ATTACHMENT A

CHANGED TECHNICAL SPECIFICATION PAGES

3.10.1.5 Except for physics tests, if a full length control rod is withdrawn as follows: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

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:

$$F_Q(Z) \leq (2.32/P) \times K(Z) \text{ for } P > 0.5 \text{ (Note 1)}$$

$$F_Q(Z) < 4.64 \times K(Z) \text{ for } P \leq 0.5 \text{ (Note 2)}$$

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

Note 1: With only two safety injection pumps operable, each capable of automatic initiation from a separate emergency bus, this formula shall be modified to read as follows:

$$F_Q(Z) \leq (2.26/P) \times K(Z) \text{ for } P > 0.5$$

Note 2: With only two safety injection pumps operable, each capable of automatic initiation from a separate emergency bus, this formula shall be modified to read as follows:

$$F_Q(Z) < 4.52 \times K(Z) \text{ for } P \leq 0.5$$

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.

---

* 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.

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:

$$F_Q(Z) \leq \left(\frac{2.32}{P}\right) \left[\frac{K(Z)}{V(Z)}\right] \text{ for } P > 0.5 \quad \text{(Note 3)}$$

$$F_Q(Z) < 4.64 \left[\frac{K(Z)}{V(Z)}\right] \text{ for } P \leq 0.5 \quad \text{(Note 4)}$$

Note 3: With only two safety injection pumps operable, each capable of automatic initiation from a separate emergency bus, this formula shall be modified to read as follows:

$$F_Q(Z) \leq \left(\frac{2.26}{P}\right) \left[\frac{K(Z)}{V(Z)}\right] \text{ for } P > 0.5$$

Note 4: With only two safety injection pumps operable, each capable of automatic initiation from a separate emergency bus, this formula shall be modified to read as follows:

$$F_Q(Z) < 4.52 \left[\frac{K(Z)}{V(Z)}\right] \text{ for } P \leq 0.5$$

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:

$$\left[\left[\text{max. over Z of } \frac{F_Q(Z) \times V(Z)}{\frac{2.32}{P} \times K(Z)}\right] -1 \right] \times 100\% \quad \text{(Note 5)}$$

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:

$$\text{APL} = \text{minimum over Z of } \frac{2.32 \times K(Z)}{F_Q(Z) \times V(Z)} \times 100\% \quad \text{(Note 6)}$$

Note 5: With only two safety injection pumps operable, each capable of automatic initiation from a separate emergency bus, this formula shall be modified to read as follows:

$$\left[\left[\text{Max. over Z of } \frac{F_Q(Z) \times V(Z)}{\frac{2.26}{P} \times K(Z)}\right] - 1\right] \times 100\%$$

Note 6: With only two safety injection pumps operable, each capable of automatic initiation from a separate emergency bus, this formula shall be modified to read as follows:

$$\text{APL} = \text{Minimum of Z of } \frac{2.26 \times K(Z)}{F_Q(Z) \times V(Z)} \times 100\%$$

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:

$$\left[F_j(Z)\ S(Z)\right]_{max} \leq \frac{2.103/P}{\bar{R}_j\ (1 + \sigma_j)} \quad \text{(Note 7)}$$

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:

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

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. σj 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.

Note 7: With only two safety injection pumps operable, each capable of automatic initiation from a separate emergency bus, this formula shall be modified to read as follows:

$$\left[F_j(Z)\ S(Z)\right] \max. \leq \frac{2.049/P}{\bar{R}_j\ (1 + \sigma_j)}$$

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

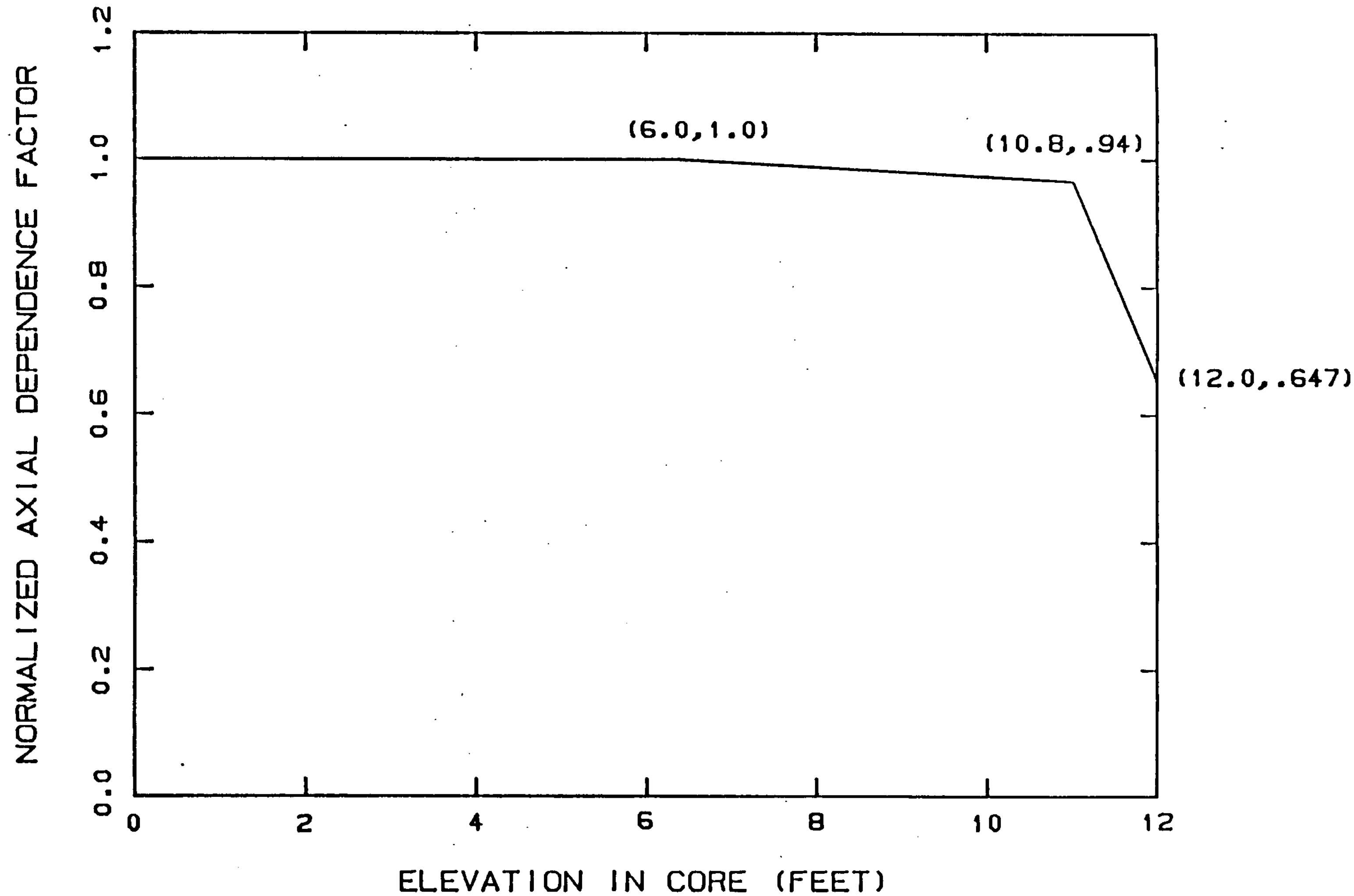


FIGURE 3.10-3 NORMALIZED AXIAL DEPENDENCE FACTOR FOR Fq VERSUS ELEVATION (PEAK Fq = 2.32)

Note: With only two safety injection pumps operable, each capable of automatic initiation from a separate emergency bus, peak Fq = 2.26.

ATTACHMENT B

SAFETY ANALYSIS

LICENSE CHANGE REQUEST
SAFETY ANALYSIS

Purpose

The allowable total core peaking limits, Fq, are being reduced in order to match the basis of a new Large Break LOCA analysis by Westinghouse. The description of this analysis is Westinghouse letter CPL-88-510, dated 2-15-88. The new Large Break analysis is required to support plant mod 951. Plant mod 951, in turn, was prompted by identification of a new, more limiting single failure that could disable two of three SI pumps.

Description

On Tech. Spec. page 3.10-12 of Tech. Spec.'s, Fq is defined as the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux.

This change reduces the maximum allowable Fq from 2.32 to 2.26 at full power conditions. The corresponding value for less than 50% power decreases from 4.64 to 4.52. The corresponding Axial Power Distribution Monitoring System (APDMS) value is reduced proportionally from 2.103 to 2.049.

Historical Information

Earlier versions of this license change request (submitted to the Plant Nuclear Safety Committee on 2-15-88 and 2-16-88) addressed the acceptability of delaying the start of the second SI pump by as much as 30 minutes in conjunction with Loss of Offsite Power. Including the SI discussion seemed appropriate then because the LOCA analysis was a direct response to identification of a new, more limiting single failure (malfunction of an Emergency Diesel Generator voltage regulator). At that time, it was expected that indication of a degraded emergency bus condition to prompt an operator to switch SI pump B to the other Emergency Diesel Generator would be sufficient. As such, no hardware change was anticipated and there was no plant modification to which the SI discussion could be tied. Since then, the potential for bus malfunction to damage a SI pump motor was identified as a problem. The solution to the problem is plant mod 951, "SI Pump B: Deletion of Auto Start". Now, with the advent of mod 951, it is much cleaner and less confusing to split the scope.
The safety implications, alone, of reducing Fq in the Tech. Spec.'s are discussed here in this license change request.
The adequacy of the reduced Fq with respect to compensating for reduced/delayed SI flow is now addressed separately as a part of plant mod 951.

LICENSE CHANGE REQUEST
SAFETY ANALYSIS

Body

Fq is a measure of power peaking in the core. Lowering the limit is reducing the extent of an allowable relative "hot spot".

This either has no effect on accident consequences (e.g., the extent of the pressure rise in the RCS following Loss of Normal Feedwater depends on the **average** or total integrated power produced in the core) or acts to reduce the severity of accident consequences (e.g., with less power being produced locally in the hotest region of the core, the same cooling flow gives a lower temperature for coolant, cladding, and fuel).

Because the extent of allowable power peaking (in itself) is only related to the effects of any accident or malfunction, the possibility or probability of occurance cannot be affected. For example, lowering the allowable Fq can no more affect the chances of initiating an accident than restricting plant load to 75% of rated power.

The adverse affects of the Tech. Spec. change include **non-safety** aspects such as
- economic or finacial penalties
- reduced operational flexibility

and
- increased surveillance requirements.

Because fairly complex computer code calculations are needed to interpret the "raw" electrical data from the incore neutron flux detectors for a core map, Fq is not regarded as a "measured" plant parameter. As such, there are no corresponding instrument alarm or trip setpoints to change in the Tech. Spec.'s.

Because the Over Temperature delta T trip, the Over Power delta T trip, and the high flux trip key on average core power to protect against DNB and fuel pellet centerline melting; it is conservative to leave these setpoints unchanged.

Although the 2.26 value for Fq was used in the Large Break LOCA analysis, 2.32 was retained as the value in the Small Break LOCA analysis. As a conservative simplification, the orginal shape of the K(z) curve in Tech. Spec. Figure 3.10-3 is being retained.

LICENSE CHANGE REQUEST
SAFETY ANALYSIS

Required Action

The revised values for Fq must be incorporated in relevent plant surviellance procedures.

If the restriction is still in effect at the end of the year, it should also be noted in FSAR Chapters 4 and 15.

Conclusion

In itself, reducing the allowable Fq is neither an Unreviewed Safety Question nor a Significant Hazard. The adequacy of this change as compensation for reduced/delayed SI flow in the Large Break LOCA analysis is addressed in plant mod 951.

ATTACHMENT C

DISCUSSION OF LOCA ANALYSIS

CPL-88-510

Westinghouse Electric Corporation **Power Systems**

Nuclear Technology Systems Division

Box 355
Pittsburgh Pennsylvania 15230-0355

February 15, 1988
NS-OPLS-OPL-II-88-092

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 WITH ALTERED HHSI FLOW PERFORMANCE

Dear Mr. Zimmerman:

At the request of Carolina Power & Light (CP&L), Westinghouse performed a Safety Evaluation to examine the effects of a reduction in the high head safety injection (HHSI) flow equivalent to an extended startup of one of the HHSI pumps on the loss-of-coolant accident (LOCA) analyses for the H. B. Robinson nuclear power plant. This evaluation reports the results of the Westinghouse large break and small break LOCA emergency core cooling system (ECCS) evaluation model analyses for a plant very similar in design to H. B. Robinson Unit 2 including the representation of Advanced Nuclear Fuels Corporation 15x15 fuel parameters.

A limiting large break LOCA analysis was performed for a double ended cold leg guillotine break with a 0.4 discharge coefficient using the NRC approved Westinghouse 1981 ECCS Evaluation Model incorporating the BART analysis methodology. The large break LOCA analysis indicates that a calculated peak cladding temperature of 2198.5°F was obtained at the 8.0-foot elevation assuming a core power level corresponding to 102% of 2300 MWth for a peak linear heat generation rate of 102% of 13.2915 kW/ft with a hot channel enthalpy rise factor of 1.65, when only one high head safety injection pump is available.

Two small break LOCA analyses were also 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 at a total core peaking factor ($F_Q T$) of 2.32 with a hot channel enthalpy rise factor of 1.65.

The analysis of a 3-inch equivalent diameter break in the cold leg resulted in the highest small break LOCA peak cladding temperature of 1771.6°F at the 12.0-foot elevation. An analysis of a 2-inch equivalent diameter break in the cold leg was performed to assure that the reduction in the safety injection flow would not shift the size of the break resulting in the highest small break LOCA peak cladding temperature calculation. The analysis of a 2-inch equivalent diameter break in the cold leg resulted in a small break LOCA peak cladding temperature of 1409.1°F at the 12.0-foot elevation, when credit is taken for manual operator action to align and start the swing high head safety injection pump.

The results of the analyses and evaluations show that the H. B. Robinson Unit 2 Nuclear Power Plant may be started and operated in compliance with the requirements of 10CFR50.46 when flow from only one high head safety injection pump is initially available and manual operator action is taken to start and align flow from a second high head safety injection pump up to 30 minutes after the start of the accident.

If you have any questions, please contact the undersigned.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

G. O. Percival, Manager
Carolina Area

D. L. Cecchett/dmr
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 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

Febrauary 1988

ATTACHMENT A

JUSTIFICATION FOR STARTUP AND 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 existed 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 until manual operator action was performed to align and start a second high head safety injection pump during a loss-of-coolant-accident (LOCA).

An interim large break LOCA analysis was performed in 1985 and 1986 for a plant similar in design to H. B. Robinson using the NRC-approved Westinghouse 1981 ECCS Evaluation Model incorporating the BART analysis methodology. The analysis indicated that the limiting peak cladding temperature of 2127°F was obtained for the double ended cold leg guillotine (DECLG) break with a discharge coefficient of 0.4 for 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 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 with 15x15 fuel manufactured by the Advanced Nuclear Fuels Corporation was performed assuming only one high head safety injection pump was operational until manual operator action was performed to align and actuate a second high head safety injection pump. The evaluation was based, in part, upon large break LOCA analyses performed using the Westinghouse 1981 ECCS Evaluation Model incorporating the BART analysis methodology and 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 large break LOCA and small break LOCA scenarios were performed assuming automatic safety injection flow delivery from only one high head safety injection pump.

The large break LOCA analysis was performed for the H.B.Robinson Unit 2 nuclear power plant using the Westinghouse 1981 ECCS Evaluation Model incorporating the BART analysis methodology. The analysis model utilized the input developed for an interim analysis of H.B.Robinson performed in 1986 for Carolina Power & Light company. The interim analysis inputs were developed from input values developed for the Turkey Point Unit 3

The small break LOCA analyses were performed using the Westinghouse Small Break LOCA ECCS Evaluation Model incorporating the NOTRUMP analysis methodology for the H.B.Robinson Unit 2 nuclear power plant. This analysis model was developed in large part using base input developed for Turkey Point Unit 3. 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. The analyses took credit for automatic safety injection flow delivery from only one high head safety injection pump. Manual operator action was credited for starting and aligning flow delivery from an additional high head safety injection pump 30 minutes into the LOCA event.

