Omaha Public Power District
1623 HARNEY • OMAHA, NEBRASKA 68102 • TELEPHONE 536-4000 AREA CODE 402

January 10, 1980

Director of Nuclear Reactor Regulation
ATTN: Mr. Robert W. Reid, Chief
Operating Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

References: (1) Docket No. 50-285
(2) NRC Letter from Robert W. Reid to W. C. Jones Regarding Automatic Initiation of Auxiliary Feedwater Systems at Fort Calhoun, Dated December 21, 1979

Gentlemen:

The referenced letter requested Fort Calhoun response to a concern regarding the applicability of the current analysis for main steam line break and requested Fort Calhoun to resolve this concern by submitting an analysis within twenty (20) days from the receipt of that letter.

The attachments to this letter provide the NRC staff with sufficient information to resolve this concern. The attachments to this letter include the following:

1. Best Estimate MSLB Analysis to Assess NSSS and Containment Response With Automatic Auxiliary Feedwater Actuation

2. Evaluation of the Impact of Automatic Initiation of Auxiliary Feedwater on MSLB Analysis

Based on the best estimate analysis provided, it is our belief that the control grade system proposed presents a safe means of operation. It is also our belief that current means of manual actuation of auxiliary feedwater is also safe. We do not share your opinion that the questions in the referenced letter are applicable to the manual mode of operation, since operators are directed to isolate the affected steam generator on a steam line break.

In addition to the best estimate analysis, the effect of automatically initiating auxiliary feedwater during MSLB was evaluated using licensing assumptions. The evaluation was based upon calculations computing return to power. The conclusions of our evaluations

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are that, with a three minute delay in the actuation of auxiliary feedwater, the conclusions of the MSLB events presented in the FSAR and subsequent reload submittals conservatively bound those with control grade automatic auxiliary feedwater system for return to power.

Sincerely,

W. C. Jones
Division Manager
Production Operations

WCJ/KJM/BJH:jmm

Attach.

cc: LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N. W.
Washington, D. C. 20036

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Enclosure (2) a to
CE-18074-706

BEST ESTIMATE MSLB ANALYSIS TO ASSESS NSSS AND CONTAINMENT RESPONSE WITH AUTOMATIC AUXILIARY FEEDWATER ACTUATION

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1.0 INTRODUCTION

A set of calculations has been performed on a generic basis with plant characteristics representative of CE operating plants to model containment building pressure and temperature response and overall NSSS behavior, including core reactivity, following a Main Steam Line Break (MSLB) inside containment. The intent of these calculations is to determine if the containment building response (pressure) and the core reactivity response (return to power) are acceptable following a MSLB when auxiliary feedwater is added without regard to the identification of the affected steam generator. The auxiliary feedwater flow is assumed to be activated at the initiation of the transient to maximize its effects. Main feedwater flow including post trip rampdown is simulated. No isolation of main or auxiliary feedwater is considered unless a high water level condition is reached.

2.0 ASSUMPTIONS AND CASES

Assumptions for the analyses are given in Table 1. The four cases analyzed are listed in Table 2.

3.0 DISCUSSION OF RESULTS

Maximum containment pressure and least negative core reactivity for the four cases are listed in Table 3. Both the containment pressure and the reactivity (return to power) values are within acceptable limits.

Main feedwater flow, auxiliary feedwater flow, core reactivity change, core power, containment pressure, primary loop temperatures, and steam generator secondary temperatures for the four cases are detailed in Figures A-1 through A-7, B-1 through B-7, C-1 through C-7, and D-1 through D-7, respectively.

The results of the analyses using best estimate models for steam generator moisture carryover and containment passive heat sink heat transfer demonstrate that the additional auxiliary feedwater has a negligible impact on containment peak pressure. The containment peak pressure is determined primarily by the initial inventory in the ruptured unit. This

inventory is released within the first few minutes, depending upon the break size, so that the contribution of auxiliary feedwater flow to the ruptured unit over this time frame is small. Over the longer time frame, the secondary inventory is boiled off at essentially the decay heat rate which the containment active heat removal systems can accommodate while reducing containment pressure. The excess feedwater which is not boiled off remains in the steam generator, causing the secondary level to rise. The containment peak pressure is essentially an initial inventory limited phenomenon.

The results of the analyses also show that the additional auxiliary feedwater has a negligible impact on core reactivity. Cases A and C assume no stuck rods and a best estimate moderator cooldown curve. For comparison, Cases B and D assume that the most reactive rod is stuck and that the moderator cooldown curve is a licensing curve. All cases took credit for boron injection via three charging pumps; however, safety injection boron credit was not taken. These cases do not have a return to power for the following reason. The initial primary loop temperature decreases are limited by the two-phase blowdown process associated with large break (≥ 2 ft^2), since much of the break flow is saturated liquid which has not absorbed significant amounts of energy from the primary loop. For smaller break areas (<2 ft^2), the blowdown is pure steam which does require large amounts of energy per unit mass to boil via primary to secondary heat transfer; however, the rate of primary-to-secondary heat transfer is controlled by the blowdown flowrate which in turn is limited by the small break area. The net result is that over approximately the first 100 seconds of the event, the amount of core and loop cooldown is about the same regardless of break size. This time frame is most important since the presence of delayed neutrons minimizes the amount of cooldown needed to produce a core criticality problem.

Without a return to power (via primary loop cooldown and delayed neutrons), the remainder of the transient is a gradual increase in reactivity due to loop cooldown which is coupled to the containment pressure, plus a decrease in reactivity due to boron injection. In time (approximately 300 seconds), the reactivity decrease due to boration overtakes the reactivity increases due to loop cooldown; thereafter, the total reactivity steadily decreases. The ruptured steam generator is at the containment backpressure and with

RCPs operating the sensible heat from the non-ruptured unit is quickly removed resulting in RCS and SG secondary temperatures essentially in equilibrium with the containment conditions in about 10 minutes.

With licensing assumptions, the peak in the reactivity transient is calculated to be within the first two minutes of the event. A three minute time delay, if added to the automatic actuation circuit, would justify a statement that automatic auxiliary feedwater actuation will not impact existing SAR core cooldown MSLB analyses.

4.0 COMPARISON WITH LICENSING CALCULATIONS

The following items are important in comparing the results contained herein with those obtained with traditional licensing models and assumptions:

1. The moisture carryover model used is a best estimate model which gives a two-phase blowdown for large break areas. The two-phase blowdown results in a lower containment pressure and less initial primary loop cooldown than a pure steam blowdown. Chapter 15 analyses assume a pure steam blowdown regardless of break size.
2. Chapter 15 analyses assume that the most reactive rod is stuck. Moreover, the remaining rod worth is assigned a conservative value in conjunction with a conservative moderator cooldown curve.
3. A best estimate containment heat transfer model provides containment pressurization results significantly lower than those provided in Chapter 6 analyses.

TABLE I

ASSUMPTIONS

NSSS Initial Conditions

Power	2700 MWt
Core Inlet Temperature	548°F
Primary Pressure	2250 PSIA
Secondary Pressure	875 PSIA
Secondary Temperature	529°F

Containment Data

Free Volume	2.5×10^6 ft^3
Design Pressure	44 psig
Heat Sinks	SAR values
Heat Transfer Model	Best estimate model
Number of Fan Coolers	4 (no single failure)
Fan Cooler Capacity, each	68×10^6 B/hr at 280°F containment temperature 100°F CCW Temperature
Fan Cooler Actuation Setpoint	Fans are operational @ t = 0
Number of Sprays	2 (no single failure)
Spray rate, each	2700 GPM
Spray Actuation Setpoint	10 PSIG + 60 seconds

Other Data

Steam Generator Isolation Signal (MSIS) setpoint	500 psia
Decay Heat Curve	ANS-5
Main Feedwater Flow	
Ruptured Unit:	Ramped to 10% over 60 seconds following Reactor Trip: (10% represents twice the bypass nominal value of 5%, this accounts for pump run-out with reduced backpressure), temperature is reduced to 100 to account for turbine off-line. Flow terminated if the elevation of upper level tap is reached. See Figures A-1, B-1, C-1, and D-1.

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Main Feedwater Flow -- continued	
Unaffected Unit:	Same as ruptured unit except that flow is ramped to 5%. See Figures A-1, B-1, C-1 and D-1.
Auxiliary Feedwater Flow	
Ruptured Unit:	Initiated at t = 0. Flow rate is a function of unit pressure. All control valves assumed to be fully opened.
Unaffected Unit:	No flow; all flow is totally diverted to the ruptured unit.
Reactor Coolant Pumps	Operating during the transient.
CEA Insertion Worth	
All rods in (ARI)	-8.9% (no stuck rod)
Most reactive rod stuck	-7.12% (best estimate)
Moderator Worth	
SAR Value	See Figure 1
Best Estimate Value	See Figure 2
Doppler Worth	See Figure 3
Moisture Carryover On Steam Generator Secondary Side	Best Estimate Model
Boron Injection Parameters	
Safety Injection	Credit Not Taken
Charging Pumps	
Number of Pumps	3
Flow Rate	44 GPM per pump
Actuation Time	SIAS
Boric Acid Concentration	8% by weight
Boron Worth	80 PPM/%
Boric Acid Conversion Factor	1749 PPM boron/% by weight boric acid
Mixing Model Used	Slug Flow Model
Loop Transit Time	10.5 seconds

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TABLE 2

CASES

Case	CEA Scram Worth (%)	Moderator Curve	Break Area (Ft^2)
A	-8.9	Figure 2	6.63(1)
B	-7.12	Figure 1	6.63(1)
C	-8.9	Figure 2	1.99(2)
D	-7.12	Figure 1	1.99(2)

(1) Double-ended severance of main steam line (two-phase blowdown).

(2) Largest break area corresponding to pure steam blowdown.

TABLE 3

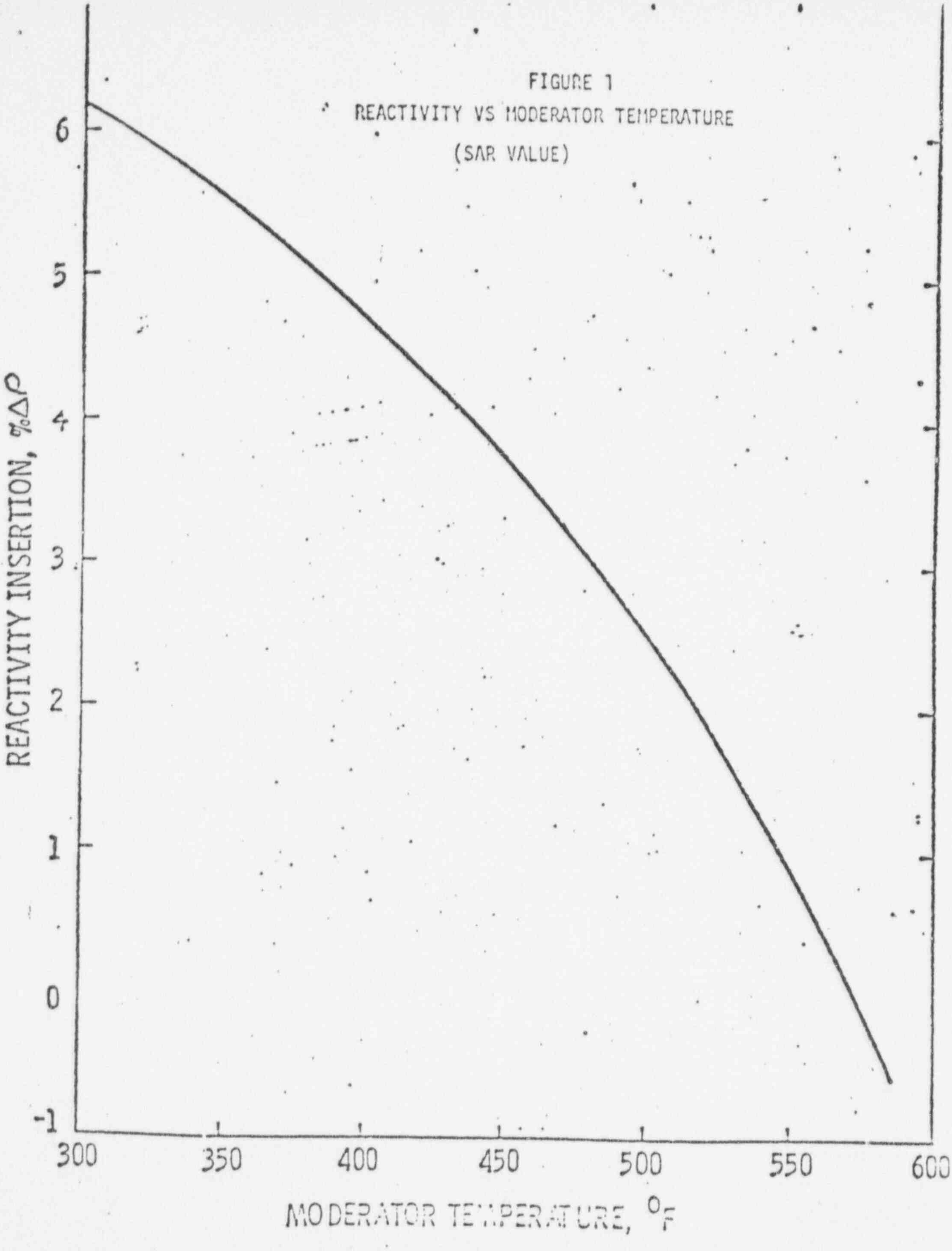
RESULTS

Case	Containment Peak Pressure (PSIG)	Least Negative Core Reactivity%
A	29.7/83.0 (sec.)	-4.31
B	29.7/83.0 (sec.)	-2.34
C	35.0/231.9 (sec.)	-3.54
D	35.0/231.9 (sec.)	-1.55

FIGURE 1

REACTIVITY VS MODERATOR TEMPERATURE

(SAR VALUE)



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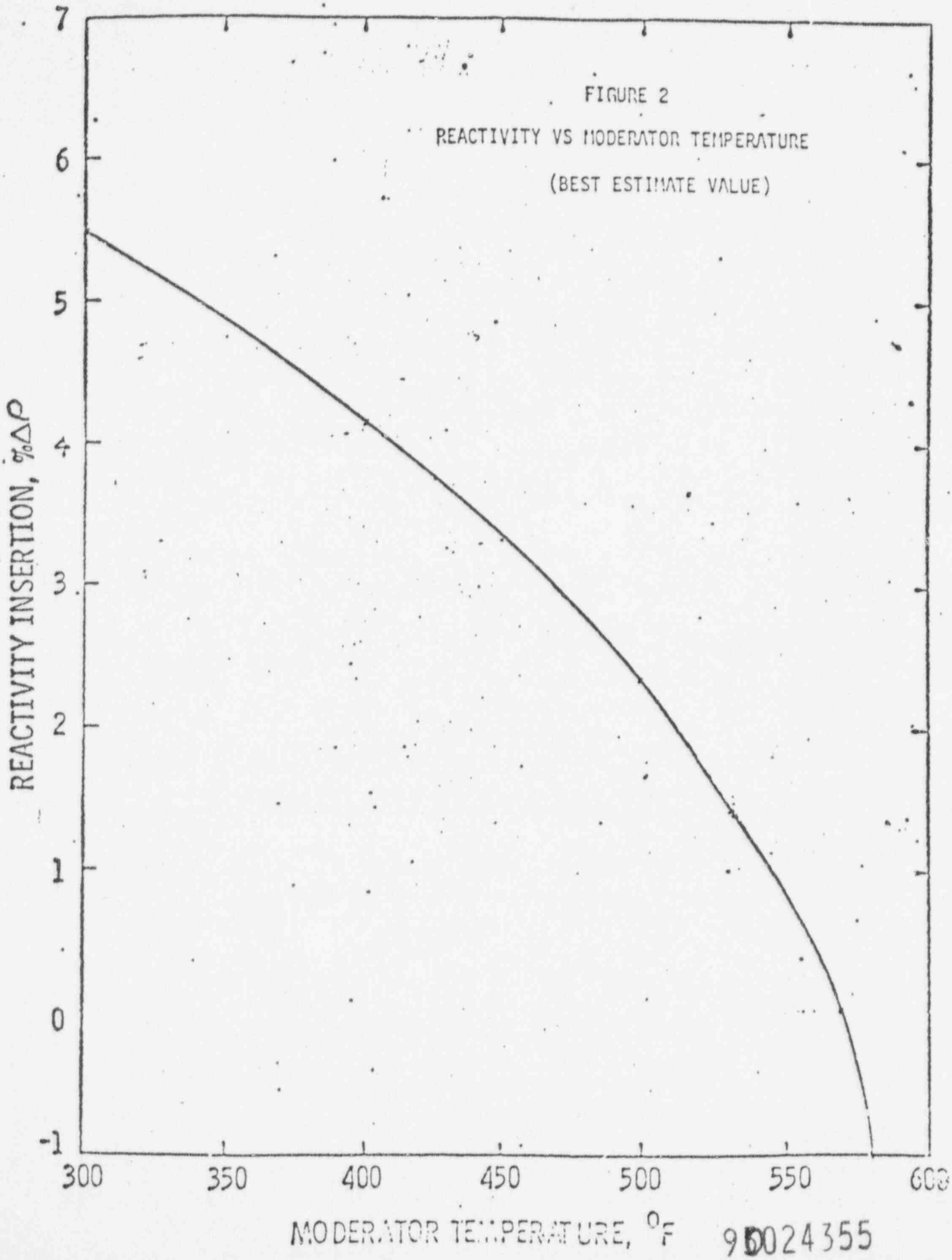
FIGURE 2
REACTIVITY VS MODERATOR TEMPERATURE
(BEST ESTIMATE VALUE)
REACTIVITY INSERTION, %Δρ
7
6
5
4
3
2
1
0
-1
300
350
400
450
500
550
600
MODERATOR TEMPERATURE, °F

