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SUBJECT: Application for amend to License DPR-58 revising Tech Specs to extend fuel peak pellet burnup & increase FQ value limit in fuel.Fee paid.

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INDIANA & MICHIGAN ELECTRIC COMPANY

P.O. BOX 16631
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August 23, 1984
AEP:NRC:0745M

Donald C. Cook Nuclear Plant Unit No. 1
Docket No. 50-315
License No. DPR-58
TECHNICAL SPECIFICATION CHANGE REQUESTS

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Denton:

By this letter and its attachments, we request changes to the Technical Specifications for the Donald C. Cook Nuclear Plant Unit No. 1. The proposed revised Technical Specification pages are contained in Attachment A. The reasons for the proposed changes to the Technical Specifications, and the justifications that the changes do not involve significant hazards considerations, are contained in Attachments B and C to this letter. The changes described in Attachment B involve extending the peak pellet burnup in fuel supplied by Exxon Nuclear Company from 42,200 MWD/MTU (42.2 MWD/KG) to 48,000 MWD/MTU (48.0 MWD/KG). These changes are supported by a LOCA Analysis and additional information regarding mechanical design, which was sent to you directly by Exxon Nuclear Company with letter JCC:113:84, dated August 21, 1984. The current burnup limit is expected to be reached on November 30, 1984. Without this burnup extension, we would be unable to continue operation of Cycle 8 because of the requirements of Technical Specification Section 3.2.2. The changes proposed in Attachment C and supported by Attachment D involve an increase in the F_Q limit in fuel supplied by Westinghouse from 1.97 to 2.10. It should be noted that part of this analysis is based on use of the BART code, which has not been previously used for the Donald C. Cook Nuclear Plant, Unit 1.

These proposed changes have been reviewed by the Plant Nuclear Safety Review Committee (PNSRC) and will be reviewed by the Nuclear Safety and Design Review Committee (NSDRC) at their next regularly scheduled meeting.

In compliance with the requirements of 10 CFR 50.91(b)(1), a copy of this letter and its attachments have been transmitted to Mr. R. C. Callen of the Michigan Public Service Commission.

A001 1/1 Rec'd w/out Check

Pursuant to 10 CFR 170.12, we have enclosed a check in the amount of $150.00 as payment for the application fee for the proposed amount.

This document has been prepared following Corporate procedures which incorporate a reasonable set of controls to insure its accuracy and completeness prior to signature by the undersigned.

Very truly yours,

M. P. Alexich
Vice President

RPK 8/28/84

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Attachments: A. Proposed Revised Technical Specifications Pages for D.C. Cook Unit 1.

B. Reasons for the extension of the peak pellet burnup allowed in fuel supplied by Exxon Nuclear Company and justification that the changes do not involve significant hazards considerations.

C. Reasons for the increase in F_Q for fuel supplied by Westinghouse.

D. "D.C. Cook Unit 1 3411 MWt Large Break LOCA Analysis", Westinghouse Electric Corporation, June, 1984.

cc: John E. Dolan
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Charnoff
E. R. Swanson, NRC Resident Inspector - Bridgman

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Attachment D

"D.C. Cook Unit 1 3411 MWt Large Break LOCA Analysis", Westinghouse Electric Corporation, June, 1983.*

* This document has been prepared by Westinghouse Electric Corporation in a format for eventual inclusion in the Donald C. Cook Nuclear Plant FSAR. Although this is not intended for that purpose at this time, the format has been retained for convenience.

14.3.1.1 Major LOCA Analyses Applicable to Westinghouse Fuel

Identification of Causes and Frequency Classification

A loss-of-coolant accident (LOCA) is the result of a pipe rupture of the RCS pressure boundary. For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is considered an ANS Condition IV event, a limiting fault, in that it is not expected to occur during the lifetime of D. C. Cook Unit 1, but is postulated as a conservative design basis.

The Acceptance Criteria for the LOCA are described in 10 CFR 50.46 (10 CFR 50.46 and Appendix K of 10 CFR 50 1974)[1] as follows:

1. The calculated peak fuel element clad temperature is below the requirement of 2,200°F.

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

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

4. The core remains amenable to cooling during and after the break.

5. 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.

These criteria were established to provide significant margin in emergency core cooling system (ECCS) performance following a LOCA. WASH-1400 (USNRC 1975)[10] presents a recent study in regards to the probability of occurrence of RCS pipe ruptures.

Sequence of Events and Systems Operations

Should a major break occur, depressurizaton of the RCS results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. A safety injection signal is generated when the appropriate setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection supplement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. However, no credit is taken in the LOCA analysis for the boron content of the injection water. In addition, the insertion of control rods to shut down the reactor is neglected in the large break analysis.

2. Injection of borated water provides for heat transfer from the core and prevents excessive clad temperatures.

Description of Large Break Loss-of-Coolant Accident Transient

The sequence of events following a large break LOCA is presented in Table 14.3.1-6.

Before the break occurs, the unit is in an equilibrium condition; that is, the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay, hot internals and the vessel, continues to be transferred to the reactor coolant. At the beginning of the blowdown phase, the entire RCS contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break develops, the time to departure from nucleate boiling is calculated, consistent with Appendix K of 10 CFR 50.[1] Thereafter the core heat transfer is unstable, with both nucleate boiling and film boiling occurring. As the core becomes uncovered, both turbulent and laminar forced convection and radiation are considered as core heat transfer mechanisms.

The heat transfer between the RCS and the secondary system may be in either direction, depending on the relative temperatures. In the case of continued heat addition to the secondary system, the secondary system pressure increases and the main steam safety valves may actuate to limit the pressure. Makeup water to the secondary side is automatically provided by the emergency feedwater system. The safety injection signal actuates a feedwater isolation signal which isolates normal feedwater flow by closing the main feedwater isolation valves, and also initiates emergency feedwater flow by starting the emergency feedwater pumps. The secondary flow aids in the reduction of RCS pressure.

When the RCS depressurizes to 600 psia, the accumulators begin to inject borated water into the reactor coolant loops. The conservative assumption is made that accumulator water injected bypasses the core and goes out through the break until the termination of bypass. This conservatism is again consistent with Appendix K of 10CFR50. Since loss of offsite power (LOOP) is assumed, the RCPs are assumed to trip at the inception of the accident. The effects of pump coastdown are included in the blowdown analysis.

The blowdown phase of the transient ends when the RCS pressure (initially assumed at 2280 psia) falls to a value approaching that of the containment atmosphere. Prior to or at the end of the blowdown, the

mechanisms that are responsible for the emergency core cooling water injected into the RCS bypassing the core are calculated not to be effective. At this time (called end-of-bypass) refill of the reactor vessel lower plenum begins. Refill is completed when emergency core cooling water has filled the lower plenum of the reactor vessel, which is bounded by the bottom of the fuel rods (called bottom of core recovery time).

The reflood phase of the transient is defined as the time period lasting from the end-of-refill until the reactor vessel has been filled with water to the extent that the core temperature rise has been terminated. From the latter stage of blowdown and then the beginning-of-reflood, the safety injection accumulator tanks rapidly discharge borated cooling water into the RCS, contributing to the filling of the reactor vessel downcomer. The downcomer water elevation head provides the driving force required for the reflooding of the reactor core. The low head and high head safety injection pumps aid in the filling of the downcomer and subsequently supply water to maintain a full downcomer and complete the reflooding process.

Continued operation of the ECCS pumps supplies water during longterm cooling. Core temperatures have been reduced to longterm steady state levels associated with dissipation of residual heat generation. After the water level of the residual water storage tank (RWST) reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching to the cold recirculation phase of operation, in which spilled borated water is drawn from the engineered safety features (ESF) containment sumps by the low head safety injection (residual heat removal) pumps and returned to the RCS cold legs. The containment spray system continues to operate to further reduce containment pressure.

Approximately 24 hours after initiation of the LOCA, the ECCS is realigned to supply water to the RCS hot legs in order to control the boric acid concentration in the reactor vessel.

Core and System Performance

Mathematical Model:

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50 (Federal Register 1974).[1]

Large Break LOCA Evaluation Model

The analysis of a large break LOCA transient is divided into three phases: (1) blowdown, (2) refill, and (3) reflood. There are three distinct transients analyzed in each phase, including the thermal-hydraulic transient in the RCS, the pressure and temperature transient within the containment, and the fuel and clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of interrelated computer codes has been developed for the analysis of the LOCA.

A description of the various aspects of the LOCA analysis methodology is given by Bordelon, Massie, and Zordan (1974).[6] This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the Acceptance Criteria. The SATAN-VI, WREFLOOD, BART and LOCTA-IV codes, which are used in the LOCA analysis, are described in detail by Bordelon et al. (1974)[5]; Kelly et al. (1974)[9]; Young et al. (1980)[16]; Bordelon and Murphy (1974)[4]; and Bordelon et al. (1974).[6] Code modifications are specified in References 2, 7 and 13. These codes assess the core heat transfer geometry and determine if the core remains amenable to cooling throughout and subsequent to the blowdown, refill, and reflood phases of the LOCA. The SATAN-VI computer code analyzes the thermal-hydraulic transient in the RCS during blowdown and the WREFLOOD computer code calculates this transient during the refill and reflood phases to the accident. The LOTIC computer code, described by Hsieh and Raymund in

WCAP-8355 (1975) and WCAP-8345 (1974)[3], calculates the containment pressure transient. The containment pressure transient is input to WREFLOOD for the purpose of calculating the reflood transient. The LOCTA-IV computer code calculates the thermal transient of the hottest fuel rod during the three phases. The standard Pad Fuel Thermal Safety Model, described in Reference 15, generates the initial fuel rod conditions input to LOCTA-IV.

SATAN-VI calculates the RCS pressure, enthalpy, density, and the mass and energy flow rates in the RCS, as well as steam generator energy transfer between the primary and secondary systems as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator water mass and internal pressure and the pipe break mass and energy flow rates that are assumed to be vented to the containment during blowdown. At the end of the blowdown phase, these data are transferred to the WREFLOOD code. Also, at the end-of-blowdown, the mass and energy release rates during blowdown are input to the LOTIC code for use in the determination of the containment pressure response during this first phase of the LOCA. Additional SATAN-VI output data from the end-of-blowdown, including the core inlet flow rate and enthalpy, the core pressure, and the core power decay transient, are input to the LOCTA-IV code.

With input from the SATAN-VI code, WREFLOOD uses a system thermal-hydraulic model to determine the core flooding rate (that is, the rate at which coolant enters the bottom of the core), the coolant pressure and temperature, and the quench front height during the reflood phase of the LOCA. WREFLOOD also calculates the mass and energy flow addition to the containment through the break. WREFLOOD is also linked to the BART and LOCTA-IV codes. The heat transfer calculation for the average fuel channel in the hot assembly during the reflood phase of the LOCA is performed by the BART[16] computer code using a mechanistic core heat transfer model. This information is then used by LOCTA-IV throughout the analysis of the LOCA transient to calculate the fuel clad temperature and metal-water reaction of the hottest rod in the core.

The large break analysis was performed with the December 1981 version of the Evaluation Model modified to incorporate the BART[16] computer code.

Input Parameters and Initial Conditions:

The analysis presented in this section was performed with a reactor vessel upper head temperature equal to the RCS hot leg temperature.

The bases used to select the numerical values that are input parameters to the analysis have been conservatively determined from extensive sensitivity studies (Westinghouse 1974[12]; Salvatori 1974[11]; Johnson, Massie, and Thompson 1975[8]). In addition, the requirements of Appendix K regarding specific model features were met by selecting models which provide a significant overall conservatism in the analysis. The assumptions which were made pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs, and include such items as the core peaking factors, the containment pressure, and the performance of the ECCS. Decay heat generated throughout the transient is also conservatively calculated.

A meeting was held at the Westinghouse Licensing Office in Bethesda on December 17, 1981 between members of the U. S. Nuclear Regulatory Commission and members of the Westinghouse Nuclear Safety Department to discuss the impact of maximum safety injection on the large break ECCS analysis on a generic basis. Further discussion of this issue is provided in a letter from E. P. Rahe, Manager of Westinghouse Nuclear Safety Department, to Robert L. Tedesco of the U. S. Nuclear Regulatory Commission.[14] A brief description of this issue is given below.

Westinghouse ECCS analyses currently assume minimum safeguards for the safety injection flow, which minimizes the amount of flow to the RCS by assuming maximum injection line resistances, degraded ECCS pump performance, and the loss of one residual heat removal (RHR) pump as the most limiting single failure. This is the limiting single failure assumption when offsite power is unavailable for most Westinghouse

plants. However, for some Westinghouse plants including D. C. Cook Unit 1, the current nature of the Appendix K ECCS evaluation models is such that it may be more limiting to assume the maximum possible ECCS flow delivery. In that case, maximum safeguards which assume minimum injection line resistances, enhanced ECCS pump performance, and no single failure, result in the highest amount of flow delivered to the RCS.

Current LOCA analysis for the D. C. Cook Unit 1 has demonstrated that maximum safeguards assumptions result in the highest peak clad temperature. Therefore, the worst break for D. C. Cook (C_D = 0.6) was re-analyzed, assuming maximum safeguards.

Results:

Based on the results of the LOCA sensitivity studies (Westinghouse 1974[12]; Salvatori 1974[11]; Johnson, Massie, and Thompson 1975[8]) the limiting large break was found to be the double ended cold leg guillotine (DECLG). Therefore, only the DECLG break is considered in the large break ECCS performance analysis. Calculations were performed for a range of Moody break discharge coefficients. The results of these calculations are summarized in Tables 14.3.1-5 and 14.3.1-6.

The containment data used to generate the LOTIC backpressure transient are shown in Table 14.3.1-1. The mass and energy release data for the minimum and maximum safeguards cases are shown in Tables 14.3.1-2 and 14.3.1-3 respectively. Nitrogen release rates to the containment are given in Table 14.3.1-4.

Figures 14.3.1-1 through 14.3.1-64 present the transients for the principal parameters for the break sizes analyzed. The following items are noted:

Figures 14.3.1-1 through 14.3.1-12

The following quantities are presented at the clad burst location and at the hot spot (location of maximum clad temperature), both on the hottest fuel rod (hot rod):

1. fluid quality;
2. mass velocity;
3. heat transfer coefficient.

The heat transfer coefficient shown is calculated by the LOCTA-IV code.

Figures 14.3.1-13 through 14.3.1-24

The system pressure shown is the calculated pressure in the core. The flow rate from the break is plotted as the sum of both ends for the guillotine break cases. The core pressure drop shown is from the lower plenum, near the core, to the upper plenum at the core outlet.

Figures 14.3.1-25 through 14.3.1-36

These figures show the hot spot clad temperature transient and the clad temperature transient at the burst location. The fluid temperature shown is also for the hot spot and burst location. The core flow (top and bottom) is also shown.

Figures 14.3.1-37 through 14.3.1-44

These figures show the core reflood transient.

Figures 14.3.1-45 through 14.3.1-52

These figures show the Emergency Core Cooling System flow for all of the cases analyzed. As described earlier, the accumulator delivery during blowdown is discarded until the end of bypass is calculated. Accumulator flow, however, is established in the refill and the reflood calculations. The accumulator flow assumed is the sum of that injected in the intact cold legs.

<u>Figures 14.3.1-53 through 14.3.1-54</u> — The containment pressure transient used in the analysis is also provided for the minimum and maximum SI cases.

<u>Figures 14.3.1-55 and 14.3.1-60</u> — These figures show the heat removal rates of the heat sinks found in the lower compartment and the heat removal by the lower containment drain, and the heat removal by the sump and LC sprays (minimum and maximum SI cases).

<u>Figures 14.3.1-61 through 14.3.1-64</u> — These figures show the temperature transients in both the upper and lower compartments of the containment and flow from the upper to lower compartments. Total heat removal in the lower compartment is the sum of all the heat removal rates shown (for minimum and maximum SI cases).

The maximum clad temperature calculated for a large break is 2163°F, which is less than the Acceptance Criteria limit of 2200°F. The maximum local metal-water reaction is 9.65 percent, which is well below the embrittlement limit of 17 percent as required by 10 CFR 50.46. The total core metal-water reaction is less than 0.3 percent for all breaks, as compared with the 1 percent criterion of 10 CFR 50.46. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

References for Section 14.3.1.1

1. "Acceptance Criteria for Emergency Core Cooling System for Light Water Cooled Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register 1974, Volume 39, Number 3.

2. Rahe, E. P. (Westinghouse), letter to J. R. Miller (USNRC), Letter No. NS-EPRS-2679, November 1982.

3. Hsieh, T., and Raymund, M., "Long Term Ice Condenser Containment LOTIC Code Supplement 1," WCAP-8355, Supplement 1, May 1975, WCAP-8345 (Proprietary), July 1974.

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

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

6. Bordelon, F. M.; Massie, H. W.; and Zordan, T. A., "Westinghouse ECCS Evaluation Model - Summary," WCAP-8339, 1974.

7. Rahe, E. P., "Westinghouse ECCS Evaluation Model, 1981 Version," WCAP-9220-P-A (Proprietary Version), WCAP-9221-P-A (Non-proprietary version), Revision 1, 1981.

8. Johnson, W. J.; Massie, H. W.; and Thompson, C. M., "Westinghouse ECCS - Four Loop Plant (17x17) Sensitivity Studies," WCAP-8565-P-A (Proprietary) and WCAP-8566-A (Non-proprietary), 1975.

9. Kelly, R. D. et al., "Calculational Model for Core Reflooding After a Loss-of-Coolant Accident (WREFLOOD Code)," WCAP-8170 (Proprietary) and WCAP-8171 (Non-proprietary), 1974.

10. U. S. Nuclear Regulatory Commission 1975, "Reactor Safety Study - An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants," WASH-1400, NUREG-75/014.

11. Salvatori, R., "Westinghouse ECCS - Plant Sensitivity Studies," WCAP-8340 (Proprietary) and WCAP-8356 (Non-proprietary), 1974.

12. "Westinghouse ECCS - Evaluation Model Sensitivity Studies," WCAP-8341 (Proprietary) and WCAP-8342 (Non-proprietary), 1974.

13. Bordelon, F. M., et al., "Westinghouse ECCS Evaluation Model - Supplementary Information," WCAP-8471 (Proprietary) and WCAP-8472 (Non-proprietary), 1975.

14. Rahe, E. P. (Westinghouse). Letter to Robert L. Tedesco (USNRC), Letter No. NS-EPR-2538, December 1981.

15. Letter from J. F. Stoltz (NRC) to T. M. Anderson (Westinghouse); subject: Review of WCAP-8720, Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations.

16. Young, M. Y., et al., "BART-A1: A Computer Code for the Best Estimate Analysis of Reflood Transients, "WCAP-9561-P-A (Proprietary) and WCAP-9695-A (Non-Proprietary) January 1980.

TABLE 14.3.1-1
LARGE BREAK
CONTAINMENT DATA
(ICE CONDENSER CONTAINMENT)

NET FREE VOLUME		
(Includes Distribution Between Upper, Lower, and Dead-Ended Compartments)	UC	746,829 ft^3
	LC	249,446
	DE	116,168
	IC	122,400
Initial Conditions		
Pressure		14.7 psia
Temperature for the Upper, Lower and Dead-Ended Compartments	UC	100°F
	LC	120°F
	DE	120°F
RWST Temperature		70°F
Service Water Temperature		40°F
Temperature Outside Containment		-7°F
Initial Spray Temperature		70°F
Spray System		
Burnout Flow for a Spray Pump		3600 gpm
Number of Spray Pumps Operating		2
Post-Accident Initiation of Spray System		40 secs
Distribution of the Spray Flow to the Upper and Lower Compartments	LC	2335 gpm
	UC	4365 gpm
Deck Fan		
Post-Accident Initiation of Deck Fans		600 secs
Flow Rate Per Fan		39,000 cfm per fan
Hydrogen Skimmer System Flow Rate		2800 cfm per fan
Assumed Spray Efficiency of Water from Ice Condenser Drains		100%

TABLE 14.3.1.1
(continued)

STRUCTURAL HEAT SINKS

Compartment		Area (ft^2)	Thickness (ft)	Material
1.	LC	12,105	0.0469/2.0	steel/concrete
2.	LC	11,700	2.0	concrete
3.	LC	65,980	1.35	concrete
4.	LC	5,481	0.0833	steel
5.	LC	4,735	0.01147	steel
6.	LC	289	0.25	lead
7.	LC	14,690	0.0079	steel
8.	LC	3,439	0.1561	steel
9.	LC	5,775	0.009	steel
10.	LC	4,966	0.0096	steel
11.	LC	7,013	0.037	steel
12.	LC	2,457	0.0334	steel
13.	UC	378	.1667/.0365	steel/concrete
14.	UC	29,772	.0092	steel
15.	UC	8,033	.0209	steel
16.	UC	420	.0052	steel
17.	UC	29,330	1.47	concrete
18.	UC	34,125	0.0469/2.0	steel/concrete
19.	UC	210	.0052	steel

UC: Upper Compartment
LC: Lower Compartment
DE: Dead-Ended Compartment
IC: Ice Condenser Compartment

TABLE 14.3.1-2

MASS AND ENERGY RELEASE RATES
MINIMUM SI

TIME (sec)	MASS (lb/sec)	ENERGY (BTU/sec)
0.	.5782E+05	.3082E+08
.2000E+01	.4783E+05	.2478E+08
.4000E+01	.3422E+05	.1798E+08
.6000E+01	.2563E+05	.1377E+08
.8000E+01	.2225E+05	.1223E+08
.1000E+02	.2044E+05	.1140E+08
.1200E+02	.1806E+05	.1037E+08
.1240E+02	.1665E+05	.9762E+07
.1400E+02	.1561E+05	.9229E+07
.1500E+02	.1436E+05	.8603E+07
.1600E+02	.1319E+05	.7990E+07
.1800E+02	.1134E+05	.6925E+07
.1900E+02	.1061E+05	.6491E+07
.2000E+02	.9917E+04	.6106E+07
.2100E+02	.8999E+04	.5628E+07
.2200E+02	.8183E+04	.5086E+07
.2400E+02	.6407E+04	.4042E+07
.2500E+02	.5476E+04	.3462E+07
.2600E+02	.4450E+04	.2730E+07
.2700E+02	.6099E+04	.2983E+07
.2800E+02	.6809E+04	.3004E+07
.2920E+02	.7005E+04	.2753E+07
.3000E+02	.6831E+04	.2513E+07
.3100E+02	.6248E+04	.2093E+07
.3200E+02	.6371E+04	.1982E+07
.3300E+02	.4868E+04	.1391E+07
.3500E+02	.4316E+04	.1019E+07
.3700E+02	.2298E+04	.6256E+06
.3800E+02	.6676E+03	.1714E-06
.3849E+02	.6587E+03	.1619E+06
.4500E+02	.1730E+03	.6583E+04
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.5325E+02	.1768E+03	.1148E+05
.5355E+02	.1768E+03	.1145E+05
.5375E+02	.1767E+03	.1135E+05
.5385E+02	.1767E+03	.1134E+05
.5973E+02	.2050E+03	.4310E+05
.7020E+02	.5402E+03	.2098E+06
.8640E+02	.5729E+03	.2153E+06
.1069E+03	.5860E+03	.2128E+06
.1302E+03	.5947E+03	.2081E+06
.1560E+03	.6022E+03	.2027E+06
.2152E+03	.6160E+03	.1907E+06
.2887E+03	.6317E+03	.1792E+06
.4107E+03	.6536E+03	.1631E+06
.4434E+03	.6593E+03	.1635E+06

TABLE 14.3.1-3

MASS AND ENERGY RELEASE RATES
MAXIMUM SI

TIME (sec)	MASS (lb/sec)	ENERGY (BTU/sec)
0		
.2000E-01	.6776E+05	.3607E+08
.1000E+01	.5500E+05	.2874E+08
.5000E+01	.3881E+05	.2069E+08
.8000E+01	.3041E+05	.1687E+08
.1000E+02	.2738E+05	.1542E+08
.1200E+02	.2382E+05	.1379E+08
.1240E+02	.1888E+05	.1129E+08
.1400E+02	.1802E+05	.1084E+08
.1500E+02	.1455E+05	.9098E+07
.1600E+02	.1282E+05	.8225E+07
.1700E+02	.1120E+05	.7433E+07
.1800E+02	.9375E+04	.6562E+07
.1900E+02	.8597E+04	.6095E+07
.2000E+02	.7564E+04	.5415E+07
.2100E+02	.5880E+04	.4147E+07
.2200E+02	.4047E+04	.2945E+07
.2300E+02	.5129E+04	.2832E+07
.2400E+02	.6880E+04	.2968E+07
.2500E+02	.7206E+04	.2679E+07
.2600E+02	.6010E+04	.1877E+07
.2700E+02	.4829E+04	.1282E+07
.2800E+02	.4337E+04	.1059E+07
.2895E+02	.3670E+04	.8232E+06
.2900E+02	.2623E+04	.4406E+06
.3000E+02	.2416E+04	.3675E+06
.3043E+02	.2380E+04	.3406E+06
.3400E+02	.2357E+04	.3196E+06
.3800E+02	.1542E+04	.8154E+05
.4000E+02	.3425E+03	.1303E+05
.4248E+02	.3425E+03	.1303E+05
.4308E+02	.3425E+03	.1303E+05
.4328E+02	.3470E+03	.1692E+05
.4338E+02	.3470E+03	.1891E+05
.4348E+02	.3470E+03	.1890E+05
.4358E+02	.3470E+03	.1889E+05
.4885E+02	.3469E+03	.1879E+05
.5941E+02	.3762E+03	.5583E+05
.7796E+02	.4579E+04	.4415E+06
.1022E+03	.1486E+04	.2436E+06
.1309E+03	.1505E+04	.2406E+06
.1639E+03	.1515E+04	.2352E+06
.2516E+03	.1524E+04	.2317E+06
	.1545E+04	.2240E+06

TABLE 14.3.1-A-3
NITROGEN MASS AND ENERGY
RELEASE RATES

Time (sec)	Flow Rate (lbs/sec)
37.5	71.9
39.5	60.7
45.5	37.2
47.5	31.6
53.5	18.8
55.5	15.6
61.5	8.5
63.5	6.9
70.3	186.0
72.3	158.0
78.5	97.3
80.5	82.4
86.3	48.5
88.3	40.0
94.3	21.9
96.3	18.2
102.2	11.7
104.2	10.5
110.2	7.6
112.2	6.8
126.2	3.3
128.2	2.9
138.2	1.8
140.2	1.6
146.2	1.2
148.2	1.1
174.2	0.25
176.2	0.075

TABLE 14.3.1-A-4

LARGE BREAK

Results	DECLG C_D=0.4 Max SI
Peak Clad Temp. °F	2162
Peak Clad Location Ft.	7.50
Local Zr/H_2O Reaction (Max)%	6.58
Local Zr/H_2O Location Ft.	7.50
Total Zr/H_2O Reaction %	< 0.3
Hot Rod Burst Time sec.	71.4
Hot Rod Burst Location Ft.	6.75

Calculation	
Licensed Core Power (Mwt) 102% of	3250
Peak Linear Power (kw/ft) 102% of	13.225
Peaking Factor (at License Rating)	1.97
Accumulator Water Volume (ft^3) per Accumulator	950

Cycle Analyzed Cycle 8

TABLE 14.3.1-A-5
LARGE BREAK
TIME SEQUENCE OF EVENTS

	Max SI DECLG C_D=0.4 (sec)
START	0.00
Reactor Trip Signal	0.60
Safety Injection Signal	4.05
Accumulator Injection	20.50
End of Blowdown	38.70
Bottom of Core Recovery	52.78
Accumulator Empty	67.45
Pump Injection	29.05

TABLE 14.3.1-4

NITROGEN MASS AND ENERGY RELEASE RATES

Time (sec)	Flow Rate (lbs/sec)
37.5	71.9
39.5	60.7
45.5	37.2
47.5	31.6
53.5	18.8
55.5	15.6
57.5	12.8
60.7	266.81
66.7	159.7
68.7	135.7
74.7	83.2
76.7	70.3
78.7	58.9
80.7	49.1
86.7	27.2
88.7	22.3
98.7	10.7
100.7	9.6
110.7	5.6
112.7	5.1
122.7	3.0
124.7	2.7
130.7	2.0
132.7	1.8
146.6	0.8
145.5	0.7

TABLE 14.3.1-5
LARGE BREAK

Results	DECLG $C_D=0.8$ Min SI	DECLG $C_D=0.6$ Min SI	DECLG $C_D=0.4$ Min SI	DECLG $C_D=0.6$ Max SI
Peak Clad Temp., °F	1942	2014	1956	2163
Peak Clad Location, ft	7.00	5.75	7.00	6.00
Local Zr/H_2O Reaction (Max),%	2.85	5.65	3.84	9.65
Local Zr/H_2O Location, ft	7.00	5.75	5.75	5.75
Total Zr/H_2O Reaction, %	<0.3	<0.3	<0.3	<0.3
Hot Rod Burst Time, sec	43.8	37.8	47.4	37.8
Hot Rod Burst Location, ft	6.00	5.75	5.75	5.75

Calculation	
Licensed Core Power (MWT) 102% of	3411
Peak Linear Power (kw/ft) 102% of	14.796
Peaking Factor (at License Rating)	2.10
Accumulator Water Volume (ft^3) per Accumulator	950

TABLE 14.3.1-6

LARGE BREAK
TIME SEQUENCE OF EVENTS

	Min SI DECLG C_D=0.8 (sec)	Min SI DECLG C_D=0.6 (sec)	Min SI DECLG C_D=0.4 (sec)	Max SI DECLG C_D=0.6 (sec)
START	0.00	0.00	0.00	0.00
Reactor Trip Signal	0.62	0.63	0.64	0.63
Safety Injection Signal	3.83	3.95	4.20	3.95
Accumulator Injection	12.90	15.50	20.80	15.50
End of Blowdown	29.68	30.43	38.49	30.43
Bottom of Core Recovery	40.66	43.29	52.64	42.47
Accumulator Empty	56.89	59.29	65.65	60.58
Pump Injection	28.83	28.95	29.20	28.95

