NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
THE HARTFORD ELECTRIC LIGHT COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

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June 16, 1980

Docket No. 50-336
A01061

Director of Nuclear Reactor Regulation
Attn: Mr. R. A. Clark, Chief
Operating Reactors Branch #3
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

References: (1) D. G. Eisenhut letter to W. G. Counsil dated October 22, 1979.
(2) W. G. Counsil letter to D. G. Eisenhut dated November 28, 1979.
(3) W. G. Counsil letter to D. G. Eisenhut dated March 10, 1980.
(4) W. G. Counsil letter to R. A. Clark dated May 20, 1980

Gentlemen:

Millstone Nuclear Power Station, Unit No. 2
Auxiliary Feedwater Systems

In Reference (1), Northeast Nuclear Energy Company was requested to respond to Enclosure 2 of that Reference regarding a generic request for additional information on auxiliary feedwater system flow requirements. As indicated in References (2) and (3), NNECO estimated completion of this effort on June 16, 1980. In fulfillment of that commitment, the attached information is being docketed regarding the design basis transient and accident conditions for the auxiliary feedwater system at Millstone Unit No. 2. In establishing the auxiliary feedwater system flow requirements, the following conditions were considered:

(1) Loss of Main Feedwater (LMFW)
(2) LMFW with Loss of Offsite A.C. Power
(3) LMFW with Loss of Onsite and Offsite A.C. Power
(4) Plant Cooldown
(5) Turbine Trip with and without Bypass
(6) Main Steam Isolation Valve Closure
(7) Main Steamline Break
(8) Small Break LOCA

In accordance with the provisions of Enclosure 2 of Reference (1), it is emphasized that the main feedline break is not considered in this evaluation as it is not a design basis event for Millstone Unit No. 2. Furthermore, the loss of main feedwater with a concurrent total loss of all A.C. power is not considered a credible event. Even though this condition was considered in the evaluation, it is incongruous to require that the auxiliary feedwater

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system be designed to automatically respond while numerous other plant systems require operator action. Therefore, NNECO's conclusion regarding the adequacy of the existing auxiliary feedwater system does not overtly apply to these postulated conditions. The analytical results presented in the attachment for the feedline break are provided for informational purposes only. Please recognize that any Staff recommendations for modifying the auxiliary feedwater system as a result of the review of the feedline break analysis or consideration of loss of offsite power and both onsite emergency diesel generators will be considered inappropriate.

The attached information further reinforces the conclusions documented in Reference (4). Specifically, delivery of 25% of the installed capacity of auxiliary feedwater flow is sufficient to ensure that design basis conditions are fulfilled.

Detailed responses to the specific requests of Enclosure 2 of Reference (1) are incorporated into the text of the attached document. Based on these analyses, NNECO has concluded that the auxiliary feedwater system is adequately sized and designed to comply with the acceptance criteria identified in Section 1.4. Therefore, no modifications are contemplated as a result of completion of this effort.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

W. G. Counsil

W. G. Counsil
Senior Vice President

Attachment

DOCKET NO. 50-336

ATTACHMENT

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

BASIS FOR AUXILIARY FEEDWATER SYSTEM FLOW REQUIREMENTS

JUNE, 1980

Millstone Point Unit 2

Basis for Auxiliary Feedwater System Flow Requirements

Introduction

Enclosure 2 of the October 22, 1979 letter from Mr. Eisenhut to Mr. W. G. Counsil (docket number 50-336) requested the Millstone Point 2 Nuclear Power Company to provide Auxiliary Feedwater (AFW) System design basis information as applicable to the design basis transients and accident conditions for their nuclear facility. This letter contains the requested information.

The following plant transients and accident conditions have been considered in establishing AFW flow requirements:

- Loss of Main Feedwater (LMFW)
- LMFW with loss of offsite AC power
- LMFW with loss of onsite and offsite AC power
- Plant Cooldown
- Turbine trip with and without bypass
- Main steam isolation valve closure
- Main steam line break
- Small break LOCA

The feedwater line break accident is not included as part of the design basis for the Millstone Point Unit 2 Nuclear Power Plant (MP2). However, we have included in this report the results of a detailed study of the AFW system performance in the event of this accident.

1. Discussion of Plant Transients Considered in AFW Design

In order to assess the performance of the auxiliary feedwater system, the adequacy of the minimum available flow during the loss of heat sink (LOHS) events listed above must be demonstrated. By "adequacy" we mean the ability of the available flow, assuming single failures and conservatisms as defined in section 3.0, to remove primary side heat to a degree that the plant acceptance criteria for the events are not violated.

In addition to LOHS type events, the adverse effects of the maximum deliverable AFW flow on the most severe RC overcooling events must be assessed.

1.1 Adequacy of Minimum AFW Flow

The following events have been addressed to assess the adequacy of minimum AFW flow:

1.1.1 LMFW - This event is the bounding case as far as this type of event is concerned. As a result, an analysis was performed in the FSAR which demonstrated the adequacy of minimum AFW flow for this event. The results of this analysis are described in section 14.10.1 of the FSAR and are discussed further in section 2.2.

1.1.2 LMFW with Loss of Offsite AC Power - In this case, the pumps have tripped and it is no longer necessary to remove pump heat through the steam generators. Therefore, the AFW flow requirement will be smaller for this case than for case 1.1.1. Primary flow heat transfer characteristics support this conclusion. This case is therefore bounded by case 1.1.1.

1.1.3 LMFW with Loss of Onsite and Offsite AC Power - This event is not part of the design basis for this plant, as it postulates an event which is not considered credible. However, functionally this case is bounded by case 1.1.1.

1.1.4 Turbine Trip with and without Bypass - A turbine trip results in an immediate reactor scram. If bypass is not available, the steam generator pressure will rise to the safety valve setpoint and remain there. The same assumption was made in case 1.1.1, with the additional conservatism of having the steam generator level at the low level setpoint. For this case, we have assumed nominal initial steam generator level. Therefore, this case is bounded by case 1.1.1. If steam bypass is available, the steam generator will stabilize at about 900 psi instead of the 1000 psi safety valve setpoint. This means that AFW is being pumped against a lower head, and hence more flow is provided. Also this case is bounded by case 1.1.1. A bounding analysis is also presented in the FSAR, Loss of Load Transient, section 14.9.

1.1.5 Main Steam Isolation Valve (MSIV) Closure - Closure of any MSIV results in reactor trip. The steam generators will pressurize up to the safety valve setpoint. This accident produces the same effects of a turbine trip without bypass as far as the AFW system is concerned, and hence it is also bounded by case 1.1.1.

1.1.6 Small Break LOCA - For a certain spectrum of sizes of small break LOCAs, the steam generators will be required to remove that fraction of decay heat not being removed through the break itself. Hence, the AFW flow rates required will be less then that required for the loss of feedwater accident (case 1.1.1) where all the decay heat must be removed through the steam generators. Therefore, it is demonstrated that a small break LOCA is bounded by case 1.1.1 as far as the AFW system is concerned.

1.2 Acceptability of Max AFW Flow

The following events have been considered to determine the acceptability of the maximum AFW flow rate.

1.2.1 Steam Line Break - Run-out flow during a main steam line break will maximize the consequences of excessive AFW flow on plant response. As a result, an analysis has been performed to determine acceptability of the maximum AFW flow for this event. The results of this analysis are described in section 2.1.

1.2.2 Plant Cooldown - The cooldown resulting from delivery of the maximum possible amount of AFW to the steam generators (1000 gpm at 900 psig) has been calculated to be less than 80°F in the first 10 minutes following a scram. Although this calculation is very conservative because it neglects the substantial contribution of decay heat, the results are within the acceptance criteria as defined in section 1.4.2.2. Plant cooldown due to spurious actuation of auxiliary feedwater has not been analyzed because the S.G. level control system will prevent an unacceptable cooldown. Failure of the control system will result in the trip discussed above.

