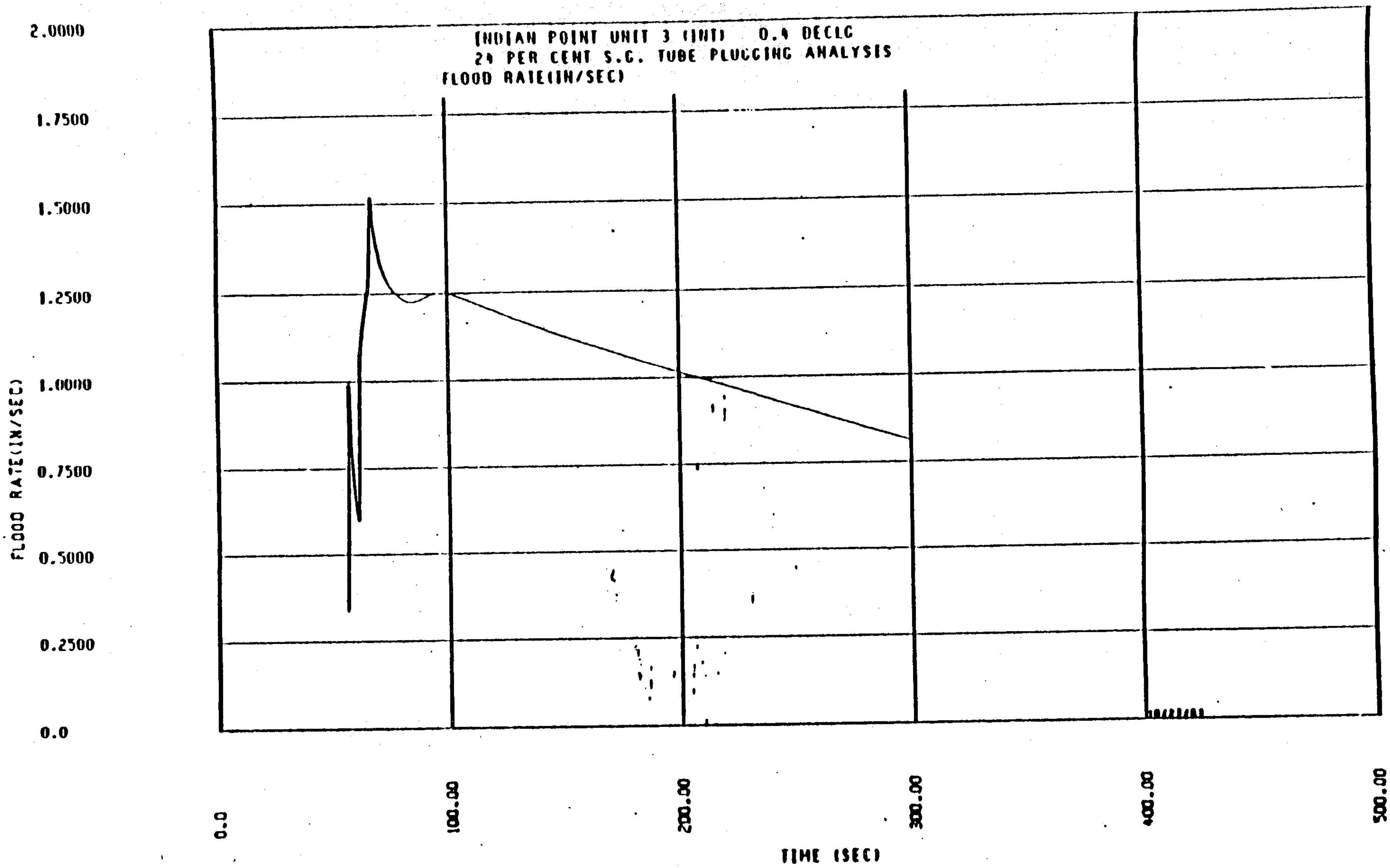

FIGURE 11C REFLOOD TRANSIENT
CORE INLET VELOCITY
DECLG(CD = 0.4)

