SECTION 15.0

ACCIDENT ANALYSES

The following design bases events were considered in the design of the Integrated Safeguards System.

15.5 INCREASE IN REACTOR COOLANT INVENTORY

Discussion and analysis of the following event is presented in this section:

1. Inadvertent operation of the emergency core cooling system during power operation

This event, considered to be ANS Condition II, does not result in an increase in reactor coolant inventory.

An increase in the reactor coolant inventory due to CVCS malfunctions will be discussed in the "Reactor Coolant System" module.

15.5.1 Inadvertent Operation of the Emergency Core Cooling System During Power Operation

15.5.1.1 Identification of Causes and Accident Description

Spurious emergency core cooling system (ECCS) operation at power could be caused by operator error or a false electrical actuation signal. A spurious signal may originate from any of the safety injection actuation channels as described in Section 7.3 of the "I&C and Electric Power" module.

Following the actuation signal, the high head safety injection pumps will start automatically. However, since the shutoff head of the safety injection pumps is less than the reactor coolant system (RCS) pressure, no flow will be delivered to the RCS. The passive accumulator injection system and the passive core reflood tanks also provide no flow at normal RCS pressure.

A safety injection system (SIS) signal normally results in a reactor trip followed by a turbine trip. However, it cannot be assumed that any single fault that actuates the SIS will also produce a reactor trip. If a reactor trip is generated by the spurious SIS signal the RCS pressure will still remain above the ISS HHSI pump shutoff head. The operator can then determine if the spurious signal was transient or steady state in nature without concern about overfilling the RCS. For a spurious occurrence, the operator would stop the safety injection and maintain the plant in the hot standby condition. If the SIS actuation instrumentation must be repaired, future plant operation would be in accordance with the Technical Specifications. If a reactor trip does occur, the RCS pressure will remain above the shutoff of the safety injection pumps so that no flow will be provided to the RCS.

15.5.1.2 Conclusions

Since the inadvertent operation of the emergency core cooling system does not result in the injection of any borated water, this event has no impact on either core or RCS integrity. This incident simply results in a reactor trip.

15.6 DECREASE IN REACTOR COOLANT INVENTORY

Events which result in a decrease in reactor coolant inventory as discussed in this section are as follows:

a) Break in instrument line or other lines from reactor coolant pressure boundary that penetrate containment (ANS Conditions II event).

b) Steam generator tube failure (ANS Condition IV event). See the "Secondary Side Safeguards" module for this event.

c) Loss of coolant accident (LOCA) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (ANS Condition IV event).

15.6.2 Break in Instrument Line or Other Lines From Reactor Coolant Pressure Boundary that Penetrate Containment

15.6.2.1 Identification of Causes and Frequency Classification

The estimated frequency of a primary sample or instrument line rupture classifies it as a limiting fault incident. A primary sample or instrument line break provides a release path for reactor coolant outside containment. The line break selected for analysis is the letdown line which penetrates the containment. This is the largest penetration whose failure could result in an event in this category. This failure would result in larger releases than would be the case for the smaller instrument and sample lines.

Following such a break, the flow out the break would be limited by the letdown orifices to approximately 100 gpm, assuming that the orifices were selected for normal letdown. The CVCS reactor makeup control system and charging pumps could maintain programmed pressurizer level and pressure indefinitely.

The operator would receive a low flow alarm from the letdown flowmeter, and would be required to isolate the break manually. The break could be isolated by remotely operated valves at the reactor coolant pressure boundary, letdown orifices, or containment boundary. The operator may, depending on break location, receive high auxiliary building radiation or sump alarms, or a high humidity alarm. The operator would also receive indication that his reactor makeup system was operating very frequently.

Since the pressurizer pressure never reaches the safety injection setpoint, the SIS is not actuated and therefore has no effect on this event.

15.6.4 LOCA Resulting From a Spectrum of Postulated Breaks Within RCS Pressure Boundary

15.6.4.1 Identification of Causes and Frequency Classification

A loss-of-coolant accident (LOCA) is the result of a pipe rupture of the Reactor Coolant System (RCS) pressure boundary. 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 a limiting fault, an ANS Condition IV event, in that it is not expected to occur during the lifetime of the plant, but is postulated as a conservative design basis.

A minor pipe break (small break) is defined as a rupture of the reactor coolant pressure boundary 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 ANS Condition III event in that it is an infrequent fault that may occur during the life of the plant.

The acceptance criteria for the loss-of-coolant accident is described in 10CFR 50 Paragraph 46 (Reference 15.6.4-1) as follows:

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

b) The amount of fuel element cladding that reacts chemically with water or steam does not exceed one percent of the total amount of Zircaloy in the fuel.

c) 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 percent are not exceeded during or after quenching.

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

e) The core temperature is maintained at an acceptably low value and decay heat is removed for an extended period of time, as required by the longlived radioactivity remaining in the core.

These criteria were established to provide significant margin in ECCS performance following a LOCA. Reference 15.6.4-2 presents a recent study in the probability of occurrence of RCS pipe ruptures.

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

15.6.4.2 Sequence of Events and Systems Operations

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 actuation signal is generated when the appropriate setpoint is reached. These countermeasures 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.

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

In addition, a high containment pressure signal is generated from the mass and energy released out the break which automatically starts the low head containment spray pumps.

Description of Large Break LOCA Transient

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 system, the secondary side pressure increases, and the main steam safety valves may actuate to limit the pressure. Make-up water to the secondary side is automatically provided by the Secondary Side Safeguards System. The safety injection actuation signal isolates the steam generators from normal feedwater flow and initiates emergency flow from the Secondary Side Safeguards System. The secondary flow aids in the reduction of reactor coolant system pressure.

The high head safety injection pumps will inject borated water whenever the reactor coolant system pressure is below 1800 psia. When the RCS depressurizes to 600 psia, the accumulators begin to inject borated water into the reactor coolant loops. When the RCS depressurize to 200 psia, the core reflood tanks begin to inject borated water into the reactor vessel. Since the loss of off-site power is assumed, the reactor coolant pumps are assumed to trip at the inception of the accident. The effects of pump coastdown are included in the blowdown analysis.

The blowdown phase of the transient ends when the RCS pressure (initially assumed at 2250 psia) falls to a value approaching that of the containment atmosphere. Prior to or at the end of the blowdown, some amount of injection water begins to enter the reactor vessel lower plenum. 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 four core reflood tanks and the high head safety injection pumps aid the filling of the downcomer and subsequently supply water to maintain a full downcomer and complete the reflooding process.

Continued operation of the ECCS pumps supplies water during long-term cooling. Core temperatures have been reduced to long-term steady state levels associated with dissipation of residual heat generation. Water for both short term and long term cooling is supplied from the in containment emergency water storage tank (EWST). The water discharged through the break will initially fill up the reactor cavity and loop compartments and then drain into the EWST. Thus, the EWST water level will always remain above the minimum required level and no switchover from the injection mode to the cold leg recirculation mode is required. The Containment Spray System continues to operate to further reduce containment pressure. Approximately 24 hours after initiation of the LOCA, the ECCS is realigned to supply water to the RCS hot legs in order to eliminate the potential for a high concentration of boric acid concentration in the reactor vessel.

Description of Small Break LOCA Transient

As contrasted with the large break, the blowdown phase of the small break occurs over a longer time period. Thus, for the small break LOCA there are only three characteristic stages, i.e., a gradual blowdown in which the decrease in water level is checked, core recovery, and long-term recirculation.

15.6.4.3 Core and System Performance

15.6.4.3.1 Mathematical Model

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10CFR 50.

Large Break LOCA Evaluation Model

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

The description of the various aspects of the LOCA analysis methodology is given in WCAP-8339, Reference 15.6.4-3. This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the acceptance criteria. The SATAN-VI, WREFLOOD, COCO, and LOCTA-IV codes, which are used in the LOCA analysis, are described in detail in References 15.6.4-4 through 15.6.4-7. These 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 and the WREFLOOD computer code is used to calculate this transient during the refill and reflood phases of the accident. The COCO computer code is used to calculate the containment pressure transient during all three phases of the LOCA analysis. Similiarly, the LOCTA-IV computer code is used to compute the thermal transient of the hottest fuel rod during the three phases.

SATAN-VI is used to calculate the RCS pressure, enthalpy, density and the mass and energy flow rates in the RCS, as well as energy transfer between the primary and steam generator secondary systems as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator mass and internal pressure and the pipe break mass and internal energy flow rates that are assumed to be vented to the Containment during blowdown. During blowdown, no credit is taken for rod insertion or boron content of the injection water. The core will shutdown due to void formation. At the end of the blowdown phase, this data is transferred to the WREFLOOD code. Also at the end of blowdown, the mass and energy release rates during blowdown are transferred to the COCO code for use in the determination of the containment pressure response during this first phase of the LOCA. Additional SATAN-VI output data from the end of blowdown, including the core inlet flow rate and enthalpy, the core pressure, and the core power decay transient, are input to the LOCTA-IV code.

With input from the SATAN-VI code, WREFLOOD uses a system thermal-hydraulic model to determine the core flooding rate (i.e., the rate at which coolant enters the bottom of the core), the coolant pressure and temperature, and the quench front height during the refill and reflood phases of the LOCA. WREFLOOD also calculates the mass and energy flow 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 WREFLOOD and COCO codes are interactively linked. WREFLOOD is also linked to the LOCTA-IV code in that thermalhydraulic parameters from WREFLOOD are used by LOCTA-IV in its calculation of the fuel temperature. During the reflood portion of the transient, the core is assumed to remain in a subcritical condition due to the boron content of the injection water. LOCTA-IV is used throughout the analysis of the LOCA transient to calculate the fuel and clad temperature and metal-water reaction of the hottest rod in the core.

Schematic representations of the computer code interfaces are given in Figure 15.6.4-2.

For the large break LOCA analyses presented here, only the Satan VI, WREFLOOD, and LOCTA IV Computer codes were used. An estimated containment pressure transient was input into the thermal-hyraulic codes due to insufficient data available for containment design.

The flow of steam and entrained water from the core is accounted for through the use of an empirical correlation developed from FLECHT (Reference 15.6.4-8.) experimental results. The significant parameters affecting the amount of steam and water leaving the core during the early part of the reflood phase of the transient are the quench front level, power density, flooding rate, pressure and inlet water subcooling. These parameters have been examined with respect to the FLECHT test data. Comparison of measured and predicted values of the mass carryover fraction for various core reflood rates shows excellent agreement. A complete discussion of this aspect of the analysis and the comparison of measured and predicted values is contained in Reference (15.6.4-5).

The large break analysis was performed with the December, 1981 version of the Evaluation Model, which includes modifications delineated in WCAP-9220-P-A and WCAP-9221-P-A (1981) (See Reference 15.6.4-9).

Also, the analysis in this section was performed with the upper head fluid temperature equal to the RCS cold leg fluid temperature which is achieved by passing a sufficient amount of bypass flow into the upper head.

Small Break LOCA Evaluation Model

The WFLASH program used in the analysis of the small break loss of coolant accident is an extension of the FLASH-4 (Reference 15.6.4-10) code developed at the Westinghouse Bettis Atomic Power Laboratory. The WFLASH (Reference 15.6.4-11) Program permits a detailed spatial representation of the Reactor Coolant System (RCS).

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 WFLASH is given in Reference 15.6.4-11.

The use of WFLASH in the analysis involves, among other things, the representation of the reactor core as a heated control volume with the associated bubble rise model to permit a transient mixture height calculation. The multi-node 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.

Clad thermal analyses are performed with the LOCTA IV Code (Reference 15.6.4-7) which uses the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core and mixture height history from the WFLASH hydraulic calculations as input.

Figure 15.6.4-38 gives the safety injection flowrate for the small break analysis. The one high head pump curve was used for the 3" and 4.313" small break analyses, which assumed a loss of one train of emergency electrical power, and that the break location was at one of the four direct vessel injection nozzles. The inside diameter for the direct vessel injection nozzle is 4.313 inches which is equivalent to the inside diameter of a 5 inch Sch 160 pipe. For breaks larger than 4.313 inches it is assumed that two high head pumps are available and that the break location is in an RCS cold leg. It should be noted that analysis has also been performed for a 6 inch break with only one high head pump delivering coolant to the Reactor Vessel and the results of that an analysis has shown no core uncovery with no credit for accumulator flow.

Figure 15.6.4-39 presents the hot rod power shape utilized to perform the small break analysis presented here. This power shape was chosen because it provides an appropriate distribution of power versus core height and also local power is maximized in the upper regions of the reactor core (10 ft. to 13 ft). This power shape is shewed to the top of the core with the peak local power occurring at the 10.7 core elevation.

This is limiting for the small break analysis because of the core uncovery process for small breaks. If the core uncovers, the cladding in the upper elevation of the core heats up and is sensitive to the local power at that elevation. The cladding temperatures in the lower elevation of the core, below the two phase mixture height, remains low. The peak clad temperature occurs above 10 ft.

Schematic representations of the computer code interfaces are given in Figure 15.6.4-3.

The small break analysis was performed with the approved October, 1975 version of the Westinghouse ECCS Evaluation Model (References 15.6.4-7, 15.6.4-11, and 15.6.4-12).

15.6.4.3.2 Input Parameters and Initial Conditions

Table 15.6.4-1 lists important input parameters and initial conditions used in the analysis.

The analysis was performed with the upper head fluid temperature equal to the reactor coolant system cold leg fluid temperature.

The bases used to select the numerical values that are input parameters to the analysis have been determined from extensive sensitivity studies on Westinghouse four loop NSSS designs (References 15.6.4-13 through 15.6.4-15). In addition, the requirements of Appendix K regarding specific model features were met by selecting models which provide a significant overall conservatism in the analysis. The assumptions made pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs and include such items as the core peaking factors, the containment pressure, and the performance of the ECCS. Decay heat generated throughout the transient is also conservatively calculated.

15.6.4.3.3 Results

Large Break Results

Based on the results of the past LOCA sensitivity studies (Reference 15.6.4-14 and 15.6.4-15), the limiting large break was found to be the double-ended cold leg guillotine (DECLG). Therefore, only the DECLG break is considered in the large break ECCS performance analysis. Calculations were performed for a range of Moody break discharge coefficients. The results of these calculations are summarized in Tables 15.6.4-2 and 15.6.4-3.

Factors affecting break flow in a Westinghouse PWR and the lower limit of break discharge coefficient based on experimental data are discussed in Reference 15.6.4-15. Conclusions in that report are that a best estimate value of the Moody discharge coefficient is about 0.6 and that varying the discharge coefficients from a maximum of 1.0 to a minimum of 0.4 covers all uncertainties associated with the prediction of the break flow in case of a guillotine type severance of a cold leg pipe. The position to limit the break discharge coefficient to that range has been reviewed and approved by the NRC. Therefore, analyzing a LOCA for break discharge coefficients less than 0.4 is not consistent with experimental data or with the established procedure for a 10CFR 50 Appendix K evaluation of ECCS performance.

Figures 15.6.4-4 through 15.6.4-30 present the parameters of principal interest from the large break ECCS analyses. For all cases analyzed transients of the following parameters are presented.

a) Hot spot clad temperature.

b) Coolant pressure in the reactor core.

c) Water level in the core and downcomer during reflood.

d) Core reflooding rate.

e) Thermal power during blowdown.

f) Containment pressure.

For the limiting break analyzed, the following additional transient parameters are presented:

a) Core flow during blowdown (inlet and outlet).

b) Core heat transfer coefficients.

c) Hot spot fluid temperature.

d) Mass released to Containment during blowdown.

e) Energy released to containment during blowdown.

f) Fluid quality in the hot assembly during blowdown.

g) Mass velocity during blowdown.

h) Accumulator water flow rate during blowdown.

i) Pumped safety injection water flow rate during reflood.

The results show that the large break LOCA transient is characterized by three distinct phases: (1) initial heat-up phase due to the core stored energy and the relatively poor positive core flow period leading to the initial clad temperature rise, (2) blowdown cooling phase characterized by influx of water into the core from the large upper plenum and upper head volumes resulting in a significant blowdown cooling effect, and (3) reflood phase with relatively high core velocities driven by water elevation head in vessel downcomer.

The maximum clad temperature calculated for a large break is [ ] which is less than the acceptance criteria limit of 2200°F of 10CFR 50.46. The maximum local metal water reaction is less than 1.0 percent which is well below the embrittlement limit of 17 percent as required by 10CFR 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 CRF 50.46, and the clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. No fuel clad damage occurs. 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. (a,c)

Small Break Results

As noted previously, the calculated peak clad temperature resulting from a small break LOCA is less than that calculated for a large break. The limiting small break was found to be less than a 10 in. diameter rupture of the RCS cold leg. A range of small break analyses are presented which establishes the limiting small break. The results of these analyses are summarized in Tables 15.6.4-4 and 15.6.4-5.

Figures 15.6.4-31 through 15.6.4-39 present the principal parameters of interest for the small break ECCS analyses. For all cases analyzed the following transient parameters are presented:

a) RCS pressure

b) Core mixture height,

c) Core power after reactor trip.

No core uncovering occurred for the small breaks analyzed. These results are well below all acceptance criteria limits of 10 CFR 50.46 and in all cases are not limiting when compared to the results presented for large breaks.

15.6.4.4 Radiological Consequences

The results of the analyses presented in this section demonstrate that the radioactivity released to the environment by a LOCA does not result in doses exceeding the limits specified in 10 CFR 100. The dose calculations take into account radioactivity released to the environment by containment leakage of gases, by leakage of the recirculating sump solution, and by containment purge at the beginning of the accident.

The major assumptions and parameters assumed in the analysis are itemized in Table 15.6.4-6.

In the licensing basis evaluation of a LOCA, the fission product release assumptions of Regulatory Guide 1.4 have been followed with some exceptions. Table 15.6.4-7 provides a comparison of the analysis to the recommendations of Regulatory Guide 1.4.

The mathematical models used to calculate the activity releases during the course of the accident and the resultant doses are described in Appendix 15A.

15.6.4.4.1 Fission Product Release to the Containment

Following a postulated double-ended rupture of a reactor coolant pipe with subsequent blowdown, the ISS limits the fuel clad temperature to well below the melting point and ensures that the reactor core remains intact and in a coolable geometry, thus minimizing the release of fission products to the containment. However, to demonstrate that the operation of a nuclear power plant does not represent an undue radiological hazard to the general public, a hypothetical accident involving a significant release of fission products to the containment is evaluated.

Both the licensing and the Westinghouse dose evaluations assume that 100 percent of the noble gases and 50 percent of the iodine equilibrium core fission product inventory is immediately released to the containment. The volatile iodine and the noble gas activity are assumed to be immediately airborn and available for leakage from the containment. The iodine source terms are described in Section 6.5.2.1.

15.6.4.4.2 Fission Product Release Due to Containment Leakage

Once the gaseous fission product activity is released to the containment atmosphere, it is subject to various mechanisms of removal which operate simultaneously to reduce the amount of activity in the containment atmosphere. The removal mechanisms include radioactive decay, containment sprays, deposition and containment leakage. For the noble gas fission products, the only removal processes considered in the containment are radioactive decay and containment leakage. Credit for radioactive decay of fission products located within the containment is assumed throughout the course of the accident. Once the activity is released to the environment, no credit is taken for radioactive decay or deposition. The containment leakage to the environment is assumed to be direct and unfiltered.

The quantity of activity released through leakage from the containment was calculated with a two-volume model of the containment to represent sprayed and unsprayed regions of the containment. This model is discussed in Appendix 15.A.

Of the total free volume of the containment, part is covered by the containment spray, while some is not. The unsprayed fraction has been estimated to be 20 percent.

The transfer rate between the sprayed and unsprayed regions is assumed to be limited to the forced convection induced by the fan cooler units. The number of units assumed in operation and the total mixing flow are presented in Table 15.6.4-6. This assumed minimum flowrate conservatively neglects the effects of natural convection, spray induced turbulence, steam condensation, and diffusion, although these effects are expected to enhance the mixing rate between the sprayed and unsprayed volumes.

For fission products other than iodine, the only removal processes considered are radioactive decay and leakage. Iodine is assumed to be removed not only by radioactive decay and leakage, but also by deposition and by the containment sprays. The effectiveness of the containment spray for the removal of the iodine in the containment atmosphere and the model used to determine the iodine removal efficiency are discussed in Section 6.5.2. The iodine removal constants are given in Table 15.6.4-6.

Limits assumed for iodine removal by containment spray and deposition are detailed in Table 15.6.4-6.

Release from the containment by containment leakage is assumed to be 0.1 percent per day for the first 24 hours and 0.05 percent per day thereafter. The offsite doses at the site boundary and at the low population zone and the doses to control room personnel are given in Table 15.6.4-8.

15.6.4.4.3 <u>Fission Product Release Due to Containment Purge Operation</u>

During normal power operation the containment purge system (described in Section 6.2) is operating, venting the containment at 5000 cfm. In the event of a LOCA, the purge system supply and exhaust isolation valves are assumed to close within five seconds of receiving a containment isolation signal as designed.

The containment airborne fission product inventory available for release is based on 100 percent of the total primary coolant iodine inventory assuming a pre-accident iodine spike level of 60 μCi/gm dose equivalent I-131 and 100 percent of the primary coolant noble gas inventory assuming the reactor has been operating with 1 percent fuel defects (i.e., defects in the cladding of fuel rods generating 1 percent of the core rated power). No credit is taken for removal of iodine by the purge filter train.

The offsite doses at the site boundary and at the low population zone and the doses to control room personnel are given in Table 15.6.4-8.

15.6.4.4.4 Radioactive Releases Resulting from Leakage from ISS Recirculation Lines

During ISS ECCS operational mode, the water from the in-containment emergency water storage tank (EWST) is injected into the RCS via the high head pumps and spilling fluid from the RCS is collected in the EWST. Because a large fraction of core iodines may be released as a result of the LOCA, and would be retained in this "recirculated" water; the radiological consequence of any postulated leakage from those portions of the ISS located outside containment exposed to this fluid must be examined. Each of the four ISS subsystems is contained in a separate containment pressure pump (CPPE) compartment adjoining the reactor containment. These pump compartments are designed to minimize the release to the environment of any radioactive material resulting from leakage in the compartments, and return it to the containment. Therefore, radiological consequences from this source are virtually eliminated.