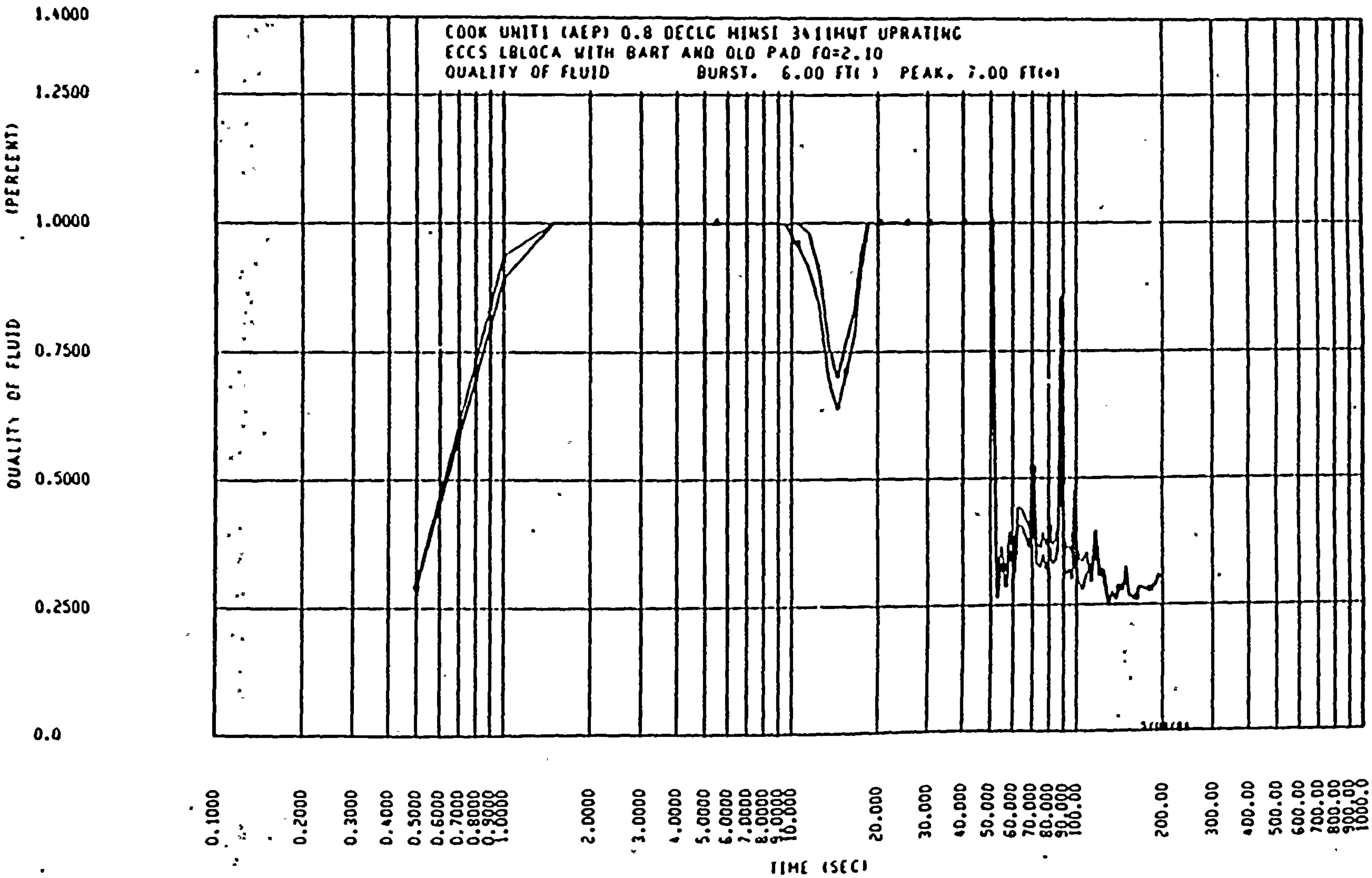


FIGURE 14.3.1-1 FLUID QUALITY, DECLG (CD=0.8) MIN SI

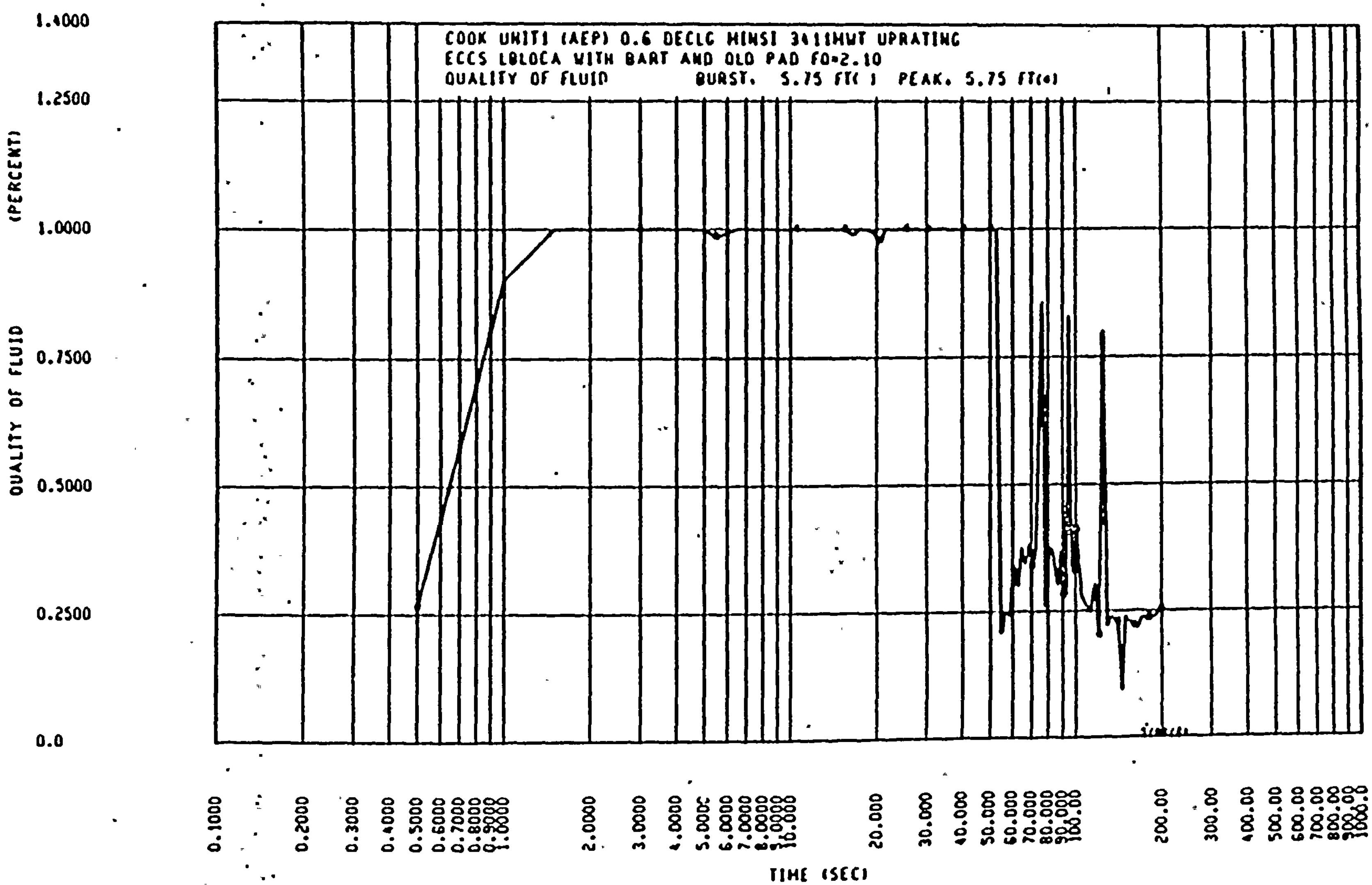


FIGURE 14.3.1-2 FLUID QUALITY, DECLG (CD=0.6) MIN SI

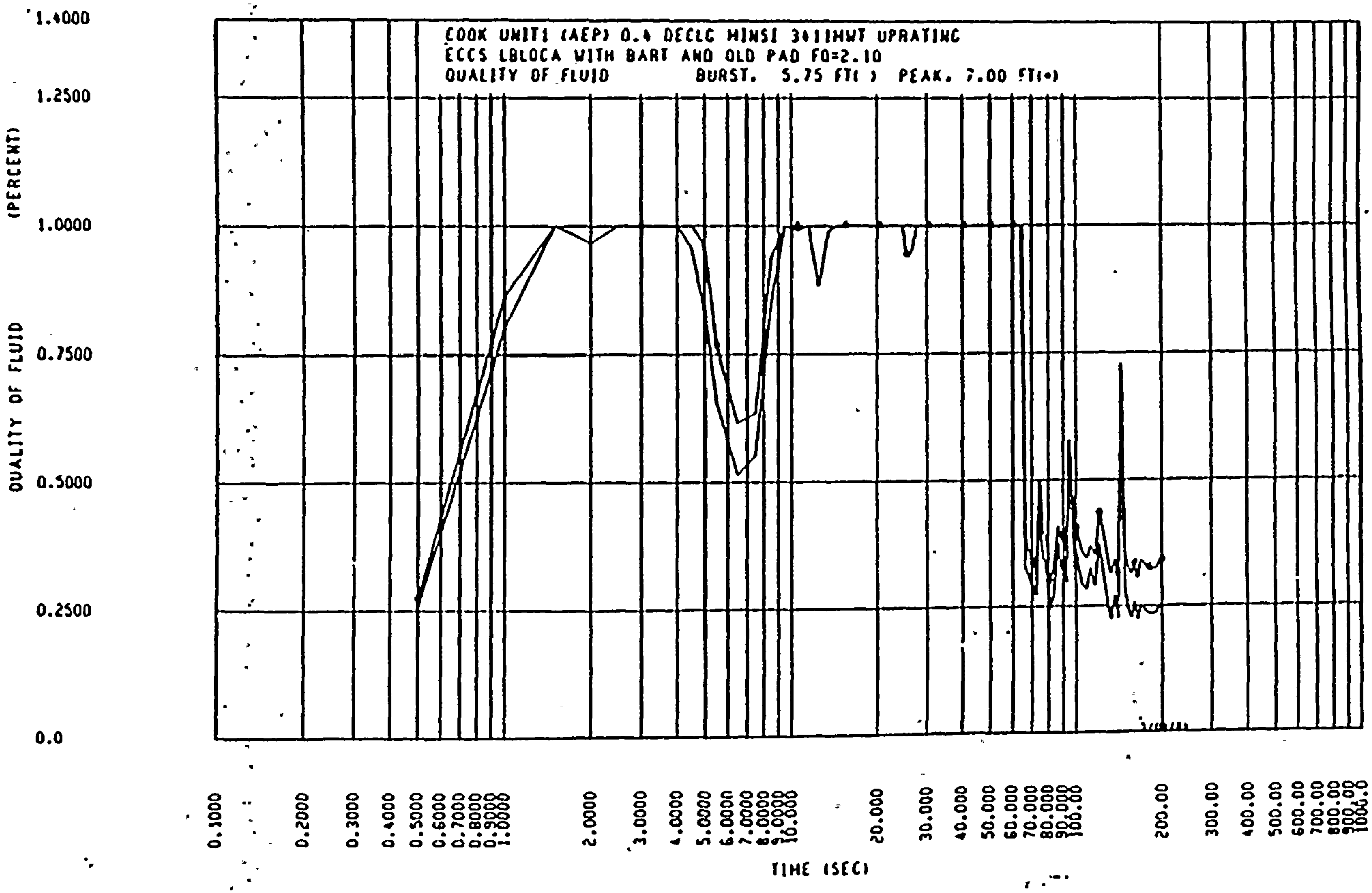


FIGURE 14.3.1-3 FLUID QUALITY, DECLG (CD = 0.4) MINSI

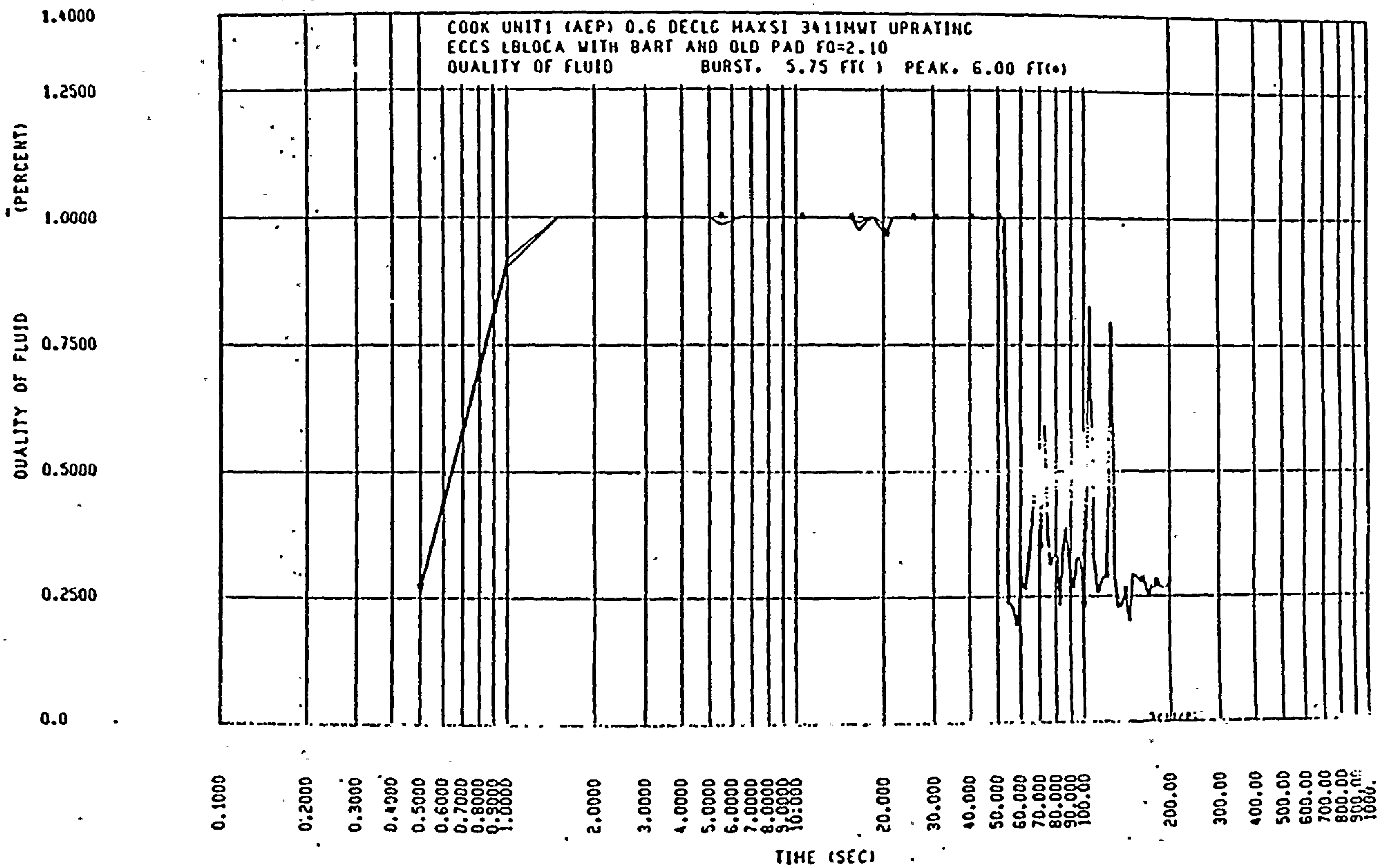


FIGURE 14.3.1-4 FLUID QUALITY DECLG (CD=0.6) MAX SI

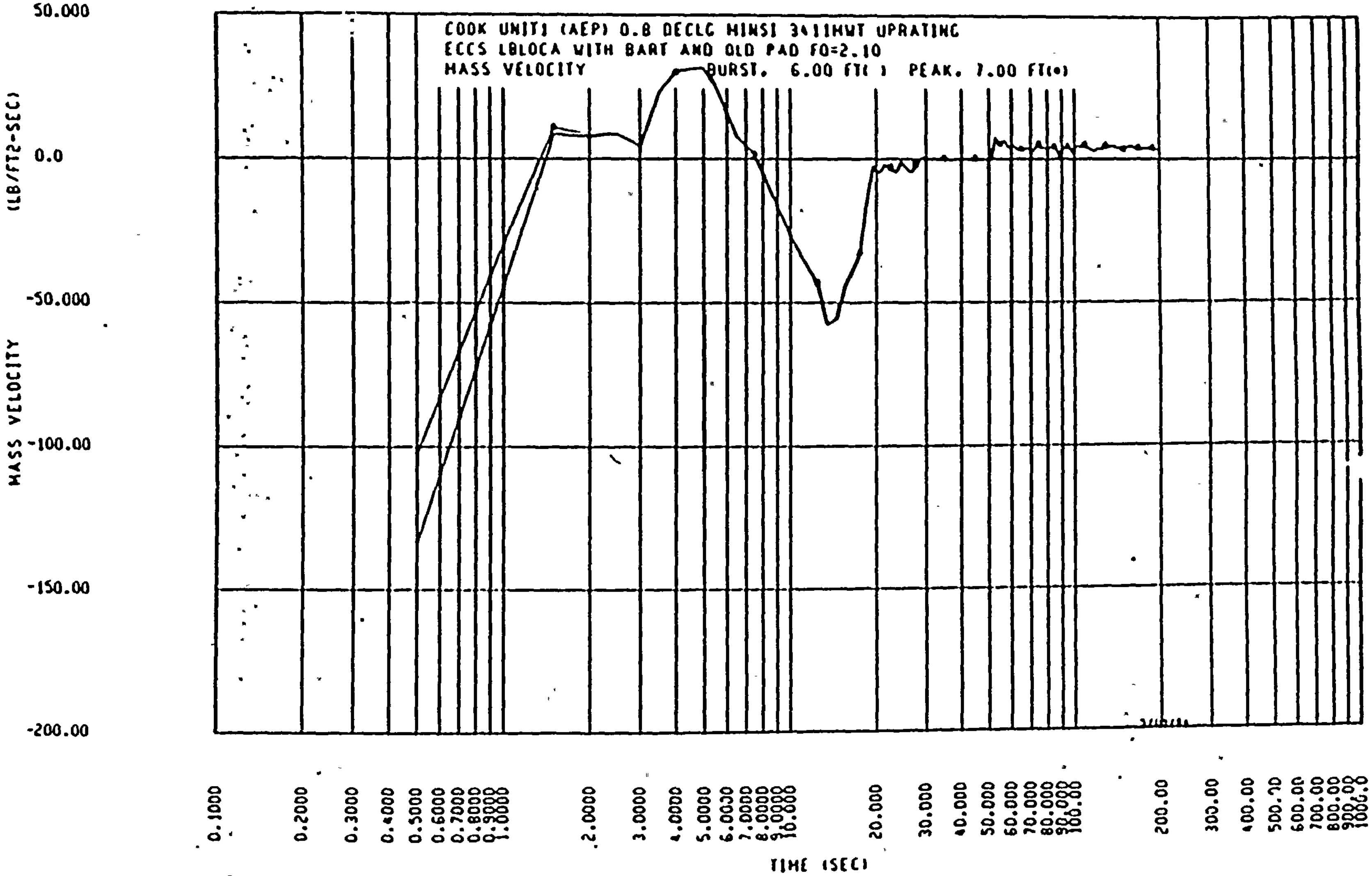


FIGURE 14.3.1-5 MASS VELOCITY
DECLG (CD=0.8) MIN SI

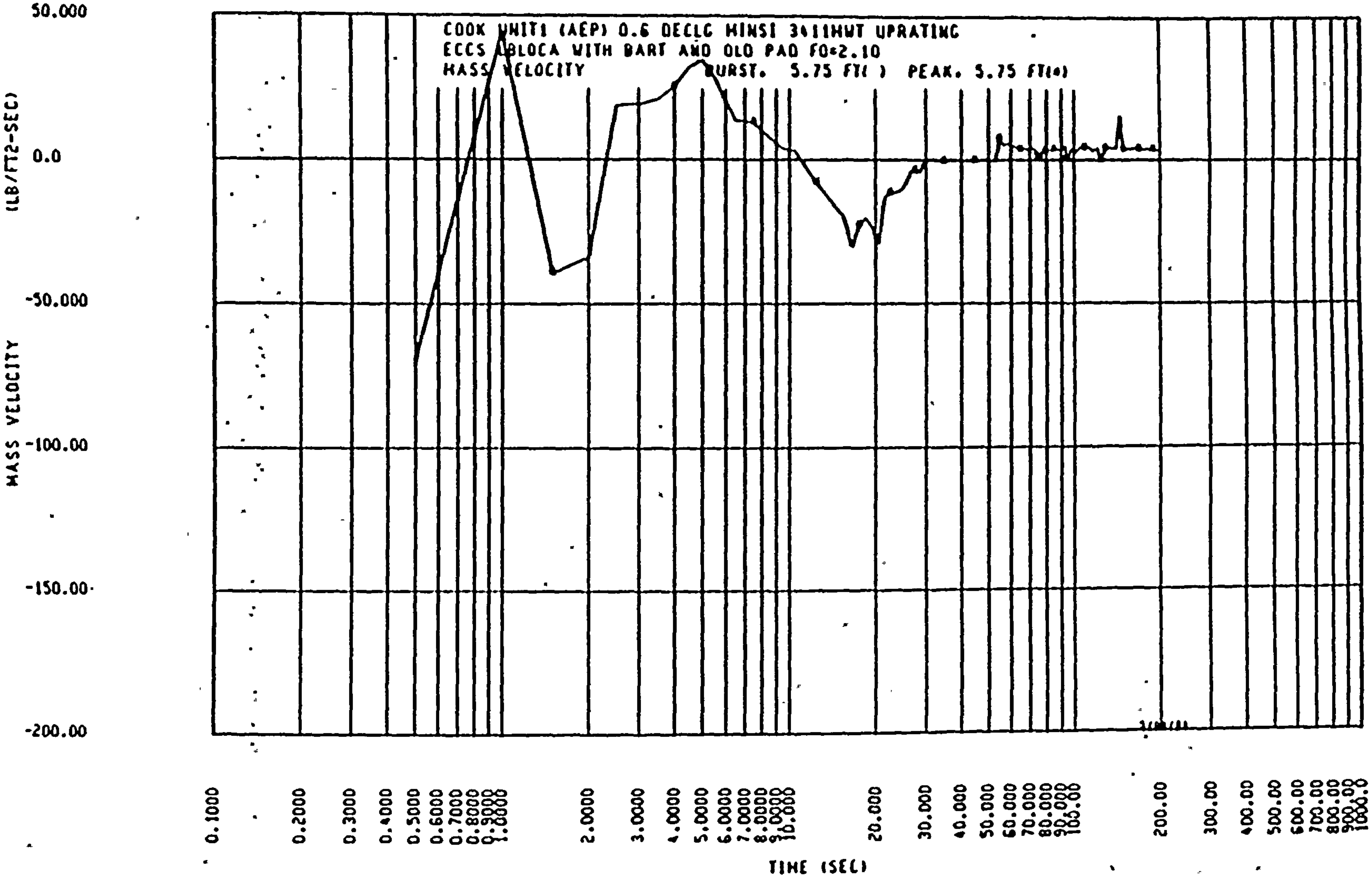


FIGURE 14.3.1-6 MASS VELOCITY
DECLG (CD=0.6) MIN SI

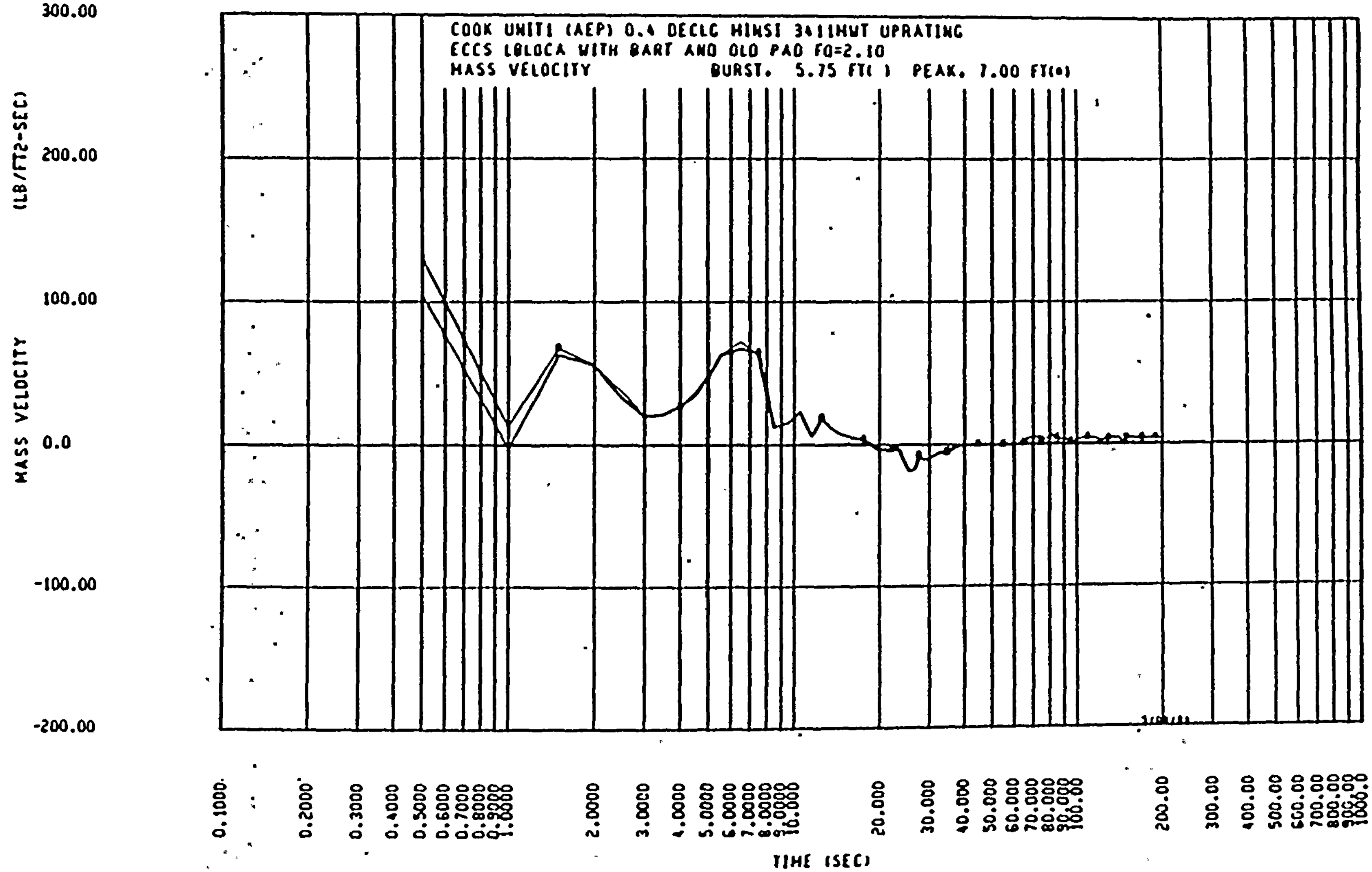


FIGURE 14.3.1-7 MASS VELOCITY
DECLG (CD=0.4) MIN SI

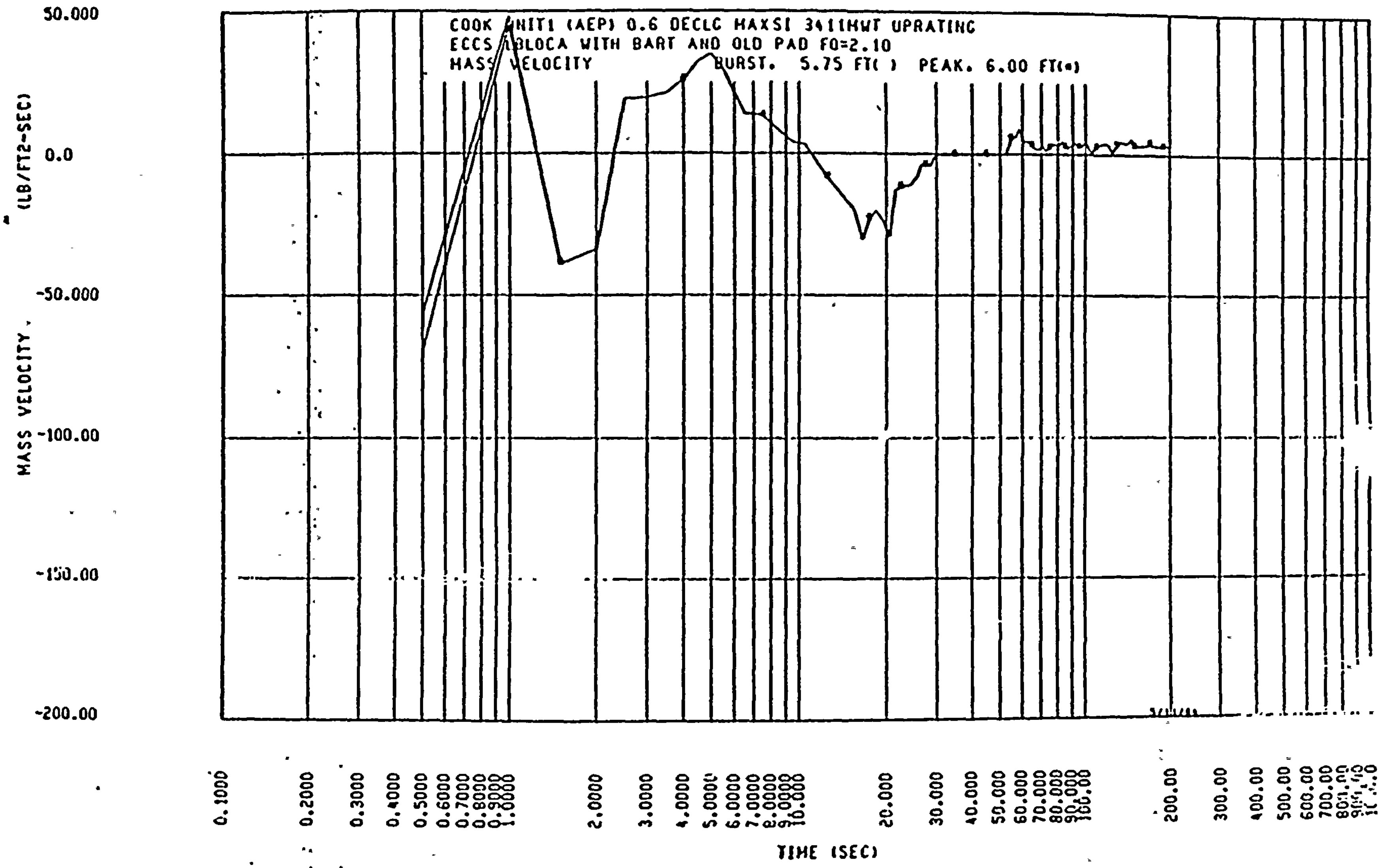


FIGURE 14.3.1-8 MASS VELOCITY
DECLG (CD = 0.6) MAX SI

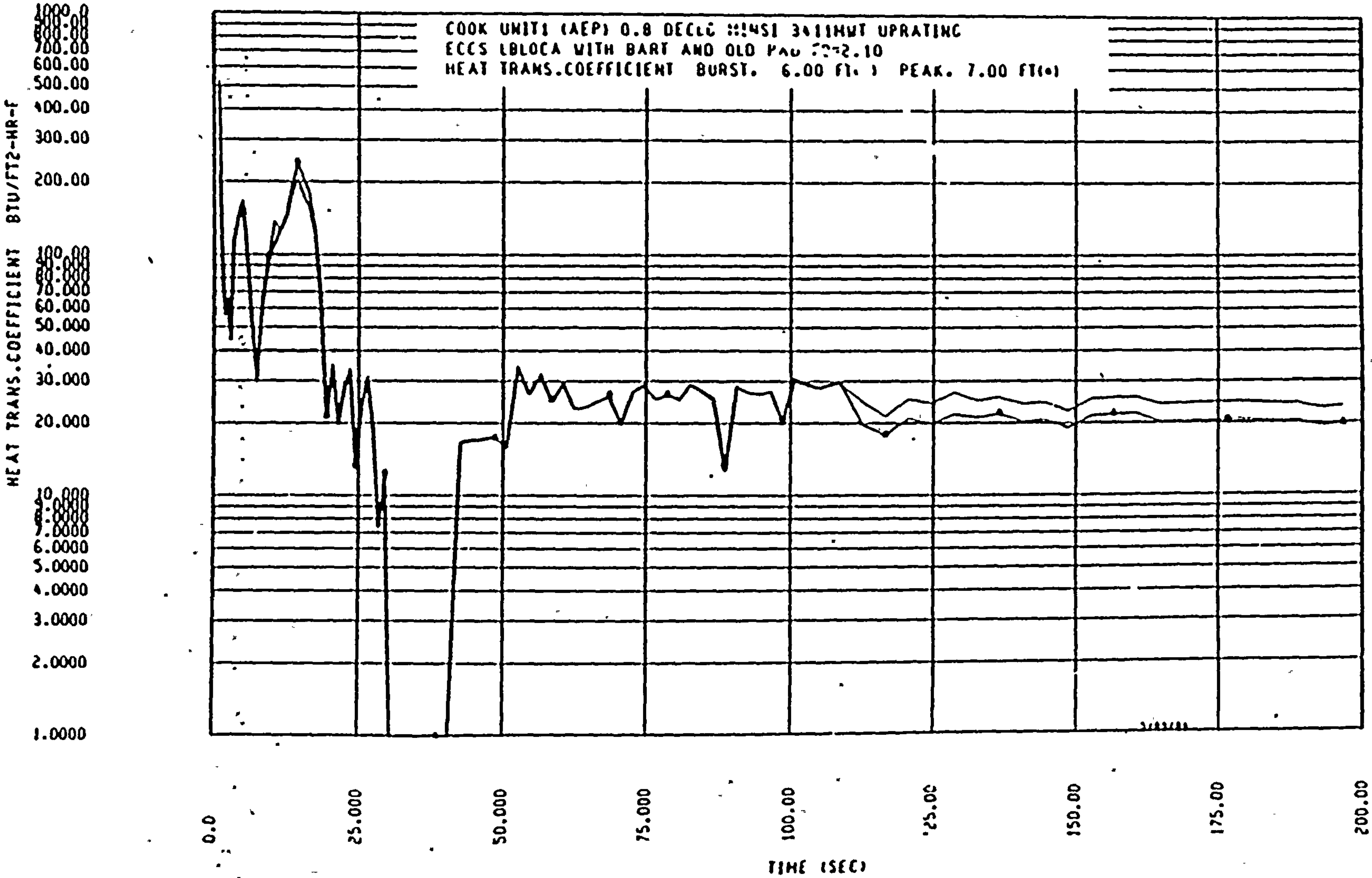


FIGURE 14.3.1-9 HEAT TRANSFER COEFFICIENT DECLG (CD=0.8) MIN SI

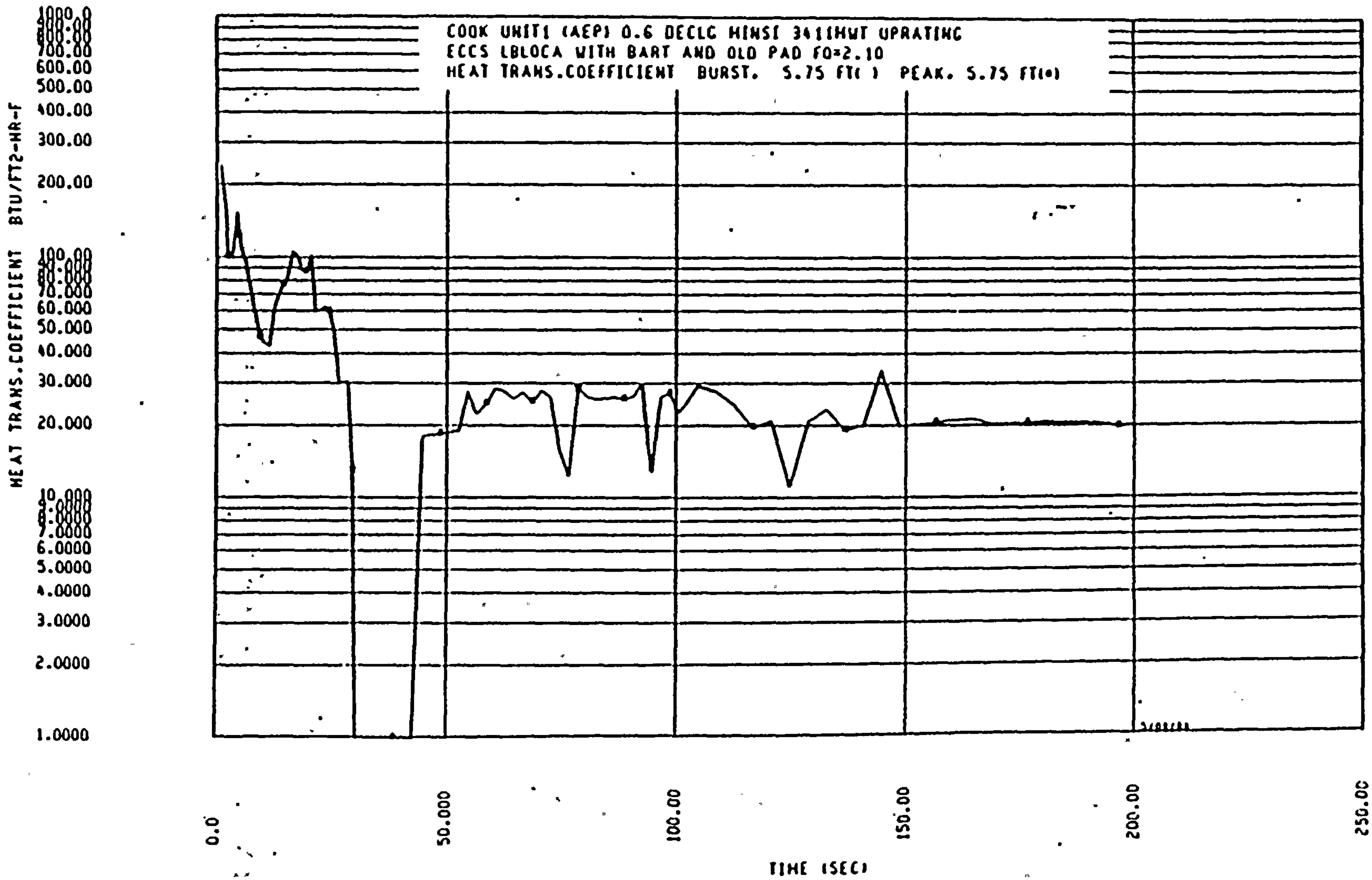


