ATTACHMENT I
Proposed Technical Specifications Changes

New York Power Authority
Indian Point 3 Nuclear Power Plant
Docket No. 50-286

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability:

Applies to the limits on core fission power distribution and to 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:

$F_Q(Z) \leq (2.20/p) \times K(Z)$ for $P > 0.5$

$F_Q(Z) \leq (4.40) \times K(Z)$ for $P \leq 0.5$

$F^N_{\Delta H} \leq 1.55\ [1 + 0.3\ (1-P)]$

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.

F^E_Q Engineering Heat Flux Hot Channel Factor, is defined as the allowance 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.

$F^N_{\Delta H}$ 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 $F^N_{\Delta H}$ 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 $F^N_{\Delta H}$.

An upper bound envelope of 2.20 times the normalized peaking factor axial dependence of Figure 3.10-2 has been determined consistent with Appendix K criteria and is satisfied for OFA transition mixed cores (3) 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 demonstrates a peak clad temperature not greater than 2049°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 movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

In the specified limit of $F^N_{\Delta H}$ there is a 8 percent allowance for uncertainies which means that normal operation of the core is expected to result in $F^N_{\Delta H} < 1.55/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape

4. Axial Power Distribution Control Procedures, which are given in terms 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 referenced 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 2.20 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

ATTACHMENT II
Safety Evaluation of Proposed
Technical Specifications

Enclosure 1: Appendix K ECCS Reanalysis
Enclosure 2: ECCS Analysis Sensitivity Study

New York Power Authority
Indian Point 3 Nuclear Power Plant
Docket No. 50-286

Attachment II
Safety Evaluation of Proposed
Technical Specifications

I. Description of Change

This revision to the Indian Point 3 Technical Specifications seeks to increase the maximum allowable total core peaking factor, F_Q. Presently, the Technical Specifications limit the full power F_Q to less than or equal to 2.13. This change will increase the full power F_Q limit to less than or equal to 2.20.

II. EVALUATION OF CHANGE

The February 11, 1980 Confirmatory Order had limited the calculated fuel peak clad temperature (PCT) to a maximum of 2000°F under large break LOCA conditons. As a result of this limit, a substantial penalty on F_Q had to be imposed.

Enclosure 1 to this Safety Evaluation provides the Appendix K Emergency Core Cooling System (ECCS) reanalysis assuming a uniform 24% steam generator tube plugging level, which was transmitted to the NRC via the Authority's May 5, 1983 letter. The limiting Final Acceptance Criteria (FAC) analysis case was the Double Ended Cold Leg Guillotine (DECLG) break, C_d=0.4. For this limiting case, the PCT was calculated to be 2039°F for a full power F_Q of 2.20. This case was re-analyzed with a full power F_Q of 2.14. The resultant PCT was 1995°F and thereby assured compliance with the February 11, 1980 Order. This F_Q limit was incorporated into the Indian Point 3 Technical Specification via Amendment 48, dated January 13, 1984.

By Rescission of Order, dated July 5, 1985, the February 11, 1980 Confirmatory Order was rescinded. As such, the aforementioned limiting case analysis with an F_Q of 2.20 can now be utilized as the resultant PCT of 2039°F satisfies the PCT requirement of 10 CFR 50.46.

However, the aforementioned Appendix K reanalysis, assuming a F_Q of 2.20, was performed in support of reactor operations with uniform 24% steam generator tube plugging. The Authority's March 14, 1986 letter transmitted Revision 1 to WCAP-10705, "Safety Evaluation for Indian Point 3 with Asymmetric Tube Plugging Among Steam Generators (Non-proprietary)", which documented sensitivity of asymmetric steam generator tube plugging level on the LOCA and non LOCA transients. Enclosure 2 to this Safety Evaluation details the sensitivity of asymmetric tube plugging on calculated ECCS performance. The full power F_Q of 2.20 for the ECCS reanalysis performed for a uniform 24% steam generator tube plugging level is not adversely impacted by asymmetric effects.

The Authority's April 23, 1985 letter transmitted proposed revisions to the Technical Specifications in support of the Cycle 4/5 refueling, which involved a fuel design transition from the Westinghouse 15 x 15 low parasitic (LOPAR) design to the 15 x 15 Optimized Fuel Assembly (OFA) design. The greater hydraulic resistance of the 15 x 15 OFA will cause an approximate reduction of 2.2% in reflood flow rate. This will result in an approximate 10°F increase in PCT under large break LOCA conditions. As a result of this increase in PCT, the F_Q limit had to be lowered from 2.14 to 2.13 so that the Confirmatory Order requirement of 2000°F PCT would not be exceeded. These revisions were incorporated into the Indian Point 3 Technical Specifications via Amendment 61, dated August 27, 1985.

Including the 10°F increase to account for the OFA design results in a PCT of 2049°F for a F_Q of 2.20 under the LOCA conditions. The limits of 10 CFR 50.46 are not exceeded. The F_Q assumed in all of the small break LOCA and non-LOCA transient analyses was 2.32. Asymmetric steam generator tube plugging and the OFA design did not necessitate any reductions in this assumed F_Q value. Hence F_Q of 2.20 assumed in the large break LOCA is limiting.

III. No Significant Hazards Evaluation

1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change seeks to increase the F_Q limit to 2.20. The revised limit will not increase the probability of an accident previously analyzed as this limit is an operational restriction to limit the consequences of the accident.

The analyses results are within the safety limits provided by 10 CFR 50.46.

2) Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Increasing the F_Q limit to 2.20 will not introduce the possibility of an accident of a different type than previously analyzed.

3) Does the proposed amendment involve a significant reduction in a margin of safety?

In issuing Amendement 48 to the Indian Point 3 Technical Specifications, dated January 13, 1984, the NRC in their safety evaluation report reviewed and approved the ECCS reanalysis assuming a F_Q of 2.14, which results in a PCT of 1995°F. The proposed change increasing the F_Q limit of 2.20 will result in a PCT of 2039°F. The presence of OFA fuel in the core will result in a PCT of 2049°.

The analyses results are within the safety limits specified in 10 CFR 50.46.

The Authority considers that the proposed changes can be classified as not likely to involve significant hazard considerations since the proposed changes constitute "a change which may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan." (Example (VI), Federal Register, Vol. 48, No. 67 dated April 6, 1983 , page 148701).

IV. Impact of Change

This change will not impact the following:

- ALARA Program
- Fire Protection Program
- Emergency Plan
- FSAR or SER Conclusions
- Overall Plant Operations

V. Conclusion

This change: a) will not increase 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 evaluated previously in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the basis for any Technical Specification; d) does not constitute an unreviewed safety question as defined in 10 CFR 50.59; e) involves no significant hazards considerations as defined in 10 CFR 50.92.

VI. References

a) NYPA letters to NRC
   1. IPN-83-37 dated May 5, 1983
   2. IPN-85-15 dated March 27, 1985
   3. IPN-85-21 dated April 23, 1985

b) WCAP - 10705 "Safety Evaluation for Indian Point 3 with Asymmetric Tube Plugging Among Steam Generators," October 1984.

c) IP-3 FSAR

d) IP-3 SER

Enclosure 1 to Safety Evaluation
ECCS Re-Analysis (24% Uniform SGTP)

New York Power Authority
Indian Point 3 Nuclear Power Plant
Docket No. 50-286

Twenty-Four Percent Tube Plugging Safety Analysis (LOCA)

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 Purge

Branch Technical Position CSB6-4 states that the evaluation of a containment purge 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 purge system will close shortly after the beginning of a postulated LOCA transient based on the response to the containment isolation signal. The containment purge system 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 purge 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 purge 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 purge 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 purge on LOCA is extended to the 24% tube plugging case (FQ = 2.20), 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°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.

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 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.

NS-TMA-2448

May 15, 1981

1981 Model Letter to NRC

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NS-TMA-2448

Mr. James R. Miller
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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 the evaluation model described in References 4 and 5. The changes in the evaluation model have either no impact on peak clad temperature or are required to properly model the system under investigation. These two letters have been referenced for every LOCA plant application since December, 1978 and have been part of our model. These items have already been discussed with the Staff and we believe also informally approved by the NRC.

Mr. James R. Miller

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, 1981.

3. Letter from T. M. Anderson, Westinghouse Electric Corporation, to D. Ross, U.S. Nuclear Regulatory Commission, NS-TMA-2354, December 22, 1980.

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).

Appendix

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-IV 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 how this is accomplished.

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

The instantaneous heat up rate is used until the cladding temperature is within 20[illegible] 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 calculations. As long as the clad temperature is above the reference temperature (Points B, C and D), the heat up rate at each succeeding time step is determined by:

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

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°F, 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.

FIGURE 1

The fuel rod uniform strain model in the current LOCTA code was developed from the data that was published by Hardy.[6] 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

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

where ε is the true strain, α 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 Westingnouse 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 consistent 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. 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 only 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-9528. 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 was 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-dimensional 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. Changing from the one-dimensional to the two dimensional downcomer resulted in only an eleven oF change in PCT.

Table 1

	REFLOOD NODE					BURST NODE			
Case	FQT	C_D	PCT	TIME	LOCATION	PCT	TIME	LOCATION	MODEL
A.1	1.93	0.6	1974	104.6	7.5	1859	47.13	5.75	FEB 78
A.2	1.93	0.8	2025	106.8	7.5	2156	51.0	5.75	FEB 78
B.1	2.20	0.6	2048	134.6	7.5	1827	50.82	6.25	UHT TECHNOLOGY
B.2	2.20	0.8	1985	143.8	7.75	1737	50.65	6.25	UHT TECHNOLOGY

TABLE 2

	Feb 78 Analysis	UHI Technology	UHI Technology
	CD = 0.6 FQ = 2.32	1-D downcomer CD = 0.6 FQ = 2.32	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	2089	1791	1780
Location, Ft.	7.5	7.5	7.5

3. Optimized Fuel FLECHT Heat Transfer

Westinghouse submitted WCAP-9500 for approval and Westinghouse has answered NRC review questions on the use of the FLECHT correlation for optimized fuel rods. Westinghouse has submitted supplementary information (Attachment 2) describing the adjustment to the FLECHT correlation for optimized fuel rods. Westinghouse 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	Paint	
	0.375	Steel	
	36.0	Concrete	49,838
2)	0.0065	Paint	
	0.500	Steel	
	36.0	Concrete	32,072
3)	12.0	Concrete	15,000
4)	0.375	Stainless Steel	
	12.0	Concrete	10,000
5)	12.0	Concrete	61,000
6)	0.0065	Paint	
	0.500	Steel	68,792
7)	0.0065	Paint	
	0.375	Steel	79,904
8)	0.0065	Paint	
	0.0250	Steel	27,948
9)	0.0065	Paint	
	0.1875	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

TABLE 5

DECLG $C_D = 0.4$

Indian Point Unit #4

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	1584.802	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	73497.388	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

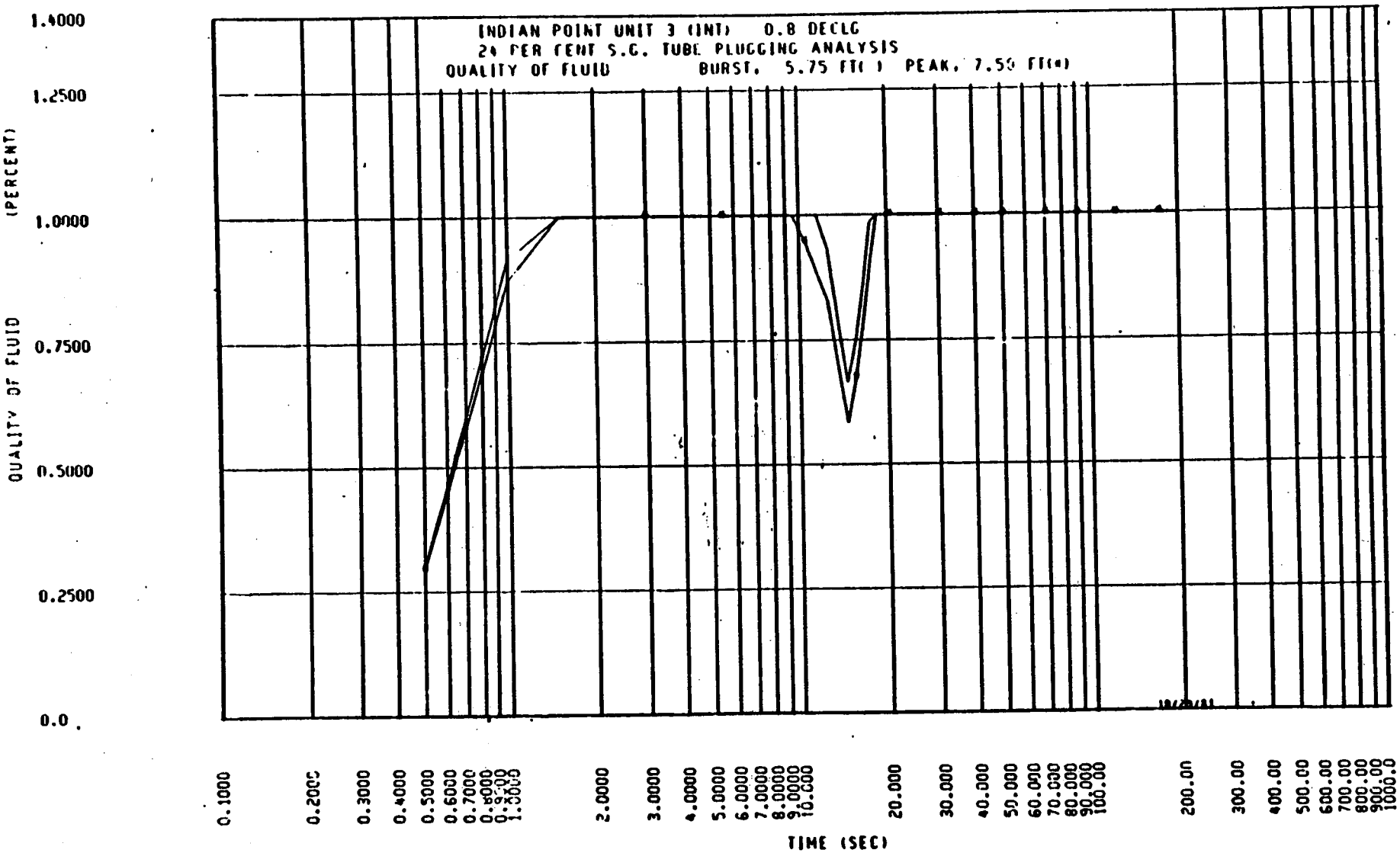

FIGURE 1A FLUID QUALITY
DECLG(CD = 0.8)

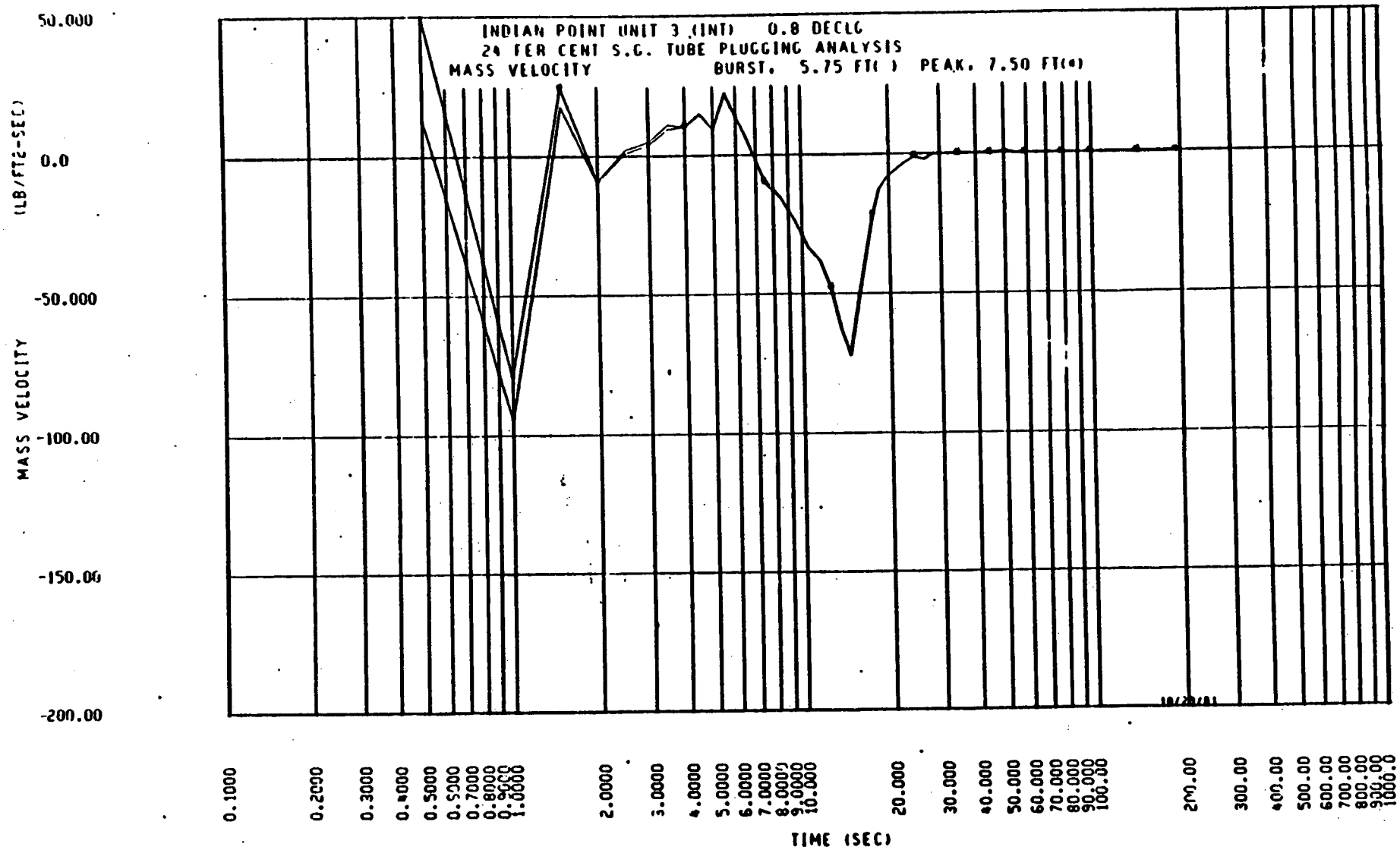

FIGURE 2A MASS VELOCITY
DECLG(CD = 0.8)

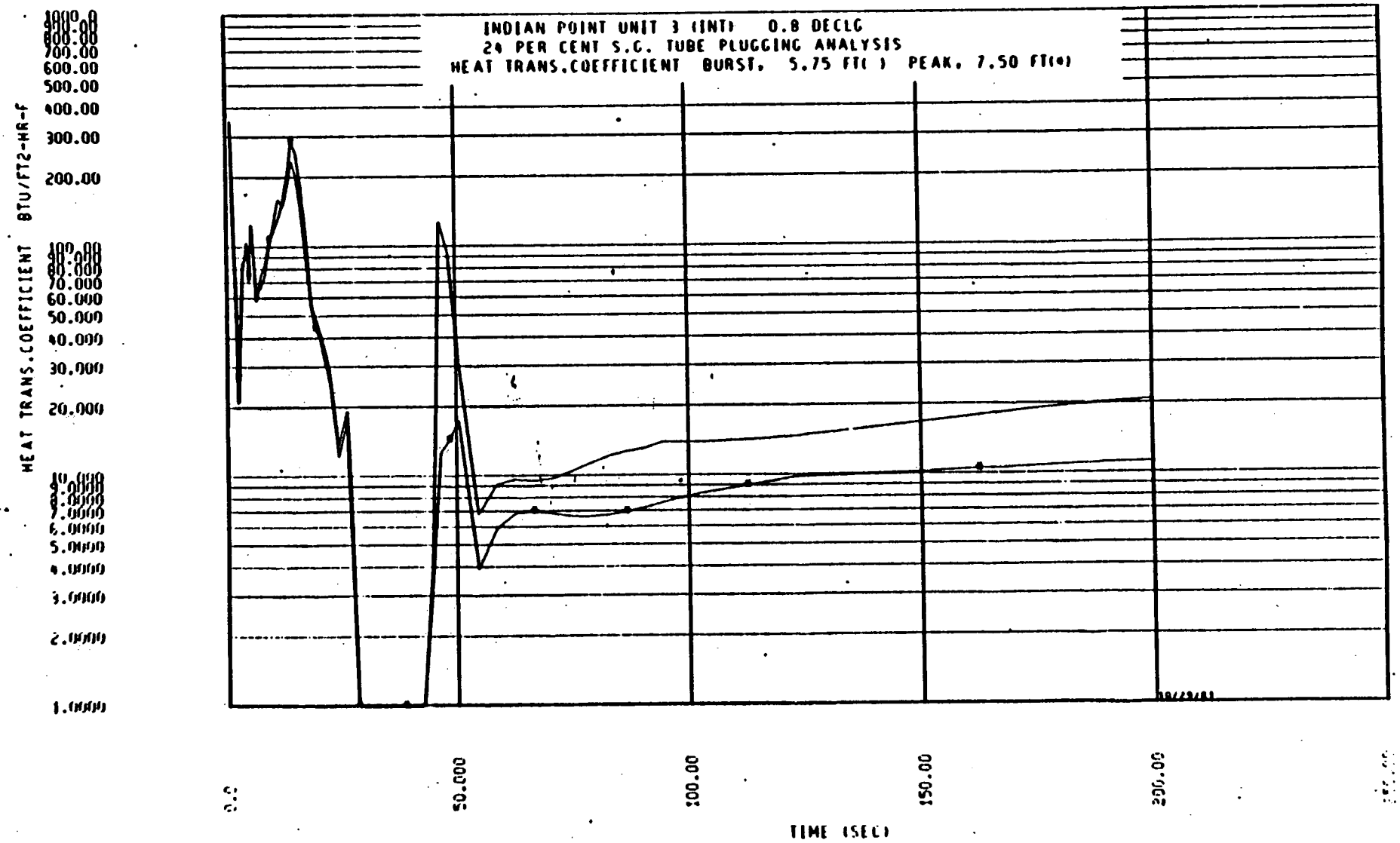

FIGURE 3A HEAT TRANSFER COEFFICIENT
DECLG(CD = 0.8)

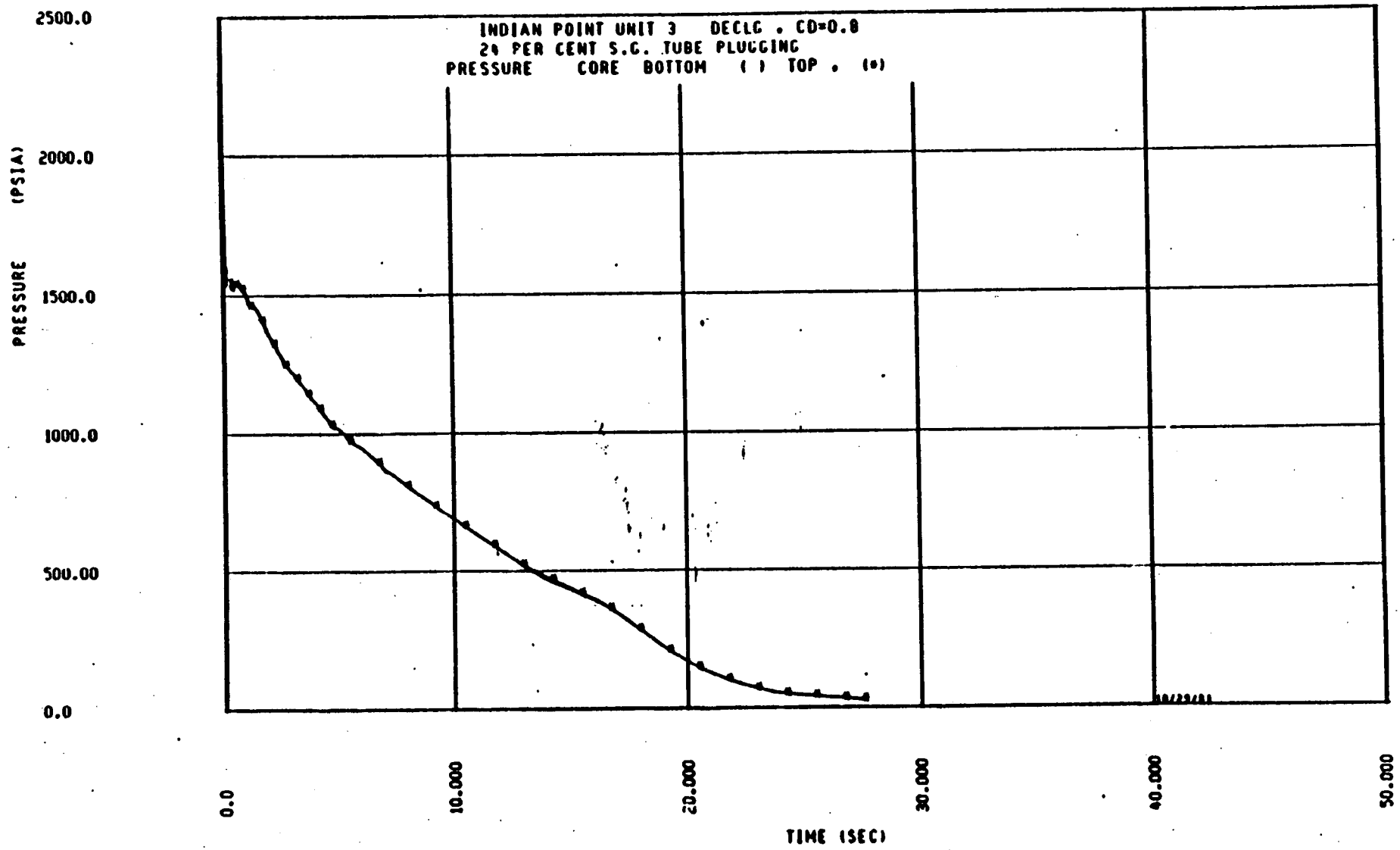

FIGURE 4A CORE PRESSURE
DECLG(CD = 0.8)

