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Mr. Carl Berlinger
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U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: James Watt

Subject: McGuire Plant BART Model ECCS Analysis with UHI Removed

Duke Power is pursuing Technical Specification changes for the McGuire Units to permit operation with the upper head injection (UHI) system removed from service. In support of this license amendment, Duke has previously submitted a large break loss of coolant accident (LOCA) 10CFR50.46 analysis utilizing the Westinghouse BASH Evaluation Model. Since BASH is not yet approved, Westinghouse has performed an ECCS analysis for the McGuire Units with UHI removed from service employing the NRC-approve BART Evalaution Model. This letter transmits the results of this large break LOCA 10CFR50.46 analysis, which demonstrate the acceptability of UHI removal at McGuire.

Also enclosed at the request of Mr. James Watt is a discussion of the system behavior calculated for McGuire with and without UHI. Please direct any questions about the submittal to Mr. Brian McIntyre of my staff at (412)-374-5506.

Very truly yours,

E.P. Rahe, Jr., Manager
Nuclear Safety Department

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RESPONSES

TO NRC STAFF

QUESTIONS ABOUT

UHI REMOVAL

AT McGUIRE

9485Q:10/031286

I. Unrealistic, highly conservative specifications of the UHI large break LOCA evaluation model cause current McGuire Plant calculated peak cladding temperatures (PCT) to be higher with UHI installed than without. Several conservative aspects of the UHI model are discussed below.

The UHI upper internals have been designed to uniformly deliver injected UHI water across the core cross-section. Almost all of the fuel assemblies (185 of 193) are located underneath a guide tube or support column; only eight low power corner assemblies do not connect via a direct flow path to the upper head region. Extensive testing demonstrated the uniformity with which UHI water is delivered to the 185 assemblies. Nevertheless, the evaluation model conservatively prohibits quenching of the hot rod independent of blowdown fluid conditions. Thus, the benefit of UHI quench cooling is limited to the non-limiting fuel rods in the core. Speaking in greater detail of the quench model, the data base for the UHI corewide quench criteria is the average quench behavior of the thermocouples at each elevation in the G-2 loop test facility; well over 6000 total data points comprise the data base.

The design quench line applied in the UHI evaluation model is not a true best estimate quench line but provides a 90% confidence that 50% of the true data population lies above the line. A true best estimate quench line would be an upper bound on the design quench line. The requirements of Appendix K are that heat transfer correlations predict conservative results in comparison to the mean of the experimental data throughout the range of parameters for which the correlations are to be used. The design quench line fulfills this requirement. The data used in developing the UHI quench criteria were obtained using boron nitride-filled stainless steel-sheathed electric heater rods which have been shown in the literature to be more difficult to quench than uranium-filled Zircaloy fuel rods. The design quench criteria would then be expected to underpredict the fraction of the core quenched during a hypothetical LOCA in a PWR equipped with UHI.

The model restriction that no quench may occur in an upflow situation restricts the time period during which quench is allowed. This restriction stems from the fact that all test data that were used in developing the quench criteria were downflow. At high flow rates there have been no significant differences noted in heat transfer behavior. Thus, even though the local fluid conditions for upflow may be the same as downflow, quench is not allowed in the calculation. It is believed this restriction results in an underprediction of the fraction of the core quenched during a UHI ECCS calculation.

An additional item that results in an underprediction of the core quenching during a LOCA in a UHI plant is the conservatism in the film boiling heat transfer coefficient calculation. To quench a fuel rod the surface temperature must be reduced to a point where it can be wet. The conservatism in the film boiling heat transfer coefficient causes higher clad temperatures and correspondingly less quenching. The UHI heat transfer model contains several instances where an arbitrary value of 1.0 Btu/Hr-°F-sqft was assigned as a default heat transfer coefficient due to a lack of test data within the range of the parameters. In these cases, the heat transfer coefficient is required to drop from a value of 7-12 Btu/Hr-°F-sqft to the 1.0 Btu/Hr-°F-sqft value for a very small change in parameters. This UHI penalty may be seen by examining Figure I-1. The oscillations between 40 and 110 seconds are a result of the artificiality of the heat transfer model. A heat transfer coefficient in the range of 6-10 Btu/Hr-sqft-°F would appear to be more appropriate during this time period.

The considerations noted above cause the current UHI ECCS model to significantly underpredict the fraction of the core quenched during blowdown. The conservatism of the evaluation model negates much of the PCT benefit UHI has exhibited in tests under LOCA blowdown conditions.

Another evaluation model requirement that diminishes the benefit of UHI is the need to model both the perfect and imperfect mixing of UHI water in the reactor vessel upper head. With the perfect mixing assumption no voids form in the upper head since the injection of the subcooled water begins prior to the system pressure reaching the saturation pressure in the upper head. During the active injection period upper head subcooling continually increases, and relatively little flow out to the core occurs until long after UHI injection is complete. The imperfect mixing assumption allows voids to form in the upper head region during the active injection period. The subcooled UHI water is assumed to fall to the bottom of the upper head (along with fluid entrained by the incoming jet) where it can be forced out through the support columns.

It is worthwhile to pause at this point to review the physical characteristics of the UHI hardware, specifically the flow paths for the UHI support columns and guide tubes. The relationship between the support column and the fuel is shown in the sketch in Figure I-2. The flow from the upper head is delivered directly to the top of the fuel via the support column and hold down assembly. The gap between the hold down assembly and the UHI nozzle is on the order of 0.030 inches.

The guide tube, Figure I-3, is observed to be relatively open to the upper plenum near the base. A close examination of the guide tube shows a significant flow area exists at each card location between the volume enclosed by the guide tube and the upper plenum. For these reasons, LOCA models assume that the flow from the guide tube enters the upper plenum. The limiting initial conditions of the UHI accumulator as regards to pressure and water volume are exactly opposite for perfect and imperfect mixing cases. The conservative evaluation model methodology specifies that the bounding accumulator operating values with uncertainties considered be applied to each case individually. Thus, the impact becomes an inability to optimize UHI accumulator setpoints for either the perfect or imperfect mixing case because of the need to secure an acceptable result for both. Were only one upper head mixing assumption necessary, improved UHI Model ECCS performance could be calculated for McGuire by revising accumulator setpoints.

The conservatisms imposed in the UHI evaluation model cited above have greatly diminished the benefits of UHI observed in testing. Nevertheless, UHI was still perceived to be a benefit under the 1981 Westinghouse Evaluation Model based upon the low flooding rates associated with the ice condenser containment pressure response together with NUREG-0630 burst/blockage models. The BART code provides a more mechanistic, physically correct prediction of core reflood phenomena which leads to improved calculated reflood phase ECCS performance. The true benefit of UHI as calculated in the UHI evaluation model is due to enhancing of the core reflood rate via quenching of fuel in the core during the blowdown. The importance of this effect is greatly reduced when a model using BART is applied, and the more important factor becomes calculated ECCS hot rod performance during blowdown. Due in part to the conservatisms imposed on the UHI evaluation model as cited above, clad temperatures at end of blowdown for McGuire are much higher with UHI than without. Since the reflood enhancement obtained from UHI is no longer nearly as significant on calculated PCT (because of BART), the end of blowdown PCT penalty makes UHI the more limiting case. Also note that in best estimate large break LOCA computations the calculated PCT typically occurs during the first few seconds of blowdown, when UHI has little if any effect one way or another. On a best estimate basis UHI adds little if any safety margin for the large break LOCA event.

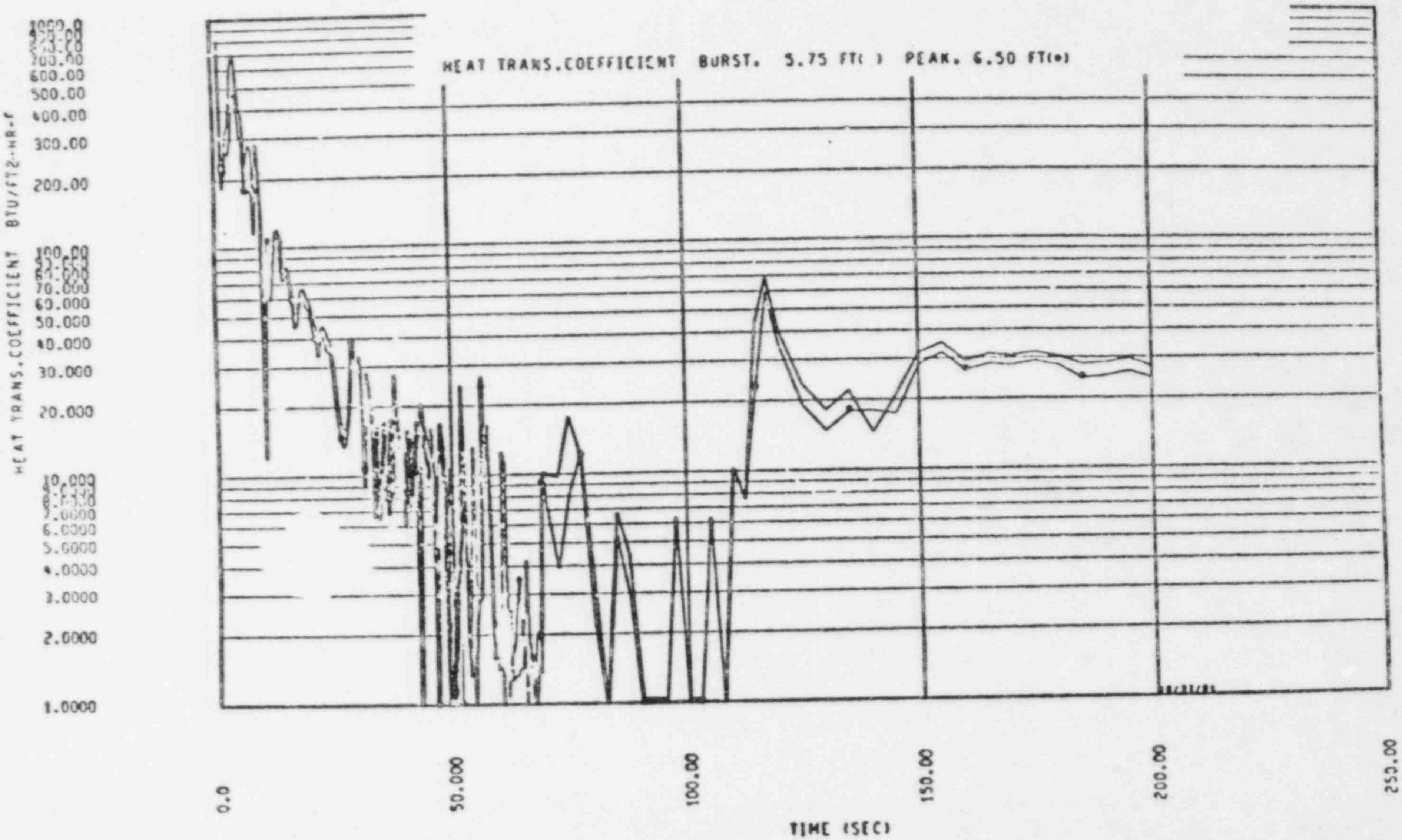

FIGURE I-1 : UHI Model Heat Transfer Coefficient
McQuire C_D=0.6 DECLG with Perfect Mixing

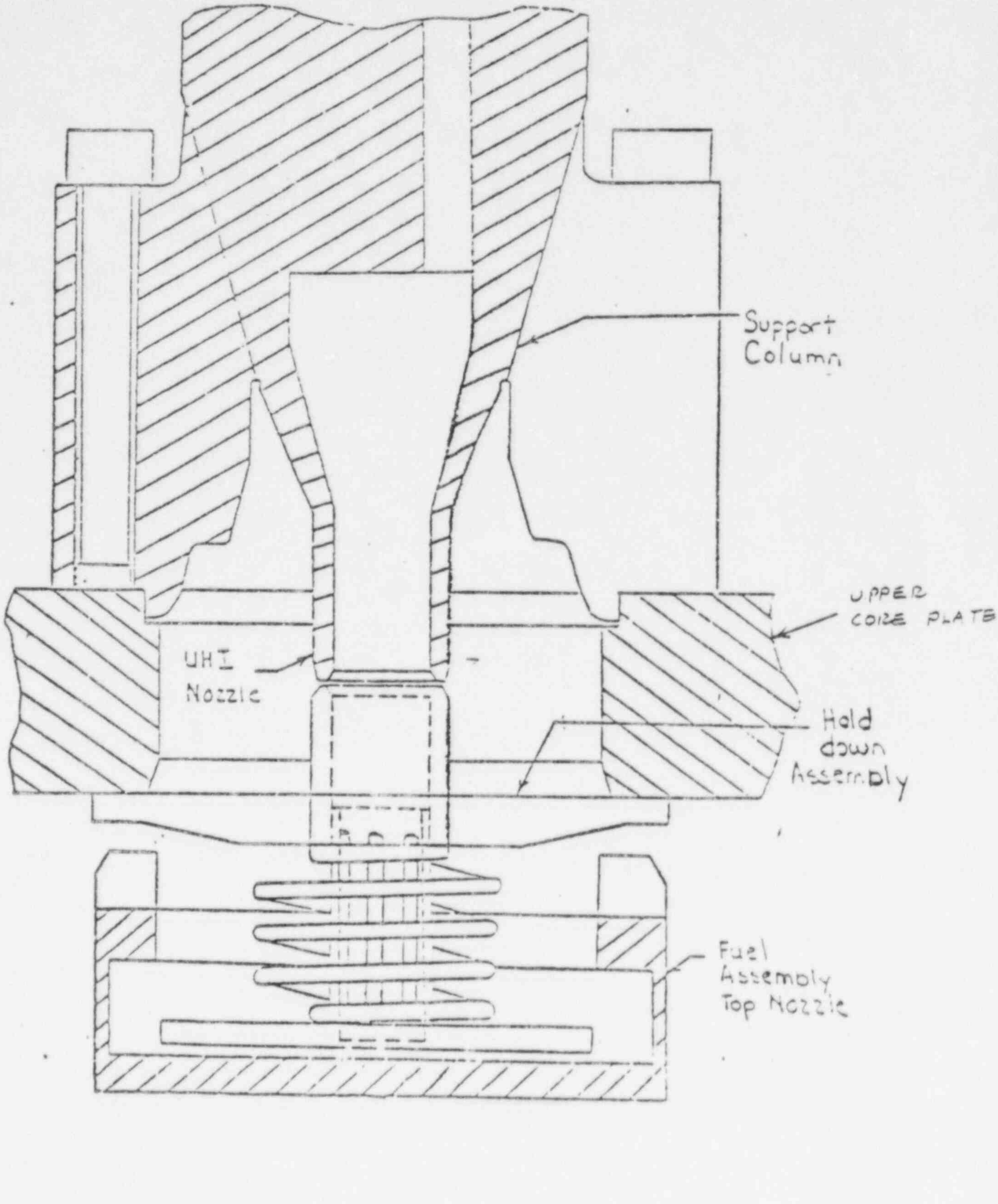

FIGURE I-2 : UHI Support Column at Upper Core Plate

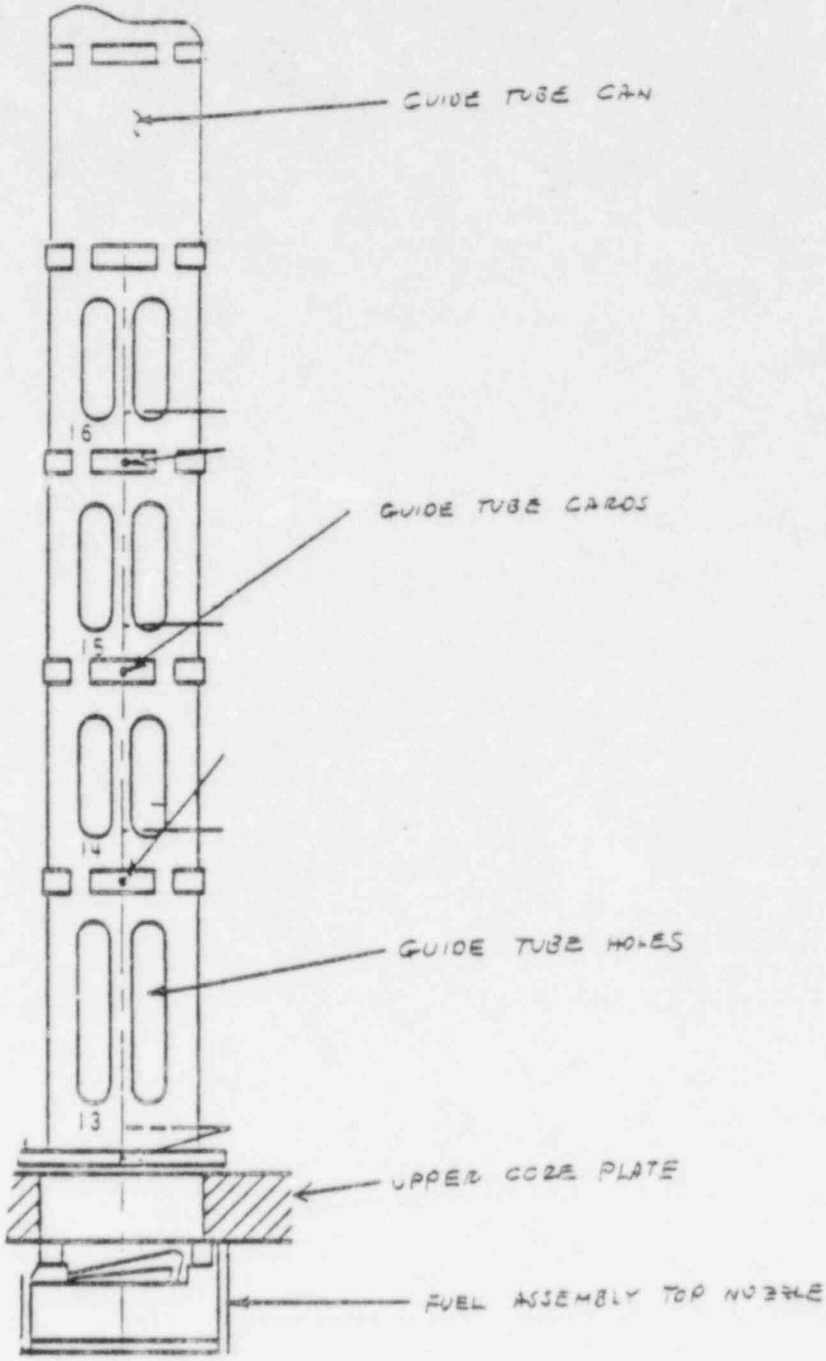

FIGURE I-3 : Lower Guide Tube Structure

II. UHI plants are equipped with very different reactor upper internals from other Westinghouse 4-loop plants. In order to distribute UHI water equitably throughout the core, 185 of the 193 fuel assemblies were located directly beneath a guide tube or support column which communicates directly with the vessel upper head. The much greater flow communication which exists between upper head and core/upper plenum with the UHI internals design produces enhanced thermal-hydraulic conditions within the fuel during a large break LOCA blowdown. As illustrated in Figures II-1 and II-2 respectively, consider the $C_D = 0.6$ DECLG core flows during blowdown for the McGuire and Callaway Plants computed by the Westinghouse SATAN code. The two units in question are 4-loop plants which are similar in design except that McGuire contains UHI-type upper internals. The core flows are similar for the first few seconds, but from five seconds onward the UHI internals are clearly beneficial. Between 5-10 seconds the UHI internals give a greater water delivery into the upper plenum which produces a notably higher positive (in Figure II-1) core flow rate for McGuire; likewise, at around 20 seconds the enhanced water delivery from the upper head at McGuire permits a much greater negative core flow surge than Callaway exhibits in Figure II-2. These greater core mass flow rates directly cause a significant (greater than 100°F) benefit in calculated peak clad temperature for McGuire relative to Callaway at the end of blowdown.

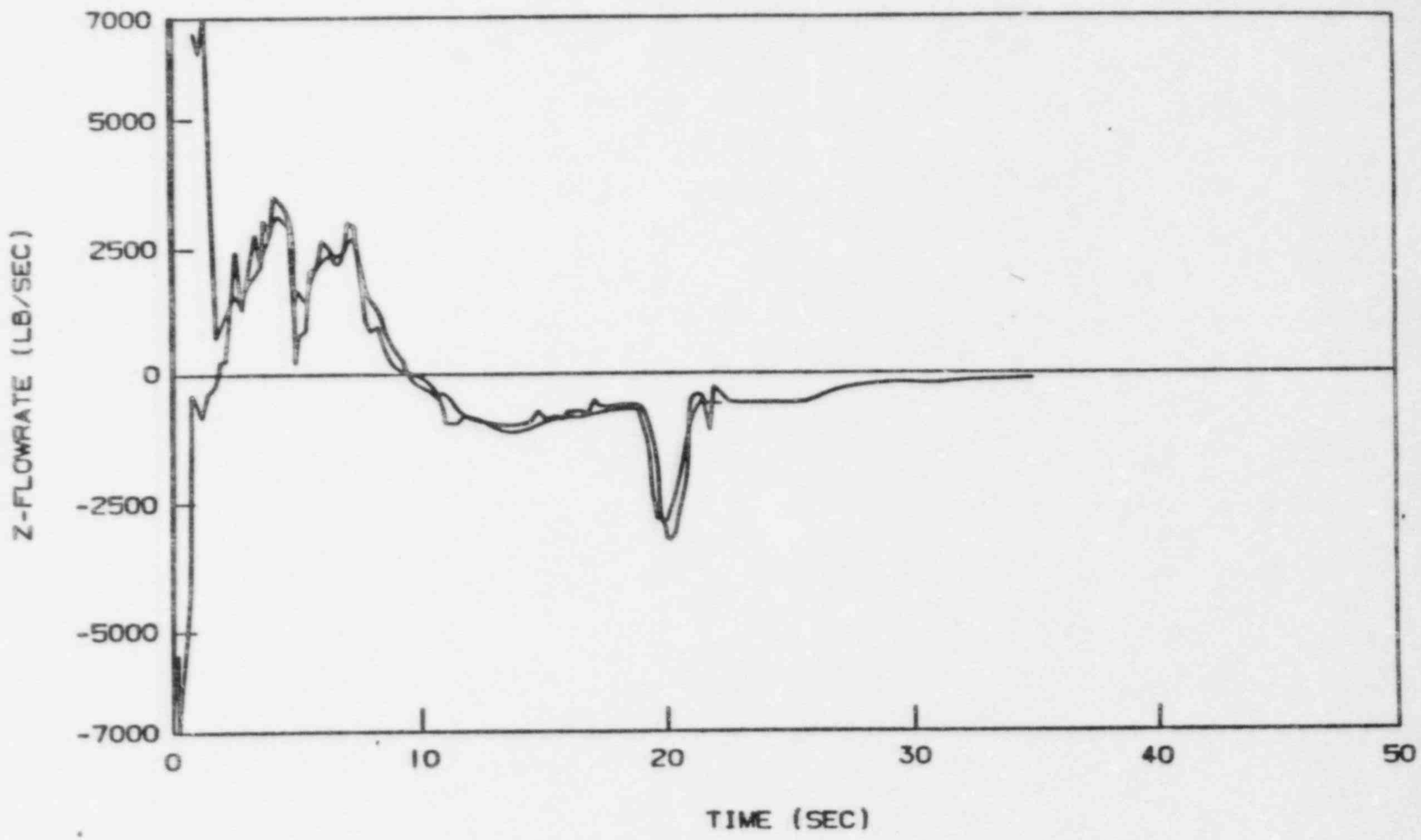

FIGURE II-1 : Core Flowrate with UHI Internals , C_D=0.6 DECLG

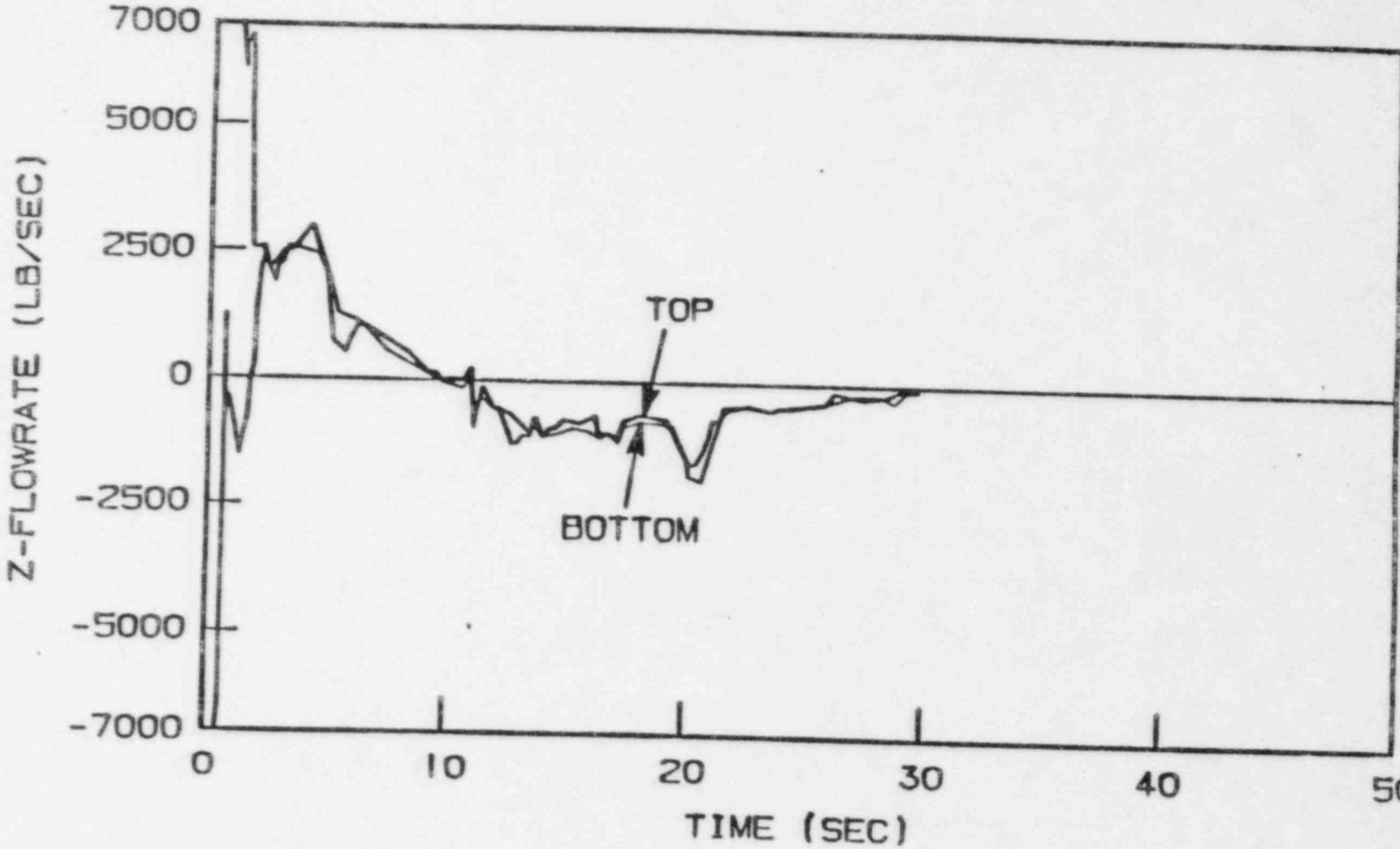

FIGURE II-2 : Core Flowrate with Standard Internals , C_D=0.6 DECLG

15.6.4.1 Identification of Causes and Accident Description 1(b)

Acceptance Criteria and Frequency Classification

A loss-of-coolant accident (LOCA) is the result of a pipe rupture of the reactor coolant system (RCS) pressure boundary. For the analyses presented here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is considered an ANS Condition IV event, a limiting fault. See Section 15.0.1 for a discussion of Condition IV events.

A minor pipe break (small break), as considered in this section, is defined as a rupture of the reactor coolant pressure boundary (Section 5.2) with a total cross-sectional area less than 1.0 ft^2 in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This is considered a Condition III event, an infrequent fault. See Section 15.0.1 for a discussion of Condition III events.

The Acceptance Criteria for the loss-of-coolant accident is described in 10 CFR 50.46 as follows:

a. The calculated peak fuel element cladding temperature is below the requirement of 2200°F.

b. The cladding 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.

c. The amount of hydrogen generated by fuel element cladding that reacts chemically with water or steam does not exceed an amount corresponding to interaction of 1% of the total amount of Zircaloy in the reactor.

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

e. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long lived radioactivity remaining in the core.

These criteria were established to provide significant margin in ECCS performance following a LOCA.

In all cases, small breaks (less than 1.0 ft^2) yield results with more margin to the Acceptance Criteria limits than large breaks.

Description of a Large Break LOCA Transient

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

a. 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. However, no credit is taken in the LOCA analysis for boron content of the injection water aiding in shutdown. In addition, the insertion of control rods to shut down the reactor is neglected in the large break analysis.

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

The sequence of events following a large break LOCA are presented in Figure 15.6.4-1.

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 fission product decay, hot internals and the vessel continues to be transferred to the reactor coolant. At the beginning of the blowdown phase, the entire RCS contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. Thereafter, the core heat transfer is based on local conditions with transition boiling and forced convection to steam as the major heat transfer mechanisms.

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

When the Reactor Coolant system depressurizes to approximately 600 psia, the accumulators begin to inject borated water into the reactor coolant loops.

Since the loss of offsite power is assumed, the reactor coolant pumps are assumed to trip at the beginning of the accident. The effects of pump coastdown are included in the blowdown analysis.

The blowdown phase of the transient ends when the RCS pressure (initially assumed at 2280 psia) falls to a value approaching that of the containment atmosphere. Prior to or at the end of the blowdown, the mechanisms that are responsible for the bypassing of emergency core cooling injection water into the RCS are calculated not to be effective. At this time (called end of bypass) refill of the reactor vessel lower plenum begins. Refill is complete when emergency core cooling water has filled the lower plenum of the reactor vessel, which is bounded by the bottom of the fuel rods (called bottom of core recovery time).

The reflood phase of the transient is defined as the time period lasting from the end of refill until the reactor vessel has been filled with water to the extent that the core temperature rise has been terminated. From the later stage of blowdown and then the beginning of reflood, the safety injection accumulator tanks rapidly discharge borated cooling water into the RCS, contributing to the filling of the reactor vessel downcomer. The downcomer water elevation head provides the driving force required for the reflooding of the reactor core. The low head and high head safety injection pumps aid in the filling of the downcomer and subsequently supply water to maintain a full downcomer and complete the reflooding process. The safety injection pumped flow as a function of pressure is given in Table 15.6.4-6 for the large break cases.

Continued operation of the ECCS pumps supplies water during long-term cooling. Core temperatures have been reduced to long-term steady-state levels associated with dissipation of residual heat. After the water level in the refueling water storage tank (RWST) reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching to the cold leg recirculation phase of operation in which spilled borated water is drawn from the containment sump by the low head safety injection (RHR) pumps and returned to the RCS cold legs. The Containment Spray System continues to operate to further reduce containment pressure. Approximately 15 hours after initiation of the LOCA, the ECCS is realigned to supply water to the RCS hot legs in order to control the boric acid concentration in the reactor vessel.

Description of Small Break LOCA Transient

Ruptures of small cross section will cause expulsion of the coolant at a rate which can be accommodated by the charging pumps. These pumps would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown. The coolant which would be released to the containment contains the fission products existing at equilibrium.

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the Reactor Coolant System through the postulated break against the charging pump makeup flow at normal Reactor Coolant System pressure, i.e., 2250 psia. A makeup flow rate from one centrifugal charging pump is typically adequate to sustain pressurizer level at 2250 psia for a break through a 0.375 inch diameter hole. This break results in a loss of approximately 17.25 lb/sec.

Should a larger break occur, depressurization of the Reactor Coolant System causes fluid to flow into the loops from the pressurizer resulting in a pressure and level decrease in the pressurizer. Reactor trip occurs when the low pressurizer pressure trip setpoint is reached. During the earlier part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as they are coasting down following reactor trip. Therefore, upward flow through the core is maintained. The Safety Injection System is actuated when the appropriate setpoint is reached. The consequences of the accident are limited in two ways:

1. Reactor trip and borated water injection complement void formation in the core and cause a rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay.

2. Injection of borated water ensures sufficient flooding of the core to prevent excessive clad temperatures.

Before the break occurs the plant 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. The heat transfer between the Reactor Coolant System and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary, system pressure increases and steam dump may occur. Makeup to the secondary side is automatically provided by the auxiliary feedwater pumps. The safety injection signal stops normal feedwater flow by closing the main feedwater line isolation valves and initiates auxiliary feedwater flow by starting auxiliary feedwater pumps. The secondary flow aids in the reduction of Reactor Coolant System pressures.

When the RCS depressurizes to 600 psia, the cold leg accumulators begin to inject water into the reactor coolant loops. Due to the loss of offsite power assumption, the reactor coolant pumps are assumed to be tripped at the time of reactor trip during the accident and the effects of pump coastdown are included in the blowdown analyses.

15.6.4.2 Analysis of Effects and Consequences

Methods of Analysis

The requirements of an acceptable ECCS Evaluation Model are presented in Appendix K of 10 CFR 50 (Reference 3). The requirements of Appendix K regarding specific model features were met by selecting models which provide a significant overall conservatism in the analysis. The assumptions made pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs and include such items as the core peaking factors, the containment pressure, and the performance of the ECCS system. Decay heat generated throughout the transient is also conservatively calculated as required by Appendix K of 10 CFR 50. The thermal-hydraulic analyses reported in this section were performed with an upper head fluid temperature of T_{cold}.

Large Break Evaluation Model

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

The description of the various aspects of the LOCA analysis methodology is given in References 4, 10, 13 and 14. These documents describe the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the Acceptance Criteria. The SATAN-VI (Reference 5), WREFLOOD (Reference 6), LOTIC (Reference 7), BART (Reference 13) and LOCTA-IV (Reference 8) codes are used to assess the core heat transfer geometry and to determine if the core remains amenable to cooling throughout and subsequent to the blowdown, refill, and reflood phases of the LOCA. The SATAN-VI computer code analyzes the thermal-hydraulic transient in the RCS during blowdown. The WREFLOOD and BART computer codes are used to calculate the thermal-hydraulic transient during the reflood phase of the accident. The BART computer code is used to calculate the fluid and heat transfer conditions in the core during reflood. The LOTIC 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. Fuel parameters input to the LOCTA-IV code were taken from a new version of the PAD code (Reference 9).

SATAN-VI is used to calculate the RCS pressure, enthalpy, density and mass and energy flow rates, as well as steam generator heat transfer between the primary and secondary systems, as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator water mass and internal pressure and the pipe break mass and energy flow rates that are assumed to be vented to the containment during blowdown. At the end of the blowdown and refill phases, these data are transferred to the WREFLOOD code. The mass and energy release rates during blowdown and reflood are transferred to the LOTIC code for use in the determination of the containment pressure response during these phases of the LOCA.

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 addition to the containment through the break. Since the mass flow rate to the containment depends upon the core flooding rate and the local core pressure, which is a function of the containment backpressure, the transient pressure computed by the LOTIC code is input to the WREFLOOD code. With input and boundary conditions from WREFLOOD, the mechanistic core heat transfer model in BART calculates the hydraulic and heat transfer conditions in the core during reflood. LOCTA-IV is used throughout the analysis of the LOCA transient to calculate the fuel clad temperature and metal-water reaction of the hottest rod in the core. A schematic representation of the computer code interfaces for large break calculations is shown in Figure 15.6-2.

The LOTIC code is a mathematical model of the ice condenser containment. LOTIC is described in detail in Reference 7. LOTIC is run using output from SATAN and WREFLOOD, which provide the necessary mass and energy releases to the containment. In this analysis the WREFLOOD/LOTIC system is used only to provide containment boundary conditions required by BASH.

