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SUBJECT: FORWARDS "ECCS ANALYSIS OF SMALL BREAKS IN CONJUNCTION W/ EMERGENCY FEEDWATER FLOW FAILURES."

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April 21, 1979

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Denwood F. Ross, Jr.
Deputy Director for Project Management

Reference: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

Dear Mr. Denton:

Attached is an analysis of small breaks for Oconee Nuclear Station, Units 1, 2, and 3, assuming a single failure in the emergency feedwater system. This analysis demonstrates acceptable core cooling performance during postulated small break events in conjunction with the loss of emergency feedwater flow, assuming operator action to establish emergency feedwater flow within 20 minutes following the transient.

Current Oconee operating procedures include a provision requiring operators to manually establish emergency feedwater flow upon loss of the normal feedwater flow and emergency feedwater flow whenever primary system cooling through the steam generators is required.

For the existing EFWS configuration the worst single failure is the failure of the turbine-driven emergency feedwater pump to start automatically. Our experience with this emergency feedwater pump is that if the pump should fail to start automatically, it can be started locally by manual action. Additionally, the manual cross-connect feature of the emergency feedwater pump discharge header of each unit permits utilization of another unit's pump, should it become necessary. These manual actions can be accomplished within the time limitations of the attached analysis.

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It is concluded from this analysis that the existing arrangement of the emergency feedwater system together with the operating procedures provide satisfactory transient mitigating capability.

Very truly yours,

William O. Parker, Jr.

William O. Parker, Jr.

PMA:vr

Attachment

OCONEE NUCLEAR STATION
ECCS ANALYSIS OF SMALL BREAKS IN CONJUNCTION
WITH EMERGENCY FEEDWATER FLOW FAILURES

1.0 Introduction

Analysis of a spectrum of small Reactor Coolant System breaks has been performed for Oconee Nuclear Station Units 1, 2, and 3, assuming a single failure in the Emergency Feedwater System (EFWS). The assumed single failure in the EFWS results in no automatic emergency feedwater flow during the postulated small break LOCA. Specifically, three cases have been examined: (1) 0.07 ft^2 break, (2) 0.02 ft^2 break, and (3) 0.01 ft^2 break.

These three cases bound all postulated small break scenarios involving the effects of a loss of automatic emergency feedwater flow.

The results of the analysis of these three cases show that for most small breaks the failure of the emergency feedwater system does not have any adverse impact on the required core cooling. For extremely small breaks, it is seen that although the failure of the EFWS delays the automatic actuation of the High Pressure Injection System (HPIS), core uncovery would not occur with operator action to establish emergency feedwater flow within 20 minutes. Therefore, the criteria of 10CFR50.46 and Appendix K of 10CFR50 are satisfied.

2.0 Method of Analysis

The analysis method used for the evaluation is that described in Chapter 5 of BAW-10104, Revision 3, "B&W's ECCS Evaluation Model," (Reference 1) along with the model modification of Reference 2. As dictated by Reference 3, the Bernoulli correlation was used for subcooled flow rather than the modified Zaloudek correlation, as proposed in Reference 2.

The following assumptions are made for conditions and system responses during the accident:

(1) The reactor is operating at 102 percent of a steady-state power level of 2772 MWt. The Oconee licensed power level is 2568 MWt and the higher power level results in a higher RCS boil-off than would occur for the Oconee units.

(2) The leak occurs instantaneously, and a discharge coefficient of 1.0 is used for the entire analysis. The Bernoulli equation was used for the subcooled portion of the transient while Moody's correlation was used in the two phase portion.

(3) The leak is assumed to occur in the bottom of the pump discharge piping. This results in a loss of approximately 30 percent of the total injected HPI once HPI is actuated.

(4) No off-site power is available.

(5) The reactor trips on low pressure.

(6) The safety rods begin entering the core after a 0.5 second delay from the time the reactor trip signal is reached.

(7) The RCS pumps trip and coastdown coincident with reactor trip.

(8) Main feedwater pump trip and coastdown coincident with reactor trip.

(9) Two complete trains of engineered safeguard systems operate when the applicable setpoints are reached. (The HPI flow assumed in this analysis was taken from the design pump head-flow curve.)

(10) The emergency feedwater system is assumed not to be available because of single failure considerations. At 20 minutes, the operator is assumed to have taken action to manually establish the emergency feedwater flow.

(11) ESFAS signal error band is considered in the analysis to signal the actuation of the HPI system.

(12) The peak linear heat generation rate in the hot pin is the maximum allowed by Technical Specifications.

