PROPOSED TECHNICAL SPECIFICATION CHANGES

RCS FLOW RATE

LIMITING CONDITION FOR OPERATION

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to 108.4×10^6 lbm/hr (Note 1).

APPLICABILITY: MODE 1.

ACTION:

With the actual Reactor Coolant System total flow rate determined to be less than the above limit, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 The actual Reactor Coolant System total flow rate shall be determined to be within its limit at least once per 12 hours.

Note 1: The value of 120.4×10^6 lbm/hr has been reduced to 108.4×10^6 lbm/hr until the steam generators are replaced. After the steam generators are replaced, this value returns to 120.4×10^6 lbm/hr.

MARKUP OF CURRENT ANO-2 TECHNICAL SPECIFICATIONS

(FOR INFO ONLY)

POWER DISTRIBUTION LIMITS

RCS FLOW RATE

LIMITING CONDITION FOR OPERATION

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to ~~120~~108. $\times 10^6$ lbm/hr (Note 1).

APPLICABILITY: MODE 1.

ACTION:

With the actual Reactor Coolant System total flow rate determined to be less than the above limit, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 The actual Reactor Coolant System total flow rate shall be determined to be within its limit at least once per 12 hours.

Note 1: The value of 120.4 $\times 10^6$ lbm/hr has been reduced to 108. $\times 10^6$ lbm/hr until the steam generators are replaced. After the steam generators are replaced, this value returns to 120.4 $\times 10^6$ lbm/hr.

ATTACHMENT 1

DESCRIPTION AND EVALUATION OF THE ANALYSES IN SUPPORT OF REDUCING THE RCS FLOW REQUIREMENTS BY 10% AND INCREASING THE STEAM GENERATOR TUBE PLUGGING LIMIT TO 30%

ATTACHMENT 1

DESCRIPTION AND EVALUATION OF THE ANALYSES IN SUPPORT OF REDUCING THE RCS FLOW REQUIREMENTS BY 10% AND INCREASING THE STEAM GENERATOR TUBE PLUGGING LIMIT TO 30%

The following is a listing of the events presented in the ANO-2 Safety Analysis Report (SAR). In the right hand column a note has been placed delineating the level of effort in addressing a reduction in Reactor Coolant System (RCS) flow. A summary of the effort for each event which is affected by a lower RCS flow is provided in this attachment. The reduction in RCS flow is attributed to steam generator tube plugging. Consideration for up to 30% steam generator tube plugging has been accounted for in the analyses described below. Occasionally, the analyses presented are unaffected by RCS flow reduction, but, are affected by tube plugging. These events are provided for information and are noted in the text for the respective event. Notes in the right hand column indicate: Evaluated, Reanalyzed, Not Reanalyzed, and Not Applicable. An event which is impacted by a reduction in RCS flow or tube plugging yet the impact can be addressed qualitatively are indicated as Evaluated. These events are addressed by qualitative arguments and some simple quantitative efforts. Events in which the effects of reducing RCS flow and plugging steam generator tubes are more involved and necessitated a new analysis have been indicated as Reanalyzed. Not Reanalyzed has been used to note events which are not impacted by a reduction in RCS flow or steam generator tube plugging. Finally, all of the events presented in Chapter 15 of the ANO-2 SAR are listed in the following table, some of which are not applicable to ANO-2 as indicated in the SAR. These events are denoted with a Not Applicable note.

SAR Section	**Section Title**	**Analysis Effort**
6.2.1	Containment Functional Design	Evaluated
6.3.3.2.2	Large Break Analysis	Reanalyzed
6.3.3.2.3	Small Break Analysis	Reanalyzed
15.1.1	Uncontrolled CEA Withdrawal from a Subcritical Condition	Reanalyzed
15.1.2	Uncontrolled CEA Withdrawal from Critical Conditions	
	Hot Zero Power (HZP)	Reanalyzed
	Hot Full Power (HFP)	Evaluated
15.1.3	CEA Misoperation	Not Reanalyzed
15.1.4	Uncontrolled Boron Dilution Incident	
	Modes 1 and 2	Reanalyzed
	Modes 3, 4, 5, and 6	Not Reanalyzed
15.1.5	Total and Partial Loss of RCS Forced Flow	
	Four Pump Loss of Flow	Reanalyzed
	Seized Rotor	Evaluated
15.1.6	Idle Loop Startup	Not Reanalyzed
15.1.7	Loss of External Load and/or Turbine Trip	Reanalyzed
15.1.8	Loss of Normal Feedwater Flow	Reanalyzed

15.1.9	Loss of All Normal and Preferred AC Power to the Station Auxiliaries	Not Reanalyzed
15.1.10	Excess Heat Removal Due to Secondary System Malfunction	Evaluated
15.1.11	Failure of the Regulating Instrumentation	Not Applicable
15.1.12	Internal and External Events Including Major and Minor Fires, Floods, Storms, and Earthquakes	Not Reanalyzed
15.1.13	Major Rupture of Pipes Containing Reactor Coolant up to and Including Double-Ended Rupture of Largest Pipe in the Reactor Coolant System (LOCA)	Not Reanalyzed
15.1.14	Major Secondary System Pipe Breaks with or without a Concurrent Loss of AC Power	
	Main Steam Line Break (MSLB)	Reanalyzed
	Feedwater Line Break (FWLB)	Reanalyzed
15.1.15	Inadvertent Loading of a Fuel Assembly into the Improper Position	Not Reanalyzed
15.1.16	Waste Gas Decay Tank Leakage or Rupture	Not Reanalyzed
15.1.17	Failure of Air Ejector Lines (BWR)	Not Applicable
15.1.18	Steam Generator Tube Rupture with or without a Concurrent Loss of AC Power (SGTR)	Evaluated
15.1.19	Failure of Charcoal of Cryogenic System (BWR)	Not Applicable
15.1.20	CEA Ejection	
	HZP	Reanalyzed
	HFP	Reanalyzed
15.1.21	The Spectrum of Rod Drop Accidents (BWR)	Not Applicable
15.1.22	Break in Instrument Line or Other Lines from Reactor Coolant Pressure Boundary that Penetrate Containment	Not Reanalyzed
15.1.23	Fuel Handling Accident	Not Reanalyzed
15.1.24	Small Spills or Leaks of Radioactive Material Outside Containment	Not Reanalyzed
15.1.25	Fuel Cladding Failure Combined with Steam Generator Leak	Not Reanalyzed
15.1.26	Control Room Uninhabitability	Not Reanalyzed
15.1.27	Failure or Over presssurization of Low Pressure Residual Heat Removal System	Not Reanalyzed
15.1.28	Loss of Condenser Vacuum (LOCV)	Not Reanalyzed (See 15.1.7)
15.1.29	Turbine Trip with Coincident Failure of Turbine Bypass Valves to Open	Not Reanalyzed (See 15.1.7)
15.1.30	Loss of Service Water System	Not Reanalyzed
15.1.31	Loss of one DC System	Not Reanalyzed
15.1.32	Inadvertent Operation of ECCS During Power Operation	Not Reanalyzed
15.1.33	Turbine Trip with Failure of Generator Breaker to Open	Not Reanalyzed
15.1.34	Loss of Instrument Air System	Not Reanalyzed
15.1.35	Malfunction of Turbine Gland Sealing System	Not Reanalyzed
15.1.36	Transients Resulting from the Instantaneous Closure of a Single MSIV	Reanalyzed

Many of the analysis presented below were performed for Cycle 13 which include the core physics data for Cycle 13. Moderator temperature coefficient, fuel temperature coefficient (Doppler curve), delayed neutron fractions, effective neutron lifetime, and control element assembly (CEA) reactivity insertion curves are core physics parameters that are typically considered on a cycle specific basis and are inputs to many of the analyses discussed below. The Cycle 13 set of physics data will be presented first, allowing the respective analyses to refer to this data as it is applied. Detailed core physics data that affects a particular analysis will be discussed for that analysis.

Figure 2 represents the Cycle 13 fuel temperature reactivity curves for Beginning of Cycle (BOC) and End of Cycle (EOC). These curves include a 0.85 multiplier for uncertainty on BOC reactivity and a 1.4 multiplier for uncertainty on EOC reactivity. This curve has been used as specified in the specific analysis.

A moderator temperature coefficient within the ranges defined in Figure 1 was assumed in the following analyses.

A new CEA reactivity insertion curve was developed for the Cycle 13 analyses. This new curve is presented in Figure 3. The scram curve is based on an Axial Shape Index (ASI) of + 0.3. A CEA insertion curve consistent with Figure 4 utilizing a 0.6 second holding coil delay time and a 3.2 second arithmetic average drop time to 90% inserted was assumed. A shutdown worth of 5% $\Delta\rho$ is incorporated into Figure 3. Figure 3 has been used as specified in the following analyses.

The following effective neutron lifetime and delayed neutron fraction were established for the Cycle 13 analyses. These parameters are used as indicated in the respective analyses.

	Neutron Lifetime (10^{-6} sec.)	Delayed Neutron Fraction
Beginning of Cycle	13	0.007252
End of Cycle	36	0.004341

The largest impact that a decrease in RCS flow has on plant operation is the reduction in operating margin to the DNBR and LHR limits. ANO-2 is a Core Operating Limits Supervisory System (COLSS) / Core Protection Calculator (CPC) plant that uses these systems to monitor the DNBR and LHR margins. The reduction in RCS flow that occurs as a result of additional steam generator tube plugging produces a reduction in the operating margin to the DNBR and LHR limits as calculated by the CPCs and COLSS. In the recent past, operating margin has been gained by reducing the cold and hot leg RCS temperatures and implementation of CEN-356(V)-P-A, Revision 01-P-A, "Modified Statistical Combinations of Uncertainty." The fuel reloads will be modified as additional margin is needed to account for future flow reductions from steam generator tube plugging. The fuel peaking factors can be controlled in the fuel reloads to ensure that adequate operating margin is maintained in the future.

CONTAINMENT FUNCTIONAL DESIGN, SAR SECTION 6.2.1

Reducing RCS flow by 10% will result in an increase in the hot leg temperature for a given cold leg temperature. Increasing the hot leg temperature may result in an increase in the energy within the RCS liquid. The energy increase in the RCS liquid is minimized due to the density decrease and resulting RCS mass reduction. Increasing the energy in the RCS could impact the containment peak pressure analysis associated with a large break LOCA.

A small increase in temperature due to a reduction in RCS flow results in a small increase in the RCS system energy and a decrease in the RCS system mass. A 10% reduction in RCS flow is attributed to 30% steam generator tube plugging. Currently, the ANO-2 steam generators are approximately 14% plugged. Mass and energy reductions due to the associated RCS volume decrease from steam generator tube plugging more than offset the effects of a small increase in hot leg temperature. Based on this assessment, the current large break LOCA peak containment pressure analysis remains bounding.

LARGE BREAK ANALYSIS, SAR SECTION 6.3.3.2.2

An analysis for Cycle 13 has been performed to support an increase in the number of plugged steam generator tubes and a decrease in the RCS flow rate. The analysis was performed using the NRC approved June 1985 version of the ABB CE large break LOCA evaluation model (Reference 13, Supplement 3-P-A). This is the same version used in the analysis of record.

The analysis was performed for the equivalent of up to 30% steam generator tube plugging per steam generator and an RCS flow rate of 107.8×10^6 lbm/hr. Table 1 lists the significant core and system parameters used in the analysis. The analysis was performed for the 0.6 Double Ended Guillotine/Pump Discharge (DEG/PD) break which is the limiting break from the previous analysis. This analysis was performed at a hot rod average burnup of 40,000 MWD/MTU.

Results of the analysis are presented in Tables 2 and 3 and Figures 5a through 5r. Table 3 lists the peak cladding temperature and oxidation percentages for the analysis. Times of interest are listed in Table 2. The figures present the transient results for the variables listed in Table 5. The results demonstrate conformance to the ECCS acceptance criteria as summarized below.

Parameter	Criterion	Result
Peak Cladding Temperature	≤ 2200°F	2158°F
Maximum Cladding Oxidation	≤ 17%	7.2%
Maximum Core-Wide Oxidation	≤ 1%	< 0.99%

Based on the results of the analysis, it is concluded that the ANO-2 ECCS design satisfies the acceptance criteria of 10 CFR 50.46 for large break LOCA for the conditions analyzed in this analysis. These include the equivalent of up to 30% steam generator tube plugging per steam generator and a minimum RCS flow rate of 107.8×10^6 lbm/hr.

SMALL BREAK ANALYSIS, SAR SECTION 6.3.3.2.3

The following Small Break LOCA analysis was presented in the Amendment No. 179 submittal. This submittal requested approval for the use of CENPD-137 Supplement 1-P (Reference 6) for performing the LOCA analysis. Included in the submittal was the application of the model to ANO-2 accounting for a 10% reduction in RCS flow and 30% steam generator tube plugging.

Evaluation Model

The small break LOCA analysis was performed using the ABB CE small break LOCA evaluation model (Reference 6, Supplement 1-P). The evaluation model was approved by the NRC in Reference 11. In the ABB CE small break LOCA evaluation model, the CEFLASH-4AS computer program (Reference 12) is used to perform the hydraulic analysis of the RCS until the time the Safety Injection Tanks (SITs) begin to inject. After injection from the SITs begins, the COMPERC-II computer program (Reference 8) is used to perform the hydraulic analysis. The hot rod cladding temperature and maximum cladding oxidation are calculated by the STRIKIN-II computer program (Reference 10) during the initial period of forced convection heat transfer and by the PARCH computer program (Reference 9) during the subsequent period of pool boiling heat transfer. Core-wide cladding oxidation is conservatively represented as the rod-average cladding oxidation of the hot rod. The initial steady state fuel rod conditions used in the analysis are determined using the FATES3B computer program (Reference 7).

Safety Injection System Parameters

The ANO-2 ECCS consists of three High Pressure Safety Injection (HPSI) pumps, two Low Pressure Safety Injection (LPSI) pumps, and four SITs. Each HPSI pump injects into one of the two high pressure injection headers which feed each cold leg. Throttle valves in each of the cold legs are used for flow balancing. The LPSI pumps inject to a common header which feeds each cold leg. Each SIT injects to a single cold leg. Two HPSI pumps and the LPSI pumps are automatically actuated by a safety injection actuation signal that is generated by either low pressurizer pressure or high containment pressure. The SITs automatically discharge when the RCS pressure decreases below the SIT pressure.

In the small break LOCA analysis it is assumed that offsite power is lost coincident with reactor trip and, therefore, the HPSI and LPSI pumps must await emergency diesel generator startup and load sequencing before they start. The total delay time assumed is 40 seconds from the time the pressurizer pressure reaches the Safety Injection Actuation Signal (SIAS) setpoint to the time that the HPSI pumps are at speed and aligned to the RCS. For breaks in the reactor coolant pump discharge leg all safety injection flow delivered to the broken discharge leg is modeled to spill out the break.

An analysis of the possible single failures that can occur within the ECCS has shown that the most damaging single failure of ECCS equipment is the failure of an emergency diesel generator to start (Reference 6). This failure causes the loss of both a HPSI and LPSI pump and results in a minimum of safety injection water being available to cool the core.

Based on the above, the following safety injection flows are credited in the small break LOCA analysis for a break in the reactor coolant pump discharge leg: 75% of the flow from one HPSI pump, 50% of the flow from one LPSI pump and 100% of the flow from three SITs. Table 6 presents the HPSI pump flow rate versus RCS pressure used in the small break LOCA analysis.

Core and System Parameters

The significant core and system parameters used in the small break LOCA analysis are presented in Table 7. For the 0.05 ft^2 and the 0.06 ft^2 break sizes, the Main Steam Safety Valve (MSSV) first bank opening pressure was assumed to be 1125 psia. For the 0.02 ft^2 and 0.04 ft^2 break sizes, the MSSV first bank opening pressure was assumed to be 1103.5 psia. The low pressurizer pressure reactor trip and SIAS setpoints were assumed to be 1400 psia for the 0.02 ft^2, 0.05 ft^2, and 0.06 ft^2 break sizes. The low pressurizer pressure reactor trip was assumed to be 1625 psia, and the low pressurizer pressure SIAS setpoint was assumed to be 1578 psia for the 0.04 ft^2 break size. The fuel rod initial conditions were taken at the burnup that produced the maximum initial stored energy. The analysis accounts for up to 30% steam generator tube plugging per steam generator.

Containment Parameters

The small break LOCA analysis does not use a detailed containment model. Therefore, other than the containment volume and the initial containment pressure, which are assumed to be 1,820,000 ft^3 and 14.7 psia, respectively, no containment parameters are employed in the analysis.

Break Spectrum

The break spectrum consisted of four reactor coolant pump breaks ranging in size from 0.02 ft^2 to 0.06 ft^2. Table 8 lists the specific break sizes that were analyzed.

The reactor coolant pump discharge leg was previously determined to be the limiting break location (Reference 6). It is limiting because it maximizes the amount of spillage from the safety injection system.

The break size range of 0.02 ft^2 to 0.06 ft^2 encompasses the break sizes for which hot rod cladding heatup is terminated solely by injection from the HPSI pump. It is within this range that the limiting small break LOCA, the 0.05 ft^2 break, resides. Breaks outside this range are either too small to experience any significant core uncovery or are sufficiently large such that injection from the SITs will recover the core and terminate cladding heatup before the cladding temperature approaches the peak cladding temperature calculated for the limiting small break LOCA.

Results and Conclusions

The peak cladding temperatures and cladding oxidation percentages for the small break LOCA analysis are summarized in Table 9. Table 10 lists times of interest for the breaks analyzed. As noted in Table 8, results for the variables listed in Table 11 are plotted as a function of time in Figures 6a through 9h for the breaks analyzed. Peak cladding temperature versus break size is presented in Figure 10.

Based on the results of the analysis, it is concluded that the ANO-2 ECCS design satisfies the Acceptance Criteria of 10CFR50.46 for a spectrum of small break LOCAs.

Energy Redistribution Factor Part 21 Issue

On July 11, 1997, ABB-CE issued Infobulletin 97-04, Revision 1 which reported the initiation of a 10 CFR 21 evaluation of the Energy Redistribution Factors (ERF) used in the ECCS performance analyses using ABB-CE's Large and Small Break LOCA ECCS performance models. ERF represents the fraction of the total energy generated by a fuel rod which is actually deposited in the rod. It was determined that the ERF used by ABB-CE in the LOCA analyses did not directly reflect the effects of moderator voiding during a LOCA and such effects have recently been calculated to be somewhat higher than previously thought. This error affects only the Large Break LOCA analysis significantly, since the Small Break LOCA analysis is insensitive to the ERF. On August 14, 1997, ABB-CE issued a Part 21 report to the NRC (LD-97-024) concerning this issue.

ABB-CE is currently working to recalculate the ERF. Once this is completed, an assessment on a plant specific basis will be made on the impact on the peak clad temperature calculated in the LOCA analysis. Due to the timing of this submittal, the LBLOCA and SBLOCA assessments presented above have not accounted for this issue. The impacts of the identified issue will be addressed consistent with the requirements of 10 CFR 50.46.

UNCONTROLLED CONTROL ELEMENT ASSEMBLY (CEA) WITHDRAWAL FROM A SUBCRITICAL CONDITION, SAR SECTION 15.1.1

Considerations for the CEA withdrawal event from subcritical conditions include minimum DNBR and fuel centerline melt. The reduction in RCS flow will have an impact on these considerations; hence, this event was reanalyzed. This event was reassessed for Cycle 13 with reduced RCS flow. The results of this effort are presented below.

The withdrawal of CEAs from subcritical conditions (less than 10^{-4} percent power) adds reactivity to the reactor core, causing both the core power level and the core heat flux to increase. Since the transient is initiated at low power levels, the normal reactor feedback mechanisms, moderator feedback, and Doppler feedback do not occur until power generation in the core is large enough to cause changes in the fuel and moderator temperatures. The Reactor Protection System (RPS) is designed to prevent such a transient from resulting in a minimum DNBR less than 1.25 by a high logarithmic power level reactor trip. The high linear power level, and the Core Protection Calculator (CPC) variable overpower trip (VOPT) high local power density and low DNBR trips provide backup protection while the high pressurizer pressure trip provides protection for the Reactor Coolant Pressure Boundary.

A continuous withdrawal of CEAs could result from a malfunction in the Control Element Drive Mechanism Control System (CEDMCS) or by operator error.

Startup of the reactor involves a planned sequence of events during which certain CEA groups are withdrawn, at a controlled rate and in a prescribed order, to increase the core reactivity gradually from subcritical to critical. To ensure that rapid shutdown by CEAs is always possible when the reactor is critical or near critical, Technical Specifications require that specified groups of CEAs be withdrawn before reaching criticality. These groups of assemblies combined with soluble boron concentration will have a total negative reactivity worth that is sufficient to provide at least the Technical Specification required shutdown margin at the hot standby condition, with the most reactive CEA assumed to remain in the fully withdrawn position.

The CEA Withdrawal from Subcritical conditions was analyzed using CENTS and CETOP computer codes. CENTS is described in Reference 2. Two reactivity addition rates were considered, 0.00025 $\Delta\rho$/sec and 0.0002 $\Delta\rho$/sec as Case 1 and 2 respectively. These reactivity addition rates are consistent with the maximum addition rates expected for bank withdrawals near critical conditions. Due to the planned sequence of events for a controlled startup, boron concentrations are maintained at levels which prevents criticality for most CEA bank withdrawals. Under certain conditions criticality can be attained with the right combination of CEA bank withdrawal and boron concentration. Only bank withdrawals which will result in critical conditions are considered for this event. The inputs used in these analyses are provided in Table 12, the Cycle 13 physics data above, and the following assumptions:

A. A steam generator tube plugging limit of 30% was considered.

B. CEA scram worth was not credited on trip, rather a CEA coil decay time of 0.6 seconds was assumed followed by negative reactivity proportional to the CEA position post trip. Reactivity is held constant for the 0.6 second delay time. After the 0.6 second delay, negative reactivity equivalent to the positive reactivity added prior to the trip is inserted, at a rate consistent with the CEA position versus time curve of Figure 4.

C. The BOC Doppler curve of Figure 2, which includes a 0.85 multiplier, is conservatively used.

D. The Cycle 13 delayed neutron fraction and effective neutron lifetime consistent with the above was assumed.

The sequence of events for these two reactivity insertion rate transients is provided in Tables 13 and 14. The maximum predicted fuel centerline temperature is less than 2800°F and the minimum DNBR is greater than 1.25. Based on these results the specified acceptable fuel design limits (SAFDLs) and the RCS pressure boundary limits are not violated.

UNCONTROLLED CEA WITHDRAWAL FROM CRITICAL CONDITIONS, SAR SECTION 15.1.2

Similar to the subcritical CEA withdrawal, considerations for the CEA withdrawal event from critical conditions include minimum DNBR and fuel centerline melt. Reducing RCS flow has a minimal impact on these considerations for this event. This event was reassessed for Cycle 13 with reduced RCS flow. The results of this effort are presented below.

The withdrawal of CEAs from a critical condition (greater than 10^{-4} percent power) adds reactivity to the reactor core, causing the core power level to increase. A continuous withdrawal of CEAs could result from a malfunction in the Reactor Regulating System (RRS), the CEDMCS or by operator error. No failure which can cause CEA withdrawal or insertion can prevent the insertion of CEA banks upon receipt of any protective system reactor trip signal.

Analyses have shown that the most adverse results for the CEA withdrawal events occur with the maximum reactivity addition rates. The analysis of the CEA withdrawal from critical conditions therefore utilizes the maximum reactivity addition rate with the CEA withdrawal speed of 30 in/minute.

The CEA withdrawal event from critical conditions is considered from hot zero power (HZP) and hot full power (HFP) conditions. An assessment of the HZP case will be presented first followed by an evaluation of the HFP condition.

CEA Withdrawal from HZP

A CEA withdrawal from HZP conditions was analyzed using CENTS and CETOP computer codes. The inputs used in this analysis are provided in Table 15, the Cycle 13 physics data above, and the following assumptions:

A. A steam generator tube plugging limit of 30% was modeled.

B. The worth of the CEAs at trip was assumed to be 2%. The CEA drop time is consistent with Figure 4 with the 0.6 second holding coil delay time; however, a more conservative normalized reactivity insertion versus CEA position for a +0.6 ASI curve was assumed.

C. The BOC Doppler curve of Figure 2, which includes a 0.85 multiplier, is conservatively used.

D. The Cycle 13 delayed neutron fraction and effective neutron lifetime consistent with the above information was assumed.

The sequence of events for this transient is provided in Table 16. The maximum fuel centerline temperature is less than 3330 °F and the minimum DNBR is greater than 1.25. Based on these results the specified acceptable fuel design limits (SAFDLs) are not violated.

CEA Withdrawal from HFP

The CEA bank withdrawal event was examined as the fastest rate of increasing power with respect to the anticipated operational occurrences (AOOs) for which the CPCs ensures that the SAFDLs would not be violated. An evaluation was performed to validate that the response of the CPC compensated neutron flux power for a CEA withdrawal event is conservative with respect to the actual rates for both the core power and core heat flux increase given this event. By ensuring the CPC protective calculations are conservative, the SAFDLs would not be violated.

As the purpose of this assessment is to ensure CPCs perform their protective function, the dynamic effects of a CEA withdrawal event that result in the most challenging rate of power increase needs to be considered. A sensitivity study was performed on RCS flow validating that high initial RCS flow rates are the most challenging; hence, the reduction in RCS flow does not have a significant impact on this event.

CEA MISOPERATION, SAR SECTION 15.1.3

The CEA drops are considered as part of the required overpower margin (ROPM) events. The analyses calculating the ROPMs are considered for each reload cycle in the determination of COLSS inputs and operating limits which assure that the DNBR SAFDL would not be exceeded. The ROPMs for CEA withdrawal, loss of RCS flow events, asymmetric steam generator transient, full length CEA drops, and other anticipated operational occurrences are determined to find the most limiting value. The full length CEA drop events produce reductions in power, relatively slow changes to the core power distribution, and are much less significant for the purposes of determining COLSS inputs and operating limits.

Although the reduced RCS flow would have a slight impact on DNBR following a CEA drop, the COLSS inputs and operating limits established with each reload will assure that the DNBR SAFDL will not be exceeded in the event of a dropped CEA. A specific reanalysis of the event to account for RCS flow reduction effects is unnecessary as other anticipated operational occurrences remain bounding. The power distortion factors resulting from a dropped rod, which are a measure of the power distribution upset, and thus the relative significance of the transient, are compared to bounding values for each reload. This assures that the CEA drop event would be reanalyzed if required.

UNCONTROLLED BORON DILUTION INCIDENT, SAR SECTION 15.1.4

The Uncontrolled Boron Dilution Incident is unaffected by the proposed reduction in RCS flow. Although flow is qualitatively assumed to exist to promote mixing, it is not a quantitative input to the analyses. The plugging of steam generator tubes, which causes the flow reduction, also reduces the RCS volume. For the more limiting dilution events in Modes 3, 4, 5 and 6, the reactor coolant pumps are assumed to be off and the stagnant volume of the steam generators is conservatively not included in the dilution volume. Thus the volume reduction has no impact in these operational modes. Since the reactor coolant pumps are running for the events in Modes 1 and 2, the full volume of the RCS (less the pressurizer and surge line) is included in the dilution volume. However, the uncontrolled boron dilution incidents in Modes 1 and 2 are much less limiting because of the large dilution volume which reduces the rate of boron dilution and a boron dilution event in Modes 1 or 2 will result in a rapid reactor shutdown by the reactor protection system. Consequently, the volume reduction does not significantly impact the events in Modes 1 and 2. For purposes of comparison, the time of 93 minutes from the start of the event to the loss of shutdown margin, included in Reference 4 for Mode 2, is decreased to 86 minutes with 30% of the steam generator tubes plugged.

TOTAL AND PARTIAL LOSS OF REACTOR COOLANT SYSTEM (RCS) FORCED FLOW, SAR SECTION 15.1.5

A loss of reactor coolant forced flow can result from the occurrence of a mechanical or electrical failure. A partial loss of flow can occur as the result of a mechanical or electrical failure in a reactor coolant pump or from a loss of power to the pump bus. A complete loss of coolant flow results from a simultaneous loss of electrical power to all operating reactor coolant pumps. A four pump loss of flow event due to a simultaneous loss of electrical power and a seized rotor event are considered separately below. Due to the Technical Specification Limiting Conditions for Operation (LCOs) on DNBR margin, by the response of the RPS which provides an automatic reactor trip as calculated by the CPCs, and Core Operating Limits Supervisory System (COLSS) calculating the power operating limit to ensure adequate thermal margin to DNB, the effects of a reduction in initial RCS flow for these events does not have a significant impact. Consideration of these events relates more to the potential impact of 30% steam generator tube plugging, as the increased system resistance could affect the post event RCS flow.

FOUR PUMP LOSS OF COOLANT FLOW ANALYSIS

To determine the impacts of a 10 percent reduction in RCS design flow and 30 percent steam generator tube plugging on the Four Pump Loss of Flow analysis, the following evaluation was performed. This evaluation has employed the HERMITE computer code (Reference 1) instead of the CESEC code used previously for this event. The CENTS computer code (Reference 2) has replaced the COAST program for calculating the RCS flow coastdown.

For a loss of flow at any power operating condition, a reactor trip will be initiated when any one of four Reactor Coolant Pump (RCP) shaft speeds drops to 95 percent of its nominal speed. In this method, the partial loss of flow resulting from a loss of electrical power to three or less RCPs is less limiting than a four pump loss of flow. This is because the reactor will trip at the same time for both cases but the partial loss of flow has a slower flow coastdown. Therefore, only the four pump loss of flow event is presented herein.

Method of Analysis

The analysis was carried out in the following steps:

A. The RCP coastdown data for the loss of flow event was generated using the CENTS code. The use of the CENTS code is a change from the original coastdown analysis which used the COAST code.

Coastdown data to account for 30% steam generator tube plugging was determined by first benchmarking the CENTS coastdown results against the original coastdown data from the COAST code and plant specific coastdown data. The CENTS basedeck was then adjusted to account for the 30% steam generator tube plugging. The CENTS coastdown analysis considered the affects of both symmetric and asymmetric steam generator tube plugging (up to 1000 tube asymetry). The coastdown analysis also considered the effects of initial RCS pressure, temperature, and flow. The resulting coastdown data generated from CENTS was used as input to the HERMITE code.

B. The HERMITE code is used to determine the reactor core response during the postulated loss of flow event. The HERMITE code solves the few-group, space and time dependent neutron diffusion equation including the feedback effects of fuel temperature, coolant temperature, coolant density, and control rod motion for a one-dimensional average fuel bundle.

C. The time dependent thermal hydraulic information generated from the HERMITE code is transferred directly to the CETOP computer code (Reference 3) for thermal margin and DNBR evaluation. The CETOP method was used to calculate both the time of occurrence and value of the minimum DNBR during the transient.

Input Parameters and Initial Conditions

The four pump loss of flow event used the conservative assumptions provided in Table 17 including the Cycle 13 physics data and the following assumptions:

A. A CEA insertion curve consistent with the CEA position versus time presented in Figure 4 was assumed. This curve accounts for a 0.6 second holding coil delay.

B. A BOC delayed neutron fraction of 0.0072546 was assumed.

C. A BOC fuel temperature coefficient of -0.0013 $\Delta\rho/\sqrt{°K}$ was assumed.

D. For this analysis, a trip on low RCP speed is the primary trip for the loss of flow event, replacing the trip on low flow-projected DNBR. A CPC trip is initiated when the RCP shaft speed drops to 95 percent of its normal speed.

The four pump loss of coolant flow produces an approach to the DNBR limit due to the decrease in the core coolant flow. Protection against the DNBR limit for this transient is provided by the initial steady state thermal margin which is maintained by adhering to the Technical Specification LCOs on DNBR margin and by the response of the RPS which provides an automatic reactor trip as calculated by the CPCs.

The principal process variables that determine thermal margin to DNB in the core are monitored by the COLSS. The COLSS computes a power operating limit which ensures that the thermal margin available in the core is equal to or greater than that needed to cause the minimum DNBR to remain greater than the DNBR limit. The minimum thermal margin required (reserved) in COLSS for the loss of flow event is set equal to the maximum thermal margin degradation observed during a loss of flow event.

The initial conditions are selected such that the system is at a very subcooled state. Initiating the event from such a state results in the least amount of negative reactivity inserted due to generation of voids in the RCS. In this manner the system undergoes the greatest amount of thermal margin degradation due to the RCP coastdown.

To demonstrate explicitly that the DNBR SAFDL is not violated during a loss of flow event, a sample case is provided in which the initial conditions are chosen such that at the onset of the event the minimum thermal margin required by the COLSS power operating limit is preserved. This analysis has used an RCS flow of 108.36 Mlbm/hr which is 90 percent of the minimum design flow corresponding to 30 percent tube-plugging. Figure 11 provides a graph of the RCS flow coastdown used for the loss of flow event with 30% steam generator tube plugging. The consequences following a total loss of forced reactor coolant flow, with respect to approaching the DNBR SAFDL, initiated from any set of initial conditions which preserve the minimum COLSS margin would be no more adverse than those presented herein.

Results

The results of this analysis is the calculation of minimum thermal margin required to be reserved in COLSS to prevent the violation of the DNBR SAFDL during a loss of flow event. With a minimum thermal margin reserved in COLSS, the minimum DNBR observed during this event is 1.29 at 2.8 seconds. The sequence of events for the four pump loss of flow assuming 30% steam generator tube plugging is provided in Table 18. Figure 12 provides a graph of DNBR versus time for the event.

For the loss of flow event, the CPC trip on pump low speed in conjunction with the initial margin reserved in COLSS is sufficient to prevent the violation of the DNBR SAFDL from any set of initial conditions.

SEIZED ROTOR

When analyzing the seized rotor event, the event is initiated from a power operating limit with the minimal acceptable thermal margin to the DNBR limit. Based on this consideration, the initial RCS flow does not have a significant impact on the analysis results. Rather, the change in flow rate from the initial value to the final flow rate is a critical parameter. Due to the potential that increased tube plugging may affect the change in flow rate, an evaluation was performed to determine the effective change in flow rate due to 30% steam generator tube plugging.

This analysis concluded that the final "steady state" flow fraction for the 30% steam generator tube plugging case is essentially equal to the "steady state" flow fraction used in the analysis of record. The coastdown data for the seized rotor event was generated using the CENTS code. The use of the CENTS code is a change from the original coastdown analysis which used the COAST code.

The analysis of record seized rotor event assumes an instantaneous drop from the initial flow rate to the reduced "steady state" flow fraction. Based on the above, this assumption remains valid; therefore, a reanalysis of the seized rotor event was not required.

IDLE LOOP STARTUP, SAR SECTION 15.1.6

Idle loop startup is defined as the startup of a reactor coolant pump, without observance of prescribed operating procedures, assuming that both reactor coolant pumps in that loop were idle. ANO-2 was originally designed to permit continued operation with one or two reactor coolant pumps idle. The Technical Specifications for ANO-2, however, precluded critical operation with any inoperative pumps. As the conditions leading to this event are not allowed by the Technical Specifications no consideration was given for a reduction in RCS flow.

LOSS OF EXTERNAL LOAD AND/OR TURBINE TRIP, SAR SECTION 15.1.7

Loss of external load and/or turbine trip results in a reduction of steam flow from the steam generators to the turbine generator. Cessation of steam flow to the turbine generator occurs because of closure of the turbine stop valves or turbine control valves. The cause of loss of load may be abnormal events in the electrical distribution network or turbine trip.

The bounding event considered is a loss of load event initiated by a turbine trip without a simultaneous reactor trip and assuming the Steam Dump and Bypass system is inoperable. If the turbine trip were caused by a Loss of Condenser Vacuum, the main feedwater pump steam turbines would trip at the same time. Therefore, a loss of load concurrent with loss of feed was analyzed to cover these events. The loss of load causes steam generator pressure to increase to the opening pressure of the main steam safety valves. The reduced secondary heat sink leads to a heatup of the RCS. The transient is terminated by a reactor trip on high pressurizer pressure.

The loss of external load and/or turbine trip was undertaken for Cycle 13 to account for Cycle 13 input parameter variations and considering the effects of 30% tube plugging and a reduction in RCS flow. For the analysis presented herein, the CENTS computer code described in Reference 2 was utilized.

Sensitivity studies were conducted on the effects of steam generator tube plugging and reductions in RCS flow. The results of these sensitivity studies indicated that RCS flow has a very minor impact on the analysis results with higher RCS flows resulting in slightly higher peak primary pressures and lower RCS flows resulting in higher peak secondary pressures. The effects of steam generator tube plugging indicated that no steam generator tube plugging was slightly more conservative for both primary and secondary peak pressures.

Input parameters from Table 19 and the Cycle 13 physics parameters above have been incorporated in the following peak RCS pressure analysis.

A summary of the principal results for the loss of external load/loss of condenser vacuum are given in Table 20. These results indicate that the peak primary pressure is 2683 psia and the peak secondary pressure is 1162 psia. A separate analysis was performed to determine a conservative peak secondary pressure, as the input assumptions described above and denoted in Table 19 are established to ensure a peak primary pressure. This second analysis is effectively the same as the peak primary analysis except the input assumptions delineated above are adjusted to ensure a conservative peak secondary pressure. The results of this second effort indicate a peak secondary pressure of 1195 psia.

The results of these analyses shows that the peak RCS and secondary side pressures are maintained less than 110% of design values.

LOSS OF NORMAL FEEDWATER FLOW, SAR SECTION 15.1.8

The loss of normal feedwater flow is defined as a reduction in feedwater flow to the steam generators when operating at power, without a corresponding reduction in steam flow from the steam generators. The result of this mismatch is a reduction in the water inventory in the steam generators.

The Emergency Feedwater (EFW) system is available to automatically provide sufficient feedwater flow to remove residual heat generation from the RCS following a reactor trip from rated power. This system consists of one motor-driven and one turbine-driven emergency feedwater pump, and a non-safety Auxiliary Feedwater pump.

A complete loss of both main feedwater pumps or all four condensate pumps and the turbine driven pump results in the loss of all normal feedwater. In manual feedwater control, closing the feedwater regulating or isolation valves also results in loss of normal feed flow.

The Plant Protection System provides protection against loss of the secondary heat sink by the steam generator low water level trip and automatic initiation of the EFW system. The high pressurizer pressure trip provides protection in the event that the RCS pressure limit is approached.

The impacts of reducing RCS flow on this event were considered. Based on a sensitivity study performed on RCS flow, lower RCS flow rates resulted in lower post trip steam generator inventories. The following analysis was performed for Cycle 13 with reduced RCS flow. This evaluation has utilized the CENTS computer code described in Reference 2.

Inputs from Table 21 and Cycle 13 physics data presented above were used in this analysis with the following clarifications:

A. An EFW response time of 97.4 seconds was assumed. EFW flow was determined based on steam generator pressure. Prior analysis efforts assumed a constant flow rate regardless of steam generator pressure.

B. The EOC Doppler curve in Figure 2 which includes a 1.4 multiplier is conservatively used.

C. The Cycle 13 delayed neutron fraction and neutron lifetime consistent with the data presented above was assumed.

D. The Cycle 13 CEA insertion curve in Figure 3 was utilized. This curve accounts for a 0.6 second holding coil delay.

E. An MSSV tolerance of -3% is conservatively assumed.

A summary of the principal results for the loss of normal feedwater flow is given in Table 22. These results support the conclusion that the steam generator heat removal capability is maintained.

LOSS OF ALL NORMAL AND PREFERRED AC POWER TO THE STATION AUXILIARIES, SAR SECTION 15.1.9

The loss of AC power is defined as a complete loss of preferred (off-site) AC electrical power and a concurrent turbine generator trip. As a result, electrical power would be unavailable for the station auxiliaries such as the reactor coolant pumps, the main feedwater pumps and the main circulating water pumps. Under such circumstances, the plant would experience a simultaneous loss of load, feedwater flow, and forced reactor coolant flow.

This event was not reanalyzed for the reduction in RCS flow rate. As indicated above for the four pump loss of flow, loss of external load, and loss of normal feedwater events, reducing RCS flow has minimal impact on these events. Additionally, the minimum DNBR considerations for this event are bounded by the consideration made in the four pump loss of flow event; hence, reducing RCS flow rate has been determined not to have a significant impact on the loss of all AC event.

EXCESS HEAT REMOVAL DUE TO SECONDARY SYSTEM MALFUNCTION, SAR SECTION 15.1.10

The excess heat removal events include several different transients that place an increased heat demand on the primary system. Steam and feedwater system malfunctions were considered for their potential impact on the fuel design limits. Various valve failures in both systems were evaluated to determine those that would cause the greatest increase in secondary heat removal. With the assumption of a negative moderator temperature coefficient, these events produce an increase in core power and a reduction in DNBR. Depending on the extent of the cooldown, the event may be ended by a trip, or a new equilibrium condition, at a higher power level could result.

As overcooling events, the dynamic impact of the transient on the primary system is directly dependent on the rate of heat transfer through the steam generators. The reduced heat transfer resulting from tube plugging will slow the cooling of the primary system. The reduced RCS flow will tend to increase the rate of primary cooldown for a given rate of heat transfer. These changes will affect the dynamics of the transients which will impact those events that lead to a reactor trip. The CPCs and RPS assure that a reactor trip will occur before the SAFDLs are exceeded by an excess heat removal event.

To assure that the CPCs can accurately sense the cooldown associated with an excess heat removal event, even with the change in transient dynamics due to tube plugging and RCS flow reductions, a CPC transient filters analysis was performed for Cycle 13. The CPC transient filters analysis verifies the CPC adjusted process parameters are conservative with respect to the expected values for a given transient event. The CPC coefficients are adjusted as necessary to assure the CPC action prevents SAFDL violation during the transient. This analysis included parametric studies on RCS flow and tube plugging to determine the limiting values of these inputs. The design minimum RCS flow reduced by 10%, and 0% tube plugging were limiting assumptions to a CENTS analysis of an excess heat removal event. The results of the analysis verifies proper detection of significant overcooling transients and conservative CPC actions. Consequently, the effects of tube plugging and reduced flow on the significant excess heat removal events have been evaluated. This evaluation ensures that the CPCs and RPS will provide the necessary trip functions to prevent the SAFDLs from being violated.

FAILURE OF THE REGULATING INSTRUMENTATION, SAR SECTION 15.1.11

A reactor coolant flow controlled malfunction is not possible. ANO-2 does not have coolant flow controllers. Therefore, a reduction in the RCS flow will not affect this event.

INTERNAL AND EXTERNAL EVENTS INCLUDING MAJOR AND MINOR FIRES, FLOODS, STORMS, AND EARTHQUAKE, SAR SECTION 15.1.12

RCS flow is not a consideration in these events as such no evaluation is necessary for a reduction in RCS flow.

MAJOR RUPTURE OF PIPES CONTAINING REACTOR COOLANT UP TO AND INCLUDING DOUBLE-ENDED RUPTURE OF LARGEST PIPE IN THE REACTOR COOLANT SYSTEM (LOCA), SAR SECTION 15.1.13

This section of the SAR, Section 15.1.13, relates only to the consequences of a LOCA. RCS flow is not a consideration with respect to the offsite releases from a LOCA. The limiting doses to the control room operator, which result from LOCA releases, are similarly unaffected by RCS flow considerations.

The requirements with respect to 10CFR50.46 are covered in Section 6.3.3 of the SAR. RCS flow is a parameter for consideration in this event which is discussed above.

MAJOR SECONDARY SYSTEM PIPE BREAKS WITH OR WITHOUT A CONCURRENT LOSS OF AC POWER - MAIN STEAM LINE BREAK (MSLB) AND FEEDWATER LINE BREAK (FWLB), SAR SECTION 15.1.14

STEAM LINE BREAK

A reduction in RCS flow will result in an increase in the RCS energy due to an increase in the hot leg temperature for a given cold leg temperature. This increase in energy results in a slightly larger cooldown following a MSLB. As a result, the MSLB has been evaluated. The Cycle 13 analysis accounts for a 10% reduction in RCS flow in addition to the affects of a low steam generator pressure setpoint of 620 psia. The following is a summary of the Cycle 13 analysis which includes the reduced RCS flow and various other conservative assumptions.

The no moisture carryover steam line break events were reanalyzed to account for a 10% reduction in the RCS design flow, a small increase in feedwater flow, a lower low steam generator pressure setpoint, and to address Cycle 13 physics data. CENTS was used to model the Nuclear Steam Supply System (NSSS) response, RCP coastdown and natural circulation, RELAP5 was used to model the feedwater system response for the hot full power (HFP or full load) cases, HRISE was used to calculate thermal margin on DNBR, and ROCS/HERMITE were used to assess reactivity feedback and peaking.

The analytical basis for the HFP and hot zero power (HZP) simulations are discussed below.

A. A double-ended guillotine break (6.357 ft^2) causes the greatest cooldown of the RCS and the most severe degradation of shutdown margin.

B. A break inside the containment building, upstream of the Main Steam Isolation Valves (MSIVs) and flow measuring venturis causes a non-isolable condition in the affected steam generator.

C. A SIAS is actuated when the pressurizer pressure drops below 1400 psia. Time delays associated with the safety injection pump acceleration and valve opening are taken into account. A 40 second HPSI response time was assumed to account for these delays. Additionally, the event was initiated from the highest pressure allowed by the Technical Specifications to delay the effect of the safety injection boron.

D. The cooldown of the RCS is terminated when the affected steam generator blows dry. As the coolant temperatures begin increasing, positive reactivity insertion from moderator reactivity feedback decreases. The decrease in moderator reactivity combined with the negative reactivity inserted via boron injection cause the total reactivity to become more negative.

E. CENTS is used to model the RCP coast down on a loss of offsite power. The CPC low DNBR (based on pump speed) trip is credited in this analysis following a loss of offsite power. A CPC low DNBR trip setpoint based on 96.5% of RCP speed with a 1.0 second response time is assumed.

F. A low steam generator pressure reactor trip setpoint of 620 psia was assumed with a 1.3 second response time.

G. Main Steam Isolation Signal (MSIS) is actuated on a low steam generator pressure setpoint of 620 psia. The MSIVs, Main Feedwater Isolation Valves (MFIVs) and Back-up MFIVs all receive an MSIS signal to close. A response time of 4.3 seconds was assumed for the MSIVs. The MFIVs and Back-up MFIVs were assumed to close in 36.4 seconds and 31.8 seconds with a loss of offsite power, and 21.4 seconds and 16.8 seconds with offsite power available, respectively.

H. The HERMITE code was used to calculate the reactivity for the post-trip return to power portion of the analysis. This was done since the HERMITE code, which is a three-dimensional coupled neutronics-open channel thermal hydraulics code, can more accurately model the effects of moderator temperature feedback on the power distribution and reactivity for the critical configuration existing during the return to power. The HERMITE results used in the ANO-2 analysis were actually obtained from a parametric study performed for Calvert Cliffs Unit 1 Cycle 7. ANO-2 specific ROCS calculations were used to confirm the applicability of these parametric results to ANO-2.

I. Three-dimensional power distribution peaks (Fq) were determined with the above mentioned ROCS and HERMITE evaluations. Axial profiles consistent with these conservative power distribution peaks were utilized in the analysis.

J. The power produced by the decay of the initial condition delayed neutron precursors and by nominal decay power is distributed according to the nominal power distribution.

K. The thermal margin on DNBR in the reactor core was simulated using the HRISE computer program. RCS conditions from CENTS (RCS temperature, pressure, flow, and power) are used in the HRISE thermal margin calculations.

The conservative assumptions included in the HZP and HFP simulations are discussed below.

The MTC assumed in the analysis corresponds to the most negative value. This negative MTC results in the greatest positive reactivity addition during the RCS cooldown caused by the steam line break. Since the coefficient of reactivity associated with moderator feedback varies significantly over the range of moderator density covered in the analysis, a curve of reactivity insertion versus moderator density rather than a single value of MTC is assumed in the analysis. The moderator cooldown curve used in the analysis (Figure 13) was conservatively calculated assuming that on reactor trip, the highest worth control element assembly is stuck in the fully withdrawn position. The effect of uneven temperature distribution on the moderator reactivity is accounted for by assuming that the moderator reactivity is a function of the lowest cold leg temperature.

For conservatism, the full steam generator heat transfer surface area is assumed to always be covered by the 2-phase level until a steam generator becomes essentially empty.

The reactivity defect associated with fuel temperature decrease is based on the most negative Fuel Temperature Coefficient (FTC). Figure 14 represents the FTC curve used in the analysis. This FTC, in conjunction with the decreasing fuel temperatures, causes the greatest positive reactivity insertion during the steam line break event. The delayed neutron fraction assumed is the maximum value including uncertainties for end-of-life conditions (total delayed neutron fraction, β, 0.005994). This too maximizes subcritical multiplication and thus increases the potential for return to power.

The minimum CEA worth assumed to be available for shutdown at the time of reactor trip at the maximum allowed power level is -7.5144 % $\Delta\rho$. For the HZP cases a shutdown CEA worth of -5.0 % $\Delta\rho$ was used. The scram worths used are consistent with the moderator cooldown curve and stuck rod assumed in the analysis. The CEA reactivity addition curve of Figure 3 adjusted to a worth of 7.5144 was used in the HFP cases. The HZP cases assumed a CEA drop time consistent with Figure 4 with the 0.6 second holding coil delay time; however, a more conservative normalized reactivity insertion versus CEA position for a +0.6 ASI curve was used.

The EFW system is conservatively modeled to initiate early with both EFW pumps available, this maximizes the potential cooling that could occur. System response times, flows and setpoints are assumed based on increasing the cooling potential of the EFW system.

The analysis assumed that, for the loss of AC power cases, one EDG failed to start. The failure of an EDG results in the failure of one HPSI pump and one of the main feedwater isolation valves to close. The faster closing back-up main feedwater isolation valves were assumed to remain open. For the HFP case with AC available, a bus fast transfer failure is the most limiting single failure as this failure is modeled as the failure of the back-up main feedwater isolation valves and a HPSI pump. A fast transfer failure would only result in the delayed actuation of the back-up main feedwater isolation valves and HPSI pump. These components would be actuated once the EDG has started. Therefore, the modeling of the fast transfer failure is conservative. This conservative modeling of a fast transfer failure is slightly more limiting than the single failure of a main feedwater pump to trip, which was determined to be more limiting in the Cycle 12 analysis. A single failure of a HPSI pump to start was assumed for the HZP case with AC available. The boration from the Safety Injection Tanks was not credited in this analysis.

The HFP feedwater addition to the steam generator assumed in this analysis is taken from the Cycle 12 analysis which used a RELAP5 model of the feedwater system. The steam generator pressure profiles and time of MSIS were verified to be consistent with respect to this analysis, thereby allowing the application of the feedwater data generated for Cycle 12. The HFP feedwater data for Cycle 12 was increased by 1% to account for a small expected increase in feedwater flow due to modifications to the high pressure turbine. For the hot zero power (HZP or no load) cases, feedwater flow is modeled by matching the energy input by the core at the start of the event.

