ATTACHMENT 1

TECHNICAL SPECIFICATIONS CHANGES

FOR

INCREASED FQ(Z) WITH 18% STEAM GENERATOR TUBE PLUGGING

NORTH ANNA UNITS 1 AND 2

8810070238 880930
PDR ADOCK 05000338
P PDC

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.19}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.38] [K(Z)] \text{ for } P \leq 0.5$$

$$\text{where } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% $F_Q(Z)$ exceeds the limit.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Satisfying the following relationship:

$$F_Q^M(z) \le \frac{2.19 \quad x \quad K(z)}{P \times N(z)} \text{ for } P > 0.5$$

$$F_Q^M(z) \le \frac{2.19 \quad x \quad K(z)}{N(z) \times 0.5} \text{ for } P \le 0.5$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, 2.19 is the F_Q limit, K(z) is given in Figure 3.2-2, P is the relative THERMAL POWER, and N(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. This function is given in the Core Surveillance Report as per Specification 6.9.1.7.

d. Measuring $F_Q^M(z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding the THERMAL POWER at which $F_Q(z)$ was last determined by 10% or more of RATED THERMAL POWER*, or

2. At least once per 31 effective full power days, whichever occurs first.

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right)$$

has increased since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

*During power escalation, the power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

SURVEILLANCE REQUIREMENTS (Continued)

1. $F_Q^M(z)$ shall be increased by 2% over that specified in 4.2.2.2.c, or

2. $F_Q^M(z)$ shall be measured at least once per 7 effective full power days until 2 successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right) \text{ is not increasing.}$$

f. With the relationships specified in 4.2.2.2.c above not being satisfied:

1. Calculate the percent $F_Q(z)$ exceeds its limit by subtracting one from the measurement/limit ratio and multiplying by 100:

$$\left\{ \underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{\frac{2.19 \times K(z)}{P \times N(z)}} \right) -1 \right\} \times 100 \text{ for } P \geq 0.5$$

$$\left\{ \underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{\frac{2.19 \times K(z)}{0.5 \times N(z)}} \right) -1 \right\} \times 100 \text{ for } P < 0.5$$

2. Either of the following actions shall be taken:

   a. Power operation may continue provided the AFD limits of Figure 3.2-1 are reduced 1% AFD for each percent $F_Q(z)$ exceeded its limit, or

   b. Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above.

g. The limits specified in 4.2.2.2.c, 4.2.2.2.e, and 4.2.2.2.f above are not applicable in the following core plane regions:

   1. Lower core region 0 to 15 percent inclusive.

   2. Upper core region 85 to 100 percent inclusive.

4.2.2.3 When $F_Q(z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2, an overall measured $F_Q(z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to acount for measurement uncertainty.

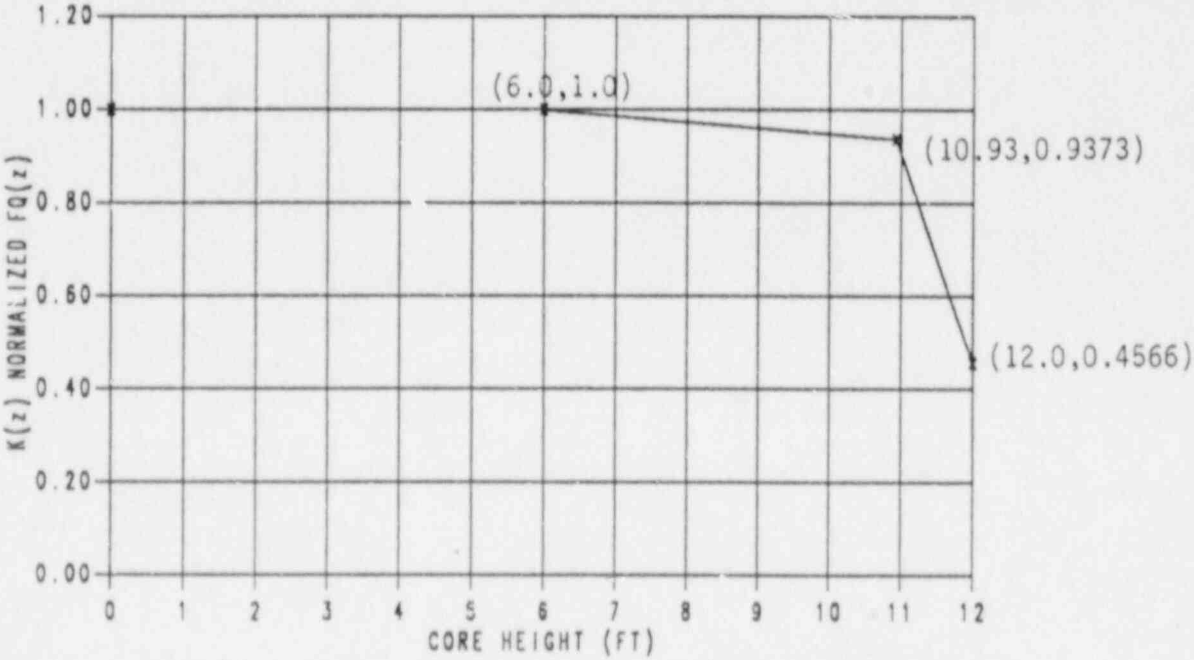

Figure 3.2-2 NORMALIZED FQ(z) AS A FUNCTION OF CORE HEIGHT

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.19}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.38] [K(Z)] \text{ for } P \leq 0.5$$

$$\text{where } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% $F_Q(Z)$ exceeds the limit.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

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a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{2.19 \times K(z)}{P \times N(z)} \text{ for } P > 0.5$$

$$F_Q^M(z) \leq \frac{2.19 \times K(z)}{N(z) \times 0.5} \text{ for } P \leq 0.5$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, 2.19 is the F_Q limit, K(z) is given in Figure 3.2-2, P is the relative THERMAL POWER, and N(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. This function is given in the Core Surveillance Report as per Specification 6.9.1.7.

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1. Upon achieving equilibrium conditions after exceeding the THERMAL POWER at which $F_Q(z)$ was last determined by 10% or more of RATED THERMAL POWER*, or

2. At least once per 31 effective full power days, whichever occurs first.

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right)$$

has increased since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

*During power escalation, the power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

SURVEILLANCE REQUIREMENTS (Continued)

1. $F_Q^M(z)$ shall be increased by 2% over that specified in 4.2.2.2.c, or

2. $F_Q^M(z)$ shall be measured at least once per 7 effective full power days until 2 successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right) \text{ is not increasing.}$$

f. With the relationships specified in 4.2.2.2.c above not being satisfied:

1. Calculate the percent $F_Q(z)$ exceeds its limit by subtracting one from the measurement/limit ratio and multiplying by 100:

$$\left\{ \underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{\frac{2.19 \times K(z)}{P \times N(z)}} \right) -1 \right\} \times 100 \text{ for } P \geq 0.5$$

$$\left\{ \underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{\frac{2.19 \times K(z)}{0.5 \times N(z)}} \right) -1 \right\} \times 100 \text{ for } P < 0.5$$

2. Either of the following actions shall be taken:

   a. Power operation may continue provided the AFD limits of Figure 3.2-1 are reduced 1% AFD for each percent $F_Q(z)$ exceeded its limit, or

   b. Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above.

g. The limits specified in 4.2.2.2.c, 4.2.2.2.e, and 4.2.2.2.f above are not applicable in the following core plane regions:

   1. Lower core region 0 to 15 percent inclusive.

   2. Upper core region 85 to 100 percent inclusive.

4.2.2.3 When $F_Q(z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2, an overall measured $F_Q(z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to acount for measurement uncertainty.

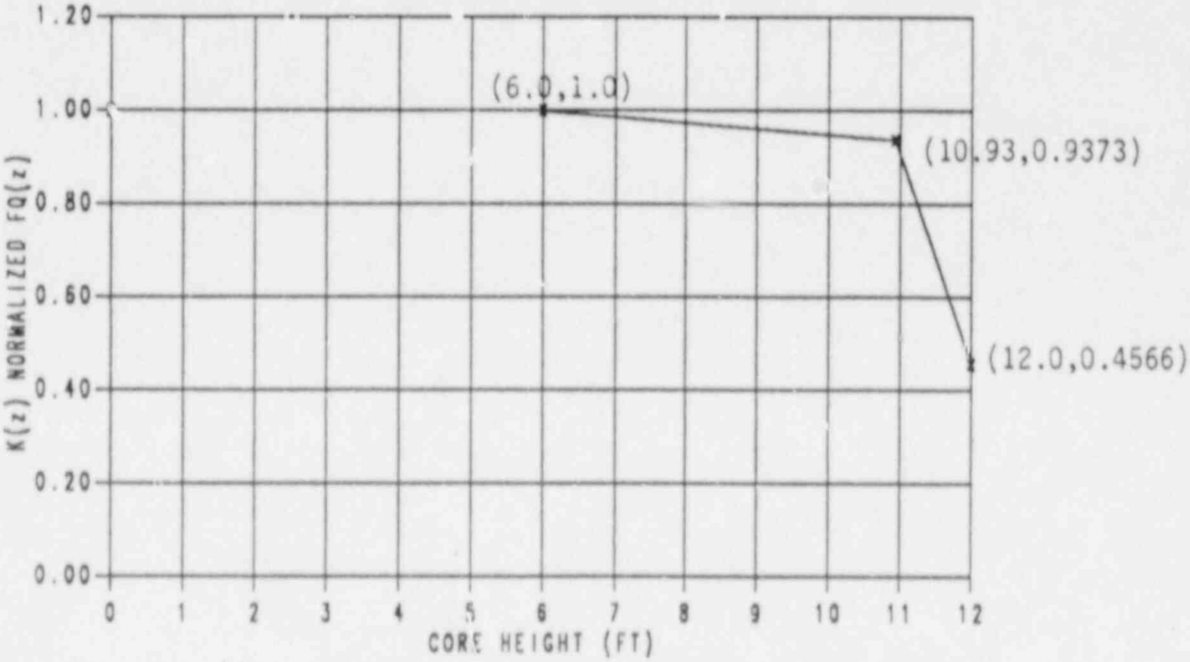


Figure 3.2-2 NORMALIZED FQ(z) AS A FUNCTION OF CORE HEIGHT

ATTACHMENT 2

LOCA-ECCS SAFETY EVALUATION

FOR

INCREASED FQ(Z) WITH 18% STEAM GENERATOR TUBE PLUGGING

NORTH ANNA UNITS 1 AND 2

1.0 INTRODUCTION

A reanalysis of the Emergency Core Cooling System (ECCS) performance for the postulated large-break LOCA has been performed in compliance with Appendix K to 10 CFR 50. The results of this re-analysis are presented here, 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 1981 model with BART version of the Westinghouse LOCA-ECCS evaluation model (Ref. 1 and 2). The analysis includes the evaluation model revisions described in Reference 16 and approved by the NRC in Reference 17. 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. All assumptions and initial operating conditions used in this reanalysis were the same as those used in previous LOCA-ECCS analyses (Ref. 3 and 19), with two exceptions. The steam generator plugging level was increased to 18% (from 7% and 15% in References 19 and 3, respectively) and the maximum core peaking factor, FQ, was increased from 2.15 to 2.19. With these changes incorporated into the analysis, it was found that the LOCA analysis results continue to meet the 10 CFR 50.46 acceptance criteria.

2.0 ACCIDENT DESCRIPTION

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 UFSAR. Sensitivity studies (Ref. 7) have indicated that a double-ended cold-leg guillotine (DECLG) pipe break is limiting. Should a DECLG break occur, rapid depressurization of the reactor coolant system occurs. 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, activating the high-head safety injection pumps. The actuation and subsequent activation of the Emergency Core Cooling System, 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. 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 temperature.

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 continue to be transferred to the reactor coolant system. At the beginning of the blowdown phase, the entire reactor coolant system contains subcooled liquid that transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break develops, the time to DNB is calculated, 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 system 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 RCS pressure. When the reactor coolant system depressurizes to 594 psia, the accumulators begin to inject borated water into the reactor coolant loops. The conservative assumption is then made that 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 refueling water storage tank (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 operate to return the containment environment to subatmospheric pressure.

3.0 ANALYSIS

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 reactor coolant system, 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 interrelated computer codes has been developed for the analysis.

The description of the various aspects of the LOCA analysis methodology is given in WCAP-8339 (Ref. 8). This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes that ensure compliance with 10 CFR 50, Appendix K. The SATAN-VI, COCO, WREFLOOD, BART, and LOCTA-IV codes, which are used in the LOCA analysis, are described in detail in WCAP-8306 (Ref. 9), WCAP-8326 (Ref 10), WCAP-8171 (Ref. 11), WCAP-9695 (Ref. 4) and WCAP-10062 (Ref. 5), and WCAP-8305 (Ref. 12), respectively. The BART code used for this analysis includes the revisions described by References 6, 16 and 17. These codes 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 reactor coolant system during blowdown, and the COCO computer code calculates the containment pressure transient during all three phases of the LOCA analysis. The thermal-hydraulic response of the reactor coolant system during refill and reflood is calculated by the WREFLOOD computer code. A mechanistic estimate of the heat transfer coefficient in the core during reflood is provided by the BART computer code. For the three phases of the LOCA, the LOCTA-IV computer code is used to compute the thermal transient of the hottest fuel rod.

SATAN-VI is used to determine the RCS pressure, enthalpy, and density, as well as the mass and energy flow rates in the reactor coolant system 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 the blowdown, including the core inlet flowrate 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 flowrate to the containment depends upon the core pressure, which is a function of the containment backpressure, the WREFLOOD and COCO codes are interactively linked. With the input and boundary conditions from WREFLOOD, the mechanistic core heat transfer model in BART calculates the fluid and heat transfer conditions in the core during reflood.

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 outputs from SATAN-VI, WREFLOOD and BART, 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 chopped cosine curve that has been previously verified (Ref. 13) 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 flowrates 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.

4.0 NON-LOCA SAFETY EVALUATION FOR 18% STEAM GENERATOR TUBE PLUGGING

This North Anna Power Station LOCA-ECCS reanalysis has evaluated plant operation at steam generator tube plugging levels of up to 18% based on the acceptance criteria delineated in 10CFR50.46. An evaluation has been performed which concluded that reanalysis of non-LOCA accidents is not required to support this increased tube plugging level provided the measured RCS flow rate remains above the thermal design flow rate assumed for the safety analyses. Steam generator tube plugging in sufficient quantity can potentially affect non-LOCA safety analysis due to reduced primary system flow, more severe pump coastdown characteristics, and the reduction of the reactor primary coolant system volume. Primary flowrate becomes a key parameter in DNB limited events (e.g., Uncontrolled RCCA Bank Withdrawal at Power) when it falls below the thermal design flowrate. Pump coastdown characteristics impact analysis results when they become more severe than the conservative values used in the loss-of-flow related analyses. The reduced primary coolant system volume affects dilution times in uncontrolled boron dilution events.

A conservative estimate of North Anna RCS flow versus tube plugging is provided in Reference 18. This estimate is based on past flow measurements taken at the North Anna Power Station for several levels of steam generator tube plugging. More recent North Anna Unit 1 measurements at greater tube plugging levels validate the conservatism of the Reference 18 curve. A re-evaluation of the projection presented in Reference 18 indicates that the conservatively estimated flow rate at the proposed 18% plugging level is approximately equal to the North Anna thermal design flow. Therefore, while measured flow exceeds the thermal design flow, the current docketed licensing analyses remain valid for those events in which flow rate is an important concern.

The loss-of-flow related analyses in Reference 15 used a limiting reactor coolant pump flow coastdown characteristic with the limiting initial thermal design flow rate. Since the conservatively estimated system flow rate equals the thermal design value, the coastdown flows for the 18% plugging level will be bounded by the coastdown flows in the Reference 15 analyses.

The impact of 18% tube plugging on dilution times in the uncontrolled boron dilution events was evaluated with respect to the analyses documented in Reference 15. Relative to the boron dilution events, the evaluation indicated:

- °For uncontrolled dilution during startup, time to criticality is 37 minutes. This is more than adequate time for the operator to recognize the high count rate signal and terminate the dilution flow.
- °For uncontrolled dilution at power, the operator has ample time (greater than 15 minutes) after the over-temperature T alarm or trip to determine the cause of dilution, isolate the water source, and initiate reboration before total shutdown margin is lost due to dilution.

Tube plugging levels exhibit no influence on dilution times for the refueling mode of operation, since the steam generator volumes are not a part of the active system.

This evaluation shows that for steam generator tube plugging levels of up to 18 percent, no reanalysis of the DNBR related non-LOCA safety events is necessary and that the currently licensed analyses remain valid. In the case of the uncontrolled boron dilution events, the available operator response times for the startup and at power evaluations are reduced but remain well above the minimum acceptance values.

5.0 LARGE BREAK LOCA 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 for the C_D = 0.4 and 0.6 discharge coefficients. The double-ended cold-leg guillotine break has been determined to be the limiting break size and location based on the sensitivity studies reported in Reference 7. The analysis resulted in a limiting peak clad temperature of 2165.2°F for the C_D = 0.4 case, a maximum local cladding oxidation level of 5.77%, 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 2A through 18B. The figures show the following:

1. Peaking Factor vs. Core Height - Figure 1 shows the chopped cosine power shape used in the analysis.

2. Mass Velocity - Figures 2A and 2B show the mass velocity at the clad burst and hot-spot locations on the hottest fuel rod for the discharge coefficient used.

3. Heat Transfer Coefficient - Figures 3A and 3B show the heat transfer coefficient at the clad burst and hot-spot locations on the hottest rod for the discharge coefficient used. The values of heat transfer coefficient that are shown were calculated by the LOCTA-IV code prior to reflooding and the BART code for the remainder of the transient. These are based on equations for heat transfer in the nucleate boiling, transition boiling, film boiling, and steam cooling regimes.

4. Core Pressure - Figures 4A and 4B show the calculated pressure in the core for the discharge coefficient used.

5. Break Flowrate - Figures 5A and 5B show the calculated flowrate out of the break for the discharge coefficient used. The flowrate out of the break is plotted as the sum of flow at both the pressure vessel end and the reactor coolant pump end of the guillotine break.

6. Core Pressure Drop - Figures 6A and 6B show the calculated core pressure drop for the discharge coefficient used. The core pressure drop is interpreted as the pressure immediately before entering the core inlet to the pressure just outside the core outlet.

7. Peak Clad Temperature - Figures 7A and 7B show the calculated hot-spot clad temperature transient and the clad temperature transient at the burst location for the discharge coefficient used. The peak clad temperature for the limiting discharge coefficient of 0.4 is 2165.2°F at the 8.00 ft elevation in the core.

8. Fluid Temperature - Figures 8A and 8B show the calculated fluid temperature for the hot spot and burst locations for the discharge coefficient used.

9. Core Flow - Figures 9A and 9B show the calculated core flow, both top and bottom, for the discharge coefficient used.

10. Reflood Transient - Figures 10A and 10B show the reactor pressure vessel downcomer and core water levels for the discharge coefficient used. Figures 11A and 11B show the core inlet velocity for the discharge coefficient used.

11. Accumulator Flow - Figures 12A and 12B show the calculated flow for the discharge coefficient used. The accumulator delivery during blowdown is discarded until the end of bypass is calculated. Accumulator flow, however, is established in the refill-reflood calculations. The accumulator flow assumed is the sum of that injected in the intact cold legs.

12. Pumped ECCS Flow (Reflood) - Figures 13A and 13B show the calculated flow of the emergency core cooling system for the discharge coefficient used.

13. Containment Pressure - Figures 14A and 14B show the calculated pressure transient for the discharge coefficient used. The analysis of this pressure transient is based on the data given in Tables 2, 5, and 6.

14. Core Power Transient - Figures 15A and 15B show the core power transient calculated by the SATAN-VI code for the discharge coefficient used.

15. Break Energy Release - Figure 16A and 16B show the break energy released to the containment for the discharge coefficient used.

16. Containment Wall Heat Transfer - Figure 17A and 17B show the containment wall heat transfer coefficient for the discharge coefficient used.

17. Fluid Quality - Figures 18A and 18B show the fluid quality at the clad burst and hot-spot locations (location of maximum clad temperature) on the hottest fuel rod (hot rod) for the limiting breaks.

6.0 CONCLUSIONS

For breaks up to and including the double-ended rupture of a reactor coolant pipe, and for the operating conditions specified in Tables 1 and 2, the emergency core cooling system will meet the acceptance criteria as presented in 10 CFR 50.46, as follows:

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% 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.