3.0 Results

3.1 Introduction

The evaluation of a small break with the complete loss of steam generator main feedwater and no automatic actuation of emergency feedwater can be broken up for consideration into two classes of breaks:

I. Those breaks which are large enough to continue depressurization after all secondary cooling is terminated to actuate the HPI system, and

II. Those breaks which are small enough such that after secondary cooling is terminated the reactor coolant system repressurization occurs and operator action is required.

Two Class I breaks have been analyzed - the 0.07 ft^2 and 0.02 ft^2. These breaks are rapidly mitigated by the actuation of the HPI system (two high pressure injection trains are available).

Class II breaks are of more significance. The limiting consequences would occur for the largest break which will not actuate the HPI system on low RCS pressure during the initial depressurization period. This break combines the largest system inventory loss rate with zero makeup (i.e., no automatic initiation of HPI) which maximizes the accident consequences. Operator action for these events is necessary to prevent core uncovery due to the loss of RCS mass inventory.

The 0.01 ft^2 break has been identified as the worst-case Class II break because the reactor coolant system repressurizes just prior to an HPI actuation signal. Operator action can be (a) manual initiation of the HPI system which will depressurize the RCS and provide makeup, or (b) manual initiation of feedwater which will depressurize the RCS and cause an automatic actuation of the HPI. The analysis provided below assumed manual initiation of the auxiliary feedwater. This assumption delays primary system depressurization and maximizes the time to automatically initiate HPI.

Breaks smaller than 0.01 ft^2 will decrease the initial RCS inventory at a slower rate, and therefore, allow longer time for operator action.

3.2 Model

The CRAFT 2 code (Reference 4) was used to calculate the reactor coolant system hydrodynamics during the small break transient. The model used for this analysis is the same as that utilized in the generic 177 fuel assembly lowered loop plant small break analysis reported in the letter J. H. Taylor to S. A. Varga dated July 18, 1978 (Reference 5). The CRAFT model uses 20 nodes to simulate the reactor coolant system, two nodes for the secondary system, and one node for the reactor building. All breaks analyzed in this report are assumed to be located at the bottom of the cold leg piping between the reactor coolant pump discharge and the reactor vessel. After the HPI's are actuated, approximately 30 percent of the total injected HPI water is lost via the break. No emergency feedwater is modeled during the early phases of the transients due to an assumed failure of the emergency feedwater pump. At 20 minutes for the 0.01 ft^2 break case, the operator is assumed to have taken action to restore emergency feedwater to the steam generators.

3.3 Break of 0.07 Ft^2

In the generic 177 FA lowered loop plant small break analysis (Reference 5), the 0.07 ft^2 break was shown to be the worst case. This break was reanalyzed assuming the loss of all auxiliary feedwater. The core pressure and core mixture height behavior during this transient are shown in Figures 1 and 2, respectively. The following table presents the key results of this analysis.

Event	Time
Break Occurs	0.0 sec.
Reactor Trip, Turbine Trip and RCP Coastdown Occur	8.0 sec.
HPI Starts	70.0 sec.
Steam Generator Cooling Essentially Zero	150 sec.

Event	Time
Long Term Cooling Established	390 sec.
Minimum Core Mixture Level	17.3 ft @ t = 75 sec.
Peak Cladding Temperature	720 ^{o}F (initial value)

As shown, the results of this analysis are improved relative to that presented in Reference 5. Because of the relatively large size for this break, heat removal via the steam generator is not necessary to depressurize the RCS. The additional makeup provided by the use of two HPI trains results in less severe consequences. No core uncovery occurred and the criteria of 10CFR50.46 are satisfied.

3.4 Break of 0.02 FT^2

This break is approximately the smallest break that will result in ESFAS actuation prior to the loss of heat removal via the steam generators. Figures 3 and 4, respectively, show the behavior of the core pressure and core mixture height for this transient. The sequence of events is provided in the table below:

Event	Time
Break Occurs	0.0 sec.
Reactor Trip, Turbine Trip, and RC Pump Coastdown Occur	29 sec.
HPI Starts	115 sec.
Steam Generator Cooling is Essentially Lost	250 sec.
Long Term Cooling Established	650 sec.
Minimum Core Mixture Level	19.0 ft @ 460 sec.
Peak Cladding Temperature	720^{o}F (initial value)

In analyzing this break, no emergency feedwater was utilized throughout the transient. After the initial depressurization to the HPI actuation pressure, system pressure stabilized at approximately 1350 psia due to the high pressure injection and energy removal via the break. Because of the two HPI trains, long term cooling was established at 650 seconds and the core never uncovered. Thus, the cladding temperature remained within a few degrees of the fluid temperature throughout the transient and no metal-water reaction occurred. Therefore, the criteria of 10CFR50.46 are satisfied.

