ENCLOSURE

(Proposed Changes to FSAR Chapter 15)

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15.3 CONDITION III - INFREQUENT FAULTS

By definition Condition III occurrences are faults which may occur very infrequently during the life of the plant. 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 the operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. A Condition Ill 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 the following faults have been grouped into this category:

1. Loss of reactor coolant, from small ruptured pipes or from cracks in large pipes, which actuates the ECCS.

2. Minor secondary system pipe breaks.

3. Inadvertent loading of a fuel assembly into an improper position.

4. Complete loss of forced reactor coolant flow.

5. Waste gas decay tank rupture.

6. Single rod cluster control assembly withdrawal at full power.

15.3.1 Loss of Reactor Coolant From Small Ruptured Pipes or From Cracks in Large Pipes Which Actuate the Emergency Core Cooling System

15.3.1.1 Identification of Causes and Accident Description

A LOCA is defined as the loss of reactor coolant at a rate in excess of the reactor coolant normal makeup rate from breaks or openings in the RCPB inside primary containment up to, and including, a break equivalent in size to the largest justified pipe rupture (or in the absence of justification, a double-ended rupture of the largest pipe) in the RCPB (ANSI/ANS-51.1-1983). See Section 3.6 for a more detailed description of the loss of reactor coolant accident boundary limits. Ruptures of small cross section will cause expulsion of the coolant at a rate which can be accommodated by the charging pumps 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 the containment contains the fission products existing ~~in it~~.

reactor coolant pressure boundary

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the RCS through the postulated break against the charging pump makeup flow at normal RCS pressure, i.e., 2250 psia. A makeup flow rate from one centrifugal charging pump is typically adequate to sustain pressurizer level at 2250 psia for a break through a 0.375 inch diameter hole. This break results in a loss of approximately 17.5 lb/sec.

Should a larger break occur, depressurization of the RCS causes fluid to flow to the RCS from the pressurizer, resulting in a pressure and level decrease in the pressurizer. A reactor trip occurs when the pressurizer low pressure trip setpoint is reached. The safety injection system is actuated when the appropriate setpoint is reached. The consequences of the accident are limited in two ways: pressure

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

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

Before the break occurs, the plant is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from decay, hot internals and the vessel continues to be transferred to the reactor coolant. The heat transfer between the RCS and the secondary system may be in either direction, depending on the relative temperatures. In the case of continued heat addition to the secondary system, pressure increases, and steam dump may occur. Makeup to the secondary side is automatically provided by the auxiliary feedwater pumps. The reactor trip signal coincident with low T_{avg} signal (with assumed coincident loss of offsite power), stops normal feedwater flow by closing the main feedwater isolation valves and flow control valves. The secondary flow aids in the reduction of RCS pressure.

When the RCS depressurizes to the cold leg accumulator tank pressure, the accumulators begin to inject water into the reactor coolant loops. The reactor coolant pumps are assumed to be tripped concurrent with the reactor trip, and effects of pump coastdown are

BACKGROUND PAGE

included in the blowdown analyses.

15.3.1.2 Analysis of Effects and Consequences

Method of Analysis

For breaks less than 1.0 ft^2, the NOTRUMP(2) digital computer code is employed to calculate the transient depressurization of the RCS as well as to describe the mass and enthalpy of flow through the break.

Small Break LOCA Analysis Using NOTRUMP

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

In NOTRUMP, the RCS is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicitly with the intact loops lumped into a second loop. The transient behavior of the system is determined from the governing conservation equation of mass, energy, and momentum applied throughout the system. A detailed description of NOTRUMP is given in References [1] and [2].

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

Cladding thermal analyses are performed with the LOCTA-IV [3] code which uses the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height history from the NOTRUMP hydraulic calculations as input.

A schematic representation of the computer code interfaces is given in Figure 15.3-1.

The small break analysis was performed with the approved Westinghouse ECCS Small Break Evaluation Model [1],[2], and [3].

Safety injection flow rate to the RCS as a function of system pressure is an input parameter. The SIS is assumed to begin delivering full flow to the RCS 30 seconds after the generation of a safety injection signal.

~~For these analyses, the SI delivery considers pumped injection flow which is depicted in Figure 15.3-2 as a function of RCS pressure. This figure represents injection flow from the SI pumps based on performance curves degraded 5 percent from the design head. The 30 second delay includes time required for diesel startup and loading of the safety injection pumps onto the emergency buses. The effect of RHR pump flow is not a factor in these analyses since their shutoff head is lower than RCS pressure during~~

~~the time portion of the transient considered here.~~ Also, minimum Safeguards ECCS capability and operability has been assumed in these analyses including conservative assumptions with regard to spillage of ECCS water from broken lines.

Hydraulic transient analyses are performed with the NOTRUMP code which determines the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core and mixture height history. The core thermal transient is performed with the LOCTA-IV (3) code. Both calculations assume the core is operating at 102% of licensed power.

15.3.1.3 Reactor Coolant System Pipe Break Results

INSERT 1

~~This section presents results of the limiting break size in terms of highest peak clad temperature. The worst break size (small break) is a 4 inch diameter break. The depressurization transient for this break is shown in Figure 15.3-3. The extent to which the core is uncovered is shown in Figure 15.3-4.~~

For all cases reported, during ~~During~~ the earlier part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as they are coasting down following reactor trip. Therefore, upward flow through the core is maintained.

The resultant heat transfer cools the fuel rod ~~and clad~~ cladding to very near the coolant temperatures as long as the core remains covered by a two phase mixture.

P

~~The maximum hot spot clad temperature calculated during the transient is 1549°F, based on a downflow barrel/baffle region, and 1446.1°F, based on an upflow barrel/baffle region. The peak clad temperature transient is shown in Figure 15.3-5 for the worst break size, i.e., the break with the highest peak clad temperature. The steam flow rate for the worst break is shown on Figure 15.3-6.~~ When the mixture level drops below the top of the core, the steam flow computed with NOTRUMP provides cooling to the upper portion of the core. ~~The rod film coefficients for this phase of the transient are given in Figure 15.3-7. The hot spot fluid temperature for the worst break is shown in Figure 15.3-8a.~~

INSERT 1

Analyses were conducted to determine the limiting break size in terms of the highest peak cladding temperature. This section contains results for a downflow barrel/baffle reactor vessel internals design (initial configuration) and for an upflow barrel/baffle design (the present configuration) in order to capture the spectrum of break sizes analyzed as well as address reduced auxiliary feedwater and safety injection flows for the present configuration.

The typical core power (dimensionless) transient following the accident (relative) to reactor scram time is shown in Figure 15.3-9. Also shown is the typical hot rod axial power shape in Figure 15.3-10.

The reactor scram time is equal to the reactor trip signal time plus 4.4 seconds for rod insertion. During this rod insertion period, the reactor is conservatively assumed to continue to operate at the initial rated power level.

~~15.3.1.3 Additional Break Sizes~~ INSERT 2

A spectrum of break sizes ~~in addition to the limiting break~~ ~~was~~ analyzed for the initial reactor vessel internals design. were a downflow barrel/baffle region. These ~~additional~~ breaks were the 3.0, ~~and~~ ~~6~~.0 and 6.0 inch breaks. Figures 15.3-11 and 12 present the RCS pressure transients. Figures 15.3-13a and 13b present the core mixture height transient for the 3.0 and 6.0 inch breaks, respectively. The peak ~~clad~~ cladding temperatures for these cases are less than the peak ~~clad~~ cladding temperature of the 4.0 inch break which was determined to be the limiting break. The peak temperatures for the 3.0 and 6.0 inch break are presented in Figures 15.3-14a and 14b.

An additional break case of 4.0 inches was analyzed ~~at the limiting 4.0 inch break~~ for an upflow barrel/baffle configuration. Results for this break can be found in Table 15.3-1, Table 15.3-2, and Figures 15.3-8b through 15.3-8g.

It should be noted that all small break sizes presented here result in calculated peak clad temperatures ~~much~~ less than those calculated for large breaks (Section 15.4.1).

The depressurization transient for the 4.0 inch break is shown in Figure 15.3-3. The extent to which the core is uncovered is shown in Figure 15.3-4. The peak cladding temperature transient is shown in Figure 15.3-5. The steam flow rate for this break is shown in Figure 15.3-6. The heat transfer coefficients for the rod for this phase of the transient are given in Figure 15.3-7, and the hot spot fluid temperature is shown in Figure 15.3-8a.

INSERT 3

INSERT 2

For the analysis without reduced auxiliary feedwater and safety injection flows, the SI injection flow is depicted in Figure 15.3-2 as a function of RCS pressure. This figure represents injection flow ~~from~~ based on performance curves degraded 5 percent from the design head. Auxiliary feedwater flow is 1410 gpm based on the design capacity of one motor driven and one turbine auxiliary feedwater pump, each delivering to two steam generators.

The 30 second delay time includes the time for diesel generator startup, loading on the 6.9 KV shudown Board, and sequential loading of the Centrifugal Charging and Safety Injection Pumps onto the emergency buses, with acceleration to full speed and capability for injection. Although included in the 30 second delay, the effect of the RHR pump flow is not a factor in these anlaysis since their shutoff head is lower than RCS pressure during the time period for this transient.

INSERT 3

For the analysis with reduced auxiliary feewater and safety injection flows only the upflow barrel/baffle reactor vessel internals configuration needed to be considered. Two break sizes were analyzed, the 4 inch which had been established in the previous analysis, and a 3 inch which was determined to be limiting in peak cladding temperatures.

For this analysis, the SI injection flow is depicted in Figure 15.3-8n as a function of RCS pressure. A reduced auxiliary feedwater flow of 1050 gpm is utilized, supplied by one motor driven and one turbine auxiliary feedwater pump, each delivering to two steam generators.

Therefore, for the upflow barrel/baffle configuration with the assumed reduction in auxiliary feedwater and safety injection flows the worst break size of 3 inches resulted in a maximum peak hot spot cladding temperature of 2089° F. Results for this break can be found in Table 15.3-1 and Table 15.3-2. Plots of the RCS upper plenum pressure (Figure 15.3-8h); mixture level (Figure 15.3-8i); peak cladding temperature (Figure 15.3-8j); steam flow (Figure 15.3-8k); heat transfer coefficient (Figure 15.3-8l) transients and hot spot fluid temperatures (Figure 15.3-8m) are provided for this case.

15.3.1.4 Conclusions - Thermal Analysis 63

For cases considered, the Emergency Core Cooling System will meet the Acceptance Criteria as presented in 10 CFR 50.46. That is:

1. The calculated peak fuel element ~~clad~~ Cladding temperature provides margin to the requirements of 2200°F, based on an Fq value of 2.40. 63

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of zircalloy in the reactor. 63

3. The ~~clad~~ Cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The ~~clad~~ oxidation limits of 17-percent of the cladding thickness are not exceeded during or after quenching.

4. 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.

The time sequence of events is shown in Table 15.3-1. Table 15.3-2 summarizes the results of these analyses. 63

TABLE 15.3-1

SMALL BREAK LOCA ANALYSIS

TIME SEQUENCE OF EVENTS

EVENT	TIME (s)				
Break size:	3-inch (3)	4-inch (3)	6-inch (3)	4-inch (1)	3-inch (2)
Break initiation	0.0	0.0	0.0	0.0	0.0
Reactor trip signal	8.36	4.72	2.39	4.735	31.5
Safety injection signal	17.97	13.19	9.19	12.915	41.0
Top of core uncovery	1322	665	271	643.0	894
Accumulator injection begins	2836	890	375	847.0	1643
Peak clad temperature occurs	2143	1050	455	935.5	1701
Top of core recovered	2770	1575	477	1440.0	N/A

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Sheet 1 of 1

(1) Upflow barrel/baffle region assumed.

(2) Upflow barrel/baffle region assumed with reduced aux feedwater & SI flows.

(3) Downflow barrel/baffle region assumed.

TABLE 15.3-2

SMALL BREAK ANALYSIS SUMMARY OF RESULTS

Parameter	Value				
Break Size:	3-inch (3)	4-inch (3)	6-inch (3)	4-inch (1)	3-inch (2)
Peak clad temperature (~~f~~) °F	1179	1549	1146	1446.1	2089
Elevation (ft)	12.0	12.0	12.0	12.0	12.0
Max. local Zr/H_2O reaction (%)	0.12	0.74	0.04	0.32	15.5
Elevation (ft)	12.0	12.0	12.0	12.0	12.0
Total Zr/H_2O reaction (%)	<0.3	<0.3	<0.3	<0.3	<1.0
Hot rod burst time (s)	N/A	N/A	N/A	N/A	N/A

Boundary Condition Assumptions

NSSS power	102% of 3411 MW
Core power (Rod heatup analysis)	102% of 3411 MW
Peak linear power	~~12.77~~ 12.84 kW/ft (4)
Cold leg accumulators:	
Water volume (each)	1050 ft^3
Pressure	600 psia

(1) Upflow barrel/baffle region assumed

(2) Upflow barrel/baffle region assumed with reduced aux feedwater & SI flows.

(3) Downflow barrel/baffle region assumed.

(4) Peak linear power for this analysis is determined from average linear power of 5.553 Kw/ft and a limiting FQ = 2.311. The FQ = 2.311 was established from the intersect of the normalized FQ vs. core height, (See Figure 4.3-21) where 1.0 is the TS FQ of 2.4, and power shape [K(Z)]. vs. core height.

