Seabrook Station Fuel Upgrade Program LOCA Safety Analysis Report

SEABROOK STATION

LOCA SAFETY ANALYSIS REPORT

August 1993

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1.0 INTRODUCTION

1.1 Classification of Faults

1.1.1 Condition IV Limiting Faults

American Nuclear Society (ANS) Condition IV occurrences are faults which are not expected to occur during the lifetime of Seabrook Station, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. They are the most drastic occurrences which must be designed against and thus represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR 100. A single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault, including those of the emergency core cooling system (ECCS) and the containment. For the purposes of this report, only the following Condition IV fault will be addressed:

> Major rupture of pipes containing reactor coolant up to and including double-ended rupture of the largest pipe in the reactor coolant system (large break loss-of-coolant accident).

1.1.2 Condition III Infrequent Faults

ANS Condition III occurrences are faults which may occur very infrequently during the lifetime of Seabrook Station. They will be accommodated with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of areas beyond the exclusion radius. A Condition III fault will not, by itself, generate a Condition IV fault or result in a consequential loss of function of the RCS or containment barriers. For the purposes of this report, only the following Condition III fault will be addressed:

> Loss of reactor coolant from small ruptured pipes or from cracks in large pipes which actuate the emergency core cooling system (small break loss-of-coolant accident).

1.2 Loss-of-Coolant Accident Acceptance Criteria

The Acceptance Criteria for a loss-of-coolant accident (LOCA) are described in 10 CFR 50.46 (Reference 1) as follows:

A. The calculated peak fuel element clad temperature does not exceed the requirement of 2200°F.

B. The amount of fuel element cladding that reacts chemically with water or steam to generate hydrogen, does not exceed one percent of the total amount of Zircaloy (or ZIRLO) in the fuel rod cladding.

C. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

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

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

These criteria were established to provide a significant margin in emergency core cooling system (ECCS) performance following a LOCA. WASH-1400 (USNRC 1975) (Reference 2) presents a study in regards to the probability of occurrence of RCS pipe ruptures.

2.0 MAJOR REACTOR COOLANT SYSTEM PIPE RUPTURES (LARGE BREAK LOCA)

This section presents a description and results of the large break loss-of-coolant accident (LOCA) in conformance with 10 CFR 50.46 and Appendix K of 10 CFR 50 (Reference 1)

2.1 Identification of Causes and Frequency Classification

A LOCA is the result of a pipe rupture of the reactor coolant system (RCS) pressure boundary. For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is considered an American Nuclear Society (ANS) Condition IV event, a limiting fault, in that it is not expected to occur during the lifetime of Seabrook Station, but is postulated as a conservative design basis.

2.2 Sequence of Events and Systems Operations

The time sequence of events following a large break LOCA is presented in Figure 2-1A.

Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. Loss-Of-Offsite Power (LOOP) is assumed coincident with the occurrence of the break. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. A safety injection signal is generated when the appropriate setpoint (high containment pressure or low pressurizer pressure) is reached. These countermeasures

will limit the consequences of the accident in two ways:

A. Reactor trip and borated water injection supplement void formation in causing rapid reduction of power to the residual level corresponding to fission product decay heat. No credit is taken in the LOCA analysis for the boron content of the injection water. However, an average RCS/sump mixed boron concentration is calculated to ensure that the post-LOCA core remains subcritical. In addition, the insertion of control rods to shut down the reactor is neglected in the large break analysis.

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

For large break LOCAs, the most limiting single failure with respect to peak clad temperature (PCT) has been shown by experience to be that which reduces safety injection while producing the lowest containment pressure. The lowest containment pressure would be obtained only if all containment spray pumps operated subsequent to the postulated LOCA. Therefore, for the purposes of large break LOCA analyses, the most limiting single failure would be the loss of one residual heat removal (RHR) pump with full operation of the spray pumps. Credit could be taken for two safety injection pumps (SIPs), two centrifugal charging pumps (CCPs) and one RHR (low head) pump for a large break. However, the Seabrook large break LOCA analysis conservatively assumes both maximum containment safeguards (lowest containment pressure) and minimum emergency core cooling system (ECCS) safeguards (the loss of one complete train of ECCS components which includes one RHR pump, one SIP and one CCP), which results in the minimum delivered ECCS flow available to the RCS. Both Emergency Diesel Generators (EDGs) are assumed to start in the modeling of the containment spray pumps. Modeling full containment heat removal systems operation is required by Branch Technical Position CSB 6-1 and is conservative for the large break LOCA. This assumption is consistent with the current methodology for large break analyses.

In the large break analysis, one ECCS train starts and delivers flow through the injection lines (one for each loop) with one branch injection line spilling to containment pressure. To minimize delivery to the reactor, the branch line chosen to spill is selected as the one with the minimum resistance (based on a 10 gpm flow imbalance of the SIP and CCP branch lines). In addition, the SIP and CCP performance curves were degraded by 5% and the RHR pump performance curve was degraded 8.75%.

Therefore, in the large break ECCS analysis presented here, single failure is conservatively accounted for via the loss of an ECCS train, and the spilling of the minimum resistance injection line while assuming full active containment heat removal system operation.

2.3 Description of Large Break LOCA Transient

Before the break occurs, the unit is in an equilibrium condition; i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay, hot internals and the vessel, continues to be transferred to the reactor coolant. At the beginning of the blowdown phase, the entire RCS contains subcooled liquid which removes heat from the core by forced convection with some fully developed nucleate boiling. Shortly after break initiation, departure from nucleate boiling is calculated to occur using a critical heat flux correlation consistent with Appendix K of 10 CFR 50. Thereafter, the heat removal from the clad surface is calculated based upon the heat transfer coefficient appropriate to the regime, which is a function of the local fluid properties and rod heat flux. Radiation heat transfer from the clad to both steam and droplets, as well as other fuel rods, is also considered.

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

When the RCS depressurizes to 600 psia, the accumulators begin to inject borated water into the reactor coolant loops. The conservative assumption is made that all of the accumulator water injected during the bypass period is subtracted from the RCS after the bypass period terminates (called end-of-bypass). End-of-bypass occurs when the expulsion or entrainment mechanisms responsible for the bypassing are calculated not to be effective. The rejection of the accumulator water delivered prior to end-of-bypass is again consistent with Appendix K of 10 CFR 50. Since LOOP is assumed, the reactor coolant pumps trip at the inception of the accident. The effects of pump coastdown are included in the blowdown analysis.

The blowdown phase of the transient ends when the RCS pressure (initially assumed at 2300 psia) falls to a value approaching that of the containment atmosphere. Prior to or at the end of the blowdown, termination of bypass occurs and refill of the reactor vessel lower plenum begins. Refill is completed when emergency core cooling water has filled the lower plenum of the reactor vessel, which is bounded by the bottom of the fuel rods (called bottom of core (BOC) recovery time).

The reflood phase of the transient is defined as the time period lasting from BOC recovery until the reactor vessel has been filled with water to the extent that the core temperature rise has been terminated. From the latter stage of blowdown to the beginning of reflood, the safety injection accumulator tanks rapidly discharge borated cooling water into the RCS thus contributing to the filling of the reactor vessel downcomer. The downcomer water elevation head provides the driving force required for the reflooding of the reactor core. The RHR pumps, CCPs and SIPs aid in the filling of the downcomer and subsequently supply water to maintain a full downcomer and complete the reflooding process.

Continued operation of the ECCS pumps supplies water during long-term cooling. Core temperatures have been reduced to long-term steady state levels associated with dissipation of residual heat generation. After the water level of the refueling water storage tank (RWST) reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching to the cold leg recirculation phase of operation. Spilled borated water is drawn from the engineered safety features (ESF) containment sumps by the RHR pumps and returned to the RCS cold legs. The containment spray pumps are manually aligned to the containment emergency sumps and continue to operate to further reduce containment pressure and temperature.

2.4 Core and System Performance

2.4.1 Mathematical Model

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

2.4.2 Large Break LOCA Evaluation Model

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

A description of the various aspects of the LOCA analysis methodology is given by Bordelon, Massie, and Zordan (1974) (Reference 3). This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the Acceptance Criteria. The SATAN-VI, WREFLOOD, BASH and LOCBART codes, which are used in the LOCA analysis, are described in detail by Bordelon et al. (1974) (Reference 4); Kelly et al. (1974) (Reference 5); Young et al. (1987) (Reference 6); and Bordelon et al. (1974) (Reference 3).

Code modifications are specified in References 7, 8, 9, and 10. These codes assess the core heat transfer geometry and determine if the core remains amenable to cooling through and subsequent to the blowdown, refill, and reflood phases of the LOCA.

SATAN-VI calculates the thermal-hydraulic transient, including the RCS pressure, enthalpy, density, and the mass and energy flow rates in the RCS, as well as steam generator energy transfer between the primary and secondary systems as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator water mass and internal pressure and the break mass and energy flow rates that are assumed to be vented to the containment during blowdown. At the end of the blowdown phase, the mass and energy release rates during blowdown are transferred to the COCO code, detailed in Reference 11, for use in determination of the containment pressure response during the first phase of the LOCA. Additional SATAN-VI output data from the end-of-blowdown, including the core inlet flow rate and enthalpy, the core pressure, and the core power decay transient, are input to the LOCBART code.

At the end of the blowdown, information from SATAN-VI on the state of the system is transferred to the WREFLOOD code which calculates the time to BOC recovery, RCS conditions at BOC and mass and energy release from the break during the reflood phase of the LOCA. Since the mass flow rate to the containment depends upon the core flooding rate and the local core pressure, which is a function of the containment pressure, the WREFLOOD and COCO codes are interactively linked. The BOC conditions calculated by WREFLOOD and the containment pressure transient calculated by COCO are used as input to the BASH code. Data from both SATAN-VI code and the WREFLOOD code out to BOC are input to the LOCBART code which calculates core average conditions at BOC for use by the BASH code.

The BART code (Reference 12) has been coupled with a loop model to form the BASH code, in which BART provides the entrainment rate based on the core flooding rate. The BASH code provides a realistic thermal-hydraulic simulation of the reactor core and RCS during the reflood phase of a large break LOCA. Instantaneous values of the accumulator conditions and safety injection flow at the time of completion of lower plenum refill are provided to BASH by WREFLOOD. Figure 2-1B illustrates how BASH has been substituted for WREFLOOD in calculating transient values of core inlet flow, enthalpy, and pressure for the detailed fuel rod model, LOCBART. A detailed description of the BASH code is available in Reference 4. The BASH code provides a sophisticated treatment of steam/water flow phenomena in the reactor coolant system during core reflood. A dynamic interaction between core thermal-hydraulics and system behavior is expected, and experiments have shown this behavior. The loop model determines the loop flows and pressure drops in response to the calculated core exit flow determined by BART. The updated inlet flow calculated by the loop model is used by BART to calculate a new entrainment rate to be fed

back to the loop code. This process of transferring data between BART, the loop code and back to BART forms the calculational process for analyzing the reflood transient. This coupling of the BART code with a loop code produces a dynamic flooding transient, which reflects the close coupling between core thermal-hydraulics and loop behavior.

The cladding heat-up transient is calculated by LOCBART which is a combination of the LOCTA code with BART. A more detailed description of the LOCBART code can be found in References 3 and 8. During reflood, the LOCBART code provides a significant improvement in the prediction of fuel rod behavior. In LOCBART the empirical FLECHT correlation has been replaced by the BART code. BART employs rigorous mechanistic models to generate heat transfer coefficients appropriate to the actual flow and heat transfer regimes experienced by the fuel rods.

The fuel rod model is initialized at steady state in the LOCA calculation to be consistent with fuel temperature and rod internal pressure data provided by the more sophisticated Westinghouse fuel temperature analysis model. The stored energy and rod internal pressure are key parameters and agreement with the fuel analysis data is obtained by matching pellet average temperatures and rod internal pressures.

Modeling features necessary to account for the reactor barrel-baffle region and the reactor fuel assembly thimbles were included in this analysis as presented in Reference 2.

2.4.3 Input Parameters and Initial Conditions

Important input parameters and initial conditions used in the analysis are listed in Tables 2-1 and 2-2. The safety injection performance, as modeled for the various large break LOCA cases, is presented in Figures 2-2A and 2-2B. Cases analyzed are given in Table 2-3.

The bases used to select the numerical values that are input parameters to the analysis have been conservatively determined from extensive sensitivity studies (Westinghouse 1974 (Reference 13); Salvatori 1974 (Reference 14); Johnson, Massie, and Thompson 1975 (Reference 15)). In addition, the requirements of Appendix K to 10 CFR 50 regarding specific model features were met by selecting models which provide a significant overall conservatism in the analysis. The assumptions which were made pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs, and include such items as the core peaking factors, the containment pressure, and the performance of the ECCS. Decay heat generated throughout the transient is also conservatively calculated as per the requirements of Appendix K to 10 CFR 50. The large break evaluation model assumes a chopped cosine power shape and a core design methodology will be applied

which assures that this shape remains bounding. Figure 2-7 contains the K(z) curve assumed in the analysis.

Westinghouse ECCS analyses currently assume minimum safeguards for the safety injection flow, which minimizes the amount of flow to the RCS by assuming that the spilling line is the branch line with the least resistance. This is the limiting single failure assumption when LOOP is assumed for most Westinghouse plants. However, for some Westinghouse plants, including Seabrook Station, the current nature of the Appendix K ECCS evaluation model is such that it may be more limiting to assume the maximum possible ECCS flow delivery. The maximum safeguards case assumes operation of both trains of ECCS pumps, minimum injection line resistances and enhanced ECCS pump performance. The worst break for Seabrook Station was reanalyzed assuming maximum safeguards. Examination of the LOCA analysis results in Table 2-5 demonstrates that minimum safeguards assumptions result in the highest peak clad temperature for Seabrook Station.

2.5 Results

Based on the results of the LOCA sensitivity studies (Westinghouse 1974 (Reference 13); Salvatori 1974 (Reference 14); Johnson, Massie, and Thompson 1975 (Reference 15)), the limiting large break was found to be the double-ended cold leg guillotine (DECLG). Therefore, only the DECLG break is considered in the large break ECCS performance analysis. Calculations were performed for a range of Moody break discharge coefficients. The results of these calculations are summarized in Tables 2-4 and 2-5.

The mass and energy release rates during reflood are shown in Table 2-6. The blowdown mass and energy release data are provided in Table 2-7.

Figures 2-3 through 2-6 present the results of the cases analyzed for the large break LOCA.

Figures A	Reactor Coolant System Pressure (Calculated Core Pressure)
Figures B	Cold Leg Break Mass Flow Rate (Sum of Both Ends of the Guillotine Break)
Figures C	Core Power Transient (Fraction of the Nominal Power)
Figures D	Core Mass Flow Rate During Blowdown
Figures E	Accumulator Mass Flow Rate During Blowdown (Sum of Injection into the Intact Loops)

2

IMAGE EVALUATION TEST TARGET (MT-3)

1.0 1.1 1.25 1.4 1.6 1.8 2.0 2.2 2.5 2.8 3.2 3.6 4.0

150mm

6"

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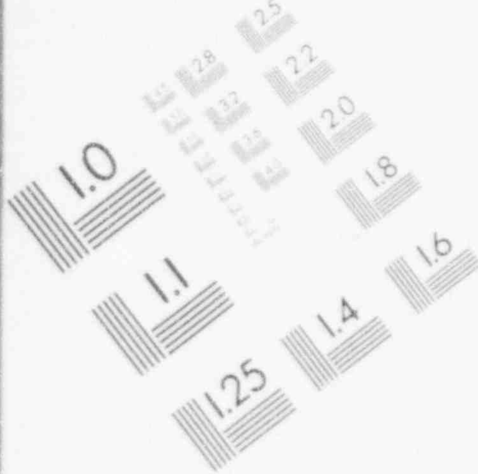
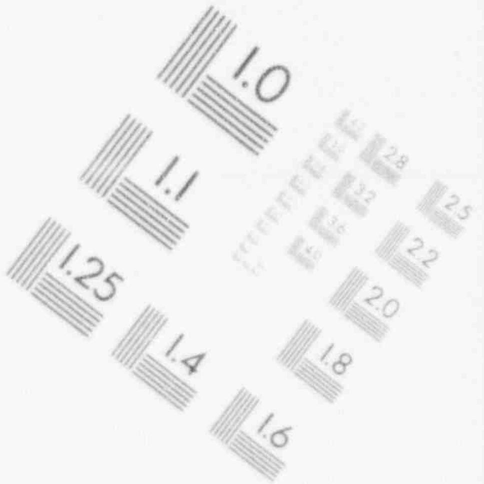
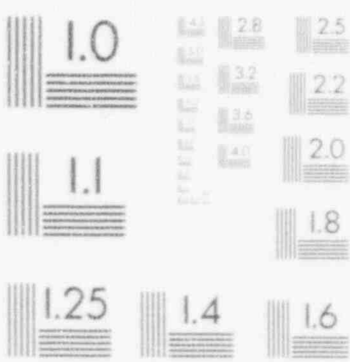
IMAGE EVALUATION TEST TARGET (MT-3)







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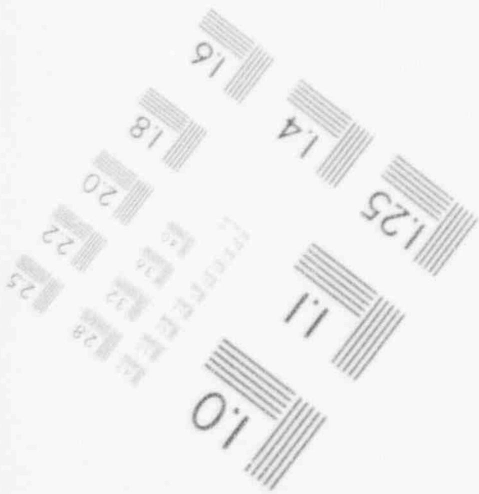

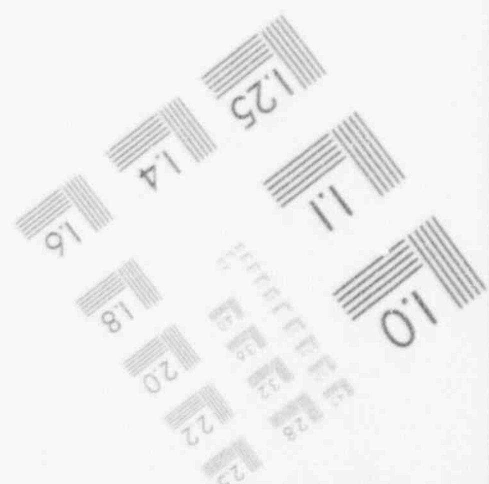

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IMAGE EVALUATION TEST TARGET (MT-3)

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Figures F	Reflood Core and Downcomer Collapsed Liquid Water Levels
Figures G	Break Energy Released to Containment as Calculated by SATAN
Figures H	Fluid Velocity Past Clad Hot Spot During Reflood
Figures I	Fluid Quality as Calculated at the Hot Spot
Figures J	Hot Rod Heat Transfer Coefficient at the Hot Spot
Figures K	Fluid Temperature at the Clad Hot Spot
Figures L	Clad Temperature at the Hot Spot on the Hot Rod
Figures M	Containment Pressure Transient
Figures N	Containment Condensing Wall Heat Transfer Coefficient

The limiting large break peak clad temperature PCT calculated is 1889°F, which is less than the acceptance criteria limit of 2200°F. This PCT, from the C_D = 0.6 minimum safeguards case, bounds all fuel types and features analyzed. Addition of the 5°F penalty for the increase in T_{AVG} associated with RTD bypass elimination yields a total PCT of 1894°F. This licensing basis PCT remains below the 2200°F acceptance criterium of 10 CFR 50.46. The maximum local metal-water reaction is well below the embrittlement limit of 17 percent as required by 10 CFR 50.46.

The total core metal-water reaction is less than 1.0 percent for all breaks analyzed, corresponding to less than 1.0 percent hydrogen generation as required by 10 CFR 50.46. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

TABLE 2-1
INPUT PARAMETERS USED IN THE
LARGE BREAK LOCA ECCS ANALYSIS

License Core Power* (MWt)	3411
Peak Linear Power for Fuel Rods* (kW/ft)	13.6
Total Peaking Factor, F_Q	2.5
Axial Peaking Factor, F_Z	1.515
Hot Channel Enthalpy Rise Factor, $F\Delta H$	1.65
Hot Assembly Average Power, P_{HA}	1.469
Power Shape	Chopped Cosine
Fuel Assembly Array	17x17 ZIRLO or Zircaloy-4, 0.374" Fuel Rods
Accumulator Water Volume (ft^3/accumulator)	850 (Nominal)
Accumulator Gas Pressure, Minimum (psia)	600
Safety Injection Pumped Flow (SIPs and CCPs degraded 5%, RHR degraded 8.75%, CCP flow imbalance = 10 gpm)	See Figures 2-2A and 2-2B
Containment Parameters	See Table 2-3
Initial Loop Flow (gpm/loop)	93800
Vessel Inlet Temperature (°F)	557.66
Vessel Outlet Temperature (°F)	619.34
Reactor Coolant Pressure (psia)	2300.0
Steam Pressure (psia)	955.7
Steam Generator Tube Plugging Level (%)	13**
Refueling Water Storage Tank Temperature for Containment Spray (°F)	50.0
Refueling Water Storage Tank Temperature for Safety Injection (°F)	100.0
Fuel Backfill Pressure (psig)	275 (100 for IFBA)
Low Pressurizer Pressure Setpoints (psia):	
Reactor Trip	1860
Safety Injection Signal	1665
Safety Injection Delay Time (sec)	30
Safety Injection Spilling Containment Pressure (psig)	0.0
Feedwater Isolation Delay after Reactor Trip (sec)	0.0
Steamline Isolation Delay after Reactor Trip (sec)	0.0
Blowdown Containment Pressure (psia)	36.5

* Two percent is added to this power to account for calorimetric error.

** Analysis is performed at 13% SGTP in order to support operation with 8% SGTP. 5% SGTP is allocated for steam generator tube crush during a combined LOCA/seismic event.

TABLE 2-2
LARGE BREAK LOCA CONTAINMENT DATA

Net Free Volume*	2,974,000 ft^3
Initial Conditions	
Pressure	14.6 psia
Temperature	90.0°F
RWST Temperature	50.0°F
Temperature Outside Containment	50.0°F
Spray Temperature	50.0°F
Spray System	
Spray System Flow Rate	6000 gpm
Starting Time for Spray	47 sec
Temperature of Spray Water	50°F

* The containment net free volume modeled in the analyses was adjusted from this value to account for containment purge effects.

TABLE 2-2 (Continued)
LARGE BREAK LOCA CONTAINMENT DATA

Structural Heat Sinks

Wall	T_{Air} [°F]	Area [ft^3]	T_{Init} [°F]	Thickness [inches]
1	50	70151	90	0.009 Paint/ 0.375 Carbon Steel/ 54.0 Concrete
2	50	33856	90	0.009 Paint/ 0.50 Carbon Steel/ 42.0 Concrete
3	90	106165	90	0.017 Paint/ 53.5 Concrete
4	90	7873	90	0.199 Stainless Steel/ 33.1 Concrete
5	90	74419	90	0.174 Carbon Steel
6	90	71157	90	0.018 Paint/ 0.56 Carbon Steel
7	90	21912	90	0.018 Paint/ 1.43 Carbon Steel
8	90	2580	90	0.009 Paint/ 1.55 Carbon Steel
9	90	12804	90	0.009 Paint/ 0.250 Carbon Steel/ 167 Concrete
10	50	3575	90	0.009 Paint/ 0.844 Carbon Steel
11	50	396	90	0.009 Paint/ 0.816 Carbon Steel

1. - Containment Cylinder
2. - Containment Dome
3. - Miscellaneous Concrete
4. - Refueling Canal
5. - Ducts and Trays
6. - Structural Steel
7. - Polar Crane
8. - Equipment Steel
9. - Containment Floor and Sump
10. - Equipment Hatch
11. - Personnel Hatch

TABLE 2-3
LARGE BREAK LOCA - CASES ANALYZED

CASE I -	C_D=0.4, 3411 MWt Core Power, F_Q=2.5, $F\Delta H$=1.65, P-BAR-HA=1.469, Minimum Safeguards.
CASE II -	C_D=0.6, 3411 MWt Core Power, F_Q=2.5, $F\Delta H$=1.65, P-BAR-HA=1.469, Minimum Safeguards. This case was found to be limiting and bounds both ZIRLO and Zircaloy-4 cladding.
CASE III -	C_D=0.8, 3411 MWt Core Power, F_Q=2.5, $F\Delta H$=1.65, P-BAR-HA=1.469, Minimum Safeguards.
CASE IV -	C_D=0.6, 3411 MWt Core Power, F_Q=2.5, $F\Delta H$=1.65, P-BAR-HA=1.469, Maximum Safeguards.

All cases model 13% steam generator tube plugging (in order to bound operation with 8% SGTP) and 2% reduction in thermal design flow (93800 gpm/loop).

TABLE 2-4
LARGE BREAK LOCA RESULTS - TIME SEQUENCE OF EVENTS

	Case I C_D=0.4 Min SI	Case II C_D=0.6 Min SI	Case III C_D=0.8 Min SI	Case IV C_D=0.6 Max SI
Start of LOCA with LOOP (sec)	0.00	0.00	0.00	0.00
Reactor Trip Setpoint Exceeded (sec)	0.757	0.735	0.720	0.735
Safety Injection Setpoint Exceeded (sec)	2.09	1.67	1.45	1.67
Accumulator Injection Begins (sec)	20.0	15.0	12.5	15.0
End-of-Bypass (sec)	39.2	33.0	28.9	33.0
End-of-Blowdown (sec)	39.2	33.3	28.9	33.3
Pump Injection Begins (sec)	32.1	31.7	31.5	31.7
Bottom of Core Recovery (sec)	54.5	47.8	43.8	47.4
Accumulators Empty (sec)	61.1	53.3	50.1	53.8

TABLE 2-5
LARGE BREAK LOCA RESULTS - FUEL CLADDING DATA

	Case I C_D=0.4 Min SI	Case II C_D=0.6 Min SI	Case III C_D=0.8 Min SI	Case IV C_D=0.6 Max SI
Peak Clad Temperature (°F)	1682	1894*	1823	1762
Peak Clad Temperature Location (ft)	8.75	7.00	7.00	6.25
Peak Clad Temperature Time (sec)	281	128	92.6	59.8
Maximum Local Zr/H_2O Reaction (%)	1.95	3.41	2.69	1.87
Maximum Zr/H_2O Reaction Location (ft)	6.75	7.00	7.00	7.00
Total Zr/H_2O Reaction (%)	<1.0	<1.0	<1.0	<1.0
Hot Rod Burst Time (sec)	88.7	55.1	47.8	46.9
Hot Rod Burst Location (ft)	6.75	6.00	6.00	5.50

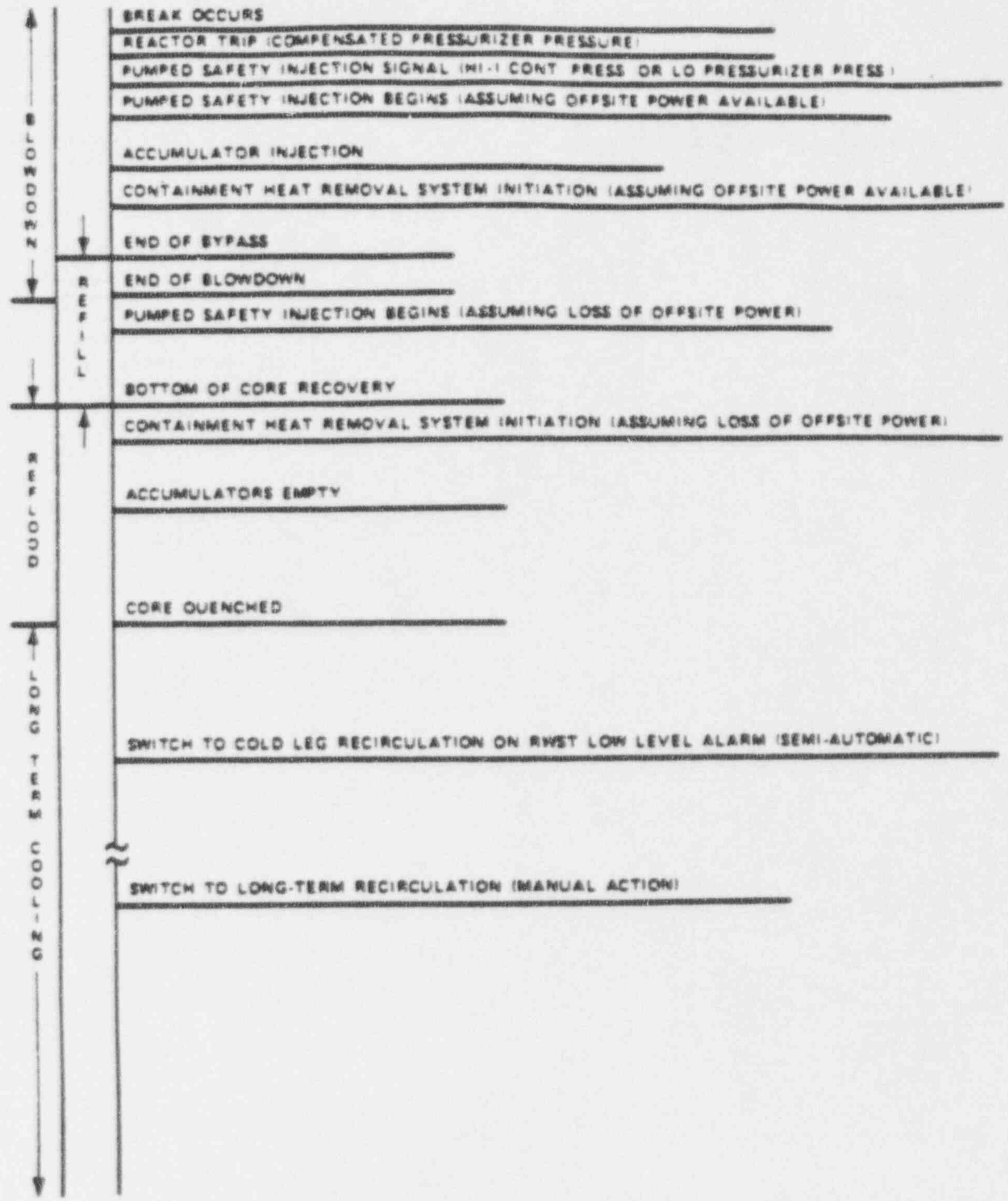
* Includes 5.0°F evaluation penalty for Increased Temperature Uncertainty (±5°F) associated with RTD Bypass Elimination.

TABLE 2-6
LARGE BREAK LOCA C_D=0.6 MINIMUM SAFEGUARDS
LOCA REFLOOD MASS AND ENERGY RELEASE RATES

Time (sec)	Mass Flow Rate (lbm/sec)	Energy Flow Rate (BTU/sec)
47.80	0.0	0.0
50.00	9.14	11859
56.78	47.08	59471
60.03	52.13	65527
70.08	66.95	83306
72.88	71.12	88310
80.08	81.51	100781
90.08	94.39	116239
90.88	95.35	117391
100.08	105.91	130062
108.88	115.17	141170
110.08	116.37	142610
120.08	126.90	154213
126.58	214.10	182311
130.08	262.26	196636
140.08	323.03	212249
143.88	330.59	213119
150.08	337.12	212641
160.08	342.59	210280
161.88	343.32	209762
170.03	389.81	220922
180.03	421.50	227259
190.03	433.60	228914
200.03	429.70	226158
203.73	430.20	225608
210.03	429.79	224385
220.03	431.49	223140
230.03	433.81	222068
240.03	437.88	221466
250.03	440.41	220476
260.03	443.17	219641
270.03	445.21	218627
274.73	445.83	218063
280.03	446.64	217485
290.03	449.22	216715
300.03	450.67	215667

TABLE 2-7
LARGE BREAK LOCA C_D=0.6 MINIMUM SAFEGUARDS
LOCA BLOWDOWN MASS AND ENERGY RELEASE RATES

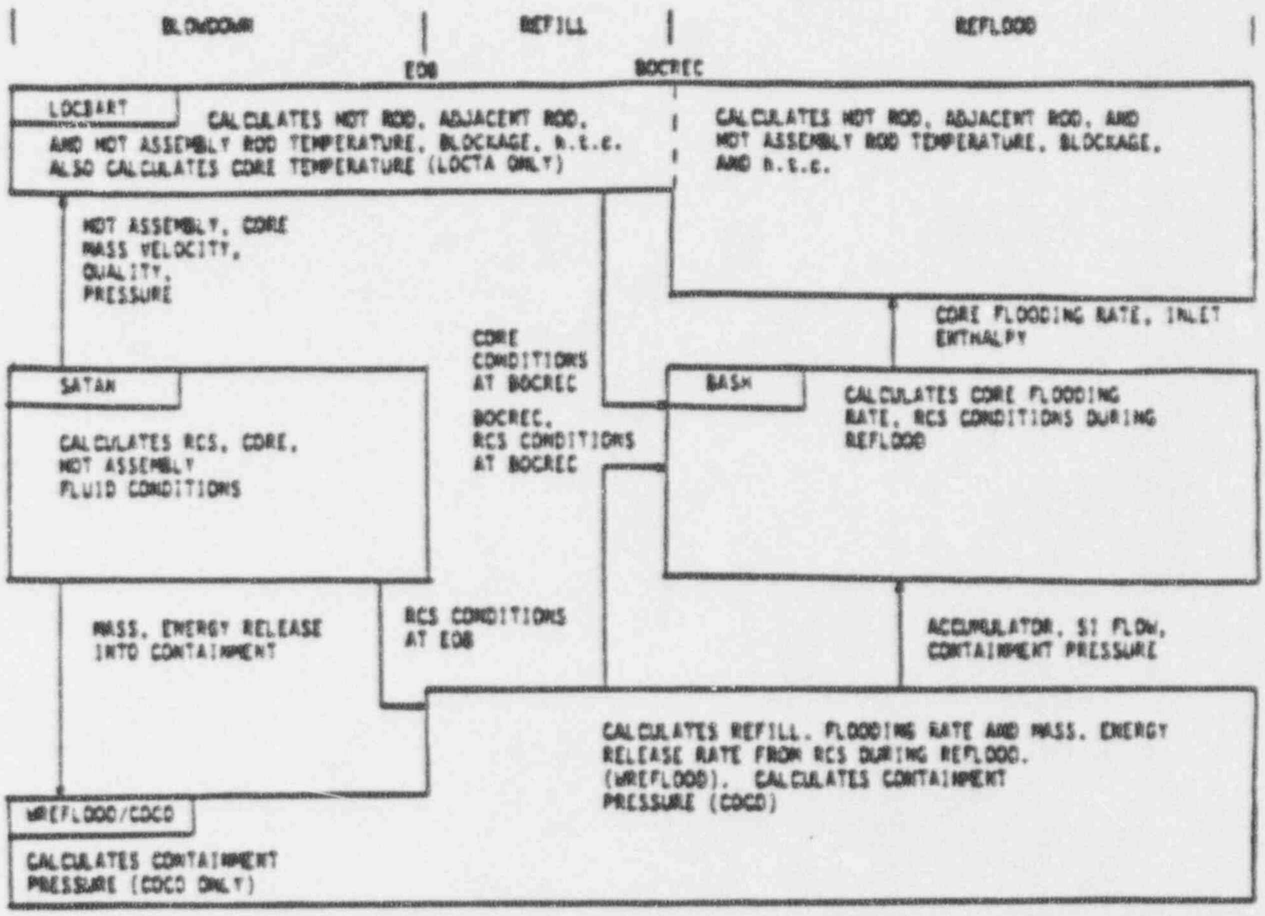
Time (sec)	Mass Flow Rate (lbm/sec)	Energy Flow Rate (BTU/sec)
0.0	2954	184950
1.0	2760	172772
2.0	2602	162860
3.0	2469	154548
4.0	2355	147421
5.0	2256	141197
6.0	2167	135669
7.0	2008	130709
8.0	2016	126227
9.0	1951	122150
10.0	1892	118420
11.0	1837	114993
12.0	1786	111835
13.0	1740	108908
14.0	1696	106182
15.0	1656	103638
16.0	1618	101265
17.0	1582	99047
18.0	1549	96975
19.0	1518	95032
20.0	1489	93195
21.0	1461	91463
22.0	1435	89831
23.0	1410	88288
24.0	1387	86827
25.0	1365	85447
26.0	1344	84146
27.0	1324	82910
28.0	1306	81727
29.0	1287	80595
29.51	0.0	0.0



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

LARGE BREAK LOCA SEQUENCE OF EVENTS

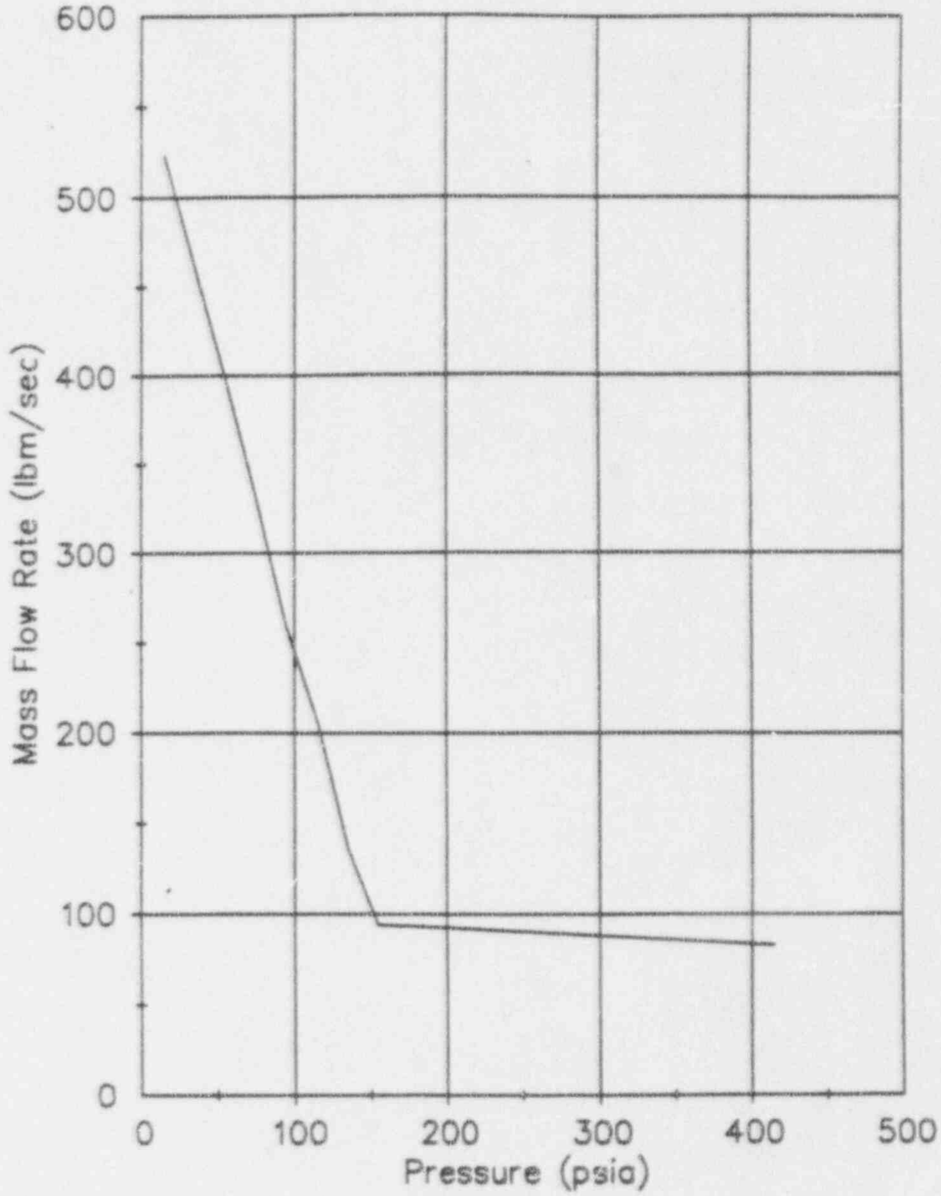
FIGURE 2-1A



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

LARGE BREAK LOCA CODE INTERFACE

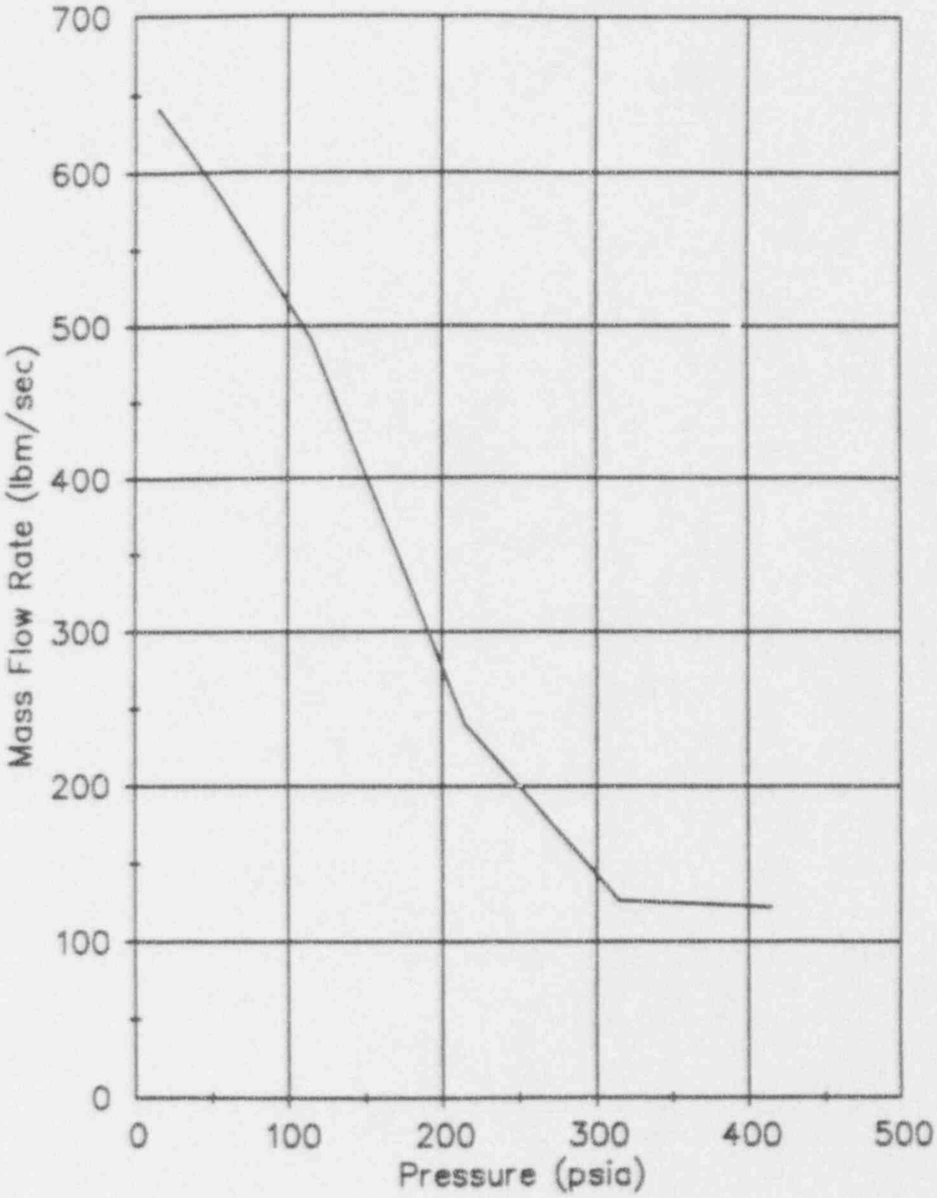
FIGURE 2-1B



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

PUMPED SAFETY INJECTION FLOW
vs. RCS PRESSURE
(MINIMUM SAFEGUARDS)

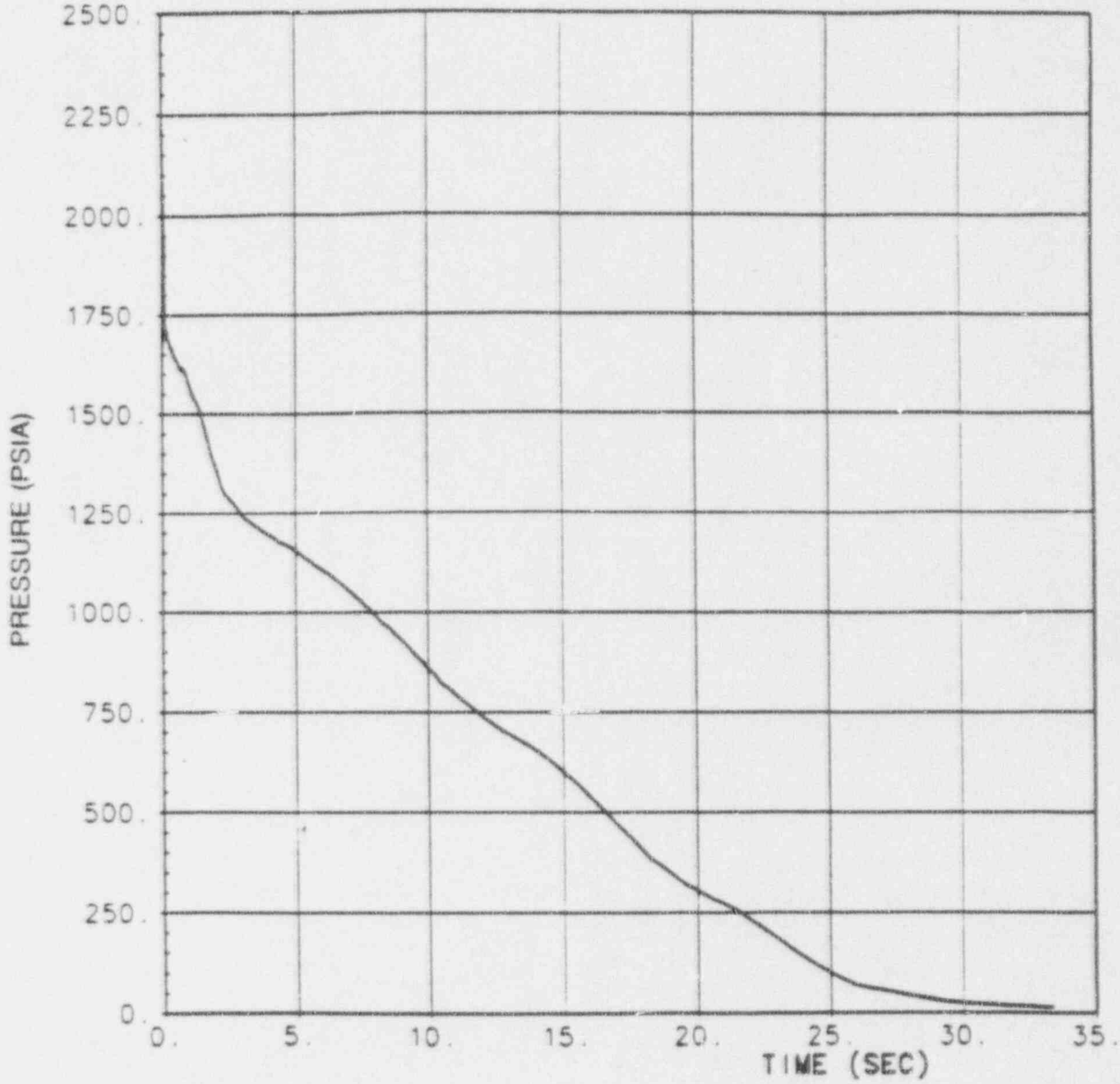
FIGURE 2-2A



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

PUMPED SAFETY INJECTION FLOW
vs. RCS PRESSURE
(MAXIMUM SAFEGUARDS)

FIGURE 2-2B

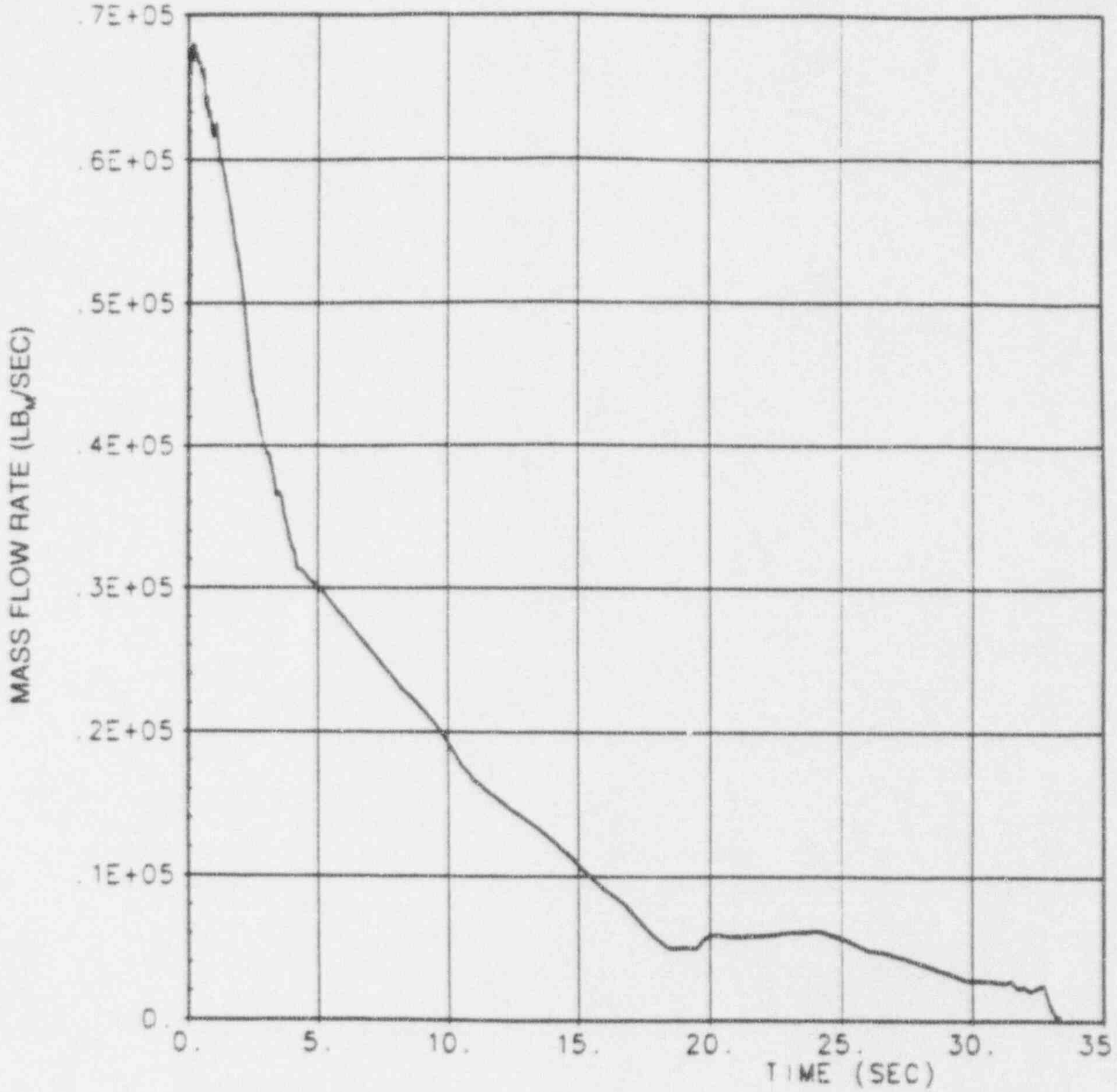


WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

RCS PRESSURE

(C_D = 0.6)

