VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

November 12, 1981

R. H. LEASBURG
VICE PRESIDENT
NUCLEAR OPERATIONS

RECEIVED
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U.S. NUCLEAR REGULATORY COMMISSION

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. Robert A. Clark, Chief
Operating Reactors Branch No. 3
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Serial No.: 627
FR/KLB: plc
Docket Nos.: 50-338
50-339
License Nos.: NPF-4
NPF-7

Gentlemen:

AMENDMENT TO OPERATING LICENSES NPF-4 AND NPF-7
NORTH ANNA POWER STATION UNIT NOS. 1 AND 2
PROPOSED TECHNICAL SPECIFICATION CHANGE

Pursuant to 10CFR50.90, the Virginia Electric and Power Company requests an amendment, in the form of changes to the Technical Specifications, to Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station Unit Nos. 1 and 2.

A LOCA-ECCS reanalysis for North Anna Unit Nos. 1 and 2 has been performed using the NRC approved February, 1978 version of the Westinghouse LOCA-ECCS Evaluation Model. The analysis has been conducted in compliance with Appendix K to 10CFR50 and meets the acceptance criteria delineated in 10CFR50.46. This analysis was performed by Vepco under supervision of Westinghouse, and the results will support continued full power operation for both North Anna Units at steam generator tube plugging levels of up to 7 percent. The results of this reanalysis also support a new $F_Q$ limit of 2.14 after consideration of current fuel rod burst and blockage penalties. These results are provided in Attachment 1. Proposed changes to the Technical Specifications, consistent with the reanalysis, are provided in Attachment 2.

This request has been reviewed by the Station Nuclear Safety and Operating Committee and the Safety Evaluation and Control staff. It has been determined that this request does not involve any unreviewed safety questions as defined [illegible] 0.59.

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s 1/40 w/check
#4,400

We have evaluated this request in accordance with the criteria in 10CFR170.22. It has been determined that this request requires a Class III amendment fee. Accordingly, a voucher check in the amount of $4400.00 is enclosed in payment of the required fee.

Very truly yours,

R. H. Leasburg

Attachments

(1) LOCA-ECCS Safety Evaluation for North Anna Unit Nos. 1 and 2
(2) Proposed Technical Specifications Changes
(3) Voucher Check for $4,400.00

cc: Mr. James P. O'Reilly, Director
Office of Inspection and Enforcement
Region II

COMMONWEALTH OF VIRGINIA )
)
CITY OF RICHMOND )

The foregoing document was acknowledged before me, in and for the City and Commonwealth aforesaid, today by R. H. Leasburg, who is Vice President-Nuclear Operations, of the Virginia Electric and Power Company. He is duly authorized to execute and file the foregoing document in behalf of that Company, and the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 12th day of November, 19 81.

My Commission expires: 2-26, 19 85.

Ann C. McBee
Notary Public

(SEAL)

Attachment 1

LOCA-ECCS Safety Evaluation

for

North Anna Unit Nos. 1 and 2

## 1.0 INTRODUCTION

A reanalysis of the ECCS performance for the postulated large break Loss of Coolant Accident (LOCA)* has been performed which is in compliance with Appendix K to 10 CFR 50. The results of this reanalysis are presented herein and are in compliance with 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors. This analysis was performed with the NRC approved (Ref. 2, 11, 12, 13) February 1978 version of the Westinghouse LOCA-ECCS evaluation model. The analytical techniques used are in full compliance with 10 CFR 50, Appendix K.

As required by Appendix K of 10 CFR 50, certain conservative assumptions were made for the LOCA-ECCS analysis. The assumptions pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA is assumed to occur and include such items as the core peaking factors, the containment pressure, and the performance of the emergency core cooling system (ECCS). All assumptions and initial operating conditions used in this reanalysis were the same as those used in the previous LOCA-ECCS analysis (Ref. 3) with the following exceptions: 1) the limiting value of the heat flux hot channel factor was increased from 2.10 to 2.20; 2) more accurate data for several containment parameters were used; 3) 7% of the steam generator tubes were assumed to be plugged; 4) the 17 x 17 generic fuel

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* The reanalysis of the small break LOCA is not necessary and therefore the analysis of this accident submitted by Reference 1 remains applicable.

fuel parameters were updated to reflect the current values, such as removal of the previously required inclusion of a 65°F uncertainty in pellet temperature; 5) a previous requirement of analysis employing a spectrum of fuel heatup rates has been eliminated; 6) a burst and blockage adjustment penalty of 0.06 (as explained in Appendix A) must be subtracted from the value for the heat flux hot channel factor.

## 2.0 DESCRIPTION OF POSTULATED MAJOR REACTOR COOLANT PIPE RUPTURE (LOSS OF COOLANT ACCIDENT - LOCA)

A LOCA is the result of a rupture of the reactor coolant system (RCS) piping or of any line connected to the system. The system boundaries considered in the LOCA analysis are defined in the FSAR. Sensitivity studies (Reference 4) have indicated that a double-end cold leg guillotine (DECLG) pipe break is limiting. In the unlikely event of a DECLG break, a rapid depressurization of the RCS will result. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. A safety injection system (SIS) signal is actuated when the appropriate setpoint is reached and the high head safety injection pumps are activated. The actuation and subsequent activation of the ECCS, which occurs with the SIS signal, assumes the most limiting single failure event. These countermeasures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. (It should be noted, however, that no credit is taken in the analysis for the insertion of control rods to shut down the reactor).

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

Before the break occurs, the unit is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from decay, hot internals and the vessel continues to be transferred to the reactor coolant system. 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 calcuated,

consistent with Appendix K of 10 CFR 50. Thereafter, the core heat transfer is based on local conditions with transition boiling and forced convection to steam as the major heat transfer mechanisms. During the refill period, it is assumed that rod-to-rod radiation is the only core heat transfer mechanism. The heat transfer between the reactor coolant system and the secondary side may be in either direction depending on the relative temperatures. For the case of continued heat addition to the secondary side, secondary side pressure increases and the main safety valves may actuate to reduce the pressure. Makeup to the secondary side is automatically provided by the auxiliary feedwater system. Coincident with the safety injection signal, normal feedwater flow is stopped by closing the main feedwater control valves and tripping the main feedwater pumps. Emergency feedwater flow is initiated by starting the auxiliary feedwater pumps. The secondary side flow aids in the reduction of reactor coolant system pressure. When the reactor coolant system depressurizes to 600 psia, the accumulators begin to inject borated water into the reactor coolant loops. A conservative assumption is then made that the injected accumulator water bypasses the core and goes out through the break until the termination of bypass. This conservatism is again consistent with Appendix K of 10 CFR 50. In addition, the reactor coolant pumps are assumed to be tripped at the initiation of the accident and effects of pump coastdown are included in the blowdown analysis.

The water injected by the accumulators cools the core and subsequent operation of the low head safety injection pumps supplies water for long term cooling. When the RWST is nearly empty, long term cooling of the core is accomplished by switching to the recirculation mode of core cooling, in which the spilled borated water is drawn from the containment sump by the low head

safety injection pumps and returned to the reactor vessel.

The containment spray system and the recirculation spray system operates to return the containment environment to a subatmospheric pressure.

The large break LOCA transient is divided, for analytical purposes, into three phases: blowdown, refill, and 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 clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of inter-related computer codes has been developed for the analysis of the LOCA.

The description of the various aspects of the LOCA analysis methodology is given in WCAP-8339(Ref. 5). This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with 10 CFR 50, Appendix K. The SATAN-VI, WREFLOOD, COCO, and LOCTA-IV codes, which are used in the LOCA analysis, are described in detail in WCAP-8306 (Ref. 6), WCAP-8326(Ref. 7), WCAP-8171(Ref. 8), and WCAP-8305(Ref. 9), respectively. These codes are able to assess whether sufficient heat transfer geometry and core amenability to cooling are preserved during the time spans applicable 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 COCO computer code is used to calculate the containment pressure transient during all three phases of the LOCA analysis. Similarly, the LOCTA-IV computer code is used to compute the thermal transient of the hottest fuel rod during the three phases.

SATAN-VI is used to determine the RCS pressure, enthalpy, and density, as well as the mass and energy flow rates in the RCS and steam generator secondary, as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator mass and 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, the mass and energy release rates during blowdown are transferred to the COCO 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 (i.e., the rate at which coolant enters the bottom of the core), the coolant pressure and temperature, and the quench front height during the refill and reflood phases of the LOCA. WREFLOOD also calculates the mass and energy flow rates that are assumed to be vented to the containment. Since the mass flow rates to the containment depends upon the core pressure, which is a function of the containment backpressure, the WREFLOOD and COCO codes are interactively linked. WREFLOOD is also linked to the LOCTA-IV code in that thermal-hydraulic parameters from WREFLOOD are used by LOCTA-IV in its calculation of the fuel temperature.

LOCTA-IV is used throughout the analysis of the LOCA transient to calculate the fuel and clad temperature of the hottest rod in the core. The input to LOCTA-IV consists of appropriate thermal-hydraulic output from SATAN-VI and WREFLOOD and conservatively selected initial RCS operating conditions. These

initial conditions are summarized in Table 1 and Figure 1. (The axial power shape of Figure 1 assumed for LOCTA-IV is a cosine curve which has been previously verified(Ref. 10) to be the shape that produces the maximum peak clad temperature).

The COCO code, which is also used throughout the LOCA analysis, calculates the containment pressure. Input to COCO is obtained from the mass and energy flow rates assumed to be vented to the containment as calculated by the SATAN-VI and WREFLOOD codes. In addition, conservatively chosen initial containment conditions and an assumed mode of operation for the containment cooling system are input to COCO. These initial containment conditions and assumed modes of operation are provided in Table 2.

## 3.0 DISCUSSION OF SIGNIFICANT INPUT

Significant differences in input between this analysis and the currently applicable analysis are delineated in Section 1.0 and discussed in more detail below. The changes made in the analysis reflect the operational conditions and limits necessary to allow full power operation at steam generator tube plugging levels of up to 7%.

The notable change for this analysis is the increase in assumed steam generator tube plugging. The currently applicable analysis allowed for 5% tube plugging. This plugging level was increased slightly to 7% for this analysis. A core inlet temperature of 548.6°F was used in the analysis. This value was adjusted from operational data to encompass this steam generator tube plugging range.

Several changes were made to the containment parameters. The thickness of one of the heat sinks in Table 2 was corrected to better represent the as-built plant containment. In addition, the previous generic value for the high-containment pressure setpoint was lowered to 18.5 psia to agree with the value in the North Anna Technical Specifications.

The calculation was performed assuming conservative generic 17 x 17 fuel parameters consistent with the current methodology. The previously required 65°F uncertainty in pellet temperature has been removed.

A previous requirement of analysis employing a spectrum of fuel heatup rates has been removed. This conforms with the NRC methodology for current ECCS analysis.

When the above changes were incorporated into the analysis, it was found that the assumed heat flux hot channel factor could be increased from 2.10 to 2.20 and still ensure compliance with the 10 CFR 50.46 acceptance criteria. This allowable increase in the assumed heat flux hot channel factor is primarily the result of the change in the generic fuel parameters, the elimination of the fuel heatup rate spectrum calculations and the higher peak clad temperature result of this analysis.

A worksheet evaluating the potential impact of using fuel rod models presented in the draft NUREG-0630 is included as Appendix A. The resulting adjustment penalty of 0.06 must be applied to the overall heat flux hot channel factor, resulting in an adjusted overall heat flux hot channel factor of 2.14.

