INDIAN POINT UNIT 3 SMALL BREAK ECCS ANALYSIS

February 1978

LIST OF TABLES

LIST OF FIGURES

Note: Figures 3 through 8 are for the limiting small break. Figures 10a, 10b, 11a, 11b, 12a, 12b are for two other non-limiting small break cases.

Method of Analysis

For small breaks less than 1.0 ft^2 the WFLASH (1, 2, 3) digital computer code is employed to calculate the transient depressurization of the Reactor Coolant System as well as to describe the mass and enthalpy of flow through the break.

The WFLASH program used in the analysis of the small break loss of coolant accident is an extension of the FLASH-4[4] code developed at the Westinghouse Bettis Atomic Power Laboratory. The WFLASH program permits a detailed spatial representation of the Reactor Coolant System.

The reactor coolant system 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 WFLASH is given in Reference 1. The modifications to WFLASH and LOCTA-IV that represent the October, 1975 Small Break Model version utilized for this analysis is given in References 2 and 3.

The use of WFLASH in the analysis involves, among other things, the representation of the reactor core as a heated control volume with the associated bubble rise model to permit a transient mixture height calculation. The multi-node 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.

For these analyses, the SI delivery considers pumped injection flow which is depicted in Figure 1 as a function of RCS pressure. This figure represents injection flow from the SI pumps based on performance curves degraded 5 percent from the design head. The 25 second delay includes time required for diesel startup and loading of the safety injection pumps onto the emergency buses. The effect of RHR pump flow is not considered since their shutoff head is lower than RCS pressure during the time portion of the transient considered here. Also minimum Safeguards Emergency Core Cooling System capability and operability has been assumed in these analyses.

Peak clad temperature analyses are performed with the LOCTA-IV (5, 2, 3) code which determines the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core and mixture height history.

Figure 2 presents the hot rod power shape utilized to perform the small break analysis presented here. This power shape was chosen because it provides an appropriate distribution of power versus core height and also local power is maximized in the upper regions of the reactor core (10' to 12'). This power shape is skewed to the top of the core with the peak local power occurring at the 10' core elevation.

This is limiting for small break analysis because of the core uncovery process for small breaks. As the core uncovers, the cladding in the upper elevation of the core heats up and is sensitive to the local power at that elevation. The cladding temperatures in the lower elevations of the core, below the two phase mixture height, remains low. The peak clad temperature occurs above 10 feet.

Results

This section presents results of the limiting break size in terms of highest peak clad temperature. The worst small break size is 6 inches. The depressurization transient for this break is shown in Figure 3. The extent to which the core is uncovered is shown in Figure 4.

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 resultant heat transfer cools the fuel rod and clad to very near the coolant temperatures as long as the core remains covered by a two phase mixture.

The maximum hot spot clad temperature calculated during the transient is 1263°F including the effects of fuel densification as described in Reference 6 and the reactor vessel upper head temperature was assumed to be equal to the hot leg temperature. The peak clad temperature transient is shown in Figure 5 for the worst break size i.e., the break with the highest peak clad temperature. The steam flow rate for the worst break is shown on Figure 6. When the mixture level drops below the top of the core, the steam flow computed in WFLASH provides cooling to the upper portion of the core. The rod film coefficient for this phase of the transient is given in Figure 7. The hot spot fluid temperature for the worst break is shown in Figure 8.

The core power (dimensionless) transient following the accident (relative to reactor scram time) is shown in Figure 9.

The reactor scram time is equal to the reactor trip signal time plus 3.4 sec for signal transmission and rod insertion. During this period the reactor is conservatively assumed to operate at rated power.

<u>Additional Break Sizes</u>

Additional break sizes were analyzed. Figures 10a and 10b present the RCS pressure transient for the 4 and 8 inch breaks respectively and Figures 11a and 11b present the volume history (mixture height) plots for both breaks. The peak clad temperatures for both cases are less than the peak clad temperature of the 6 inch break. The peak clad temperatures for both cases are given in Figures 12a and 12b.

Conclusions

Analyses presented in this report show that the high head portion of the Emergency Core Cooling System, together with accumulators, provide sufficient core flooding to keep the calculated peak clad temperatures below required limits of 10CFR50.46. Hence, adequate protection is afforded by the Emergency Core Cooling System in the event of a small break loss of coolant accident.

References

1. V. J. Esposito, K. Kesavan, B. A. Maul WFLASH-A FORTRAN IV Computer Program for Simulation of Transients in a Multi-Loop PWR, WCAP-8261 Rev. 1, July 1974.

2. R. J. Skwarek, W. J. Johnson, P. E. Meyer, "Westinghouse Emergency Core Cooling System Small Break October 1975 Model", WCAP-8970, April 1977.

3. Letter from C. Eicheldinger to J. Stolz, NS-CE-1672, dated February 10, 1978.

4. Porsching, T. A., Murphy, J. H., Redfield, J. A., and Davis, V. C., "FLASH-4: A Fully Implicit FORTRAN-IV Program for the Digital Simulation of Transients in a Reactor Plant," WAPD-TM-84; Bettis Atomic Power Laboratory (March, 1969).

5. F. M. Bordelon, et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8305, June 1974.

6. J. M. Hellman, "Fuel Densification Experimental Results and Model for Reactor Application," WCAP-8219, October, 1973.

TABLE 1

SMALL BREAK

TIME SEQUENCE OF EVENTS

	4 in.	6 in.	8 in.
Start			
Reactor Trip Signal (Sec.)	19.0	11.5	10.25
Top of Core Uncovered (Sec.)	372.	153.	82.
Accumulator Injection Begins (Sec.)	898.	352.	186.
PCT Occurs (Sec.)	944.	566.	147.
Top of Core Covered (Sec.)	1198.	574.	227.

TABLE 2

SMALL BREAK RESULTS

Case Analyzed	4 IN	6 IN	8 IN
Results			
Peak Clad Temp. °F	1222.	1263	990
Peak Clad Temp. Location Ft.	12.0	11.0	10.75
Local Zr/H_2O Rxn(max)%	0.333	0.33	0.32
Local Zr/H_2O Location Ft.	11.75	11.0	10.75
Total Zr/H_2O Rxn %	<0.3	<0.3	<0.3
Hot Rod Burst Time sec	N/A	N/A	N/A
Hot Rod Burst Location Ft.	N/A	N/A	N/A

Calculation

Core Power Mwt 102% of	3025.
Peak Linear Power kw/ft	See Figure 2
Accumulator Water Volume (ft^3 per accumulator)	800.

Fuel region + cycle analyzed	Cycle	Region
Unit 3	2	4

FIGURE 1

Safety Injection Flowrate (Total)

