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. Two safety injections pumps are operable, each capable of automatic initiation from a separate emergency bus.

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.

~~8805100295~~ 50pp

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 safety injection pump is 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.

3.10.1.5 Except for physics tests, if a full length control rod is withdrawn as follows:

- at positions $\geq$ 200 steps and is > 15 inches out of alignment with its bank position, or

- at positions < 200 steps and is > 7.5 inches out of alignment with the average of its bank position

then within two hours, perform the following:

a. Correct the situation, or

b. Determine by measurement the hot channel factors and apply Specification 3.10.2.1, or

c. Limit power to 70 percent of rated power

3.10.1.6 Insertion limits do not apply during physics tests or during period exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-2 must be maintained, except during the low power physics test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one full length control rod inserted.

3.10.2 <u>Power Distribution Limits</u>

3.10.2.1 At all times except during low power physics tests, the hot channel factors, $F_Q(Z)$ and $F_{\Delta H}$, defined in the basis, must meet the following limits:

$F_Q(Z) \leq (2.32/P) \times K(Z)$ for $P > 0.5$

$F_Q(Z) < 4.64 \times K(Z)$ for $P \leq 0.5$

$F_{\Delta H} < 1.65\ (1 + 0.2(1-P))$

where P is the fraction of rated power (2300 Mwt) at which the core is operating. $F_Q(Z)$ is the measured $F_Q(Z)$ including the measurement uncertainty factor $F_u^N = 1.05$ and the engineering factor $F_Q^E = 1.03$. $F_{\Delta H}$ is the measured $F_{\Delta H}$ including a 1.04 measurement uncertainty factor. K(Z) is based on the function given in Figure 3.10-3, and Z is the axial location of F_Q.

3.10.2.1.1 Following initial loading, or upon achieving equilibrium conditions after exceeding by 10% or more of rated power, the power $F_Q(Z)$ was last determined, and at least once per effective full power month, power distribution maps using the movable detector system, shall be made to confirm that the hot channel factor limits of Specification 3.10.2.1 are satisfied and to establish the target axial flux difference as a function of power level (called the target flux difference).*

If either measured hot channel factor exceeds the specified limit, the reactor power shall be reduced so as not to exceed a fraction equal to the ratio of the $F_Q(Z)$ or $F_{\Delta H}$ limit to the measured value, whichever is less, and the high neutron flux trip setpoint shall be reduced by the same ratio.

If subsequent incore mapping cannot, within a 24-hour period, demonstrate that the hot channel factors are met, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

3.10.2.2 $F_Q(Z)$ shall be determined to be within the limit given in 3.10.2.1 by satisfying the following relationship for the middle axial 80% of the core at the time of the target flux determination:

$$F_Q(Z) \leq \left(\frac{2.32}{P}\right) \left[\frac{K(Z)}{V(Z)}\right] \text{ for } P > 0.5$$

$$F_Q(Z) < 4.64 \left[\frac{K(Z)}{V(Z)}\right] \text{ for } P \leq 0.5$$

* During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

where V(Z) is defined in Figure 3.10-4 which corresponds to the target band and P > 0.5.

3.10.2.2.1 If the relationship specified in 3.10.2.2 cannot be satisfied, one of the following actions shall be taken:

a) Place the core in an equilibrium condition where the limit in 3.10.2.2 is satisfied and re-establish the target axial flux difference

b) Reduce the reactor power by the maximum percent calculated with the following expression for the middle axial 80% of the core:

$$\left[\left[\text{max. over Z of } \frac{F_Q(Z) \times V(Z)}{\frac{2.32}{P} \times K(Z)} \right] - 1 \right] \times 100\%$$

c) Comply with the requirements of Specification 3.10.2.2.2.

3.10.2.2.2 The Allowable Power Level above which initiation of the Axial Power Distribution Monitoring System (APDMS) is required is given by the relation:

$$\text{APL} = \text{minimum over Z of } \frac{2.32 \times K(Z)}{F_Q(Z) \times V(Z)} \times 100\%$$

where $F_Q(Z)$ is the measured $F_Q(Z)$, including the engineering factor $F_Q^E = 1.03$ and the measurement uncertainty factor $F_u^N = 1.05$ at the time of target flux determination from a power distribution map using the movable incore detectors. V(Z) is the variation function defined in Figure 3.10-4 which corresponds to the target band. K(Z) is the function defined in Figure 3.10-3.

The above limit is not applicable in the following core plane regions.

1) Lower core region 0% to 10% inclusive.
2) Upper core region 90% to 100% inclusive.

At power levels in excess of APL of rated power, the APDMS will be employed to monitor $F_Q(Z)$. The limiting value is expressed as:

$$[F_j(Z)\ S(Z)]_{max} \leq \frac{2.103/P}{\bar{R}_j\ (1 + \sigma_j)}$$

where:

a. P is the fraction of rated power (2300 Mwt) at which the core is operating ($P \leq 1.0$).

b. $\bar{R}_j$ for thimble j, is determined from core power maps and is by definition:

$$\bar{R}_j = \frac{1}{6} \sum_{i=1}^{6} \frac{F_{Qj}}{[F(Z)_{ij}\ S(Z)]_{max}}$$

F_{Qj} is the value obtained from a full core map including S(Z), but without the measurement uncertainty factor F_u^N or the engineering uncertainty factor, F_Q^E. The quantity $F(Z)_{ij}$ S(Z) is the measured value without inclusion of the instrument uncertainty factors F_Q^a. Those uncertainty factors, $F_u^N = 1.05$, $F_Q^a = 1.02$, as well as the engineering factor $F_Q^E = 1.03$, have been included in the limiting value of 2.103/P.

c. σ_j is the standard deviation associated with the determination of $\bar{R}_j$.

d. S(Z) is the inverse of the K(Z) function given in Figure 3.10-3.

This limit is not applicable during physics tests and excore detector calibrations.

3.10.2.2.3 With successive measurements indicating the enthalpy rise hot channel factor, $F_{\Delta H}^N$, to be increasing with exposure, the total peaking factor, $F_Q(Z)$, shall be further increased by two percent over that specified in Specifications 3.10.2.2, 3.10.2.2.1, and

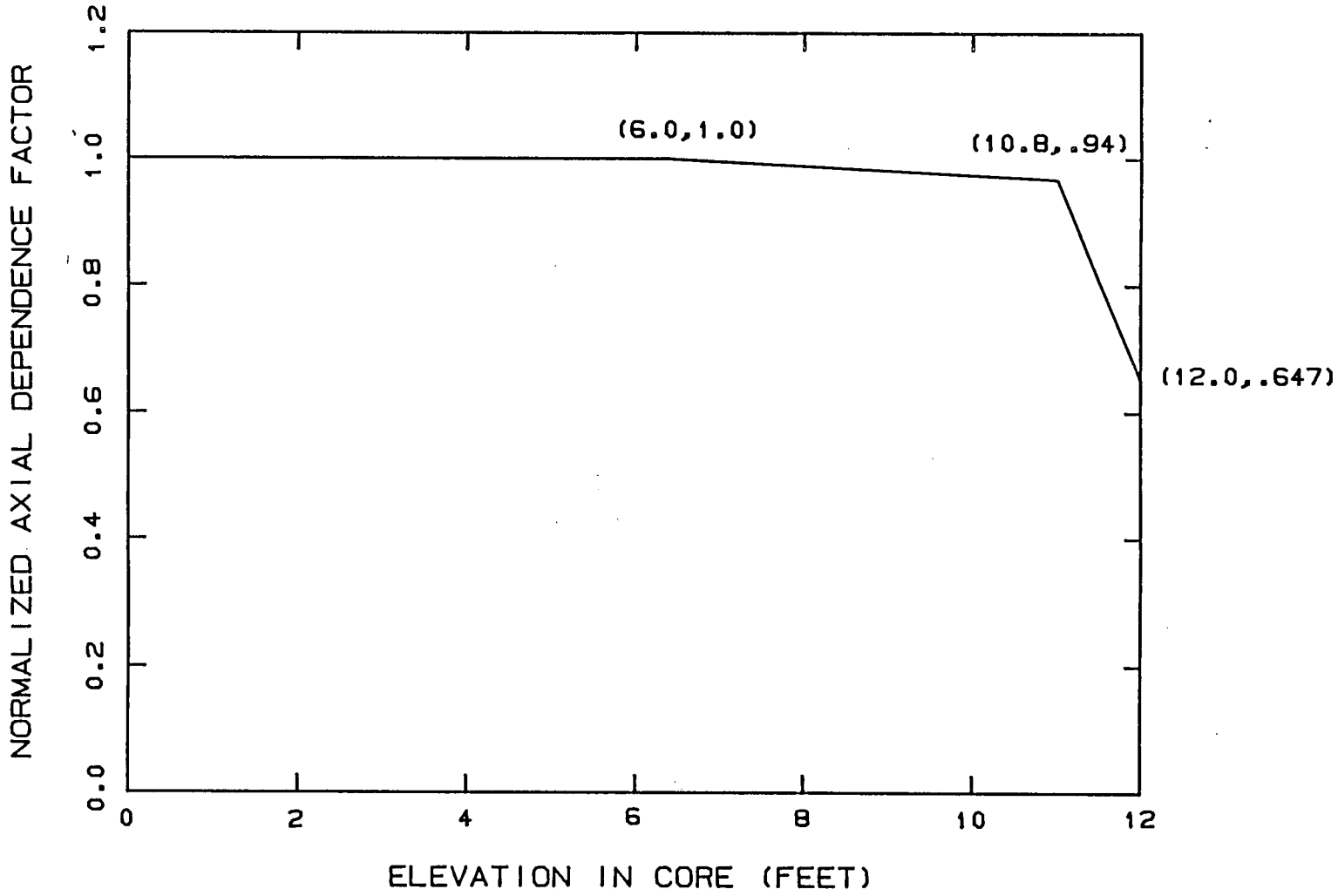


FIGURE 3.10-3 NORMALIZED AXIAL DEPENDENCE FACTOR FOR Fq VERSUS ELEVATION (PEAK Fq = 2.32)

CPL-88-538

Westinghouse Electric Corporation

Power Systems

Nuclear Technology
Systems Division

Box 355
Pittsburgh Pennsylvania 15230-0355

May 5, 1988
NS-OPLS-OPL-II-88-322

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
JUSTIFICATION FOR STARTUP AND OPERATION OF
H. B. ROBINSON AT 100% POWER WITH ONE
HIGH HEAD SAFETY INJECTION PUMP AVAILABLE

Dear Mr. Zimmerman:

Please find attached the results of small break loss-of-coolant accident (LOCA) analyses performed by Westinghouse for the H. B. Robinson nuclear power plant assuming that only one high head safety injection (HHSI) pump is available to provide pumped ECCS flow during a small break LOCA event. Attachment 1 provides background information describing how the current analyses differ from previous H. B. Robinson small break LOCA analyses.

The current analyses consist of a spectrum of three small break LOCA analyses 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 2300 MWth and included the representation of Advanced Nuclear Fuels Corporation (ANF) 15x15 fuel parameters.

The power shapes used for these analyses were chosen by Westinghouse from a data base of power shapes provided by ANF using Westinghouse supplied criteria to provide a good census of limiting small break LOCA power shapes. Attachment 2 provides supporting information describing the power shape development.

Analyses were performed for 1.0-inch, 1.5-inch, and 2-inch equivalent diameter breaks. Of these three cases, the 1.5-inch case resulted in the highest peak cladding temperature of 2003.8°F. The complete results for all three breaks are provided in Attachment 3.

The results of the analyses show that the H.B.Robinson Unit 2 Nuclear Power Plant may be operated at 100 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.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

G. O. Percival, Manager
Carolina Area

D. L. Cecchett/dmr
Attachment

cc: S. R. Zimmerman (CP&L) 1L, 1A
T. M. Dresser (CP&L) 1L, 1A
T. B. Clements (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

CPL-88-538
NS-OPLS-OPL-II-88-322

bcc: File: CPL-88-538 (WEC-W-238) 1L, 1A
G. O. Percival (WEC-W-238) 1L, 1A
T. R. Croasdaile (MMOB 2-11) 1L, 1A
G. J. Murray (WEC-W-238) 1L, 1A
D. L. Cecchett (WEC-E-413) 1L, 1A
D. P. Dominicis (WEC-E-414) 1L, 1A
OPL Letter File (Denise Rohaus) WEC-E-413 1L, 1A
W. D. Tauche (WEC-E-425) 1L, 1A
J. M. Brennan (WEC E4) 1L, 1A
R. A. Osterrieder (WEC E4) 1L, 1A

ATTACHMENT 1

SMALL BREAK LOCA ANALYSIS BACKGROUND
H. B. ROBINSON UNIT 2 NUCLEAR POWER PLANT

A small break loss-of-coolant-accident (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 for 15x15 fuel manufactured by the Advanced Nuclear Fuels Corporation 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.

Early in 1988, Carolina Power and Light personnel 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 during a LOCA event. 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 LOCA.

Carolina Power and Light then instructed Westinghouse to perform small break analyses assuming that only one high head safety injection pump is available during the small break LOCA. Due to the time constraints of the situation, there was no time to do detailed calculations of the flow delivered to the reactor coolant system (RCS) from only one HHSI pump. Therefore, a conservatively low estimate of the amount of safety injection flow delivered to the RCS from one high head safety injection pump was used for the analyses.

The results of the analyses showed that the H.B.Robinson Unit 2 Nuclear Power Plant could be 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.

Since that time calculations have been performed by Westinghouse to determine what flow will be delivered to the RCS from one HHSI. The calculations incorporated the standard FSAR ECCS assumption of minimum safeguards. The calculated delivery data was developed based on as-built piping layout information and a composite minimum pump curve (based on system test performance) degraded by 5 percent of the design TDH. Other assumptions used for the development of the delivery data include no branch line header balancing, the branch line with the least resistance spills to RCS pressure, and the pump minimum flow path remains open throughout the entire injection phase. The results of these calculations are compared to the conservatively low values assumed for the 60 percent power cases in Figure 1.1.

New small break LOCA analyses have been performed with the calculated HHSI delivery data for one HHSI pump. These analyses consist of a spectrum of three small break LOCA analyses performed assuming a core power level corresponding to 102% of 2300 MWth.

