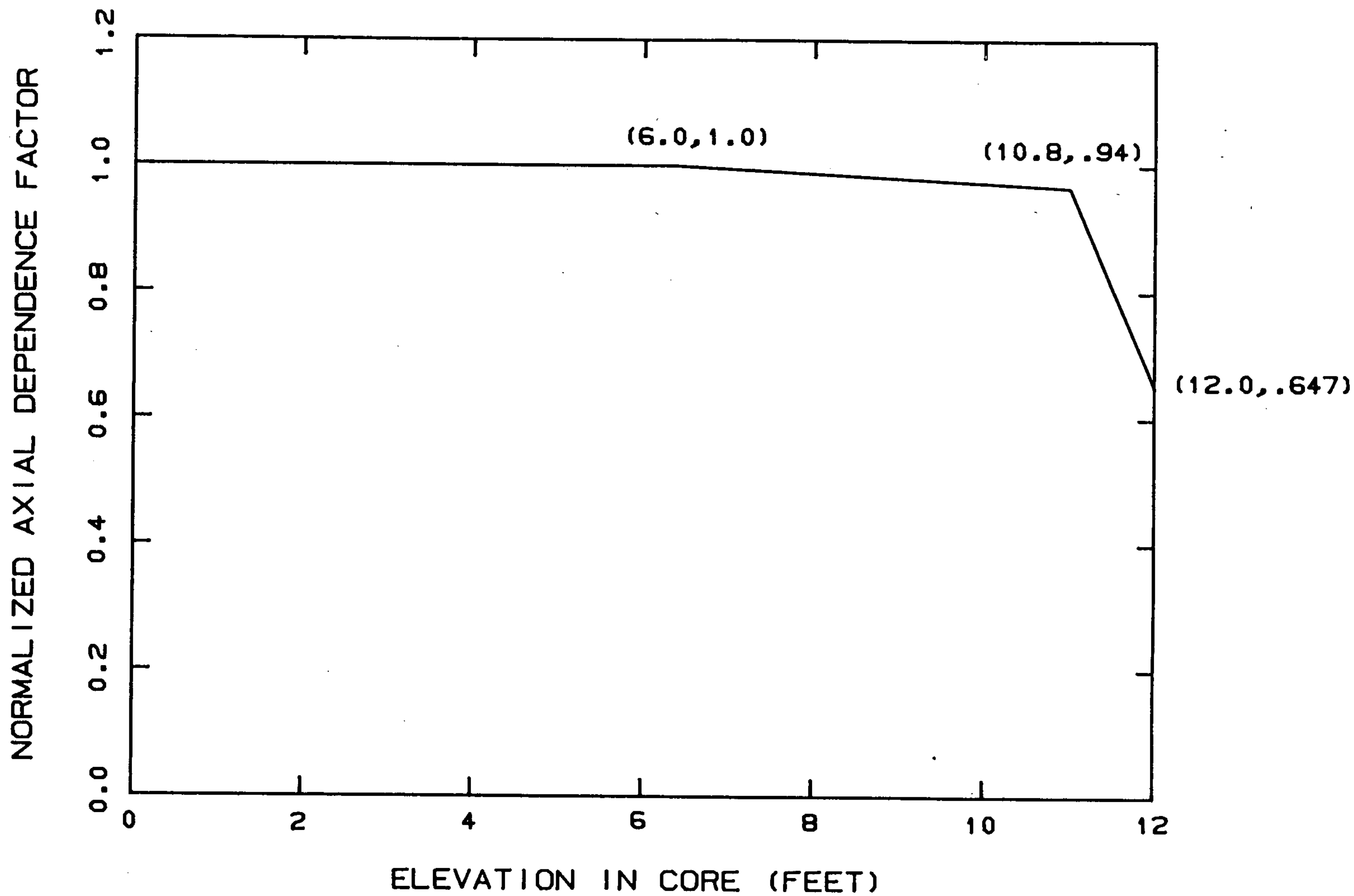

FIGURE 3.10-3 NORMALIZED AXIAL DEPENDENCE FACTOR FOR Fq VERSUS ELEVATION (PEAK Fq = 2.32)

ENCLOSURE 2

15.6.2 SMALL BREAK LOSS-OF-COOLANT ACCIDENTS

15.6.2.1 Identification of Causes and Frequency Classification/Acceptance Criteria and

Frequency Classification

A loss-of-coolant accident (LOCA) is the result of a pipe rupture of the Reactor Coolant System pressure boundary. A major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 sq. ft. This event is considered an ANS Condition IV event, a limiting fault. See Section 15.0.1 for a discussion of Condition IV events.

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

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

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

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

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

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

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

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

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

Description of Small Break LOCA Transient

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

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the Reactor Coolant

System through the postulated break against the charging pump makeup flow at normal Reactor Coolant System pressure, i.e., 2250 psia. A makeup flow rate from one positive displacement charging pump is typically adequate to sustain pressurizer level at 2250 PSIA for a break through a 0.295 inch diameter hole. This break results in a loss of approximately 10.6 lb/sec.

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

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

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

Before the break occurs, the plant is in an equilibrium condition; i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from decay, hot internals, and the vessel continues to be transferred to the Reactor Coolant System. The heat transfer between the Reactor Coolant System and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary, system pressure increases and steam dump may occur. Makeup to the secondary side is automatically provided by the auxiliary feedwater pumps. The safety injection signal stops normal feedwater flow by closing the main feedwater line isolation valves and initiates auxiliary feedwater flow by starting auxiliary feedwater pumps. The secondary flow aids in the reduction of Reactor Coolant System pressures.

When the RCS depressurizes to 615 psia, the cold leg accumulators begin to inject water into the reactor coolant loops. Due to the loss of offsite power assumption, the reactor coolant pumps are assumed to be tripped at the time of reactor during the accident and the effects of pump coastdown are included in the blowdown analyses.

15.6.2.2 Analysis of Effects and Consequences

Method of Analysis

The requirements of an acceptable ECCS Evaluation Model are presented in Appendix K of 10 CFR 50 (Reference 15.6.2-1). The requirements of Appendix K regarding specific model features were met by selecting models which provide a significant overall conservatism in the analysis. The assumptions made pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs and include such items as the core peaking factors, the containment pressure, and the performance of the ECCS system. Decay heat generated throughout the transient is also conservatively calculated as required by Appendix K of 10 CFR 50.

Small Break LOCA Evaluation Model

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

In NOTRUMP, the RCS is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicitly with the intact loops lumped into a second loop. The transient behavior of the system is determined from the governing conservation equations of mass, energy and momentum applied throughout the system. A detailed description of NOTRUMP is given in References 15.6.2-2 and 15.6.2-3.

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

Cladding thermal analyses are performed with the LOCTA-IV (Reference 15.6.2-4) code which used 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.

The small break analysis was performed with the approved Westinghouse ECCS Small Break Evaluation Model (References 15.6.2-2, 2-3, and 2-4).

Small Break Input Parameters and Initial Conditions

Table 15.6.2-1 lists important input parameters and initial conditions used in the small break analyses. The axial power distribution and core decay power assumed for the small break analyses are shown in Figures 15.6.2-13 and 15.6.2-14.

Safety injection flow to the Reactor Coolant System as a function of the system pressure is used as part of the input. The SI delivery considers pumped injection flow which is depicted in Figure 15.6.2-12 as a function of RCS pressure. This figure represents injection flow from the SI pumps based on H. B. Robinson degraded delivery data. The degraded delivery data incorporates the standard FSAR ECCS assumption of minimum safeguards. The effects of flow from the RHR pumps is not considered here since their shutoff head is lower than RCS pressure during the time portion of the transient considered here.

The Safety Injection system was also assumed to be delivering to the RCS 25 seconds after the generation of a safety injection signal. This delay time includes the time required for diesel startup and loading of the safety injection pumps onto the emergency buses.

The hydraulic analyses are performed with the NOTRUMP code using 102 percent of the licensed core power plus the 8 MWt energy added by the three reactor coolant pumps. The core thermal transient analyses using LOCTA-IV are performed using the 102 percent licensed core power assumption and incorporating Exxon 15x15 fuel data. This fuel data is summarized in Table 15.6.2-2.