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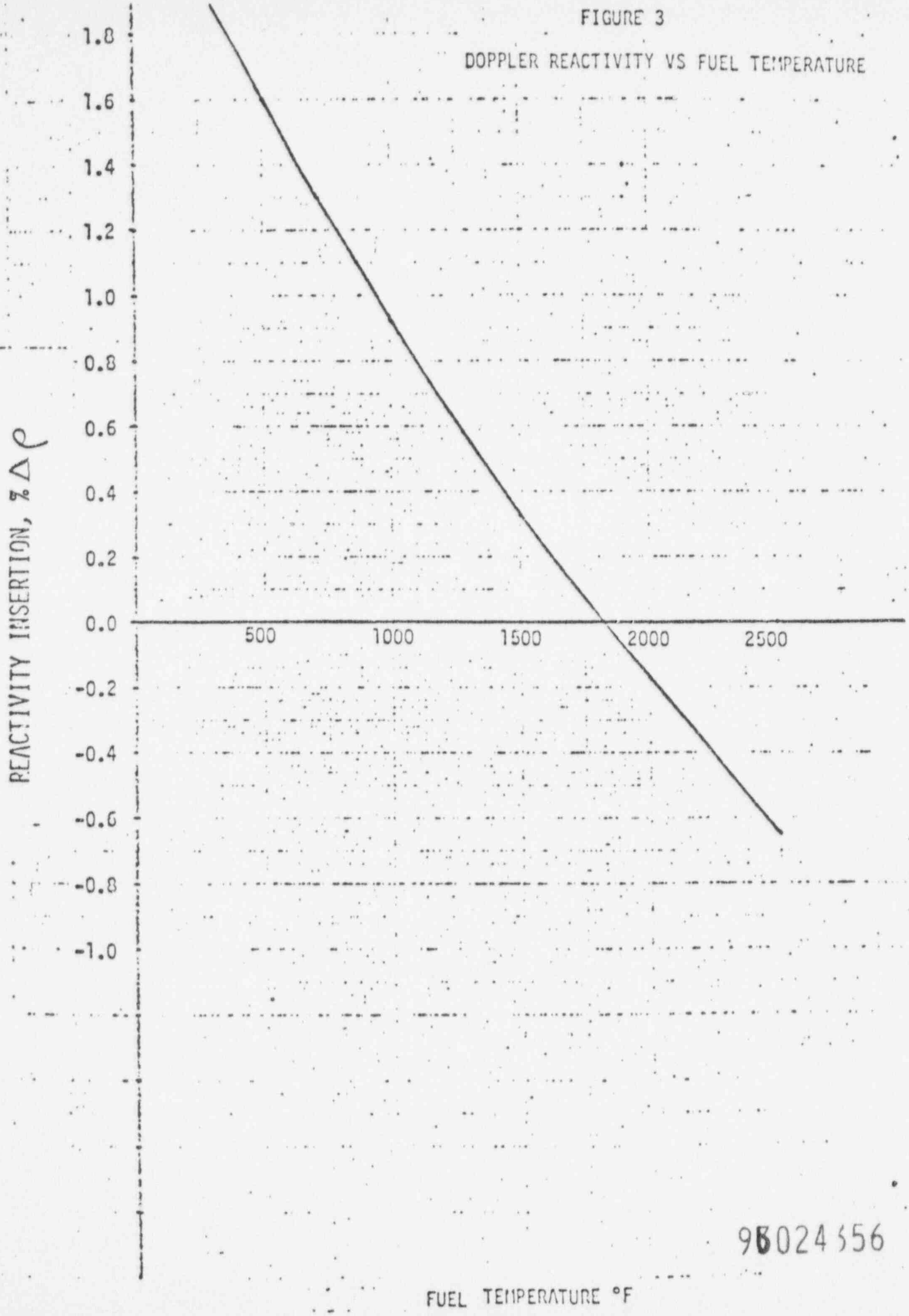
FIGURE 3

DOPPLER REACTIVITY VS FUEL TEMPERATURE



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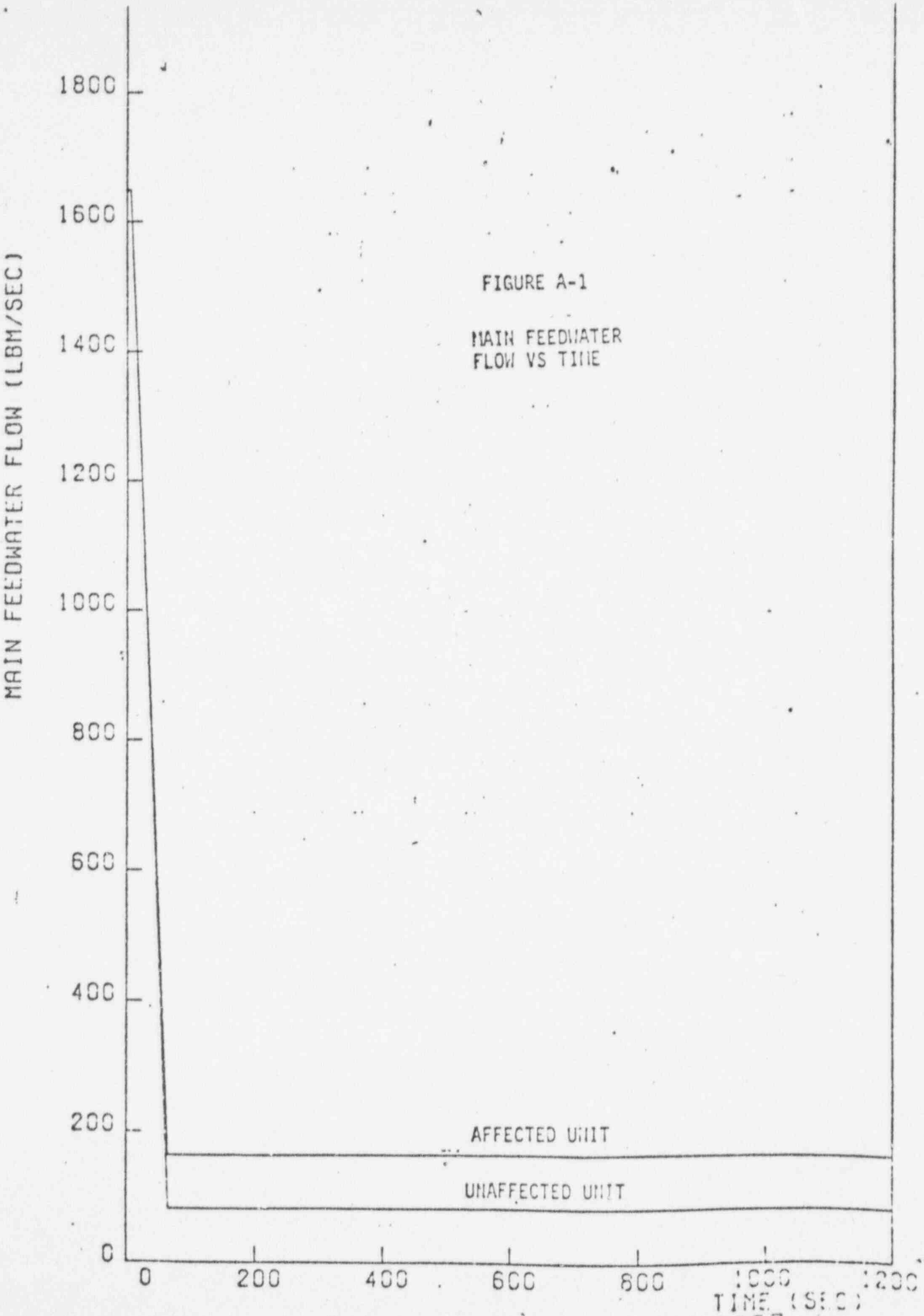
FIGURE A-1
MAIN FEEDWATER
FLOW VS TIME
MAIN FEEDWATER FLOW (LBM/SEC)
1800
1600
1400
1200
1000
800
600
400
200
0
AFFECTED UNIT
UNAFFECTED UNIT
0
200
400
600
800
1000
1200
TIME (SEC)

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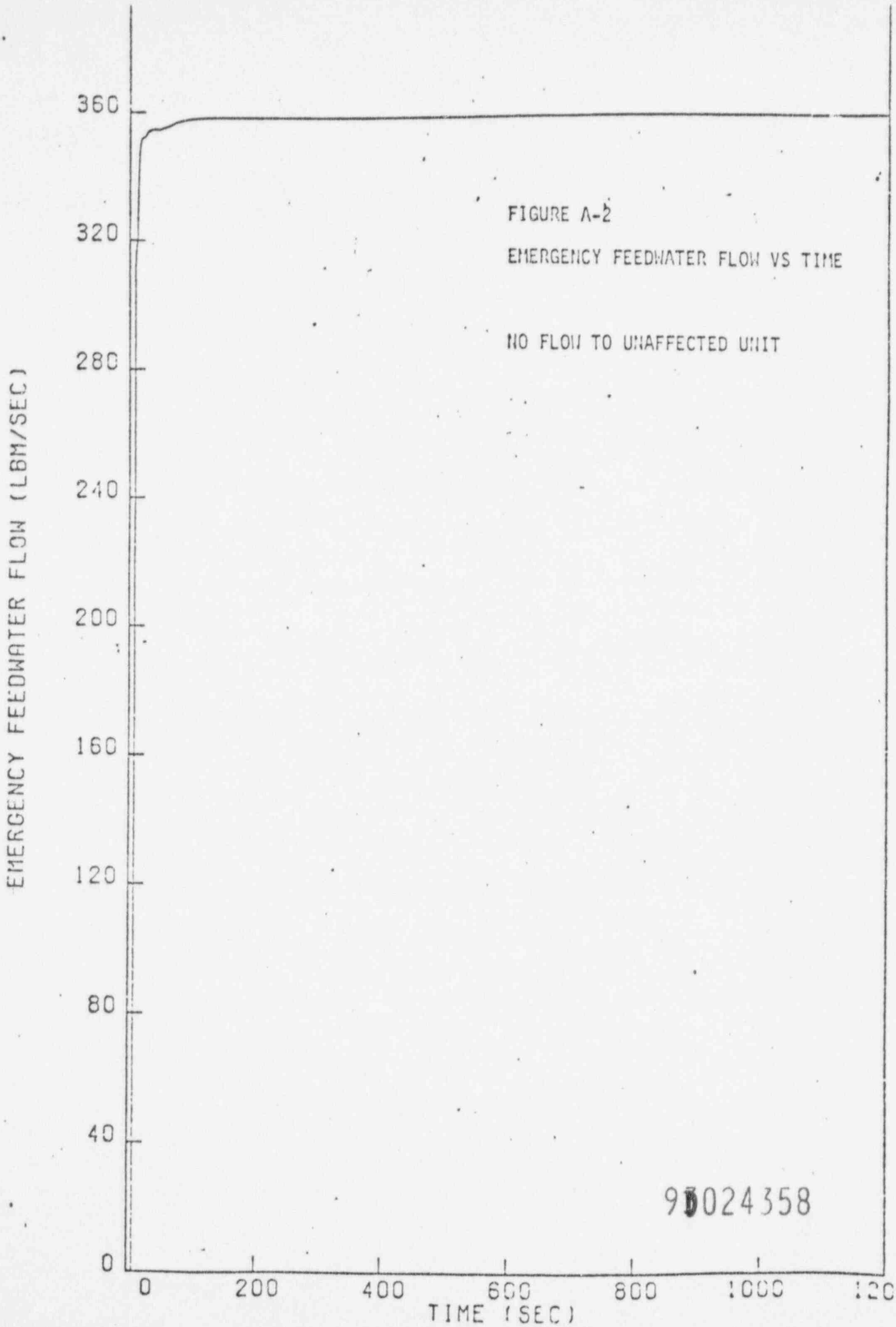