The effects of increasing the allowable steam generator tube plugging to 18% has been assessed for existing non-LOCA event analyses. This evaluation has concluded:

1. Current analyses for which RCS flow is an important concern remain valid as long as measured flow is greater than the thermal design flow assumed in safety analyses.

2. The existing loss-of-flow related analyses assume a conservative reactor coolant pump flow coastdown characteristic which accommodates the effect of increased tube plugging on loop flow resistance.

3. Boron dilution analyses assuming the reduced RCS volume associated with tube plugging result in dilution times which remain adequate for the required operator actions to be performed.

10 CFR 50.59 SAFETY EVALUATION

The proposed limit changes for steam generator tube plugging and FQ have been reviewed against the criteria of 10 CFR 50.59 and were concluded not to involve any unreviewed safety question. The specific bases for this determination are as follows:

1. Since the proposed changes involve parameters which are not accident initiators, they will not increase the probability of occurrence of any malfunction or accident previously addressed. The reanalyzed large break LOCA analysis verifies that operation under the revised specifications would also not result in any increase in accident consequences over those in previously accepted analyses.

2. No new accident types or equipment malfunction scenarios will be introduced as a result of operating in accordance with the revised specifications. The change which potentially affects physical components in the plant systems (steam generator tube plugging) was explicitly included in the analysis and shown not to produce any new or unique accident precursors.

3. The margin of safety, as defined in the basis for the plant Technical Specifications, is not reduced. The revised ECCS analysis meets the acceptance criteria of 10 CFR 50.46. Additionally, since evaluation of non-LOCA accidents concluded that acceptance criteria are met when considering the proposed changes, the current margin of safety is maintained for LOCA and non-LOCA accidents.

8.0 REFERENCES

1. Letter from J. R. Miller, NRC, to E. P. Rahe, Westinghouse, "Acceptance for Referencing of the 1981 Version of the Westinghouse Large Break ECCS Evaluation Model," December 1, 1981.

2. Letter from C. O. Thomas, NRC to E. P. Rahe, Westinghouse, "Acceptance for Referencing of Licensing Topical Report WCAP-9561, BART A-1: A Computer Code for Best Estimate Analyses of Reflood Transients," December 21, 1983, and Addenda 1 and 2.

3. Letter from W. L. Stewart, Vepco, to U.S. Nuclear Regulatory Commission, Serial No. 87-486, dated September 11, 1987.

4. Young, M. Y. et al., BART-A1: A Computer Code for the Best Estimate Analysis of Reflood Transients, WCAP-9695, January 1980.

5. Chiou, J. S. et al., Models for PWR Reflood Calculations using the BART Code, WCAP-10062, December 1981.

6. Letter from C. O. Thomas, NRC, to E. P. Rahe, Westinghouse, "Acceptance for Referencing of Licensing Topical Report WCAP-10484(P), Spacer Grid Heat Transfer Effects During Reflood," June 21, 1984.

7. R. Salvatori, Westinghouse ECCS Sensitivity Studies, WCAP-8356, July 1974.

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

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

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

11. R. D. Kelly et al., Calculational Model for Core Reflooding After a Loss-of-Coolant Accident (WREFLOOD Code), WCAP-8171, June 1974.

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

13. Letter from C. M. Stallings, Vepco, to E. G. Case, NRC, Serial No. 092, dated February 17, 1978.

14. Letter from C. M. Stallings, Vepco, to E. G. Case, NRC, Serial No. 344, August 9, 1977.

15. Updated Final Safety Analysis Report - North Anna Power Station Units 1 and 2, Virginia Electric and Power Company, Rev. 6, June 1987.

16. M. Y. Young, "Addendum to BART-A1: A Computer Code for the Best Estimate Analysis of Reload Transients " (Special Report: Thimble Modeling in Westinghouse ECCS Evaluation Model): WCAP-9561-P, Addendum 3, Revision 1, July, 1986.

17. Letter from Charles E. Rossi, NRC, to E. P. Rahe, Westinghouse, "Acceptance for Referencing of Licensing Topical Report WCAP-9561, Addendum 3, Revision 1," August 25, 1986.

18. Letter from R. H. Leasburg, Vepco, to H. R. Denton, NRC, Serial No. 080, February 12, 1982.

19. Letter from W. L. Stewart, Vepco, to H. R. Denton, NRC, Serial No. 85-077, dated May 2, 1985.

TABLE 1

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

Calculational Input

Core Power (MWt) 102% of 2893	2951
Peak linear power (kW/ft), 102% of 12.45	12.70
Heat flux hot-channel factor (F_Q)	2.19
Enthalpy rise hot-channel factor ($F_{\Delta H}^N$)	1.55
Accumulator water volume (ft^3, each)	1025
Reactor vessel upper head temperature equal to T_{hot}	

Limiting Fuel Region and Cycle	Cycle	Region
Unit 1	All	All regions
Unit 2	All	All regions

TABLE 2 CONTAINMENT DATA (DRY CONTAINMENT)

Net Free Volume	1.916×10^6 ft^3
Initial Conditions	
Pressure (total), psia	9.50
Temperature, °F	90
RWST temperature, °F	35
Outside temperature, °F	-10
Containment Quench Spray System	
Number of pumps operating	2
Runout flowrate (each), gpm	2000
Actuation time, sec	59
Structural Heat Sinks	
Type/thickness (in.)	Area (ft^2), with uncertainty
Concrete/6	8,393
Concrete/12	62,271
Concrete/18	55,365
Concrete/24	11,591
Concrete/27	9,404
Concrete/36	3,636
Carbon steel/0.375, Concrete/54	22,039
Carbon Steel/0.375, Concrete/54	28,933
Carbon steel/0.50, Concrete/30	25,673
Concrete/26.4 (floor), Carbon Steel/0.25, Concrete/120	12,110
Carbon steel/0.371	160,328
Stainless Steel/0.407	10,527
Carbon Steel/0.882	9,894
Carbon Steel/0.059	60,875

TABLE 3
TIME SEQUENCE OF EVENTS FOR DECLG

	$C_D = 0.4$ (sec)	$C_D = 0.6$ (sec)
Start	0.0	0.0
Reactor trip	0.630	0.615
Safety injection signal	2.60	2.05
Accumulator injection	16.7	12.7
Pump injection	27.60	27.05
End of bypass	32.036	26.286
End of blowdown	32.036	26.286
Bottom of core recovery	45.843	39.826
Accumulator empty	56.516	51.364

TABLE 4
RESULTS FOR DECLG

	$C_D = 0.4$	$C_D = 0.6$
Peak clad temperature, °F	2165.2	1971.7
Peak clad locati n, ft	8.0	7.25
Local Zr/H_2O reac ion (max), %	5.77	3.38
Local Zr/H_2O location, ft	5.50	6.50
Total Zr/H_2O reaction, %	<0.3	<0.3
Hot-rod burst time, sec	40.60	63.80
Hot-rod burst location, ft	5.50	6.50

TABLE 5

REFLOOD MASS AND ENERGY RELEASES DECLG (C_D = 0.4)

Time (sec)	Total Mass Flow Rate (lb/sec)	Total Energy Flow Rate (10^5 Btu/sec)
45.843	0.0	0.0
46.468	0.66	0.009
56.810	86.77	1.078
71.860	141.93	1.243
90.360	240.25	1.454
110.760	257.99	1.425
132.860	264.43	1.386
169.510	308.53	1.415

TABLE 6

BROKEN LOOP ACCUMULATOR FLOW TO CONTAINMENT DECLG (C_D = 0.4)

Time (sec)	Mass Flow Rate[a] (lbm/sec)
0.00	4095.55
1.01	3691.57
3.01	3155.57
5.01	2801.97
7.01	2542.34
10.01	2250.67
15.01	1913.20
20.01	1681.07
25.01	1519.24
30.01	1552.21

[a] For energy flowrate, multiply mass flow rate by a constant of 59.62 Btu/lbm.

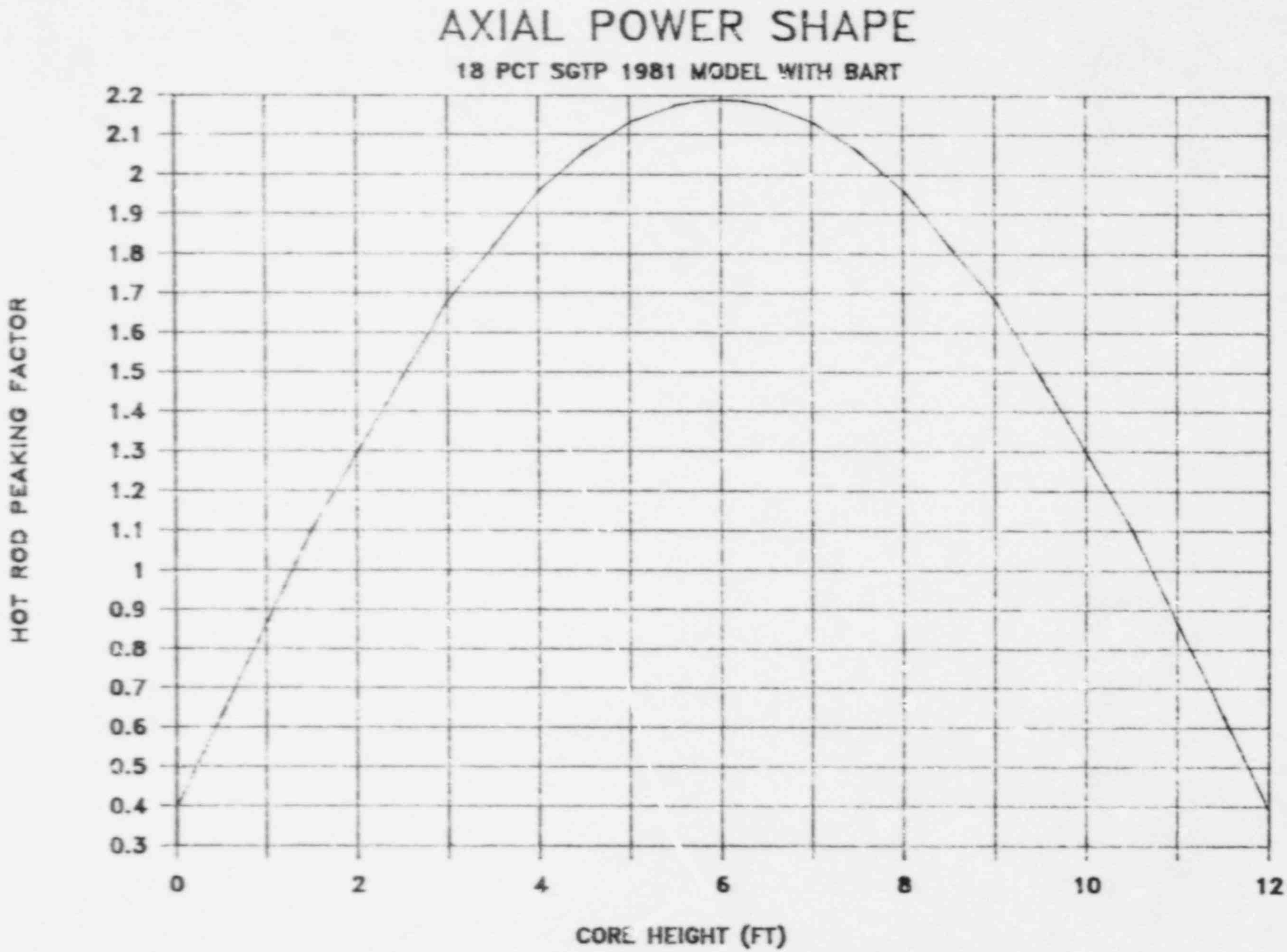


FIGURE 1

PEAKING FACTOR VERSUS CORE HEIGHT - FQ = 2.19

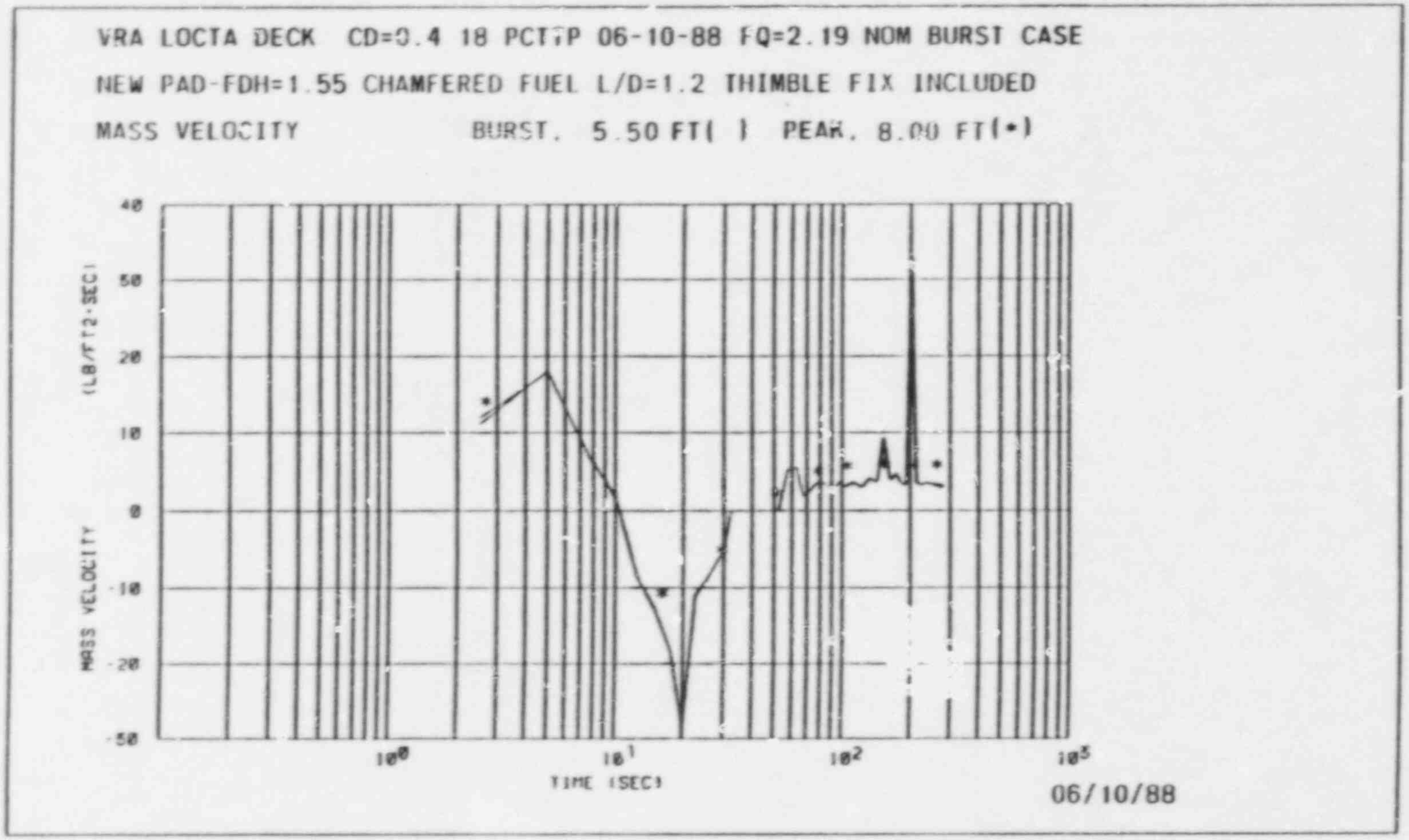


FIGURE 2A
MASS VELOCITY VERSUS TIME
DECLG (CD = 0.4)

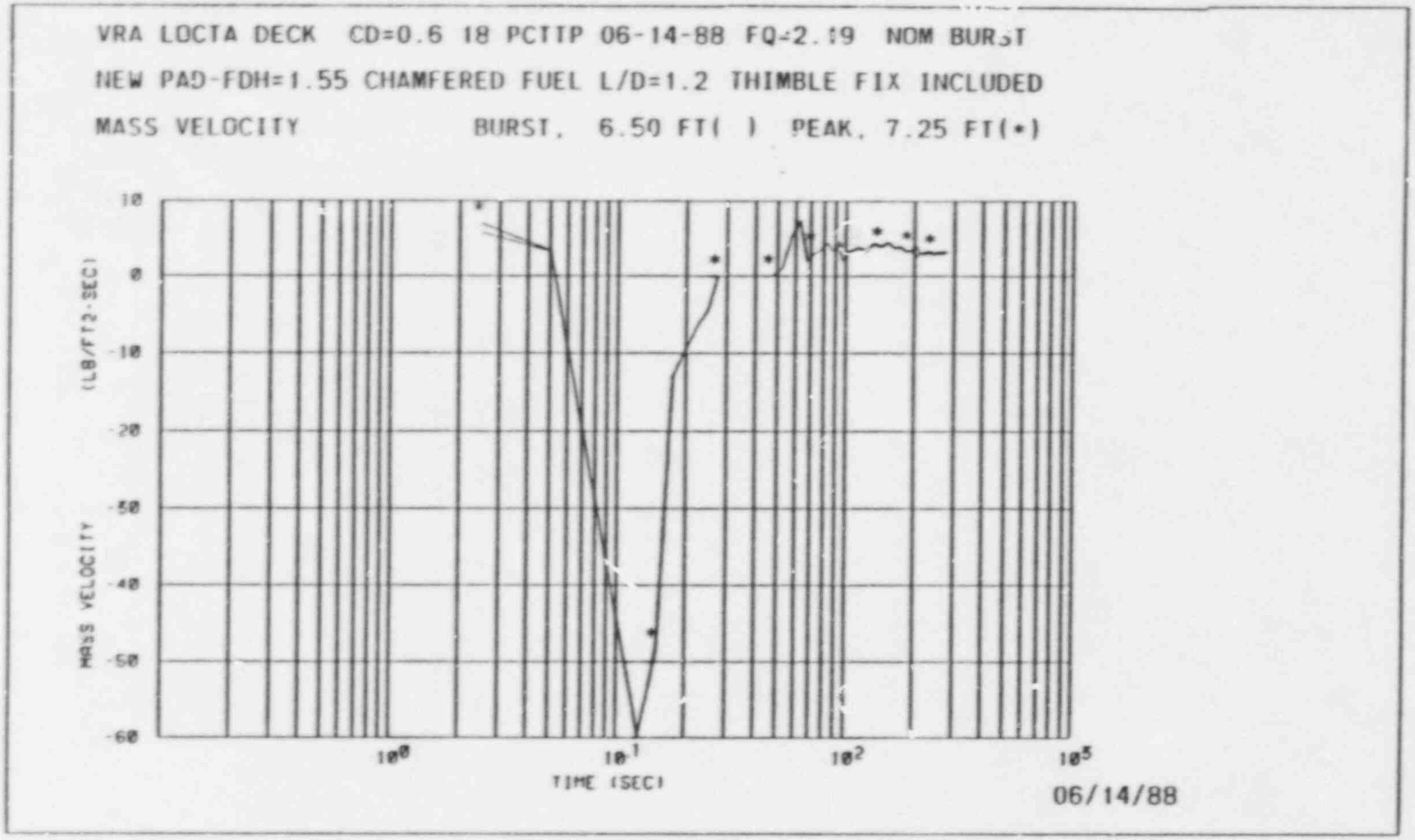


FIGURE 2B
MASS VELOCITY VERSUS TIME
DECLG (CD = 0.6)