More realistic but conservative power shapes were used in these analyses. The power shapes used were chosen by Westinghouse from a data base of power shapes provided by ANF using Westinghouse supplied criteria to provide a good census of limiting small break LOCA power shapes. Attachment 2 provides supporting information describing the power shape development.

Analyses were performed for 1.0-inch, 1.5-inch, and 2-inch equivalent diameter breaks. Of these three cases, the 1.5-inch case resulted in the highest peak cladding temperature of 2003.8°F.

The results of the analyses show that the H.B.Robinson Unit 2 Nuclear Power Plant may be operated at 100 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.

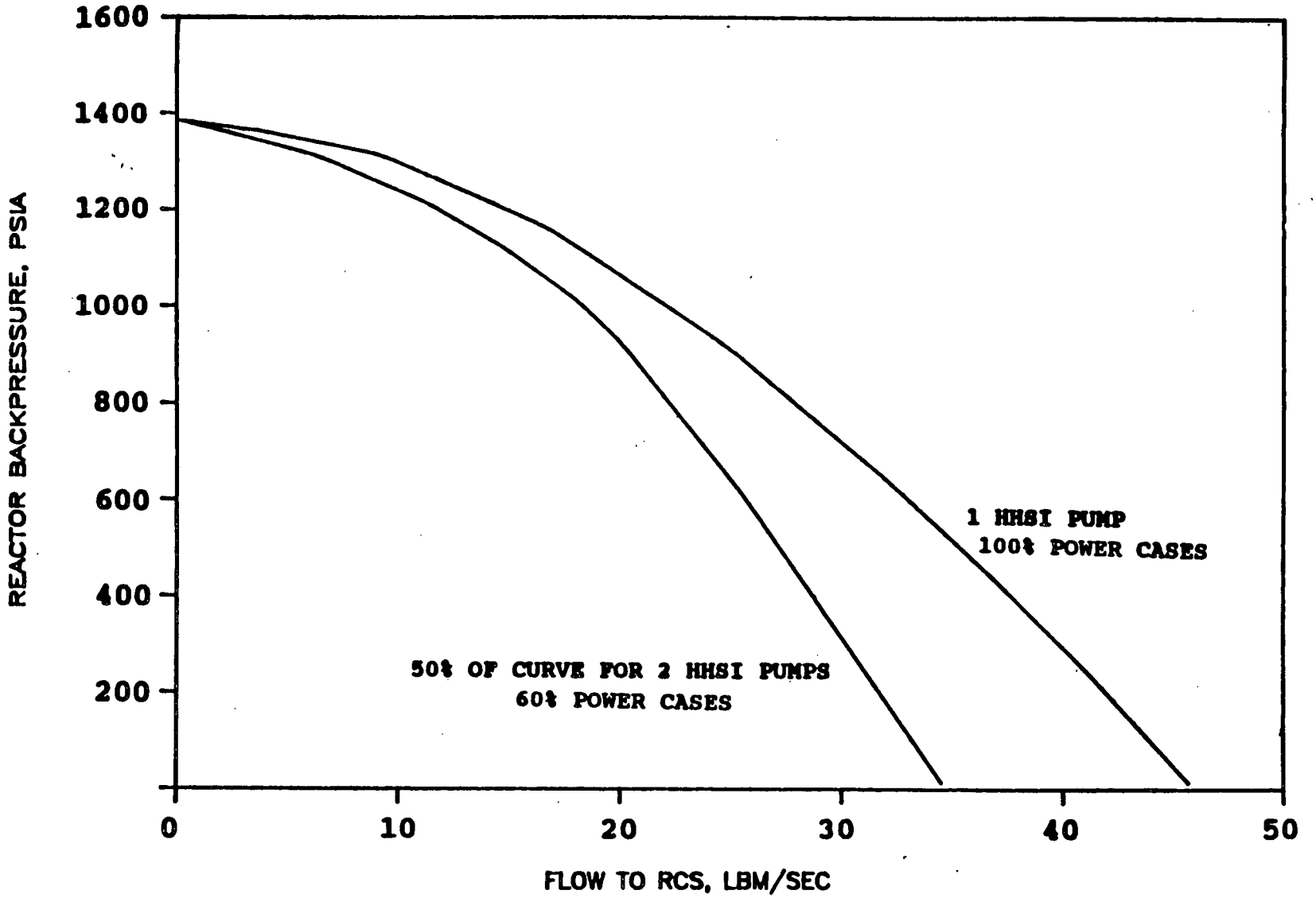


FIGURE 1.1 COMPARISON OF SAFETY INJECTION FLOWRATES USED FOR 60% POWER ANALYSES TO SAFETY INJECTION FLOWRATES USED FOR 100% POWER ANALYSES

ATTACHMENT 2

Supporting Information For Power Shape Development

L. POWER SHAPE SELECTION AND SCALING

L.1 Background

For nuclear plant licensing basis LOCA calculations the requirements of 10CFR50.46, Appendix K, require that:

> ". . . A range of power distribution shapes and peaking factors representing power distributions that may occur over core lifetime shall be studied and the one selected should be that which results in the most severe calculated consequences . . ."

To comply with this requirement standard Westinghouse methodology regarding generic power shapes for SBLOCA analysis is based upon the concept of enveloping all possible shapes in the top of the core through the use of the Tech Spec axial peaking factor [K(z)] curve. This methodology is grossly conservative since it is based upon artificially adding, or skewing power into the top elevations of the core far in excess of that which is known to be possible based upon nuclear design calculations. It is also standard methodology that, should excessive peak clad temperatures be calculated with the generic K(z) envelope methodology, plant specific power shapes may be used to perform the licensing calculations. Since the H.B. Robinson plant is fueled with Advanced Nuclear Fuels (ANF) corporation fuel, and the core nuclear design is performed by ANF, power shape data was obtained from ANF. Westinghouse reviewed the data base to select a limiting basis shape, and scaled that shape into a conservative input for the LOCA calculations.

L.2 Data Base

ANF supplied a data base of approximately 120 power shapes for further review. Axial power distributions were generated for several core power levels at beginning, middle and end of cycle conditions. The calculations were performed with a three-dimensional code (24 axial and 4 radial nodes per assembly) using standard ANF PWR neutronics design methodology. Specifically shapes were obtained from bounding load follow cases. These cases assumed operation under PDC-2 operating guidelines for $\pm 5\%$ ΔI operating bands. Many of the shapes in the data base represent conditions at limiting positive ΔI values. Runs were made with both nominal and increased control rod worth to encompass the range of top-peaked power distributions possible. In the opinion of ANF, the data base represents a good census of limiting SBLOCA power shapes. The power shapes were transmitted in the form of composite rod curves, representing the highest peaking factor for each elevation in the core model.

L.3 Basis Power Shape Selection and Scaling

Westinghouse evaluated the power shapes for SBLOCA severity using a weighting technique that quantitatively gages the topward skewing of the shapes on a common normalized basis. Analysis of the results showed that the weighting factor for all of the curves varied over a range of approximately 5%, and that generally the weighting increased with burnup. The highest weighted curve from the data base for 100% power, which was an end of cycle burnup case, was selected, and the highest peaked single fuel pin from the composite curve was used to obtain the basis power shape for further calculations. This shape in terms of FQ(z) is shown in Figure L.1 along with the corresponding core average shape.

The core average shape is transformed into an equivalent 4-node average power distribution for use in the NOTRUMP, system hydraulics code. The Tech Spec axial peaking factor limits allow the core hot rod peak power to reach the limits shown by the K(z) curve on Figure L.1. Since the worst case power shape from the data base still does not challenge these limits, the power shape is scaled to provide an input suitable for LOCA rod heat transfer licensing calculations. The scaling procedure involves forcing the peak hot rod power to match the appropriate K(z) peaking factor limit, while maintaining the Tech Spec limiting hot channel enthalpy rise factor (Fdh). This final curve is shown in Figure L.2 in terms of the linear heat generation rate for the hot rod. This technique results in a LOCA code input power shape which is at the limiting Tech Spec values, but which still preserves the the basic shape of the basis curve from the limiting nuclear design calculations.

L.4 Comparison to Standard Curve and Conservatism

Figure L.3 compares the hot rod power shape for the present analysis to the standard K(z) envelope shape used generically, with both in terms of the hot rod linear heat generation rate. The obvious difference between the two is in the form of the power shape in the top two feet of the core. As stated previously, the standard shape represents an envelope of shapes above the 10.8 ft elevation, and does not reflect any obtainable individual shape. Nuclear design calculations show conclusively that heat generation rates as high as the K(z) envelope are not obtainable in the top of the core. By evaluating a large number of power shapes in terms of their SBLOCA severity, the limiting shape from a broad range of possible shapes is determined directly. In the process of scaling this basis curve, additional conservatism is applied such that the Tech Spec limits are validated. In this respect, in accordance with Appendix K, a range of shapes has been examined and a limiting power shape selected without the need to apply a third layer of additional gross conservatism represented by the K(z) envelope technique.

