CEN-201 (S)

NATURAL CIRCULATION TEST PROGRAM

SAN ONOFRE NUCLEAR GENERATING STATION UNIT 2

SAFETY EVALUATION

Prepared for
Southern California Edison Company

NUCLEAR POWER SYSTEMS DIVISION
APRIL, 1982

POWER SYSTEMS
COMBUSTION ENGINEERING, INC.

LEGAL NOTICE

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CONTENTS

1.0 INTRODUCTION AND SUMMARY

Consistent with the requirements of Item I.G.1 of NUREG 0737, "Special Low-Power Testing and Training," a special natural circulation test program has been prepared for San Onofre Nuclear Generating Station, Unit 2. The basic objective of the program is to provide meaningful technical information beyond that obtained in the normal startup test program of Regulatory Guide 1.68 and to provide supplemental operator training. The test program was prepared in accordance with the requirements of 10 CFR 50.59. In addition to the Introduction and Summary Section, the evaluation contains five other sections as follows: Description of Tests and Pretest Simulations, Safety Analysis, Operational and Test Termination Criteria, Impact on Technical Specifications, and References.

In order to perform the various tests of the natural circulation program, certain reactor trips and certain automatic safety functions must be bypassed or altered. In addition, relief from several of the technical specifications contained in Appendix A of San Onofre Unit 2 Operating License Number NPF-10 will be required. A detailed review of the proposed tests has been performed. This review has determined that the tests are acceptable and that they can be performed with a minimum of risk provided that close operator surveillance of plant parameters is maintained and that prompt operator action is taken, as outlined in the specific test procedures, to terminate testing as required by the test termination criteria discussed in Section 4.0 of this report.

Since the natural circulation test program involves testing which impacts the San Onofre Unit 2 technical specifications and since portions of the proposed tests were not previously evaluated in the Chapter 15 safety analysis of Reference 1, a detailed review of the program, including the specific test procedures, was performed in order to determine their effects on the design basis events. The

results of the review, which demonstrate full compliance with the requirements of 10 CFR 50.59, are presented in Section 3.0 of this report. In most cases, it was possible to demonstrate through qualitative analysis that the FSAR discussion bounded the consequences of the design basis event occuring under conditions of natural circulation. For certain cases, quantitative analysis was performed in order to demonstrate that the proposed tests did not involve an unreviewed safety question. These analyses indicated that the automatic protective functions available during the tests in conjunction with the operator actions specified by the detailed test procedures were adequate to prevent violation of the acceptance criteria of the Standard Review Plan (Reference 2), should a design basis event occur during natural circulation. A complete list of the specific operator actions, i.e., the mandatory test termination criteria and the operational criteria which the operators should try to maintain for each of the tests, is presented in Section 4.0 of this report. These operational safety criteria were developed based upon the safety analysis discussed in Section 3.0. Additionally, the results of the safety analysis conducted for the natural circulation test program also identified certain technical specifications for which temporary relief is required in order to facilitate the performance of the tests. The specific technical specifications requiring relief are listed in Section 5.0.

The conclusion, the results of the safety evaluation performed for the natural circulation test program in conjunction with the operational and test termination criteria, as discussed in this report, demonstrate that the proposed tests do not involve an unreviewed safety question as defined in 10 CFR 50.59. Specifically, it was determined that the proposed program (i) does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Chapter 15 safety analysis, and (ii) does not create an accident of malfunction of a different type than evaluated previously in the safety analysis report, and (iii) does not reduce the margin of safety as defined in the basis for the technical specifications.

2.0 DESCRIPTION OF TESTS AND PRETEST SIMULATIONS

2.1 Basic Description of Tests

2.1.1 Natural Circulation Demonstration Program - Verification of Natural Circulation (Test A1)

Objective - To establish natural circulation flow conditions.

Method - The reactor is critical at approximately 3% power with all four reactor coolant pumps operating. Reactor protective system trips are bypassed, as required, to allow the reactor to be maintained critical with forced circulation secured. At time zero, all four reactor coolant pumps are simultaneously tripped and natural circulation is established.

2.1.2 Natural Circulation Demonstration Program - Verification of Natural Circulation at Reduced Pressures (Test A2)

Objective - To demonstrate the capability to maintain natural circulation and adequate margin to saturation without the use of pressurizer heaters. To determine the reactor coolant system depressurization rate following the loss of all four reactor coolant pumps and pressurizer heaters. To demonstrate the ability to maintain natural circulation at reduced reactor coolant system pressures. To demonstrate the ability to control the subcooled margin through the use of the chemical and volume control system and the steam bypass system.

Method - Natural circulation conditions are established in accordance with Test A1 above. All pressurizer heaters are secured allowing a slow cooling of the pressurizer to begin with a corresponding decrease in the reactor coolant system pressure.

2.1.3 Natural Circulation Demonstration Program - Verification of Natural Circulation with Reduced Heat Removal Capacity (Test A3)

Objective - To demonstrate the capability to maintain natural circulation with one steam generator isolated. To demonstrate that full natural circulation flow can be reestablished when the isolated steam generator is returned to service.

Method - Natural circulation conditions are established in accordance with Test A1 above with indicated reactor power approximately 1%. A controlled cooldown is performed to lower the pressure in both steam generators to approximately 800 psia. Steam Generator 2 is isolated by closing its main steam isolation valve and securing feedwater flow. When the operator is ready, Steam Generator 2 is returned to service by opening its main steam isolation valve and restoring feedwater. A demonstration is then made that full natural circulation flow can be reestablished.

2.1.4 Loss of Offsite Power and Simulated Loss of Both Offsite and Onsite Power (Tests B1 & B2) (Tests B1 and B2 of Table I.G.1-1 of Reference 1 were combined into one test.)

Objective - To demonstrate the ability to shutdown and maintain the reactor in hot standby using only emergency power following a loss of offsite power. To demonstrate system operation under conditions that simulate a total loss of AC power.

Method - The reactor is critical at 20 ± 1% indicated power with all four reactor coolant pumps operating. Previous power history is such that adequate decay heat is available for natural circulation. All plant safety related and auxiliary loads are powered from their normal sources. At time zero, the reactor is tripped and offsite power is simultaneously interrupted. Emergency diesel operation is demonstrated, natural circulation is established, and plant conditions

are stabilized. Selected safety related loads are secured to simulate a loss of both offsite and onsite AC power. (Total loss of AC power is simulated to an extent that will not risk plant damage.)

2.1.5 80% Total Loss of Flow/Natural Circulation (Test B3)

Objective - To measure plant response to a total loss of forced reactor coolant flow from 80% power. To demonstrate natural circulation following the reactor trip from 80% power. To determine that adequate boron mixing can be achieved under natural circulation conditions. To demonstrate the ability to perform a natural circulation cooldown to shutdown cooling initiation conditions. To demonstrate that automatic control systems operate satisfactorily under steady-state and transient conditions.

Method - The reactor is critical at 80 $\pm$ 0.5% indicated power with all four reactor coolant pumps operating. Previous power history is such that adequate decay heat is available for natural circulation. All plant safety related and auxiliary loads are powered from external sources. At time zero, all four reactor coolant pumps are simultaneously tripped followed by an automatic reactor trip. Natural circulation is established and conditions are stabilized. Boron concentration in the primary is increased and proper boron mixing is demonstrated. A natural circulation cooldown is performed and plant pressure is lowered until shutdown cooling entry conditions are attained.

2.2 Pretest Simulations

Because of the unusual conditions under which the special natural circulation tests will be performed, a simulation of plant responses was made using the Long Term Cooling (LTC) digital computer code. This code has been successfully used to calculate NSSS natural circulation transients and was used to calculate the results published

in Reference 3. For each of the simulations performed by the LTC code, certain assumptions concerning plant behavior were made. These assumptions are listed below along with the corresponding test results. Except where noted, all assumptions were chosen to model actual plant conditions as accurately as possible.

2.2.1 Natural Circulation Verification (Test A1)

Figures 2-1 through 2-36 (pp. 15 to 50) contain the results produced by the LTC digital computer code for Test A1. Figures 2-1 through 2-12 show the results for a power level of 1%, Figures 2-13 through 2-24 show the results for a power level of 3%, and Figures 2-25 through 2-36 show the results for a power level of 5%. Steady-state results for other power levels were obtained by performing hand calculations. Table 2-1 (p. 14) contains numerical results for Test A1 at various power levels. The following assumptions were used for the computer simulations:

1. Reactor power is held constant at 1, 3, or 5 percent for the entire transient. Input from decay heat negligible.

2. Natural circulation is initiated at time 100 seconds by simultaneously tripping all four reactor coolant pumps.

3. Pressure in both steam generators is held constant at 1000 psia by use of the steam bypass control system. This corresponds to a natural circulation T_c of 545°F.

4. Steam generator water levels are maintained at approximately 69% via manual control of the auxiliary feedwater system.

5. Pressurizer pressure and level controls are maintained in automatic.

6. Auxiliary pressurizer spray is not used.

7. Heat losses from the reactor coolant system to the containment are negligible.

8. For the simulation at 5% power only, the reactor trip on high pressurizer pressure and the action of the pressurizer safety relief valves were eliminated. This was done for continuity in order to predict the maximum pressurizer pressure that could be seen and does not represent actual test conditions.

Following reactor coolant pump trip, flow through the core decreases smoothly until rotor lockup at about 400 seconds, see Figures 2-8, 2-20, and 2-32. Loop resistance increases sharply at rotor lockup causing core flow to decrease suddenly until it eventually stabilizes at about 800 seconds. Final steady-state flow rates are dependent on power level as shown in Table 2-1 and correspond to loop cycle times of five (5) to twelve (12) minutes. T_h increases gradually after pump trip, see Figures 2-5, 2-6, 2-17, 2-18, 2-29, and 2-30, and stabilizes at a maximum value at approximately 800 seconds. The maximum T_h is dependent upon power level as shown in Table 2-1. Pressurizer level, initially at 33%, will increase automatically as T_{avg} increases. The maximum pressurizer pressure during each of the transients occurs at about 600 seconds. For an initial power level of 3%, a maximum value of 2354 psia is reached assuming no auxiliary spray. Maintaining a constant steam generator pressure will result in a constant cold leg temperature as shown in Figures 2-5, 2-17, and 2-29. During the actual tests, however, the operator is permitted to vary T_c, if desired to provide increased operational flexibility, within the operational criteria of Section 4.0 of this report.

LTC code results indicate that although power levels as high as 5% can be tolerated for short transients, continuous operation at power levels greater than 4% could result in exceeding certain of the operational criteria of Section 4.0 of this report. Specifically, the maximum core exit temperature of 600°F could be exceeded at continuous

power levels of greater than 4%. (This limit is based upon maintaining a sufficient margin to critical heat flux as discussed in 3.1.3.2 below.) In addition, at a power level of approximately 5.0%, a power-to-flow ratio of 1.0 will be exceeded. (See Table 2-1 on page 14.) To ensure that the plant is operated safely at all times, close operator surveillance of plant parameters and strict compliance with the operational and test termination criteria of Section 4.0 must be maintained.

2.2.2 Verification of Natural Circulation at Reduced Pressures (Test A2)

The results of the LTC computer simulation for Test A2 are shown in Figures 2-37 through 2-45 (pp. 51 to 59). The simulation was initiated from 3% power with all four reactor coolant pumps operating and an initial system pressure of 1700 psia. These conditions differ slightly from the conditions of the actual test in that initial system pressure will be 2250 psia and a slow pressure reduction will be conducted after natural circulation has been initiated. This slight deviation from the true test conditions was chosen since it resulted in a more significant plant transient than will actually take place and thus produced a more conservative set of simulation results. Other assumptions which were made concerning plant behavior are as follows:

1. Reactor power is held constant at 3%. Input from decay heat is negligible.

2. Natural circulation is initiated at time 100 seconds by simultaneously tripping all four reactor coolant pumps.

3. Pressure in both steam generators is held constant at 1000 psia by use of the steam bypass control system.

4. Steam generator water levels are maintained at approximately 69% via manual control of the auxiliary feedwater system.

5. Pressurizer level control is in automatic.

6. Pressurizer pressure is controlled in automatic at 1700 psia.

As can be seen from Figures 2-37 through 2-45, no significant deviations from the final steady-state parameters obtained for Test A1 were noted during the simulation of Test A2. Although pressurizer pressure was controlled at 1700 psia for the simulation, reactor coolant system pressure during the performance of the actual test should not be allowed to drop below 1750 psia. Maintaining system pressure at or above 1750 psia during the actual test will ensure that an adequate subcooled margin exists at all times in order to prevent void formation in the upper head region.

2.2.3 Natural Circulation with Reduced Heat Removal Capacity (Test A3)

The results of the LTC computer simulation for Test A3 are shown in Figures 2-46 through 2-55 (pp. 60 to 69). During the performance of the actual test, a natural circulation cooldown will be performed to lower pressure in both steam generators to approximately 800 psia before isolating Steam Generator 2. For the computer run, however, the LTC code was initiated at 800 psia since no additional information would be gained by simulating a plant cooldown. Other assumptions which were made concerning plant behavior are as follows:

1. Reactor power is held constant at 1%. Input from decay heat is negligible.

2. Transient is initiated at time 1000 seconds by isolating Steam Generator 2.

3. Pressure in Steam Generator 1 is maintained constant at 800 psia by use of the steam bypass system.

4. Water level in Steam Generator 1 is maintained at approximately 69% via manual control of the auxiliary feedwater system.

5. Pressurizer pressure and level controls are in automatic.

6. Heat losses from the RCS to the containment are negligible.

The initial conditions for the simulation and the actual performance of Test A3 were chosen such that the pressure in the isolated steam generator would not rapidly approach secondary safety valve setpoints. The LTC code results show that a slow plant transient will take place following the isolation of Steam Generator 2. Assuming a constant power level, T_h in both loops will increase gradually due to the reduced heat removal capacity and a small reduction in core flow, see Figures 2-50, 2-51, and 2-53. Loop B ΔT decreases continuously, reaching 9°F at about 3000 seconds, and eventually will approach zero as the secondary temperature in Steam Generator 2 approaches T_h. An increase in letdown flow, see Figure 2-55, will take place following steam generator isolation as the pressurizer level instrument automatically compensates for thermal expansion due to primary system temperature increases. Pressure in Steam Generator 2 increases gradually, see Figure 2-48, reaching 988 psia at time 3000 seconds, and will continue to increase slowly until eventually the relief valve setpoint could be approached. As long as actual reactor power is approximately 1%, however, the high pressure situation in Steam Generator 2 will not occur before two hours into the transient. Sufficient time is thus available for appropriate operator action, e.g., operation of atmospheric dump valves and/or lowering pressure in the operating steam generator, to prevent lifting the secondary safety valves on Steam Generator 2.

2.2.4 Loss of Offsite Power and Simulated Loss of Both Offsite and Onsite Power (Tests B1 & B2)

The results produced by the LTC program for Tests B1 and B2 are contained in Figures 2-56 through 2-71 (pp. 70 to 85). The simulation was performed for the initial part of the tests only, i.e., the loss of all AC portion was not simulated. The following assumptions were used for the computer runs:

1. Reactor coolant system is initially stable at 20% power. Adequate decay heat is available for natural circulation following the reactor trip.

2. Initial steam generator pressure is approximately 950 psia.

3. At time zero, the reactor is tripped and offsite power is simultaneously interrupted.

4. A turbine generator trip occurs 0.5 seconds after the reactor trip.

5. Following the turbine trip, steam is dumped via the atmospheric dumps as necessary to maintain steam generator pressure less than or equal to 1000 psia.

6. Emergency feedwater flow at 800 gpm and 70°F is set to initiate automatically if generator level reaches 23%.

7. Pressurizer pressure and level controls are in automatic.

8. Decay heat values are obtained by multiplying the standard decay heat curve by the initial power level in percent. (See Figure 2-56.)

Following reactor coolant pump trip, flow through the core decreases smoothly until rotor lockup at about 250 seconds, see Figures 2-59 and 2-60. Loop resistance increases sharply at rotor lockup causing core flow to decrease suddenly until it eventually stablilzes at about 900 seconds. Steam generator levels decrease rapidly due to shrinkage. As shown in Figures 2-65 and 2-66, the minimum levels obtained are above the emergency feedwater actuation setpoints. In order to regain normal steam generator water levels in this situation, manual initiation of the auxiliary feedwater system would be required. This was not performed for the simulation; however, during the actual tests levels should be restored slowly in order to limit plant cooldown.

2.2.5 80% Total Loss of Flow/Natural Circulation (Test B3)

Figures 2-72 through 2-92 (pp. 86 to 106) show the results produced by th LTC program for Test B3. The simulation was performed for the first portion of the test only, i.e., boration and natural circulation cooldown were not simulated. The assumptions used for the computer runs are as follows:

1. Reactor coolant system is initially stable at 80% power. Adequate decay heat is available for natural circulation following the reactor trip.

2. Initial steam generator pressure is approximately 910 psia.

3. Transient is initiated at time zero by simultaneously tripping all four reactor coolant pumps. The reactor is tripped four seconds after pump trip.

4. A turbine trip occurs 0.5 seconds after the reactor trip.

5. Following the turbine trip, pressure in the steam generators is maintained less than or equal to 1000 psia by use of the steam bypass system.

6. Pressurizer pressure and level controls are in automatic.

7. Main feedwater flow is reduced to 5% per steam generator following the reactor trip.

8. Emergency feedwater flow at 800 gpm and 70°F is initiated when steam generator levels reach 23%.

9. When steam generator levels reach 69%, main feedwater flow is secured and levels are maintained via manual control of the auxiliary feedwater system.

10. Decay heat values are obtained by multiplying the standard decay heat curve by the initial power level in percent. (See Figure 2-72.)

Following reactor coolant pump trip, flow through the core decreases smoothly until rotor lockup at about 250 seconds, see Figures 2-75 and 2-76. Loop resistance increases sharply at rotor lockup causing core flow to decrease suddenly until it eventually stabilizes at about 1500 seconds. Steam generator water levels decrease rapidly, see Figure 2-81 and 2-82, due to shrinkage. An emergency feedwater actuation signal is obtained when levels reach 23%. Levels in both generators continue to fall to approximately 12% at which time they begin to recover. Main feedwater flow, see Figures 2-85 and 2-86, falls to 5% of normal full flow and emergency feedwater is throttled to approximately 400 gpm per steam generator, see Figures 2-88 and 2-89. As can be seen from Figures 2-77 and 2-78, limiting feedwater flow reduces the initial plant cooldown to acceptable values. Once normal water levels are restored, main feedwater flow in secured and levels are maintained via manual control of the auxiliary feedwater system. Due to expansion as the steam generators begin to heat up, levels will continue to increase as shown in Figures 2-81 and 2-82. Operator action may be required to further control feedwater flow in order to prevent a high level situation from occuring.

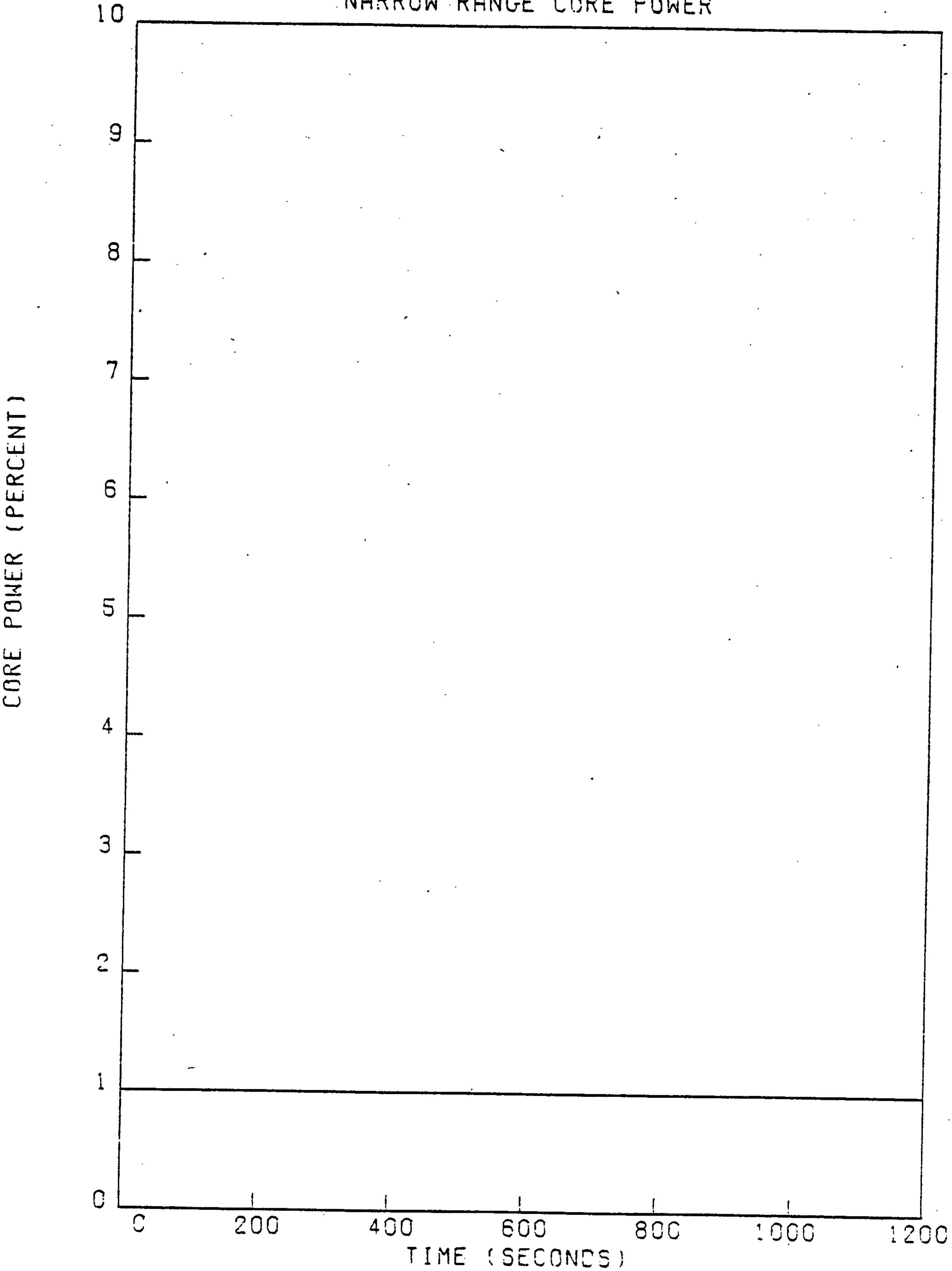
Table 2-1

Verification of Natural Circulation
Numerical Results(1)

Plant Parameter	Plant response vs power levels 1%	2%(2)	3%	4%(2)	5%
T_h(°F)	567	579	589	598	605
T_c(°F)	545	545	545	545	545
T_{avg}(°F)	556	562	567	572	575
Core flow (lbm/sec)	1167	1470	1702	1890	2033
Core flow (%)	2.80	3.53	4.09	4.53	4.88
Power-to-flow ratio	0.36	0.57	0.73	0.88	1.03
Pzr. pressure (psia)(3)	2291	2323	2354	2383	2411
Pzr. level (%)	37	41	45	48	52

1) Results given are final steady-state values, except where indicated.
2) Values for this power level were obtained from hand calculations.
3) Values given are the maximum pressurizer pressure obtained during the simulated test transients. Auxiliary spray flow for each is zero.