15.6.4.4.5 Identification of Uncertainties and Conservatisms in the Analysis

The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of a LOCA result principally from assumptions made involving the amount of the gaseous fission products available for release to the environment and the meteorology present at the site during the course of the accident. The most significant of these assumptions are:

A. The ISS is designed to prevent fuel cladding damage that would release the fission products, contained in the fuel, to the reactor coolant. Severe degradation of the ISS (simultaneous multiple failures of redundant components) would be necessary to release the quantity of fission products assumed in the analysis.

B. The release of fission products to the containment is assumed to occur instantaneously.

C. The activity released to the containment atmosphere is assumed to leak to the environment at the containment leakage rate of 0.1 volume percent/day for the first 24 hours and 0.05 volume percent/day thereafter. The initial containment leakage rate is based on the peak calculated internal containment pressure anticipated after a LOCA. The pressure within the containment actually decreases with time. Taking into account that the containment leak rate is a function of pressure, the calculated doses could be reduced significantly.

D. The meteorological conditions assumed to be present at the site during the course of the accident are based on X/Q values which are worse than those which will exist at the site 95 percent of the time. This condition results in the poorest values of atmospheric dispersion calculated for the exclusion area boundary and the low population zone outer boundary. Furthermore, no credit has been taken for the transit time required for activity to travel from the point of release to the exclusion area boundary and to the low population zone outer boundary. Hence, the radiological consequences evaluated under these conditions are conservative.

15.6.4.4.6 Conclusions

15.6.4.4.6.1 Filter Loadings

No recirculating or single-pass filters are used for fission product cleanup and control within the containment following a postulated LOCA. The only engineered safety features filtration systems expected to be operating under post-LOCA conditions are the control room heating, ventilation, and air-conditioning (HVAC) system (Reference 15.6.4-16) and the safeguard component area air cleanup system.

Activity loadings on the control room charcoal adsorbers are based on the flowrate through the adsorber, the concentration of activity at the adsorber inlet, and the adsorber efficiency. Based on the radioactive iodine release assumptions previously described, the assumption that 50 percent of the core inventory of isotopes I-127 and I-129 is available for release from the containment atmosphere, and the assumption that the charcoal adsorber is 100 percent efficient, the calculated filter loadings are in accordance with Regulatory Guide 1.52, which limits the maximum loading to 2.5 milligrams of iodine per gram of activated charcoal.

15.6.4.4.6.2 Doses to a Receptor at the Exclusion Area Boundary and Low Population Zone Outer Boundary

The potential radiological consequences resulting from the occurrence of the postulated LOCA have been conservatively analyzed, using assumptions and models described in previous sections.

The total body immersion dose and the thyroid inhalation dose have been analyzed for the 0 to 2 hour dose at the exclusion area boundary and for the duration of the accident at the low population zone. The results are listed in Table 15.6.4-3. The resultant doses are within the guideline values of 10 CFR 100.

15.6.4.4.6.3 Doses to the Control Room Personnel

Radiation doses to control room personnel following a postulated LOCA are based on the ventilation, cavity dilution, and dose model discussed in Section 15A.3.

Control room personnel are subject to a total body immersion dose and a thyroid inhalation dose. These doses have been analyzed and are provided in Table 15.6.4-8. The resultant doses are within the limits established by GDC-19.

15.6.7 References

15.6.4-1 "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10CFR 50.46 and Appendix K of 10CFR 50. Federal Register, Volume 39, Number 3, January 4, 1974.

15.6.4-2 "Reactor Safety Study - An Assessment of Accident Risk in U.S. Commercial Nuclear Power Plants," WASH-1400, NUREG 75/014, October, 1975.

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TABLE 15.6.4-1

INPUT PARAMETERS USED IN THE ECCS ANALYSIS

(a,c)

Core Power*	
Peak linear power (includes 102% factor	
Total peaking factor, F_Q	
Power shape Large break-chopped cosine	
Small break-see Figure 15.6.4-39	
Full assembly array	
Accumulator water volume (nominal)	
Accumulator tank volume (nominal)	
Accumulator gas pressure (minimum)	
Core reflood tank water volume minimal	
Core reflood tank volume	
Core reflood tank gas pressure	
Safety injection pumped flow	
Initial loop flow	
Vessel inlet temperature	
Vessel outlet temperature	
Reactor coolant pressure	
Steam pressure	
Steam generator tube plugging level	

* 2% is added to this power level to account for calorimetric error.

TABLE 15.6.4-2

LARGE BREAK - TIME SEQUENCE OF EVENTS

Event	Occurrence Time (second) DECLG, C_D = .6	DECLG, C_D = 0.8	DECLG, C_D = 1.0
Accident initiation			(a,c)
Reactor trip signal			
S.I. Actuation signal			
Start Accumulator injection			
Start CRT injection			
Start pumped ECC injection			
End of ECC bypass			
End of blowdown			
Bottom of core recovery			
Accumulators empty			

TABLE 15.6.4-3

LARGE BREAK RESULTS

(a,c

Results	DECLG, C_D = .6	DECLG, C_D = 0.8	DECLG, C_D = 1.0
Peak clad temperature (F)			
Location (ft)			
Maximum local clad/water reaction (%)			
Location (ft)			
Total core clad/water reaction (%)			
Hot rod burst time (seconds)			
Location (ft)			

TABLE 15.6.4-4 (Sheet 1 of 3)
(a)

3 INCH SMALL BREAK LOCA TIME SEQUENCE OF EVENTS

Event	Time (sec)
Break occurs	0.0
Reactor trip signal, SI signal, loss of A/C power to reactor coolant pumps	74.9
SI injection flow begins	134.9
Minimum reactor vessel water level reached,* vessel begins refilling	1475
Accumulator setpoint reached**	> 1800

* No core uncovery occurs.

** No credit for accumulator flow was assumed in the analysis.

TABLE 15.6.4-4 (Sheet 2 of 3)
(b)

4.313 INCH SMALL BREAK LOCA TIME SEQUENCE OF EVENTS

Event	Time (sec)
Break occurs	0.0
Reactor trip signal, SI signal, loss of A/C power to reactor coolant pumps	40.2
SI injection flow begins	65.2
Accumulator setpoint reached*	1037
Minimum reactor vessel water level reached, vessel begins refilling**	1176

* No credit for accumulator flow was assumed in the analysis.

** No core uncovery occurs.

TABLE 15.6.4-4 (Sheet 3 of 3)
(c)

6 INCH SMALL BREAK LOCA TIME SEQUENCE OF EVENTS

Event	Time (sec)
Break occurs	0.0
Reactor trip signal, SI signal, loss of A/C power to reactor coolant pumps	22.2
SI injection flow begins	82.2
Minimum reactor vessel water level reached, vessel begins refilling*	169
Accumulator setpoint reached**	453

* No core uncovery occurs.

** No credit for accumulator flow was assumed in the analysis.

TABLE 15.6.4-5

SMALL BREAK LOCA RESULTS

Results	3 in.	4.313 in.	6 in.
Peak clad temp., F	N/A	N/A	N/A
Peak clad location, ft	N/A	N/A	N/A
Local Zr/H_2O reaction, (max)%	N/A	N/A	N/A
Local Zr/H_2O location, ft	N/A	N/A	N/A
Local Zr/H_2O reaction, %	N/A	N/A	N/A
Hot rod burst time, sec	N/A	N/A	N/A
Hot rod burst location, ft	N/A	N/A	N/A
Minimum mixture level above the bottom of the core (ft)*	19.5	16.1	20.9
	(no core uncovering occurs)		

* Active fuel length equals 12' 9".

TABLE 15.6.4-6 (Sheet 1 of 3)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL
CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

Source Data	Licensing Basis	Westinghouse Basis
Core power level (MWt)	3800	3800
Burnup, full power (days)	900	900
Core activity initially released to the containment (%)		
Noble gas	100	100
Iodine	50	50
Core inventories	Table 15A-3	Table 15A-3
Iodine distribution (%)		
Elemental	95.5	48*
Organic	2	2
Particulate	2.5	2.5
Atmospheric Dispersion Factors	Table 15A-2	Table 15A-2
Control Room Parameters	Tables 15A-1 and 15A-2	Table 15A-1 and 15A-2
Containment Leakage of Activity		
Containment leak rate (volume %/day)		
0 to 24 hours	0.1	0.1
1 to 30 days	0.05	0.05
Credit for deposition		
Deposition iodine removal constants, elemental iodine only ($hour^{-1}$)		
Sprayed volume	2.4**	2.4**
Unsprayed volume	1.9**	1.9**

* Remaining iodine is composed of nonvolatile iodides.

** Value assumed to be 0.0 after a decontamination factor of 200 is achieved through the combined effects of all iodine removal mechanisms.

TABLE 15.6.4-6 (Sheet 2 of 3)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

Credit for containment sprays

Spray iodine removal constants ($hour^{-1}$)

Elemental	5.5**
Organic	0.0
Particulate	1.3

Duration of sprays (hours)	2.0	
Sprayed volume (percent)	80	
Unsprayed volume (percent)	20	
No. of fan coolers operating	2	
Sprayed-unsprayed mixing rate (ft^3/min)	[90,000	(a,c)
Containment volume (ft^3)	]	

Activity released to containment atmosphere from the core

Isotope	Licensing Basis Curies	Westinghouse Basis Curies
I-131	4.9×10^7	2.6×10^7
I-132	7.2×10^7	3.8×10^7
I-133	1.0×10^8	5.3×10^7
I-134	1.1×10^8	5.7×10^7
I-135	9.4×10^7	4.9×10^7
Xe-131m	7.0×10^5	7.0×10^5
Xe-133m	2.9×10^7	2.9×10^7
Xe-133	1.9×10^8	1.9×10^8
Xe-135m	4.0×10^7	4.0×10^7
Xe-135	4.2×10^7	4.2×10^7
Xe-138	1.6×10^8	1.6×10^8
Kr-85m	2.7×10^7	2.7×10^7
Kr-85	6.6×10^5	6.6×10^5
Kr-87	4.9×10^7	4.9×10^7
Kr-88	2.0×10^7	2.0×10^7

* Value assumed to be 0.0 after a decontamination factor of 100 is achieved for elemental iodine.

** Value assumed to be 0.0 after a decontamination factor of 200 is achieved through the combined effects of all iodine removal mechanisms.

TABLE 15.6.4-6 (Sheet 3 of 3)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

Containment Purge of Activity

Purge flow rate (cfm)	5000
Duration of purge, from accident initiation, (sec)	5
Reactor coolant iodine activity (μCi/gm I-131 dose equivalent)	60*
Reactor coolant noble gas activity (% defects)	1**
Reactor coolant activity airborne in the containment (%)	
Noble gas	100
Iodine	100

Activity released to the containment atmosphere from the reactor coolant (for both Westinghouse basis and Licensing basis)

Isotope	Curies
I-131	1.05×10^4
I-132	1.05×10^4
I-133	1.58×10^4
I-134	2.69×10^3
I-135	8.66×10^3
Xe-131m	5.05×10^2
Xe-133m	3.91×10^3
Xe-133	6.21×10^4
Xe-135m	1.09×10^2
Xe-135	1.66×10^3
Xe-138	1.49×10^2
Kr-85m	4.60×10^2
Kr-85	1.68×10^3
Kr-87	2.98×10^2
Kr-88	8.29×10^2

* See Table 15A-5 for specific nuclide concentrations.
** See Table 15A-7 for specific nuclide concentrations.

TABLE 15.6.4-7 (SHEET 1 OF 7)

DESIGN COMPARISON TO THE REGULATORY POSITIONS OF REGULATORY GUIDE 1.4, ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT FOR PRESSURIZED WATER REACTORS, REVISION 2, JUNE 1974

Regulatory Guide 1.4 Position	Design
1. The assumptions related to the release of radioactive material from the fuel and containment are as follows:	
a. Twenty-five percent of the equilibrium radioactive iodine inventory developed from maximum full power operation of the core should be assumed to be immediately available for leakage from the primary reactor containment. Ninety-one percent of this 25 percent is to be assumed to be in the form of elemental iodine; 5 percent of this 25 percent in the form of particulate iodine; and 4 percent of this 25 percent in the form of organic iodides.	Fifty percent of core inventory of iodine is assumed to be immediately available for leakage from the containment The iodine is assumed to be 95.5% elemental, 2.5% particulate and 2% organic. These assumptions are in accordance with Section 6.5.2 of NUREG-0800.
b. One hundred percent of equilibrium radioactive noble gas inventory developed from maximum full power operation of the core should be assumed to be immediately available for leakage from the reactor containment.	Conforms.
c. The effects of radiological decay during holdup in the containment or other buildings should be taken into account.	Conforms. Credit for radioactive decay is taken until the activity is assumed to be released.
d. The reduction in the amount of radioactive material available for leakage to the environment by contairment sprays, recirculating filter systems, or other engineered safety features may be taken into account, but the amount of reduction in concentration of radioactive materials should be evaluated on an individual case basis.	Conforms.

Regulatory Guide 1.4 Position	Design
e. The primary reactor containment should be assumed to leak at the leak rate incorporated or to be incorporated as a technical specification requirement at peak accident pressure for the first 24 hours and at 50 percent of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the Technical Specifications for containment leak testing.	Conforms.
2. Acceptable assumptions for atmospheric diffusion and dose conversion are:	
a. The 0 to 8 hour ground level release concentrations may be reduced by a factor ranging from 1 to a maximum of 3 (see Figure 1) for additional dispersion produced by the turbulent wake of the reactor building in calculating potential exposures. The volumetric building wake correction, as defined in Section 3-3.5.2 of Meteorology and Atomic Energy 1968, should be used only in the 0 to 8 hour period; it is used with a shape factor of 1/2 and the minimum cross-sectional area of the reactor building only.	Short-term accident atmospheric dispersion factors were calculated based on onsite meteorological measurement program described in Section 2.3. These factors are for ground level releases and are based on Regulatory Guide 1.145 methodology and represent the worst of the 5 percent site meteorology and the 0.5 percent worst sector meteorology.
b. No correction should be made for depletion of the effluent plume of radioactive iodine resulting from deposition on the ground or for the radiological decay of iodine in transit.	Same as response to 2a above.
c. For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.47×10^{-4} m^3/s. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.75×10^{-4} m^3/s. After that, until the end of the accident, the rate should be assumed to be 2.32×10^{-4} m^3/s. (These	Conforms.

Regulatory Guide 1.4 Position	Design
values were developed from the average daily breathing rate (2×10^7 cm^3/day) assumed in the report of ICRP, Committee II-1959.)	
d. The iodine dose conversion factors are given in ICRP Publication 2, Report of Committee II, Permissible Dose for Internal Radiation, 1959.	The dose conversion factors provided in Regulatory Guide 1.109 are used.
e. External whole body doses should be calculated using "infinite cloud" assumptions; i.e., the dimensions of the cloud are assumed to be large compared to the distance that the gamma rays and beta particles travel. "Such as a cloud would be considered an infinite cloud for a receptor at the center because any additional (gamma and) beta emitting material beyond the cloud dimensions would not alter the flux of (gamma rays and) beta particles to the receptor" (Meteorology and Atomic Energy, Section 7.4.1.1; editorial additions made so that gamma and beta emitting material could be considered). Under these conditions the rate of energy absorption per unit volume is equal to the rate of energy released per unit volume. For an infinite uniform cloud containing χ curies of beta radioactivity per cubic meter the beta dose in air at the cloud center is:	The dose factors in Regulatory Guide 1.109 are used.

$$_{\beta}D'_{\infty} = 0.457\ \bar{E}_{\beta}\ \chi$$

The surface body dose rate from beta emitters in the infinite cloud can be approximated as being one-half this amount (i.e.,

$$_{\beta}D'_{\infty} = 0.23\ \bar{E}_{\beta}\ \chi$$

TABLE 15.6.4-7 (SHEET 4 OF 7)

Regulatory Guide 1.4 Position	Design
For gamma emitting material, the dose rate in air at the cloud center is:	

$$D'_\infty = 0.507\,\bar{E}\,\chi$$

From a semi-infinite cloud, the gamma dose rate in air is:

$$D'_\infty = 0.25\,\bar{E}\,\chi$$

where:

$_\epsilon D'_\infty$ = Beta dose rate from an infinite cloud (rad/s).

D'_∞ = Gamma dose rate from an infinite cloud (rad/s).

$\bar{E}_\epsilon$ = Average beta energy per disintegration (Mev/dis).

$\bar{E}$ = Average gammer energy per disintegration (Mev/dis).

χ = Concentration of beta of gamma emitting isotope in the cloud (curie/m^3).

f. The following specific assumptions are acceptable with respect to the radioactive cloud dose calculations:

TABLE 15.6.4-7 (SHEET 5 OF 7)

Regulatory Guide 1.4 Position	Design
(1) The dose at any distance from the reactor should be calculated based on the maximum concentration in the plume at that distance, taking into account specific meteorological, topographical, and other characteristics which may affect the maximum plume concentration. These site related characteristics must be evaluated on an individual case basis. In the case of beta radiation, the receptor is assumed to be exposed to an infinite cloud at the maximum ground level concentration at that distance from the reactor. In the case of gamma radiation, the receptor is assumed to be exposed to only one-half of the cloud owing to the presence of the ground. The maximum cloud concentration always should be assumed to be at ground level.	See response to 2e above.
(2) The appropriate average beta and gamma energies emitted per disintegration, as given in the Table of Isotopes, Sixth Edition, by C. M. Lederer, J. M. Hollander, and I. Perlman; University of California, Berkeley, Lawrence Radiation Laboratory, should be used.	See response to 2e above.
g. The atmospheric diffusion model should be as follows:	
(1) The basic equation for atmospheric diffusion from a ground level point source is: $\chi/Q = \frac{1}{\pi u \sigma_y \sigma_z}$ where: χ = The short term average centerline value of ground level concentration (curie/m^3).	Short-term accident atmospheric dispersion factors were calculated based on onsite meteorological measurements program These factors are for ground described in Section 2.3. based on Regulatory Guide 1.145.

TABLE 15.6.4-7 (SHEET 6 OF 7)

Regulatory Guide 1.4 Position	Design
Q = Amount of material released (curie/s). u = Windspeed (m/s). σ_y = The horizontal standard deviation of the plume (m). (See Figure V-1, page 48, Nuclear Safety, June 1961, Volume 2, Number 4, Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion, F. A. Gifford, Jr.) σ_z = The vertical standard deviation of the plume (m). (See Figure V-2, page 48, Nuclear Safety, June 1961, Volume 2, Number 4, Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion, F. A. Gifford, Jr.)	methodology and represent the worst of the 5 percent site meteorology and the 0.5 percent worst sector meteorology.
(2) For time period of greater than 8 hours the plume should be assumed to meander and spread uniformly over a 22.5° sector. The resultant equation is: $x/Q = \frac{2.032}{x_z ux}$ where: x = Distance from point of release to the receptor; other variables are given in 2g(1).	See response to 2g(1) above.

TABLE 15.6.4-7 (SHEET 7 OF 7)

Regulatory Guide 1.4 Position

(3) The atmospheric diffusion model[2] for ground level release is based on the information below:

See response to 2g(1) above. 2g(1) above.

