Southern California Edison Company

P. O. BOX 800

2244 WALNUT GROVE AVENUE

ROSEMEAD, CALIFORNIA 91770

M. O. MEDFORD
MANAGER OF NUCLEAR ENGINEERING
AND LICENSING

TELEPHONE
(818) 302-1749

November 20, 1987

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Subject: Docket No. 50-206
Engineered Safety Features Single Failure Analysis
San Onofre Nuclear Generating Station
Unit 1

By letter dated November 6, 1987, the NRC was provided with the Engineered Safety Features Single Failure Analysis Report for San Onofre Unit 1. SCE committed to provide the design descriptions for the modifications to the auxiliary feedwater system, the steam/feedwater flow mismatch trip and additional modifications to resolve the identified single failure concerns. The purpose of this letter is to submit the above information.

Accordingly, Enclosure 1 to this letter provides design descriptions for the integration of the third auxiliary feedwater pump, the steam/feedwater flow mismatch trip, and main feedwater isolation. Additional modifications to resolve other single failure concerns are currently being evaluated. Design descriptions for the final modifications will be provided by March 31, 1988.

Enclosure 2 provides the revised auxiliary feedwater system transient analysis. This analysis reflects the Cycle X auxiliary feedwater system configuration with the third automated auxiliary feedwater pump. The analysis also credits the availability of the single failure-proof steam/feedwater flow mismatch trip above 50% power (see design description). The revised analysis eliminates credit for the safety injection/high containment pressure trip for feedline breaks inside containment as discussed in SCE's July 2, 1987 submittal. The evaluation of the radiological consequences of a feedline break outside containment is currently being reviewed. The results of this review will be provided as soon as it is completed.

If you have any questions or require additional information regarding this subject please contact me.

Very truly yours,

M O Medford

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cc: J. O. Bradfute, NRR Project Manager, San Onofre Unit 1
J. B. Martin, Regional Administrator, NRC Region V
F. R. Huey, NRC Senior Resident Inspector, San Onofre Units 1, 2 and 3

Enclosure 1

CYCLE X MODIFICATION

DESIGN DESCRIPTIONS

for

AUXILIARY FEEDWATER SYSTEM UPGRADE

STEAM/FEEDWATER FLOW MISMATCH TRIP

MAIN FEEDWATER SYSTEM ISOLATION

SAN ONOFRE UNIT 1

Cycle X Modification
Design Descriptions
for
Auxiliary Feedwater System Upgrade
Steam/Feedwater Flow Mismatch Trip
Main Feedwater System Isolation
San Onofre Unit 1

Introduction

SCE previously committed to upgrade the auxiliary feedwater system (AFW) by automating and integrating the third AFW pump (installed to meet fire protection requirements during the Cycle IX refueling outage). Subsequently, SCE identified single failure susceptibilities of the Reactor Protection System (Steam/Feedwater Flow mismatch trip) and the Engineered Safety Features (main feedwater isolation and recirculation). Accordingly, design description for the integration of the third AFW pump, the steam/feedwater flow mismatch trip and main feedwater isolation are provided below. Modifications to correct the single failure susceptibilities of the recirculation system will be provided at a later date.

I. Auxiliary Feedwater System (AFW) Upgrades

A) Add two new AFW flow control valves (FCVs) to the existing configuration so that two FCVs in parallel are provided on each AFW line. The parallel valves on each line will be on separate electrical trains. Train F FCVs will fail closed on loss of control power and the Train G FCVs will fail open on loss of control power.

B) The control system for the new FCVs will be identical to that for existing FCVs.

C) Install a venturi/orifice downstream of the AFW flow control valves in each of the three AFW lines. The venturis/orifices are to be sized so as to prevent Pump G10S run-out and exceeding water hammer flow restrictions to depressurized steam generators. A locked closed bypass valve will be provided for each venturi/orifice.

D) A venturi/orifice is to be placed in the discharge of the G10W pump so as to prevent exceeding the maximum flow limit to each steam generator under any sequence of events independent of the number of steam generators available. A manual locked closed bypass shall be provided for Dedicated Safe Shutdown operation of G10W.

E) Realign G10 to the same electrical train as G10S (i.e., Train F).

F) Add G10W to other electrical train (i.e., Train G).

G) Revise the existing control room panel to include the same controls, indication and alarms for G10W as exist for the other motor driven AFW pump. The control room panel will be revised in accordance with human factors guidelines as established by the Control Room Design Review. This will include:

1. Suction and discharge pressure indication and low suction pressure alarm.

2. Control switches and position indication of G10W discharge valve, FV-3110. When AFWS is in auto mode, FV-3110 shall open on a Train G AFWS auto initiation signal.

3. Pump manual Start/Stop switches with running lights and ammeter.

H) Remove the G10S low suction/discharge pressure trips, but retain the indication and alarm functions of the instrumentation. Pump runout protection will be provided by the venturis/orifices in each of the three AFW discharge lines and verified by pump testing.

I) Provide a manual transfer switch for selecting Dedicated Safe Shutdown (DSD) or normal Safety Related power for G10W locally. This switch will provide isolation of the normal and DSD power supplies.

J) Modify the auto-mode control circuit of each pump and respective discharge valve to operate as follows upon receipt of the steam generator low level signal (AFWS auto initiation).

1. Lead Train G Pump (G10W) immediately provides flow and tubing drive Train F pump (G10) placed in "warm-up mode."

2. Lag Train F pumps (G10S and G10) start upon a no flow signal from the G10W pump discharge manifold and then the Train F pump discharge valves open.

3. Train F pumps (G10S and G10) will stop and discharge valves close when there is a flow signal from the G10W discharge manifold.

4. Separate Train F flow switches will be provided on the G10W discharge manifold for control of the Train F pumps and control of the Train F pump discharge valves to prevent postulated single failures from causing inadvertent operation of both trains.

K) All work will be classified as Safety-Related, Seismic Category A. (See Figure 1.)

II. Steam/Feedwater Flow Mismatch Reactor Trip

A) The current mismatch trip logic will be revised to provide a trip signal to the reactor trip circuit, two out of three reactor trip logic, for a high steam/feedwater flow mismatch as well as the original low flow mismatch. The setpoint for the high mismatch will be defined as part of final design.

B) The high pressurizer level trip will be retained at the 50% setpoint.

C) A P-8 permissive will be added to the revised steam/feedwater flow mismatch trip. This permissive will arm the trip at or above 50% power. This feature is to improve plant availability by reducing the probability of unnecessary plant trips during reduced power operation and startup.

D) A minimum floor value will be provided for the main steam header pressure signal in each of the channelized steam flow calculator modules to prevent loss or spurious initiation of the steam/feedwater flow mismatch trip due to a downscale failure of PT-459.

E) The power supplies and signal paths for each steam/feedwater flow mismatch instrument loop will be channelized. Channelization of the power supplies and signal paths will prevent loss of more than one steam/feedwater flow mismatch channel due to a postulated single failure.

F) Isolation will be provided between the PT-459 instrument loop and each steam/feedwater flow mismatch channel and its associated feedwater control loop to prevent loss of the steam/feedwater flow mismatch channels due to postulated single failures of the PT-459 loop or the non-qualified control loops.

G) All work will be classified as Safety-Related, Seismic Category A. (See Figure 2.)

III. Main Feedwater System

A) Replace the solenoid valves on the main feedwater control valves (FCV-456, -457 and -458) and their bypass valves (CV-141, -142 and -143) with replacements environmentally qualified (EQ) to 10 CFR 50.49 requirements.

B) Replace the actuators on MOVs 20, 21 and 22 with replacements environmentally qualified (EQ) to 10 CFR 50.49 requirements.

C) Change the power source on the solenoid valve circuits for feedwater valves FCV-457 and FCV-458 and respective bypass valves CV-144 and CV-143 to Train 2.

D) Provide N_2 backup to the main feedwater control valves.

E) Modify or replace the actuators for the main feedwater FCVs, MOVs and bypass CVs to ensure valve closing time is sufficient to meet transient analysis requirements.

F) Provide a new solenoid valve (redundant) to each of the bypass CVs, powered and sequenced from the opposite train. This will provide redundant actuation for each bypass CV.

G) Change motive and control power for MOV-22 from MCC-3 to MCC-1, 1A or 1B (which are outside harsh areas during main steam line breaks).

H) All work will be classified as Safety-Related, Seismic Category A.

Figure 1
AUXILIARY FEEDWATER SYSTEM UPGRADE

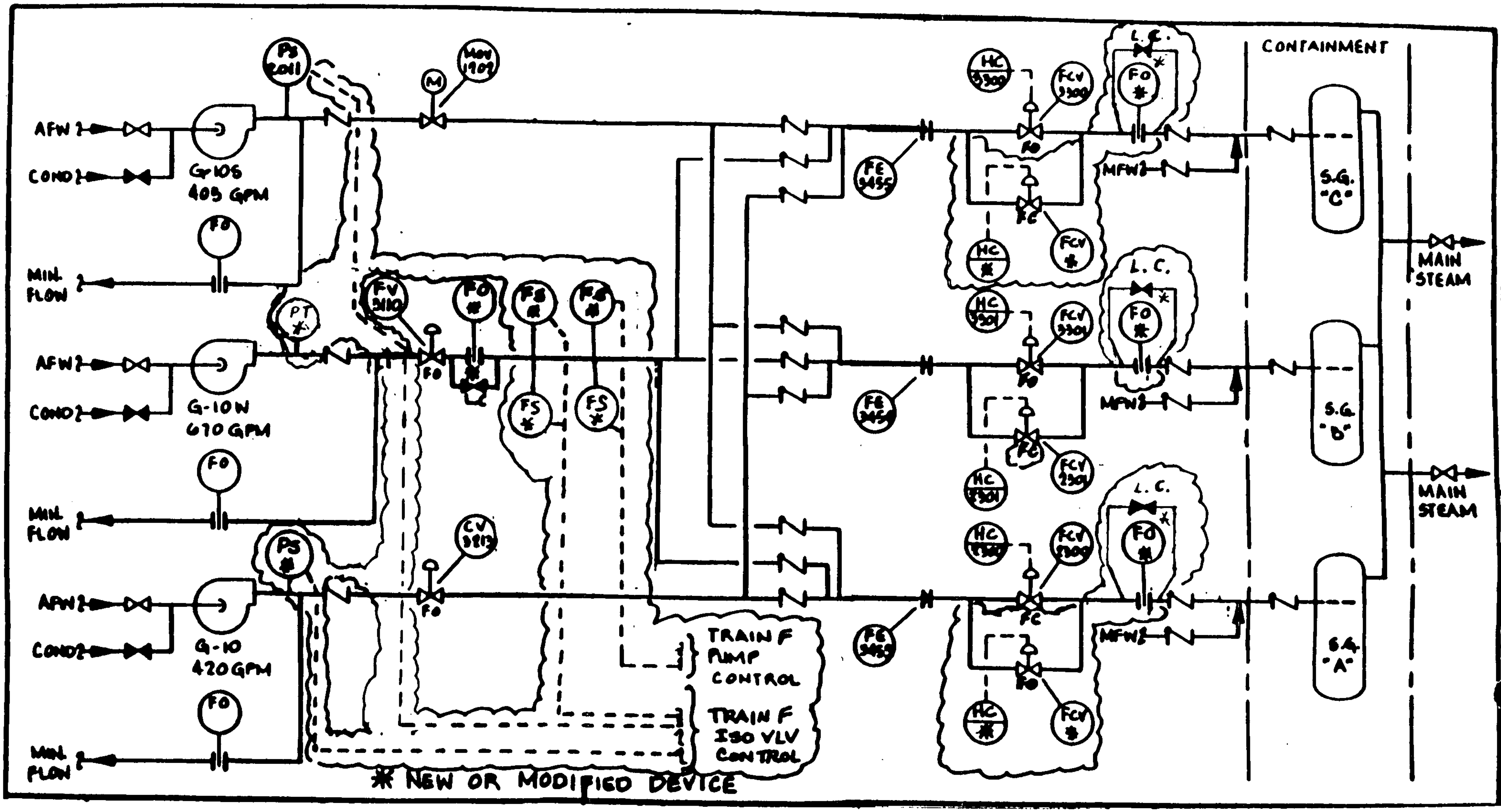
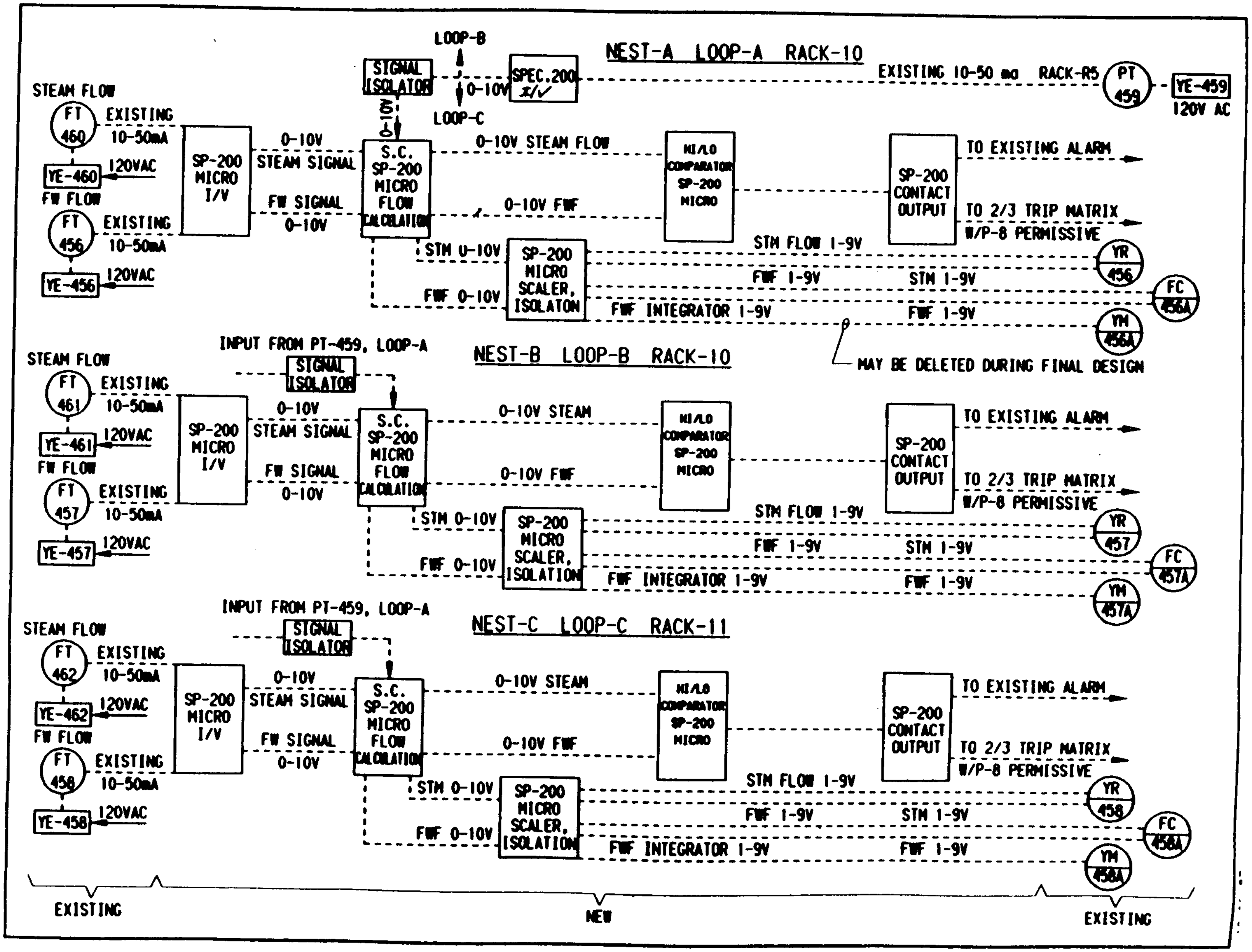


Figure 2

STEAM/FEEDWATER FLOW MISMATCH TRIP



ENCLOSURE 2

Cycle X

Auxiliary Feedwater System

Transient Analysis

San Onofre Unit 1

Auxiliary Feedwater System
Transient Analysis
for Cycle X
San Onofre Unit 1

BACKGROUND

During the Cycle X refueling outage, SCE committed to automating and integrating the third AFW pump (installed to meet fire protection requirements during the Cycle IX refueling outage) into the AFW system. The AFW system must be capable of meeting minimum flow requirements established by analysis of Loss of Normal Feedwater (LONF) and Feedline Break (FLB) events, and of not exceeding maximum flow limits set by water hammer. LONF and FLB analysis was necessary to supplement the previous analysis and AFW design alternatives along with Reactor Protection System alternatives for these events.

ANALYSIS

The LOFTRAN code was used to simulate the transients. The assumptions applicable to all cases are presented below. The assumptions specific to each of the five cases are presented separately. All assumptions, including initial conditions, were selected so to maximize the consequences of the applicable transient.

General Assumptions

1. The initial pressurizer pressure is 30 psi above its nominal value of 2100 psia.

2. Initial steam generator water level is at the nominal value.

3. A High Pressurizer Water Level reactor trip setpoint of 50% narrow range span (NRS) plus 4% NRS for uncertainties is assumed with a delay time of 2 seconds.

4. A High Pressurizer Pressure reactor trip setpoint of 2260 psia (including uncertainties) is assumed with a delay time of 2 seconds.

5. A loss of reactor coolant pumps with SONGS 1 specific RCP coastdown characteristics is modeled. An operating pump heat addition to the RCS of 3 MWth/pump is assumed.

6. 1979 ANS 5.1 Decay Heat is modeled.

7. An AFW temperature of 100°F is assumed.

8. A feedwater system purge volume of 73 ft^3/loop is assumed. This piping volume must be purged of the relatively hot main feedwater before the colder AFW enters the steam generators.

CASE A: Partial Loss of Normal Feedwater at 100% power

Specific Assumptions

The plant is initially operating at 103% of rated power.

Initial reactor coolant average temperature is 4°F above the nominal full-power value (575.15°F).

Initial pressurizer water level is 37.5% NRS.

Main feedwater to effected steam generator is assumed to stop at the time of the failure of the flow control valve. Main feedwater to the two unaffected steam generators continues until time of reactor trip.

Pressurizer power-operated relief valves and pressurizer sprays are available.

Auxiliary feedwater (AFW) is assumed to be manually actuated and the system manually aligned to deliver flow of 185 gpm to three steam generators 30 minutes after the initiation of the event (FW flow control valve failure).

The steam flow/feed flow mismatch reactor trip is assumed not to function since required logic (2 out of 3 loops) will not be met.

Results and Conclusions

The results of the Partial LONF transient at full power are shown in Figures 1 through 9. The time sequence of events is presented in Table 1. The results show that reactor trip occurs on high pressurizer water level and that the high pressurizer water level setpoint (50% NRS) prevents the pressurizer from filling. Thus, the acceptance criterion for a LONF event is met.

CASE B: Complete Loss of Normal Feedwater at 100% Power

CASE B analysis was previously submitted by letter from M. O. Medford (SCE) to G. E. Lear (NRC) dated May 1, 1986. This analysis showed that with AFW flow of 165 gpm to 3 steam generators initiated at 30 minutes, the acceptance criterion of no pressurizer fill was achieved. Reactor trip occurs on steam flow/feedwater flow mismatch.

CASE C: Complete Loss of Normal Feedwater at 50% power.

Specific Assumptions

The plant is initially operating at 53% of rated power.

Initial reactor coolant average temperature is 4°F above the nominal value (551.5°F) corresponding to 50% power level on the nominal average temperature program (575.15°F at full power).

Initial pressurizer water level is 30.0% NRS.

Main feedwater to all steam generators is assumed to stop at the time of the complete loss of normal feedwater.

Pressurizer power-operated relief valves and pressurizer sprays are available.

AFW is assumed to be manually actuated and the system manually aligned to deliver flow of 185 gpm to three steam generators 30 minutes after the initiation of the event (loss of normal feedwater).

The steam flow/feedwater flow mismatch reactor trip is assumed unavailable (bypassed).

Results and Conclusions

The results of the complete LONF at 50% power transient are shown in Figures 10 through 18. The time sequence of events is presented in Table 2. The results show that reactor trip occurred on high pressurizer water level and that the high pressurizer water level setpoint (50% NRS) prevents the pressurizer from filling. Thus, the acceptance criterion for a LONF event is met.