The LOCTA-IV code is a computer program that evaluates fuel, cladding and coolant temperatures during a LOCA. A more complete description than is presented here can be found in Reference 8. In the LOCTA detailed fuel rod model, for the calculation of local heat transfer coefficients, the empirical FLECHT correlation is replaced by the BART code. BART employs rigorous mechanistic models to generate heat transfer coefficients appropriate to the actual flow and heat transfer regimes experienced by the LOCTA fuel rods. This is considered a more dynamic realistic approach than relying on a static empirical correlation.

Small Break LOCA Evaluation Model

The NOTRUMP computer code is used in the analysis of loss-of-coolant accidents due to small breaks in the reactor coolant system. The NOTRUMP computer code is a state-of-the-art one-dimensional general network code consisting of a number of advanced features. Among these features are the calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. The NOTRUMP small break LOCA emergency core cooling system (ECCS) evaluation model was developed to determine the RCS response to design basis small break LOCAs and to address the NRC concerns expressed in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plants."

In NOTRUMP, the RCS is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicitly with the intact loops lumped into a second loop. The transient behavior of the system is determined from the governing conservation equations of mass, energy and momentum applied throughout the system. A detailed description of NOTRUMP is given in References 11 and 15.

The use of NOTRUMP in the analysis involves, among other things, the representation of the reactor core as heated control volumes with an associated bubble rise model to permit a transient mixture height calculation. The multinode capability of the program enables an explicit and detailed spatial representation of various system components. In particular, it enables a proper calculation of the behavior of the loop seal during a loss-of-coolant transient.

Cladding thermal analyses are performed with the LOCTA-IV (Reference 8) code which uses the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height history from the NOTRUMP hydraulic calculations, as input.

A schematic representation of the computer code interfaces is given in Figure 15.6.4-3.

The small break analysis was performed with the approved Westinghouse ECCS Small Break Evaluation Model (References 8, 11 and 15).

Large Break Input Parameters and Initial Conditions

Table 15.6.4-1 lists important input parameters and initial conditions used in the large break analyses.

Small Break Input Parameters and Initial Conditions

Table 15.6.4-1 lists important input parameters and initial conditions used in the small break analyses.

The axial power distribution and core decay power assumed for the small break analyses are shown in Figures 15.6.4-60 and 15.6.4-61.

Safety injection flow rate to the Reactor Coolant System as a function of the system pressure is used as part of the input. The Safety Injection (SI) system was assumed to be delivering to the RCS 25 seconds after the generation of a safety injection signal.

For these analyses, the SI delivery considers pumped injection flow which is depicted in Figure 15.6.4-62 as a function of RCS pressure. This figure represents injection flow from the SI pumps based on performance curves degraded 5 percent from the design head. The 25 second delay includes time required for diesel startup and loading of the safety injection pumps onto the emergency buses. The effect of flow from the RHR pumps is not considered here since their shutoff head is lower than RCS pressure during the time portion of the transient considered here. Also, minimum safeguards Emergency Core Cooling System capability and operability has been assumed in this analysis.

The hydraulic analyses are performed with the NOTRUMP code using 102% of the licensed NSSS core power. The core thermal transient analyses are performed with the LOCTA-IV code using 102% of licensed NSSS core power.

Large Break Results

Based on the results of the LOCA sensitivity studies (Reference 12), the limiting large break was found to be double-ended cold leg guillotine (DECLG). Therefore, only the DECLG break is considered in the large break ECCS performance analysis. Calculations were performed for a range of Moody break discharge coefficients (C_D). Consistent with the methodology described in Reference 16 the break size which resulted in the worst case for minimum safety injection was used in a calculation in which no failures of the ECCS were assumed (Maximum safeguards). The results of these calculations are summarized in Tables 15.6.4-2 through 15.6.4-5.

Figures 15.6.4-4 through 15.6.4-44 present the parameters of principal interest from the large break ECCS analyses. Transients of the following parameters are presented for each discharge coefficient analyzed, and where appropriate for the worst break maximum safeguards case.

Figure 15.6.4-4 through Figure 15.6.4-15	The following quantities are presented for the hot spot (location of maximum clad temperature) and the burst elevation on the hottest fuel rod (hot rod): 1. fluid quality 2. mass velocity 3. heat transfer coefficient The heat transfer coefficient shown is calculated by the BART code.
Figure 15.6.4-16 through Figure 15.6.4-21	The system pressure shown is the calculated pressure in the core. Core flowrates are also presented.
Figure 15.6.4-22 through Figure 15.6.4-29	These figures show the hot spot clad temperature transient and the clad temperature transient at the burst location. The fluid temperature shown is also both locations. The nodal notation of the figures is defined in Table 15.6.5-7.
Figure 15.6.4-30 through Figure 15.6.4-37	These figures show the core reflood transient.
Figure 15.6.4-38 through Figure 15.6.4-40	These figures show the cold leg accumulator delivery during blowdown.
Figure 15.6.4-41 through Figure 15.6.4-44	The pumped safety injection during reflood and the calculated containment pressure are presented for the C_D = 0.6 DECLG maximum and minimum safeguards cases.

The maximum cladding temperature calculated for a large break is 2132°F which is less than the Acceptance Criteria limit of 2200°F of 10 CFR 50.46. The maximum local metal water reaction is 5.1 percent, which is well below the embrittlement limit of 17 percent as required by 10 CFR 50.46. The total core metal water reaction is less than 0.3 percent for all breaks, as compared with the 1 percent criterion of 10 CFR 50.46, and the cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

<u>Small Break Results</u>

As noted previously, the calculated peak cladding temperature resulting from a small break LOCA is less than that calculated for a large break. A range of small break analyses are presented which establishes the limiting break size. The results of these analyses are summarized in Tables 15.6.4-7 and 15.6.4-8. Figures 15.6.4-63a through 15.6.4-71 present the principal parameters of interest for the small break ECCS analyses. For all cases analyzed, the following transient parameters are included:

a. RCS pressure
b. core mixture height
c. hot spot clad temperature

For the limiting break analyzed (3 inch), the following additional transient parameters are presented (Figures 15.6.4-72 through 15.6.4-74):

a. core steam flow rate
b. core heat transfer coefficient
c. hot spot fluid temperature

The maximum calculated peak cladding temperature for the small breaks analyzed is 1488°F. These results are well below all Acceptance Criteria limits of 10 CFR 50.46 and no case is limiting when compared to the results presented for large breaks.

<u>Transition Core Impact</u>

The large break loss-of-coolant accident (LOCA) analysis presented herein for McGuire Units 1 and 2 considered a fuel core of optimized fuel. This is consistent with the methodology employed in the Reference Core Report 17 x 17 Optimized Fuel Assembly (OFA) for 17 x 17 OFA Transition (WCAP-9500).

When assessing the impact of transition cores on large break LOCA analysis, it must be determined whether the transition core can have a greater calculated peak clad temperature (PCT) than either a complete core of the reference design or a complete core of the new fuel design. For a given peaking factor, the only mechanism available to cause a transition core to have a greater calculated PCT than a full core of either fuel is the possibility of flow redistribution due to fuel assembly hydraulic resistence mismatch. This hydraulic resistance mismatch may exist only for transition cores and is the only unique difference between a complete core of either fuel type and the transition core.

The difference in fuel assembly resistance (K/A^2) for the two assembly designs [17 x 17 Standard/17 x 17 OFA] may impact two portions of the large break LOCA analysis model. One is the reactor coolant system (RCS) blowdown portion of the transient analyzed with the SATAN-VI computer code, where the higher resistance 17 x 17 OFA assembly has less cooling flow than the 17 x 17 standard fuel assembly. While the SATAN-VI code models the crossflows between the average core flow channel (N-1 fuel assemblies) and a hot assembly flow channel (one fuel assembly), experience has shown that SATAN-VI results are not significantly affected by small differences in the hydraulic resistance between these two channels.

To better understand the transition core large break LOCA blowdown transient phenomena, conservative blowdown fuel clad heatup calculations have been performed to determine the clad temperature effect on the new fuel design for mixed core configurations. The effect was determined by reducing the axial flow in the hot assembly at the appropriate elevations to simulate the effects of the transition core hydraulic resistance mismatch. In addition, the <u>W</u> blowdown evaluation model was modified to account for grid heat transfer enhancement during blowdown for this evaluation. The results of this analysis have shown that no peak clad temperature penalty is observed during blowdown. Therefore, it is not necessary to perform a new blowdown calculation for transition core configurations because the Evaluation Model blowdown calculation performed for the full 17 x 17 OFA core is conservative and bounding.

The other portion of the LOCA calculation impacted by hydraulic resistance mismatch is the core reflood transient. Fuel assembly design specific analyses have been performed with a version of the BART computer code which accurately models mixed core cases during reflood. Westinghouse transition core designs including specific 14 x 14, 15 x 15 and 17 x 17 standard to OFA transition core cases were analyzed. For each of these cases, BART modelled both fuel assembly types and predicted the reduction in axial flow at the appropriate elevations. As expected, the increase in hydraulic resistance mismatch for the 17 x 17 OFA assembly was shown to produce a reduction in reflood steam flow rate for the 17 x 17 OFA assembly at the mixing vane grid elevations during the transition core period. This reduction in steam flow rate is offset by the fuel grid heat transfer enhancement predicted during reflood. The various fuel assembly specific transition core analyses performed resulted in peak clad temperature increases of up to 10°F for core axial elevations where PCTs can possibly occur. Therefore, the maximum PCT penalty possible for 17 x 17 OFA during transition cores is 10°. Once a full core of the 17 x 17 OFA fuel is achieved, the large break LOCA analysis with UHI removed will apply without the crossflow penalty.

15.6.4.3 Environmental Consequences

The postulated consequences of a LOCA are calculated for 1) offsite and 2) control room operators.

Offsite Dose Consequences

The offsite radiological consequences of a LOCA are calculated based on the following assumptions and parameters.

1. 100 percent of the core noble gases and 25 percent of the core iodines are released to the containment atmosphere.
2. 50 percent of the core iodines are released to the containment.
3. Annulus activity which is exhausted prior to the time at which the annulus reaches a negative pressure of -0.25 in. w.g. is unfiltered.
4. ECCS leakage begins at the earliest possible time sump recirculation can begin.
5. ECCS leakage occurs at twice the maximum operational leakage.
6. Bypass leakage is 7 percent of total containment leakage.
7. The effective annulus volume is 50 percent of the actual volume.
8. The annulus filters become faulted at 900 seconds resulting in a 15 percent reduction in flow.
9. Elemental iodine removal by the ice condenser begins at 600 seconds and continues for 2540 seconds with a removal efficiency of 30 percent.

10. One of the containment air return fans is assumed to fail.

11. The containment leak rate is fifty percent of the Technical Specification limit after 1 day.

12. Iodine partition factor for ECCS leakage is 0.1 for the course of the accident.

13. No credit is taken for the auxiliary building filters for ECCS leakage.

14. The redundant hydrogen recombiners and igniters fail. Therefore, purges are required for hydrogen control.

15. The annulus reaches equilibrium after 200,000 seconds such that the only discharge is due to inleakage.

16. Water density at 160°F is used to calculate the sump water mass.

17. Other assumptions are listed in Table 15.6.4-10.

Based on the model in Appendix 15A, the thyroid and whole body doses are calculated at the exclusion area boundary and the low population zone. The doses are presented in Table 15.6.5-10 and are within the limits of 10 CFR 100.

Control Room Operator Dose

The maximum postulated dose to a control room operator is determined based on the releases of a Design Basis Accident. In addition to the parameters and assumptions listed above, the following apply:

1. The control room pressurization rate is 1,000 cfm; the filtered recirculation rate is 1,000 cfm.

2. The unfiltered inleakage into the control room is 10 cfm.

3. Other assumptions are listed in Table 15.6.4-11.

15.6.5 A NUMBER OF BWR TRANSIENTS

Not applicable to McGuire.

REFERENCES FOR SECTION 15.6

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2. Chelemer, H., Boman, L. H., Sharp, D. R., "Improved Thermal Design Procedures", WCAP-8567, July 1975.

3. "Acceptance Criteria for Emergency Core Cooling System for Light Water Cooled Nuclear Power Reactors", 10 CFR 50.46 and Appendix K of 10 CFR 50. Federal Register, Volume 39, Number 3, January 4, 1974.

4. Bordelon, F. M., Massie, H. W. and Borden, T. A., "Westinghouse ECCS Evaluation Model-Summary", WCAP-8339, (Non-Proprietary), July 1974.

5. Bordelon, F. M., et. al., "SATAN-VI Program: Comprehensive Space Time Dependent Analysis of Loss of Coolant", WCAP-8302, (Proprietary) June 1974, and WCAP-8303, (Non-Proprietary), June 1974.

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9. Rahe, E. P., Westinghouse letter to Thomas, C. O., U.S.N.R.C., Letter Number NS-EPR-2673, October 27,1982, Subject: "Westinghouse Revised PAD Code Thermal Safety Model", WCAP-8720, Addendum 2 (Proprietary).

10. Westinghouse ECCS Evaluation Model, 1981 Version", WCAP-9220-P-A, Rev. 1 (Proprietary), WCAP-9221-A, Rev. 1 (Non-Proprietary), February, 1982.

11. Lee, H., Tauche, W. D., Schwarz, W. R., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code", WCAP-10081-A, August 1985.

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13. Young, M., et. al., "BART-1A: A Computer Code for the Best Estimate Analyzed Reflood Transients", WCAP-9561-P-A, 1984 (Westinghouse Proprietary).

14. Chiou, J. S., et.al., "Models for PWR Reflood Calculations Using the BART Code", WCAP-10062.

15. Meyer, P. E. and Kornfilt, J., "NOTRUMP, A Nodal Transient Small Break and General Network Code", WCAP-10080-A, August 1985.

16. Rahe, E. P. (Westinghouse), letter to Tedesco, R. L. (USNRC), No. NS-EPR-2538, December 1981.

TABLE 15.6.4-1

Input Parameters Used in the ECCS Analyses

Parameter	Large Break	Small Break
Peak Linear Power (kw/ft) (includes 102% factor)	12.545	12.21
Total Peaking Factor, F_Q	2.26	2.32
Power Shape	Chopped Cosine	See Figure 15.6.4-60
Fuel Assembly Array	17 X 17 Optimized	17 X 17 Optimized
Nominal Cold Leg Accumulator Water Volume (ft^3/accumulator)	950	950
Nominal Cold Leg Accumulator Tank Volume (ft^3/accumulator)	1350	1350
Minimum Cold Leg Accumulator Gas Pressure (psia)	600	600
Pumped Safety Injection Flow	See Table 15.6.4-6	See Figure 15.6.4-62
Steam Generator Initial Pressure (psia)	987.0	987.0
Steam Generator Tube Plugging Level (%)	3	5

TABLE 15.6.4-2

Large Break LOCA Time Sequence of Events

	C_D = 0.8 DECLG (sec)	C_D = 0.6 DECLG (sec)	C_D = 0.4 DECLG (sec)
Start	0.0	0.0	0.0
Reactor Trip Signal	0.46	0.46	0.47
Safety Injection Signal	2.6	2.7	2.9
Cold Leg Accumulator Injection	12.7	15.3	21.0
Pump Injection	27.6	27.7	27.9
End of Bypass	28.42	33.1	44.5
End of Blowdown	28.6	34.8	44.6
Bottom of Core Recovery	45.3	50.9	64.1
Cold Leg Accumulator Empty	67.2	71.1	79.2

TABLE 15.6.4-3

Large Break LOCA Time Sequence of Events

Maximum Safeguards

	C_D= 0.6 DECLG (sec)
Start	0.0
Reactor Trip Signal	0.46
Safety Injection Signal	2.7
Cold Leg Accumulator Injection	15.3
Pump Injection	27.7
End of Bypass	33.1
End of Blowdown	34.8
Bottom of Core Recovery	49.6
Cold Leg Accumulator Empty	78.0

TABLE 15.6.4-4

Large Break LOCA Results

Fuel Cladding Data

	C_D = 0.8 DECLG	C_D = 0.6 DECLG	C_D = 0.4 DECLG
RESULTS			
Peak Clad Temperature (°F)	1865	1895	1863
Peak Clad Temperature Location (ft)	6.75	6.75	6.75
Local Zr/H_2O Reaction (max), (%)	2.53	2.12	2.16
Local Zr/H_2O Location (ft)	5.50	6.00	5.50
Total Zr/H_2O Reaction, (%)	<0.3	<0.3	<0.3
Hot Rod Burst Time, (sec)	61.4	62.2	88.8
Hot Rod Burst Location, (ft)	5.50	6.00	5.50

TABLE 15.6.4-5

Large Break LOCA Results

Fuel Cladding Data

Maximum Safeguards

	$C_D = 0.6$ DECLG
RESULTS	
Peak Clad Temperature (°F)	2132
Peak Clad Temperature Location (ft)	6.50
Local Zr/H_2O Reaction (max), (%)	5.05
Local Zr/H_2O Location (ft)	6.50
Total Zr/H_2O Reaction, (%)	<0.3
Hot Rod Burst Time, (sec)	63.0
Hot Rod Burst Location, (ft)	6.00