A comparison of Turkey Point Unit 3 and H.B. Robinson Unit 2 plant design parameters and components was performed to support the application of base input assumptions developed for Turkey Point Unit 3 for use in the formulation of the H.B. Robinson Unit 2 LOCA analyses. Table 1 provides the plant specific component comparison for H.B. Robinson Unit 2 and Turkey Point Unit 3 which was performed to establish that Turkey Point Unit 3 is representative of the H.B. Robinson Unit 2 plant design. The H.B. Robinson plant specific values which greatly influence the LOCA results such as power level, safety injection flow, auxilliary feedwater flow and accumulator volume were modeled in the H.B. Robinson LOCA analyses.

III. EVALUATION RESULTS

The results of the LOCA evaluations performed to address the reduction in ECCS performance resulting from a single failure event which results in only one high head safety injection pump being available to deliver safety injection flow are provided as seperate attachments. The results of the large break LOCA evaluation are provided in Attachment B and the small break LOCA results are provided in Attachment C.

ATTACHMENT A

TABLE 1

COMPARISON OF DESIGN PARAMETERS

FOR H.B. ROBINSON UNIT 2 AND TURKEY POINT UNIT 3

PARAMETER	H.B. ROBINSON UNIT 2	TURKEY POINT UNIT 3
Core Power (MWth)	2300	2200
Fuel Type	EXXON 15X15	W 15X15 OFA
Barrel Baffel Design	Downflow	Downflow
Upper Head Temperature	Thot	Thot
Upper Support Plate Design	Flat	Flat
Lower Support Plate Design	Flat	Flat
Steam Generator Type	Model 44F	Model 44F
Pressurizer Volume (ft^3)	1300	1300
Reactor Coolant Pump	Model 93 6000 hp	Model 93 6000 hp
Accumulator Total Volume (ft^3)	1200	1200
Accumulator Gas Pressure, psia	615	615

15.6.5 LOSS-OF-COOLANT ACCIDENTS

15.6.5.1 Identification of Causes and Event Consequences

For the purpose of LOCA analyses, a major LOCA is defined as a rupture 1.0 ft^2 or larger of the Reactor Primary Coolant System piping, including the double-ended rupture of the largest pipe in the RCS or of any line connected to that system up to the first closed valve.

Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. Reactor trip signal occurs when the pressurizer low pressure trip setpoint is reached. A SIS signal is actuated when the appropriate setpoint (high containment pressure) is reached. These countermeasures will limit the consequences of the accident in two ways:

a) Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat, and

b) Injection of borated water provides heat transfer from the core and prevents excessive cladding temperatures.

15.6.5.2 Method of Analysis

The mathematical model used was the revised Westinghouse 1981 Evaluation Model with BART, which has been approved for use by the NRC as meeting the requirements of an acceptable ECCS Evaluation Model as presented in Appendix K of 10CFR50. This evaluation model is comprised of the SATAN-VI, WREFLOOD, COCO, BART and LOCTA-IV codes, which are described in References 15.6.5-1 through 15.6.5-7. These codes assess the core heat transfer and determine if the core remains amenable to cooling throughout and subsequent to the blowdown, refill, and reflood phases of the LOCA. The SATAN-VI code is employed for the thermal-hydraulic transient during blowdown, while the WREFLOOD code computes this transient during refill and reflood. The COCO code is used for the complete containment pressure history for dry containments. Reflood thermal-hydraulic conditions are supplied to the BART code which performs the heat transfer calculation for the average fuel channel in the hot assembly using a mechanistic core heat transfer model. This information is then used by LOCTA-IV to calculate the fuel clad temperature and metal-water reaction of the hottest rod in the core. Additional information on the Westinghouse Evaluation Model and methodology is in References 15.6.5-8 through 15.6.5-13. 5

A double-ended guillotine break of the cold leg with a discharge coefficient of 0.4 was selected as the limiting break. The analysis was performed assuming a chopped cosine power shape, which peaked at the six-foot elevation. Additional input data is presented in Table 15.6.5-1 (Reference 15.6.5-14).

15.6.5.3 Results

Table 15.6.5-2 presents the peak clad temperature and hot spot metal reaction for the C_D = 0.4 break size. The calculated PCT was ~~2127~~ 2198.5°F occurring at ~~111~~ 135.2 seconds at an elevation of ~~7.5~~ 8.0 feet relative to the bottom of the active 5

core. The maximum local metal-water reaction was 6.76 percent, which is well below the embrittlement limit of 17 percent, as required by 10CFR50.46. The analysis was performed at 102 percent of the licensed core power of 2300 MWt at the total peaking factor of ~~2.32~~ 2.26 and enthalpy rise factor of 1.65. Table 15.6.5-3 presents the time sequence of events for the large break LOCA. Figures 15.6.5-1 through 15.6.5-16 present the transients for the principal parameters for the break analyzed.

15.6.5.4 Conclusions

This analysis demonstrates that the H. B. Robinson Unit 2 nuclear power plant
with Exxon fuel operating at 100% power, with the ~~2.32~~ 2.26 $F_Q T$ and
1.65 $F_{\Delta H}$ limits, conforms to the Acceptance Criteria as presented in
5| 10CFR50.46 when analyzed with the revised Westinghouse 1981 Evaluation Model
with BART. That is:

a) The calculated peak fuel element clad temperature provides margin to the requirement of 2200°F, based on an $F_Q T$ value of ~~2.32~~. 2.26

b) The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircaloy in the reactor.

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

d) 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.

15.6.5.5 Radiological Consequences

The results of analyses presented in this section demonstrate that the amount of radioactivity released to the environment in the event of a LOCA does not exceed the limits specified in 10CFR100.

The event causing the postulated releases is a double-ended rupture of a reactor coolant pipe, with subsequent blowdown, as described in Section 15.6.5.3. As demonstrated by the analysis described in Section 15.6.5.3, the ECCS, using emergency power, keeps cladding temperatures well below melting and limits zirconium - water reactions to an insignificant level, assuring that the core remains intact and in place. As a result of the increase in cladding temperature and the rapid depressurization of the core, however, some cladding failure may occur in the hottest regions of the core. For this reason, the entire inventory of volatile fission products contained in the pellet-cladding gap is assumed to be released during the time the core is being flooded by the ECCS. Of this gap inventory, 50 percent of the halogens and 100 percent of the noble gases are assumed to be released to the containment vessel atmosphere.

TABLE 15.6.5-1

CALCULATION BASIS

License Core Power, MWt	2300
Power Used for Analysis, MWt*	2346
Peak Linear Power for Analysis, kw/ft*	13.56
Total Peaking Factor, F_QT	2.26
Enthalpy Rise, Nuclear, $F^T{}_H$	1.65
Steam Generator Tube Plugging (%)	5.00

* Including 1.02 Factor for Power Uncertainties

TABLE 15.6.5-2

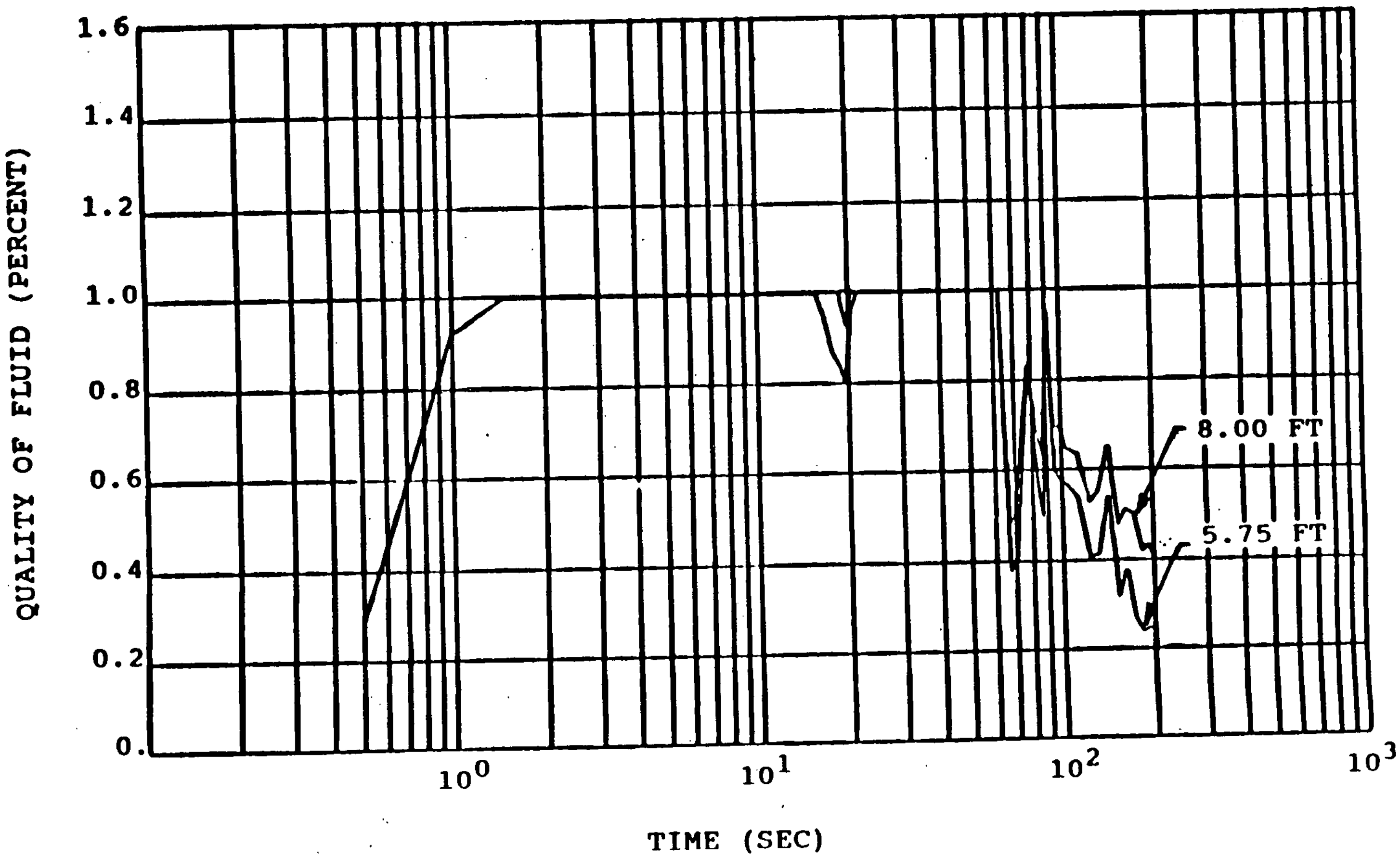
ANALYSIS RESULTS

Peak Clad Temperature (PCT), °F	2198.5
Peak Clad Temperature Reached, (sec)	135.2
Peak Clad Temperature Location, ft.	8.0
Local Zr/H_2O Reactor (max.) %	7.14
Local Zr/H_2O Location, ft. from Bottom	5.75
Total H_2 Generation, % of Total Zr Reacted	< 0.3
Hot Rod Burst Time, sec.	49.0
Hot Rod Burst Location, ft.	5.75

TABLE 15.6.5-3

LOCA/ECCS TIME SEQUENCE OF EVENTS
C_D = 0.4 DECLG BREAK

Event	Time (sec)
Start	0.0
Safety Injection Signal	0.92
Accumulator Injection	15.1
End-of-Bypass	31.22
Safety Injection Pump	25.92
Bottom-of-Core Recovery	50.068
Accumulators Empty	56.47
Peak Clad Temperature Reached	135.2

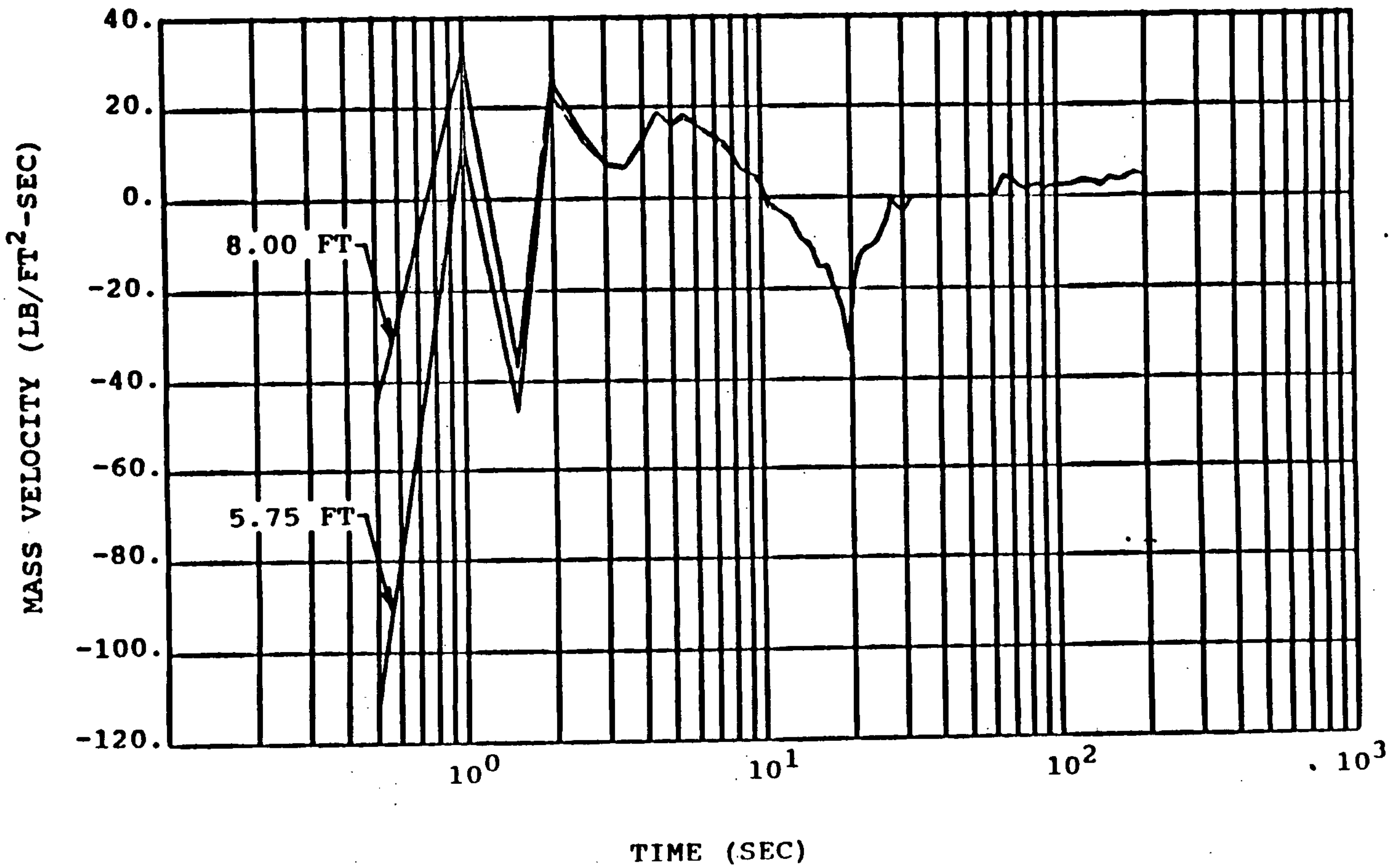


AMENDMENT NO. 6

H. B. ROBINSON
UNIT 2
Carolina Power & Light Company
UPDATED FINAL
SAFETY ANALYSIS REPORT

FLUID QUALITY
DECLG (C_D = 0.4)

