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STOLZ, J. F. Operating Reactors Branch 4

SUBJECT: Forwards evaluation of effect of low core flood tank boron concentrations on steam line break events. Rept provides supplemental info for review of proposed rev to Tech. Spec 3. 3. 3.

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November 22, 1985

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

Attention: Mr. J. F. Stolz, Chief
Operating Reactors Branch No. 4

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

Dear Sir:

Please find attached Duke's evaluation of the effect of low core flood tank (CFT) concentrations on steam line break events. The results of this evaluation were discussed during a conference call between the NRC staff and Duke Power Company on October 15, 1985.

The attached report is intended to provide supplemental information for staff's review of a proposed revision to the Oconee Nuclear Station Technical Specification 3.3.3 submitted by a letter dated September 12, 1984.

Also, as was discussed with the NRC staff on October 15, 1985, the CFT will be sampled for boron concentration determination following any change in the CFT inventory that may affect the CFT boron concentration to ensure the required minimum concentration is maintained.

Very truly yours,

Hal B. Tucker / ww

Hal B. Tucker

MAH:slb

Attachment

A001
1/1

cc: Dr. J. Nelson Grace, Regional Administrator
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EVALUATION OF THE EFFECT OF LOW CORE FLOOD TANK BORON CONCENTRATIONS ON STEAM LINE BREAK EVENTS

Duke Power Company

Oconee Nuclear Station

October 1985

I. INTRODUCTION

The effect of low core flood tank (CFT) boron concentrations on a steam line break event was analyzed. Currently, Oconee Technical Specification 3.3.3 requires that the boron concentrations in both core flood tanks exceed 1835 ppm. When deviations from the minimum required boron concentration occur, the current Technical Specification does not allow sufficient time to restore the boron concentration and has led to unit shutdown in the past. On March 1, 1984, Oconee 1 suffered such a forced outage lasting 23 hours. Certain design basis steam line break transients described in the Oconee FSAR take credit for the injected boron from the core flood tanks limiting a return to power following the break. Consequently, a more realistic steam line break analysis was performed to examine the sensitivity of the reactor shutdown margin to the CFT boron concentration. The results of this analysis justify continued operation for a limited period of time with one or both of the core flood tanks below the Technical Specification minimum boron concentration.

II. DESCRIPTION OF ANALYSIS

The RETRAN02-MOD003 computer code (Reference 1) was used to analyze the thermal-hydraulic response of an Oconee unit to a double-ended guillotine break of a main steam line. The Reactor Coolant System (RCS) was modeled with the two loop, four cold leg nodalization shown on Figure 1. A best estimate approach was taken in general; however, conservative assumptions were made in some areas. A list of important assumptions is given in Table 1.

The base case considered the core flood tanks to be at nominal pressure and level conditions with the Technical Specification minimum boron concentration of 1835 ppm. The second case was a sensitivity study which also assumed nominal pressure and level in the tanks, but the initial boron concentration in both tanks was set at 0 ppm. Therefore this represents the worst possible CFT conditions for the bounding steam line break transient. The sensitivity study was a hand calculation based on the flows calculated in the base case. The resulting RCS boron concentration was calculated assuming homogeneous mixing in the primary system, which is valid for the high flow rates seen in this transient.

III. RESULTS

The thermal-hydraulic response of both cases was identical, since no return to power occurs. The primary system cooldown resulting from a double-ended steam line break is severe, but there is no return to power or core uncovery and therefore no fuel damage calculated for this transient. The cooldown and contraction of the primary coolant following the break leads to a rapid reactor trip and subsequent High Pressure Injection System actuation. Thirty seconds following loss of primary system subcooled margin the operators trip all four reactor coolant pumps per the Emergency Operating Procedure. Maximum main feedwater injection to the affected (A) steam generator continues until operator action to isolate it occurs at five minutes. Operator action to fill the intact (B) steam generator to 95% following loss of subcooled margin also occurs. The blowdown of the affected steam generator is complete one minute after it is isolated. Pressurizer level is back onscale soon thereafter and the transient is effectively terminated. A sequence of events is included in Table 2, and plots of pertinent parameters are shown on Figures 2-10.

The maximum post-trip k-effective is 0.995 at 253 seconds for both cases. At that time an increase in the intact loop flow turns around the core temperature decrease. The core flood tanks begin to deliver flow at 256 seconds, and the RCS boron concentration begins to differ between the two cases at that point. CFT injection ends at 332 seconds with a total of 242 ft^3 being delivered to the RCS from the tanks. Figures 11 and 12 show the RCS boron concentration and core k-effective for both cases. Even with the large reactivity penalty taken for the worst case stuck rod, and the unrealistically low CFT boron concentration in Case 2, no return to critically occurs.

IV. CONCLUSION

No return to power is calculated following a double-ended steam line break with a boundingly conservative assumption for CFT boron concentration. In addition, the five minute time assumed for feedwater isolation by the operators is very conservative in light of the clear and unambiguous indications of a steam line break and explicit procedural guidance to isolate the affected generator. The 1.76% $\Delta k/k$ stuck rod penalty provides added assurance core recriticality would not occur.

This analysis is not intended to supplant the existing FSAR design basis. It is intended to provide further assurance that the realistic result of a worst case steam line break would be acceptable even with the lowest possible boron concentration in the core flood tanks. Should another instance of low CFT boron concentration arise it would be very preferable to maintain the plant in a stable condition and correct the problem, as opposed to inducing a transient (reactor shutdown) to correct a situation which poses no real threat to the health and safety of the public. Thus, a Technical Specification change which would allow a degraded mode for core flood tank boron concentration is justified.

V. REFERENCES

1. Cecil O. Thomas (NRC--Division of Licensing) to Dr. Thomas W. Schnatz (Utility Group for Regulatory Application); Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, "RETRAN--A Program for One Dimensional Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," and EPRI NP-1850-CCM, "RETRAN-02--A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," September 4, 1984.

2. Oconee Nuclear Station Final Safety Analysis Report, Chapter 15, Section 13.

3. Oconee Nuclear Station Technical Specifications, Section 3.3.3.

TABLE 1

Important Assumptions in ONS Steam Line Break Analysis

- Initial core power = 100% (2568 MW thermal)
- Nominal full power initial steam generator inventory
- Nominal end of cycle kinetics parameters and control rod worths
- Maximum worth control rod (1.7% $\Delta k/k$) remains stuck out of the core following reactor trip **
- Moderator temperature coefficient = -3.0×10^{-4} $\Delta k/k/°F$, as assumed in the FSAR
- Double-ended guillotine break of steam line A modeled with Moody choked flow and a discharge coefficient of 1.0 **
- No main feedwater pump trip following break **
- Full decay heat based on 1971 ANS Standard
- Automatic main feedwater realignment to emergency feedwater header following RCP trip
- Full HPI flow available following Engineered Safeguards actuation
- 1835 ppm boron concentration in borated water storage tank **
- Homogeneous mixing of injected boron
- Boron reactivity worth = 110 ppm per % $\Delta k/k$ (HFP value) **
- Operator action to