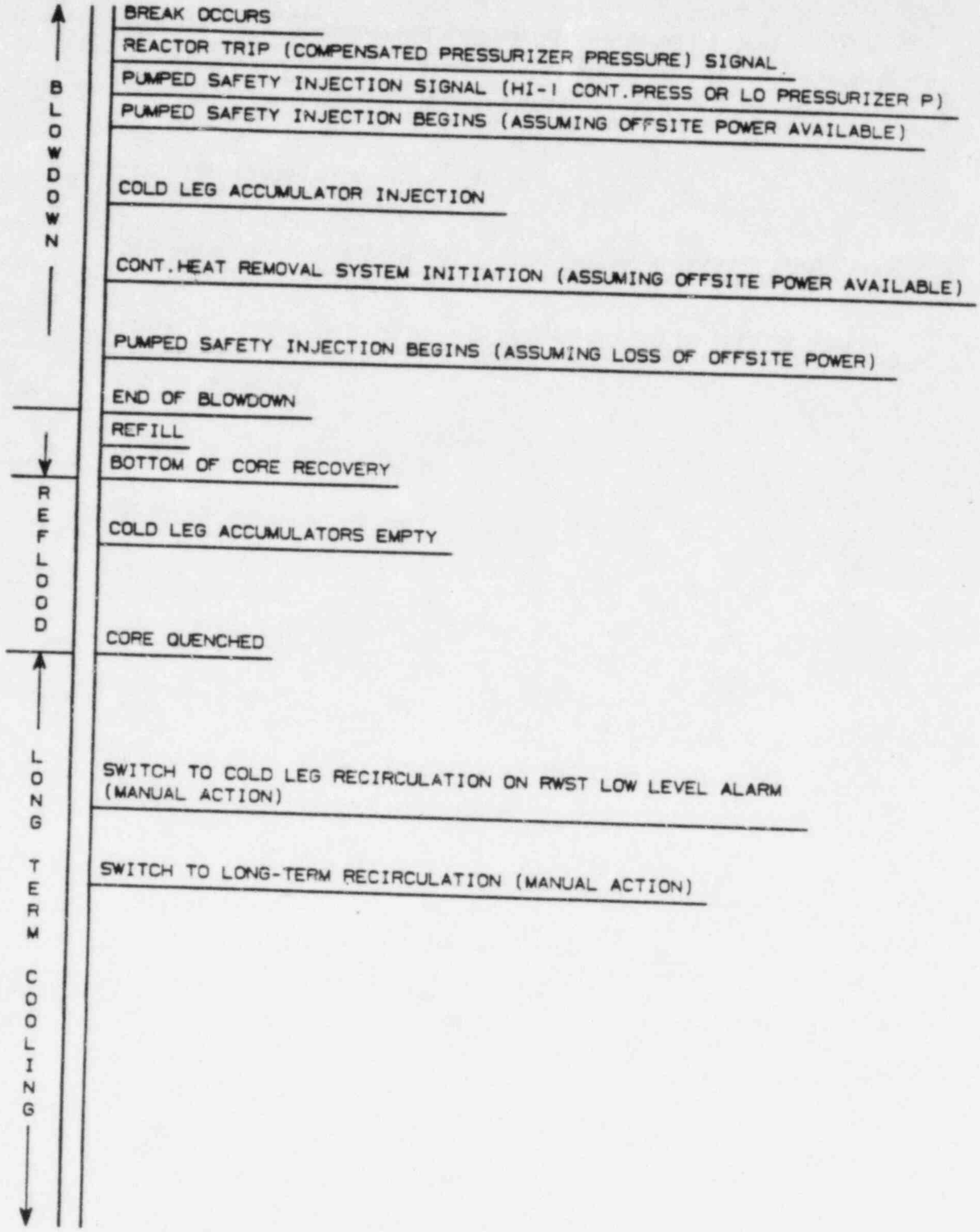

Figure 15.6.4-1: Sequence of Events for Large Break Loss-of-Coolant Analysis

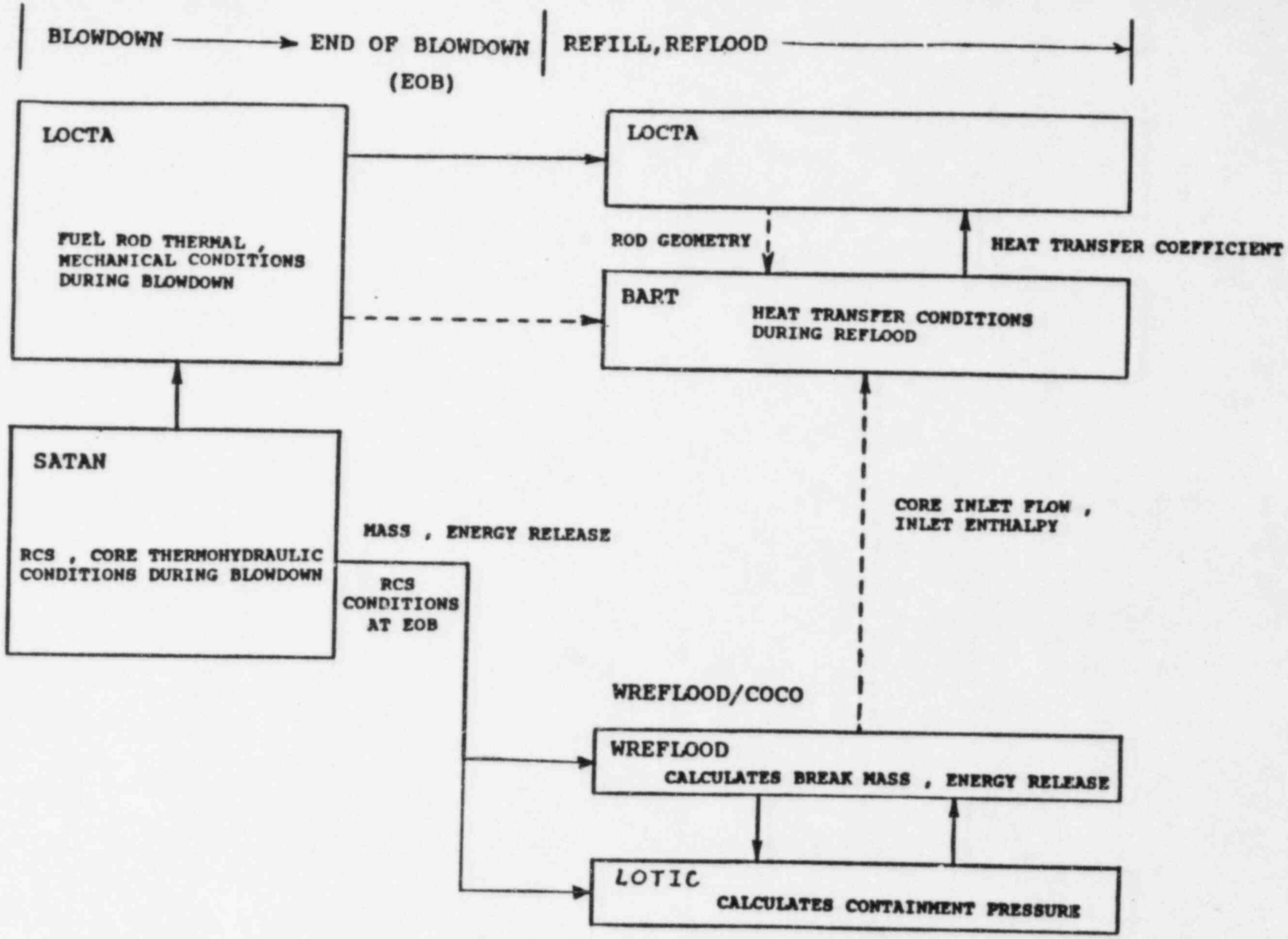

Figure 15.6-2 APPROVED WESTINGHOUSE APPENDIX K LOCA EVALUATION MODEL WITH BART

14715.54

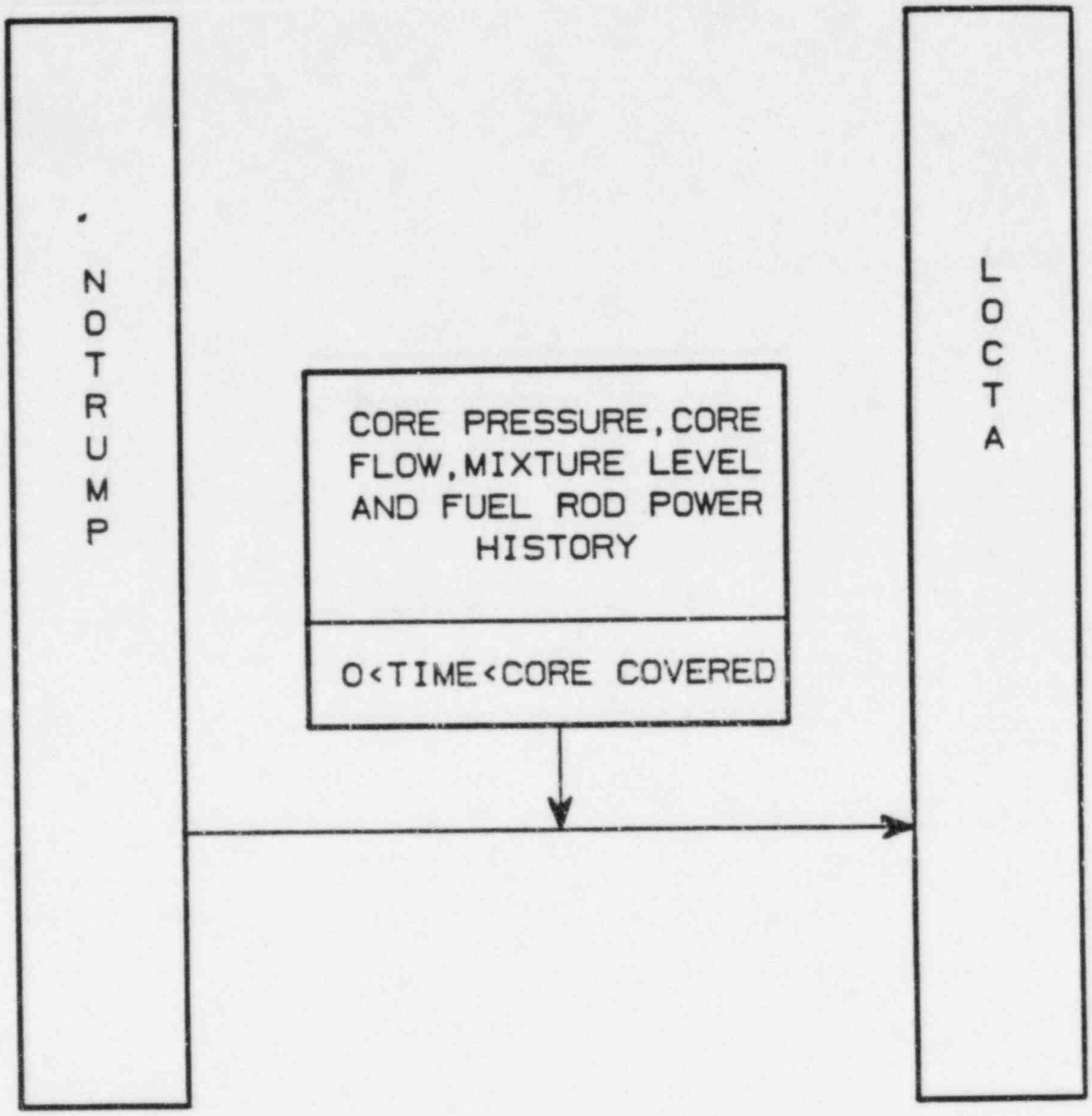

Figure 15.6.4-3: Code Interface Description for Small Break Model

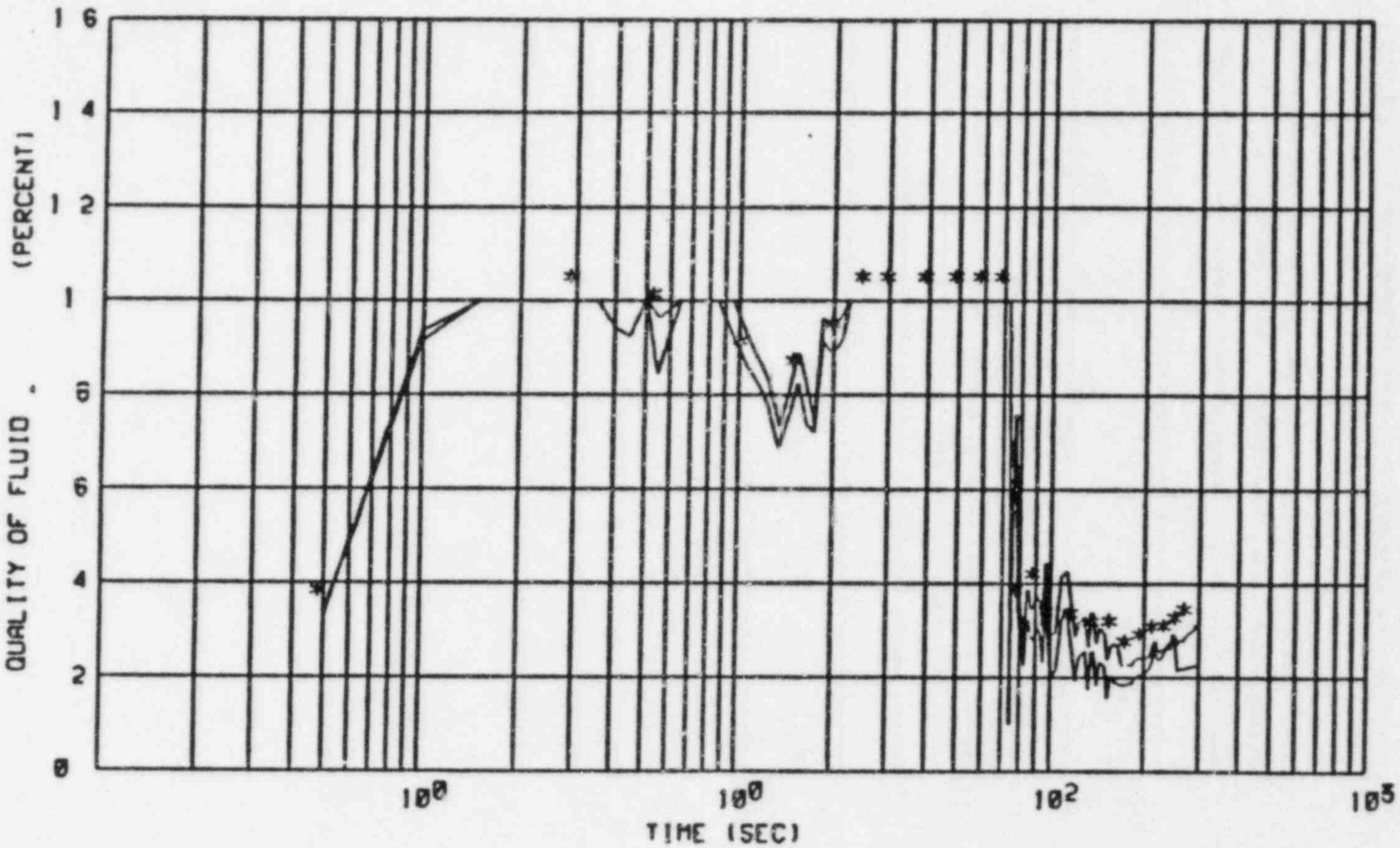

Figure 15.6.4-4: Fluid Quality, C_D=0.8 DECLG

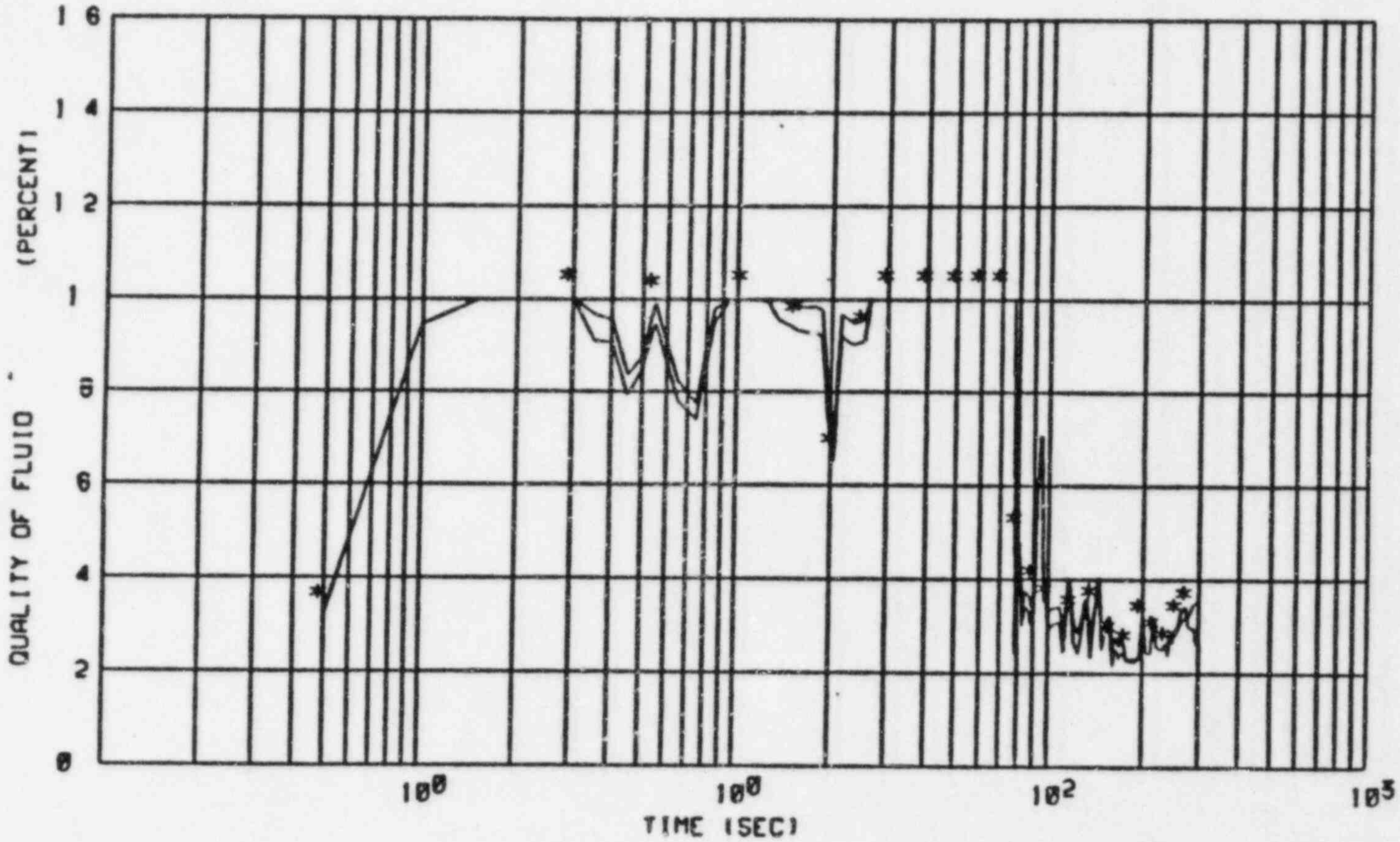

Figure 15.6.4-5: Fluid Quality, C_D=0.6 DECLG

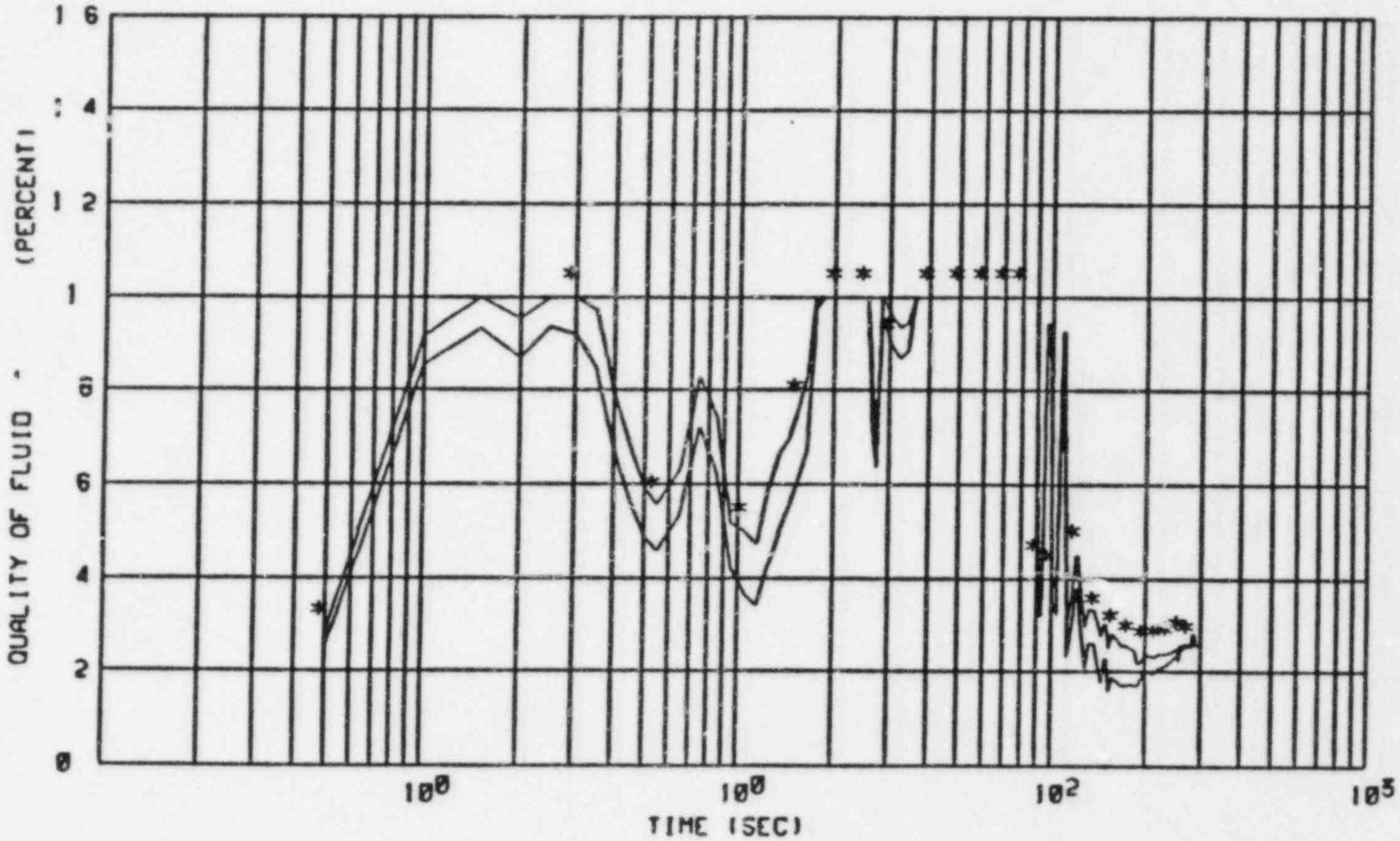

Figure 15.6.4-6: Fluid Quality, C_D=J.4 DECLG

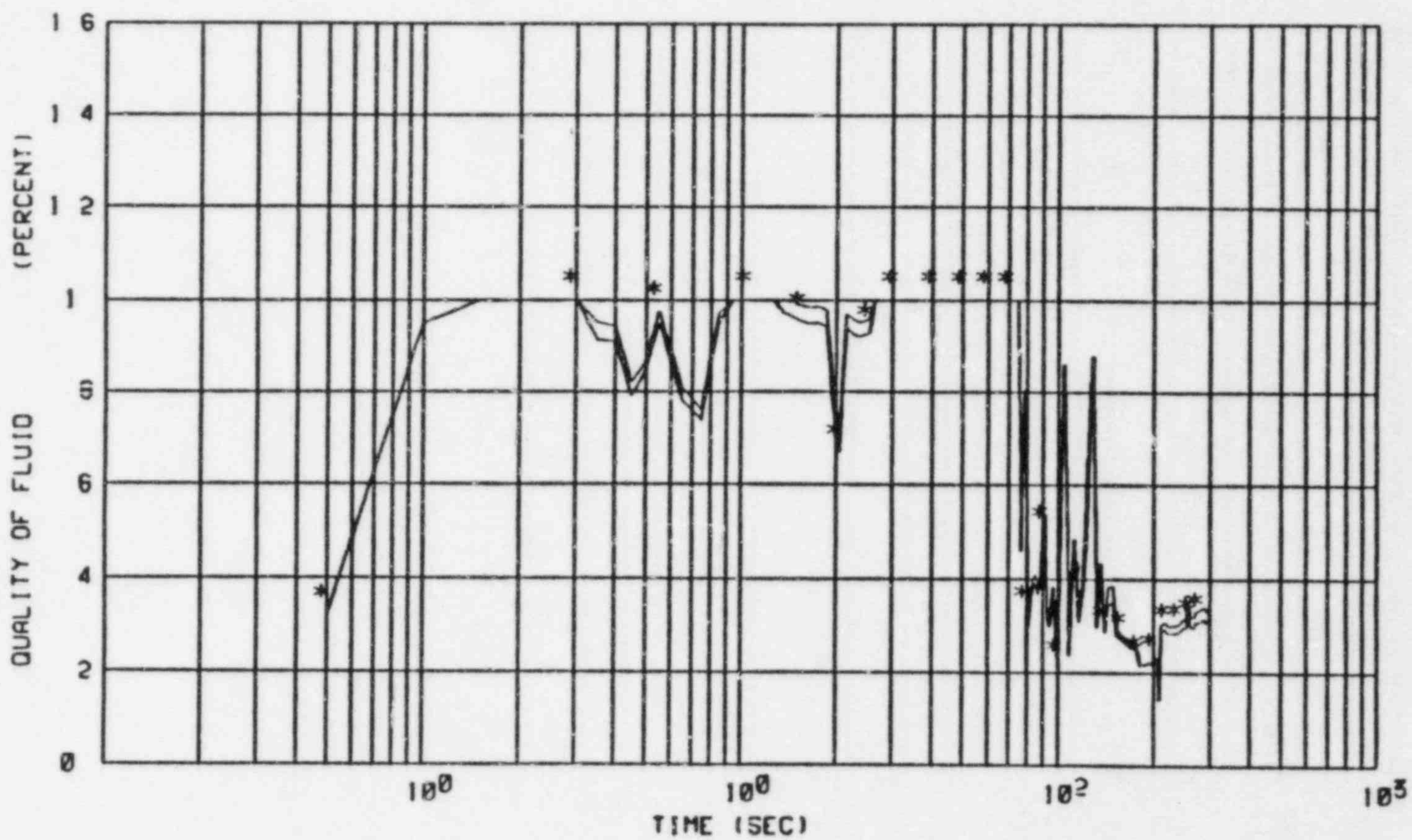

Figure 15.6.4-7: Fluid Quality, C_D=0.6 Max. SI DECLG

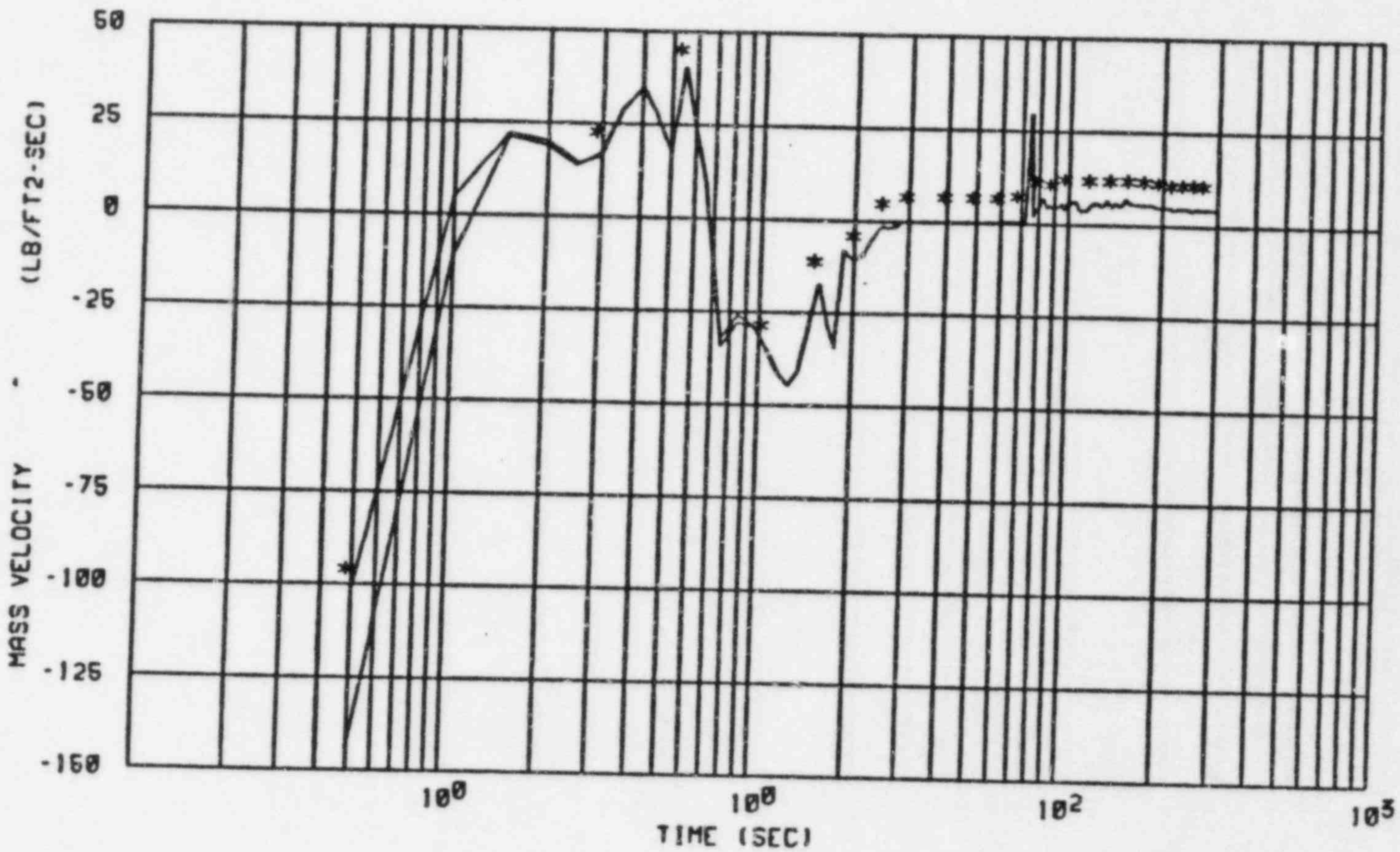

Figure 15.4.6-8: Mass Velocity, C_D=0.8 DECLG

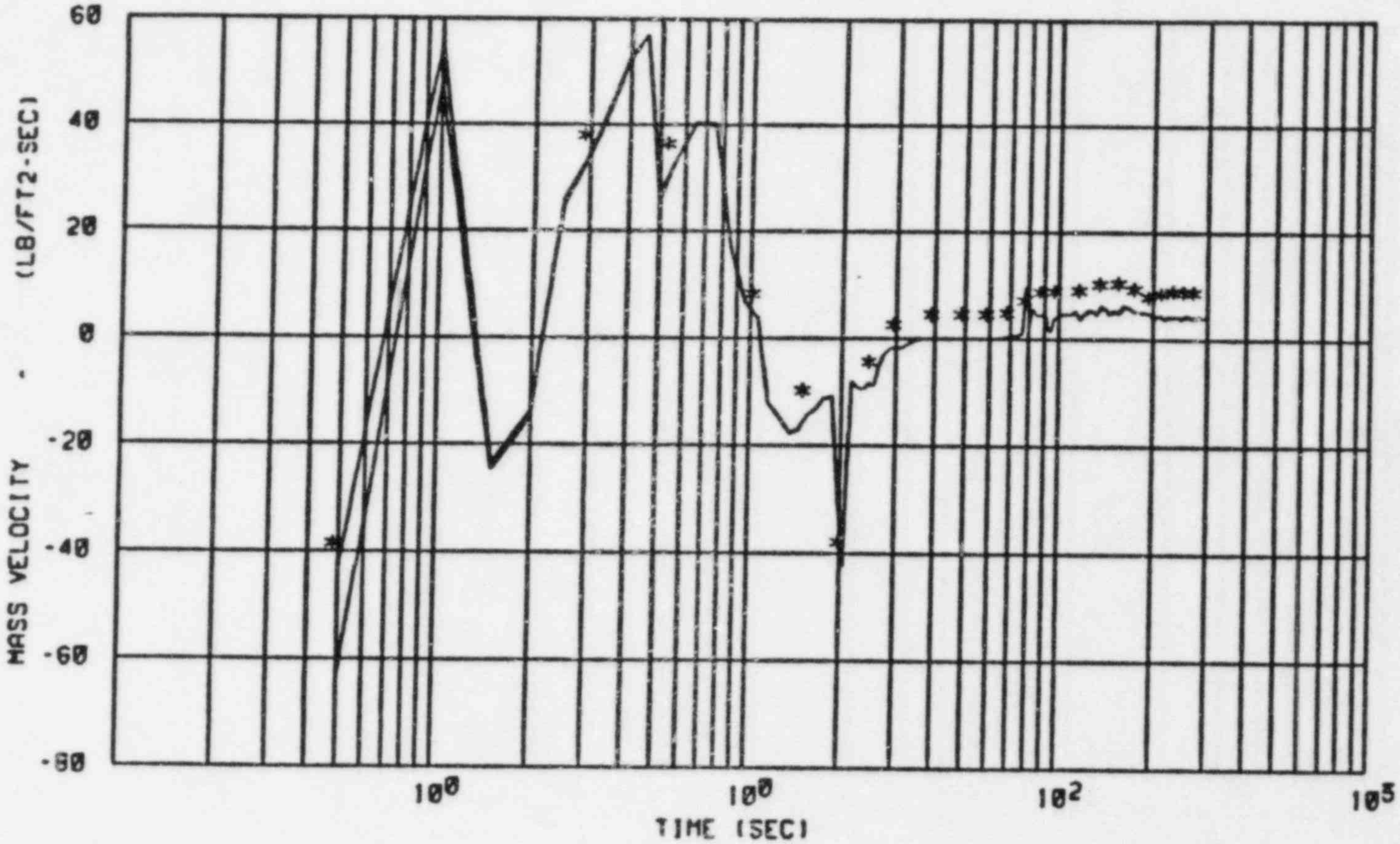

Figure 15.6.4-9: Mass Velocity, C_D=0.6 DECLG

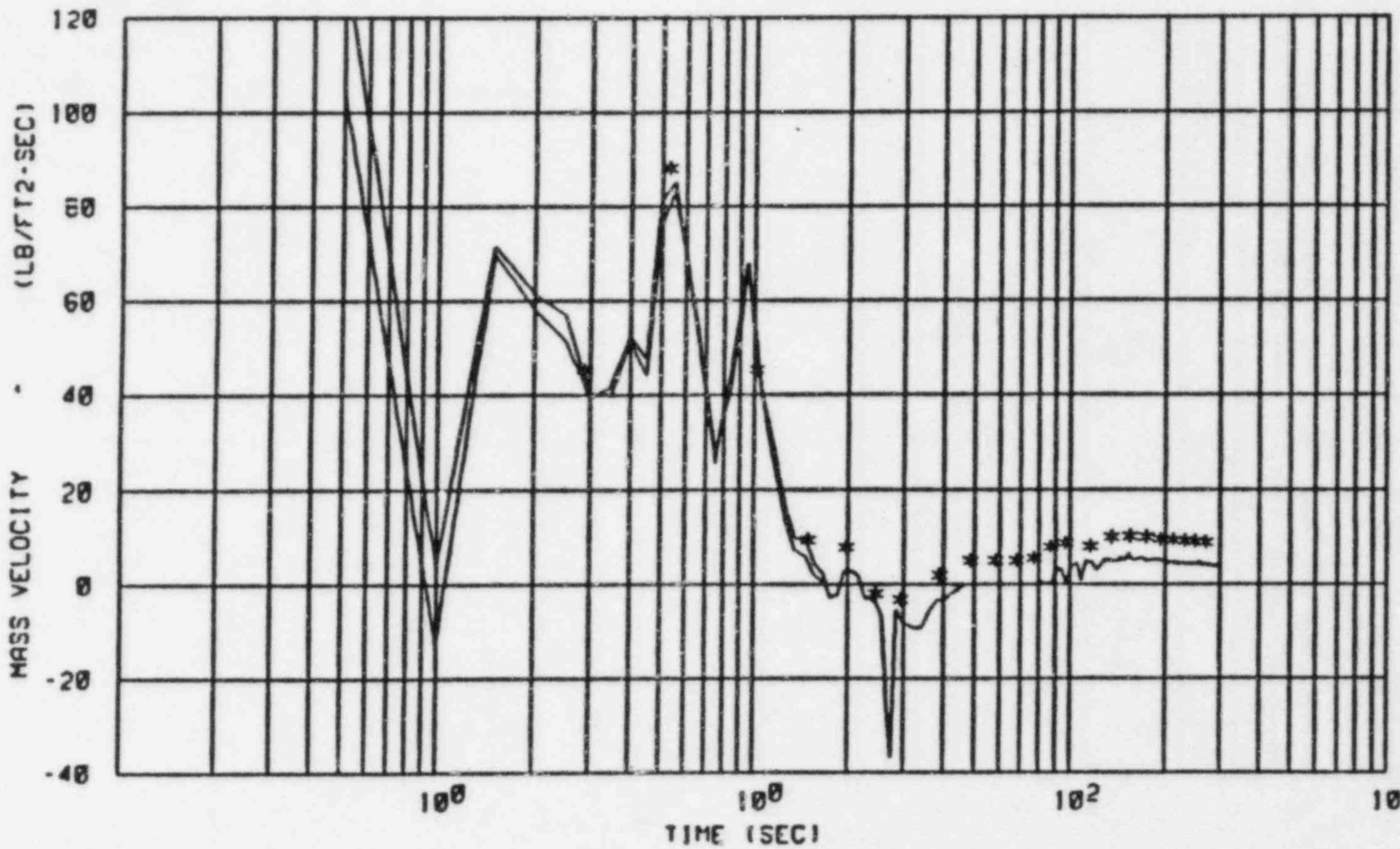

Figure 15.6.4-10: Mass Velocity, C_D=0.4 DECLG

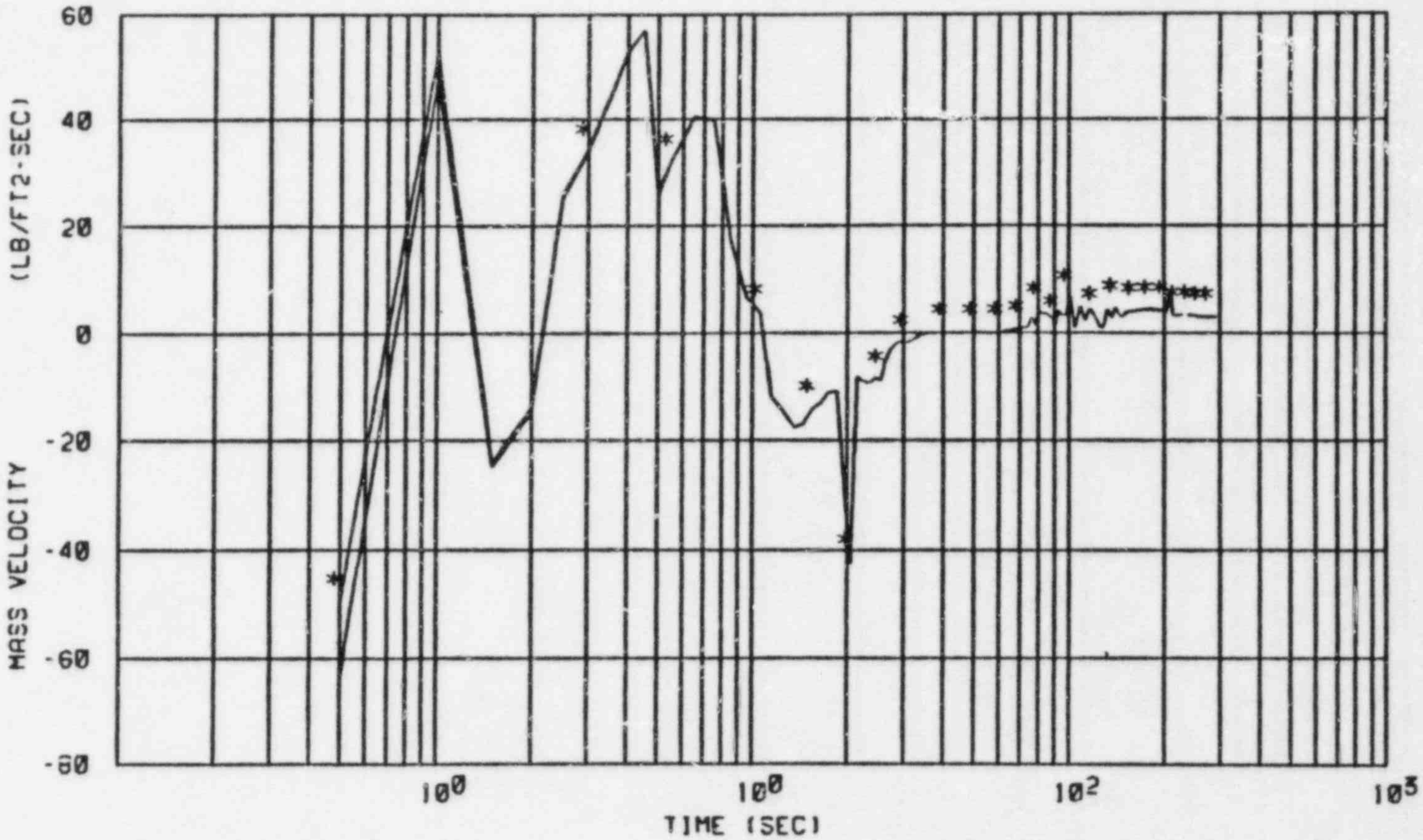

Figure 15.6.4-11: Mass Velocity, C_D=0.6 DECLG Max. SI

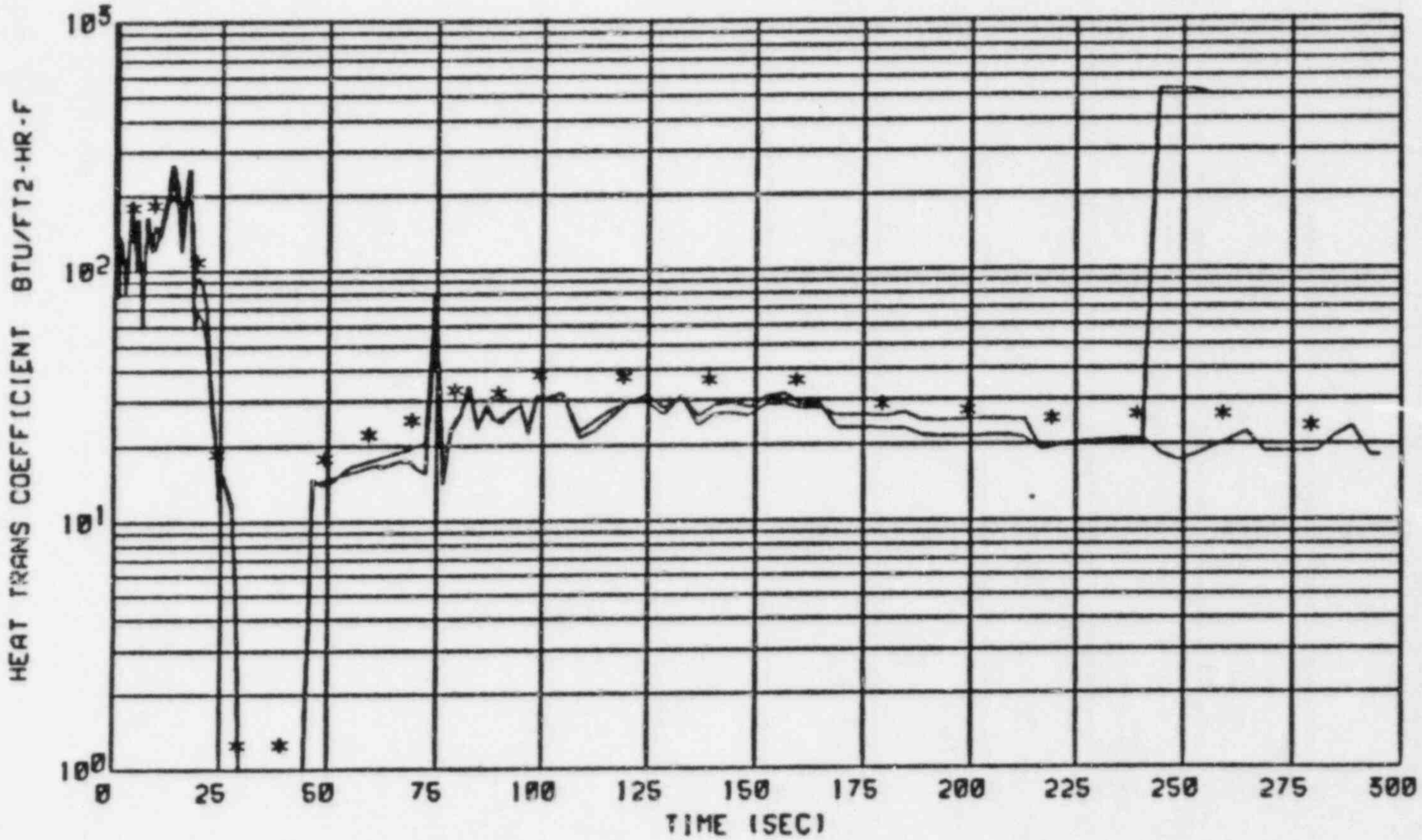