FIGURE 2-3A

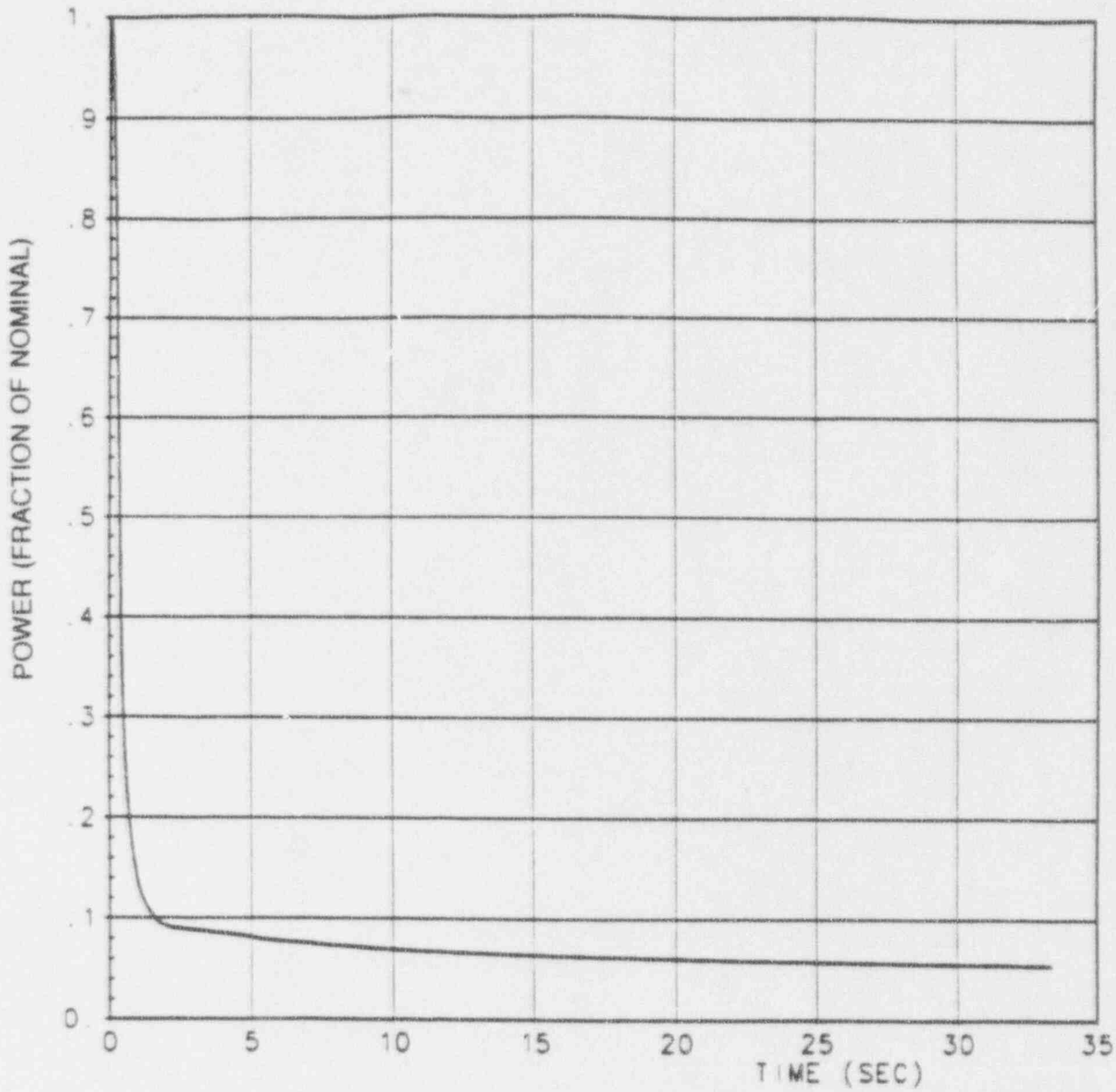


WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

COLD LEG BREAK MASS FLOW RATE

(C_D = 0.6)

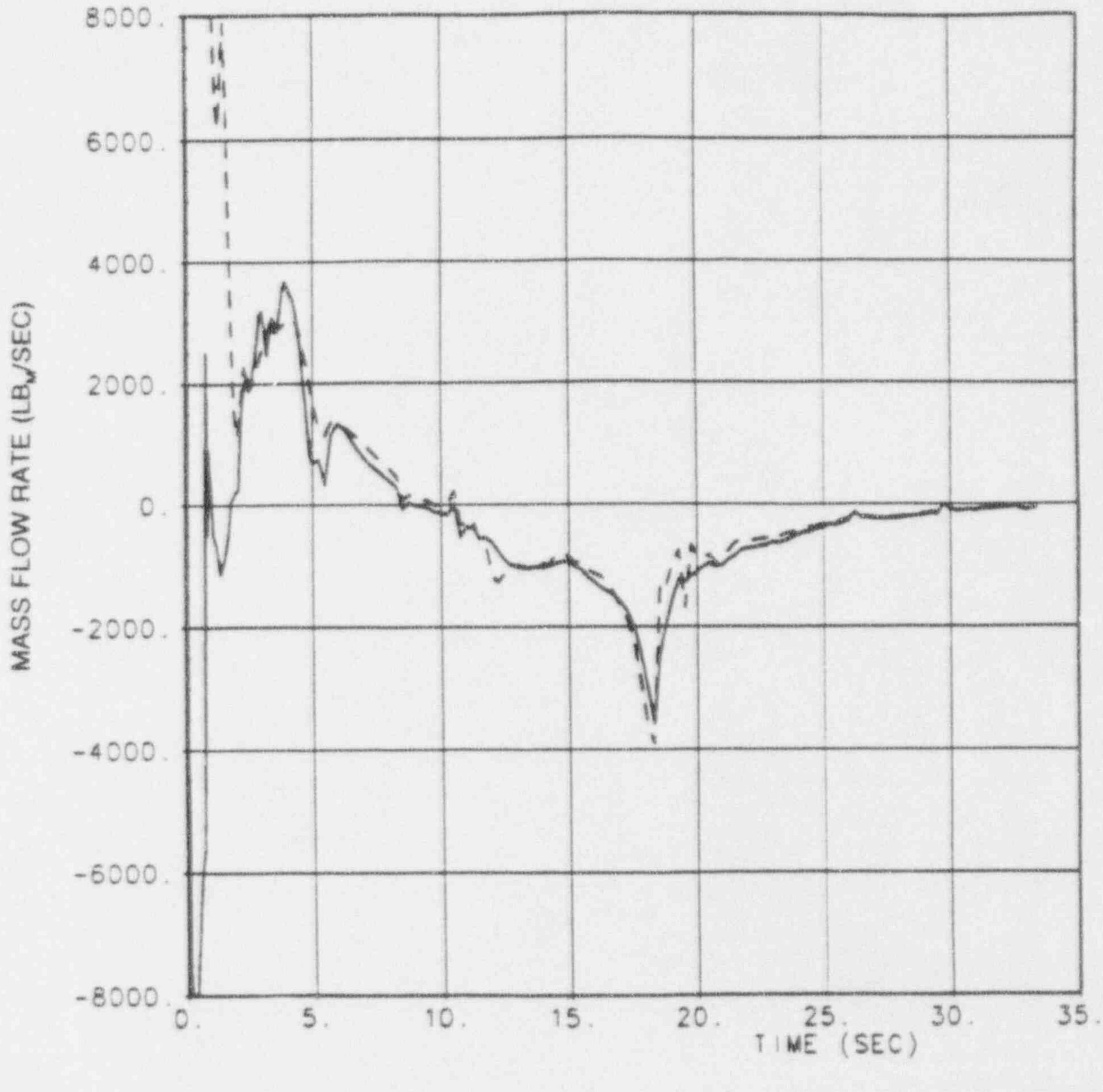
FIGURE 2-3B



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

CORE POWER
(FRACTION OF NOMINAL)
(C_D = 0.6)

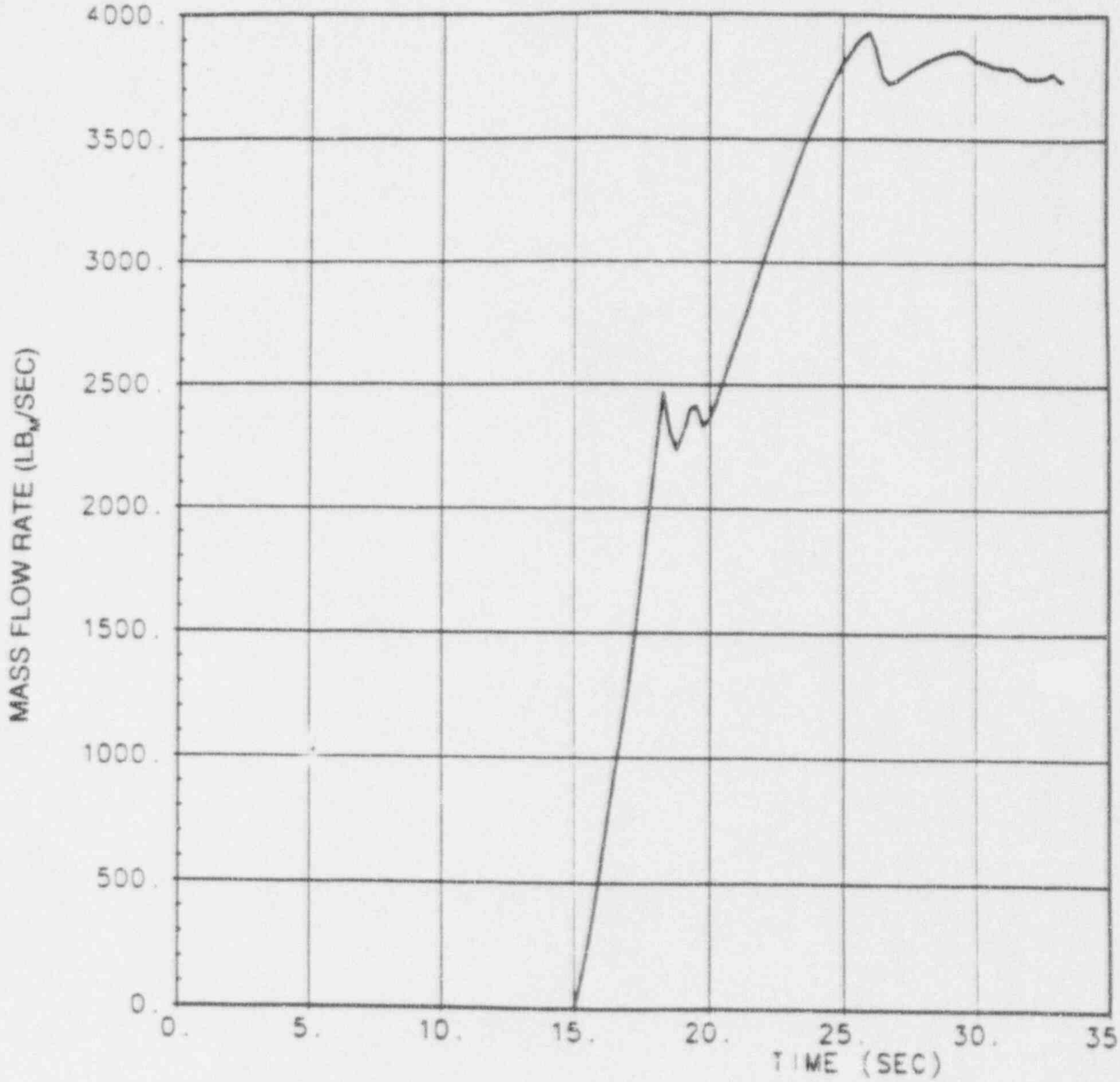
FIGURE 2-3C



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

CORE MASS FLOW RATE
(TOP AND BOTTOM)
(C_D = 0.6)

FIGURE 2-3D

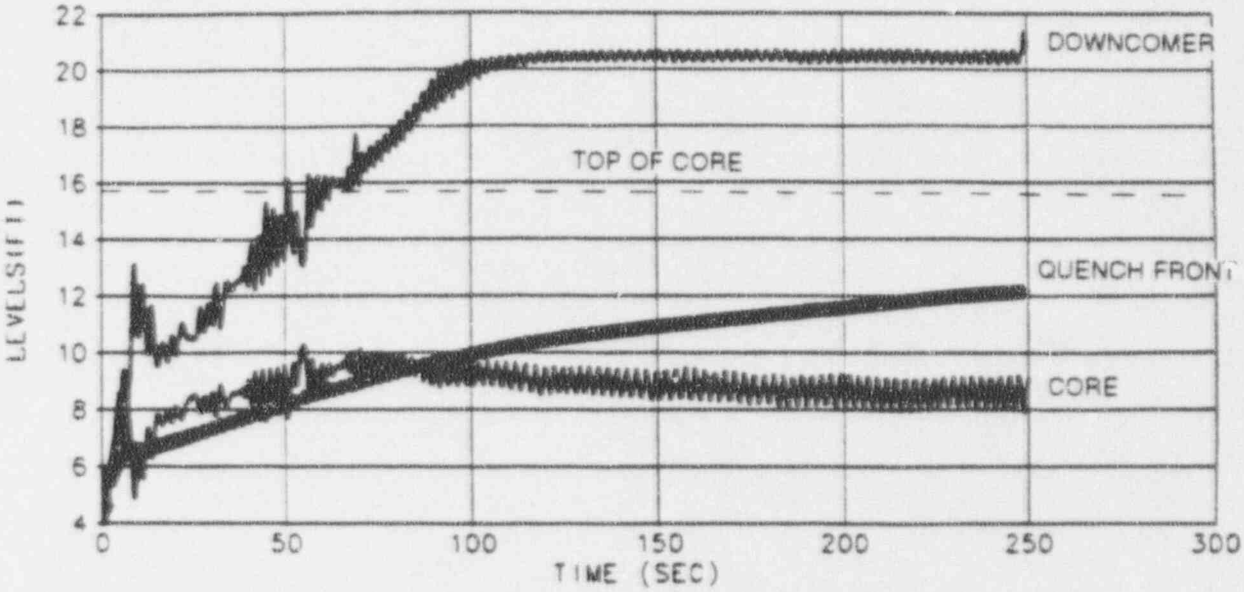


WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

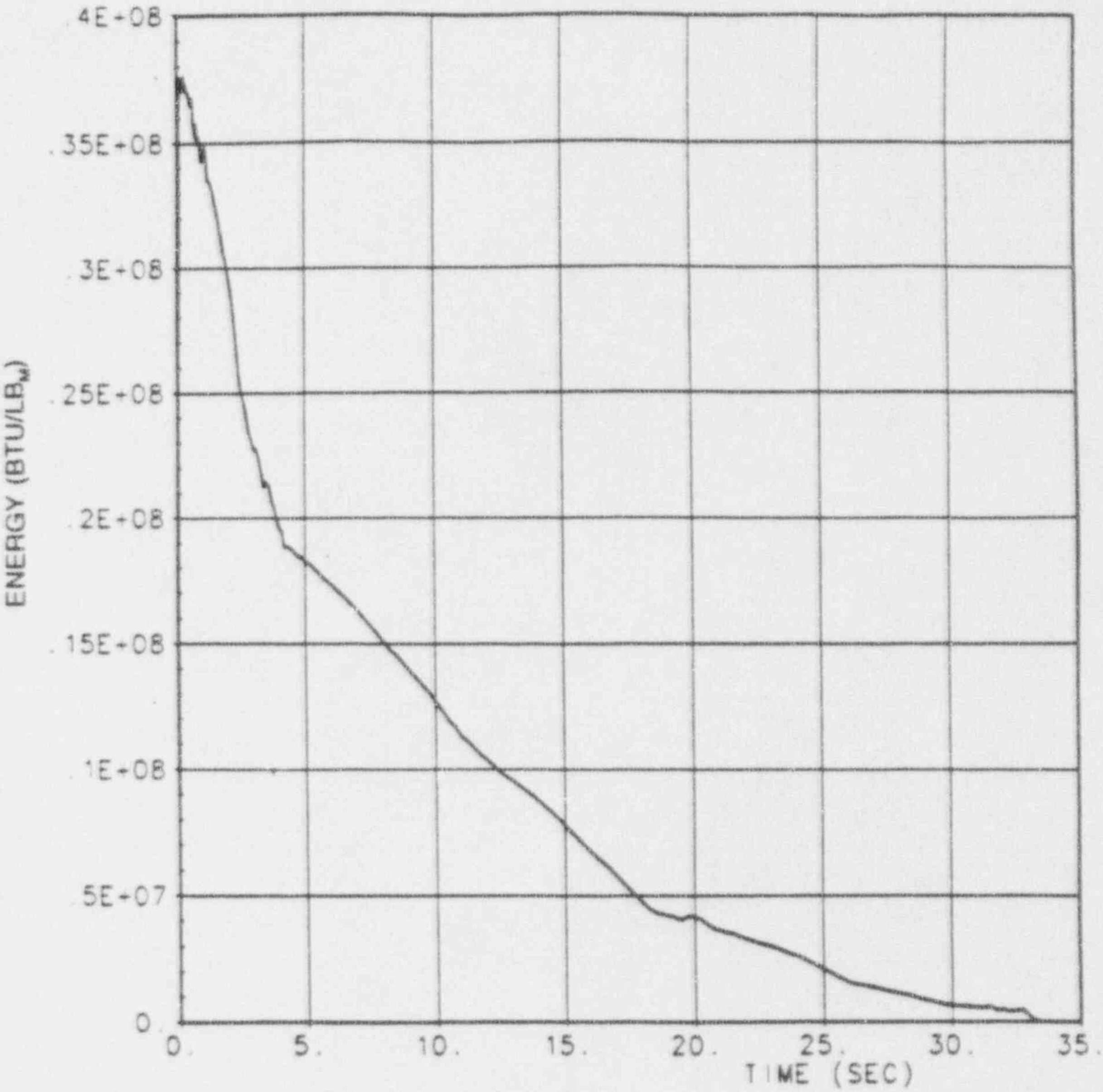
ACCUMULATOR MASS FLOW RATE

(C_D = 0.6)

FIGURE 2-3E



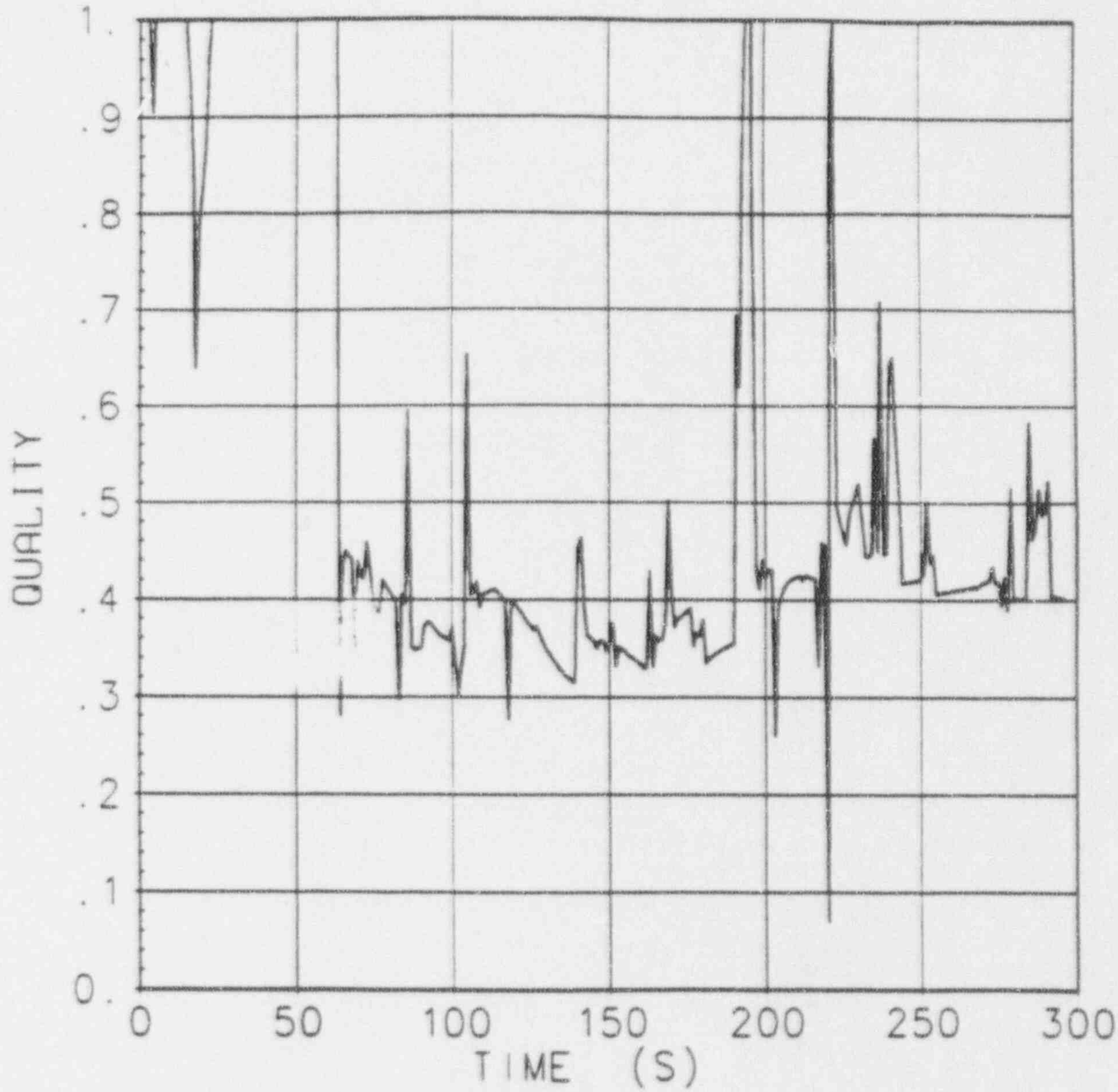
WESTINGHOUSE ELECTRIC CORPORATION Nuclear and Advanced Technology Divisions	REFLOOD CORE AND DOWNCOMER WATER LEVELS (C_D = 0.6)
	FIGURE 2-3F



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

BREAK ENERGY RELEASED TO CONTAINMENT (C_D = 0.6)

FIGURE 2-3G

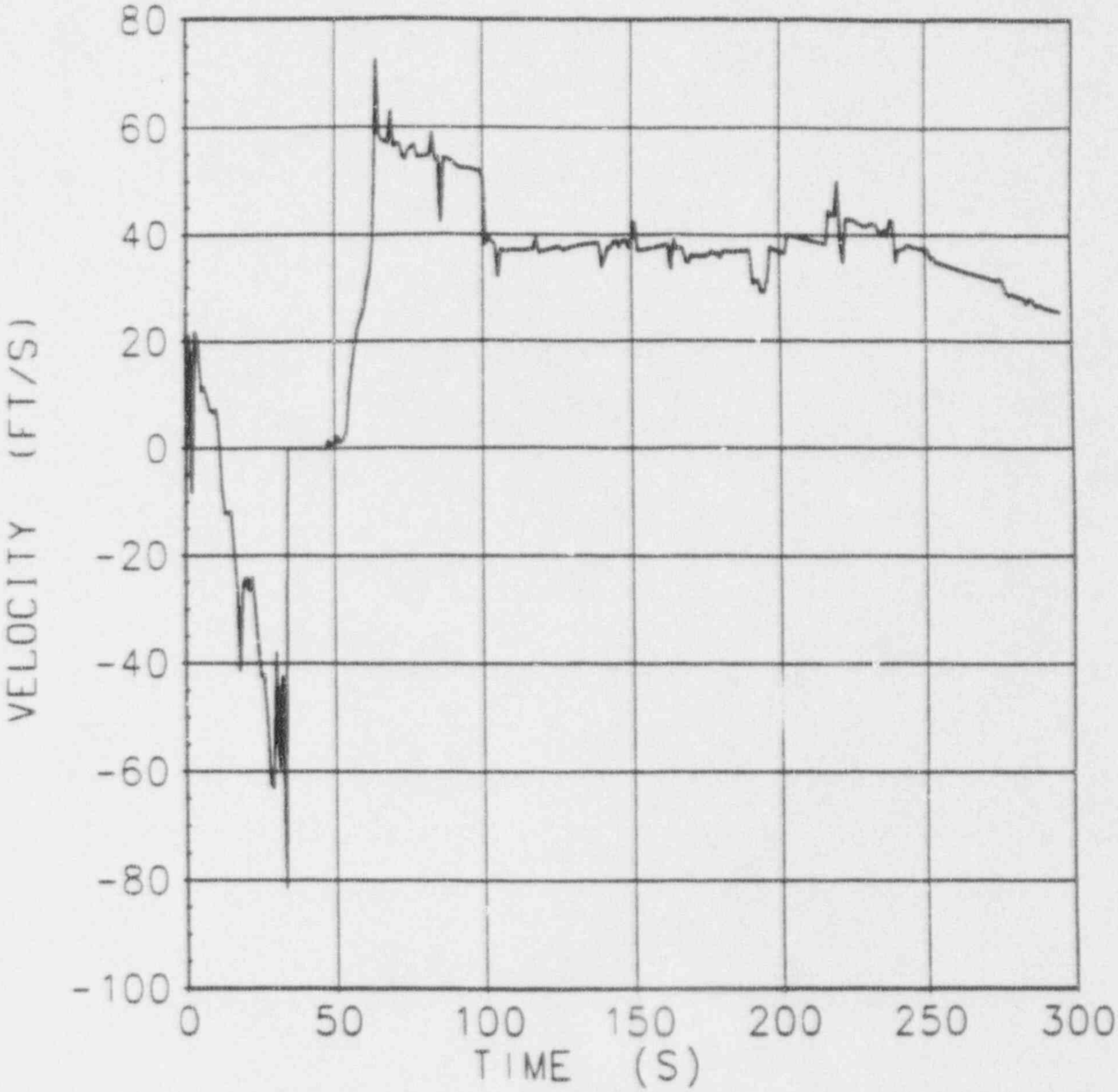


WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

FLUID QUALITY

(C_D = 0.6)

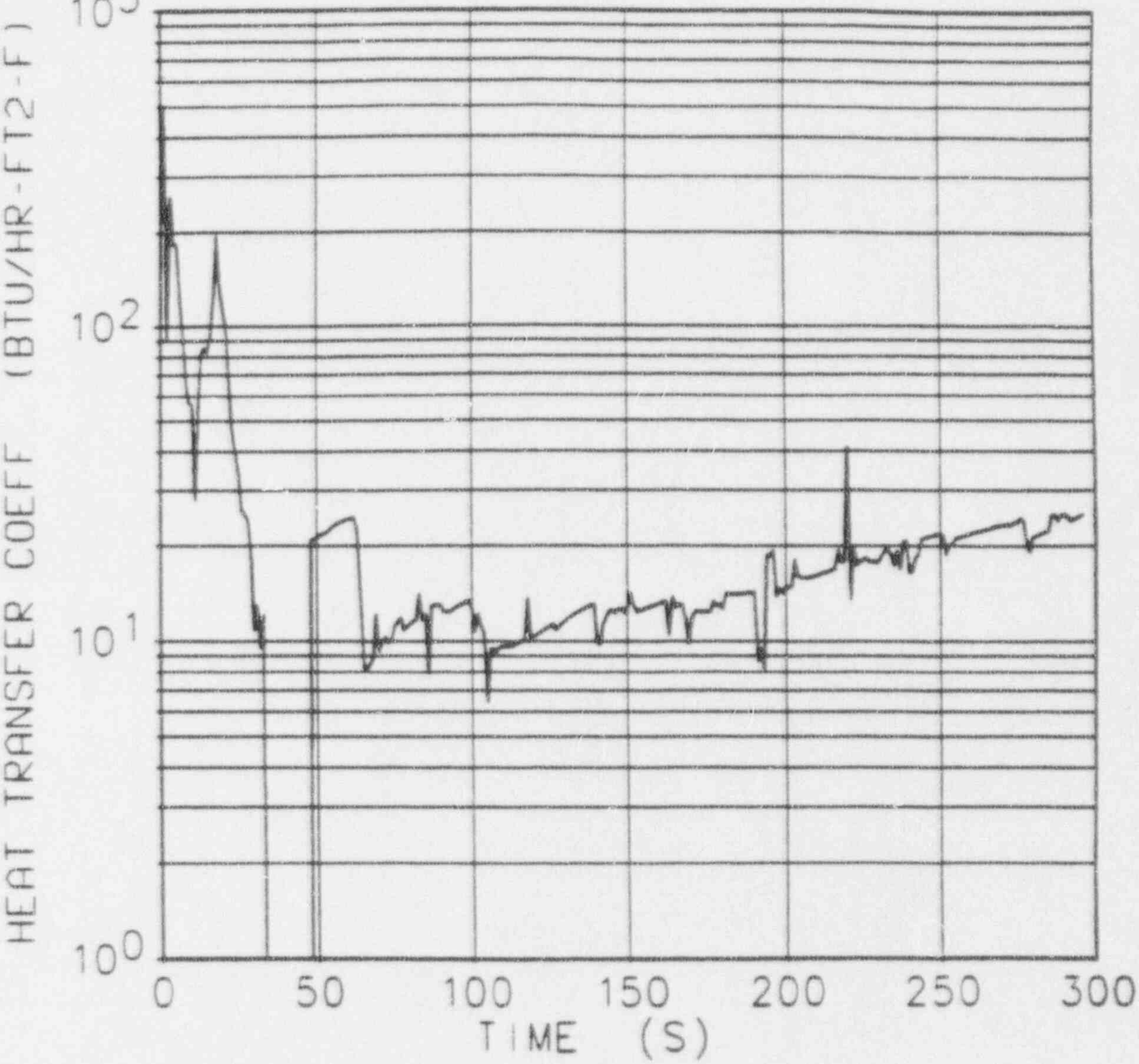
FIGURE 2-3H



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

FLUID VELOCITY PAST CLAD HOT SPOT (C_D = 0.6)

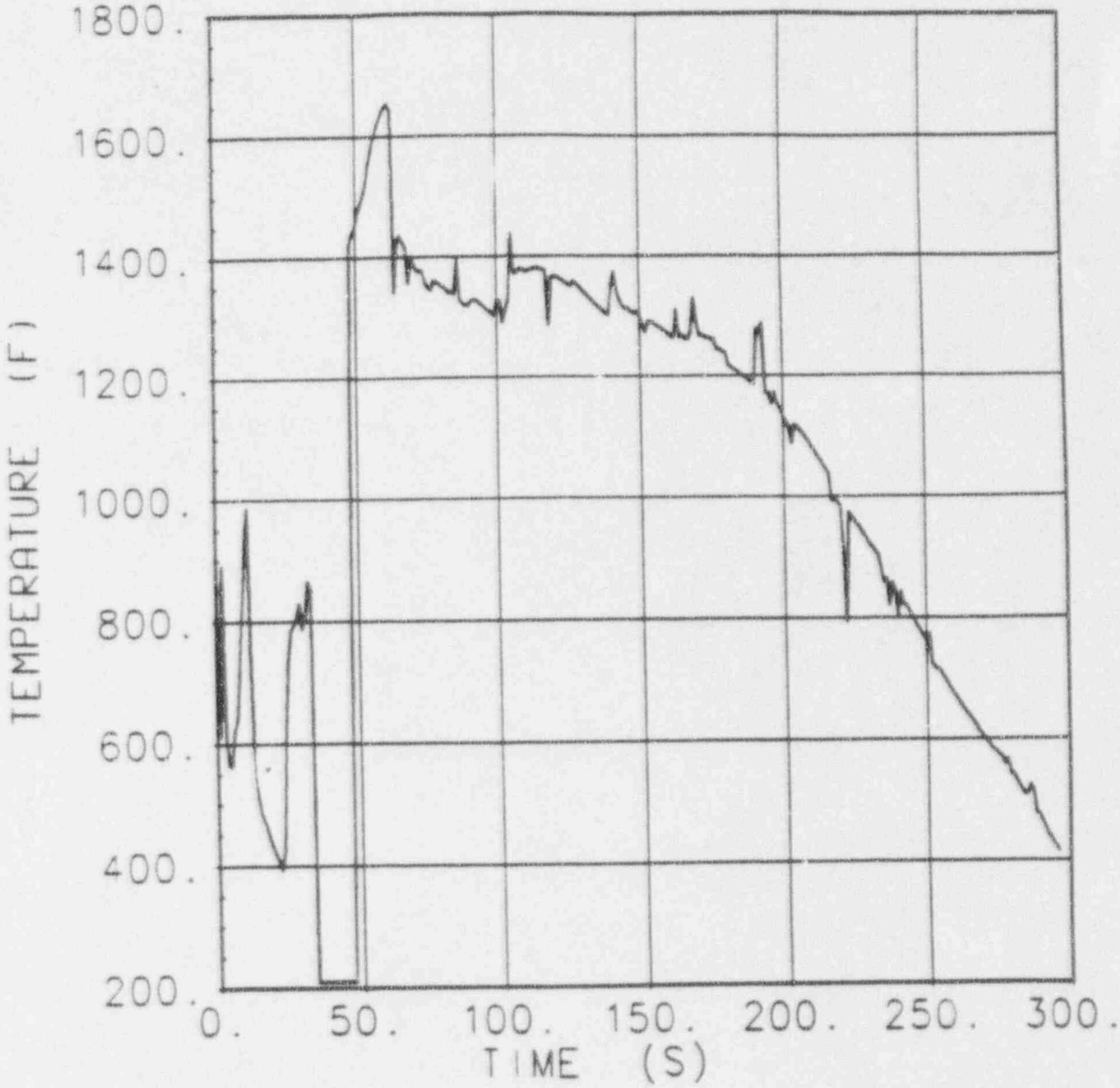
FIGURE 2-31



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

HOT ROD HEAT TRANSFER COEFFICIENT (C_D = 0.6)

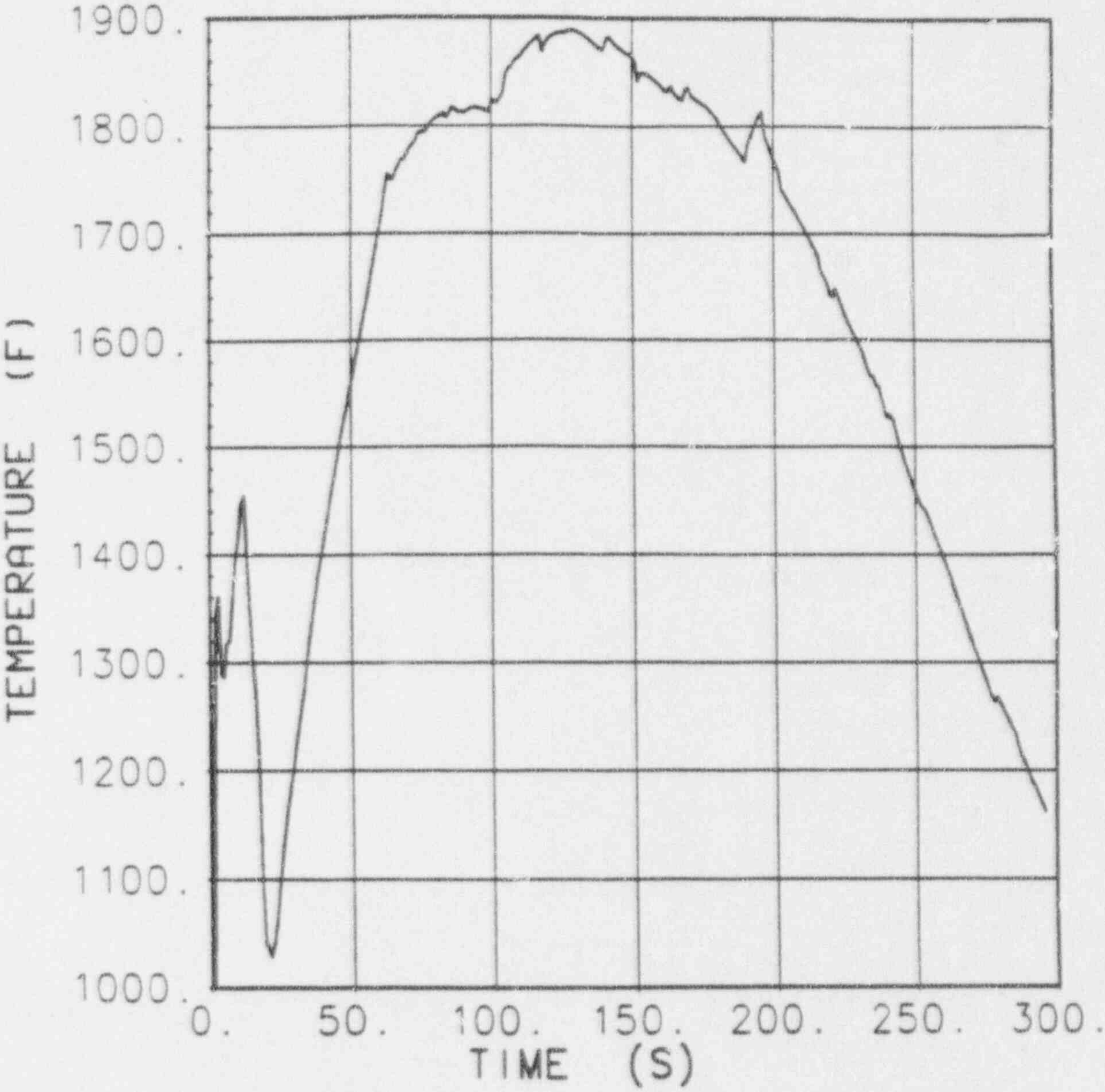
FIGURE 2-3J



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

CLAD HOT SPOT FLUID TEMPERATURE
(C_D = 0.6)

FIGURE 2-3K

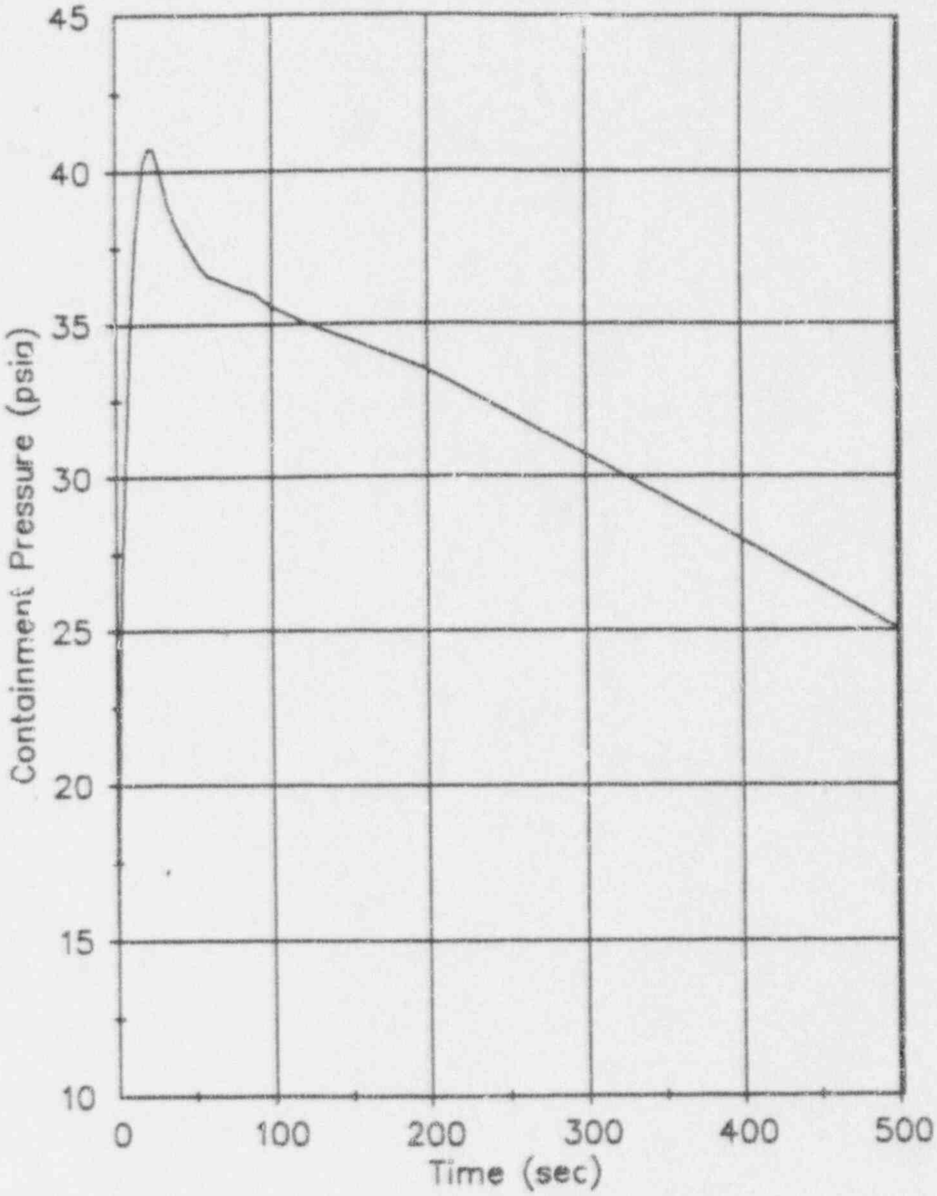


WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

HOT ROD PEAK CLAD TEMPERATURE

(C_D = 0.6)

FIGURE 2-3L

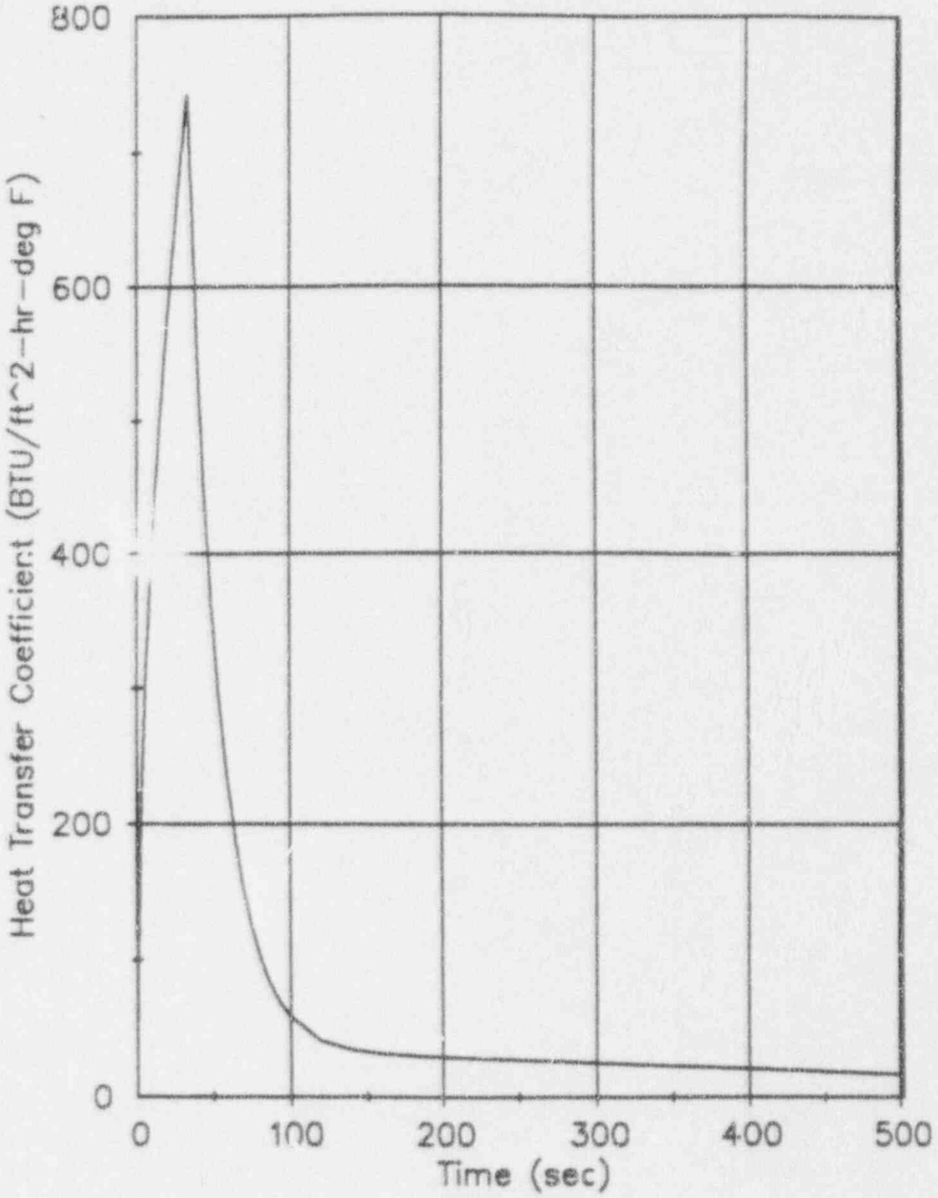


WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

CONTAINMENT PRESSURE

(C_D = 0.6)

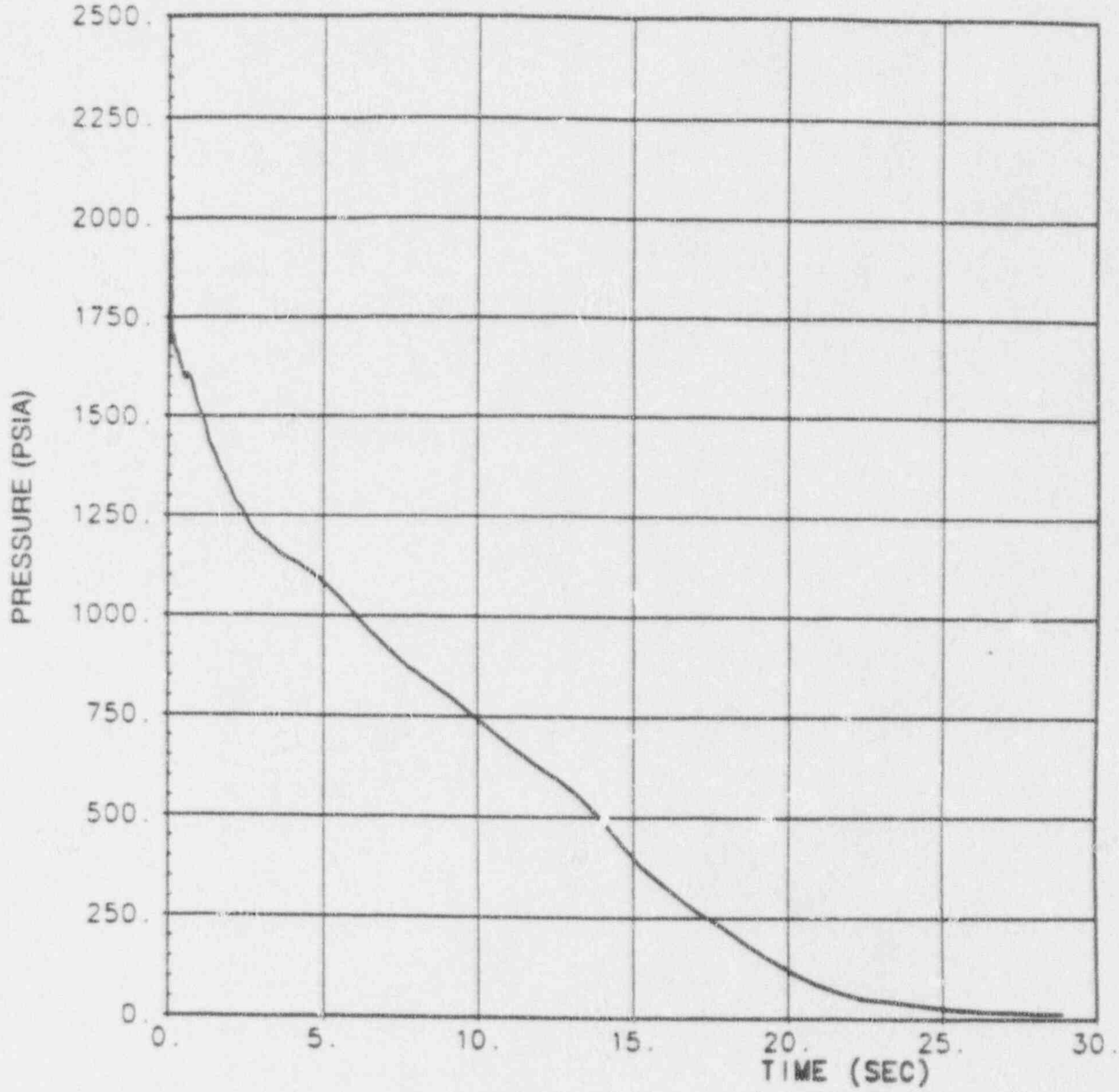
FIGURE 2-3M



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

CONTAINMENT CONDENSING WALL HEAT TRANSFER COEFFICIENT (C_D = 0.6)