1.3 Accident Not in the Design Basis

1.3.1 Feedline Break - This accident is not in the design basis for the AFW system at MP2. However, an analysis was performed as requested and its results are included in this submittal (section 2.3) for informational purposes.

1.4 Acceptance Criteria

The criteria for determining if the AFW flow rate is acceptable for the events described in sections 1.1 and 1.2 are given below:

1.4.1 Adequacy of Minimum AFW Flow (section 1.1) - The AFW flow rate shall be considered adequate for events 1.1.1 through 1.1.5 providing:

1.4.1a - The pressurizer pressure will not exceed 110 percent of the RCS design pressure (2750 psig).

1.4.1b - No DNB condition is experienced at the clad surface of any fuel rod in the core.

1.4.1c - Sufficient steam generator level remains to remove the primary side heat generated. Minimum available AFW is sufficient to provide long term recovery of steam generator level.

For event 1.1.6 (small break LOCA) the criterion will be that sufficient steam generator liquid level will remain to remove that fraction of primary side heat generated which is not removed via the break, such that 10CFR50.46 limits are not exceeded.

1.4.2 Acceptability of Maximum AFW Flow (section 1.2) - Two separate criteria are to be considered here; one for the cooldown resulting from a steam side accident and one for the cooldown resulting from a normal plant shutdown.

1.4.2.1 Effect of maximum run-out flow on the limiting secondary steam release accident: The AFW flow rate shall be considered not to exceed the maximum permissible limit providing that DNB criteria are not exceeded.

1.4.2.2 Cooldown following normal reactor shutdown: The AFW flow rate shall be considered not to exceed the maximum permissible providing the primary side coolant does not cool down more than 100°F in the ten minute period following the shutdown.

For the above cases, it is assumed that the operator would take action at ten minutes into the transient to control the AFW system which is causing the overcooling.

2. Analyzed Events

Sections 1.1 and 1.2 have reduced the number of transients to be included in the design basis to two limiting events. These events are the steam line break and the loss of feedwater. A summary of the method of analysis and the results obtained are included in this section. In addition, a description of the feedline break (which is not included in the design basis) is included for informational purposes.

2.1 Steam Line Break - The analysis of the steam line break accident with automatic AFW initiation has previously been docketed (ref. 1). This analysis assumed a circumferential rupture of a 34 inch diameter steam line. A conservatively high value of AFW flow was calculated assuming all pumps are operable and automatically initiated and deliver water in a run-out condition due to reduced back pressure at the broken steam generator. The worst pressure split between the steam generators (1000 psia intact, 0 psia broken) was assumed in order to divert the maximum amount of AFW flow into the broken steam generator. This flow was conservatively increased by over 30% to maximize cooldown. It was assumed that the operator acted to isolate the AFW system from the break ten minutes into the transient. The results showed that the DNB limits are not exceeded, even with the most reactive rod stuck in the withdrawn position. Hence, the AFW system met the acceptance criteria as defined in section 1.4.2 for this transient.

2.2 Loss of Feedwater - The loss of feedwater accident has been determined to be the limiting design basis event in terms of minimum AFW flow required.

The existing FSAR analysis of this accident showed through a very conservative calculation that, without operator action to feed auxiliary feedwater at 10m., the S.G.'s would dry out in over 15m. At 10m., both steam generators were shown to have still ample water inventory to provide adequate decay heat removal. Addition of 600 gpm of auxiliary feedwater (corresponding to 50 percent of full system capacity) was shown to quickly recover steam generator level. Automatic initiation of auxiliary feedwater will further improve auxiliary feedwater system performance by adding water at an earlier time and preventing depletion of existing inventory. Analyses recently performed (Reference 2) have shown that, assuming automatic initiation of auxiliary feedwater, only 300 gpm (or 1/4 of the total capacity of the AFW system) would be adequate for the loss of feedwater event.

Therefore, the present AFW system meets the criteria set forth in section 1.4.1 for this accident.

2.3 Feedline Rupture - This accident is not in the design basis of the plant. The results of the analyses of this accident are included for informational purposes.

2.3.1 Method of Analysis - The feedwater line break analyses were performed utilizing two different computer codes, namely, SGN III and LTC. The need to correctly predict the steam generator response dictated the use of a steam generator design code such as SGN III. This code, described in Reference 3, is used when the specific emphasis of an analysis requires maximization of primary-to-secondary heat transfer.

The major conservatisms imbedded in SGN III are:

a. Steam generator tube heat transfer coefficients are not degraded from operating conditions as a function of time.

b. All primary side walls are kept in thermal equilibrium with RCS liquid inventory.

c. Loop transit times are neglected.

The limitations of the SGN III in predicting primary system response required, in turn, the use of a loop model to calculate RC pressure during the accident. LTC, which is a best estimate code described in Reference 4 was selected for this purpose. This code was selected because it is the most accurate model of the primary system available and is capable of providing the most accurate response of the primary system to the steam generator transient being studied. The use of a best estimate code is also supported by the fact that in a long-term event, resulting in subsequent pressure decreases and pressure increases, it becomes unclear what constitutes a "conservative" approach. However, the input data were selected utilizing the standard conservatisms to maximize primary system pressure during the repressurization phase (i.e., no pressurizer pressure sprays in use, etc.). Pressurizer pressure response was determined using a "piston" model, to maximize pressure rise time.

Table 1 presents the analysis assumptions used in these calculations. Table 2 presents the assumptions specific to the main feedline break analysis.

2.3.2 Case 1 - no AFW

This case was analyzed with SGN III only in order to determine the minimum time to intact steam generator dryout. A dryout time of 21 minutes from event initiation was determined. This result was obtained

a steam generator model which accurately reflects the "as built" internal structure of the steam generator. This event begins as a water blowdown changing to a steam blowdown as steam generator level falls below the feedwater ring. Table 3 gives the sequence of events for this case. Figures 3 to 6 show reactor coolant temperature, steam generator pressure, steam generator temperature and steam generator two-phase volume, vs. time.

2.3.3 Case 2 - 600 GPM AFW delivered at 10 minutes

This case was analyzed as described with SGN III and represents automatic initiation of AFW with manual isolation of the broken steam generator at 10 minutes. Prior to isolation run-out flow to the break prevents any auxiliary feedwater from entering the unaffected steam generator. This case, therefore, also covers manual actuation at 10 minutes. Assuming isolation at 10 minutes 600 GPM are delivered to the unaffected steam generator. This amount of AFW is sufficient to remove the primary system heat and begin steam generator refill. Figures 7 to 10 show reactor coolant temperature, steam generator pressure, steam generator temperature and steam generator two-phase volume vs. time. Figure 10 shows a steady increase in steam generator level following break isolation, showing the adequacy of the 600 GPM flow to remove decay heat and re-establishing steam generator level. (A calculation was also performed assuming delivery of 300 GPM only, and even this minimum flow showed adequate steam generator inventory recovery.)

Figure 11 shows pressurizer pressure vs. time (LTC); figure 8 shows steam generator pressure vs. time. Pressurizer pressure decreases as a result of the blowdown then increases and opens the PORV at about 1600 seconds. Although steam generator level and pressure are rapidly increasing the lack of pressurizer spray and the assumption of no turbine bypass available force reactor coolant system pressure to stay at 2400 psia until the secondary relief opens at 1000 psia. The SGN III code predicts this to happen at about 2200 seconds; the LTC code predicts in excess of 2400 seconds. The reasons for this difference are that SGN III maximizes heat transfer and therefore repressurizes sooner and pumps were left running, also to maximize heat transfer. LTC, on the contrary, is a best estimate code and pumps were tripped at 1600 psia. The minimum inventory in the intact steam generator for this case is 8192 lbm. The primary pressure does not violate the criteria established in section 1.4.1.

