"EVALUATION OF TRANSIENT BEHAVIOR AND
SMALL REACTOR COOLANT SYSTEM BREAKS
IN THE 177 FUEL ASSEMBLY PLANT"

VOLUME 1 SECTION 6.0 - SUPPLEMENT 1

MAY 12, 1979

"SMALL BREAK IN THE PRESSURIZER (PORV)
WITH NO AUXILIARY FEEDWATER AND SINGLE
FAILURE OF THE ECCS"

Small Break in the Pressurizer (PORV) With No Auxiliary Feedwater and a Single Failure of the ECCS

1. Introduction

The evaluation of small breaks in the pressurizer in Chapter 6.2.3 of the main report covered the following two cases:

1. Loss of main feedwater resulting in a stuck open PORV, no loss of offsite power, and one HPI train available for emergency core cooling.
2. Stuck open PORV with loss of offsite power and one HPI train.

This supplement provides the additional analysis of a small break of the PORV without feedwater availability and a single failure in the emergency core cooling system. For this assumption, the 177 fuel assembly lowered-loop plants fall into two categories:

1. Oconee will have one HPI pump injecting through one train for 10 minutes and two HPI pumps injecting through two trains thereafter.
2. All others will have one HPI pump injecting through one train for 10 minutes and one HPI pump injecting through two trains thereafter.

This evaluation is based on the availability of one HPI. Consideration of the results for Oconee is given in Section 4.

This evaluation was performed with the conservative assumption that decay heat equals 120% of the ANS 5.1 (1971) values. The evaluation shows no core uncover and no cladding temperature excursion for a period of 30 minutes. Extrapolation of these results by hand calculation lead us to expect core uncovery by 40 minutes if no further action is taken. Operator action to start feedwater or initiate a second HPI train by 30 minutes would prevent core uncovery and assure compliance to the criteria of 10 CFR 50.46.

2. Method of Analysis

The analysis method used for the evaluation is that described in Chapter 5 of reference 4, BAW-10104, Rev. 3, "B&W's ECCS Evaluation Model," along with the model modifications of reference 5. As dictated by reference 6, the Bernoulli correlation was used for subcooled flow rather than the modified Zaloudek correlation as proposed in reference 5. The following conditions and system responses were assumed during the transient:

a. The reactor is operating at 102% of a steady-state power level of 2772 MWt. Decay heat is based on 1.2 times the 1971 ANS 5.1 standard for infinite reactor operation.

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

c. No offsite power is available.

d. The reactor trips on low pressure (1900 psig).

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

f. The RC pumps trip and coastdown coincident with reactor trip.

g. One complete train of the emergency safeguards system fails to operate, leaving two CFTs and only one HPI and one LPI system available for pumped injection to mitigate the consequences of the LOCA.

h. The auxiliary feedwater (FW) system is assumed not to be available during the transient.

i. The ESFAS trip, including signal errors, occurs at a RC pressure of 1415 psia.

30 minutes core boil-off is exceeded by HPI injection and the RCS loops still contain significant liquid inventory at elevations above the top of the core. Actuation of feedwater to control and decrease the RCS pressure at this time will maintain that balance and prevent uncovery.

3. Results

Figures 2 through 15 show the evolution of several key reactor coolant system variables for this event. The CRAFT2 noding diagram for this analysis is presented in Figure 1. The following table presents key results of the analysis:

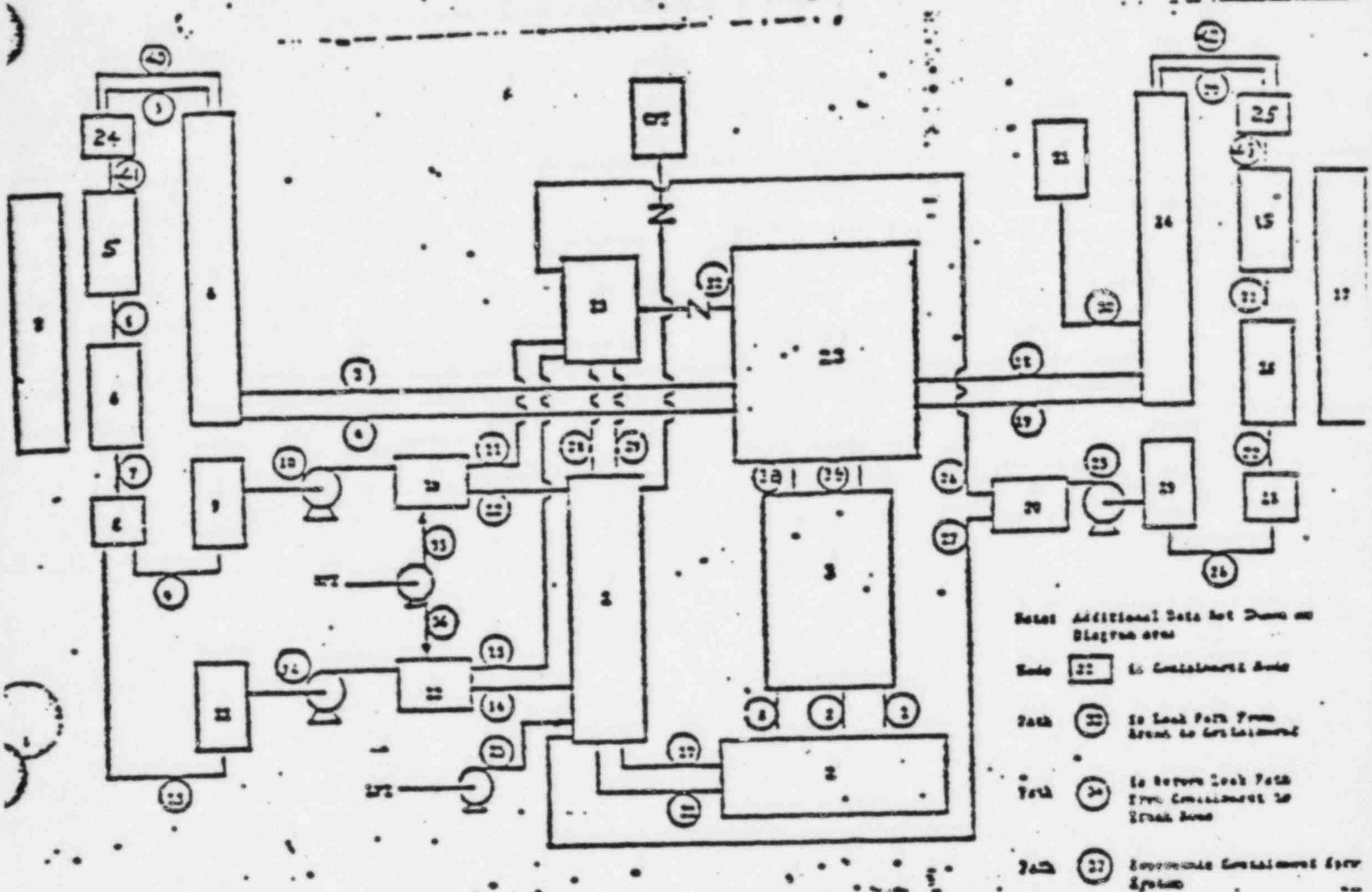
Sequence of events	Time, s
Break occurs (1.05 in.^{2} @ top of pressurizer equal to a PORV break)	0.0
Reactor trip	60.0
Loss of offsite power, RC pumps coastdown occurs	60.0
Main feedwater coastdown ends	74.0
HPI injection starts	250.0
Natural circulation essentially lost	1440.0
Peak cladding temperature	720F (initial value)

The RCS depressurizes with the initiation of the break as shown in Figure 2. Since the break is in the steam space of the pressurizer, the RCS depressurizes rapidly to the ESFAS setpoint, thus initiating the HPI injection. At 600 seconds into the transient, the RCS repressurized because the pressurizer goes solid as shown in Figure 3. As shown in Figure 4, the SG secondary side liquid volume goes dry (approximately 570 sec) just prior to the pressurizer going solid. The reduced primary to secondary heat transfer also precipitates the increase in RCS pressure. Extrapolation of the results at 30 minutes to extended times shows that the system will eventually repressurize to 2500 psig (code safety setpoint). At this pressure HPI injection will not match core boiling until approximately 10,000 seconds. Therefore, unless further action is taken, core uncovery will occur. Operator action to initiate feedwater at 30 minutes or to establish a second HPI pump will prevent core uncovery. At

4. Small Break in Oconee Pressurizer Steam Space Without Feedwater

The discussion of system response in sections 2 and 3 was based on one operating HPI pump. Even with a single active failure, Oconee will have two pumps available within 10 minutes following ESFAS actuation. The Oconee HPI pumps are approximately 10% lower in capacity than the generic HPI capacity used in sections 2 and 3. For the first 250 seconds up to ESFAS action system response will be identical. Following that, for 10 minutes, the Oconee system liquid inventory will lag the generic calculation slightly (maximum effect - 70 ft^3 at 950 seconds). At 950 seconds Oconee will establish two HPI pumps or about 180% of the injection used in the generic evaluation. Therefore, Oconee will achieve long term cooling much earlier than the generic evaluation and the remainder of that evaluation will bound the results for Oconee.