FIGURE 2-3N

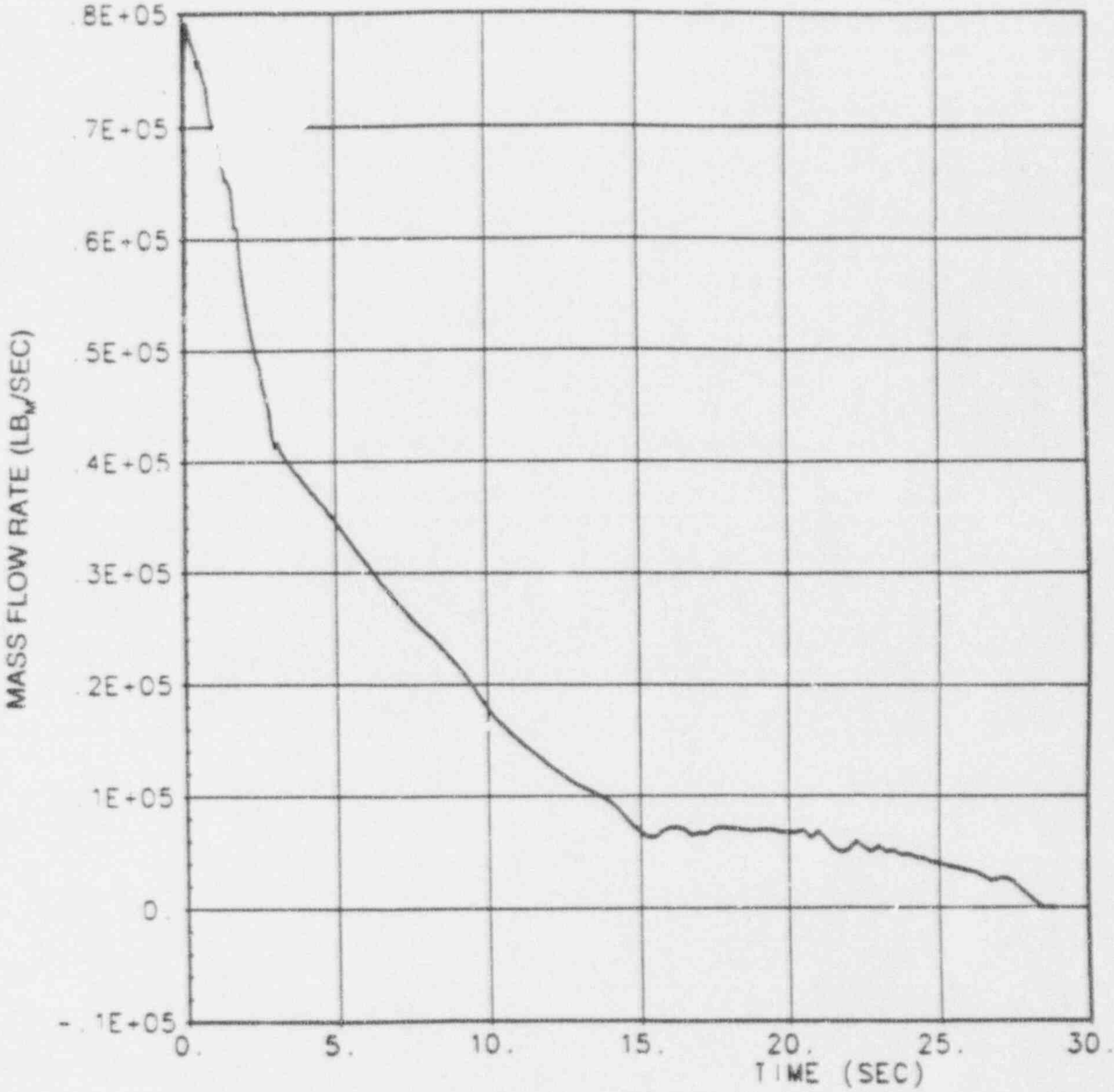


WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

RCS PRESSURE

(C_D = 0.8)

FIGURE 2-4A

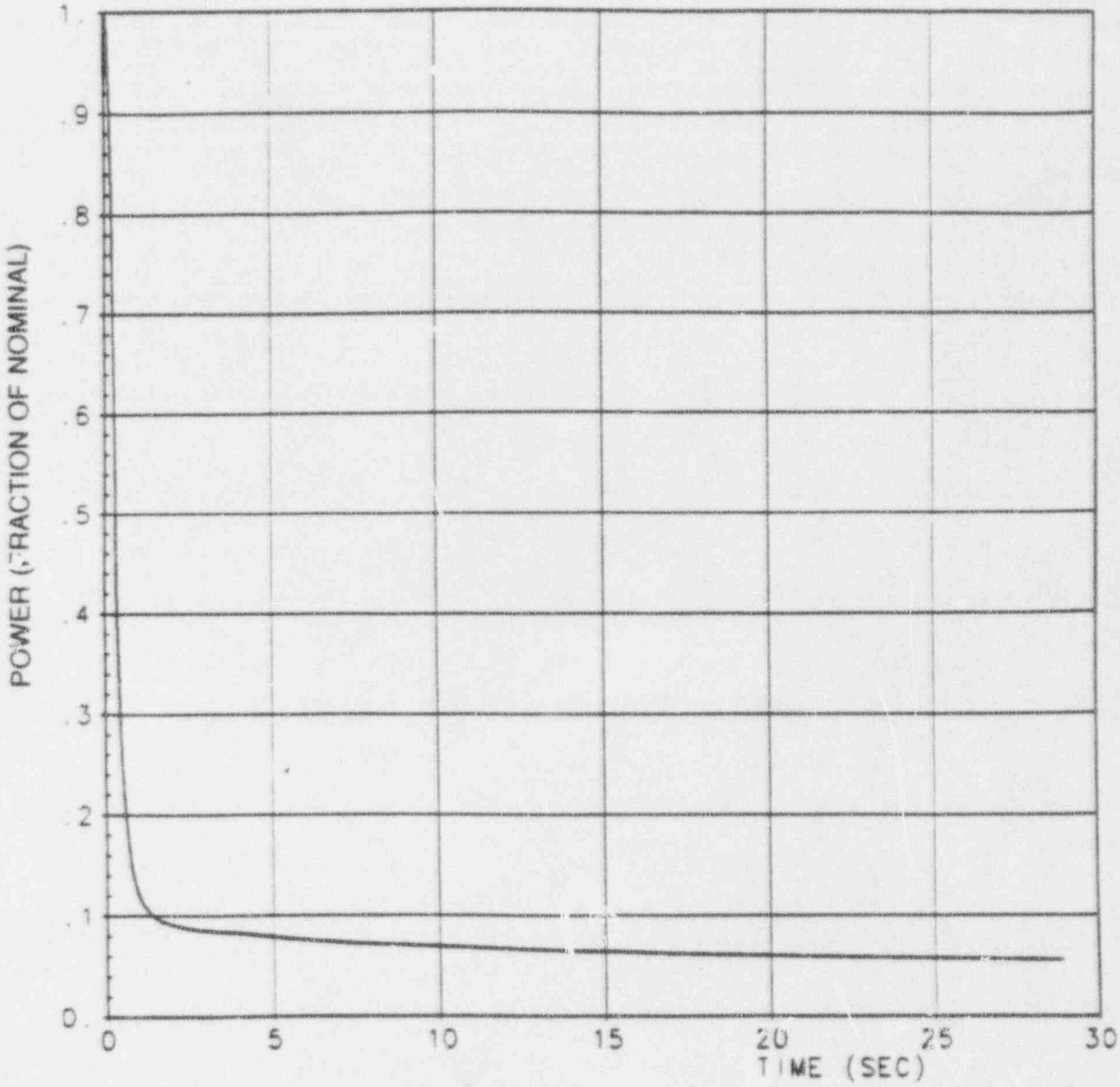


WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

COLD LEG BREAK MASS FLOW RATE

(C_D = 0.8)

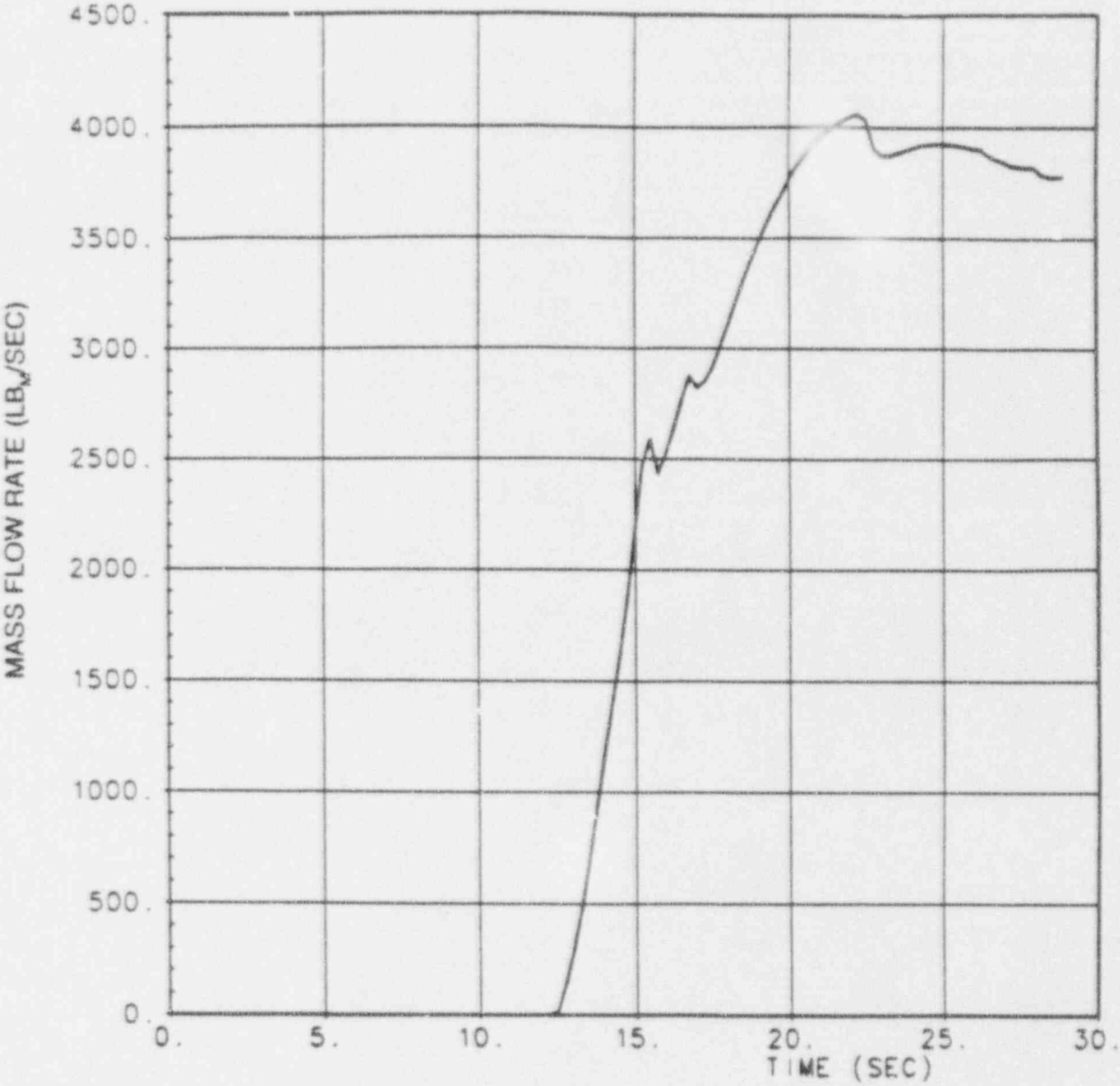
FIGURE 2-4B



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

CORE POWER
(FRACTION OF NOMINAL)
(C_D = 0.8)

FIGURE 2-4C

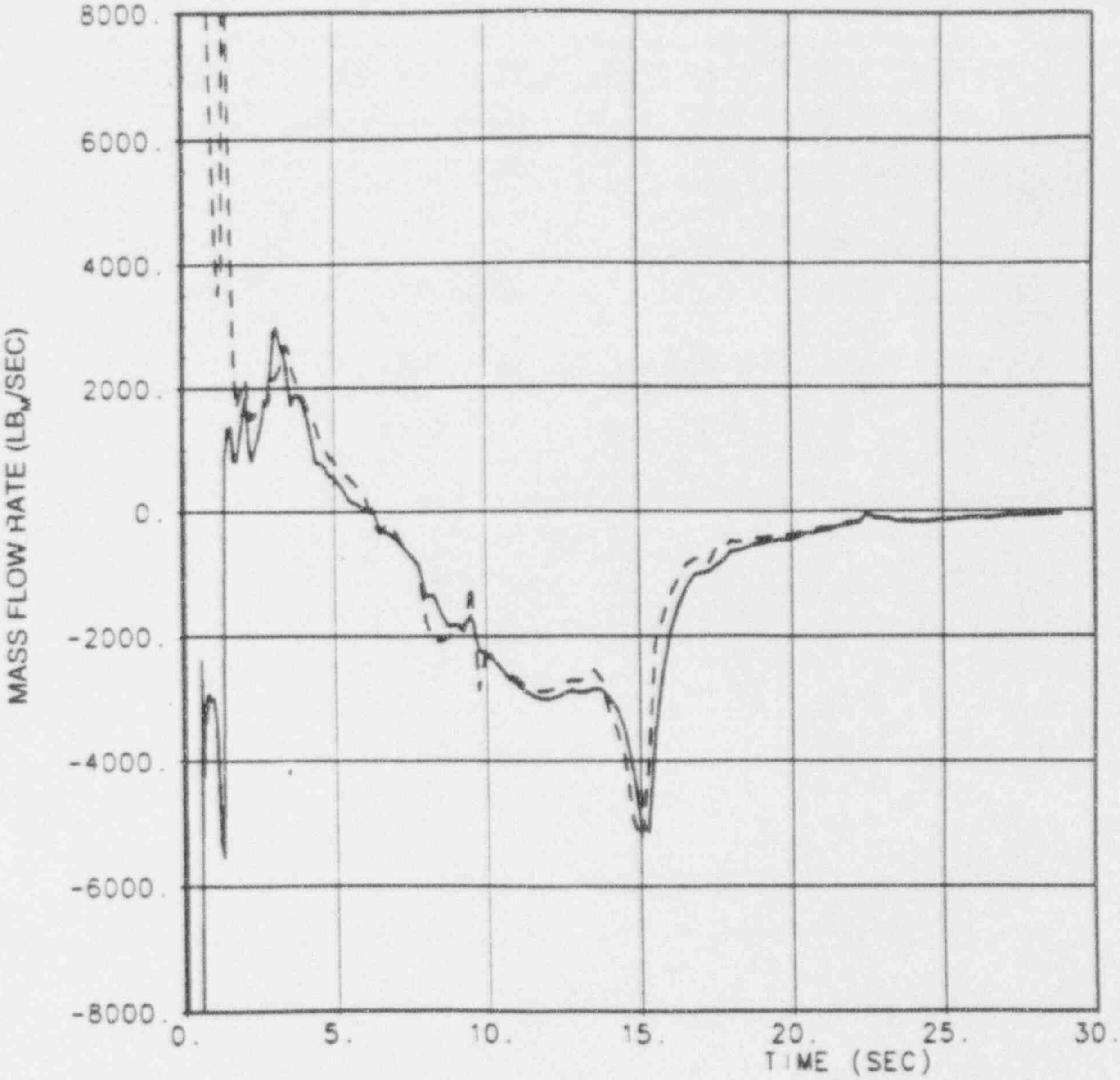


WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

ACCUMULATOR MASS FLOW RATE

(C_D = 0.8)

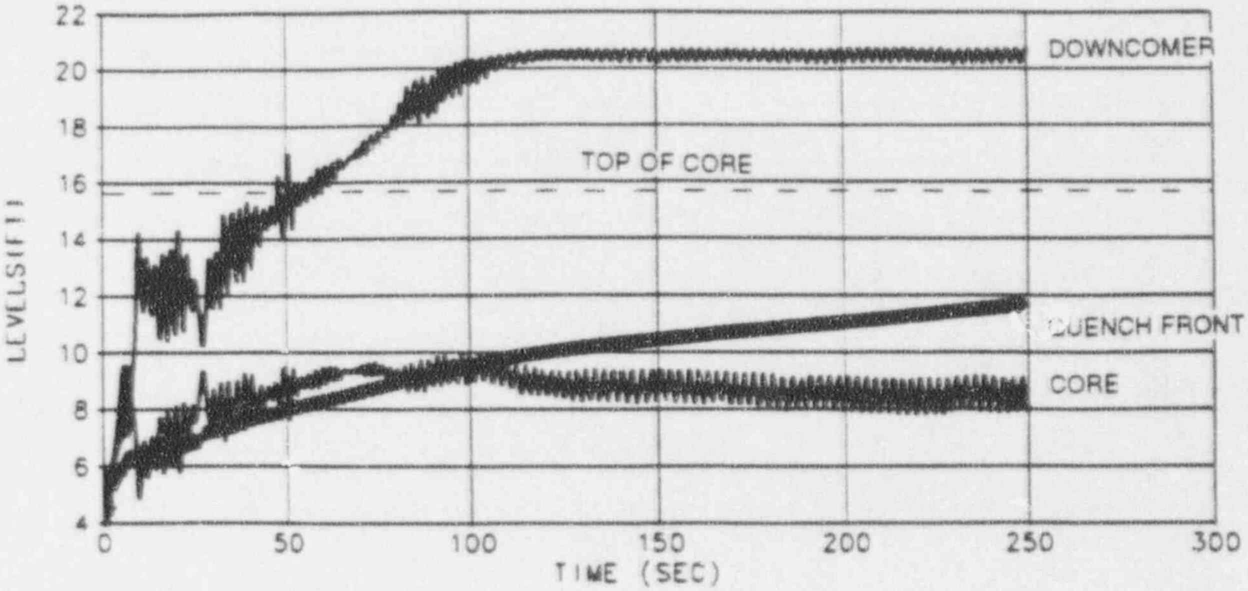
FIGURE 2-4D



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

CORE MASS FLOW RATE
(TOP AND BOTTOM)
(C_D = 0.8)

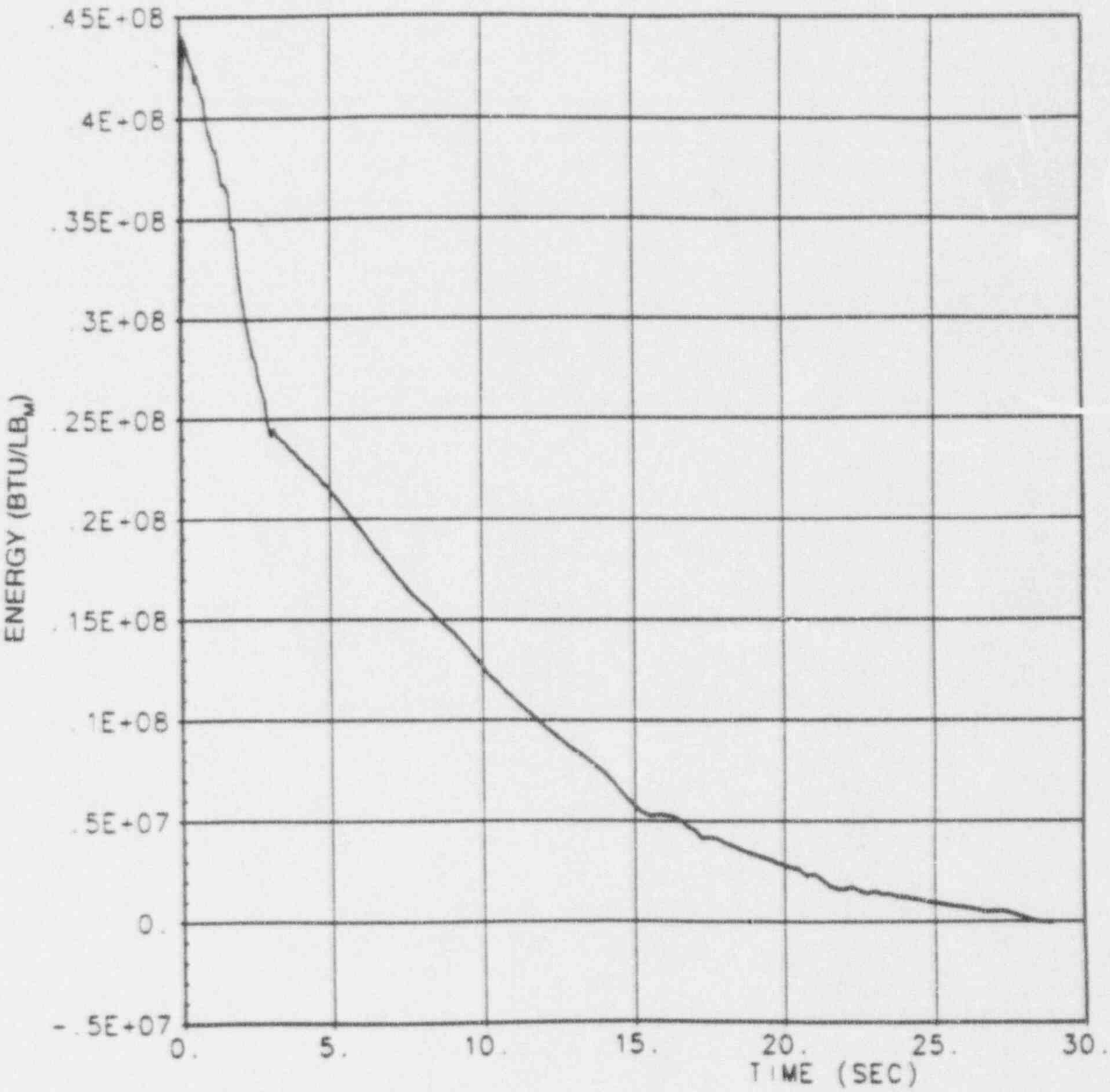
FIGURE 2-4E



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

REFLOOD CORE AND DOWNCOMER WATER LEVELS (C_D = 0.8)

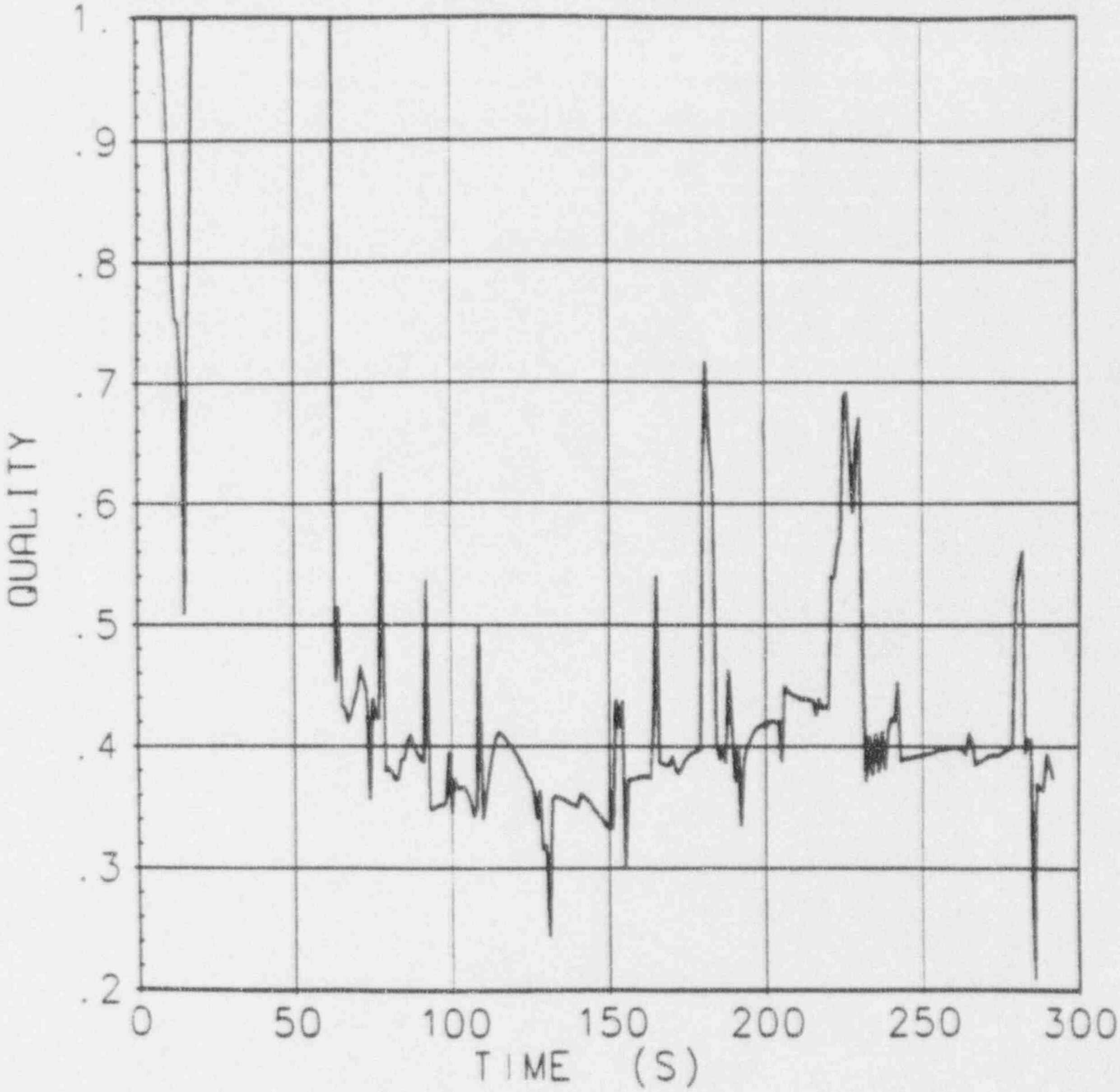
FIGURE 2-4F



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

BREAK ENERGY RELEASED TO CONTAINMENT (C_D = 0.8)

FIGURE 2-4G

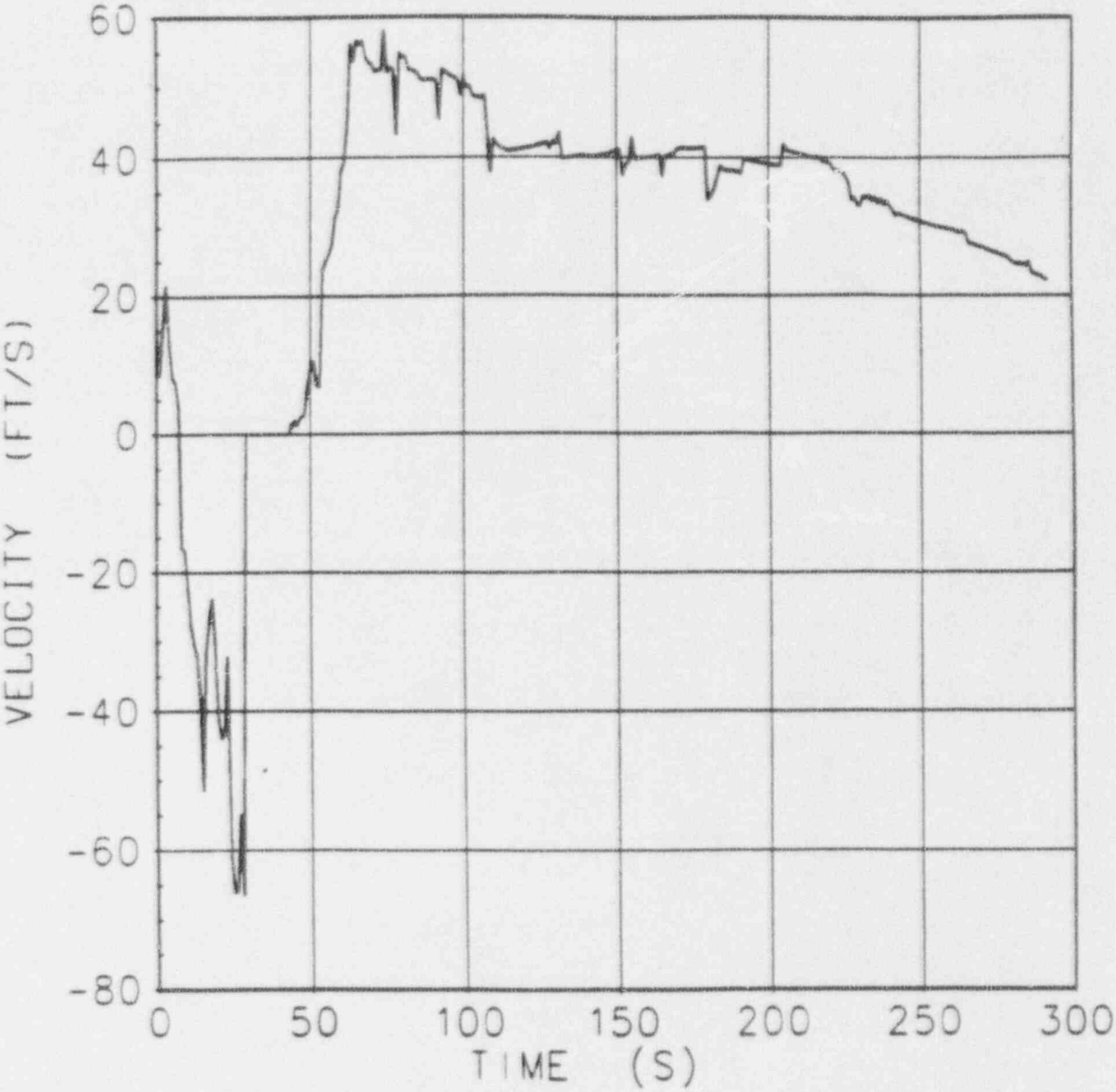


WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

FLUID QUALITY

(C_D = 0.8)

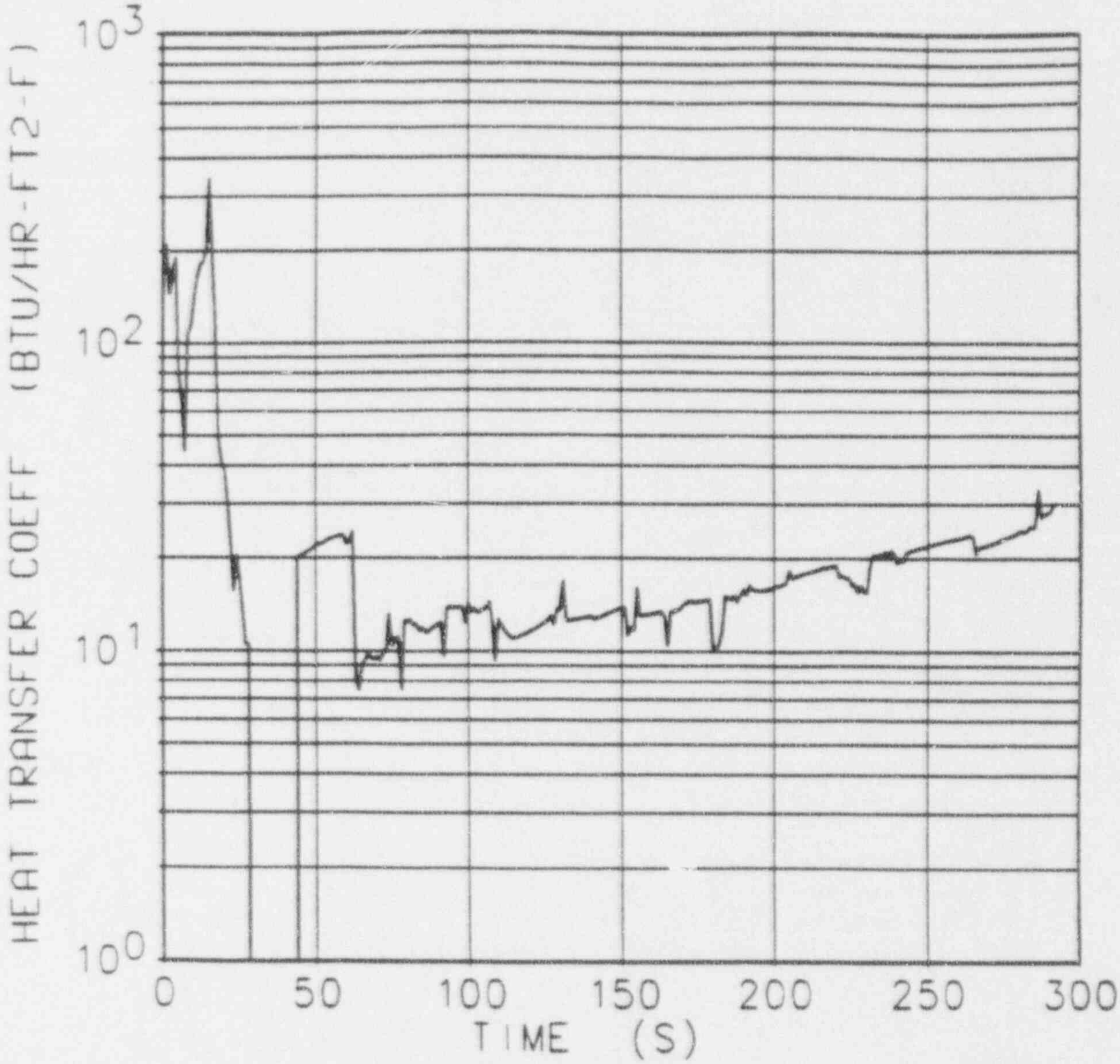
FIGURE 2-4H



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

FLUID VELOCITY PAST CLAD HOT SPOT
(C_D = 0.8)

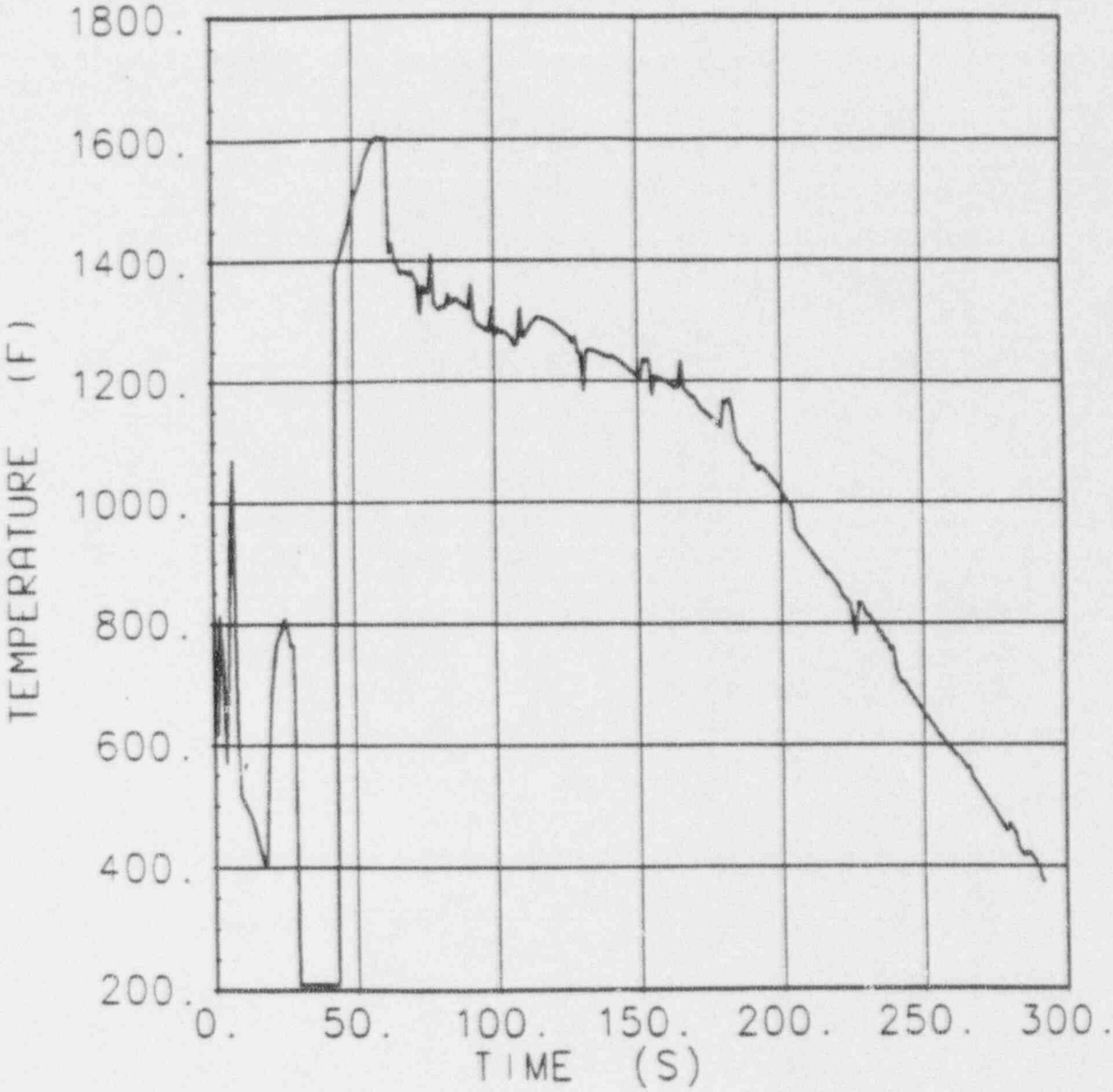
FIGURE 2-4I



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

HOT ROD HEAT TRANSFER COEFFICIENT
(C_D = 0.8)

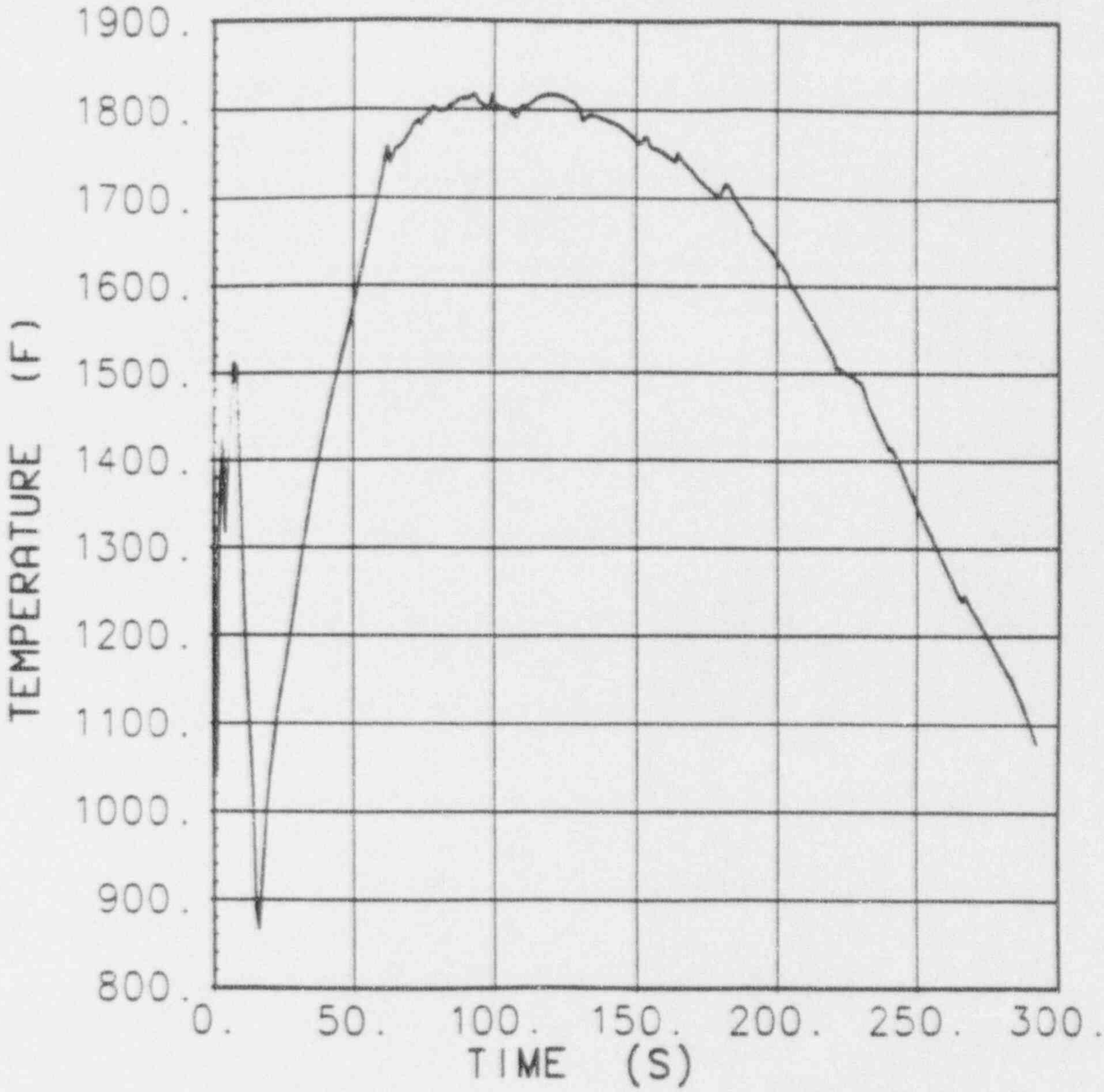
FIGURE 2-4J



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

CLAD HOT SPOT FLUID TEMPERATURE
(C_D = 0.8)

FIGURE 2-4K

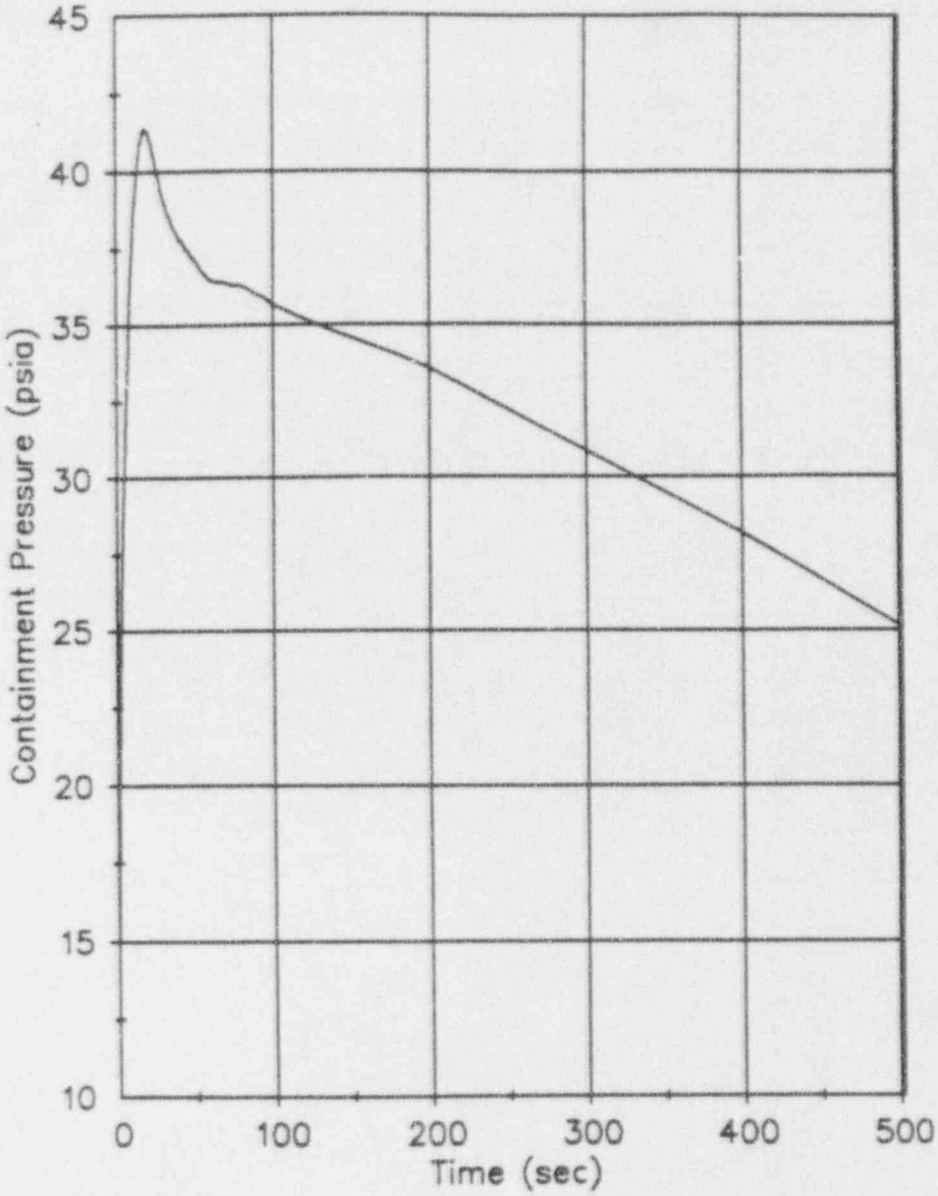


WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

HOT ROD PEAK CLAD TEMPERATURE

(C_D = 0.8)

FIGURE 2-4L

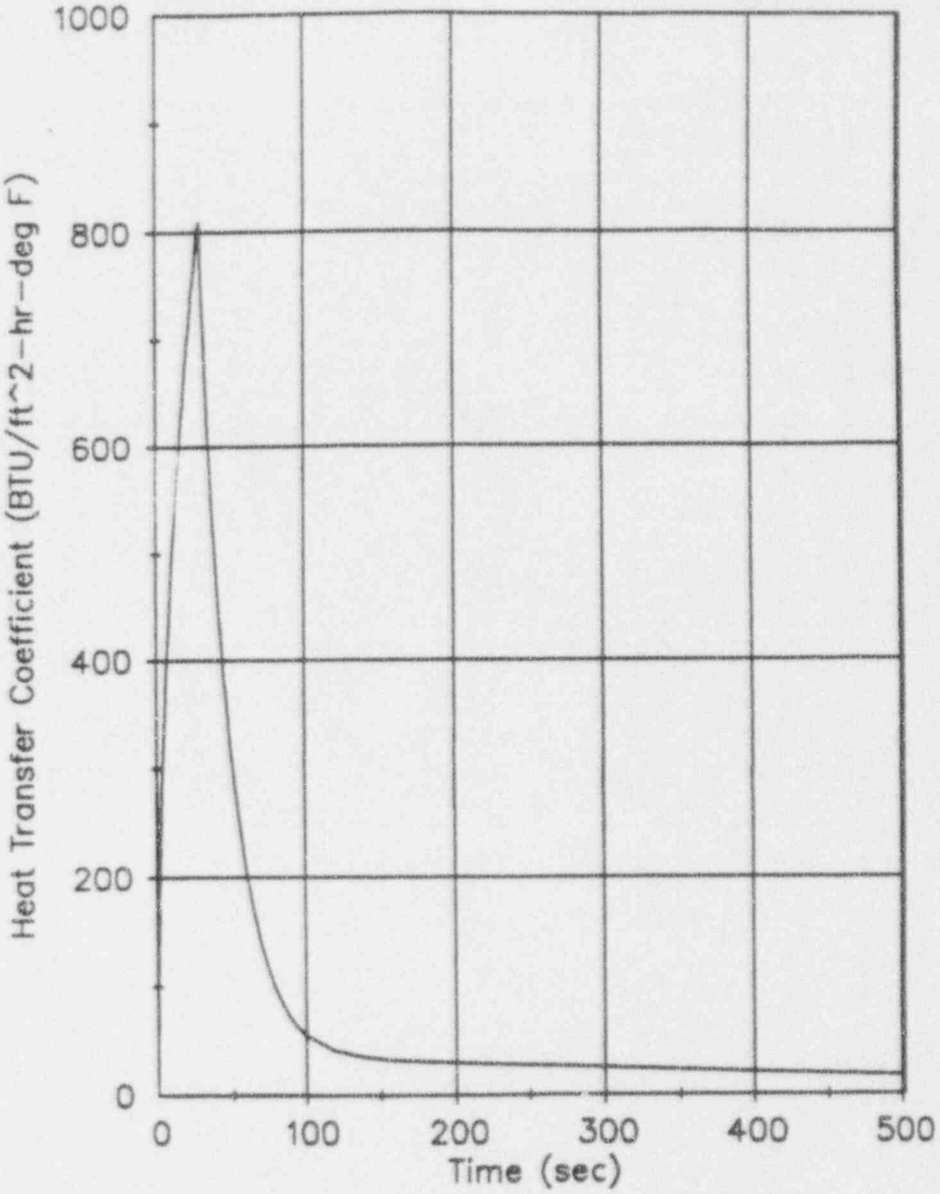


WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

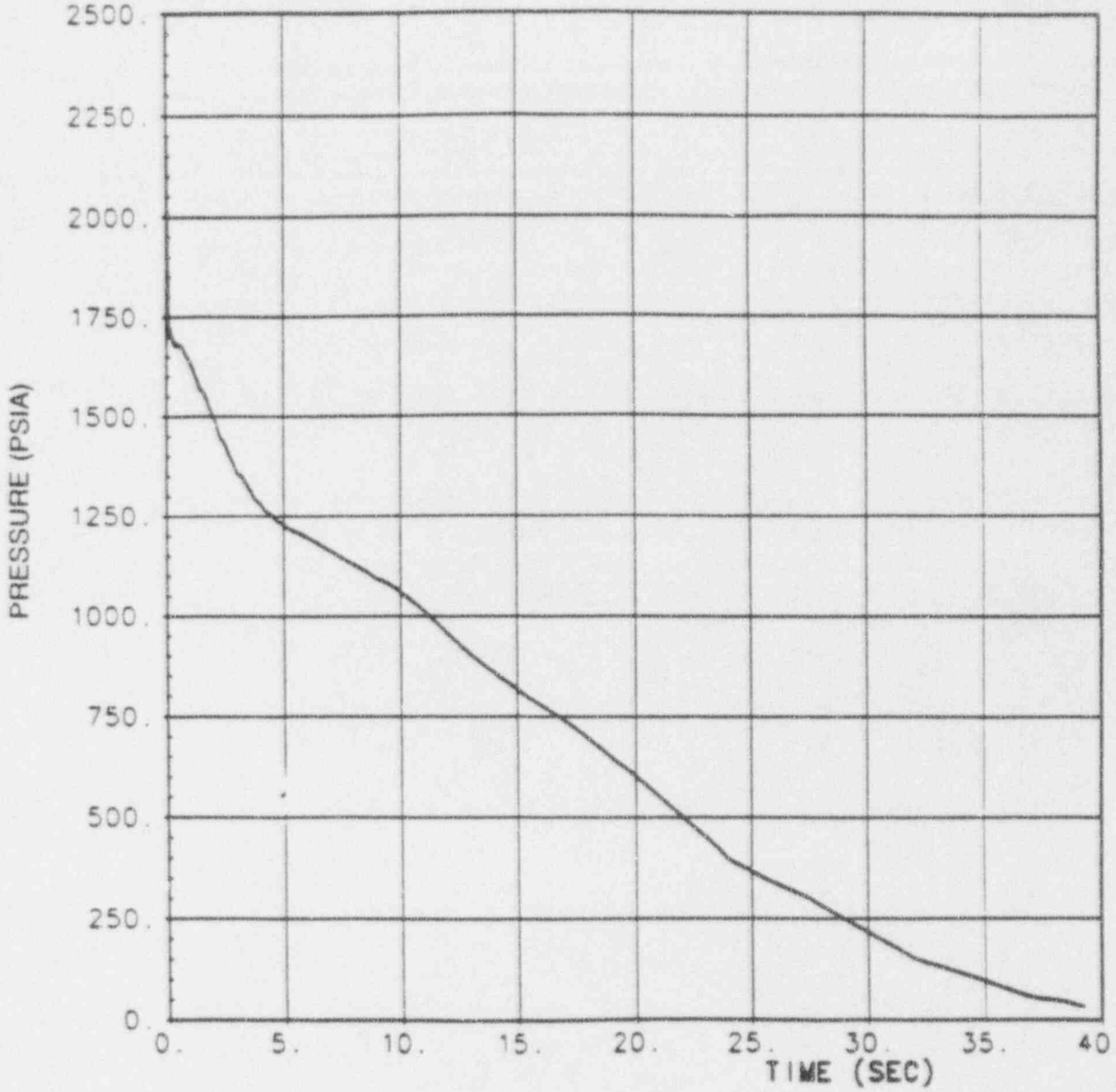
CONTAINMENT PRESSURE

(C_D = 0.8)

FIGURE 2-4M



WESTINGHOUSE ELECTRIC CORPORATION Nuclear and Advanced Technology Divisions	CONTAINMENT CONDENSING WALL HEAT TRANSFER COEFFICIENT (C_D = 0.8)
	FIGURE 2-4N

