ATTACHMENT A

DEMONSTRATION OF THE CONFORMANCE
TO THE 10CFR50.46 ACCEPTANCE CRITERIA
FOR THE LARGE BREAK LOSS-OF-COOLANT ACCIDENT
FOR THE
H. B. ROBINSON UNIT 2 NUCLEAR POWER PLANT
WITH EXXON NUCLEAR COMPANY FUEL

WESTINGHOUSE ELECTRIC CORPORATION
NUCLEAR TECHNOLOGY DIVISION
NUCLEAR SAFETY DEPARTMENT
SAFEGUARDS ENGINEERING AND DEVELOPMENT

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I. INTRODUCTION

A safety evaluation was performed to justify the resumption of operation of the H. B. Robinson Unit 2 nuclear power plant with Exxon fuel to 100% of the licensed core power level of 2300 MWt. The evaluation was performed using the Westinghouse 1981 Evaluation Model with BART for a double-ended guillotine break of the cold leg with a discharge coefficient of 0.4. The results of the analysis demonstrate conformance with the requirements of the 10CFR50.46 acceptance criteria.

II. METHOD OF ANALYSIS

As a technical basis for the safety evaluation, an analysis of a postulated large break LOCA was performed for H. B. Robinson Unit 2 using the 1981 Evaluation Model with BART. In order to perform the analysis on an expedited basis, the analysis was performed using an input model that was previously assembled for evaluation of Florida Power and Light's Turkey Point Unit 3 nuclear power plant as a basis. The H. B. Robinson and Turkey Point Unit 3 design parameters are compared in Table 1. This comparison indicates the similarities of the physical layout and characteristics of the two plants' Nuclear Steam Supply Systems (NSSS), and that the control volume representation of Turkey Point's NSSS represents the H. B. Robinson NSSS, when modified with the following plant specific items. The input was modified to incorporate H. B. Robinson's power level, primary system operating conditions, Emergency Core Cooling System (ECCS), steam generator secondary side pressure and steam flow inputs, and fuel. The information used to model the fuel was supplied by the Exxon Nuclear Company.

The mathematical model used was the Westinghouse 1981 Evaluation Model with BART, which has been approved for use by the NRC as meeting the requirements of an acceptable ECCS Evaluation Model as presented in Appendix K of 10CFR50. This evaluation model is comprised of the SATAN-VI, WREFLOOD, COCO, BART and LOCTA-IV codes, which are described in References 1-7. These codes assess the core heat transfer 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 code is employed for the thermal-hydraulic transient during blowdown, while the WREFLOOD code computes this transient during refill and reflood. The COCO code is used for the complete containment pressure history for dry containments. Reflood thermal-hydraulic conditions are supplied to the BART code which performs the heat transfer calculation for the average fuel channel in the hot assembly using a mechanistic core heat transfer model. This information is then used by LOCTA-IV to calculate the fuel clad temperature and metal-water reaction of the hottest rod in the core. Additional information on the Westinghouse Evaluation Model and methodology is in References 8-13.

Based on past Westinghouse analyses for both plants, a double-ended guillotine break of the cold leg with a discharge coefficient of 0.4 was selected for the evaluation. This was the limiting break size not only for the last Westinghouse analysis performed for H. B. Robinson, which used the October 1975 Evaluation Model, but was also limiting for the Turkey Point Unit 3 analyses, including analyses using the October 1975 Evaluation Model and the 1981 Evaluation Model with BART.

The analysis was performed assuming a chopped cosine power shape, which peaked at the six foot elevation. This power shape has been determined to be the limiting power shape for large break LOCA analyses using Westinghouse Evaluation Models (Reference 11). Recent sensitivity studies have demonstrated that the chopped cosine power shape is also limiting for large break LOCA analysis for Exxon fuel using Westinghouse Evaluation Models. These studies have shown that the calculated peak cladding temperature predicted for the Exxon fuel using Westinghouse methodology is over one hundred degrees higher for a chopped cosine power shape than it is for power shapes that are skewed toward the top of the core. While these studies were not performed for a 3-loop plant, there is no reason to believe that H. B. Robinson Unit 2 would perform any differently.

III. RESULTS

Table 2 presents the peak clad temperature and hot spot metal reaction for the $C_d = 0.4$ break size. The calculated PCT was 2199°F occurring at 108 seconds at an elevation of 6.0 feet relative to the bottom of the active core. The maximum local metal-water reaction was 7.09 percent, which is well below the embrittlement limit of 17 percent, as required by 10CFR50.46. The analysis was performed at 102 percent of the licensed core power of 2300 MWt at the total peaking factor of 2.32 and enthalpy rise factor of 1.65. Table 3 presents the time sequence of events for the large break LOCA. Figures 1-16 present the transients for the principal parameters for the break analyzed.

IV. CONCLUSIONS

This analysis demonstrates that the H. B. Robinson Unit 2 nuclear power plant with Exxon fuel operating at 100% power, with the 2.32 $F_Q T$ and 1.65 $F_{\Delta H}$ limits, conforms to the Acceptance Criteria as presented in 10CFR50.46 when analyzed with the Westinghouse 1981 Evaluation Model with BART. That is:

1. The calculated peak fuel element clad temperature provides margin to the requirement of 2200°F, based on an $F_Q T$ value of 2.32.

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 clad 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

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

TABLE 1

COMPARISON OF DESIGN PARAMETERS
FOR H. B. ROBINSON UNIT 2 AND TURKEY POINT UNIT 3

PARAMETER	H. B. ROBINSON UNIT 2	TURKEY POINT UNIT 3
Core Power (MW_{th})	2300	2200
Fuel Type	Exxon 15 x 15	W 15 x 15 OFA
Barrel Baffle Design	Downflow	Downflow
Upper Head Temperature	T_{hot}	T_{hot}
Upper Support Plate Design	Flat	Flat
Lower Support Plate Design	Flat	Flat
Steam Generator Type	Model 44F	Model 44F
Pressurizer Volume (ft^3)	1300	1300
Reactor Coolant Pump	Model 93 6000 hp	Model 93 6000 hp
Accumulator Total Volume (ft^3)	1200	1200
Accumulator Gas Pressure, psia	615	615
Thermal Design Flow (GPM)	88,600*	89,500

* At 5% Steam Generator Tube Plugging Level

(3015BAT/ccc)

TABLE 2

LOCA-ECCS ANALYSIS RESULTS

Calculation Basis	
License Core Power, MWt	2300
Power Used for Analysis, MWt*	2346
Peak Linear Power for Analysis, kw/ft*	14.20
Total Peaking Factor, F_Q^T	2.32
Enthalpy Rise, Nuclear, $F^T_{\Delta H}$	1.65
Steam Generator Tube Plugging (%)	5.00
Analysis Results	**C_D = 0.4 DECLG**
Peak Clad Temperature (PCT), °F	2199
Peak Clad Temperature Reached, (sec)	108.
Peak Clad Temperature Location, ft.	6.0
Local Zr/H_2O Reactor (max.), %	7.09
Local Zr/H_2O Location, ft. from Bottom	6.25
Total H_2 Generation, % of Total Zr Reacted	< 0.3
Hot Rod Burst Time, sec.	41.40
Hot Rod Burst Location, ft.	6.25

* Including 1.02 Factor for Power Uncertainties

TABLE 3

LOCA/ECCS TIME SEQUENCE OF EVENTS
C_D = 0.4 DECLG BREAK

Event	Time (sec)
Start	0.0
Safety Injection Signal	0.9
Accumulator Injection	15.1
End-of-Bypass	31.31
Safety Pump Injection	25.9
Bottom-of-Core Recovery	50.11
Accumulators Empty	56.73
Peak Clad Temperature Reached	108.