## 4.0 RESULTS

Tables 1 and 2 and Figure 1 present the initial conditions and modes of operation that were assumed in the analysis. Table 3 presents the time sequence of events and Table 4 presents the results for the double-ended cold leg guillotine break (DECLG) for the CD=0.4 discharge coefficient. The DECLG has been determined to be the limiting break size and location based on the sensitivity studies reported in Reference 4. Further, all previous LOCA-ECCS submittals for the North Anna units have resulted in the CD=0.4 discharge coefficient being the limiting break size. The applicability of this conclusion (i.e. CD=0.4 is the limiting break size) for this analysis was explicitly verified. Consequently, only the results of the most limiting break size are presented in the figures and remaining tables in this submittal. The current analysis resulted in a limiting peak clad temperature of 2180.2°F, a maximum local cladding oxidation level of 7.75%, and a total core metal-water reaction of less than 0.3%. The detailed results of the LOCA reanalysis are provided in Tables 3 through 6 and Figures 2 through 18.

## 5.0 CONCLUSIONS

For breaks up to and including the double-ended severance of a reactor coolant pipe and for the operating conditions specified in Table 1 and 2, the Emergency Core Cooling System will meet the Acceptance Criteria as presented in 10 CFR 50.46. That is:

1. The calculated peak fuel rod clad temperature is below the requirement of 2200°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 limits of 17% are 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 the long-term decay heat is removed for an extended period of time.

## 6.0 REFERENCES

1. Final Safety Analysis Report, North Anna Power Station, Units 1 and 2, Virginia Electric and Power Company.

2. Letter from J. F. Stolz(NRC) to T. M. Anderson(Westinghouse), dated August 29, 1978.

3. Letter from C. M. Stallings(Vepco) to H. R. Denton(NRC), Serial No. 986C, January 2, 1980.

4. Buterbaugh, T. L., Johnson, W. J., and Kopelic, S. D., "Westinghouse ECCS Plant Sensitivity Studies," WCAP-8356, July 1974.

5. Bordelon, F. M., et. al., "Westinghouse ECCS Evaluation Model- Summary," WCAP-8339, July, 1974.

6. Bordelon, F. M., et.al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," WCAP-8306, June 1974.

7. Bordelon, F. M., and Murphy, E. T., "Containment Pressure Analysis Code (COCO)," WCAP-8326, June 1974.

8. Kelly, R. D., et. al., "Calculation Model for Core Reflooding after a Loss-of-Coolant Accident (WREFLOOD Code)," WCAP-8171, June 1974.

9. Bordelon, F. M., et. al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8305, June 1974.

10. Letter from C. M. Stallings(Vepco) to E. G. Case (NRC), Serial No. 092, February 17, 1978.

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

12. Letter from T. M. Anderson(Westinghouse) to J. F. Stolz(NRC), Serial No. NS-TMA-1981, November 1, 1978.

13. Letter from T. M. Anderson(Westinghouse) to R. Tedesco(NRC), Serial No. NS-TMA-2014, December 11, 1978.

TABLE 1

INITIAL CORE CONDITIONS ASSUMED FOR THE DOUBLE-ENDED COLD LEG GUILLOTINE BREAK (DECLG)

CALCULATIONAL INPUT

| | |
|---|---|
| Core Power (MWt, 102% of) | 2775 |
| Peak Linear Power (kw/ft, 102% of) | 11.98 |
| Heat Flux Hot Channel Factor ($F_Q$) | 2.20 |
| Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) | 1.55 |
| Accumulator Water Volume ($ft^3$, each) | 1025 |
| Reactor Vessel Upper Head Temperature Equal to Thot | |

| LIMITING FUEL REGION AND CYCLE | CYCLE | REGION |
|---|---|---|
| Unit 1 | ALL | ALL Regions |
| Unit 2 | ALL | ALL Regions |

TABLE 2

CONTAINMENT DATA

| | |
|---|---|
| NET FREE VOLUME | $1.916 \times 10^6$ $ft^3$ |
| INITIAL CONDITIONS[1] | |
| Pressure | 9.5 psia |
| Temperature | 90°F |
| RWST Temperature | 35°F |
| Outside Temperature | -10°F |
| SPRAY SYSTEM[1] | |
| Number of Pumps Operating | 2 |
| Runout Flow Rate (per pump) | 2000 gpm |
| Time in which spray is effective | 59 secs |

STRUCTURAL HEAT SINKS[1]

| Thickness (In) | Area ($Ft^2$), w/uncertainty |
|---|---|
| 6 Concrete | 8,393 |
| 12 Concrete | 62,271 |
| 18 Concrete | 55,365 |
| 24 Concrete | 11,591 |
| 27 Concrete | 9,404 |
| 36 Concrete | 3,636 |
| .375 Steel, 54 Concrete | 22,039 |
| .375 Steel, 54 Concrete | 28,933 |
| .500 Steel, 30 Concrete | 25,673 |
| 26.4 Concrete, .25 Steel, 120 Concrete | 12,110 |
| .407 Stainless Steel | 10,527 |
| .371 Steel | 160,328 |
| .882 Steel | 9,894 |
| .059 Steel | 60,875 |

[1]See the response to Comment S6.106 of the FSAR for a detailed breakdown of the containment heat sinks and for justification of the other input parameters used to calculate containment pressure.

TABLE 3

TIME SEQUENCE OF EVENTS

| | DECLG CD=0.4 (Sec) |
|---|---|
| Start | 0.0 |
| Reactor Trip | 0.71 |
| S. I. Signal | 2.07 |
| Acc. Injection | 16.28 |
| End of Bypass | 26.41 |
| Pump Injection | 27.07 |
| End of Blowdown | 29.60 |
| Bottom of Core Recovery | 39.85 |
| Acc. Empty | 52.34 |

TABLE 4

RESULTS FOR DECLG

| | CD=0.4 |
|---|---|
| Peak Clad Temp, °F | 2180.2 |
| Peak Clad Location, Ft. | 7.5 |
| Local Zr/H20 RXN (max), % | 7.75 |
| Local Zr/H20 Location, Ft. | 7.5 |
| Total Zr/H20 RXN, % | <0.3 |
| Hot Rod Burst Time, sec. | 37.00 |
| Hot Rod Burst Location, Ft. | 6.0 |

TABLE 5

REFLOOD MASS AND ENERGY RELEASES

DECLG (CD= 0.4)

| TIME(SEC) | TOTAL MASS FLOWRATE (LB/SEC) | TOTAL ENERGY FLOWRATE ($10^5$BTU/SEC) |
|---|---|---|
| 39.85 | 0.0 | 0.0 |
| 40.6 | 0.785 | 0.0099 |
| 46.3 | 35.87 | 0.4675 |
| 55.9 | 221.0 | 1.421 |
| 70.8 | 253.9 | 1.439 |
| 89.7 | 266.3 | 1.406 |
| 111.2 | 274.4 | 1.361 |
| 134.9 | 280.9 | 1.312 |
| 189.6 | 293.2 | 1.209 |
| 257.6 | 307.2 | 1.099 |

TABLE 6

BROKEN LOOP ACCUMULATOR FLOW TO CONTAINMENT

DECLG, CD=0.4

| TIME(SEC) | MASS FLOWRATE* (LBM/SEC) |
|---|---|
| 0.0 | 4010 |
| 1.0 | 3622 |
| 3.0 | 3104 |
| 5.0 | 2761 |
| 7.0 | 2509 |
| 10.0 | 2226 |
| 15.0 | 1895 |
| 20.0 | 1674 |
| 25.0 | 1523 |
| 30.0 | 1415 |

*For energy flowrate multiply mass flowrate by a constant of 59.60 BTU/LBM.