FIGURE 2-1
SONGS 1 PCT PWR NATURAL CIRCULATION
NARROW RANGE CORE POWER



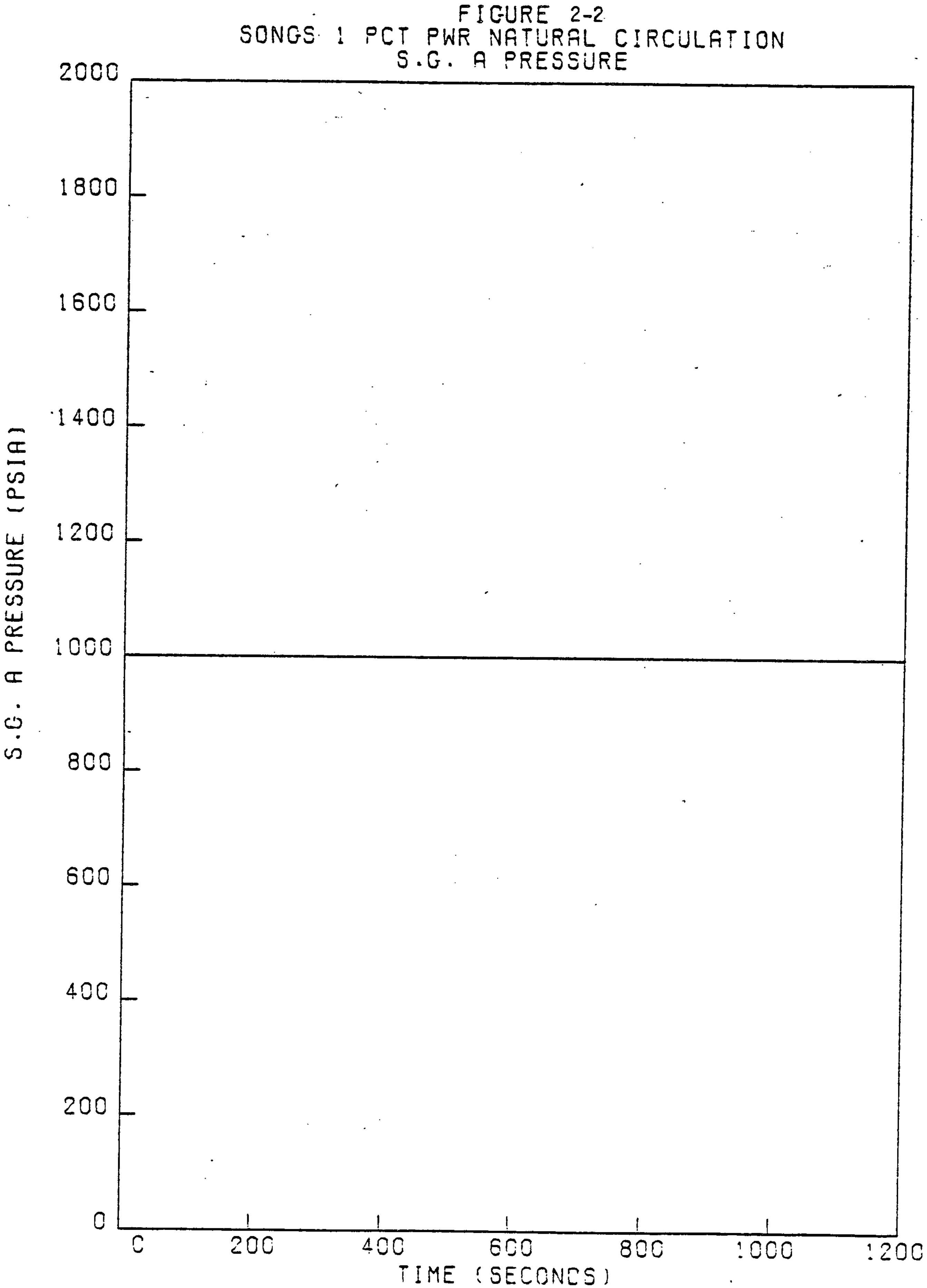


FIGURE 2-2
SONGS 1 PCT PWR NATURAL CIRCULATION
S.G. A PRESSURE
2000
1800
1600
1400
1200
1000
800
600
400
200
0
S.G. A PRESSURE (PSIA)
0
200
400
600
800
1000
1200
TIME (SECONDS)