FIGURE A-2

EMERGENCY FEEDWATER FLOW VS TIME

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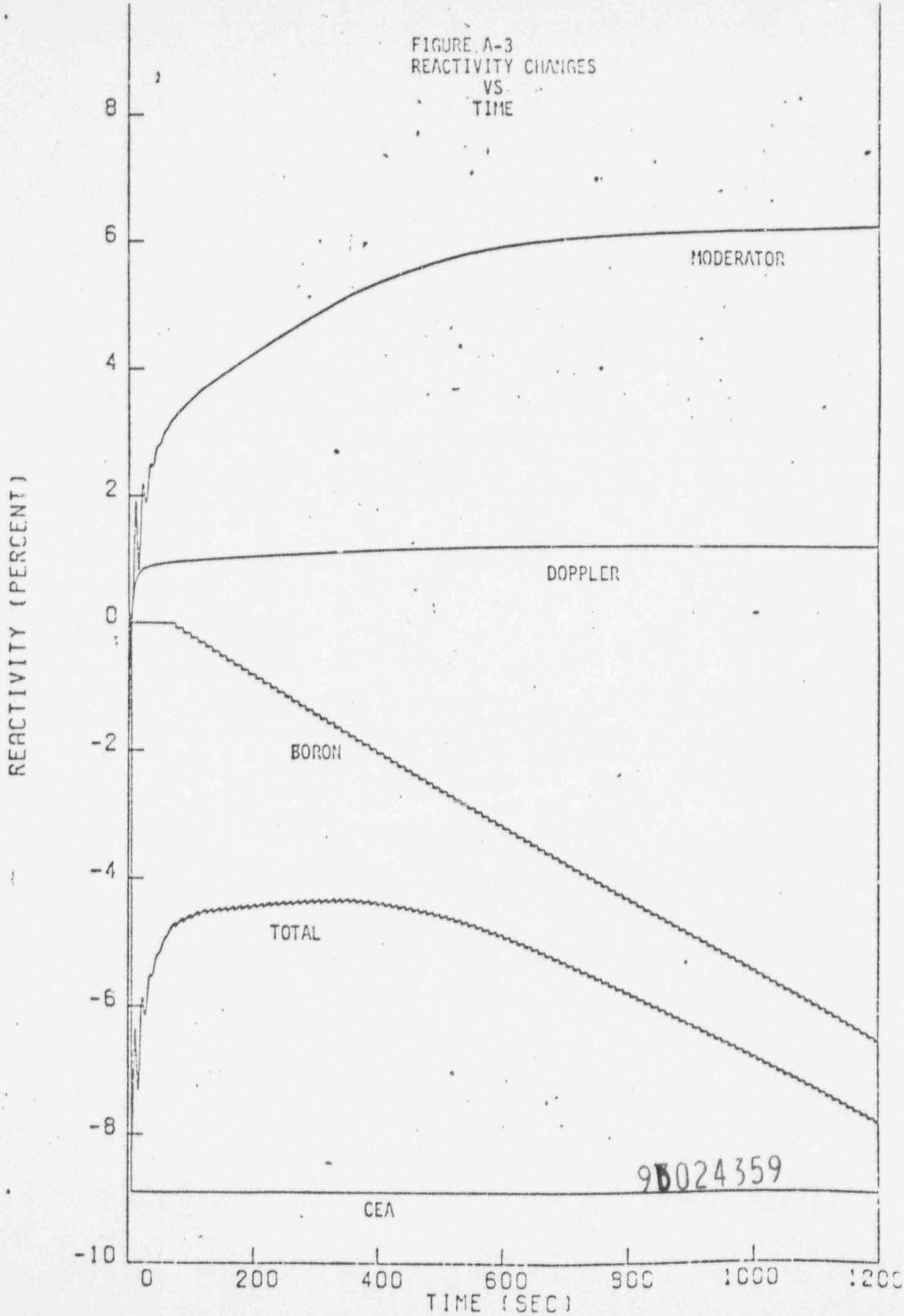
FIGURE A-3
REACTIVITY CHANGES
VS
TIME
REACTIVITY (PERCENT)
8
6
4
2
0
-2
-4
-6
-8
-10
MODERATOR
DOPPLER
BORON
TOTAL
CEA
0
200
400
600
800
1000
1200
TIME (SEC)
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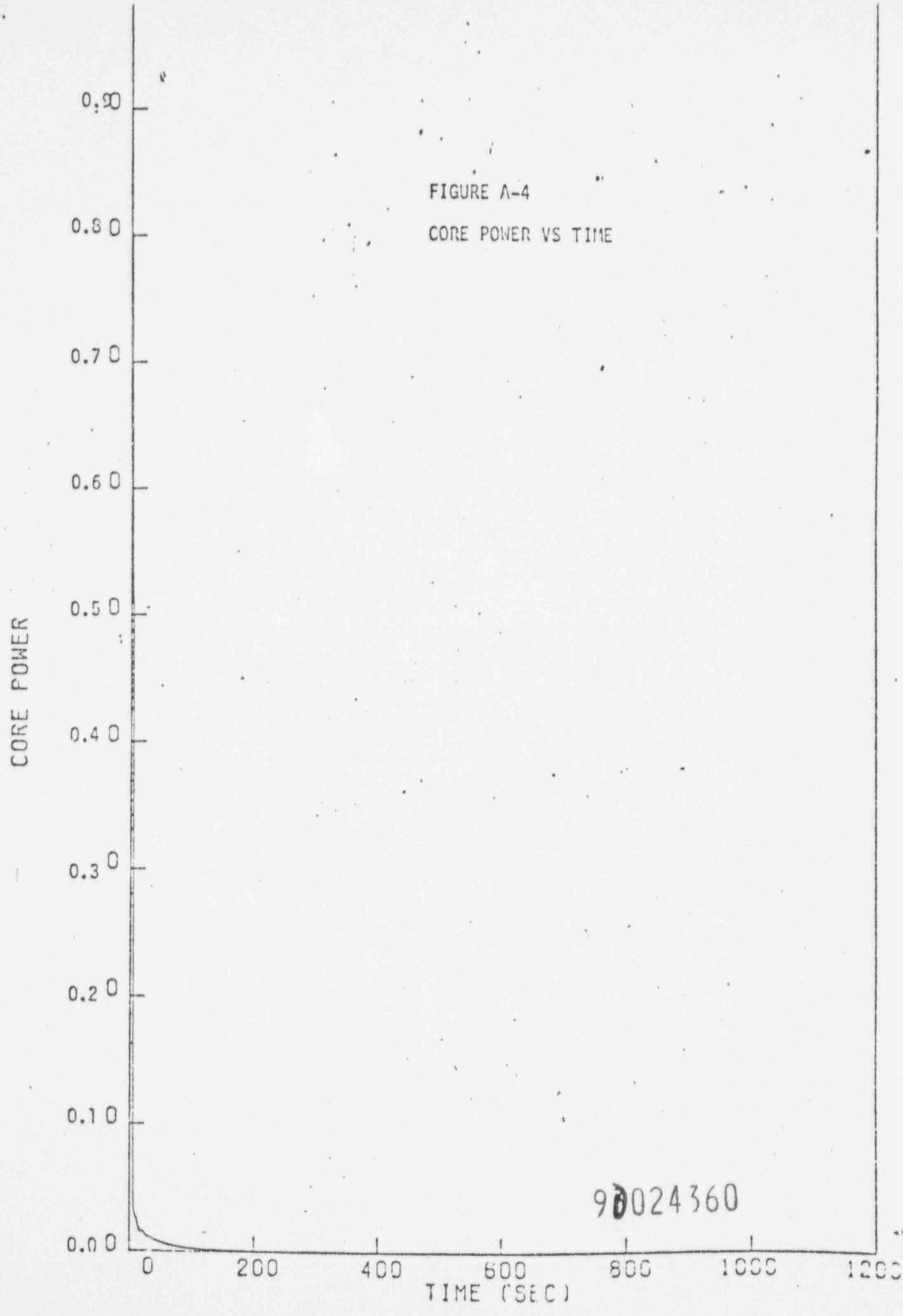
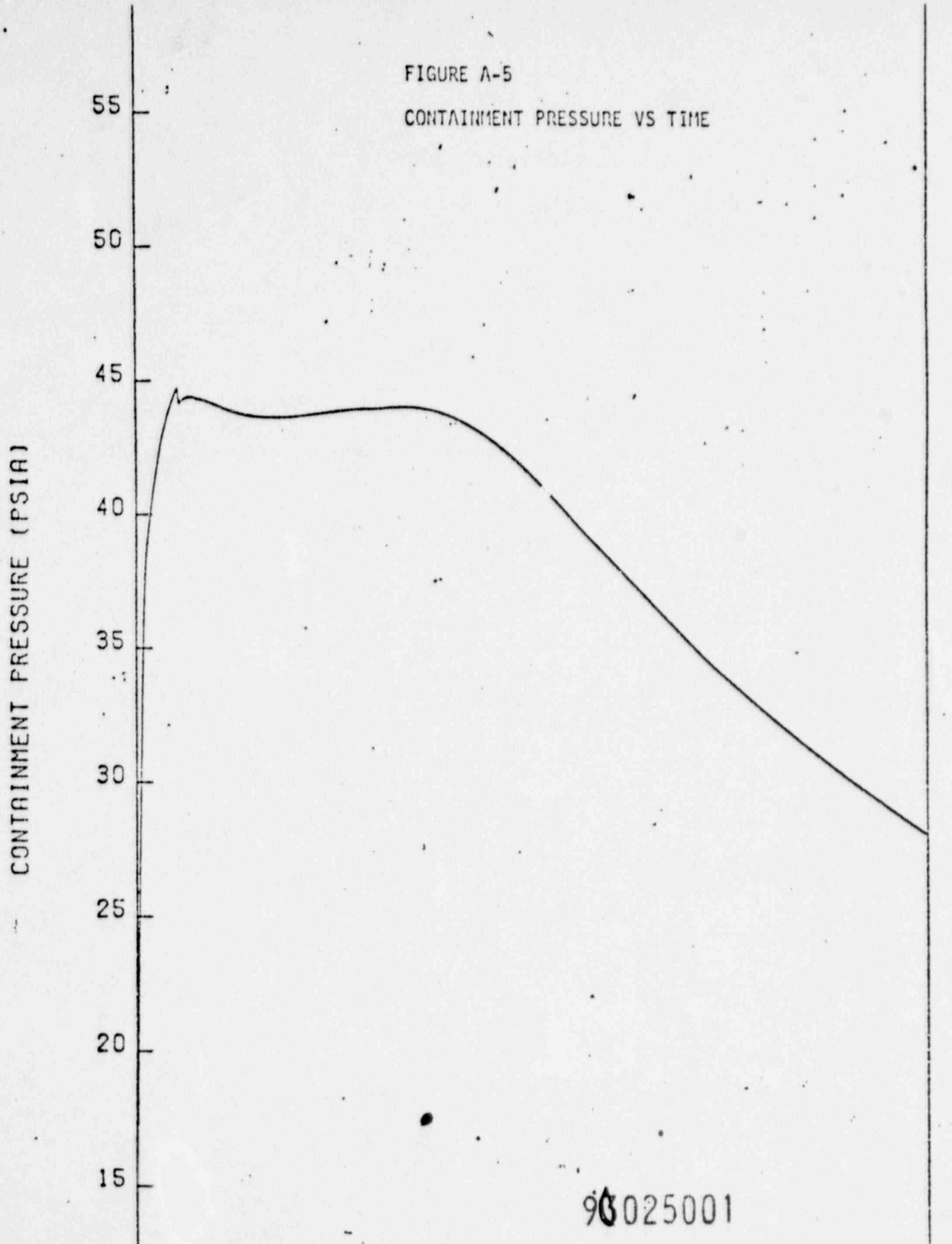


FIGURE A-4

CORE POWER VS TIME

FIGURE A-5

CONTAINMENT PRESSURE VS TIME



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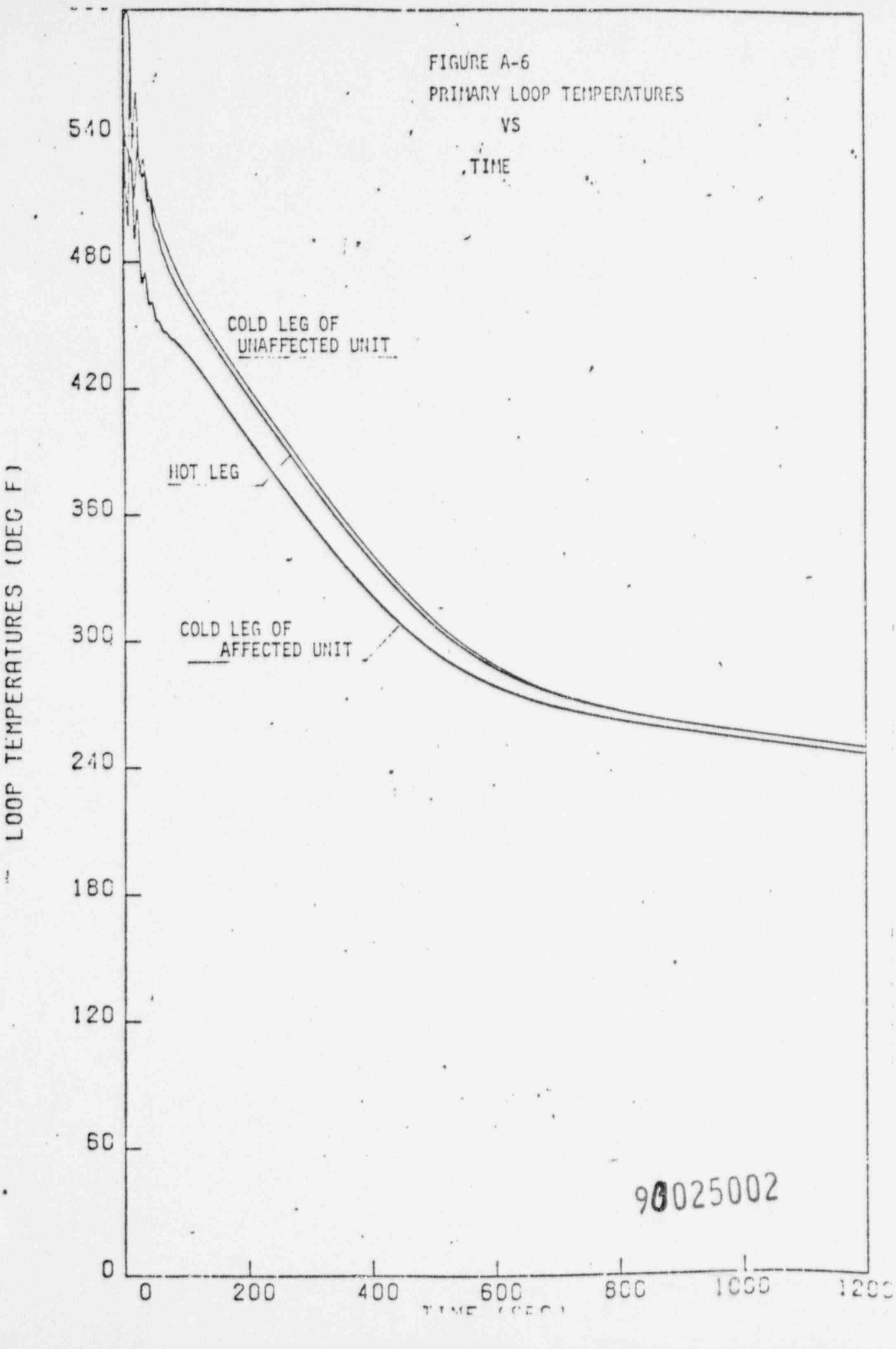

FIGURE A-6
PRIMARY LOOP TEMPERATURES
VS
TIME
COLD LEG OF
UNAFFECTED UNIT
HOT LEG
COLD LEG OF
AFFECTED UNIT
LOOP TEMPERATURES (DEG F)
540
480
420
360
300
240
180
120
60
0
0
200
400
600
800
1000

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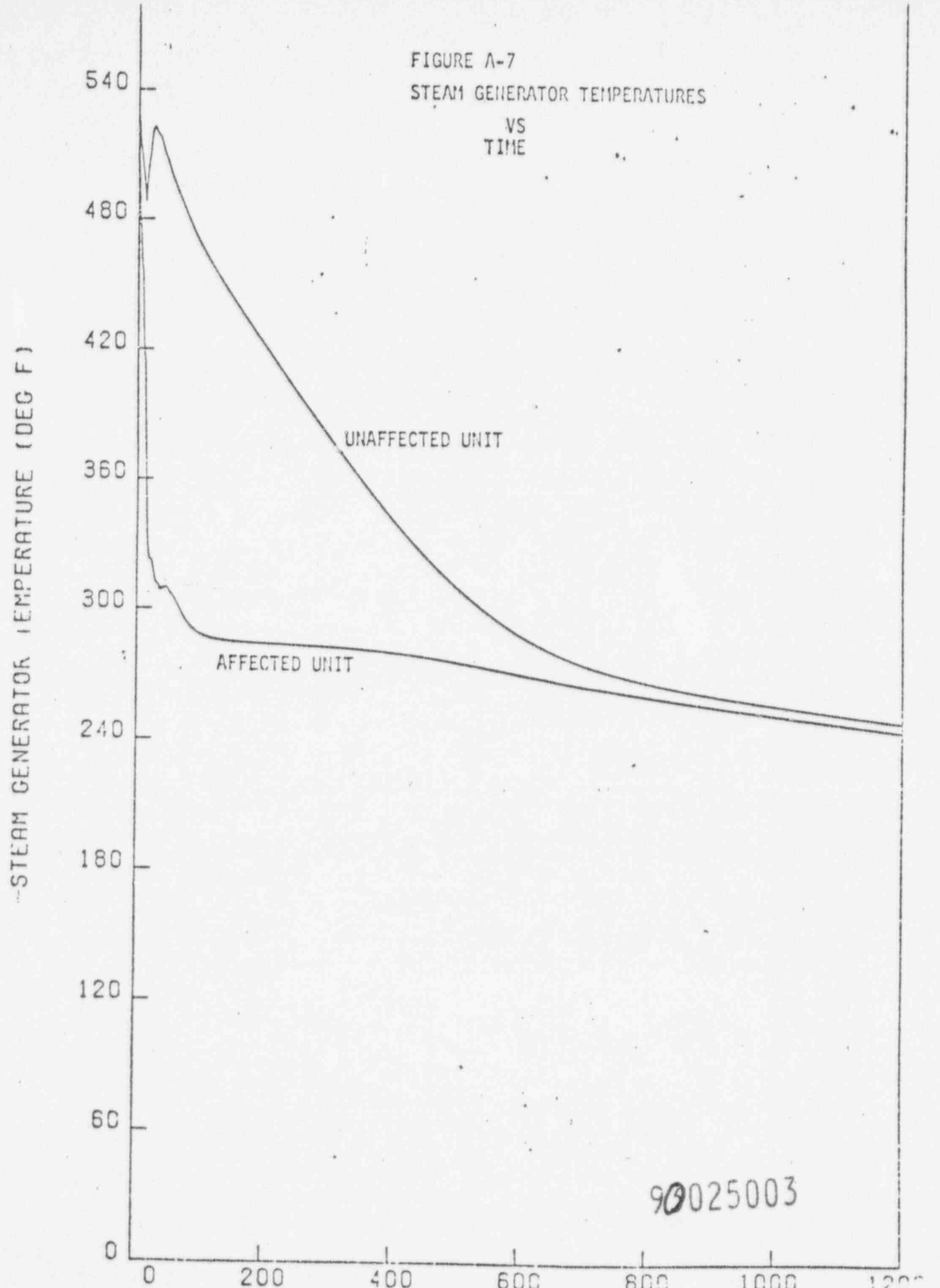

FIGURE A-7
STEAM GENERATOR TEMPERATURES
VS
TIME
STEAM GENERATOR TEMPERATURE (DEG F)
540
480
420
360
300
240
180
120
60
0
UNAFFECTED UNIT
AFFECTED UNIT
0
200
400
600
800
1000
1200
TIME (SEC)

9002500З

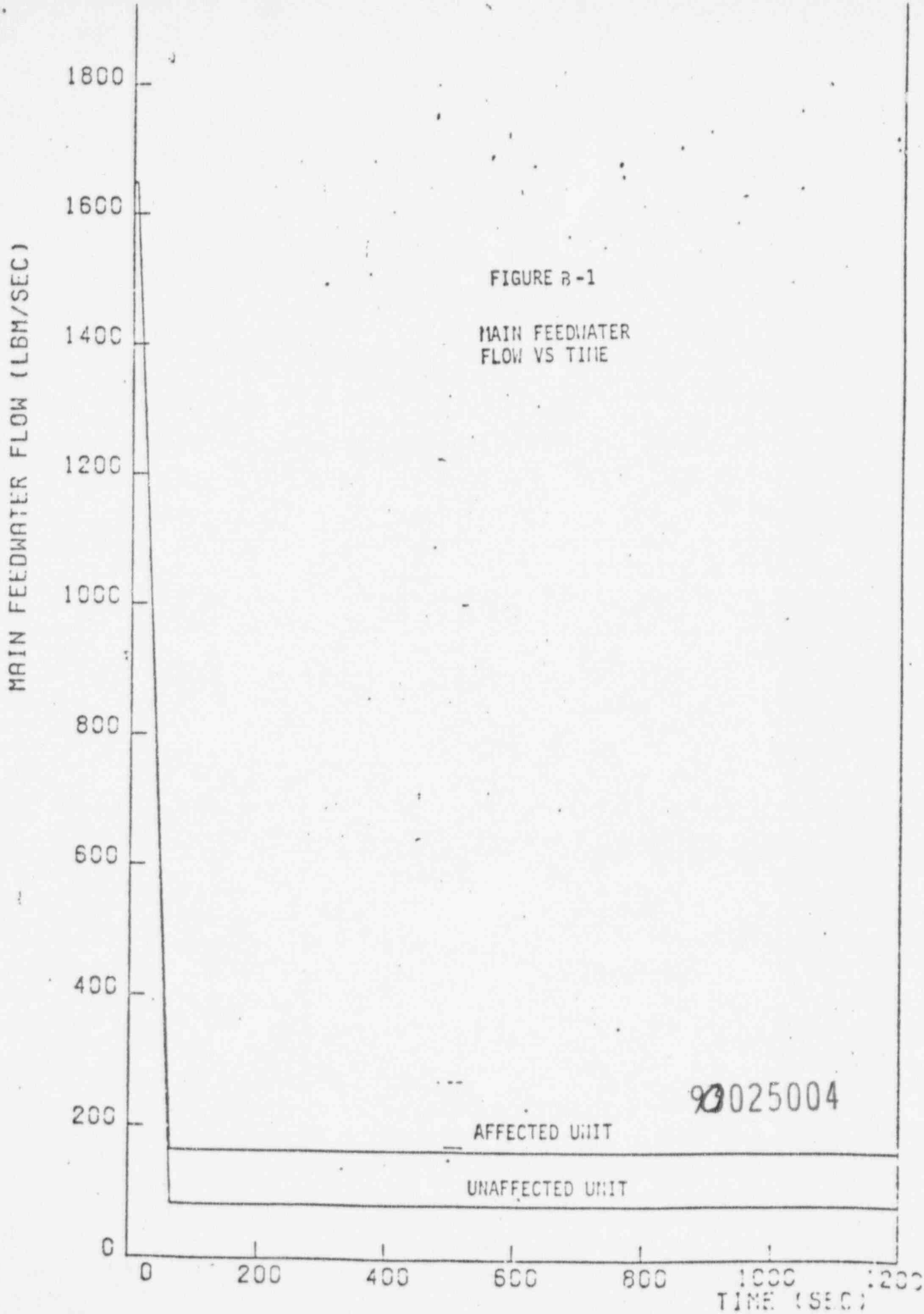
FIGURE B-1
MAIN FEEDWATER
FLOW VS TIME
MAIN FEEDWATER FLOW (LBM/SEC)
1800
1600
1400
1200
1000
800
600
400
200
0
AFFECTED UNIT
UNAFFECTED UNIT
0
200
400
600
800
1000
1200
TIME (SEC)