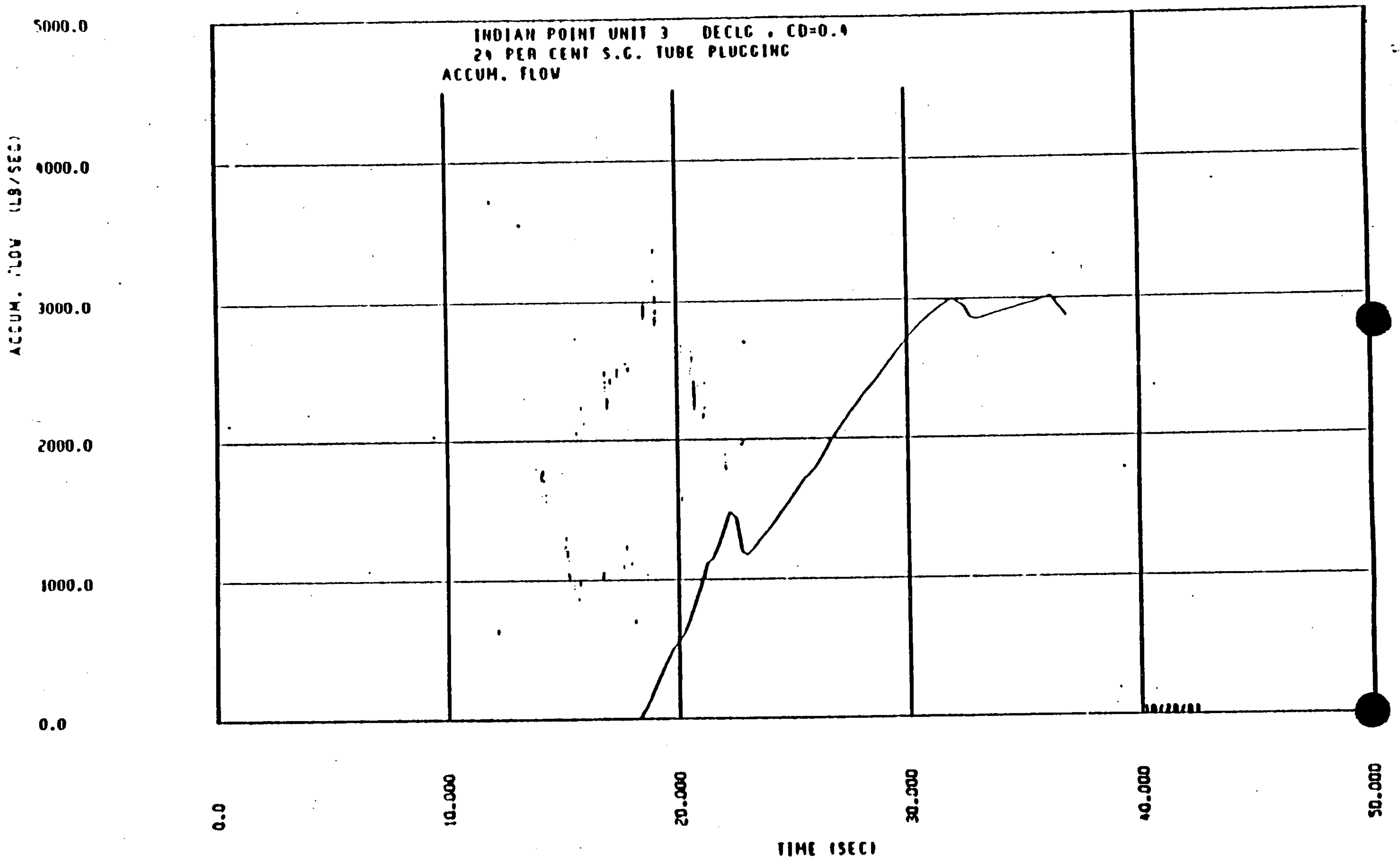

FIGURE 12C ACCUMULATOR FLOW(BLOWDOWN)
DECLG(CD = 0.4)

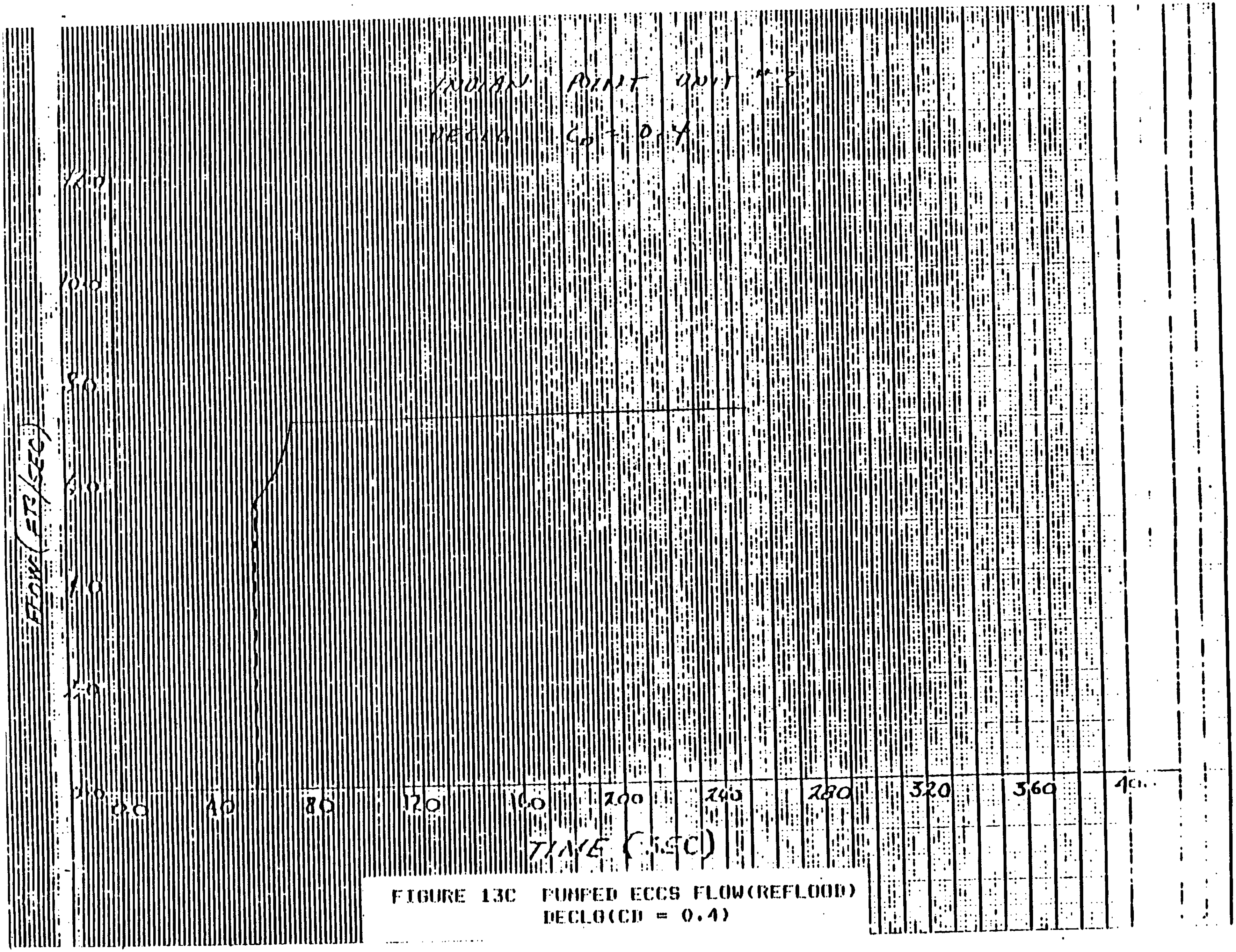

FIGURE 13C PUMPED ECCS FLOW(REFLOOD)
DECLO(CD = 0.4)