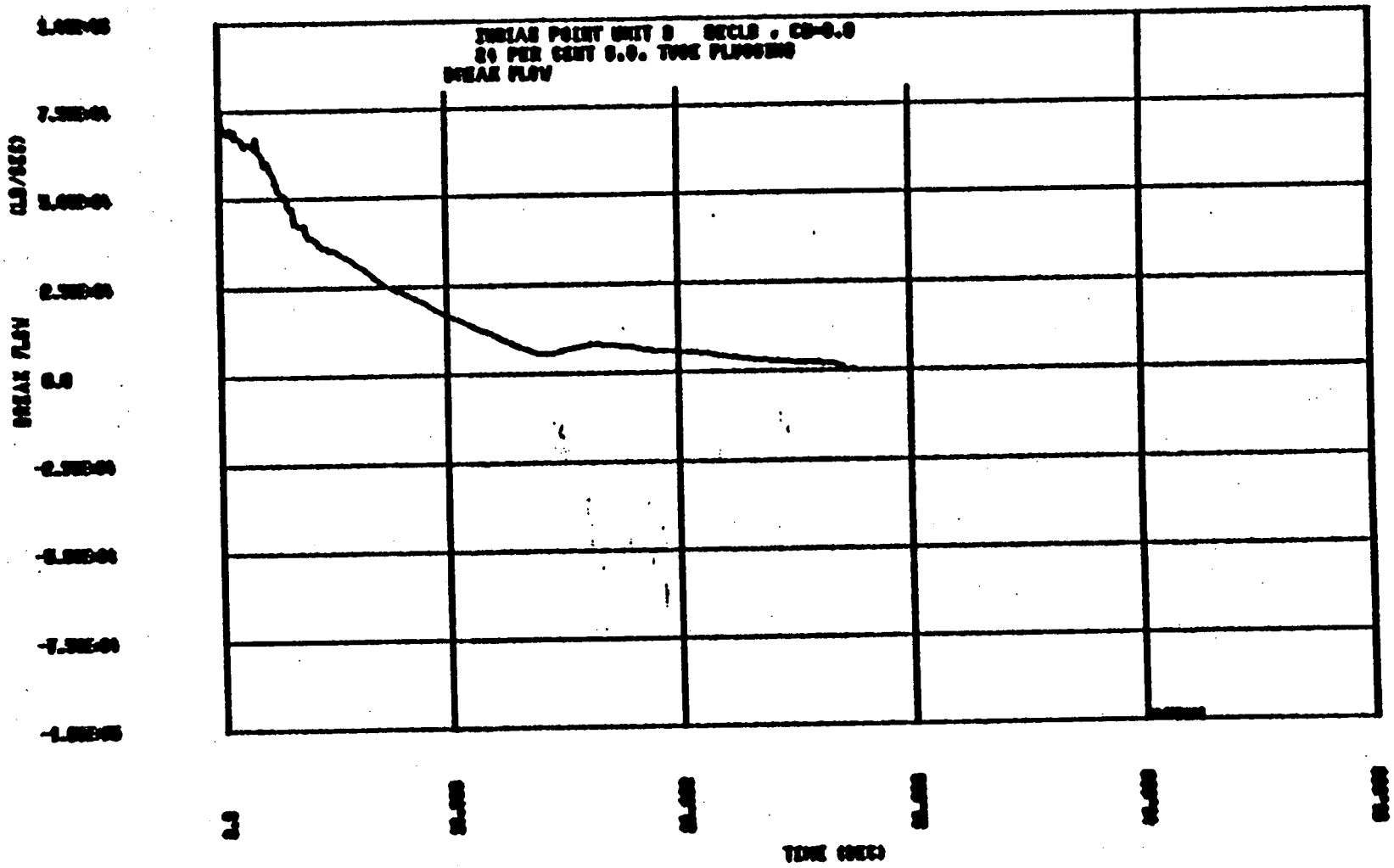

FIGURE 5A BREAK FLOW RATE
DECLG(CD = 0.8)

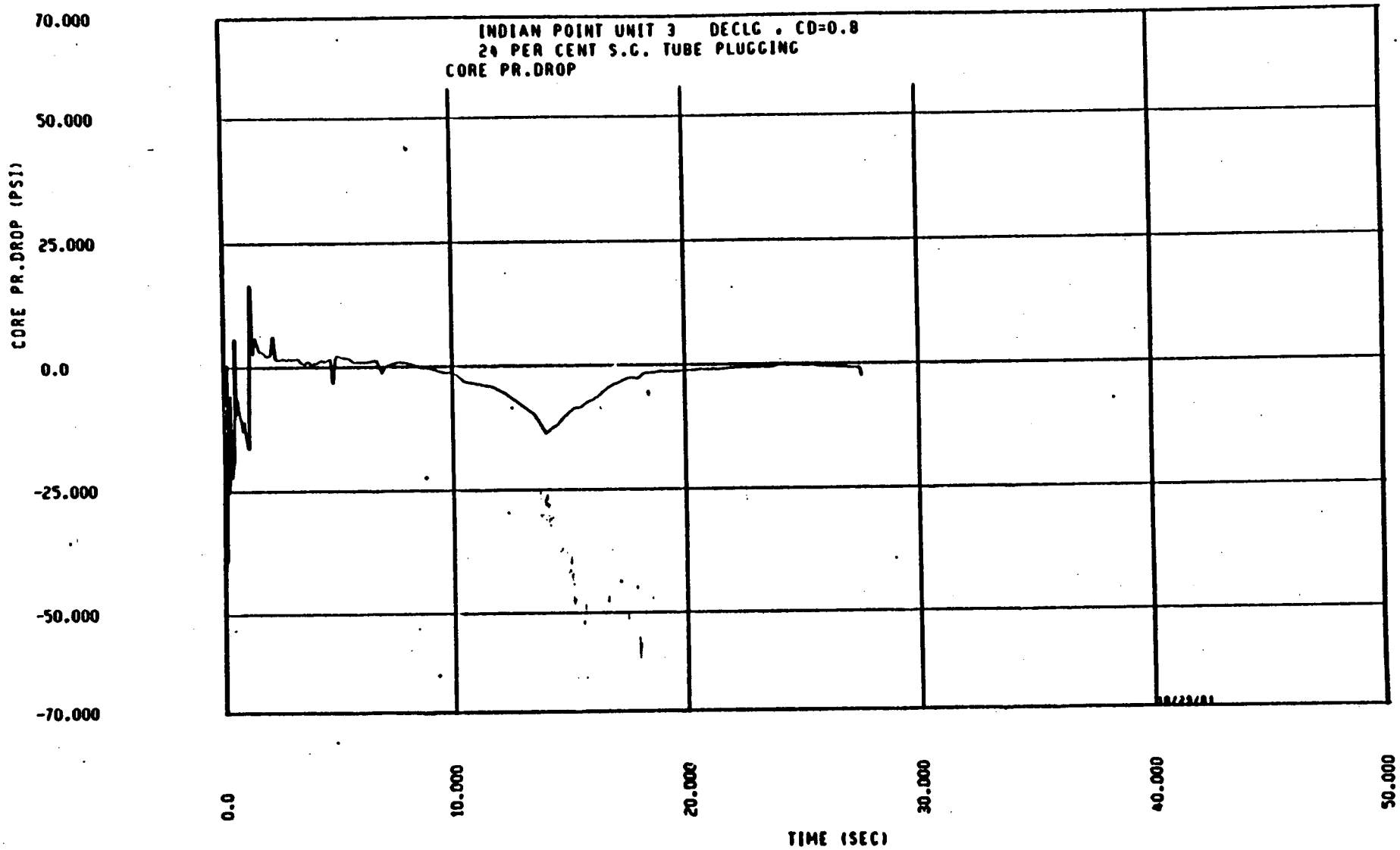

FIGURE 6A CORE PRESSURE DROP
DECLG(CD = 0.8)

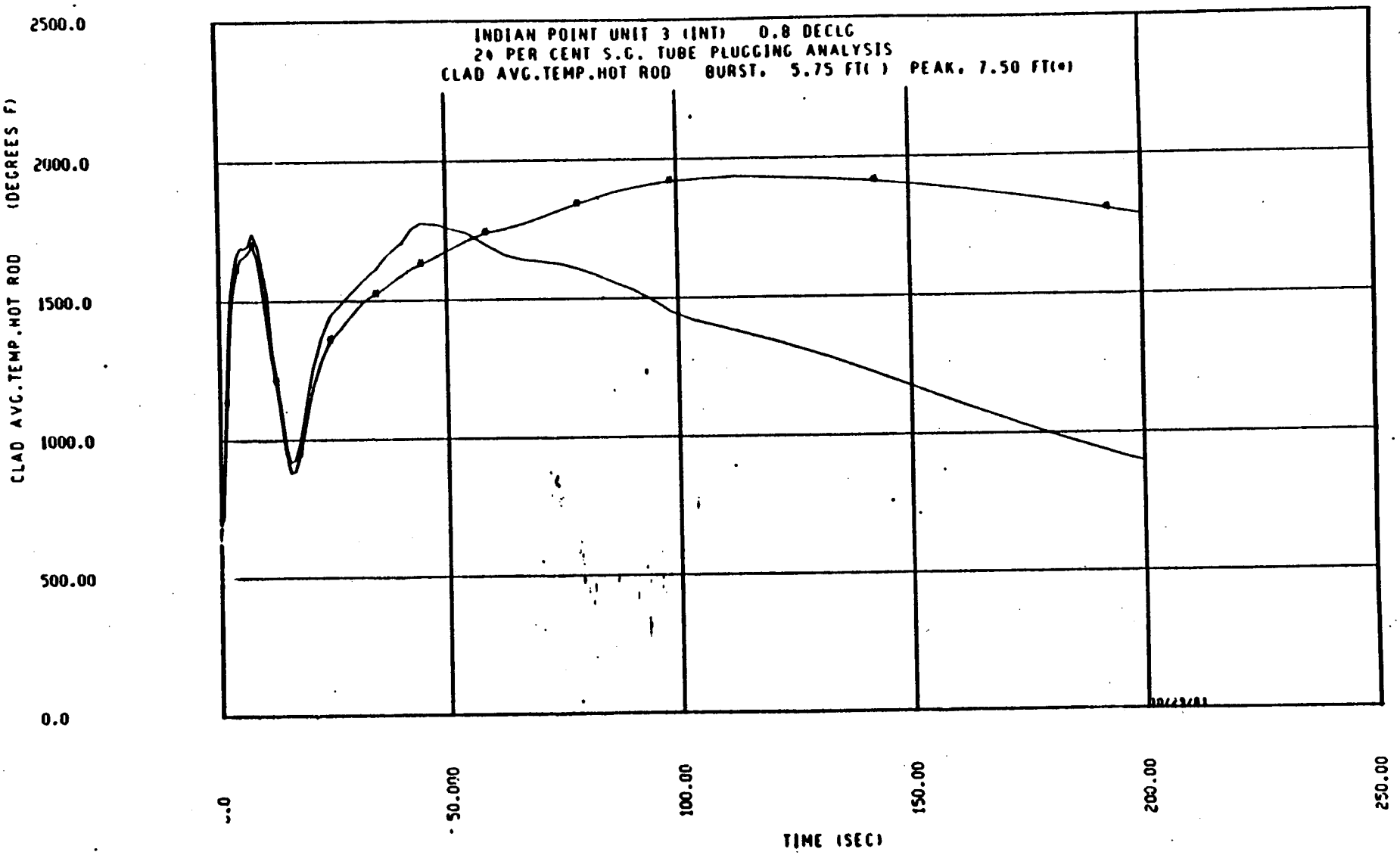

FIGURE 7A PEAK CLAD TEMPERATURE
DECLG(CD = 0.8)

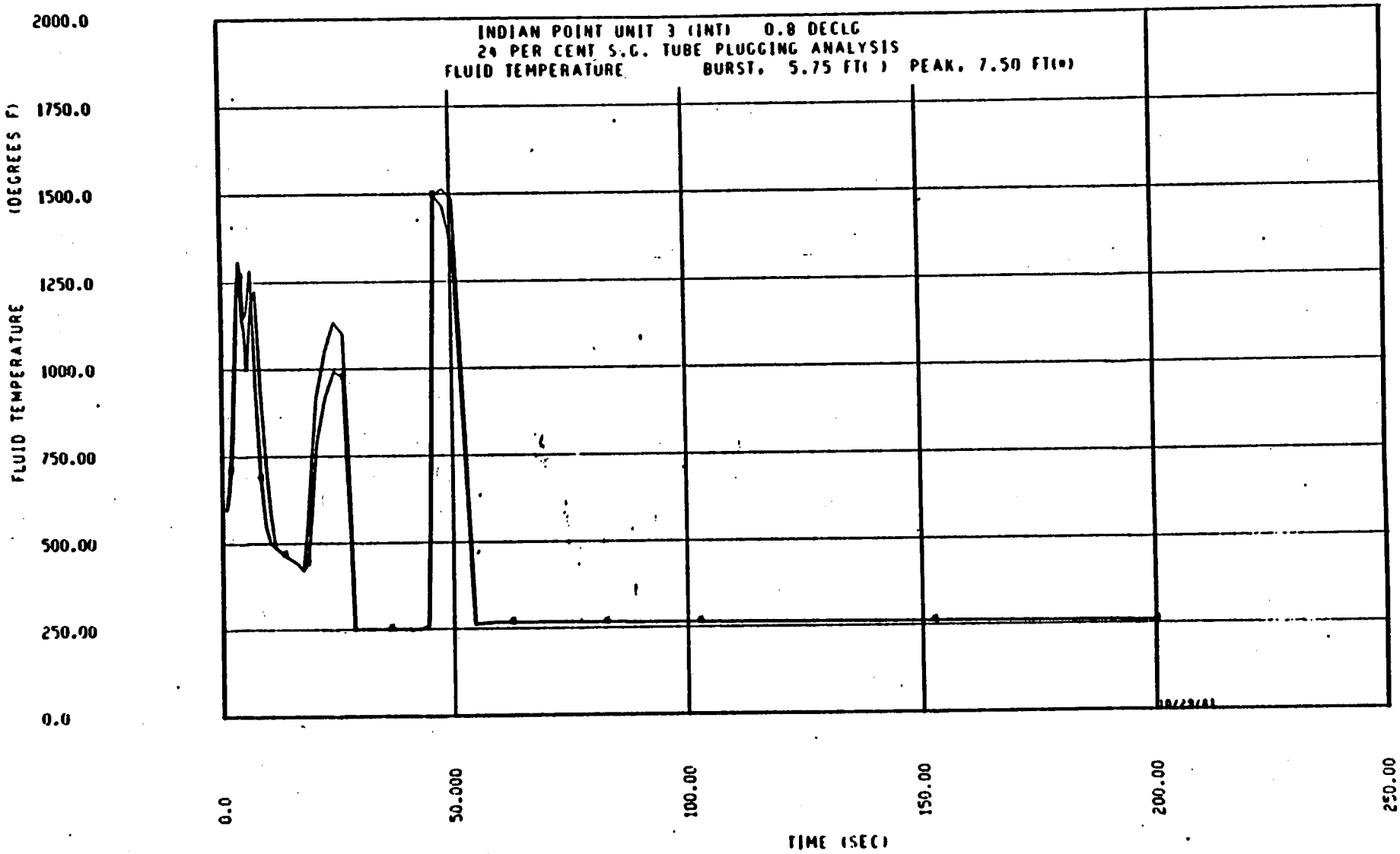

FIGURE 8A FLUID TEMPERATURE
DECLG(CD = 0.8)

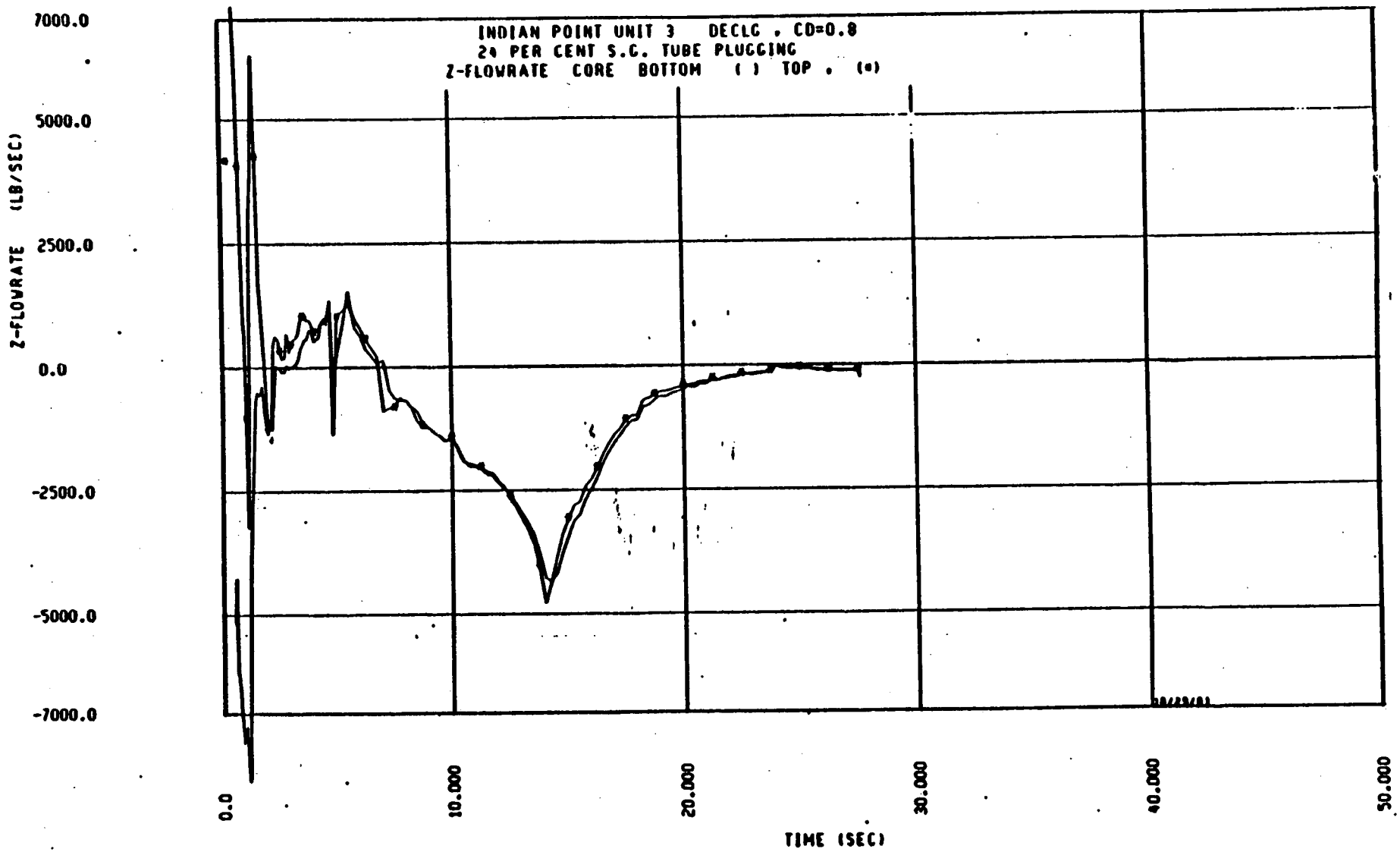

FIGURE 9A CORE FLOW (TOP AND BOTTOM)
DECLG(CD = 0.8)

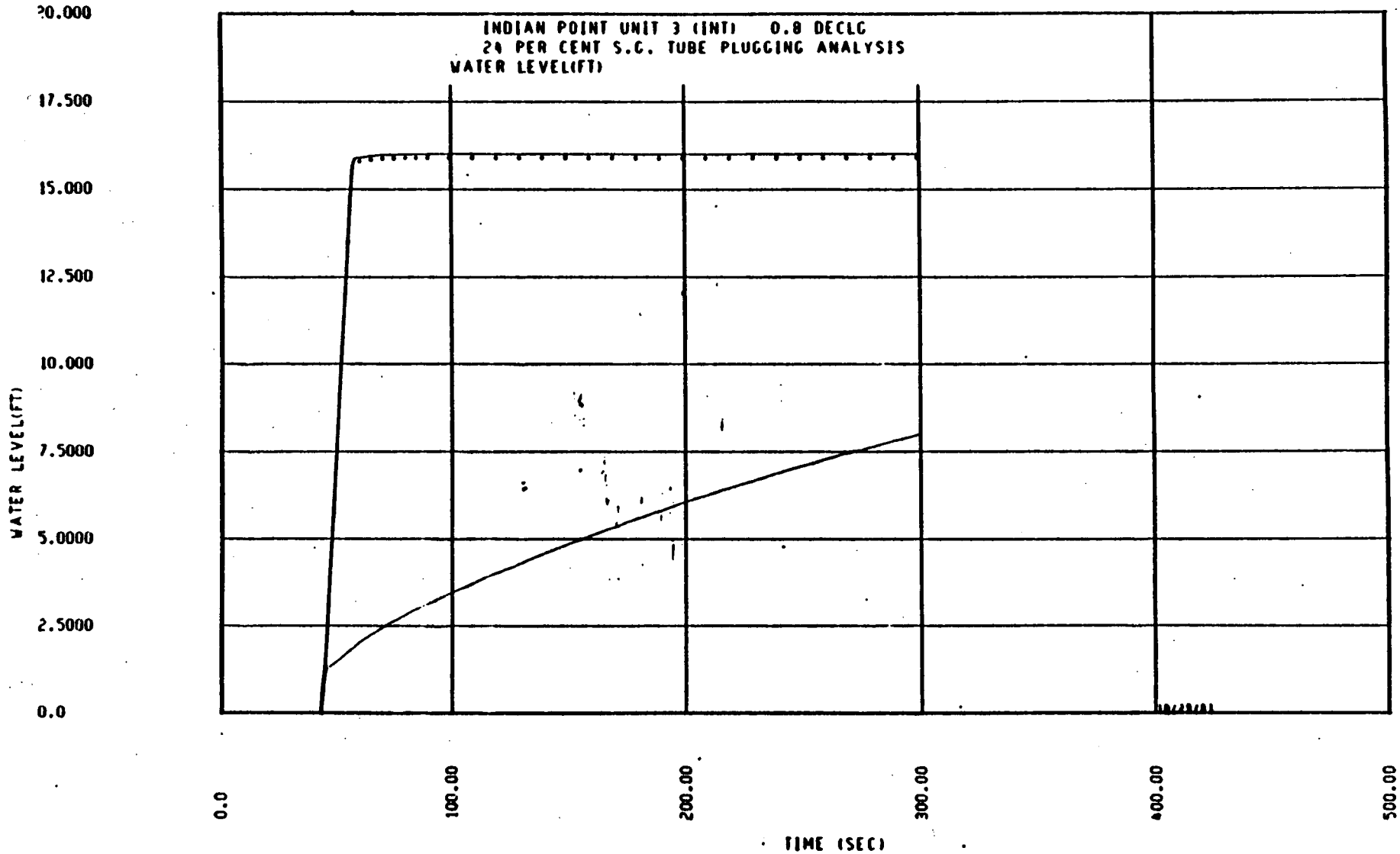

FIGURE 10A REFLOOD TRANSIENT - CORE & DOWNCOMER WATER LEVELS DECLG(CD = 0.8)

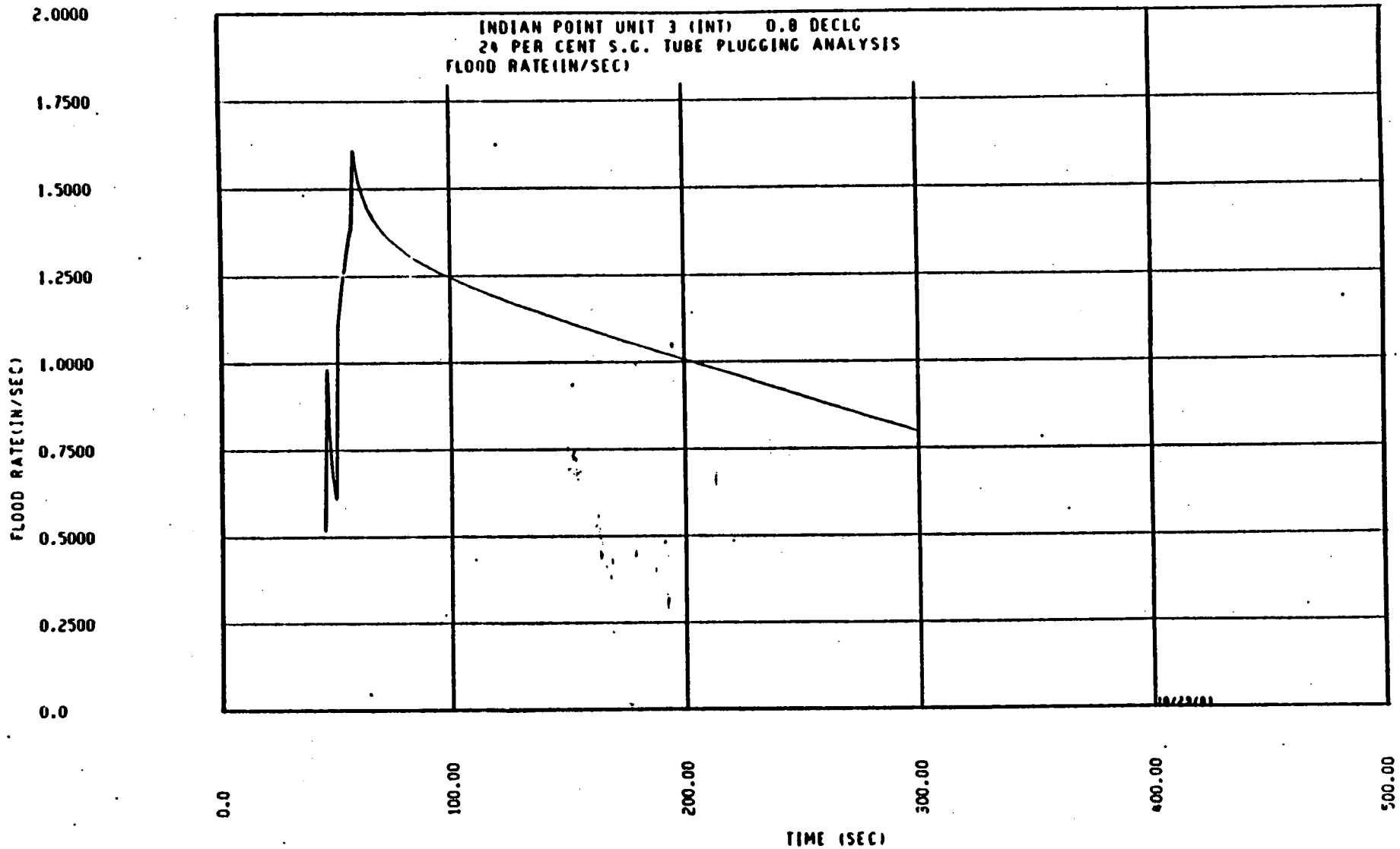

FIGURE 11A REFLOOD TRANSIENT
CORE INLET VELOCITY
DECLG(CD = 0.8)