(3015BAT/ccc)

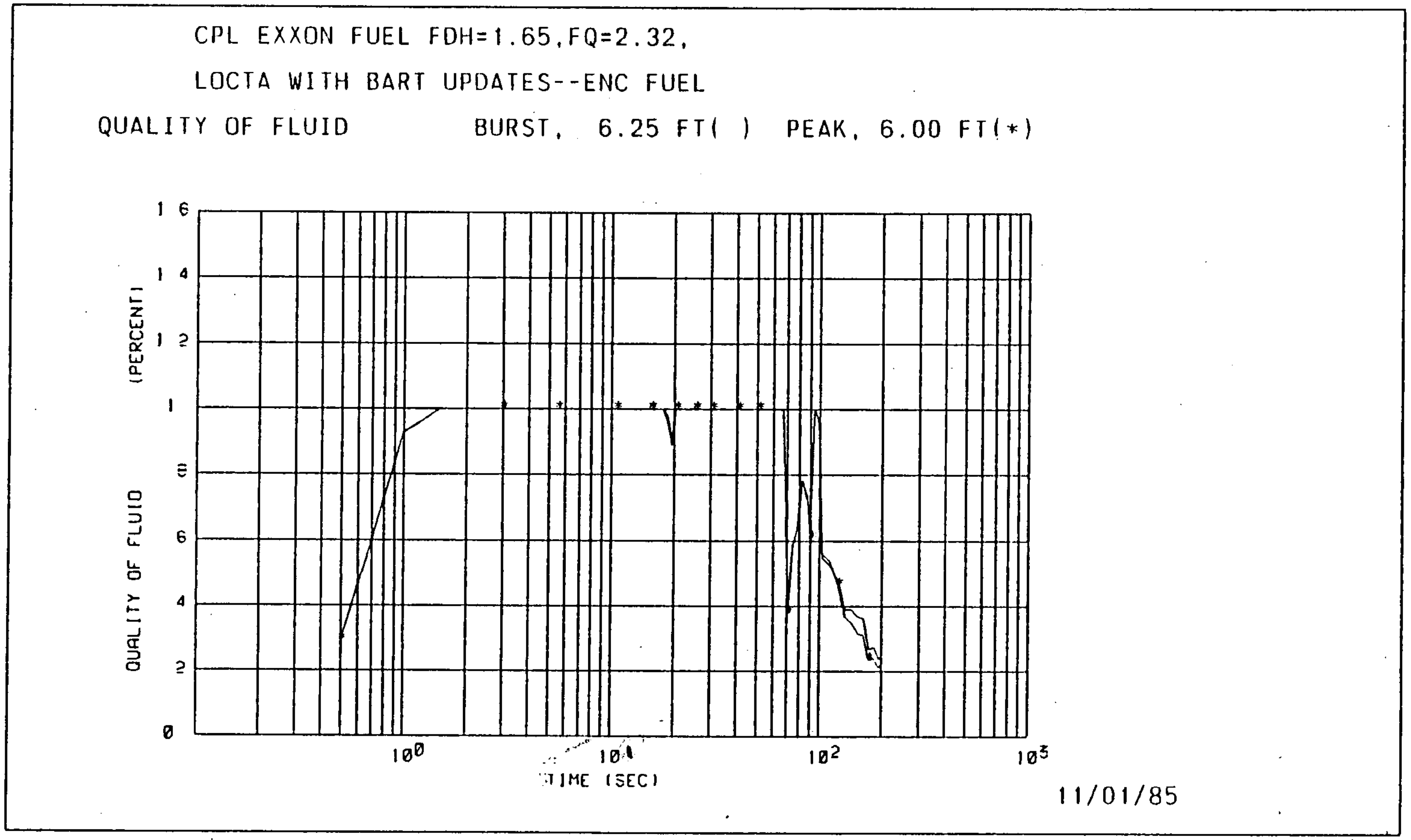

Figure 1. FLUID QUALITY - DECLG (C_D = 0.4)

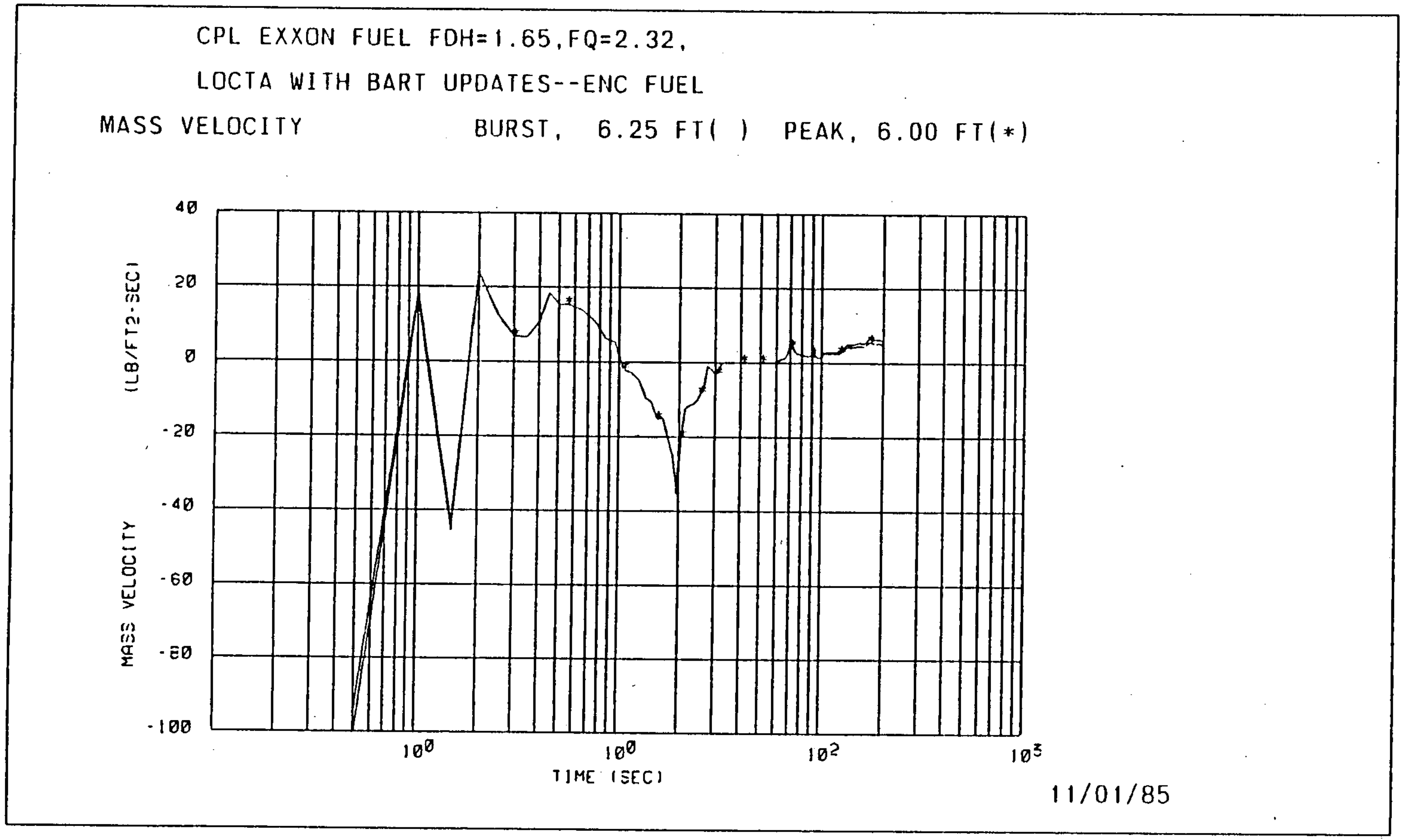

Figure 2. MASS VELOCITY - DECLG (C_D = 0.4)