FIGURE
15.6.5-1



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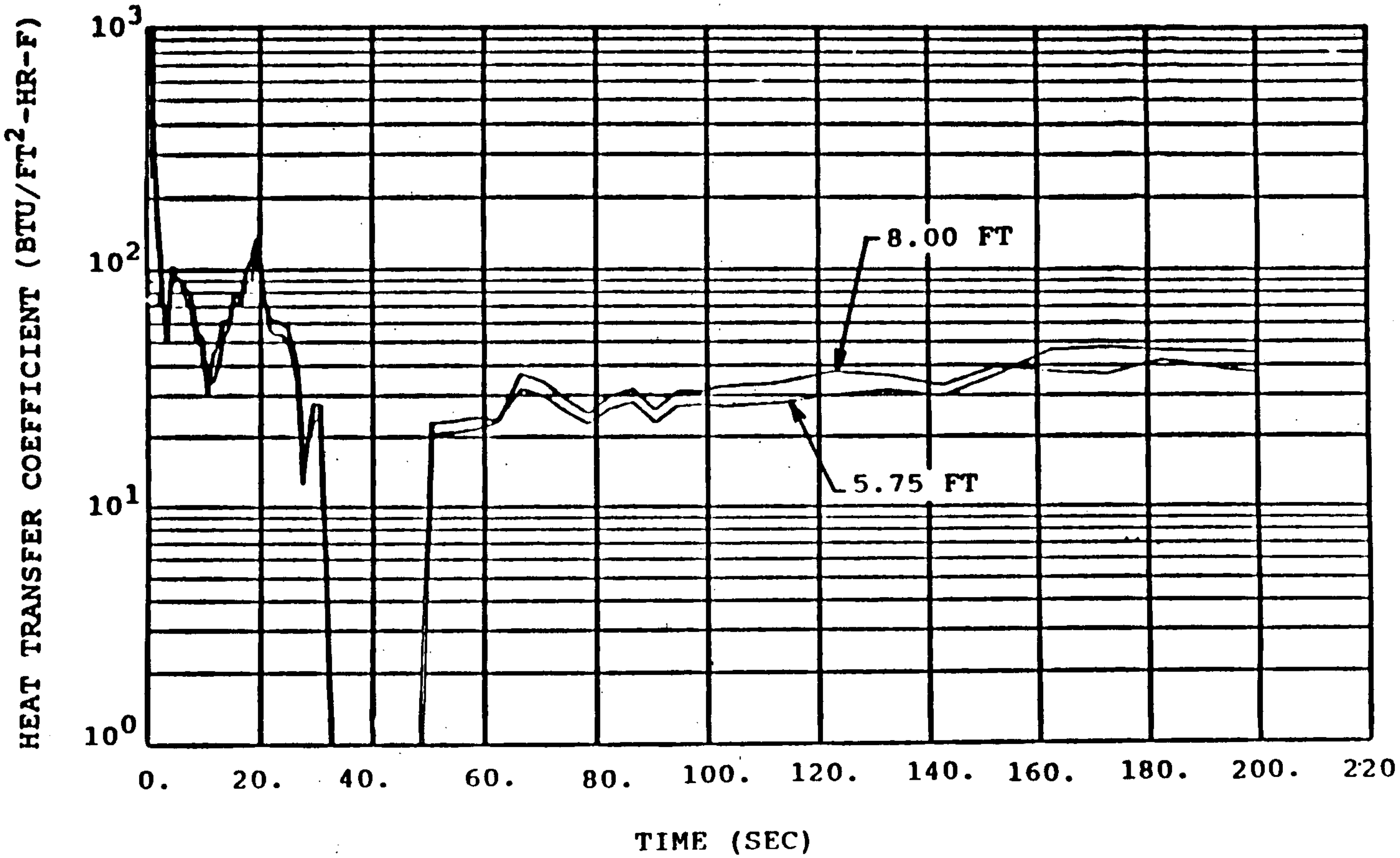
MASS VELOCITY
DECLG (C_D = 0.4)

FIGURE
15.6.5-2



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HEAT TRANSFER COEFFICIENT
DECLG (C_D = 0.4)

FIGURE
15.6.5-3

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CORE PRESSURE
DECLG (C_D = 0.4)

FIGURE
15.6.5-4

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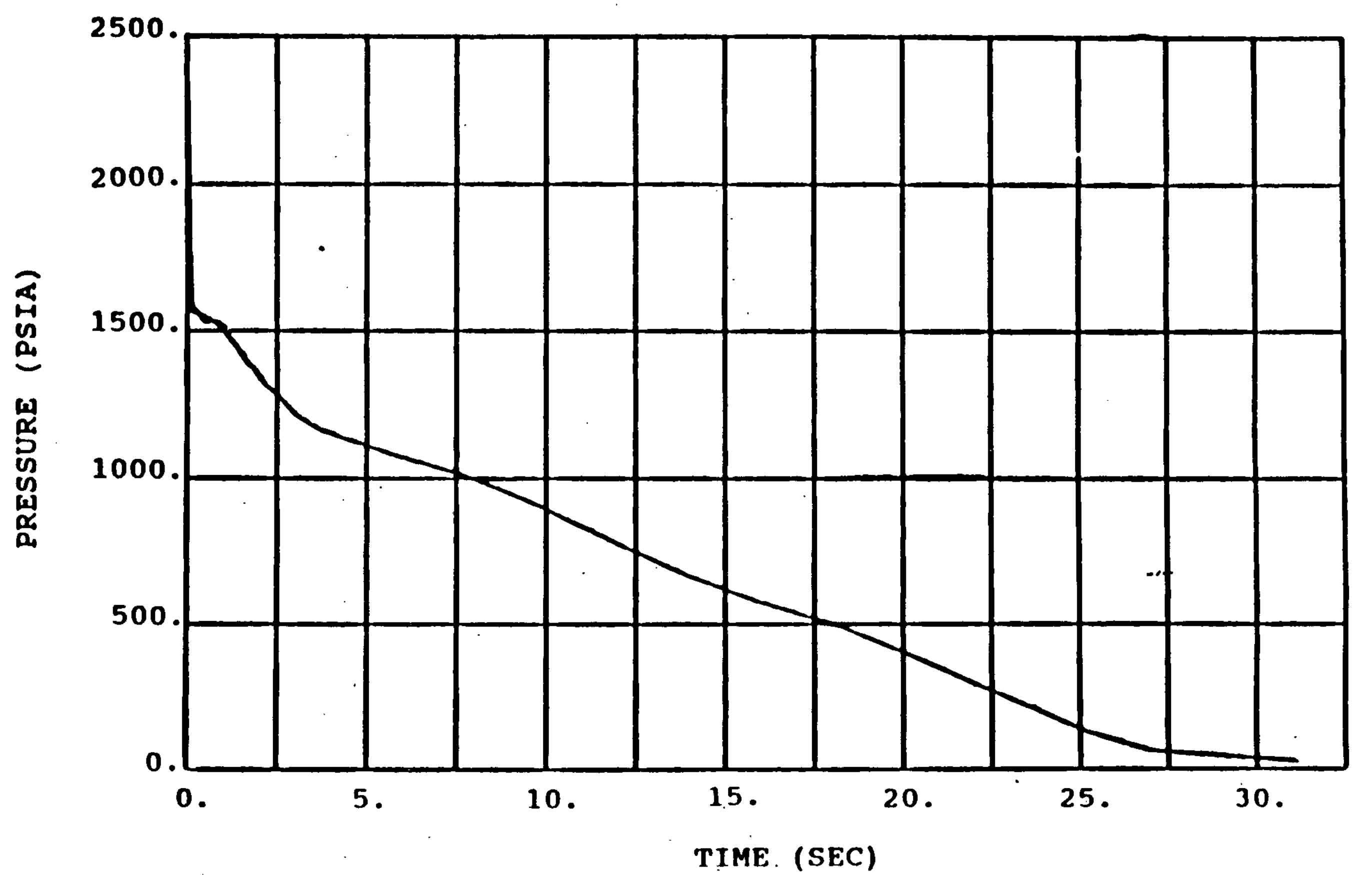

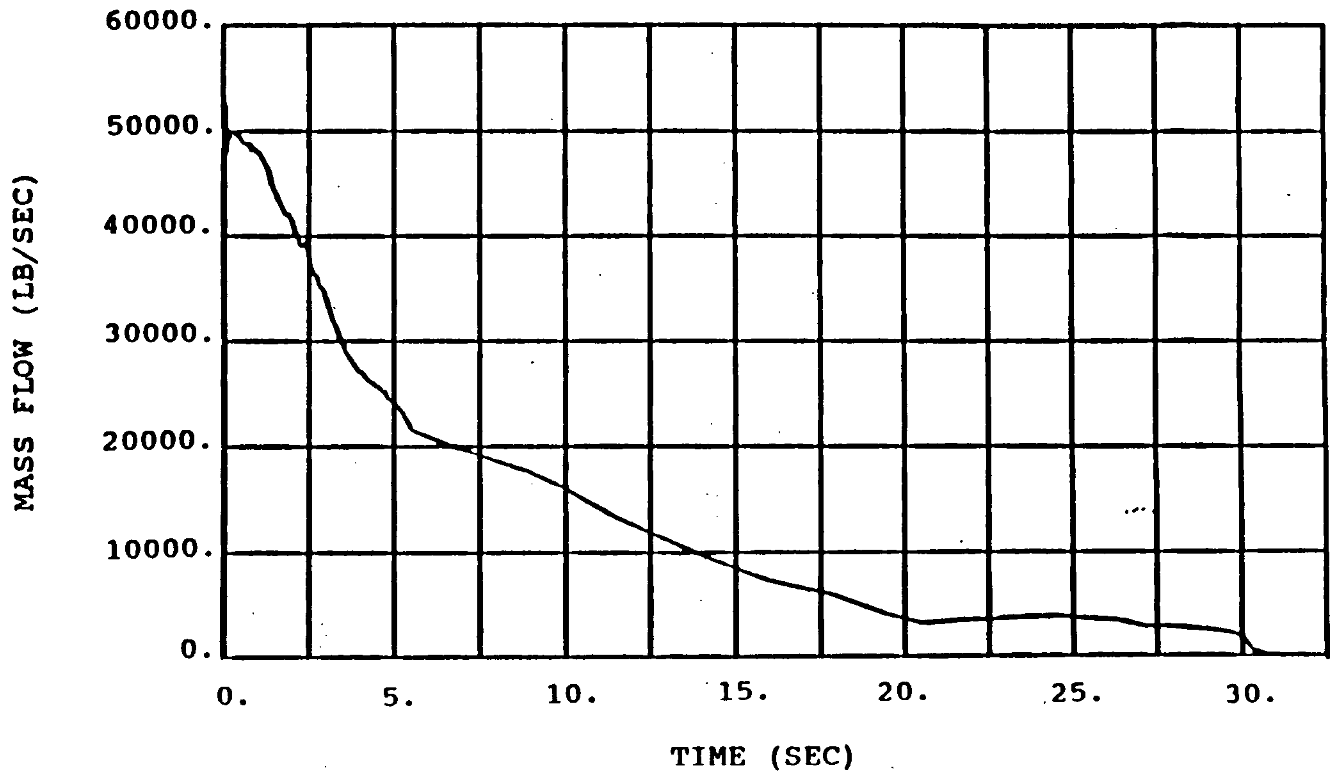

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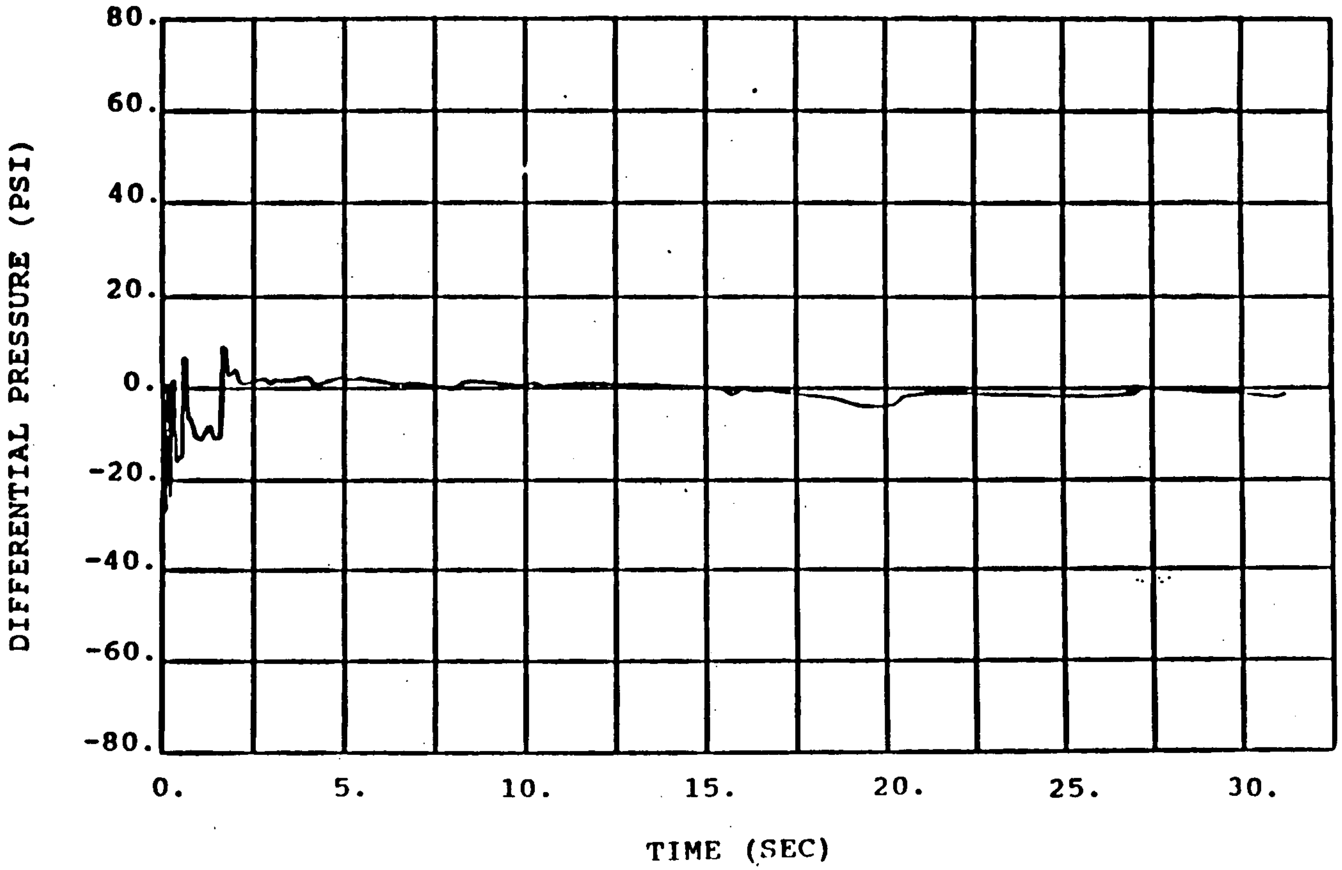
BREAK FLOW RATE
DECLG (C_D = 0.4)