    1) trip all reactor coolant pumps 30 seconds after loss on indicated subcooled margin **

    2) isolate the broken steam generator five minutes after the initiating event **

** Denotes conservative assumption

TABLE 2

Steam Line Break Sequence of Events

Time (sec)	Event
0	Double-ended break of steam line A
4.72	Reactor trip on variable low pressure
4.77	Turbine trip on reactor trip
19	Engineered Safeguards channels 1 and 2 actuate on RCS pressure $\leq$ 1600 psig
22	Pressurizer level offscale low
29	High Pressure Injection flow begins following ES 1 and 2
34	Loss of indicated subcooled margin
64	Operators trip the reactor coolant pumps due to loss of subcooled margin;
	MFW flow redirected to EFW header in both steam generators
253	Maximum post-trip k-effective = 0.995
256	Core flood tanks begin to deliver flow to the RCS
300	Operators isolate feedwater to SG A
332	Core flood tank injection ends
334	Minimum RCS pressure = 443 psig
356	SG A blowdown complete (SG A pressure < 1 psig)
370	Indicated subcooled margin regained
374	Pressurizer level back onscale
420	End of analysis

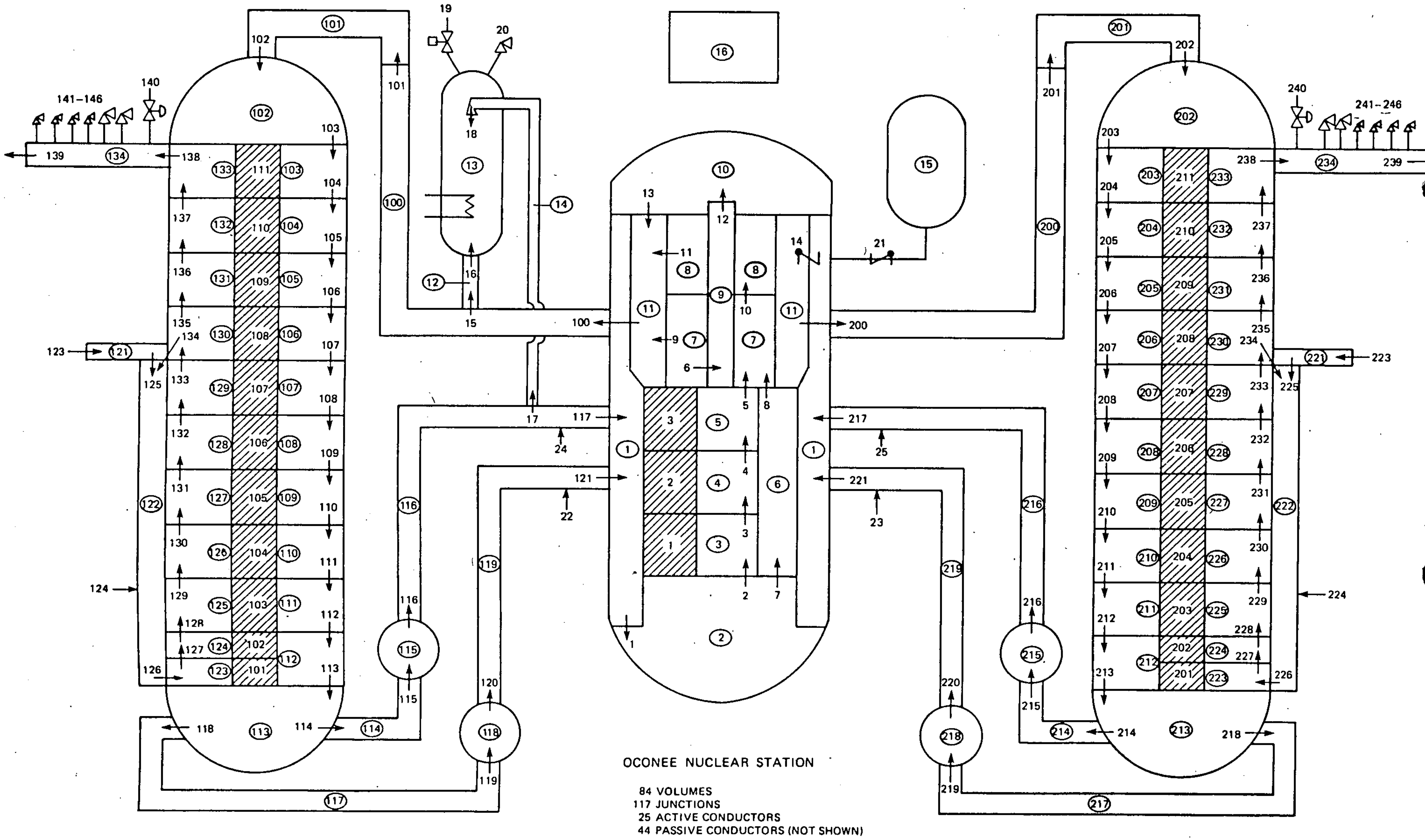
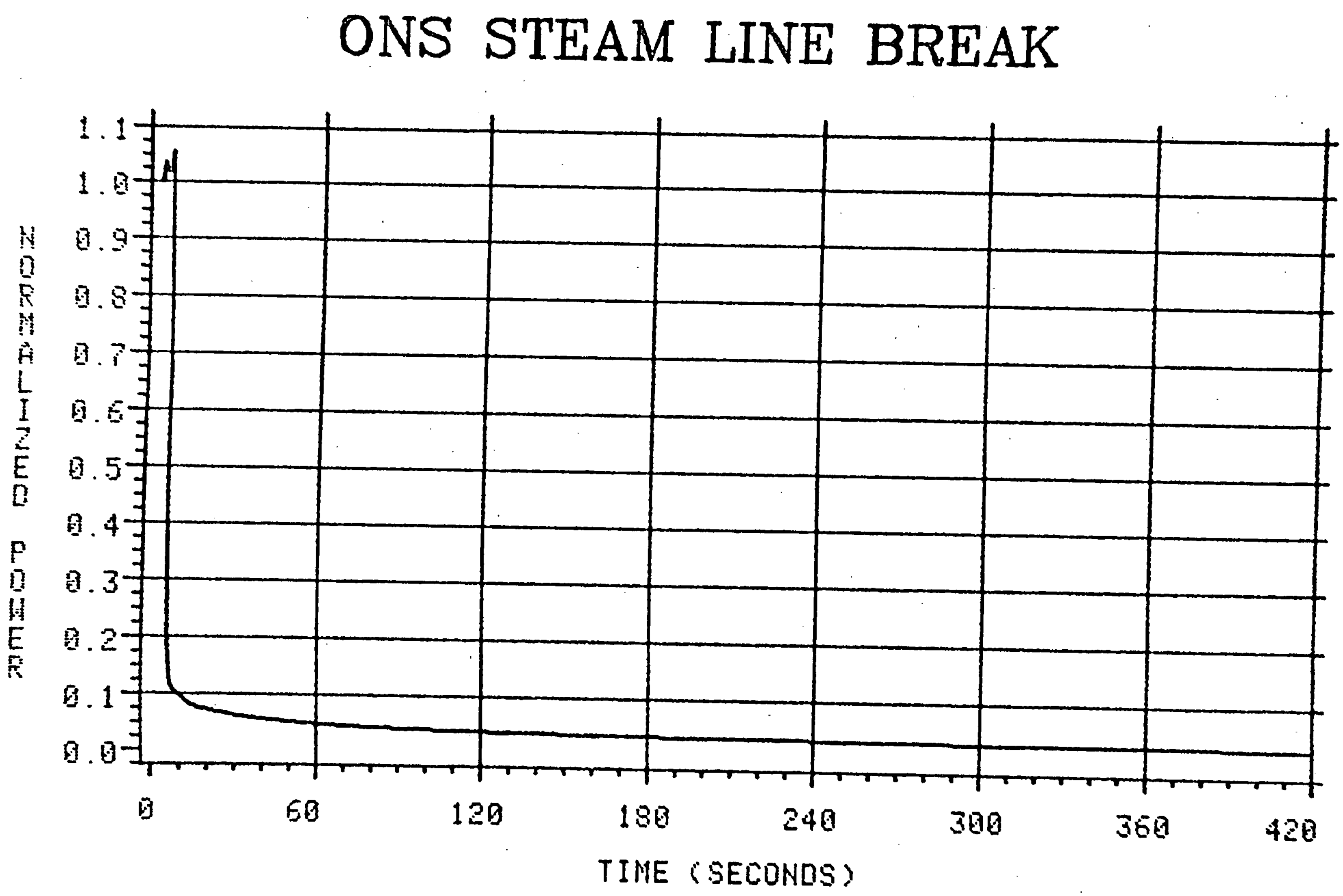

