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MIRAGLIA, F. J. Division of Pressurized Water Reactor Licensing - B (post 8

SUBJECT: Forwards results of C-E large break LOCA EECS performance analysis which demonstrates compliance w/10CFR50.46 criteria, per 850930 commitment.

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FLORIDA POWER & LIGHT COMPANY

JAN 3 1 1986

L-86-37

Office of Nuclear Reactor Regulation
Attention: Mr. Frank J. Miraglia, Director
Division of PWR Licensing - B
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Miraglia:

Re: St. Lucie Unit No. 2
Docket No. 50-389
C-E Large Break LOCA Analysis

Florida Power and Light Company Letter L-85-373, dated September 30, 1985, stated that Combustion Engineering was redoing the St. Lucie Unit 2 Large Break LOCA Analysis, and that the results were expected to be available by January 31, 1986. Attached are these results which demonstrate compliance with 10 CFR 50.46 criteria.

Very truly yours,

C. O. Woody
Group Vice President
Nuclear Energy

COW/RJS/gp

Attachment

A001 1/1

Large Break LOCA ECCS Performance

Introduction and Summary

On July 3, 1985, Florida Power & Light Company (FPL) received notification from Combustion Engineering (C-E) of a potential non-conservatism in one element of C-E's large break loss-of-coolant accident (LOCA) evaluation model. That notification was received by copy of a letter to NRC from C-E (A. E. Scherer to G. W. Knighton, LD-85-031, July 2, 1985). The particular element in question was the treatment of axial power distribution and peaking factor.

In response to NRC concerns raised relating to the selection of the power distribution used in the C-E large break ECCS analysis, a sensitivity study was performed to determine the limiting power distribution for use in the large break ECCS analysis of C-E NSSS's with the currently approved C-E Evaluation Model. The results of analyses performed determined the most adverse axial shape to be a 1.55 peak located at 65% of core height (Reference 12).

With respect to the Cycle 2 license (Reference 2) the limiting large break LOCA was reanalyzed with the limiting axial power shape (Reference 12), 1250 plugged tubes per steam generator, fuel parameters which bound current and expected conditions, and a lower augmentation penalty to demonstrate compliance with 10CFR50.46 which presents the NRC Acceptance Criteria for Emergency Core Cooling Systems for Light Water cooled reactors (Reference 1). The analysis justifies an allowable peak linear heat generation rate (PLHGR) of 13.0 kw/ft. This PLHGR is equal to the existing limit for St. Lucie Unit 2. The method of analysis and detailed results which support this value are presented herein.

Method of Analysis

The method of analysis is based upon the NRC approved C-E large break LOCA

ECCS evaluation model which is described in References 3 through 8. Blowdown, and refill/reflood hydraulics, and hot rod temperature calculations were performed with the fuel parameters which bound the current fuel cycle and expected conditions for future cycles at a reactor power level of 2754 Mwt. The blowdown hydraulic calculations were performed with the CEFLASH-4A (Reference 5) code while the refill/reflood hydraulic calculations were performed with the COMPERC-II (Reference 6) code. The hot rod clad temperature and clad oxidation calculations were performed with the STRIKIN-II (Reference 7) and PARCH (Reference 8) codes. Fuel performance calculations were performed using the FATES-3A Version of the C-E's NRC approved fuel performance code (References 9 and 10) with the fuel grain size restriction as approved by the NRC (Reference 11).

Results

Table 1 presents the analysis results for the 1.0 DEG/PD* break which produces the highest peak clad temperature. The results of the evaluation confirm that 13.0 kw/ft is an acceptable value for the PLHGR in the present analysis. The peak clad temperature and maximum local and core wide clad oxidation values, as shown in Table 1, are below the 10CFR50.46 acceptance limits of 2200°F, 17% and 1% respectively.

Table 2 presents a list of the significant parameters displayed graphically for the limiting 1.0 DEG/PD break. The significant parameters and initial conditions for this analysis are shown in Table 3.

This analysis accounts for steam generator U-tube plugging of 1250 average length tubes per steam generator. Steam generator U-tube plugging increases system resistance to flow and hence the ability of the Reactor Coolant System (RCS) to vent steam during reflood. The analysis also accounts for the decreased heat transfer area and primary side coolant volume caused by the tube plugging.