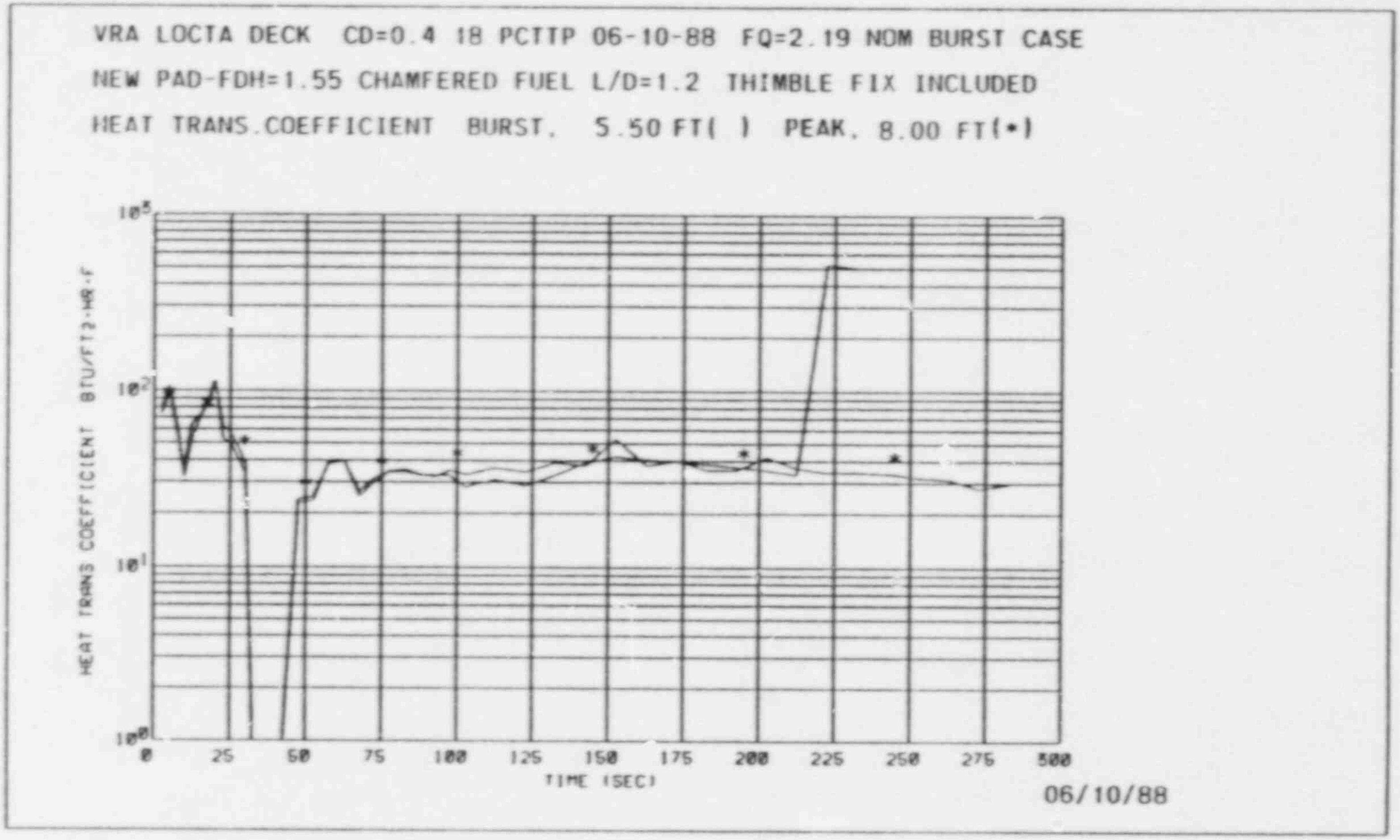


FIGURE 3A
HEAT TRANSFER COEFFICIENT VERSUS TIME
DECLG (CD = 0.4)

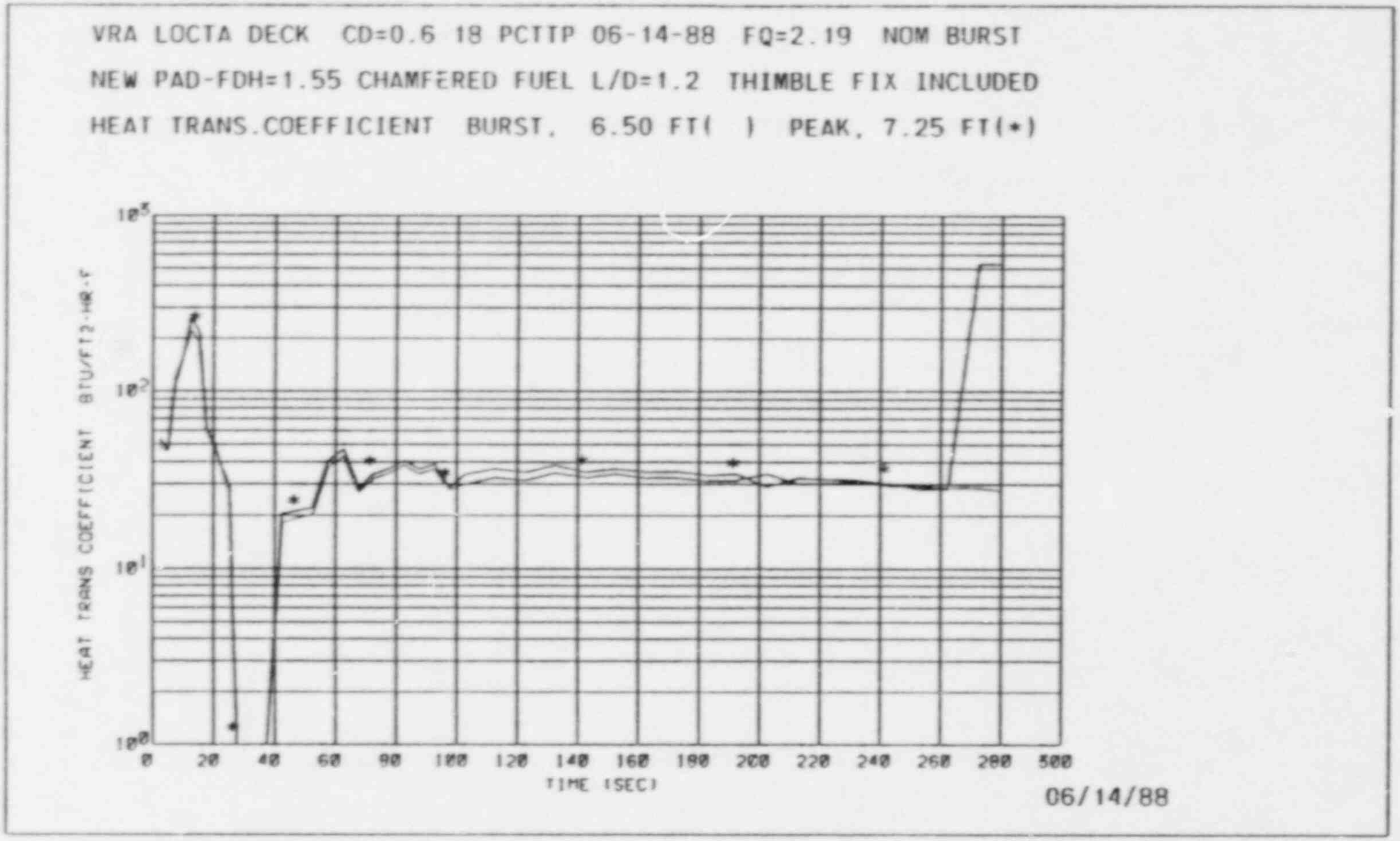


FIGURE 3B
HEAT TRANSFER COEFFICIENT VERSUS TIME
DECLG (CD = 0.6)

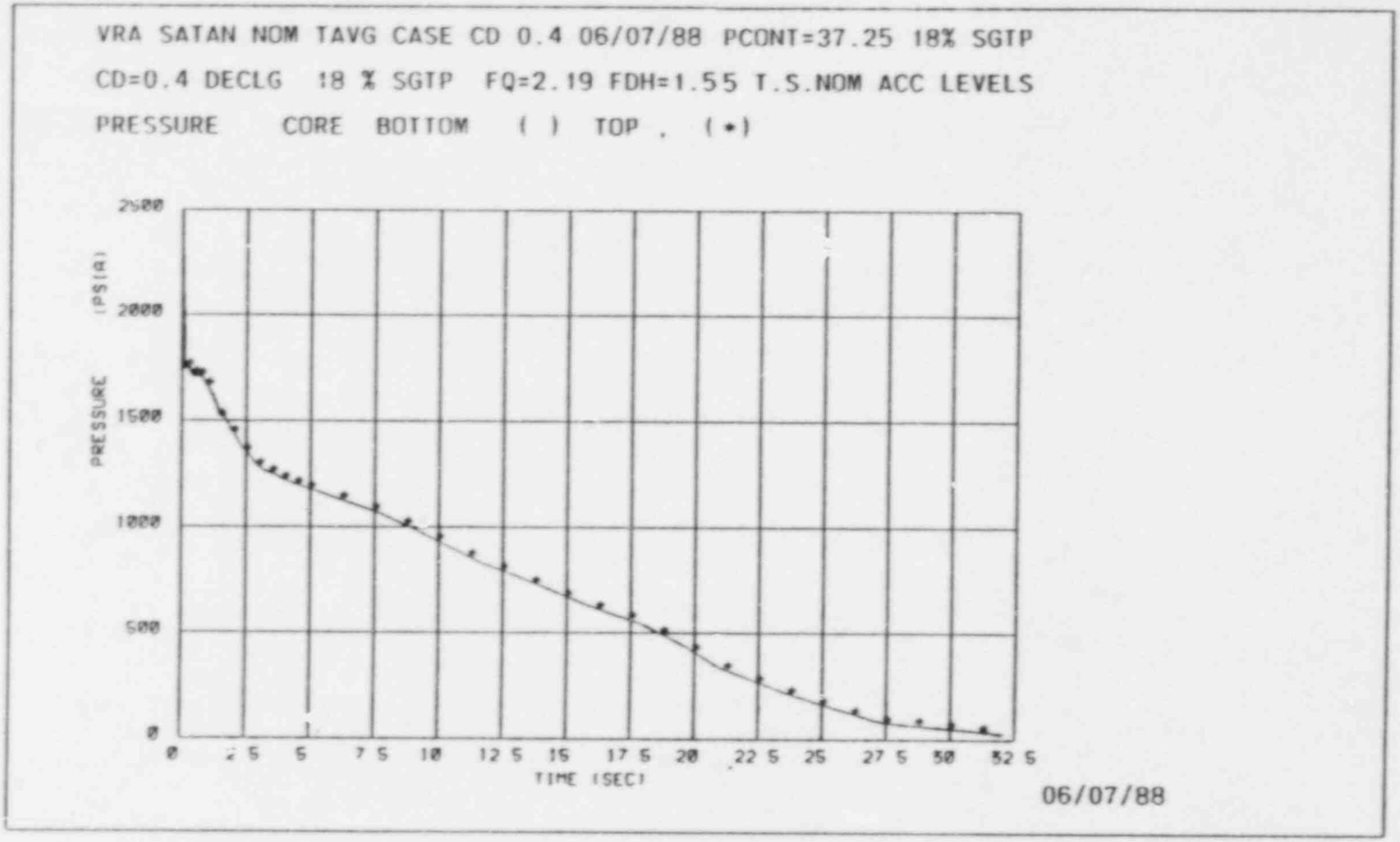


FIGURE 4A

CORE PRESSURE VERSUS TIME

DECLG (CD = 0.4)

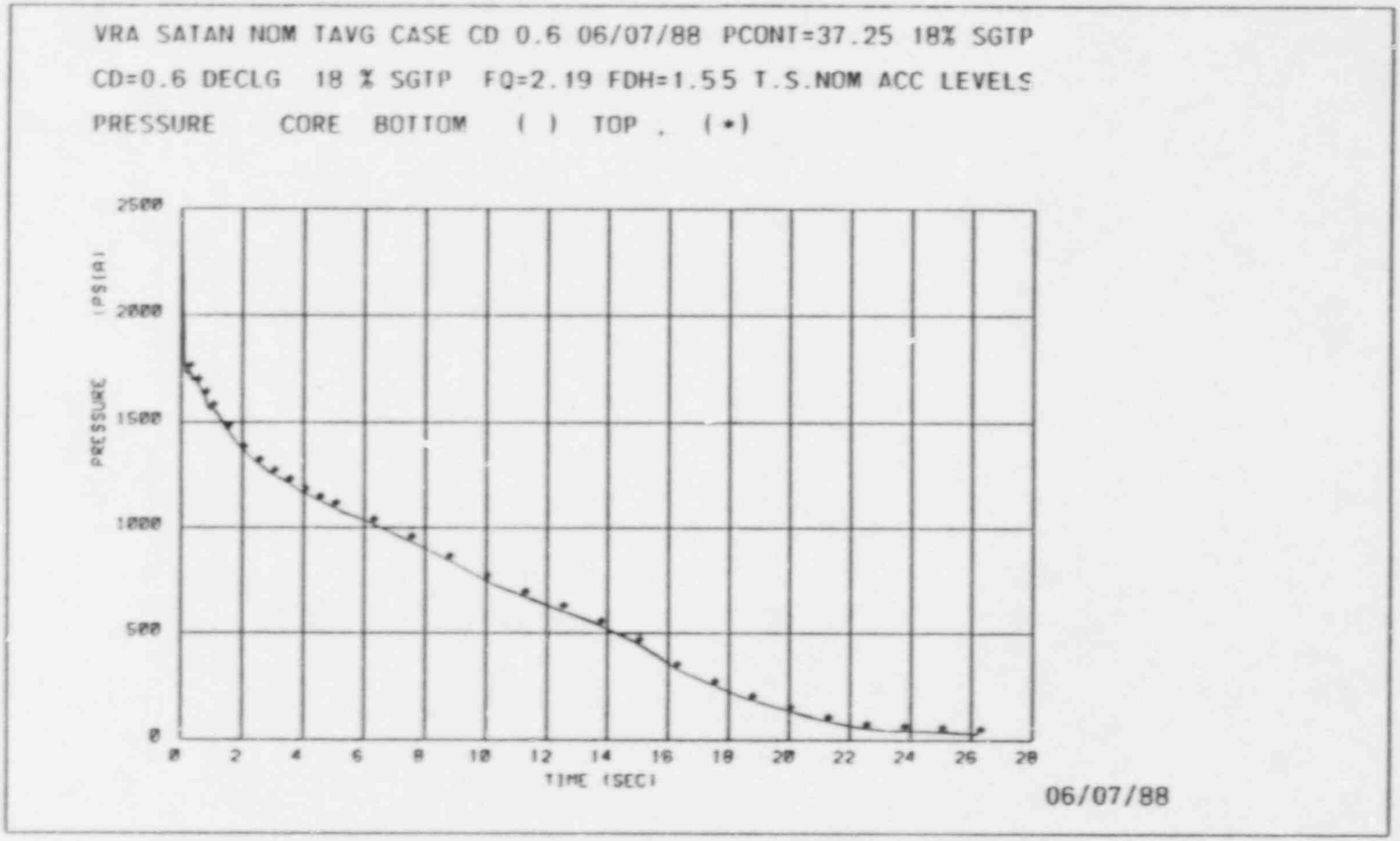


FIGURE 4B

CORE PRESSURE VERSUS TIME

DECLG (CD = 0.6)

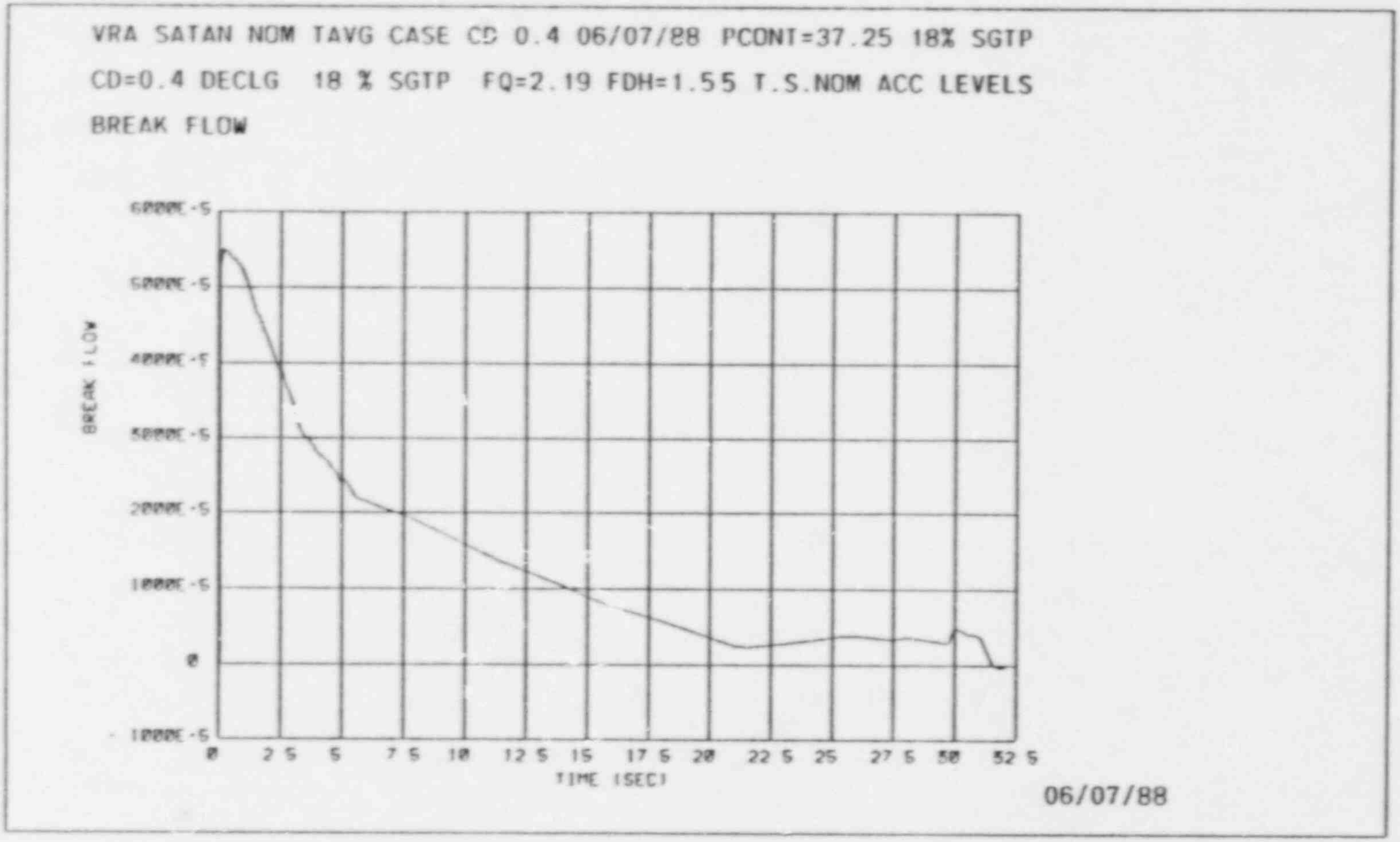


FIGURE 5A

BREAK FLOW RATE VERSUS TIME

DECLG (CD = 0.4)

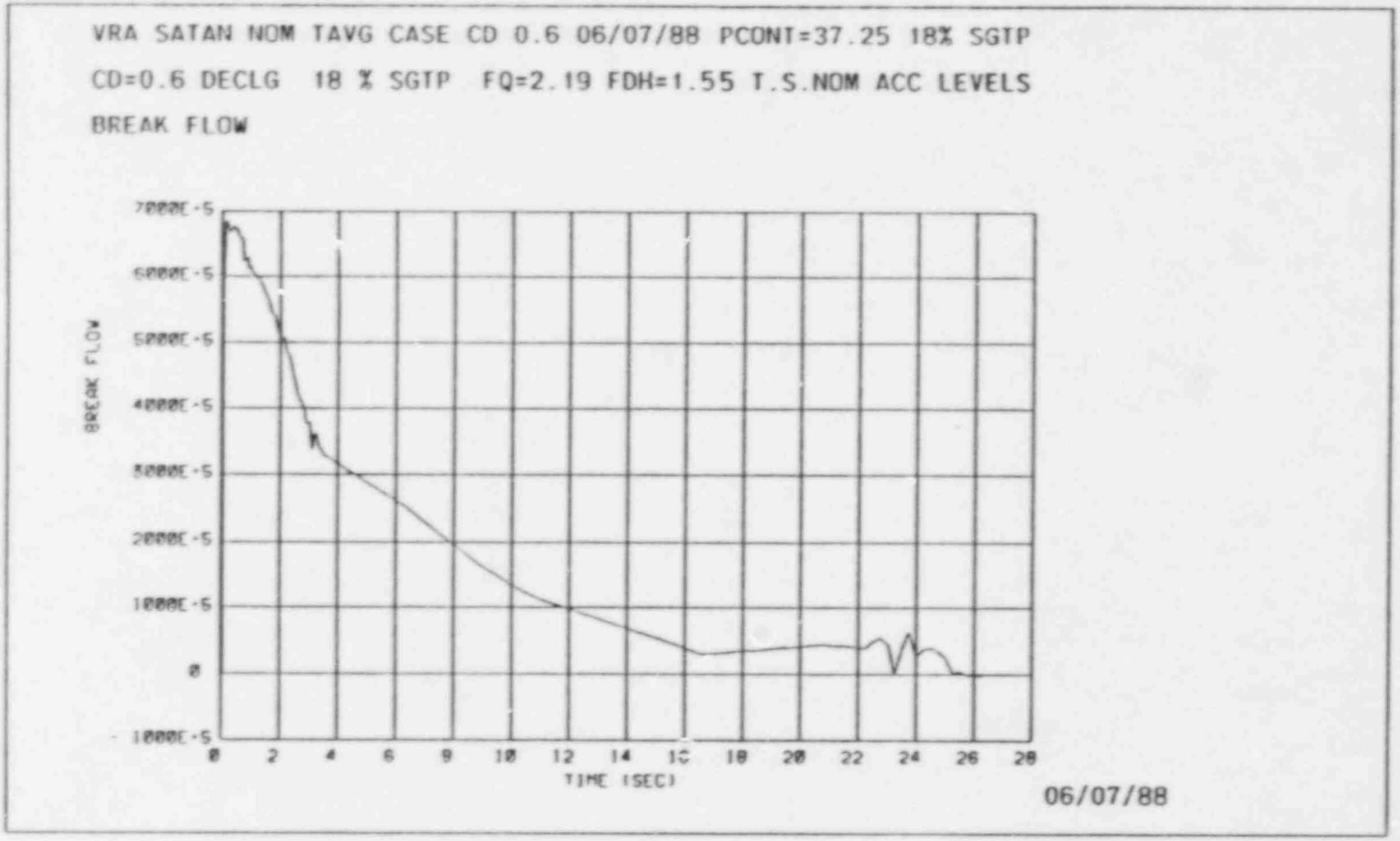


FIGURE 5B

BREAK FLOW RATE VERSUS TIME

DECLG (CD = 0.6)