3.5 Break of 0.01 FT^2

This break was analyzed because it is approximately the largest break that will not result in ESFAS actuation prior to the loss of steam generator heat removal. By analyzing the largest break, the loss rate of system inventory via the break is maximized. Figures 5 and 6, respectively, show the behavior of the core pressure and core mixture height during this transient. Key events during this transient are shown below.

Event	Time
Break Occurs	0.0 sec.
Reactor Trip, Turbine Trip, RC Pump Coastdown Occur	54.5 sec.
Steam Generator Cooling Essentially Lost	280 sec.
Emergency Feedwater Reestablished	1250 sec.
HPI Initiation	1730 sec.
Long Term Cooling Established	1730 sec.
Minimum Core Mixture Level	15.5 ft @ 1640 sec.
Peak Cladding Temperature	720°F (initial value)

For this break, the system pressure decreases to a minimum value of 1385 psia prior to loss of the steam generator heat removal. This pressure is 20 psia higher than the HPI actuation point including the allowance for instrument error. If the break is too small to remove the energy being generated within the core, the system repressurizes. At 1250 seconds, the system pressure has increased to 2415 psia. Operator action was then assumed to actuate emergency feedwater resulting in a depressurization of the system. At 1730 seconds, the HPI actuation pressure was reached and the subsequent HPI injection was sufficient to establish long term cooling. During the transient, the core remained covered, thus satisfying the criteria of 10CFR50.46.

4.0 Conclusion

As demonstrated, core mixture heights above the top of the core are assured for all small breaks when corrective operator action is provided by 20 minutes. This insures that the peak cladding temperature is merely the initial value at the start of the transient. Long term cooling is established for all breaks. No core damage is incurred and the core, therefore, remains amenable to cooling. Because of the low core temperatures, no metal-water reaction occurs. Thus, all of the criteria of 10CFR50.46 are satisfied.

5.0 References

1. B. M. Dunn, et al., "B&W's ECCS Evaluation Model," BAW-10104, Rev. 3, Babcock & Wilcox, August, 1977.

2. J. H. Taylor to S. A. Varga, Proposed Modifications to Topical Report BAW-10104, "B&W's ECCS Evaluation Model," May 26, 1978.

3. S. A. Varga to J. H. Taylor, Evaluation of BAW-10104, Rev. 3 and 4, September 5, 1978.

4. R. A. Hedrick, J. J. Cudlin, and R. C. Foltz, CRAFT 2, FORTRAN Program for Digital Simulation of a Multi-Node Reactor Plant During Loss of Coolant, BAW-10092, Rev. 2, Babcock & Wilcox, April, 1975.

5. J. H. Taylor to S. A. Varga, Additional ECCS Small Break Analysis for B&W's 177 Fuel Assembly Lowered Loop NSSS, July 18, 1978.