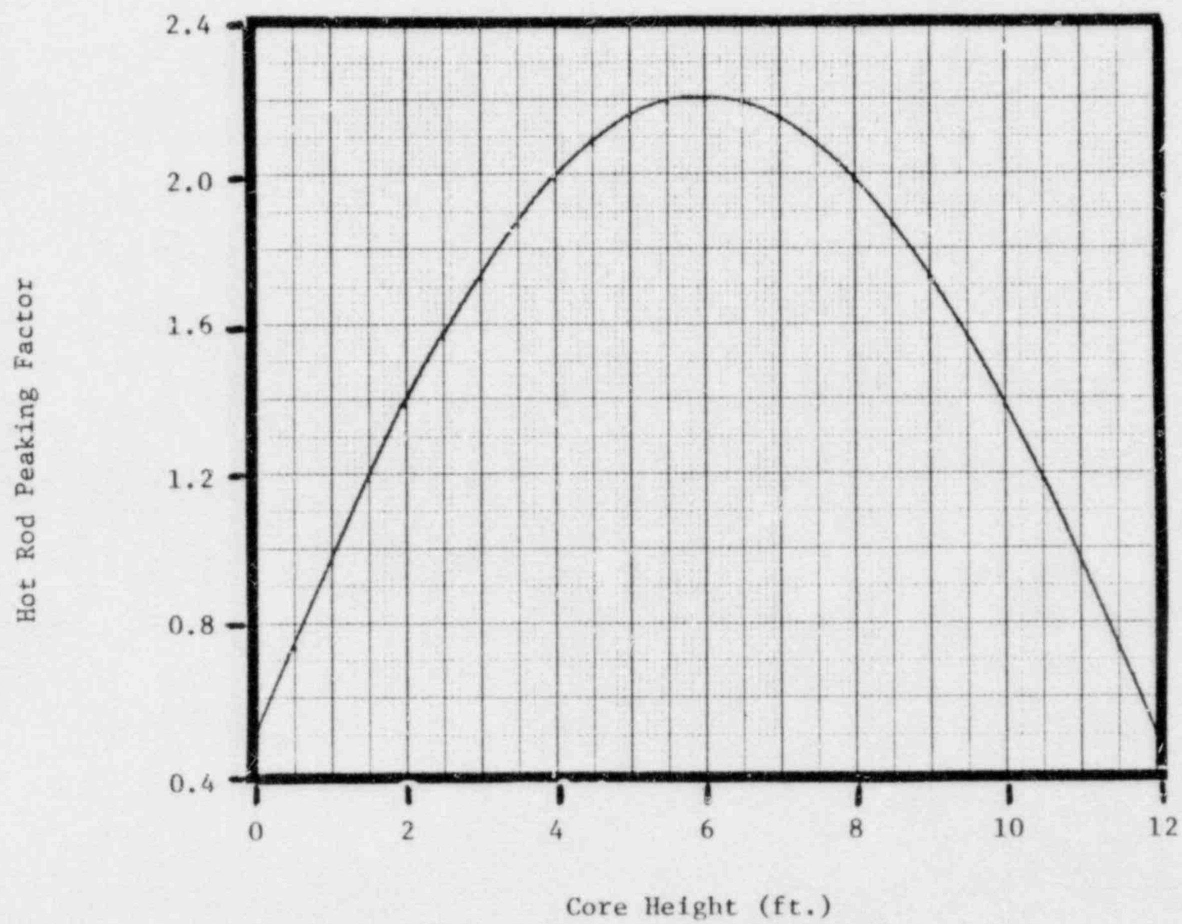


Figure 1: Peaking Factor versus Core Height - FQ=2.20

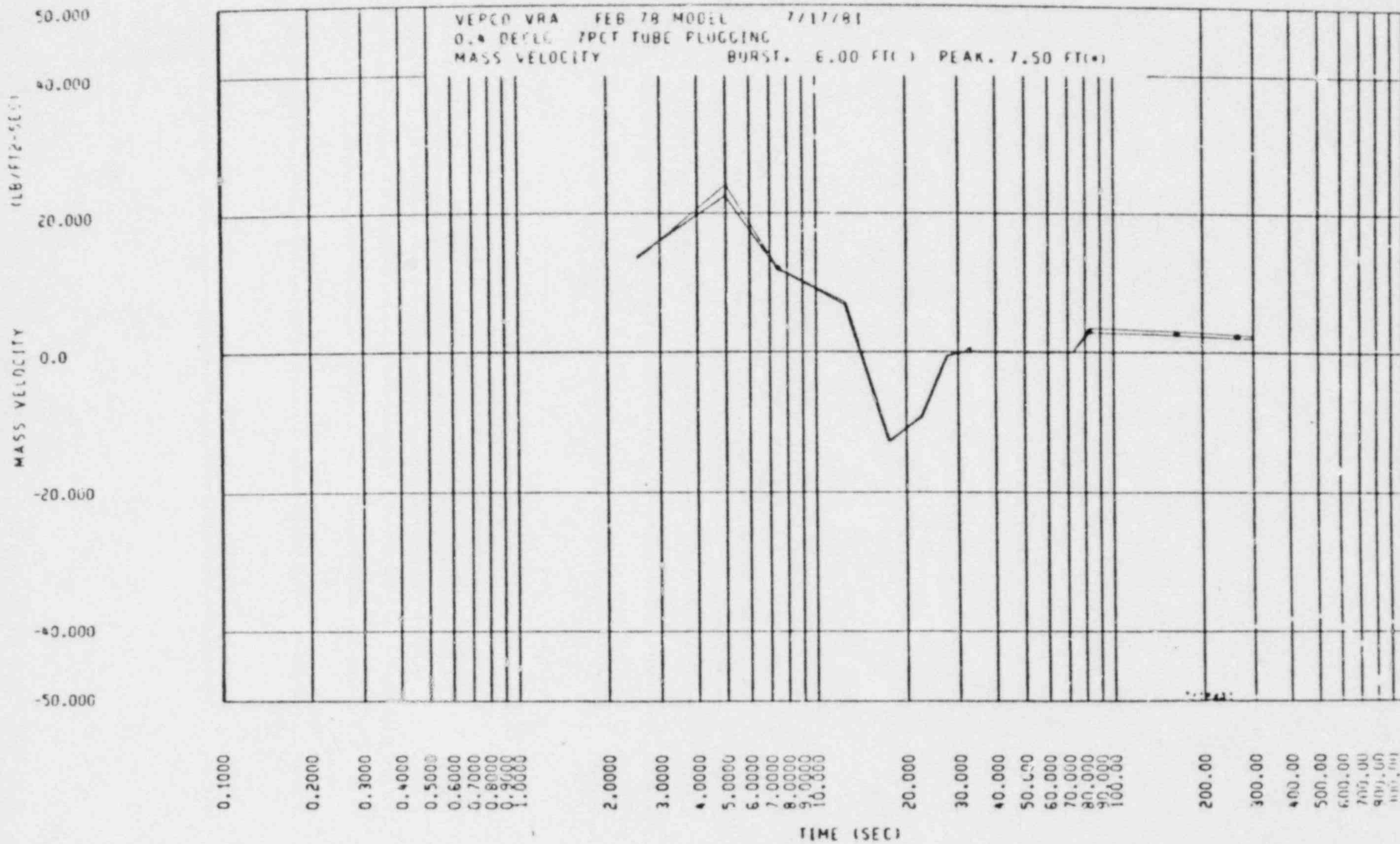


Figure 2: Mass Velocity - DECLG($C_D$ = 0.4)

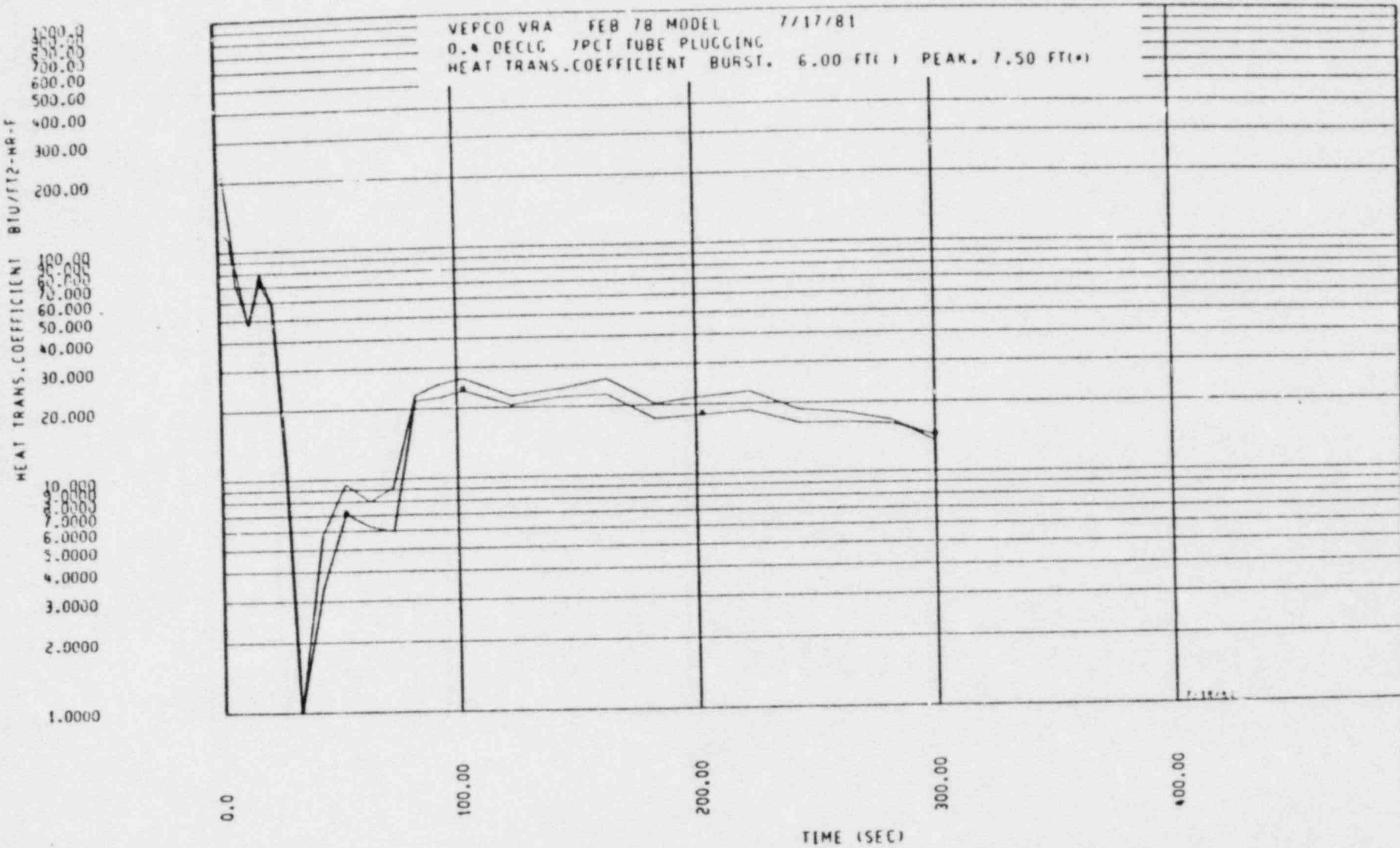


Figure 3: Heat Transfer Coefficient - DECLG($C_D$ = 0.4)

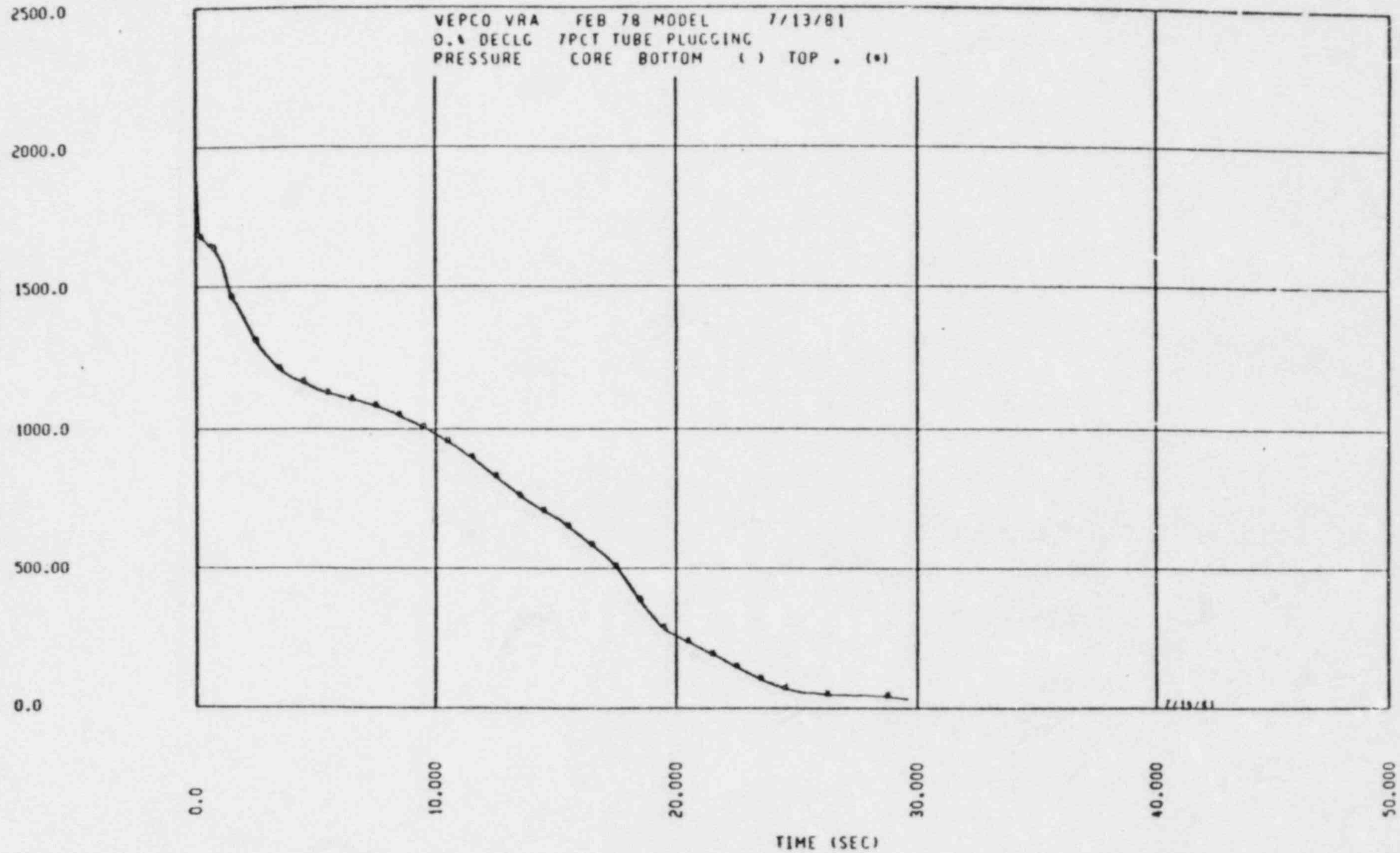


Figure 4: Core Pressure - DECLG($C_D$ = 0.4)