FIGURE 14.3.1-10 HEAT TRANSFER COEFFICIENT
DECLG (CD = 0.6) MIN SI

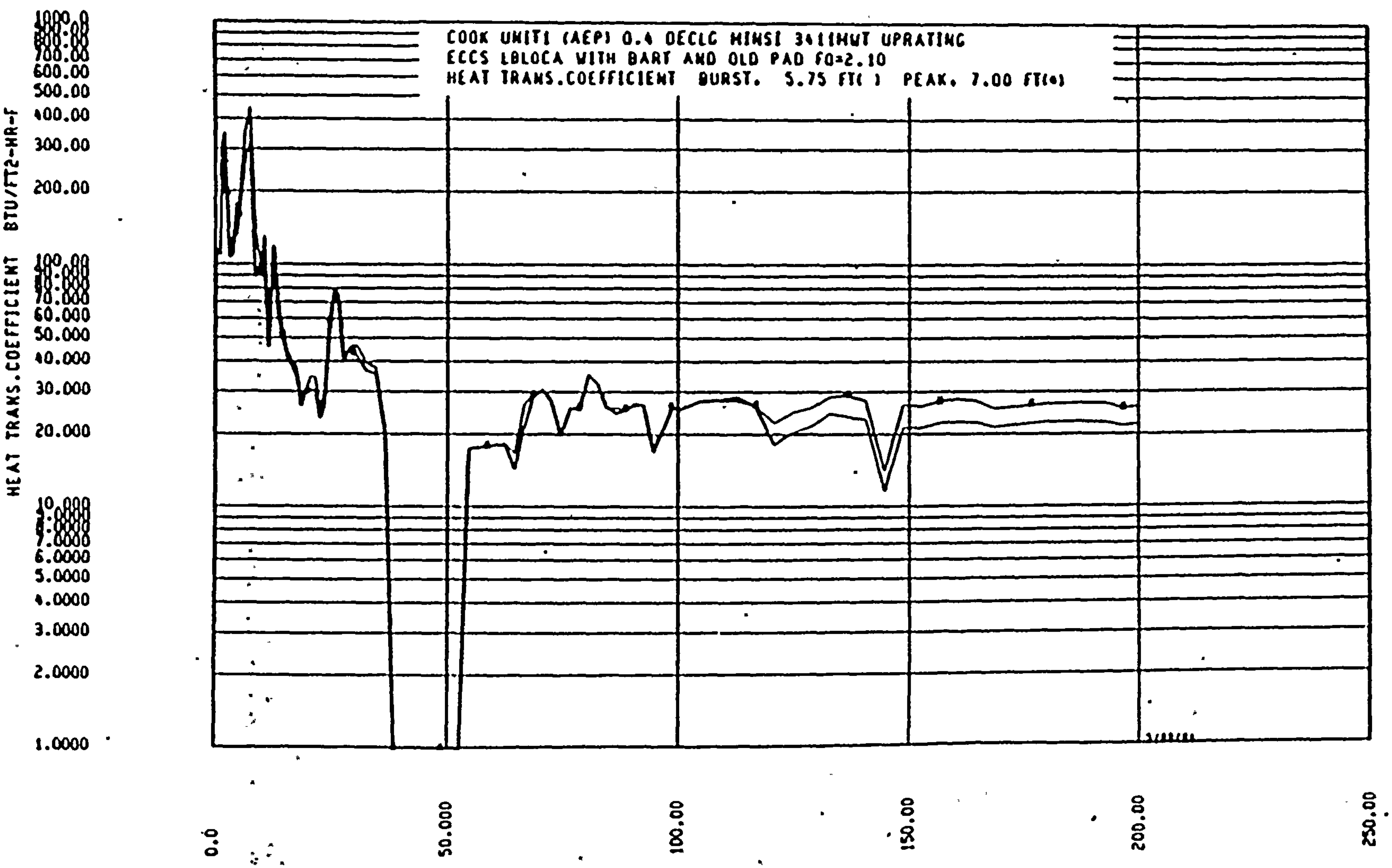


FIGURE 14.3.1-11 HEAT TRANSFER COEFFICIENT
DECLG(CD=0.4) MIN SI

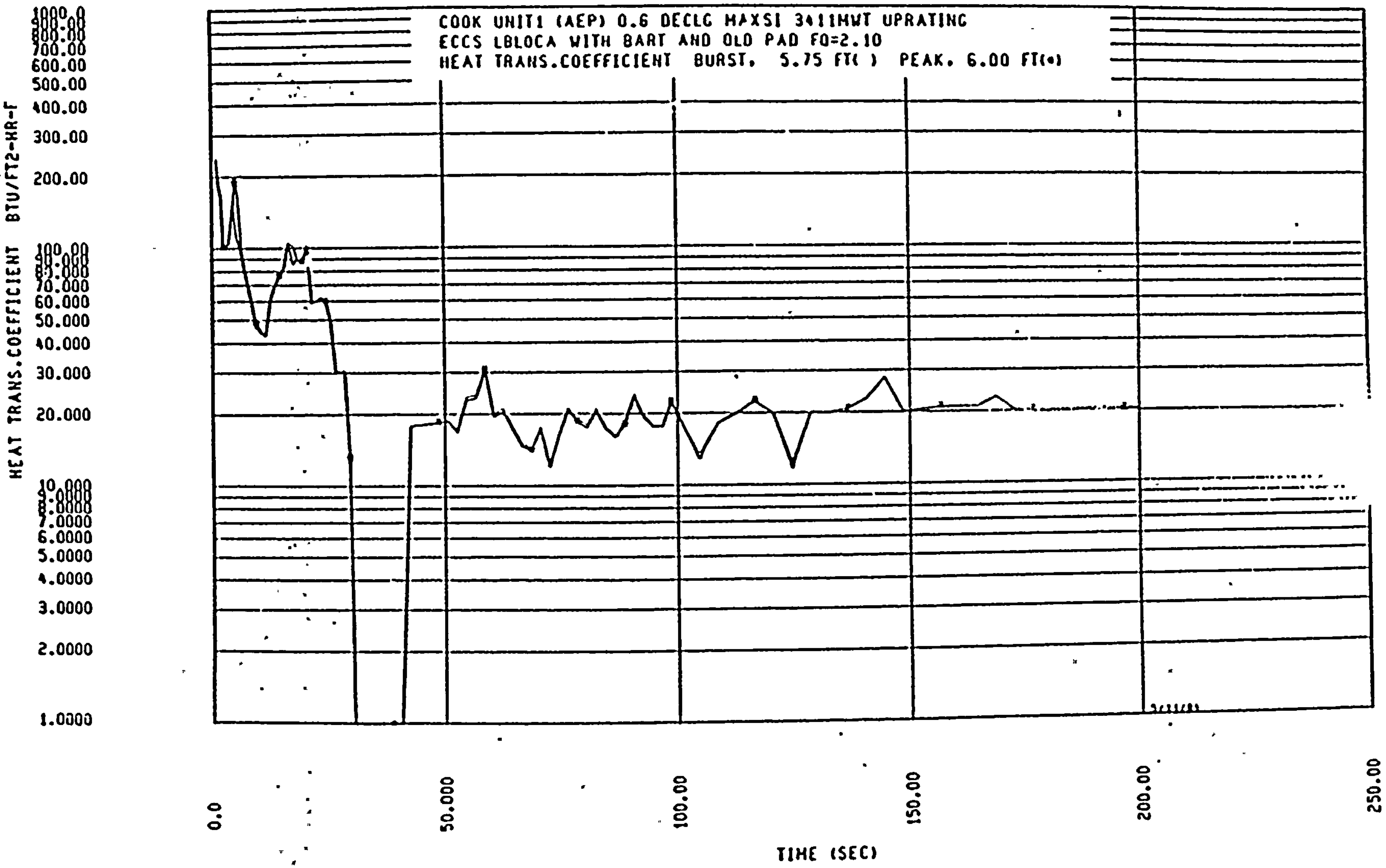


FIGURE 14.3.1-12 HEAT TRANSFER COEFFICIENT DECLG (CD=0.6) MAX SI

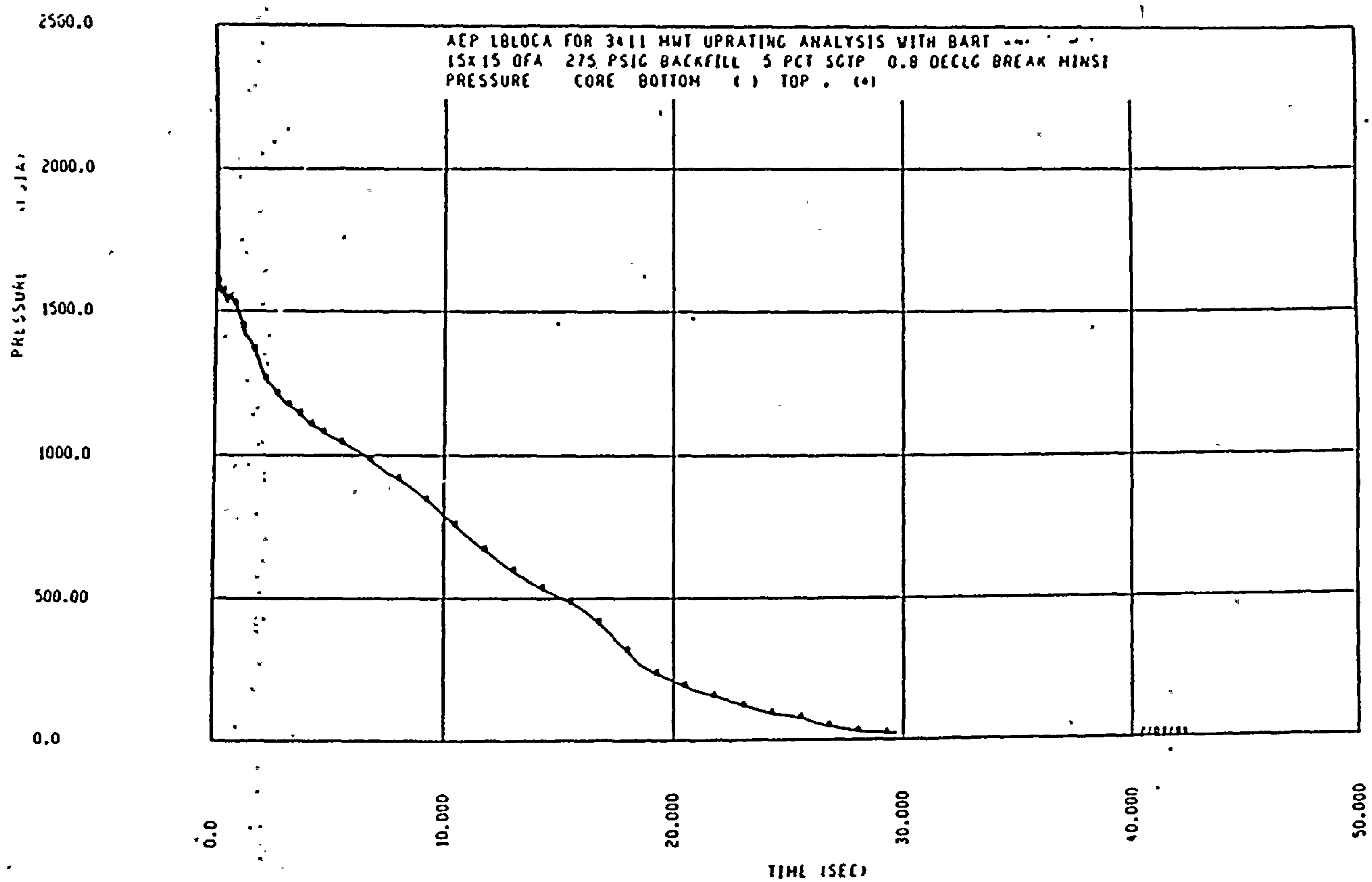


FIGURE 14.3.1-13 CORE PRESSURE
DECLG (CD=0.8) MIN SI

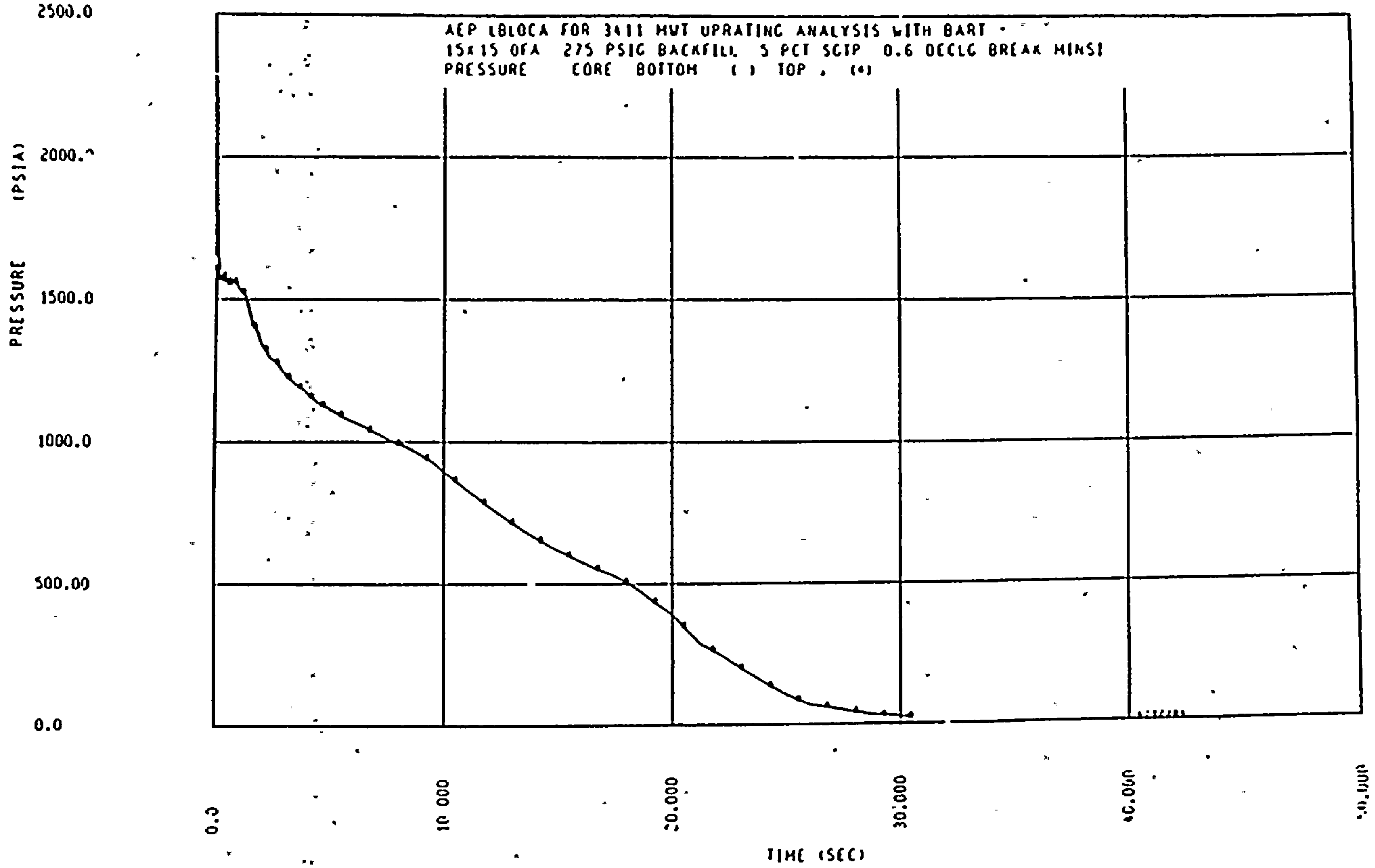


FIGURE 14.3.1-14 CORE PRESSURE
DECLG (CD=0.6) MIN SI

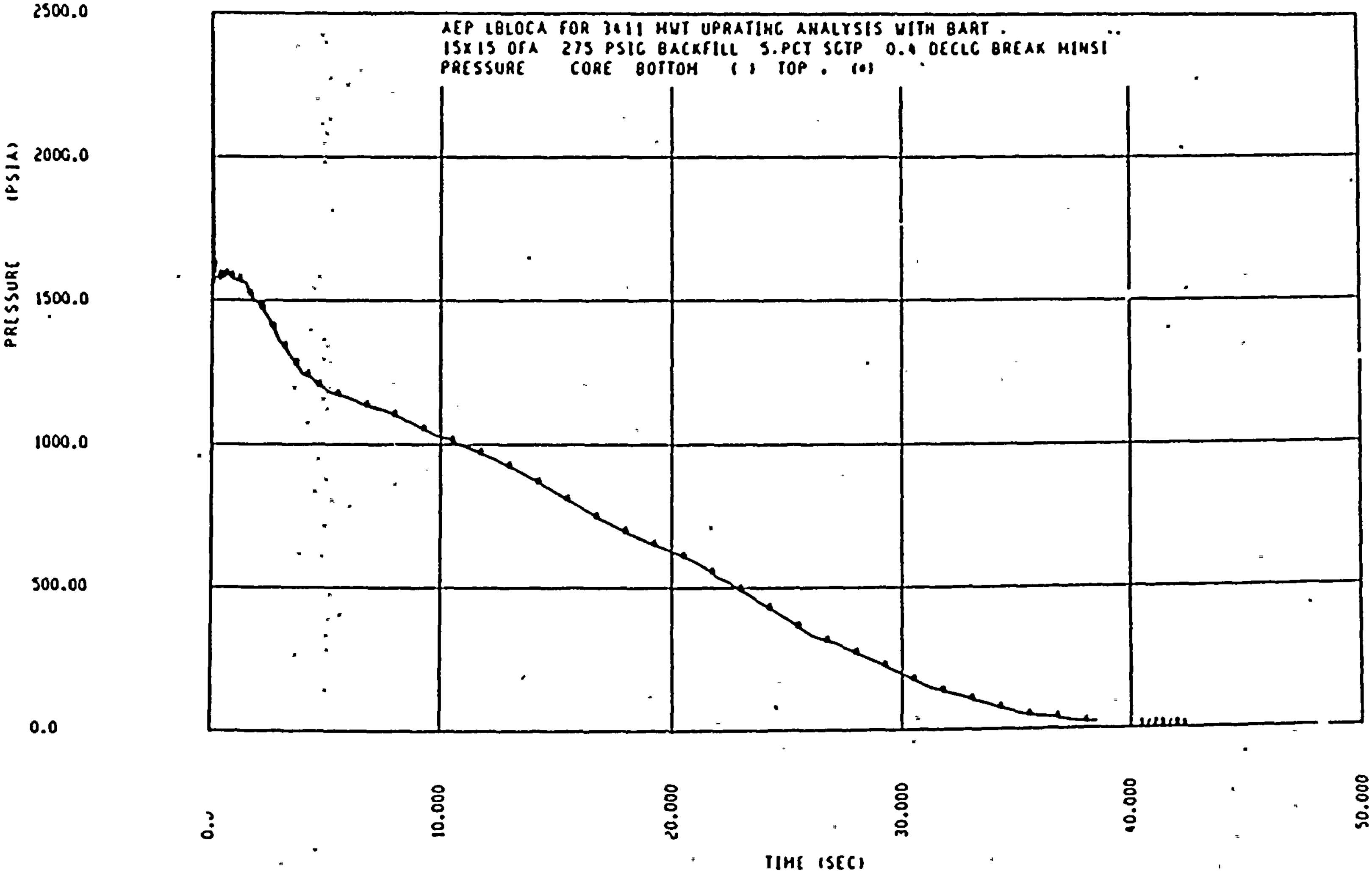


FIGURE 14.3.1-15 CORE PRESSURE
DECLG (CD=0.4) MIN SI

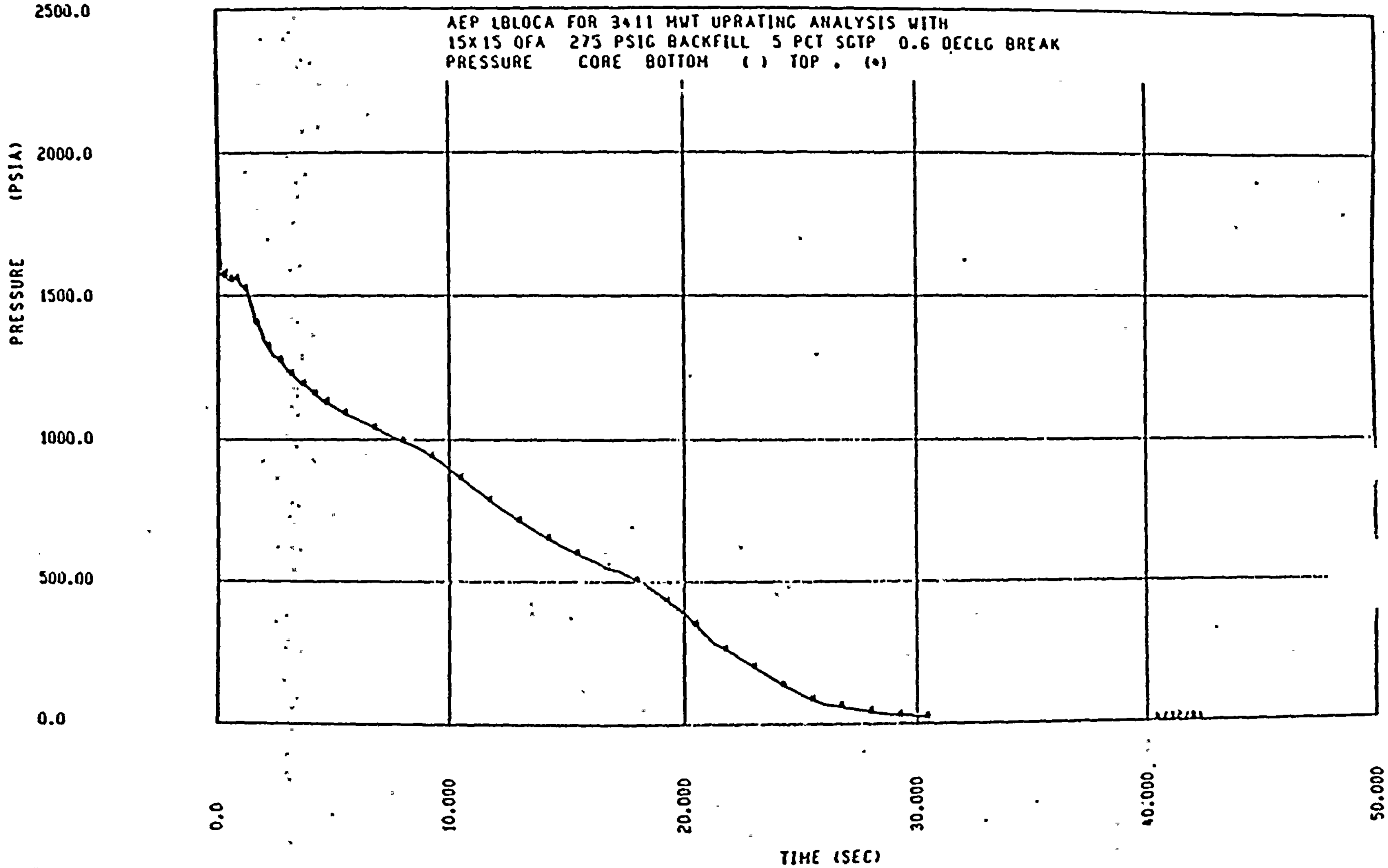


FIGURE 14.3.1-16 CORE PRESSURE
DECLG (CD=0.6) MAX SI

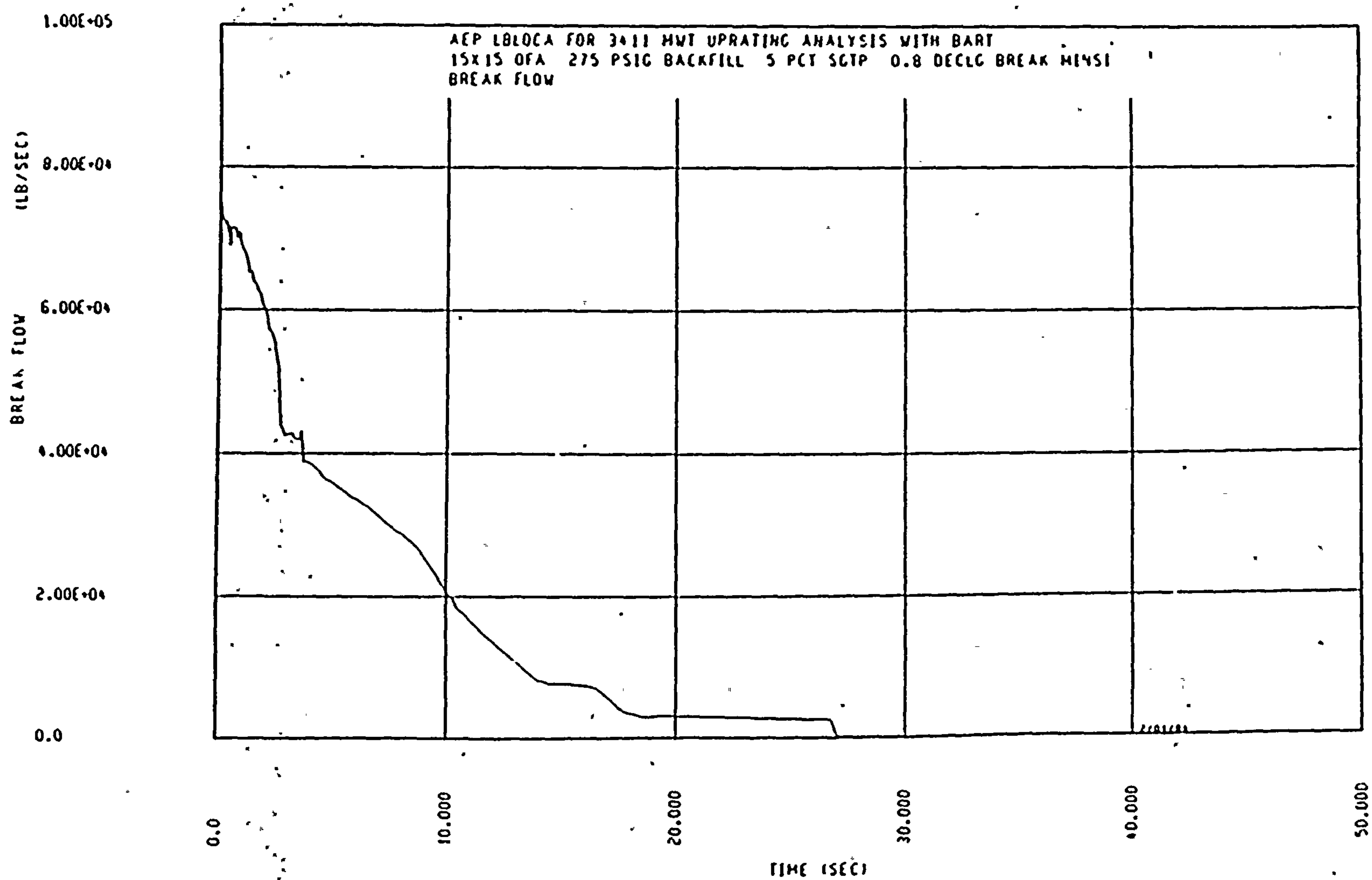


FIGURE 14.3.1-17 BREAK FLOW RATE
DECLG (CD=0.8) MIN SI

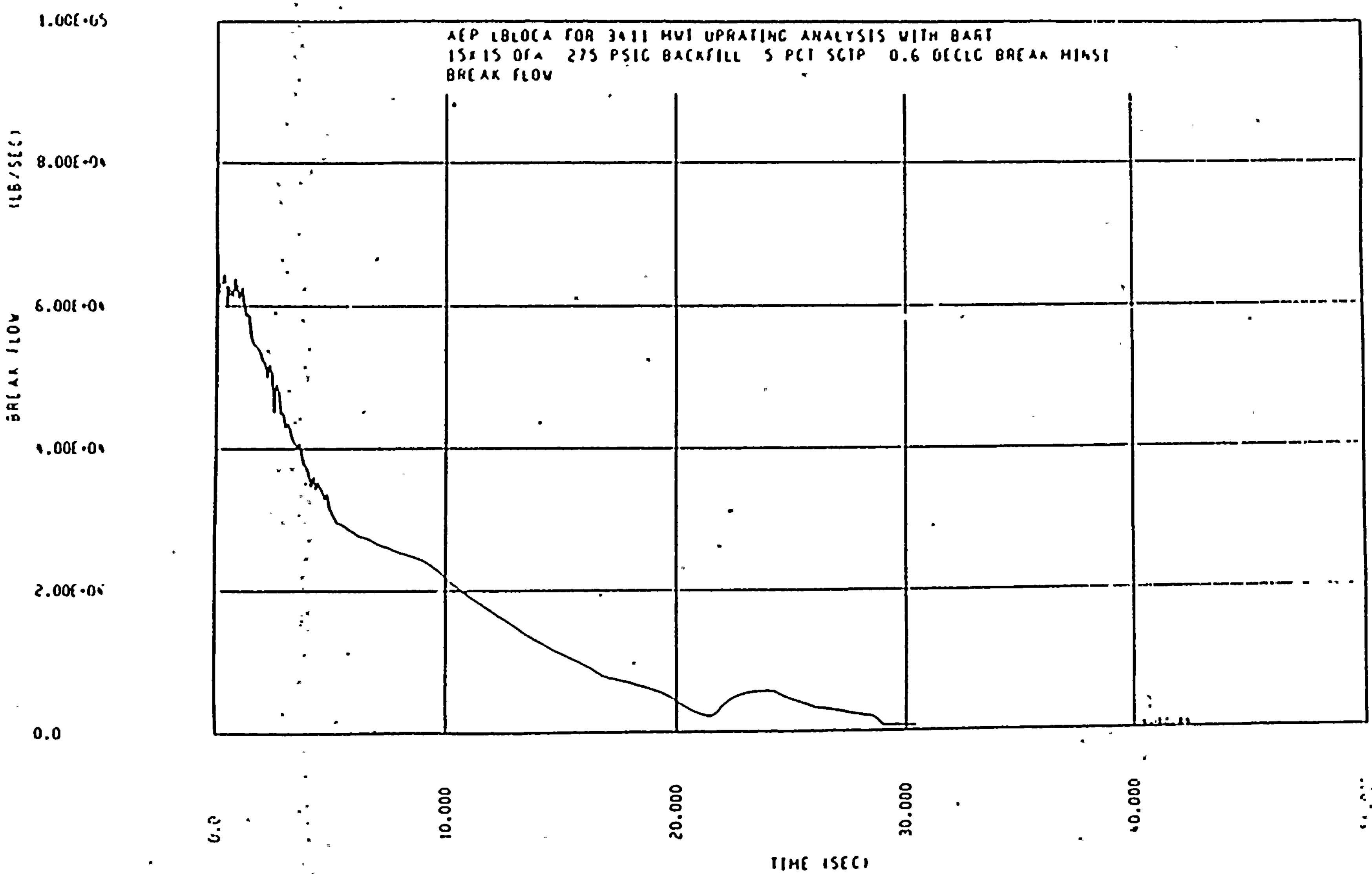


FIGURE 14.3.1-18 BREAK FLOW RATE
DECLG (CD=0.6) MIN SI

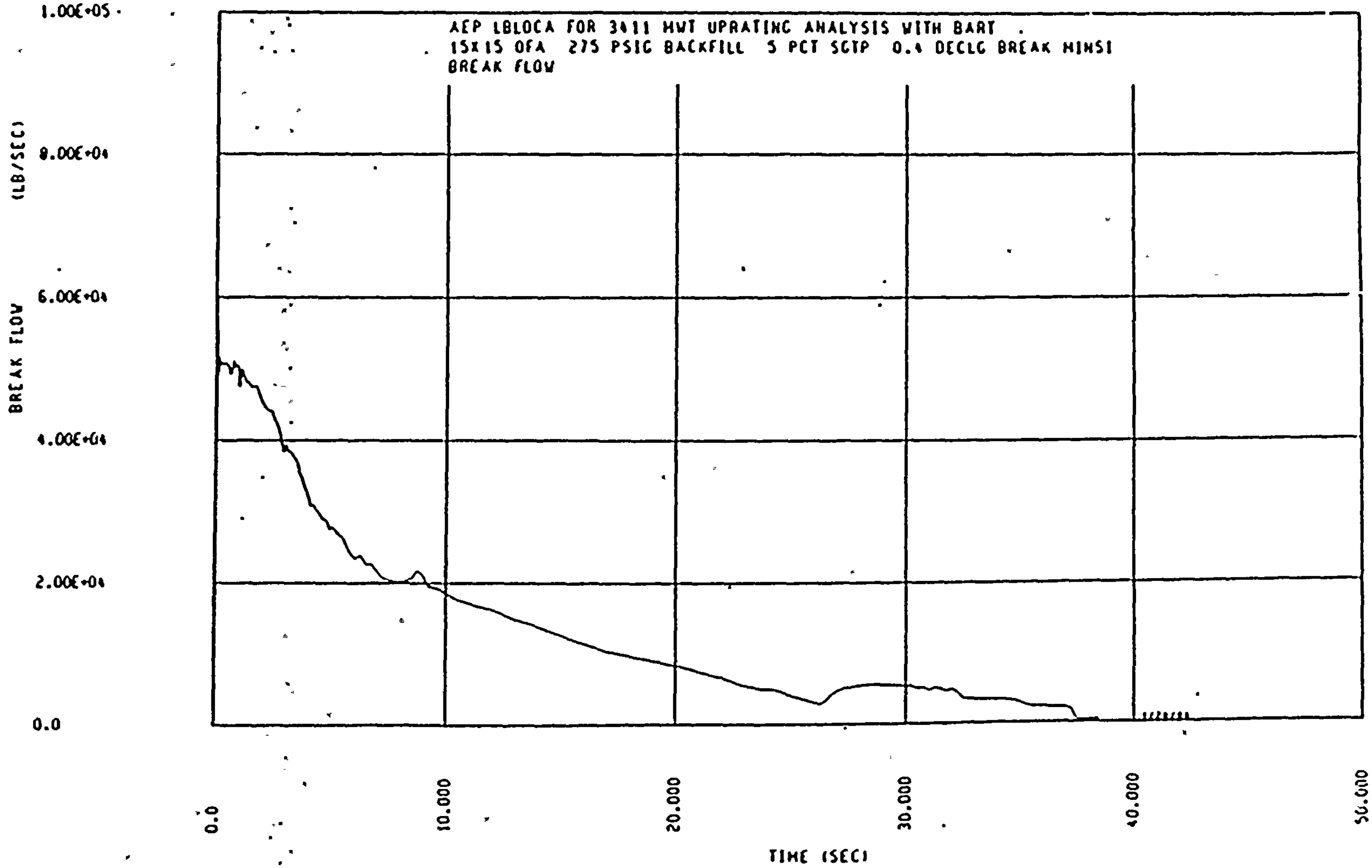