Time Following Accident	Atmospheric Conditions	Design
0 to 8 hours	Pasquill type F, windspeed 1 m/s, uniform direction	
8 to 24 hours	Pasquill type F, windspeed 1 m/s, variable direction within a 22.5° sector	
1 to 4 days	(a) 40 percent Pasquill type D, windspeed 3 m/s	
	(b) 60 percent Pasquill type F, windspeed 2 m/s	
	(c) wind direction variable within a 22.5° sector	
4 to 30 days	(a) 33.3 percent Pasquill type C, windspeed 3 m/s	
	(b) 33.3 percent Pasquill type D, windspeed 3 m/s	
	(c) 33.3 percent Pasquill type F, windspeed 2 m/s	
	(d) windspeed direction 33.3 percent frequency in a 22.5° sector	

TABLE 15.6.4-8

DOSES RESULTING FROM A LOSS-OF-COOLANT ACCIDENT

Site Boundary Dose (0-2 hr)	Licensing Basis	Westinghouse Basis
a. Containment leakage		
thyroid, rem	61.5	39.2
gamma-body, rem	0.8	0.8
beta-skin, rem	0.4	0.4
b. Containment purge		
thyroid, rem	0.2	0.2
gamma-body, rem	3.3×10^{-5}	3.3×10^{-5}
beta-skin, rem	3.0×10^{-5}	3.0×10^{-5}
c. Total		
thyroid, rem	61.7	39.4
gamma-body, rem	0.8	0.8
beta-skin, rem	0.4	0.4
Low Population Zone (0-30 days)		
a. Containment leakage		
thyroid, rem	49.7	38.5
gamma-body, rem	0.5	0.5
beta-skin, rem	0.3	0.3
b. Containment purge		
thyroid, rem	0.08	0.08
gamma-body, rem	1.3×10^{-5}	1.3×10^{-5}
beta-skin, rem	1.2×10^{-5}	1.2×10^{-5}
c. Total		
thyroid, rem	49.8	38.6
gamma-body, rem	0.5	0.5
beta-skin, rem	0.3	0.3
Control Room (0-30 days)		
a. Containment leakage		
thyroid, rem	9.3	7.9
gamma-body, rem	1.1	1.1
beta-skin, rem	17.8	17.8
b. Containment purge		
thyroid, rem	0.006	0.006
gamma-body, rem	2.5×10^{-5}	2.5×10^{-5}
beta-skin, rem	5.2×10^{-4}	5.2×10^{-4}
d. Total		
thyroid, rem	9.3	7.9
gamma-body, rem	1.1	1.1
beta-skin, rem	17.8	17.8

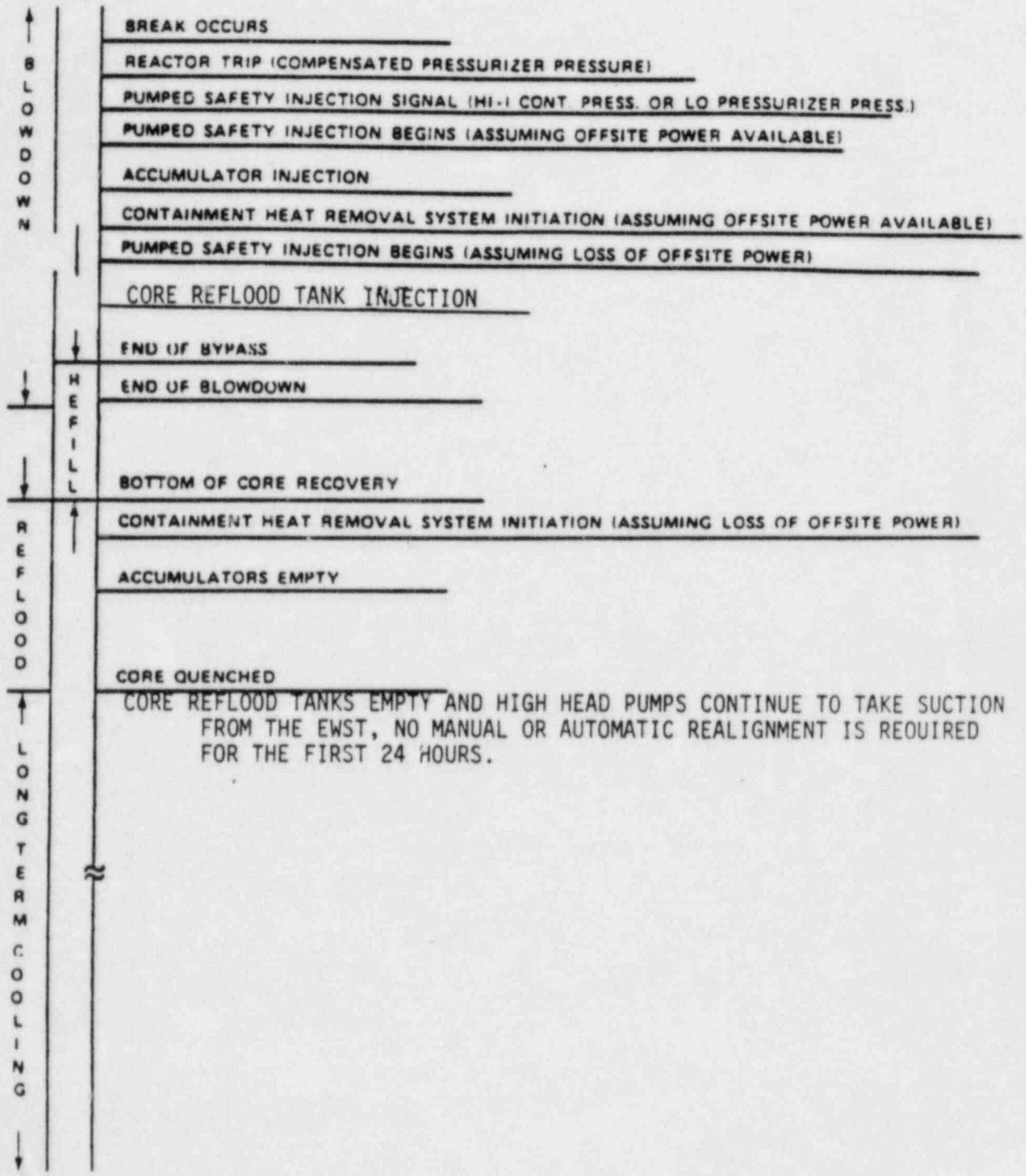


Figure 15.6.4-1 Sequence of Events for Large Break Loss-of-Coolant Analysis

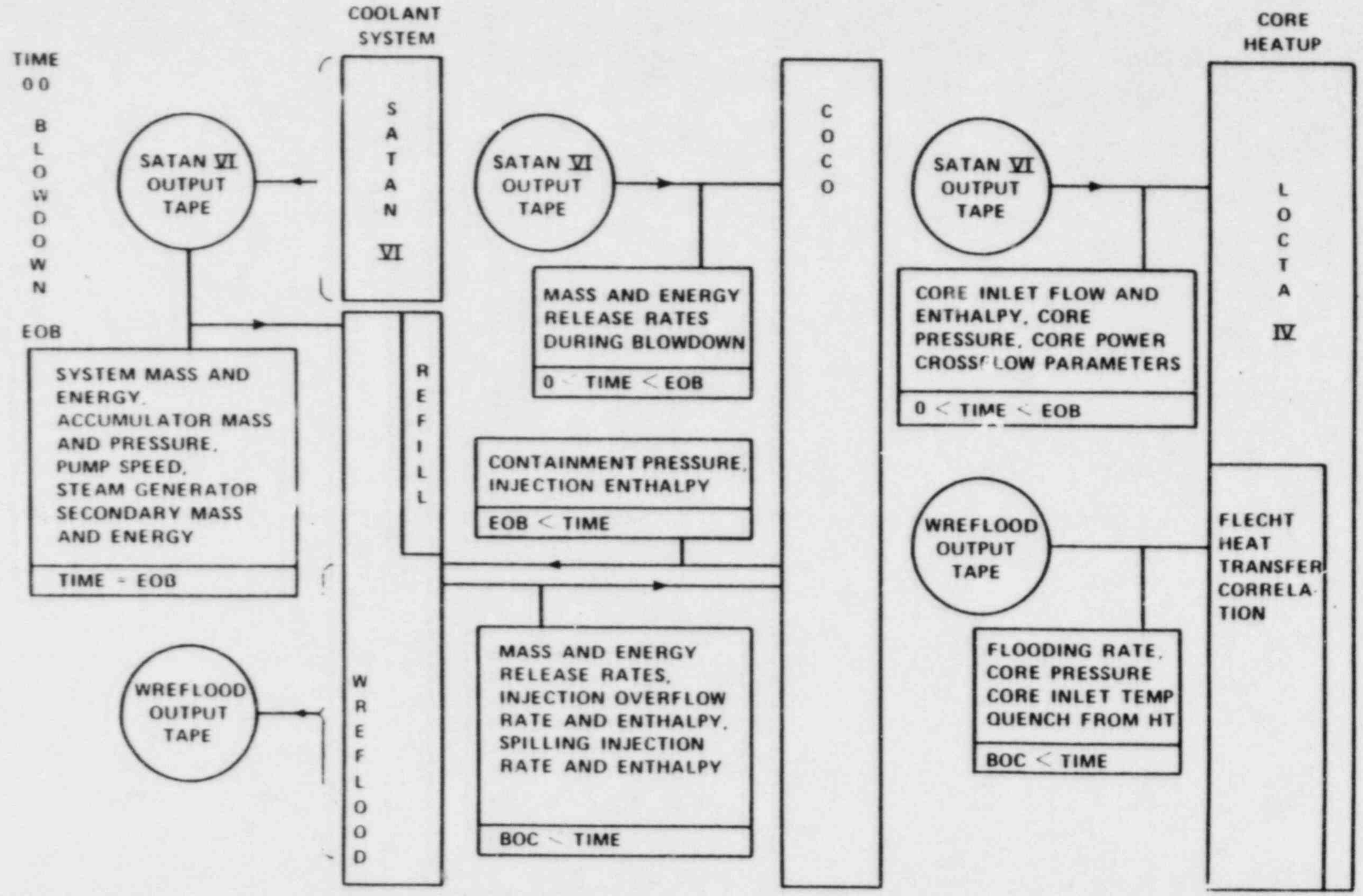


Figure 15.6.4-2 Code Interface Description for Large Break Model

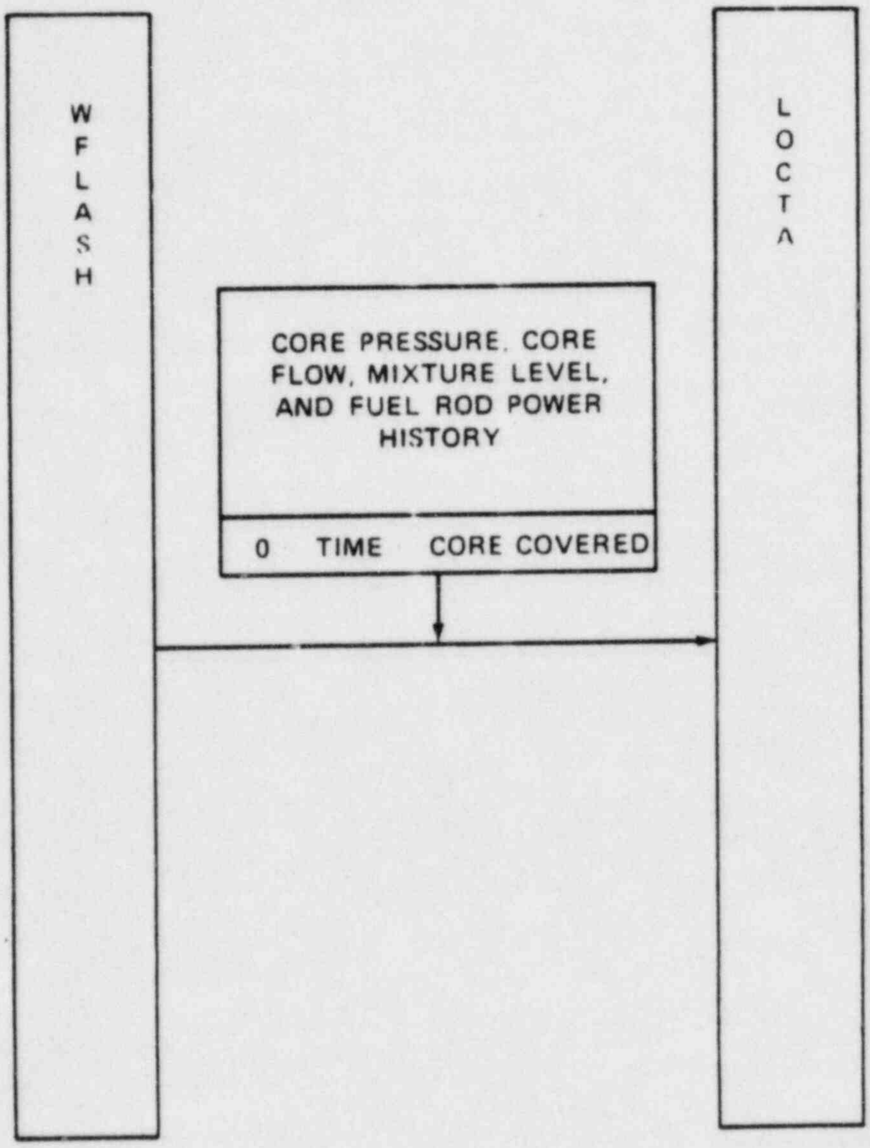


Figure 15.6.4-3 Code Interface Description for Small Break Model

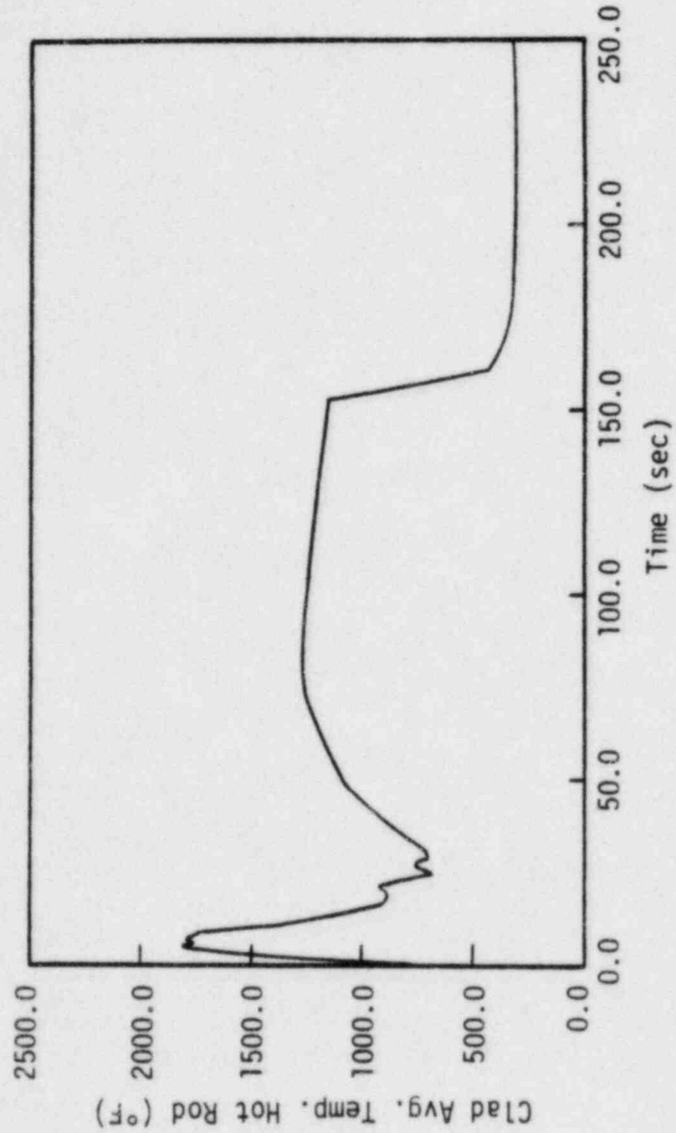


Figure 15.6.4-4 Peak Clad Temperature, Elevation 6.5 Feet DECLG (C_D = 1.0)

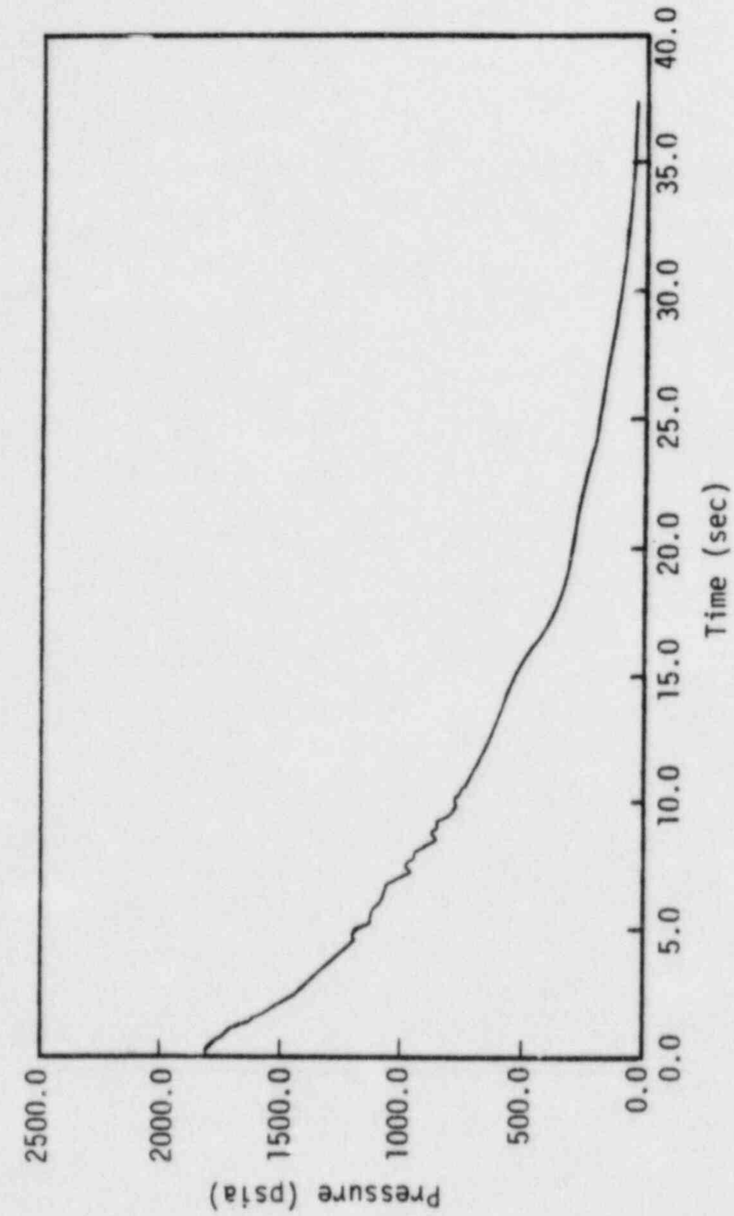


Figure 15.6.4-5 Core Pressure
DECLG (C_D = 1.0)

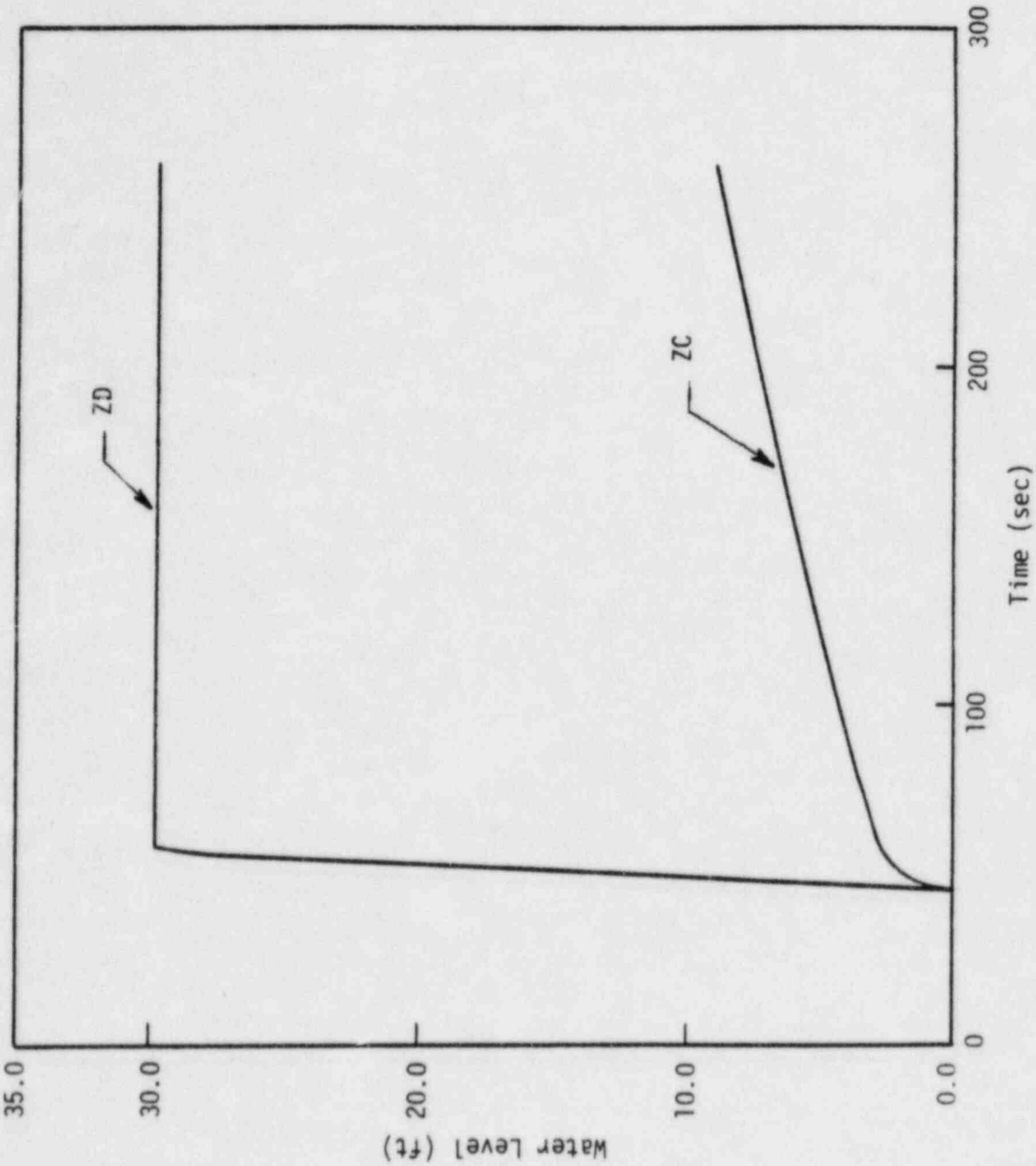


Figure 15.6.4-6 Reflood Transient - Core and Downcomer Water Levels DECLG (C_D = 1.0)

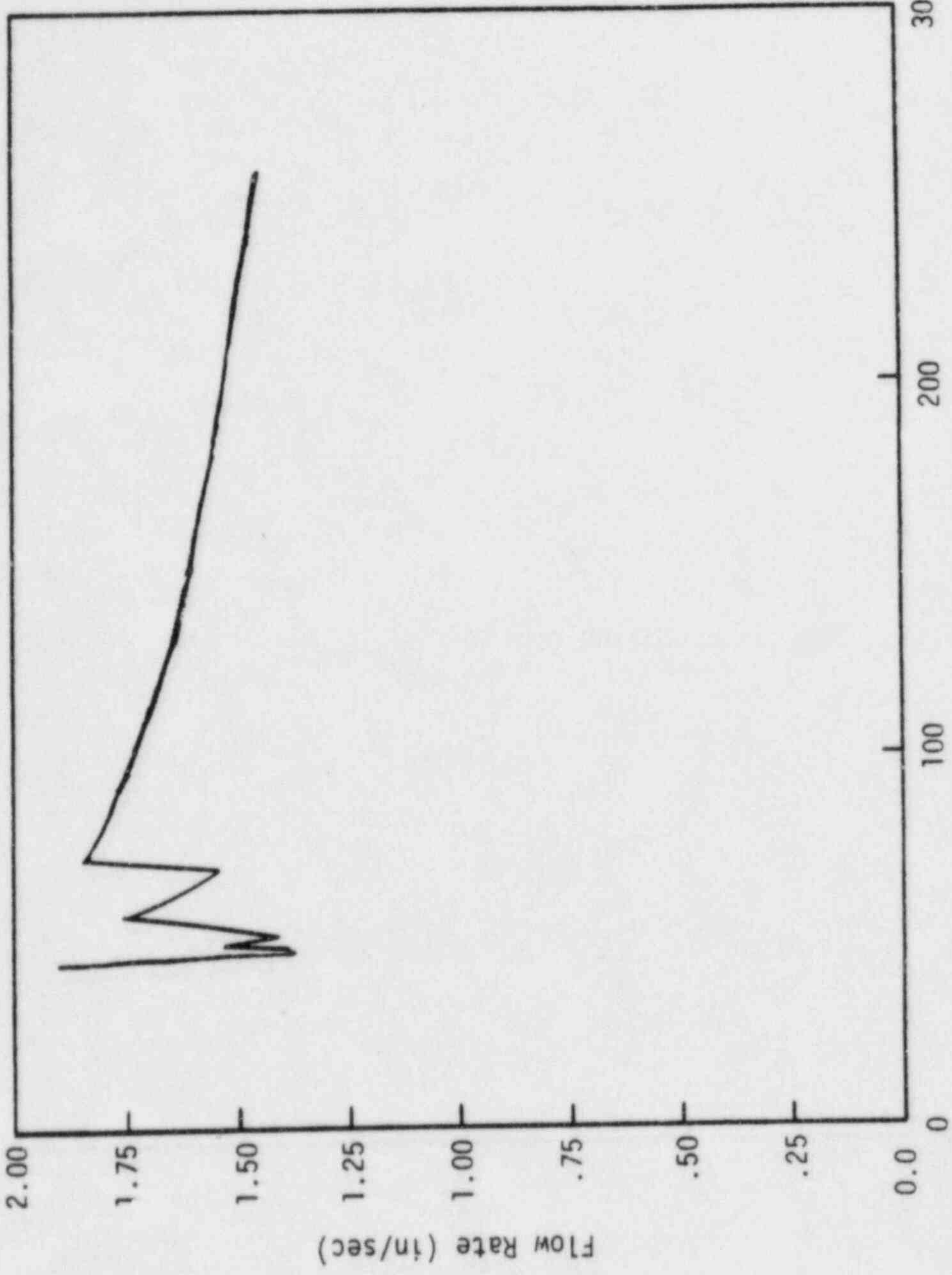


Figure 15.6.4-7 Reflood Transient Core Inlet Velocity DECLG (C_D = 1.0)