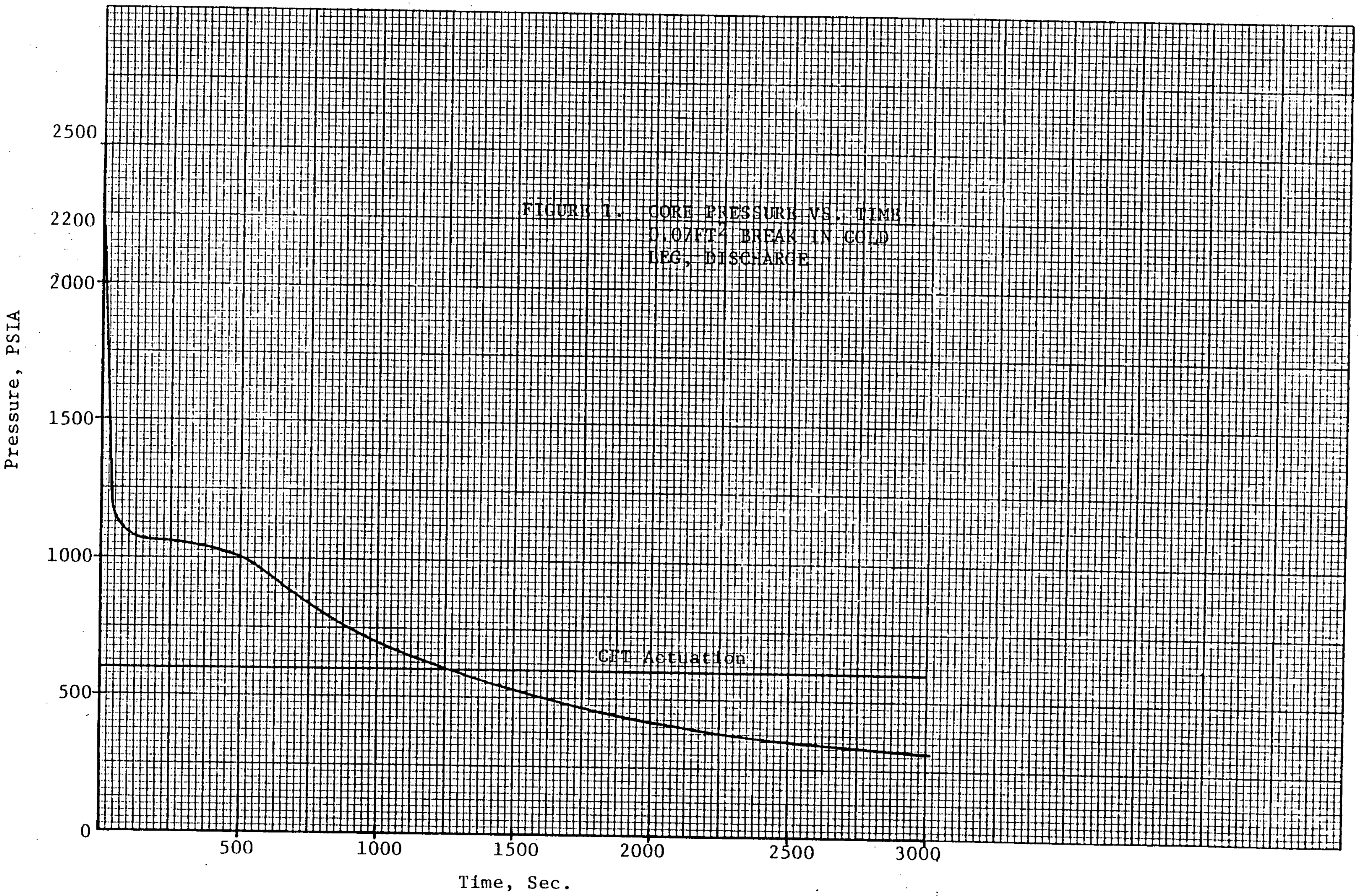
FIGURE 1. CORE PRESSURE VS. TIME
0.07FT² BREAK IN COLD
LEG, DISCHARGE
CFT Actuation
Pressure, PSIA
0
500
1000
1500
2000
2200
2500
Time, Sec.
500
1000
1500
2000
2500
3000