FIGURE 14.3.1-19 BREAK FLOW RATE
DECLG (CD=0.4) MIN SI

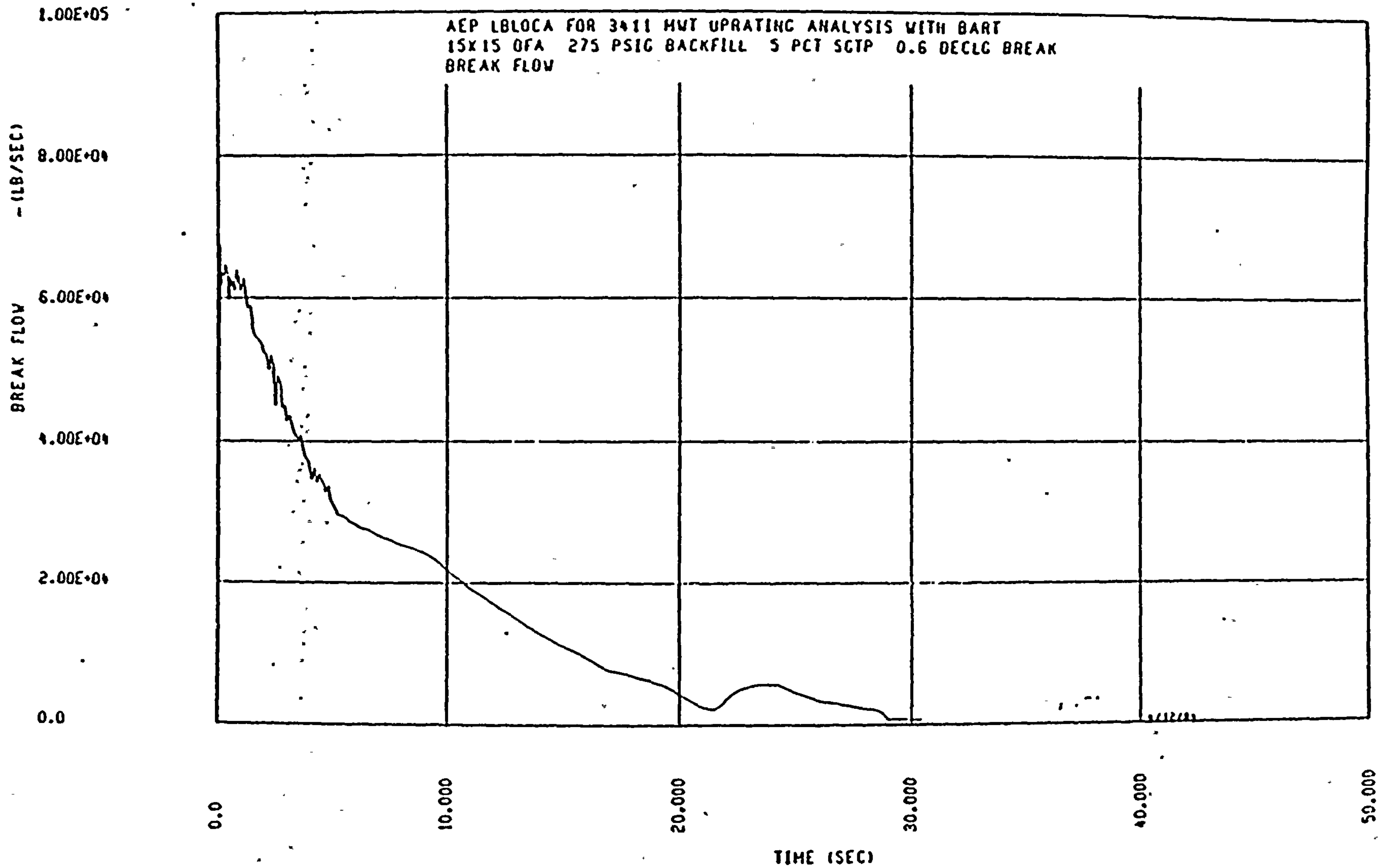


FIGURE 14.3.1-20 BREAK FLOW RATE
DECLG (CD=0.6) MAX SI

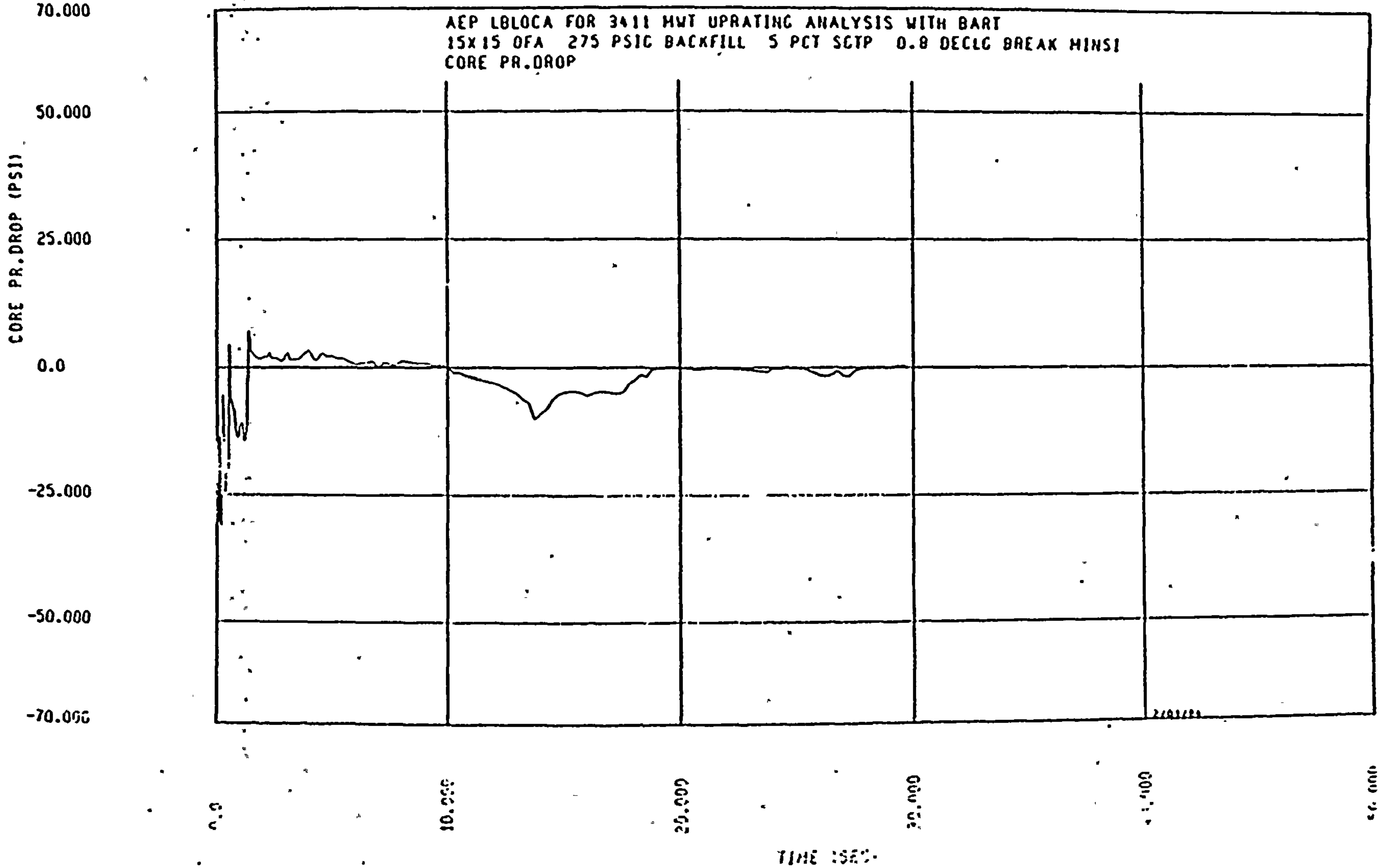


FIGURE 14.3.1-21 CORE PRESSURE DROP
DECLG(CD=0.8) MIN SI

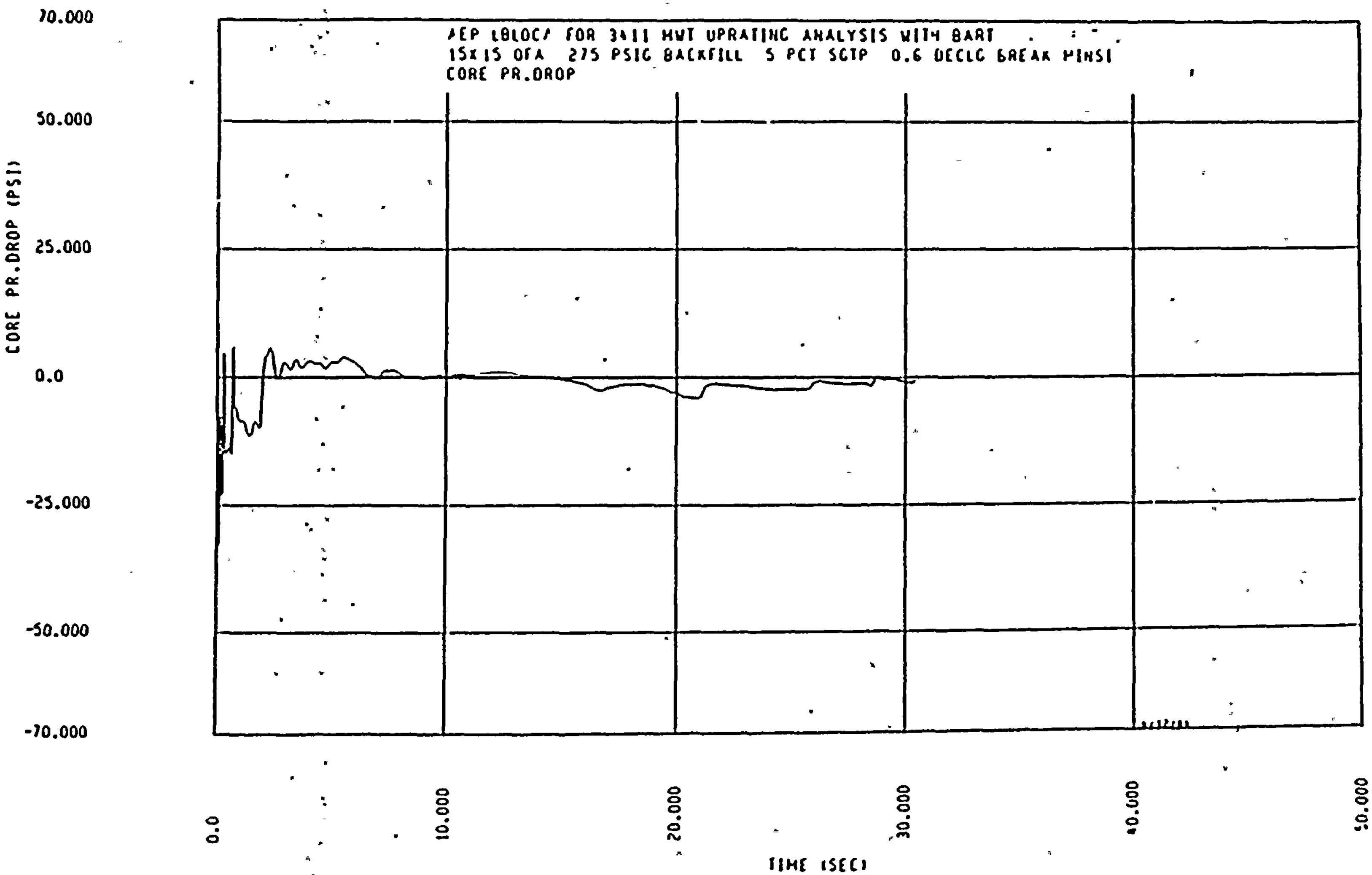


FIGURE 14.3-1-22 CORE PRESSURE DROP
DECLG (CD=0.6) MIN SI

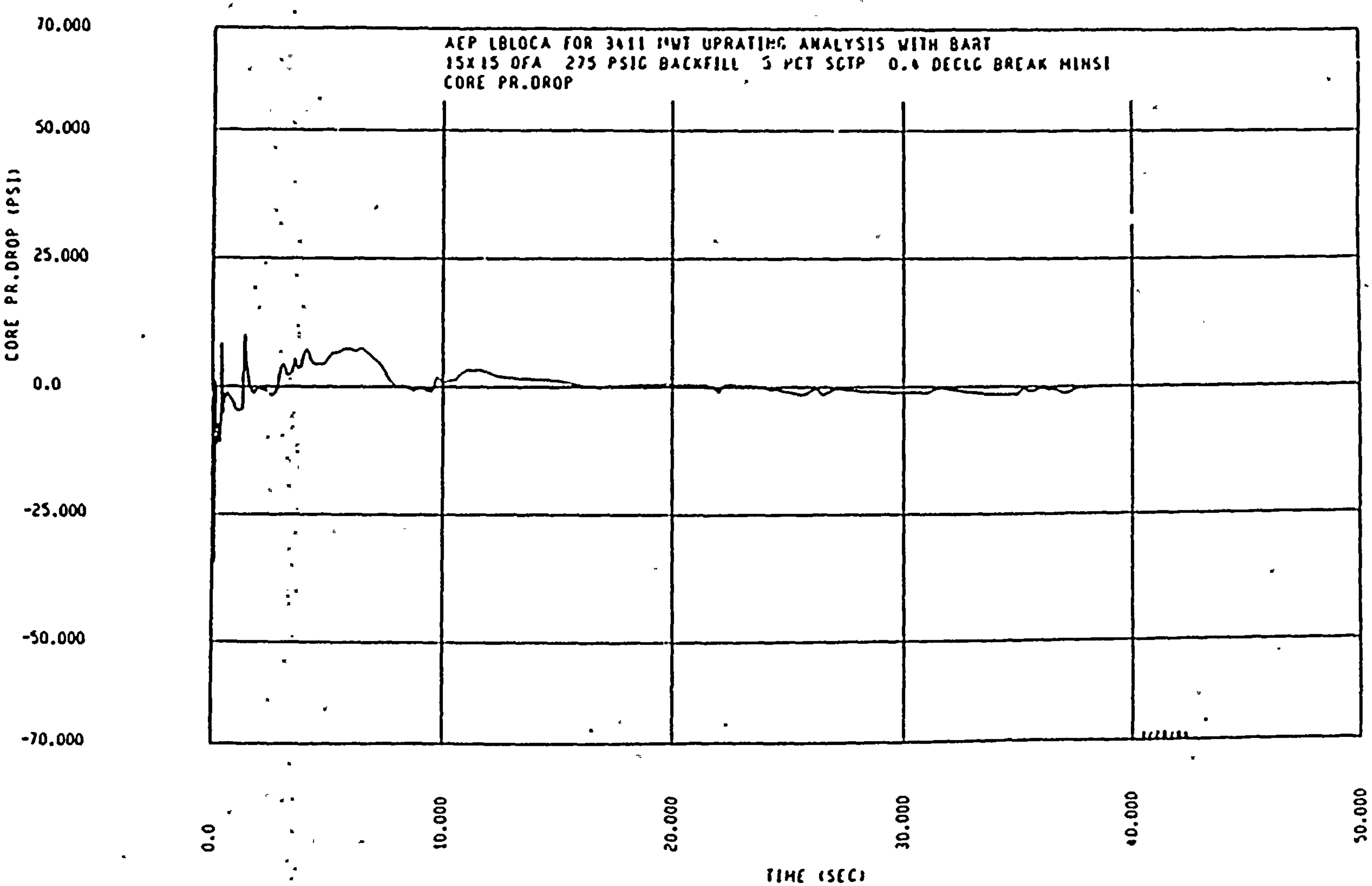


FIGURE 14.3.1-23 CORE PRESSURE DROP
DECLG (CD=0.4) MIN SI

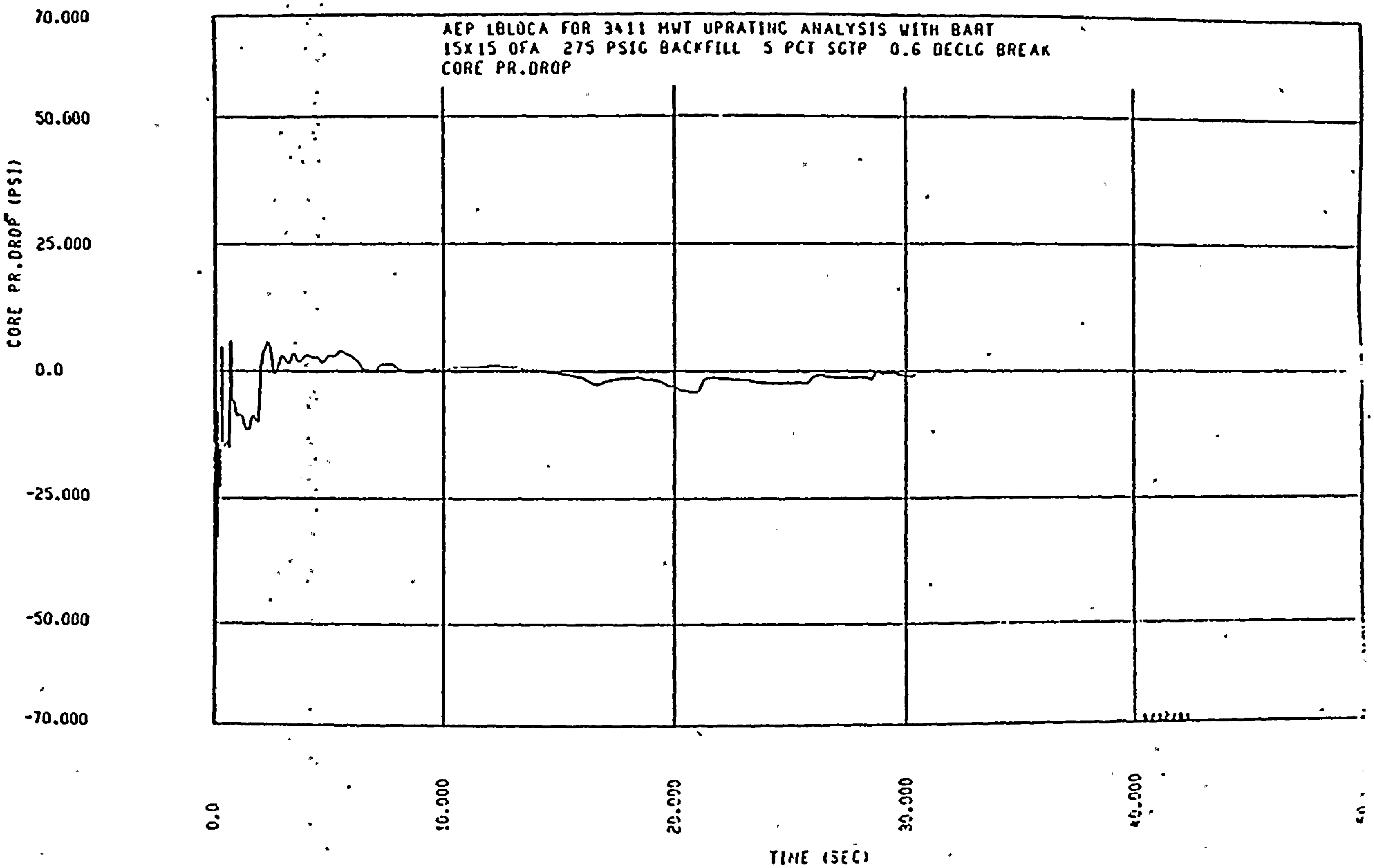


FIGURE 14.3.1-24 CORE PRESSURE DROP
DECLG(CD=0.6) MAX SI

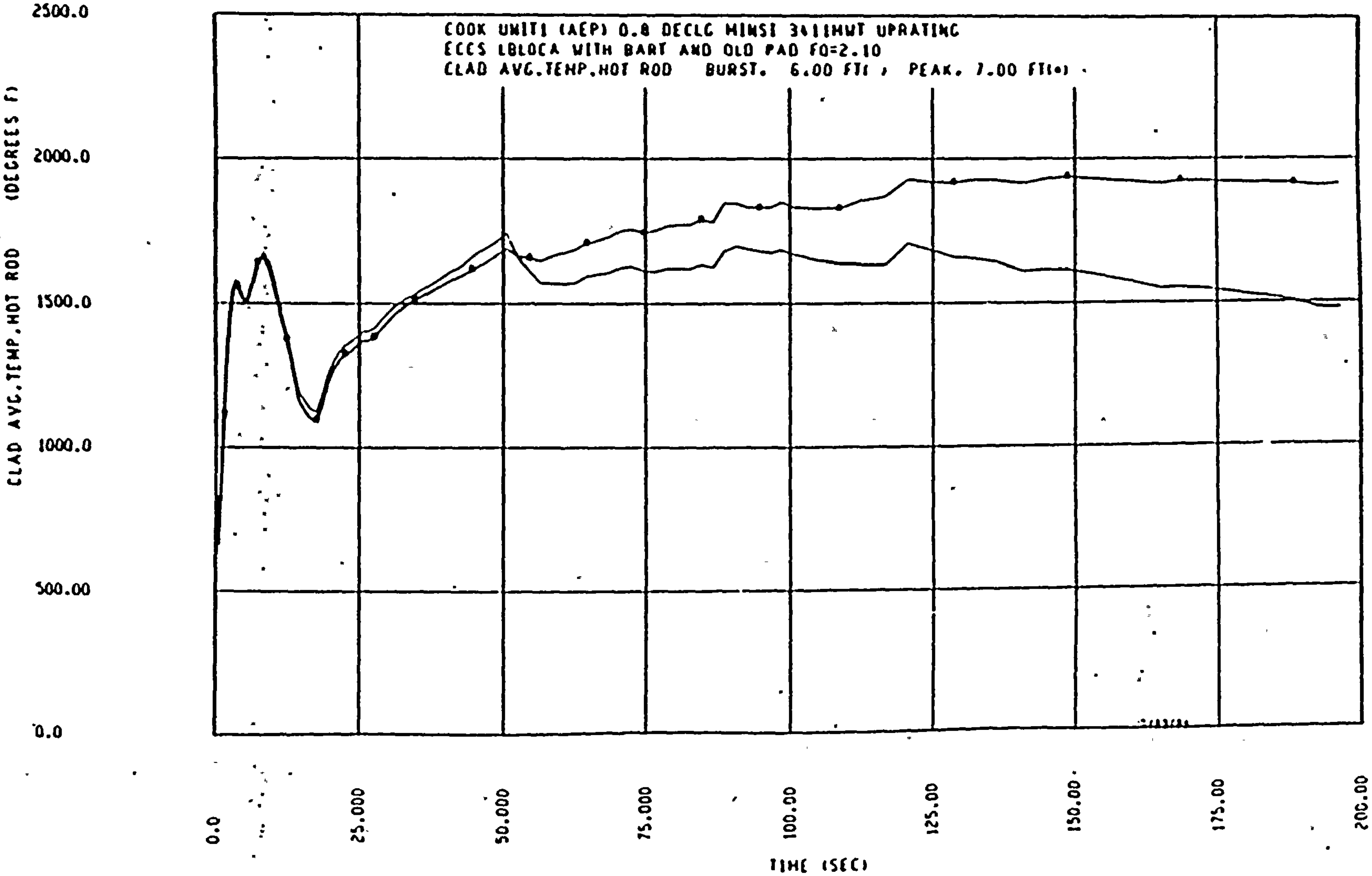


FIGURE 14.3.1-25 PEAK CLAD TEMPERATURE

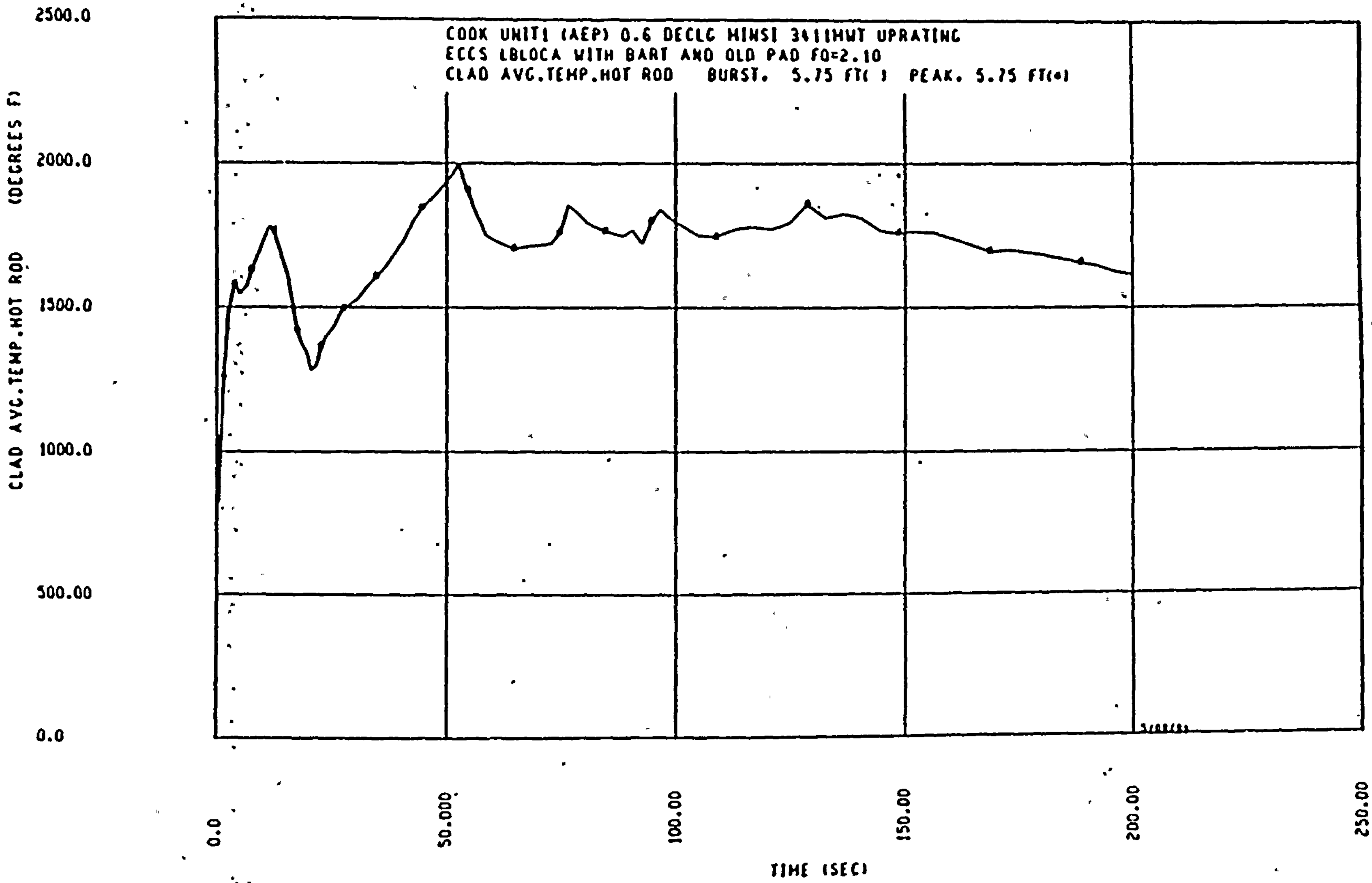


FIGURE 14.3.1-26 PEAK CLAD TEMPERATURE
DECLG (CD=0.6) MIN SI

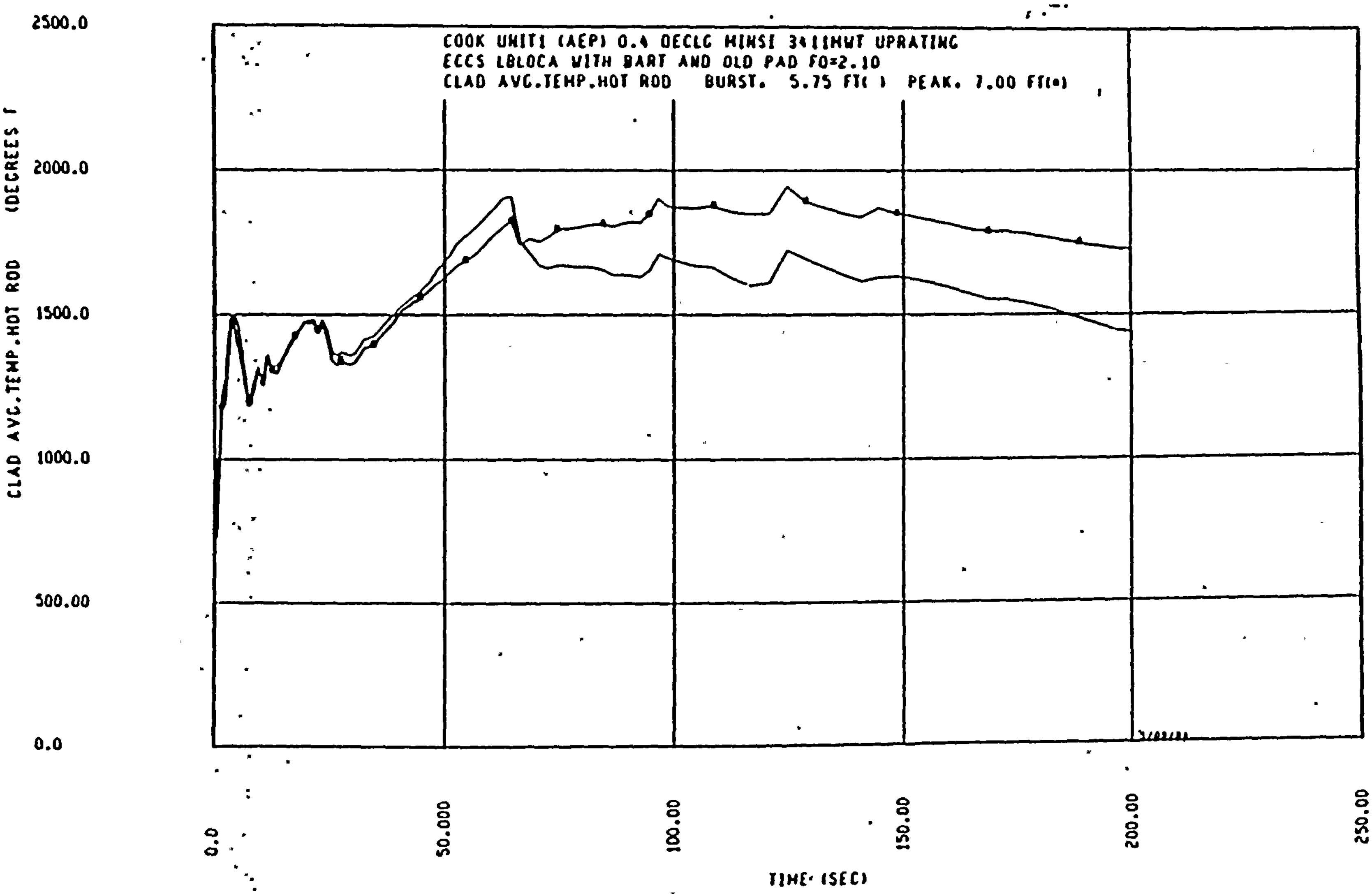


FIGURE 14.3.1-27 PEAK CLAD TEMPERATURE
DECLG (CD=0.4) MIN SI

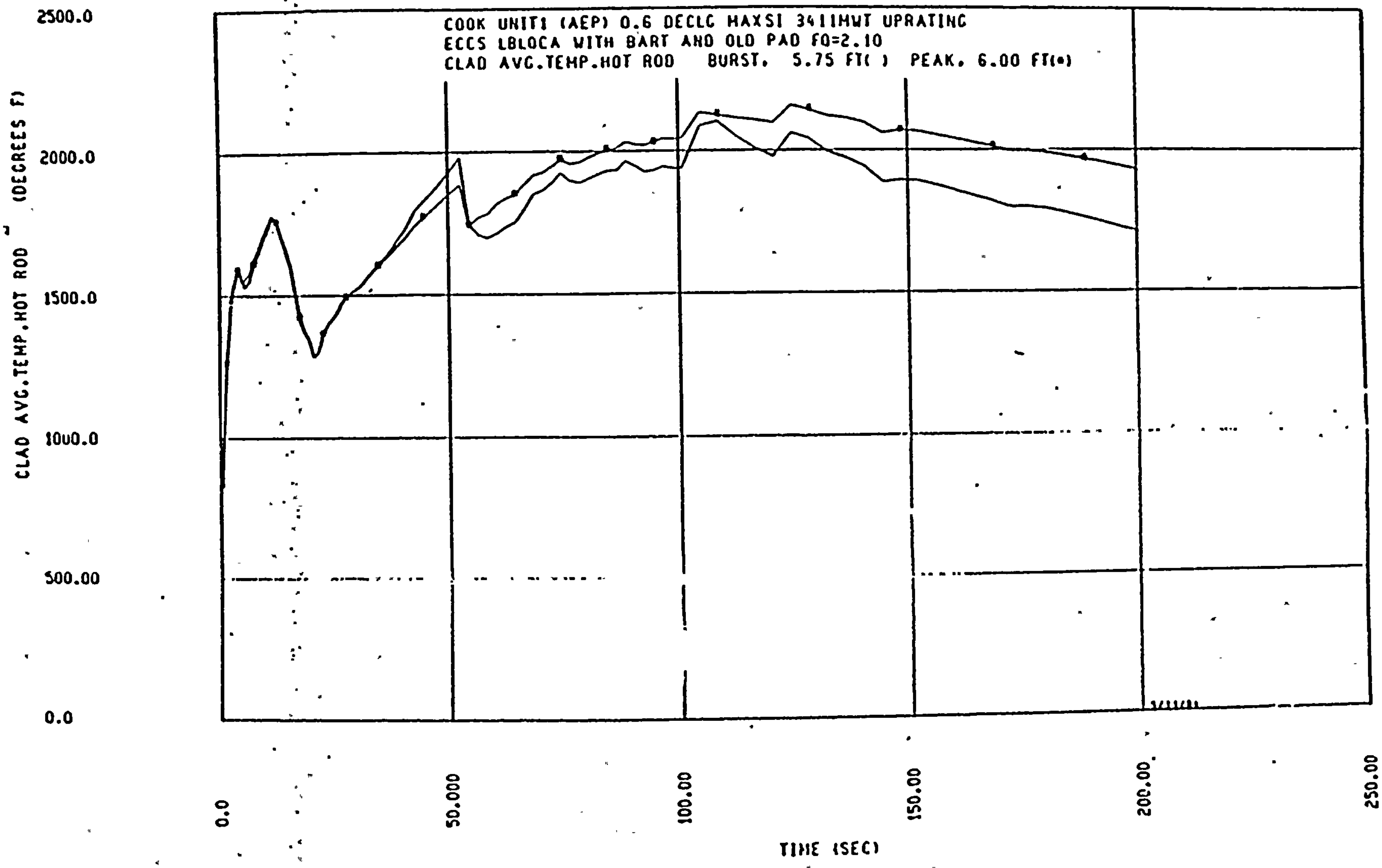


FIGURE 14.3.1-28 PEAK CLAD TEMPERATURE