*DEG/PD = Double-Ended Guillotine at Pump Discharge

The reduction in the augmentation penalty results in an increase of the hot assembly average channel PLHGR. The hot assembly average channel PLHGR influences the radiation heat transfer between the hot rod of the hot assembly and the average rod of the hot assembly. Higher power of the average rod of the hot assembly results in reduced heat transfer from the hot rod to its surrounding rods.

To determine the limiting power distribution for use in the large break ECCS analysis, a sensitivity study was performed and is documented in Reference 12. Based on this study the limiting power shape was used for the present analysis.

The 1.0 DEG/PD produced the highest peak clad temperature of 2106°F and a peak local oxidation of 16.12% compared to the acceptance criteria of 2200°F and 17%, respectively. The 1.0 DEG/PD also resulted in the highest core wide oxidation of less than 0.70% which is below the 1% acceptance criterion.

Conclusions

The results of the ECCS performance evaluation for the present analysis for St. Lucie Unit 2, demonstrated a peak clad temperature of 2106°F, a peak local clad oxidation percentage of 16.12%, and a peak core wide clad oxidation percentage of less than 0.70% compared to the acceptance criteria of 2200°F, 17%, and 1%, respectively. Therefore, operation of St. Lucie Unit 2 at a core power level of 2754 Mwth (102% of 2700 Mwth) and a PLHGR of 13.0 kw/ft is in conformance with 10CFR50.46.

REFERENCES

1. Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Reactors, Federal Register, Vol. 39, No. 3, Friday, January 4, 1974.

2. Letter from J. W. Williams, Jr., (FP&L), to E. G. Eisenhut (US NRC), St. Lucie Unit 2, Docket No. 50-389, "Proposed License Amendment Cycle 2 Reload," L-84-148, June 4, 1984.

3. CENPD-132, "Calculative Methods for the C-E Large Break LOCA Evaluation Model," August 1974 (Proprietary).

   CENPD-132, Supplement 1, "Updated Calculative Methods for the C-E Large Break LOCA Evaluation Model," December 1974 (Proprietary).

4. CENPD-132, Supplement 2, "Calculational Methods for the C-E Large Break LOCA Evaluation Model, " July 1975 (Proprietary)

5. CENPD-133, "CEFLASH-4A, A FORTRAN IV Digital Computer Program for Reactor Blowdown Analysis", April 1974 (Proprietary).

   CENPD-133, Supplement 2, "CEFLASH-4A, A FORTRAN IV Digital Computer Program for Reactor Blowdown Analysis (Modification)", December 1974 (Proprietary).

6. CENPD-134, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," April 1974 (Proprietary).

   CENPD-134, Supplement 1, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core (Modification)", December 1974 (Proprietary).

7. CENPD-135, "STRIKIN, A Cylindrical Geometry Fuel Rod Heat Transfer Program, April 1974 (Proprietary).

CENPD-135, Supplement 2, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program (Modification)", February 1975.

CENPD-135, Supplement 4, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program", August 1976 (Proprietary).

8. CENPD-138, and Supplement 1 "PARCH, A FORTRAN IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup, " February 1975.

CENPD-138 Supplement 2 (P), "PARCH - A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup", January 1977 (Proprietary).

9. CENPD-139-P-A, "C-E Fuel Evaluation Model Topical Report", July 1974.

10. CEN-161(B)-P, "Improvements to Fuel Evaluation Model Topical Report," July 1981.

11. Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr. (BG&E), dated March 31, 1983.

12. Letter #F2-CE-R-043, "St. Lucie 2 Large Break LOCA Axial Shape Sensitivity Study Results Based on the August 1974 Evaluation Medel", November 14, 1985.

Table 1

St. Lucie - Unit 2

Limiting Break Size (1.0 DEG/PD)

Parameter	Value
Peak Linear Heat Generation Rate (kw/ft)	13.0
Peak Clad Temperature (°F)	2106
Time of Peak Clad Temperature (Seconds)	259
Time of Clad Rupture (Seconds)	55.85
Peak Local Clad Oxidation (%)	$<$ 16.12
Total Core-Wide Clad Oxidation (%)	$<$ 0.70

Table 2

St. Lucie - Unit 2

Analysis Plots

Variables	Figure Designation
Peak Clad Temperature	1
Hot Spot Gap Conductance	2
Peak Local Clad Oxidation	3
Clad, Fuel Centerline, Average Fuel, and Coolant Temperatures for Hottest Node	4
Hot Spot Heat Transfer Coefficient	5
Hot Rod Internal Gas Pressure	6