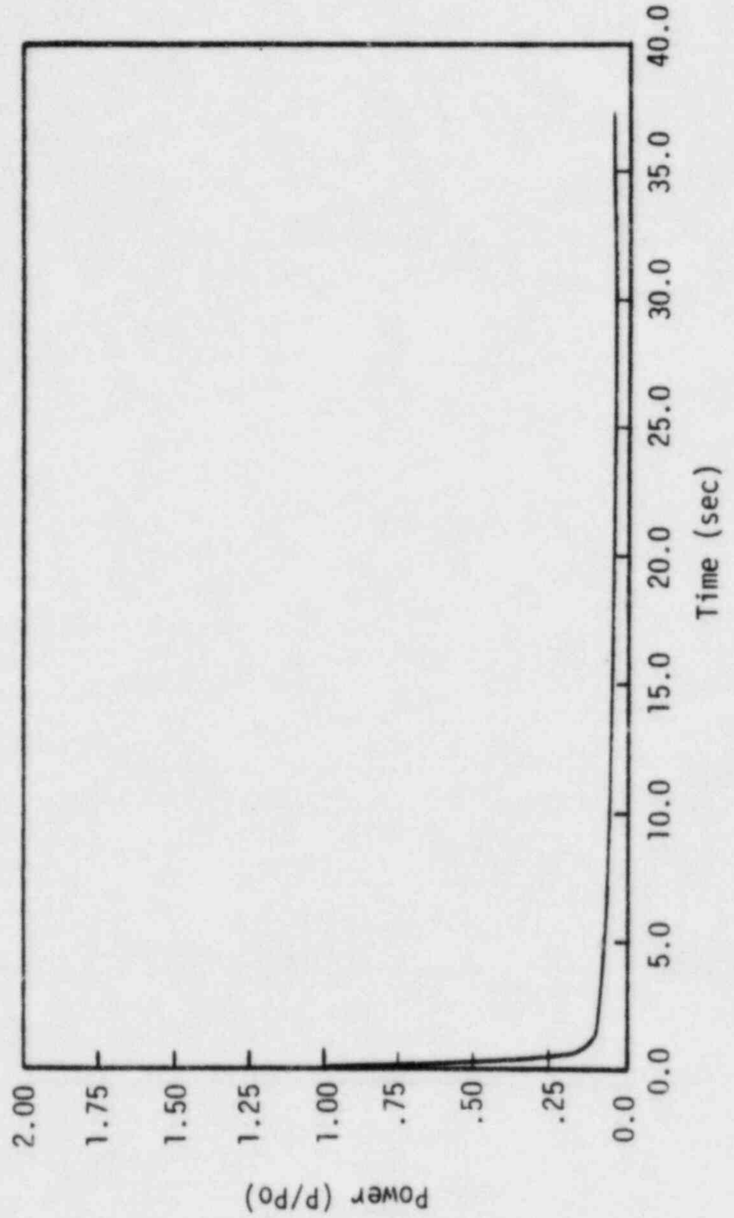


Figure 15.6.4-8 Core Power Transient DECLG (C_D = 1.0)

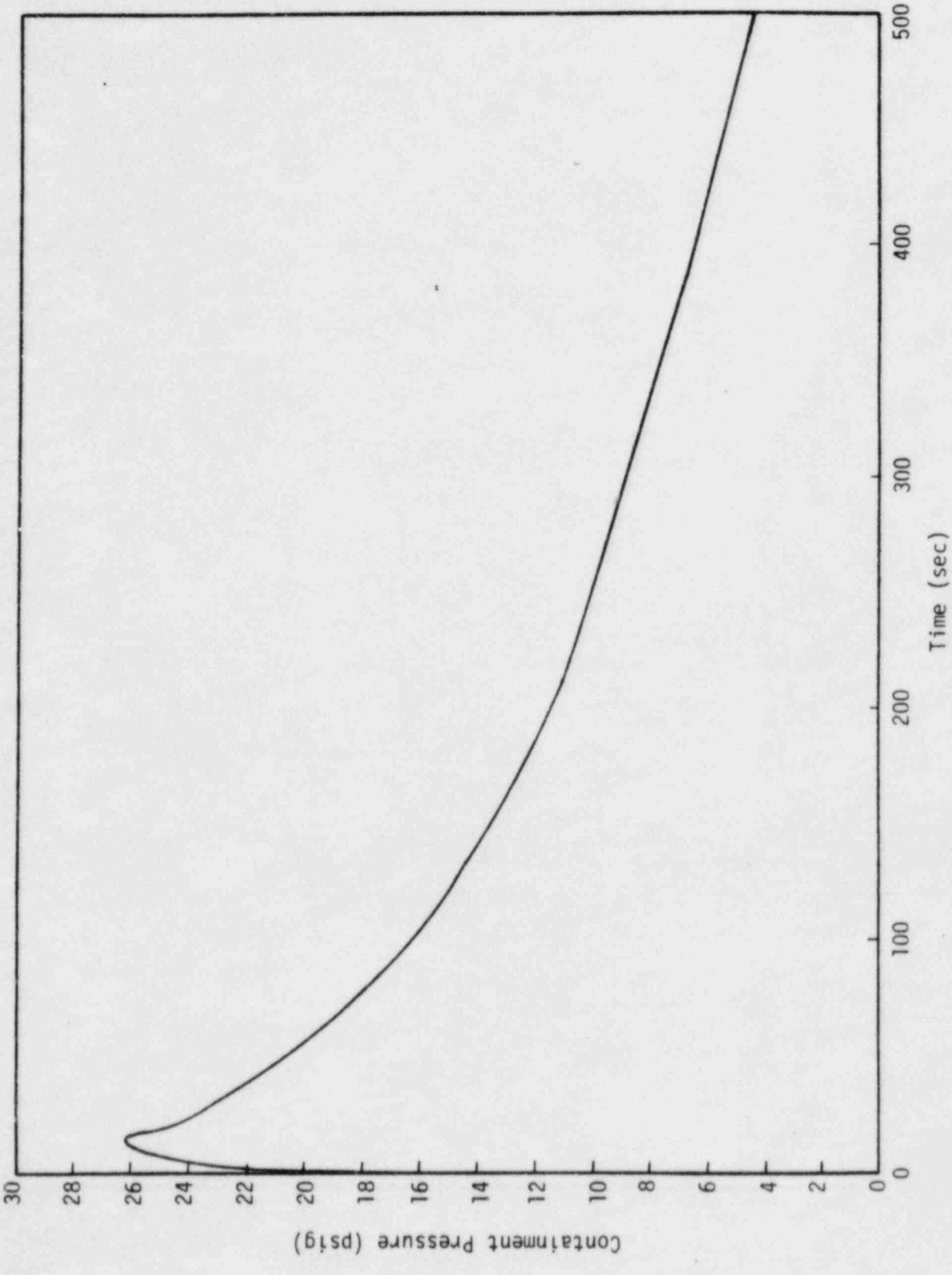
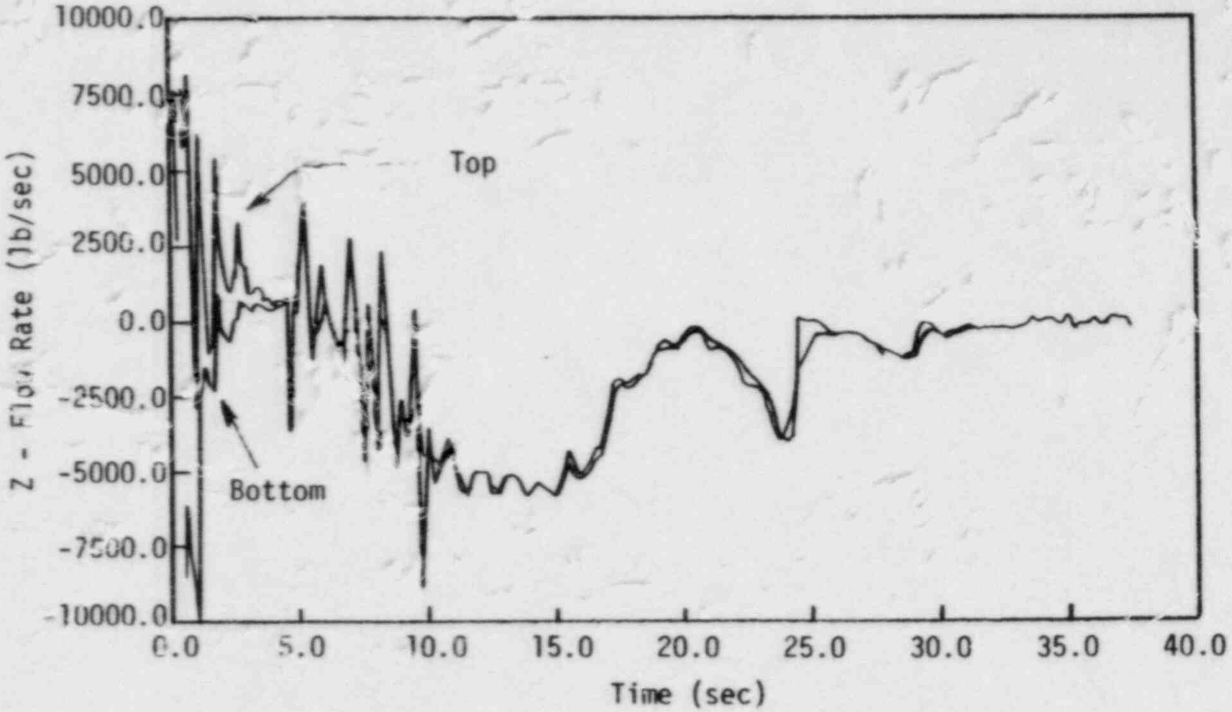


Figure 15.6.4-9 Containment Pressure
DECLG (C_D = 1.0)

Figure 15.6.4-10 Core Flow (Top and Bottom) DECLG (C_D = 1.0)



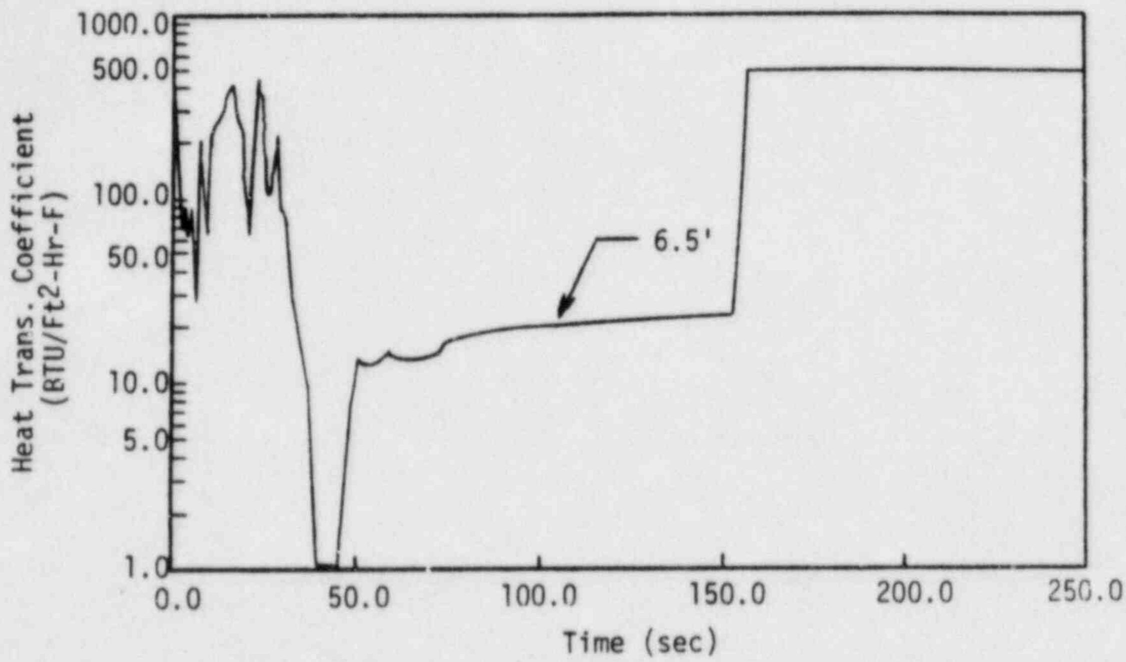


Figure 15.6.4-11 Heat Transfer Coefficient
Peak Clad Temperature Elevation
DECLG (C_D = 1.0)

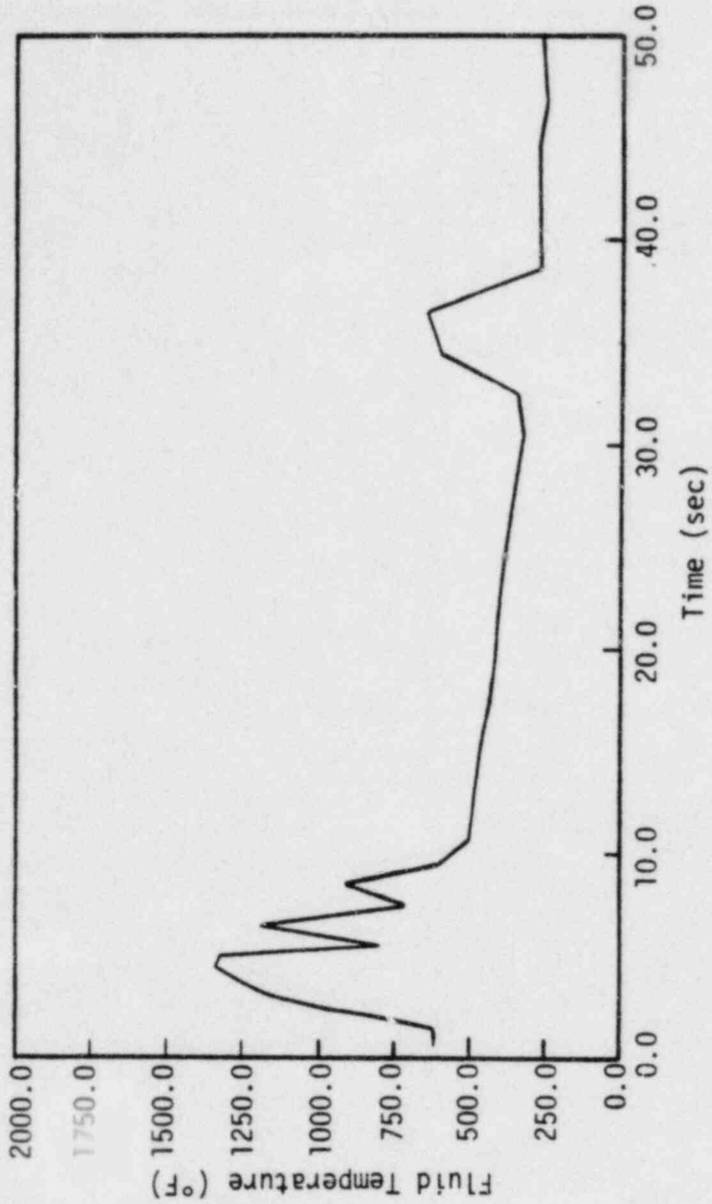


Figure 15.6.4-12 Fluid Temperature
Elevation 6.5 Ft
DECLG (C_D = 1.0)

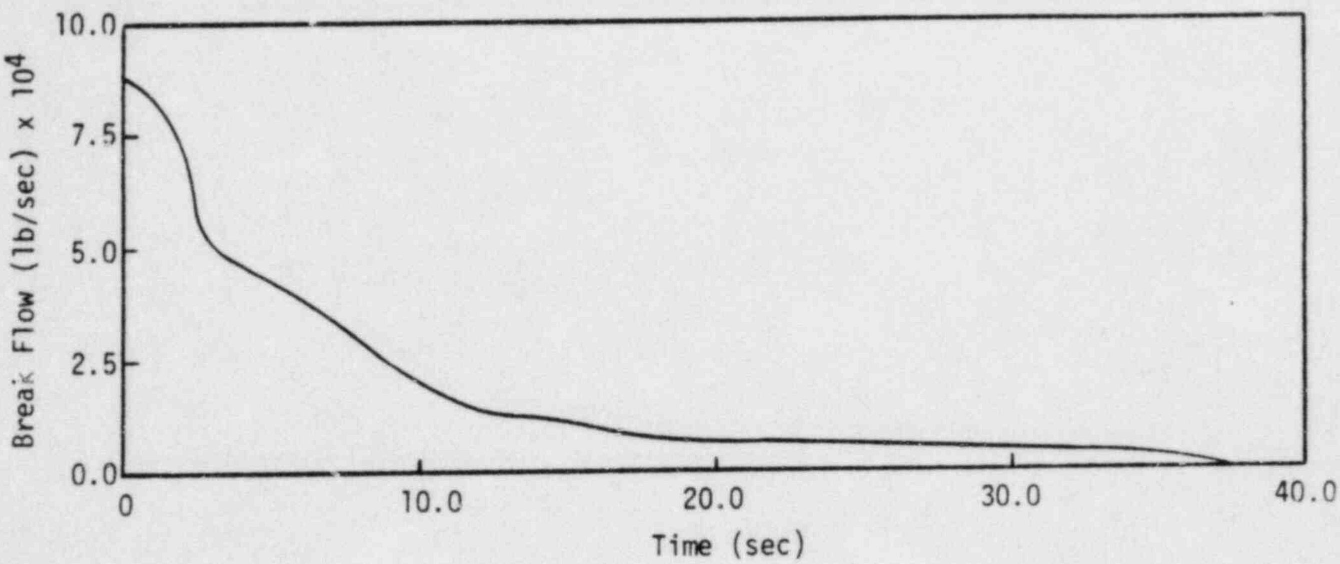


Figure 15.6.4-13 Break Flow Rate
DECLG (C_D = 1.0)

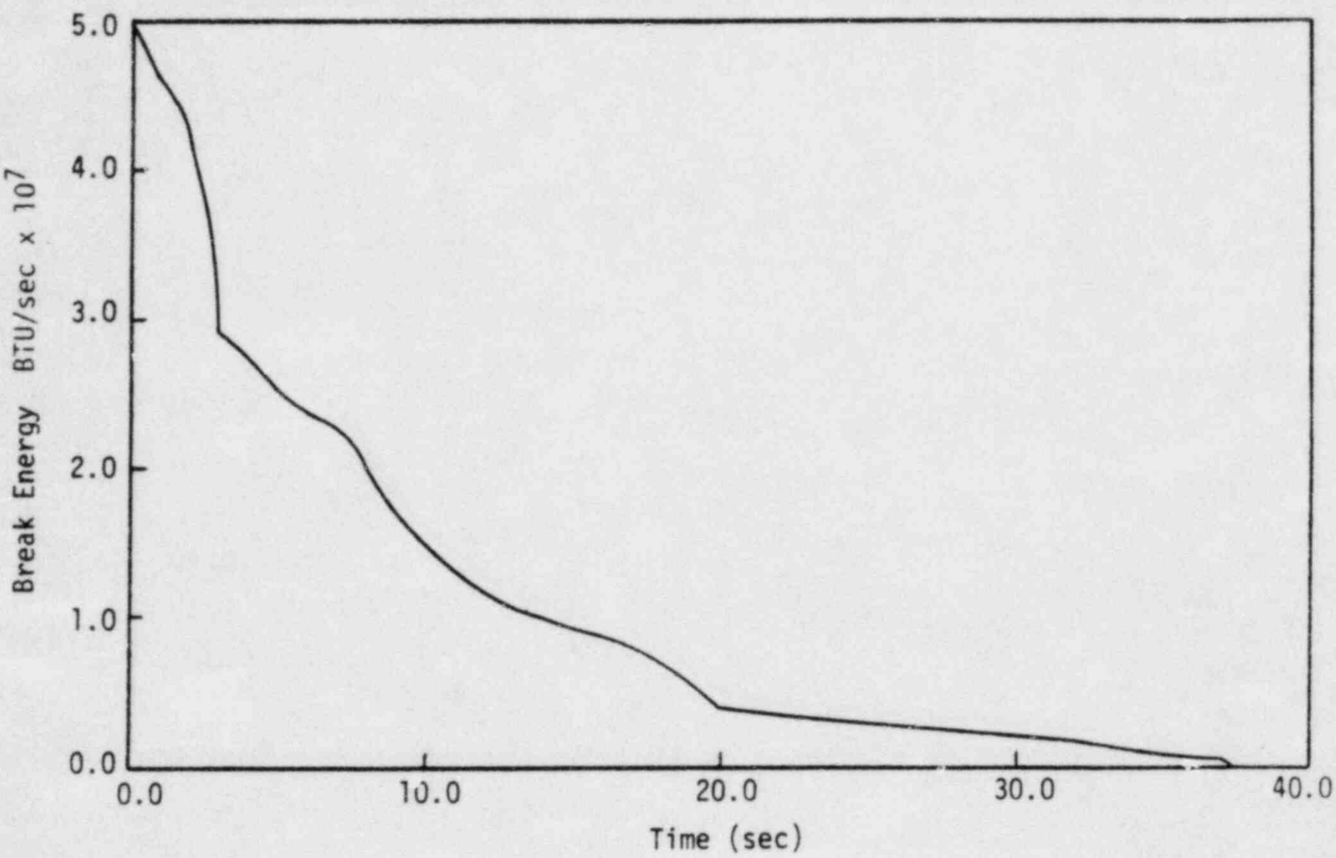


Figure 15.6.4-14 Break Energy Released to Containment DECLG (C_D = 1.0)