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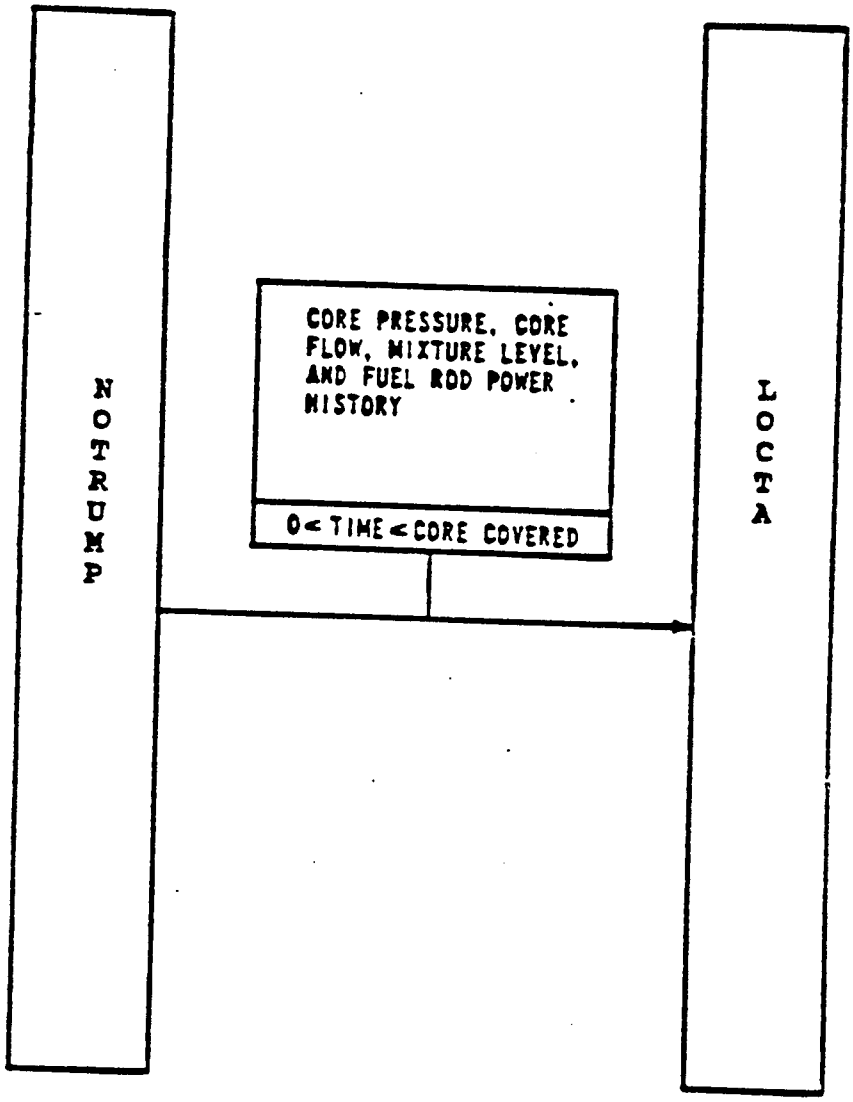
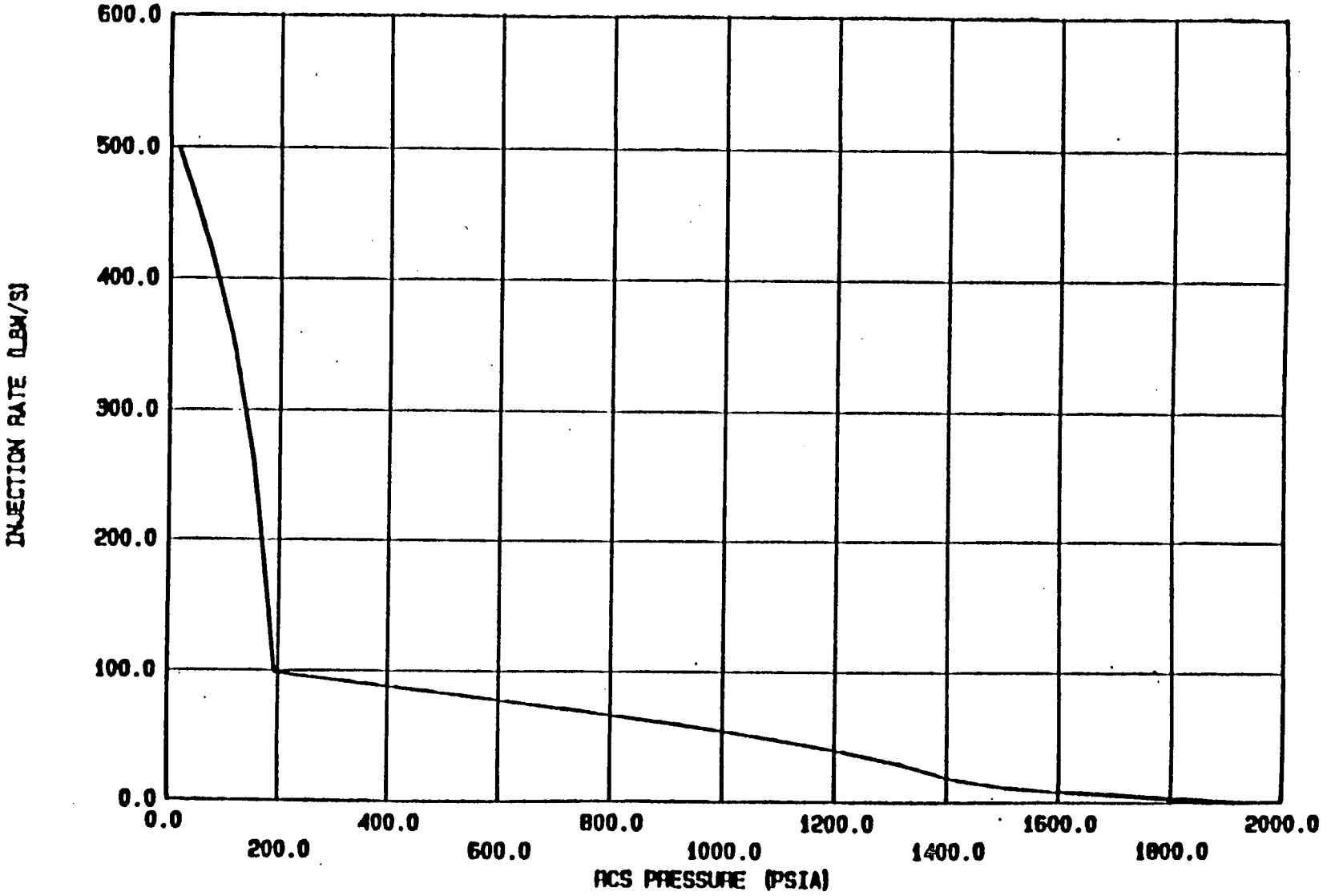


FIGURE 15.3-1 CODE INTERFACE DESCRIPTION FOR SMALL BREAK MODEL

FIGURE 15.3-2 SAFETY INJECTION VS. RCS PRESSURE



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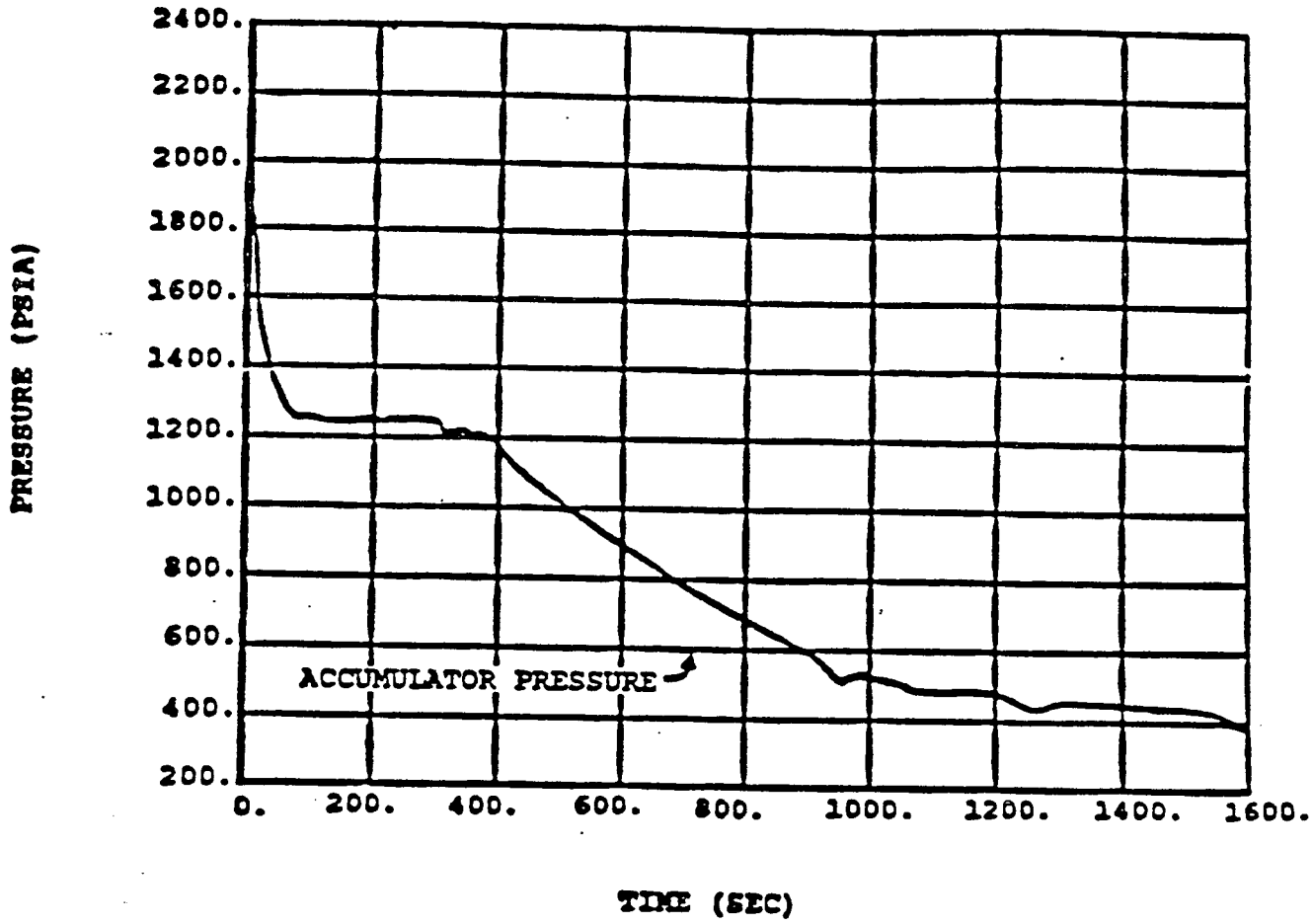

FIGURE 15.3-3 REACTOR COOLANT SYSTEM PRESSURE FOR THE 4-INCH BREAK

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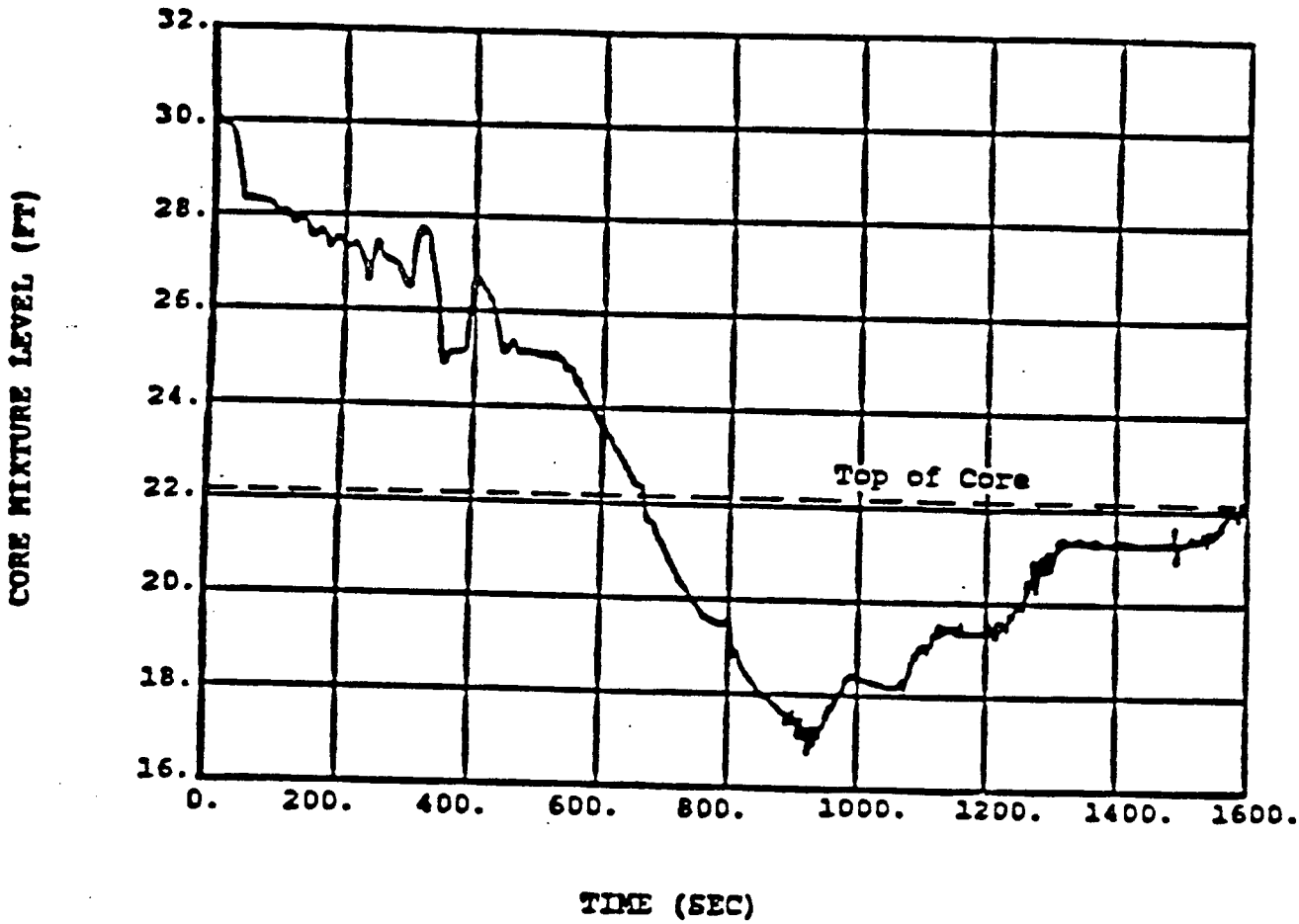

FIGURE 15.3-4 CORE MIXTURE HEIGHT TRANSIENT FOR THE 4-INCH BREAK

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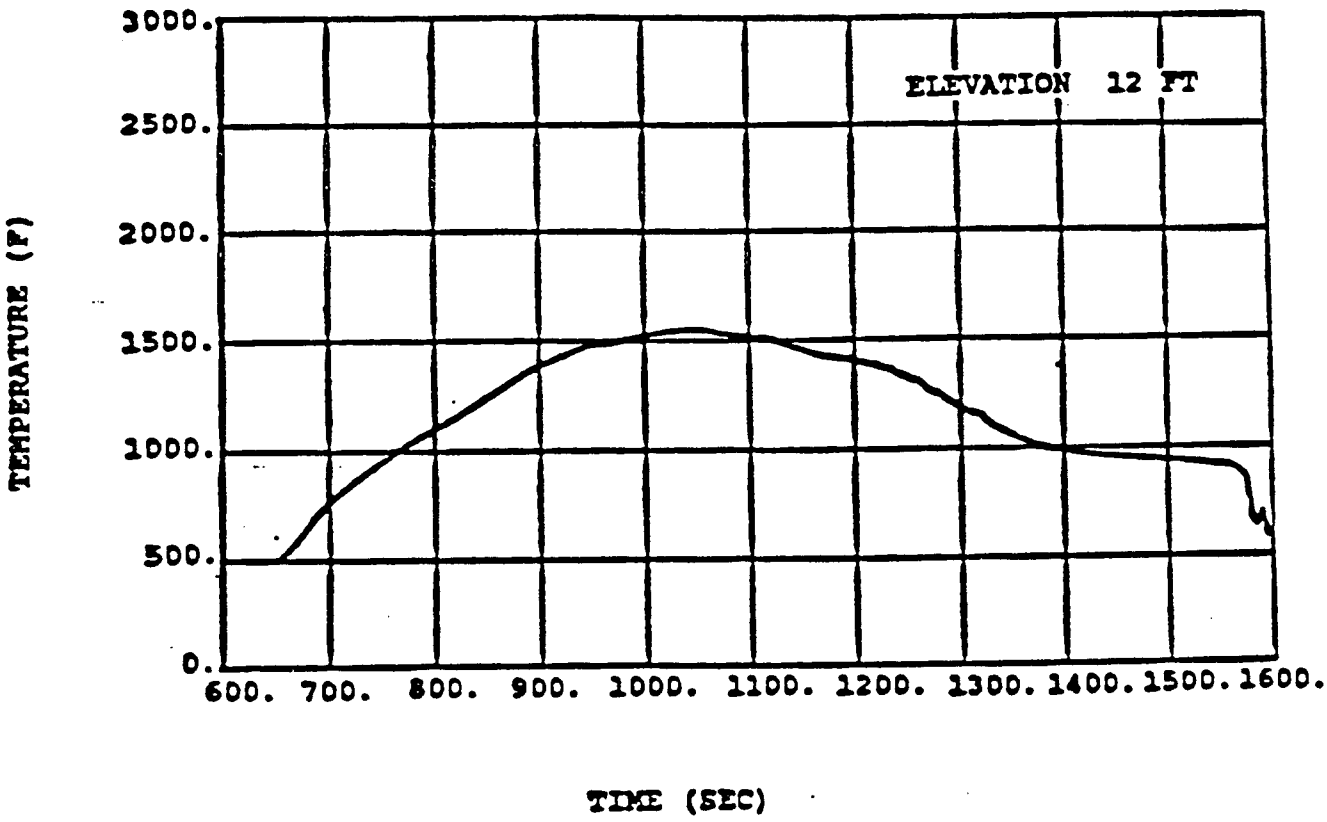