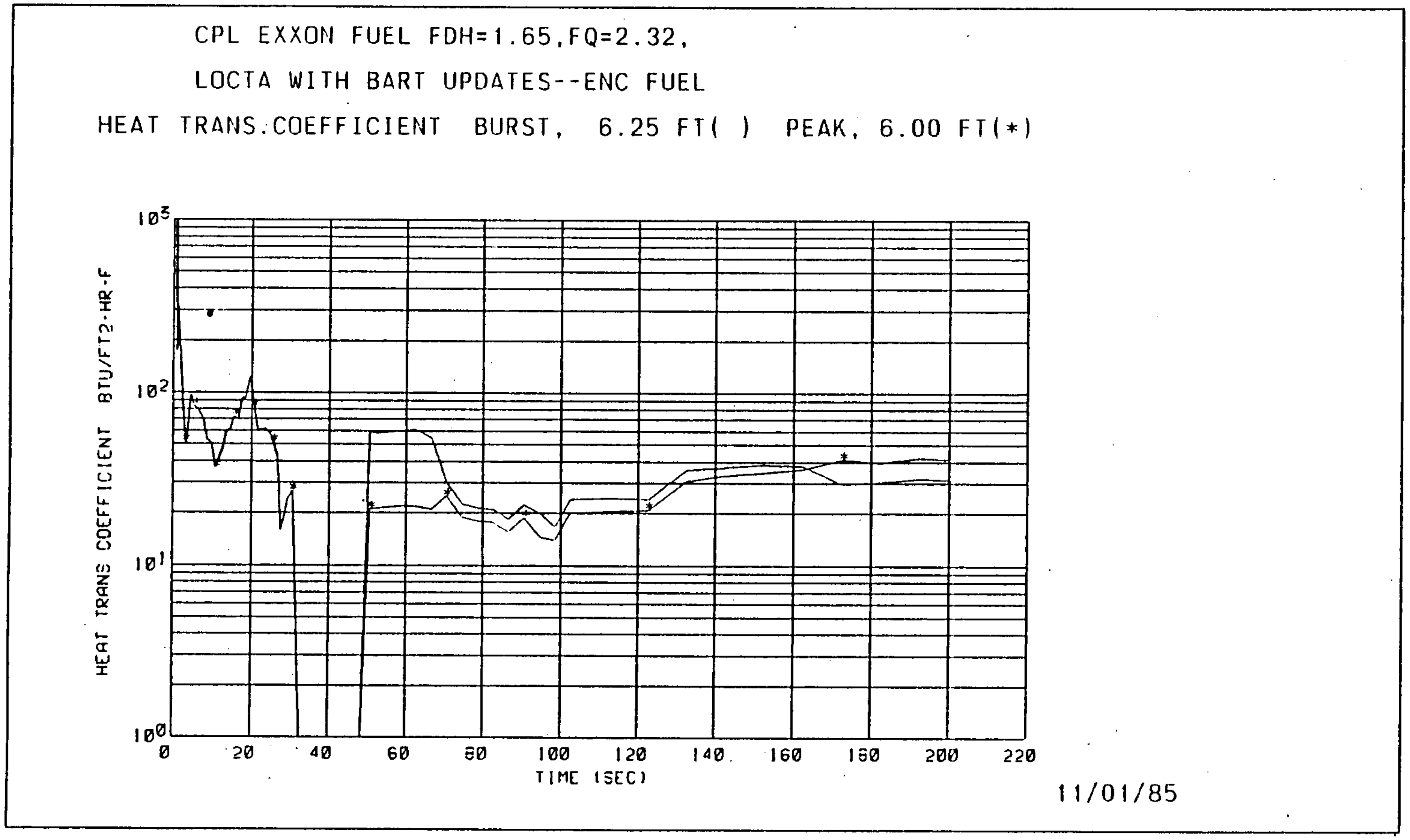

Figure 3. HEAT TRANSFER COEFFICIENT - DECLG (C_D = 0.4)

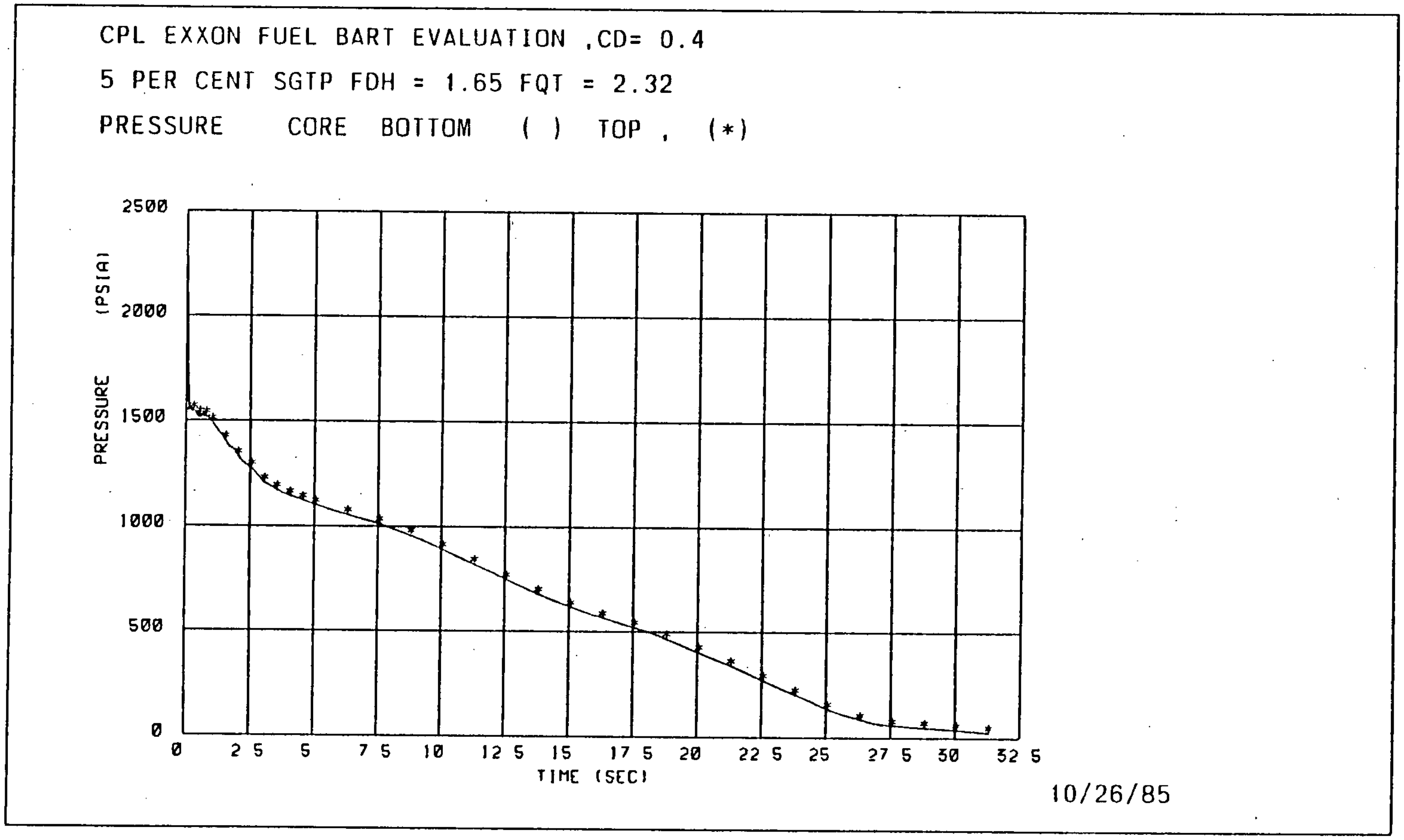

Figure 4. CORE PRESSURE - DECLG (C_D = 0.4)