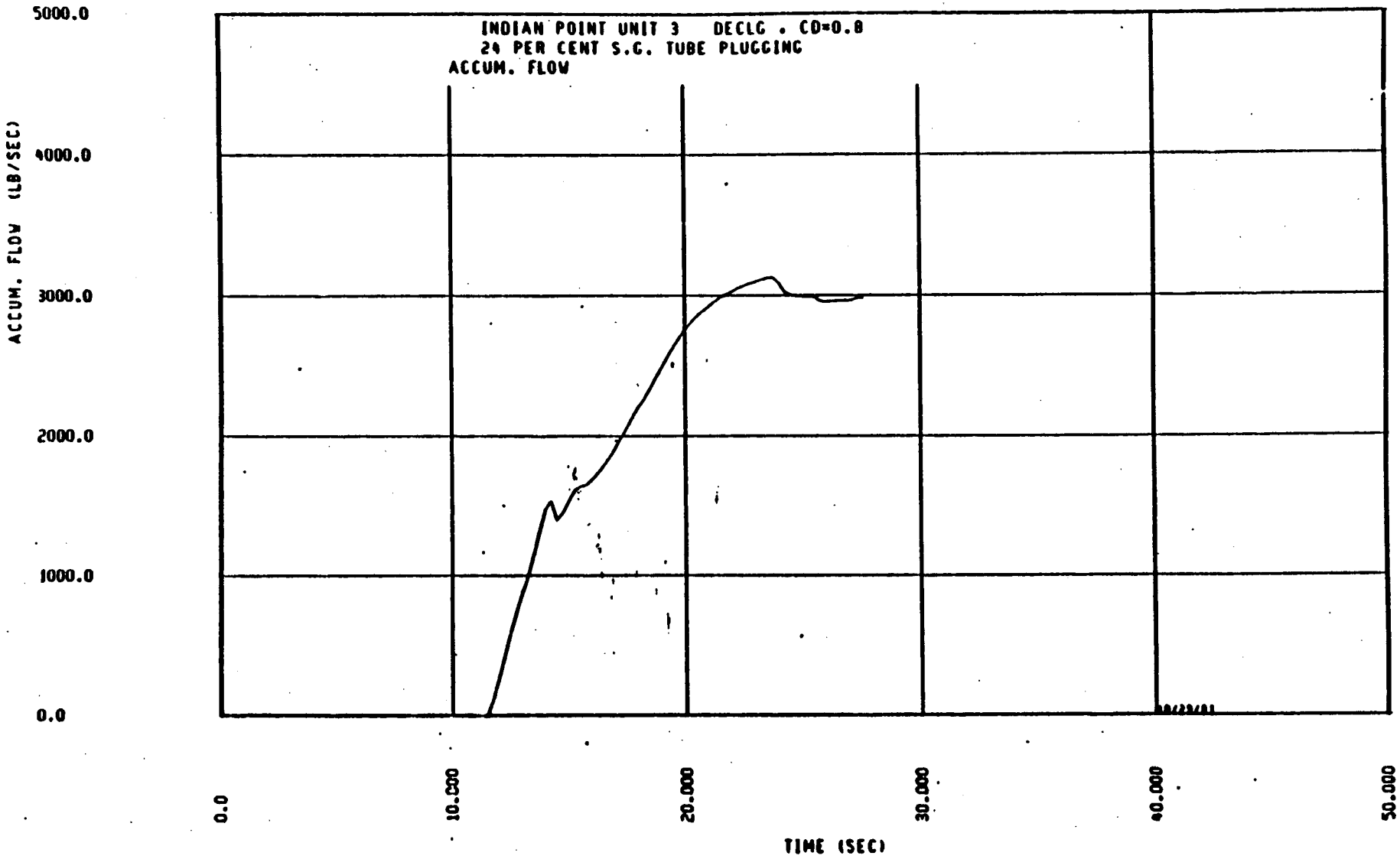

FIGURE 12A ACCUMULATOR FLOW(BLOWDOWN)
DECLG(CD = 0.8)

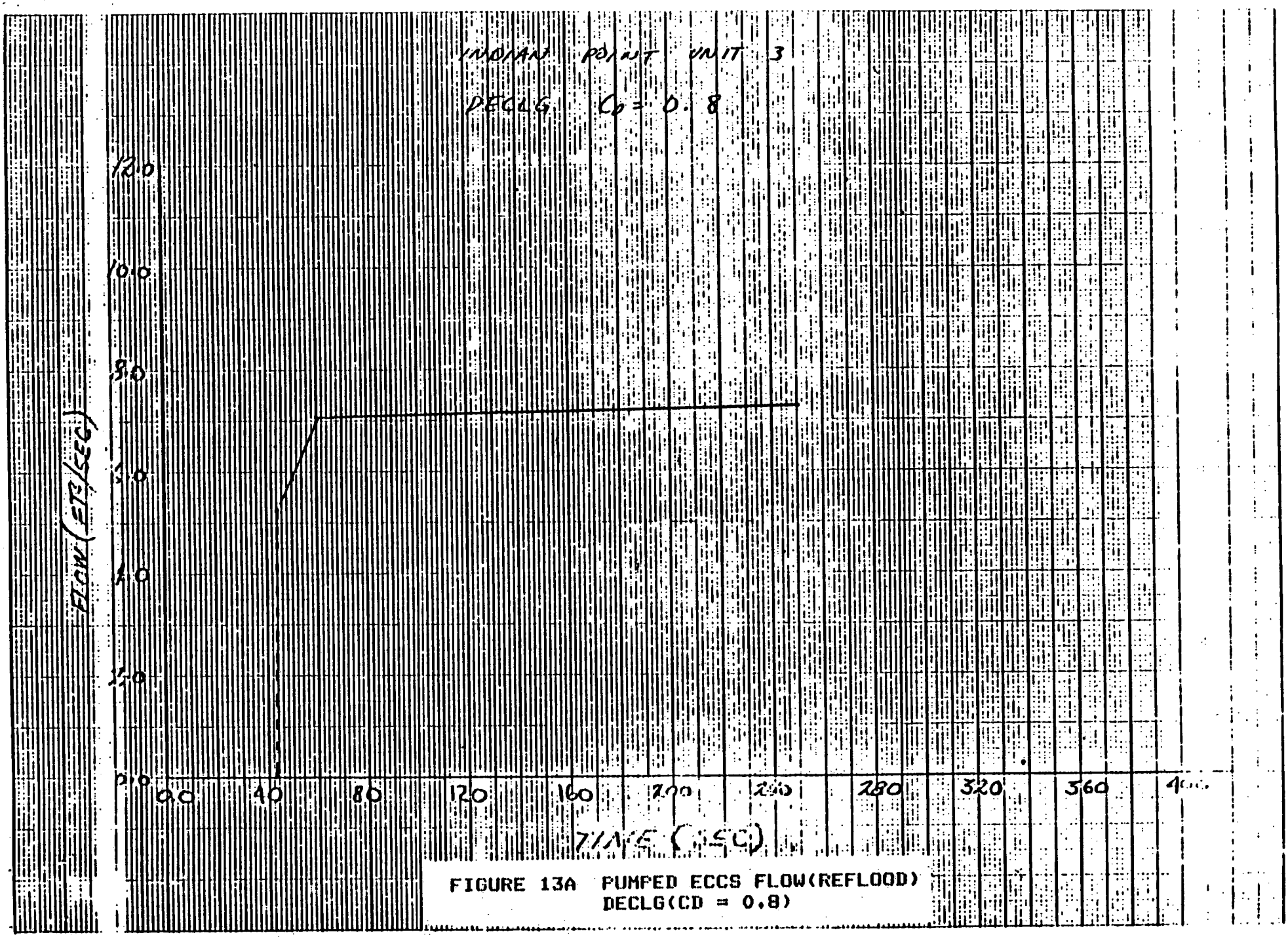

FIGURE 13A PUMPED ECCS FLOW(REFLOOD)
DECLG(CD = 0.8)

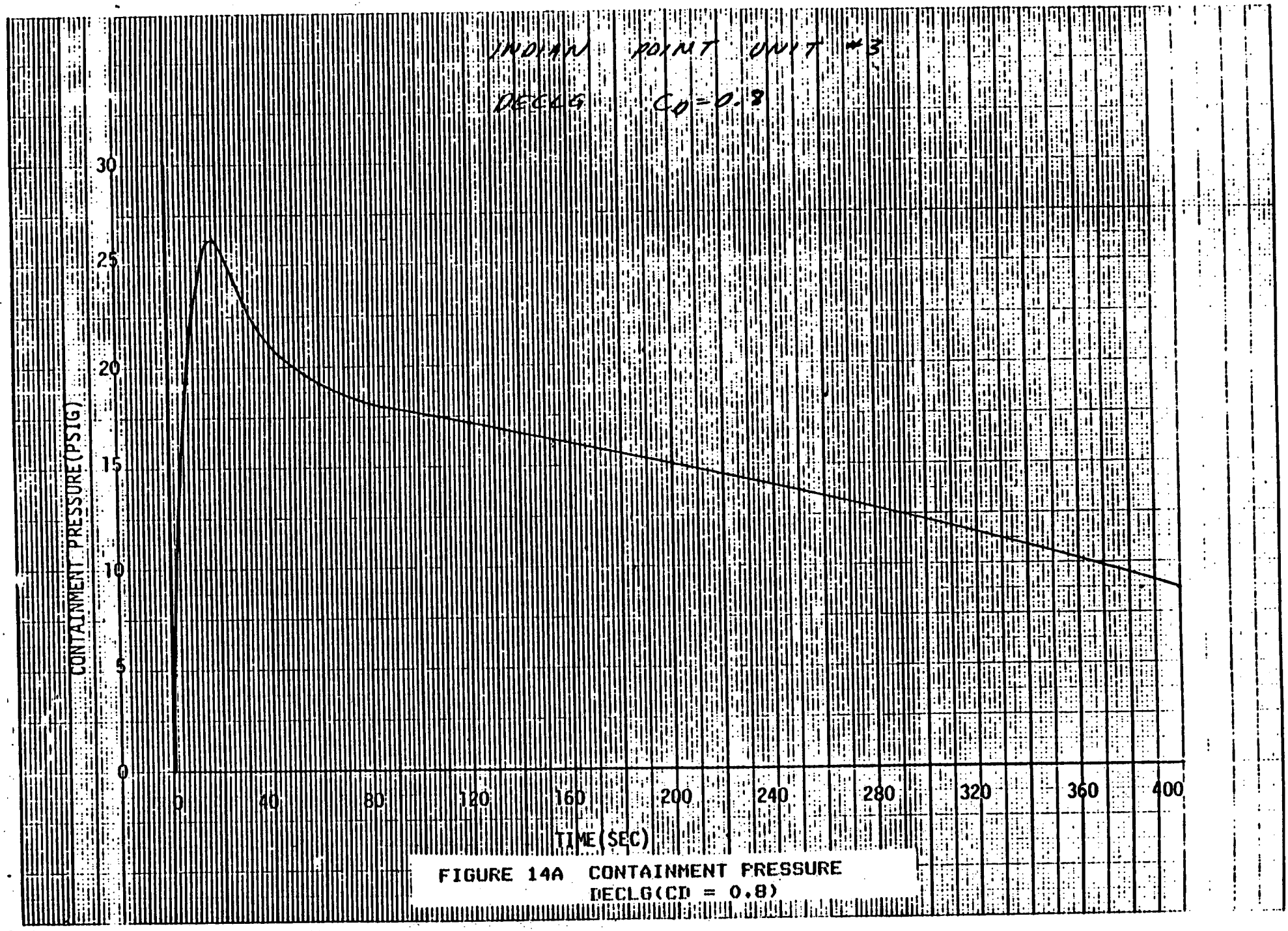

FIGURE 14A CONTAINMENT PRESSURE
DECLG(CD = 0.8)

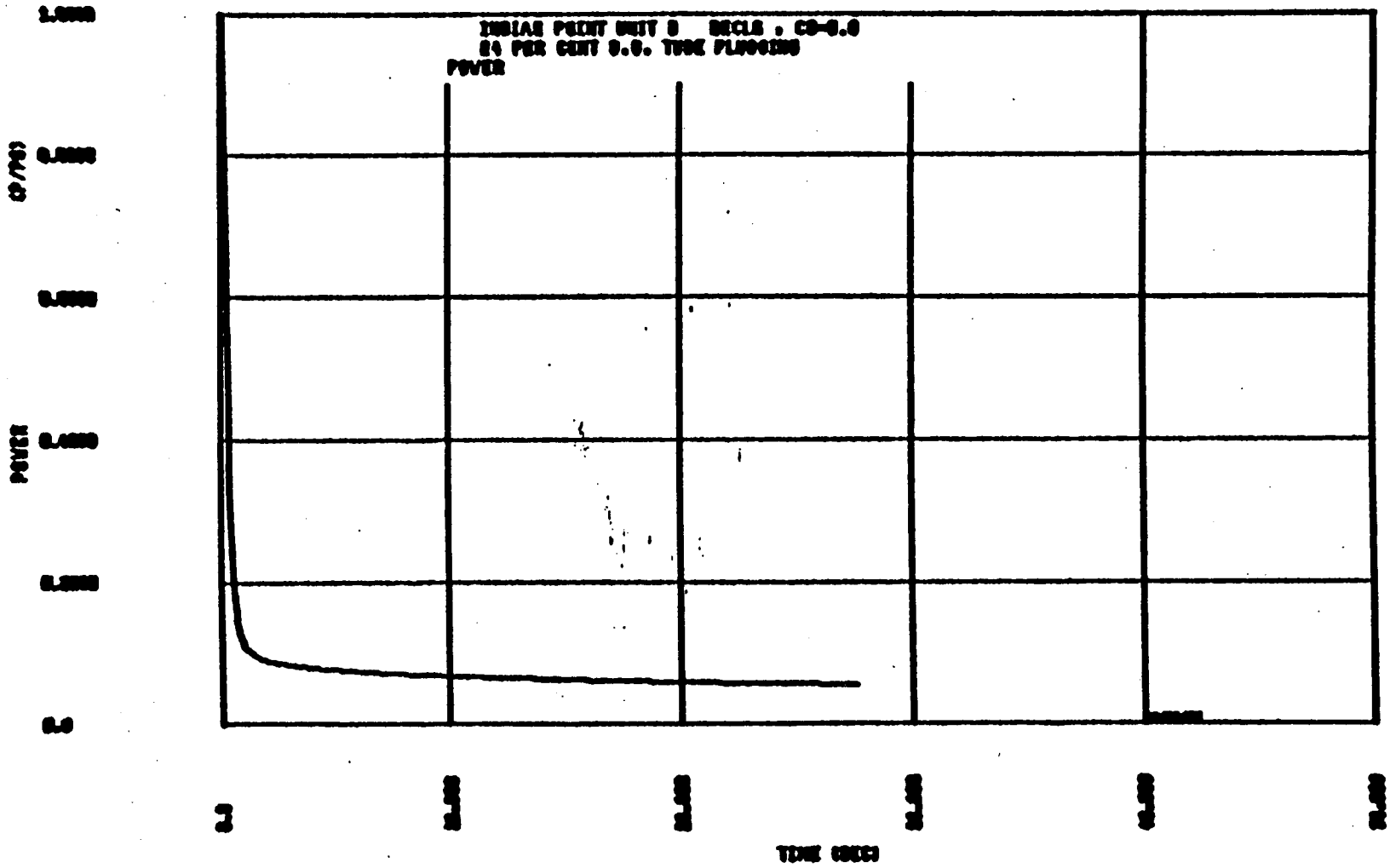

FIGURE 15A CORE POWER TRANSIENT
DECLG(CD = 0.8)

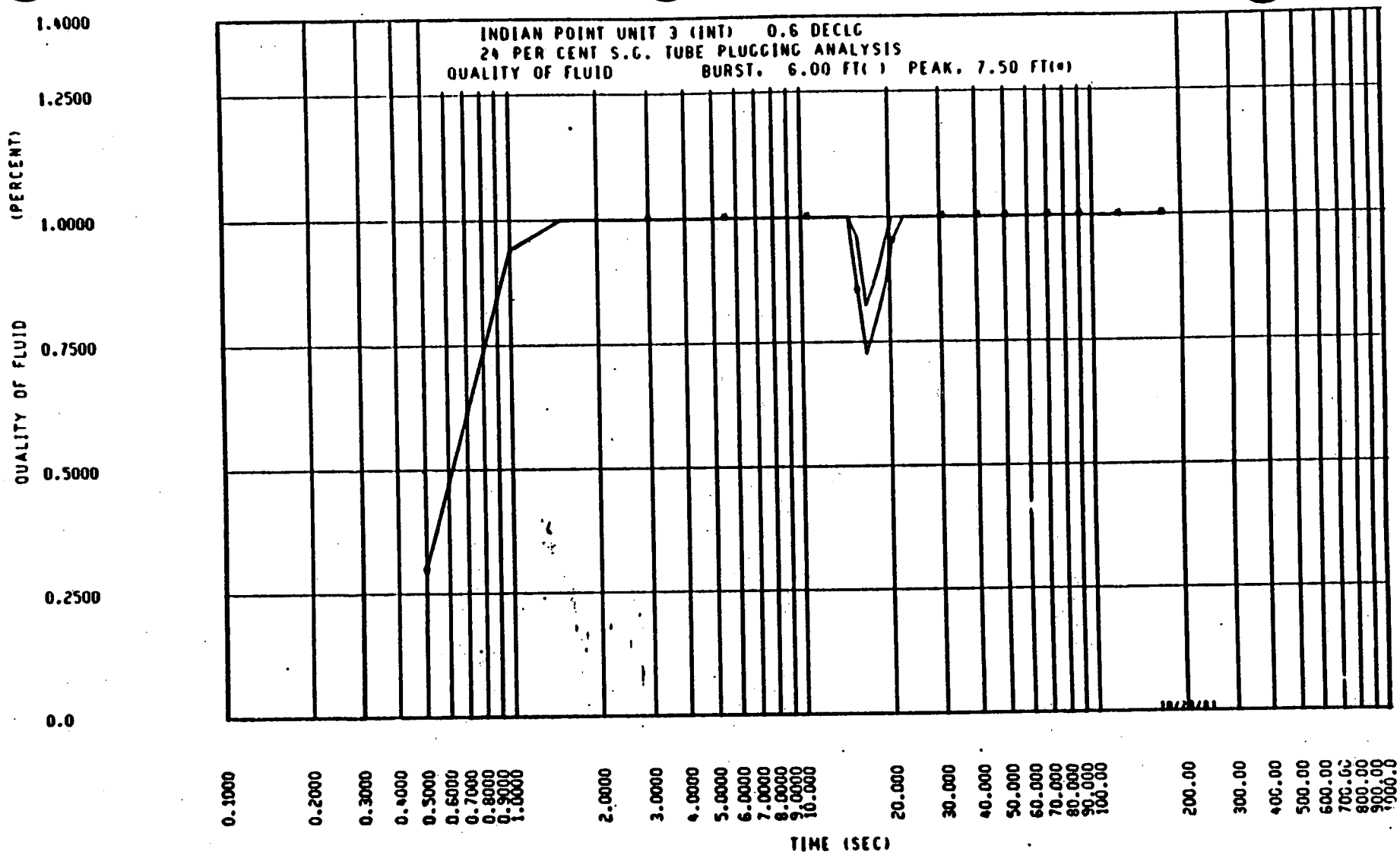

FIGURE 1B FLUID QUALITY
DECLG(CD = 0.6)

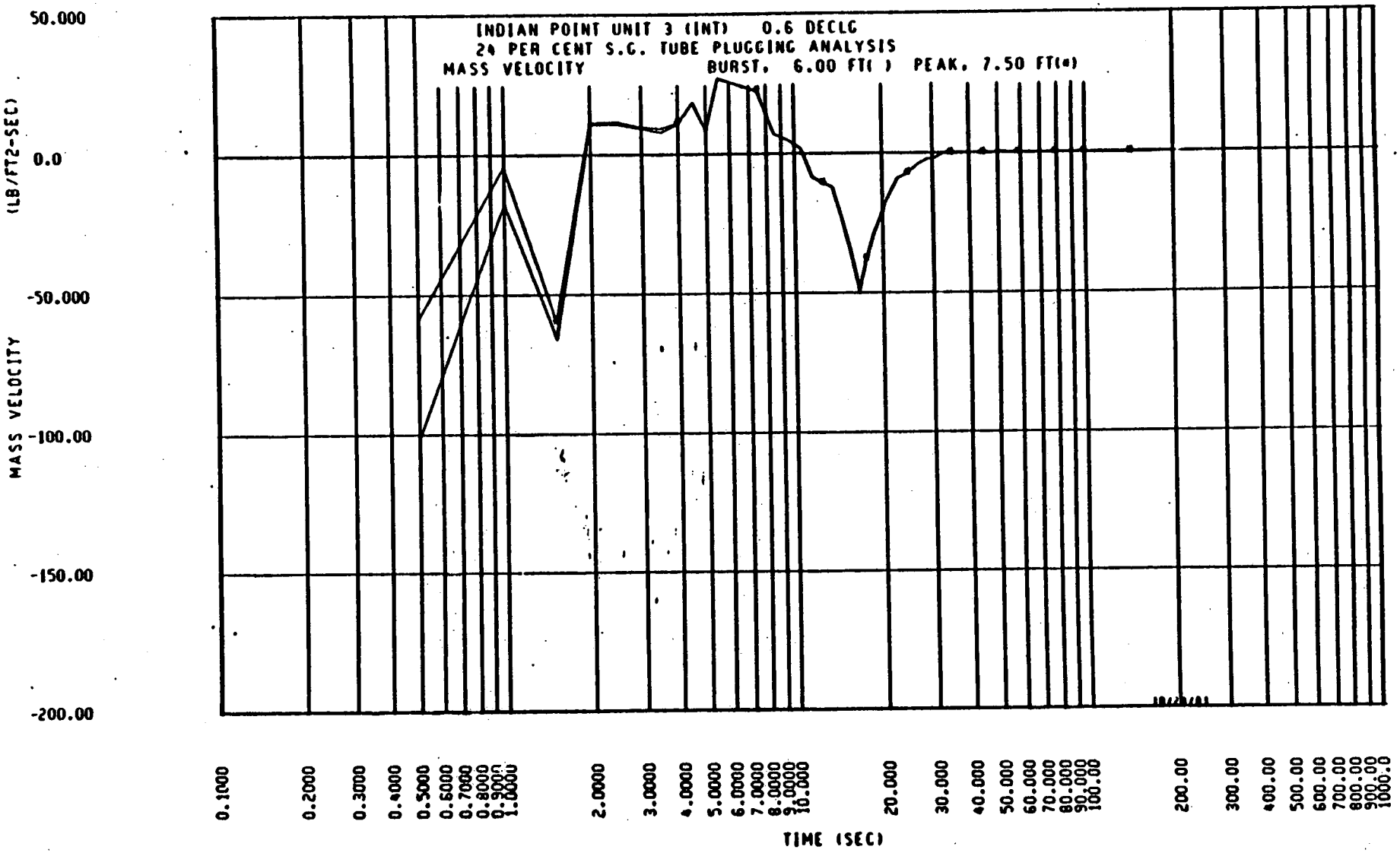

FIGURE 2B MASS VELOCITY
DECLG(CD = 0.6)

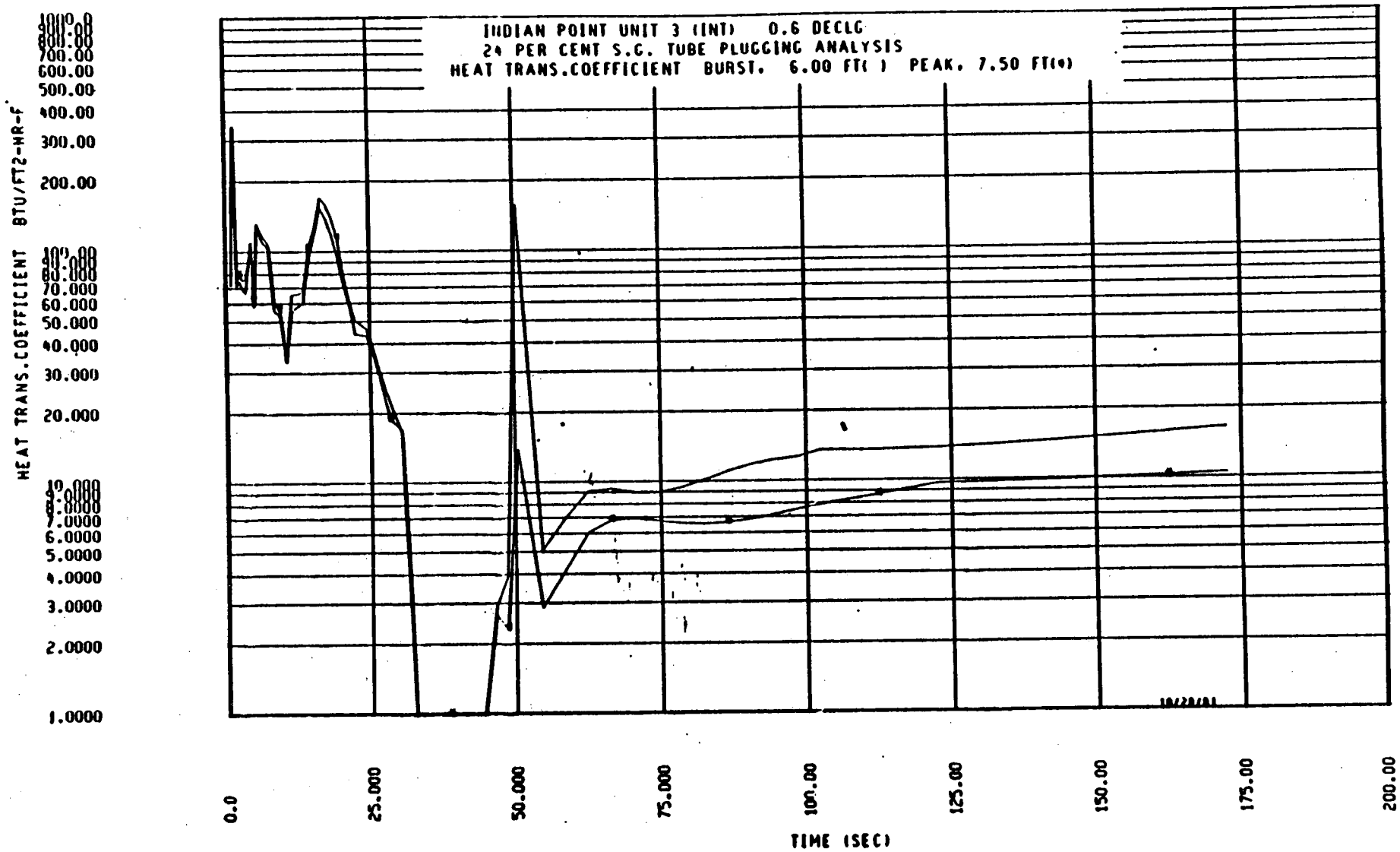

FIGURE 3B HEAT TRANSFER COEFFICIENT
DECLG(CD = 0.6)