FIGURE 15.3-5 CLAD TEMPERATURE TRANSIENT FOR THE 4-INCH BREAK

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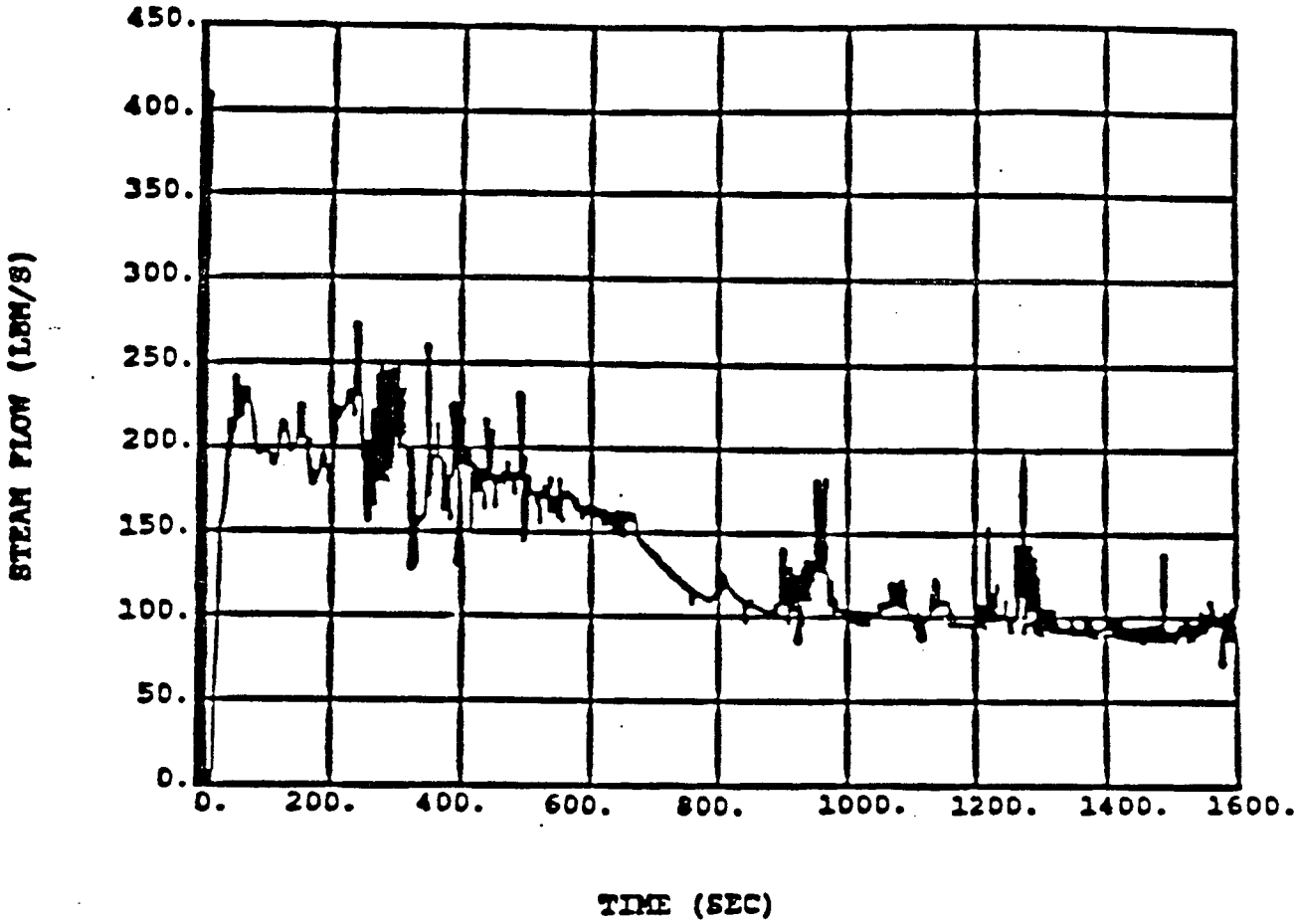

FIGURE 15.3-6 CORE STEAM FLOW RATE FOR THE 4-INCH BREAK

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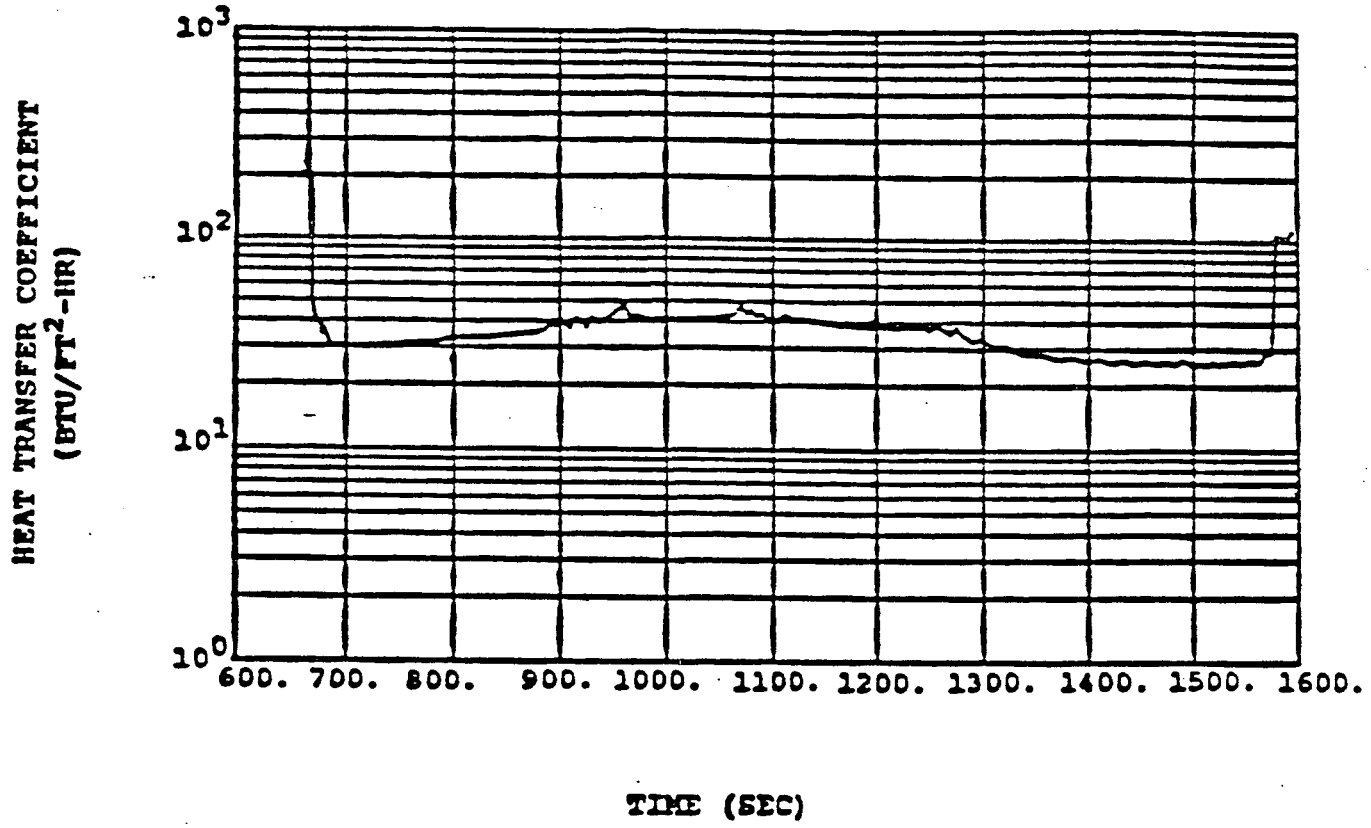

FIGURE 15.3-7 ROD HEAT TRANSFER COEFFICIENT AT THE PEAK TEMPERATURE ELEVATION FOR THE 4-INCH BREAK

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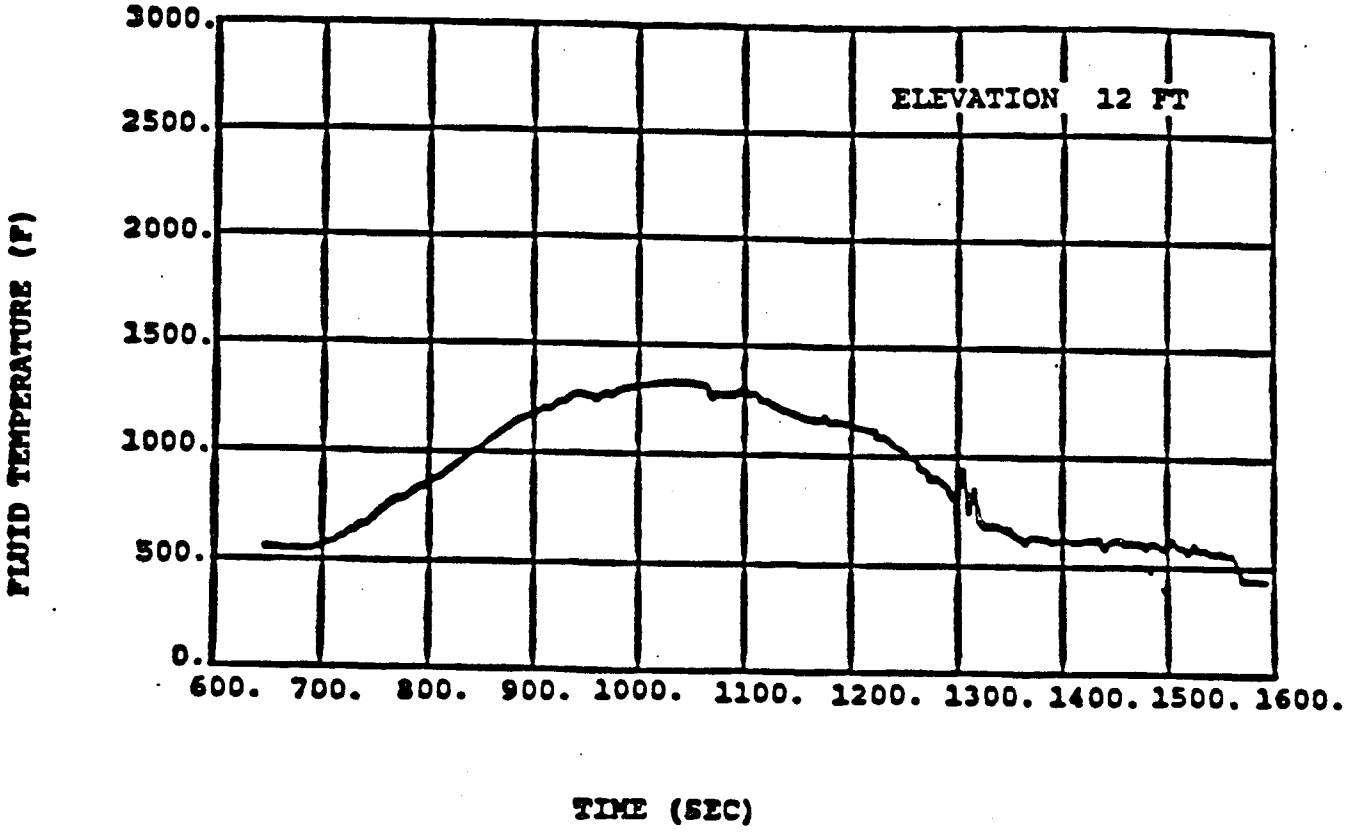

FIGURE 15.3-8a HOT SPOT FLUID TEMPERATURE FOR THE 4-INCH BREAK

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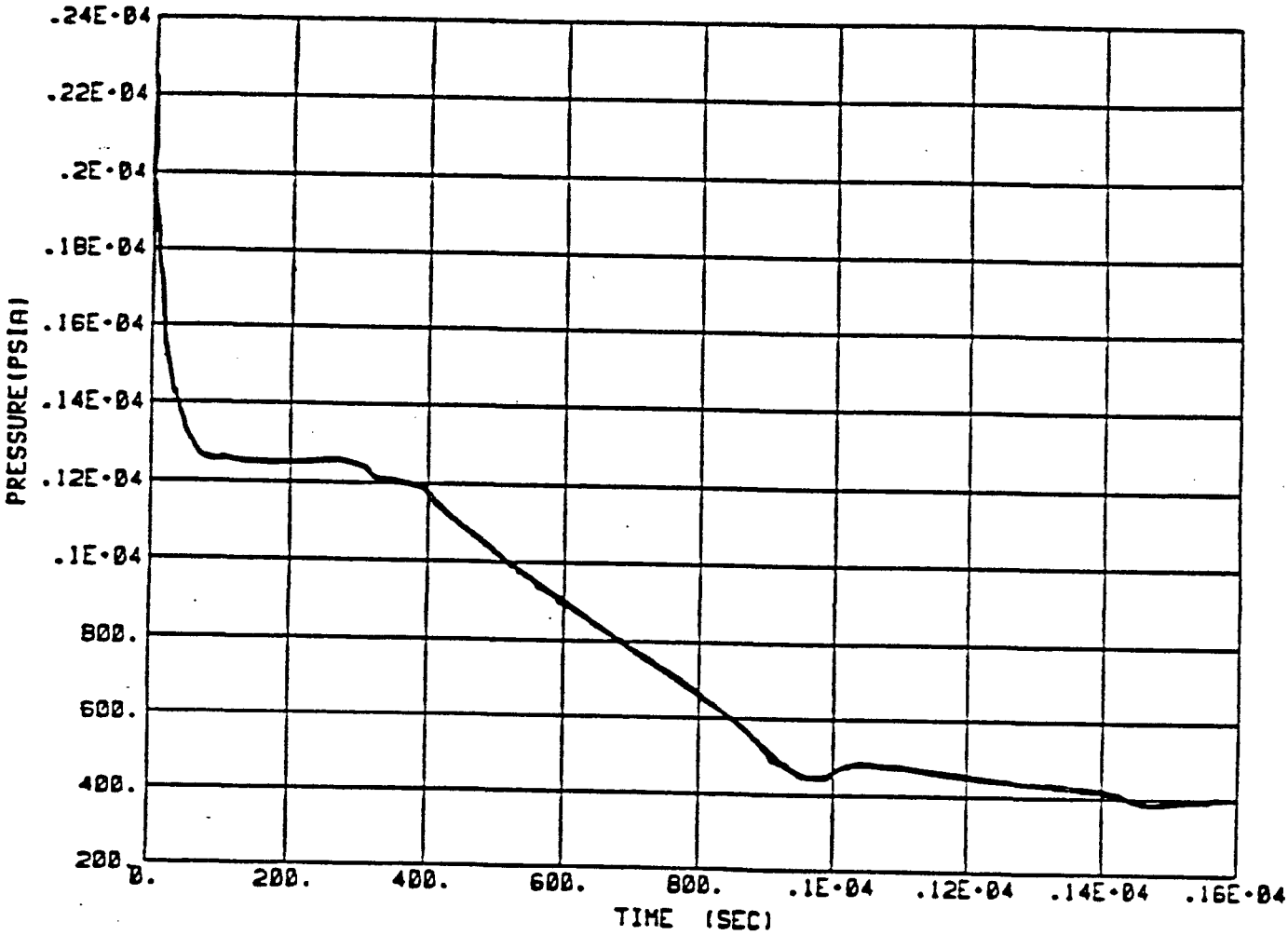