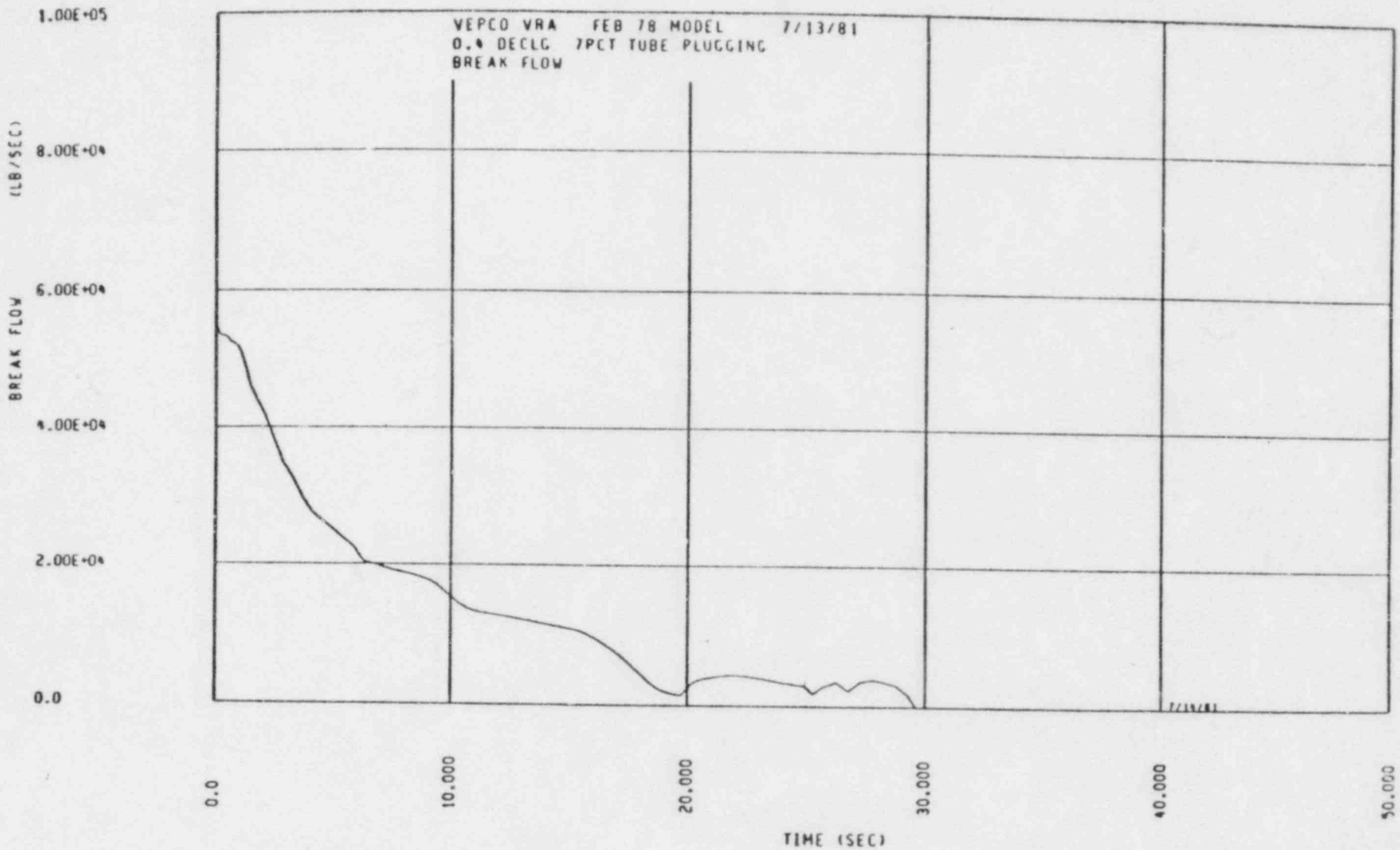


Figure 5: Break Flow Rate - DECLG($C_D$ = 0.4)

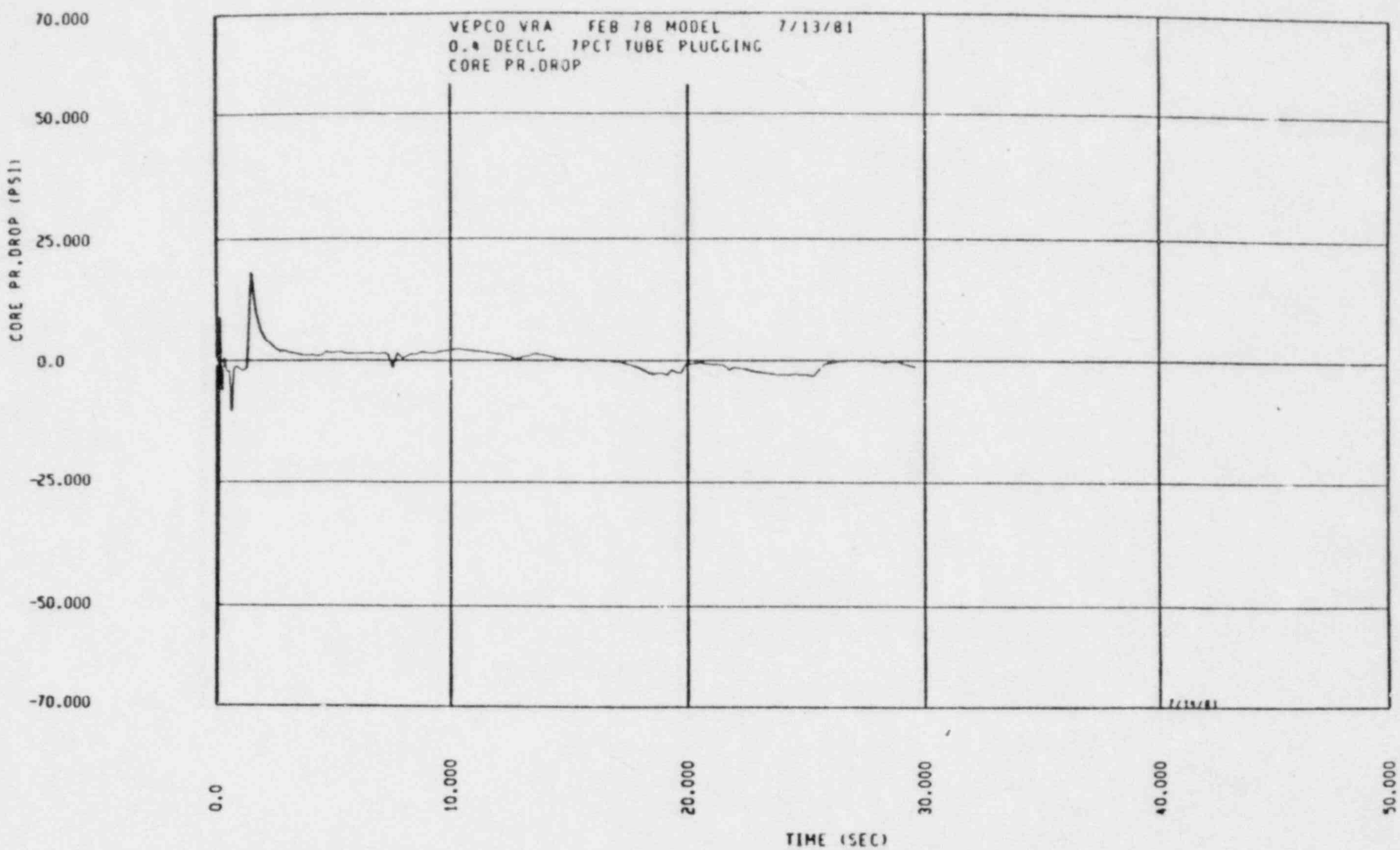


Figure 6: Core Pressure Drop - DECLG($C_D$ = 0.4)

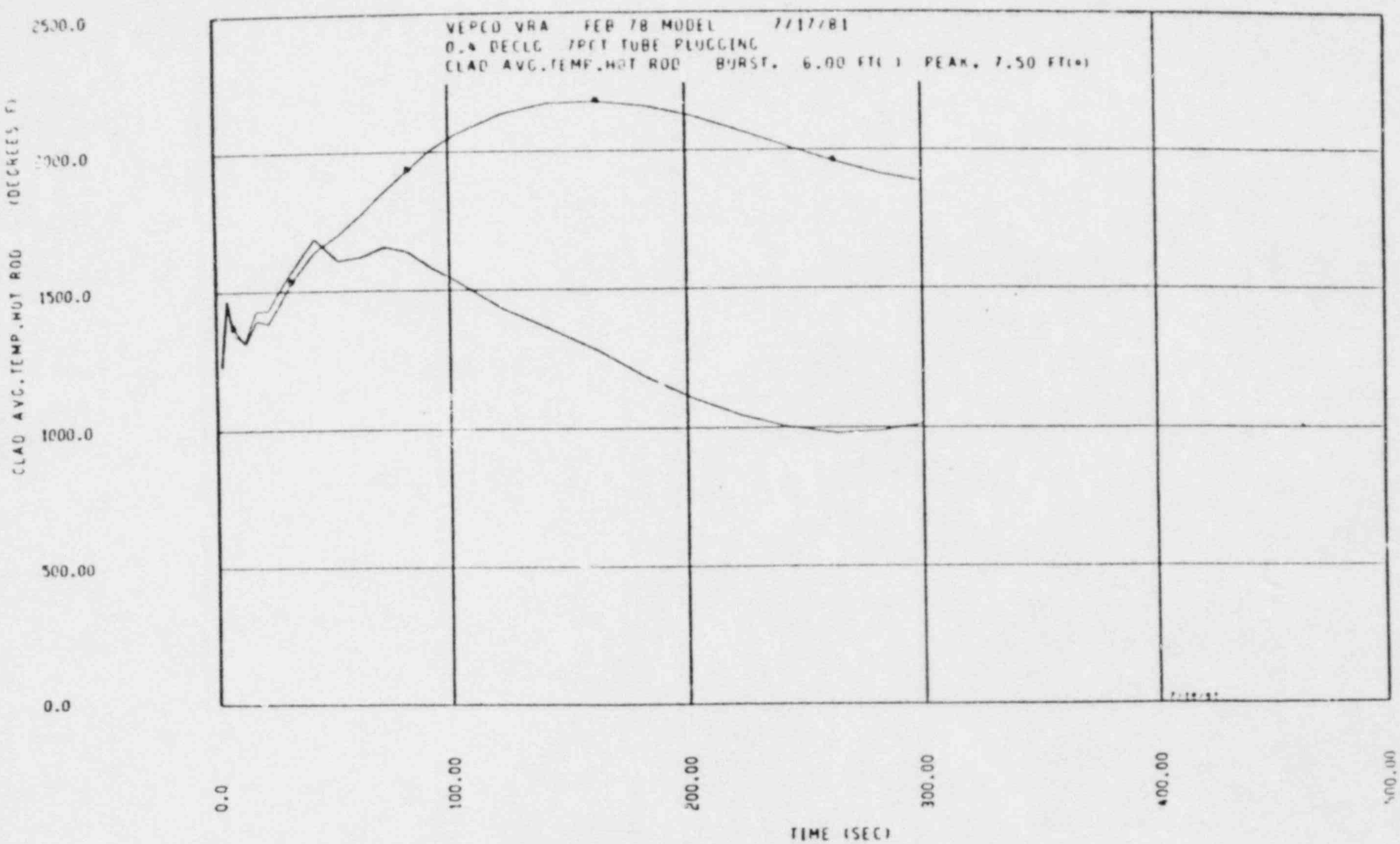


Figure 7: Peak Clad Temperature - DECLG($C_D$ = 0.4)