OCONEE NUCLEAR STATION
84 VOLUMES
117 JUNCTIONS
25 ACTIVE CONDUCTORS
44 PASSIVE CONDUCTORS (NOT SHOWN)

FIGURE 1



ONS STEAM LINE BREAK
NORMALIZED POWER
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0.0
0
60
120
180
240
300
360
420
TIME (SECONDS)

FIGURE 2

ONS STEAM LINE BREAK

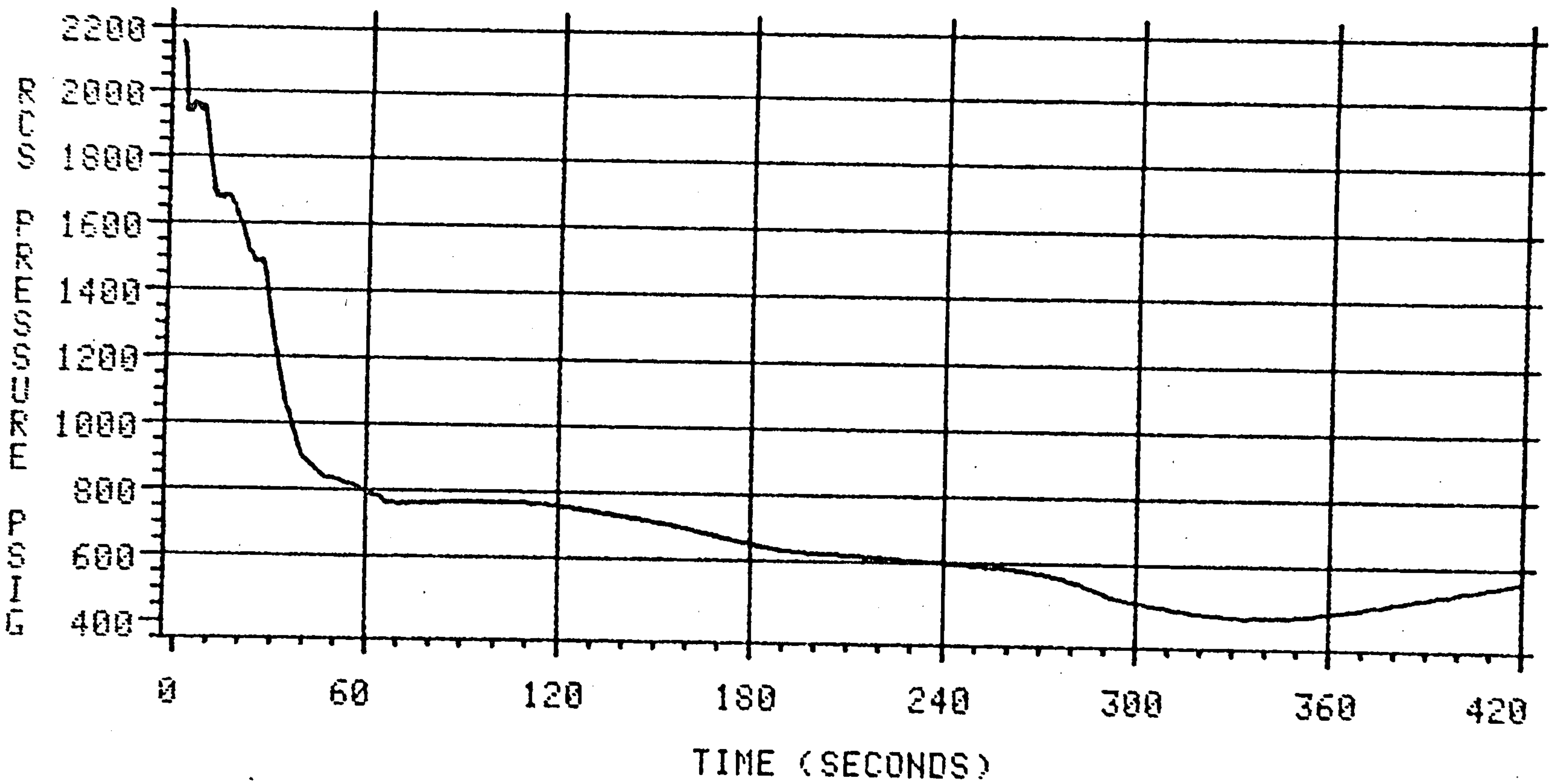


FIGURE 3

ONS STEAM LINE BREAK

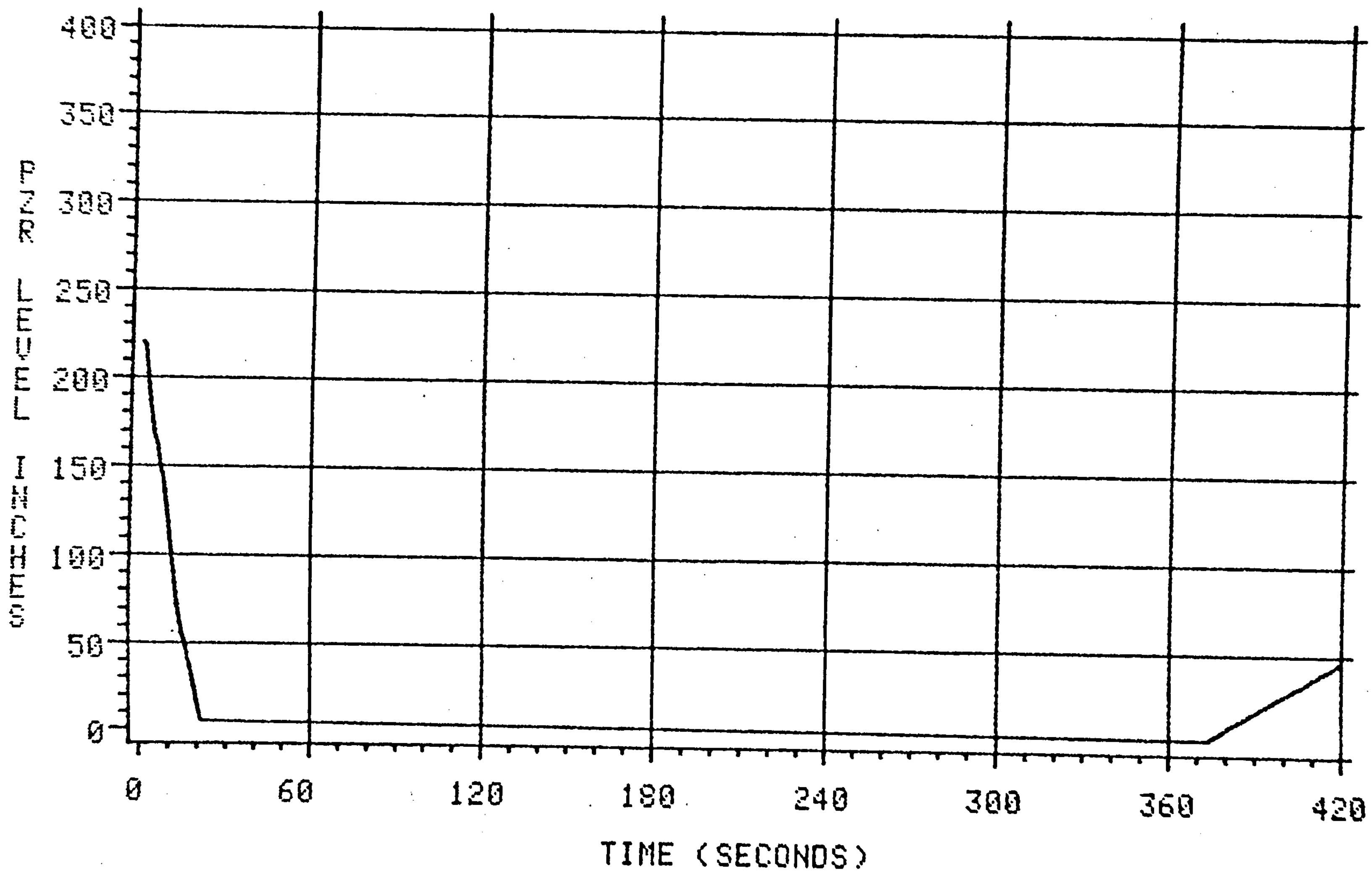
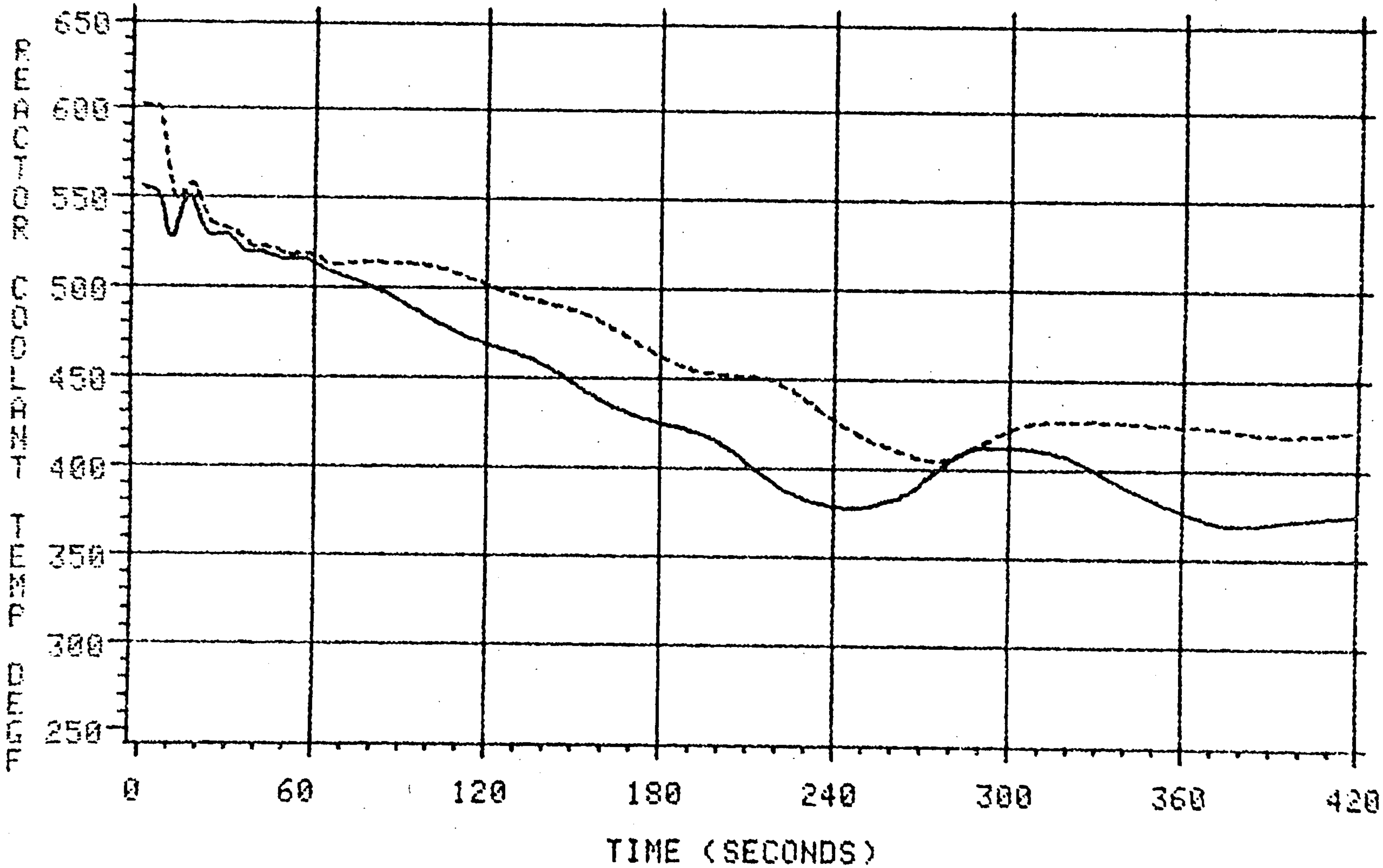