Figure 15.6.4-12 Heat Transfer Coefficient, C_D=0.8 DECLG

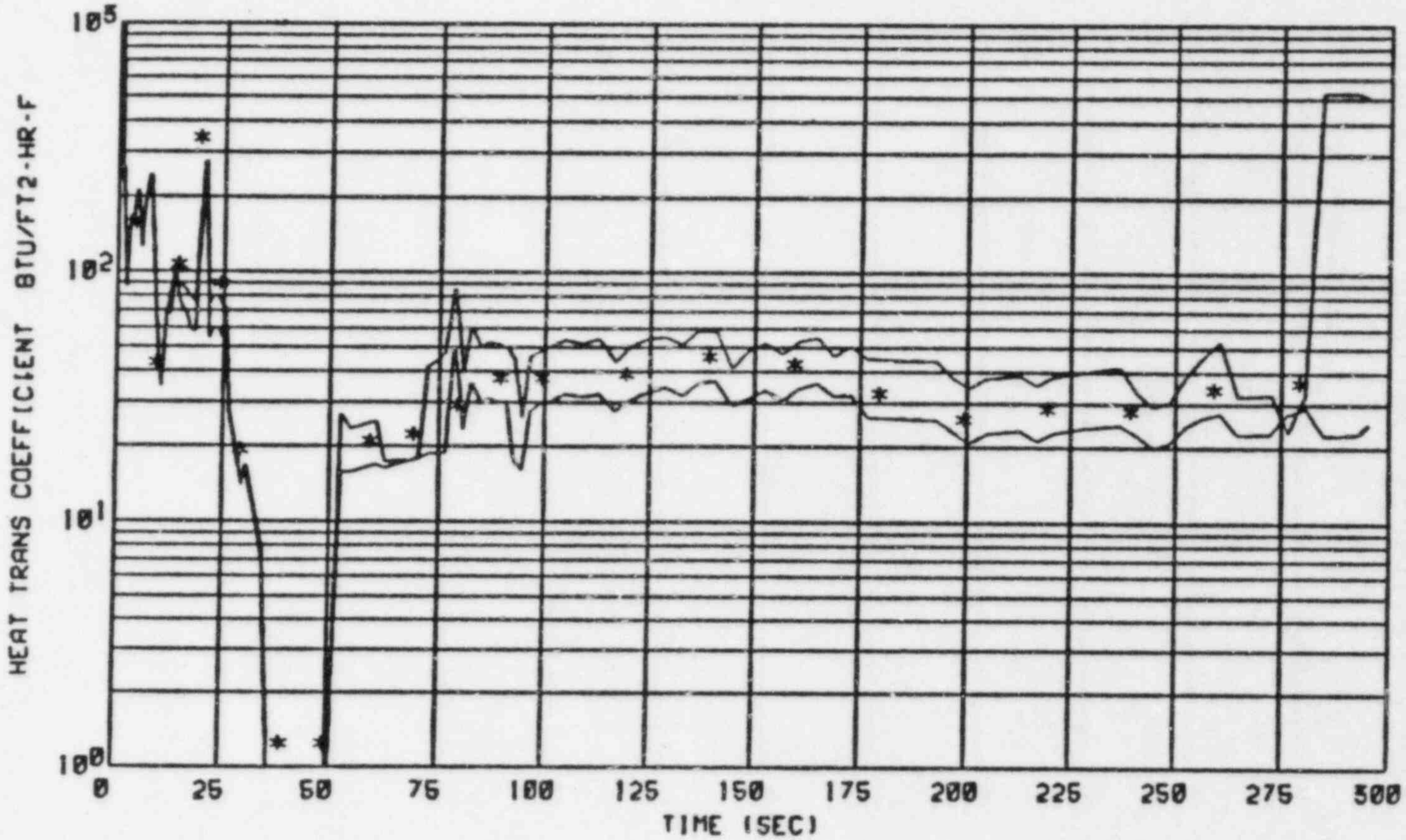

Figure 15.6.4-13: Heat Transfer Coefficient, C_D=0.6 DECLG

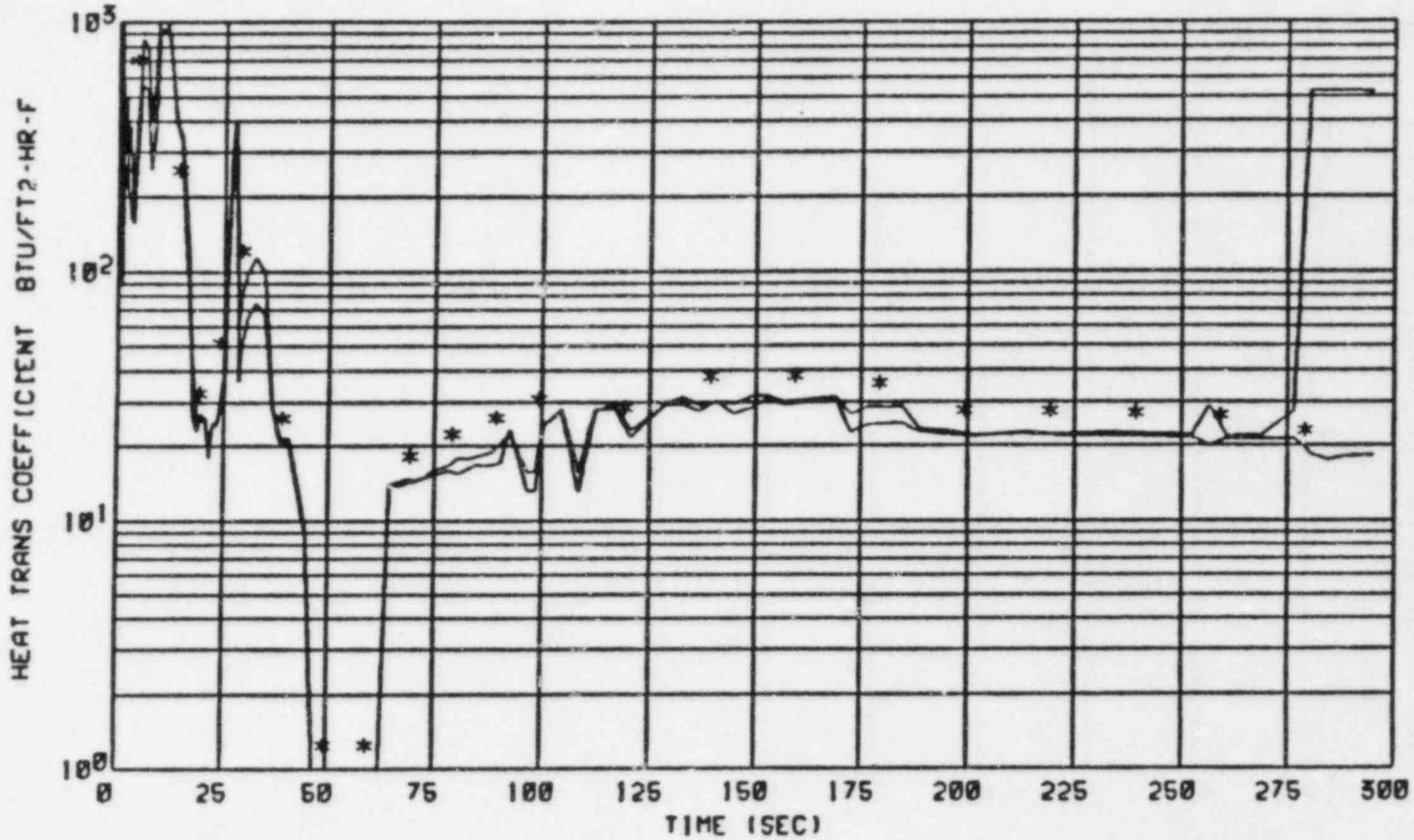

Figure 15.6.4-14: Heat Transfer Coefficient, C_D=0.4 DECLG

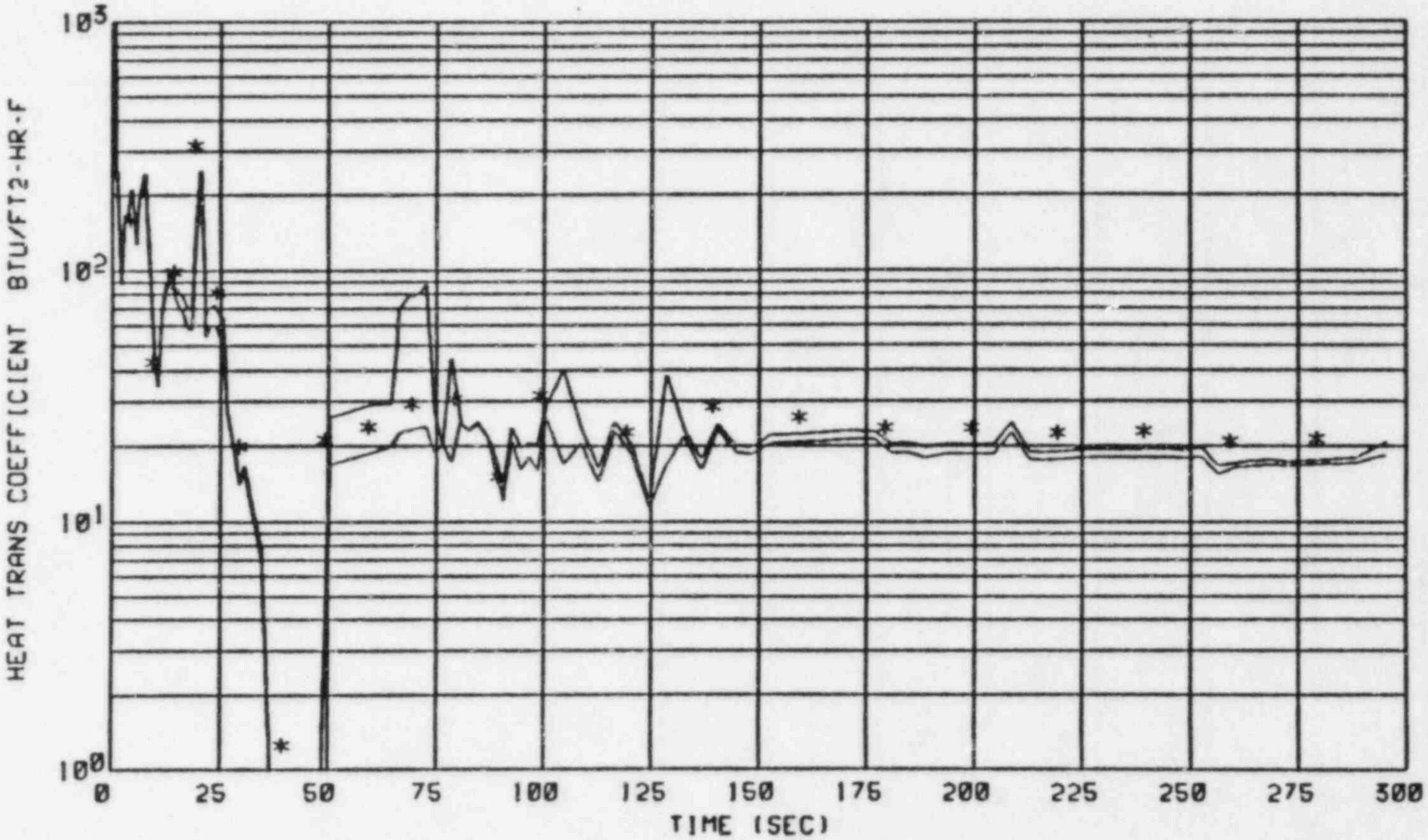

Figure 15.6.4-15: Heat Transfer Coefficient, C_D=0.6 DECLG, Max. SI

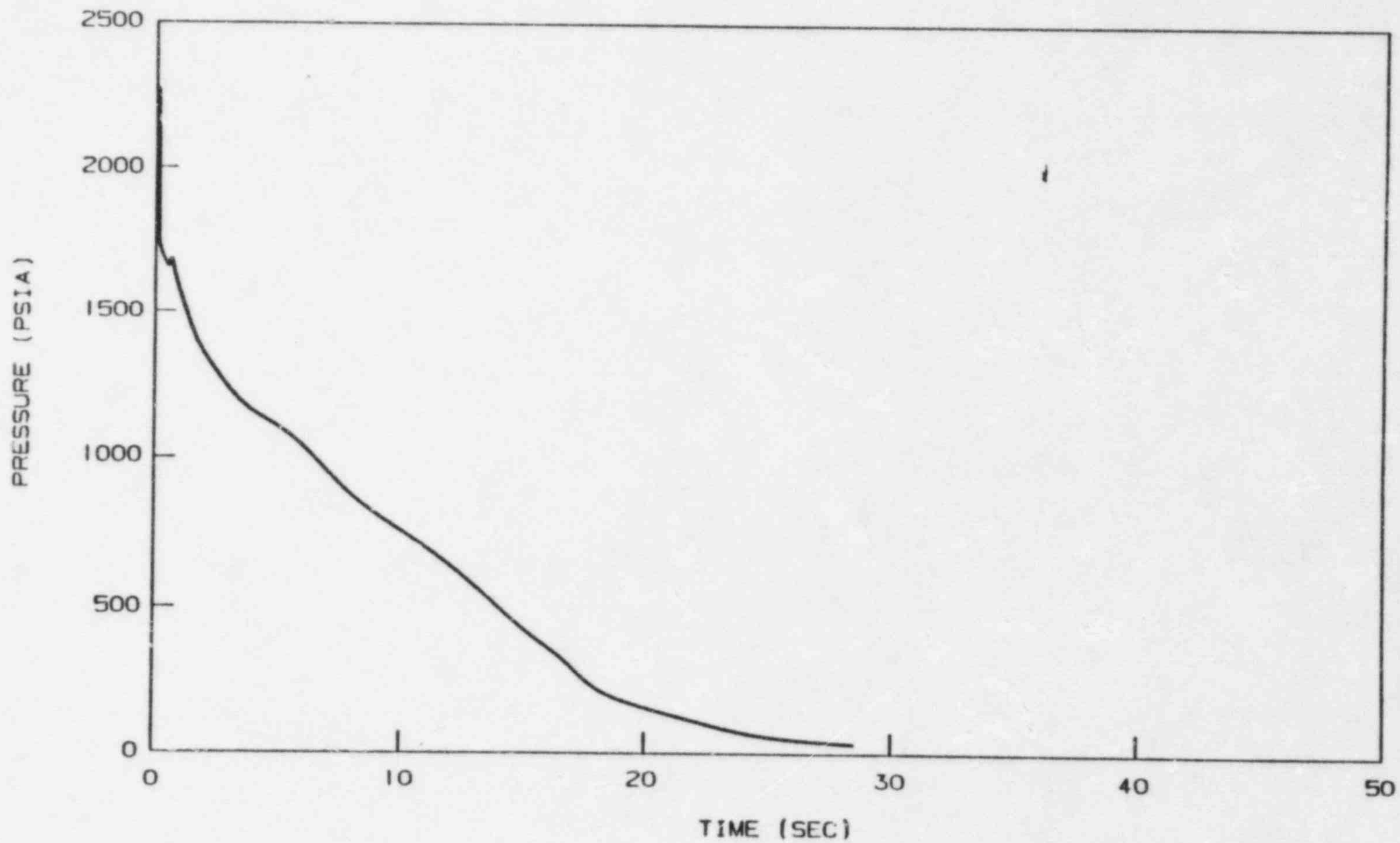

Figure 15.6.4-16: Core Pressure, C_D=0.8 DECLG

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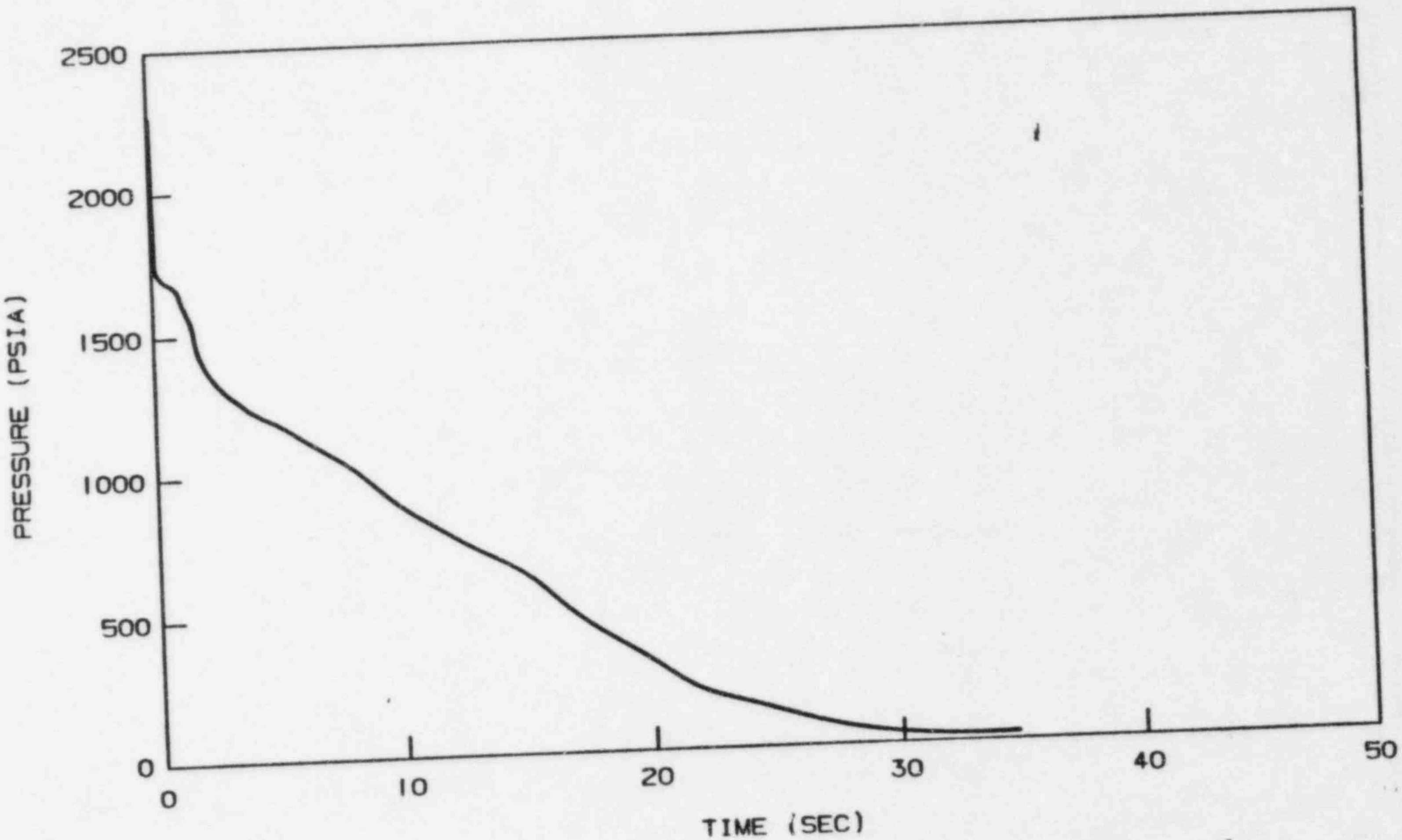

Figure 15.6.4-17: Core Pressure, C_D=0.6 DECLG

14715.8

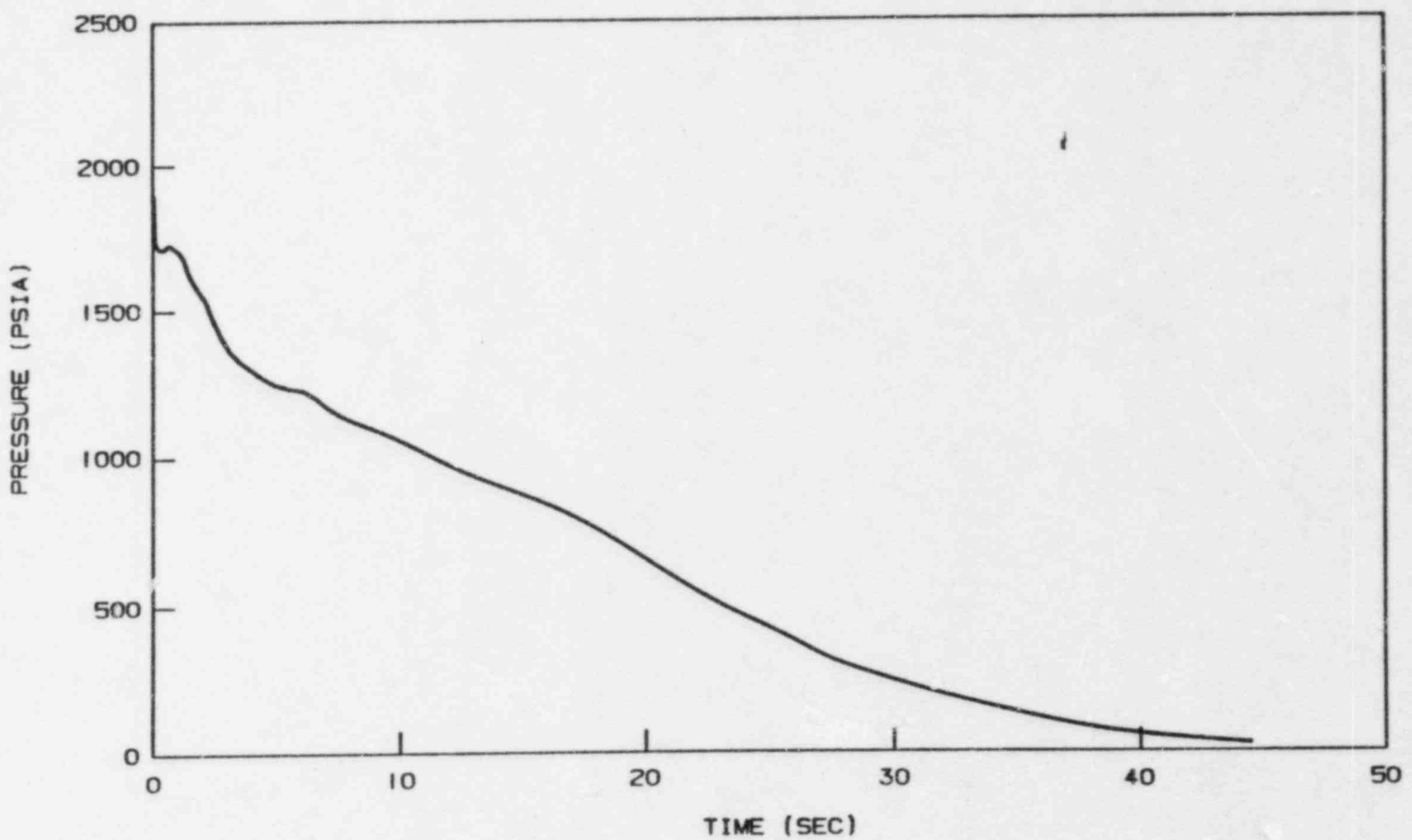

Figure 15.6.4-18: Core Pressure, C_D=0.4 DECLG

14715.9

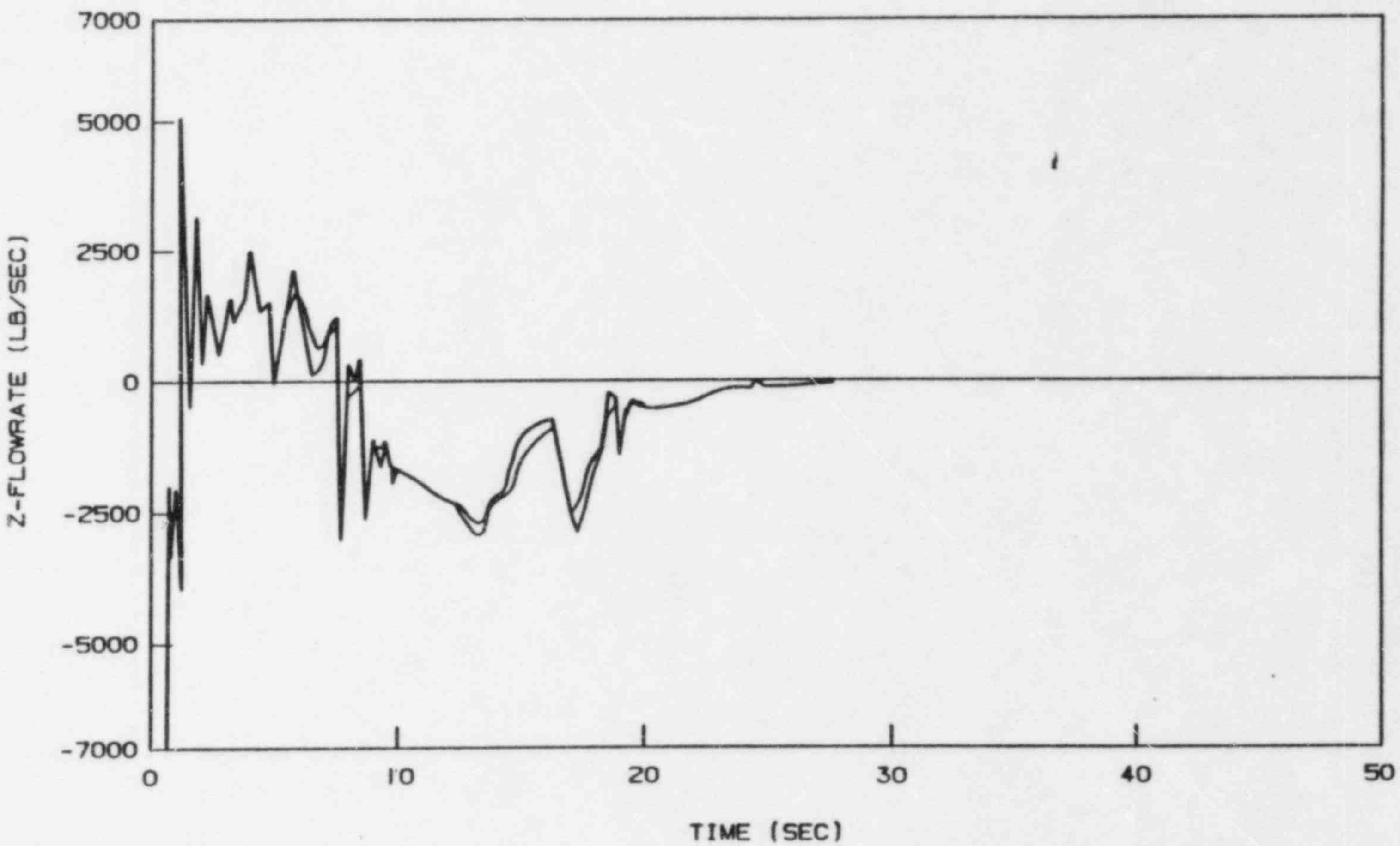

Figure 15.6.4-19: Core Flowrate, C_D=0.8 DECLG

14715.10

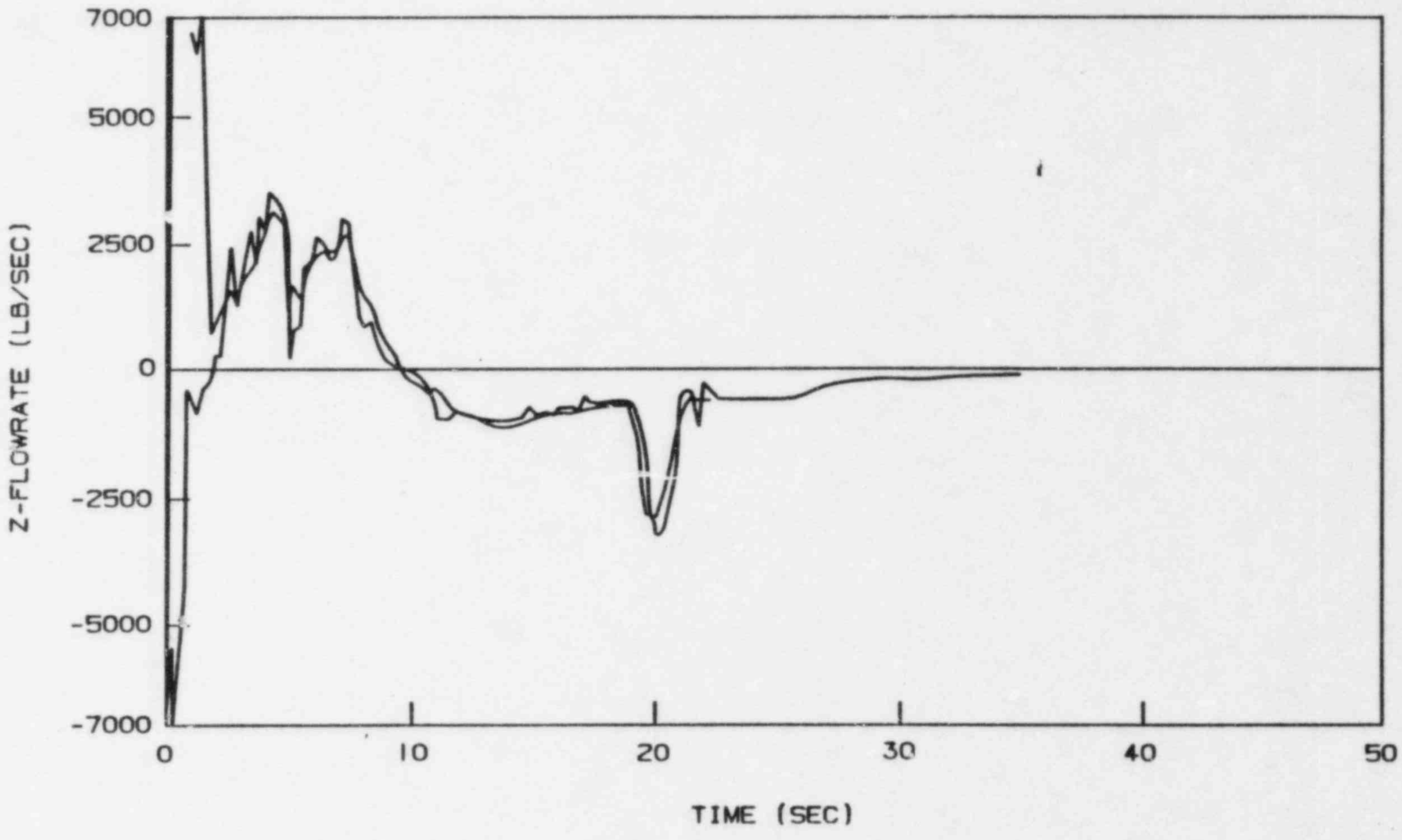

Figure 15.6.4-20: Core Flowrate, C_D=0.6 DECLG

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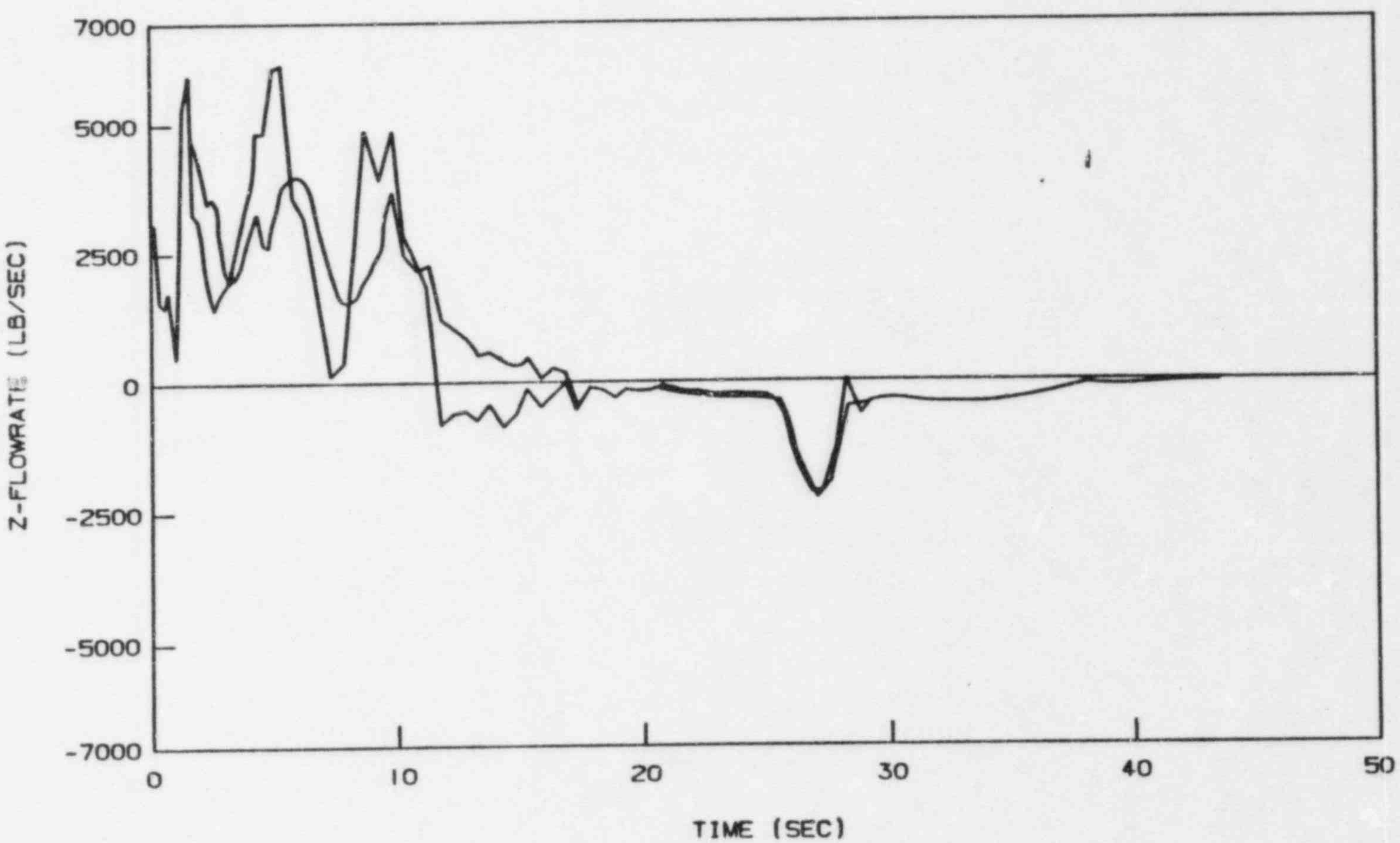

Figure 15.6.4-21: Core Flowrate, C_D=0.4 DECLG

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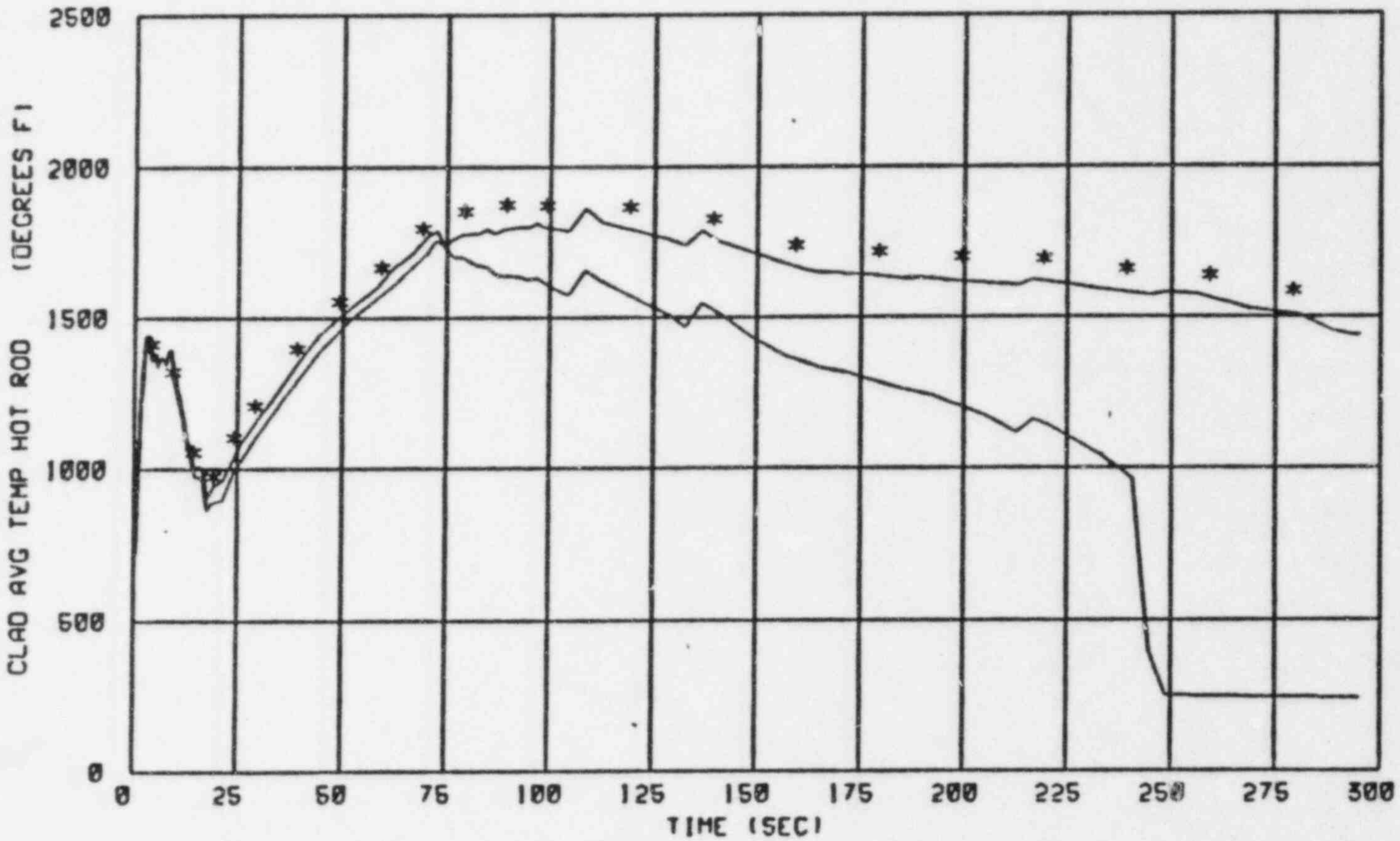