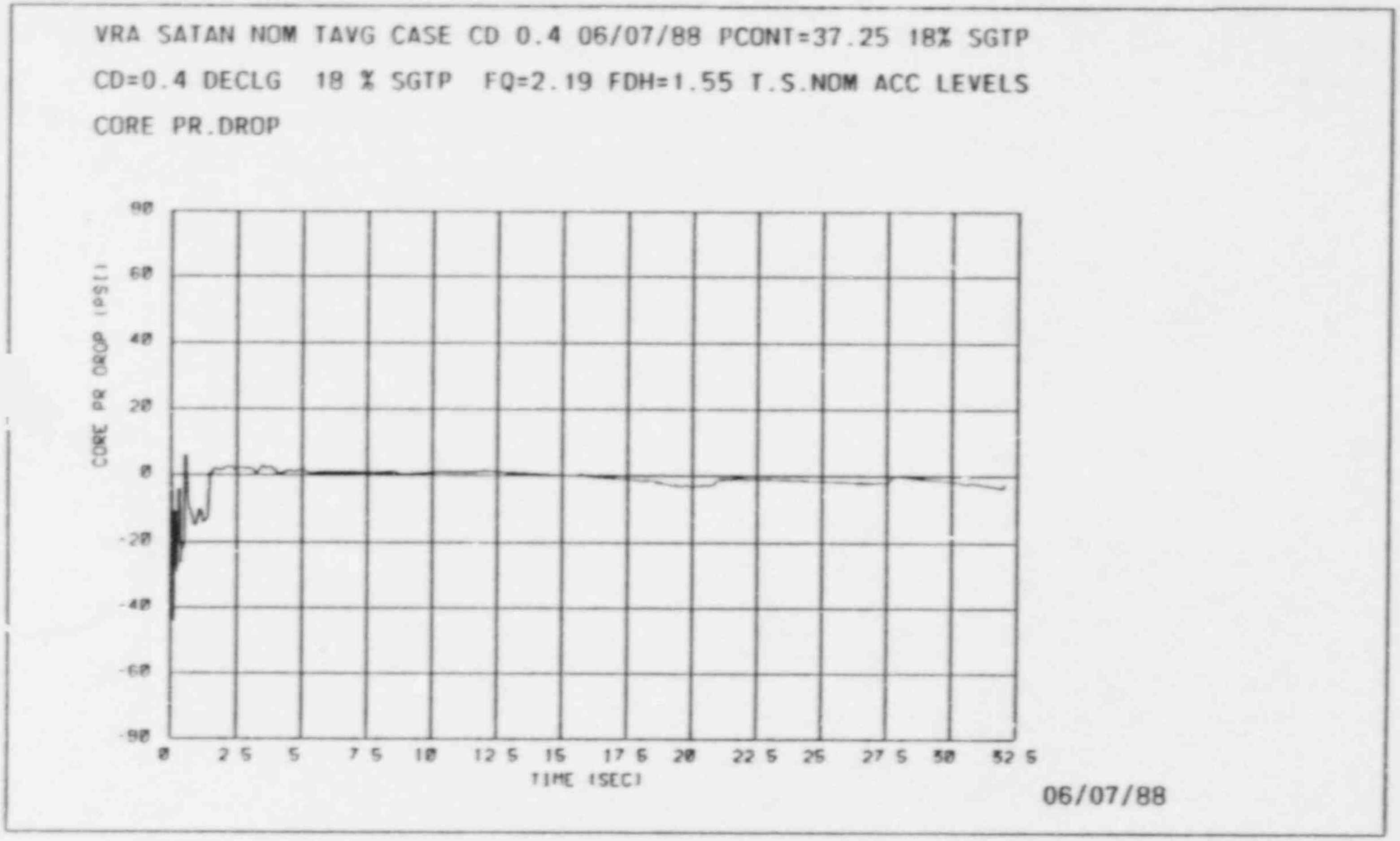
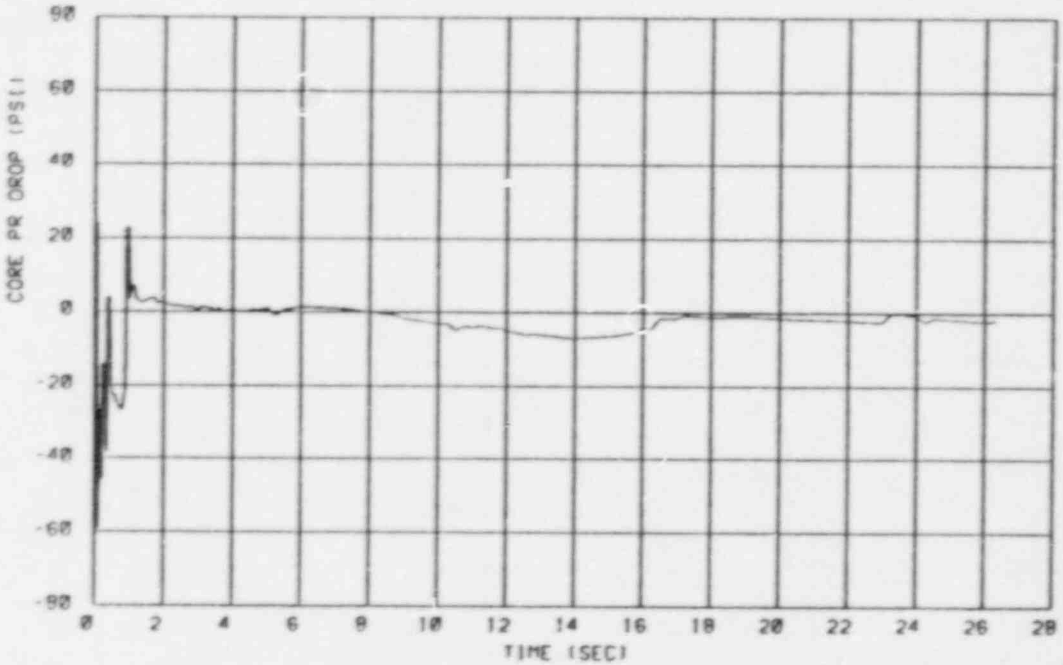


FIGURE 6A

CORE PRESSURE DROP VERSUS TIME

DECLG (CD = 0.4)

VRA SATAN NOM TAVG CASE CD 0.6 06/07/88 PCONT=37.25 18% SGTP
CD=0.6 DECLG 18 % SGTP FQ=2.19 FDH=1.55 T.S.NOM ACC LEVELS
CORE PR.DROP



06/07/88

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

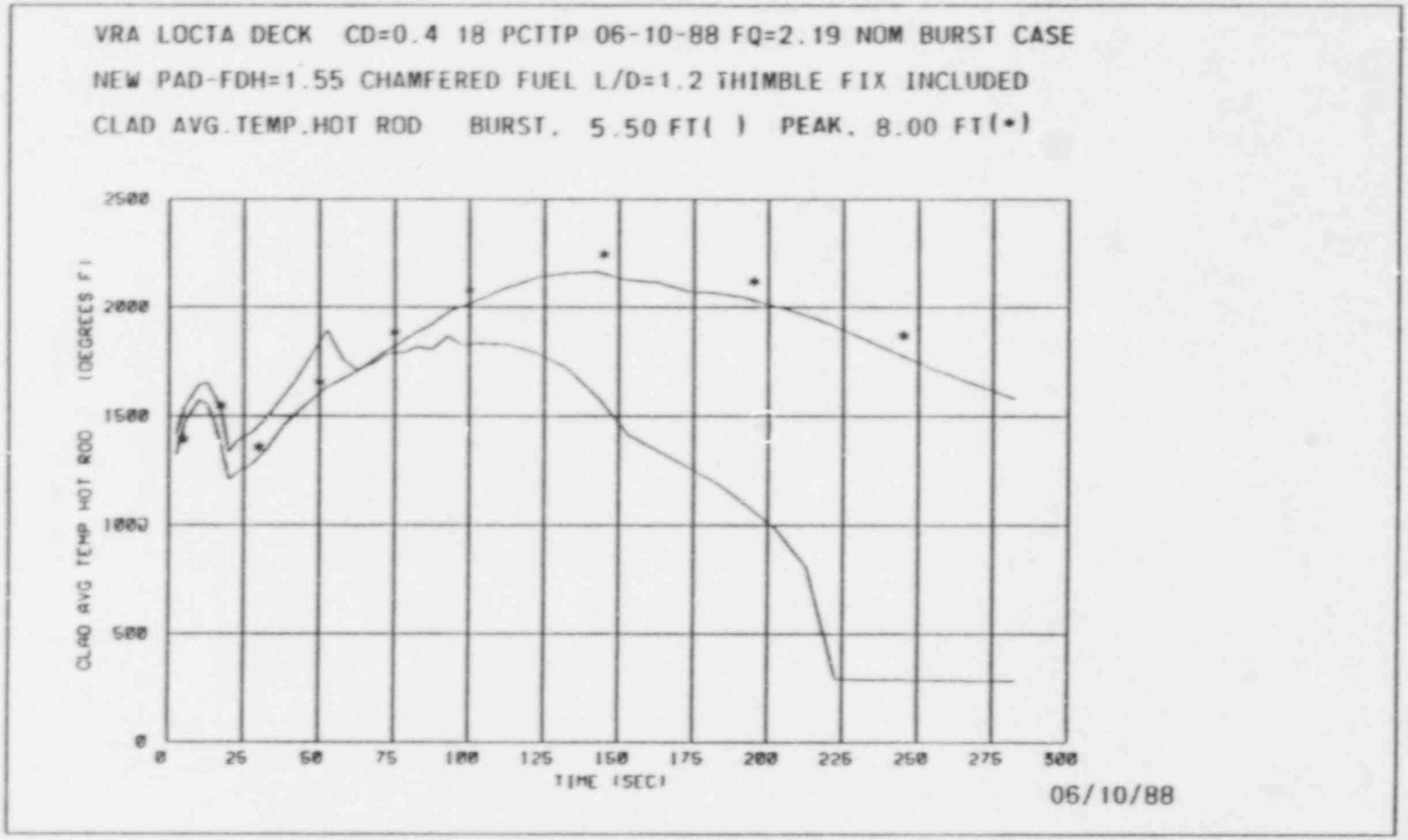


FIGURE 7A

PEAK CLAD TEMPERATURE TRANSIENT

DECLG (CD = 0.4)

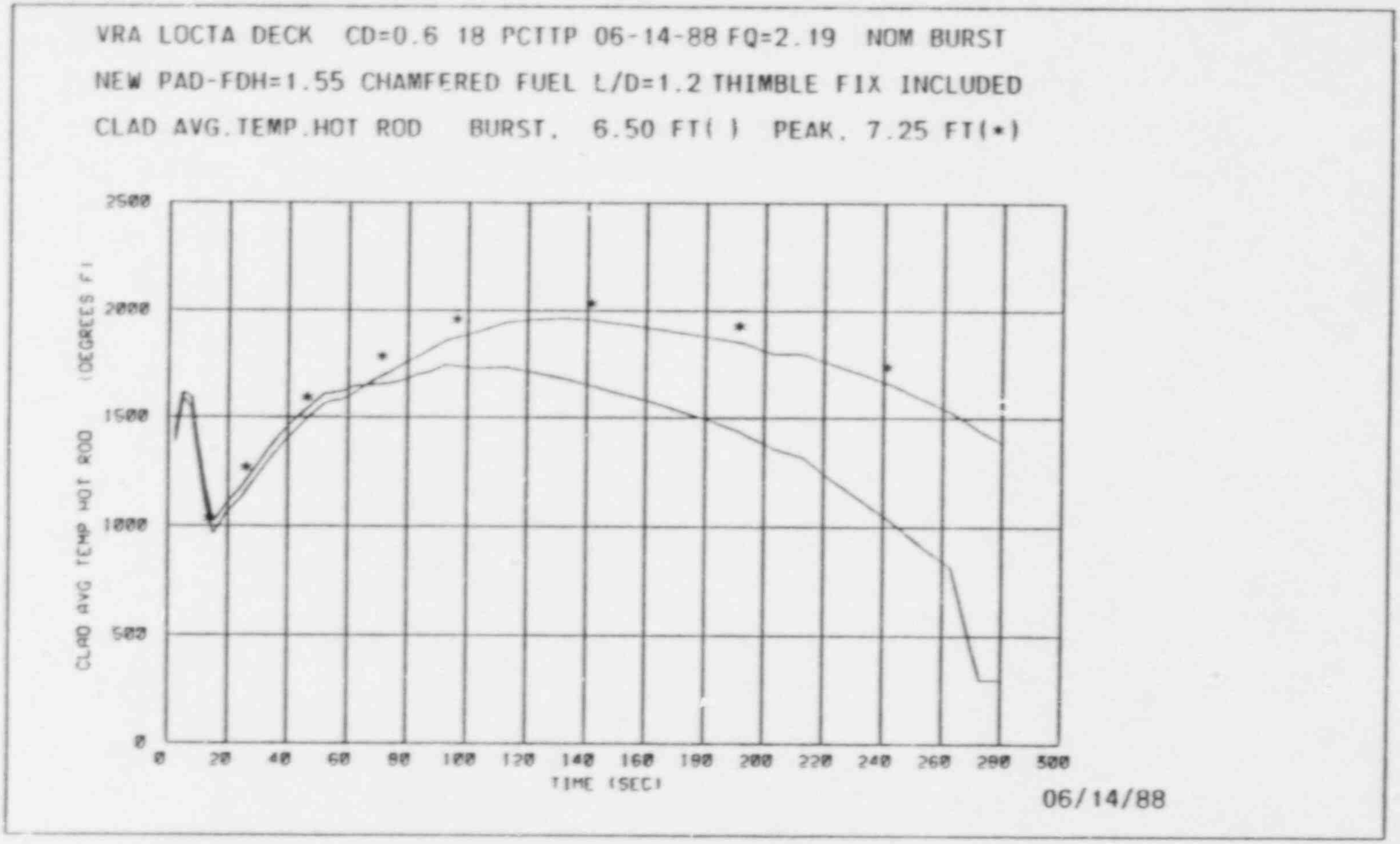


FIGURE 7B

PEAK CLAD TEMPERATURE TRANSIENT

DECLG (CD = 0.6)

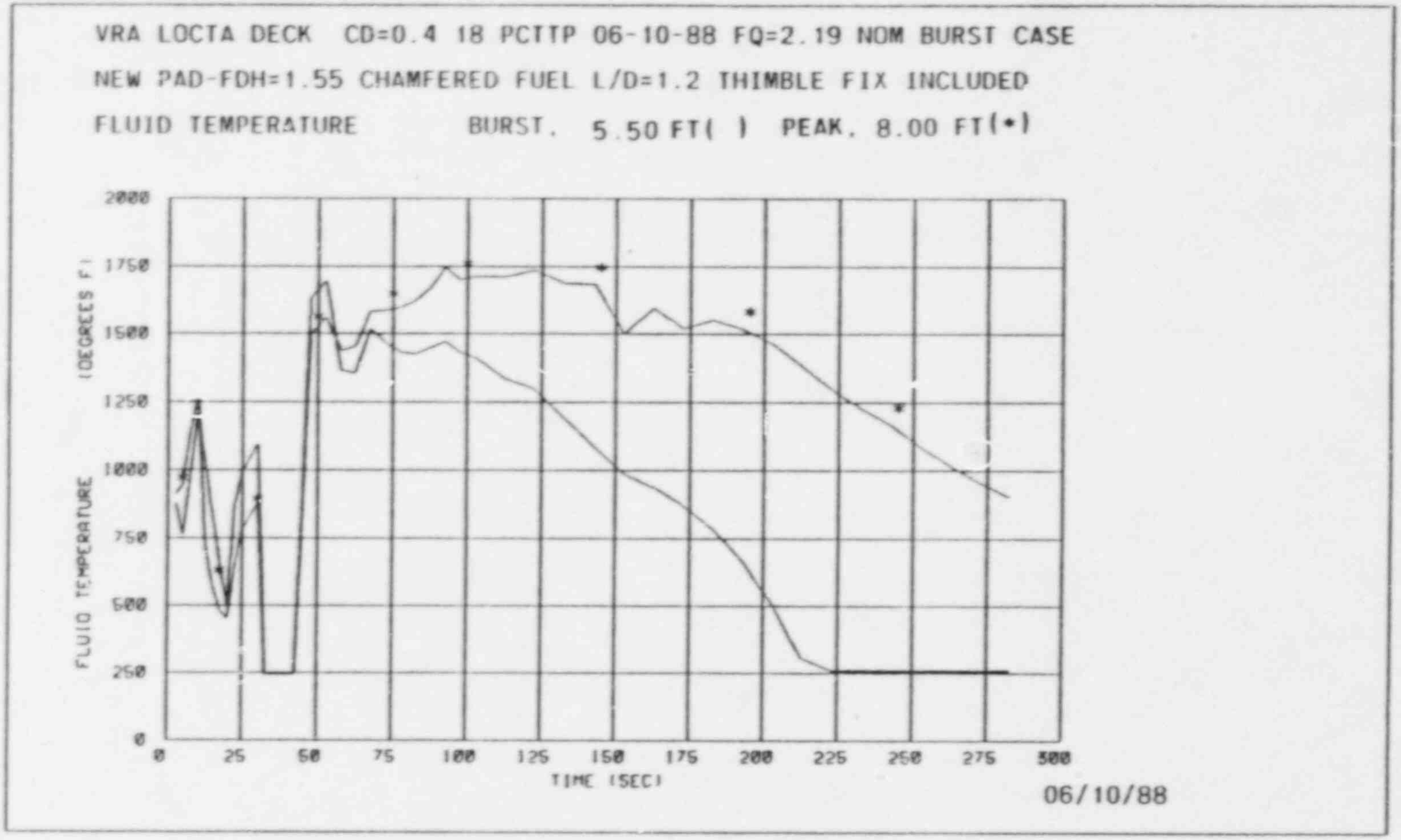


FIGURE 8A

FLUID TEMPERATURE VERSUS TIME

DECLG (CD = 0.4)

