ATTACHMENT A

CHANGED TECHNICAL SPECIFICATION PAGES

b. Each accumulator is pressurized to at least 600 psig and contains at least 825 ft^3 and no more than 841 ft^3 of water with a boron concentration of at least 1950 ppm. No accumulator may be isolated.

c. Three safety injections pumps are operable.*

d. Two residual heat removal pumps are operable.

e. Two residual heat exchangers are operable.

f. All essential features including valves, interlocks, and piping associated with the above components are operable.

g. During conditions of operation with reactor coolant pressure in excess of 1000 psig the A.C. control power shall be removed from the following motor operated valves with the valve in the specified position:

Valves	Position
MOV 862 A&B	Open
MOV 864 A&B	Open
MOV 865 A,B,&C	Open
MOV 878 A&B	Open
MOV 863 A&B	Closed
MOV 866 A&B	Closed

h. During conditions of operation with reactor coolant pressure in excess of 1000 psig, the air supply to air operated valves 605 and 758 shall be shut off with valves in the closed position.

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* With only two safety injection pumps operable, each capable of automatic initiation from a separate emergency bus, the reactor may be made critical; however, steady state reactor core power level shall not exceed 1380 megawatts thermal.

i. Power operation with less than three loops in service is prohibited.

3.3.1.2 During power operation, the requirements of 3.3.1.1 may be modified to allow any one of the following components to be inoperable. If the system is not restored to meet the requirements of 3.3.1.1 within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.1.1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

a. One accumulator may be isolated for a period not to exceed four hours.

b. If one safety injection pump becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours, provided the remaining two safety injection pumps are demonstrated to be operable prior to initiating repairs.*

c. If one residual heat removal pump becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours, provided the other residual heat removal pump is demonstrated to be operable prior to initiating repairs.

---

* For reactor core power levels up to and including 1380 megawatts thermal, Specification 3.3.1.2.b shall be modified to read: "If one of the two automatically initiated safety injection pumps becomes inoperable, the reactor may remain in operation for a period not to exceed 24 hours, provided the remaining automatically initiated safety injection pump is demonstrated to be operable prior to initiating repairs."

d. If one residual heat exchanger becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours.

e. If any one flow path including valves of the safety injection or residual heat removal system is found to be inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours, provided the other flow path(s) are demonstrated to be operable prior to initiating repairs. The hot leg injection paths of the Safety Injection System, including valves, are not subject to the requirements of this specification.

f. Power or air supply may be restored to any valve referenced in 3.3.1.1.g. and 3.3.1.1.h. for the purpose of valve testing or maintenance providing no more than one valve has power restored and provided that testing and maintenance is completed and power removed within 24 hours except for accumulator isolation valves (MOV 865 A,B,&C) which will have this time period limited to four hours.

ATTACHMENT B

SAFETY ANALYSIS

(5382JSK/crs)

SAFETY ANALYSIS

Purpose

This Safety Analysis is intended to determine whether operation at 60% of rated power with only two SI pumps capable of automatic initiation from separate emergency buses constitutes either an Unreviewed Safety Question or a Significant Hazard. Without taking credit for operator action to start the third pump, necessary consideration of a single active failure would leave only one of three SI pumps delivering flow under accident conditions.

Historical Information

The current licensing basis for H. B. Robinson Unit 2 takes credit for two of three Safety Injection (SI) pumps in mitigating accidental depressurization of the RCS. With concurrent Loss of Offsite Power, SI pump "A" is automatically powered by "A" Emergency Diesel Generator, and pump "C" is automatically loaded onto "B" Emergency Diesel Generator. SI pump "B" is the "swing" pump in that while it is automatically loaded onto "A" diesel generator, if that emergency power bus is not energized, it is provided with an automatic start feature to switch to the "B" diesel generator without operator action.

On January 28, 1988, after reviewing plant design documentation for formulating a response to an NRC letter sent to CP&L on January 14, 1988, requesting additional information regarding the design of the H. B Robinson Unit 2 electical distribution system, the plant staff discovered that at least one postulated single failure existed which could delay actuation of "B" SI pump by 10 seconds. In a meeting at the NRC offices on February 10, 1988, CP&L discussed the vulnerability of the on-site emergency electrical distribution system to single failures under certain conditions.

This license change is necessitated by the identification on of a type of scenario which would render two of the three safety injection pumps vulnerable to a single failure (e.g., malfunction of a diesel generator voltage regulator or other diesel generator voltage or frequency malfunctions).

Historical Information (continued)

In a meeting with CP&L in the NRC offices on February 16, 1988, disconnecting the automatic start feature of SI pump B was discussed as a possible solution to this unresolved scenario. This is effectively opening the power supply breakers to this swing pump so that it is to be loaded onto the diesel generator of choice by manual action instead of automatically. This design change is plant mod 951. The main point of deleting the automatic start feature is that it avoids the chance of damaging two SI pump motors as the result of undetected irregularities in one emergency power bus.

While earlier versions of this license change focused on

- reducing Fq as compensation for reduced SI flow in the reanalysis of Large Break LOCA, and
- taking credit for operator action to verify proper SI operation and to manually start SI pump B upon failure of pump A or C in the reanalysis of Small Break LOCA,

subsequent refinement of the possible time of occurance of the single active failure in the Small Break LOCA resulted in an allowable operator response time that could not be addressed without additional review. As a result, another reanalysis of Small Break LOCA was based on operation of only one SI pump. This Small Break LOCA analysis with no credit for operator action represents the current limiting case and requires reduction to 60% power (1380 MWt) as a necessary condition to calculate acceptable consequences. The different approach of reducing power instead of the core peaking factor is due to both the magnitude of the restriction (reducing Fq from 2.32 to 2.26 may or may not have required power reduction) and also the increased importance of decay heat in Small Break LOCA analysis relative to Large Break analysis.

SAFETY ANALYSIS

Body of Evaluation

The response to the first three questions in Section "B" of Attachment 6.2 is provided in two parts. Reduced SI flow is addressed first, followed by reduced power.

I. regarding reduced automatically actuated SI flow

1.a. This does not increase the probability of any analyzed accident because the SI system serves a mitigating function only after event initiation. The probability of initiating any FSAR Chapter 15 accident is unaffected by the availability of the SI pumps.

1.b. This does not increase the consequences of any analyzed accident because:

i.) A review of the accidents involving SI (Attachment 1) shows that only Large Break and Small Break LOCA's require formal reanalysis.

ii.) As documented in Westinghouse letter CPL-88-510, dated 2-15-88, Westinghouse has reanalyzed Large Break LOCA based on operation of one of three SI pumps and one of two RHR pumps. Their calculations show that a slight reduction in total power peaking in the core (i.e., changing the value of Fq from 2.32 to 2.26) is sufficient compensation for the reduced SI flow.

iii.) As documented in Westinghouse letter CPL-88-513, dated 2-23-88, Westinghouse has reanalyzed Small Break LOCA based on operation of one of three SI pumps. Their calculations show that restricting core power to 60% is sufficient compensation for the reduced SI flow.

2. This does not introduce the possibility of a new or different kind of accident from any previously evaluated because (as explained in "1.a.") SI is soley a mitigating action in response to previously analyzed accident scenarios.

I. regarding reduced automatically actuated SI flow (continued)

3. This does not reduce the margin of safety because (as explained in "1.b.") LOCA reanalysis by Westinghouse show a slightly reduced Fq for Large Break and a 60% power level for Small Break to be adequate compensation for the reduced SI flow. Westinghouse used NRC approved analysis methods to show compliance with the 10CFR50.46 acceptance criteria:

- a calculated peak fuel clad temperature below 2200 degrees F
- less than 1% of the total amount of Zircaloy in the core, reacts chemically with water or steam,
- less than 17% of cladding is oxidized at any location, and
- the core geometry remains amenable to long term cooling.

SAFETY ANALYSIS

II. **No other changes to Limiting Conditions for Operation, setpoints, or operating parameters in the Technical Specification are required for this SI configuration if power is administratively limited to 60%.**

1.a. The probability of any analyzed accident is not increased because continued operation at 60% load is well within the design capability of the NSSS.

1.b. The consequences of any analyzed accident is not increased because

i.) Tech. Spec.'s are designed to cover all modes of plant operation or shutdown. As such, there are automatic provisions for the natural transition from startup to full power operation. Operation at reduced power simply relies on formulas and conditions dependent on core power that are already built into the Tech. Spec.'s.

For example, with the additional restraint of holding core power to 60%, all the safety limit restrictions of Tech. Spec. section 2.1 and Figure 2.1-1 still apply. Although Figure 2.1-1 still presents information about full power operation, operation in the region represented in the right half of the figure will be administrativey prohibited. This is not much different than currently relying on operator action to avoid actuation of turbine runback or trip on high power.

ii.) For non-LOCA's, the restricted power level either

- is not applicable (e.g., accidents conservatively initiated from zero power)

- is consistent with the current initial condition (e.g., Uncontrolled Rod Bank Withdrawl at Power)

or

- reduces the severity of the accident by reducing the initial temperature of fuel, cladding, and coolant as well as reducing decay heat.

In other words, since Attachment 1 shows that the reduced SI pump operability does nothing to aggravate the approach to non-LOCA safety limits such as DNB, fuel centerline melting, and RCS overpressurization; restricting power while keeping the same Over Temperature delta T, Over Power delta T, High Flux, Low RCS Flow, and other trip setpoints will either maintain or increase the margin of safety in these respects.

2. The possibility of an unanalyzed accident is not introduced because

   i.) Technical Specifications and the FSAR Chapter 15 accident analyses already cover operation at reduced power.

   ii.) Although without resetting the High Flux trip to ~60% power, there is a remote possibility that an operator error (or a Condition II event as outlined in FSAR section 15.0.1) may briefly violate the administrative limit on power. However, this in itself would not cause a LOCA and postulation of multiple, independent, simultaneous or coincident accidents is not required.

3. The margin of safety is not reduced because

   i.) As explained in "1.b.", restricting core power can only maintain or increase the margin to non-LOCA safety limits.

   ii.) As explained in Part I., the Westinghouse reanalysis of Small Break LOCA shows that restricting power to 60% is sufficient to compensate for reduced SI flow. This Small Break reanalysis correctly accommodates the increased allowable Fq at 60% power in accordance with Tech. Spec. section 3.10.2.2.

While the allowable Fq of 3.87 at 60% power allows a slightly higher linear heat generation rate than was used in the corresponding Large Break analysis, i.e.,

$$\left[\frac{\text{rated thermal power produced in the fuel}}{\text{combined length of all fuel pellet stacks in the core}} = 6 \frac{\text{KW}}{\text{ft.}} \right] (Fq = 2.26) = 13.56 \text{ KW/ft.}$$

peak linear power in Large Break LOCA analysis

vs.

$$(6 \text{ KW/ft.}) \times (60\% \text{ power}) \times \left[Fq = \frac{2.32}{0.6} \right] = 13.92 \text{ KW/ft.}$$

allowable at 60% power.

However, in qualitative comparison to full power conditions; the lower fuel temperatures, lower decay heat, and lower amount of stored energy in the core at 60% power more than compensate for this slight non-conservatism.

SAFETY ANALYSIS

Required Action

1. Other items included in the "Confirmation of Action" letter (NRC-88-066), dated 2-11-88.

2. If this condition (i.e., only two of three SI pumps being automatically actuated) continues to the end of the year, it should be incorporated (at least) in FSAR sections 6.3, 8.3, 15.6.2, and 15.6.5.

Conclusion

This license change constitutes neither an Unreviewed Safety Question nor a Significant Hazard.

SAFETY ANALYSIS

ATTACHMENT 1

Review of FSAR Chapter 15 Events Involving Safety Injection

From FSAR Table 15.0.9-1, the events that may have the potential for actuation of Safety Injection are:

- Inadvertent Opening of a Steam Generator PORV or Safety Valve
- Main Steamline Break
- Feedwater Line Break
- Inadvertent Operation of the ECCS
- Steam Generator Tube Rupture
- Small Break LOCA

and - Large Break LOCA

FSAR section 15.1.4 explains how the consequences of Inadvertent Opening of a Steam Generator PORV or Safety Valve is bounded by Main Steamline Break as a similiar, but more severe event.

Although Main Steamline Break is classified as a Postulated Accident, analysis from XN-NF-85-17(P) presented in FSAR section 15.1.5 shows how DNB is avoided. Because continued RCP flow is necessary for the rapid cooldown of the RCS, offsite power is considered to be availible throughout this event. The current analysis is based on loss of one of three SI pumps as the worst single failure. With no credit for short term operator action, this plant modification will change the limiting fault to automatic actuation of one of two SI pumps. The effect of this change is to introduce additional delay in delivering borated water to the core. From FSAR Table 15.1.5-4 and Figure 15.1.5-14, although it is delivery of the borated water at 143 seconds that gradually reduces the power level to approximately 20% at 240 seconds, it is temperature reactivity feedback that limits the maximum power level to 34% at 128 seconds. In other words, Doppler reactivity limits the extent of the power increase well before boration acts to slowly shut the reactor down. Even if delivery of SI from a single pump doubles the transport time delay from 124 to 248 seconds, the effect would be to merely extend the time period during which the reactor is fairly stable between approximately 30% and 40% power. With continued RCP flow and with Figure 15.1.5-12 showing constant RCP pressure after 100 seconds, core power will control any approach to DNB. Since Doppler limits the peak power level, there is no reason that extending the time duration of the return to power would significantly lower the calculated DNB ratio. Moreover, with a calculated minimum DNBR of 1.869 vs. the limit of 1.135 for the Modified Barnett correlation, there is considerable margin with respect to the conclusion of no fuel failure stated in the FSAR.

SAFETY ANALYSIS

ATTACHMENT 1

Review of FSAR Chapter 15 Events Involving Safety Injection
(continued)

As explained in FSAR section 15.2.8, the effects of a Feedwater Line Break are bounded by the Main Steamline Break accident: instead of heating up the RCS, the location of the feedwater sparger or feedring within the Steam Generator makes Feedwater Line Break more like an overcooling event.

For Inadvertent Operation of the ECCS in FSAR Section 15.5.1, this plant modification mitigates the severity of the accident by decreasing the SI flow resulting from either spurious automatic actuation or a single operator error.

For Steam Generator Tube Rupture, FSAR section 15.6.3 explains that (except for radiological consequences) this event is bounded by Inadvertent Opening of the Pressurizer PORV or Safety Valve (in section 15.6.1). For both events, the short term effects with regard to the possibility of fuel failure are evaluated without taking credit for SI. As such, there is nothing in the analysis to change as a result of this plant modification.

ATTACHMENT C

WESTINGHOUSE LETTER

Westinghouse Electric Corporation
Power Systems

Nuclear Technology Systems Division

Box 355
Pittsburgh Pennsylvania 15230-0355

CPL-88-513

February 23, 1988

NS-OPLS-OPL-II-88-114

Ref: 1) CPL-86-531

Mr. S. R. Zimmerman, Manager
Nuclear Fuel
Carolina Power & Light Company
P. O. Box 1551
Raleigh, NC 27602

ATTENTION: T. Clements

CAROLINA POWER & LIGHT COMPANY
H. B. ROBINSON
Small Break LOCA at Reduced Power

Dear Mr. Zimmerman:

Attached is a safety evaluation that examines the effects of having only one high head safety injection (HHSI) pump providing pumped ECCS flow during a small break loss-of-coolant accident (LOCA) for the H. B. Robinson nuclear power plant. The safety evaluation reports the results of the Westinghouse small break LOCA emergency core cooling system (ECCS) evaluation model analyses performed at 60 percent of licensed core power for H. B. Robinson Unit 2 including the representation of Advanced Nuclear Fuels Corporation 15x15 fuel parameters.

Three small break LOCA analyses were performed using the NRC approved Westinghouse small break LOCA ECCS Evaluation Model incorporating the NOTRUMP analysis methodology. The analyses assumed a core power level corresponding to 102% of 1380 MWth (60% power operation).

These analyses were for 1.0-inch, 1.5-inch, and 2-inch equivalent diameter breaks. Of these three 60% power cases, the 2-inch case had the highest peak cladding temperature of 965.4°F.

The analysis of a 3-inch equivalent diameter break in the cold leg at 100 percent power (102% of 2300 Mwt) was also performed with only one HHSI pump available which resulted in a peak cladding temperature of 1771.6°F at the 12.0-foot elevation.

Small break LOCA analyses performed previously for H. B. Robinson (documented in reference 1) at 100% power with two HHSI pumps available resulted in the 3-inch case having the highest peak cladding temperature of the 2-inch, 3-inch, and 4-inch breaks. Lowering the safety injection flow by taking credit for only one HHSI pump lowers the limiting break size. Therefore, the limiting break size for 100% power with only one HHSI available will be less than or equal to 3 inches.

The peak cladding temperature for the 3-inch case at 60% power will be lower than the peak cladding temperature for the 3-inch case at 100% power. Because the 3-inch case at 100% power did not exceed the limits of 10CFR50.46, the 3-inch case at 60% power will not exceed the limits. Therefore, the 3-inch case was not performed for the 60% power level.

The results of the analyses and evaluations show that the H.B.Robinson Unit 2 Nuclear Power Plant may be started and operated at 60 percent of licensed core power in compliance with the requirements of 10CFR50.46 when flow from only one high head safety injection pump is available.

Any questions regarding this evaluation should be directed to the undersigned.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

DL Cecchett - for
G. O. Percival, Manager
Carolina Area

D.L. Cecchett/cgl
Attachment

cc: L. H. Martin (CP&L) 1L, 1A
T. M. Dresser (CP&L) 1L, 1A
B. G. Rieck (CP&L - HBR) 1L, 1A
B. M. Slone (CP&L -HBR) 1L, 1A
R. J. Muth (CP&L - HBR) 1L, 1A
R. S. Pollock (W - Raleigh) 1L, 1A

ATTACHMENT A

JUSTIFICATION FOR STARTUP AND 60% POWER OPERATION OF H. B. ROBINSON UNIT 2 NUCLEAR POWER PLANT WITH 15 X 15 ADVANCED NUCLEAR FUELS COMPANY FUEL IN CONFORMANCE WITH THE 10CFR50.46 ACCEPTANCE CRITERIA

Westinghouse Electric Corporation
Nuclear Technology Systems Division
Nuclear Safety Department
Safeguards Engineering and Development

February 1988

ATTACHMENT A

JUSTIFICATION FOR STARTUP AND 60% POWER OPERATION WITH
15 X 15 ADVANCED NUCLEAR FUELS COMPANY FUEL
IN CONFORMANCE WITH THE 10CFR50.46 ACCEPTANCE CRITERIA
FOR THE
H. B. ROBINSON UNIT 2 NUCLEAR POWER PLANT

I. BACKGROUND

In the process of reviewing plant documents for formulating a response to NRC letter NRC-88-017, it was discovered that at least one postulated single failure event exists which could result in the loss of the ability to automatically start two high head safety injection pumps. Upon thorough review and examination of the problem, failure events were postulated in which flow from only one high head safety injection pump would be available during a loss-of-coolant-accident (LOCA).

A small break LOCA analysis was performed in 1986 for H.B.Robinson using the NRC-approved Westinghouse small break LOCA ECCS Evaluation Model incorporating the NOTRUMP analysis methodology. The spectrum of 2-inch, 3-inch, and 4-inch equivalent diameter cold leg small break analyses resulted in the highest calculated peak cladding temperature of 1398°F for the 3-inch break. The analysis was performed assuming a core power level corresponding to 102 % of 2300 MWth at a total core peaking factor (FQT) of 2.32 with a hot channel enthalpy rise factor of 1.65. The analysis assumed flow was delivered automatically from two high head safety injection pumps.

A safety evaluation to justify the resumption of operation of the H.B.Robinson Unit 2 nuclear power plant to a maximun of 60% power with 15x15 fuel manufactured by the Advanced Nuclear Fuels Corporation was performed assuming only one high head safety injection pump was operational. The evaluation was based upon small break LOCA analyses using the Westinghouse Small Break LOCA ECCS Evaluation Model incorporating the NOTRUMP analysis methodology.

II. METHOD OF EVALUATION

As a technical basis for the safety evaluation, analysis of postulated small break LOCA scenarios were performed assuming automatic safety injection flow delivery from only one high head safety injection pump.

The small break LOCA analyses were performed using the Westinghouse Small Break LOCA ECCS Evaluation Model incorporating the NOTRUMP analysis methodology to justify 60% power level operation for the H.B.Robinson Unit 2 nuclear power plant. The analyses assumed conservatively low estimate of the amount of safety injection flow delivered from one high head safety injection pump. The analysis model utilized the input developed in 1986 for the Carolina Power & Light company for the H.B.Robinson Unit 2 nuclear power plant performed to address the requirements of NUREG-0737 II.K.3.31.

Four small break LOCA analyses were performed using the NRC approved Westinghouse small break LOCA ECCS Evaluation Model incorporating the NOTRUMP analysis methodology. One of the analyses, a 3-inch equivalent diameter cold leg break, assumed 100% power. Three other analyses assumed a core power level corresponding to 102% of 1380 MWth (60% power operation). These analyses were for 1.0-inch, 1.5-inch, and 2-inch equivalent diameter cold leg breaks.

III. EVALUATION RESULTS

Of the three 60% power cases, the 2-inch case resulted in the highest peak cladding temperature of 965.4°F.

The analysis of the 3-inch equivalent diameter break at 100 percent power (102% of 2300 Mwt) resulted in a peak cladding temperature of 1771.6°F.

Small break LOCA analyses performed previously for H. B. Robinson at 100% power with two HHSI pumps available resulted in the 3-inch case having the highest peak cladding temperature of the 2-inch, 3-inch, and 4-inch breaks. Lowering the safety injection flow by taking credit for only one HHSI pump lowers the limiting break size. Therefore, the limiting break size for 100% power with only one HHSI available will be less than or equal to 3 inches.

The peak cladding temperature for the 3-inch case at 60% power will be lower than the peak cladding temperature for the 3-inch case at 100% power. Because the 3-inch case at 100% power did not exceed the limits of 10CFR50.46, the 3-inch case at 60% power will not exceed the limits. Therefore, the 3-inch case was not performed for the 60% power level.

The results of the analyses and evaluations show that the H.B.Robinson Unit 2 Nuclear Power Plant may be started and operated at 60 percent of licensed core power in compliance with the requirements of 10CFR50.46 when flow from only one high head safety injection pump is available.

A more detailed set of the small break LOCA analyses results are provided in Attachment B.

ATTACHMENT B

H.B. ROBINSON UNIT 2

SMALL BREAK LOCA ANALYSIS RESULTS

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 (RCS) pressure boundary. A major pipe break (large break) is defined as a rupture with a total cross sectional area equal to or greater than 1.0 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:

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

b. 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% are not exceeded during or after quenching.

c. 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% of the total amount of Zircaloy in the reactor.

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 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 trip 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 countercurrent 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 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.

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 small break LOCA power shape 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 conservative injection flow from one High Head Safety Injection (HHSI) pump. The conservative delivery data incorporates the standard FSAR ECCS assumption of minimum safeguards. The effect 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 busses.

The hydraulic analyses are typically performed with the NOTRUMP code using 102% of the licensed core power plus the 8 MWt energy added by the three reactor coolant pumps. However, due to the degraded HHSI system, only those results for the 3-inch break were performed at full power. The analyses for the 2, 1.5 and 1-inch break were performed at 60% of licensed core power. The core thermal transient analyses using LOCTA-IV were performed in a similar manner, i.e., 102% of licensed core power for the 3-inch break and 102% of 60% of licensed core power for the 2, 1.5 and 1-inch break. The LOCTA-IV core thermal analyses incorporated Exxon 15x15 fuel data which 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 that the limits of 10CFR 50.46 will not be exceeded at 60% of licensed core power operation. The results of these analyses are summarized in Tables 15.6.2-3 and 15.6.2-4. Figures 15.6.2-1 through 15.6.2-8b and 15.6.2-25 through 15.6.2-33 present the principal parameters of interest for the small break ECCS analyses. For the 2-inch 1.5-inch and 1-inch break sizes analyzed at 60% power, the following transient parameters are included:

a. RCS Pressure
b. Core Mixture Height
c. Hot Spot Clad Temperature
d. Intact Loop Pumped SI Flow
e. Break Vapor Flow

As indicated in the results for clad heat up, the 3-inch case is limiting. However, the 3-inch case is for 102% of licensed core power. For the limiting break size analyzed (3-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 1772°F. These results are well below all Acceptance Criteria limits of 10 CFR 50.46 and demonstrate acceptability of operation with one HHSI pump at 60% of licensed core power.

TABLE 15.6.2-1

Input Parameters Used in the SBLOCA Analysis

Core Power[1]	1407.6 MWt
Pump Heat	8 MWt
NSSS Power	1415.6 MWt[2]
Peak Linear Power (includes 102% factor)	13.197 kW/ft
Total Peaking Factor, F	2.32[3]
Power Shape	Fig. 15.6.2-13
Fuel Assembly Array	Exxon 15x15
Nominal Accumulator Water Volume	825 ft /accum.
Nominal Accumulator Tank Volume	1200 ft /accum.
Minimum Accumulator Gas Pressure	615 psia
Pumped Safety Injection Flow	Fig. 15.6.2-12
Steam Generator Initial Pressure	863 psia
Auxiliary Feedwater Flow	41.22 lb/sec/SG
Steam Generator Tube Plugging Level	5%

1 - 2% has been added to this power to account for calorimetric uncertainty

2 - As noted in the text, the 3-inch break was performed for 100% power or NSSS power of 2354 MWt

3 - 2.32 is for 100% Power. At reduced power levels the allowable peaking factor will increase in accordance with plant Technical Specifications.

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

Event	100% Power 3 in (sec)	-------- 60% Power------- 2 in (sec)	1.5 in (sec)	1 in (sec)
Start	0.0	0.0	0.0	0.0
Reactor Trip	5.79	12.00	21.31	47.14
S-signal	9.85	19.39	33.84	76.77
Loop Seal Venting	450.3	1336.6	2088.0	4286.1
Top of Core Uncovered	798.2	1312.9	2048.3	4560.1
Accumulator Injection	1099.6	3006.9	N/A	N/A
Maximum Core Uncovery	1182.1	3005.9	2085.0	4645.5
Peak Clad Temperature Occurs	1229.9	3101.6	2352.6	N/A
Top of Core Covered	N/A	3465.3	2355.4	4657.7

TABLE 15.6.2-4

Small Break LOCA Fuel Cladding Results

Results	100% Power 3 in	60% Power 2 in	60% Power 1.5 in	60% Power 1 in
Peak clad temperature (°F)	1771.6	965.4	743.5	N/A
Peak clad temperature location (ft)	12.0	12.0	12.0	N/A
Local Zr/H_2O reaction, maximum (%)	2.31	0.20	0.20	N/A
Local Zr/H_2O location (ft)	12.0	12.0	12.0	N/A
Total Zr/H_2O reaction (%)	<0.3	<0.3	<0.3	N/A
Hot rod burst time (sec)	N/A	N/A	N/A	N/A
Hot rod burst location (ft)	N/A	N/A	N/A	N/A

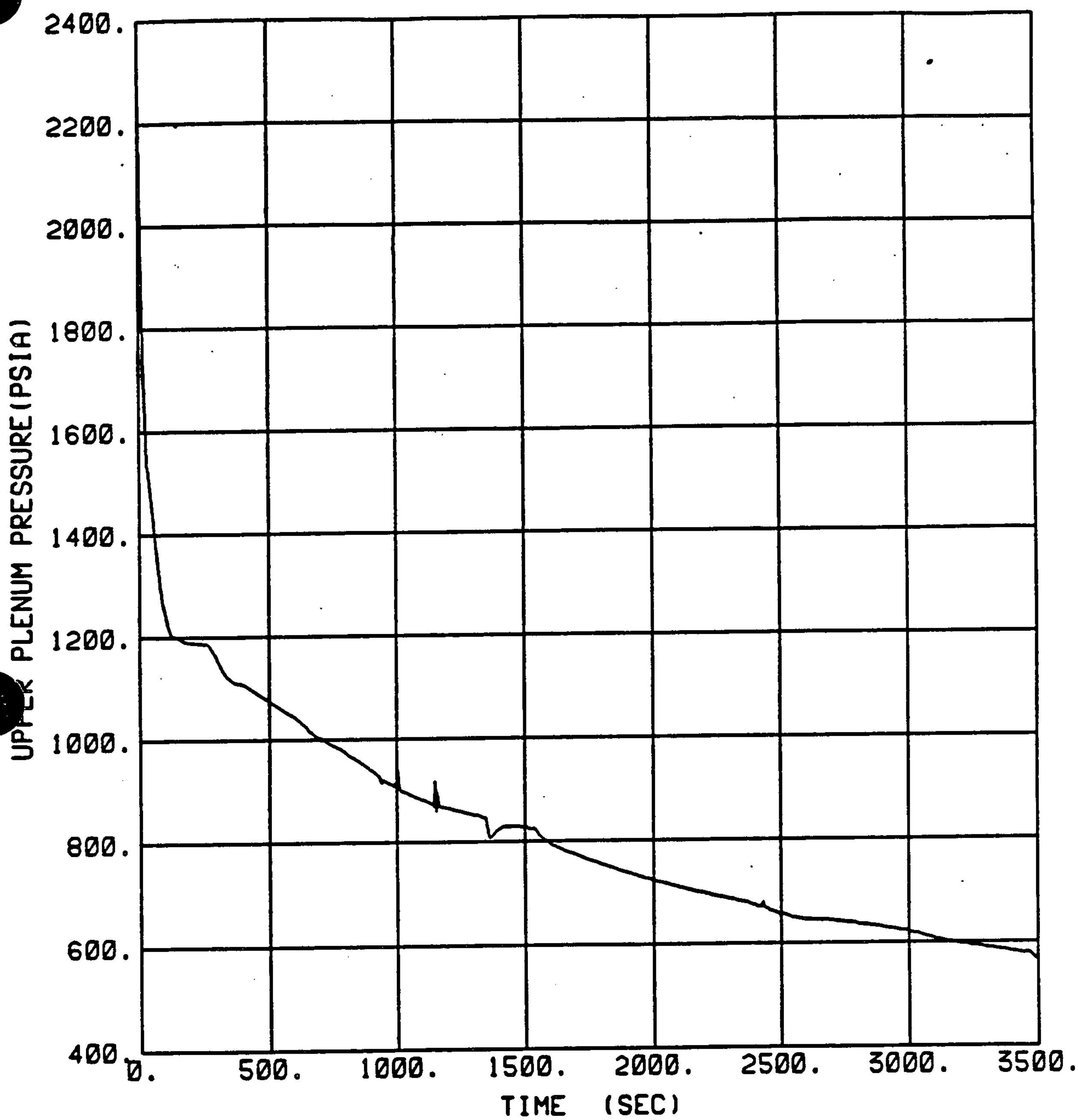
ERENCES FOR SECTION 15.6.2

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, 1974.