Small Break LOCA Results

As noted previously, the calculated peak cladding temperature resulting from a small break LOCA is less than that calculated for a large break. A range of small break analyses is presented which establishes the limiting break size. The results of these analyses are summarized in Tables 15.6.2-3 and 15.6.2-4. Figures 15.6.2-1 through 15.6.2-11 present the principal parameters of interest for the small break ECCS analyses. For the three-inch and four-inch break sizes analyzed, the following transient paraments are included:

a. RCS Pressure
b. Core Mixture Height
c. Hot Spot Clad Temperature

The two-inch break LOCTA results indicate that no clad heat-up occurs which is greater than the initial clad temperature of 719°F. Thus, the minimal core uncovery results in an insignificant clad heat-up and the two-inch case would be bounded by the three-inch PCT. The RCS pressure and core mixture height plots are included for the two-inch case.

For the limiting break size analyzed (three-inch). the following additional transient parameters are presented (Figures 15.6.2-6 through 15.6.2-8):

a. Core Steam Flow Rate
b. Core Heat Transfer Coefficient
c. Hot Spot Fluid Temperature

The maximum calculated peak cladding temperature for the small breaks analyzed is 1398°F. These results are well below all Acceptance Criteria limits of 10 CFR 50.46 and no case is limiting when compared to the results presented for large breaks.

REFERENCES FOR SECTION 15.6.2

1. "Acceptance Criteria for Emergency Core Cooling System for Light Water Cooled Nuclear Power Reactors", 10 CFR 50.46 and Appendix K of 10 CFR 50. Federal Register, Volume 39, Number 3, January 4, 1984.

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

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

4. Bordelon, F. M., et. al., "LOCTA-IV Program: Loss of Coolant Transient Analysis", WCAP-8301, (Proprietary) and WCAP-8305, (Non-Prprietary), June 1974.

TABLE 15.6.2-1

Input Parameters Used in the SBLOCA Analysis

Core Power*	2346 MWt
Pump Heat	8 MWt
NSSS Power	2354 MWt
Peak Linear Power (includes 102% factor)	11.064 kW/ft
Total Peaking Factor, F	2.32
Power Shape	Fig. 15.6.2-13
Fuel Assembly Array	Exxon 15x15
Nominal Accumulator Water Volume	825 ft^3/accum.
Nominal Accumulator Tank Volume	1200 ft^3/accum.
Minimum Accumulator Gas Pressure	615 psia
Pumped Safety Injection Flow	Fig. 15.6.2-12
Steam Generator Initial Pressure	787 psia
Auxiliary Feedwater Flow	41.22 lb/sec/SG
Steam Generator Tube Plugging Level	5%

* 2% is added to this power to account for calorimetric uncertainty

TABLE 15.6.2-2

Fuel Design Parameters

Parameter	Exxon Fuel
Cladding, O.D.	0.424 in.
Cladding, I.D.	0.364 in.
Pellet O.D.	0.3565 in.
Fuel Active Length	144 in.
Fuel Rod Pitch	0.563 in.
Fuel Enrichment	3.34%
Pellet Theoretical Density	95.3%

TABLE 15.6.2-3

Small Break LOCA Time Sequence of Events

	2 in (sec)	3 in (sec)	4 in (sec)
Start	0.0	0.0	0.0
Reactor Trip	14.91	7.32	4.68
S-signal	23.61	13.80	9.54
Top of Core Uncovered	N/A	593.51	480.00
Accumulator Injection	N/A	1090.00	594.00
Peak Clad Temperature Occurs	N/A	1207.00	690.56
Top of Core Covered	N/A	1565.00	880.00

TABLE 15.6.2-4

Small Break LOCA Fuel Cladding Results

	2 in	3 in	4 in
Peak Clad Temperature (^{o}F)	N/A	1398	1115
Peak Clad Location (ft)	N/A	12.0	12.0
Local Zr/H_2O Reaction (max), (%)	N/A	1.23	10E-5
Local Zr/H_2O Reaction Location (ft)	N/A	12.0	10.0
Total Zr/H_2O Reaction (%)	N/A	<0.3	<0.3
Hot Rod Burst Time (sec)	N/A	N/A	N/A
Hot Rod Burst Location (ft)	N/A	N/A	N/A