3.0 Capability of AFW System - This section is included to describe the capabilities of the AFW system at MP2.

3.1 AFW Minimum Flow Rates - Table 4 gives the minimum flow rates that can be expected to reach the intact steam generators for various combinations of flow, isolation, and pressure. These values are determined assuming a recirculation flow rate of 50 gpm at 1040 psig for the turbine driven pump, and 25 gpm at 1040 psig for each electric driven pump. Pump wear has been assumed to be negligible since these pumps are used only during plant startup and shutdown.

It has been determined that 600 gpm of AFW flow will be needed to remove decay heat and pump heat after shutdown. This value assumes ANS+20 decay heat. As can be seen from Table 4, only one steam generator must be available to be able to receive this much flow. Note that if more realistic decay heat values were to be used and both pumps were assumed to start, then steam generator inventory loss would be minimized and level recovered rapidly. This is consistent with the experience at the plant.

3.2 AFW Source Inventory - The primary source of water is the CST which has a minimum capacity of 150,000 gallons by technical specifications. The long term water source is fire water system which can provide adequate water to feed the AFW system indefinitely.

Maximum design conditions required to switch to the RHR system are 300 psig and 300°F. A maximum of 99,300 gallons of AFW are required to cool down the reactor to 300°F in the four hour period starting from the beginning of any transient. Hence, we do not need any more water than the volume of the CST.

4.0 Conclusions - Based on the analyses presented here, we conclude that the auxiliary feedwater system for the Millstone Point Unit 2 Nuclear Power Plant is adequately sized to meet the acceptance criteria defined in section 1.4 for the events in its design basis (section 1.1 and 1.2). The water inventory for this system is large enough to remove primary side heat until well past the point where the RHR system operation starts.

Reference

1) Letter from W. G. Counsil to R. Reid, Docket No. 50-336, dated January 25, 1980.

2) Letter from W. G. Counsil to R. A. Clark, Docket No. 50-336, dated May 20, 1980.

3) System 80 PSAR, Standard PWR NSSS, CESSAR-P, App. 68.

4) CEN-128, "Response of Combustion Engineering Nuclear Steam Supply Systems to Transients and Accidents".

Table 1

Analysis Assumptions

Core Power, MWt	2700
Pump Power, MWt	10
Primary Pressure psia	2250
T_{in}, ^{o}F	551
Core Flow Rate, gpm	369929.
S.G. Initial Pressure, psia	877.9
S.G. Initial Inventory (per S.G., total water and steam) lbm	144612.
Initial Water Level Assumed in S.G. Downcomer	Normal (24" below can deck)
Decay Heat Used	ANS + 20%
Moderator Cooldown Curve	Figure 1
Fuel Doppler Curve	Figure 2
Scram Rod Worth	-5.31 $\%\Delta\rho$

Table 2

Assumptions for Main Feedwater Line Break

Main Feedwater Flow, GPM	None
Auxiliary Feedwater Flow, GPM	600 (1)
Reactor Coolant Pumps	On (2)
Feedwater Line Break Size, FT^2 (guillotine)	1.48 (3)
Decay Heat Margin, %	20%
All other data	CEN-128

Notes:

(1) Intact unit only, starting at 10 minutes. No flow to ruptured unit.

(2) Pumps were tripped after SIAS, as per current MP2 operating procedures.

(3) Complete guillotine break assumed.

Table 3

Feedline Rupture Case 1

Time (seconds)	Event
0	Feedline Rupture
9.08	Reactor Trip
212.4	MSIS
247.0	Dryout Affected S.G.
1260.0	Dryout Unaffected S.G.

Table 4

Minimum Expected AFW Flow Rates

Condition	Total Flow to All Intact Steam Generators (GPM)		
	1 m.d. pump	2 m.d. pumps	2 m.d. + 1 s.d. pump
One Ruptured SG and 1 Intact SG Unisolated at 1000 psig	0	0	0
2 Intact SG's at 1000 psig	300	600	1000
1 Intact SG at 1000 psig with the Other Isolated	300	600	1000