Table 3

St. Lucie - Unit 2

Significant Parameters and Initial Conditions

Parameter	Value
Core Power Level at 102% of Nominal, MWt	2754
Core Average Linear Heat Rate (102% of Nominal) (kw/ft)	4.90
Peak Linear Heat Generation Rate (PLHGR) Hot Assembly, Hot Channel (kw/ft)	13.0
Peak Linear Heat Generation Rate (PLHGR) Hot Assembly, Average Channel (kw/ft)	11.45
Core Inlet Temperature (°F)	552
Core Outlet Temperature (°F)	603.8
System Flow Rate (lbm/hr)	136.1×10^6 (1)
Core Flow Rate (lbm/hr)	131.1×10^6
Gap Conductance at PLHGR(2) (Btu/hr-ft^2-°F)	1416
Fuel Centerline Temperature at PLHGR(2) (°F)	3228
Fuel Average Temperature at PLHGR(2) (°F)	2078
Hot Rod Gas Pressure (psia)(2)	1118
Hot Rod Burnup (MWD/MTU)	1038
Number of Tubes Plugged Per Steam Generator	1250
Augmentation Factor	1.01

(1) System flowrate consistent with 363,000 gpm.
(2) STRIKIN-II values at hot rod burnup which yields highest peak clad temperature.