FIGURE 2-3
SONGS 1 PCT PWR NATURAL CIRCULATION
S.G. B PRESSURE

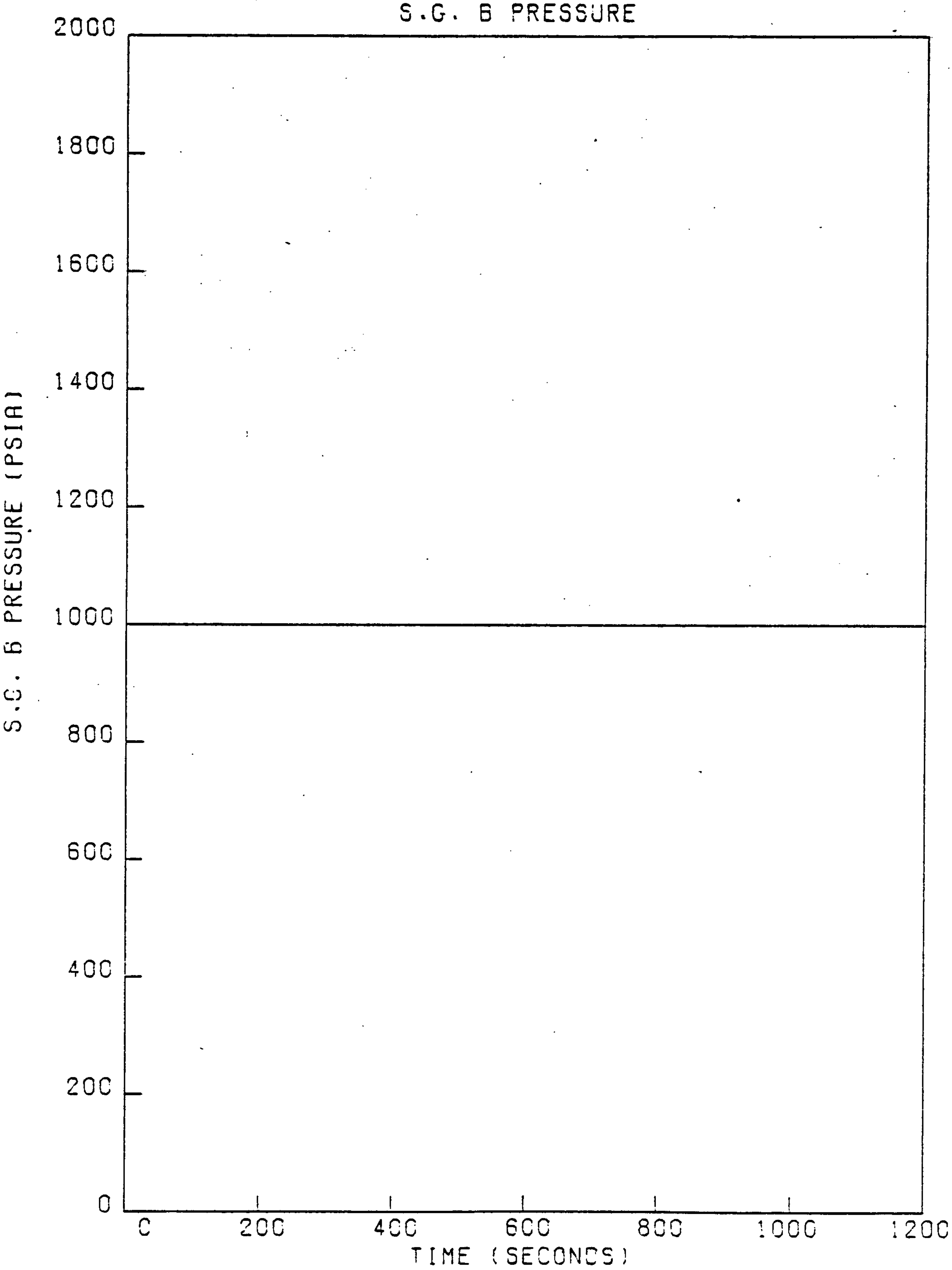


FIGURE 2-4
SONGS 1 PCT PWR NATURAL CIRCULATION
PZR LEVEL

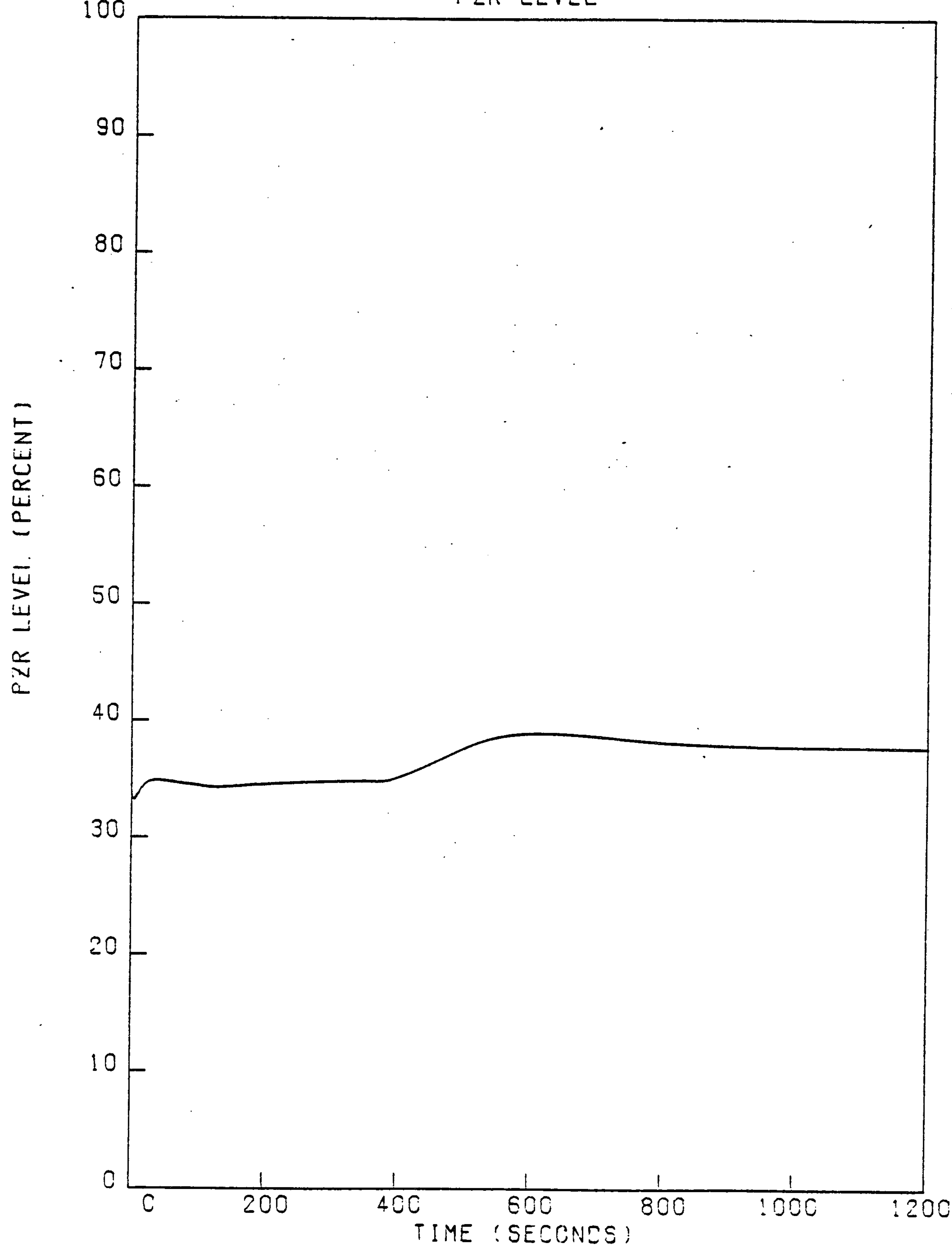


FIGURE 2-5
SONGS 1 PCT PWR NATURAL CIRCULATION
LOOP A COOLANT TEMPERATURES

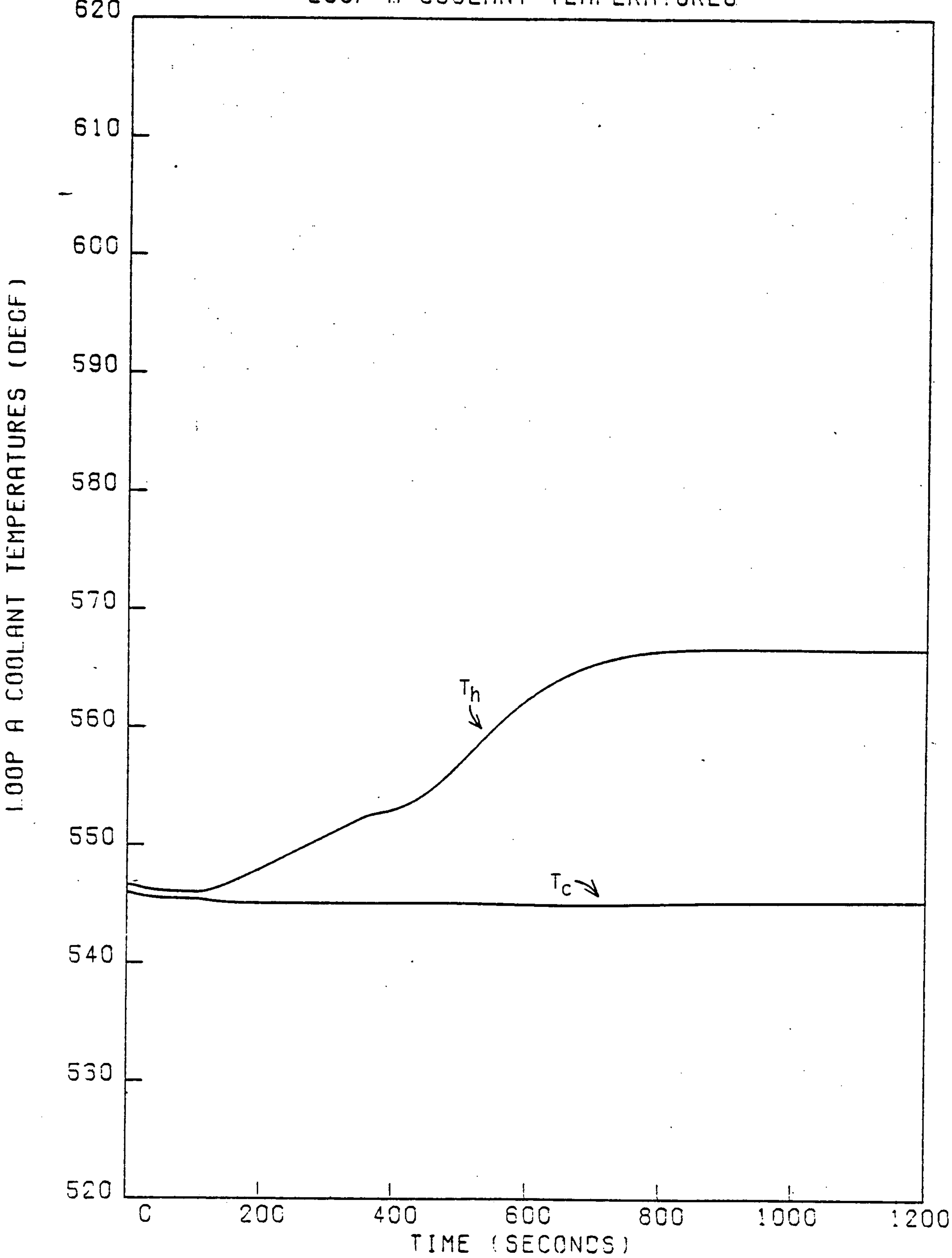


FIGURE 2-6
SONGS 1 PCT PWR NATURAL CIRCULATION
LOOP B COOLANT TEMPERATURES

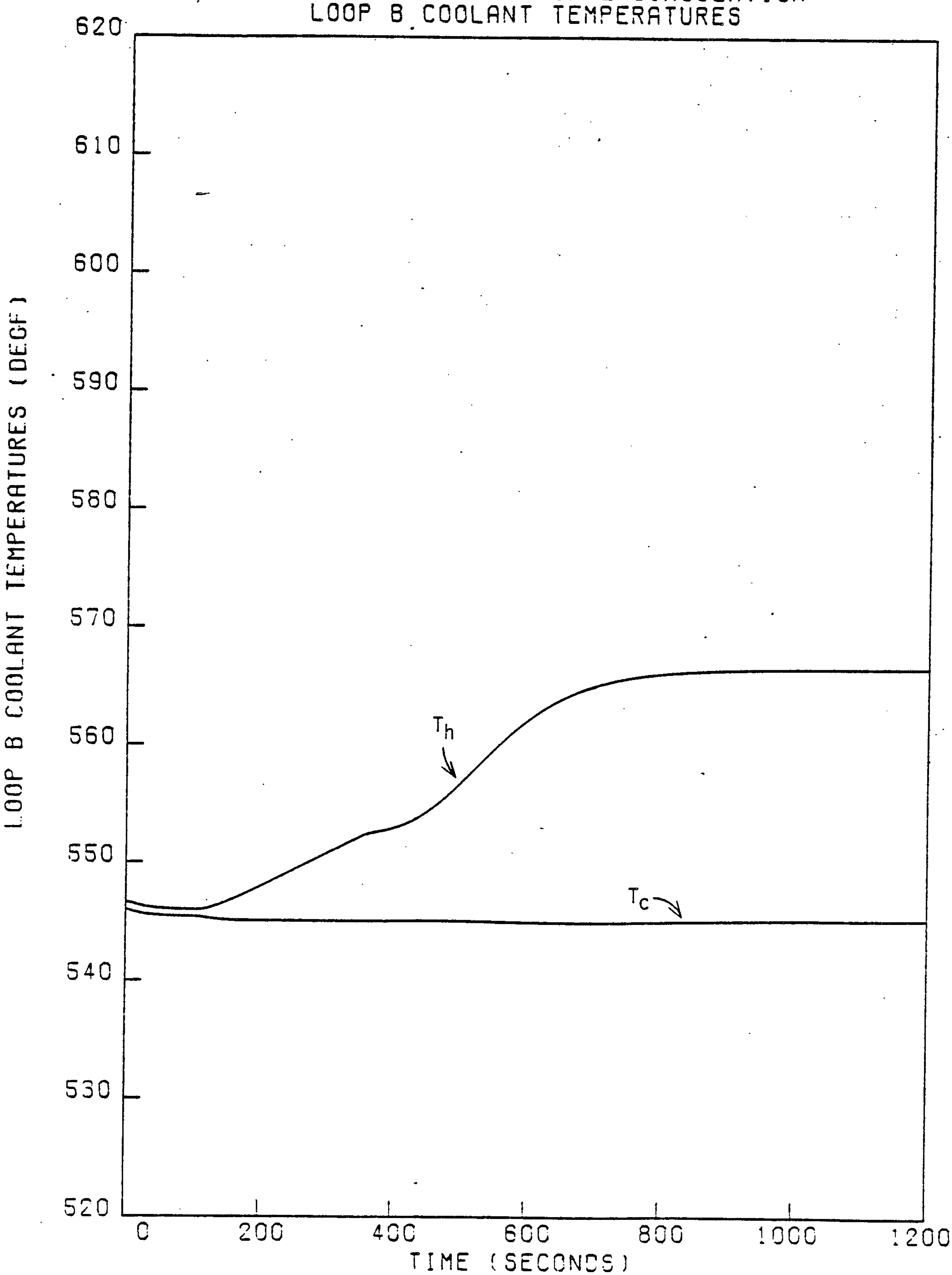


FIGURE 2-7
SONGS 1 PCT PWR NATURAL CIRCULATION
PRESSURIZER PRESSURE

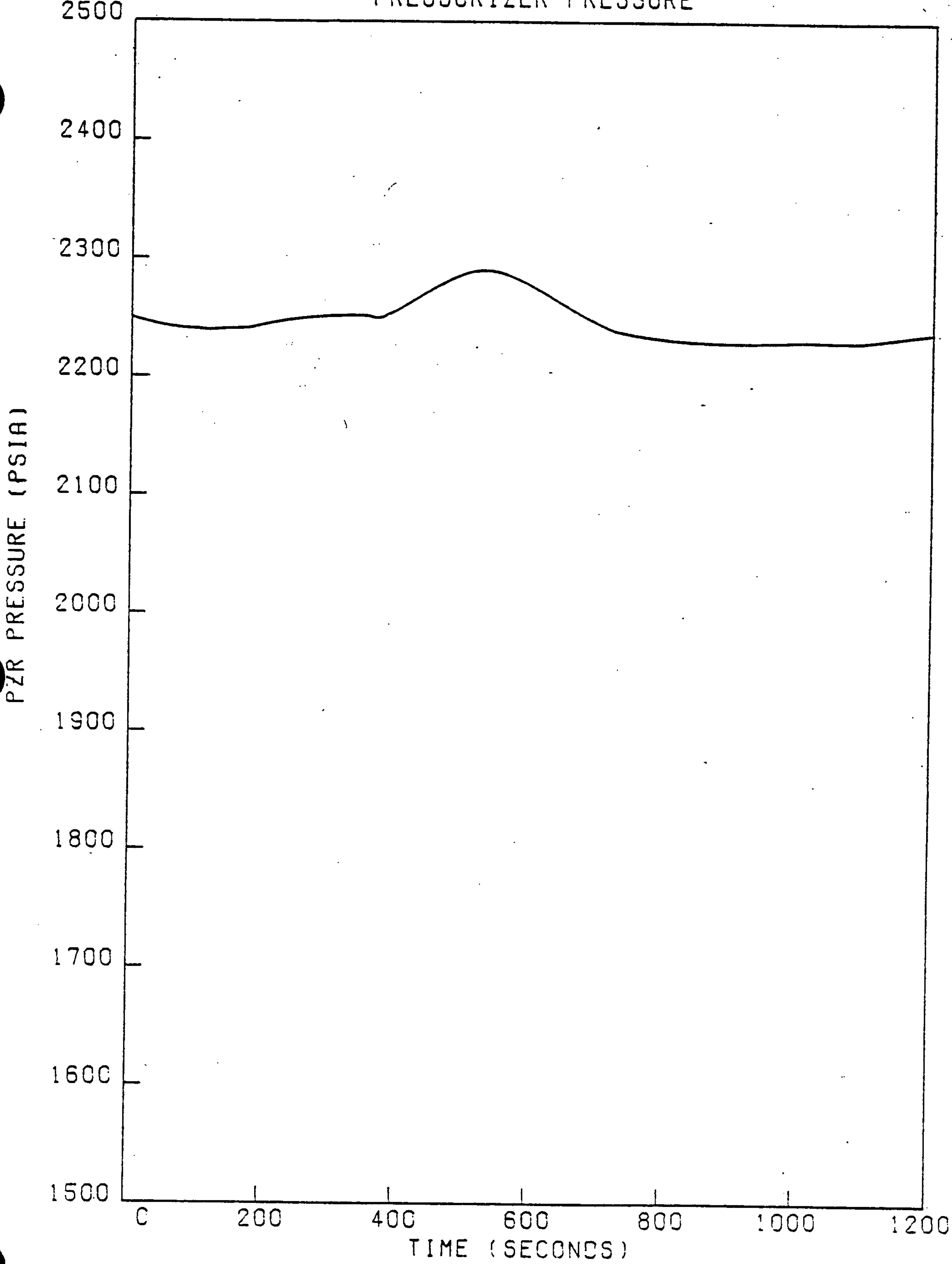


FIGURE 2-8
SONGS 1 PCT PWR NATURAL CIRCULATION
REACTOR COOLANT CORE FLOW

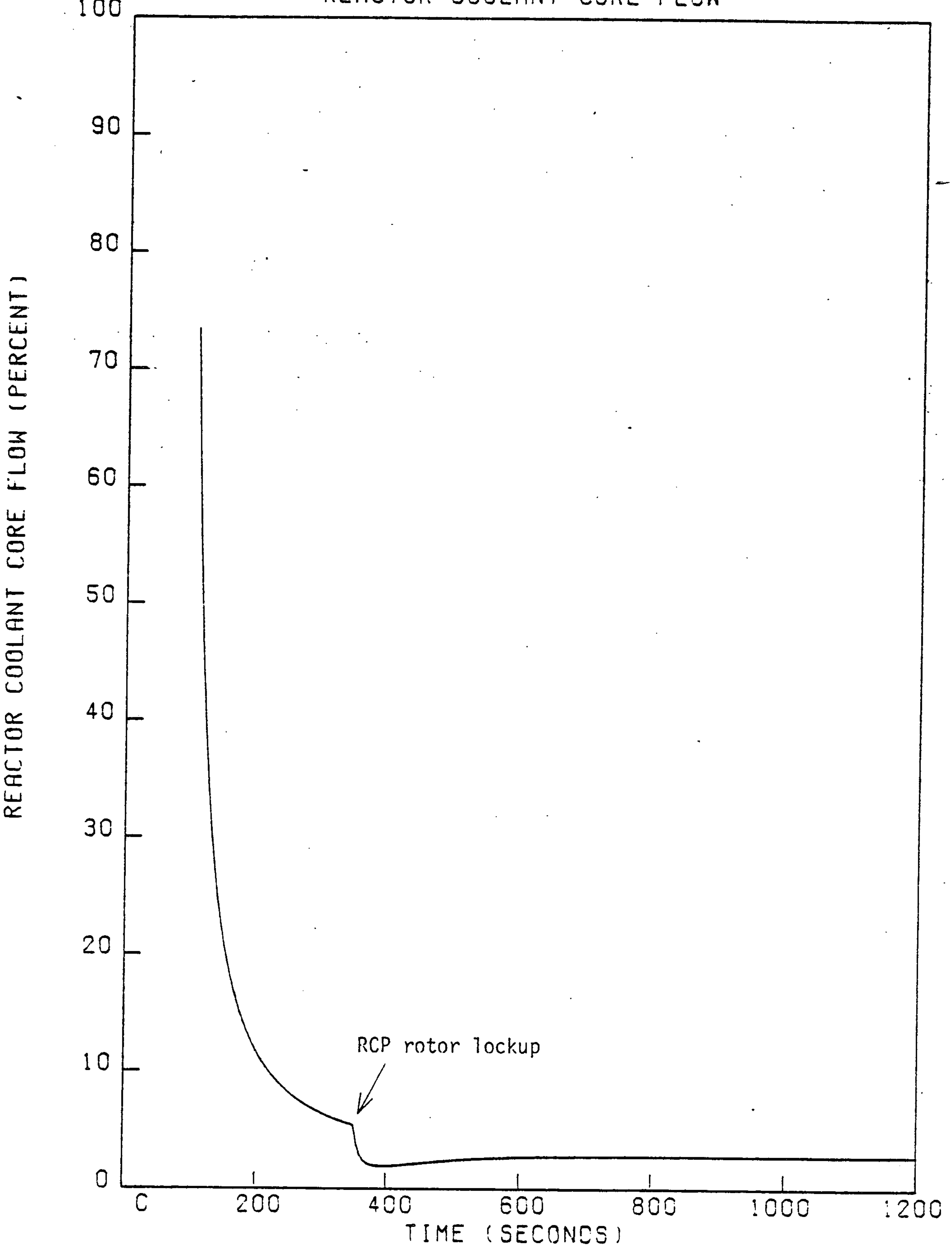


FIGURE 2-9
SONGS 1 PCT PWR NATURAL CIRCULATION
NARROW RANGE CORE FLOW

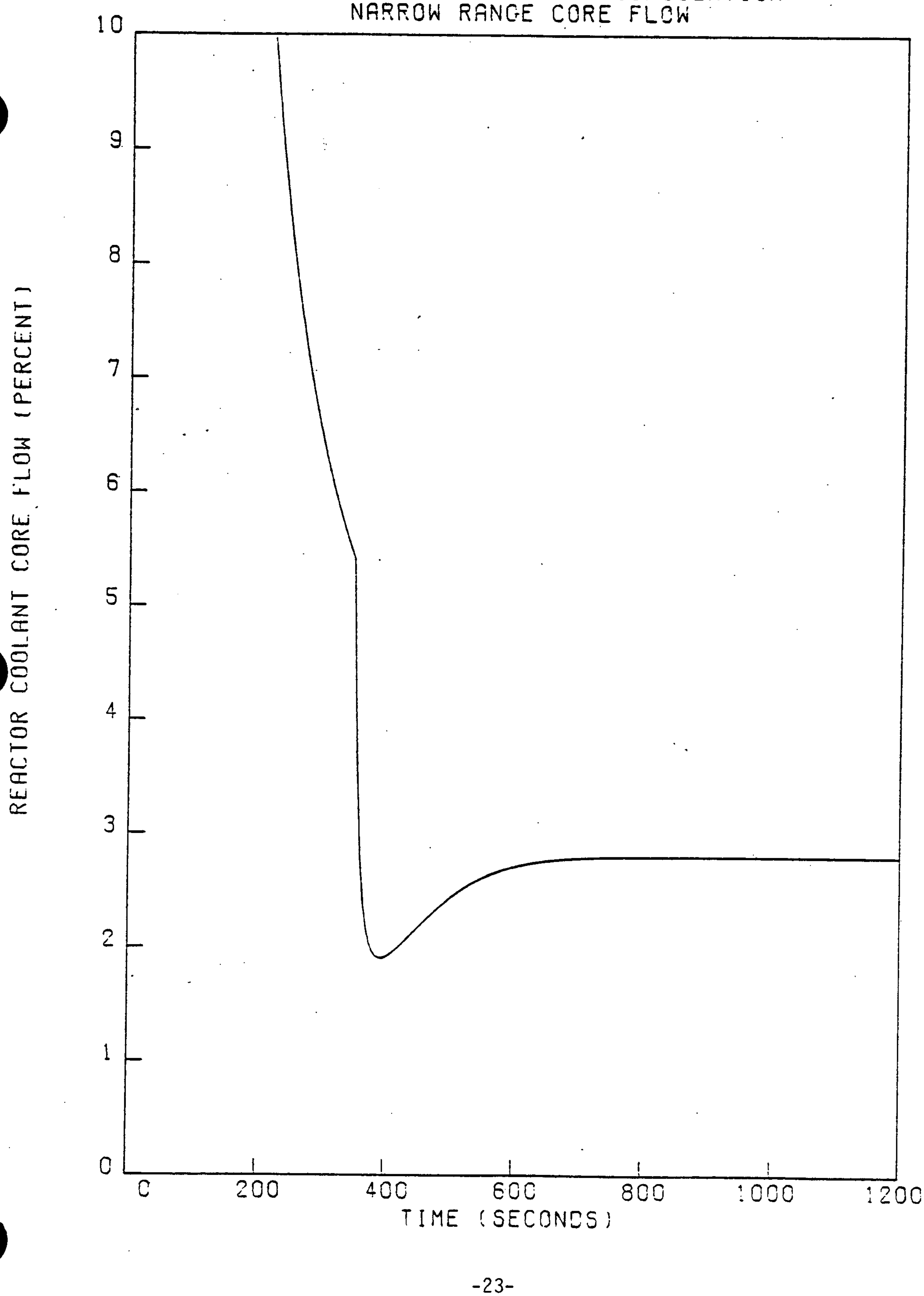