The key parameters used for the post-trip steam line break analyses are listed in Table 23. Tables 24 through 27 present the sequence of events for the HFP and HZP steam line break cases with and without a concurrent loss of AC power. Figures 15 through 38 show the transient response for key parameters.

The results of this analysis indicate that the HFP cases remain subcritical through out the post trip event. The new maximum post trip reactivity values are -0.029 and -0.338 considering a loss of AC and offsite power available, respectively. The peak return to power and minimum DNBR values are 2.61% and 1.81, and 4.98% and 2.46 considering a loss of AC and offsite power available, respectively.

The HZP results of this analysis indicate a slight return to critical; however, this return to critical is bounded by the FSAR results. The new maximum post trip reactivity values are +0.252 and +0.227 considering a loss of AC and offsite power available, respectively. These values are bounded by the FSAR analysis results of +0.43 and +0.34. The peak return to power and minimum DNBR values are 0.41% and 12.3, and 1.275% and 11.2 considering a loss of AC and offsite power available, respectively.

As these results indicate acceptable DNBR values, no fuel failure is predicted. The results of the steam line break analyses demonstrated that there was no calculated fuel failure, thus the coolable geometry is maintained.

FEEDWATER LINE BREAK

The FWLB event was assessed for a lower low steam generator trip setpoint of 620 psia. During this effort the effects of a 10% reduction in RCS flow and 30% steam generator tube plugging were considered. The results of sensitivity studies on RCS flow indicated minimal effects on the analysis results due to changes in RCS flow. Steam generator tube plugging effects indicated that 0% tube plugging resulted in slightly higher peak RCS pressures. Steam generator tube plugging results in a slightly slower RCP coastdown due to the increased system resistance which allows for improved heat transfer to the secondary system; thereby, producing slightly lower peak primary pressures.

As the limiting case for peak RCS pressure is not affected by RCS flow and steam generator tube plugging, the analysis results are not presented here.

INADVERTENT LOADING OF A FUEL ASSEMBLY INTO THE IMPROPER POSITION, SAR SECTION 15.1.15

Two accidents are considered in this section: 1) the erroneous loading of fuel pellets or fuel rods of different enrichment in a fuel assembly; and, 2) the erroneous placement or orientation of fuel assemblies. Neither of these events consider RCS flow as a parameter; hence, reducing RCS flow will not affect this event.

WASTE GAS DECAY TANK LEAKAGE OR RUPTURE, SAR SECTION 15.1.16

The most limiting waste gas accident is an unexpected and uncontrolled release to the atmosphere of the radioactive xenon and krypton fission gases that are stored in one waste gas decay tank. This event is unaffected by RCS flow.

FAILURE OF AIR EJECTOR LINES (BWR), SAR SECTION 15.1.17

This event is not applicable to ANO-2.

STEAM GENERATOR TUBE RUPTURE WITH OR WITHOUT A CONCURRENT LOSS OF AC POWER (SGTR), SAR SECTION 15.1.18

The steam generator tube rupture accident with or without a loss of AC power is a penetration of the barrier between the RCS and the main steam system. Integrity of this barrier is significant from a radiological standpoint, since a leaking steam generator tube would allow transport of reactor coolant into the main steam system. Radioactivity contained in the reactor coolant would mix with shell side water in the affected steam generator. This radioactivity would be transported through the turbine to the condenser, where the non-condensable radioactive materials would be released to the auxiliary building ventilation system via the condenser vacuum pumps if AC power is available.

Reducing RCS flow by 10% and plugging 30% of the steam generator tubes will result in slightly higher hot leg temperatures for a given cold leg temperature, and lower steam generator pressures. Increased hot leg temperatures will result in a greater flashing fraction for the primary system fluid entering the steam generator. The increased hot leg temperatures will also result in more energy being stored in the RCS. Both of these factors will slightly increase the radioactivity released to the environment during a steam generator tube rupture. Lower steam generator pressures at the start of the event, due to tube plugging and RCS flow reductions, will allow for an increase in the break flow prior to reactor trip. All of these factors have been evaluated with respect the radioactivity released for a SGTR event. The offsite dose could increase by as much as 30%, but the result would remain well within 10 CFR 100 limits.

FAILURE OF CHARCOAL OF CRYOGENIC SYSTEM (BWR), SAR SECTION 15.1.19

This event is not applicable to ANO-2.

CEA EJECTION, SAR SECTION 15.1.20

The CEA Ejection Event at both HFP and HZP conditions were reanalyzed in Cycle 13 accounting for a 10% reduction in the RCS design flow. RCS flow reduction has an adverse effect on the deposited energy during the event. Methods consistent with those identified in Reference 5 were employed in this analysis. The HFP and HZP analyses were performed based on the parameters in Table 28, the Cycle 13 core physics data provided above, and the following input assumptions.

A. A Doppler curve consistent with Figure 2 (BOC) was assumed in both the HFP and HZP analyses.

B. A CEA insertion curve consistent with Figure 3 with a 0.6 holding coil delay time was assumed for the HFP case. For the HZP case, the CEA position versus time of Figure 4 is consistent with the analysis assumption; however, a more conservative normalized reactivity insertion versus CEA position for a +0.6 ASI curve was used.

C. The axial power distribution provided in Table 29 was assumed in both cases.

D. A CPC DNBR trip (based on VOPT) setpoint of 47% and 134% (of 2815 MWt) with a response time of 0.59 seconds was assumed in the HZP and HFP analyses, respectively.

E. A minimum EOC delayed neutron fraction was assumed.

Table 30 lists the acceptable 3D peak F_qs versus ejected CEA worth that was generated based on the above parameters and the following acceptance criteria.

Clad Damage Threshold:
Total Average Enthalpy ≤ 200 cal/gm

Fully Molten Centerline Threshold:
Total Centerline Enthalpy ≤ 310 cal/gm

Cycle specific calculations of the maximum ejected Fq and ejected worth are performed and verified to fall within the limits calculated above.

Based on the above, the maximum total energy deposited during the event is less than the criterion for clad damage and molten centerline temperature. Therefore, results of this analysis are bounded by the prior analyses.

THE SPECTRUM OF ROD DROP ACCIDENTS (BWR), SAR SECTION 15.1.21

This event is not applicable to ANO-2.

BREAK IN INSTRUMENT LINE OR OTHER LINES FROM REACTOR COOLANT PRESSURE BOUNDARY THAT PENETRATE CONTAINMENT, SAR SECTION 15.1.22

There are no instrument lines from the RCS which penetrate the containment.

FUEL HANDLING ACCIDENT, SAR SECTION 15.1.23

This analysis assumes that a fuel assembly is dropped during fuel handling. RCS flow has no effect on this event.

SMALL SPILLS OR LEAKS OF RADIOACTIVE MATERIAL OUTSIDE CONTAINMENT, SAR SECTION 15.1.24

RCS flow is not a consideration for small spills or leaks of radioactive material outside containment.

FUEL CLADDING FAILURE COMBINED WITH STEAM GENERATOR LEAK, SAR SECTION 15.1.25

Releases resulting from operation with leaking steam generator tubes and defective cladding are not affected by RCS flow.

CONTROL ROOM UNINHABITABILITY, SAR SECTION 15.1.26

RCS flow is not consideration in the control room uninhabitability event.

FAILURE OR OVERPRESSSURIZATION OF LOW PRESSURE RESIDUAL HEAT REMOVAL SYSTEM, SAR SECTION 15.1.27

RCS flow is not a consideration for a failure or overpressurization of low pressure residual heat removal system.

LOSS OF CONDENSER VACUUM (LOCV), SAR SECTION 15.1.28

Loss of condenser vacuum is sensed by the turbine emergency trip system and results in a turbine-generator trip. An analysis of the effects and consequences of a turbine-generator trip is provided in Section 15.1.7.

TURBINE TRIP WITH COINCIDENT FAILURE OF TURBINE BYPASS VALVES TO OPEN, SAR SECTION 15.1.29

This event is described and analyzed in Section 15.1.7.

LOSS OF SERVICE WATER SYSTEM, SAR SECTION 15.1.30

RCS flow is not a consideration in a loss of service water system.

LOSS OF ONE DC SYSTEM, SAR SECTION 15.1.31

RCS flow is not a consideration in a loss of one DC system.

INADVERTENT OPERATION OF ECCS DURING POWER OPERATION, SAR SECTION 15.1.32

RCS flow is not a consideration for an inadvertent operation of ECCS during power operation.

TURBINE TRIP WITH FAILURE OF GENERATOR BREAKER TO OPEN, SAR SECTION 15.1.33

RCS flow is not a consideration for a turbine trip with failure of generator breaker to open.

LOSS OF INSTRUMENT AIR SYSTEM, SAR SECTION 15.1.34

RCS flow is not a consideration for a loss of instrument air system.

MALFUNCTION OF TURBINE GLAND SEALING SYSTEM, SAR SECTION 15.1.35

RCS flow is not a consideration for a malfunction of turbine gland sealing system.

TRANSIENTS RESULTING FROM THE INSTANTANEOUS CLOSURE OF A SINGLE MSIV, SAR SECTION 15.1.36

The Cycle 13 evaluation of the Asymmetric Steam Generator Transient (ASGT) event has been performed considering a 10% reduction in RCS flow. Assuming minimum RCS flow is not necessarily bounding for consideration in the ASGT event when determining the required overpower margin (ROPM). However, an ASGT event is typically not limiting with respect to ROPM requirements. The following event was assessed to demonstrate that acceptable results are expected when considering a 10% reduction in RCS flow and the ASGT event is non-limiting with respect to ROPM.

This evaluation has utilized the CENTS computer code described in Reference 2. Input parameters from Table 31 and the Cycle 13 physics data presented above have been incorporated in this analysis with these following clarifications:

A. The BOC Doppler curve in Figure 2 which includes a 0.85 multiplier is conservatively used.

B. The Cycle 13 delayed neutron fraction and neutron lifetime consistent with those defined above were assumed.

C. The Cycle 13 CEA insertion curve in Figure 3 was utilized. This curve accounts for a 0.6 second holding coil delay and a CEA worth of 5%.

D. A CPC asymmetric steam generator trip setpoint of 11°F was assumed. Cold and hot leg RTD response times of 8 seconds and 13 seconds, respectively, were accounted for along with a CPC trip delay time of 0.59 seconds.

E. The Cycle 13 analysis was performed at 90% power and assumed a nominal RCS pressure of 2250 psia.

A summary of the principal results for the ASGT are given in Table 32. The combined effects of the input modifications and the improved models utilized in the CENTS codes have shown that there are no adverse impacts due to the reduced RCS flow and other changes (ASGT remains non-limiting with respect to ROPM requirements). Thus the ASGT trip setpoint incorporated in the CPCs ensures that acceptable DNBR limits will not be exceeded during an ASGT event.

REFERENCES:

1) "A Multi-Dimensional Space-Time Kinetics Code for PWR Transients", CENPD-188-A, July 1976.

2) "Technical Manual for the CENTS Code," CENPD 282-P-A, February 1991.

3) "CETOP-D Code Structures and Modeling Methods for Arkansas Nuclear One - Unit 2, CEN-214(A)-P, July 1982.

4) 2CAN108705, T. G. Campbell of AP&L to J. Calvo of the NRC, "Technical Specification Change Request, Refueling Water Tank and Safety Injection Tank Boron Concentration Increase," dated October 28, 1987.

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Table 1

SYSTEM PARAMETERS AND INITIAL CONDITIONS FOR THE LARGE BREAK LOCA ECCS PERFORMANCE EVALUATION WITH INCREASED TUBE PLUGGING AND REDUCED RCS FLOW RATE

Quantity	Value	Units
Reactor power level (103% of rated power)	2900	MWt
Peak linear heat generation rate (PLHGR) of the hot rod	13.5	kW/ft
PLHGR of the average rod in assembly with hot rod	12.73	kW/ft
Gap conductance at the PLHGR[1]	2136	BTU/hr-ft^2-°F
Fuel centerline temperature at the PLHGR[1]	3204	°F
Fuel average temperature at the PLHGR[1]	1984	°F
Hot rod gas pressure [1]	2647	psia
Moderator temperature coefficient at initial density	$+0.5\times10^{-4}$	$\Delta\rho$/°F
RCS flow rate	107.8×10^{6}	lbm/hr
Core flow rate	104.0×10^{6}	lbm/hr
RCS pressure	2250	psia
Cold leg temperature	556.7	°F
Hot leg temperature	622.7	°F
Safety injection tank pressure	550	psia
Safety injection tank water volume (min/max)	1350/1600	ft^3
Low pressure safety injection pump flow rate (min/max)	3222/5000	gpm
High pressure safety injection pump flow rate (min/max)	678/825	gpm

(1) These quantities correspond to the rod average burnup of the hot rod (40,000 MWD/MTU) that yields the highest peak cladding temperature.

Table 2

**TIMES OF INTEREST
FOR THE LARGE BREAK LOCA ECCS PERFORMANCE EVALUATION
(Seconds after Break)**

Evaluation	SITs On	End of Bypass	Start of Reflood	SITs Empty	Hot Rod Rupture
0.6 DEG/PD Increased tube plugging and reduced RCS flow rate	11.6	17.5	29.3	57.2	23.5

Table 3

**PEAK CLADDING TEMPERATURES AND OXIDATION PERCENTAGES
FOR THE LARGE BREAK LOCA ECCS PERFORMANCE EVALUATION**

Evaluation	Peak Cladding Temperature (°F)	Maximum Cladding Oxidation (%)	Core-Wide Cladding Oxidation (%)
0.6 DEG/PD Increased tube plugging and reduced RCS flow rate	2158	7.2	<0.99

Table 4 (not used)

Table 5

VARIABLES PLOTTED AS A FUNCTION OF TIME FOR THE LIMITING BREAK OF THE LARGE BREAK LOCA ECCS PERFORMANCE EVALUATION

Variable	Figure
Core Power	5a
Pressure in Center Hot Assembly Node	5b
Leak Flow Rate	5c
Hot Assembly Flow Rate (Below Hot Spot)	5d.1
Hot Assembly Flow Rate (Above Hot Spot)	5d.2
Hot Assembly Quality	5e
Containment Pressure	5f
Mass Added to Core During Reflood	5g
Peak Cladding Temperature and Temperature of the Rupture Node	5h
Mid Annulus Flow Rate`	5i
Quality Above and Below the Core	5j
Core Pressure Drop	5k
Safety Injection Flow Rate into Intact Discharge Legs	5l
Water Level in Downcomer During Reflood	5m
Hot Spot Gap Conductance	5n
Local Cladding Oxidation Percentage	5o
Fuel Centerline, Fuel Average, Cladding and Coolant Temperature at the Hot Spot	5p
Hot Spot Heat Transfer Coefficient	5q
Hot Pin Pressure	5r

Table 6

**HIGH PRESSURE SAFETY INJECTION PUMP
MINIMUM DELIVERED FLOW TO RCS
(ASSUMING ONE EMERGENCY GENERATOR FAILED)**

RCS Pressure, psia	Flow Rate, gpm
1348	0.0
1321	82.6
1284	138.6
1248	186.5
1142	264.4
1071	314.1
990	361.5
899	407.6
800	458.5
692	507.7
577	554.7
456	602.6
327	651.6
191	702.3
46	750.6
31	755.1
22	757.8
14.7	760.0

Notes:

1. The flow is assumed to be split equally to each of the four discharge legs.
2. The flow to the broken discharge leg is assumed to spill out the break.

Table 7

SYSTEM PARAMETERS AND INITIAL CONDITIONS FOR THE SMALL BREAK LOCA ECCS PERFORMANCE EVALUATION

Quantity	Value	Units
Reactor power level (103% of rated power)	2900	MWt
Peak linear heat generation rate (PLHGR)	13.5	kW/ft
Axial shape index	-0.3	asiu
Gap conductance at PLHGR	1582	BTU/hr-ft^2-°F
Fuel centerline temperature at PLHGR	3334	°F
Fuel average temperature at PLHGR	2115	°F
Hot rod gas pressure	1123	psia
Moderator temperature coefficient at initial density	0.0×10^{-4}	$\Delta\rho$/°F
RCS flow rate	108.4×10^6	lbm/hr
Core flow rate	104.6×10^6	lbm/hr
RCS pressure	2250	psia
Cold leg temperature	556.7	°F
Hot leg temperature	622.7	°F
Plugged tubes per steam generator	30	%
MSSV first bank opening pressure	1103.5	psia
Low pressurizer pressure reactor trip setpoint	1625*	psia
Low pressurizer pressure SIAS setpoint	1578*	psia
Safety injection tank pressure	550	psia

* Various values were assumed for these setpoints as noted in the text. These values are the bounding assumptions.

Table 8

**BREAK SPECTRUM
FOR THE SMALL BREAK LOCA ECCS PERFORMANCE EVALUATION**

Break Size and Location	Abbreviation	Figure No.
0.06 ft^2 Break in Pump Discharge Leg	0.06 ft^2/PD	6
0.05 ft^2 Break in Pump Discharge Leg	0.05 ft^2/PD	7
0.04 ft^2 Break in Pump Discharge Leg	0.04 ft^2/PD	8
0.02 ft^2 Break in Pump Discharge Leg	0.02 ft^2/PD	9

Table 9

**PEAK CLADDING TEMPERATURES AND OXIDATION PERCENTAGES
FOR THE SMALL BREAK LOCA ECCS PERFORMANCE EVALUATION**

Break	Peak Cladding Temperature (°F)[a]	Maximum Cladding Oxidation (%)[b]	Hot Rod Oxidation (%)[c]
0.06 ft^2/PD	2003	4.78	< 0.726
0.05 ft^2/PD	2011	5.47	< 0.835
0.04 ft^2/PD	1870	3.37	< 0.567
0.02 ft^2/PD	1671	1.73	< 0.318

(a) Acceptance criterion is ≤ 2200°F.
(b) Acceptance criterion is ≤ 17%
(c) Acceptance criterion is ≤ 1.0% core-wide cladding oxidation. Rod-average oxidation of the hot rod is given as a conservative representation of the core-wide cladding oxidation.

Table 10

**TIMES OF INTEREST
FOR THE SMALL BREAK LOCA ECCS PERFORMANCE EVALUATION
(Seconds after Break)**

Break	HPSI Flow Delivered to RCS (sec)	LPSI Flow Delivered to RCS (sec)	SIT Flow Delivered to RCS (sec)	Peak Cladding Temperature Occurs (sec)
0.06 ft^2/PD	169	(a)	1290[b]	1541
0.05 ft^2/PD	192	(a)	1592[b]	1624
0.04 ft^2/PD	179	(a)	(c)	1943
0.02 ft^2/PD	389	(a)	(c)	3411

(a) Calculation completed before LPSI flow delivery to RCS begins.
(b) SIT injection calculated to begin but not credited in analysis.
(c) Calculation completed before SIT injection begins.

Table 11

**VARIABLES PLOTTED AS A FUNCTION OF TIME FOR EACH BREAK OF
THE SMALL BREAK LOCA ECCS PERFORMANCE EVALUATION**

Variable	Figure 6 Through 9 Designation
Normalized Total Core Power	a
Inner Vessel Pressure	b
Break Flow Rate	c
Inner Vessel Inlet Flow Rate	d
Inner Vessel Two-Phase Mixture Level	e
Heat Transfer Coefficient at Hot Spot	f
Coolant Temperature at Hot Spot	g
Cladding Temperature at Hot Spot	h

Table 12

ASSUMPTIONS FOR THE UNCONTROLLED CEA WITHDRAWAL FROM A SUBCRITICAL CONDITION

Parameter	Units	Assumptions Case 1	Assumptions Case 2
Initial Core Power	(MWt)	896×10^{-9}	896×10^{-9}
RCP Heat	(MWt)	18	18
Core Inlet Temperature	(°F)	552	552
Reactor Coolant System Pressure	(psia)	2000	2000
Steam Generator Pressure	(psia)	1055	1055
Reactor Coolant System Flow	(lbm/hr)	108.36×10^{6}	108.36×10^{6}
Total Nuclear Heat Flux Factor	-	6.8	9
Moderator Temperature Coefficient	(10^{-4} $\Delta\rho$/°F)	+0.5	+0.5
Doppler Multiplier	-	0.85	0.85
CEA Maximum Reactivity Addition Rate	(10^{-4} $\Delta\rho$/sec)	2.5	2.0
Steam Bypass System	-	Manual	Manual
Feedwater Regulating System	-	Manual	Manual

Table 13

SEQUENCE OF EVENTS FOR THE UNCONTROLLED CEA WITHDRAWAL FROM SUBCRITICAL CONDITIONS CASE 1

Time (sec)	Event	Setpoint or Value
0.0	Initiation of withdrawal	-
256.6	High Logarithmic power level trip condition	4% of full power
257.0	Trip breakers open, and Rod withdrawal stops	-
257.4	Maximum Power occurs	97.4% of full power
257.6	CEAs begin to drop	-
257.7	Maximum heat flux, and Minimum DNBR	34.5% of full power 1.27
261.2	Maximum RCS Pressure	2119.6 psia
300	End of transient	-

Table 14

SEQUENCE OF EVENTS FOR THE UNCONTROLLED CEA WITHDRAWAL FROM SUBCRITICAL CONDITIONS CASE 2

Time (sec)	Event	Setpoint or Value
0.0	Initiation of withdrawal	-
320.2	High Logarithmic power level trip condition	4% of full power
320.6	Trip breakers open, and Rod withdrawal stops	-
321.2	CEAs begin to drop, and Maximum Power occurs	77.7% of full power
321.3	Maximum heat flux, and Minimum DNBR	24.84% of full power 1.42
324.7	Maximum RCS Pressure	2099.4 psia
350	End of transient	-

Table 15

**ASSUMPTIONS FOR THE
UNCONTROLLED CEA WITHDRAWAL
FROM HOT ZERO POWER**

Parameter	Units	Assumptions
Initial Core Power	(MWt)	0.002815
RCP Heat	(MWt)	18
Core Inlet Temperature	(°F)	552
Reactor Coolant System Pressure	(psia)	2000
Steam Generator Pressure	(psia)	1055
Reactor Coolant System Flow	(lbm/hr)	108.36×10^6
Total Nuclear Heat Flux Factor	-	7.5
Moderator Temperature Coefficient	(10^{-4} $\Delta\rho$/°F)	+0.5
Doppler Multiplier	-	0.85
CEA Worth on Trip	(% $\Delta\rho$)	-2
CEA Maximum Reactivity Addition Rate	(10^{-4} $\Delta\rho$/sec)	1.8
Steam Bypass System	-	Manual
Feedwater Regulating System	-	Manual
Automatic Withdrawal Prohibit	-	Inoperative

Table 16

SEQUENCE OF EVENTS FOR THE UNCONTROLLED CEA WITHDRAWAL FROM HOT ZERO POWER

Time (sec)	Event	Setpoint or Value
0.0	Initiation of withdrawal	-
22.2	VOPT trip conditions occurs	41% of full power
22.8	Trip breakers open, and Rod withdrawal stops	-
23.1	Maximum power occurs	71.3% of full power
23.4	CEAs begin to drop	-
23.5	Maximum heat flux, and Minimum DNBR	38% of full power (see values below)
27.2	Maximum RCS Pressure	2174.2 psia

Minimum DNBR Results for Various Power Shapes

ASI	Fr	DNBR
0	3.95	1.31
-0.3	3.52	1.33
-0.6	3.26	1.34
-0.75	2.97	1.34
-0.9	2.90	1.33

Table 17

ASSUMPTIONS FOR THE LOSS OF COOLANT FLOW ANALYSIS ASSUMING 30% STEAM GENERATOR TUBE PLUGGING

Parameter	Units	Conservative Assumptions
Initial Core power Level	(MWt)	2900
Core Inlet Coolant Temperature	(°F)	556.7
Core Mass Flow Rate	(10^6 lbm/hr)	104.57
RCS Pressure	(psia)	2200
Radial Peaking Factor, Fr	-----	1.71
Axial Shape Index	-----	0.3
Moderator Temperature Coefficient	(10^{-4} $\Delta\rho$/°F)	0.0
Scram Worth	(% $\Delta\rho$)	-5.0

Table 18

**SEQUENCE OF EVENTS
FOR THE 4-PUMP LOSS OF COOLANT FLOW ANALYSIS
ASSUMING 30% STEAM GENERATOR TUBE PLUGGING**

Time (sec)	Event	Setpoint or Value
0.0	Loss of power to all four reactor coolant pumps	-------
0.8	CPC Low RCP Speed Trip (95%)	95% nominal speed
1.1	Trip breakers open	-------
1.7	Shutdown CEAs begin to drop into core	-------
2.8	Minimum CE-1 DNBR	1.29

Table 19

**ASSUMPTIONS FOR THE
LOSS OF EXTERNAL LOAD/LOSS OF CONDENSER VACUUM**

Parameter	Units	Conservative Assumptions
Initial Core Power Level	(MWt)	2900
RCP Heat	(MWt)	18
Core Inlet Coolant Temperature	(°F)	540
Reactor Coolant System Flow	(10^6 lbm/hr)	135.3
Reactor Coolant System Pressure	(psia)	2000
Steam Generator Pressure	(psia)	795
Moderator Temperature Coefficient	(10^{-4} $\Delta\rho$/°F)	0
Doppler Multiplier	-	0.85
CEA Worth on Trip	(% $\Delta\rho$)	-5.0
Steam Generator tube Plugging	%	0
Tolerance on MSSV Setpoint	%	3
Tolerance on PSV Setpoint	%	3
Steam Bypass System	-	Inoperative
Feedwater Regulating System	-	Manual

Table 20

SEQUENCE OF EVENTS FOR THE LOSS OF EXTERNAL LOAD/LOSS OF CONDENSER VACUUM

Time (sec)	Event	Setpoint or Value
0.0	Loss of Condenser Vacuum, Turbine Stop Valves Close, and Main Feedwater Valves Close	-
8.1	High Pressurizer Pressure Trip Condition Occurs	2422 psia
9.0	Trip Breakers Open	-
9.6	CEAs Begin to Drop	-
9.9	Pressurizer Safety Valves Open	2575 psia
10.5	Maximum RCS Pressure Occurs	2683 psia
11.4	Main Steam Safety Valves Open	1125.5 psia
13.6	Peak Secondary Pressure Occurs	1162 psia
13.9	Pressurizer Safety Valves Close	2472 psia

Table 21

**ASSUMPTIONS FOR THE
CYCLE 13 LOSS OF NORMAL FEEDWATER FLOW**

Parameter	Units	Conservative Assumptions
Initial Core Power Level	(MWt)	2900
RCP Heat	(MWt)	18
Core Inlet Coolant Temperature	(°F)	556.7
Reactor Coolant System Flow	(10^6 lbm/hr)	108.4
Reactor Coolant System Pressure	(psia)	2000
Steam Generator Pressure	(psia)	922
Moderator Temperature Coefficient	(10^{-4} $\Delta\rho$/°F)	-3.5
Doppler Multiplier	-	1.4
CEA Worth On Trip	(% $\Delta\rho$)	-5.0
Steam Bypass System	-	Automatic
Feedwater Regulating System	-	Malfunction

Table 22

PRINCIPAL RESULTS FOR THE LOSS OF NORMAL FEEDWATER FLOW

Time (sec)	Event	Setpoint or Value
0.0	Loss of Feedwater Flow	-
18.5	Steam Dump and Bypass Begins to Open	Variable
47.2	Low Steam Generator Water Level Trip Condition	5%
48.5	Trip Breakers Open	-
49.1	CEAs Begin to Drop	-
51.9	Peak RCS Pressure Occurs	2229 psia
53.0	MSSVs Open	1059.9 psia
57.1	Peak Steam Generator Pressure Occurs	1084.5 psia
68.5	MSSVs Close	1006.9 psia
144.6	EFW Begins to Inject	-
203	Minimum Liquid Inventory in Steam Generator A	-
260.5	Minimum Liquid Inventory in Steam Generator B	-

Table 23

ASSUMPTIONS FOR THE STEAM LINE BREAK ANALYSIS FROM HOT FULL POWER AND HOT ZERO POWER

		Assumptions	
Parameter	**Units**	**Hot Full Power**	**Hot Zero Power**
Initial Incore Power Level	MWt	2900	1
RCP Heat	MWt	10	10
Initial Core Inlet Temperature	°F	556.7	552
Initial Reactor Coolant Flow	10^6 lbm/hr	108.36	108.36
Initial RCS Pressure	psia	2300	2300
CEA Worth at Trip	% $\Delta\rho$	-7.5144	-5.0
Initial Steam Generator Pressure	psia	922	1058
Doppler Coefficient		1.22	1.22
Moderator Temperature Coefficient	10^{-4} $\Delta\rho$/°F	-3.4	-3.4
Feedwater Control System	----	Automatic	Manual

Table 24

SEQUENCE OF EVENTS FOR THE STEAM LINE BREAK HOT FULL POWER WITH LOSS OF AC

Time Seconds	Event	Setpoint or Value
0	Steam line break occurs, Loss of AC power occurs, RCPs begin coasting down	----
0.31	CPC Low pump speed trip signal, fraction	0.965
1.31	Trip breakers open	----
1.91	CEAs begin to drop	----
2	MSIS setpoint has been reached, psia	620
3.3	MSIV begin to close	----
3.4	MFIV begin to close	----
6.3	Complete Closure of the MSIV	----
7.2	SG delta pressure isolation reached, psid	220
12.5	Intact SG level reaches EFW actuation setpoint, % of narrow range	35.0
21	Pressurizer empties	----
24.9	SIAS setpoint is reached, psia	1400
37.6	EFW enters intact SG (steam pump)	----
38.4	Complete closure of the MFIV	----
64.9	SIAS pumps reach full speed and begin injecting	----
100.9	EFW to intact SG is increased (electric pump)	----
106.6	Boron reaches RCS	----
204	Maximum post-trip fission power, % of 2815 MWt	2.61
210	Minimum DNBR	1.81
302	Maximum post trip reactivity, %$\Delta\rho$	-0.029
325	Ruptured steam generator empties, lbm	<2510
390	Cooldown ends, Minimum inlet temperature, °F	387.1
500	End of calculation	----
1800	Operator initiates cooldown (not simulated)	----

Table 25

SEQUENCE OF EVENTS FOR THE STEAM LINE BREAK HOT FULL POWER WITH AC AVAILABLE

Time Seconds	Event	Setpoint or Value
0	Steam line break occurs	---
2.07	SG low pressure trip condition and MSIS setpoint has been reached, psia	620
3.34	MSIVs begin to close	----
3.37	Trip breakers open	----
3.47	MFIV begin to close	----
3.97	CEAs begin to drop	----
6.34	Complete Closure of the MSIVs	----
7.1	SG delta pressure isolation reached, psid	220
13.7	Intact SG level reaches EFW actuation setpoint, % of narrow range	35.0
17.2	Pressurizer empties	----
18.67	SIAS setpoint is reached, psia	1400
23.47	Complete closure of the MFIV	----
38.8	EFW enters intact SG (steam pump)	
58.7	SIAS pumps reach full speed and begin injecting	----
80	Maximum post-trip fission power, % of 2815 MWt	4.98
80	Minimum DNBR	2.46
83	Maximum post trip reactivity, $\%\Delta\rho$	-0.338
84	Cooldown ends, Minimum inlet temperature, °F	405.1
87.4	Boron reaches RCS	
96.5	EFW to intact SG is increased (electric pump)	----
100.6	Ruptured steam generator empties, lbm	<2510
350	End of calculation	----
1800	Operator initiates cooldown (not simulated)	----

Table 26

SEQUENCE OF EVENTS FOR THE STEAM LINE BREAK HOT ZERO POWER WITH LOSS OF AC

Time Seconds	Event	Setpoint or Value
0	Steam line break occurs Loss of AC power occurs RCPs begin coasting down	----
0.32	CPC Low flow trip signal, Fraction of pump speed	0.965
1.32	Trip breakers open	----
1.92	CEAs begin to drop	----
3.2	MSIS initiation setpoint has been reached, psia	620
4.47	MSIVs begin to close	----
7.47	Complete Closure of the MSIV	----
8.8	SG delta pressure isolation reached, psid	220
27.3	Pressurizer empties	----
28.4	SIAS setpoint is reached, psia	1400
54.5	Emergency Feed valves close	----
68.4	SIAS pumps reach full speed and begin injecting	----
106.7	Boron enters RCS	----
159	Maximum post trip reactivity (first peak), %$\Delta\rho$	.252
253	Maximum post trip reactivity (second peak), %$\Delta\rho$	.126
334	Maximum post-trip fission power, % of 2815 MWt	.41
343	Minimum DNBR	12.3
555	Ruptured steam generator empties, lbm	<2520
610	Cooldown ends, Minimum inlet temperature, °F	269.4
650	End of calculation	----
1800	Operator initiates cooldown (not simulated)	----

Table 27

SEQUENCE OF EVENTS FOR THE STEAM LINE BREAK HOT ZERO POWER WITH AC AVAILABLE

Time Seconds	Event	Setpoint or Value
0	Steam line break occurs	---
3.22	SG low pressure trip condition and MSIS initiation setpoint has been reached, psia	620
4.49	MSIVs begin to close	----
4.49	Trip breakers open	----
5.09	CEAs begin to drop	----
7.49	Complete Closure of the MSIV	----
8.8	SG Delta pressure isolation reached, psid	220
20.3	Pressurizer empties	----
20.94	SIAS setpoint is reached, psia	1400
39.52	Emergency Feed Valves close	----
61.0	SIAS pumps reach full speed and begin injecting	----
87.3	Boron enters RCS	----
122	Maximum post trip reactivity, $\%\Delta\rho$	.227
145	Maximum post-trip fission power, % of 2815 MWt	1.275
145	Minimum DNBR	11.2
146	Ruptured steam generator empties, lbm	<2500
146	Cooldown ends, Minimum inlet temperature, °F	348.6
250	End of calculation	----
1800	Operator initiates cooldown (not simulated)	----

Table 28

**ASSUMPTIONS FOR THE
CEA EJECTION ACCIDENT ANALYSIS**

Parameter	Units	HZP	HFP
Initial Core Power	(MWt)	29	2900
Core Inlet Temperature	(°F)	552	556.7
Reactor Coolant System Pressure	(psia)	2000	2000
Reactor Coolant System Flow	(10^6 lbm/hr)	108.36	108.36
Total Delayed Neutron Fraction (β)	-	0.0043414	0.0043414
Moderator Temperature Coefficient	(10^{-4} Δρ/°F)	+0.5	0.0
CEA Ejection Time	(sec)	0.05	0.05
Doppler Multiplier	-	0.85	0.85
CEA Worth at Trip	% Δρ	-2	-5

Table 29

AXIAL POWER DISTRIBUTION USED FOR THE CEA EJECTION ACCIDENT ANALYSES

Fractional Distance from the Bottom of the Reactor Core	Power Fraction, Fz
0.025	0.5
0.075	0.8
0.125	1.0
0.175	1.1
0.225	1.1
0.275	1.1
0.325	1.1
0.375	1.1
0.425	1.1
0.475	1.1
0.525	1.1
0.575	1.1
0.625	1.1
0.675	1.1
0.725	1.1
0.775	1.1
0.825	1.1
0.875	1.0
0.925	0.8
0.975	0.5

Table 30

RESULTS FOR THE CEA EJECTION ACCIDENT ANALYSIS

Initial Power, % of 2815 MWt	Ejected CEA Worth (10^{-2} $\Delta\rho$)	Acceptable Ejected 3D Peak, F_q
100	0.30	4.98
	0.20	5.94
	0.17	6.27
0	0.85	14.7
	0.70	15.6

Table 31

**ASSUMPTIONS FOR THE
LOSS OF LOAD TO ONE STEAM GENERATOR**

Parameter	Units	Conservative Assumptions
Initial Core Power Level	(MWt)	2534
Core Inlet Coolant Temperature	(°F)	556.7
Reactor Coolant System Flow	(10^6 lb_m/hr)	108.36
Reactor Coolant System Pressure	(psia)	2250
Moderator Temperature Coefficient	(10^{-4} $\Delta\rho$/°F)	-3.5
Doppler Multiplier	-	0.85
CEA Worth on Trip	(% $\Delta\rho$)	-5.0
Steam Generator tube Plugging	%	30
Tolerance on MSSV lift Setpoint	%	3
Axial Shape Index	asiu	-0.3

Table 32

**SEQUENCE OF EVENTS FOR THE
LOSS OF LOAD TO ONE STEAM GENERATOR**

Time (sec)	Event	Setpoint or Value
0.0	Spurious closure of a single MSIV	-
5.72	ASGT trip setpoint reached	11°F
6.0	Main steam safety valves open on affected steam generator	1125.5 psia
6.31	Trip breakers open	-
6.91	CEAs begin to drop into core	-
7.90	Time of minimum DNBR	≥ 1.25
9.8	Maximum steam generator pressure	1160 psia

Figure 1

Moderator Temperature Coefficient

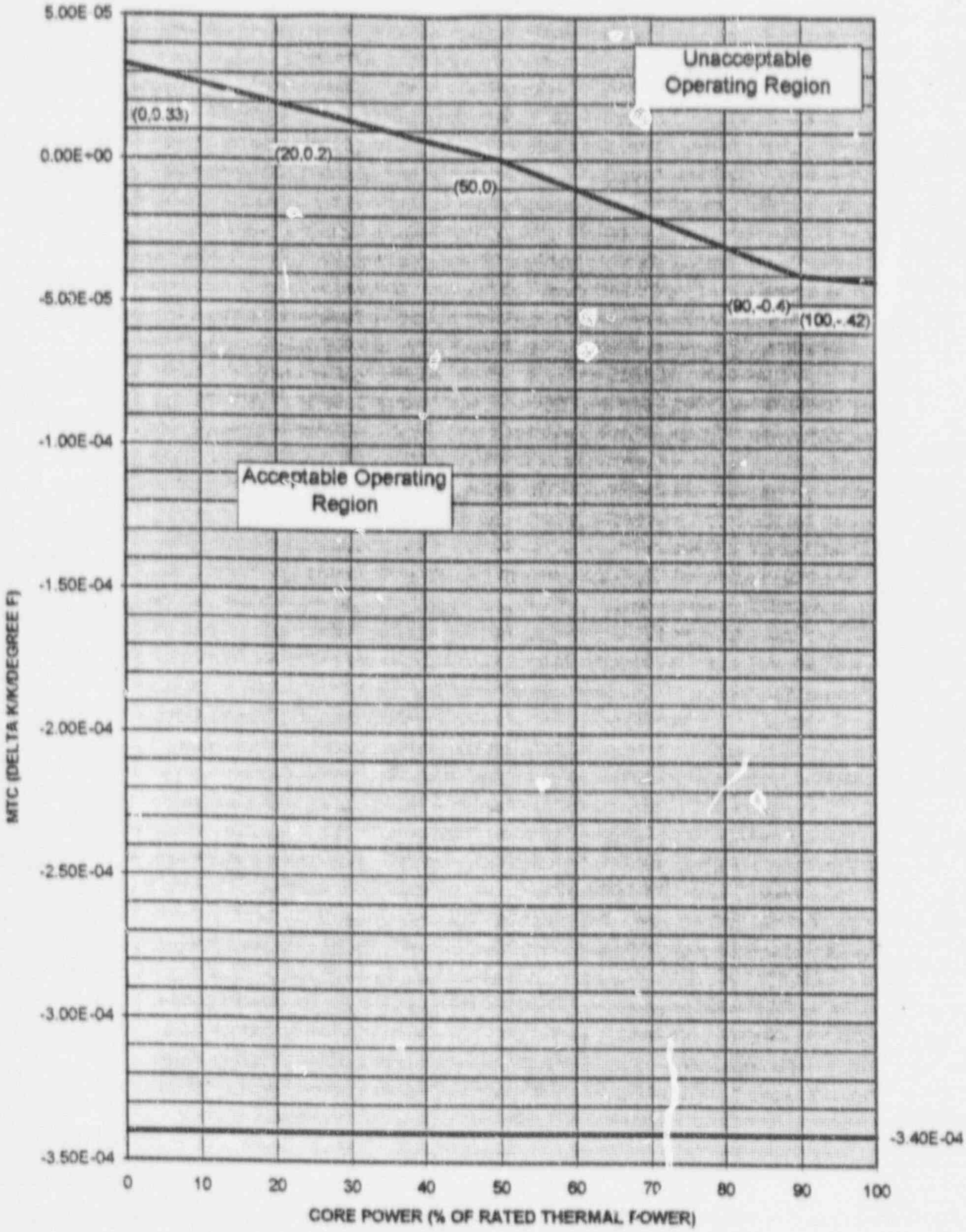


Figure 2

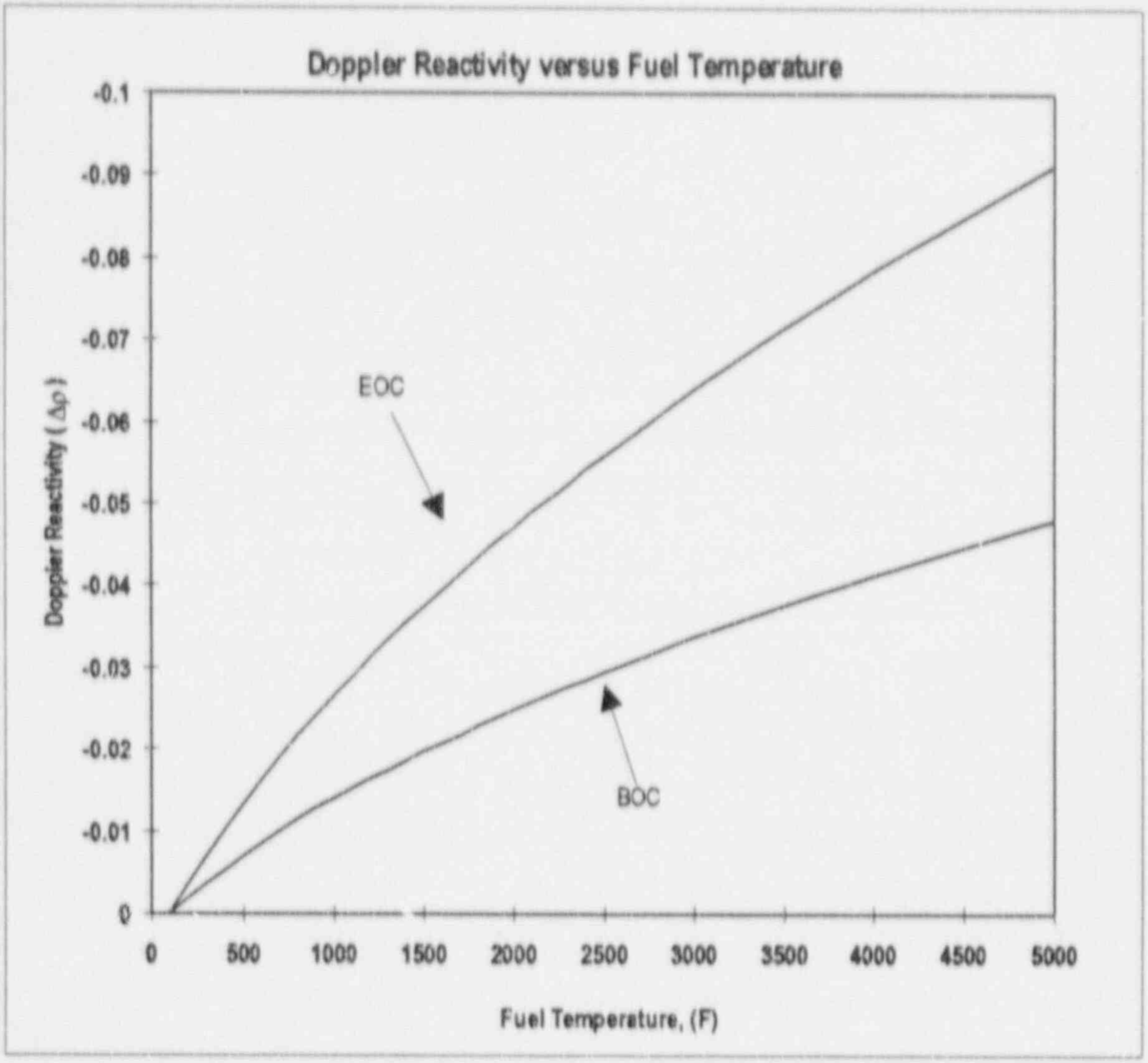

Doppler Reactivity versus Fuel Temperature
Doppler Reactivity (Δρ)
Fuel Temperature, (F)
EOC
BOC
-0.1
-0.09
-0.08
-0.07
-0.06
-0.05
-0.04
-0.03
-0.02
-0.01
0
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000

Figure 3

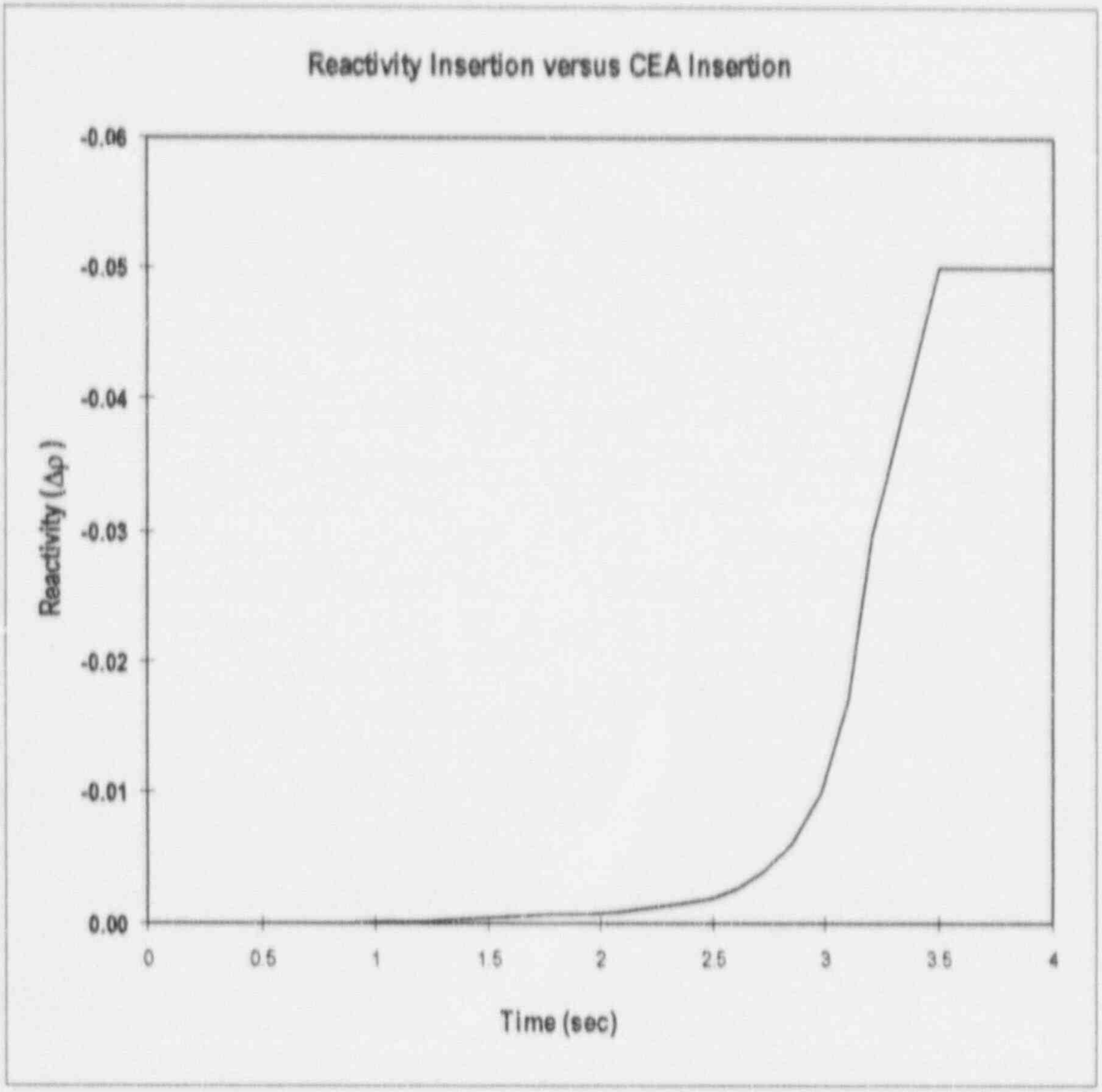

Reactivity Insertion versus CEA Insertion
-0.06
-0.05
-0.04
-0.03
-0.02
-0.01
0.00
Reactivity (Δρ)
0
0.5
1
1.5
2
2.5
3
3.5
4
Time (sec)