FIGURE 15.3-8b REACTOR COOLANT SYSTEM PRESSURE, 4-INCH BREAK, UPFLOW BARREL/BAFFLE REGION

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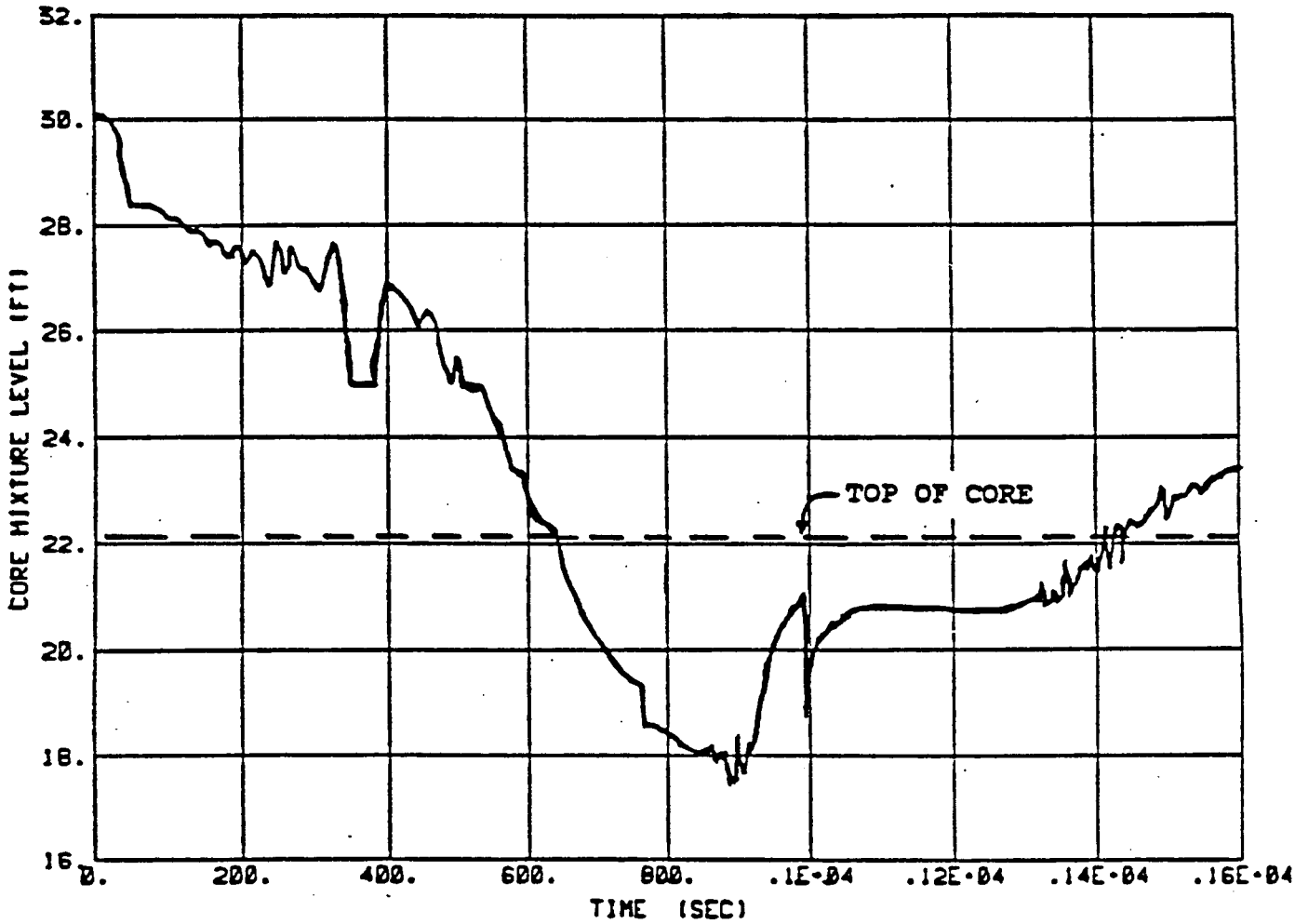

FIGURE 15.3-8c CORE MIXTURE HEIGHT TRANSIENT, 4-INCH BREAK, UPFLOW BARREL BAFFLE REGION

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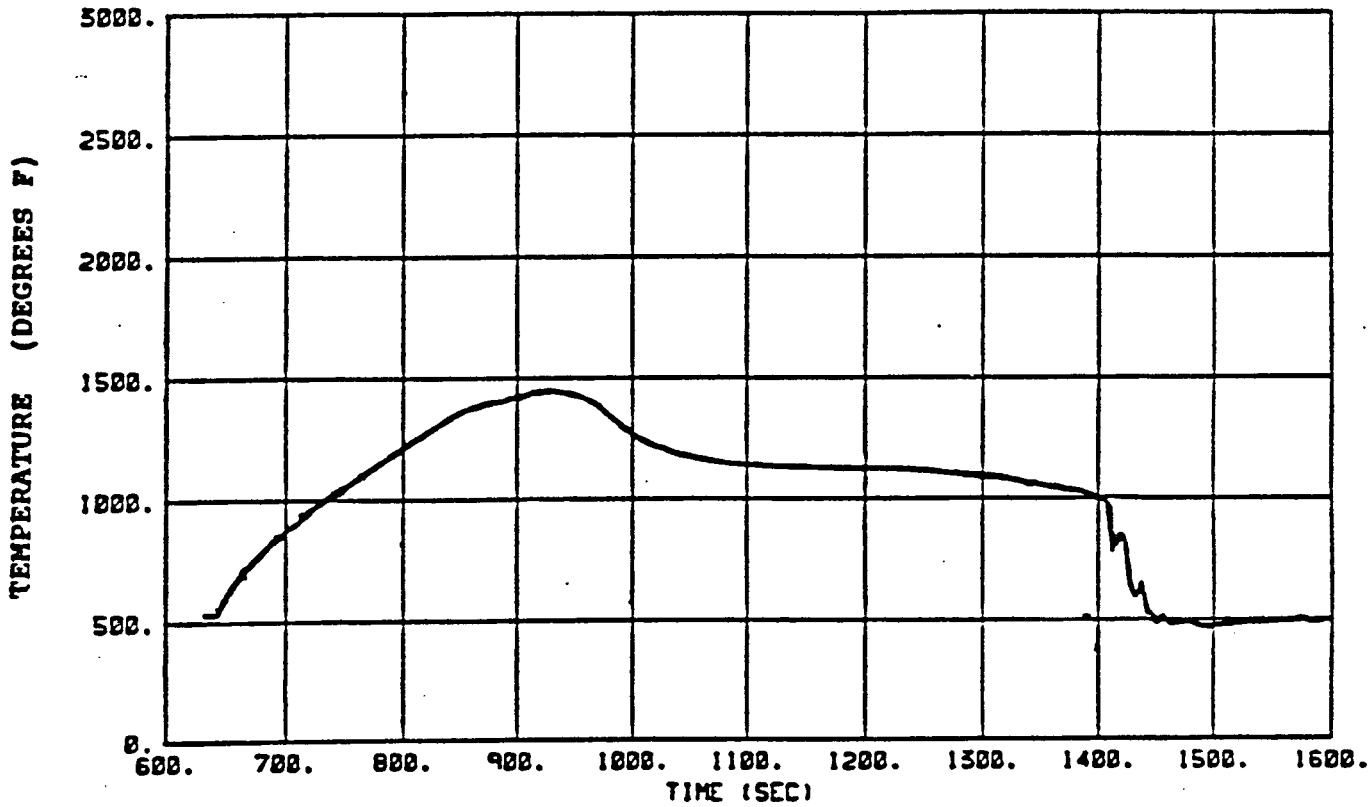

FIGURE 15.3-8d CLAD TEMPERATURE TRANSIENT, 4-INCH BREAK, UPFLOW BARREL/BAFFLE REGION

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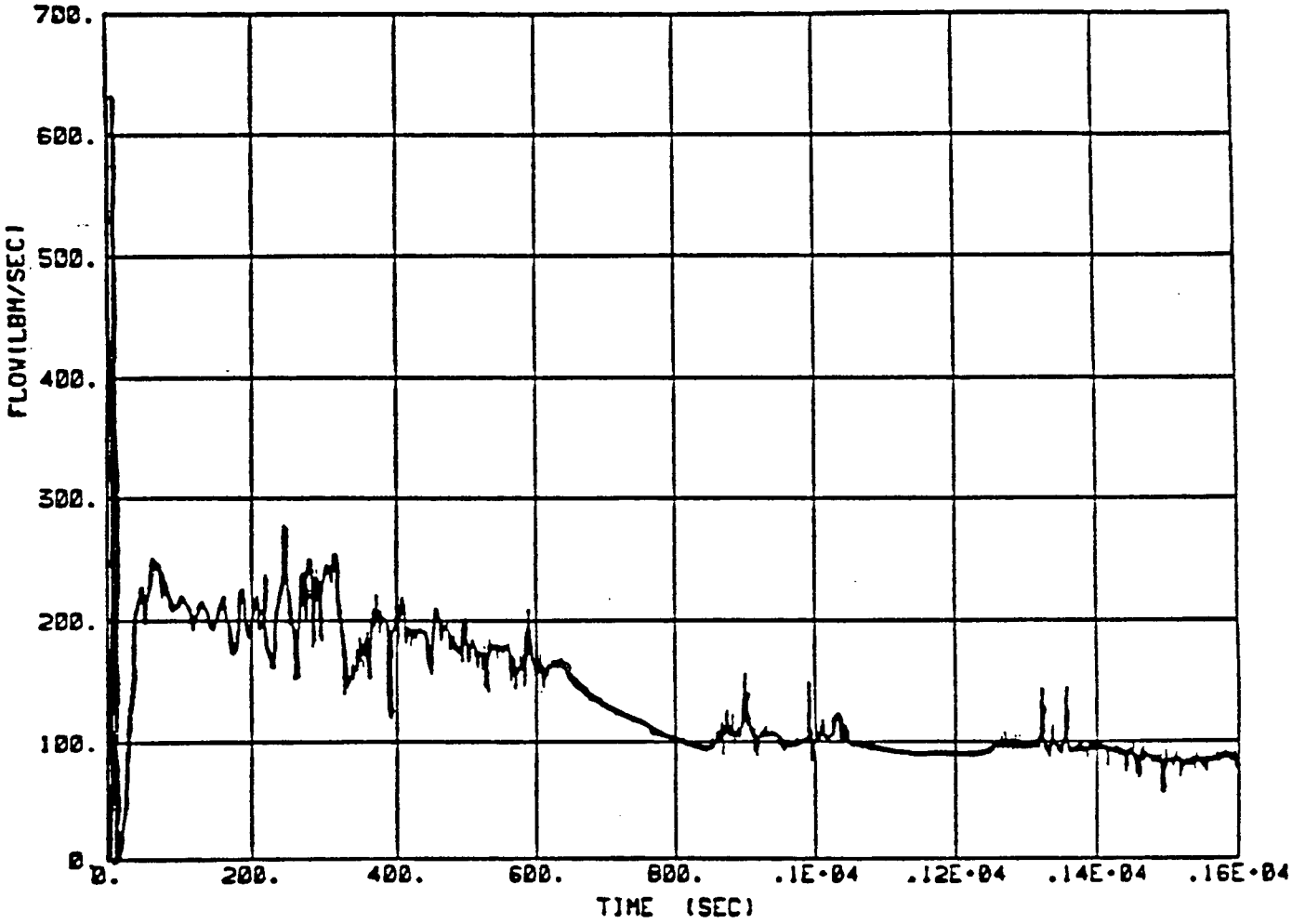

FIGURE 15.3-8e CORE STEAM FLOWRATE, 4-INCH BREAK, UPFLOW BARREL/BAFFLE REGION

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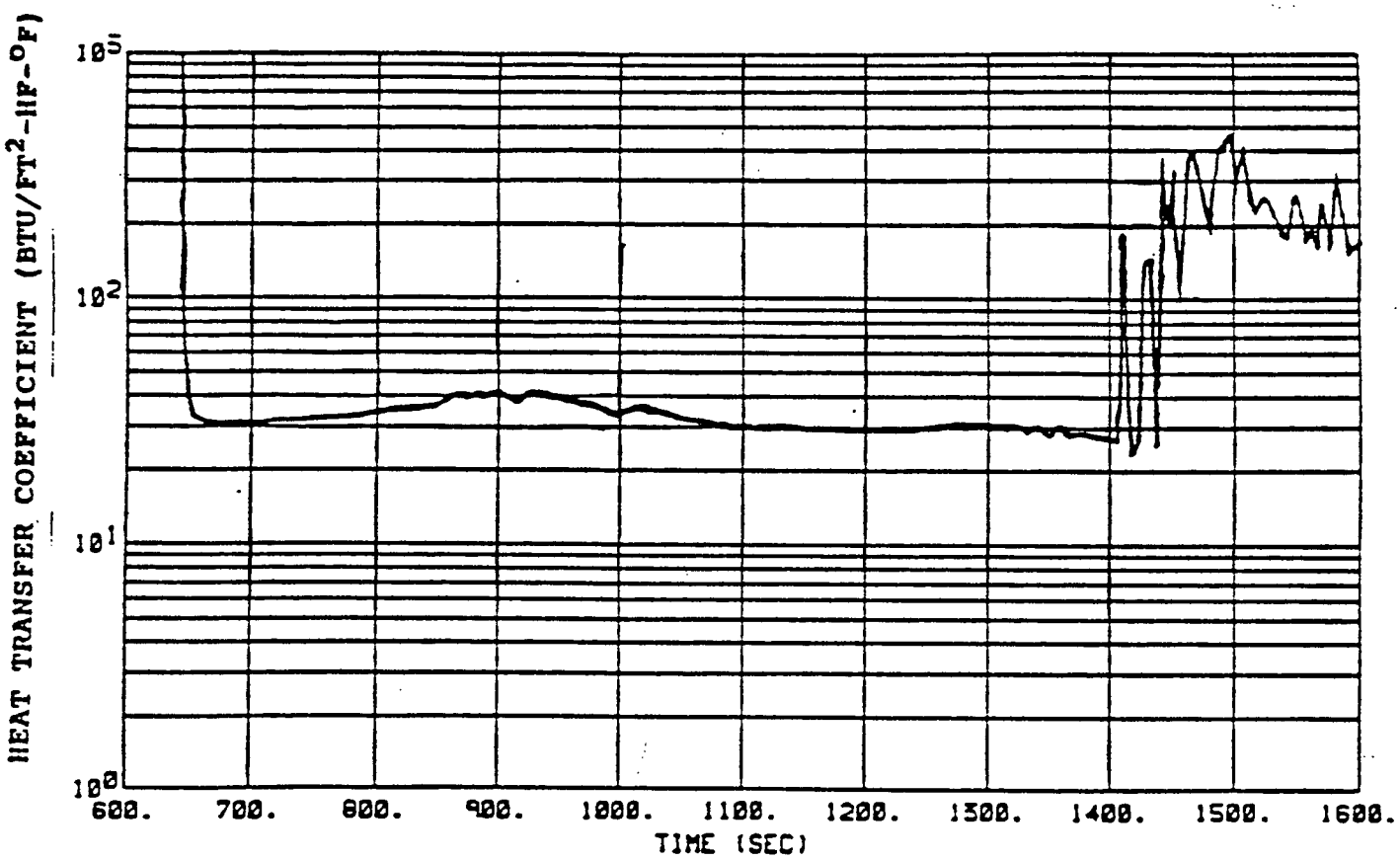