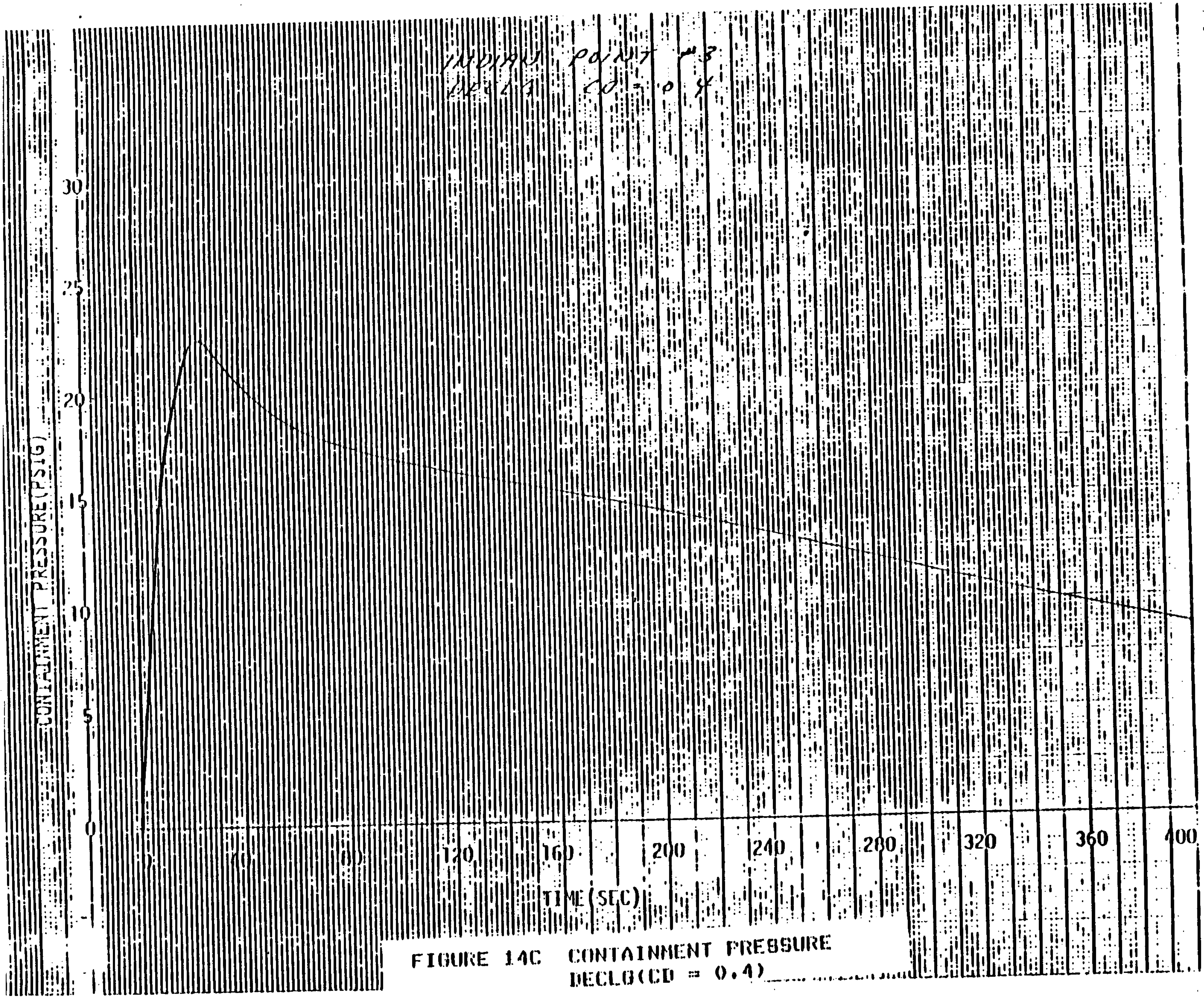

FIGURE 14C CONTAINMENT PRESSURE
DECLG(CD = 0.4)