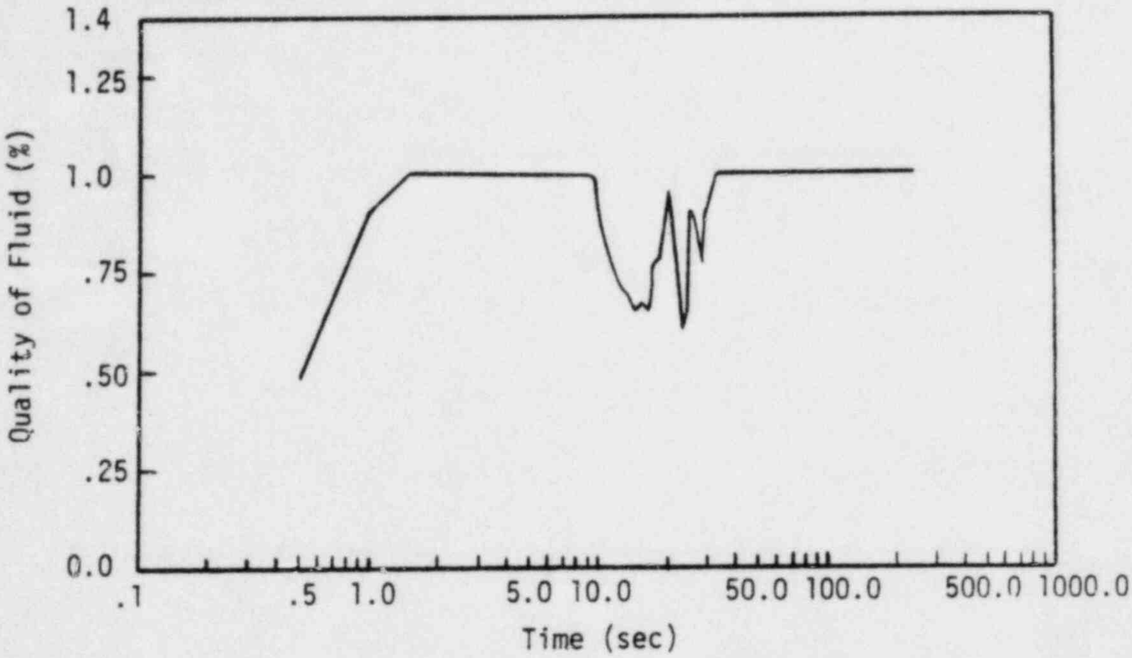


Figure 15.6.4-15 Fluid Quality Peak Clad Temperature Elevation, 6.5 Feet DECLG (C_D = 1.0)

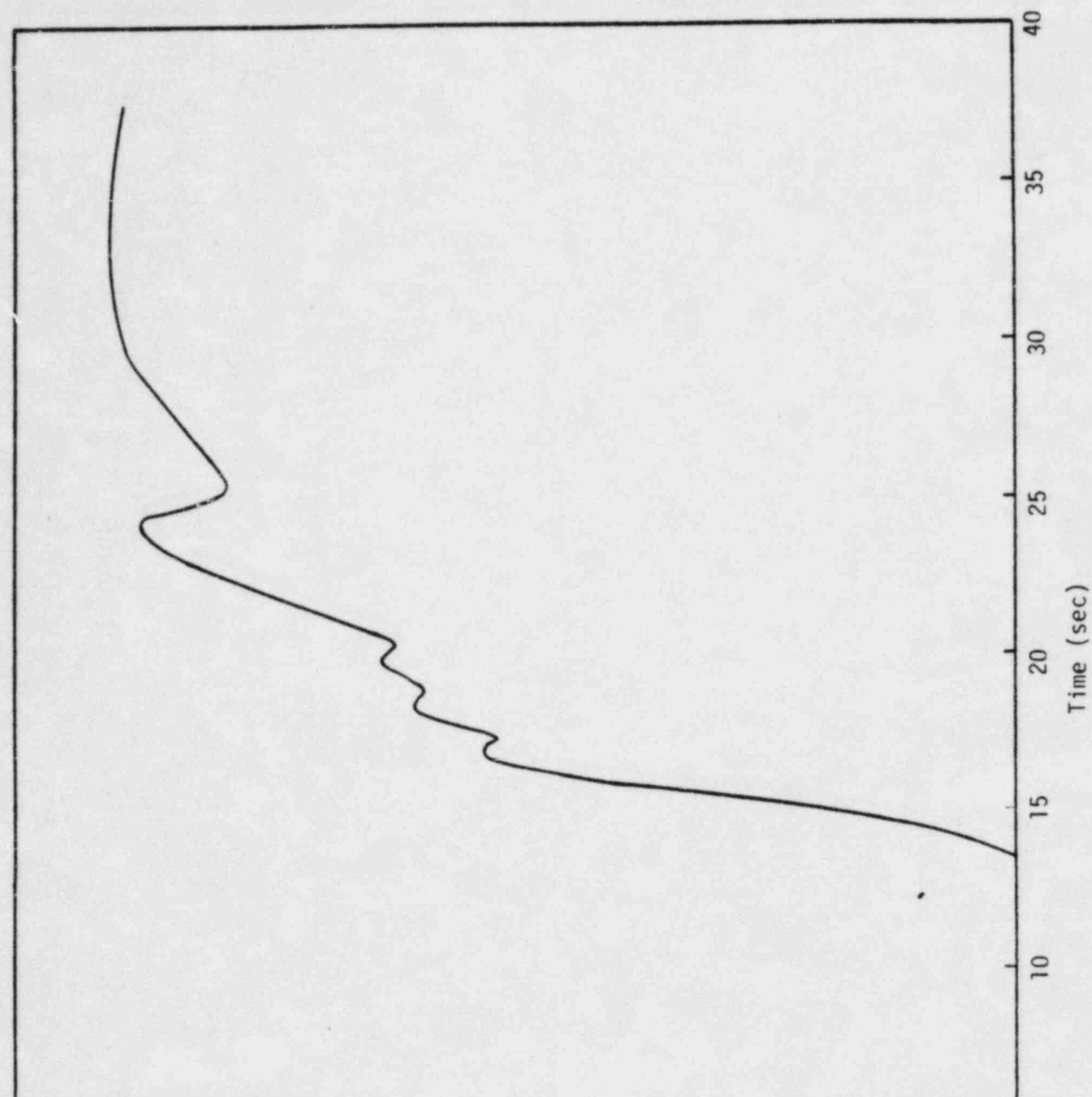


Figure 15.6.4-16 Accumulator Flow (Blowdown)
DECLG (CD = 1.0)

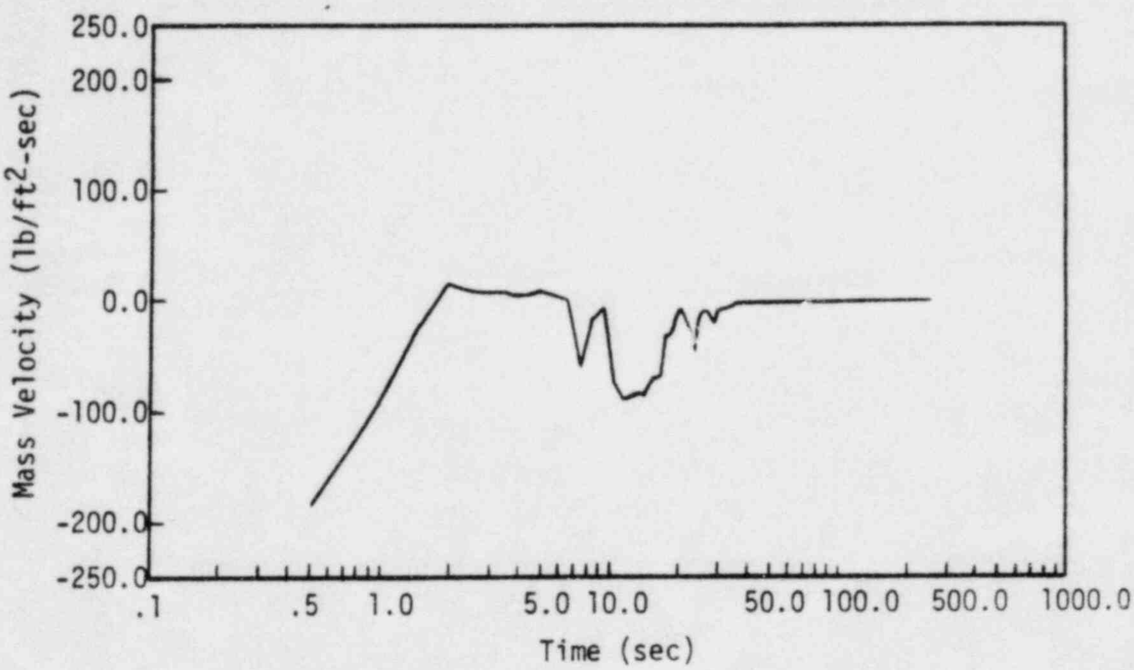
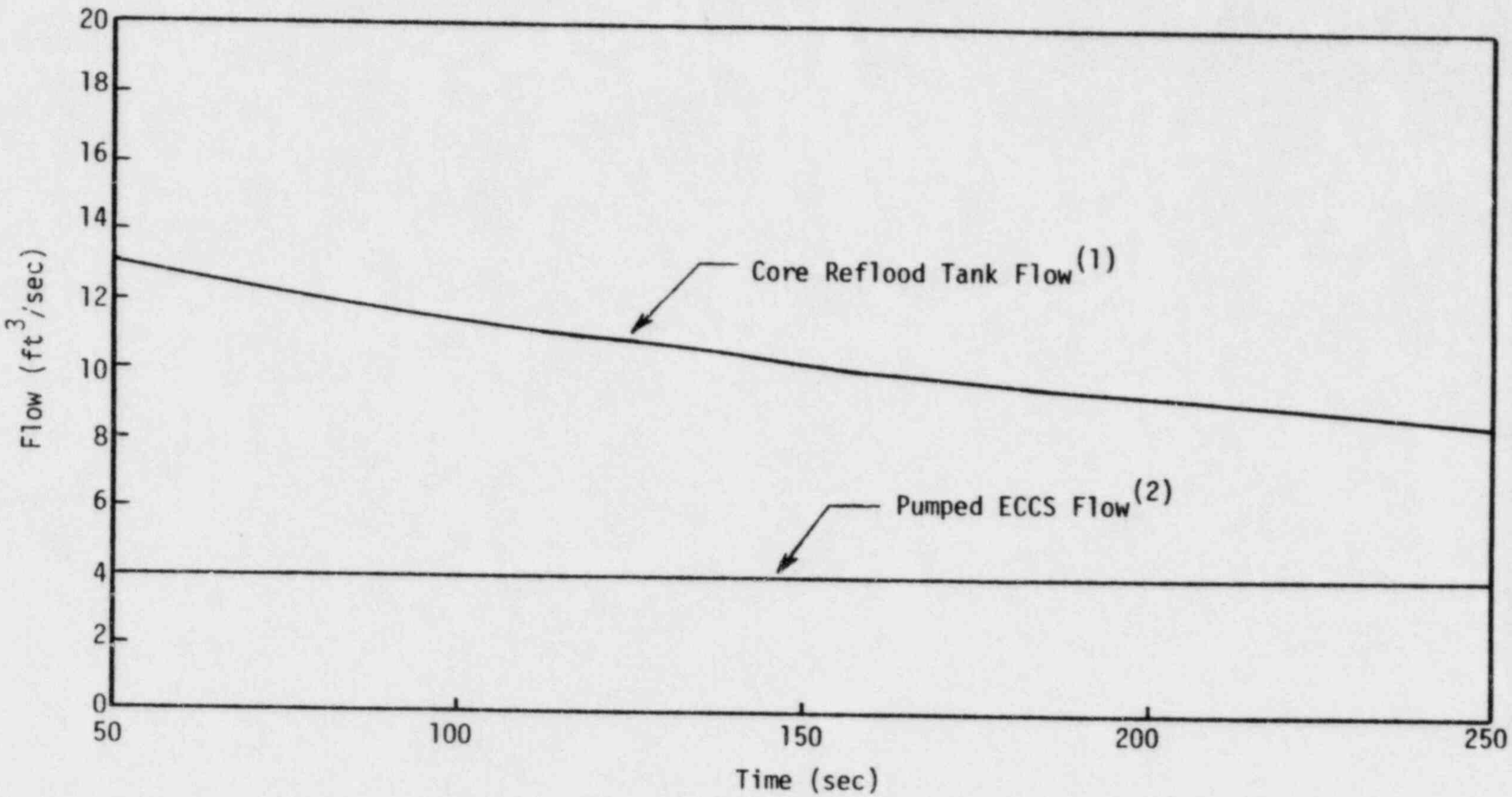


Figure 15.6.4-17 Mass Velocity, Elevation 6.5 Feet
DECLG (C_D = 1.0)

Figure 15.6.4-18 ECCS Flow During Reflood DECLG (C_D = 1.0)



(1) BASED ON 4 CRT'S DELIVERING
(2) BASED ON 2 OF 4 HIGH HEAD PUMPS DELIVERING

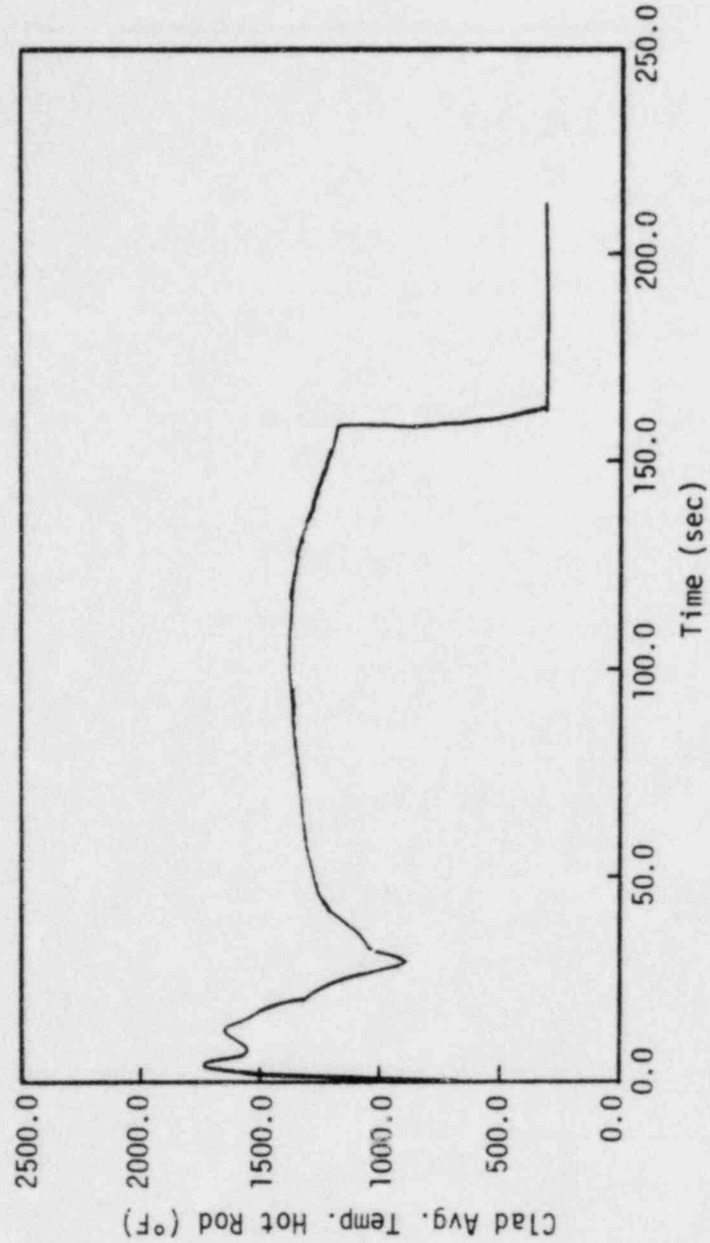


Figure 15.6.4-19 Peak Clad Temperature, Elevation 6.5 Feet DECLG (C_D = 0.6)

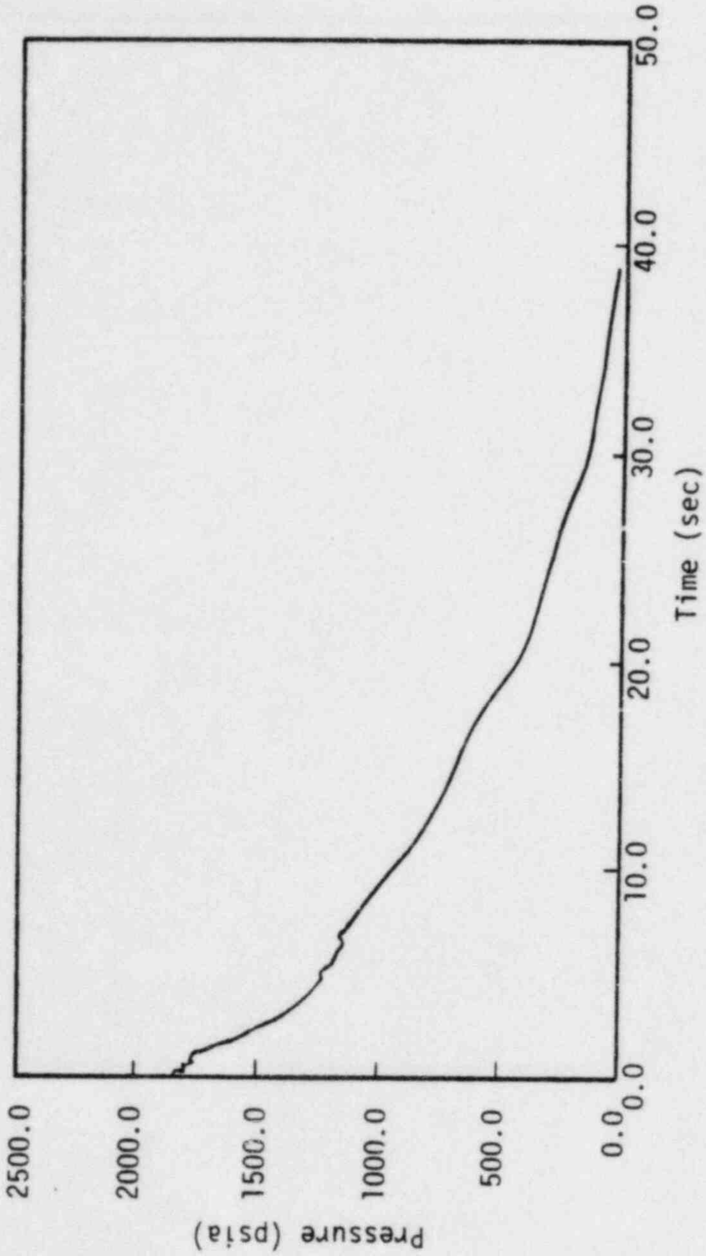


Figure 15.6.4-20 Core Pressure DECLG (C_D = 0.6)

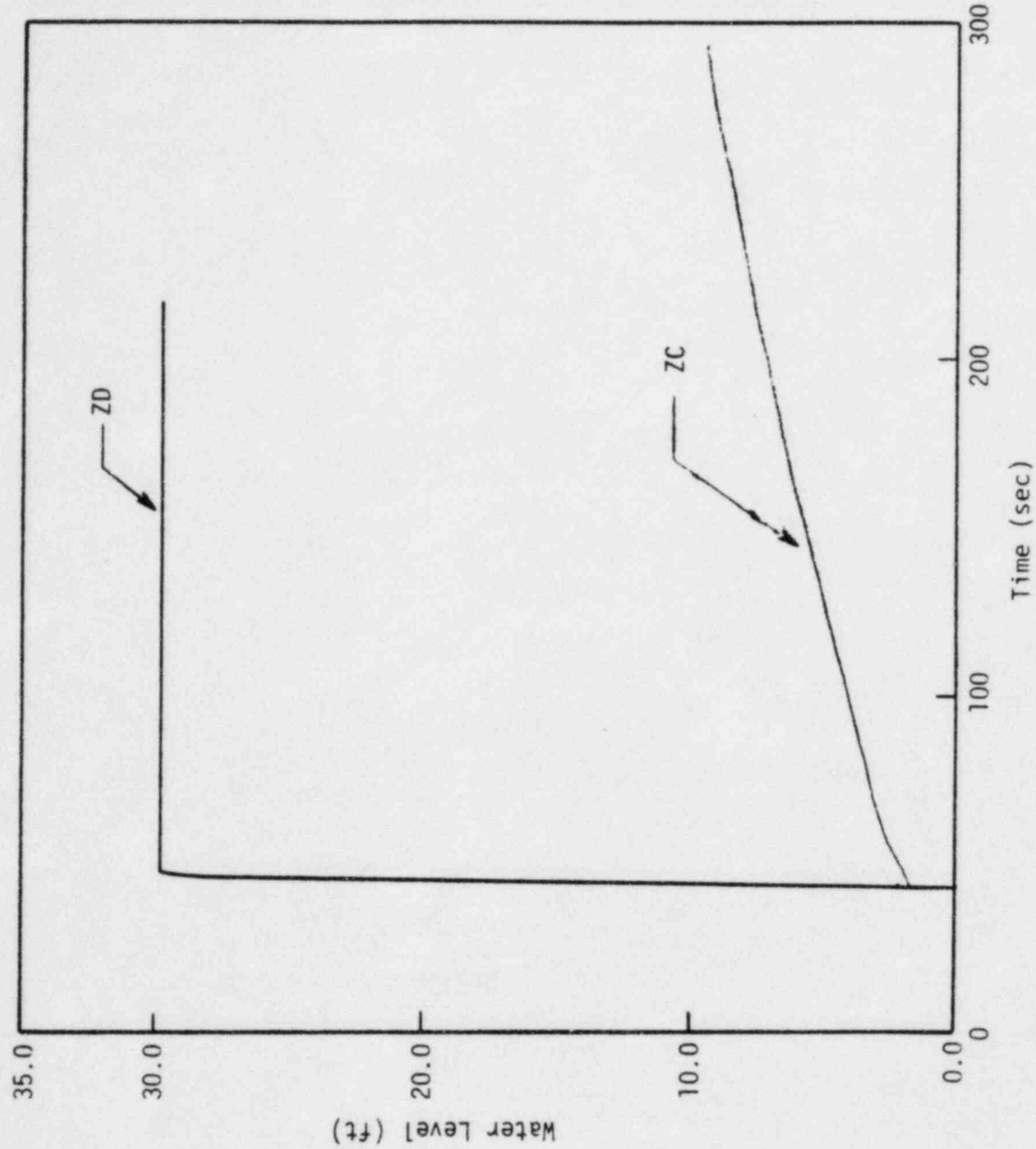


Figure 15.6.4-21 Reflood Transient - Core and Downcomer Water Levels DECLG (C_D = 0.6)