CASE D: Main Feedwater Line Break Upstream of In-Containment Check Valves at 100% Power

Specific Assumptions

The plant is initially operating at 103% of rated power.

Initial reactor coolant average temperature is 4°F above the nominal full-power value (575.15°F).

Initial pressurizer water level is 37.5% NRS.

Main feedwater to all steam generators is assumed to stop at the time of the feedline break.

Pressurizer power-operated relief valves are available, but no credit is taken for the pressurizer sprays.

AFW is assumed to be manually actuated and the system manually aligned to deliver flow of 125 gpm to two intact steam generators 30 minutes after the initiation of the event (feedline break).

The steam flow/feedwater flow mismatch reactor trip is assumed to occur 10 seconds after the feedline break.

The steam generators will remain pressurized due to the in-containment check valves. This scenario initially behaves as a complete loss of normal feedwater.

Results and Conclusions

The results of the feedline break at full power located upstream of inside containment check valve transient are shown in Figures 19 through 27. The time sequence of events is presented in Table 3. Reactor trip is provided by the steam flow/feedwater flow mismatch signal. The results show that an AFW flow of 125 gpm initiated 30 minutes after the break is sufficient to remove core decay heat. The reactor coolant system (RCS) remains subcooled and the pressurizer does not fill. Thus, the core remains covered with water, and the acceptance criterion for a feedline break event is met.

CASE E: Main Feedwater Line Break Upstream of In-Containment Check Valves at 50% Power

Specific Assumptions

The plant is initially operating at 53% of rated power.

Initial reactor coolant average temperature is 4°F above the nominal value (551.5°F) corresponding to 50% power level on the nominal average temperature program (575.15°F at full power).

Initial pressurizer water level is 30.0% NRS.

Main feedwater to all steam generators is assumed to stop at the time of the feedline break.

Pressurizer power-operated relief valves are available, but no credit is taken for the pressurizer sprays.

AFW is assumed to be manually actuated and the system manually aligned to deliver flow of 125 gpm to two steam generators 15 minutes after the initiation of the event (feedline break).

The steam flow/feedwater flow mismatch reactor trip is assumed unavailable (by-passed).

The steam generators will remain pressurized due to the in-containment check valves. This scenario initially behaves as a complete loss of normal feedwater.

Results and Conclusions

The results of the feedline break at 50% power located upstream of inside containment check valve transient are shown in Figures 28 through 36. The time sequence of events is presented in Table 4. Reactor trip is provided by the high pressurizer water level (50% NRS) signal. The results show that an AFW flow of 125 gpm initiated 15 minutes after the break is sufficient to remove core decay heat. The reactor coolant system (RCS) remains subcooled and the pressurizer does not fill. Thus, the core remains covered with water, and the acceptance criterion for a feedline break event is met.

CASE F: Main Feedwater Line Break Downstream of In-Containment Check Valves at 100% Power

Case F analysis was previously submitted by letter from M. O. Medford (SCE) to G. E. Lear (NRC) dated May 1, 1986. This analysis showed that with AFW flow of 250 gpm to 2 steam generators initiated at 15 minutes, the acceptance criterion of no pressurizer fill was achieved. Reactor trip occurred on steam flow/feedwater flow mismatch.

CASE G: Main Feedwater Line Break Downstream of the In-Containment Check Valves at 50% power

Specific Assumptions

The plant is initially operating at 53% of rated power.

Initial reactor coolant average temperature is 4°F above the nominal value (551.5°F) corresponding to 50% power level on the nominal average temperature program (575.15°F at full power).

Initial pressurizer water level is 30.0% narrow range span (NRS).

Main feedwater to all steam generators is assumed to stop at the time of the feedline break.

Pressurizer power-operated relief valves are available, but no credit is taken for the pressurizer sprays.

AFW is assumed to be manually actuated and the system manually aligned to deliver flow of 250 gpm to two steam generators 15 minutes after the initiation of the event (feedline break).

The steam flow/feedwater flow mismatch reactor trip is assumed unavailable (by-passed).

All three steam generators depressurize since SONGS 1 does not have main steamline isolation valves.

Results and Conclusions

The results of the feedline break at 50% power located downstream of inside containment check valve transient are shown in Figures 37 through 45. The time sequence of events is presented in Table 5. Reactor trip is provided by the high pressurizer pressure signal. The results show that an AFW flow of 250 gpm initiated 15 minutes after the break is sufficient to remove core decay heat. Calculations of this case show that the core remained in a coolable geometry during this feedwater line break scenario. The detailed calculations involve showing that the mass relieved through the pressurizer PORVs (between the time of initial relief through the PORVs and the time the PORVs reseat due to the heat removal capability of the AFW exceeds the core decay heat) was not sufficient to uncover the core. As such, the acceptance criterion for a FLB event that the core remains in a coolable geometry during the transient was shown to be met.

TABLE 1

TIME SEQUENCE OF EVENTS FOR CASE A PARTIAL LONF

Event	Time, sec
Feedwater flow control valve fails — terminating feed flow to one steam generator	10.
Reactor trip on high pressurizer water level	96.
Rods begin to drop	98.
Main feedwater flow is terminated for remaining two steam generators	98.
Pressurizer PORVs open (2200 psia)	1191.
AFW manually started of 185 gpm to 3 steam generators	1810.
Cold AFW reaches 3 steam generators	2341.
Pressurizer PORVs close	2357.
Heat removal of AFW is capable of removing core decay heat	2366.
Peak pressurizer water volume (1228 ft3)	2389.

TABLE 2

TIME SEQUENCE OF EVENTS FOR CASE C LONF 50% POWER

Event	Time, sec
Complete loss of normal feedwater	10.
Reactor trip on high pressurizer water level	136.
Rods begin to drop	138.
Pressurizer PORVs open (2200 psia)	1055.
AFW manually started of 185 gpm to 3 steam generators	1810.
Pressurizer PORVs close	1871.
Peak pressurizer water volume (916 ft3)	1871.
Heat removal of AFW is capable of removing core decay heat	1876.
Cold AFW reaches 3 steam generators	2341.

TABLE 3

TIME SEQUENCE OF EVENTS FOR CASE D FLB FULL POWER

Event	Time, sec
Feedline Break between the 2 MFW check valves	10.
Reactor trip on steam flow/feed flow mismatch	20.
Rods begin to drop	20.
Pressurizer PORVs open (2200 psia)	1227.
AFW manually started of 125 gpm to 2 steam generators	1810.
Cold AFW reaches 2 steam generators	2341.
Pressurizer PORVs close	4373.
Heat removal of AFW is capable of removing core decay heat	4550.

TABLE 4

TIME SEQUENCE OF EVENTS FOR CASE E FLB 50% POWER

Event	Time, sec
Feedline Break between the 2 MFW check valves	10.
Reactor trip on high pressurizer water level (50% NRS)	160.
Rods begin to drop	162.
Pressurizer PORVs open (2200 psia)	476.
AFW manually started of 125 gpm to 2 steam generators	910.
Cold AFW reaches 2 steam generators	1437.
Pressurizer PORVs close	1460.
Heat removal of AFW is capable of removing core decay heat	1460.

TABLE 5

TIME SEQUENCE OF EVENTS FOR CASE G FLB 50% POWER

Event	Time, sec
Feedline Break downstream of the check valves inside containment	10.
Pressurizer PORVs open (2200 psia)	76.
Reactor trip on high pressurizer pressure	84.
Rods begin to drop	86.
AFW manually started of 250 gpm to 2 steam generators	910.
Cold AFW reaches 2 steam generators	1173.
Heat removal of AFW is capable of removing core decay heat	1177.
Pressurizer PORVs close	1185.

CASE A

SCE Partial LONF at Full Power

AFW Flow of 185 gpm @ 30 min.

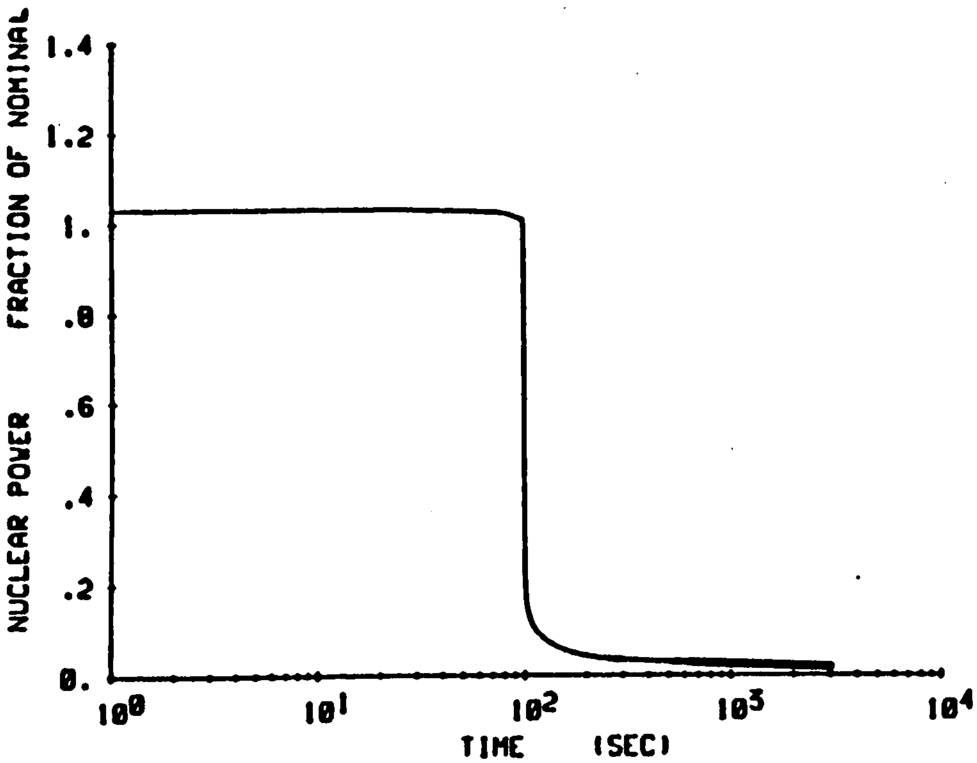


Figure 1

CASE A

SCE Partial LONF at Full Power

AFW Flow of 185 gpm @ 30 min.

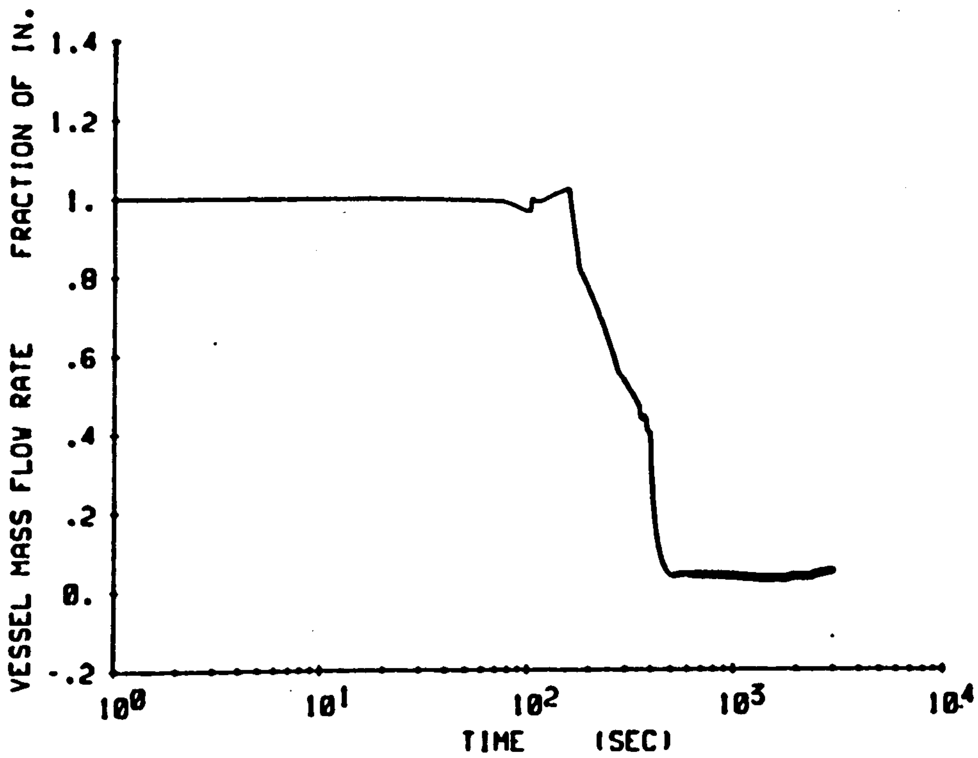


Figure 2

CASE A

SCE Partial LONF at Full Power

AFW Flow of 185 gpm @ 30 min.

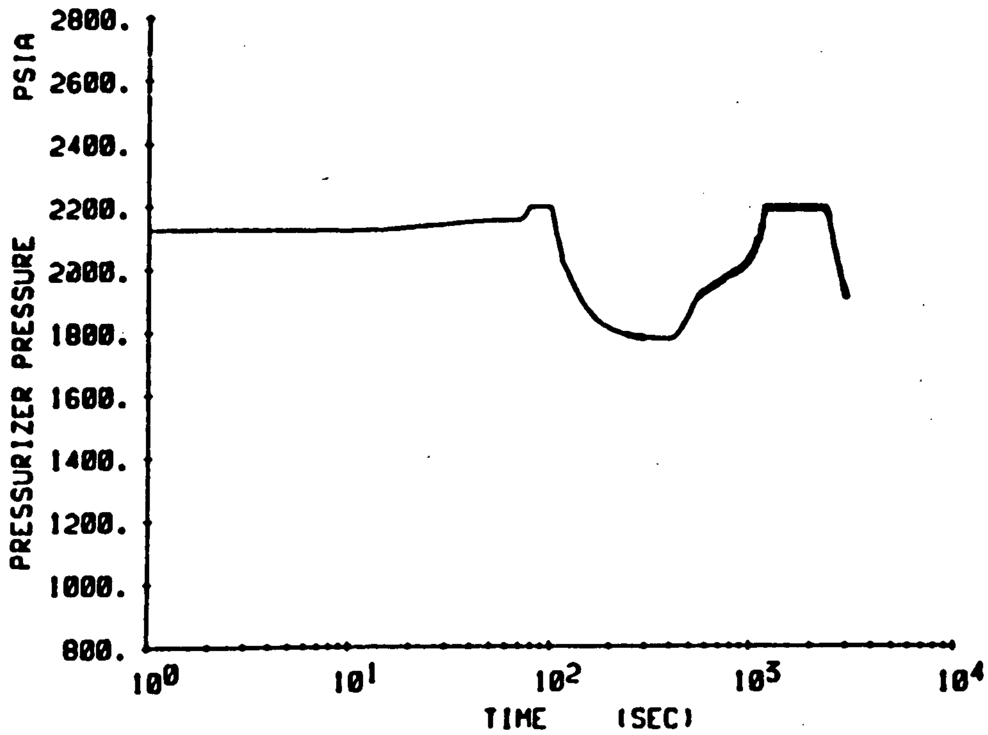


Figure 3

CASE A

SCE Partial LONF at Full Power
AFW Flow of 185 gpm @ 30 min.

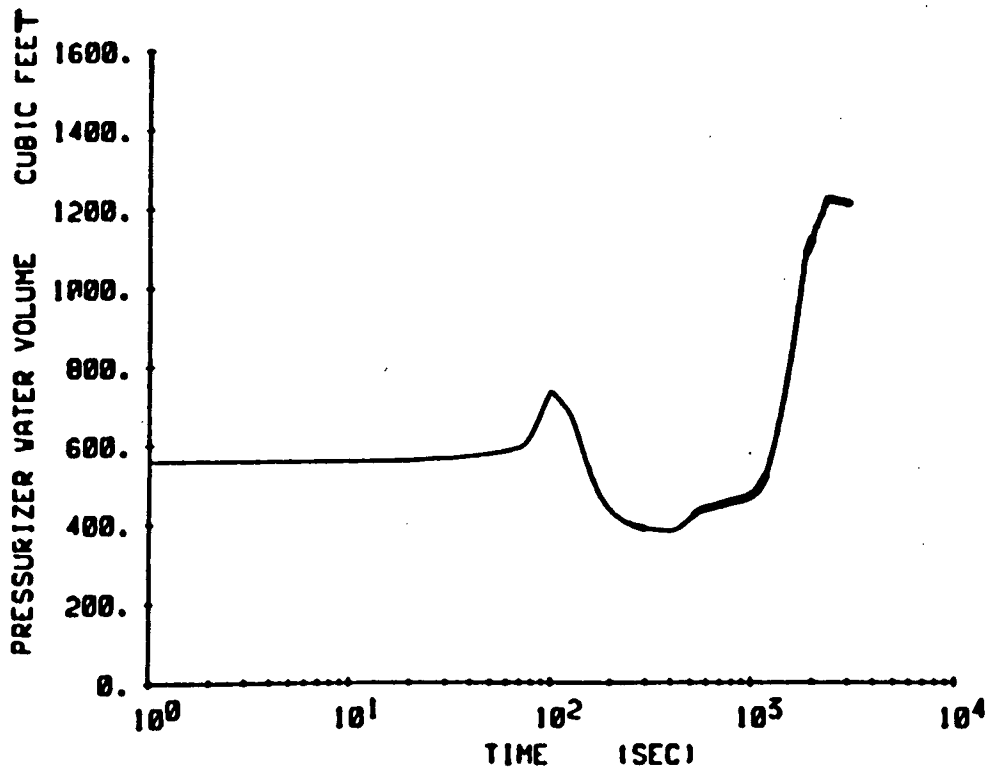
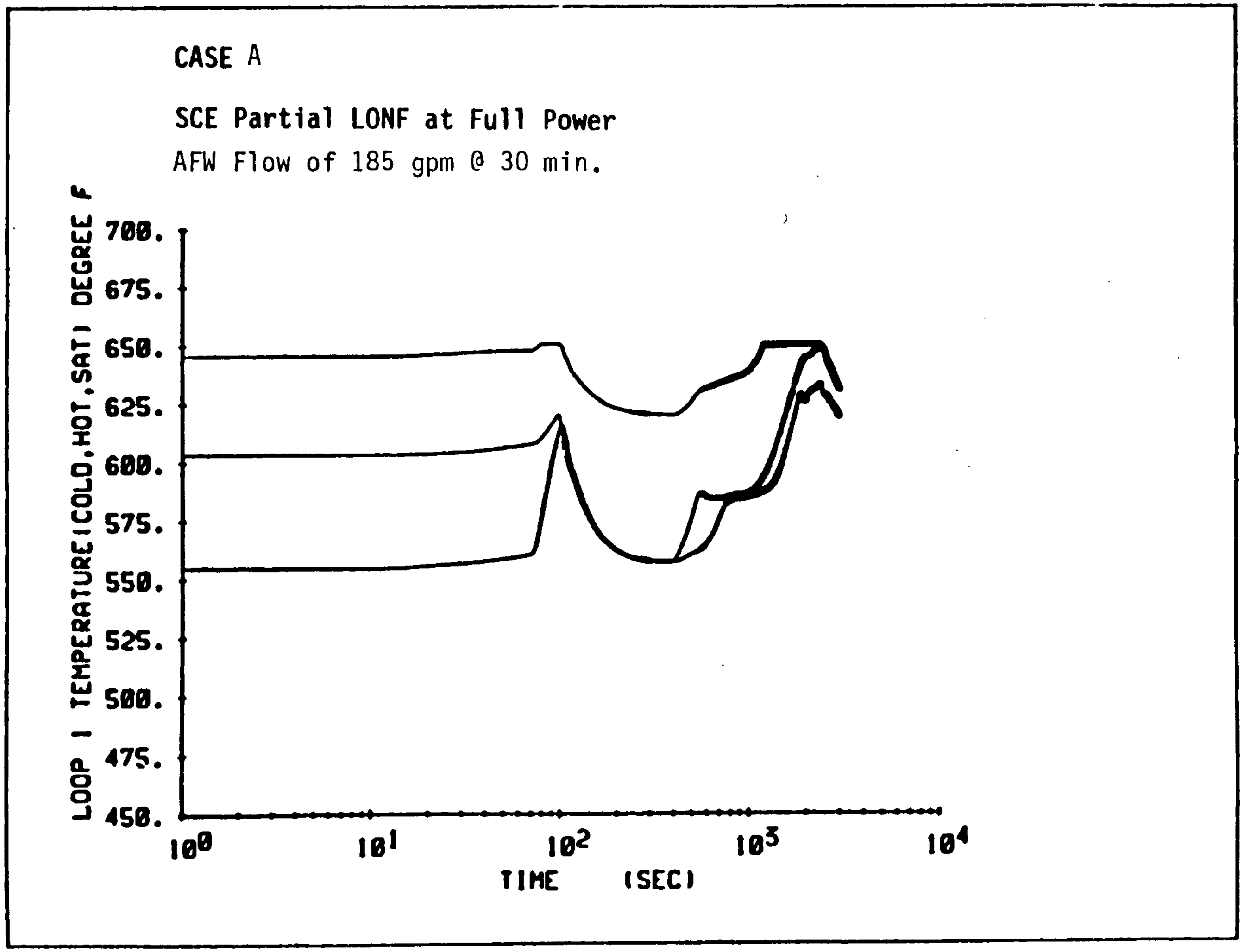


Figure 4



CASE A
SCE Partial LONF at Full Power
AFW Flow of 185 gpm @ 30 min.
LOOP 1 TEMPERATURE(COLD,HOT,SAT) DEGREE F
700.
675.
650.
625.
600.
575.
550.
525.
500.
475.
450.
10^0
10^1
10^2
10^3
10^4
TIME (SEC)

Figure 5

CASE A

SCE Partial LONF at Full Power

AFW Flow of 185 gpm @ 30 min.