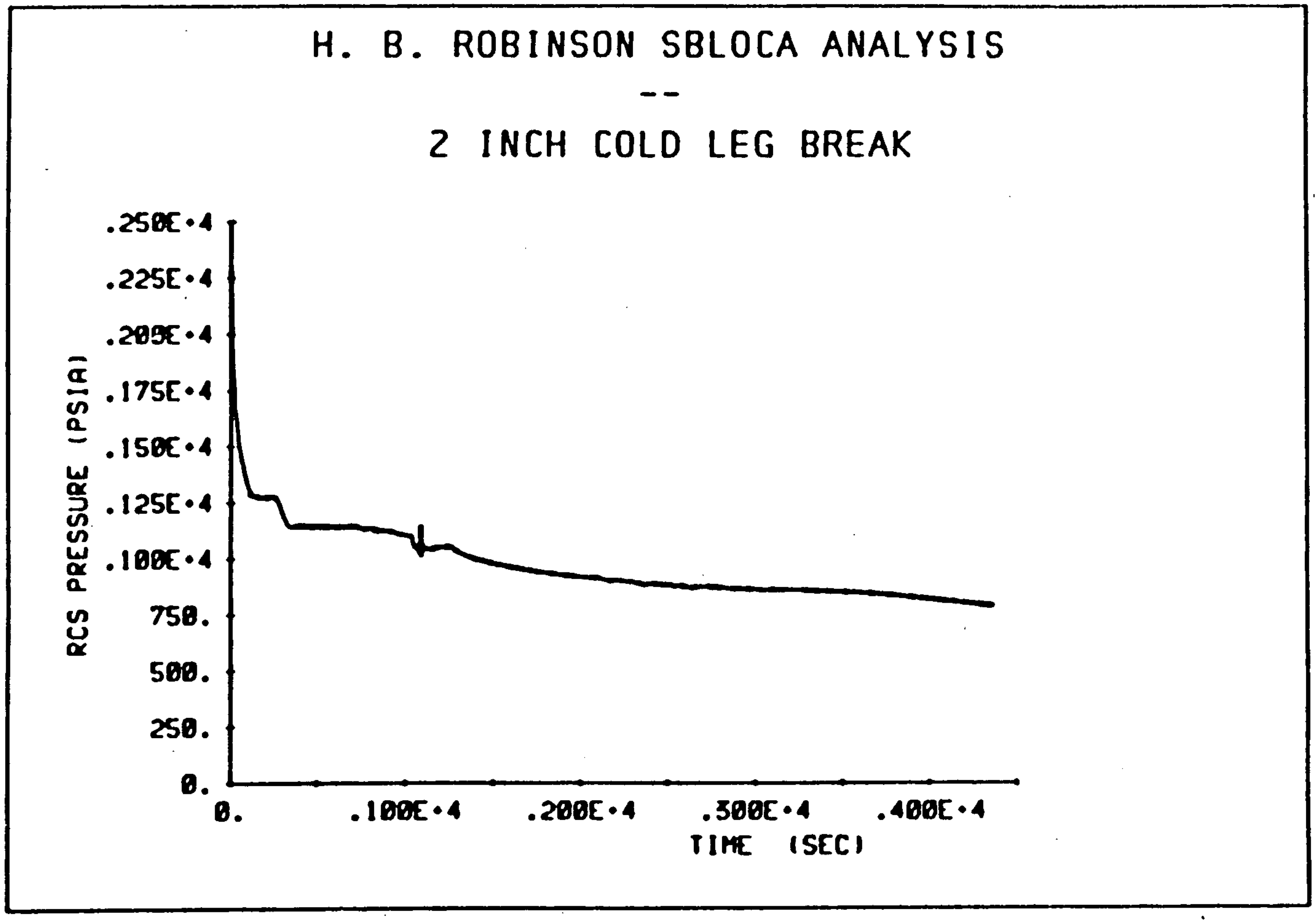

Figure 15.6.2-1 RCS Pressure for 2-inch SBLOCA

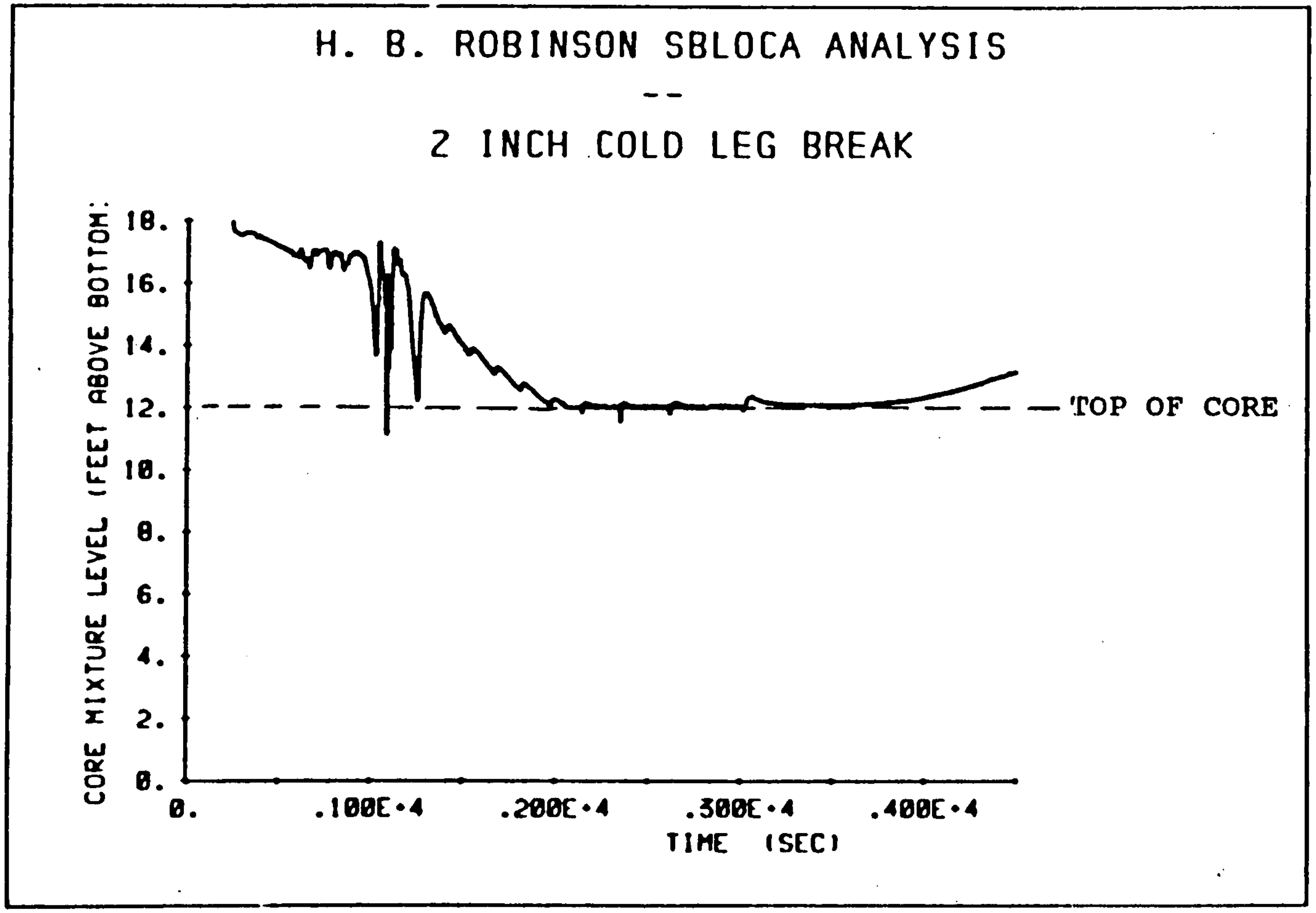

Figure 15.6.2-2 Core Mixture Height for 2-inch SBLOCA

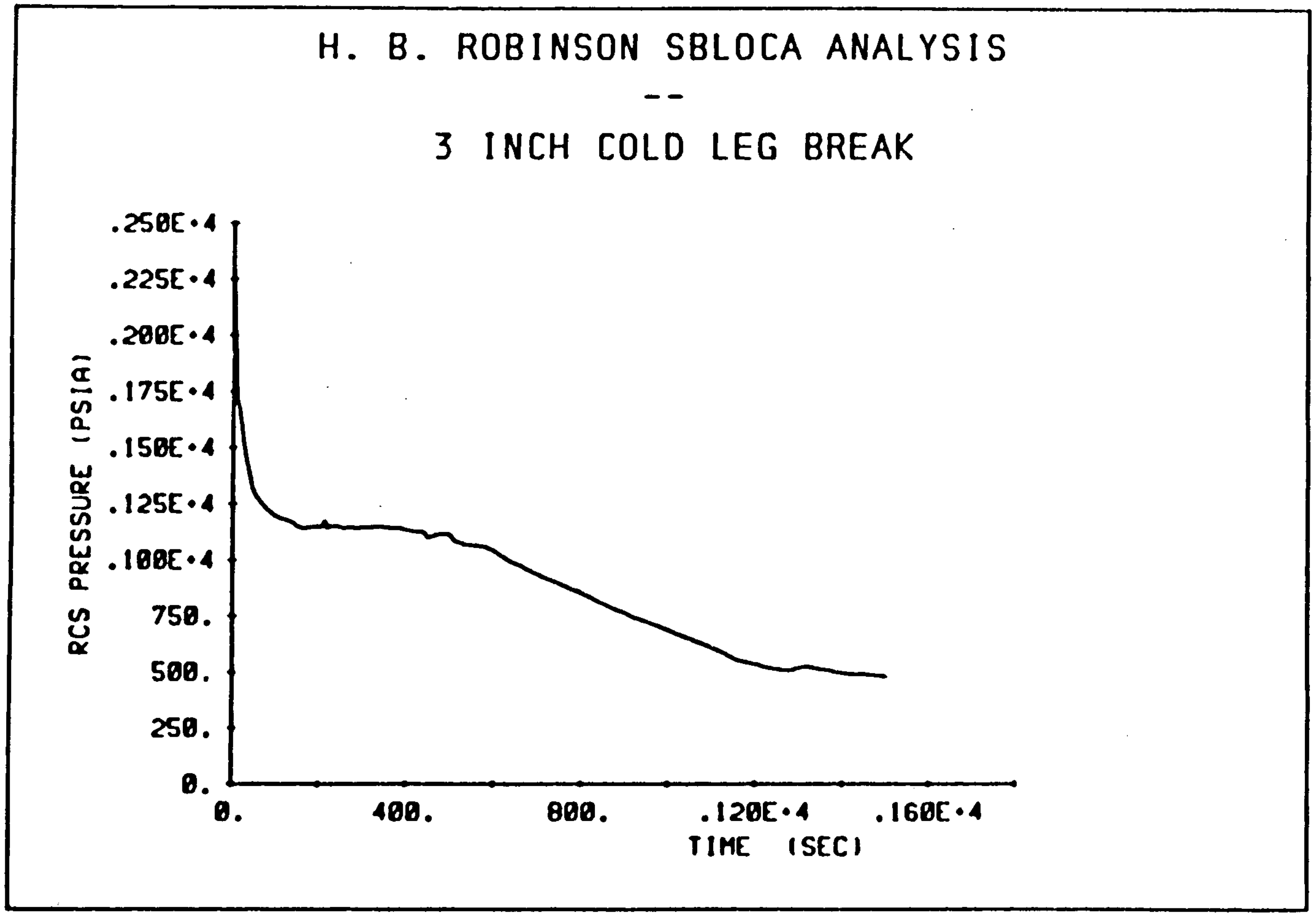