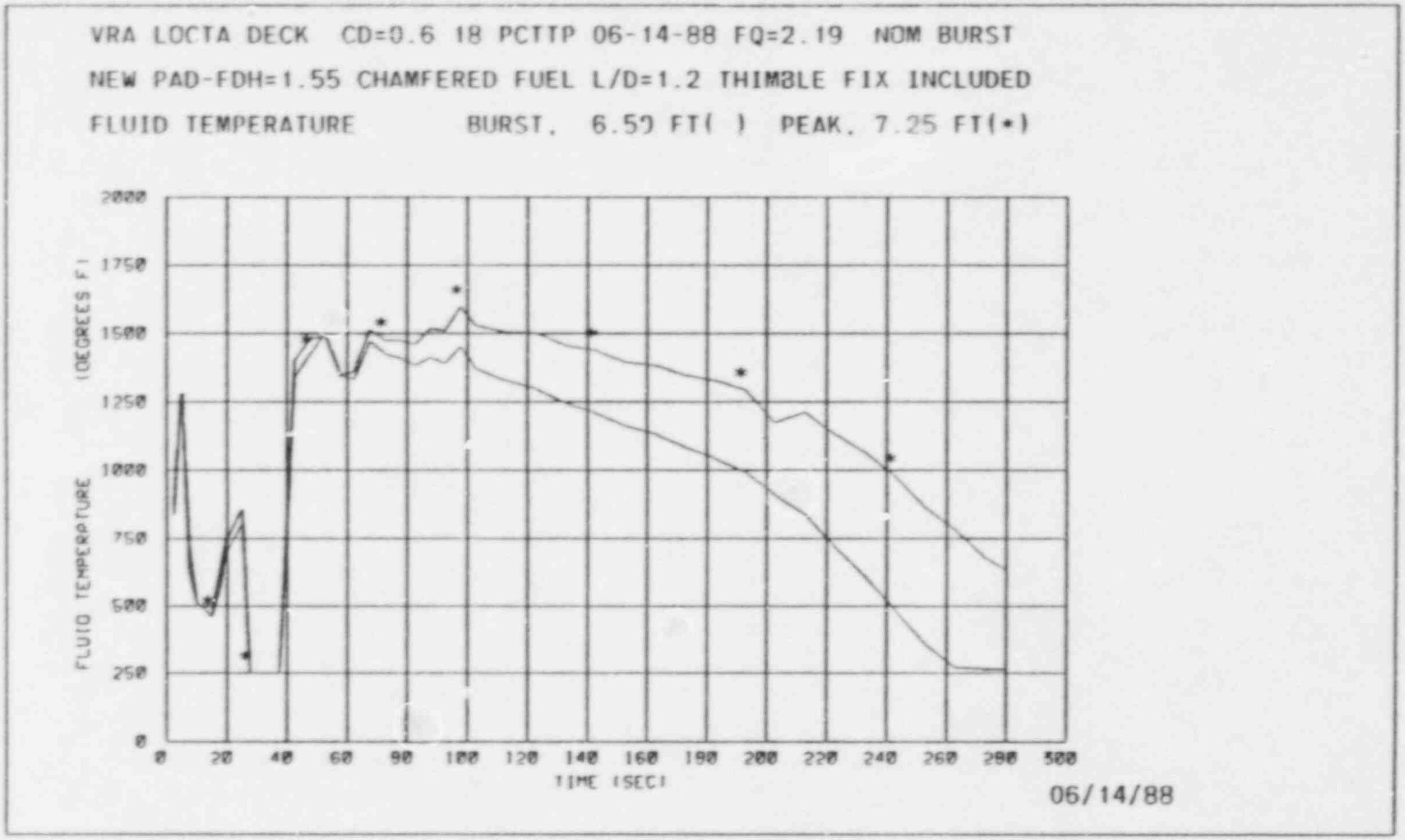


FIGURE 8B

FLUID TEMPERATURE VERSUS TIME

DECLG (CD = 0.6)

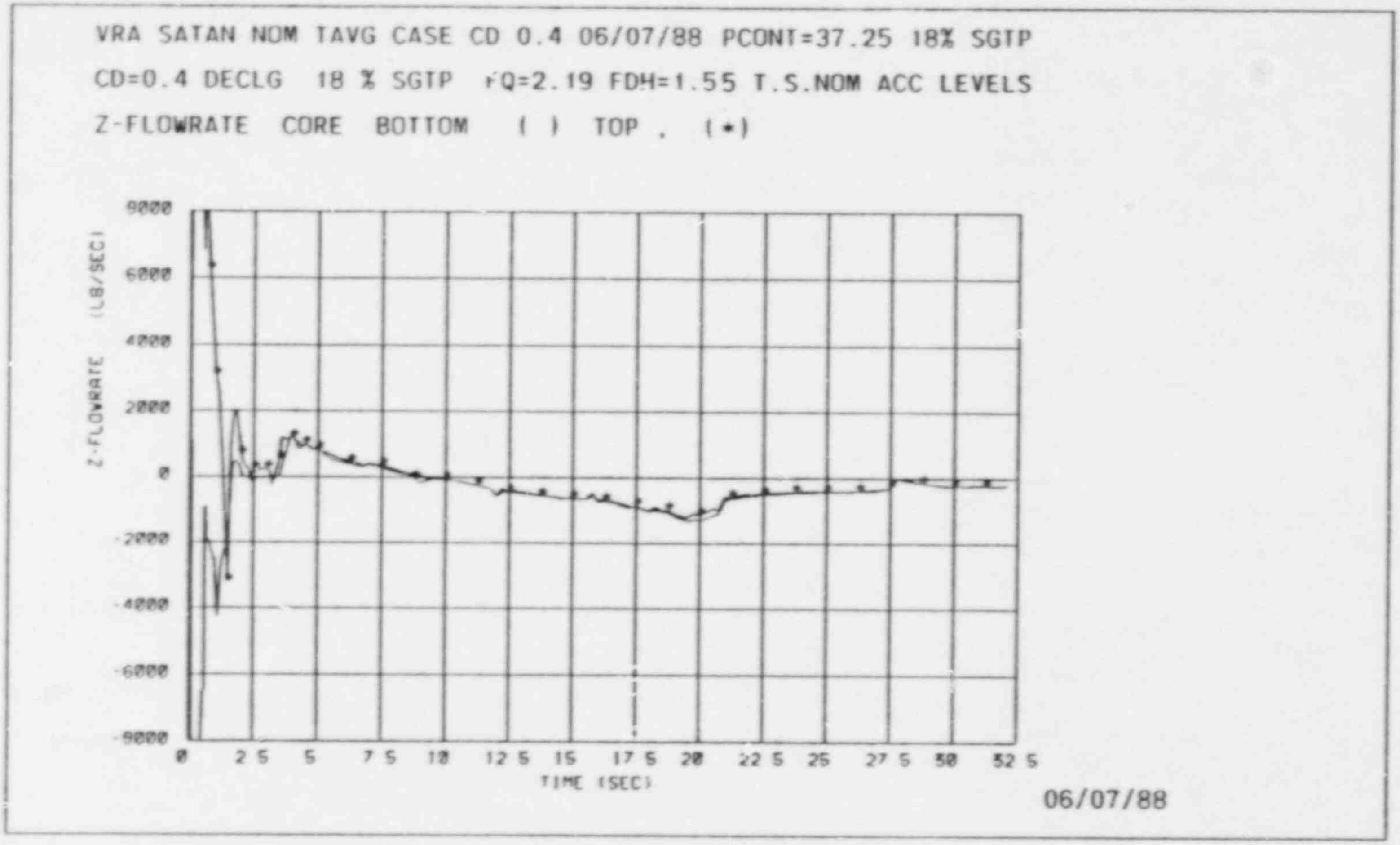


FIGURE 9A

CORE FLOW VERSUS TIME (TOP AND BOTTOM)

DECLG (CD = 0.4)

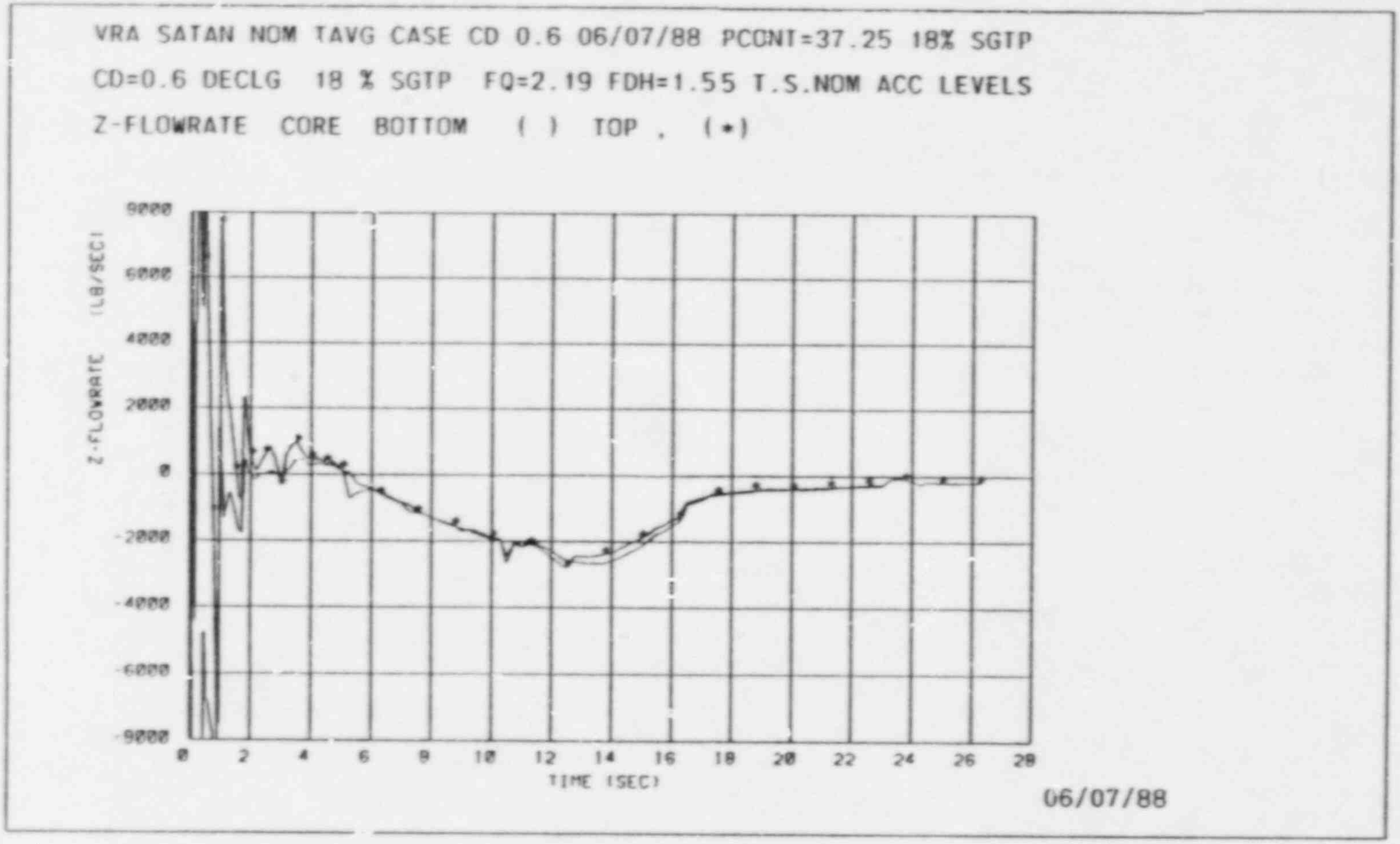


FIGURE 9B

CORE FLOW VERSUS TIME (TOP AND BOTTOM)

DECLG (CD = 0.6)

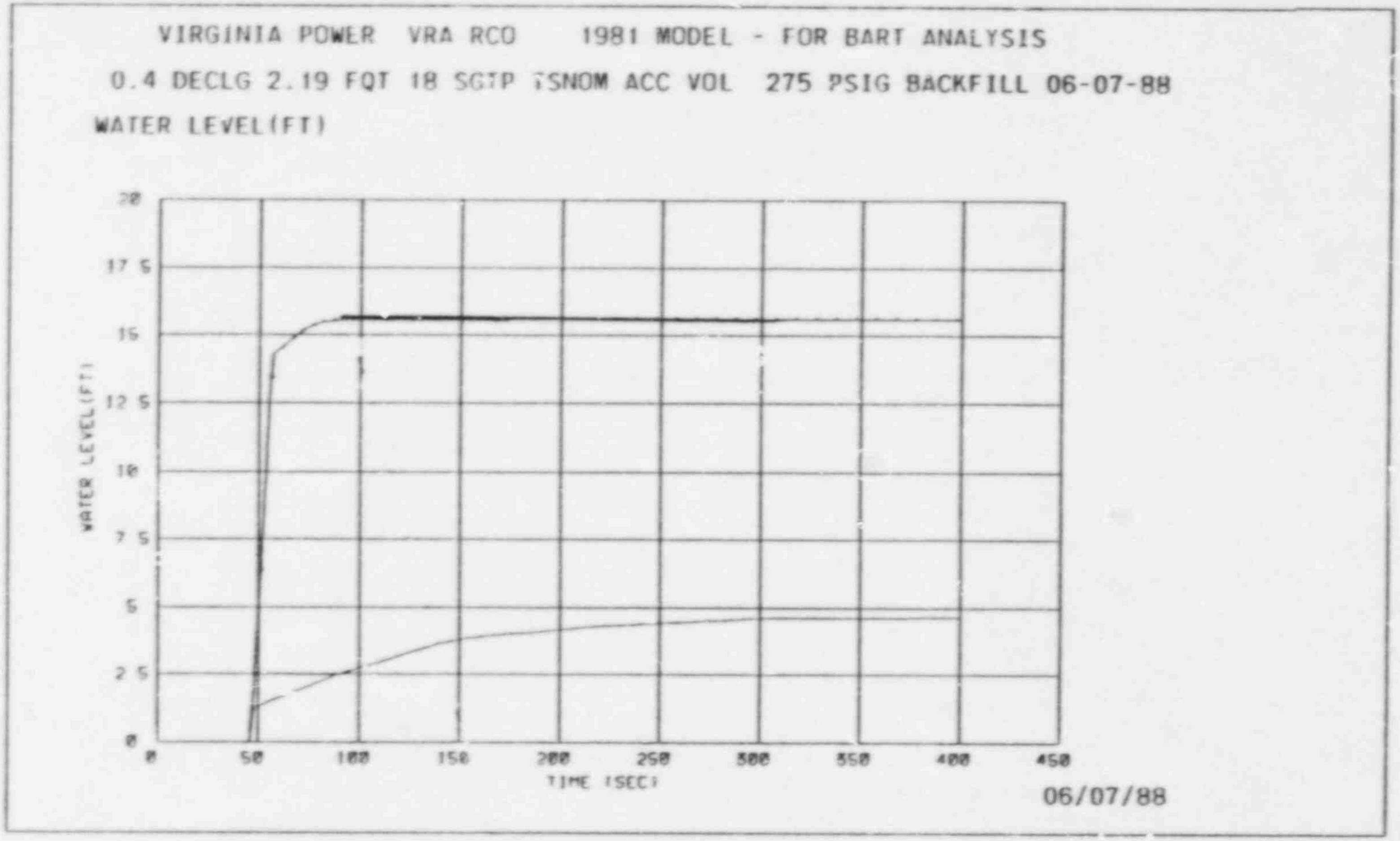


FIGURE 10A

REFLOOD TRANSIENT - CORE AND DOWNCOMER WATER LEVELS
DECLG (CD = 0.4)

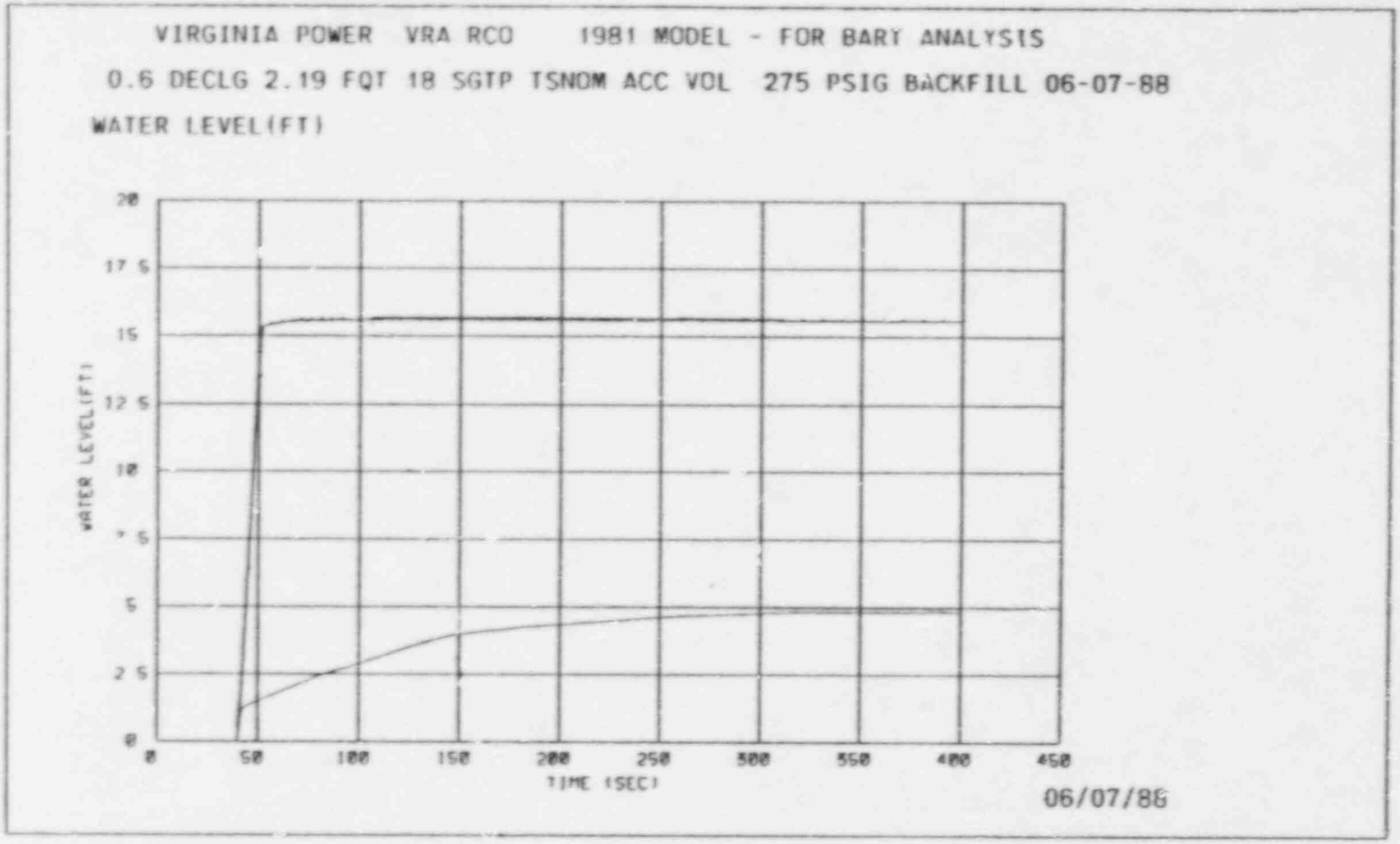


FIGURE 10B

REFLOOD TRANSIENT - CORE AND DOWNCOMER WATER LEVELS

DECLG (CD = 0.6)

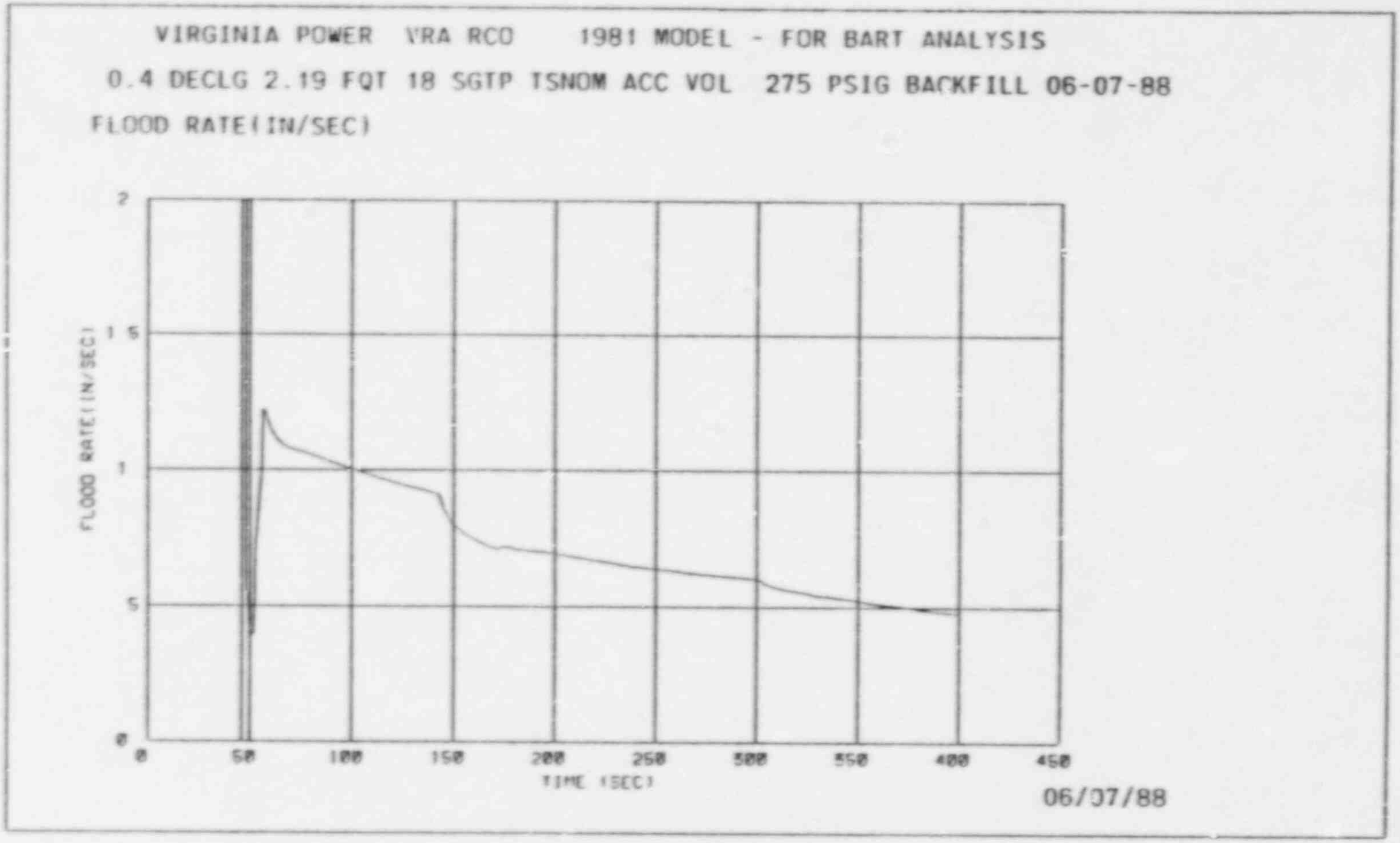


FIGURE 11A

REFLOOD TRANSIENT - CORE INLET VELOCITY

DECLG (CD = 0.4)