Figure 1. System Noding



Node No.	Identification
1	Downcomer
2	Lower Plenum
3	Core
4,14	Hot Leg Piping
5,15	SG & Upper Head
6,16	Steam Generator Tubes
7,17	Secondary, SG
8,18	SG Lower Head
9,11,19	Cold Leg Piping
10,12,20	Cold Leg Piping
13	Upper Downcomer
21	Pressurizer
22	Containment
23	Upper Plenum
24,25	SG Upper Head

Path No.	Identification
1,2	Core
3,4,18,19	Hot Leg Piping
5,20,40,42	Hot Leg, Upper
6,21	SG Tubes
7,22	SG Lower Head
8	Core Bypass
9,13,24	Cold Leg Piping
10,14,25	Pumps
11,12,15,16,26,27	Cold Leg Piping
17,31	Downcomer
23	LPI
28,29	Upper Downcomer
30	Pressurizer
32	Vent Valve
33,34	Leak & Return Path
35,36	HPI
37	Containment Sprays
41,43	SG Upper Head

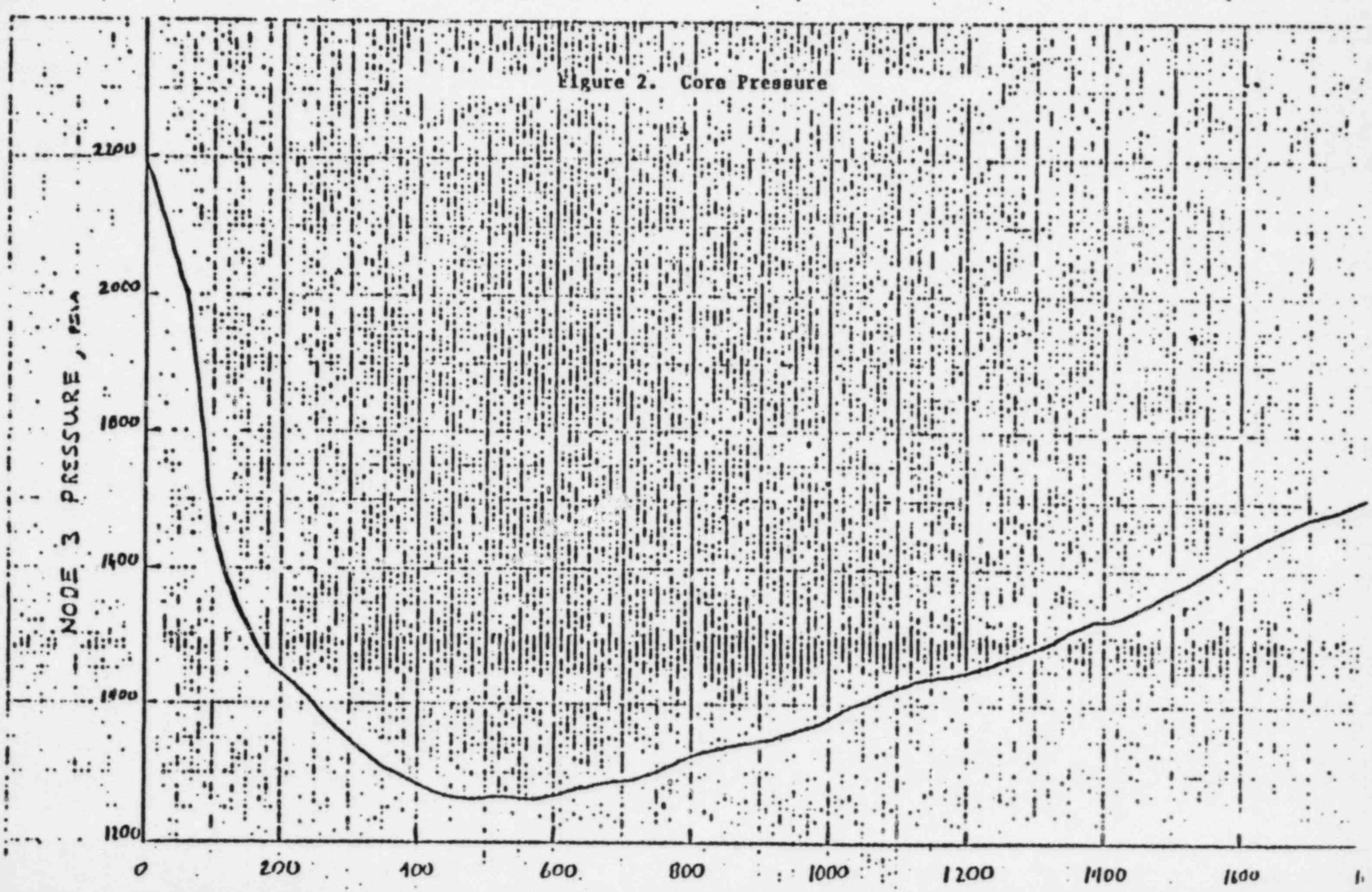


Figure 2. Core Pressure

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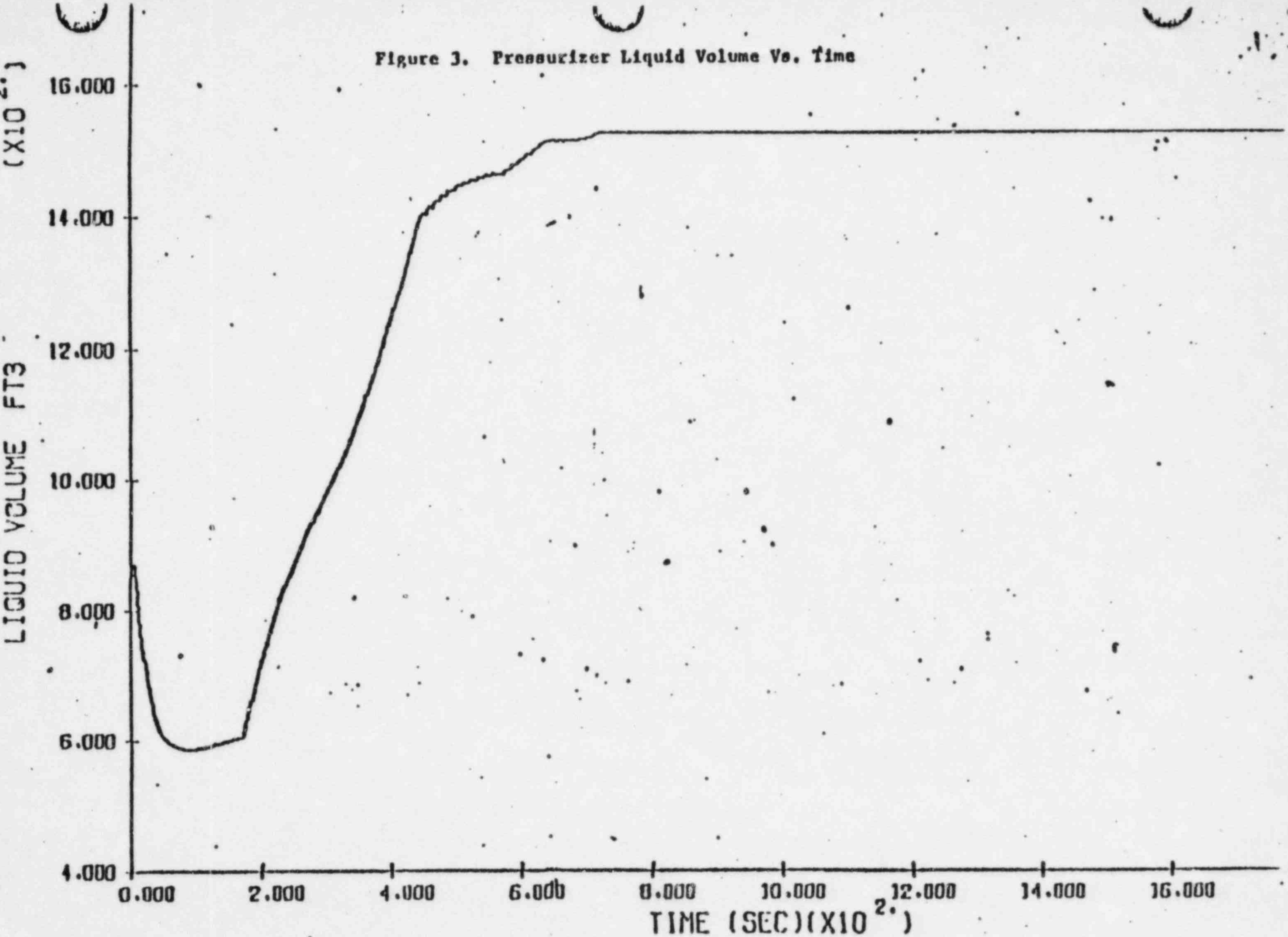
Figure 3. Pressurizer Liquid Volume Vs. Time