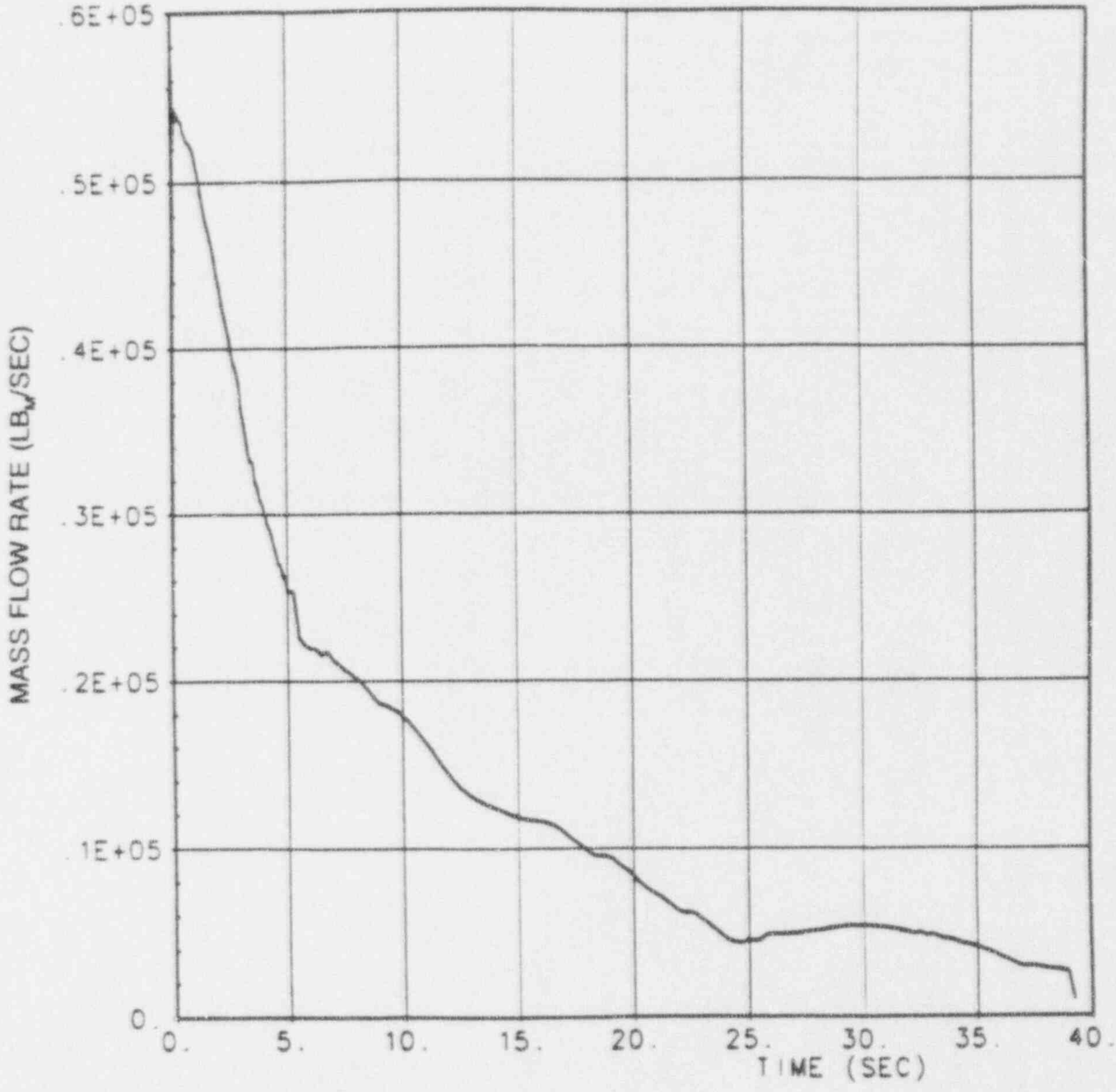


WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

RCS PRESSURE

(C_D = 0.4)

FIGURE 2-5A

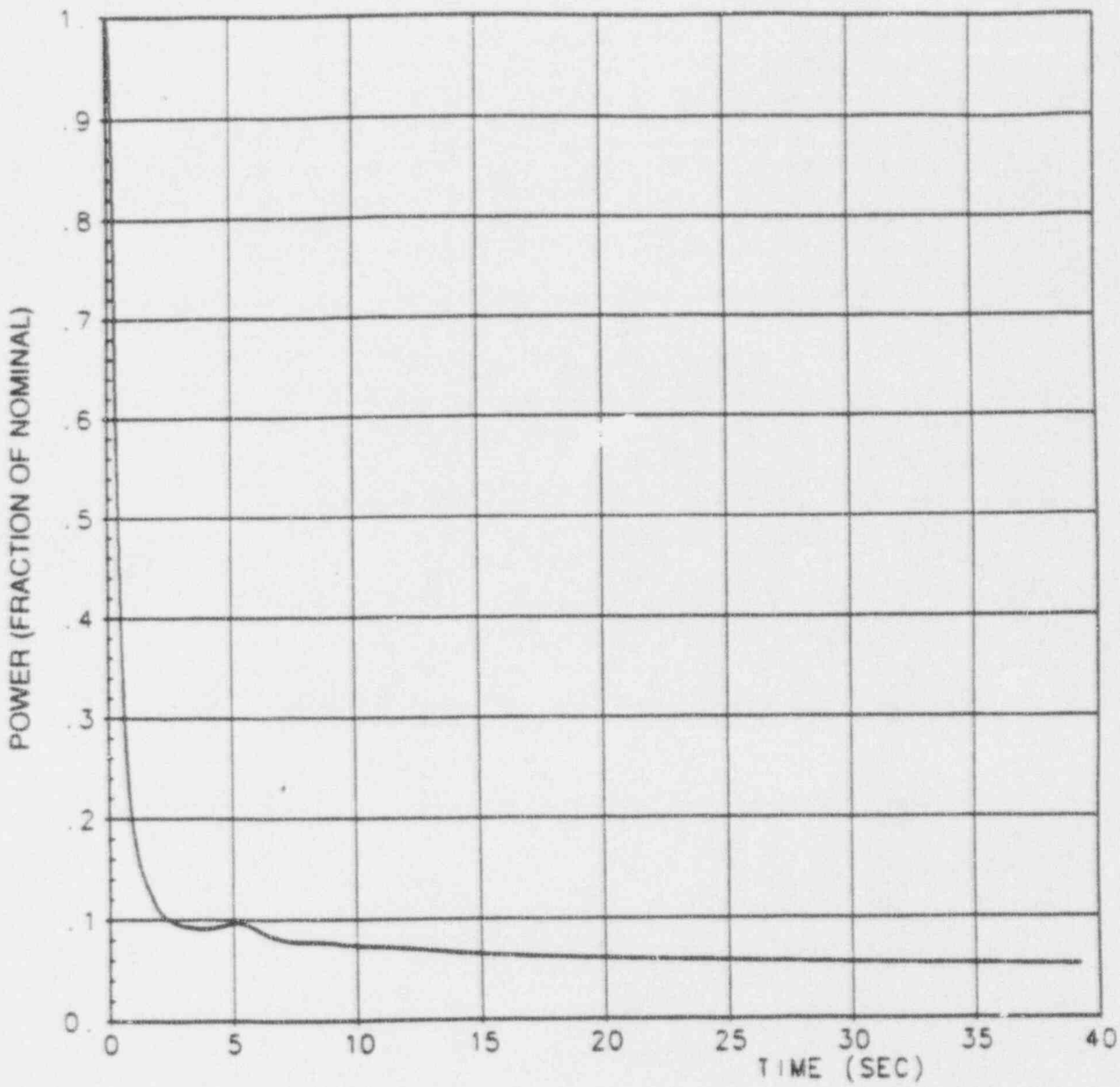


WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

COLD LEG BREAK MASS FLOW RATE

(C_D = 0.4)

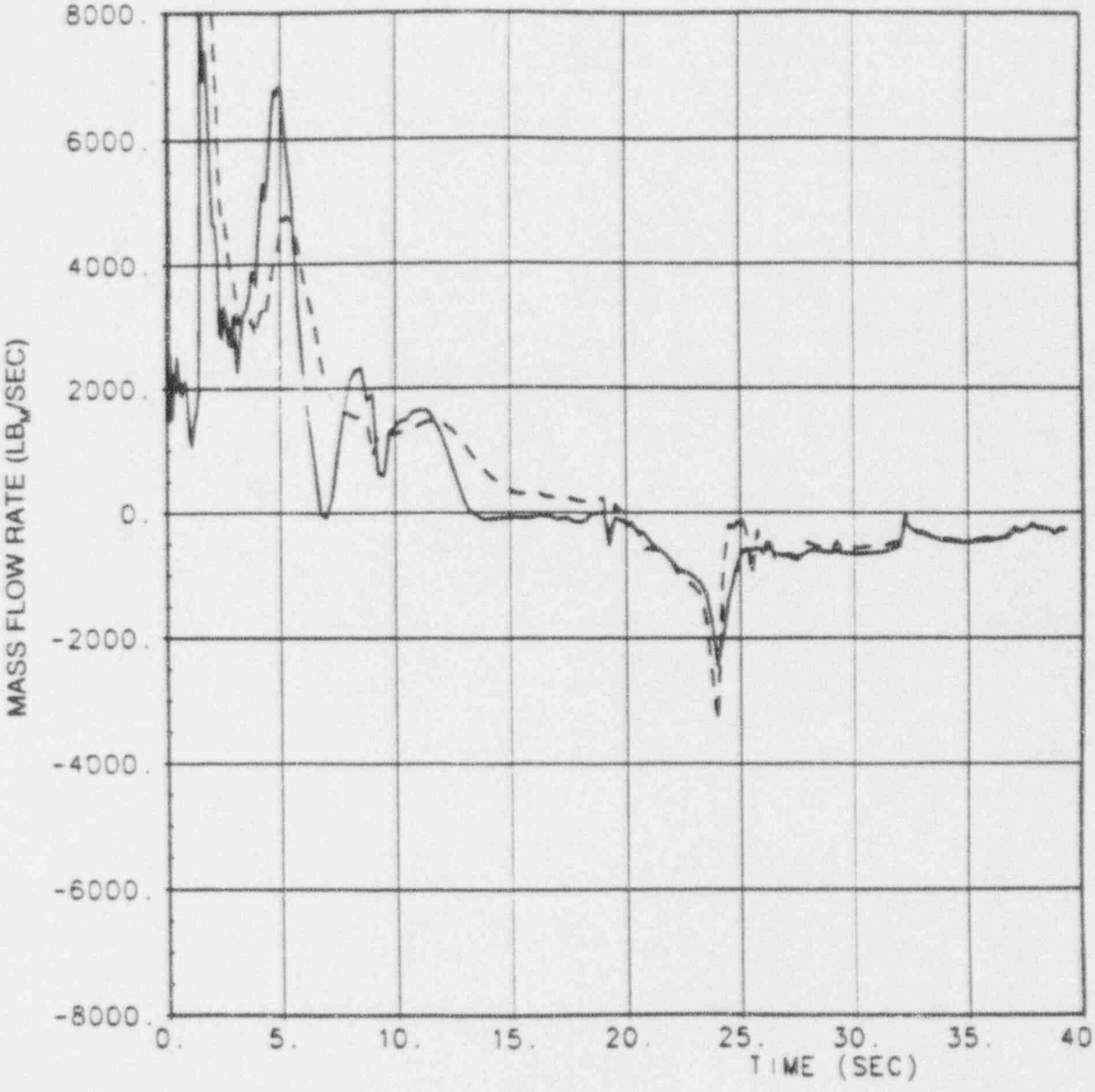
FIGURE 2-5B



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

CORE POWER
(FRACTION OF NOMINAL)
(C_D = 0.4)

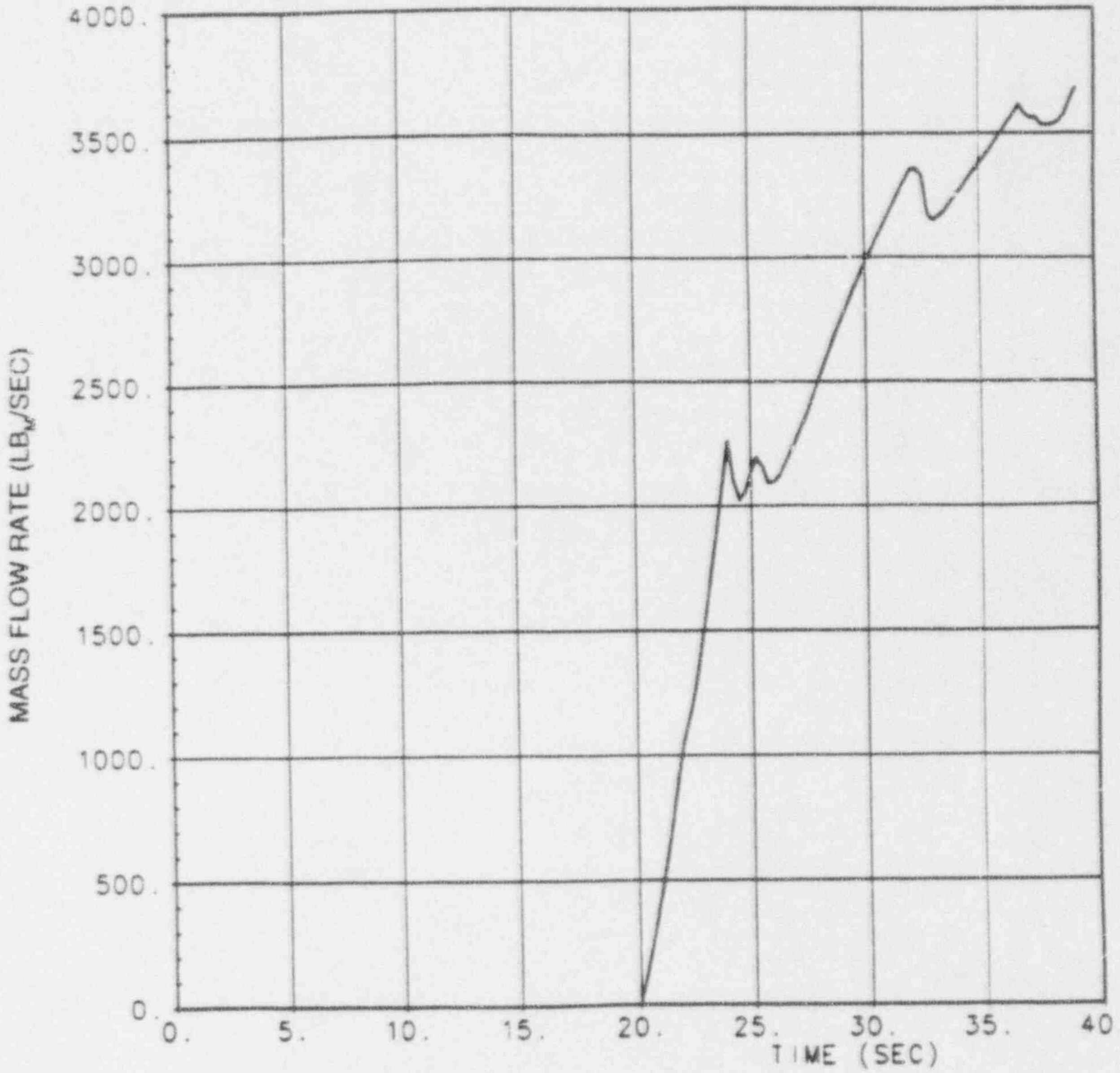
FIGURE 2-5C



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

CORE MASS FLOW RATE
(TOP AND BOTTOM)
(C_D = 0.4)

FIGURE 2-5D

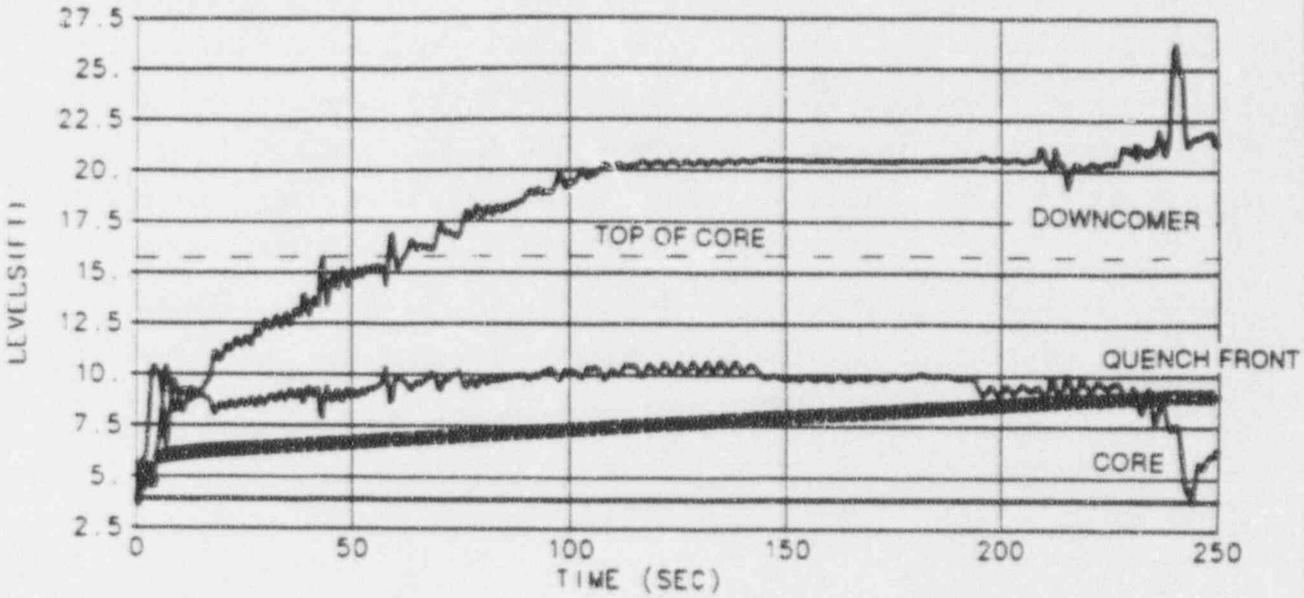


WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

ACCUMULATOR MASS FLOW RATE

(C_D = 0.4)

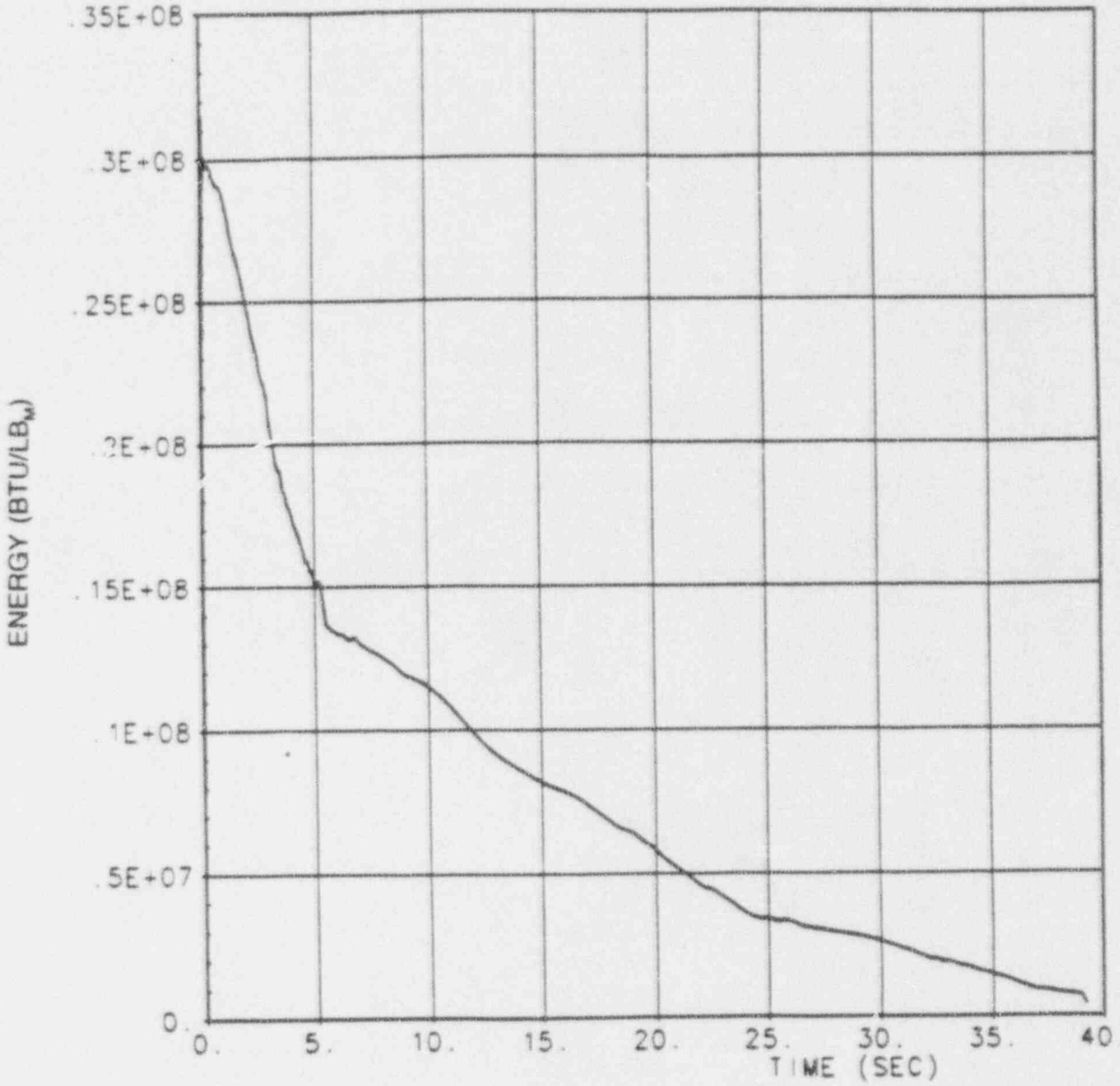
FIGURE 2-5E



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

REFLOOD CORE AND DOWNCOMER WATER LEVELS (C_D = 0.4)

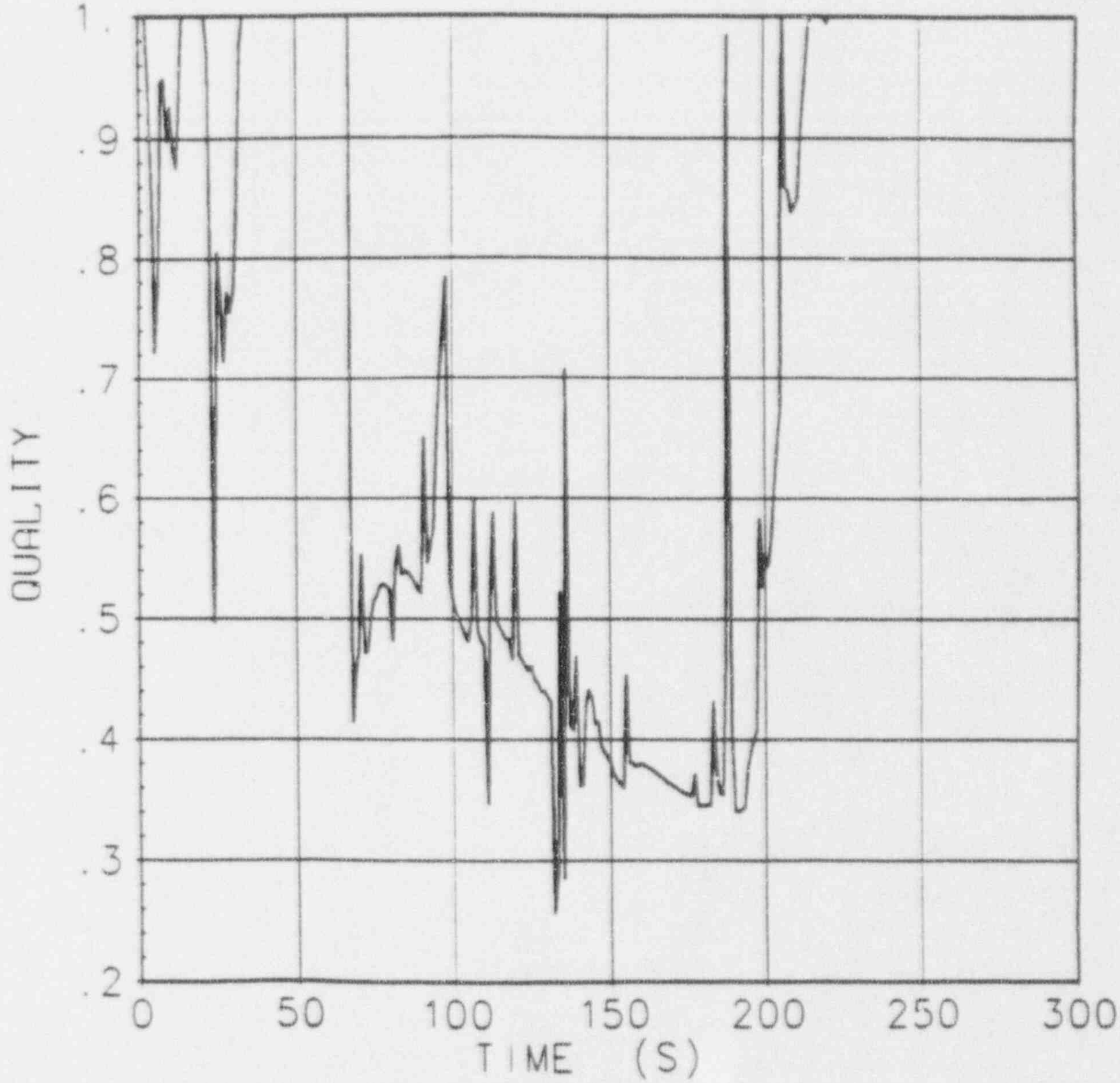
FIGURE 2-5F



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

BREAK ENERGY RELEASED TO CONTAINMENT (C_D = 0.4)

FIGURE 2-5G

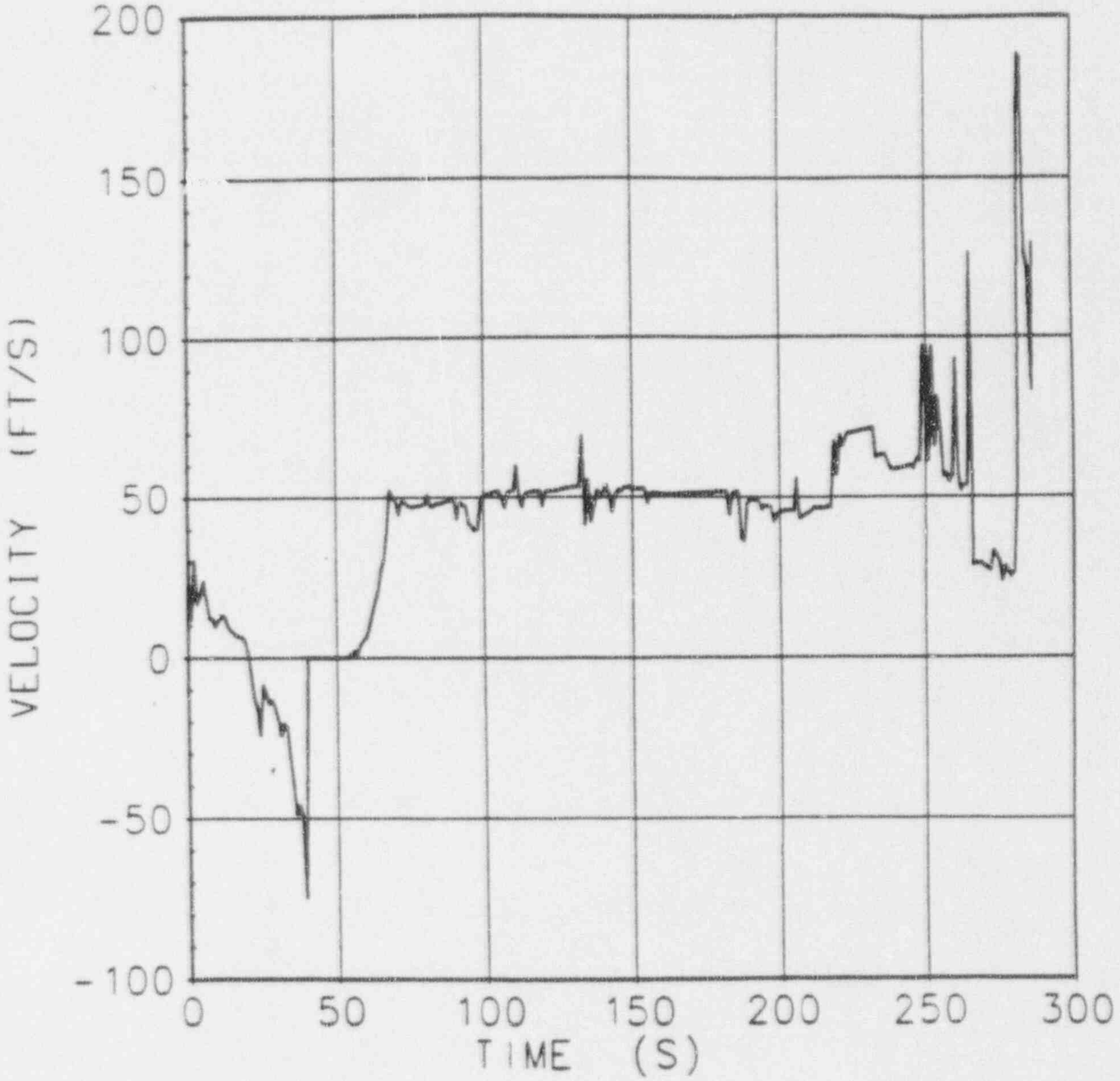


WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

FLUID QUALITY

(C_D = 0.4)

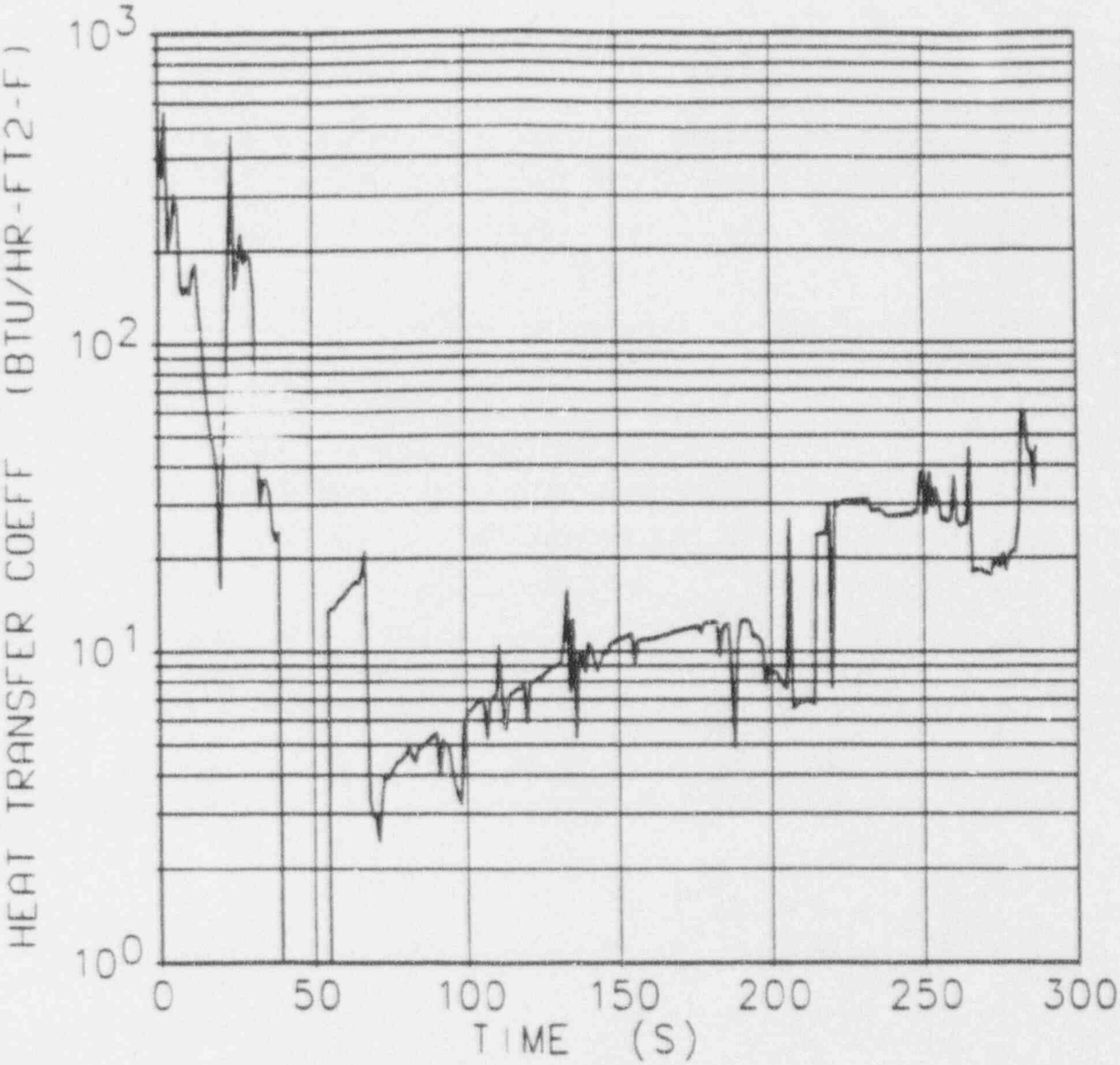
FIGURE 2-5H



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

FLUID VELOCITY PAST CLAD HOT SPOT
(C_D = 0.4)

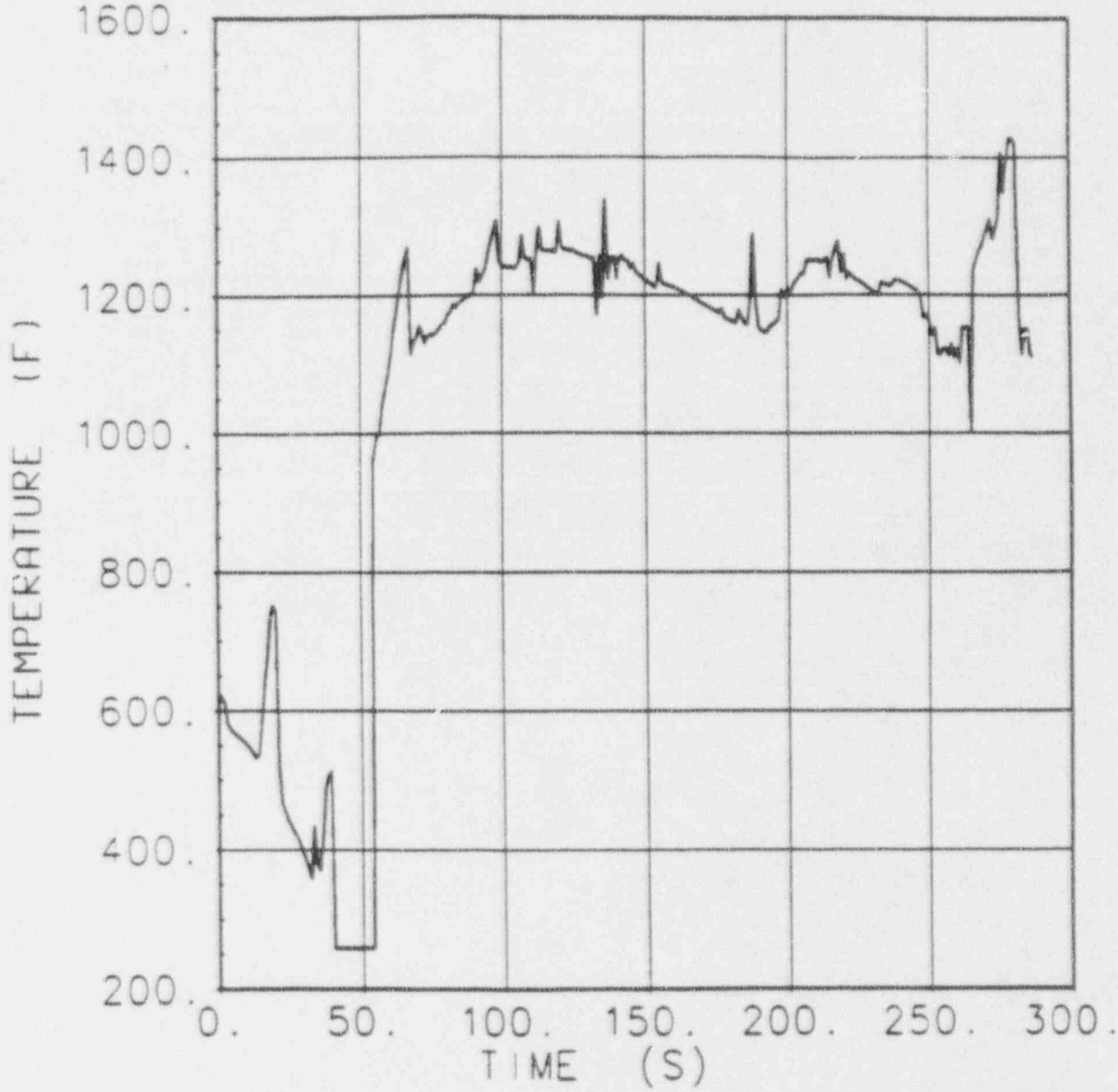
FIGURE 2-5I



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

HOT ROD HEAT TRANSFER COEFFICIENT
(C_D = 0.4)

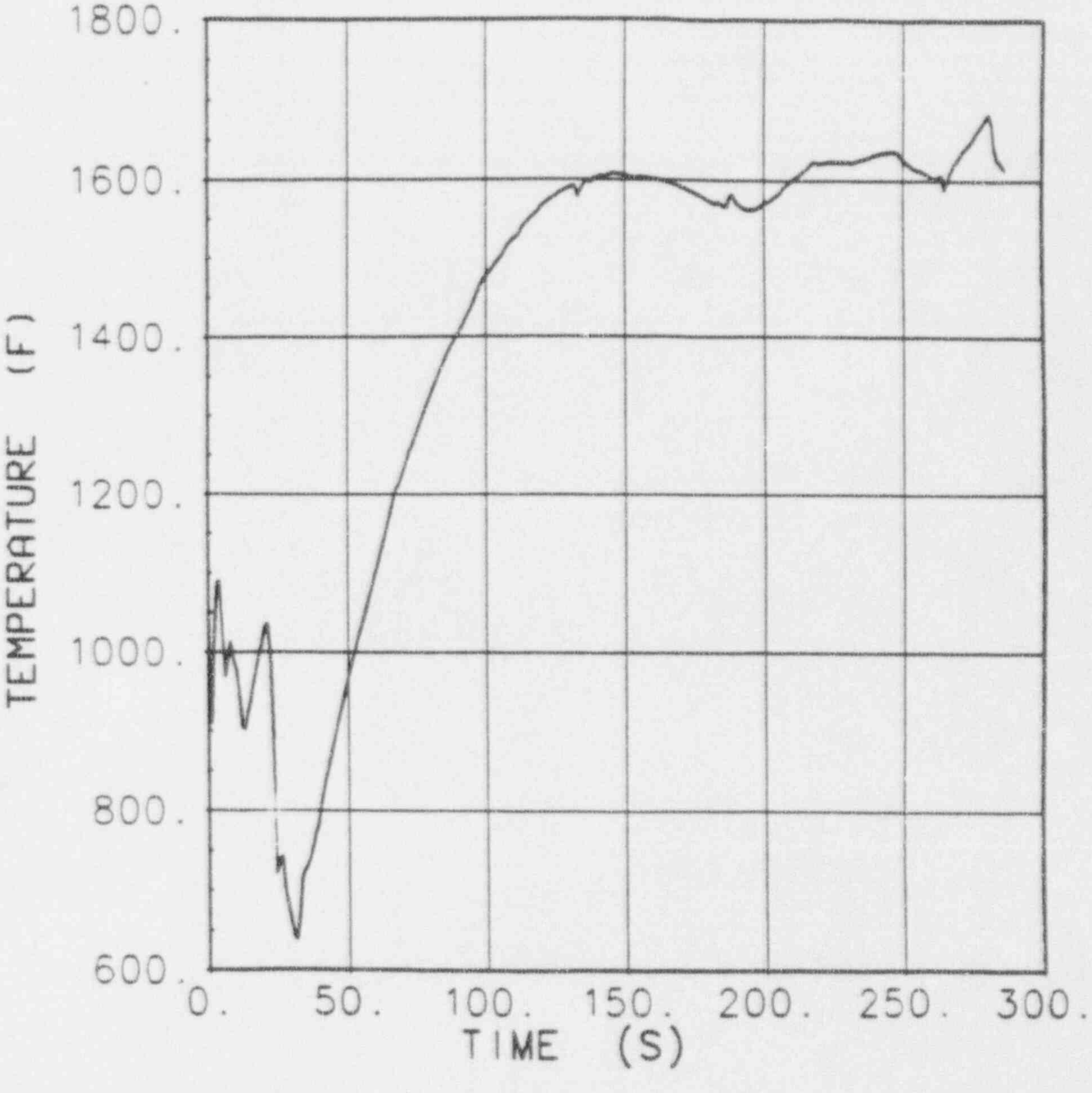
FIGURE 2-5J



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

CLAD HOT SPOT FLUID TEMPERATURE
(C_D = 0.4)

FIGURE 2-5K

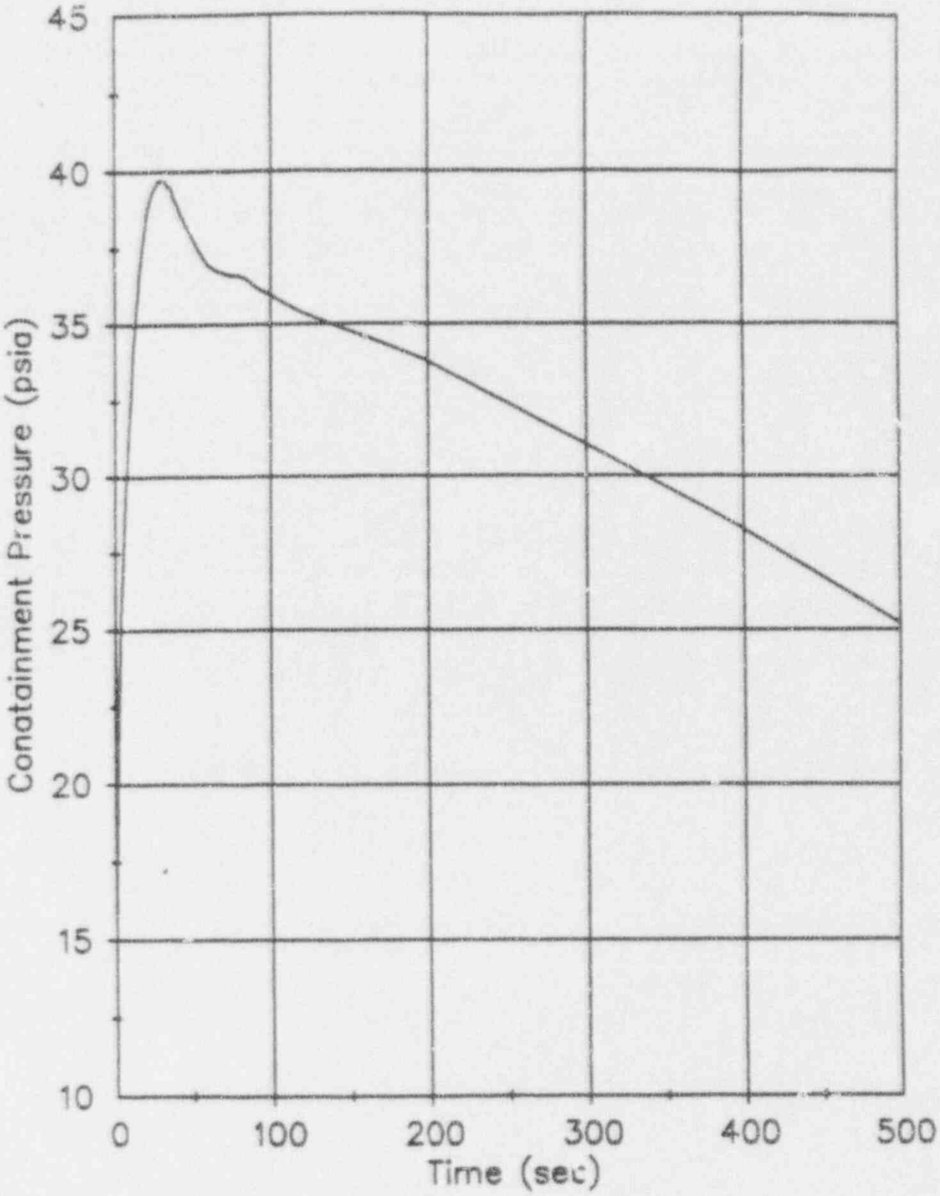


WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

HOT ROD PEAK CLAD TEMPERATURE

(C_D = 0.4)

FIGURE 2-5L

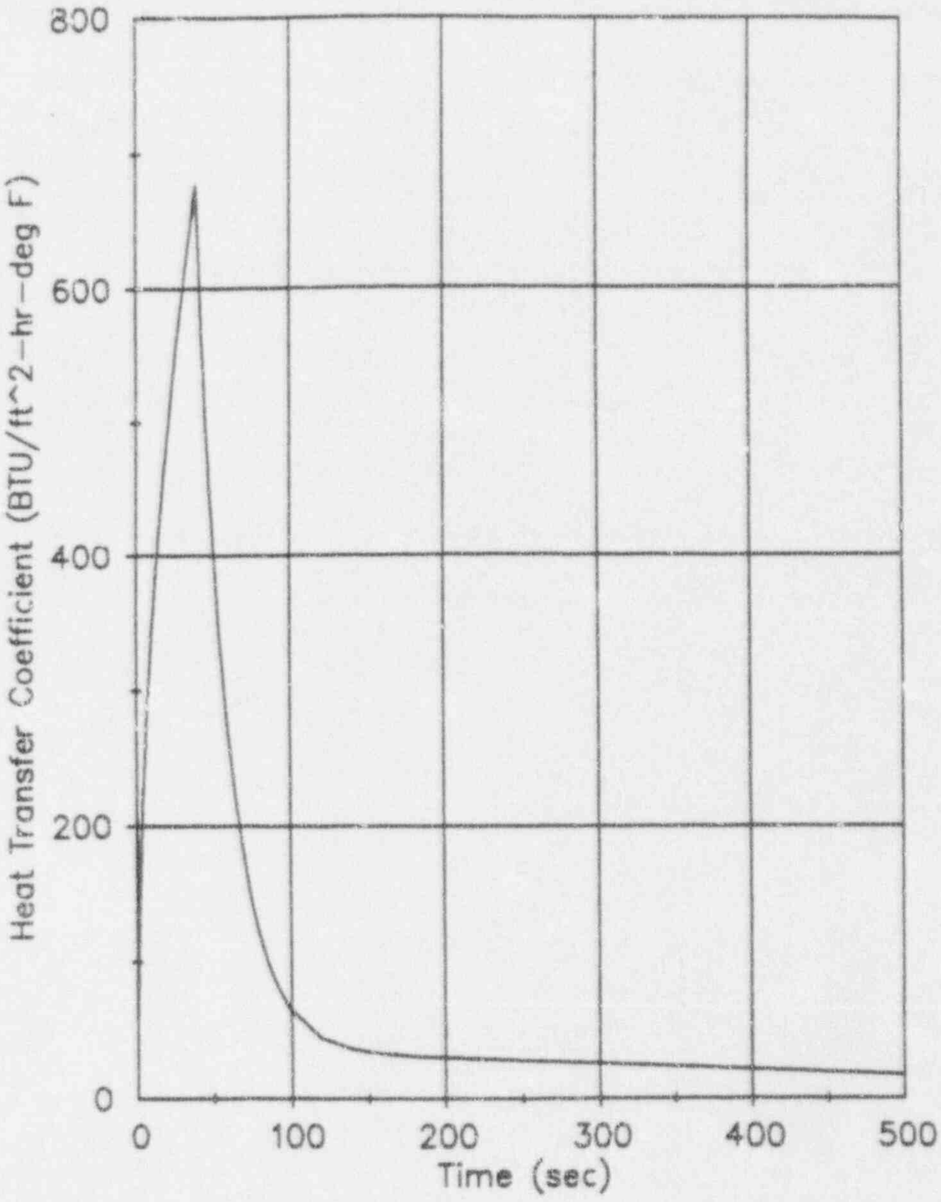


WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

CONTAINMENT PRESSURE

(C_D = 0.4)

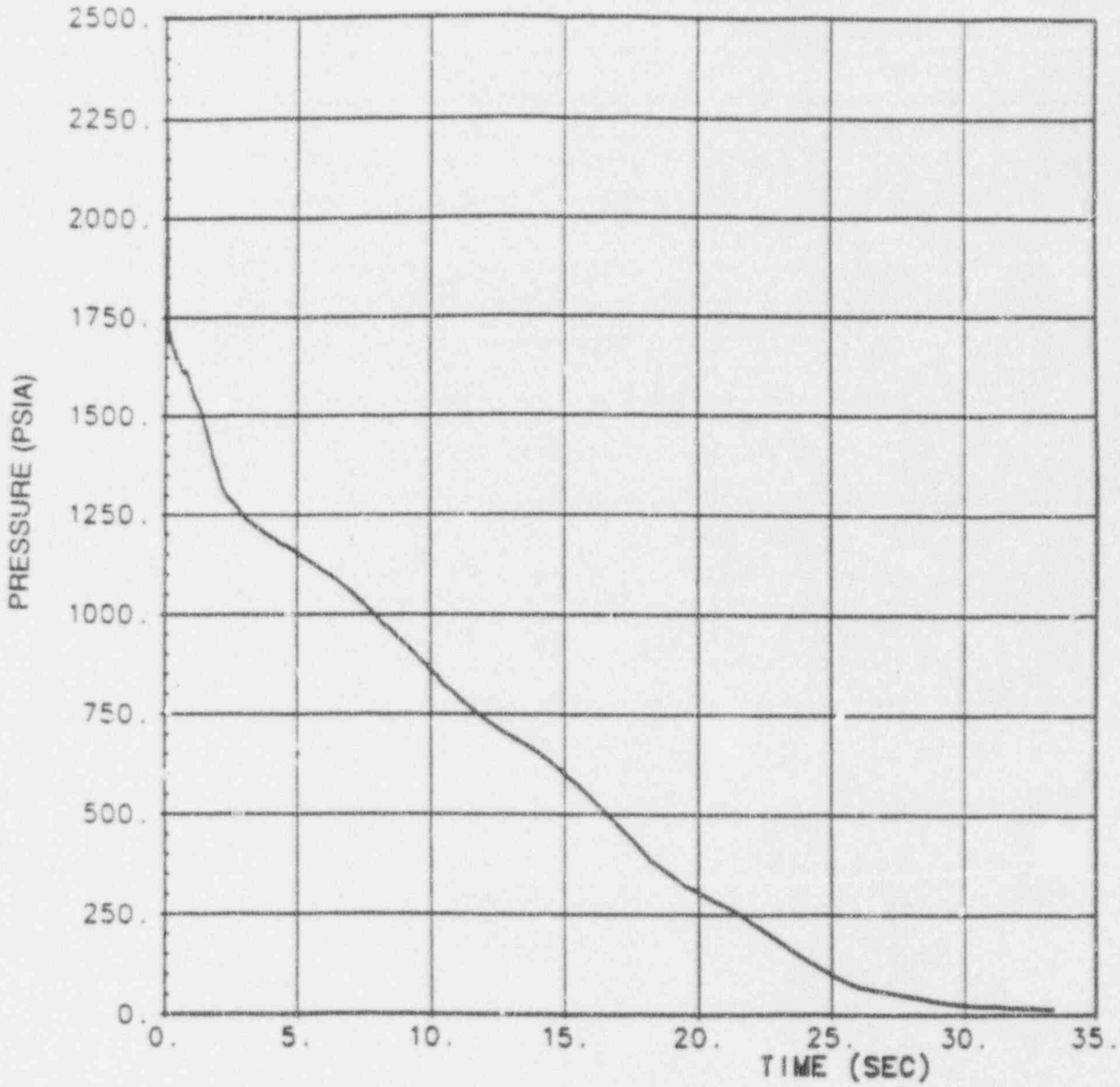
FIGURE 2-5M



WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

CONTAINMENT CONDENSING WALL HEAT TRANSFER COEFFICIENT (C_D = 0.4)