FIGURE
15.6.5-5



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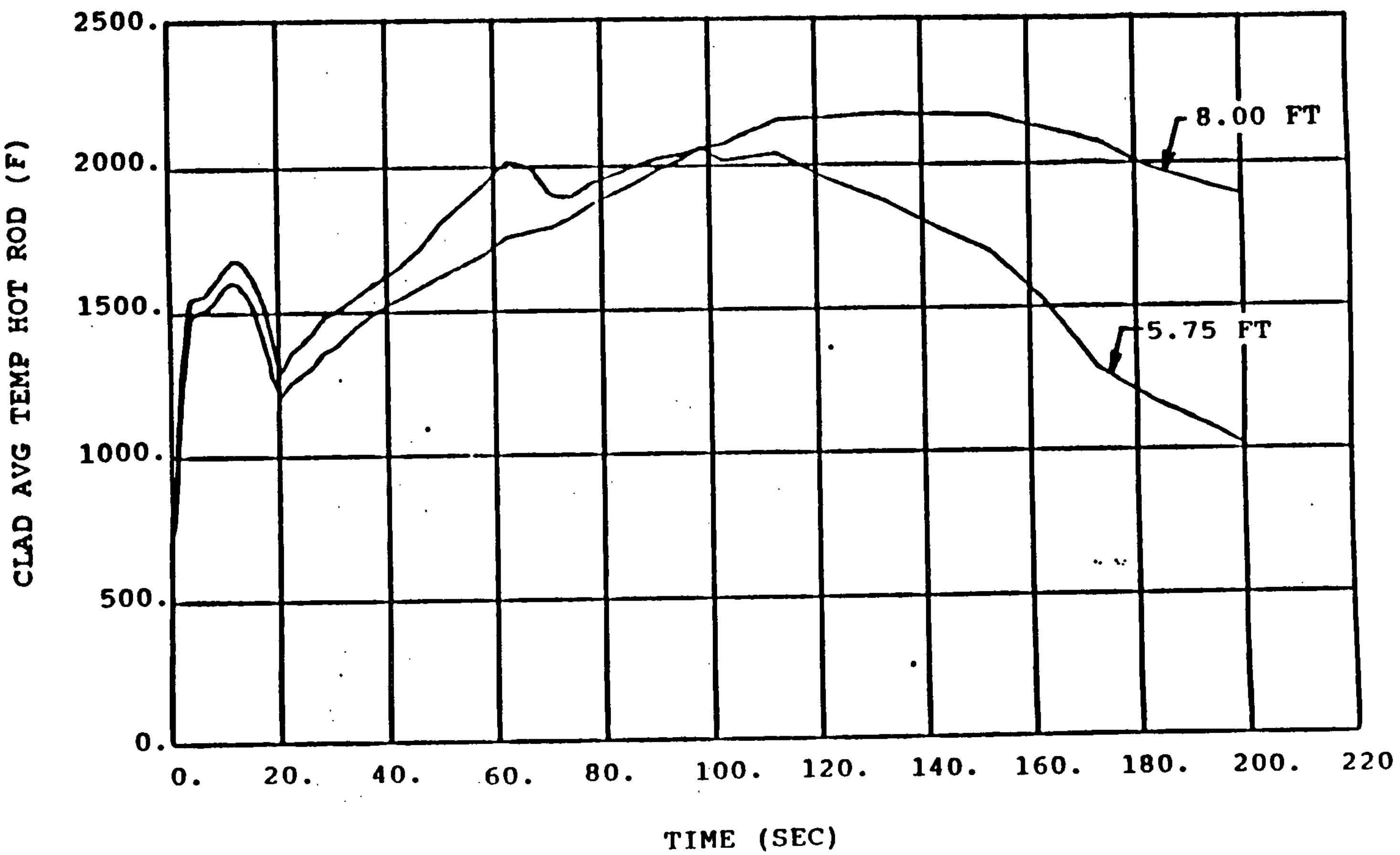
CORE PRESSURE DROP
DECLG (C_D = 0.4)

FIGURE
15.6.5-6



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PEAK CLAD TEMPERATURE
DECLG (C_D=0.4)

FIGURE
15.6.5-7

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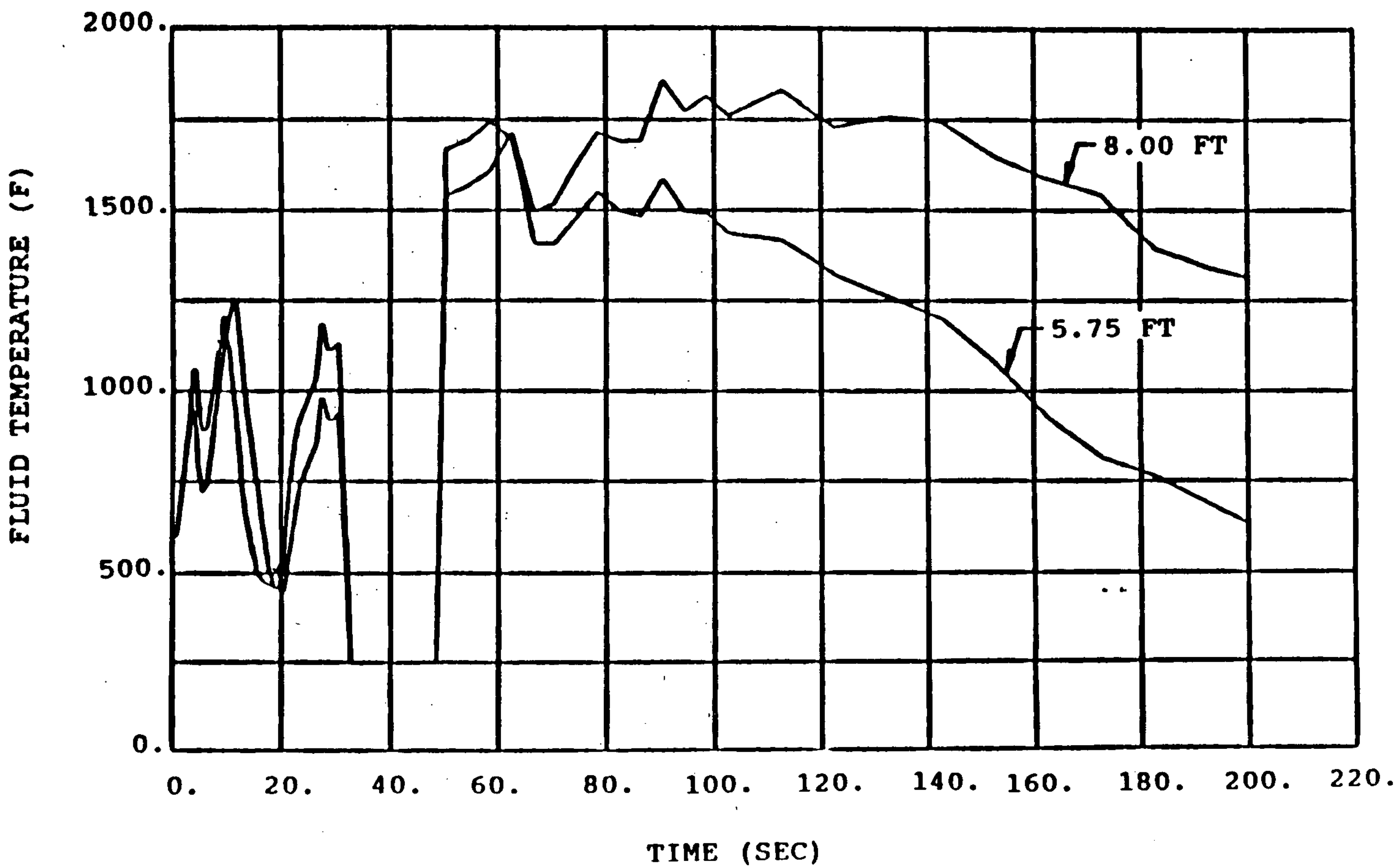

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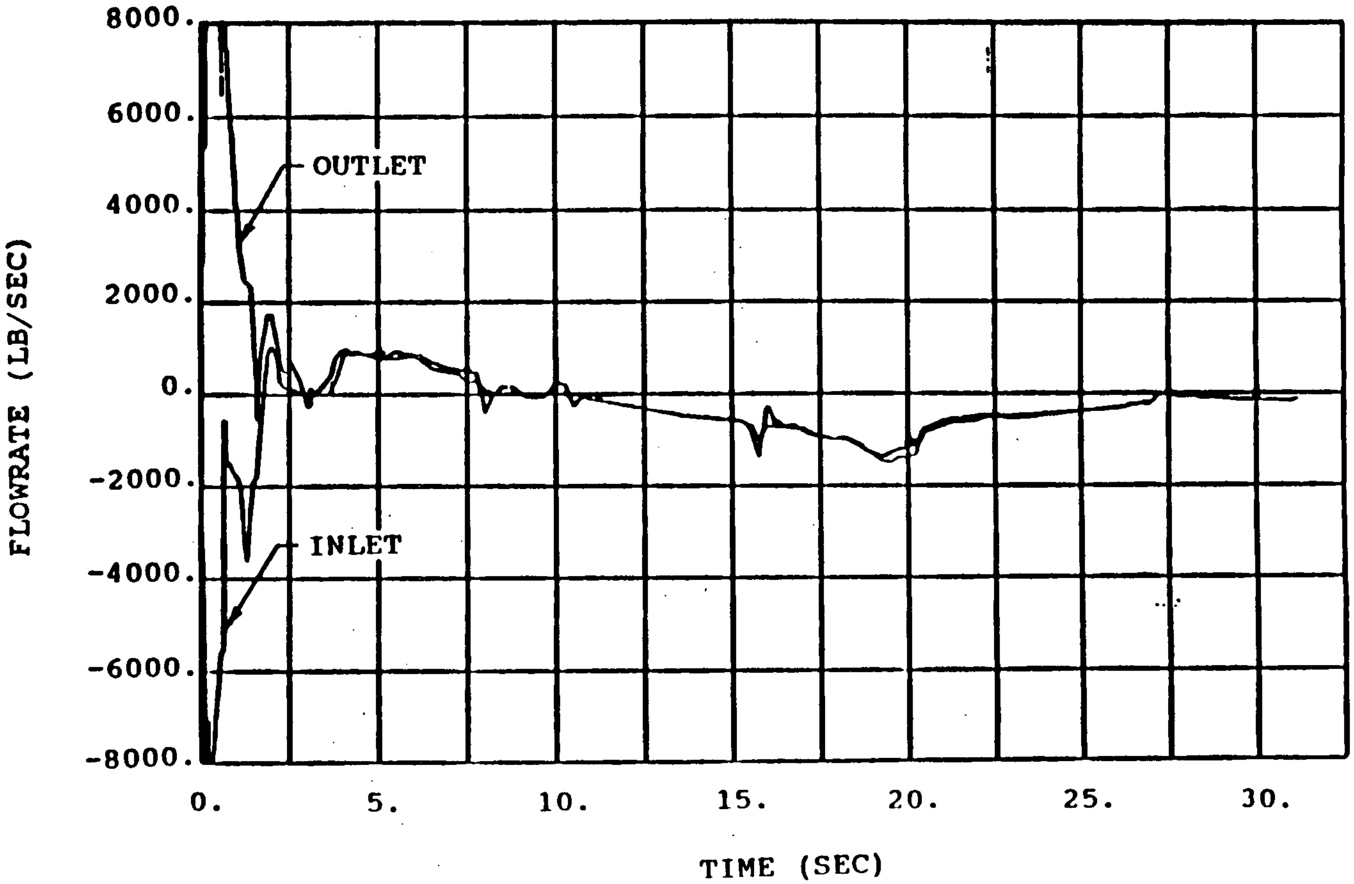
FLUID TEMPERATURE
DECLG (C_D = 0.4)

FIGURE
15.6.5-8



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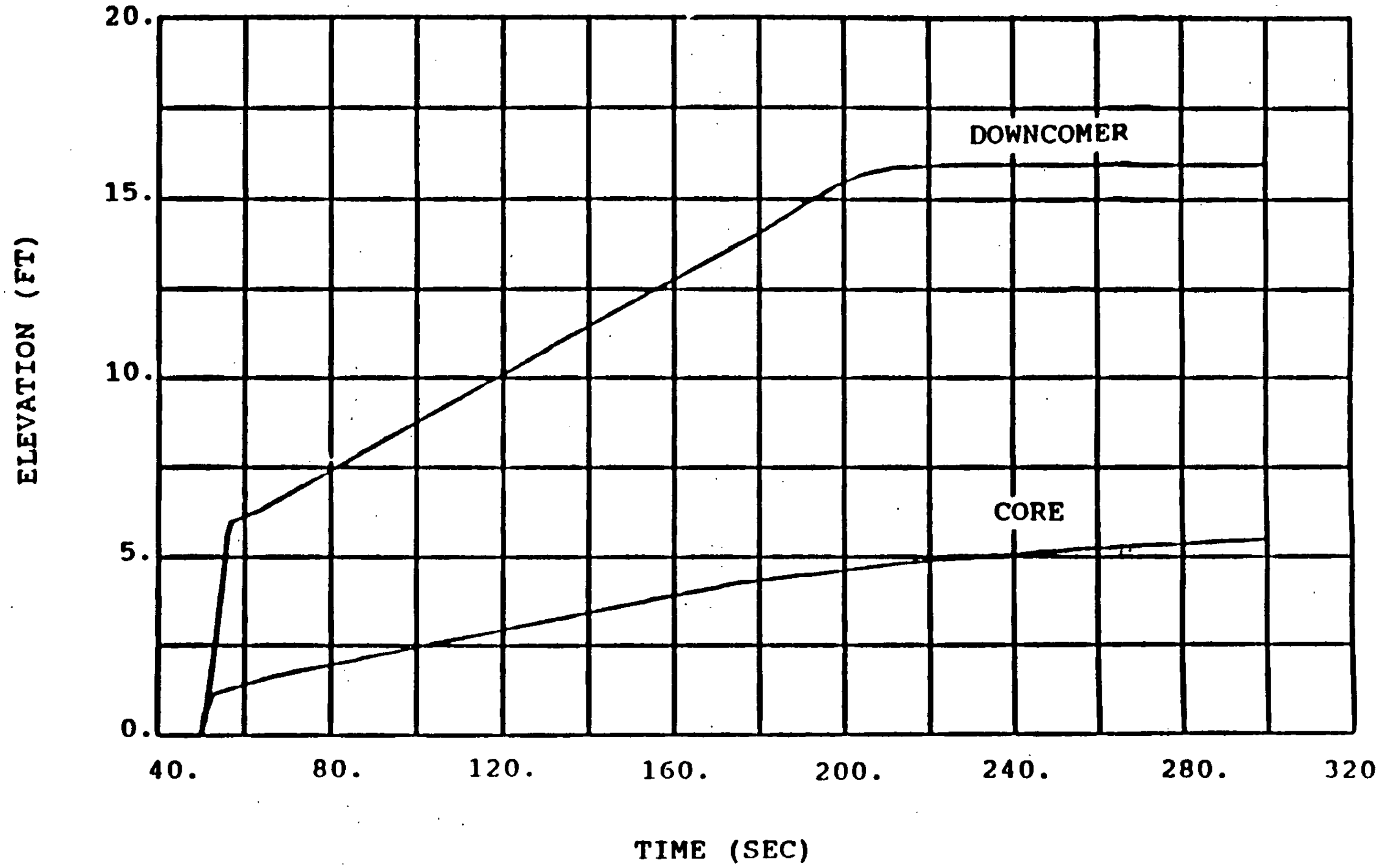
CORE FLOW TOP AND BOTTOM
DECLG (C_D=0.4)