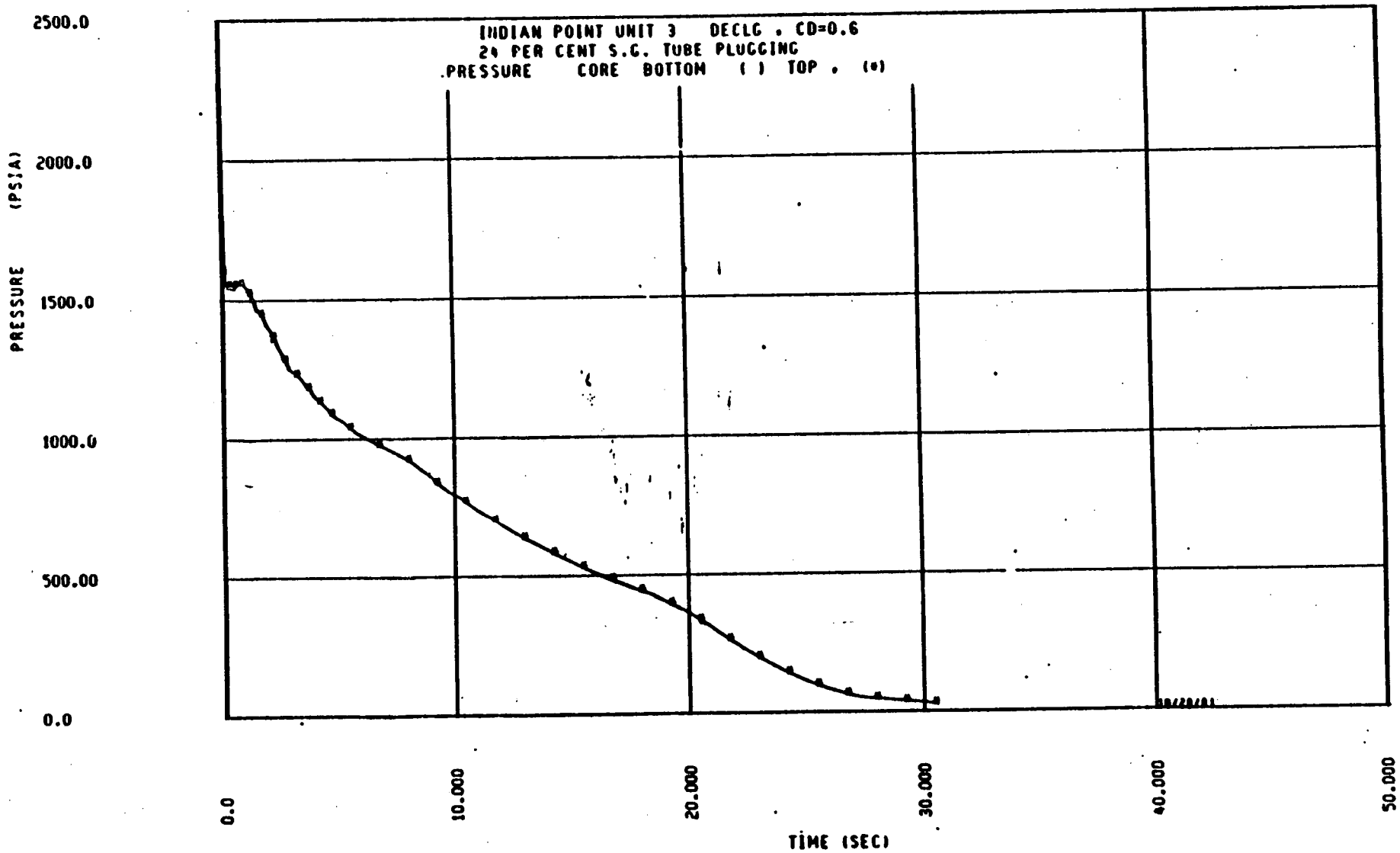

FIGURE 4B CORE PRESSURE
DECLG(CD = 0.6)

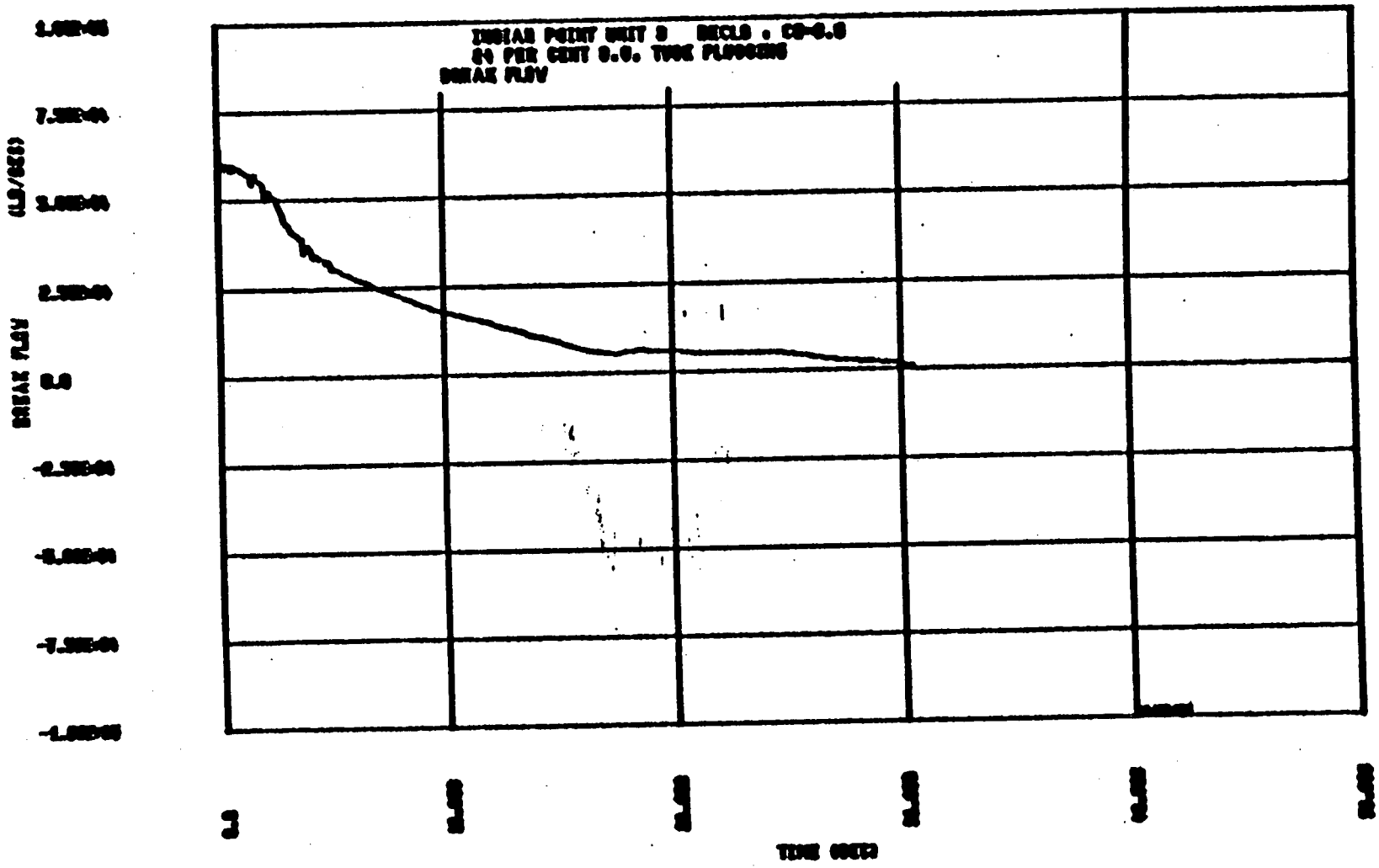

FIGURE 5B BREAK FLOW RATE
DECLG(CD = 0.6)

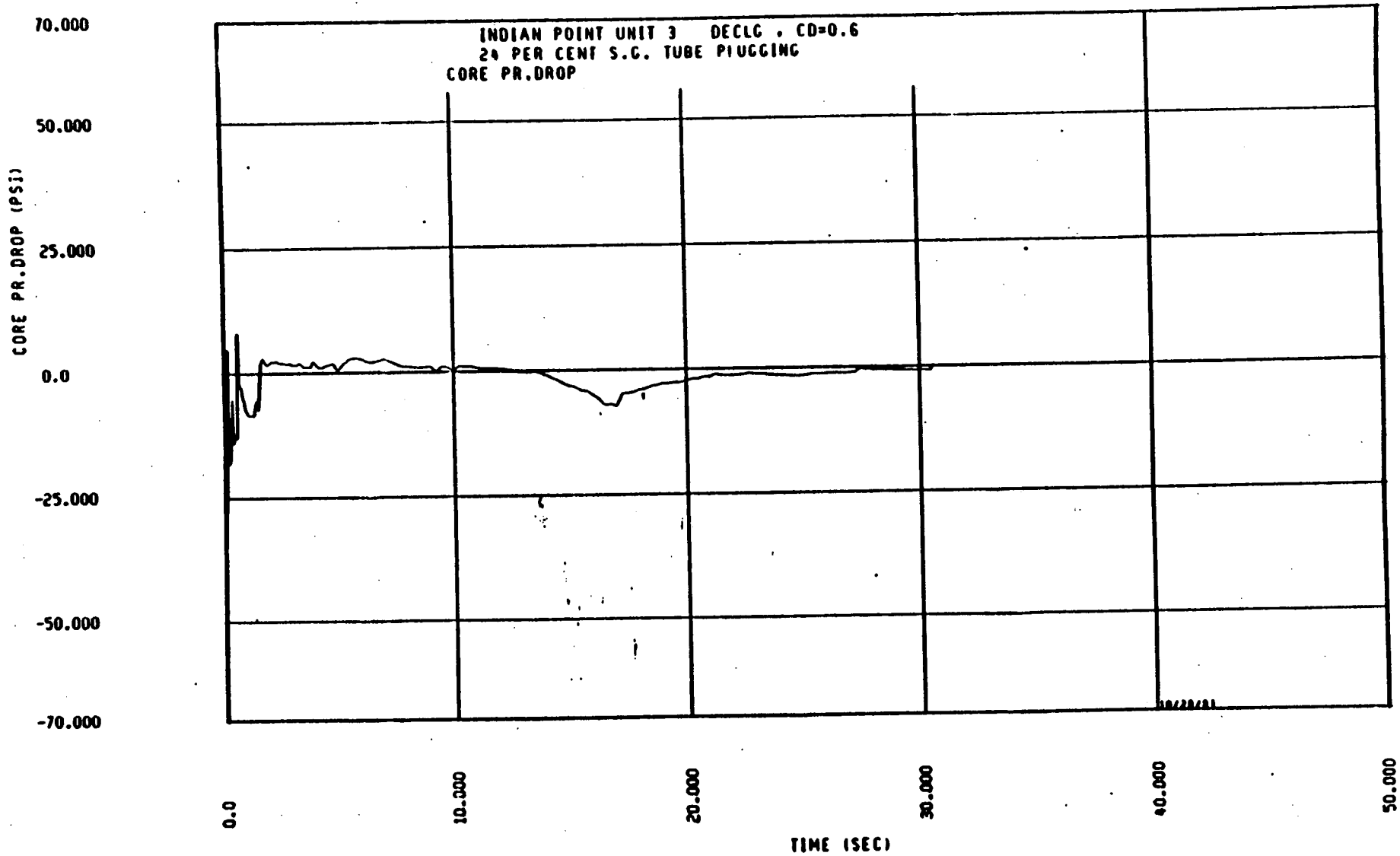

FIGURE 6B CORE PRESSURE DROP
DECLG(CD = 0.6)

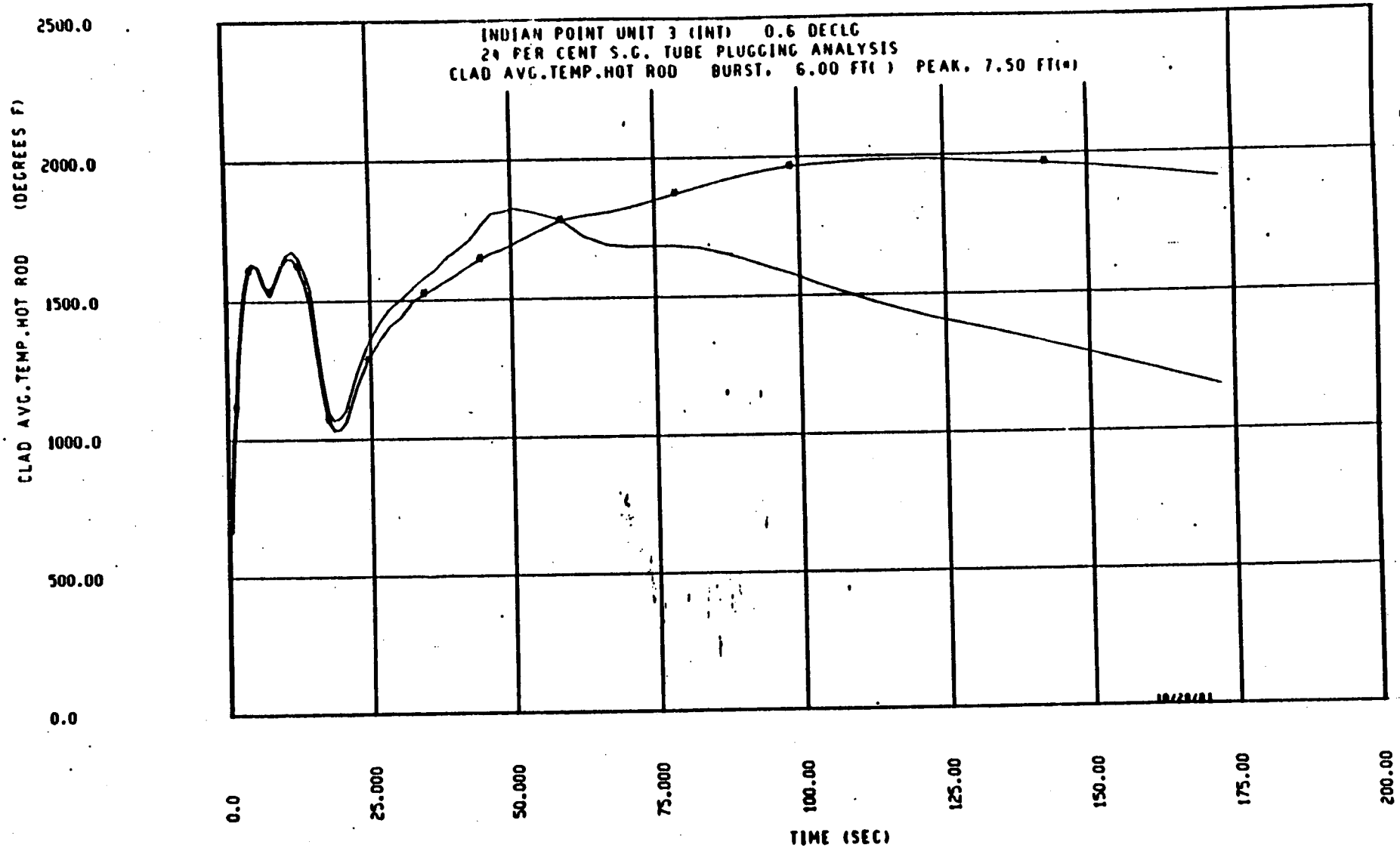

FIGURE 7B PEAK CLAD TEMPERATURE
DECLG(CD = 0.6)

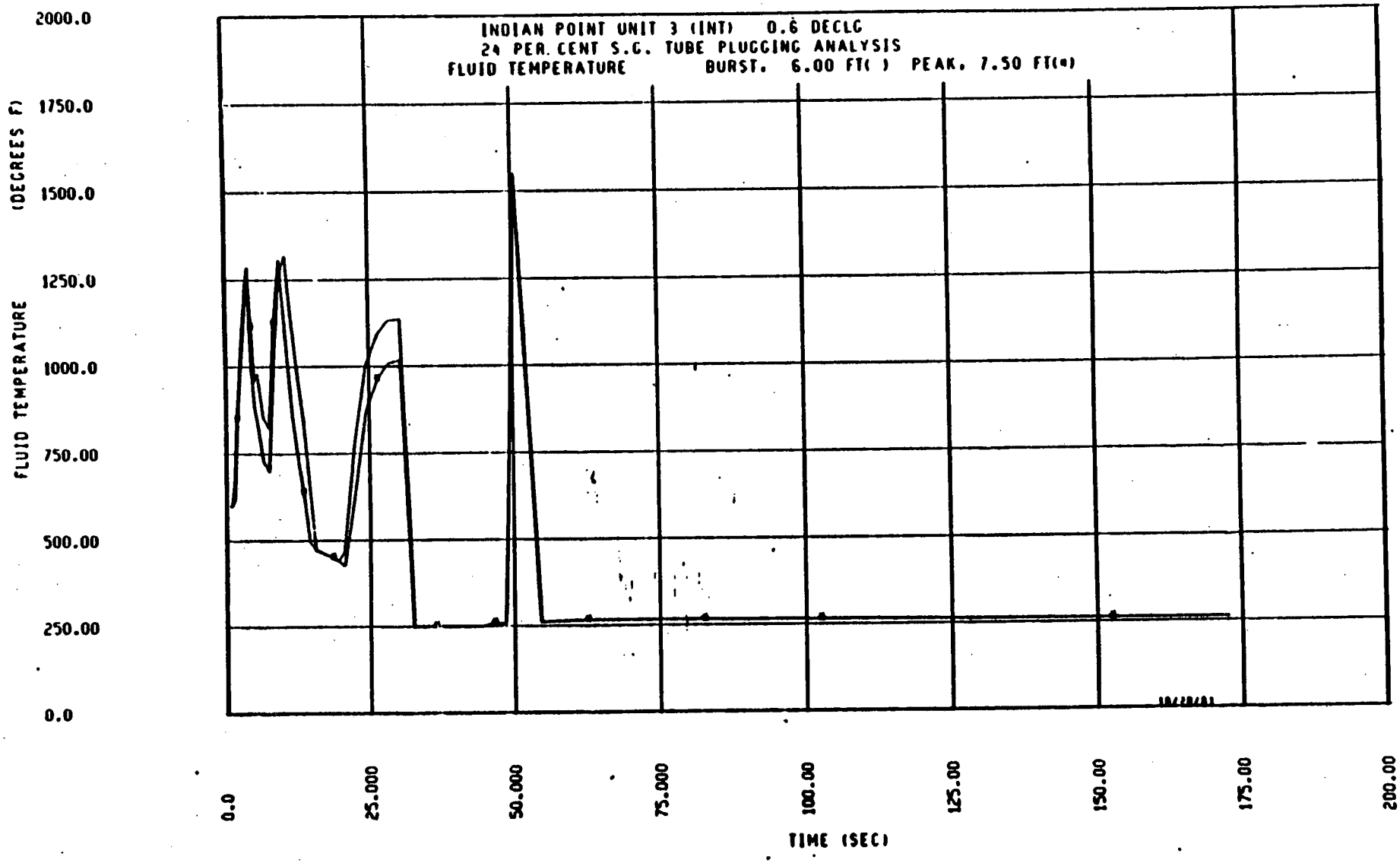

FIGURE 8B FLUID TEMPERATURE
DECLG(CD = 0.6)

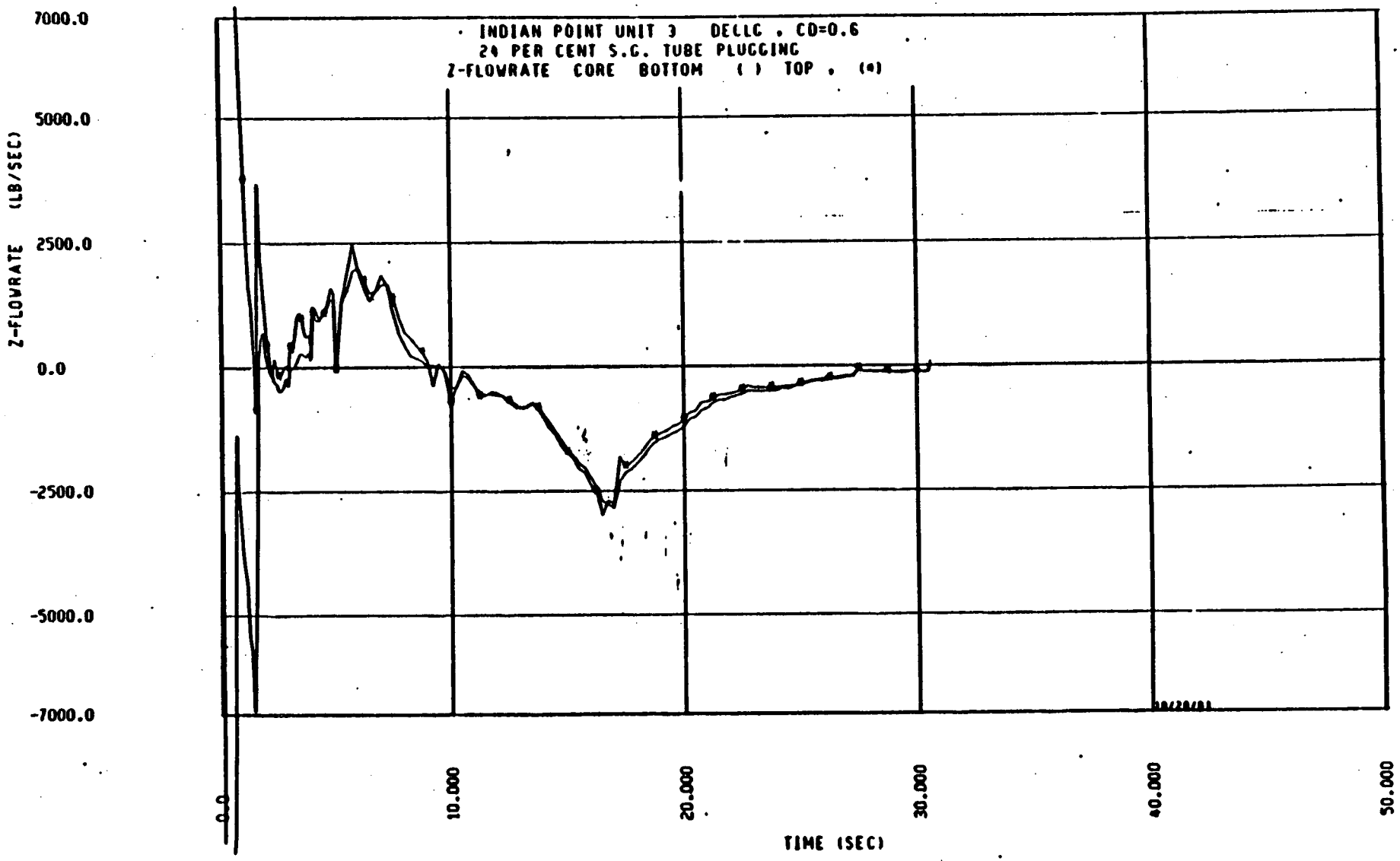

FIGURE 9B CORE FLOW (TOP AND BOTTOM)
DECLG(CD = 0.6)

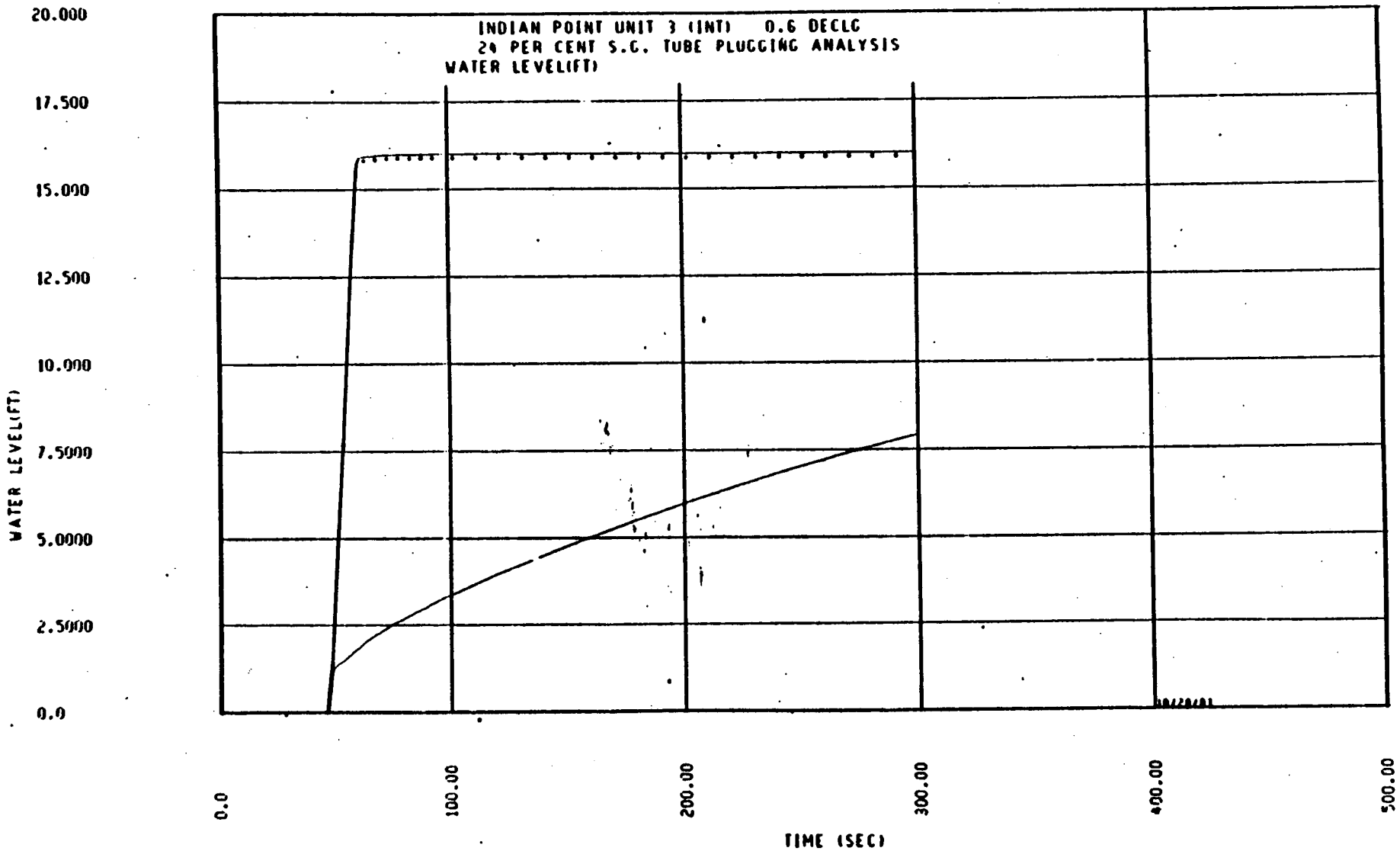

FIGURE 10B REFLOOD TRANSIENT - CORE & DOWNCOMER WATER LEVELS DECLG(CD = 0.6)