FIGURE 2-10
SONGS 1 PCT PWR NATURAL CIRCULATION
CHARGING FLOW

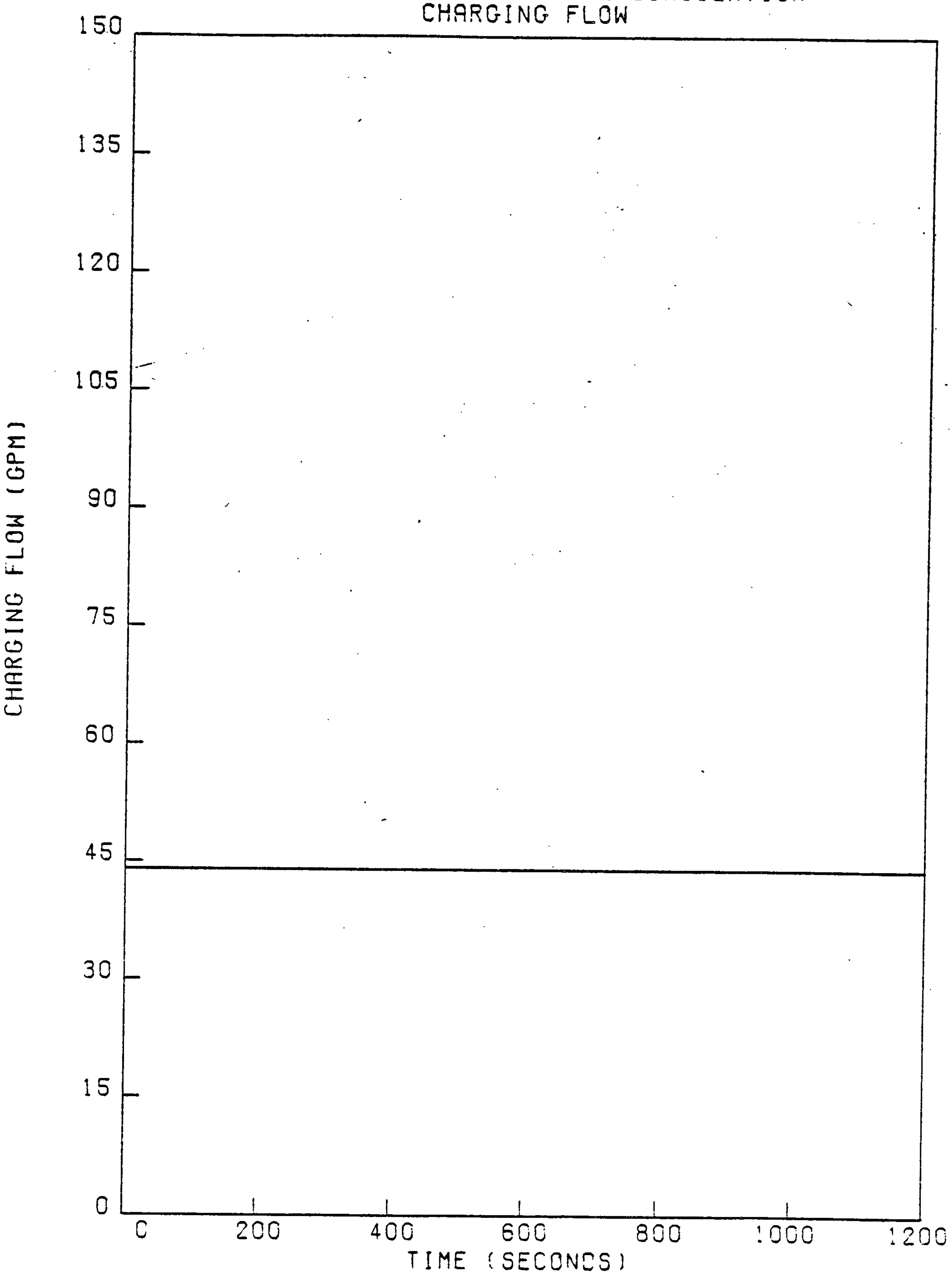


FIGURE 2-11
SONGS 1 PCT PWR NATURAL CIRCULATION
LETDOWN FLOW

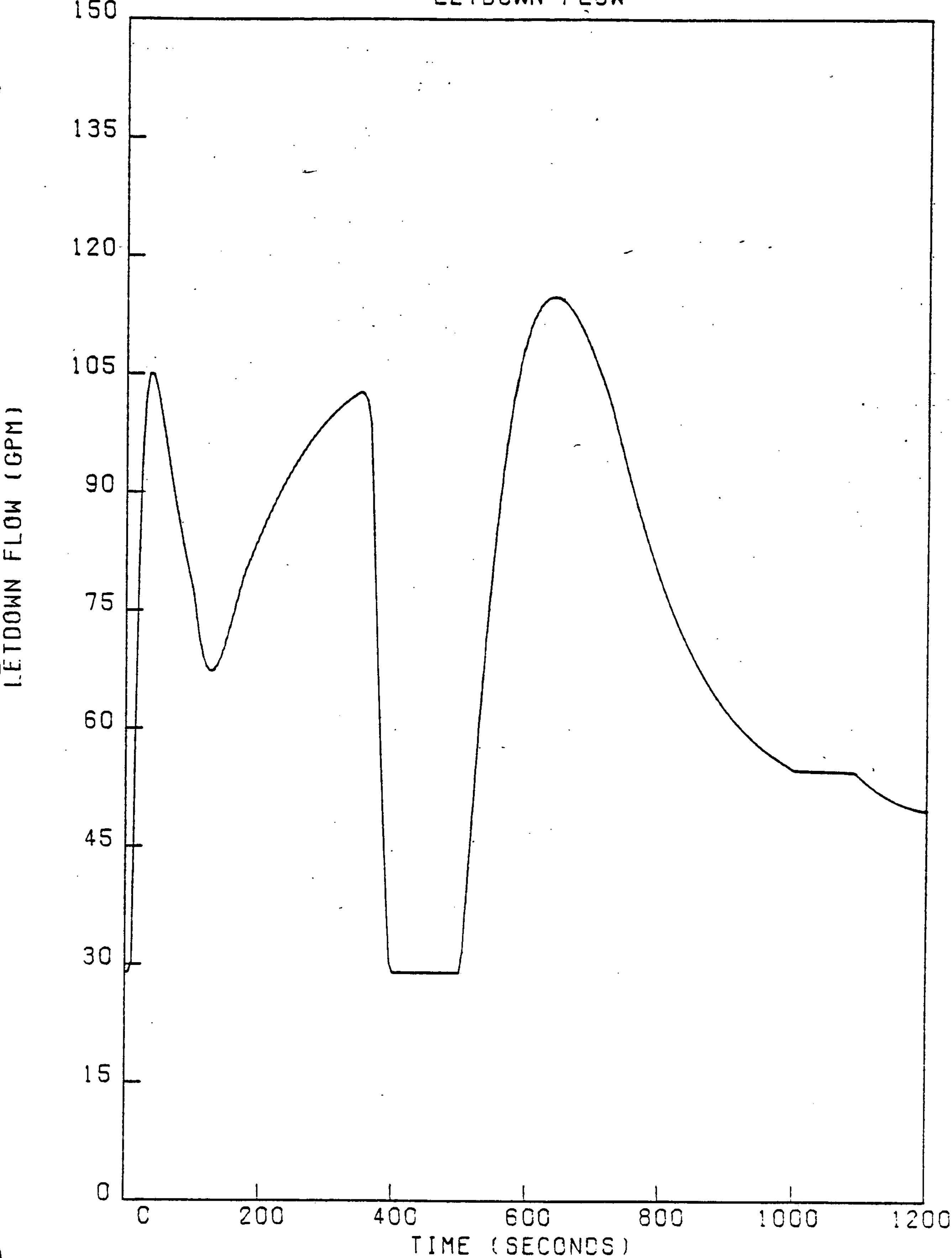


FIGURE 2-12
SONGS 1 PCT PWR NATURAL CIRCULATION
PZR HEATER OUTPUTS

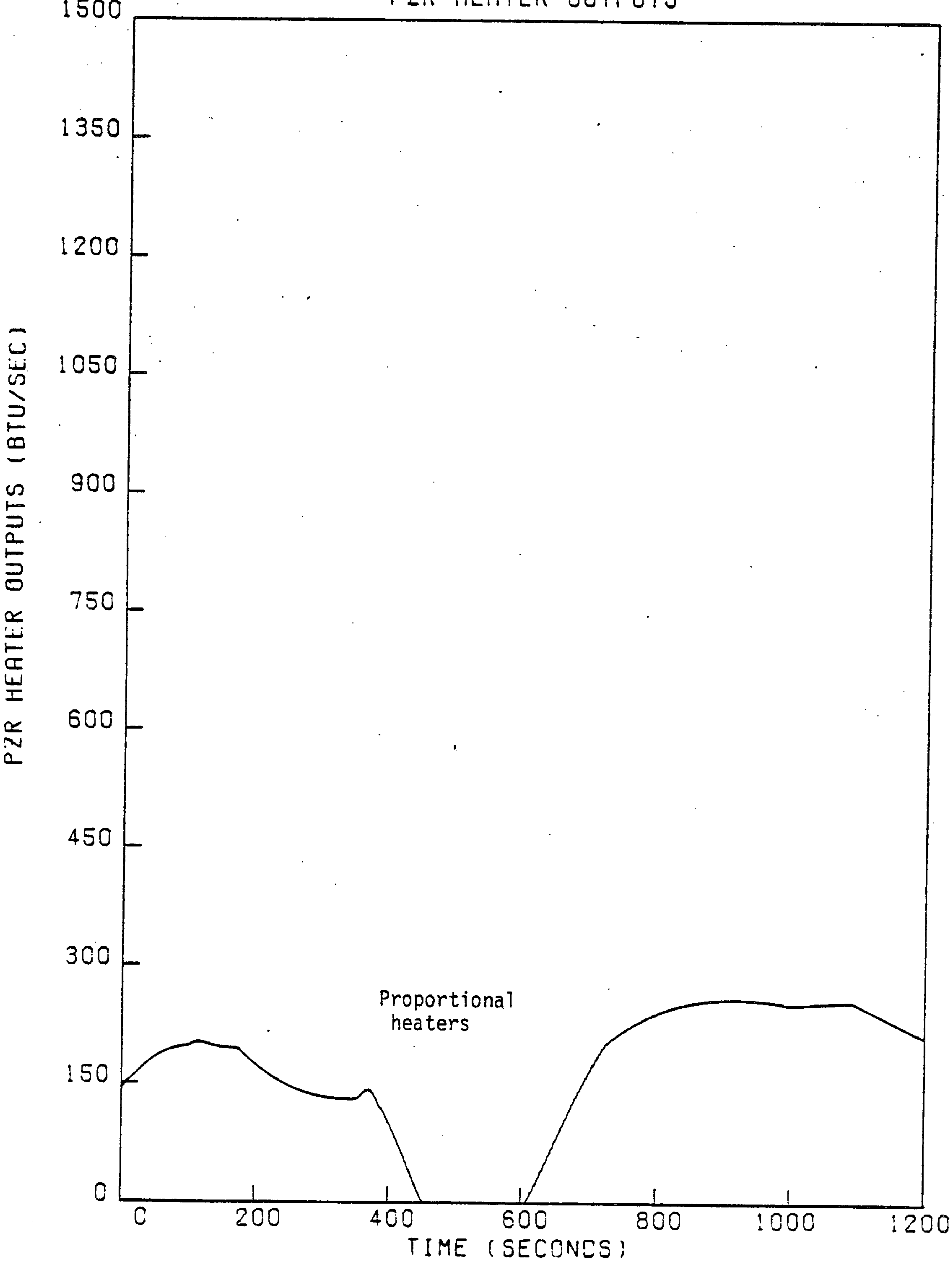


FIGURE 2-13
SONGS 3 PCT PWR NATURAL CIRCULATION
NARROW RANGE CORE POWER

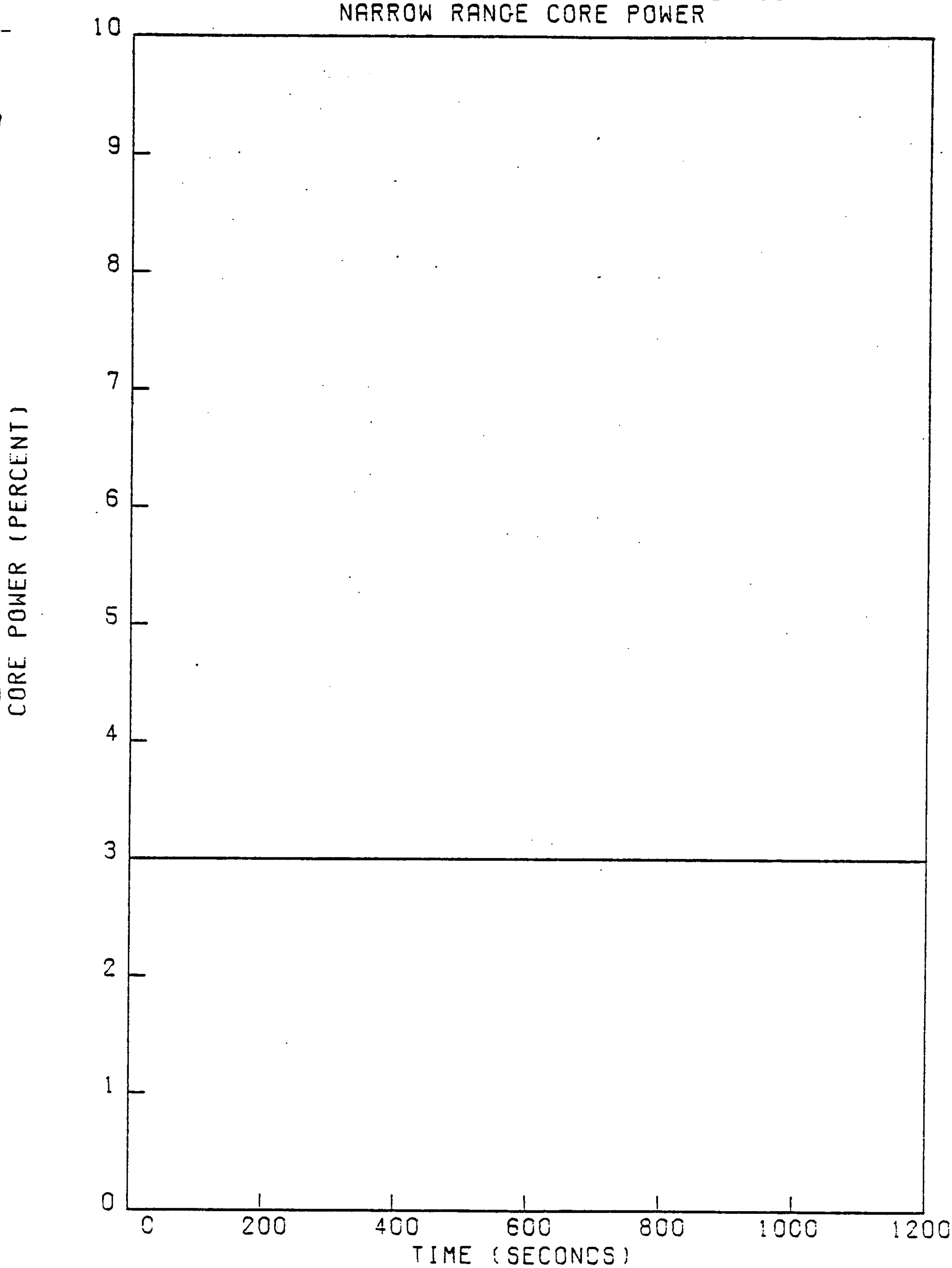


FIGURE 2-14
SONGS 3 PCT PWR NATURAL CIRCULATION
S.G. A PRESSURE

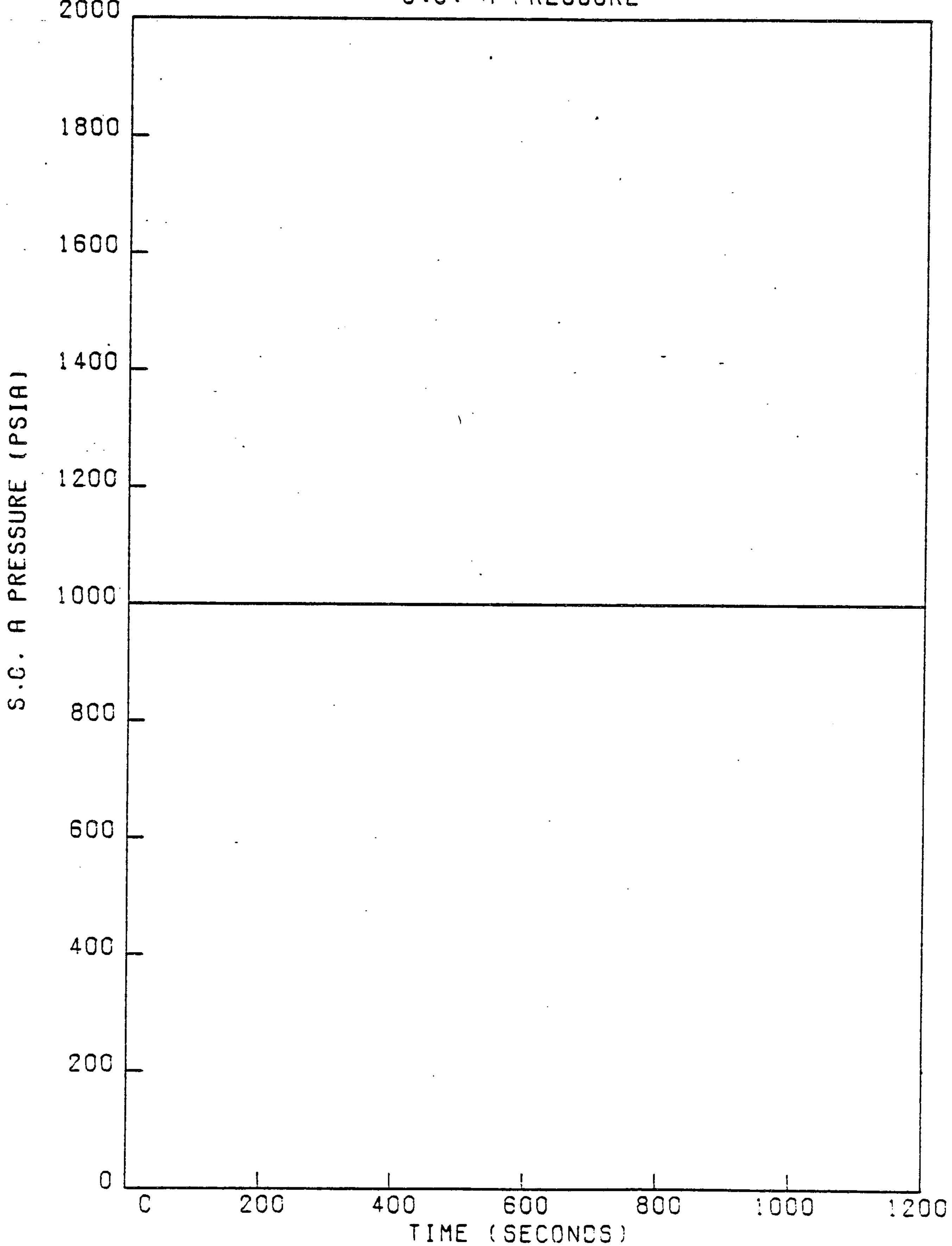


FIGURE 2-15
SONGS 3 PCT PWR NATURAL CIRCULATION
S.G. B PRESSURE

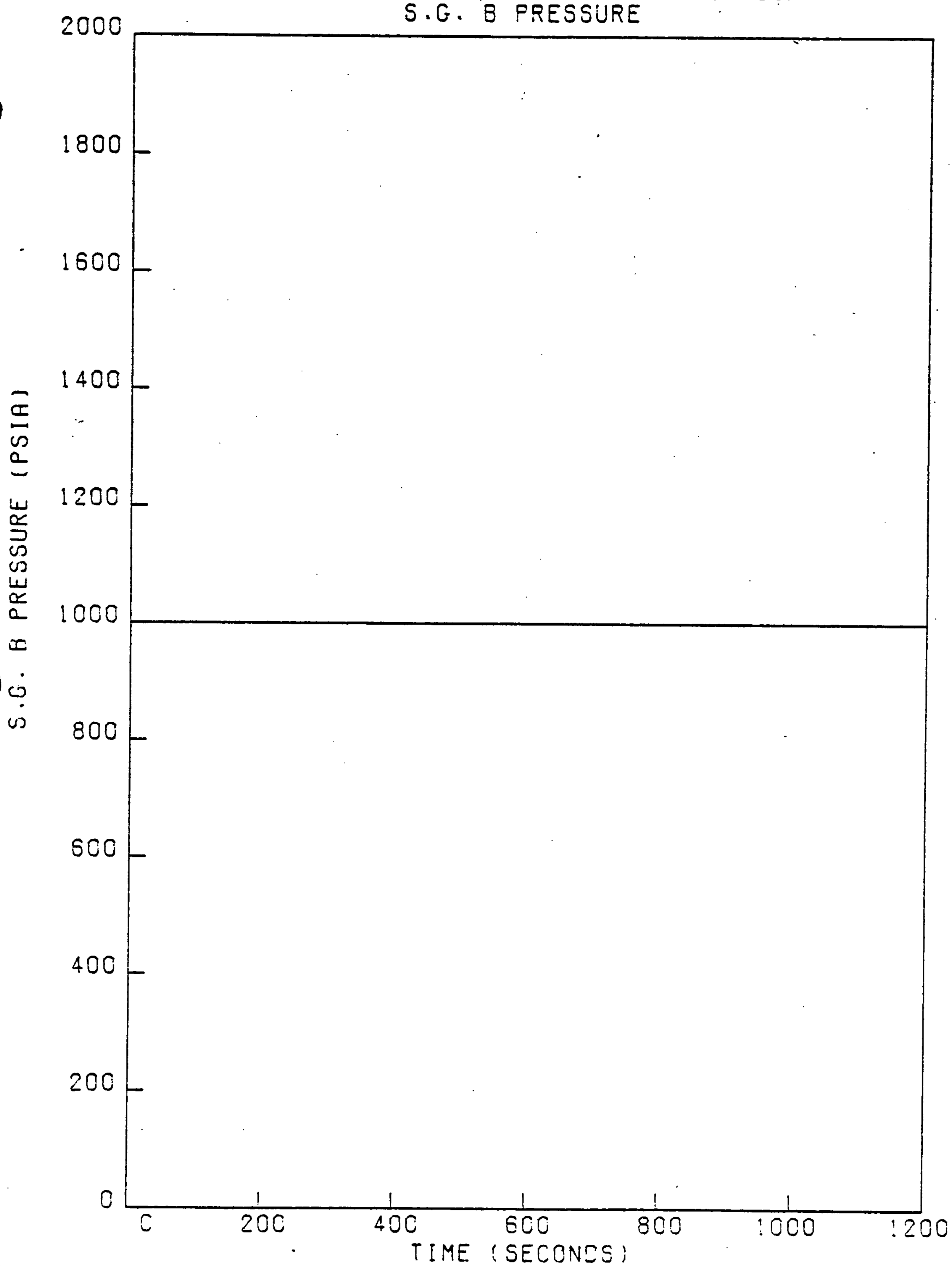


FIGURE 2-16
SONGS 3 PCT PWR NATURAL CIRCULATION
PZR LEVEL

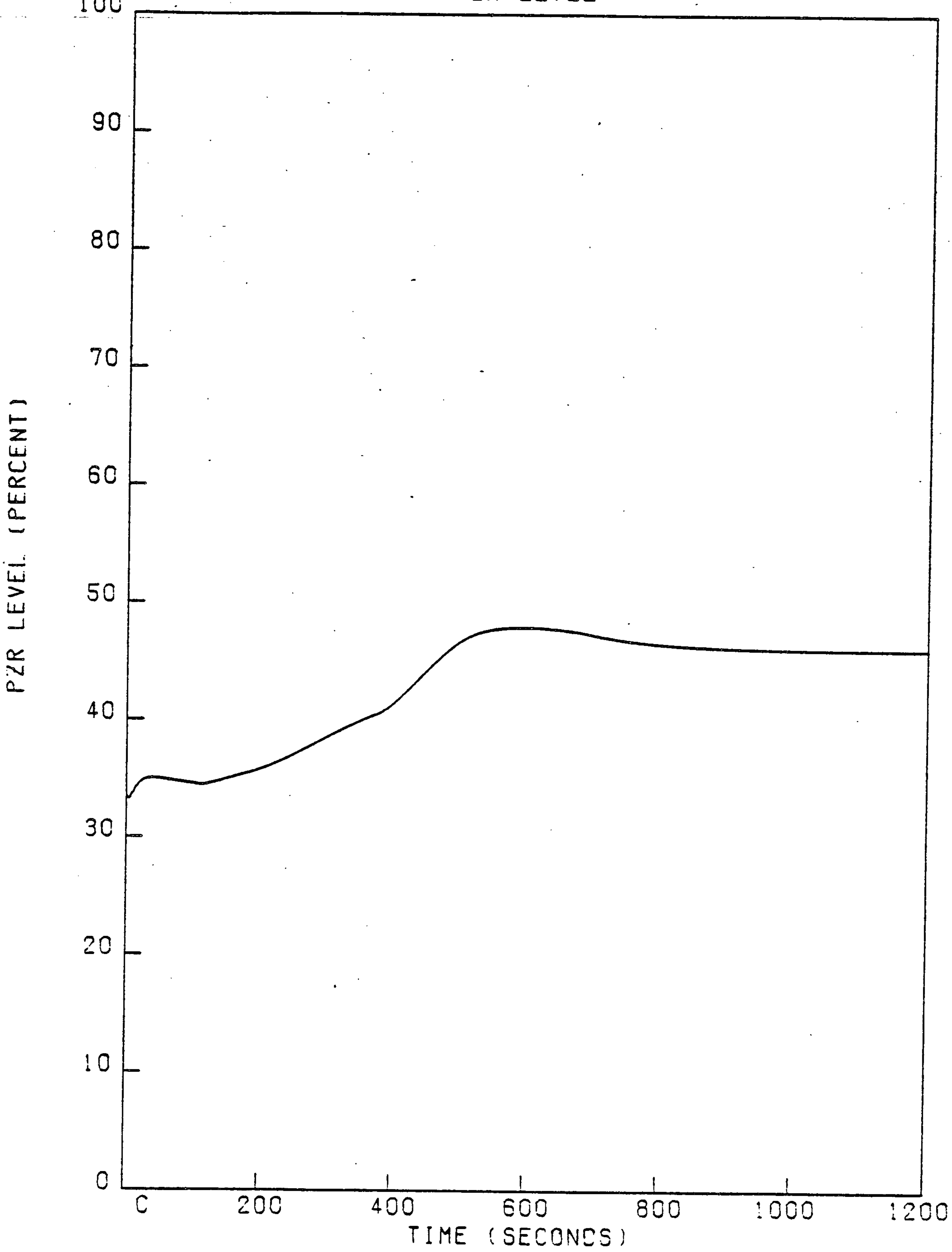