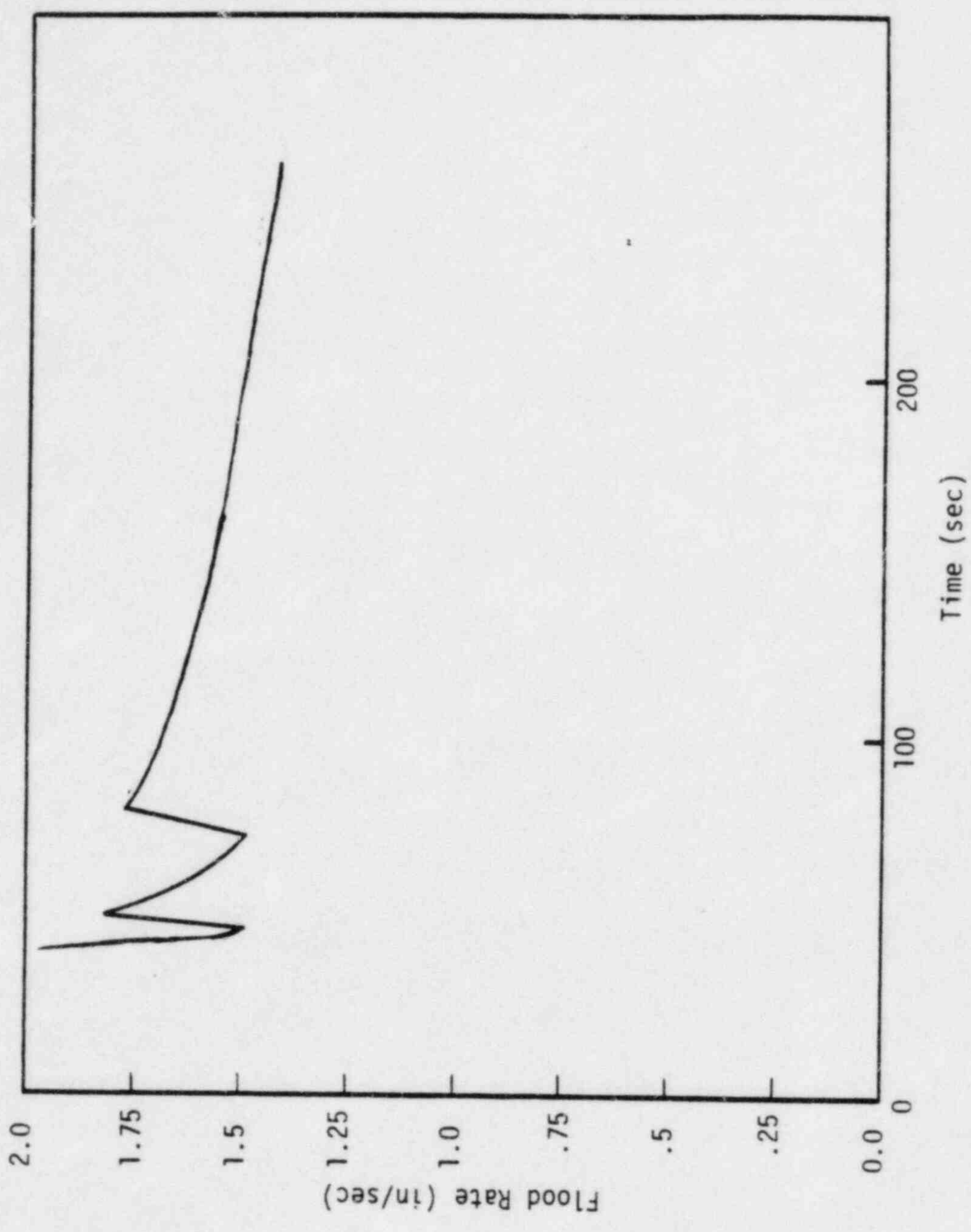


Figure 15.6.4-22 Reflood Transient Core Inlet Velocity
DECLG (C_D = 0.6)

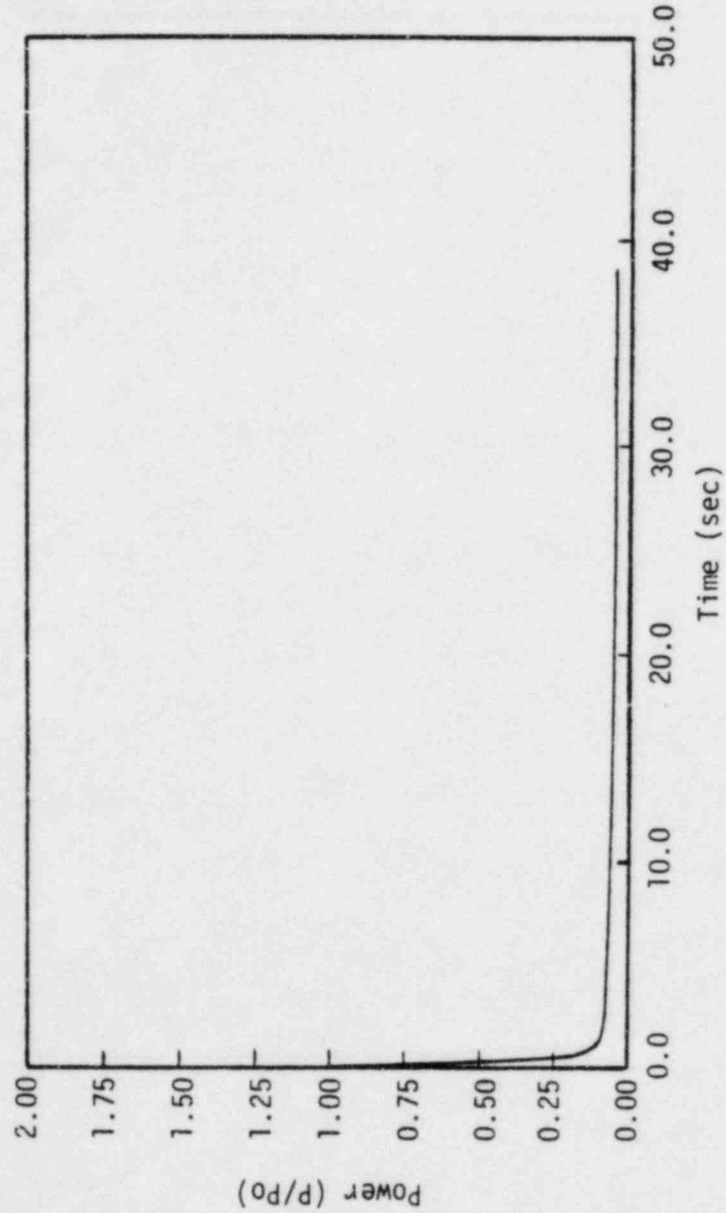


Figure 15.6.4-23 Core Power Transient DECLG (C_D = 0.6)

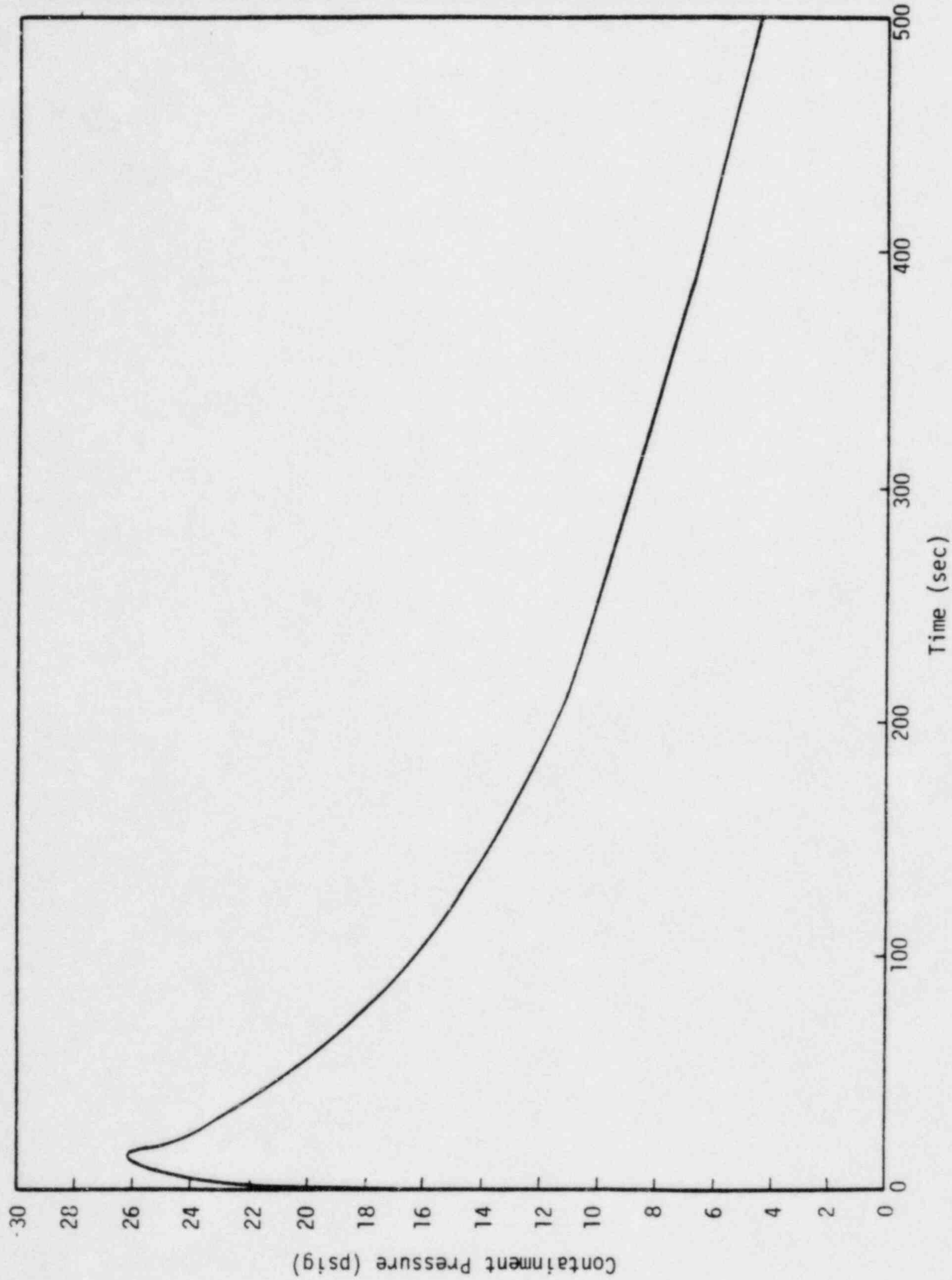


Figure 15.6.4-24 Containment Pressure DECLG (C_D = 0.6)

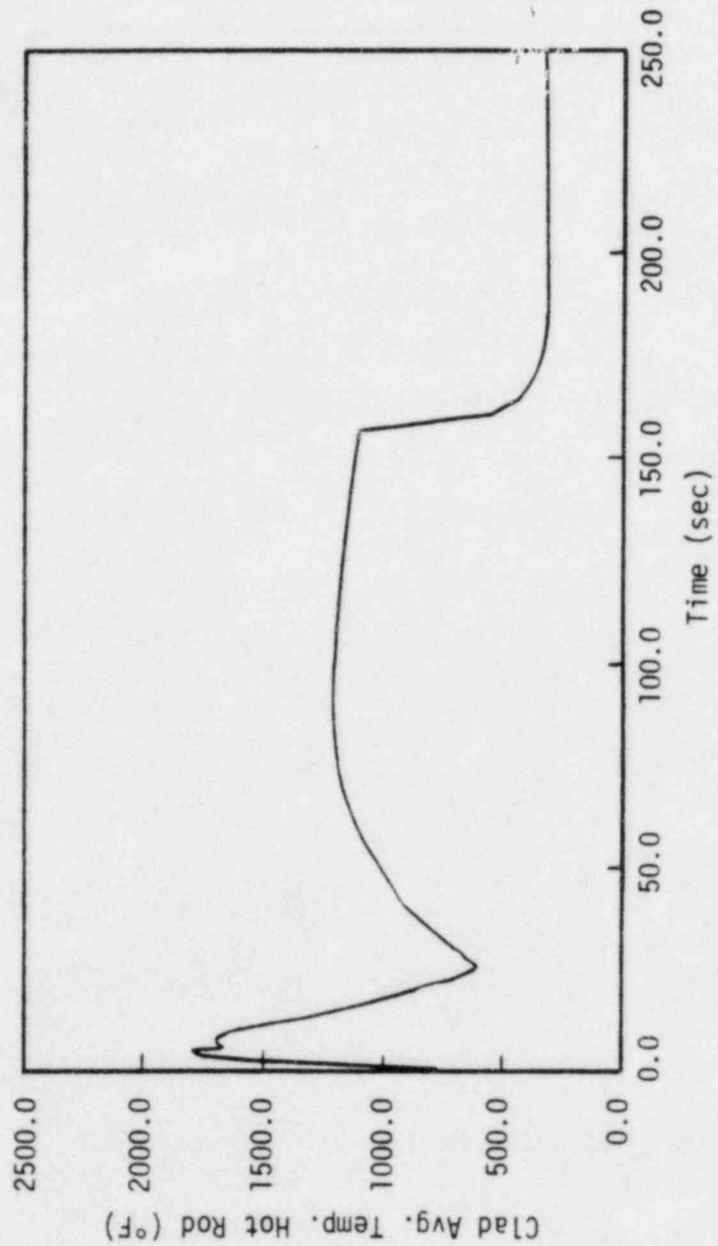


Figure 15.6.4-25 Peak Clad Temperature, Elevation 6.5 Feet
DECLG (C_D = 0.8)

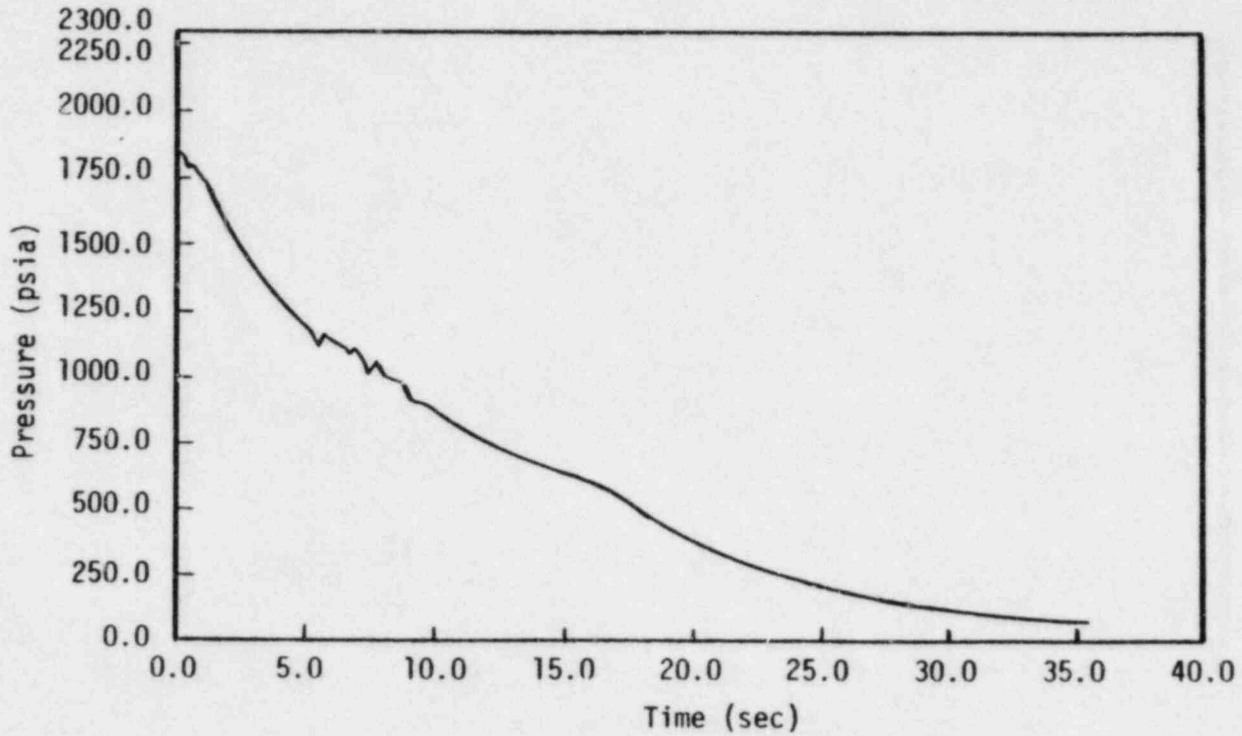


Figure 15.6.4-26 Core Pressure
DECLG (C_D = 0.8)

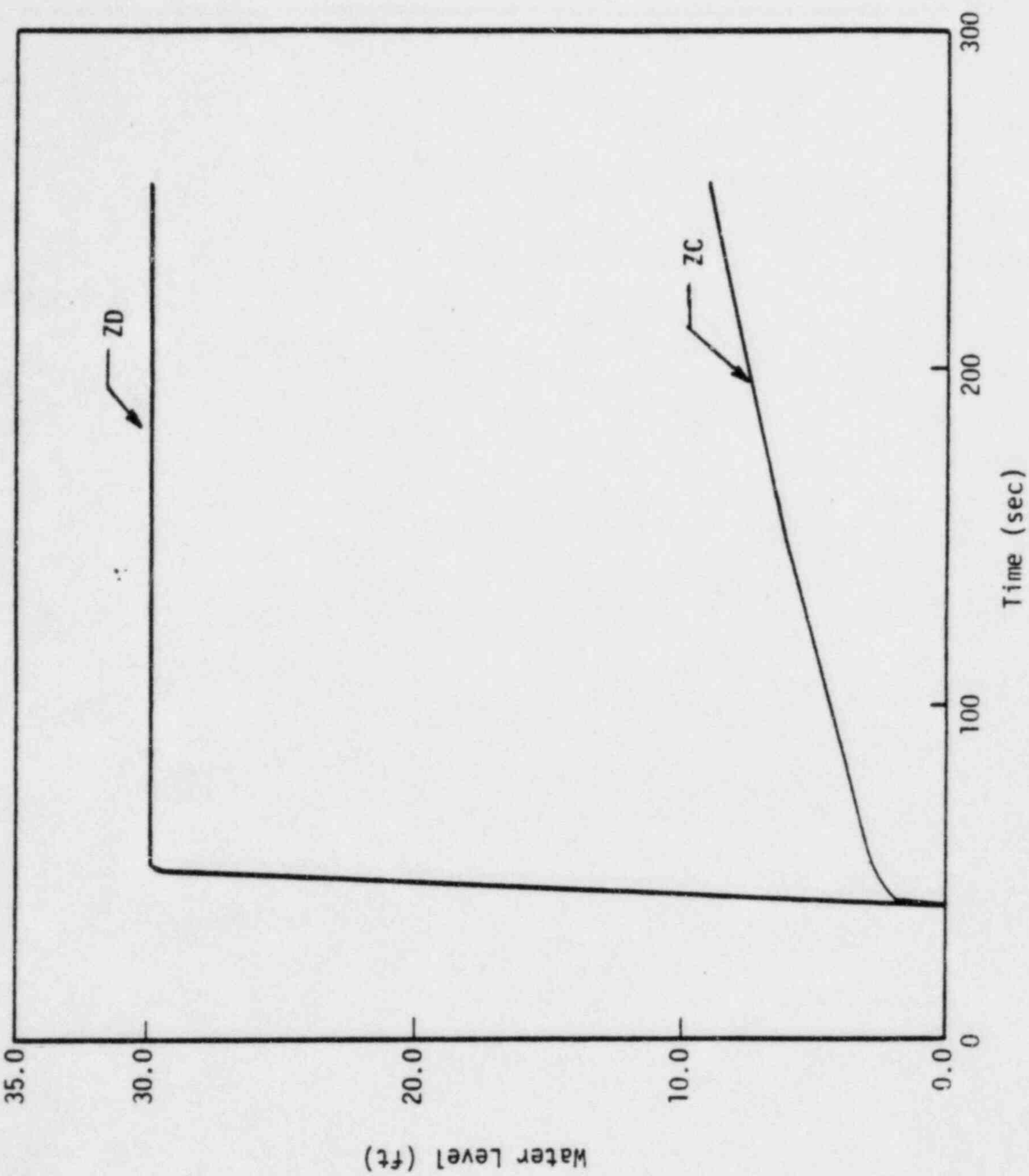


Figure 15.6.4-27 Reflood Transient Core and Downcomer Water Levels DECLG (C_D = 0.8)