DECLG (CD=0.6) MAX SI

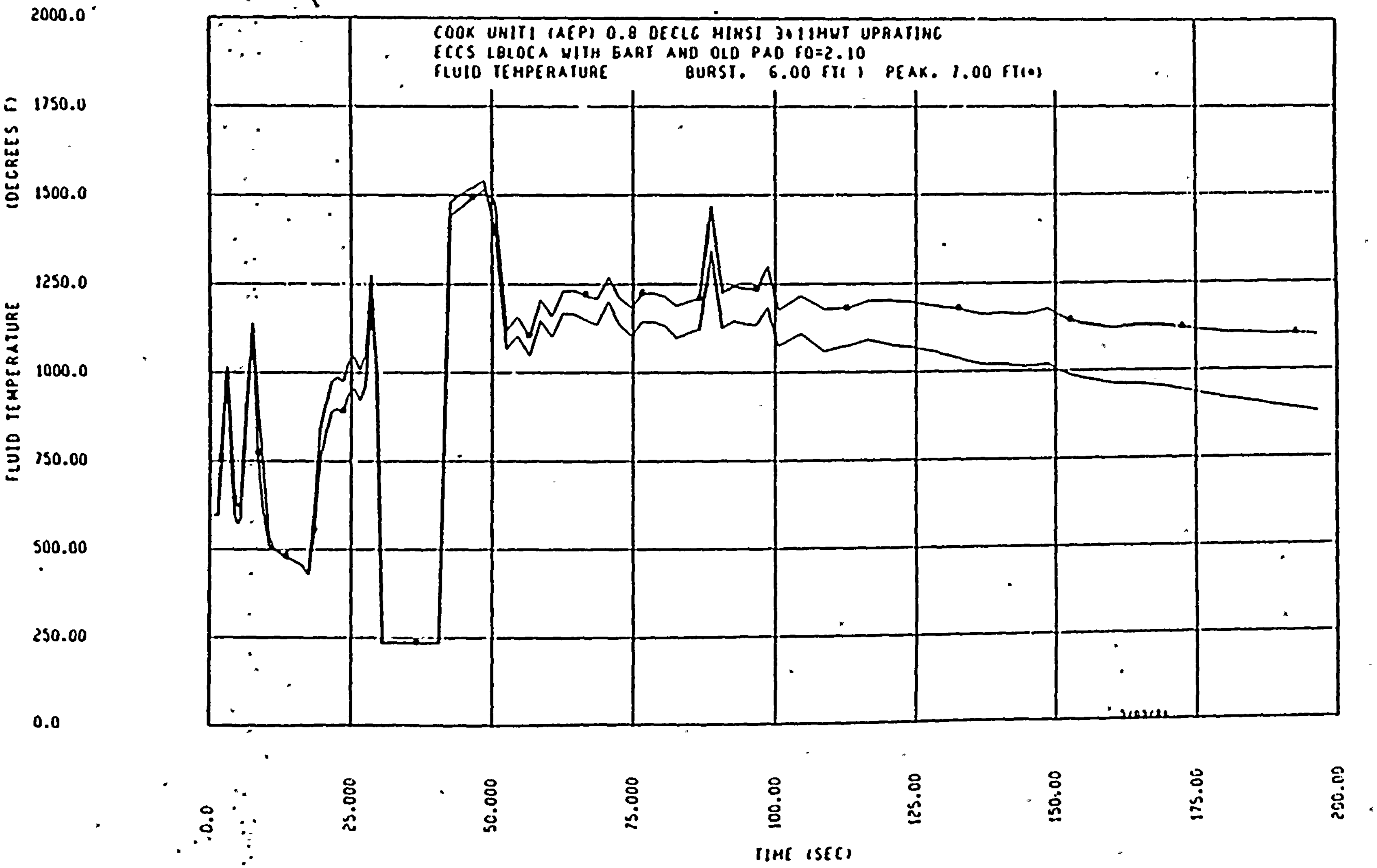


FIGURE 14.3.1-29 FLUID TEMPERATURE
DECLG (CD=0.8) MIN SI

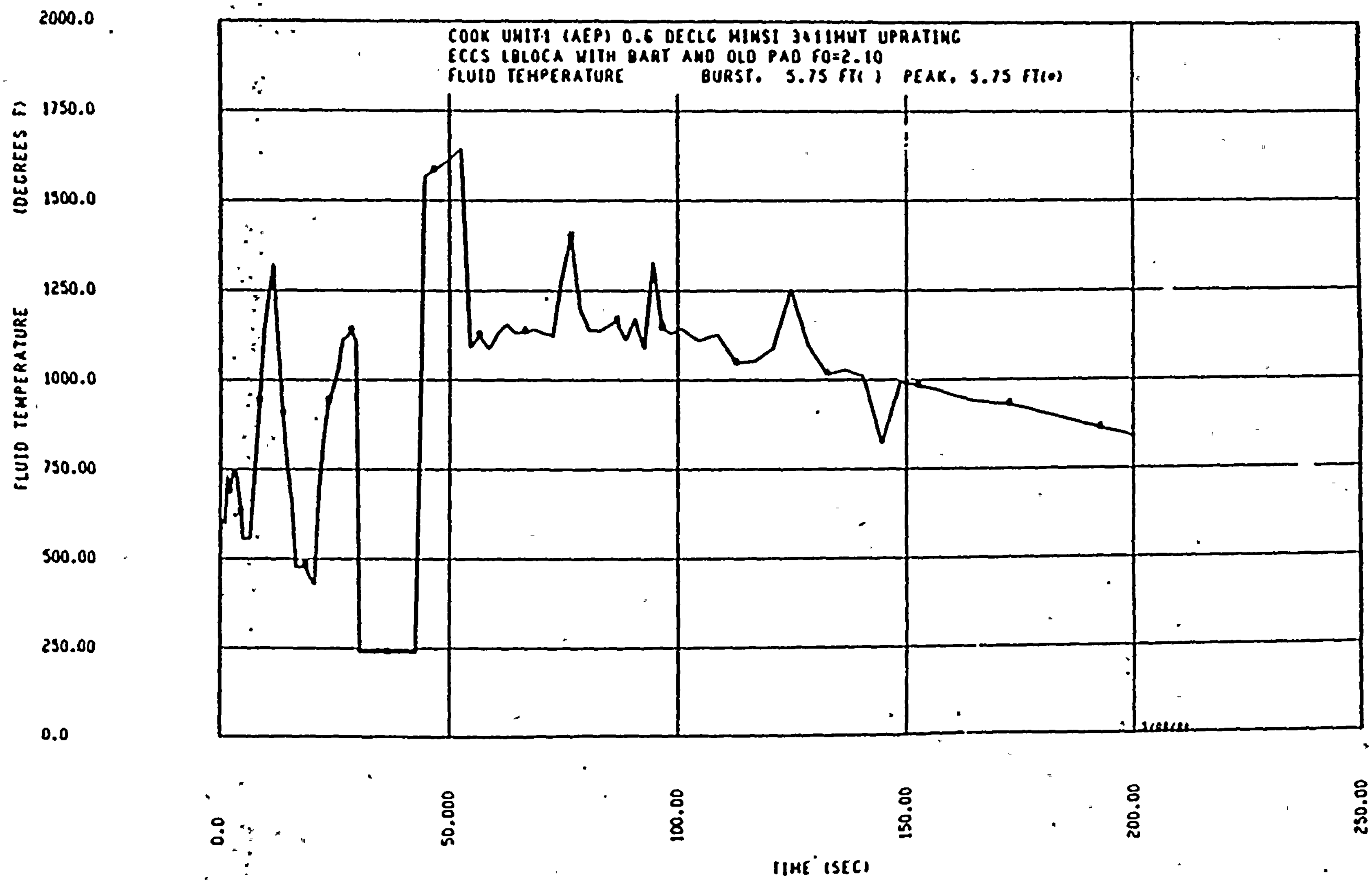


FIGURE 14.3.1-30 FLUID TEMPERATURE
DECLG (CD=0.6) MIN SI

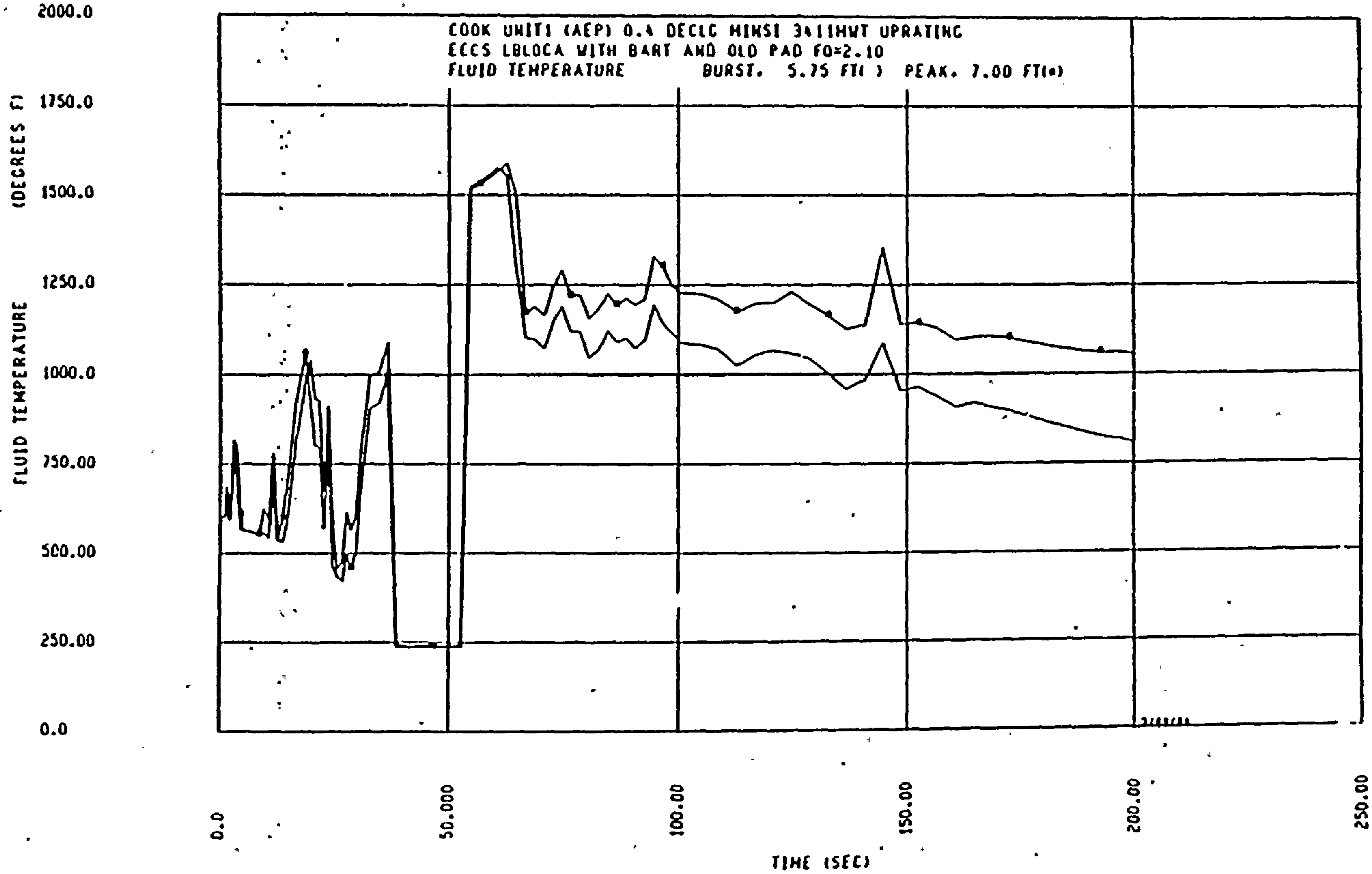


FIGURE 14.3.1-31 FLUID TEMPERATURE
DECLG (CD=0.4) MIN SI

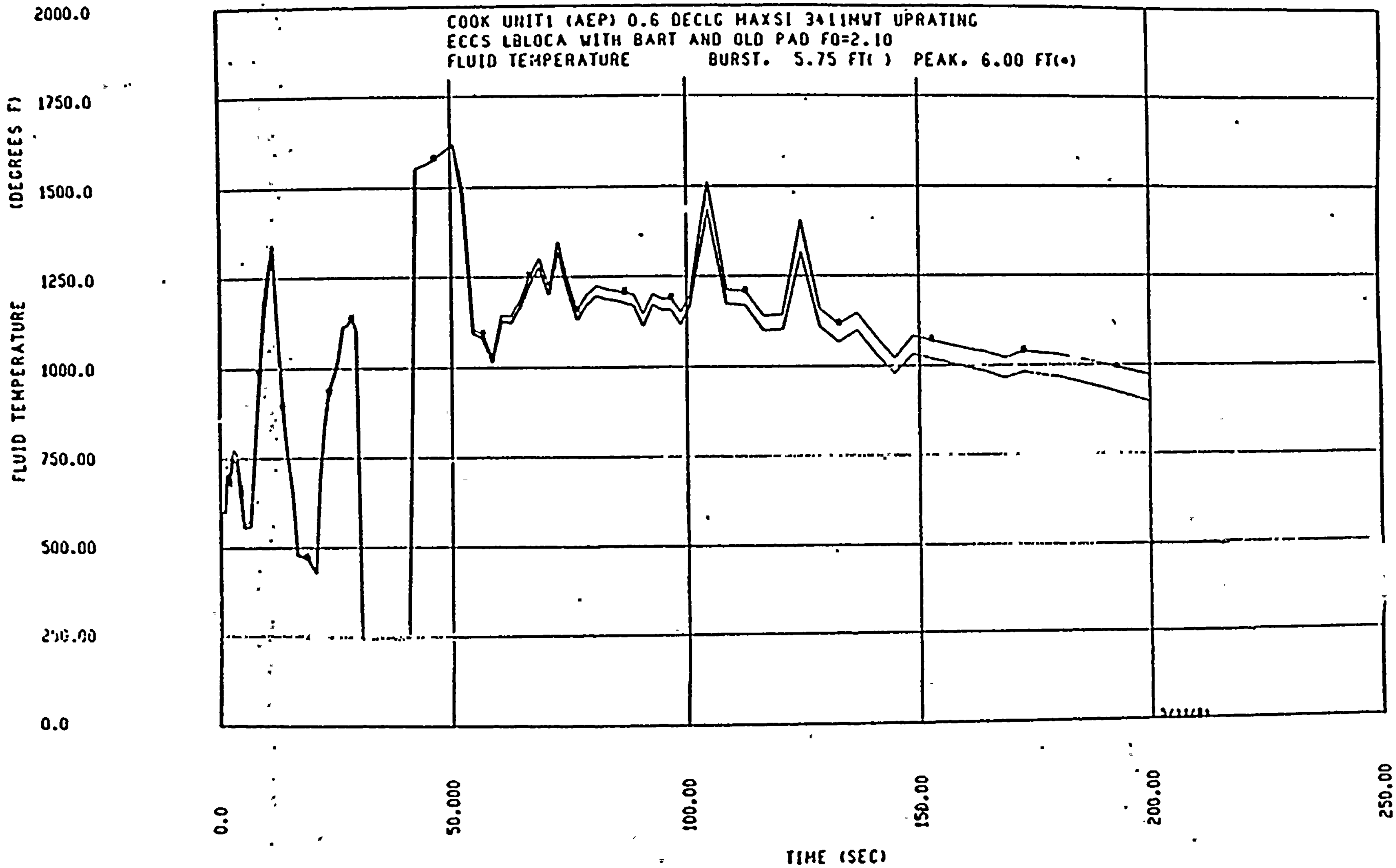


FIGURE 14.3.1-32 FLUID TEMPERATURE
DECLG (CD=0.6) MAX SI

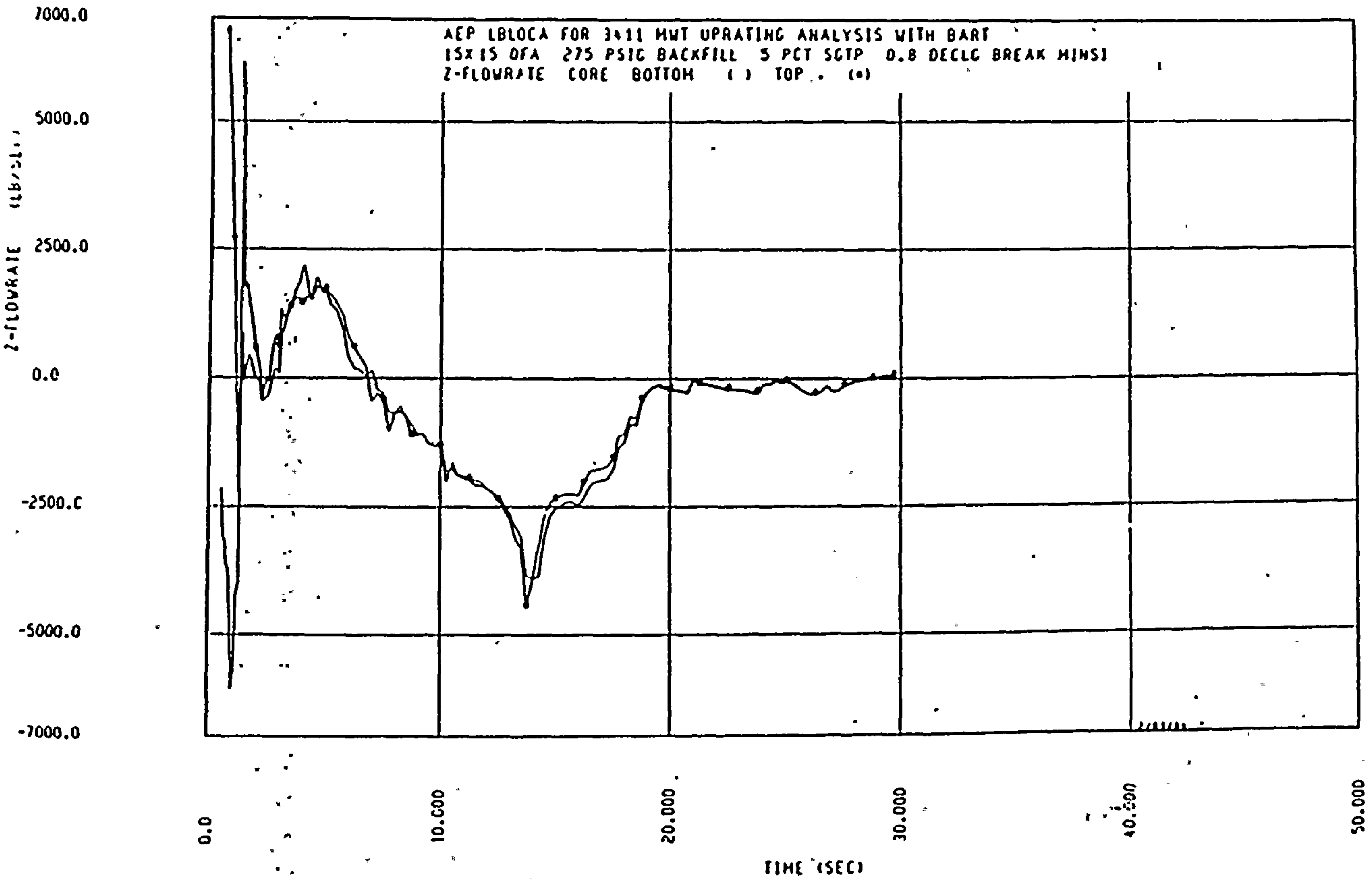


FIGURE 14.3.1-33 CORE FLOW (TOP AND BOTTOM)
DECLG (CD=0.8) MIN SI

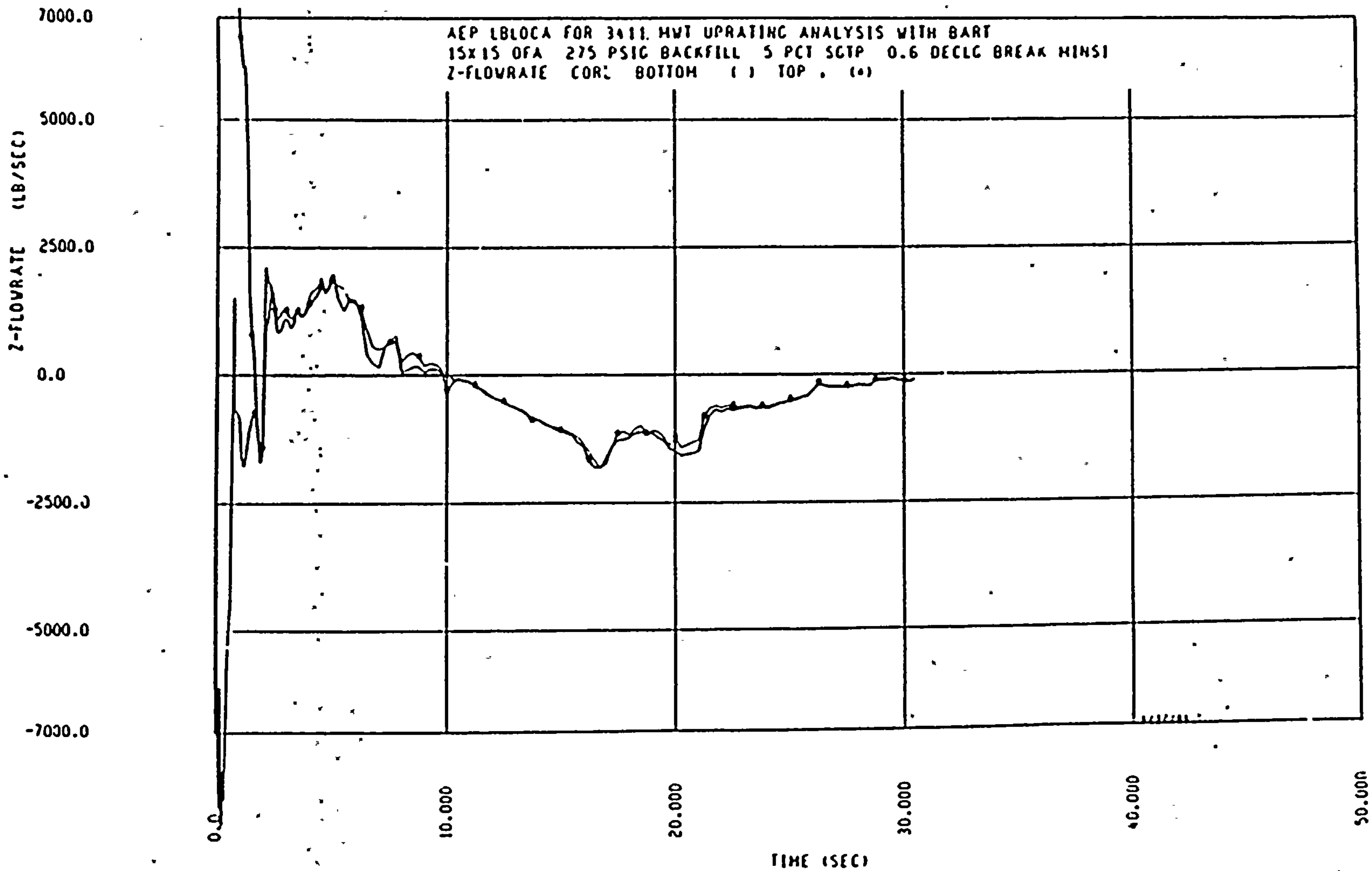


FIGURE 14.3.1-34 CORE FLOW (TOP AND BOTTOM)
DECLG (CD=0.6) MIN SI

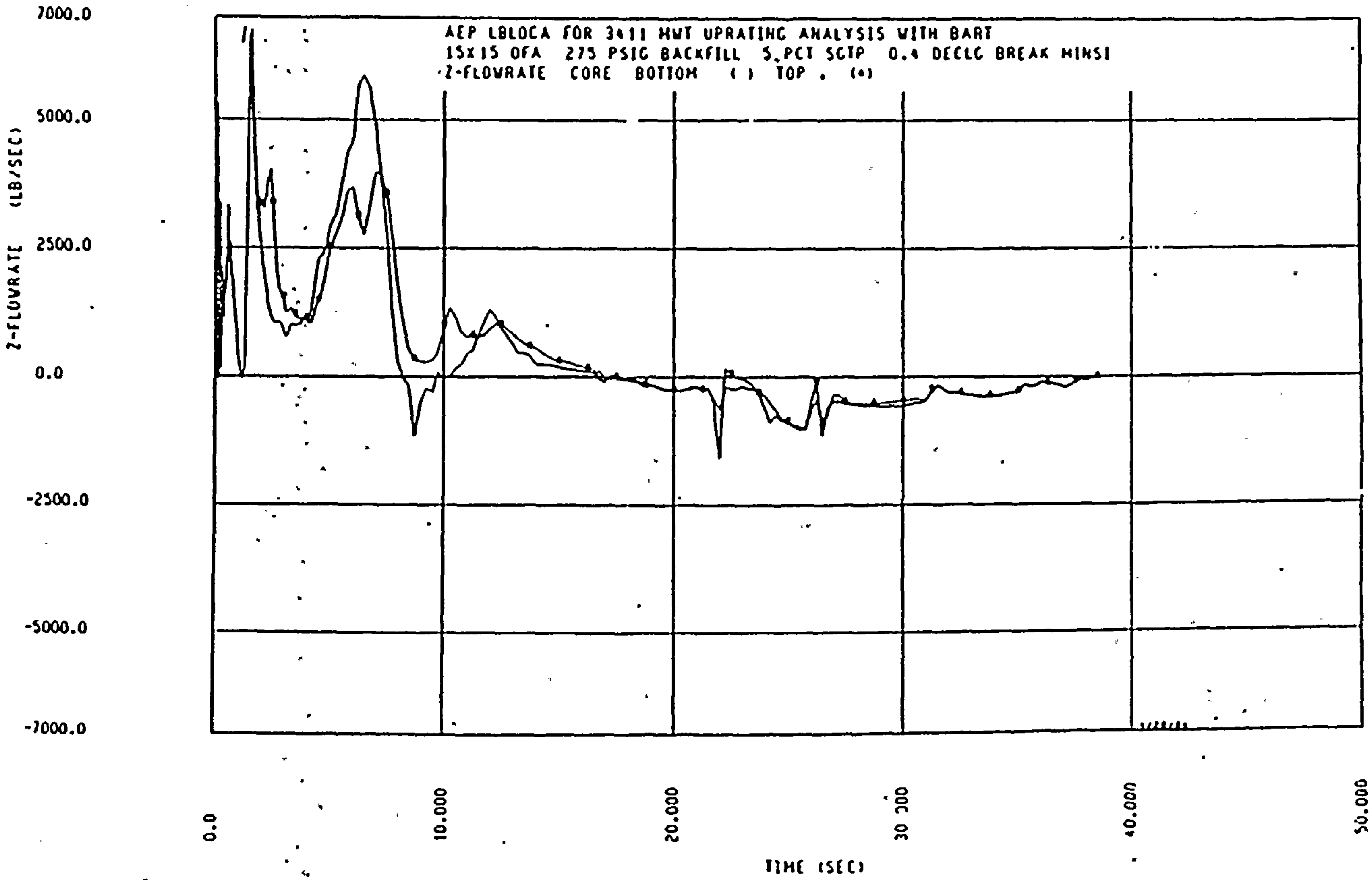


FIGURE 14.3.1-35 CORE FLOW (TOP AND BOTTOM)
DECLG (CD=0.4) MIN SI

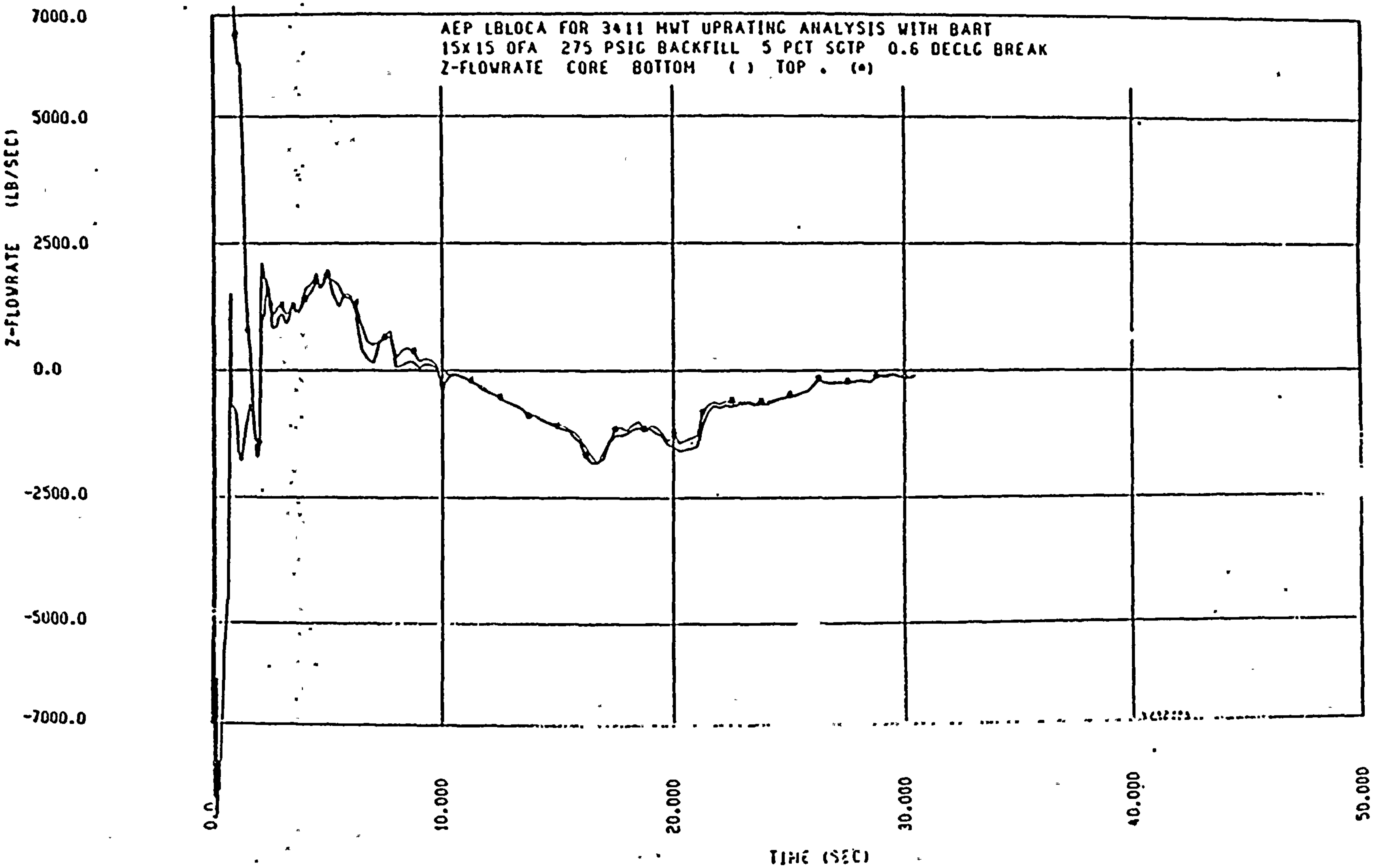


FIGURE 14.3.1-36 CORE FLOW (TOP AND BOTTOM)
DECLG(CD =0.6) MAX SI

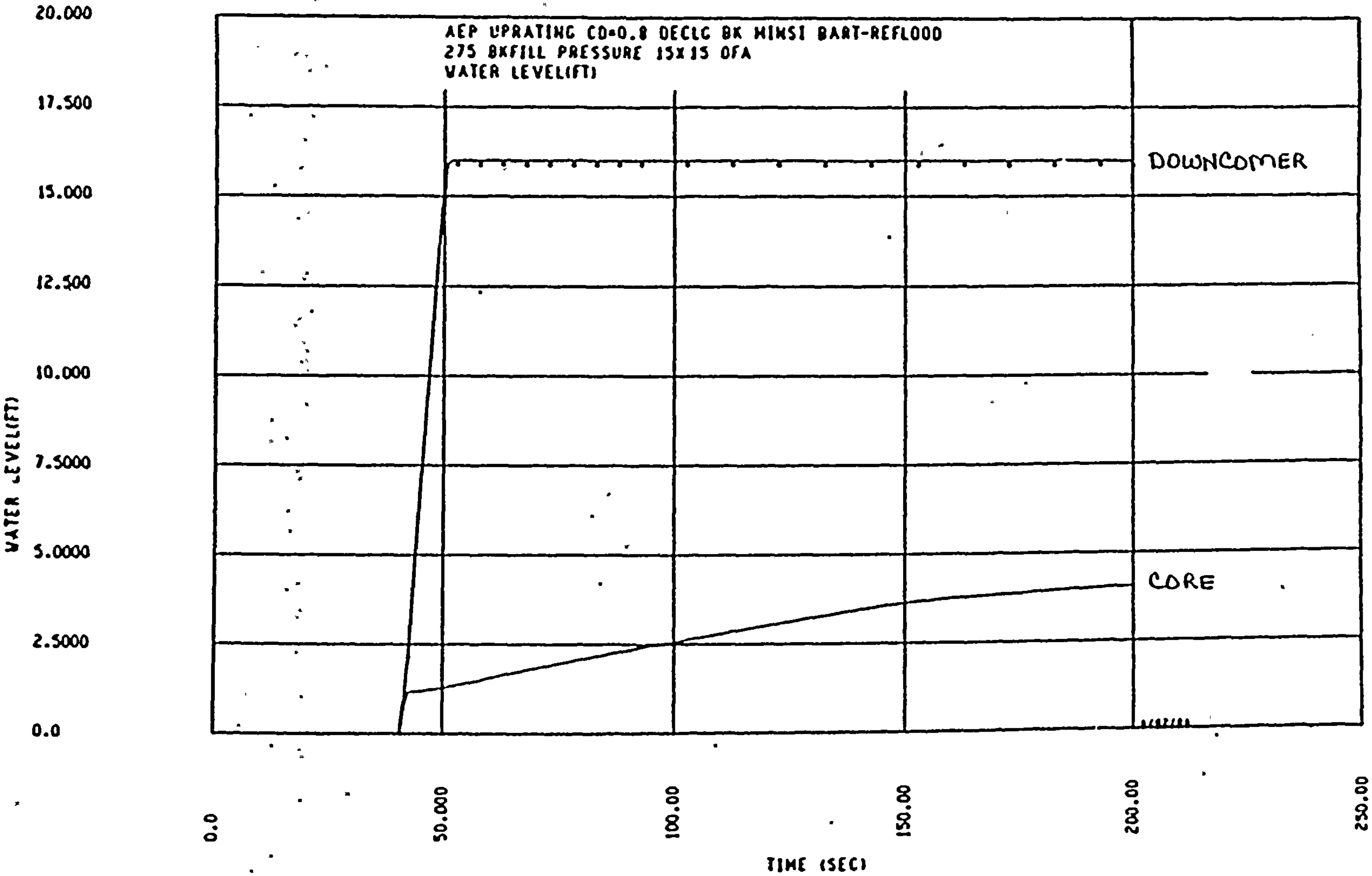