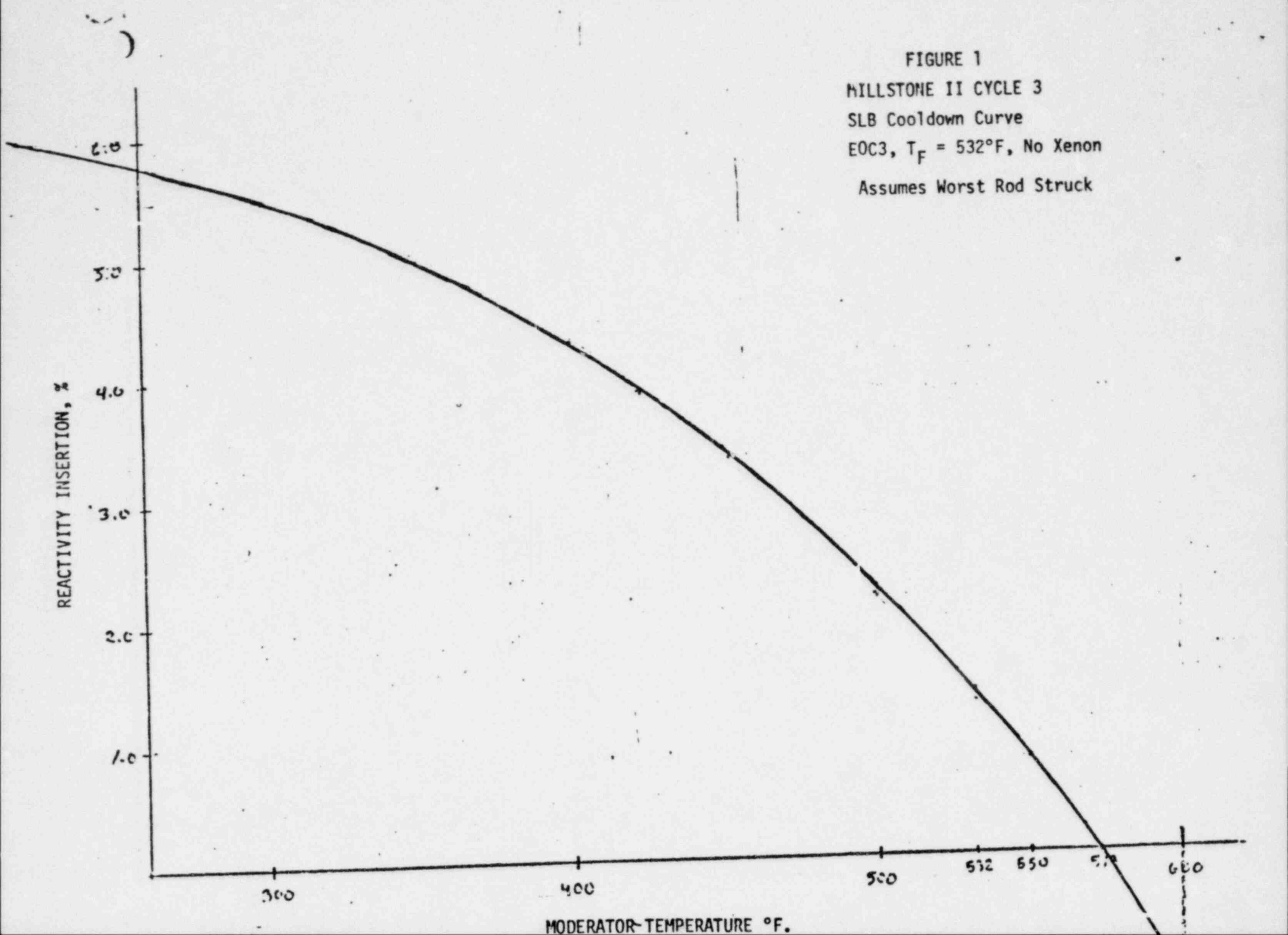
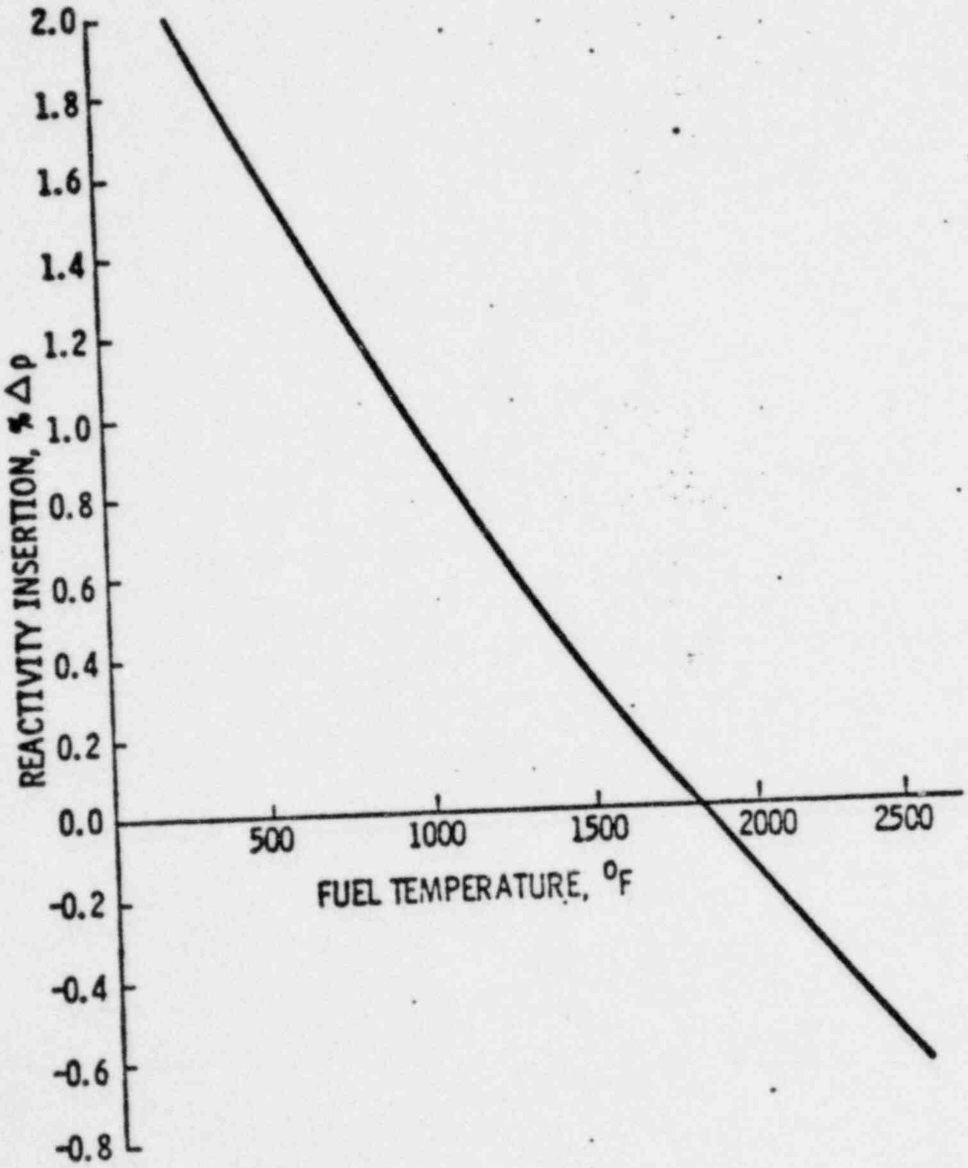


FIGURE 1
MILLSTONE II CYCLE 3
SLB Cooldown Curve
EOC3, T_F = 532°F, No Xenon
Assumes Worst Rod Struck
REACTIVITY INSERTION, %
6.0
5.0
4.0
3.0
2.0
1.0
300
400
500
532
550
600
MODERATOR TEMPERATURE °F.

Figure 2
DOPPLER REACTIVITY vs FUEL TEMPERATURE



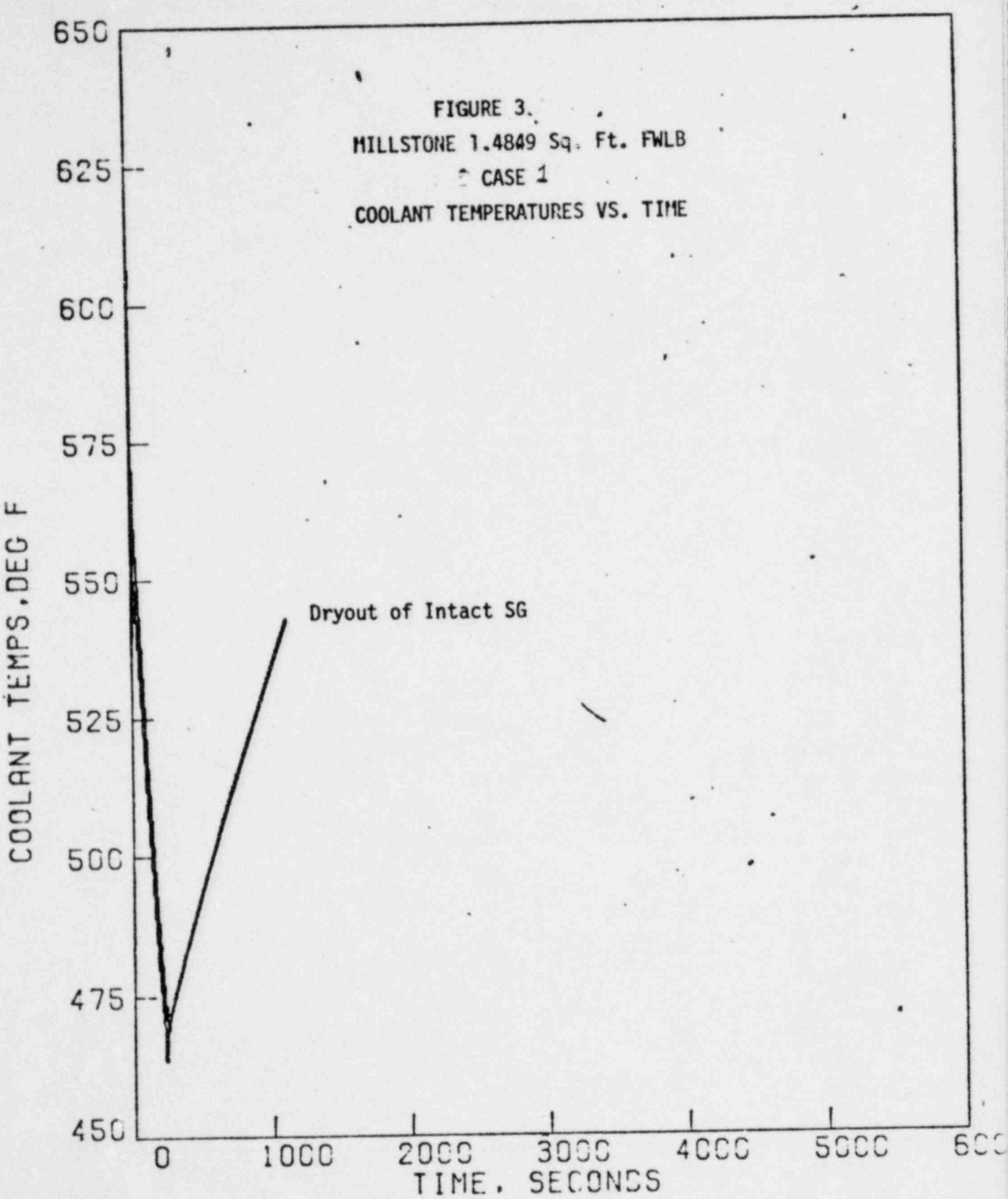


FIGURE 3.
MILLSTONE 1.4849 Sq. Ft. FWLB
CASE 1
COOLANT TEMPERATURES VS. TIME
Dryout of Intact SG
COOLANT TEMPS, DEG F
650
625
600
575
550
525
500
475
450
0
1000
2000
3000
4000
5000
TIME, SECONDS