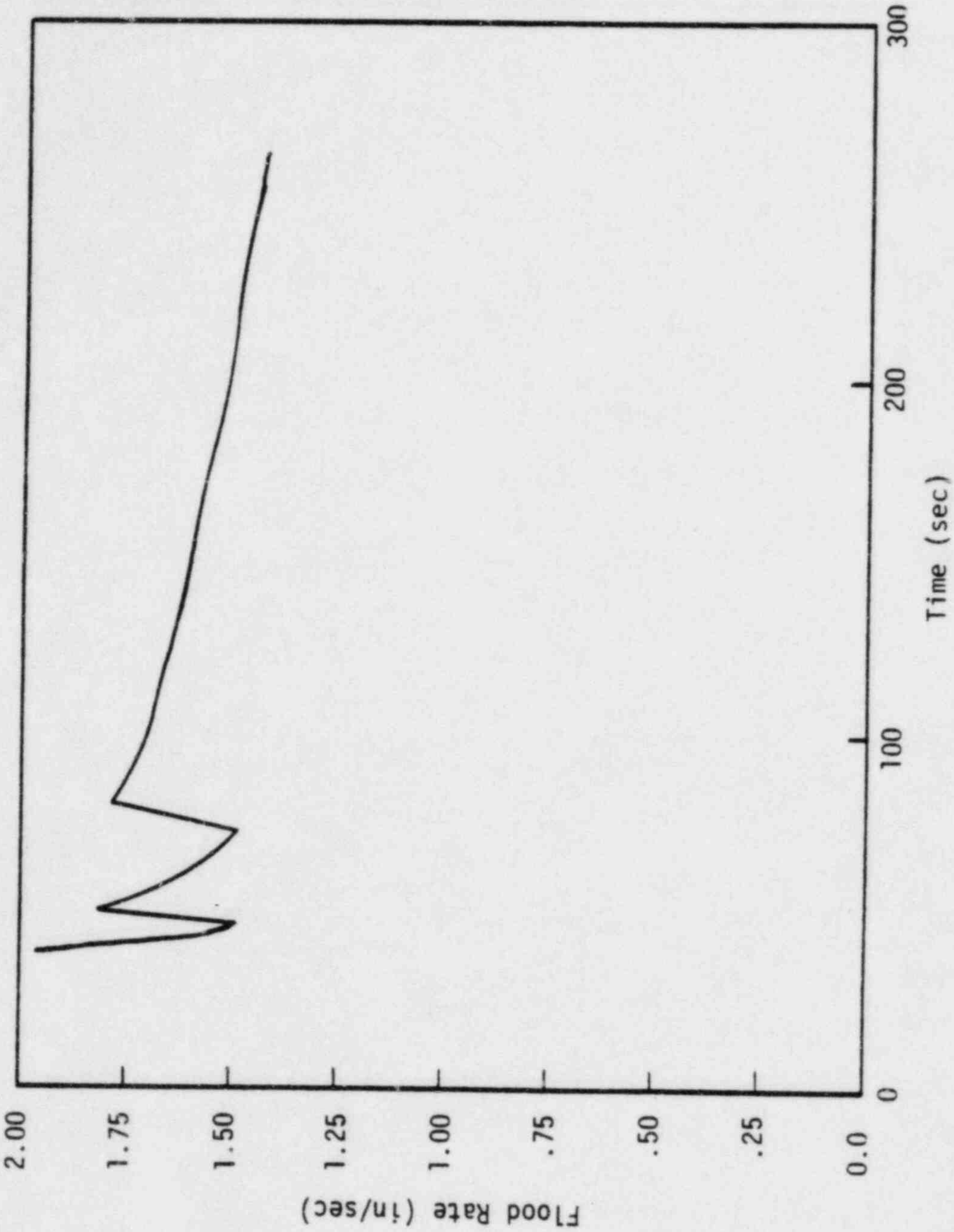


Figure 15.6.4-28 Reflood Transient Core Inlet Velocity DECLG (C_D = 0.8)

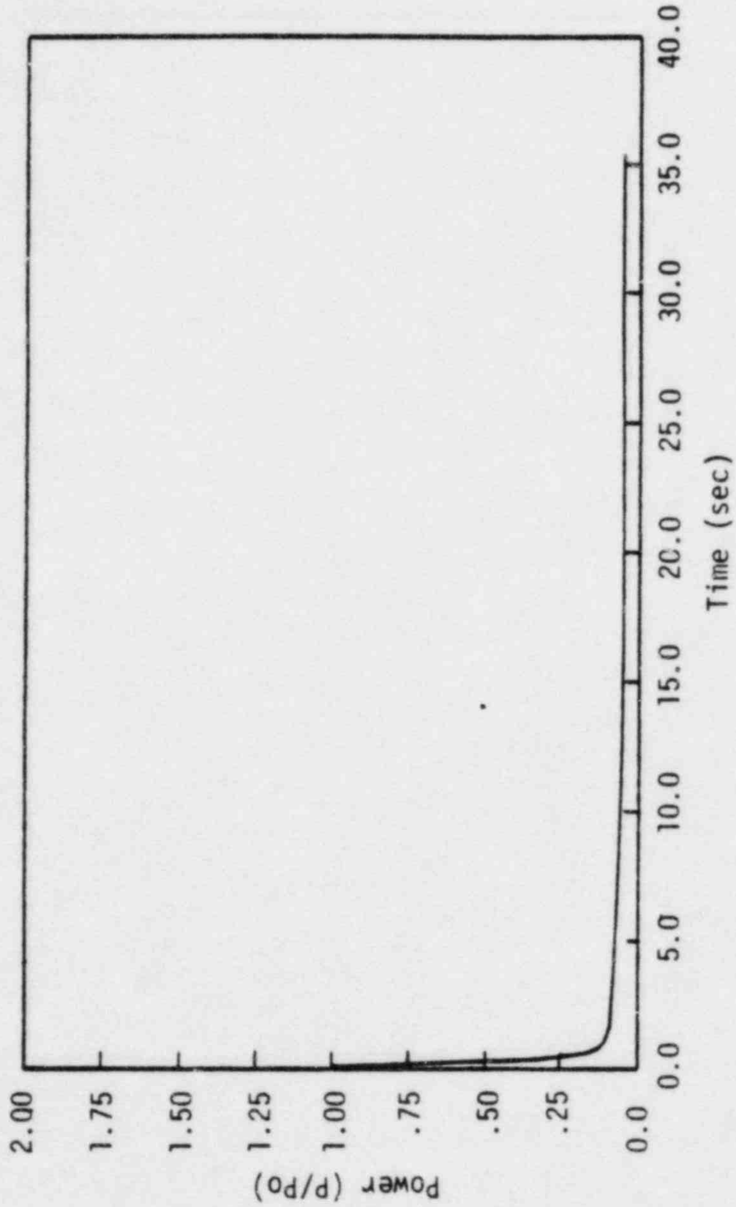


Figure 15.6.4-29 Core Power Transient
DECLG (C_D = 0.8)

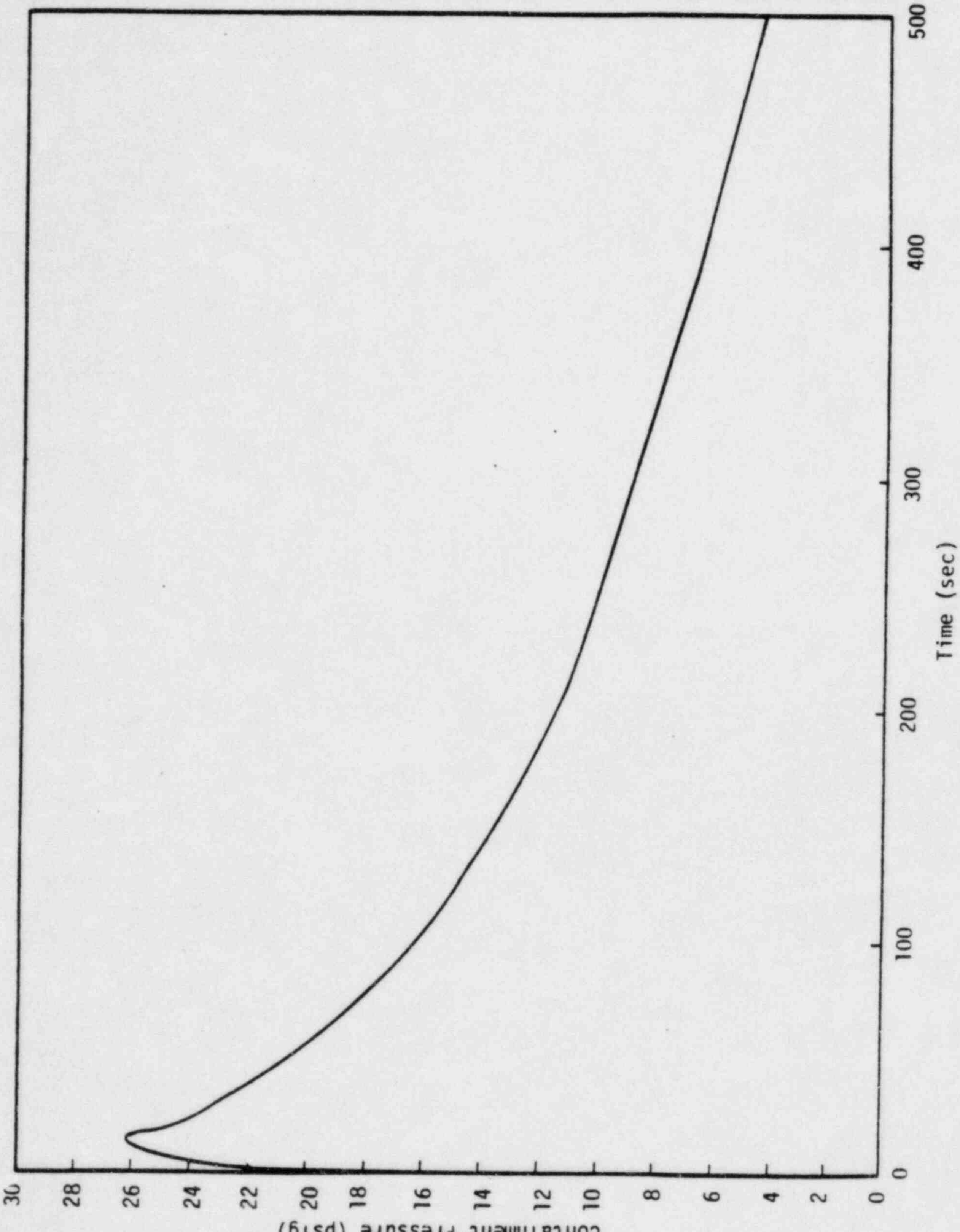


Figure 15.6.4-30 Containment Pressure DECLG (C_D = 0.8)

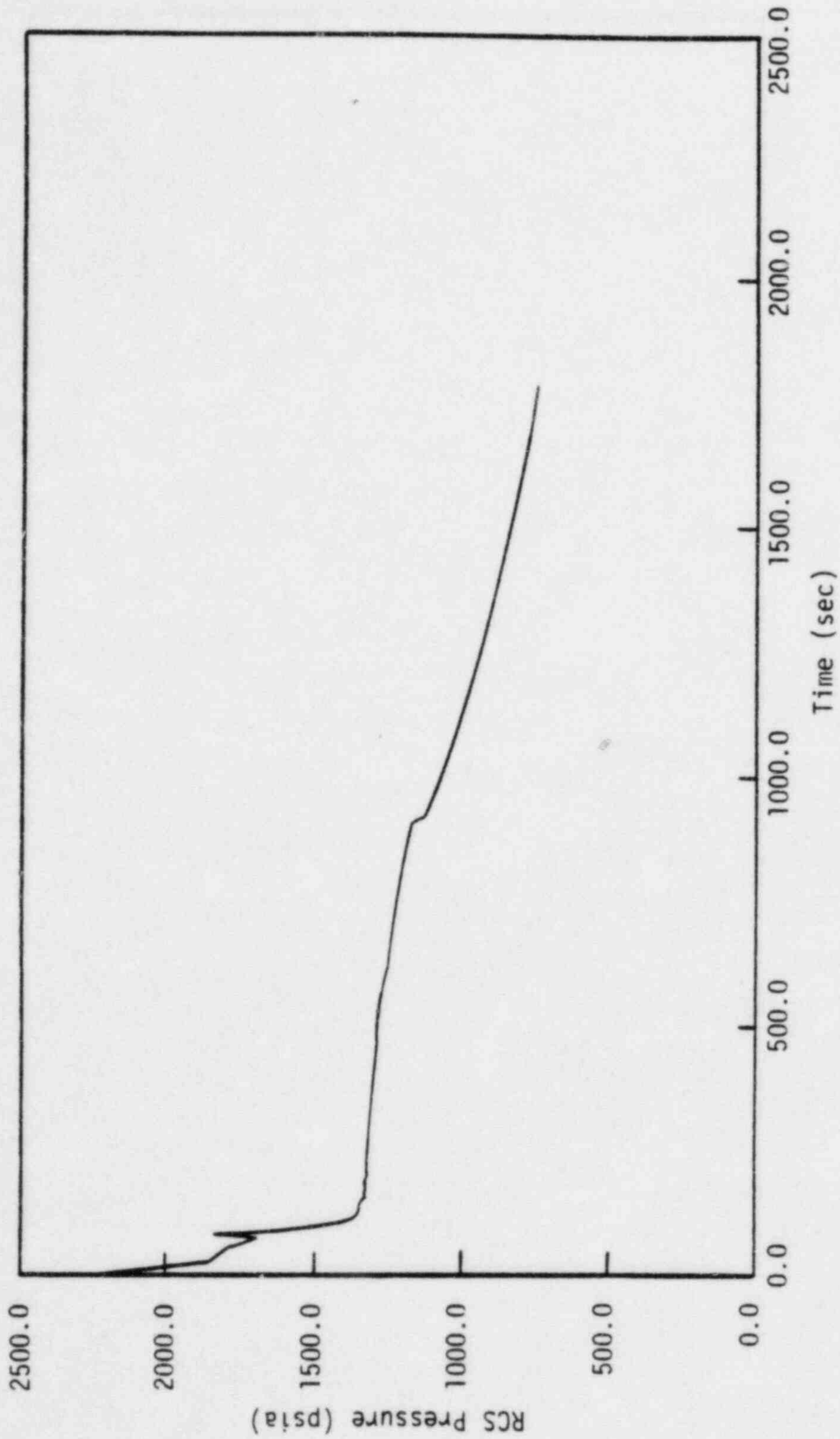


Figure 15.6.4-31 RCS Depressurization Transient (3 inch)

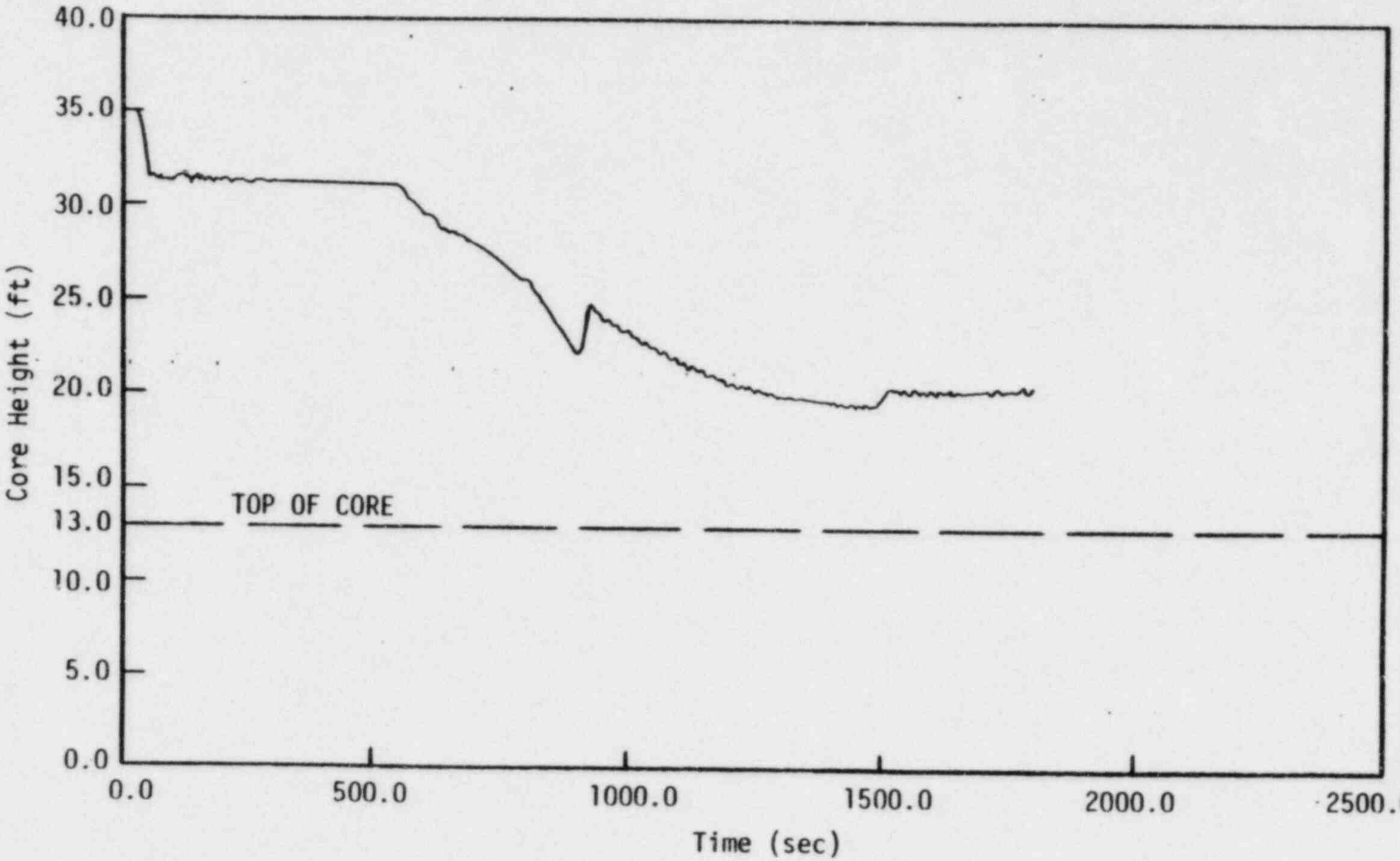


Figure 15.6.4-32 Core Mixture Height (3 inch)

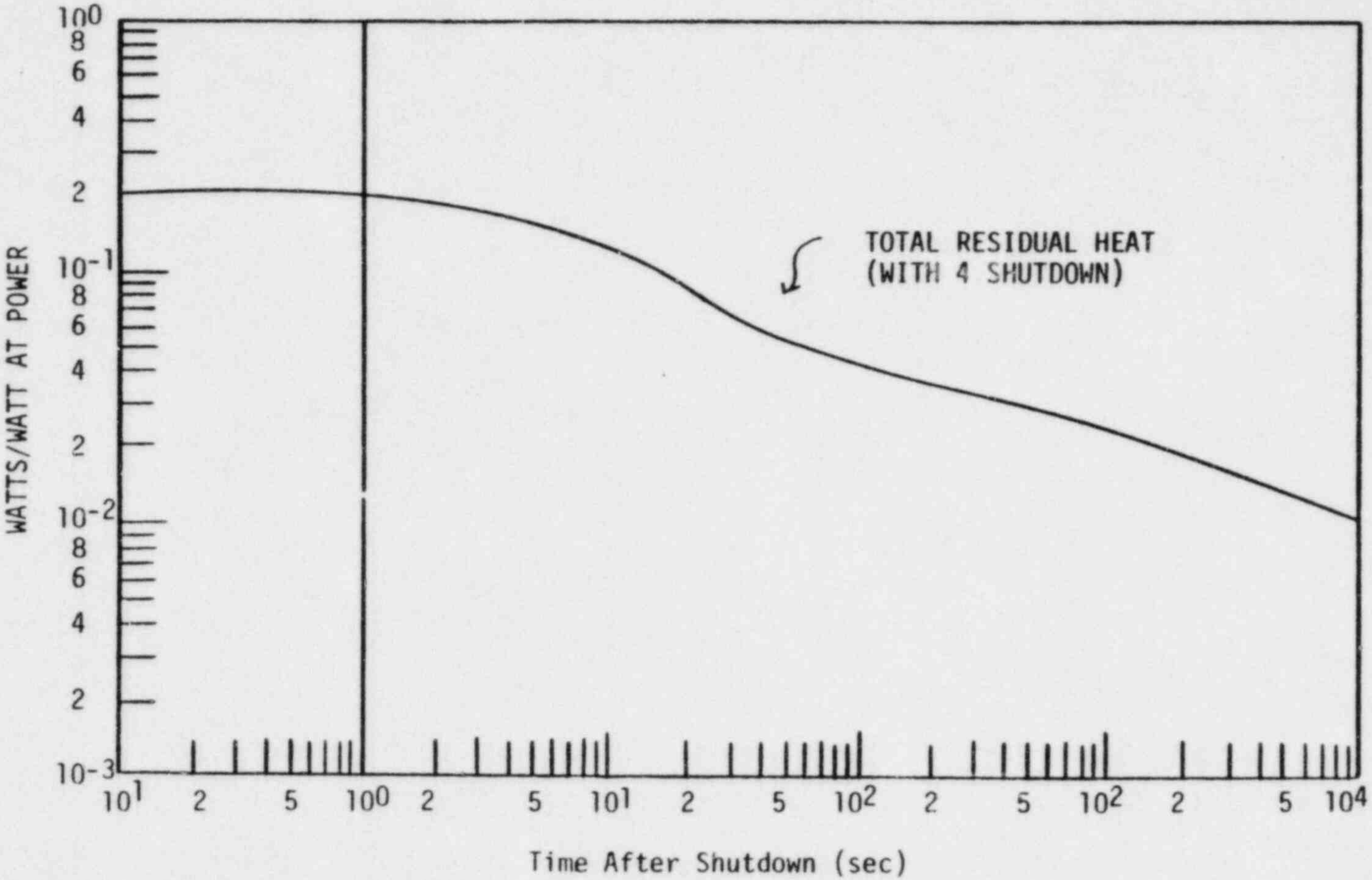


Figure 15.6.4-33 Core Power After Reactor Trip

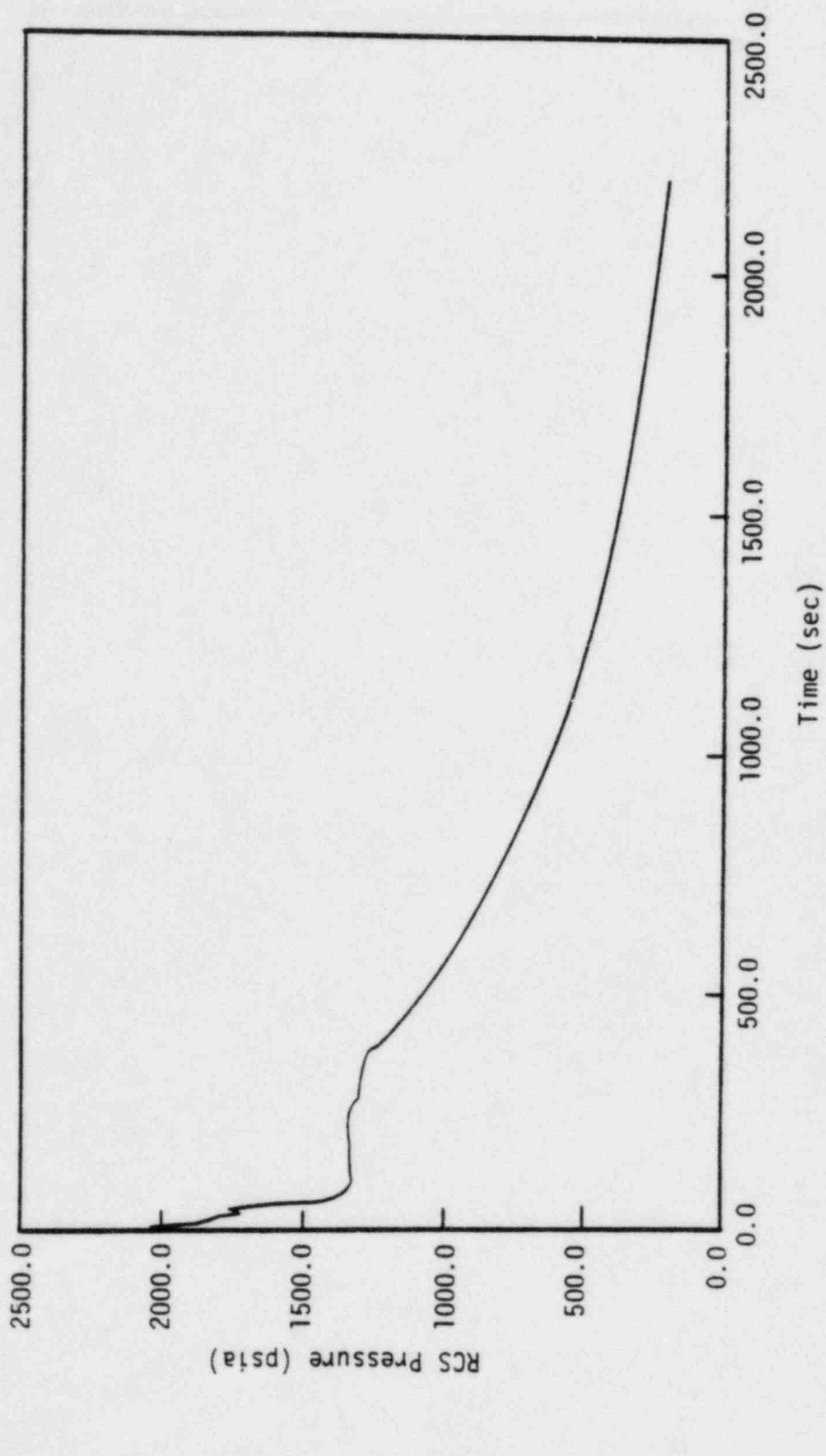


Figure 15.6.4-34 RCS Depressurization Transient (4.313 inch)