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P21410 NAF 1415HP 1.2ANS

NODE 21

Figure 4. Steam Generator Liquid Volume Vs. Time
(SECONDARY SIDE)

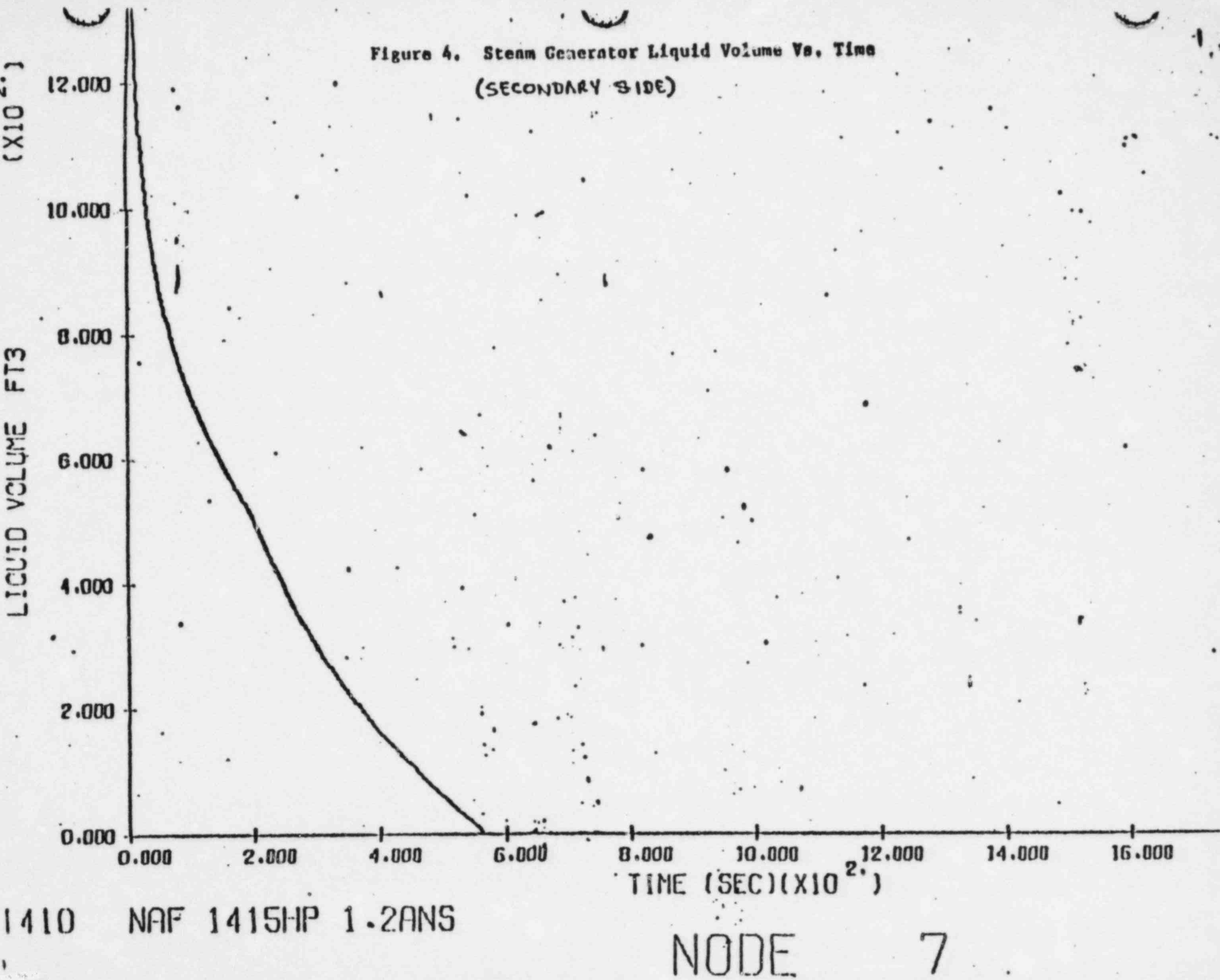
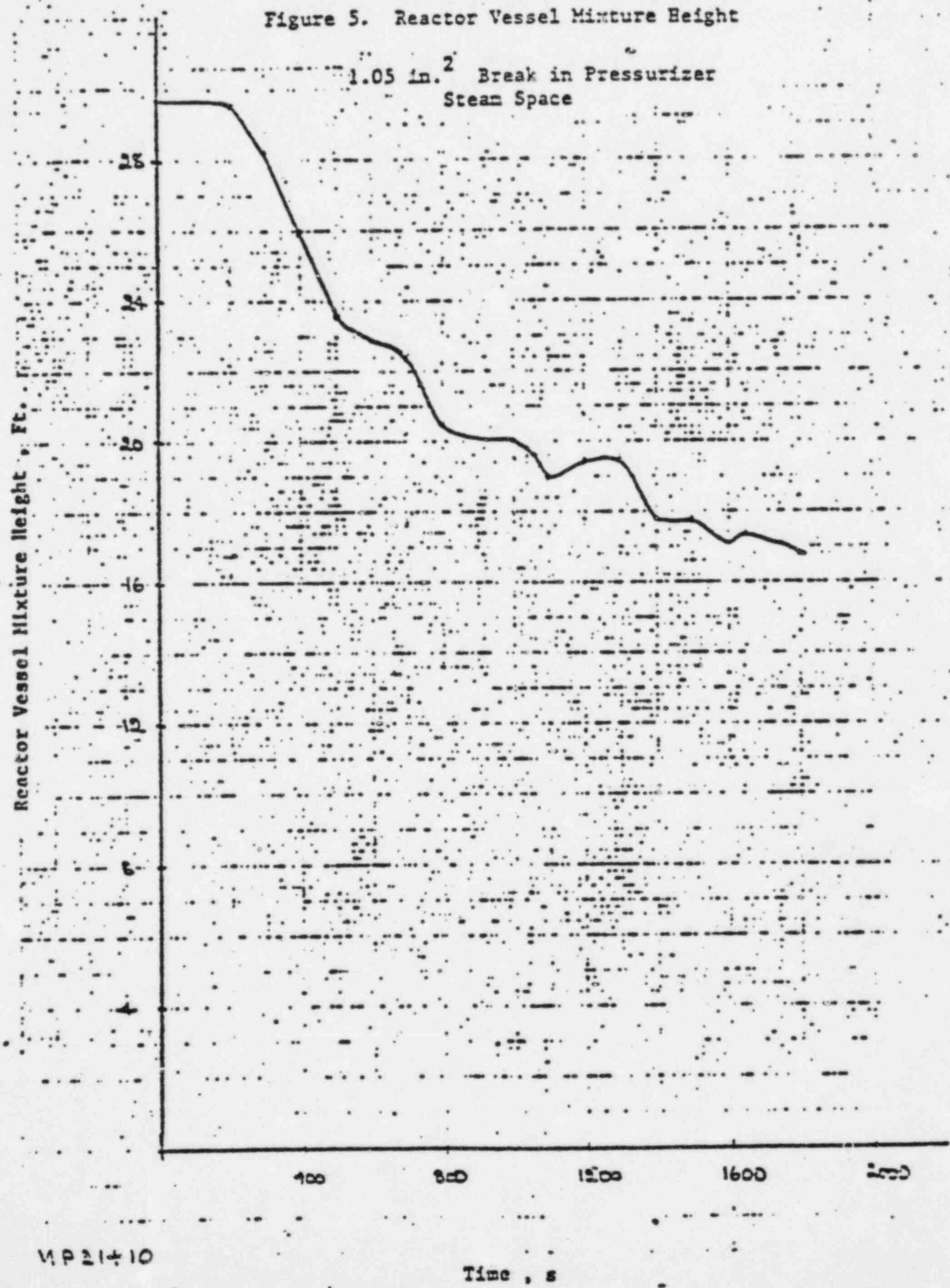