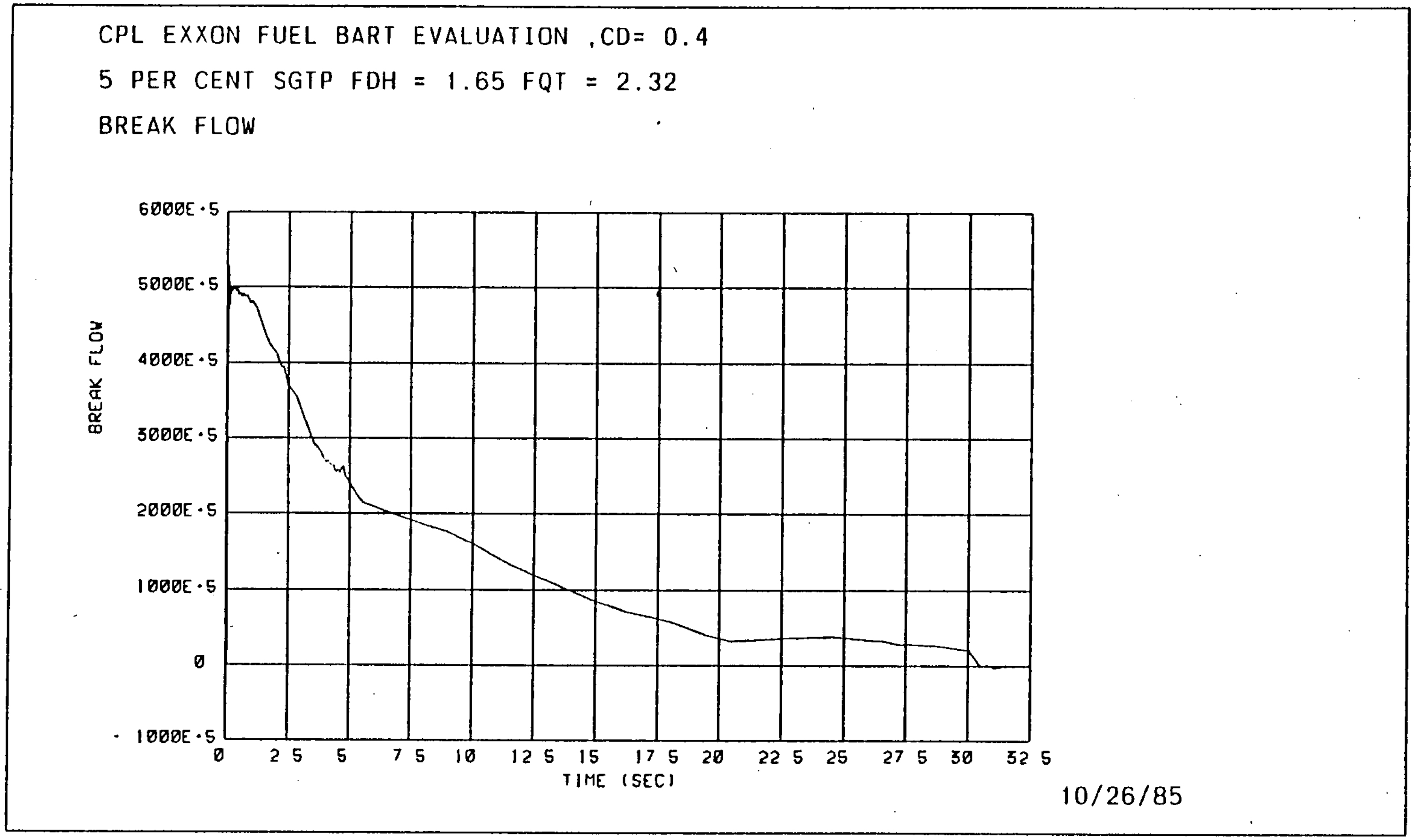

Figure 5. BREAK FLOW RATE - DECLG (C_D = 0.4)

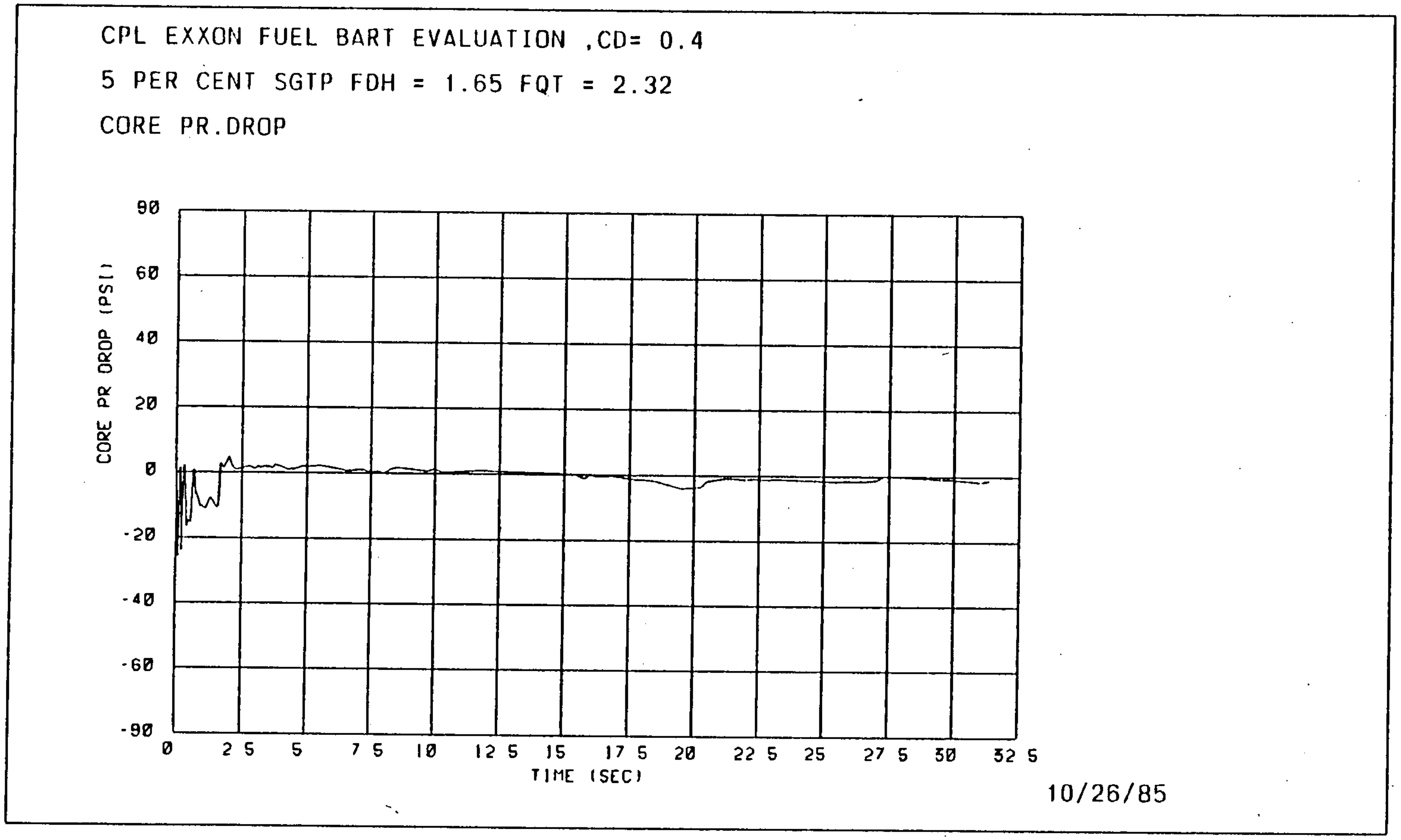

Figure 6. CORE PRESSURE DROP - DECLG (C_D = 0.4)

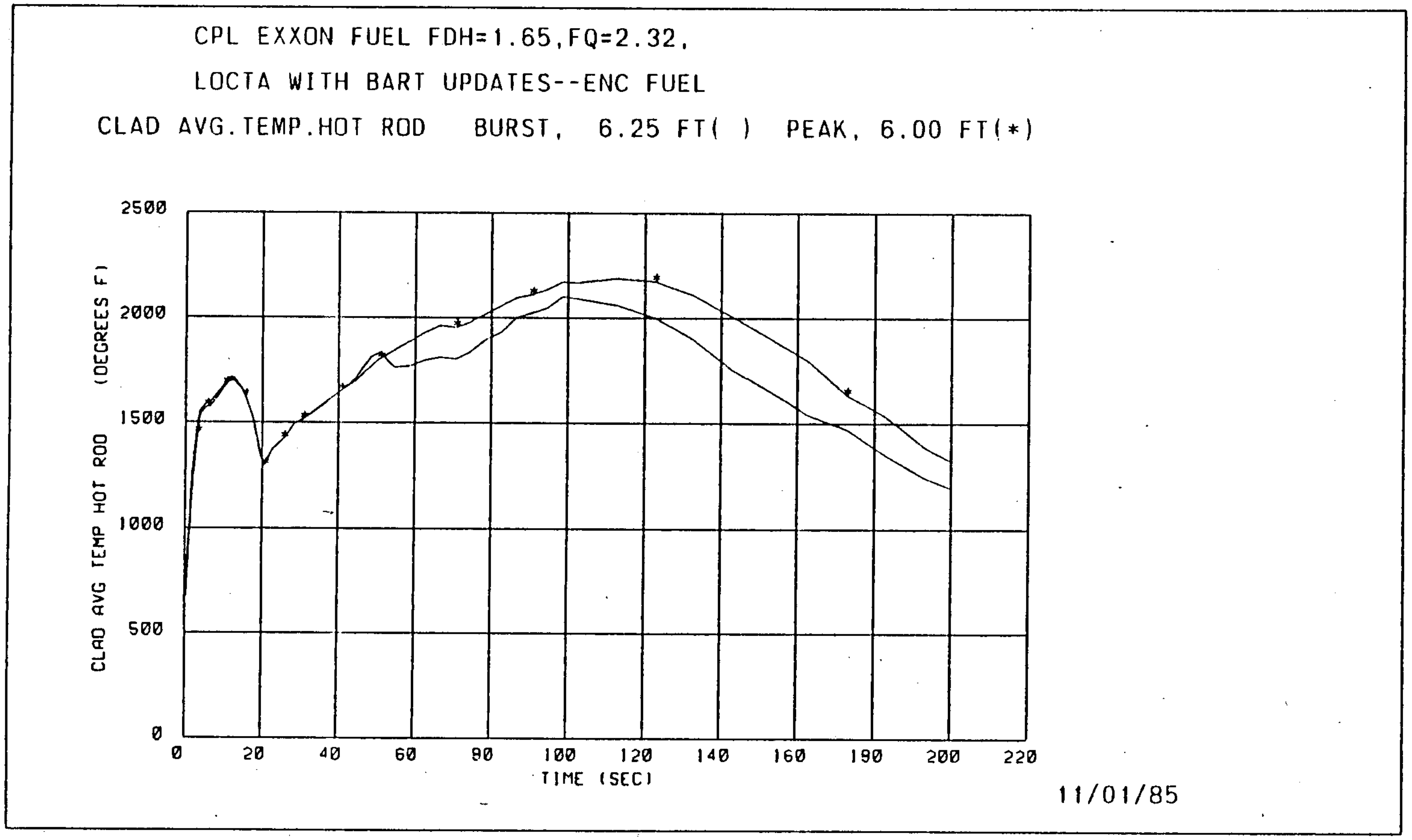

Figure 7. PEAK CLAD TEMPERATURE - DECLG (C_D = 0.4)