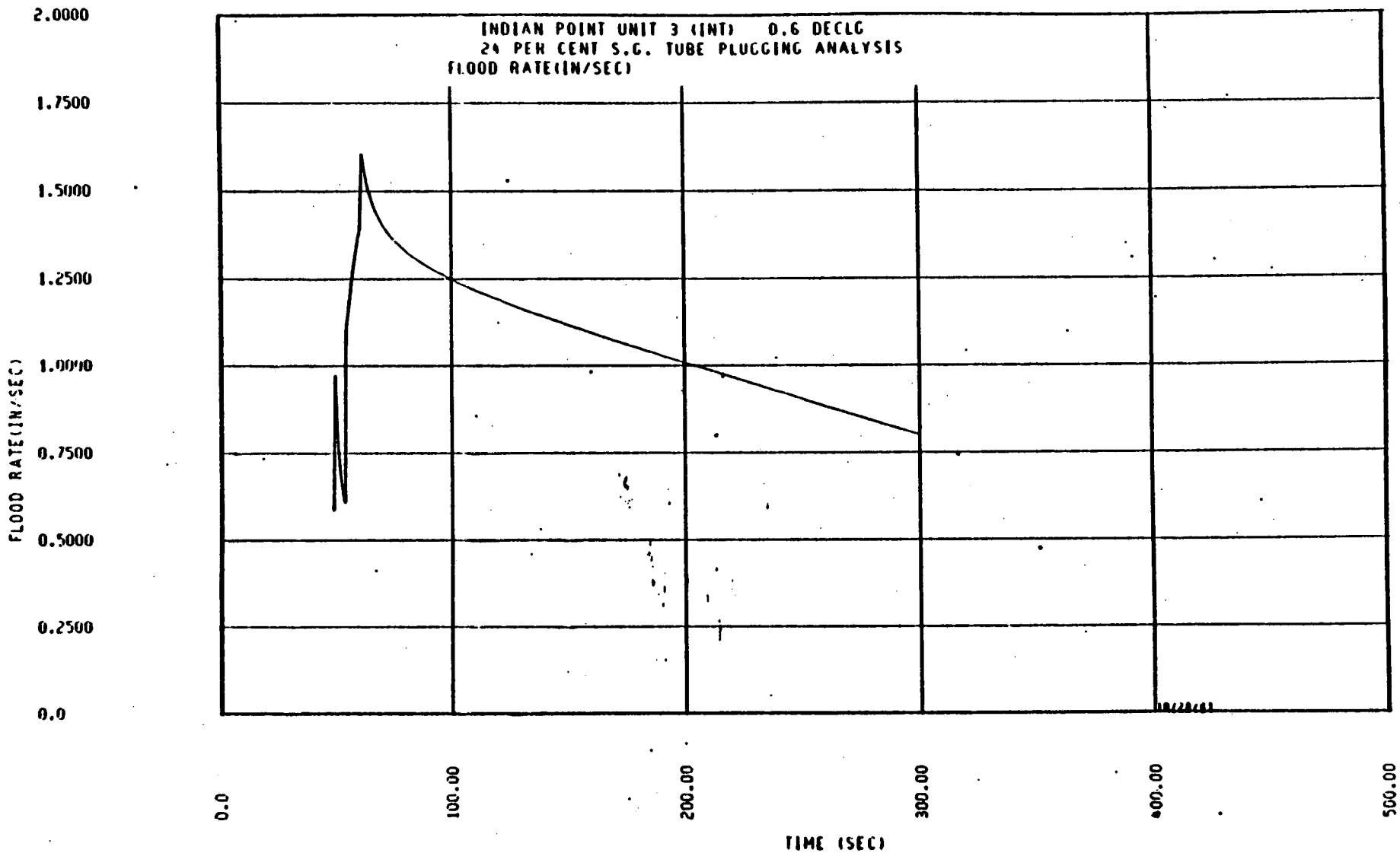

FIGURE 11B REFLOOD TRANSIENT
CORE INLET VELOCITY
DECLG(CD = 0.6)

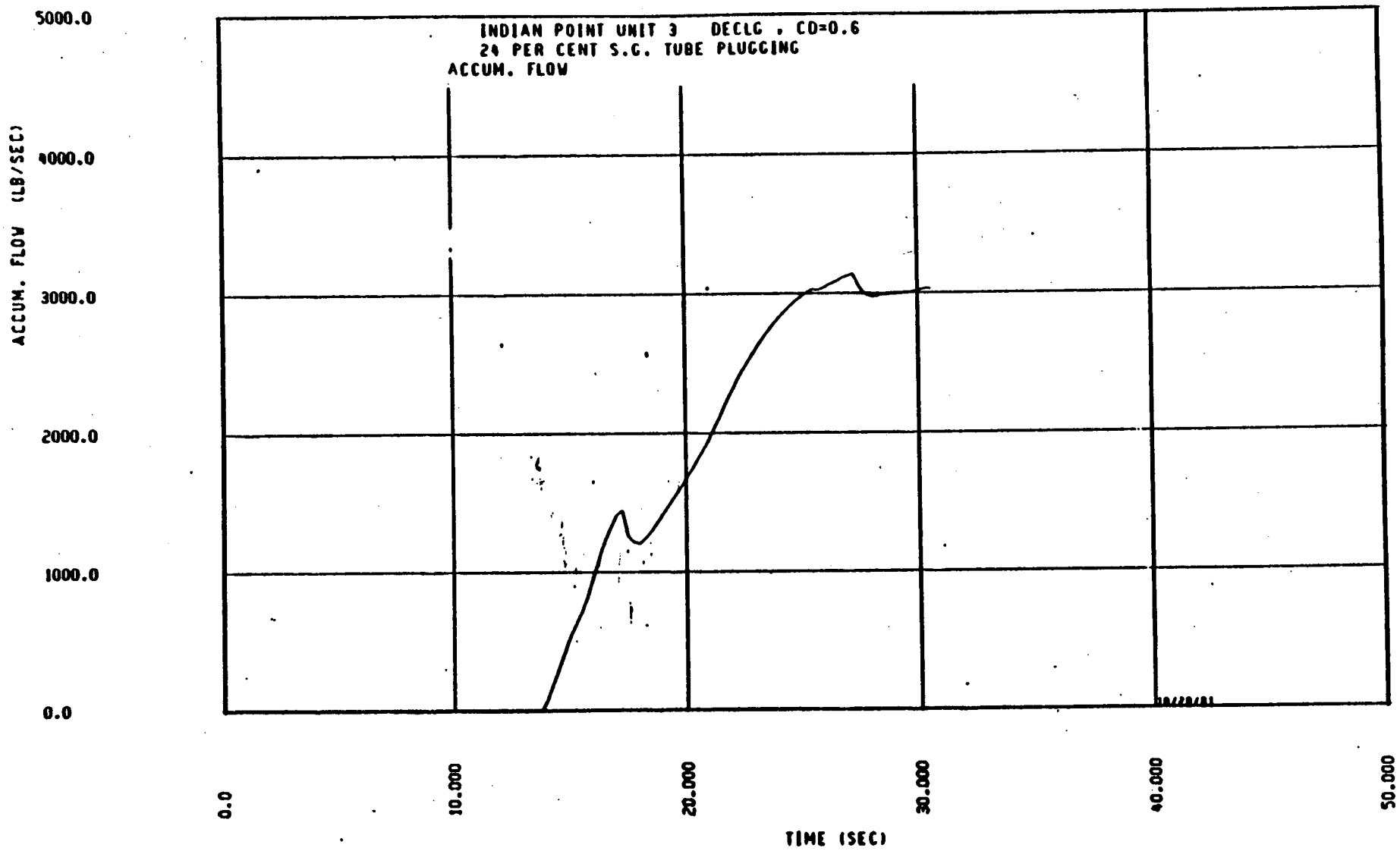

FIGURE 12B ACCUMULATOR FLOW(BLOWDOWN)
DECLG(CD = 0.6)

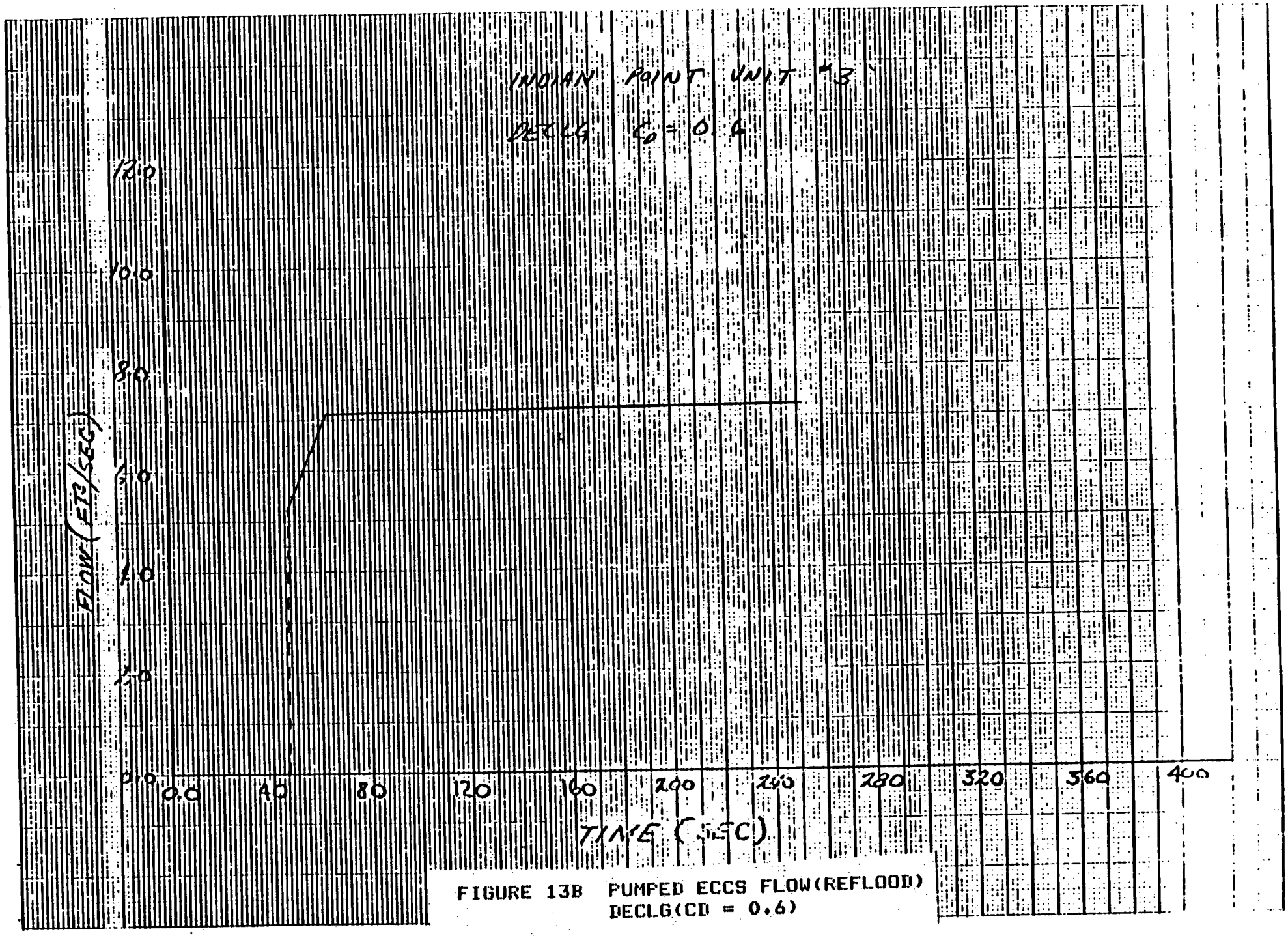

FIGURE 13B PUMPED ECCS FLOW(REFLOOD)
DECLG(CD = 0.6)

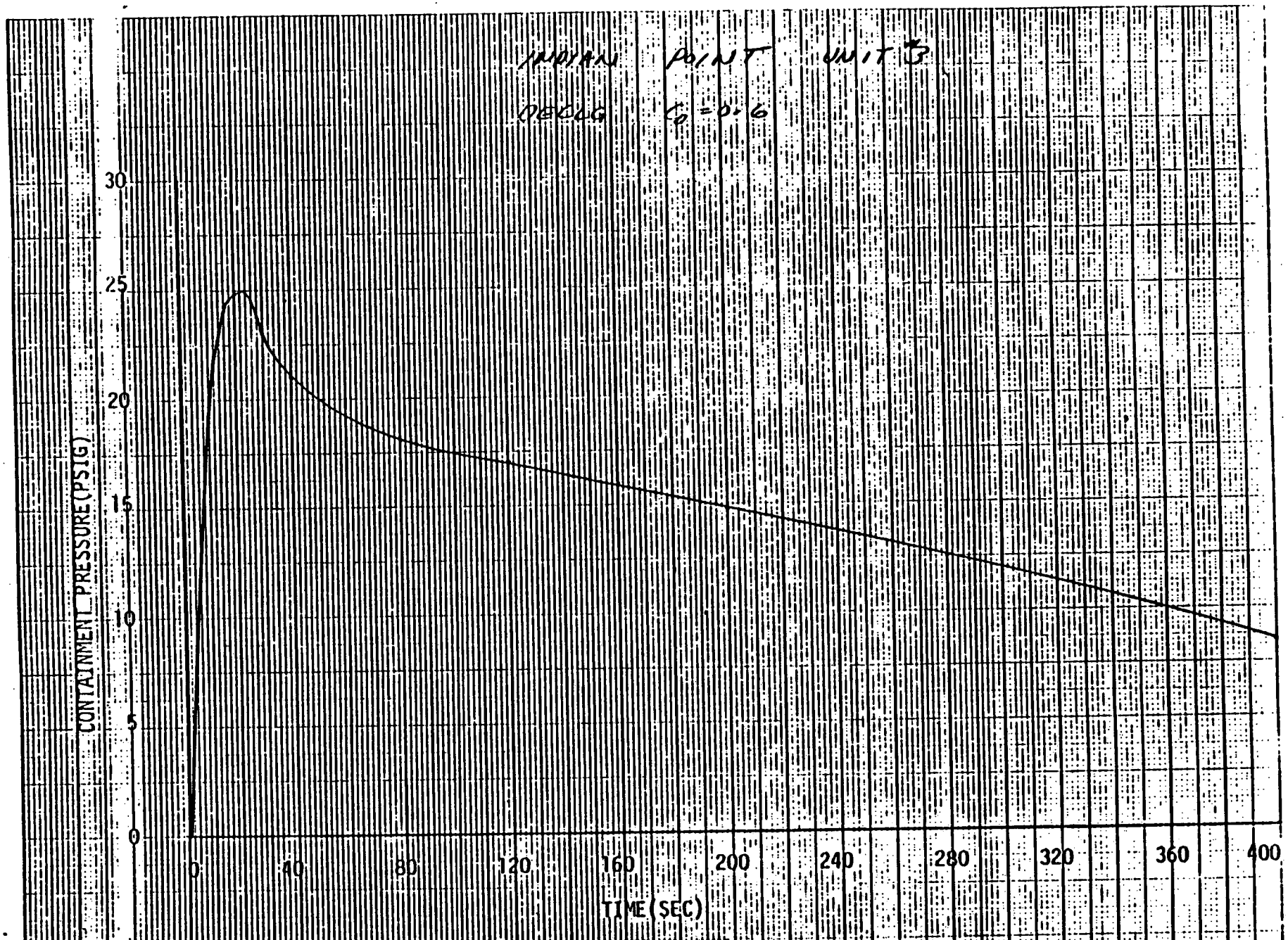

FIGURE 14B CONTAINMENT PRESSURE
DECLG(CD = 0.6)

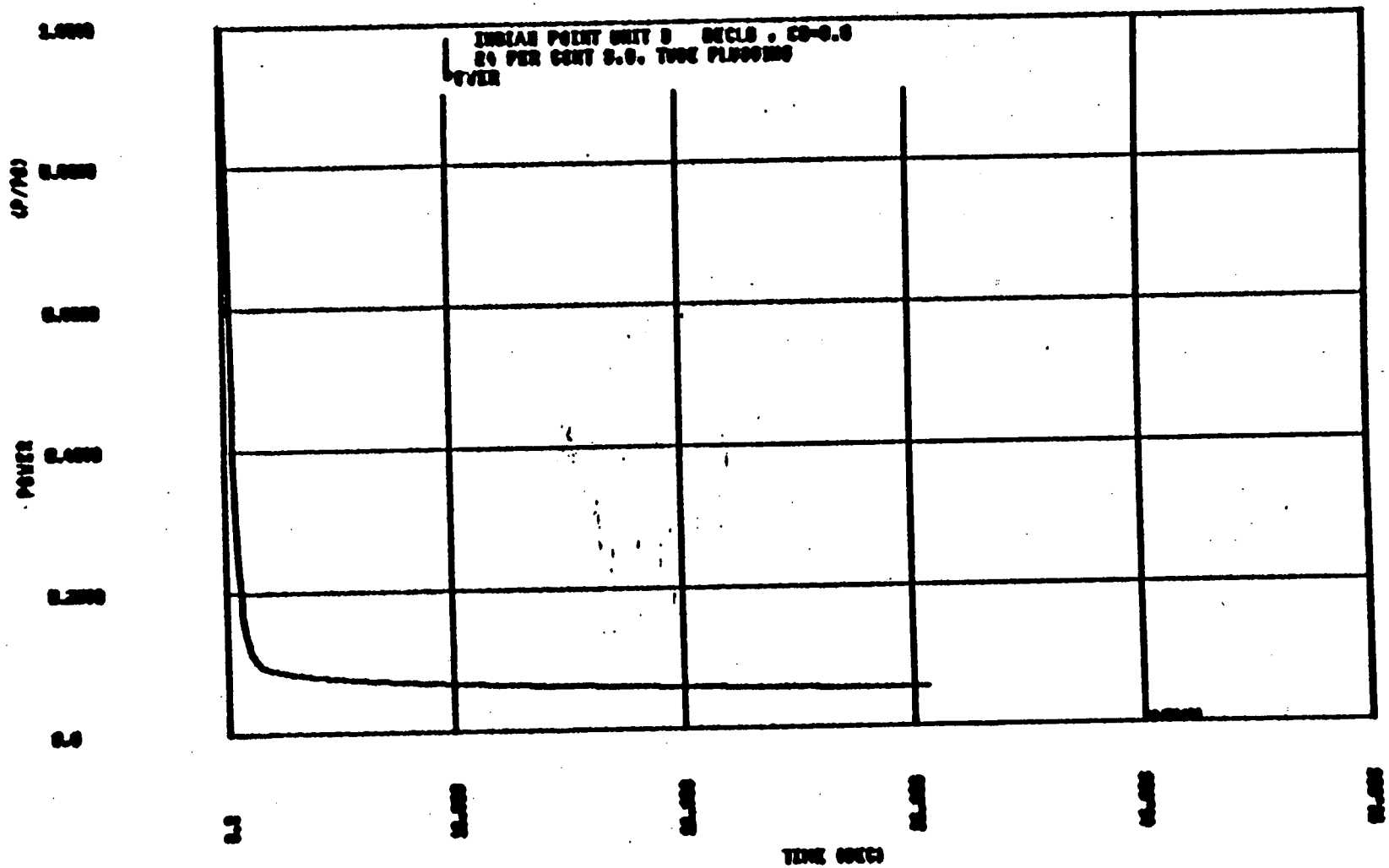

FIGURE 15B CORE POWER TRANSIENT
DECLG(CD = 0.6)

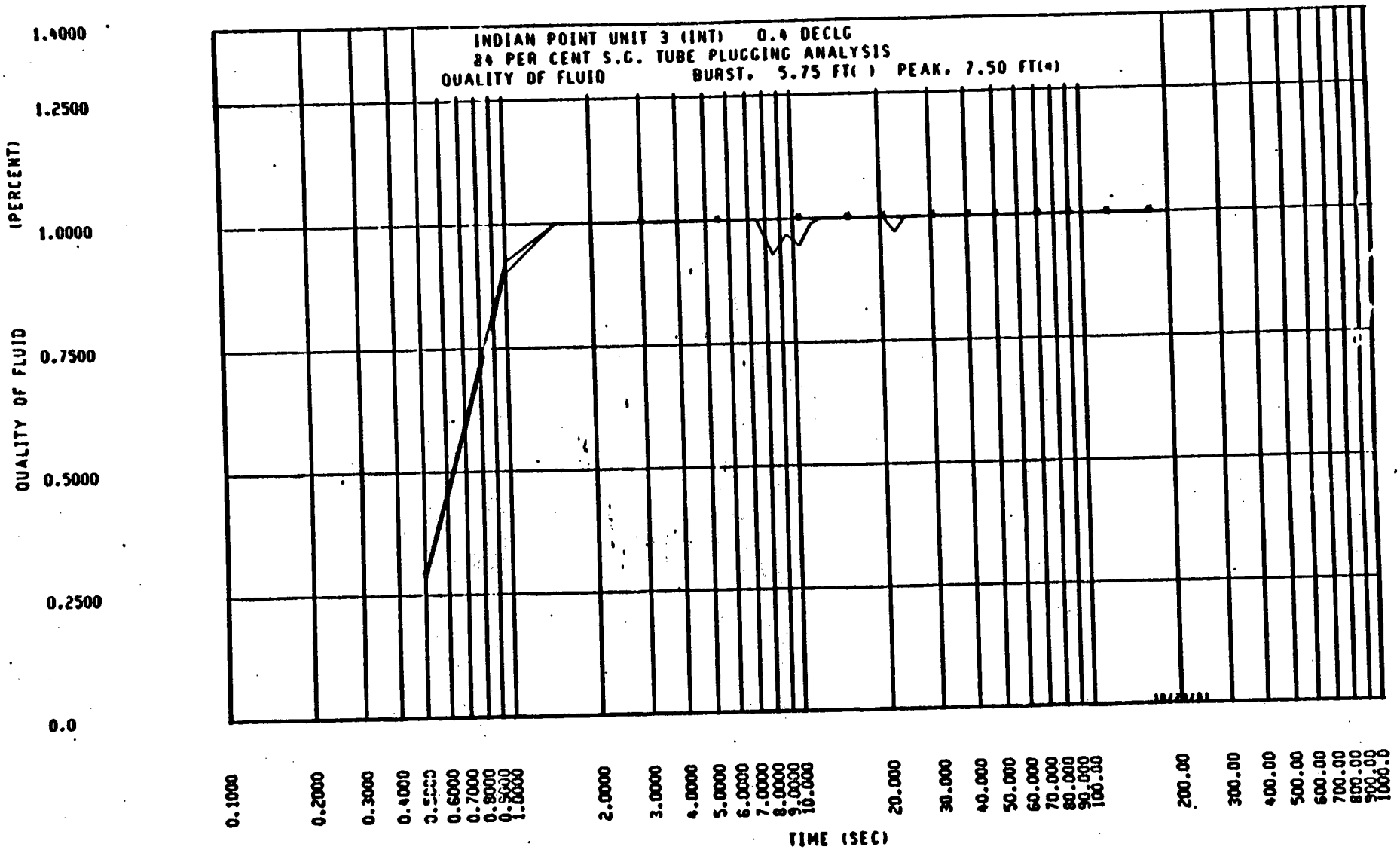

FIGURE 1C FLUID QUALITY
DECLG(CD = 0.4)

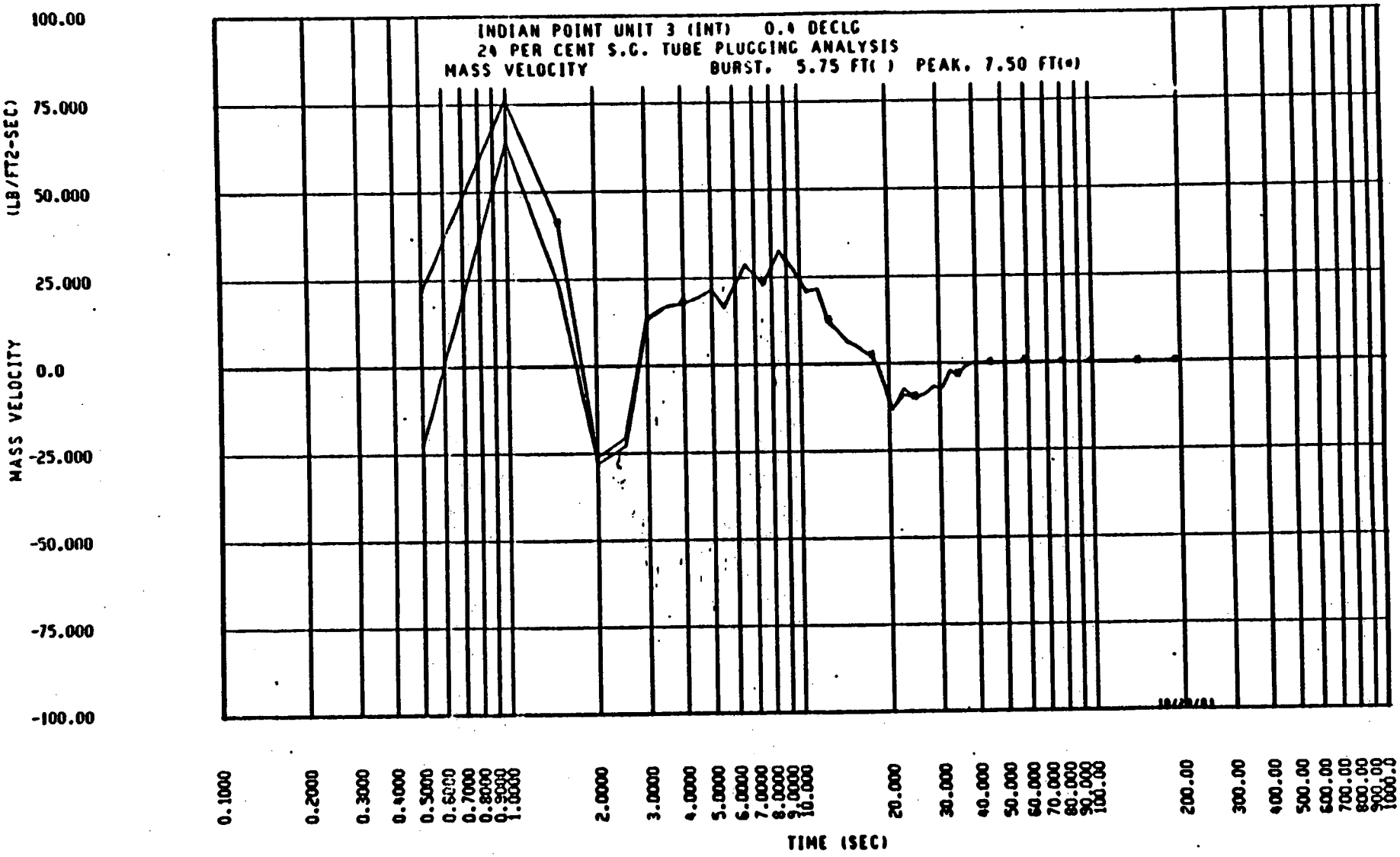

FIGURE 2C MASS VELOCITY
DECLG(CD = 0.4)