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 ECCS Evaluation Model Using the NOTRUMP Code", WCAP-10081-A, August 1985.

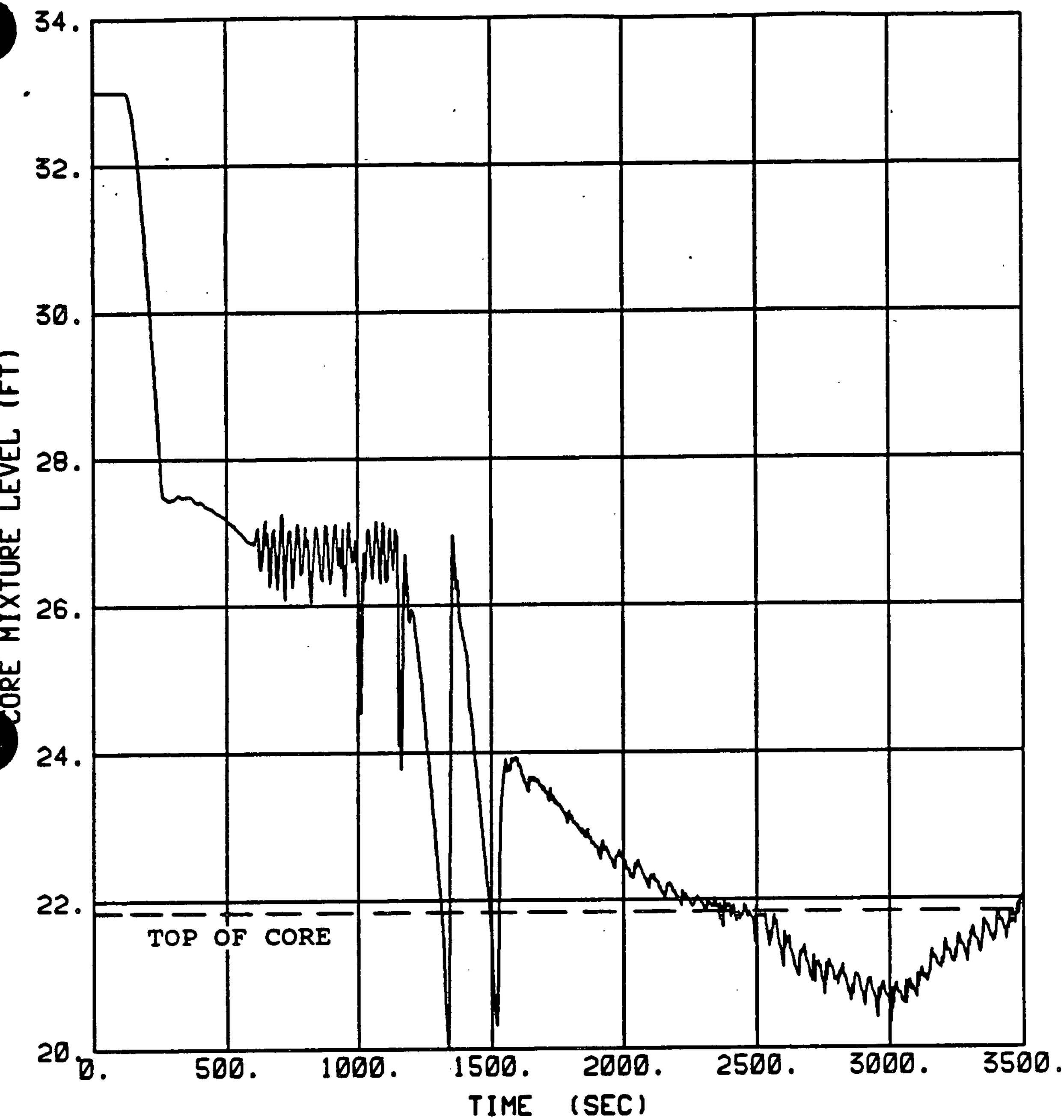
4. Bordelon, F. M., et. al., "LOCTA-IV Program: Loss of Coolant Transient Analysis", WCAP-8301, (Proprietary) and WCAP-8305, (Non-Proprietary), June 1974.