FIGURE 15.3-8f FUEL ROD HEAT TRANSFER COEFFICIENT, 4-INCH BREAK, UPFLOW BARREL/BAFFLE

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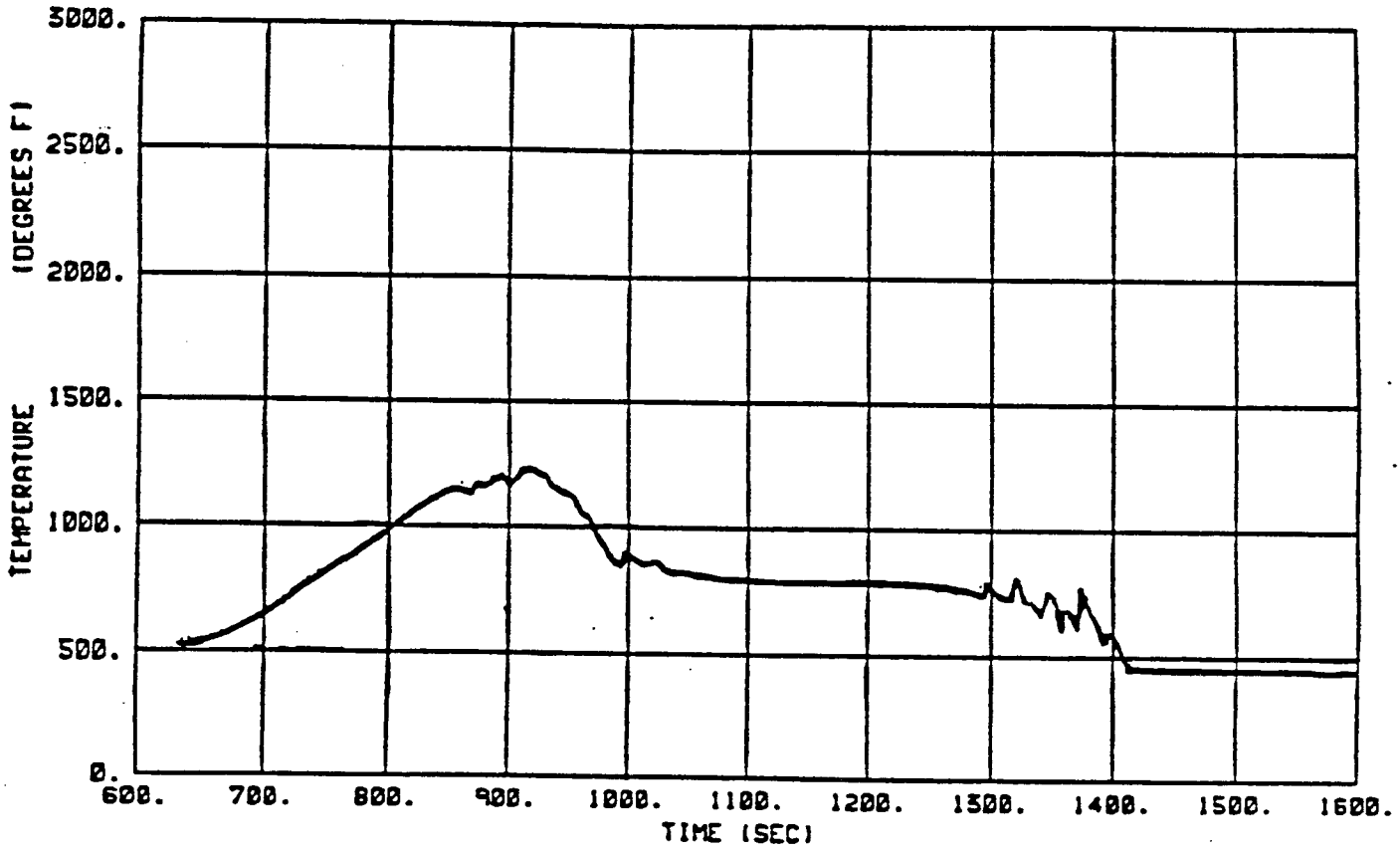
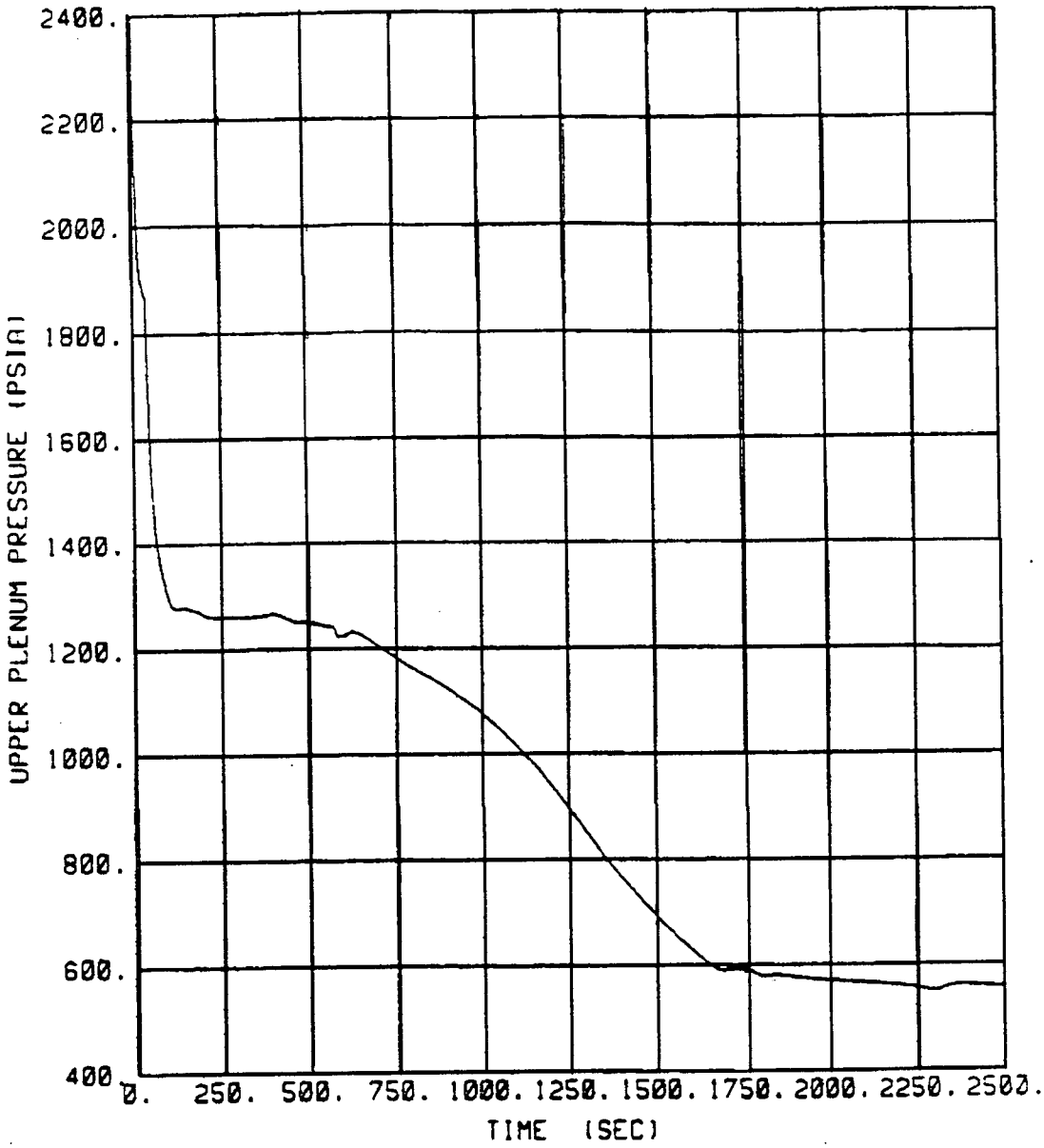


FIGURE 15.3-8g HOT SPOT FLUID TEMPERATURE, 4-INCH BREAK, UPFLOW BARREL/BAFFLE REGION



AMENDMENT 72

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

REACTOR COOLANT SYSTEM
PRESSURE, 3-INCH BREAK
UPFLOW BARREL/BAFFLE REGION
FIGURE 15.3-8h

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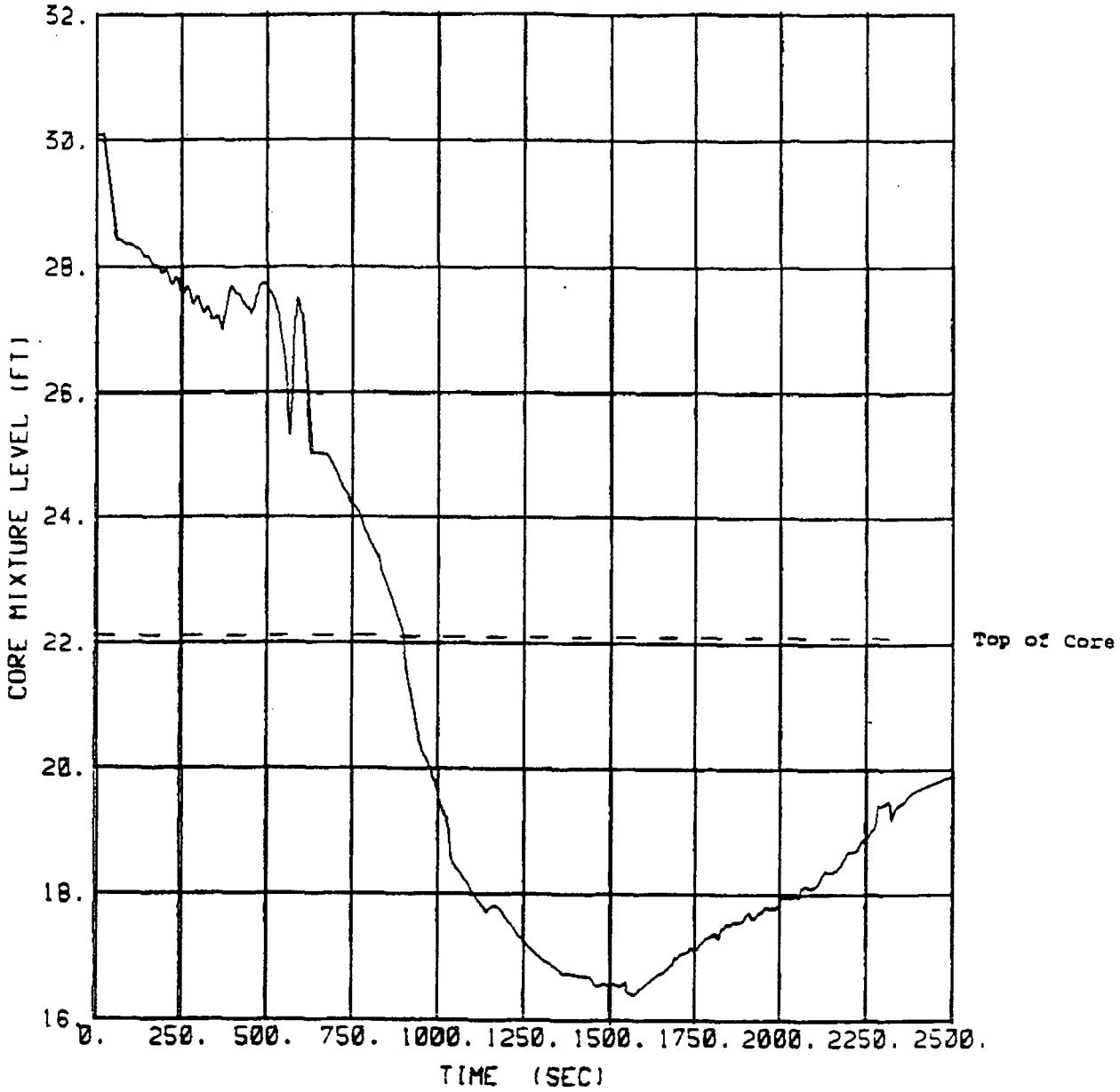

AMENDMENT 72

WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT

MIXTURE LEVEL TRANSIENT
3-INCH BREAK
UPFLOW BARREL/BAFFLE REGION

FIGURE 15.3-8i

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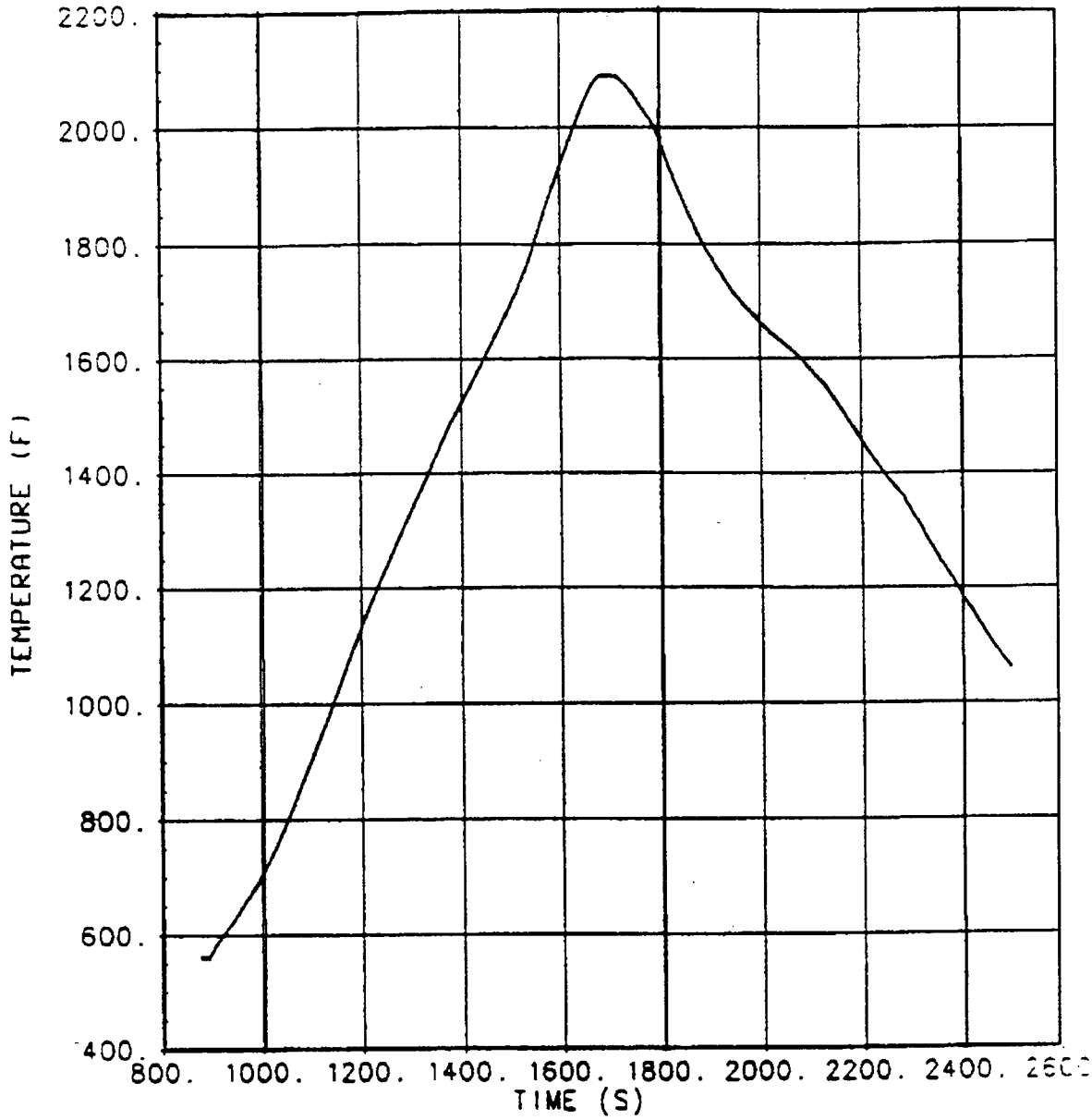