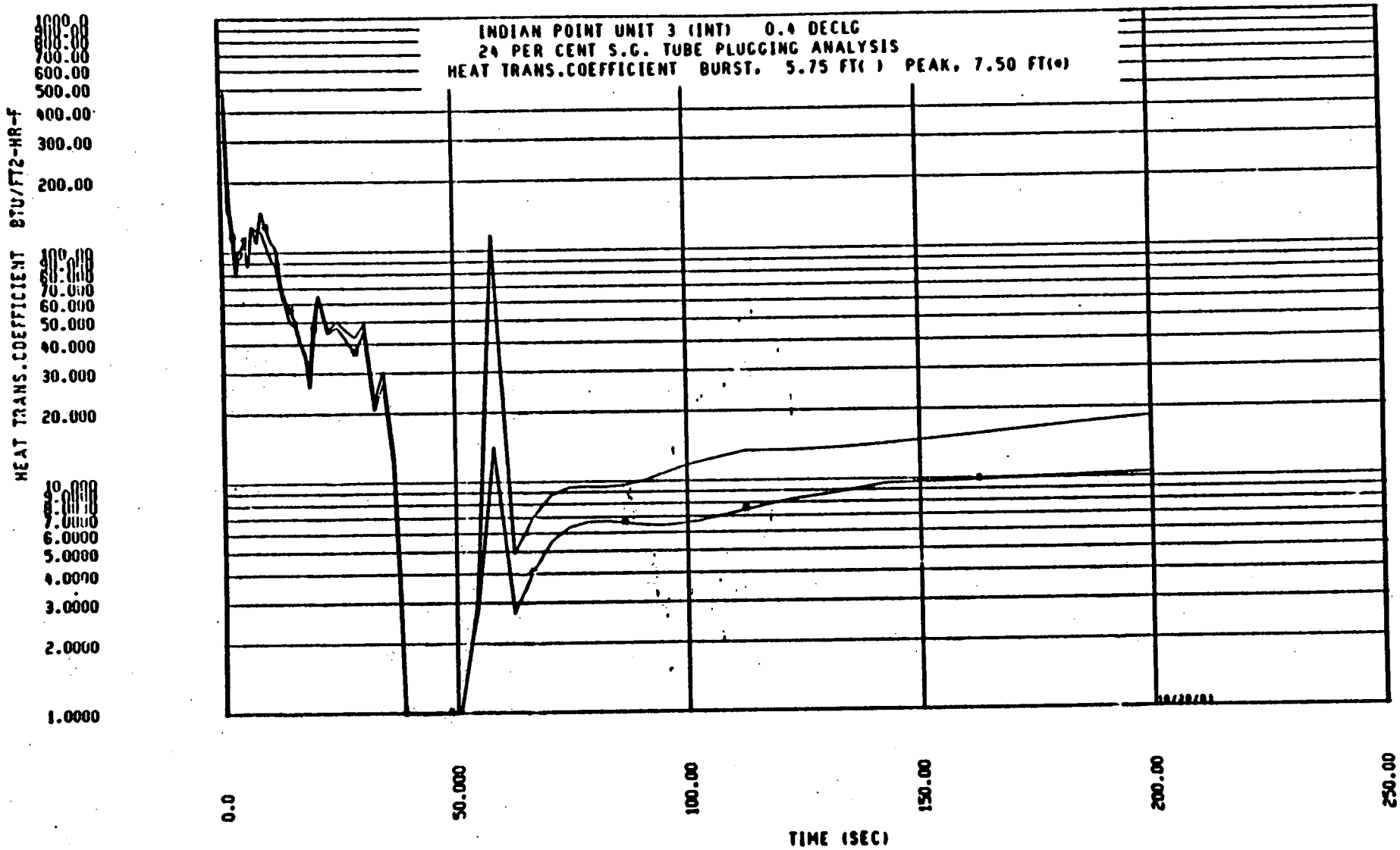

FIGURE 3C HEAT TRANSFER COEFFICIENT
DECLG(CD = 0.4)

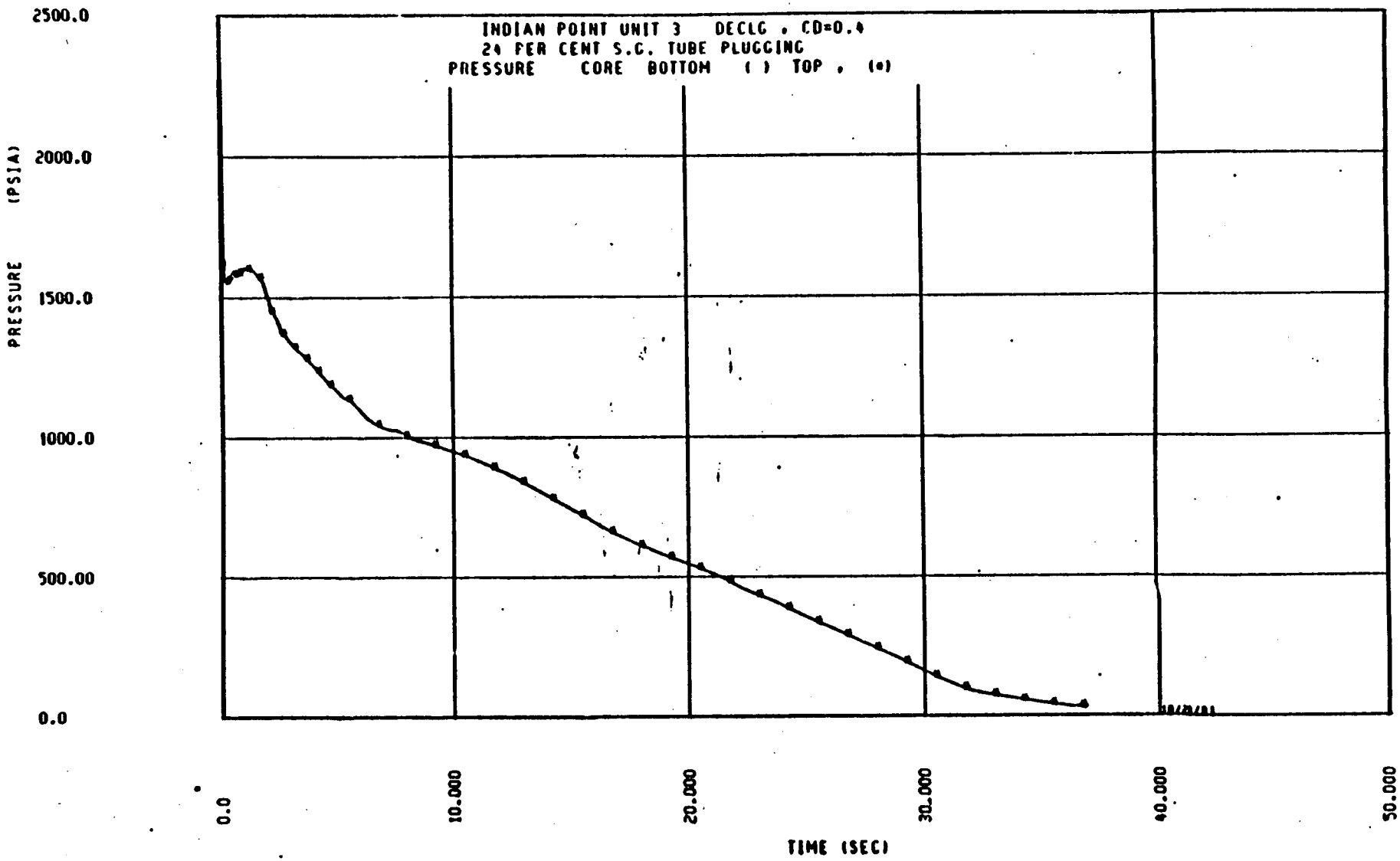

FIGURE 4C CORE PRESSURE
DECLG(CD = 0.4)

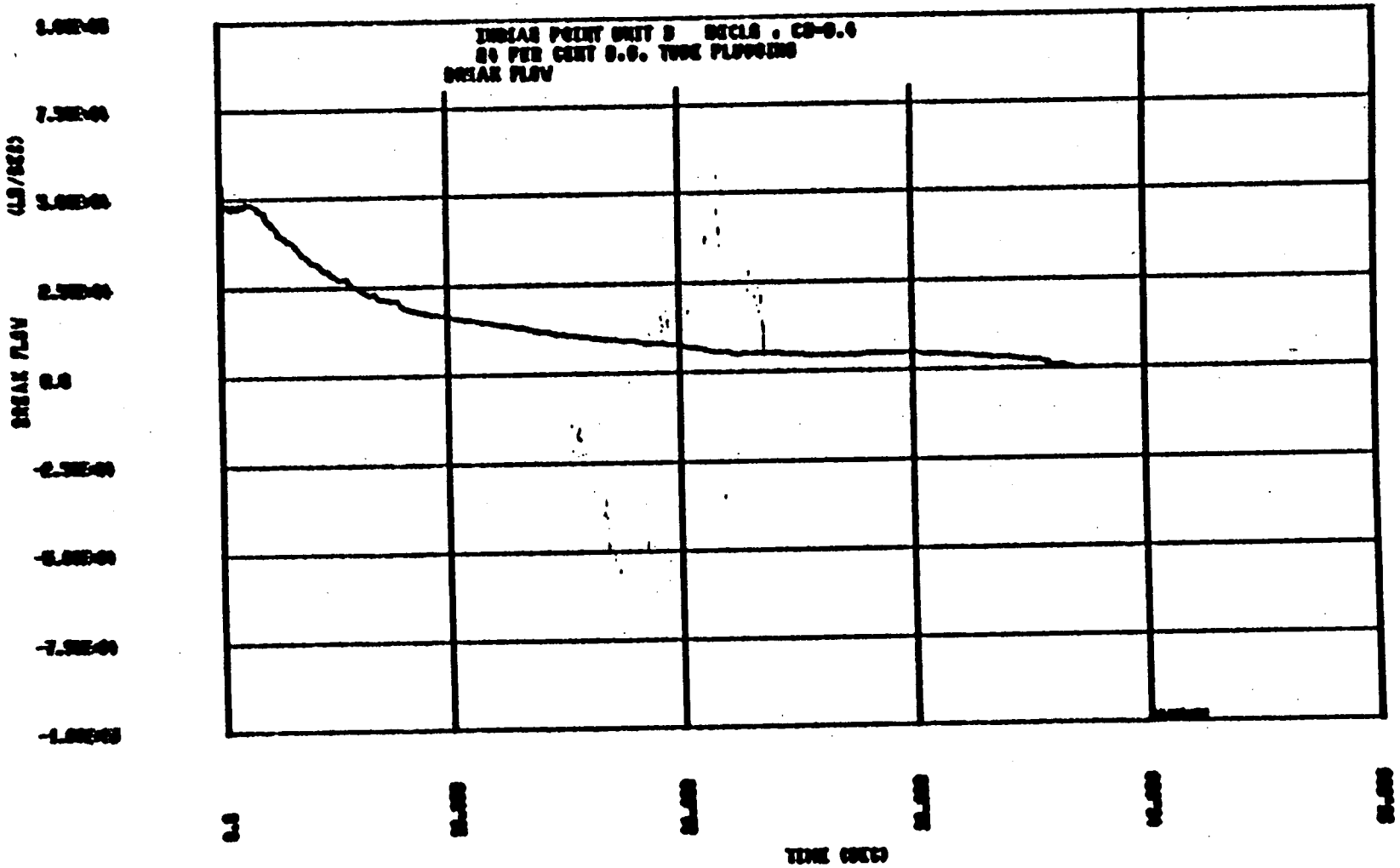

FIGURE 5C BREAK FLOW RATE
DECLG(CD = 0.4)

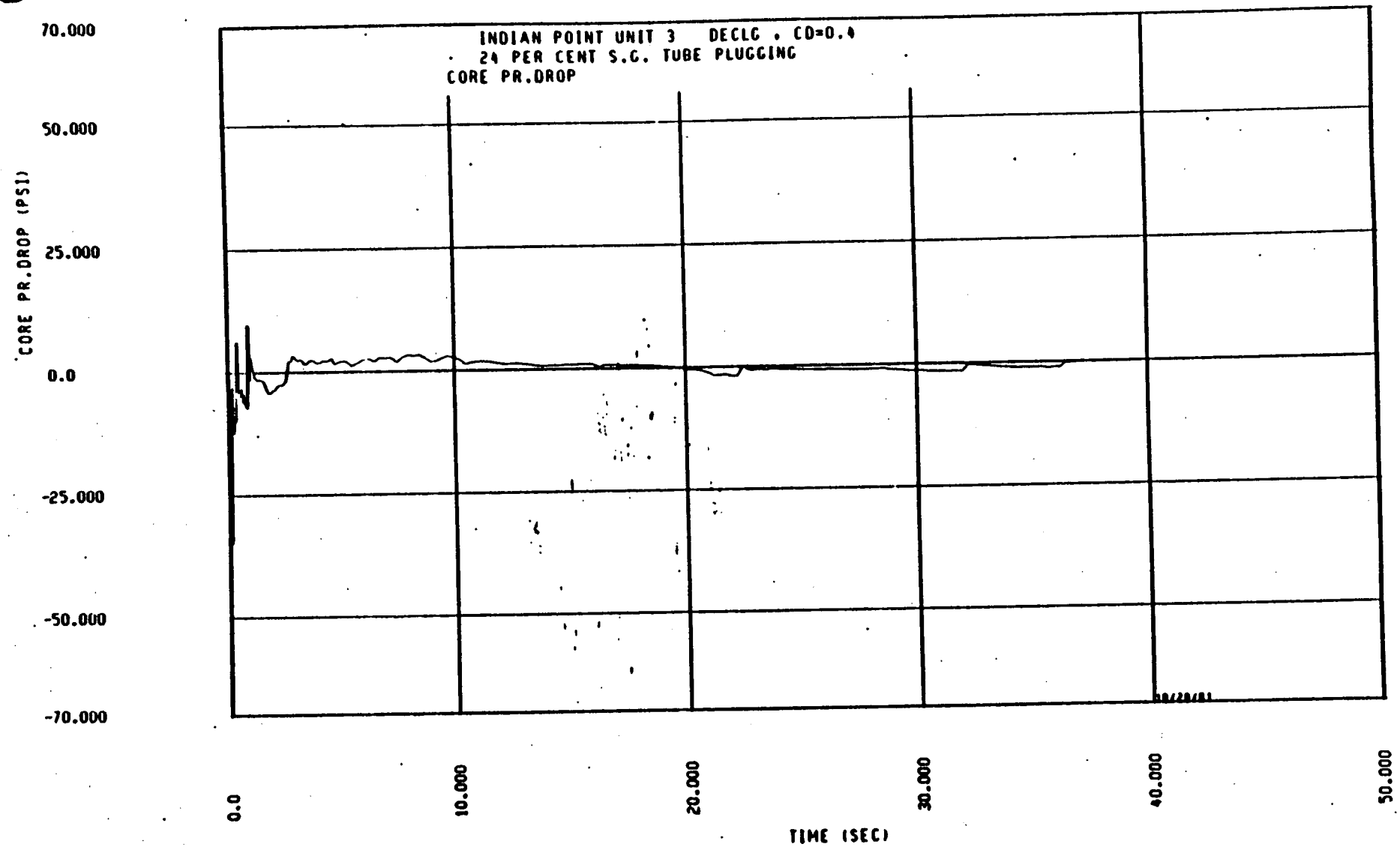

FIGURE 6C CORE PRESSURE DROP
DECLG(CD = 0.4)

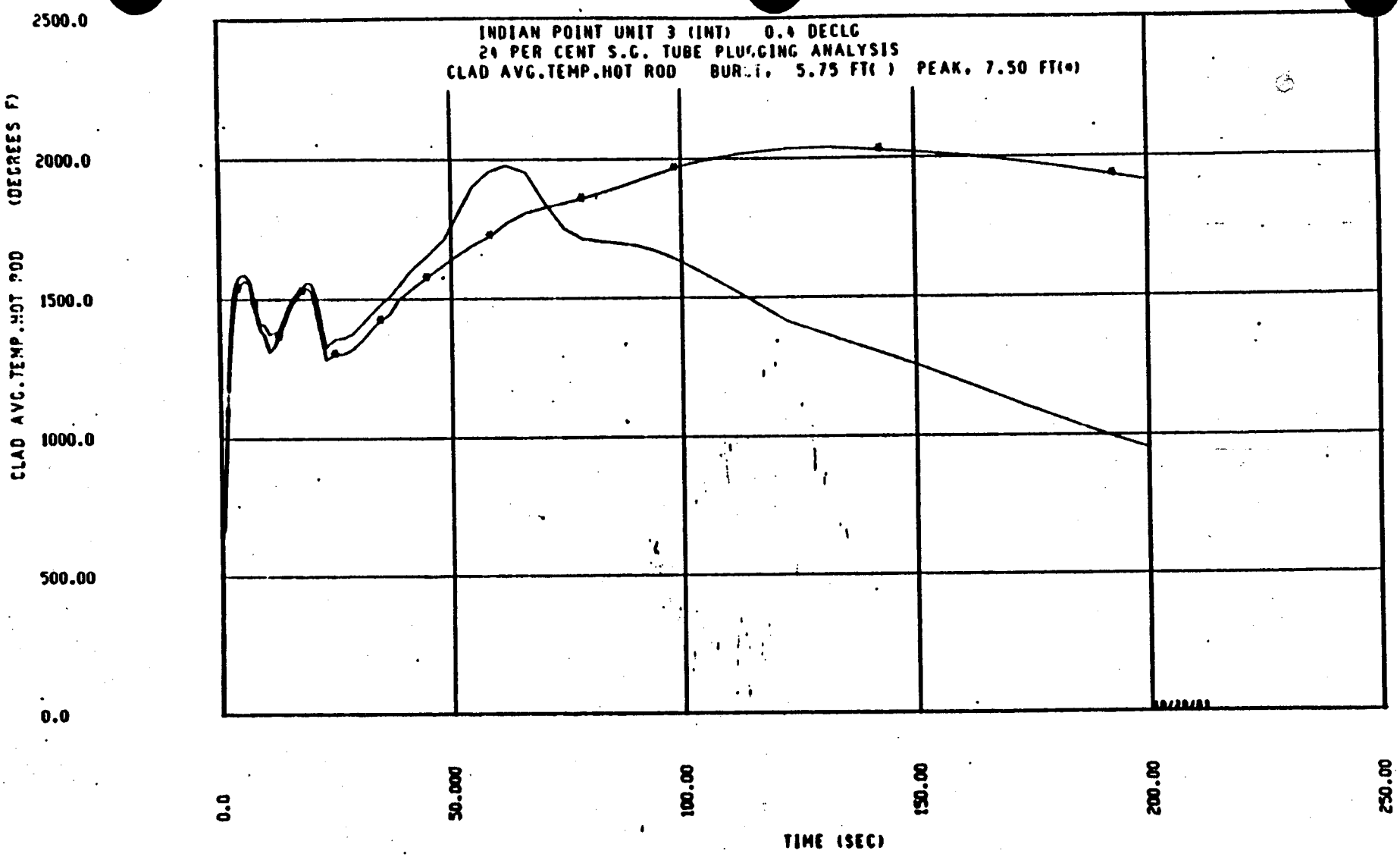

FIGURE 7C PEAK CLAD TEMPERATURE
DECLG(CD = 0.4)

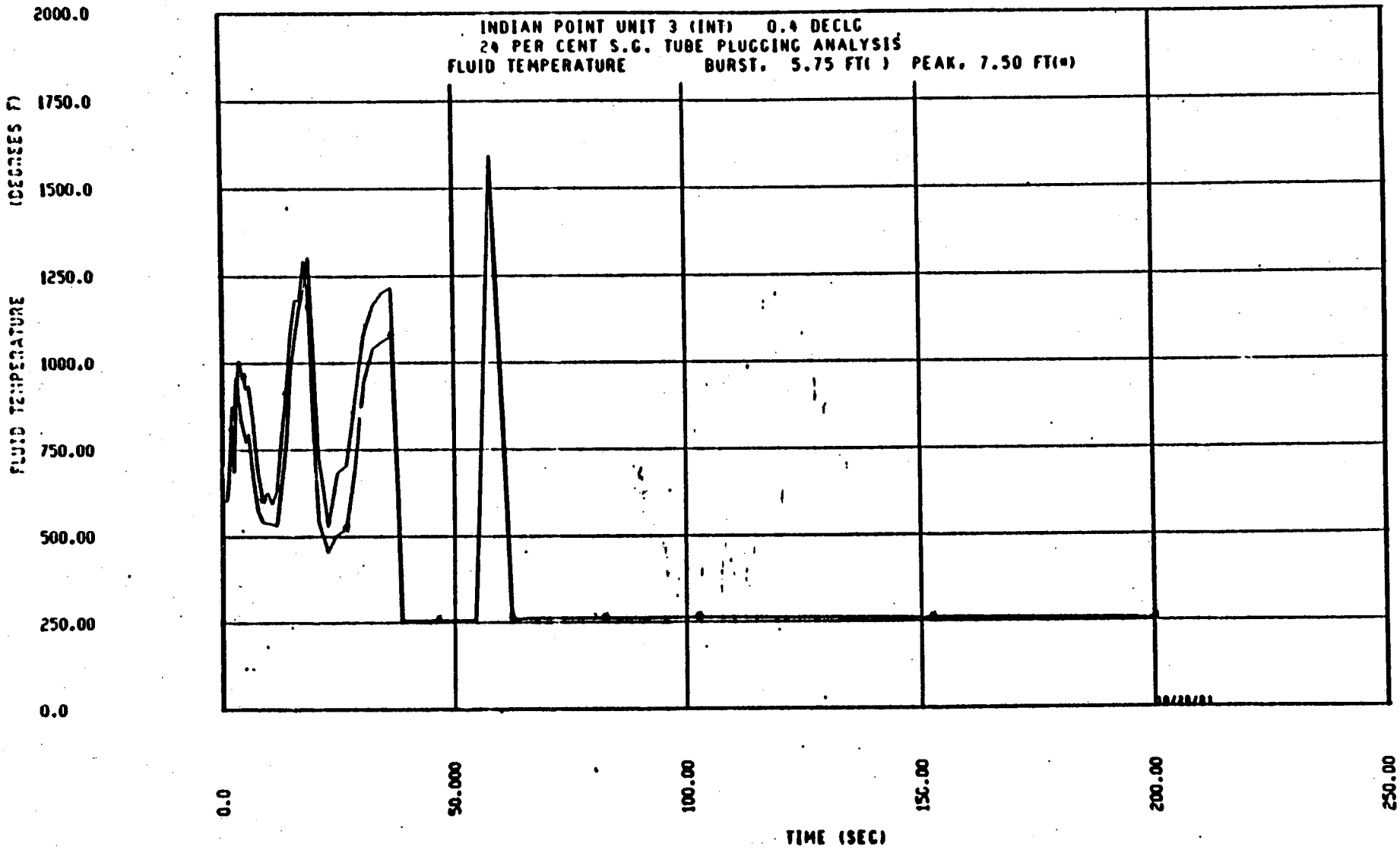

FIGURE 8C FLUID TEMPERATURE
DECLG(CD = 0.4)

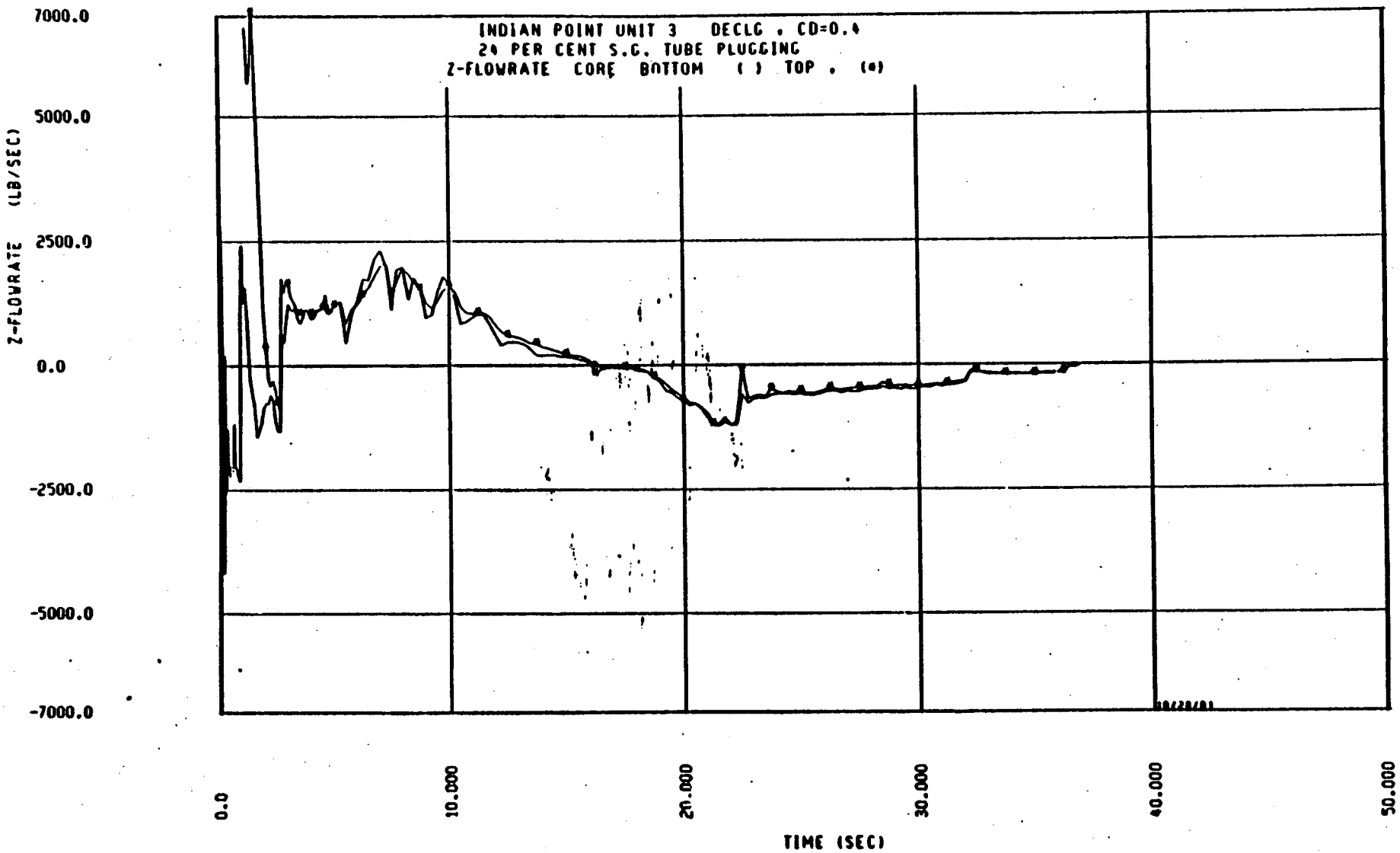

FIGURE 9C CORE FLOW (TOP AND BOTTOM)
DECLG(CD = 0.4)

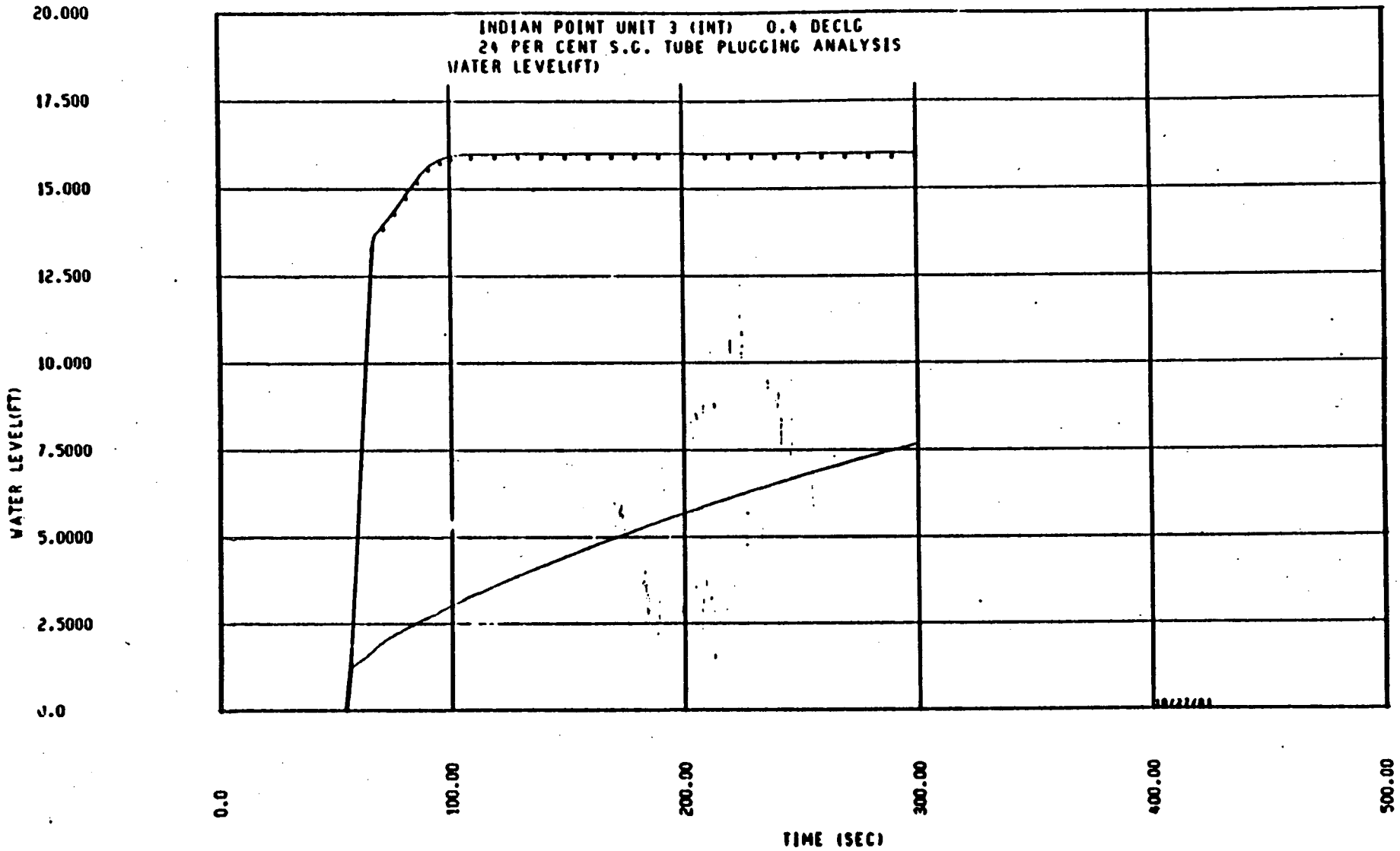

FIGURE 10C REFLOOD TRANSIENT - CORE & DOWNCOMER WATER LEVELS DECLG(CD = 0.4)