FIGURE 2-17
SONGS 3 PCT PWR NATURAL CIRCULATION
LOOP A COOLANT TEMPERATURES

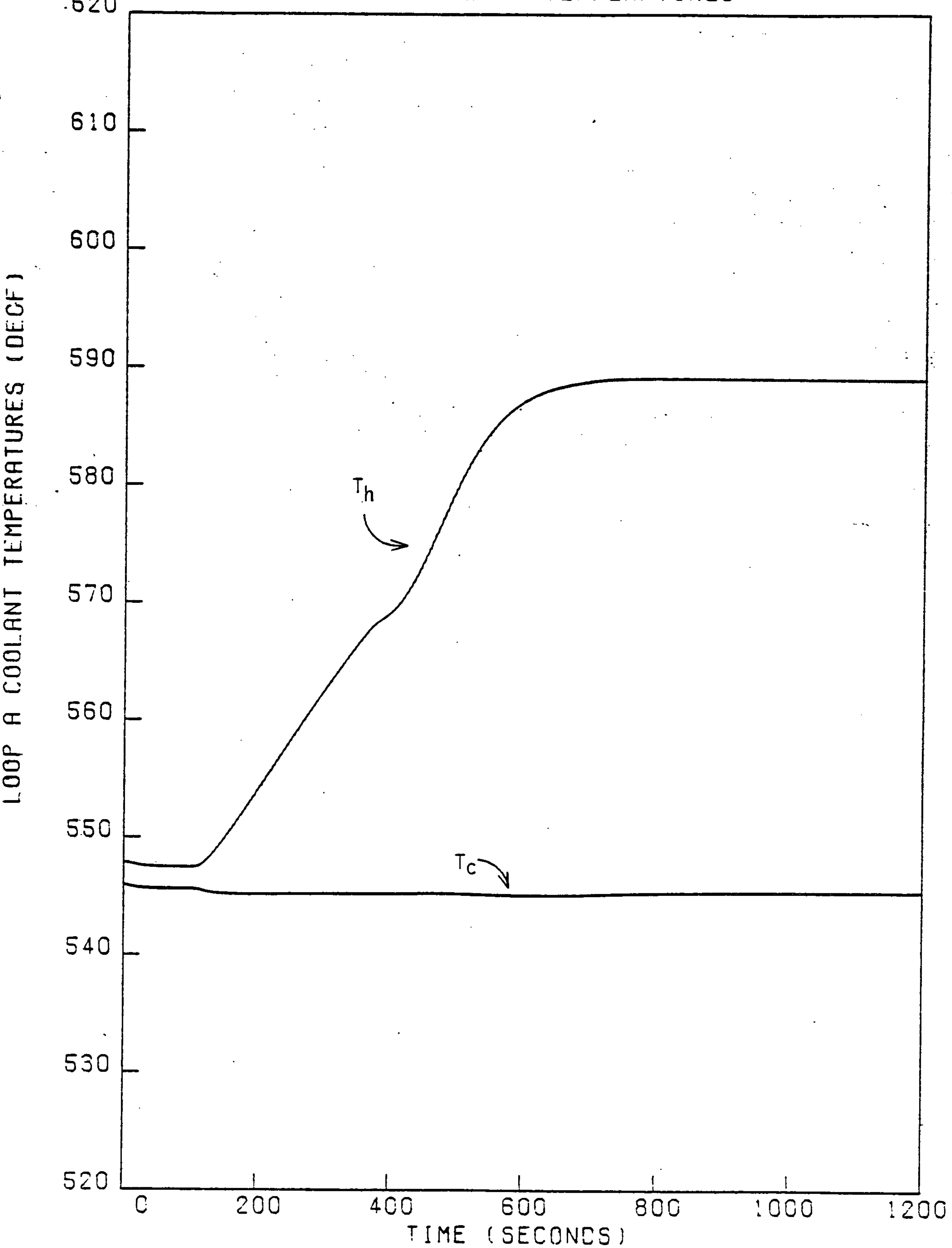


FIGURE 2-18
SONGS 3 PCT PWR NATURAL CIRCULATION
LOOP B COOLANT TEMPERATURES

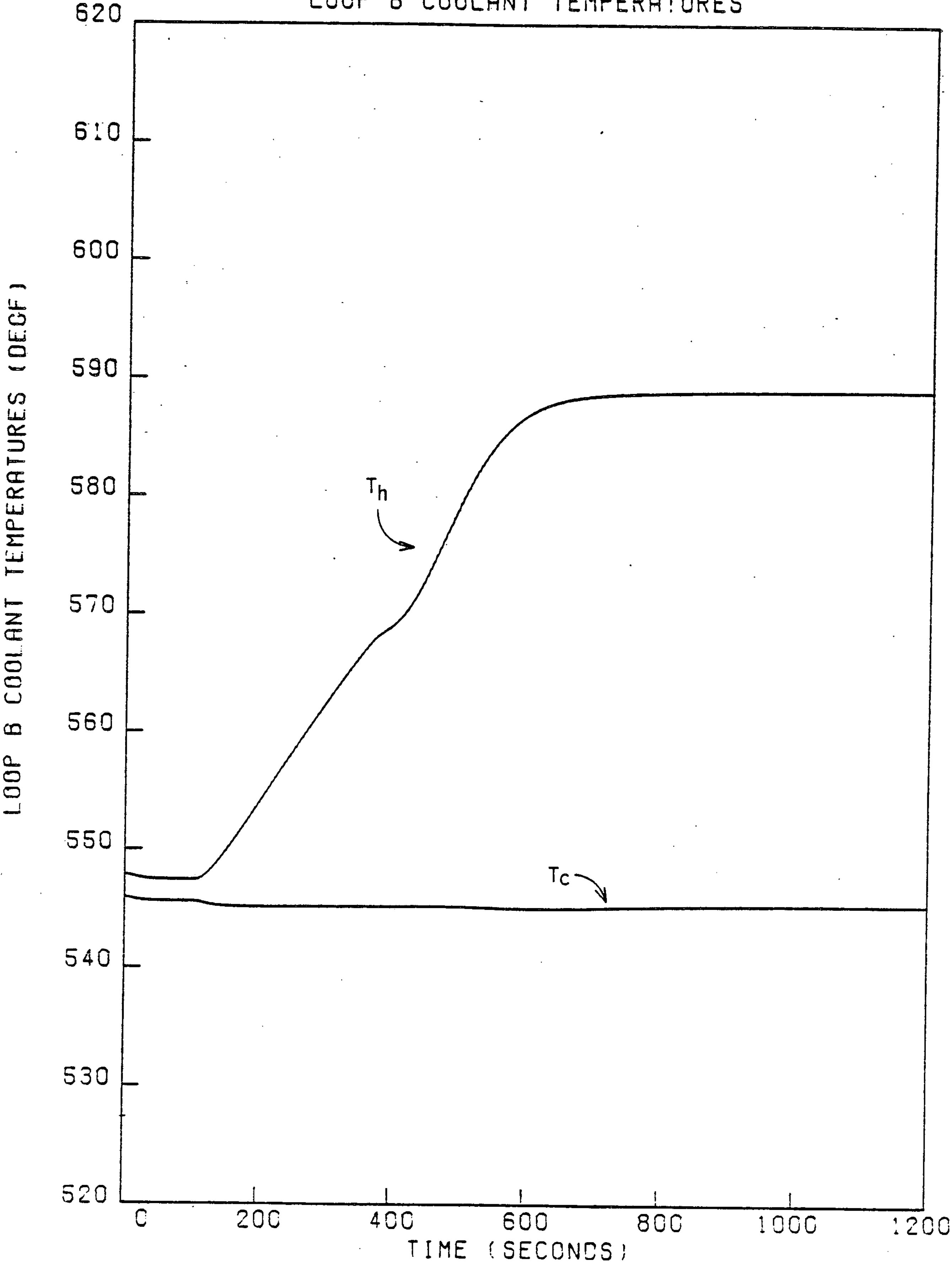


FIGURE 2-19
SONGS 3 PCT PWR NATURAL CIRCULATION
PRESSURIZER PRESSURE

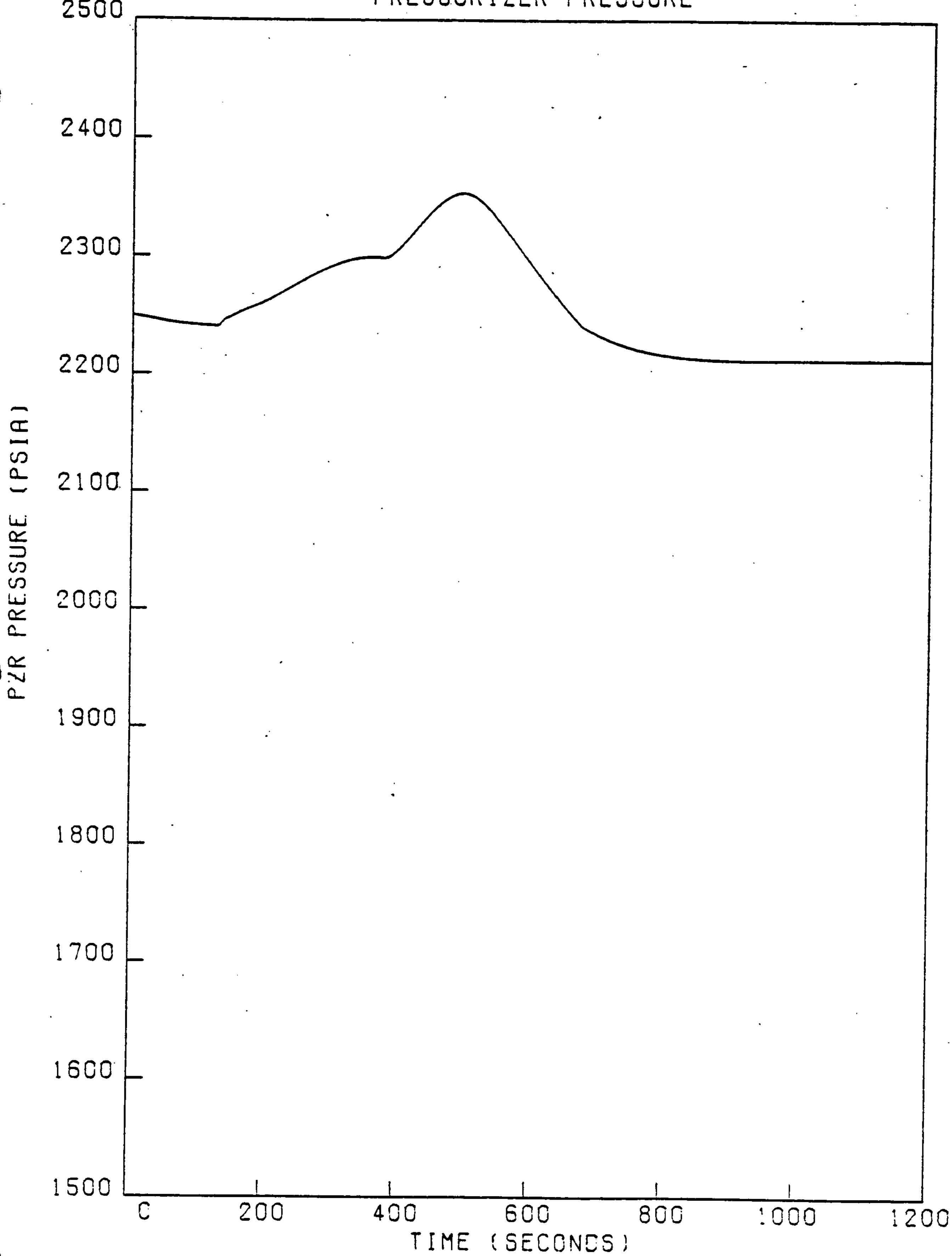


FIGURE 2-20
SONGS 3 PCT PWR NATURAL CIRCULATION
REACTOR COOLANT CORE FLOW

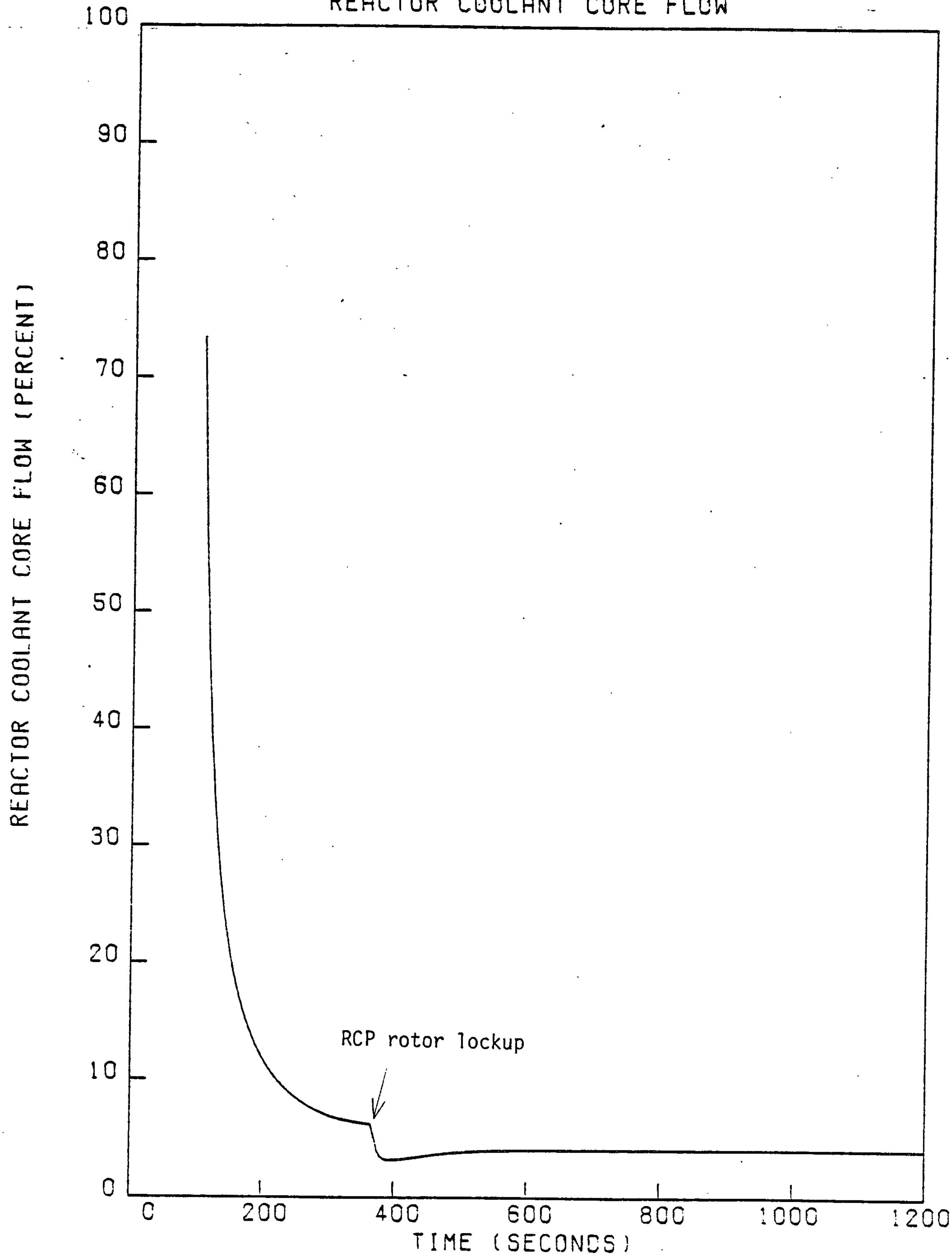


FIGURE 2-21
SONGS 3 PCT PWR NATURAL CIRCULATION
NARROW RANGE CORE FLOW

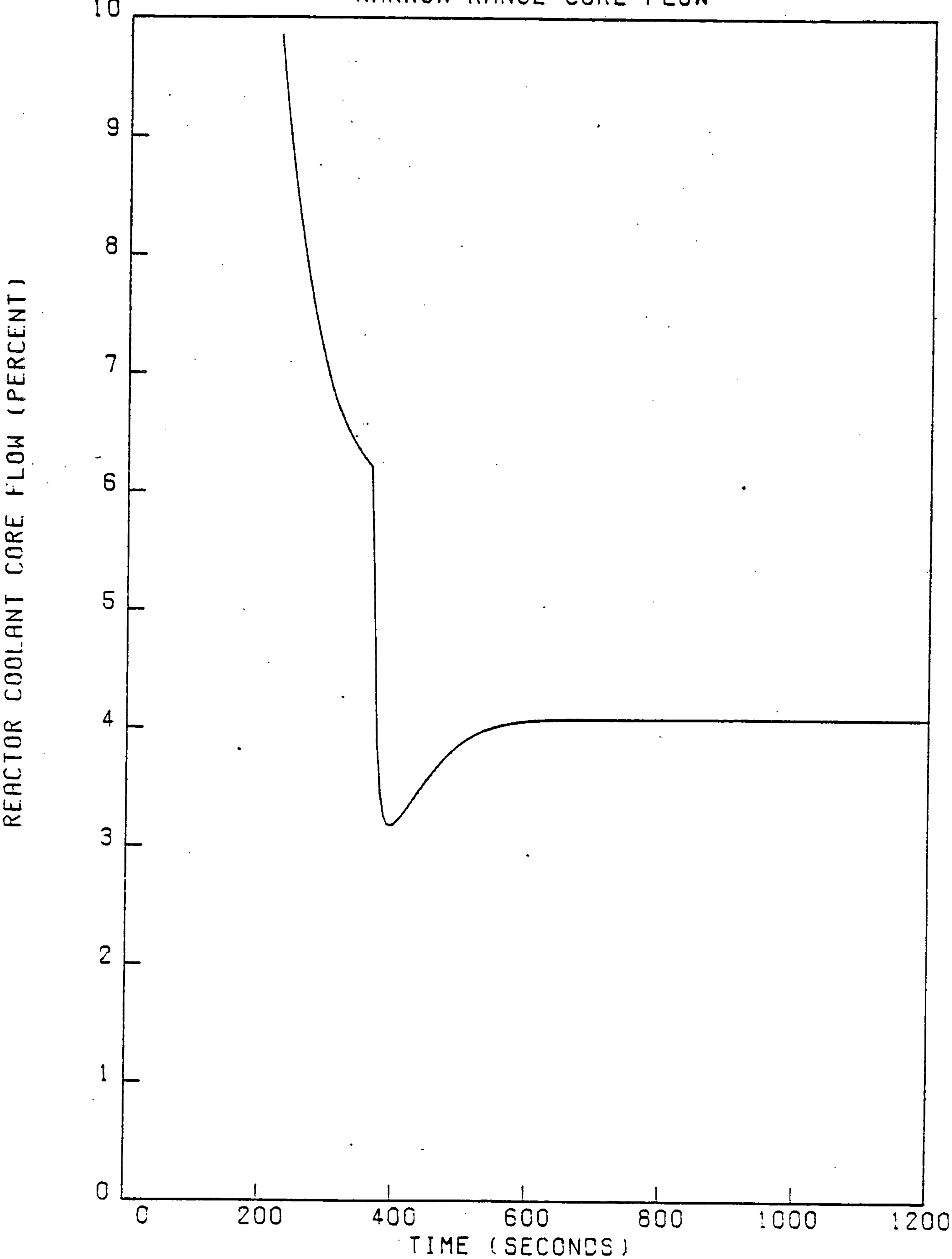


FIGURE 2-22
SONGS 3 PCT PWR NATURAL CIRCULATION
CHARGING FLOW

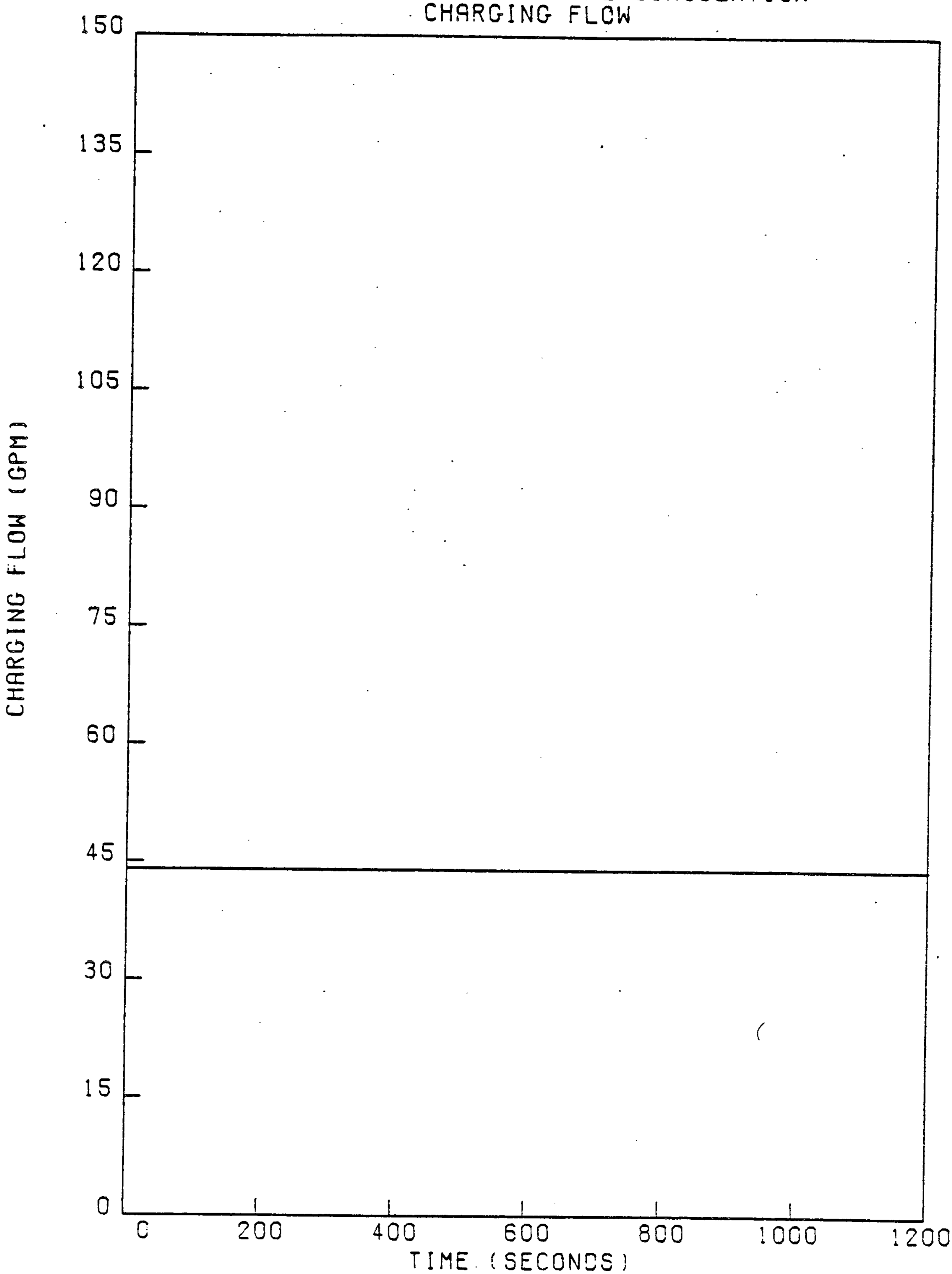


FIGURE 2-23
SONGS 3 PCT PWR NATURAL CIRCULATION
LETDOWN FLOW

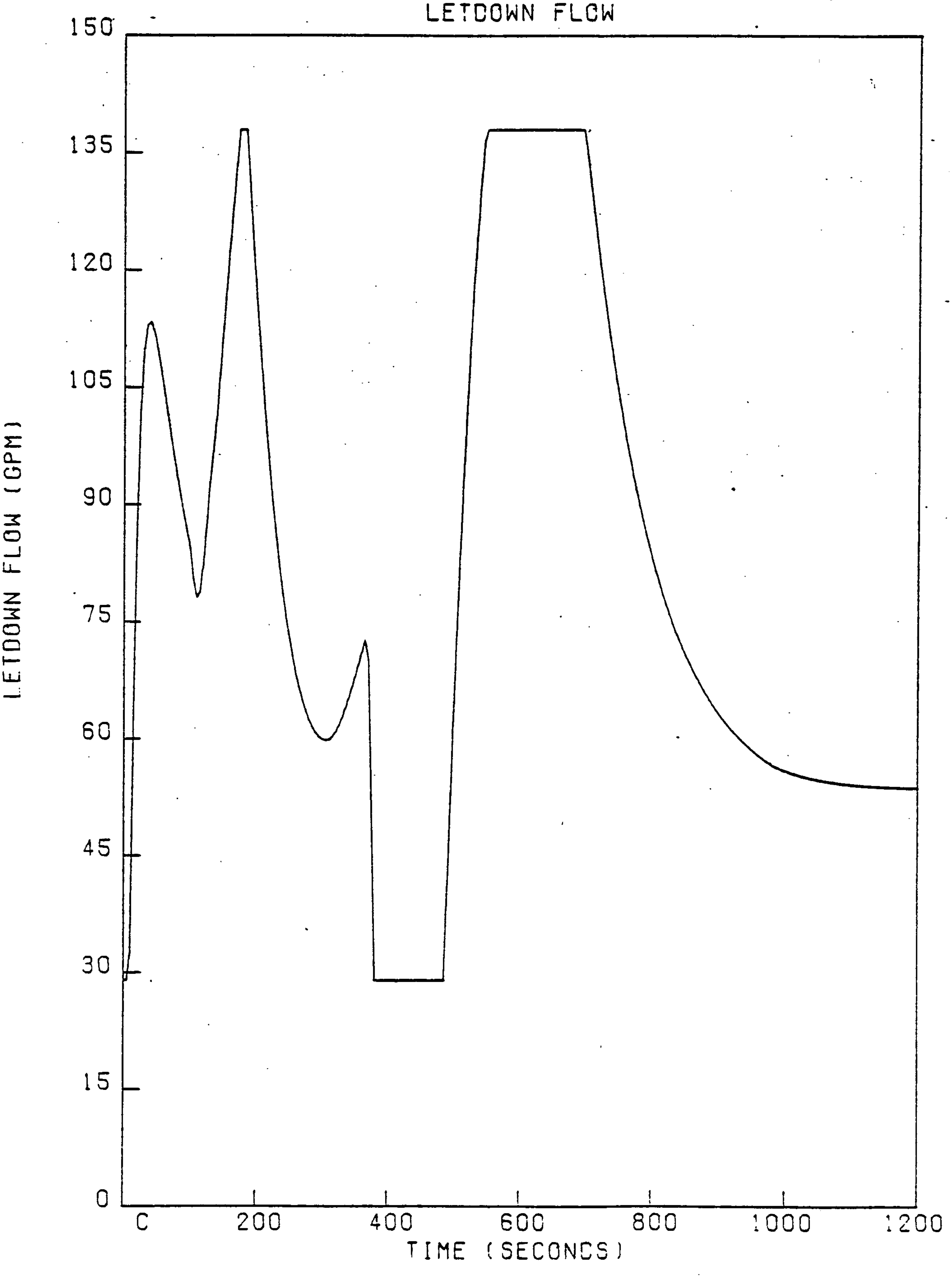