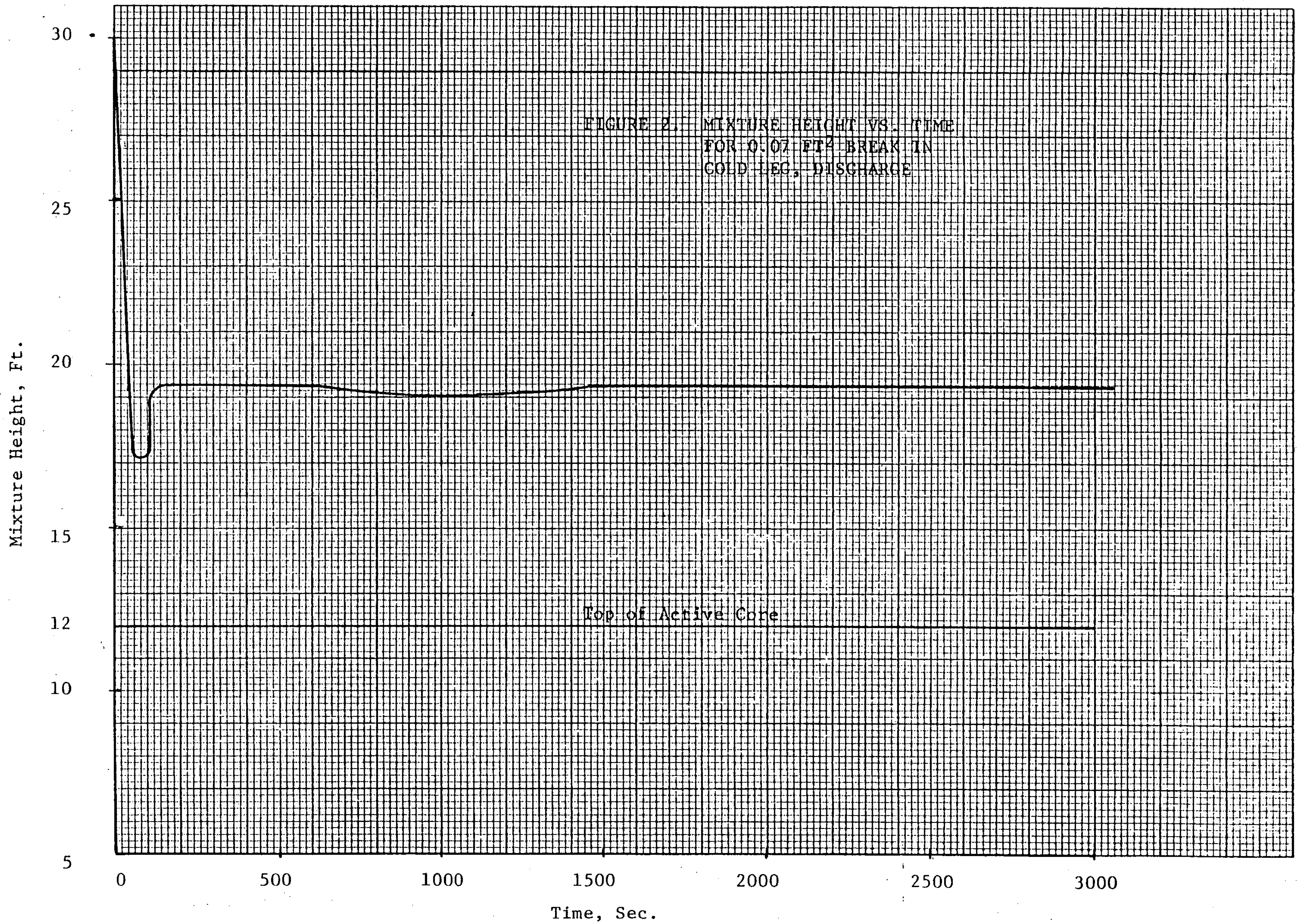
FIGURE 2. MIXTURE HEIGHT VS. TIME
FOR 0.07 FT^2 BREAK IN
COLD LEG, DISCHARGE
Top of Active Core
Mixture Height, Ft.
30
25
20
15
12
10
5
0
500
1000
1500
2000
2500
3000
Time, Sec.