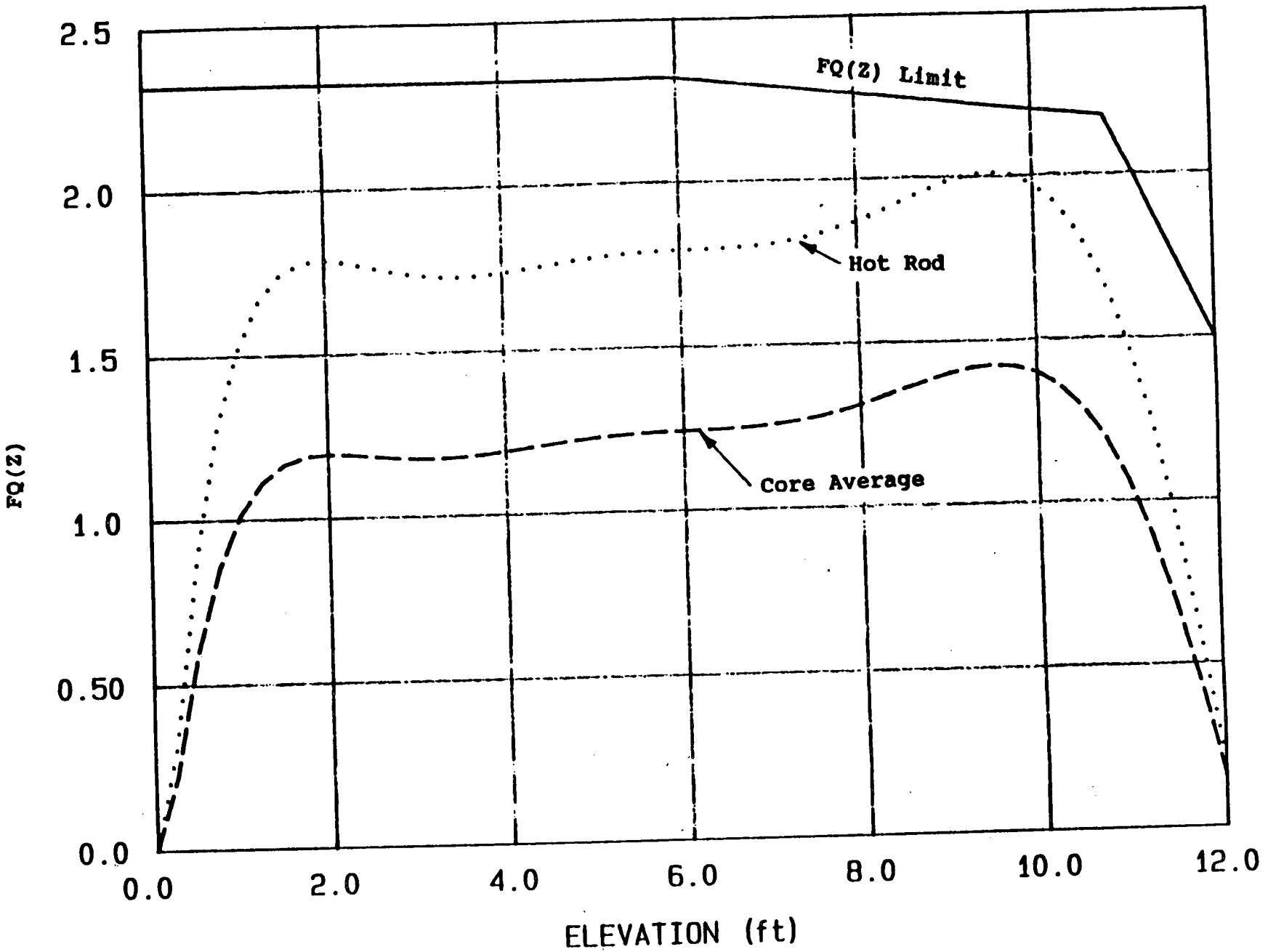


Figure L.1 - H. B. Robinson
SBLOCA Basis Power Shapes

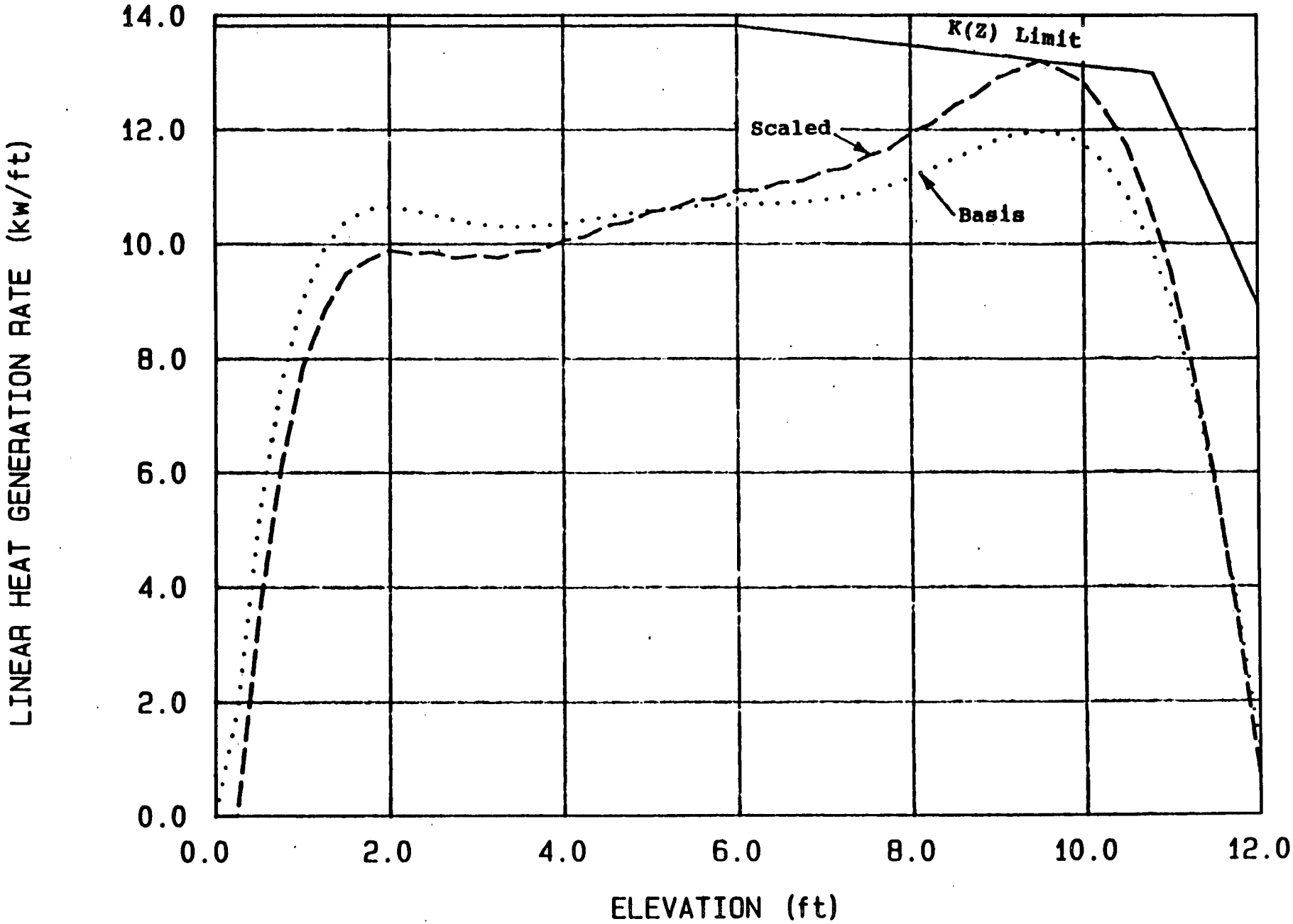


Figure L.2 - H. B. Robinson
SBLOCA Analysis Basis and Scaled Hot Rod Power Shapes

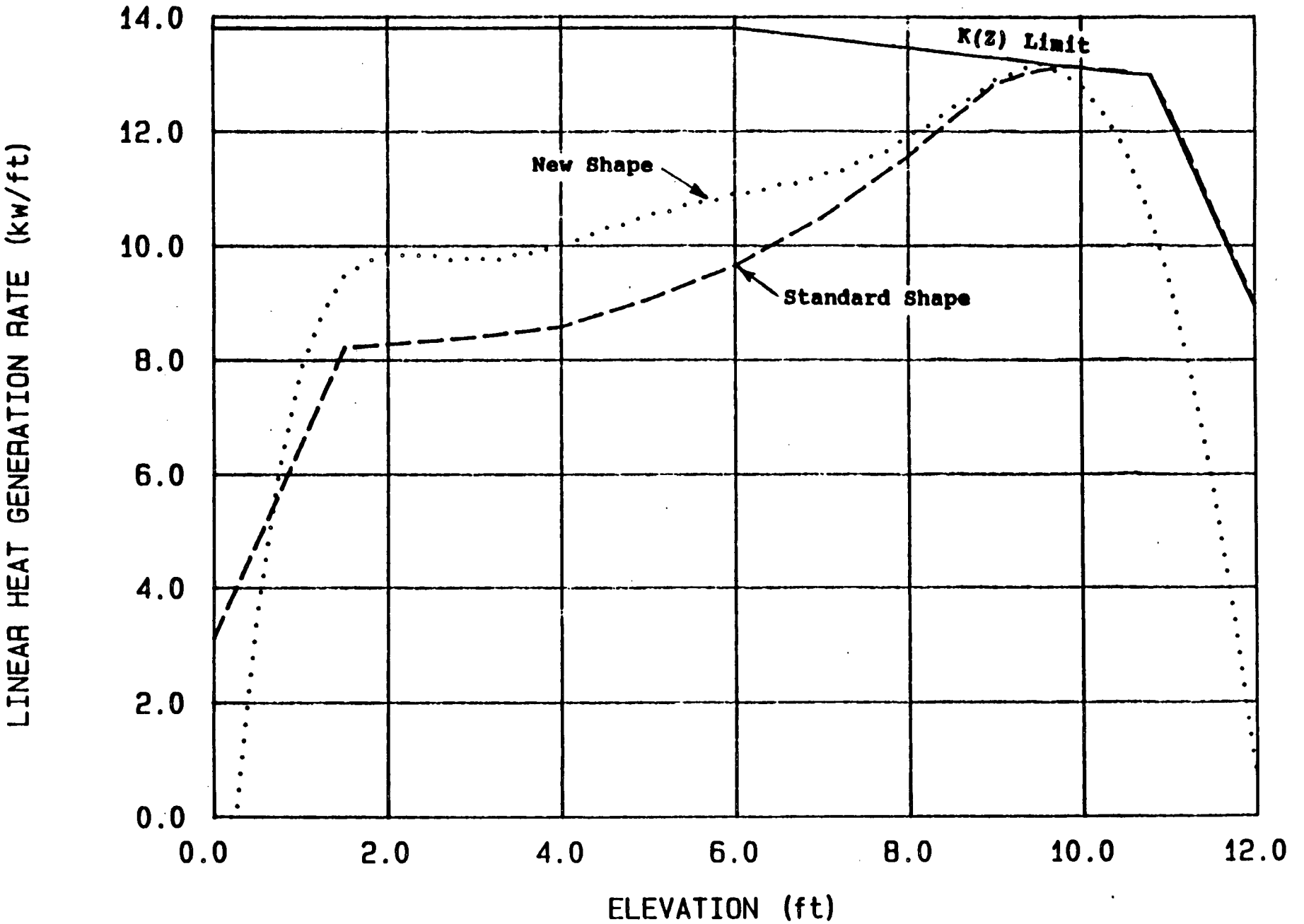


Figure 1.3 - H. B. Robinson SBLOCA Analysis Hot Rod Power Shape Compared to the Standard K(Z) Envelope Shape

ATTACHMENT 3

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.

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 core average and hot rod power shapes and core decay power assumed for the small break analyses are shown in Figures 15.6.2-2 and 15.6.2-3.

Safety injection flow to the Reactor Coolant System (RCS) as a function of the system pressure is used as part of the input. The SI delivery curve used for these analyses is depicted in Figure 15.6.2-1 as a function of RCS pressure. This figure represents injection flow from one High Head Safety Injection (HHSI) pump. The delivery data incorporates the standard FSAR ECCS assumption of minimum safeguards. The delivery data was developed based on as-built piping layout information and a composite minimum pump curve (based on system test performance) degraded by 5 percent of the design TDH. Other assumptions used for the development of the delivery data include no branch line header balancing, the branch line with the least resistance spills to RCS pressure, and the pump minimum flow path remains open throughout the entire injection phase. 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 performed with the NOTRUMP code using 102% 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 were performed in a similar manner, i.e., 102% of

licensed core power. The LOCTA-IV core thermal analyses incorporated Exxon 15x15 fuel data which is summarized in Table 15.6.2-2.

Small Break LOCA Results

A range of small break analyses is presented which establishes that the limits of 10CFR 50.46 will not be exceeded at 100% of licensed core power operation. The results of these analyses are summarized in Tables 15.6.2-3 and 15.6.2-4. Figures 15.6.2-4 through 15.6.2-20 present the principal parameters of interest for the small break ECCS analyses. For the 2-inch, 1.5-inch and 1-inch break sizes the following transient parameters are included:

a. RCS Pressure
b. Core Mixture Height
c. Hot Spot Clad Temperature (2.0 and 1.5 inch cases only)
d. Intact Loop Pumped SI Flow
e. Break Vapor Flow

As indicated in the results for clad heat up, the 1.5-inch case is limiting. For the limiting 1.5 inch break size, the following additional transient parameters are presented :

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 2003.8°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 100% of licensed core power.

TABLE 15.6.2-1

Input Parameters Used in the SBLOCA Analysis

Core Power[1]	2346 MWt
Pump Heat	8 MWt
NSSS Power	2354 MWt
Peak Linear Power (includes 102% factor)	13.318 kW/ft
Total Peaking Factor, F	2.32
Power Shape	Fig. 15.6.2-2
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. 15.6.2-1
Steam Generator Initial Pressure	787 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

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.
Pellet Theoretical Density	95.3%

TABLE 15.6.2-3

Small Break LOCA Time Sequence of Events

Event	2 in (sec)	1.5 in (sec)	1 in (sec)
Start	0.0	0.0	0.0
Reactor Trip	33.21	63.07	150.33
S-signal	46.56	85.15	184.76
Loop Seal Venting	970.5	1746.4	4400.3
Top of Core Uncovered	1884.5	3166.9	4369.7
Accumulator Injection	N/A	N/A	N/A
Maximum Core Uncovery	3299.9	4546.3	4397.3
Peak Clad Temperature Occurs	3518.5	5045.9	0.0

TABLE 15.6.2-4

Small Break LOCA Fuel Cladding Results

Results	2 in	1.5 in	1 in
Peak clad temperature (°F)	1887.6	2003.8	N/A
Peak clad temperature location (ft)	11.5	12.0	N/A
Local Zr/H_2O reaction, maximum (%)	5.04	8.59	N/A
Local Zr/H_2O location (ft)	11.5	11.75	N/A
Total Zr/H_2O reaction (%)	<0.3	<0.3	N/A
Hot rod burst time (sec)	N/A	N/A	N/A
Hot rod burst location (ft)	N/A	N/A	N/A

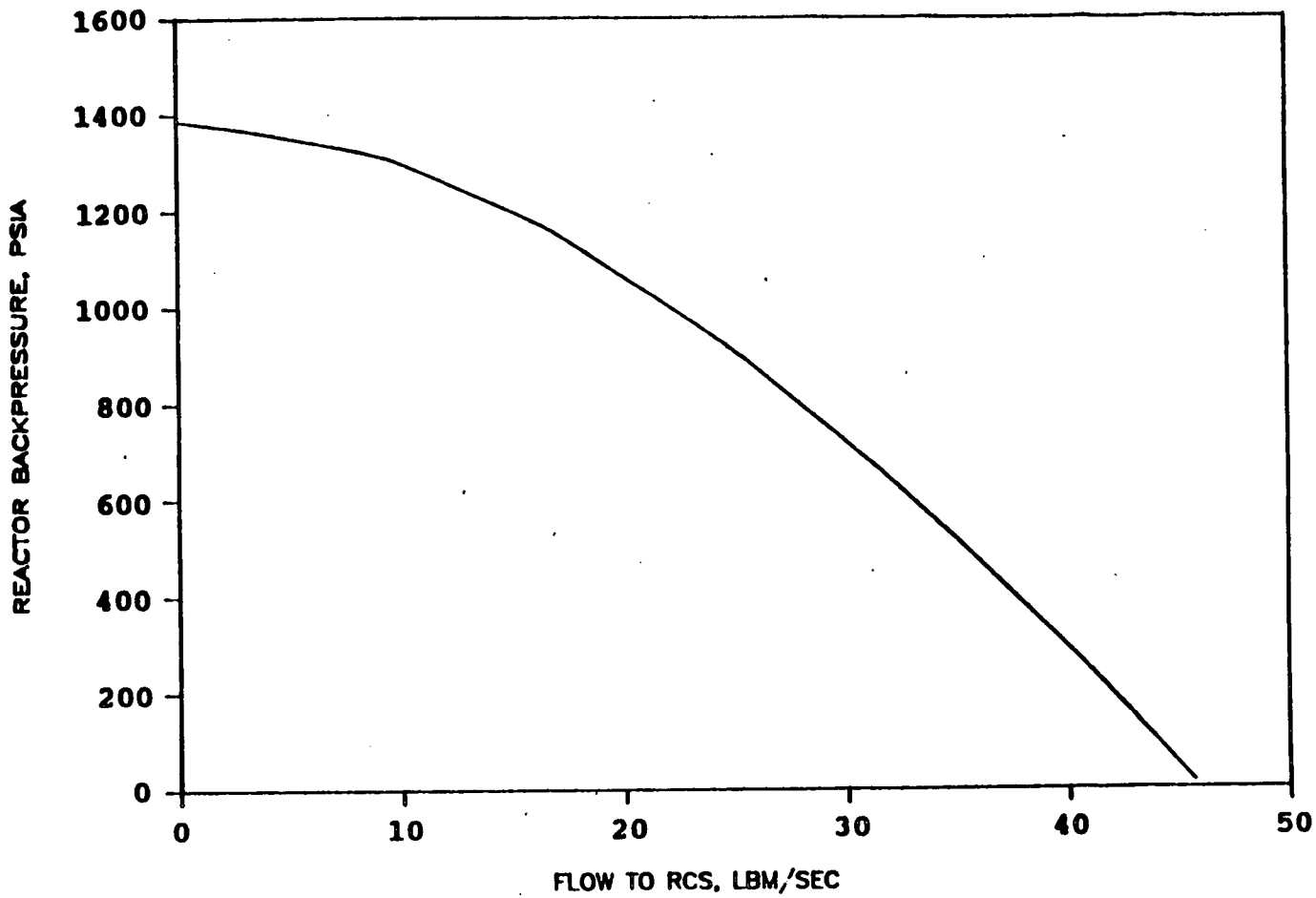
REFERENCES FOR SECTION 15.6.2

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, 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.