H. B. ROBINSON UNIT 2

UPPER PLENUM PRESSURE
2-INCH COLD LEG BREAK - 60% POWER

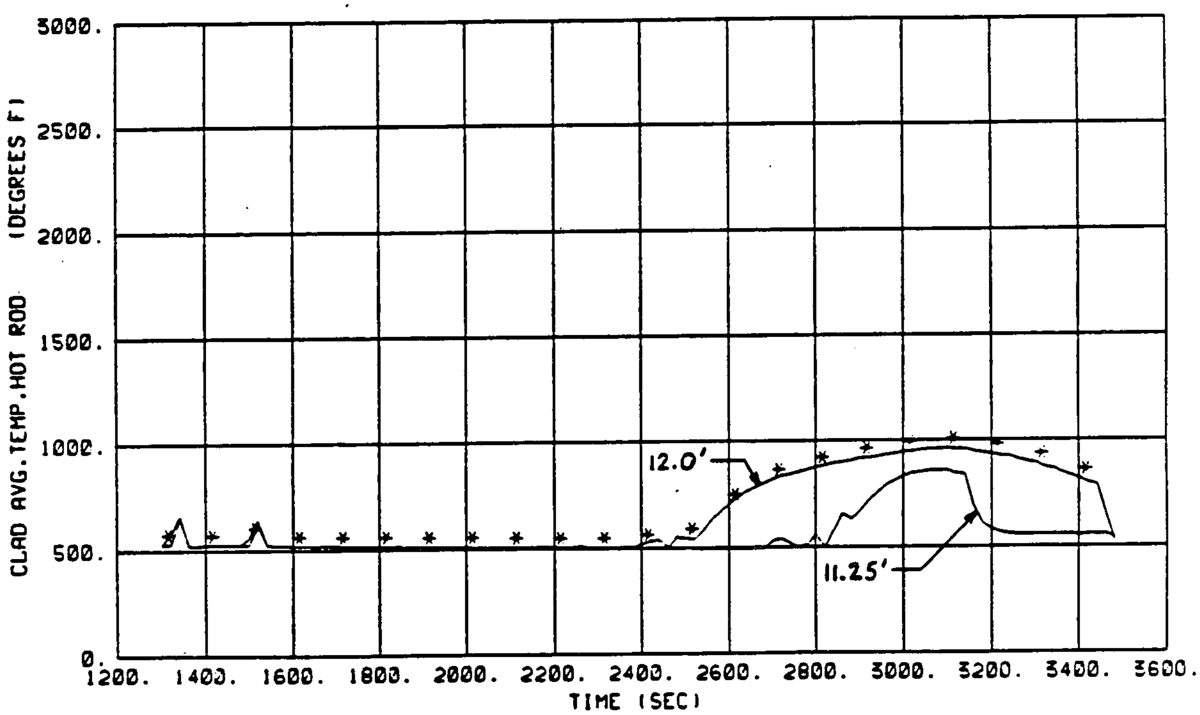
FIGURE 15.6.2-1



H. B. ROBINSON UNIT 2

CORE MIXTURE LEVEL
2-INCH COLD LEG BREAK - 60% POWER

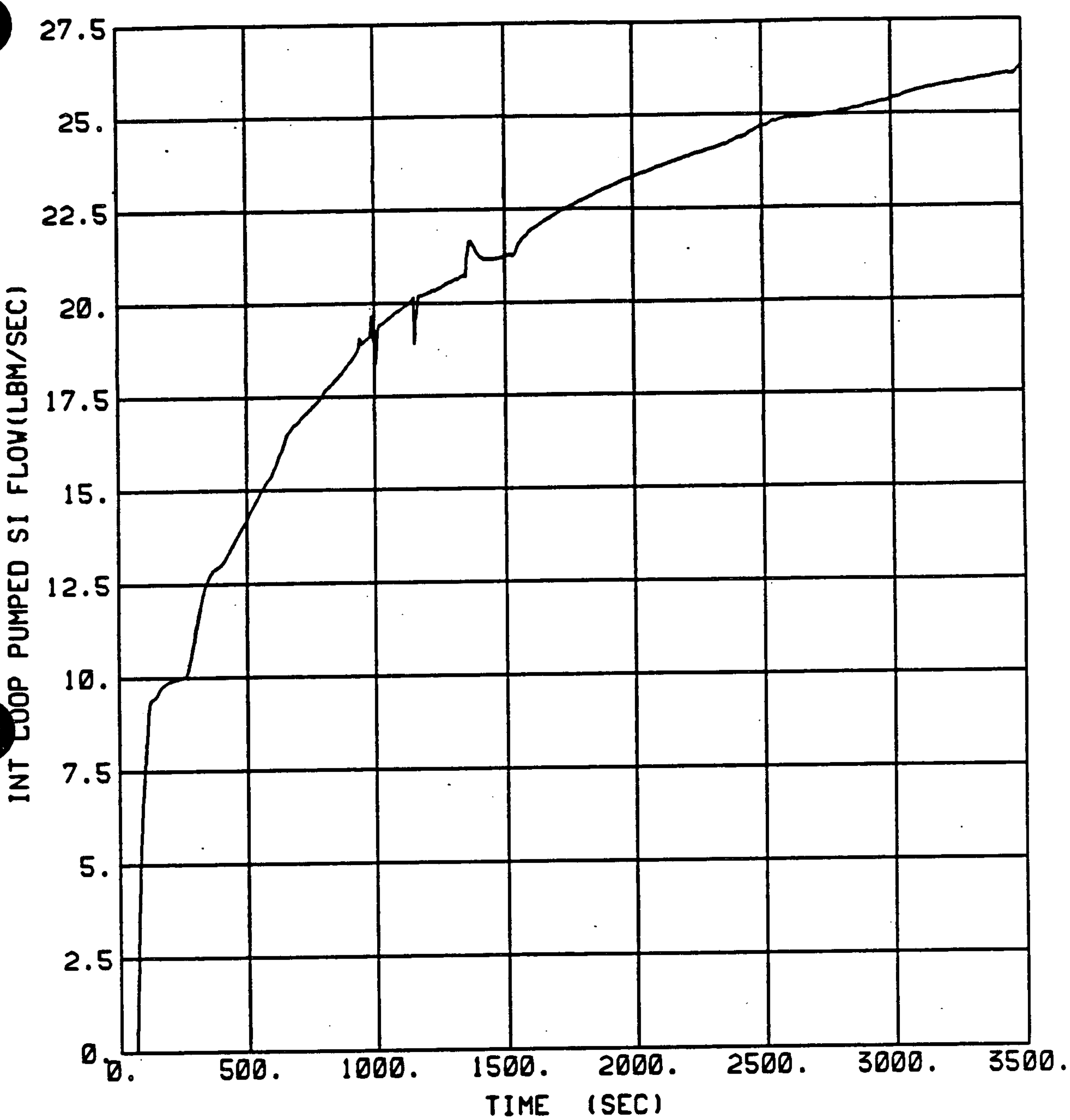
FIGURE 15.6.2-2



H. B. ROBINSON UNIT 2

HOT SPOT CLAD TEMPERATURE
2-INCH COLD LEG BREAK - 60% POWER

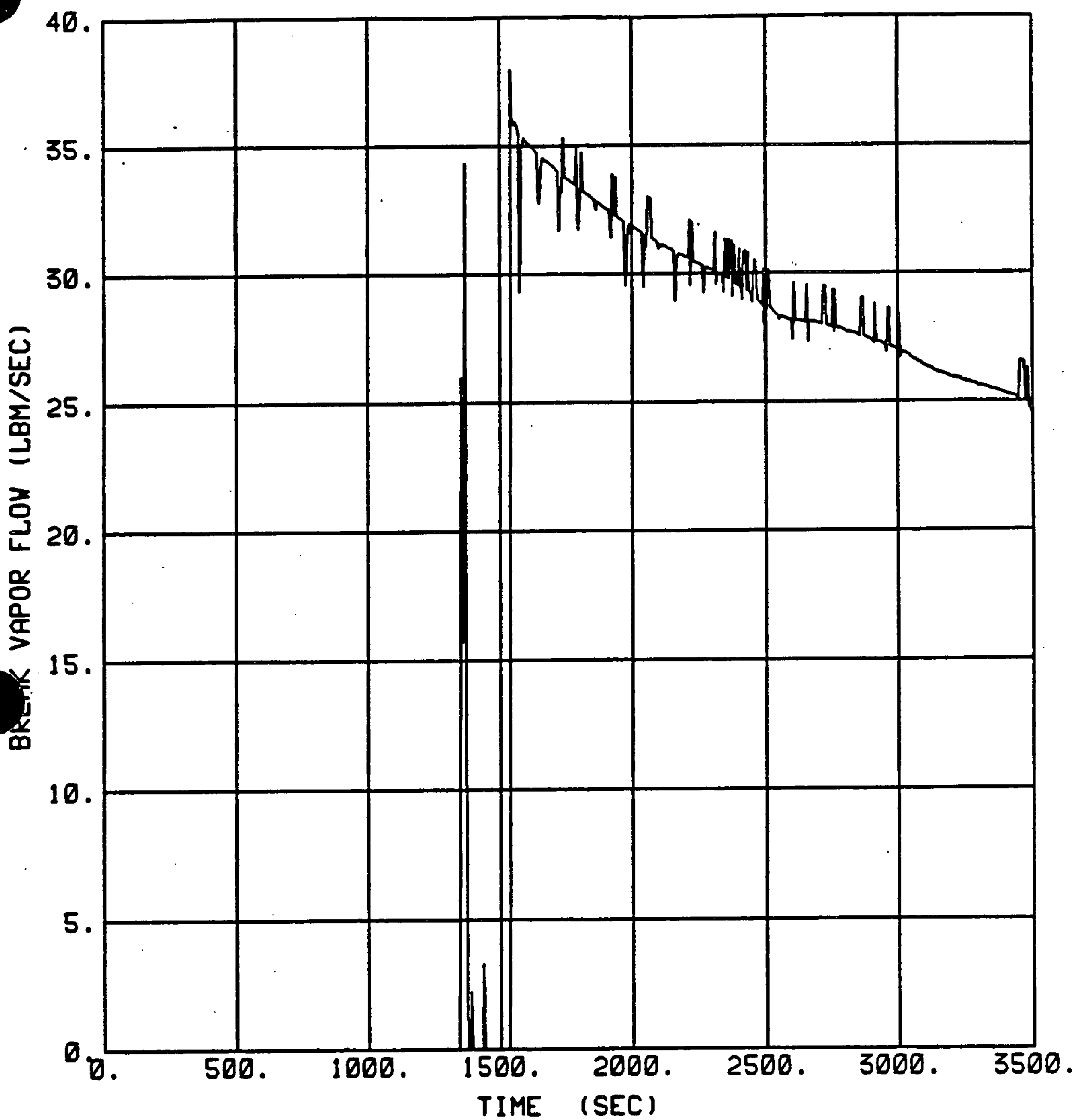
FIGURE 15.6.2-2a



H. B. ROBINSON UNIT 2

INTACT LOOP PUMPED SI FLOW
2-INCH COLD LEG BREAK - 60% POWER

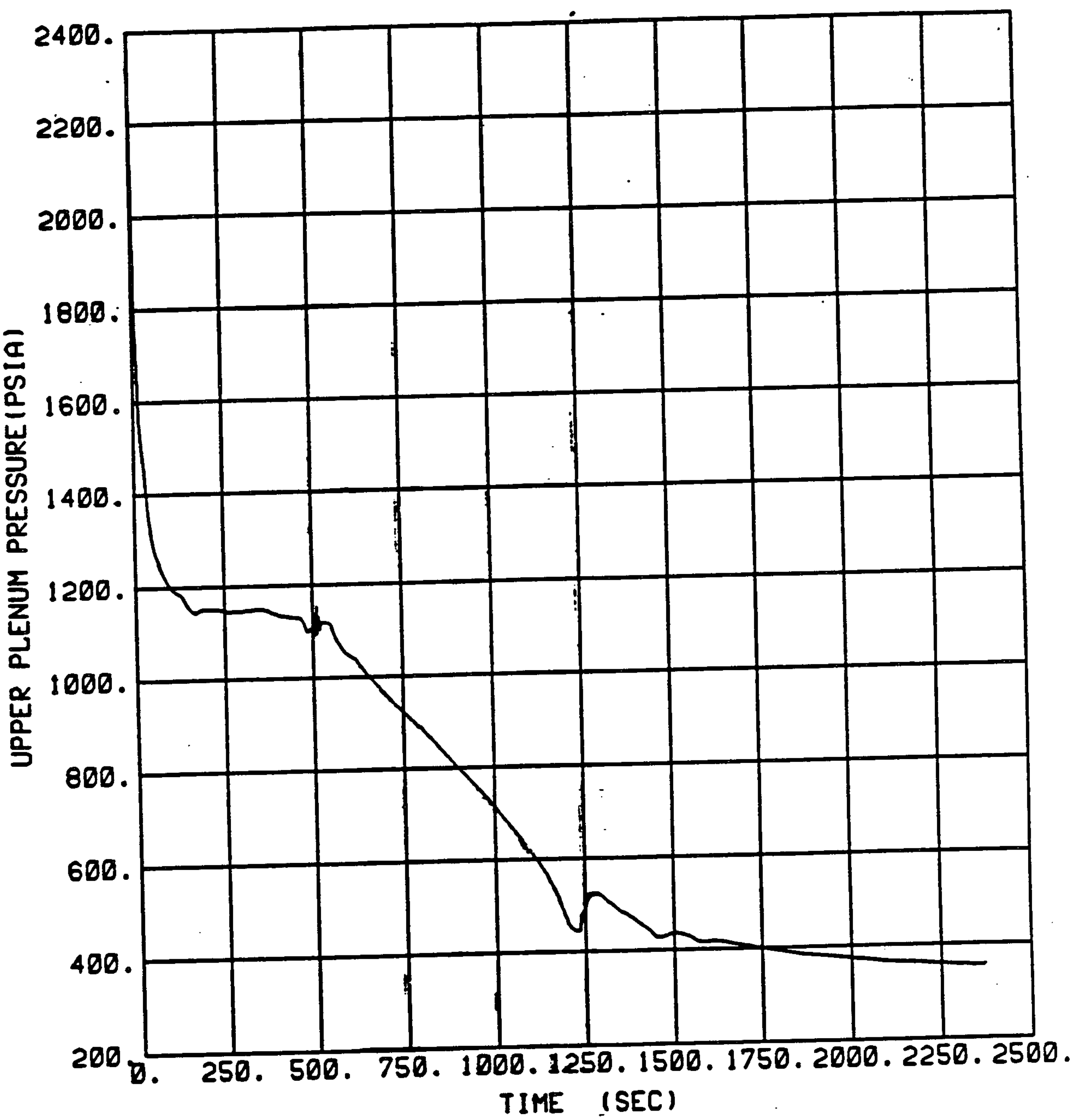
FIGURE 15.6.2-2b



H. B. ROBINSON UNIT 2

BREAK VAPOR FLOW
2-INCH COLD LEG BREAK - 60% POWER

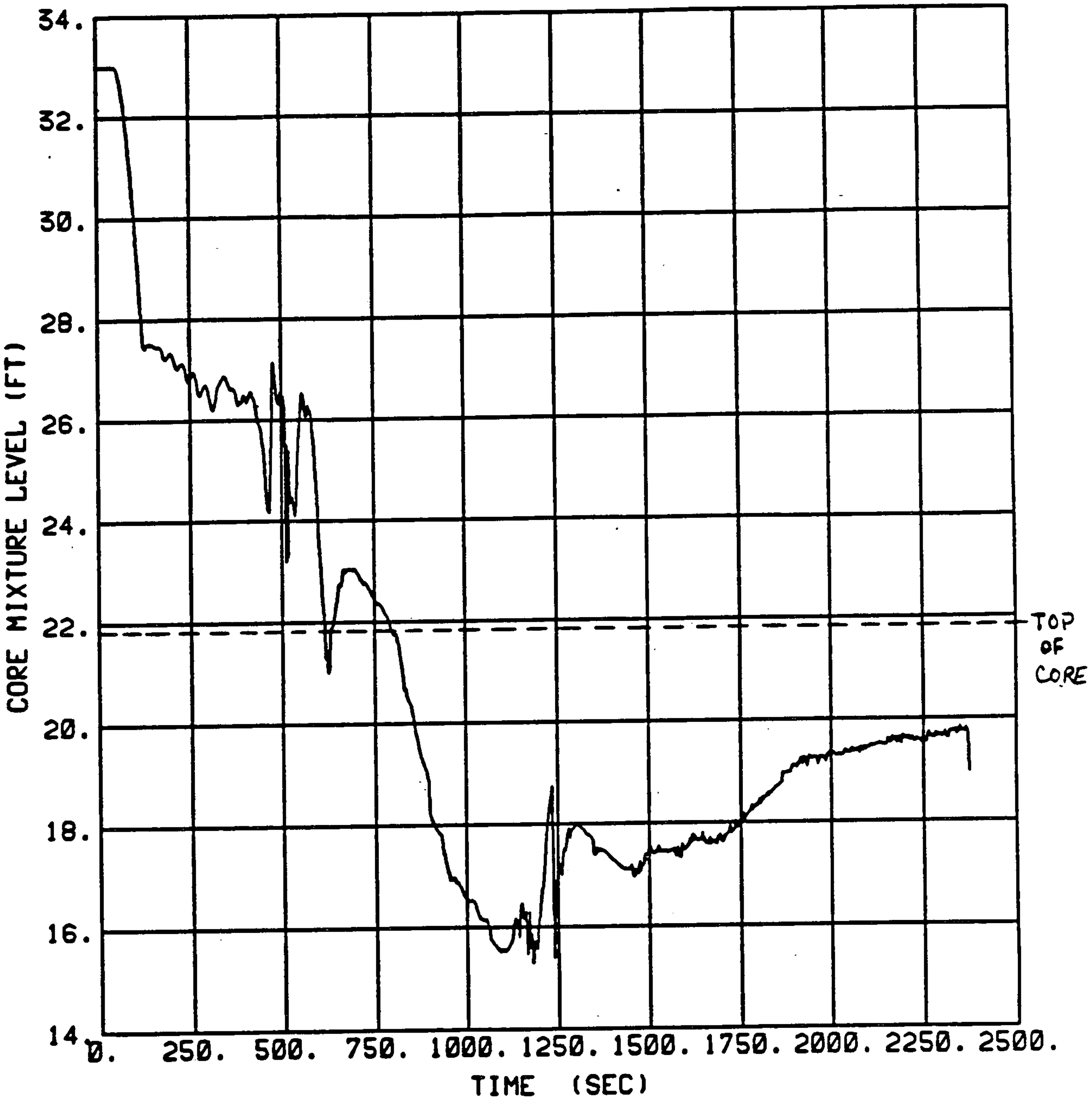
FIGURE 15.6.2-2c



H. B. ROBINSON UNIT 2

UPPER PLENUM PRESSURE
3-INCH COLD LEG BREAK

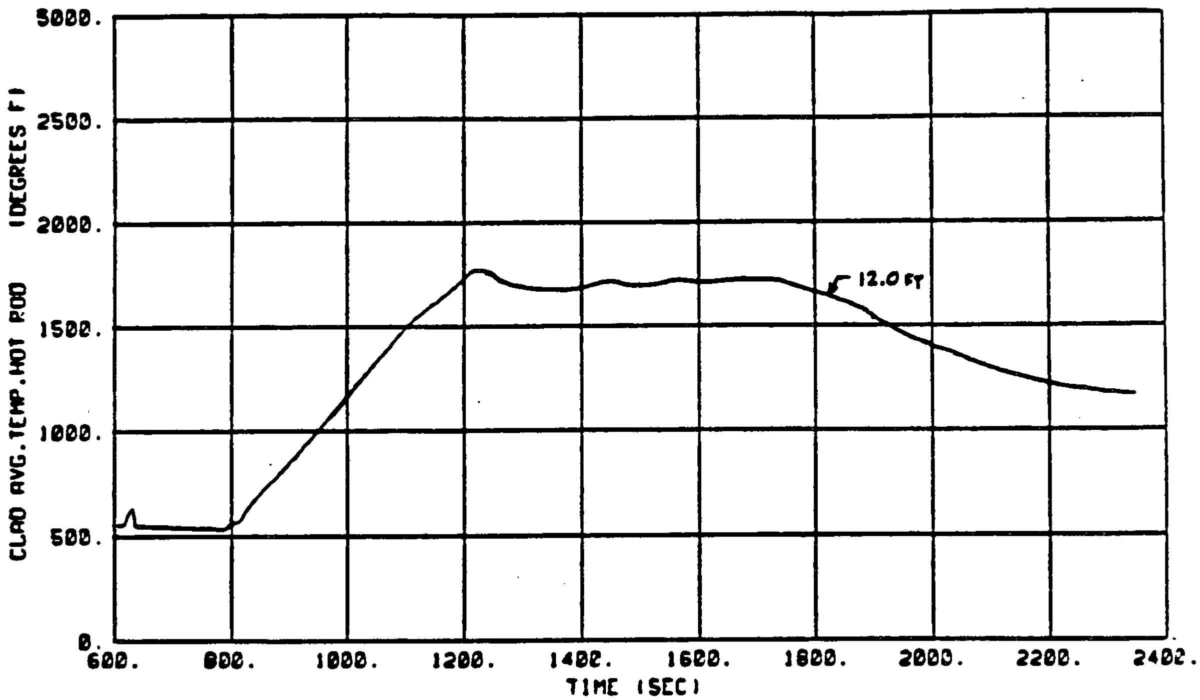
FIGURE 15.6.2-3



H. B. ROBINSON UNIT 2

CORE MIXTURE LEVEL
3-INCH COLD LEG BREAK

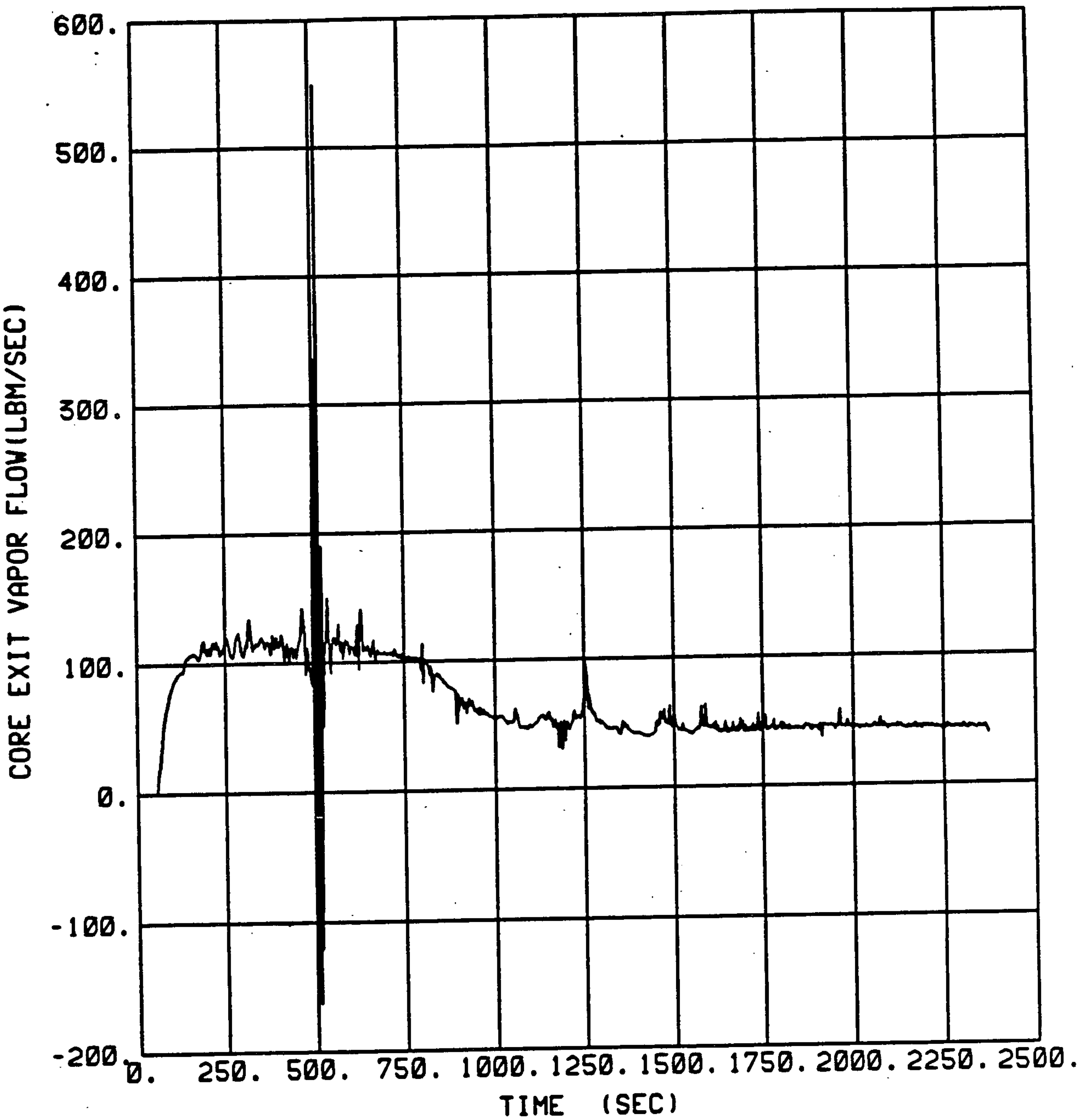
FIGURE 15.6.2-4



H. B. ROBINSON UNIT 2

HOT SPOT CLAD TEMPERATURE
3-INCH COLD LEG BREAK

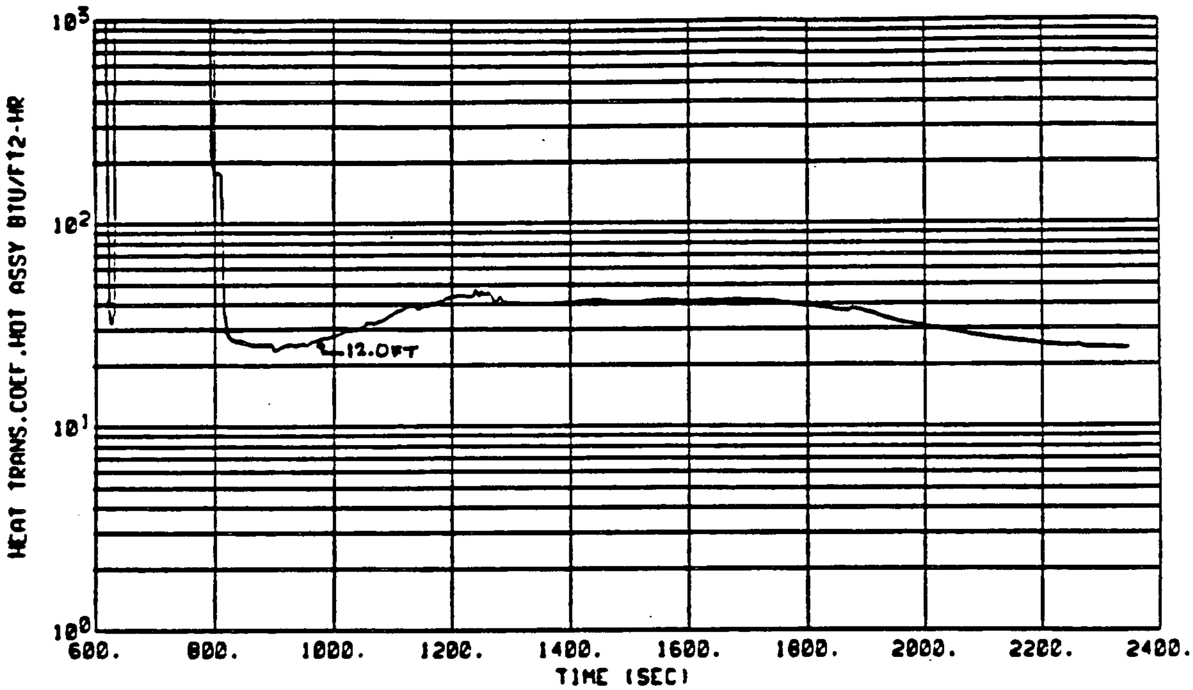
FIGURE 15.6.2-5



H. B. ROBINSON UNIT 2

CORE EXIT VAPOR FLOW
3-INCH COLD LEG BREAK

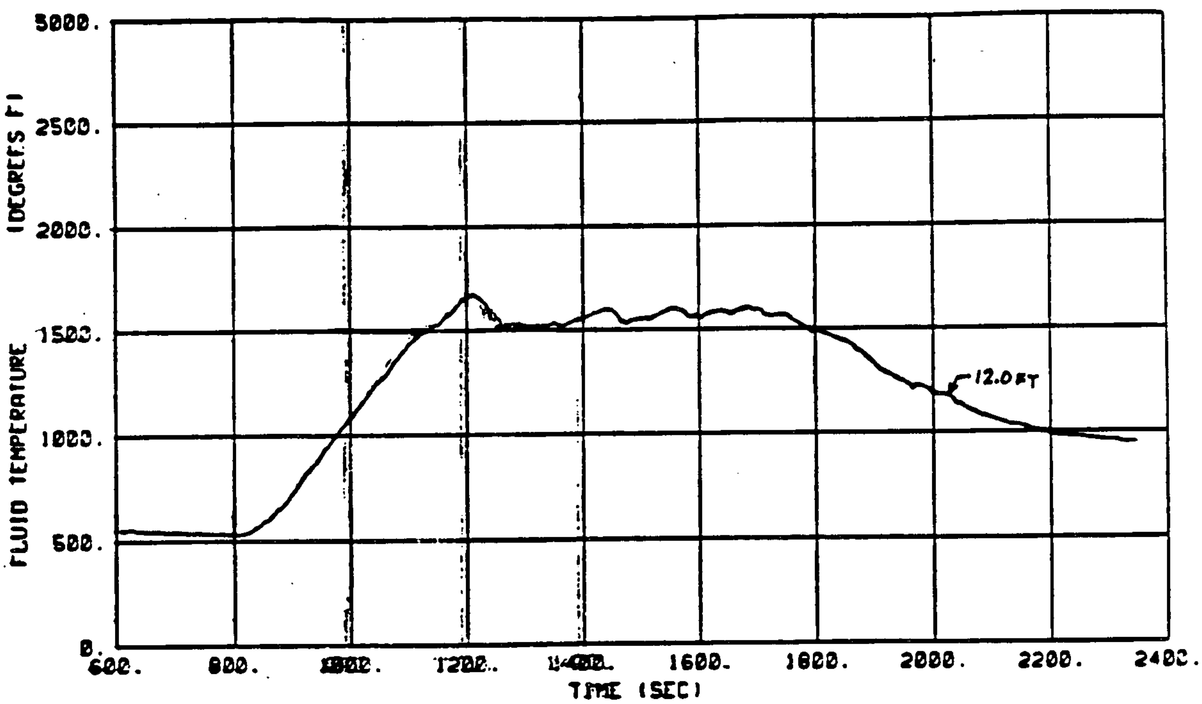
FIGURE 15.6.2-6



H. B. ROBINSON UNIT 2

CORE HEAT TRANSFER COEFFICIENT
3-INCH COLD LEG BREAK

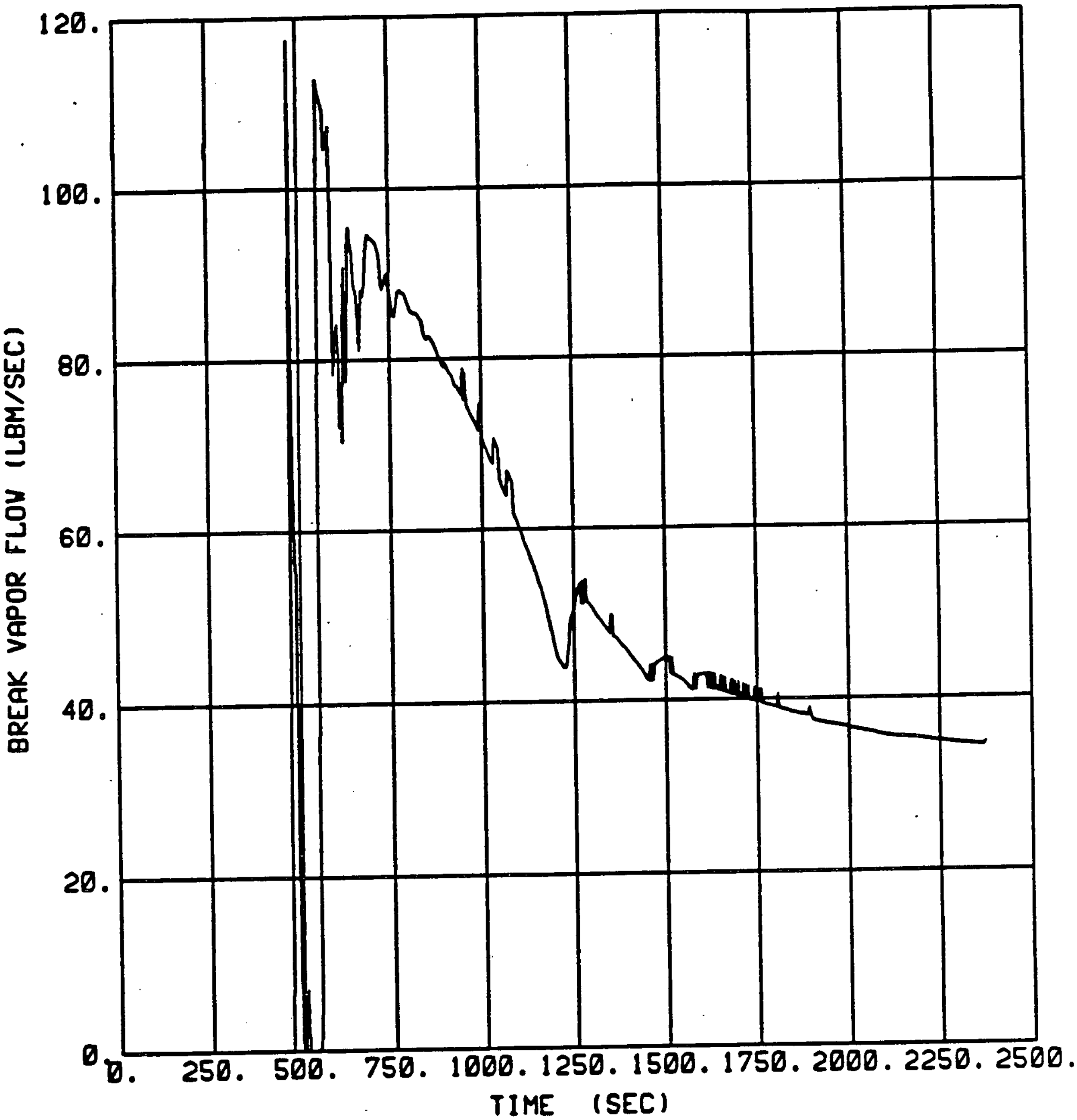
FIGURE 15.6.2-7



H. B. ROBINSON UNIT 2

HOT SPOT FLUID TEMPERATURE
3-INCH COLD LEG BREAK

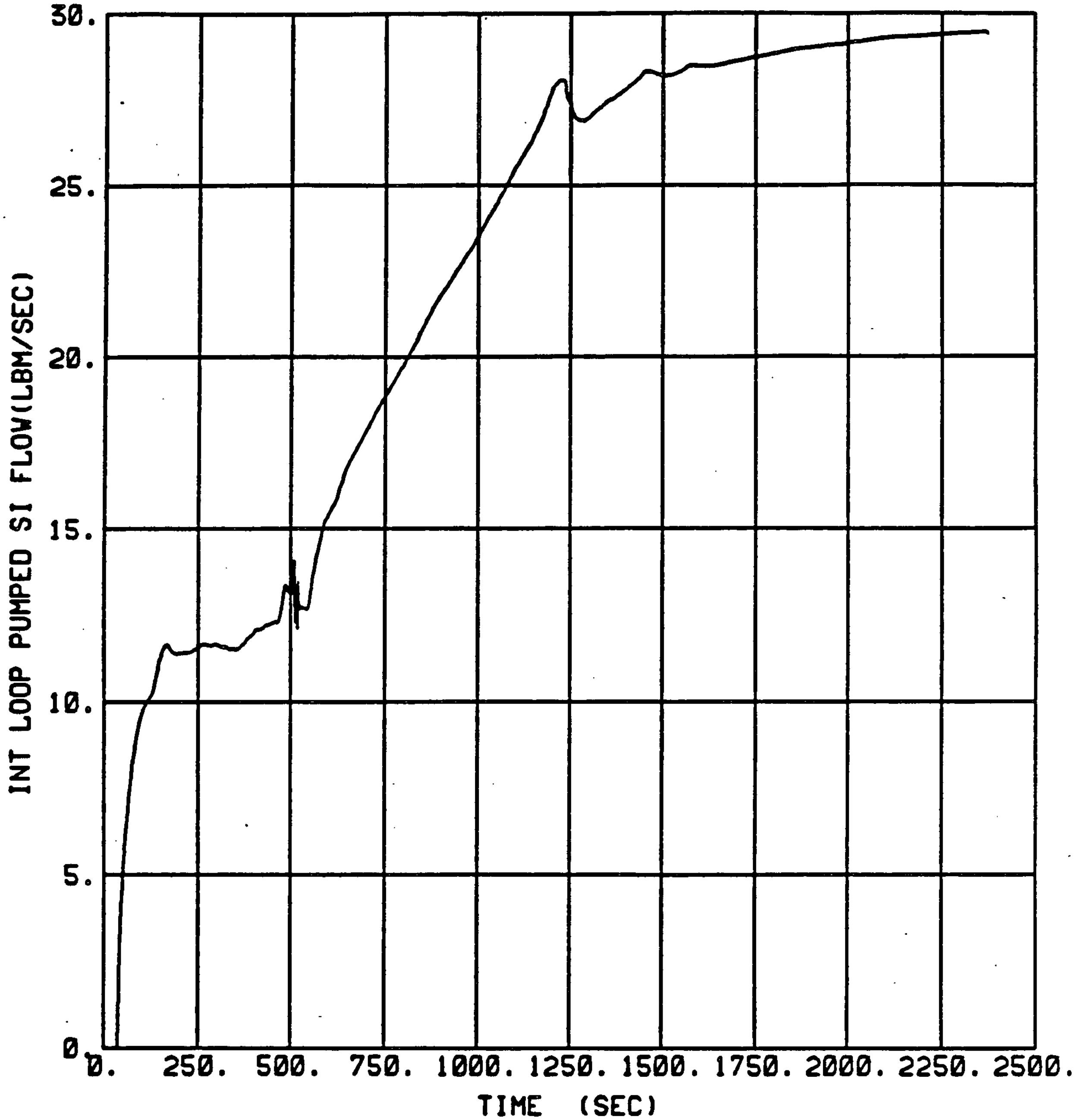
FIGURE 15.6.2-8



H. B. ROBINSON UNIT 2

VAPOR BREAK MASS FLOW RATE
3-INCH COLD LEG BREAK - 100% POWER

FIGURE 15.6.2 - 8a



H. B. ROBINSON UNIT 2

INTACT LOOP PUMPED SAFETY INJECTION MASS FLOW RATE
3-INCH COLD LEG BREAK - 100% POWER

FIGURE 15.6.2 - 8b

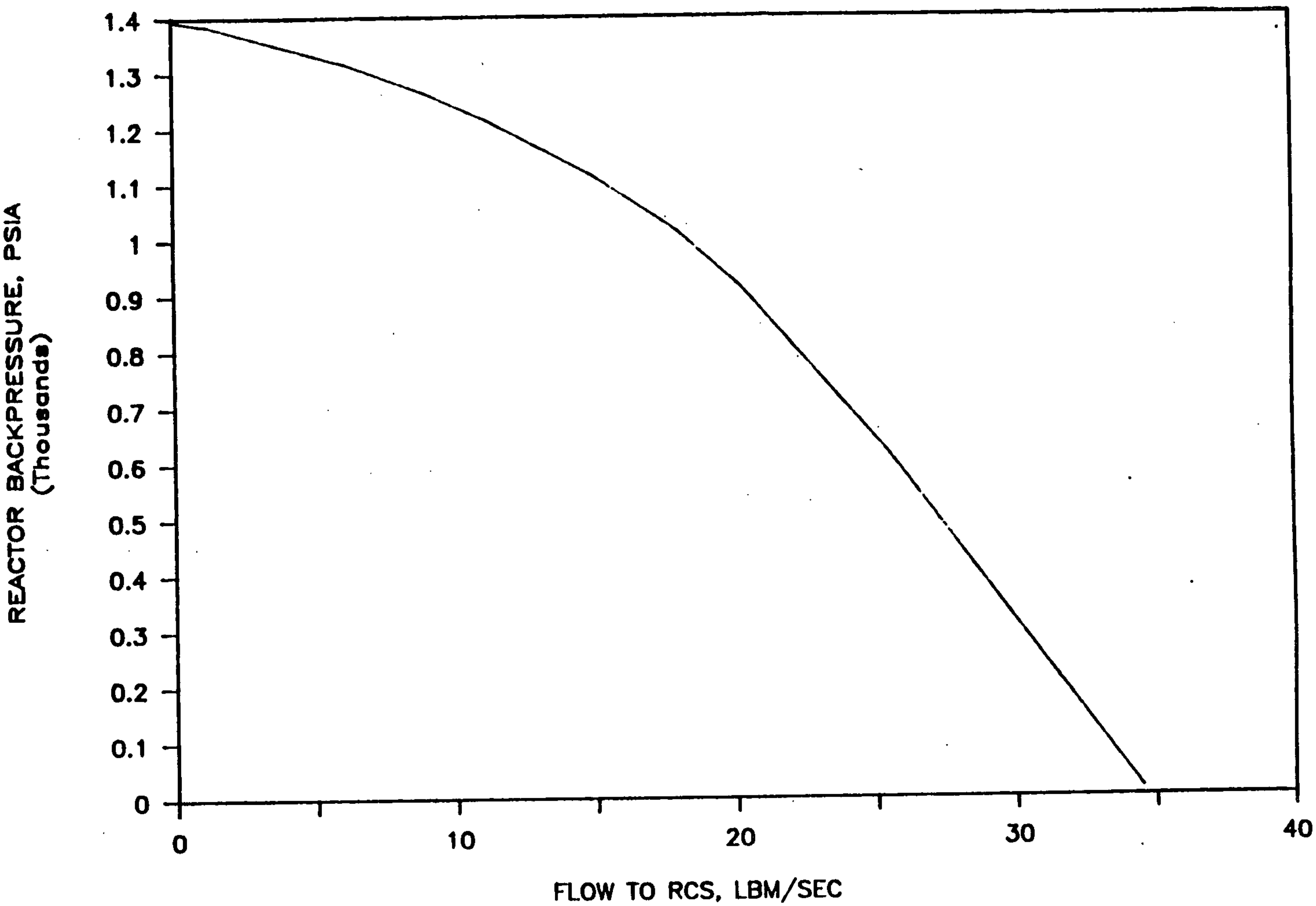


Figure 15.6.2-12 H. B. Robinson Pumped Safety Injection Flow

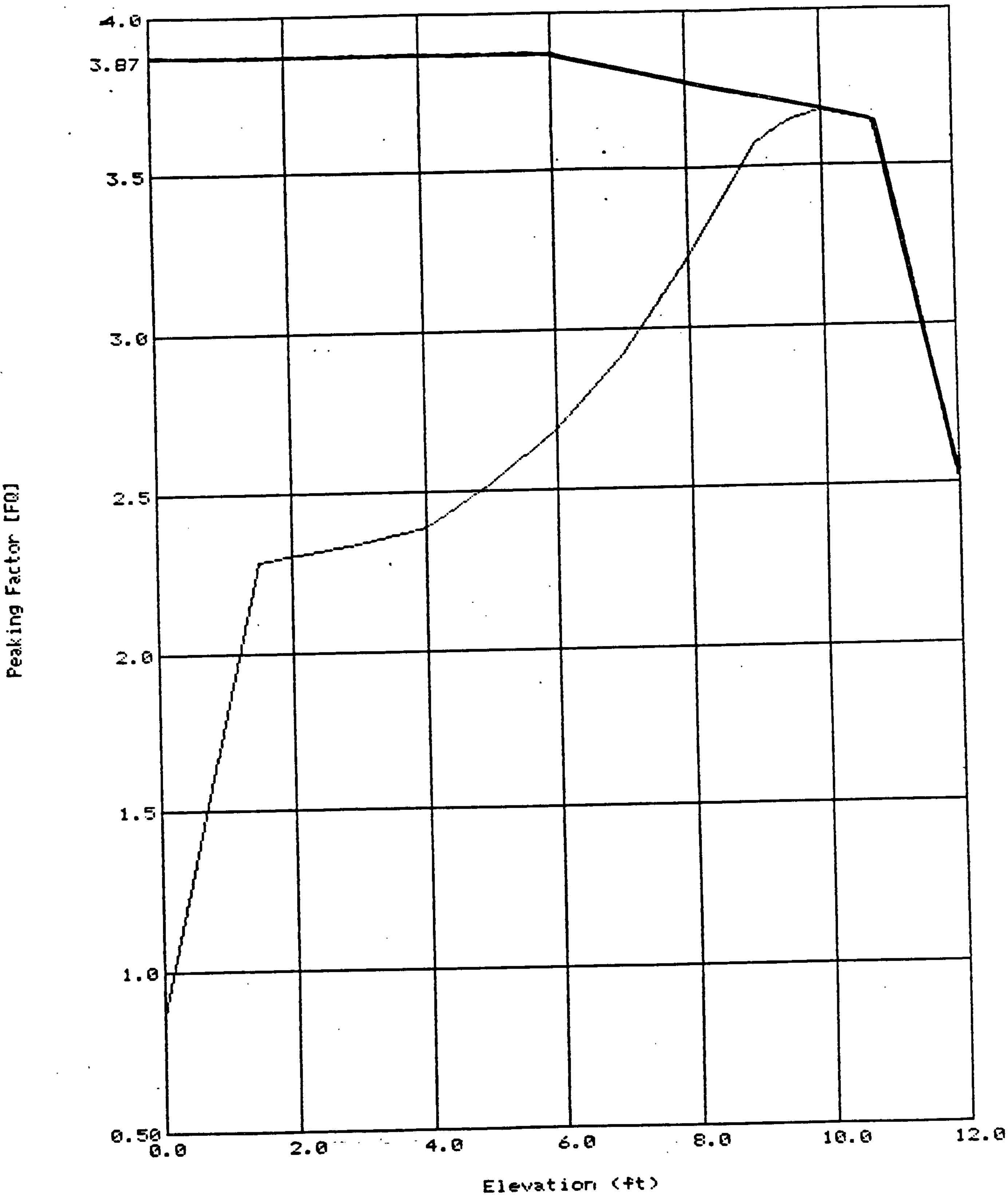


Figure 15.6.2-13 H.B. Robinson Small-Break LOCA Power Shape for 60% Power.