Figure 15.6.2-3 RCS Pressure for 3-inch SBLOCA

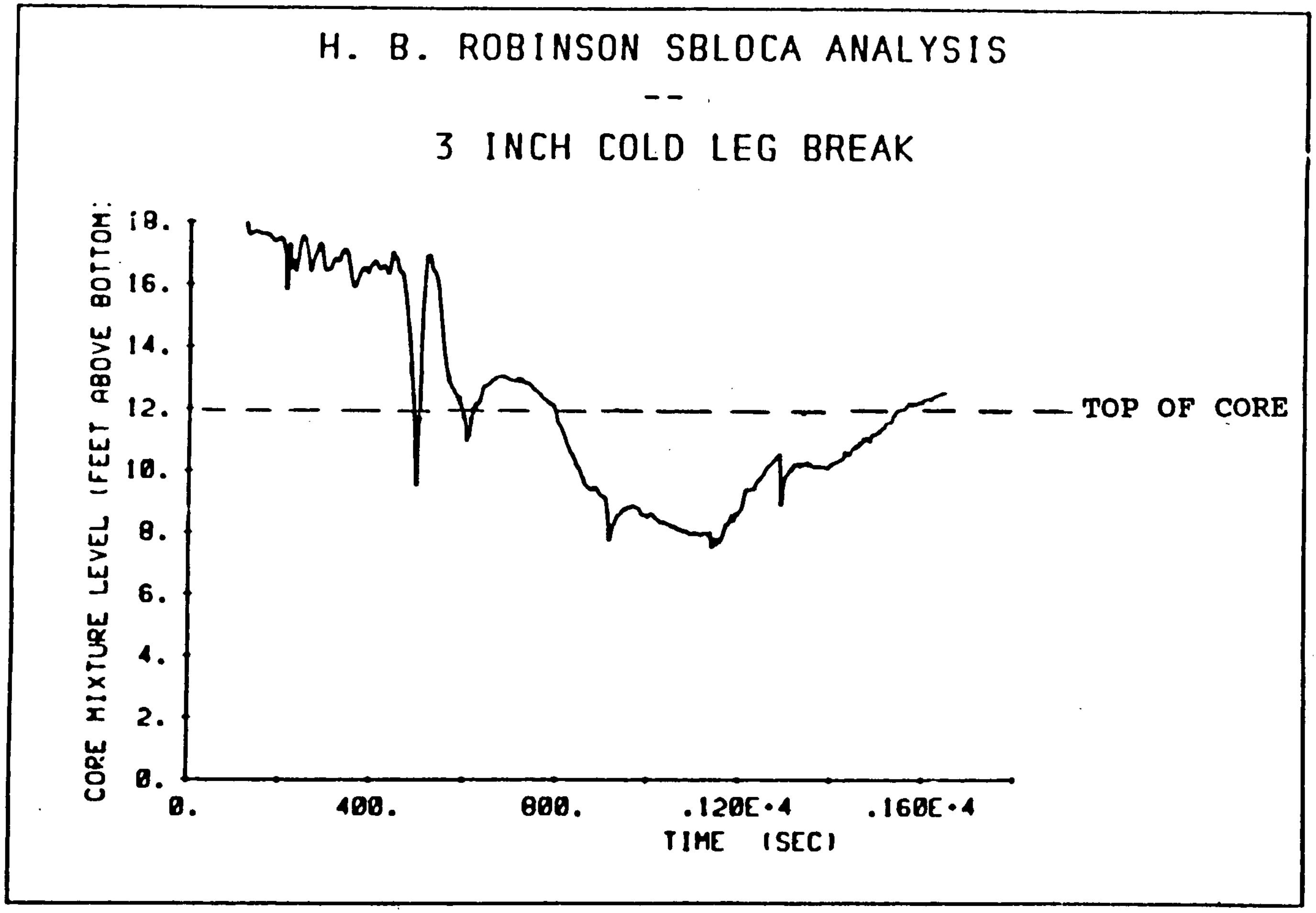

Figure 15.6.2-4 Core Mixture Height for 3-inch SBLOCA

H. B. ROBINSON 3-INCH LOCTA

CLAD AVG. TEMPERATURE HOT ROD, PEAK AT 12 FT.