Figure 5. Reactor Vessel Mixture Height

$1.05\ in.^2$ Break in Pressurizer Steam Space



MP21+10

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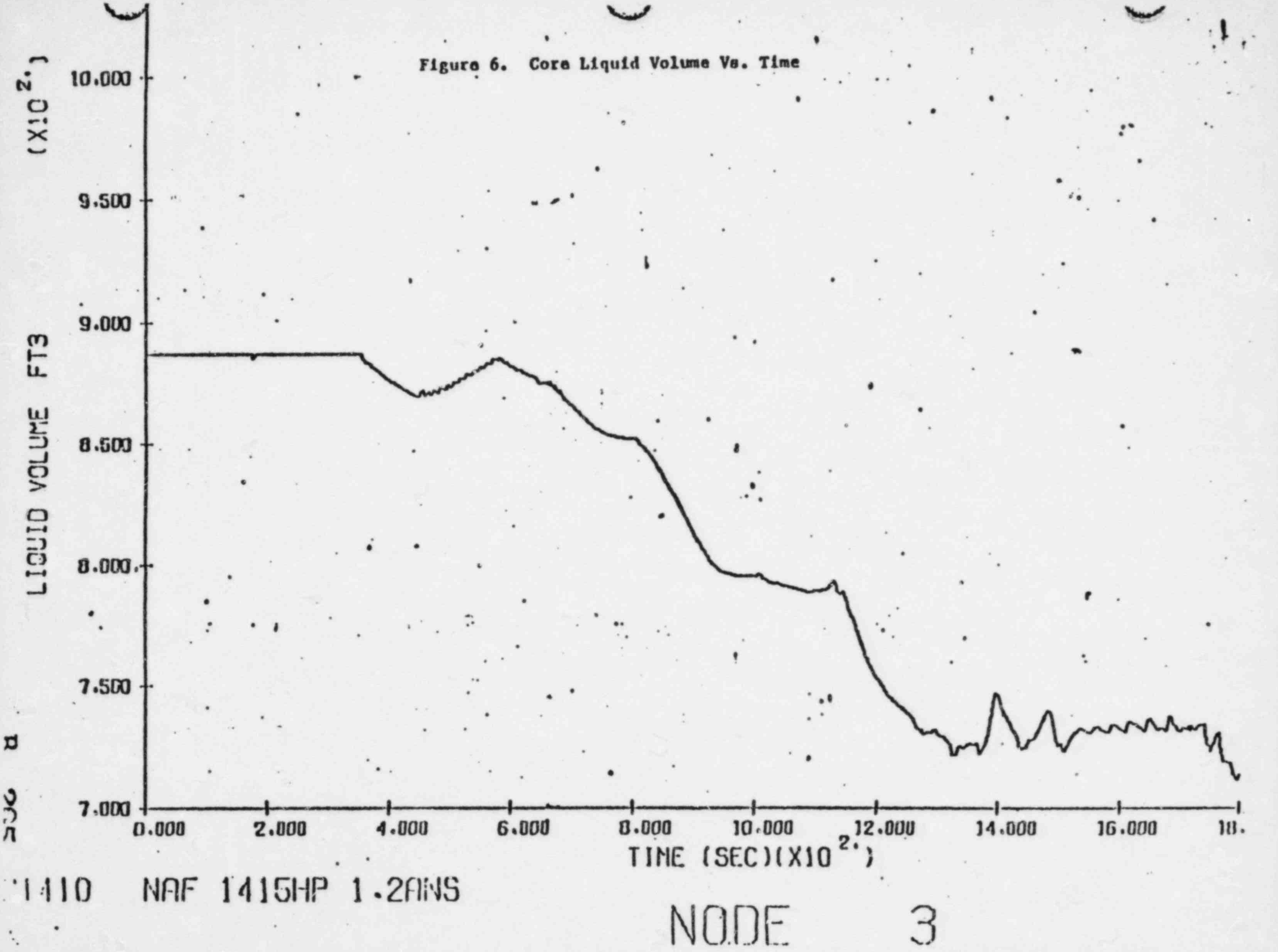