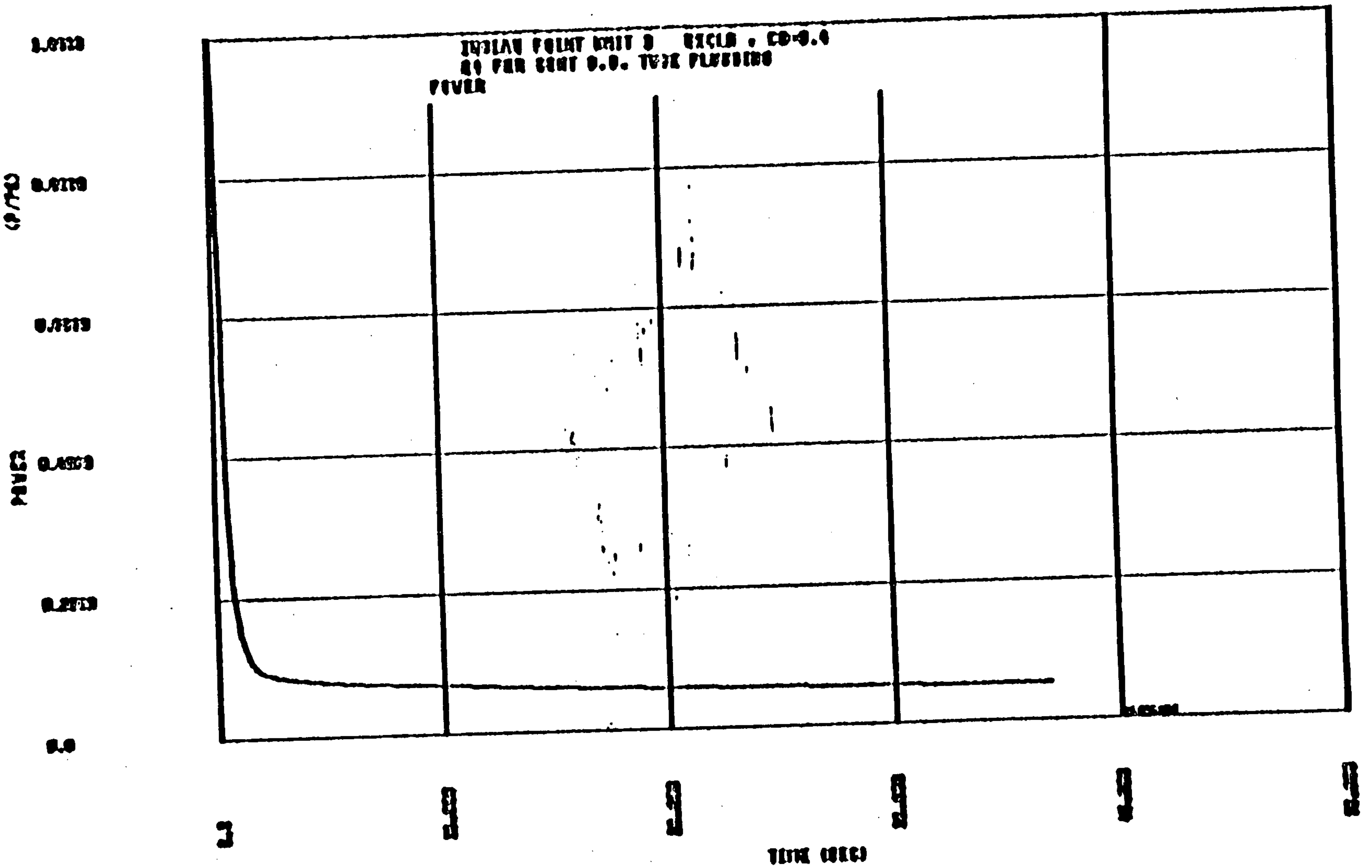

FIGURE 15C CORE POWER TRANSIENT
DECLG(CD = 0.4)