Figure 4

CEA Insertion vs. Time

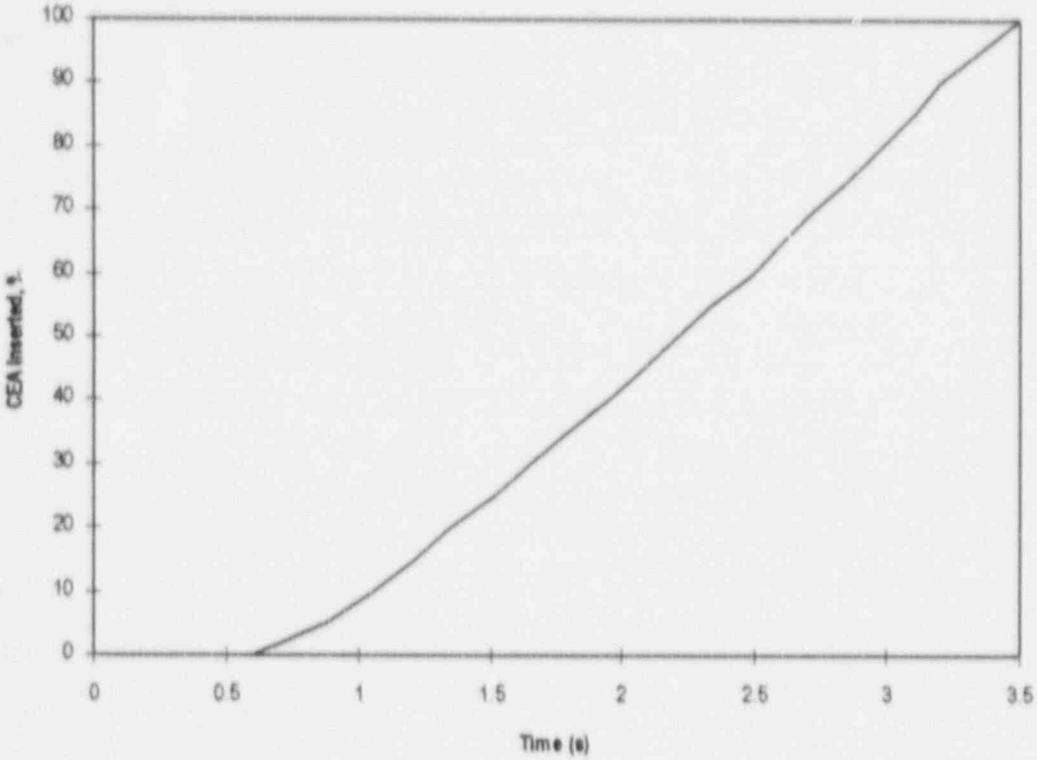


Figure 5a

0.6 Double Ended Guillotine Break in Pump Discharge Leg

Core Power vs. Time

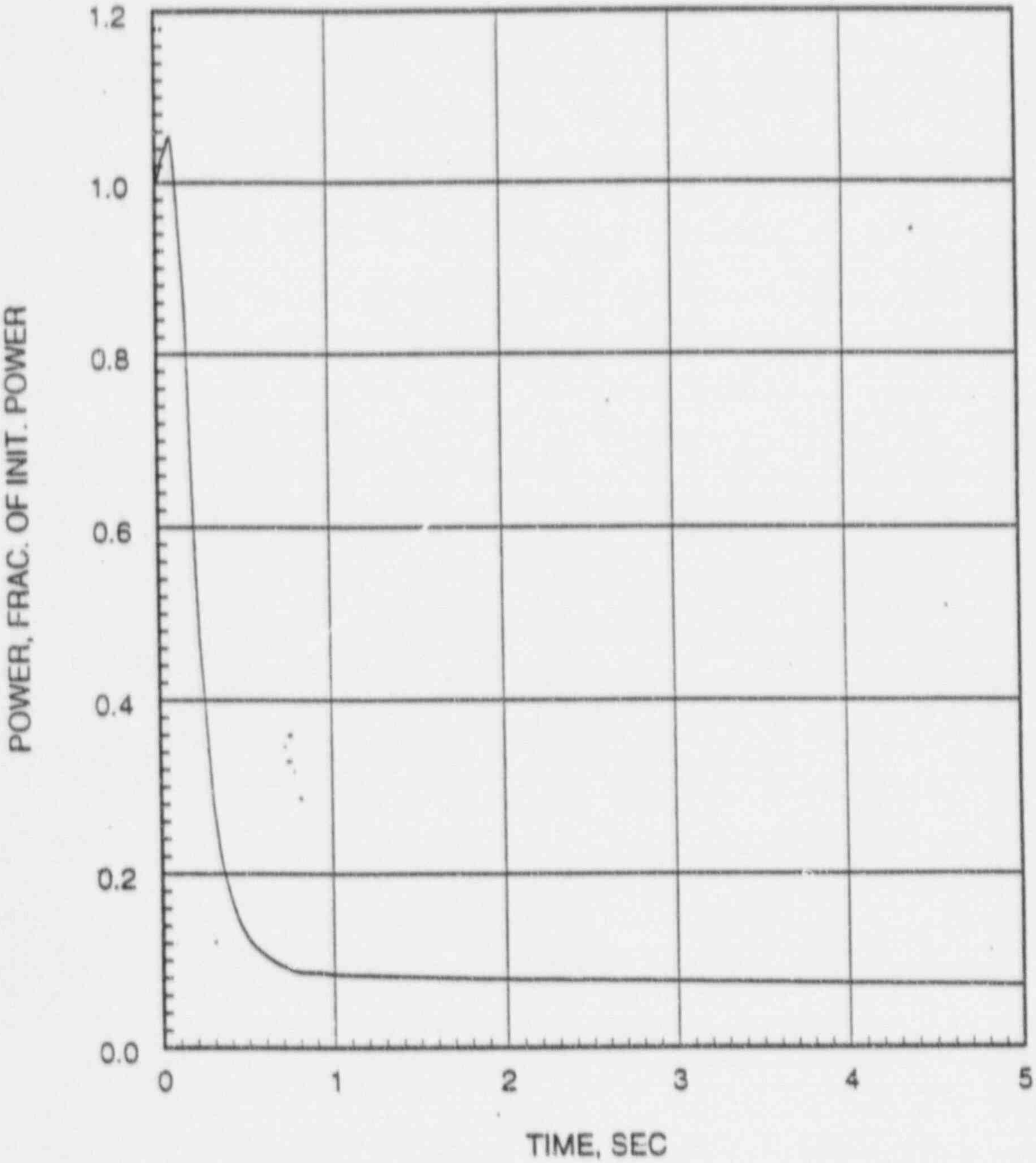


Figure 5b

0.6 Double Ended Guillotine Break in Pump Discharge Leg

Pressure in Center Hot Assembly Node vs. Time

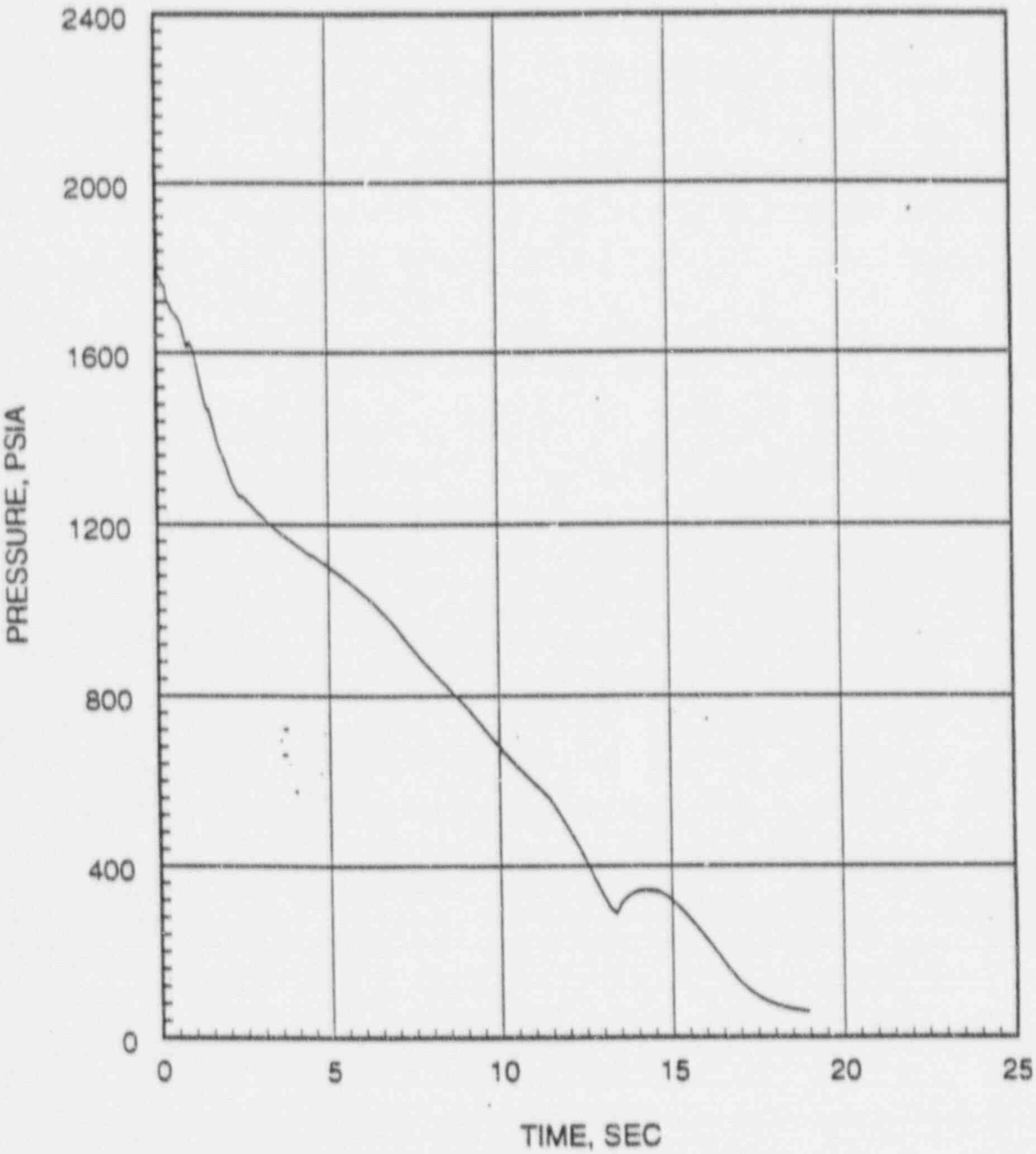


Figure 5c

0.6 Double Ended Guillotine Break in Pump Discharge Leg

Leak vs. Time

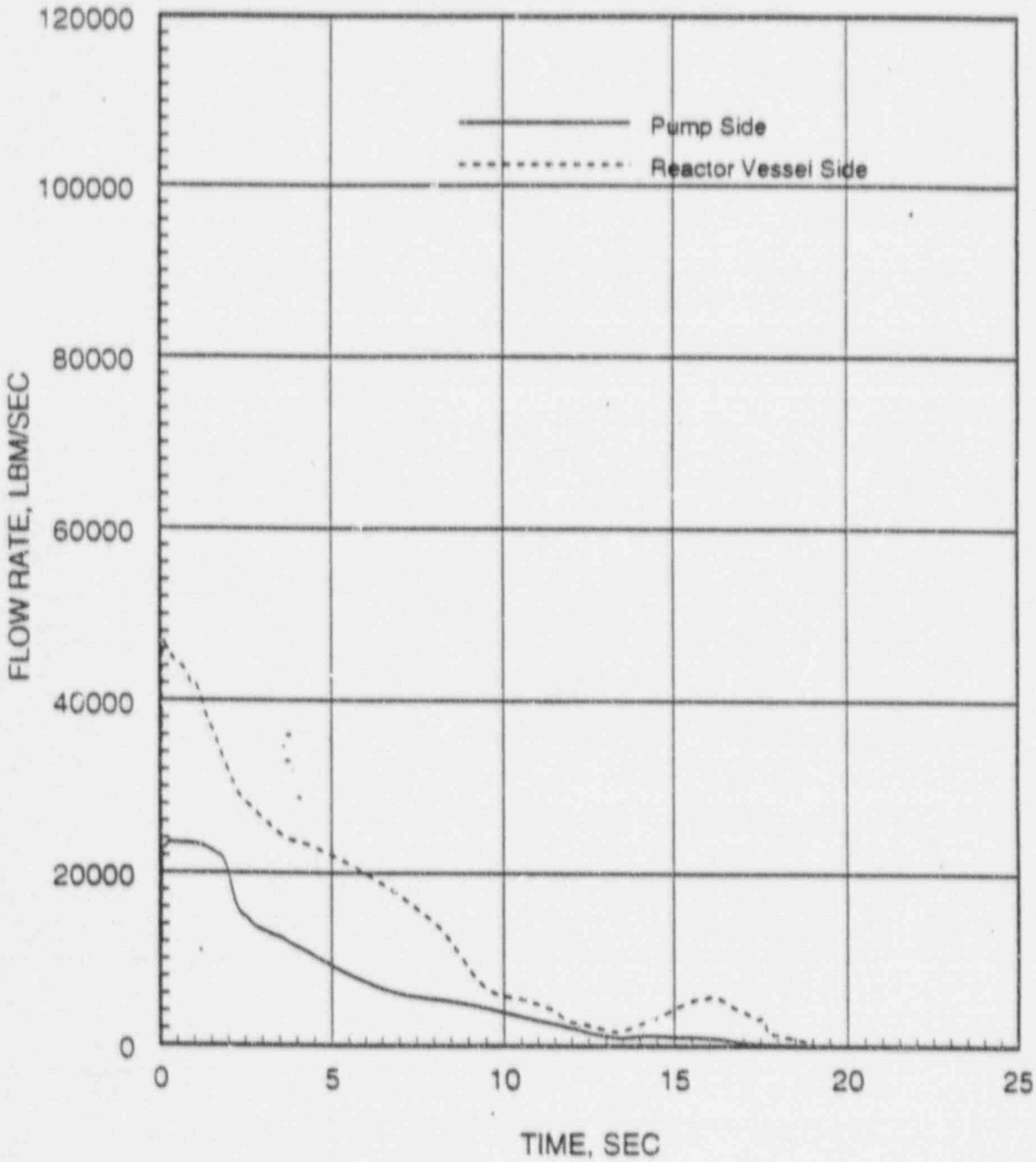


Figure 5d.1

0.6 Double Ended Guillotine Break in Pump Discharge Leg

Hot assembly Flow Rate (Below Hot Spot) vs. Time

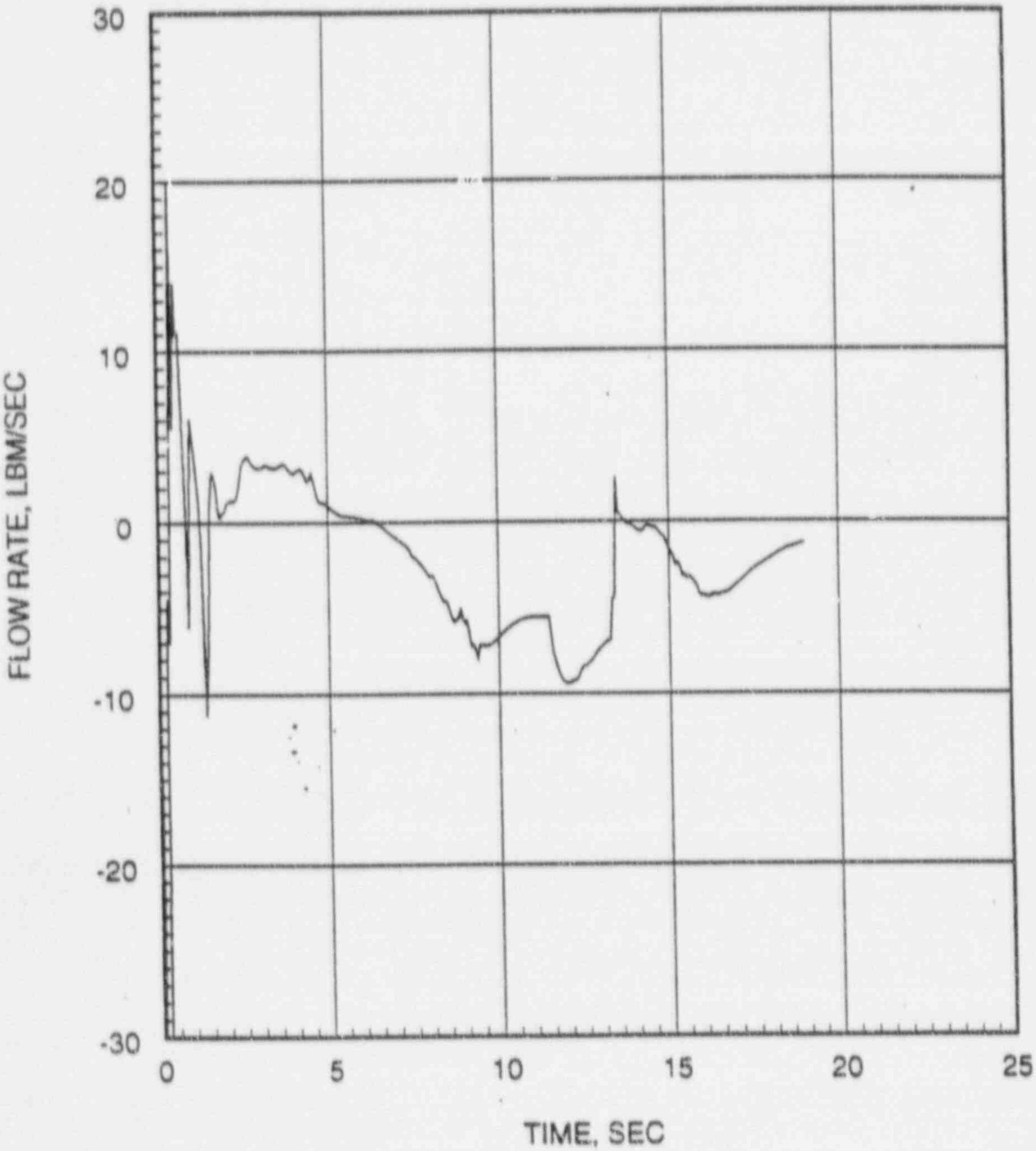


Figure 5d.2

0.6 Double Ended Guillotine Break in Pump Discharge Leg

Hot assembly Flow Rate (Above Hot Spot) vs. Time

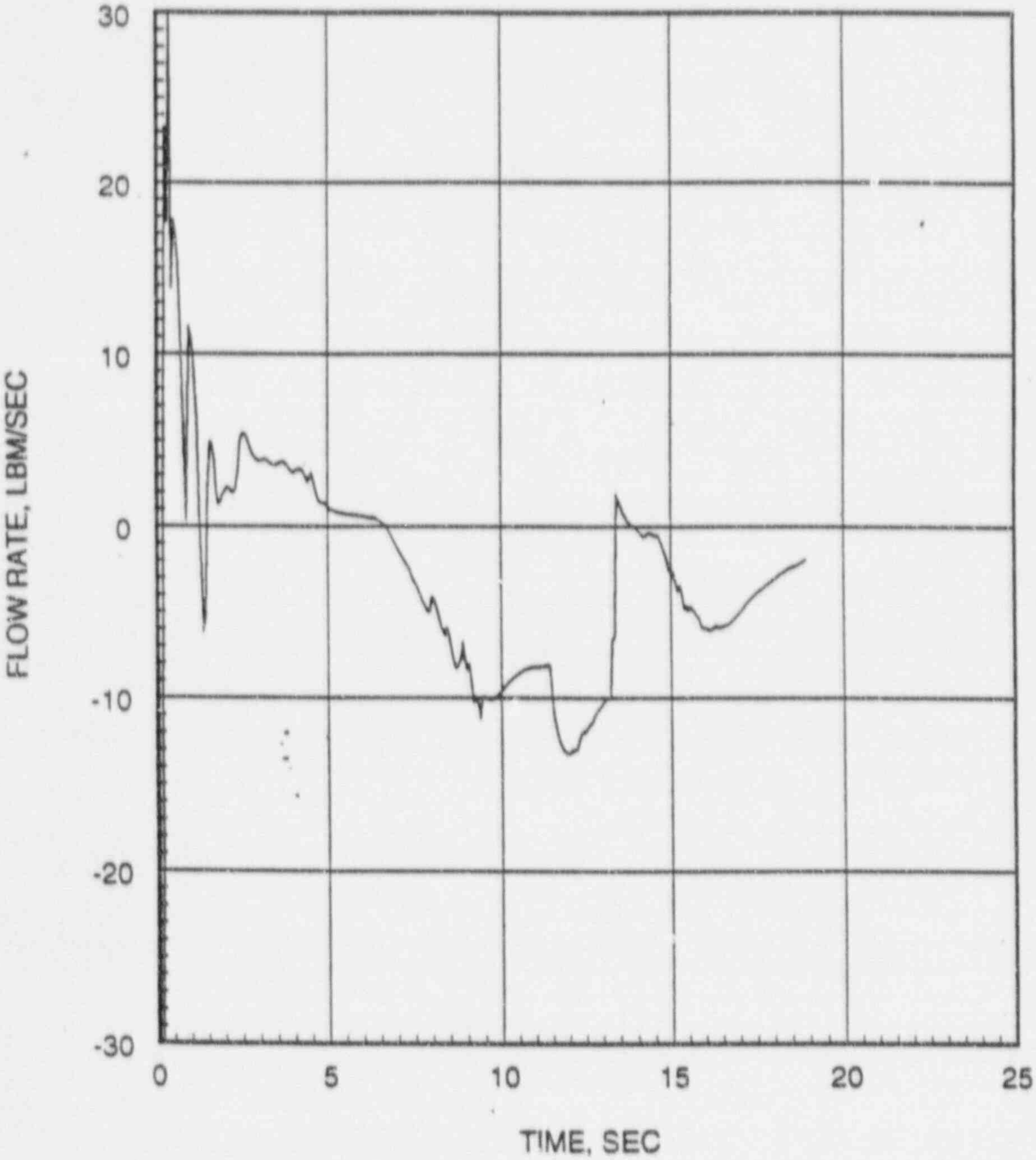


Figure 5e

0.6 Double Ended Guillotine Break in Pump Discharge Leg

Hot assembly Quality vs. Time

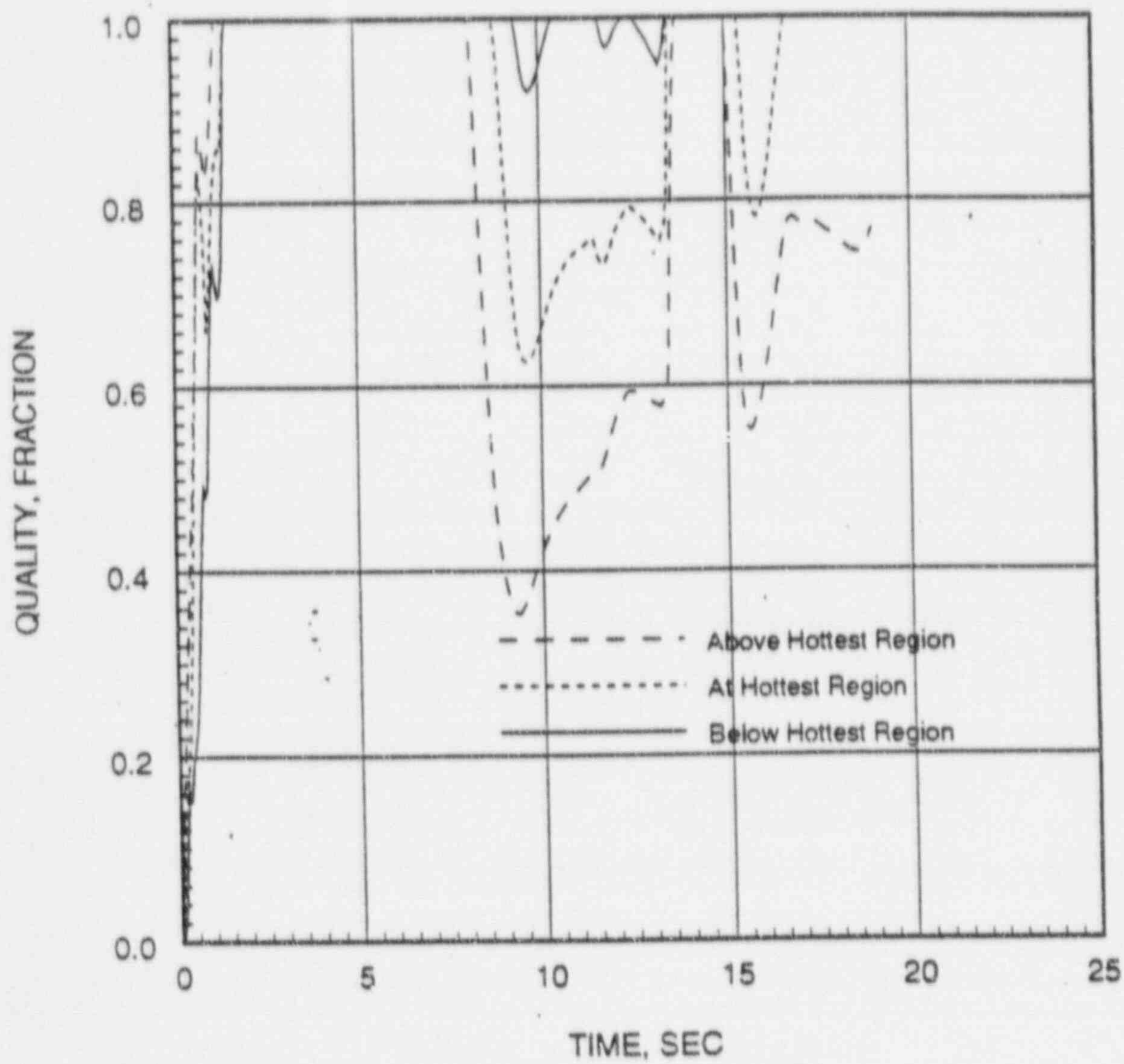


Figure 5f

0.6 Double Ended Guillotine Break in Pump Discharge Leg

Containment Pressure vs. Time

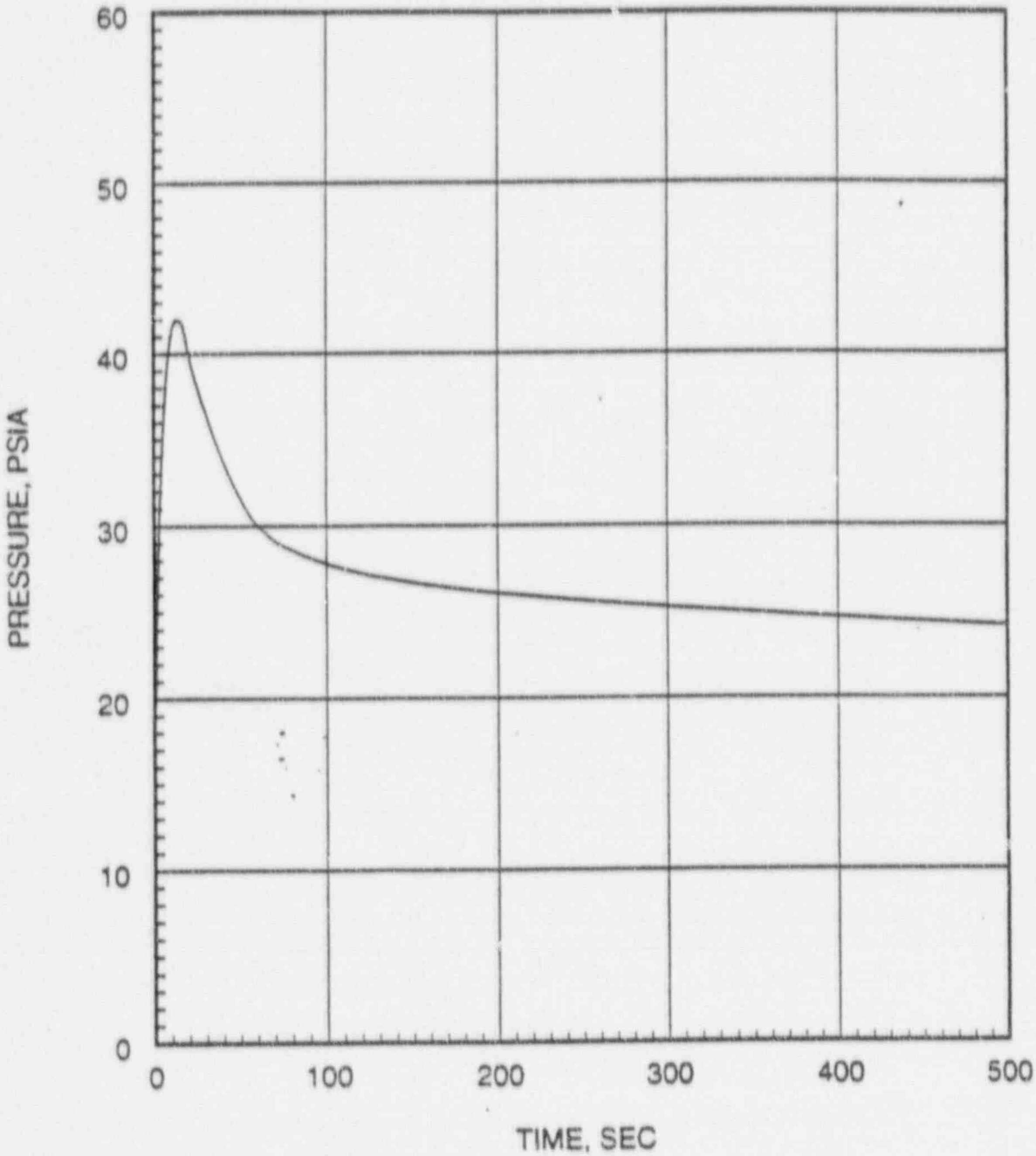


Figure 5g

0.6 Double Ended Guillotine Break in Pump Discharge Leg

Mass Added to Core During Reflood vs. Time

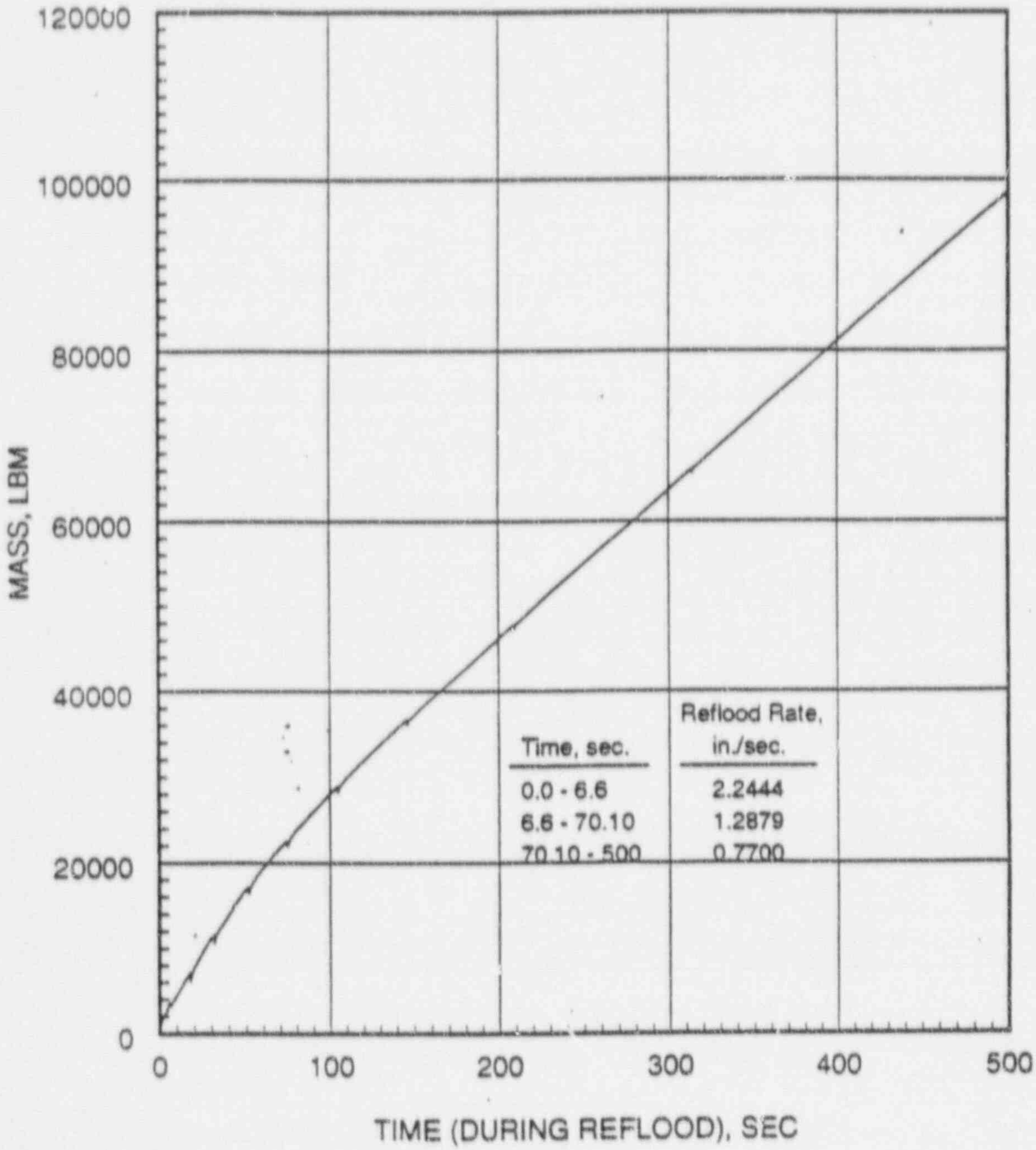


Figure 5b

0.6 Double Ended Guillotine Break in Pump Discharge Leg

Peak Cladding Temperature vs. Time

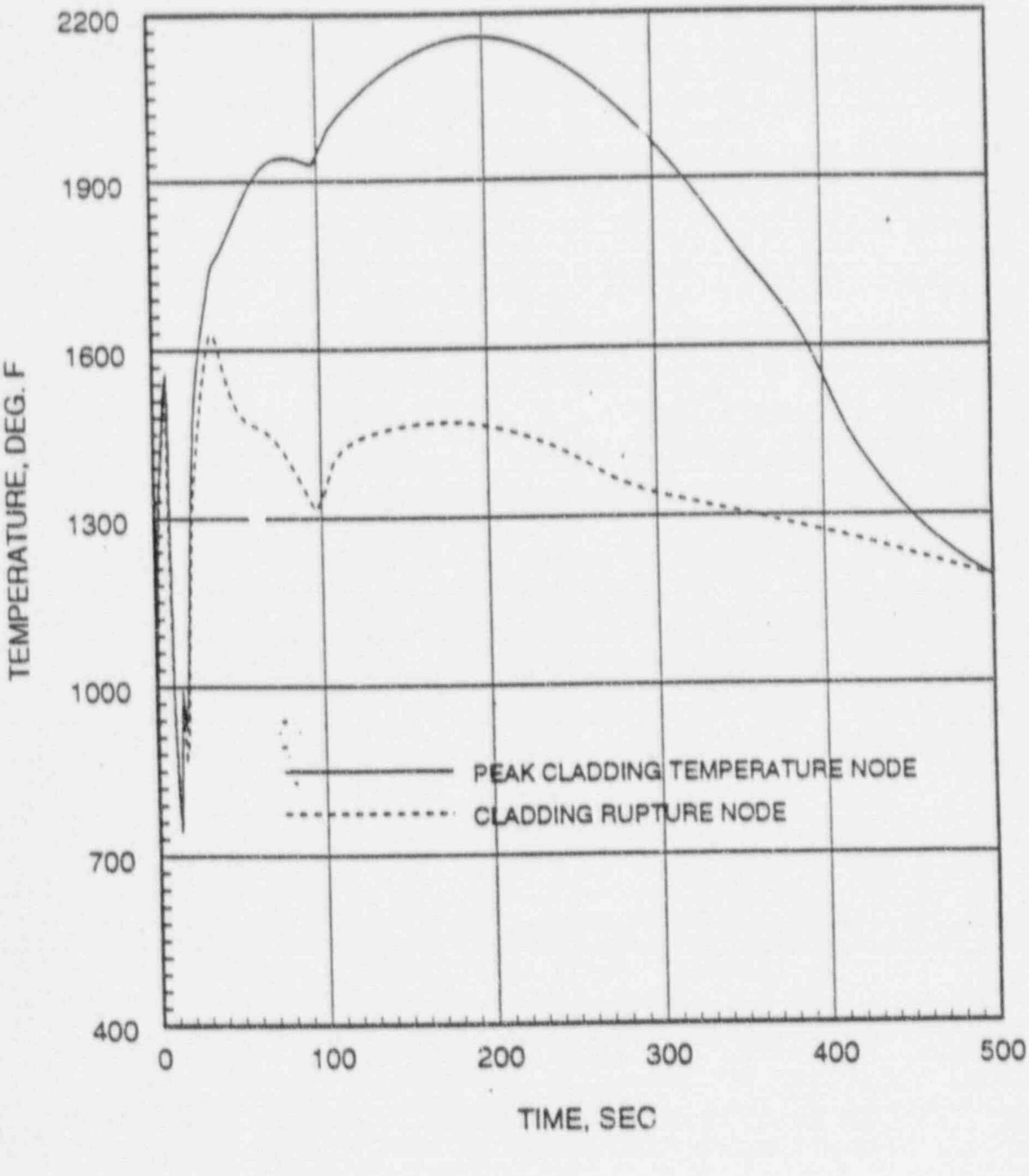


Figure 5i

0.6 Double Ended Guillotine Break in Pump Discharge Leg

Mid Annulus Flow Rate vs. Time

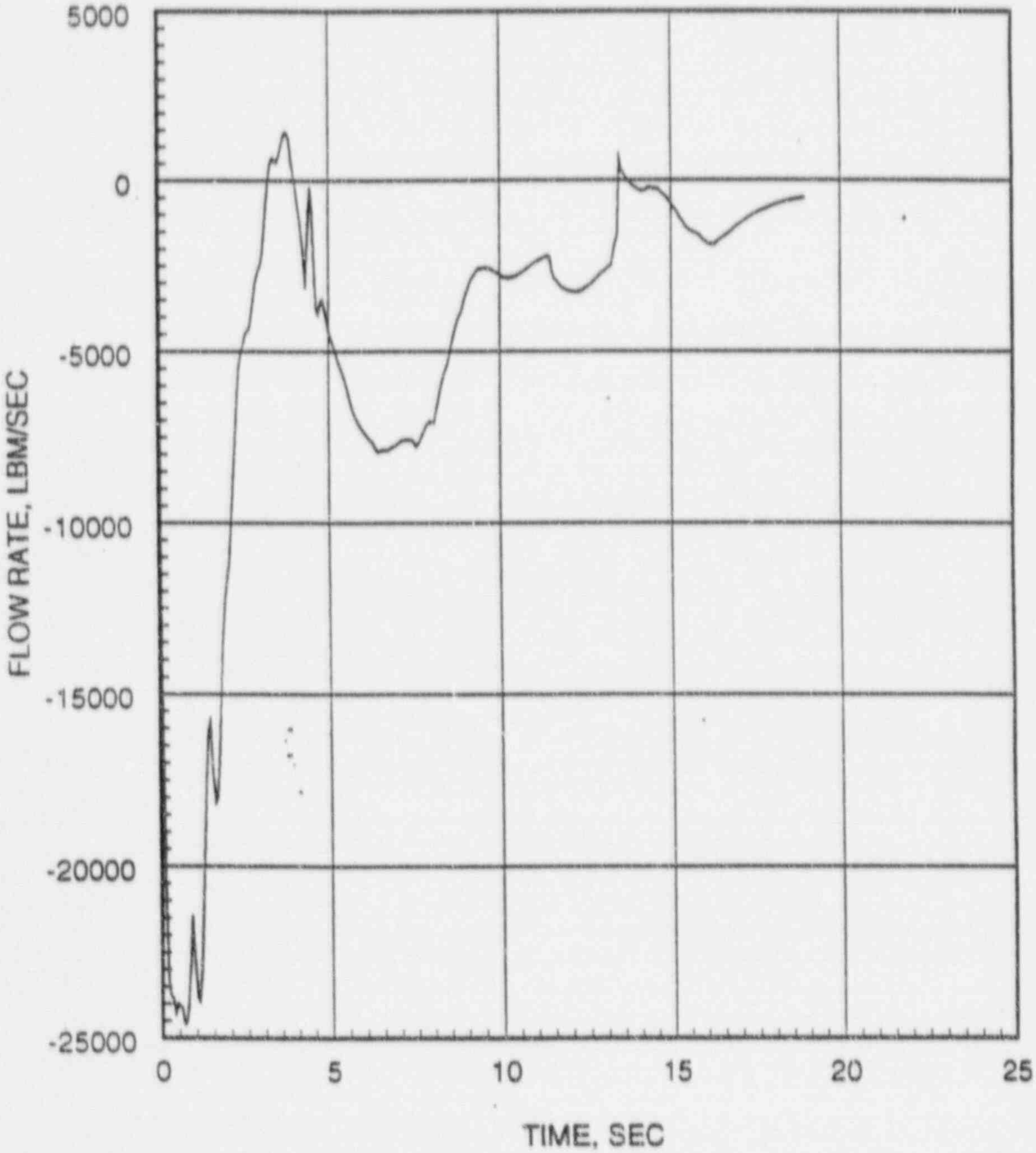


Figure 5j

0.6 Double Ended Guillotine Break in Fump Discharge Leg

Quality Above and Below the Core vs. Time

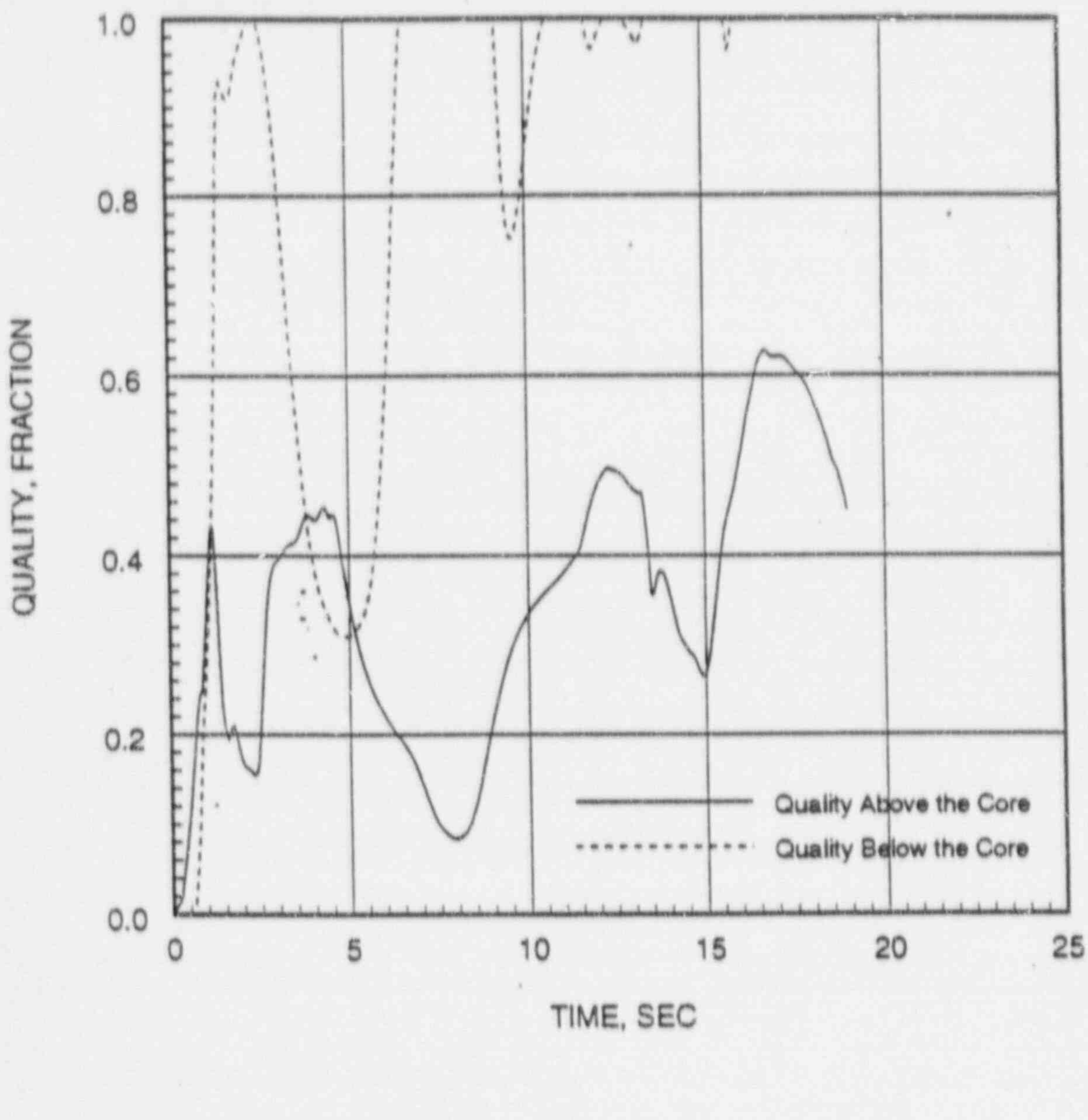


Figure 5k

0.6 Double Ended Guillotine Break in Pump Discharge Leg

Core Pressure Drop vs. Time

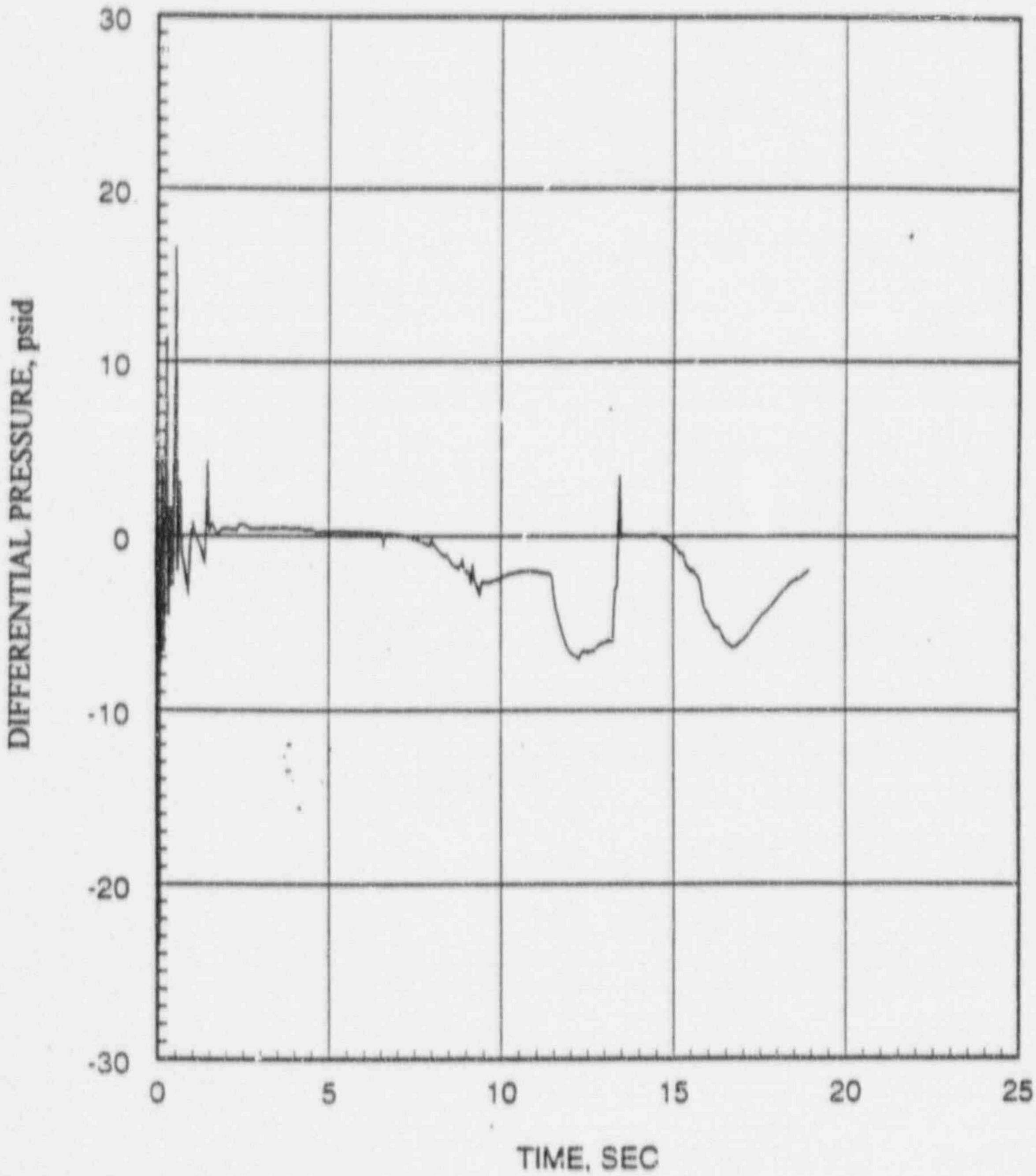


Figure 51

0.6 Double Ended Guillotine Break in Pump Discharge Leg

Safety Injection Flow Rate into Intact Discharge Legs vs. Time

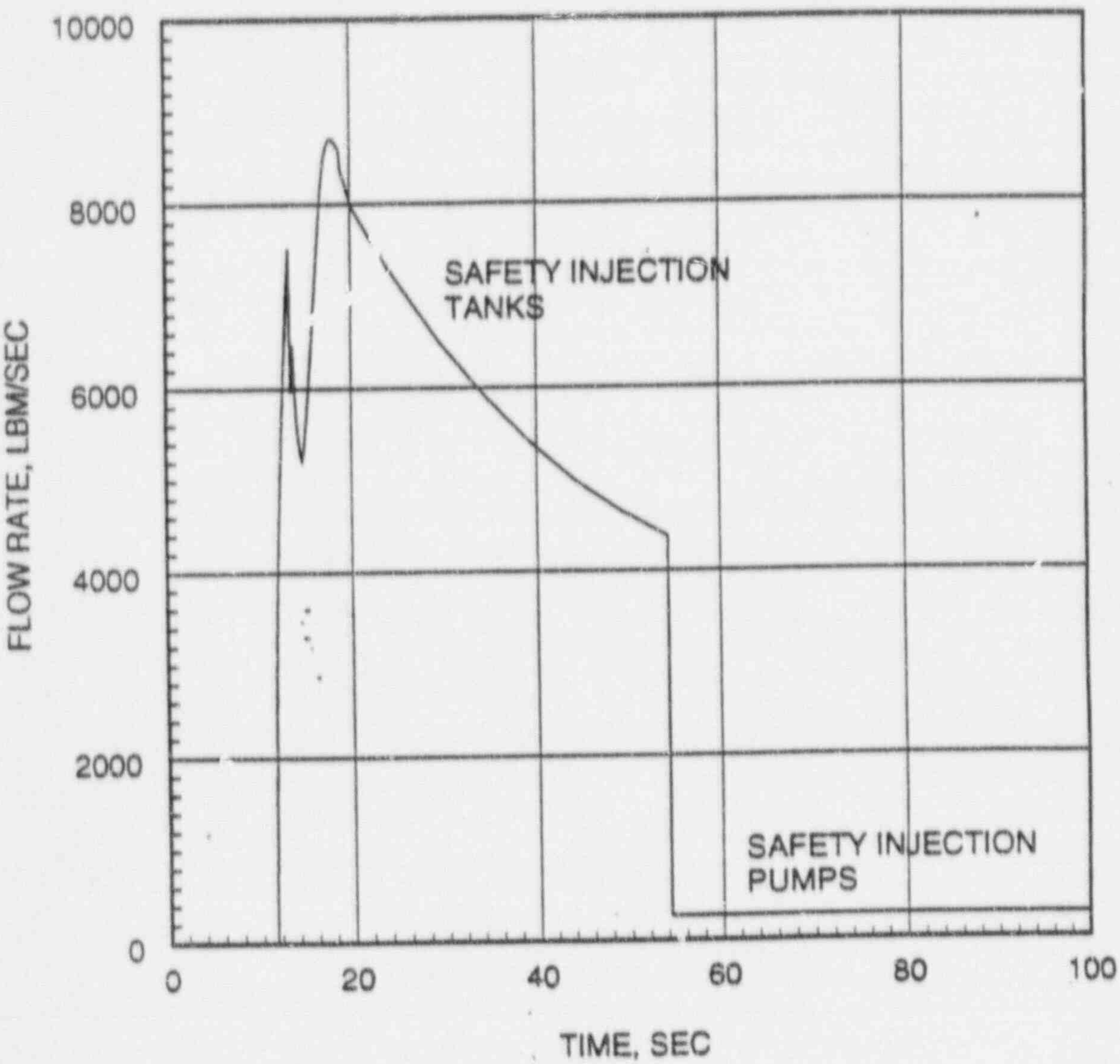


Figure 5m

0.6 Double Ended Guillotine Break in Pump Discharge Leg

Water Level in Downcomer During Reflood vs. Time

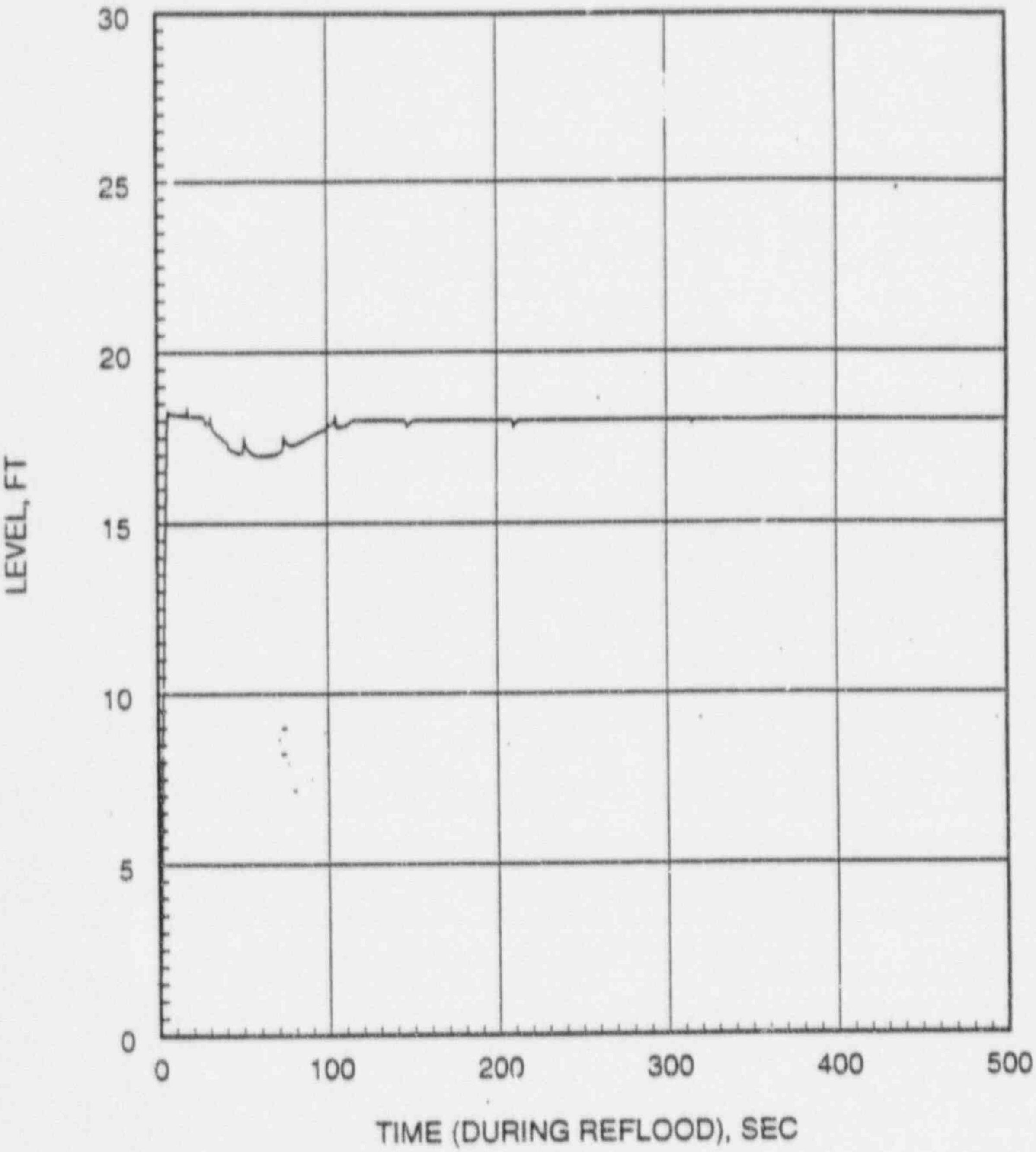


Figure 5n

0.6 Double Ended Guillotine Break in Pump Discharge Leg

Hot Spot Gap Conductance vs. Time

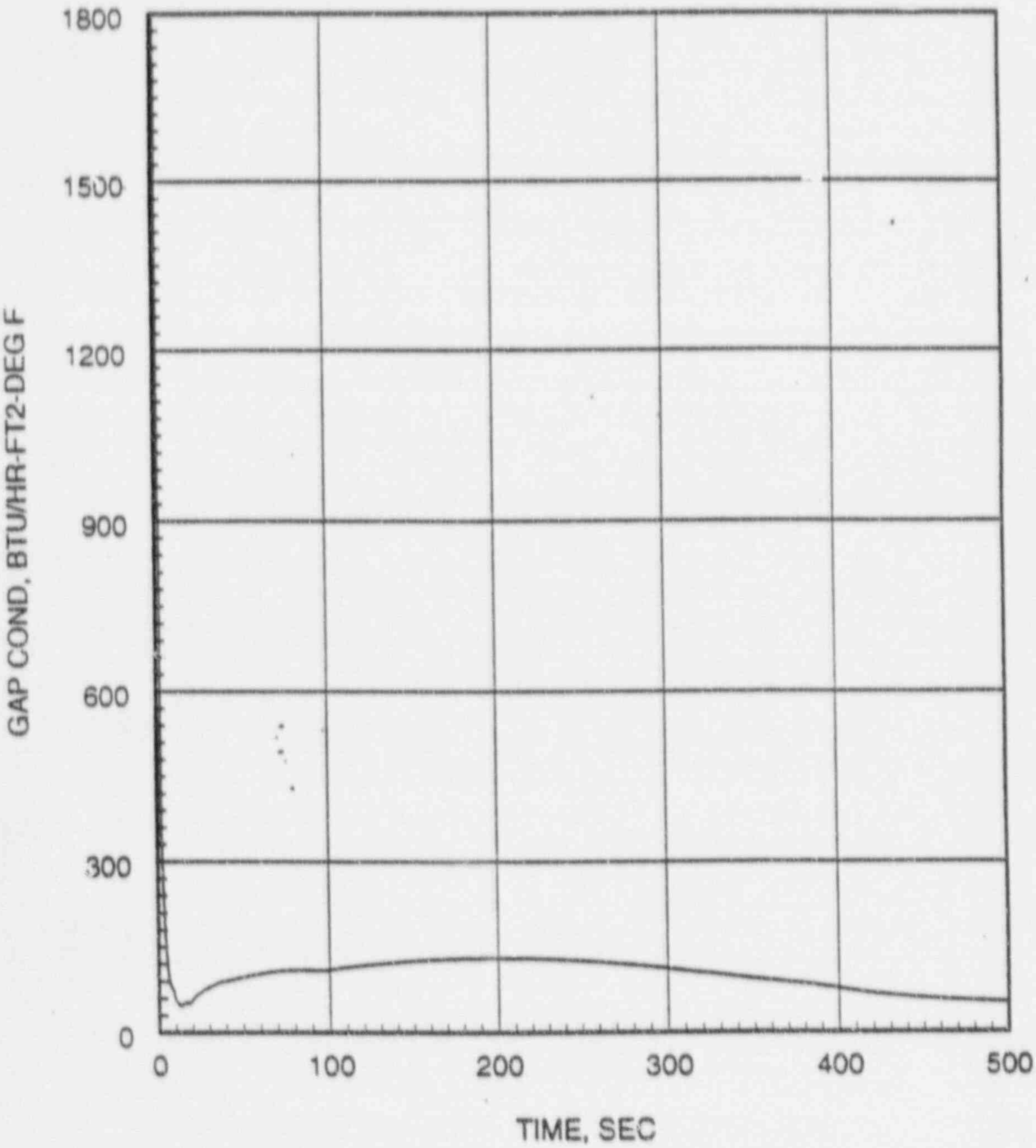
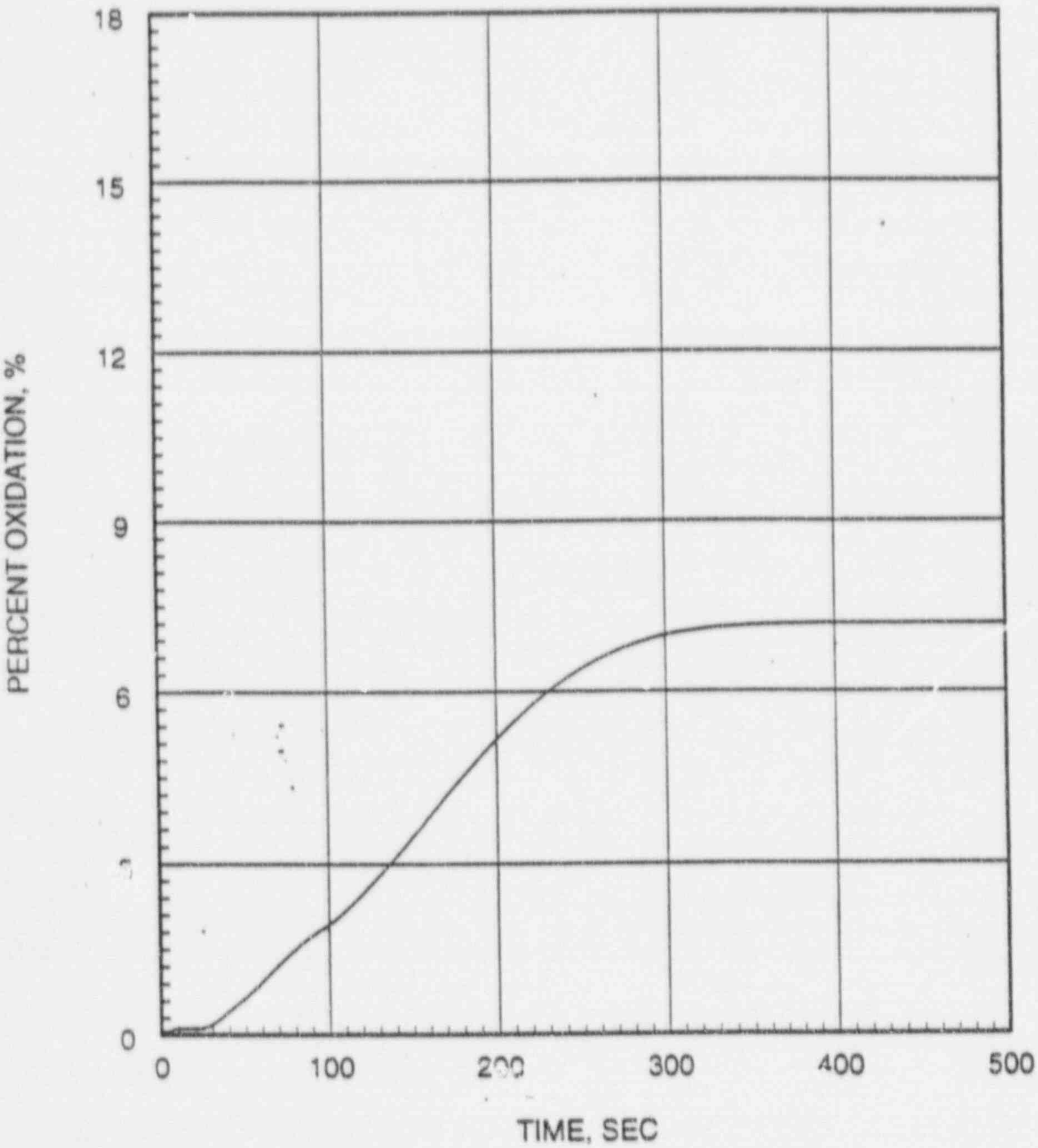


Figure 5o

0.6 Double Ended Guillotine Break in Pump Discharge Leg

Local Cladding Oxidation Percentage vs. Time



'ure 5p

0.6 Double Ended Guil. 'ine Break in Pump Discharge Leg

Fuel Centerline, Fuel Average, Cladding and Coolant Temperature at the Hot Spot vs. Time