FIGURE 4

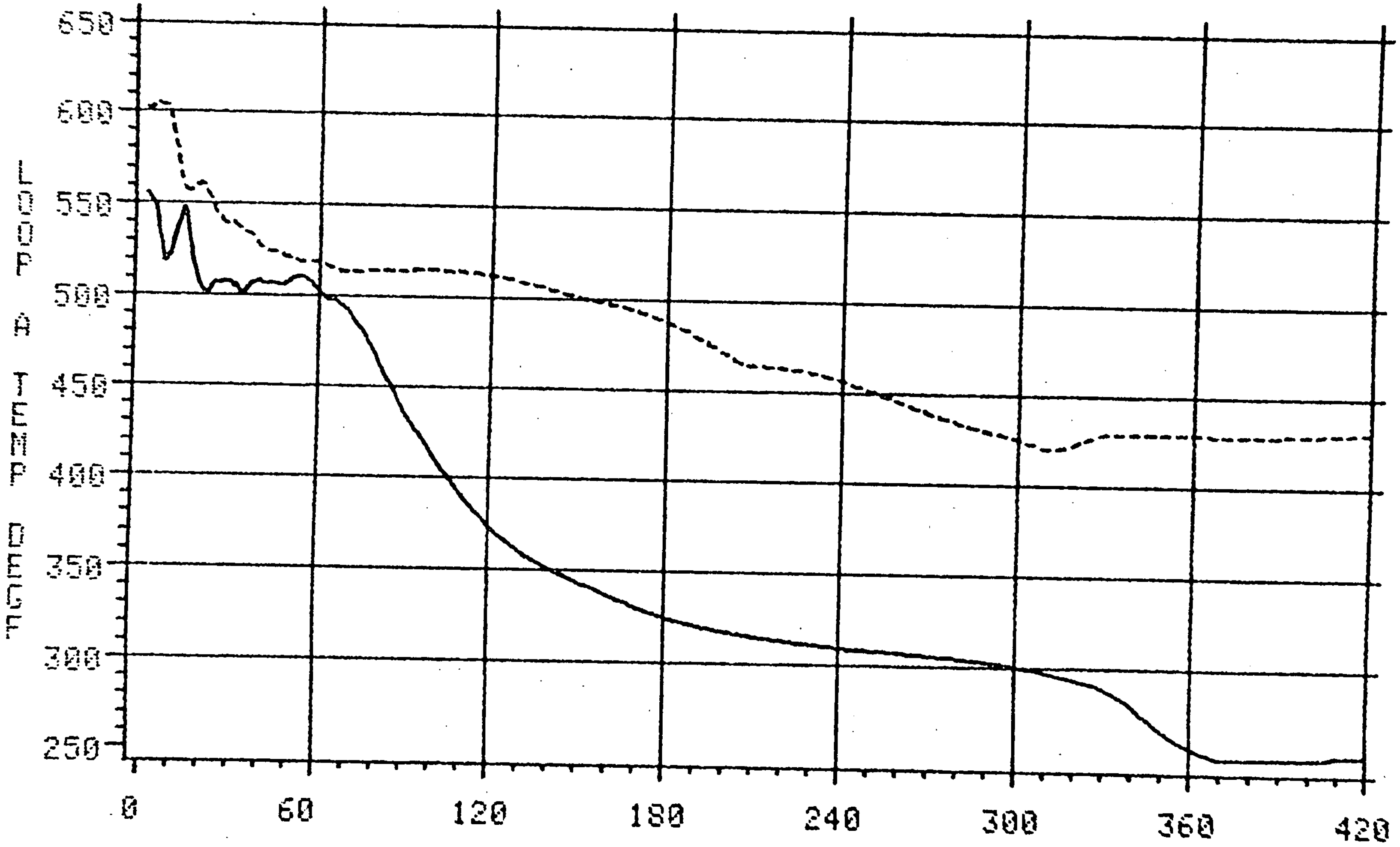
ONS STEAM LINE BREAK



SOLID LINE CORE INLET DASHED LINE CORE EXIT

FIGURE 5

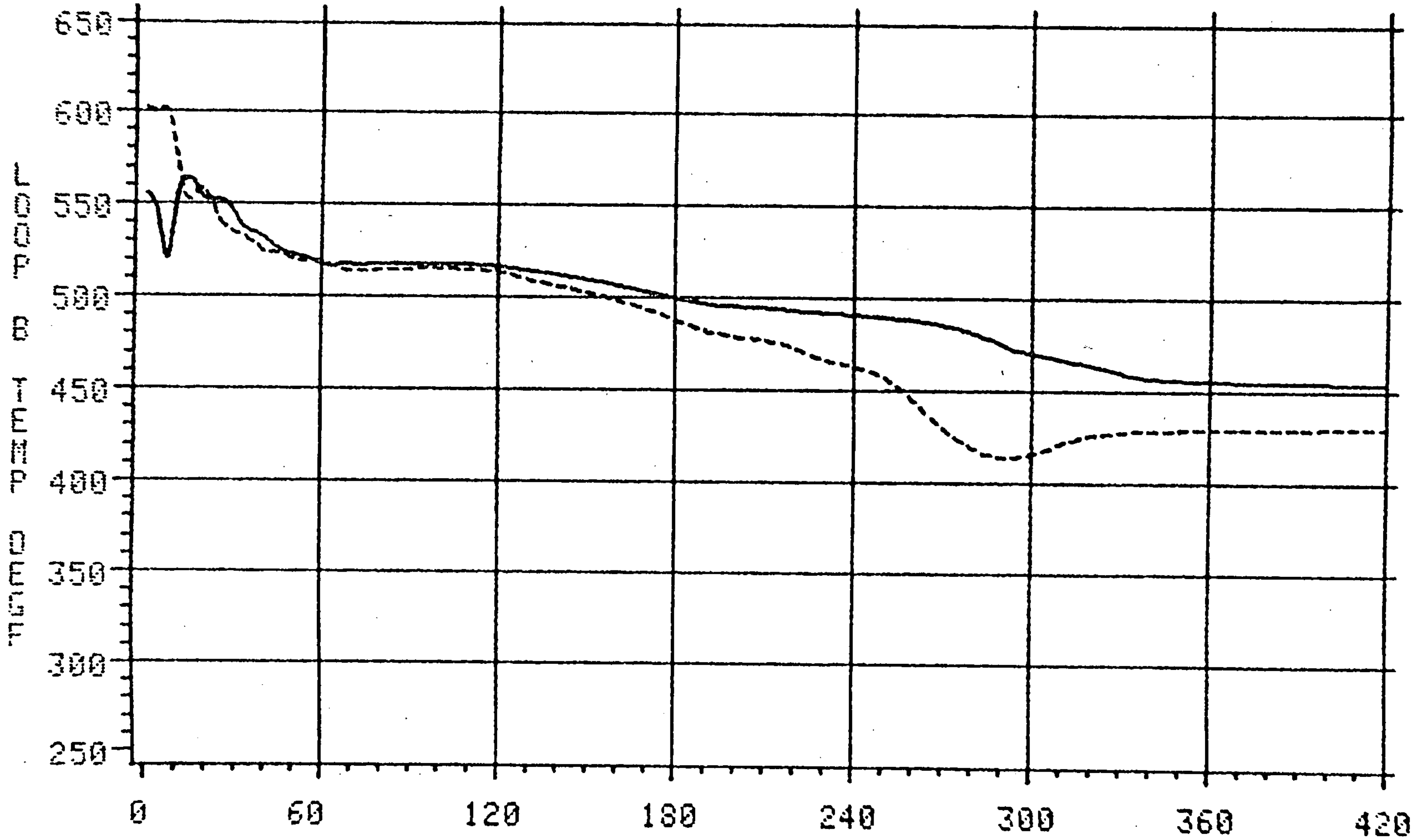
ONS STEAM LINE BREAK