FIGURE 2-24
SONGS 3 PCT PWR NATURAL CIRCULATION
PZR HEATER OUTPUTS

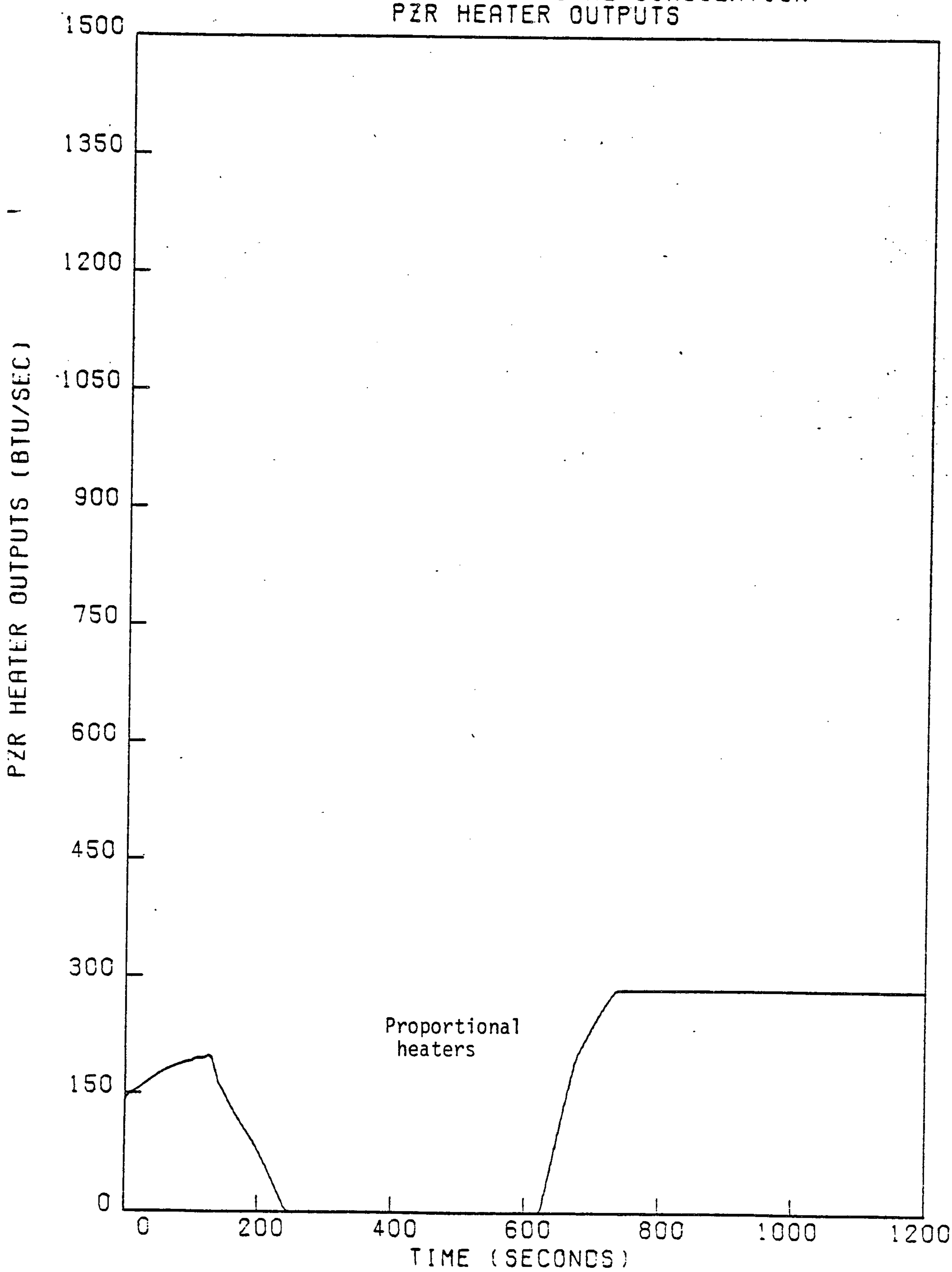


FIGURE 2-25
SONGS 5 PCT PWR NATURAL CIRCULATION
NARROW RANGE CORE POWER

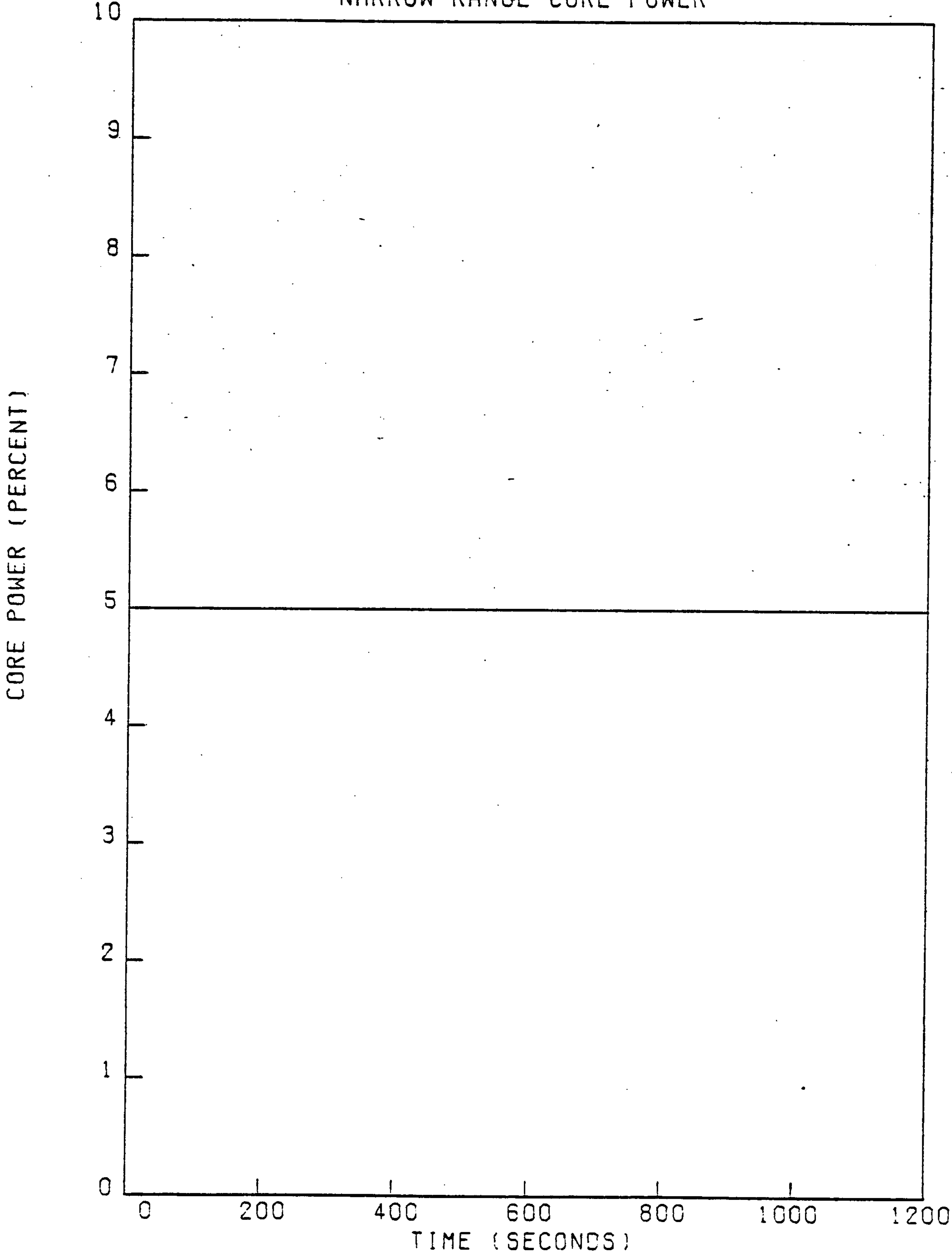


FIGURE 2-26
SONGS 5 PCT PWR NATURAL CIRCULATION
S.G. A PRESSURE

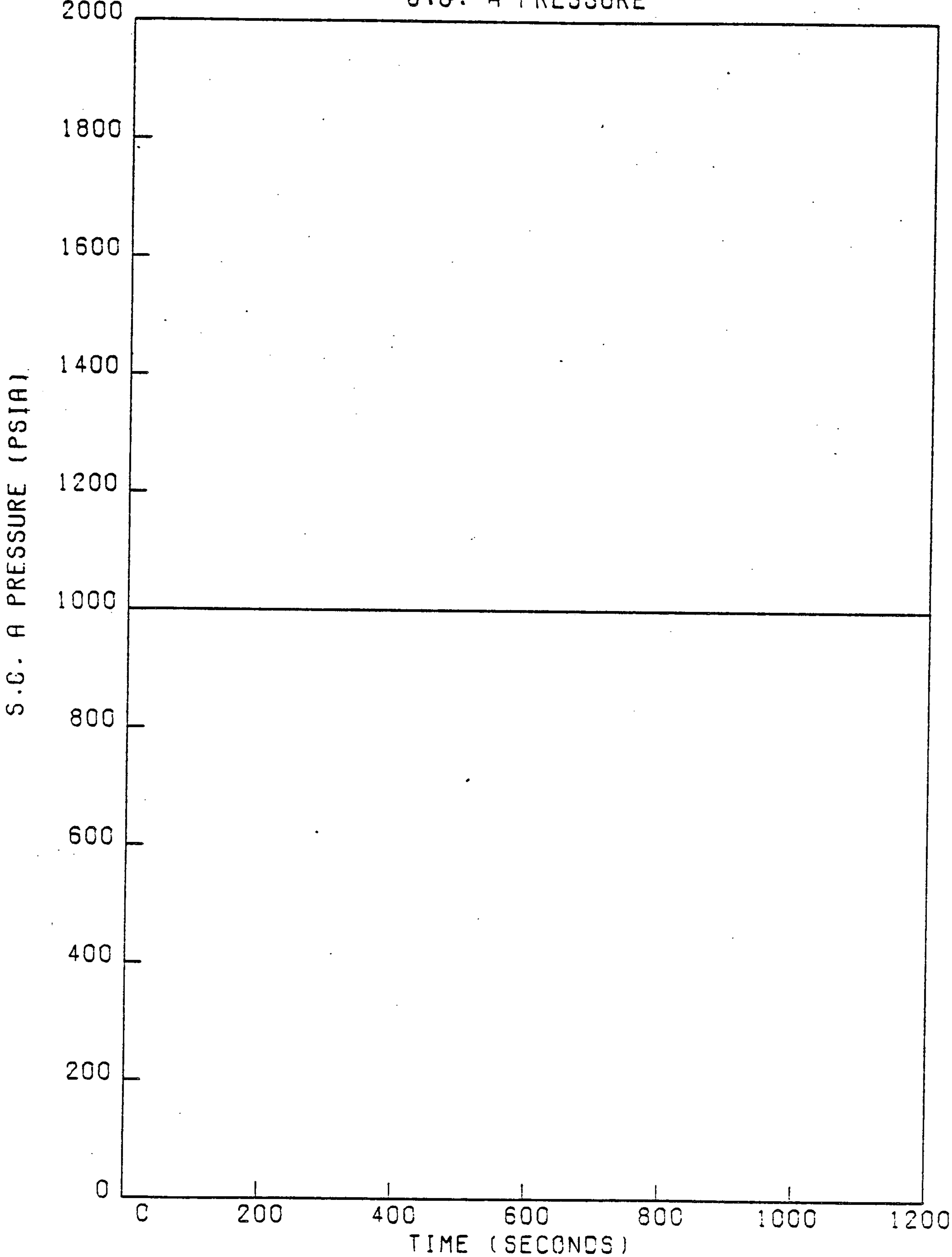


FIGURE 2-27
SONGS 5 PCT PWR NATURAL CIRCULATION
S.G. B PRESSURE

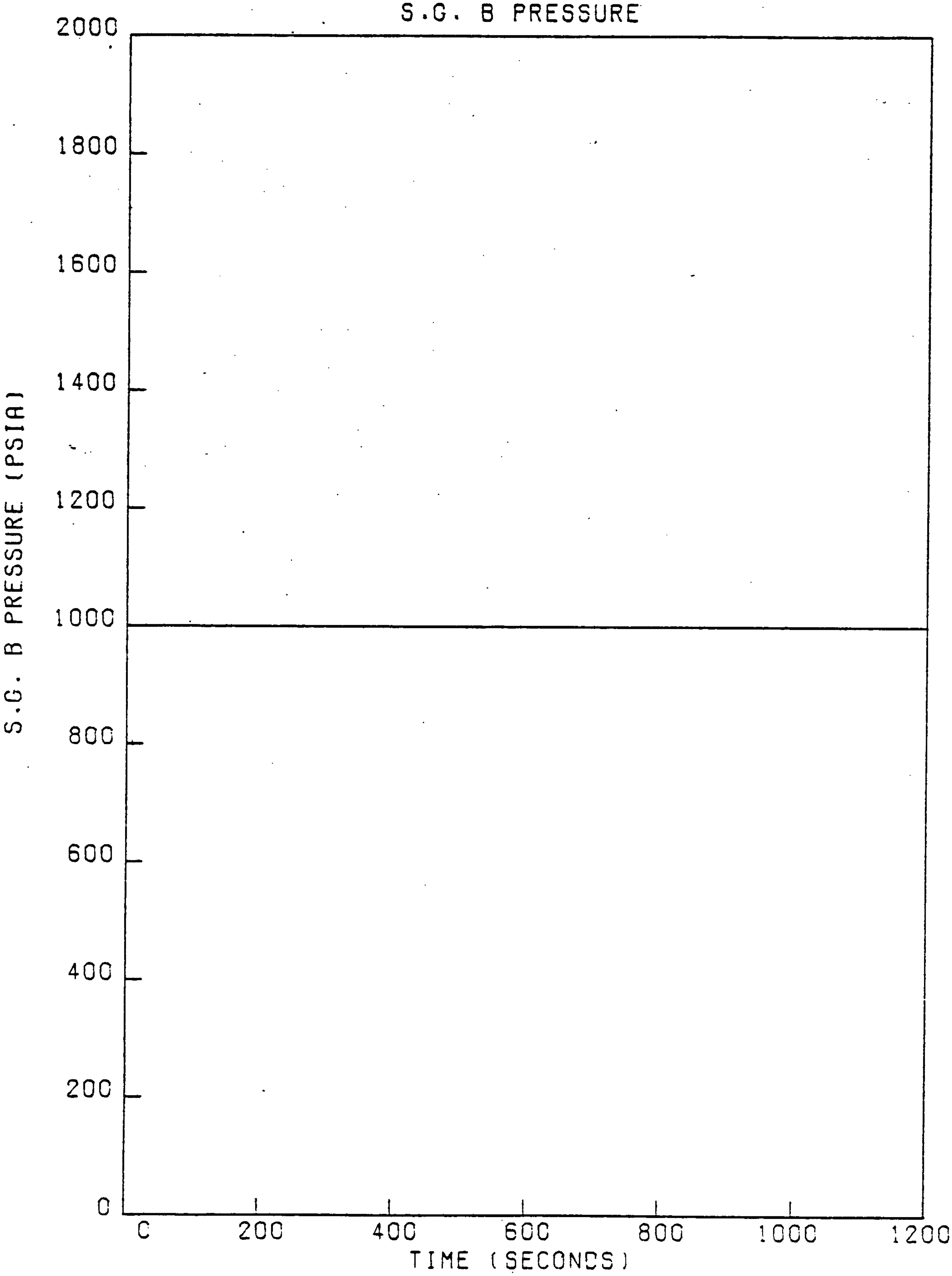


FIGURE 2-28
SONGS 5 PCT PWR NATURAL CIRCULATION
PZR LEVEL

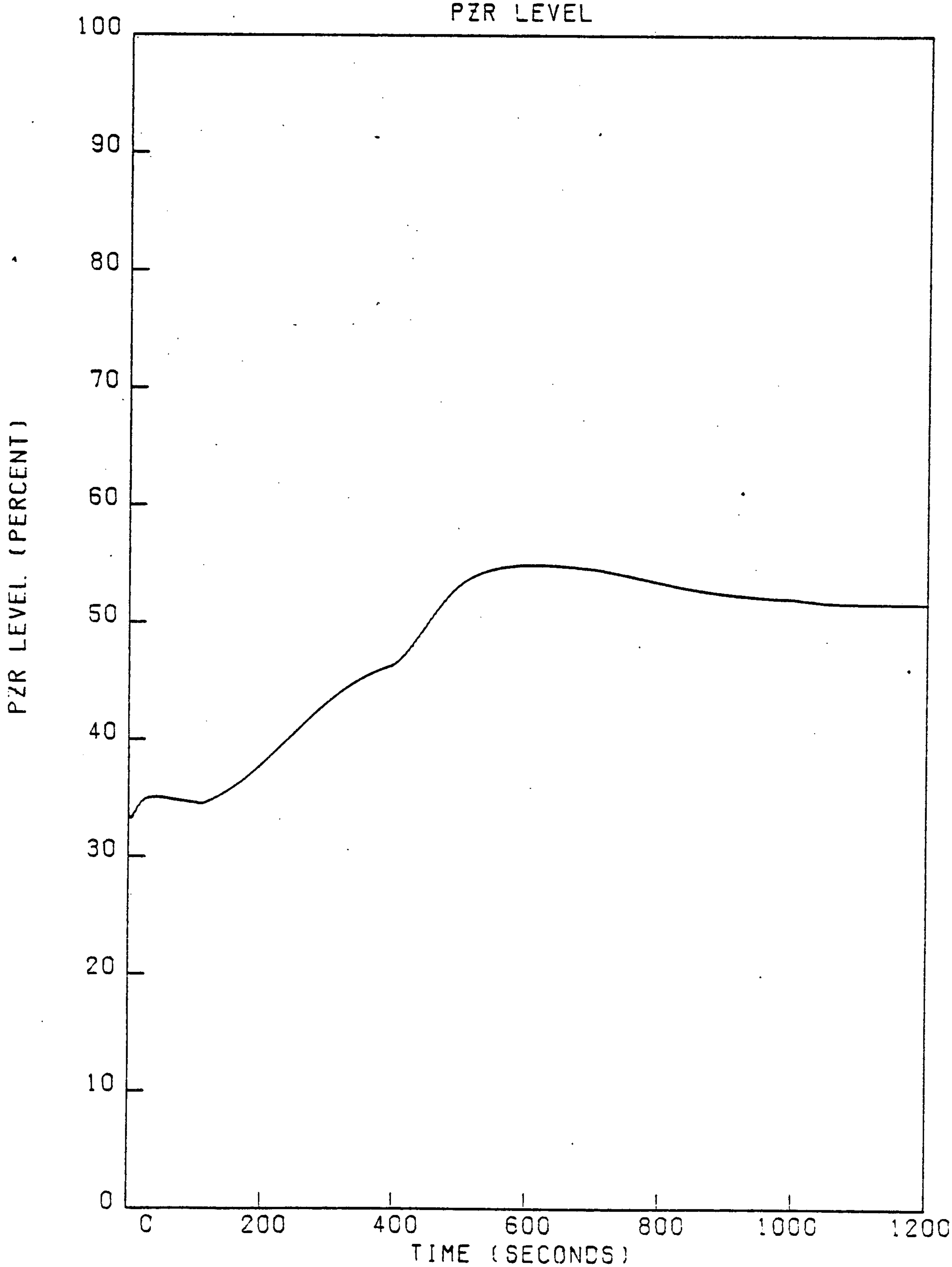


FIGURE 2-29
SONGS 5 PCT PWR NATURAL CIRCULATION
LOOP A COOLANT TEMPERATURES

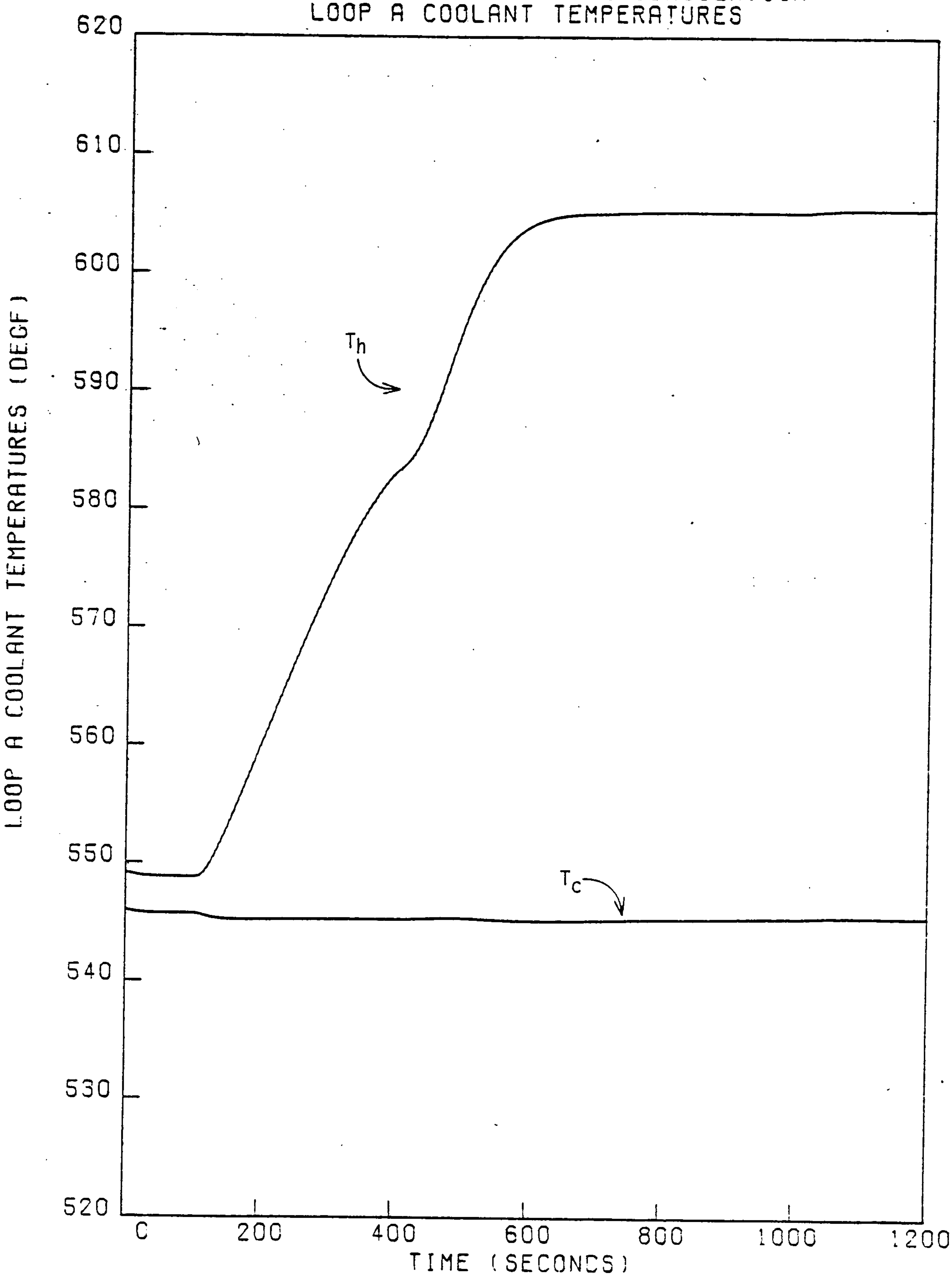


FIGURE 2-30
SONGS 5 PCT PWR NATURAL CIRCULATION
LOOP B COOLANT TEMPERATURES