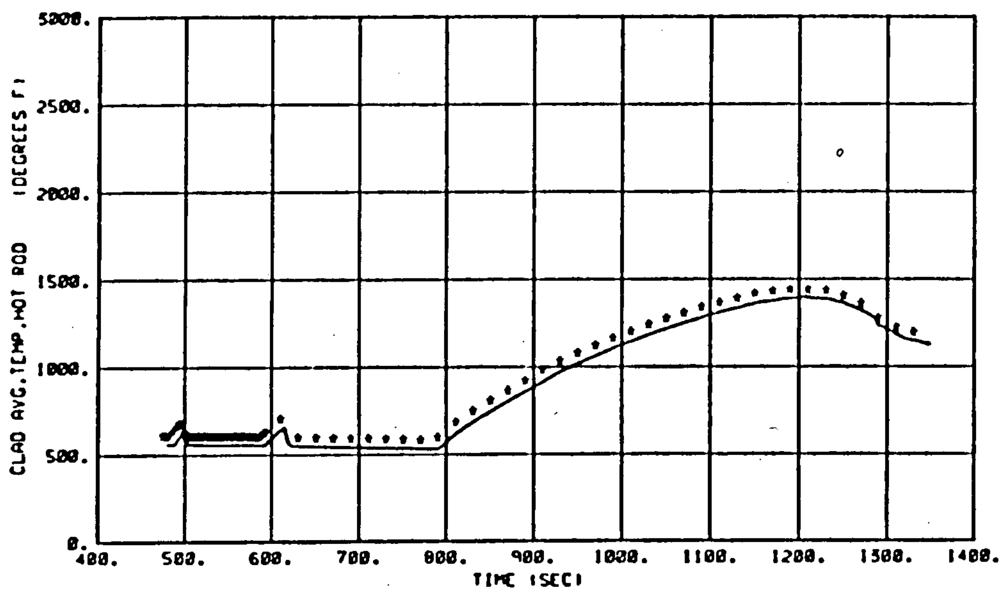

Figure 15.6.2-5 Hot Spot Clad Temperature for 3-inch SBLOCA

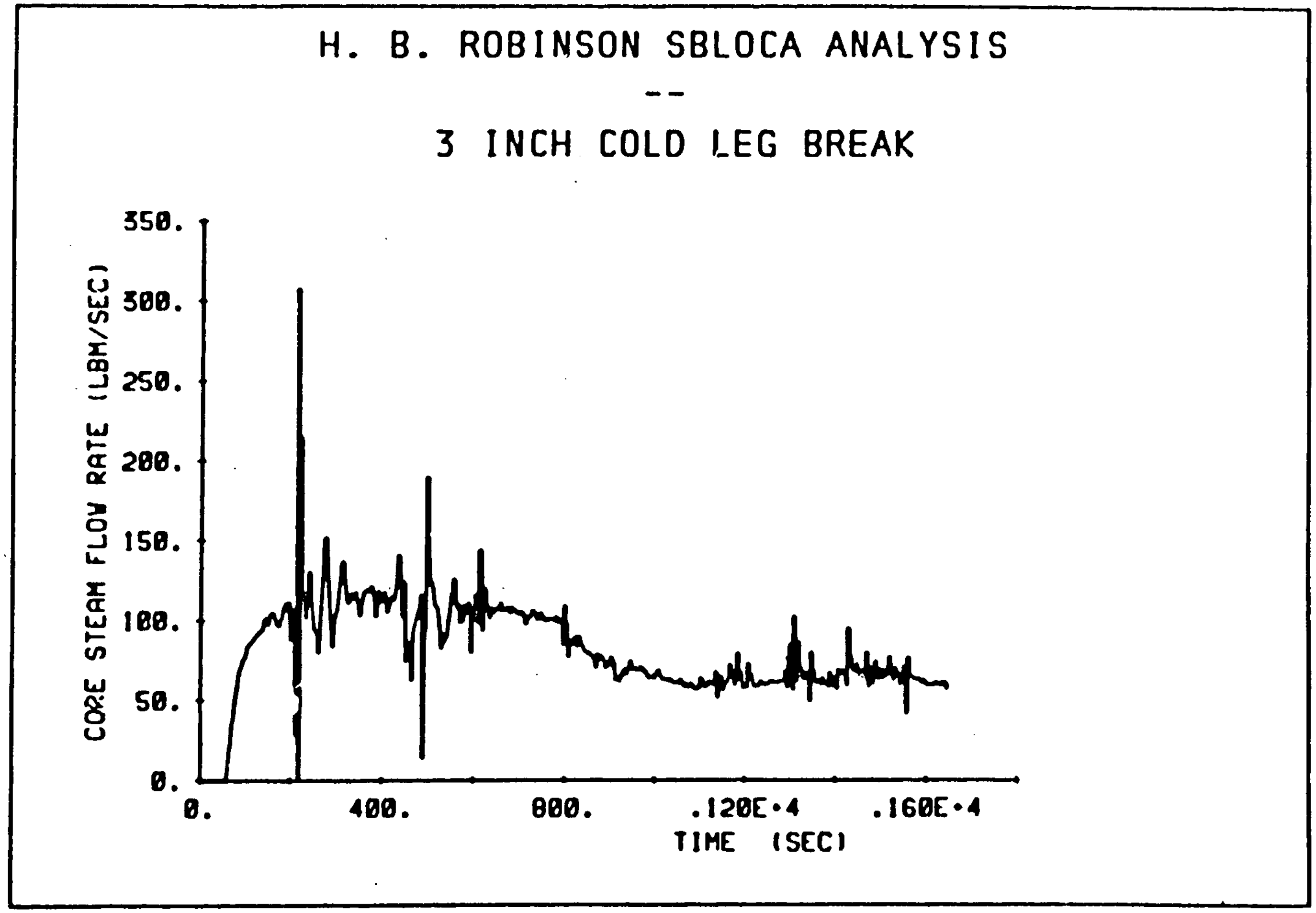

Figure 15.6.2-6 Core Steam Flow Rate for 3-inch SBLOCA

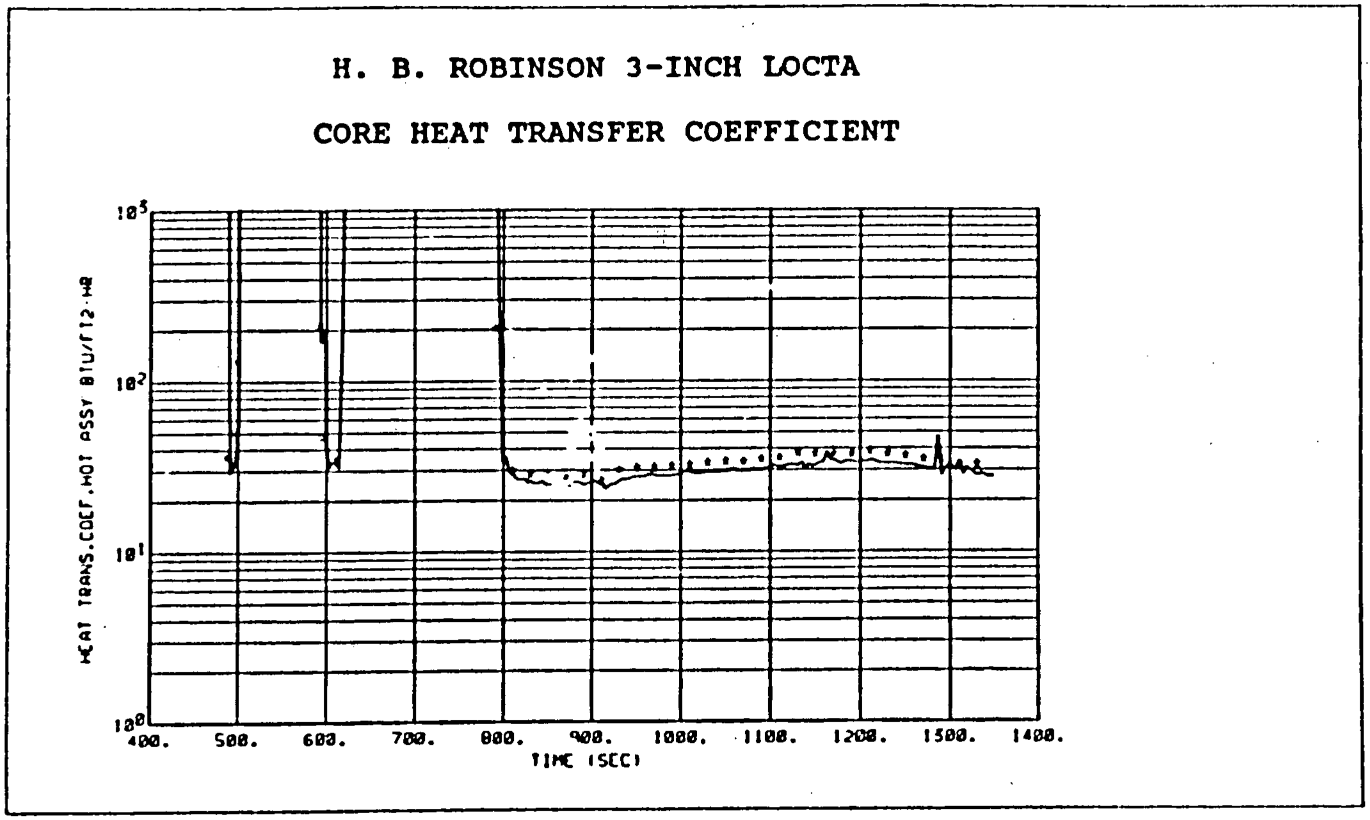