FIGURE 14.3.1-37 REFLOOD TRANSIENT - CORE
+ DOWNCOMER WATER LEVELS
DECLG (CD=0.8) MIN SI

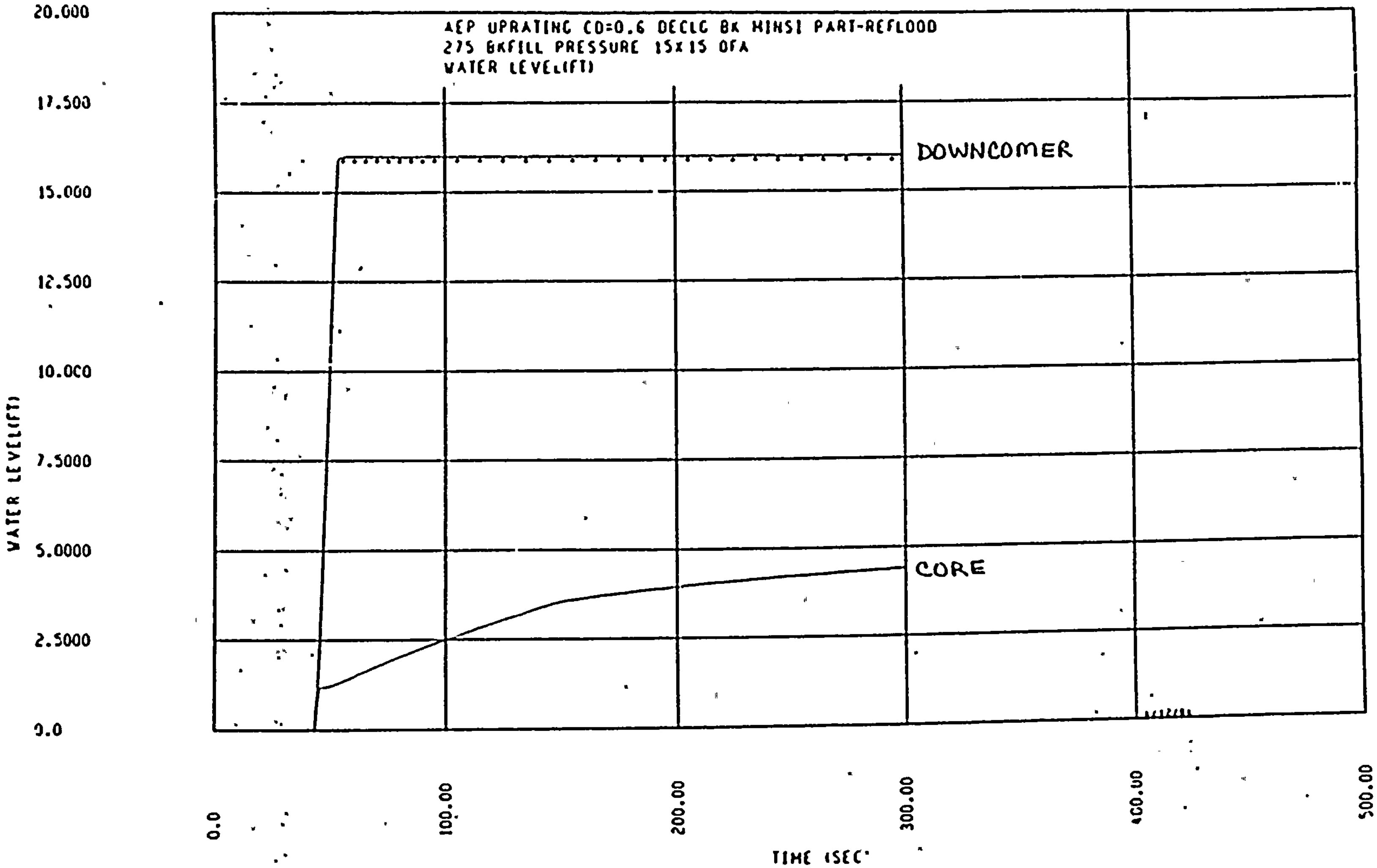


FIGURE 14.3.1-38 REFLOOD TRANSIENT – CORE & DOWNCOMER WATER LEVELS DECLG (CD=0.6) MIN SI

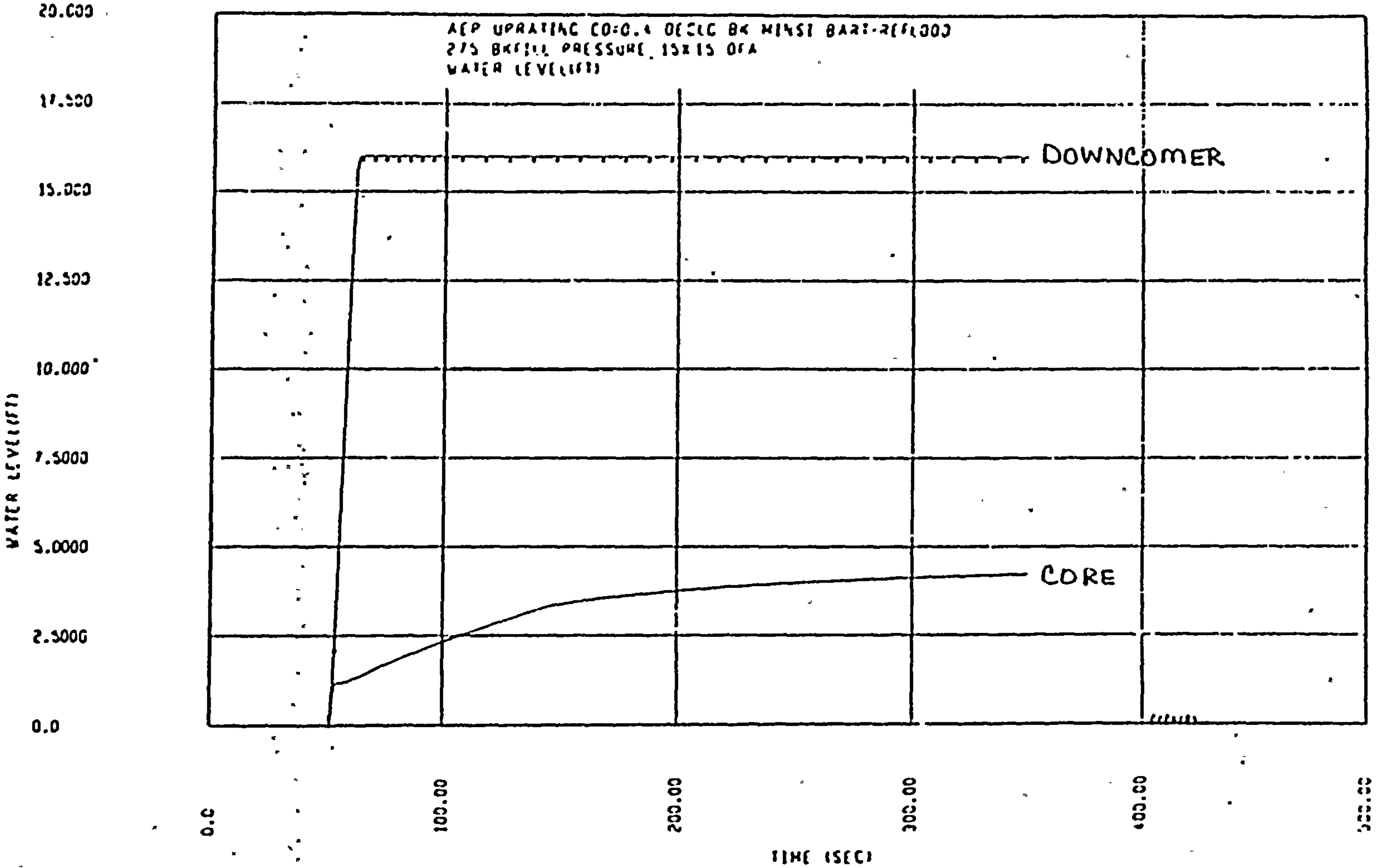


FIGURE 14.3.1-39 REFLOOD TRANSIENT - CORE
+ DOWNCOMER WATER LEVELS
DECLG (CD=0.4) MIN SI

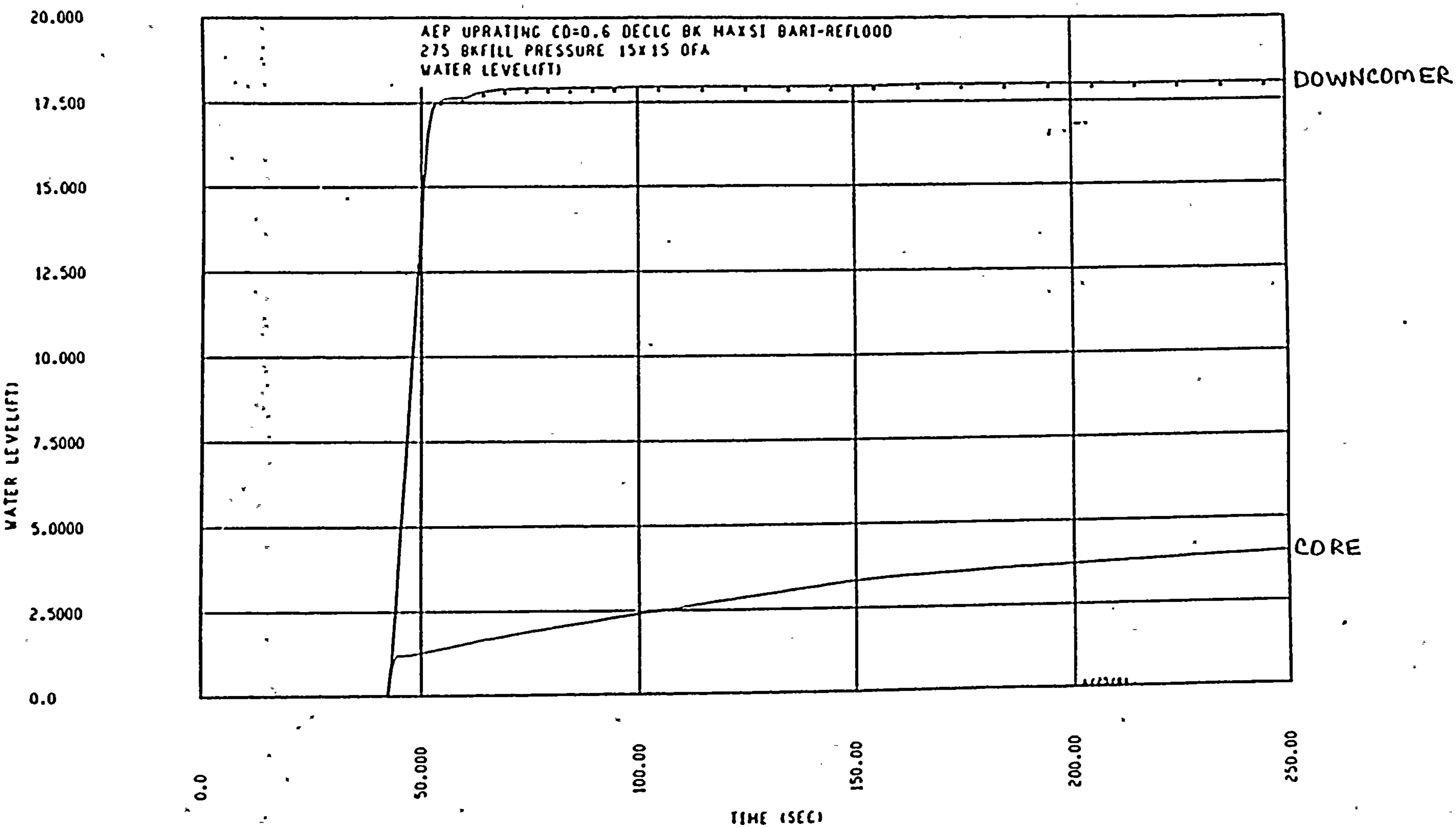


FIGURE 14.3.1-40 REFLOOD TRANSIENT - CORE + DOWNCOMER WATER LEVELS DECLG (CD=0.6) MAX SI

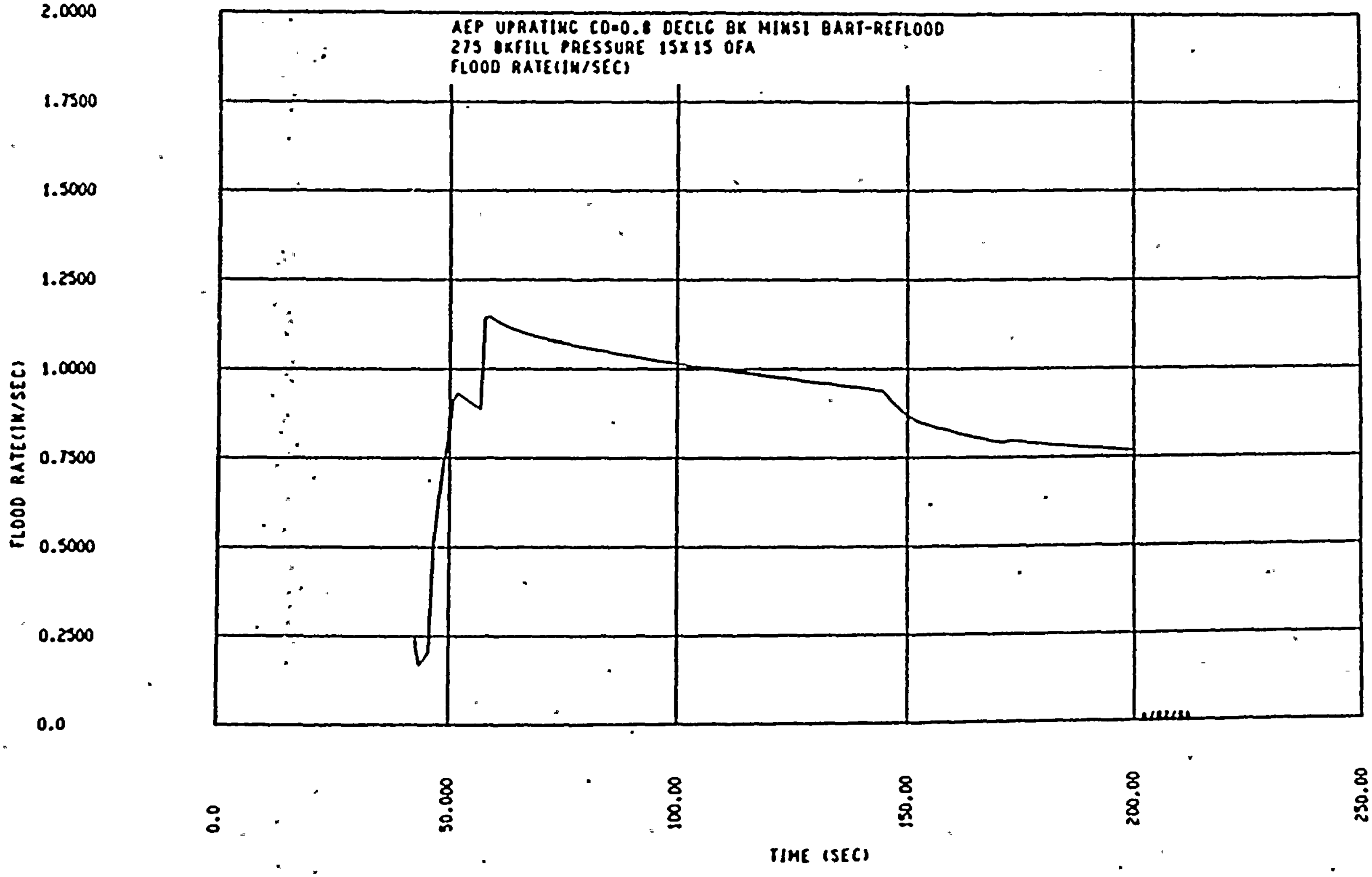


FIGURE 14.3.1-41 REFLOOD TRANSIENT
CORE INLET VELOCITY
DECLG (CD=0.8) MIN SI

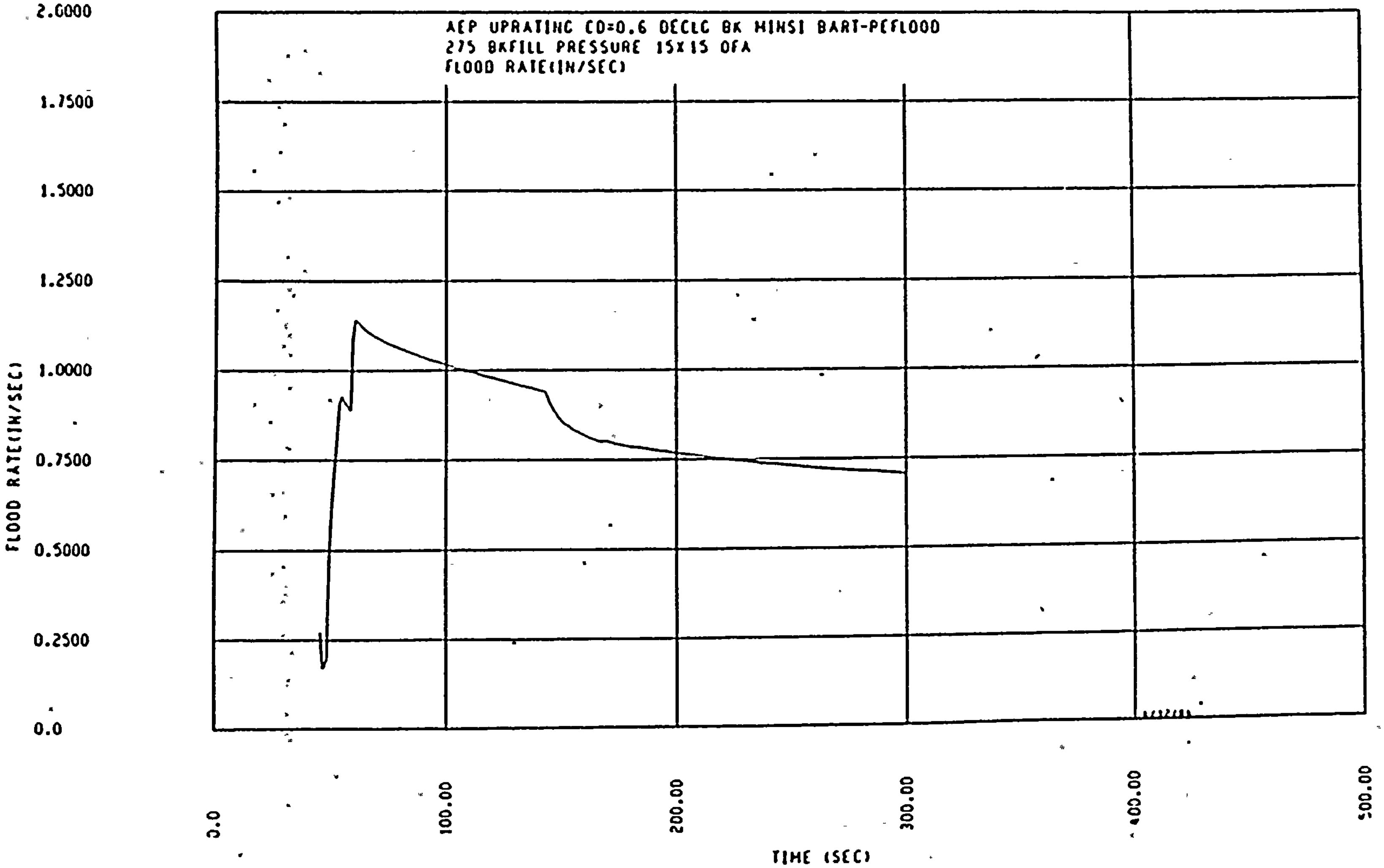


FIGURE 14.3.1-42 REFLOOD TRANSIENT
CORE INLET VELOCITY
DECLG (CD = 0.6) MIN. SI

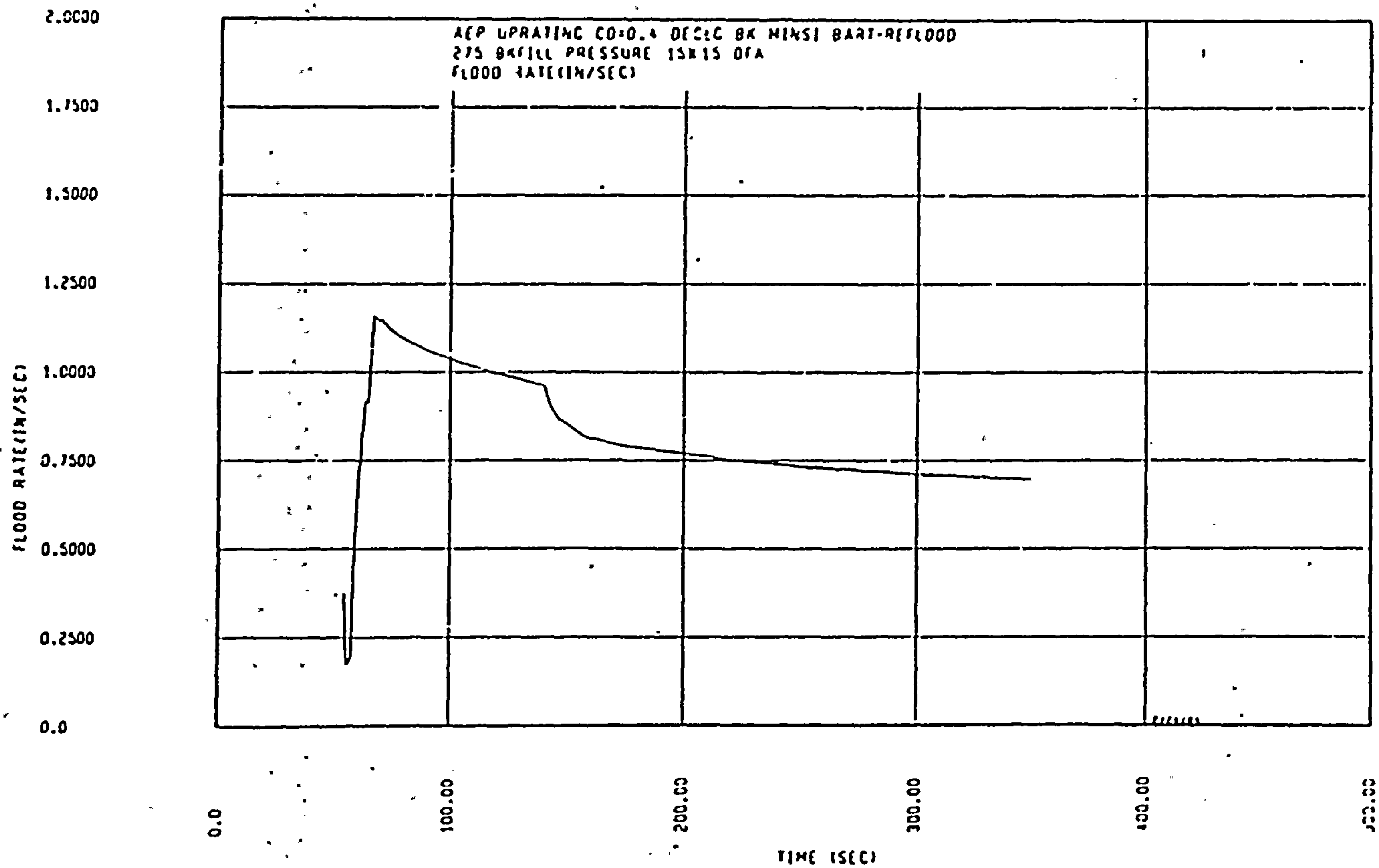


FIGURE 14.3.1-43 REFLOOD TRANSIENT.
CORE INLET VELOCITY
DECLG (CD=0.4) MIN SI

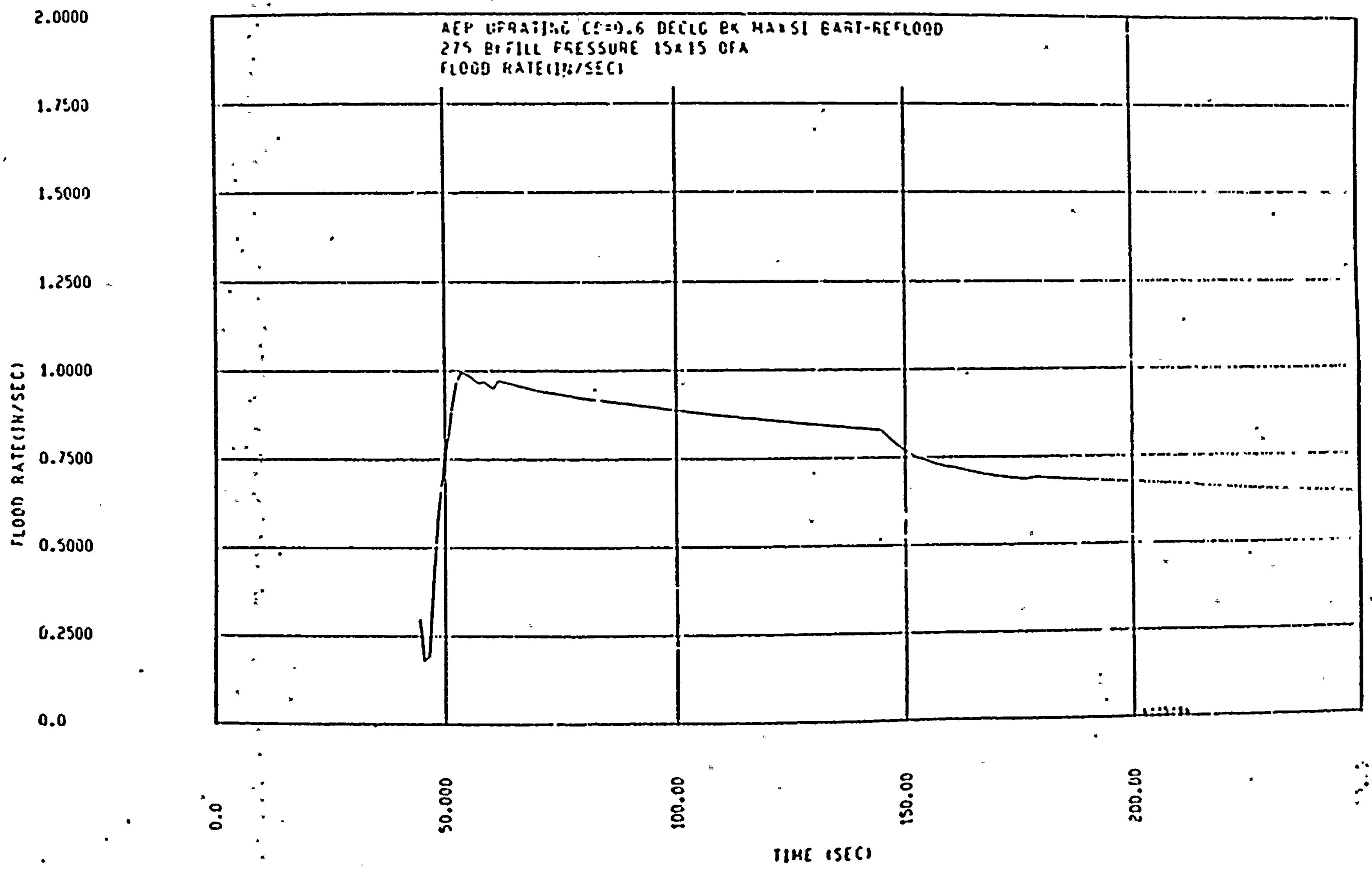