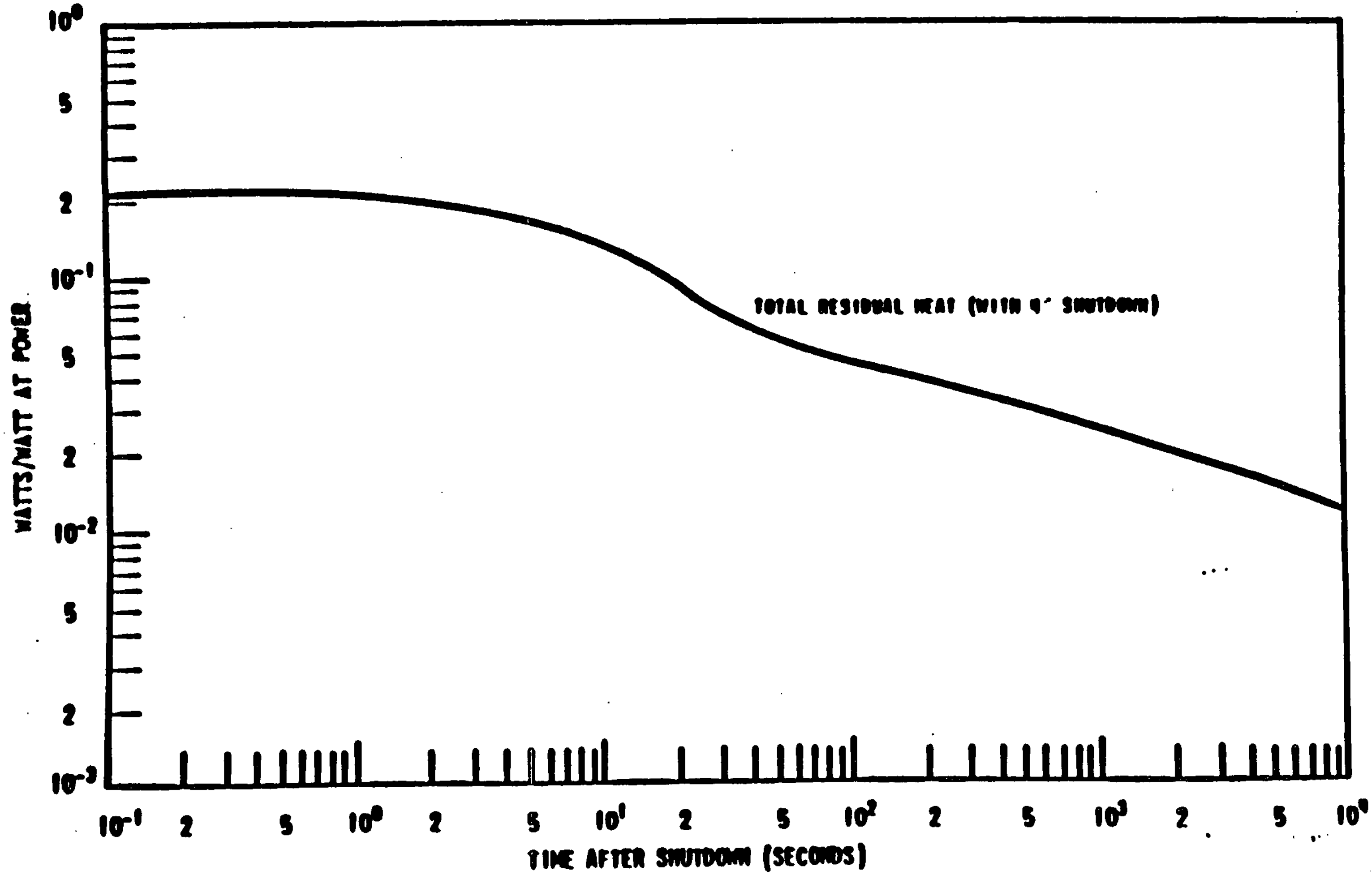
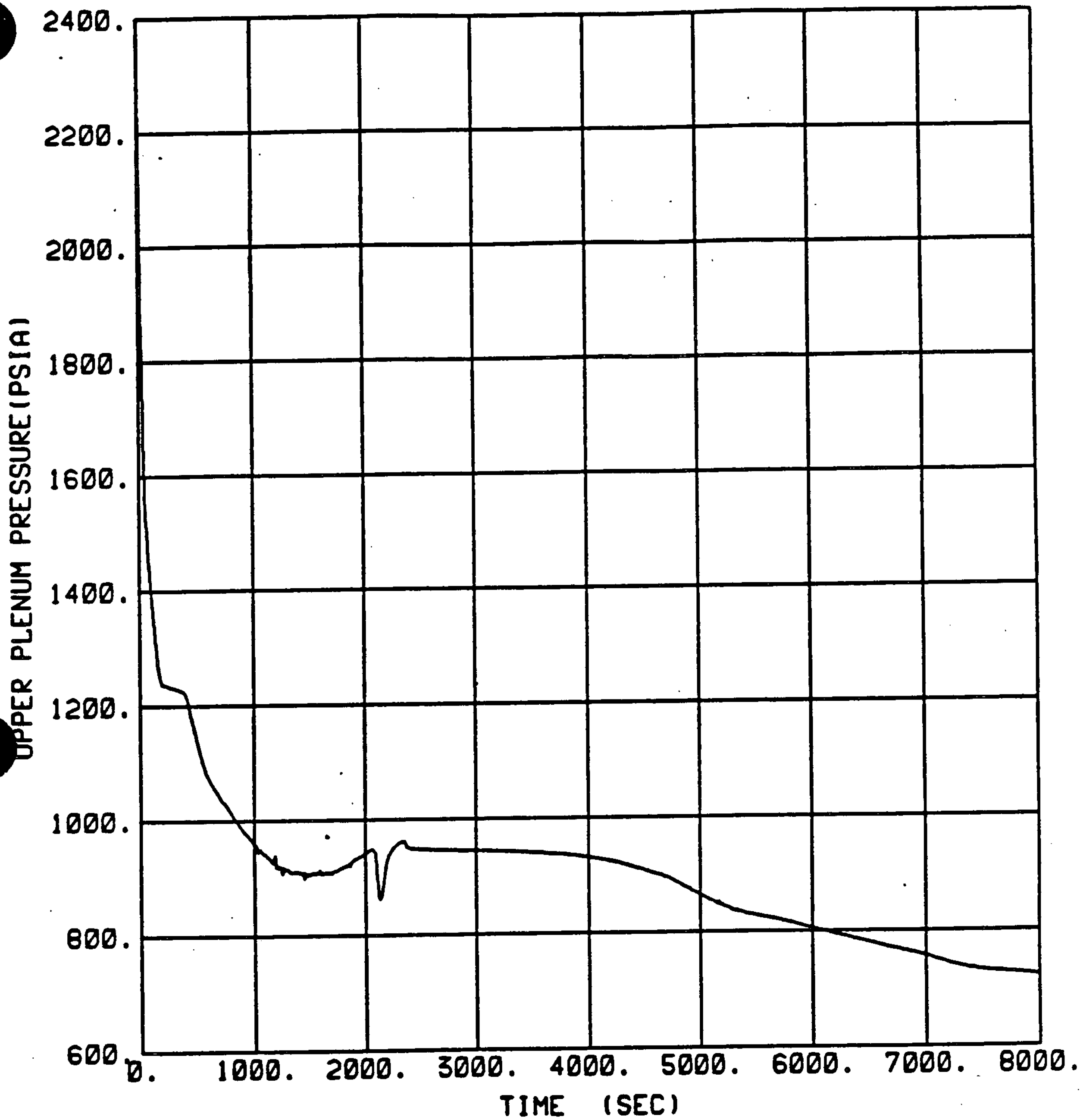


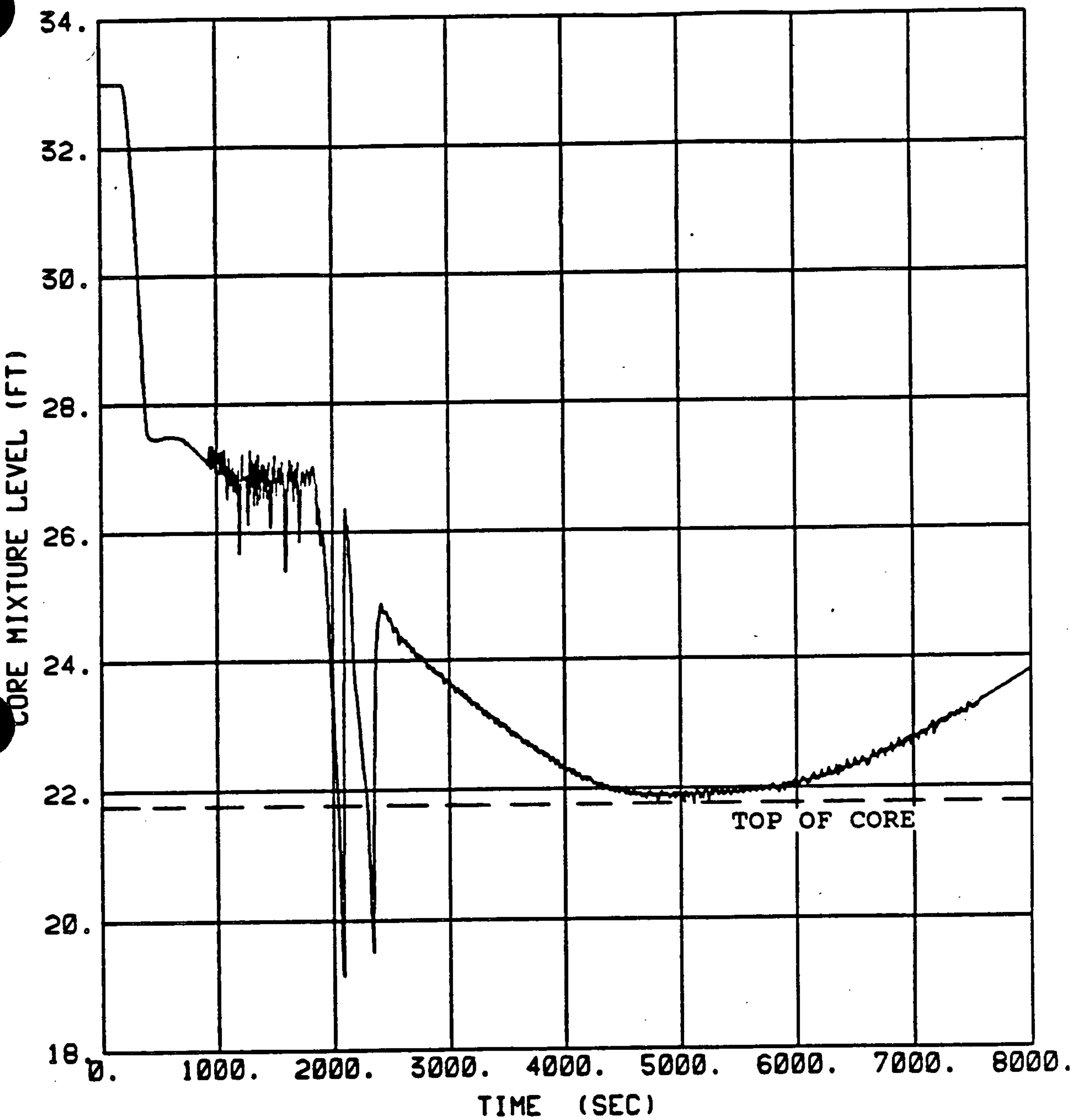
Figure 15.6.2-14 Core Power After Reactor Trips