FIGURE 2-5N

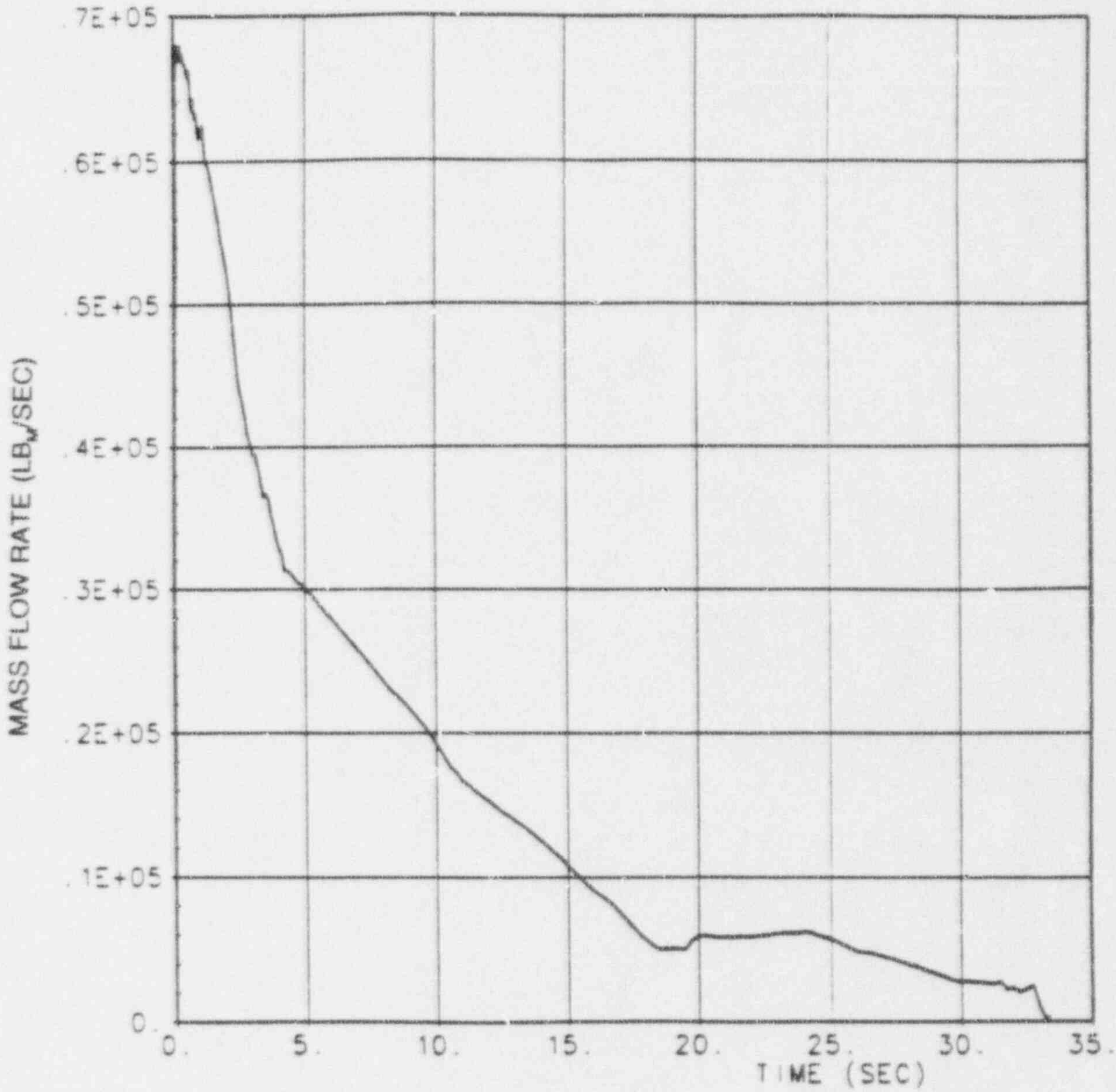


WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Divisions

RCS PRESSURE

(C_D = 0.6, MAX SI)

FIGURE 2-6A



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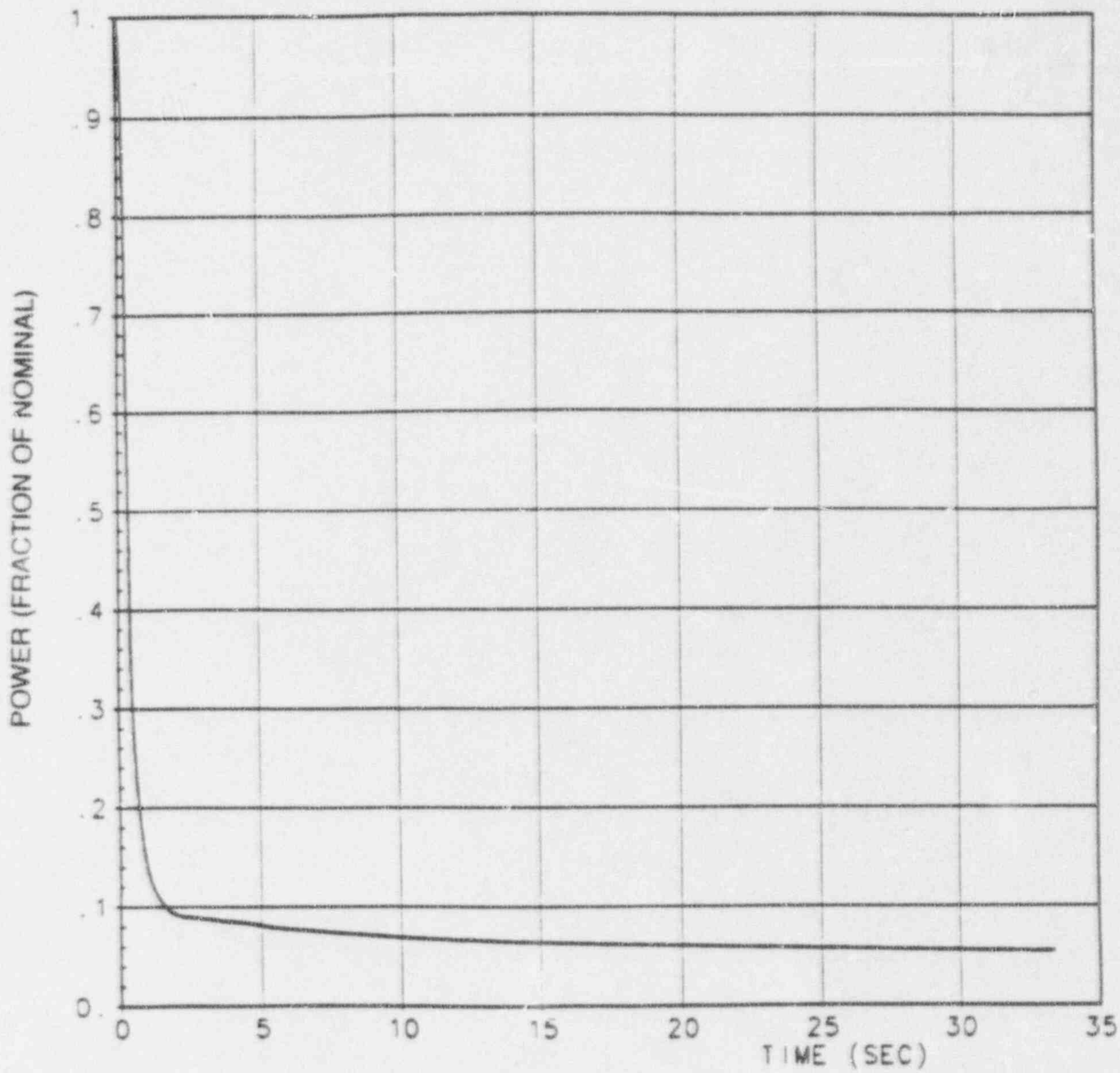
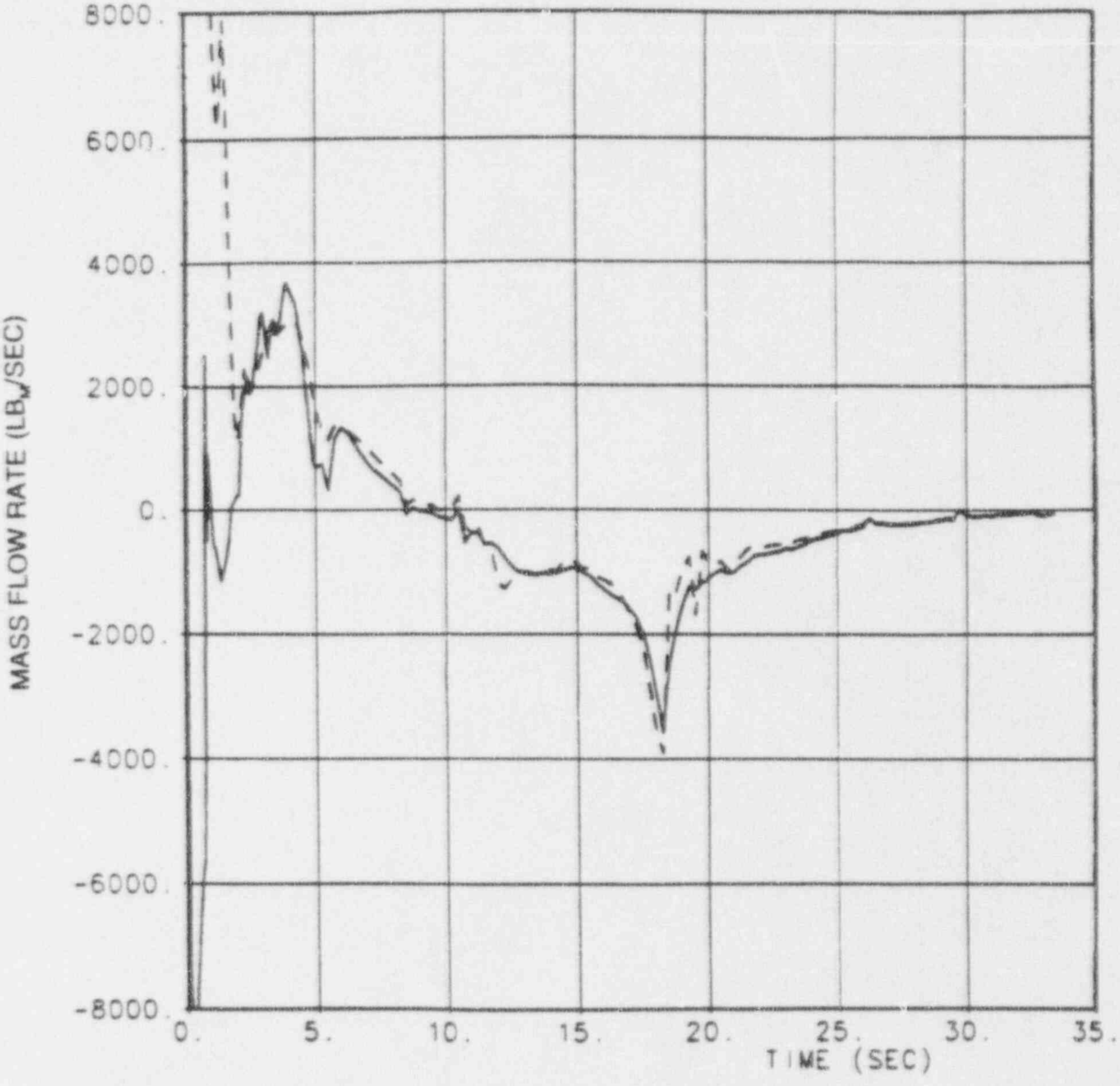
COLD LEG BREAK MASS FLOW RATE

(C_D = 0.6, MAX SI)

FIGURE 2-6B



WESTINGHOUSE ELECTRIC CORPORATION Nuclear and Advanced Technology Divisions	CORE POWER (FRACTION OF NOMINAL) (C_D = 0.6, MAX SI)
	FIGURE 2-6C



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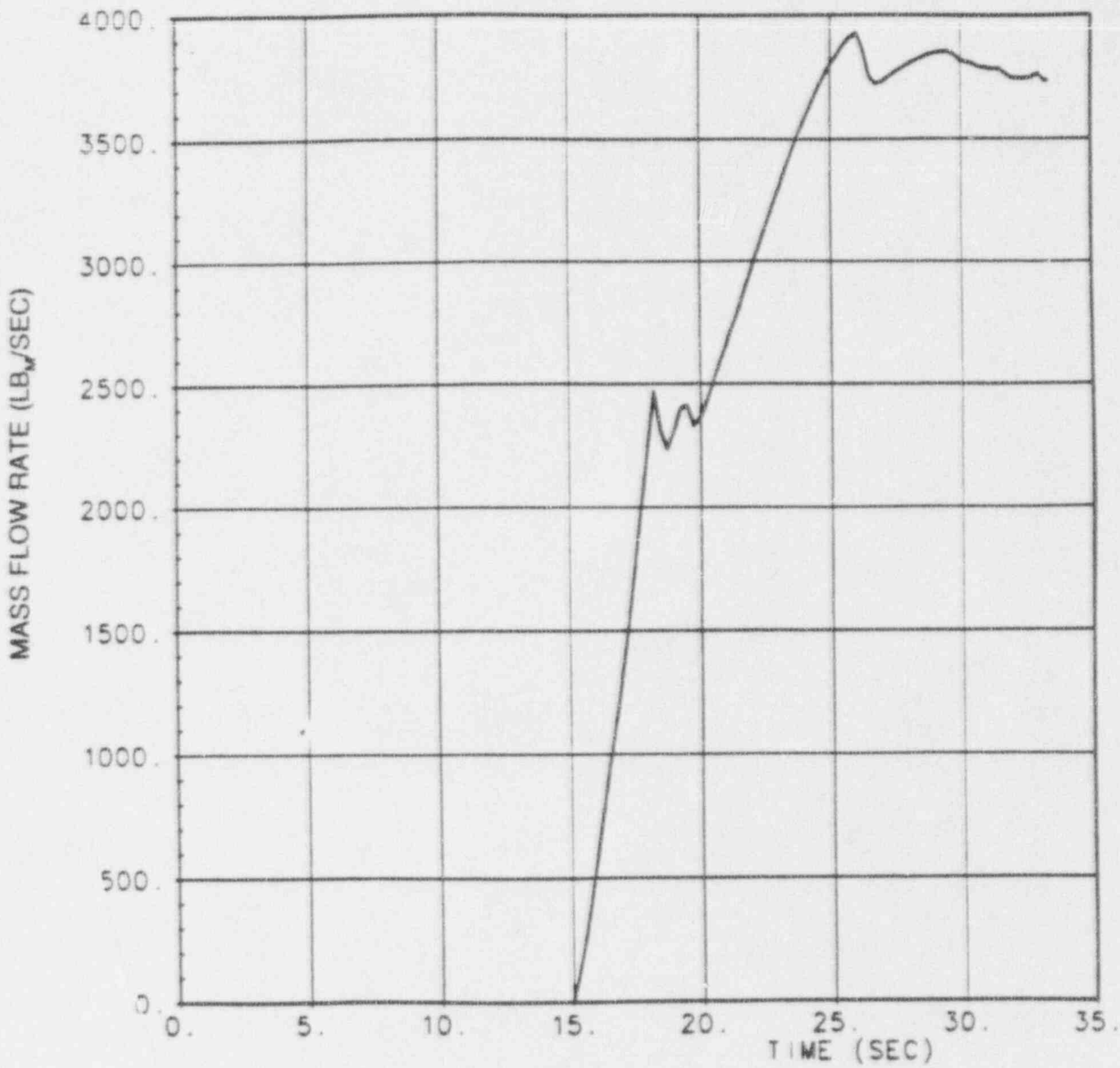
CORE MASS FLOW RATE
(TOP AND BOTTOM)
(C_D = 0.6, MAX SI)

FIGURE 2-6D



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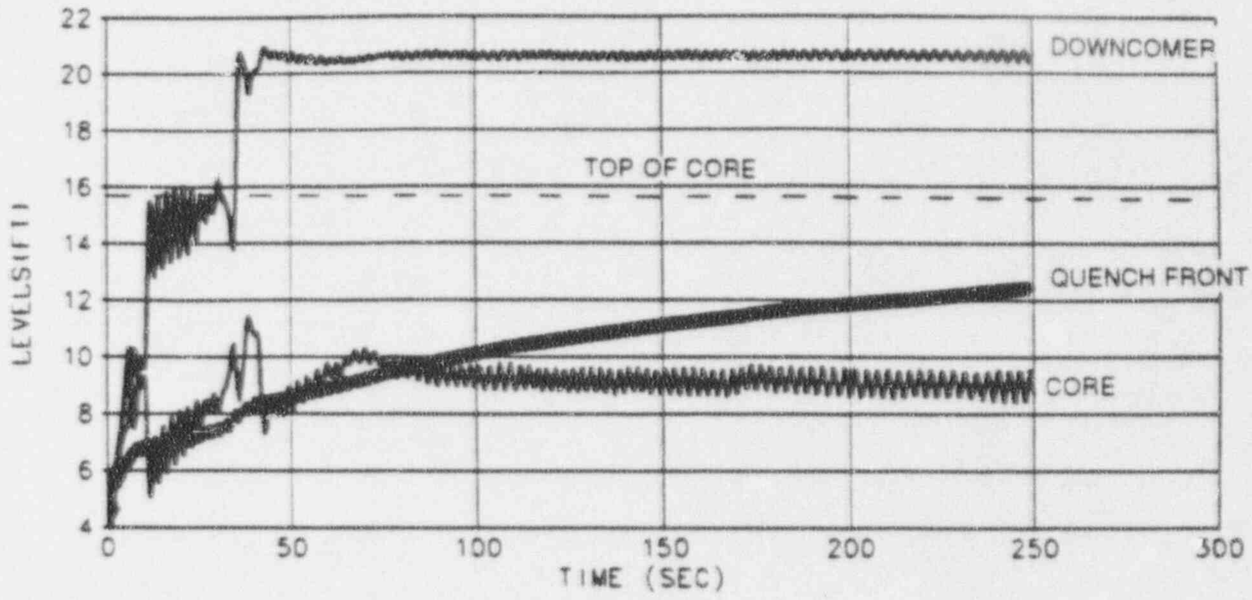
ACCUMULATOR MASS FLOW RATE

(C_D = 0.6, MAX SI)

FIGURE 2-6E



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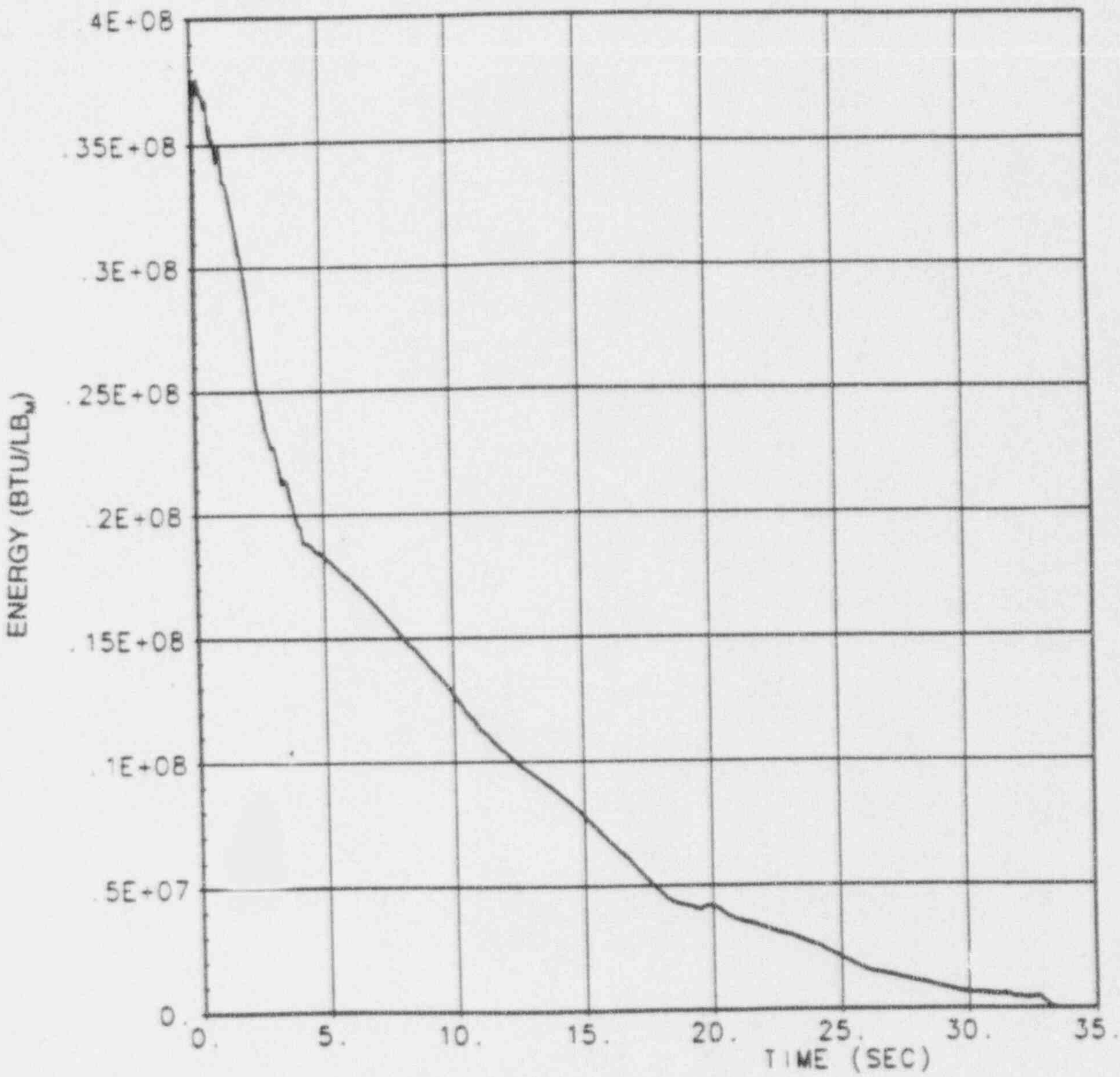
REFLOOD CORE AND DOWNCOMER WATER LEVELS (C_D = 0.6, MAX SI)

FIGURE 2-6F



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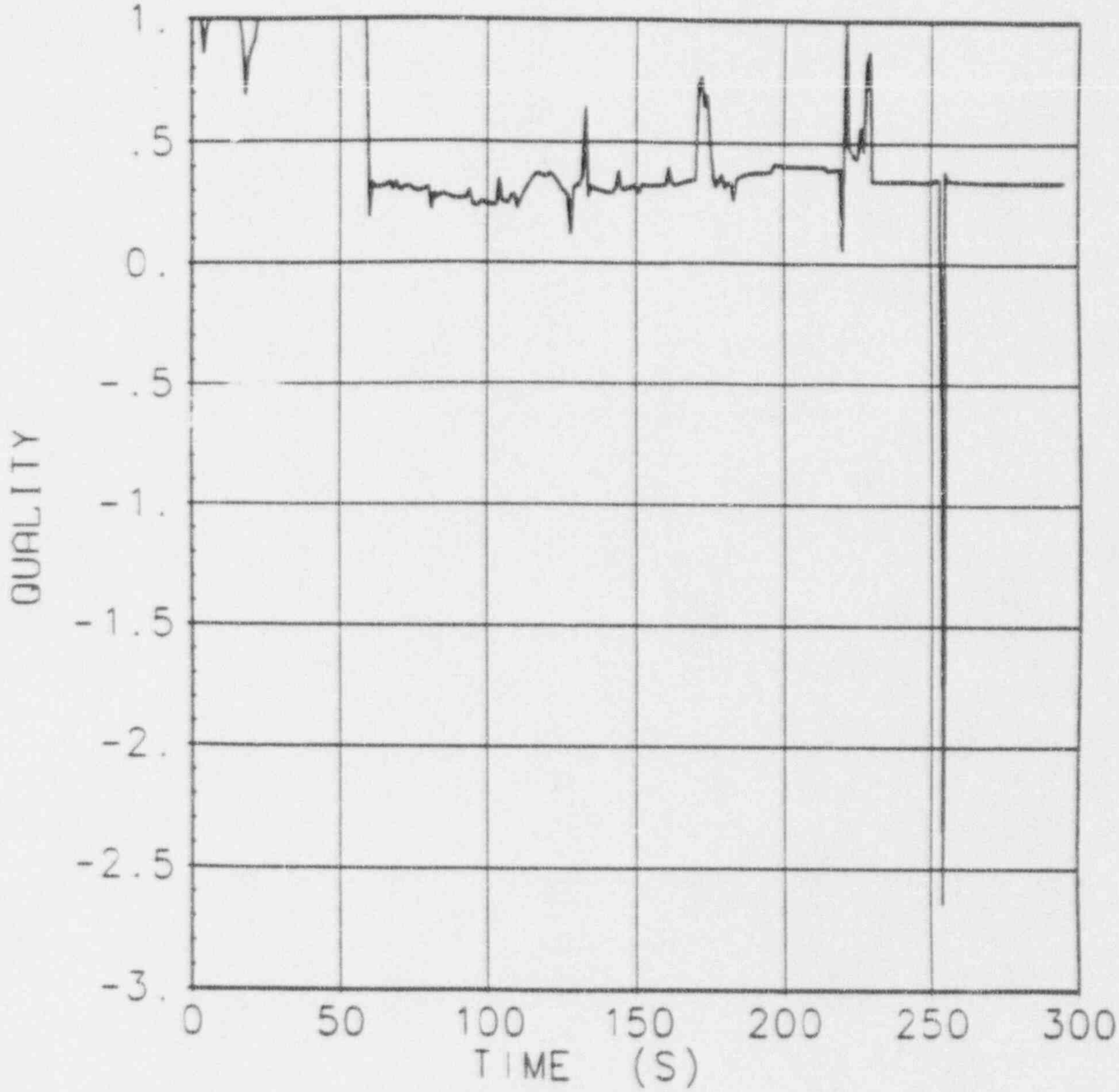
BREAK ENERGY RELEASED TO CONTAINMENT (C_D = 0.6, MAX SI)

FIGURE 2-6G



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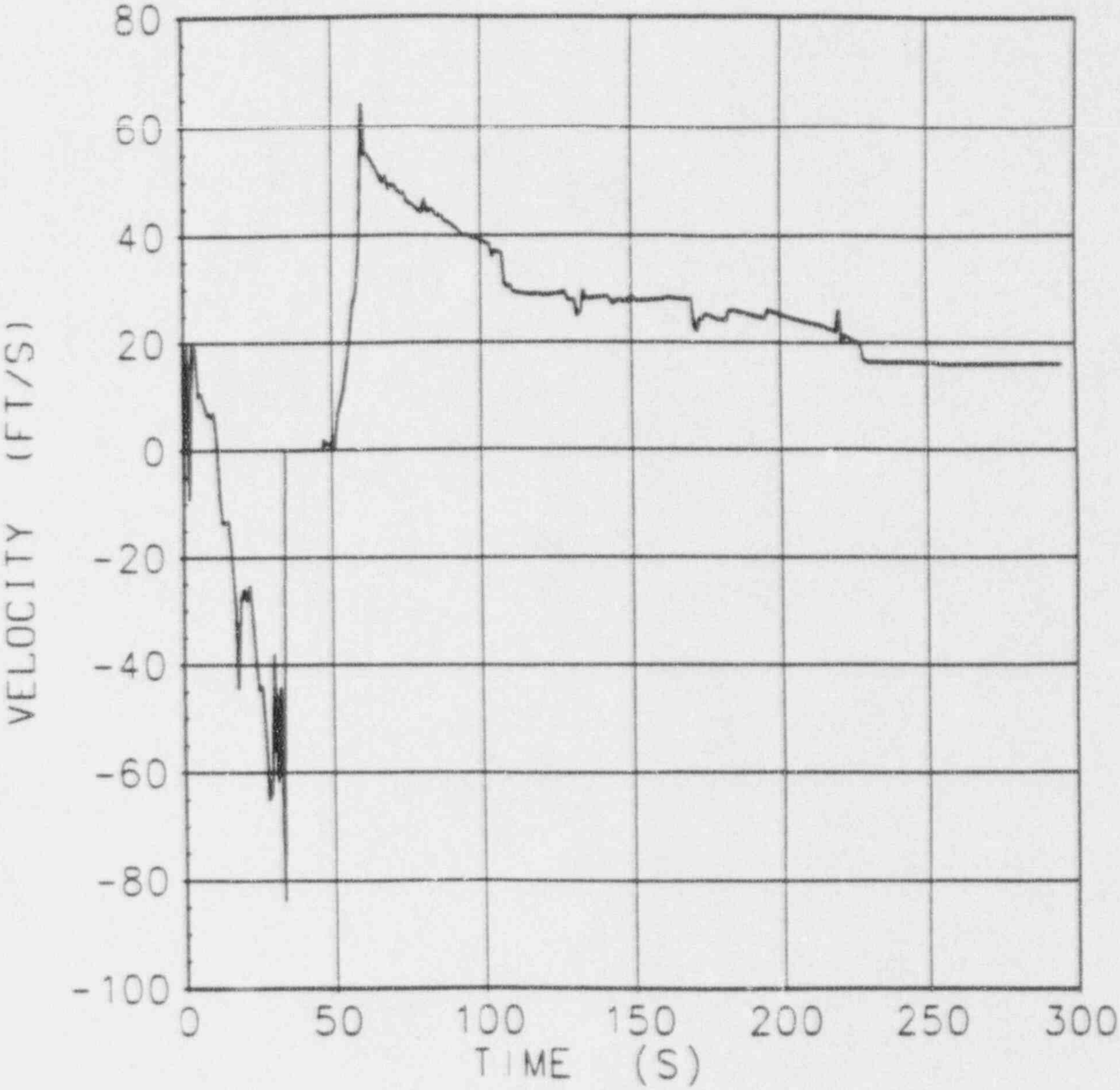
FLUID QUALITY

(C_D = 0.6, MAX SI)

FIGURE 2-6H



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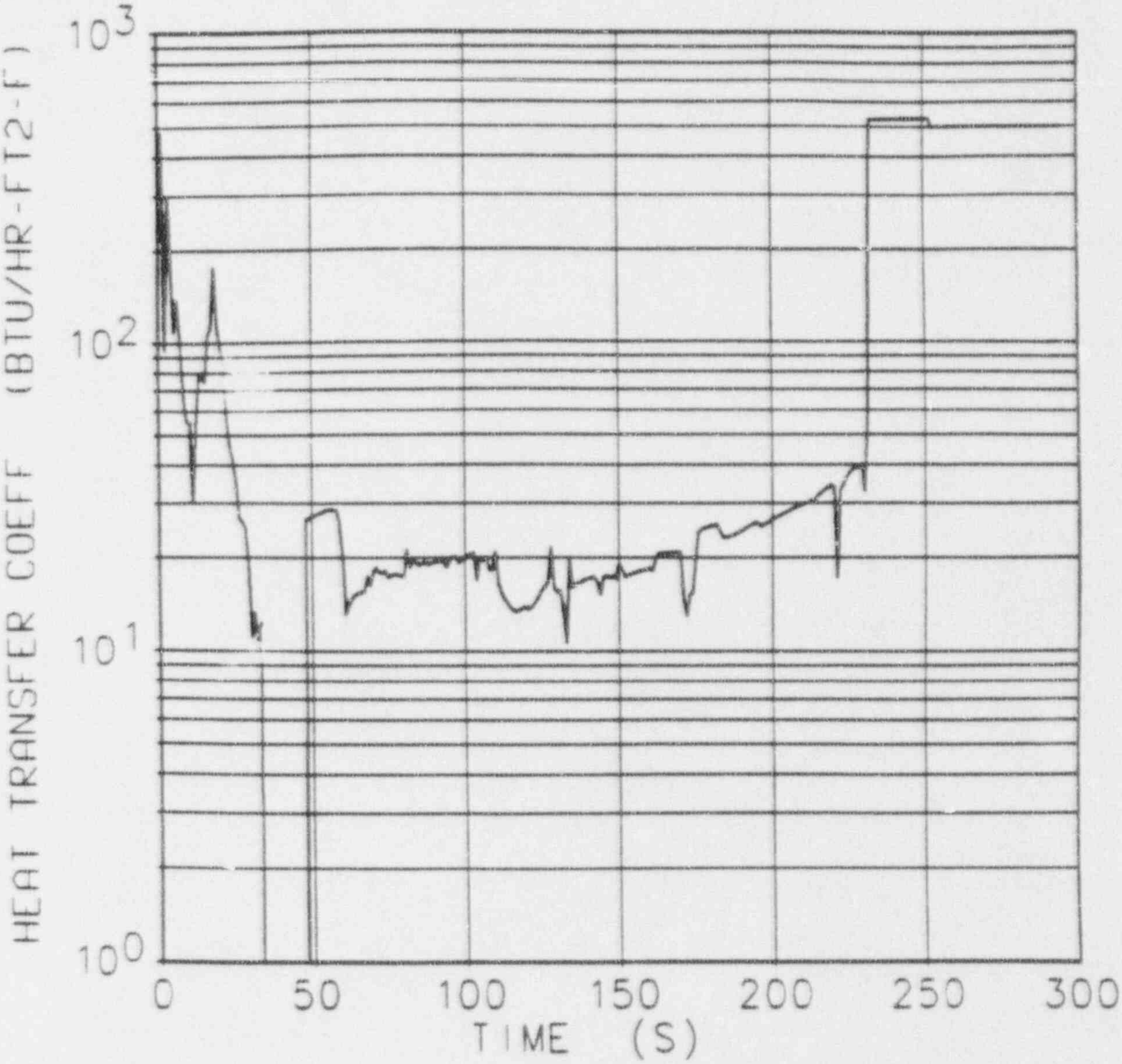
FLUID VELOCITY PAST CLAD HOT SPOT (C_D = 0.6, MAX SI)

FIGURE 2-6I



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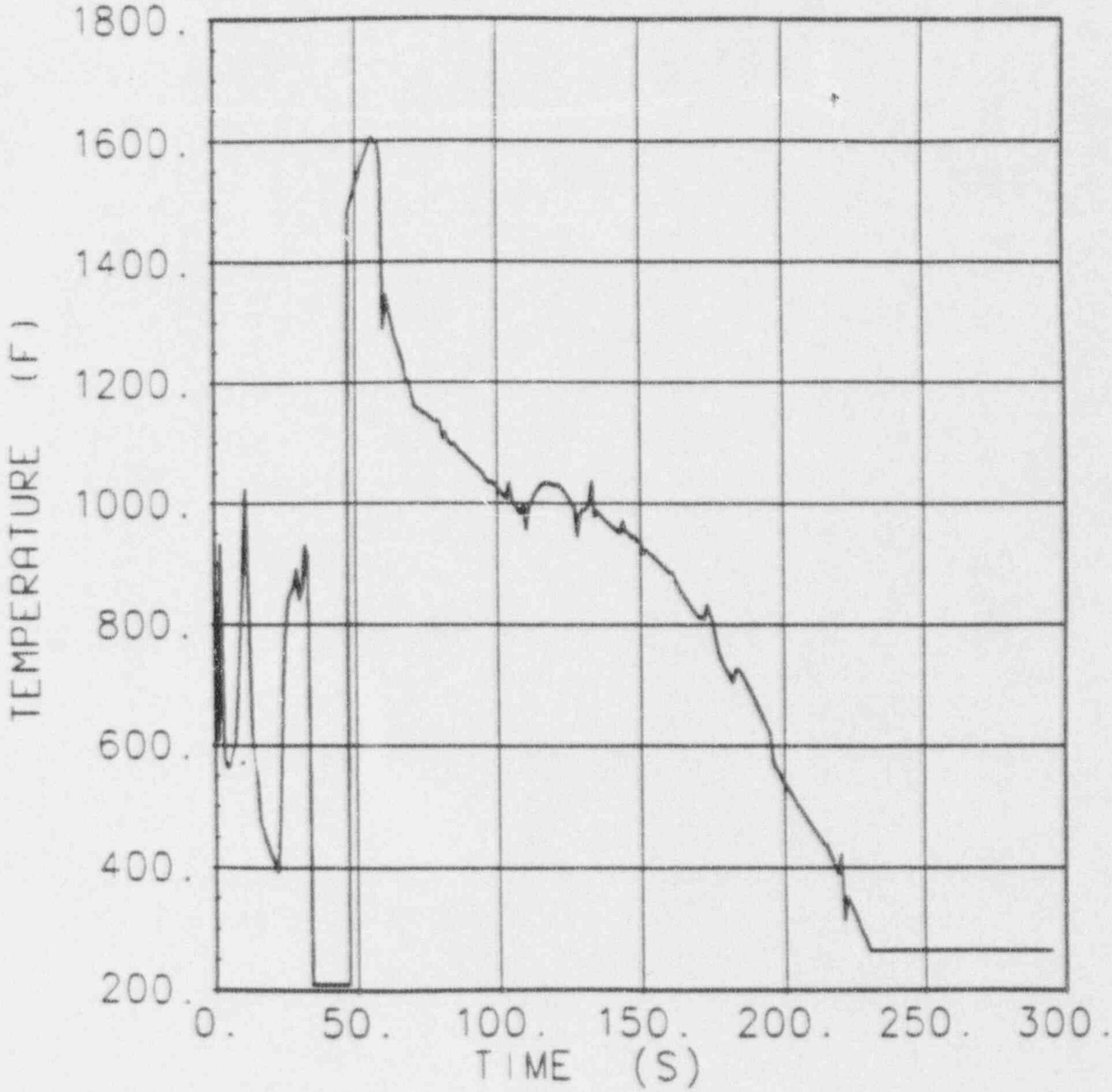
HOT ROD HEAT TRANSFER COEFFICIENT (C_D = 0.6, MAX SI)

FIGURE 2-6J



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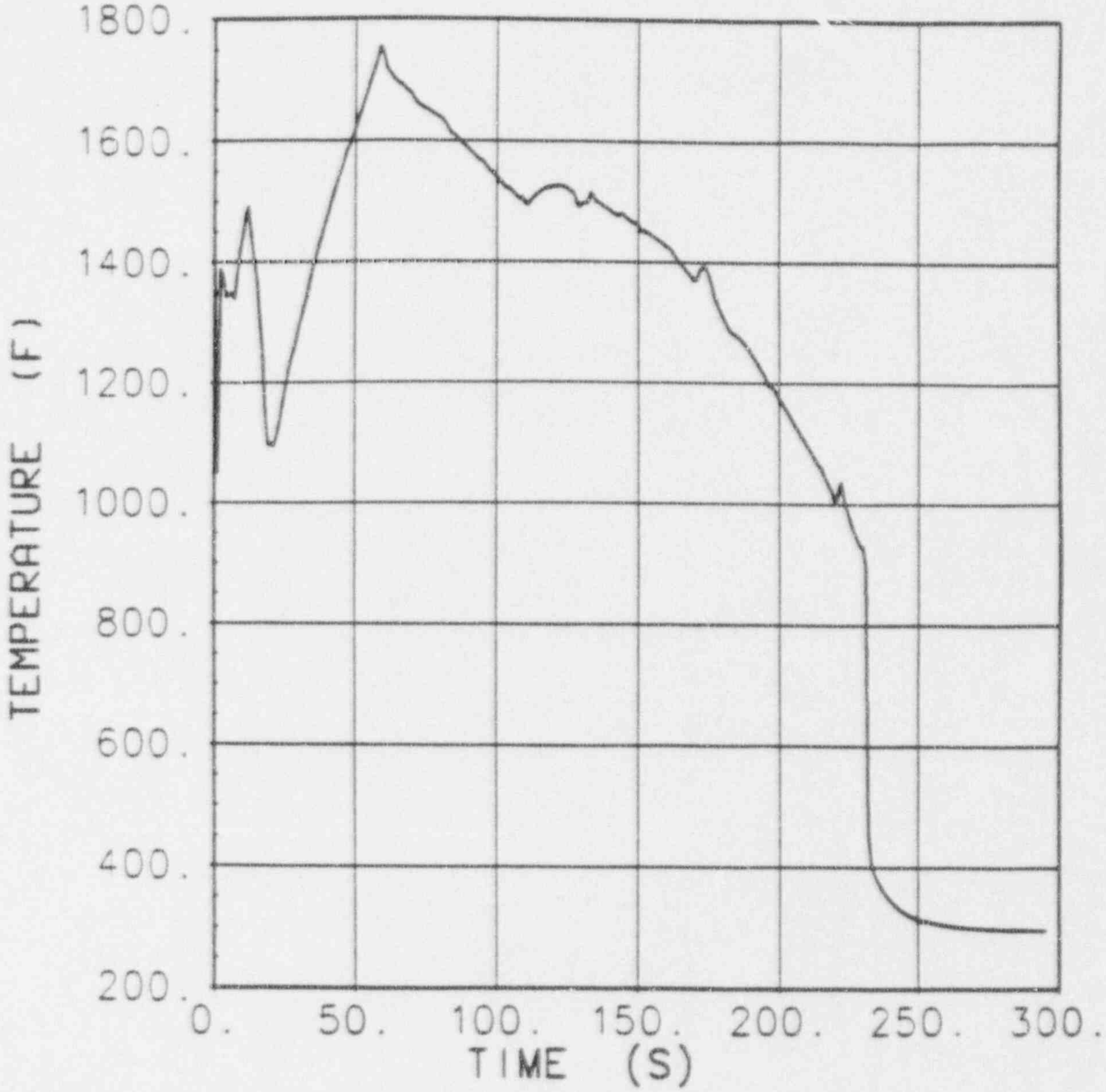
CLAD HOT SPOT FLUID TEMPERATURE
(C_D = 0.6, MAX SI)

FIGURE 2-6K



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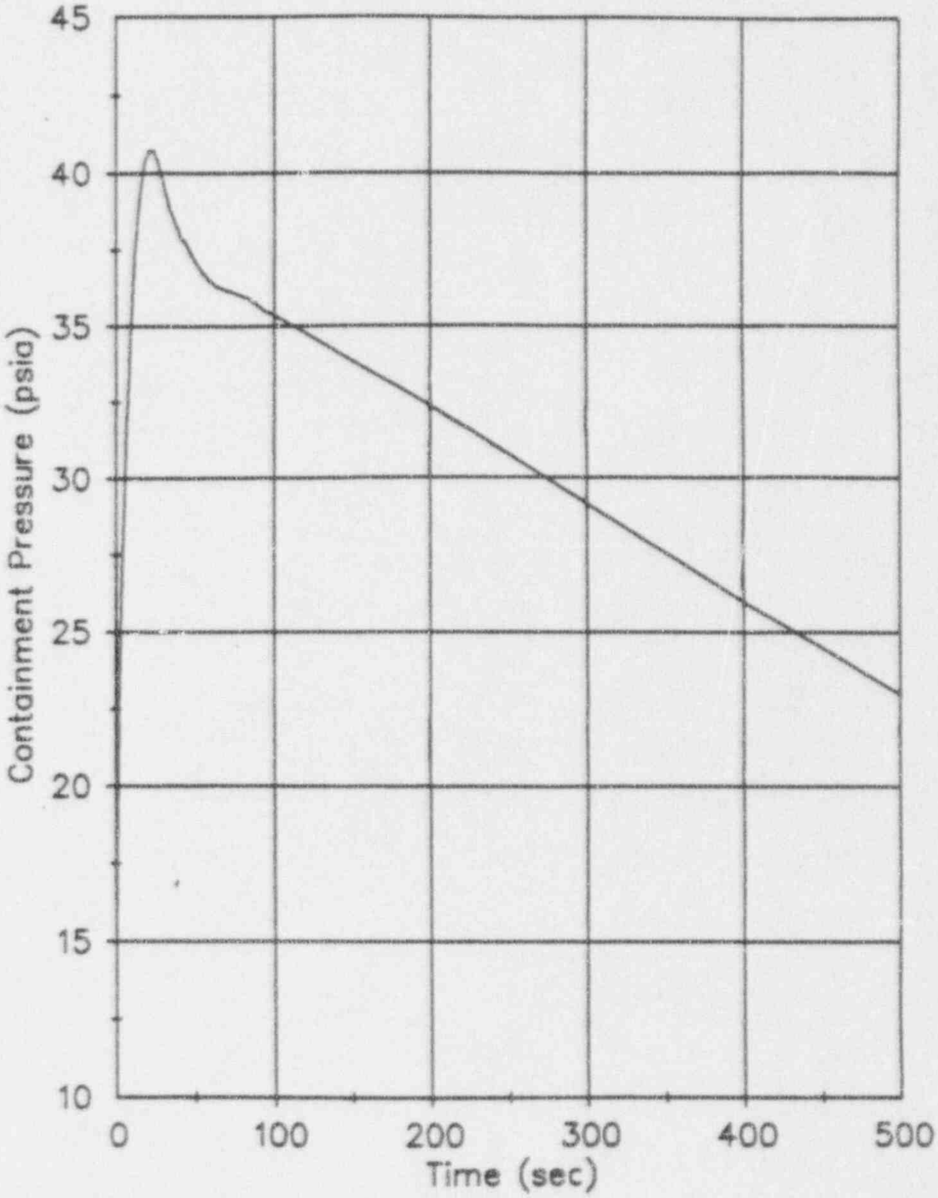
HOT ROD PEAK CLAD TEMPERATURE

(C_D = 0.6, MAX SI)

FIGURE 2-6L



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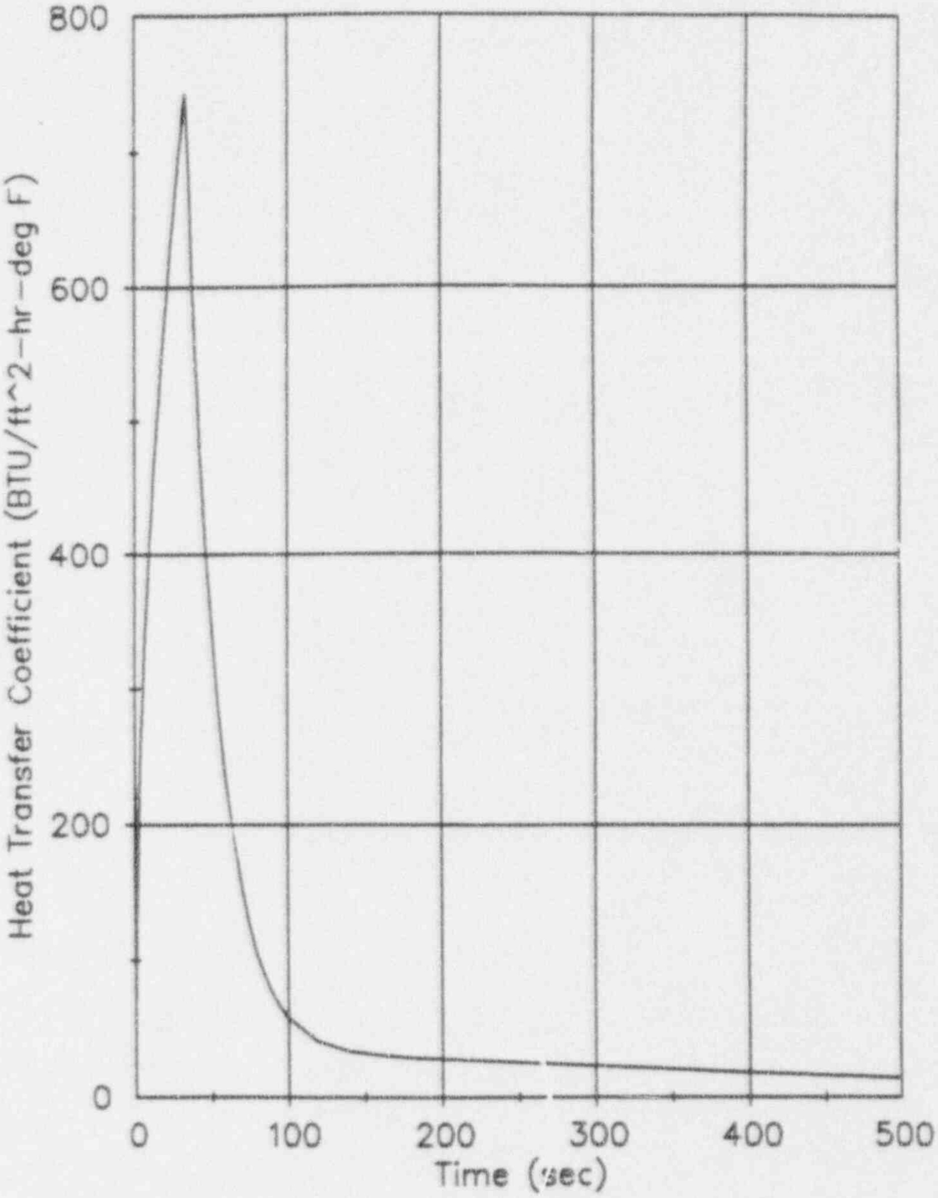
CONTAINMENT PRESSURE

(C_D = 0.6, MAX SI)

FIGURE 2-6M



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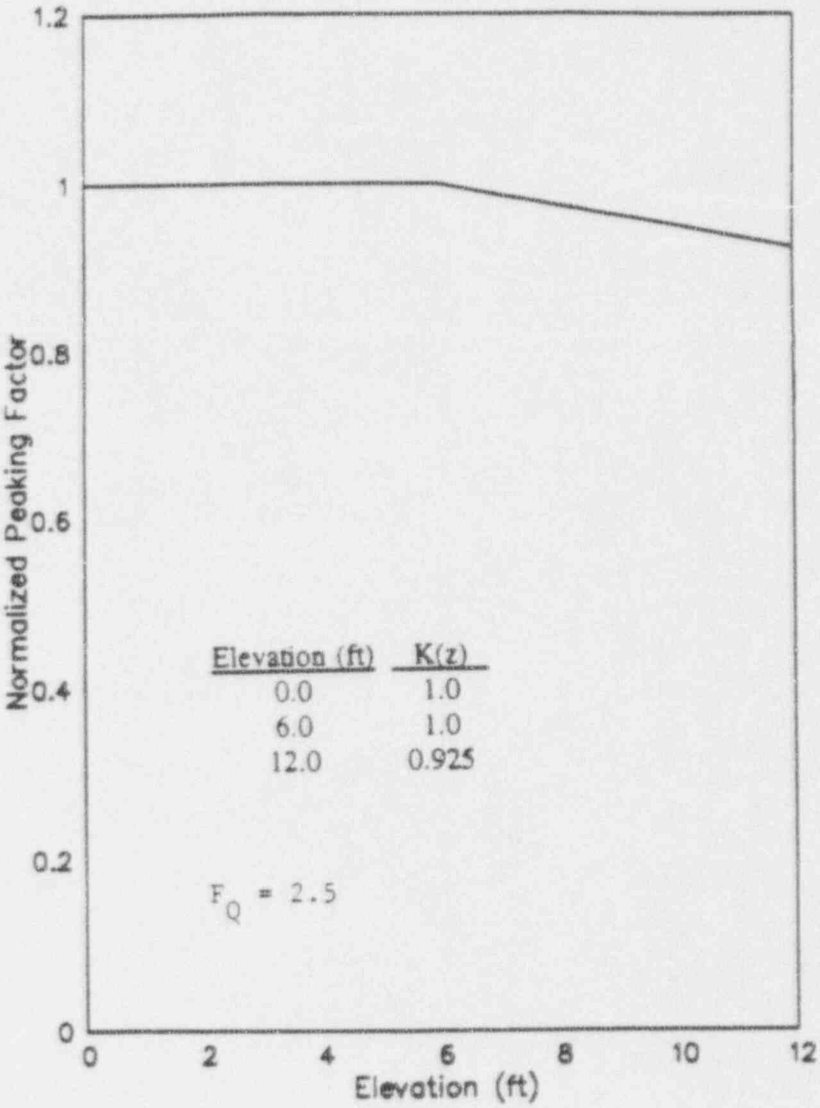
CONTAINMENT CONDENSING WALL HEAT TRANSFER COEFFICIENT (C_D = 0.6, MAX SI)

FIGURE 2-6N



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K(z) CURVE

FIGURE 2-7

3.0 LOSS OF REACTOR COOLANT FROM SMALL RUPTURED PIPES OR FROM CRACKS IN LARGE PIPES WHICH ACTUATES THE ECCS (SMALL BREAK LOCA)

This section presents a description and results of the small break loss-of-coolant accident (LOCA) in conformance with 10 CFR 50.46 and Appendix K of 10 CFR 50 (Reference 1).