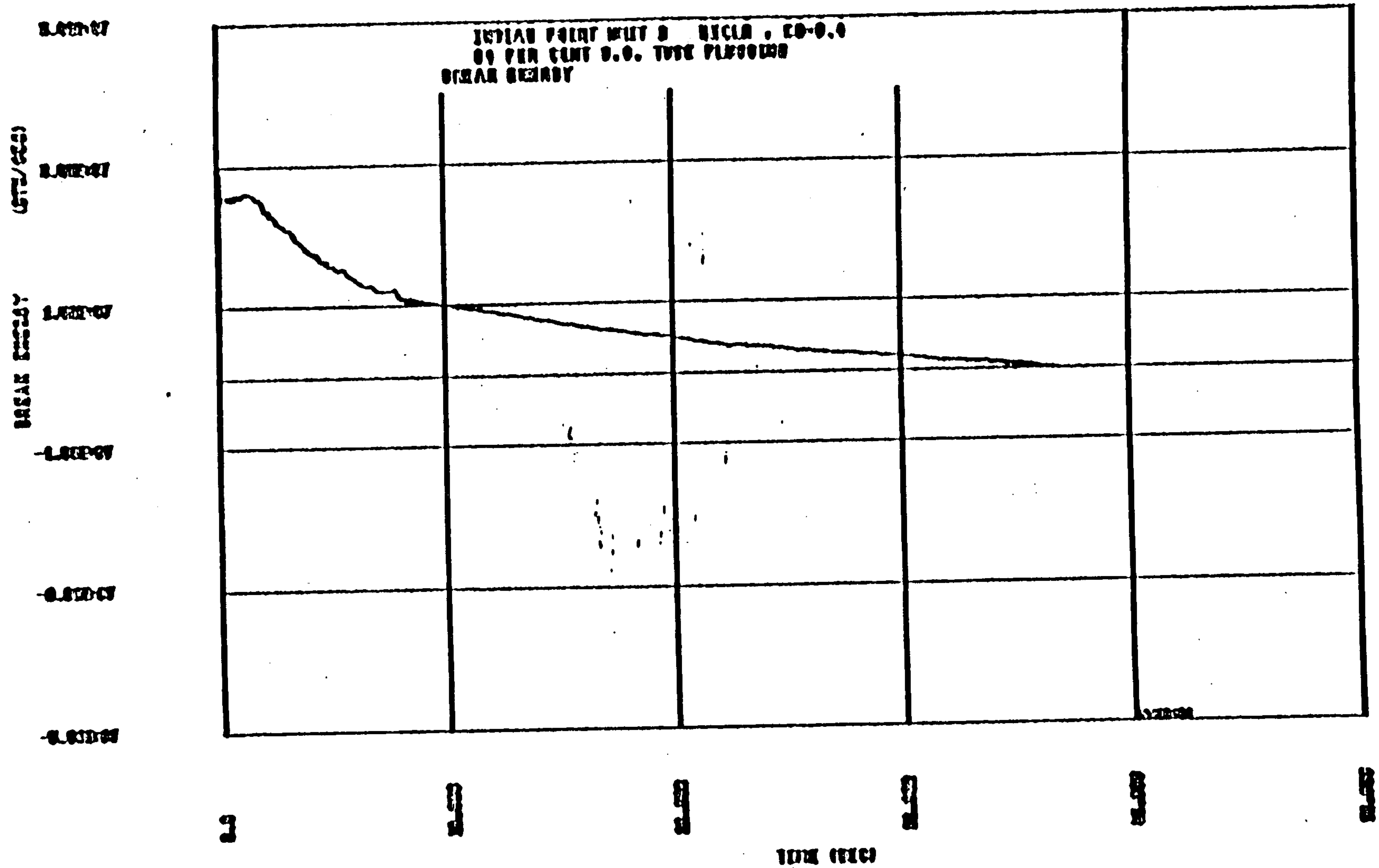

FIGURE 16 BREAK ENERGY RELEASED TO CONTAINMENT

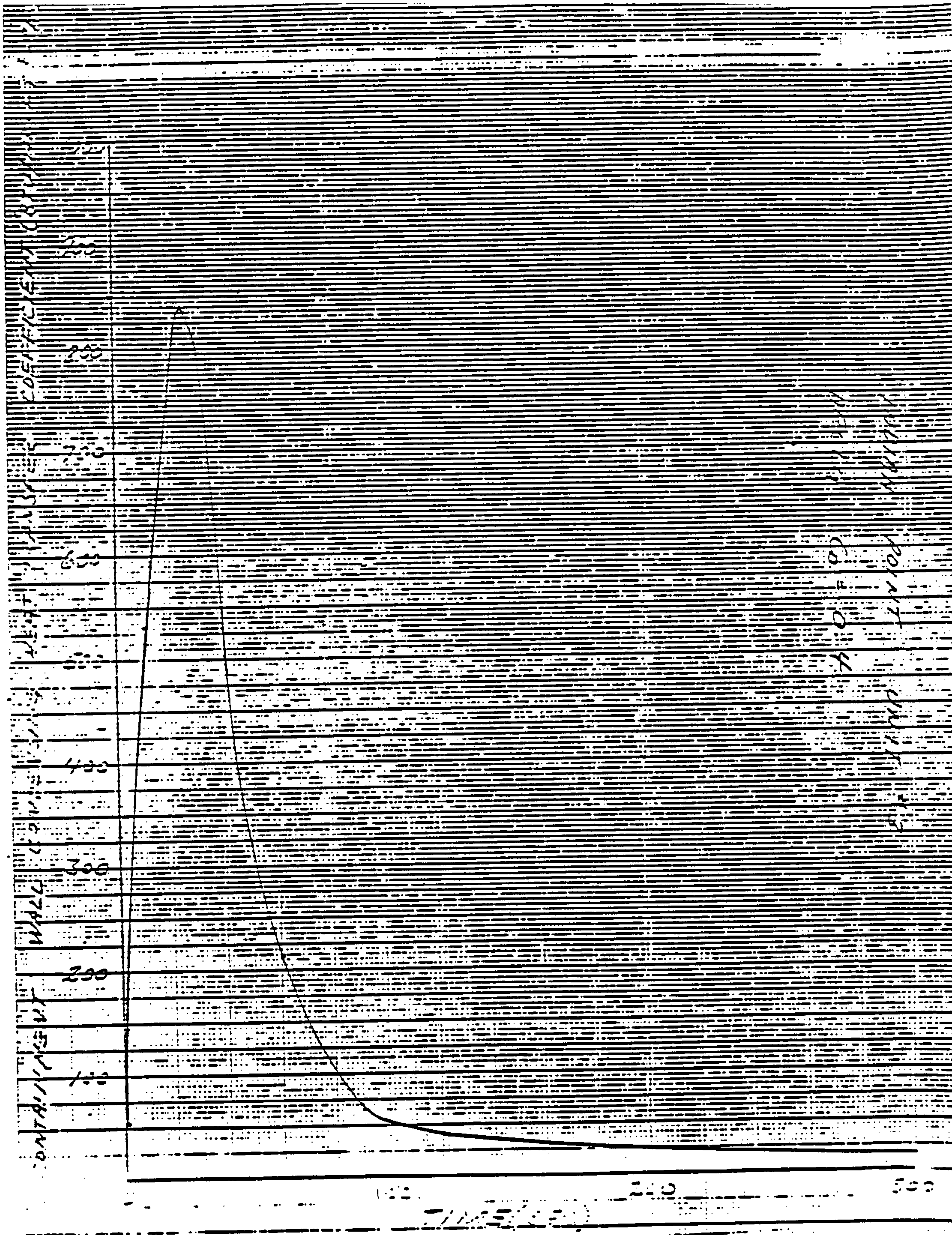

FIGURE 17 CONTAINMENT WALL
HEAT TRANSFER COEFFICIENT