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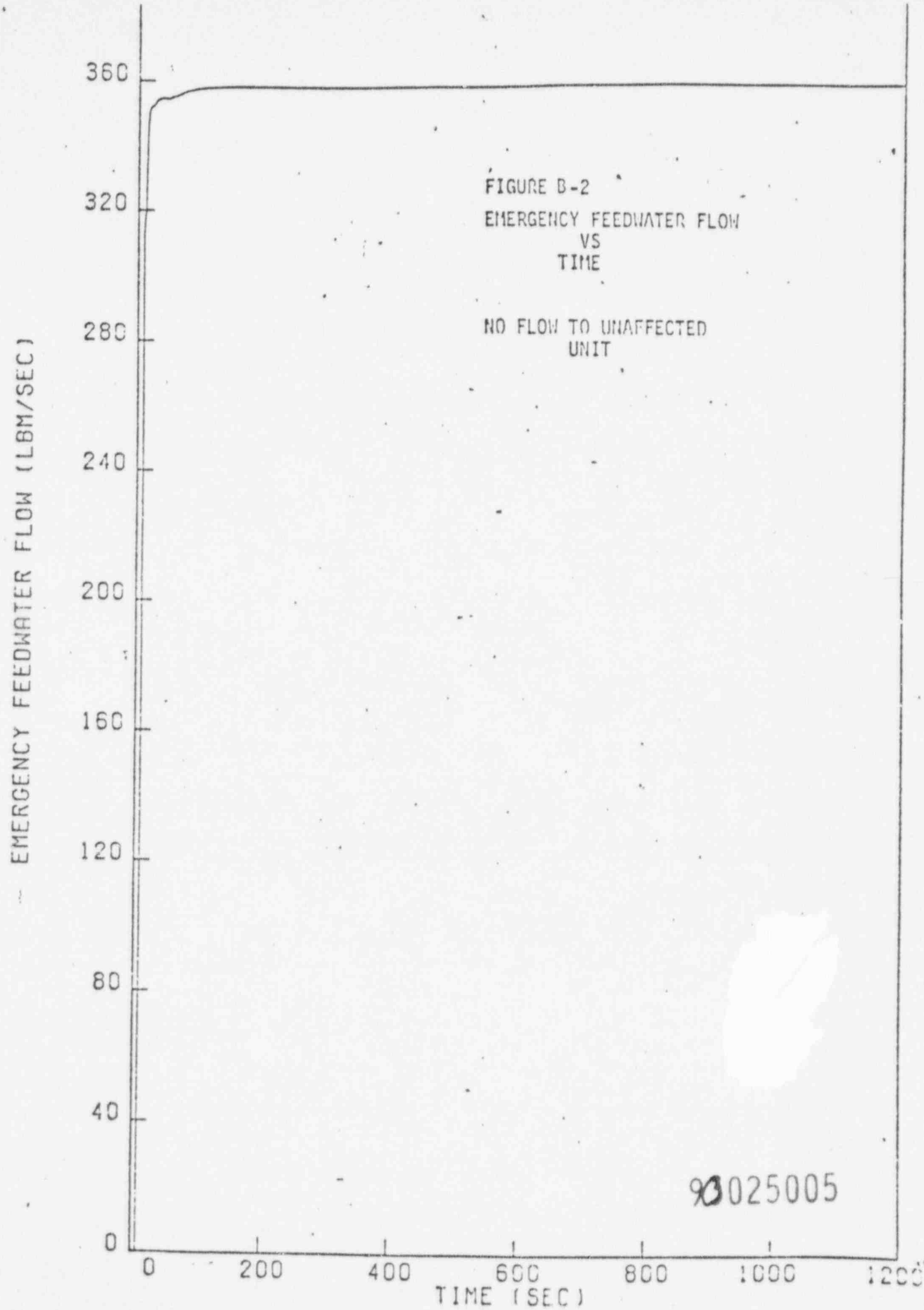
FIGURE B-2
EMERGENCY FEEDWATER FLOW
VS
TIME
NO FLOW TO UNAFFECTED
UNIT
EMERGENCY FEEDWATER FLOW (LBM/SEC)
360
320
280
240
200
160
120
80
40
0
0
200
400
600
800
1000
1200
TIME (SEC)

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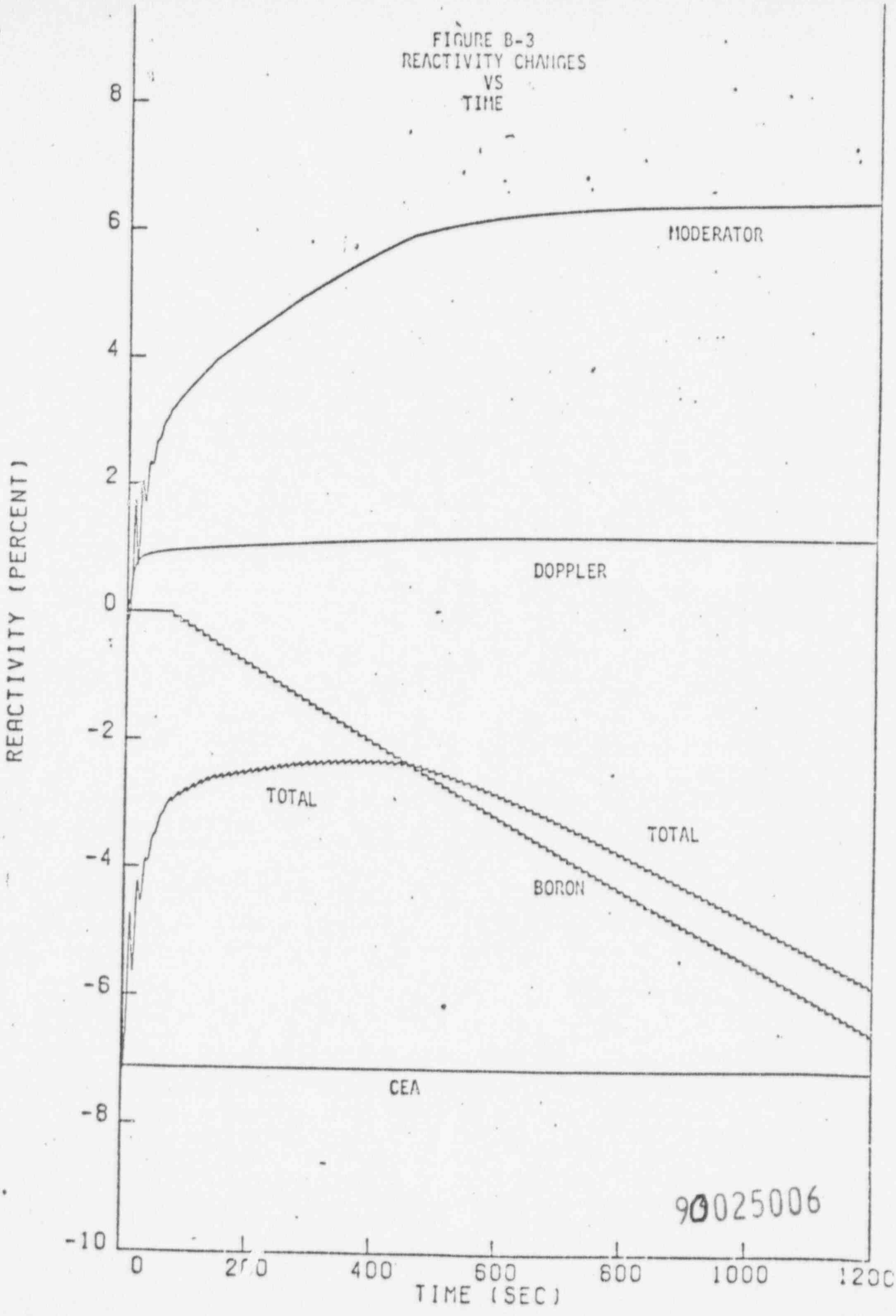
FIGURE B-3
REACTIVITY CHANGES
VS
TIME
MODERATOR
DOPPLER
TOTAL
TOTAL
BORON
CEA
REACTIVITY (PERCENT)
8
6
4
2
0
-2
-4
-6
-8
-10
0
400
600
800
1000
TIME (SEC)

9002500б

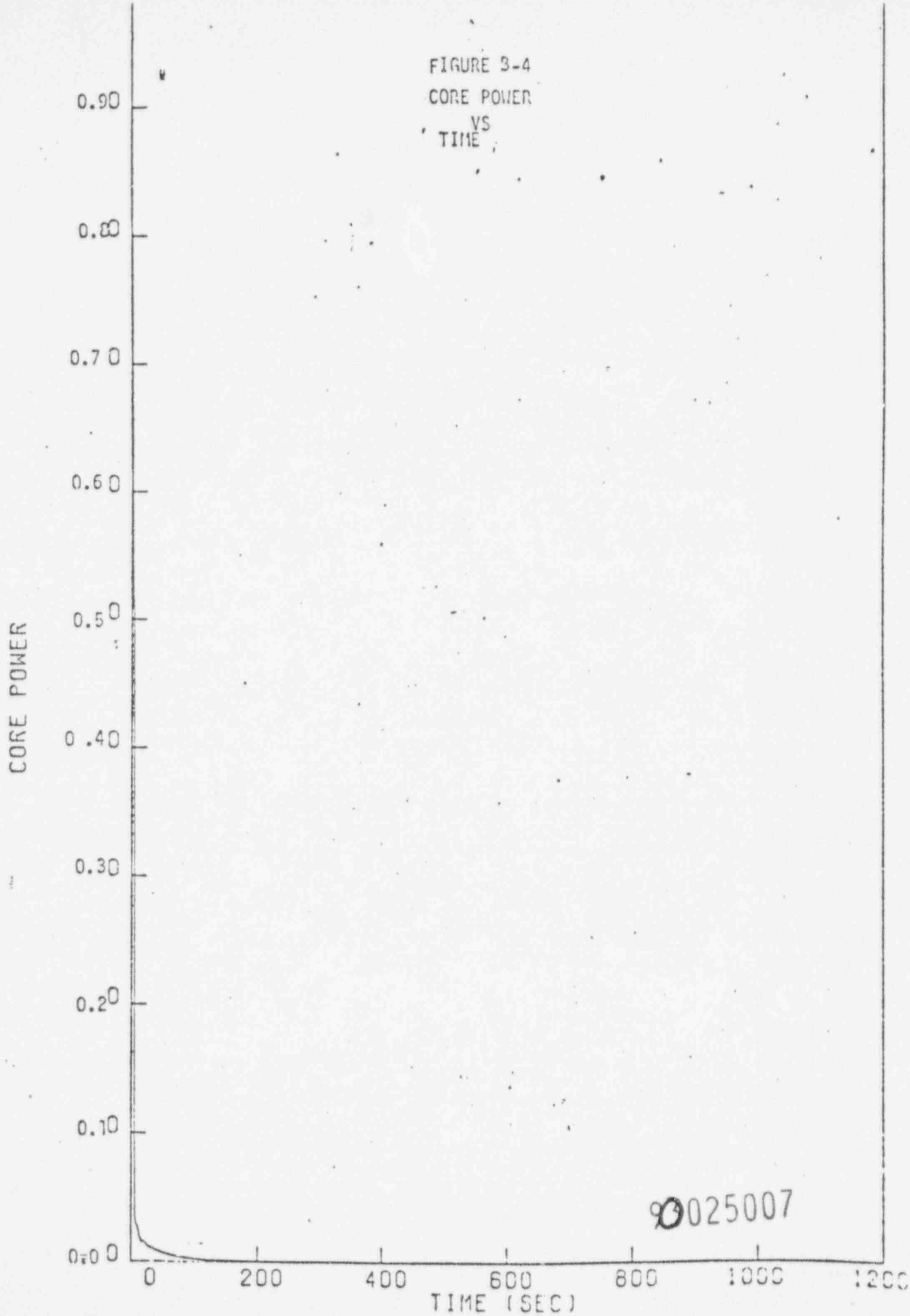
FIGURE 3-4
CORE POWER
VS
TIME
CORE POWER
0.90
0.80
0.70
0.60
0.50
0.40
0.30
0.20
0.10
0.00
0
200
400
600
800
1000
1200
TIME (SEC)
9002500 7