FIGURE 14.3.1-44 REFLOOD TRANSIENT
CORE INLET VELOCITY
DECLG (CD = 0.6) MAX SI

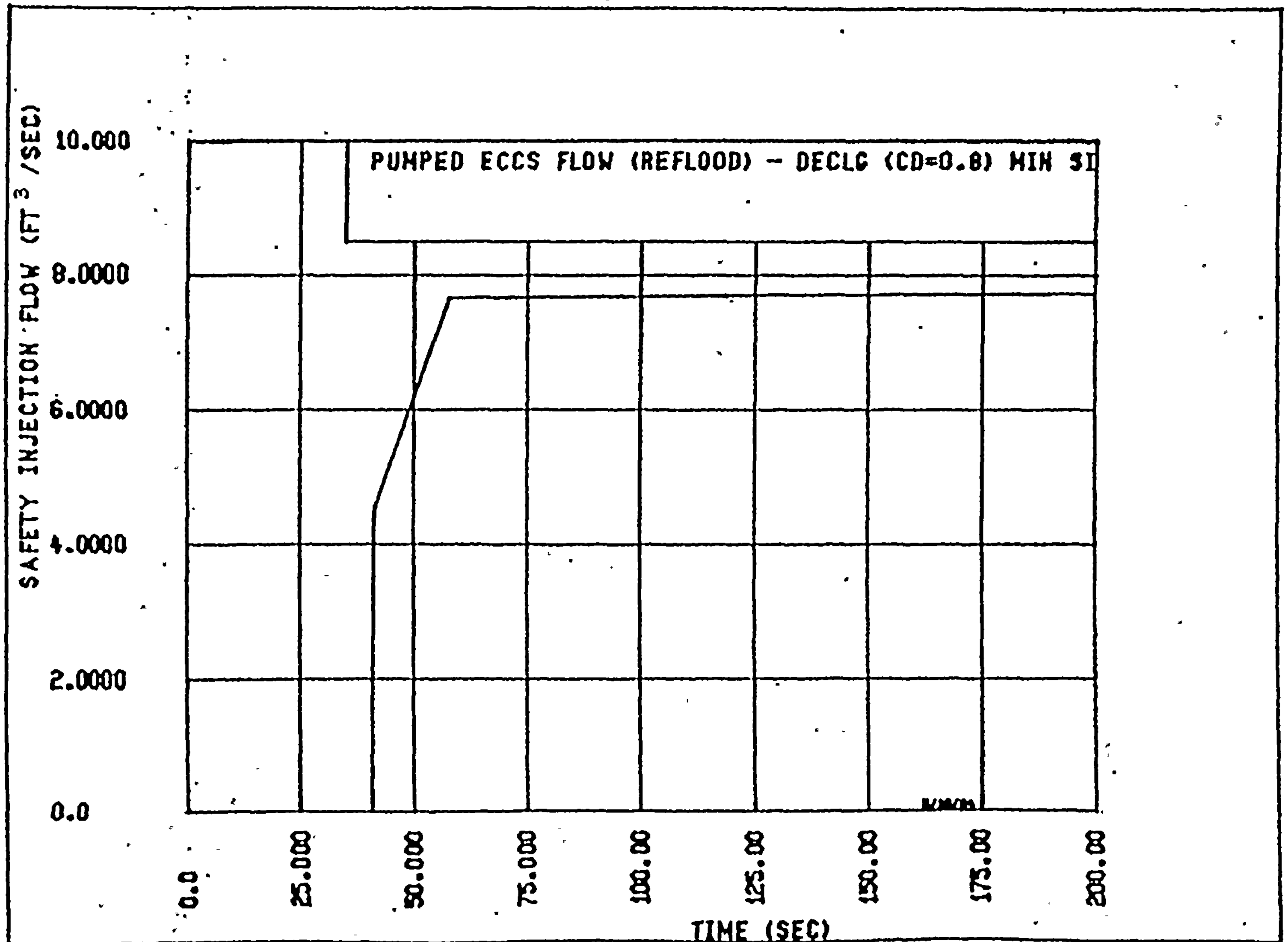


FIGURE 14.3.1-45 PUMPED ECCS FLOW (REFLOOD)
DECLG (CD=0.8) MIN SI

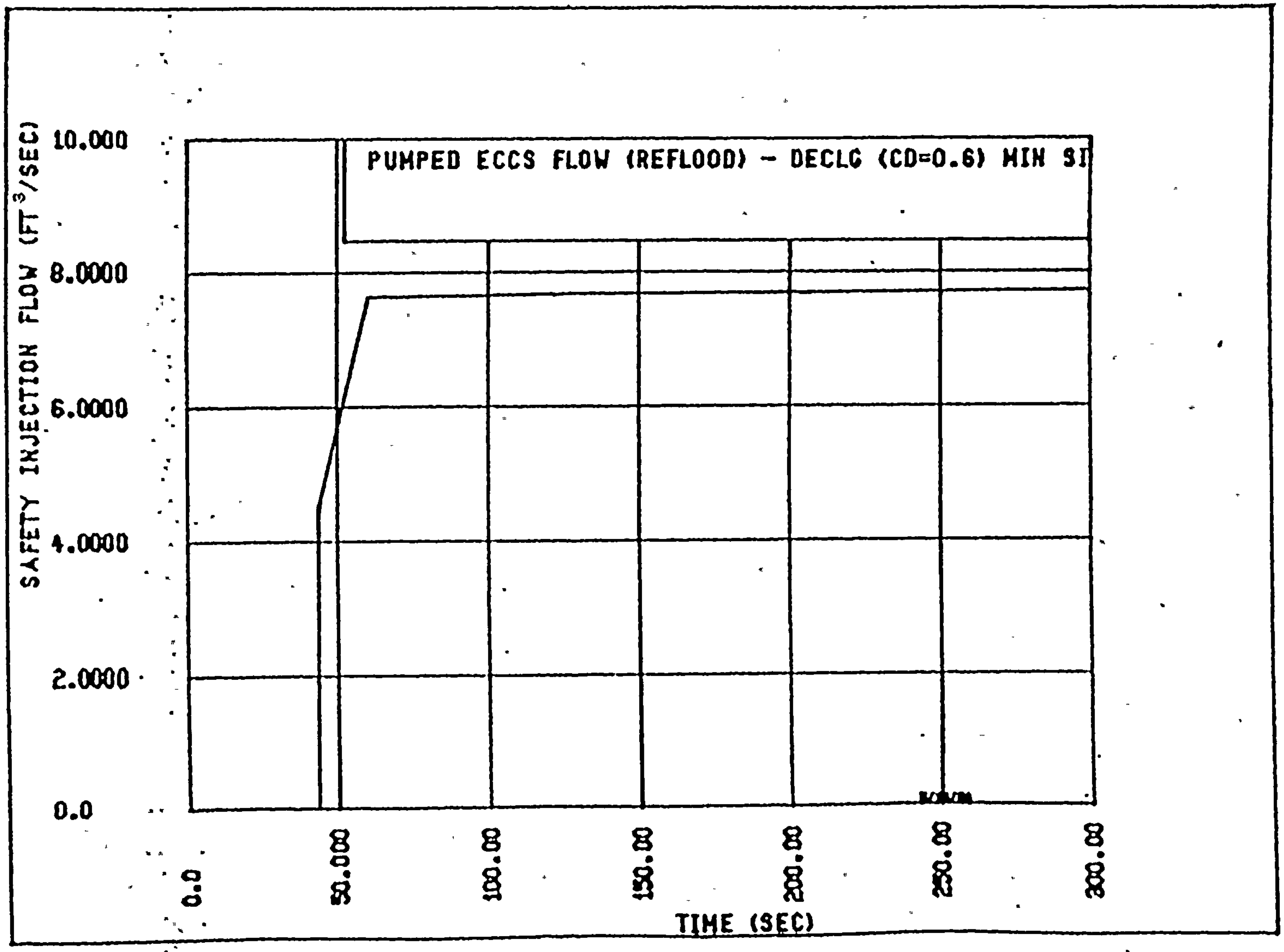


FIGURE 14.3.1-46 PUMPED ECCS FLOW (REFLOOD) DECLG (CD=0.6) MIN SI

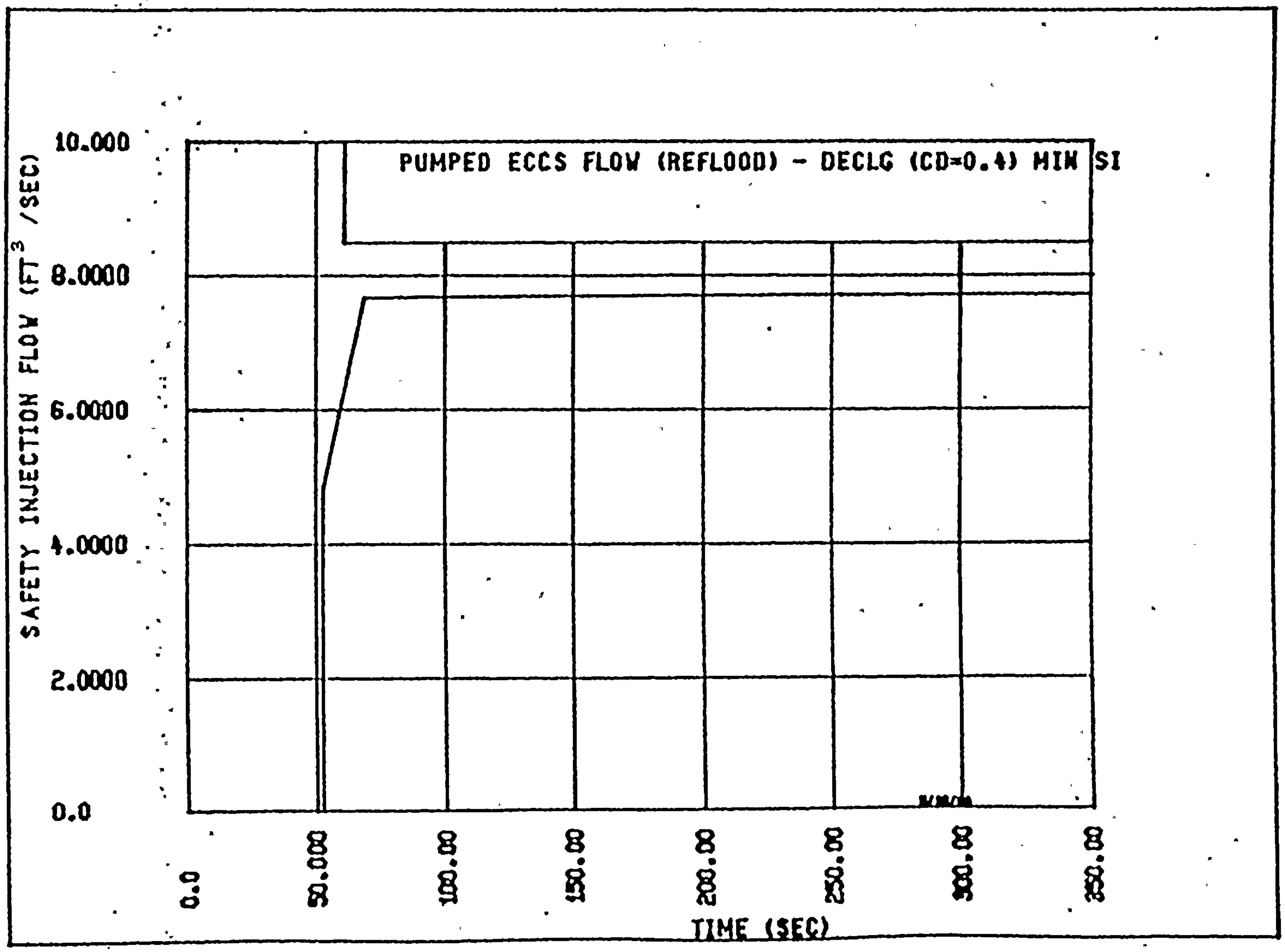


FIGURE 14.3.1-47 PUMPED ECCS FLOW (REFLOOD) DECLG (CD=0.4) MIN SI

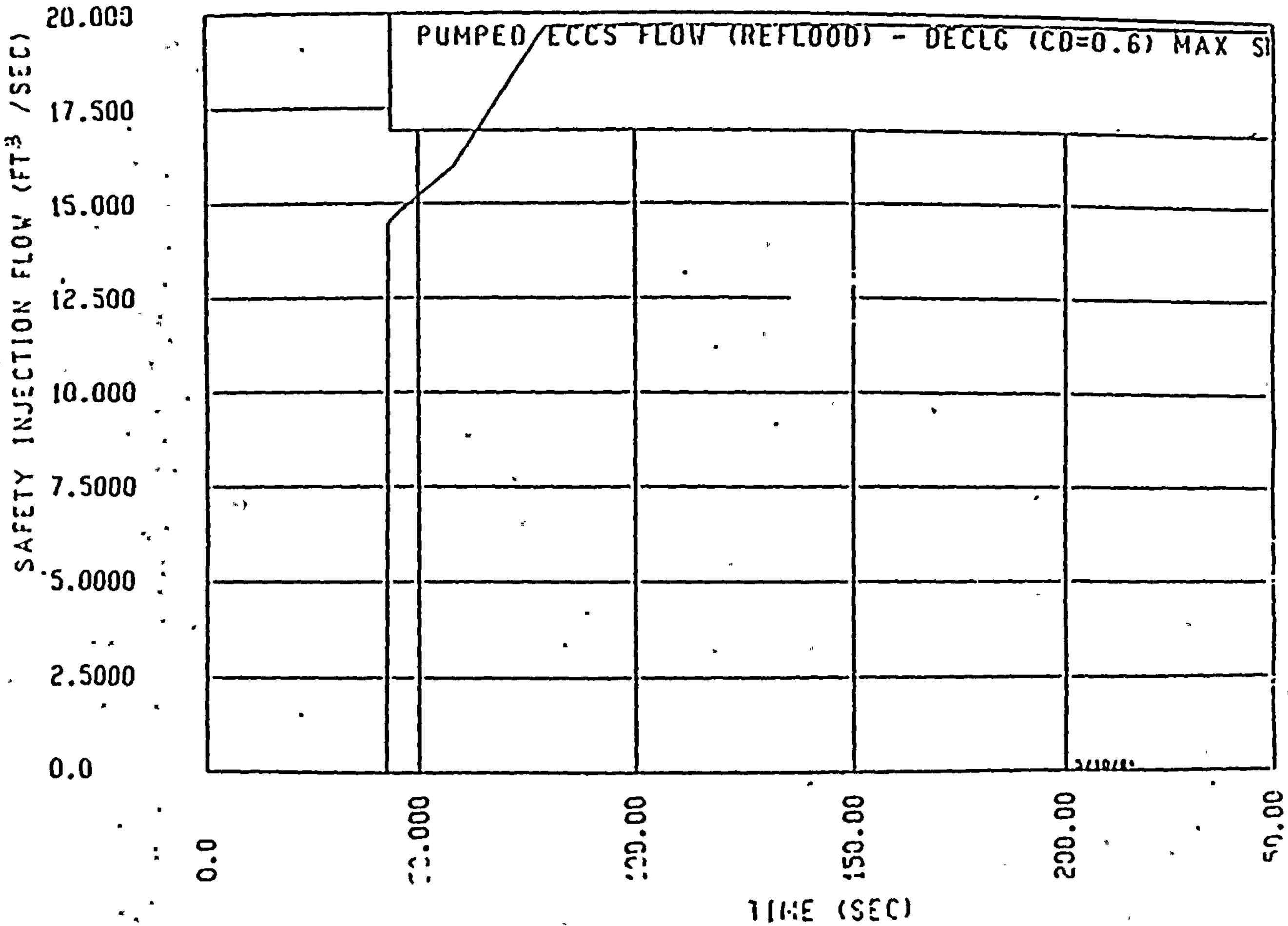


FIGURE 14.3.1-48 PUMPED ECCS FLOW (REFLOOD)
DECLG (CD=0.6) MAX SI

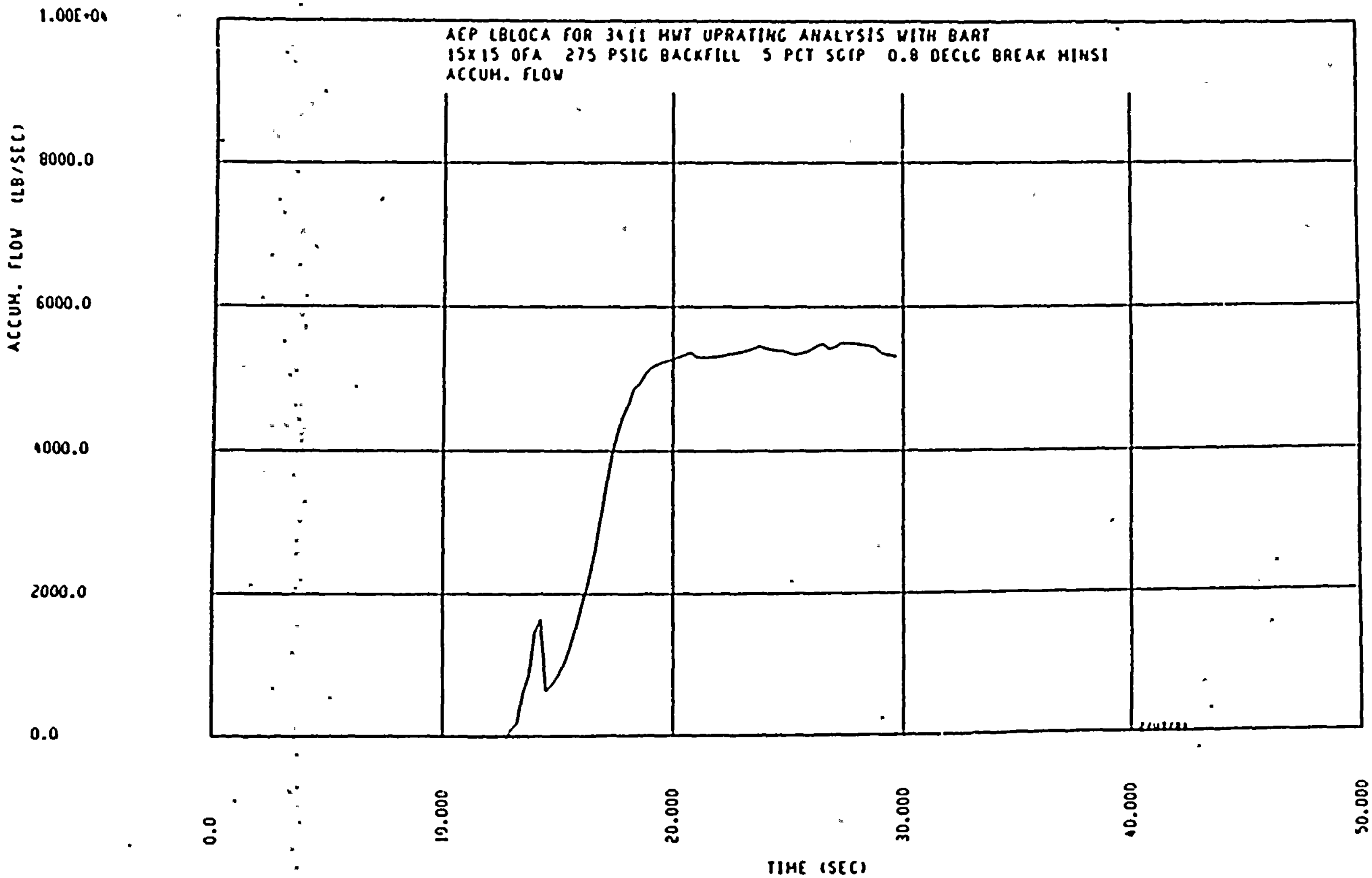


FIGURE 14.3.1-49 ACCUMULATOR FLOW (BLOWDOWN)
DECLG (CD=0.8) MIN SI

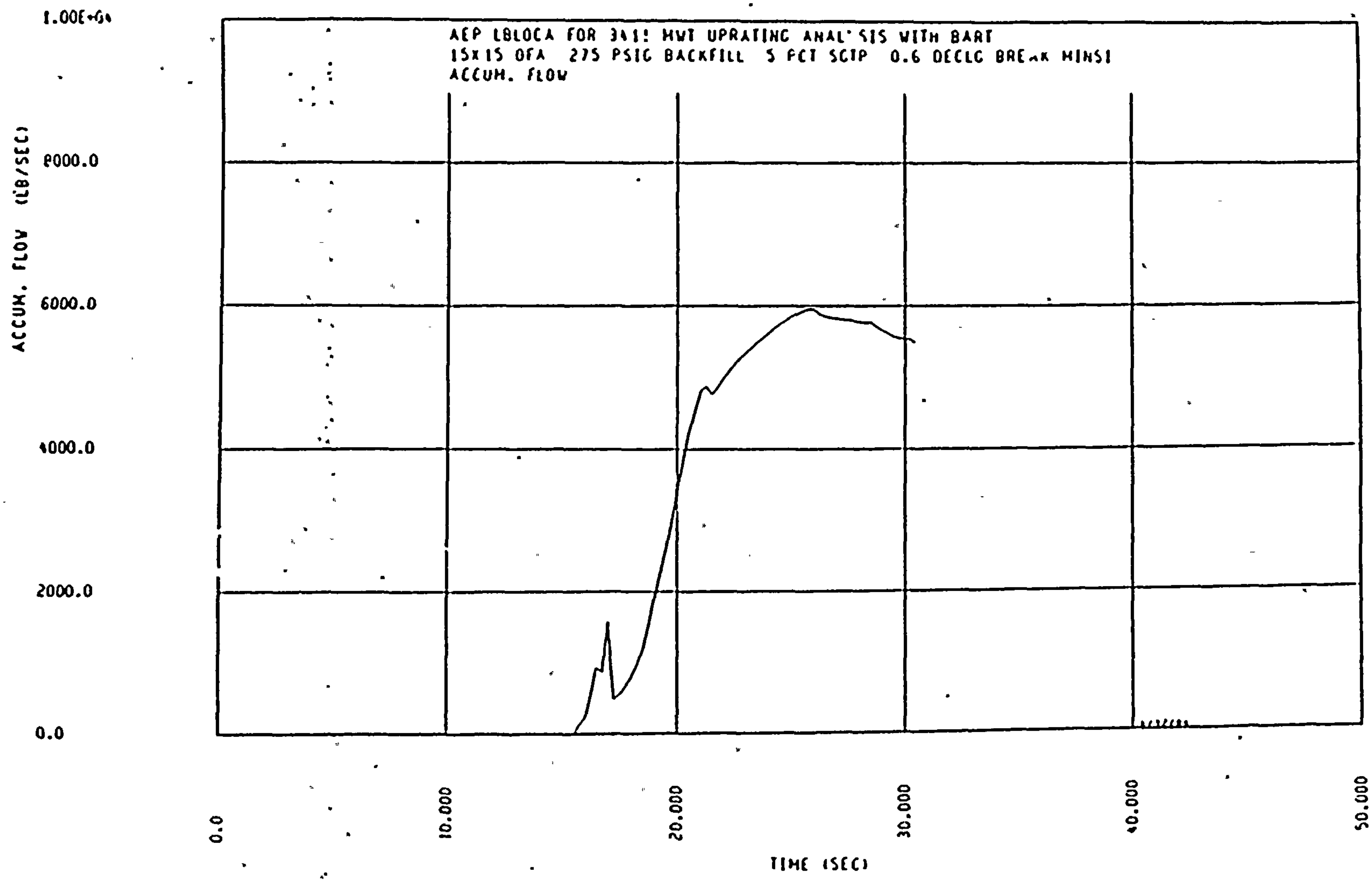


FIGURE 14.31-50 ACCUMULATOR FLOW (BLOWDOWN)
DECLG (CD=0.6) MIN SI

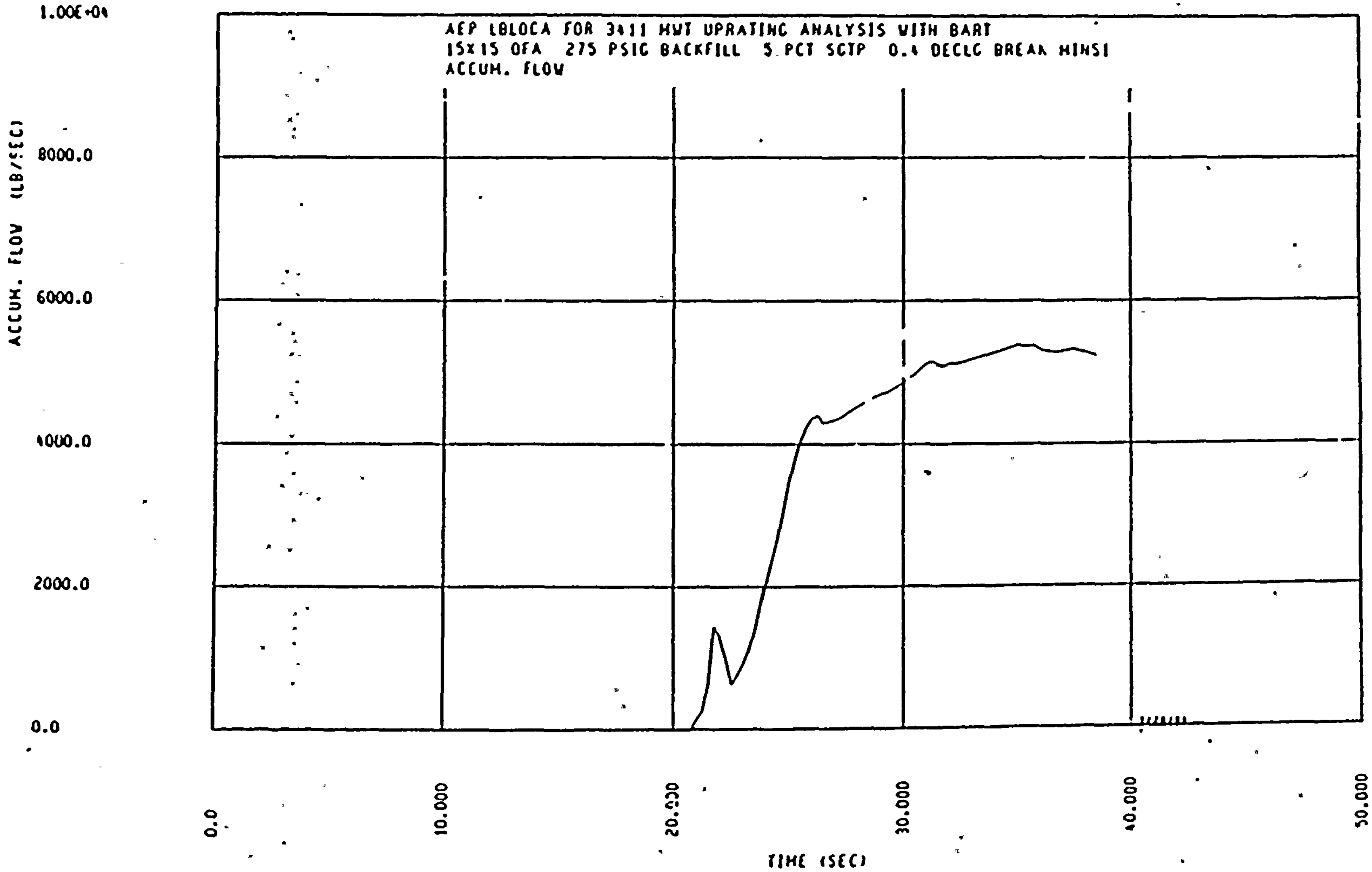


FIGURE 14.3.1-51 ACCUMULATOR FLOW (BLOWDOWN)
DECLG (CD=0.4) MIN SI

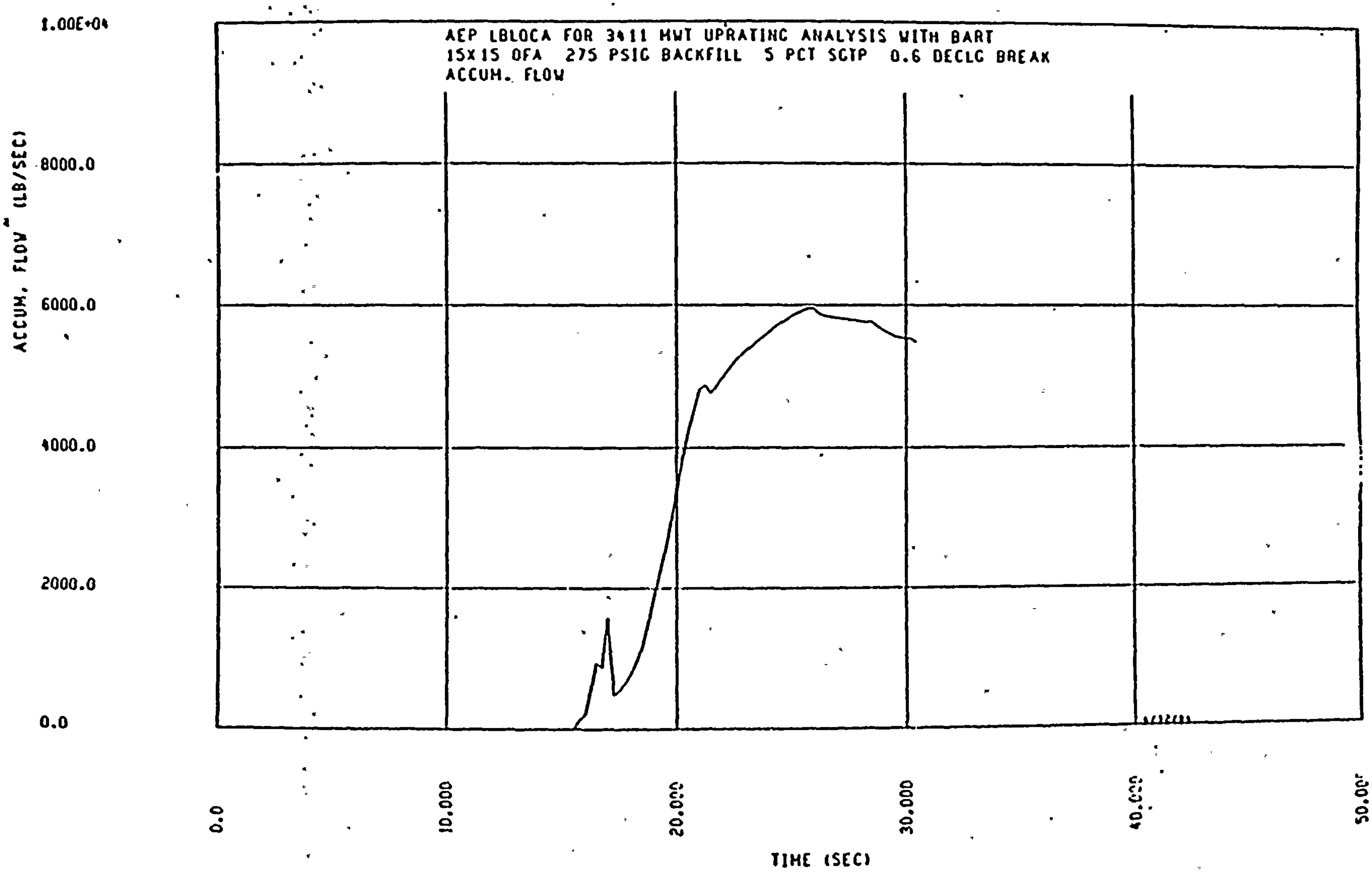
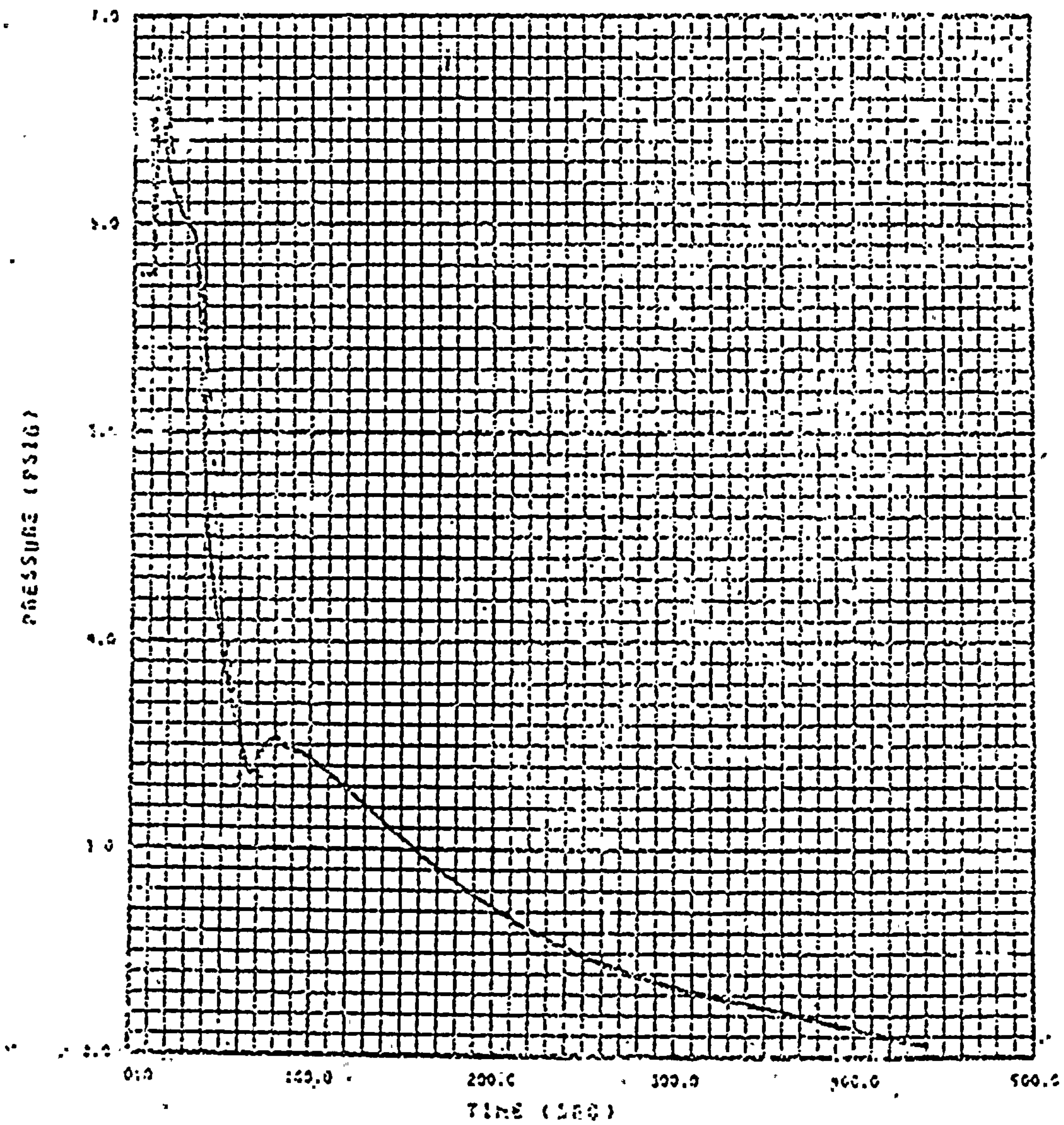