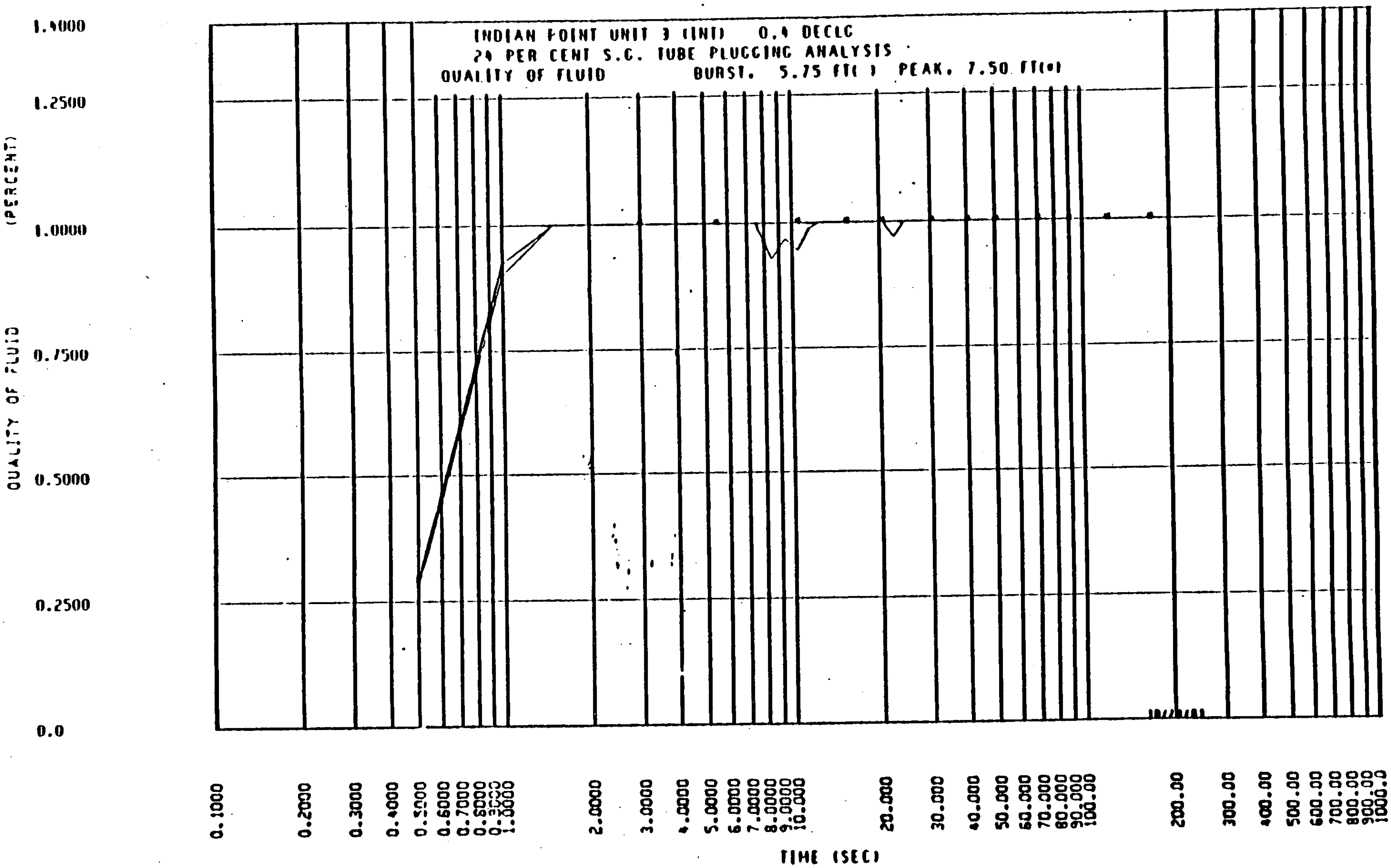

FIGURE 1D FLUID QUALITY
DECLG(CD = 0.4)

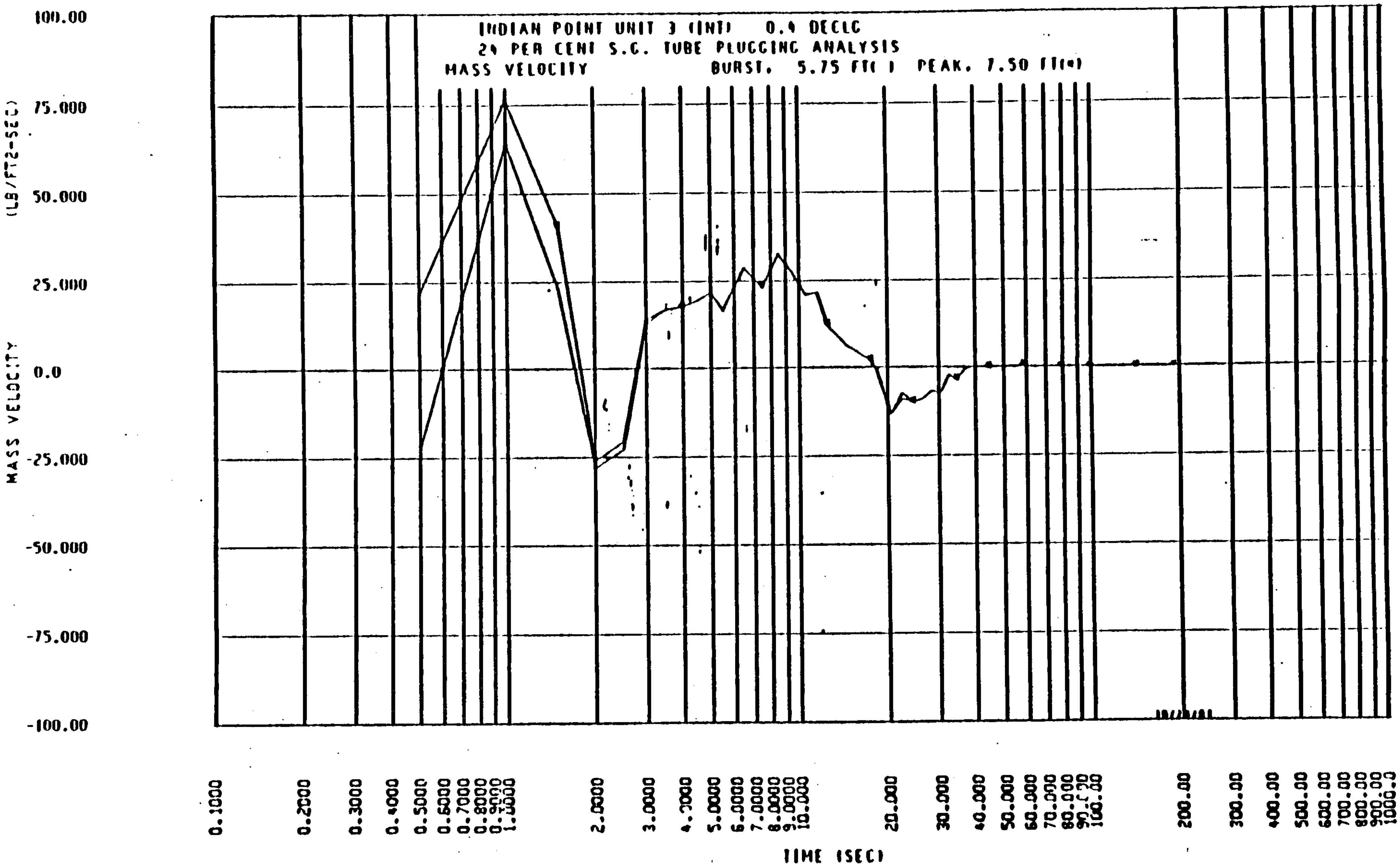

FIGURE 2D MASS VELOCITY
DECLG(CD = 0.4)