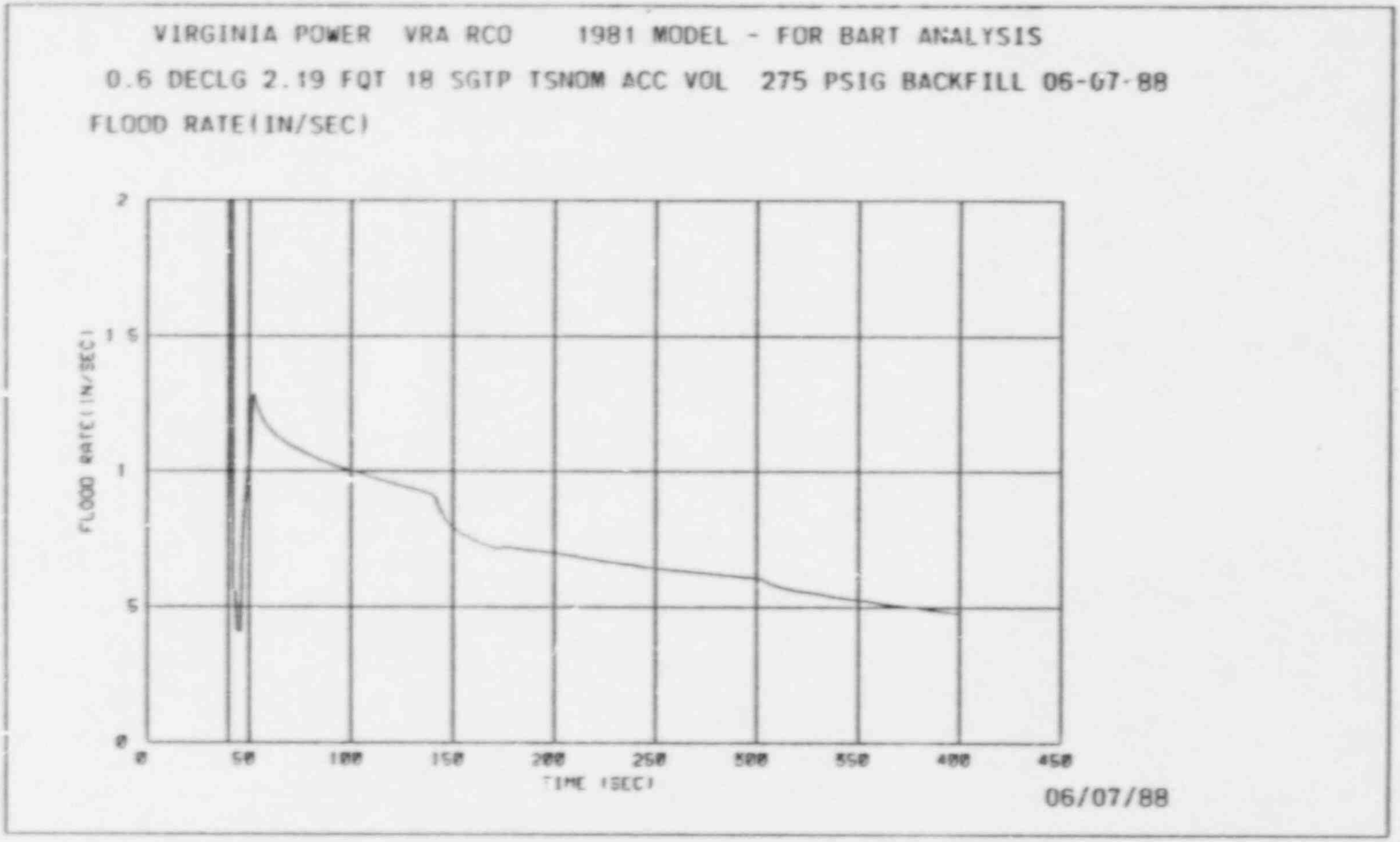


FIGURE 11B

REFLOOD TRANSIENT - CORE INLET VELOCITY

DECLG (CD = 0.6)

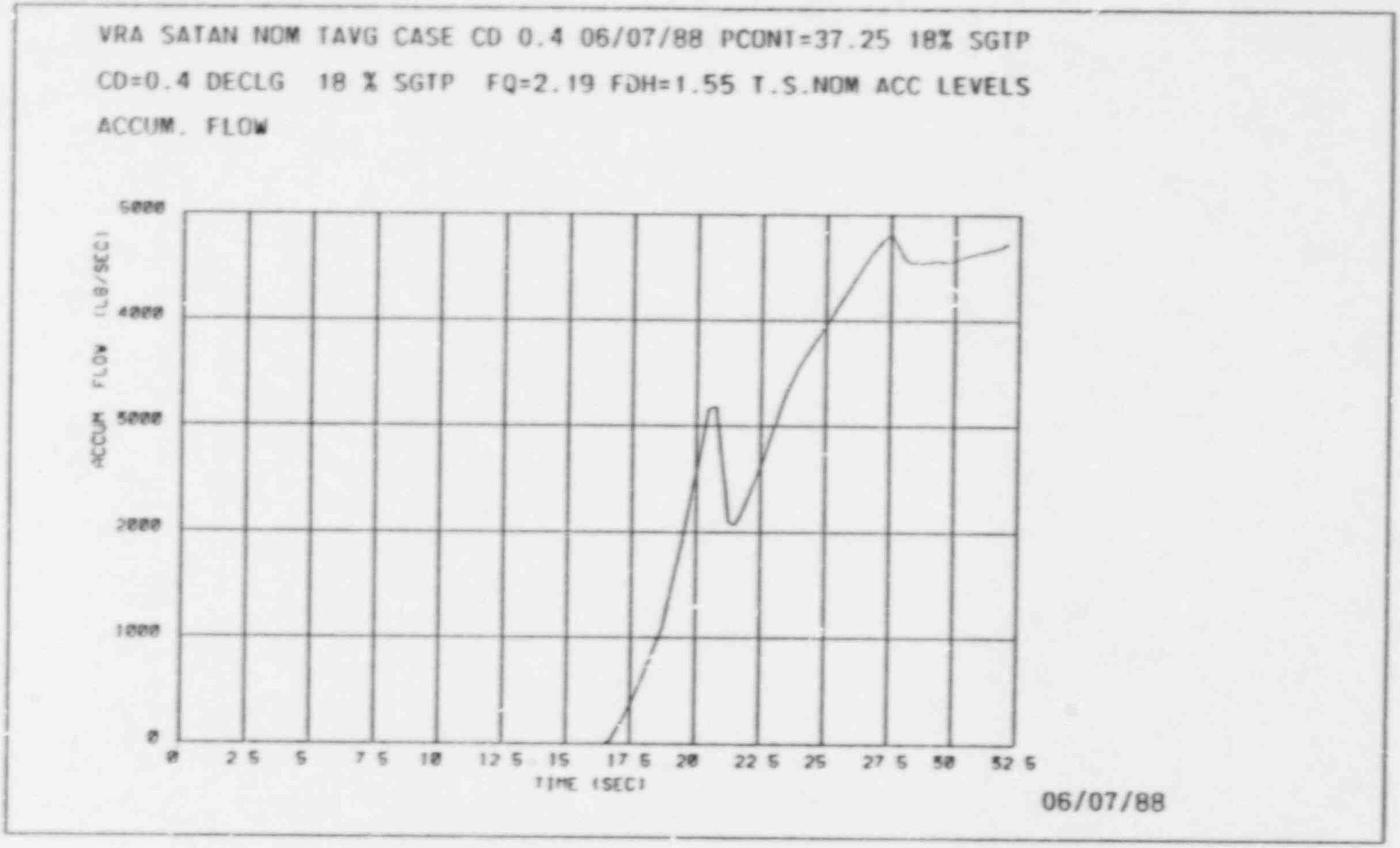


FIGURE 12A

ACCUMULATOR FLOW VERSUS TIME (BLOWDOWN)

DECLG (CD = 0.4)

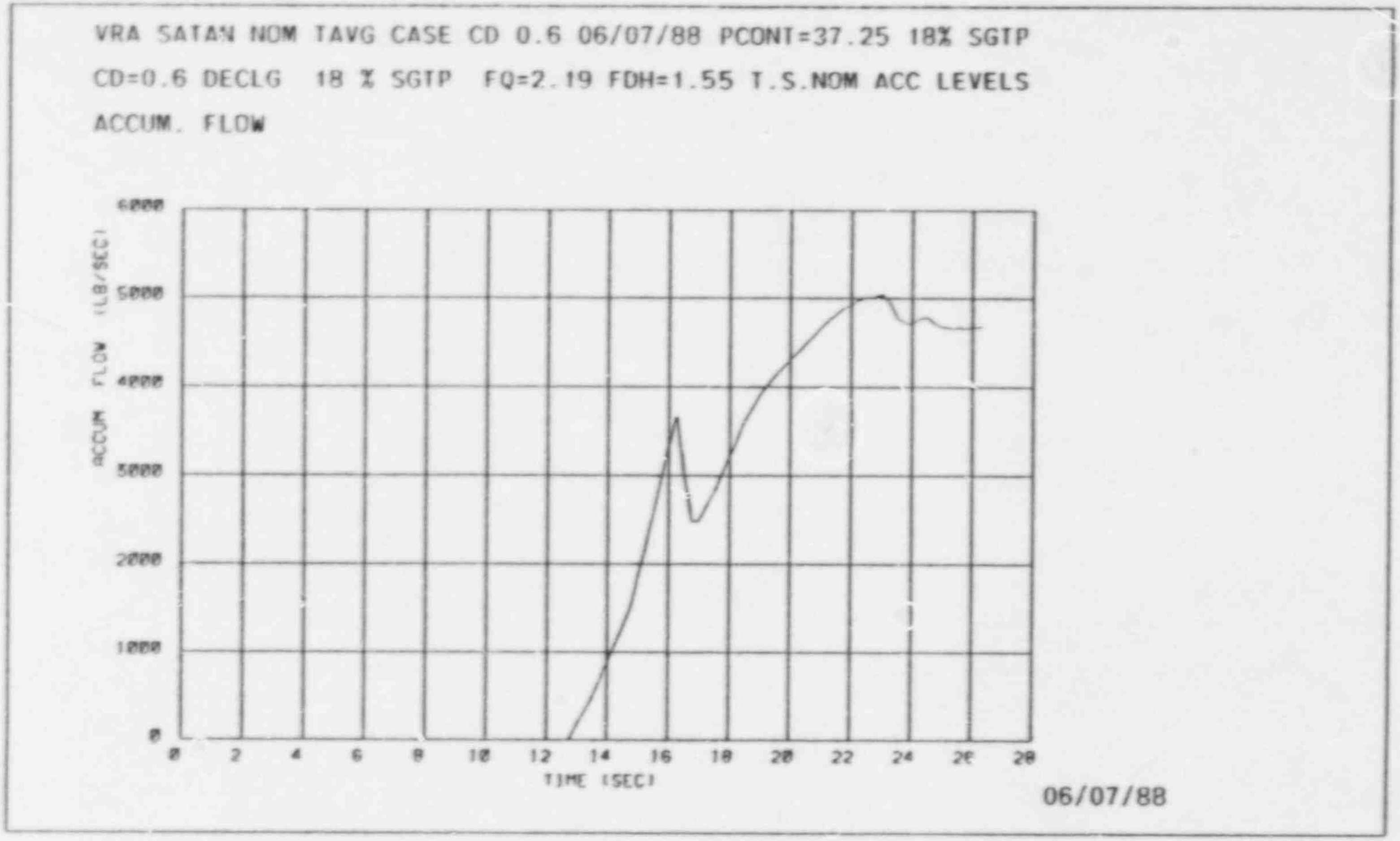


FIGURE 12B

ACCUMULATOR FLOW VERSUS TIME (BLOWDOWN)

DECLG (CD = 0.6)

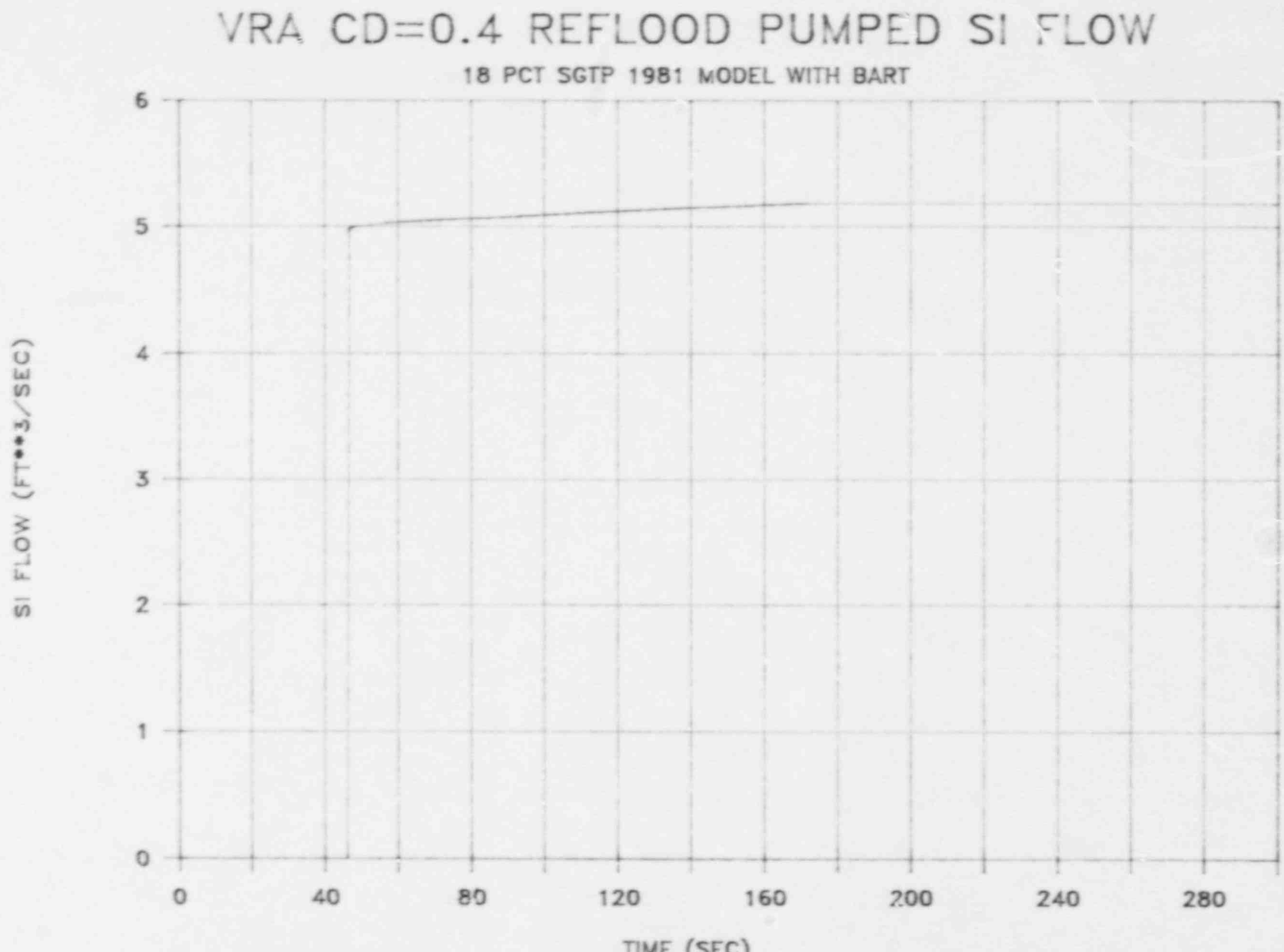


FIGURE 13A
PUMPED ECCS FLOW VERSUS TIME (REFLOOD)

VRA CD=0.6 REFLOOD PUMPED SI FLOW

18 PCT SGTP 1981 MODEL WITH BART

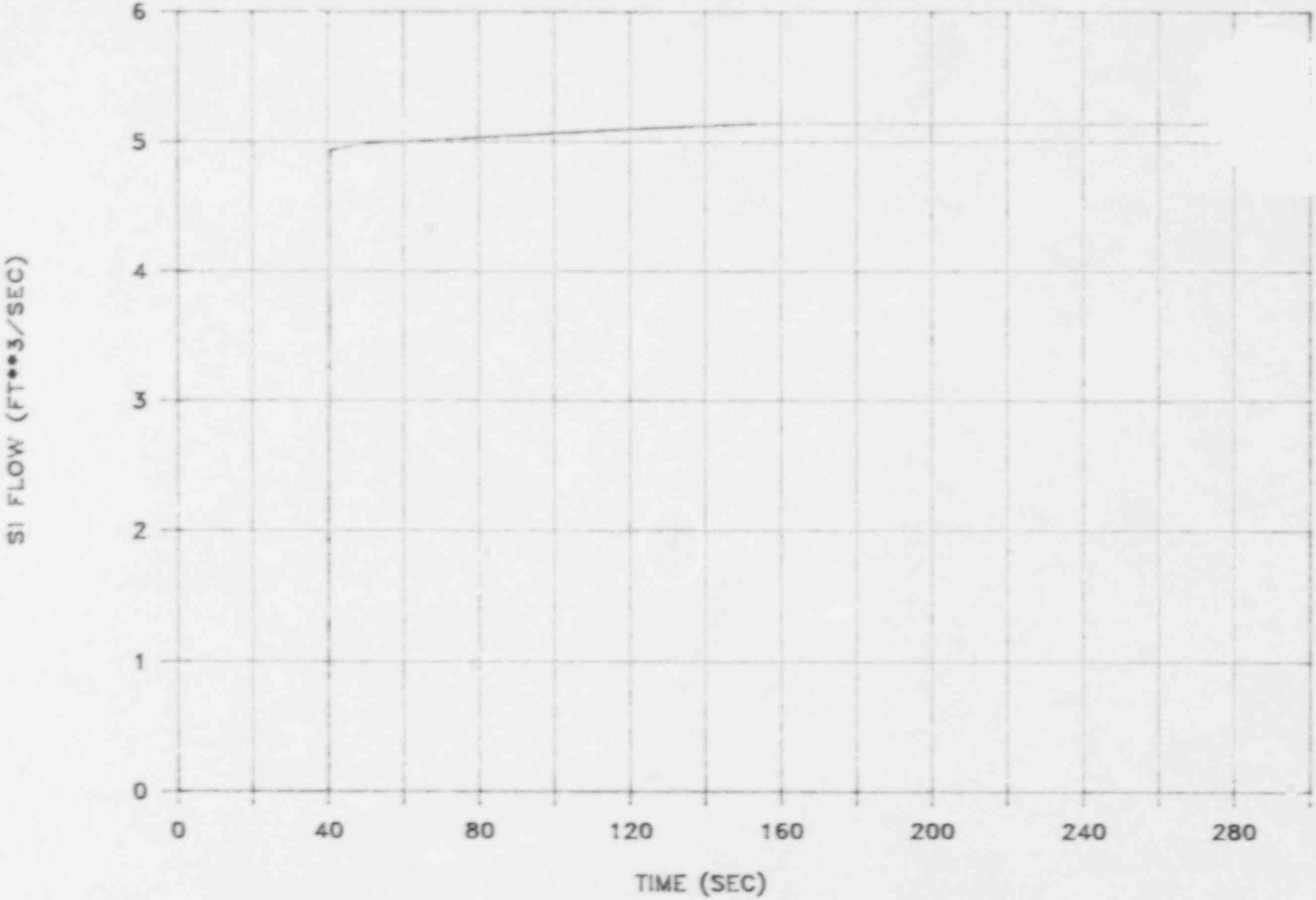


FIGURE 13B
PUMPED LCCS FLOW VERSUS TIME (REFLOOD)