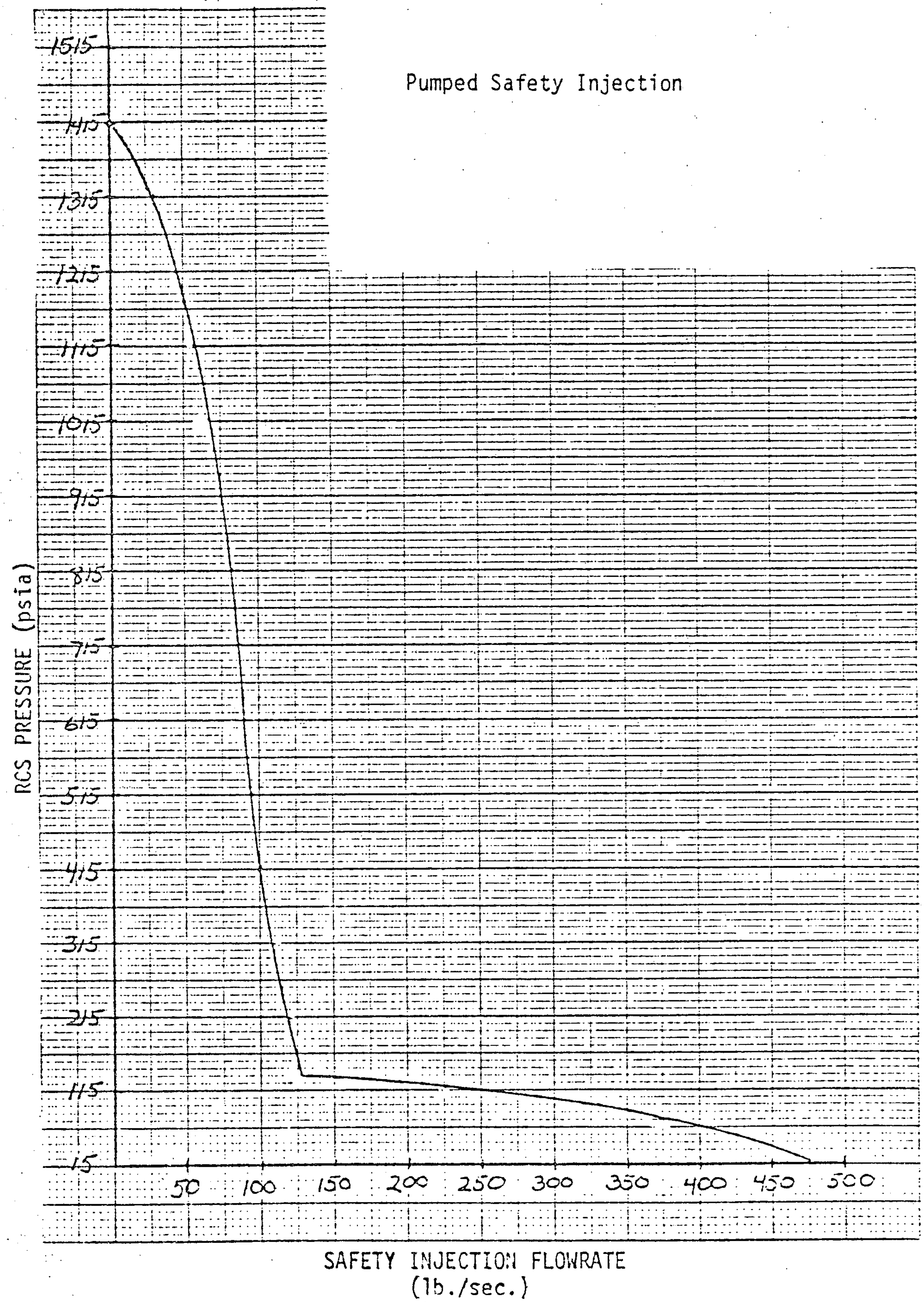

FIGURE 2

Core Power Distribution

Axial Power Shape

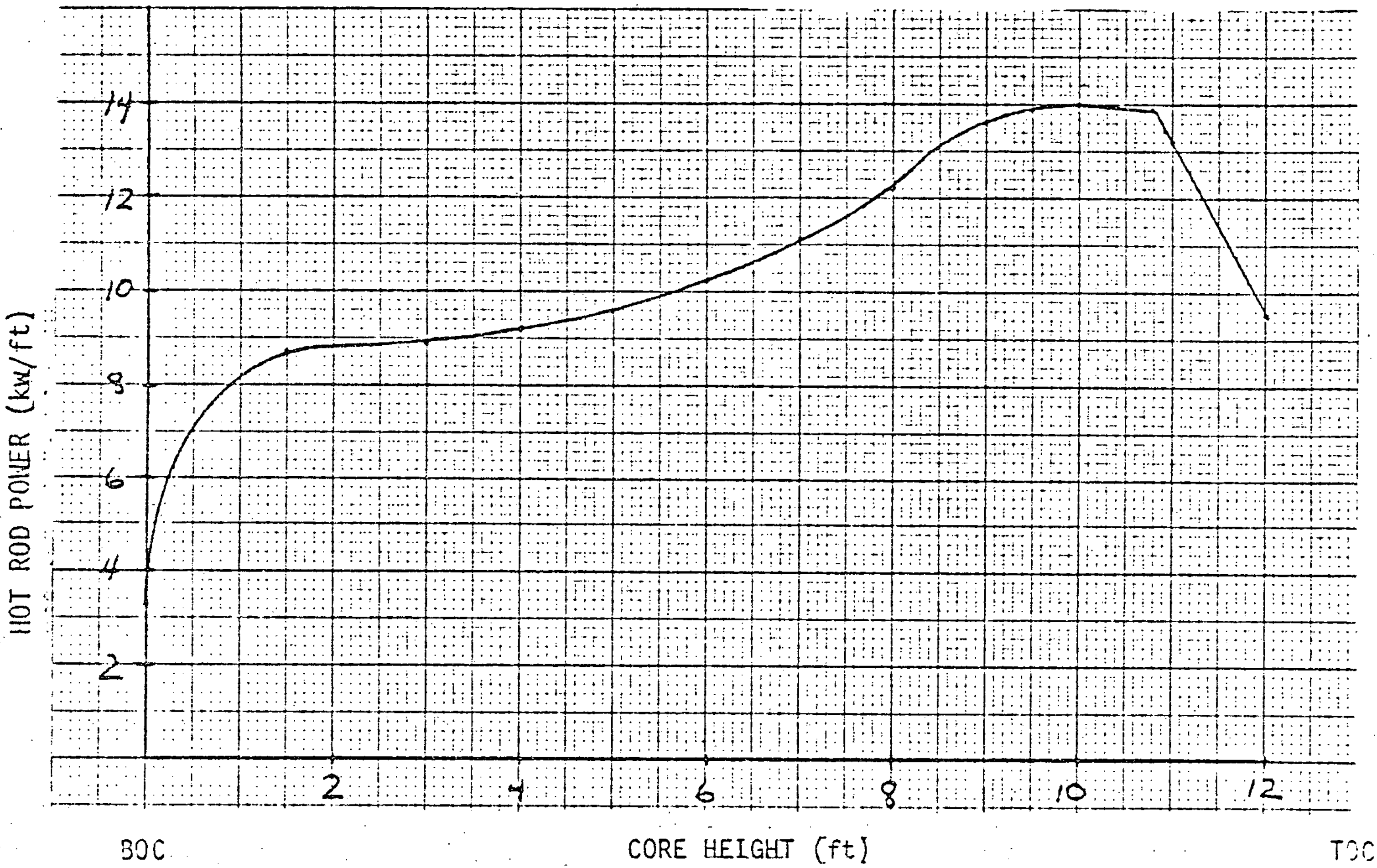

FIGURE 3