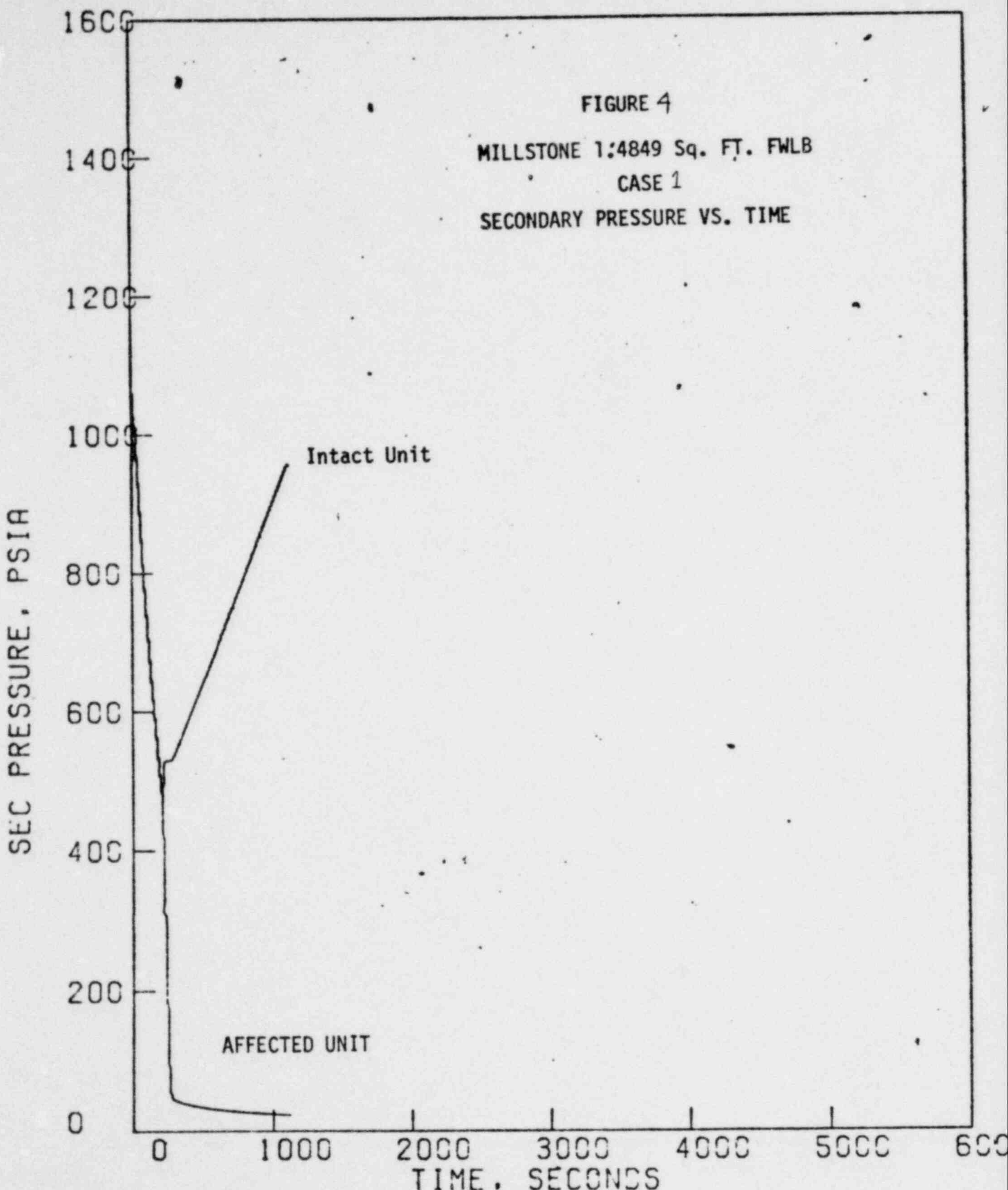


FIGURE 4
MILLSTONE 1:4849 Sq. FT. FWLB
CASE 1
SECONDARY PRESSURE VS. TIME
Intact Unit
AFFECTED UNIT
SEC PRESSURE, PSIA
TIME, SECONDS
0
200
400
600
800
1000
1200
1400
1600
1000
2000
3000
4000
5000