FIGURE
15.6.5-9



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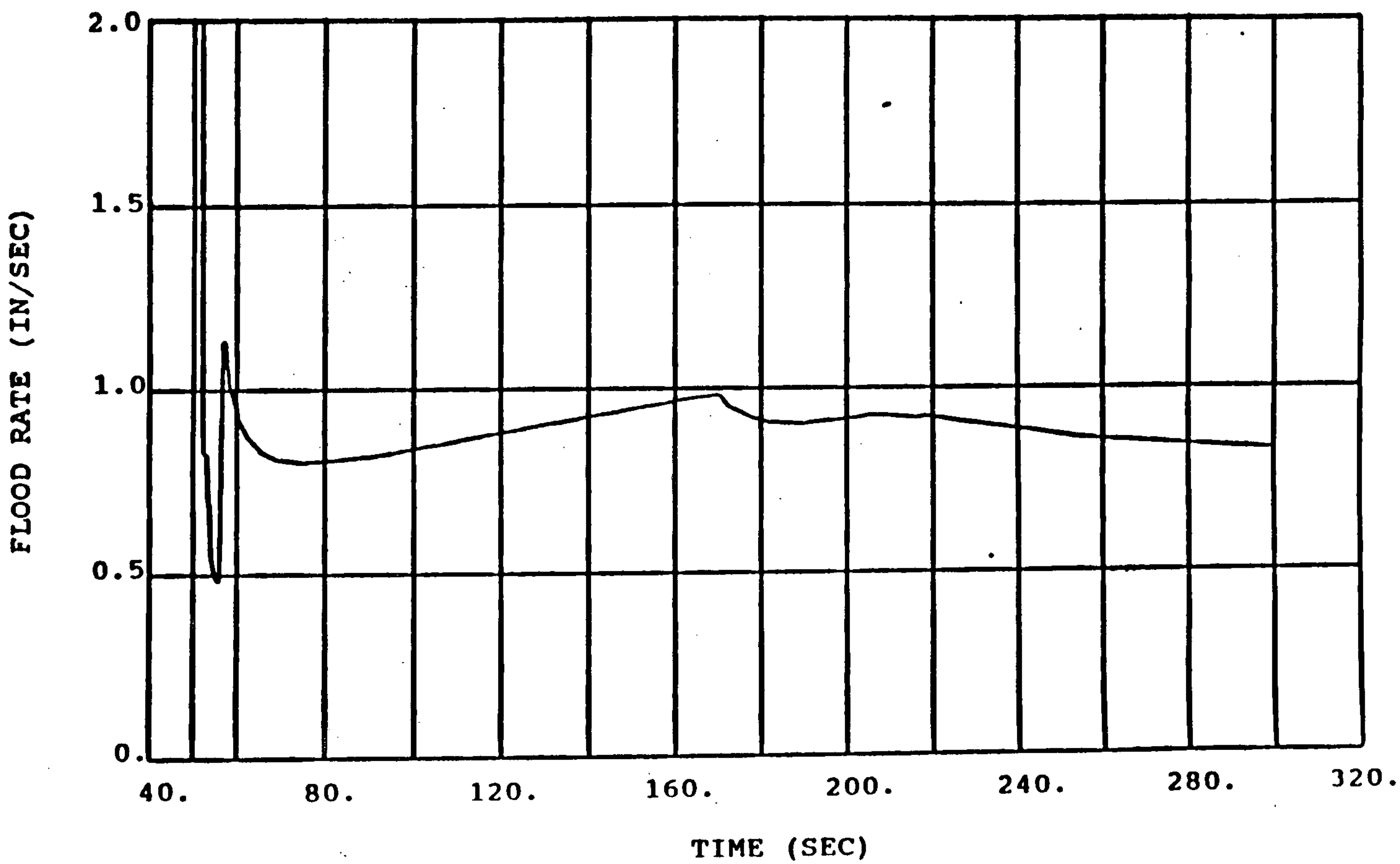
REFLOOD TRANSIENT
DECLG (C_D=0.4)
DOWNCOMBER AND CORE
WATER LEVELS

FIGURE
15.6.5-10



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REFLOOD TRANSIENT
DECLG (C_D = 0.4)
CORE INLET VELOCITY

FIGURE
15.6.5-11

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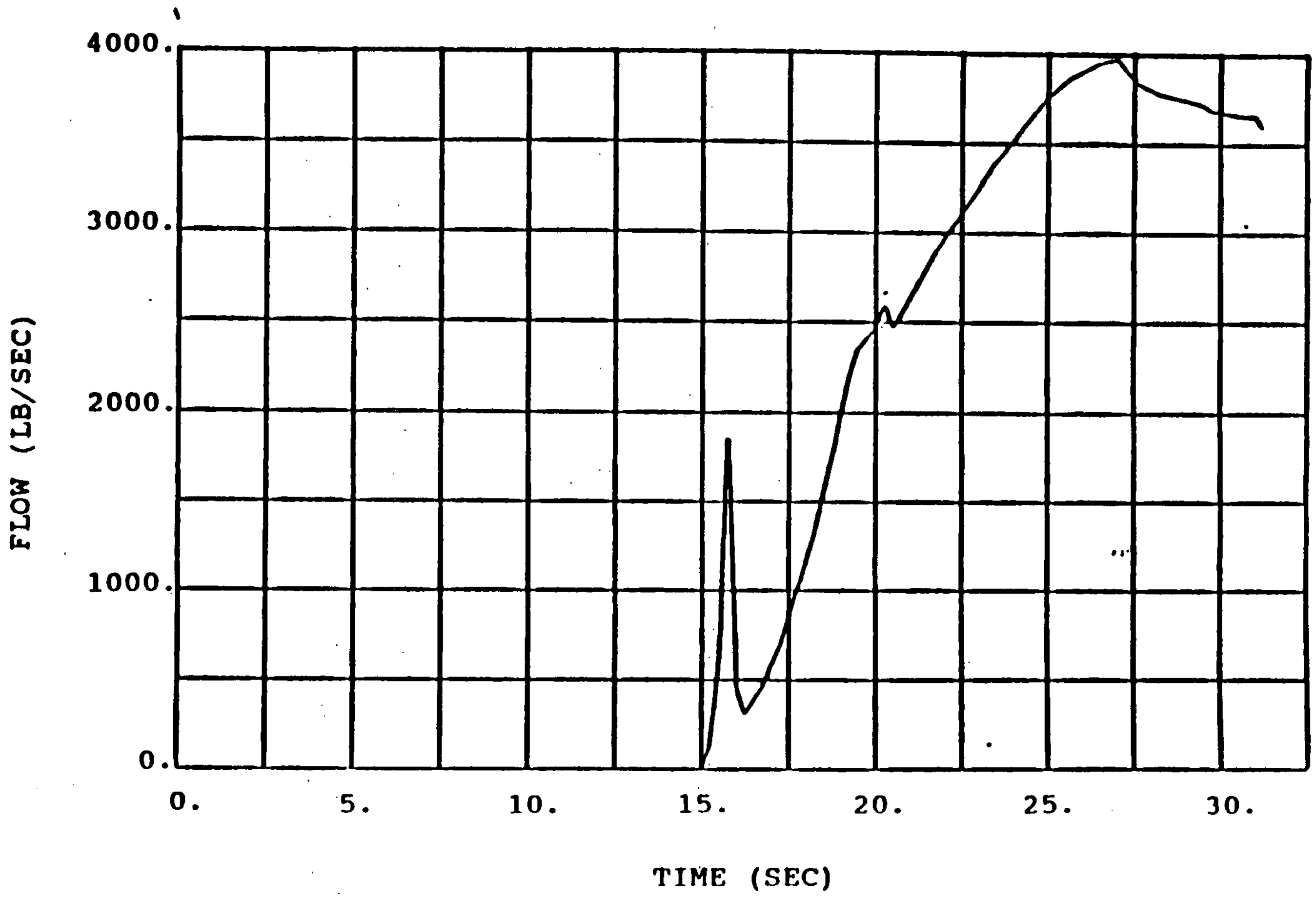

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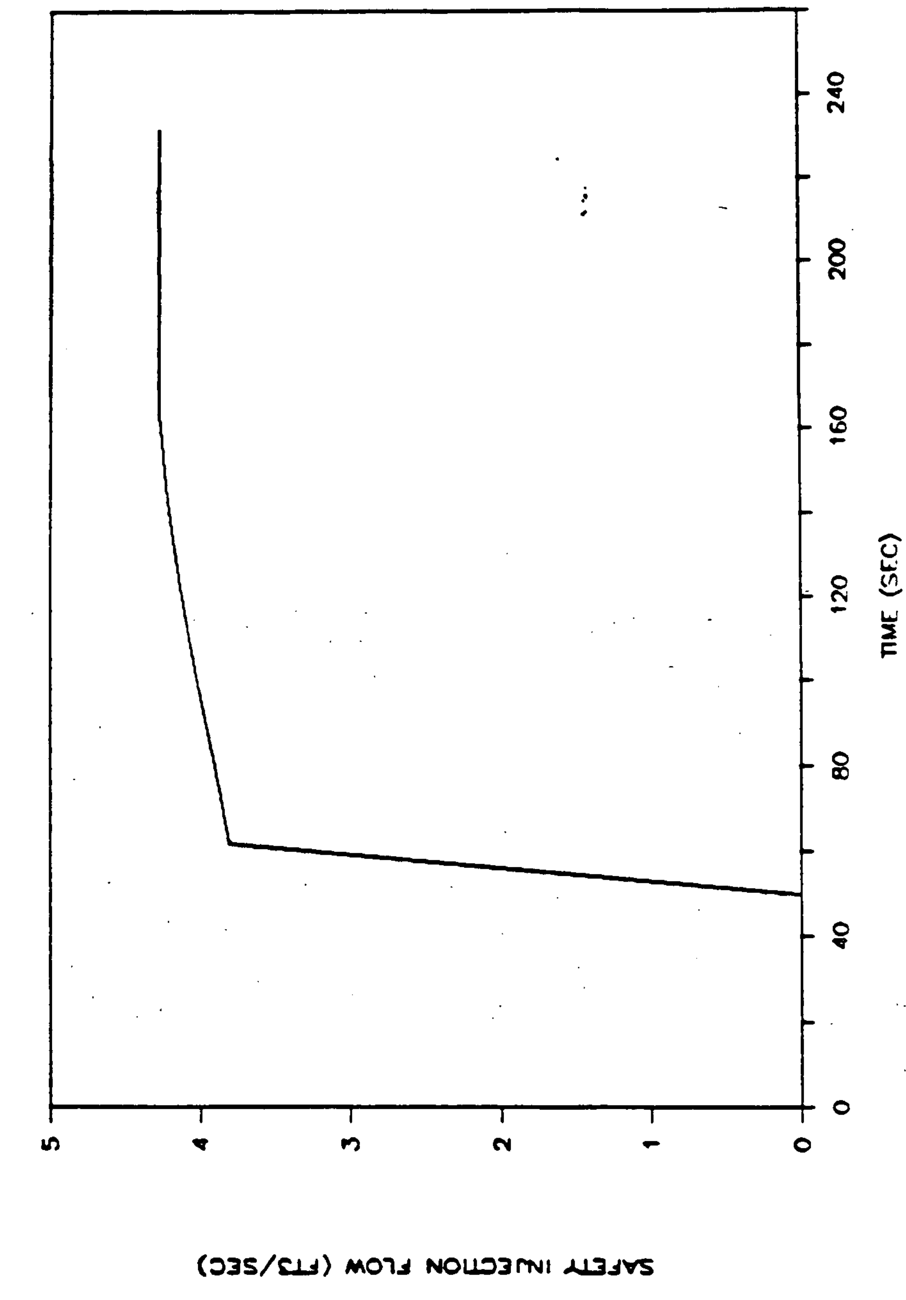
ACCUMULATOR FLOW
(BLOWDOWN)-DECLG (C_D = 0.4)

FIGURE
15.6.5-12



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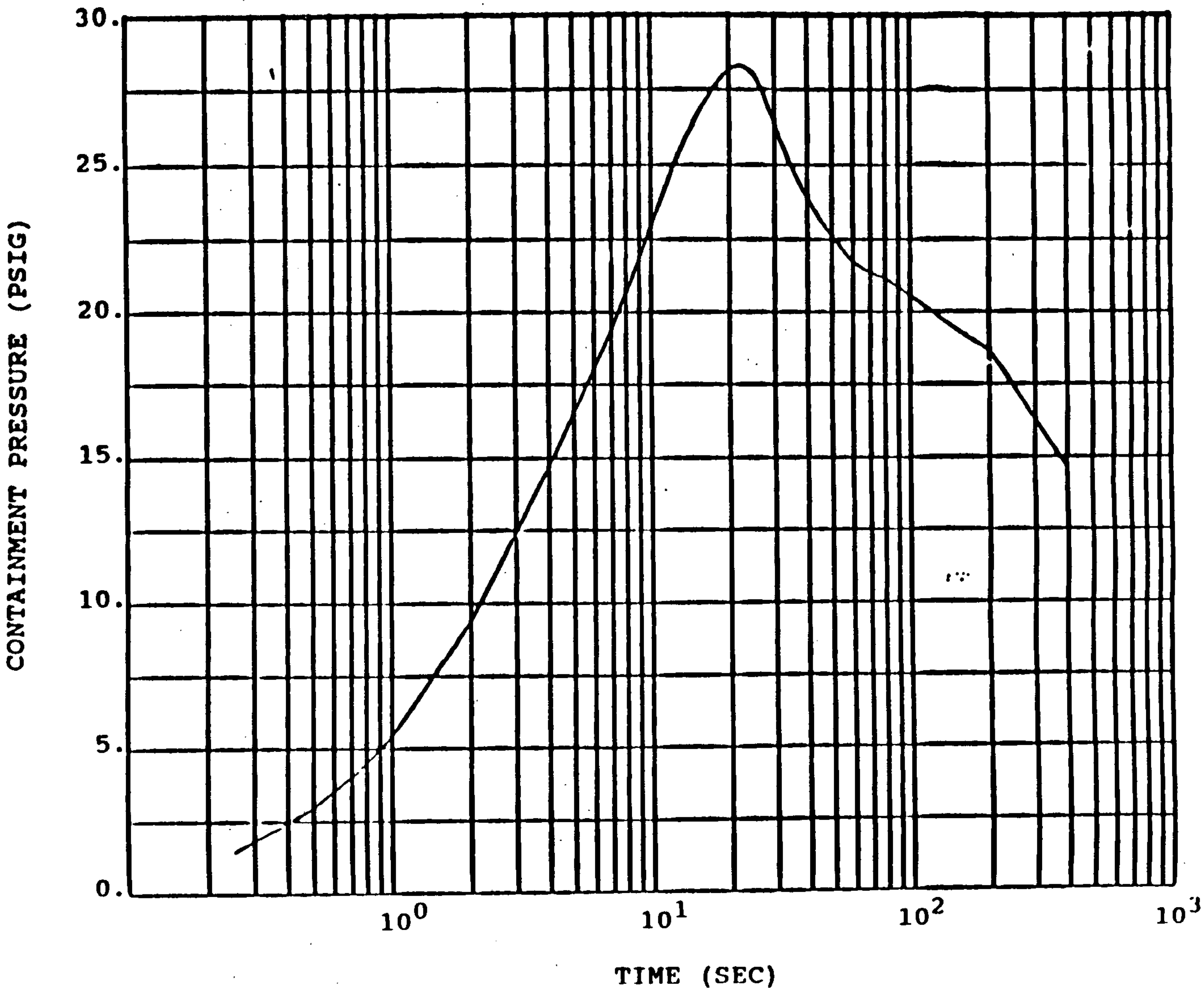
H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	PUMPED ECCS FLOW (REFLOOD) DECLG ($C_D = 0.4$)	FIGURE 15.6.5-13