RCS Depressurization Transient

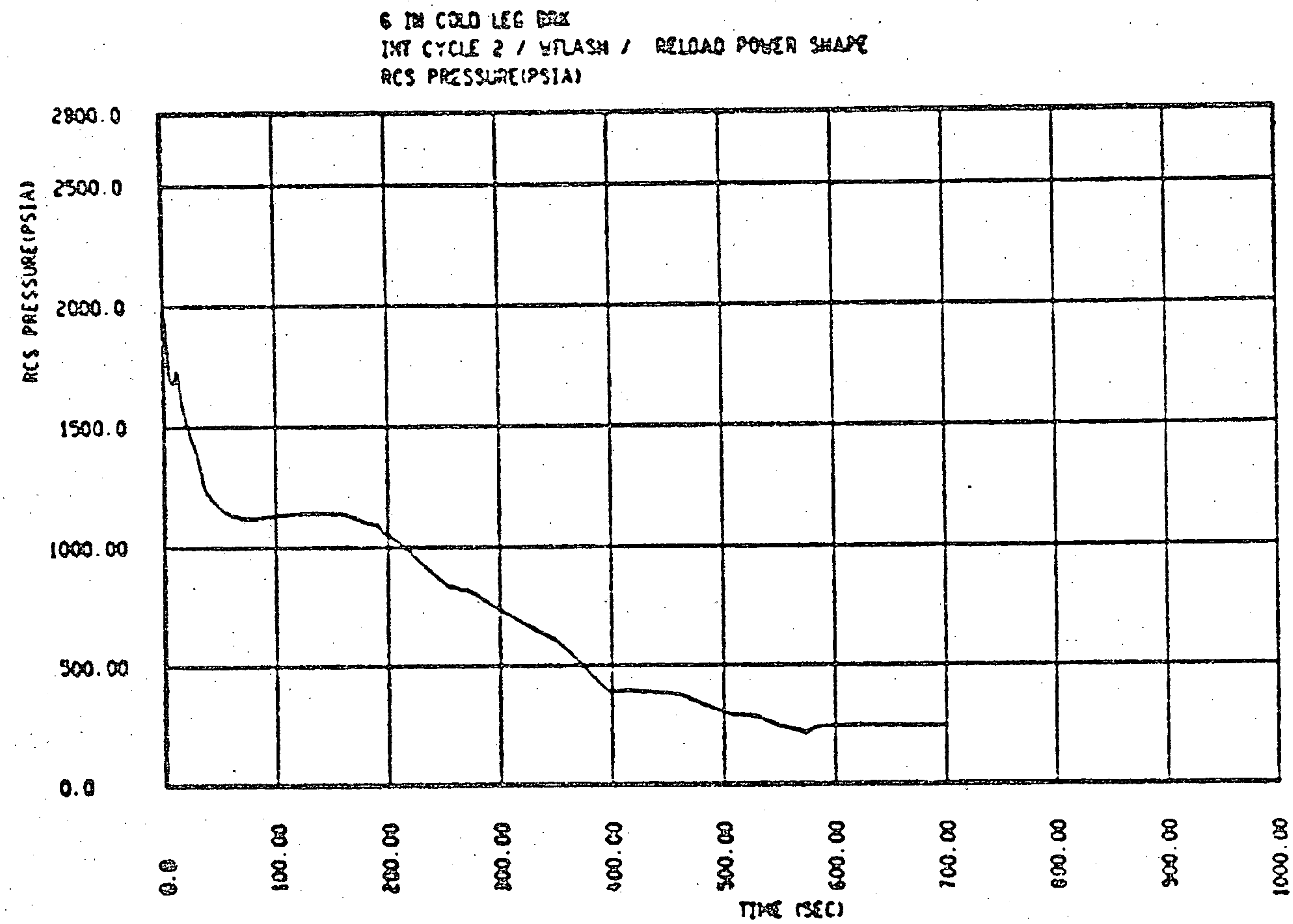

FIGURE 4

Core Volume History

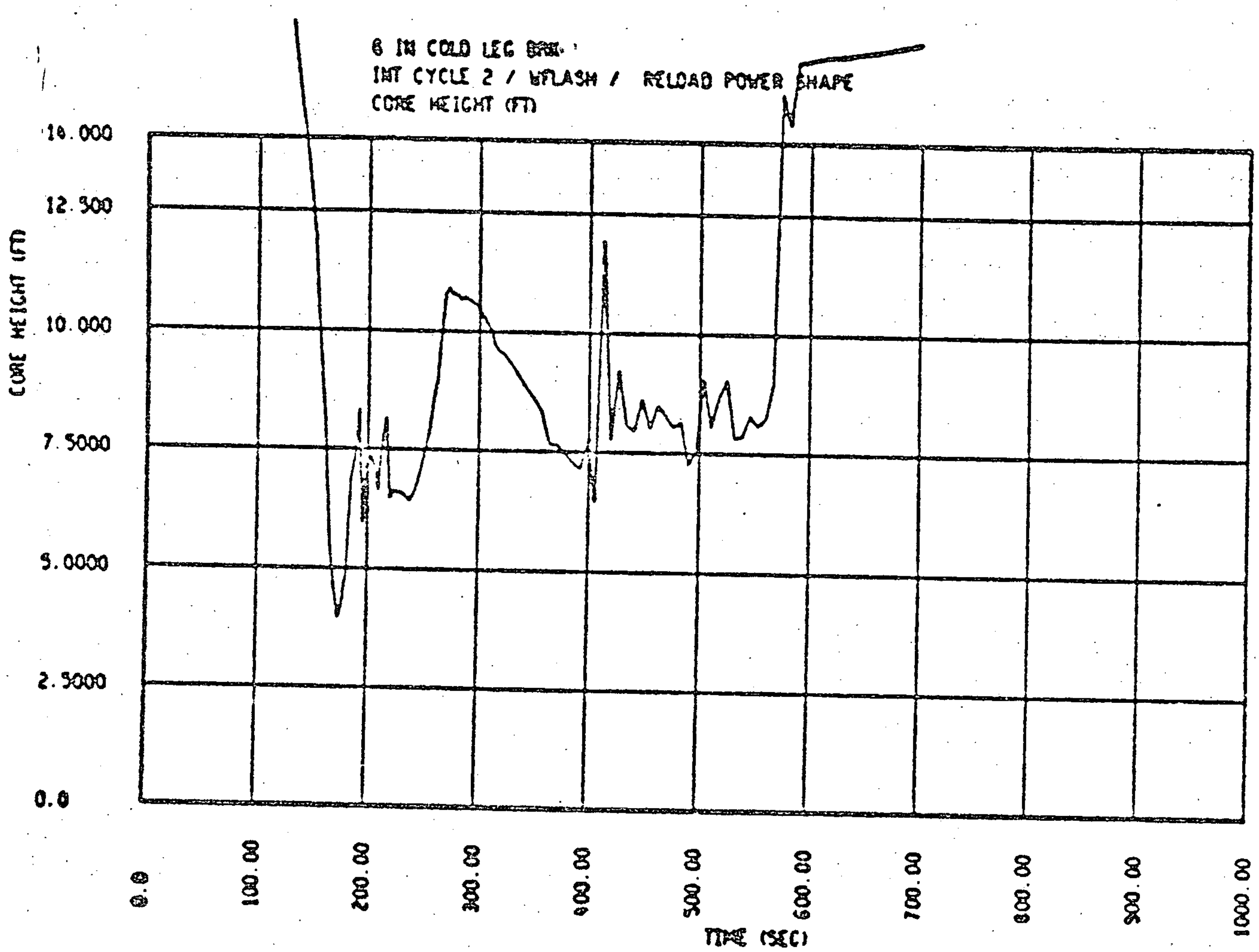

FIGURE 5

Peak Clad Temperature