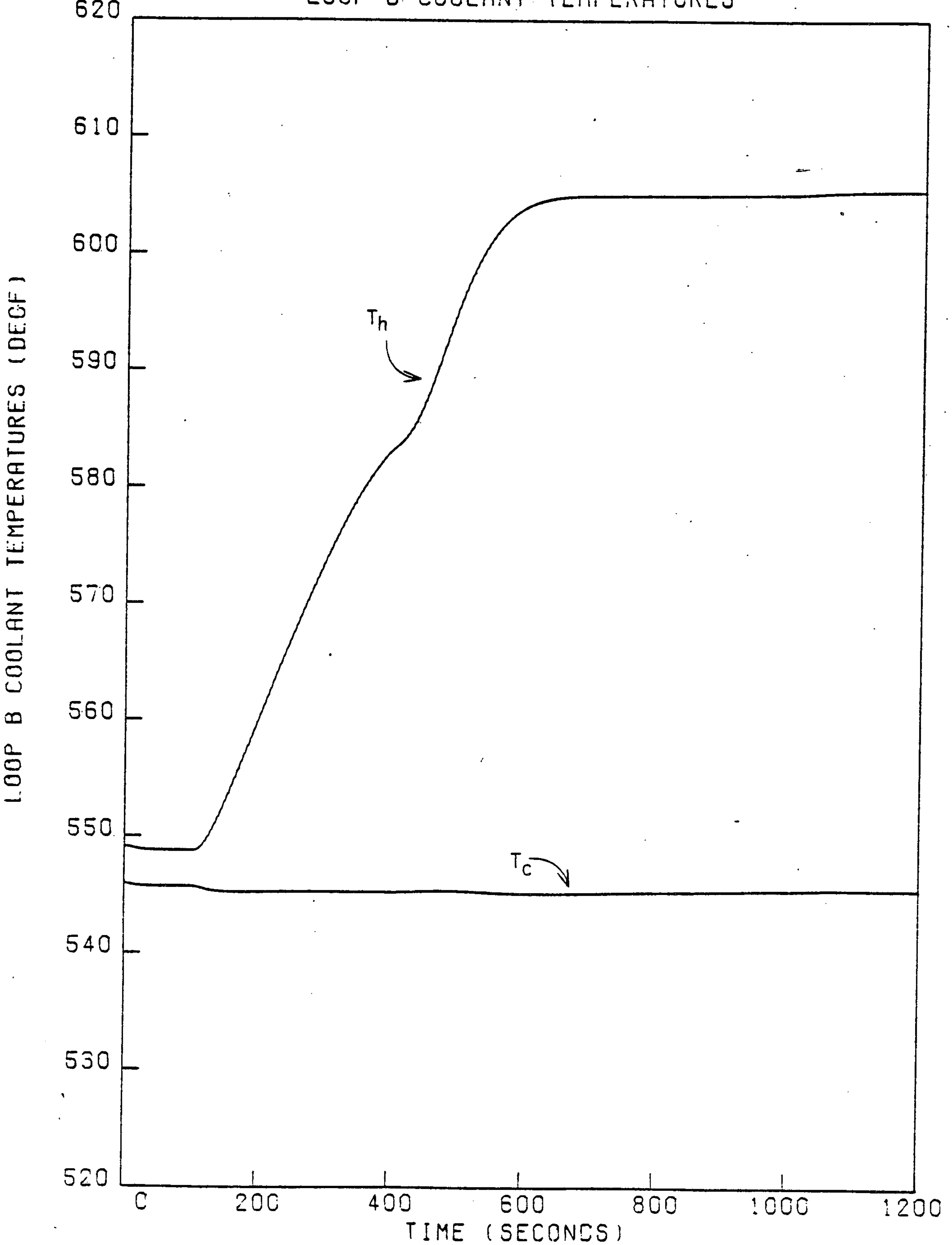


FIGURE 2-31
SONGS 5 PCT PWR NATURAL CIRCULATION
PRESSURIZER PRESSURE

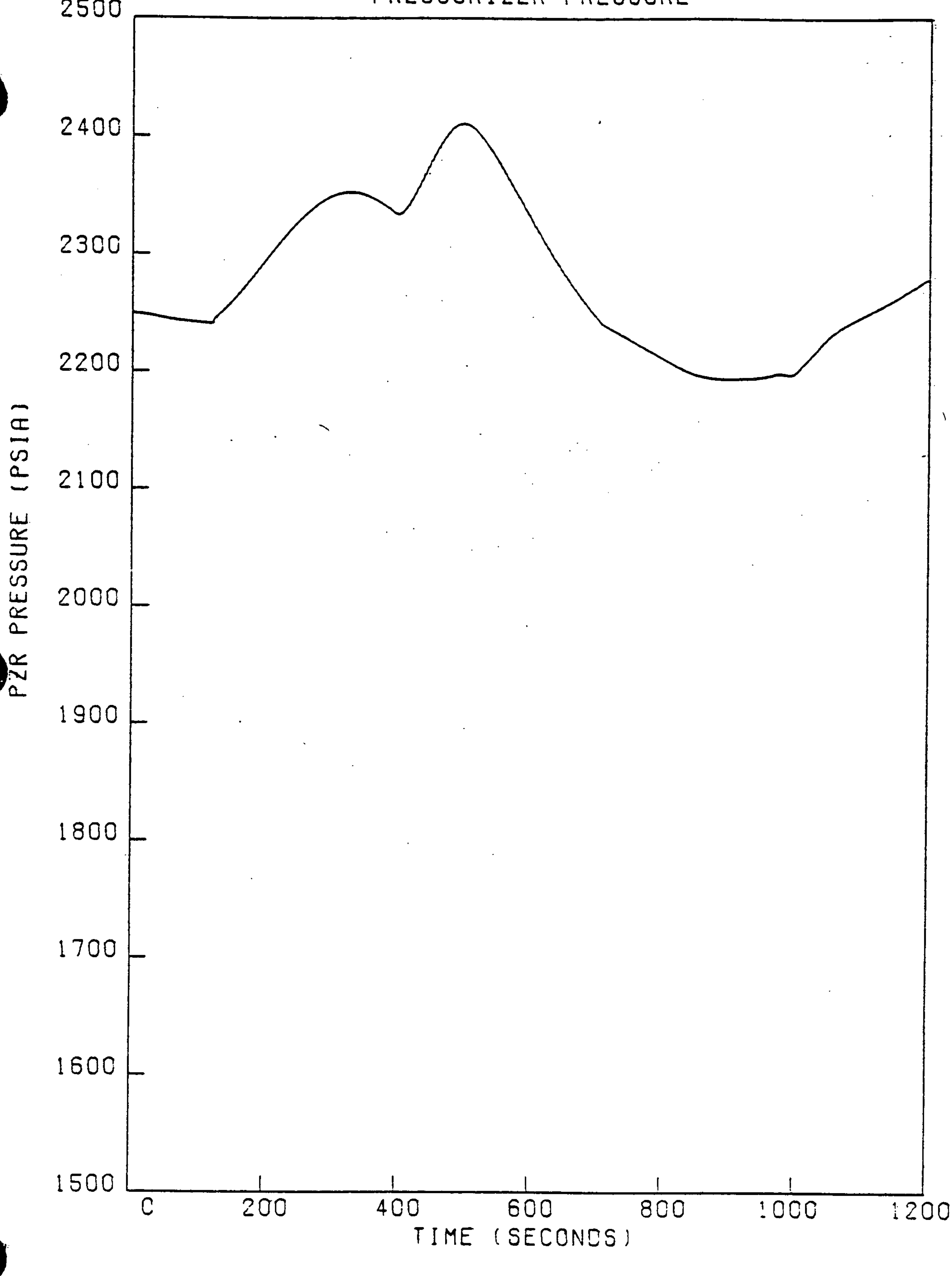


FIGURE 2-32
SONGS 5 PCT PWR NATURAL CIRCULATION
REACTOR COOLANT CORE FLOW

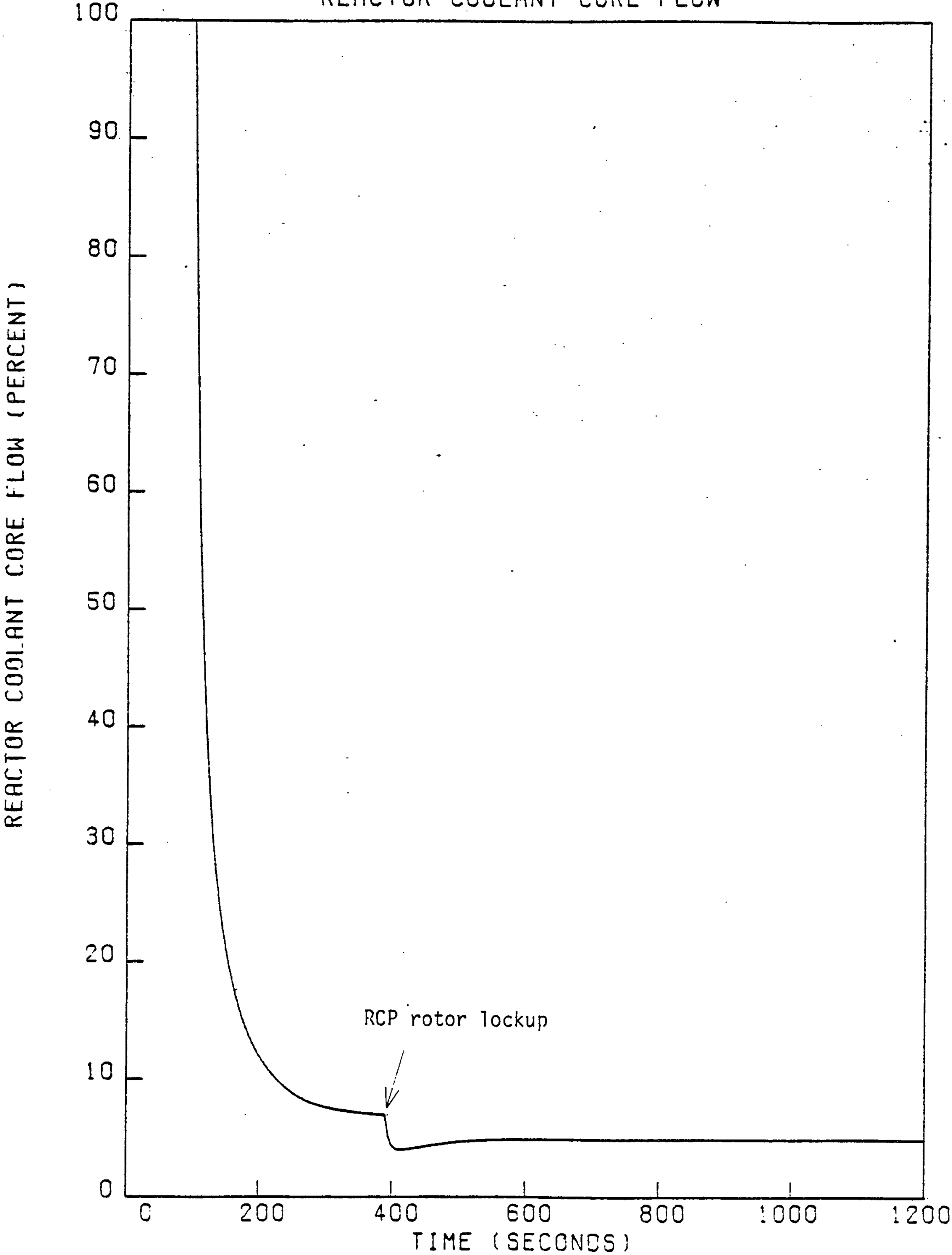


FIGURE 2-33
SONGS 5 PCT PWR NATURAL CIRCULATION
NARROW RANGE CORE FLOW

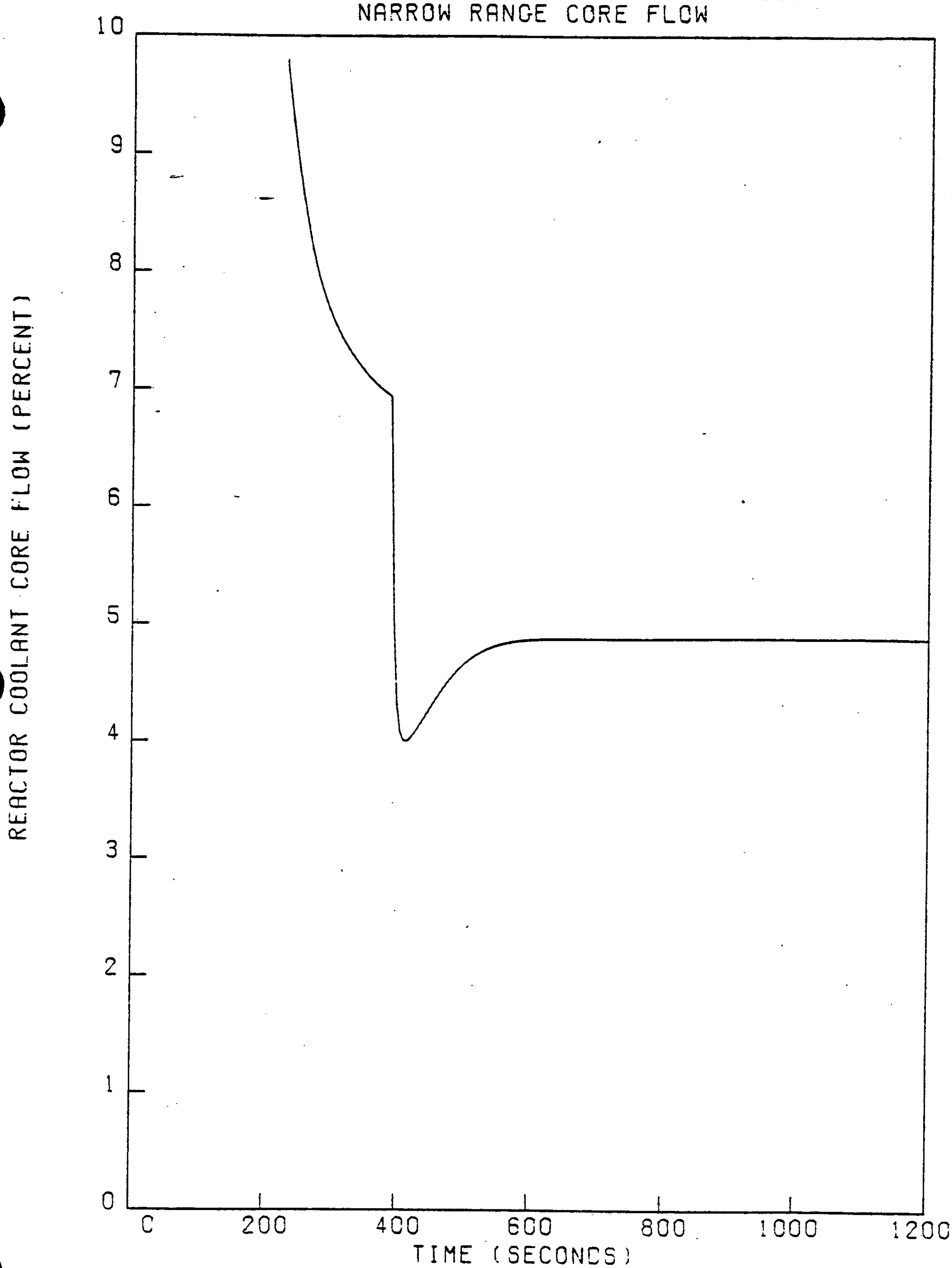


FIGURE 2-34
SONGS 5 PCT PWR NATURAL CIRCULATION
CHARGING FLOW

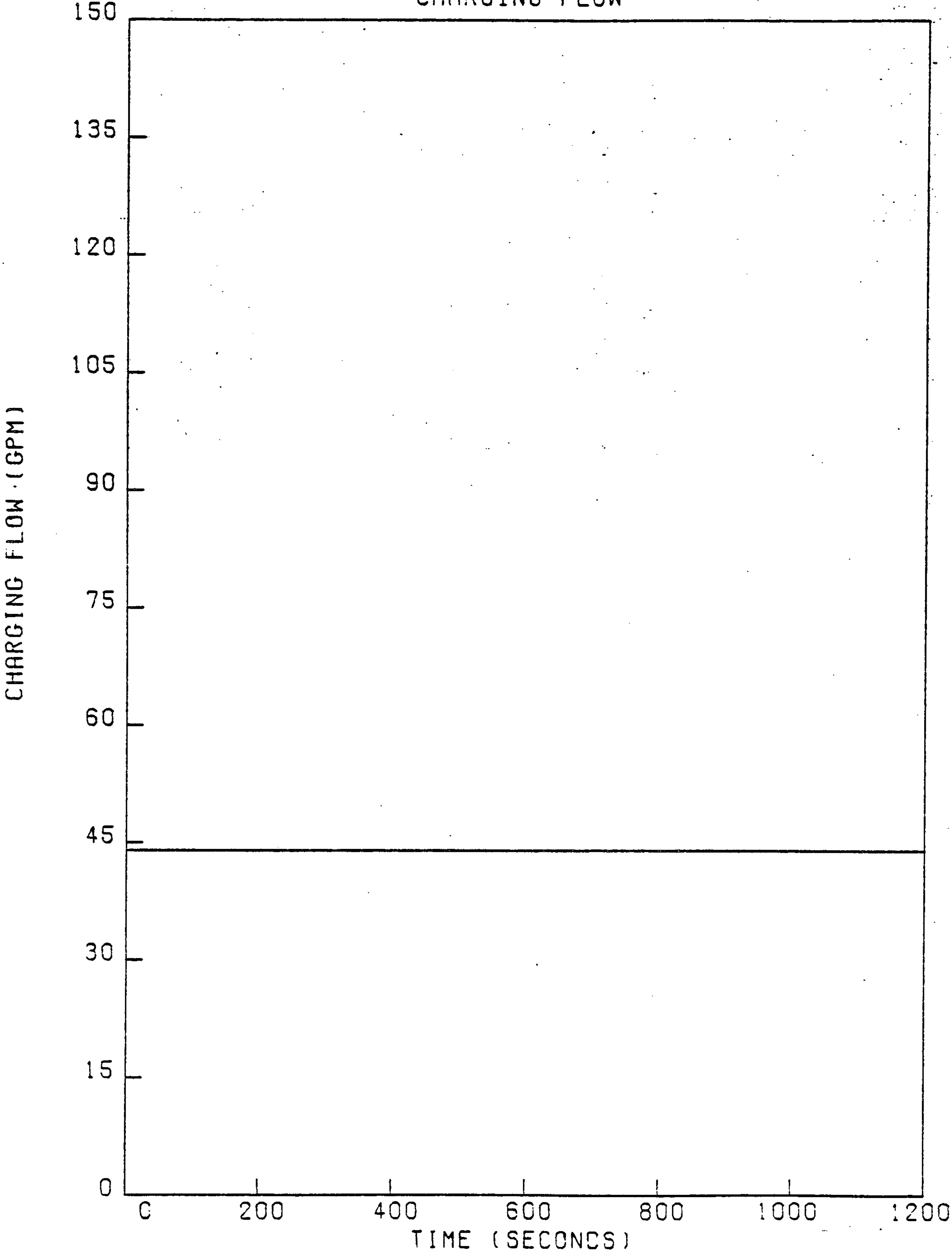


FIGURE 2-35
SONGS 5 PCT PWR NATURAL CIRCULATION
LETDOWN FLOW

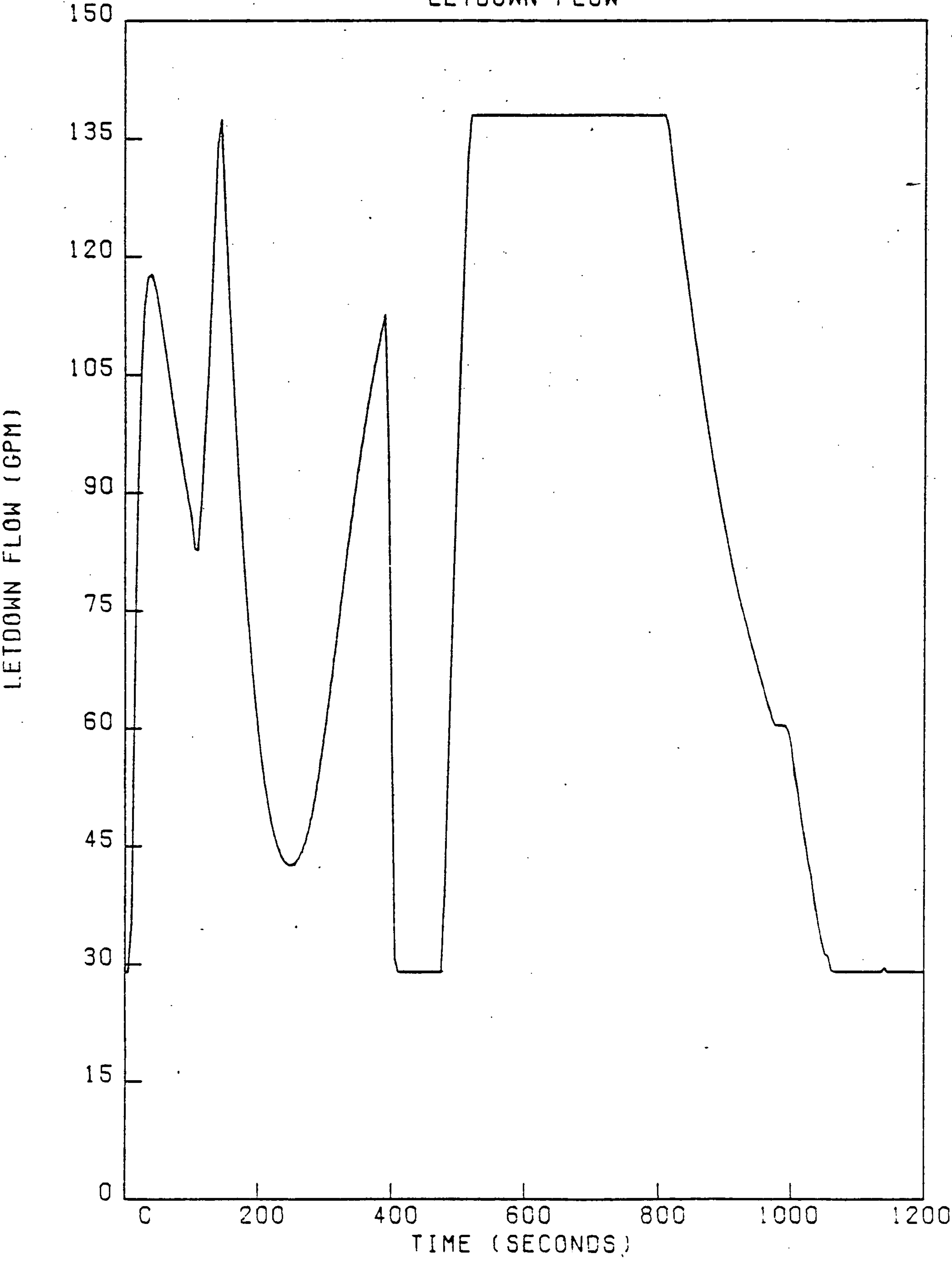


FIGURE 2-36
SONGS 5 PCT PWR NATURAL CIRCULATION
PZR HEATER OUTPUTS

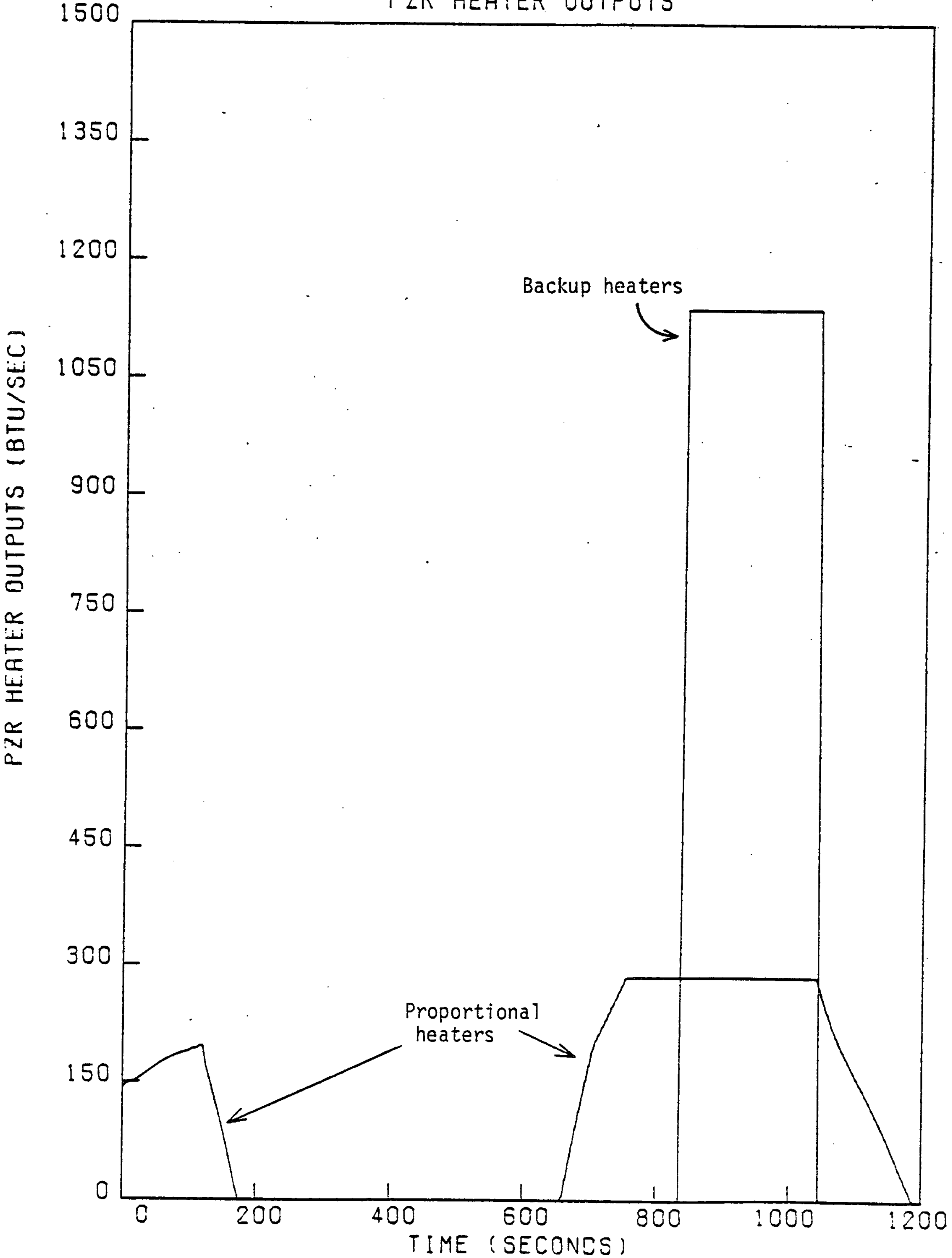


FIGURE 2-37
SONGS NAT CIRC AT REDUCED PRESSURE
NARROW RANGE CORE POWER