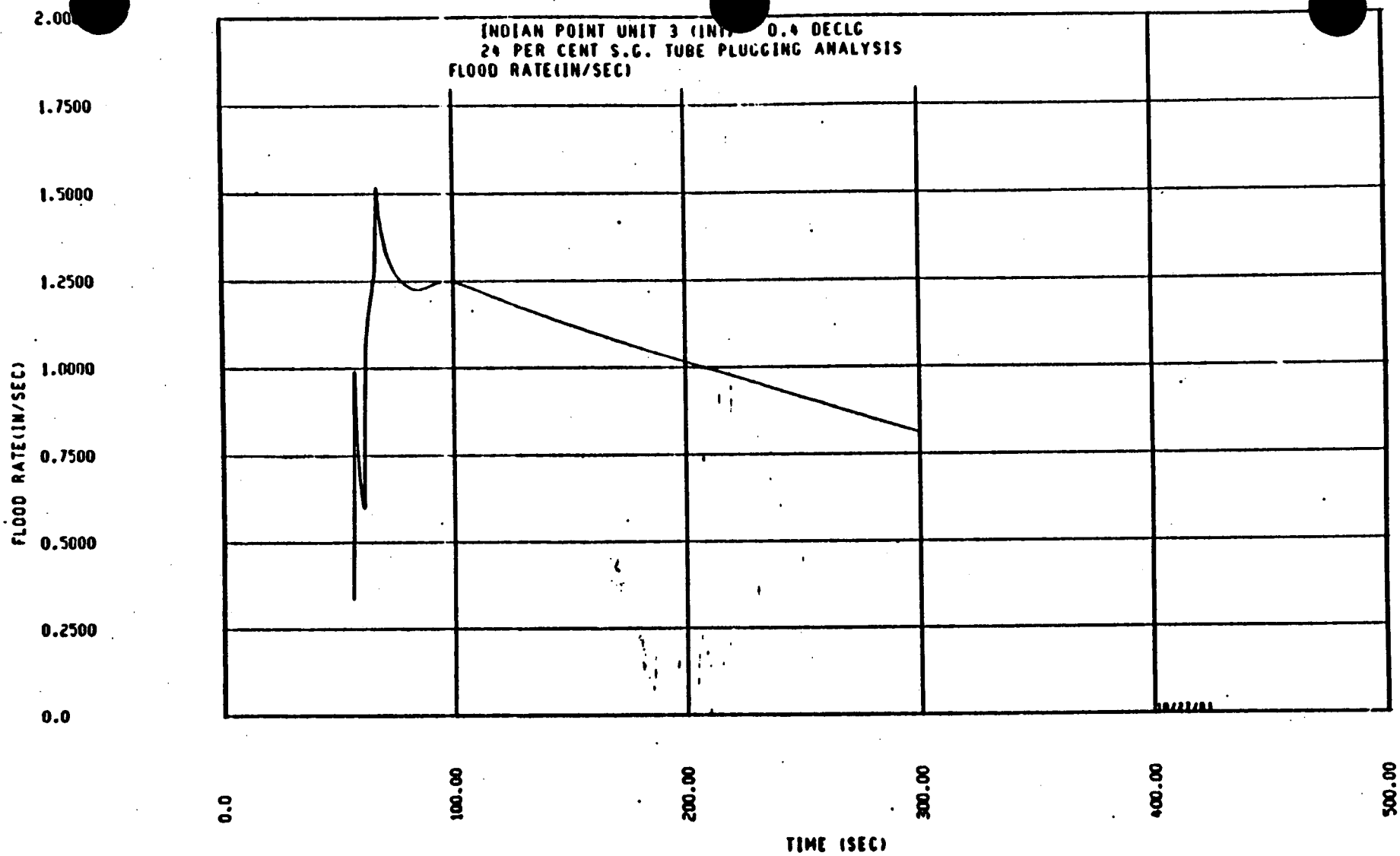

FIGURE 11C REFLOOD TRANSIENT
CORE INLET VELOCITY
DECLG(CD = 0.4)

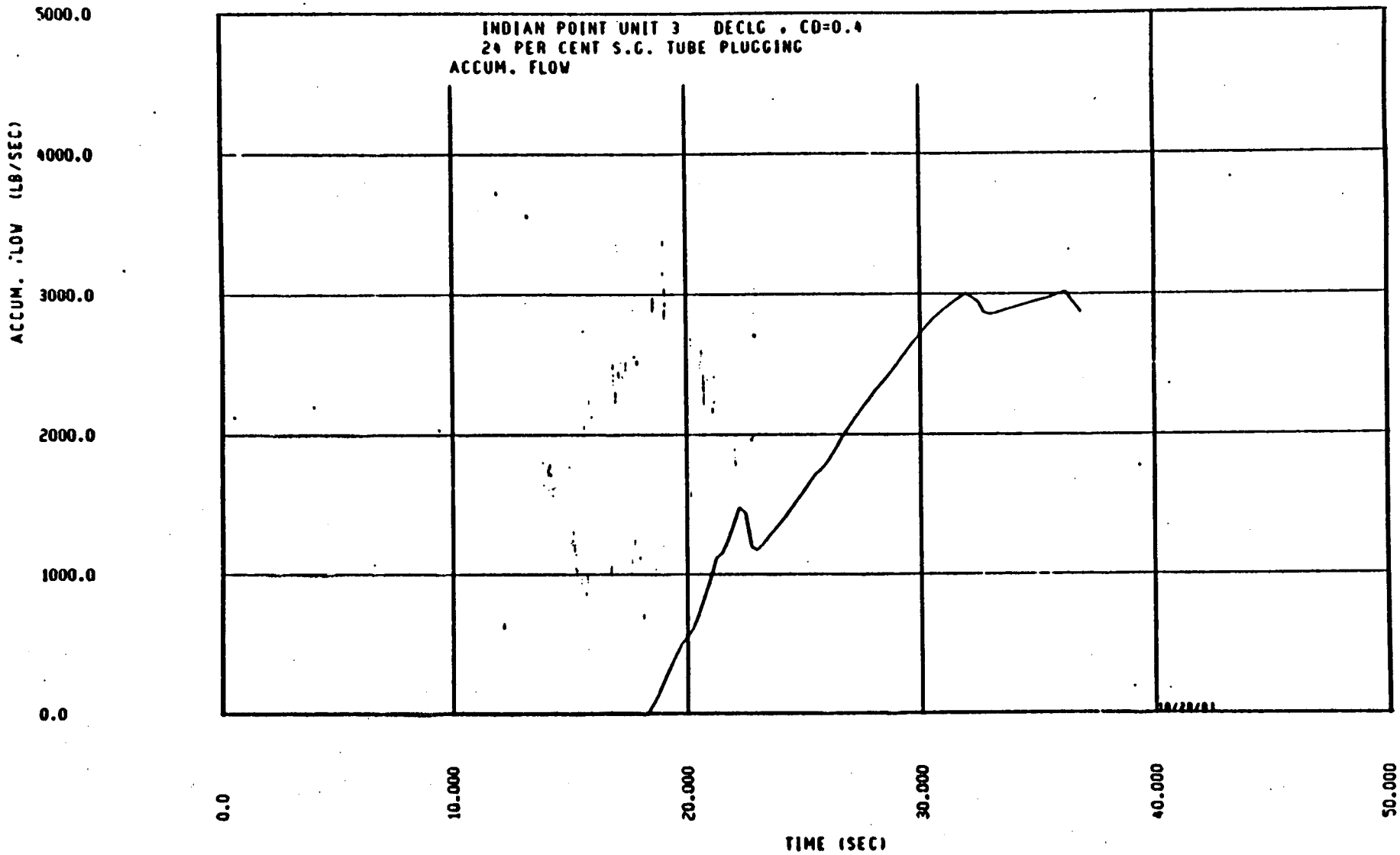

FIGURE 12C ACCUMULATOR FLOW(BLOWDOWN)
DECLG(CD = 0.4)

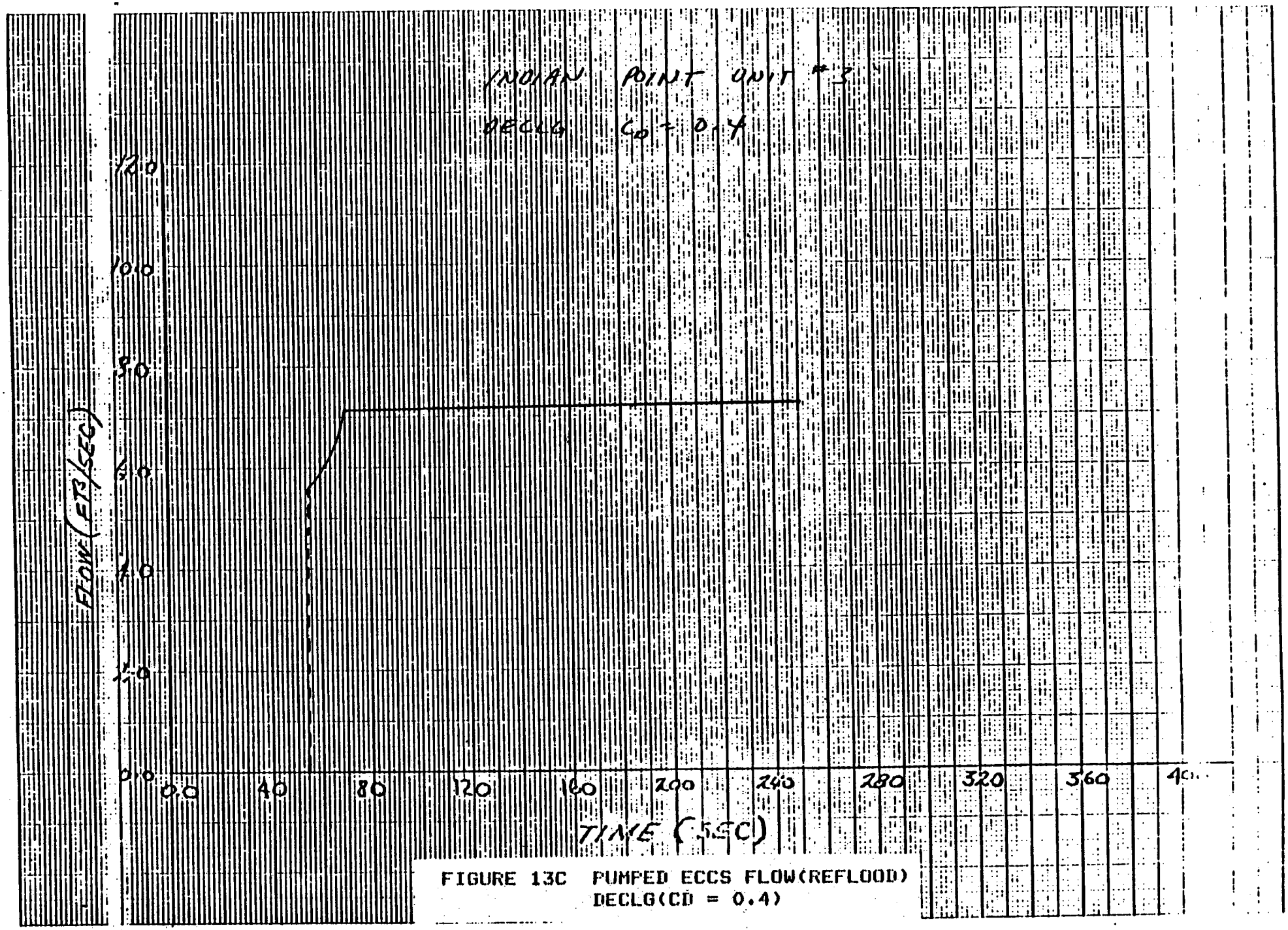

FIGURE 13C PUMPED ECCS FLOW(REFLOOD)
DECLG(CD = 0.4)

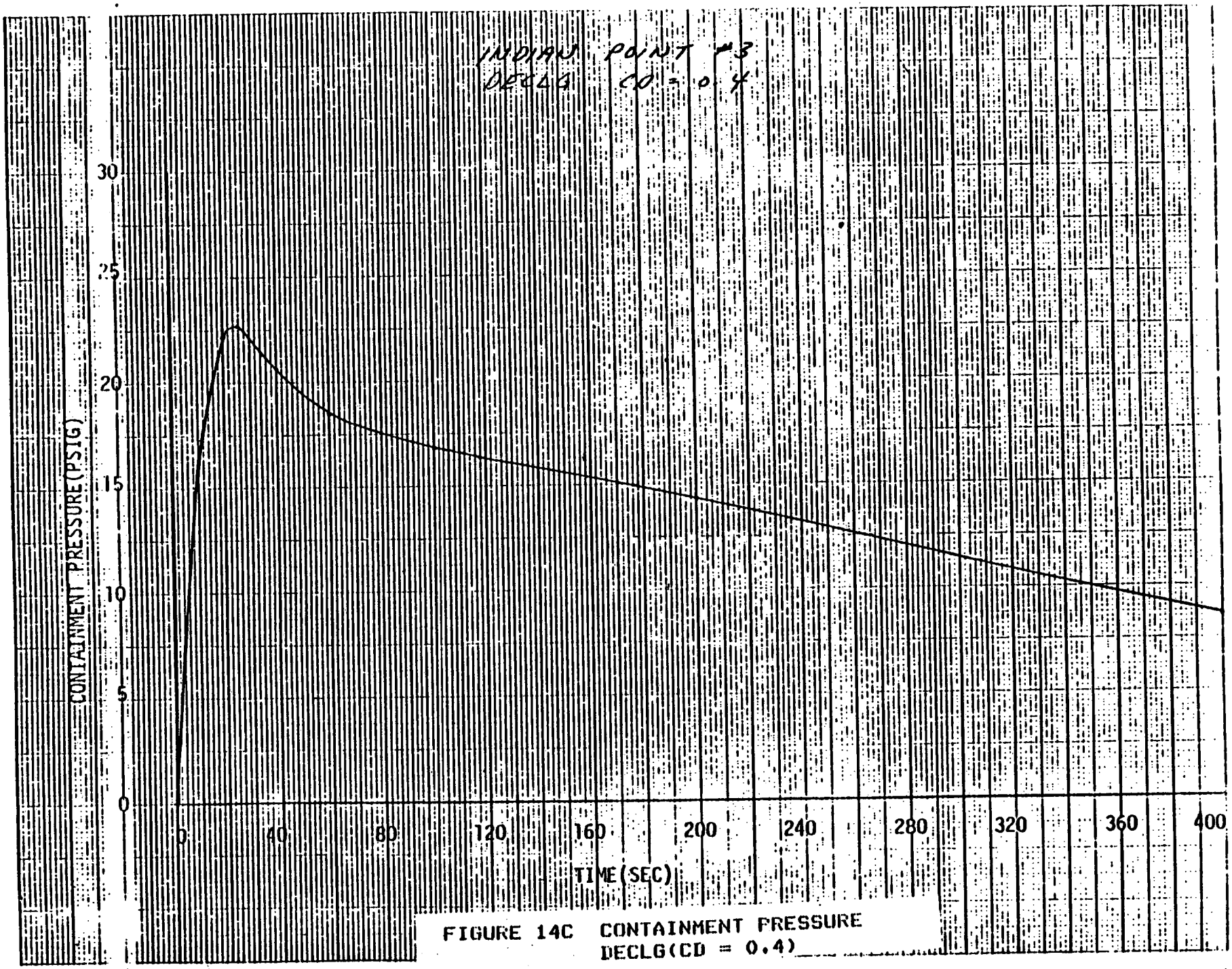

FIGURE 14C CONTAINMENT PRESSURE DECLG(CD = 0.4)

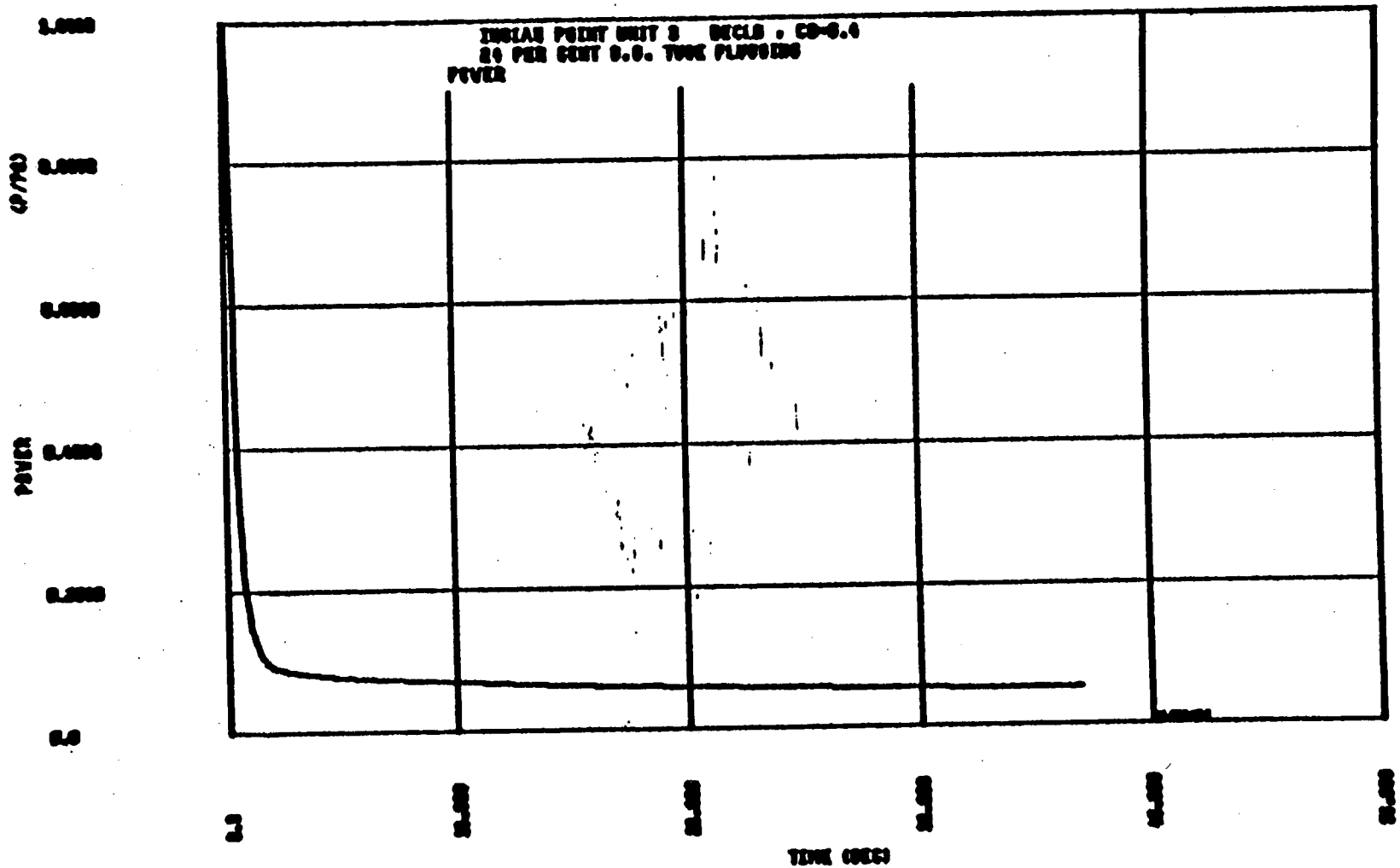

FIGURE 15C CORE POWER TRANSIENT
DECLG(CD = 0.4)

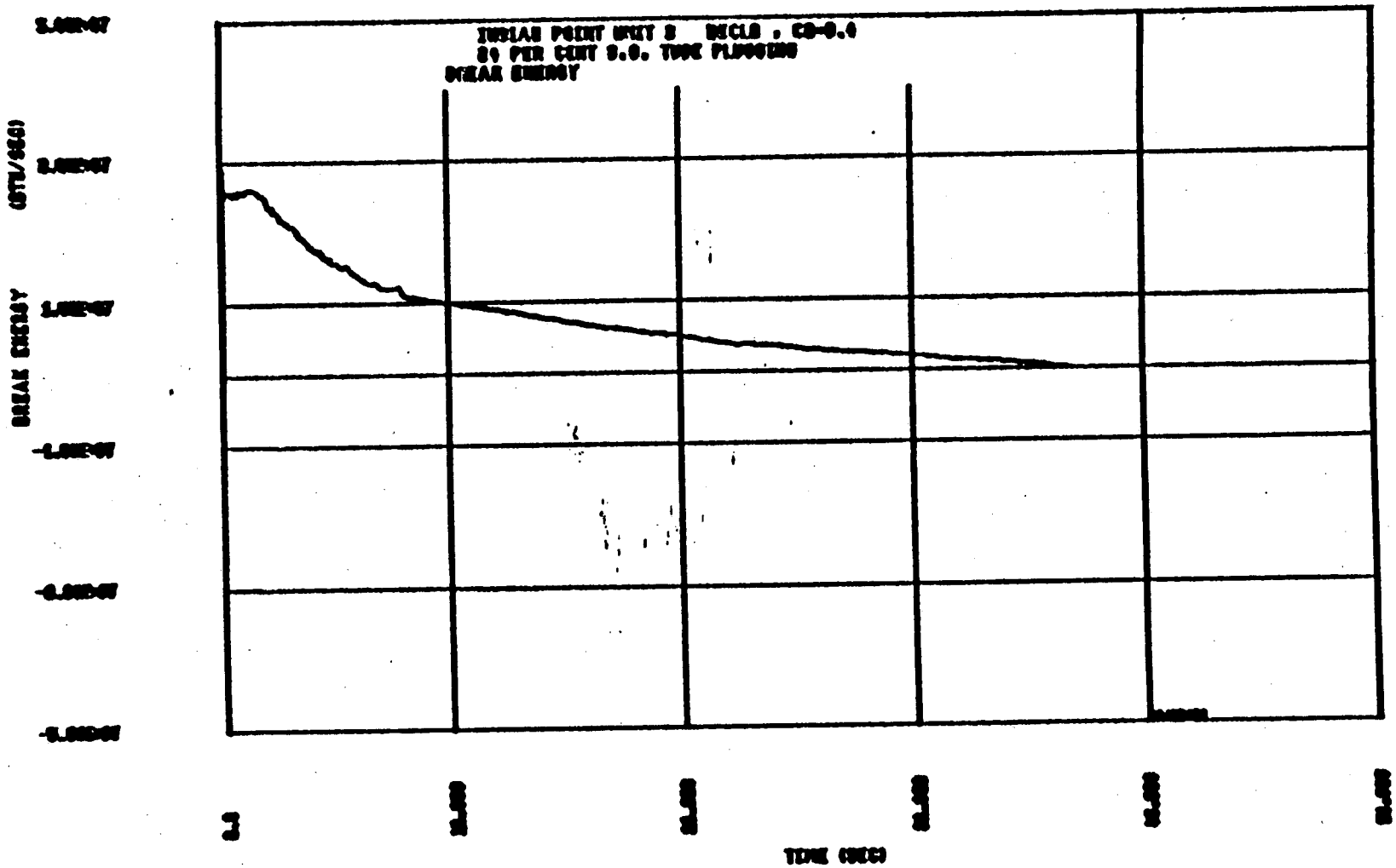

FIGURE 16 BREAK ENERGY RELEASED TO CONTAINMENT

FIGURE 17 CONTAINMENT WALL HEAT TRANSFER COEFFICIENT

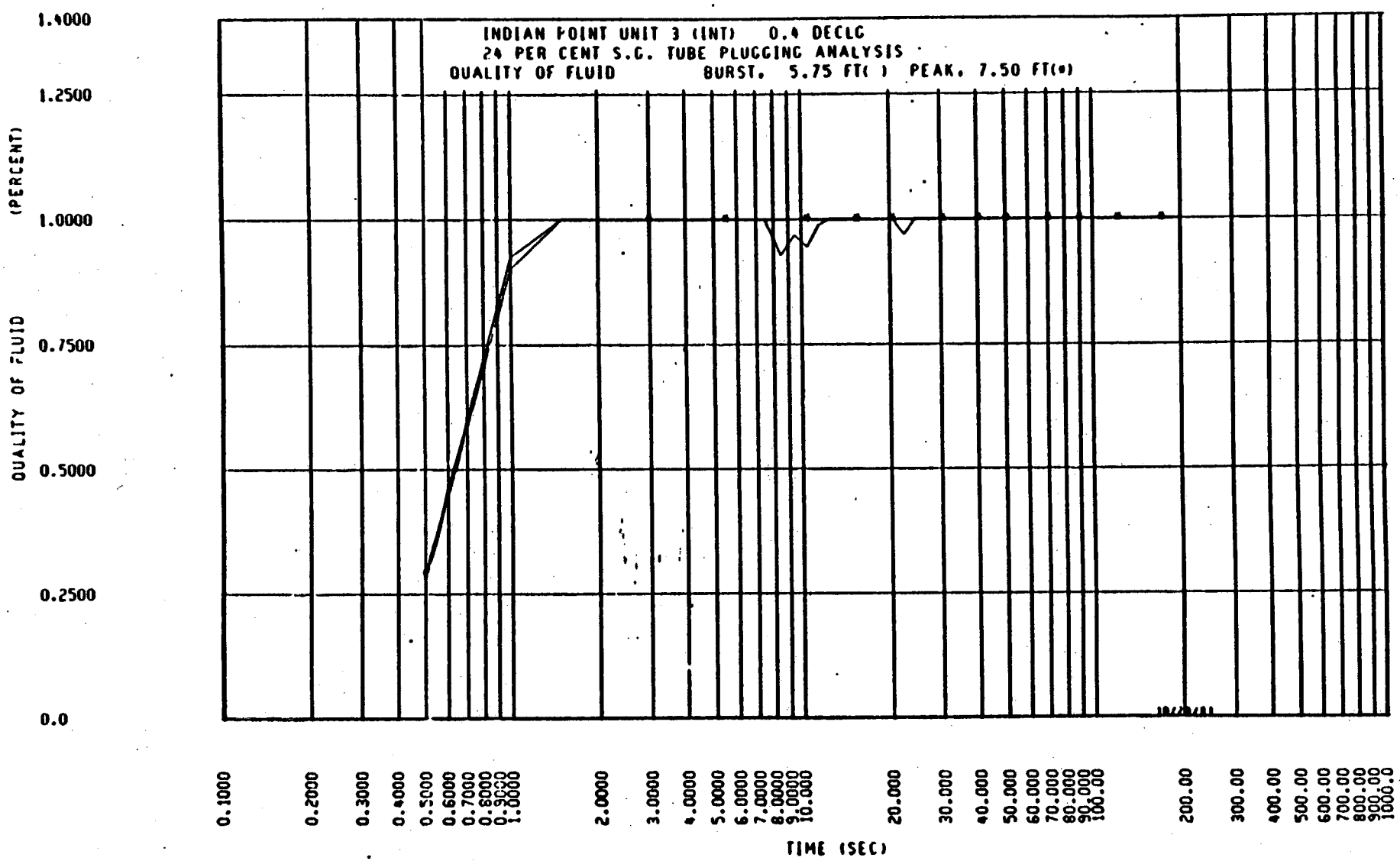

FIGURE 1D FLUID QUALITY
DECLG(CD = 0.4)