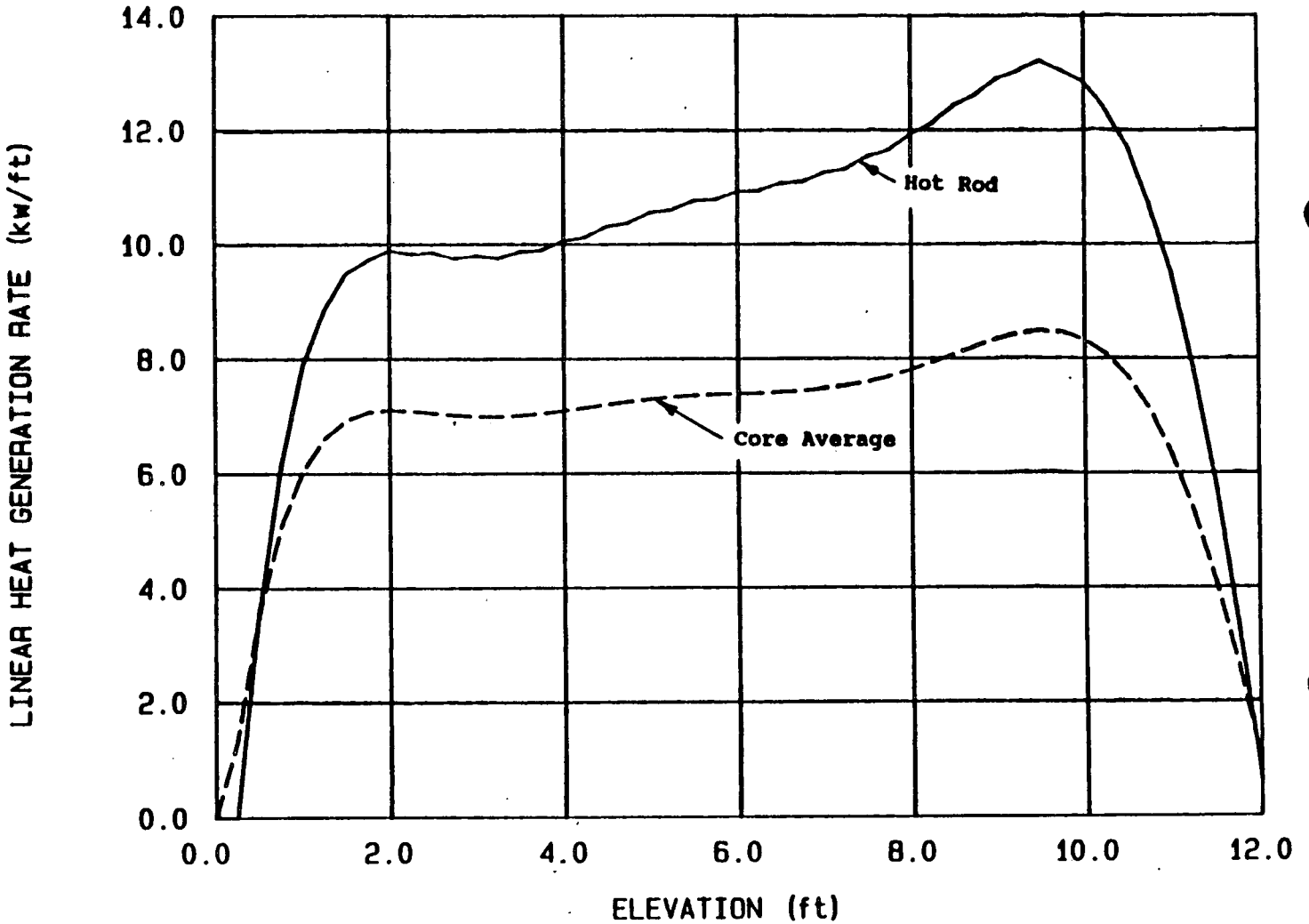
4. Bordelon, F. M., et. al., "LOCTA-IV Program: Loss of Coolant Transient Analysis", WCAP-8301, (Proprietary) and WCAP-8305, (Non-Proprietary), June 1974.