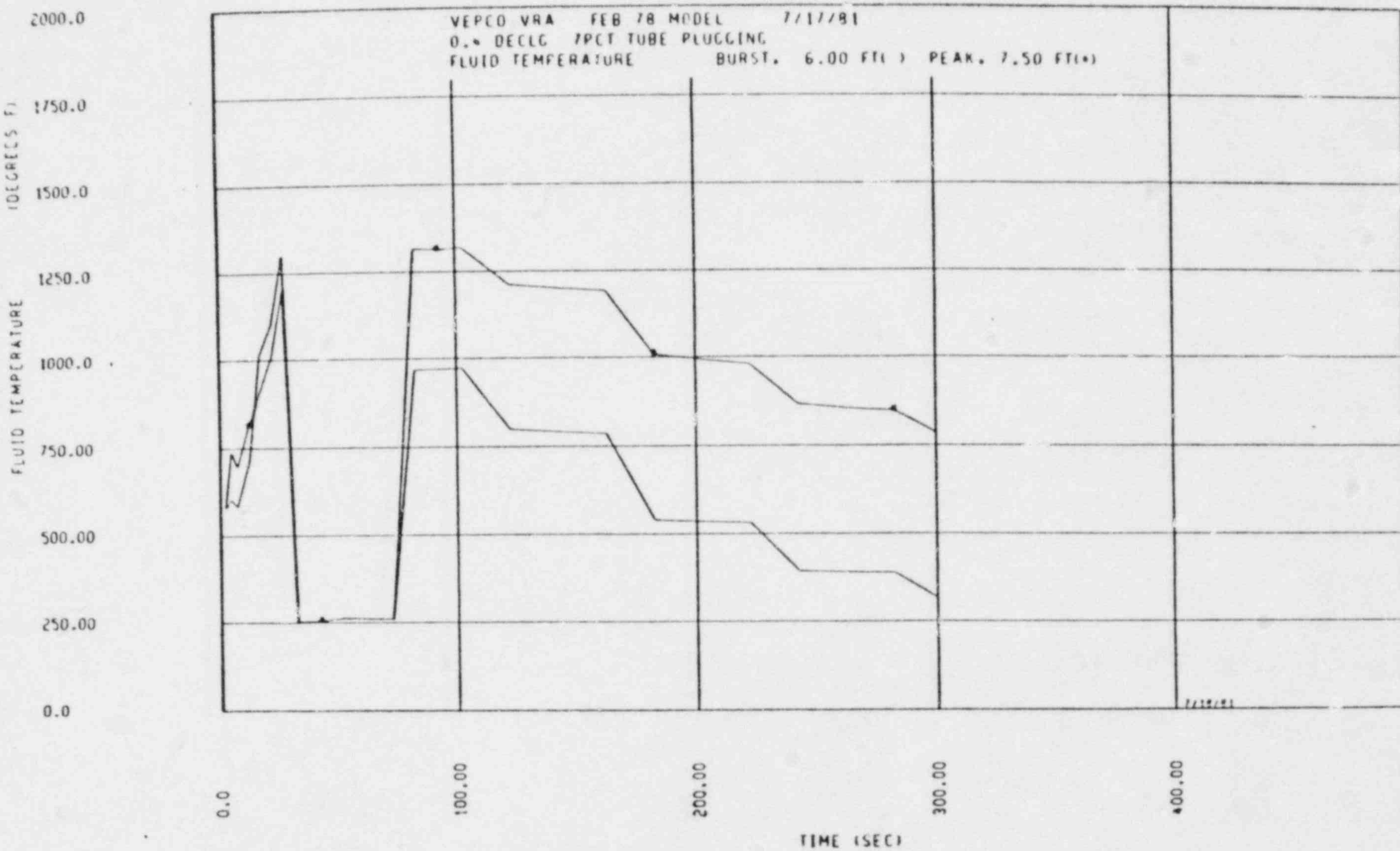


Figure 8: Fluid Temperature - DECLG($C_D$ = 0.4)

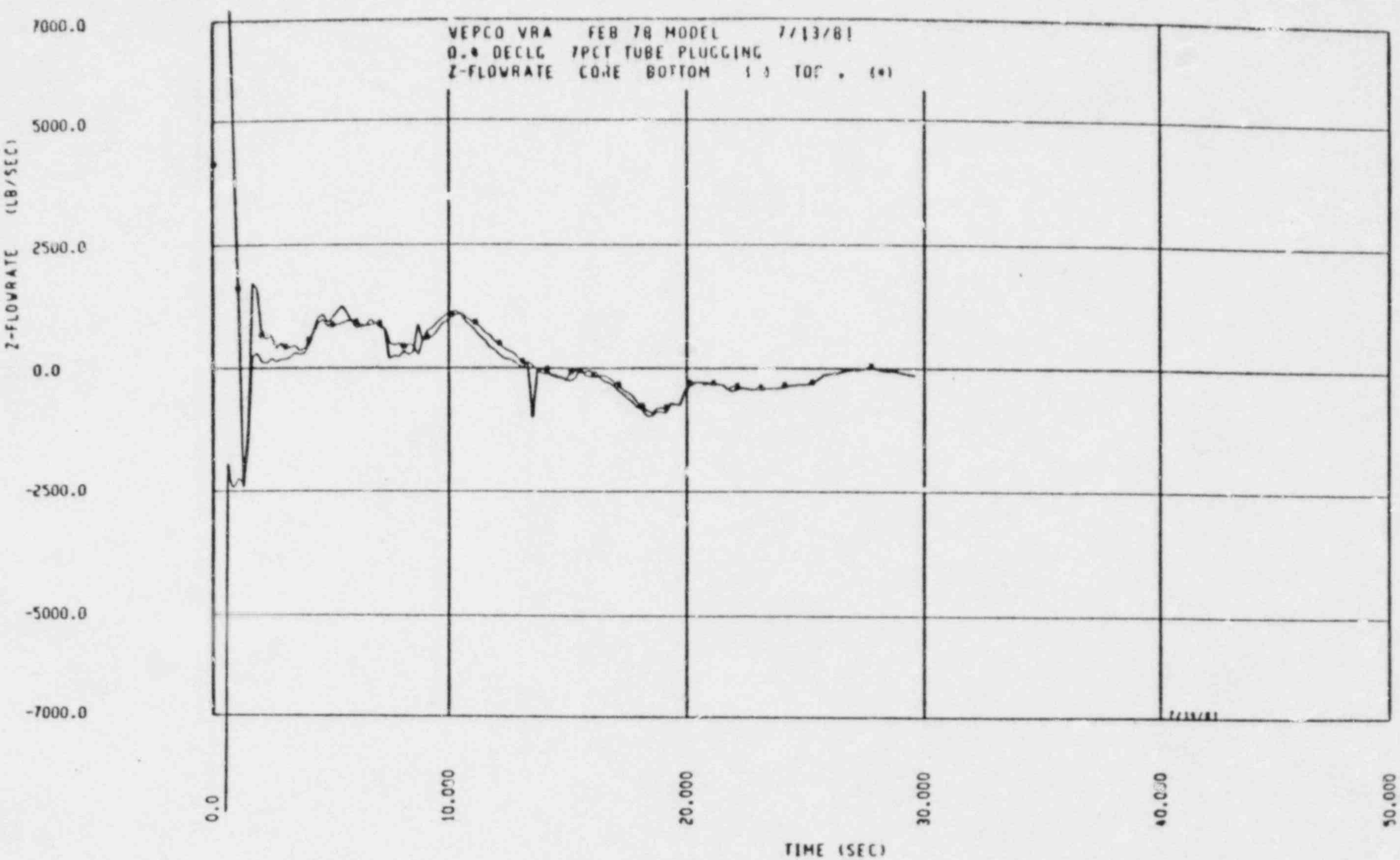


Figure 9: Core Flow (Top and Bottom) - DECLG($C_D$ = 0.4)

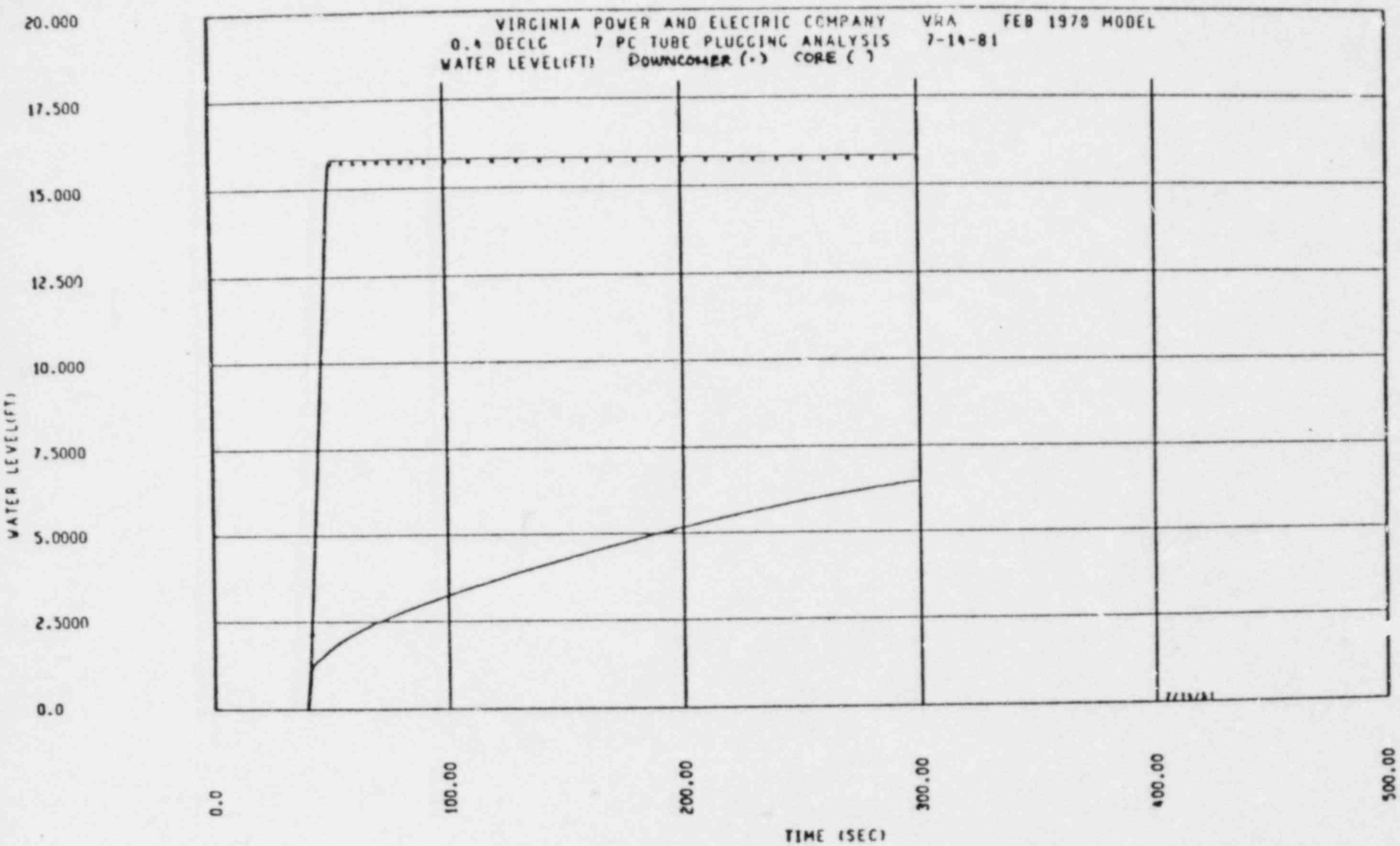


Figure 10: Reflood Transient, Core and Downcomer Water Levels - DECLG($C_D$ = 0.4)