H. B. ROBINSON UNIT 2

UPPER PLENUM PRESSURE
1.5-INCH COLD LEG BREAK - 60% POWER

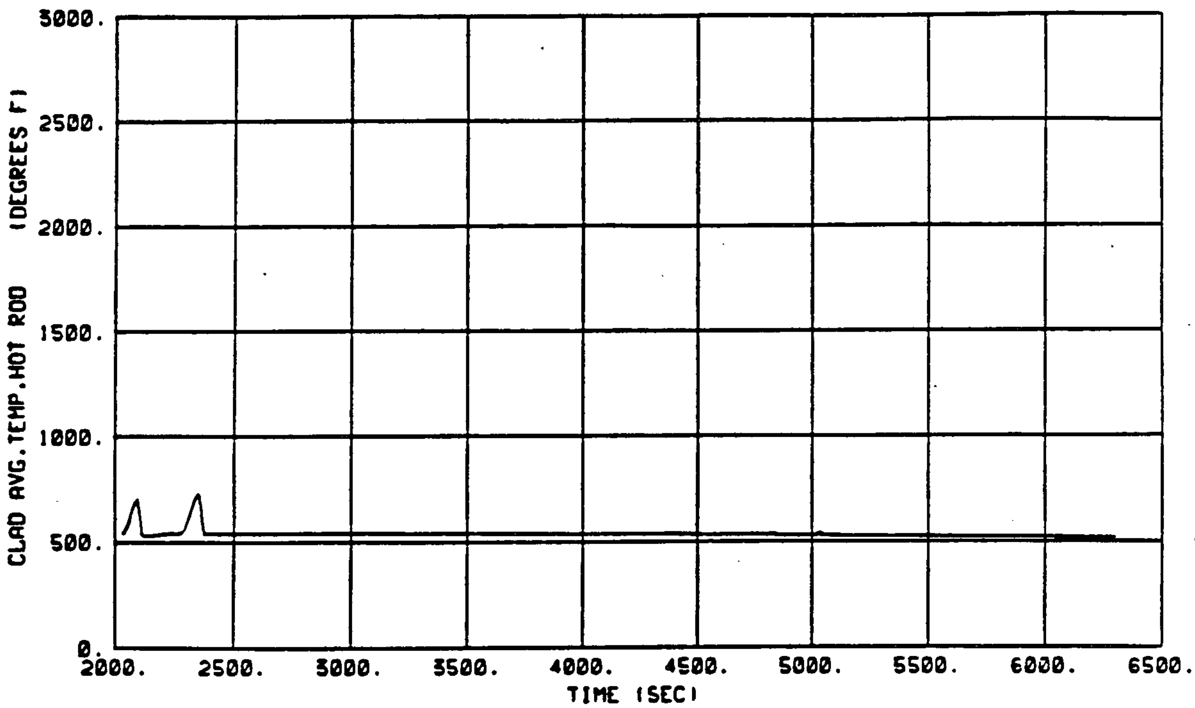
FIGURE 15.6.2-25



H. B. ROBINSON UNIT 2

CORE MIXTURE LEVEL
1.5-INCH COLD LEG BREAK - 60% POWER

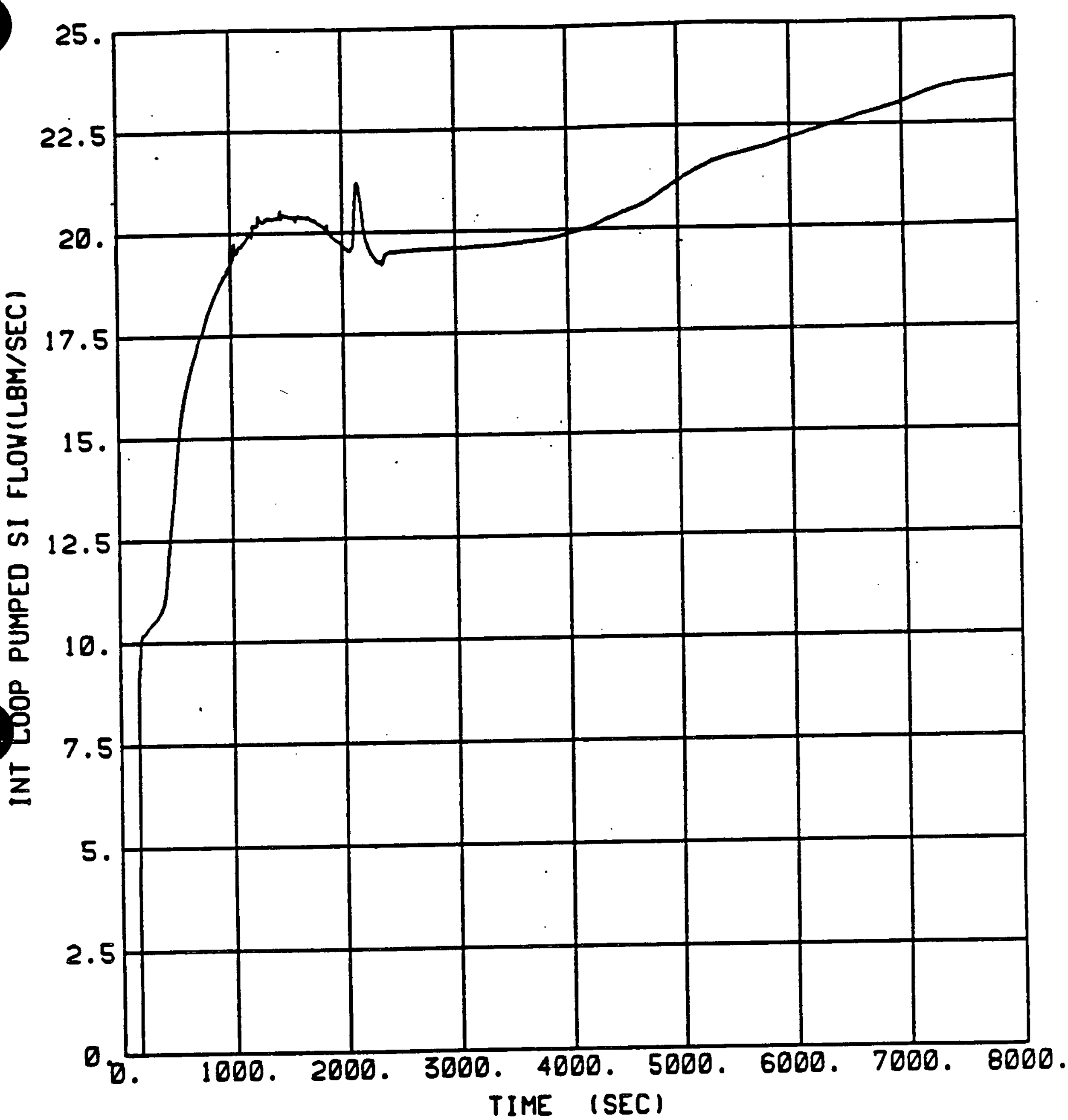
FIGURE 15.6.2-26



H. B. ROBINSON UNIT 2

HOT SPOT CLAD TEMPERATURE
1.5-INCH COLD LEG BREAK - 60% POWER

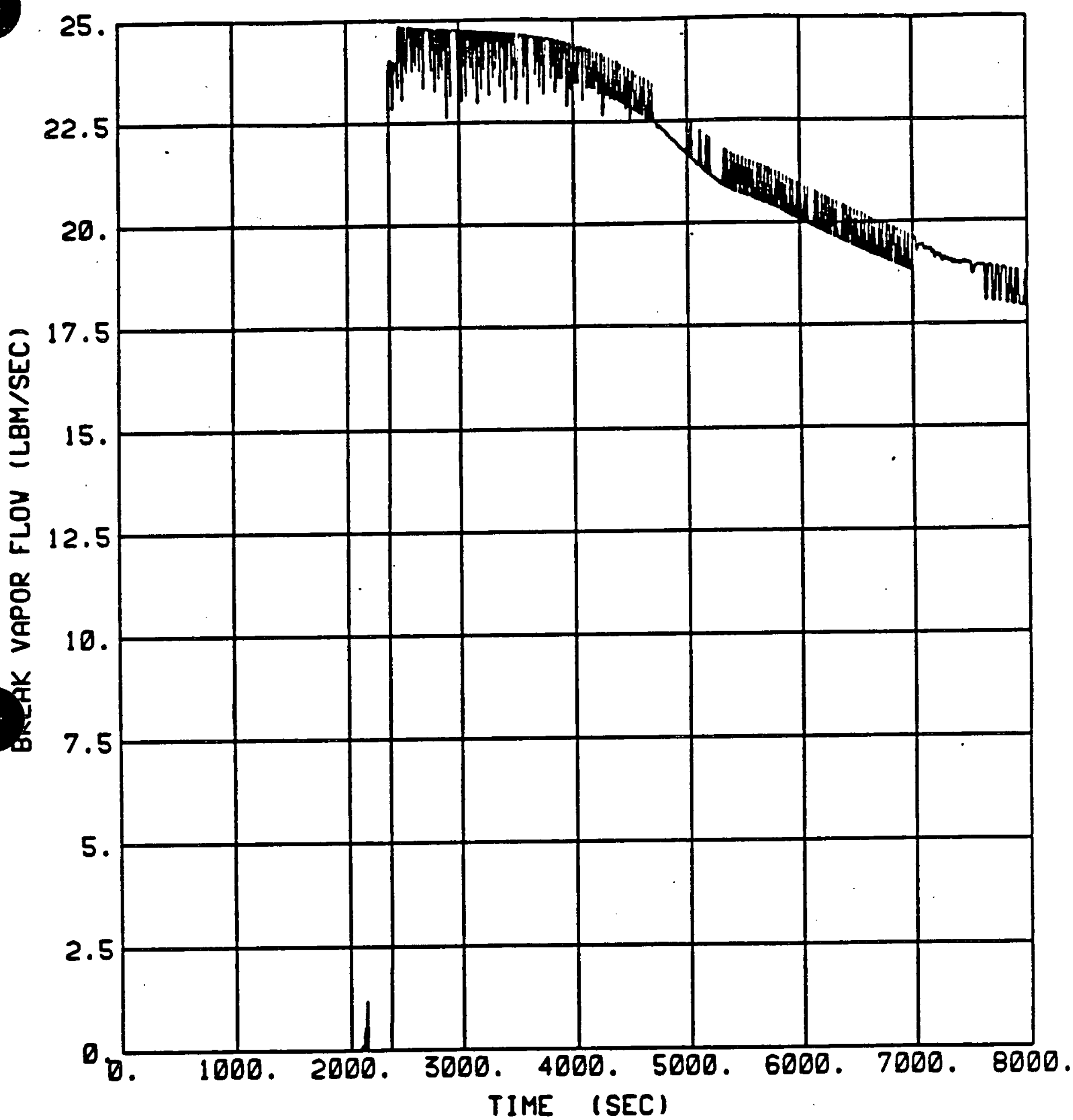
FIGURE 15.6.2-27



H. B. ROBINSON UNIT 2

INTACT LOOP PUMPED SI FLOW
1.5-INCH COLD LEG BREAK - 60% POWER

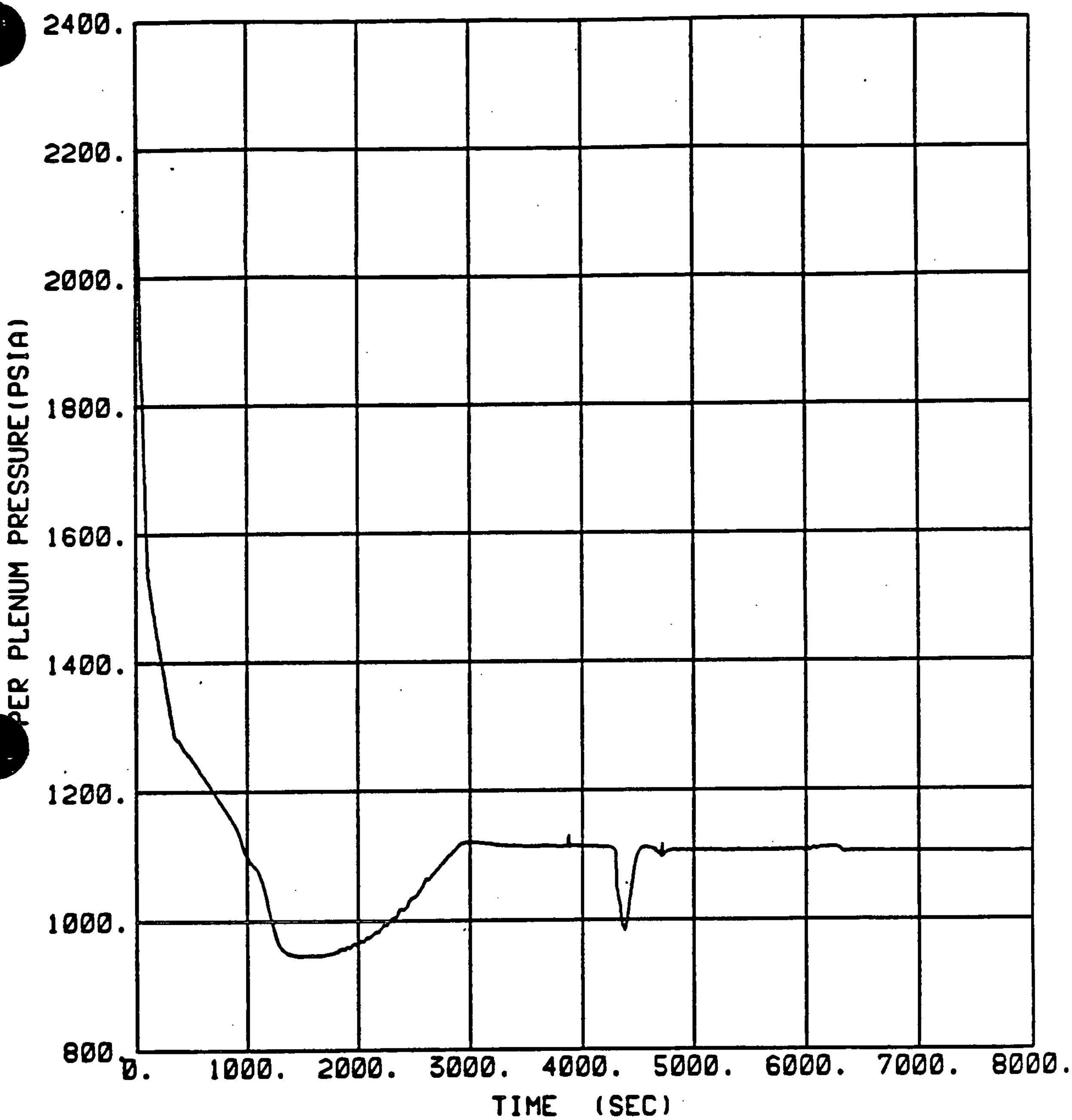
FIGURE 15.6.2-28



H. B. ROBINSON UNIT 2

BREAK VAPOR FLOW
1.5-INCH COLD LEG BREAK - 60% POWER

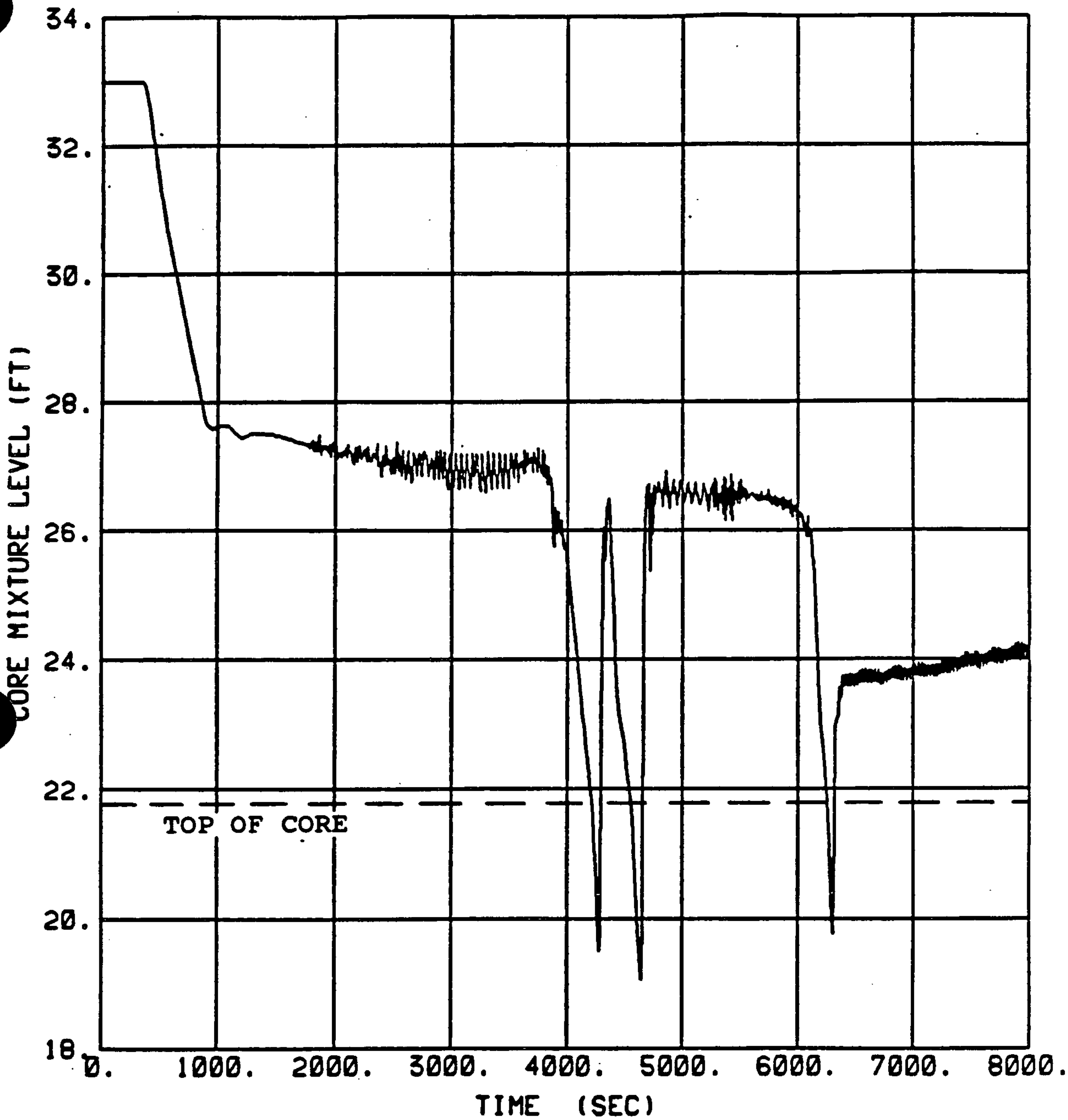
FIGURE 15.6.2-29



H. B. ROBINSON UNIT 2

UPPER PLENUM PRESSURE
1-INCH COLD LEG BREAK - 60% POWER

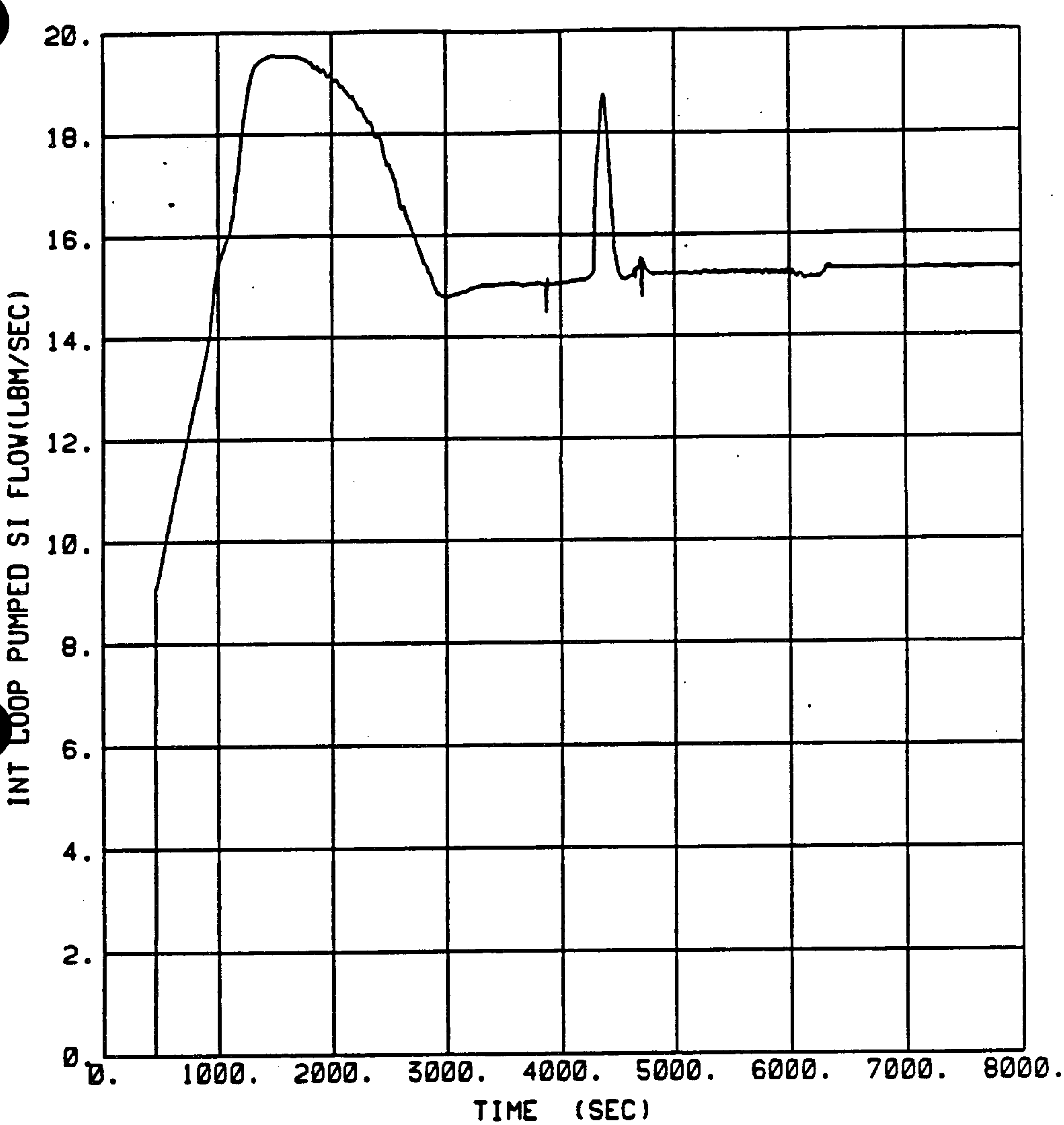
FIGURE 15.6.2-30



H. B. ROBINSON UNIT 2

CORE MIXTURE LEVEL
1-INCH COLD LEG BREAK - 60% POWER

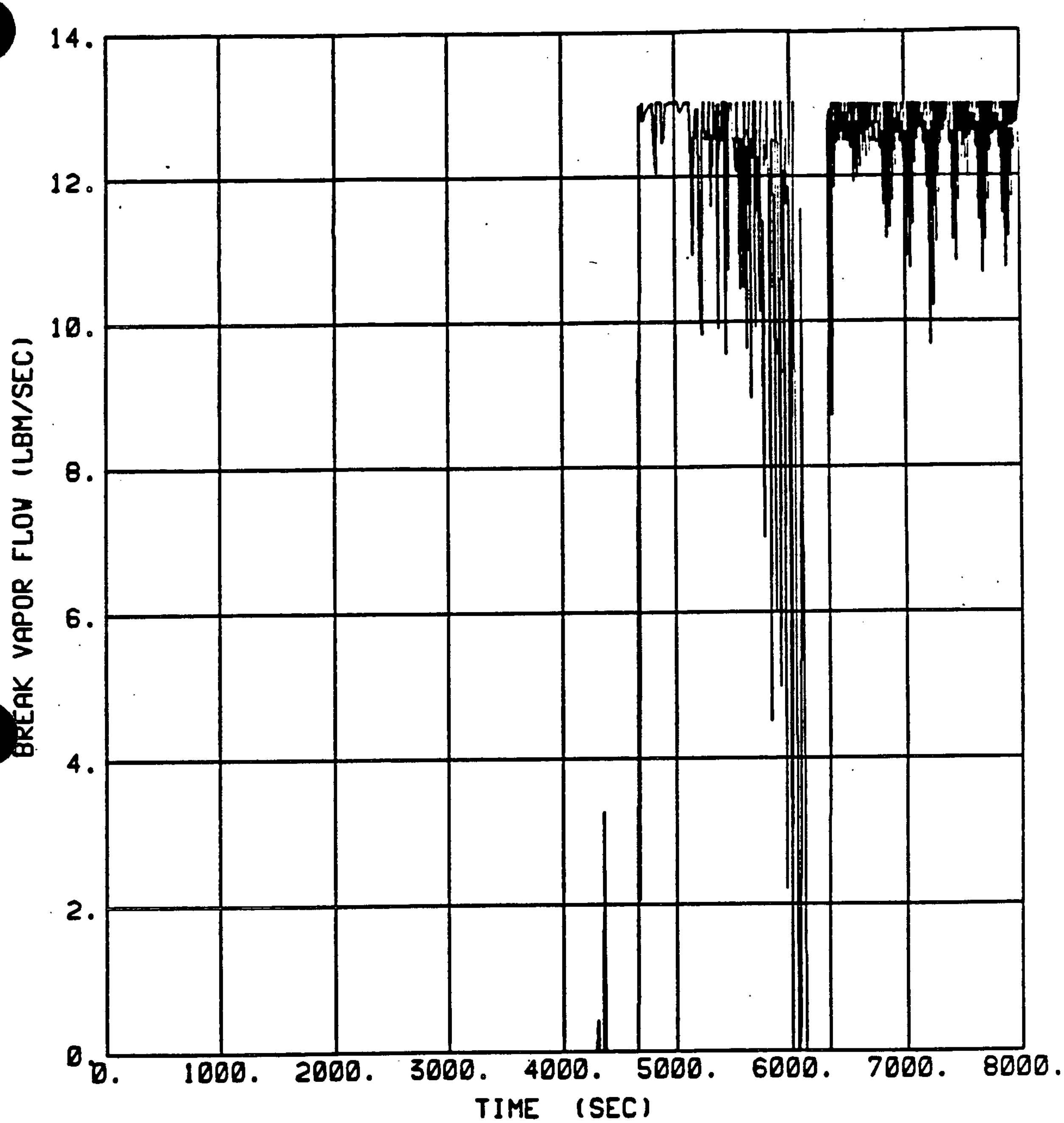
FIGURE 15.6.2-31



H. B. ROBINSON UNIT 2

INTACT LOOP PUMPED SI FLOW
1-INCH COLD LEG BREAK - 60% POWER

FIGURE 15.6.2-32



H. B. ROBINSON UNIT 2

BREAK VAPOR FLOW
1-INCH COLD LEG BREAK - 60% POWER

FIGURE 15.6.2-33

CPL-88-513

Westinghouse Electric Corporation   Power Systems

Nuclear Technology Systems Division

Box 355
Pittsburgh Pennsylvania 15230-0355

February 23, 1988

NS-OPLS-OPL-II-88-114

Ref: 1) CPL-86-531

Mr. S. R. Zimmerman, Manager
Nuclear Fuel
Carolina Power & Light Company
P. O. Box 1551
Raleigh, NC 27602

ATTENTION: T. Clements

CAROLINA POWER & LIGHT COMPANY
H. B. ROBINSON
Small Break LOCA at Reduced Power

Dear Mr. Zimmerman:

Attached is a safety evaluation that examines the effects of having only one high head safety injection (HHSI) pump providing pumped ECCS flow during a small break loss-of-coolant accident (LOCA) for the H. B. Robinson nuclear power plant. The safety evaluation reports the results of the Westinghouse small break LOCA emergency core cooling system (ECCS) evaluation model analyses performed at 60 percent of licensed core power for H. B. Robinson Unit 2 including the representation of Advanced Nuclear Fuels Corporation 15x15 fuel parameters.

Three small break LOCA analyses were performed using the NRC approved Westinghouse small break LOCA ECCS Evaluation Model incorporating the NOTRUMP analysis methodology. The analyses assumed a core power level corresponding to 102% of 1380 MWth (60% power operation).

These analyses were for 1.0-inch, 1.5-inch, and 2-inch equivalent diameter breaks. Of these three 60% power cases, the 2-inch case had the highest peak cladding temperature of 965.4°F.

The analysis of a 3-inch equivalent diameter break in the cold leg at 100 percent power (102% of 2300 Mwt) was also performed with only one HHSI pump available which resulted in a peak cladding temperature of 1771.6°F at the 12.0-foot elevation.

Small break LOCA analyses performed previously for H. B. Robinson (documented in reference 1) at 100% power with two HHSI pumps available resulted in the 3-inch case having the highest peak cladding temperature of the 2-inch, 3-inch, and 4-inch breaks. Lowering the safety injection flow by taking credit for only one HHSI pump lowers the limiting break size. Therefore, the limiting break size for 100% power with only one HHSI available will be less than or equal to 3 inches.

The peak cladding temperature for the 3-inch case at 60% power will be lower than the peak cladding temperature for the 3-inch case at 100% power. Because the 3-inch case at 100% power did not exceed the limits of 10CFR50.46, the 3-inch case at 60% power will not exceed the limits. Therefore, the 3-inch case was not performed for the 60% power level.

The results of the analyses and evaluations show that the H.B.Robinson Unit 2 Nuclear Power Plant may be started and operated at 60 percent of licensed core power in compliance with the requirements of 10CFR50.46 when flow from only one high head safety injection pump is available.

Any questions regarding this evaluation should be directed to the undersigned.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

DL Cecchett - for
G. O. Percival, Manager
Carolina Area

D.L. Cecchett/cgl
Attachment

cc: L. H. Martin (CP&L) 1L, 1A
T. M. Dresser (CP&L) 1L, 1A
B. G. Rieck (CP&L - HBR) 1L, 1A
B. M. Slone (CP&L - HBR) 1L, 1A
R. J. Muth (CP&L - HBR) 1L, 1A
R. S. Pollock (W - Raleigh) 1L, 1A

ATTACHMENT A

JUSTIFICATION FOR STARTUP AND 60% POWER OPERATION OF H. B. ROBINSON UNIT 2 NUCLEAR POWER PLANT WITH 15 X 15 ADVANCED NUCLEAR FUELS COMPANY FUEL IN CONFORMANCE WITH THE 10CFR50.46 ACCEPTANCE CRITERIA

Westinghouse Electric Corporation
Nuclear Technology Systems Division
Nuclear Safety Department
Safeguards Engineering and Development

February 1988

ATTACHMENT A

JUSTIFICATION FOR STARTUP AND 60% POWER OPERATION WITH
15 X 15 ADVANCED NUCLEAR FUELS COMPANY FUEL
IN CONFORMANCE WITH THE 10CFR50.46 ACCEPTANCE CRITERIA
FOR THE
H. B. ROBINSON UNIT 2 NUCLEAR POWER PLANT

I. BACKGROUND

In the process of reviewing plant documents for formulating a response to NRC letter NRC-88-017, it was discovered that at least one postulated single failure event exists which could result in the loss of the ability to automatically start two high head safety injection pumps. Upon thorough review and examination of the problem, failure events were postulated in which flow from only one high head safety injection pump would be available during a loss-of-coolant-accident (LOCA).

A small break LOCA analysis was performed in 1986 for H.B.Robinson using the NRC-approved Westinghouse small break LOCA ECCS Evaluation Model incorporating the NOTRUMP analysis methodology. The spectrum of 2-inch, 3-inch, and 4-inch equivalent diameter cold leg small break analyses resulted in the highest calculated peak cladding temperature of 1398°F for the 3-inch break. The analysis was performed assuming a core power level corresponding to 102 % of 2300 MWth at a total core peaking factor (FQT) of 2.32 with a hot channel enthalpy rise factor of 1.65. The analysis assumed flow was delivered automatically from two high head safety injection pumps.

A safety evaluation to justify the resumption of operation of the H.B.Robinson Unit 2 nuclear power plant to a maximun of 60% power with 15x15 fuel manufactured by the Advanced Nuclear Fuels Corporation was performed assuming only one high head safety injection pump was operational. The evaluation was based upon small break LOCA analyses using the Westinghouse Small Break LOCA ECCS Evaluation Model incorporating the NOTRUMP analysis methodology.

II. METHOD OF EVALUATION

As a technical basis for the safety evaluation, analysis of postulated small break LOCA scenarios were performed assuming automatic safety injection flow delivery from only one high head safety injection pump.

The small break LOCA analyses were performed using the Westinghouse Small Break LOCA ECCS Evaluation Model incorporating the NOTRUMP analysis methodology to justify 60% power level operation for the H.B.Robinson Unit 2 nuclear power plant. The analyses assumed conservatively low estimate of the amount of safety injection flow delivered from one high head safety injection pump. The analysis model utilized the input developed in 1986 for the Carolina Power & Light company for the H.B.Robinson Unit 2 nuclear power plant performed to address the requirements of NUREG-0737 II.K.3.31.

Four small break LOCA analyses were performed using the NRC approved Westinghouse small break LOCA ECCS Evaluation Model incorporating the NOTRUMP analysis methodology. One of the analyses, a 3-inch equivalent diameter cold leg break, assumed 100% power. Three other analyses assumed a core power level corresponding to 102% of 1380 MWth (60% power operation). These analyses were for 1.0-inch, 1.5-inch, and 2-inch equivalent diameter cold leg breaks.

III. EVALUATION RESULTS

Of the three 60% power cases, the 2-inch case resulted in the highest peak cladding temperature of 965.4°F.

The analysis of the 3-inch equivalent diameter break at 100 percent power (102% of 2300 Mwt) resulted in a peak cladding temperature of 1771.6°F.

Small break LOCA analyses performed previously for H. B. Robinson at 100% power with two HHSI pumps available resulted in the 3-inch case having the highest peak cladding temperature of the 2-inch, 3-inch, and 4-inch breaks. Lowering the safety injection flow by taking credit for only one HHSI pump lowers the limiting break size. Therefore, the limiting break size for 100% power with only one HHSI available will be less than or equal to 3 inches.

The peak cladding temperature for the 3-inch case at 60% power will be lower than the peak cladding temperature for the 3-inch case at 100% power. Because the 3-inch case at 100% power did not exceed the limits of 10CFR50.46, the 3-inch case at 60% power will not exceed the limits. Therefore, the 3-inch case was not performed for the 60% power level.

The results of the analyses and evaluations show that the H.B.Robinson Unit 2 Nuclear Power Plant may be started and operated at 60 percent of licensed core power in compliance with the requirements of 10CFR50.46 when flow from only one high head safety injection pump is available.

A more detailed set of the small break LOCA analyses results are provided in Attachment B.

ATTACHMENT B

H.B. ROBINSON UNIT 2

SMALL BREAK LOCA ANALYSIS RESULTS

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 (RCS) pressure boundary. A major pipe break (large break) is defined as a rupture with a total cross sectional area equal to or greater than 1.0 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:

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

b. 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% are not exceeded during or after quenching.

c. 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% of the total amount of Zircaloy in the reactor.

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 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.

n the RCS depressurizes to 615 psia, the cold leg accumulators gin 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 trip 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 s-of-coolant accidents due to small breaks in the Reactor lant 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 countercurrent 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 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.

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 small break LOCA power shape 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 conservative injection flow from one High Head Safety Injection (HHSI) pump. The conservative delivery data incorporates the standard FSAR ECCS assumption of minimum safeguards. The effect 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 busses.

The hydraulic analyses are typically performed with the NOTRUMP code using 102% of the licensed core power plus the 8 MWt energy added by the three reactor coolant pumps. However, due to the degraded HHSI system, only those results for the 3-inch break were performed at full power. The analyses for the 2, 1.5 and 1-inch break were performed at 60% of licensed core power. The core thermal transient analyses using LOCTA-IV were performed in a similar manner, i.e., 102% of licensed core power for the 3-inch break and 102% of 60% of licensed core power for the 2, 1.5 and 1-inch break. The LOCTA-IV core thermal analyses incorporated Exxon 15x15 fuel data which 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 that the limits of 10CFR 50.46 will not be exceeded at 60% of licensed core power operation. The results of these analyses are summarized in Tables 15.6.2-3 and 15.6.2-4. Figures 15.6.2-1 through 15.6.2-8b and 15.6.2-25 through 15.6.2-33 present the principal parameters of interest for the small break ECCS analyses. For the 2-inch 1.5-inch and 1-inch break sizes analyzed at 60% power, the following transient parameters are included:

a. RCS Pressure
b. Core Mixture Height
c. Hot Spot Clad Temperature
d. Intact Loop Pumped SI Flow
e. Break Vapor Flow

As indicated in the results for clad heat up, the 3-inch case is limiting. However, the 3-inch case is for 102% of licensed core power. For the limiting break size analyzed (3-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 1772°F. These results are well below all Acceptance Criteria limits of 10 CFR 50.46 and demonstrate acceptability of operation with one HHSI pump at 60% of licensed core power.

TABLE 15.6.2-1

Input Parameters Used in the SBLOCA Analysis

Core Power[1]	1407.6 MWt
Pump Heat	8 MWt
NSSS Power	1415.6 MWt[2]
Peak Linear Power (includes 102% factor)	13.197 kW/ft
Total Peaking Factor, F	2.32[3]
Power Shape	Fig. 15.6.2-13
Fuel Assembly Array	Exxon 15x15
Nominal Accumulator Water Volume	825 ft /accum.
Nominal Accumulator Tank Volume	1200 ft /accum.
Minimum Accumulator Gas Pressure	615 psia
Pumped Safety Injection Flow	Fig. 15.6.2-12
[illegible]am Generator Initial Pressure	863 psia
Auxiliary Feedwater Flow	41.22 lb/sec/SG
Steam Generator Tube Plugging Level	5%

1 - 2% has been added to this power to account for calorimetric uncertainty

2 - As noted in the text, the 3-inch break was performed for 100% power or NSSS power of 2354 MWt

3 - 2.32 is for 100% Power. At reduced power levels the allowable peaking factor will increase in accordance with plant Technical Specifications.

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

Event	100% Power 3 in (sec)	-------- 60% Power------- 2 in (sec)	1.5 in (sec)	1 in (sec)
Start	0.0	0.0	0.0	0.0
Reactor Trip	5.79	12.00	21.31	47.14
S-signal	9.85	19.39	33.84	76.77
Loop Seal Venting	450.3	1336.6	2088.0	4286.1
Top of Core Uncovered	798.2	1312.9	2048.3	4560.1
Accumulator Injection	1099.6	3006.9	N/A	N/A
Maximum Core Uncovery	1182.1	3005.9	2085.0	4645.5
Peak Clad Temperature Occurs	1229.9	3101.6	2352.6	N/A
Top of Core Covered	N/A	3465.3	2355.4	4657.7

TABLE 15.6.2-4

Small Break LOCA Fuel Cladding Results

Results	100% Power 3 in	-------- 60% Power 2 in	1.5 in	------- 1 in
Peak clad temperature (°F)	1771.6	965.4	743.5	N/A
Peak clad temperature location (ft)	12.0	12.0	12.0	N/A
Local Zr/H_2O reaction, maximum (%)	2.31	0.20	0.20	N/A
Local Zr/H_2O location (ft)	12.0	12.0	12.0	N/A
Total Zr/H_2O reaction (%)	<0.3	<0.3	<0.3	N/A
Hot rod burst time (sec)	N/A	N/A	N/A	N/A
Hot rod burst location (ft)	N/A	N/A	N/A	N/A

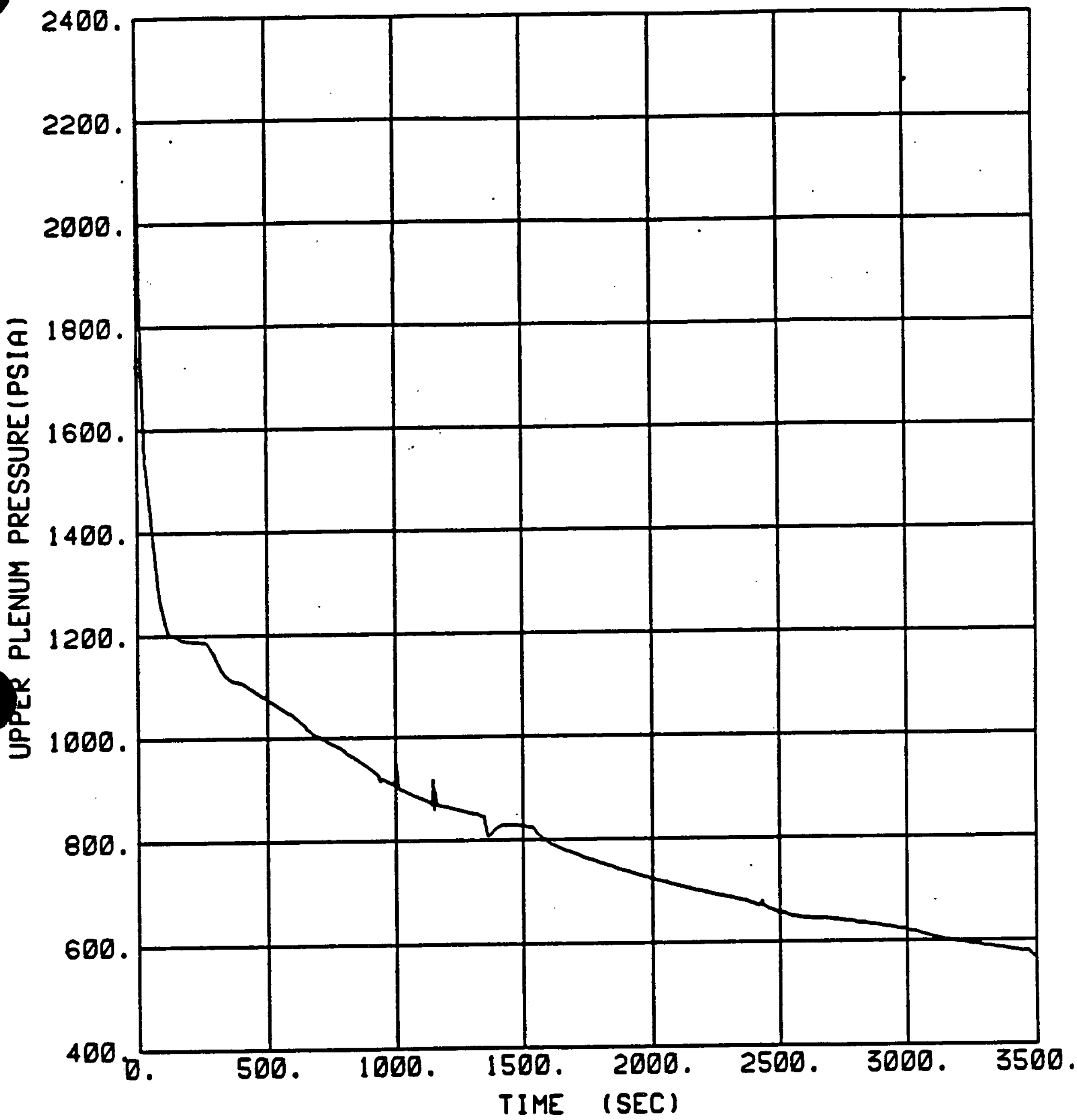
ERENCES FOR SECTION 15.6.2

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, 1974.

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 ECCS Evaluation Model Using the NOTRUMP Code", WCAP-10081-A, August 1985.

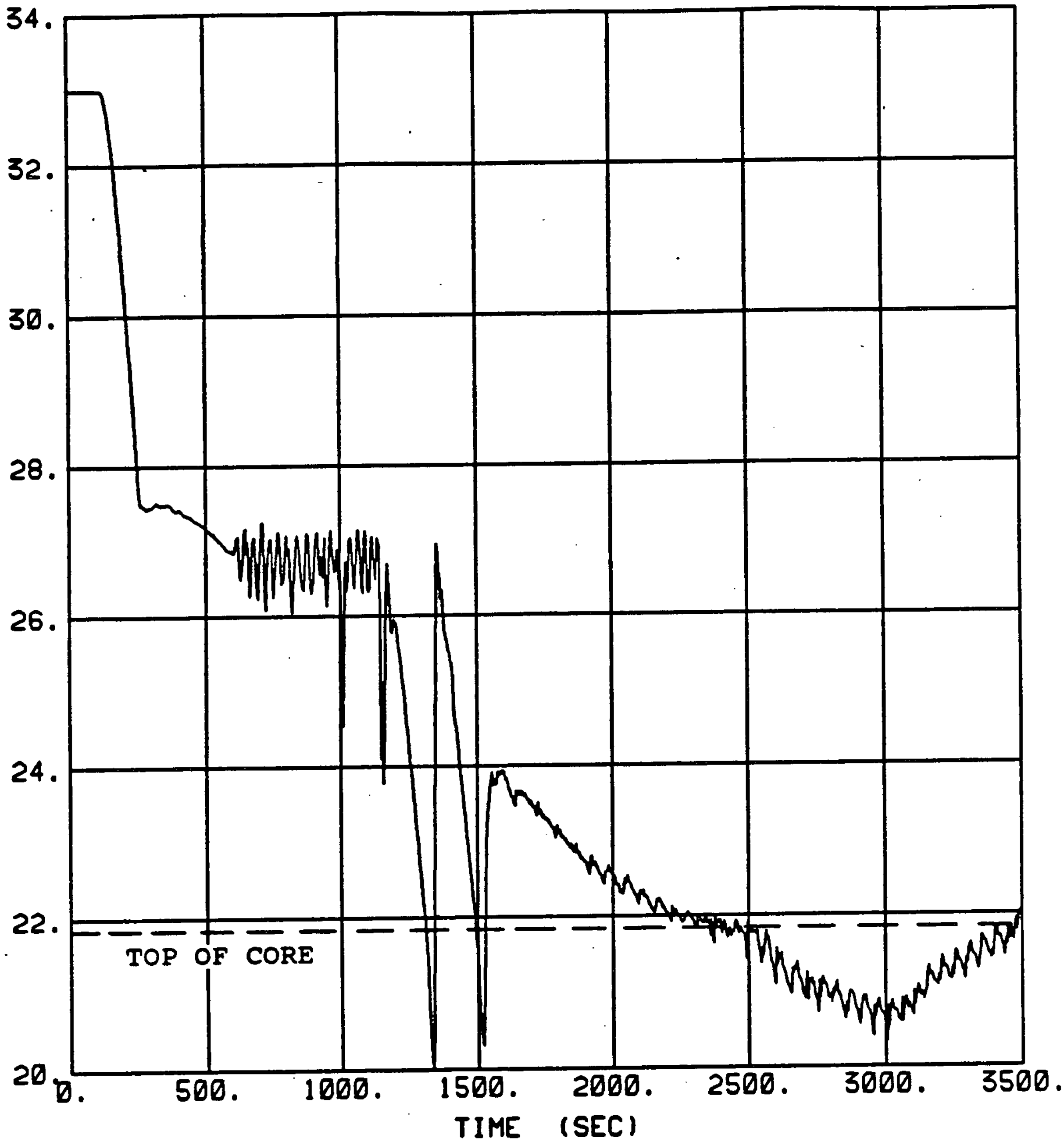
4. Bordelon, F. M., et. al., "LOCTA-IV Program: Loss of Coolant Transient Analysis", WCAP-8301, (Proprietary) and WCAP-8305, (Non-Proprietary), June 1974.