AMENDMENT 72

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

PEAK CLAD TEMPERATURE TRANSIENT
3-INCH BREAK
UPFLOW BARREL/BAFFLE REGION
FIGURE 15.3-8j

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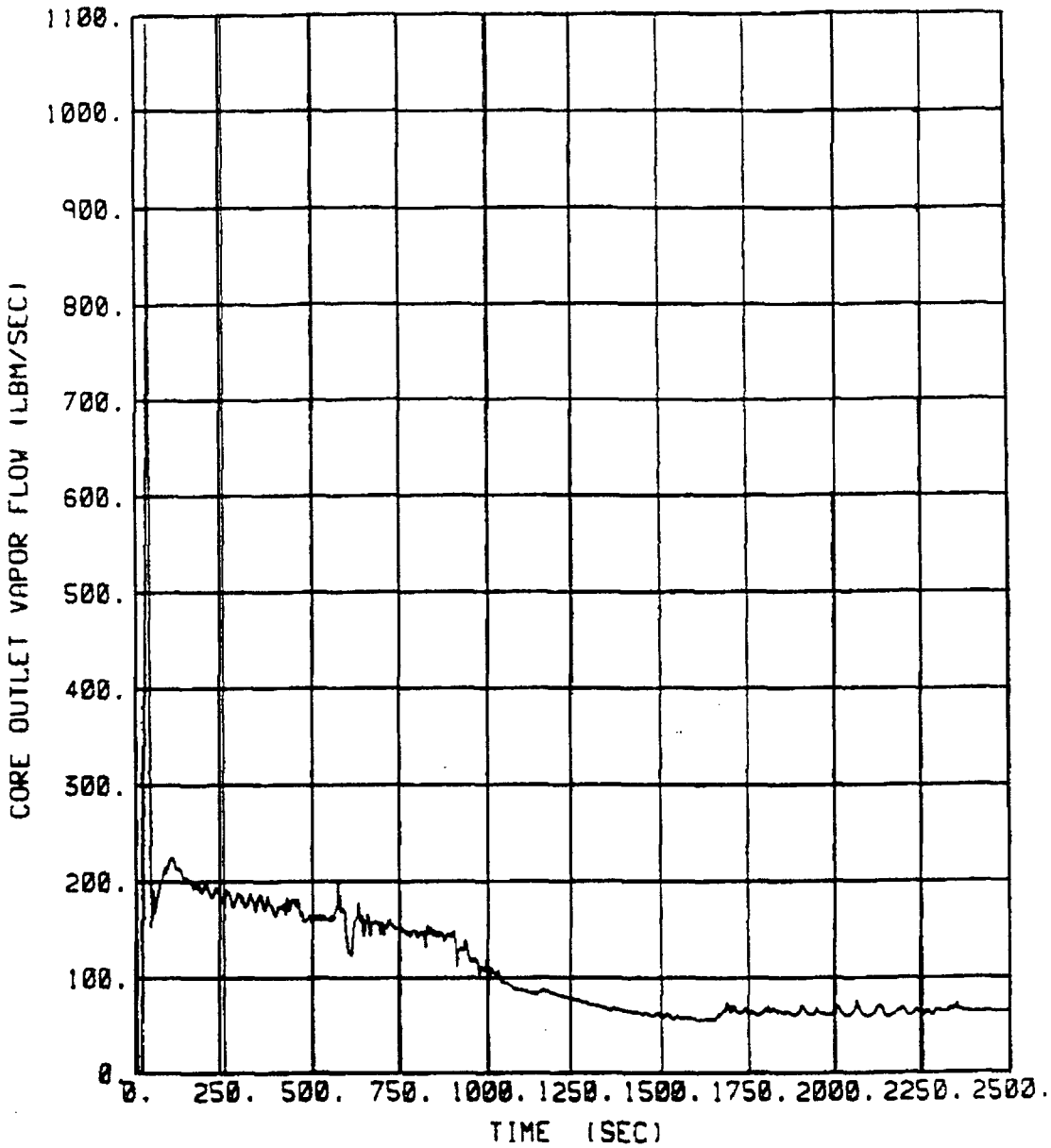

AMENDMENT 72

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

STEAM FLOW RATE
3-INCH BREAK
UPFLOW BARREL/BAFFLE REGION
FIGURE 15.3-8k

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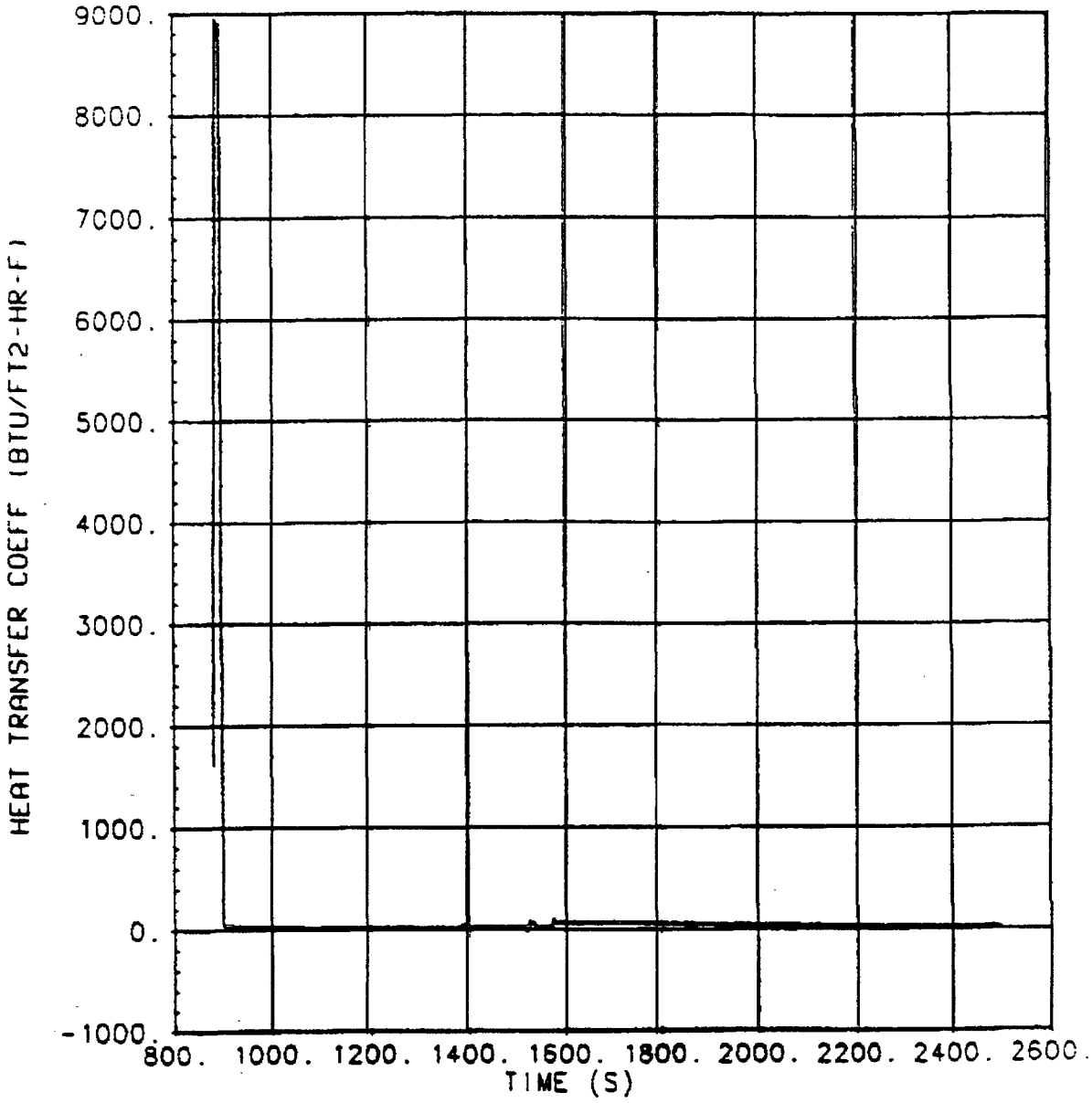

AMENDMENT 72

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

HEAT TRANSFER COEFFICIENT
3-INCH BREAK
UPFLOW BARREL/BAFFLE REGION
FIGURE 15.3-8L

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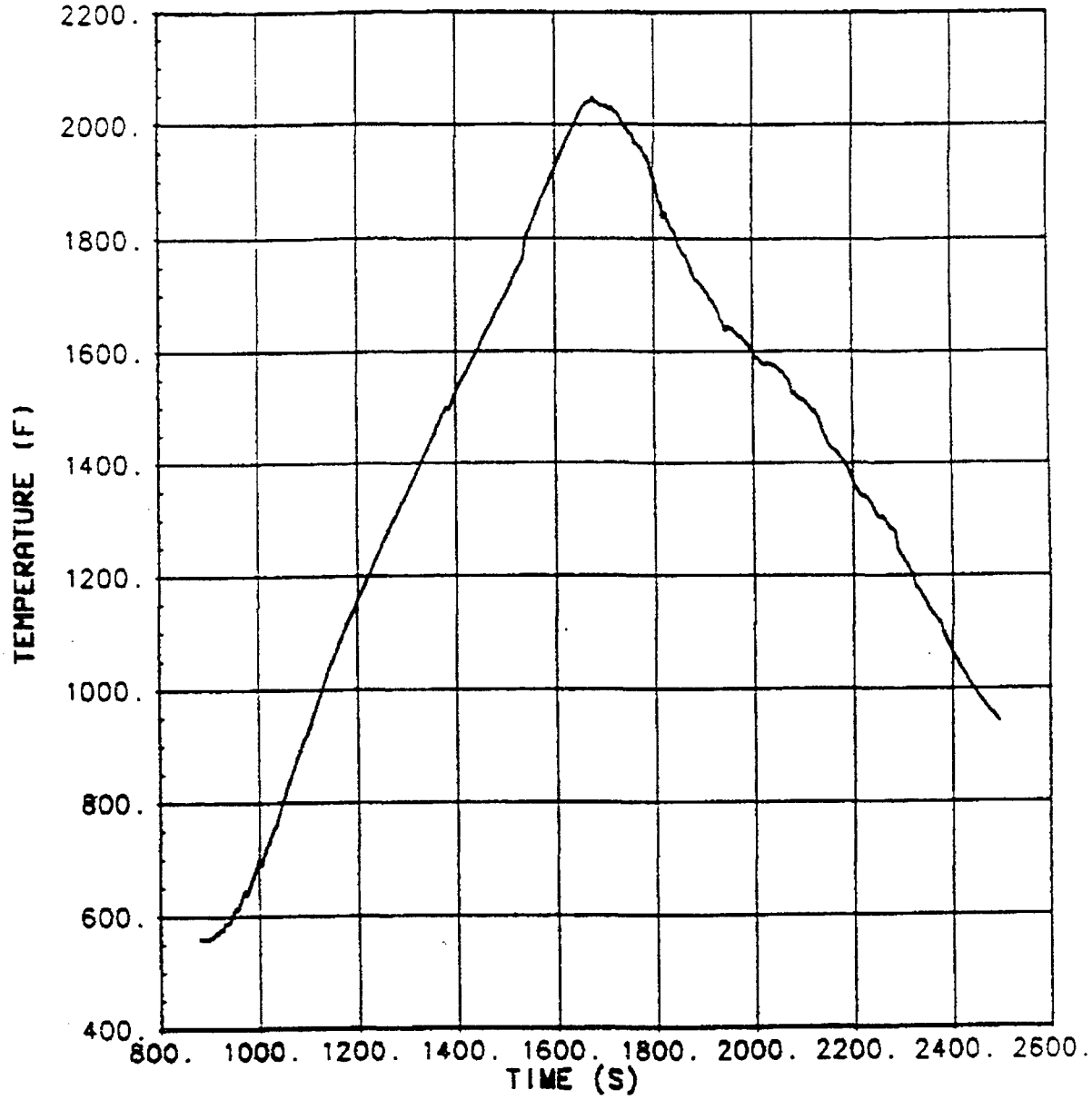

AMENDMENT 72

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

HOT SPOT FLUID TEMPERATURE
3-INCH BREAK
UPFLOW BARREL/BAFFLE REGION
FIGURE 15.3-8m

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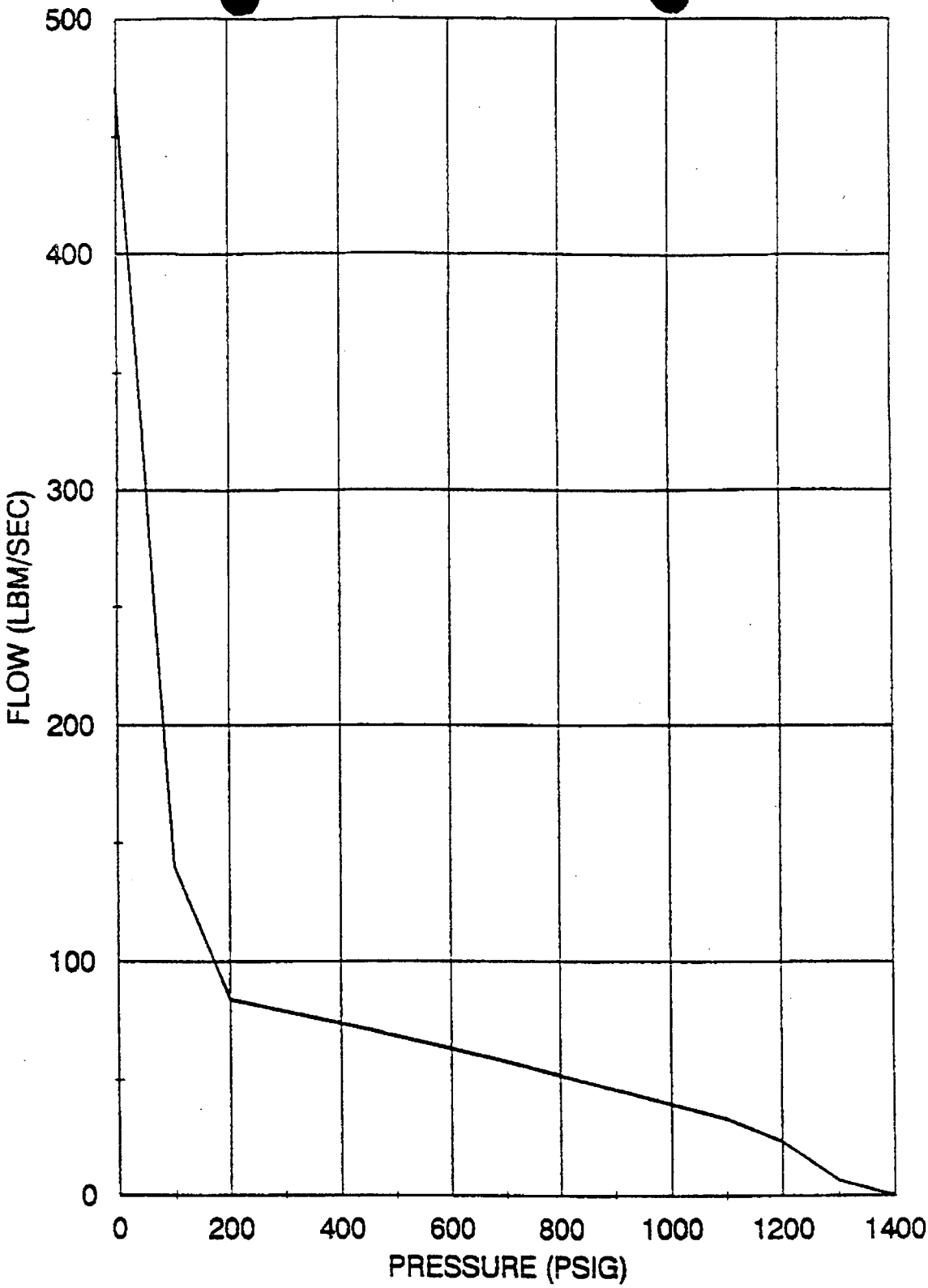