Figure 15.6.2-7 Core Heat Transfer Coefficient for 3-inch SBLOCA

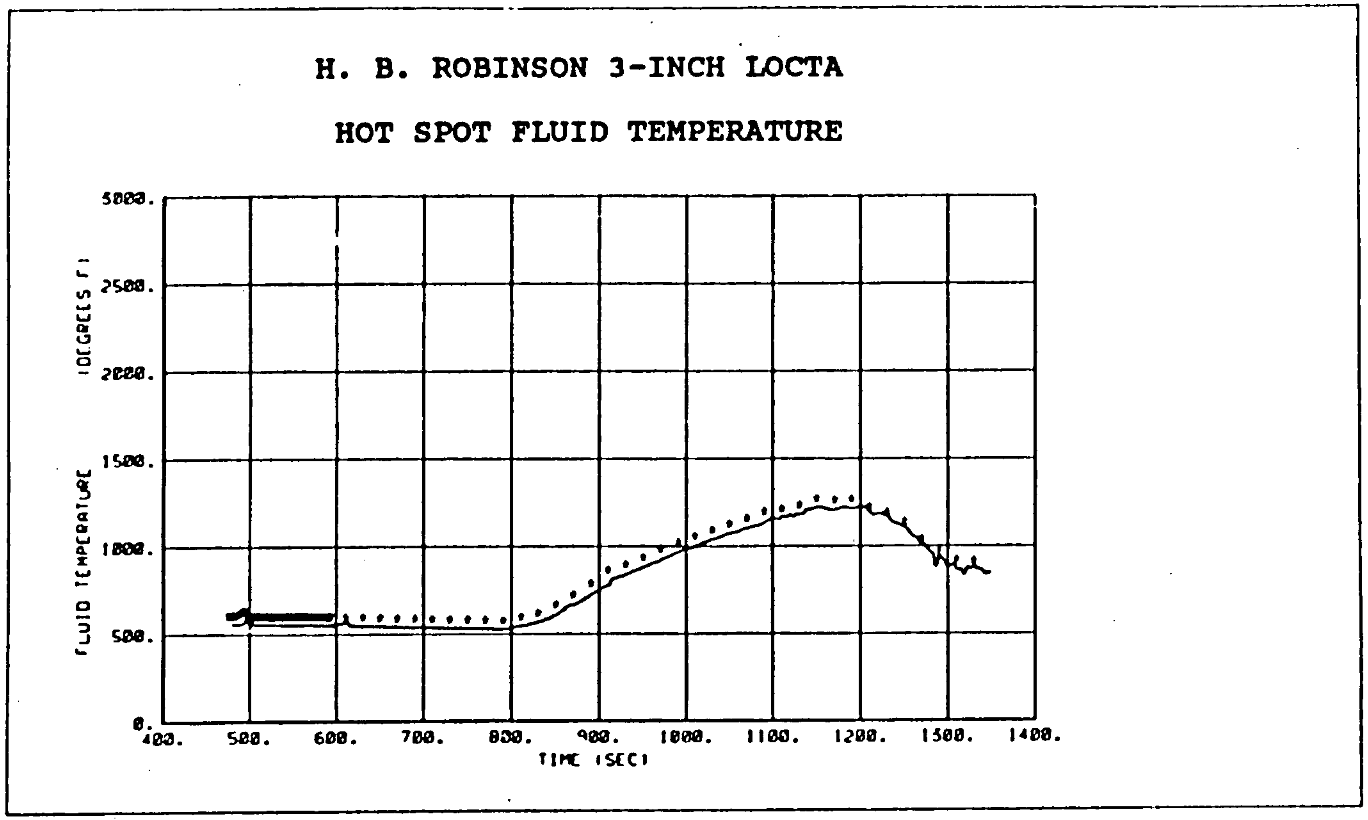

Figure 15.6.2-8 Hot Spot Fluid Temperature for 3-inch SBLOCA

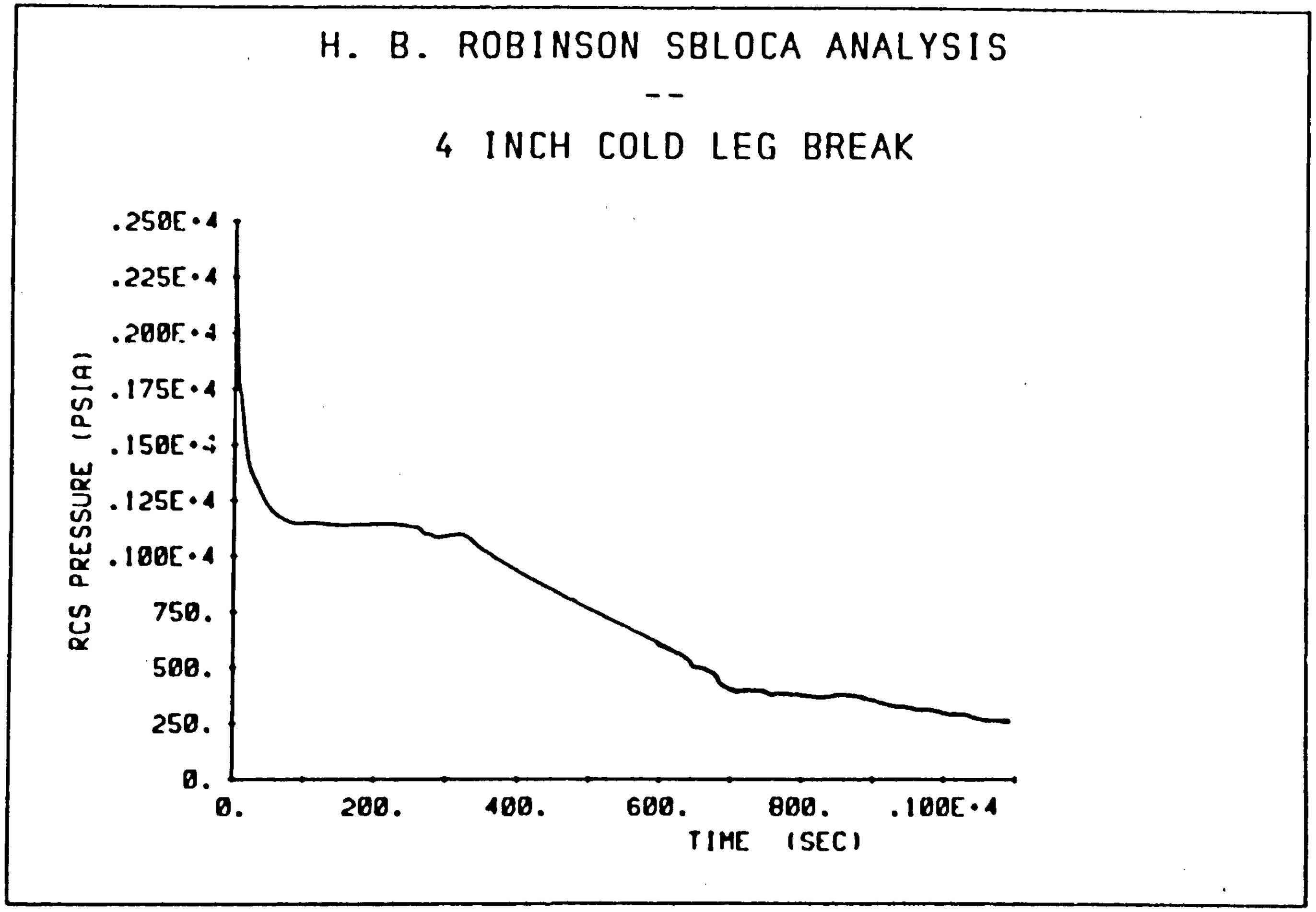

Figure 15.6.2-9 RCS Pressure for 4-inch SBLCOA