Figure 6. Core Liquid Volume Vs. Time

Figure 7. Hot Leg Liquid Volume Vs. Time

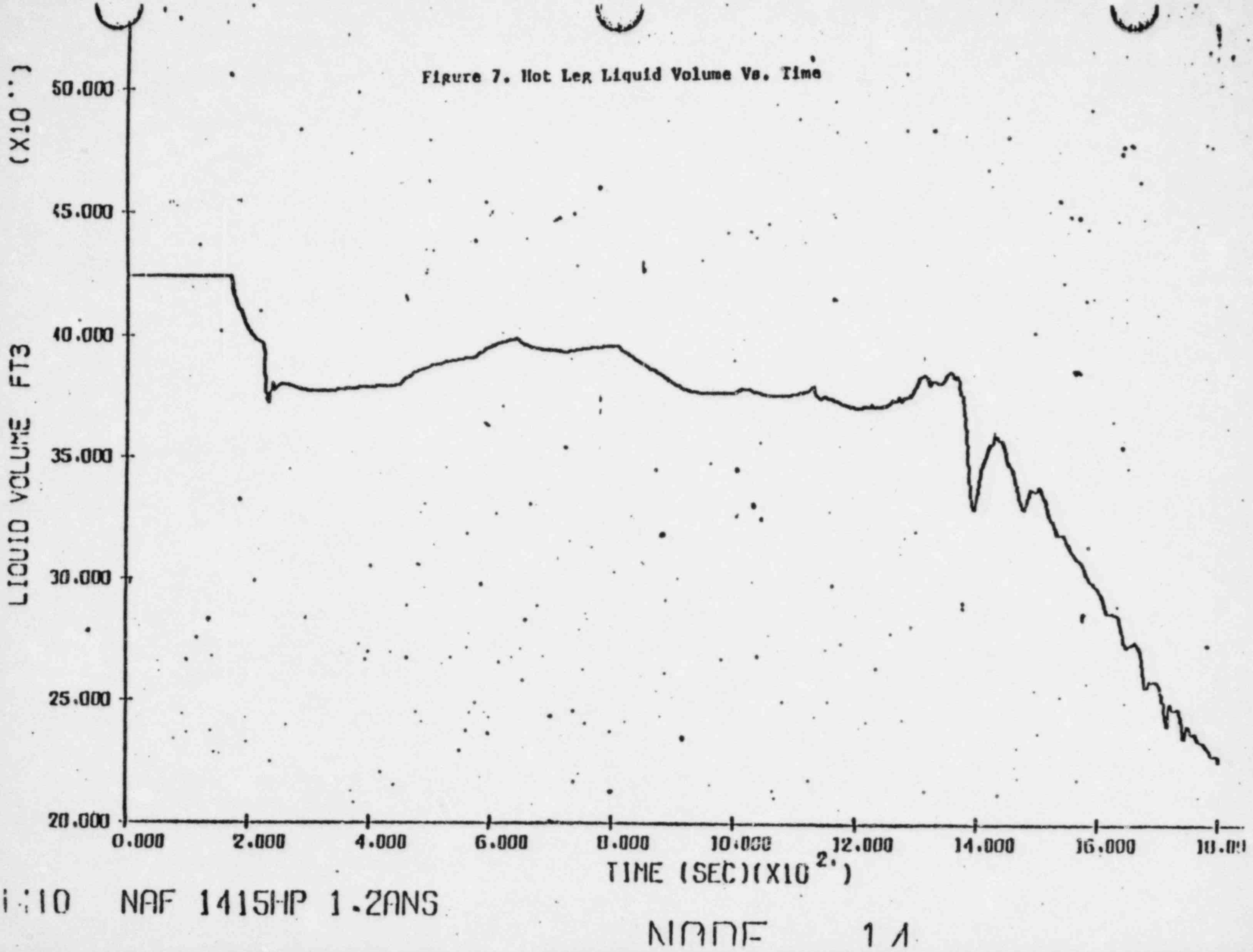


Figure 8. Lower Half of Steam Generator Liquid Volume

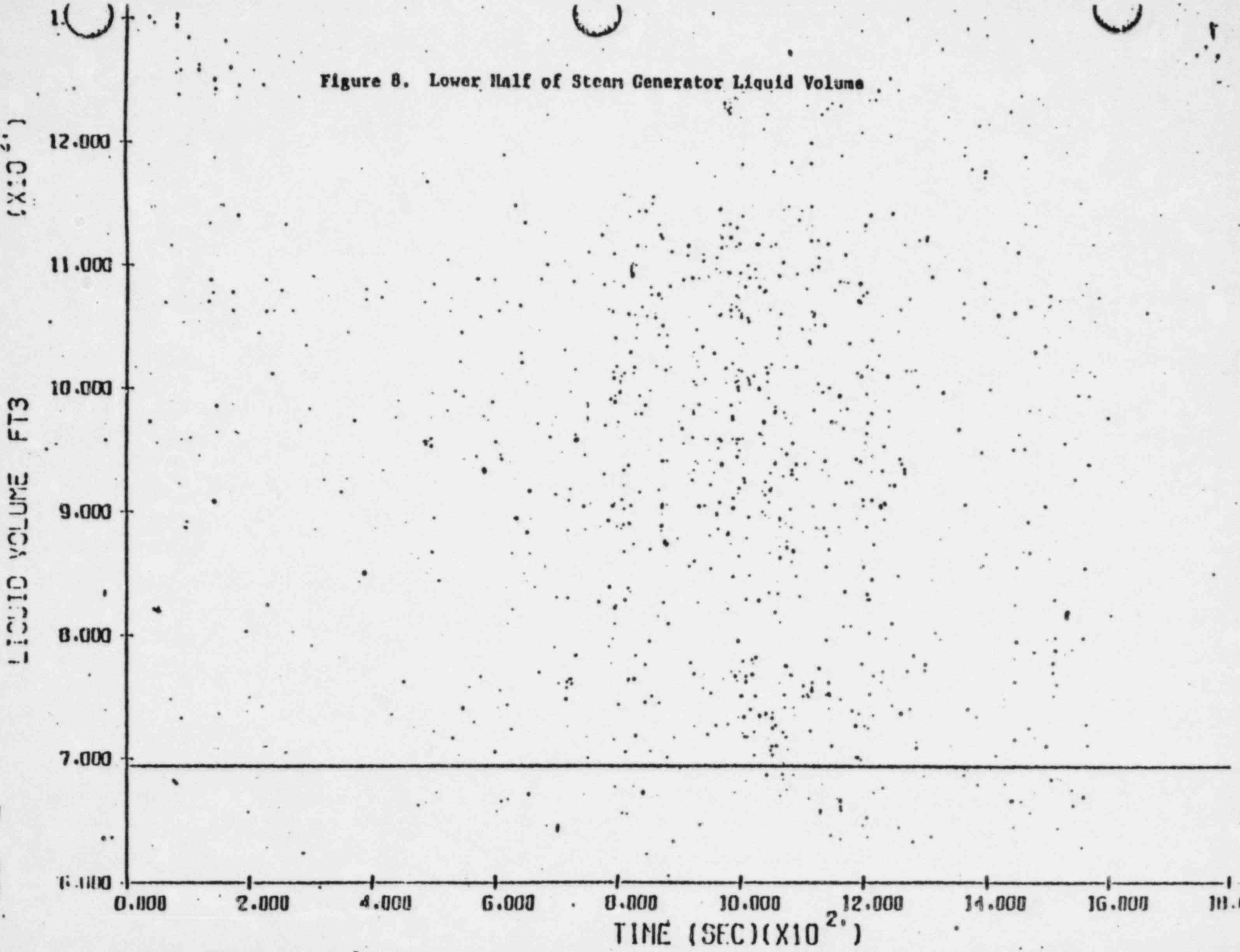