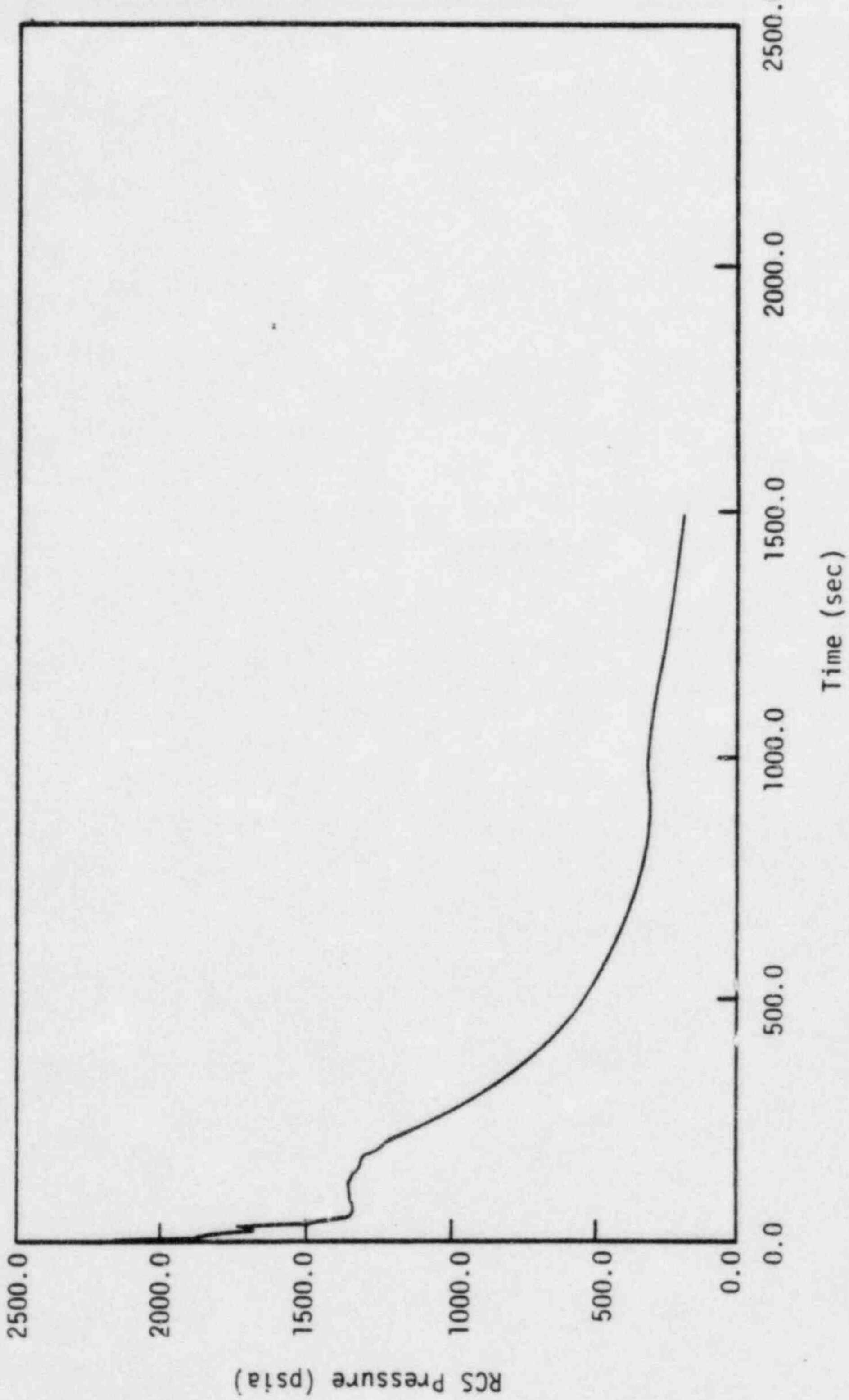


Figure 15.6.4-35 RCS Depressurization Transient (6 inch)

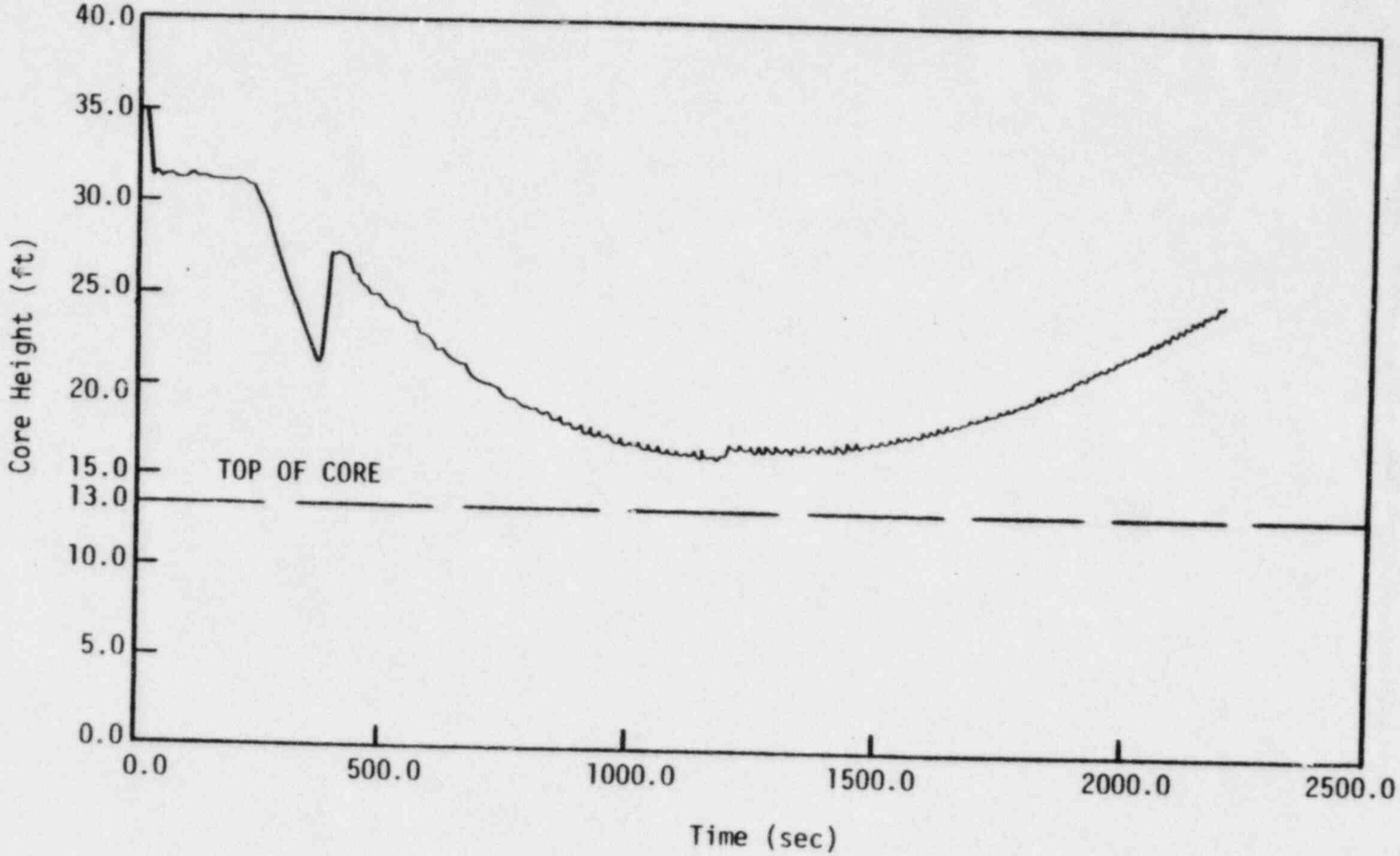


Figure 15.6.4-36 Core Mixture Height (4.313 inch)

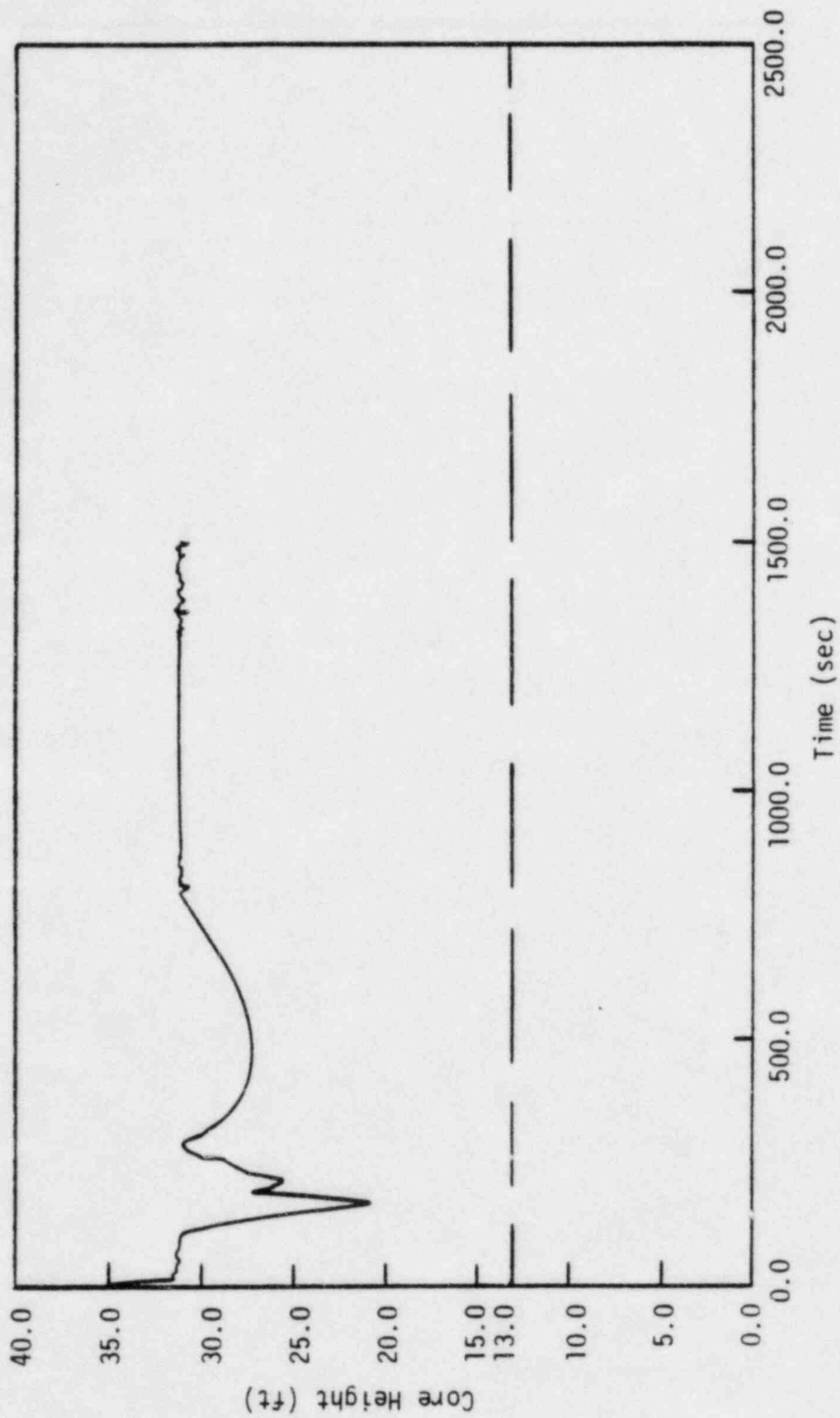


Figure 15.6.4-37 Core Mixture Height (6 inch)

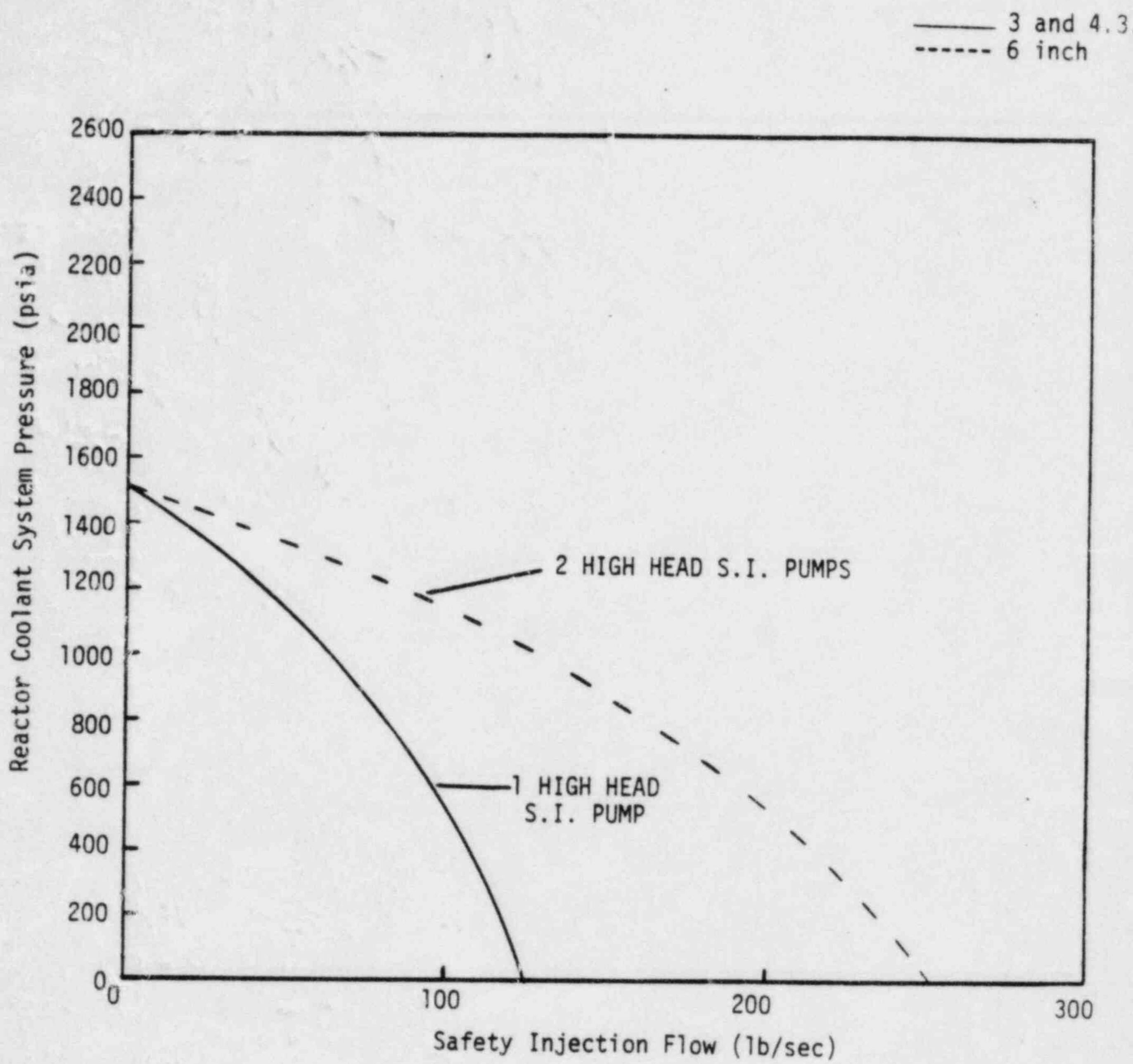


Figure 15.6.4-38 Small Break Safety Injection Flow Rates

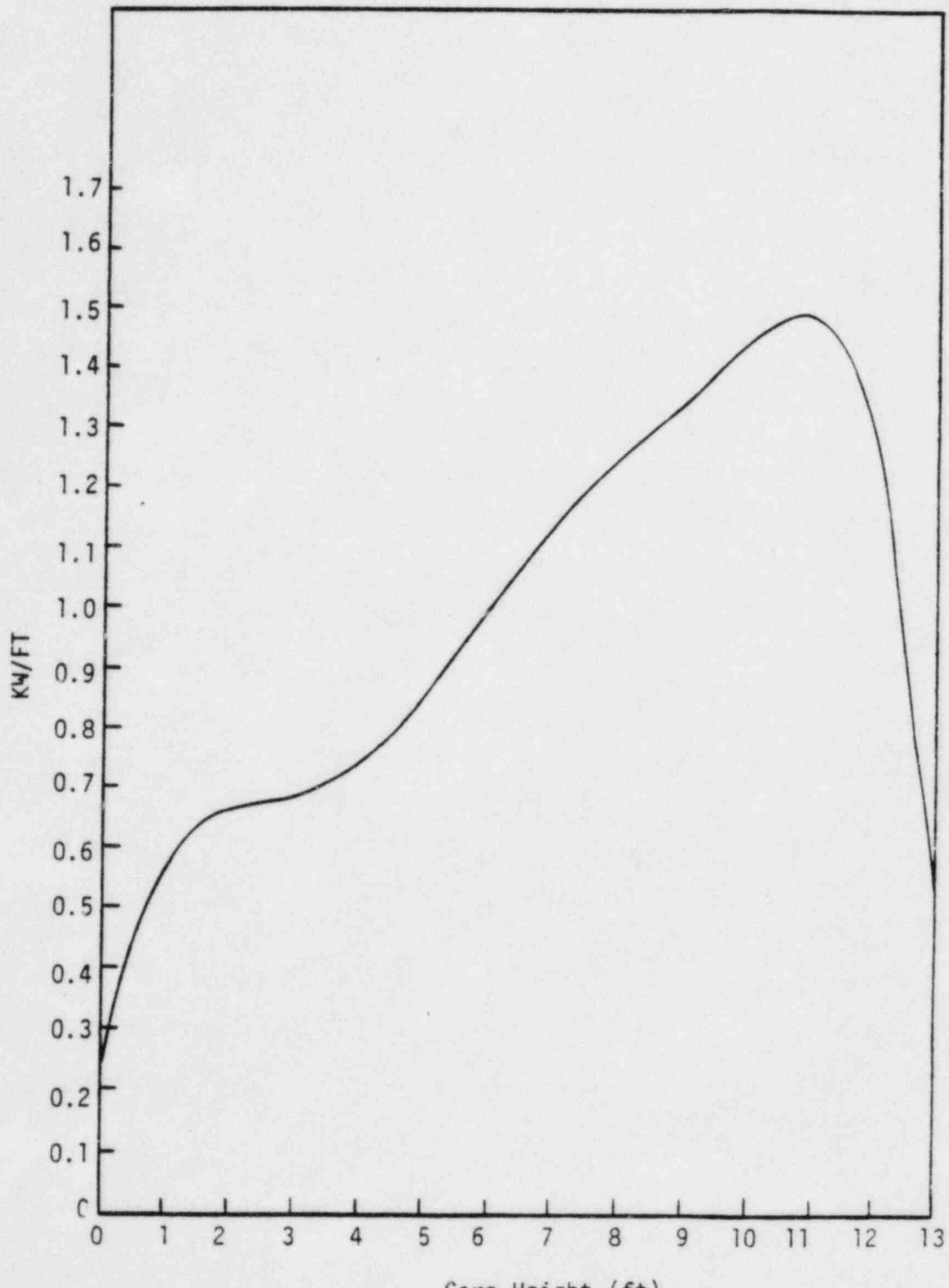


Figure 15.6.4-39 Small Break Power Distribution