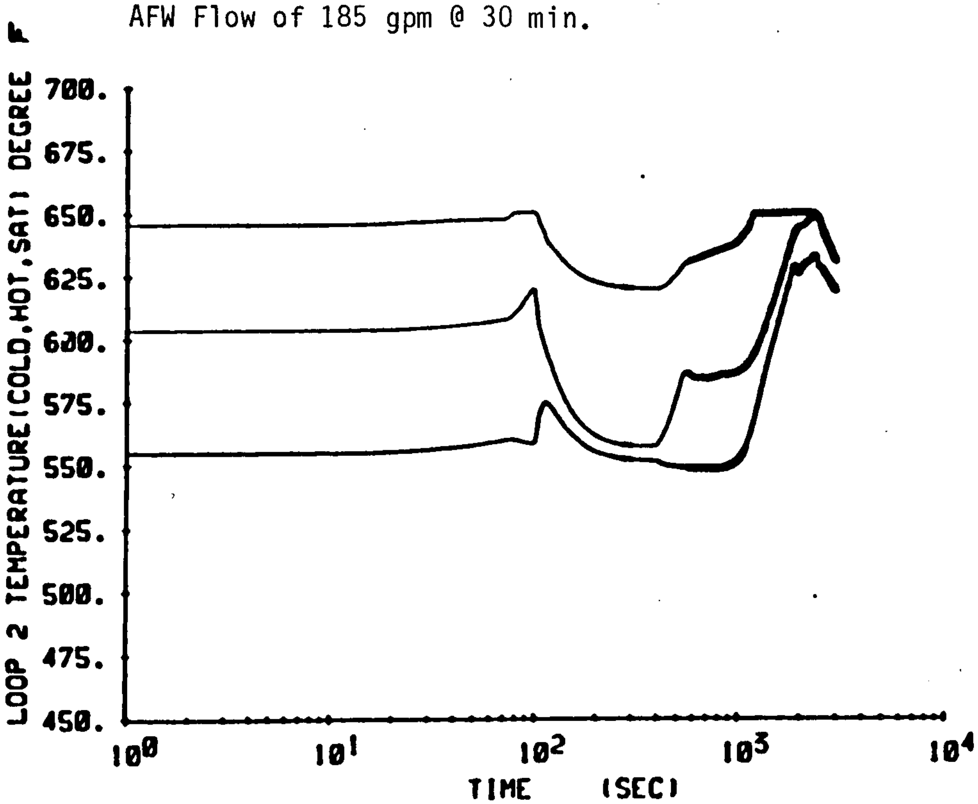
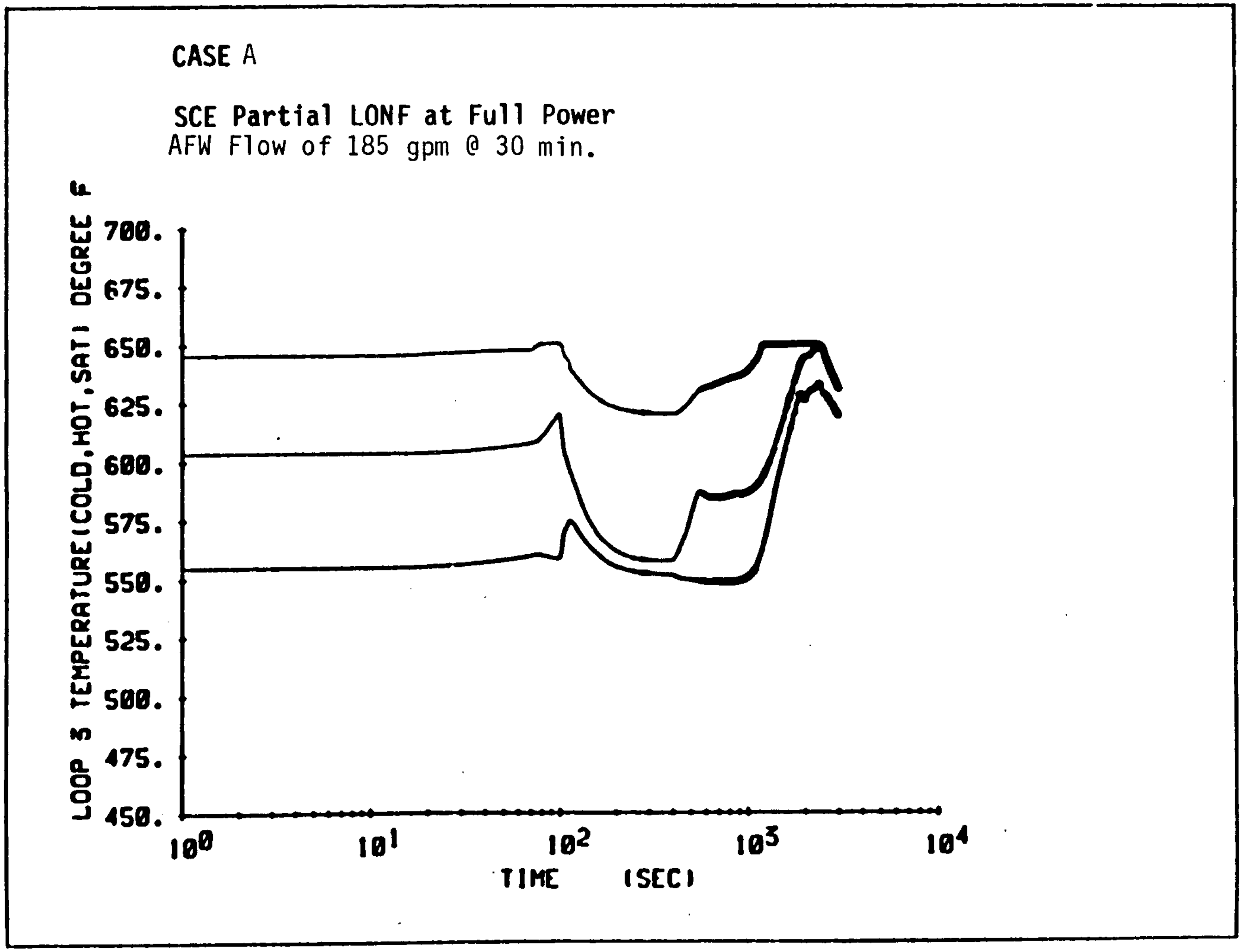


Figure 6



CASE A
SCE Partial LONF at Full Power
AFW Flow of 185 gpm @ 30 min.
LOOP 3 TEMPERATURE(COLD,HOT,SAT) DEGREE F
700.
675.
650.
625.
600.
575.
550.
525.
500.
475.
450.
10^0
10^1
10^2
10^3
10^4
TIME (SEC)

Figure 7

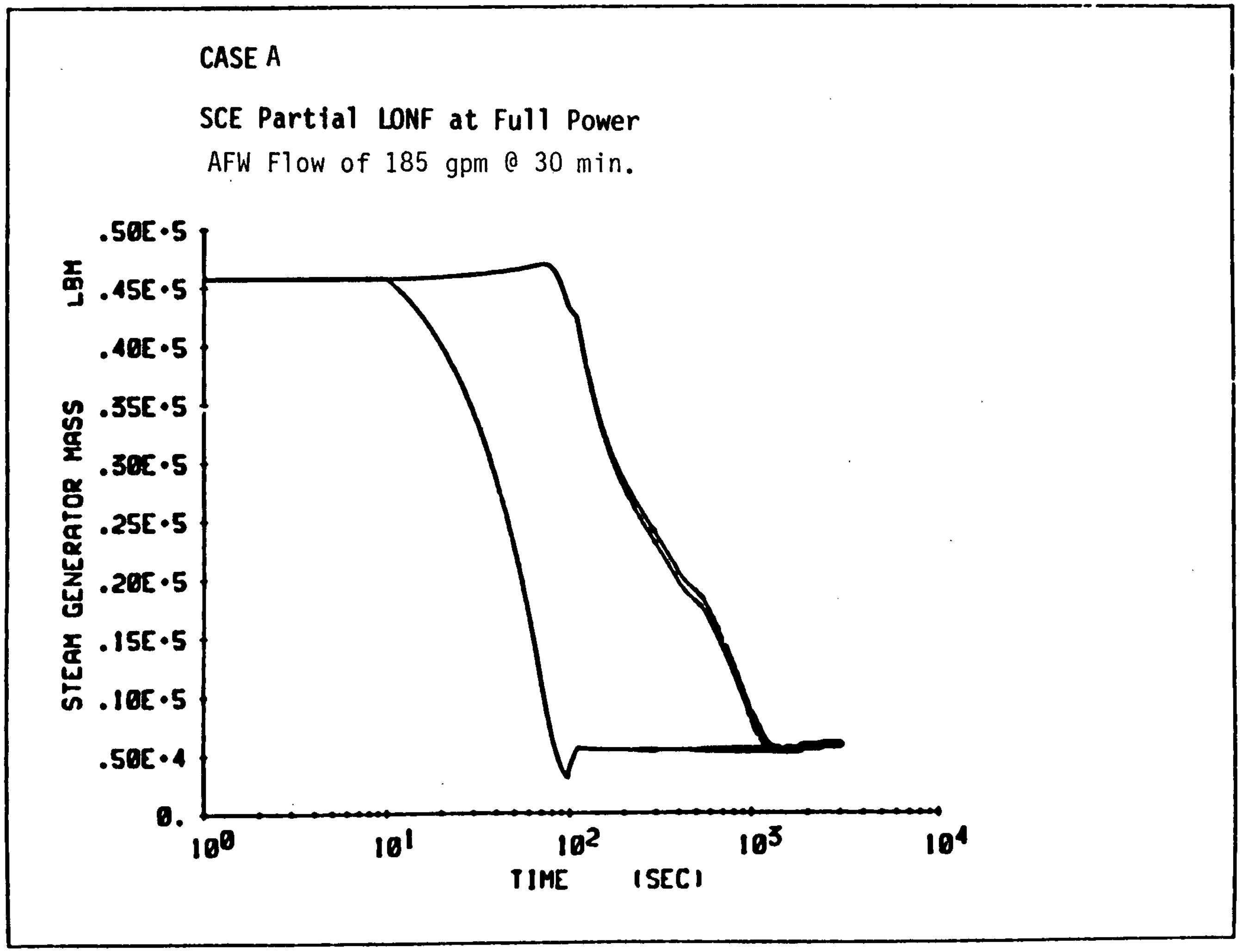
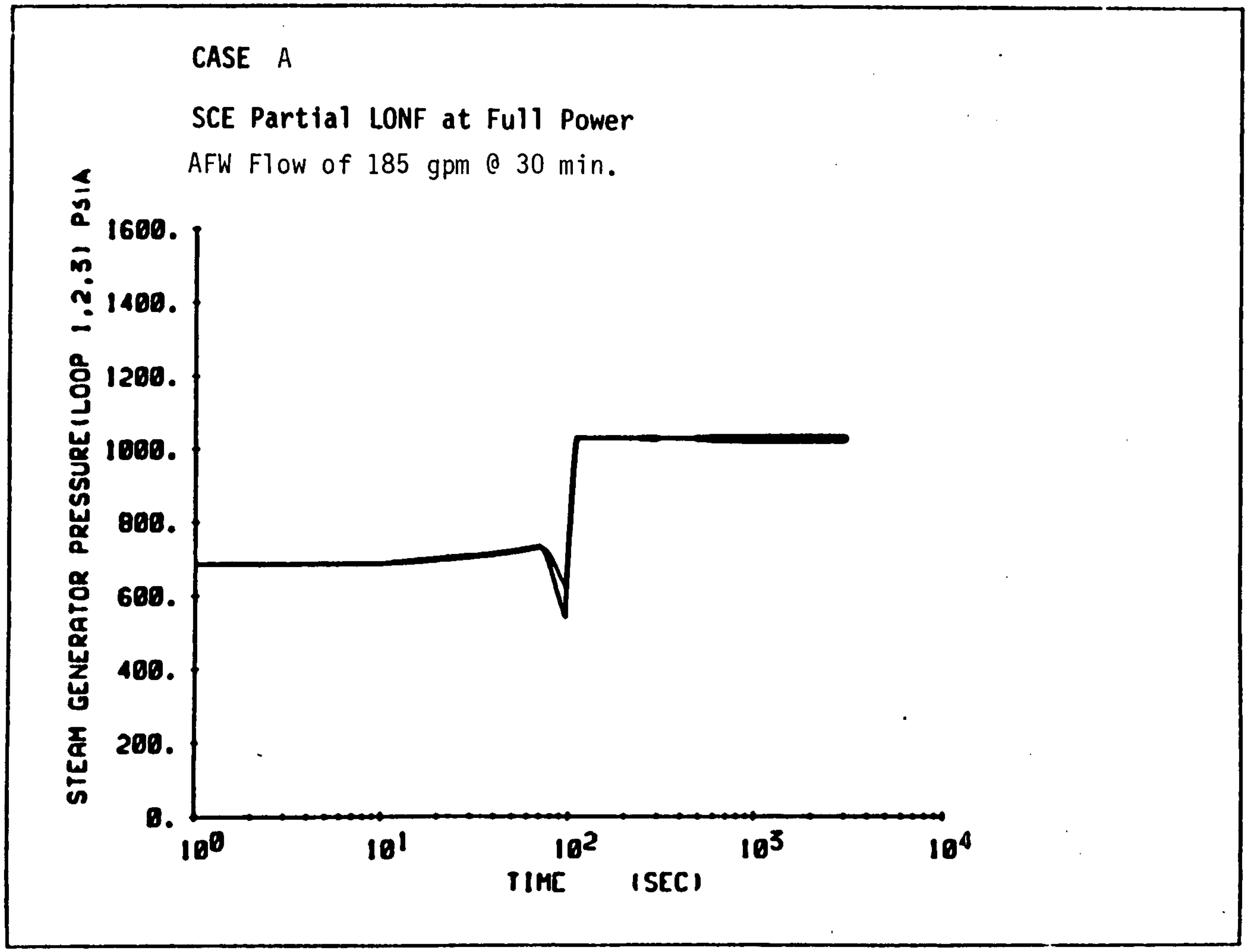

CASE A
SCE Partial LONF at Full Power
AFW Flow of 185 gpm @ 30 min.
STEAM GENERATOR MASS LBM
.50E+5
.45E+5
.40E+5
.35E+5
.30E+5
.25E+5
.20E+5
.15E+5
.10E+5
.50E+4
0.
10^0
10^1
10^2
10^3
10^4
TIME (SEC)

Figure 8



CASE A
SCE Partial LONF at Full Power
AFW Flow of 185 gpm @ 30 min.
STEAM GENERATOR PRESSURE(LOOP 1,2,3) PSIA
1600.
1400.
1200.
1000.
800.
600.
400.
200.
0.
10^0
10^1
10^2
10^3
10^4
TIME (SEC)

Figure 9

CASE C

SCE Complete LONF at 50% Power
AFW Flow of 185 gpm @ 30 min.

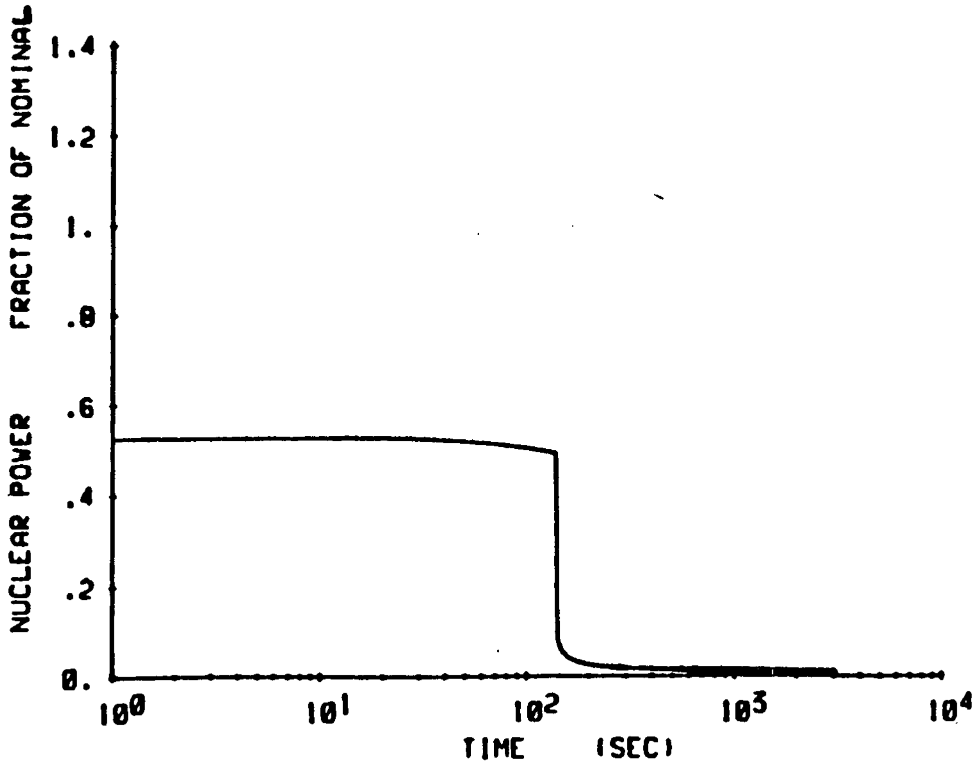


Figure 10

CASE C

SCE Complete LONF at 50% Power

AFW Flow of 185 gpm @ 30 min.

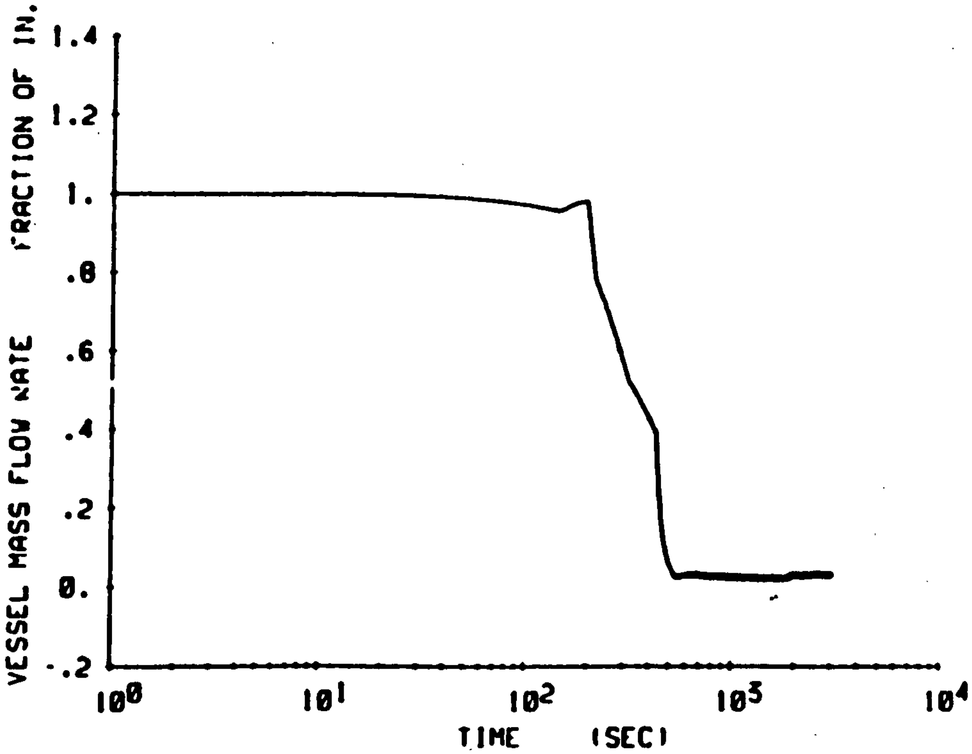


Figure 11

CASE C

SCE Complete LONF at 50% Power
AFW Flow of 185 gpm @ 30 min.