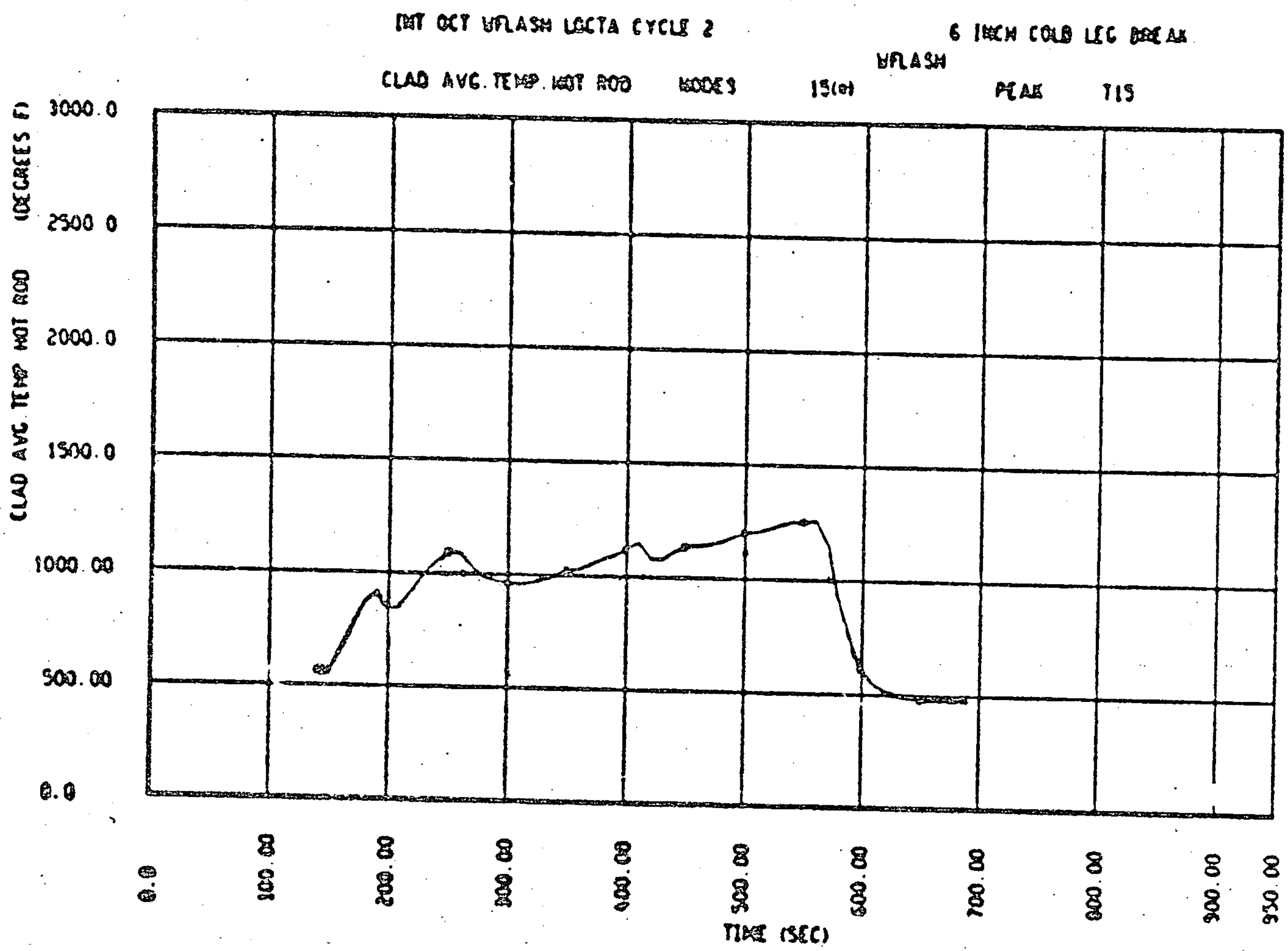

FIGURE 6

Core Steam Flowrate

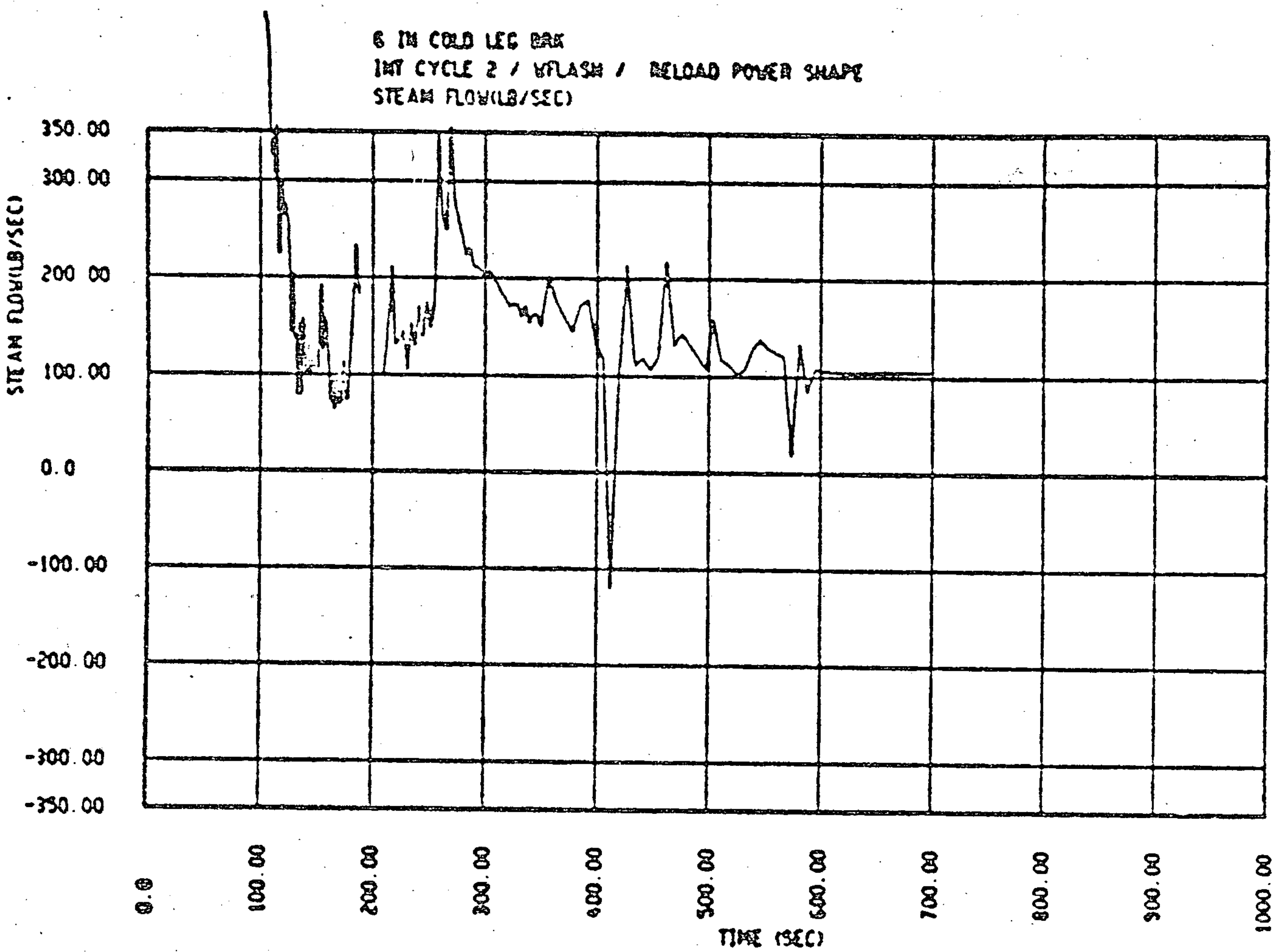

FIGURE 7

Rod Film Coefficient