H. B. ROBINSON UNIT 2

PUMPED SAFETY INJECTION FLOW

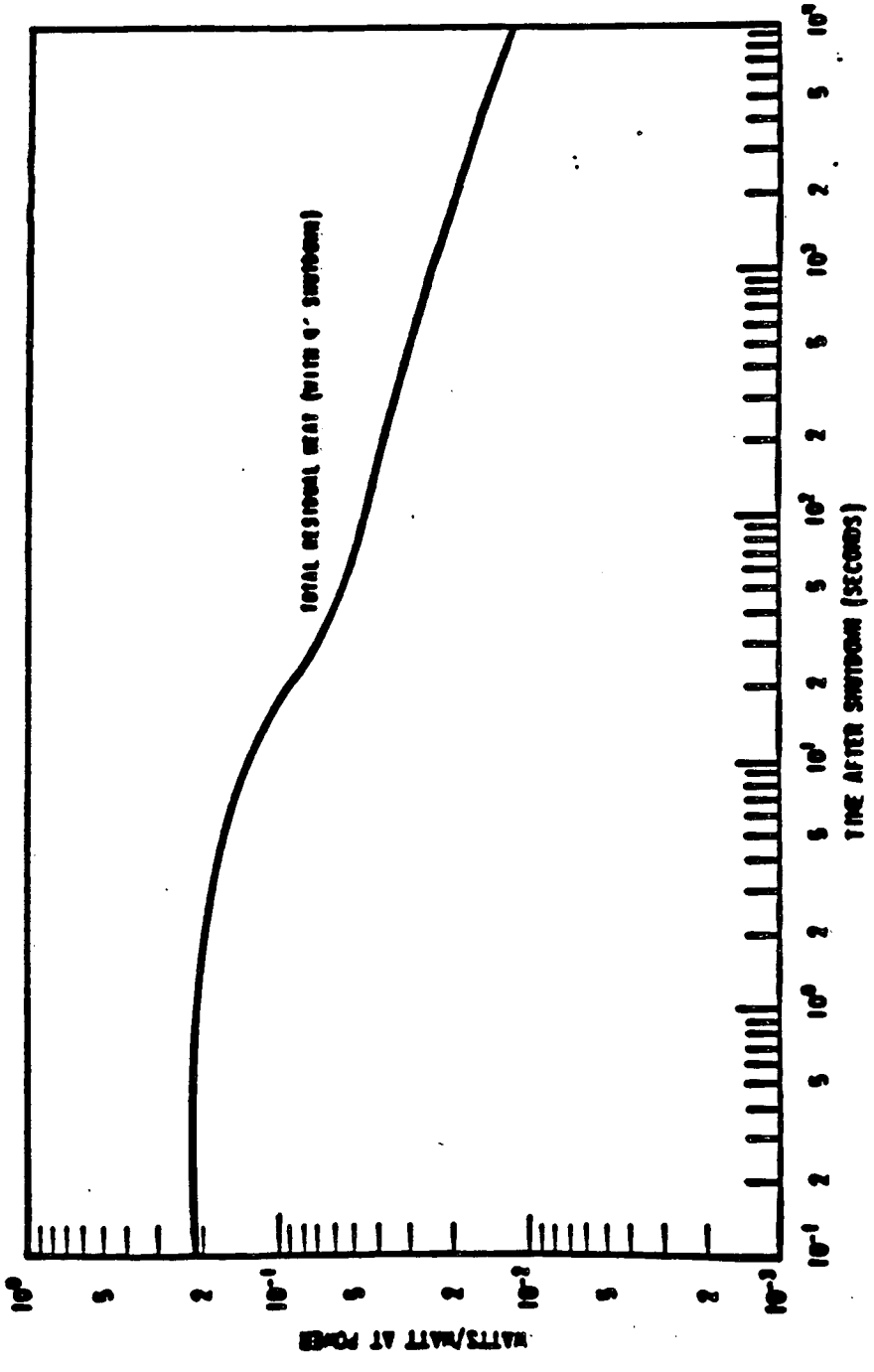
FIGURE 15.6.2-1



H. B. ROBINSON UNIT 2

SMALL BREAK LOCA AXIAL POWER SHAPES

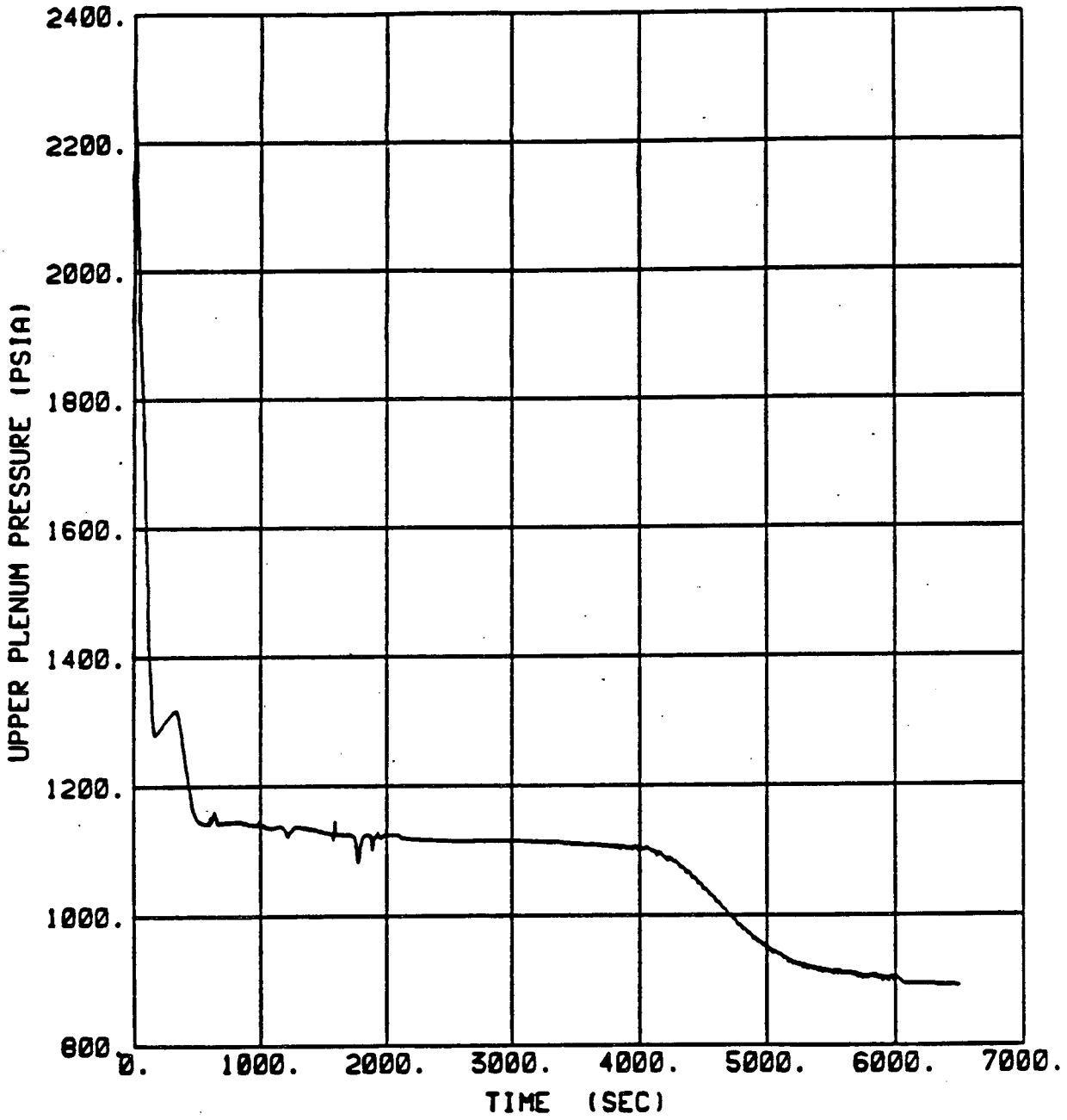
FIGURE 15.6.2-2



H. B. ROBINSON UNIT 2

CORE POWER AFTER REACTOR TRIPS

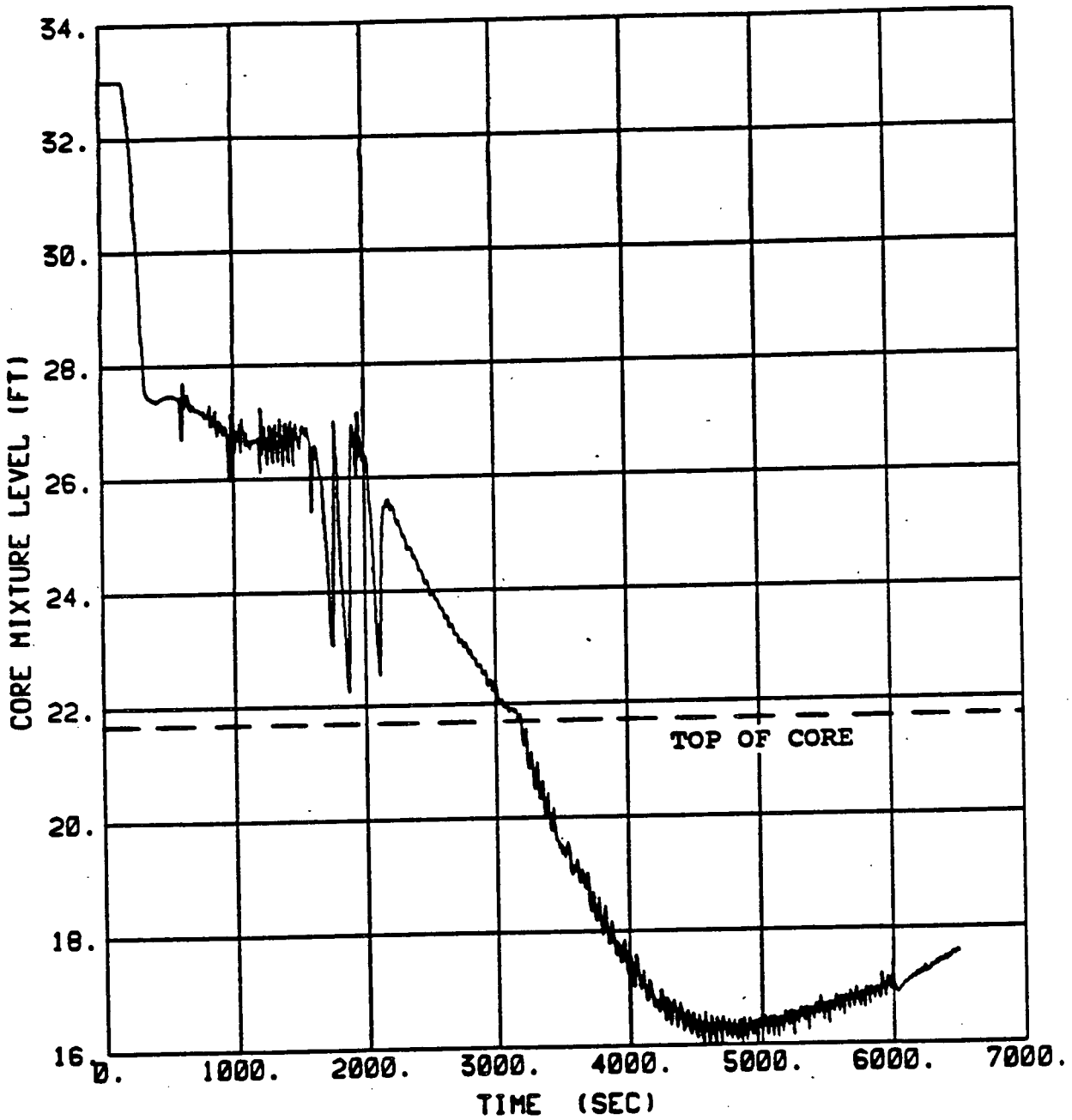
FIGURE 15.6.2-3



H. B. ROBINSON UNIT 2

UPPER PLENUM PRESSURE
1.5-INCH COLD LEG BREAK - 100% POWER

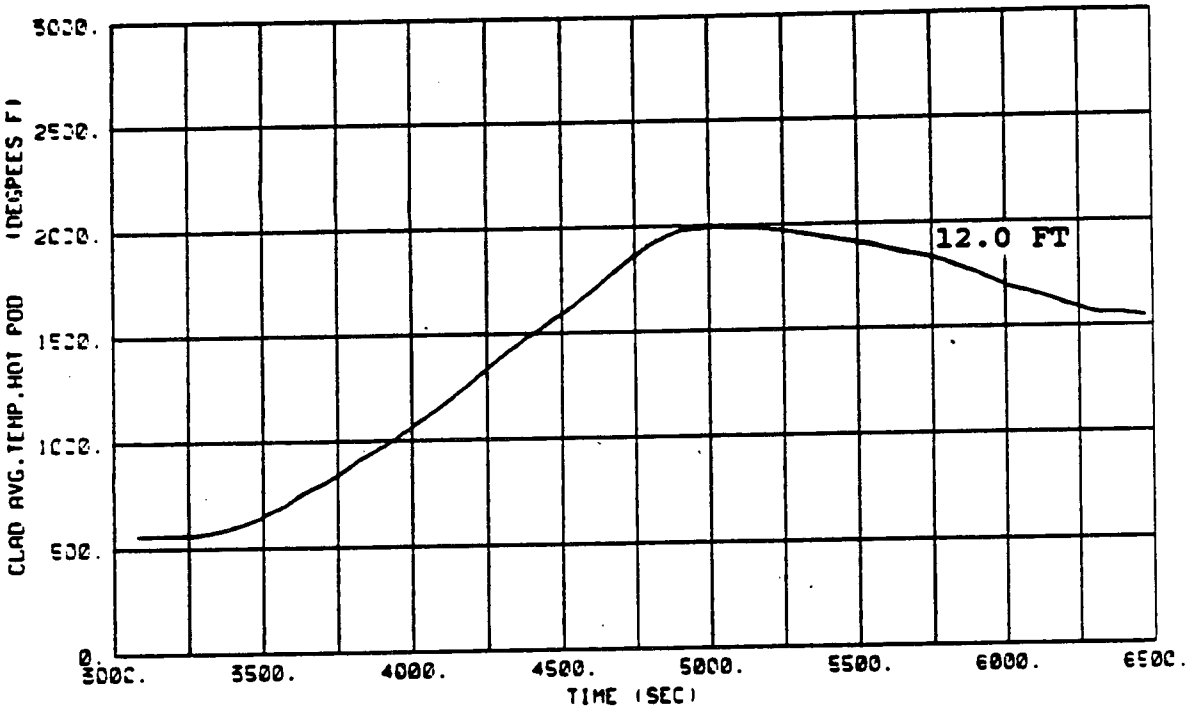
FIGURE 15.6.2-4



H. B. ROBINSON UNIT 2

CORE MIXTURE LEVEL
1.5-INCH COLD LEG BREAK - 100% POWER

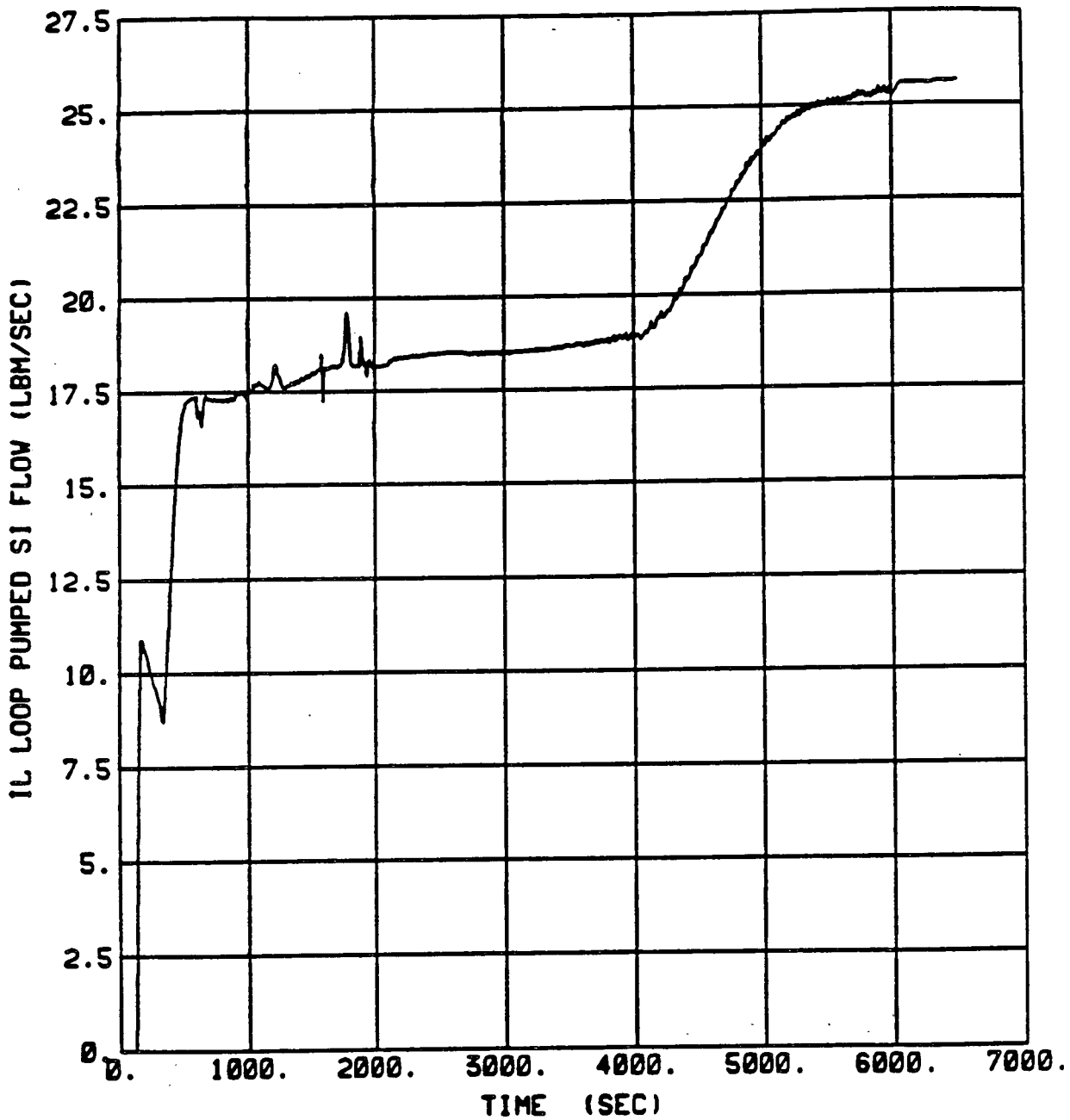
FIGURE 15.6.2-5



H. B. ROBINSON UNIT 2

HOT SPOT CLAD TEMPERATURE