Figure 15.6.4-22: Peak Clad Temperature. C_D=0.8 DECLG

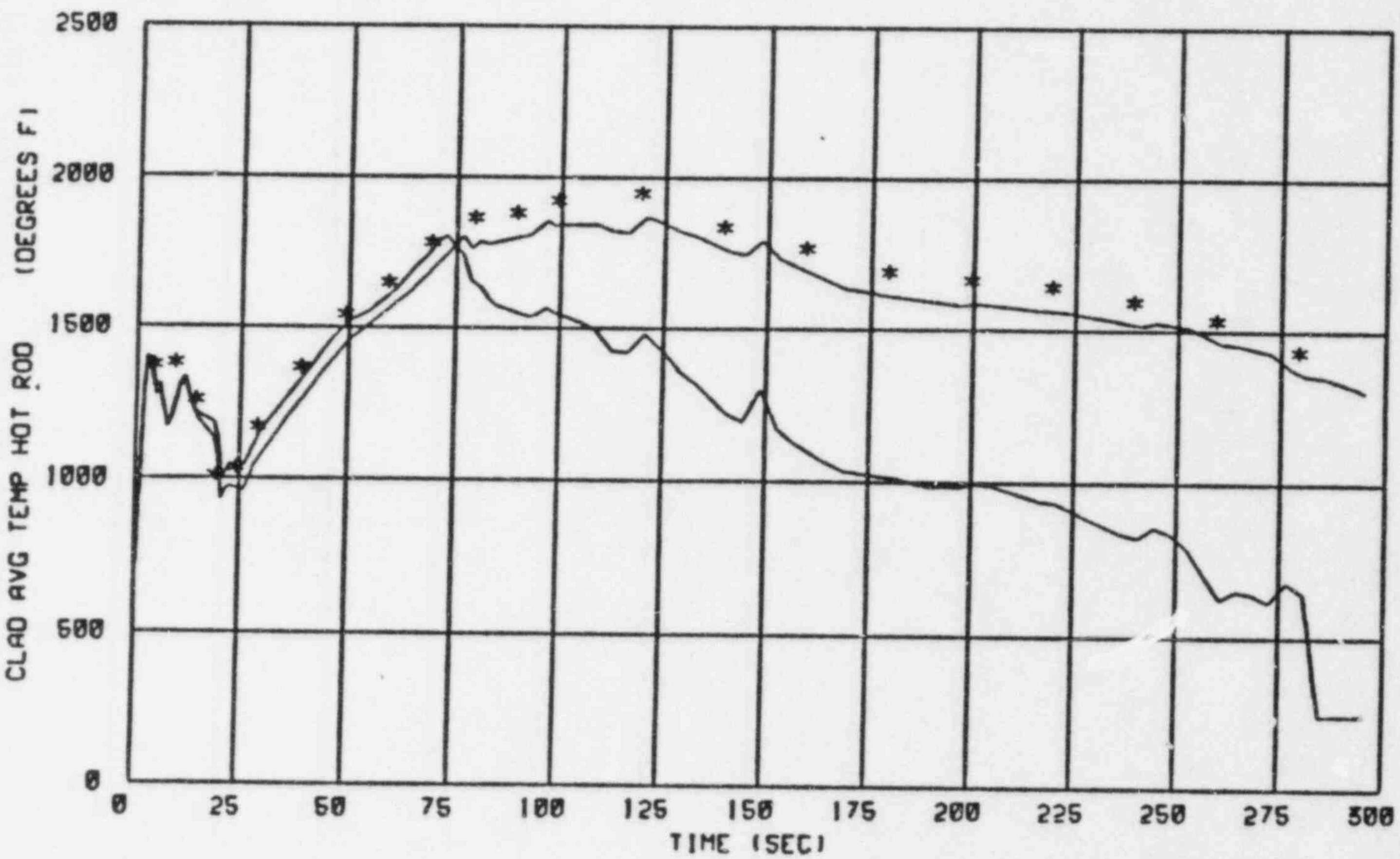


Figure 15.6.4-23: Peak Clad Temperature, C_D=0.6 DECLG

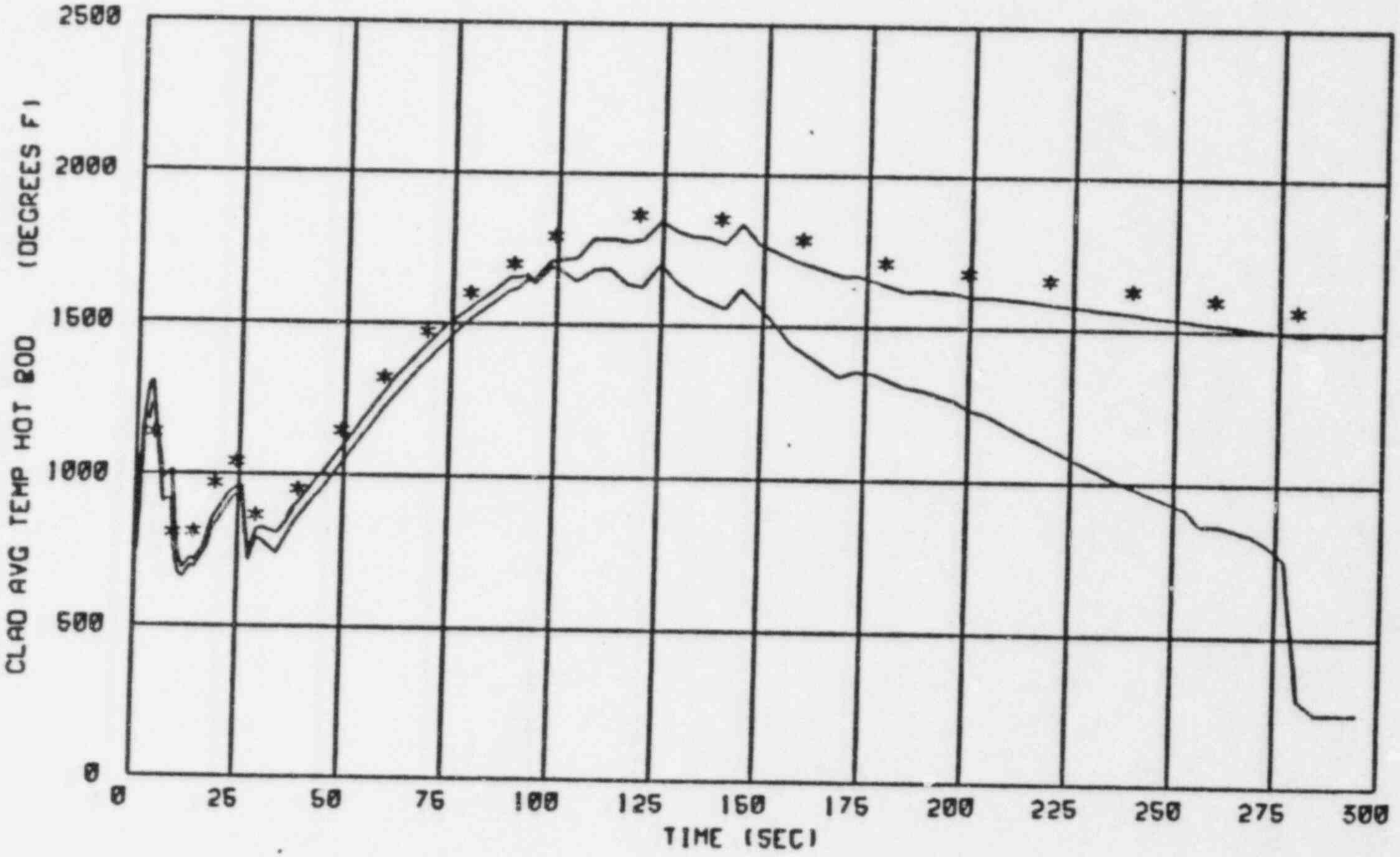


Figure 15.6.4-24: Peak Clad Temperature, C_D=0.4 DECLG

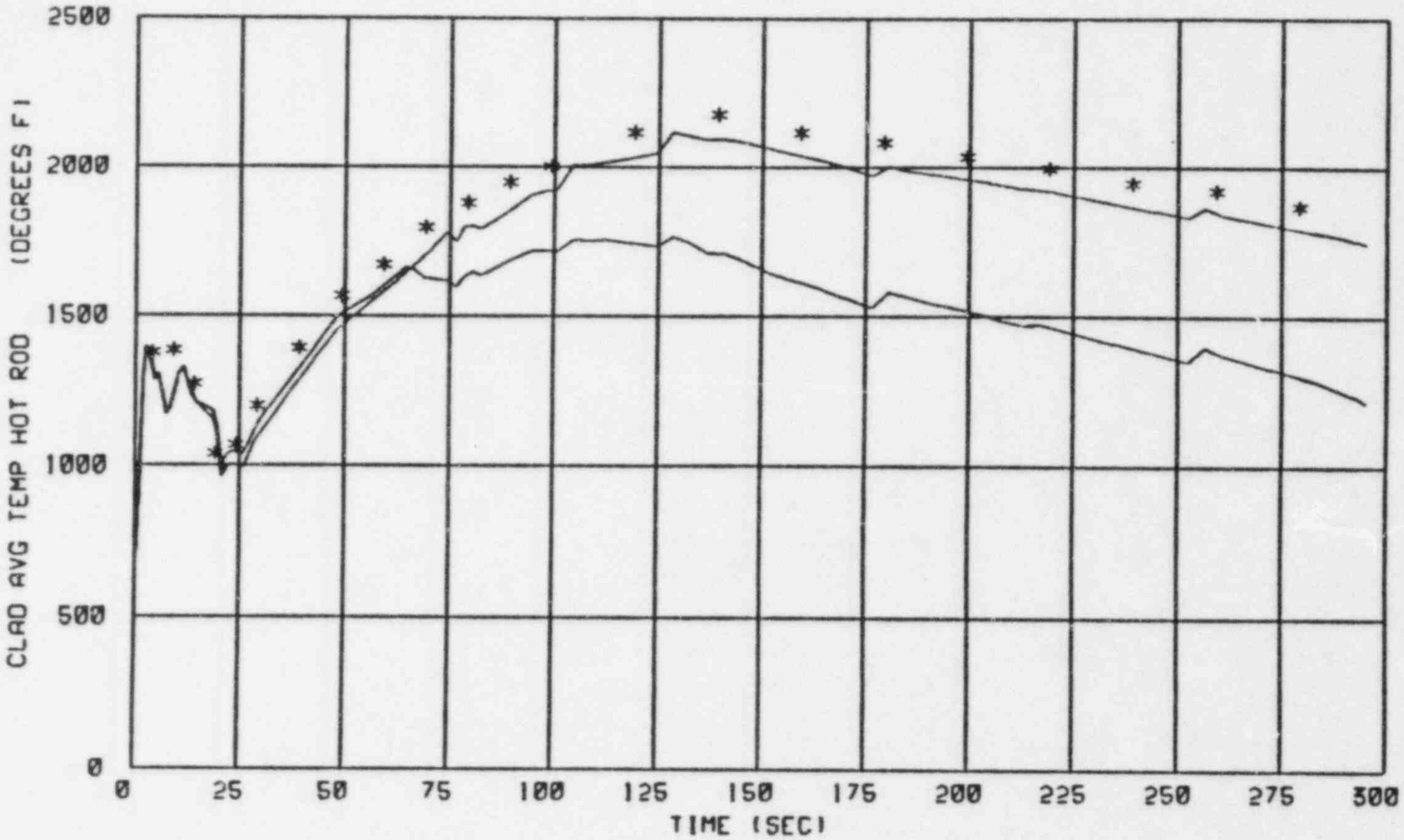


Figure 15.6.4-25: Peak Clad Temperature, C_D=0.6 DECLG; Max. SI

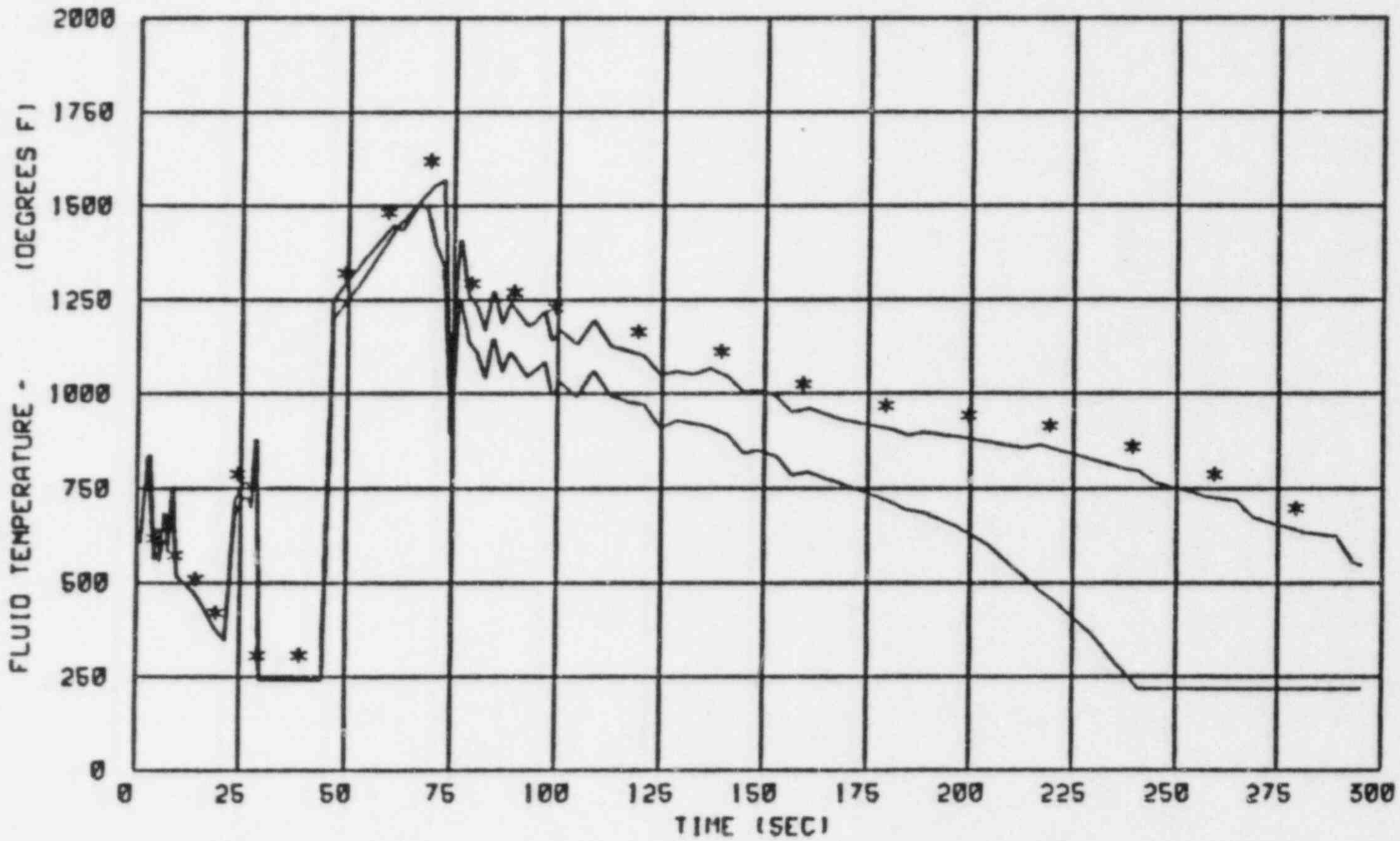


Figure 15.6.4-26: Fluid Temperature, C_D=0.8 DECLG

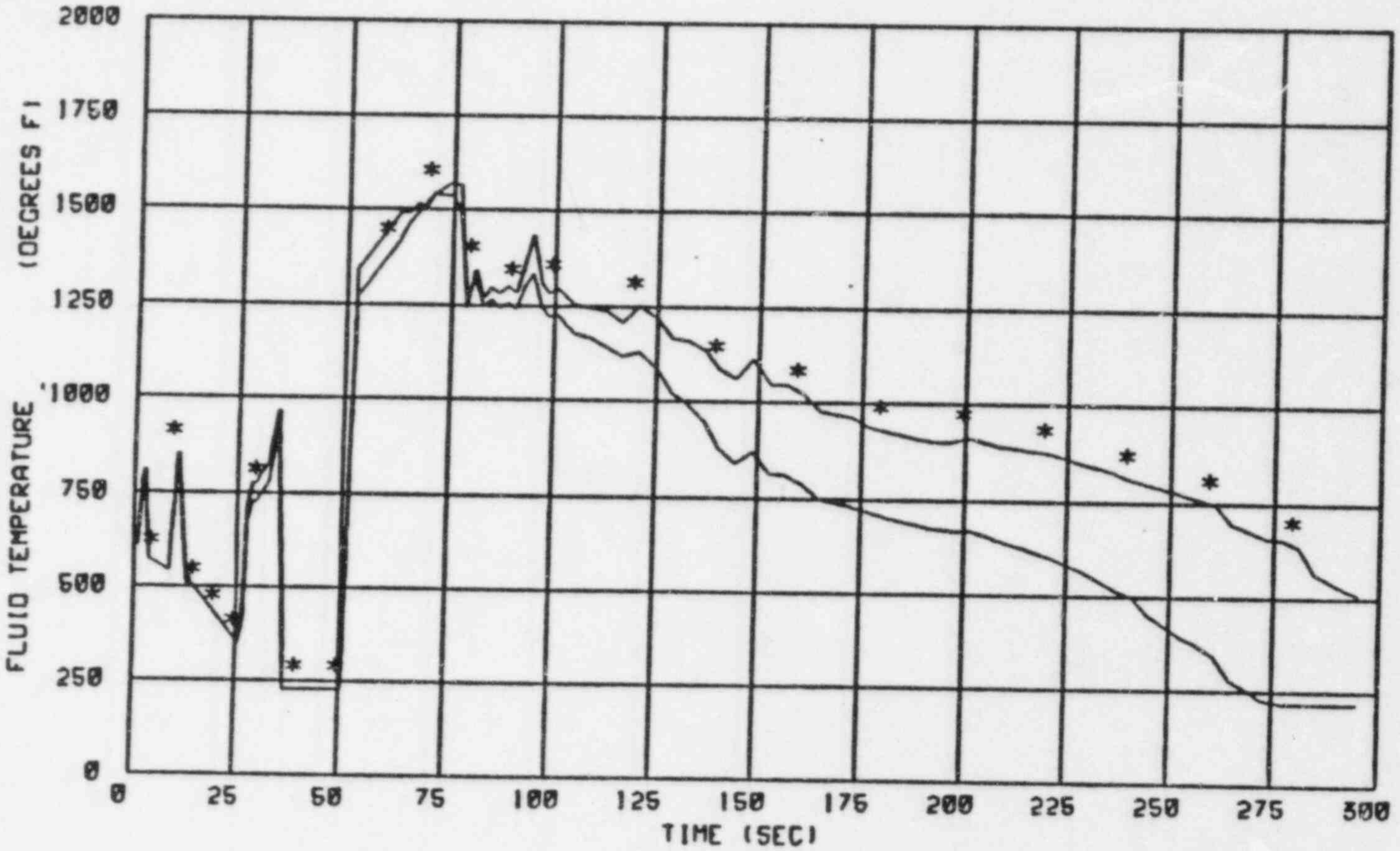


Figure 15.6.4-27: Fluid Temperature, C_D=0.6 DECLG

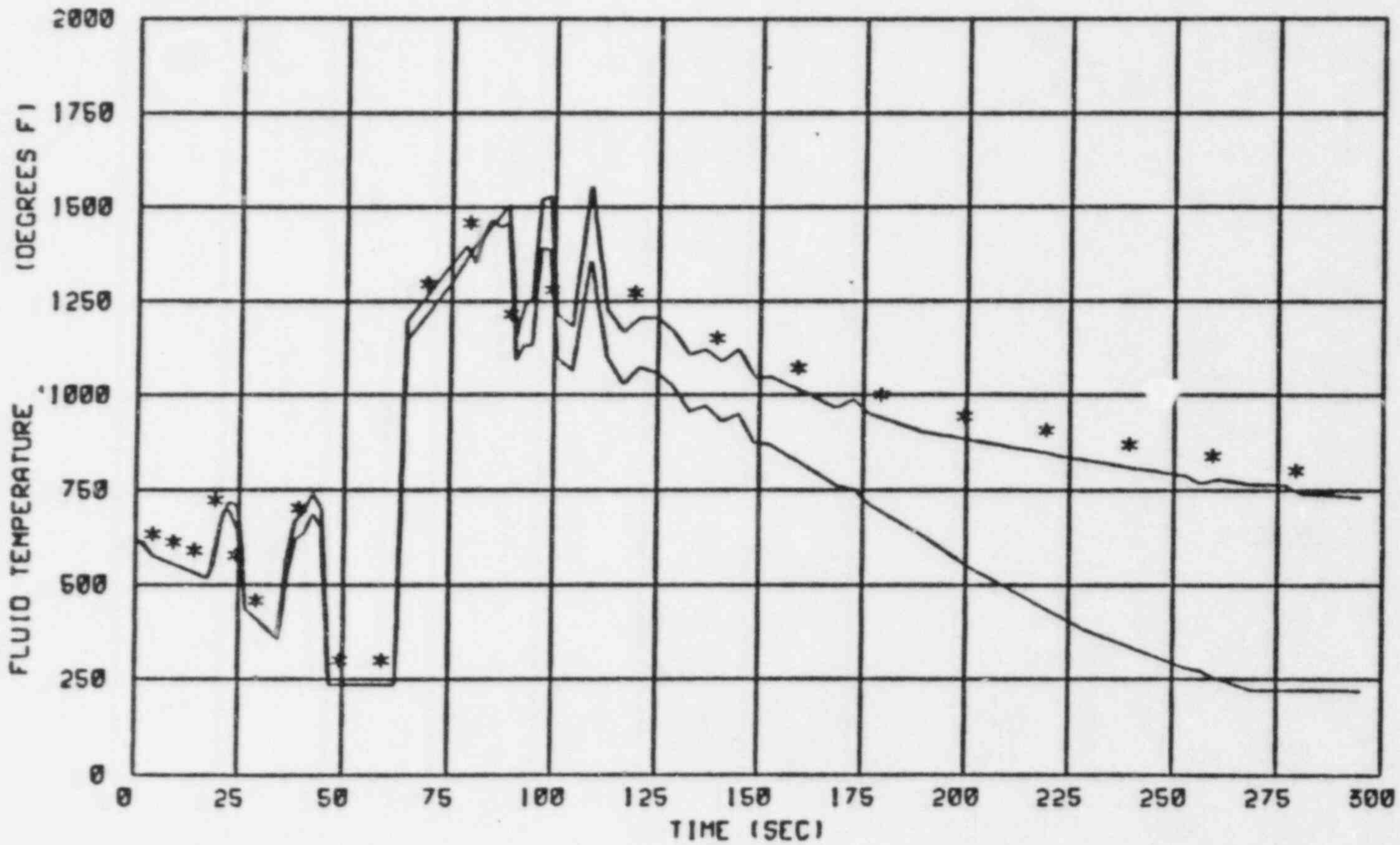


Figure 15.6.4-28: Fluid Temperature, C_D=0.4 DECLG

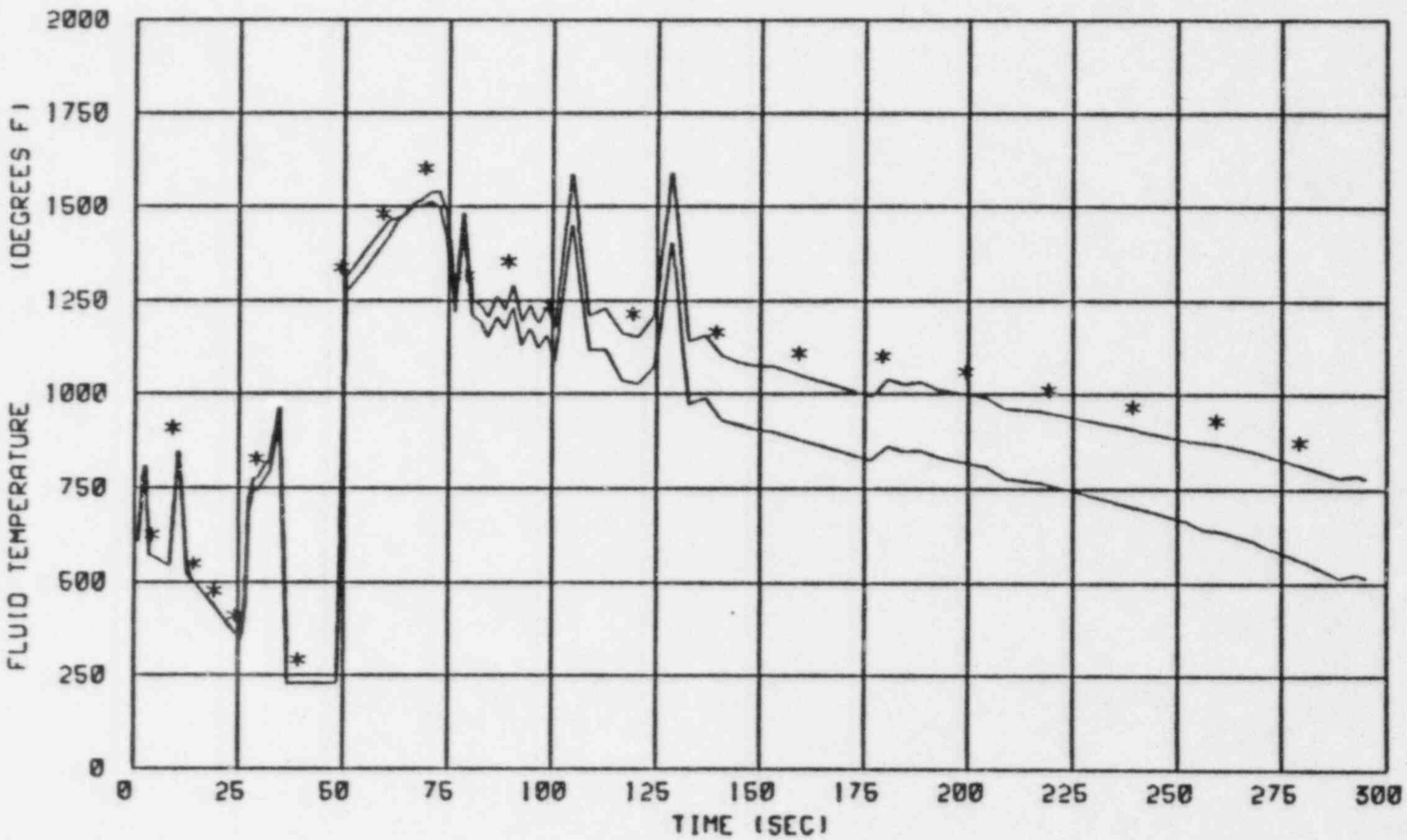


Figure 15.6.4-29: Fluid Temperature, C_D=0.6 DECLG, Max. SI

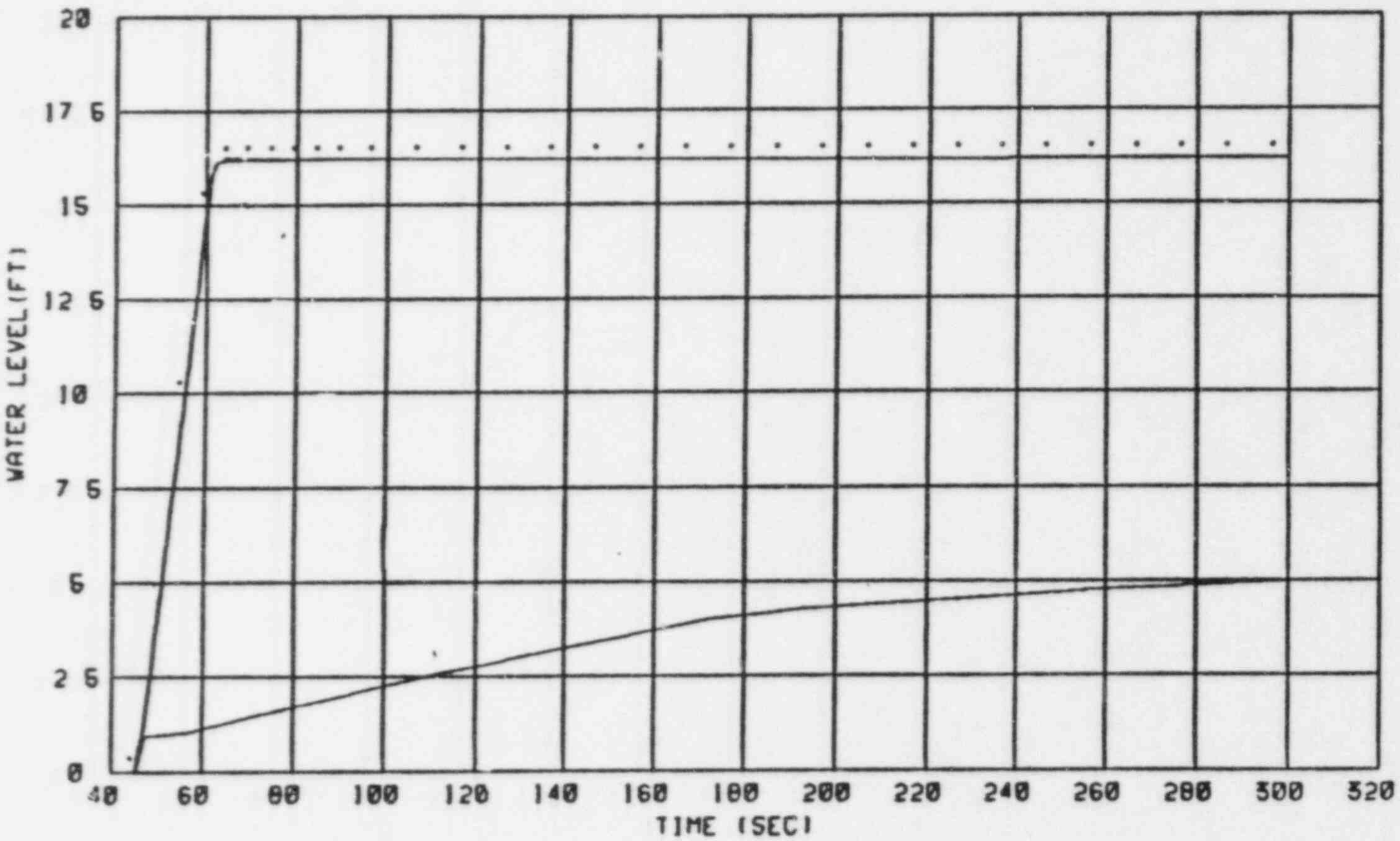


Figure 15.6.4-30: Reflood ~~Mixture~~ Liquid Levels, C_D=0.8 DECLG

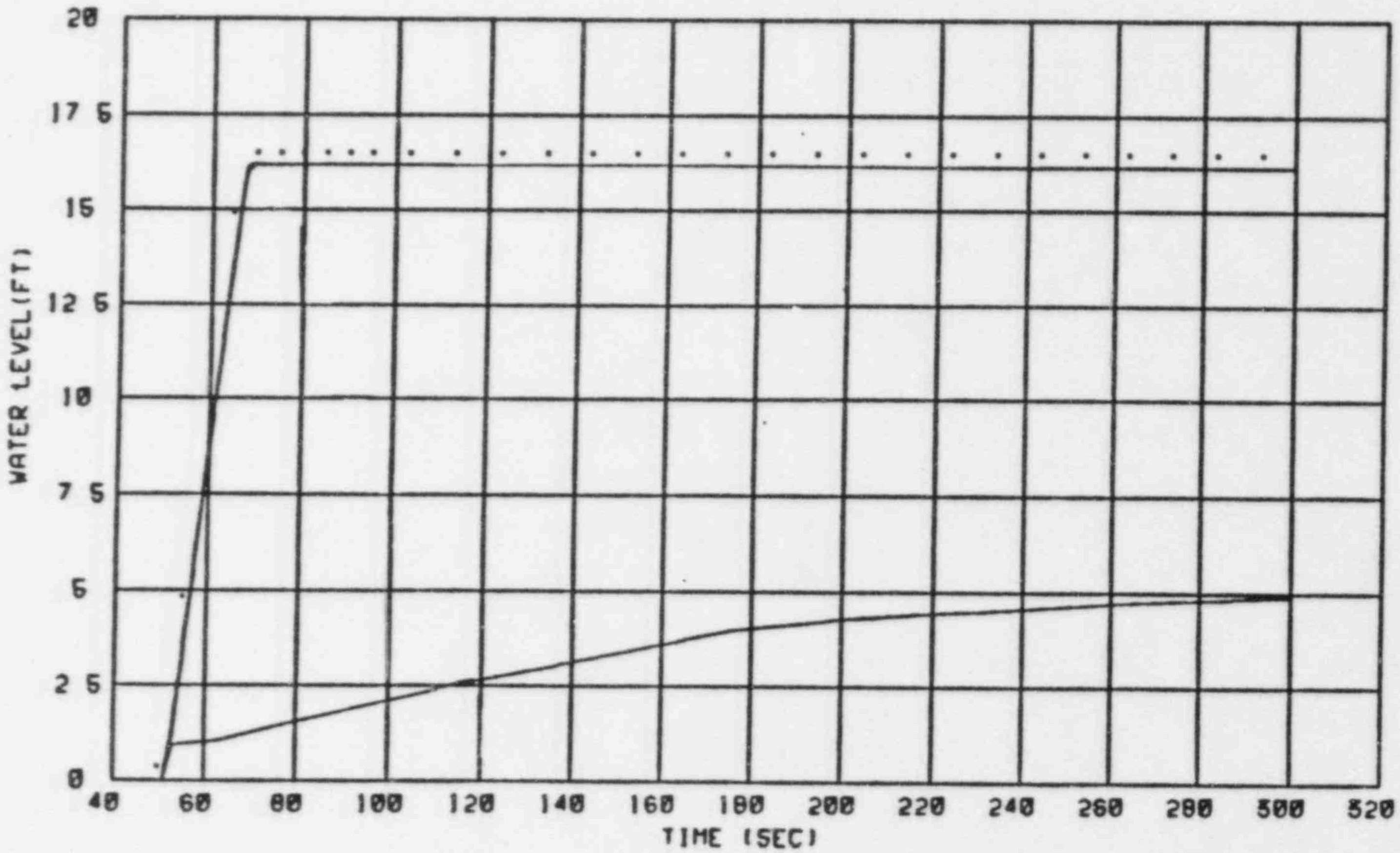


Figure 15.6.4-31: Reflood ~~Mixture~~ Levels, C_D=0.6 DECLG

Liquid

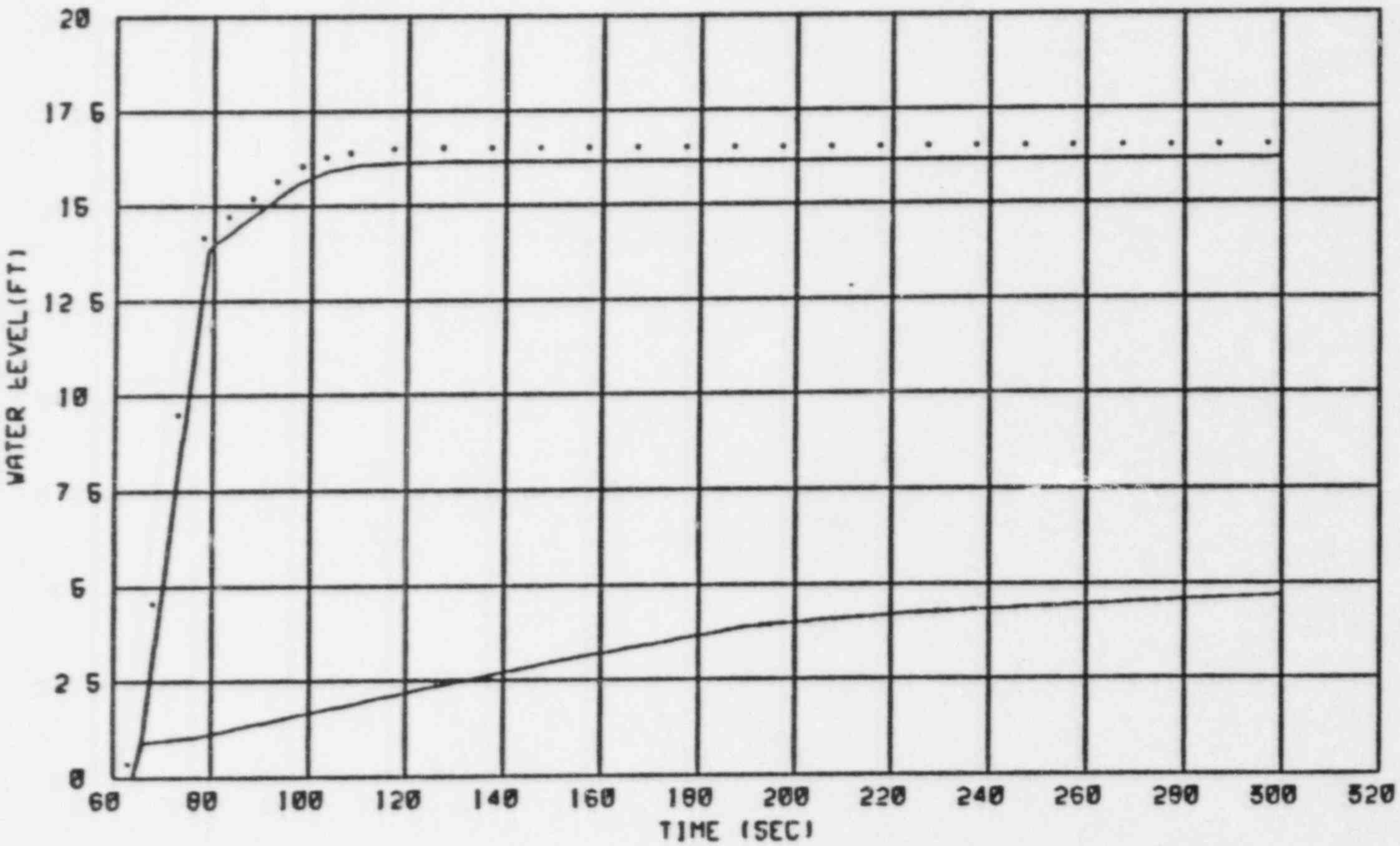


Figure 15.6.4-32: Reflood Mixture Levels, C_D=0.4 DECLG

Liquid

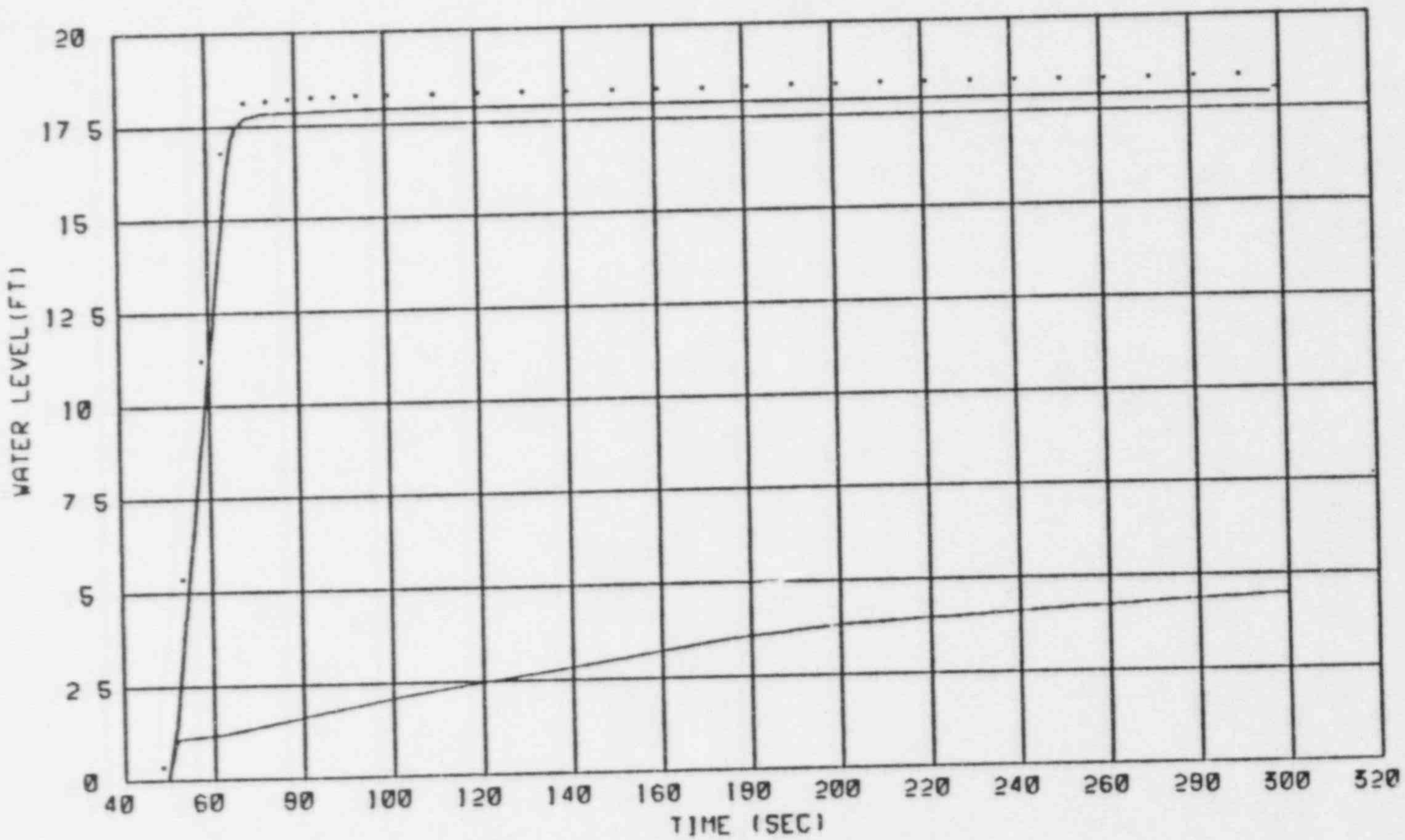


Figure 15.6.4-33: Reflood Mixture Levels, C_D=0.6 DECLG Max. SI

Liquid

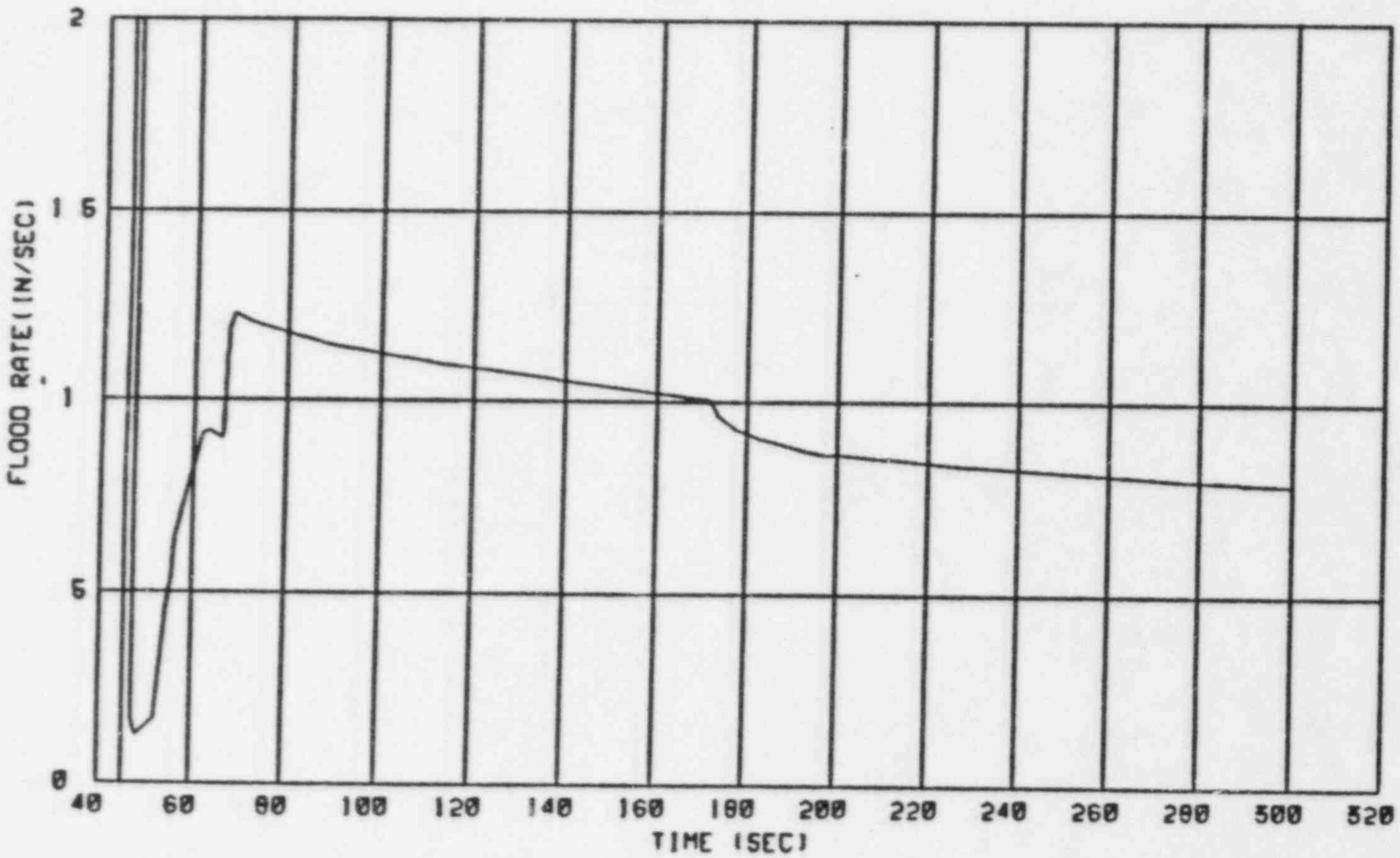


Figure 15.6.4-34: Core Inlet Velocity, C_D=0.8 DECLG

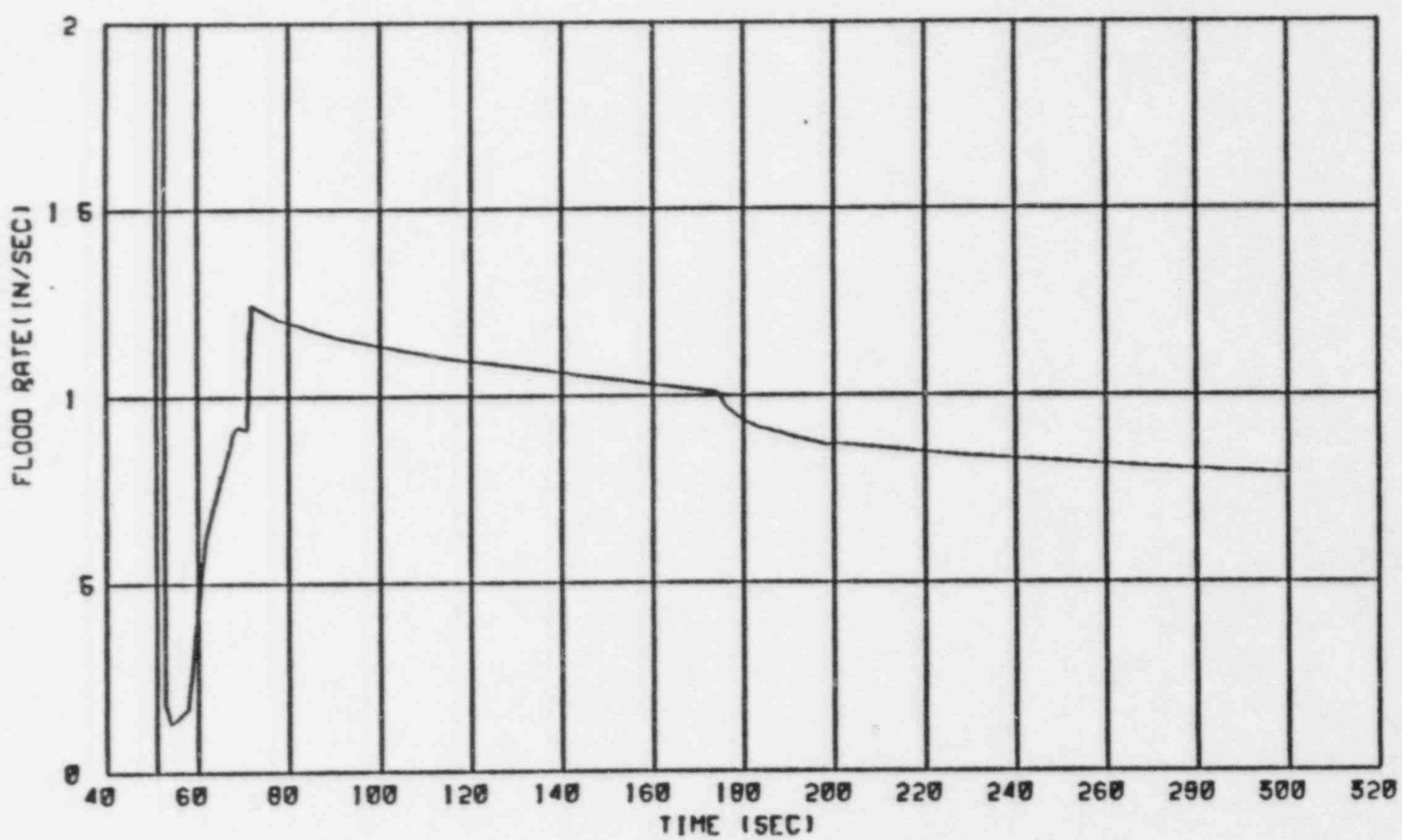