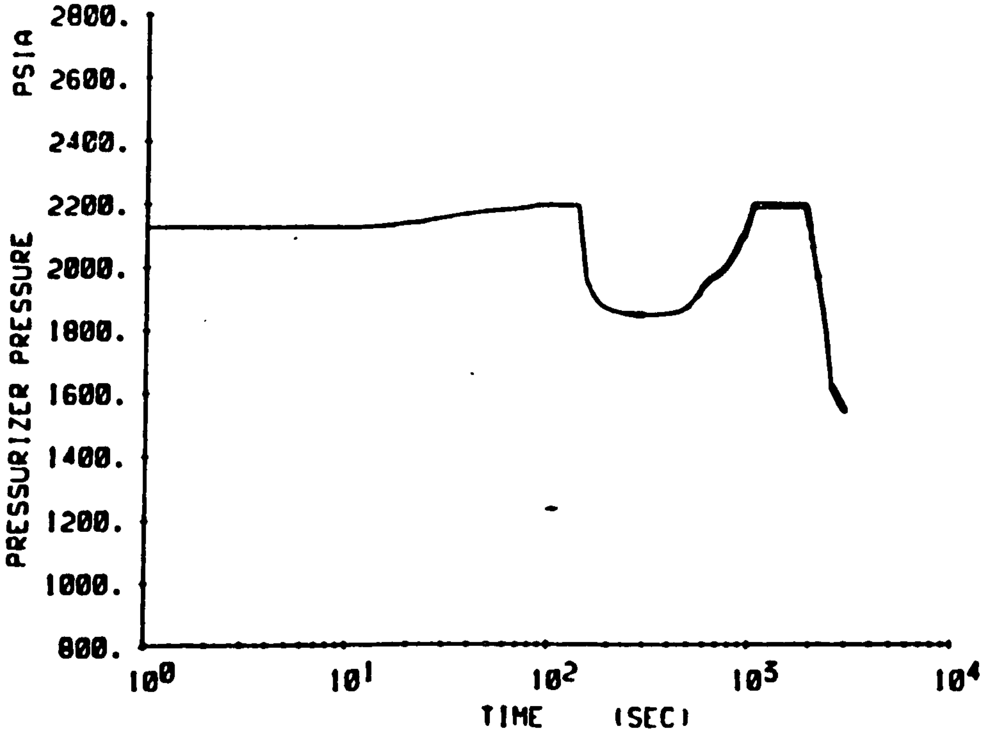


Figure 12

CASE C

SCE Complete LONF at 50% Power
AFW Flow of 185 gpm @ 30 min.

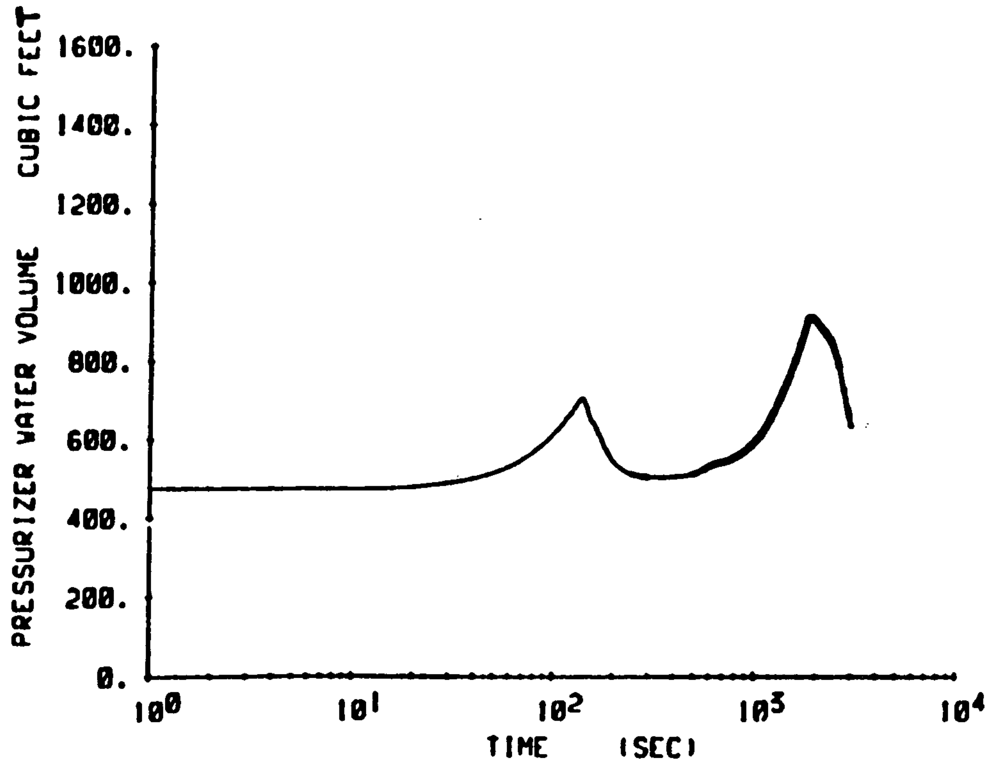


Figure 13

CASE C

SCE Complete LONF at 50% Power
AFW Flow of 185 gpm @ 30 min.

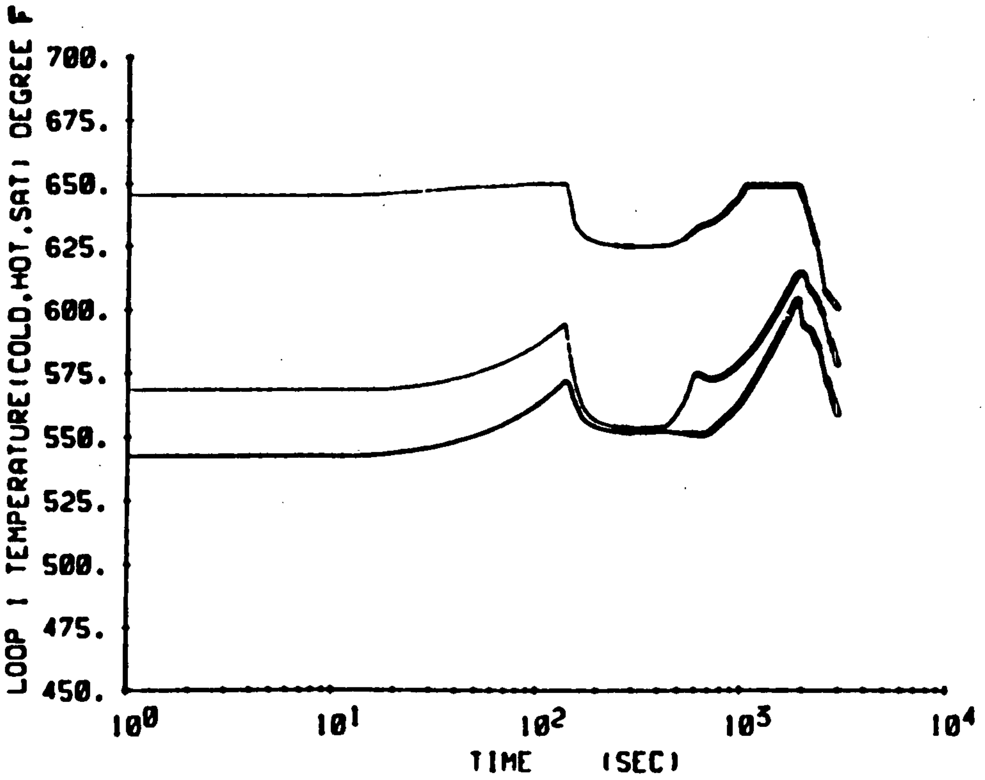


Figure 14

CASE C

SCE Complete LONF at 50% Power
AFW Flow of 185 gpm @ 30 min.

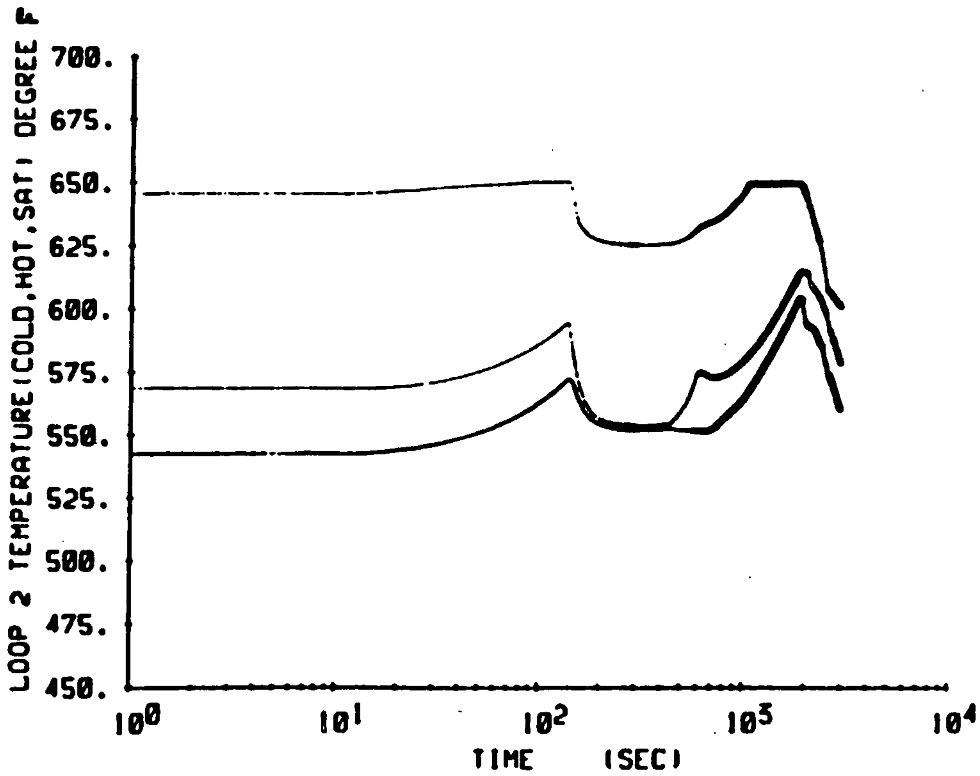


Figure 15

CASE C

SCE Complete LONF at 50% Power
AFW Flow of 185 gpm @ 30 min.

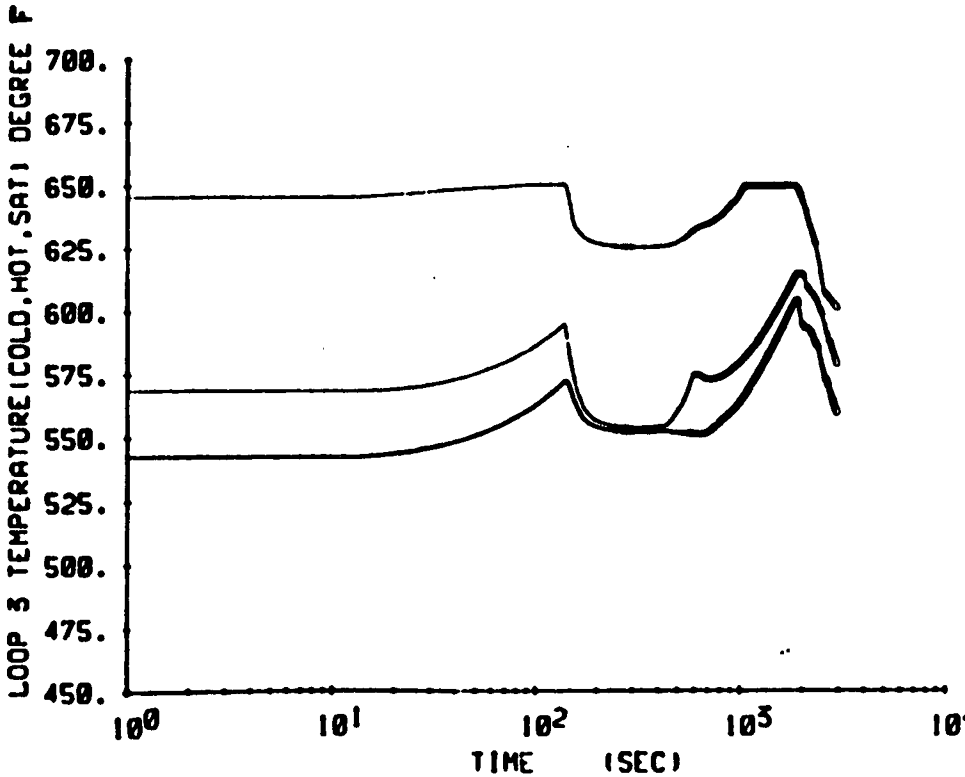


Figure 16

CASE C

SCE Complete LONF at 50% Power
AFW Flow of 185 gpm @ 30 min.

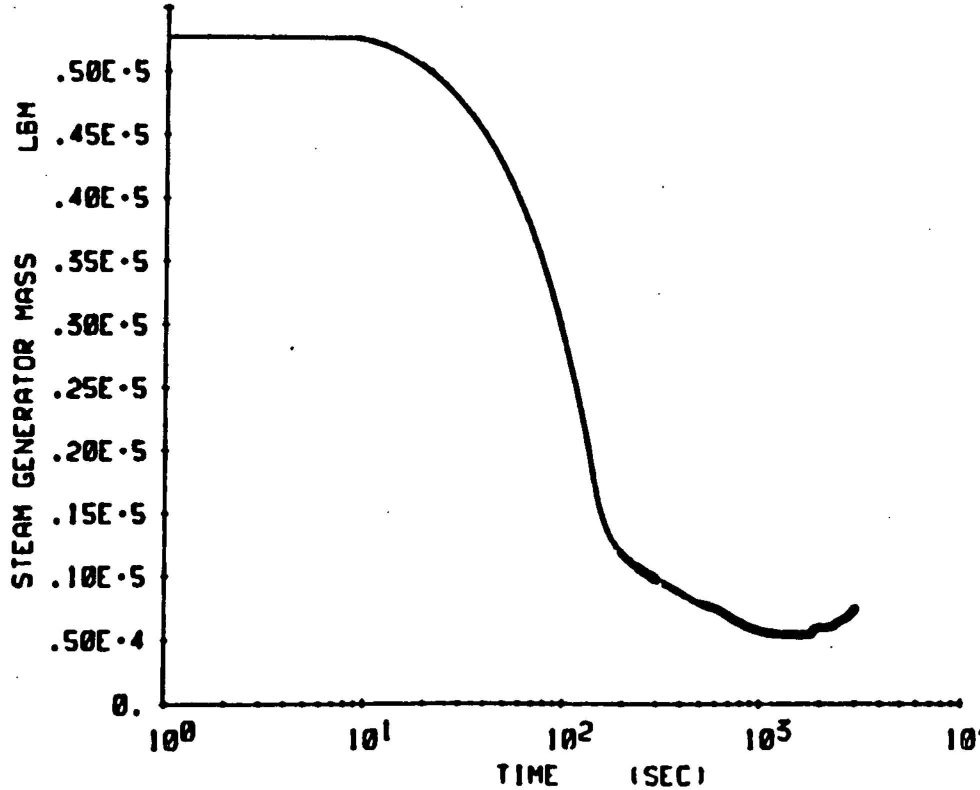


Figure 17

CASE C

SCE Complete LONF at 50% Power
AFW Flow of 185 gpm @ 30 min.

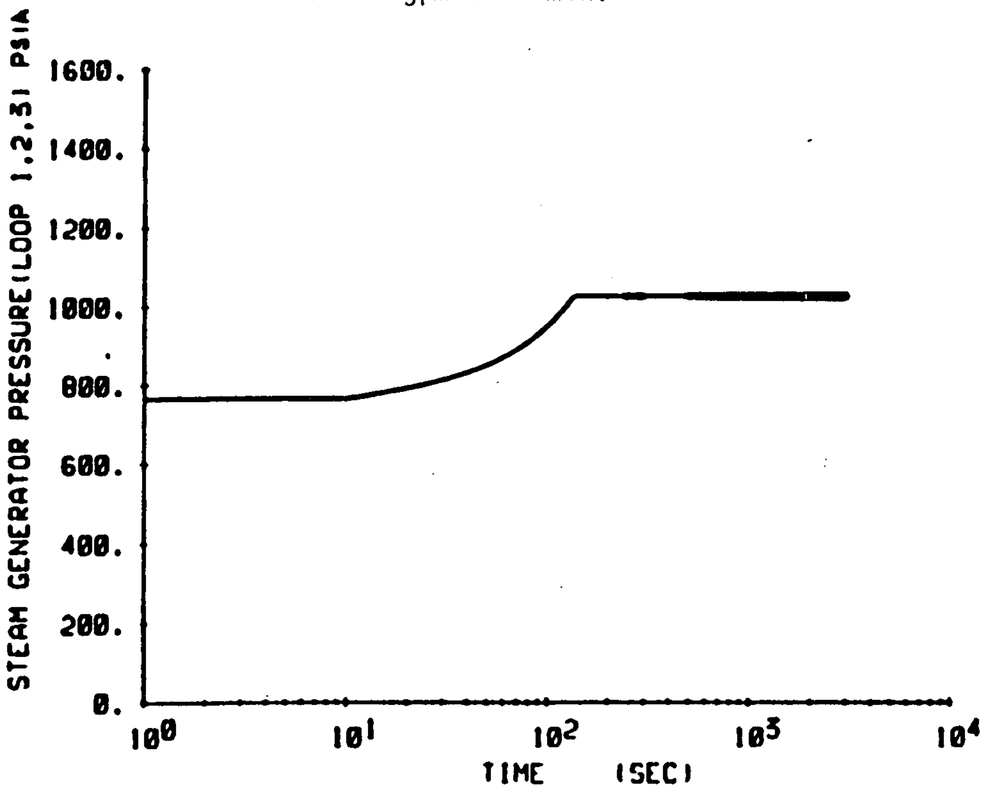


Figure 18

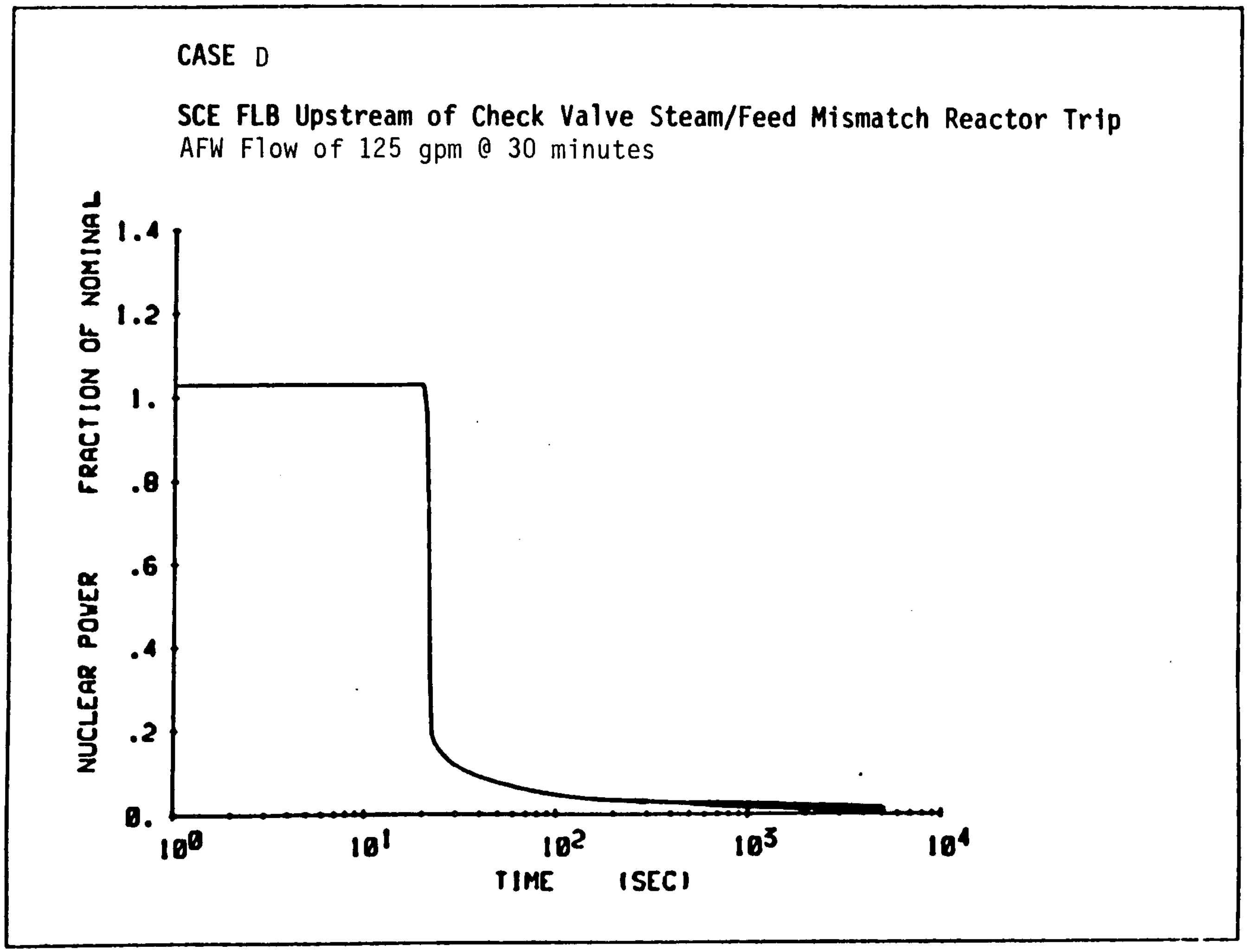

CASE D
SCE FLB Upstream of Check Valve Steam/Feed Mismatch Reactor Trip
AFW Flow of 125 gpm @ 30 minutes
NUCLEAR POWER FRACTION OF NOMINAL
1.4
1.2
1.
.8
.6
.4
.2
0.
10^0
10^1
10^2
10^3
10^4
TIME (SEC)