Figure 9. Cold Leg Liquid Volume

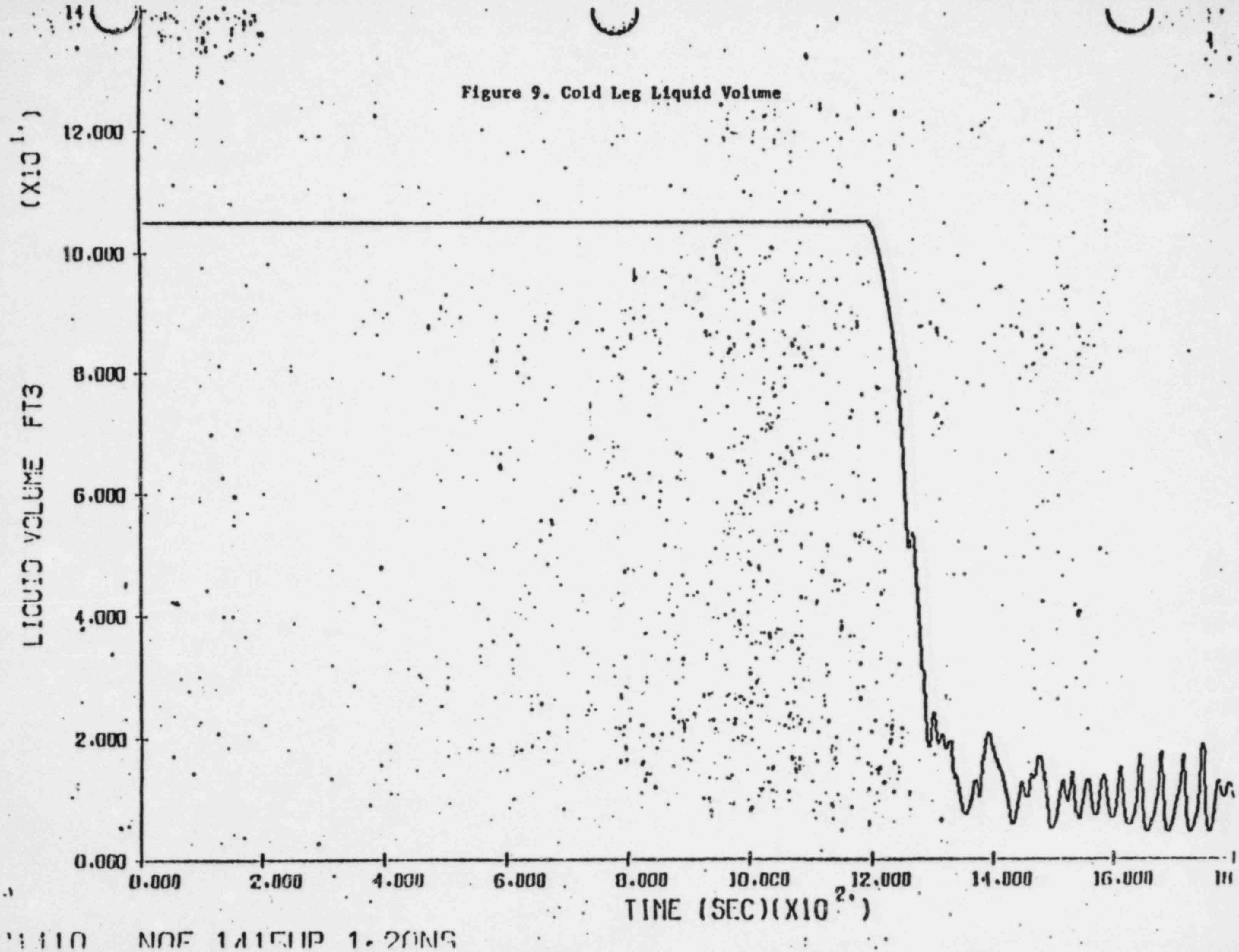


Figure 10. Downcomer Region Above Vessel Nozzles Liquid Volume

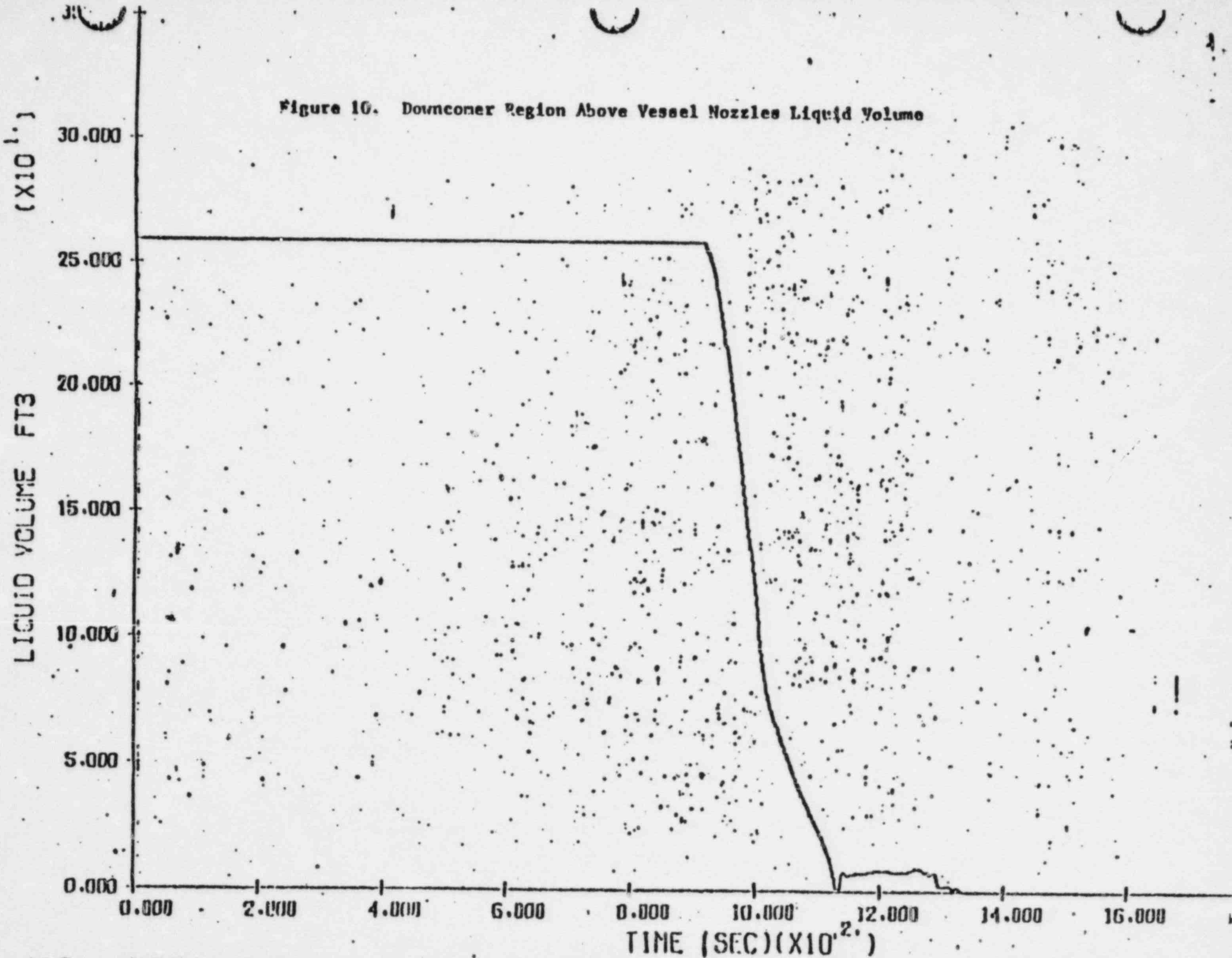
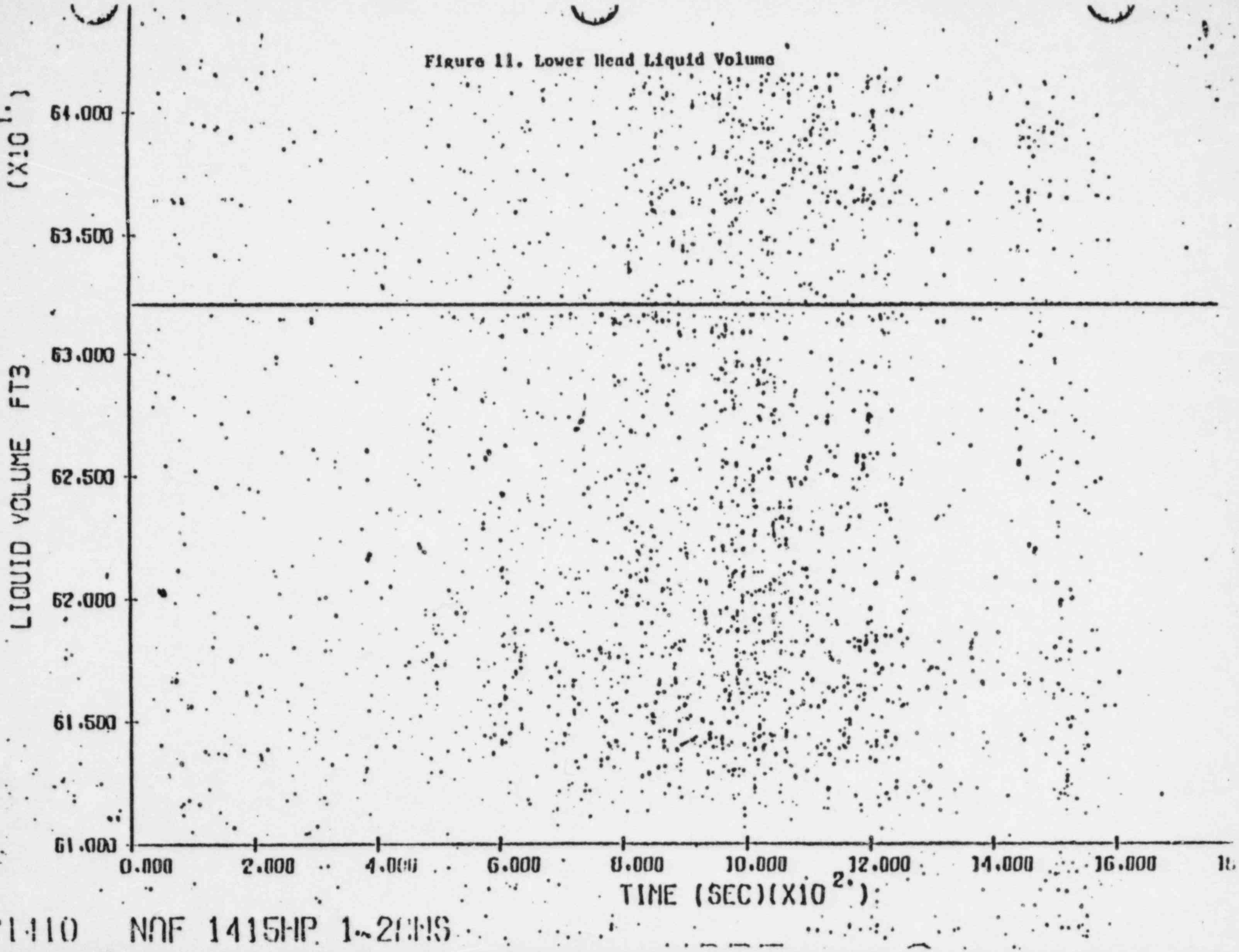