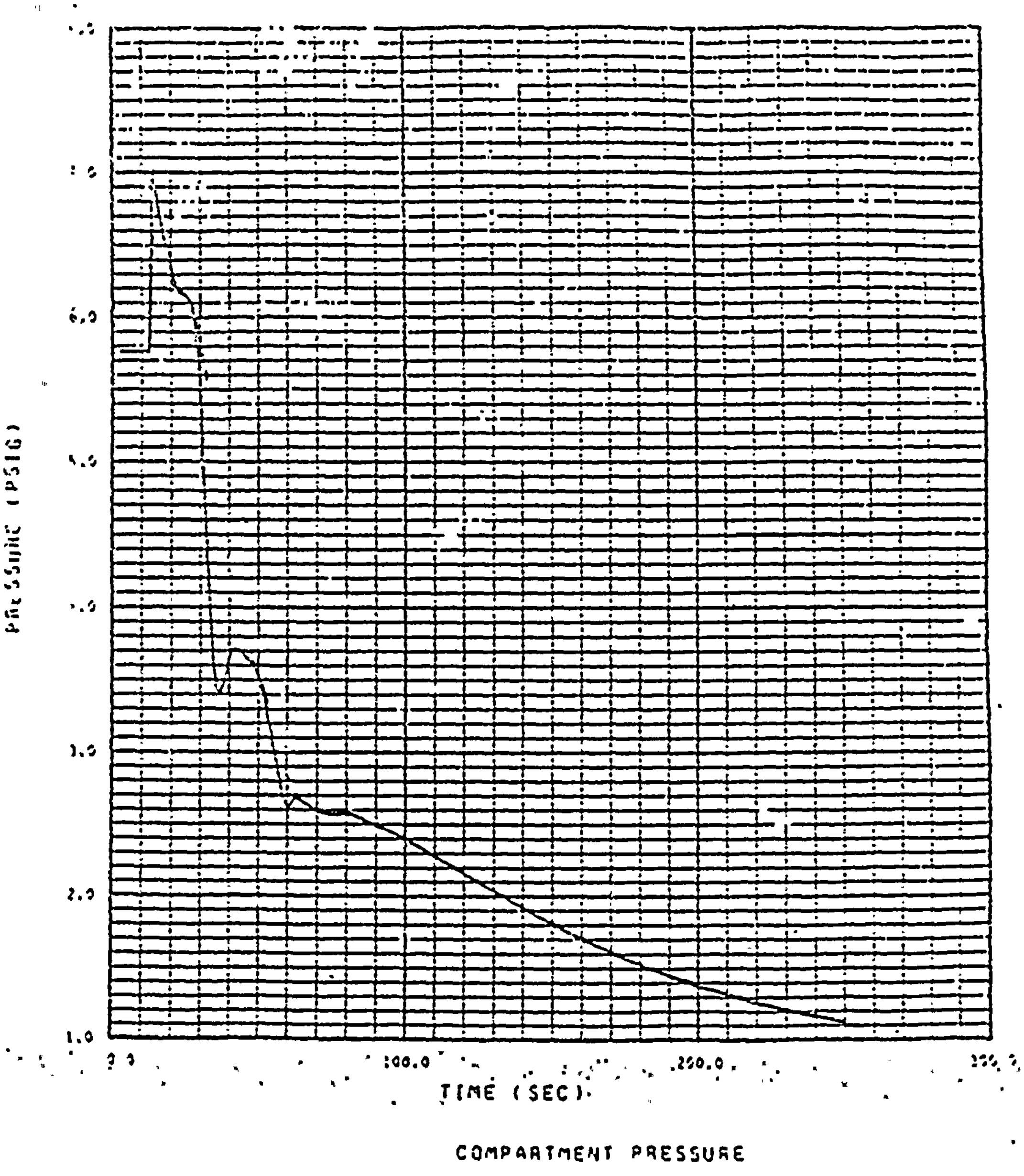
FIGURE 14.3.1-52 ACCUMULATOR FLOW (BLOWDOWN)
DECLG (CD=0.6) MAX SI



COMPARTMENT PRESSURE

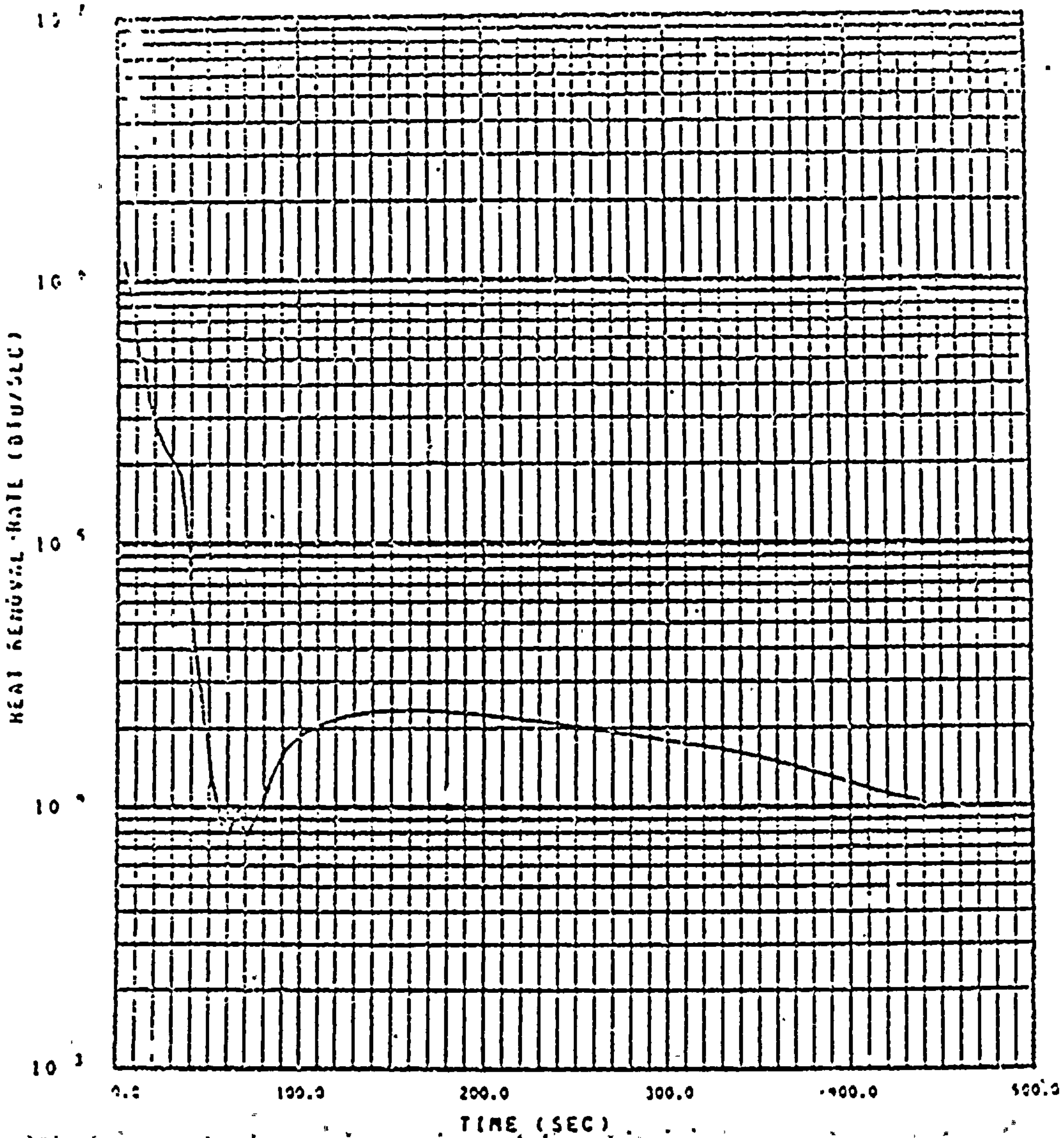
MINIMUM SI

FIGURE 14.3.1-53 CONTAINMENT PRESSURE



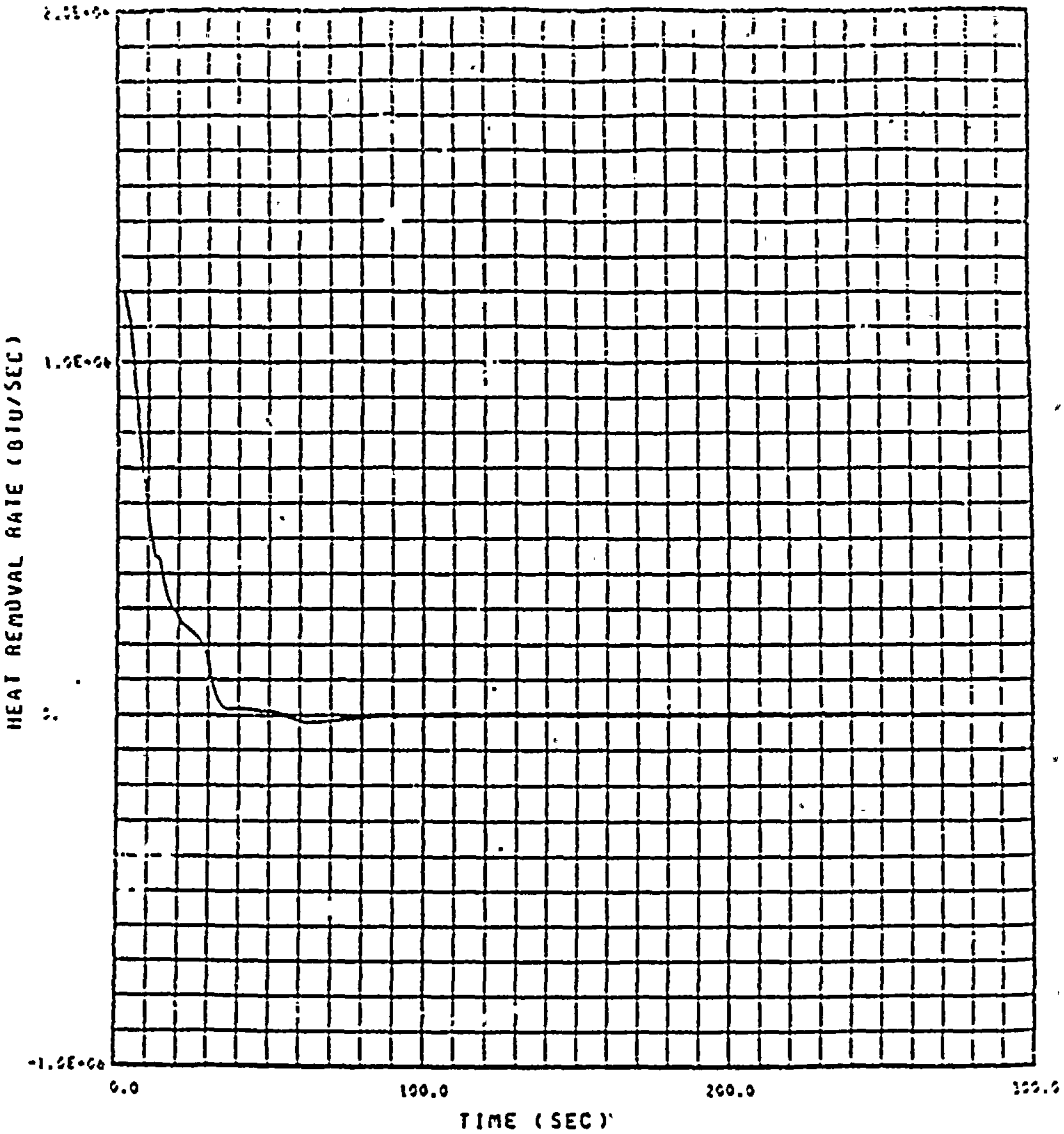
MAXIMUM SI

FIGURE 14.3.1-54 CONTAINMENT PRESSURE



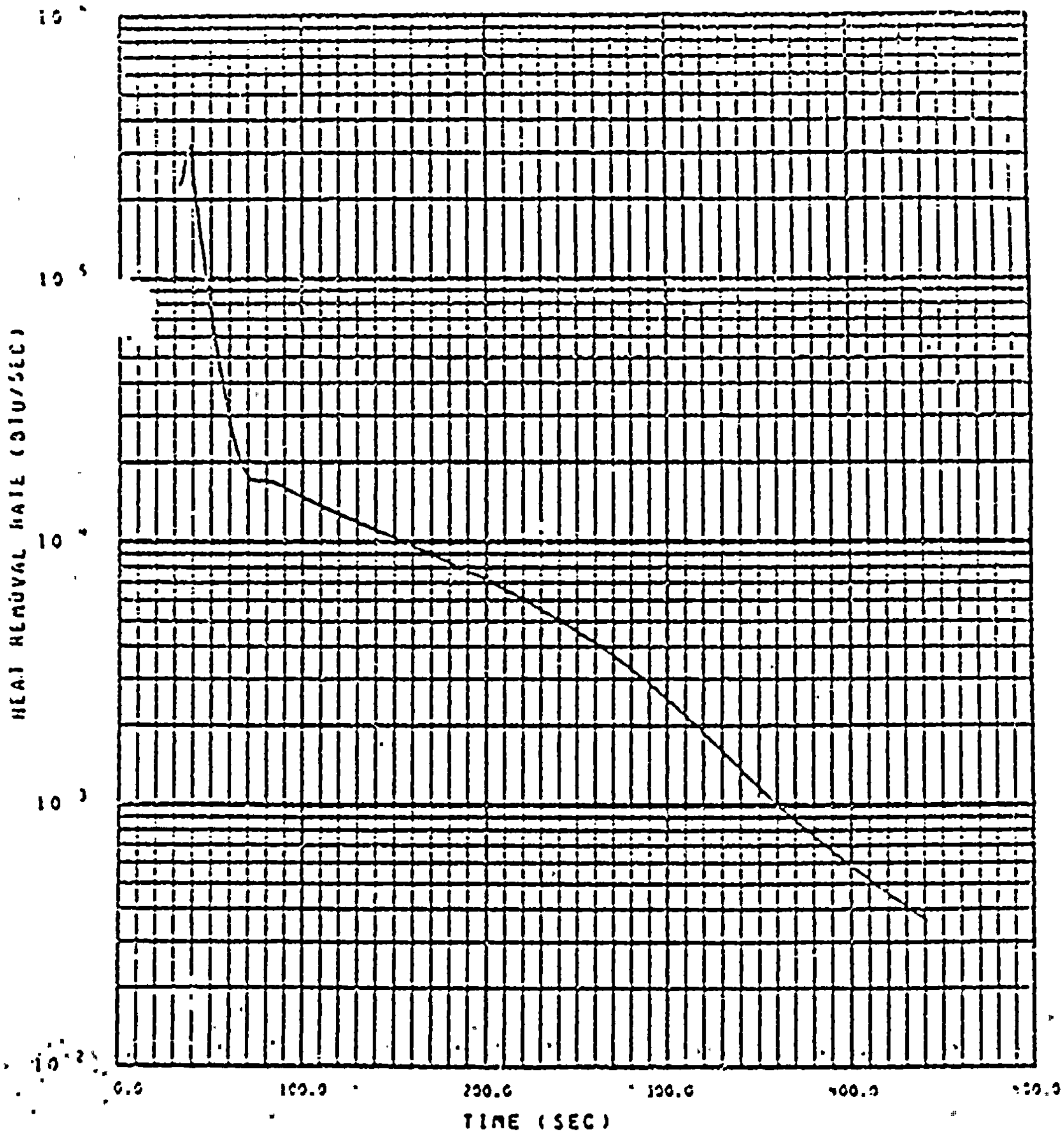
MINIMUM SI

FIGURE 14.3.1-55 LOWER COMPARTMENT STRUCTURAL HEAT REMOVAL RATE



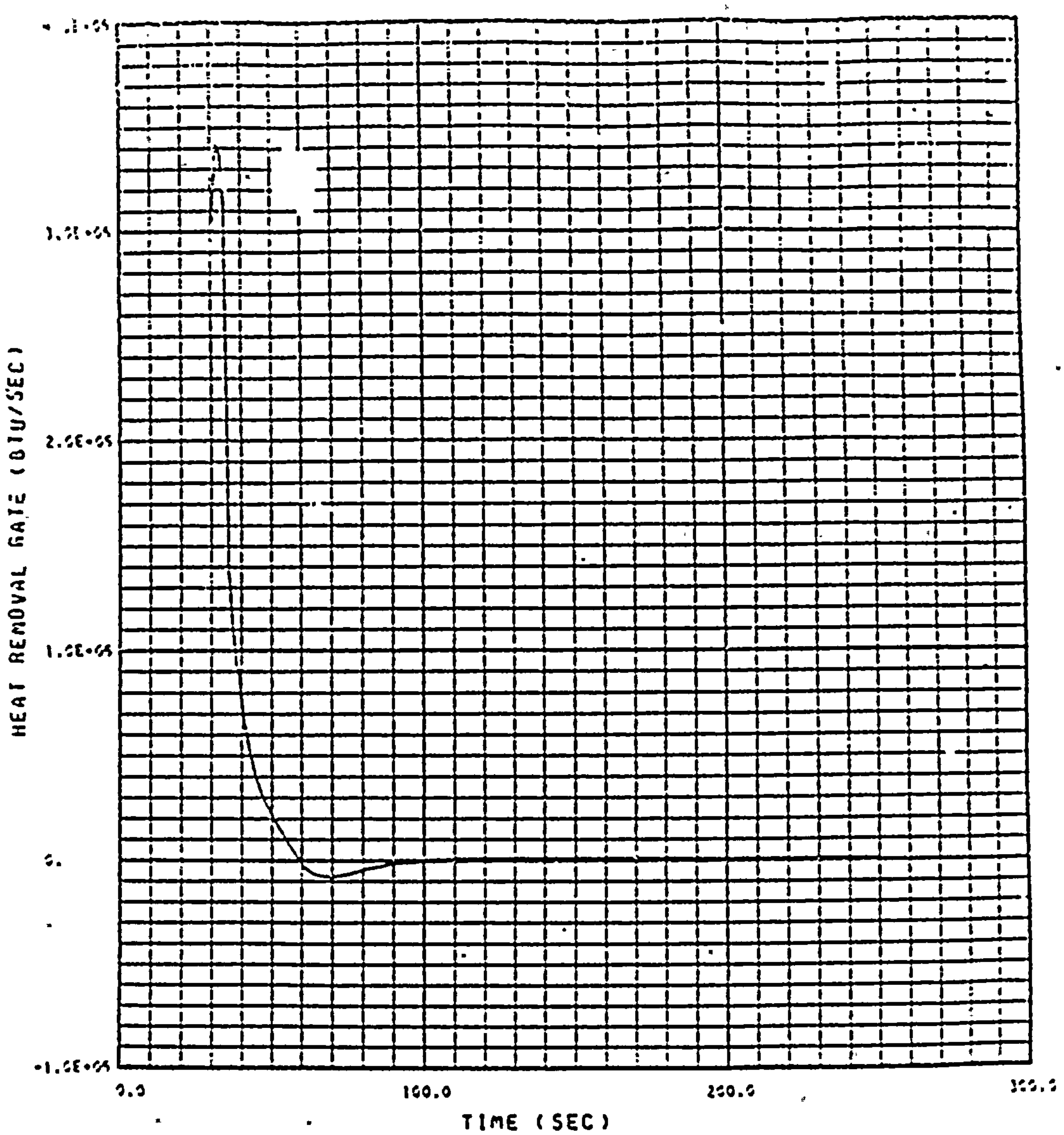
MAXIMUM SI

FIGURE 14.3.1-56. LOWER COMPARTMENT STRUCTURAL HEAT REMOVAL RATE



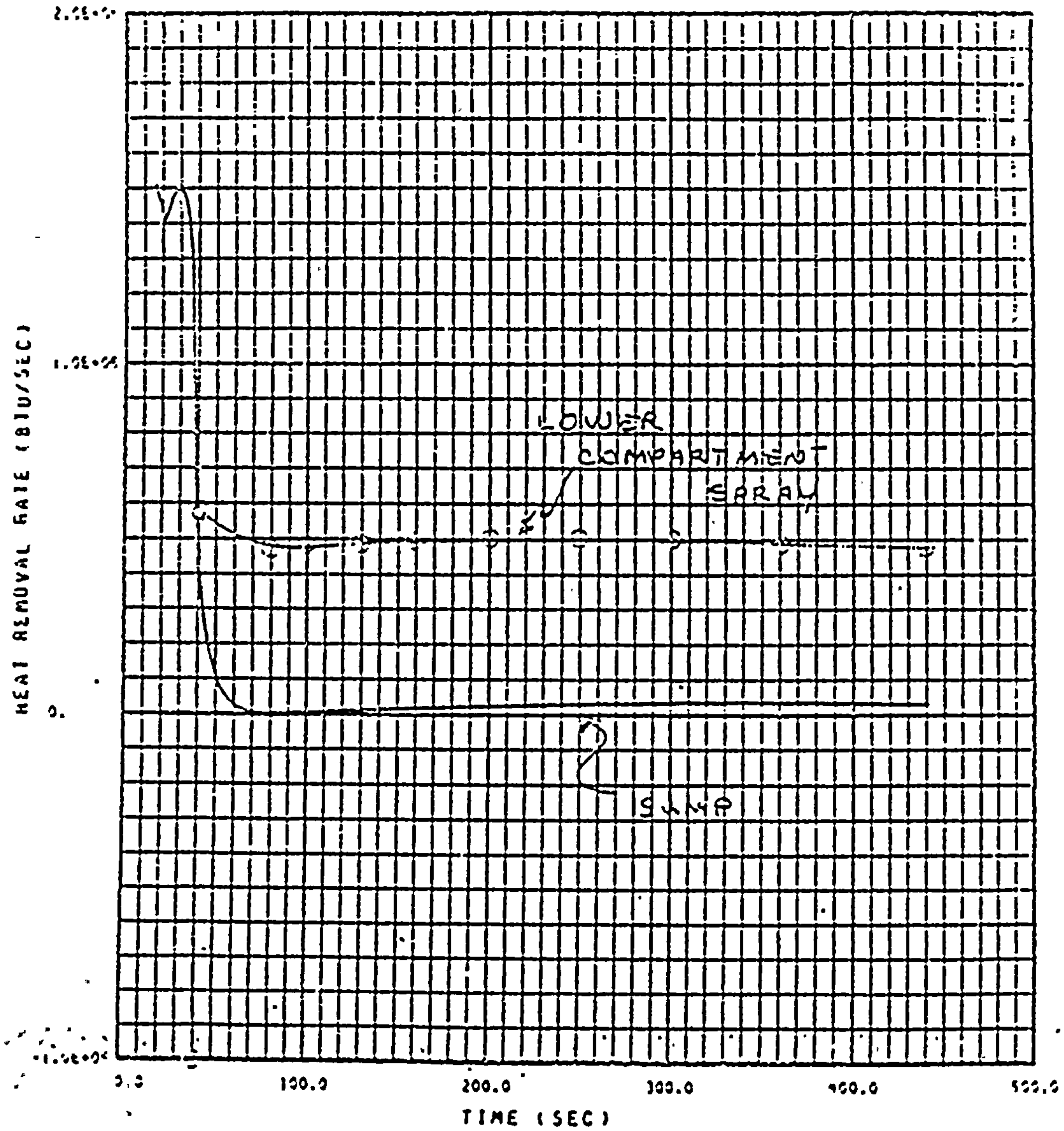
MINIMUM SI

FIGURE 14.3.1-57 HEAT REMOVAL BY LC DRAIN



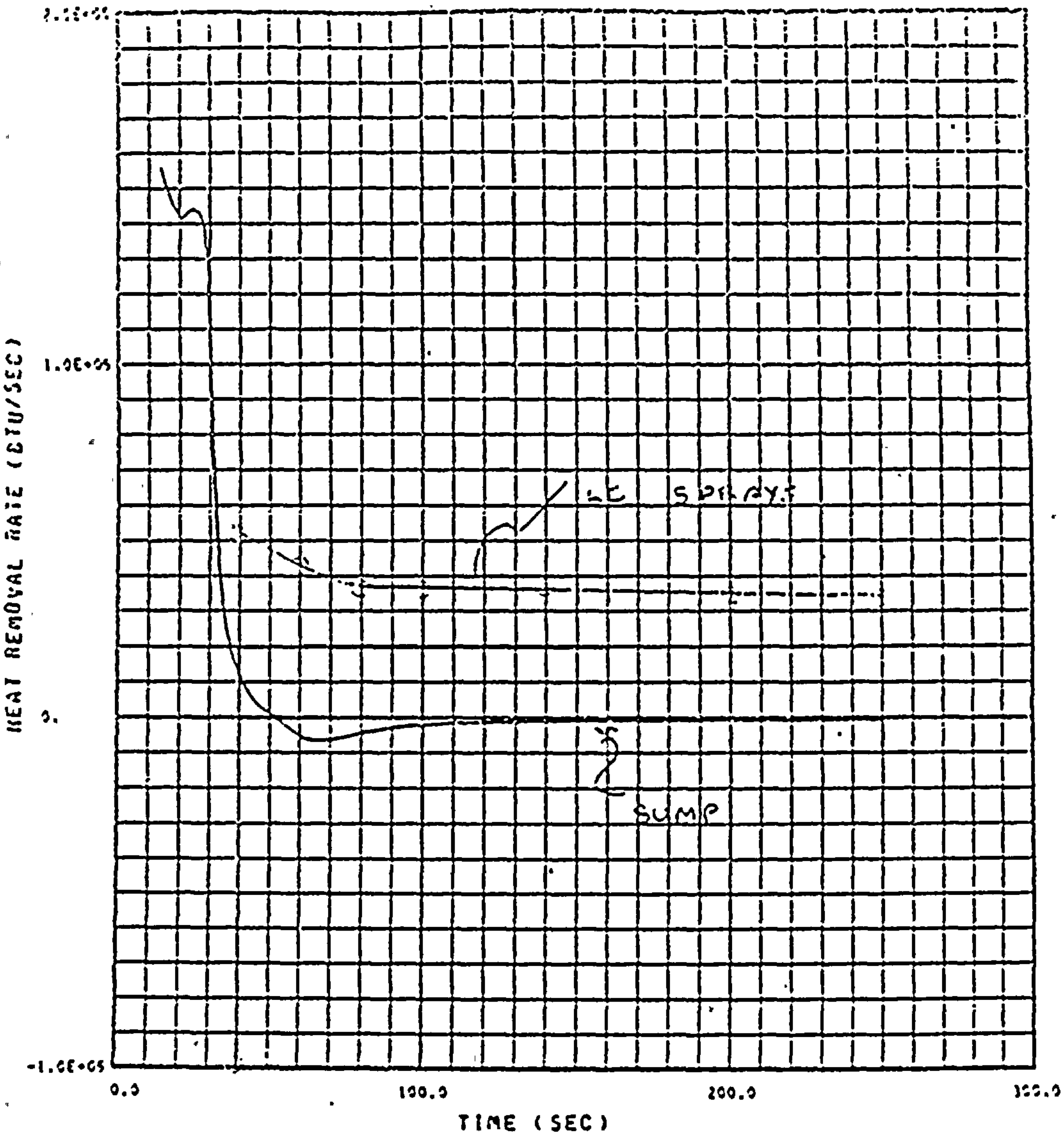
MAXIMUM SI

FIGURE 14.3.1-58 HEAT REMOVAL BY LC DRAIN

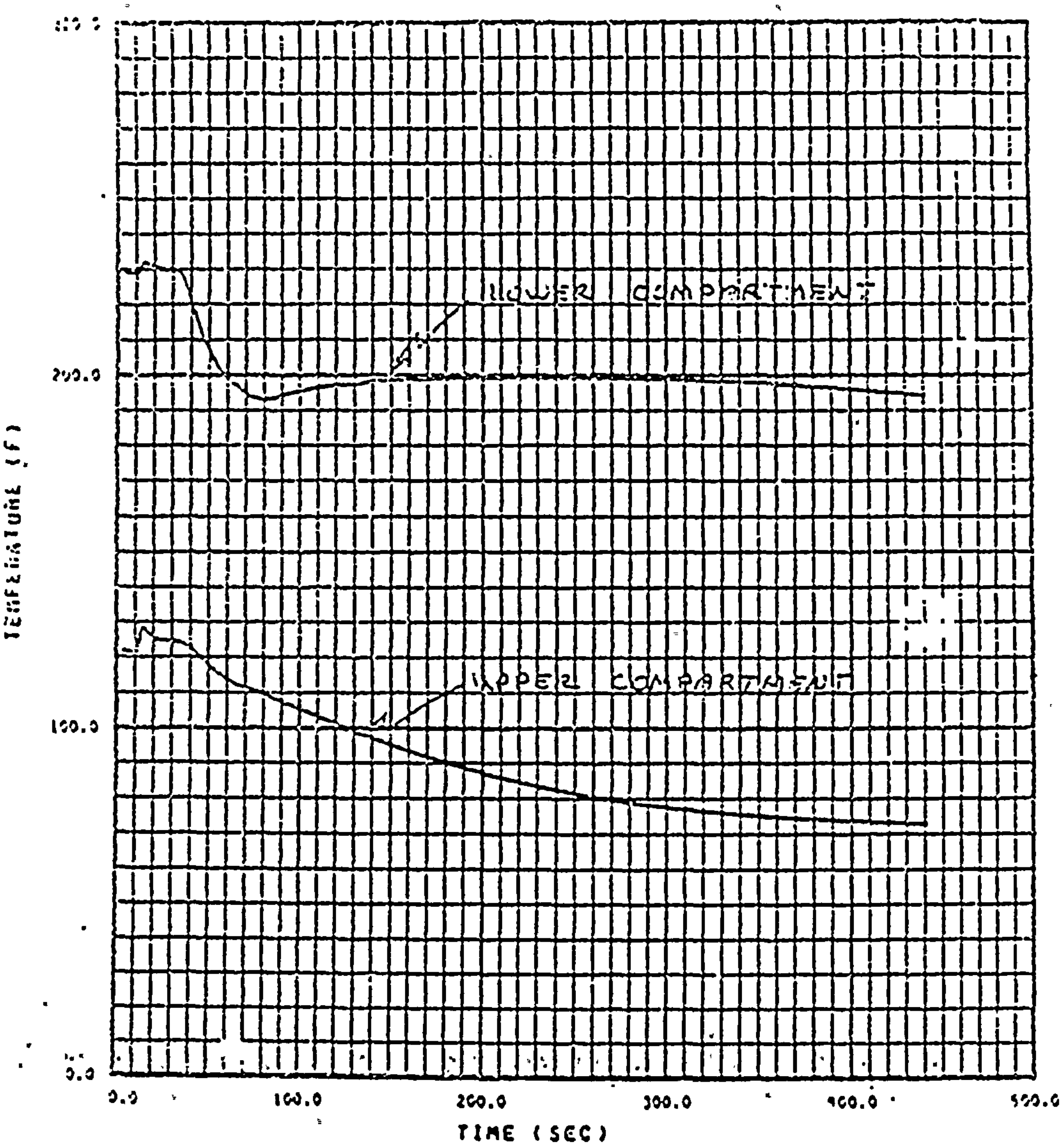


MINIMUM SI

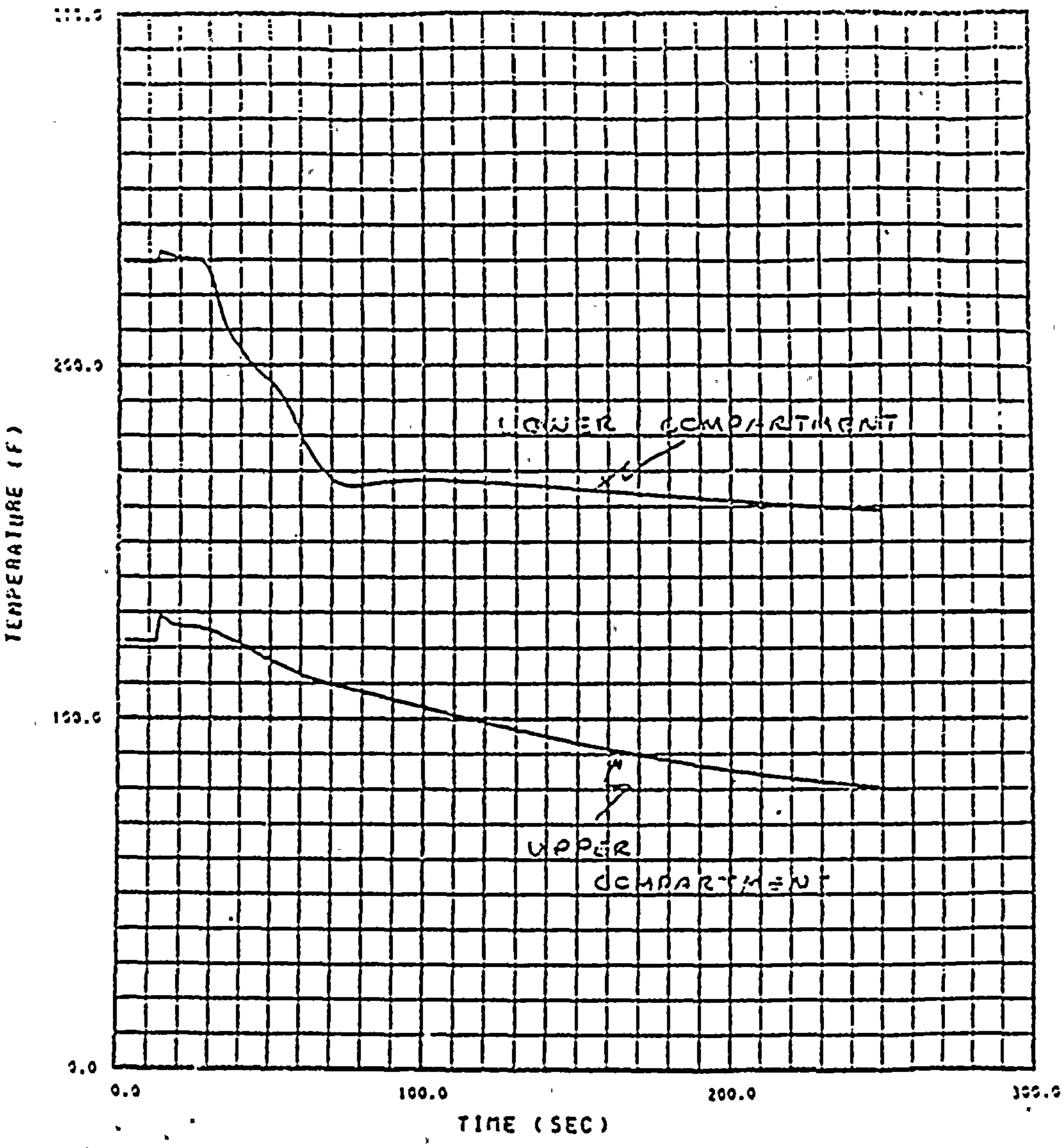
FIGURE 14.3.1-59 HEAT REMOVAL BY SUMP AND L.C SPRAY



MAXIMUM SI
FIGURE 14.3.1-60 HEAT REMOVAL BY SUMP AND LC SPRAYS



MINIMUM SI
FIGURE 14.3.1-61 COMPARTMENT TEMPERATURE



MAXIMUM SI

FIGURE 14.3.1-62 COMPARTMENT TEMPERATURE

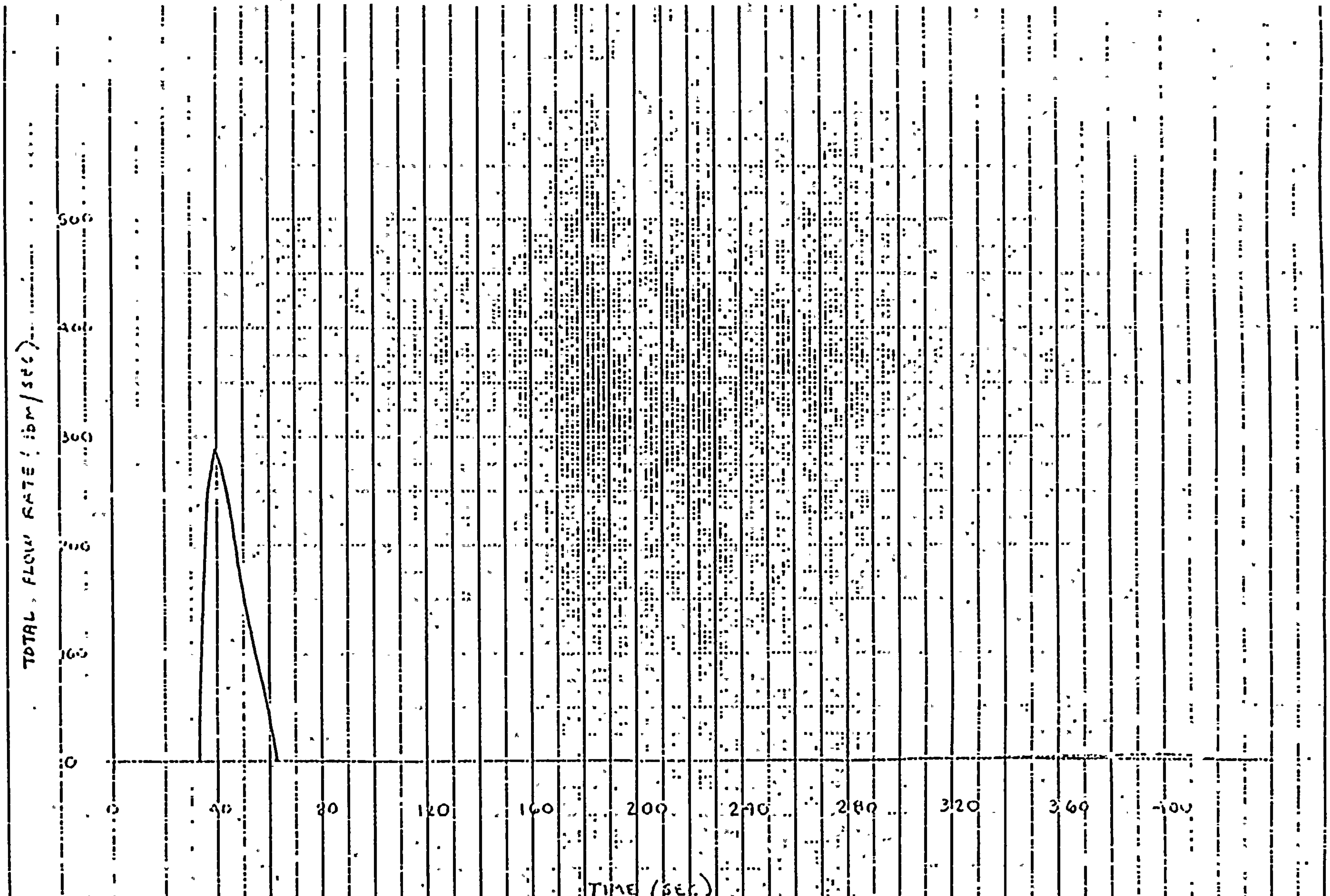


FIGURE 14.3.1-63 FLOW FROM UPPER TO LOWER COMPARTMENT

MINIMUM SI

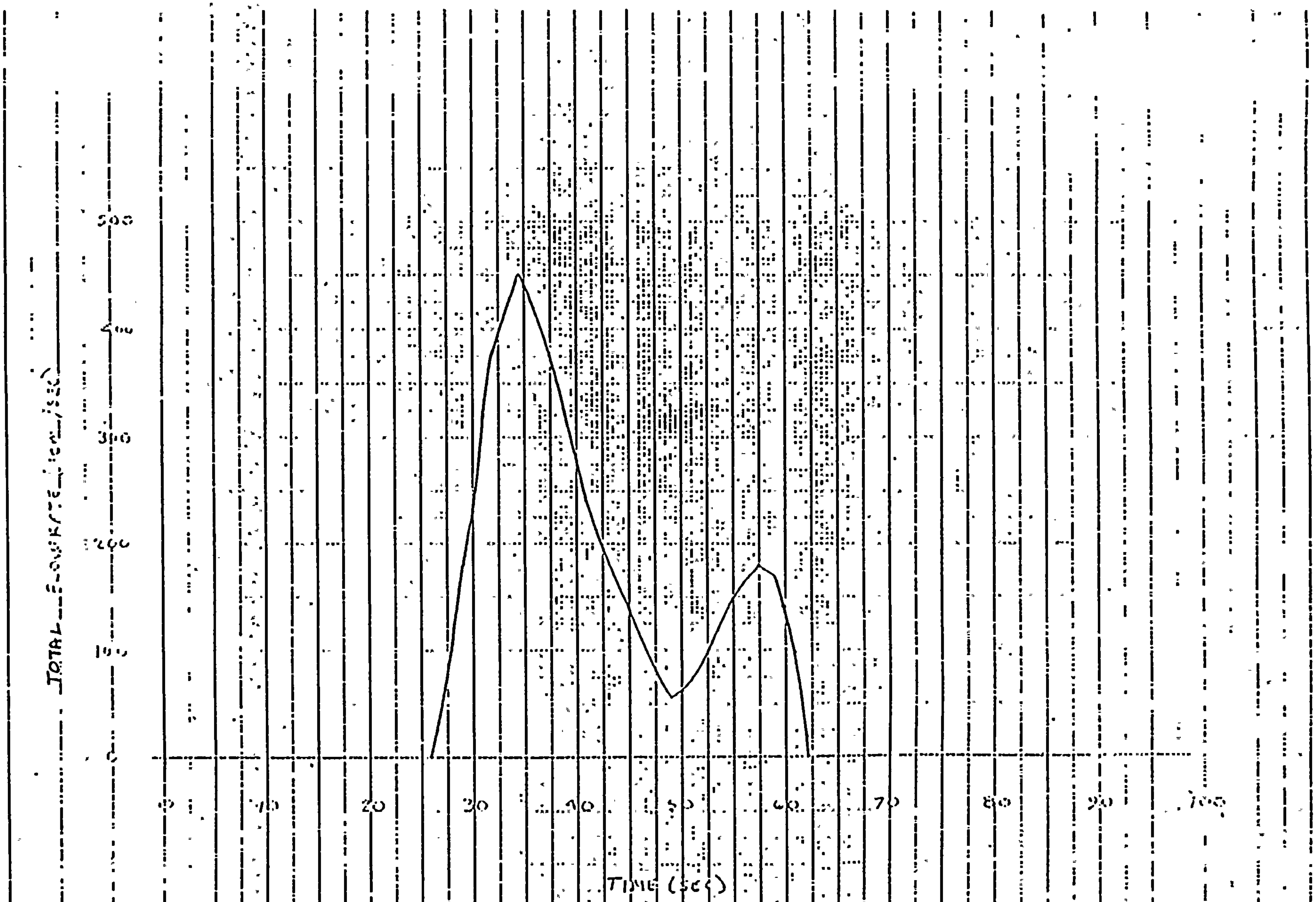
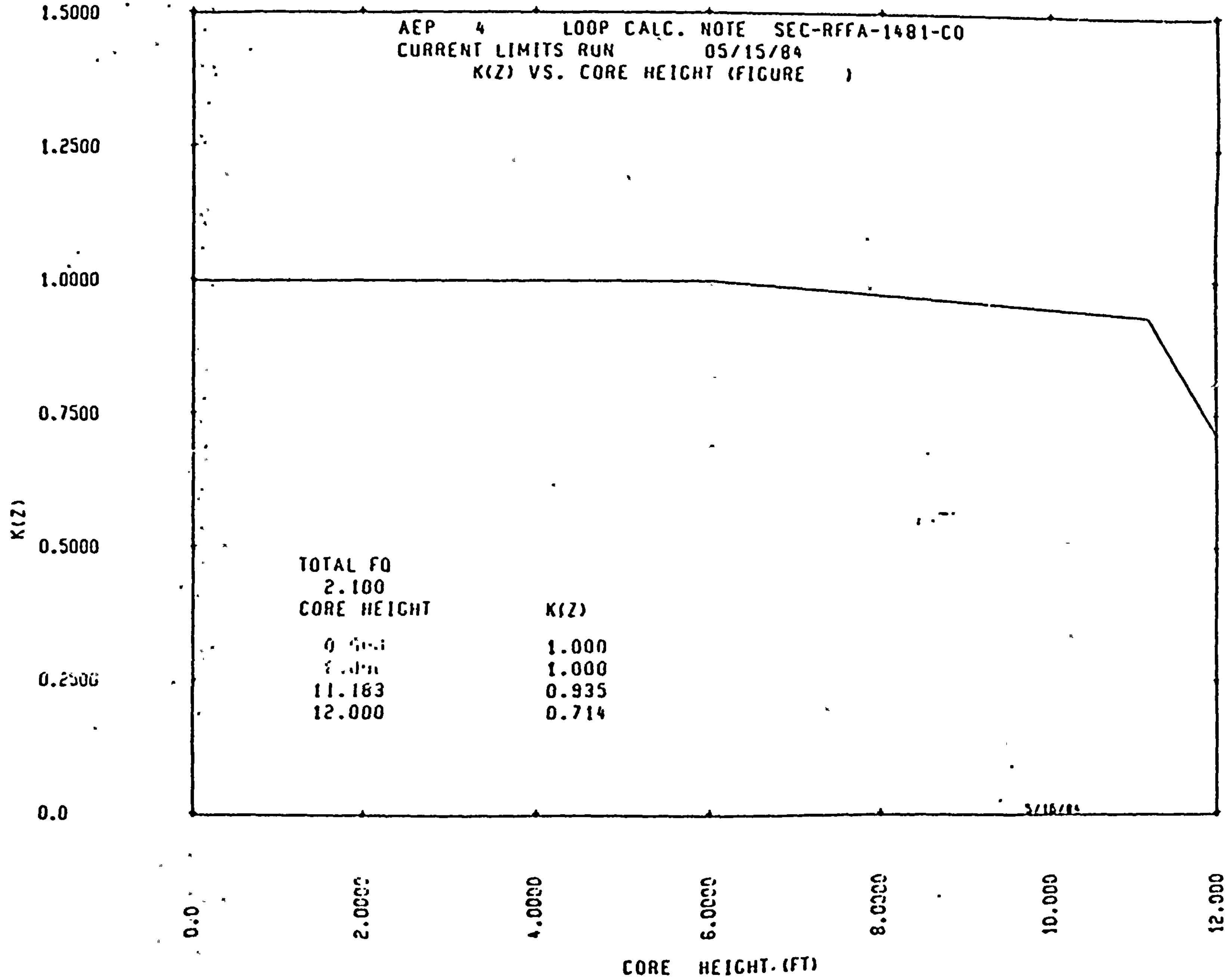


FIGURE 14.3.1-64 FLOW FROM UPPER TO LOWER COMPARTMENT
MAXIMUM SI

ATTACHMENT B

LOCA RELATED TECH SPECS

Plant Name:	Donald C. Cook Unit 1 (AEP)
Type/Date of LOCA Analysis	D_C = ~~0.4~~ 0.6 (max SI case) Large Break LOCA Analysis ~~7/83~~ 5/84
Total Peaking Factor F_Q^T:	~~1.97~~ 2.10
Cold Leg Accumulator Water Volume:	950 ft^3/accumulator (nominal) unchanged from current tech specs
Cold Leg Accumulator Gas Pressure:	600 psia (minimum) unchanged from current tech specs
K(z) Curve:	See next page


AEP 4 LOOP CALC. NOTE SEC-RFFA-1481-CO
CURRENT LIMITS RUN 05/15/84
K(Z) VS. CORE HEIGHT (FIGURE)
1.5000
1.2500
1.0000
0.7500
0.5000
0.2500
0.0
K(Z)
TOTAL FQ
2.100
CORE HEIGHT K(Z)
1.000
1.000
11.163 0.935
12.000 0.714
0.0
2.0000
4.0000
6.0000
8.0000
10.000
12.000
CORE HEIGHT (FT)