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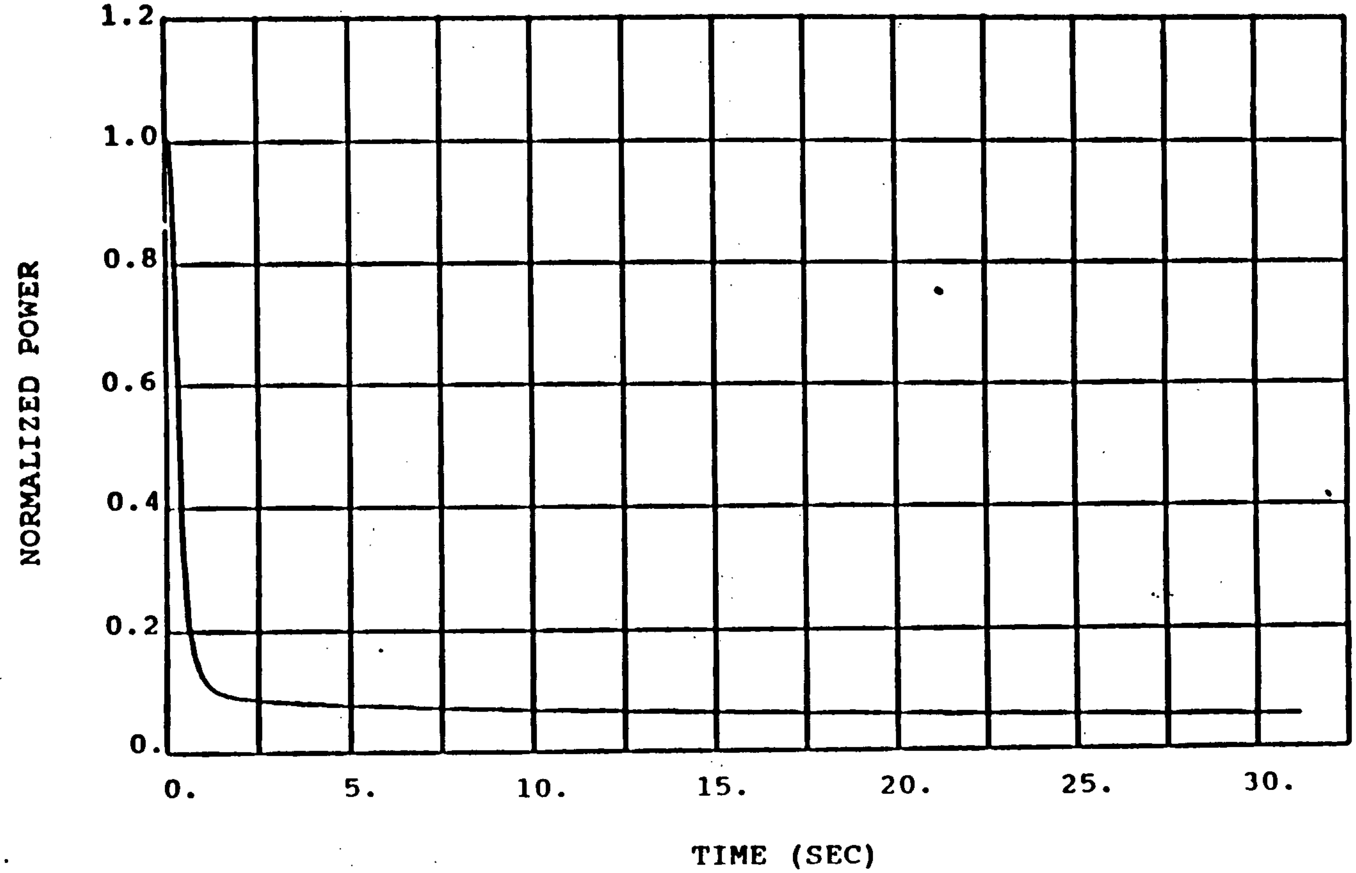
CONTAINMENT PRESSURE
DECLG (C_D=0.4)

FIGURE
15.6.5-14



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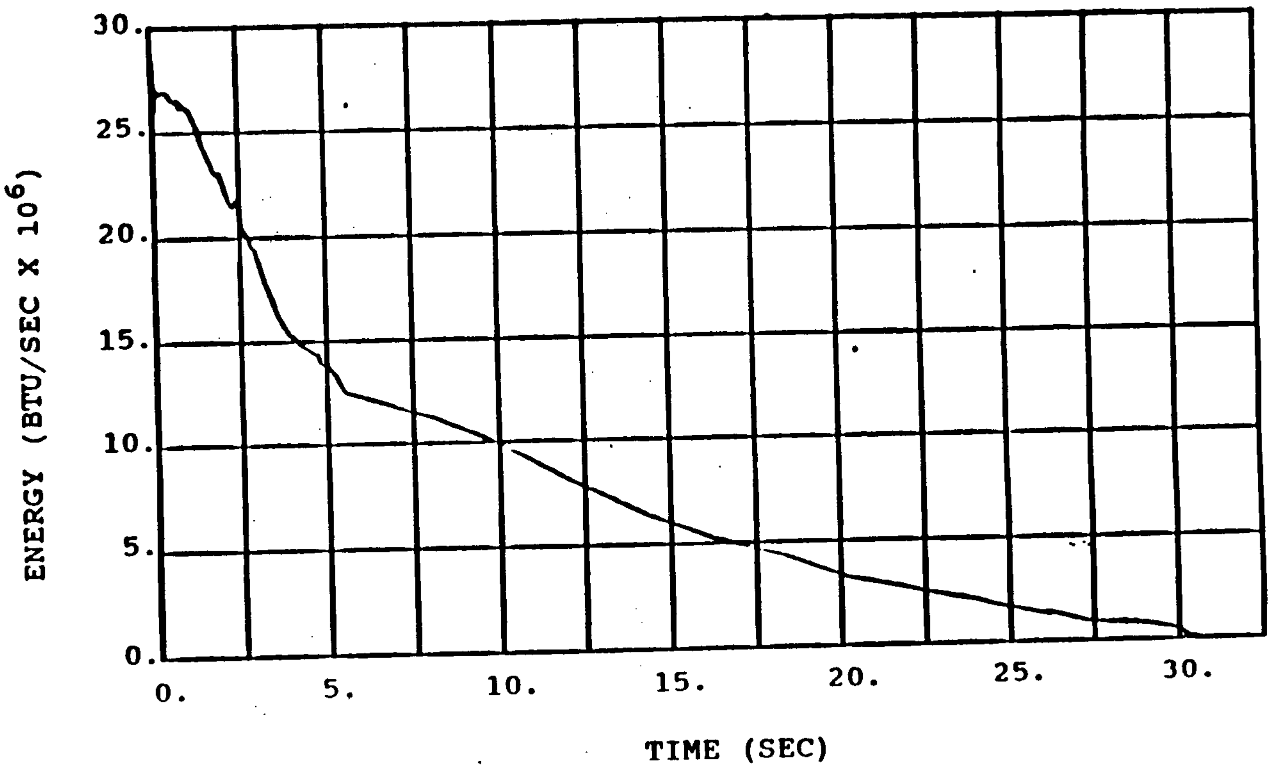
CORE POWER TRANSIENT
DECLG (C_D=0.4)

FIGURE
15.6.5-15



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BREAK ENERGY RELEASED
TO CONTAINMENT
DECLG (C_D = 0.4)

FIGURE
15.6.5-16

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 Evaluaiton Model

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

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

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

Cladding thermal analyses are performed with the LOCTA-IV (Reference 15.6.2-4) code which 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 injection flow from the SI pumps based on H. B. Robinson degraded delivery data. The degraded delivery data incorporates the standard FSAR ECCS assumption of minimum safeguards. For this analysis, the SI flow assumed for the first 1800 seconds of the transient were 50% of the flows presented in Figure 15.6.2-12. At 1800 seconds the flow increases to those presented in the figure. 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 injeciton 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 are performed using the 102% licensed core power assumption and incorporating Exxon 15x15 fuel data. This fuel data is summarized in Table 15.6.2-2.

Small Break LOCA Results

As noted previously, the calculated peak cladding temperature resulting from a small break LOCA is less than that calculated for a large break. A range of small break analyses is presented which establishes the limiting break size. The results of these analyses are summarized in Tables 15.6.2-3 and 15.6.2-4. Figures 15.6.2-1 through 15.6.2-5 present the principal parameters of interest for the small break ECCS analyses. For the 2-inch and 3-inch break sizes analyzed, the following transient parameters are included:

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

As indicated in the results for clad heat up, the 2-inch case is bounded by the 3-inch PCT. For the limiting break size analyzed (3-inch), the following additional transient parameters are presented (Figures 15.6.2-6 through 15.6.2-8a):

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

The maximum calculated peak cladding termperature for the small breaks analyzed is 1772°F. These results are well below all Acceptance Criteria limits of 10 CFR 50.46 and no case is limiting when compared to the results presented for large breaks.

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)	13.197 kW/ft
Total Peaking Factor, F	2.32
Power Shape	Fig. 15.6.2-13
Fuel Assembly Array	Exxon 15x15
Nominal Accumulator Water Volume	825 ft /accum.
Nominal Accumulator Tank Volume	1200 ft /accum.
Minimum Accumulator Gas Pressure	615 psia
Pumped Safety Injection Flow[2]	Fig. 15.6.2-12
Steam Generator Initial Pressure	787 psia
Auxiliary Feedwater Flow	41.22 lb/sec/SG
Steam Generator Tube Plugging Level	5%

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

2 - As noted in the text, 50% of this flow was assumed for the first 1800 seconds of the transient

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	2 in (sec)	3 in (sec)
Start	0.0	0.0
Reactor Trip	12.99	5.79
S-signal	21.70	9.85
Loop Seal Venting	1009.1	450.3
Top of Core Uncovered	1683.5	798.2
Accumulator Injection	N/A	1099.6
Maximum Core Uncovery	2114.3	1182.1
Peak Clad Temperature Occurs	2491.9	1229.9
Top of Core Covered	3709.8	2231.2

TABLE 15.6.2-4

Small Break LOCA Fuel Cladding Results

Results	--1 HHSI Pump--- 2 Inch*	3 Inch*
Peak clad temperature (°F)	1409.1	1771.6
Peak clad temperature location (ft)	12.0	12.0
Local Zr/H_2O reaction, maximum (%)	0.44	2.31
Local Zr/H_2O location (ft)	12.0	12.0
Total Zr/H_2O reaction (%)	<0.3	<0.3
Hot rod burst time (sec)	N/A	N/A
Hot rod burst location (ft)	N/A	N/A

* - 2 HHSI Pumps at 1800 seconds

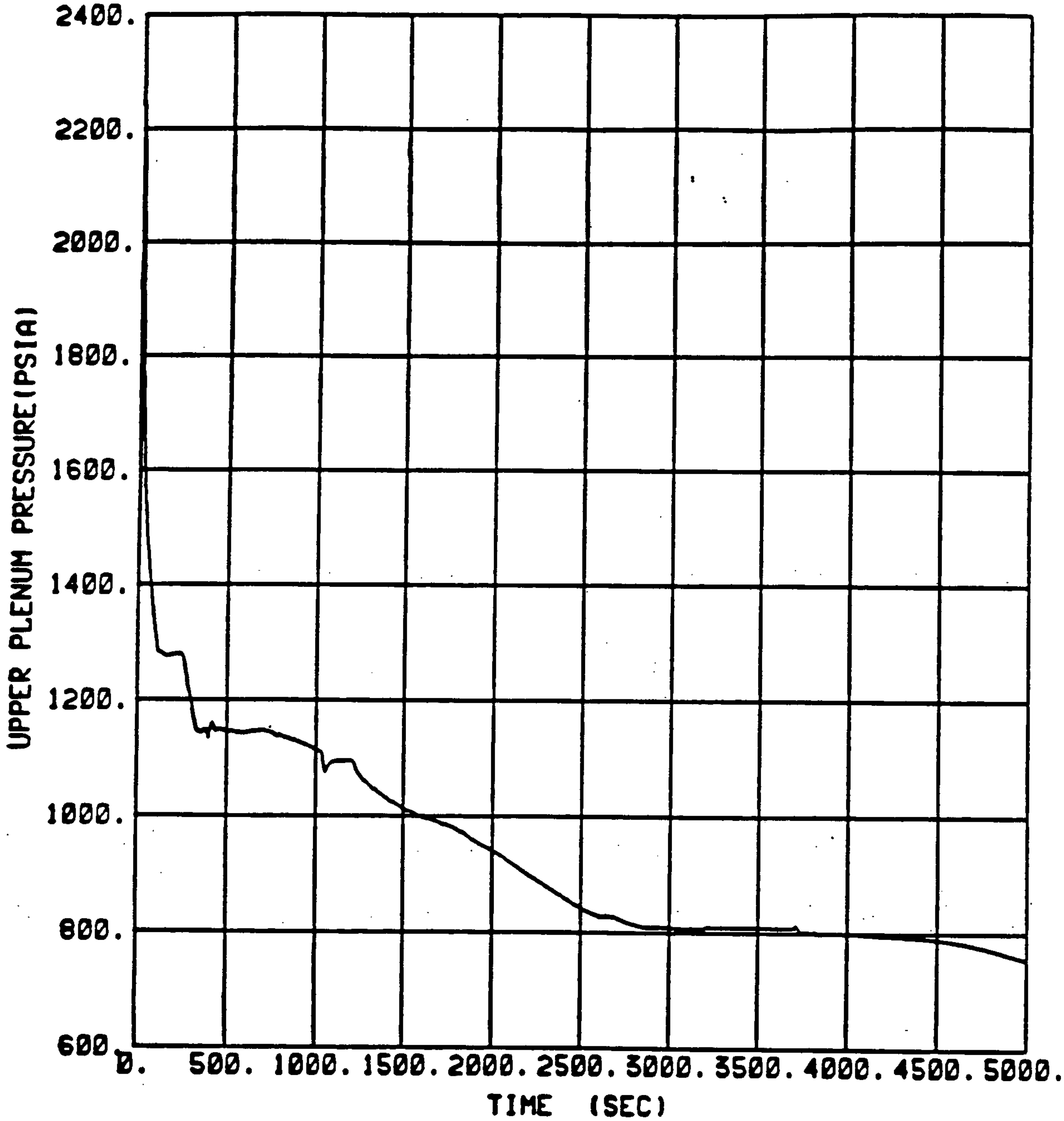
REFERENCES FOR SECTION 15.6.2

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.

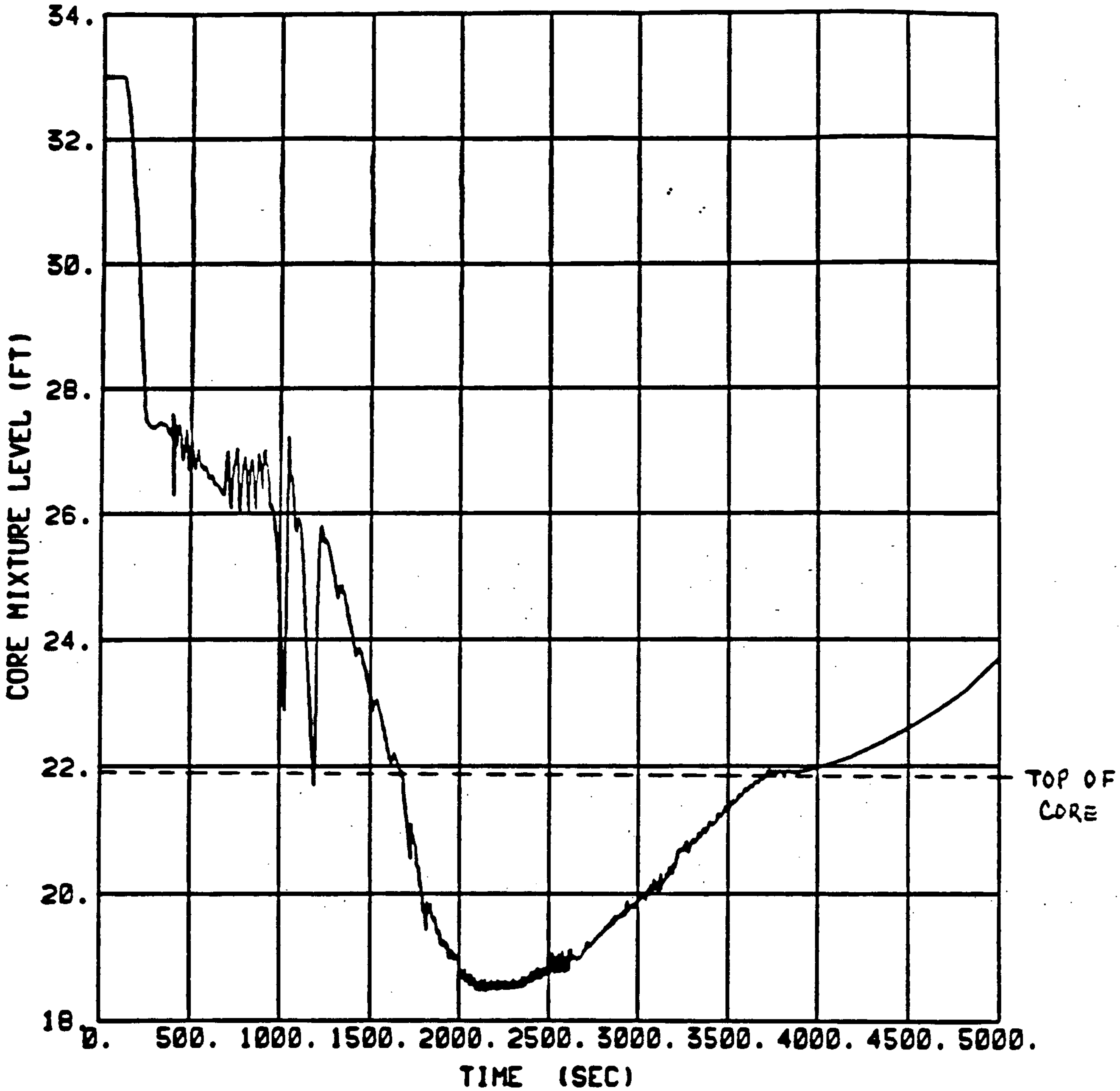
4. Bordelon, F. M., et. al., "LOCTA-IV Program: Loss of Coolant Transient Analysis", WCAP-8301, (Proprietary) and WCAP-8305, (Non-Proprietary), June 1974.