Figure 15.6.4-35: Core Inlet Velocity, C_D=0.6 DECLG

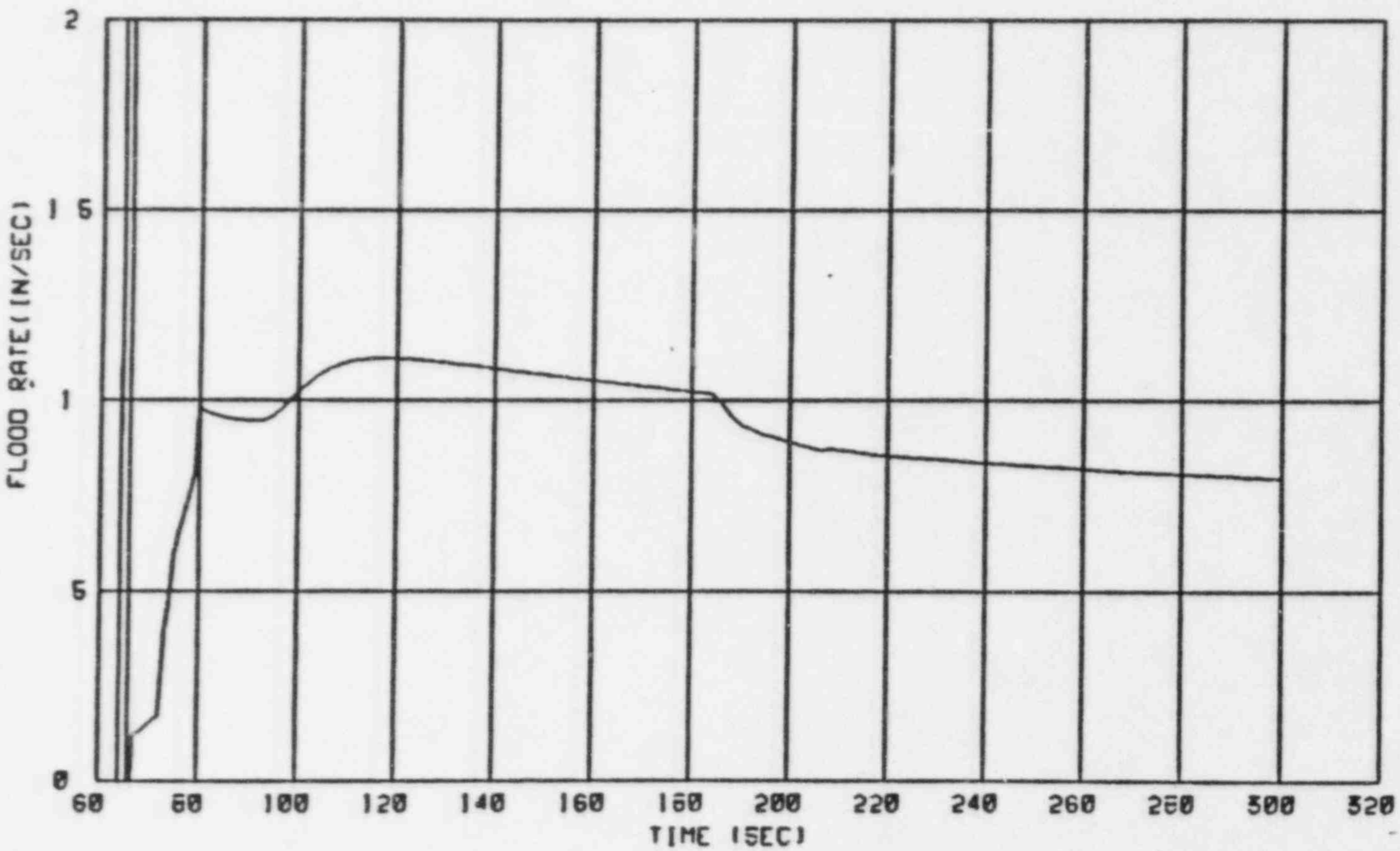


Figure 15.6.4-36: Core Inlet Velocity, C_D=0.4 DECLG

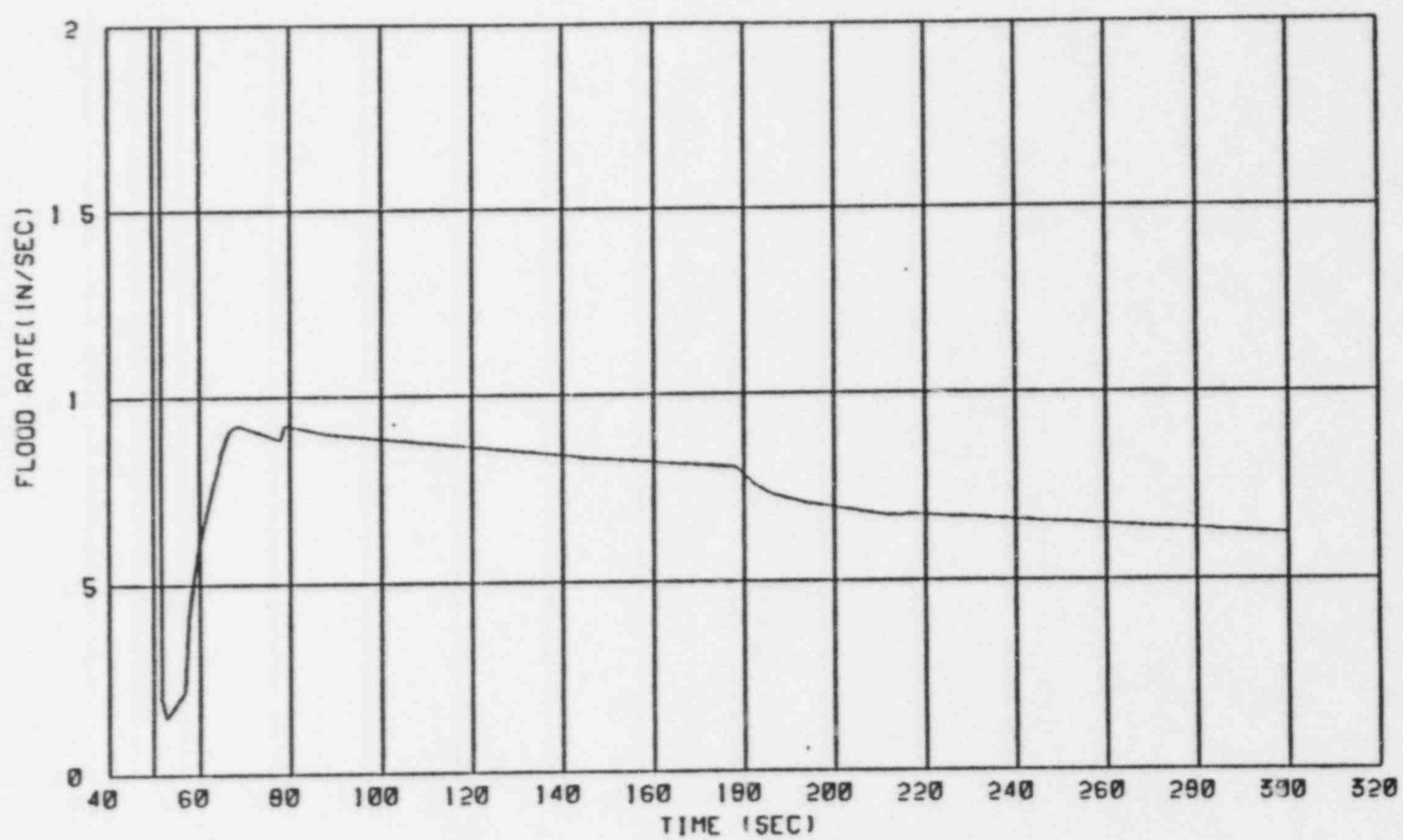


Figure 15.6.4-37: Core Inlet Velocity, C_D=0.6 DECLG; Max. SI

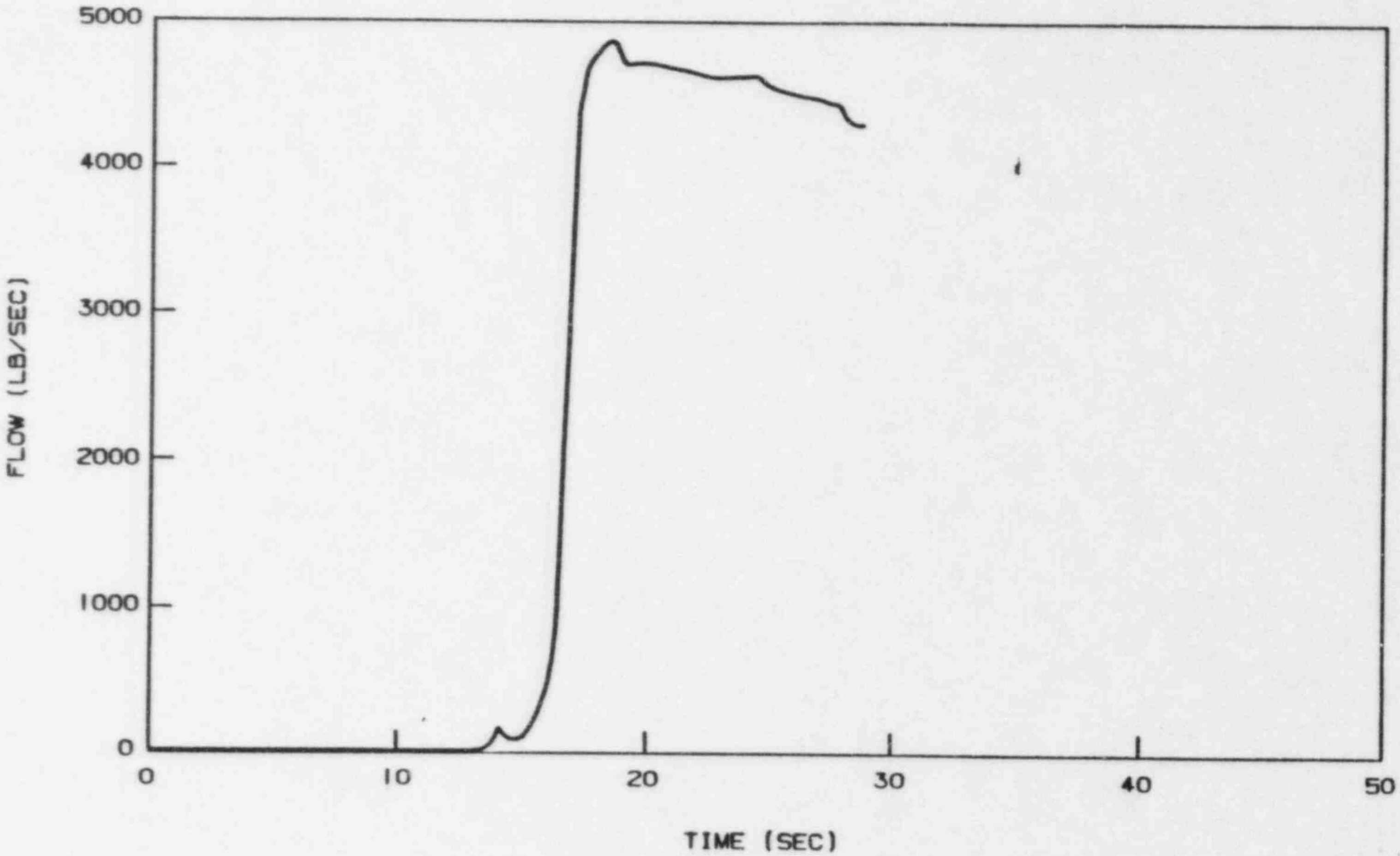


Figure 15.6.4-38: Accumulator Injection, C_D=0.8 DECLG

14715.21

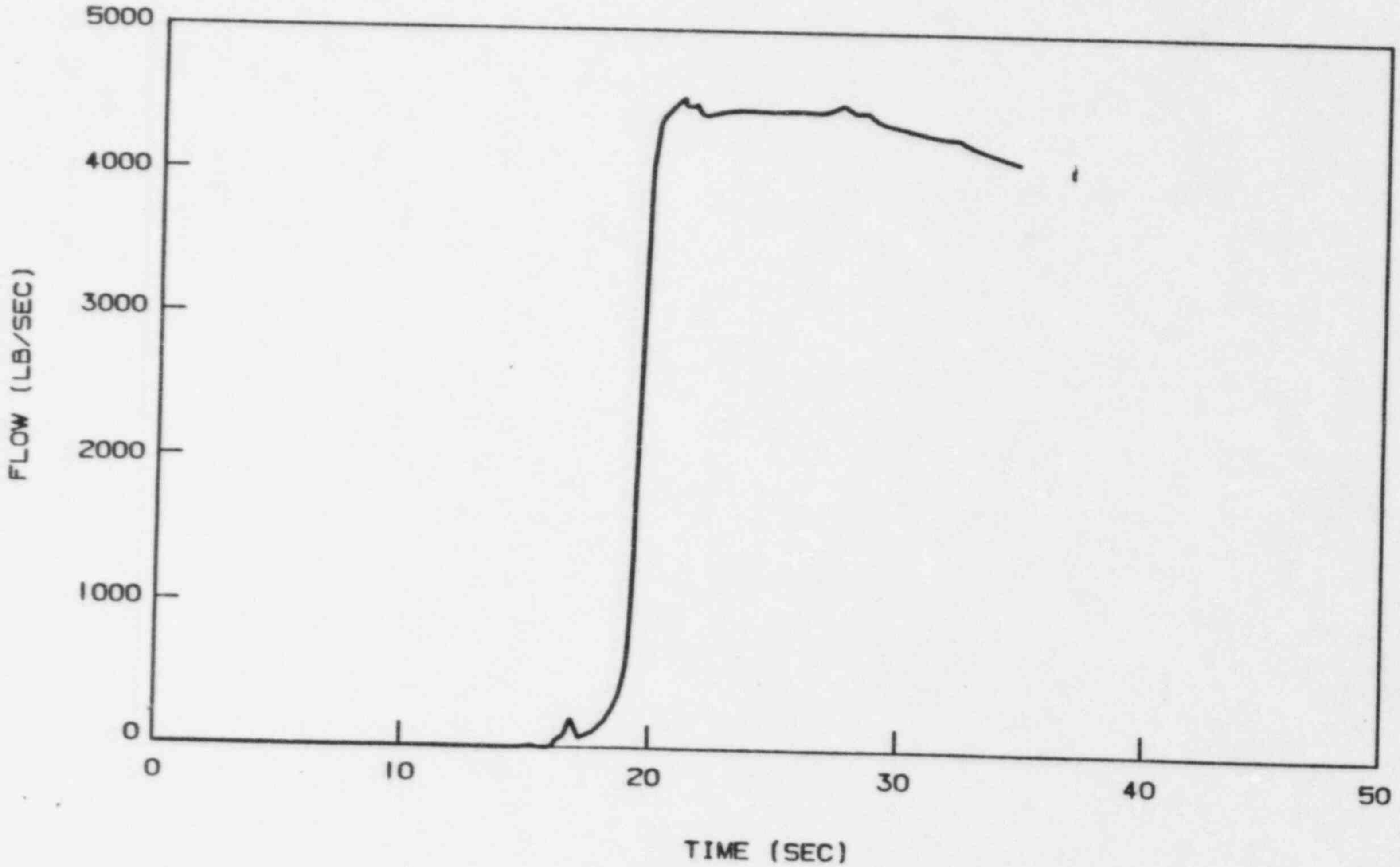


Figure 15.6.4-39: Accumulator Injection, C_D=0.6 DECLG

14715.22

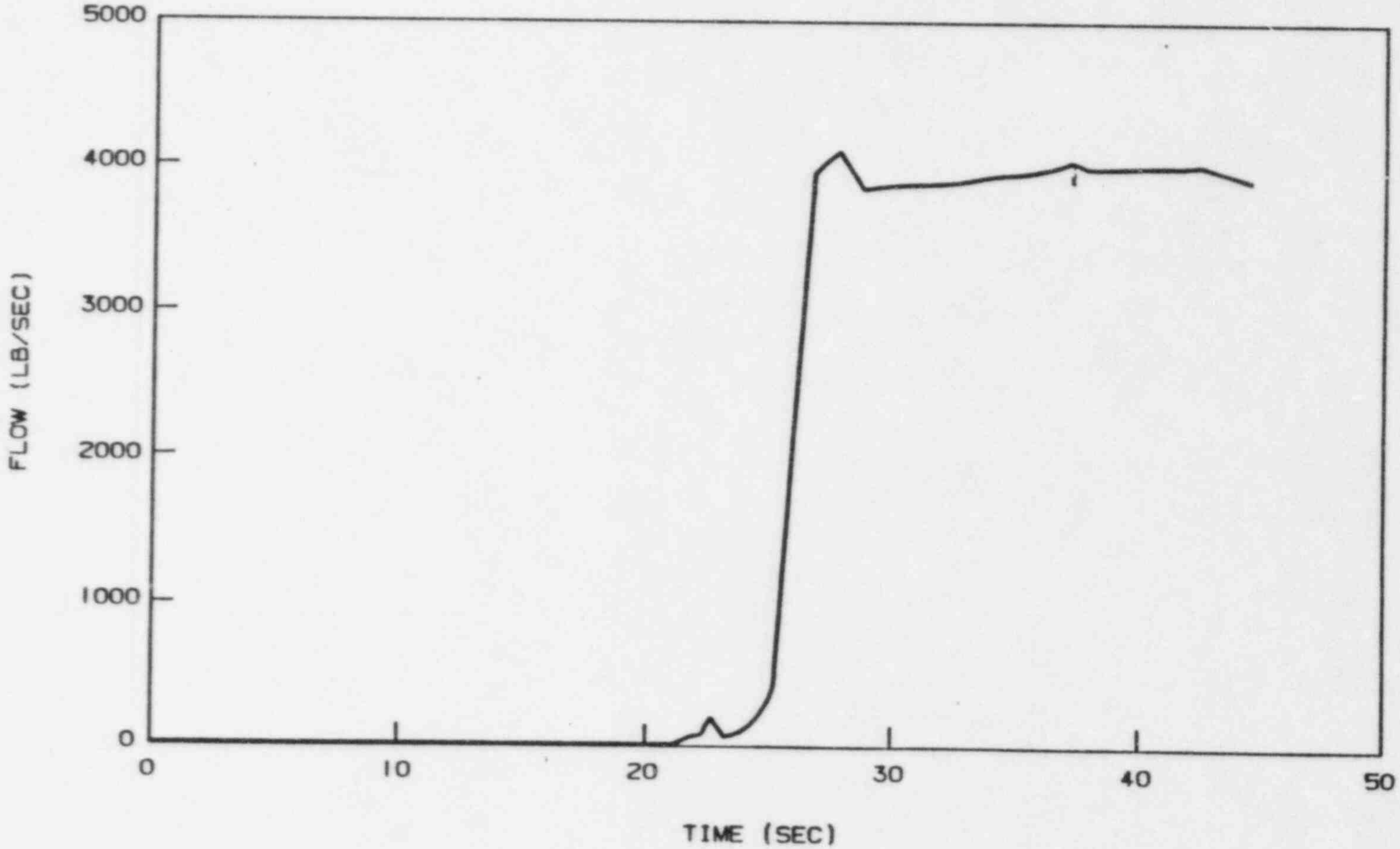


Figure 15.6.4-40: Accumulator Injection, C_D=0.4 DECLG

14715.22

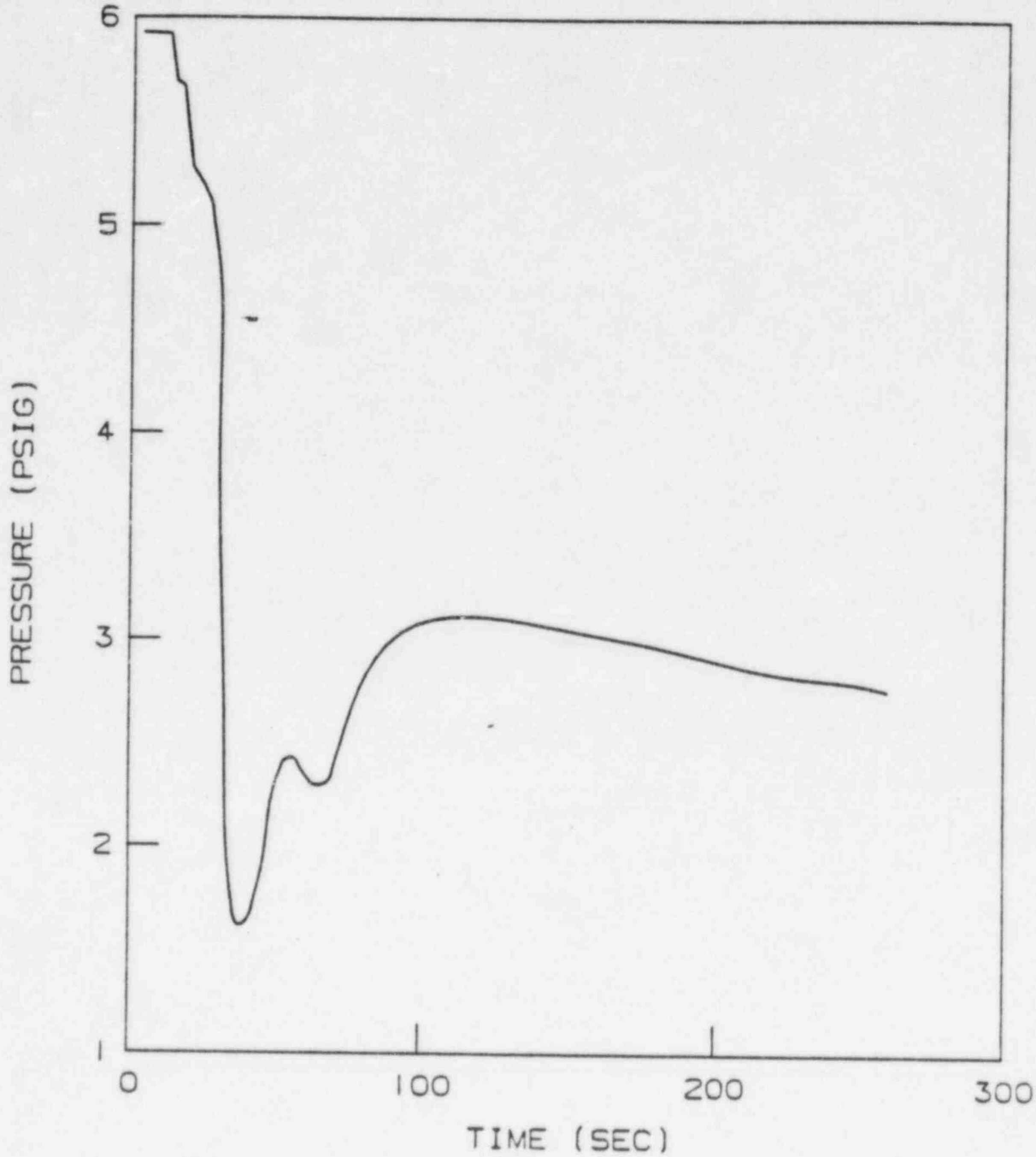


Figure 15.6.4-41 Compartment Pressure, Minimum SI

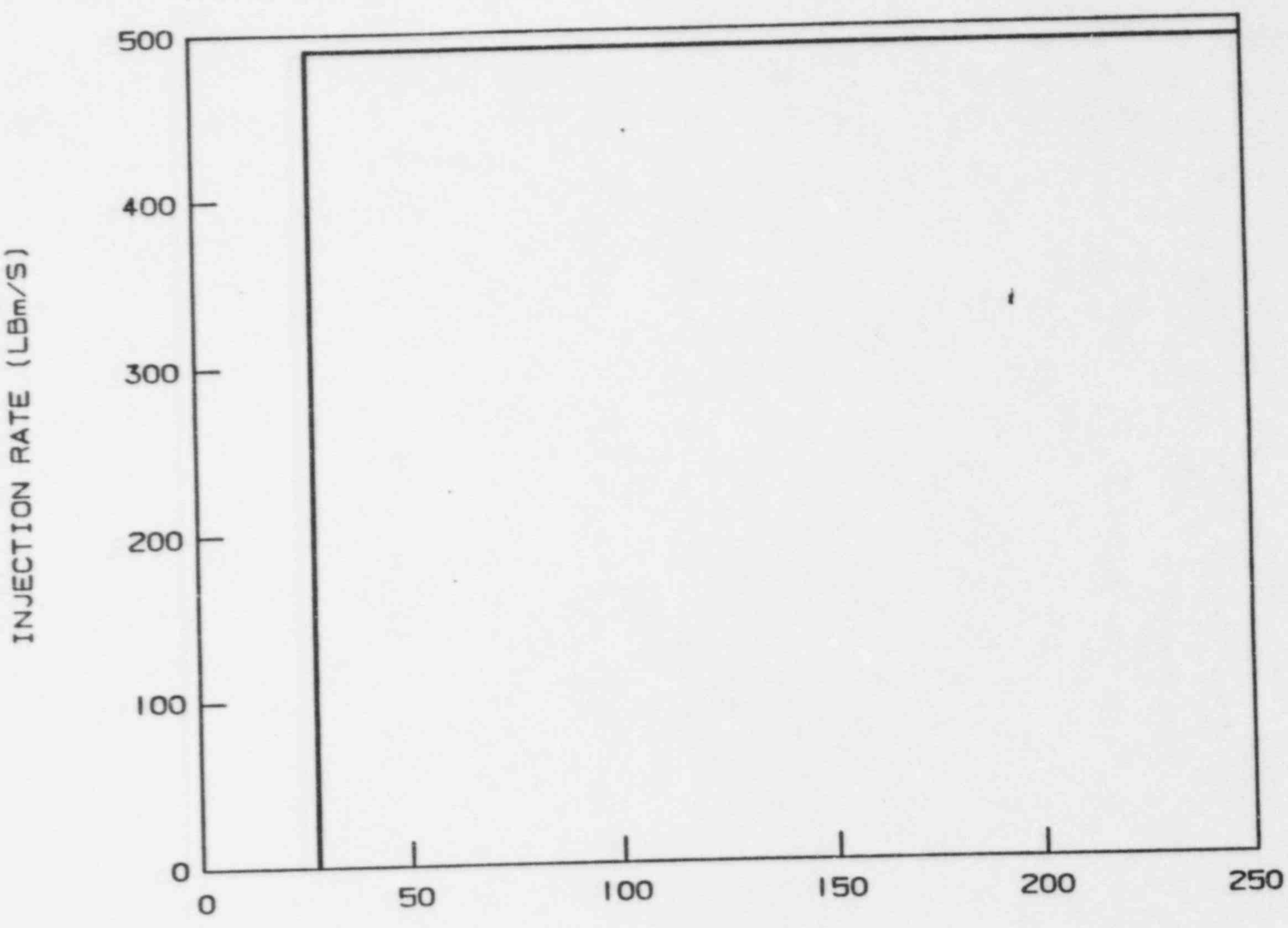


Figure 15.6.4-42: Pumped ECC Injection Rate, C_D=0.6 DECLG

14715.56

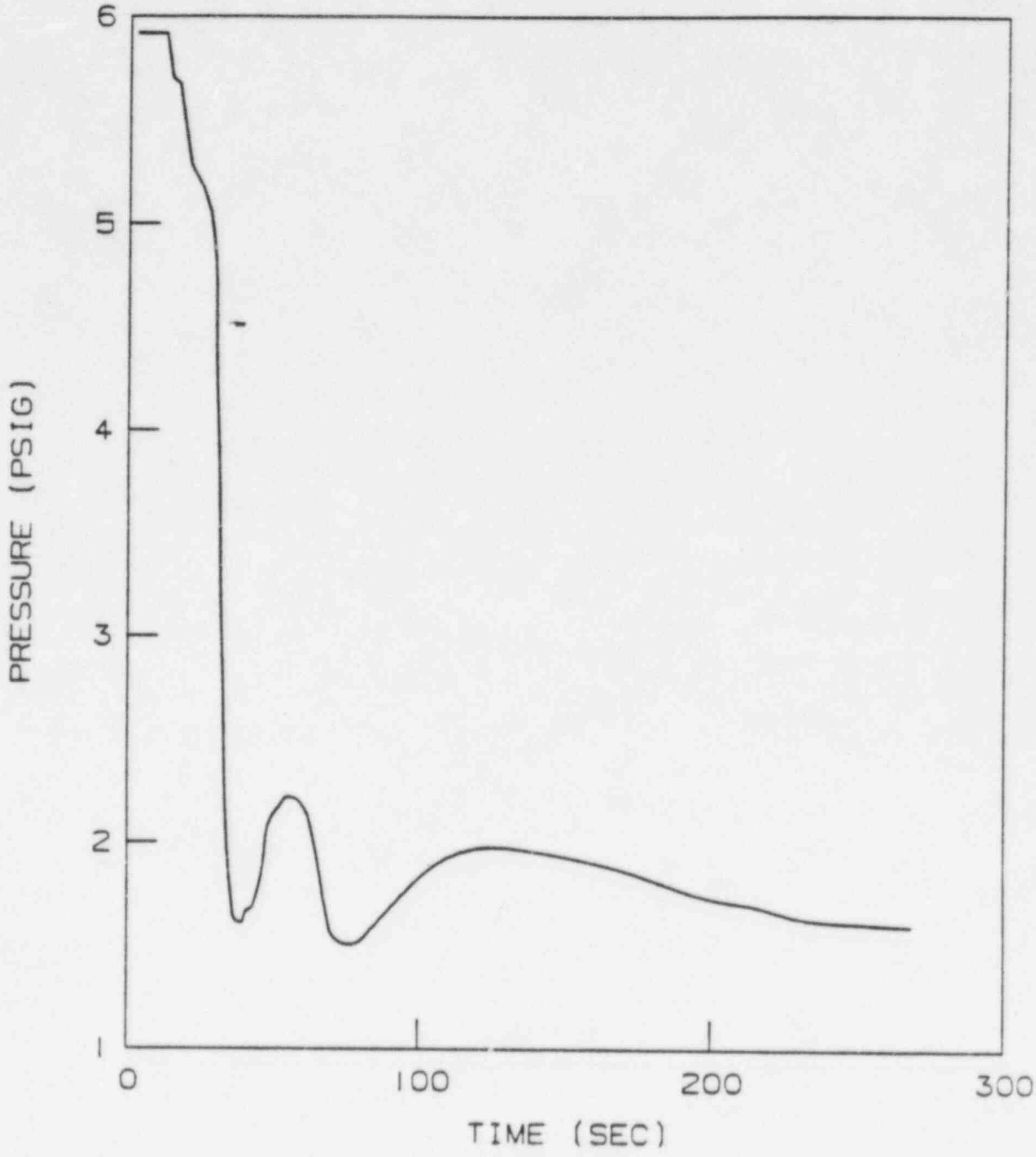


Figure 15.6.4-43 Compartment Pressure, Maximum SI

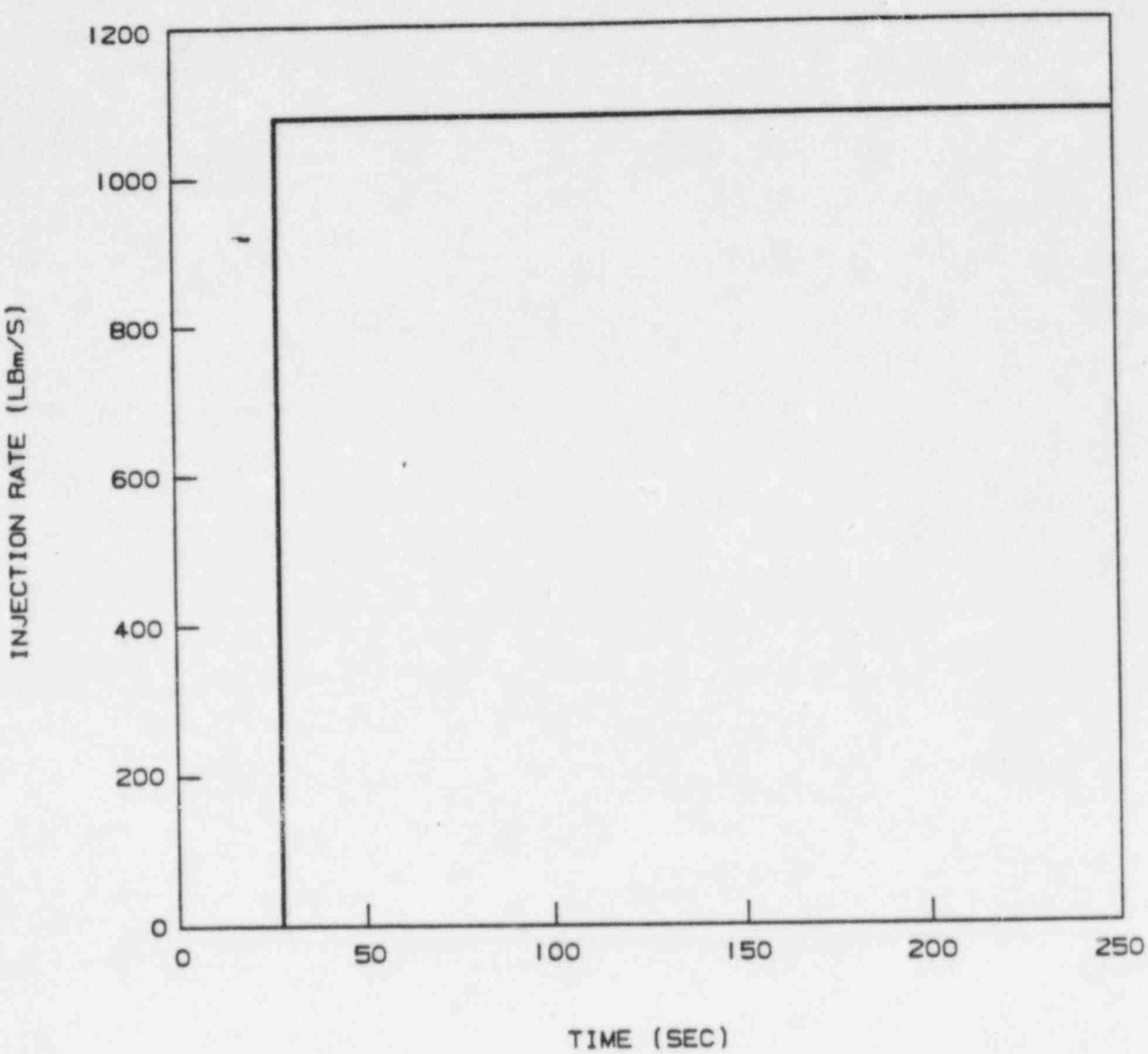
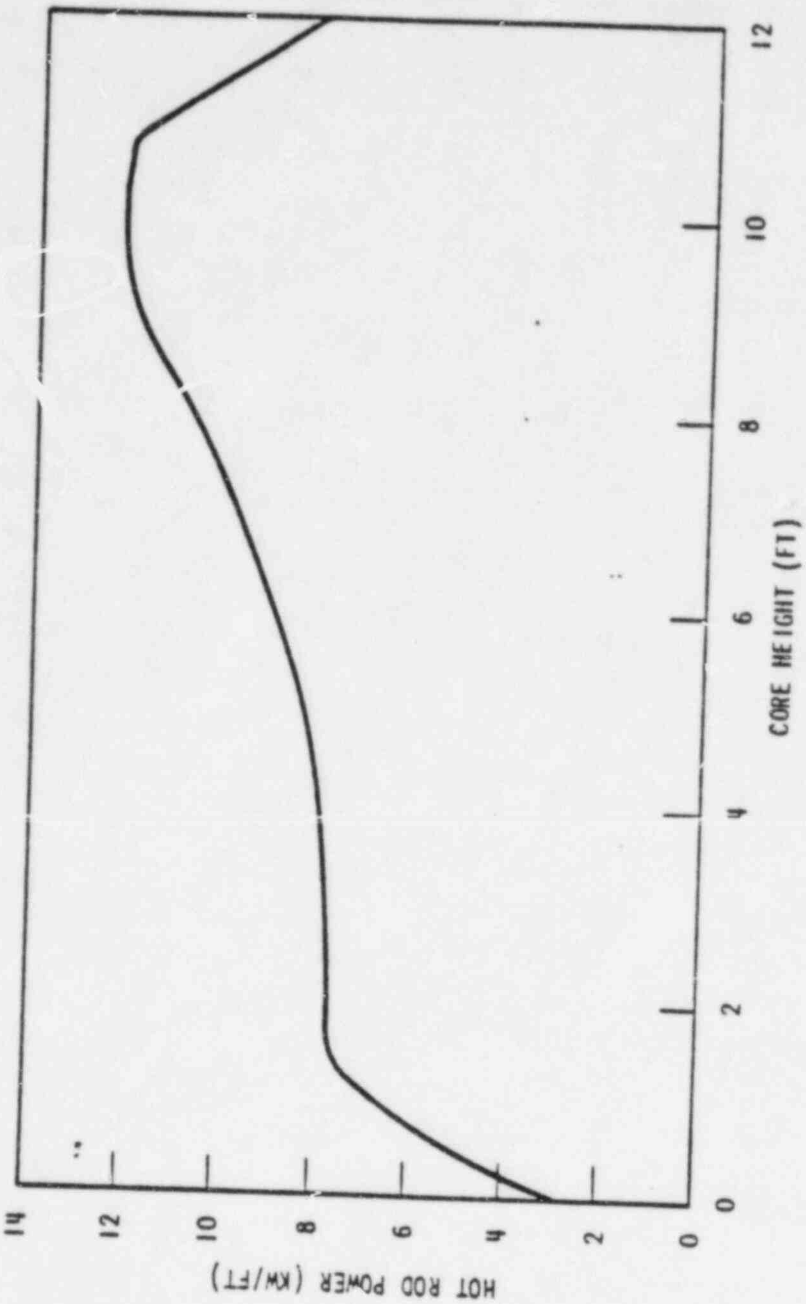


Figure 15.6.4-44: Pumped ECC Injection Rate, C_D=0.6 DECLG Max. SI

FIGURES 15.6.4-45 - 15.6.4-59

HAVE BEEN DELETED

No Change From
1984 FSAR Update

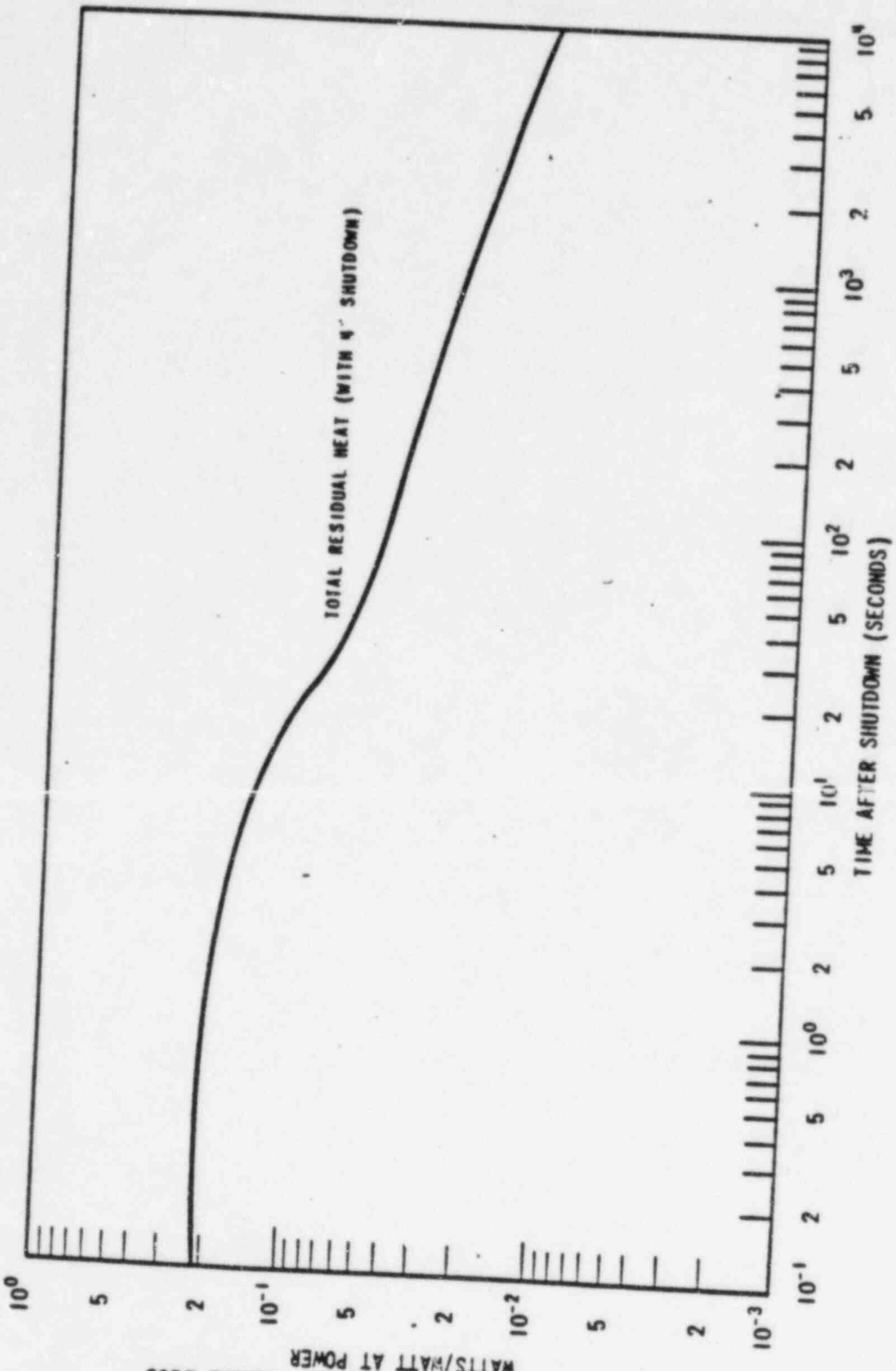


SMALL BREAK POWER DISTRIBUTION
ASSUMED FOR LOCA ANALYSES

McGUIRE NUCLEAR STATION

Figure 15.6. 4-60
1984 Update

No Change From
1984 FSAR Update



CORE POWER AFTER REACTOR TRIP (APPLIES TO ALL SMALL BREAKS)