SOLID LINE COLD LEG DASHED LINE HOT LEG

FIGURE 6

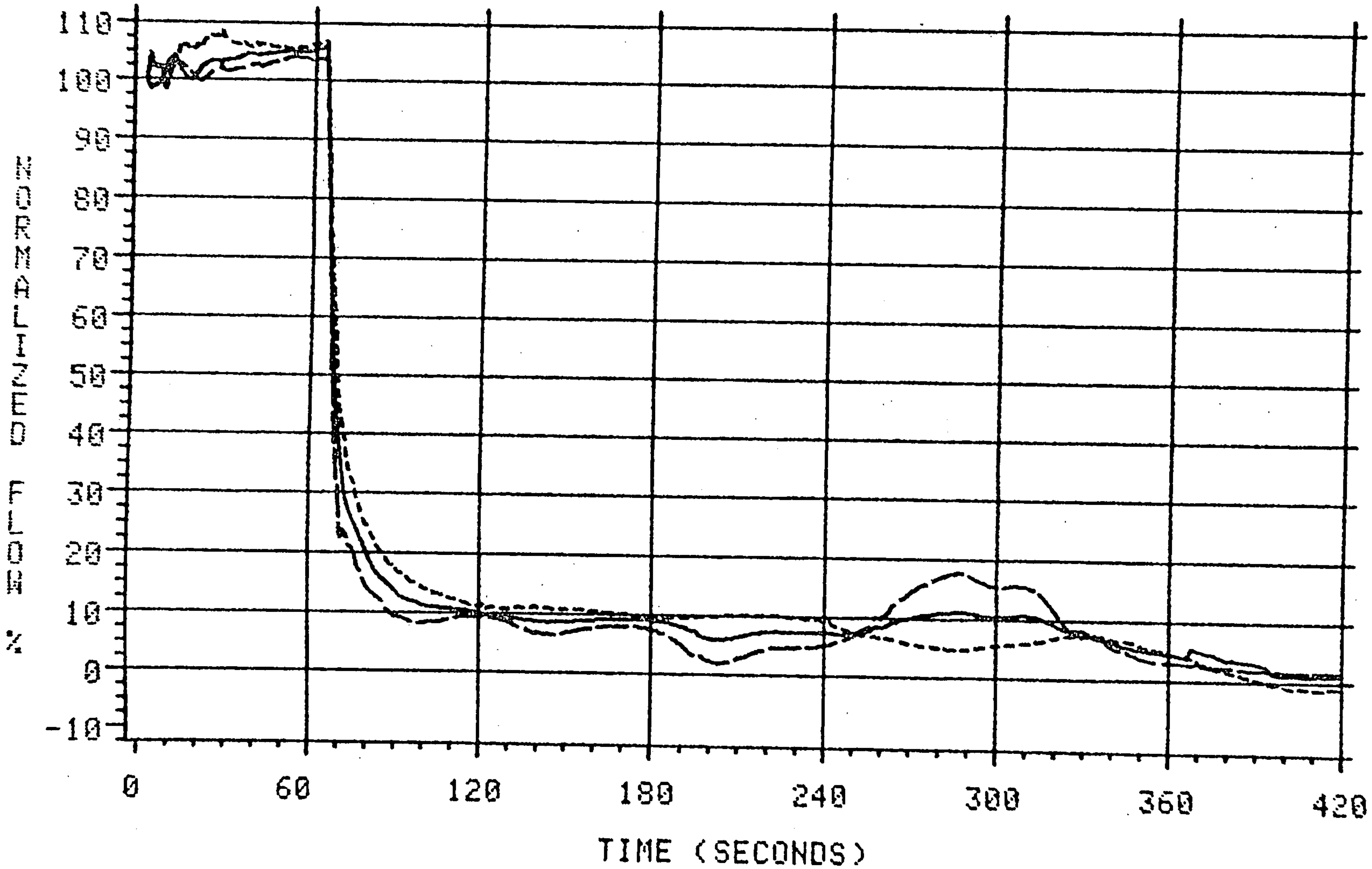
ONS STEAM LINE BREAK



SOLID LINE COLD LEG   DASHED LINE HOT LEG

FIGURE 7

ONS STEAM LINE BREAK