1.5-INCH COLD LEG BREAK - 100% POWER

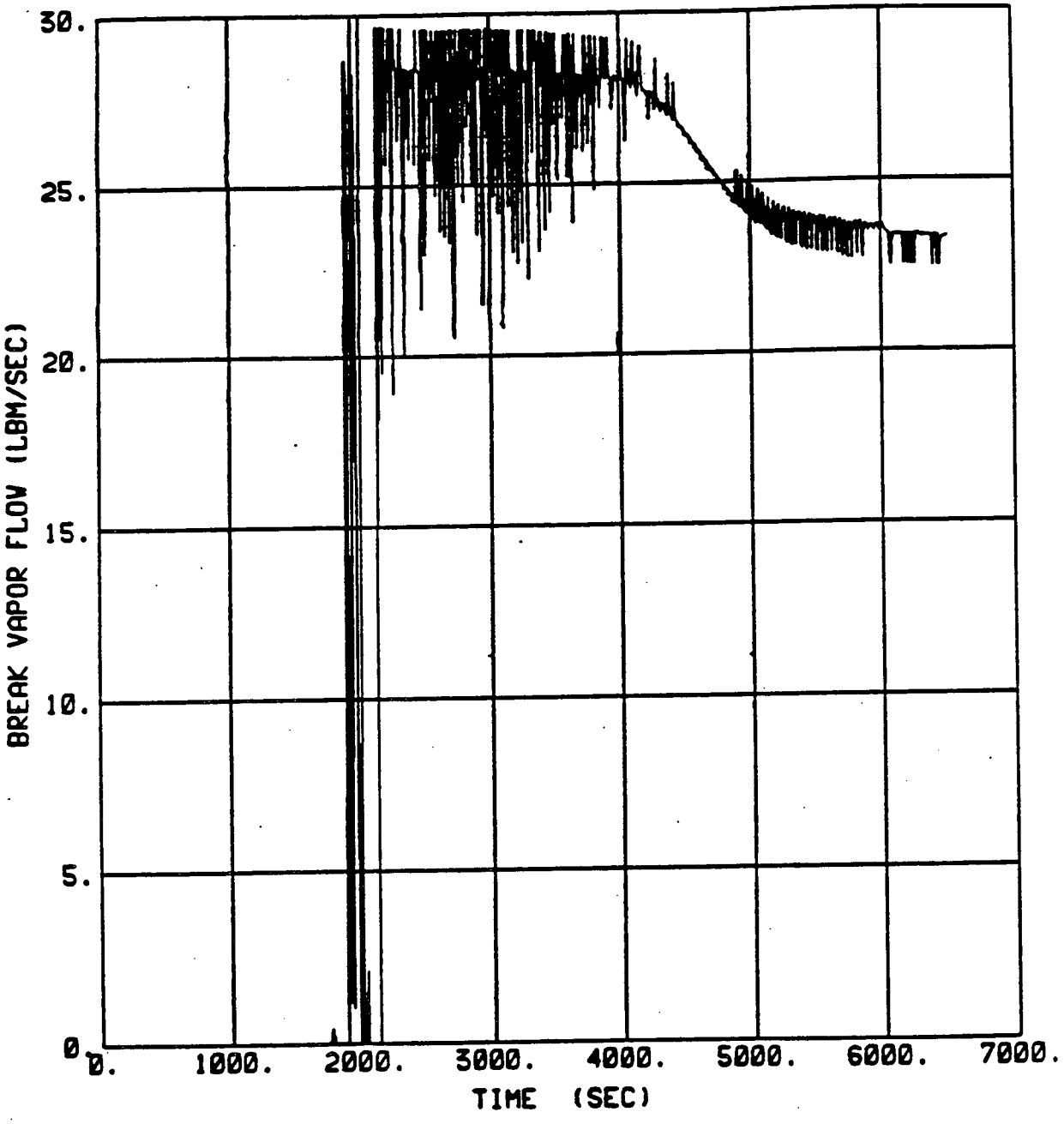
FIGURE 15.6.2-6



H. B. ROBINSON UNIT 2

INTACT LOOP PUMPED SI FLOW
1.5-INCH COLD LEG BREAK - 100% POWER

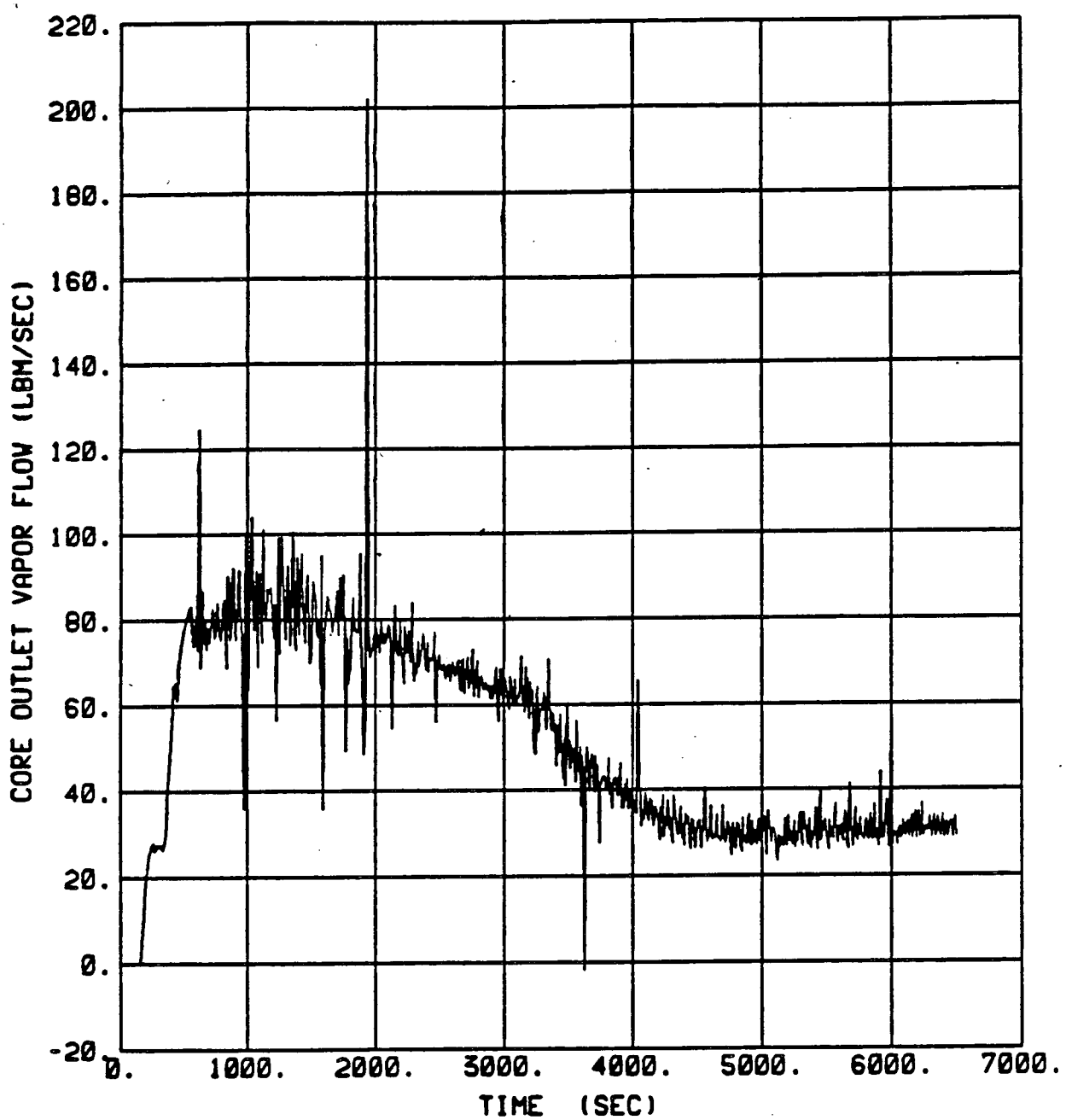
FIGURE 15.6.2-7



H. B. ROBINSON UNIT 2

BREAK VAPOR FLOW
1.5-INCH COLD LEG BREAK - 100% POWER

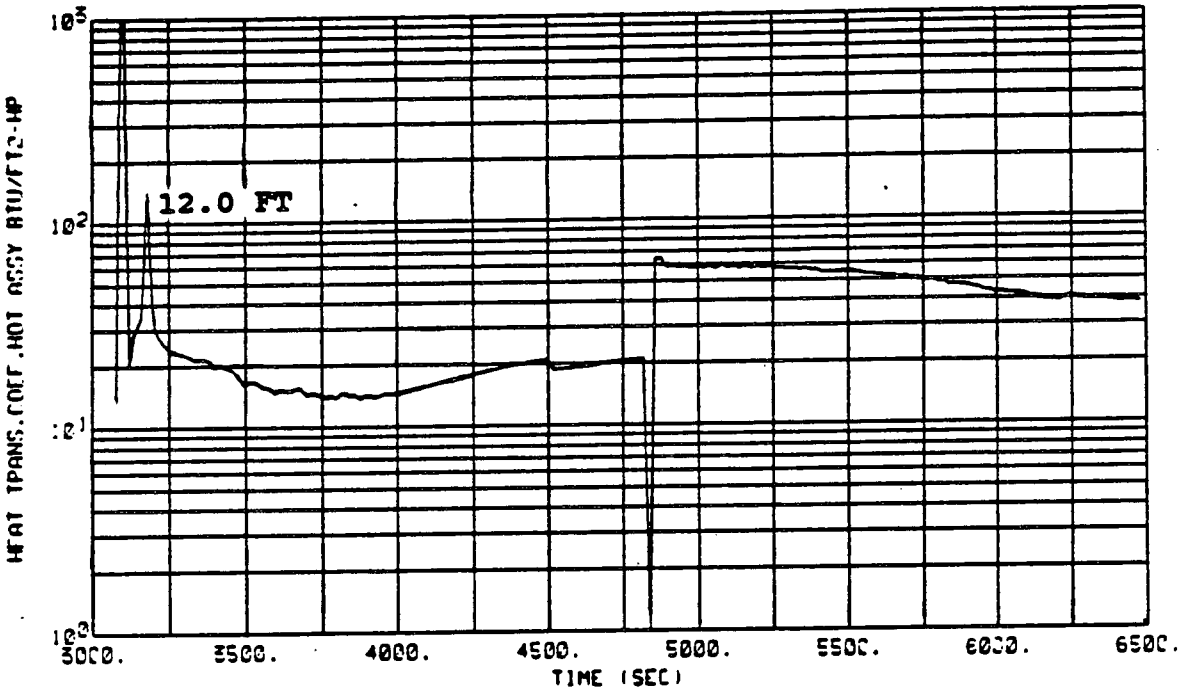
FIGURE 15.6.2-8



H. B. ROBINSON UNIT 2

CORE OUTLET VAPOR FLOW
1.5-INCH COLD LEG BREAK - 100% POWER

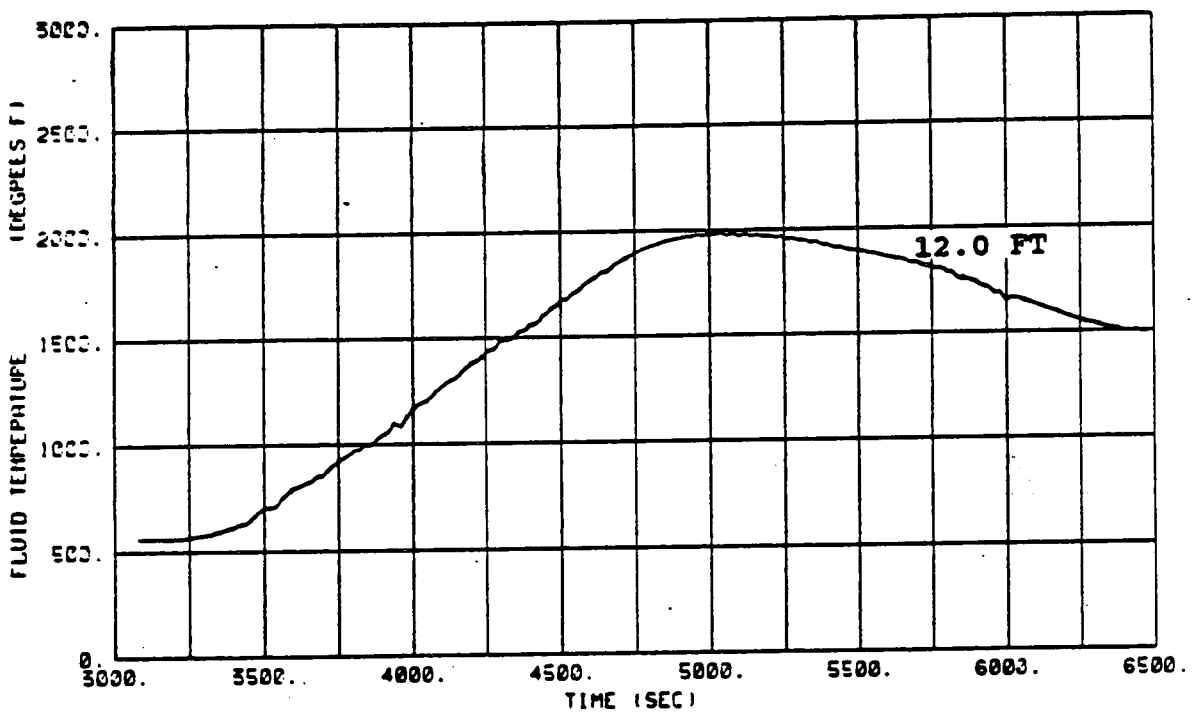
FIGURE 15.6.2-9



H. B. ROBINSON UNIT 2

CORE HEAT TRANSFER COEFFICIENT
1.5-INCH COLD LEG BREAK - 100% POWER

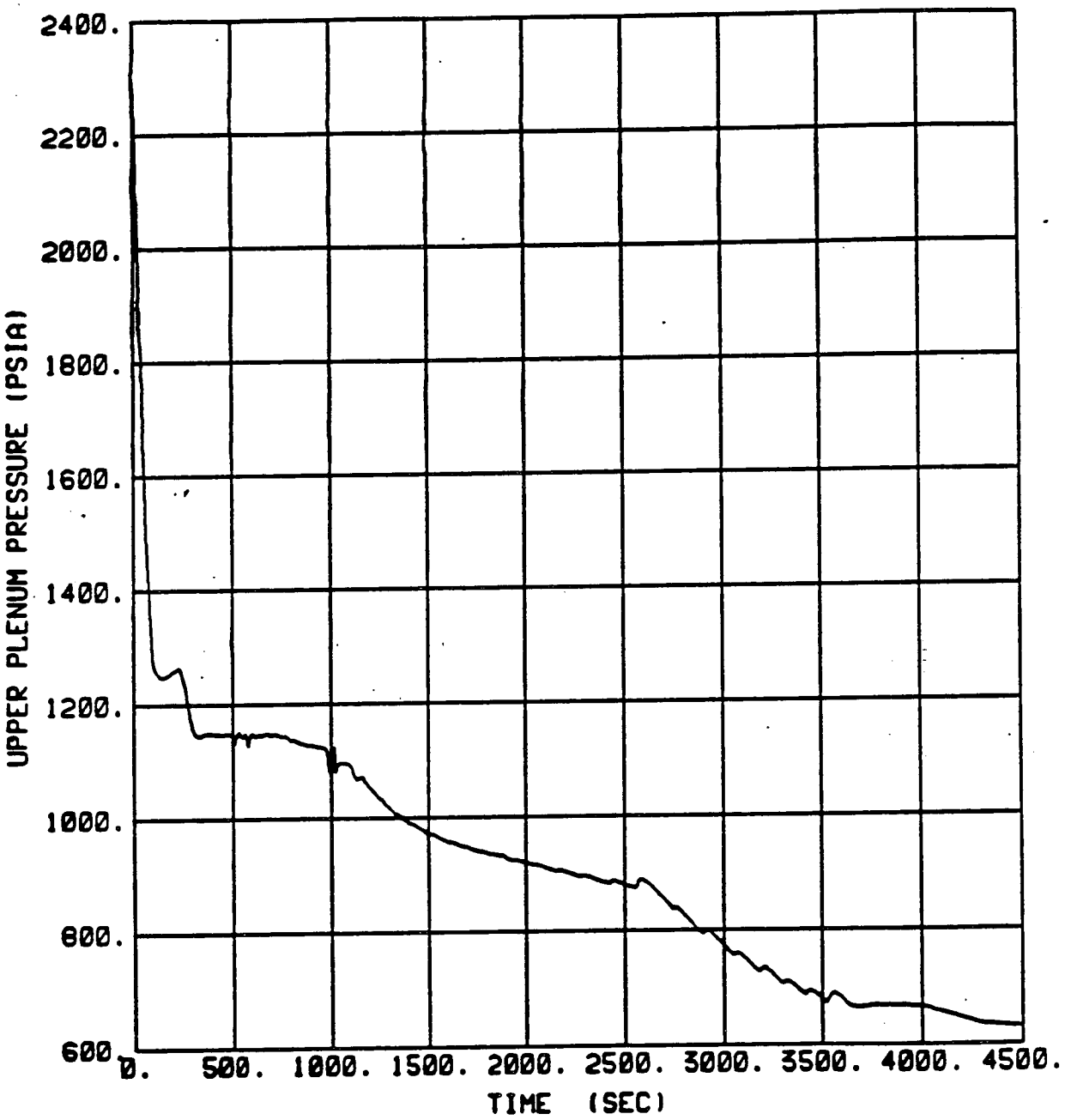