Figure 19

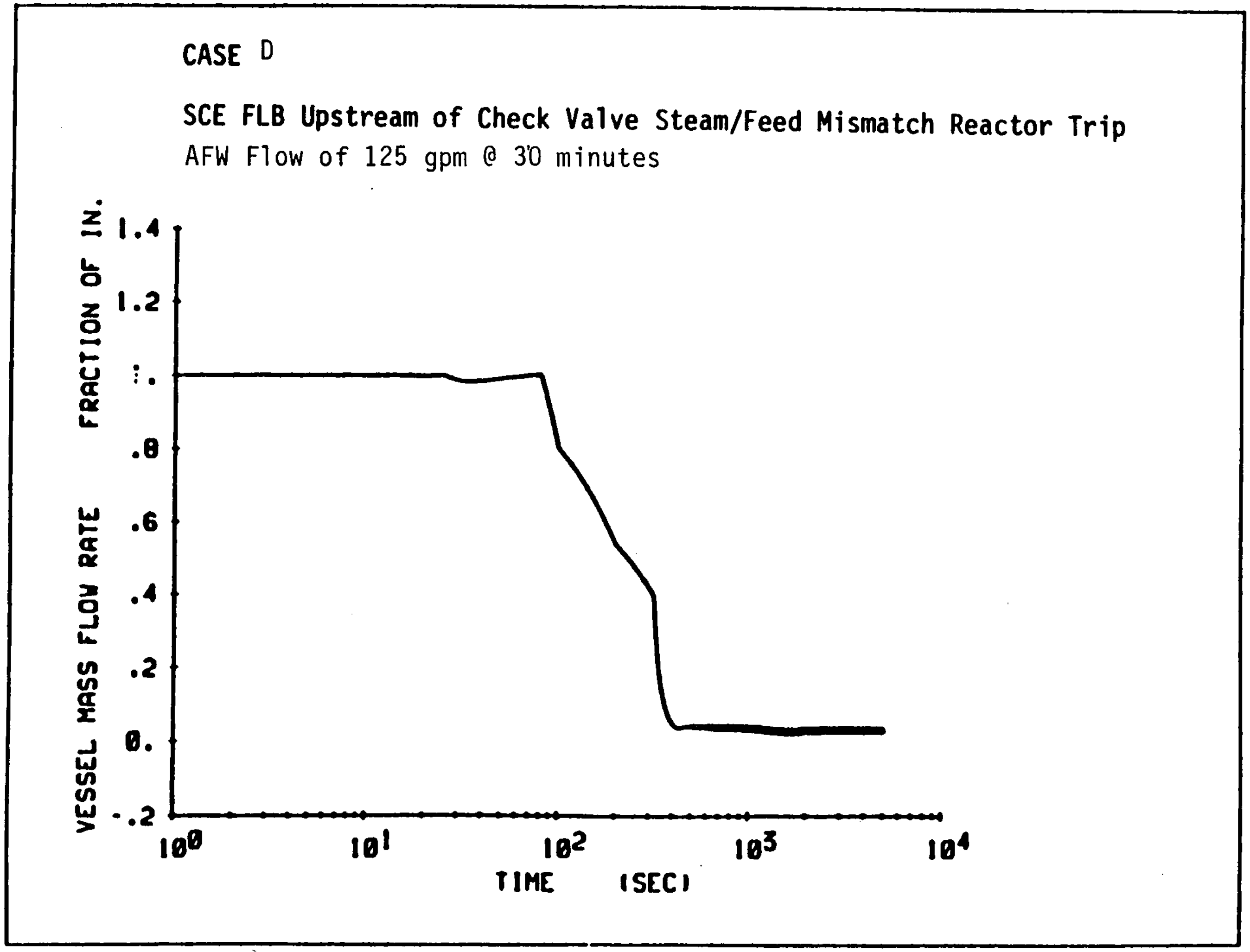

CASE D
SCE FLB Upstream of Check Valve Steam/Feed Mismatch Reactor Trip
AFW Flow of 125 gpm @ 30 minutes
VESSEL MASS FLOW RATE FRACTION OF IN.
1.4
1.2
1.
.8
.6
.4
.2
0.
-.2
10^0
10^1
10^2
10^3
10^4
TIME (SEC)

Figure 20

CASE D

SCE FLB Upstream of Check Valve Steam/Feed Mismatch Reactor Trip
AFW Flow of 125 gpm @ 30 minutes

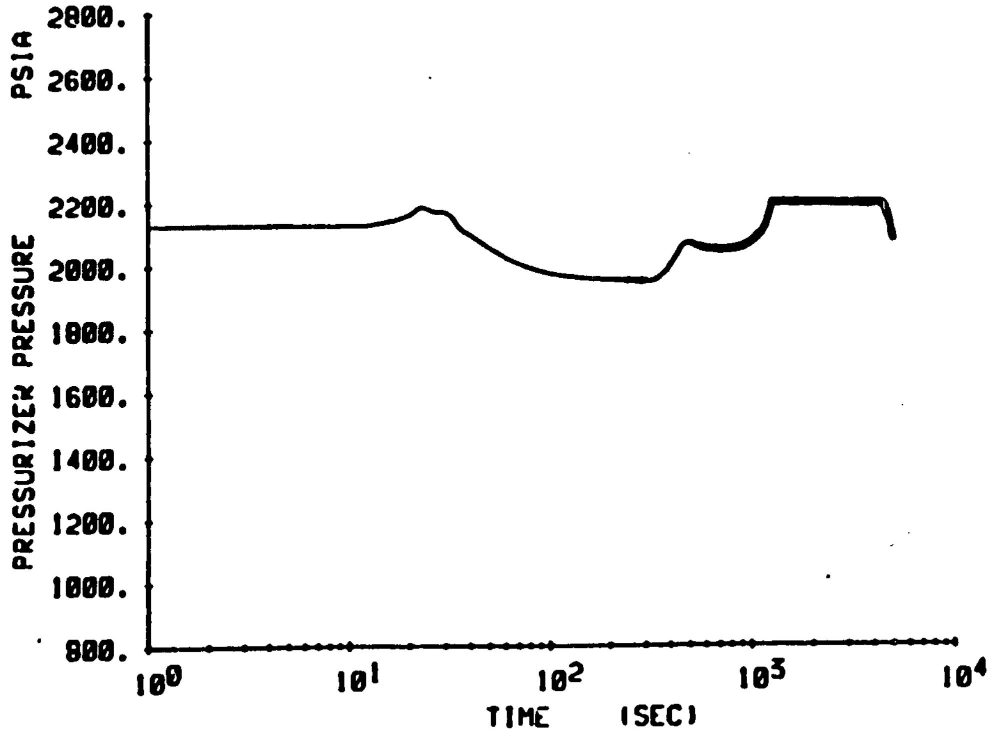


Figure 21

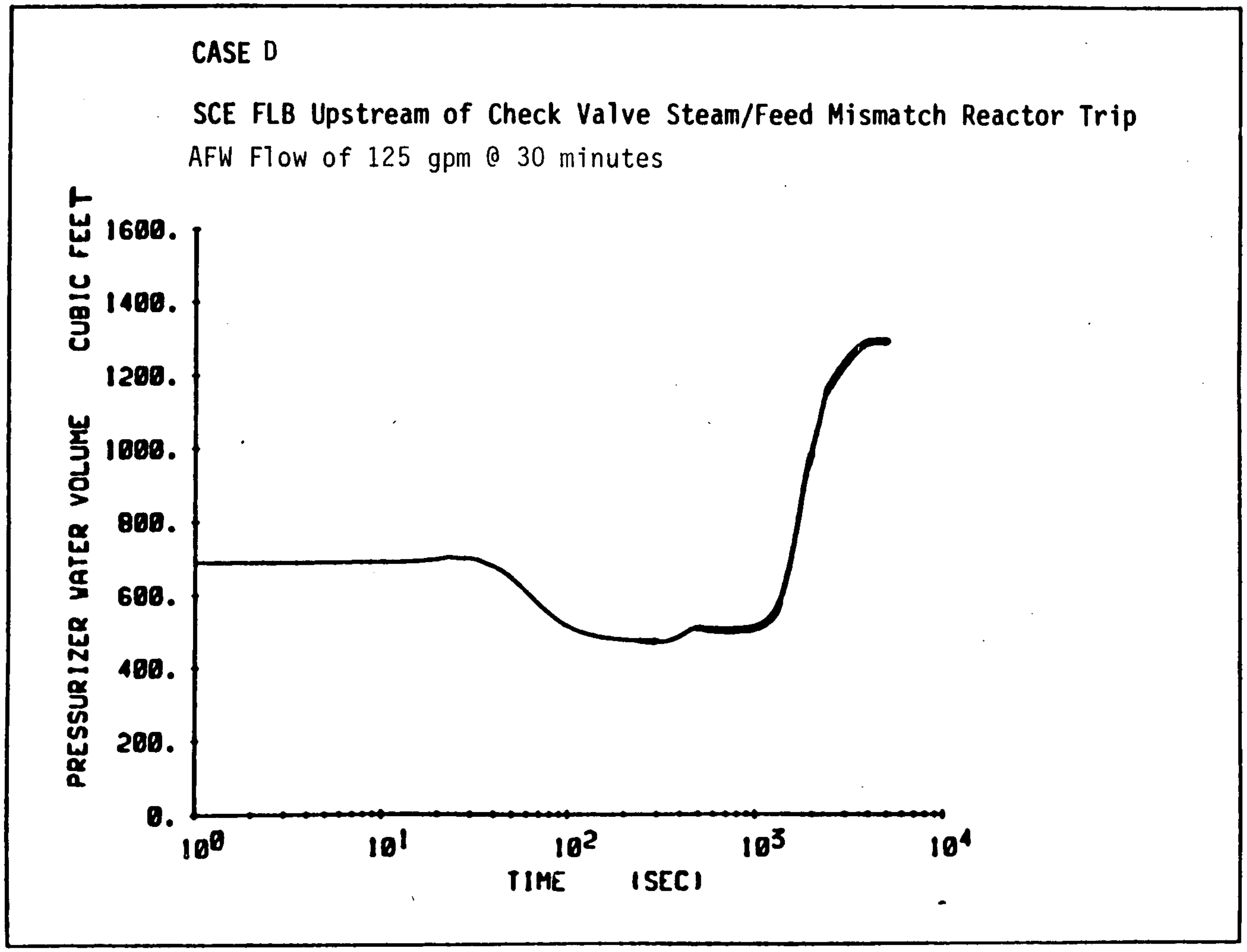
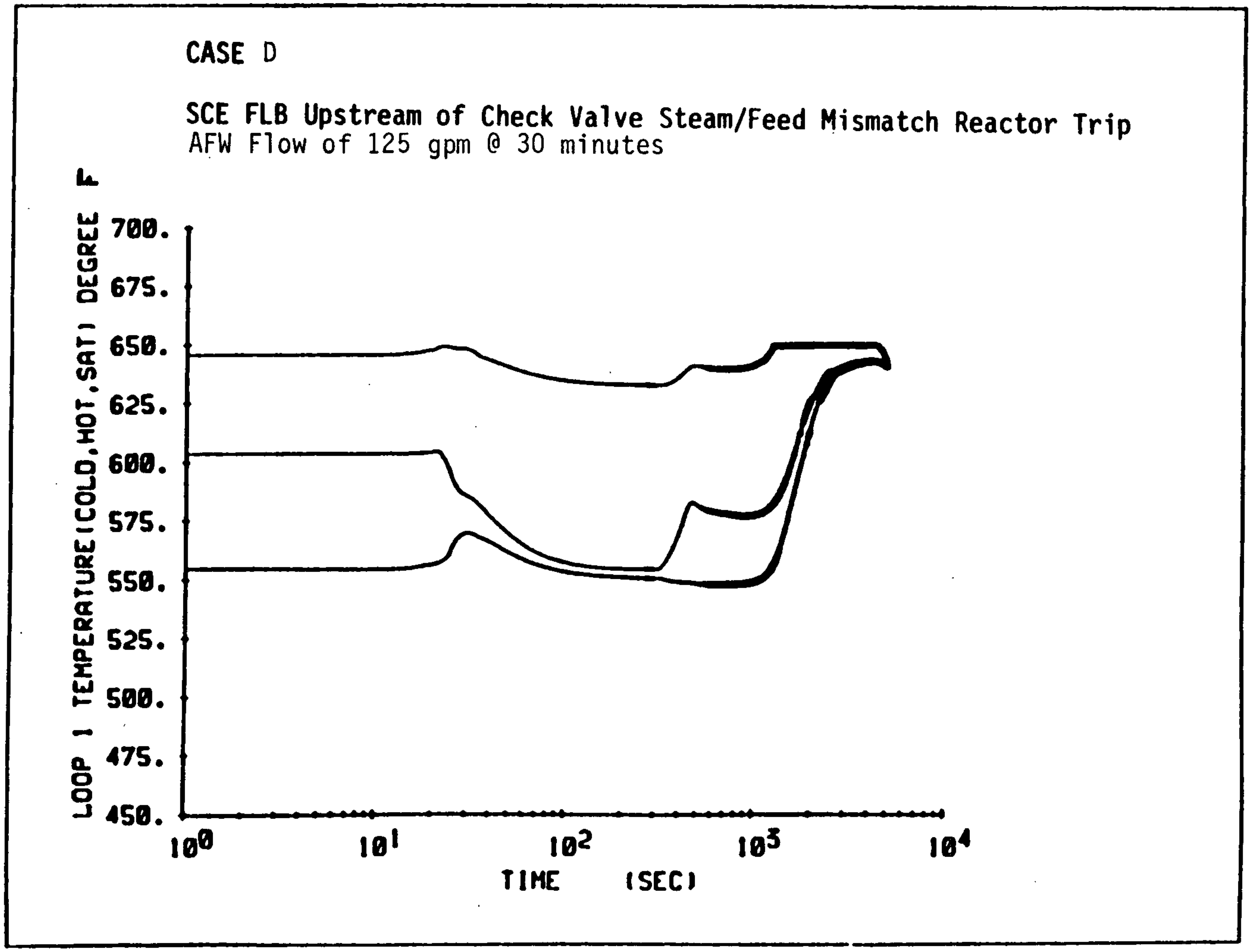

CASE D
SCE FLB Upstream of Check Valve Steam/Feed Mismatch Reactor Trip
AFW Flow of 125 gpm @ 30 minutes
PRESSURIZER WATER VOLUME CUBIC FEET
1600.
1400.
1200.
1000.
800.
600.
400.
200.
0.
10^0
10^1
10^2
10^3
10^4
TIME (SEC)

Figure 22



CASE D
SCE FLB Upstream of Check Valve Steam/Feed Mismatch Reactor Trip
AFW Flow of 125 gpm @ 30 minutes
LOOP 1 TEMPERATURE(COLD,HOT,SAT) DEGREE F
700.
675.
650.
625.
600.
575.
550.
525.
500.
475.
450.
10^0
10^1
10^2
10^3
10^4
TIME (SEC)

Figure 23

CASE D

SCE FLB Upstream of Check Valve Steam/Feed Mismatch Reactor Trip
AFW Flow of 125 gpm @ 30 minutes

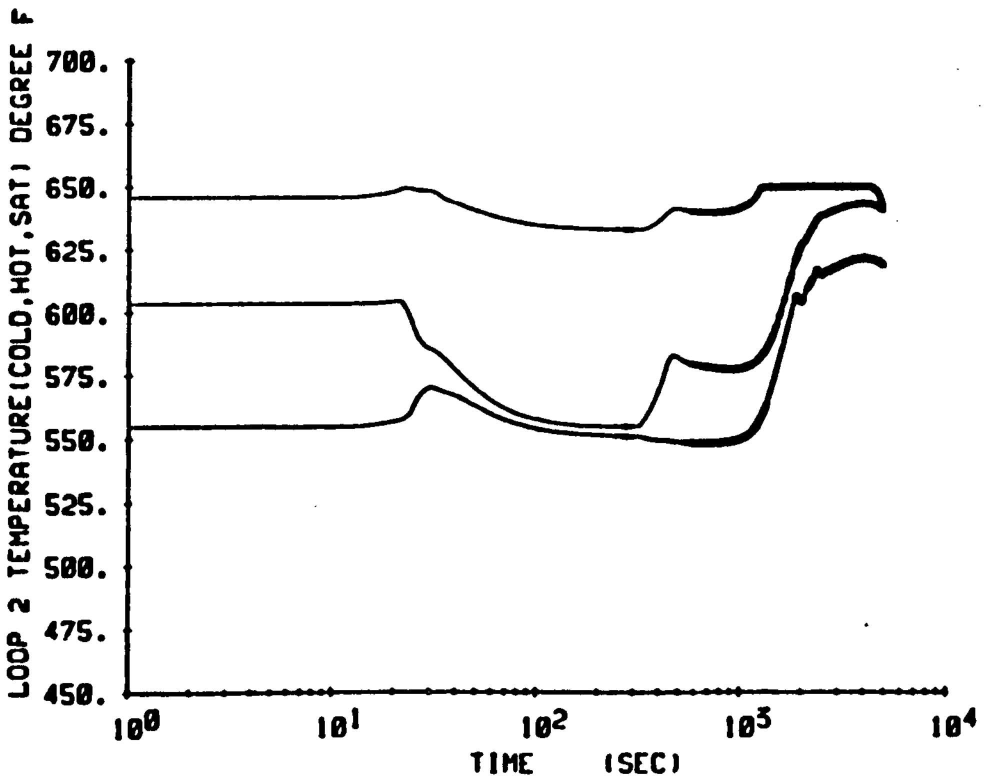


Figure 24

CASE D

SCE FLB Upstream of Check Valve Steam/Feed Mismatch Reactor Trip
AFW Flow of 125 gpm @ 30 minutes