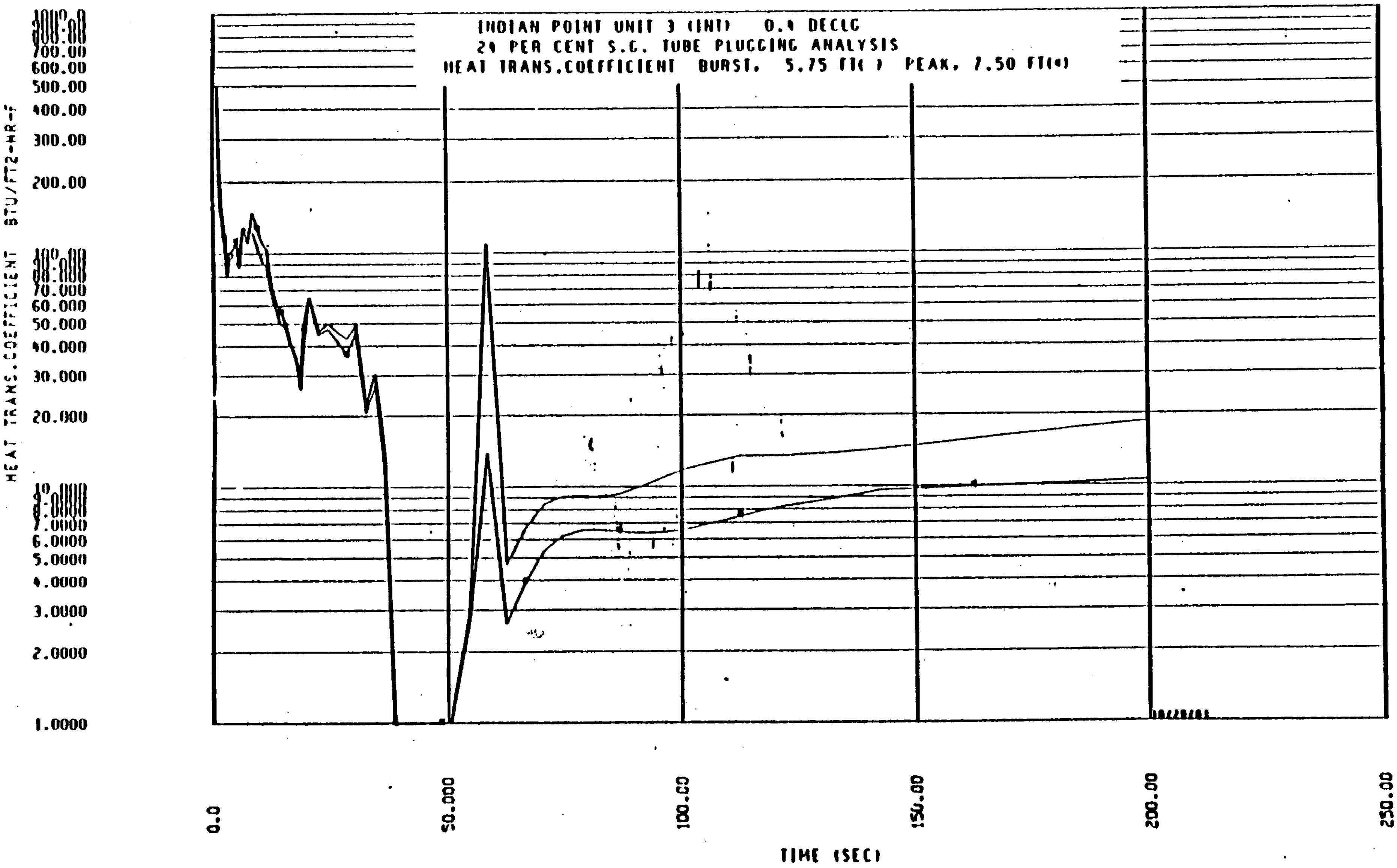

FIGURE 3D HEAT TRANSFER COEFFICIENT
DECLG(CD = 0.4)