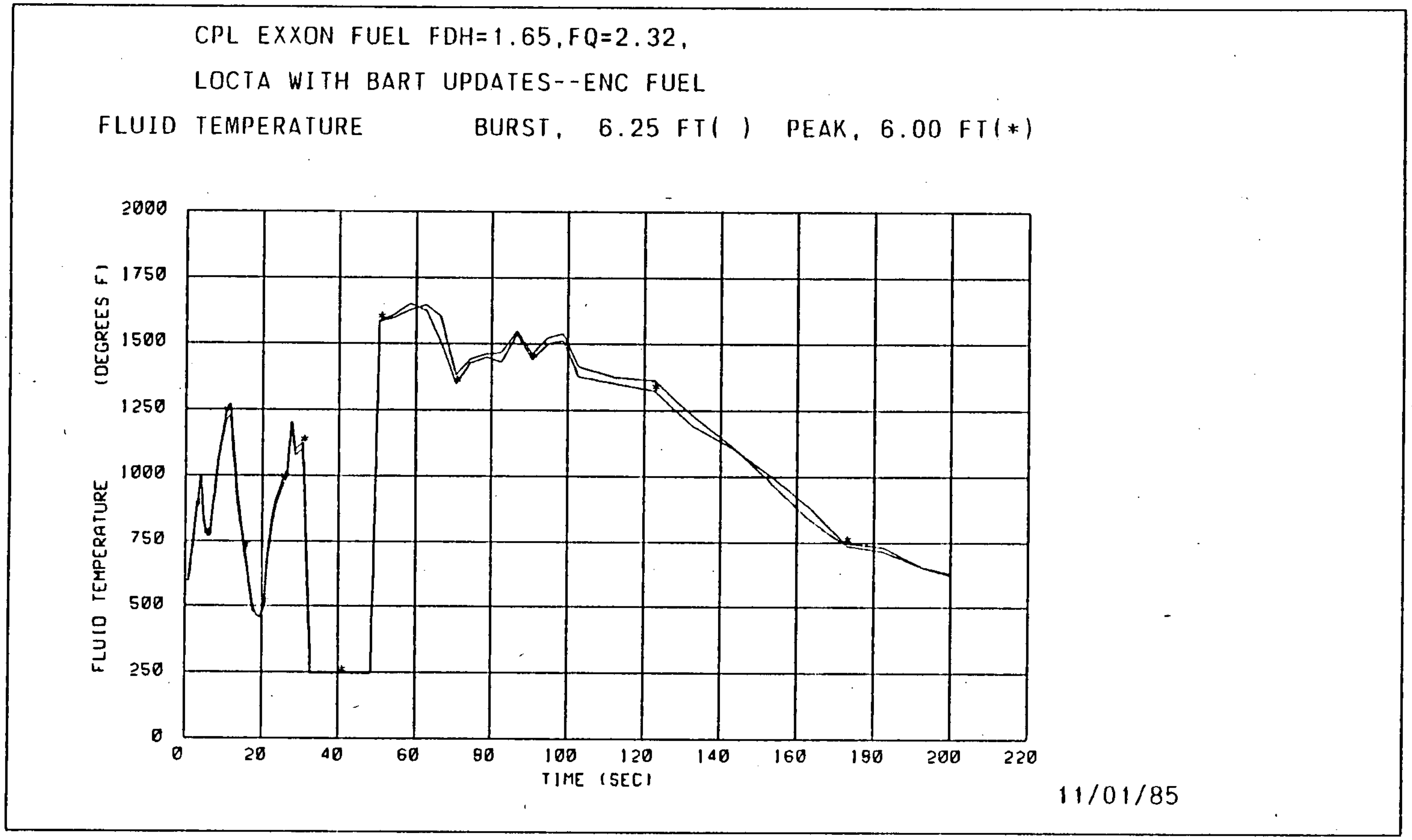

Figure 8. FLUID TEMPERATURE - DECLG (C_D = 0.4)

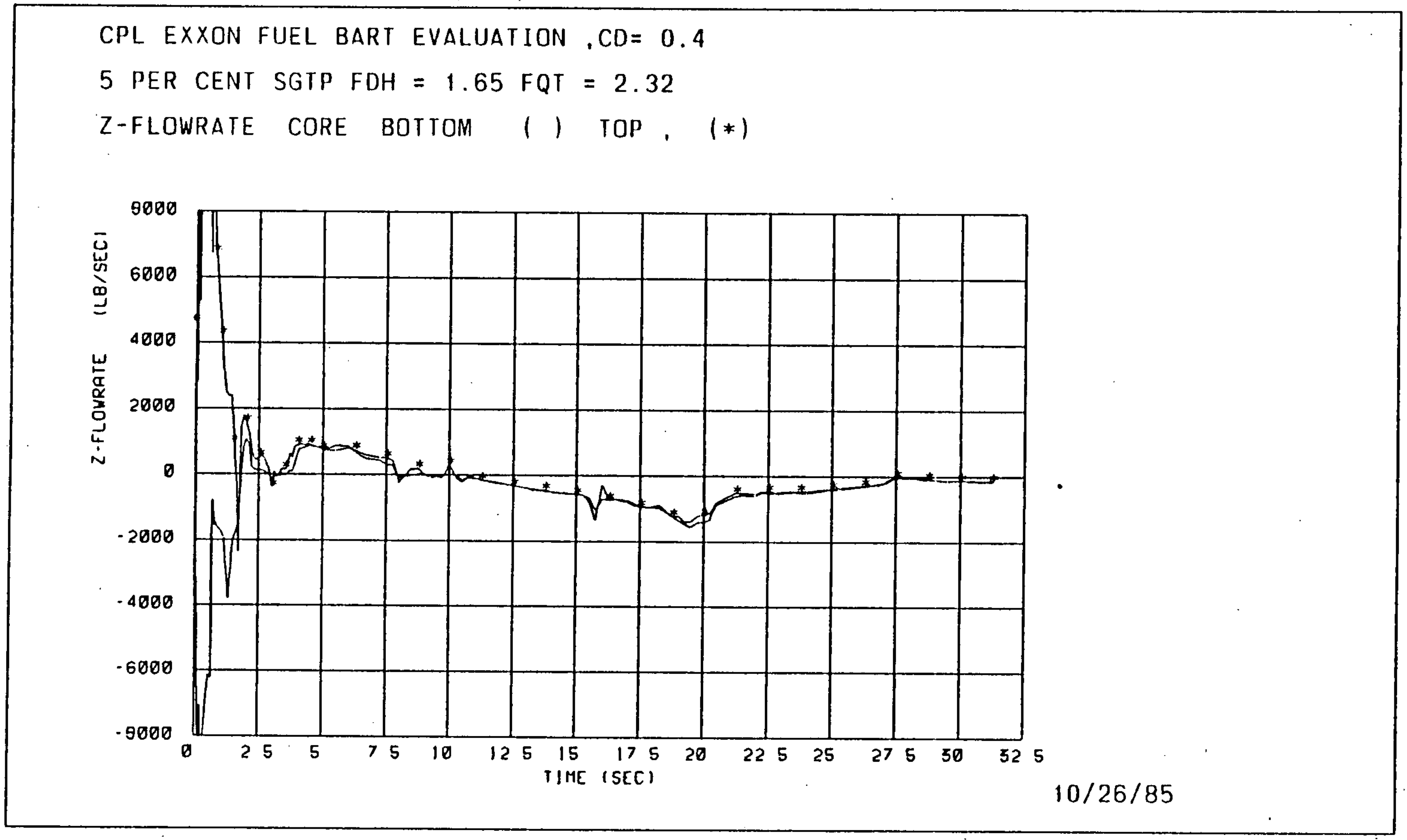

Figure 9. CORE FLOW - TOP AND BOTTOM - DECLG (C_D = 0.4)