FIGURE 1

PEAK CLAD TEMPERATURE
1.0 x DOUBLE ENDED GUILLOTINE BREAK
IN PUMP DISCHARGE LEG

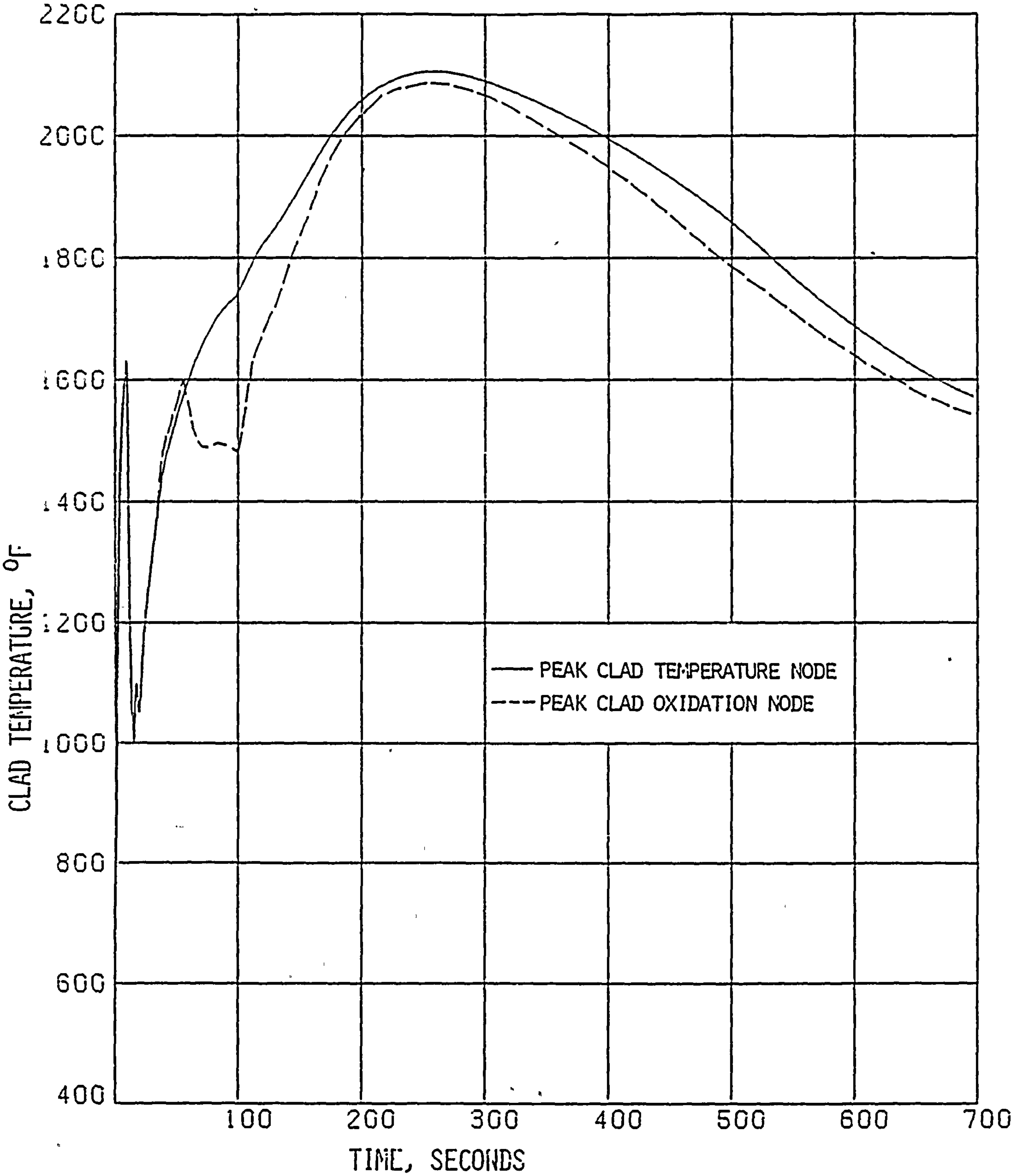

FIGURE 2

HOT SPOT GAP CONDUCTANCE
1.0 x DOUBLE ENDED GUILLOTINE BREAK
IN PUMP DISCHARGE LEG

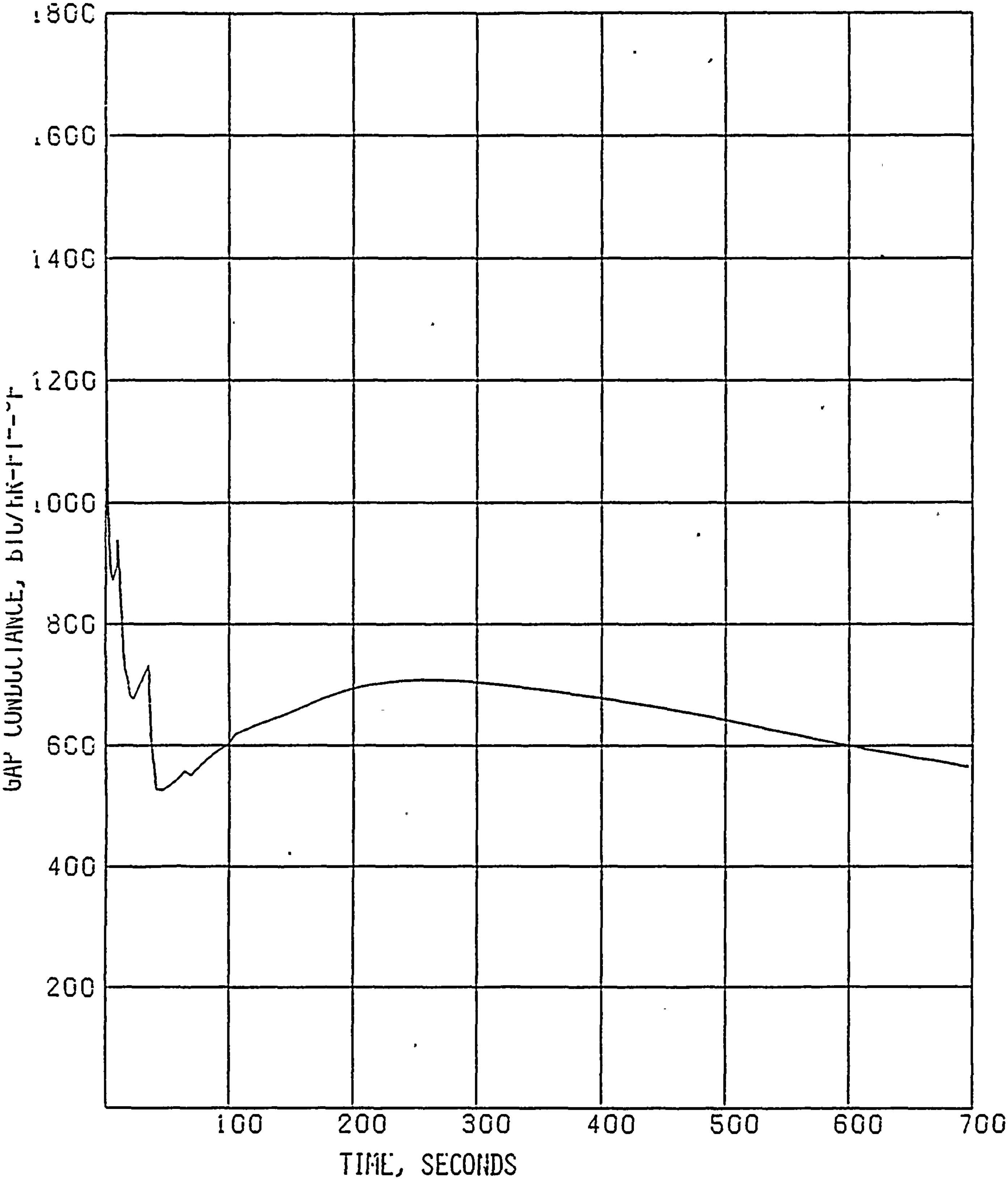