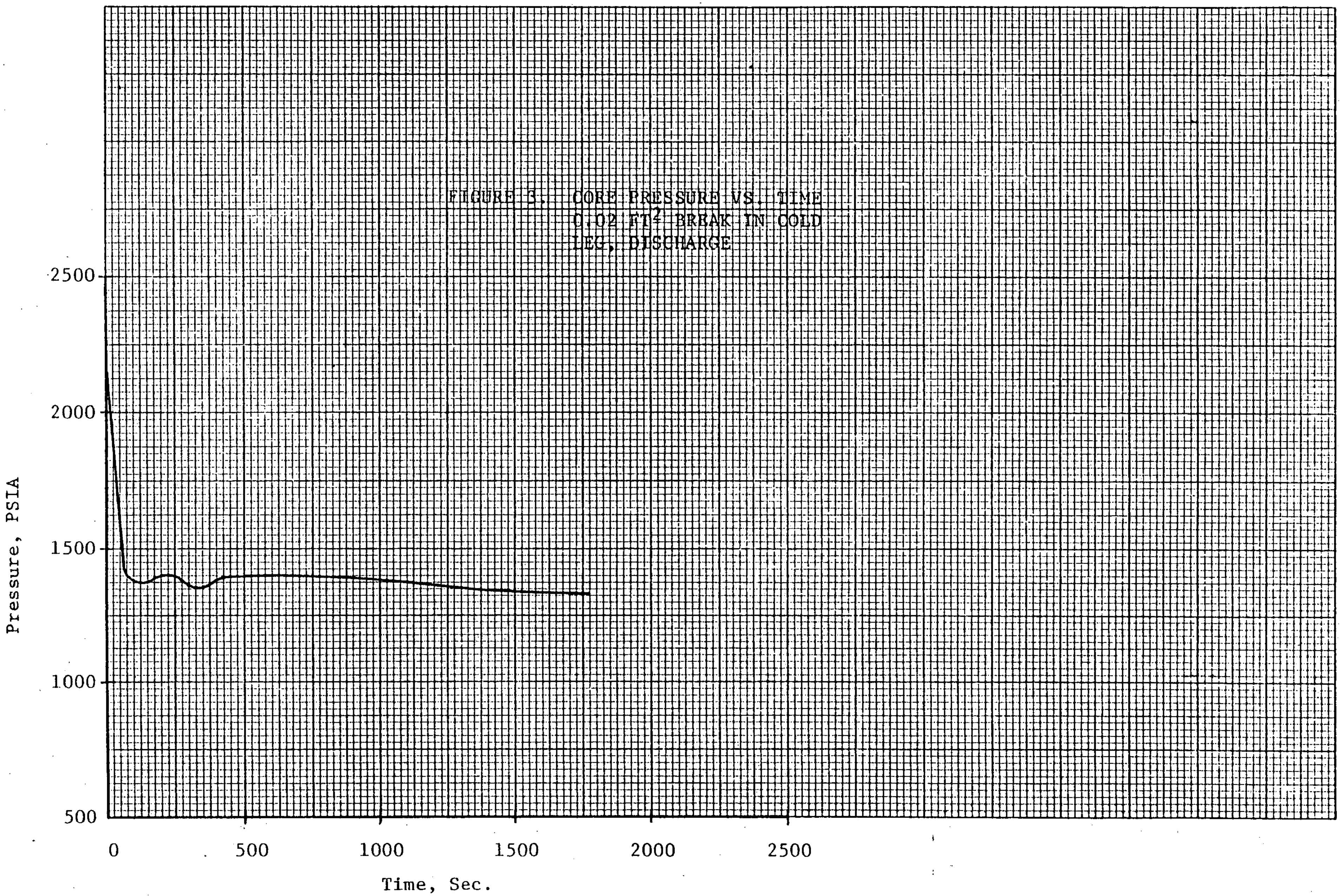

FIGURE 3. CORE PRESSURE VS. TIME
0.02 FT^2 BREAK IN COLD
LEG, DISCHARGE

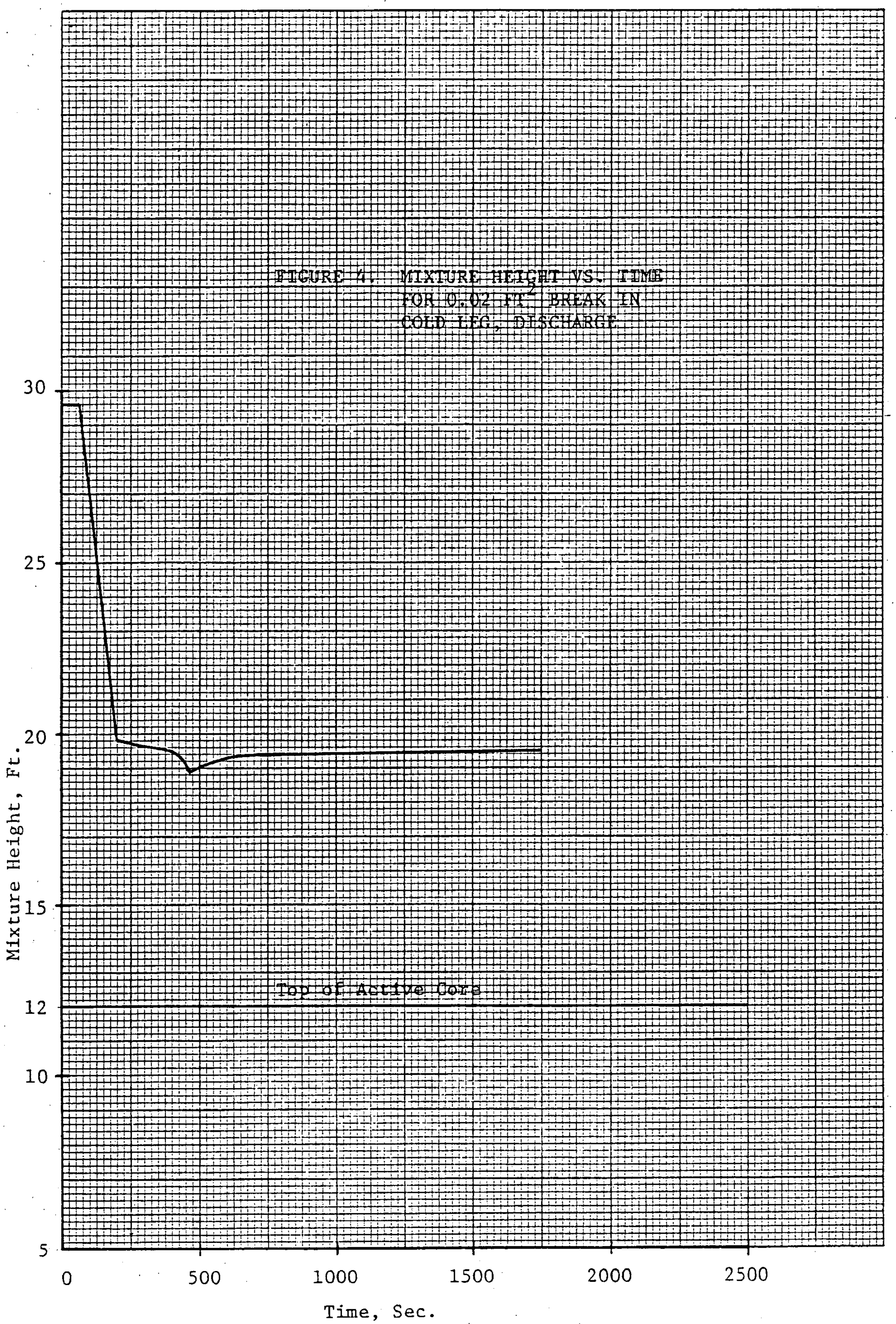