McGUIRE NUCLEAR STATION

Figure 15.6.4-61
1984 Update

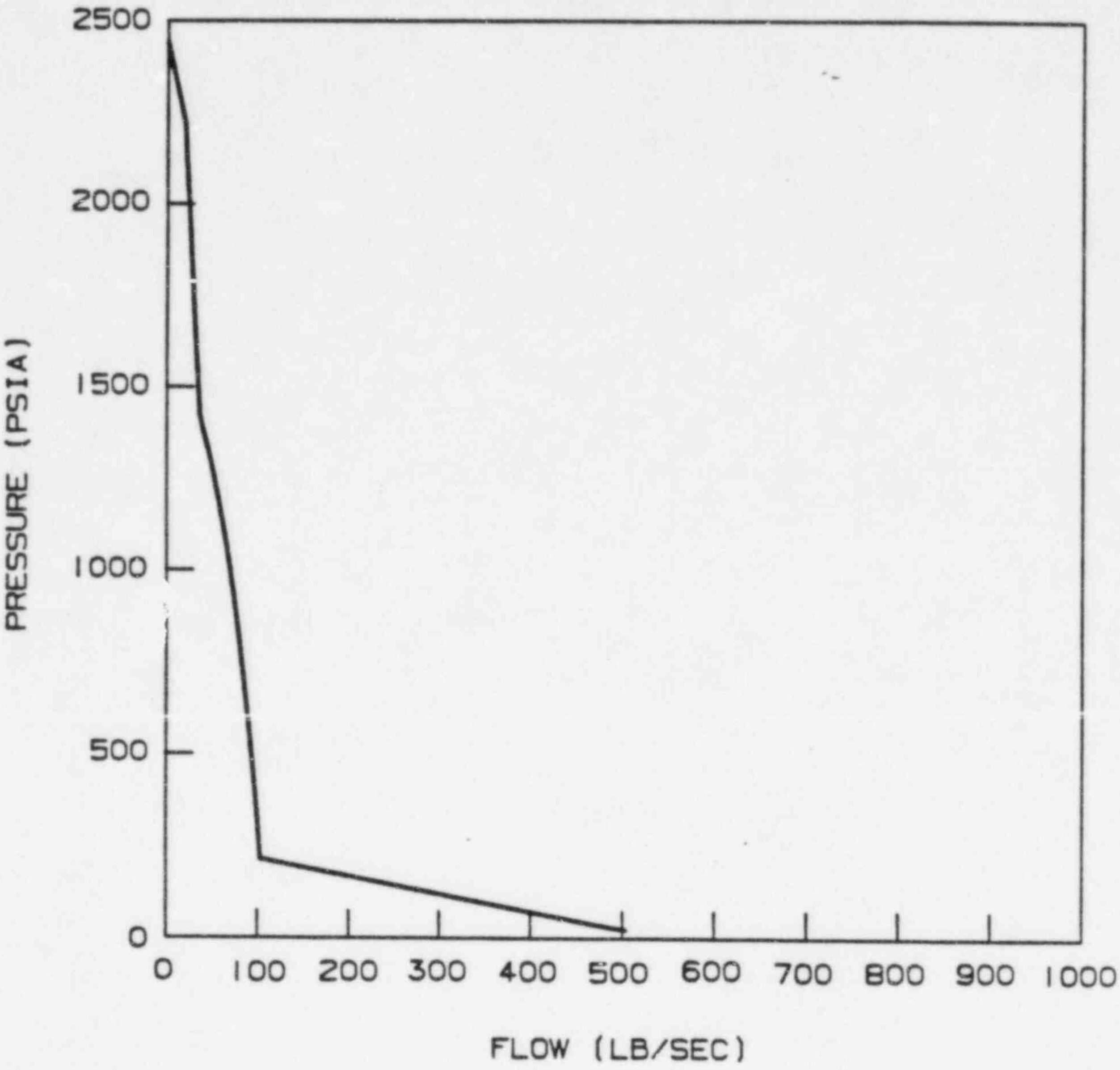


Figure 15.6.4-62: Safety Injection Flowrate vs. Pressure

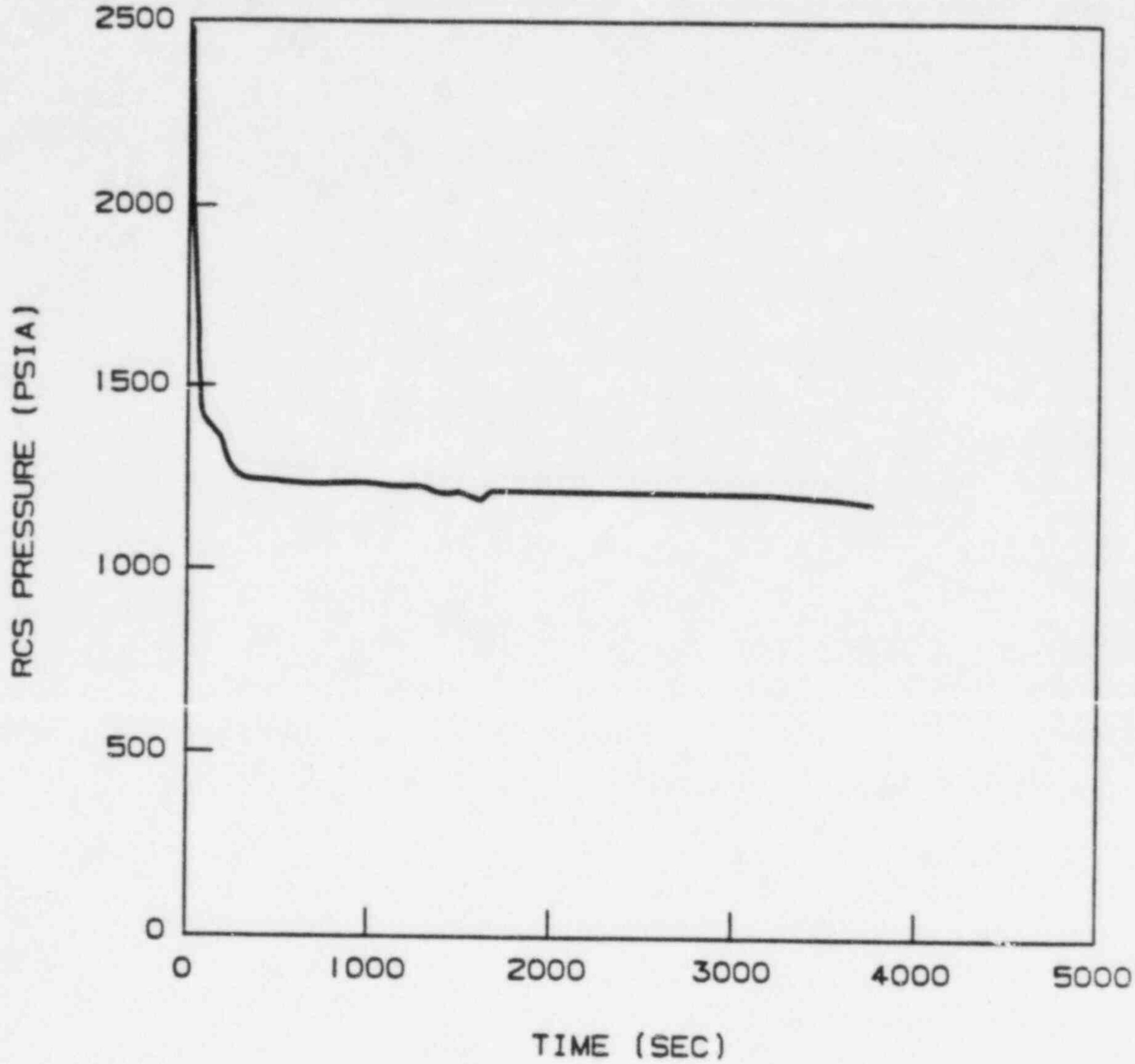


Figure 15.6.4-63a: 2" Cold Leg Break RCS Pressure vs. Time

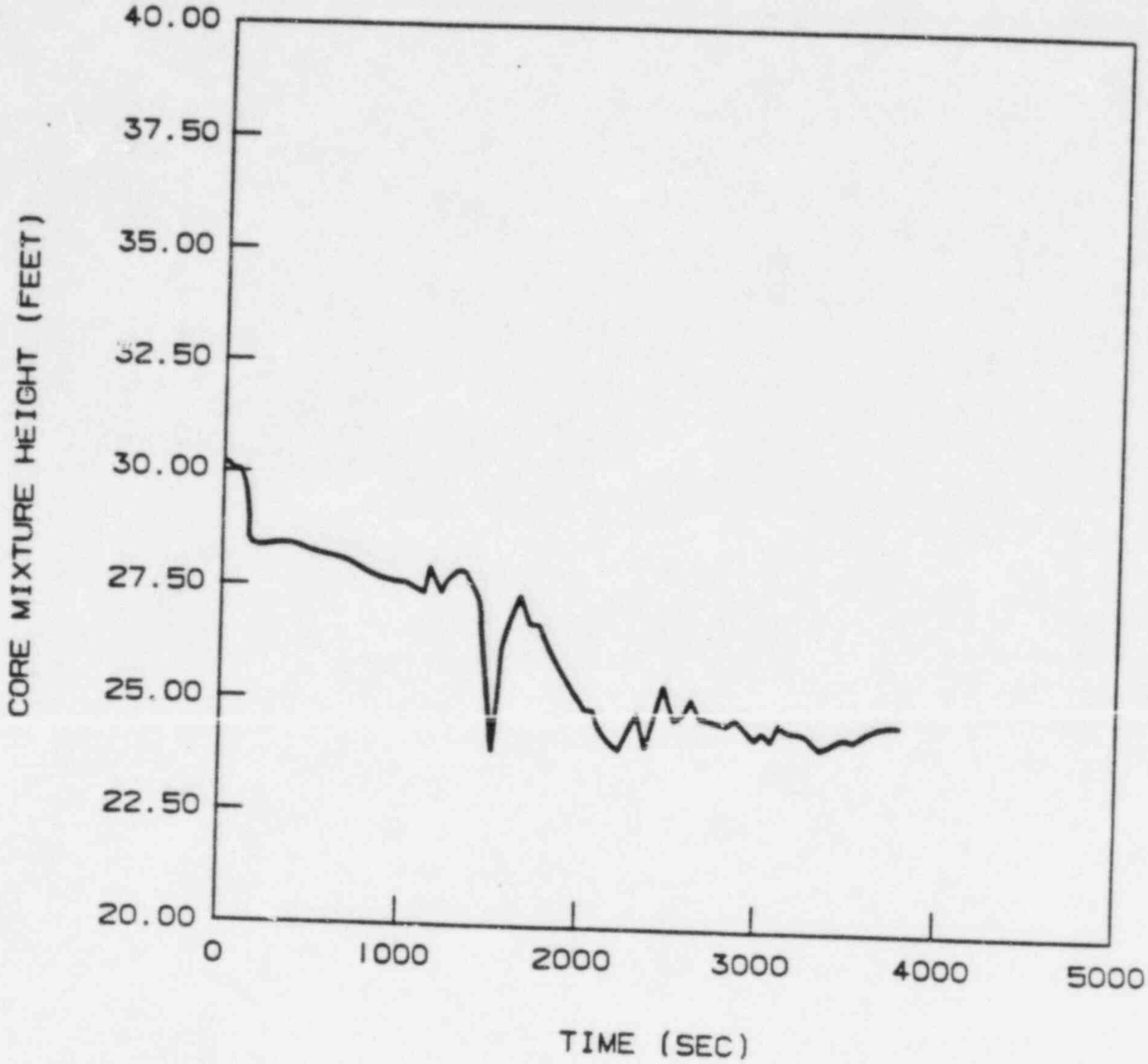


Figure 15.6.4-63b: 2" Cold Leg Break Core Mixture Height vs. Time

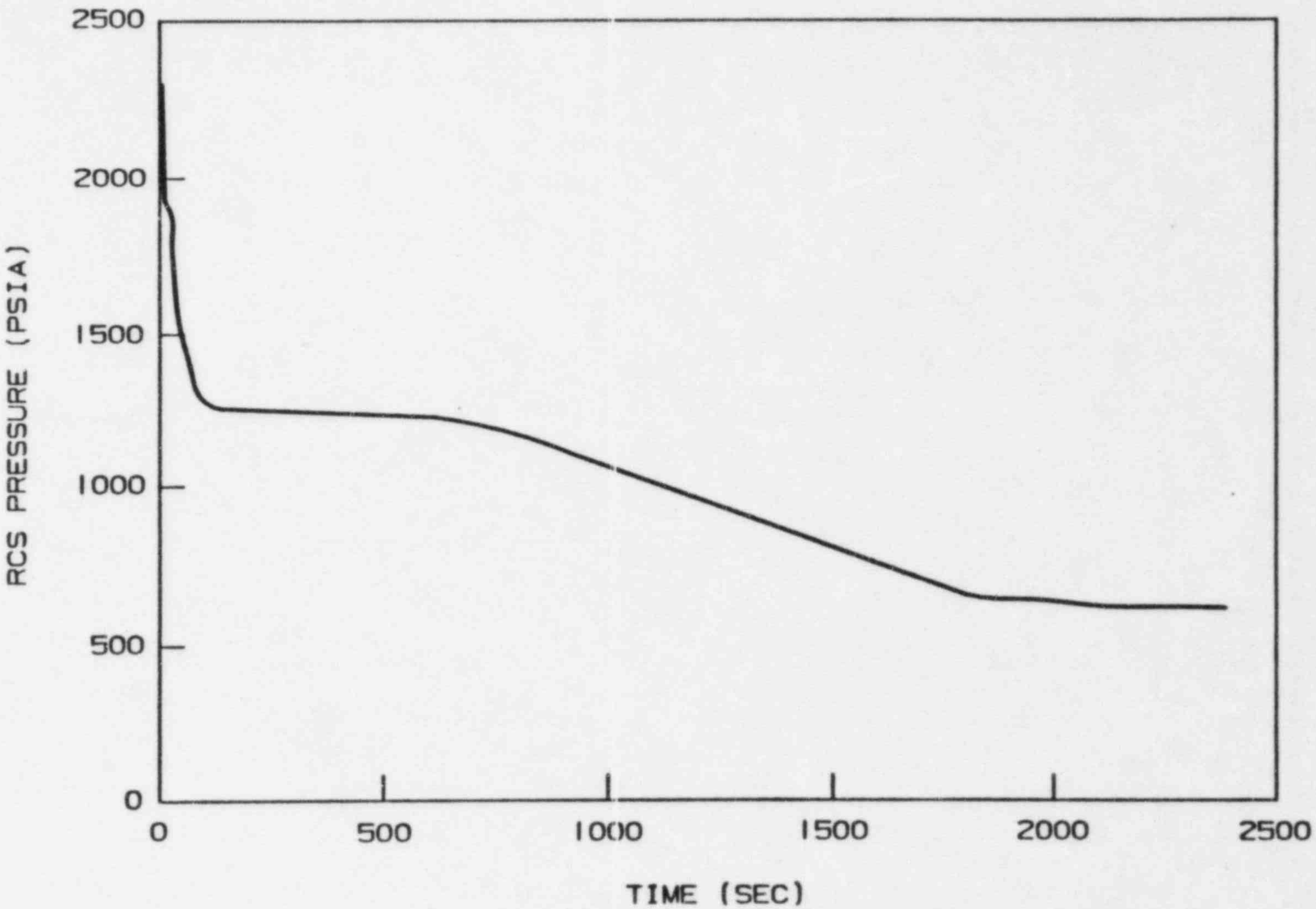


Figure 15.6.4-63c: 3" Cold Leg Break RCS Pressure vs. Time

14715.24

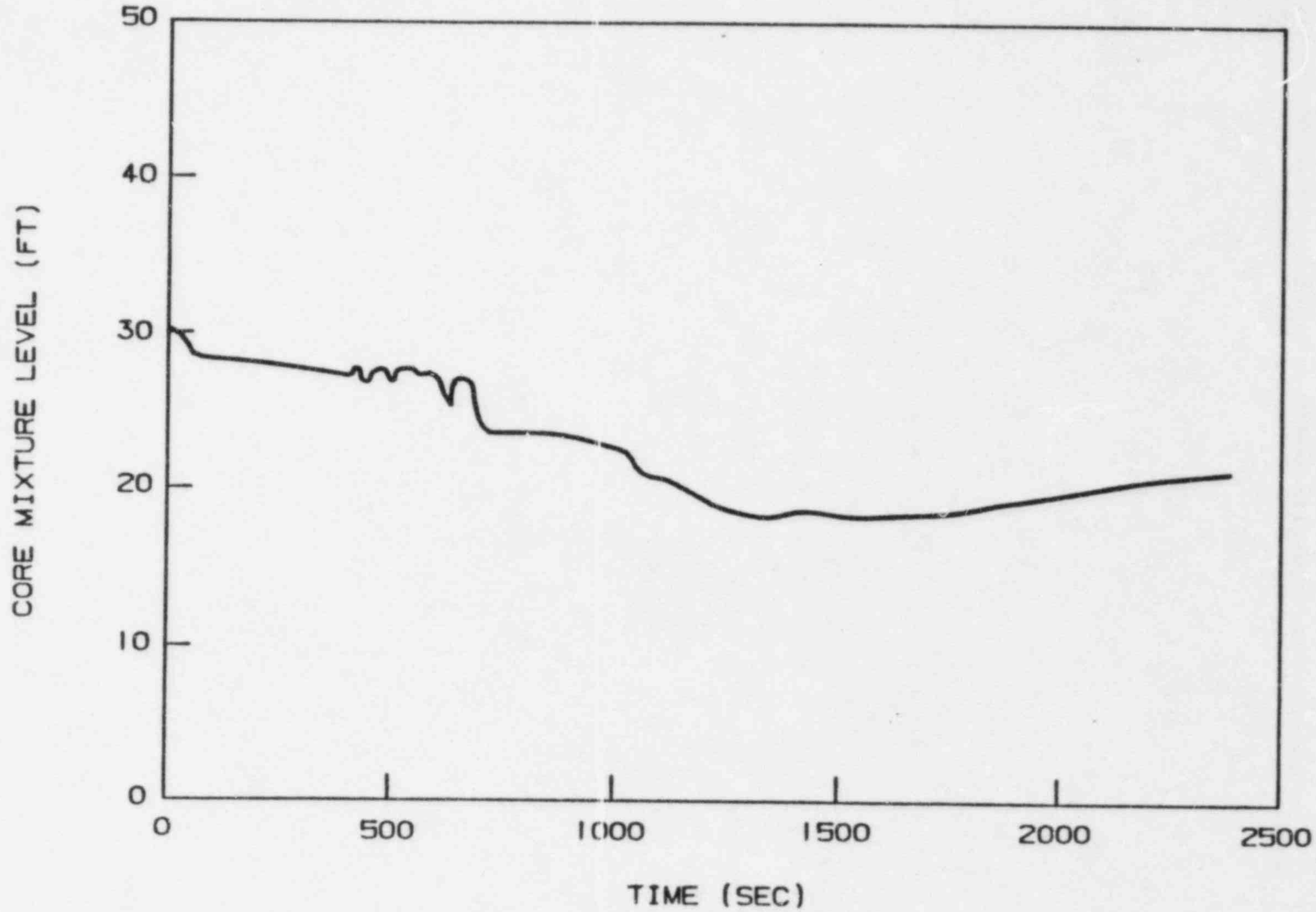


Figure 15.6.4-64: 3" Cold Leg Break Core Mixture Height vs. Time

14715.25

14715.59

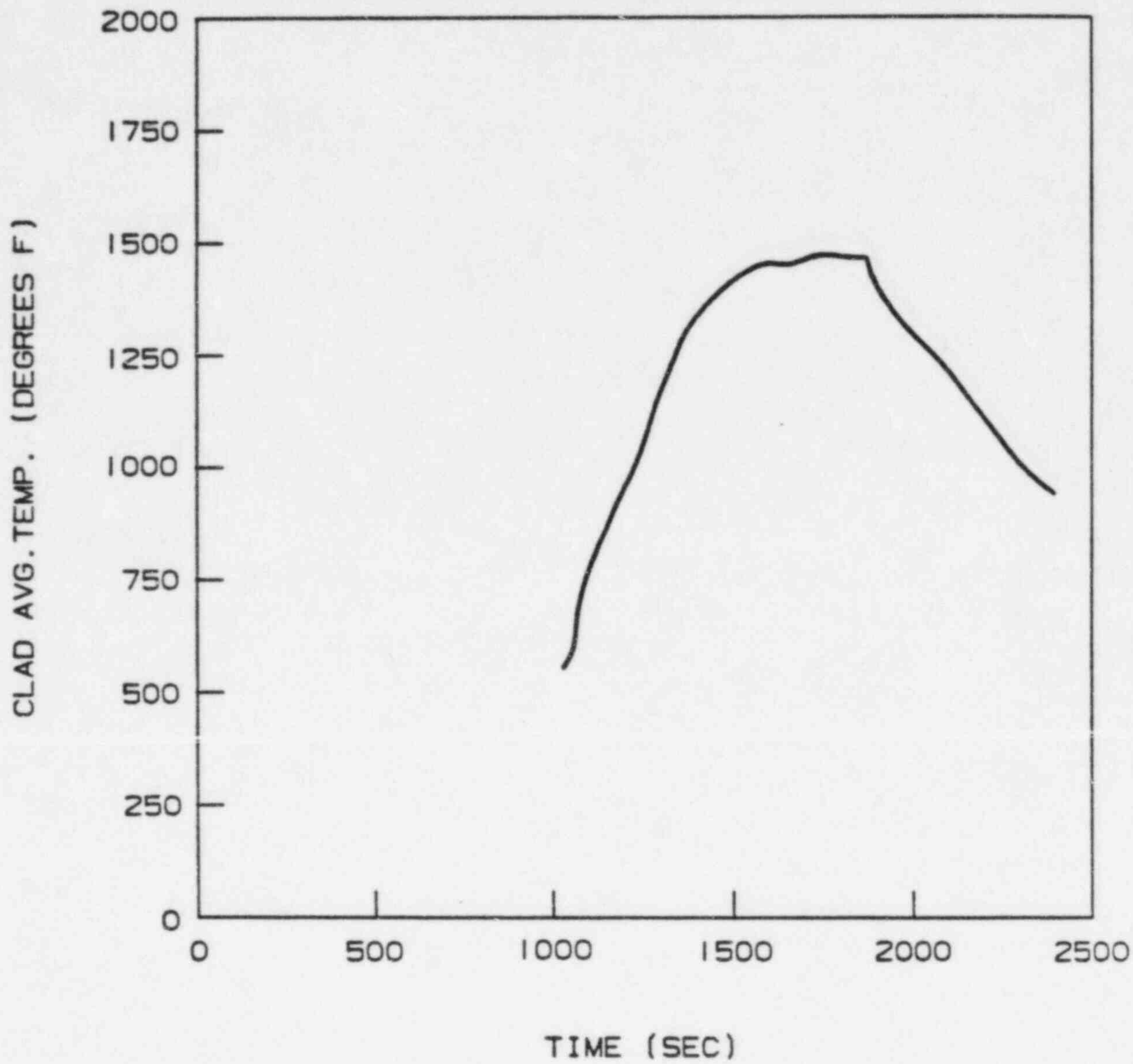


Figure 15.6.4-65: 3" Cold Leg Break Hot Spot Clad Temperature vs. Time

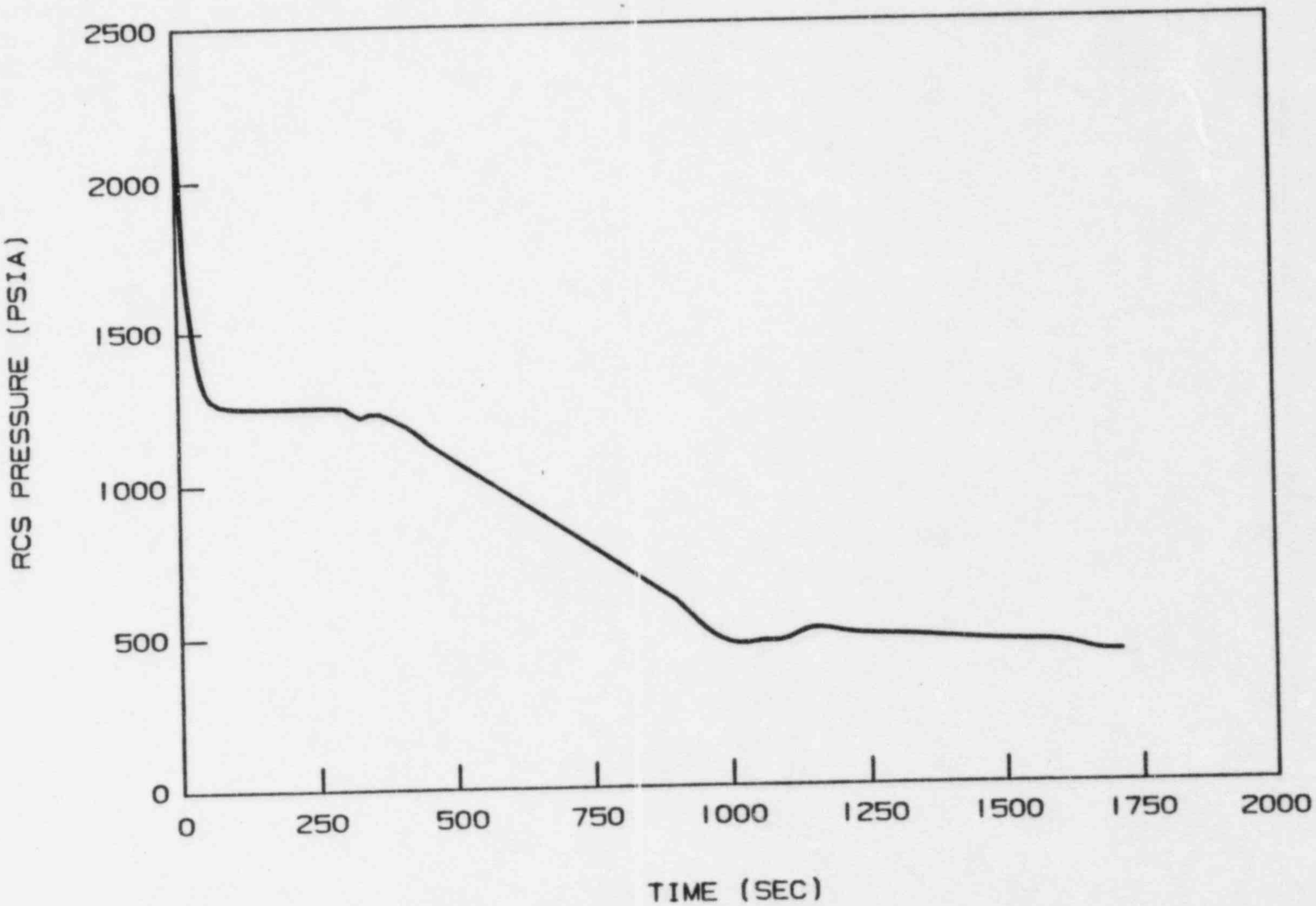


Figure 15.6.4-66: 4" Cold Leg Break RCS Pressure vs. Time

14715.27

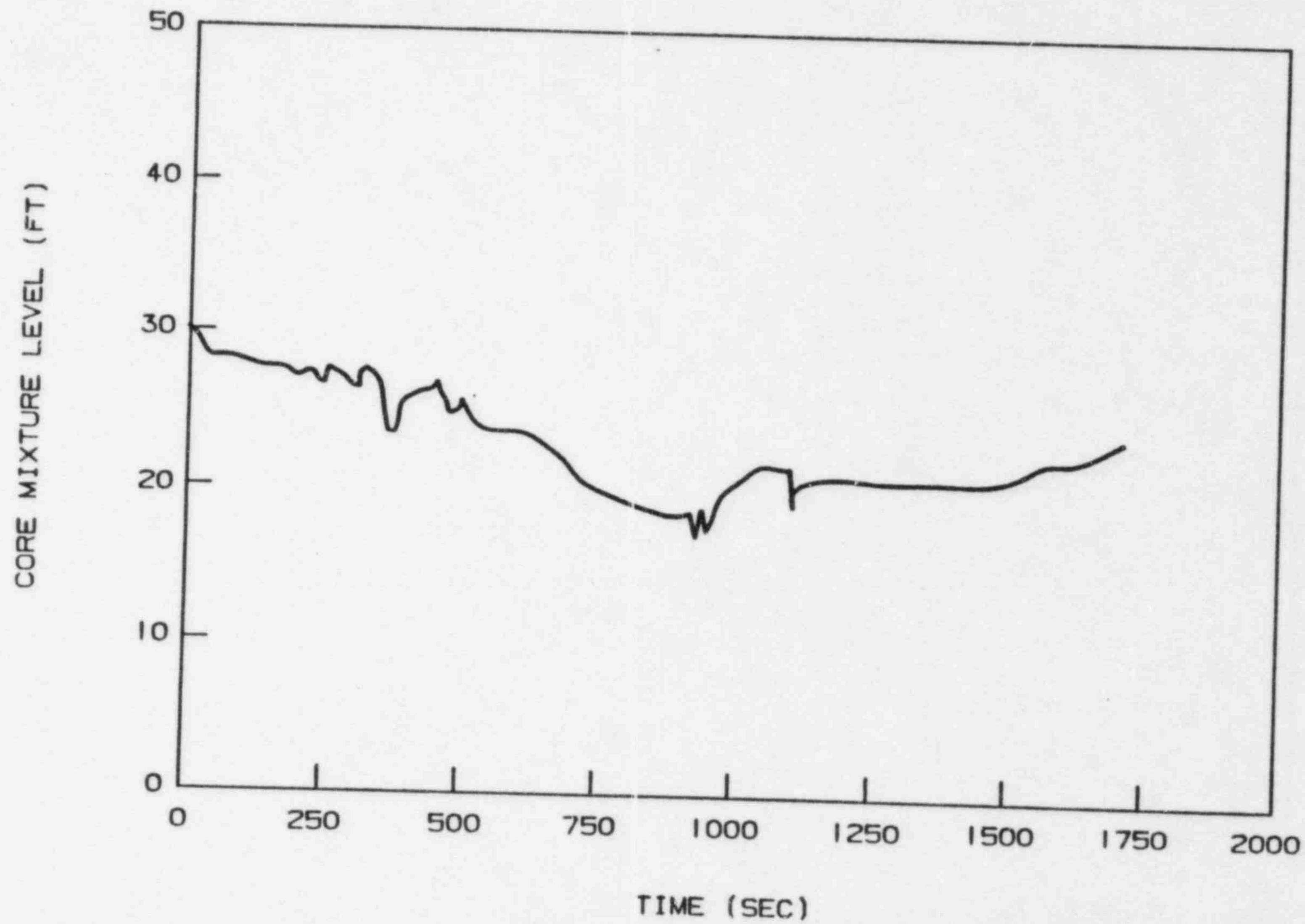