3.1 Identification of Causes and Frequency Classification

A LOCA is the result of a pipe rupture of the reactor coolant system (RCS) pressure boundary. Ruptures of small cross-section will cause expulsion of the coolant at a rate which can be accommodated by the high head safety injection pumps and which would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown. The coolant which would be released to containment contains the fission products present in it.

A small break, as considered in this section, is defined as a rupture of the RCS piping with a cross-sectional area less than 1.0 ft^2, in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This event is considered an American Nuclear Society (ANS) Condition III event, which is a fault which may occur very infrequently during the life of Seabrook Station.

3.2 Sequence of Events and Systems Operations

Should a small break LOCA occur, depressurization of the RCS causes fluid to flow into the loops from the pressurizer resulting in a pressure and level decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low-pressure trip setpoint is reached. Loss-of-Offsite Power (LOOP) is assumed to occur coincident with reactor trip. A safety injection signal is generated when the pressurizer low-low-pressure setpoint is reached. These counter measures limit the consequences of the accident in two ways:

A. Reactor trip and borated water injection supplement void formation in causing rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay. No credit is taken in the LOCA analysis for the boron content of the injection water. However, an average RCS/sump mixed boron concentration is calculated to ensure that the post-LOCA core remains subcritical. In addition, in the small break LOCA analysis, credit is taken for the insertion of Rod Cluster Control Assemblies (RCCAs) subsequent to the reactor trip signal, while assuming the most reactive RCCA is stuck in the full out position.

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

For small break LOCAs, the most limiting single active failure is the one that results in the minimum emergency core cooling system (ECCS) flow delivered to the RCS. This has been determined to be the loss of an emergency power train which results in the loss of one complete train of ECCS components. This means that credit can be taken for only one centrifugal charging pump (CCP), one safety injection pump (SIP), and one residual heat removal (RHR) (or low head) pump. During the small break transient, one ECCS train is assumed to start and deliver flow through the injection lines (one for each loop) with one branch injection line (RHR and SIP) spilling to the RCS backpressure. Because the line diameter may be less than the break size for small breaks, one charging pump injection line spills to containment backpressure. To minimize delivery to the reactor, the branch line chosen to spill is selected as the one with the minimum resistance. In addition, the SIP and CCP performance curves were degraded by 5%, the RHR pump performance curve was degraded 8.75%, and a 10 gpm flow imbalance was assumed for the high head safety injection pumps.

3.3 Description of Small Break LOCA Transient

Before the break occurs the plant is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. After the small break LOCA is initiated, reactor trip occurs due to a low pressurizer signal. During the earlier part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as the pumps coast down following LOOP. Upward flow through the core is maintained. However, the core flow is not sufficient to prevent a partial core uncovery. Subsequently, the ECCS provides sufficient core flow to cover the core.

During blowdown, heat from fission product decay, hot internals, and the vessel continues to be transferred to the RCS. The heat transfer between the RCS and the secondary system may be in either direction depending on the relative temperatures. Continued heat addition to the secondary system results in increased secondary system pressure which leads to steam relief via the main steam safety valves. The safety injection signal isolates normal feedwater flow by closing the main feedwater control and bypass valves. Makeup to the secondary is automatically provided by the auxiliary feedwater pumps. Loss-of-Offsite Power, assumed concurrent with reactor trip, initiates auxiliary feedwater flow by starting the auxiliary feedwater pumps. The secondary flow aids in the reduction of RCS pressure.

When the RCS depressurizes to approximately 600 psia, the cold leg accumulators begin to inject borated water into the reactor coolant loops. However, for most small breaks the vessel mixture level starts to increase, covering the fuel with ECCS pumped injection before accumulator injection begins.

3.4 Core and System Performance

3.4.1 Mathematical Model

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

3.4.2 Small Break LOCA Evaluation Model

For small breaks (less than 1.0 ft^2) the NOTRUMP digital computer code (References 16 and 17) is employed to calculate the transient depressurization of the RCS as well as to describe the mass and energy of the fluid flow through the break. The NOTRUMP computer code is a state-of-the-art one-dimensional general network code incorporating a number of advanced features. Among these are calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes and regime-dependent drift flux calculations with multiple-stacked fluid nodes and regime-dependent heat transfer correlations. The NOTRUMP small break LOCA ECCS evaluation model was developed to determine the RCS response to design basis small break LOCAs, and to address NRC concerns expressed in NUREG-0611 (Reference 18).

The RCS model is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicitly, while the intact loops are lumped into a single second loop. Transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum. The multinode capability of the program enables explicit, detailed spatial representation of various system components which, among other capabilities, enables a proper calculation of the behavior of the loop seal during a LOCA. The reactor core is represented as heated control volumes with associated phase separation models to permit transient mixture height calculations.

Clad thermal analyses are performed with the LOCTA-IV code (Reference 19) using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions (Figure 3-1). Figure 3-2 depicts the hot rod axial power shape used to perform the small break LOCA analysis. The shape was chosen because it represents a distribution with power concentrated in the upper regions of the core. Such a distribution is limiting for small break LOCA because it minimizes

coolant level swell, while maximizing vapor superheating and fuel rod heat generation at the uncovered elevations. The small break analysis assumes that the core continues to operate at full power until the control rods are completely inserted. For conservatism, it is assumed that the most reactive RCCA does not insert.

3.4.3 Input Parameters and Initial Conditions

Important input parameters and initial conditions used in the analysis are listed in Table 3-1. The safety injection performance, as modeled in the small break analysis, is presented in Figure 3-3. Cases analyzed are given in Table 3-2.

3.5 Results

NUREG-0737 Section II.K.3.31 (Reference 20) requires a plant-specific small break LOCA analysis using an Evaluation Model revised per Section II.K.3.30. In accordance with NRC generic letter 83-35 (Reference 21), generic analyses using NOTRUMP (References 16 and 17) were performed and are presented in References 22 and 23. Those results demonstrate that in a comparison of cold leg, hot leg and pump suction leg break locations, the cold leg break of less than 10 inches in diameter is limiting. Calculations were made for the 3, 4 inch and 6 inch break sizes. The results of these calculations are summarized in Tables 3-3 and 3-4.

It was determined that, because of the low calculated PCT, rod burst and blockage effects would not have a significant effect on the small break results for Seabrook Station. Therefore a fuel assembly burnup sensitivity study was not required.

Figures 3-4 through 3-6 present the results of the cases analyzed for the small break LOCA.

Figures A	Reactor Coolant System Pressure (Calculated Core Pressure)
Figures B	Core Mixture Level
Figures C	Hot Spot Clad Temperature
Figures D	Core Exit Steam Mass Flow Rate
Figures E	Hot Rod Heat Transfer Coefficient
Figures F	Fluid Temperature at the Clad Hot Spot
Figures G	Break Mass Flow Rate
Figures H	Pumped Safety Injection Mass Flow Rate

The limiting small break PCT is 1082°F, which is less than the acceptance criteria limit of 2200°F. This PCT, from the 4-inch case, bounds all fuel types and features analyzed. Addition of the 8°F penalty for the increase in T_{AVG} associated with RTD bypass elimination yields a total PCT of 1090°F. This licensing basis PCT remains below the 2200°F acceptance criterium of 10 CFR 50.46. The maximum local metal-water reaction is well below the embrittlement limit of 17 percent as required by 10 CFR 50.46.

The total core metal-water reaction is less than 1.0 percent for all breaks analyzed, corresponding to less than 1.0 percent hydrogen generation as required by 10 CFR 50.46. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

TABLE 3-1
INPUT PARAMETERS USED IN THE
SMALL BREAK LOCA ECCS ANALYSIS

License Core Power (MWt)	3411*
Total Peaking Factor, F_Q	2.5
Hot Channel Enthalpy Rise Factor, FΔH	1.65
Power Shape	See Figure 3-2 (+20% A.O.)
Fuel Assembly Array	17x17 ZIRLO or Zircaloy-4, 0.374" Fuel Rods
Accumulator Water Volume (ft^3/accumulator)	850 (Nominal)
Accumulator Gas Pressure, Minimum (psia)	600
Safety Injection Pumped Flow (SIPs and CCPs degraded 5%, RHR degraded 8.75% CCP flow imbalance = 10 gpm)	See Figure 3-3
Total System Flow Rate (lbm/sec)	38788
Vessel Inlet Temperature (°F)	557.64
Vessel Outlet Temperature (°F)	619.34
Reactor Coolant Pressure (psia)	2300.0
Steam Pressure (psia)	955.7
Steam Generator Tube Plugging Level (%)	13**
Minimum Refueling Water Storage Tank Temperature (°F)	50.0
Fuel Backfill Pressure (psig)	275 (100 for IFBA)
Low Pressurizer Pressure Setpoints (psia):	
Reactor Trip	1860
Safety Injection Signal	1665
Rod Drop Time (sec)	4.4
Safety Injection Delay Time (sec)	30
Feedwater Isolation Delay after Reactor Trip (sec)	2.0
Feedwater Isolation Valve Closure Time (sec)	0.0

* Two percent is added to this power to account for calorimetric error.
** Analysis is performed at 13% SGTP, however, the large break analysis limits operation to 8% SGTP. 5% SGTP is allocated for steam generator tube crush during a combined LOCA/seismic event.

TABLE 3-2
SMALL BREAK LOCA - CASES ANALYZED

CASE I -	3-Inch Break, 3411 MWt Core Power, F_Q=2.5, FΔH=1.65, P-BAR-HA=1.469.
CASE II -	4-Inch Break, 3411 MWt Core Power, F_Q=2.5, FΔH=1.65, P-BAR-HA=1.469. This case was found to be limiting and bounds both ZIRLO and Zircaloy-4 cladding.
CASE III -	6-Inch Break, 3411 MWt Core Power, F_Q=2.5, FΔH=1.65, P-BAR-HA=1.469.

All cases model 13% steam generator tube plugging (but are limited by the large break analysis to operation with 8% SGTP) and 2% reduction in thermal design flow ([illegible] gpm/loop).

TABLE 3-3
SMALL BREAK LOCA RESULTS - TIME SEQUENCE OF EVENTS

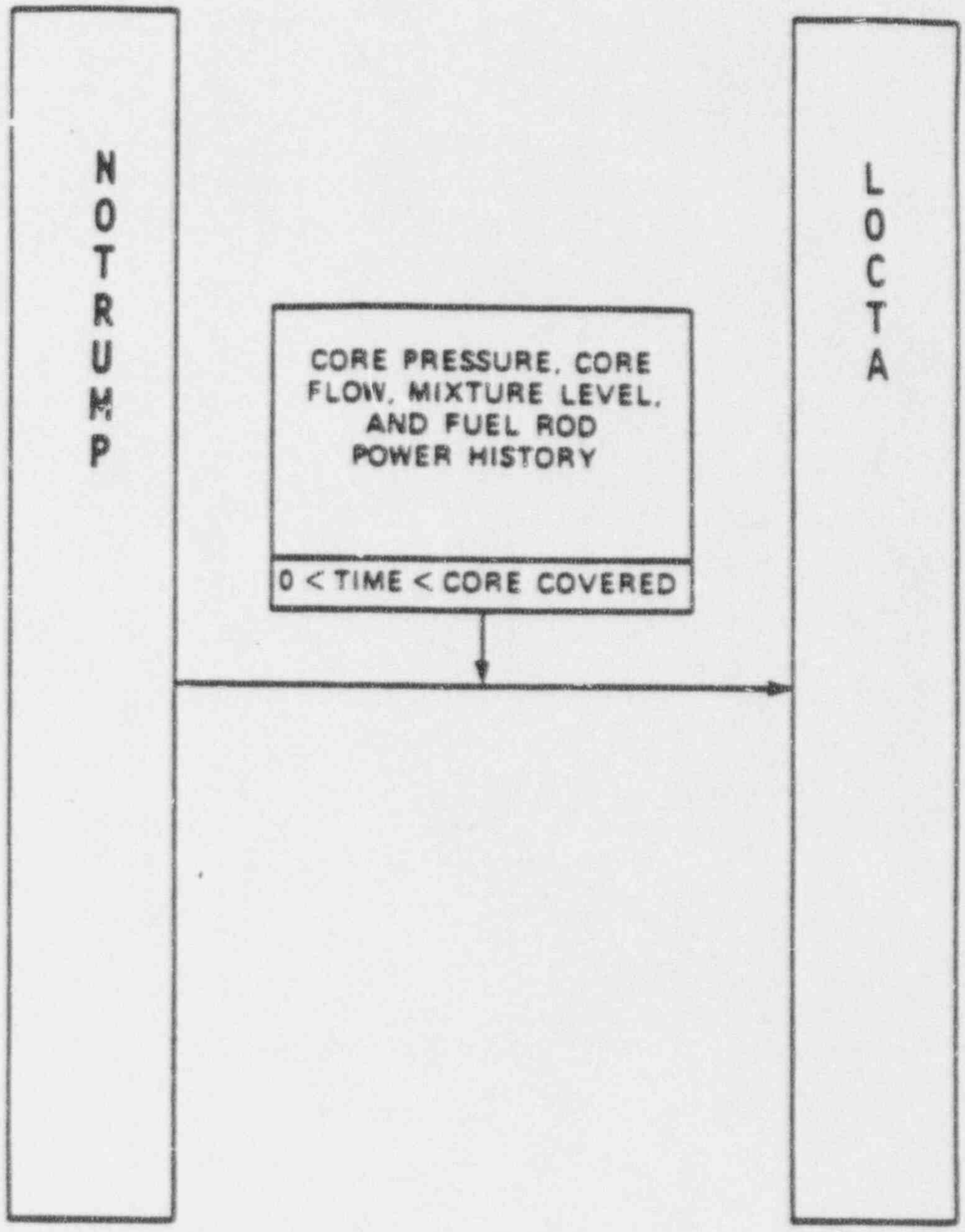
Seconds	Case I 3-inch	Case II 4-inch	Case III 6-inch
Start of LOCA	0.00	0.00	0.00
Reactor Trip Setpoint Exceeded	20.0	11.8	7.4
Safety Injection Setpoint Exceeded	28.9	20.2	13.5
Pump Injection Begins	58.9	50.2	43.5
Start of Auxiliary Feedwater Delivery	95.0	86.8	82.4
Initial Loop Seal Venting	420	283	144
Loop Seal Uncovery	N/A	321	166
Loop Seal Recovery	N/A	345	172
Boil-off Core Uncovery	1210	687	220
Accumulator Injection Begins	(1)	925	363
Peak Clad Temperature	1710	942	421
Top of Core Recovered	(2)	1459	436
Si Flow Rate Exceeds Break Flow Rate	1775	1335	(3)

(1) System pressure never drops below the accumulator cut-in pressure (600 psia).
(2) Although the core is not yet covered in this case, SI flow exceeds break flow and the clad temperature transient is over.
(3) Although SI flow has not yet matched break flow, the core is covered, the clad temperature transient has ended, and the total RCS mass is increasing.

TABLE 3-4
SMALL BREAK LOCA RESULTS - FUEL CLADDING DATA

	Case I 3-inch	Case II 4-inch	Case III 6-inch
Peak Clad Temperature (°F)	1052	1090*	962
Peak Clad Temperature Location (ft)	11.25	11.00	11.00
Peak Clad Temperature Time (sec)	1710	942	421
Maximum Local Zr/H_2O Reaction (%)	0.0492	0.0396	0.0335
Maximum Zr/H_2O Reaction Location (ft)	11.25	11.25	11.00
Total Zr/H_2O Reaction (%)	<1.0	<1.0	<1.0
Rod Burst	None	None	None

* Includes 8.0°F evaluation penalty for Increased Temperature Uncertainty (±5°F) associated with RTD Bypass Elimination.



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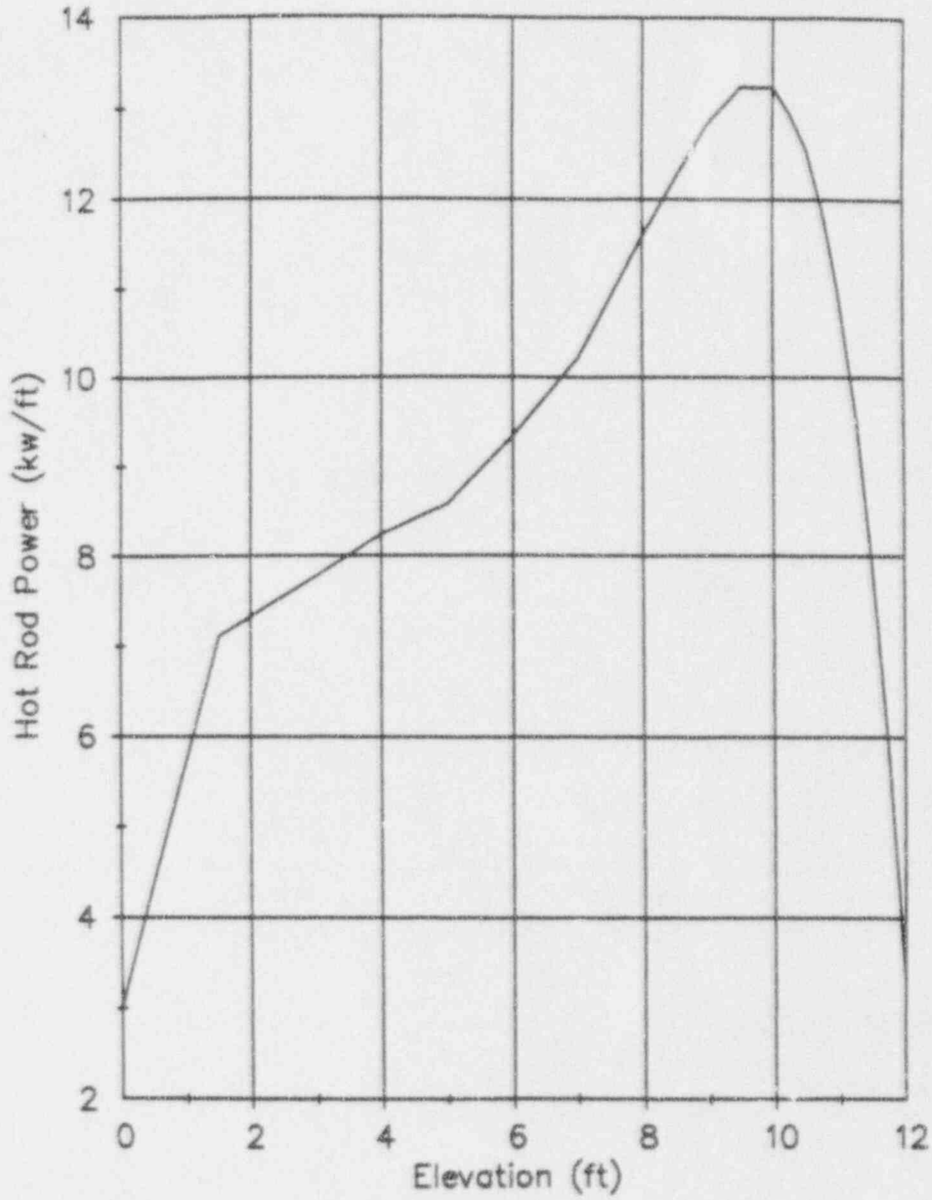
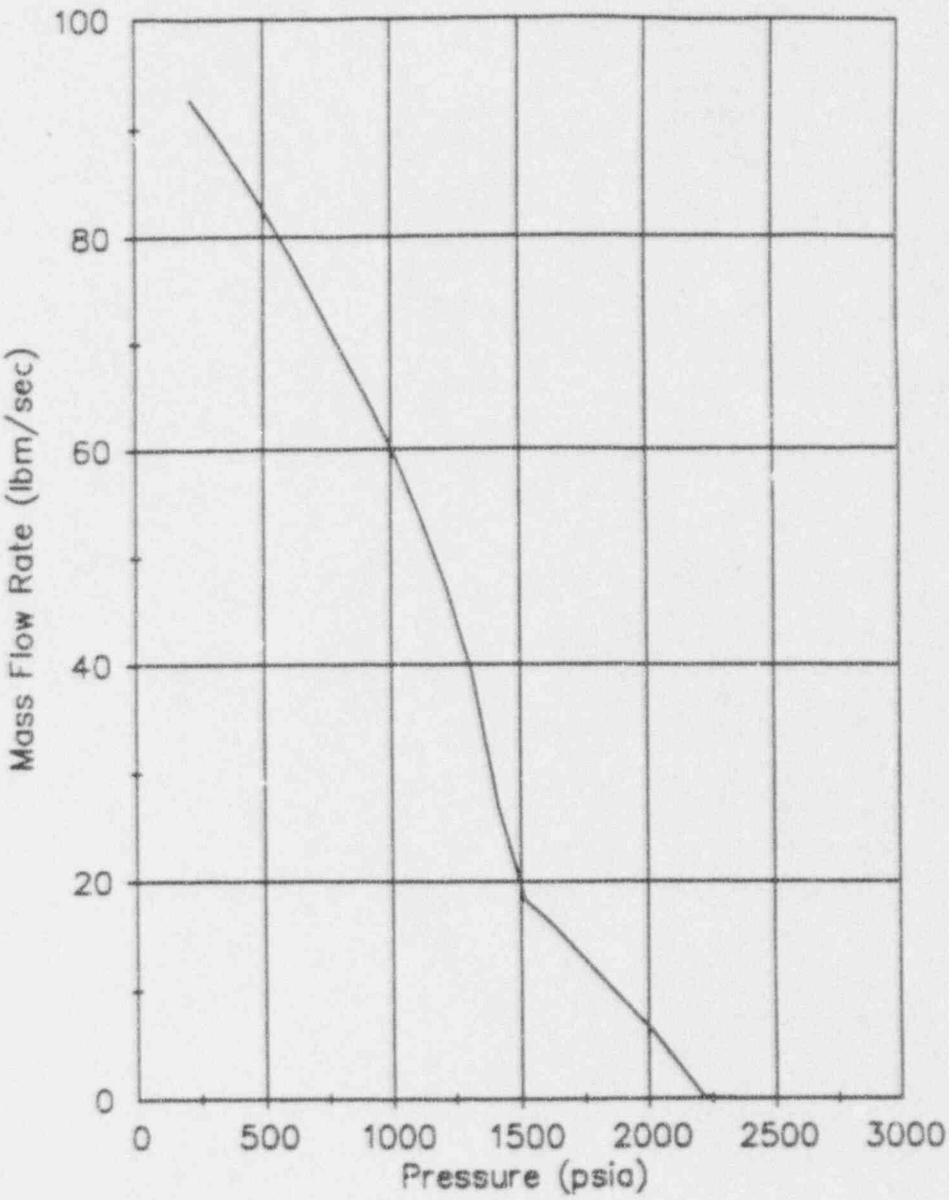
SMALL BREAK LOCA CODE INTERFACE

FIGURE 3-1



WESTINGHOUSE ELECTRIC CORPORATION Nuclear and Advanced Technology Divisions	SMALL BREAK LOCA HOT ROD POWER SHAPE
	FIGURE 3-2



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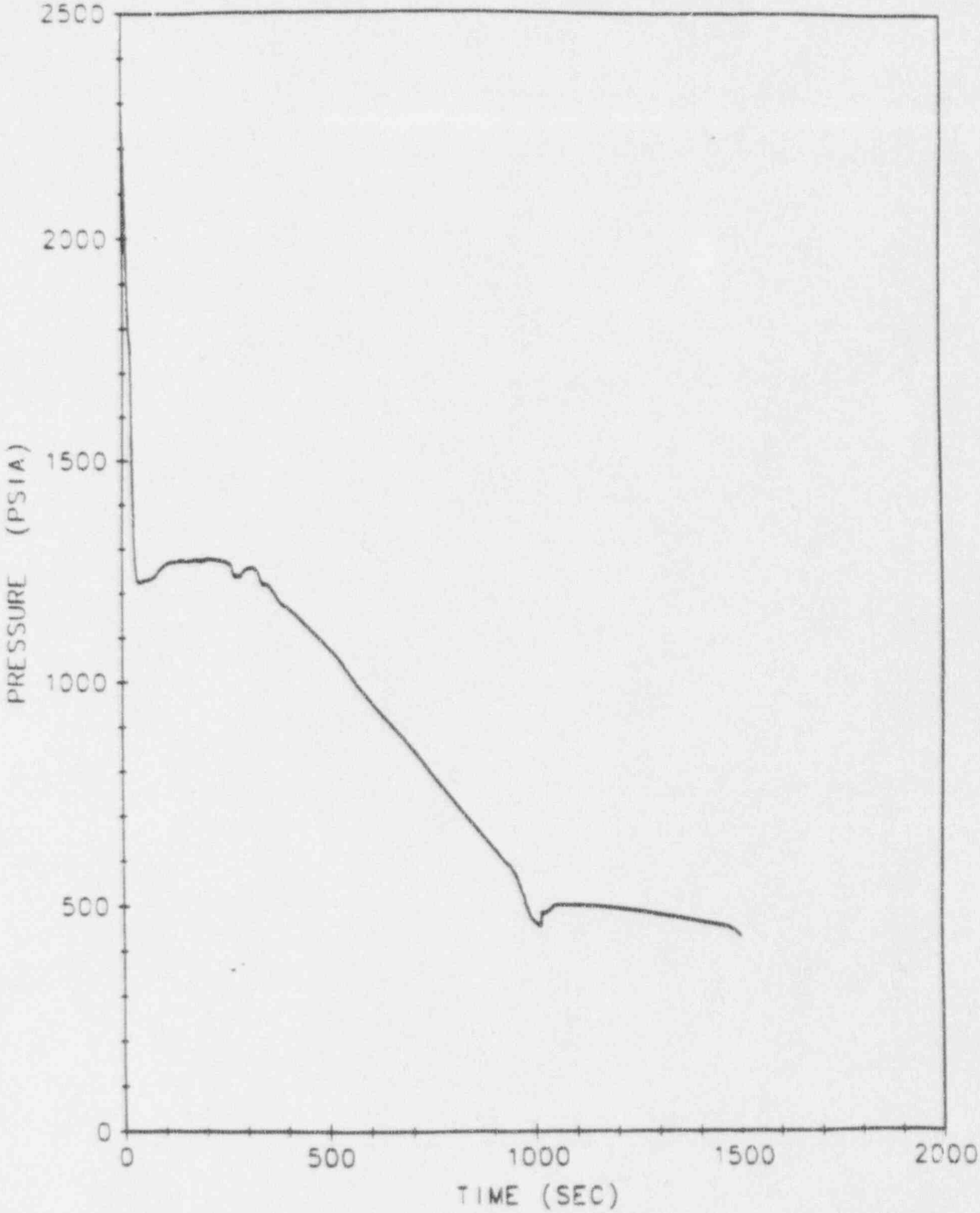
PUMPED SAFETY INJECTION FLOW vs. RCS PRESSURE

FIGURE 3-3



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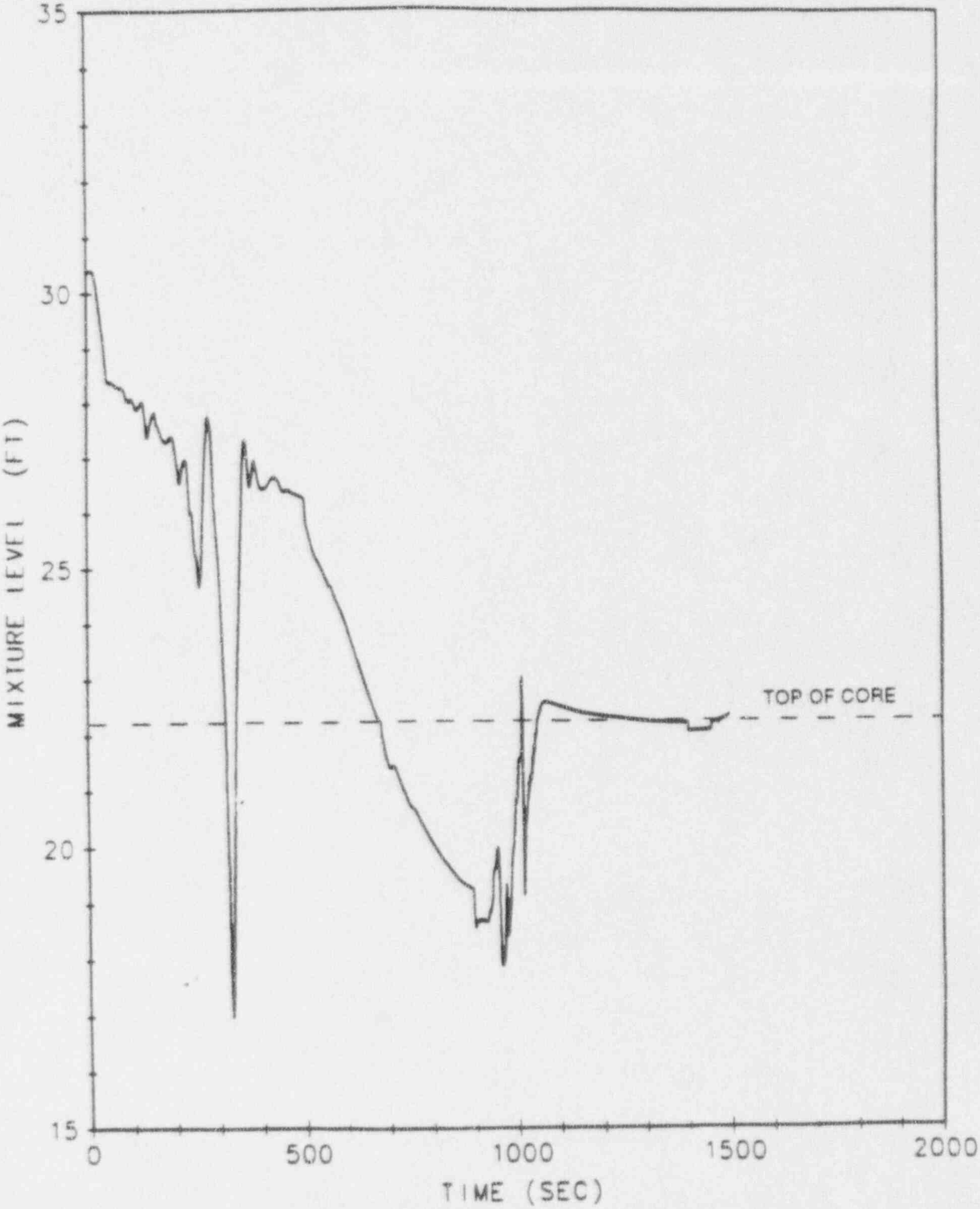
RCS PRESSURE

(4 INCH BREAK)

FIGURE 3-4A



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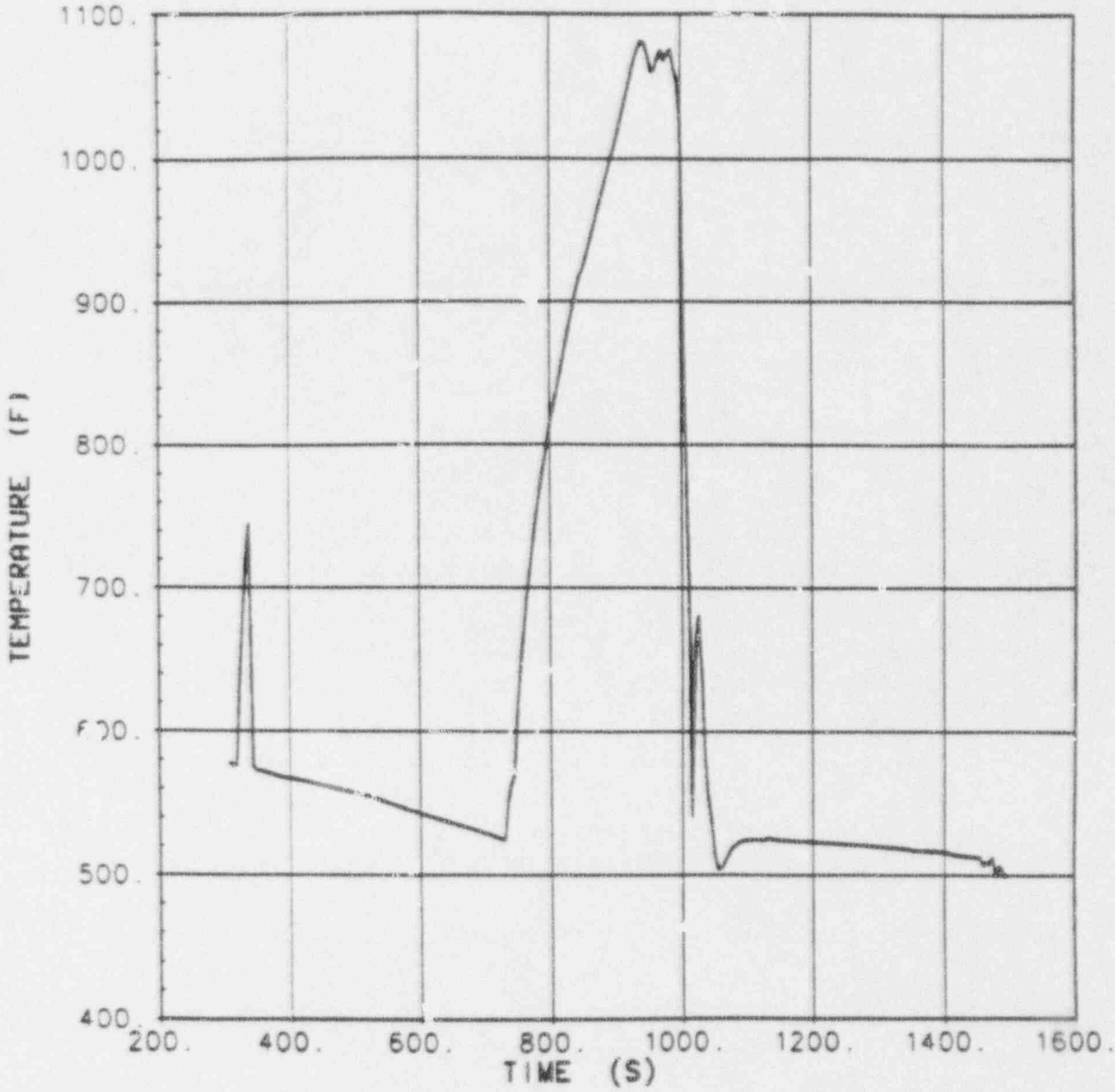
Nuclear and Advanced Technology Divisions

CORE MIXTURE LEVEL

(4 INCH BREAK)

FIGURE 3-4B



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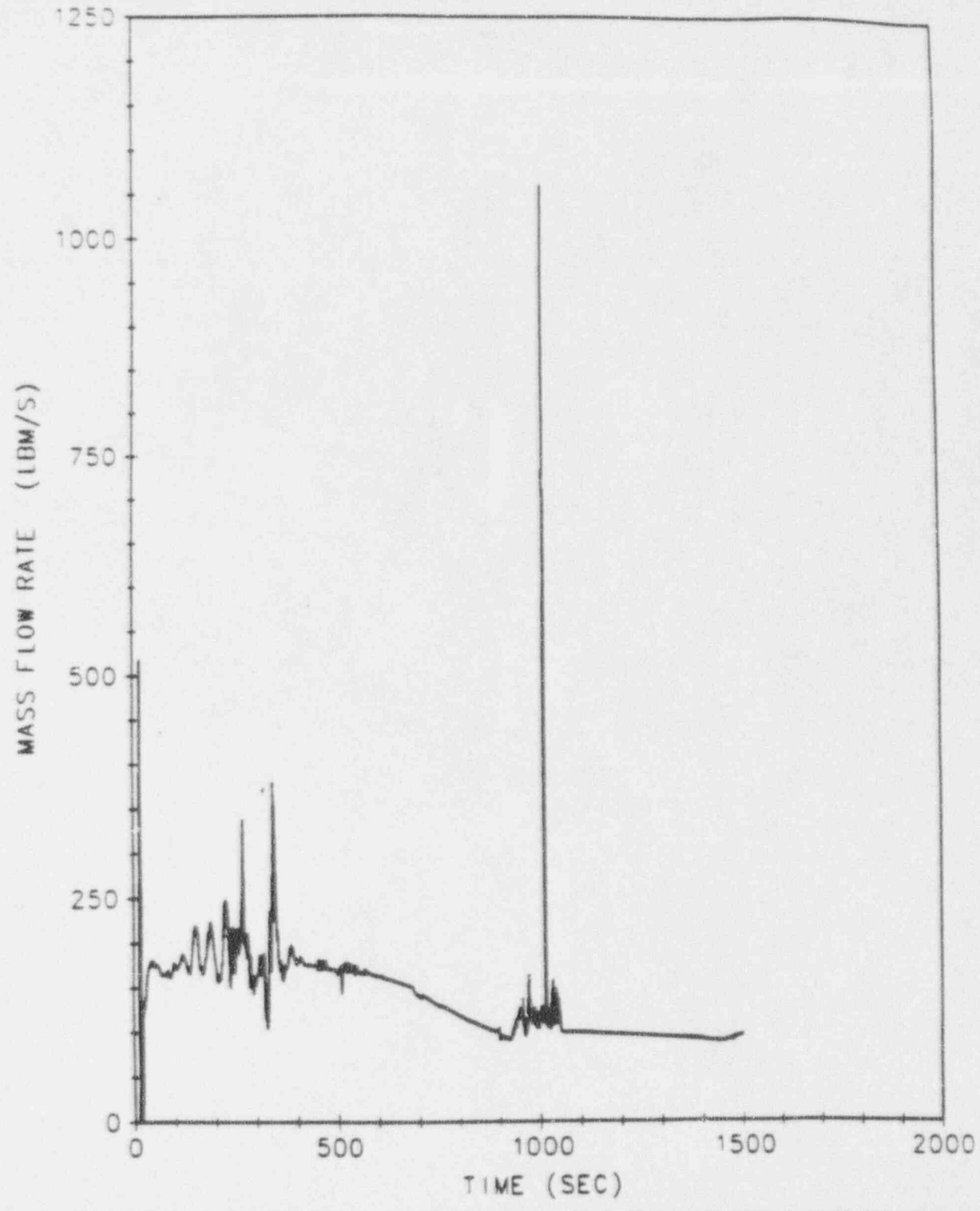
PEAK CLAD TEMPERATURE

(4 INCH BREAK)

FIGURE 3-4C



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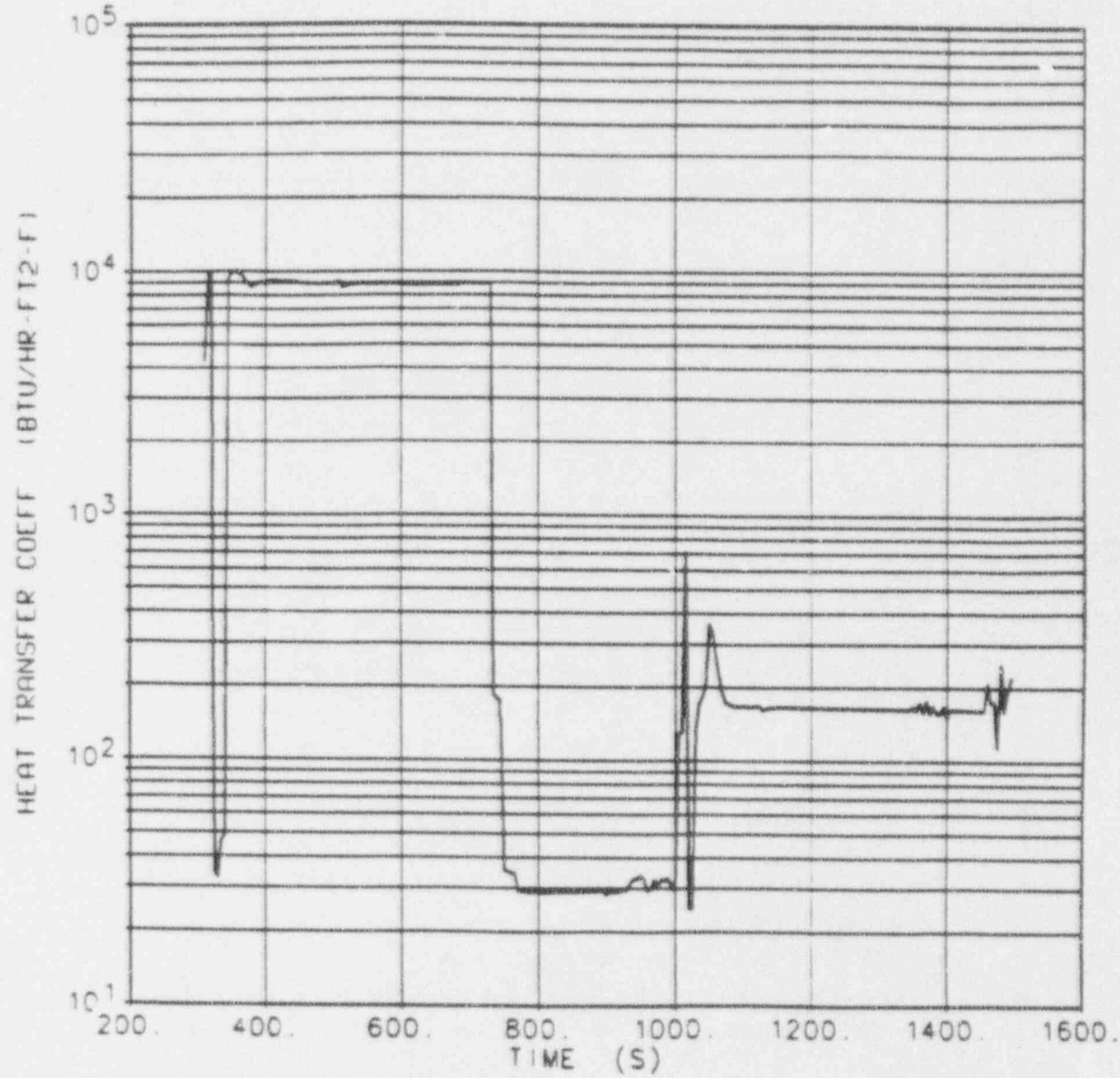
CORE EXIT STEAM MASS FLOW RATE (4 INCH BREAK)

FIGURE 3-4D



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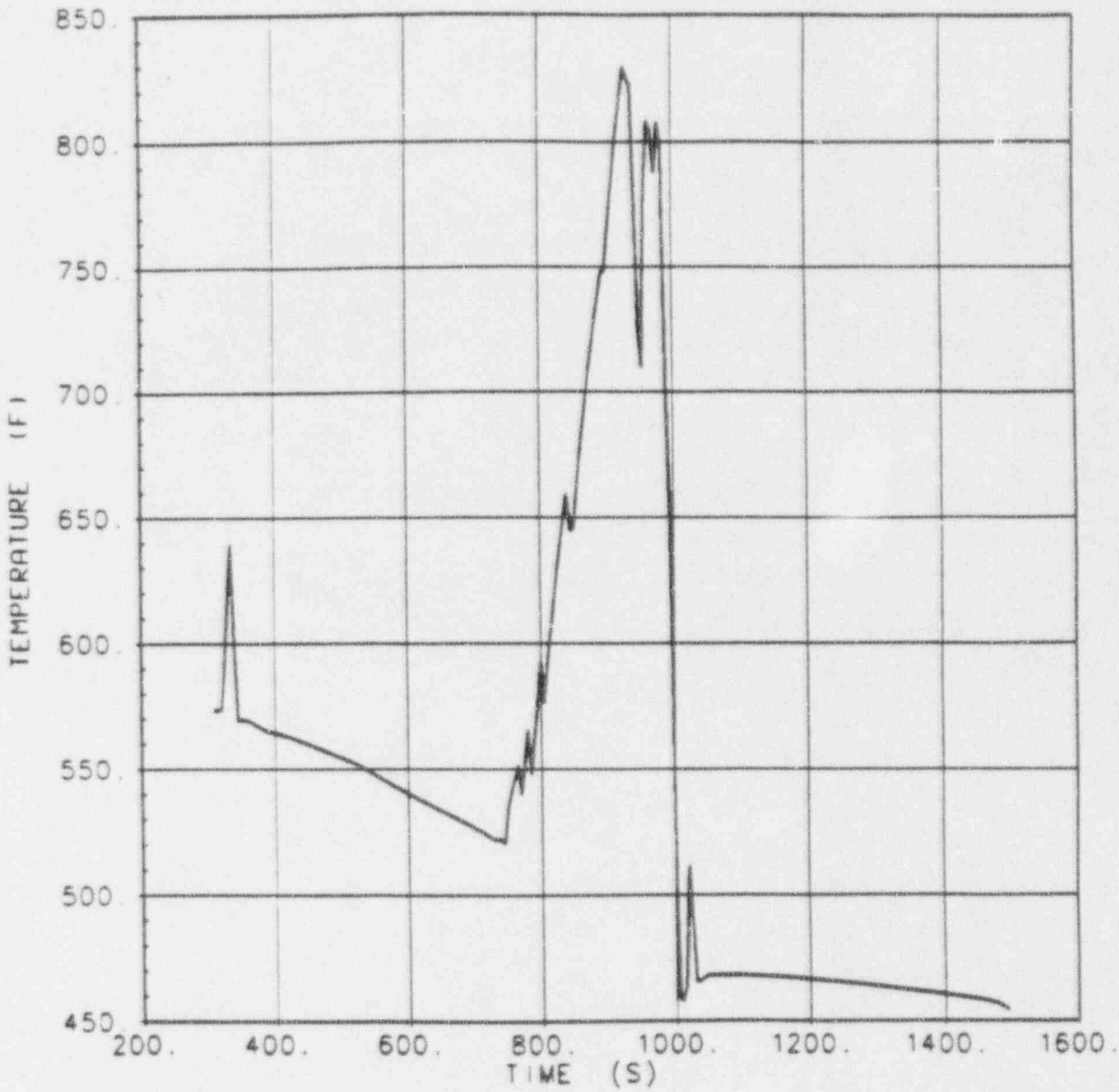
HOT ROD HEAT TRANSFER COEFFICIENT (4 INCH BREAK)

FIGURE 3-4E



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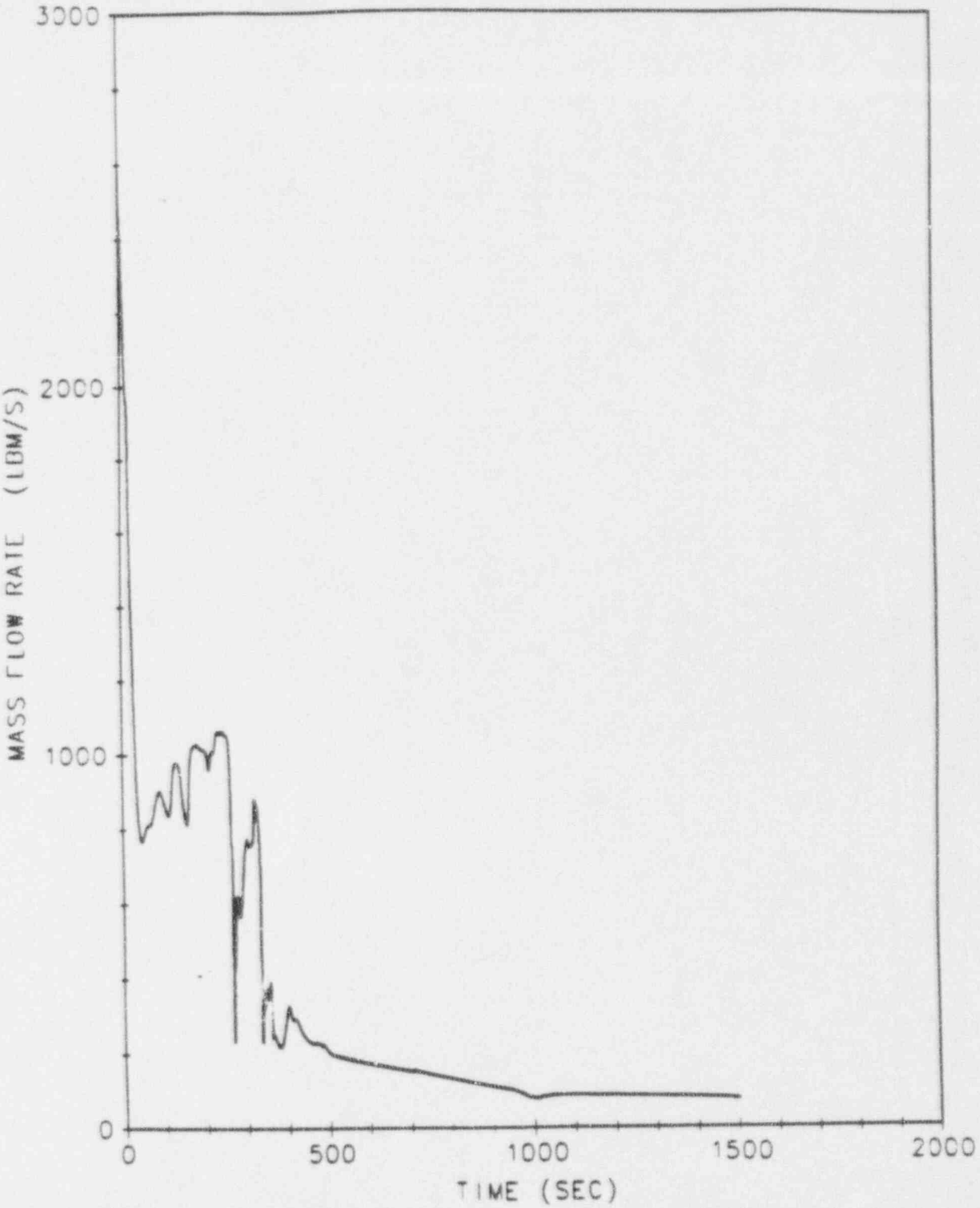
CLAD HOT SPOT FLUID TEMPERATURE (4 INCH BREAK)