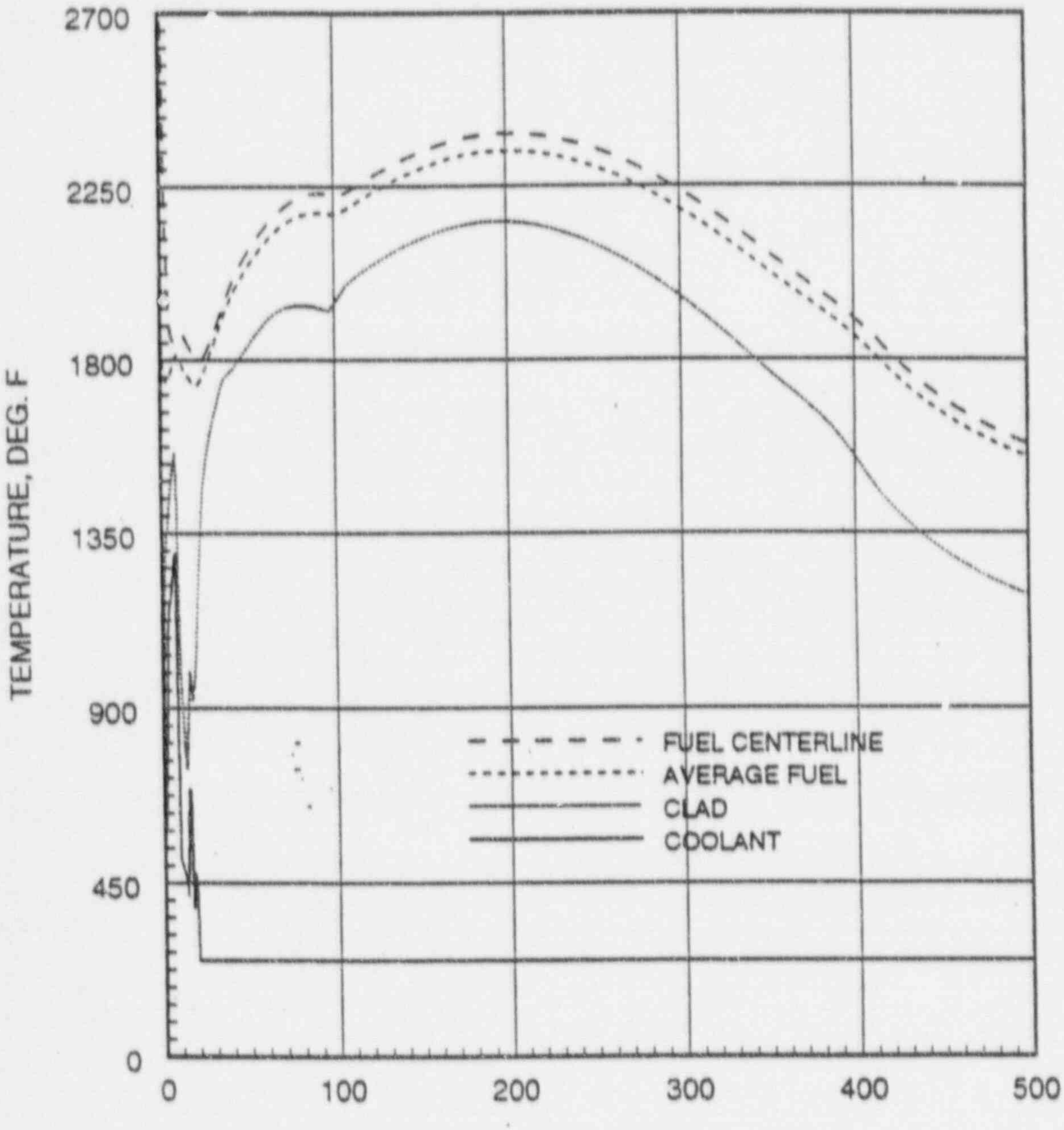


Figure 5q

0.6 Double Ended Guillotine Break in Pump Discharge Leg

Hot Spot Heat Transfer Coefficient vs. Time

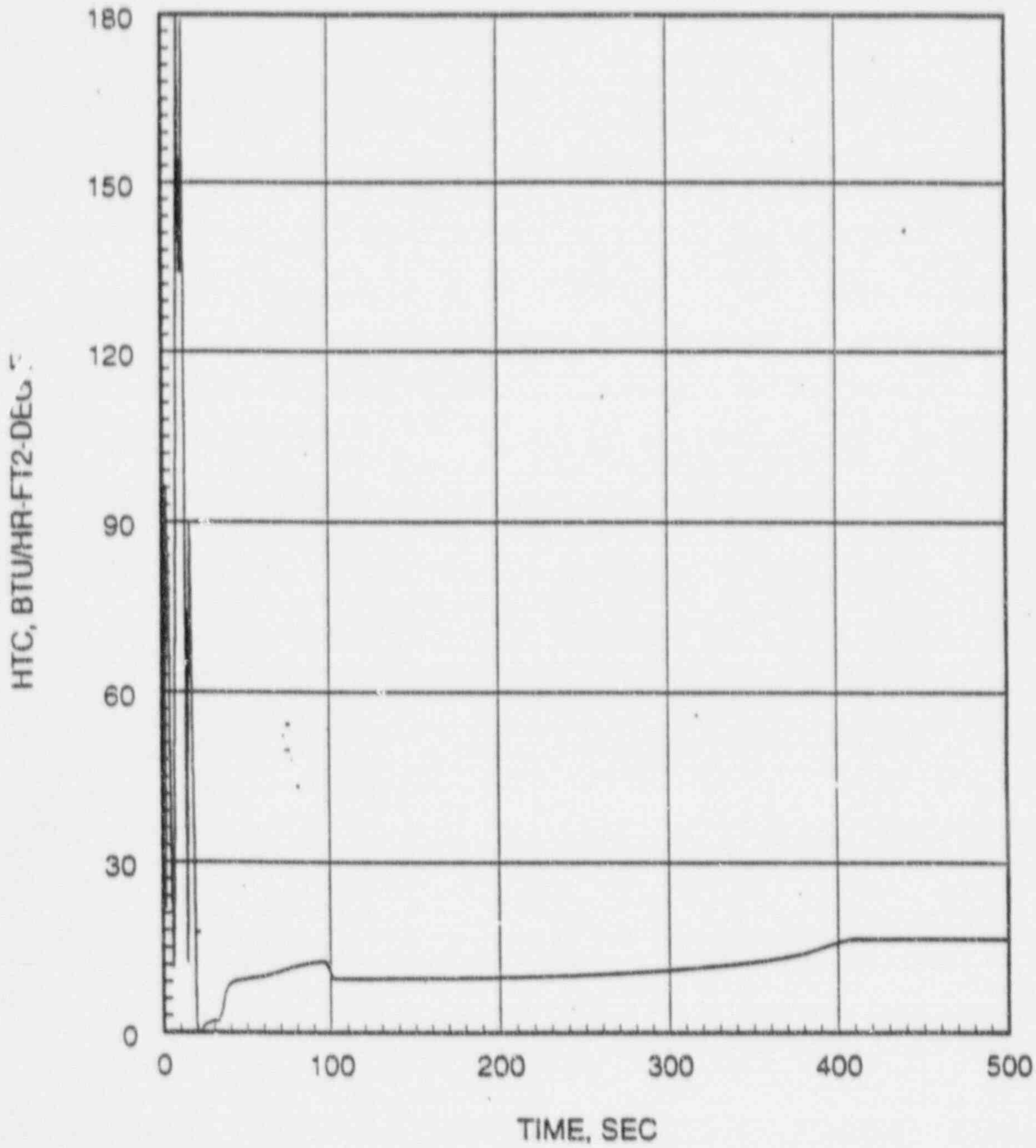


Figure 5r

0.6 Double Ended Guillotine Break in Pump Discharge Leg

Hot Pin Pressure vs. Time

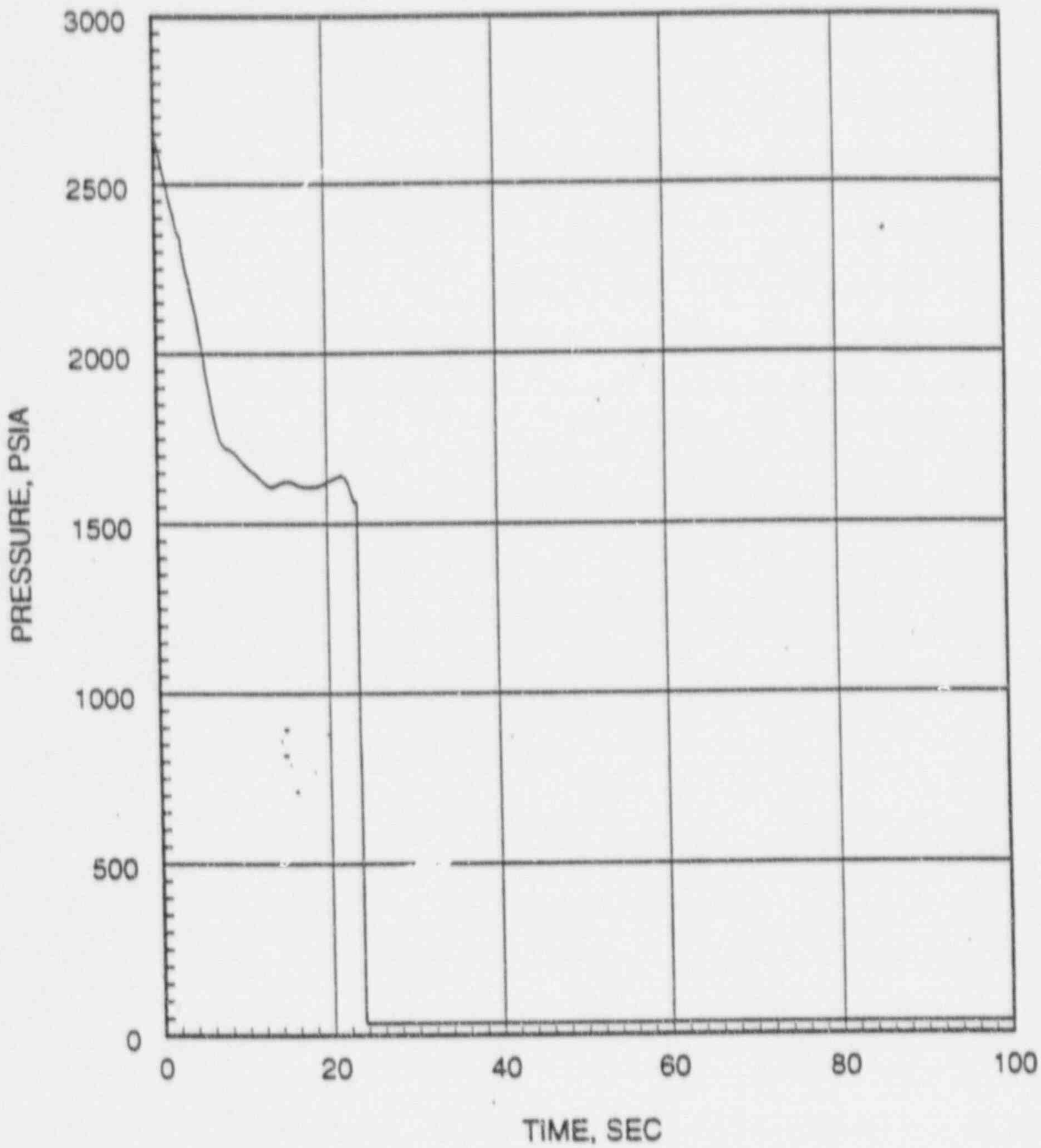


Figure 6a

0.06 FT² Break in Pump Discharge Leg

Normalized Core Power vs. Time

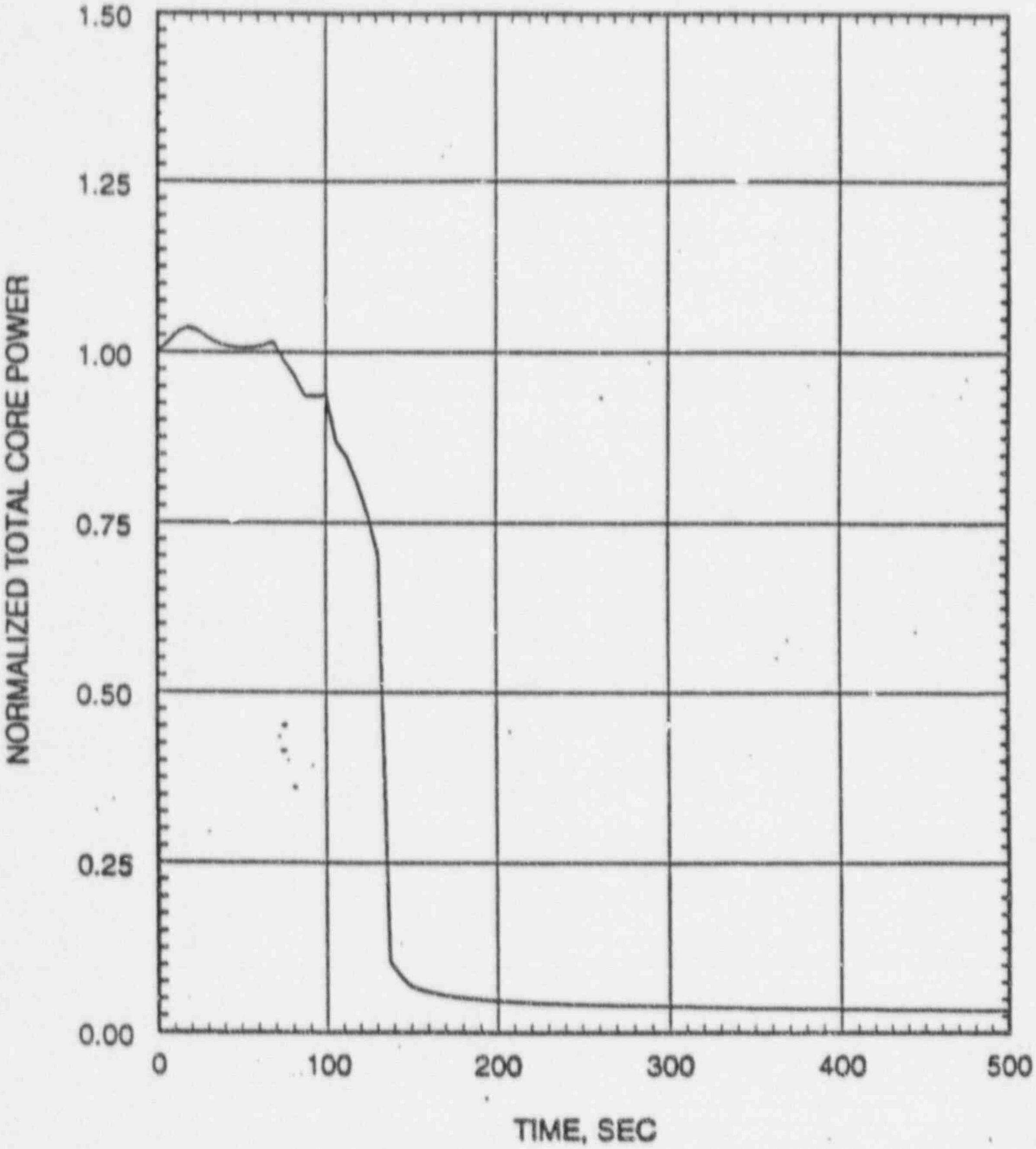


Figure 6b

0.06 FT^2 Break in Pump Discharge Leg

Inner Vessel Pressure vs. Time

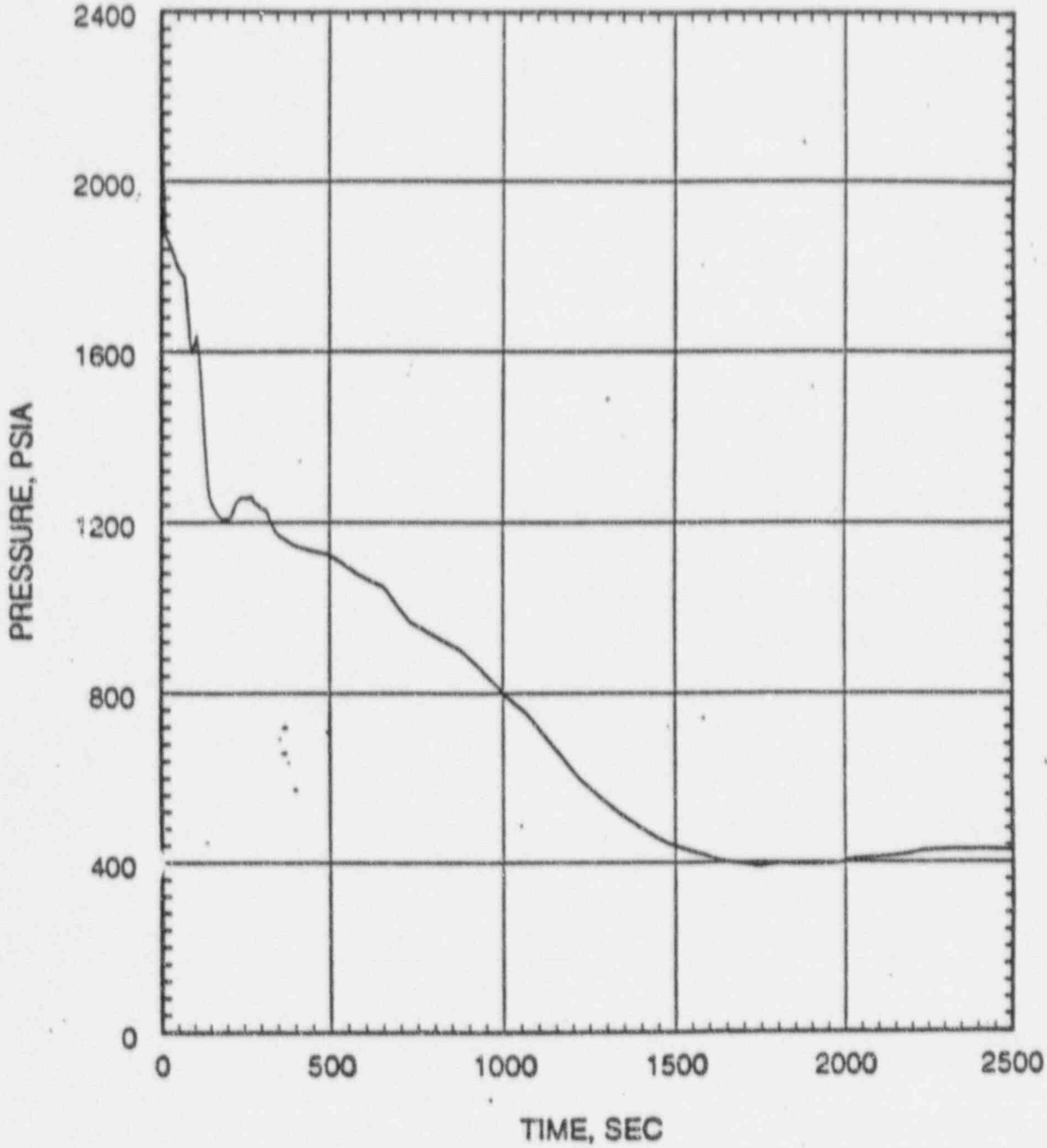


Figure 6c

0.06 FT^2 Break in Pump Discharge Leg

Break Flow Rate vs. Time

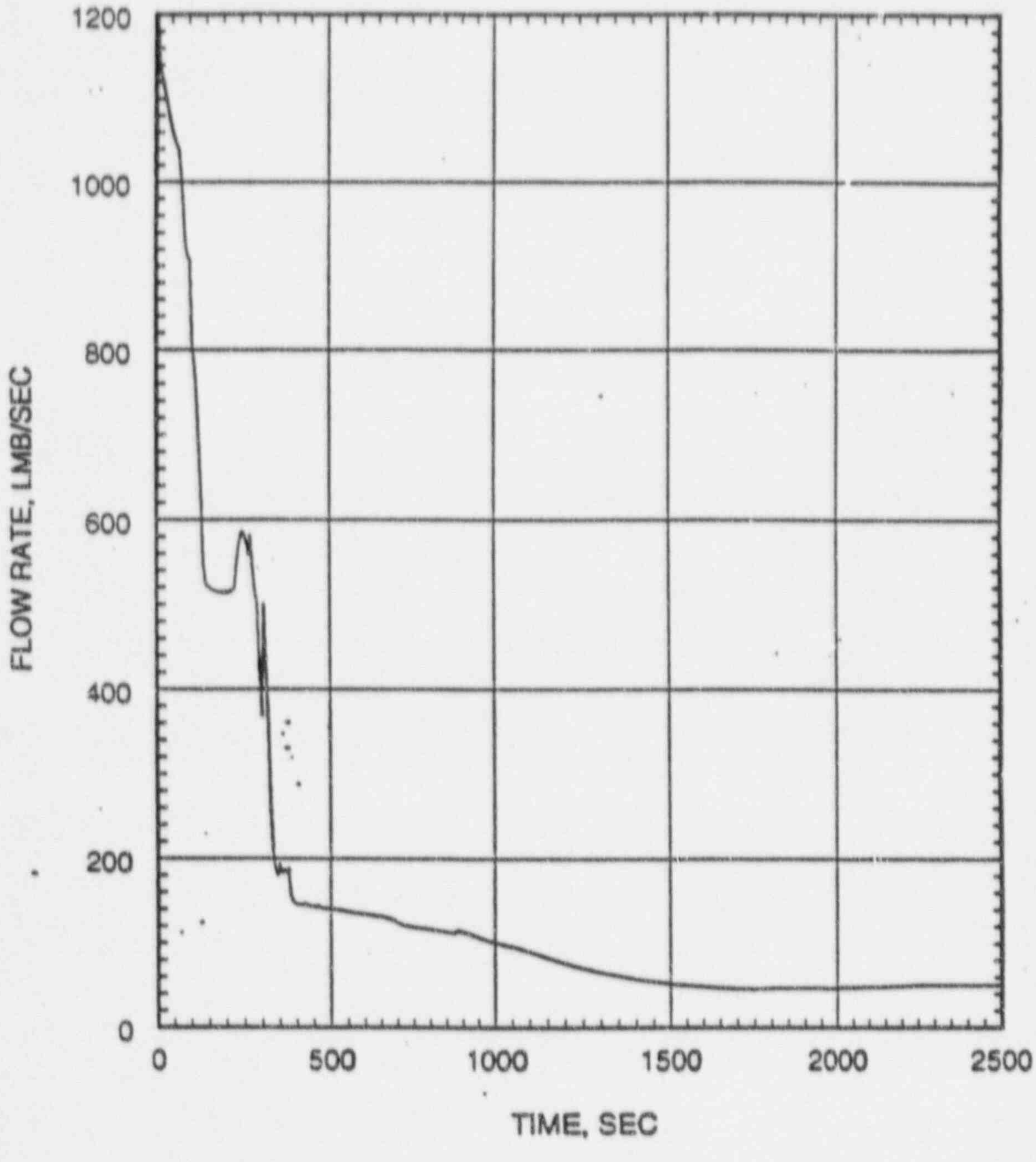


Figure 6d

0.06 FT² Break in Pump Discharge Leg

Inner Vessel Inlet Flow Rate vs. Time

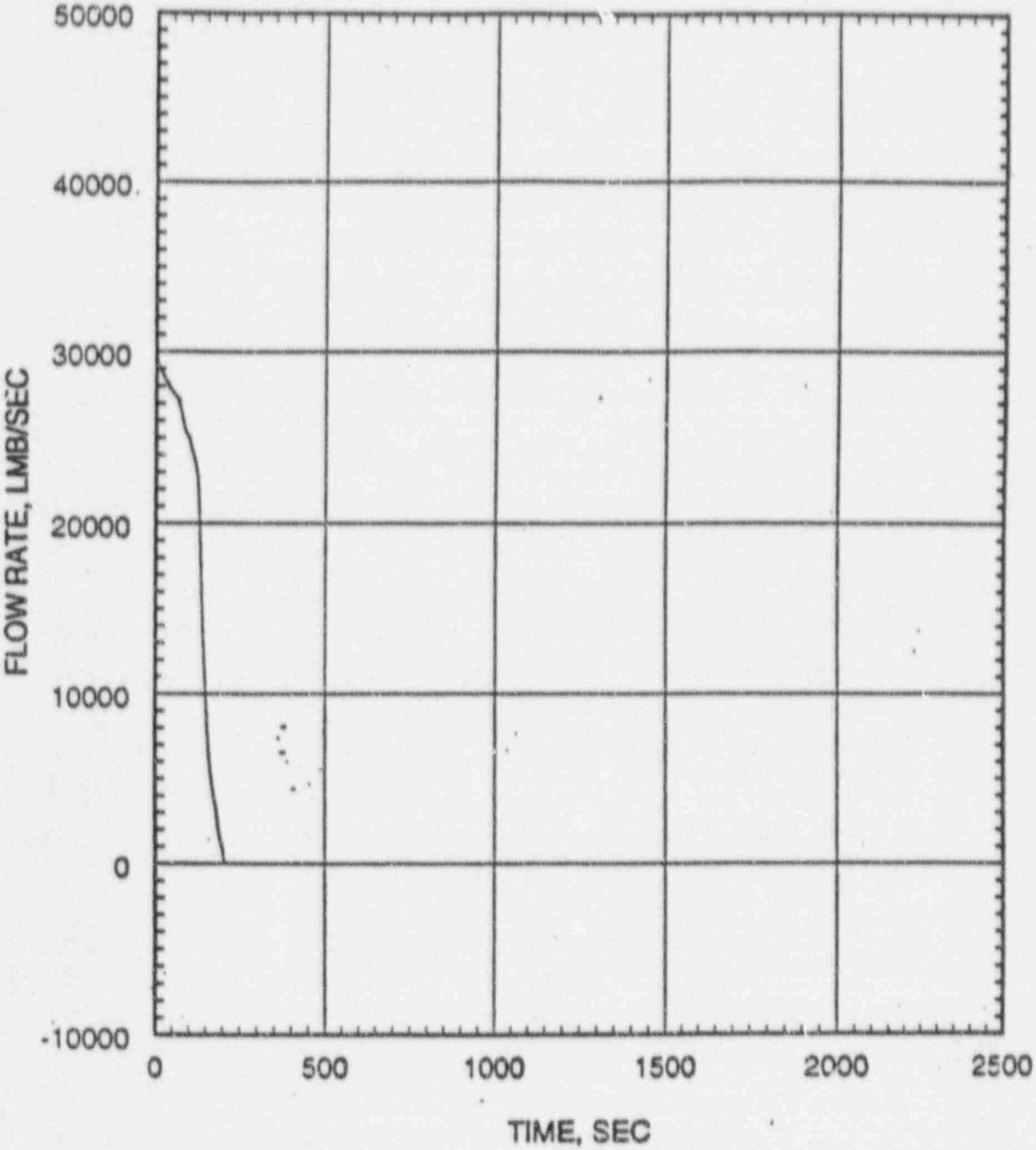


Figure 6e

0.06 FT^2 Break in Pump Discharge Leg

Inner Vessel Two-Phase Mixture Level vs. Time

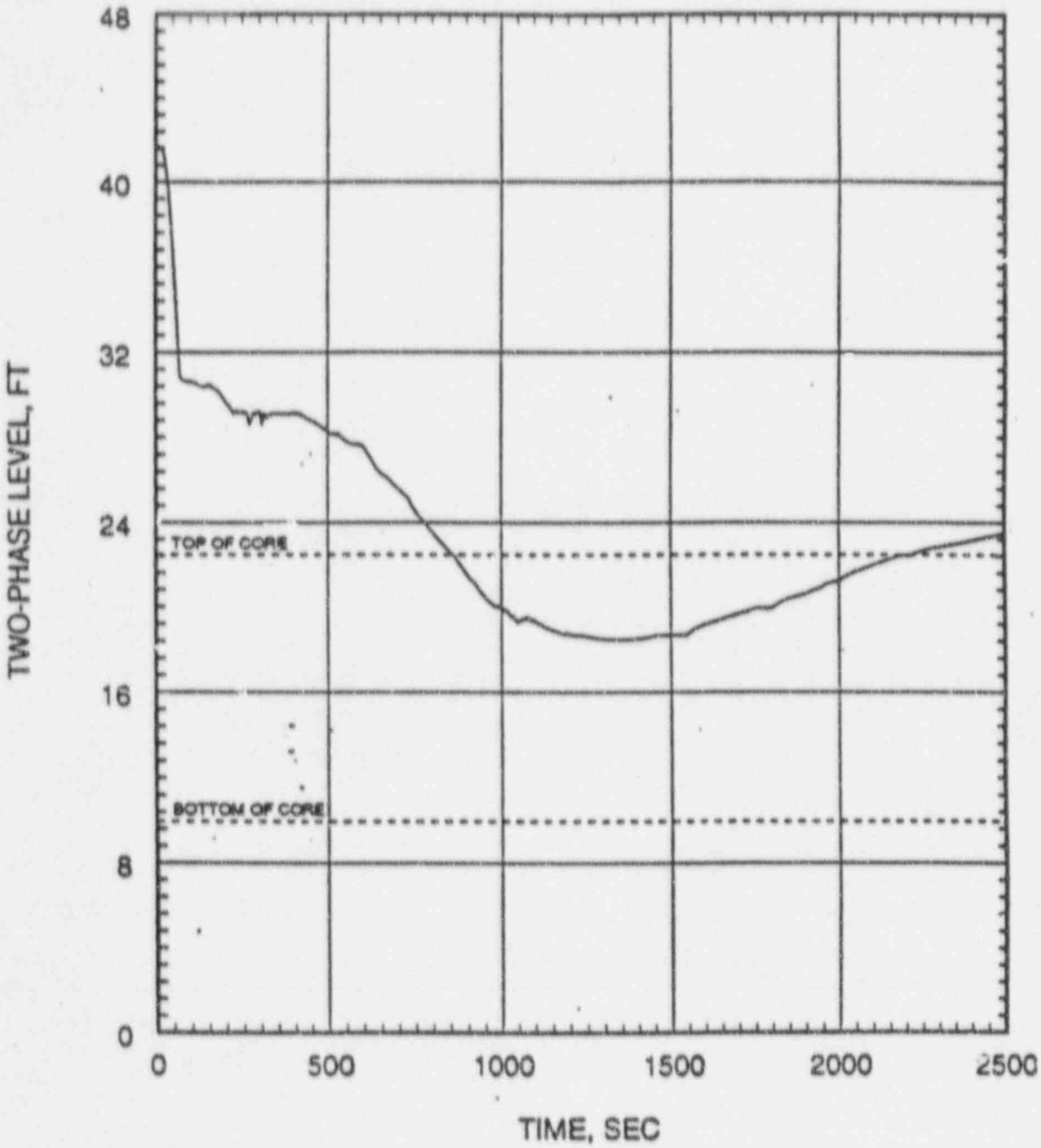


Figure 6f

0.06 FT^2 Break in Pump Discharge Leg

Heat Transfer Coefficient at Hot Spot vs. Time

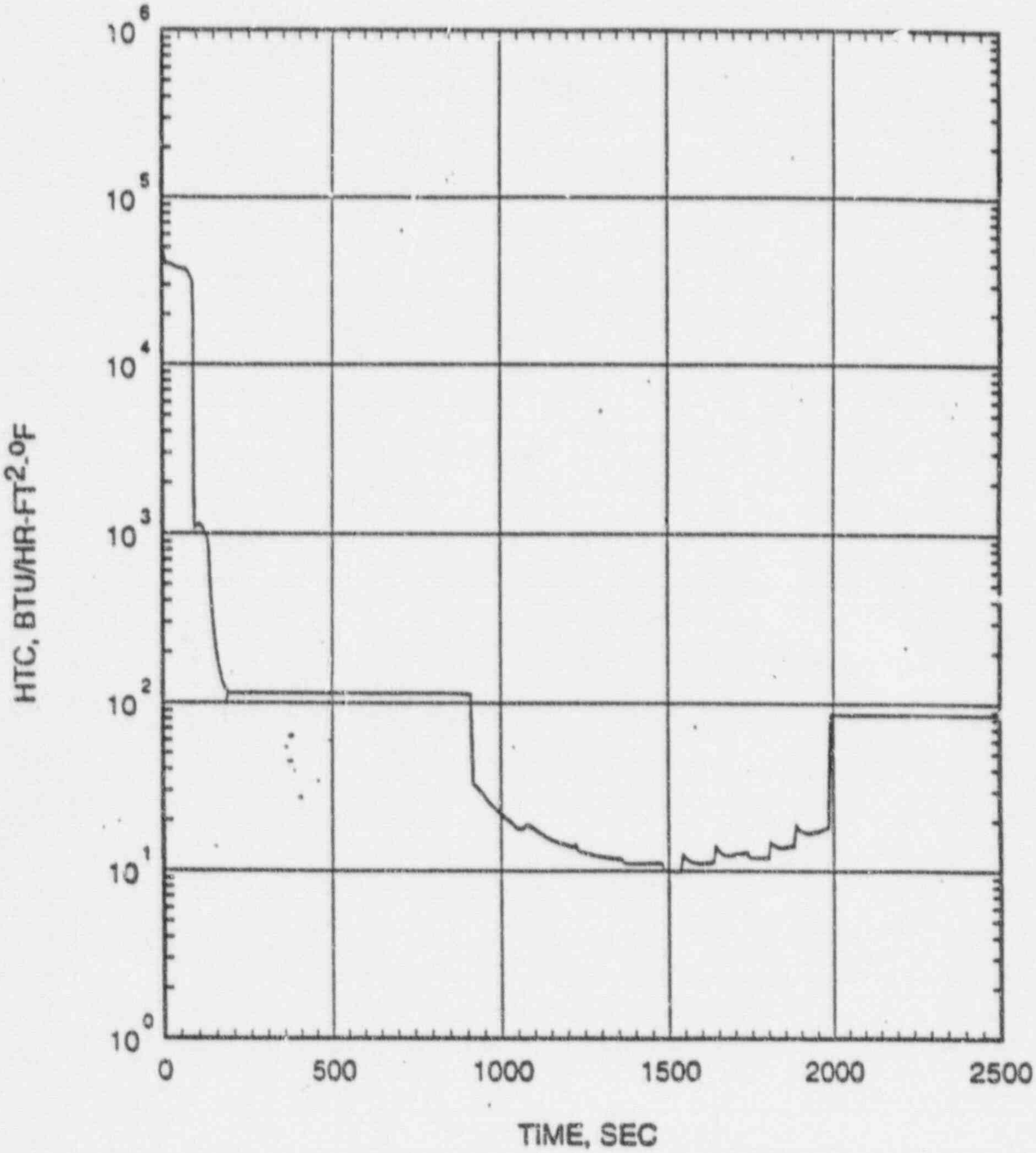


Figure 6g

0.06 FT² Break in Pump Discharge Leg

Coolant Temperature at Hot Spot vs. Time

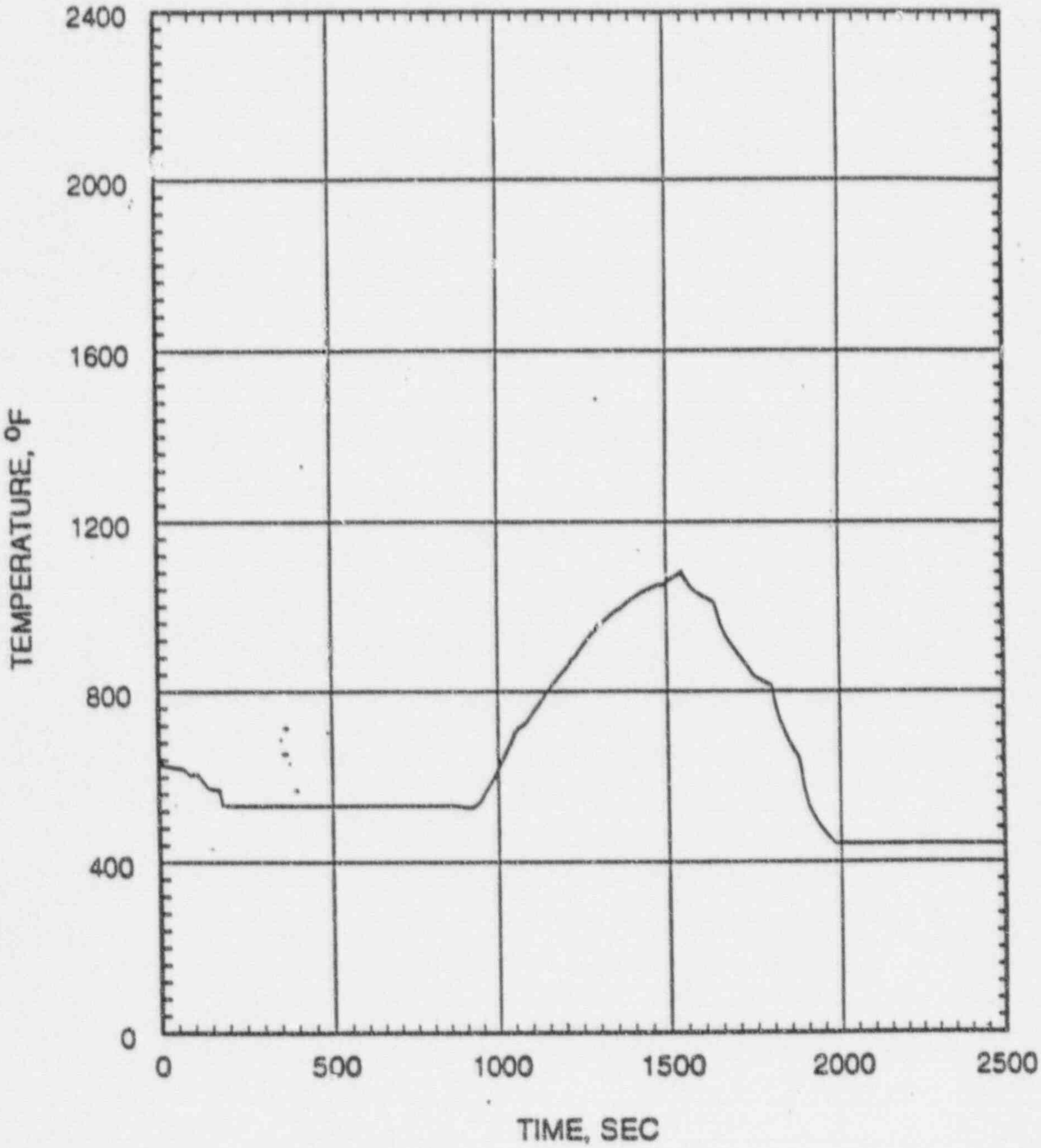


Figure 6h

0.06 FT^2 Break in Pump Discharge Leg

Cladding Temperature at Hot Spot vs. Time

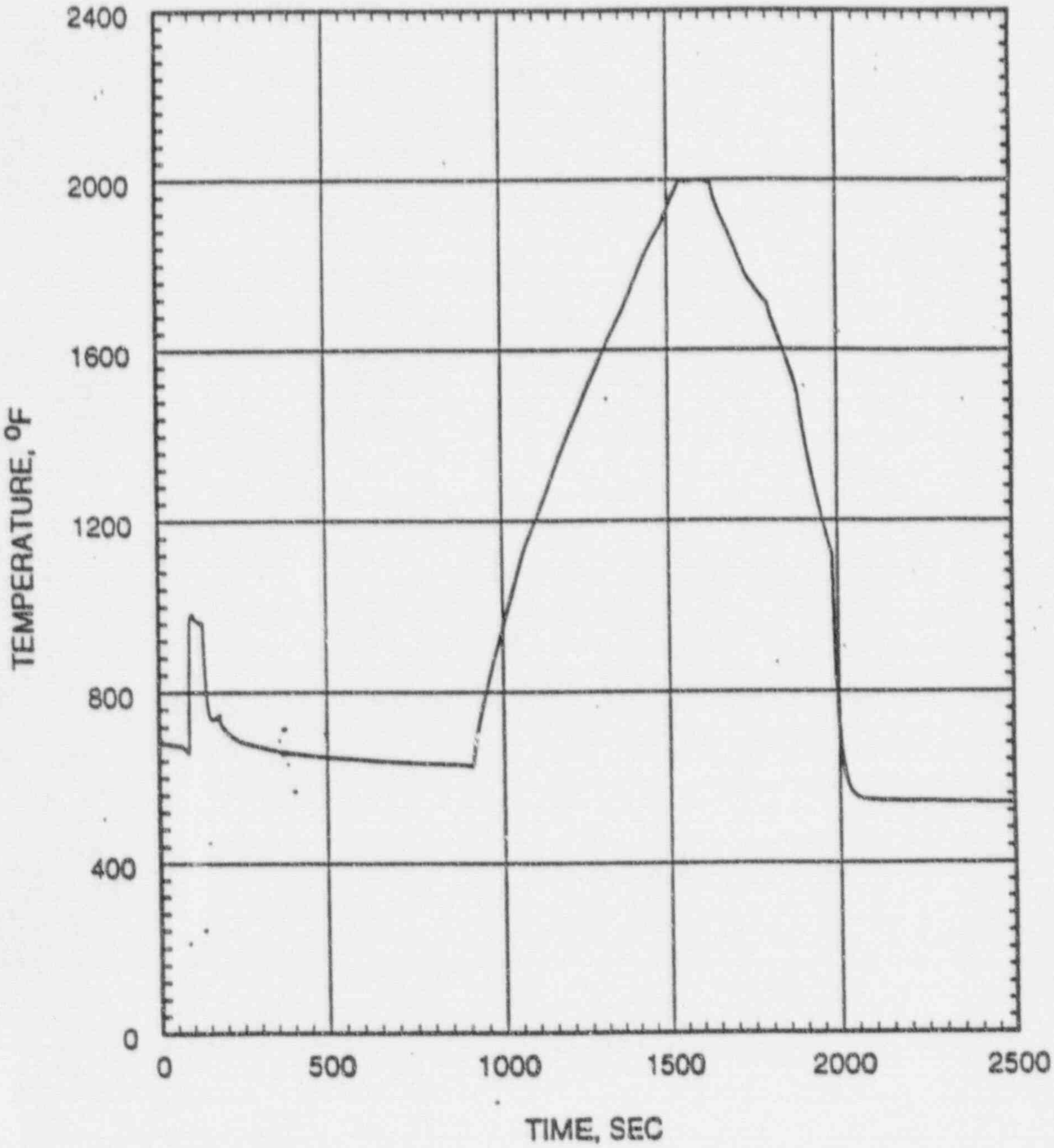


Figure 7a

0.05 FT^2 Break in Pump Discharge Leg

Normalized Core Power vs. Time

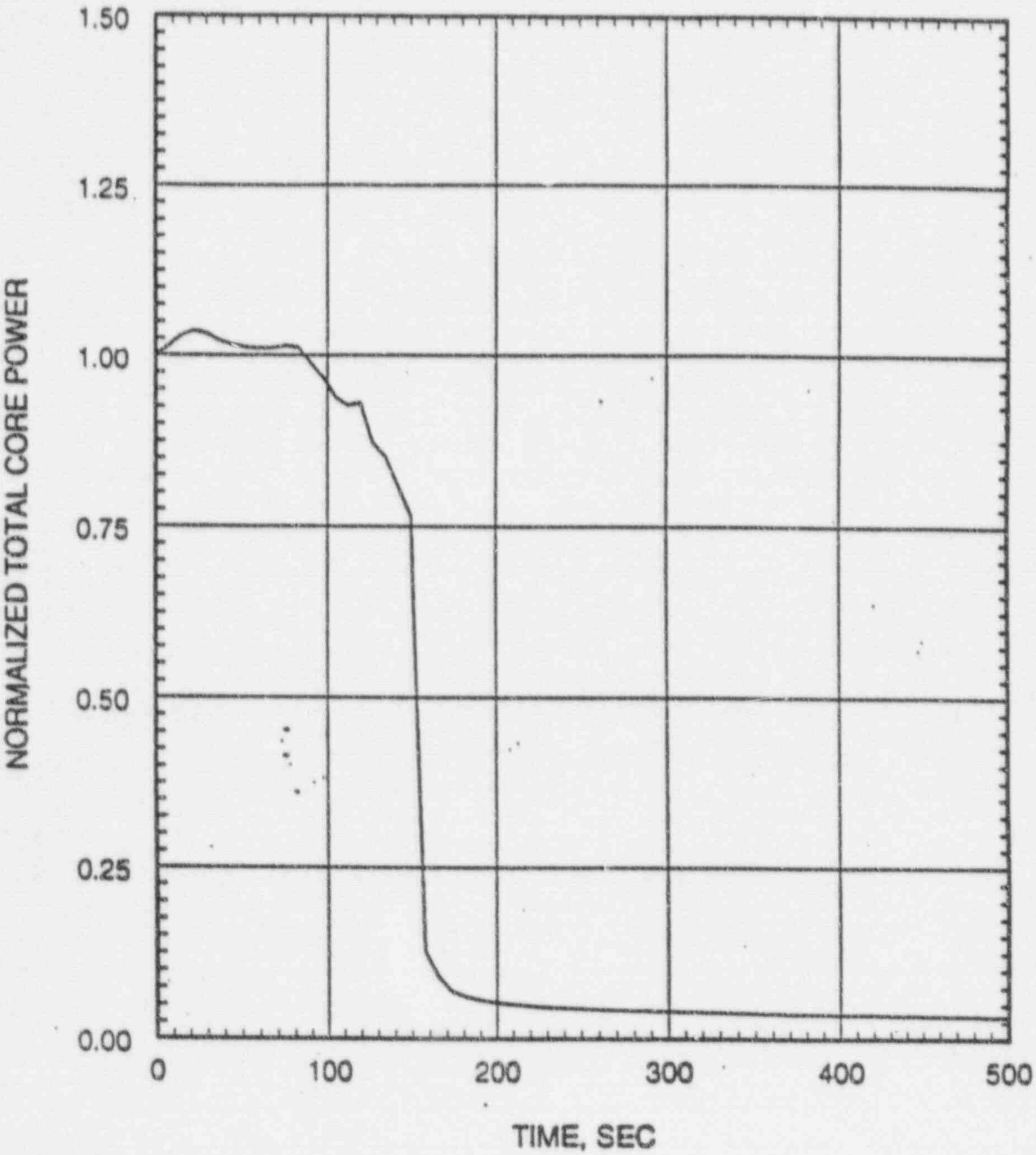


Figure 7b

0.05 FT^2 Break in Pump Discharge Leg

Inner Vessel Pressure vs. Time

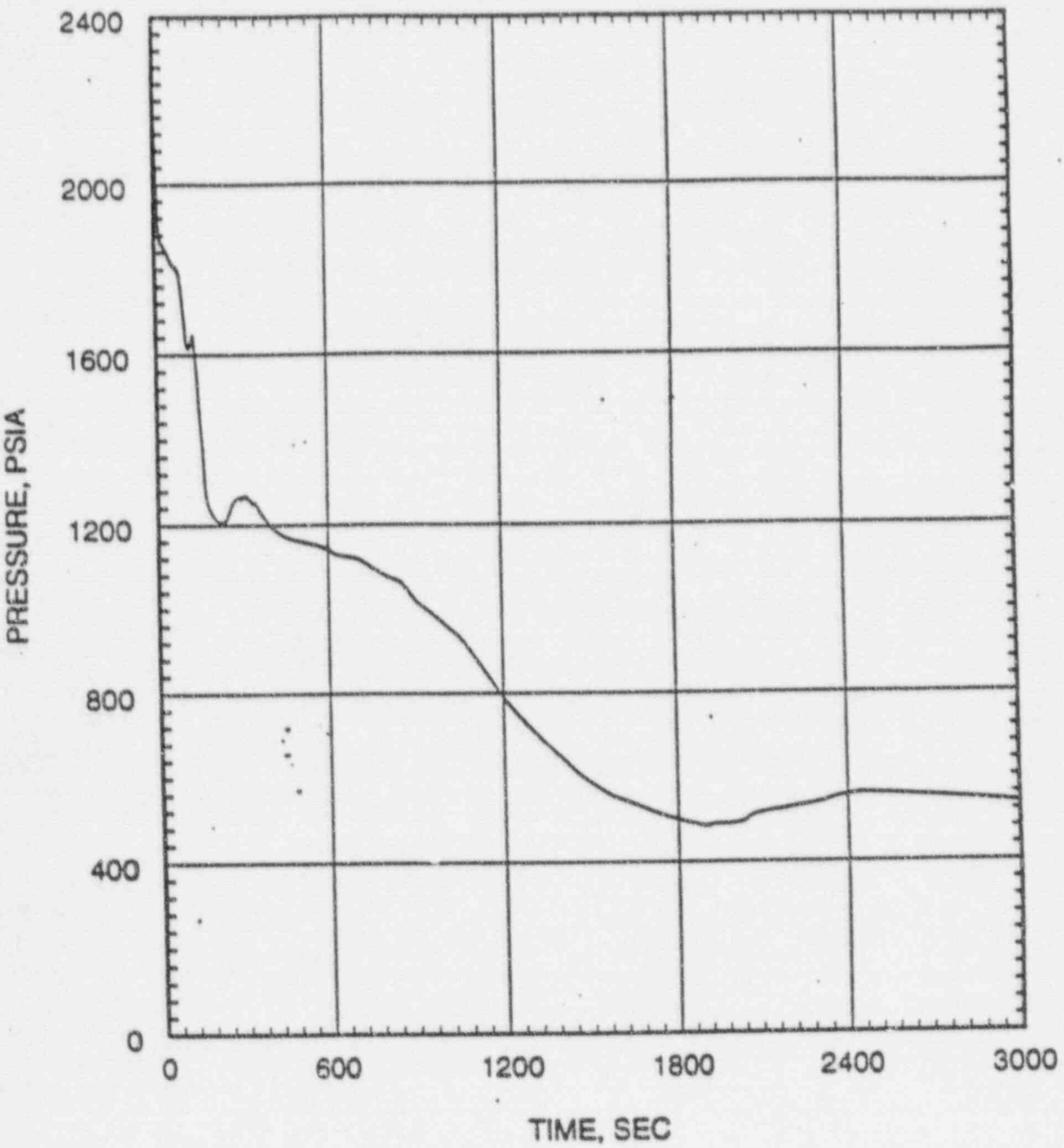


Figure 7c

0.05 FT² Break in Pump Discharge Leg

Break Flow Rate vs. Time

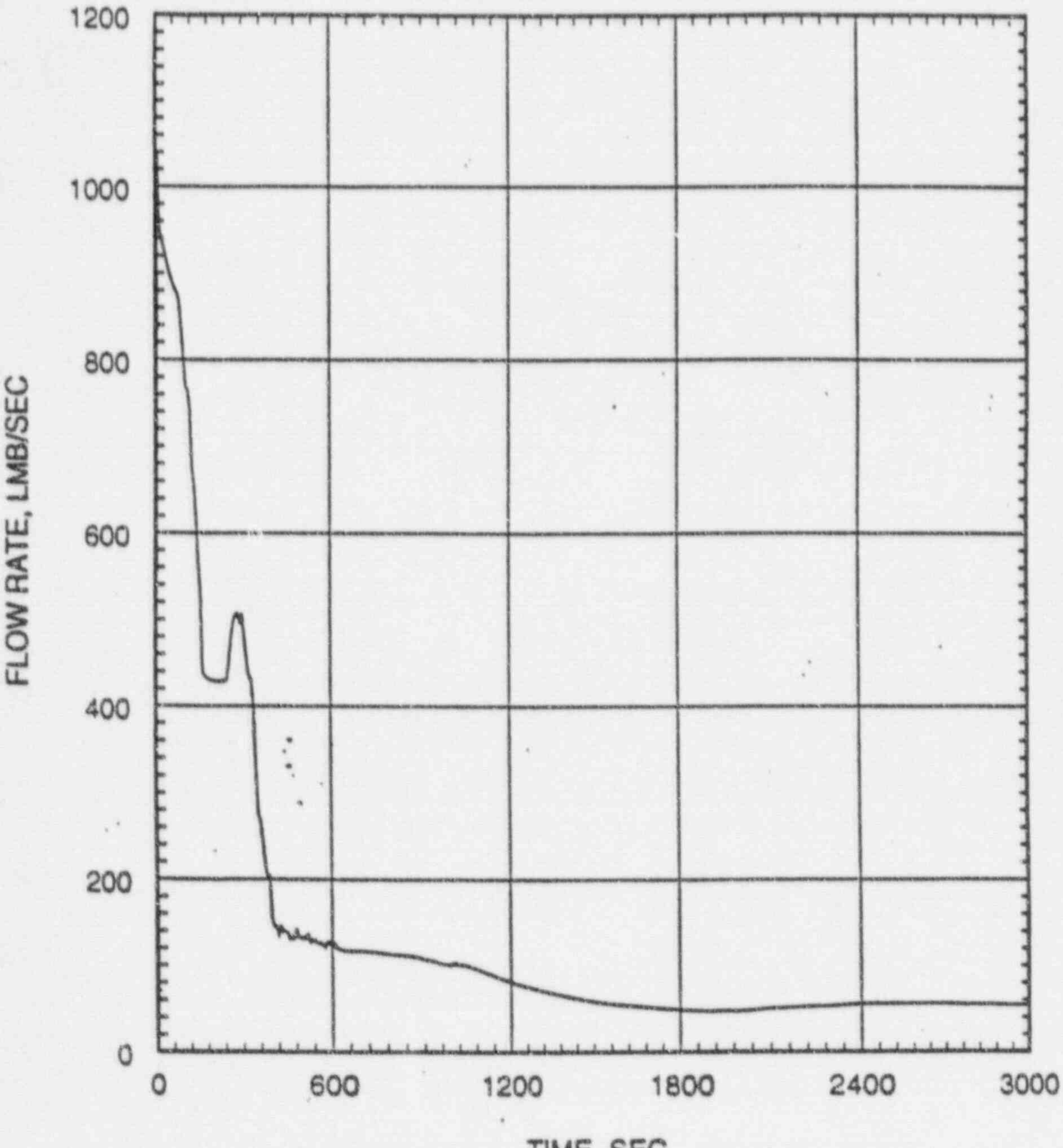


Figure 7d