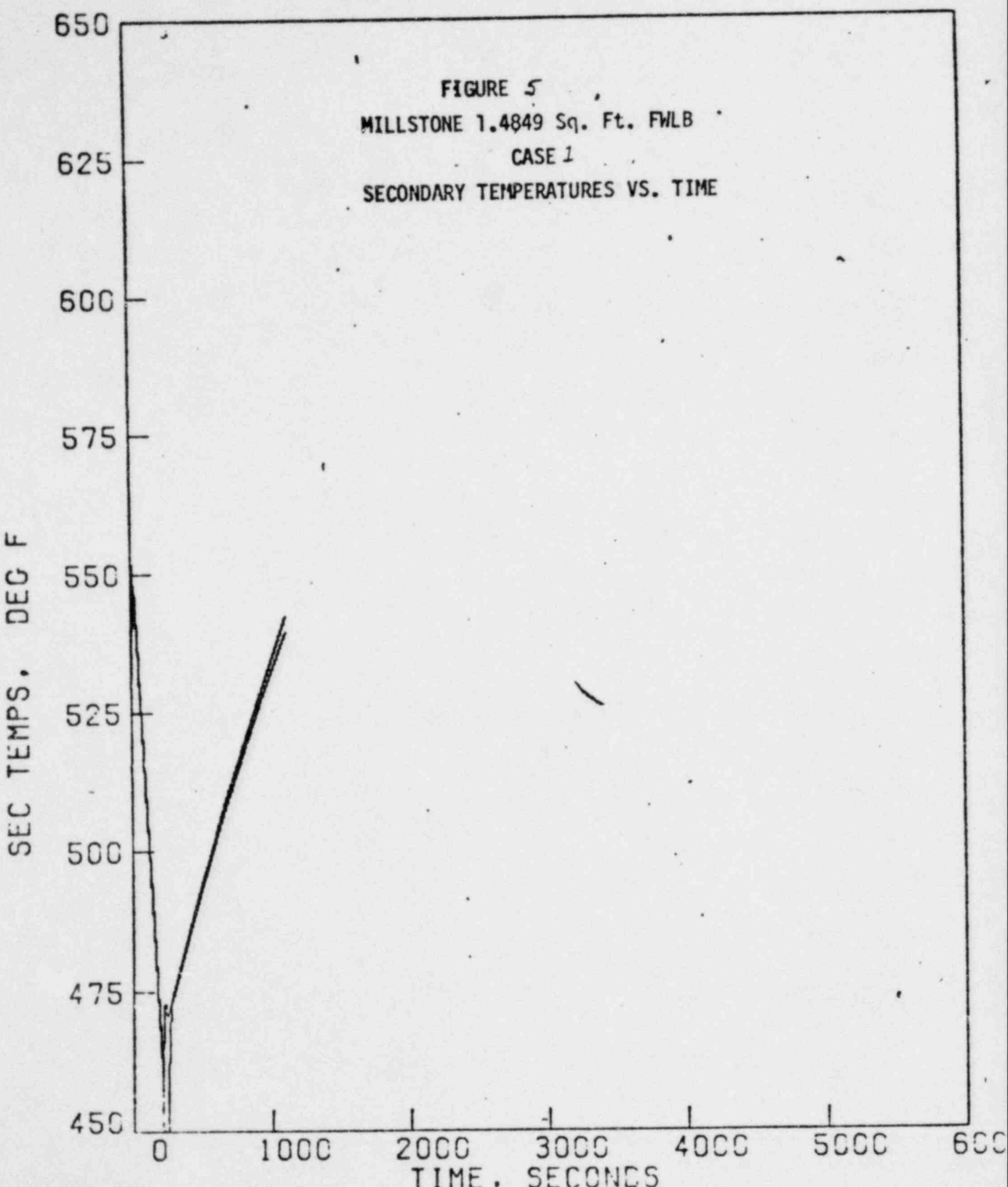


FIGURE 5
MILLSTONE 1.4849 Sq. Ft. FWLB
CASE 1
SECONDARY TEMPERATURES VS. TIME
SEC TEMPS, DEG F
650
625
600
575
550
525
500
475
450
0
1000
2000
3000
4000
5000
600
TIME, SECONDS

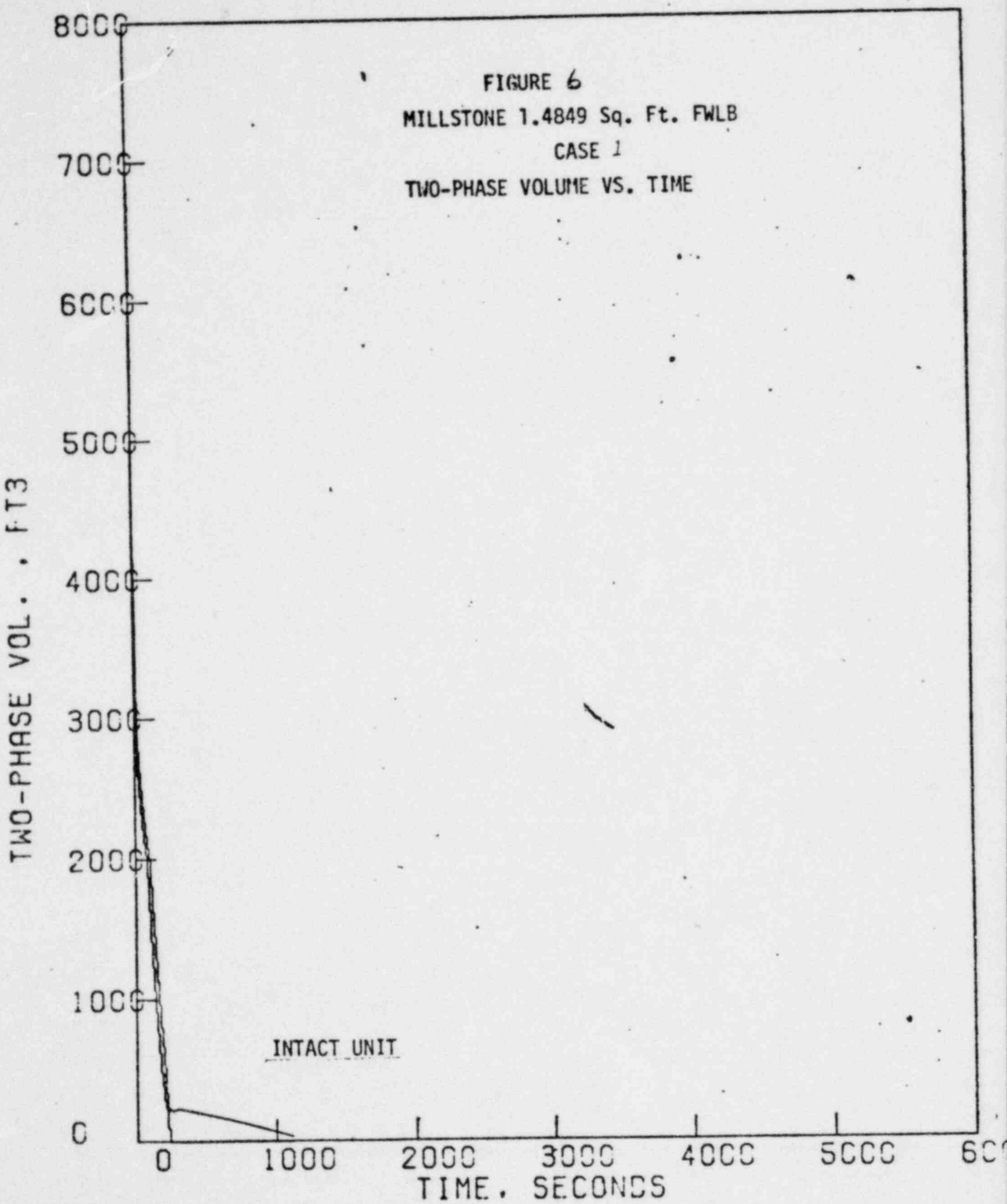


FIGURE 6
MILLSTONE 1.4849 Sq. Ft. FWLB
CASE 1
TWO-PHASE VOLUME VS. TIME
TWO-PHASE VOL. , FT3
8000
7000
6000
5000
4000
3000
2000
1000
0
INTACT UNIT
0
1000
2000
3000
4000
5000
TIME, SECONDS