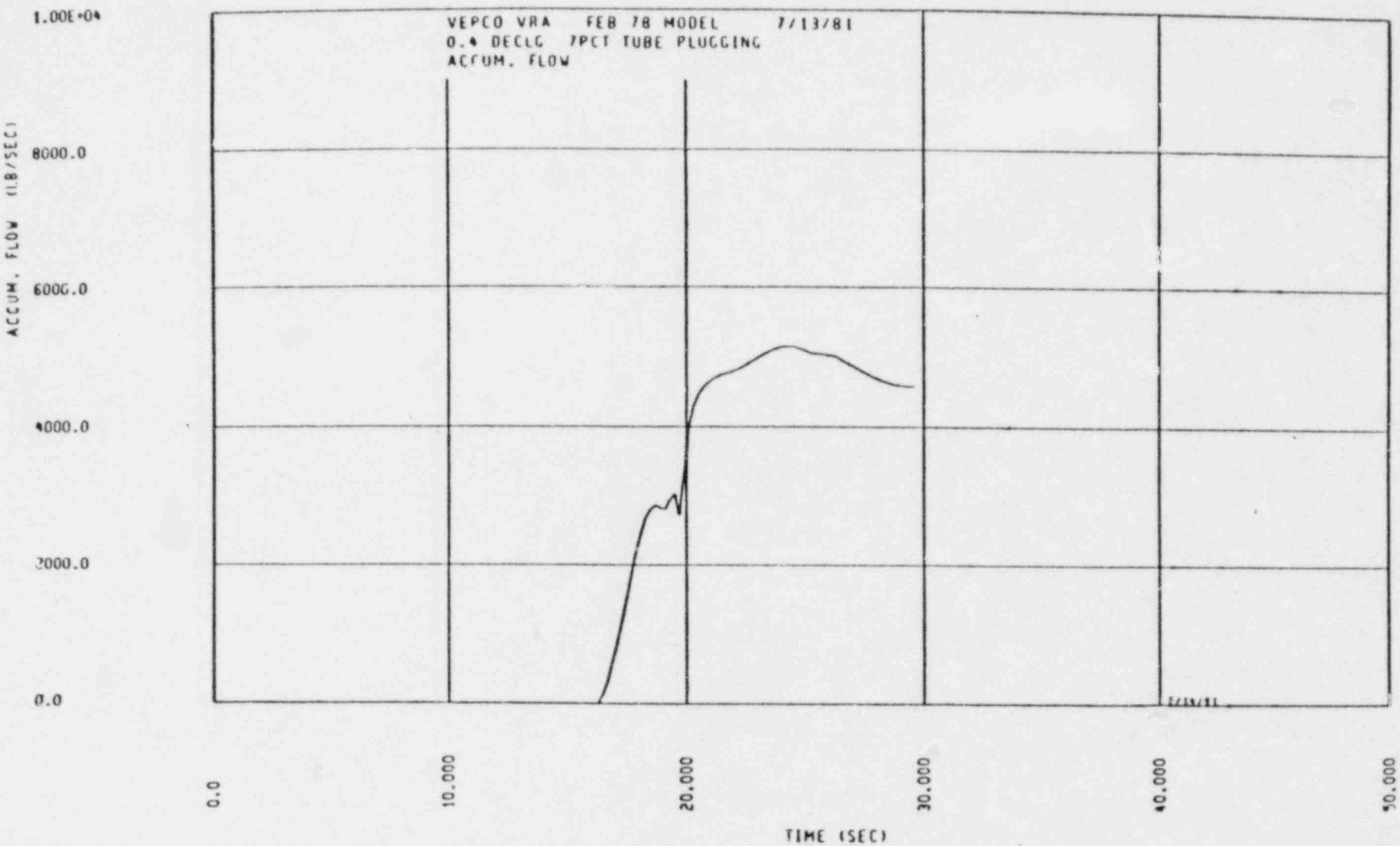


Figure 11: Accumulator Flow (Blowdown) - DECLG($C_D$ = 0.4)

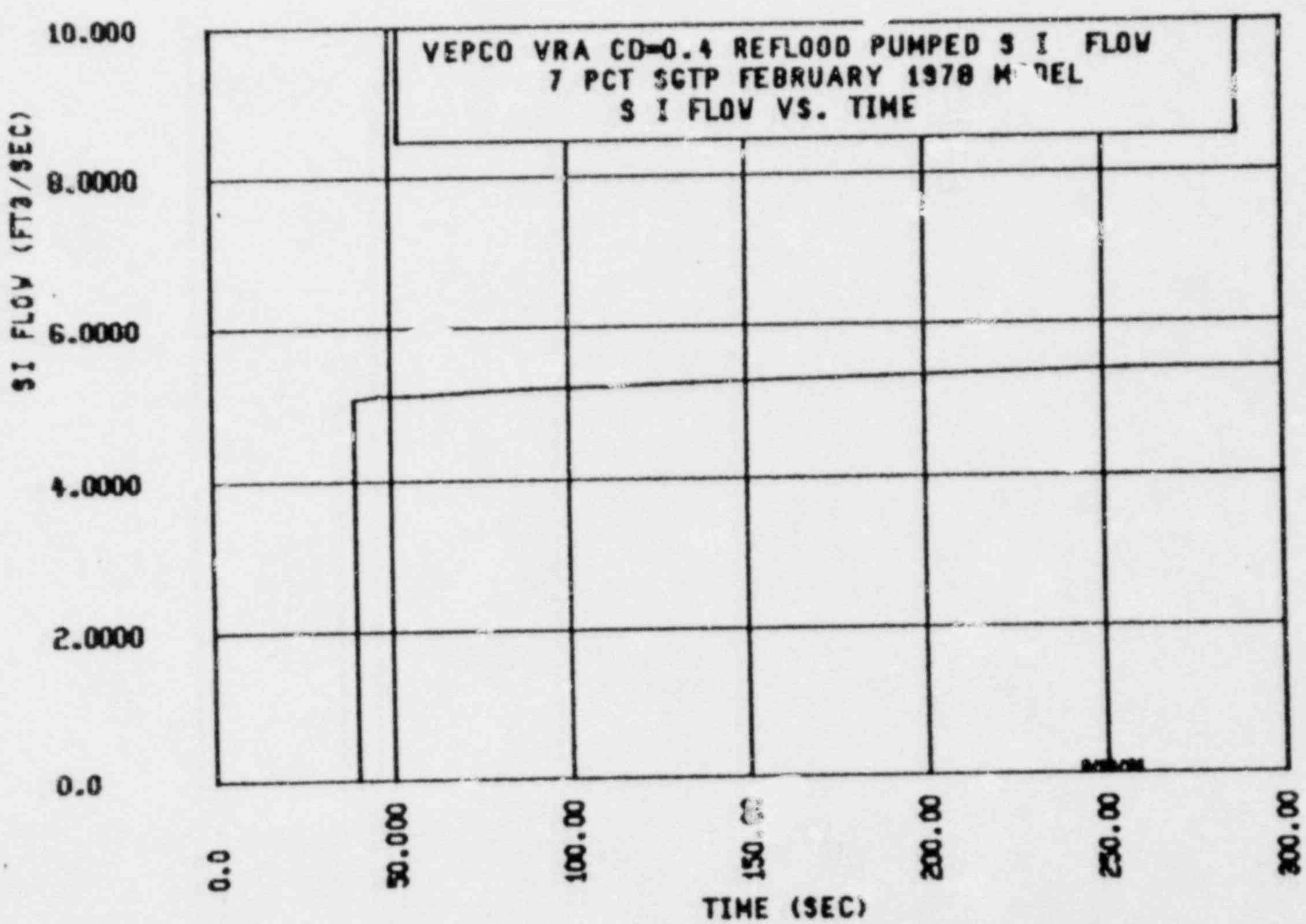


Figure 12: Pumped ECCS Flow (Reflood) - DECLG($C_D$ = 0.4)

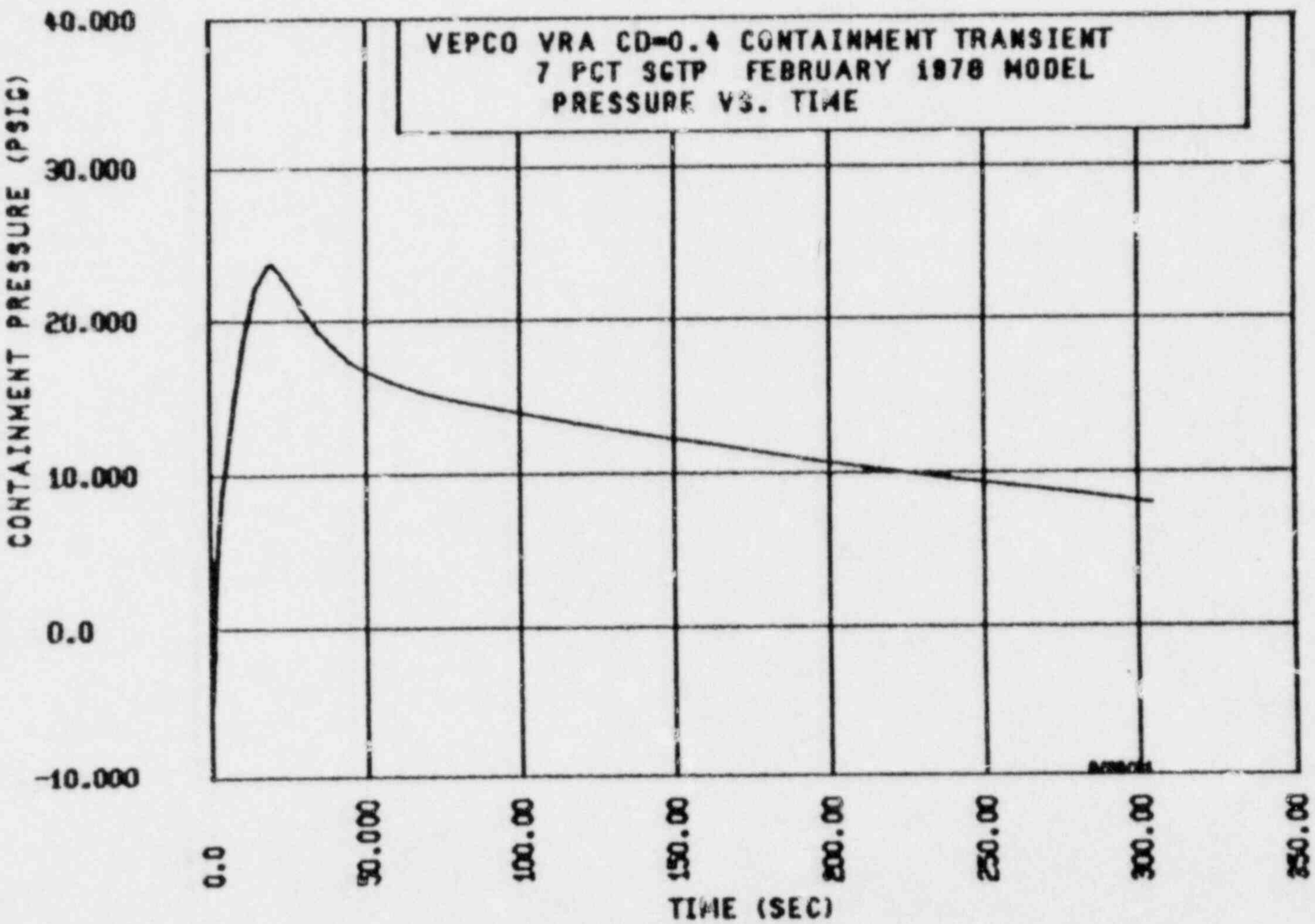


Figure 13: Containment Pressure - DECLG($C_D$ = 0.4)