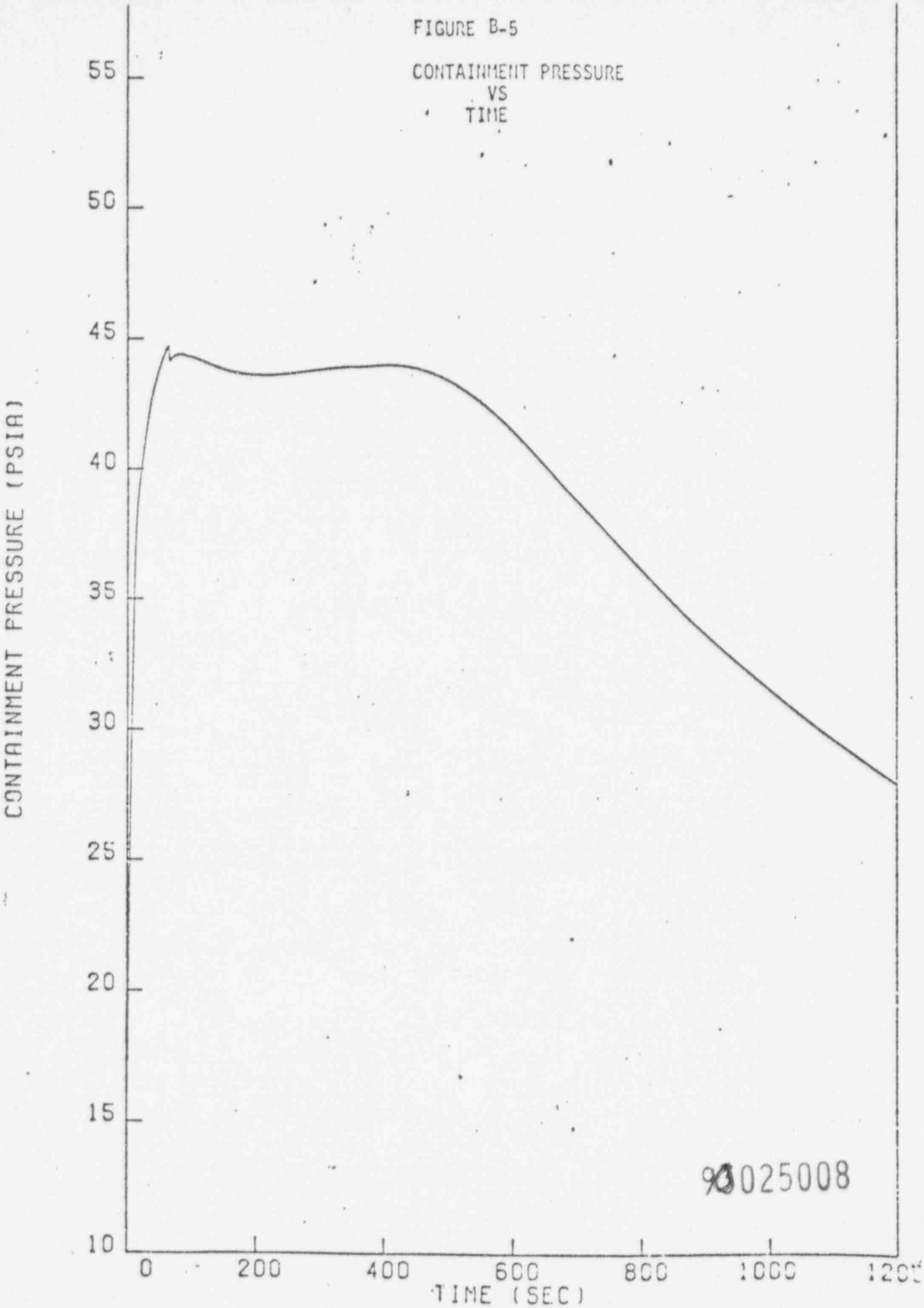

FIGURE B-5
CONTAINMENT PRESSURE
VS
TIME
CONTAINMENT PRESSURE (PSIA)
55
50
45
40
35
30
25
20
15
10
0
200
400
600
800
1000
TIME (SEC)
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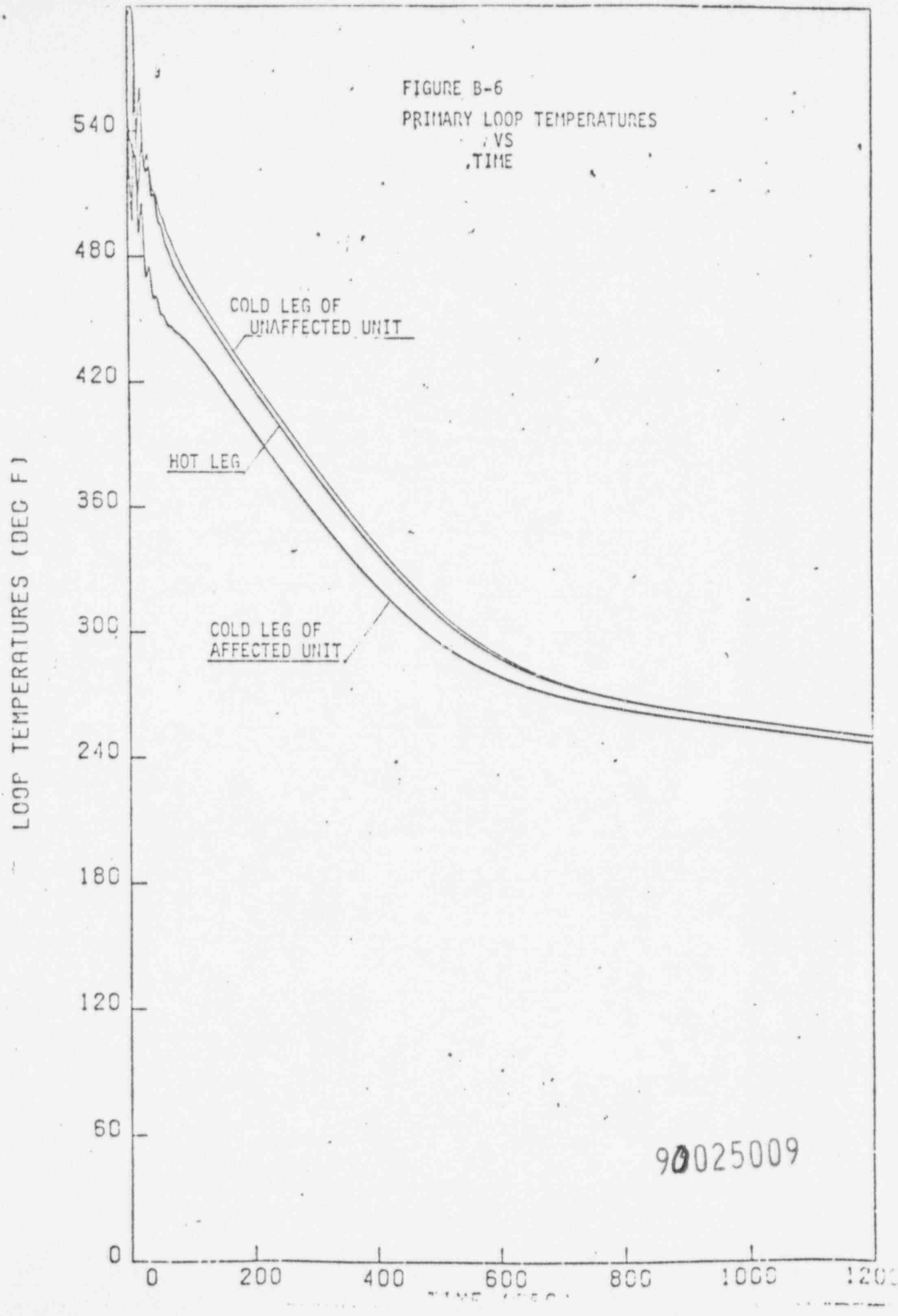

FIGURE B-6
PRIMARY LOOP TEMPERATURES
VS
TIME

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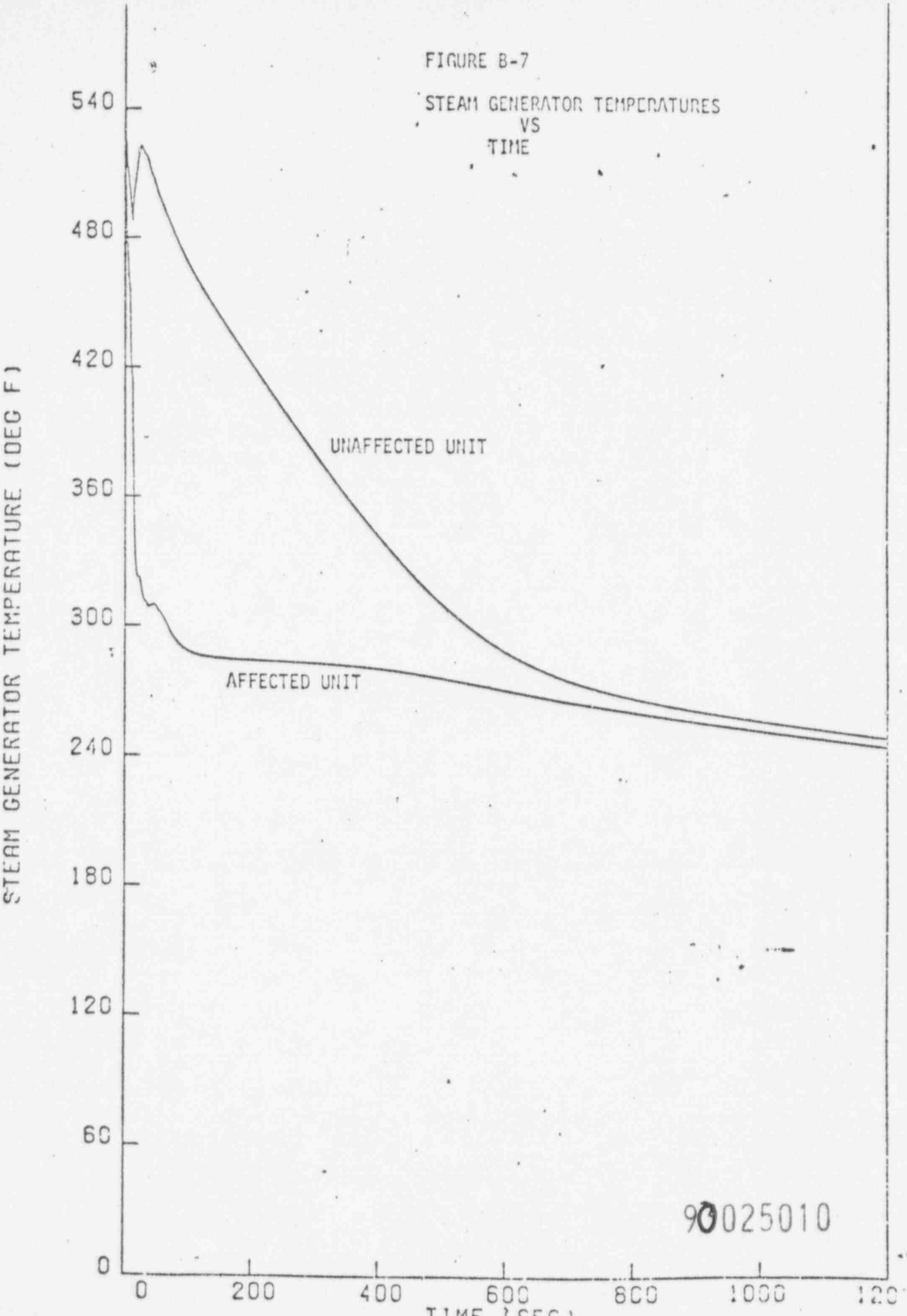
FIGURE B-7
STEAM GENERATOR TEMPERATURES
VS
TIME
UNAFFECTED UNIT
AFFECTED UNIT
STEAM GENERATOR TEMPERATURE (DEG F)
540
480
420
360
300
240
180
120
60
0
0
200
400
600
800
1000
TIME (SEC)
9002501O

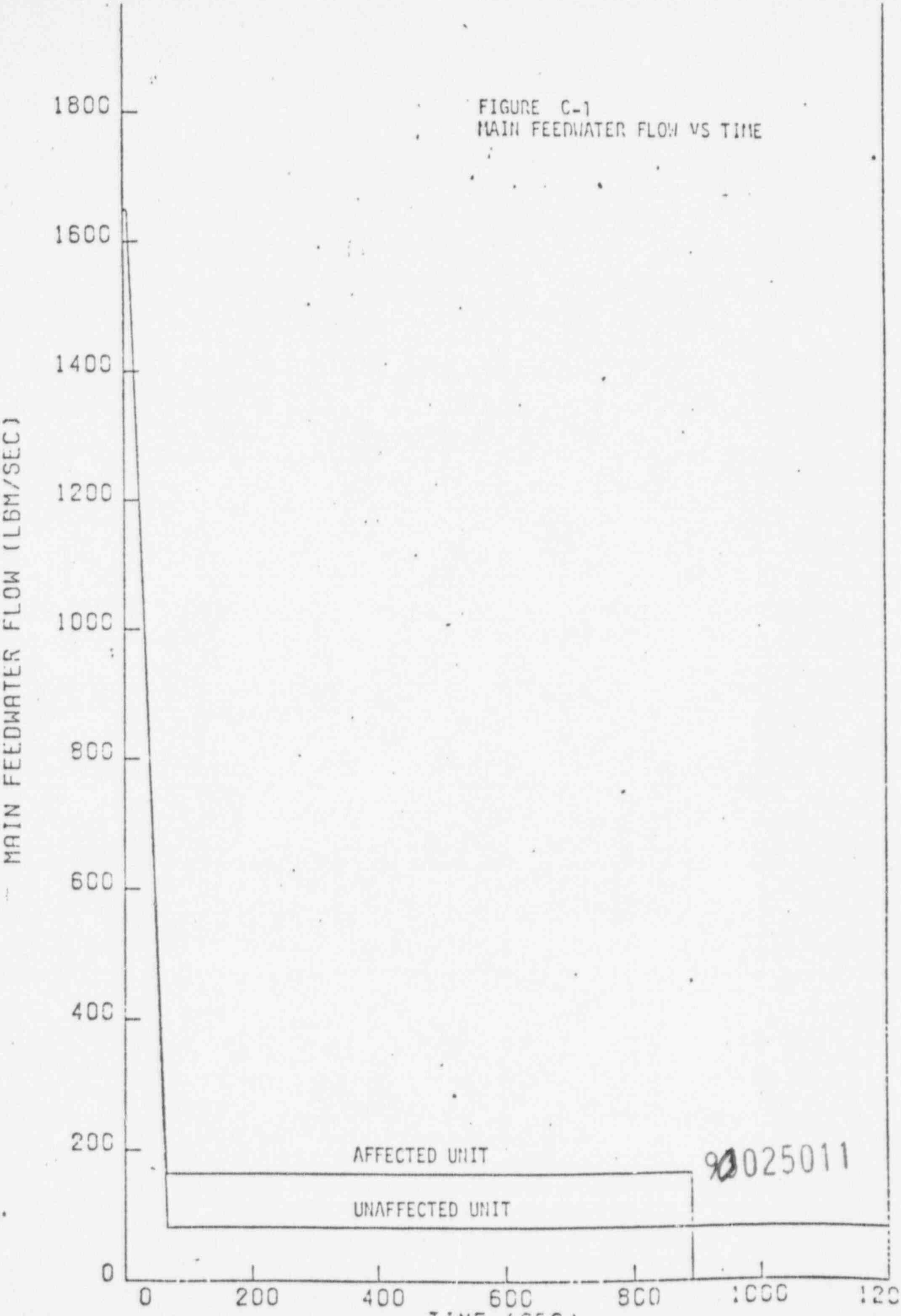
FIGURE C-1
MAIN FEEDWATER FLOW VS TIME
MAIN FEEDWATER FLOW (LBM/SEC)
1800
1600
1400
1200
1000
800
600
400
200
0
AFFECTED UNIT
UNAFFECTED UNIT
0
200
400
600
800
1000
1200
TIME (SEC)
90025011

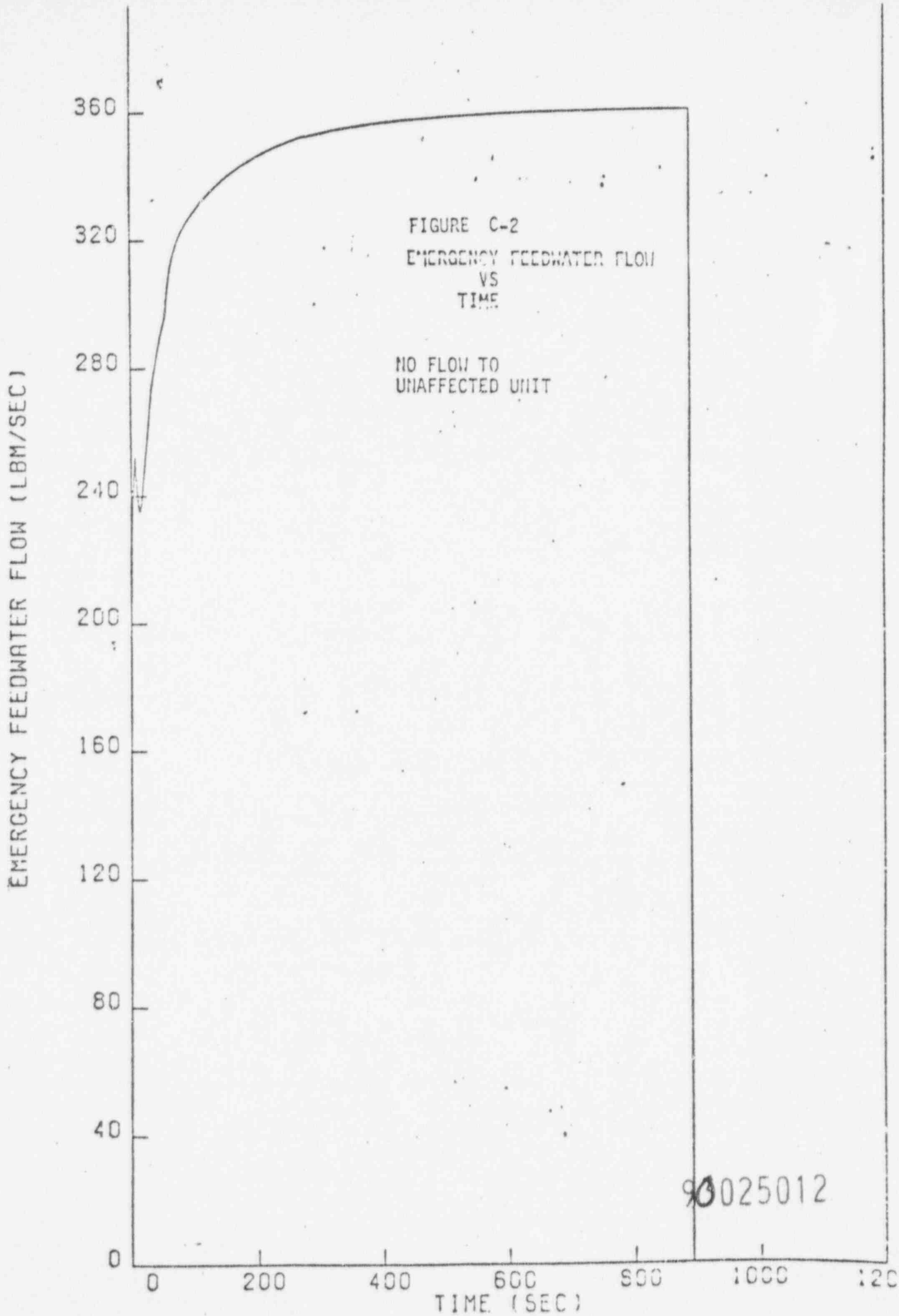
FIGURE C-2
EMERGENCY FEEDWATER FLOW
VS
TIME
NO FLOW TO
UNAFFECTED UNIT
EMERGENCY FEEDWATER FLOW (LBM/SEC)
TIME (SEC)
0
40
80
120
160
200
240
280
320
360
0
200
400
600
800
1000
90025012