H. B. ROBINSON UNIT 2

UPPER PLENUM PRESSURE
2-INCH COLD LEG BREAK

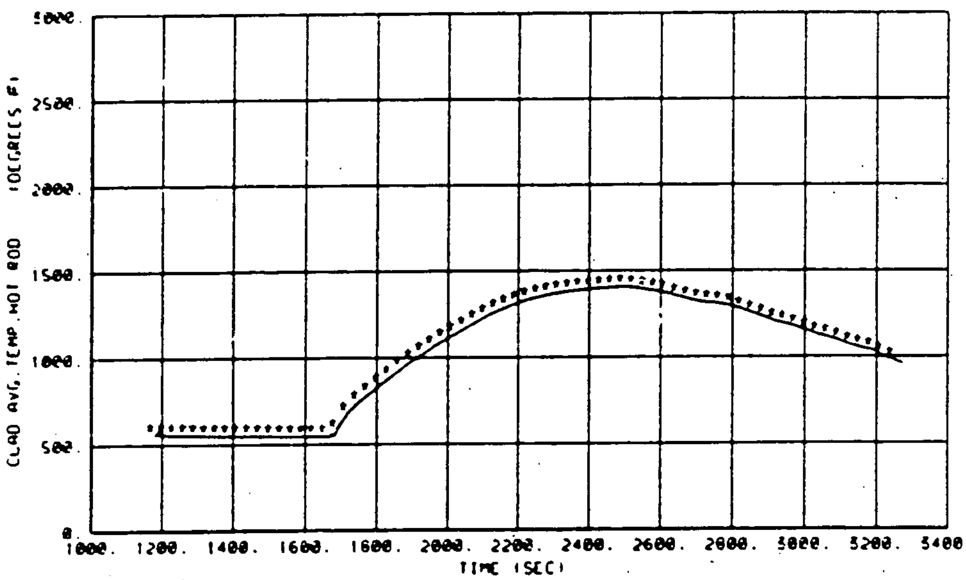
FIGURE 15.6.2-1



H. B. ROBINSON UNIT 2

CORE MIXTURE LEVEL
2-INCH COLD LEG BREAK

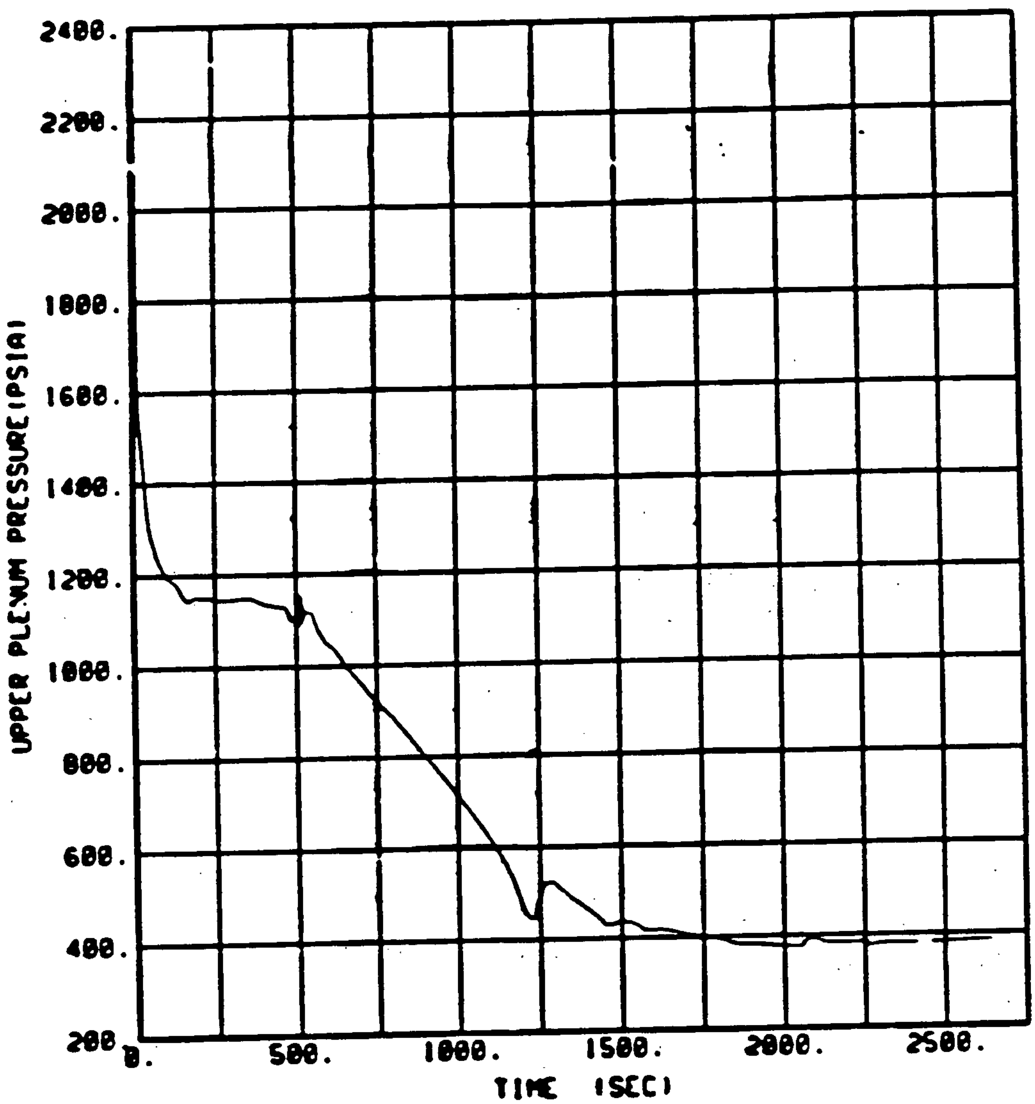
FIGURE 15.6.2-2



H. B. ROBINSON UNIT 2

HOT SPOT CLAD TEMPERATURE
2-INCH COLD LEG BREAK

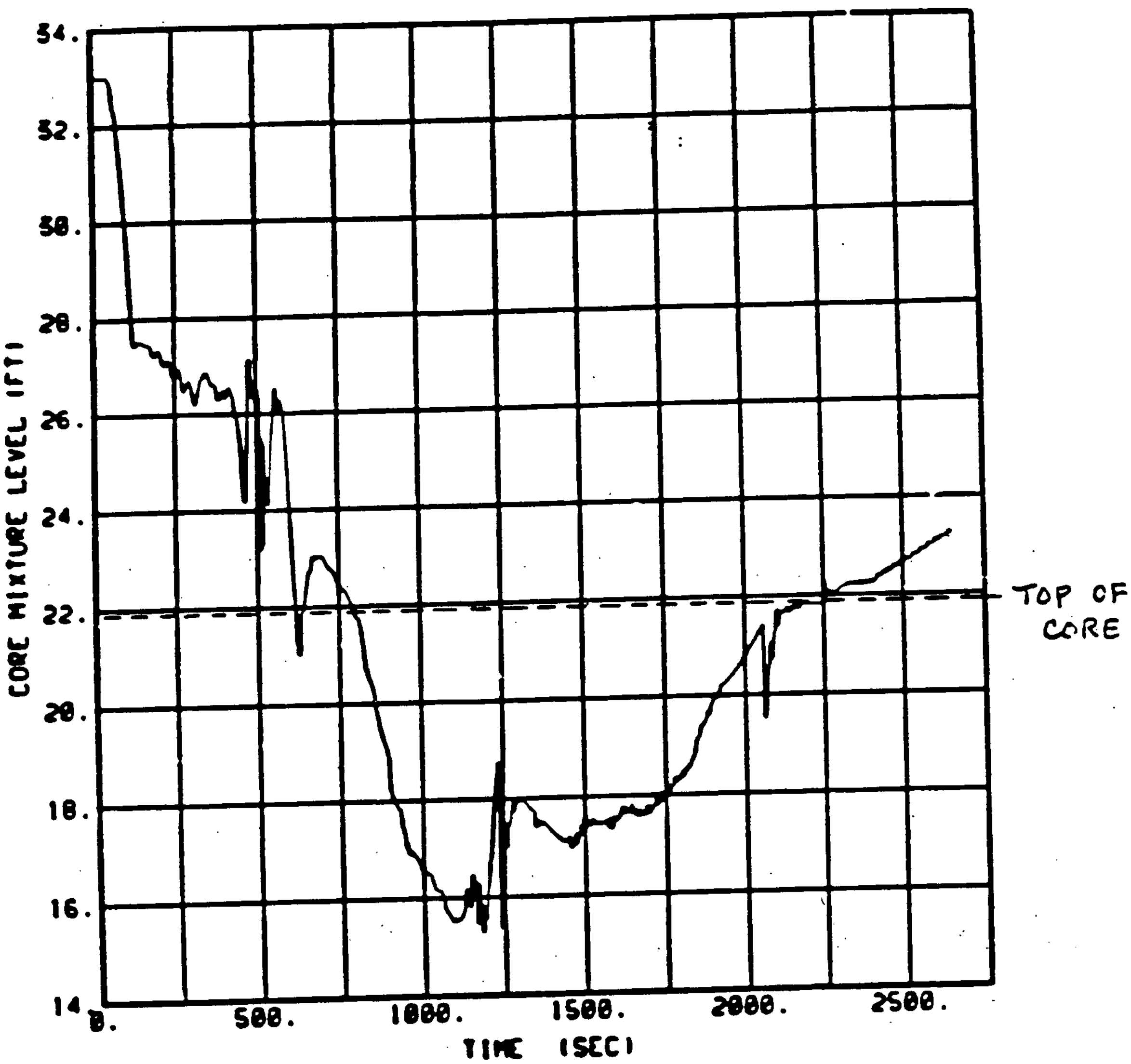
FIGURE 15.6.2-2a



H. B. ROBINSON UNIT 2

UPPER PLENUM PRESSURE
3-INCH COLD LEG BREAK

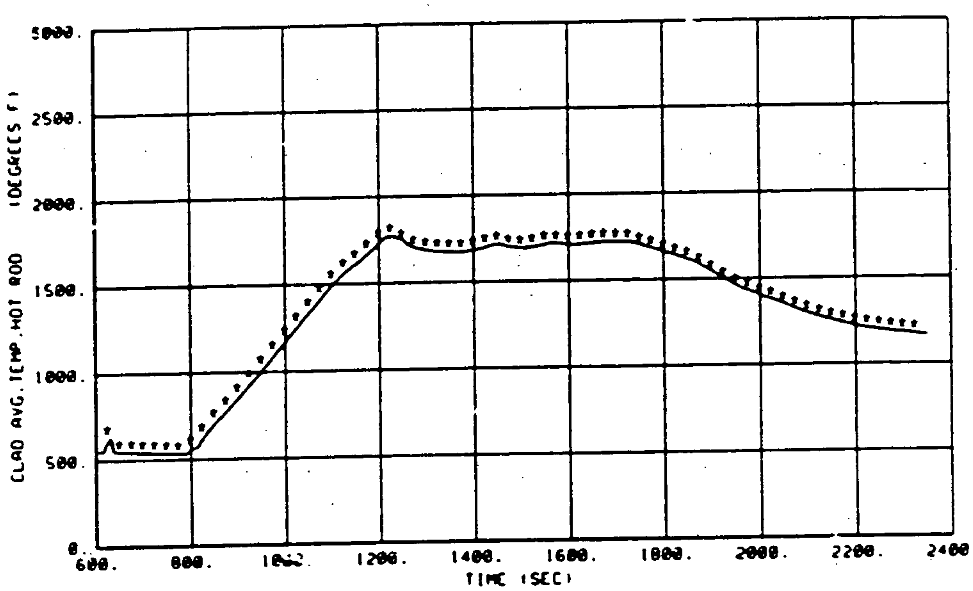
FIGURE 15.6.2-3



H. B. ROBINSON UNIT 2

CORE MIXTURE LEVEL
3-INCH COLD LEG BREAK

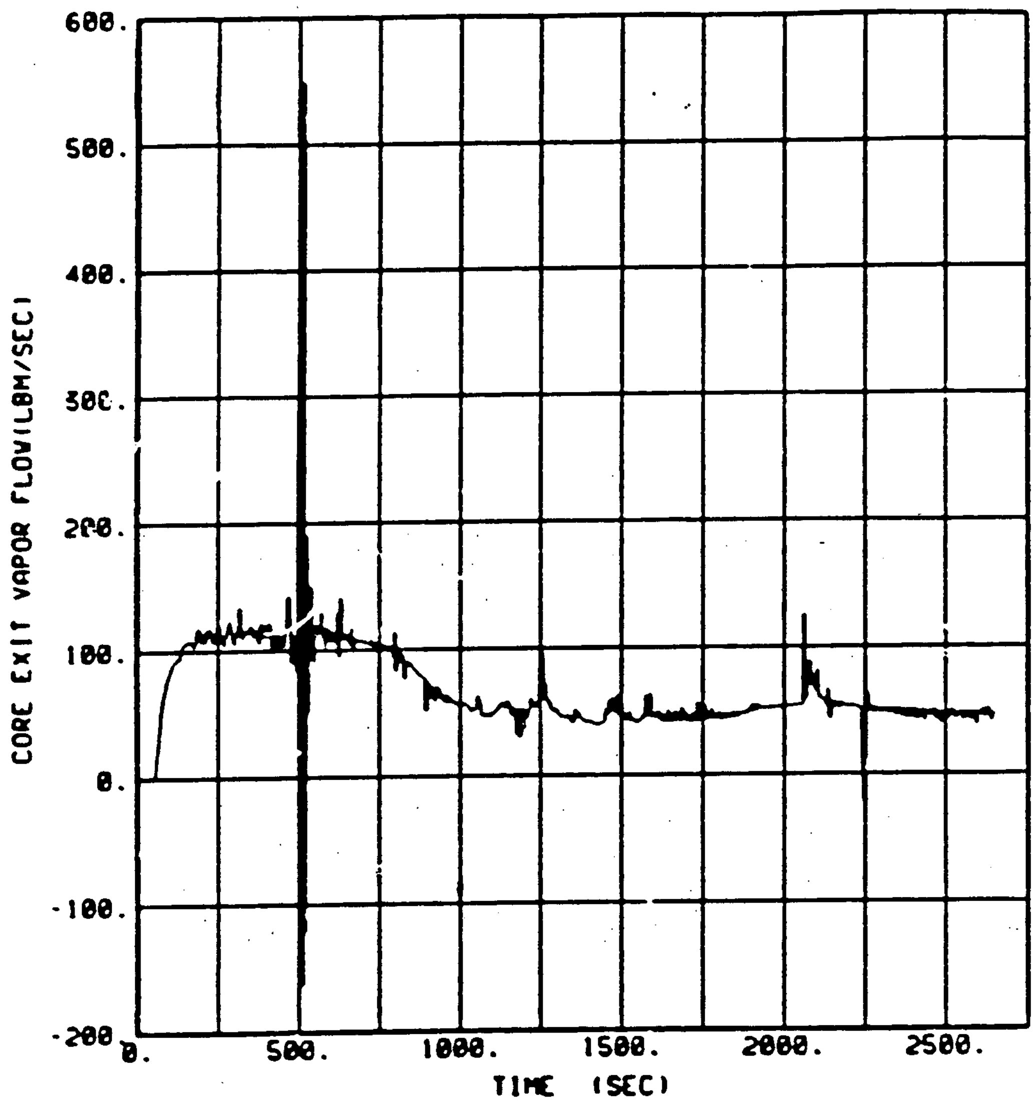
FIGURE 15.6.2-4



H. B. ROBINSON UNIT 2

HOT SPOT CLAD TEMPERATURE
3-INCH COLD LEG BREAK