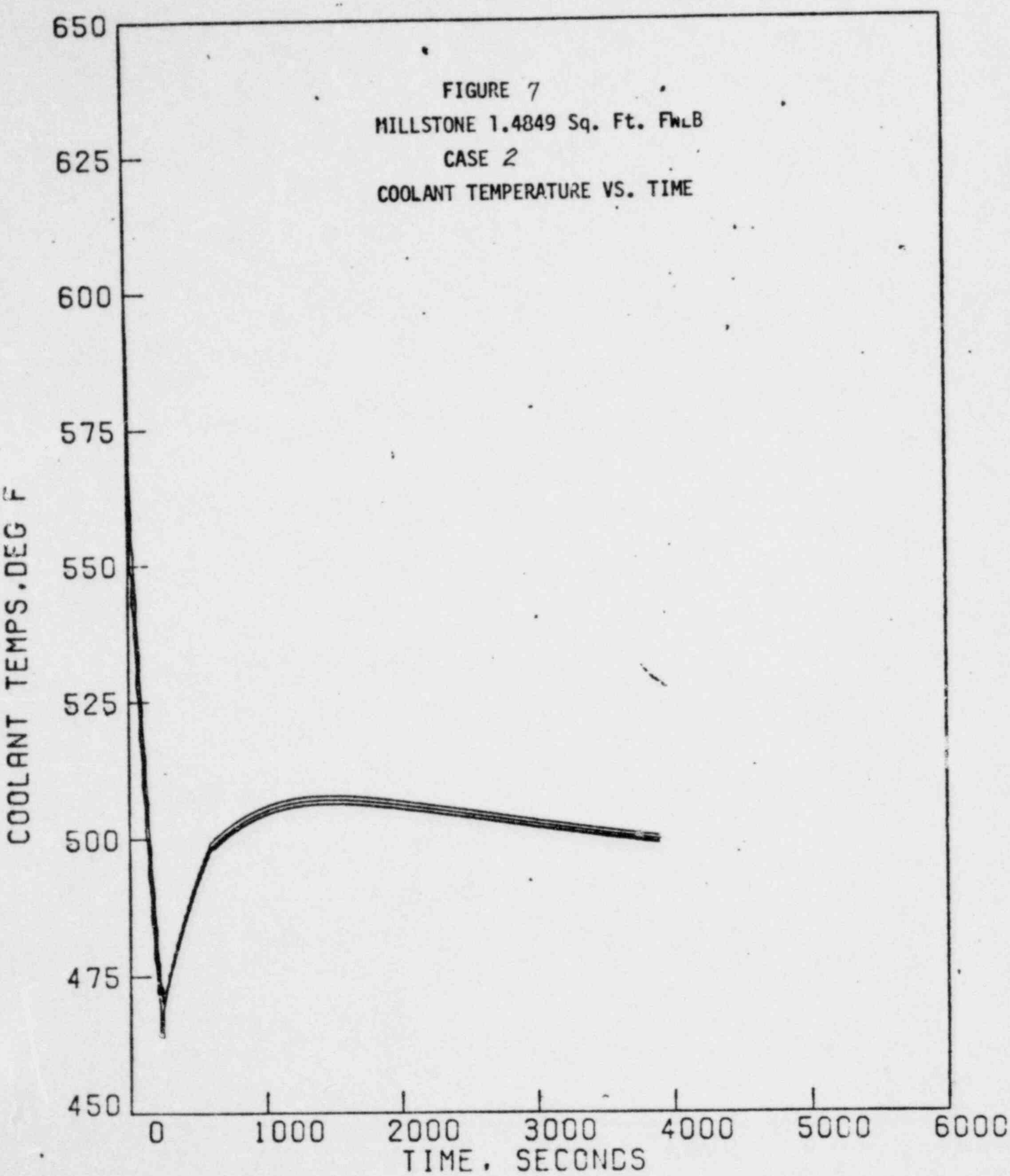


FIGURE 7
MILLSTONE 1.4849 Sq. Ft. FWLB
CASE 2
COOLANT TEMPERATURE VS. TIME
COOLANT TEMPS.DEG F
650
625
600
575
550
525
500
475
450
0
1000
2000
3000
4000
5000
6000
TIME, SECONDS

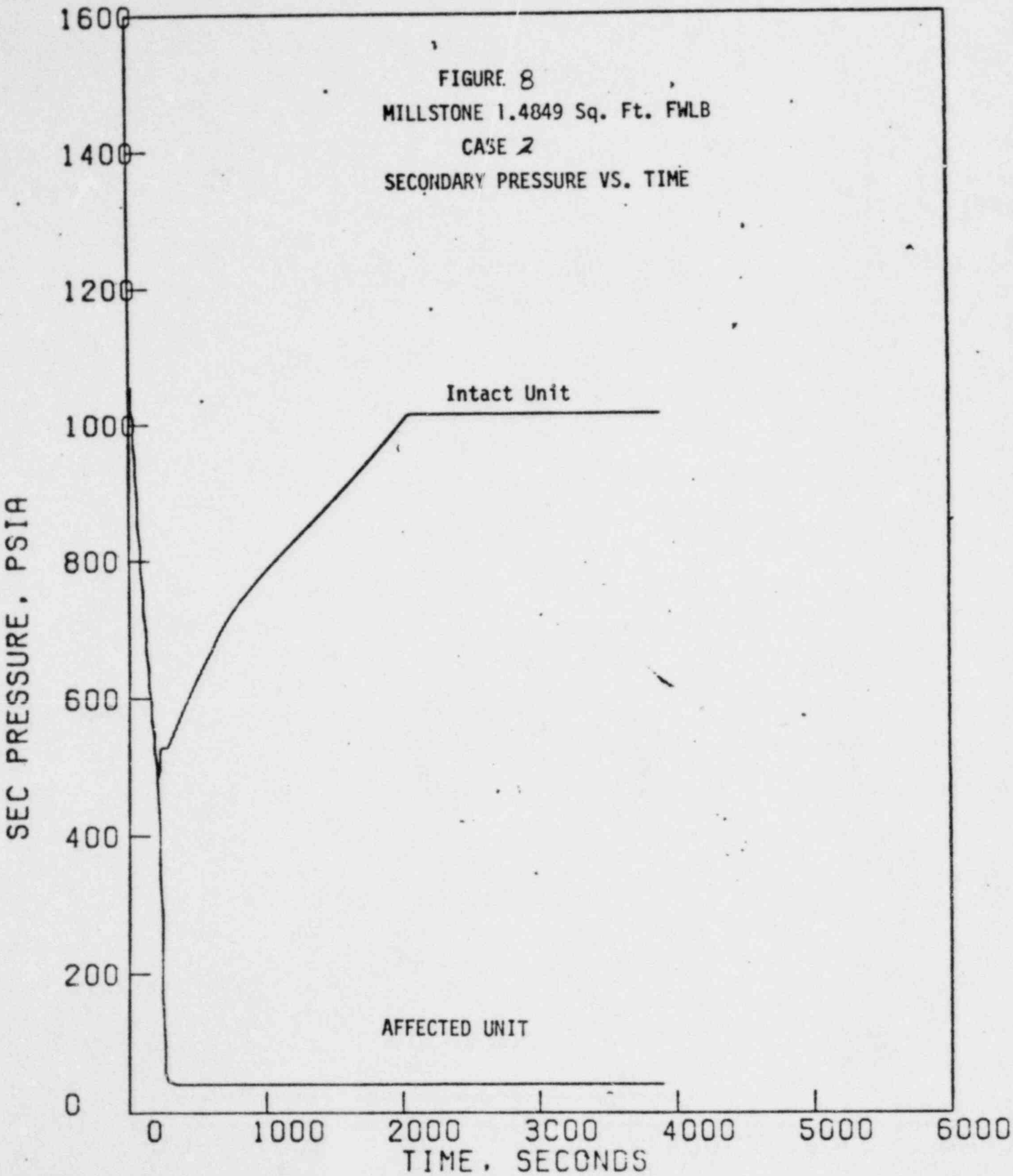


FIGURE 8
MILLSTONE 1.4849 Sq. Ft. FWLB
CASE 2
SECONDARY PRESSURE VS. TIME
Intact Unit
AFFECTED UNIT
SEC PRESSURE, PSIA
TIME, SECONDS
0
200
400
600
800
1000
1200
1400
1600
0
1000
2000
3000
4000
5000
6000