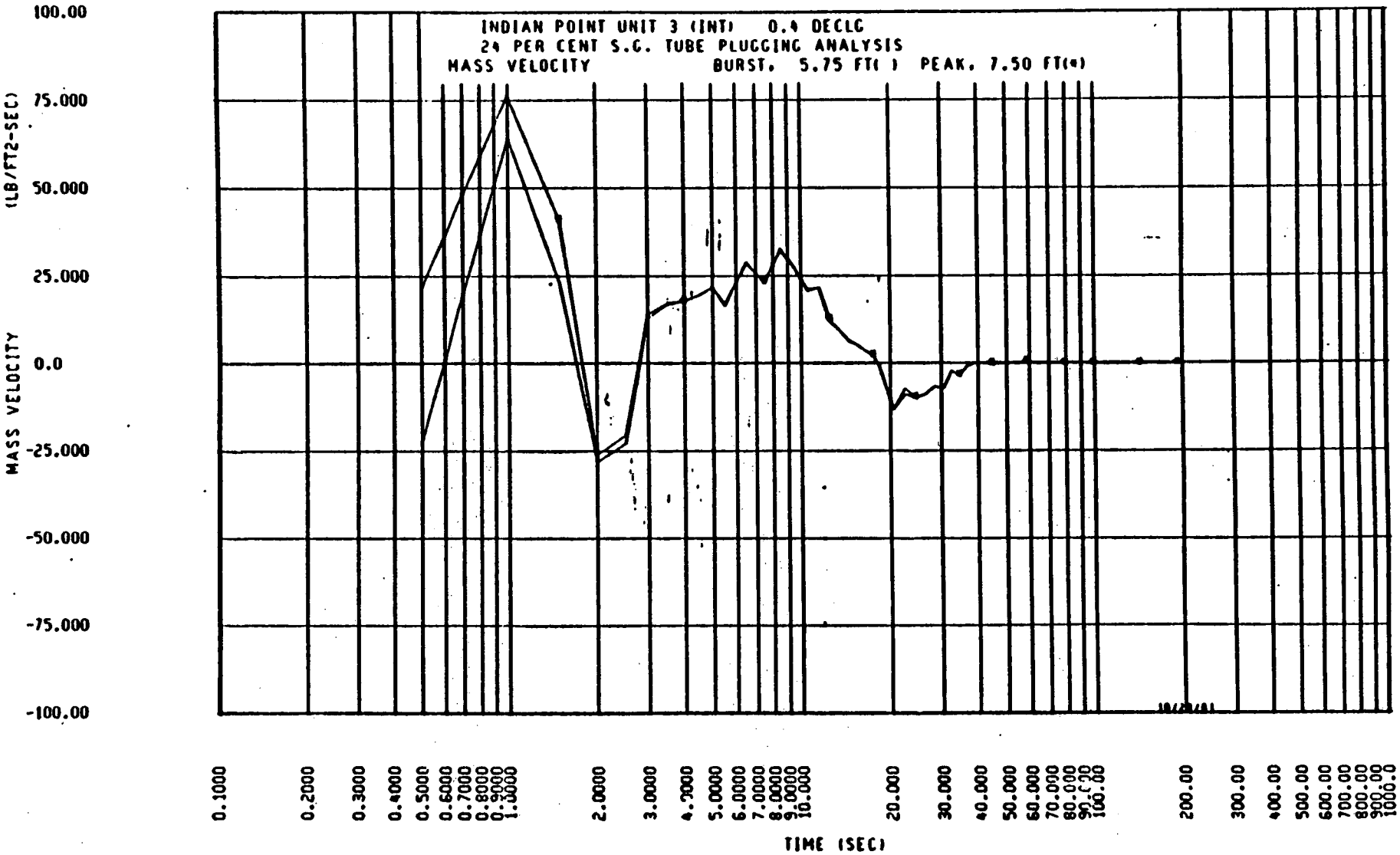

FIGURE 2D MASS VELOCITY
DECLG(CD = 0.4)

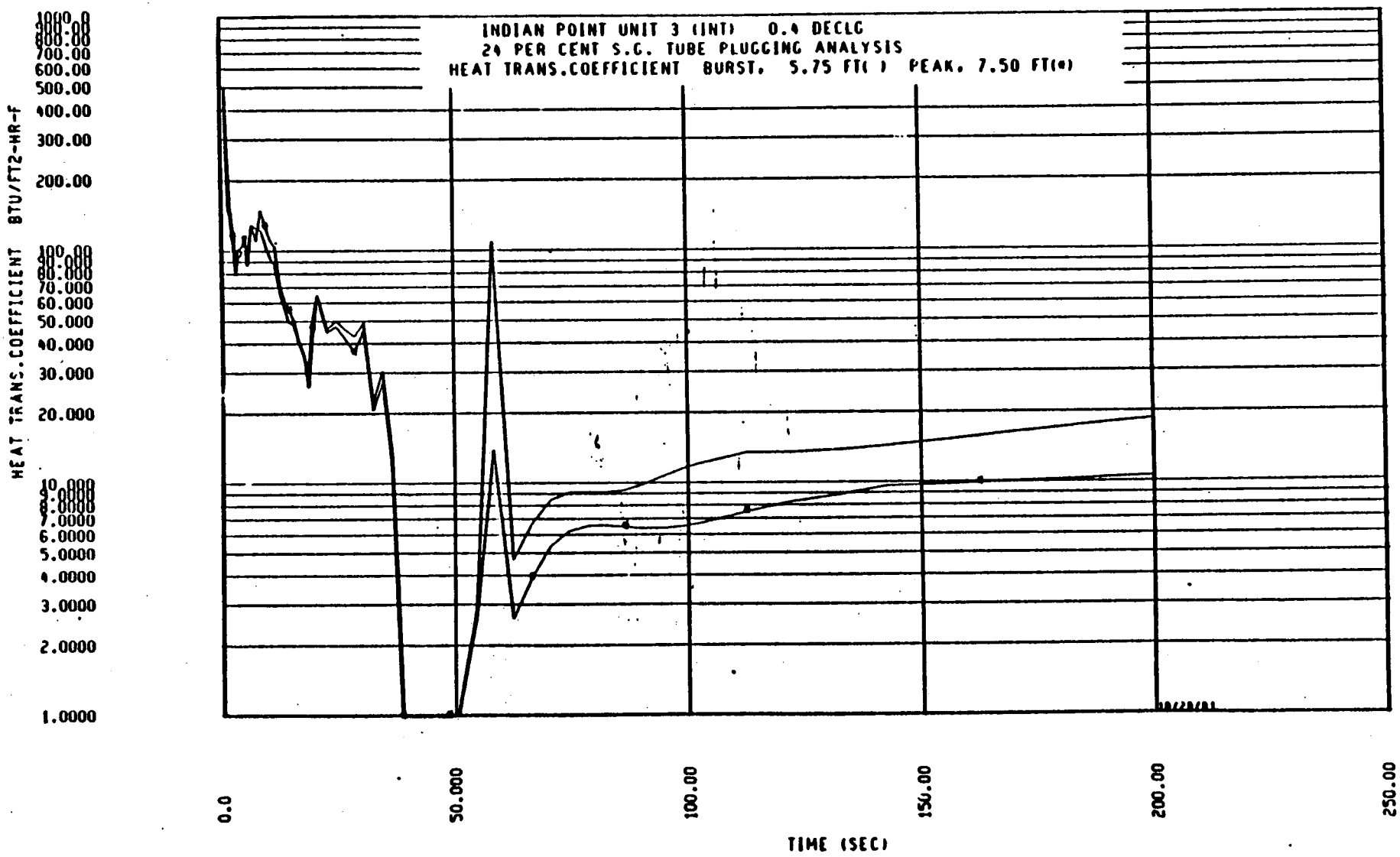

FIGURE 3D HEAT TRANSFER COEFFICIENT
DECLG(CD = 0.4)

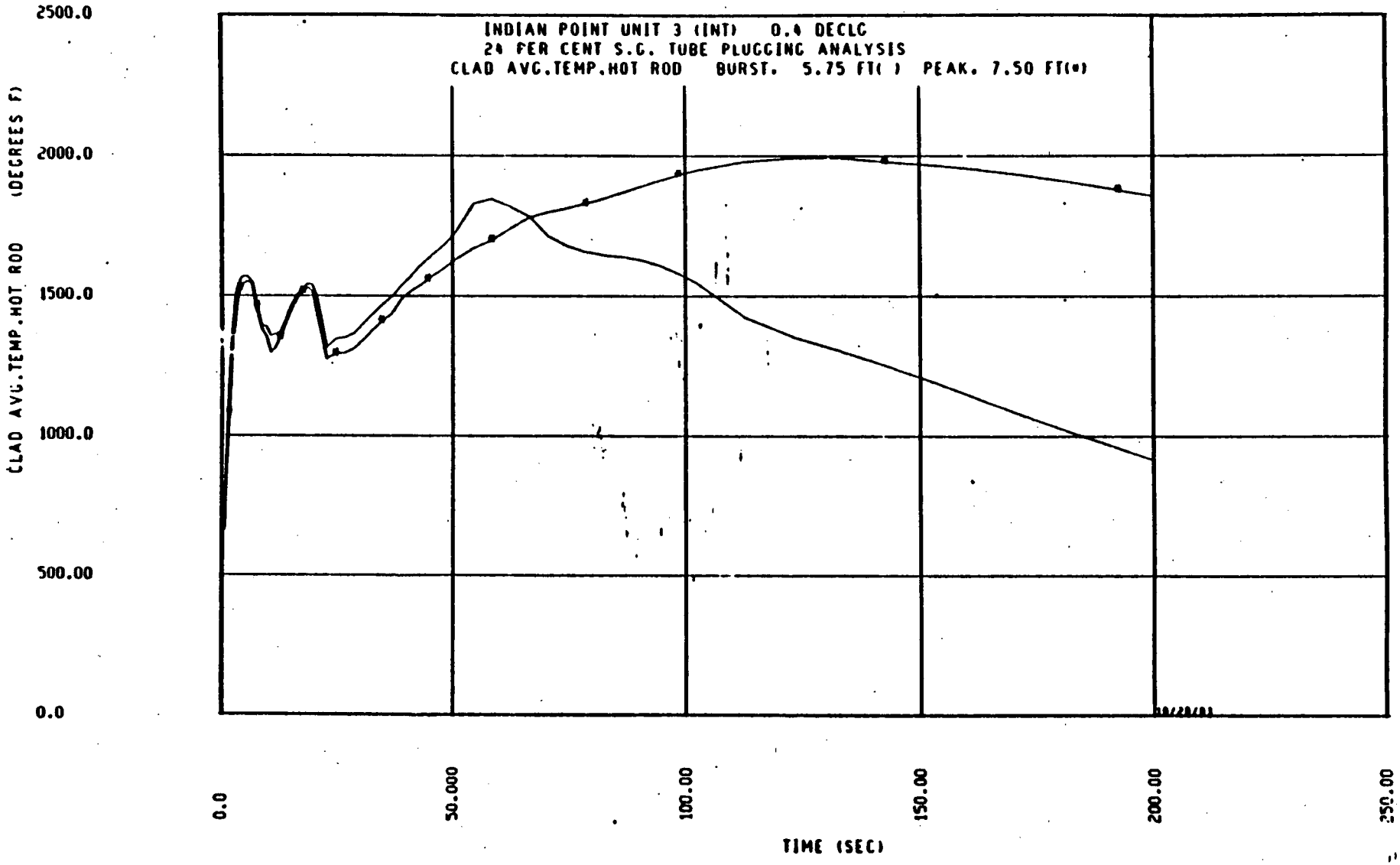

FIGURE 7D PEAK CLAD TEMPERATURE
DECLG(CD = 0.4)

Enclosure 2 to Safety Evaluation

LOCA Sensitivity to Asymmetric Steam Generator Tube Plugging

New York Power Authority
Indian Point 3 Nuclear Power Plant
Docket No. 50-286

SAFETY EVALUATION - LOCA

Sensitivity studies performed in the past and documented in WCAP-8986 have demonstrated that the increase in calculated peak clad temperature with uniform steam generator tube plugging is linear for many different Westinghouse PWR designs (2, 3 and 4-loop plants). These sensitivities to tube plugging are for an equal amount of plugging in each steam generator, hence the term uniform plugging.

The increase in PCT observed with increasing steam generator tube plugging is primarily a consequence of the added resistance to fluid flow through the coolant loops during core reflood. Because the added resistance represents the predominant phenonmenon associated with tube plugging and because of the linear nature of the PCT relationship, deviations in plugging from one steam generator to another do another do not significantly affect LOCA analysis results.

The impact of asymmetric tube plugging upon calculated ECCS performance may be determined by a review of the equations which describe the system behavior during core reflooding. The WREFLOOD model,is described in WCAP-8170. The WREFLOOD model as shown in Figure III-1, represents the loops (lumped intact and broken) and the reactor vessel. As Figure III-2 indicates, the intact loops constitute a resistance network which connects core and downcomer regions. Resistance networks also model the broken loop piping. Nomenclature of Figure III-2 is as follows:

P_D is downcomer static pressure

P_C is core static pressure

P_X is containment pressure

K is the resistance loss coefficient

Subscripts to K refer to loop (intact loop or broken loop) and location (hot leg, steam generator, etc.)

WREFLOOD is a quasi-steady-state code which models the venting of a core-generated steam-water mixture through the loops. The pertinent equations are presented below using the following additional nomenclature:

ΔP STUB is the pressure difference between vessel downcomer and containment

ρ_L is the liquid density in the downcomer

g is the gravitational constant

ΔZ is the difference in water level between downcomer and core

w is the mass flow rate through a reactor coolant loop

ρ_G is the gas density through the loops

A is the loop flow area; A_T is the total flow area in all loops

V_c is the core inlet velocity

F_{out} is the mass effluent fraction, the fraction of mass entering the core which is expelled

G_{core} is the mass velocity at the core exit

Consider loop behavior during the core reflood transient:

WREFLOOD equations state pressure relationships are

$$P_D = P_X + \Delta P_{STUB}$$
$$P_C = P_D + \rho_L g \Delta Z$$

The driving force for intact loop flow is

$$\Delta P = P_C - P_D = \rho_L g \Delta Z$$

Simplify the loop equation of WCAP-8170 p. 2-2 by eliminating small magnitude terms:

this gives $\Delta P = \dfrac{K_{IL} (w_{IL})^2}{2 \rho_G A_{IL}^2}$ for the intact loop

then $\rho_L g \Delta Z = \dfrac{K_{IL} (w_{IL})^2}{2 \rho_G A_{IL}^2}$

and $w_{IL} = A_{IL} \left[\dfrac{2 \rho_G \rho_L g \Delta Z}{K_{IL}} \right]^{1/2}$

Thus at any particular point ΔZ during the core reflood process

$$w_{IL} \propto A_{IL} \left(\frac{1}{K_{IL}}\right)^{1/2}$$

$$w_{IL} \propto (\text{frictional resistance})^{-1/2}$$

Apply the simplified equation to broken loop:

$$\Delta P_{BL} = \frac{K_{BL} (w_{BL})^2}{2 \rho_G A_{BL}^2}$$

for the broken loop

$$\Delta P_{BL} = P_C - P_X = \rho_L g \Delta Z + \Delta P_{STUB}$$

$$\text{then } \rho_L g \Delta Z + \Delta P_{STUB} = \frac{K_{BL} \, (w_{BL})^2}{2\rho_G \, A_{BL}^{\,2}}$$

$$\text{so } w_{BL} = A_{BL} \left[\frac{2\rho_G \rho_L g \Delta Z + 2\rho_G \Delta P_{STUB}}{K_{BL}}\right]^{1/2}$$

A review of the IP3 limiting break (C_D = 0.4 DECLG) reveals that ΔP_{STUB} is small compared to $\rho_L g \Delta Z$ until calculated clad temperature has increased to a value near PCT. Therefore, the ΔP_{STUB} term may be ignored to obtain

$$w_{BL} \cong A_{BL} \left[\frac{2\rho_G \, \rho_L \, g \, \Delta Z}{K_{BL}}\right]^{1/2}$$

$$w_{BL} \propto A_{BL} \left(\frac{1}{K_{BL}}\right)^{1/2}$$

From p. 2-6 of WCAP-8170, the loop flow boundary condition at the core is

$$G_{core} = V_C \cdot \rho_L \cdot F_{out}$$

which may be written as

$$\frac{w_{BL} + w_{IL}}{A_C} = V_C \cdot \rho_L \cdot F_{out}$$

$$\text{Therefore } V_C = \frac{w_{BL} + w_{IL}}{A_C \, \rho_L \, F_{out}}$$

Core flooding rate V_C is determined by the ability to vent core-generated steam through the loops and is directly proportional to the sum (w_{BL} + w_{IL}). IP3 exhibits its calculated PCT well into the core reflood portion of the LOCA transient: PCT is directly related to the magnitude of the flooding rate. The higher the value of V_C[and (w_{BL} + w_{IL})] the lower the calculated PCT in the IP3 1981 Model analysis.

The effect of tube plugging configurations upon total flow exiting the core (w_{BL} + w_{IL}) can now be assessed from the proportionality relationships. For a 4-loop plant the total flow through the loops, w_τ, is given as

$$w_\tau = w_{IL} + w_{BL} \propto \left(\frac{A_{IL}^2}{K_{IL}}\right)^{1/2} + \left(\frac{A_{BL}^2}{K_{BL}}\right)^{1/2}$$

$$\text{or } w_\tau \propto .75\ A_\tau\ K_{IL}^{-1/2} + .25\ A_\tau\ K_{BL}^{-1/2}$$

in an original, unplugged state $K_{IL_0} = K_{BL_0}$:

$$w_\tau \propto 1.0\ A_\tau\ K_{IL_0}^{-1/2} \equiv w_0$$

When SG tube plugging is introduced, w_0 will be diminished due to an increase in frictional resistance. In the following presentation changes in resistance caused by SG tube plugging will be applied to the loss coefficient (K) term of the [$A*K^{-1/2}$] expressions while A is held constant for ease of computation. Since no critical flow effects are involved the flow impact of SG tube plugging can be properly represented in this fashion. The uniform plugging case and two bounding asymmetric plugging cases are considered.

I. Uniform SG Tube Plugging Case

An added resistance (considered to be due to SG tube plugging) is introduced into each loop at IP3. Assume conservatively that the magnitude of the added resistance to flow is 10% of the original total loop resistance. In the 24% uniform SG tube plugging WREFLOOD cases, the steam generator accounts for slightly more than 30% of the total IP3 loop resistance to flow.

$$W_\tau \propto .75A_\tau(K_{IL})^{-1/2} + .25A_\tau(K_{BL})^{-1/2}$$

$$W_\tau \propto .75A_\tau\ (K_{IL_0}*1.1)^{-1/2} + .25A_\tau\ (K_{IL_0}*1.1)^{-1/2}$$

$$W_\tau \propto 1.0A_\tau\ (K_{IL_0}*1.1)^{-1/2} \propto W_0*1.1^{-1/2}$$

$$W_\tau \propto .9535\ W_0$$

Total flow exiting the core is reduced by .0465 in this uniform resistance case.

II. All Plugging in Broken Loop

The [4*(10% of individual loop resistance)] added resistance is placed totally into the broken loop in WREFLOOD. Then

$$W_\tau \propto .75A_\tau\ K_{IL}^{-1/2} + .25A_\tau K_{BL}^{-1/2}$$

$$W_\tau \propto .75A_\tau\ (K_{IL_0})^{-1/2} + .25A_\tau\ (1.4\ K_{IL_0})^{-1/2}$$

$$W_\tau \propto (.75 + \frac{.25}{\sqrt{1.4}})\ A_\tau\ K_{IL_0}^{-1/2}$$

$$W_\tau \propto (.75 + .2113)\ W_0 = .9613\ W_0$$

Total flow exiting the core is reduced by .0387 in this case. The reduction in w_τ and V_C (and the subsequent rise in PCT) is predicted to be less for this configuration. Therefore, there is no difference of significance relative to the uniform plugging configuration.

III. No Plugging in Broken Loop

None of the added resistance is placed into the broken loop. Thus 4(0.1) = 0.4 of the base loop resistance is added to the lumped intact loop, so its K value becomes 0.4/3 = 1.133 of its original value on a lumped basis.

$$W_\tau \propto .75A_\tau K_{IL}^{-1/2} + .25A_\tau K_{BL}^{-1/2}$$

$$\propto .75 A_\tau (K_{IL_0} * 1.133)^{-1/2} + .25 A_\tau K_{IL_0}^{-1/2}$$

$$W_\tau \propto [.75 (1.133)^{-1/2} + .25] W_0$$

$$W_\tau \propto .9545 W_0$$

Total flow exiting the core is reduced by .0455. The reduction in W_τ and V_c is a bit less for this configuration. The above discussion has shown that asymmetry presumed in steam generator tube plugging causes no adverse effects based on the WREFLOOD equations. However, the arguments presented here should only be applied to the established range of applicability in which WREFLOOD has been employed in Evaluation Model ECCS computations. The indicated upper bound is a 30% steam generator tube plugging level in any SG unit.

The above discussion has demonstrated based upon the pertinent WREFLOOD equations that presumed asymmetry in steam generator tube plugging does not adversely impact calculated ECCS performance at a given plugging level. Therefore, the existing 24% uniform tube plugging ECCS performance analyses for Indian Point 3 will support continued plant operation as long as:

1. The number of tubes plugged in all four steam generators remains less than 24% of the total number of SG tubes present in the plant.

2. The number of equivalent tubes plugged does not exceed 30% of the 3260 tubes present in any steam generator.

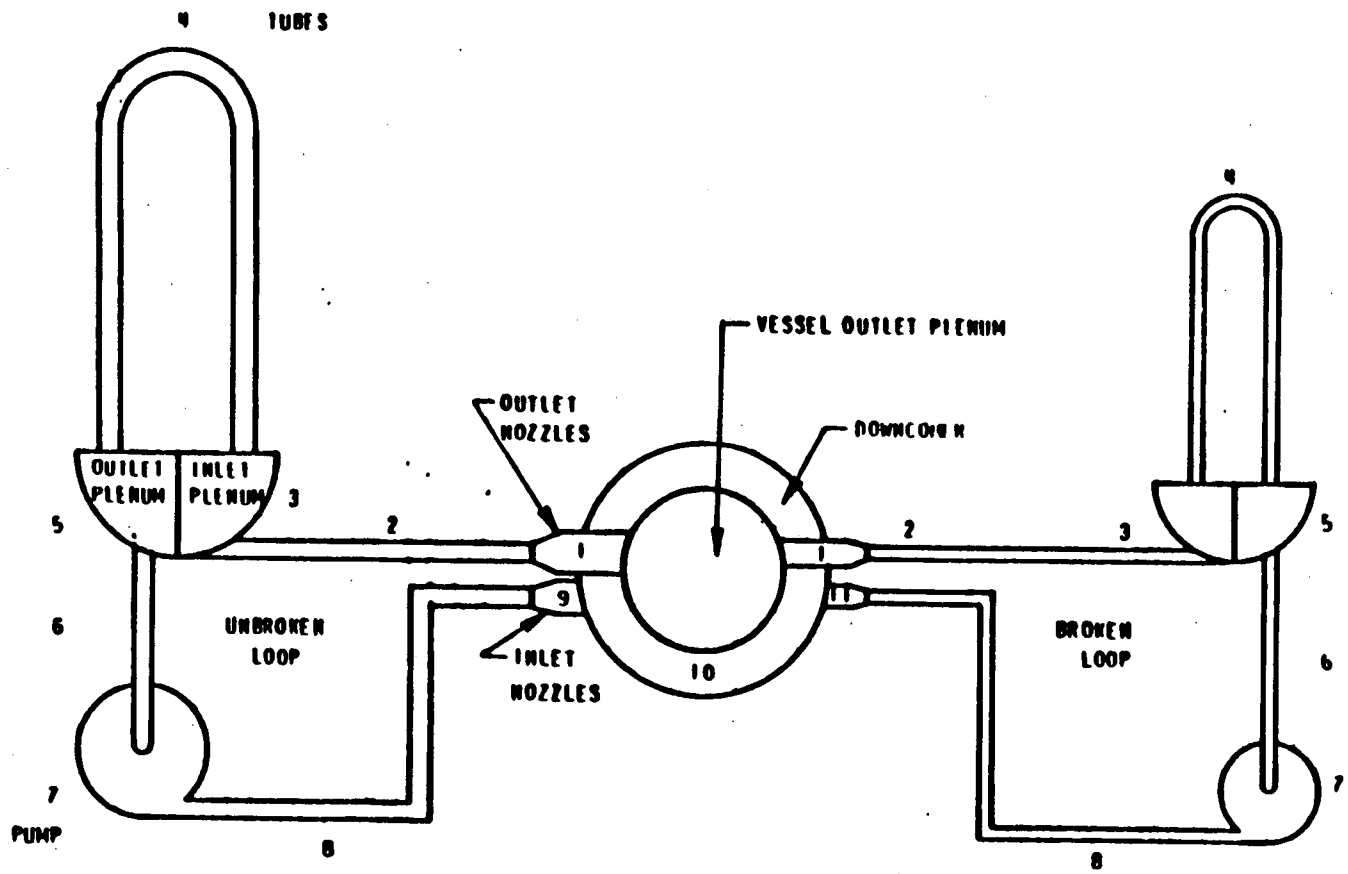

FIGURE III-1: SCHEMATIC OF WREFLOOD MODEL OF WESTINGHOUSE PWR

FIGURE III-2: WREFLOOD RESISTANCE NETWORK REPRESENTATION OF A PWR