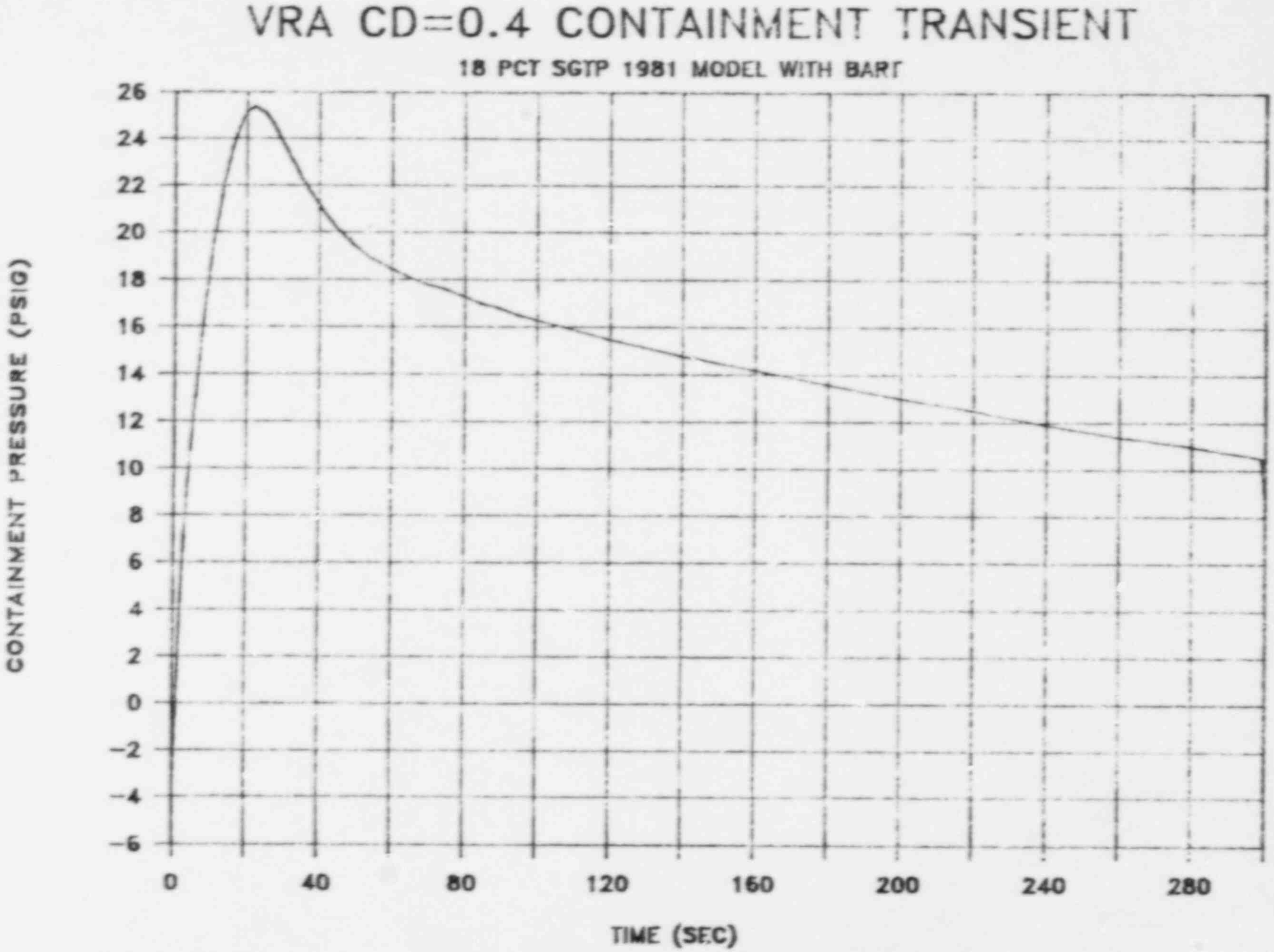


FIGURE 14A

CONTAINMENT PRESSURE TRANSIENT

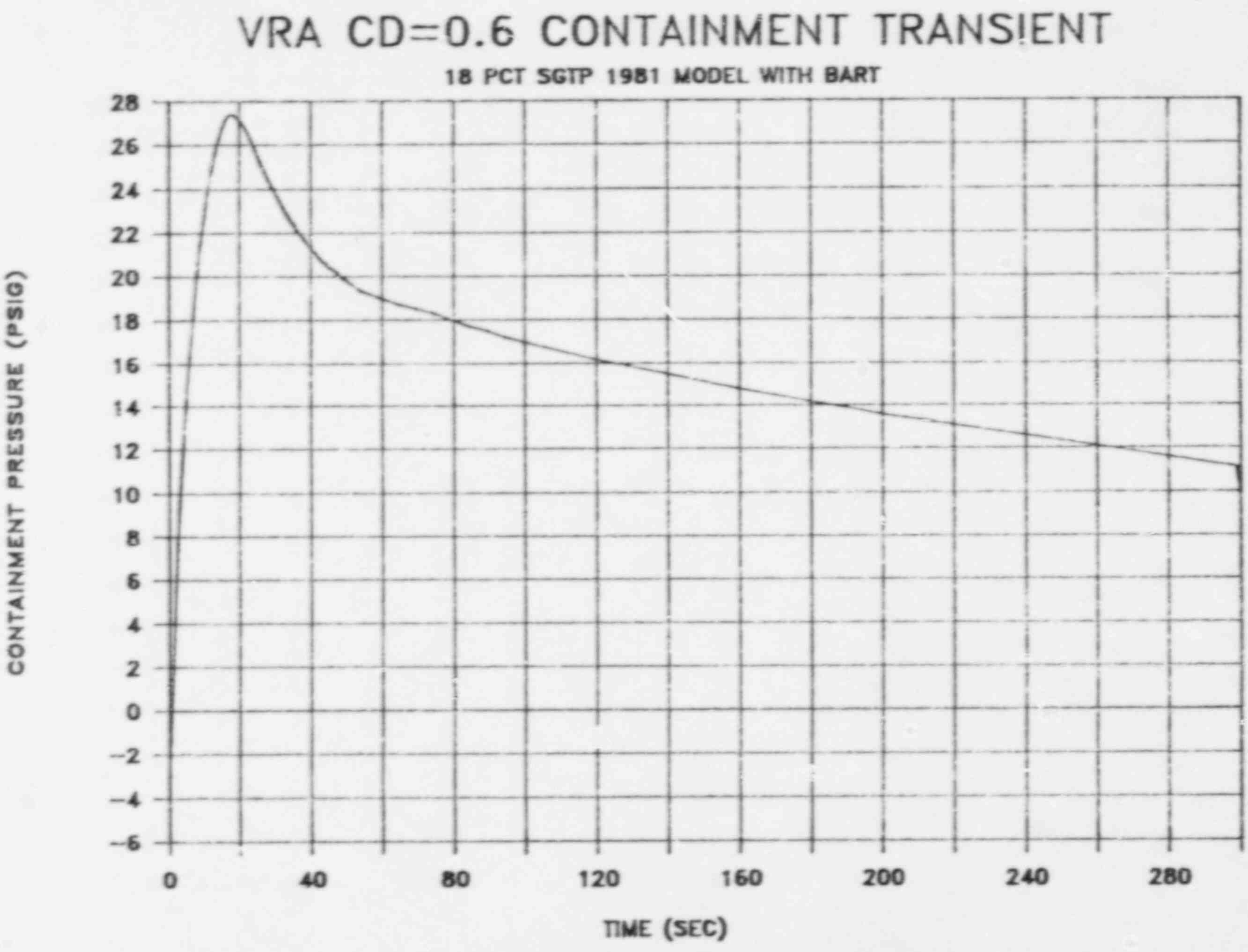


FIGURE 14B
CONTAINMENT PRESSURE TRANSIENT

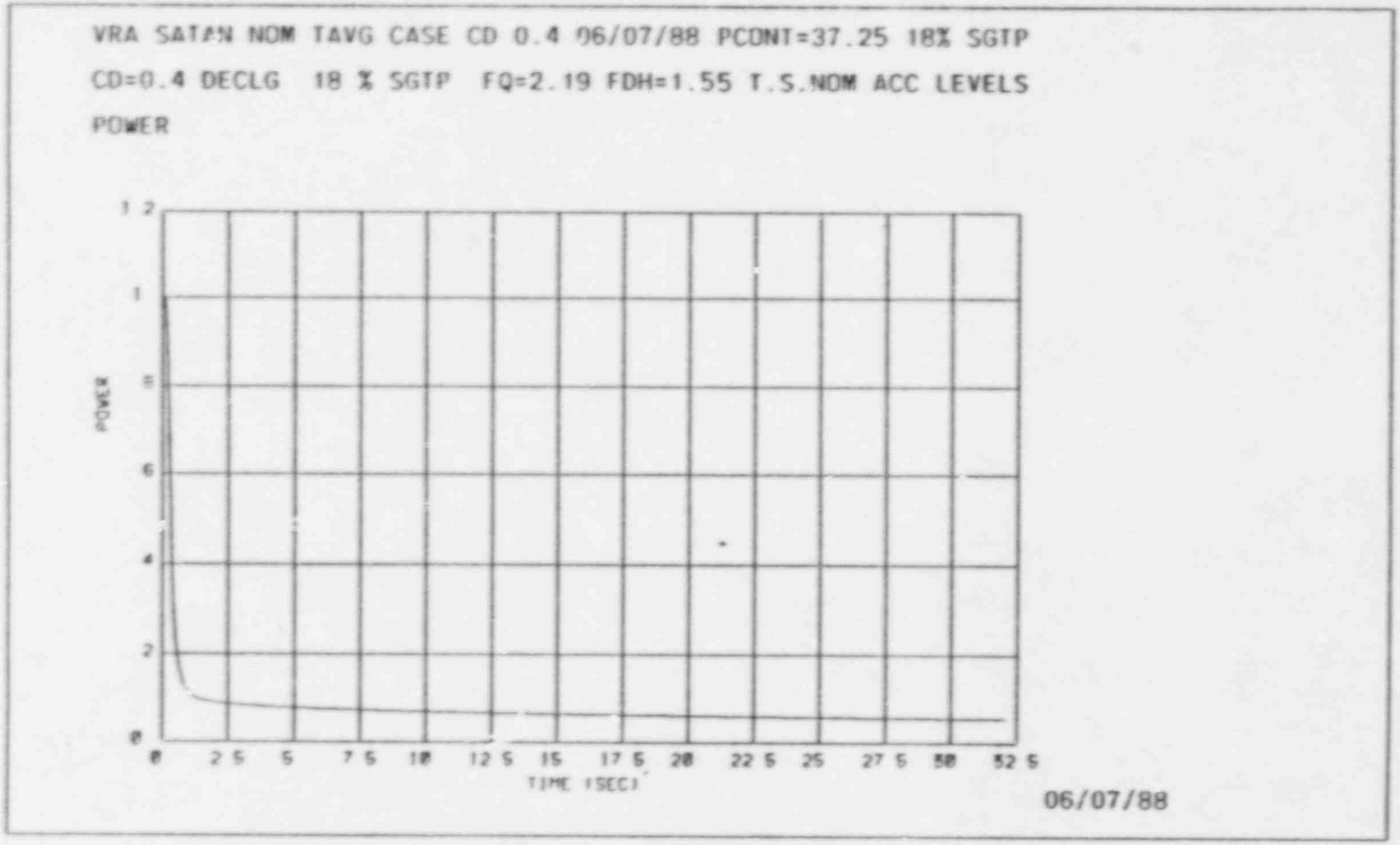


FIGURE 15A

CORE POWER TRANSIENT

DECLG (CD = 0.4)

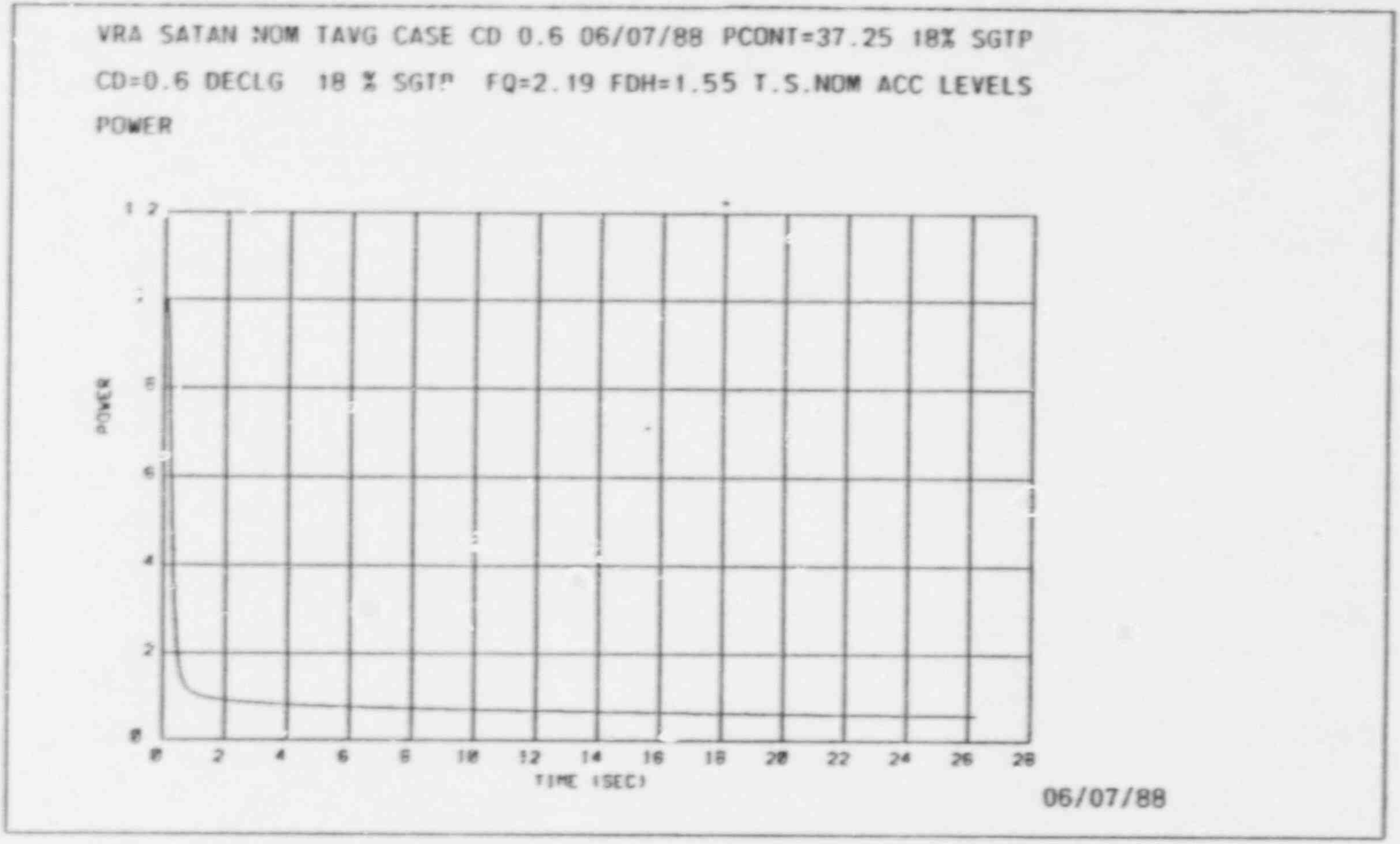


FIGURE 15B

CORE POWER TRANSIENT

DECLG (CD = 0.6)

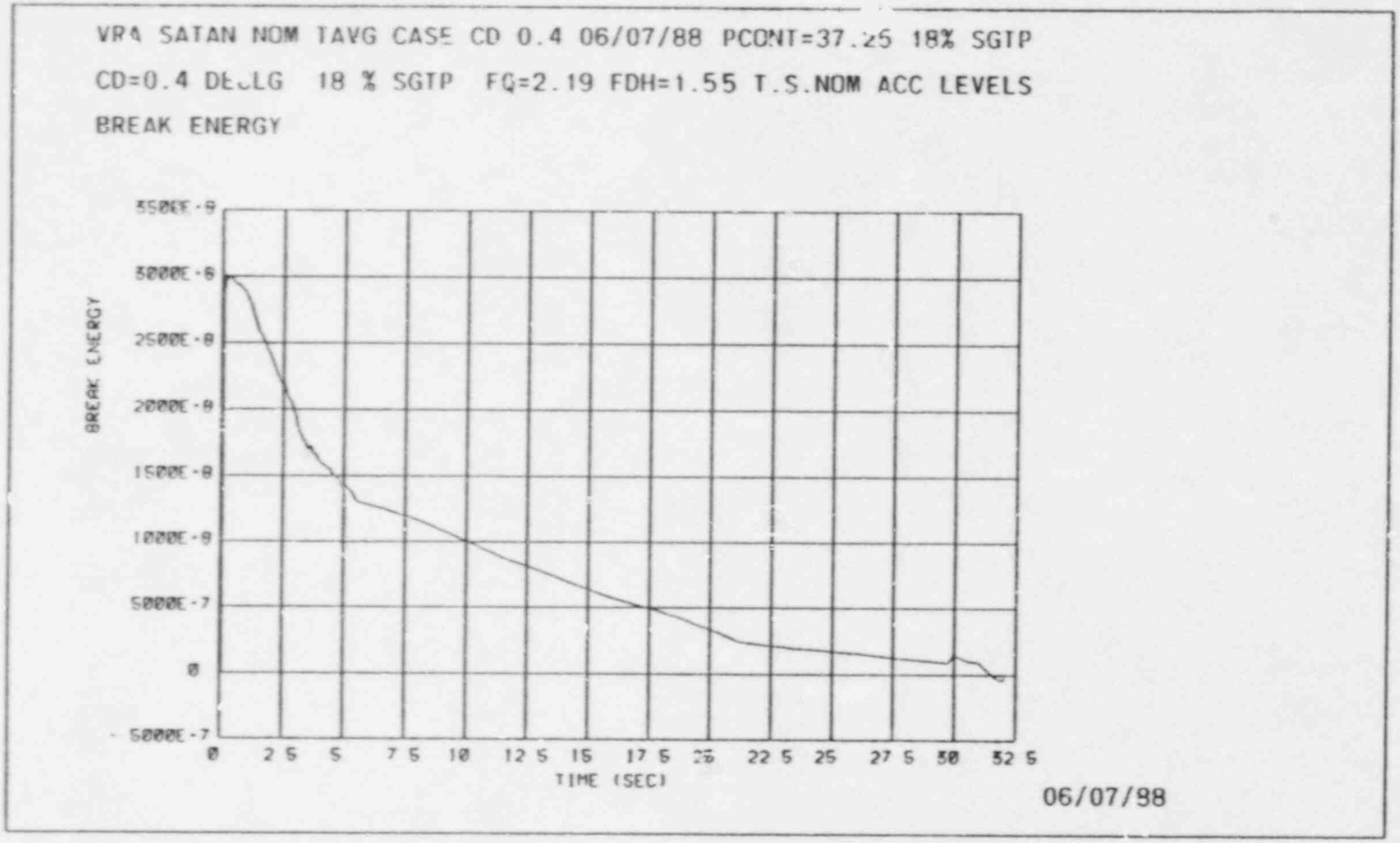


FIGURE 16A

BREAK ENERGY RELEASED TO CONTAINMENT

DECLG (CD = 0.4)