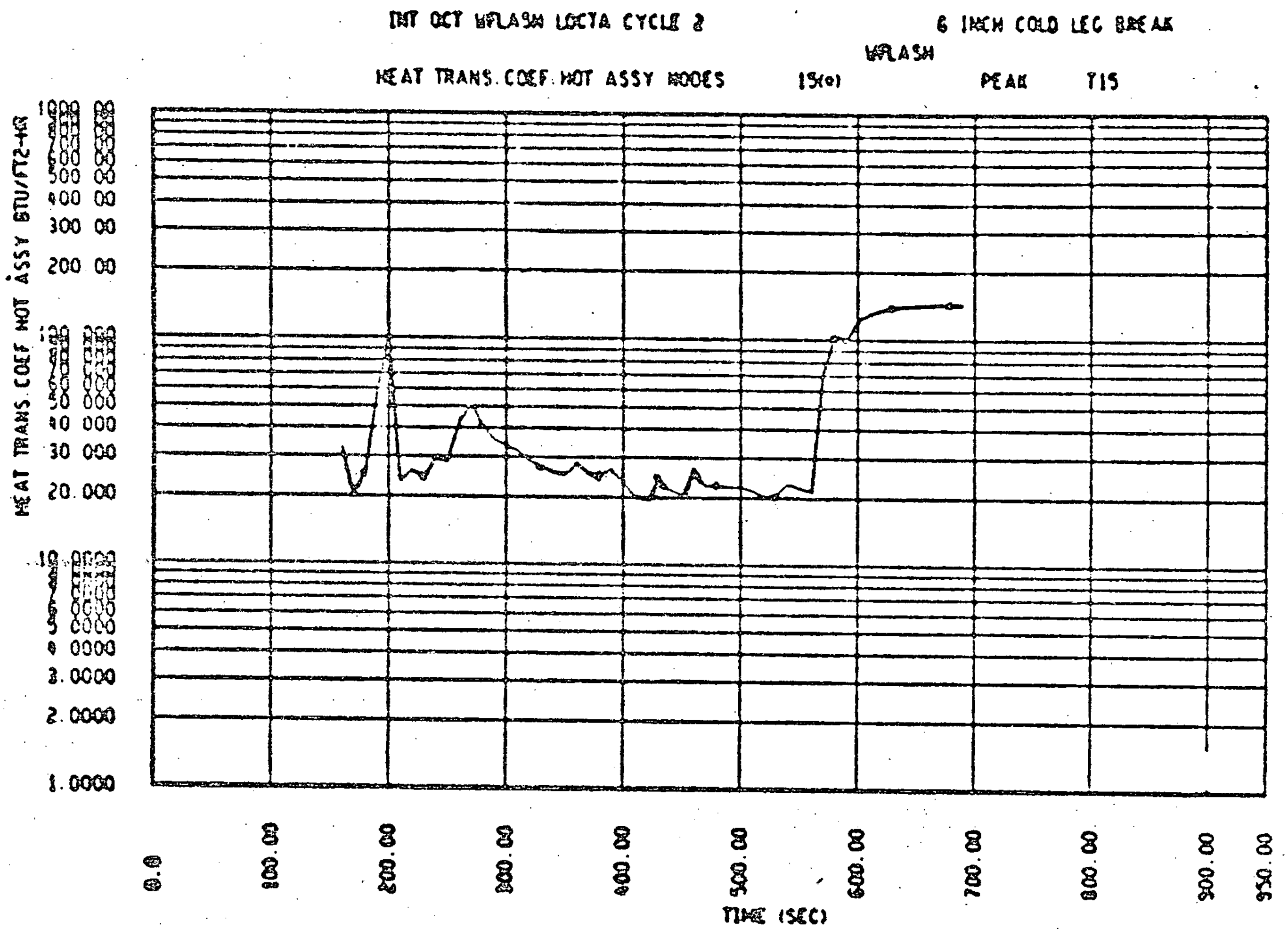

FIGURE 8

Hot Spot Fluid Temperature

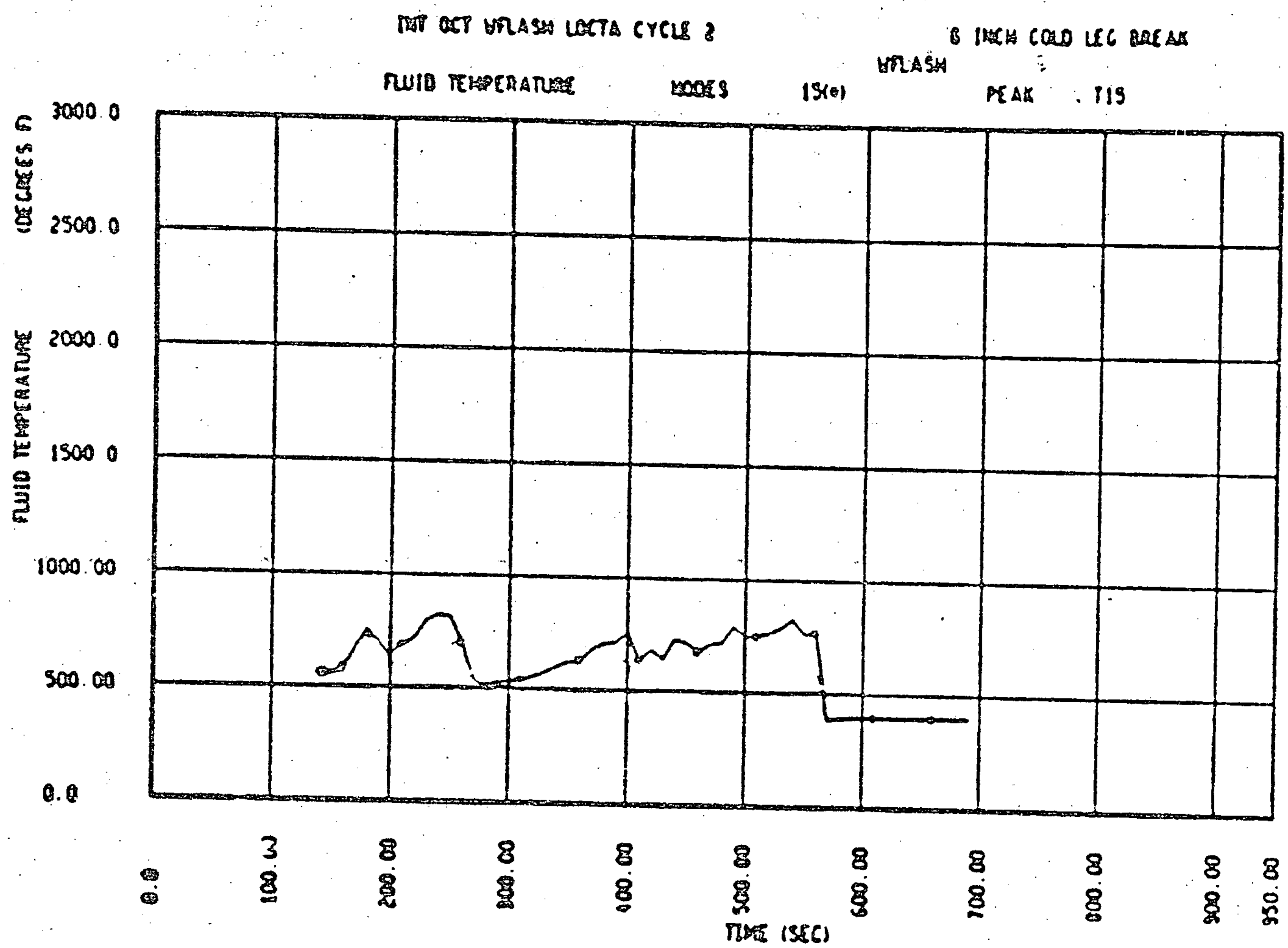

FIGURE 9

Core Power After Reactor Trip