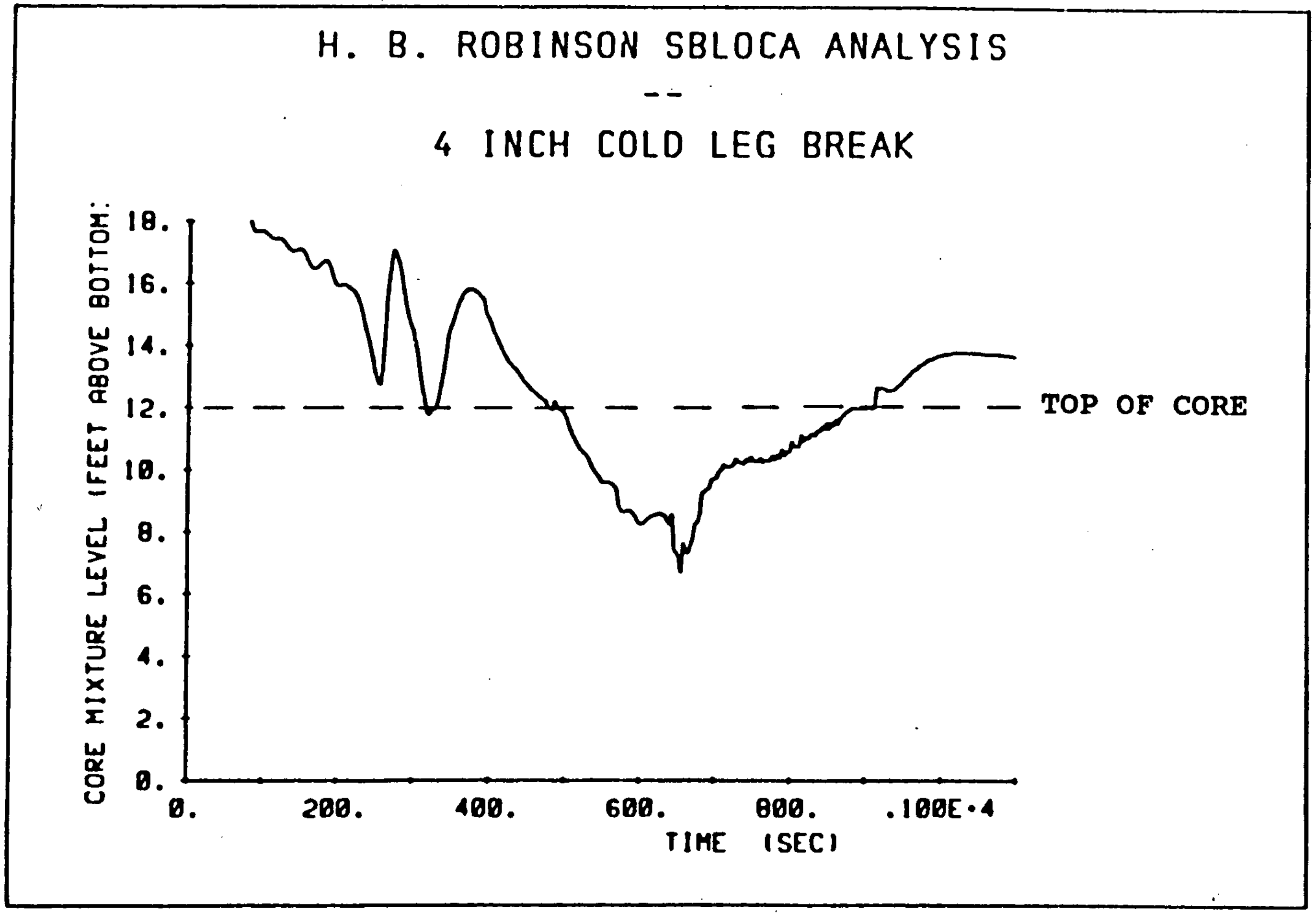

Figure 15.6.2-10 Core Mixture Height for 4-inch SBLOCA

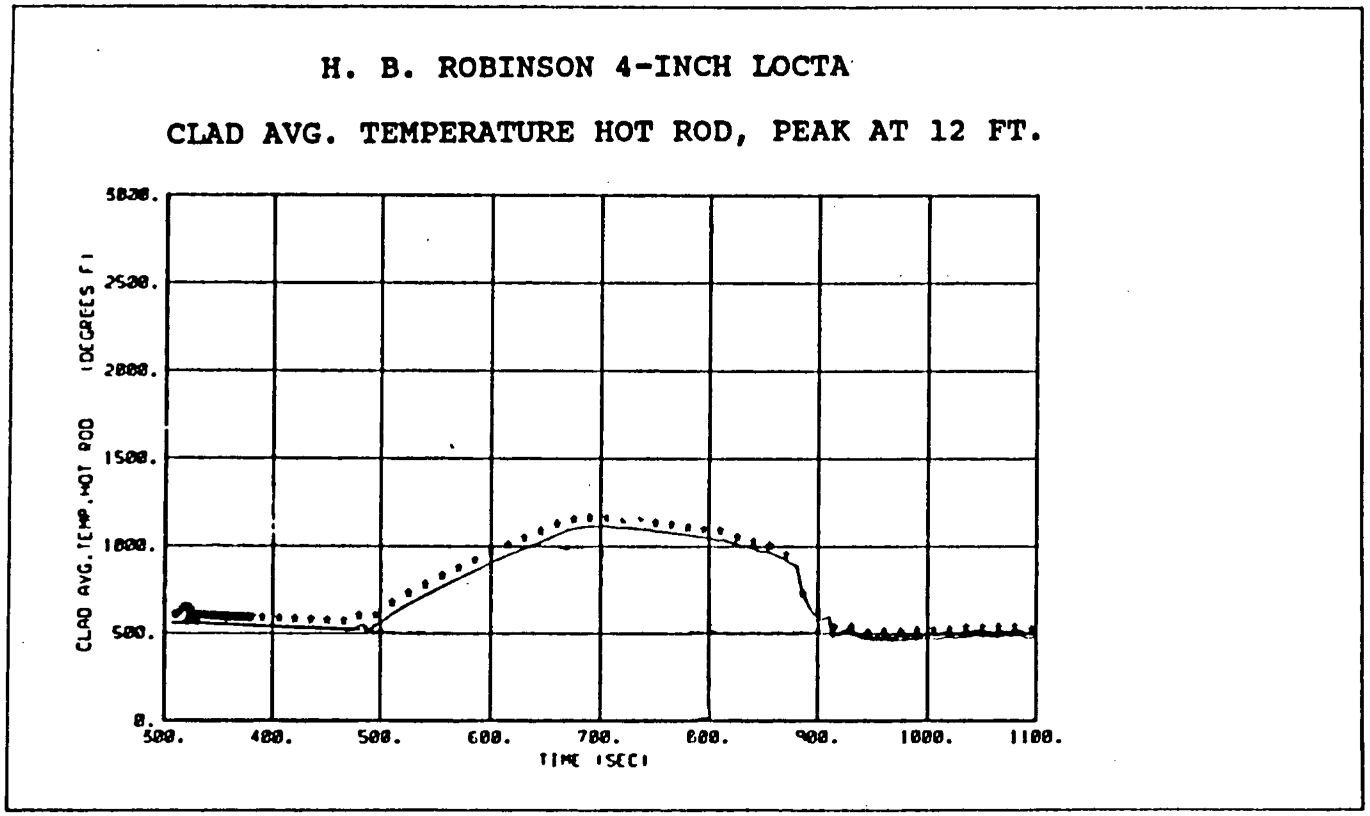

Figure 15.6.2-11 Hot Spot Clad Temperature for 4-inch SBLOCA

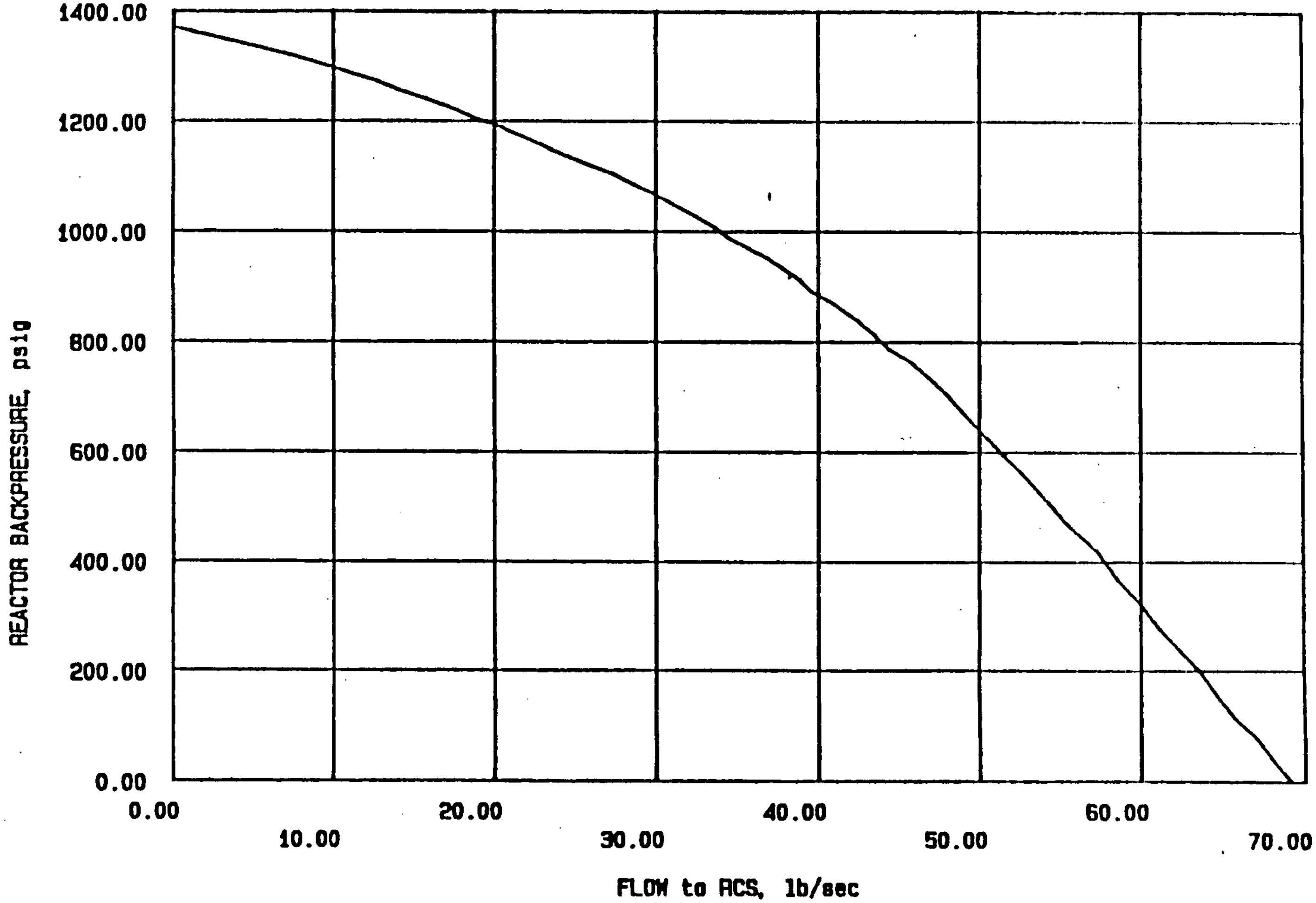