FIGURE 3

PEAK LOCAL CLAD OXIDATION
1.0 x DOUBLE ENDED GUILLOTINE BREAK
IN PUMP DISCHARGE LEG

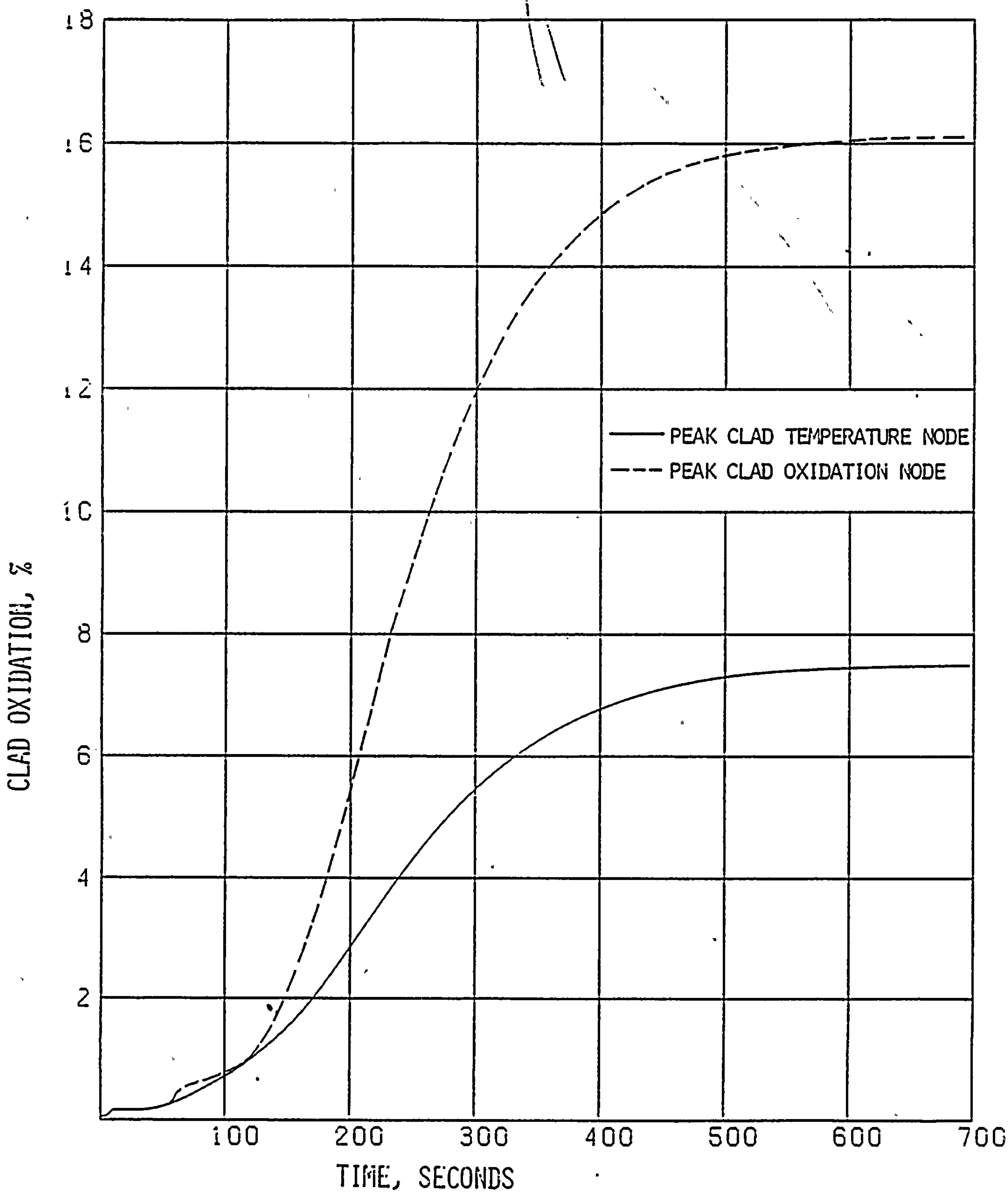

FIGURE 4

CLAD, CENTERLINE, AVERAGE FUEL AND COOLANT TEMPERATURE FOR HOTTEST NODE 1.0 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG