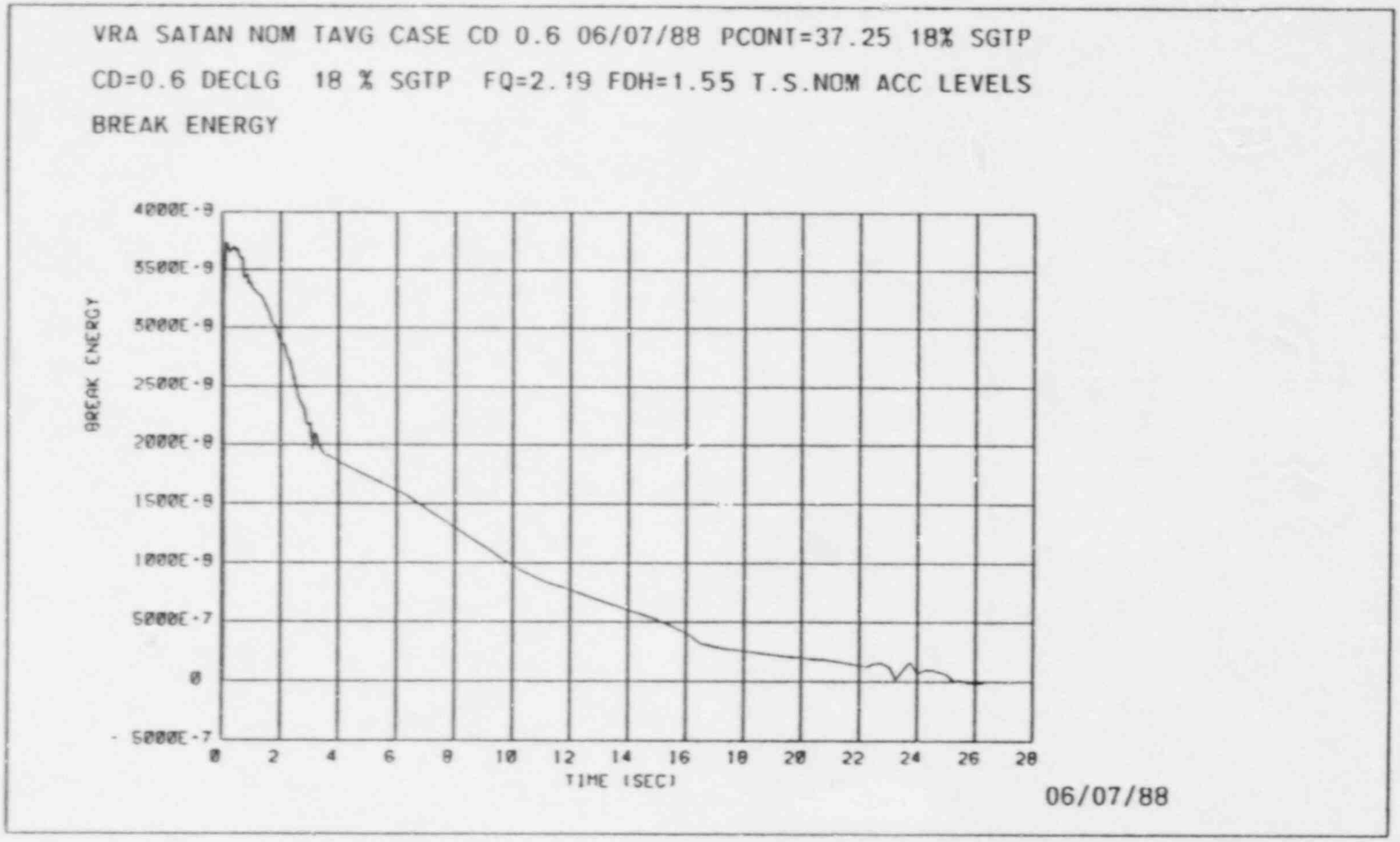


FIGURE 16B

BREAK ENERGY RELEASED TO CONTAINMENT

DECLG (CD = 0.6)

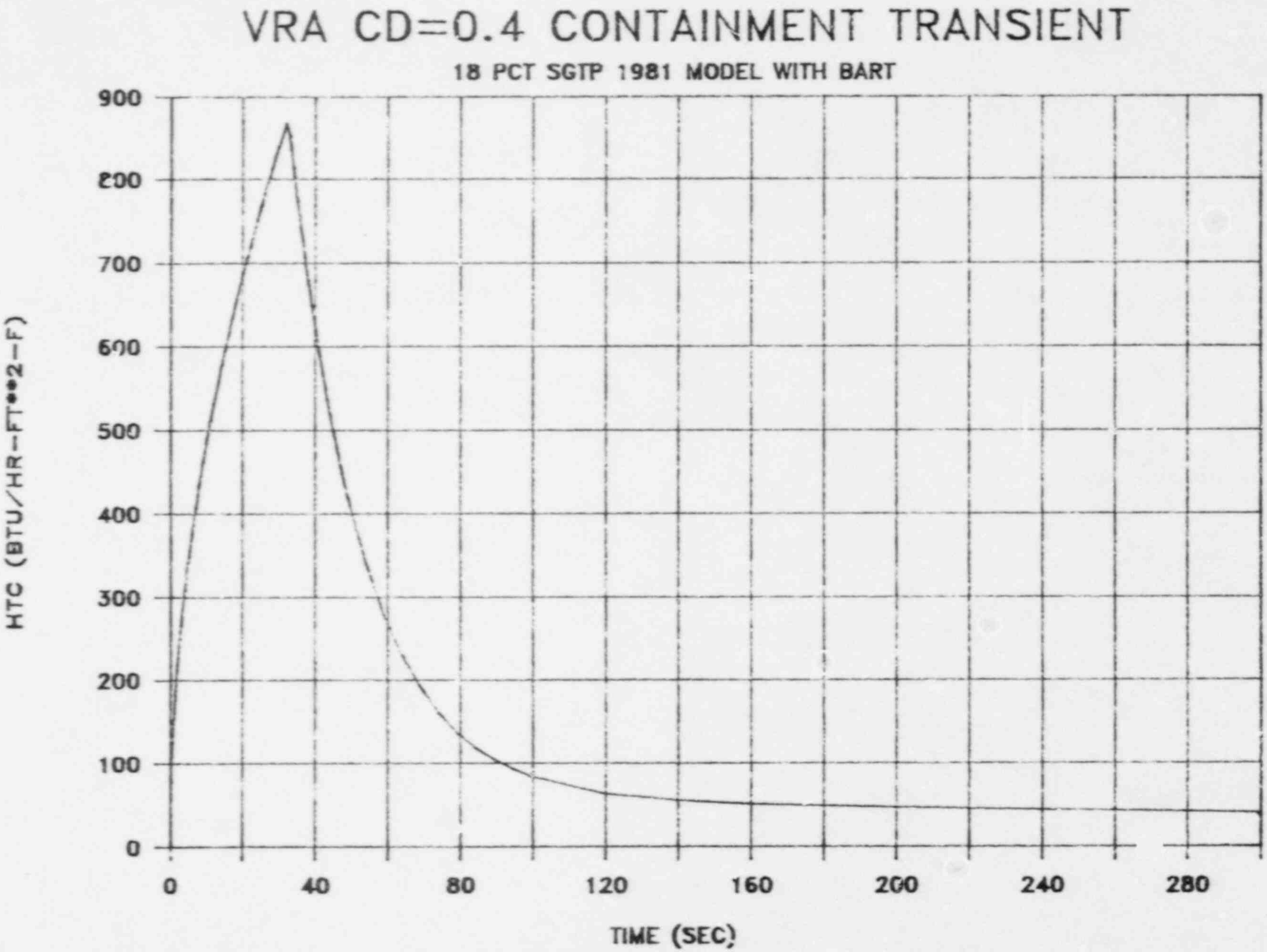


FIGURE 17A
CONTAINMENT WALL HEAT TRANSFER COEFFICIENT

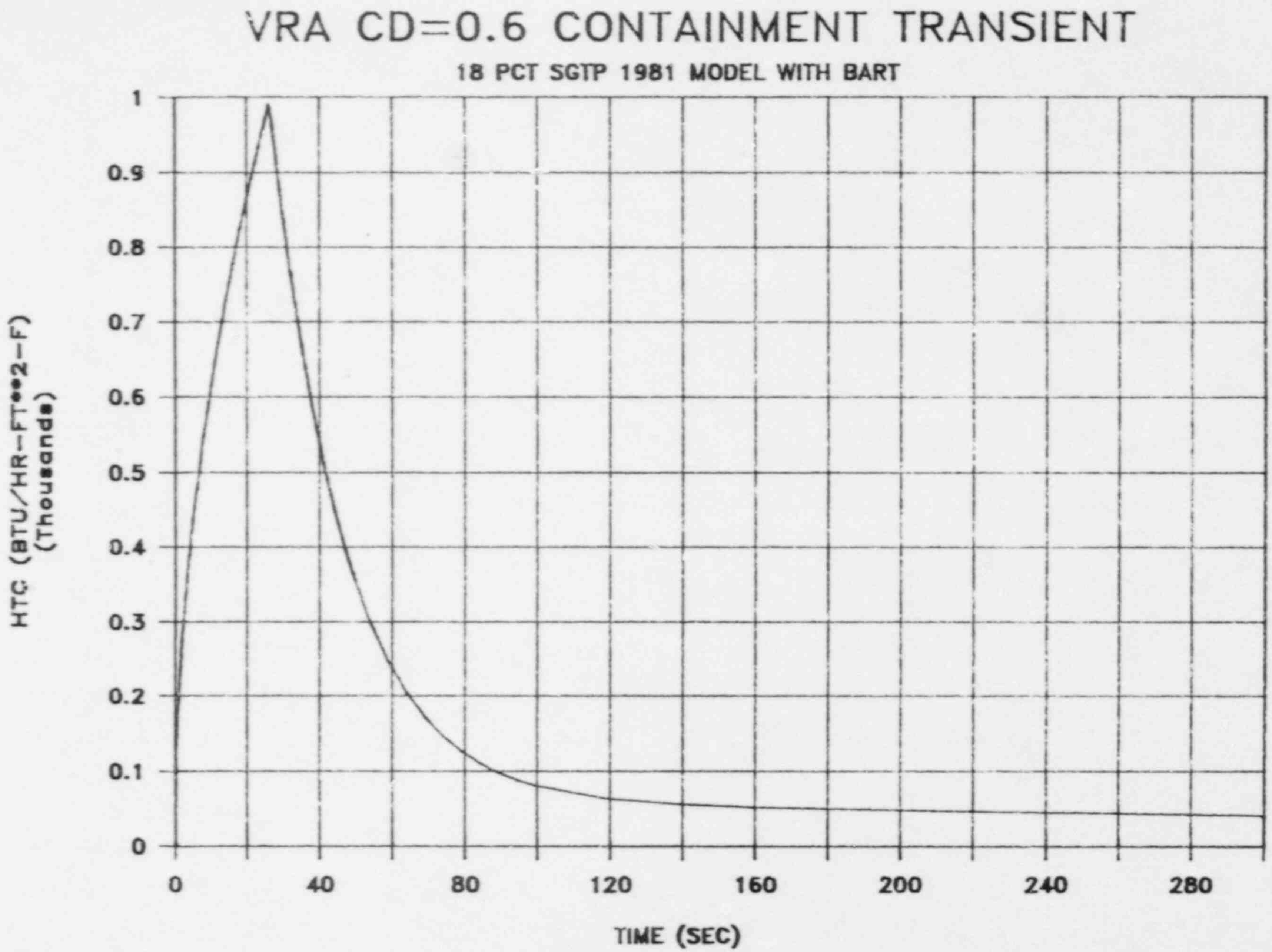


FIGURE 17B
CONTAINMENT WALL HEAT TRANSFER COEFFICIENT

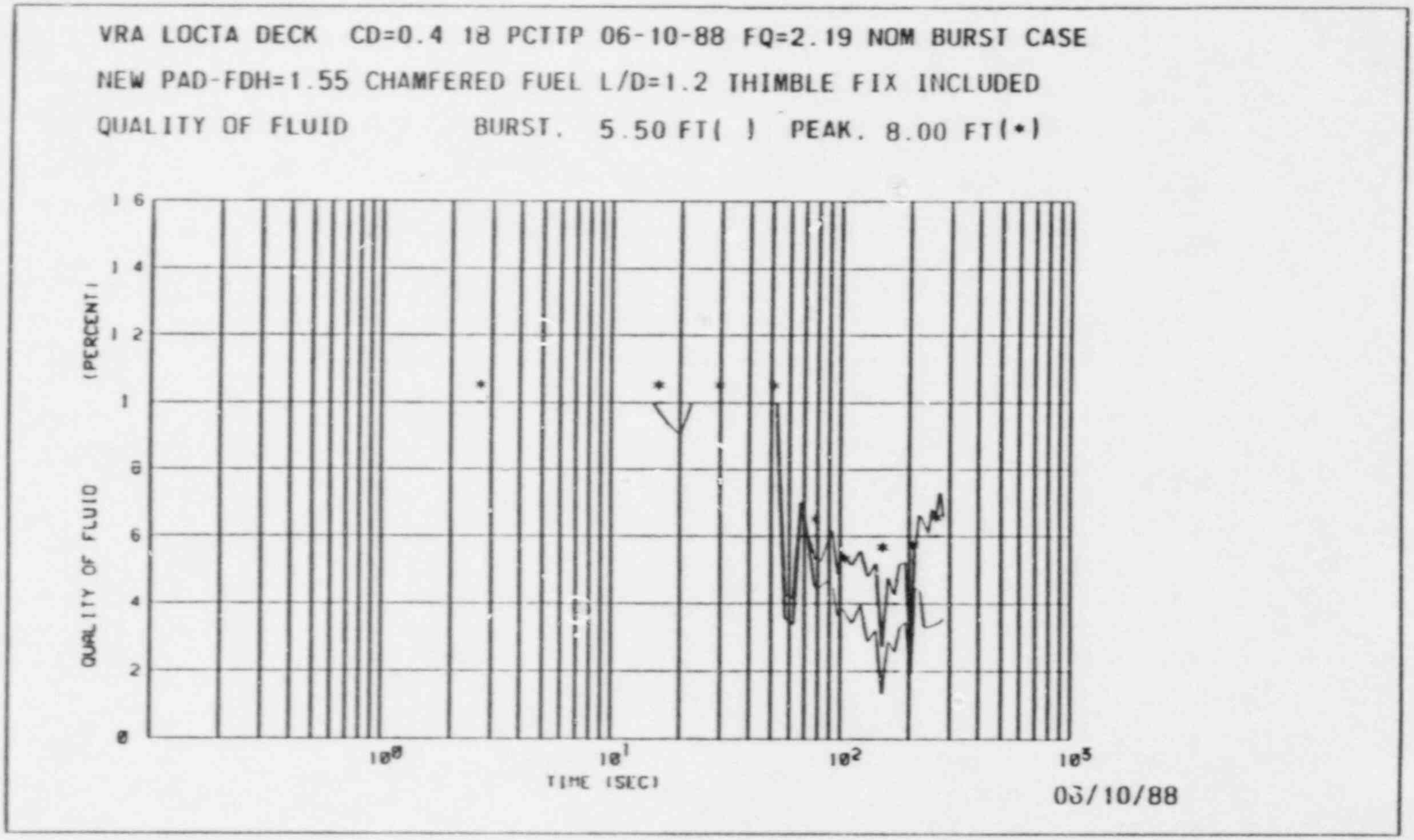


FIGURE 18A
FLUID QUALITY VERSUS TIME
DECLG (CD = 0.4)

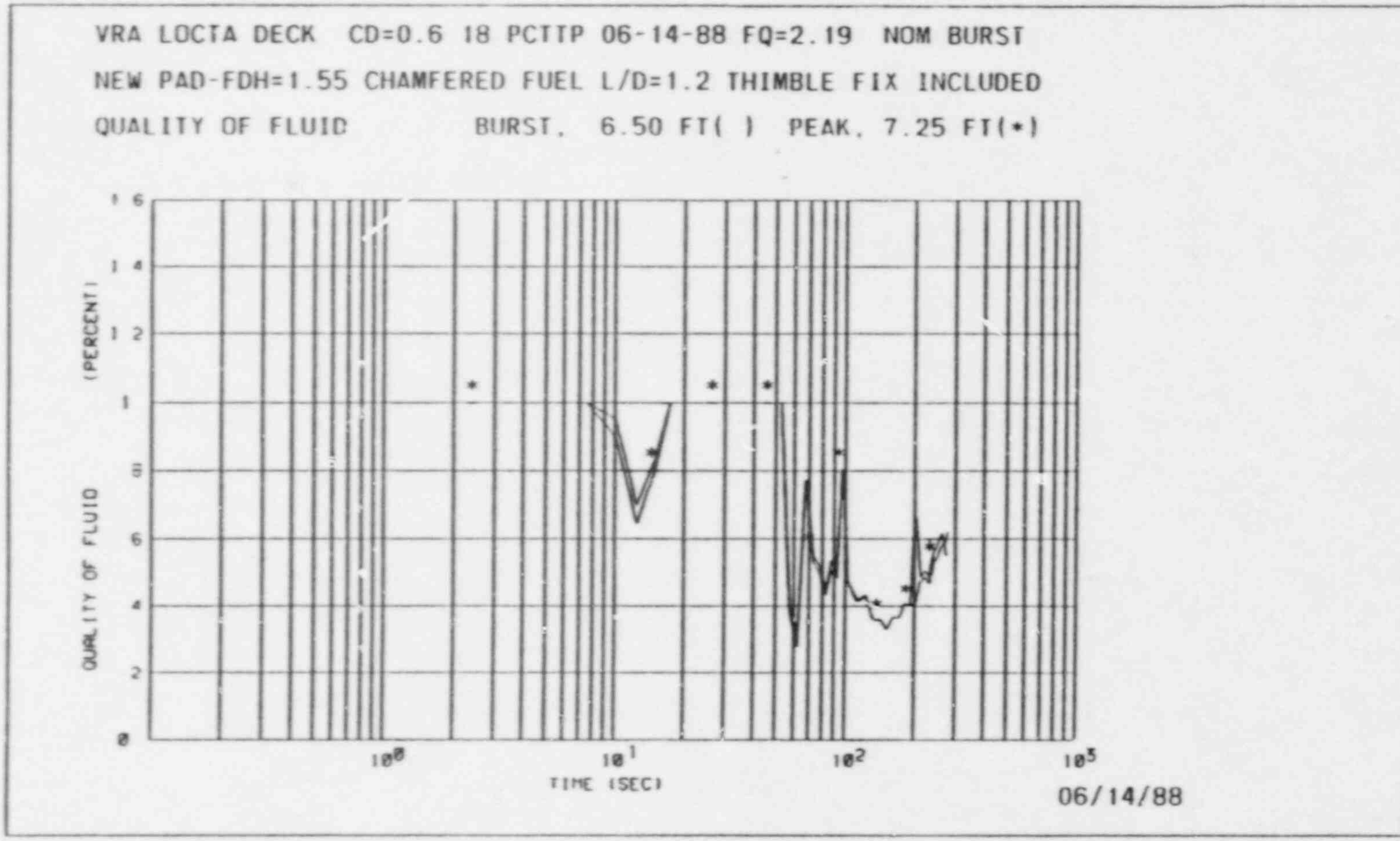


FIGURE 18B

FLUID QUALITY VERSUS TIME

DECLG (CD = 0.6)

ATTACHMENT 3

10 CFR 50.92 SIGNIFICANT HAZARDS EVALUATION

FOR

INCREASED FQ(Z) WITH 18% STEAM GENERATOR TUBE PLUGGING

NORTH ANNA UNITS 1 AND 2

BASIS FOR NO SIGNIFICANT HAZARDS DETERMINATION

The proposed changes do not involve a significant hazards consideration because operation of North Anna Units 1 and 2 in accordance with these change would not:

1. involve a significant increase in the probability or consequence of an accident previously evaluated. The revised LOCA analysis which supports these changes demonstrated that the ECCS acceptance criteria of 10 CFR 50.46 were met. Evaluation of the non-LOCA accidents has shown that the acceptance criteria for these accidents are also met with no increase in accident consequences.

2. create the possibility of a new or different kind of accident from any accident previously identified. The proposed changes involve changes in assumptions made for previously evaluated LOCA accidents. The revised analysis included these parameter changes and demonstrated that they would not cause a new accident. In addition, the increase in steam generator tube plugging was evaluated for impact upon RCS flow and RCS coolant volume. It has been demonstrated that the non-LOCA accidents for which these parameters are significant meet applicable acceptance criteria when considering the proposed changes. Thus, the proposed changes will not create the possibility of a new or different kind of accident.

3. involve a significant reduction in a margin of safety. The revised ECCS analysis meets the requirements of 10 CFR 50.46. Additionally, the non-LOCA accidents affected by the proposed changes meet their acceptance criteria. The current margin of safety as established by meeting regulatory requirements (e.g., 10 CFR 50.46) is therefore maintained for LOCA and non-LOCA accidents.