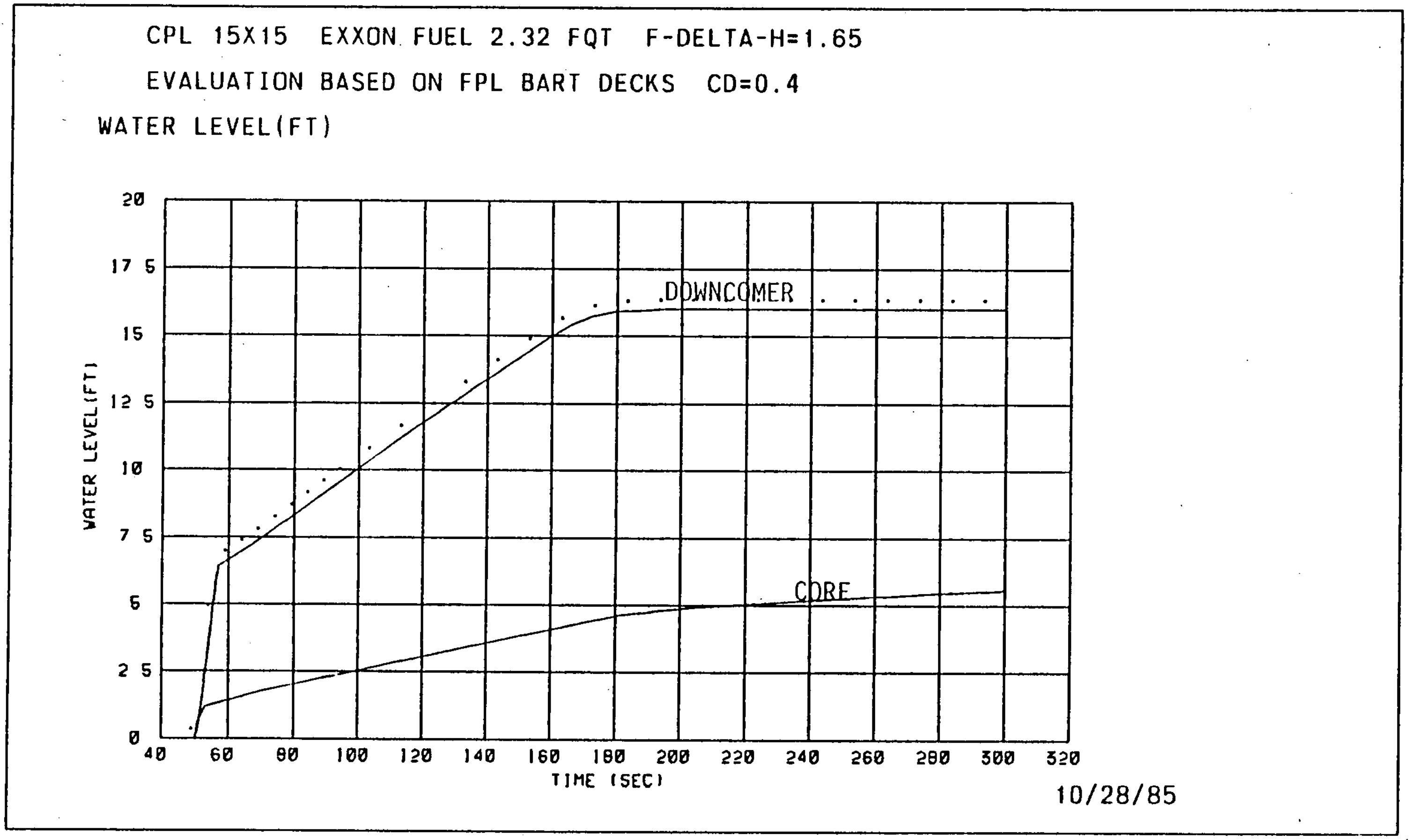

Figure 10. REFLOOD TRANSIENT - DECLG (C_D = 0.4)
DOWNCOMER AND CORE WATER LEVELS

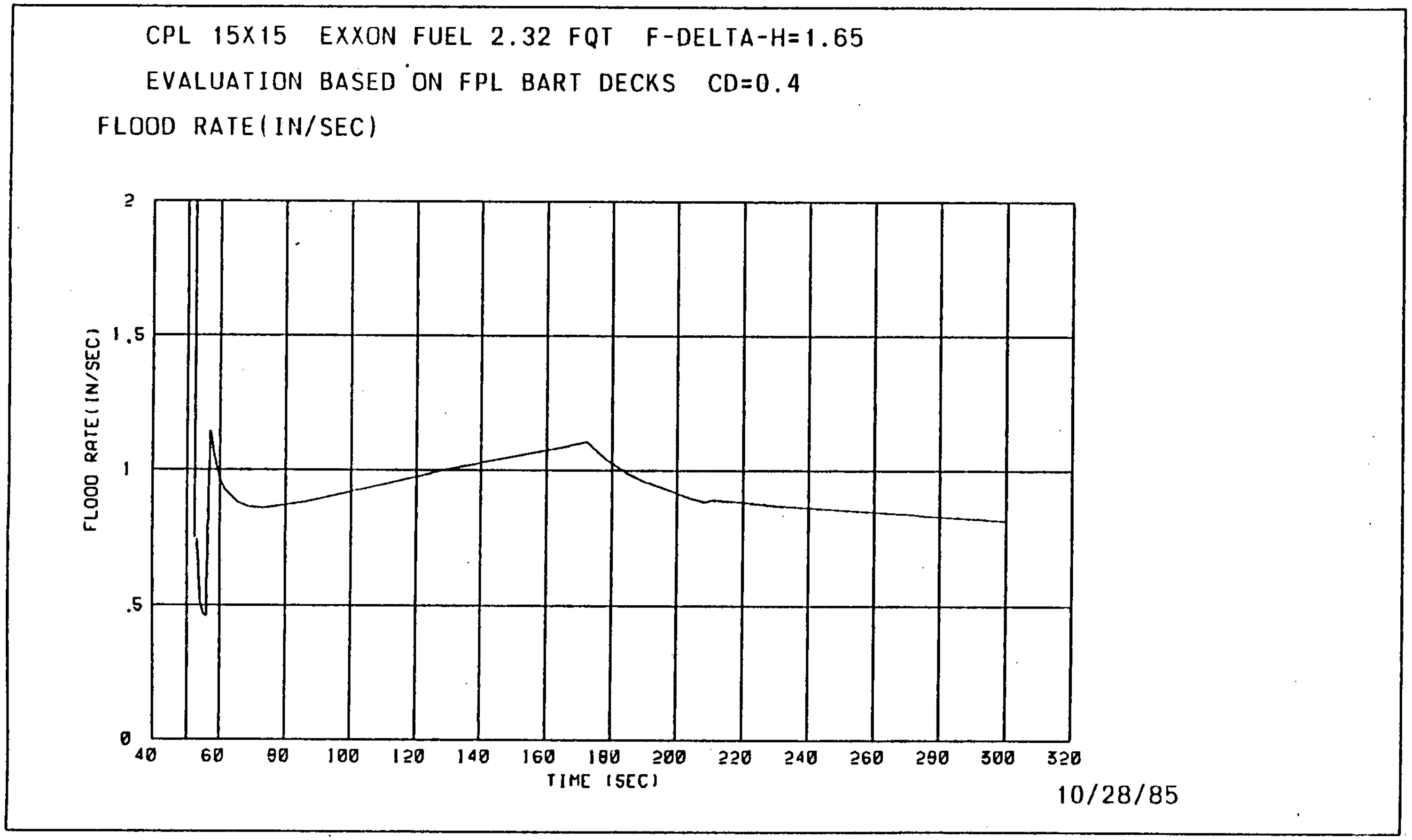

Figure 11. REFLOOD TRANSIENT - DECLG (C_D= 0.4)