SOLID LINE CORE DASHED LINE LOOP A WIDE DASHED LINE LOOP B

FIGURE 8

ONS STEAM LINE BREAK

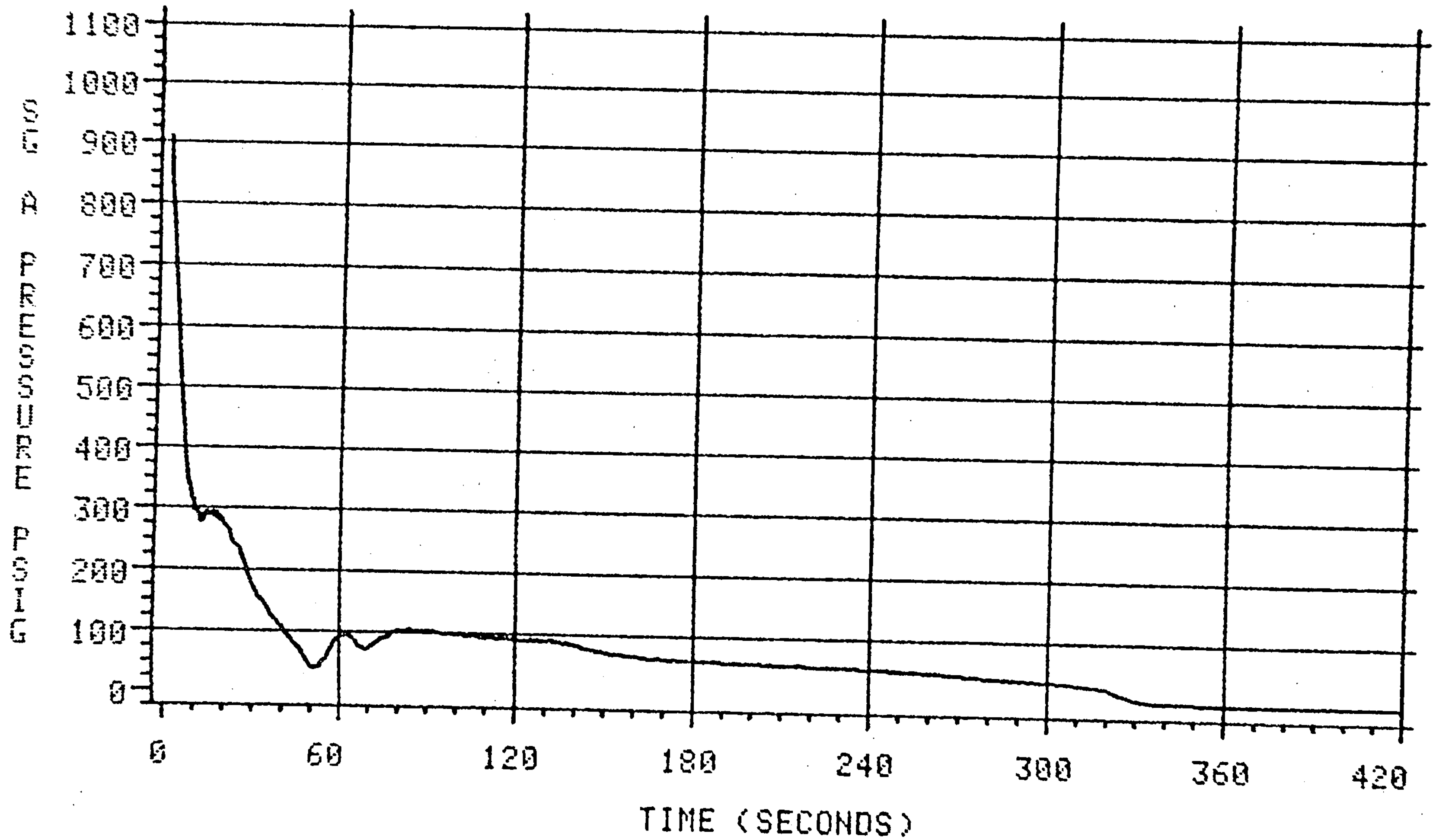


FIGURE 9

ONS STEAM LINE BREAK