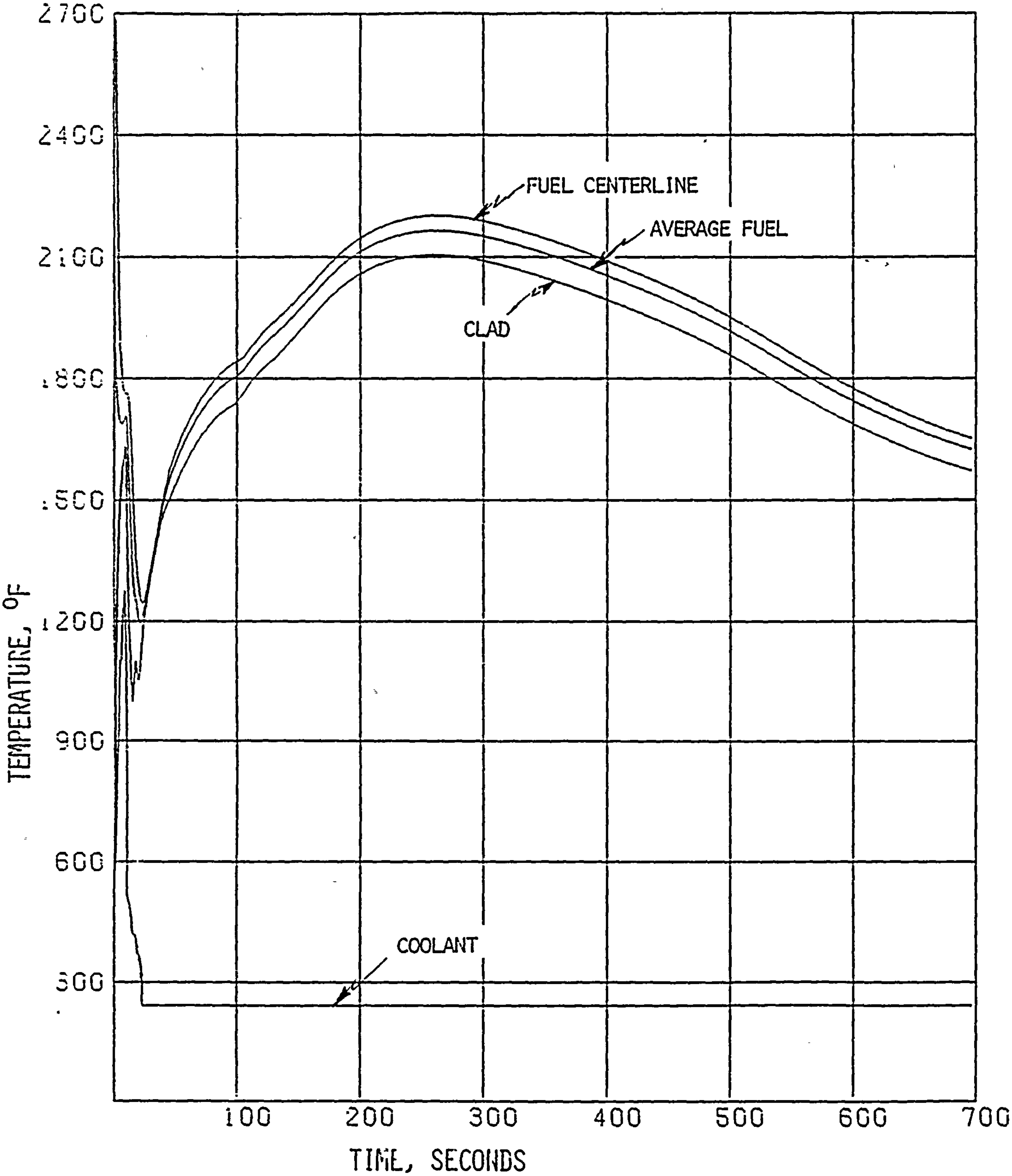

FIGURE 5

HOT SPOT HEAT TRANSFER COEFFICIENT
1.0 x DOUBLE ENDED GUILLOTINE BREAK
IN PUMP DISCHARGE LEG