H. B. ROBINSON UNIT 2

UPPER PLENUM PRESSURE
2-INCH COLD LEG BREAK - 60% POWER

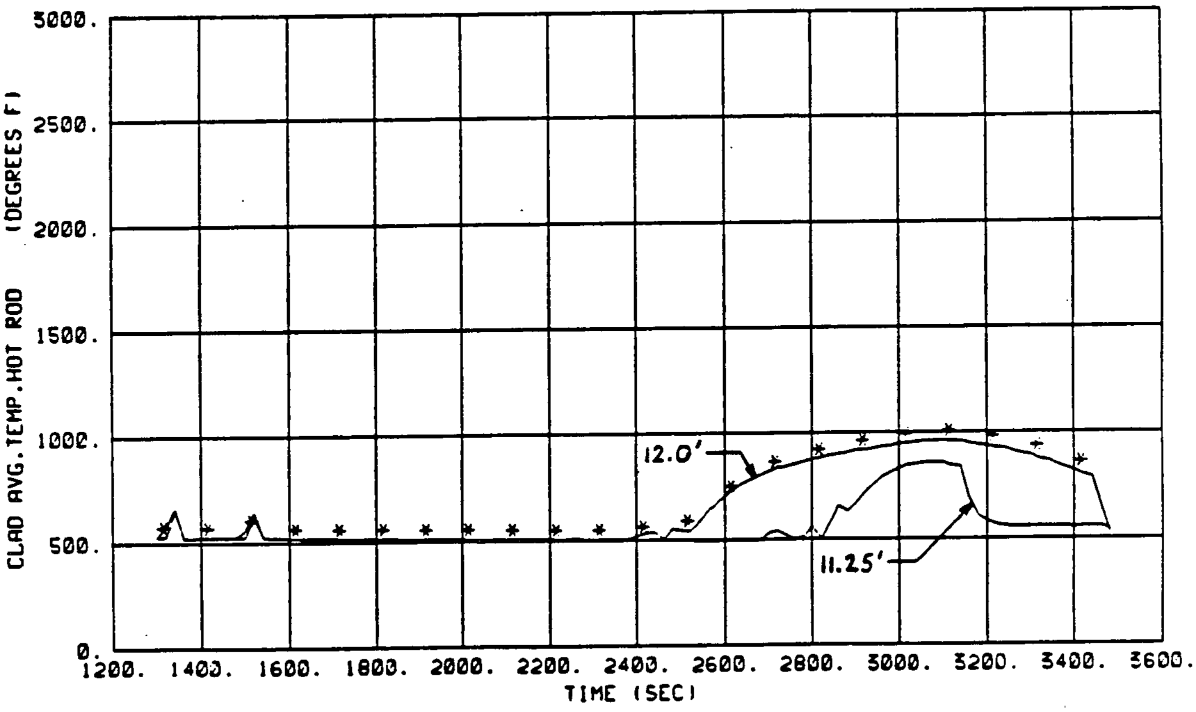
FIGURE 15.6.2-1



H. B. ROBINSON UNIT 2

CORE MIXTURE LEVEL
2-INCH COLD LEG BREAK - 60% POWER

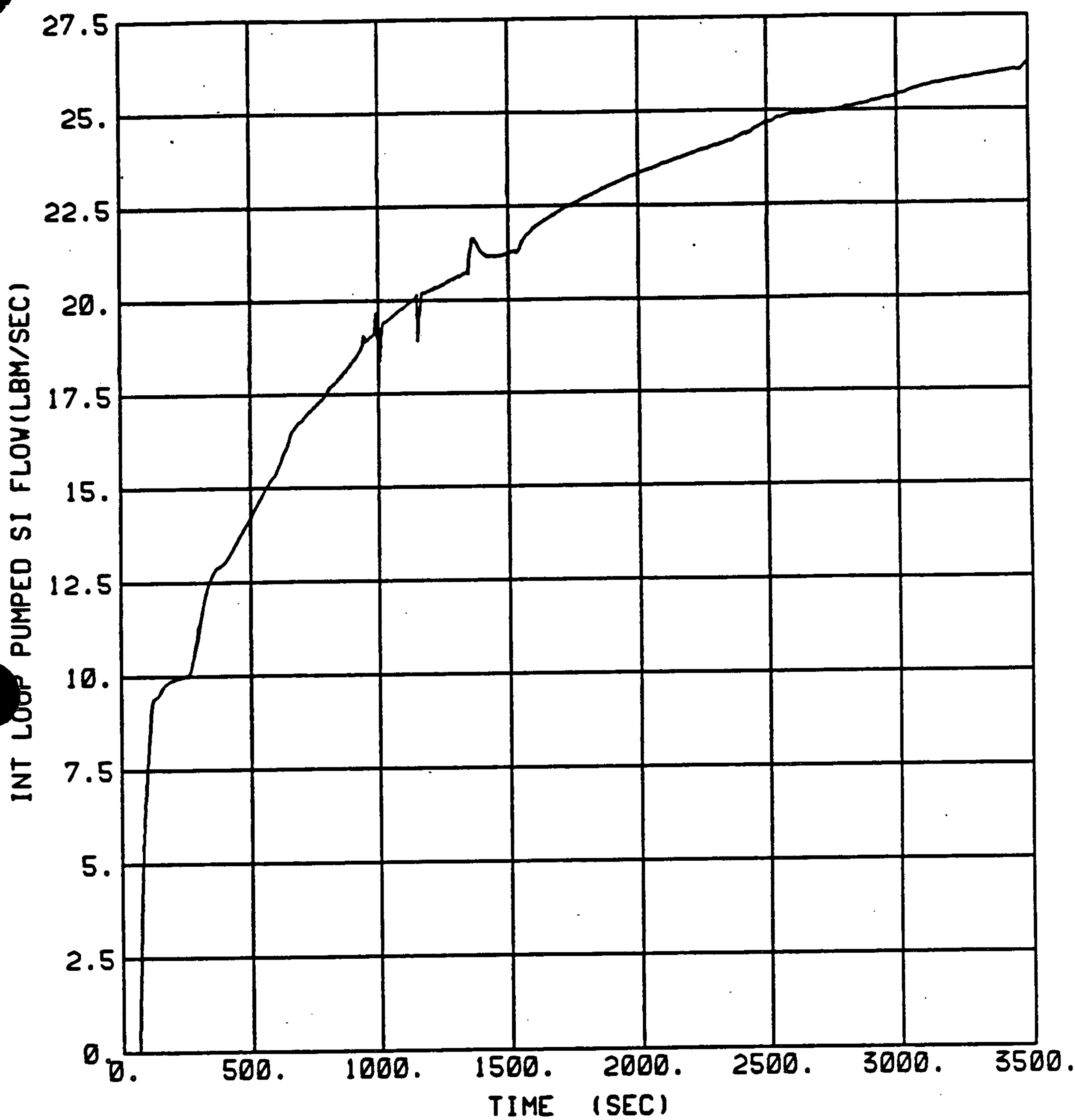
FIGURE 15.6.2-2



H. B. ROBINSON UNIT 2

HOT SPOT CLAD TEMPERATURE
2-INCH COLD LEG BREAK - 60% POWER

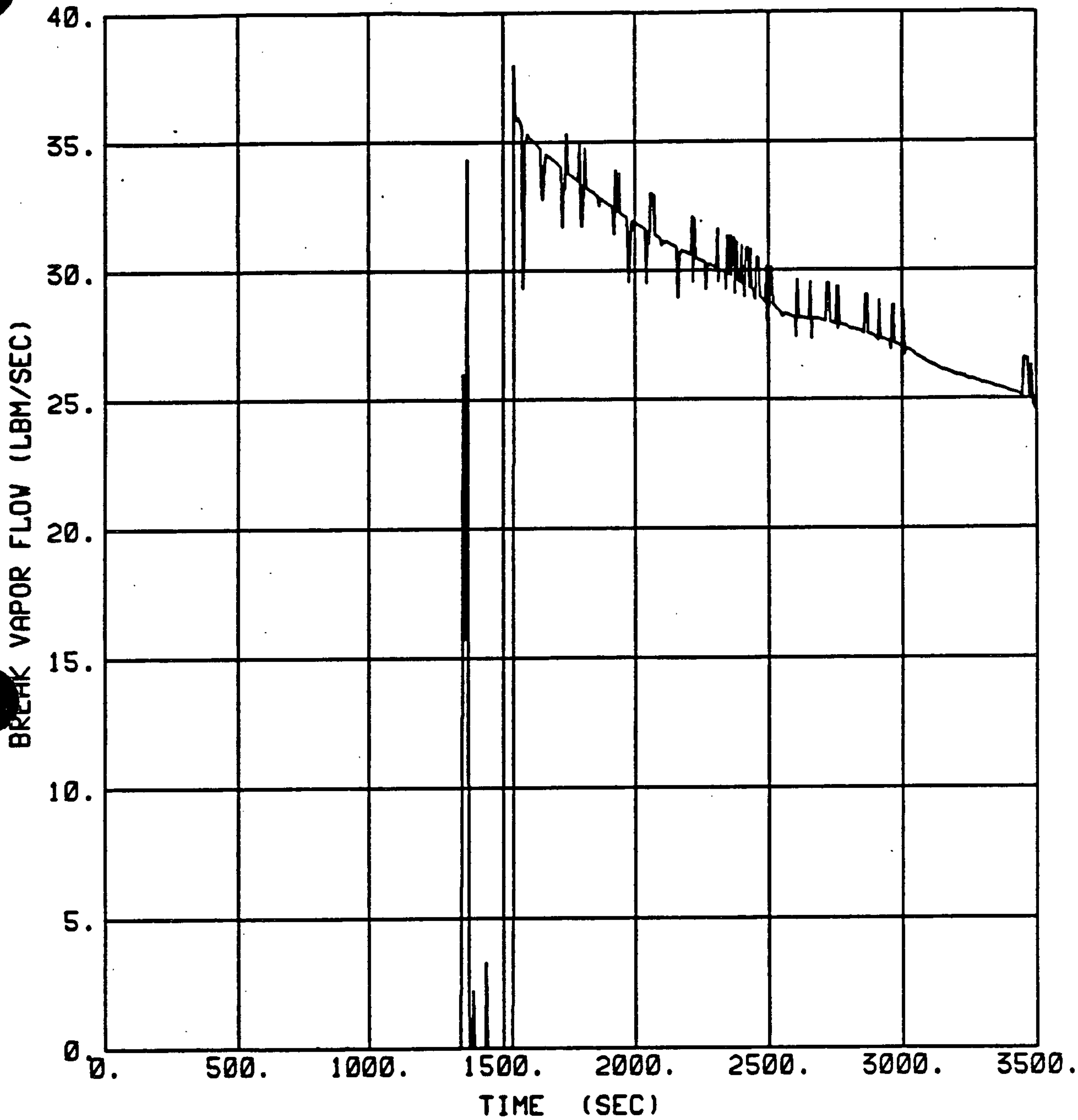
FIGURE 15.6.2-2a



H. B. ROBINSON UNIT 2

INTACT LOOP PUMPED SI FLOW
2-INCH COLD LEG BREAK - 60% POWER

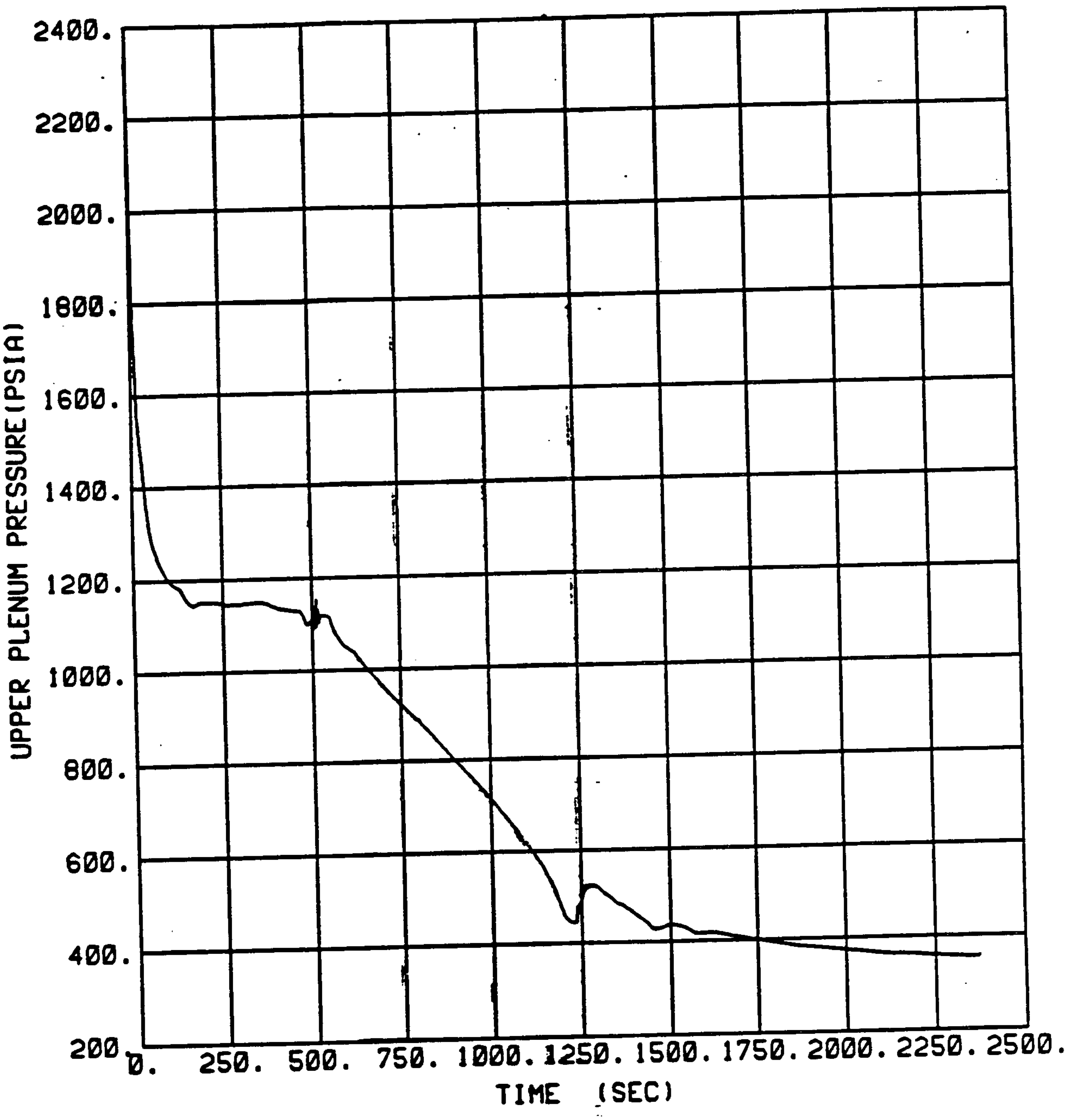
FIGURE 15.6.2-2b



H. B. ROBINSON UNIT 2

BREAK VAPOR FLOW
2-INCH COLD LEG BREAK - 60% POWER

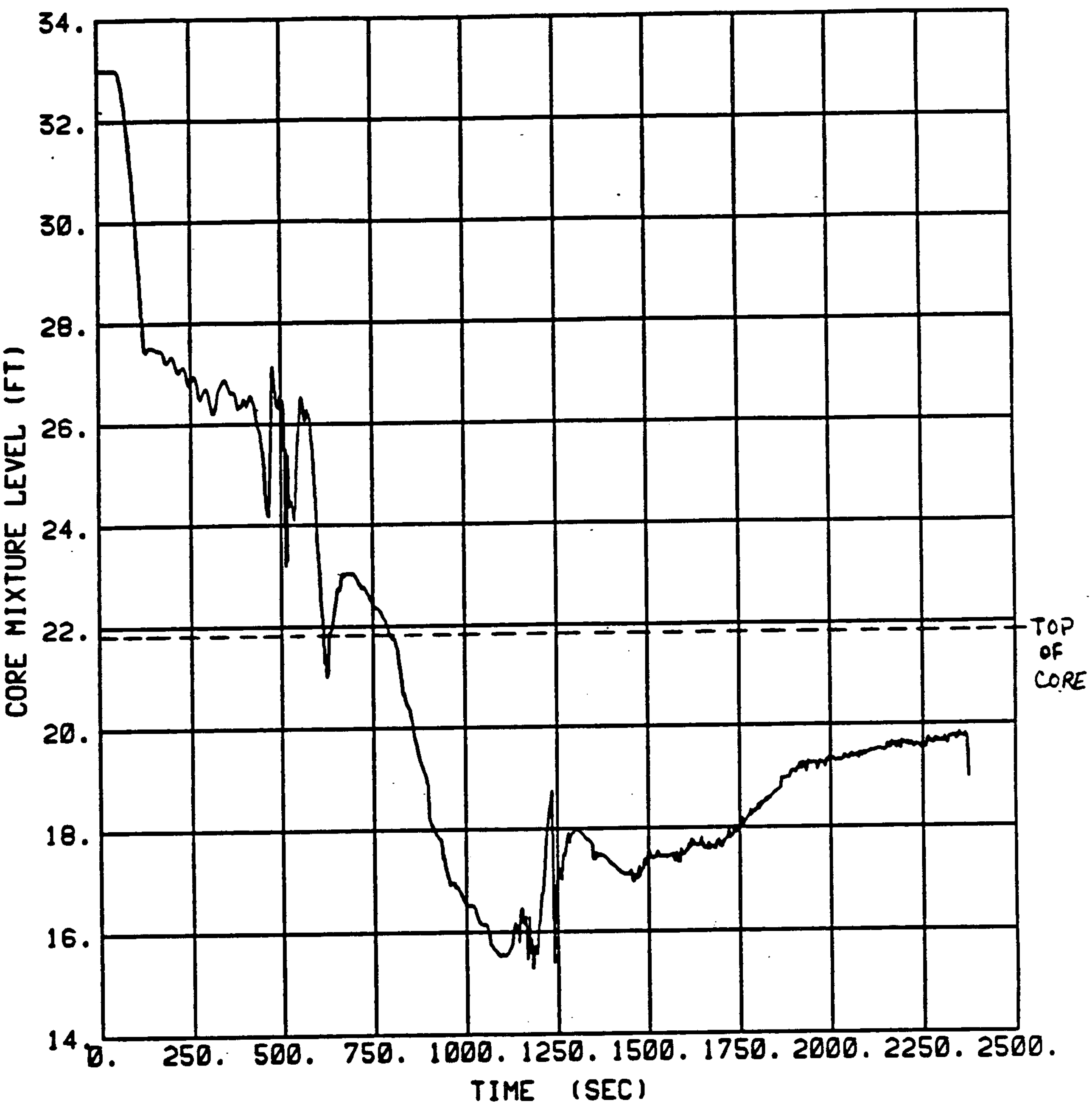
FIGURE 15.6.2-2c



H. B. ROBINSON UNIT 2

UPPER PLENUM PRESSURE
3-INCH COLD LEG BREAK

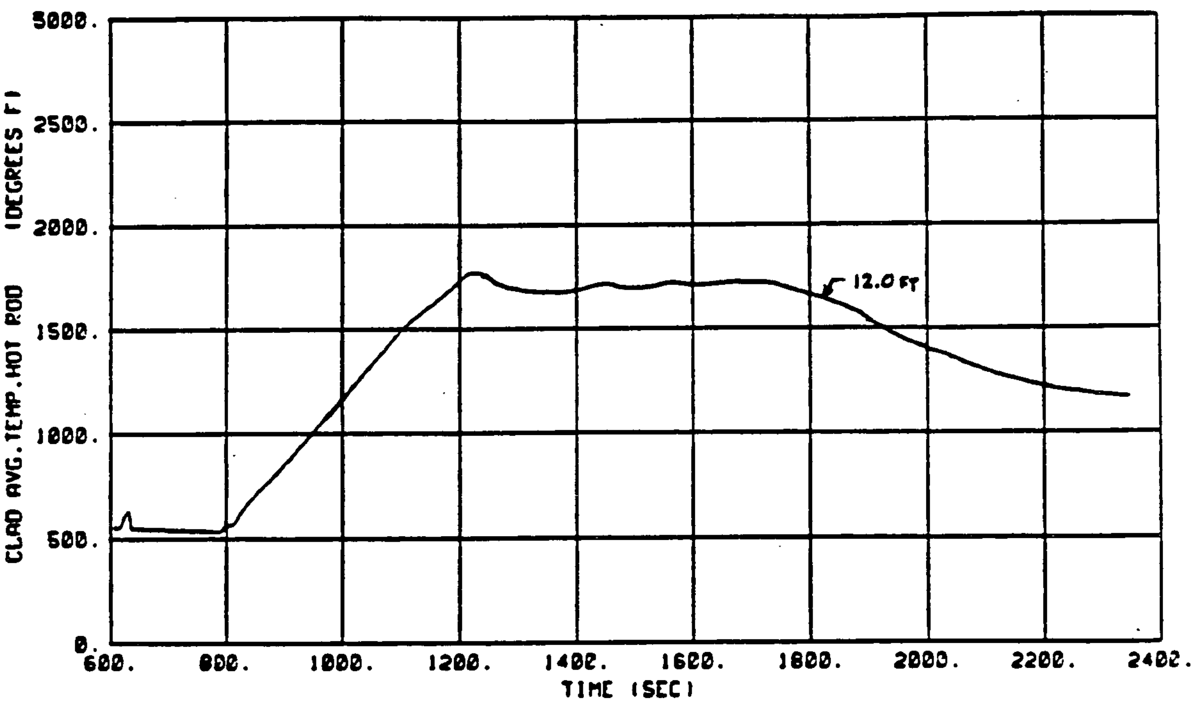
FIGURE 15.6.2-3



H. B. ROBINSON UNIT 2

CORE MIXTURE LEVEL
3-INCH COLD LEG BREAK

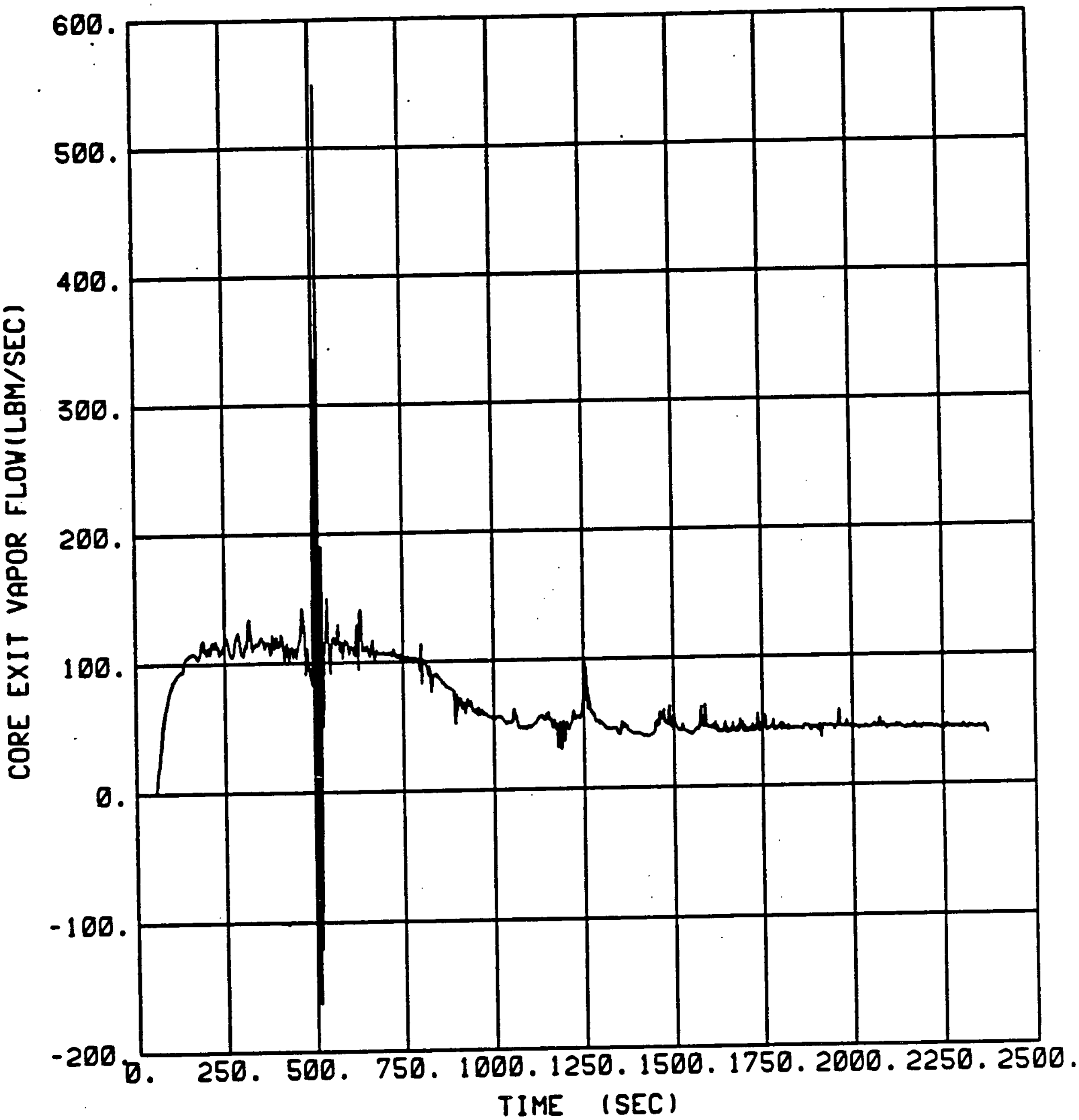
FIGURE 15.6.2-4



H. B. ROBINSON UNIT 2

HOT SPOT CLAD TEMPERATURE
3-INCH COLD LEG BREAK

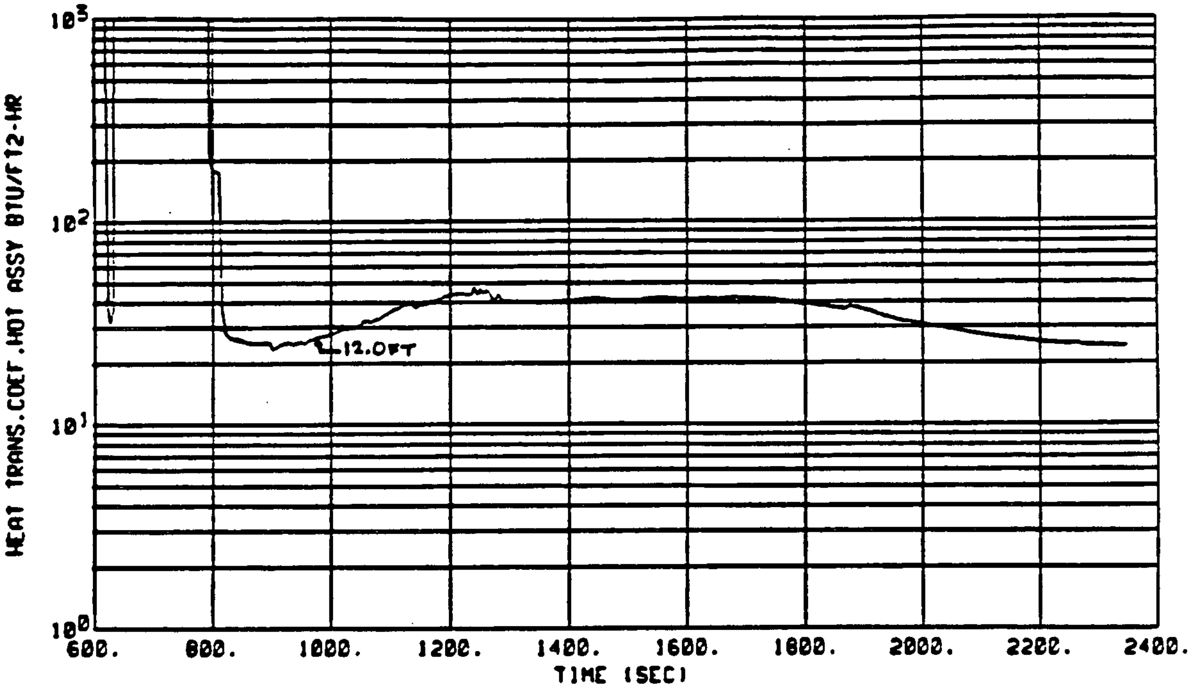