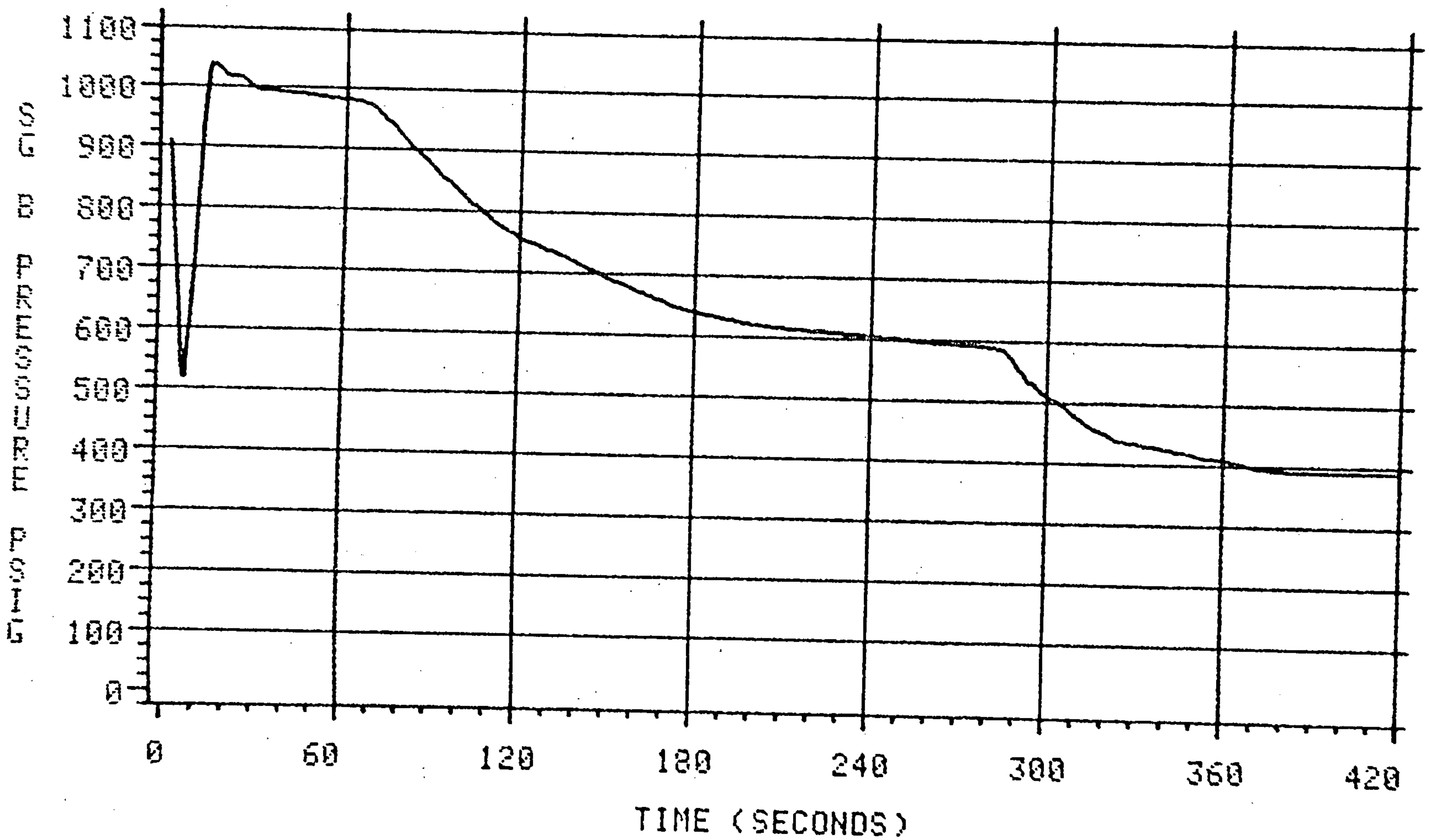


FIGURE 10

ONS STEAM LINE BREAK

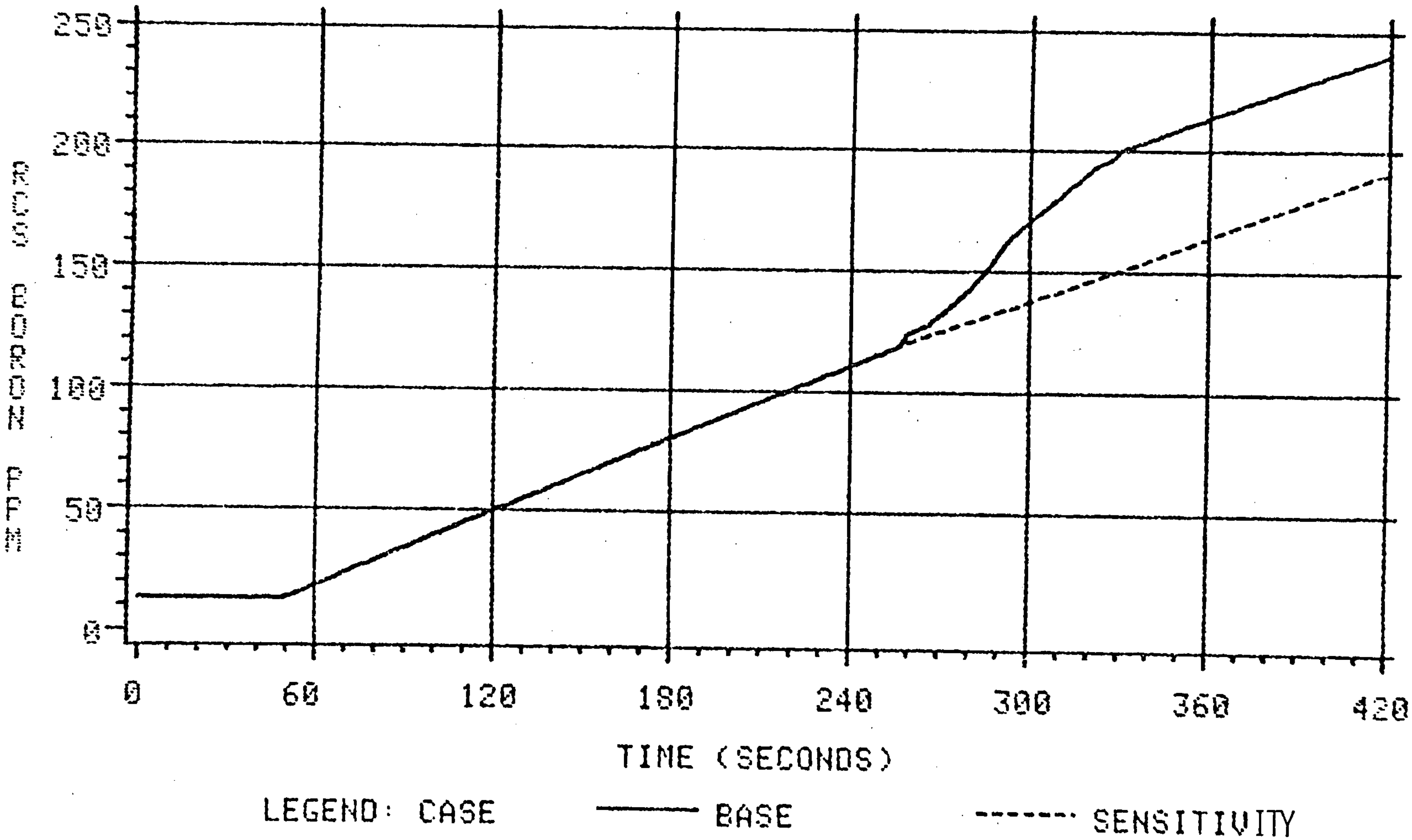


FIGURE 11

ONS STEAM LINE BREAK

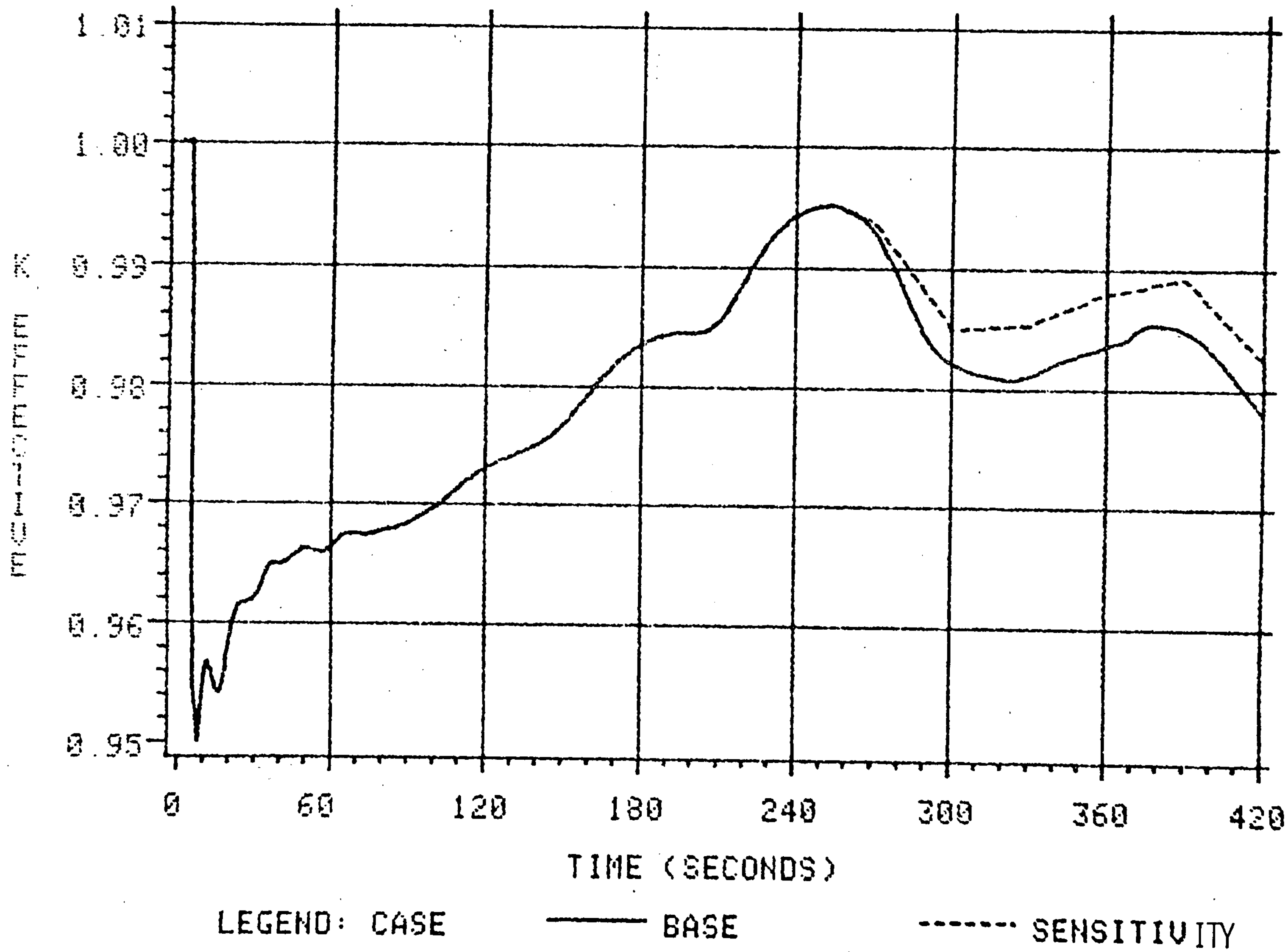


FIGURE 12