FIGURE 15.6.2-10



H. B. ROBINSON UNIT 2

HOT SPOT FLUID TEMPERATURE
1.5-INCH COLD LEG BREAK - 100% POWER

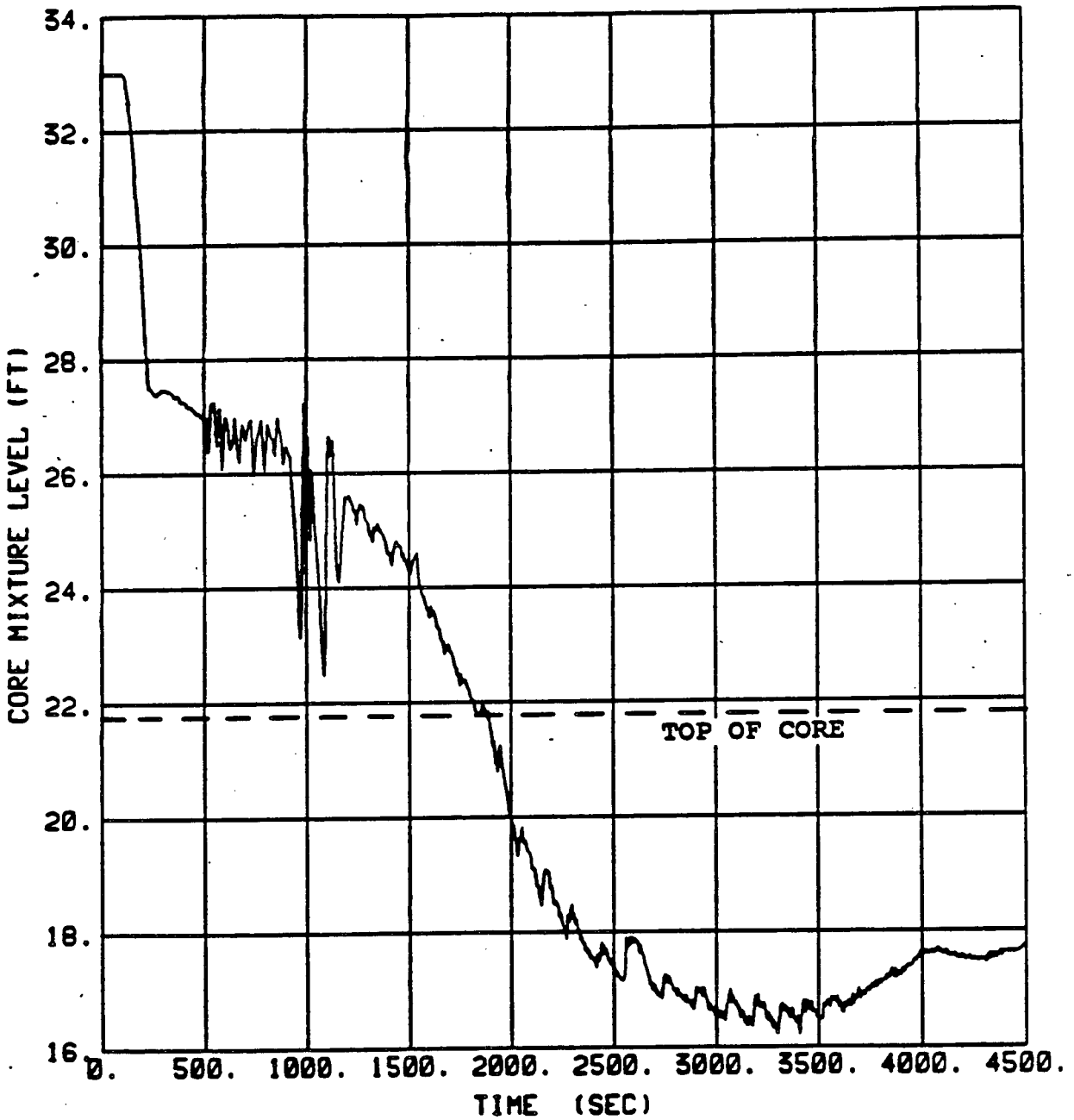
FIGURE 15.6.2-11



H. B. ROBINSON UNIT 2

UPPER PLENUM PRESSURE
2-INCH COLD LEG BREAK - 100% POWER

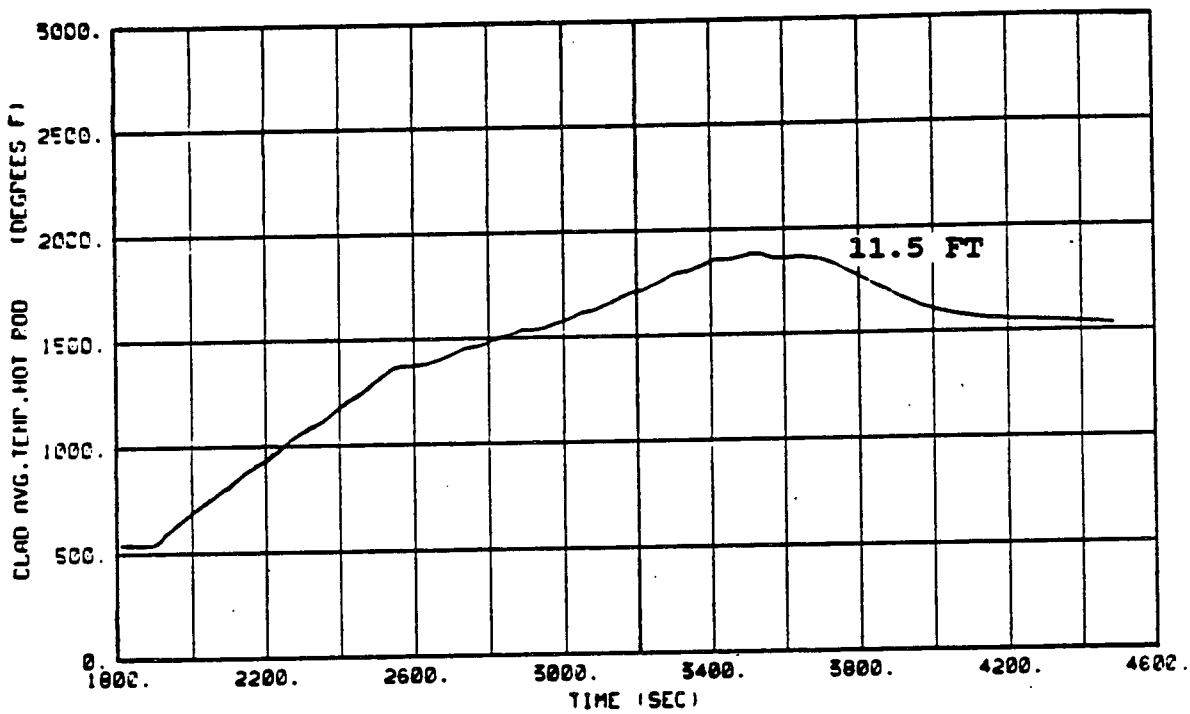
FIGURE 15.6.2-12



H. B. ROBINSON UNIT 2

CORE MIXTURE LEVEL
2-INCH COLD LEG BREAK - 100% POWER

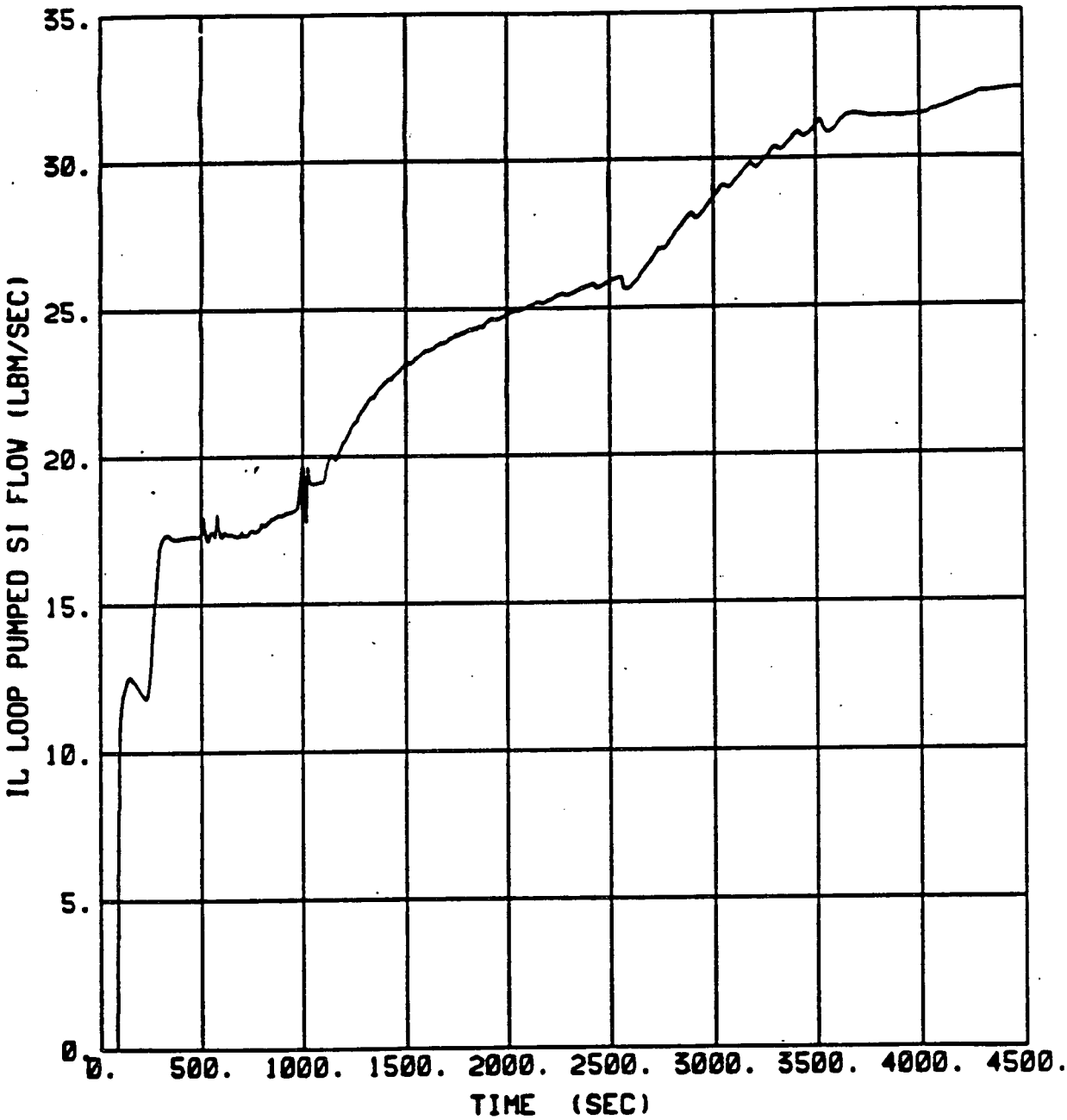
FIGURE 15.6.2-13



H. B. ROBINSON UNIT 2

HOT SPOT CLAD TEMPERATURE
2-INCH COLD LEG BREAK - 100% POWER

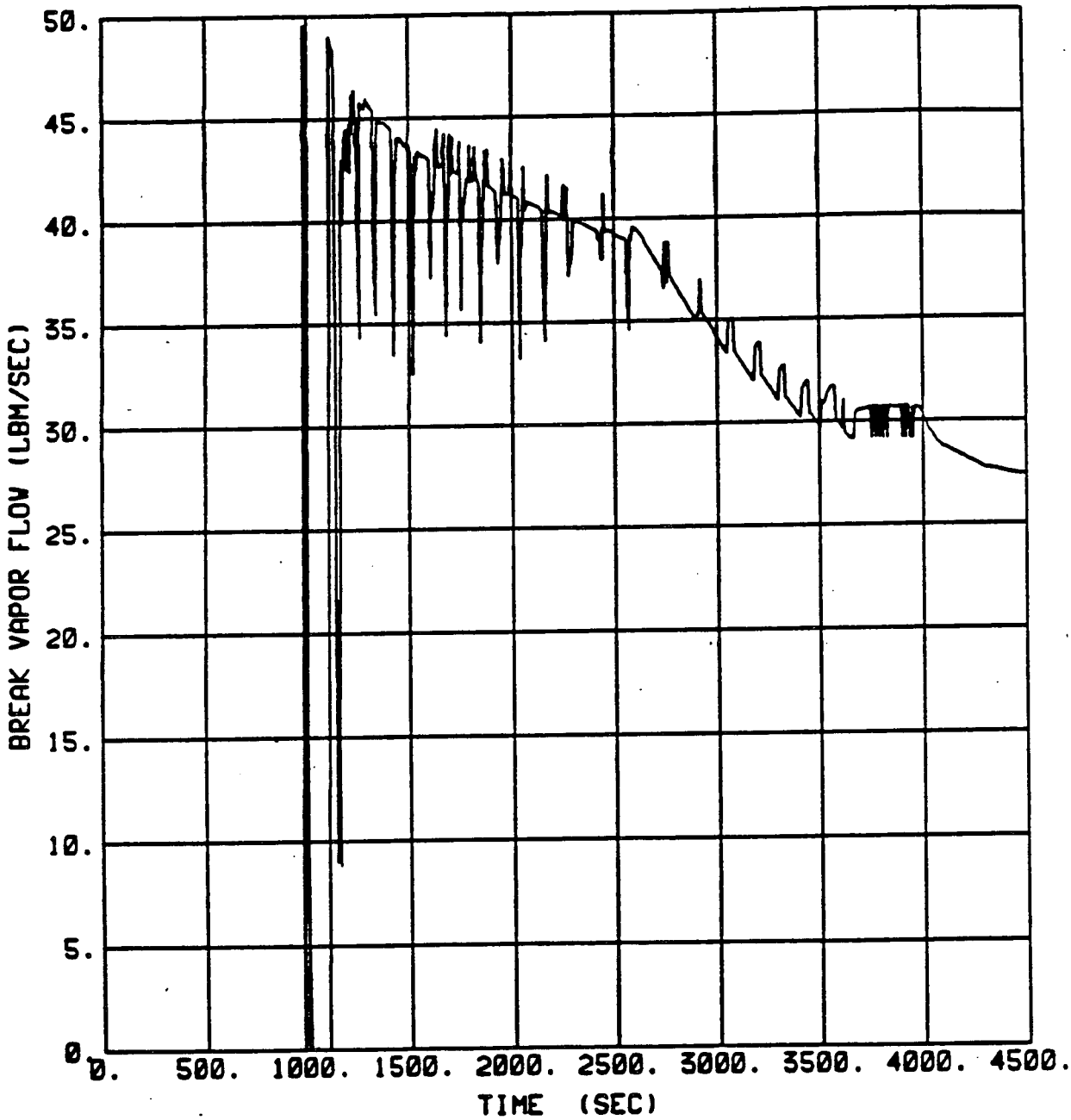
FIGURE 15.6.2-14



H. B. ROBINSON UNIT 2

INTACT LOOP PUMPED SI FLOW
2-INCH COLD LEG BREAK - 100% POWER