Figure 15.6.4-67: 4" Cold Leg Break Core Mixture Height vs. Time

14715.28

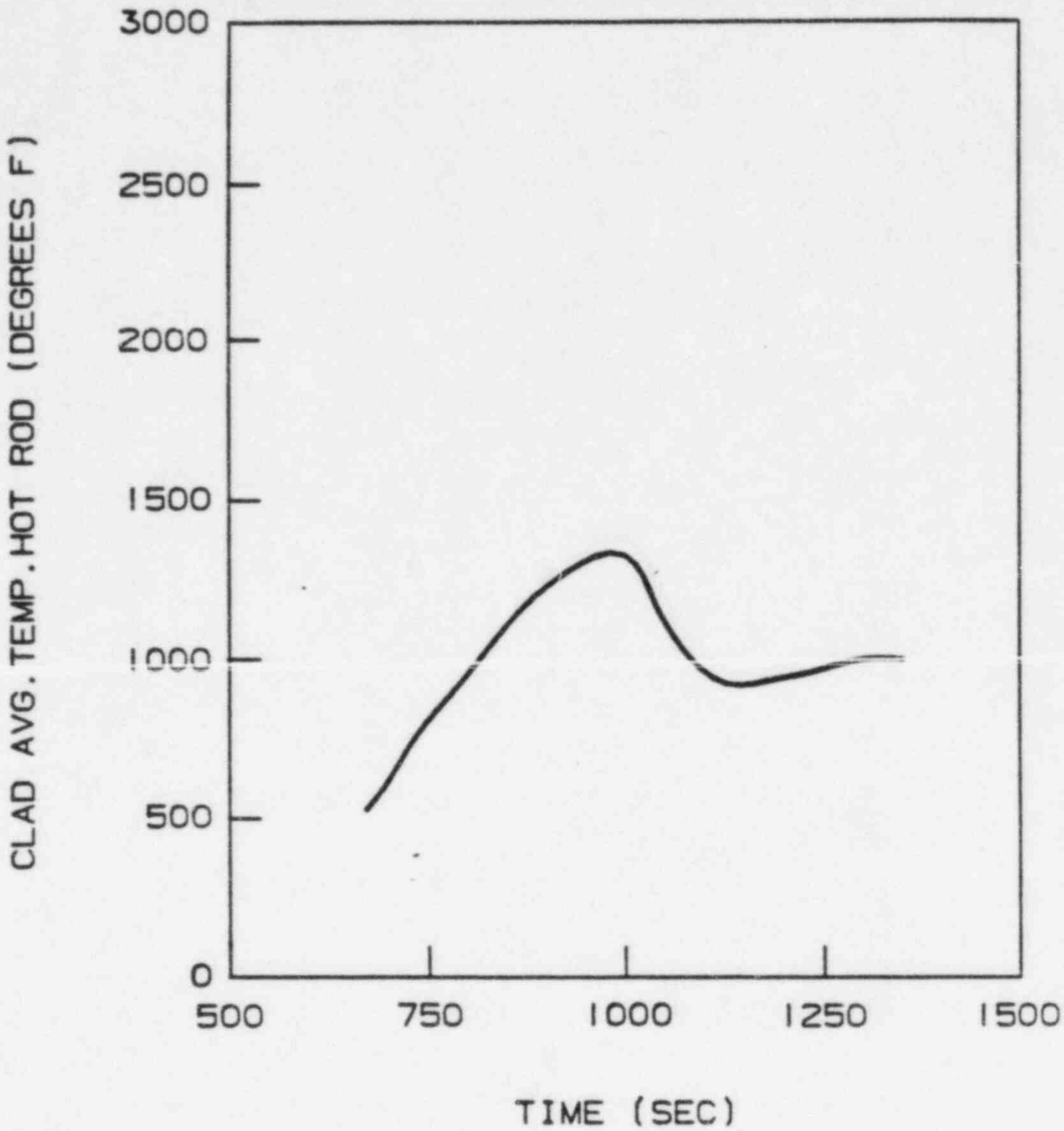


Figure 15.6.4-68: 4" Cold Leg Break Hot Spot Clad Temperature vs.Time

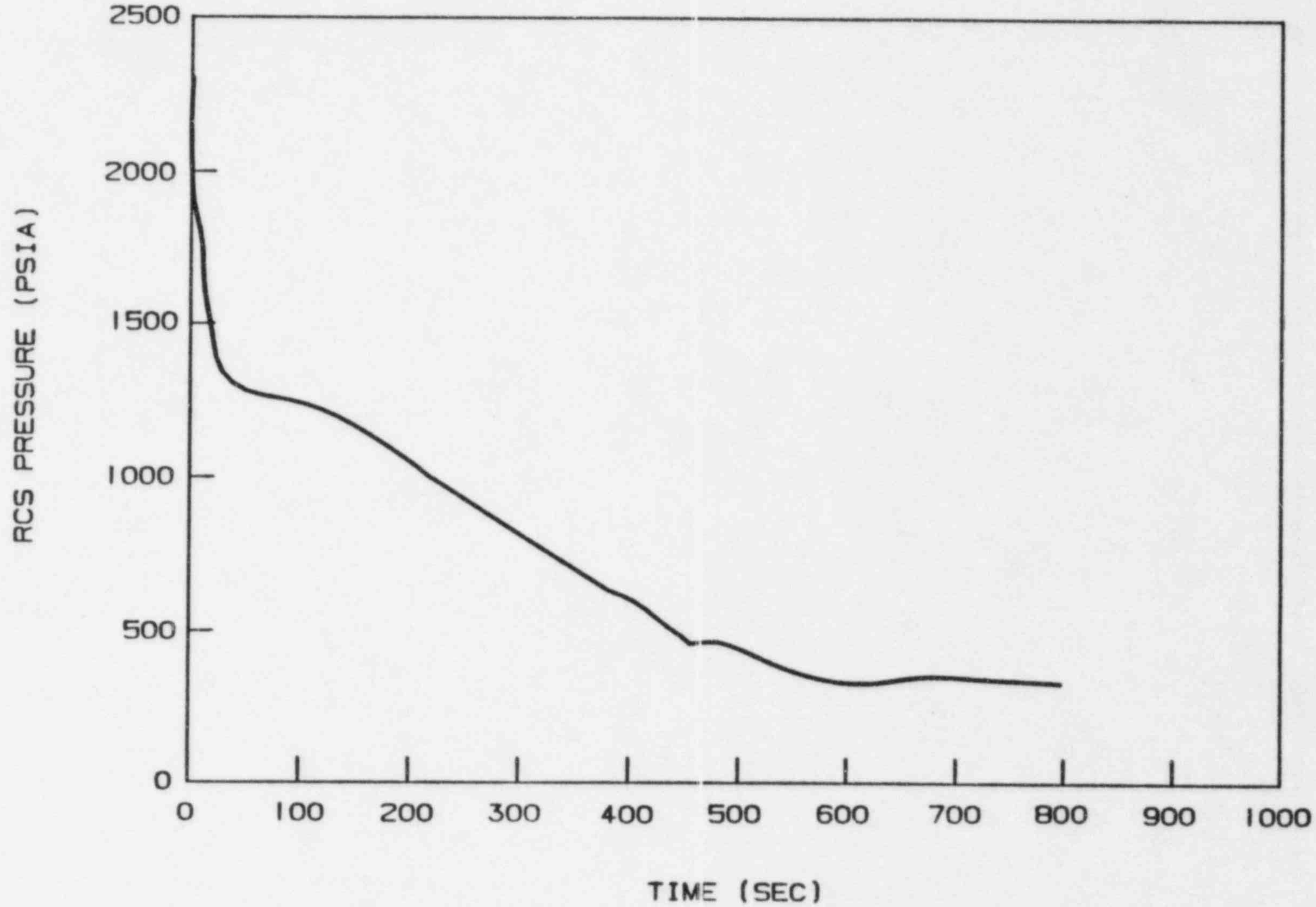


Figure 15.6.4-69: 6" Cold Leg Break RCS Pressure vs. Time

14715.30

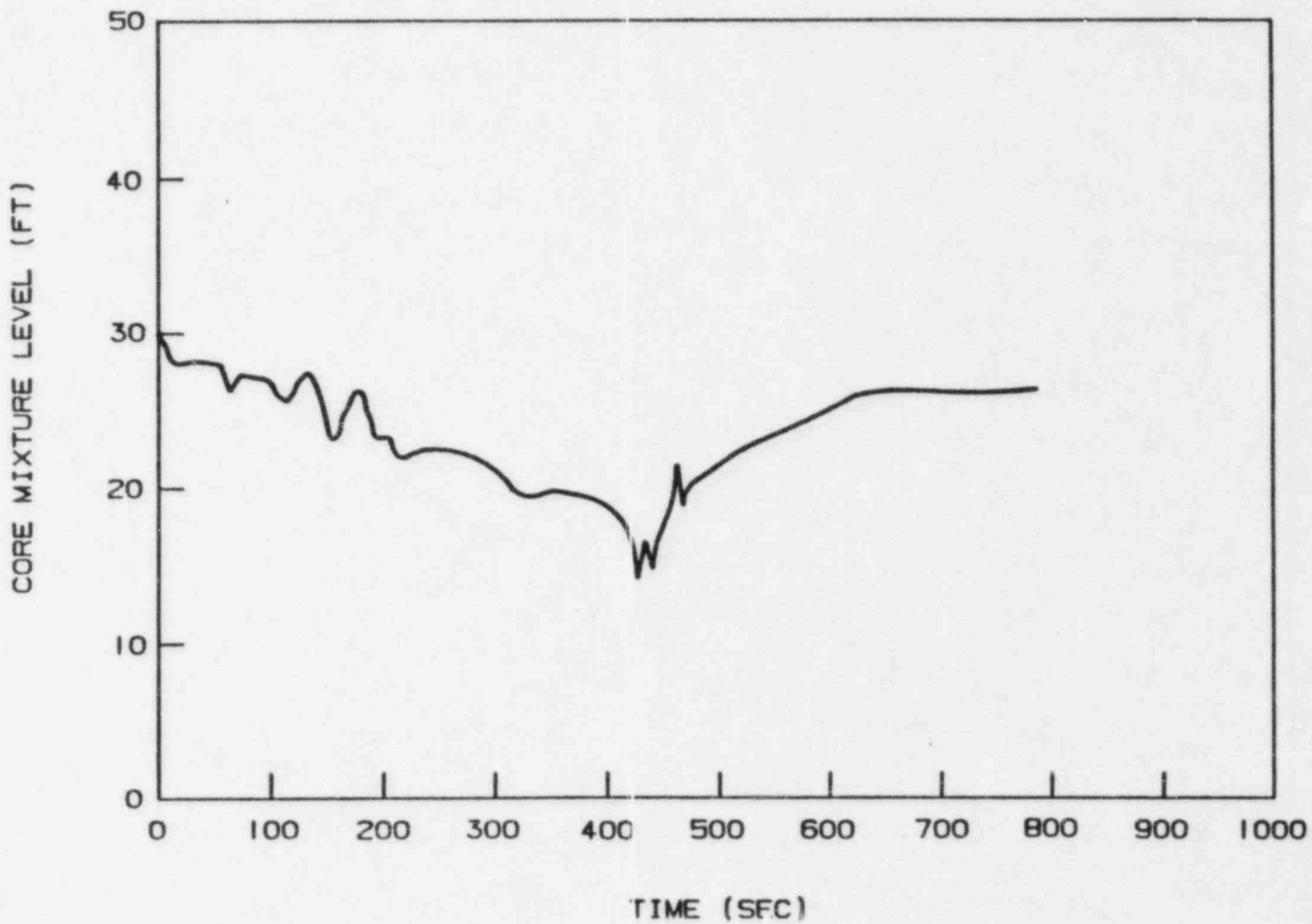


Figure 15.6.4-70: 6" Cold Leg Break Core Mixture Height vs. Time

14715.31

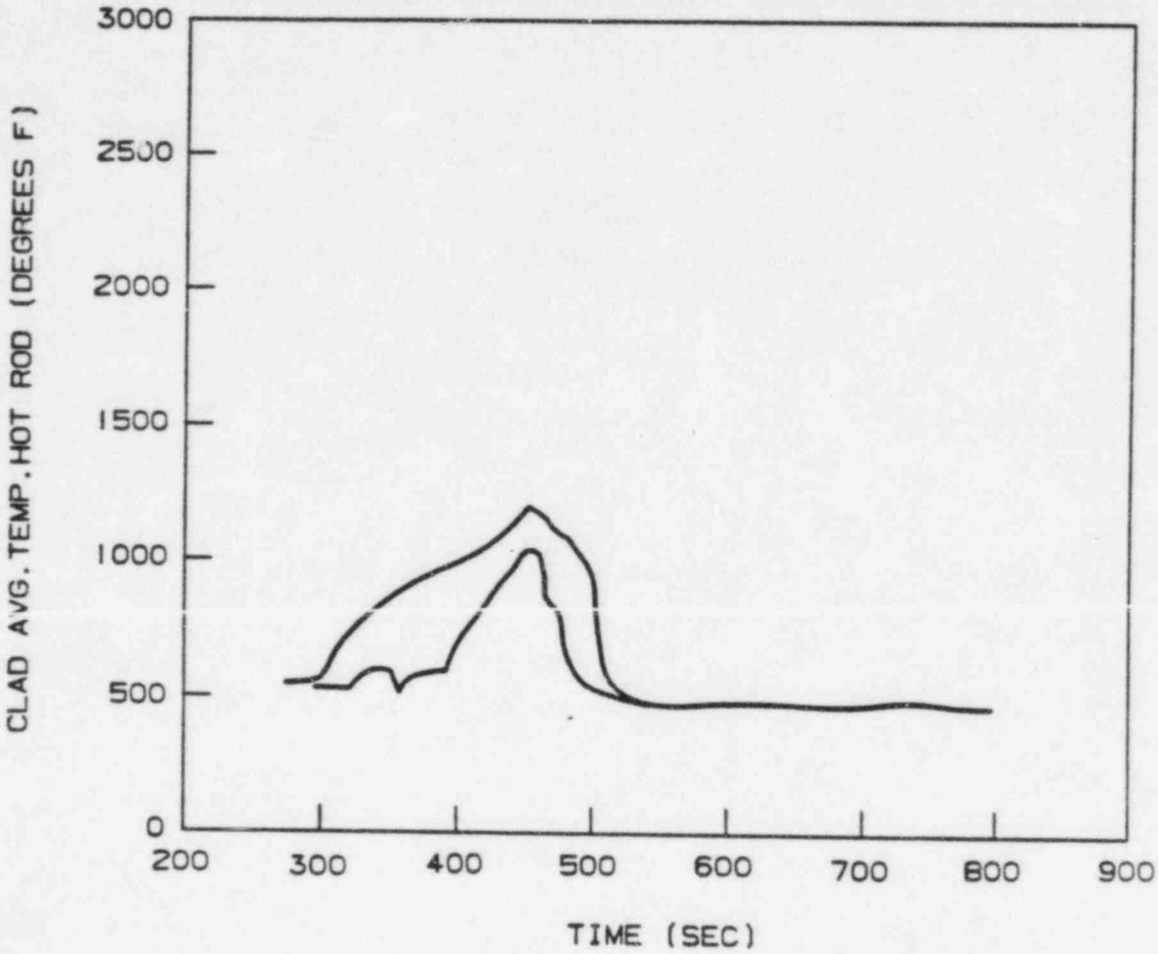


Figure 15.6.4-71: 6" Cold Leg Break Hot Spot Clad Temperature vs. Time

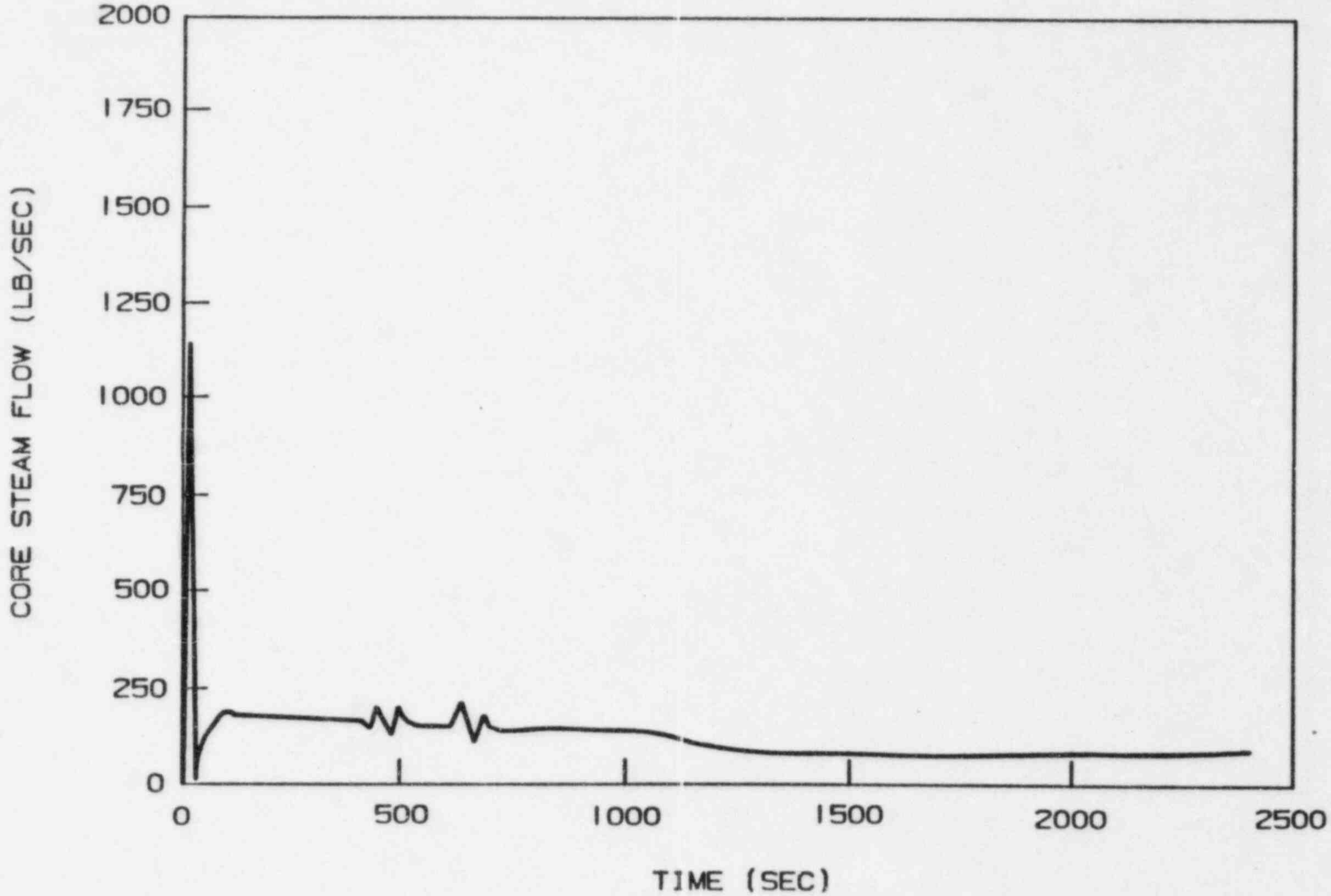


Figure 15.6.4-72: 3" Cold Leg Break Core Steam Flow vs. Time

14715.33

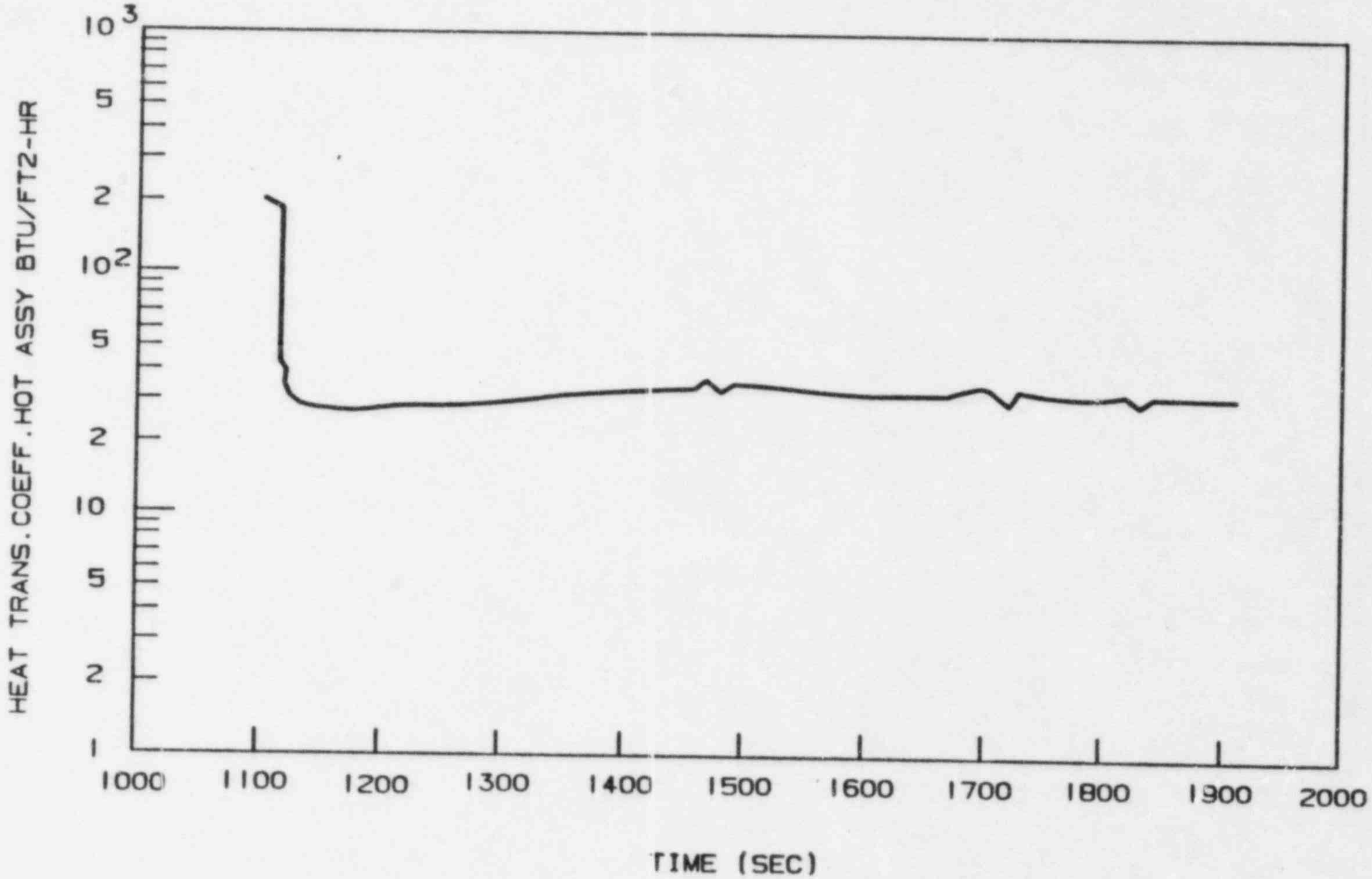


Figure 15.6.4-73: 3" Cold Leg Break Core Heat Transfer Coefficient vs. Time

14715.34

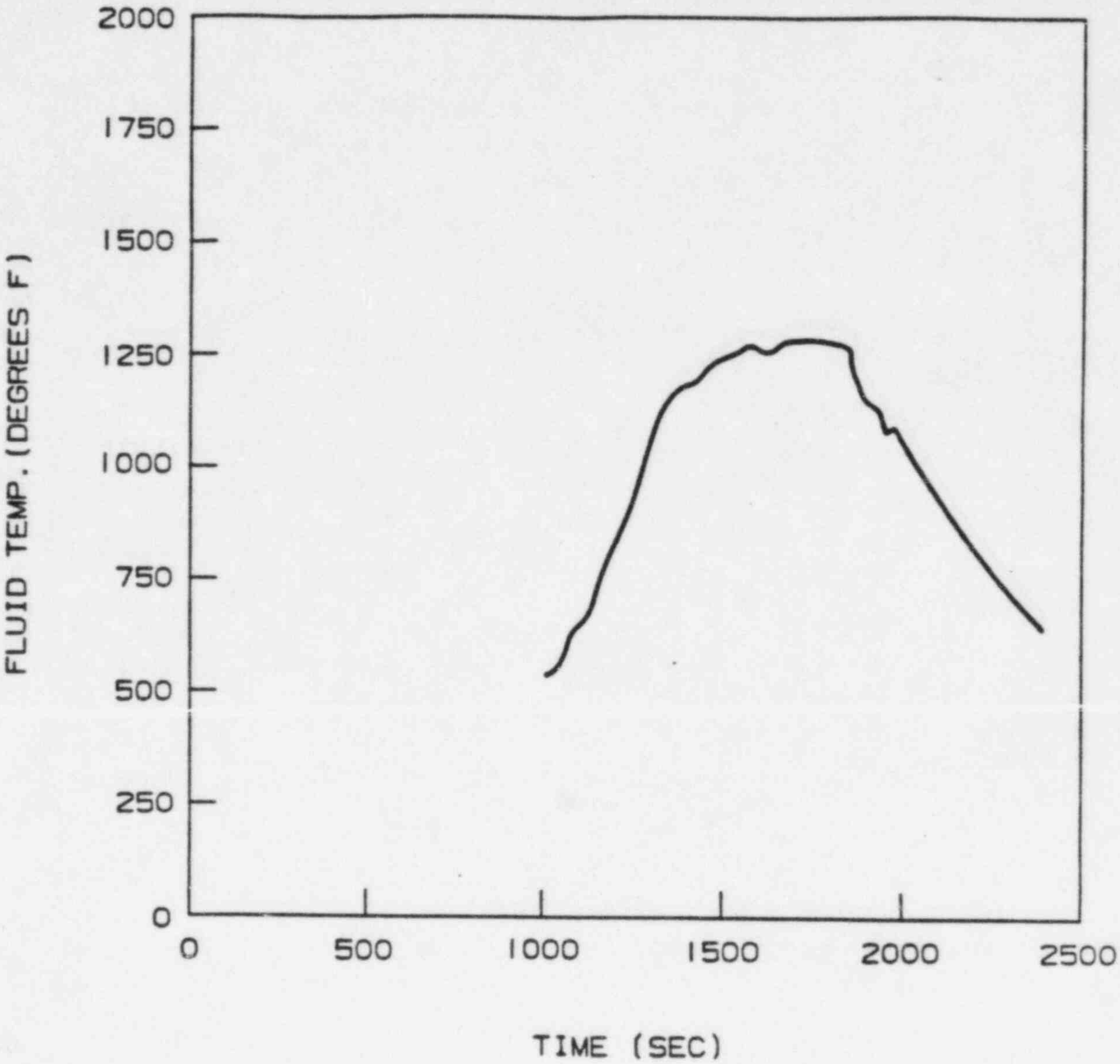


Figure 15.6.4-74: 3" Cold Leg Break Hot Spot Fluid Temperature vs. Time