0.05 FT² Break in Pump Discharge Leg

Inner Vessel Inlet Flow Rate vs. Time

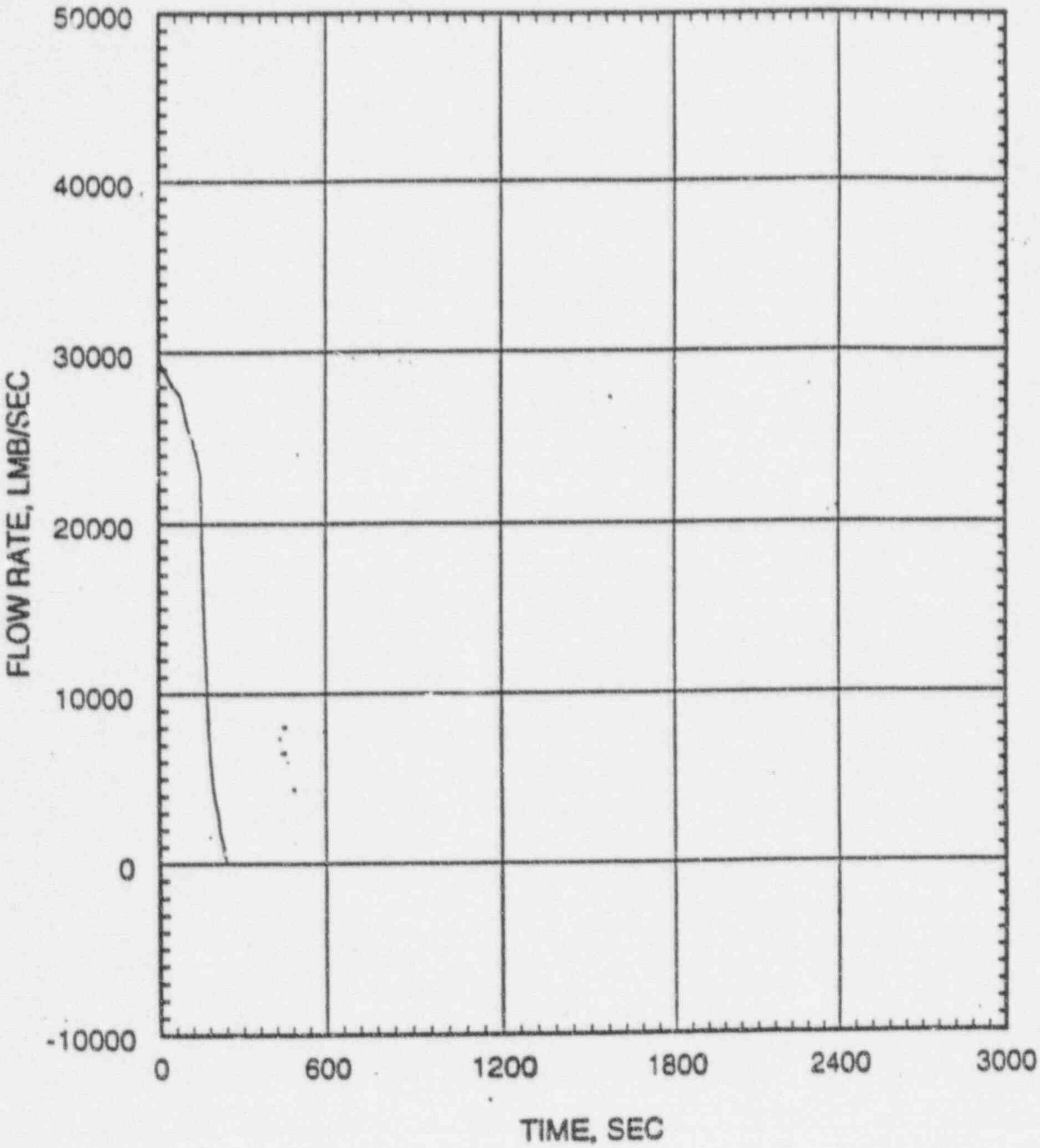


Figure 7e

0.05 FT^2 Break in Pump Discharge Leg

Inner Vessel Two-Phase Mixture Level vs. Time

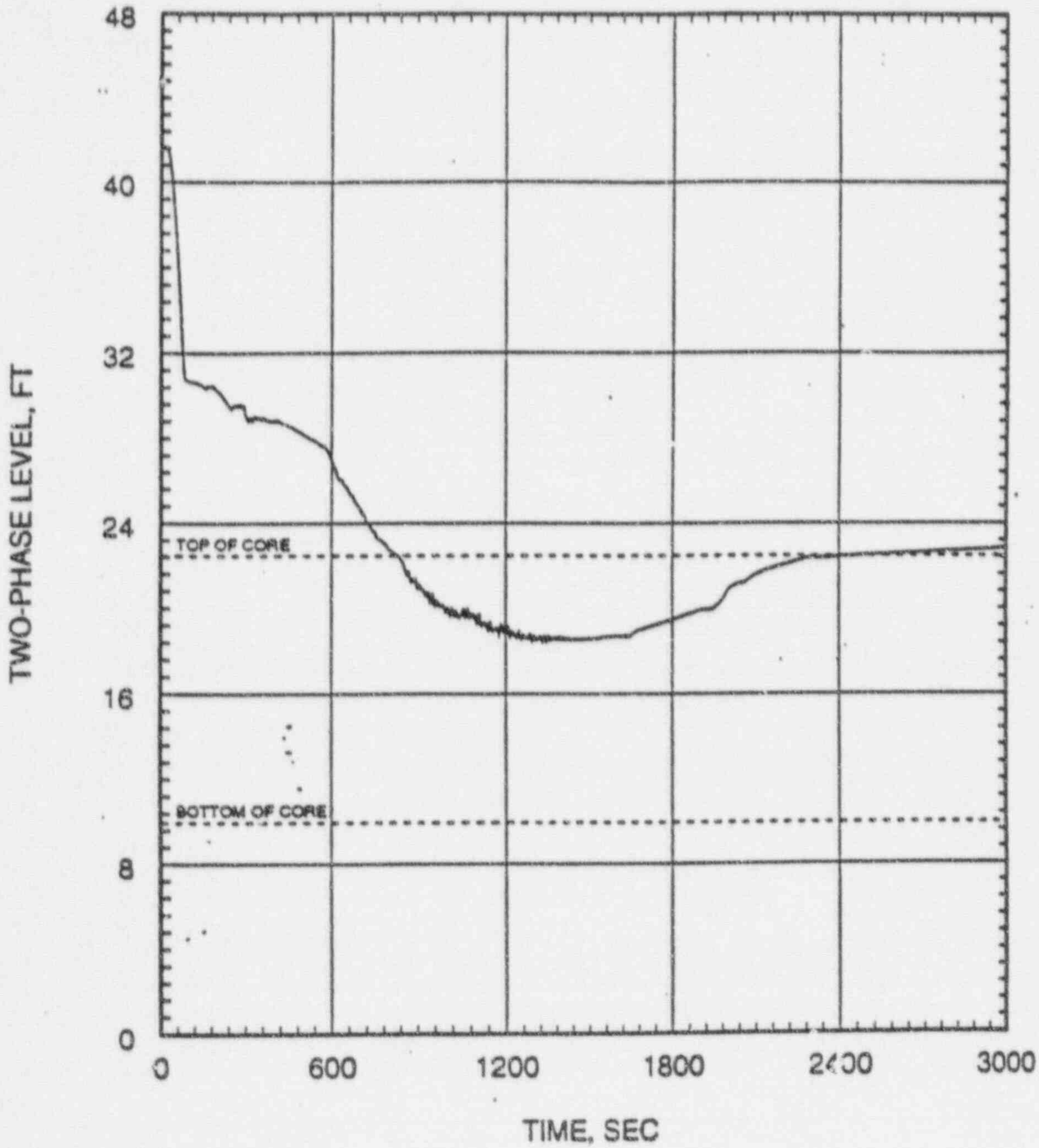


Figure 7f

0.05 FT^2 Break in Pump Discharge Leg

Heat Transfer Coefficient at Hot Spot vs. Time

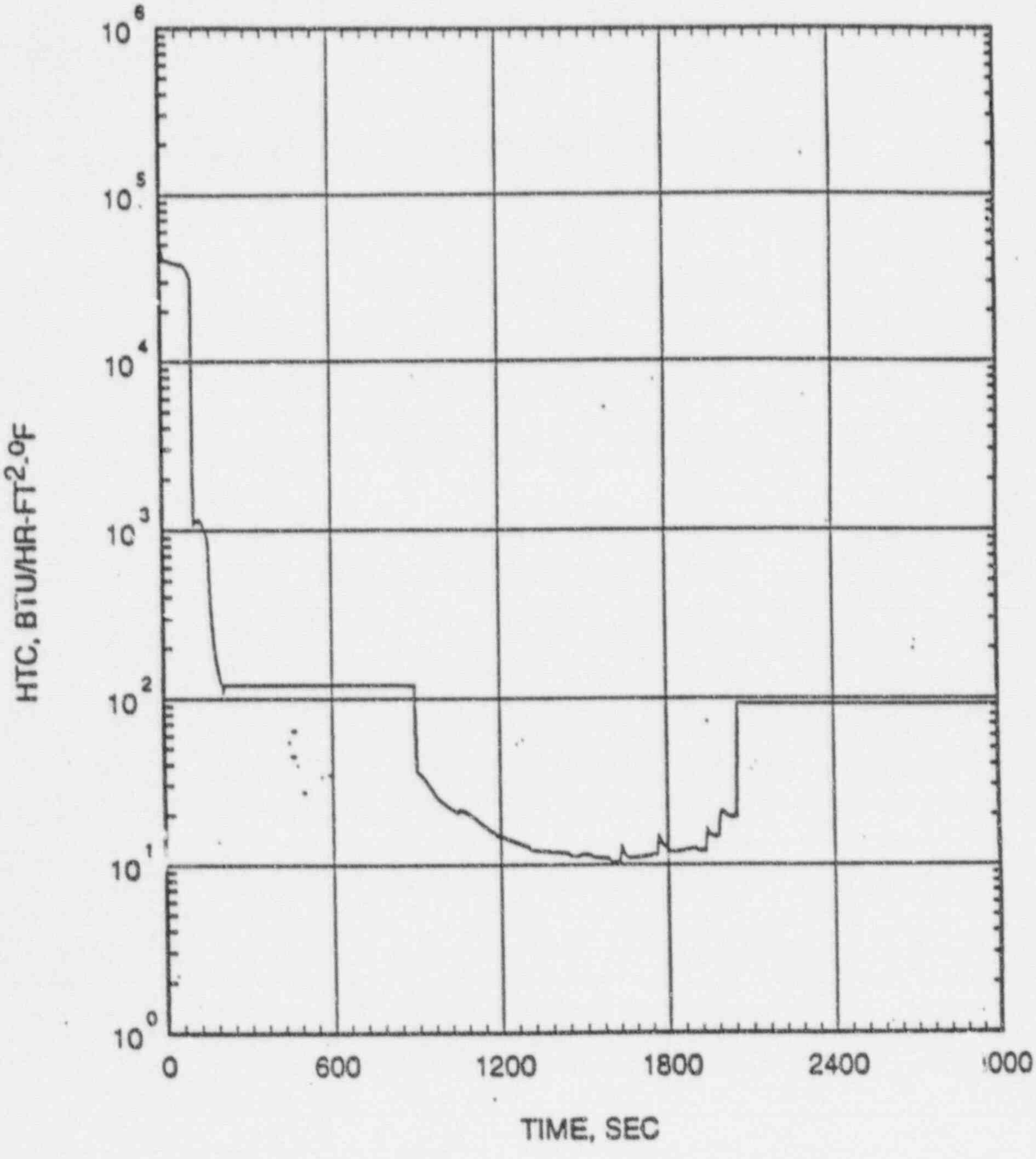


Figure 7g

0.05 FT^2 Break in Pump Discharge Leg

Coolant Temperature at Hot Spot vs. Time

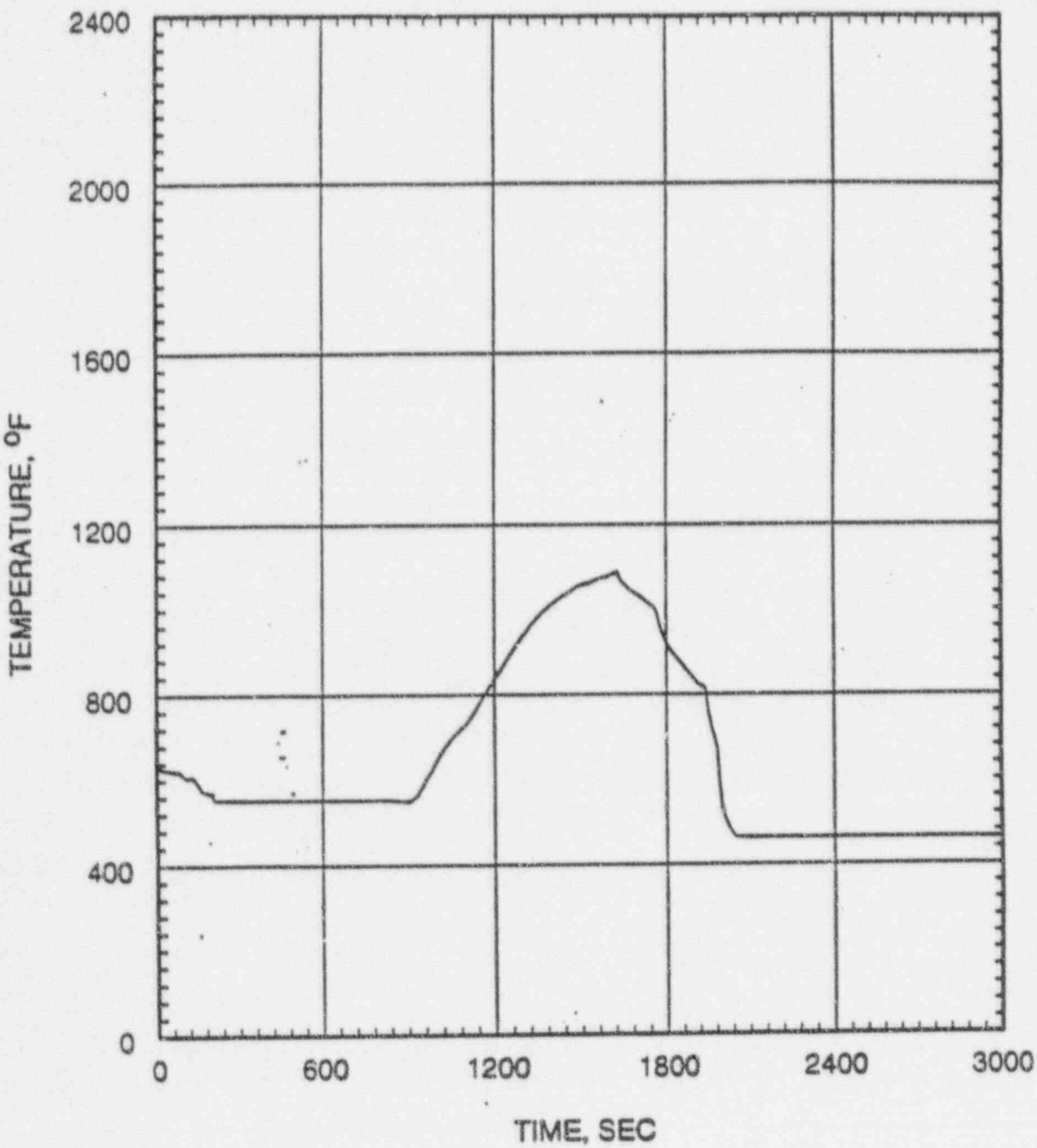


Figure 7h

0.05 FT^2 Break in Pump Discharge Leg

Cladding Temperature at Hot Spot vs. Time

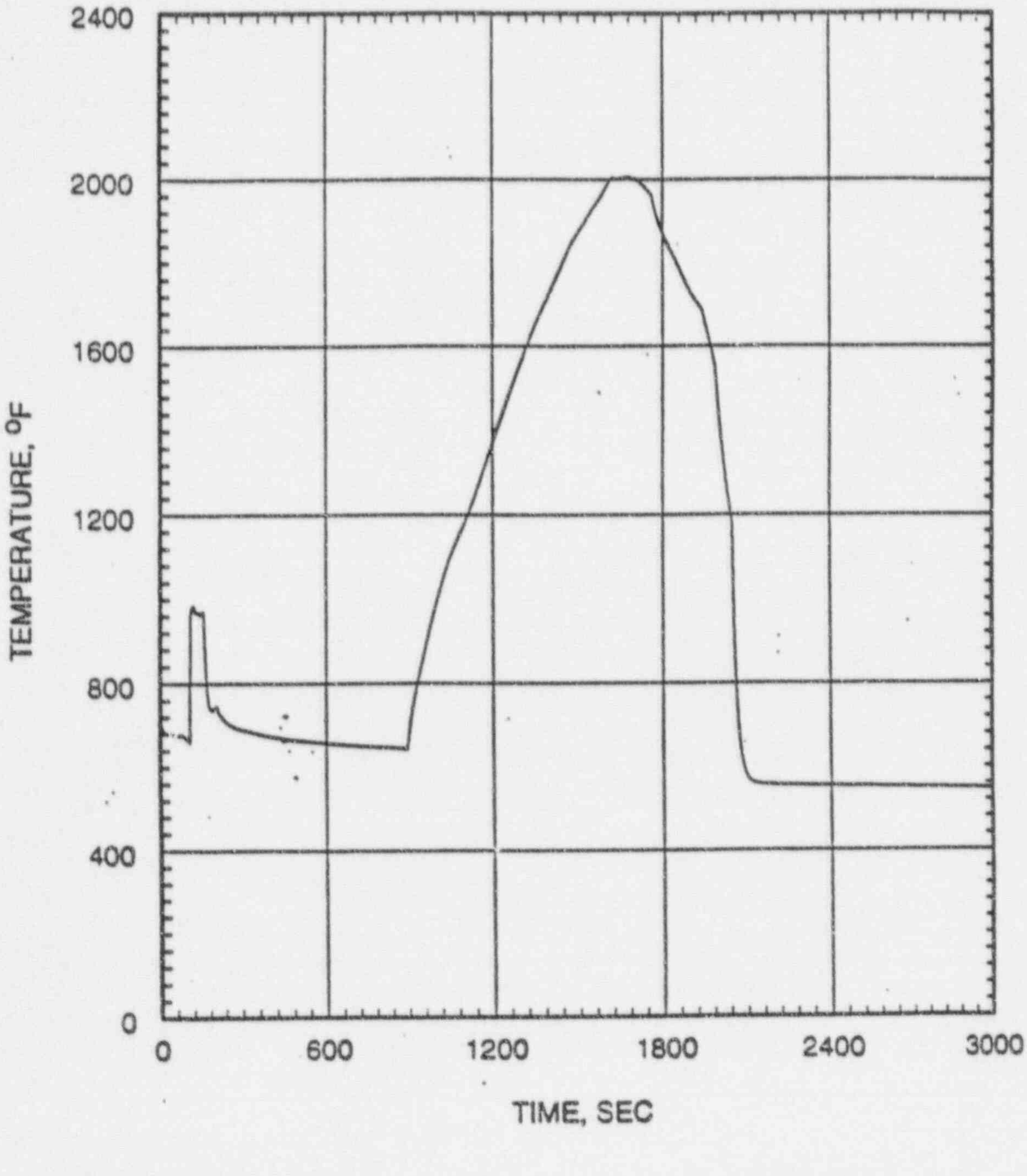


Figure 8a

0.04 FT^2 Break in Pump Discharge Leg

Normalized Core Power vs. Time

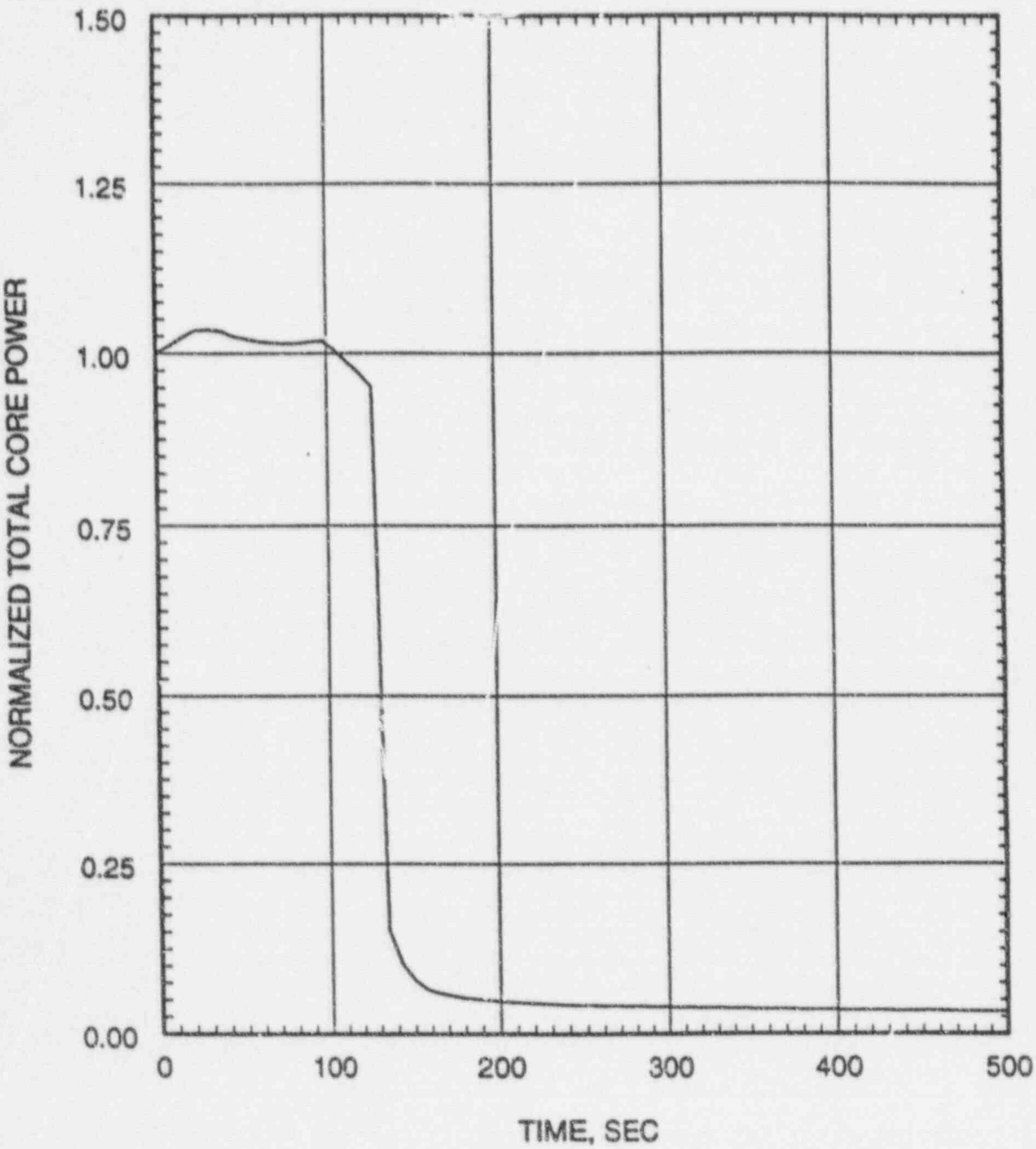


Figure 8b

0.04 FT^2 Break in Fump Discharge Leg

Inner Vessel Pressure vs. Time

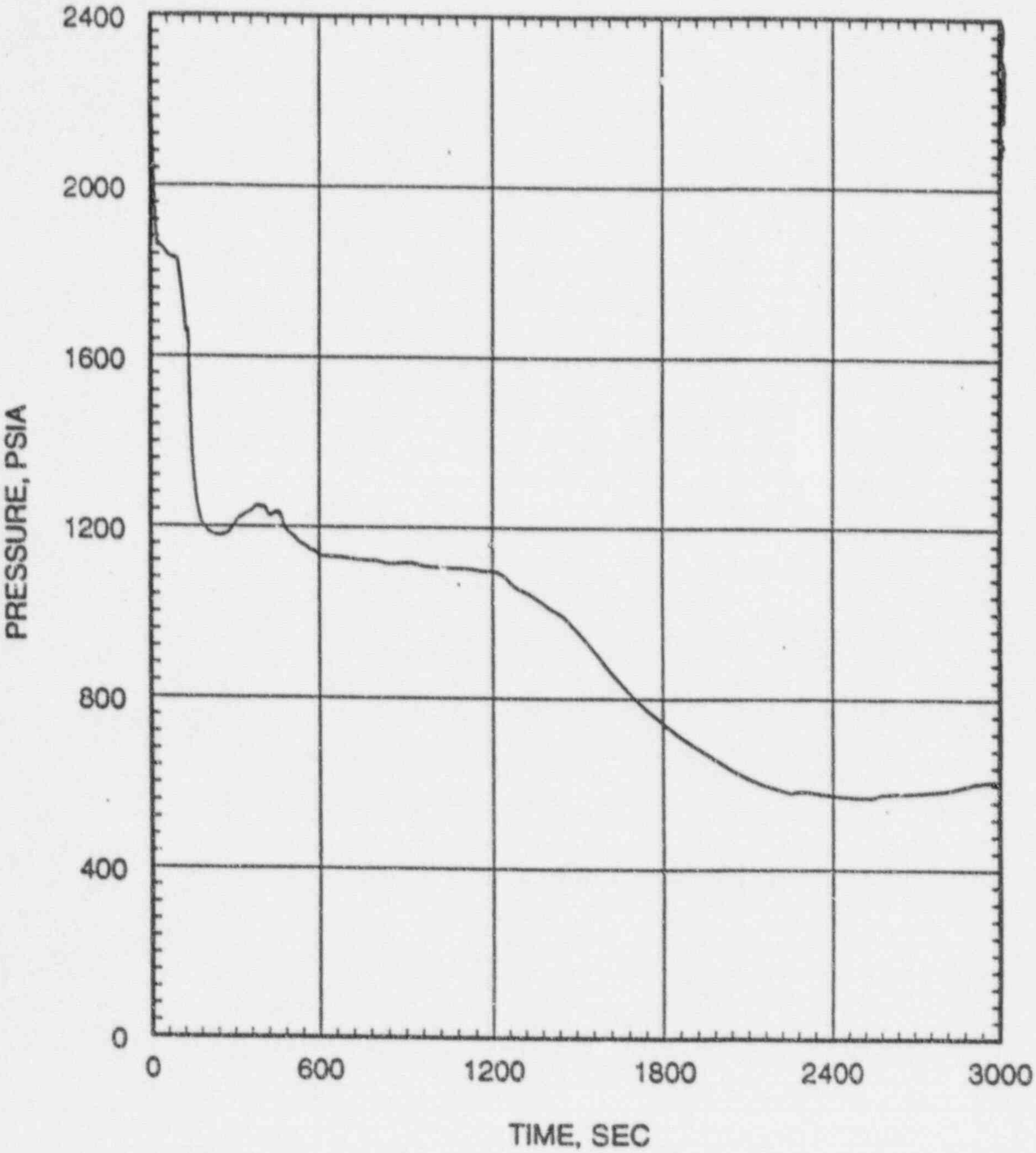


Figure 8c

0.04 FT^2 Break in Pump Discharge Leg

Break Flow Rate vs. Time

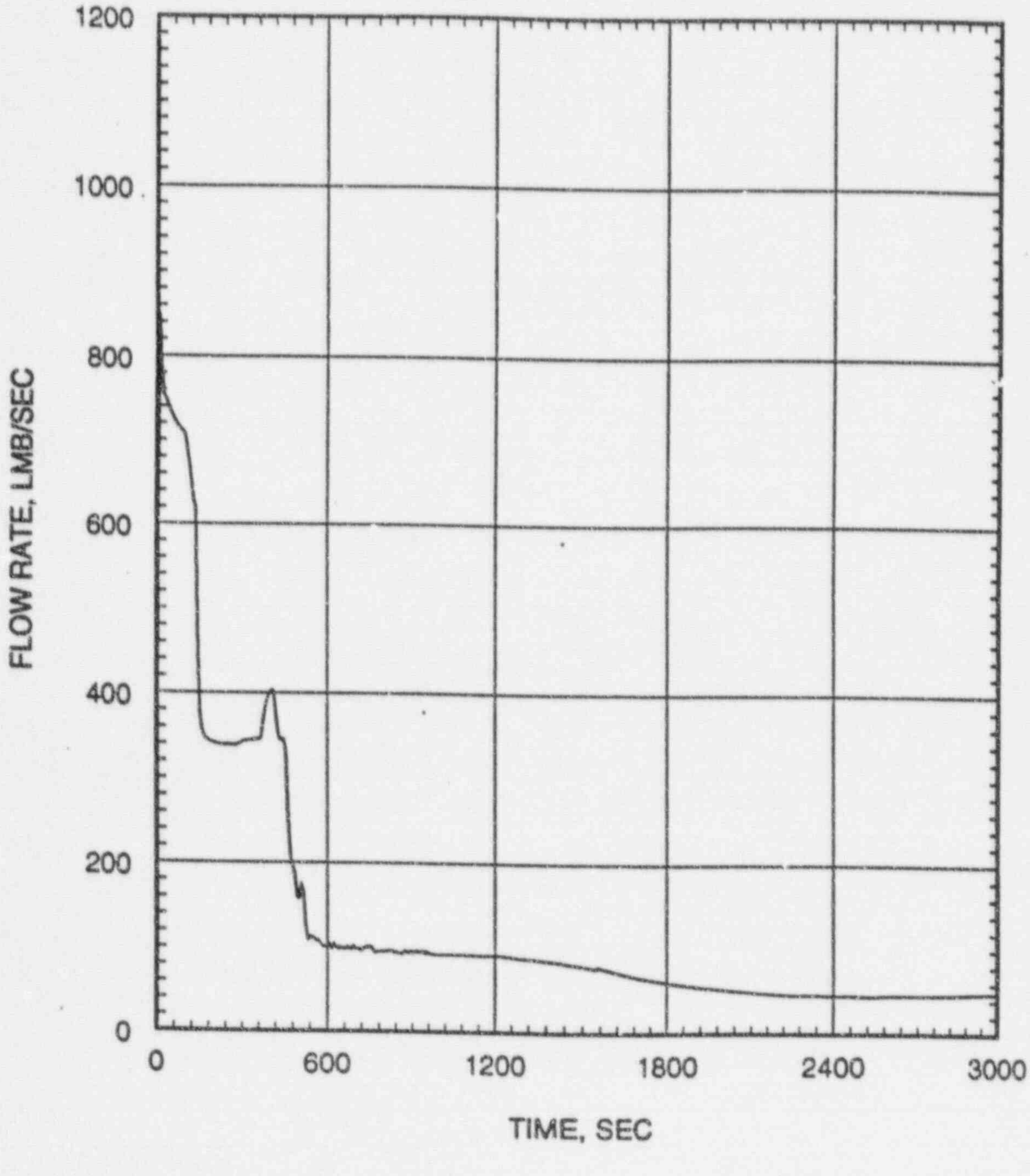


Figure 8d

0.04 FT^2 Break in Pump Discharge Leg

Inner Vessel Inlet Flow Rate vs. Time

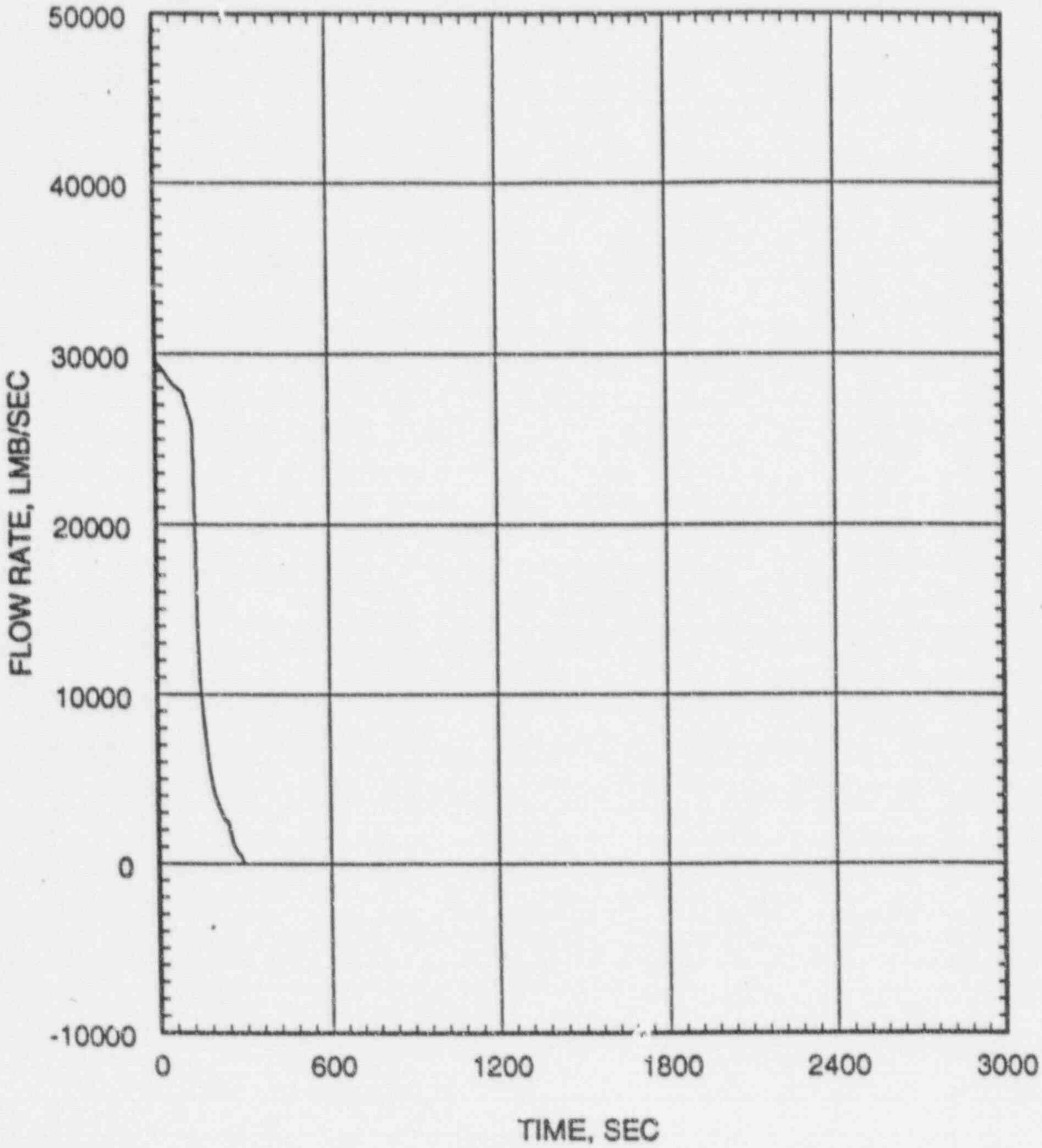


Figure 8e

0.04 FT^2 Break in Pump Discharge Leg

Inner Vessel Two-Phase Mixture Level vs. Time

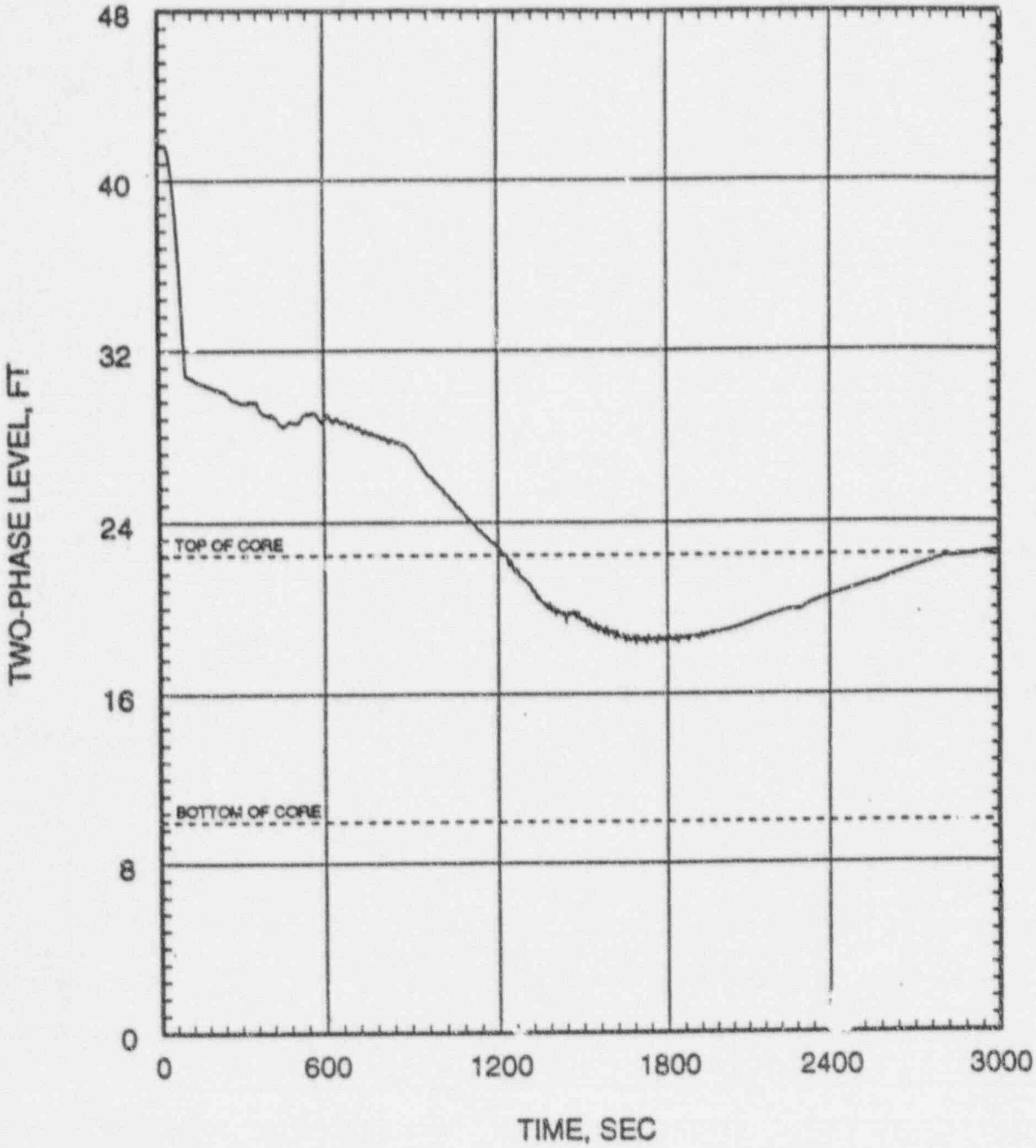


Figure 8f

0.04 FT^2 Break in Pump Discharge Leg

Heat Transfer Coefficient at Hot Spot vs. Time

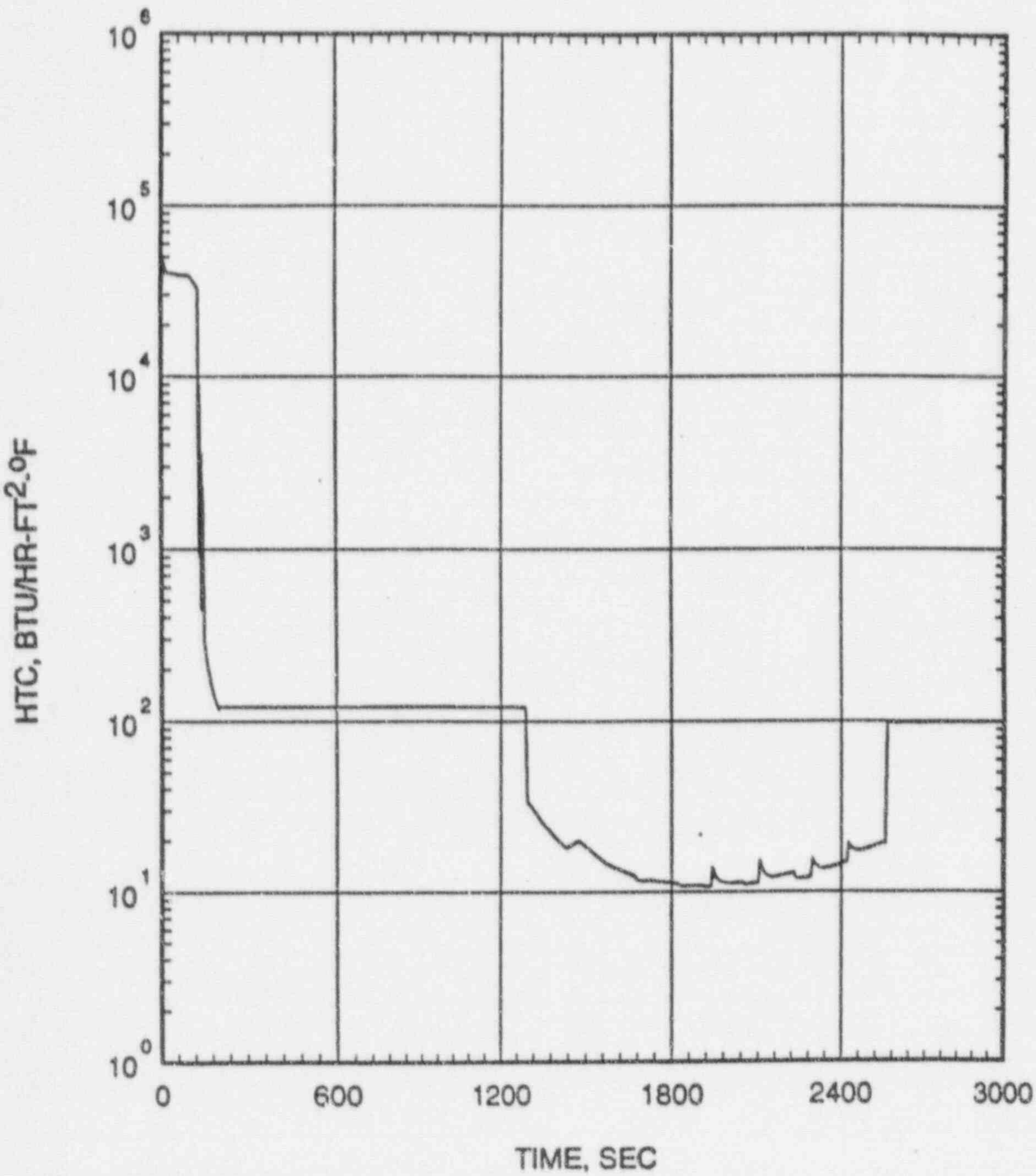


Figure 8g

0.04 FT^2 Break in Pump Discharge Leg

Coolant Temperature at Hot Spot vs. Time

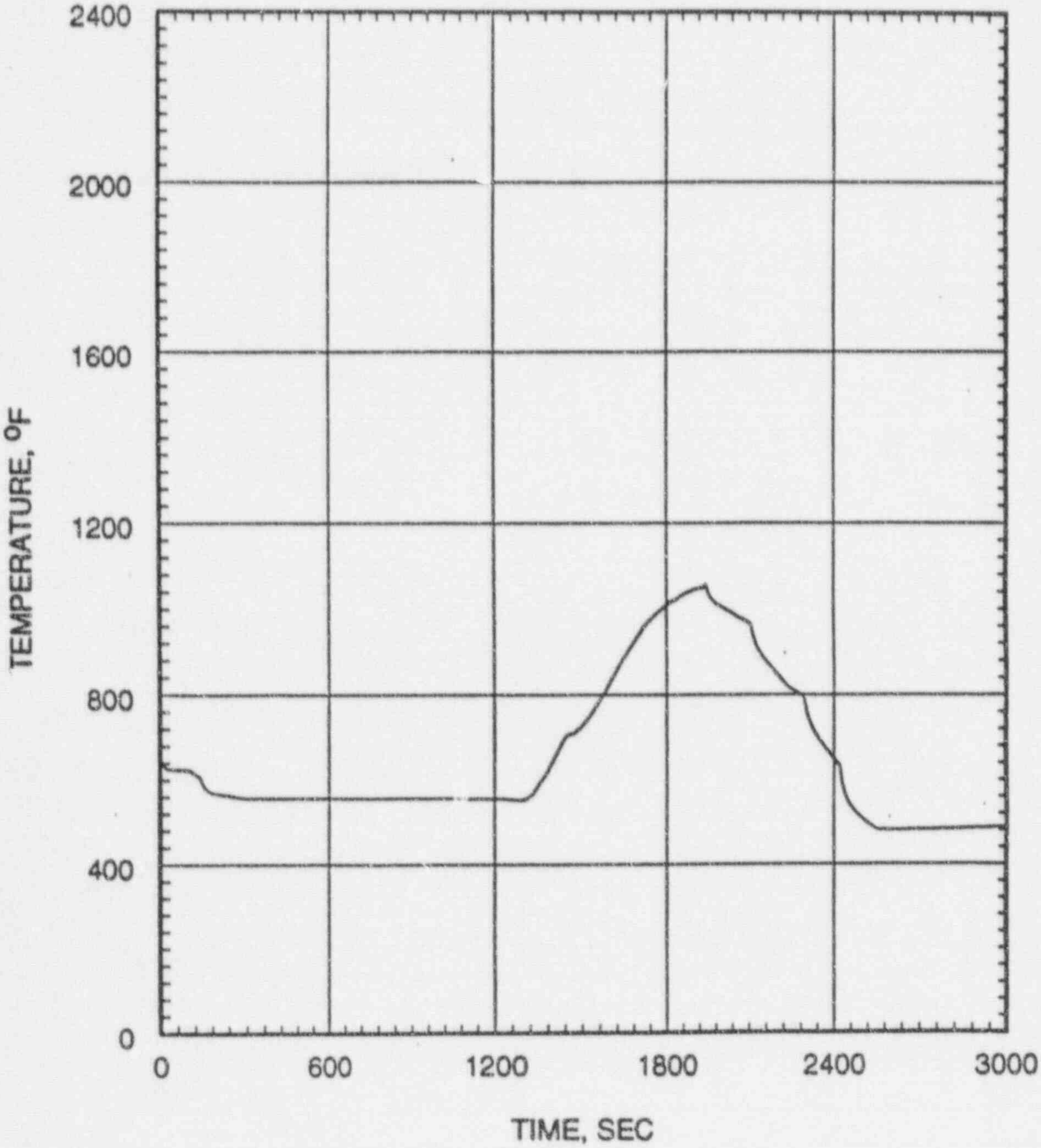


Figure 8h

0.04 FT^2 Break in Pump Discharge Leg

Cladding Temperature at Hot Spot vs. Time

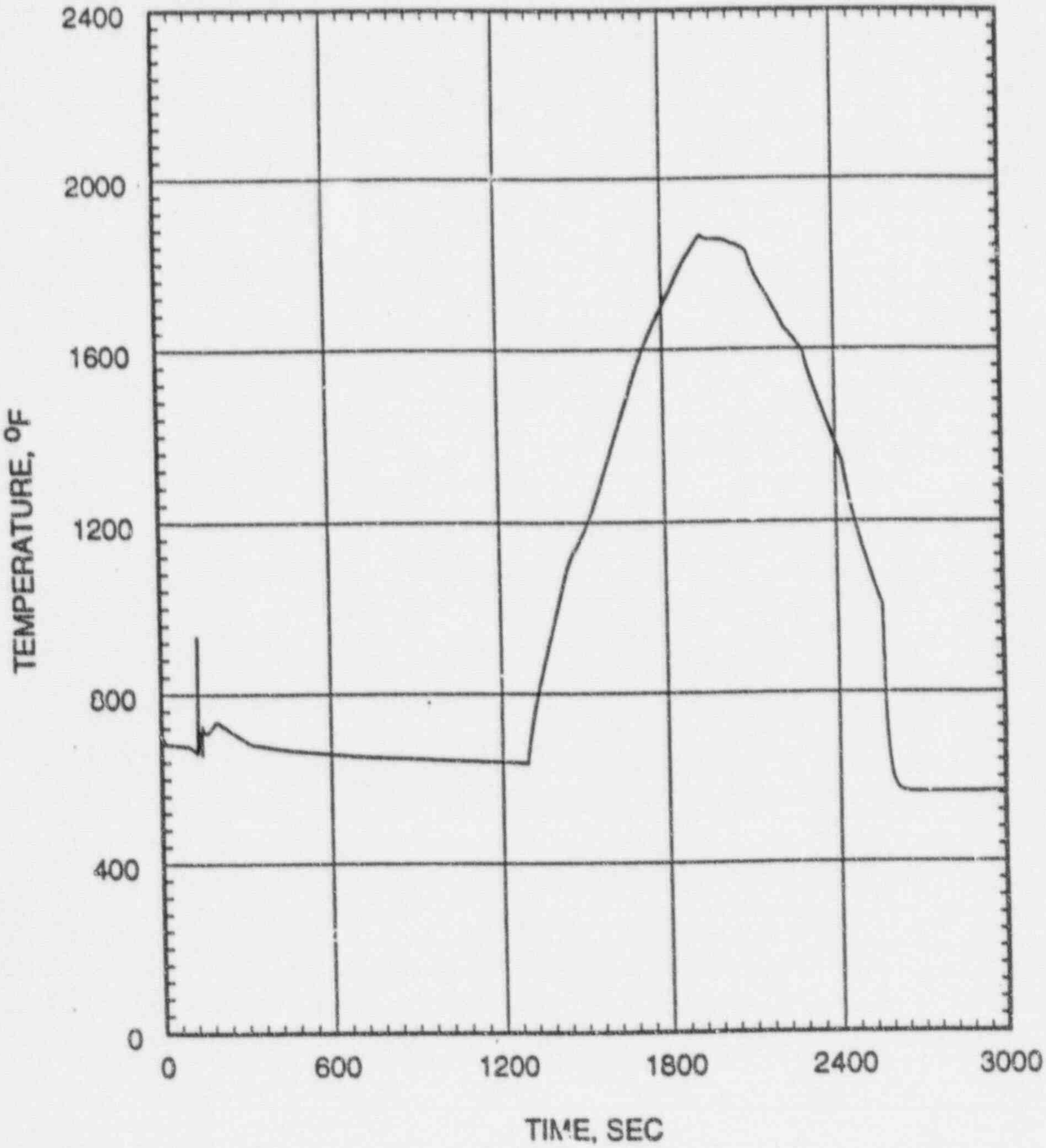


Figure 9a

0.02 FT^2 Break in Pump Discharge Leg