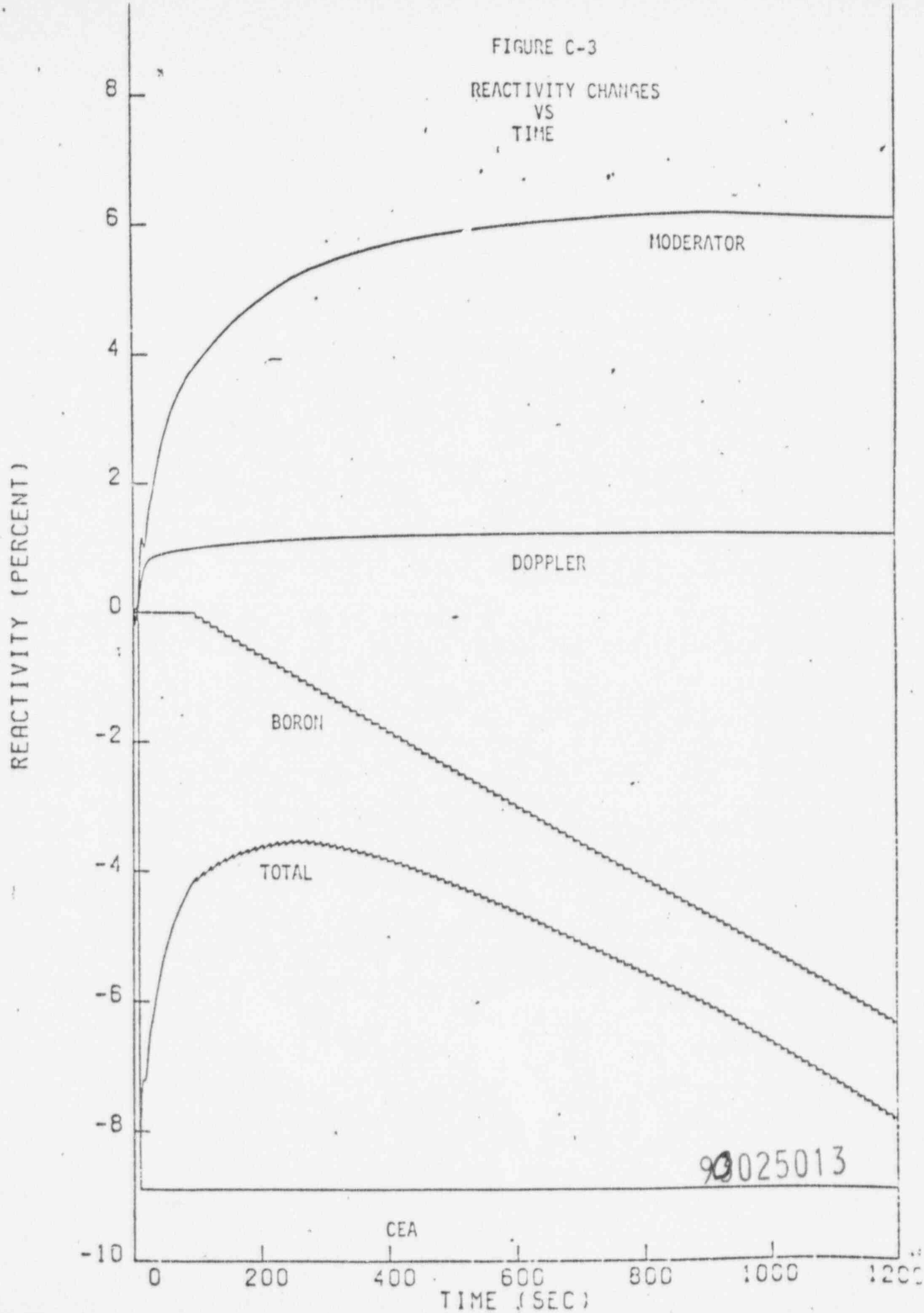

FIGURE C-3

REACTIVITY CHANGES VS TIME

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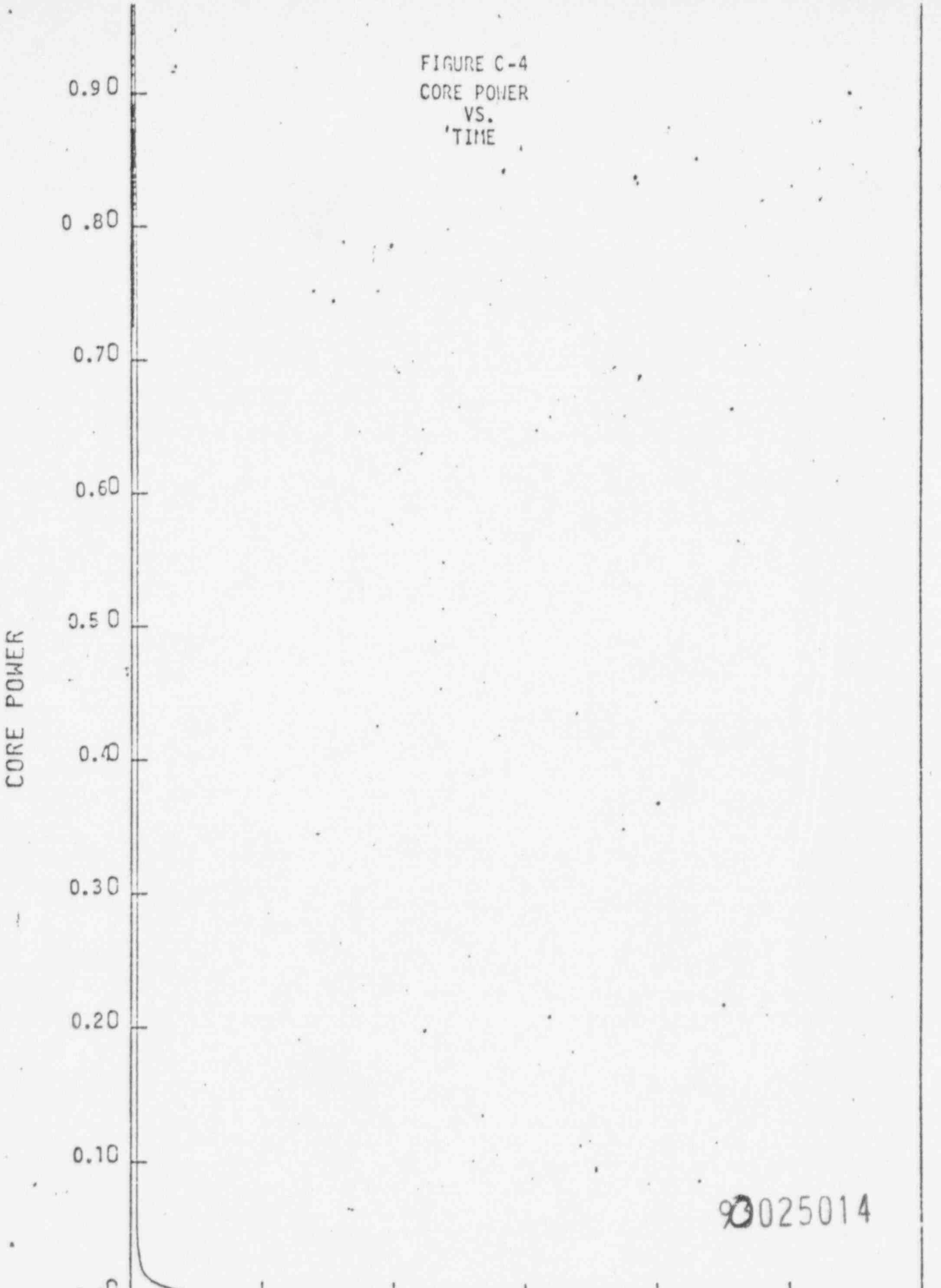
FIGURE C-4
CORE POWER
VS.
TIME
CORE POWER
0.90
0.80
0.70
0.60
0.50
0.40
0.30
0.20
0.10
0.00
0
200
400
600
800
1000
1200
TIME (SEC)

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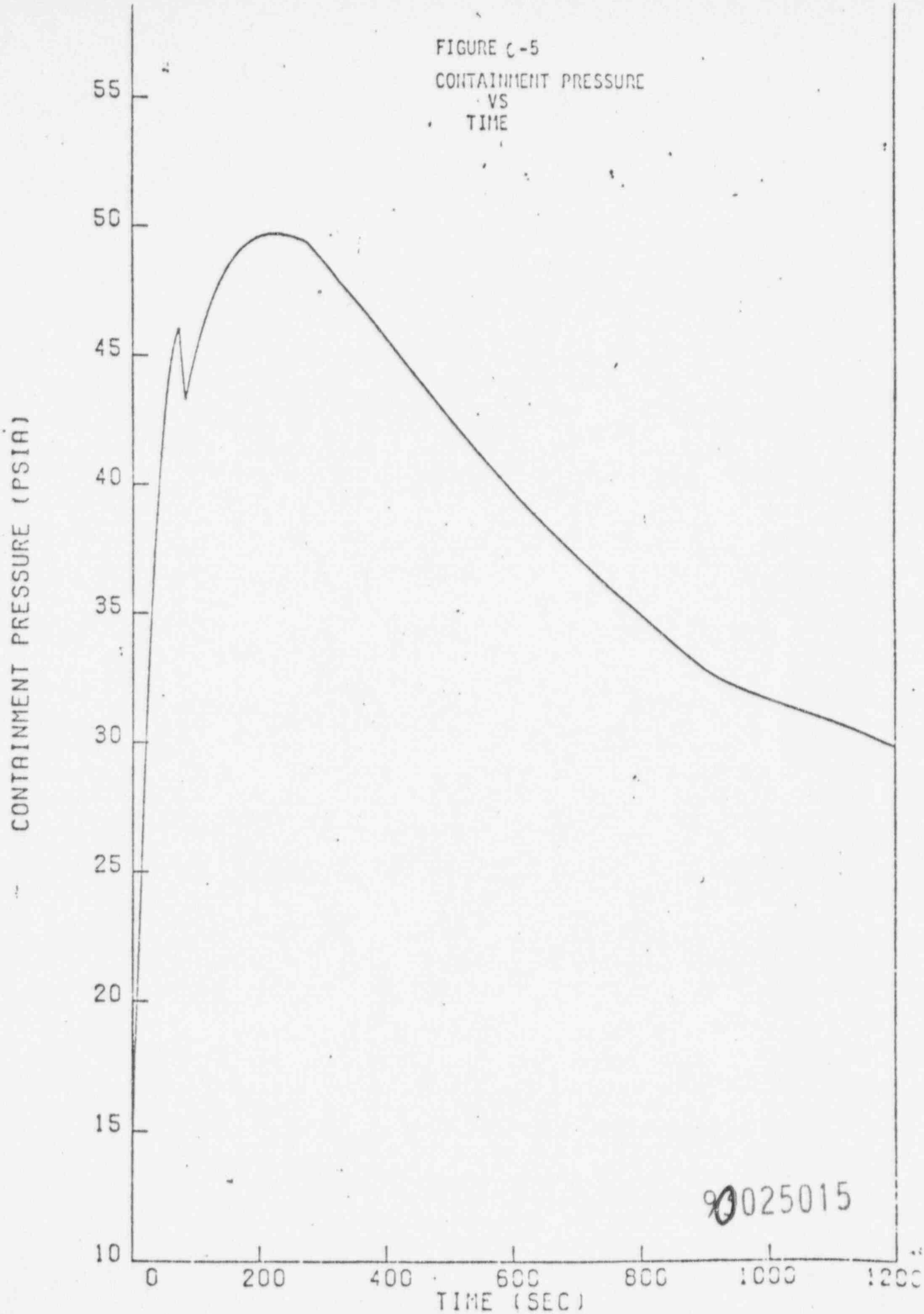
FIGURE C-5
CONTAINMENT PRESSURE
VS
TIME
CONTAINMENT PRESSURE (PSIA)
55
50
45
40
35
30
25
20
15
10
0
200
400
600
800
1000
1200
TIME (SEC)
90025015

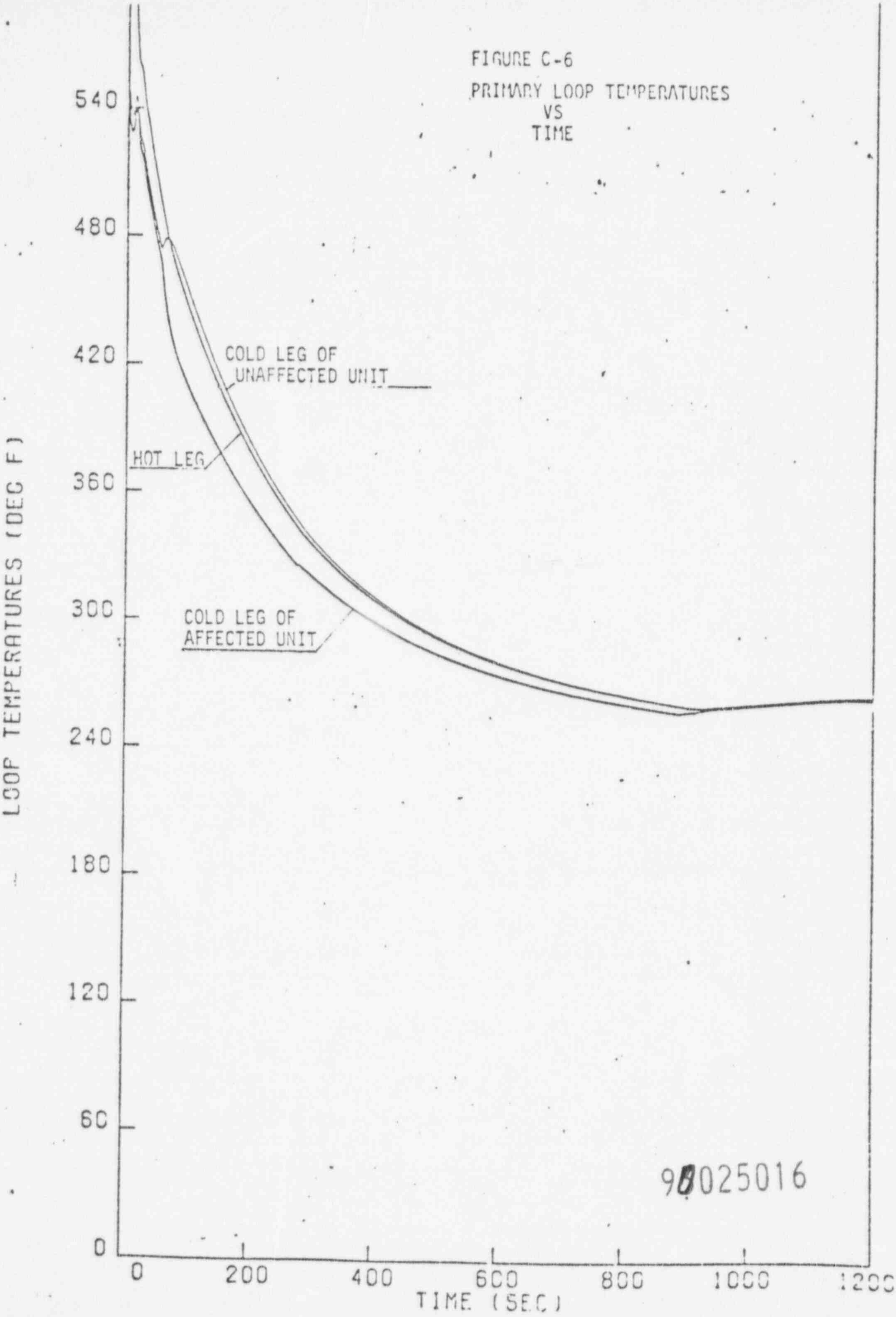
FIGURE C-6
PRIMARY LOOP TEMPERATURES
VS
TIME
COLD LEG OF
UNAFFECTED UNIT
HOT LEG
COLD LEG OF
AFFECTED UNIT
LOOP TEMPERATURES (DEG F)
540
480
420
360
300
240
180
120
60
0
0
200
400
600
800
1000
1200
TIME (SEC)

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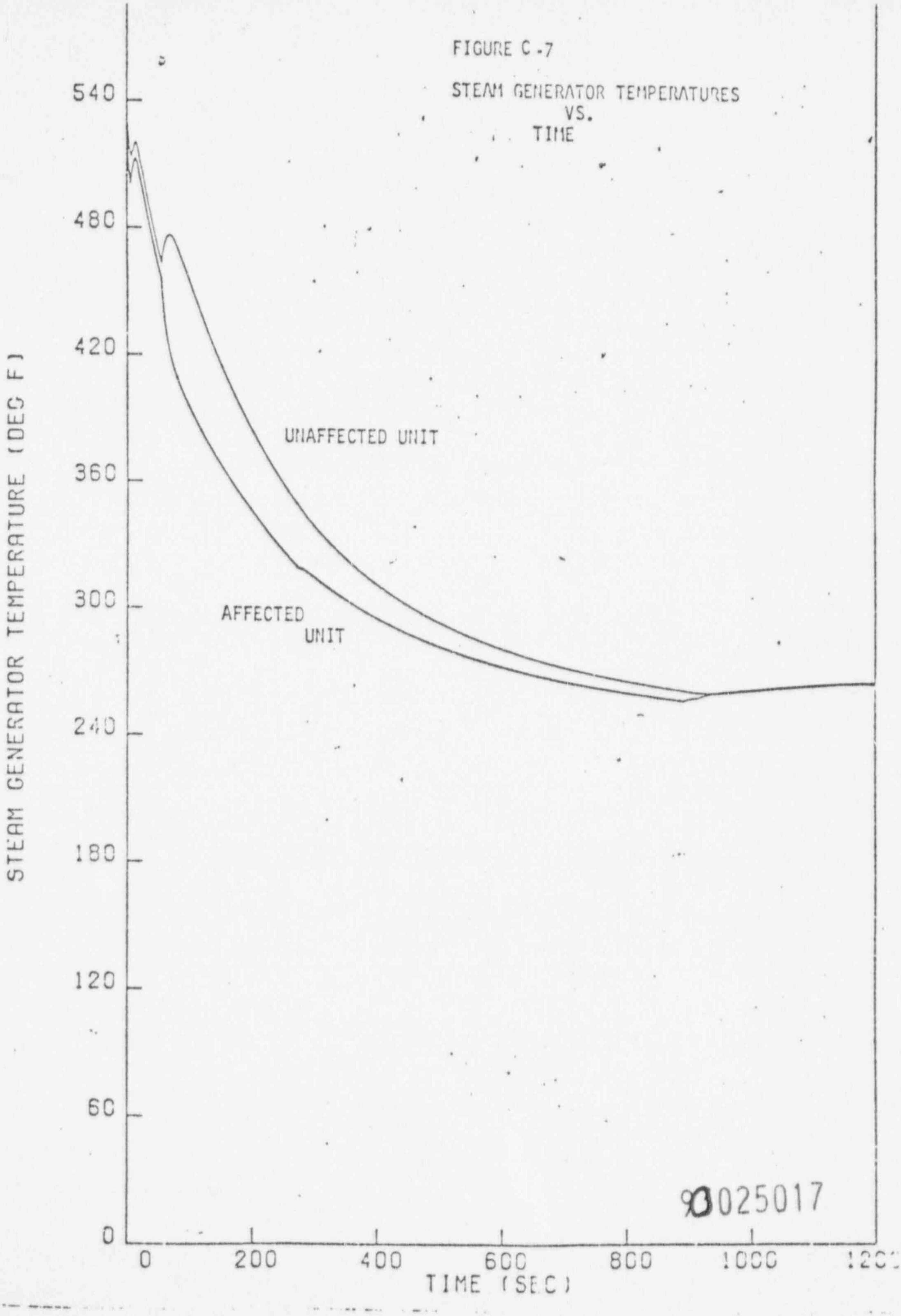
FIGURE C-7
STEAM GENERATOR TEMPERATURES
VS.
TIME
STEAM GENERATOR TEMPERATURE (DEG F)
540
480
420
360
300
240
180
120
60
0
UNAFFECTED UNIT
AFFECTED
UNIT
0
200
400
600
800
1000
TIME (SEC)