FIGURE 15.6.2-5



H. B. ROBINSON UNIT 2

CORE EXIT VAPOR FLOW
3-INCH COLD LEG BREAK

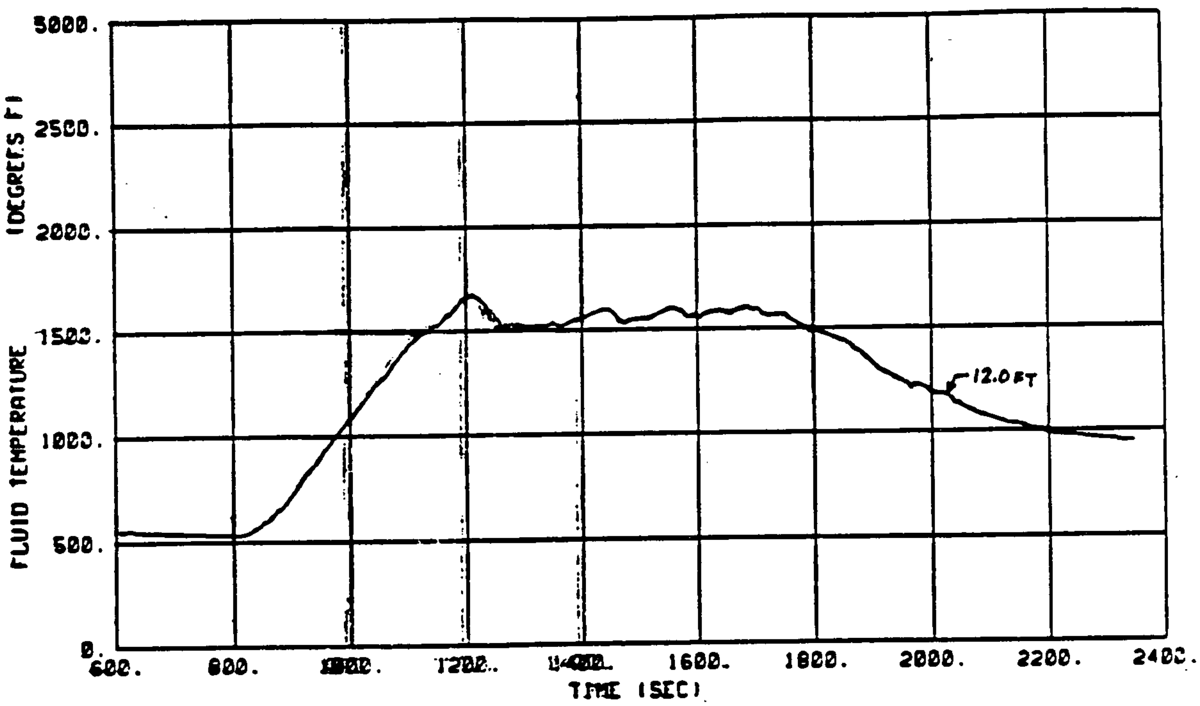
FIGURE 15.6.2-6



H. B. ROBINSON UNIT 2

CORE HEAT TRANSFER COEFFICIENT
3-INCH COLD LEG BREAK

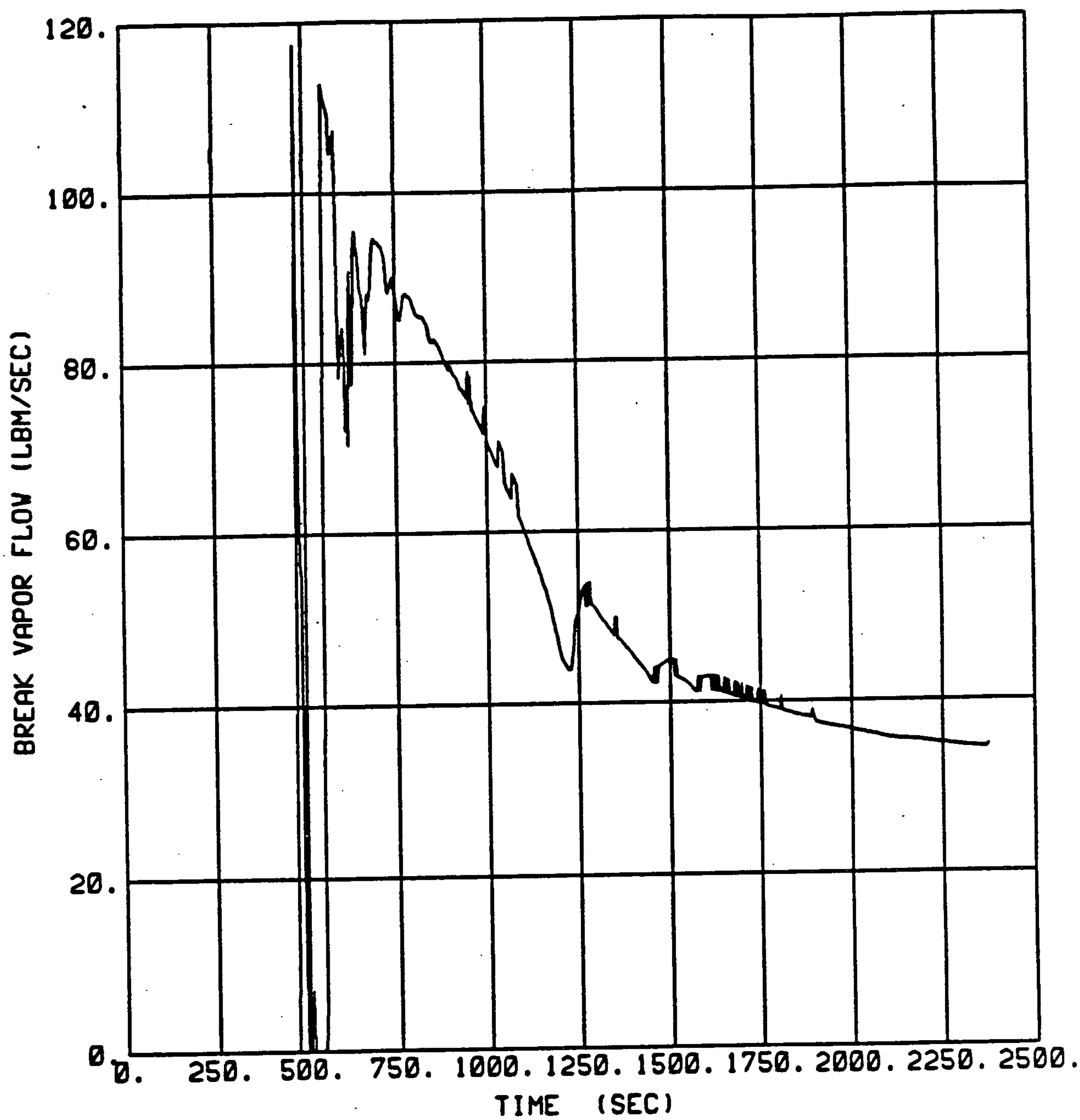
FIGURE 15.6.2-7



H. B. ROBINSON UNIT 2

HOT SPOT FLUID TEMPERATURE
3-INCH COLD LEG BREAK

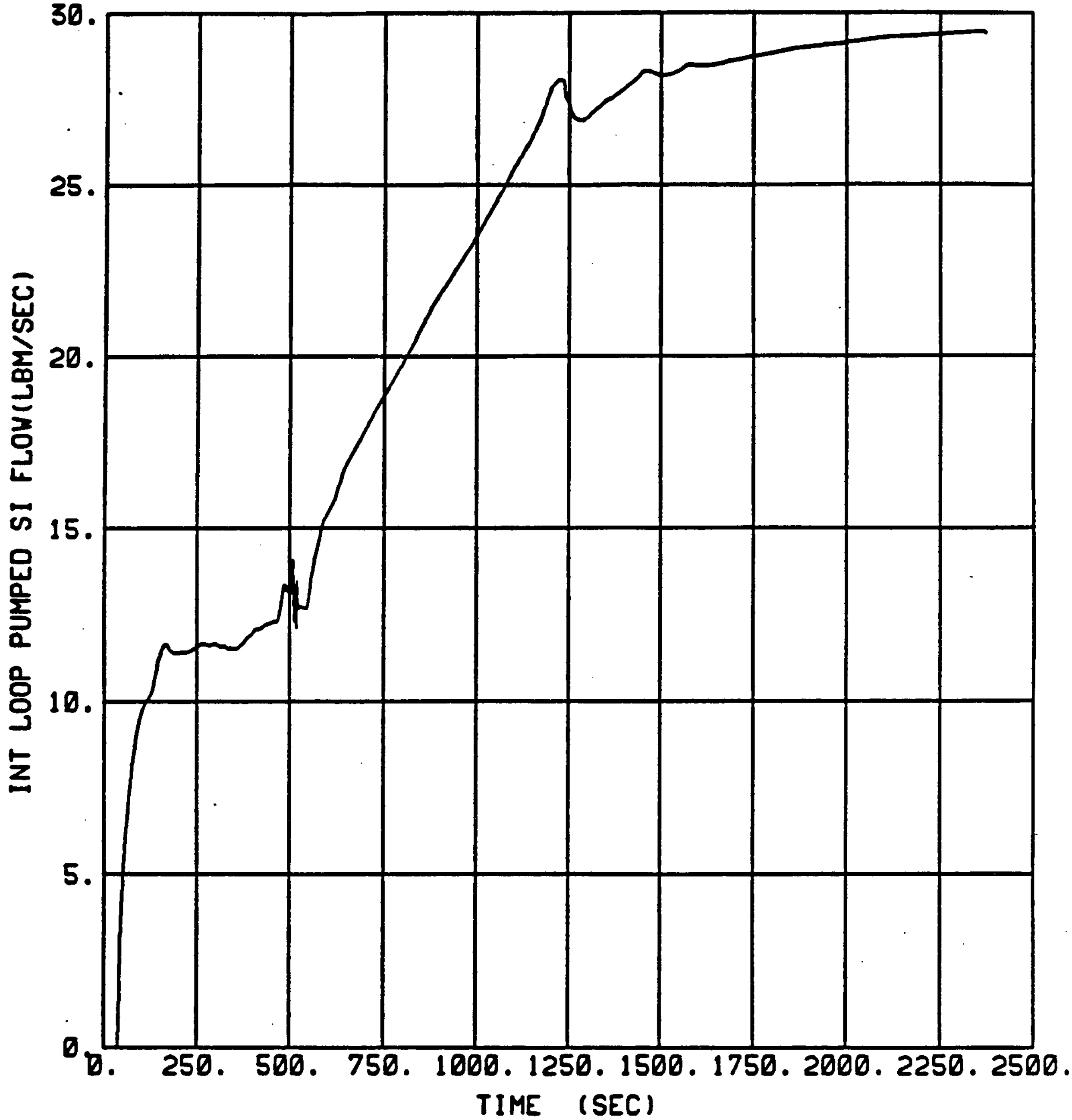
FIGURE 15.6.2-8



H. B. ROBINSON UNIT 2

VAPOR BREAK MASS FLOW RATE
3-INCH COLD LEG BREAK - 100% POWER

FIGURE 15.6.2 - 8a



H. B. ROBINSON UNIT 2

INTACT LOOP PUMPED SAFETY INJECTION MASS FLOW RATE
3-INCH COLD LEG BREAK - 100% POWER

FIGURE 15.6.2 - 8b

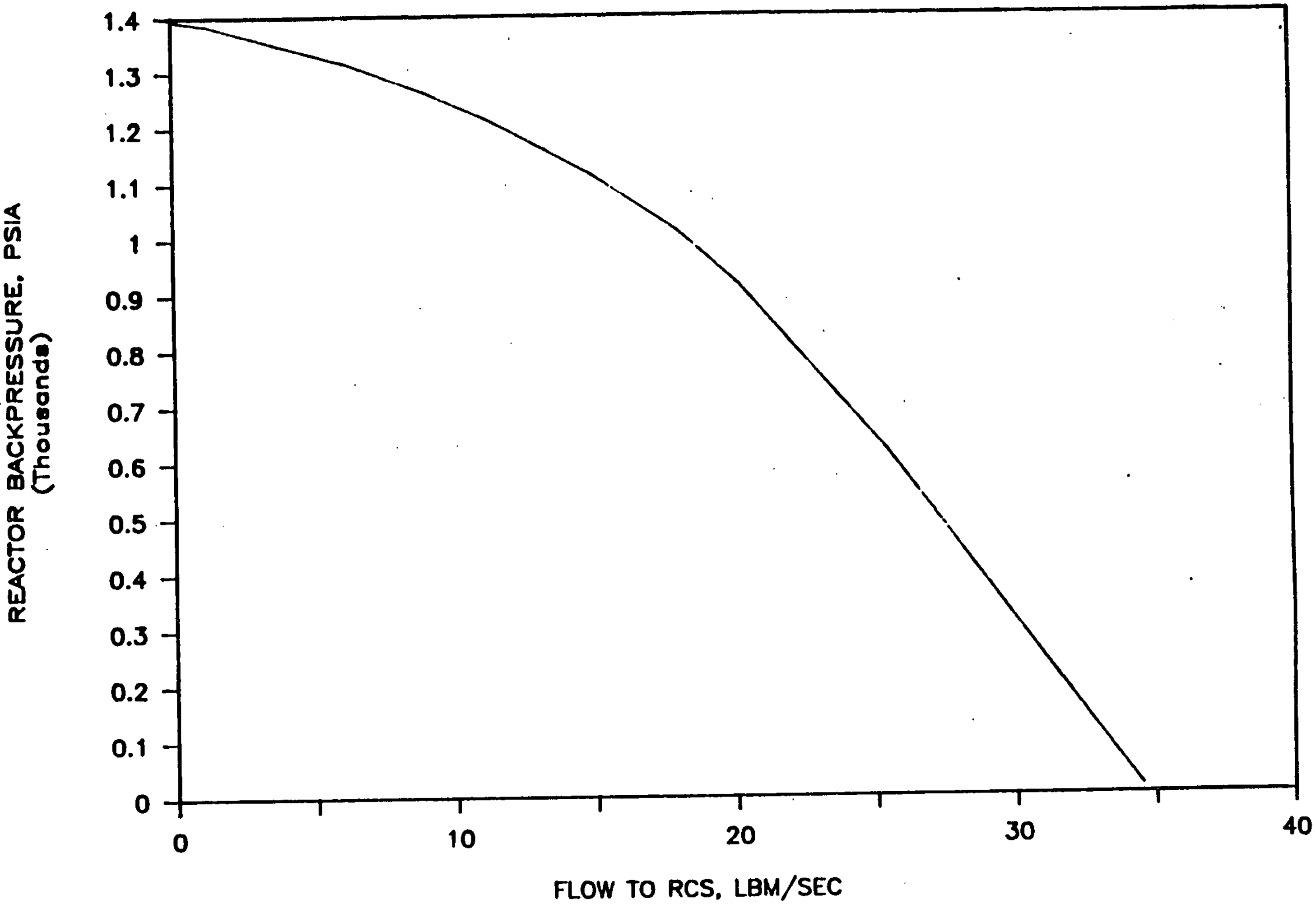


Figure 15.6.2-12 H. B. Robinson Pumped Safety Injection Flow

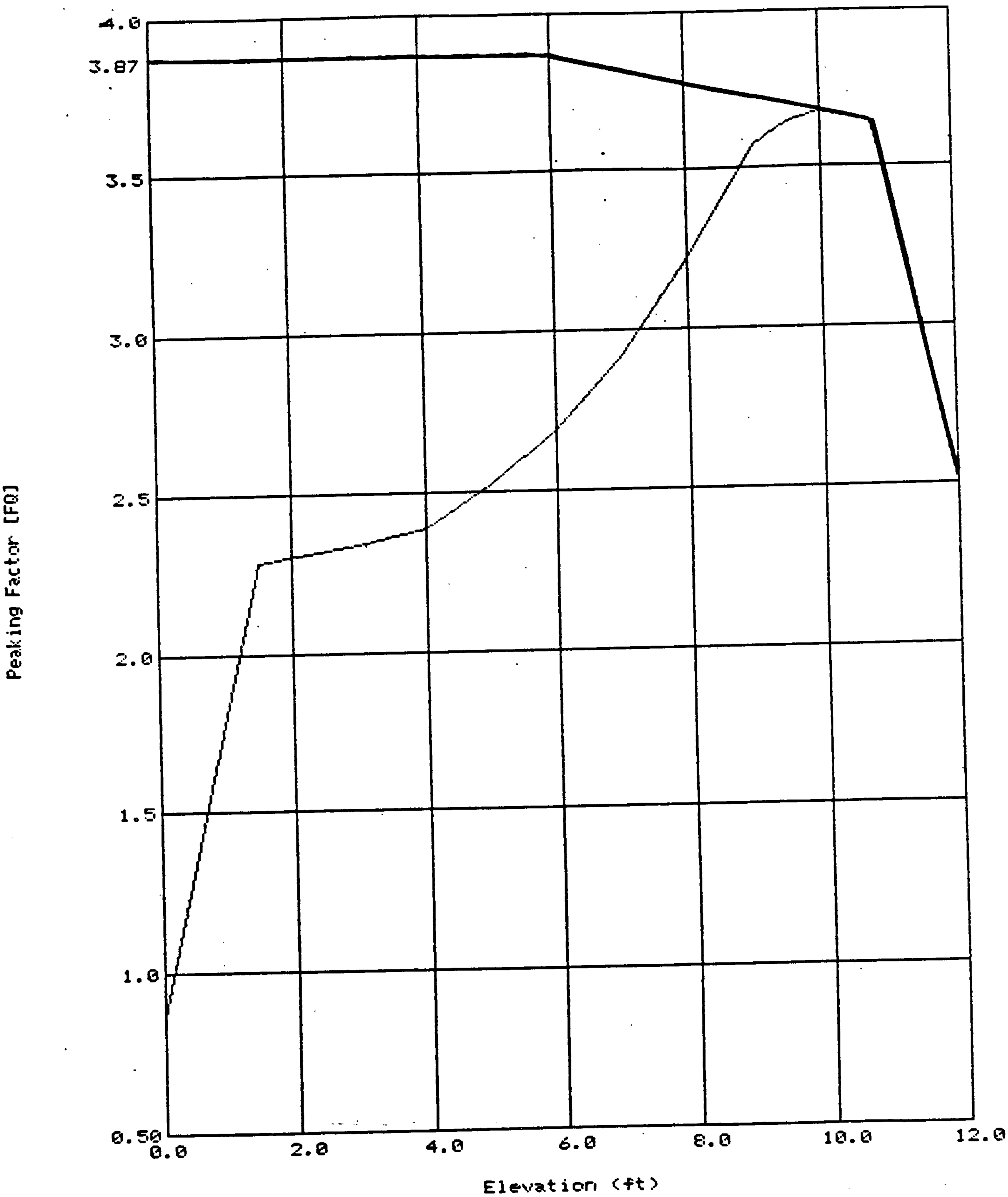


Figure 15.6.2-13 H.B. Robinson Small-Break LOCA Power Shape for 60% Power.

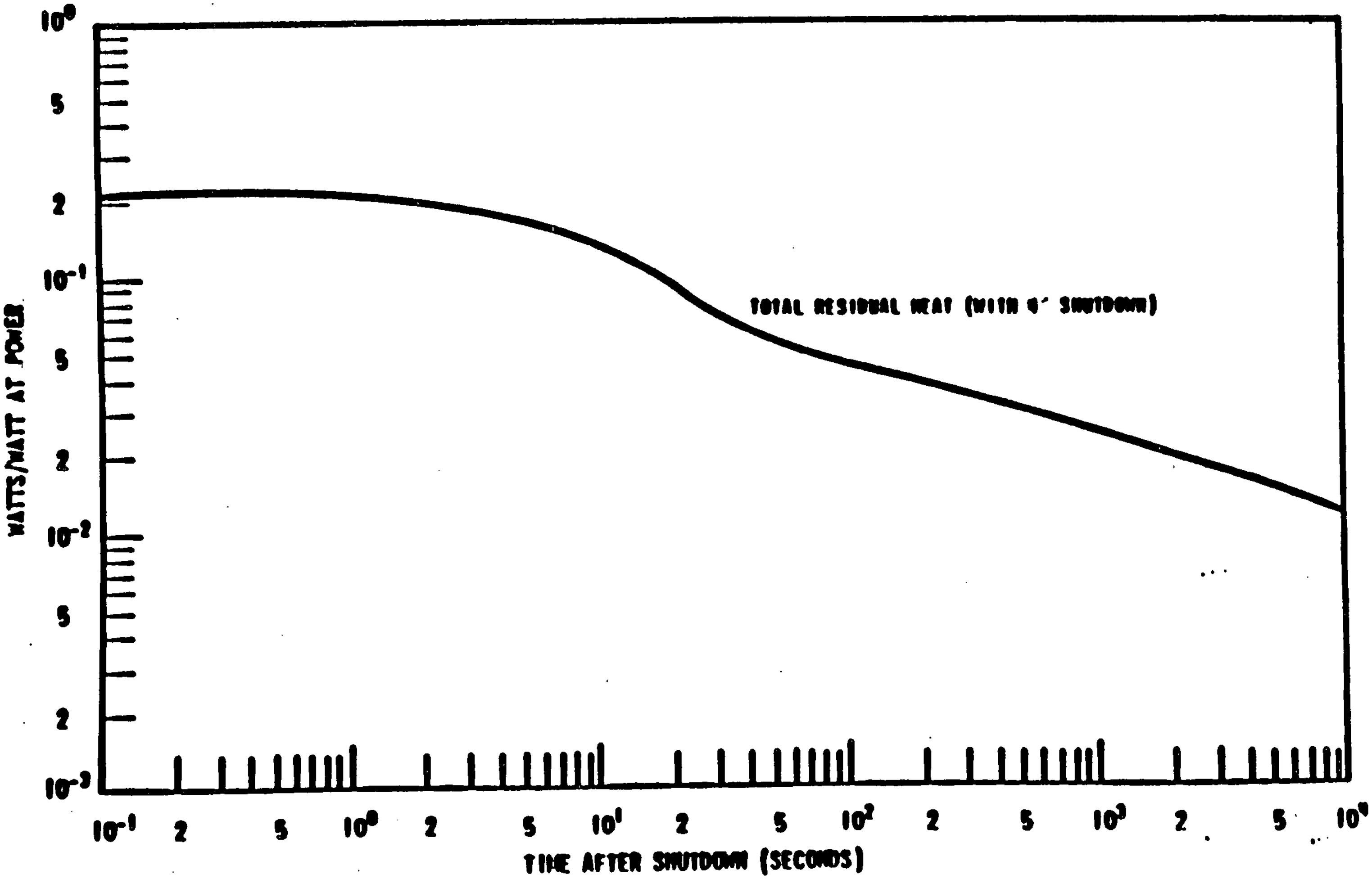
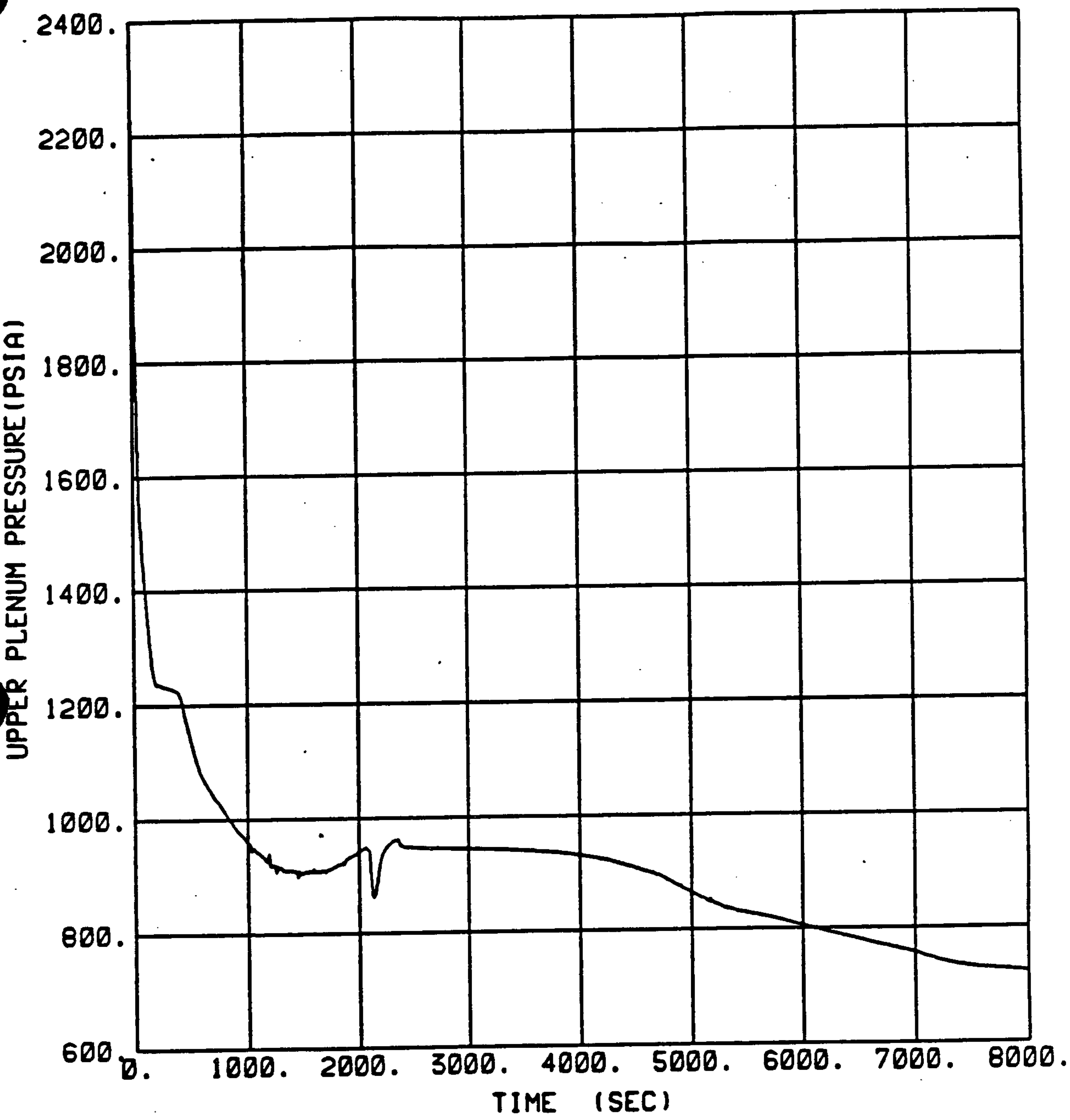


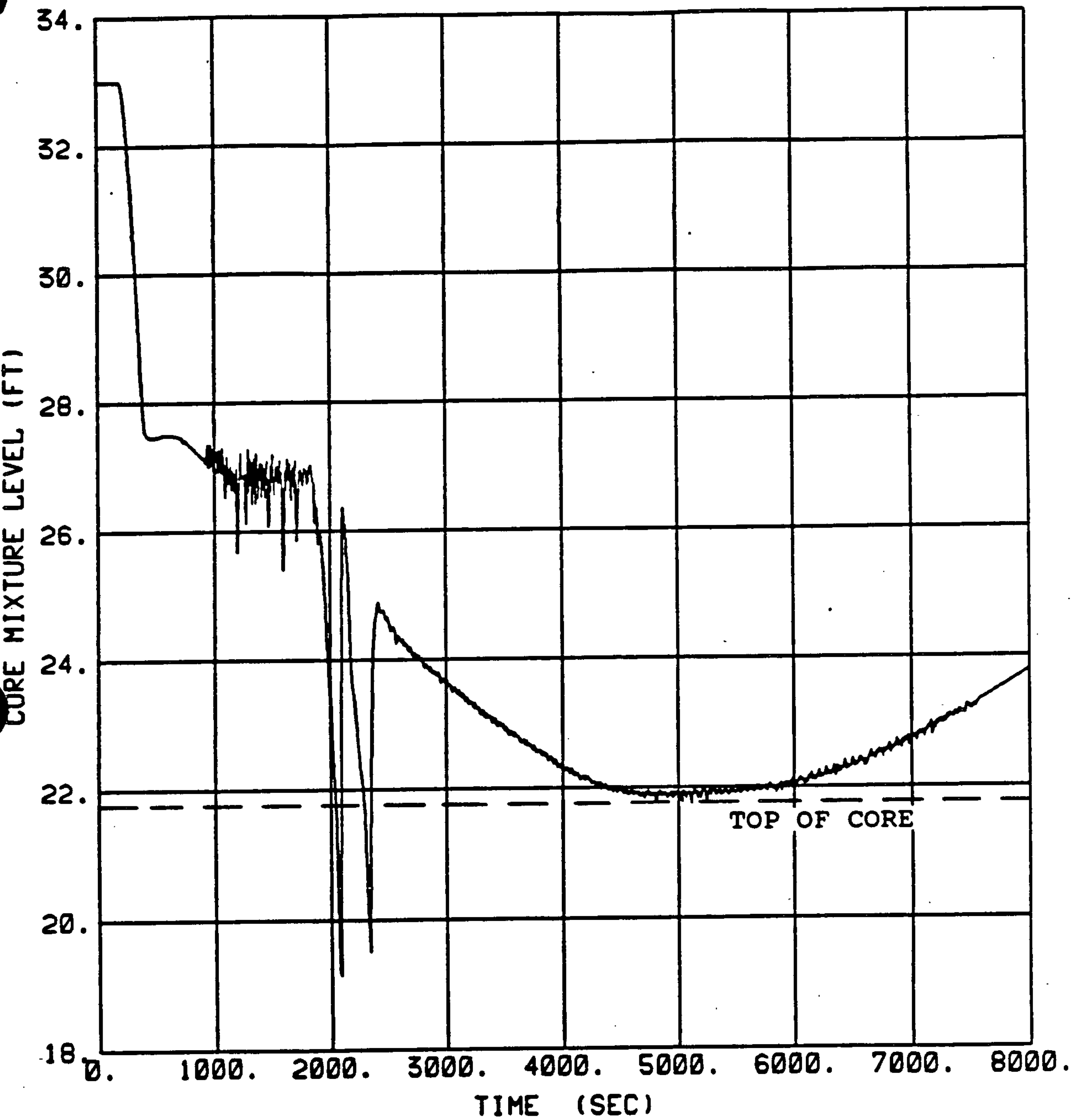
Figure 15.6.2-14 Core Power After Reactor Trips