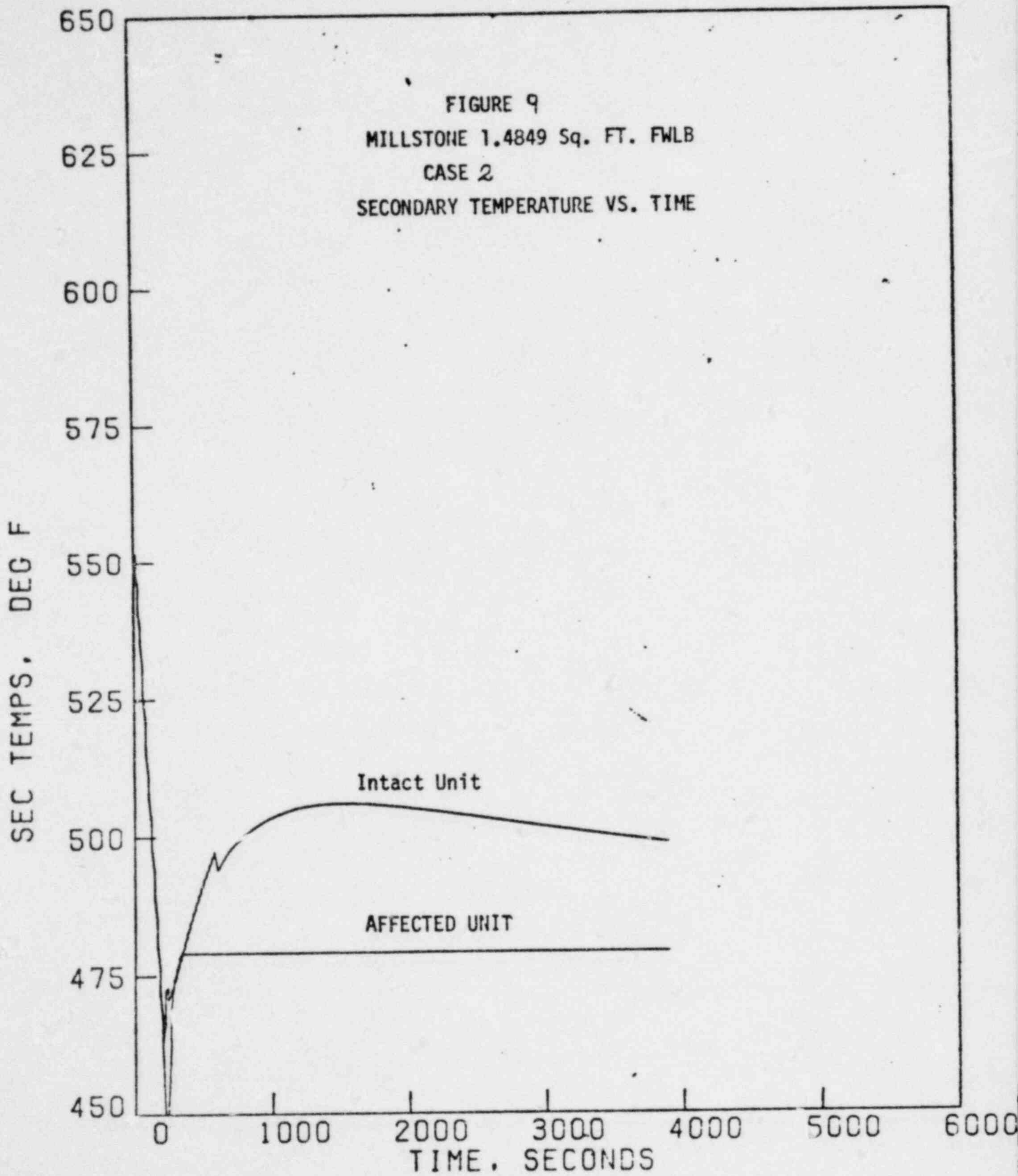


FIGURE 9
MILLSTONE 1.4849 Sq. FT. FWLB
CASE 2
SECONDARY TEMPERATURE VS. TIME
Intact Unit
AFFECTED UNIT
SEC TEMPS, DEG F
650
625
600
575
550
525
500
475
450
0
1000
2000
3000
4000
5000
6000
TIME, SECONDS

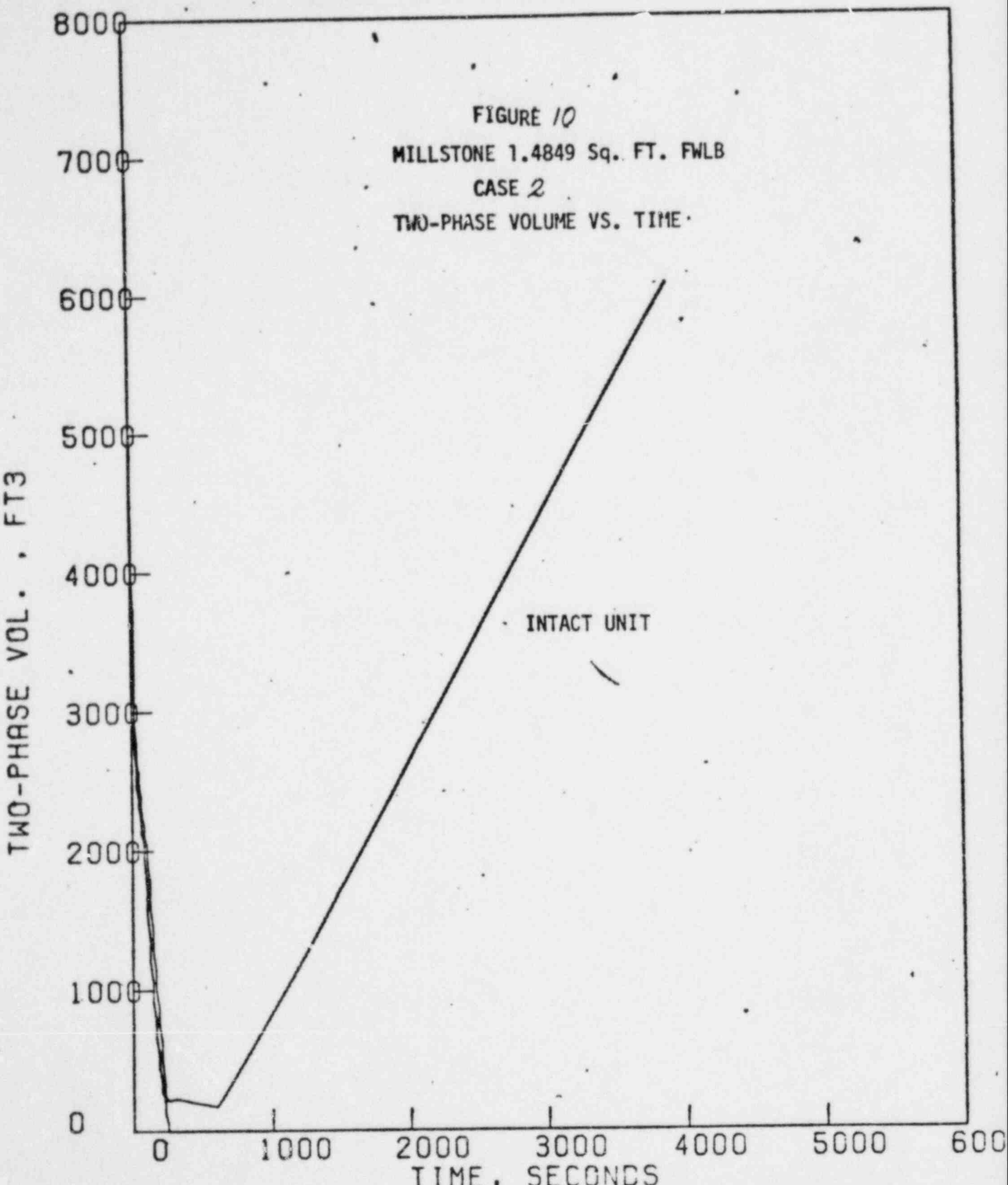


FIGURE 10
MILLSTONE 1.4849 Sq. FT. FWLB
CASE 2
TWO-PHASE VOLUME VS. TIME
INTACT UNIT
TWO-PHASE VOL. , FT3
TIME, SECONDS
8000
7000
6000
5000
4000
3000
2000
1000
0
0
1000
2000
3000
4000
5000
600

FIGURE #11
1.48 FT2 GUILLOTINE MFWLB - NO CONDENSATION
PZR WIDE RANGE PRESSURE (LTC CODE)