FIGURE 4. MIXTURE HEIGHT VS. TIME FOR 0.02 FT^2 BREAK IN COLD LEG, DISCHARGE

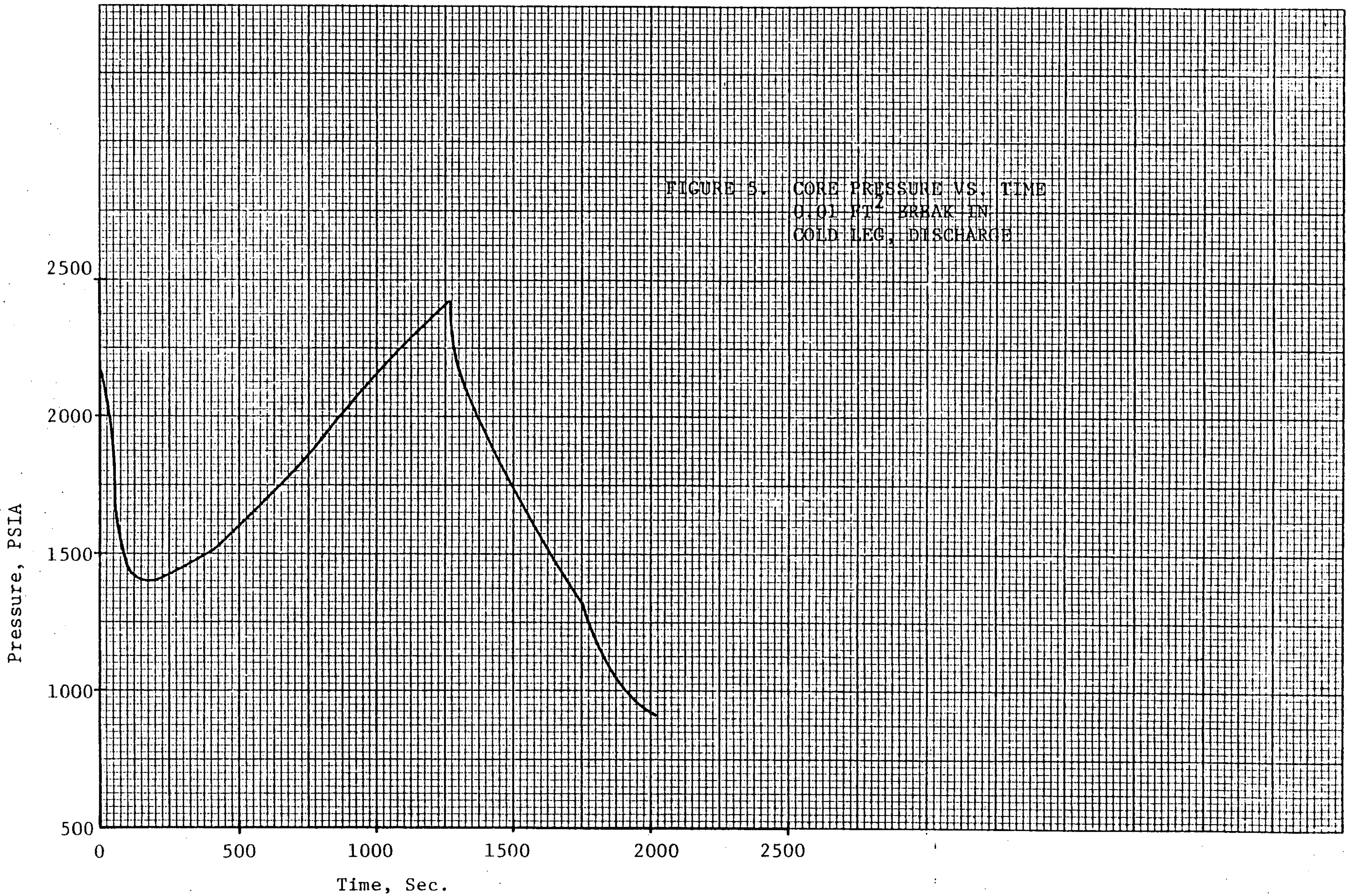

FIGURE 5. CORE PRESSURE VS. TIME