Figure 15.6.2-12 H. B. Robinson Pumped Safety Injection Flow

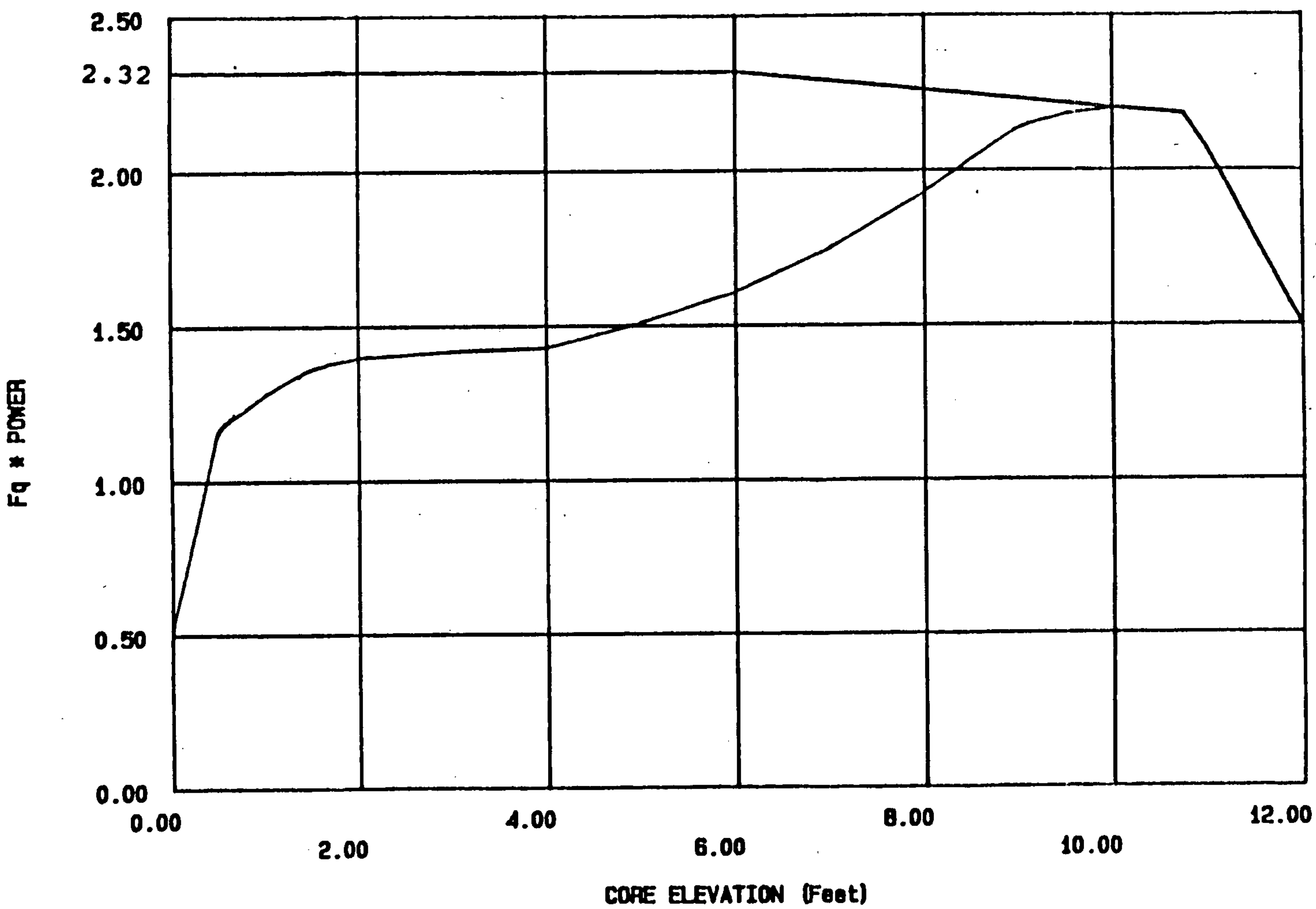

Figure 15.6.2-13 H. B. Robinson Small Break LOCA Power Shape

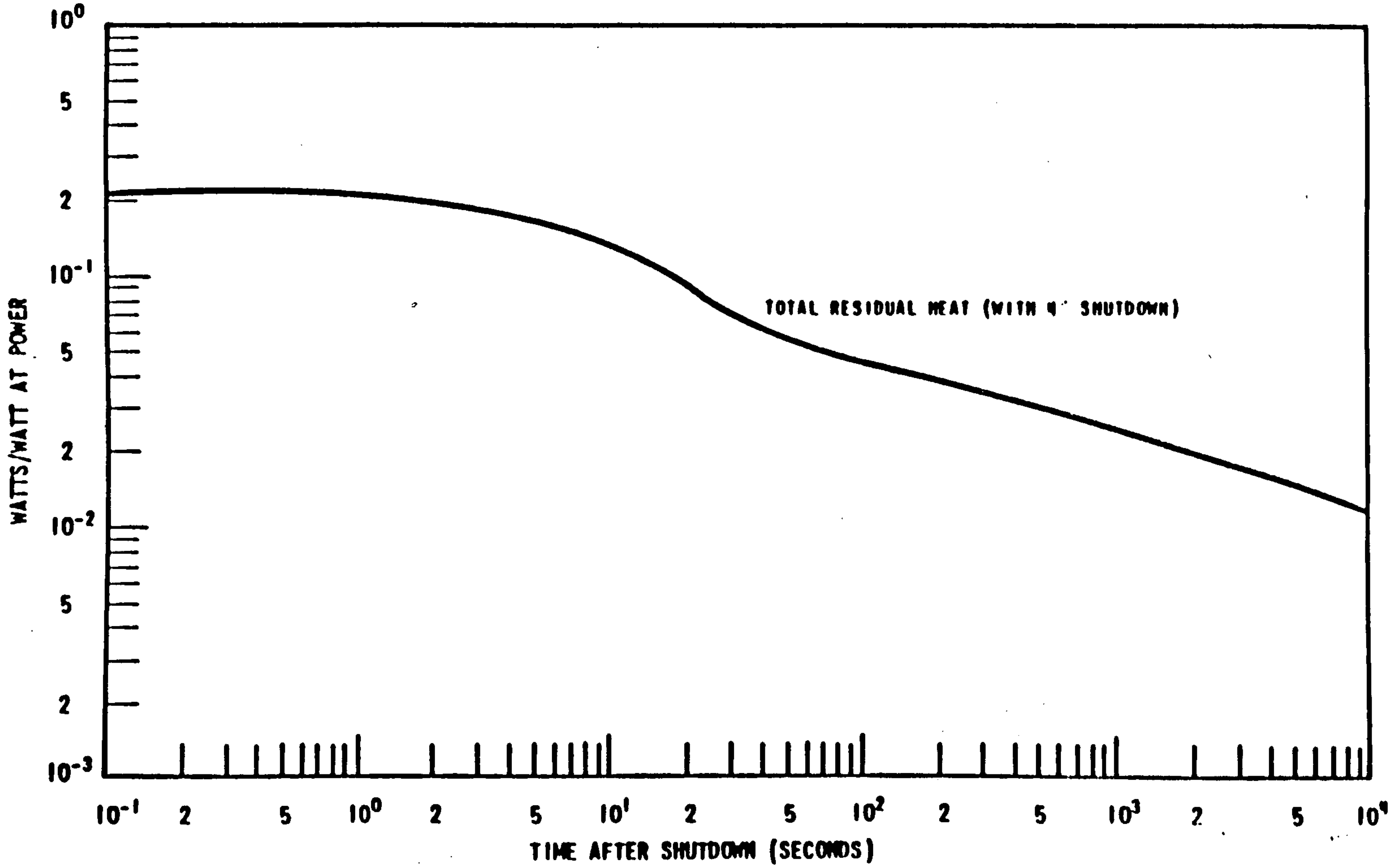

Figure 15.6.2-14 Core Power After Reactor Trips