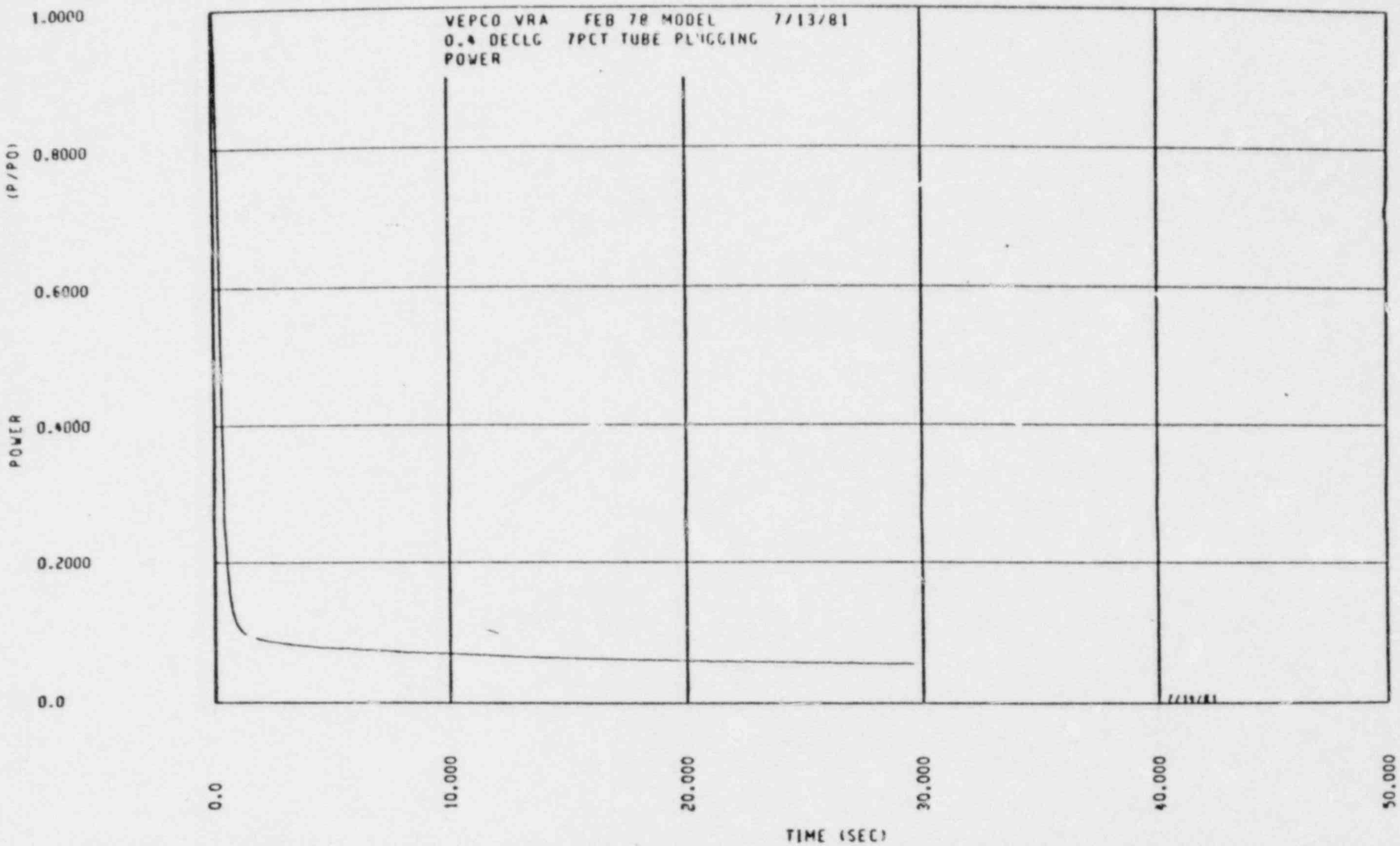


Figure 14: Core Power Transient - DECLG($C_D$ = 0.4)

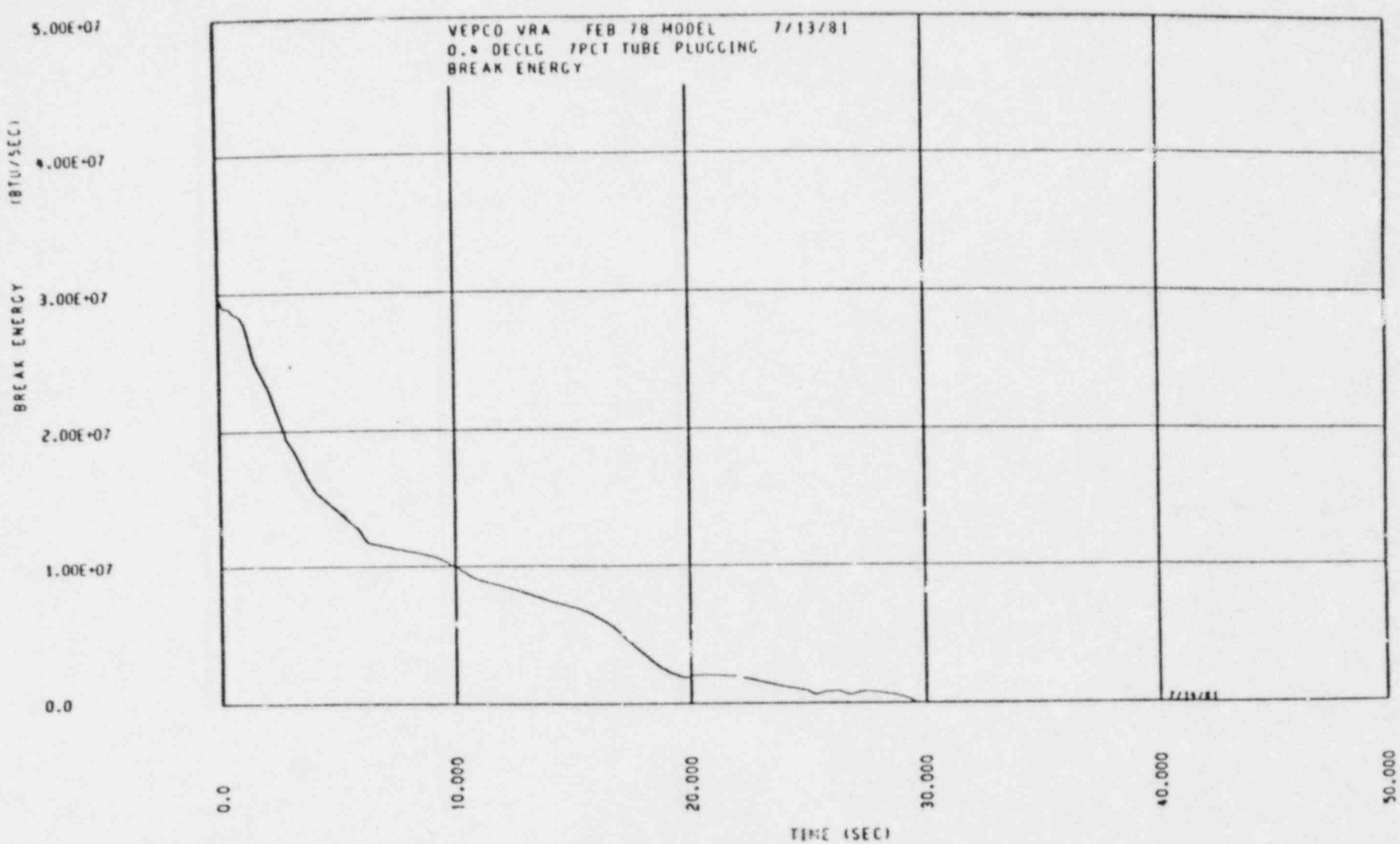


Figure 15: Break Energy Released to Containment - DECLG($C_D$ = 0.4)

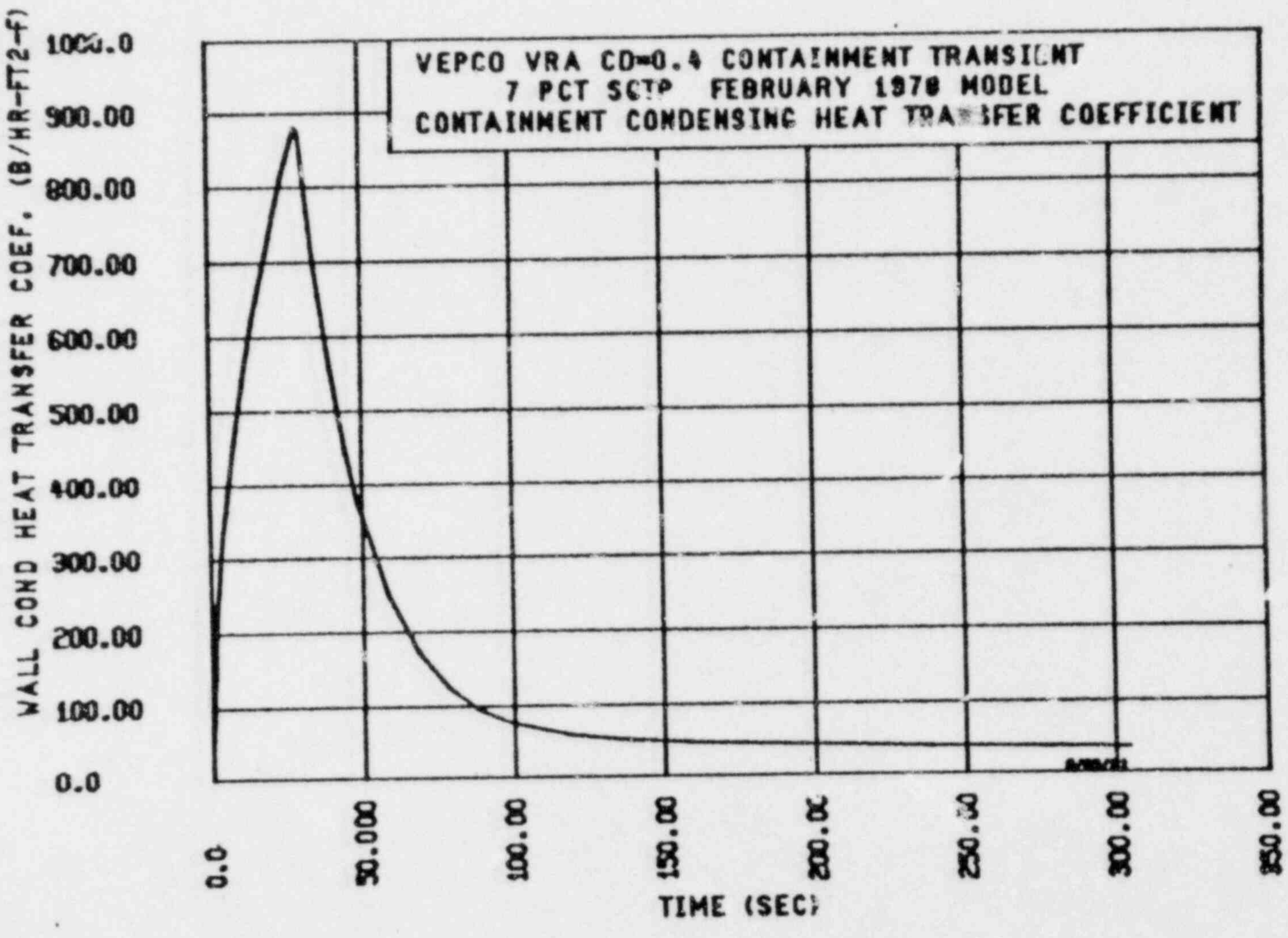


Figure 16: Containment Wall Heat Transfer Coefficient - DECLG($C_D$ = 0.4)