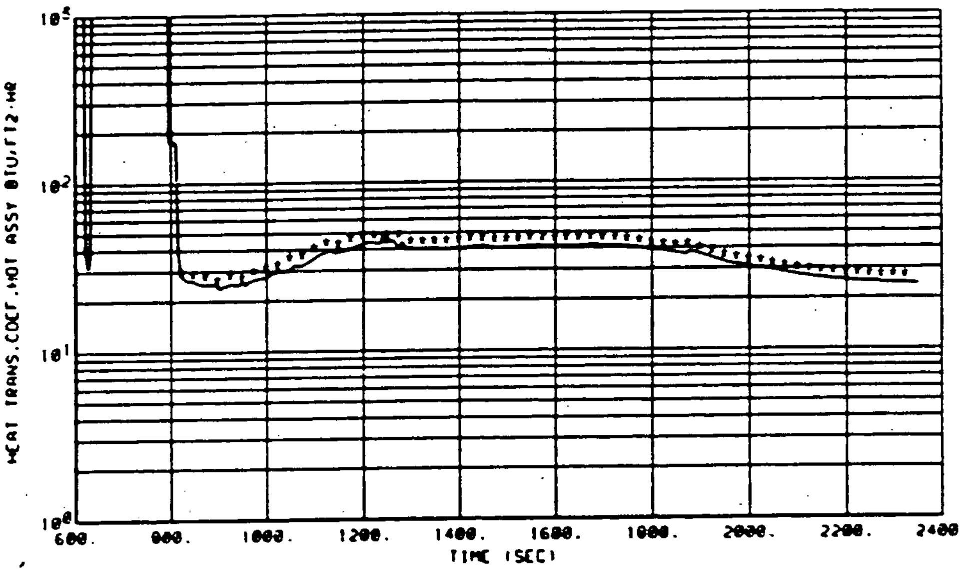
FIGURE 15.6.2-5



H. B. ROBINSON UNIT 2

CORE EXIT VAPOR FLOW
3-INCH COLD LEG BREAK

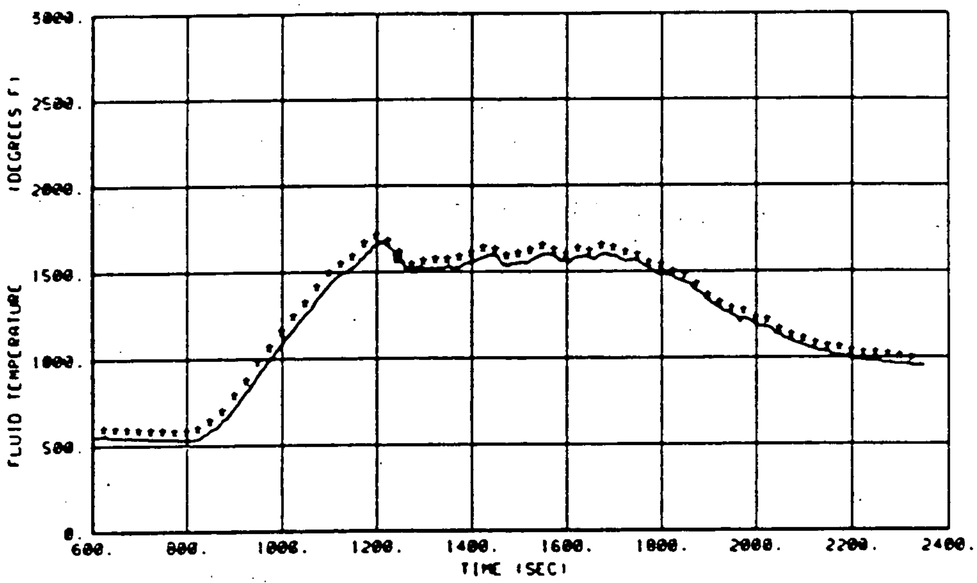
FIGURE 15.6.2-6



H. B. ROBINSON UNIT 2

CORE HEAT TRANSFER COEFFICIENT
3-INCH COLD LEG BREAK

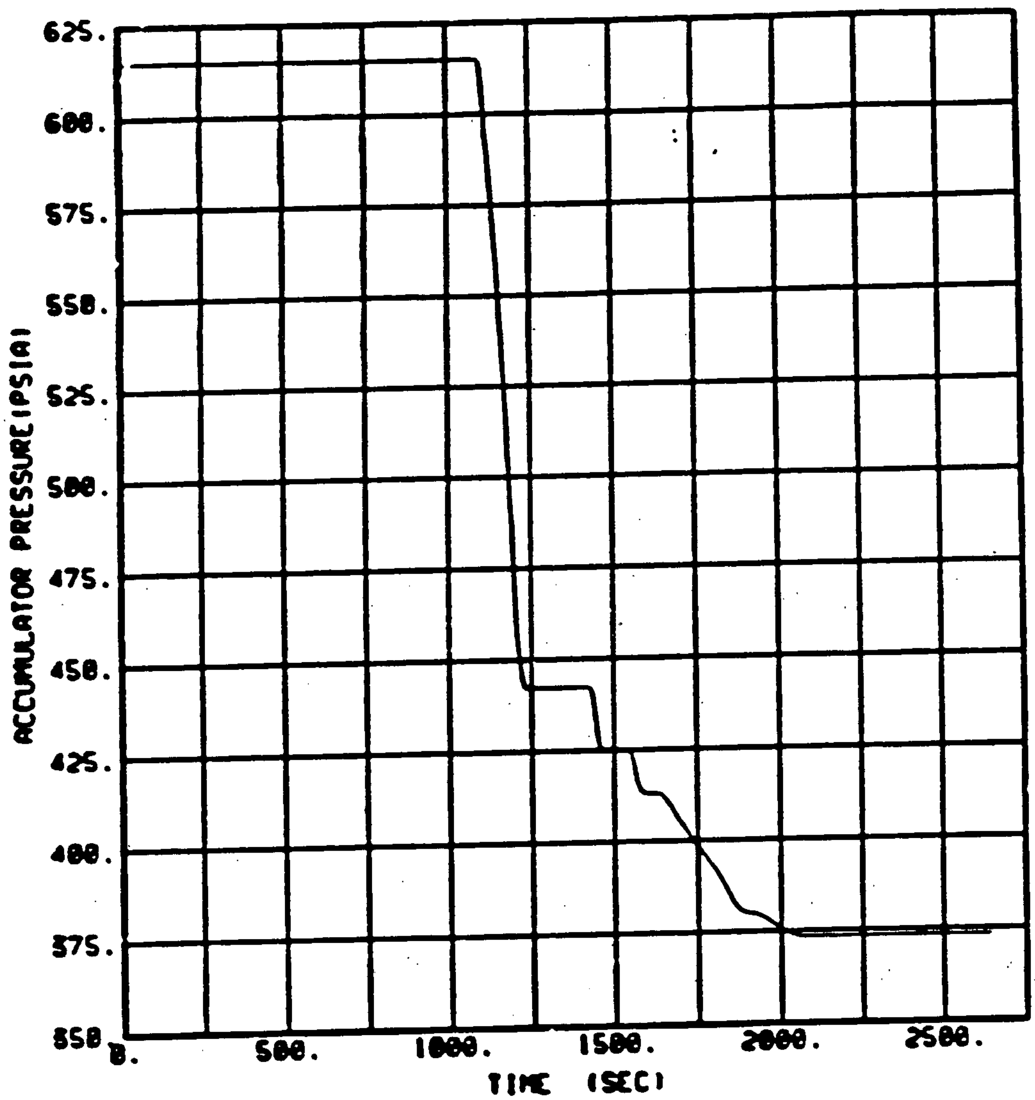
FIGURE 15.6.2-7



H. B. ROBINSON UNIT 2

HOT SPOT FLUID TEMPERATURE
3-INCH COLD LEG BREAK

FIGURE 15.6.2-8



H. B. ROBINSON UNIT 2

ACCUMULATOR PRESSURE
3-INCH COLD LEG BREAK

FIGURE 15.6.2-8a

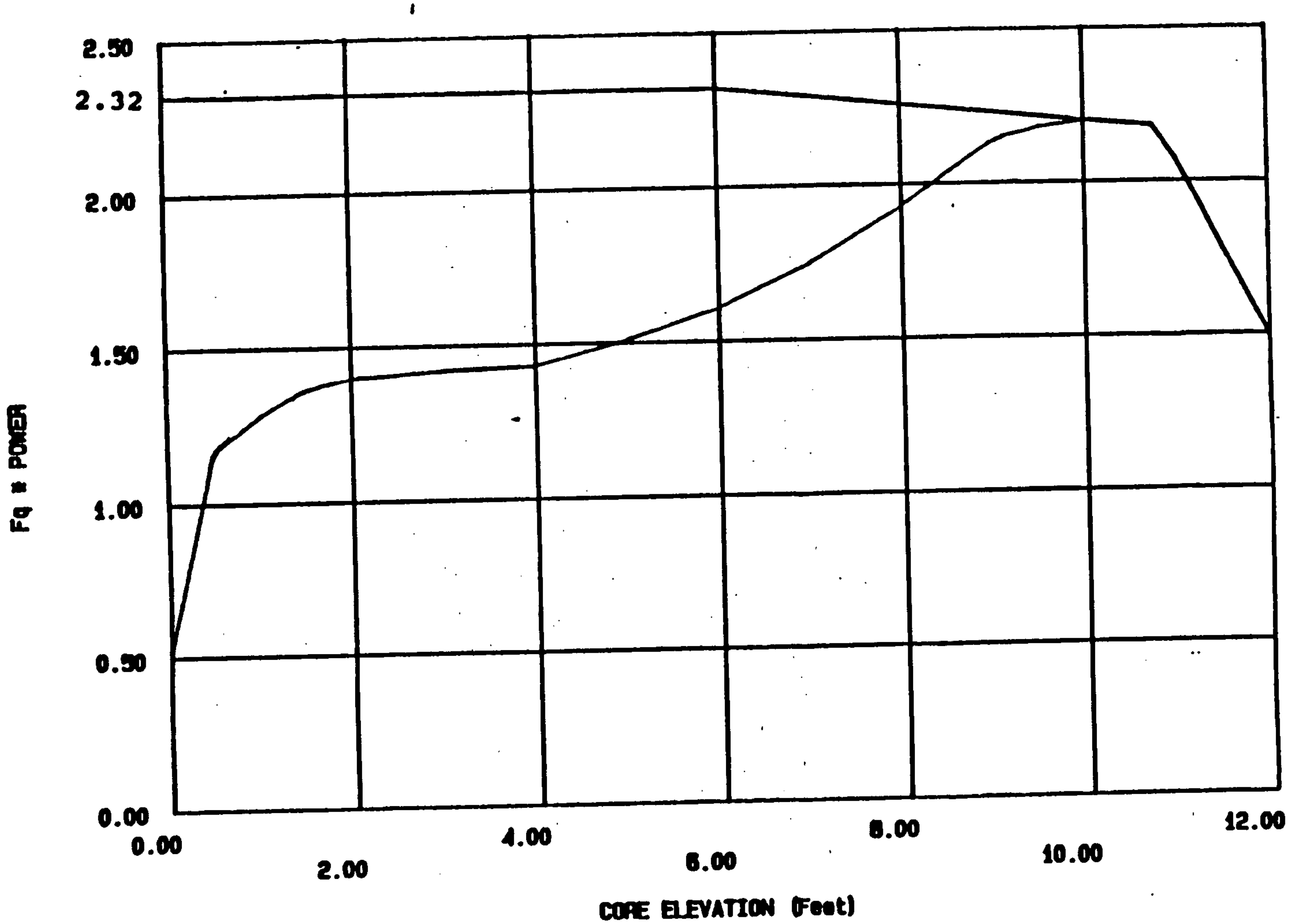


Figure 15.6.2-13 H. B. Robinson Small Break LOCA Power Shape

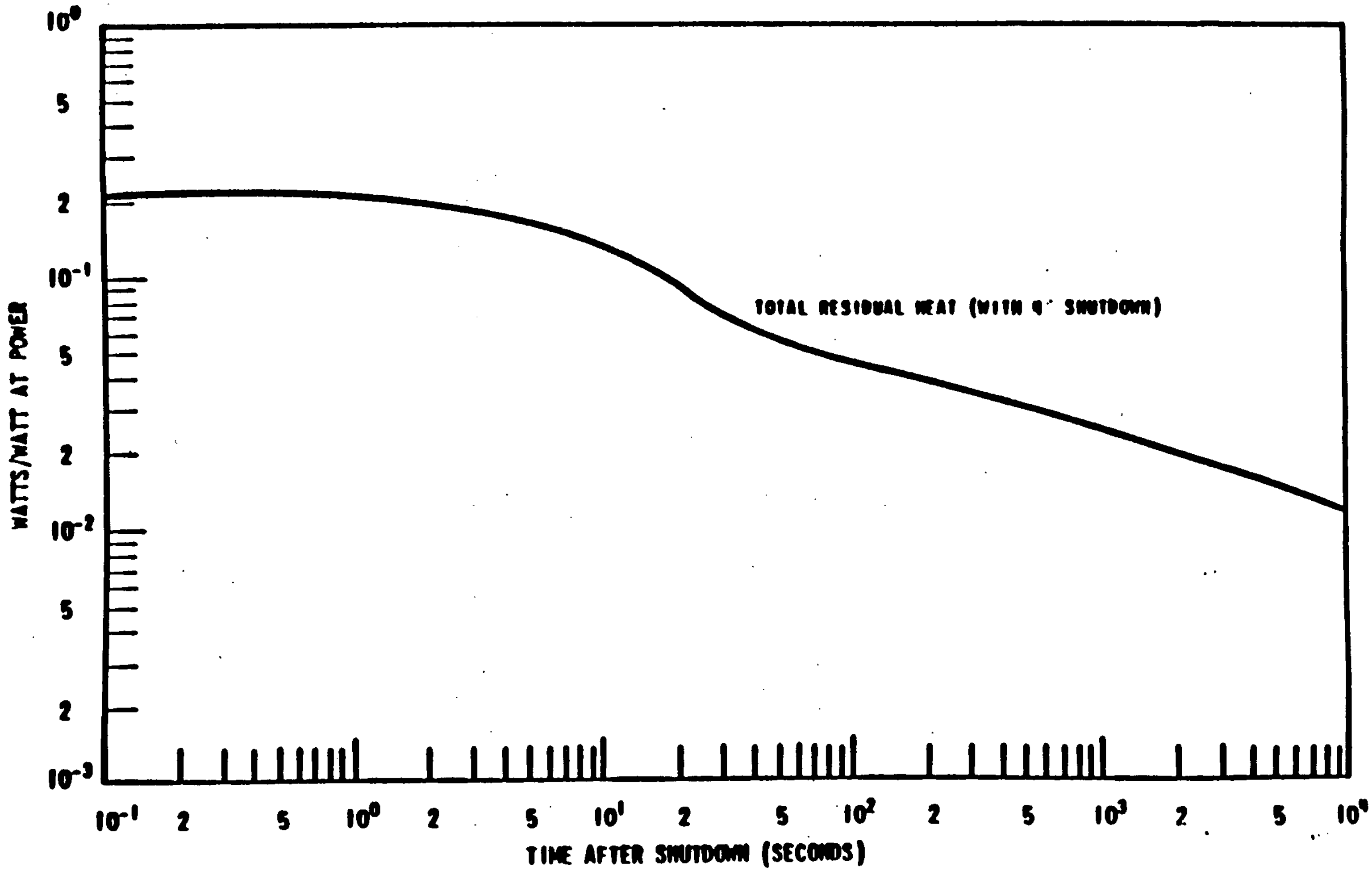


Figure 15.6.2-14 Core Power After Reactor Trips