Normalized Core Power vs. Time

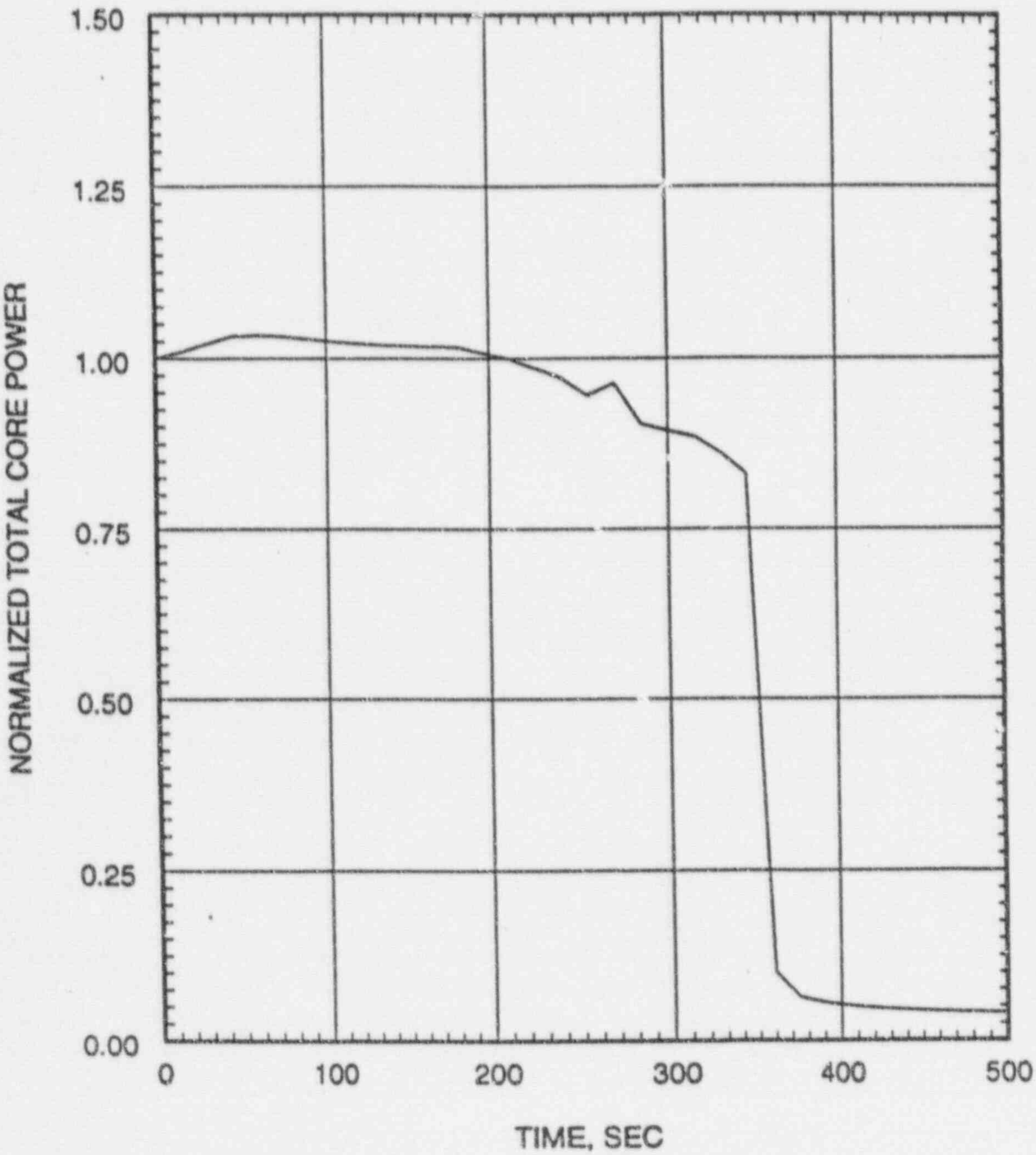


Figure 9b

0.02 FT^2 Break in Pump Discharge Leg

ınner Vessel Pressure vs. Time

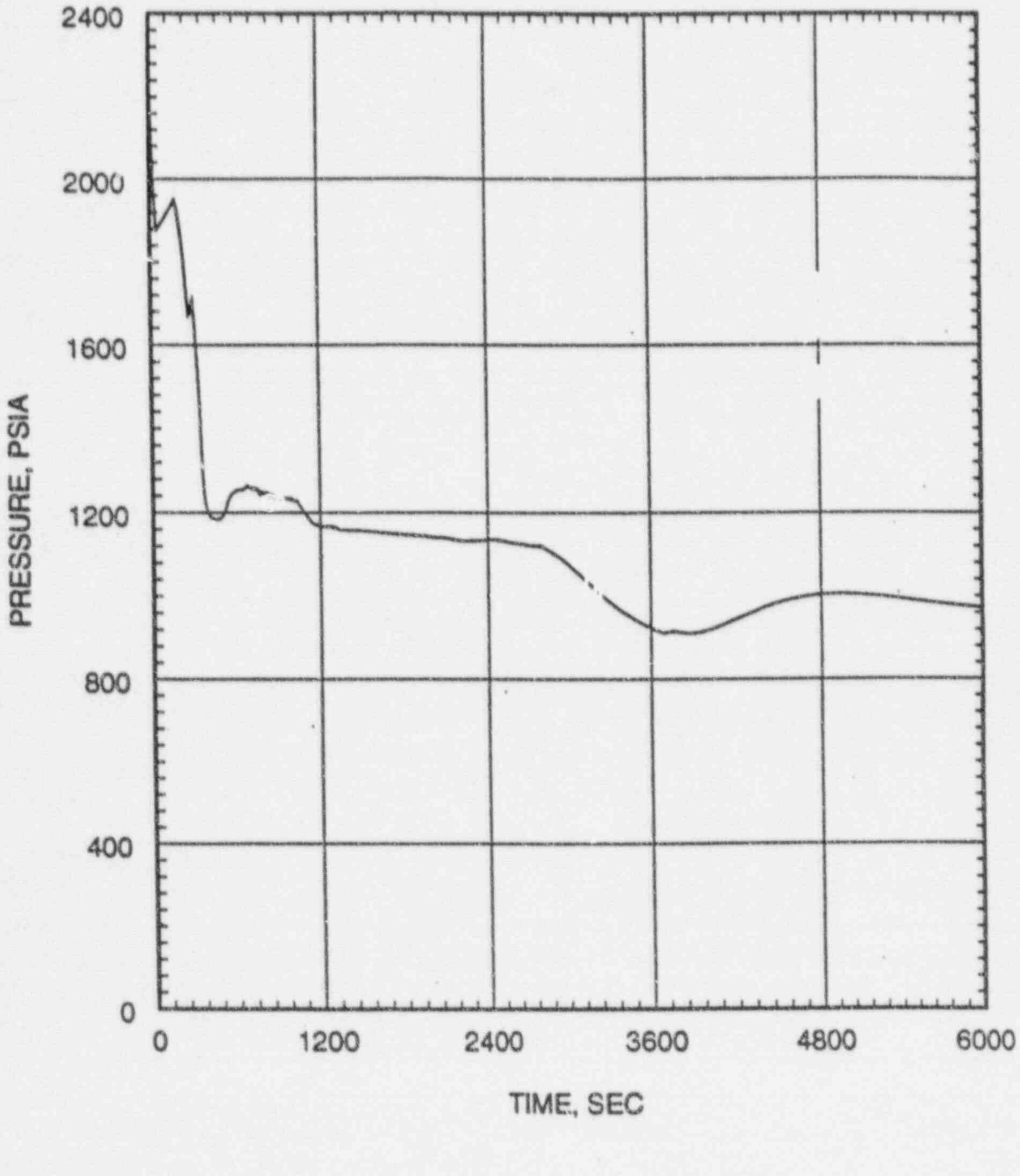


Figure 9c

0.02 FT^2 Break in Pump Discharge Leg

Break Flow Rate vs. Time

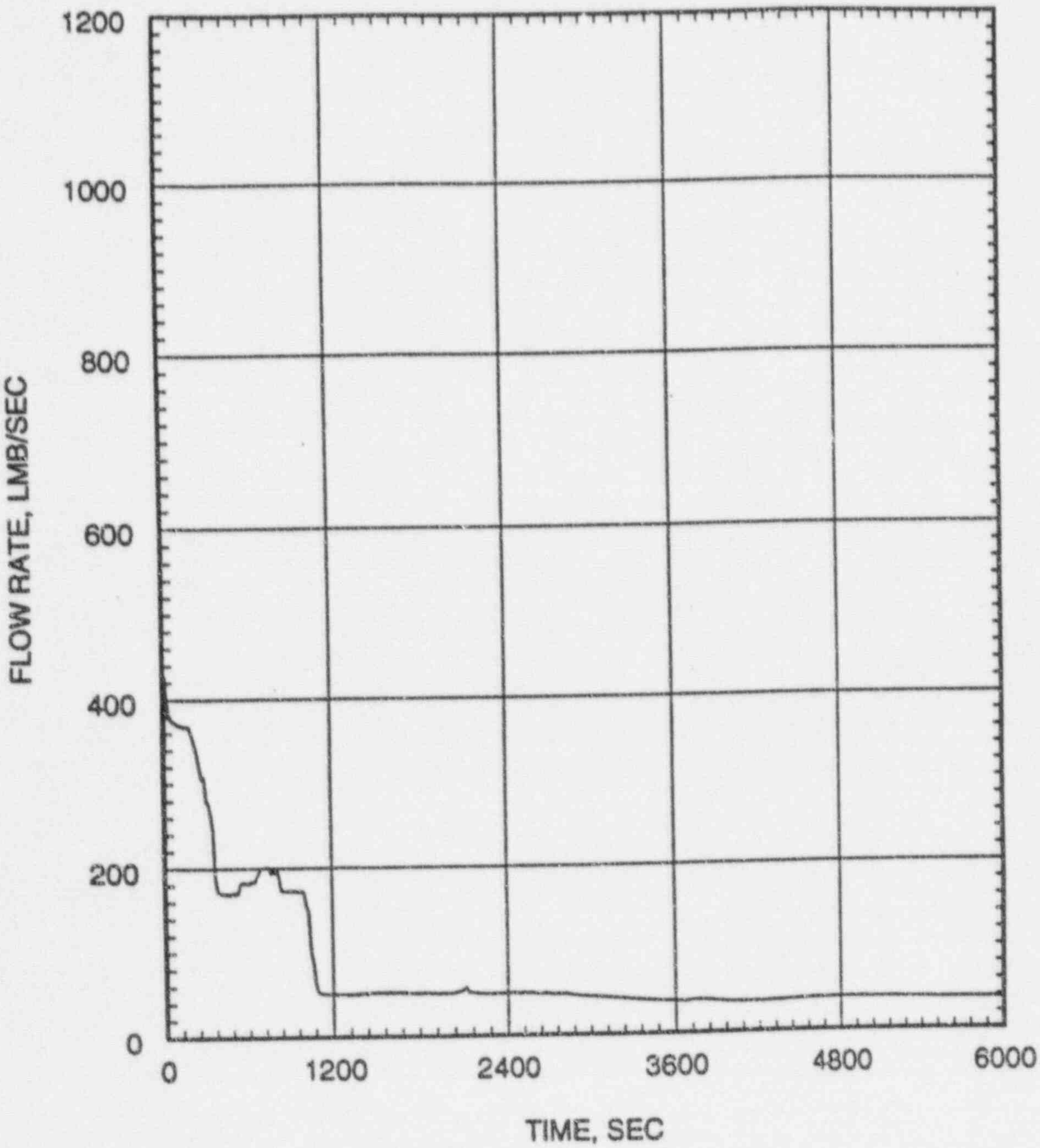


Figure 9d

0.02 FT² Break in Pump Discharge Leg

Inner Vessel Inlet Flow Rate vs. Time

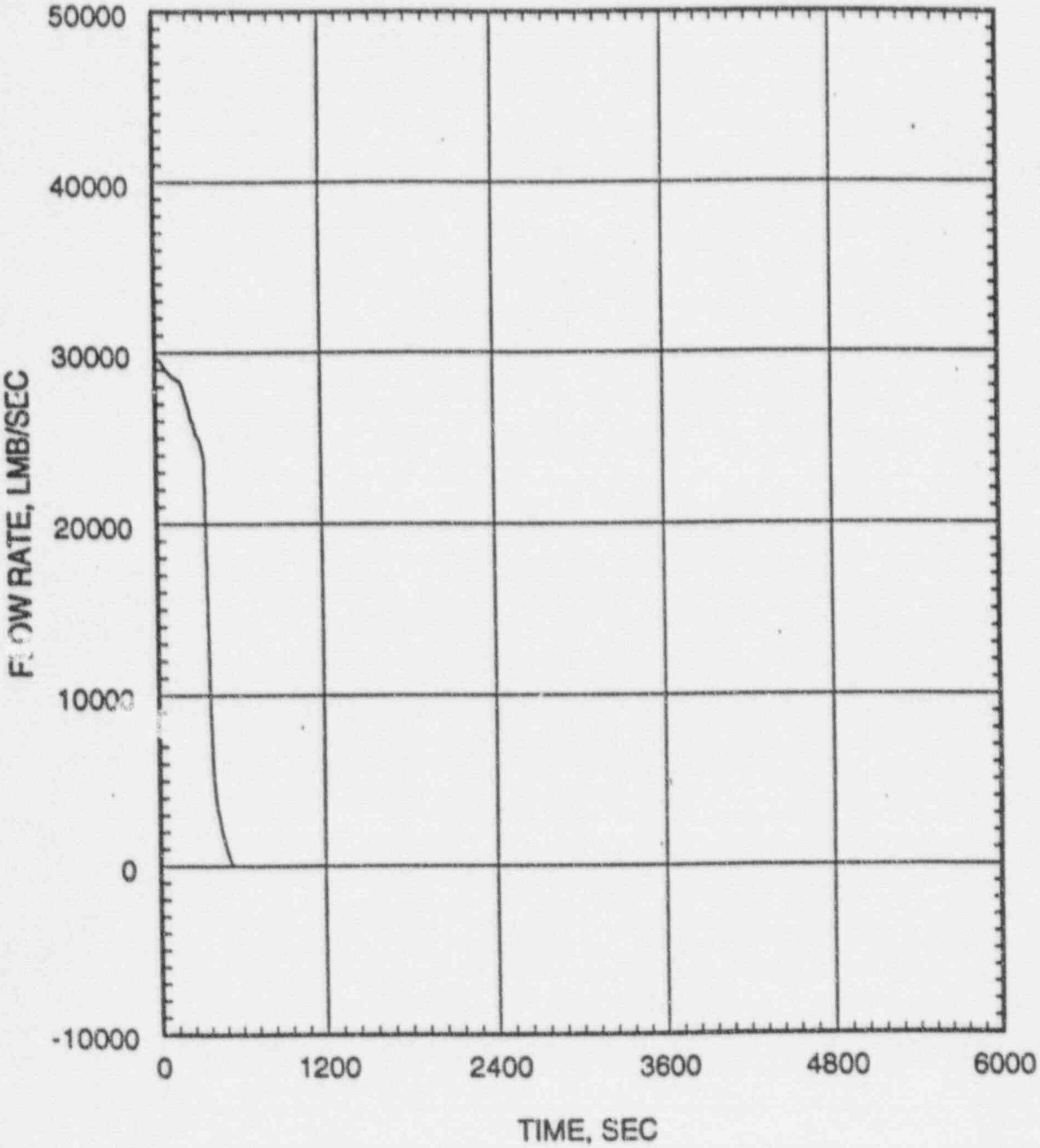


Figure 9e

0.02 FT^2 Break in Pump Discharge Leg

Inner Vessel Two-Phase Mixture Level vs. Time

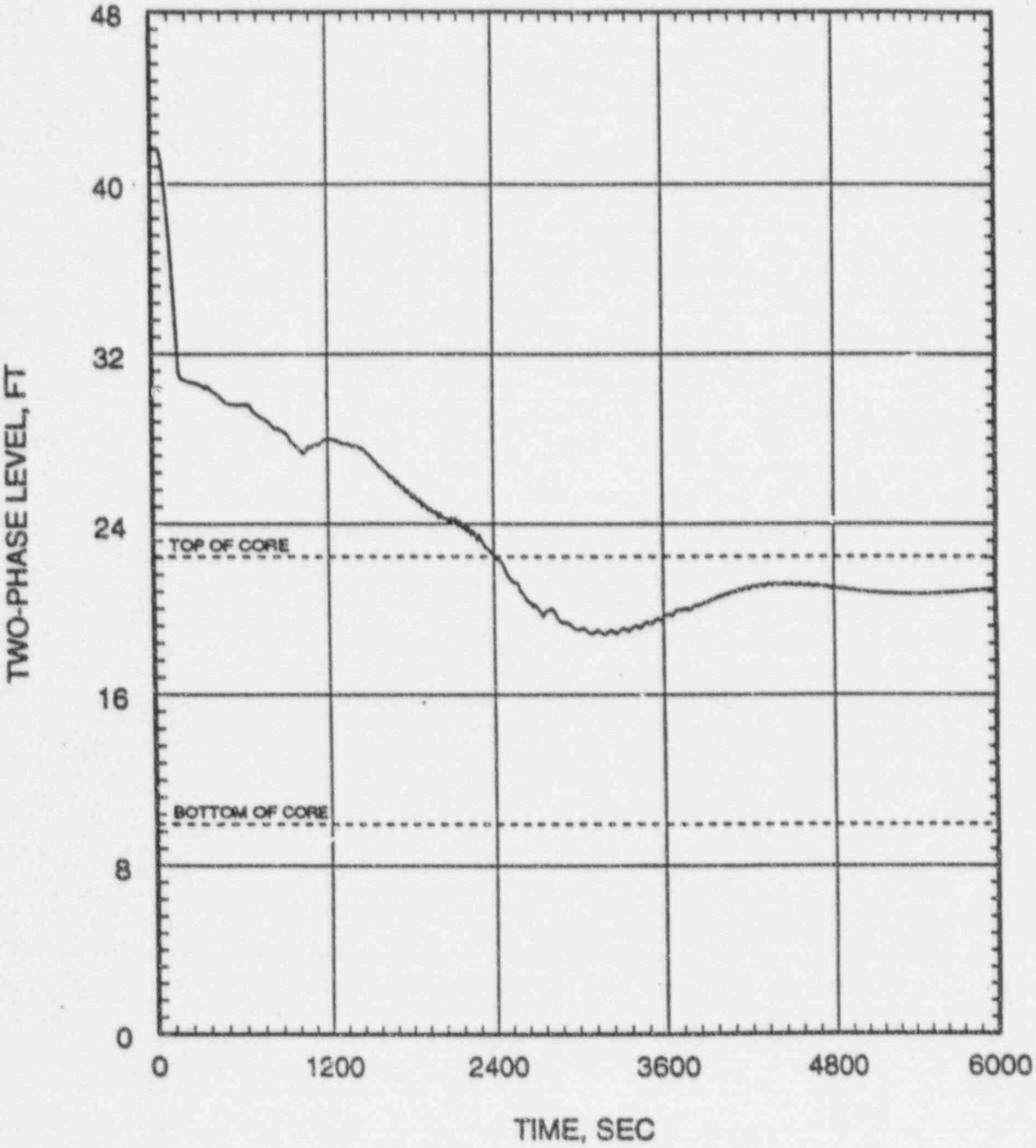


Figure 9f

0.02 FT^2 Break in Pump Discharge Leg

Heat Transfer Coefficient at Hot Spot vs. Time

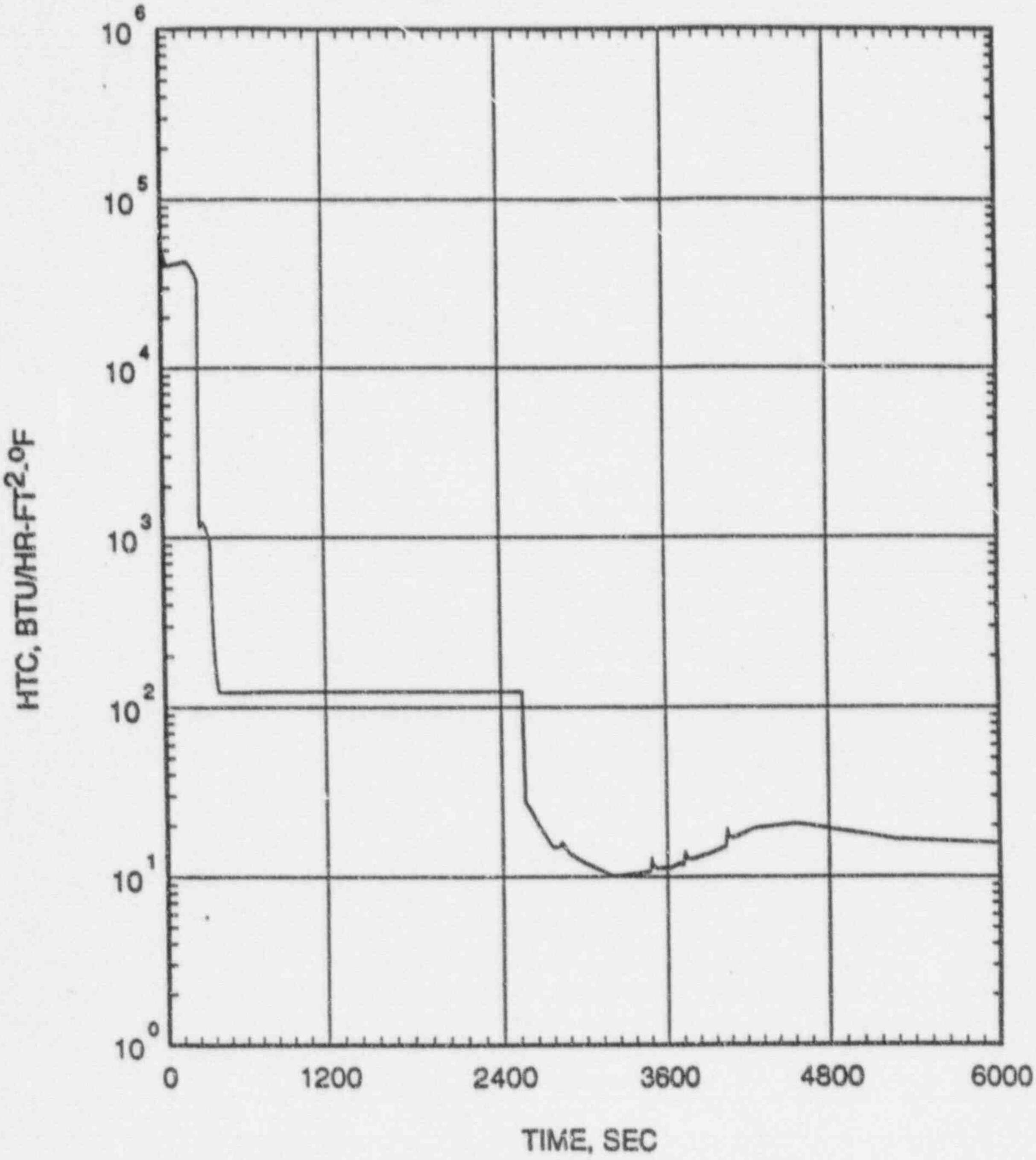


Figure 9g

0.02 FT^2 Break in Pump Discharge Leg

Coolant Temperature at Hot Spot vs. Time

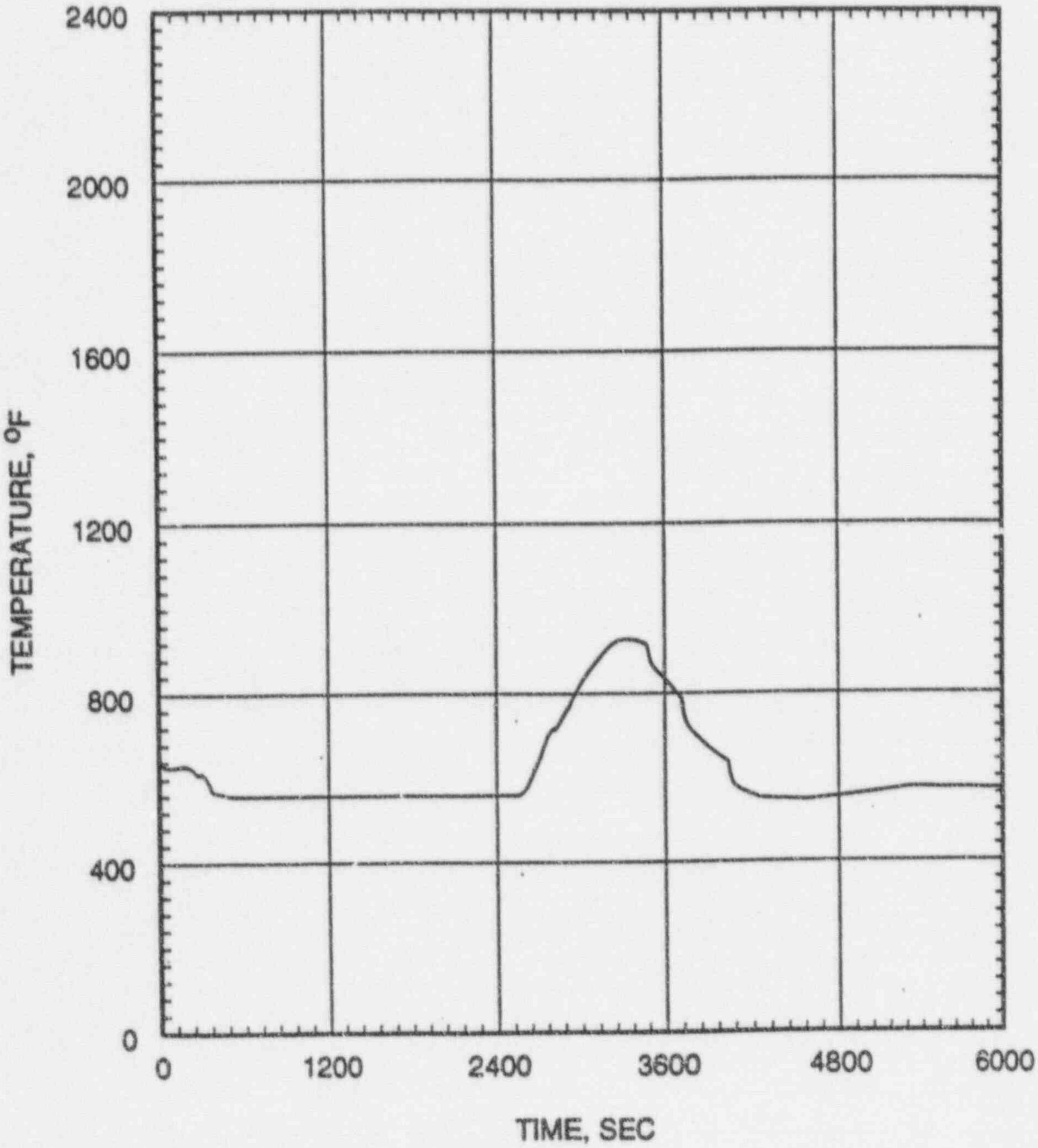


Figure 9h

0.02 FT^2 Break in Pump Discharge Leg

Cladding Temperature at Hot Spot vs. Time

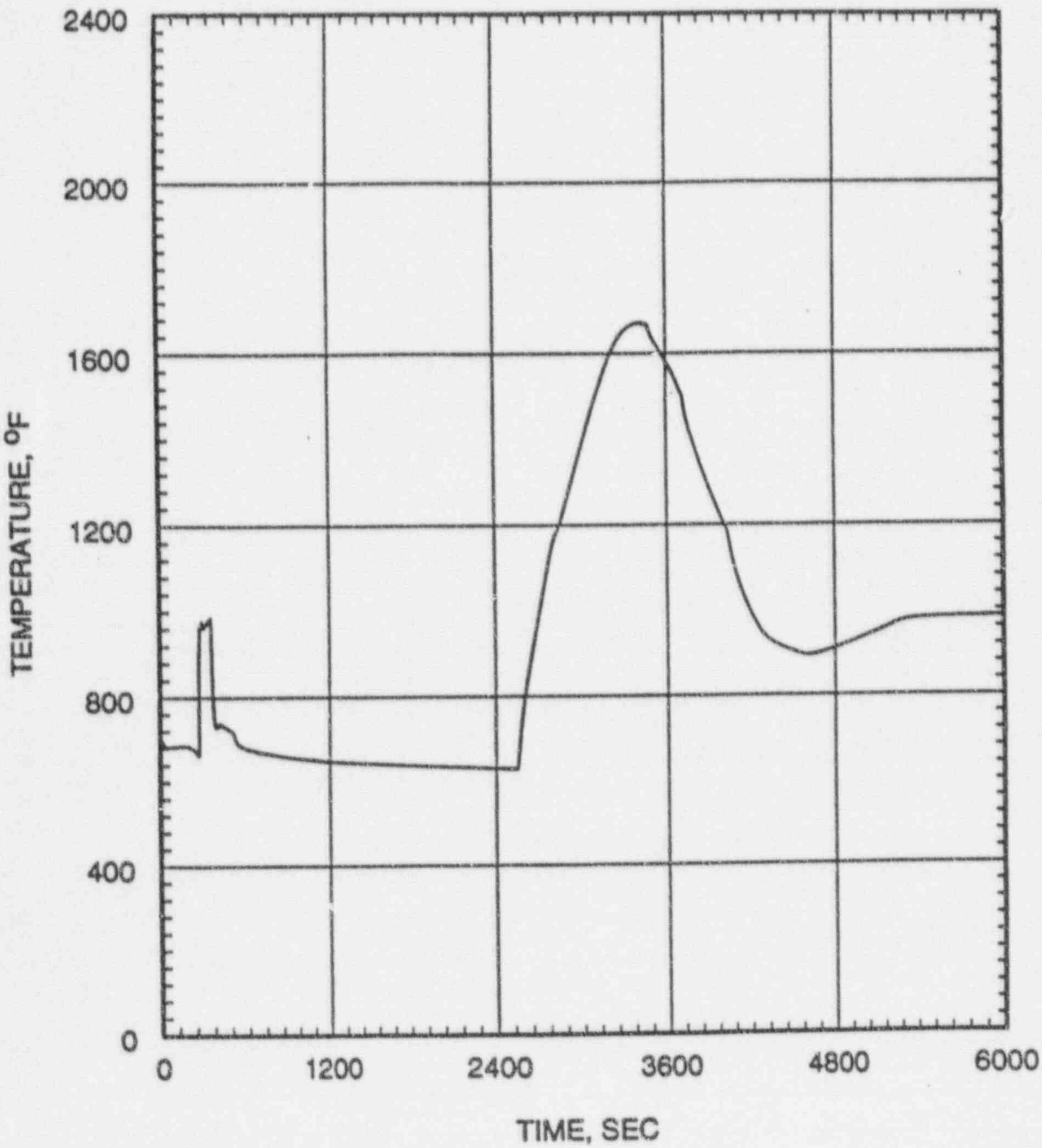


Figure 10

Peak Cladding Temperature vs. Break Size for SBLOCAs

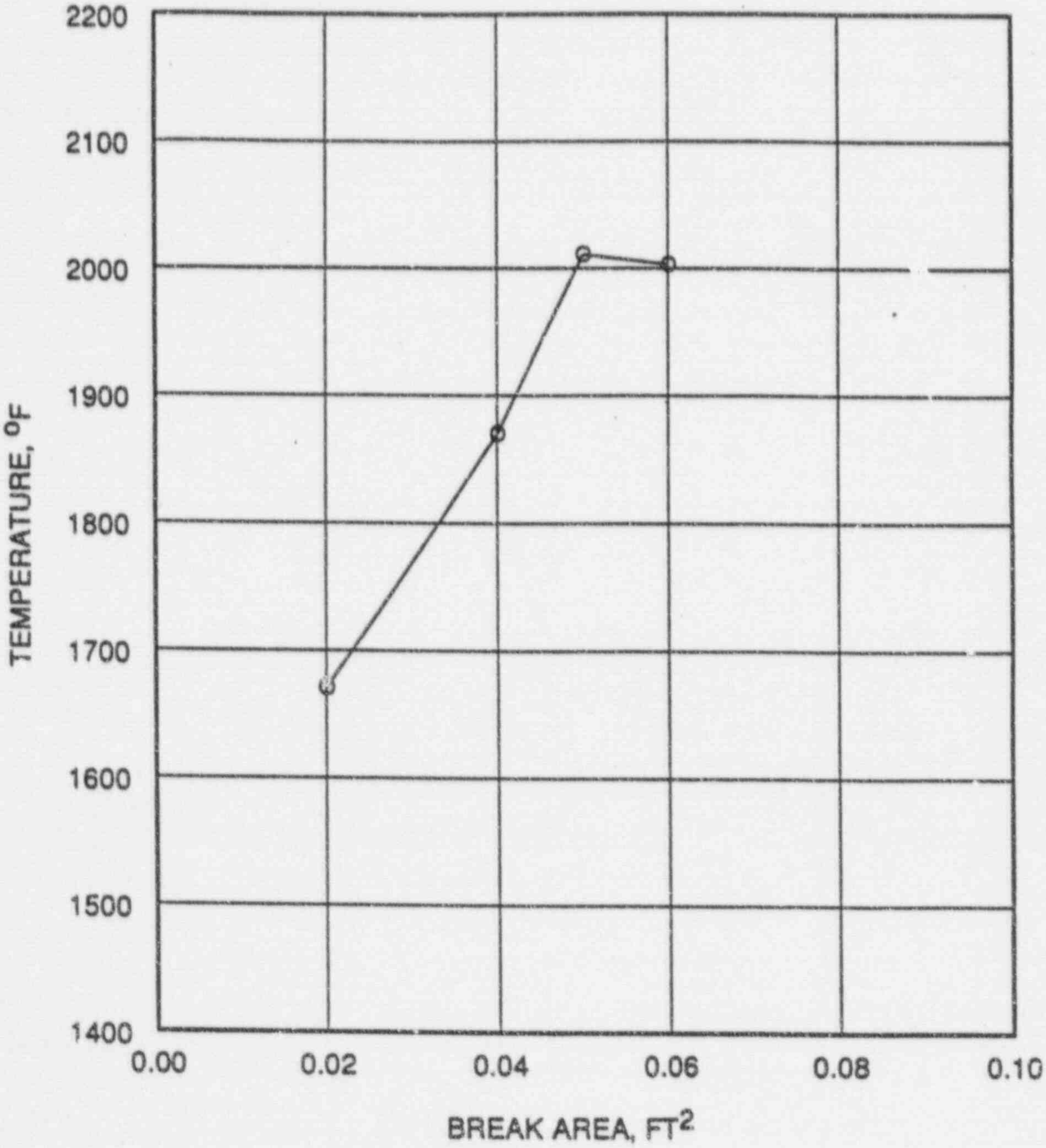


Figure 11

RCP FLOW COAST DOWN WITH 30% S/G TUBE PLUGGING

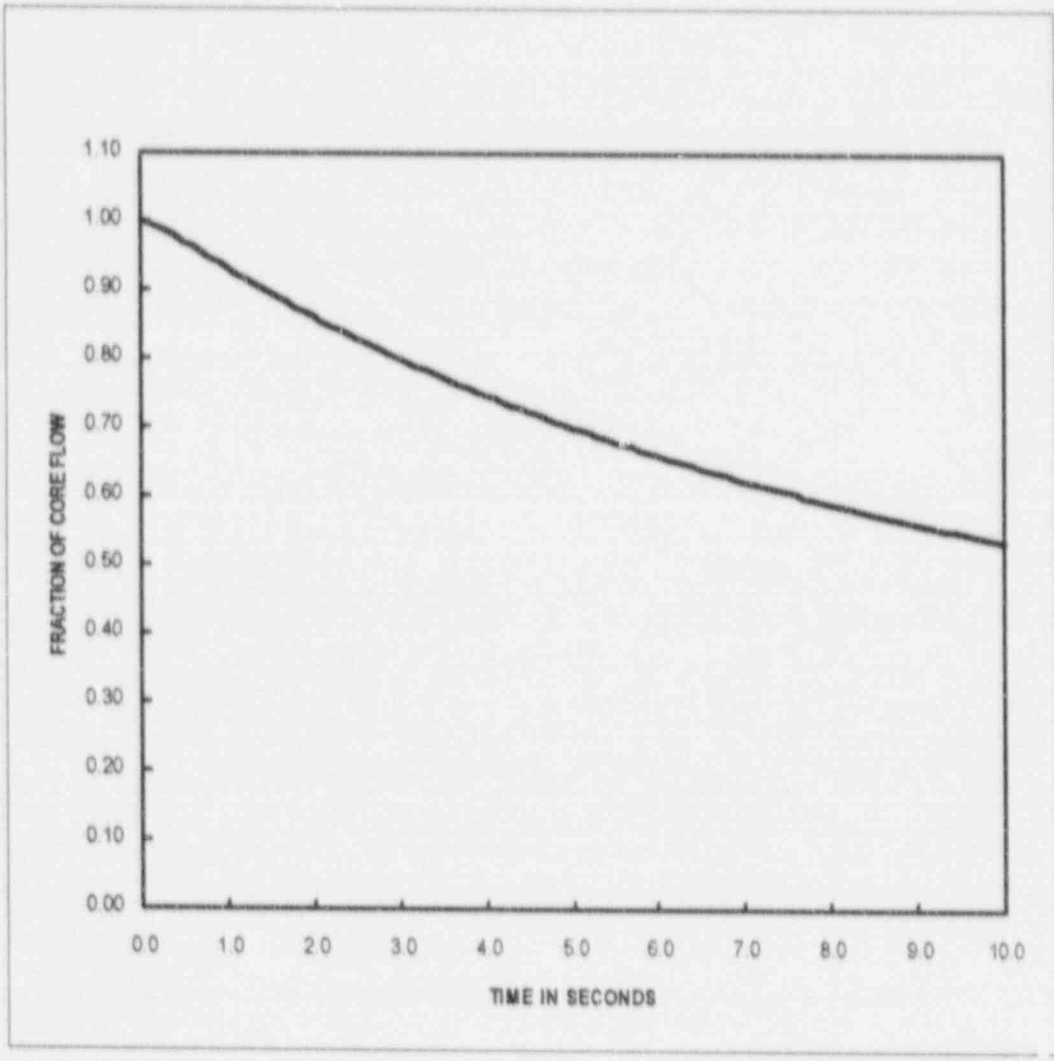


Figure 12

DNBR vs. Time

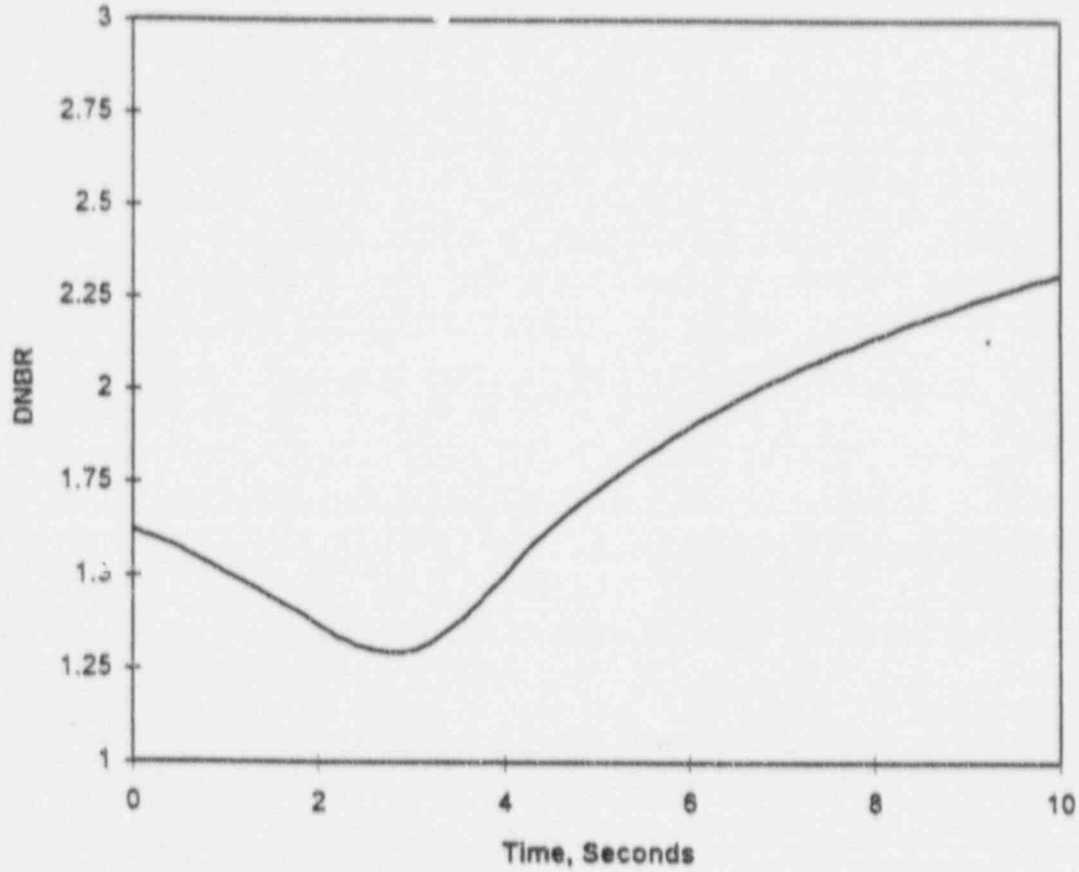


Figure 13

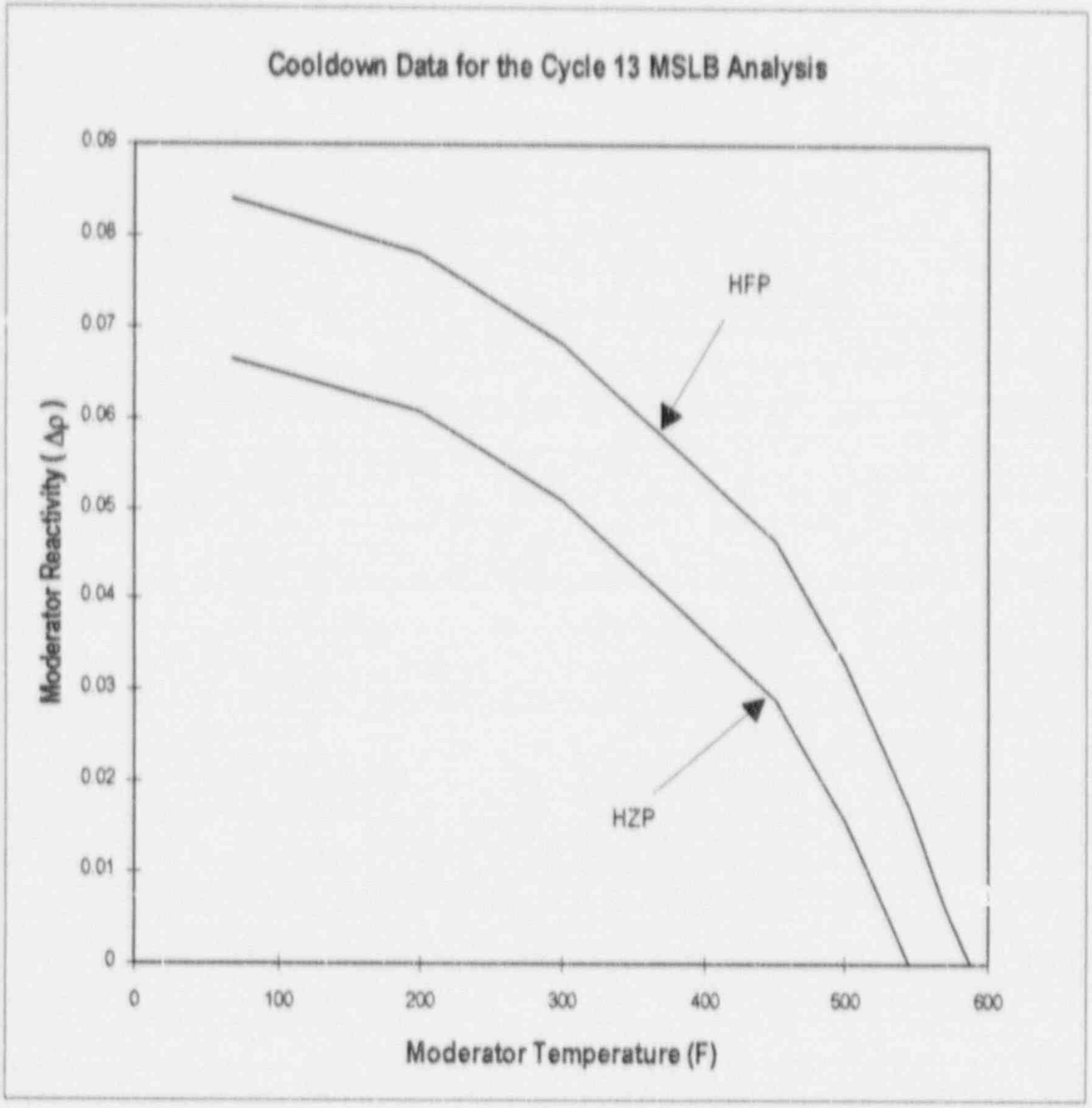

Cooldown Data for the Cycle 13 MSLB Analysis
Moderator Reactivity (Δρ)
0.09
0.08
0.07
0.06
0.05
0.04
0.03
0.02
0.01
0
HFP
HZP
0
100
200
300
400
500
600
Moderator Temperature (F)

Figure 14

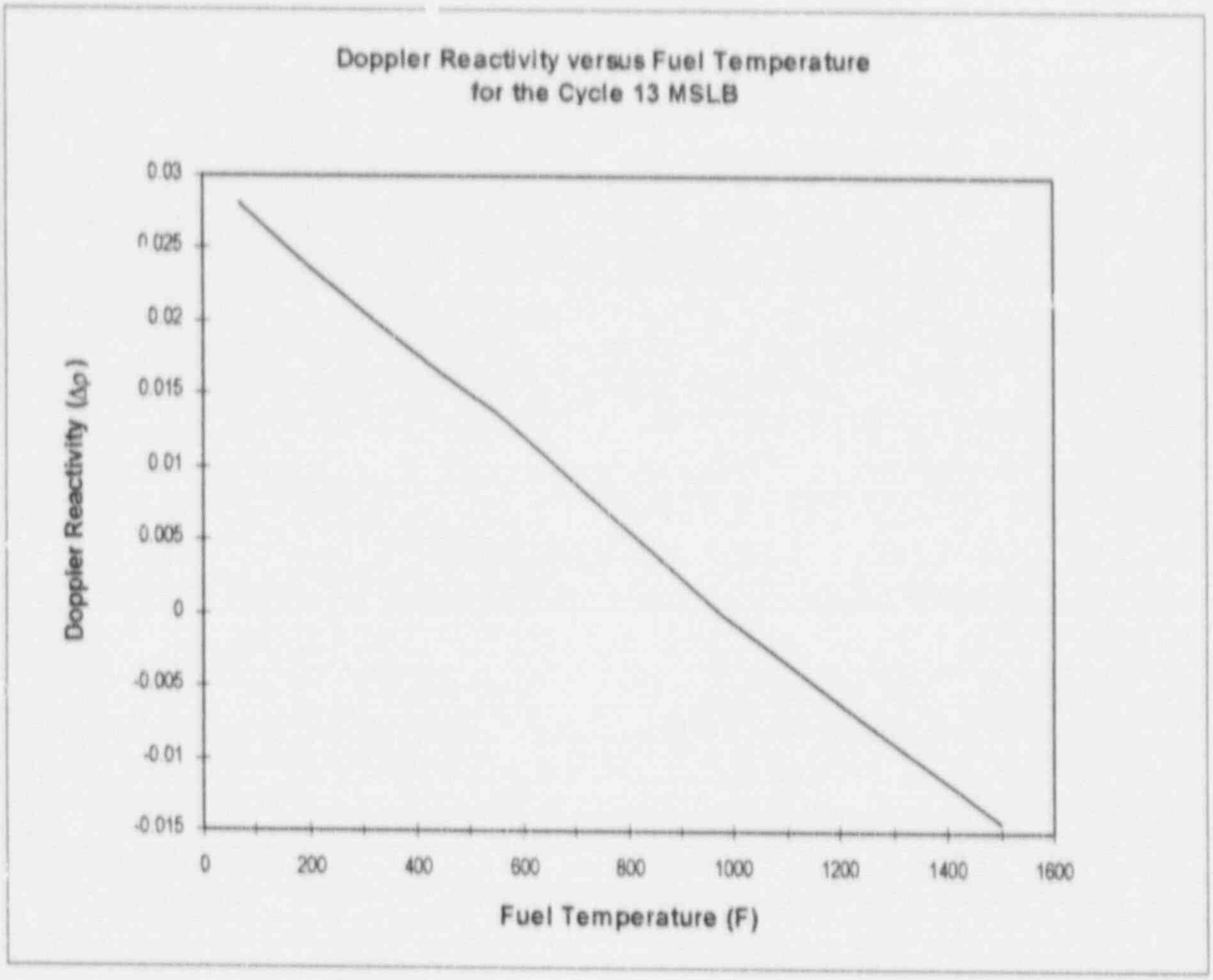

Doppler Reactivity versus Fuel Temperature
for the Cycle 13 MSLB
Doppler Reactivity (Δρ)
0.03
0.025
0.02
0.015
0.01
0.005
0
-0.005
-0.01
-0.015
0
200
400
600
800
1000
1200
1400
1600
Fuel Temperature (F)