AMENDMENT 72

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

SAFETY INJECTION VS. RCS PRESSURE
3-INCH BREAK
UPFLOW BARREL/BAFFLE REGION
FIGURE 15.3-8n

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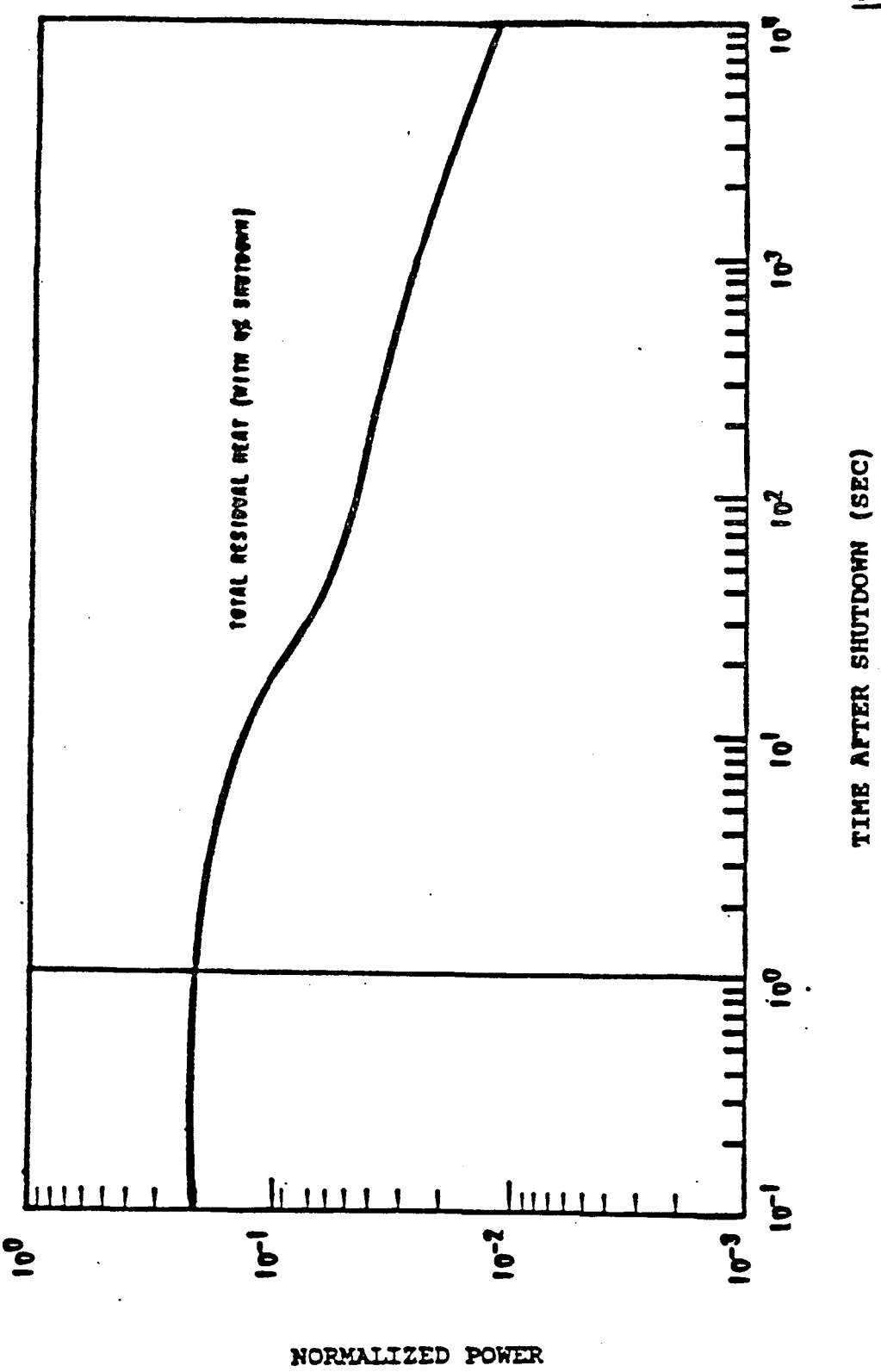
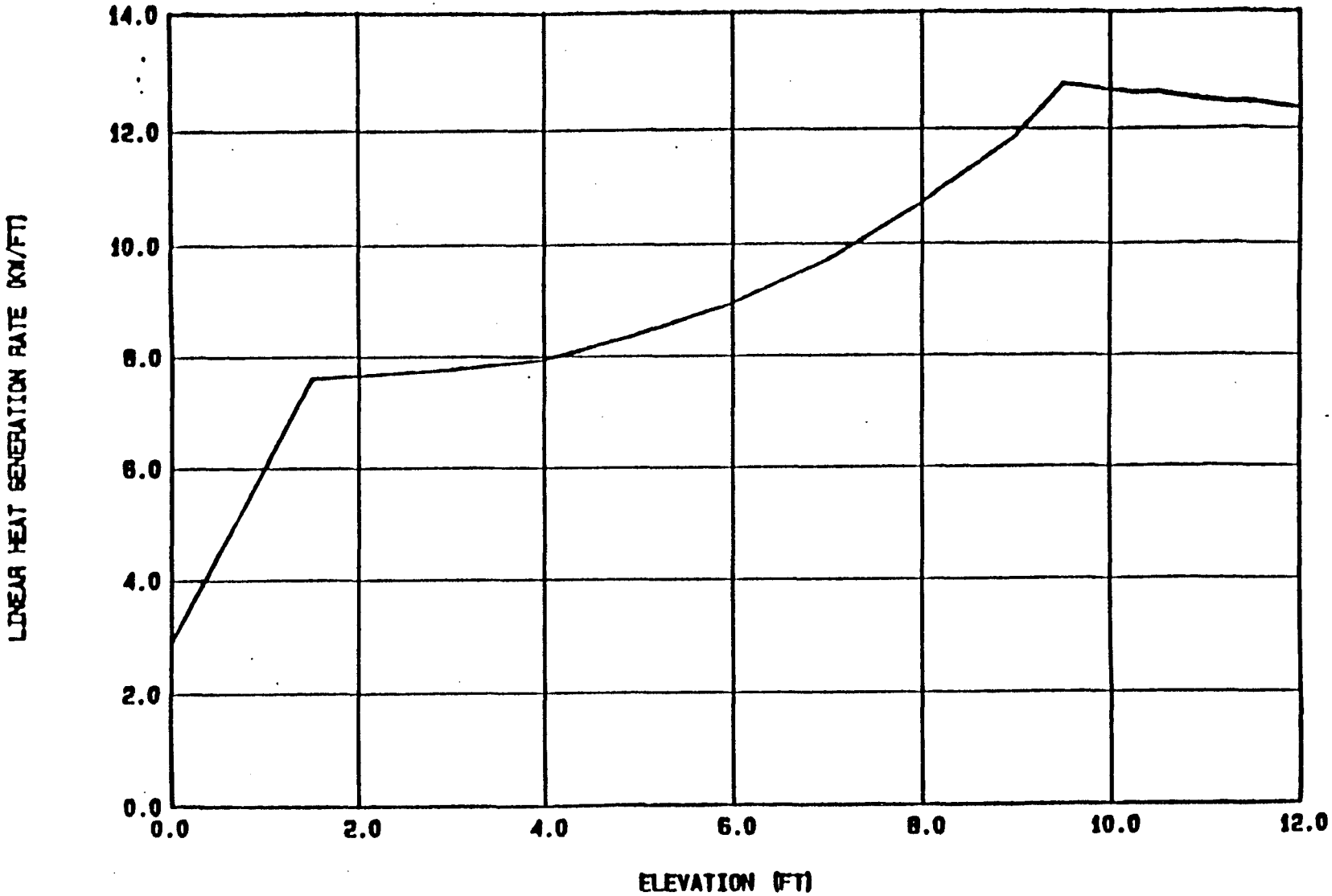


FIGURE 15.3-9 CORE POWER TRANSIENT

FIGURE 15.3-10 HOT ROD AXIAL POWER SHAPE



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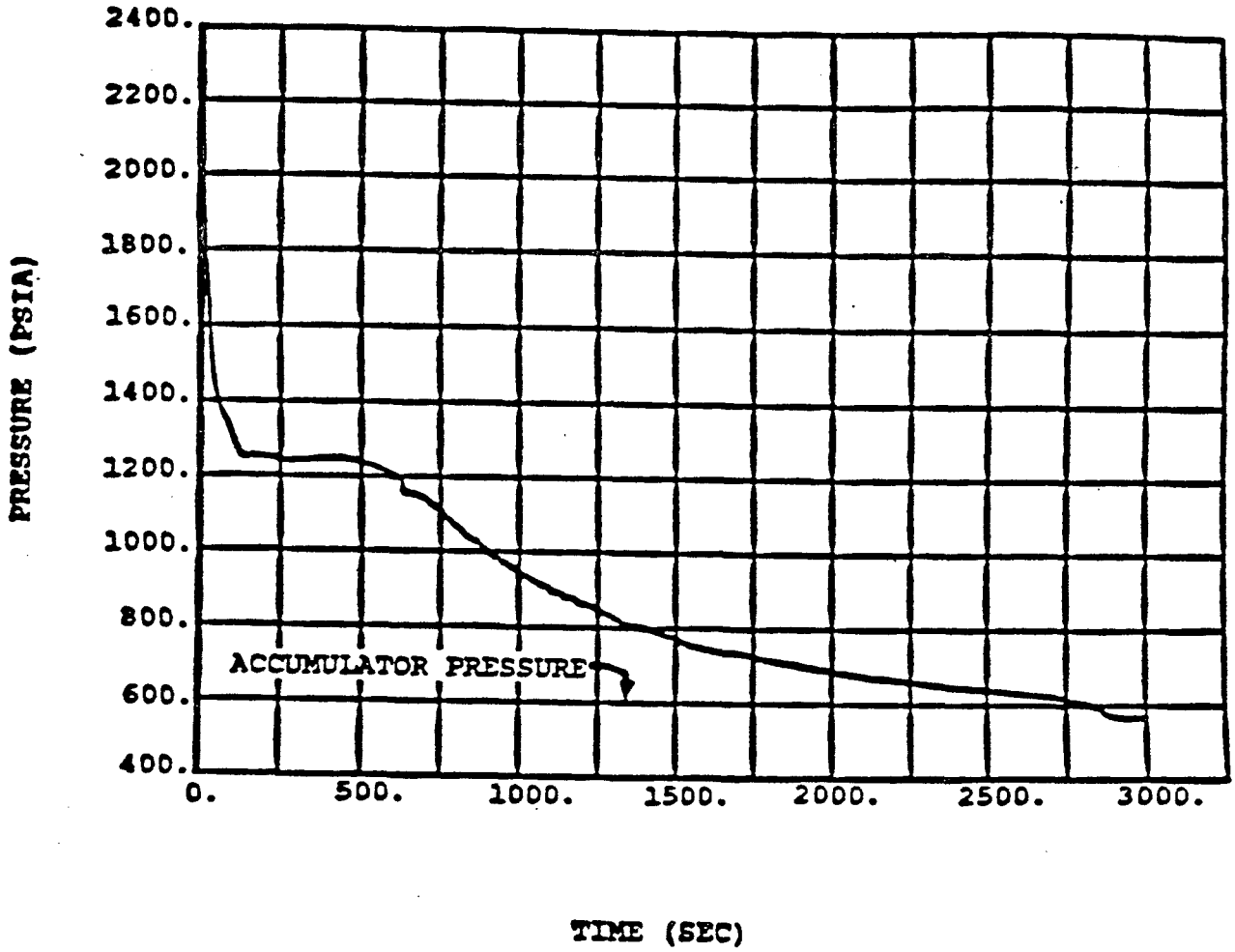

FIGURE 15.3-11 REACTOR COOLANT SYSTEM PRESSURE FOR THE 3-INCH BREAK

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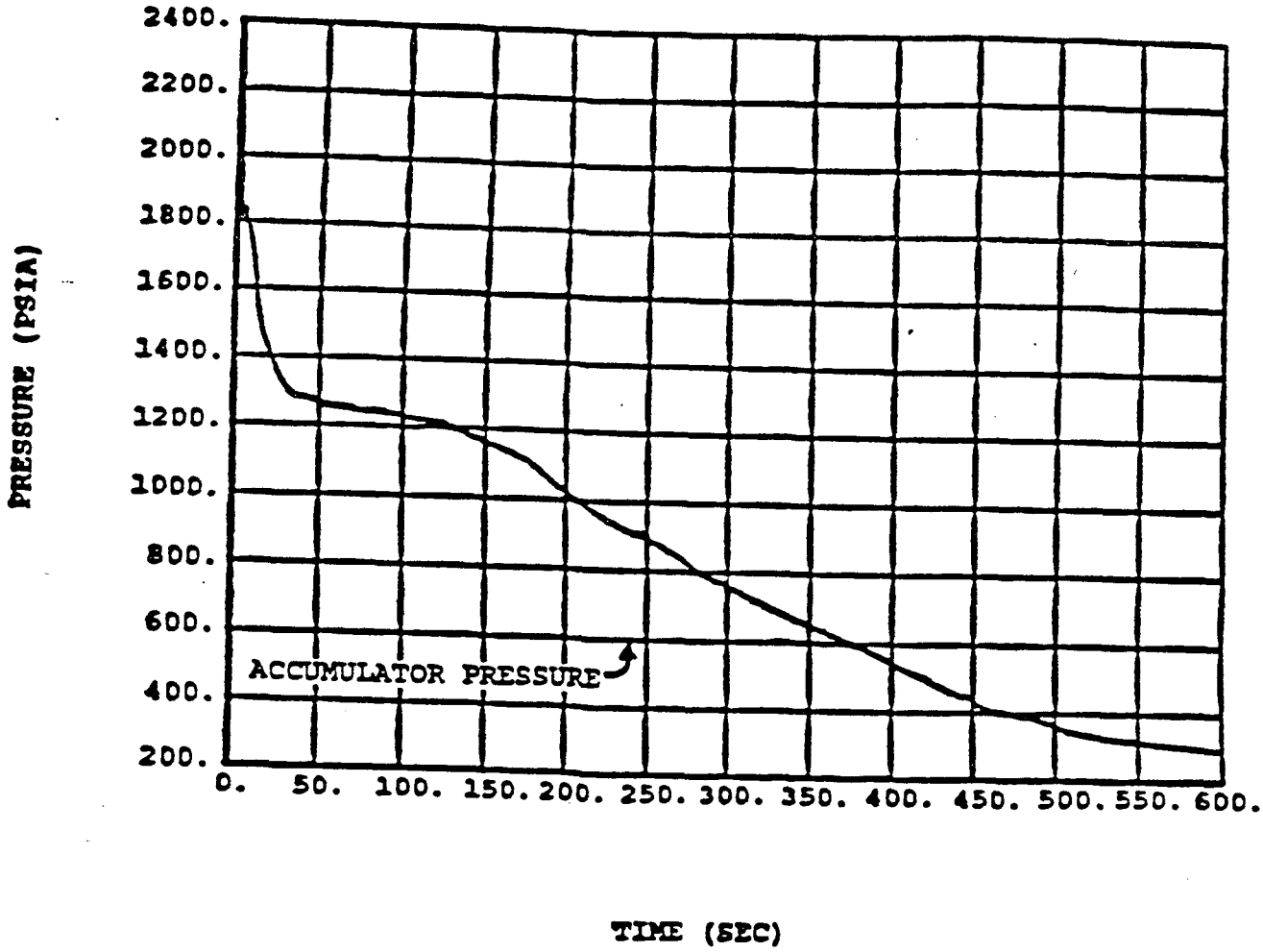