0.01 FT^2 BREAK IN
COLD LEG, DISCHARGE

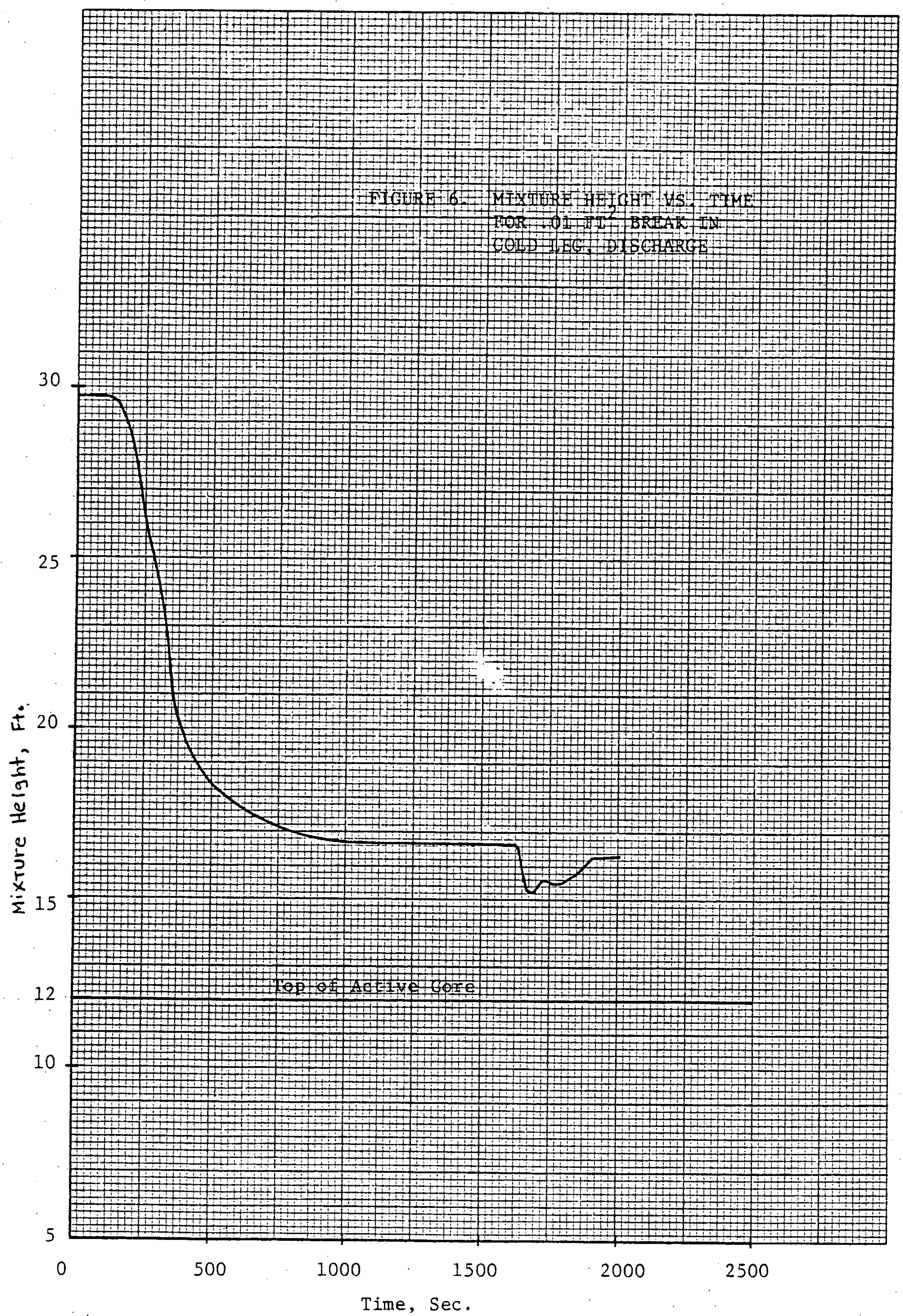

FIGURE 6. MIXTURE HEIGHT VS. TIME FOR .01 FT^2 BREAK IN COLD LEG, DISCHARGE