Figure 11. Lower Head Liquid Volume



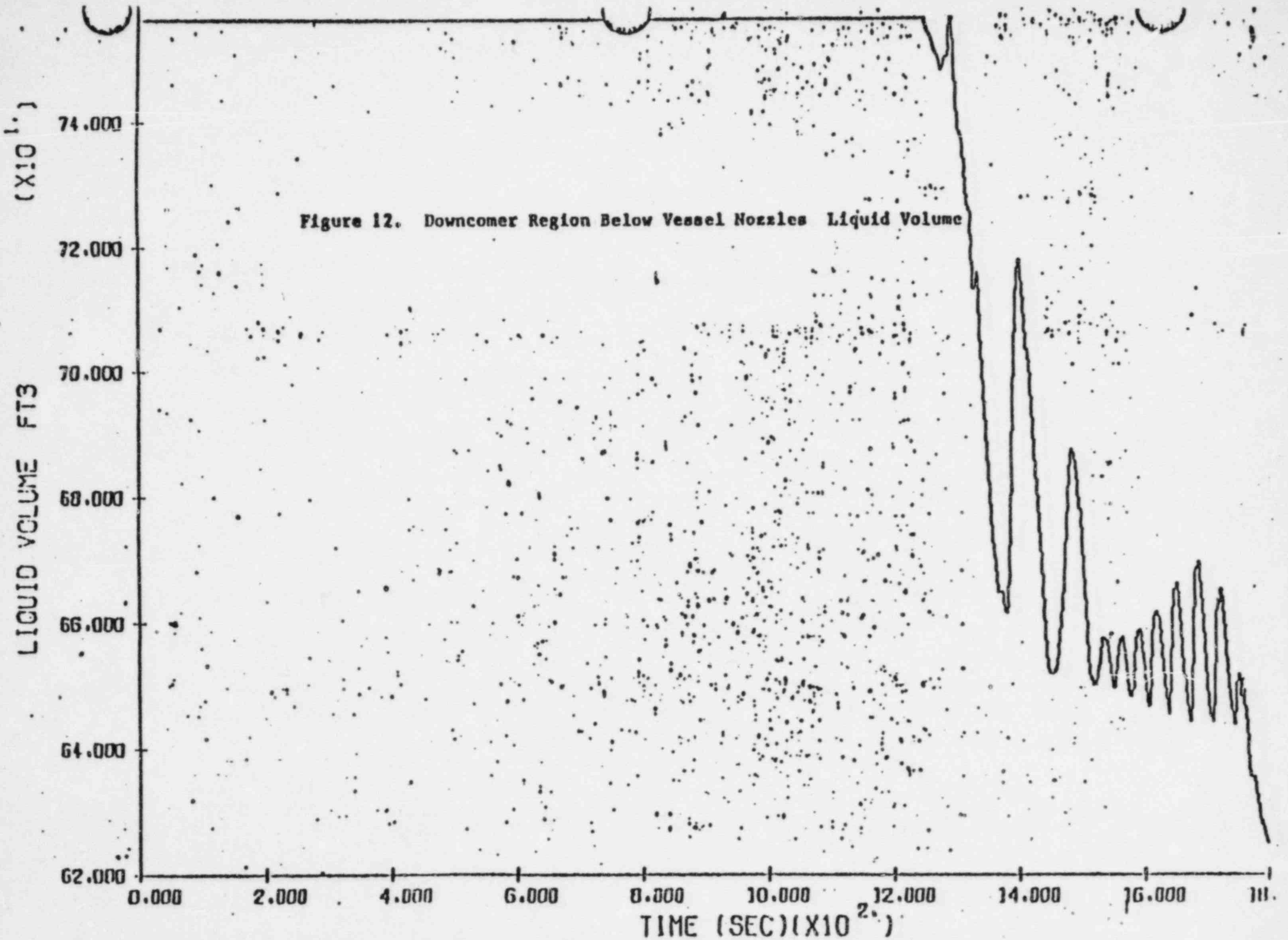
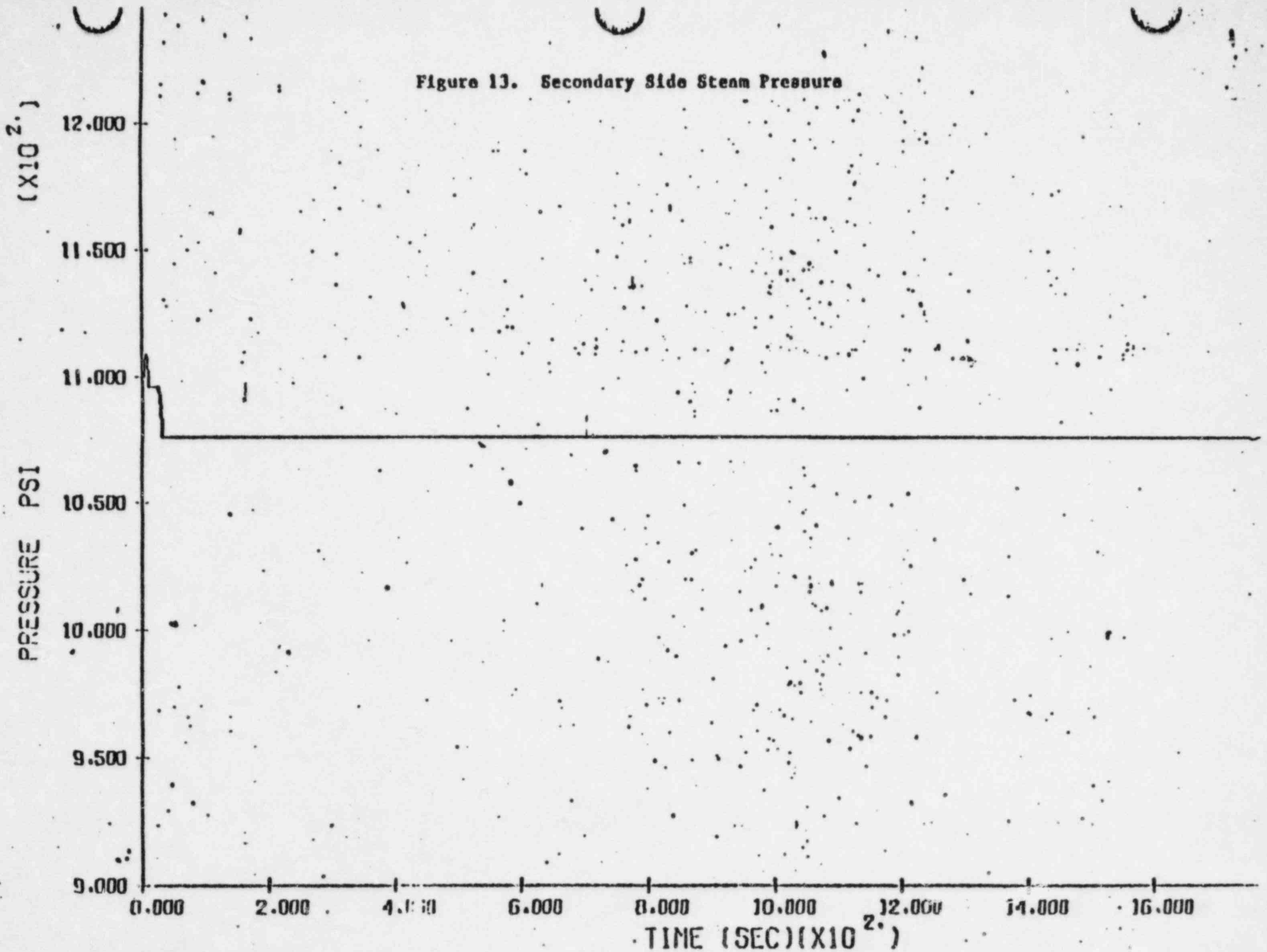