H. B. ROBINSON UNIT 2

UPPER PLENUM PRESSURE
1.5-INCH COLD LEG BREAK - 60% POWER

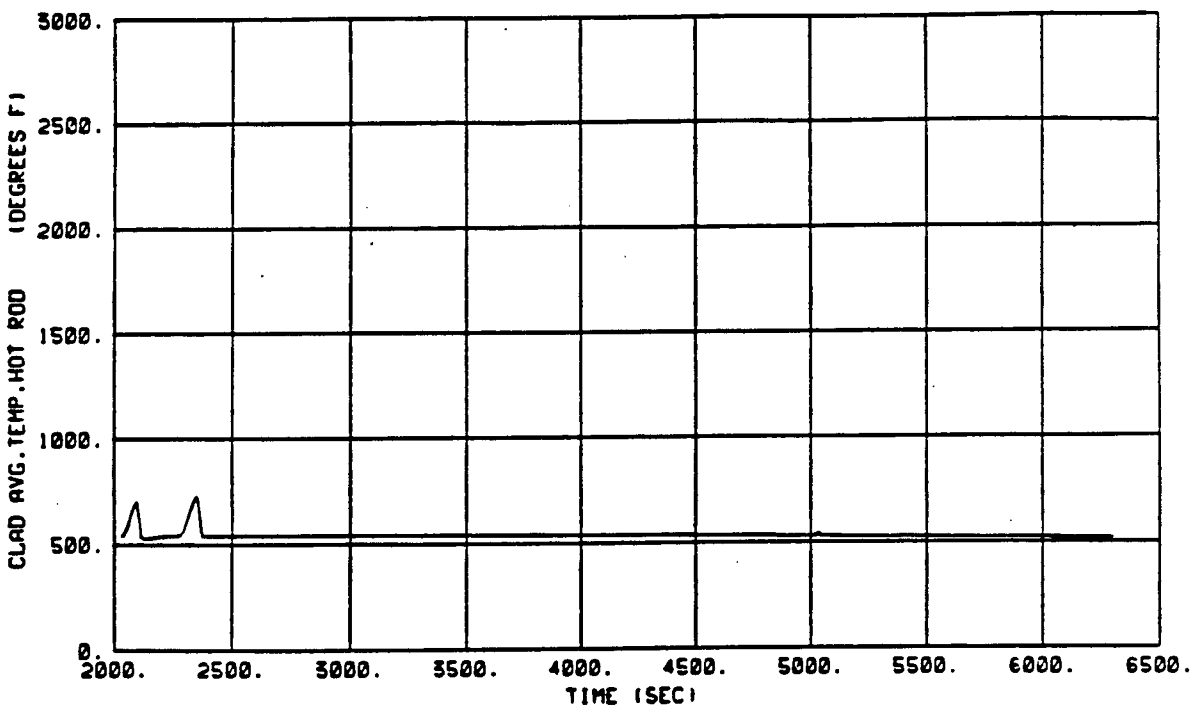
FIGURE 15.6.2-25



H. B. ROBINSON UNIT 2

CORE MIXTURE LEVEL
1.5-INCH COLD LEG BREAK - 60% POWER

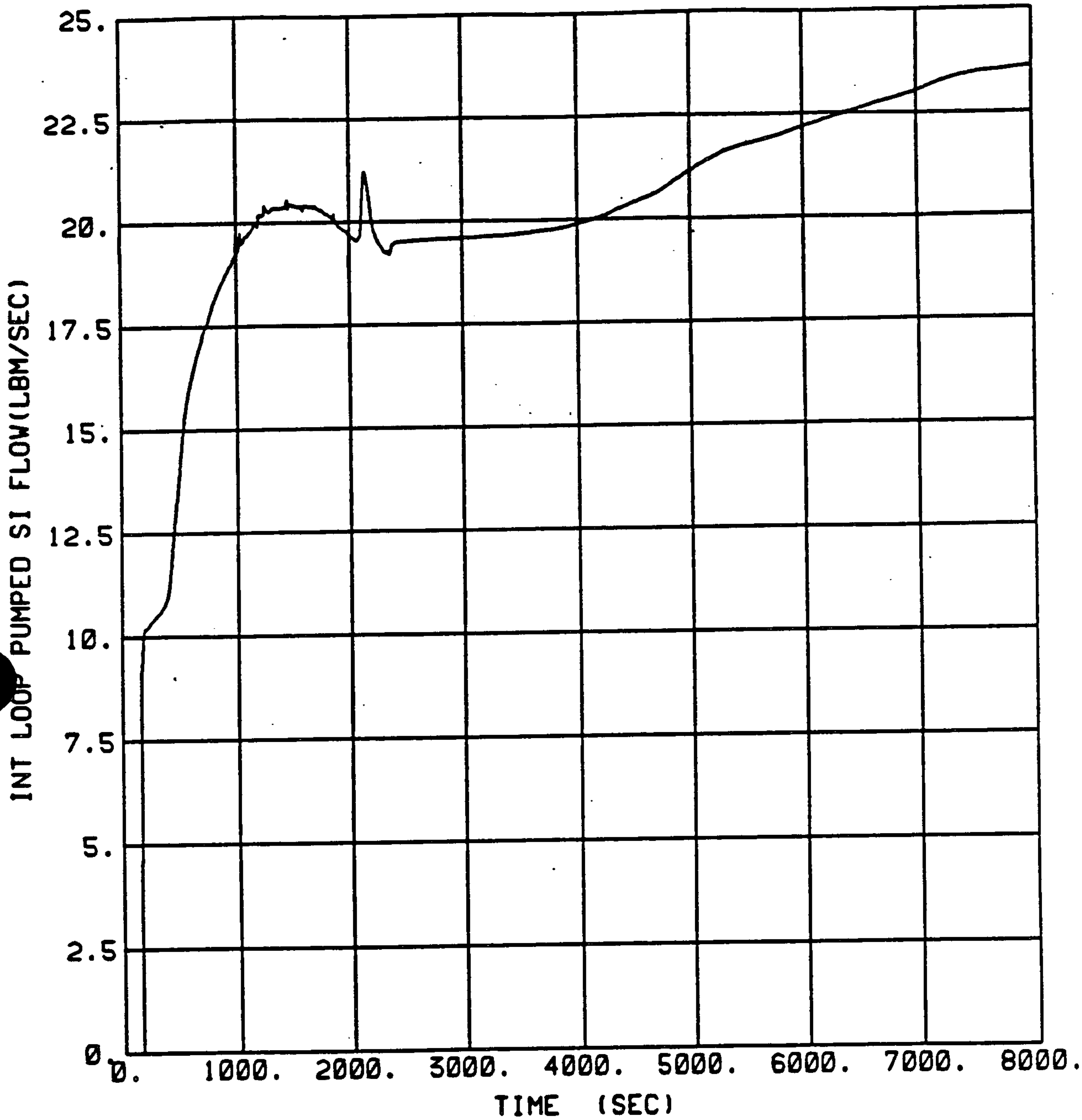
FIGURE 15.6.2-26



H. B. ROBINSON UNIT 2

HOT SPOT CLAD TEMPERATURE
1.5-INCH COLD LEG BREAK - 60% POWER

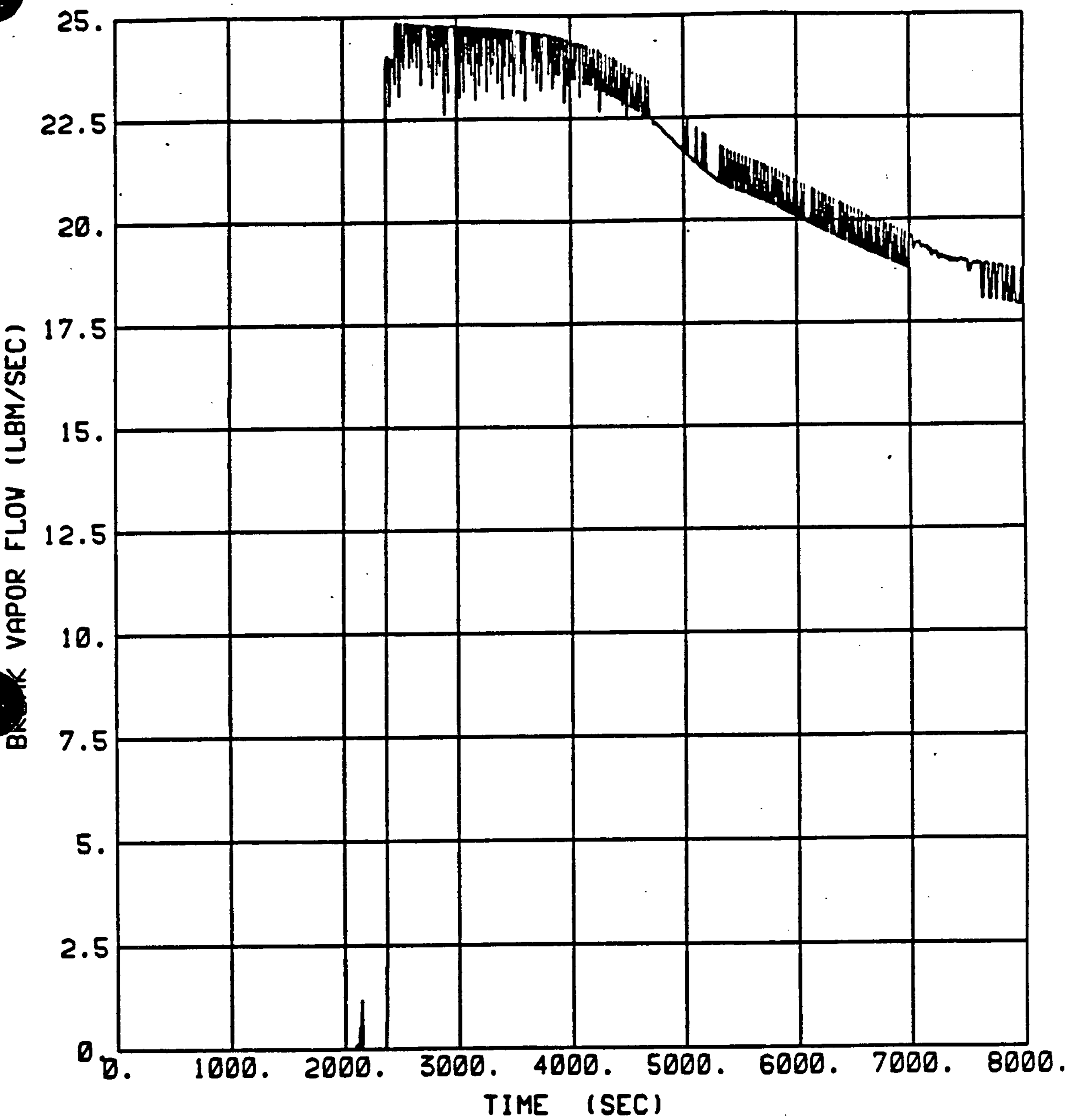
FIGURE 15.6.2-27



H. B. ROBINSON UNIT 2

INTACT LOOP PUMPED SI FLOW
1.5-INCH COLD LEG BREAK - 60% POWER

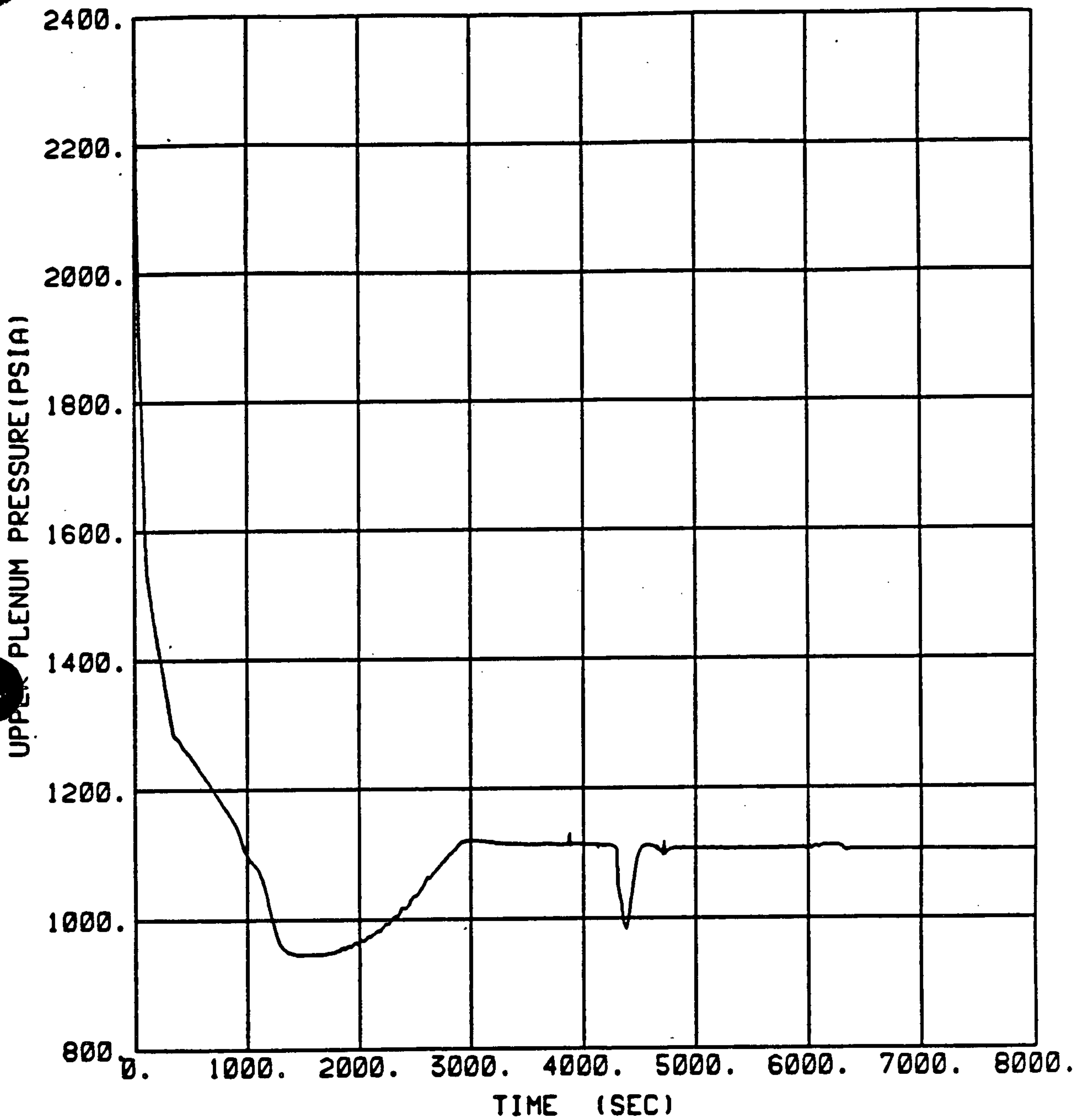
FIGURE 15.6.2-28



H. B. ROBINSON UNIT 2

BREAK VAPOR FLOW
1.5-INCH COLD LEG BREAK - 60% POWER

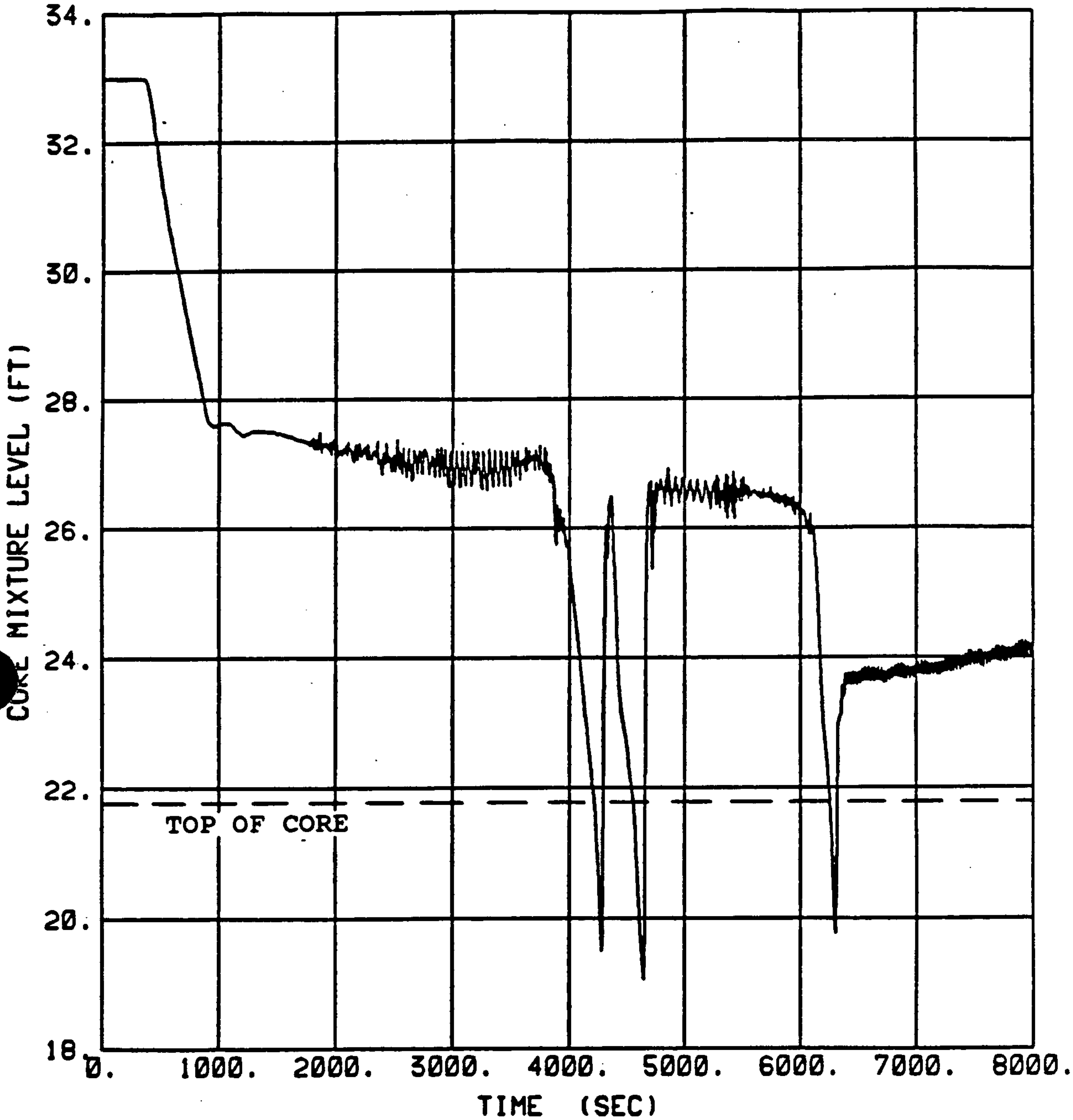
FIGURE 15.6.2-29



H. B. ROBINSON UNIT 2

UPPER PLENUM PRESSURE
1-INCH COLD LEG BREAK - 60% POWER

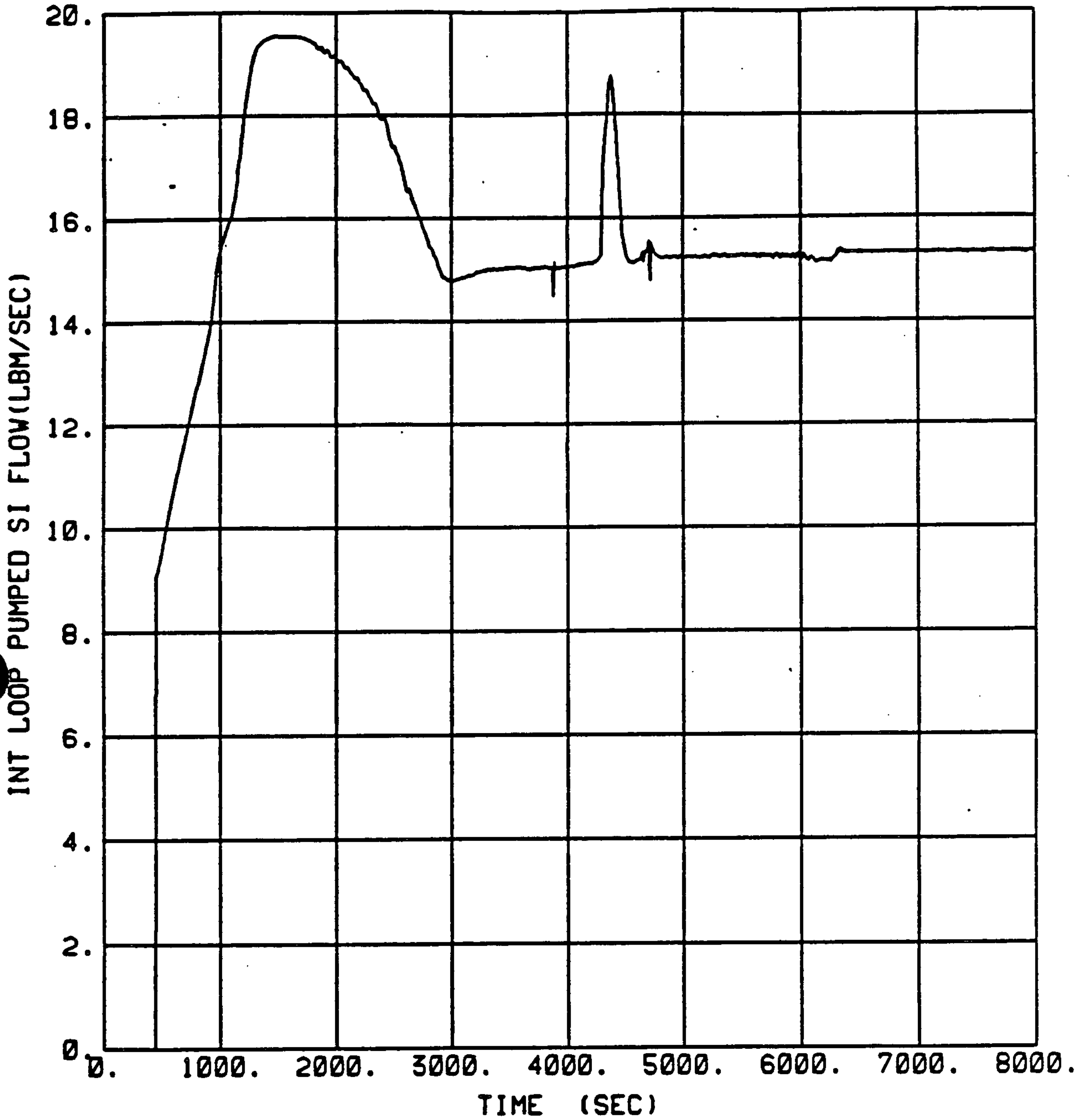
FIGURE 15.6.2-30



H. B. ROBINSON UNIT 2

CORE MIXTURE LEVEL
1-INCH COLD LEG BREAK - 60% POWER

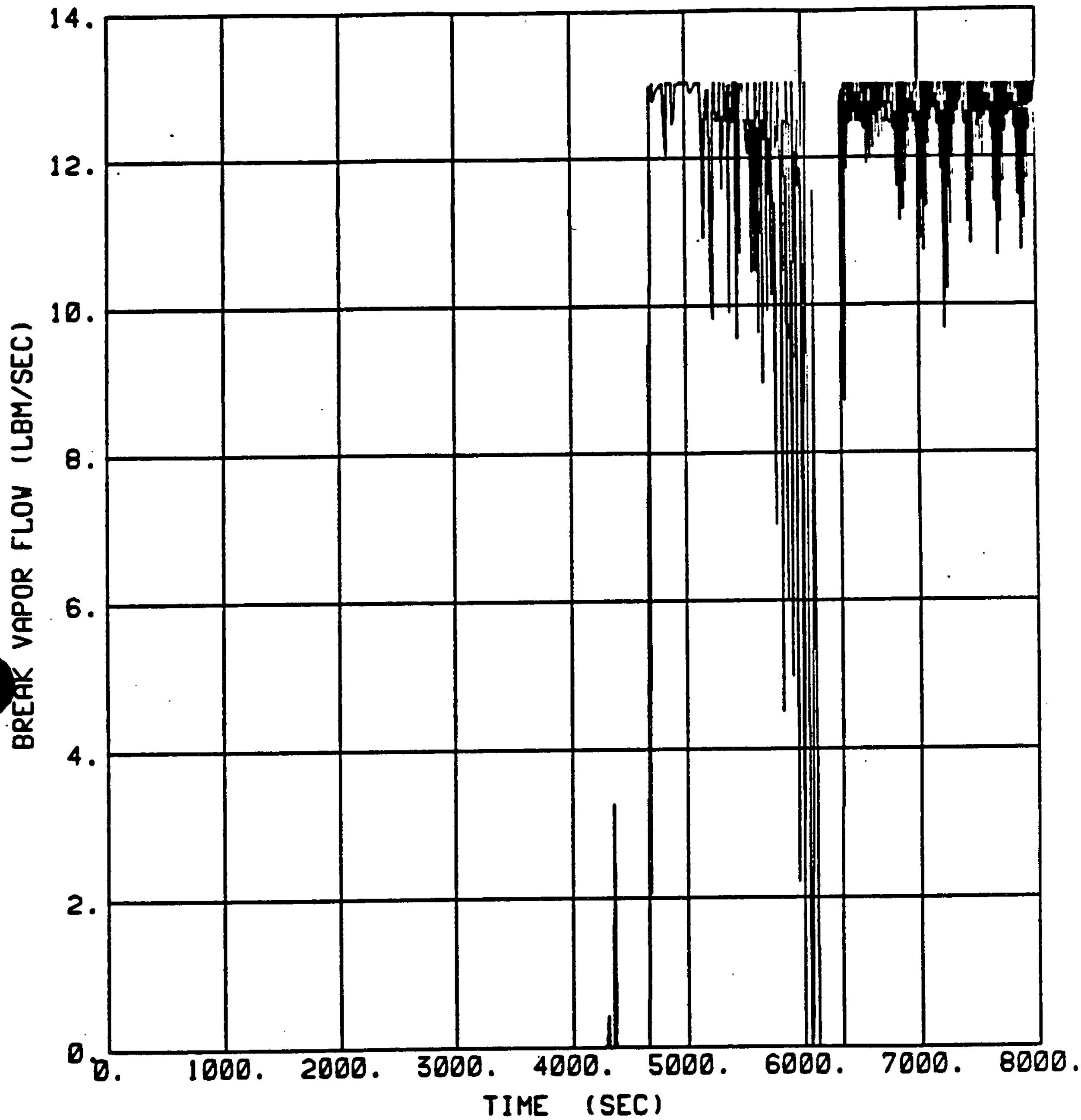
FIGURE 15.6.2-31



H. B. ROBINSON UNIT 2

INTACT LOOP PUMPED SI FLOW
1-INCH COLD LEG BREAK - 60% POWER

FIGURE 15.6.2-32



H. B. ROBINSON UNIT 2

BREAK VAPOR FLOW
1-INCH COLD LEG BREAK - 60% POWER

FIGURE 15.6.2-33

ATTACHMENT D

DISCUSSION OF CONFIRMATORY ACTION ITEMS

Attachment D

ITEM 1

Carolina Power & Light Company has completed a detailed review of the design basis and as-built configuration of the H. B. Robinson Unit 2 (HBR2) onsite electrical distribution system, as it applies to sequencing of the Safety Injection (SI) pumps, in order to evaluate the adequacy of the system to meet the plant licensing basis regarding single-failure vulnerability. This review resulted in one single failure scenario that could not be easily resolved without extensive engineering. This scenario was identified as a postulated failure of the voltage regulator on the emergency diesel generator which provided power to the two SI pumps. The scenario, however, can be more broadly stated as a postulated failure of a diesel control feature that could cause a voltage or frequency anomaly which would result in loss of two of three SI pumps. To eliminate this single failure susceptibility in the short term, an analysis was performed of the most limiting accident, a Small Break Loss of Coolant Accident, which demonstrates that only one SI pump is required to operate up to 1380 MWt (60 percent of rated reactor power) to provide an adequate margin of safety. Operation of one SI pump is assured by the configuration of the SI system which incorporates two automatically initiated SI pumps, "A" powered from emergency bus E-1 and "C" powered from emergency bus E-2. The automatic sequencing of these two SI pumps will remain unchanged. Although not required, "B" SI pump will be a manually operated pump which can be powered from either emergency bus if necessary.

ITEM 2.a

Detailed in our February 12, 1988 letter.

ITEM 2.b

A plant modification (Mod 951) in addition to those described in our February 12, 1988 letter will bring the electrical distribution system into compliance with single-failure criteria. The modification will remove the automatic start capability for the "B" SI pump to assure single failure protection during a safety injection. The modification will also result in operation with the "B" SI pump breaker (52/29C) open. Only one SI pump is required for operation up to 60 percent of rated power. The "A" and "C" SI pumps provide this capability automatically with sufficient redundancy to account for a single failure. Although not required, operation of the "B" SI pump, if necessary, would be through manual operator action. The current plan for implementation shows completion of Mod 951 by February 26, 1988 prior to RCS heat up above 200°F.

ITEM 2.c

Post-modification testing will demonstrate that the electrical distribution system meets its design basis by verifying that the automatic start feature from each train's SI sequencing logic has been removed with regard to the "B"

SI pump. Manual operation of the "B" SI pump will also be verified. The 22B/29B interlock which prevents paralleling E1 and E2 will be verified. The testing will be completed prior to declaring Mod 951 satisfactorily complete.

ITEM 2.d

Training of plant personnel as a result of Mod 951 will be conducted for all licensed shift personnel by a senior instructor from the Robinson Training Unit prior to assuming shift duties at or above 200°F. In addition, the physical fidelity of the Robinson plant-specific simulator will be preserved. The Modification Package M-951, and the acceptance test, SP-798, will be utilized to develop a lesson plan and handout. This training material will cover the following training objectives: (1) describe the hardware changes implemented under the modification, (2) describe the single-failure scenario that necessitated the modification and the correction action taken, and (3) describe operator impact as a result of the change.

ITEM 2.e

Technical Specification changes are requested by this letter.

(5382JSK/crs)