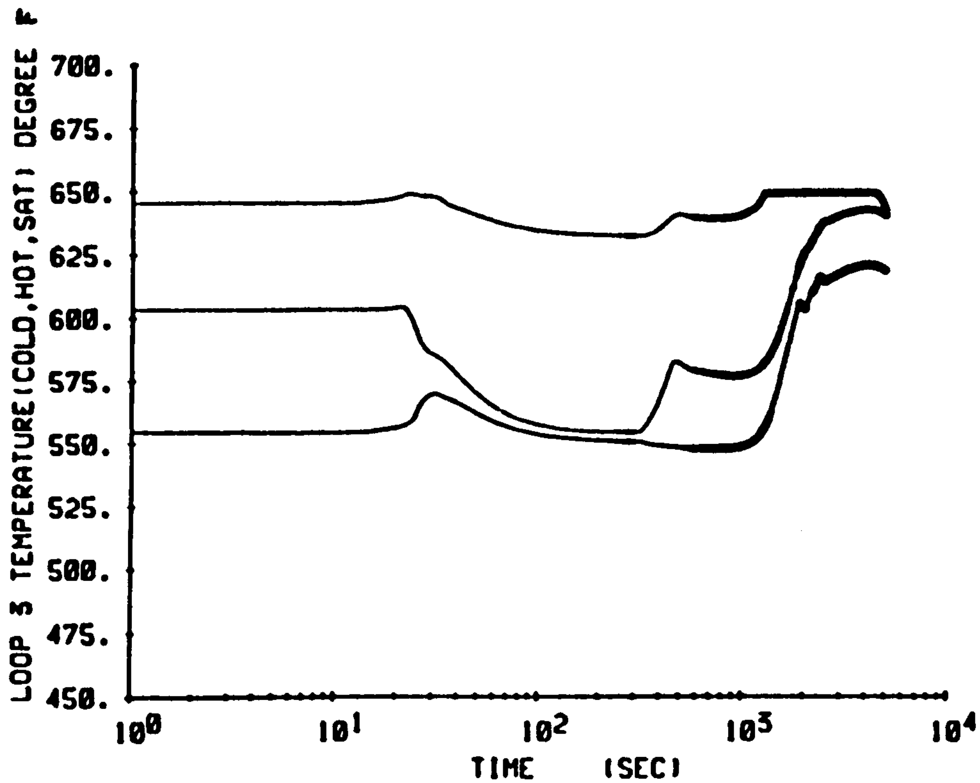


Figure 25

CASE D

SCE FLB Upstream of Check Valve Steam/Feed Mismatch Reactor Trip
AFW Flow of 125 gpm @ 30 minutes

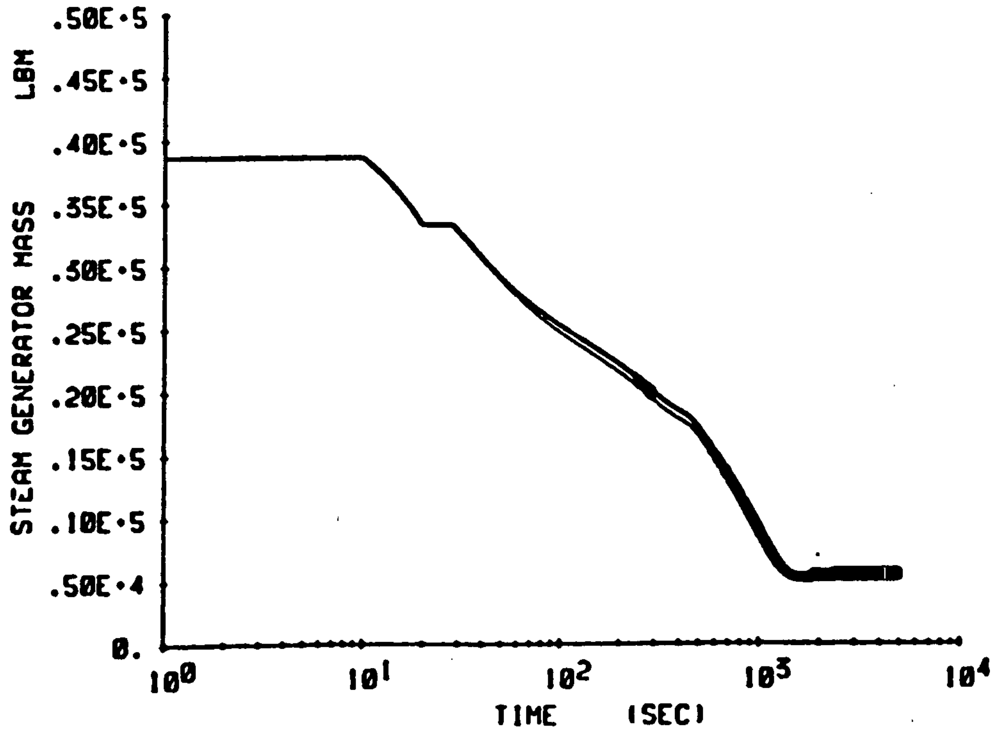


Figure 26

CASE D

SCE FLB Upstream of Check Valve Steam/Feed Mismatch Reactor Trip
AFW Flow of 125 gpm @ 30 minutes