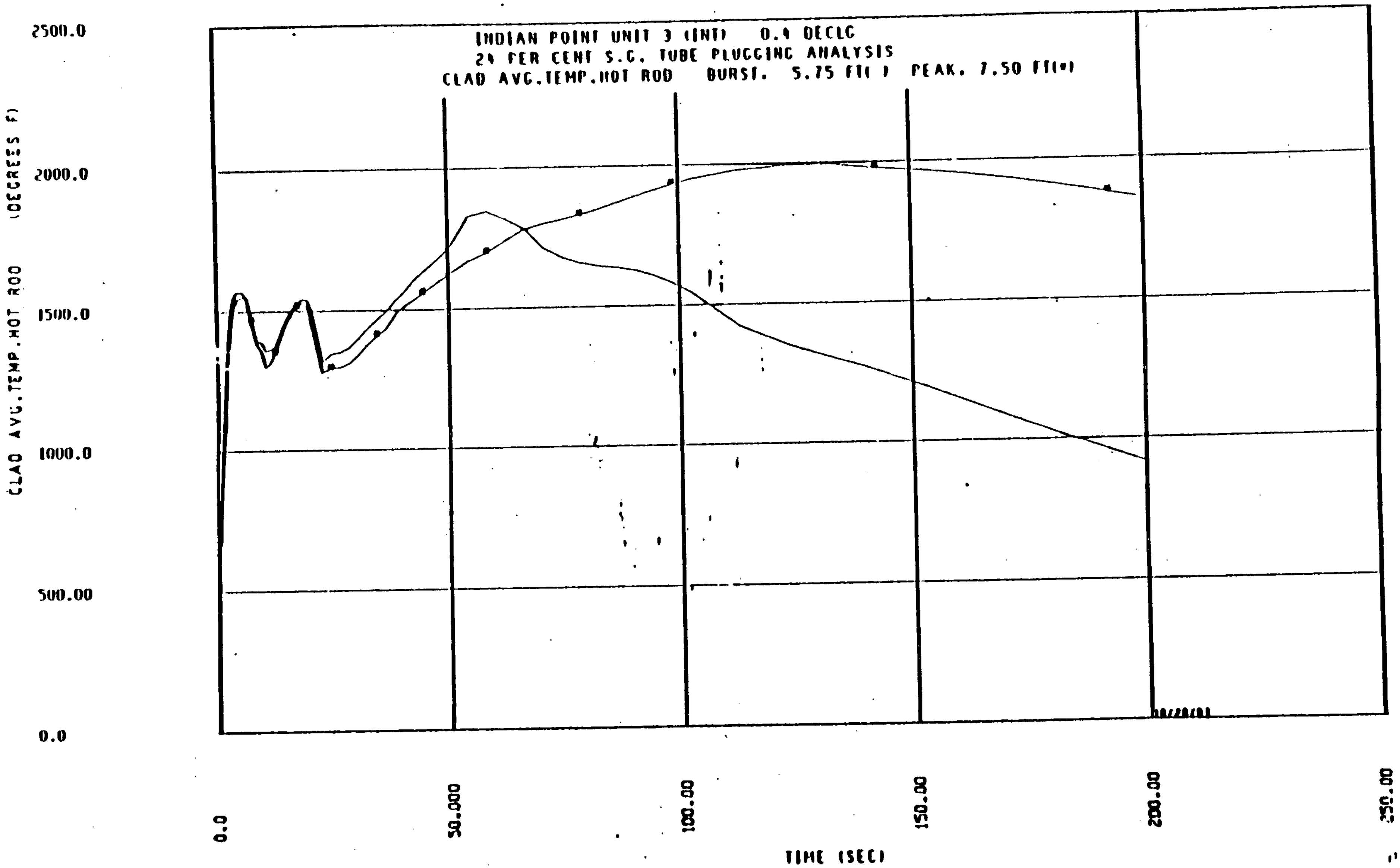

FIGURE 7D PEAK CLAD TEMPERATURE
DECLG(CD = 0.4)

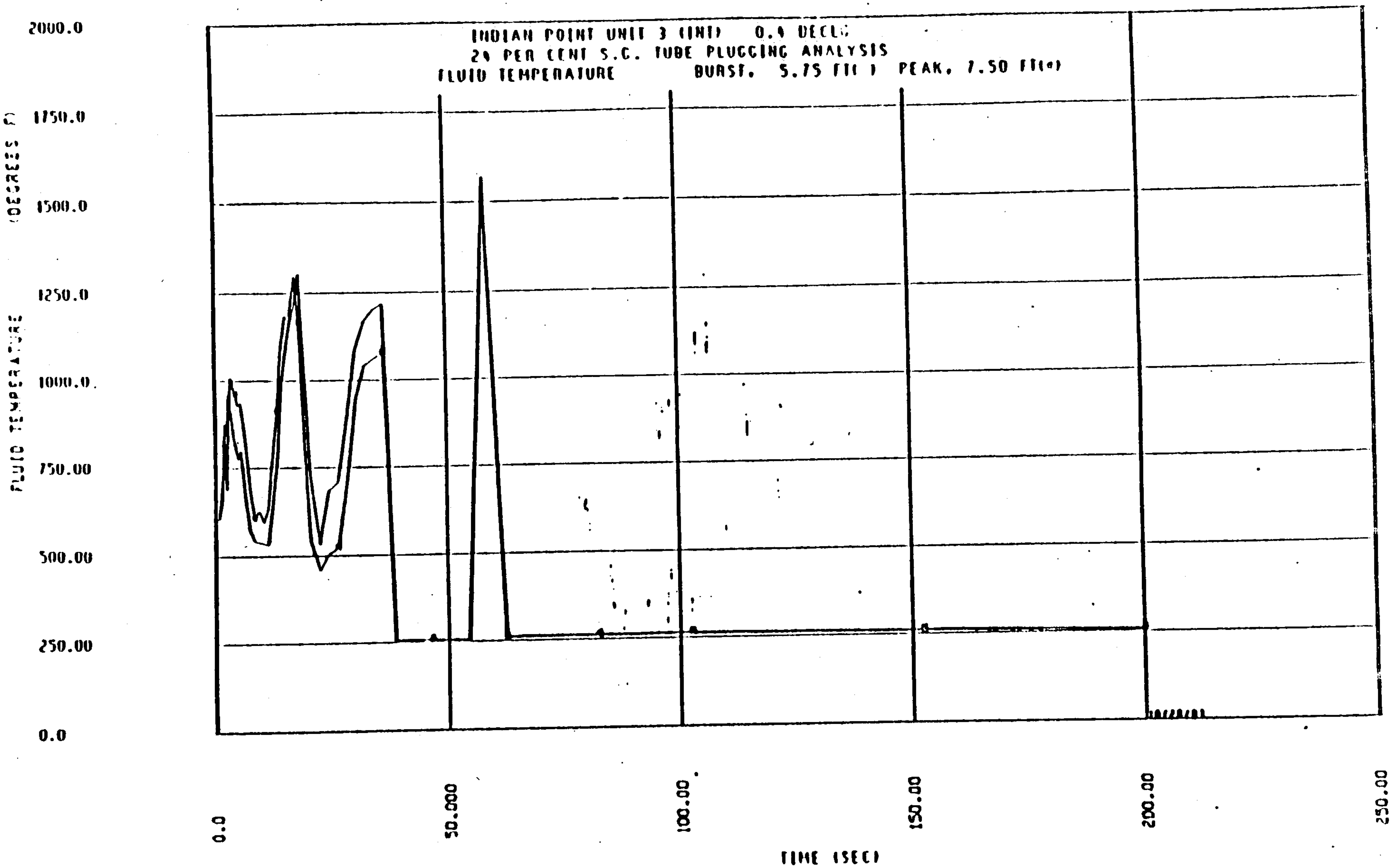

FIGURE 8D FLUID TEMPERATURE
DECLG(CD = 0.4)