Figure 15

SLB HFP Loss of AC 1 HPSI

Core Power vs. Time

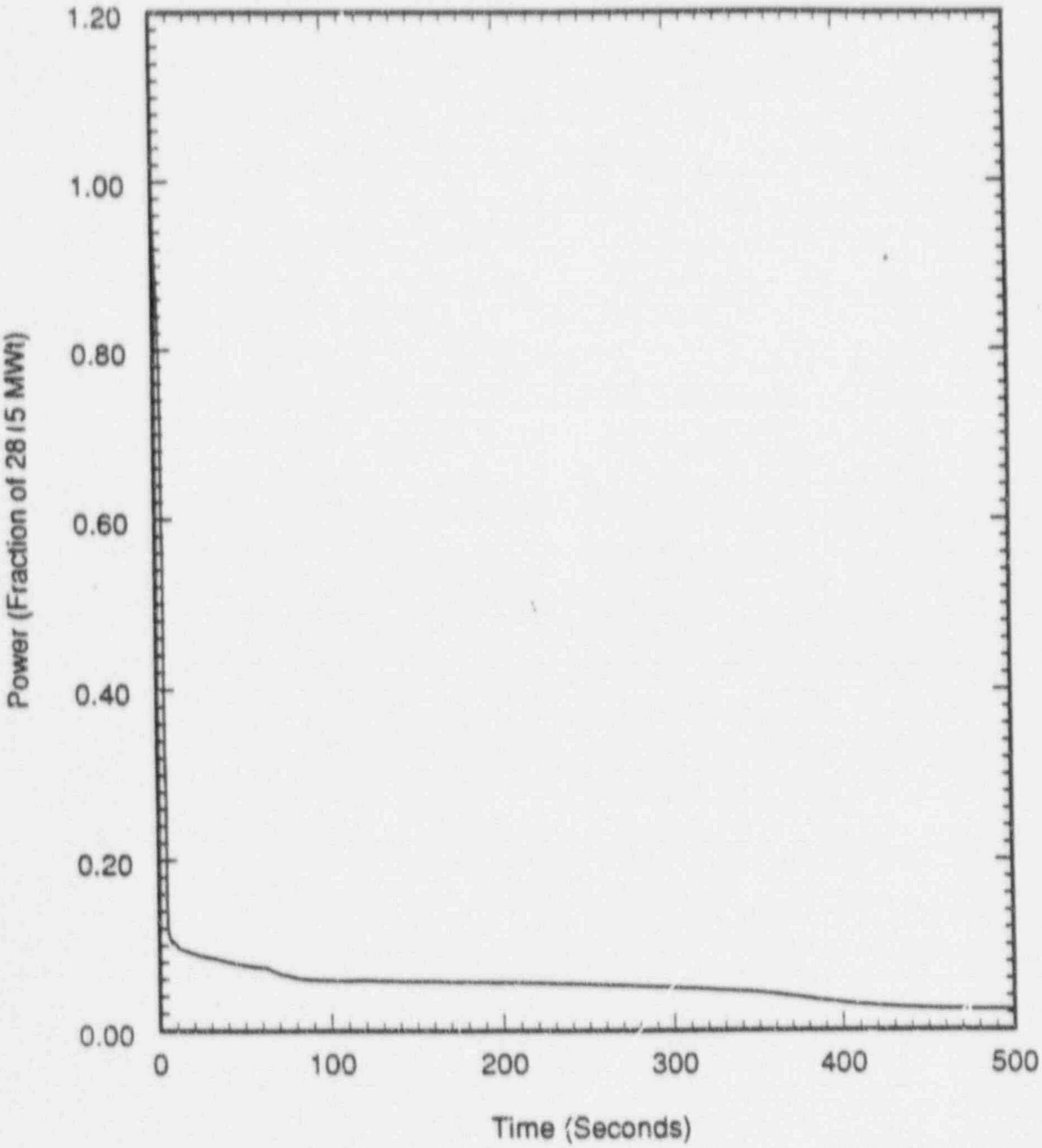


Figure 16

SLB HFP Loss of AC 1 HPSI

Heat Flux vs. Time

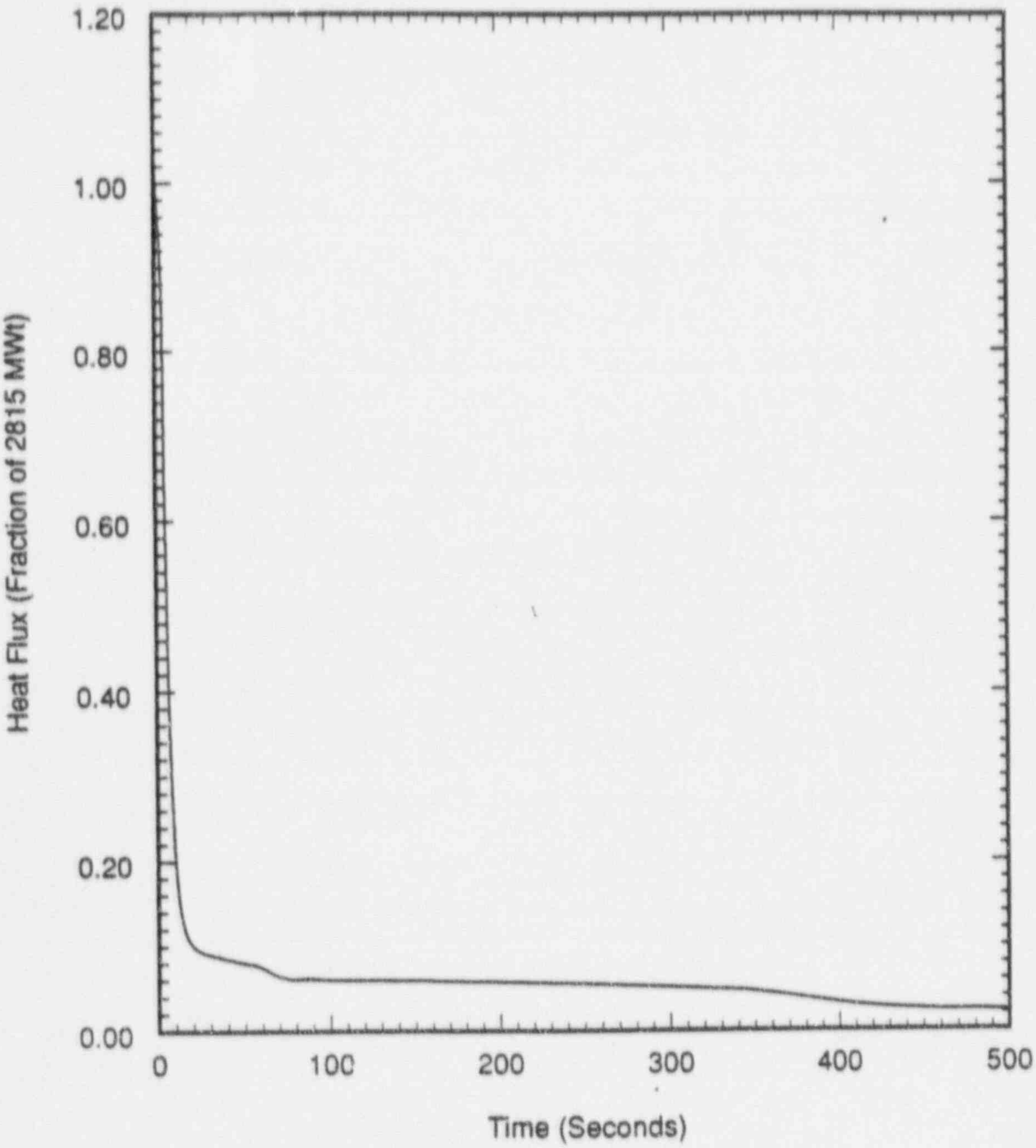


Figure 17

SLB HFP Loss of AC 1 HPSI

Pressurizer Pressure vs. Time

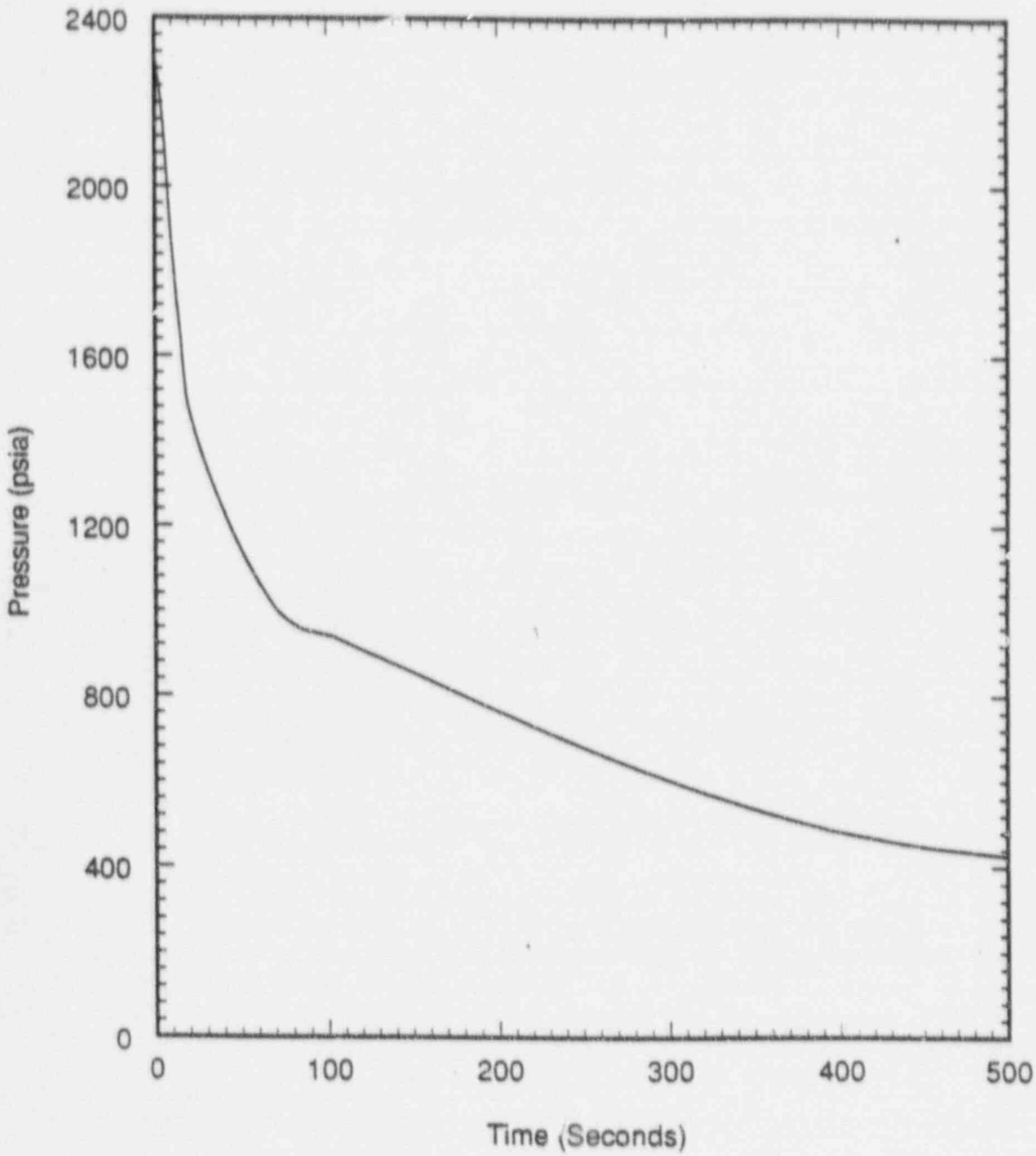


Figure 18

SLB HFP Loss of AC 1 HPSI

Steam Generator Pressure vs. Time

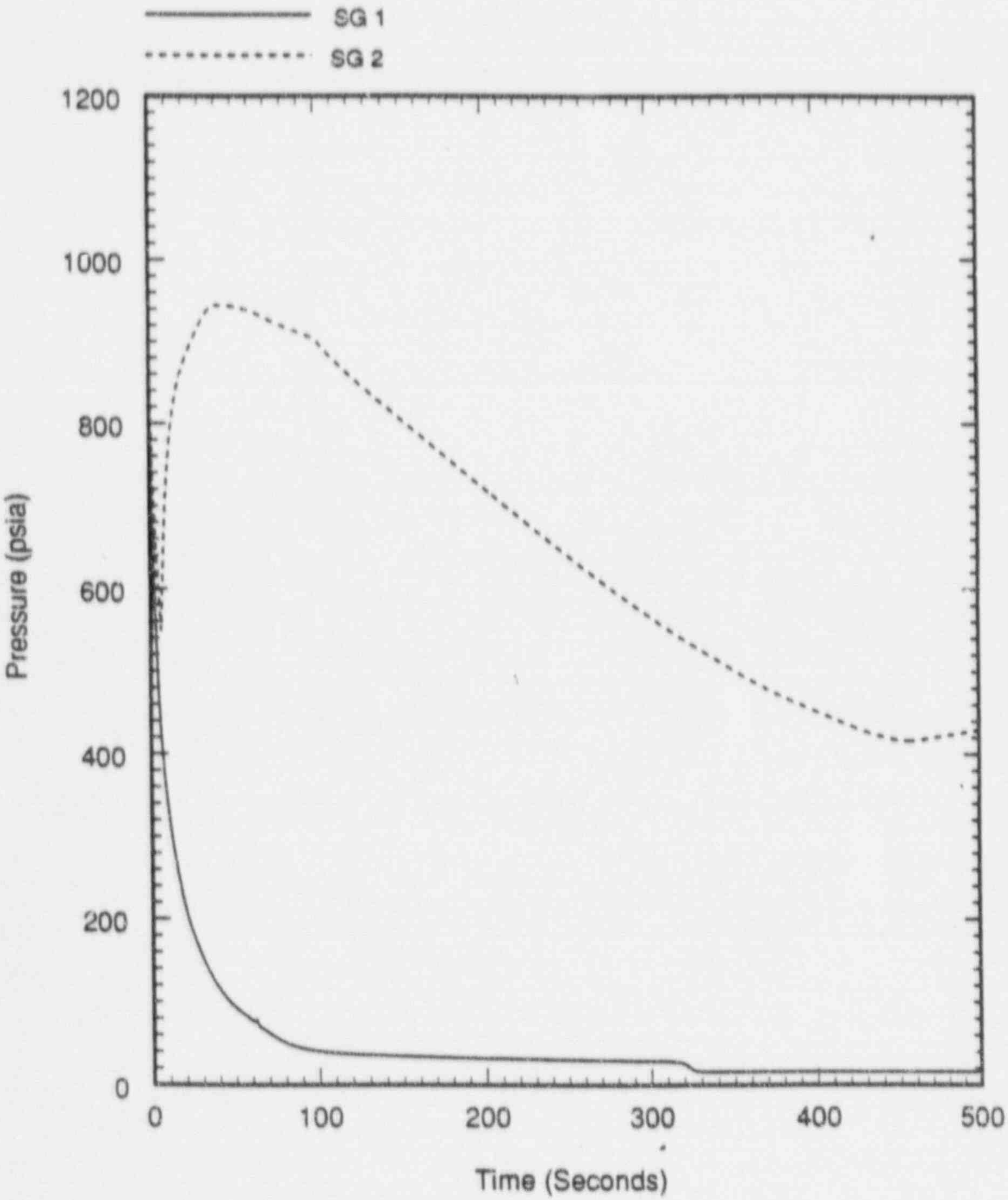


Figure 19

SLB HFP Loss of AC 1 HPSI

RCS Temperatures vs. Time

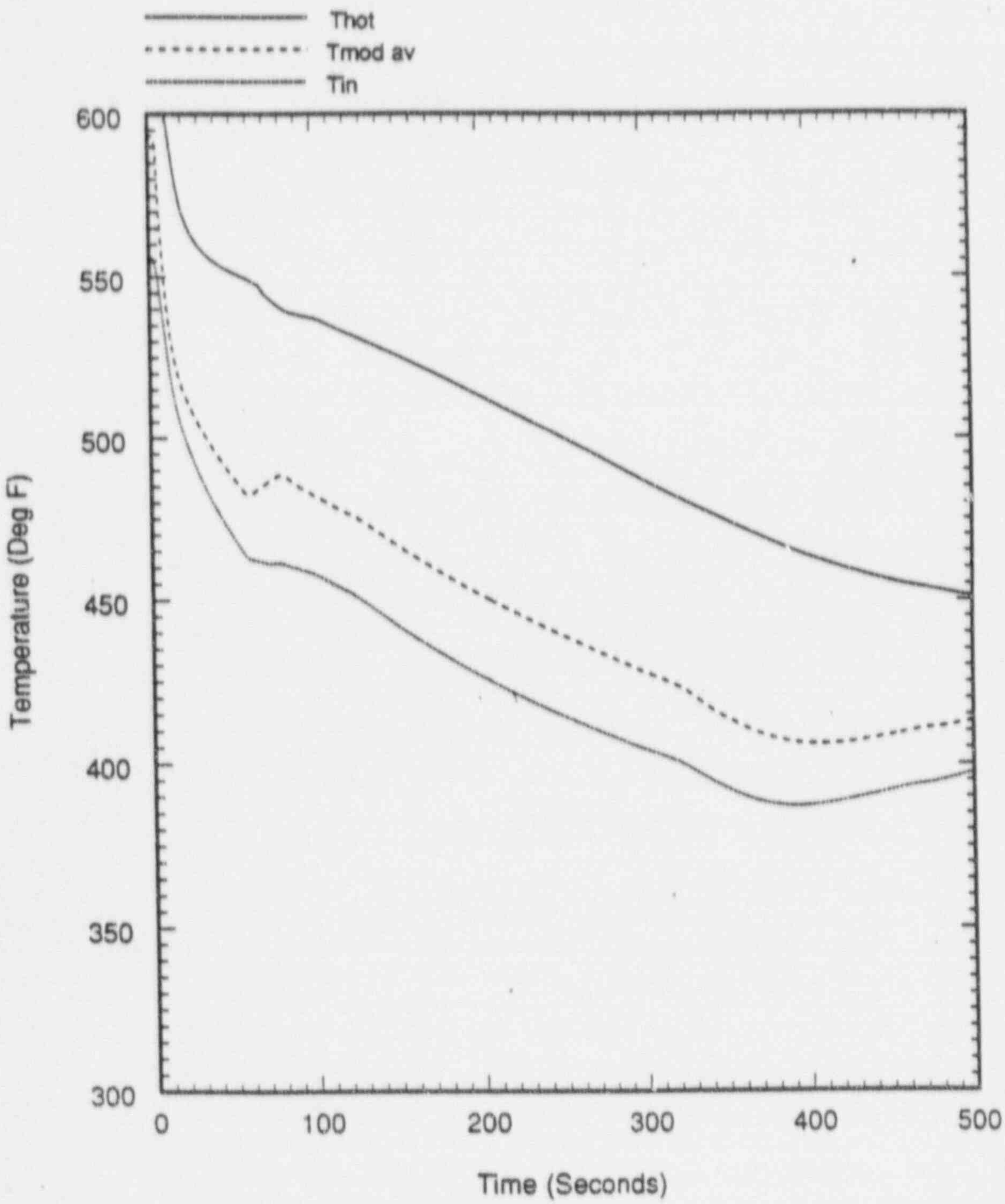


Figure 20

SLB HFP Loss of AC 1 HPSI

Reactivities vs. Time

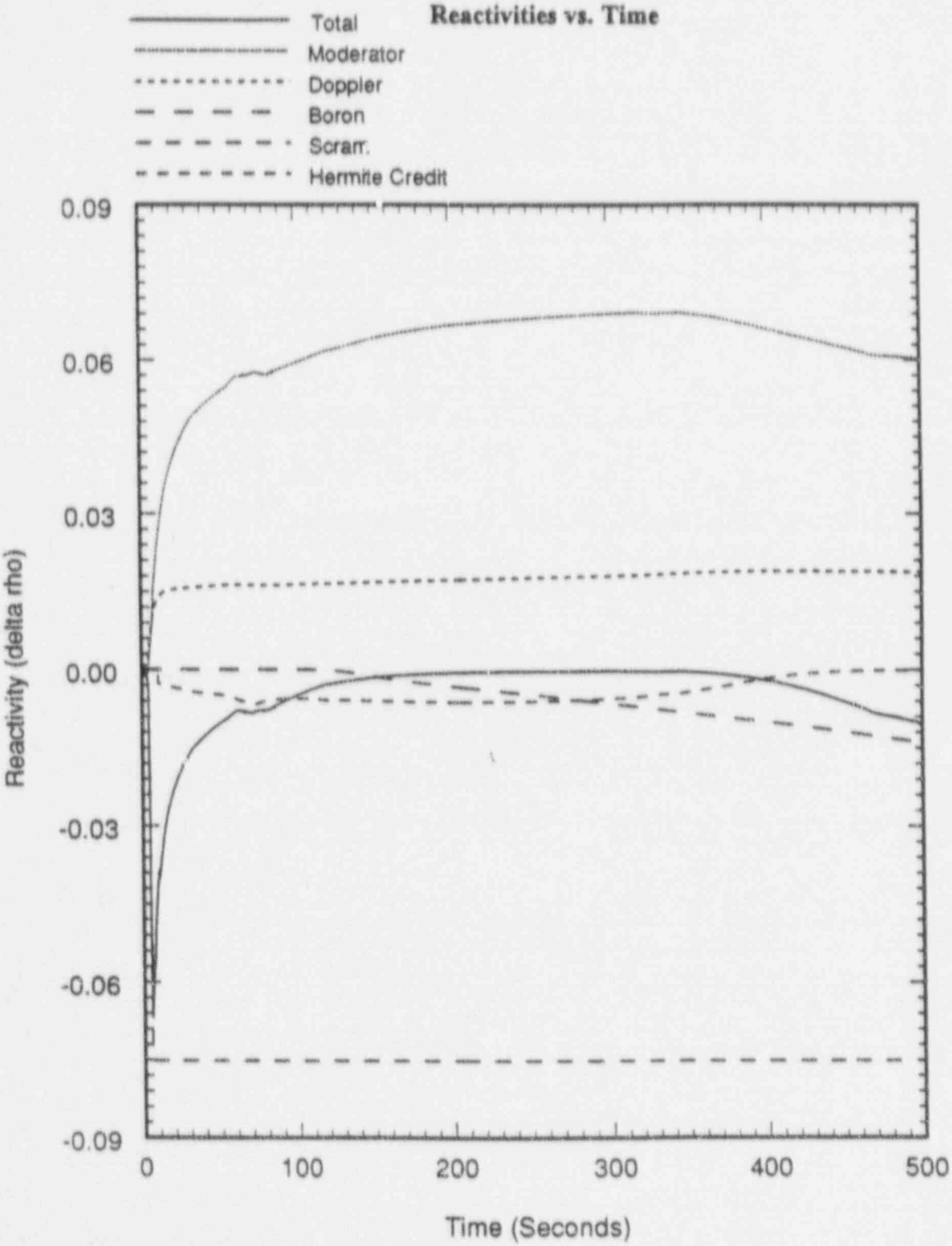


Figure 21

SLB HFP AC Available 1 HPSI

Core Power vs. Time

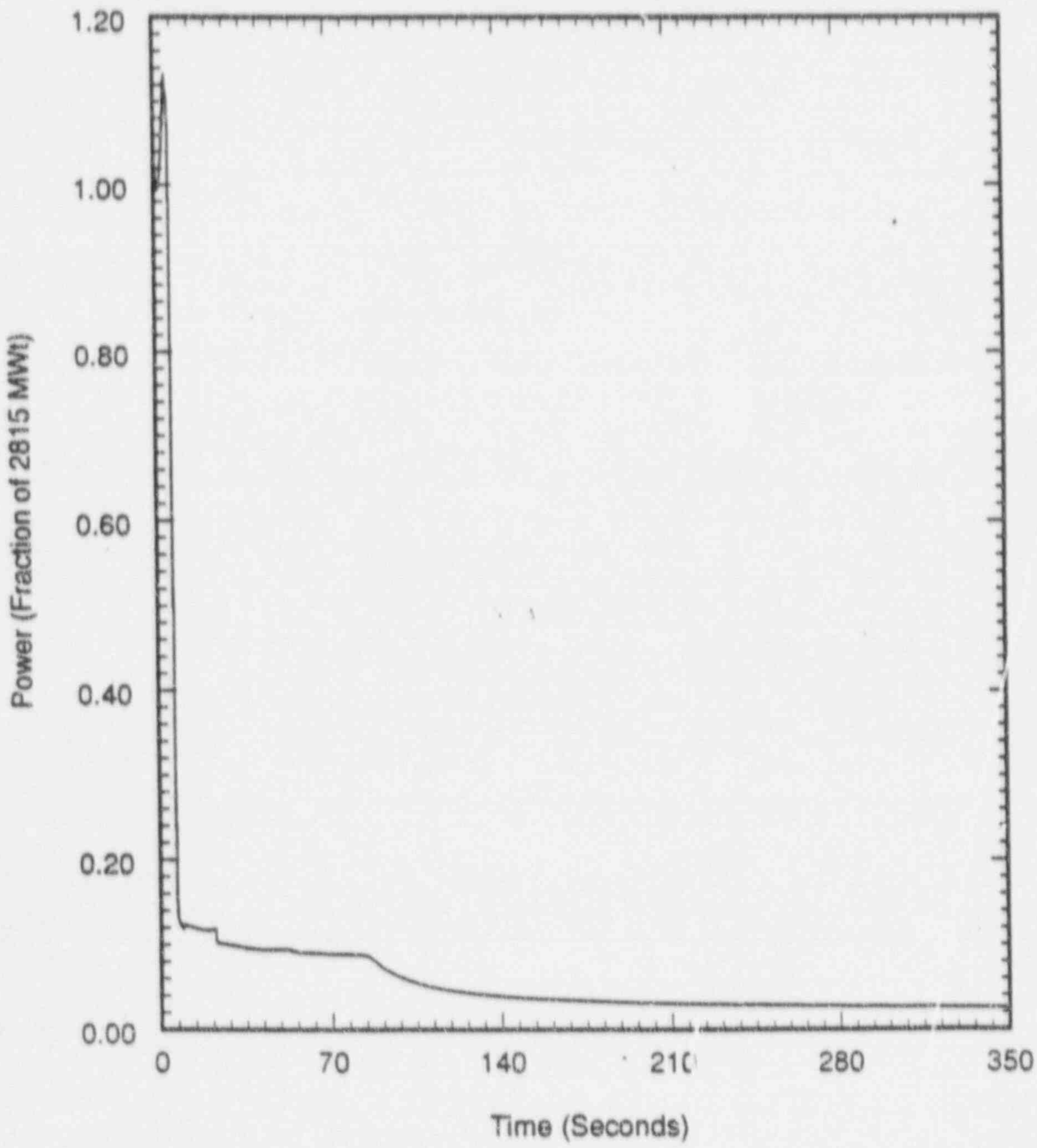


Figure 22

SLB HFP AC Available 1 HPSI

Heat Flux vs. Time

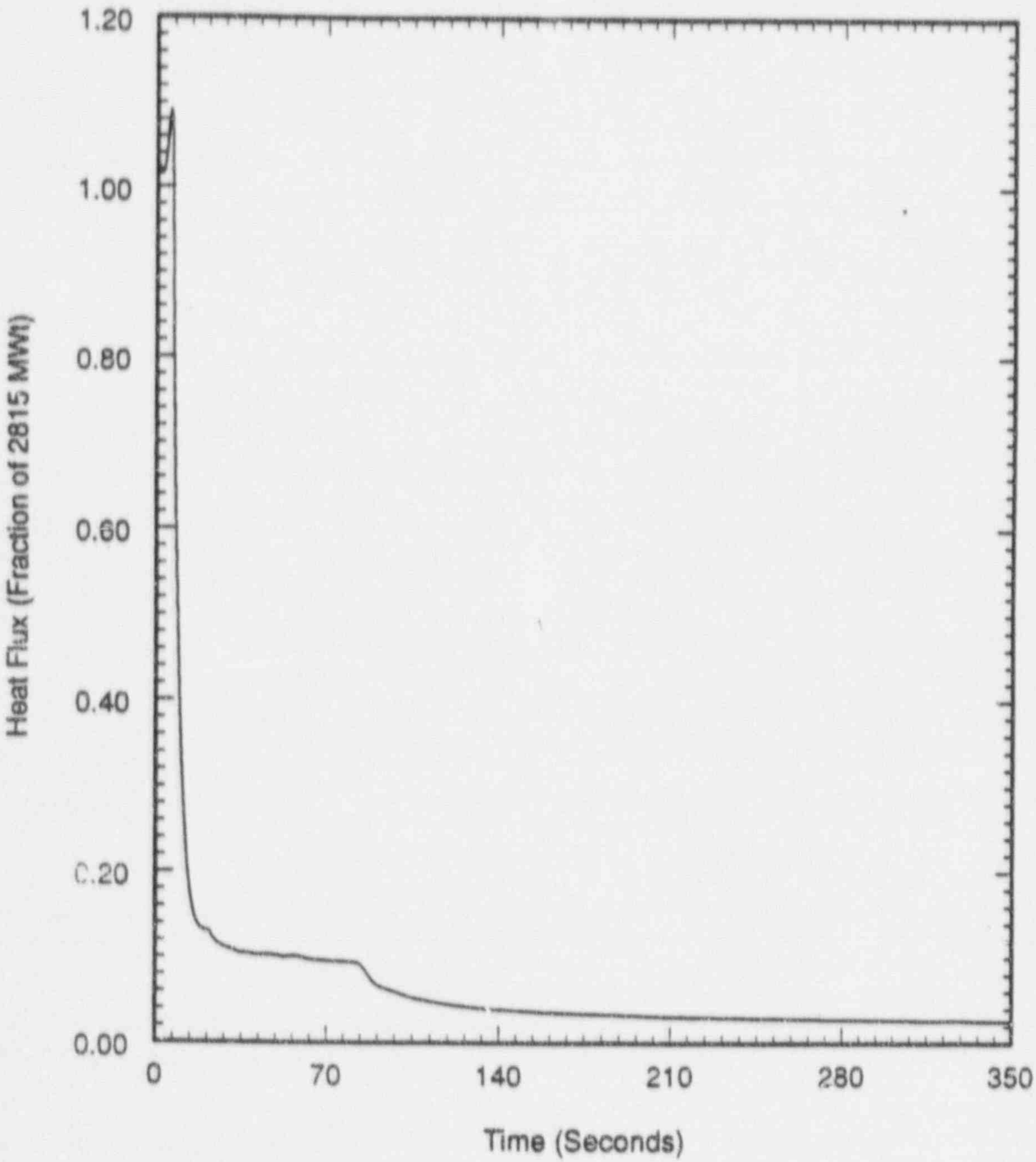


Figure 23

SLB HFP AC Available 1 HPSI

Pressurizer Pressure vs. Time

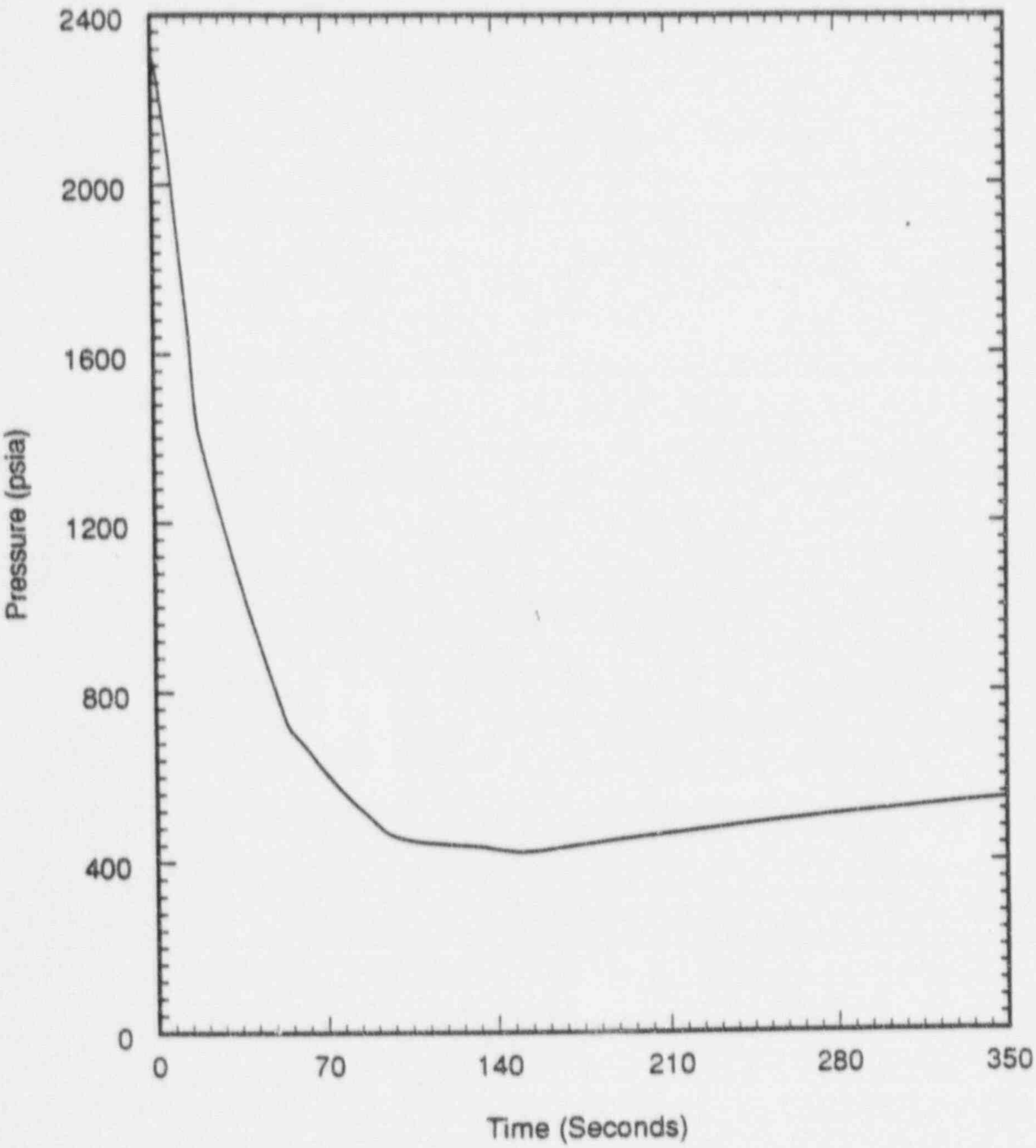


Figure 24

SLB HFP AC Available 1 HPSI

Steam Generator Pressure vs. Time

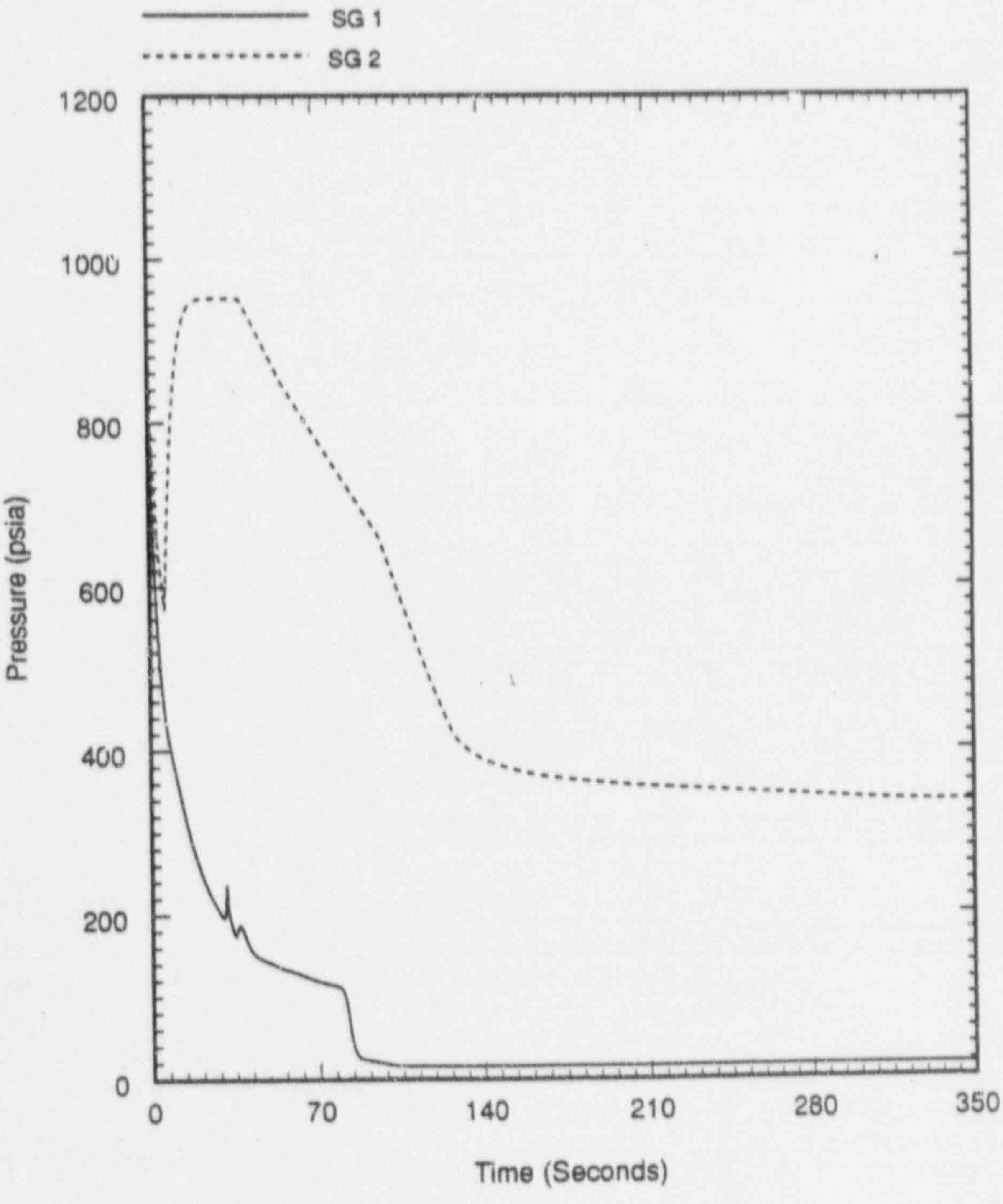


Figure 25

SLB HFP AC Available 1 HPSI

RCS Temperatures vs. Time

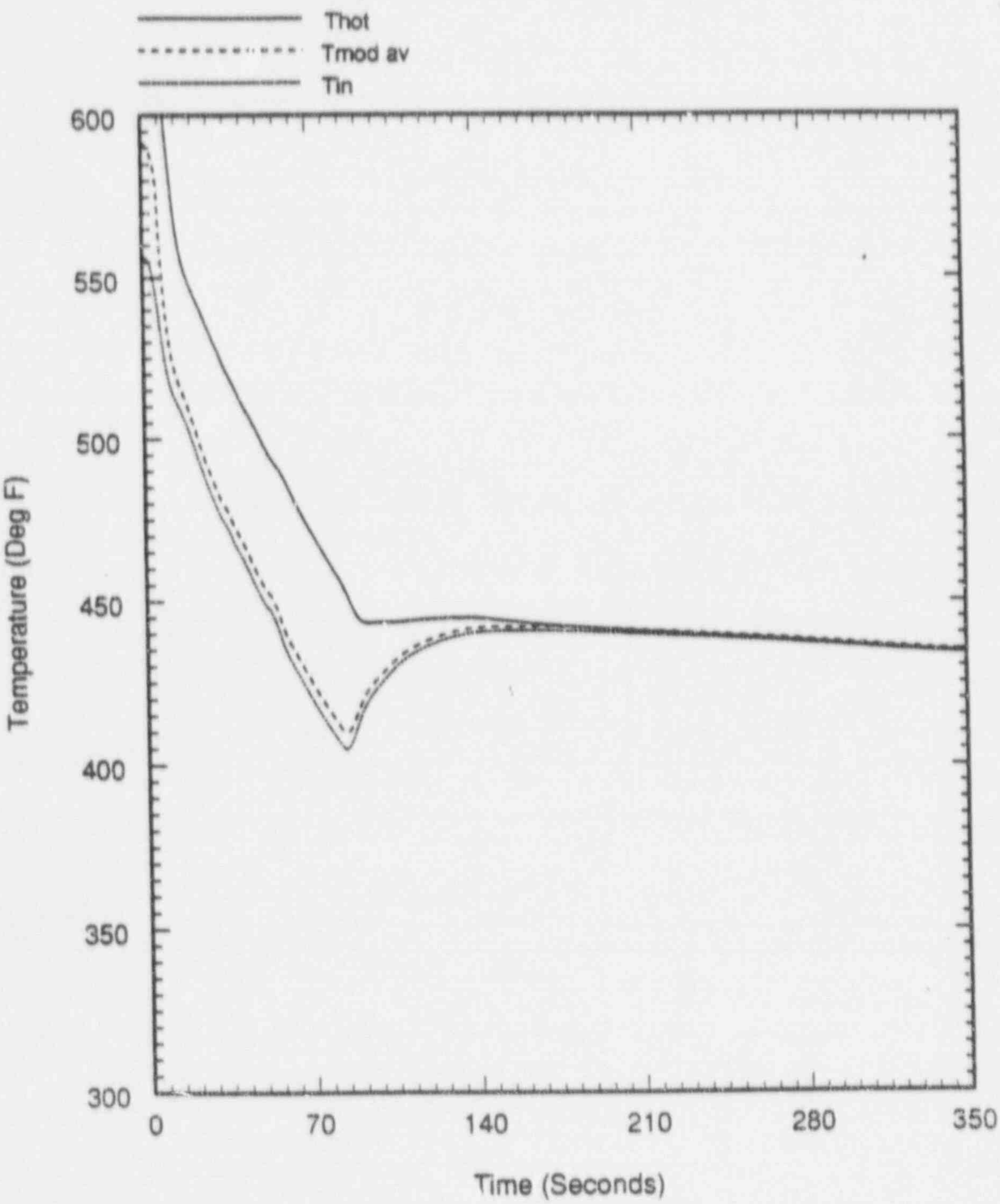


Figure 25

SLB HFP AC Available 1 HPSI

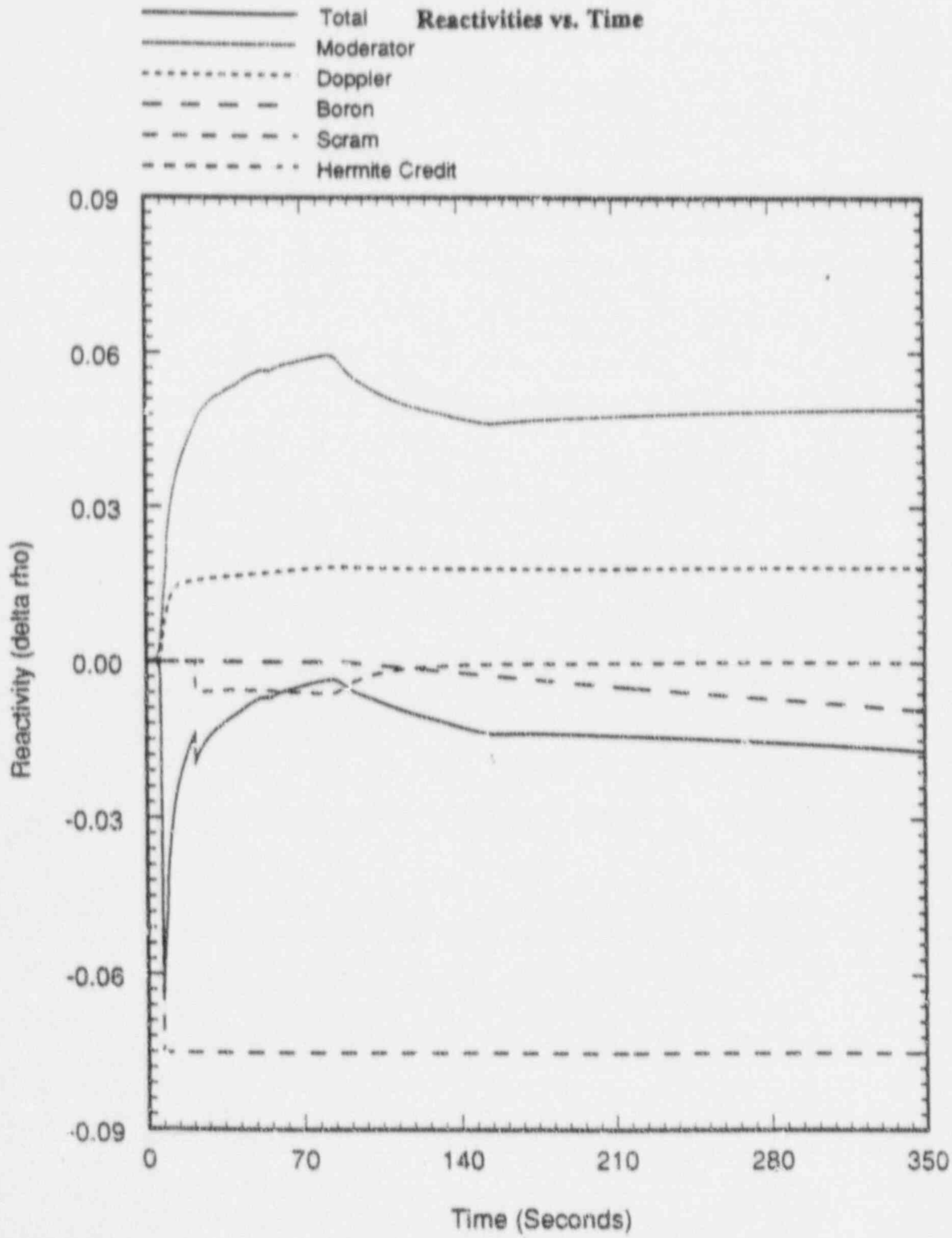


Figure 27