FIGURE 3-4F



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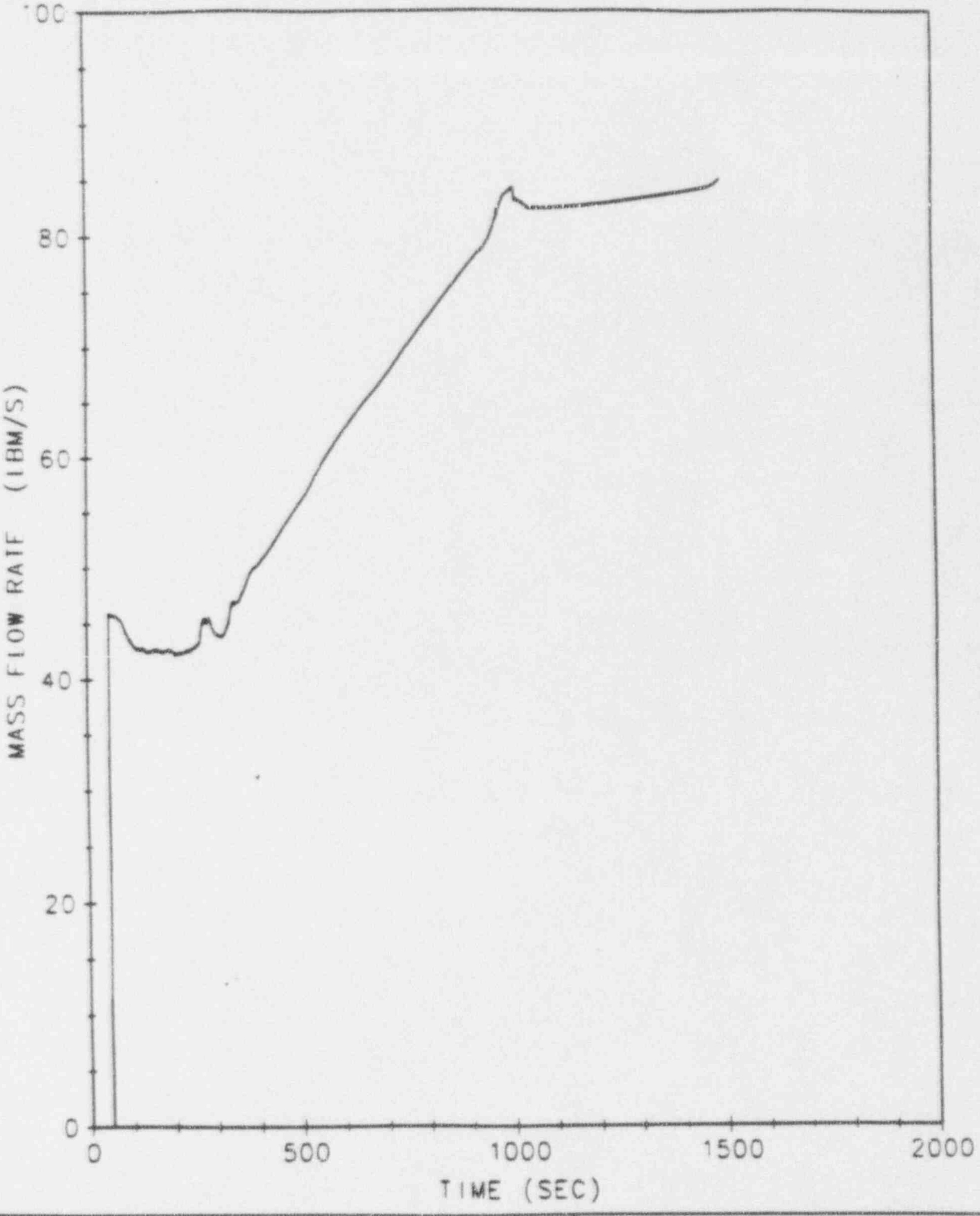
BREAK MASS FLOW RATE

(4 INCH BREAK)

FIGURE 3-4G



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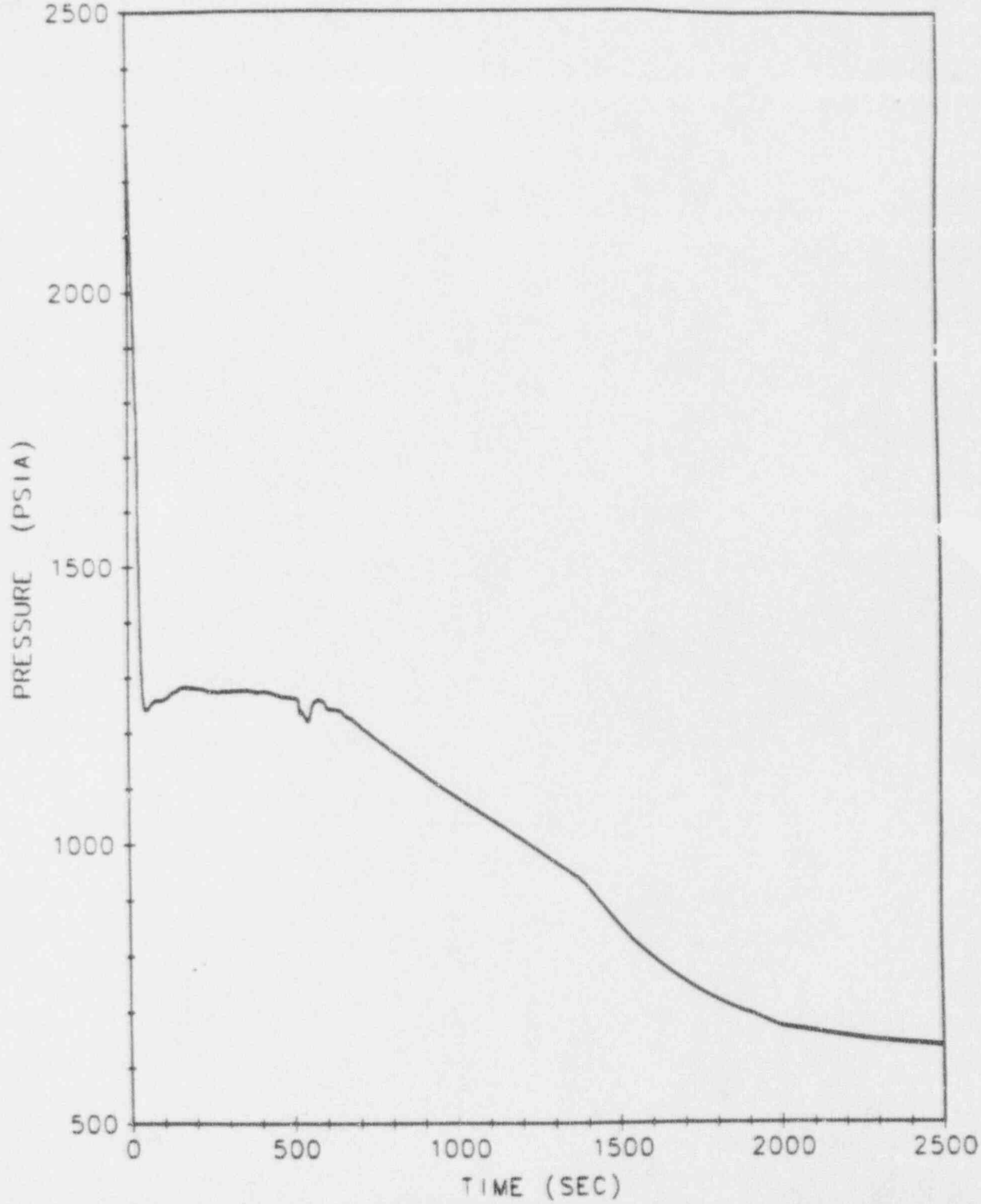
PUMPED SAFETY INJECTION MASS FLOW RATE (4 INCH BREAK)

FIGURE 3-4H



WESTINGHOUSE ELECTRIC CORPORATION
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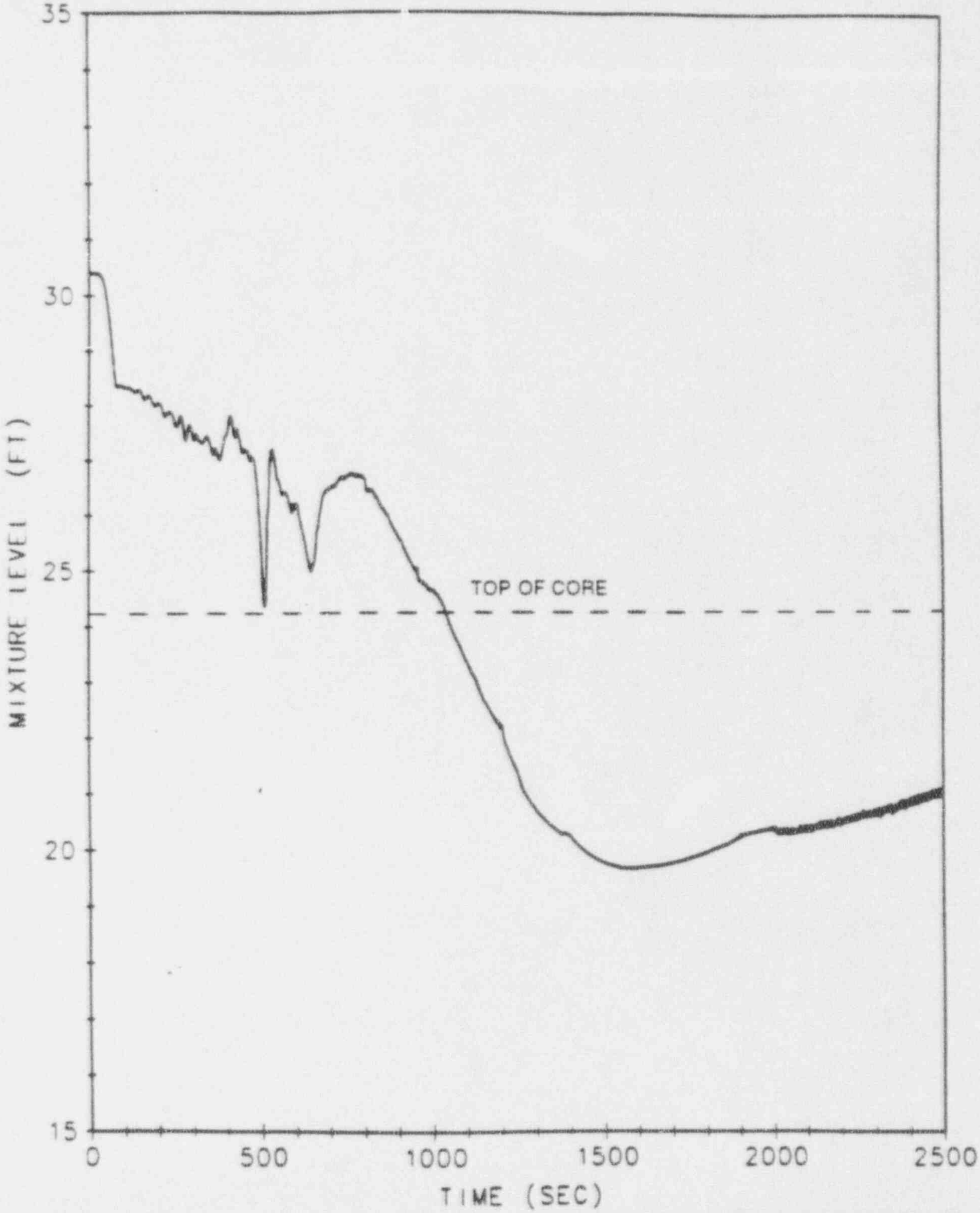
RCS PRESSURE

(3 INCH BREAK)

FIGURE 3-5A



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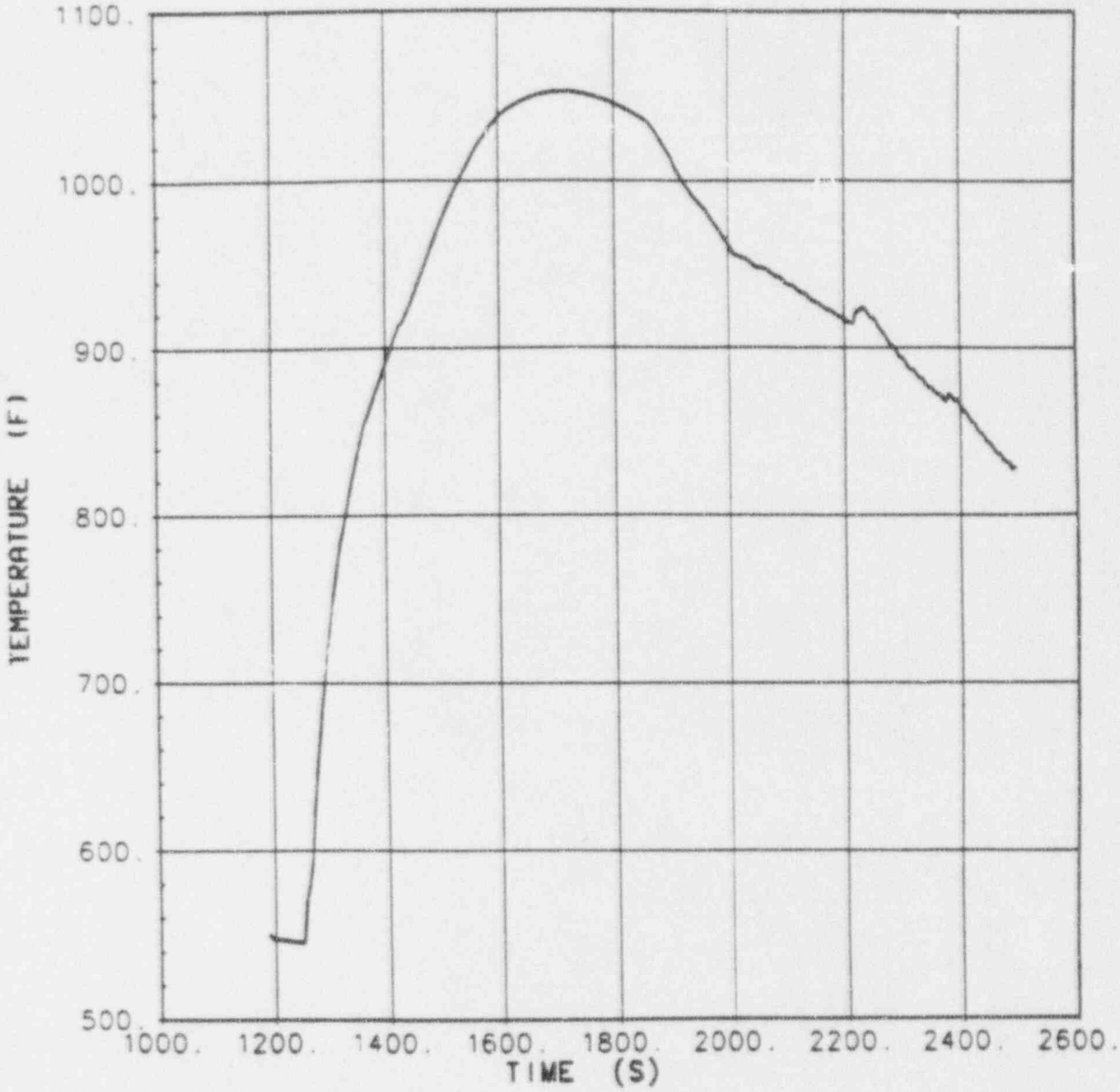
CORE MIXTURE LEVEL

(3 INCH BREAK)

FIGURE 3-5B



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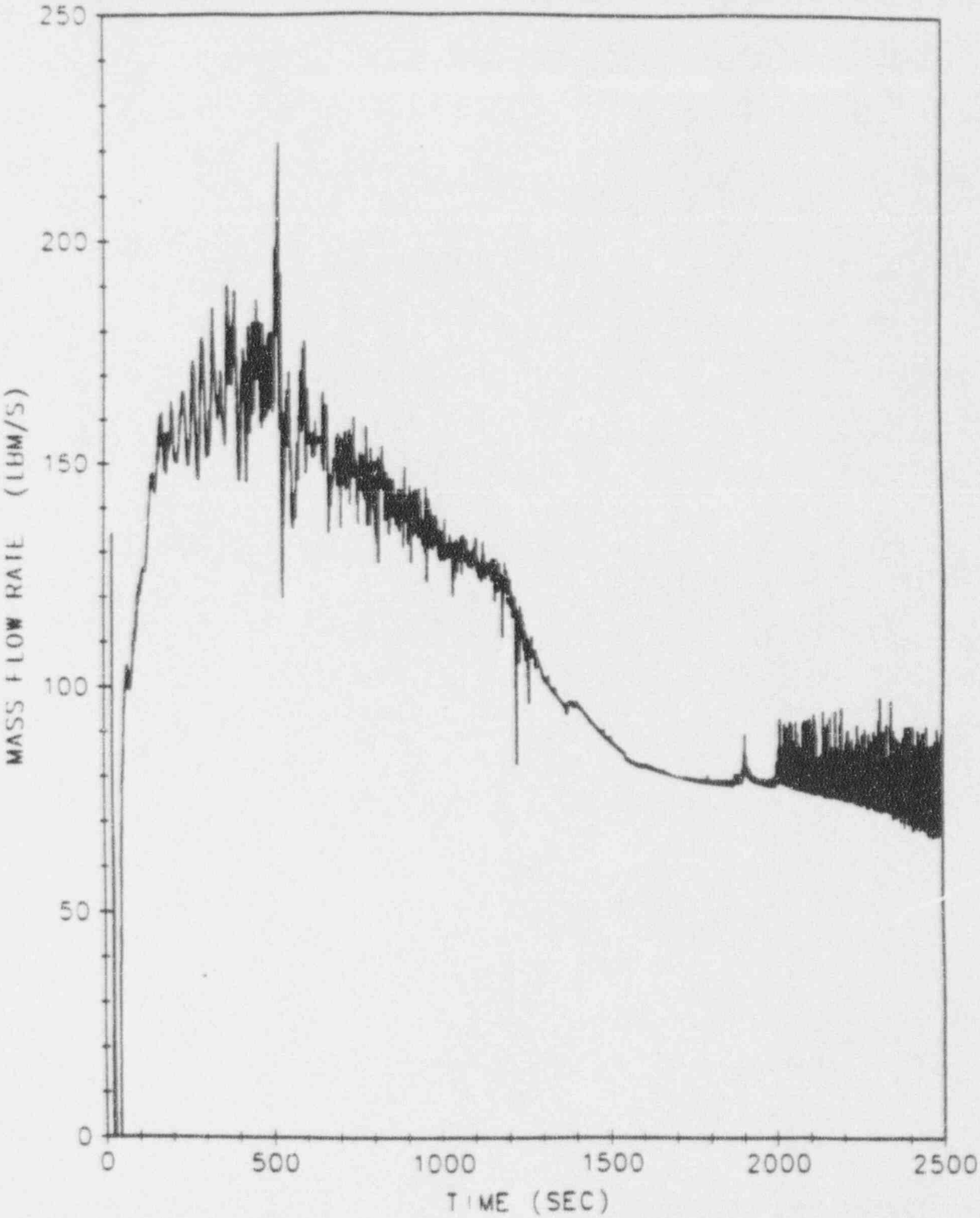
PEAK CLAD TEMPERATURE

(3 INCH BREAK)

FIGURE 3-5C



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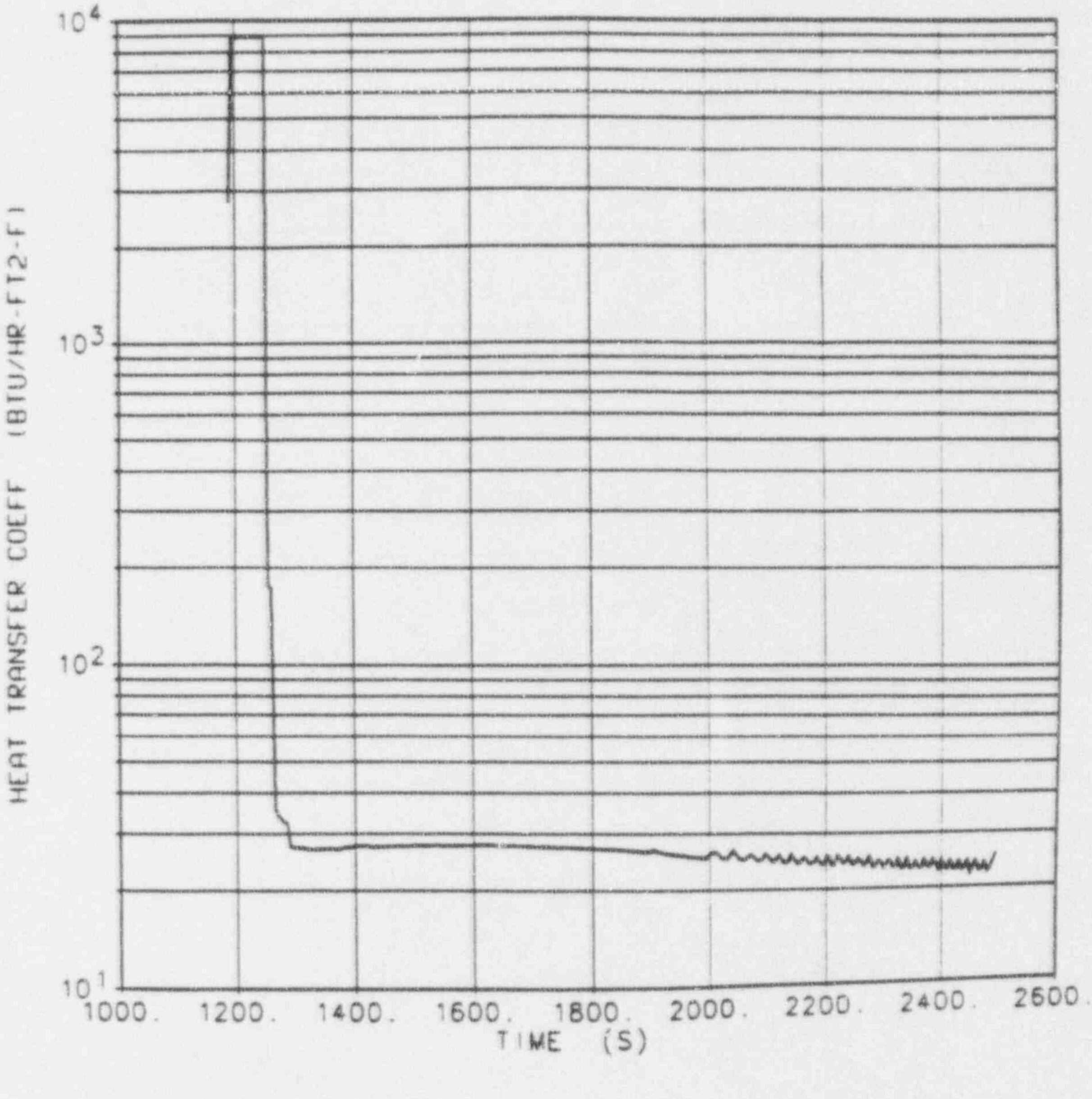
CORE EXIT STEAM MASS FLOW RATE (3 INCH BREAK)

FIGURE 3-5D



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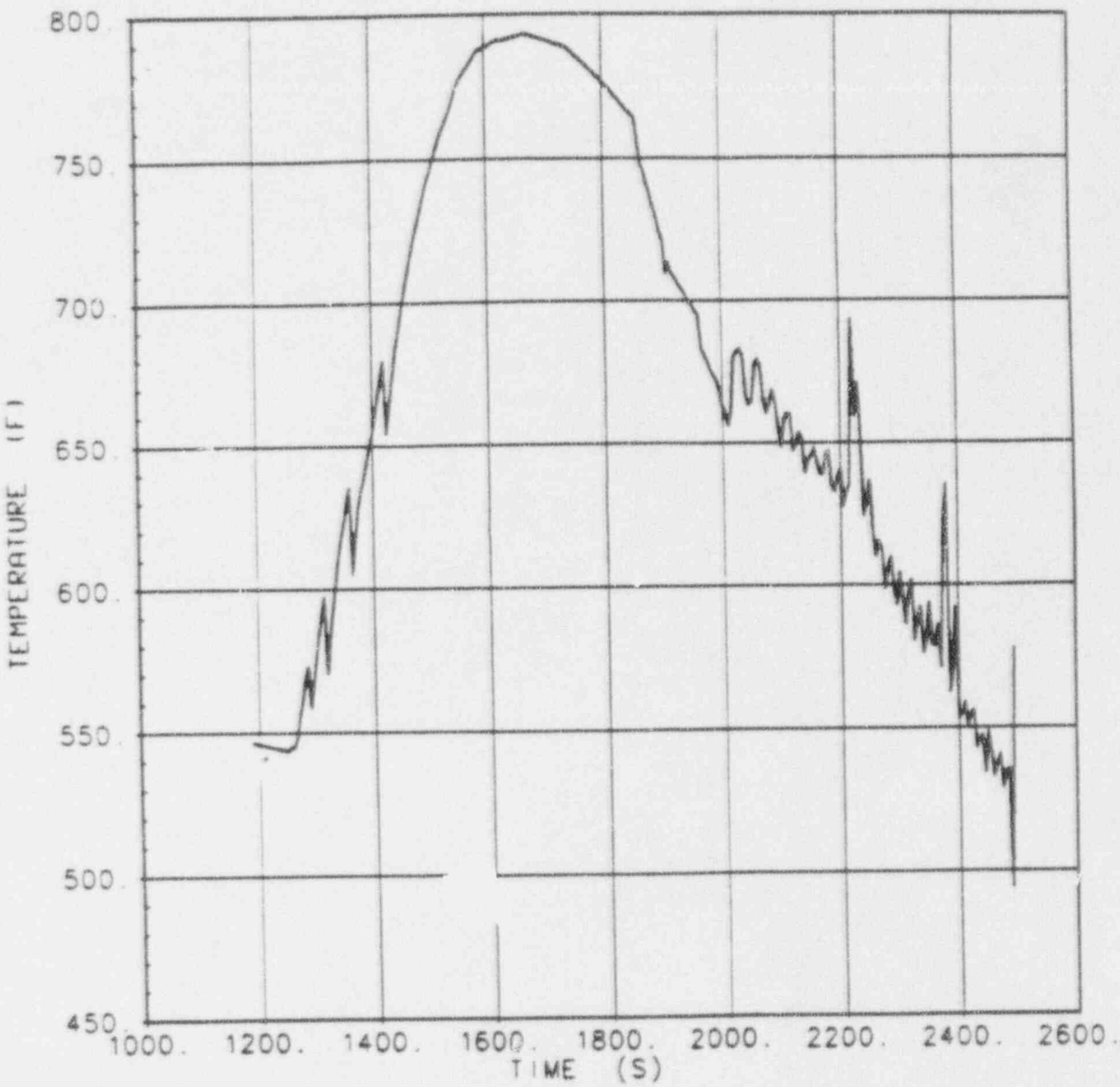
HOT ROD HEAT TRANSFER COEFFICIENT (3 INCH BREAK)

FIGURE 3-5E



WESTINGHOUSE ELECTRIC CORPORATION
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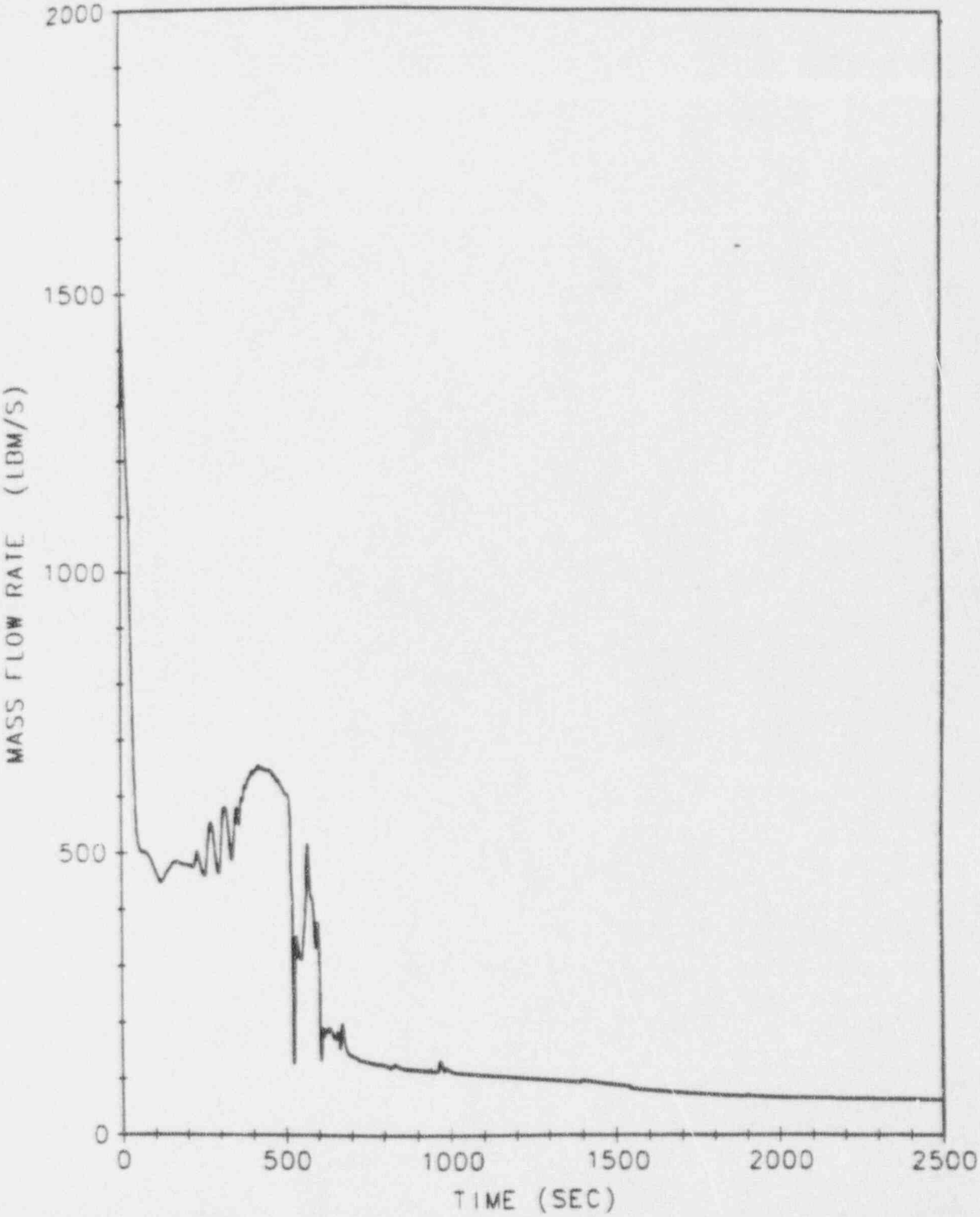
CLAD HOT SPOT FLUID TEMPERATURE (3 INCH BREAK)

FIGURE 3-5F



WESTINGHOUSE ELECTRIC CORPORATION
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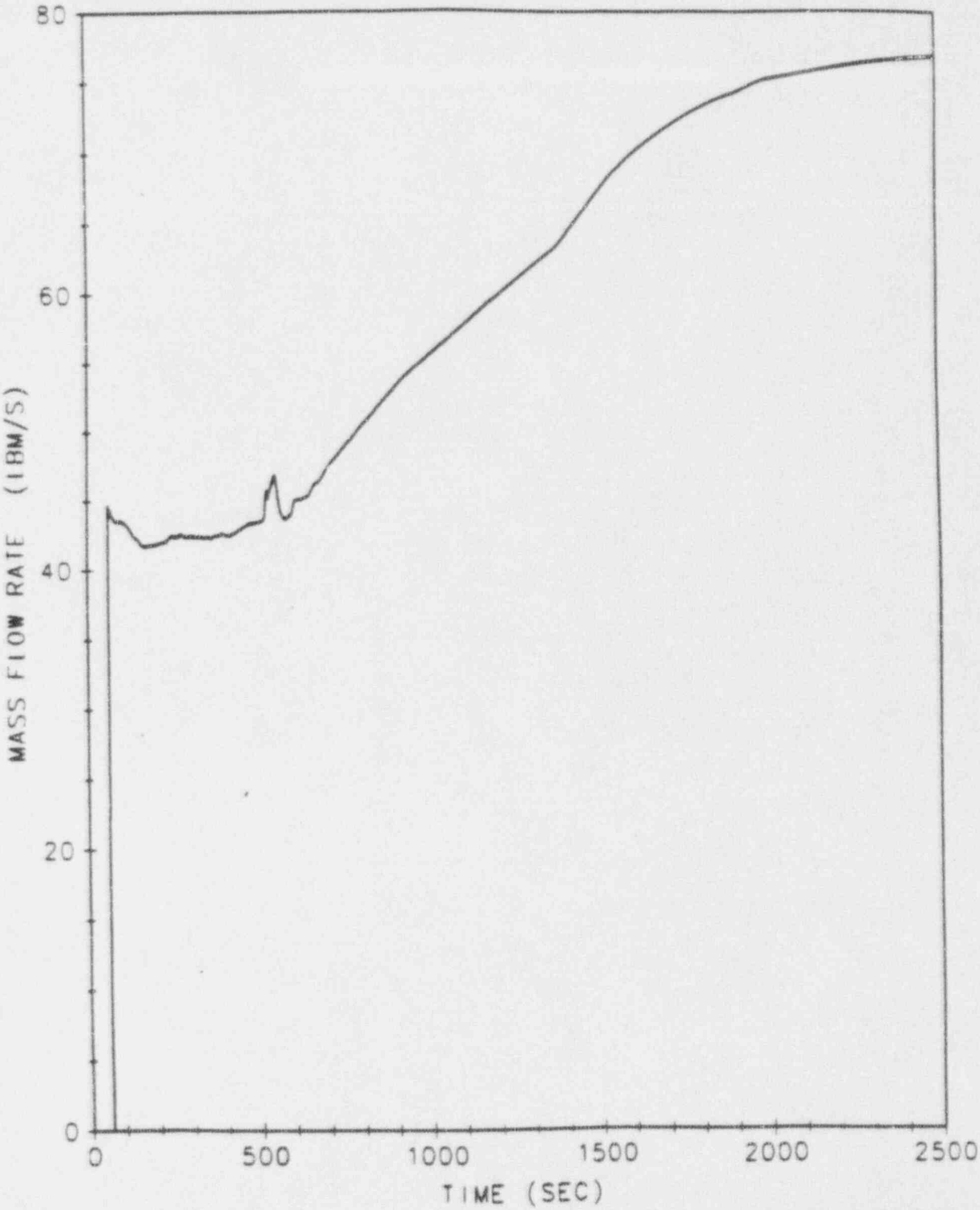
BREAK MASS FLOW RATE
(3 INCH BREAK)

FIGURE 3-5G



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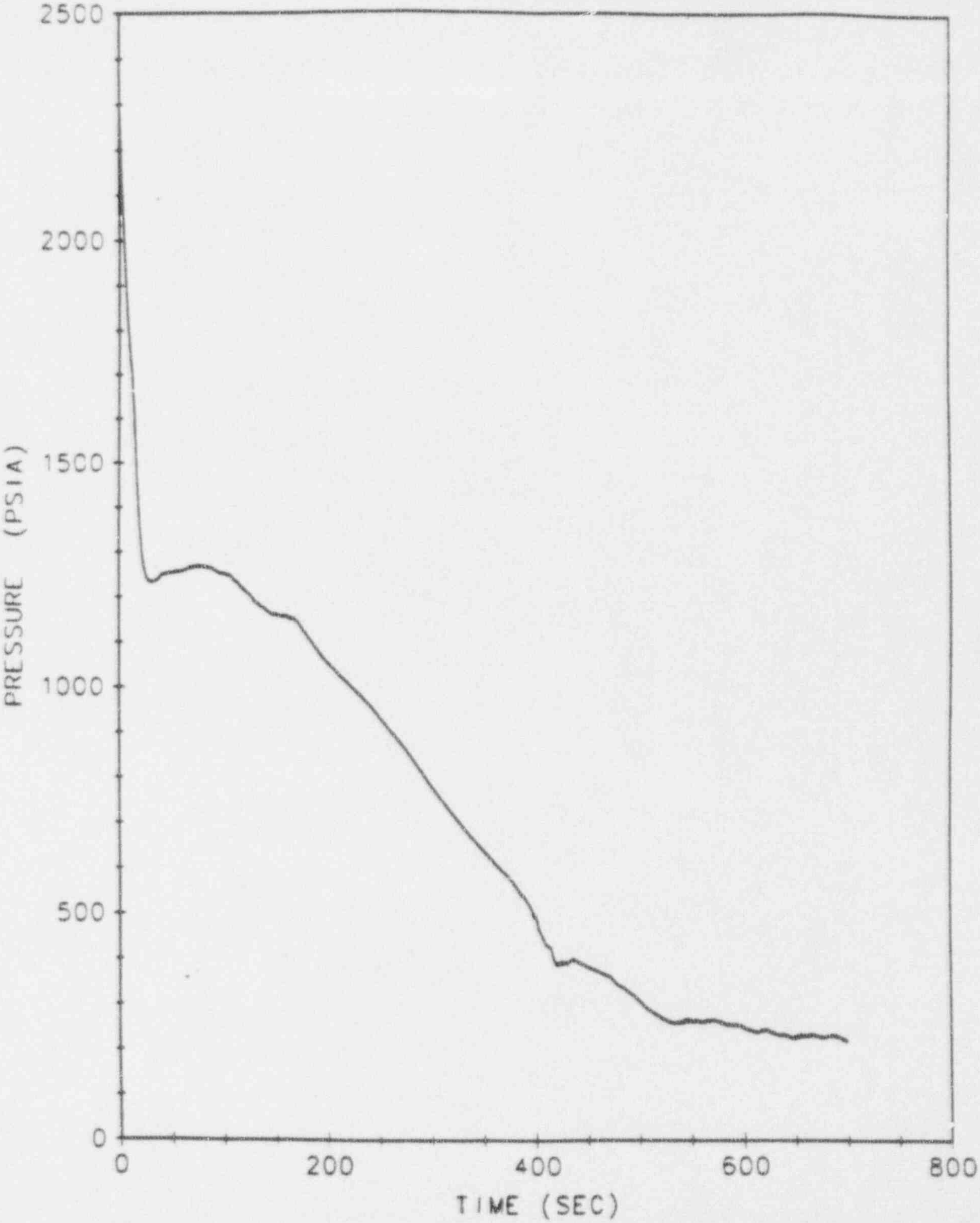
PUMPED SAFETY INJECTION MASS FLOW RATE (3 INCH BREAK)

FIGURE 3-5H



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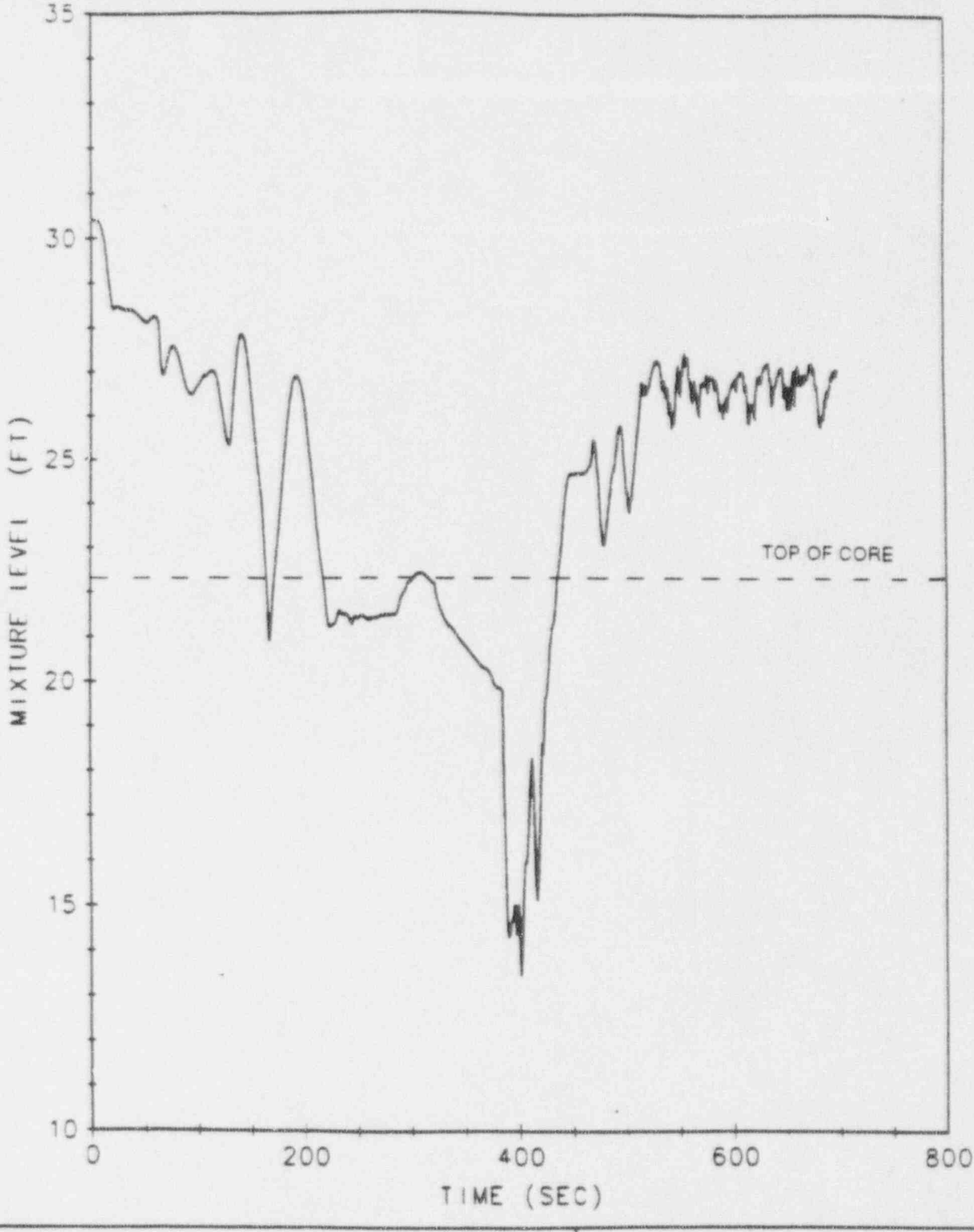
RCS PRESSURE

(6 INCH BREAK)

FIGURE 3-6A



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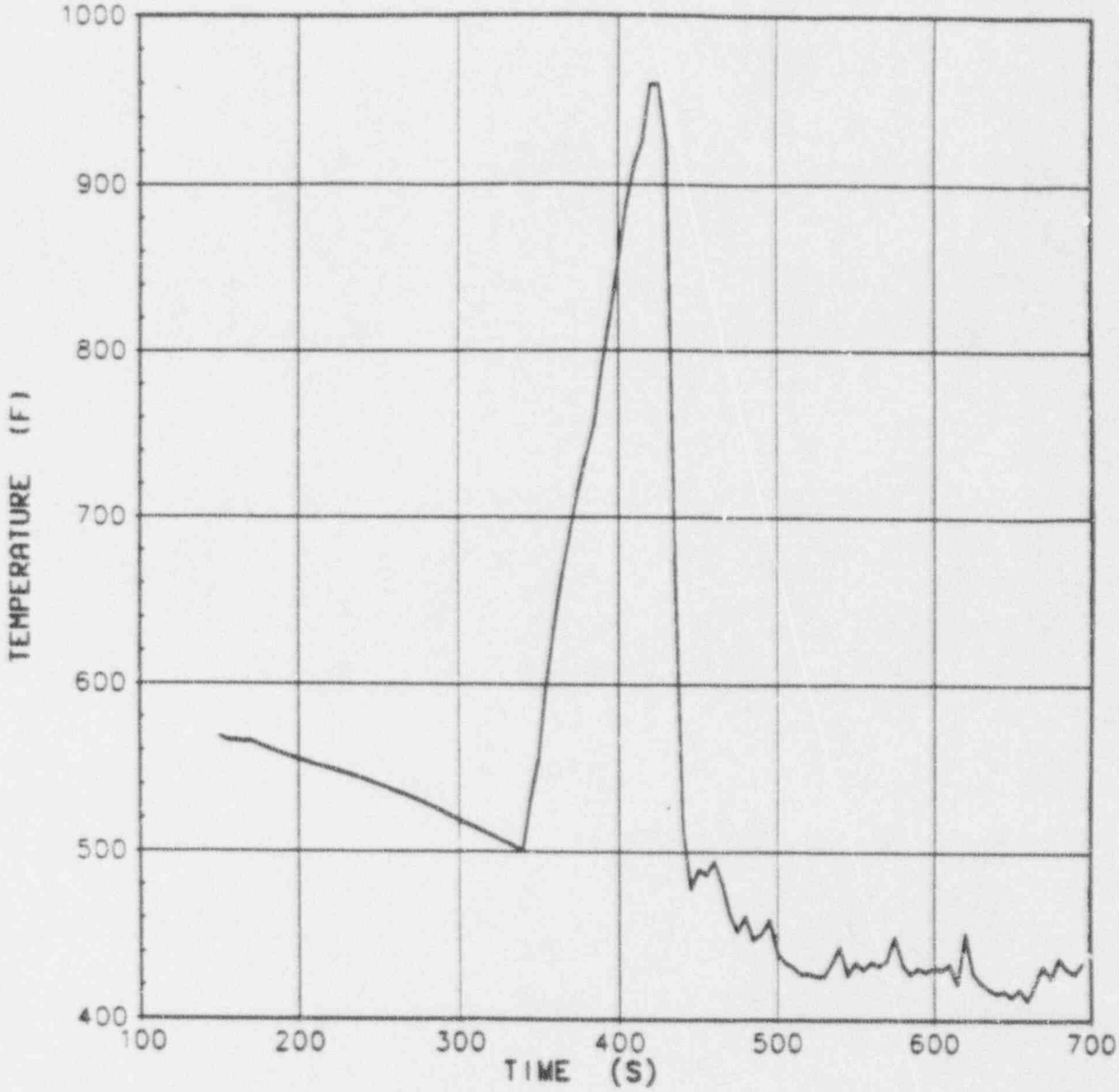
CORE MIXTURE LEVEL

(6 INCH BREAK)

FIGURE 3-6B



WESTINGHOUSE ELECTRIC CORPORATION
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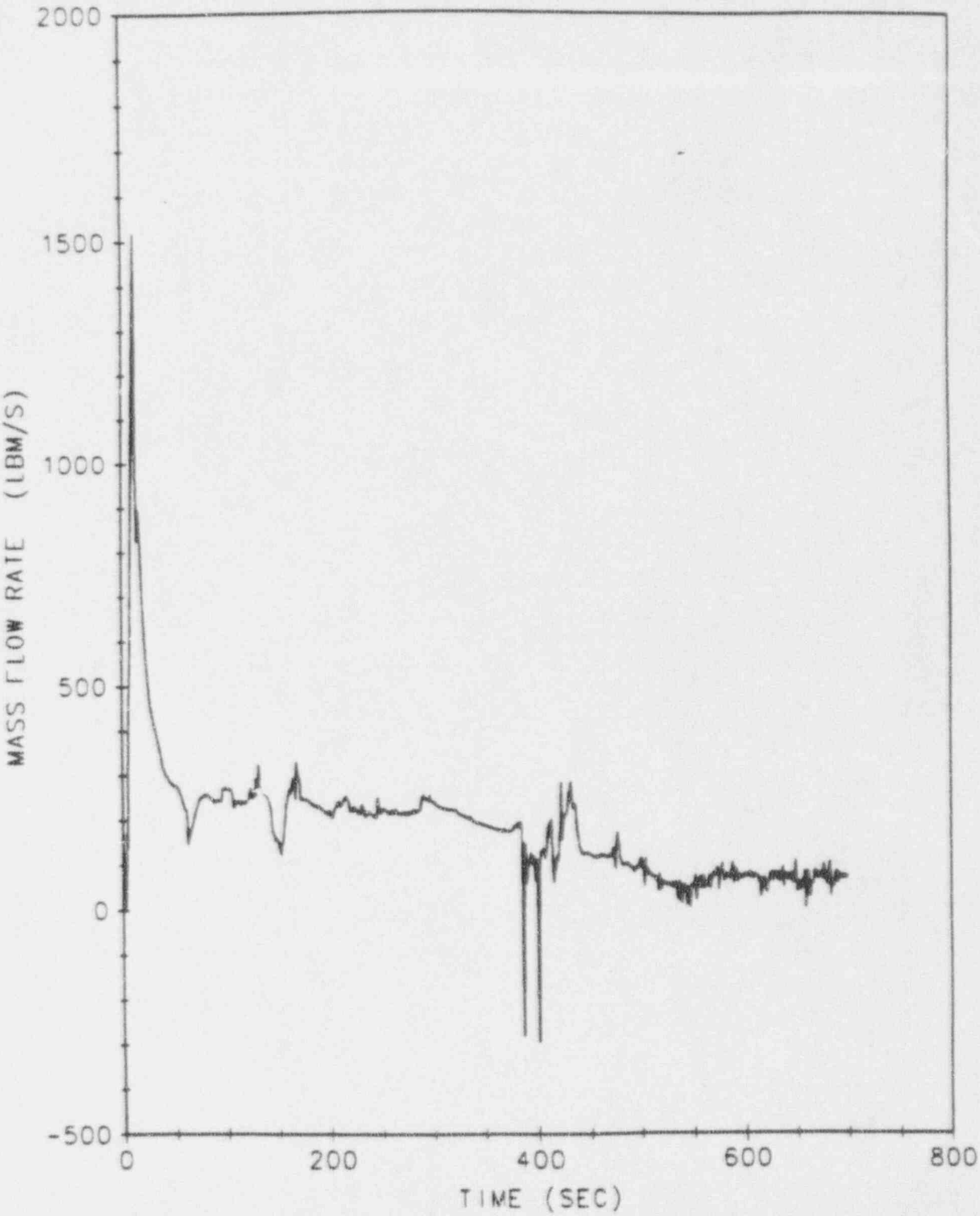
PEAK CLAD TEMPERATURE