CORE INLET VELOCITY

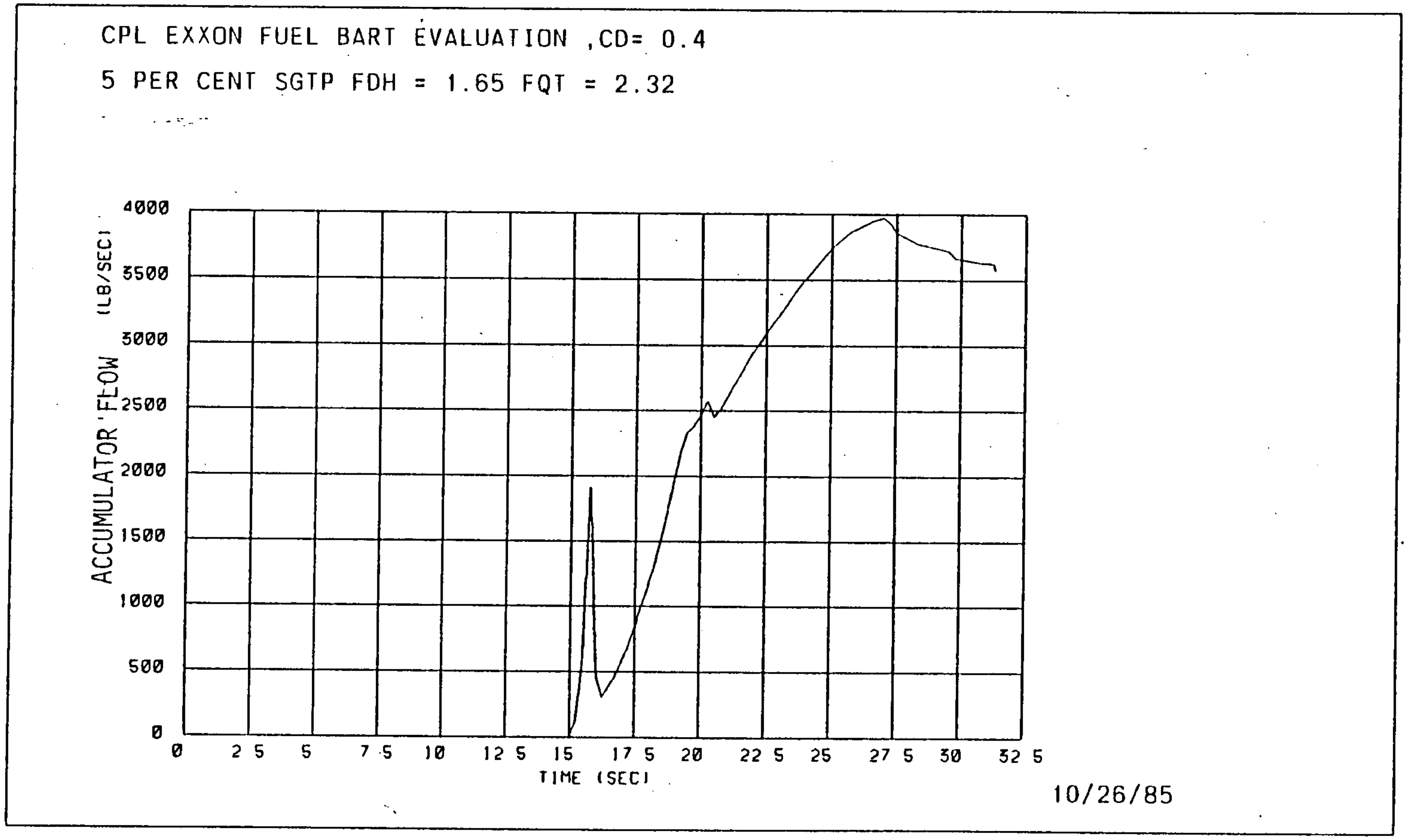

Figure 12. ACCUMULATOR FLOW (BLOWDOWN) - DECLG (C_D = 0.4)

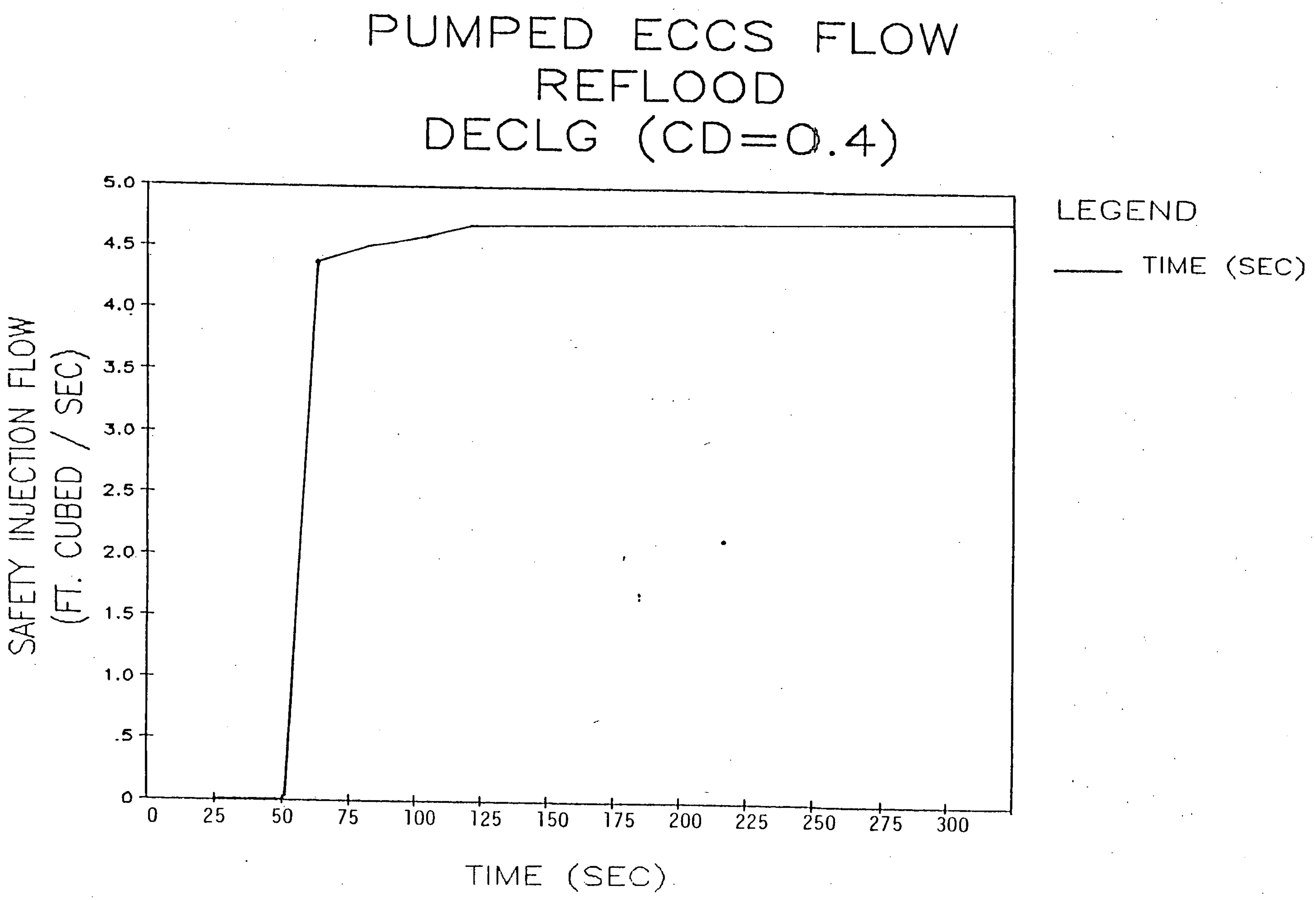

FIGURE 13 PUMPED ECCS FLOW (REFLOOD) DECLG (CD=0.4)