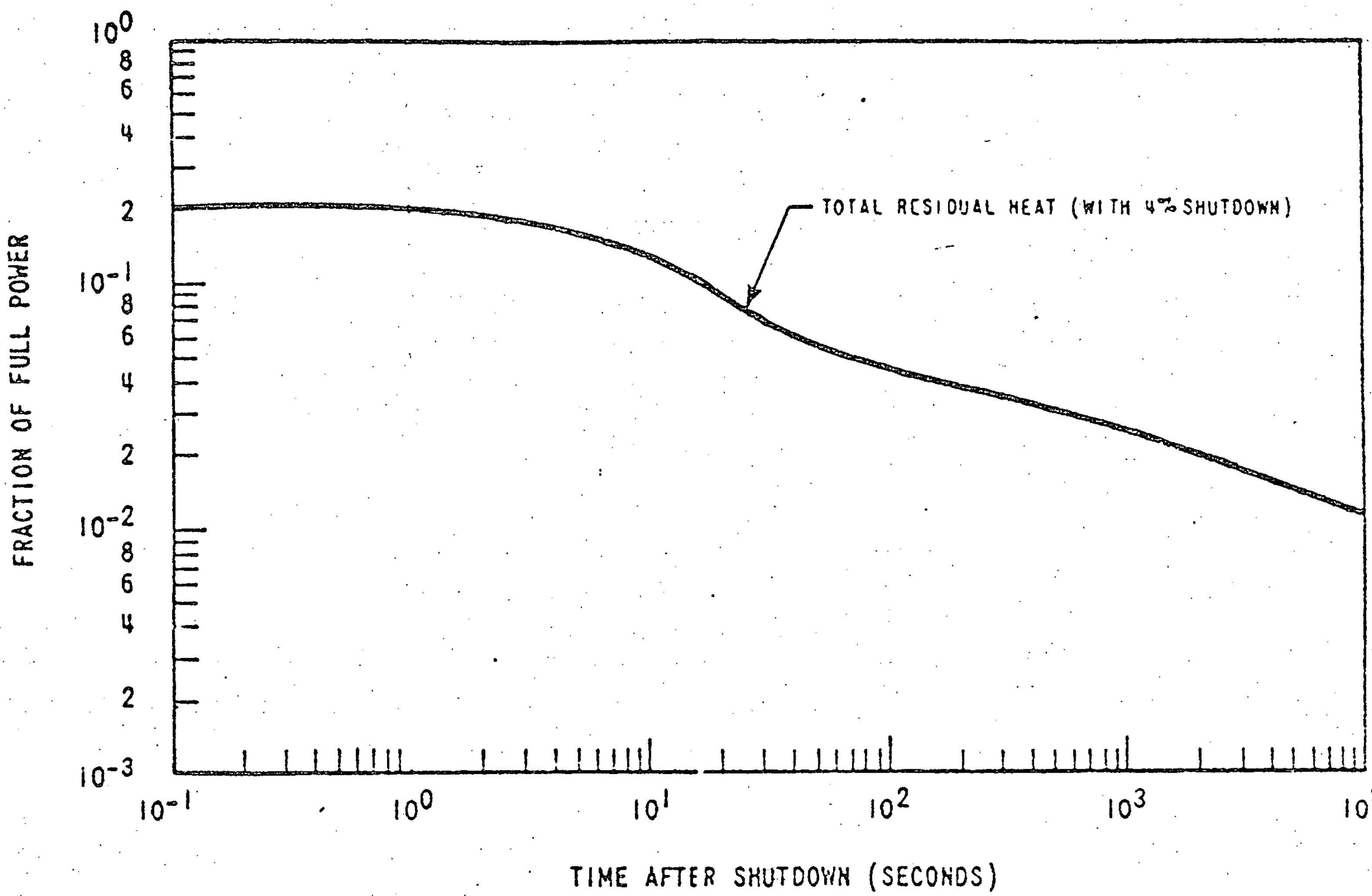

FIGURE 10a

RCS Depressurization Transient

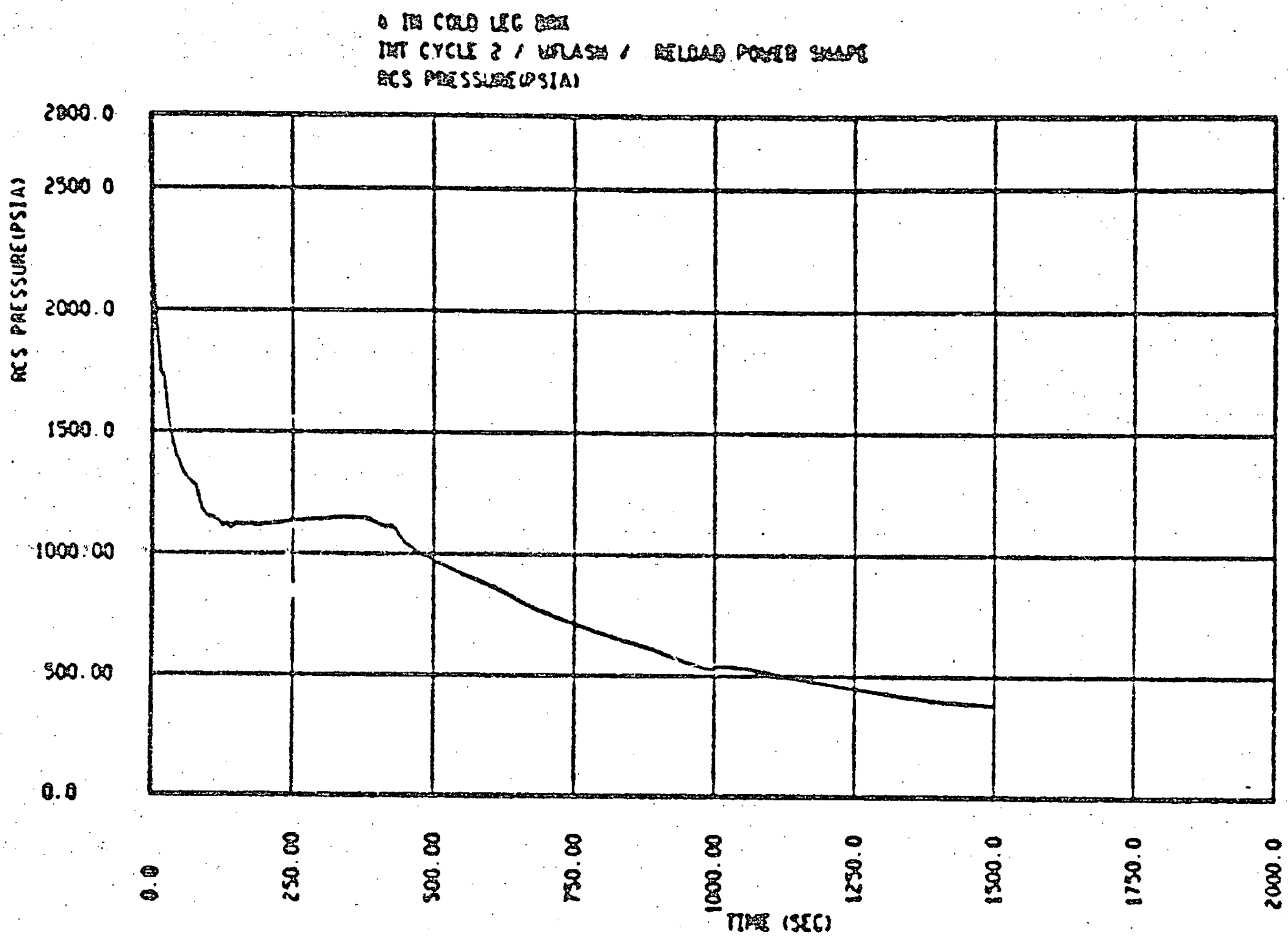

FIGURE 10b

RCS Depressurization Transient