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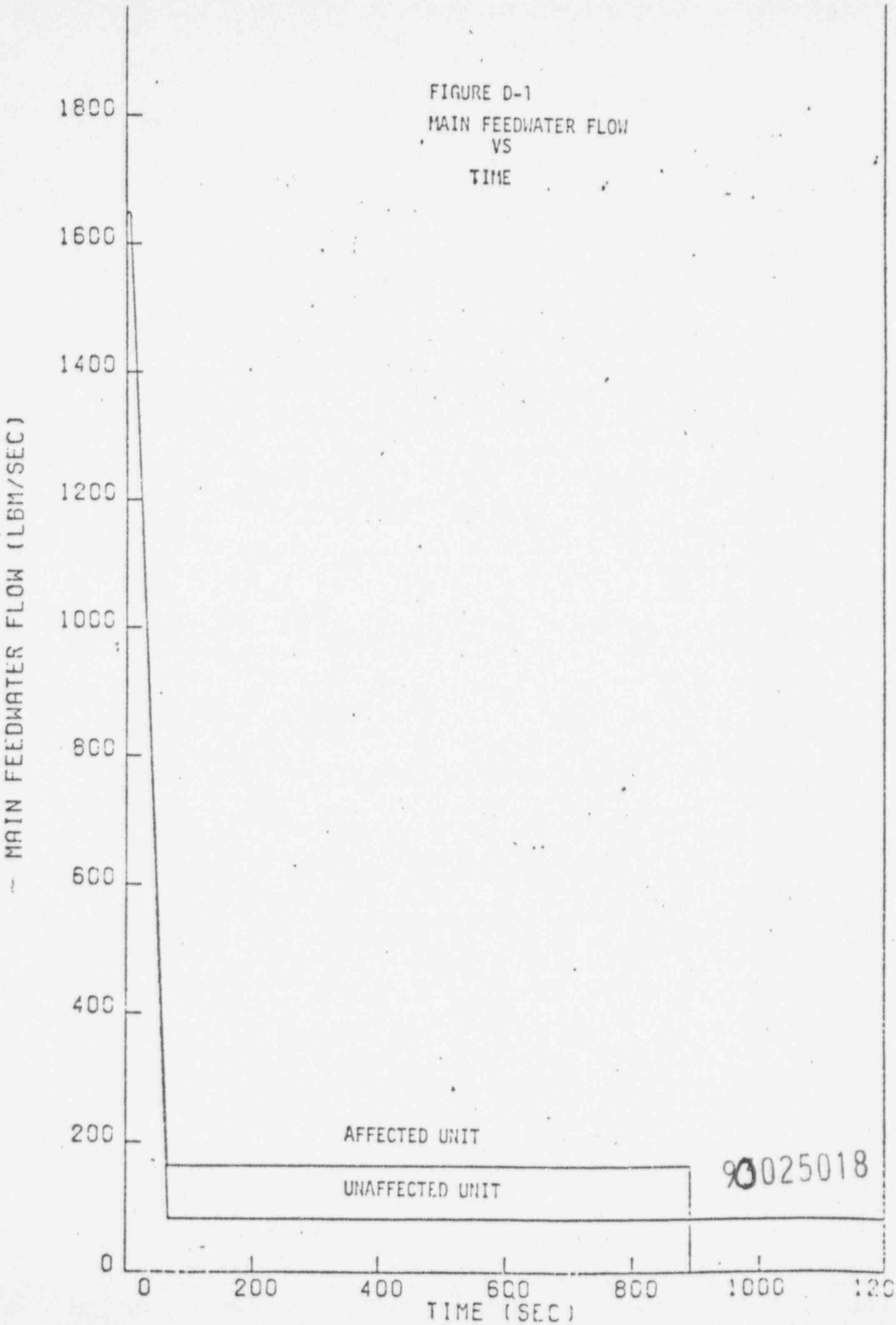

FIGURE D-1
MAIN FEEDWATER FLOW
VS
TIME
MAIN FEEDWATER FLOW (LBM/SEC)
1800
1600
1400
1200
1000
800
600
400
200
0
AFFECTED UNIT
UNAFFECTED UNIT
0
200
400
600
800
1000
TIME (SEC)

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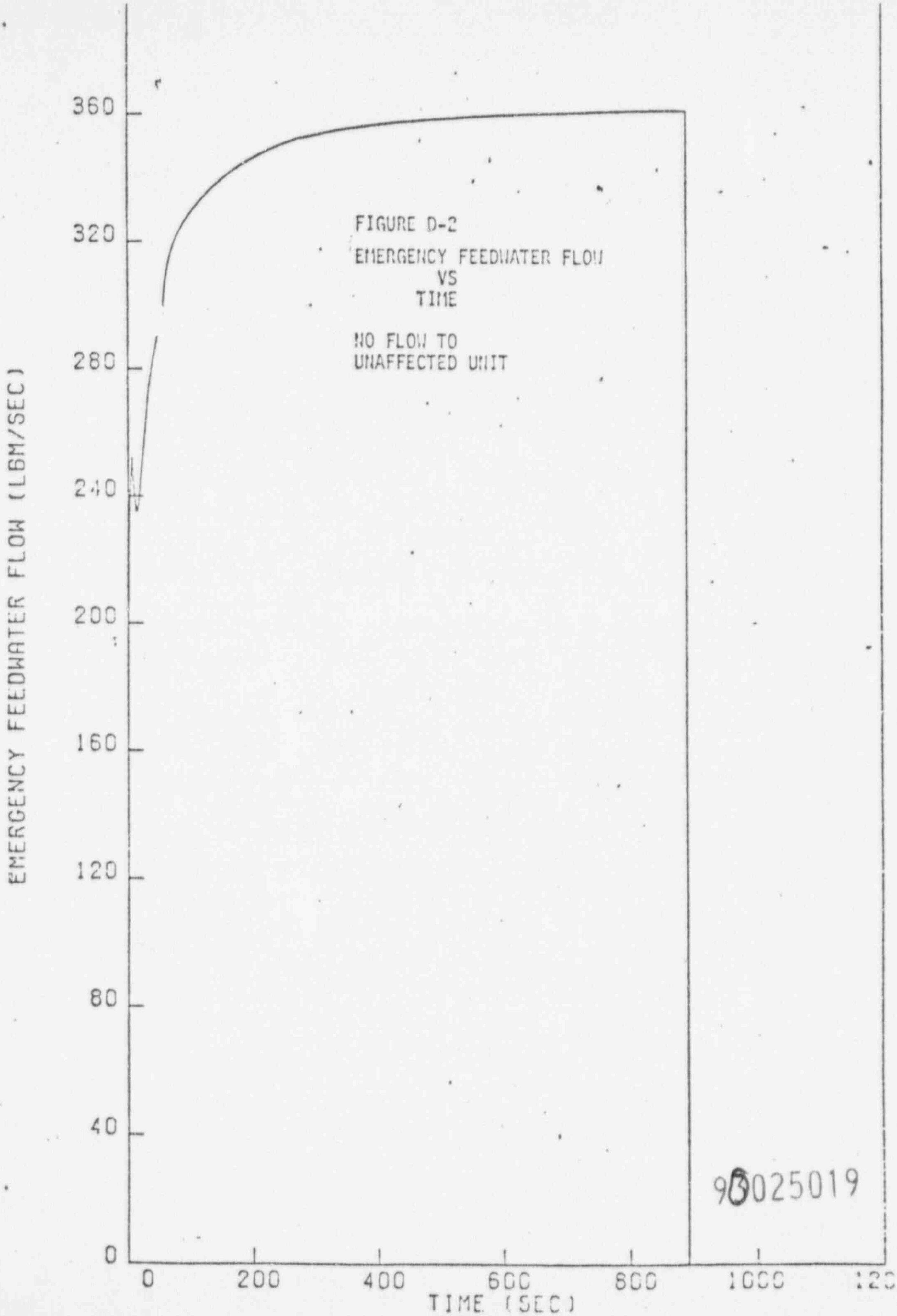

FIGURE D-2
EMERGENCY FEEDWATER FLOW
VS
TIME
NO FLOW TO
UNAFFECTED UNIT
EMERGENCY FEEDWATER FLOW (LBM/SEC)
360
320
280
240
200
160
120
80
40
0
0
200
400
600
800
1000
TIME (SEC)

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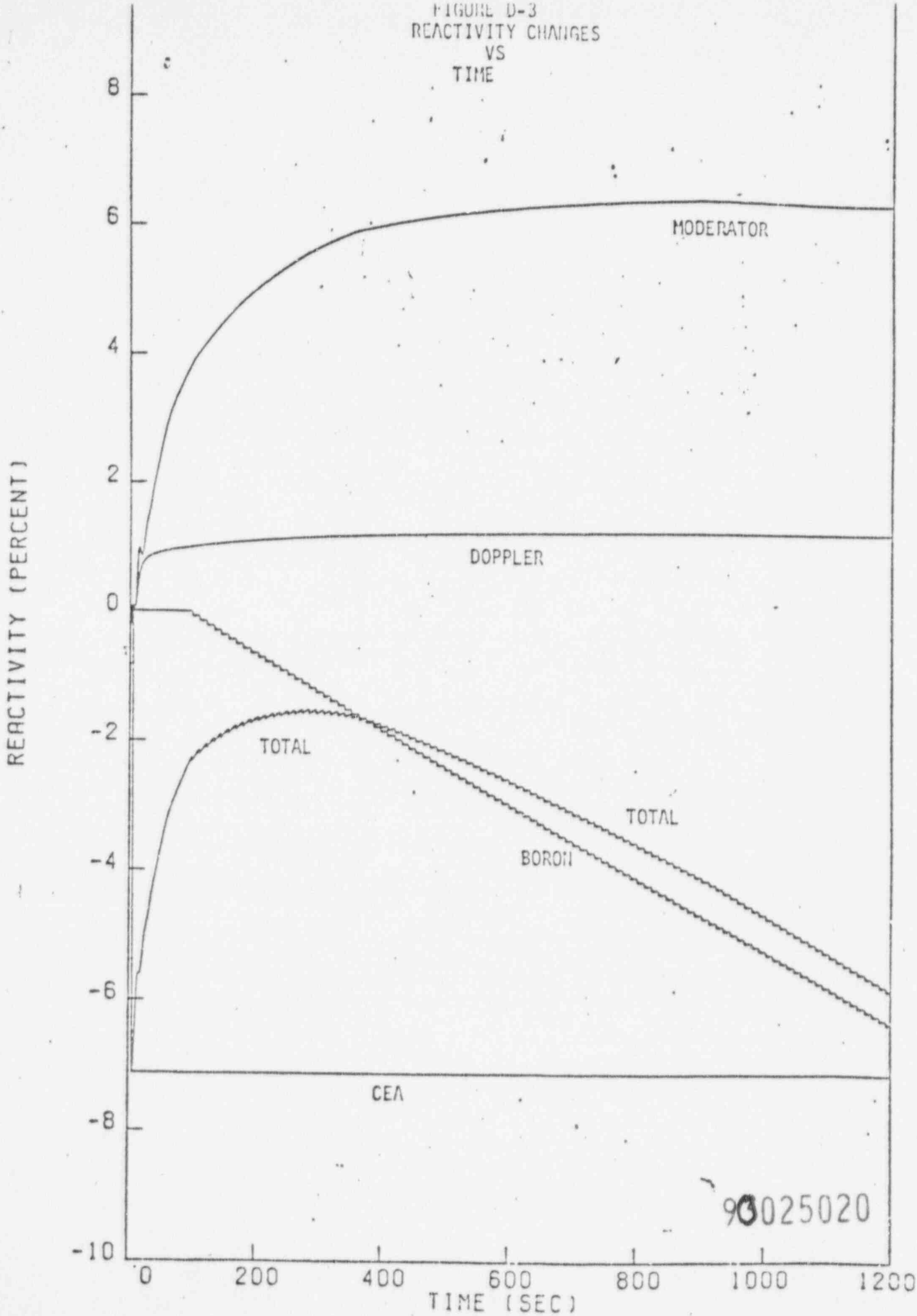

FIGURE D-3
REACTIVITY CHANGES
VS
TIME
MODERATOR
DOPPLER
TOTAL
TOTAL
BORON
CEA
8
6
4
2
0
-2
-4
-6
-8
-10
REACTIVITY (PERCENT)
0
200
400
600
800
1000
1200
TIME (SEC)

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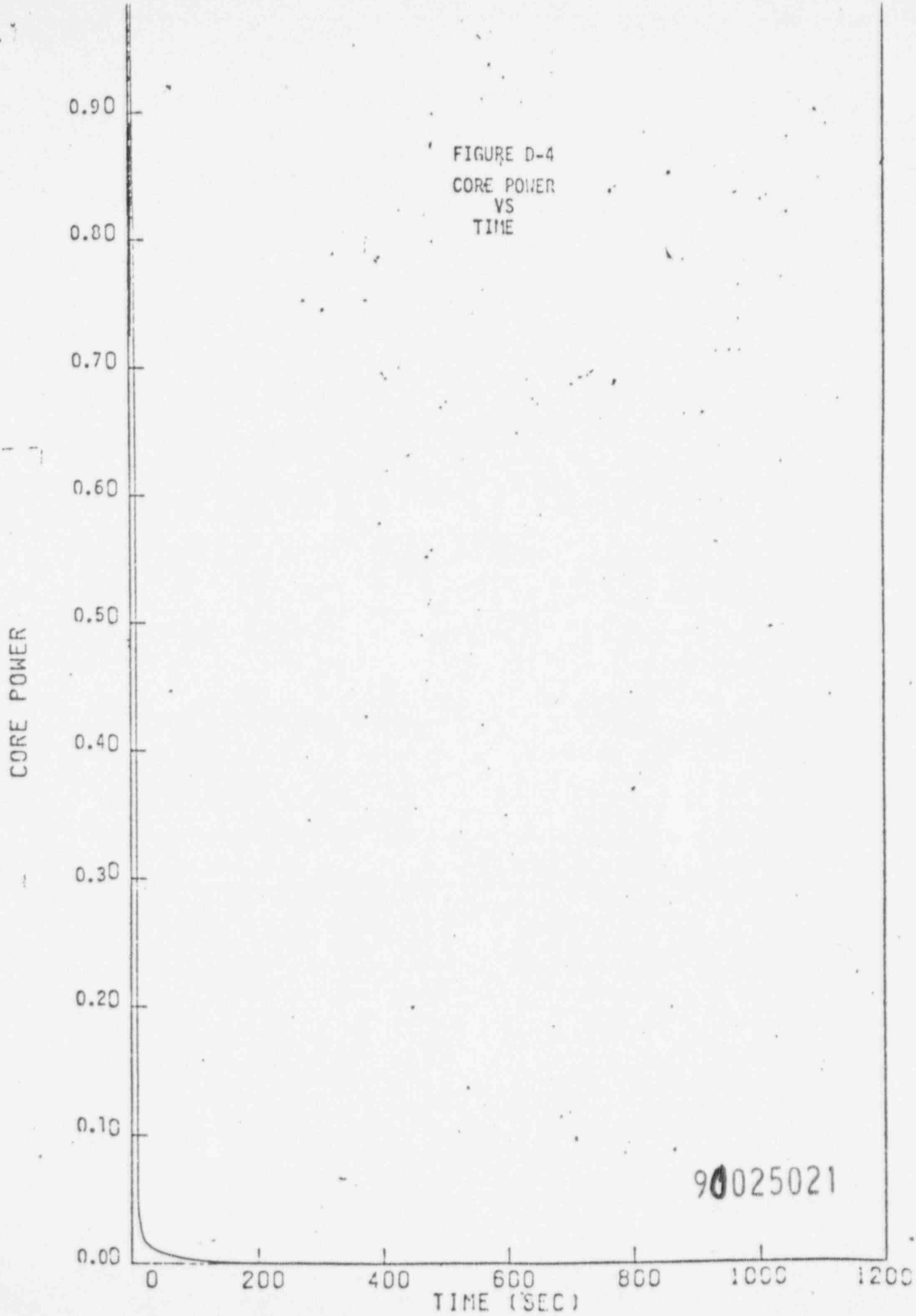