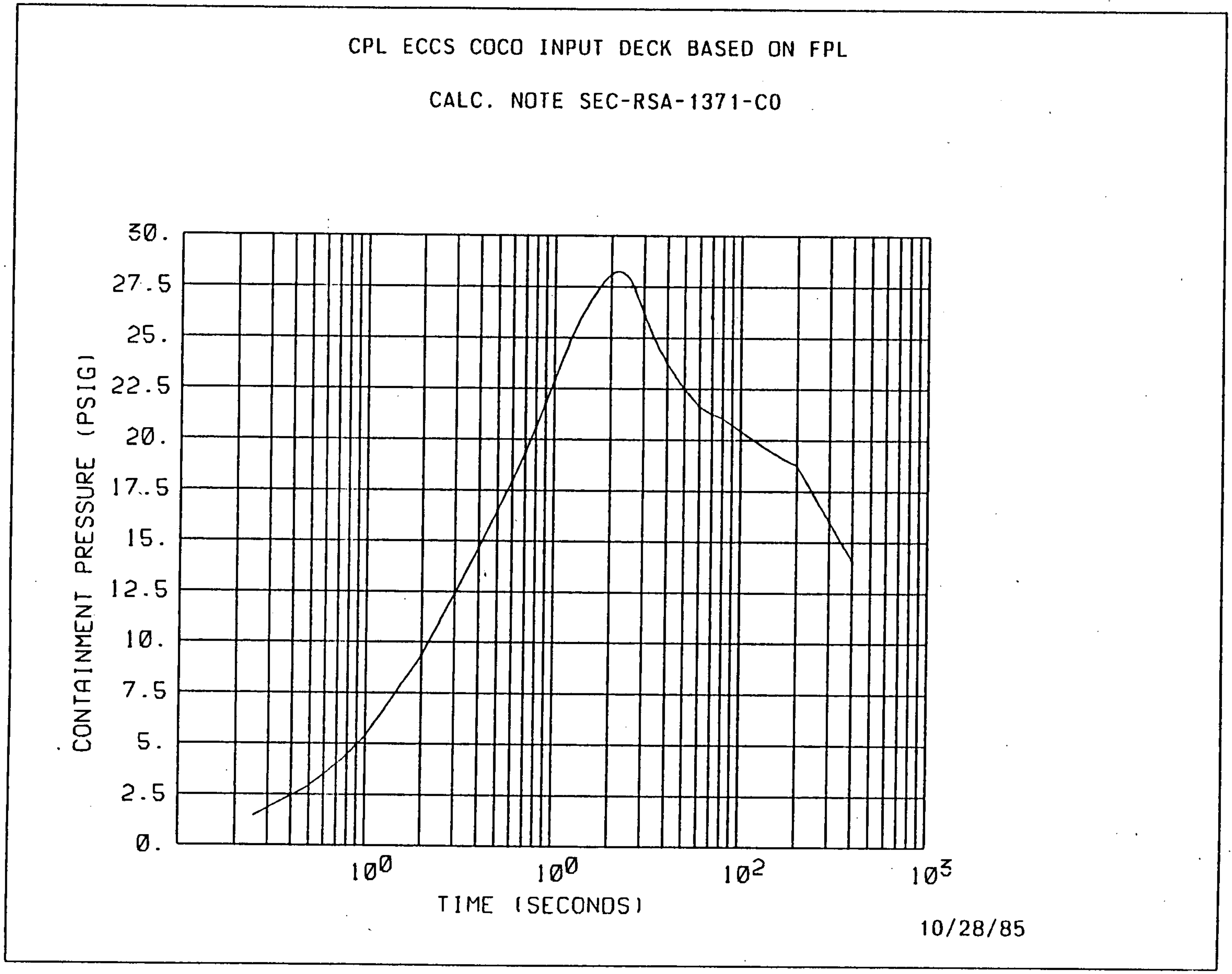

Figure 14. CONTAINMENT PRESSURE - DECLG (C_D = 0.4)

CPL EXXON FUEL BART EVALUATION ,CD= 0.4

5 PER CENT SGTP FDH = 1.65 FQT = 2.32

POWER

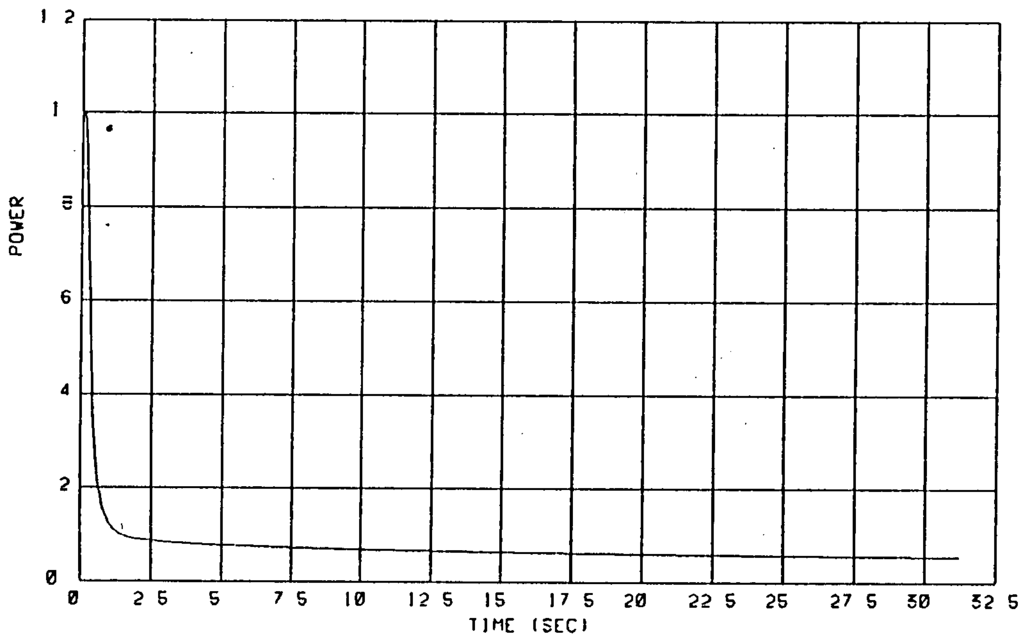

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Figure 15. CORE POWER TRANSIENT - DECLG (C_D = 0.4)

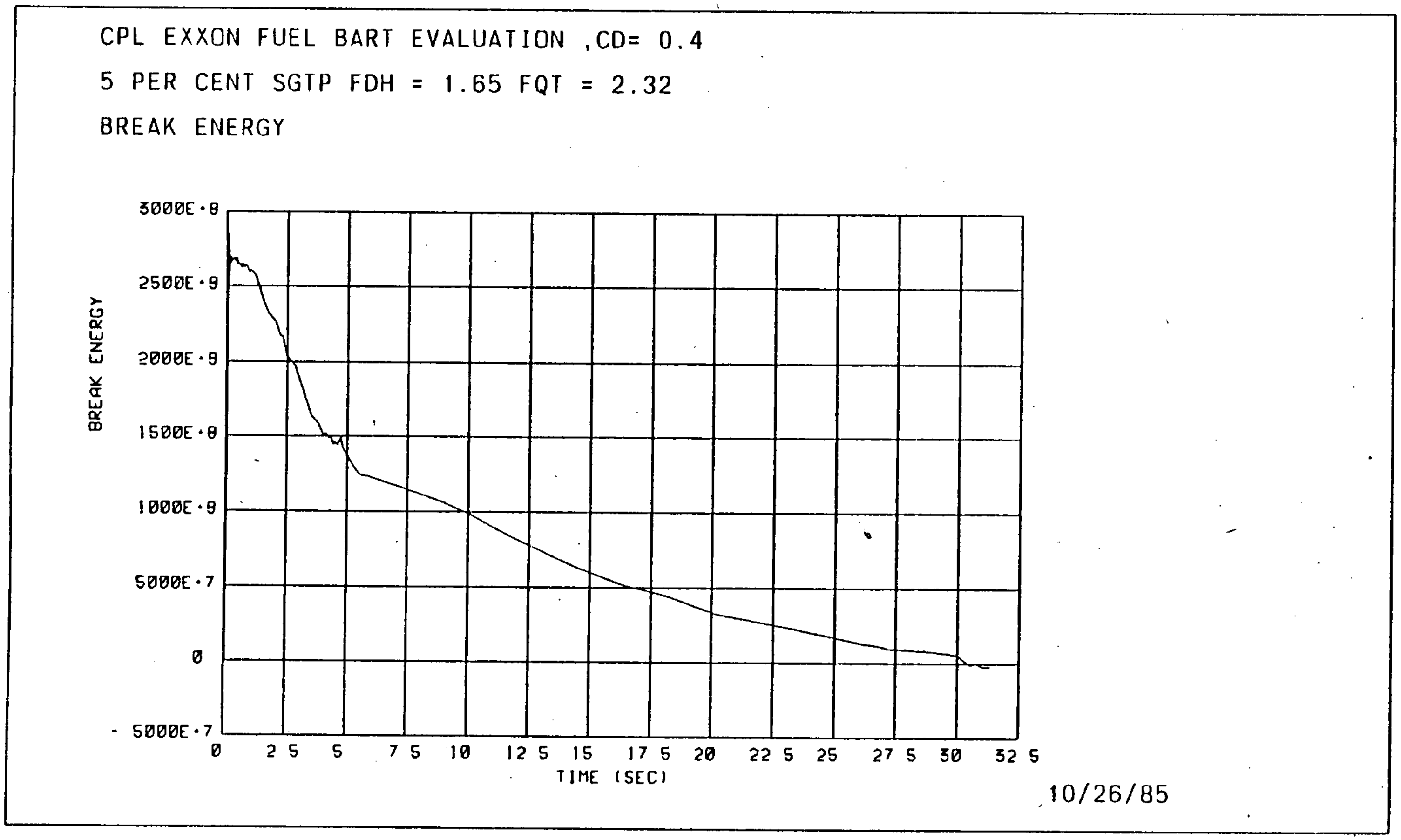

Figure 16. BREAK ENERGY RELEASED TO CONTAINMENT - DECLG (C_D = 0.4)