(6 INCH BREAK)

FIGURE 3-6C



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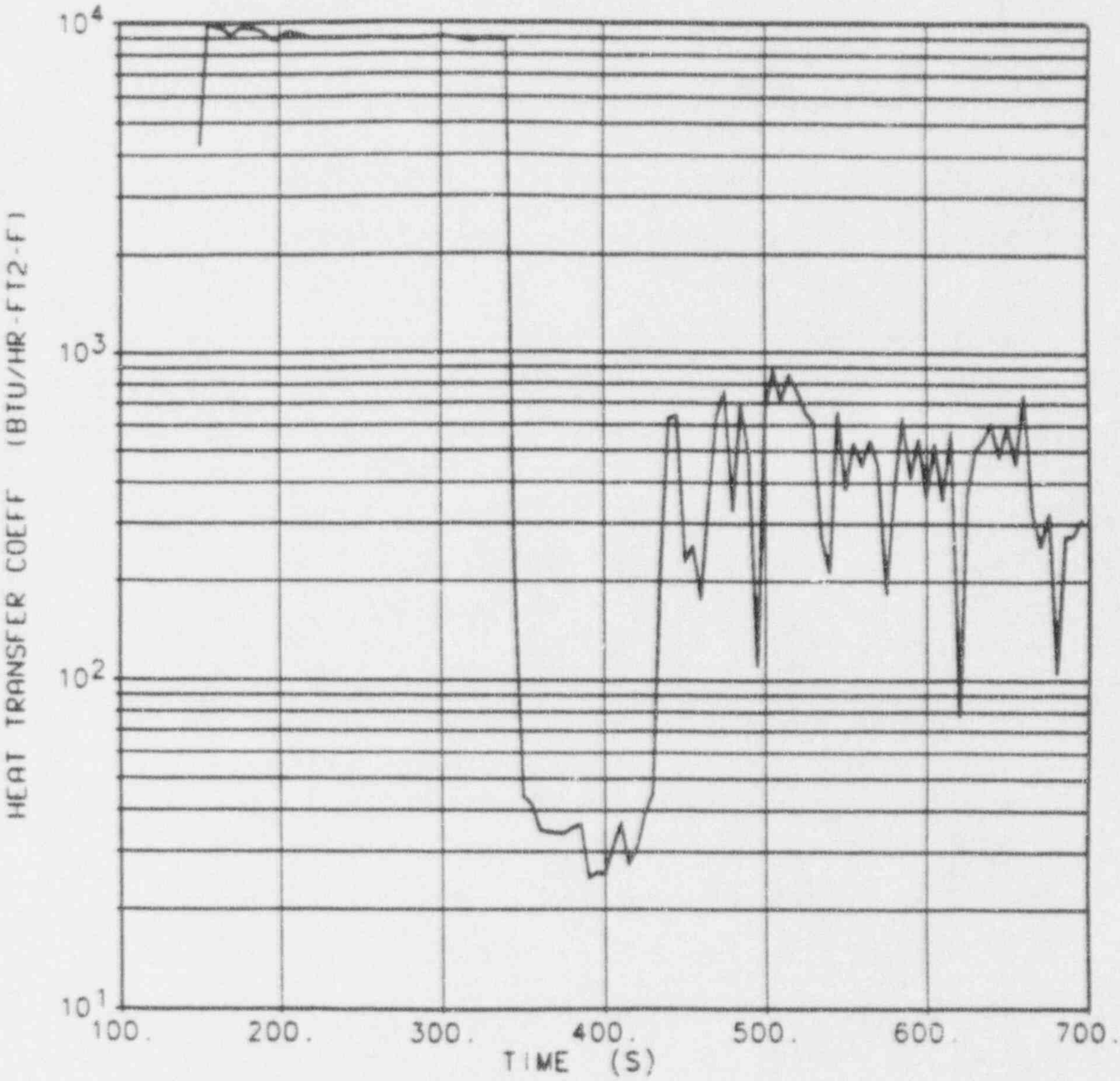
CORE EXIT STEAM MASS FLOW RATE (6 INCH BREAK)

FIGURE 3-6D



WESTINGHOUSE ELECTRIC CORPORATION
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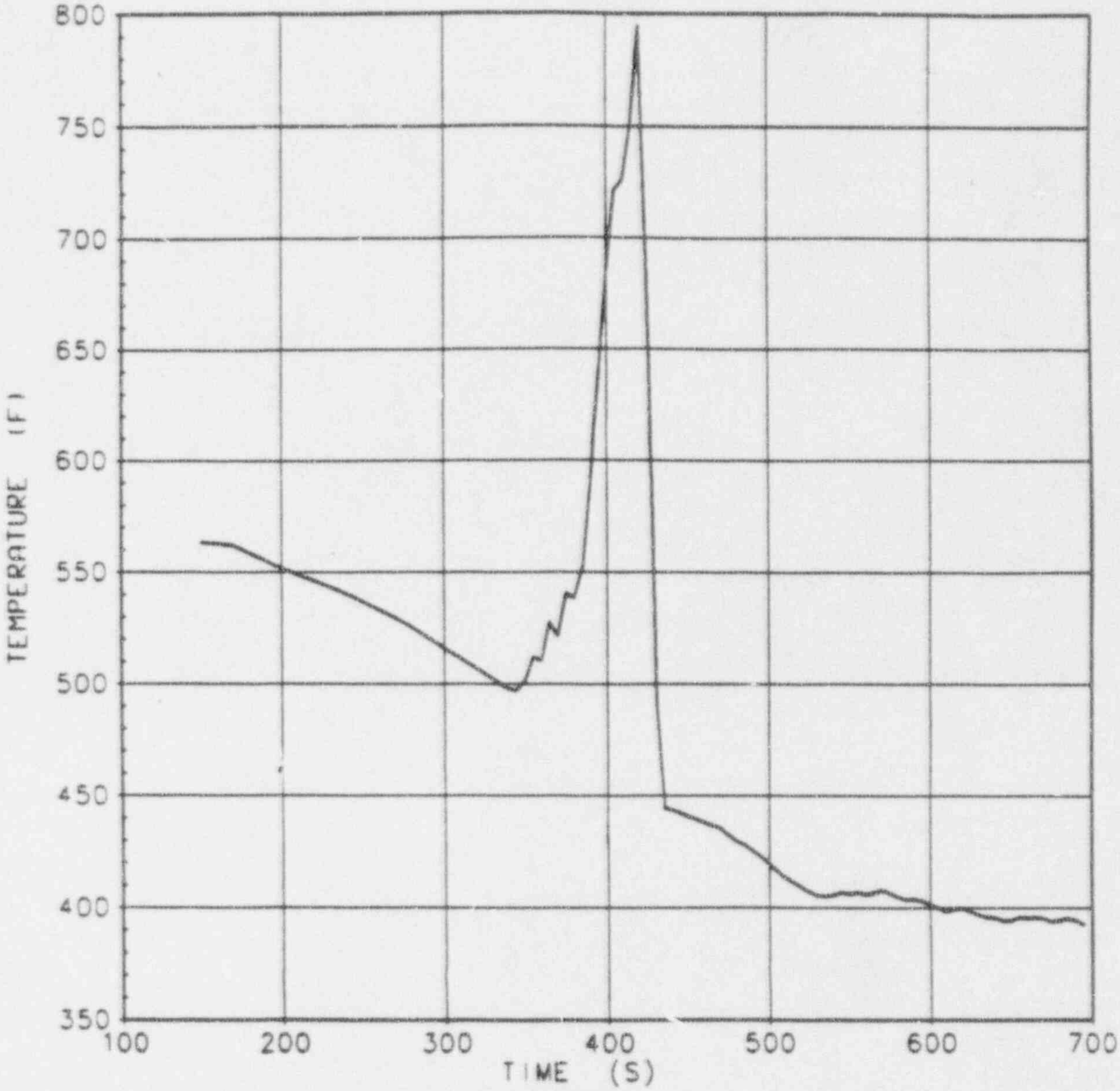
HOT ROD HEAT TRANSFER COEFFICIENT (6 INCH BREAK)

FIGURE 3-6E



WESTINGHOUSE ELECTRIC CORPORATION
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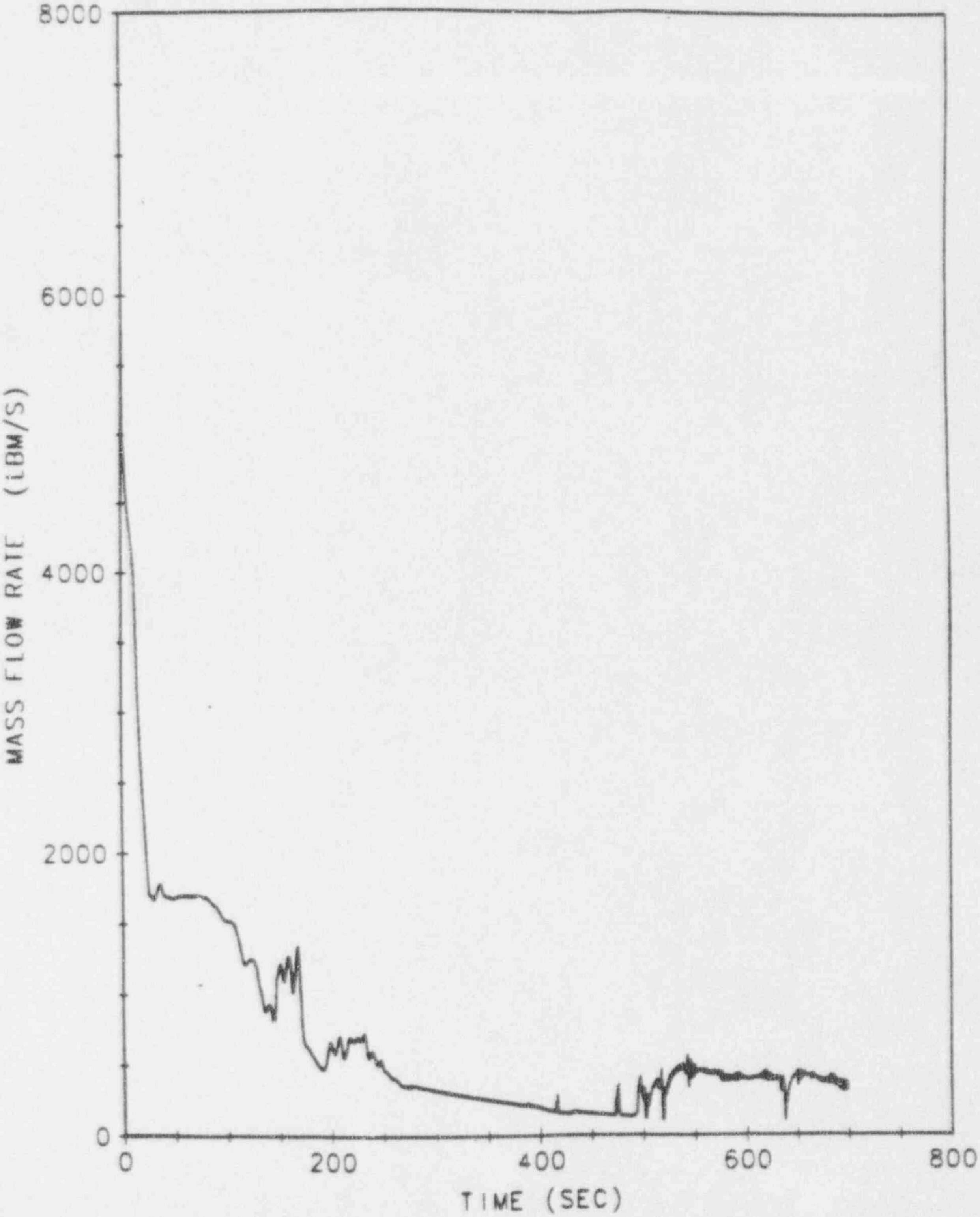
CLAD HOT SPOT FLUID TEMPERATURE (6 INCH BREAK)

FIGURE 3-6F



WESTINGHOUSE ELECTRIC CORPORATION
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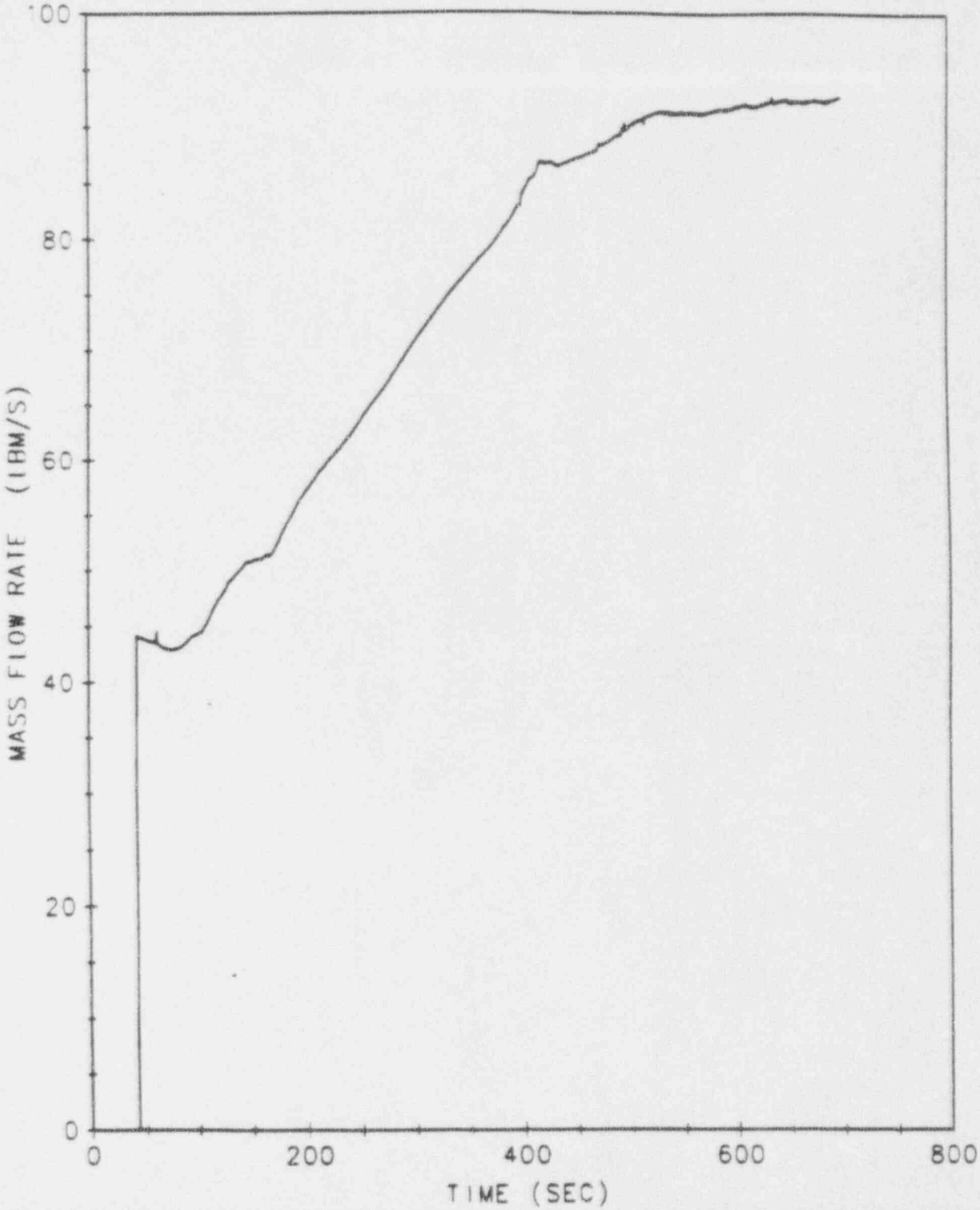
BREAK MASS FLOW RATE
(6 INCH BREAK)

FIGURE 3-6H



WESTINGHOUSE ELECTRIC CORPORATION
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PUMPED SAFETY INJECTION MASS FLOW RATE (6 INCH BREAK)

FIGURE 3-6G

4.0 CONCLUSION

The analyses presented in Sections 2 and 3 show that the Seabrook Station emergency core cooling system provides sufficient core flooding to meet the requirements of 10 CFR 50.46 in the event of a large break or small break loss-of-cooling accident.

5.0 REFERENCES

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Proposed Core Operating Limits Report

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report for Seabrook Station Unit 1, Cycle 3 has been prepared in accordance with the requirements of Technical Specification 6.3.1.6.

The Technical Specifications affected by this report are:

1)	2.2.1	Limiting Safety System Settings
2)	3.1.1.1	Shutdown Margin limit for MODES 1, 2, 3, 4
3)	3.1.1.2	Shutdown Margin limit for MODE 5
4)	3.1.1.3	Moderator Temperature Coefficient
5)	3.1.3.5	Shutdown Rod Insertion Limit
6)	3.1.3.6	Control Rod Insertion Limits
7)	3.2.1	Axial Flux Difference
8)	3.2.2	Heat Flux Hot Channel Factor
9)	3.2.3	Nuclear Enthalpy Rise Hot Channel Factor

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in section 1.0 are presented in the following subsections. These limits have been developed using the NRC-approved methodologies specified in Technical Specification 6.8.1.6.

2.1 Limiting Safety System Settings: (Specification 2.2.1)

2.1.1 Cycle Dependent Overtemperature ΔT Trip Setpoint Parameters and Function Modifier:

2.1.1.1 $K_1 = 1.145$

2.1.1.2 $K_2 = 0.020/°F$

2.1.1.3 $K_3 = 0.001/psig$

2.1.1.4 Channel Total Allowance (TA) = N.A.

2.1.1.5 Channel Z = N.A.

2.1.1.6 Channel Sensor Error (S) = N.A.

2.1.1.7 Allowable Value - The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.2% of ΔT span.

2.1.1.8 $F_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers with gains to be selected based on measured instrument response during plant startup test. $F_1(\Delta I)$ is specified in Figure 1.1.

2.1.2 Cycle Dependent Overpower ΔT Trip Setpoint Parameters and Function Modifier:

2.1.2.1 K_4 = 1.080 (1.085 with modified $F_2(\Delta I)$ NCH PC card)

2.1.2.2 K_5 = 0.020/°F for increasing average temperature and K_5 = 0.0 for decreasing average temperature.

2.1.2.3 K_6 = -0.00196/°F for T > T", and K_6 = 0.0 for T ≤ T", where:
T = Average temperature (°F), and
T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 588.5°F).

2.1.2.4 Channel Total Allowance (TA) = N.A.

2.1.2.5 Channel Z = N.A.

2.1.2.6 Channel Sensor Error (S) = N.A.

2.1.2.7 Allowable Value - The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.1%* of ΔT span.

* (2.0% with modified $F_2(\Delta I)$ NCH PC card)

2.1.2.8 $F_2(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers with gains to be selected based on measured instrument

response during plant startup tests. $F_2(\Delta I)$ is specified in Figure 1.2.

2.2 Shutdown Margin Limit For MODES 1, 2, 3, AND 4: (Specification 3.1.1.1)

A) The Shutdown Margin shall be greater than or equal to 1.3% ΔK/K in MODES 1, 2, and 3.

B) The Shutdown Margin shall be greater than or equal to 1.8% ΔK/K in MODE 4.

2.3 Shutdown Margin Limit For MODE 5: (Specification 3.1.1.2)

The Shutdown Margin shall be greater than or equal to 1.8% ΔK/K.

2.4 Moderator Temperature Coefficient: (Specification 3.1.1.3)

2.4.1 The Moderator Temperature Coefficient (MTC) shall be less positive than $+0.5 \times 10^{-4}$ ΔK/K/°F for all the rods withdrawn, Beginning of Cycle Life (BOL), for power levels up to 70% RATED THERMAL POWER with a linear ramp to 0 ΔK/K/°F at 100% RATED THERMAL POWER.

2.4.2 MTC shall be less negative than -4.2×10^{-4} ΔK/K/°F for End of Cycle Life (EOL), ARO, Rated Thermal Power conditions.

2.4.3 The 300 ppm ARO, Rated Thermal Power MTC shall be less negative than -3.3×10^{-4} ΔK/K/°F (300 ppm Surveillance Limit).

2.5 Shutdown Rod Insertion Limit: (Specification 3.1.3.5)

2.5.1 The shutdown rods shall be fully withdrawn. The fully withdrawn position is defined as the interval within 225 steps withdrawn to the mechanical fully withdrawn position inclusive.

2.6 Control Rod Insertion Limits: (Specification 3.1.3.6)

2.6.1 The control rod banks shall be limited in physical insertion as specified in Figure 1.

2.7 Axial Flux Difference: (Specification 3.2.1)

2.7.1 For operation with the Fixed Incore Detector Alarm OPERABLE, the indicated AFD must be within the Acceptable Operation Limits specified in Figure 2.1.

2.7.2 For operation with the Fixed Incore Detector Alarm inoperable, the indicated AFD must be within the Acceptable Operation Limits specified in Figure 2.2.

2.8 Heat Flux Hot Channel Factor: (Specification 3.2.2)

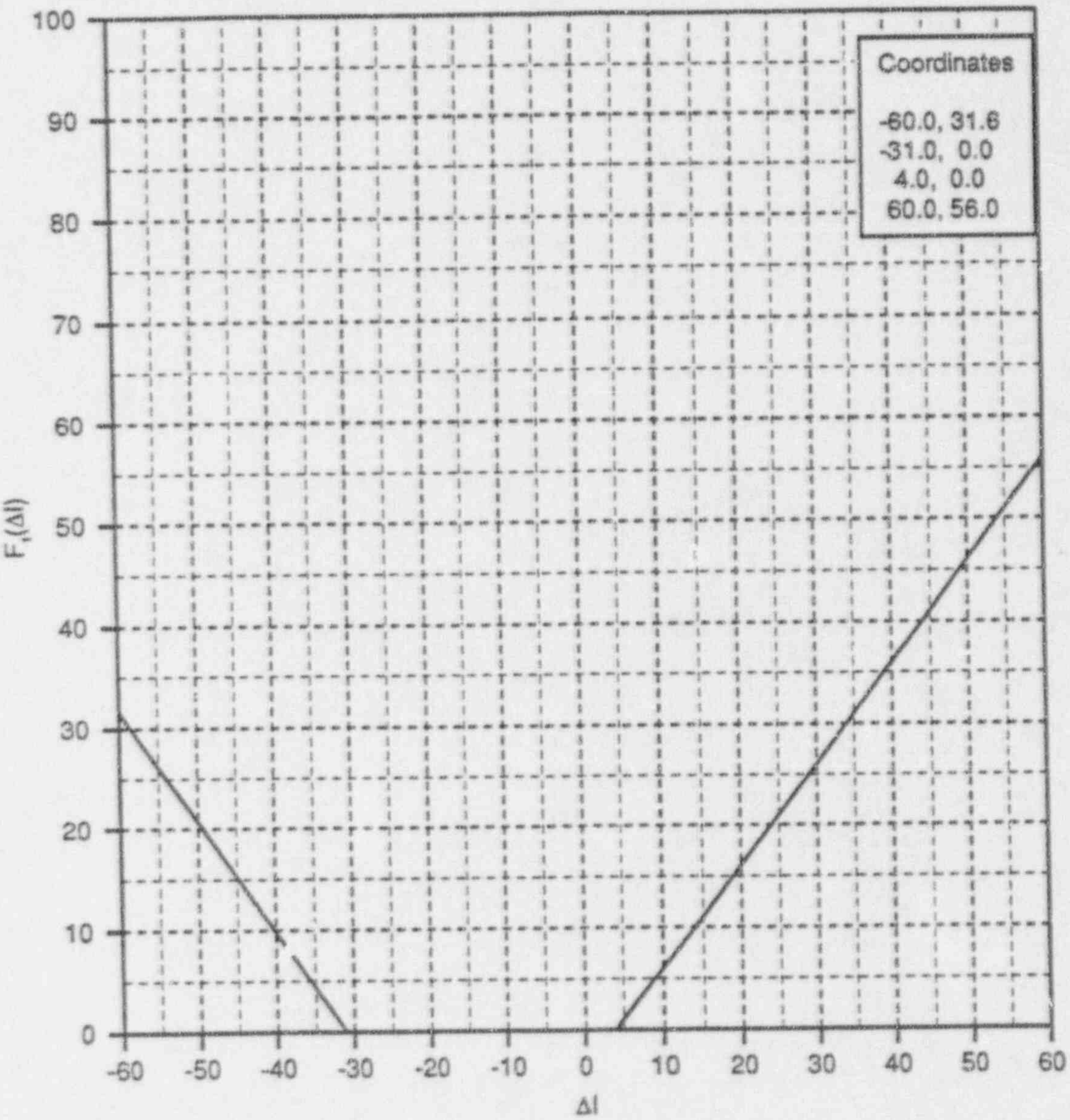
2.8.1 $F_Q^{RTP} = 2.50$

2.8.2 For operation with the Fixed Incore Detector Alarm OPERABLE, K(Z) is specified in Figure 3.

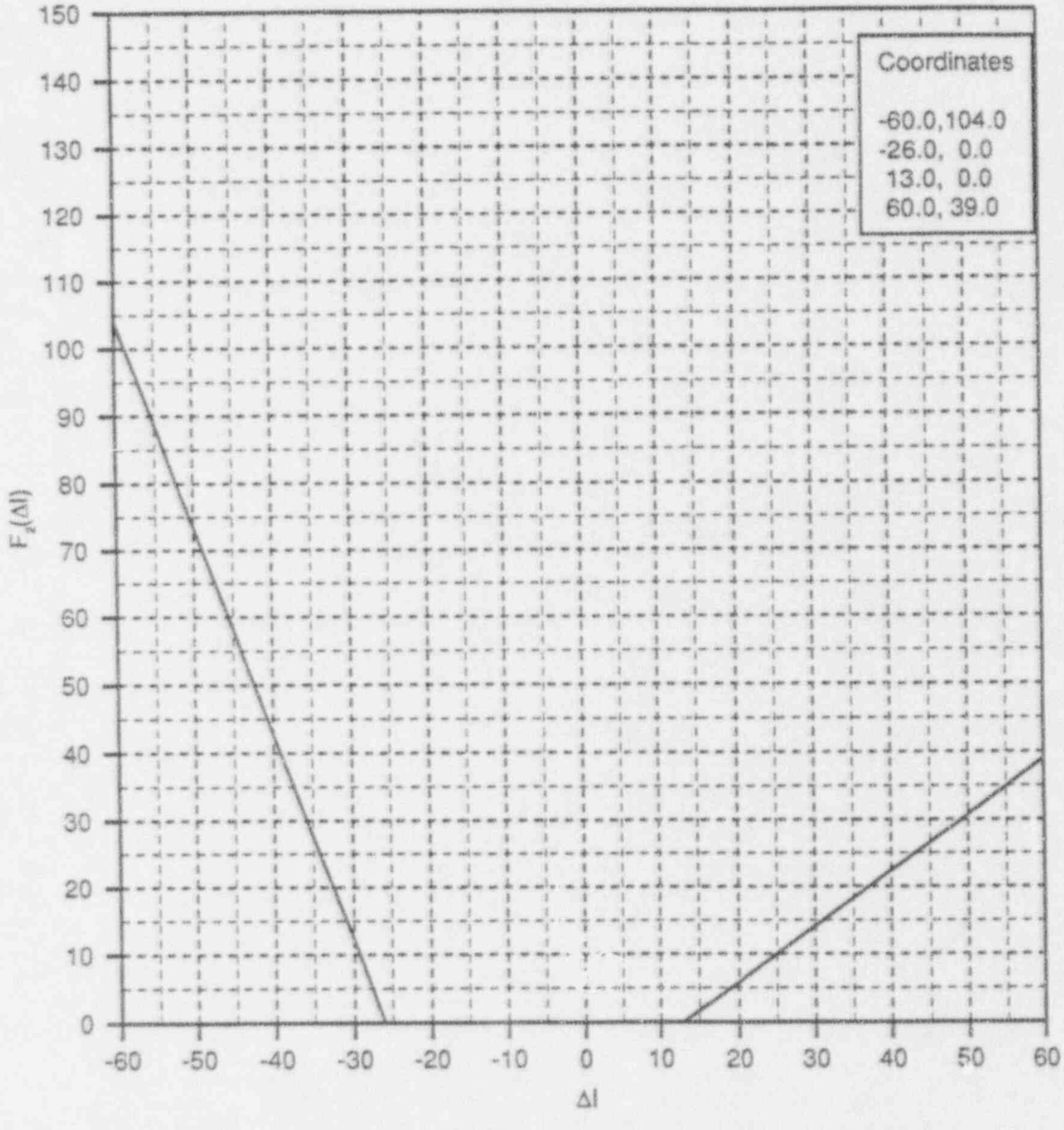
2.8.3 For operation with the Fixed Incore Detector Alarm inoperable, K(Z) is specified in Figure 4.

2.9 Nuclear Enthalpy Rise Hot Channel Factor: (Specification 3.2.3)

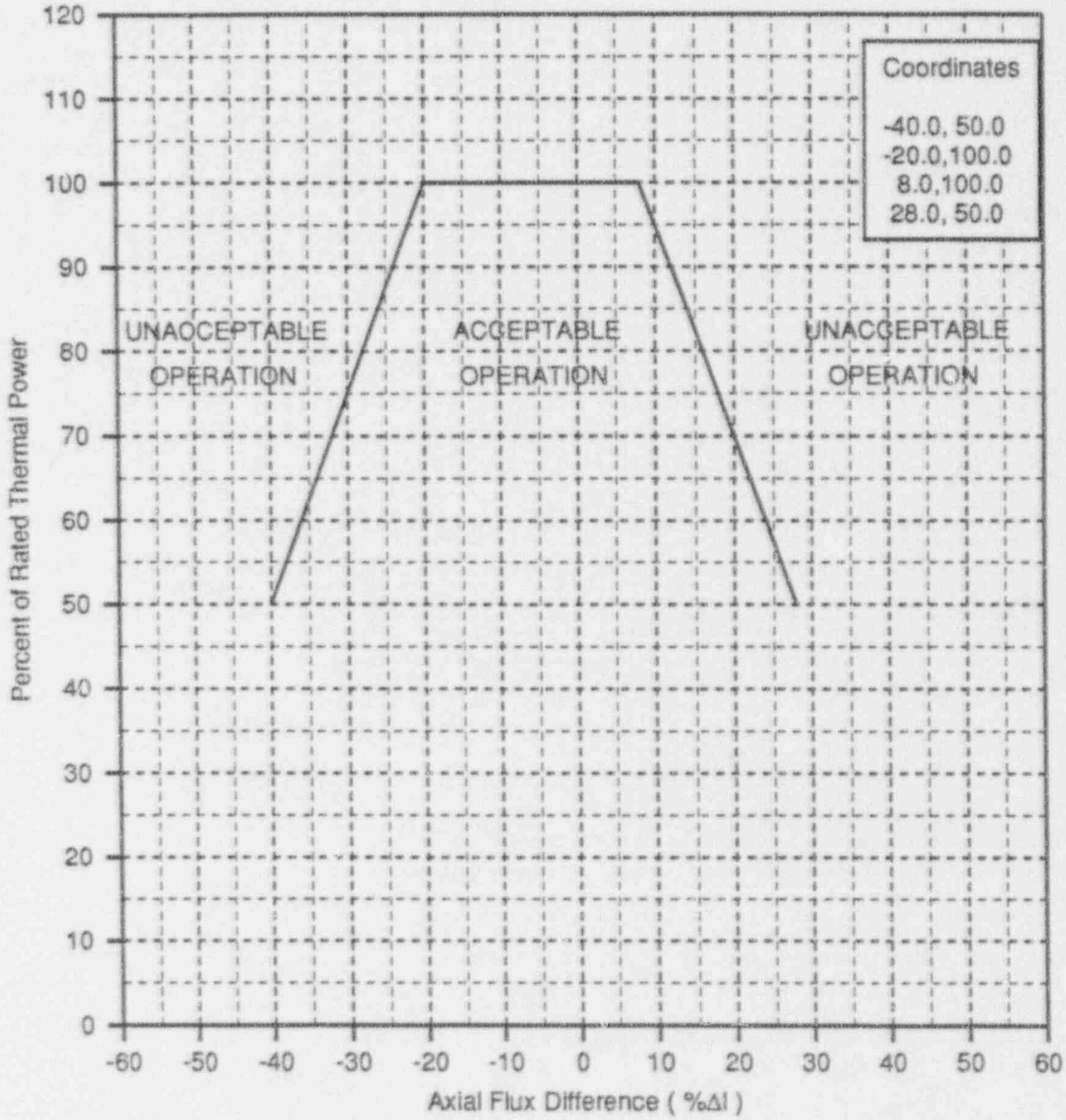
The limits on $F^N_{\Delta H}$ are specified in Figure 5. The limits apply to $F^N_{\Delta H}$ measured using either the fixed or movable incore detectors since a bounding measurement error has been allowed for in determination of the design DNBR limit value.



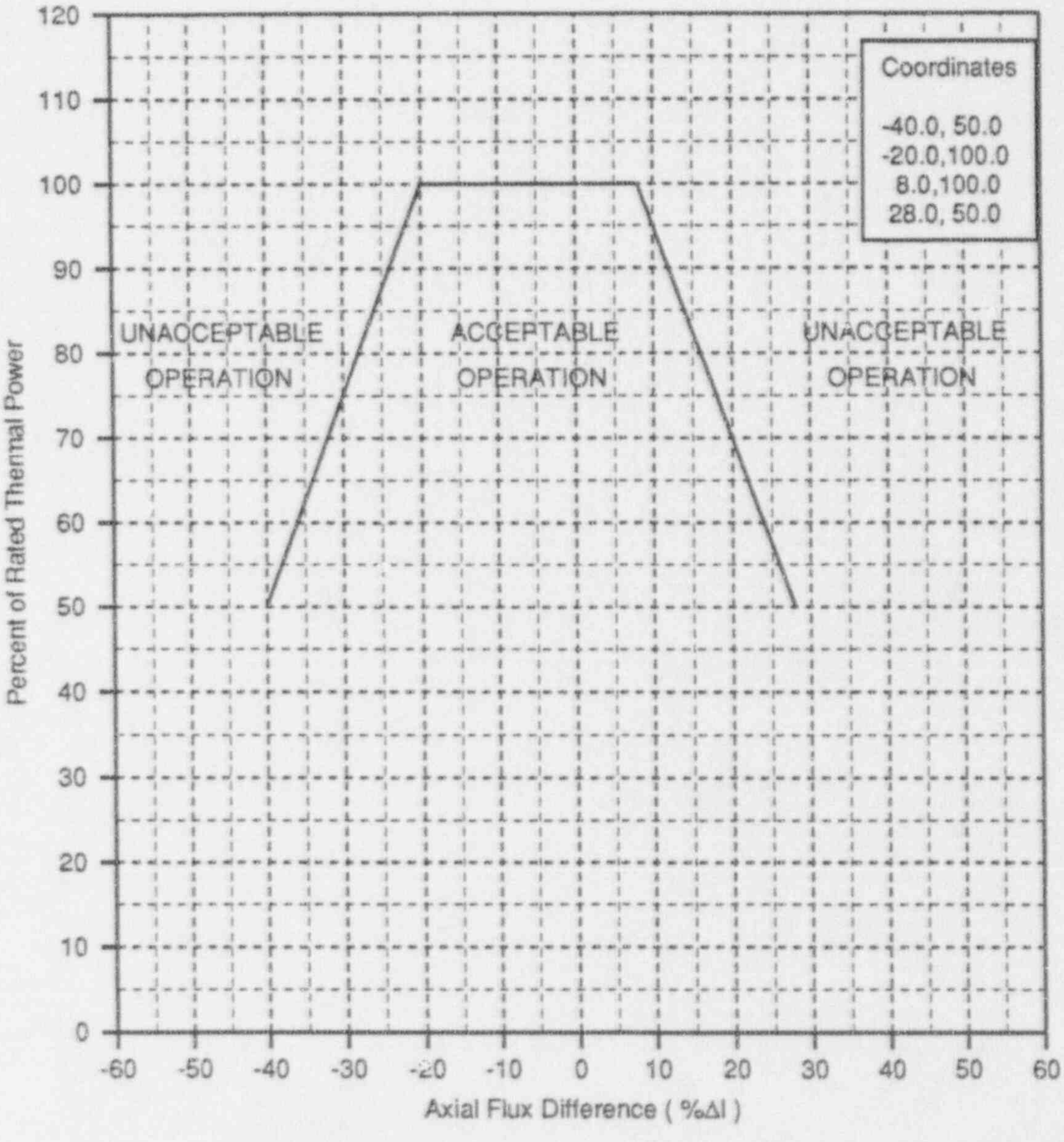
SEABROOK STATION CYCLE 4 CORE OPERATING LIMITS REPORT	Overtemperature ΔT Trip $F_1(\Delta I)$ Axial Flux Imbalance Penalty Function FIGURE 1.1



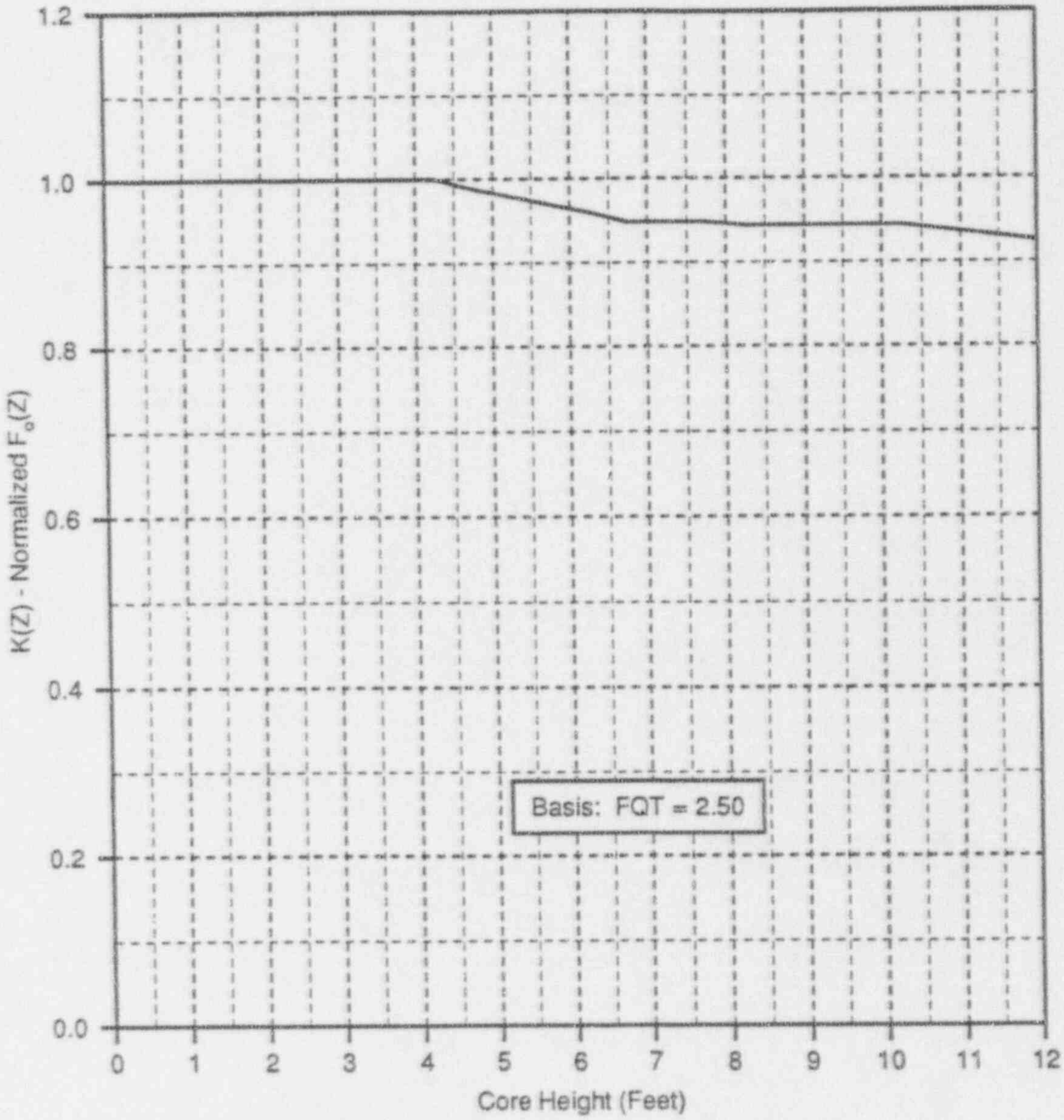
SEABROOK STATION CYCLE 4 CORE OPERATING LIMITS REPORT	Overpower ΔT Trip $F_2(\Delta I)$ Axial Flux Imbalance Penalty Function FIGURE 1.2



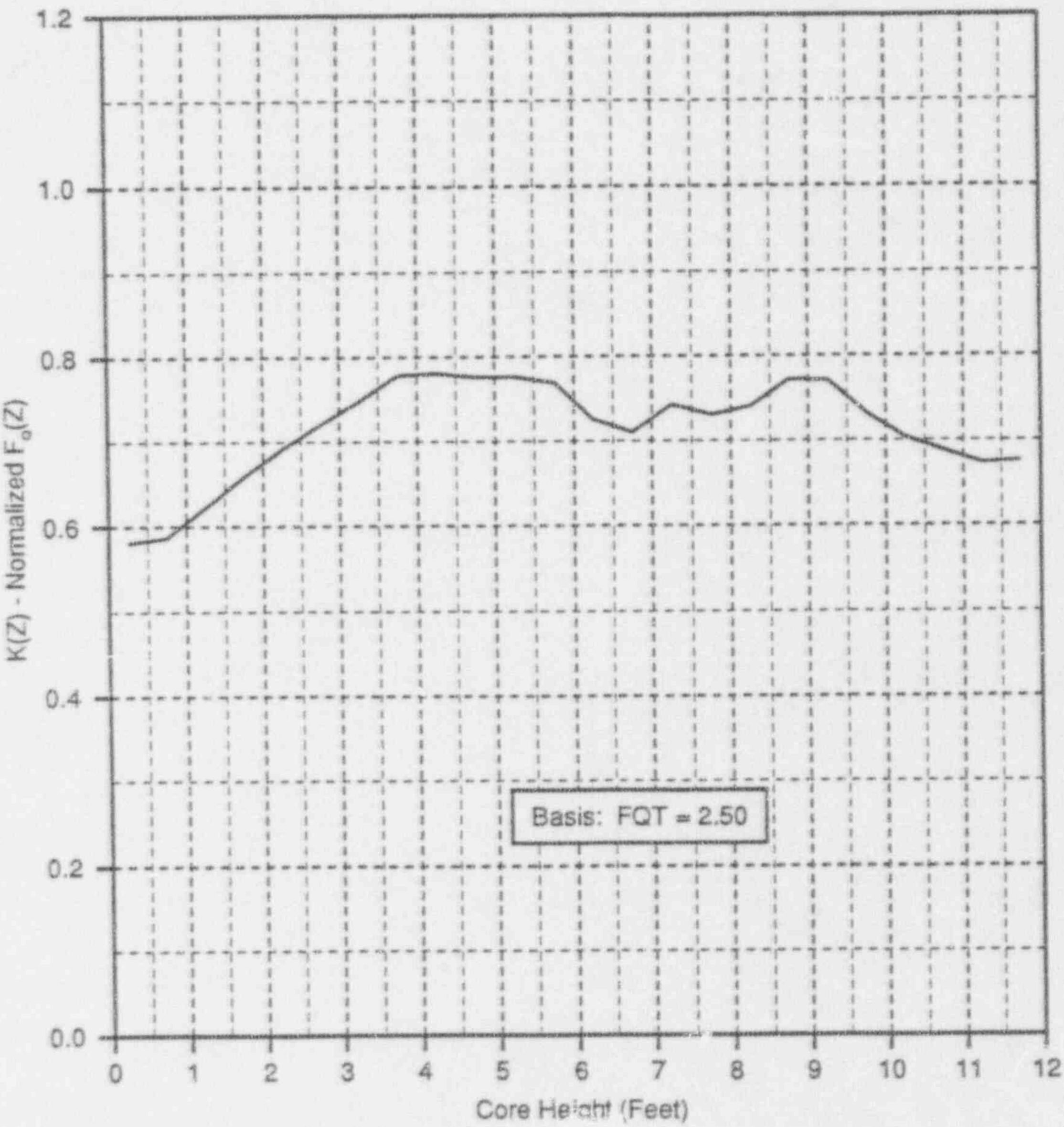
SEABROOK STATION CYCLE 4 CORE OPERATING LIMITS REPORT	Axial Flux Difference Limits as a Function of Rated Thermal Power for Operation With Fixed Incore Detector System Alarm OPERABLE FIGURE 2.1



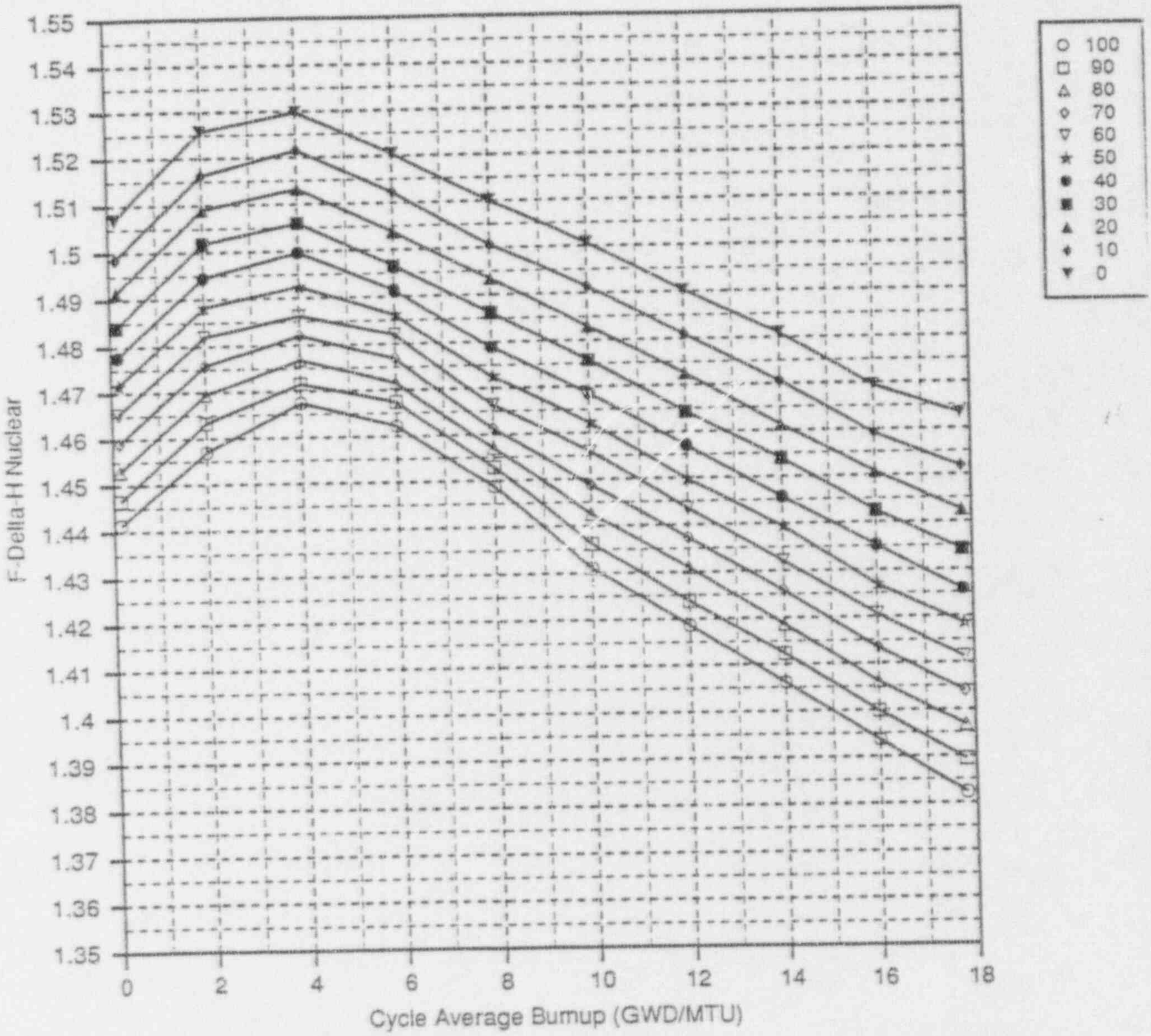
SEABROOK STATION CYCLE 4 CORE OPERATING LIMITS REPORT	Axial Flux Difference Limits as a Function of Rated Thermal Power for Operation With Fixed Incore Detector System Alarm Inoperable FIGURE 2.2



SEABROOK STATION CYCLE 4 CORE OPERATING LIMITS REPORT	K(Z) - Normalized $F_Q(Z)$ As A Function of Core Height FIGURE 3



SEABROOK STATION CYCLE 4 CORE OPERATING LIMITS REPORT	K(Z) - Normalized $F_Q(Z)$ As A Function of Core Height With Fixed Incore Detector Alarm Inoperable FIGURE 4



SEABROOK STATION CYCLE 4 CORE OPERATING LIMITS REPORT	All-Rods-Out Nuclear Enthalpy Rise Hot Channel Factor Versus Power Level FIGURE 5

10CFR50.46 LOCA Model Assessments on the PCT Margin Utilization

10CFR50.46 LOCA MODEL ASSESSMENTS ON THE PCT MARGIN UTILIZATION

The new small break Loss of Coolant Accident (LOCA) analyses for Seabrook Station utilize the NOTRUMP Emergency Core Cooling System (ECCS) analysis model (see proposed Technical Specification 6.8.1.6.b2). Westinghouse Electric Corporation (Westinghouse) has identified the following non-conservatisms/conservatisms associated with the NOTRUMP model:

1. The Seabrook Station LOCA analysis assumes no Safety Injection (SI) into the broken loop. Sensitivity calculations indicate that a Peak Clad Temperature (PCT) penalty of 150°F is typical to address this affect. To offset this affect, a more realistic condensation model based on COSI test results has been evaluated by Westinghouse. The COSI test facility is a 1/100 scale representation of the cold leg SI injection ports in a Westinghouse designed PWR. This model demonstrates that analysis performed with spillage to the broken loop using the original condensation model is conservative compared to the improved condensation model with injection into the broken loop.

2. The Seabrook Station LOCA analysis was performed with an error in the NOTRUMP drift flux floe regime map. The net impact of this error is estimated to decrease the calculated PCT in the range of 13°F to 55°F.

Thus, the reference small break LOCA PCTs will be modified slightly to account for the above items. The modified PCTs will be reported to the NRC pursuant to the requirements of 10CFR50.46.