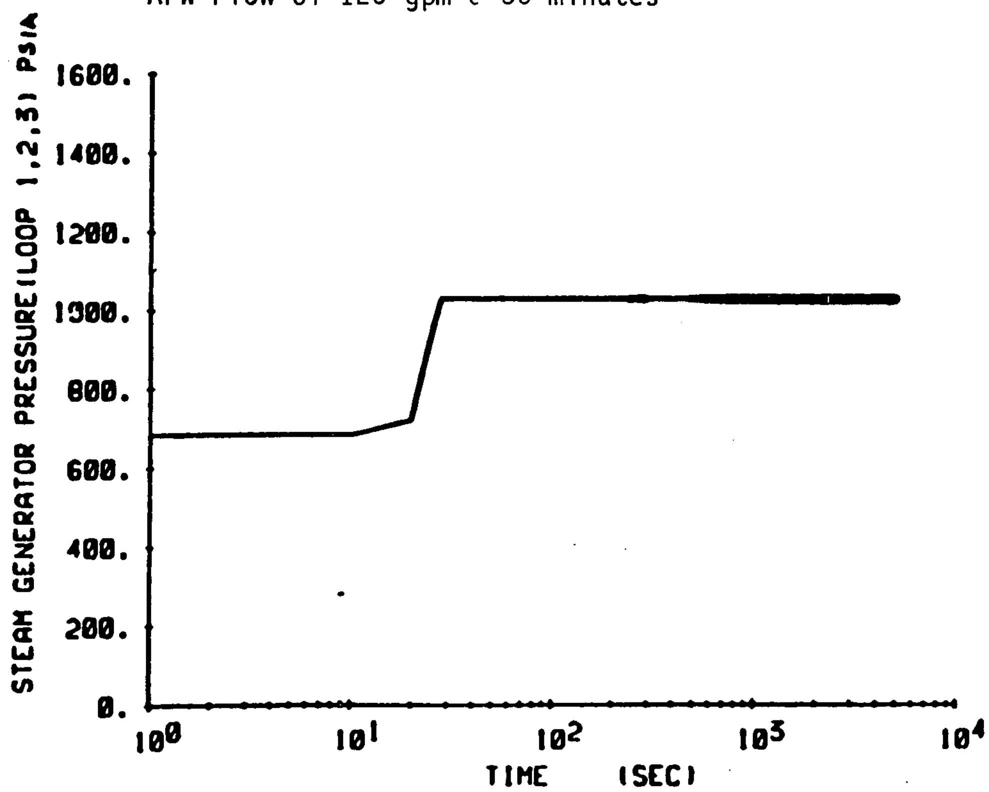


Figure 27

CASE E

SCE FLB Upstream Check Valve at 50% Power
AFW Flow of 125 gpm @ 15 minutes

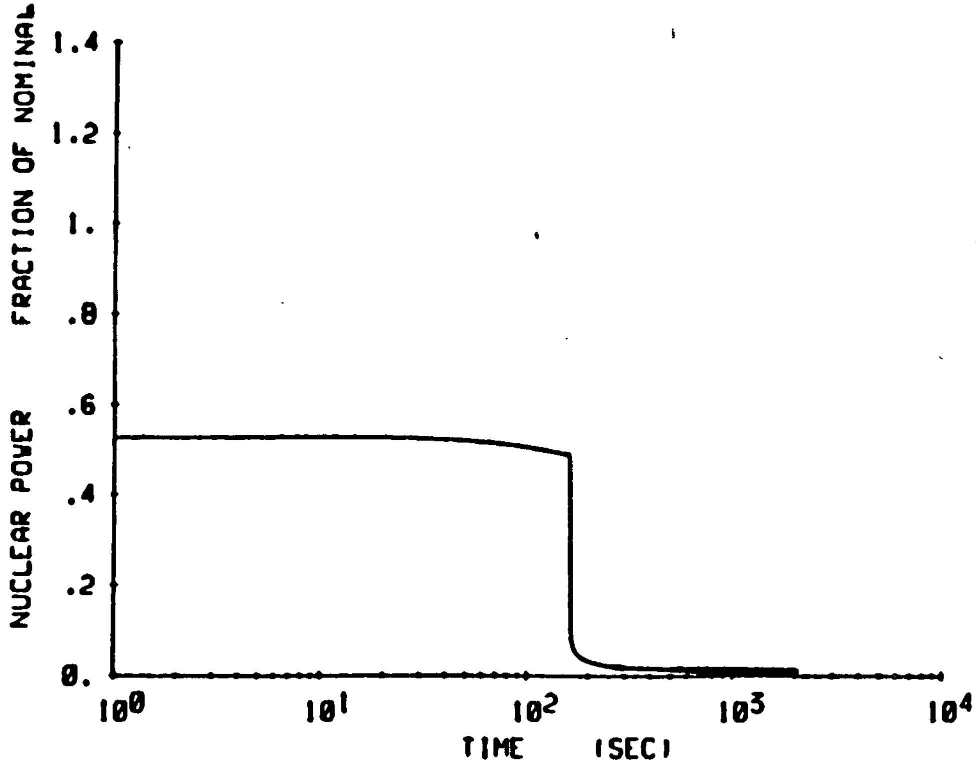


Figure 28

CASE E

SCE FLB Upstream Check Valve at 50% Power

AFW Flow of 125 gpm @ 15 minutes

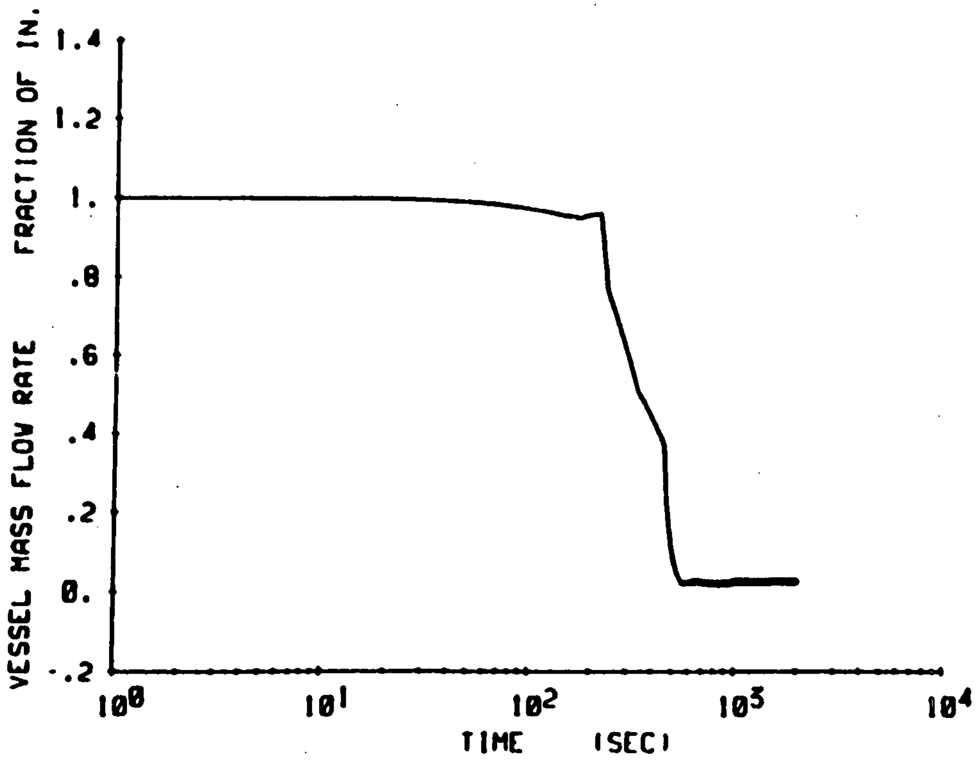


Figure 29

CASE E

SCE FLB Upstream Check Valve at 50% Power

AFW Flow of 125 gpm @ 15 minutes

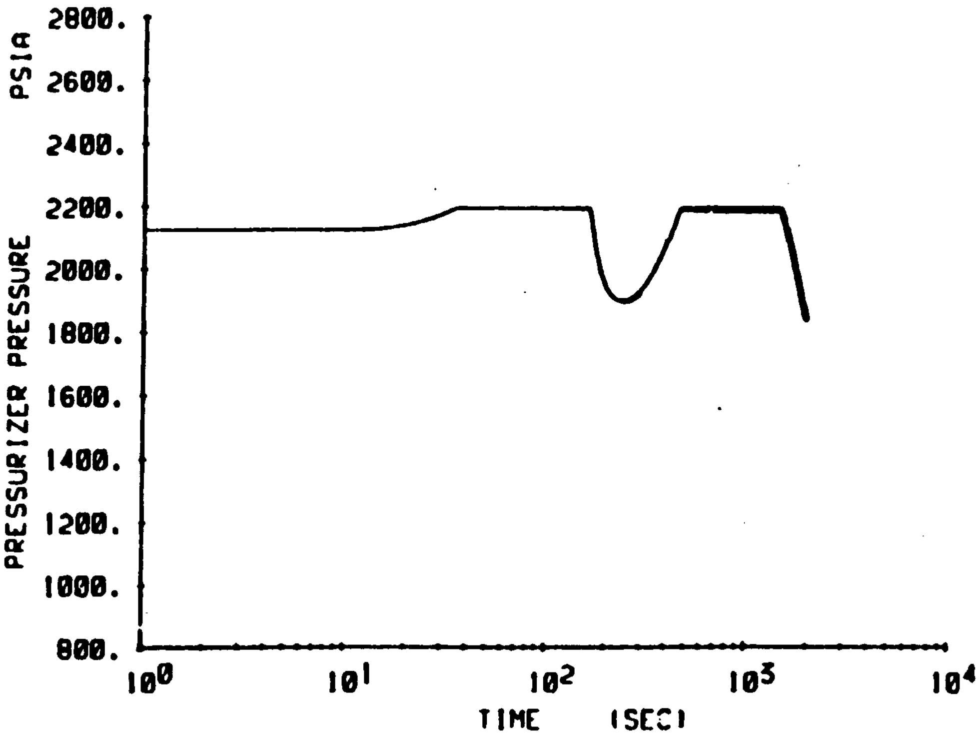


Figure 30

CASE E

SCE FLB Upstream Check Valve at 50% Power
AFW Flow of 125 gpm @ 15 minutes

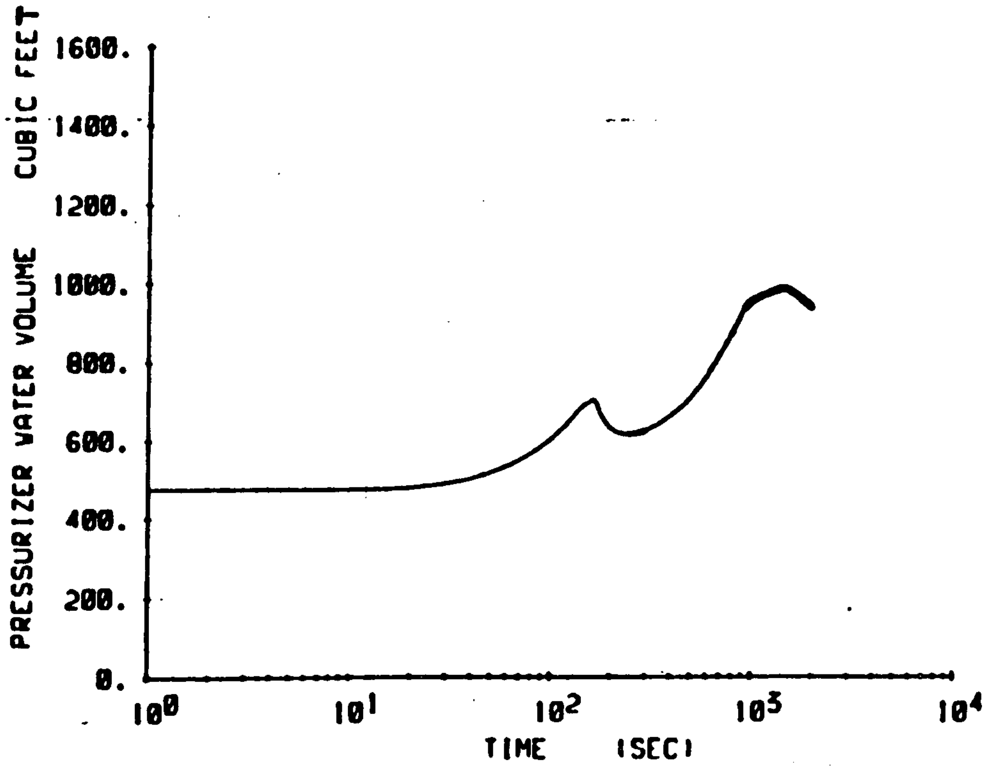


Figure 31

CASE E

SCE FLB Upstream Check Valve at 50% Power
AFW Flow of 125 gpm @ 15 minutes

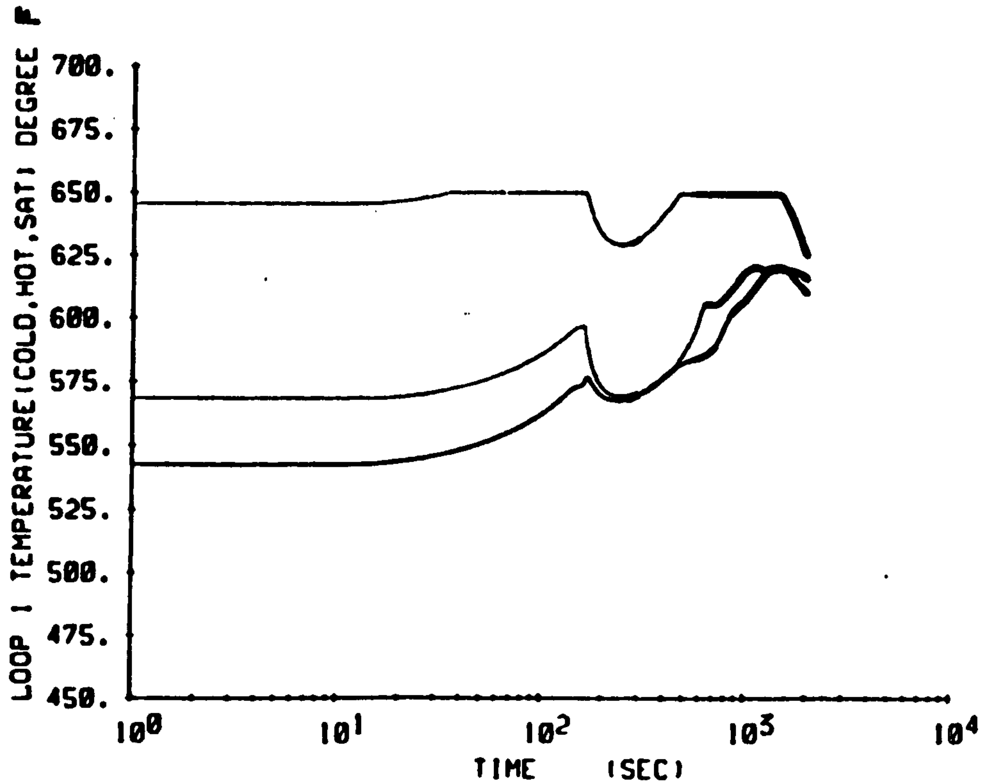


Figure 32

CASE E

SCE FLB Upstream Check Valve at 50% Power
AFW Flow of 125 gpm @ 15 minutes

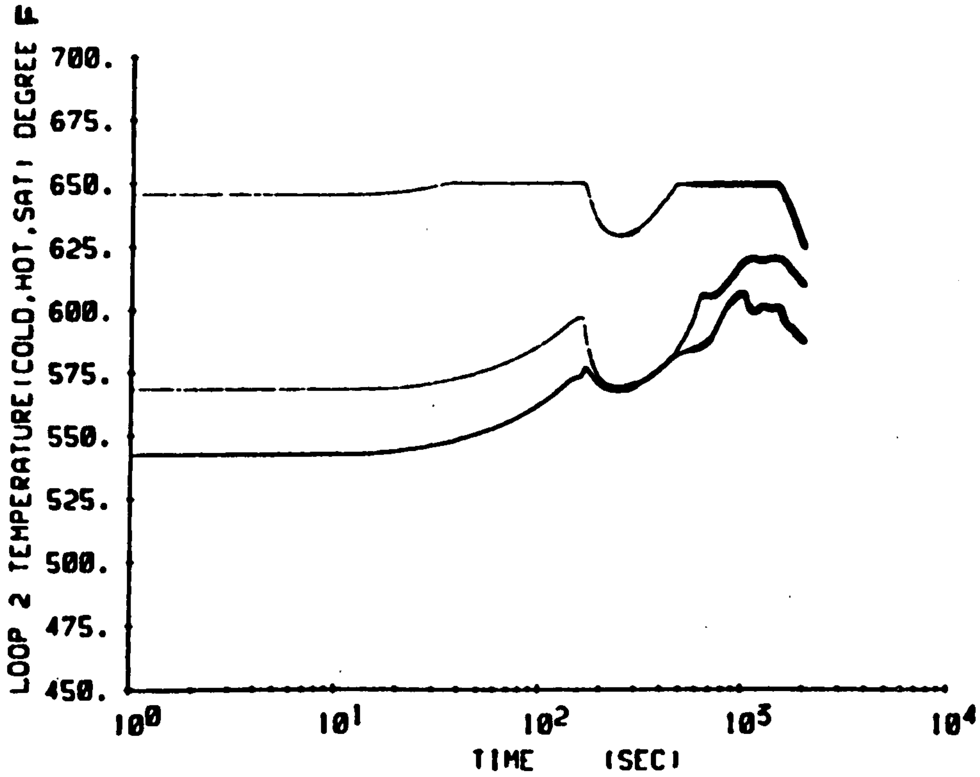


Figure 33

CASE E

SCE FLB Upstream Check Valve at 50% Power

AFW Flow of 125 gpm @ 15 minutes

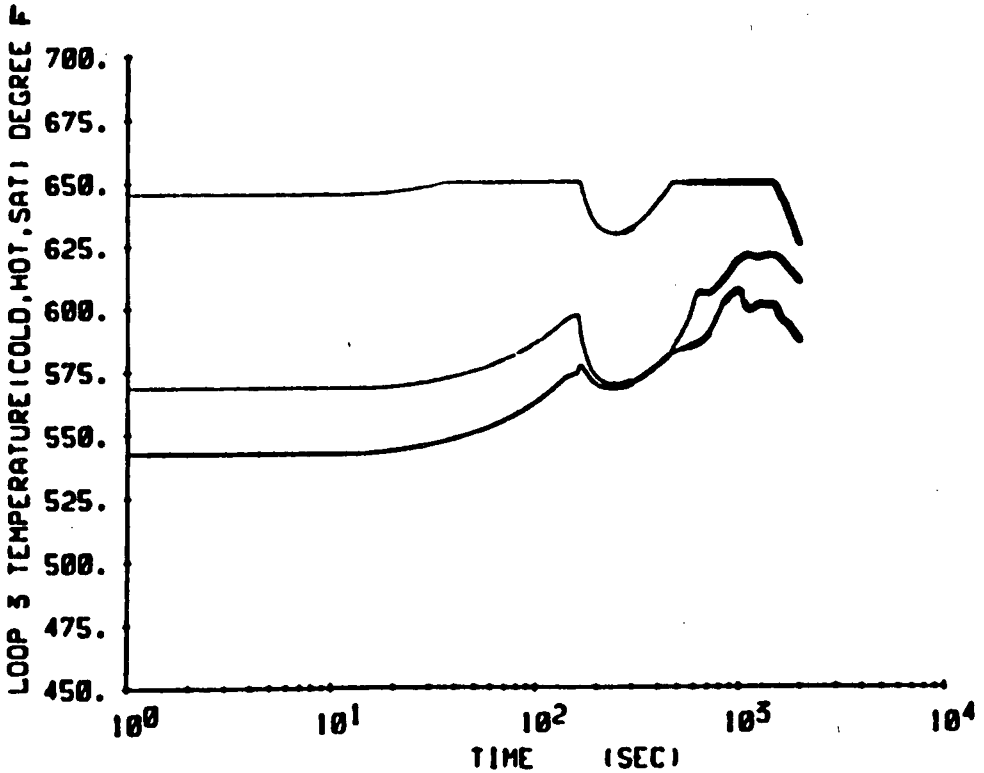


Figure 34

CASE E

SCE FLB Upstream Check Valve at 50% Power
AFW Flow of 125 gpm @ 15 minutes

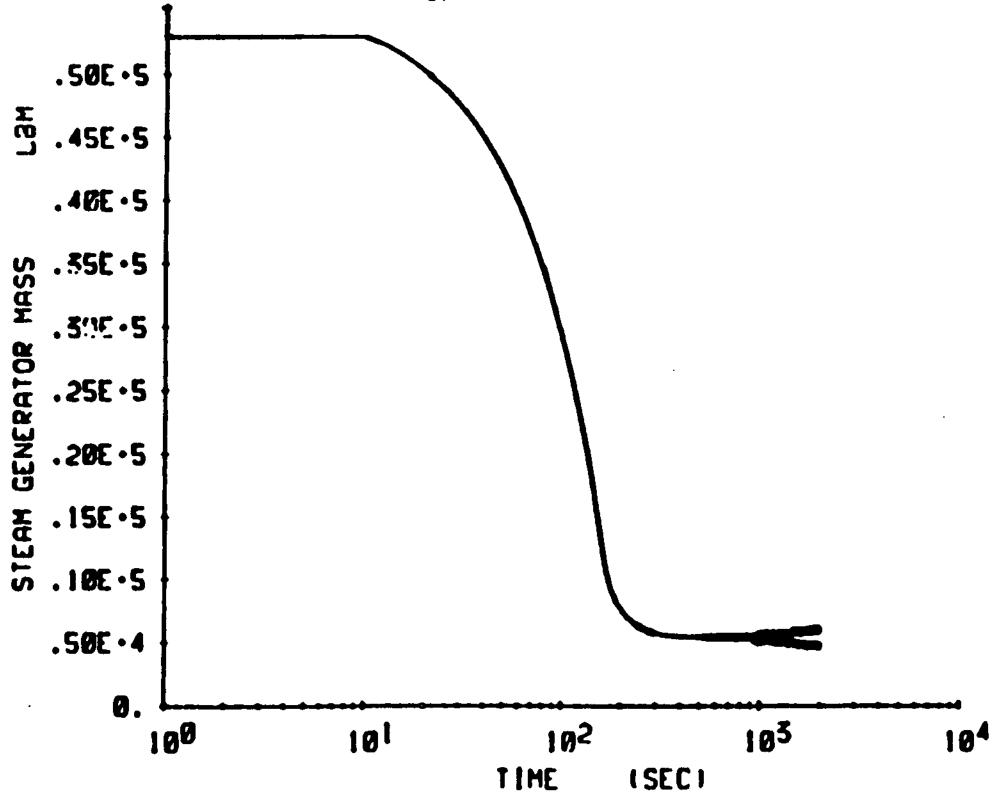


Figure 35

CASE E

SCE FLB Upstream Check Valve at 50% Power
AFW Flow of 125 gpm @ 15 minutes

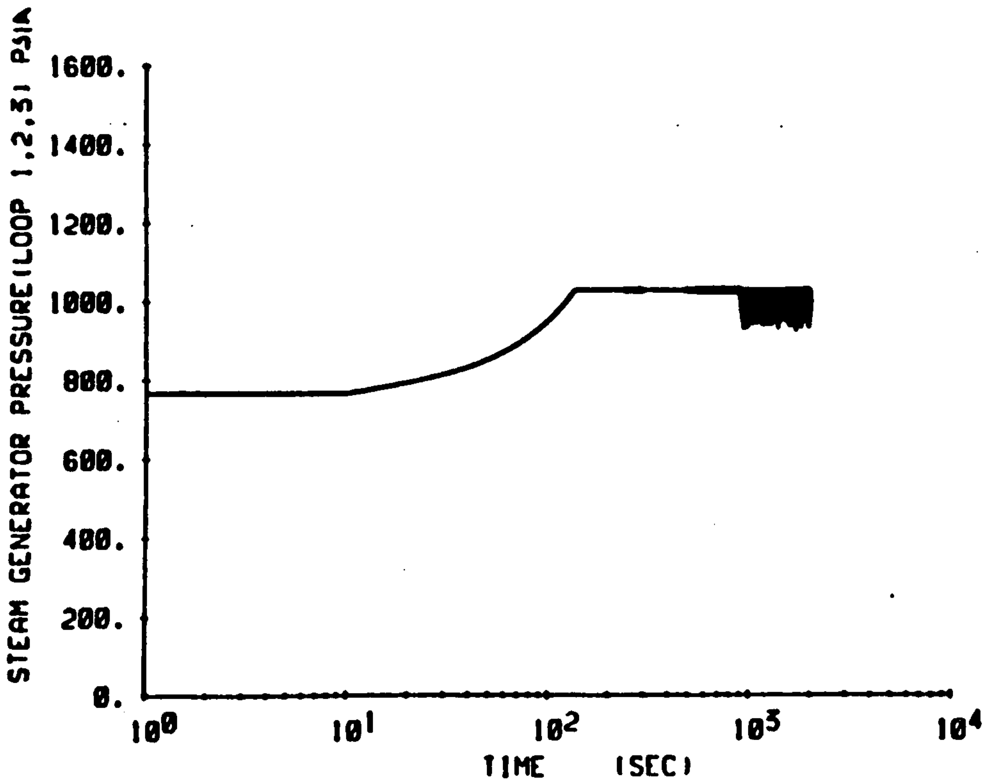


Figure 36

CASE G

SCE FLB DOWNSTREAM CHECK VALVE AT 50% POWER

AFW Flow of 250 gpm @ 15 minutes

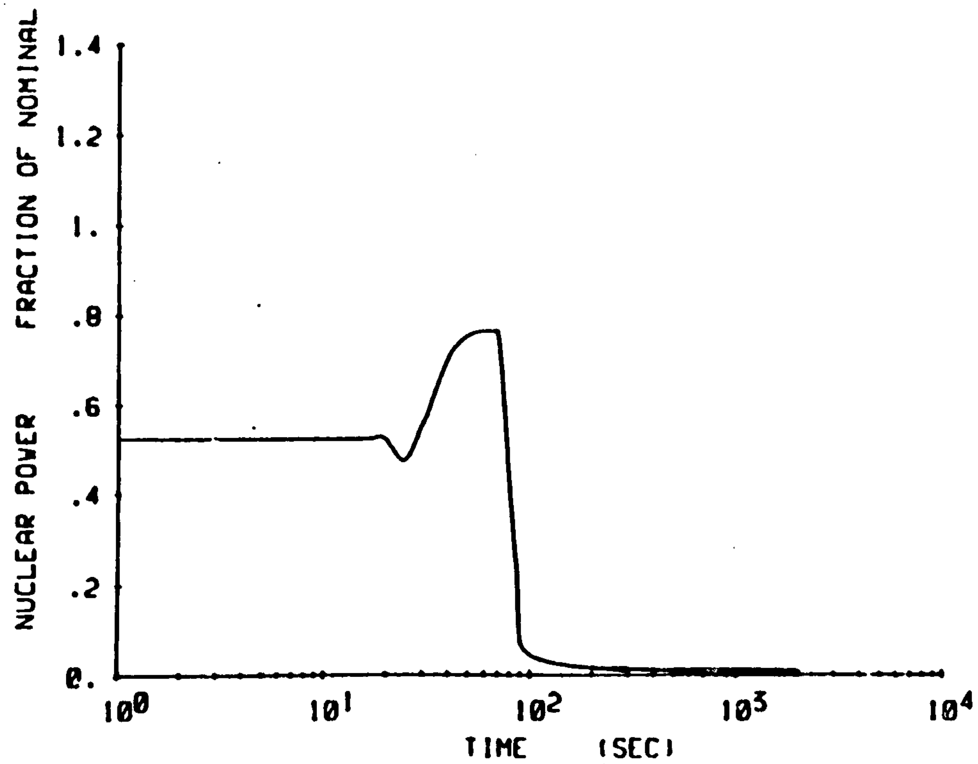


FIGURE 37

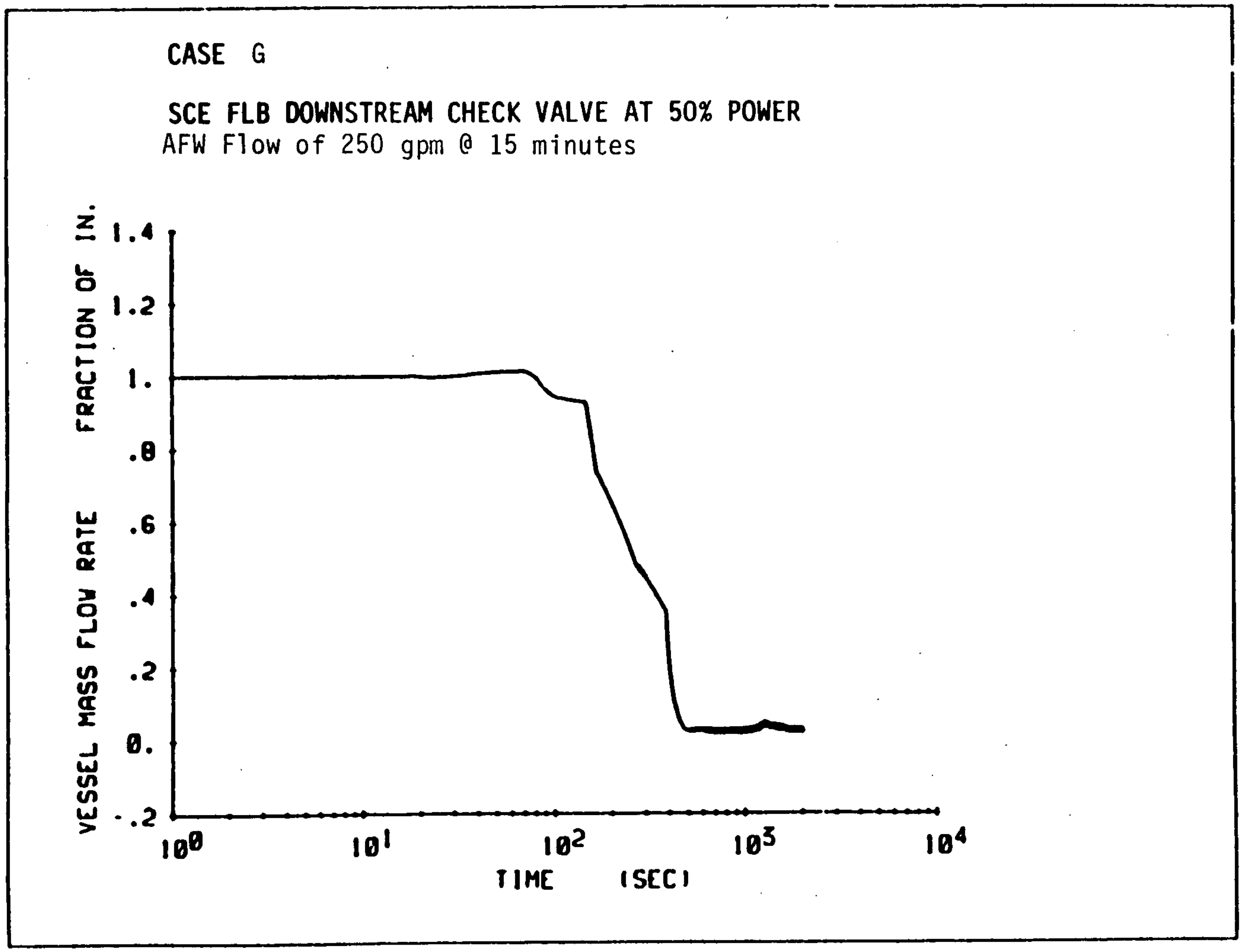

CASE G
SCE FLB DOWNSTREAM CHECK VALVE AT 50% POWER
AFW Flow of 250 gpm @ 15 minutes
VESSEL MASS FLOW RATE FRACTION OF IN.
1.4
1.2
1.
.8
.6
.4
.2
0.
-.2
10^0
10^1
10^2
10^3
10^4
TIME (SEC)

FIGURE 38

CASE G

SCE FLB DOWNSTREAM CHECK VALVE AT 50% POWER
AFW Flow of 250 gpm @ 15 minutes

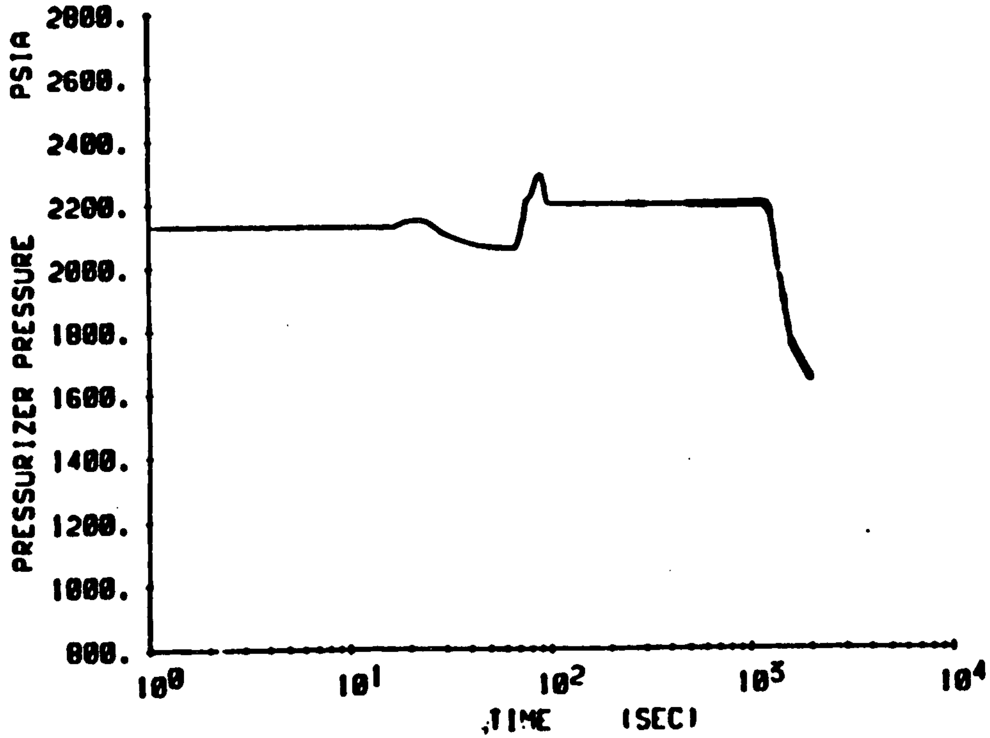


FIGURE 39

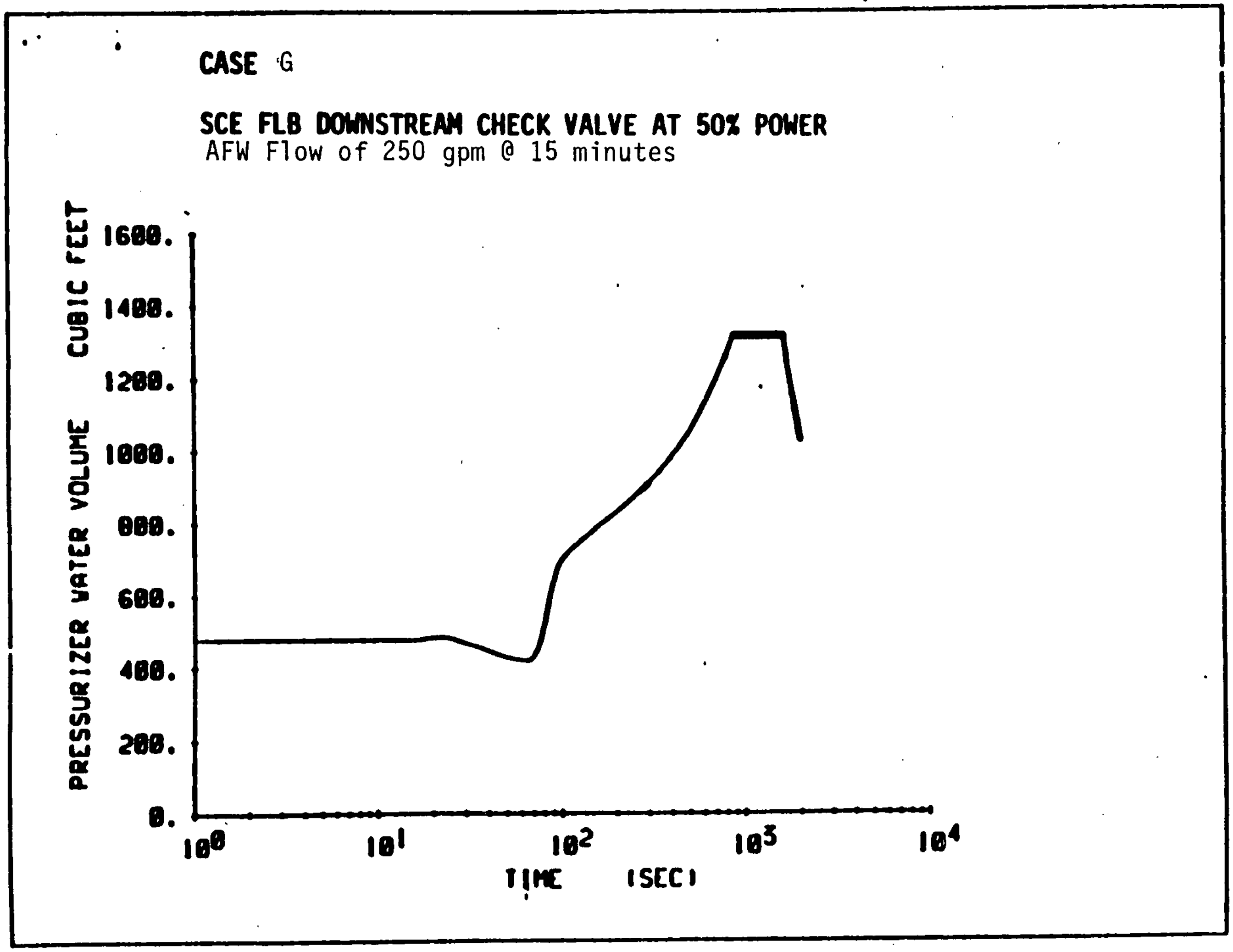

CASE G
SCE FLB DOWNSTREAM CHECK VALVE AT 50% POWER
AFW Flow of 250 gpm @ 15 minutes
PRESSURIZER WATER VOLUME CUBIC FEET
1600.
1400.
1200.
1000.
800.
600.
400.
200.
0.
10^0
10^1
10^2
10^3
10^4
TIME (SEC)