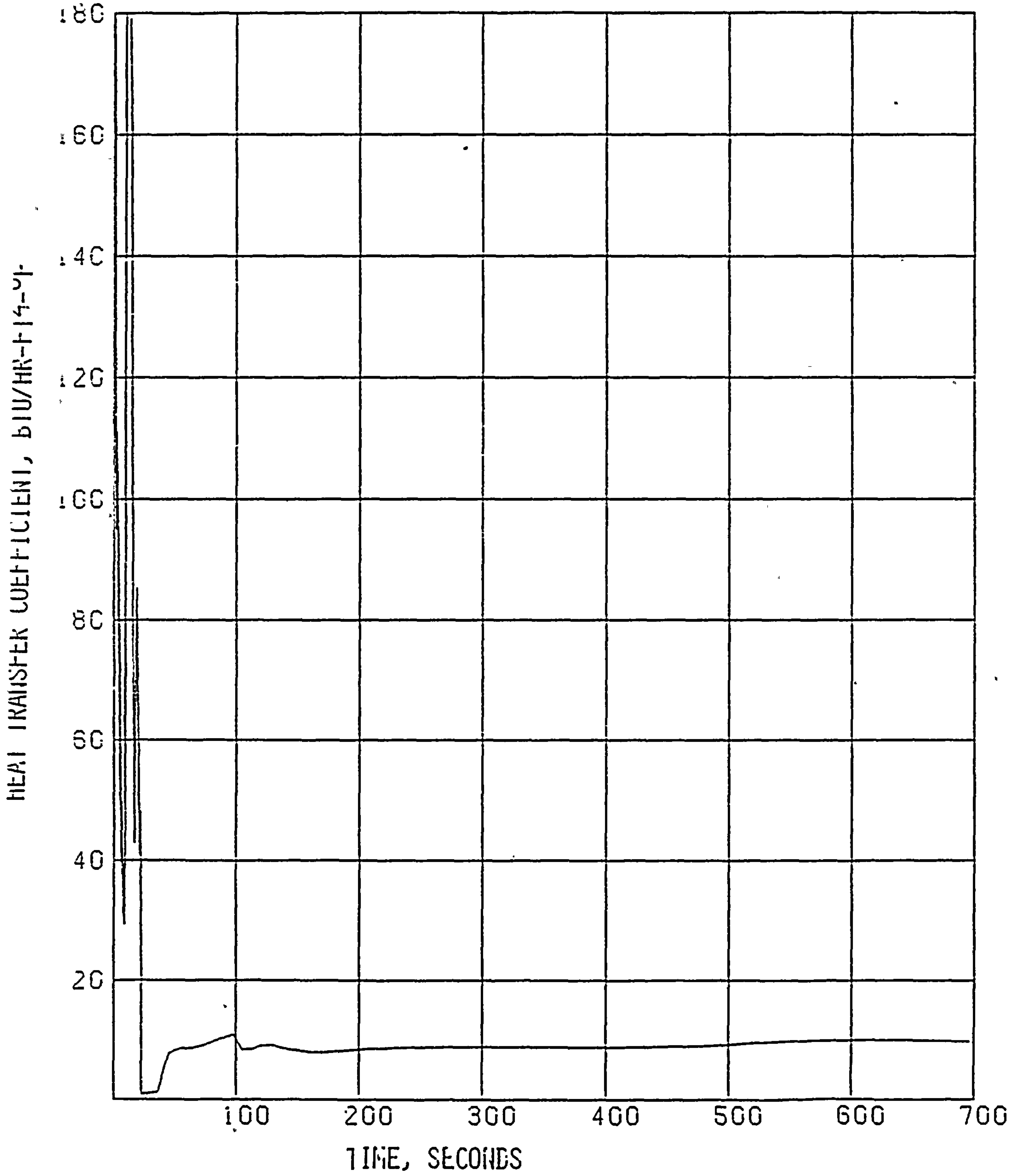

FIGURE 6

HOT ROD INTERNAL GAS PRESSURE
1.0 x DOUBLE ENDED GUILLOTINE BREAK
IN PUMP DISCHARGE LEG