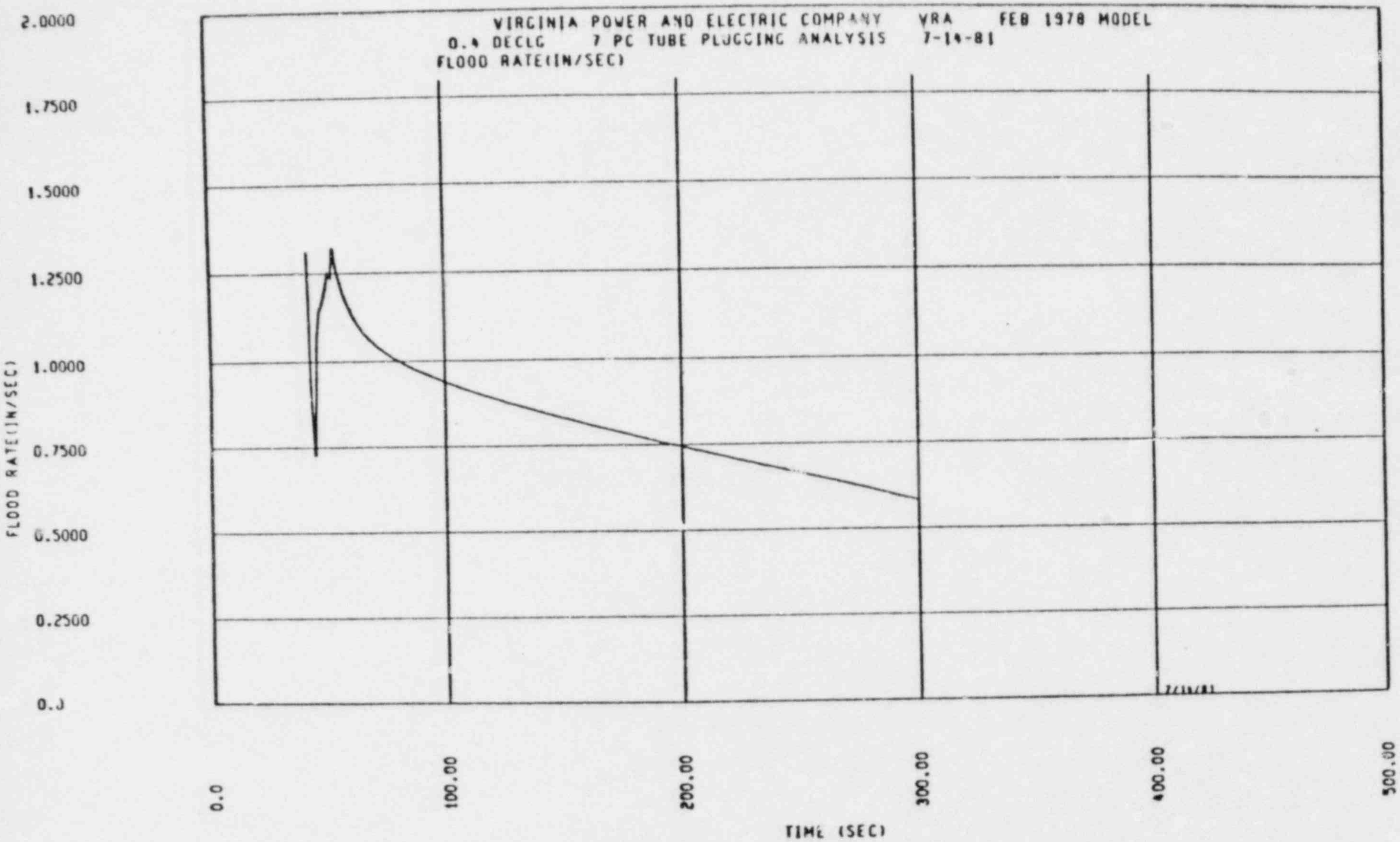


Figure 17: Reflood Transient, Core Inlet Velocity - DECLG($C_D$ = 0.4)

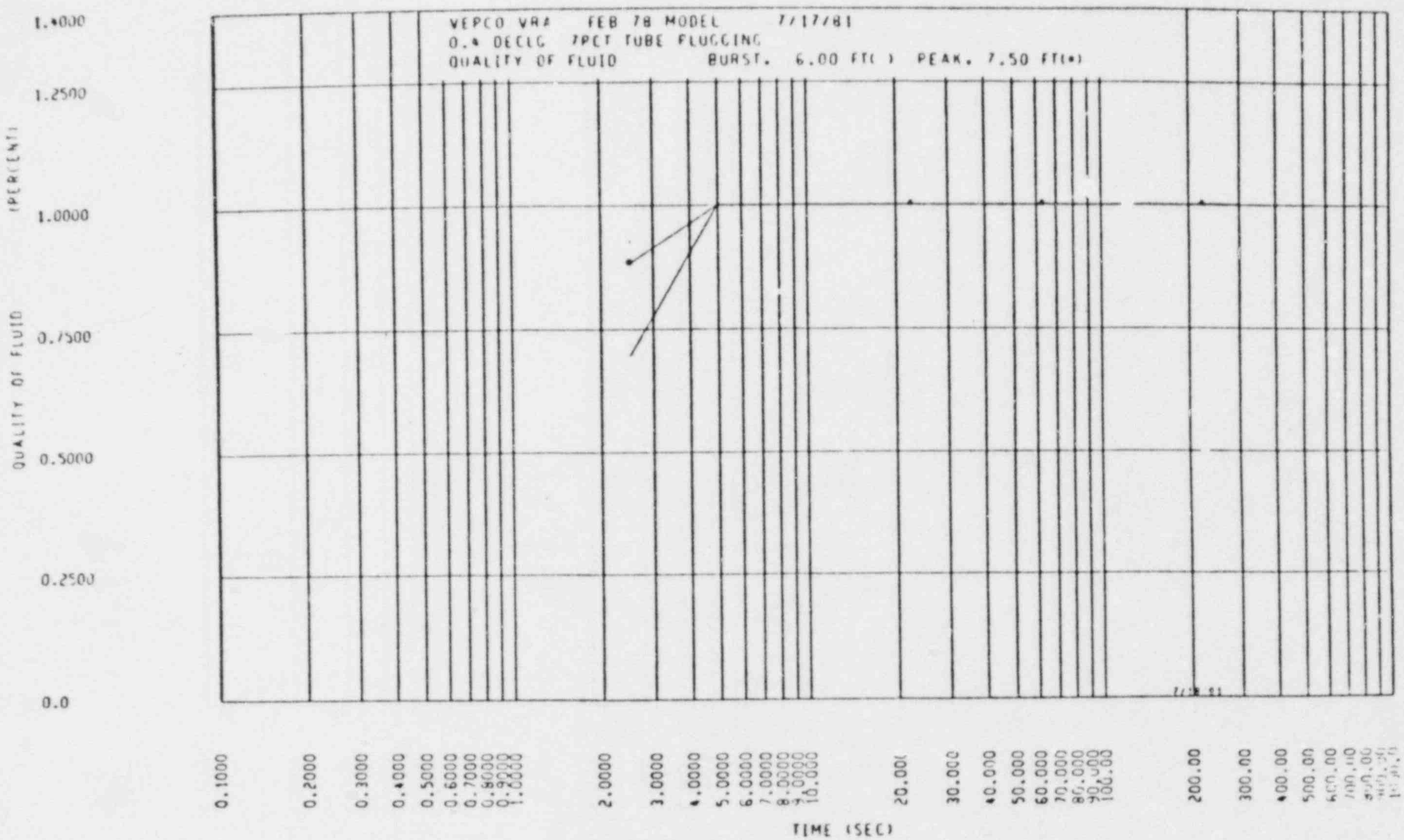


Figure 18: Fluid Quality - DECLG($C_D$ = 0.4)

APPENDIX A

A. Evaluation of the potential impact of using fuel rod models presented in dra^ft NUREG-0630 on the Loss of Coolant Accident (LOCA) analysis for North Anna Unit 1 (VRA).

Th·- evaluation is based on the limiting break LOCA analysis identified as follows:

BREAK TYPE - DOUBLE ENDED COLD LEG GUILLOTINE

BREAK DISCHARGE COEFFICIENT 0.4

WESTINGHOUSE ECCS EVALUATION MODEL VERSION February '78 model

CORE PEAKING FACTOR 2.20

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR THE BURST REGION OF THE CLAD - 1693.91 $^{o}F = PCT_{B}$

ELEVATION - 6.00 Feet.

HOT ROD MAXIMUM TEMPERATURE CLACULATED FOR A NON-RUPTURED REGION OF THE CLAD - 2180.2 $^{o}F = PCT_{N}$

ELEVATION - 7.50 Feet

CLAD STRAIN DURING BLOWDOWN AT THIS ELEVATION 2.47 Percent
MAXIMUM CLAD STRAIN AT THIS ELEVATION - 9.50 Percent

Maximum temperature for this non-burst node occurs when the core reflood rate is (LESS) than 1.0 inch per second and reflood heat transfer is based on the (STEAM COOLING) calculation.

AVERAGE HOT ASSEMBLY ROD BURST ELEVATION - 7.25 Feet

HOT ASSEMBLY BLOCKAGE CALCULATED - 27.2 Percent

1. BURST NODE

The maximum potential impact on the ruptured clad node is expressed in letter NS-TMA-2174 in terms of the change in the peaking factor limit (FQ) required to maintain a peak clad temperature (PCT) of $2200^{\circ}F$ and in terms of a change in PCT at a constant FQ. Since the clad-water reaction rate increases significantly at temperatures above $2200.^{\circ}F$, individual effects (such as $\Delta PCT$ due to changes in several fuel rod models) indicated here may not accurately apply over large ranges, but a simultaneous change in FQ which causes the PCT to remain in the neighborhood of $2200\ ^{\circ}F$ justifies use of this evaluation procedure.

From NS-TMA-2174:

For the Burst Node of the clad:

- $0.01\ \Delta FQ \rightarrow \sim 150^{\circ}F$ BURST NODE $\Delta PCT$

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

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

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

$$\Delta PCT_1 = (.027 + .03)\ (150^{\circ}F/.01) = 855^{\circ}F$$

Margin to the $2200^{\circ}F$ limit is:

$$\Delta PCT_2 = 2200.^{\circ}F - PCT_B = \underline{\ \ 506.09\ \ }$$

The FQ reduction required to maintain the $2200^{\circ}F$ clad temperature limit is:

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

$$\Delta FQ_B = \frac{855. - 506.09}{} \left(\frac{.01}{150}\right)$$

$= \underline{0.023}$ (but not less than zero).

2. NON-BURST NODE

The maximum temperature calculated for a non-burst section of clad typically occurs at an elevation above the core mid-plane during the core reflood phase of the LOCA transient. The potential impact on that maximum clad temperature of using the NRC fuel rod models can be estimated by examining two aspects of the analyses. The first aspect is the change in pellet-clad gap conductance resulting from a difference in clad strain at the non-burst maximum clad temperature node elevation. Note that clad strain all along the fuel rod stops after clad burst occurs and use of a different clad burst model can change the time at which burst is calculated. Three sets of LOCA analysis results were studied to establish an acceptable sensitivity to apply generically in this evaluation. The possible PCT increase resulting from a change in strain (in the Hot Rod) is +20.°F per percent decrease in strain at the maximum clad temperature locations. Since the clad strain calculated during the reactor coolant system blowdown phase of the accident is not changed by the use of NRC fuel rod models, the maximum decrease in clad strain that must be considered here is the difference between the "maximum clad strain" and the "clad strain at the end of RCS blowdown" indicated above.

Therefore:

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

$$= \left(\frac{20}{.01}\right) \left(\underline{.0950} - \underline{.0247}\right)$$

$$= \underline{140.6^{\circ}F}$$

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

Therefore,

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

$$= 1.25\ (50 - \underline{27.2}) + 2.36\ (75-50)$$

$$= \underline{87.5}^{\circ}F$$

If $PCT_N$ occurs when the core reflood rate is greater than 1.0 inch per second $\Delta PCT_4 = 0$. The total potential PCT increase for the non burst node is then

$$\Delta PCT_5 = \Delta PCT_3 + \Delta PCT_4 = 228.1^{\circ}F$$

Margin to the 2200°F limit is

$$\Delta PCT_6 = 2200^{\circ}F - PCT_N = 19.8^{\circ}F$$

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

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

$\Delta FQ_N = \underline{0.208}$ but not less than zero.

The peaking factor reduction required to maintain the 2200°F clad temperature limit is therefore the greater of $\Delta FQ_B$ and $\Delta FQ_N$,

or; $\Delta FQ_{PENALTY} = \underline{0.21}$

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

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

$$FQ\ ADJUSTMENT = \underline{0.15} - \underline{0.21}$$

$$= -0.06$$