SLB HZP Loss of AC 1 HPSI

Core Power vs. Time (Semi Log scale)

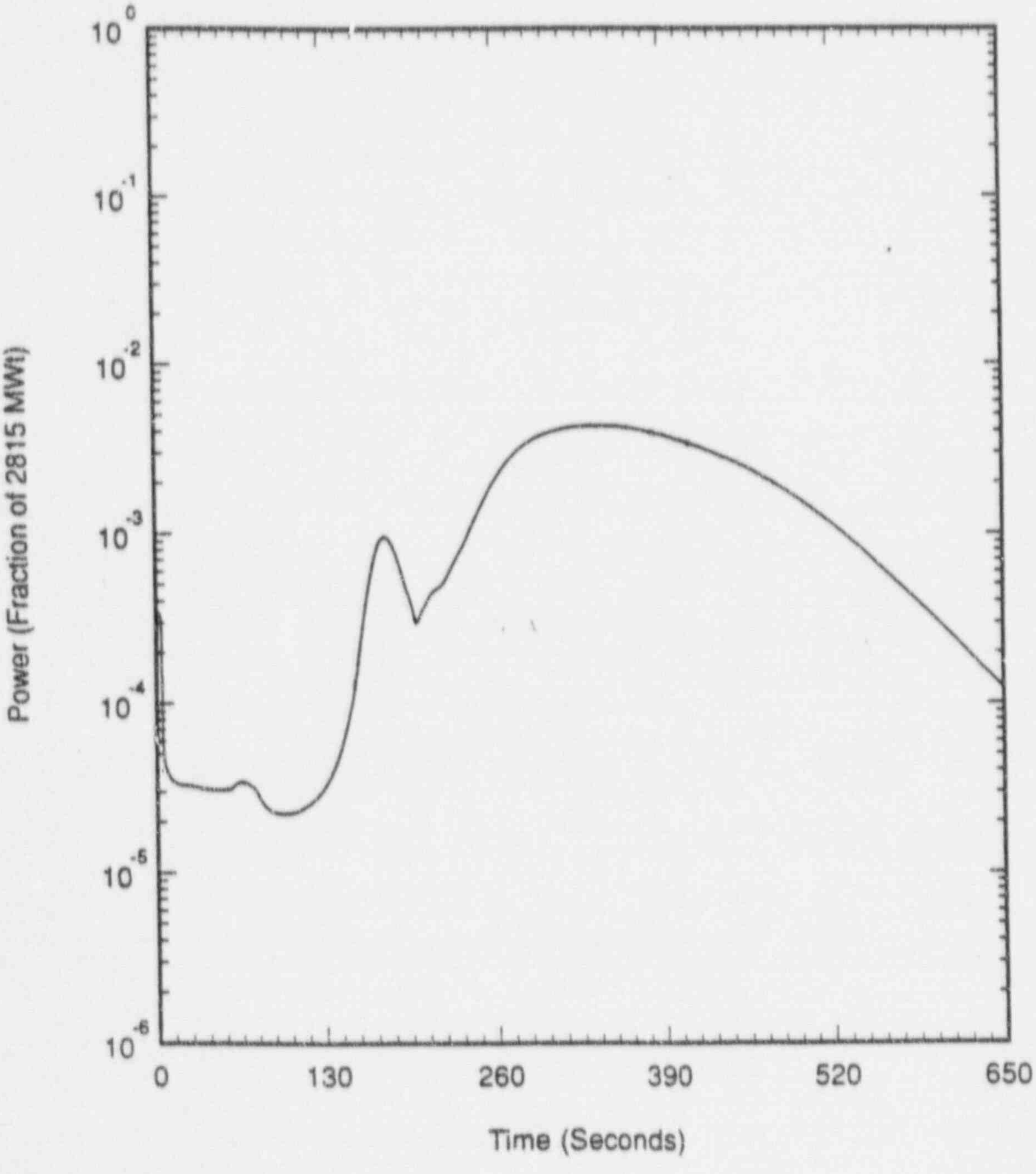


Figure 28

SLB HZP Loss of AC 1 HPSI

Heat Flux vs. Time (Semi Log scale)

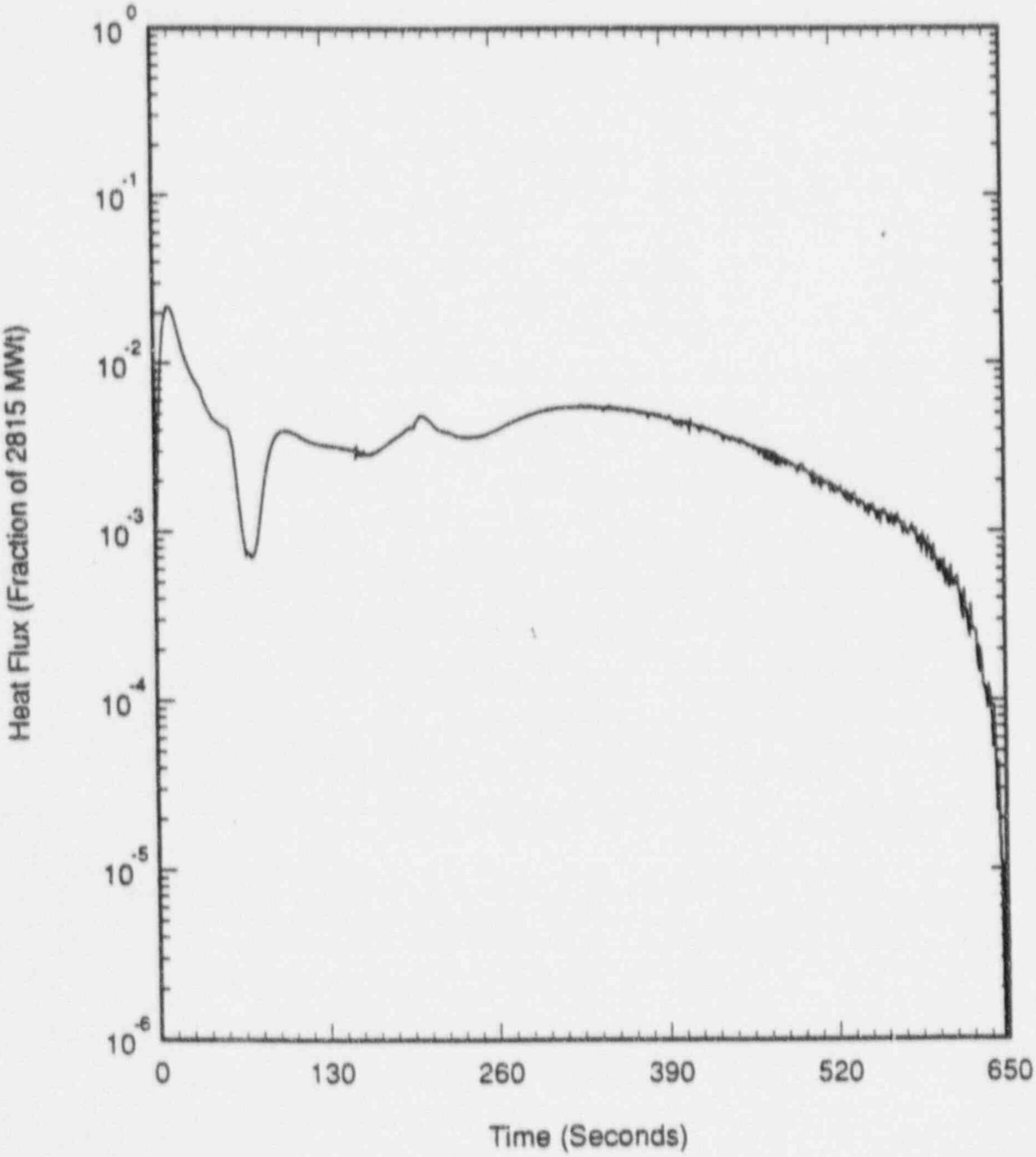


Figure 29

SLB HZP Loss of AC 1 HPSI

Pressurizer Pressure vs. Time

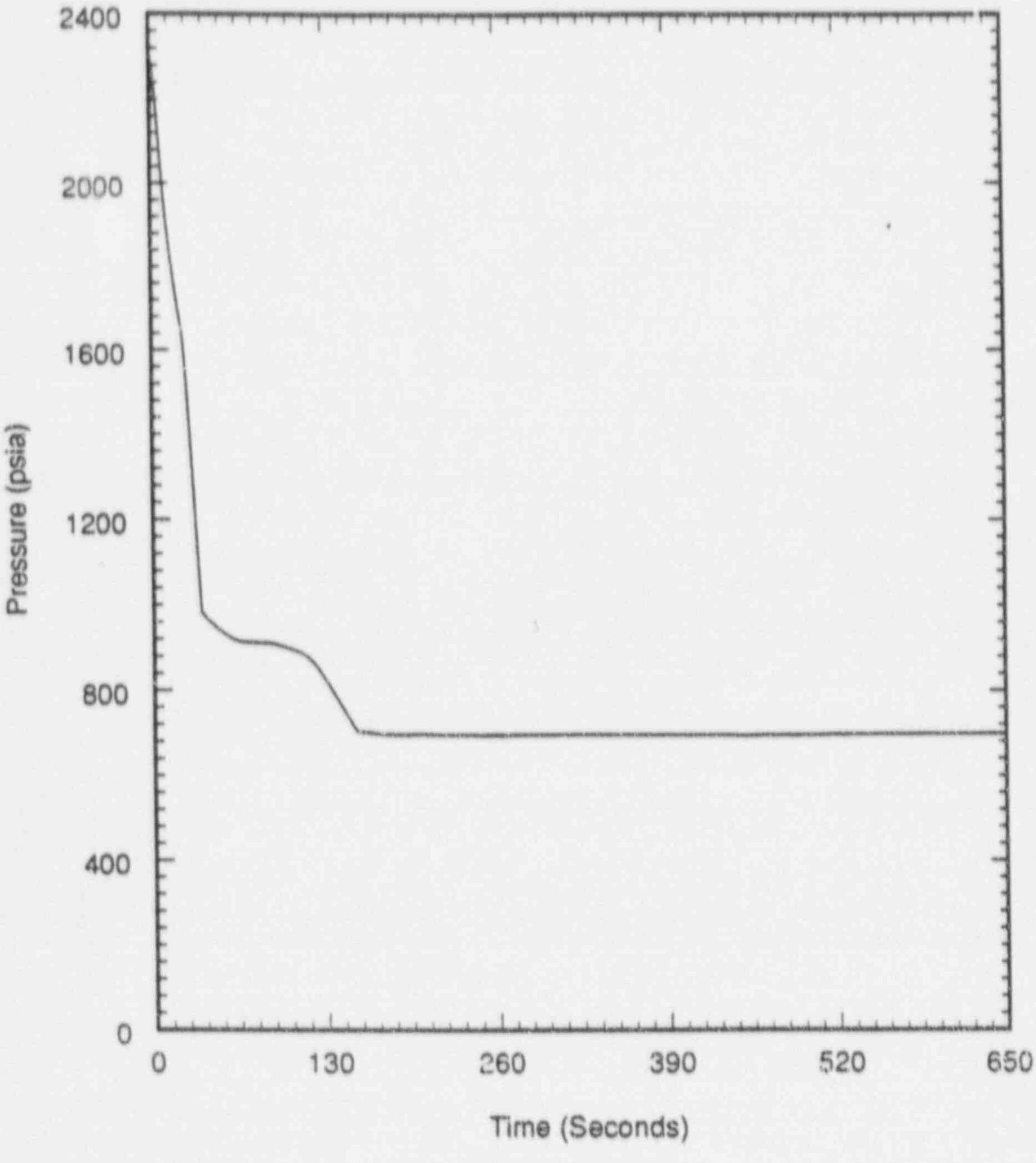


Figure 30

SLB ..ZP Loss of AC 1 HPSI

Steam Generator Pressure vs. Time

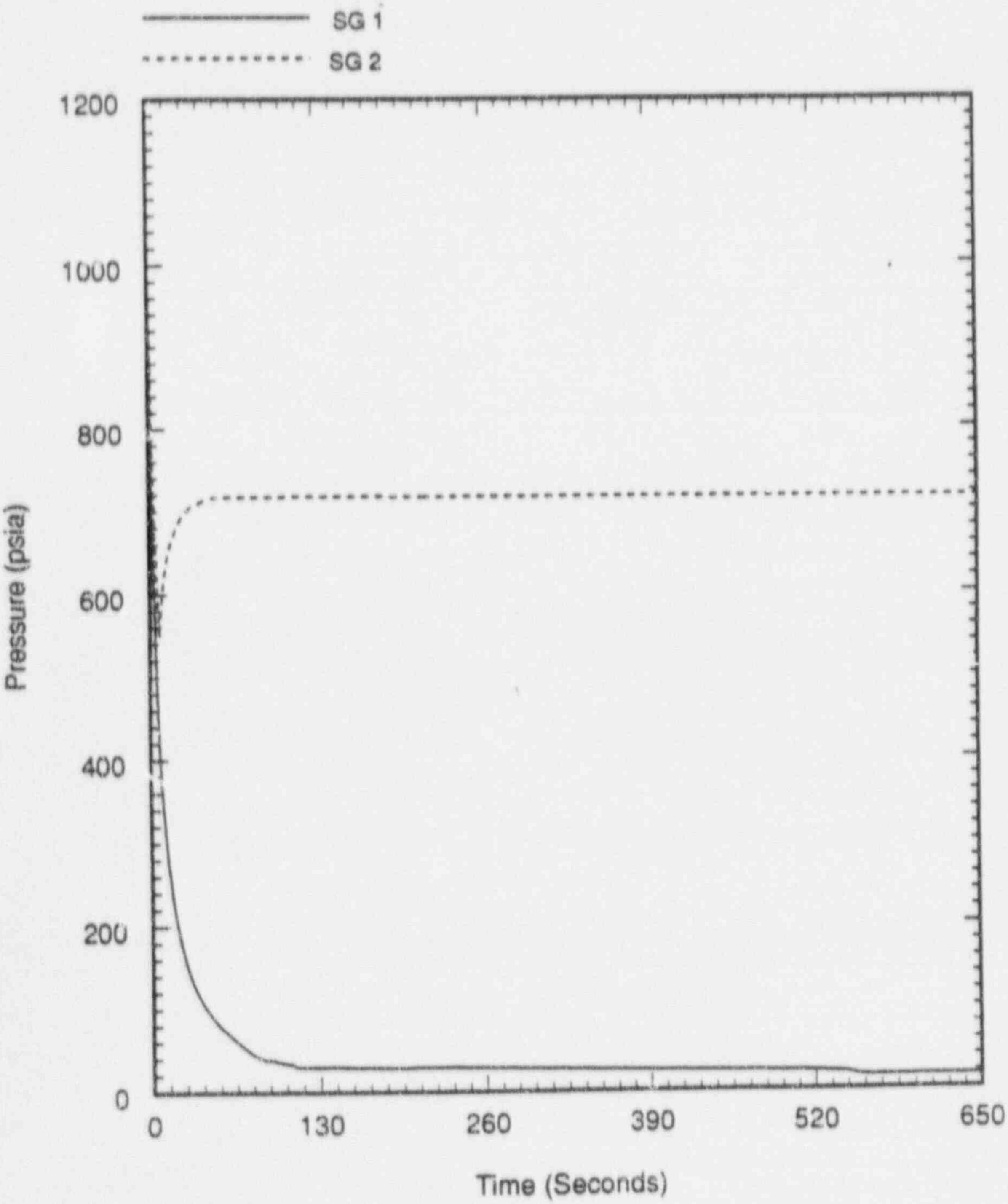


Figure 31

SLB HZP Loss of AC 1 HPSI

RCS Temperatures vs. Time

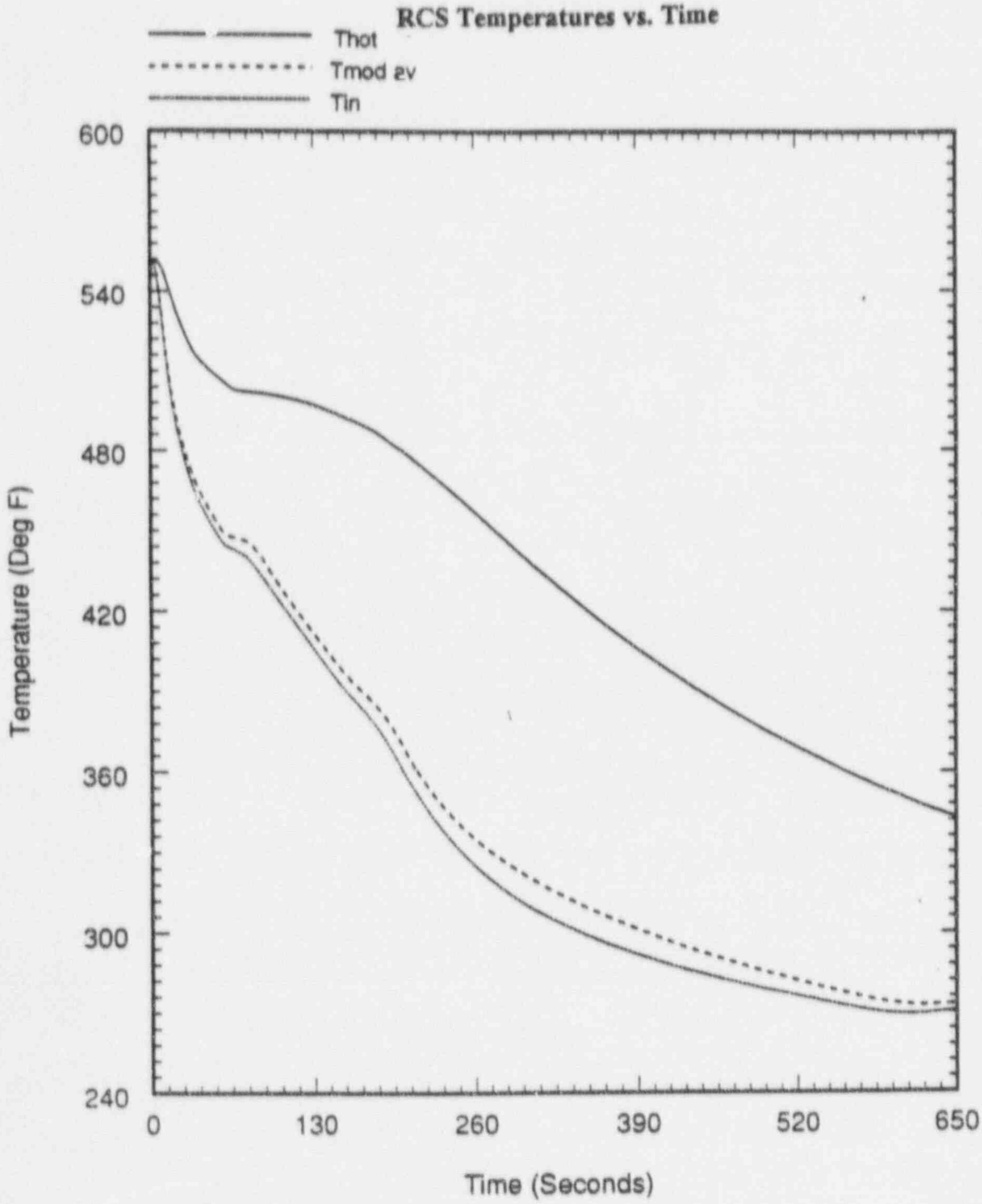


Figure 32

SLB HZP Loss of AC 1 HPSI

Reactivities vs. Time

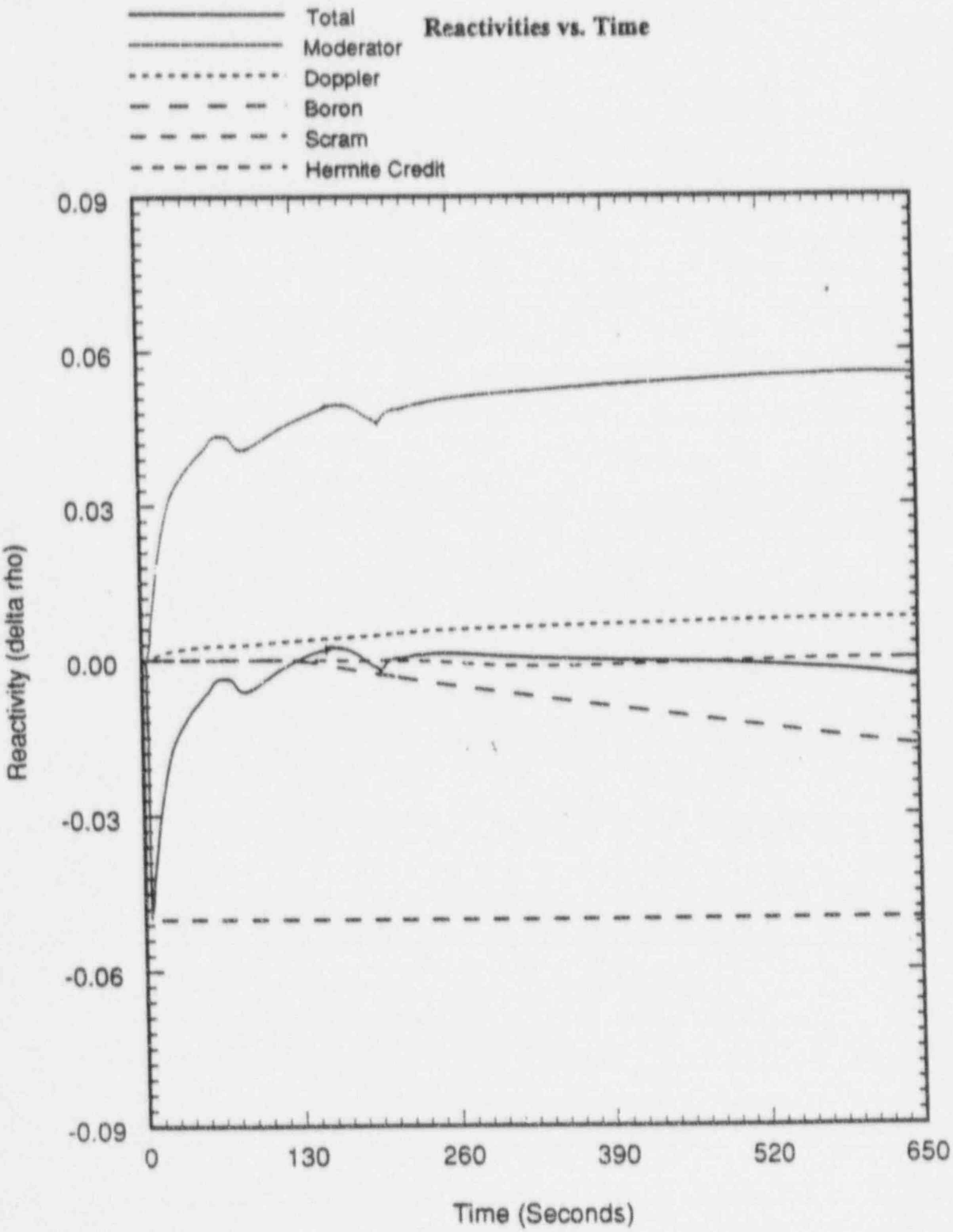


Figure 33

SLB HZP AC Available 1 HPSI

Core Power vs. Time (Semi Log scale)

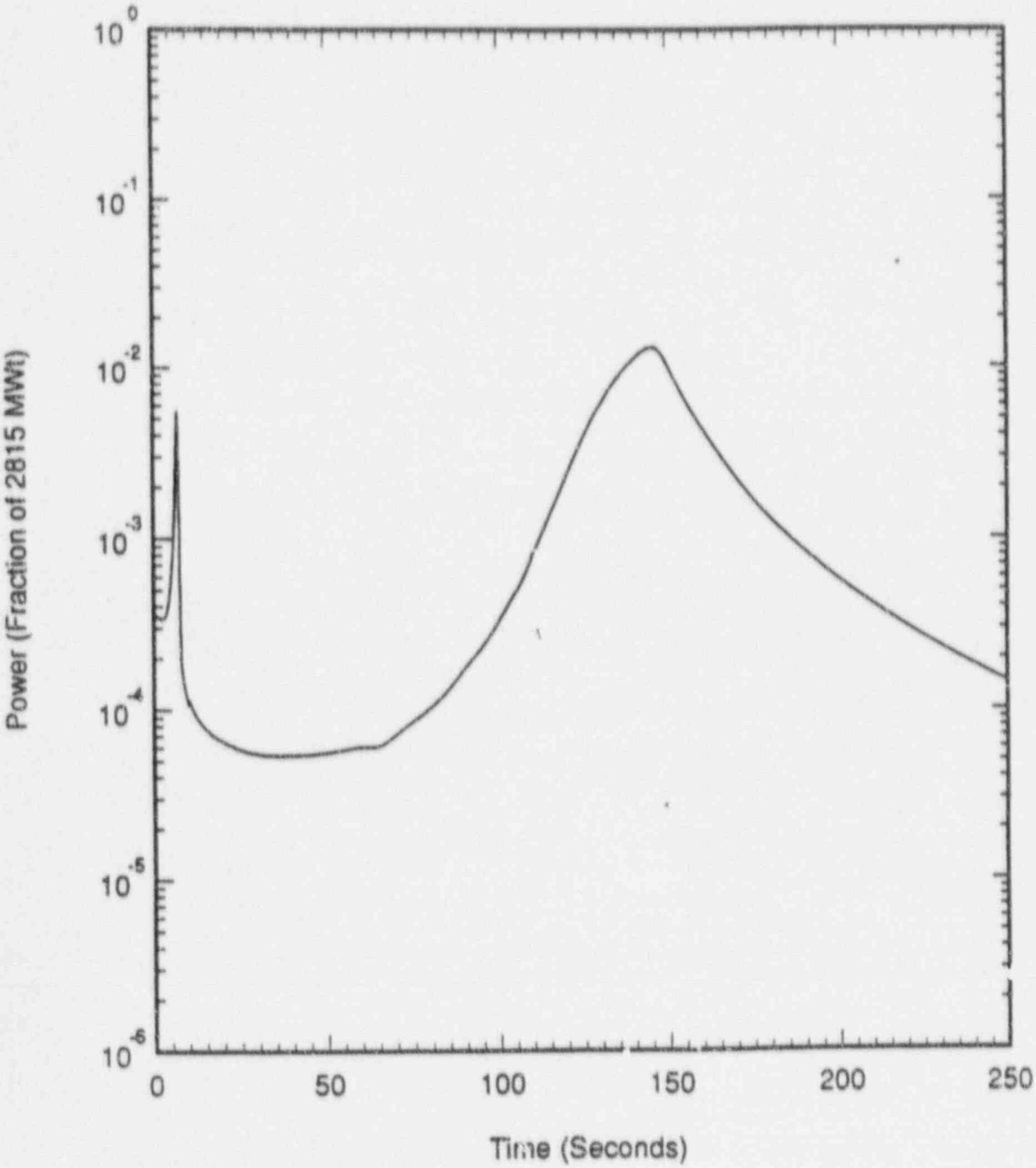


Figure 34

SLB HZP AC Available 1 HPSI

Heat Flux vs. Time (Semi Log scale)

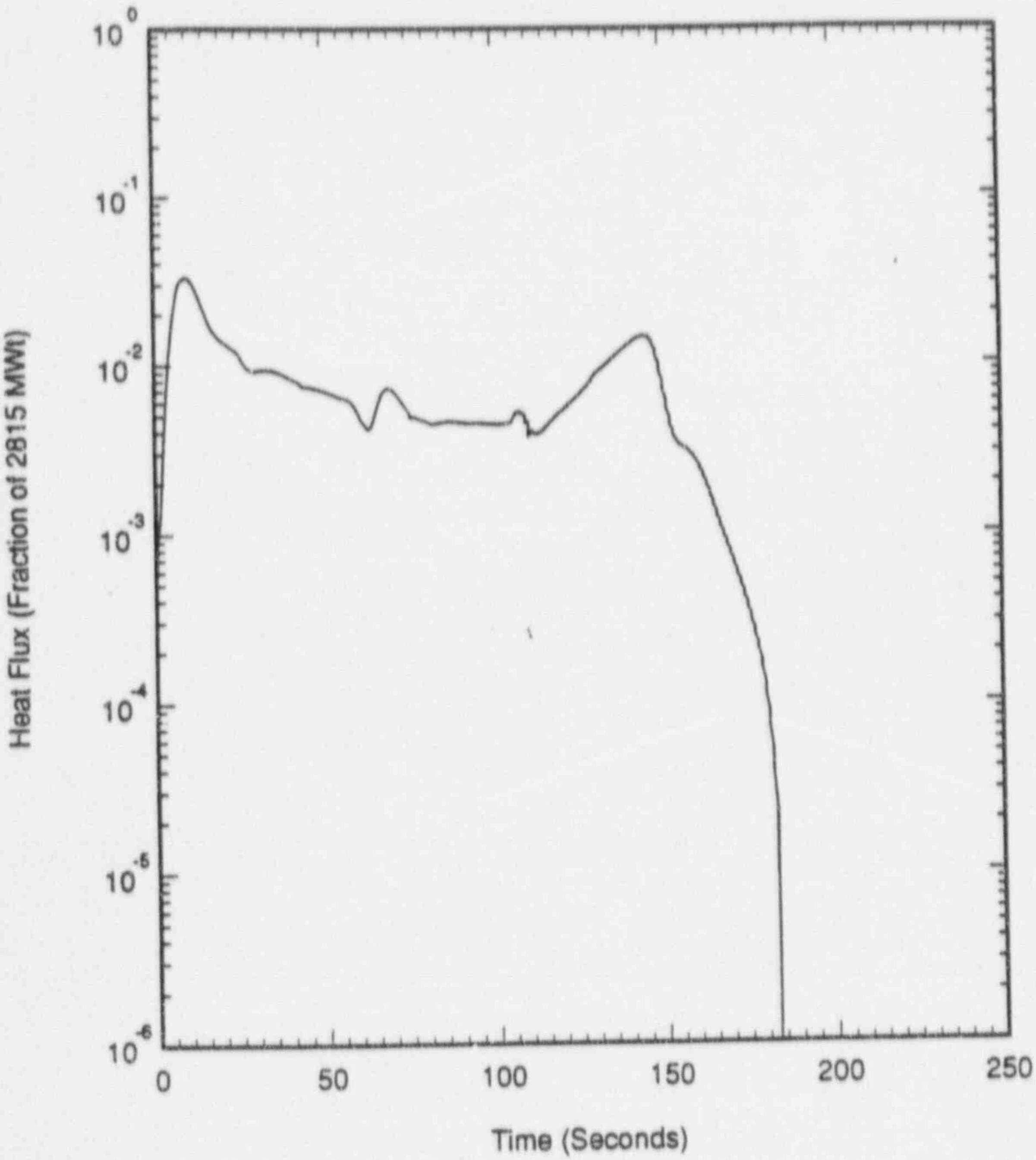


Figure 35

SLB HZP AC Available 1 HPSI

Pressurizer Pressure vs. Time

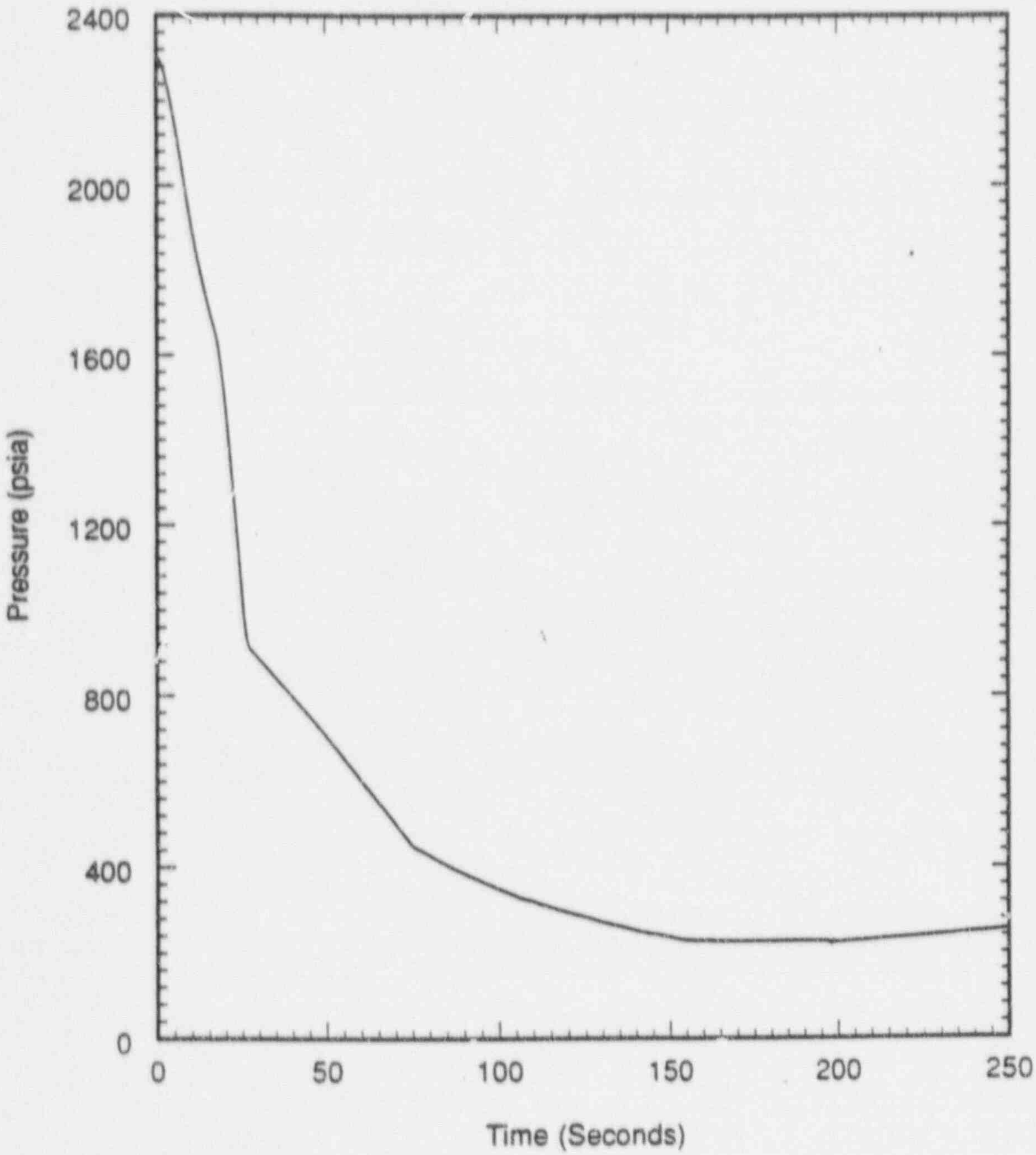


Figure 36

SLB HZP AC Available 1 HPSI

Steam Generator Pressure vs. Time

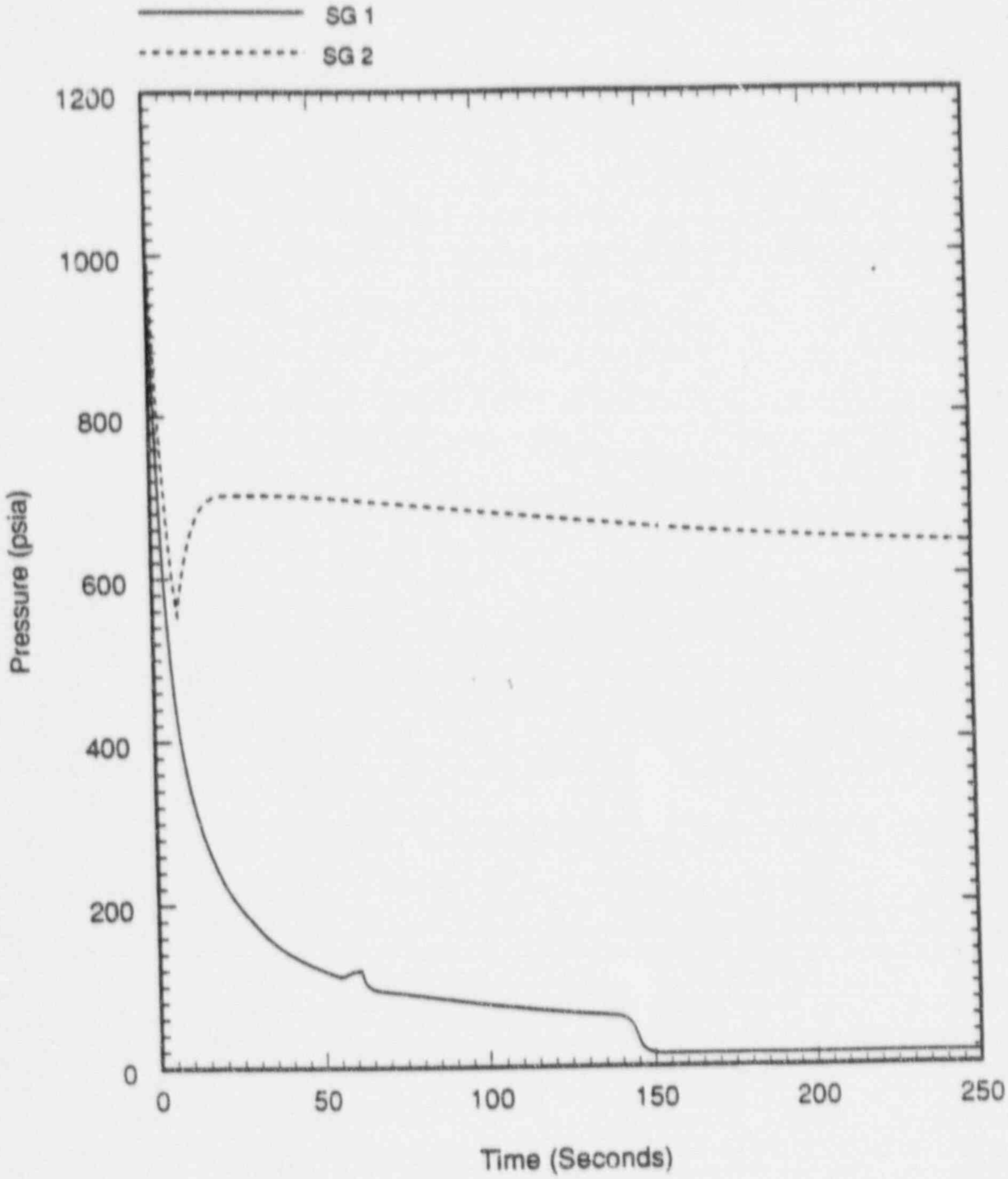


Figure 37

SLB HZP AC Available 1 HPSI

RCS Temperatures vs. Time

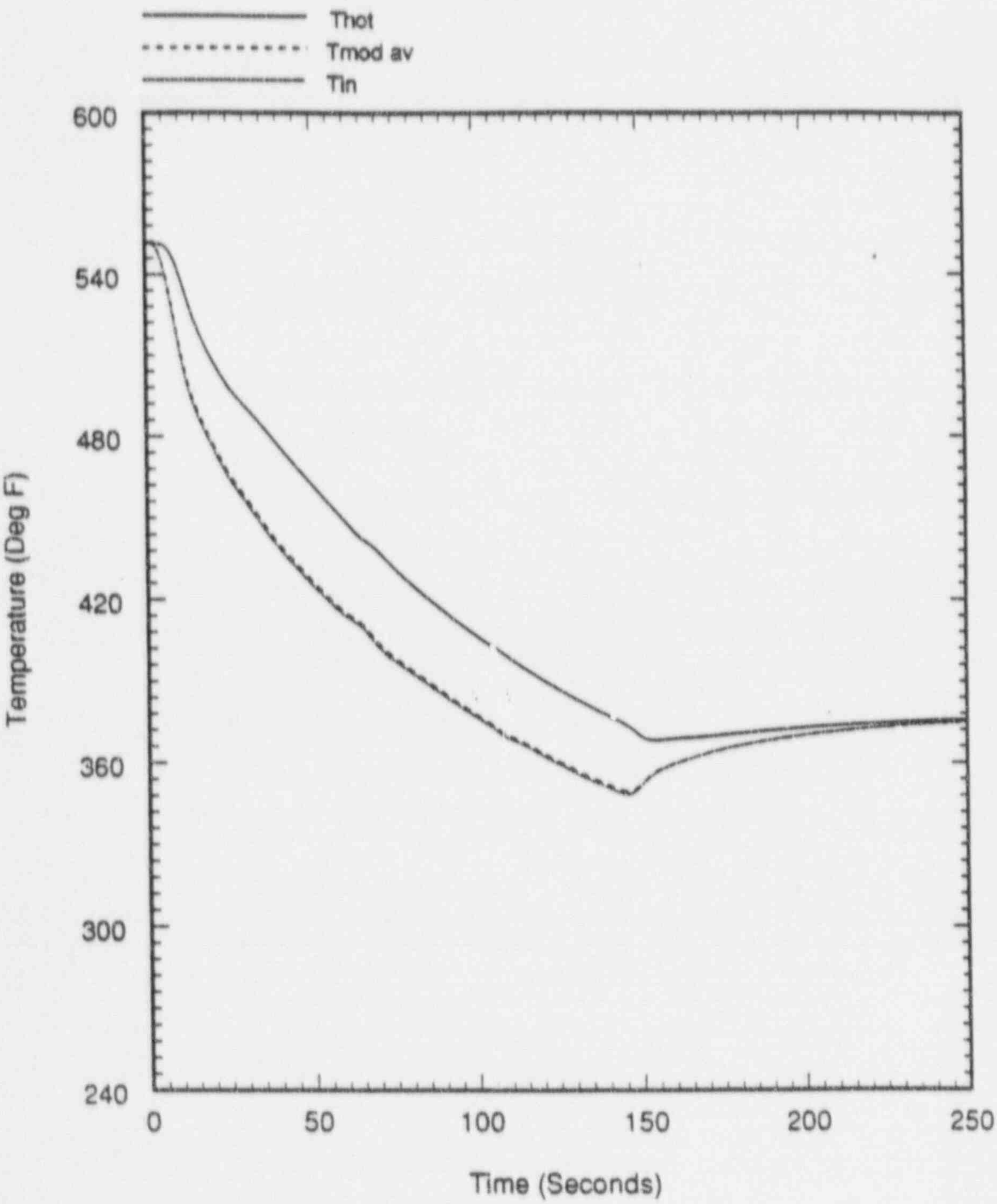


Figure 38

SLB HFP AC Available 1 HPSI

Reactivities vs. Time