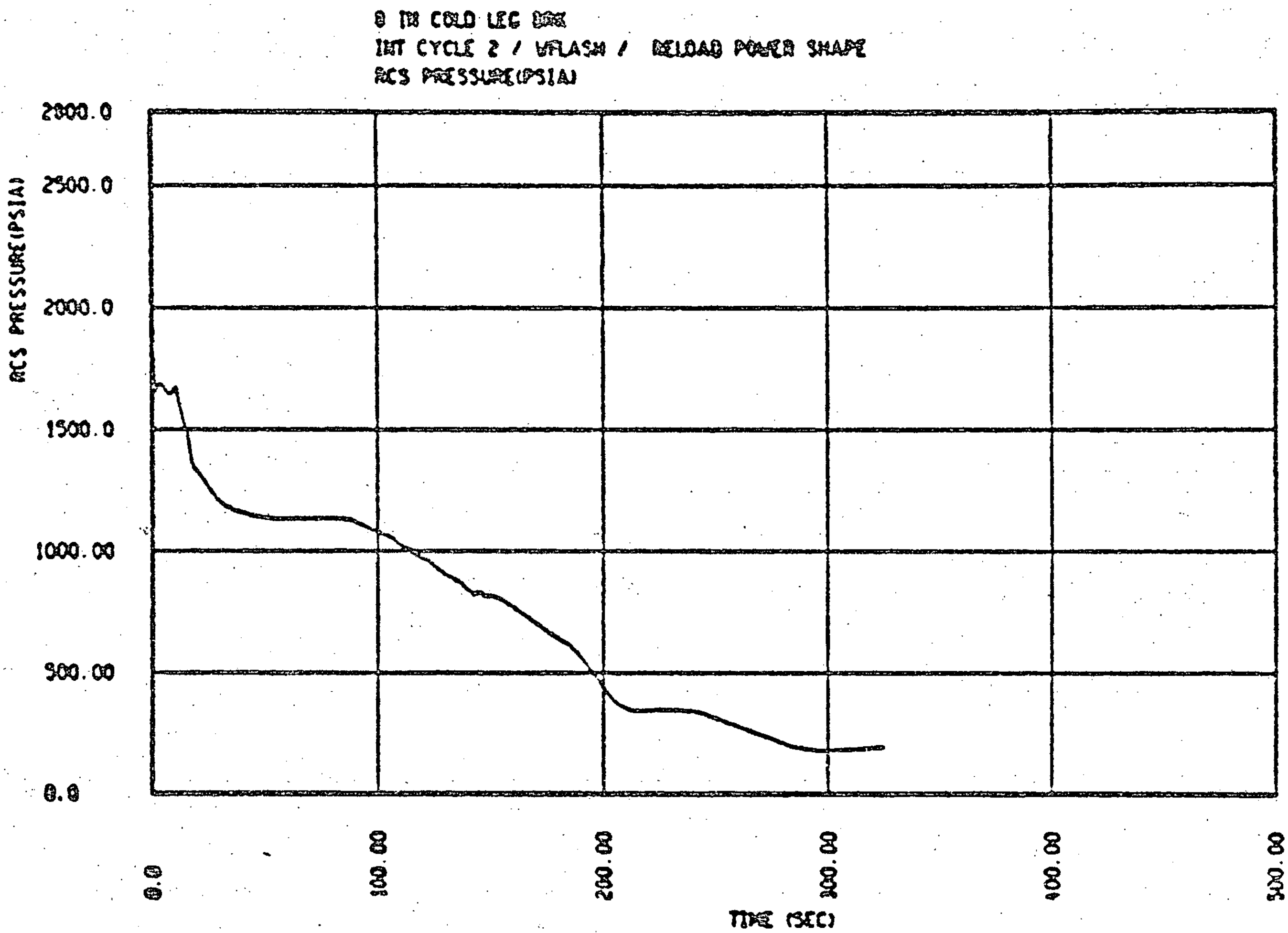

FIGURE 11a

Core Volume History

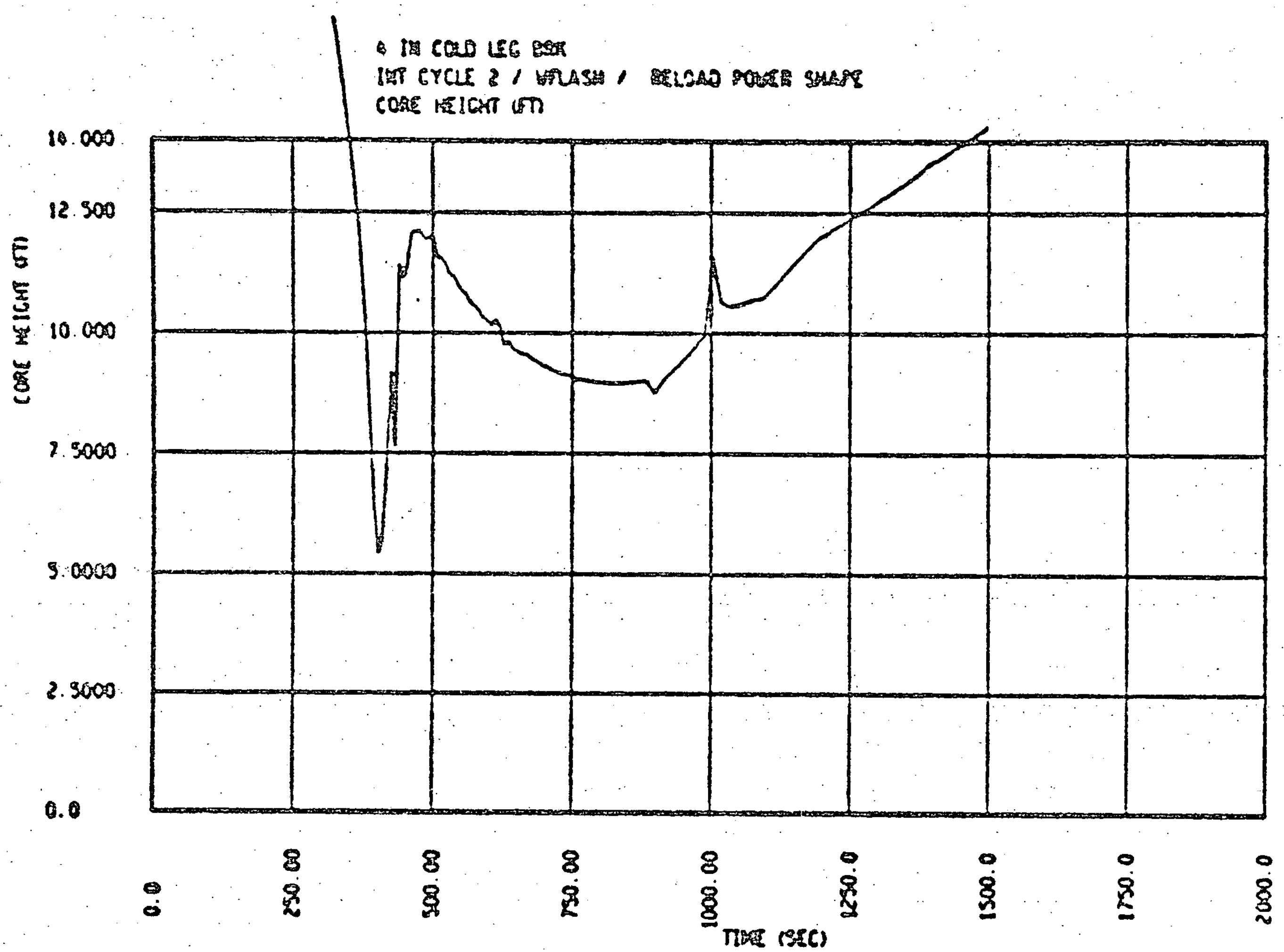

FIGURE 11b

Core Volume History