Figure 12. Downcomer Region Below Vessel Nozzles Liquid Volume

1410 NAF 1415HP 1.20NS



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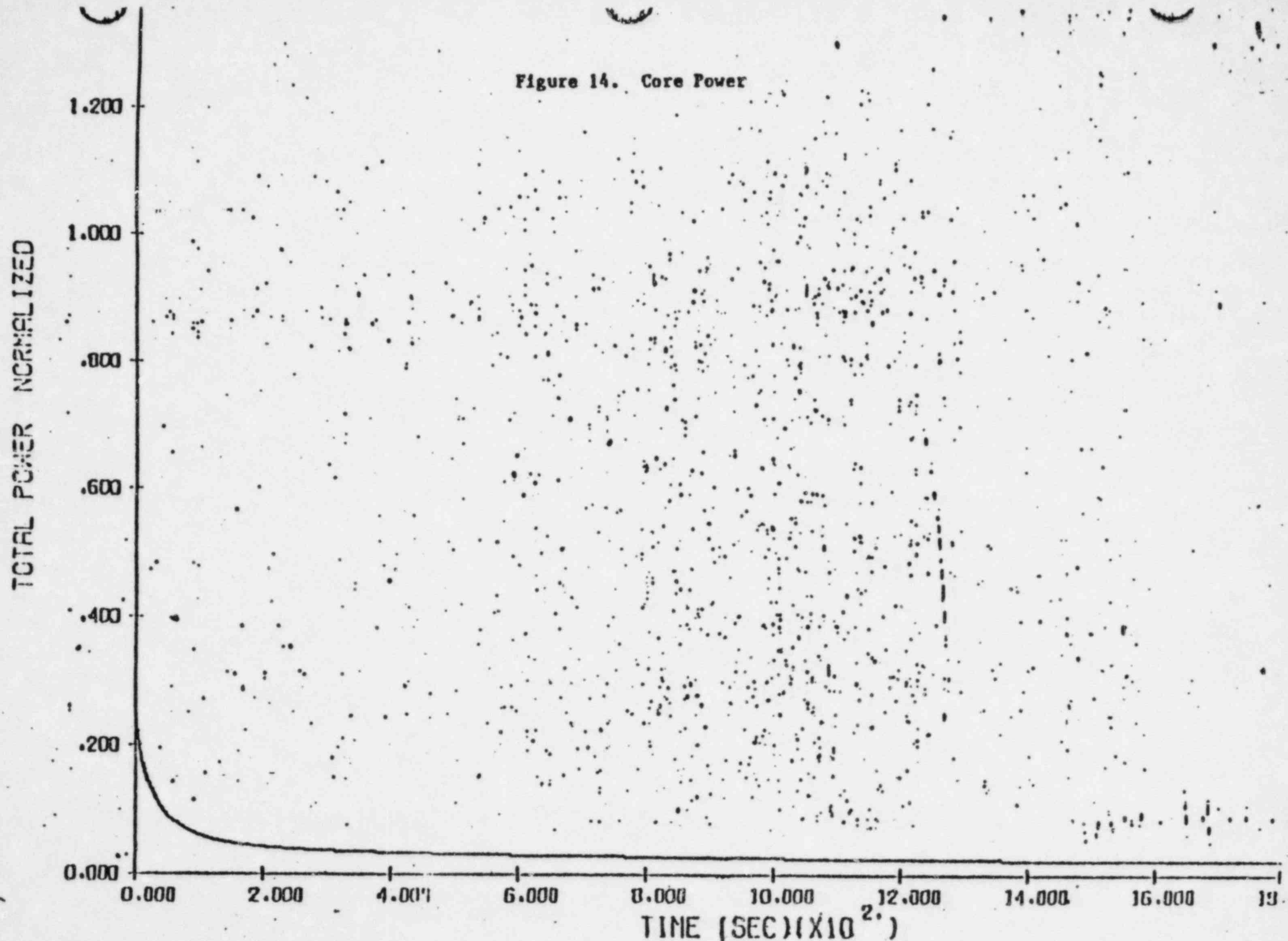
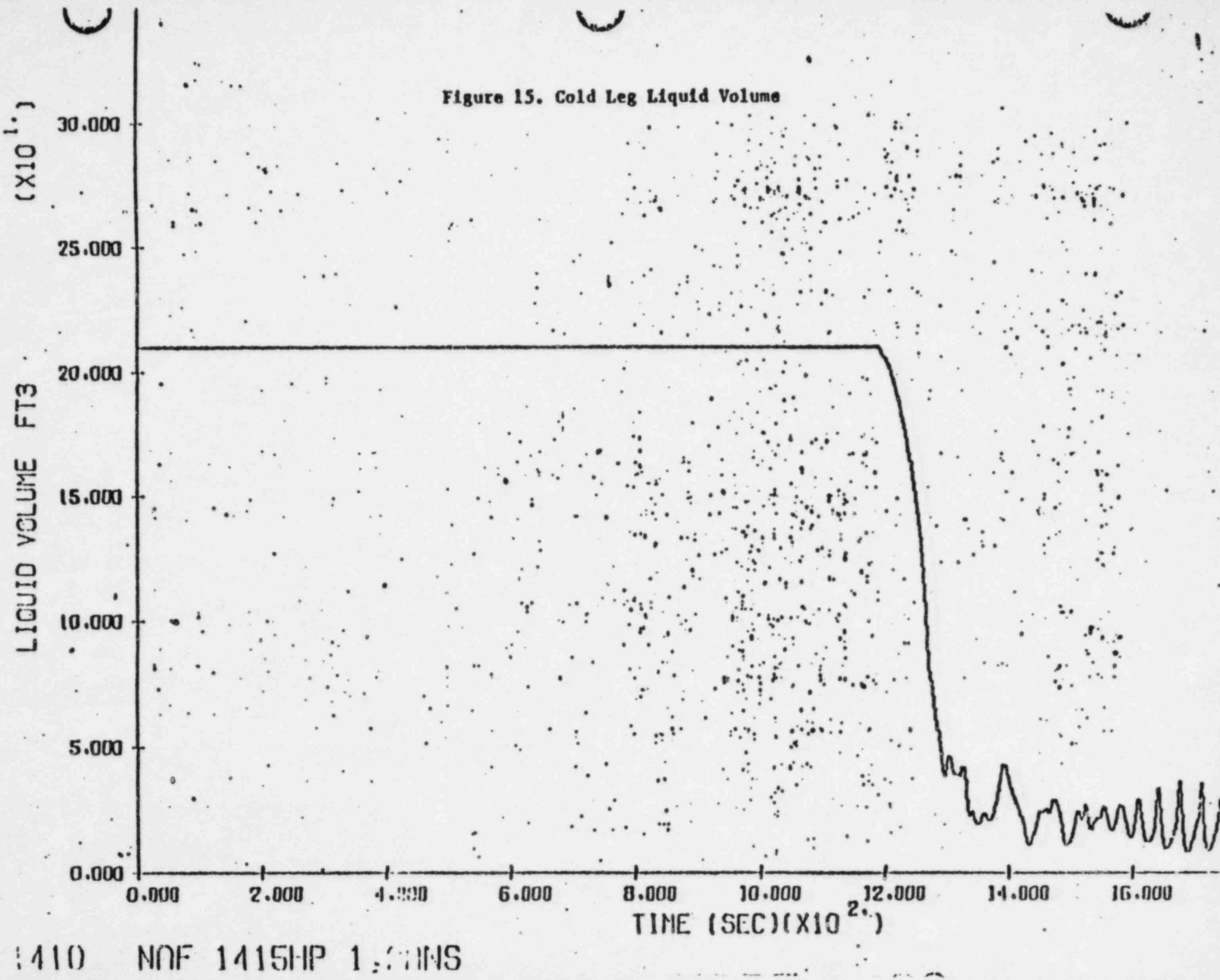
1410 NOF 1415HP

Figure 14. Core Power



NAF.1415HP 1.20MS

Figure 15. Cold Leg Liquid Volume



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