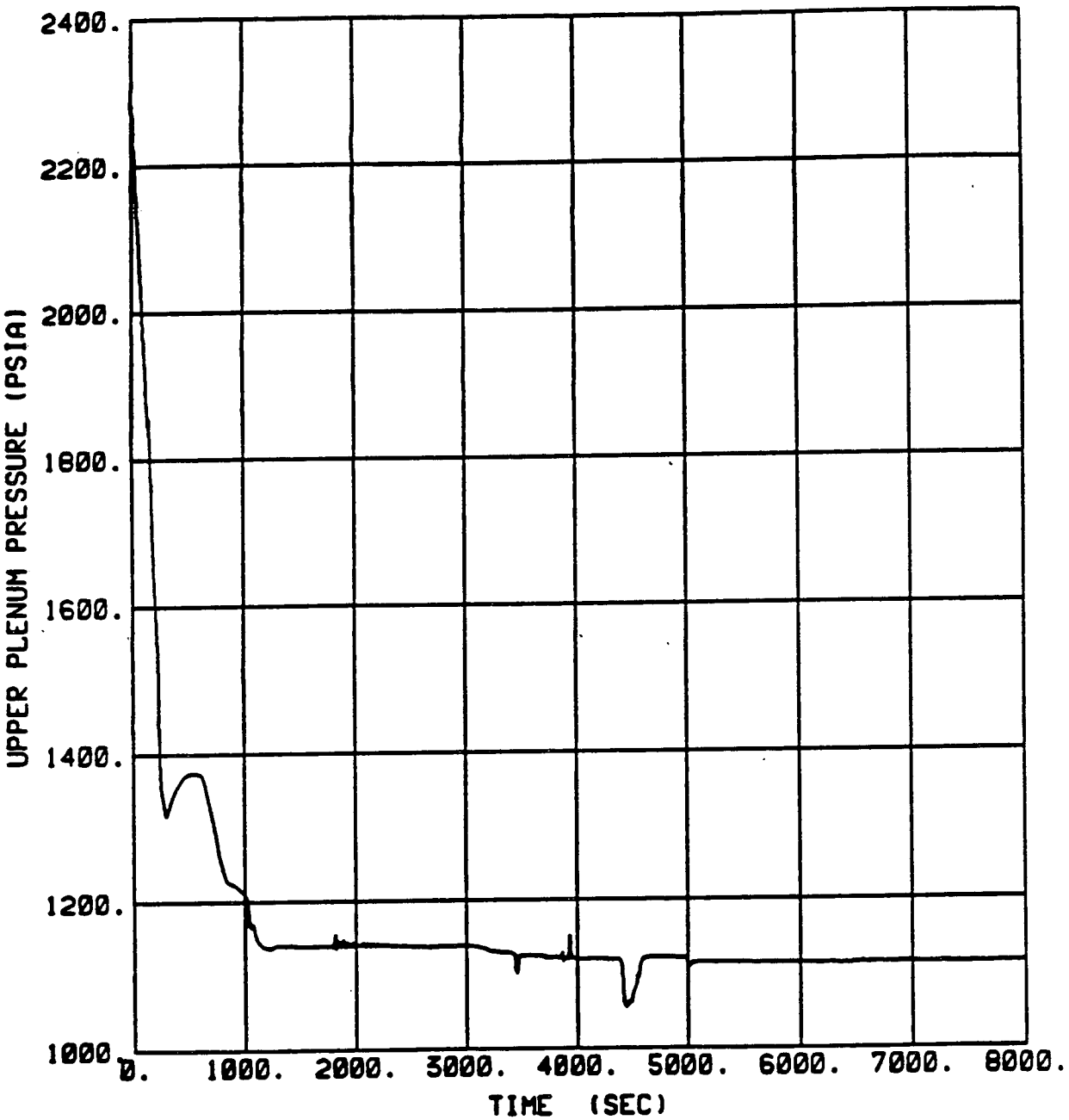
FIGURE 15.6.2-15



H. B. ROBINSON UNIT 2

BREAK VAPOR FLOW
2-INCH COLD LEG BREAK - 100% POWER

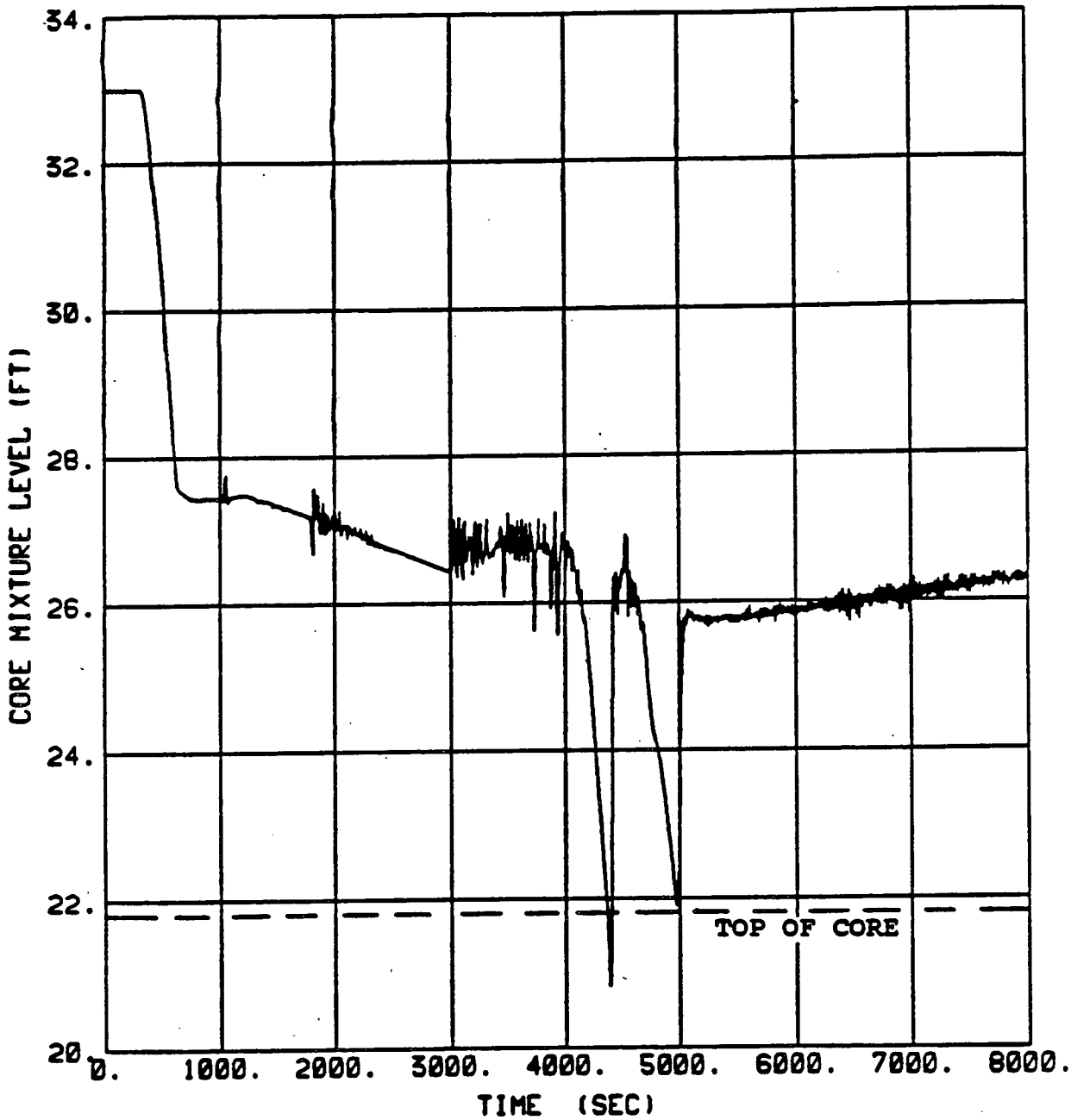
FIGURE 15.6.2-16



H. B. ROBINSON UNIT 2

UPPER PLENUM PRESSURE
1-INCH COLD LEG BREAK - 100% POWER

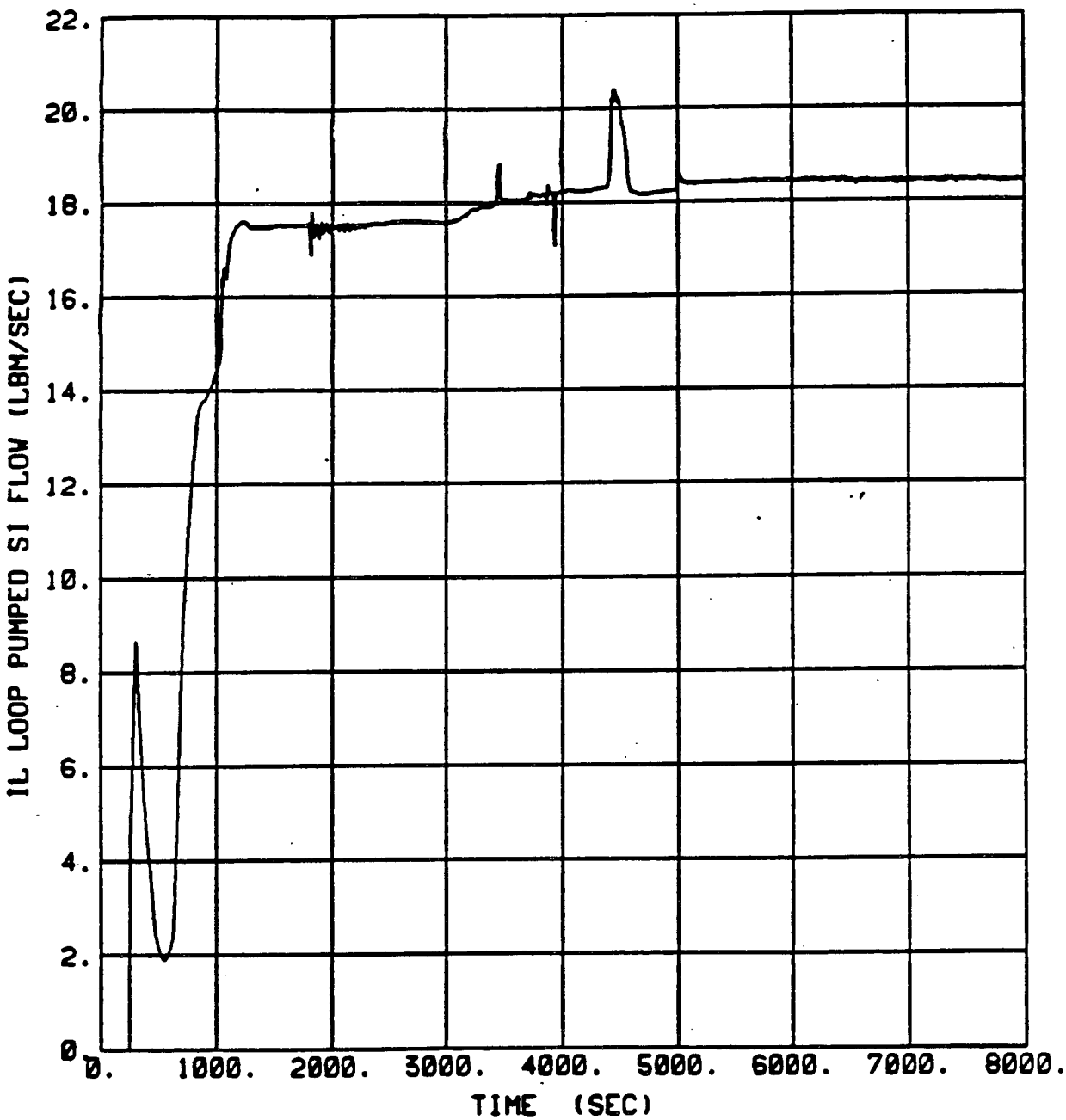
FIGURE 15.6.2-17



H. B. ROBINSON UNIT 2

CORE MIXTURE LEVEL
1-INCH COLD LEG BREAK - 100% POWER

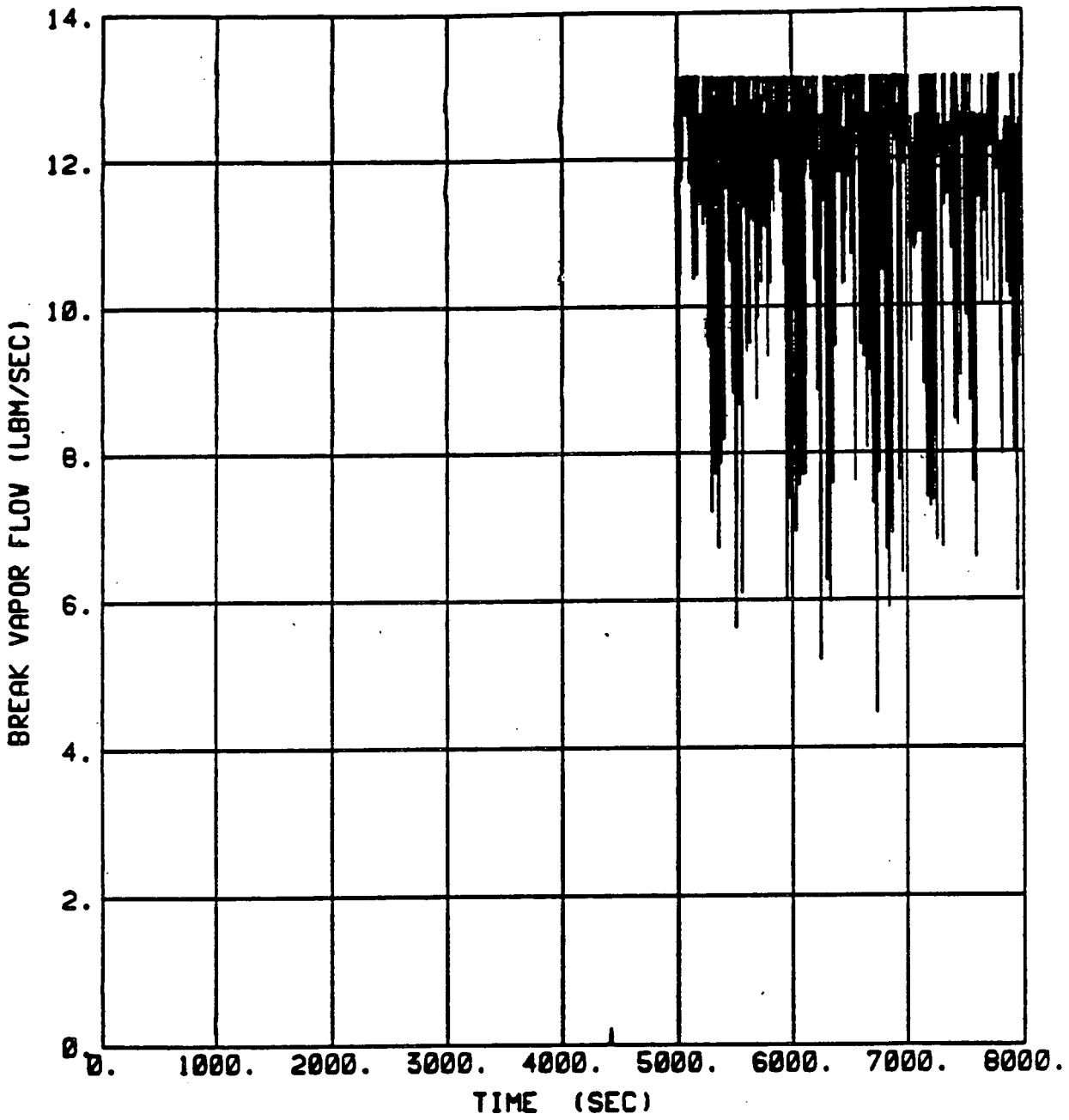
FIGURE 15.6.2-18



H. B. ROBINSON UNIT 2

INTACT LOOP PUMPED SI FLOW
1-INCH COLD LEG BREAK - 100% POWER

FIGURE 15.6.2-19



H. B. ROBINSON UNIT 2

BREAK VAPOR FLOW
1-INCH COLD LEG BREAK - 100% POWER

FIGURE 15.6.2-20