FIGURE D-4
CORE POWER
VS
TIME
CORE POWER
0.90
0.80
0.70
0.60
0.50
0.40
0.30
0.20
0.10
0.00
0
200
400
600
800
1000
1200
TIME (SEC)

9002502

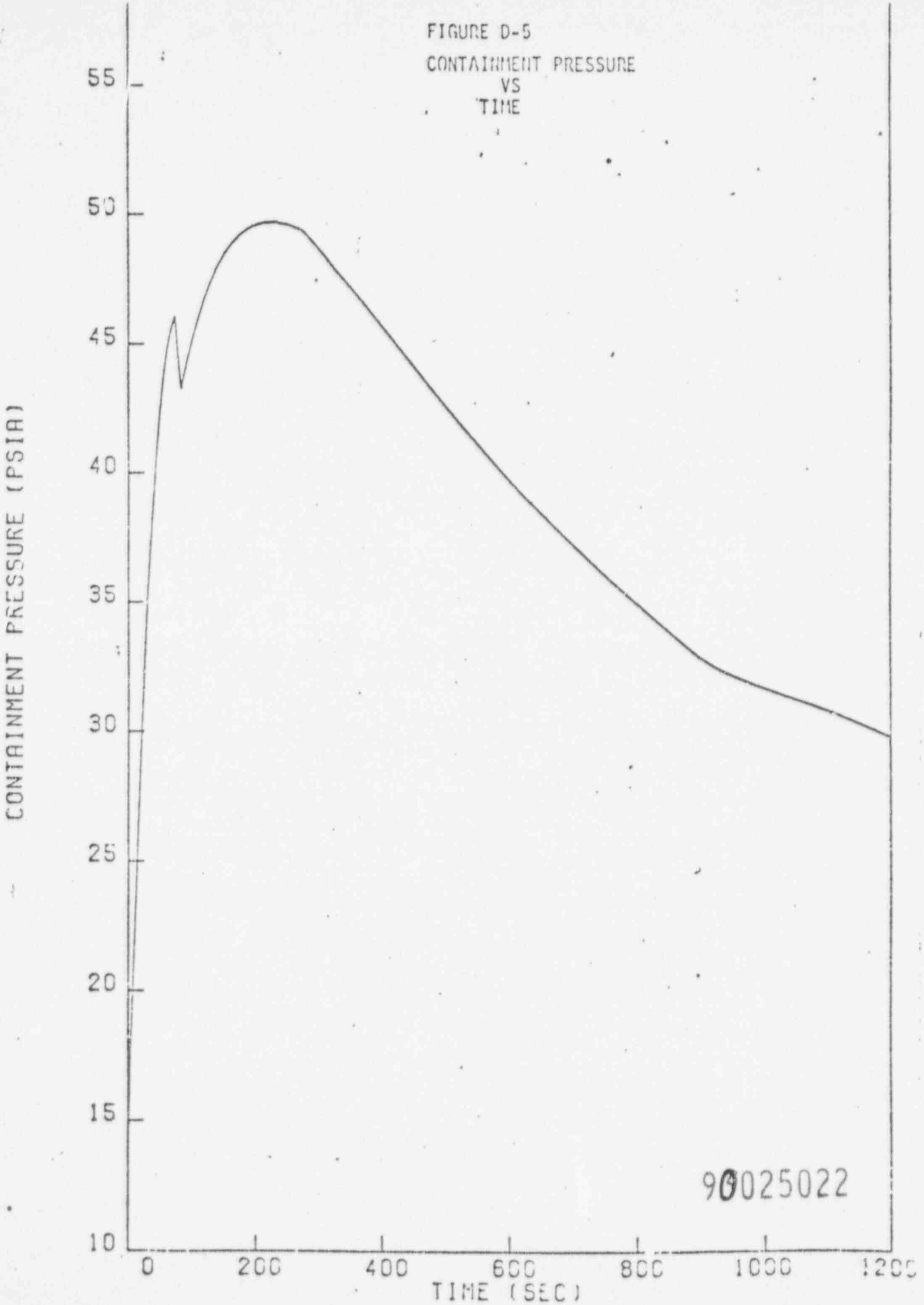

FIGURE D-5
CONTAINMENT PRESSURE
VS
TIME
CONTAINMENT PRESSURE (PSIA)
55
50
45
40
35
30
25
20
15
10
0
200
400
600
800
1000
1200
TIME (SEC)

90025022

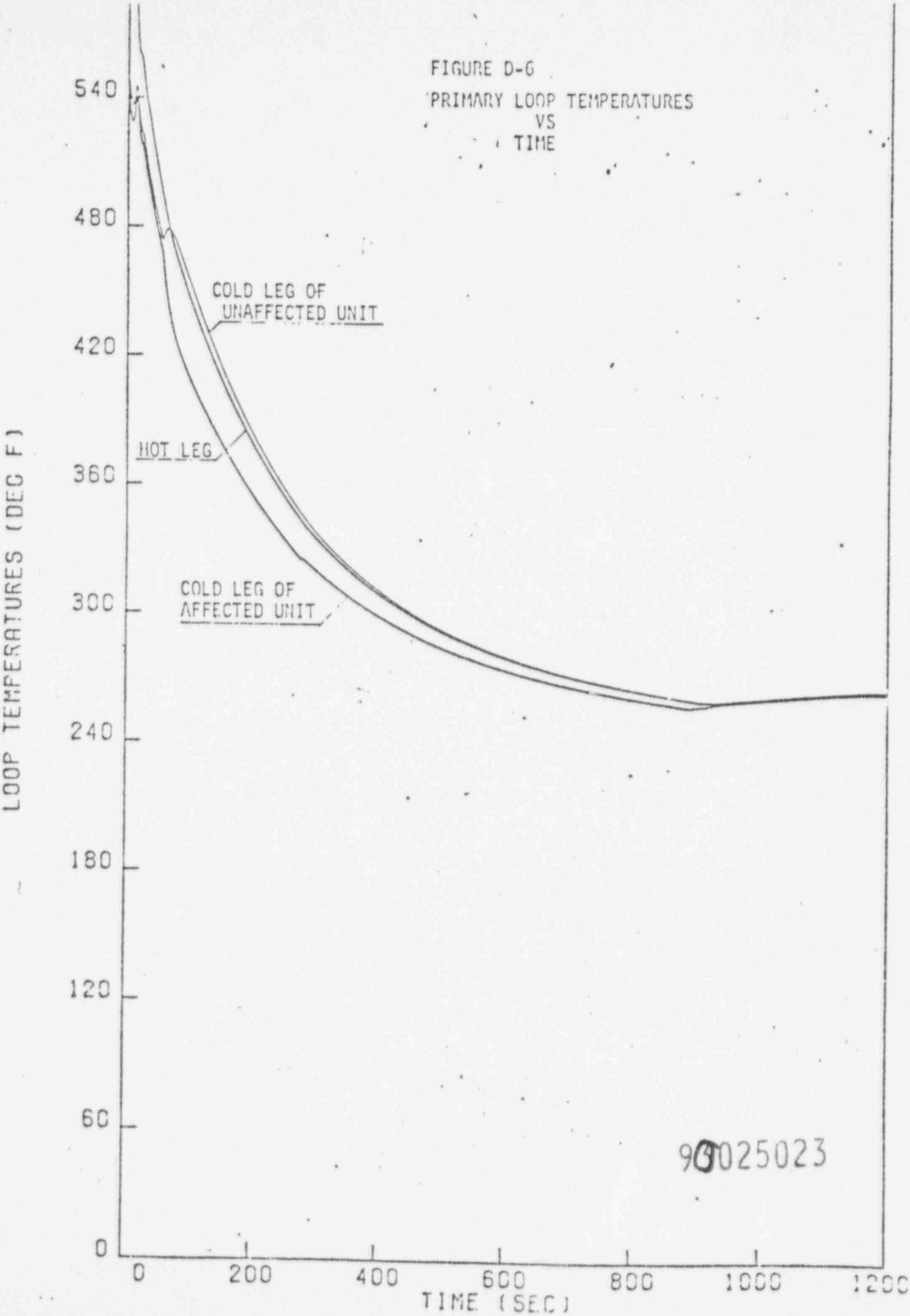

FIGURE D-6
PRIMARY LOOP TEMPERATURES
VS
TIME
COLD LEG OF
UNAFFECTED UNIT
HOT LEG
COLD LEG OF
AFFECTED UNIT
LOOP TEMPERATURES (DEG F)
540
480
420
360
300
240
180
120
60
0
0
200
400
600
800
1000
1200
TIME (SEC)
9Ø025023

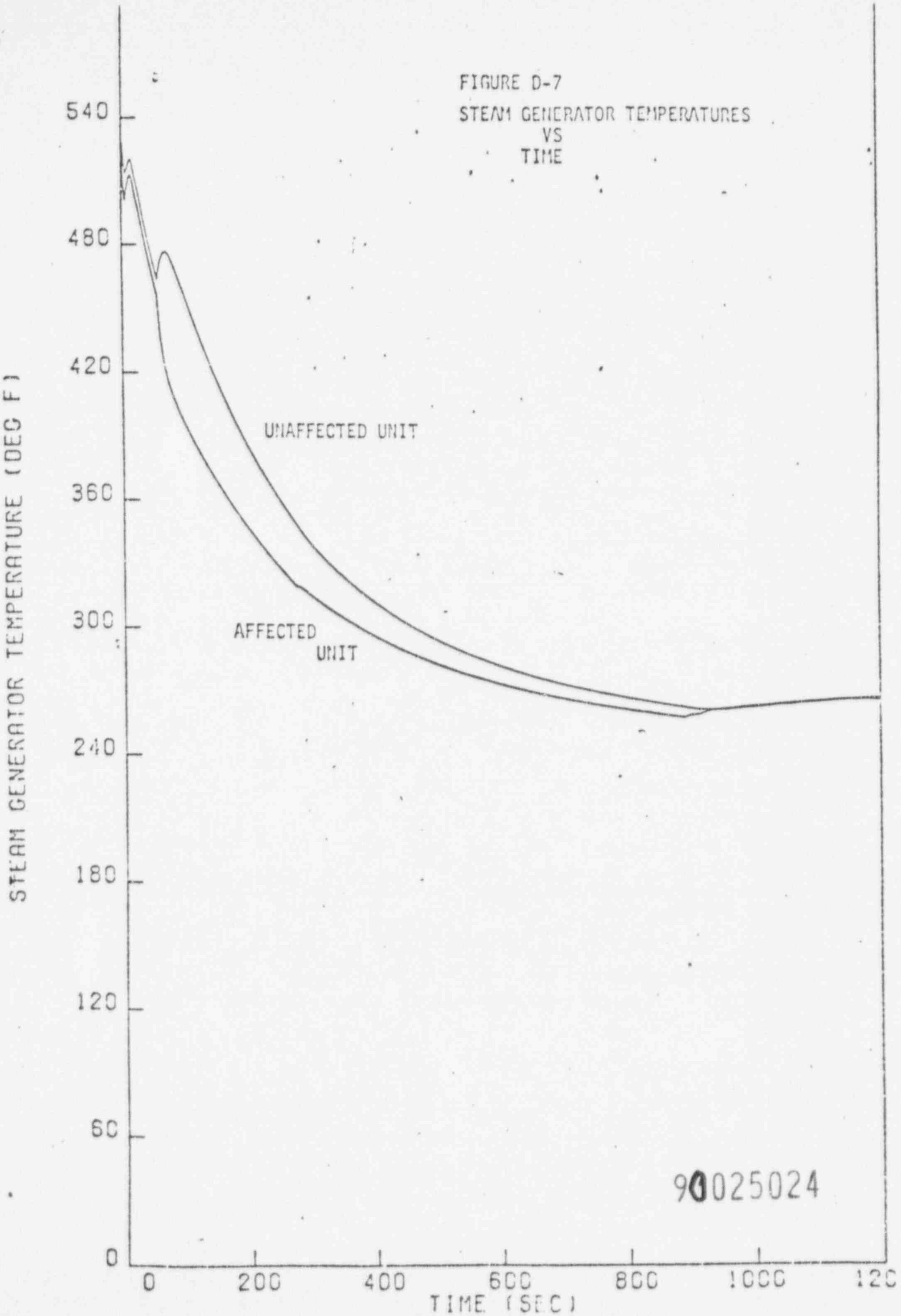

FIGURE D-7
STEAM GENERATOR TEMPERATURES VS TIME

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RETURN TO POWER ANALYSIS

This attachment illustrates the impact of automatic initiation of auxiliary feedwater system (AFWS) on the licensing cases analyzed for a Main Steam Line Break Event. The provided results of analysis for the most limiting Main Steam Line Breaks (MSLB) with respect to return to power assume this system is designed with a delay in the automatic delivery of the auxiliary feedwater to the steam generators. The analyses performed for this evaluation, therefore, allowed for a three (3) minute delay in the actuation of the AFWS subsequent to receiving a low steam generator water level trip signal. The conclusions of this analysis are considered to be applicable to Fort Calhoun.

The most limiting cases analyzed were those selected from the Design Basis Events analyzed in the FSAR or subsequent reload licensing submittals, as appropriate. The most limiting case was found to be a full power-full flow Main Steam Line Break inside containment with automatic initiation of AFWS after a 3 minute delay. The analysis was continued until the subcriticality margin was continuously increasing. The delay of 3 minutes assumed as part of the design of the automatically initiated auxiliary feed system was modeled conservatively in the analysis. The MSLB outside containment is less limiting because the blowdown rate of the steam generators is restricted by the flow venturies located in the steam lines thus leading to a less severe reactivity insertion and a smaller potential for return-to-power than the results presented herein.

The results of the limiting case show that the affected steam generator blows dry in about 70 seconds and begins Reactor Coolant System (RCS) cooldown with feedwater only. The peak power level attained including decay heat and subcritical multiplication is 12%. From the time the steam generator runs dry until the actuation of auxiliary feedwater system, boron injected by High Pressure Safety Injection (HPSI), actuated at about 16 seconds into the transient, continues to add more negative reactivity to the core. After the initiation of AFW flow, the cooldown of the reactor coolant system (RCS) is resumed. The auxiliary flow is conservatively assumed to feed the affected steam generator only. The assumed auxiliary flow to the affected steam generator was conservatively taken to be about 20% of the full power feedwater flow. The continued cooldown of the RCS adds more positive reactivity which is eventually terminated by the Low Pressure Safety Injection (LPSI) flow injected due to low RCS pressure. The negative reactivity inserted via LPSI flow terminates the reactivity excursion. The return-to-power attained after the AFWS delivery is 10.7%. Thus, with the 3 minute delay in the actuation of AFWS, the auxiliary feedwater will be introduced away from the most critical time frame with respect to return-to-power and the conclusions of the MSLB events presented in the FSAR and the subsequent reload submittals conservatively bound those with the control grade automatic initiation of auxiliary feedwater systems included. A typical sequence of events, typical for operating C-E plants, for the limiting case are presented in Table 1.

The MSLB results presented in the FSAR and subsequent reload licensing submittals assumed the following consequential failures in addition to the single failure which initiates the event (i.e., the double ended pipe break inside containment):

(a) On reactor scram, the highest worth Control Element Assembly is assumed to stick in the fully withdrawn position,

(b) On Safety Injection Actuation, on the HPSI and one of the LPSI safety injection pumps are assumed to fail to start.

(c) No main feedwater isolation is assumed on MSIS. The main feed flow is assumed to coastdown to 5% of full power flow in 60 seconds. (More realistically flow would ramp to zero in about 20 seconds.)

Single failures were considered in the design basis to the extent that a failure initiates the event and safety grade equipment is designed to accommodate single failures as described above and is consistent with the design basis presented in the FSAR. No consequential failures other than previously identified were considered. All control systems considered were assumed to function in the manner consistent with the FSAR.

Single failures concurrent with the MSLB (other than those identified above), as well as loss of offsite power concurrent with MSLB, are not, and have not been part of the design basis as described in the FSAR and, therefore, were not considered.

TABLE A-1

Sequence of Events for the Main Steam Line Break Event with Automatic Initiation of Auxiliary Feedwater System (Full load, Two-Loop Condition, Nozzle Break)

Time (sec.)	Event	Safety System Initiated	Setpoint or Value
0.0	Initiation of break	----	----
3.4	Low steam Generator Pressure trip signal occurs, MSIS initiated and Main Steam Isolation Valves begin to close.	Reactor Protection System Main Steam Isolation System	478 psia
4.3	Trip breakers open	----	----
4.8	CEAs begin to drop into core	Reactor Protection System	----
10.7	Complete closure of Main Steam Isolation Valves to terminating blowdown from the intact steam generator	----	----
15.9	Pressurizer empties	----	----
16.2	Low RCS pressure, SIAS Initiated	Safety Injection System	1563 psia
22.8	High Pressure Safety Injection flow Initiated	Safety Injection System	1220 psia
64.8	Main feedwater flow completes ramp down to 5%	----	----
68.7	Affected steam generator liquid inventory depleted and beginning of blowdown of feedwater only	----	----
71.9	Peak return-to-power* occurs with a peak reactivity of -.186%Δρ	----	12%

* return-to-power includes decay heat and subcritical multiplication

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TABLE A-1 (Continued)

Time (sec.)	Event	Safety System Initiated	Setpoint or Value
150.0	Auxiliary Feedwater flow to affected steam generator initiated	Auxiliary Feedwater System	----
318.7	Low Pressure Safety Injection flow Initiated	Safety Injection System	207 psia
319.9	Peak reactivity post auxiliary feedwater delivery	----	+.13%$\Delta\rho$
345.1	Peak return to power post auxiliary feedwater delivery	----	10.7%

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