FIGURE 15.3-12 REACTOR COOLANT SYSTEM PRESSURE FOR THE 6-INCH BREAK

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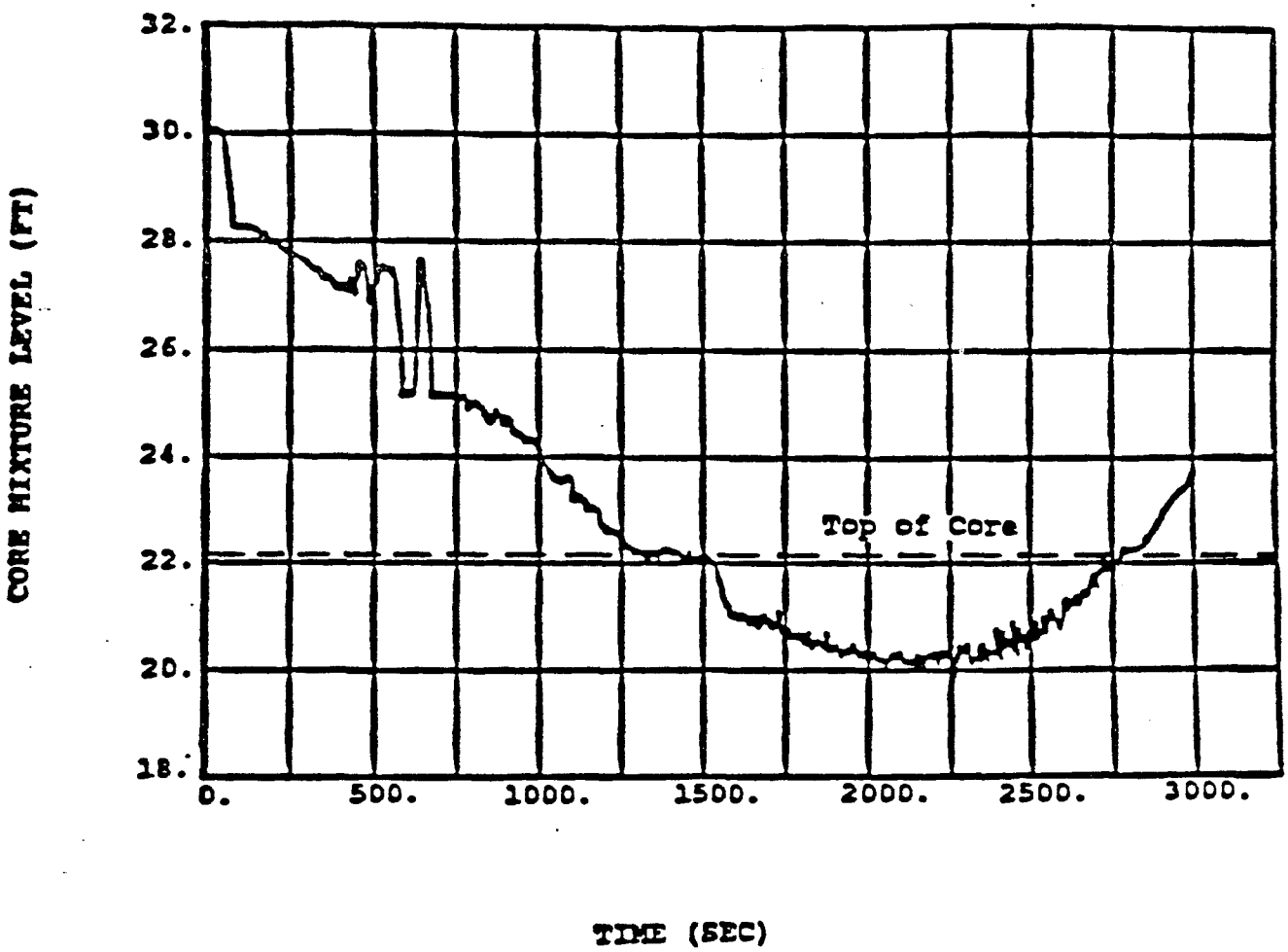

FIGURE 15.3-13a CORE MIXTURE HEIGHT TRANSIENT FOR THE 3-INCH BREAK

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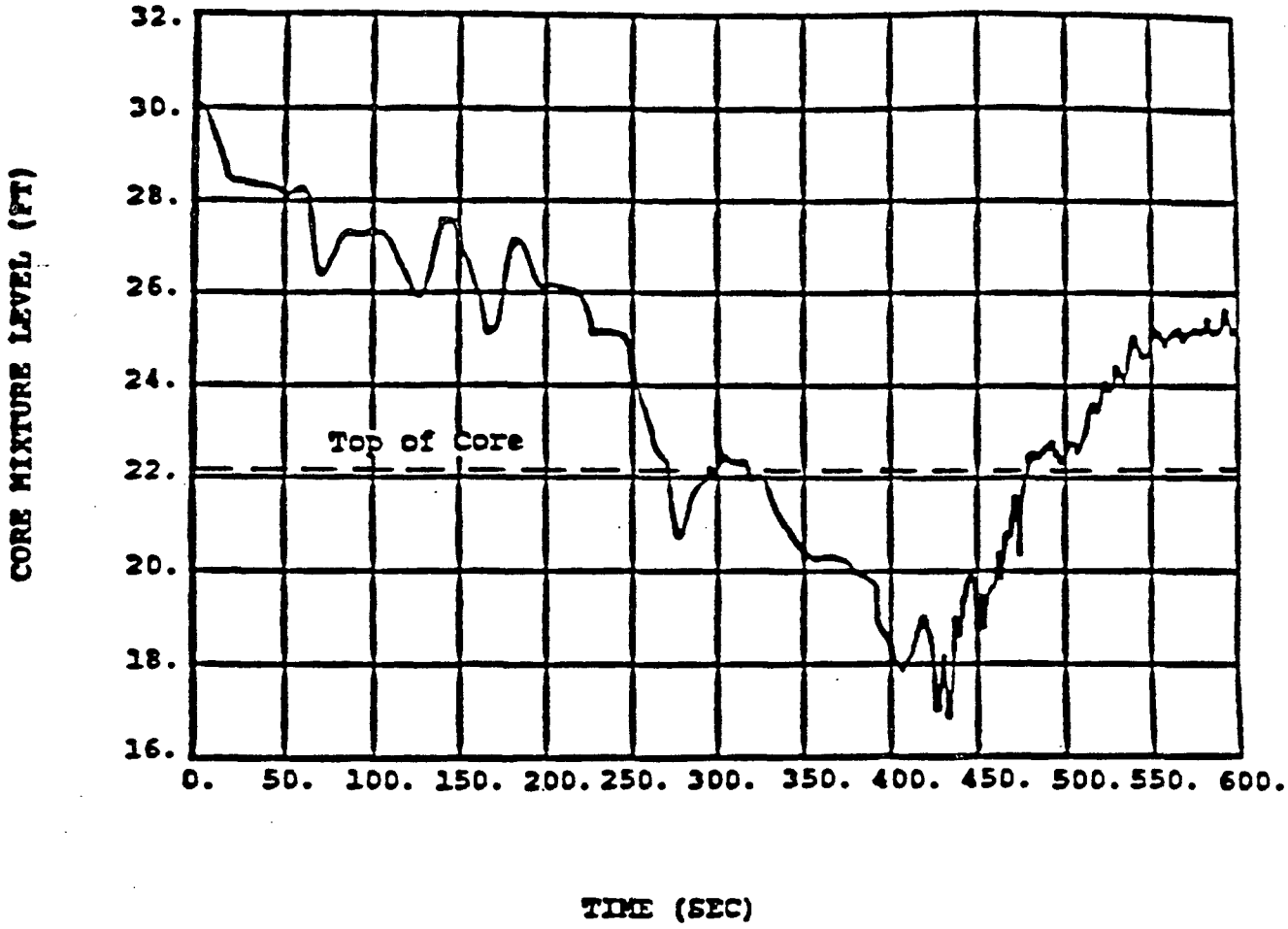

FIGURE 15.3-13b CORE MIXTURE HEIGHT TRANSIENT FOR THE 6-INCH BREAK

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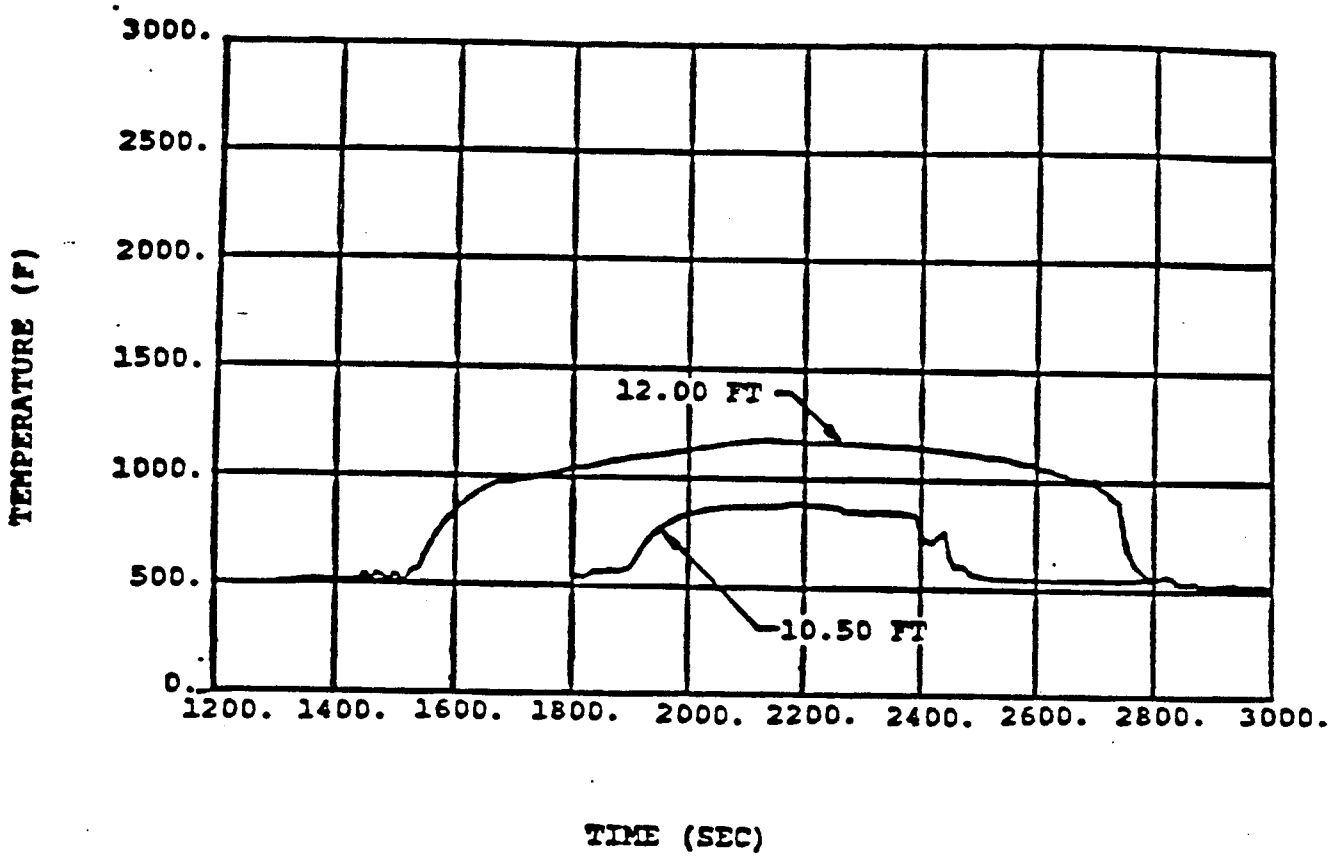

FIGURE 15.3-14 a CLAD TEMPERATURE TRANSIENT FOR THE 3-INCH BREAK

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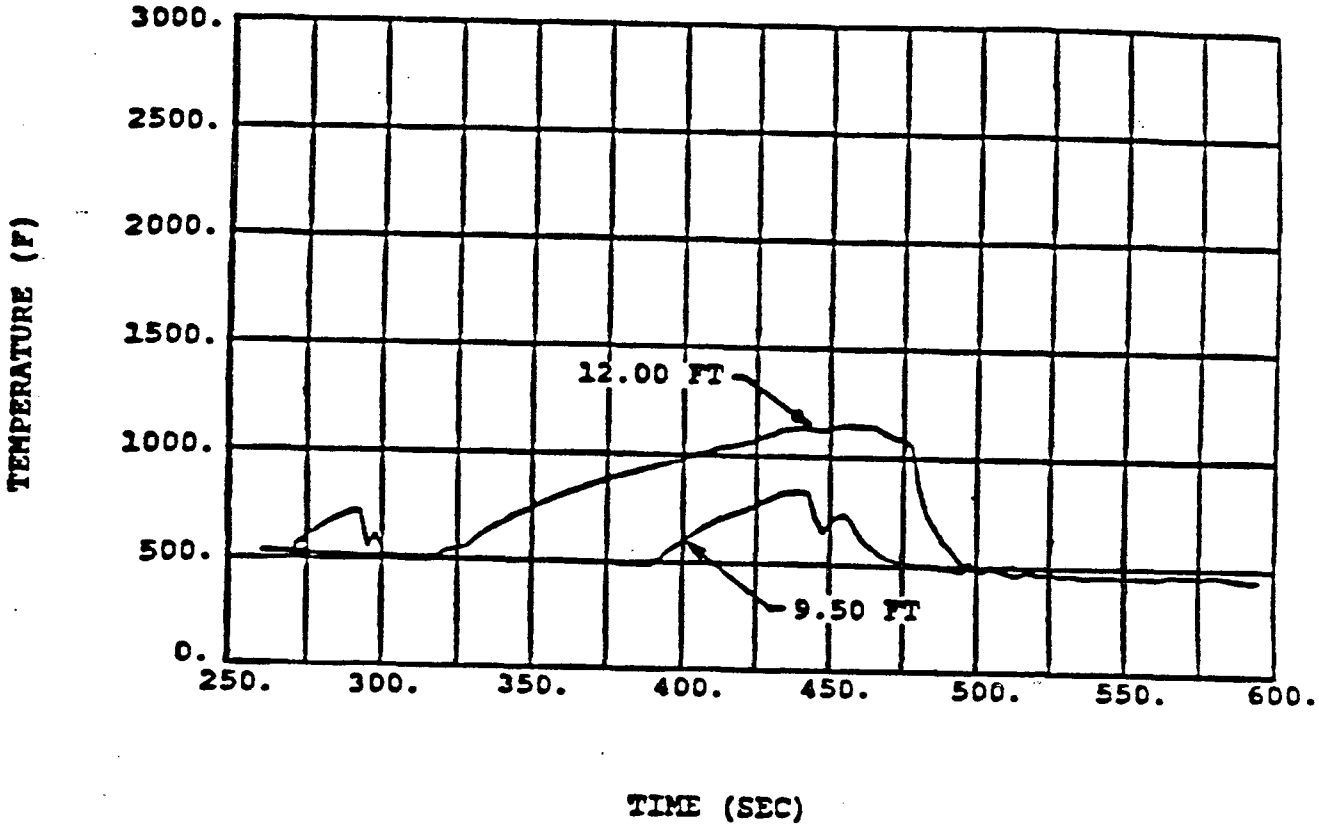

FIGURE 15.3-14 b CLAD TEMPERATURE TRANSIENT FOR THE 6-INCH BREAK