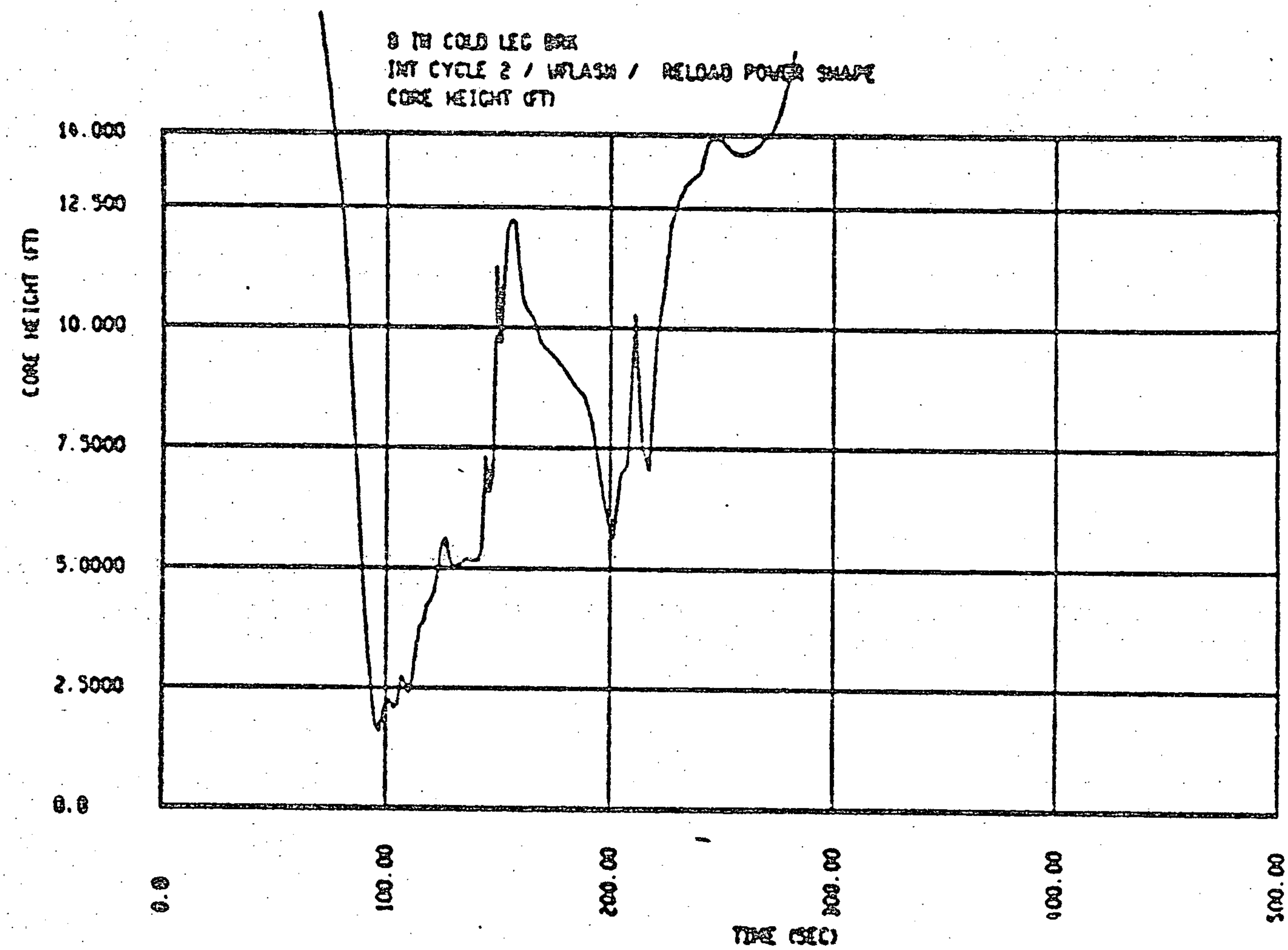

FIGURE 12a

Peak Clad Temperature

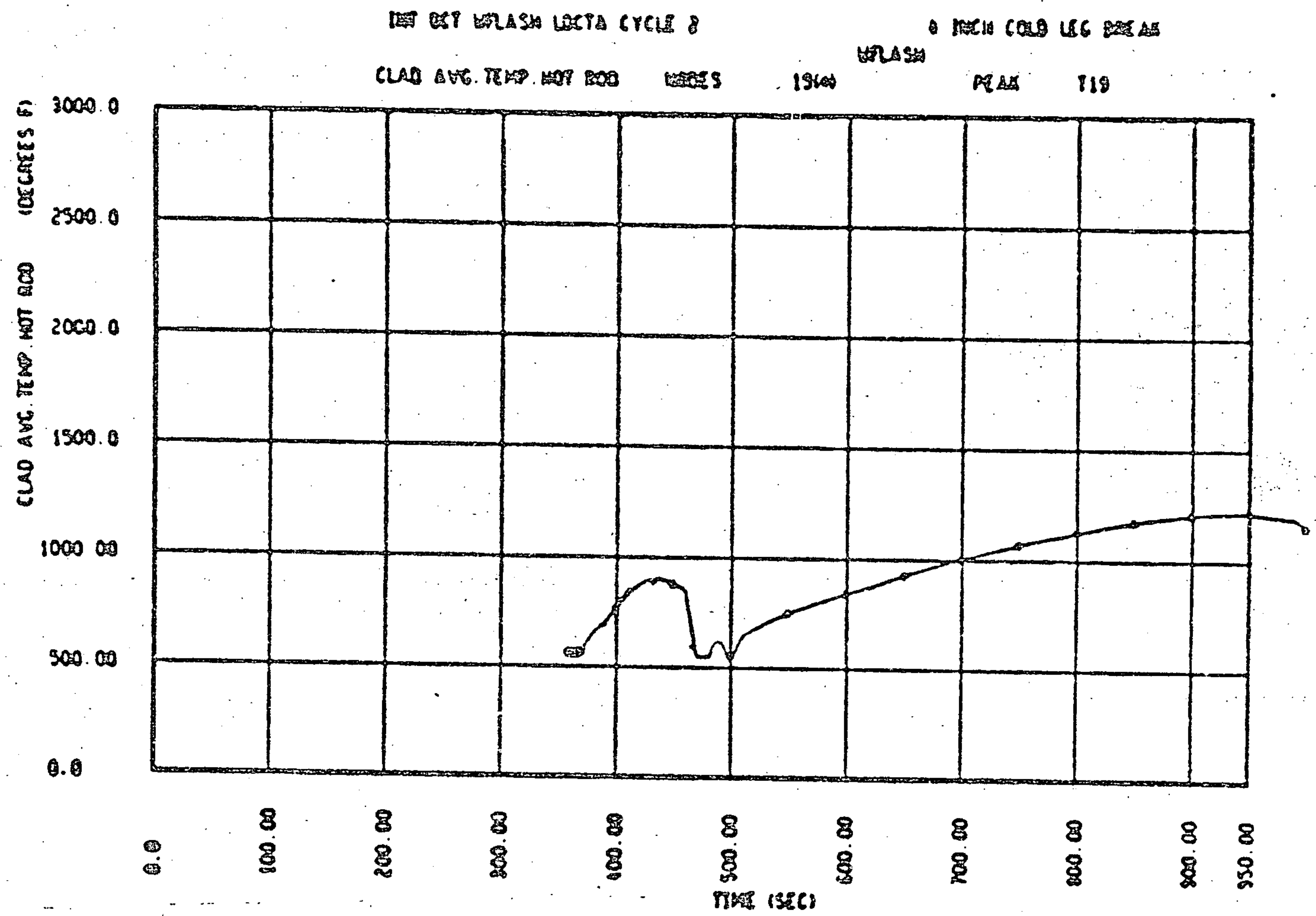

FIGURE 12b

Peak Clad Temperature