FIGURE 40

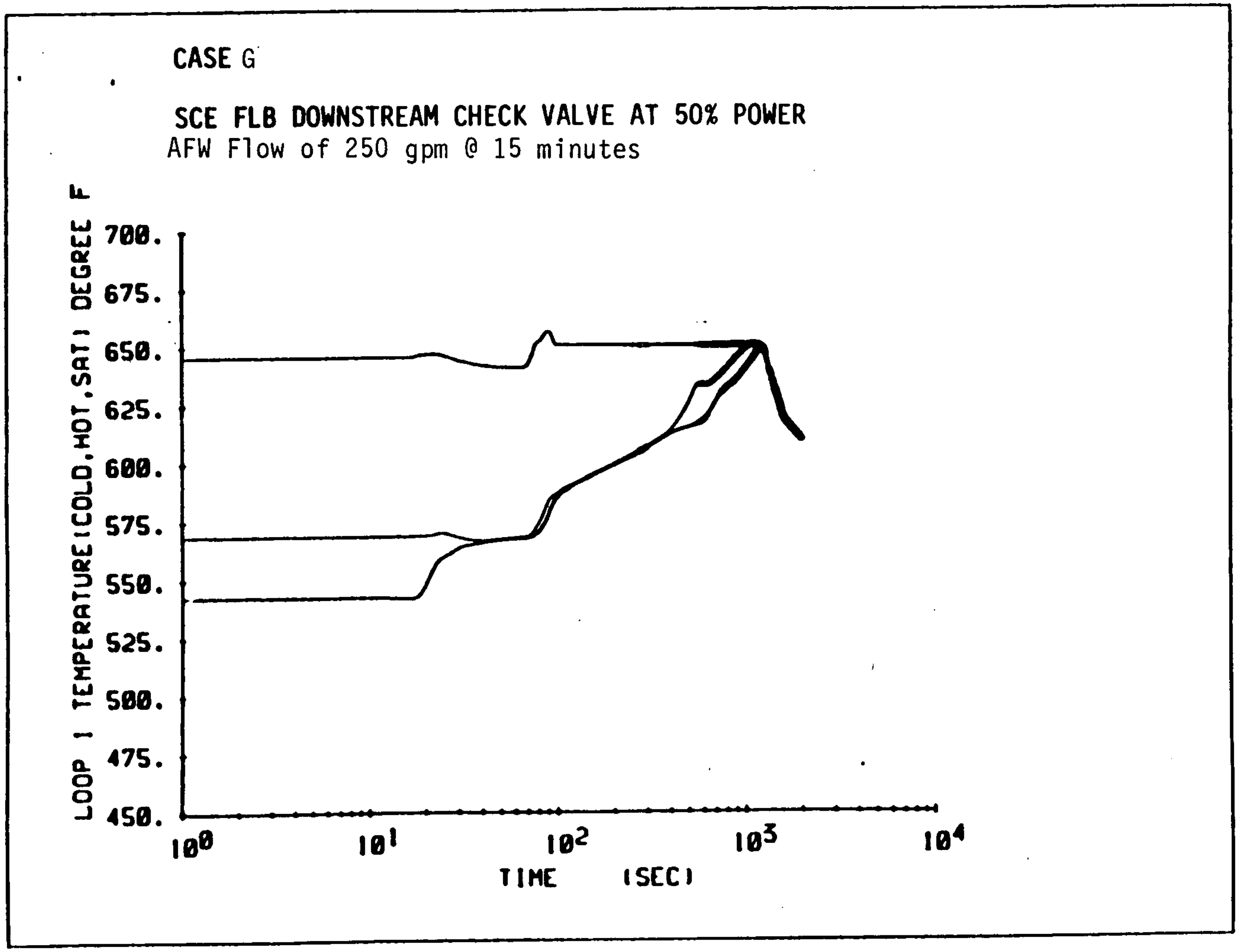

CASE G
SCE FLB DOWNSTREAM CHECK VALVE AT 50% POWER
AFW Flow of 250 gpm @ 15 minutes
LOOP 1 TEMPERATURE(COLD,HOT,SAT) DEGREE F
700.
675.
650.
625.
600.
575.
550.
525.
500.
475.
450.
10^0
10^1
10^2
10^3
10^4
TIME (SEC)

FIGURE 41

CASE G

SCE FLB DOWNSTREAM CHECK VALVE AT 50% POWER
AFW Flow of 250 gpm @ 15 minutes

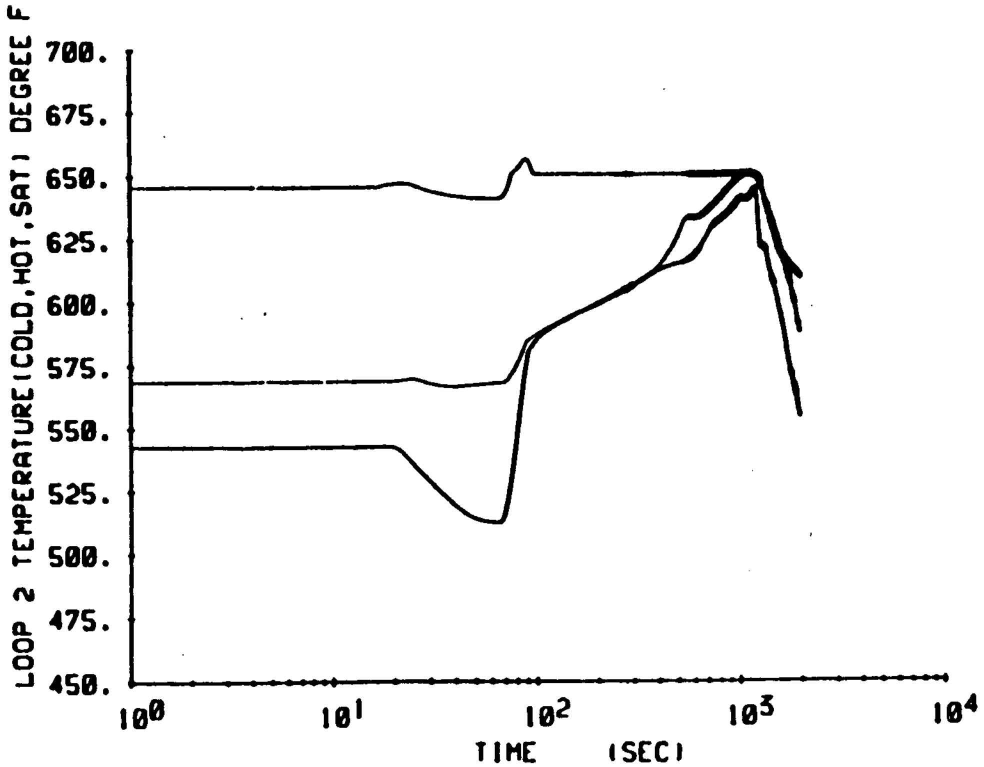


FIGURE 42

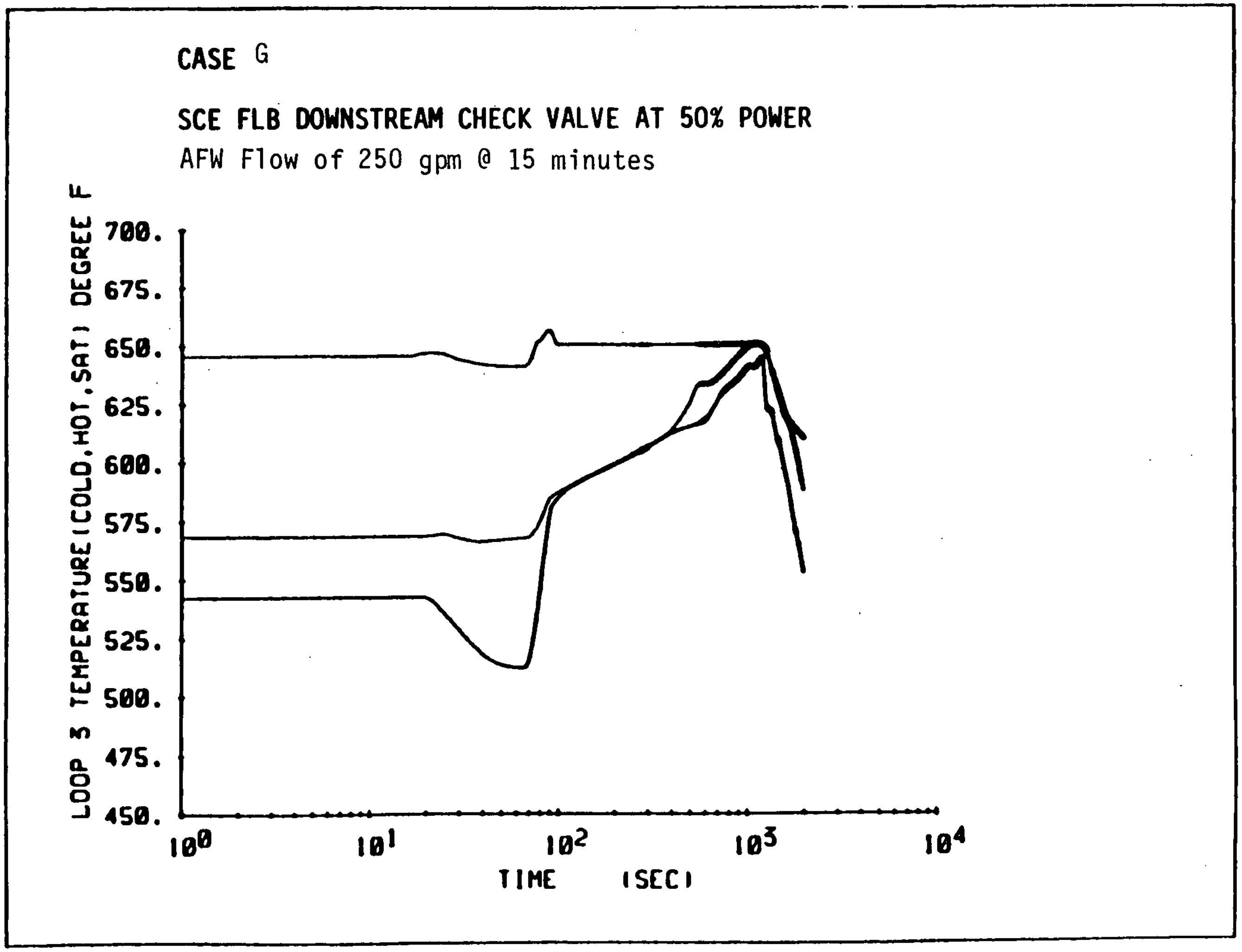

CASE G
SCE FLB DOWNSTREAM CHECK VALVE AT 50% POWER
AFW Flow of 250 gpm @ 15 minutes
LOOP 3 TEMPERATURE (COLD, HOT, SAT) DEGREE F
700.
675.
650.
625.
600.
575.
550.
525.
500.
475.
450.
10^0
10^1
10^2
10^3
10^4
TIME (SEC)

FIGURE 43

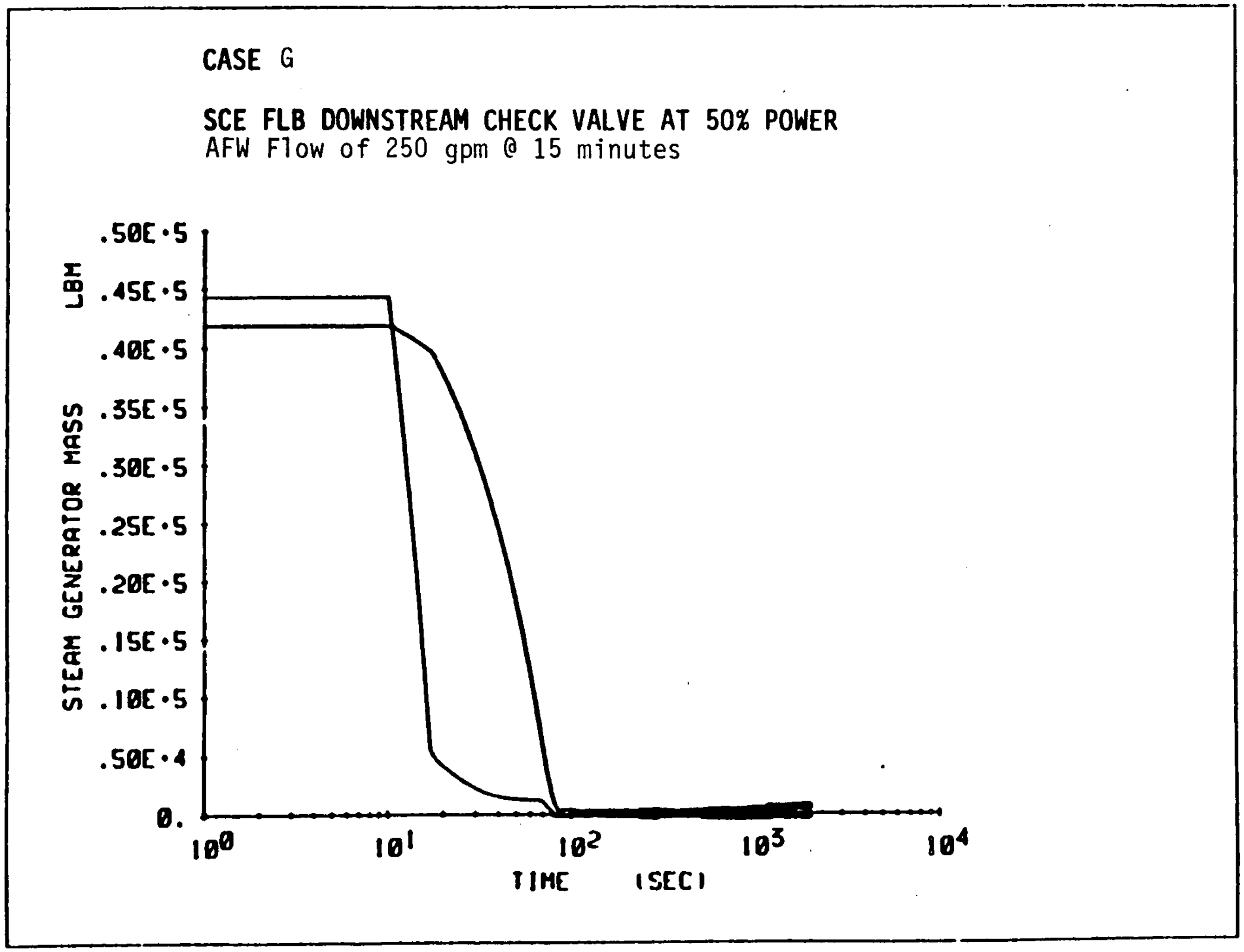

CASE G
SCE FLB DOWNSTREAM CHECK VALVE AT 50% POWER
AFW Flow of 250 gpm @ 15 minutes
STEAM GENERATOR MASS LBM
.50E+5
.45E+5
.40E+5
.35E+5
.30E+5
.25E+5
.20E+5
.15E+5
.10E+5
.50E+4
0.
10^0
10^1
10^2
10^3
10^4
TIME (SEC)

FIGURE 44

CASE G

SCE FLB DOWNSTREAM CHECK VALVE AT 50% POWER
AFW Flow of 250 @ 15 minutes

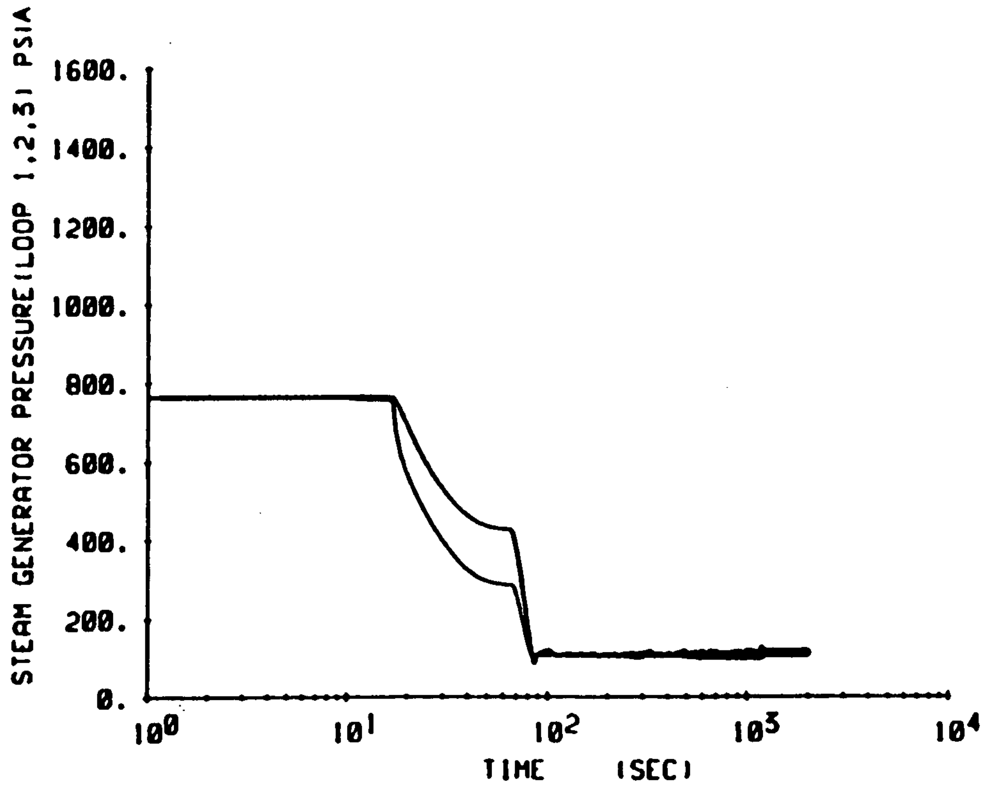


FIGURE 45

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