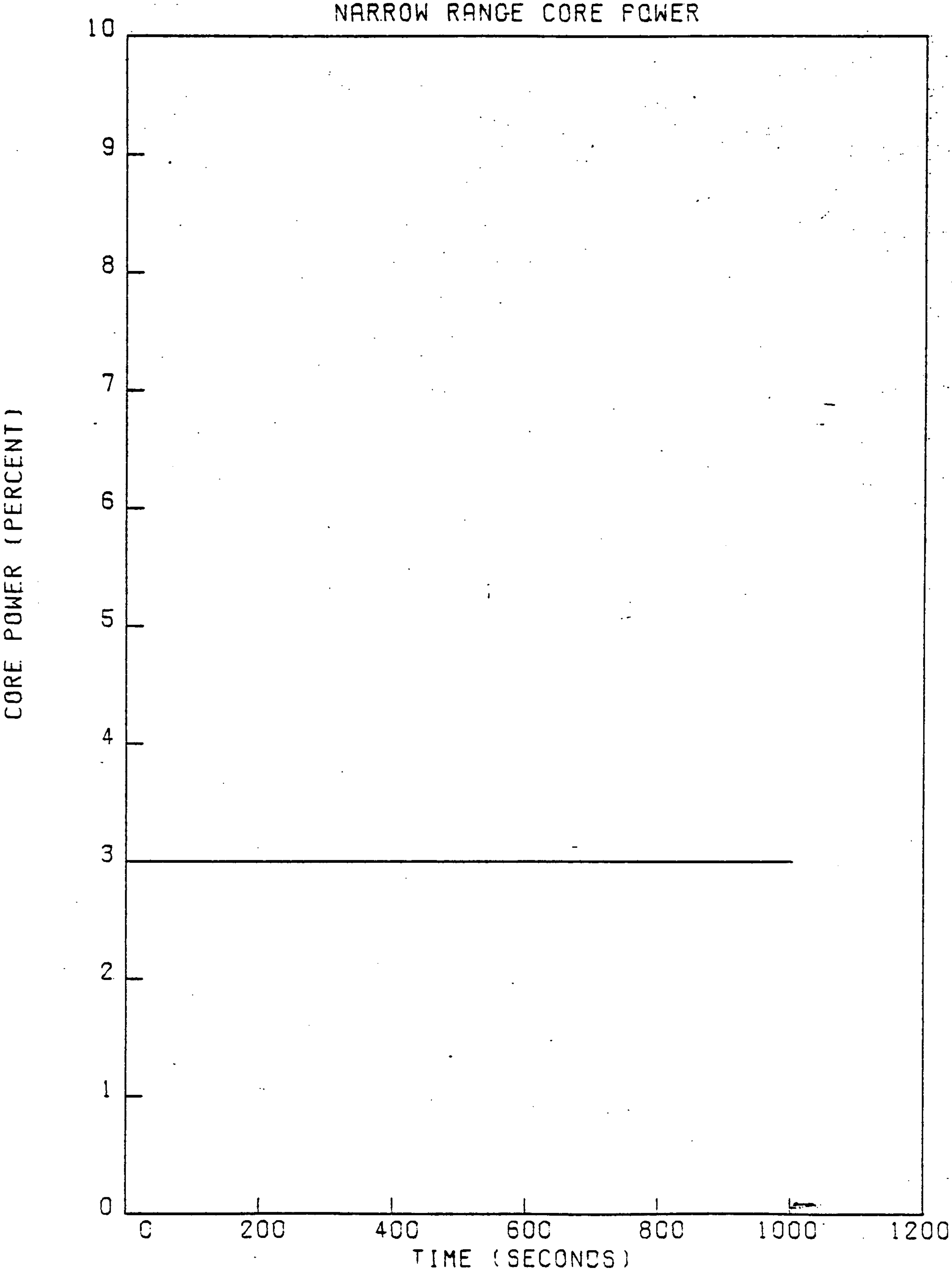


FIGURE 2-38
SONGS NAT CIRC AT REDUCED PRESSURE
S.G. A PRESSURE

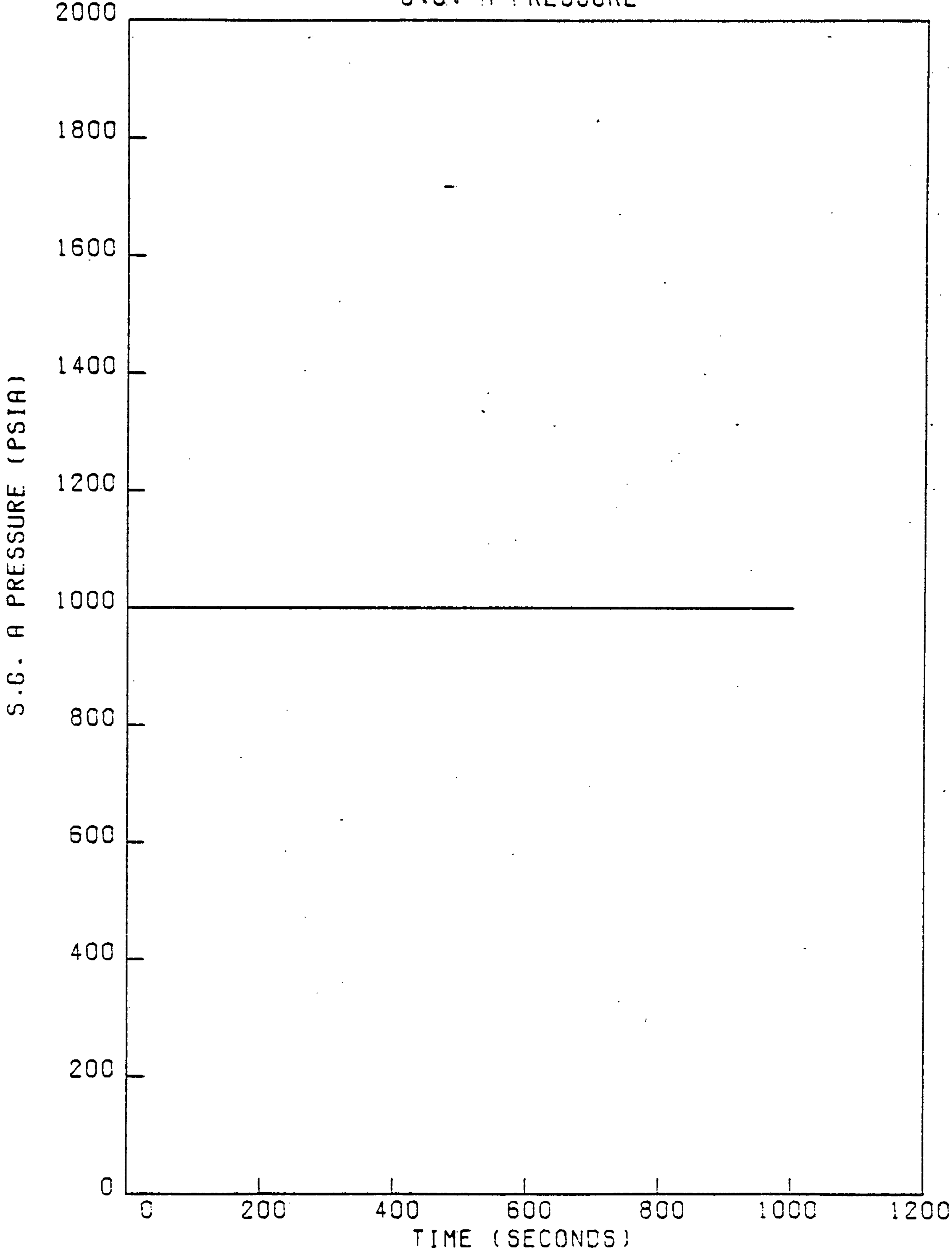


FIGURE 2-39
SONGS NAT CIRC AT REDUCED PRESSURE
S.G. B PRESSURE

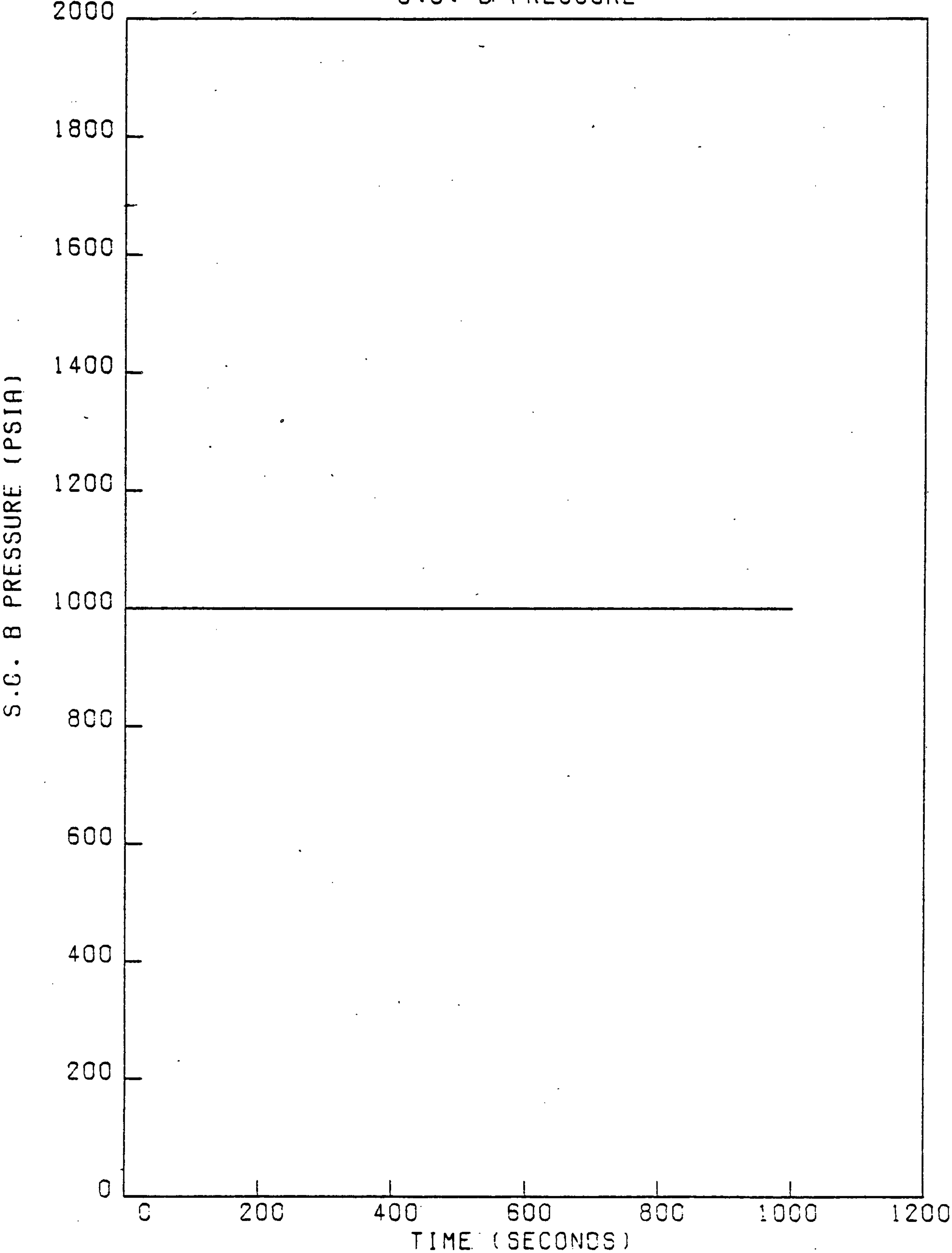


FIGURE 2-40
SONGS NAT CIRC AT REDUCED PRESSURE
PZR LEVEL

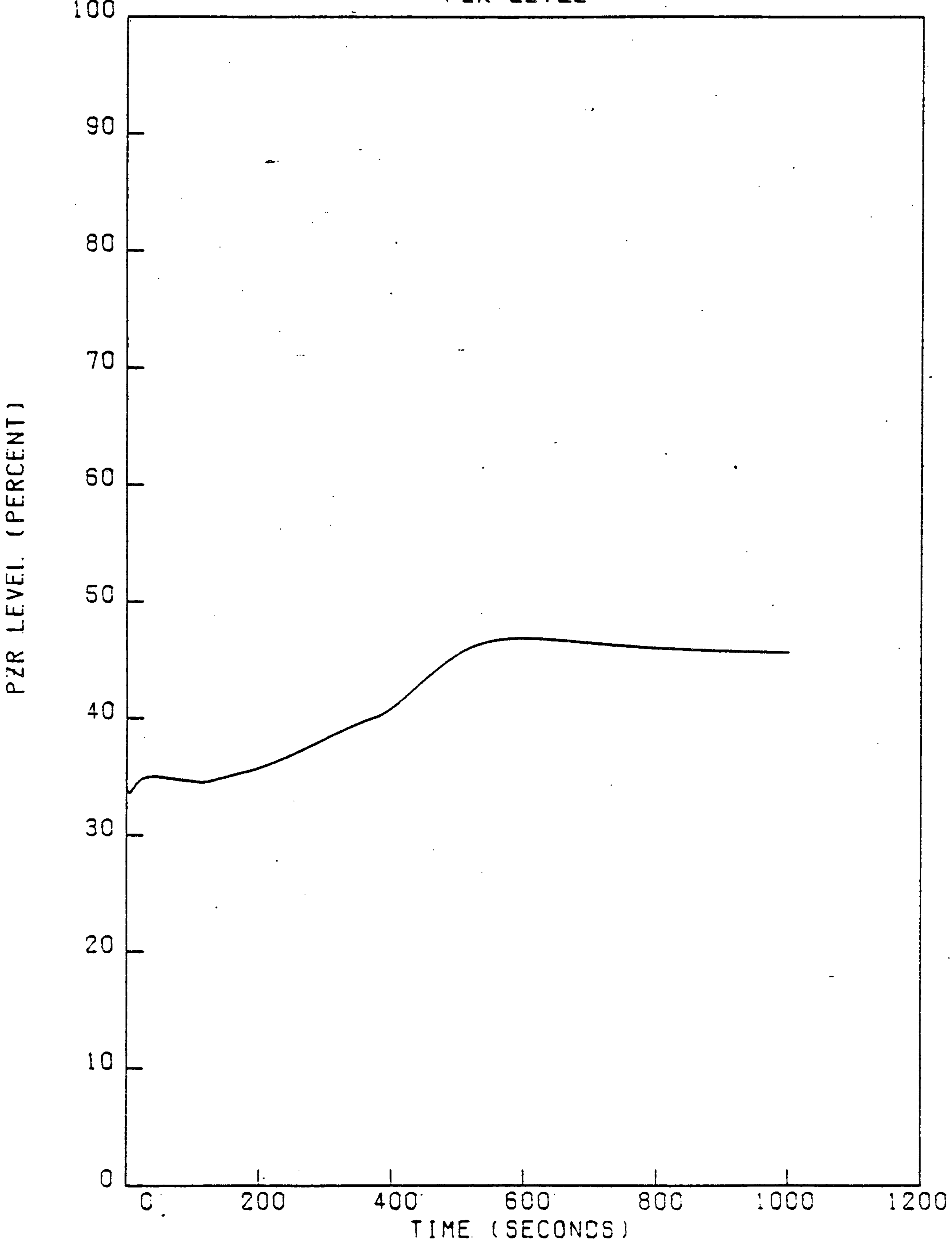


FIGURE 2-41
SONGS NAT CIRC AT REDUCED PRESSURE
LOOP A COOLANT TEMPERATURES

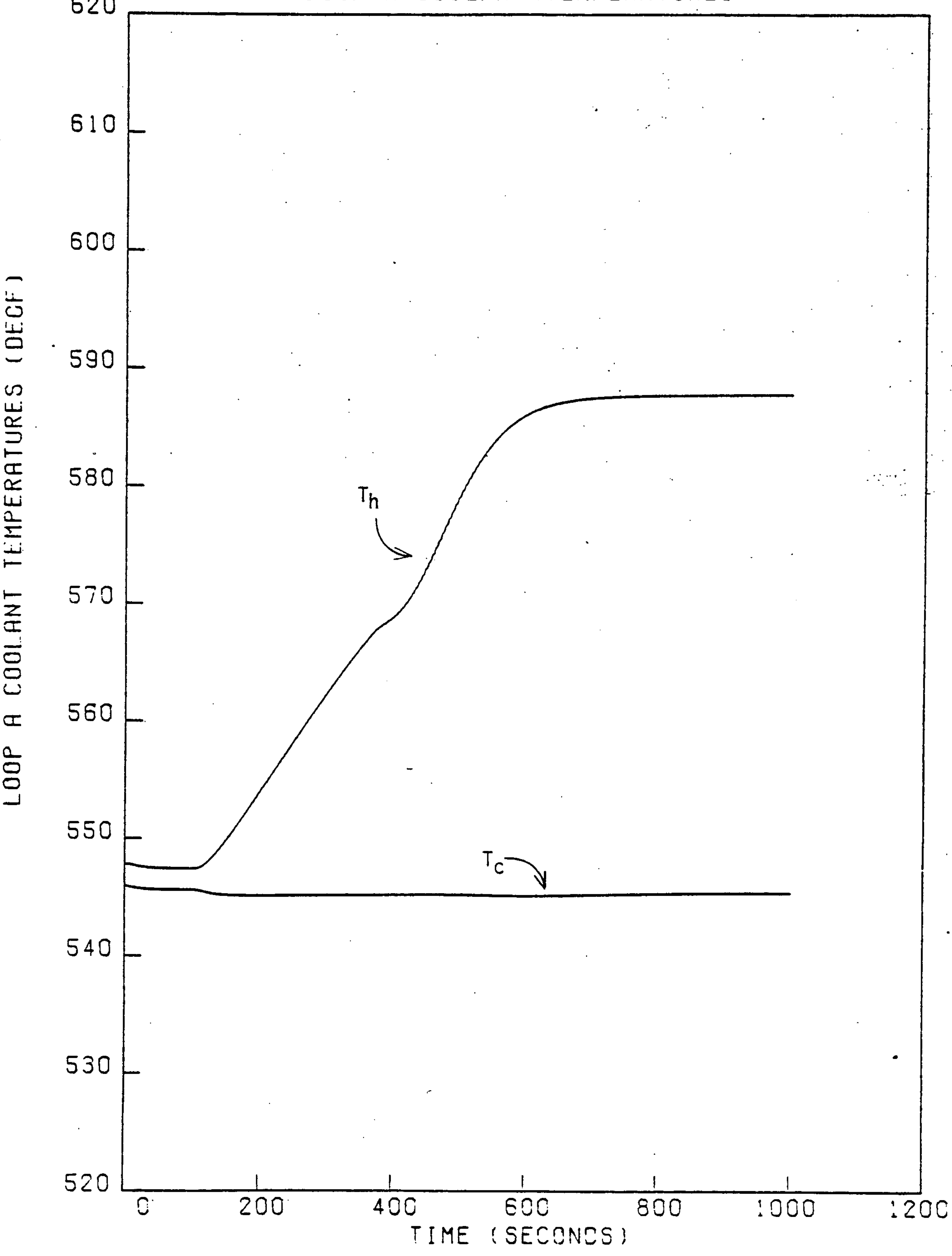


FIGURE 2-42
SONGS NAT CIRC AT REDUCED PRESSURE
LOOP B COOLANT TEMPERATURES

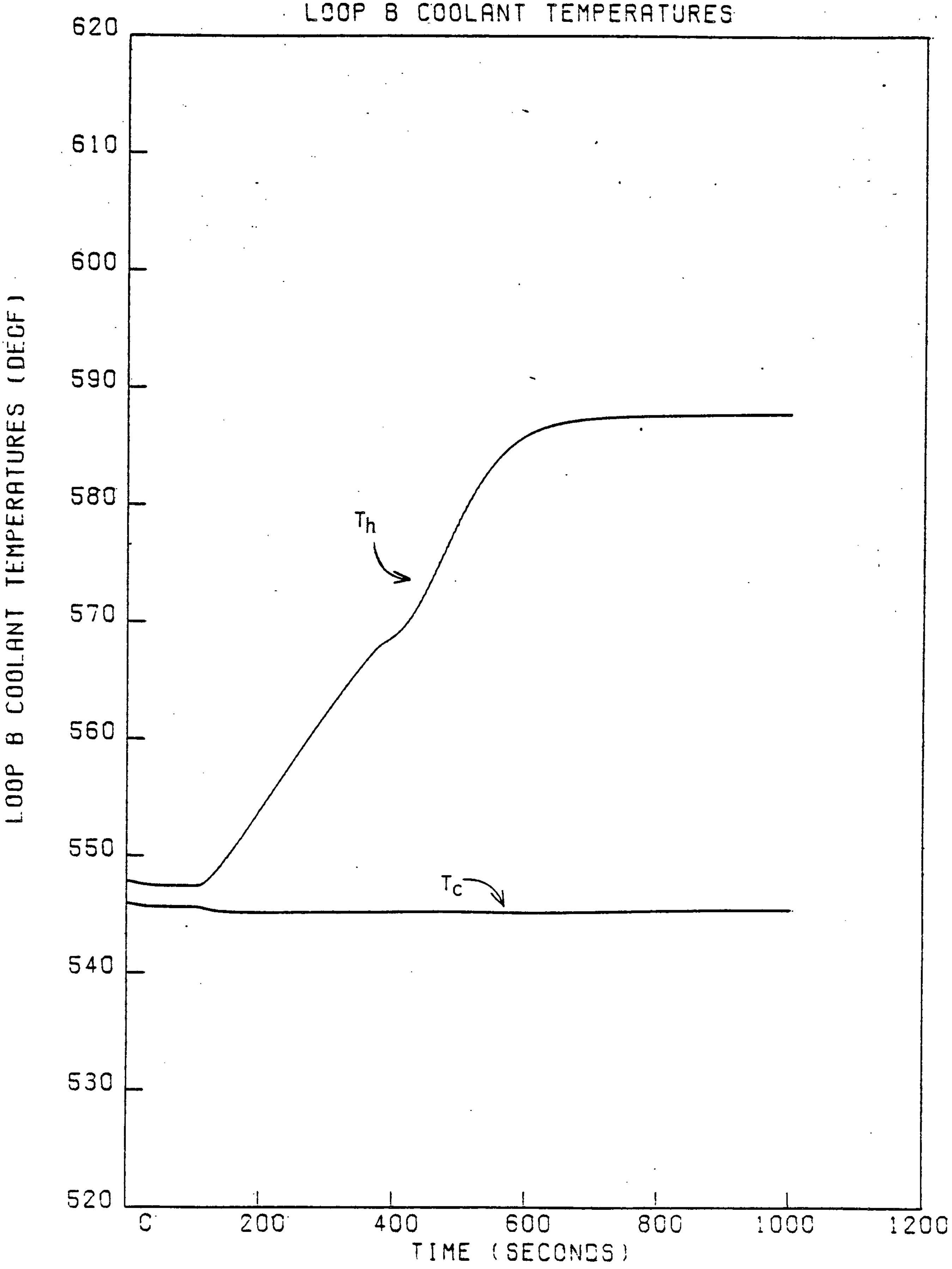


FIGURE 2-43
SONGS NAT CIRC AT REDUCED PRESSURE
PRESSURIZER PRESSURE

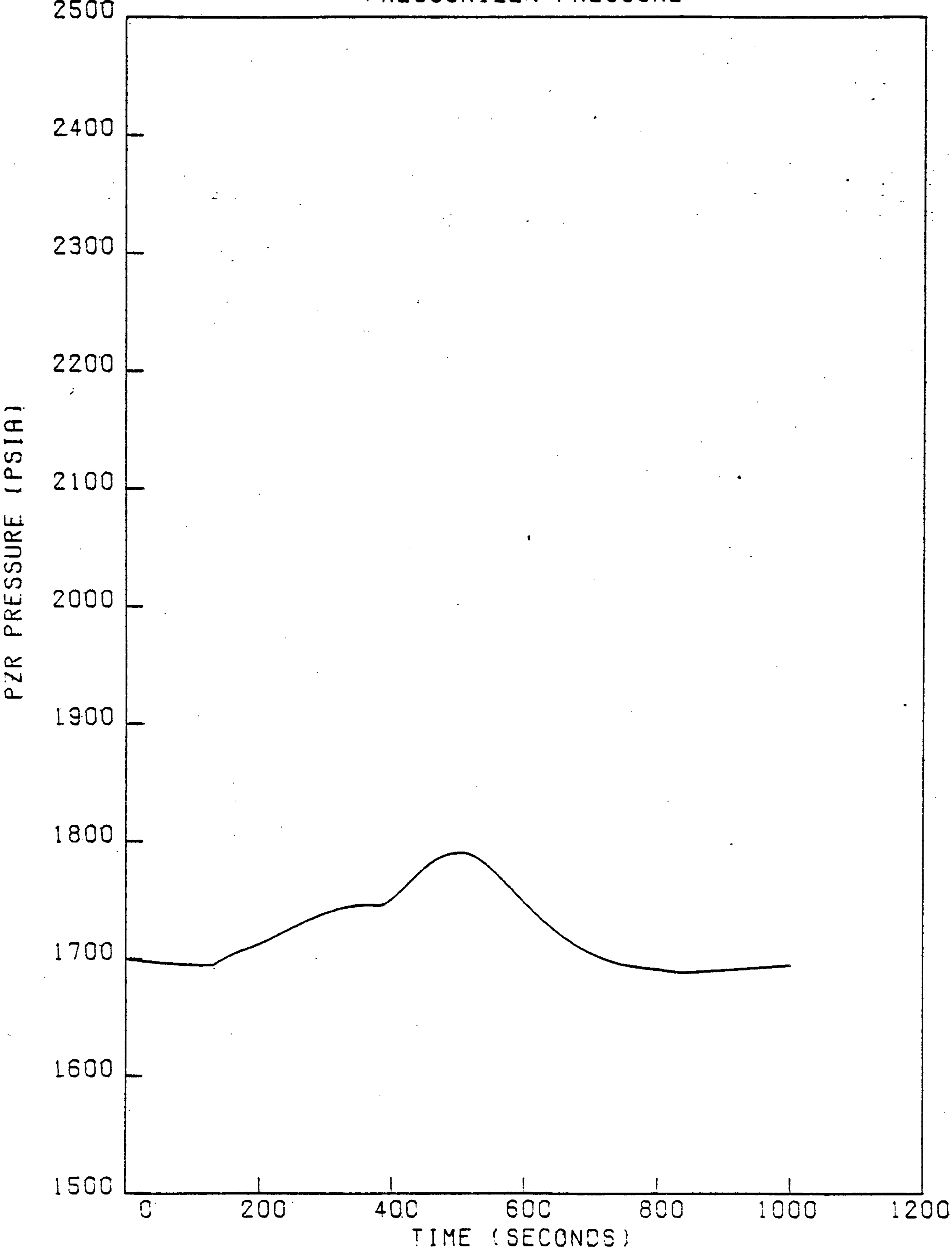


FIGURE 2-44
SONGS NAT CIRC AT REDUCED PRESSURE
REACTOR COOLANT CORE FLOW

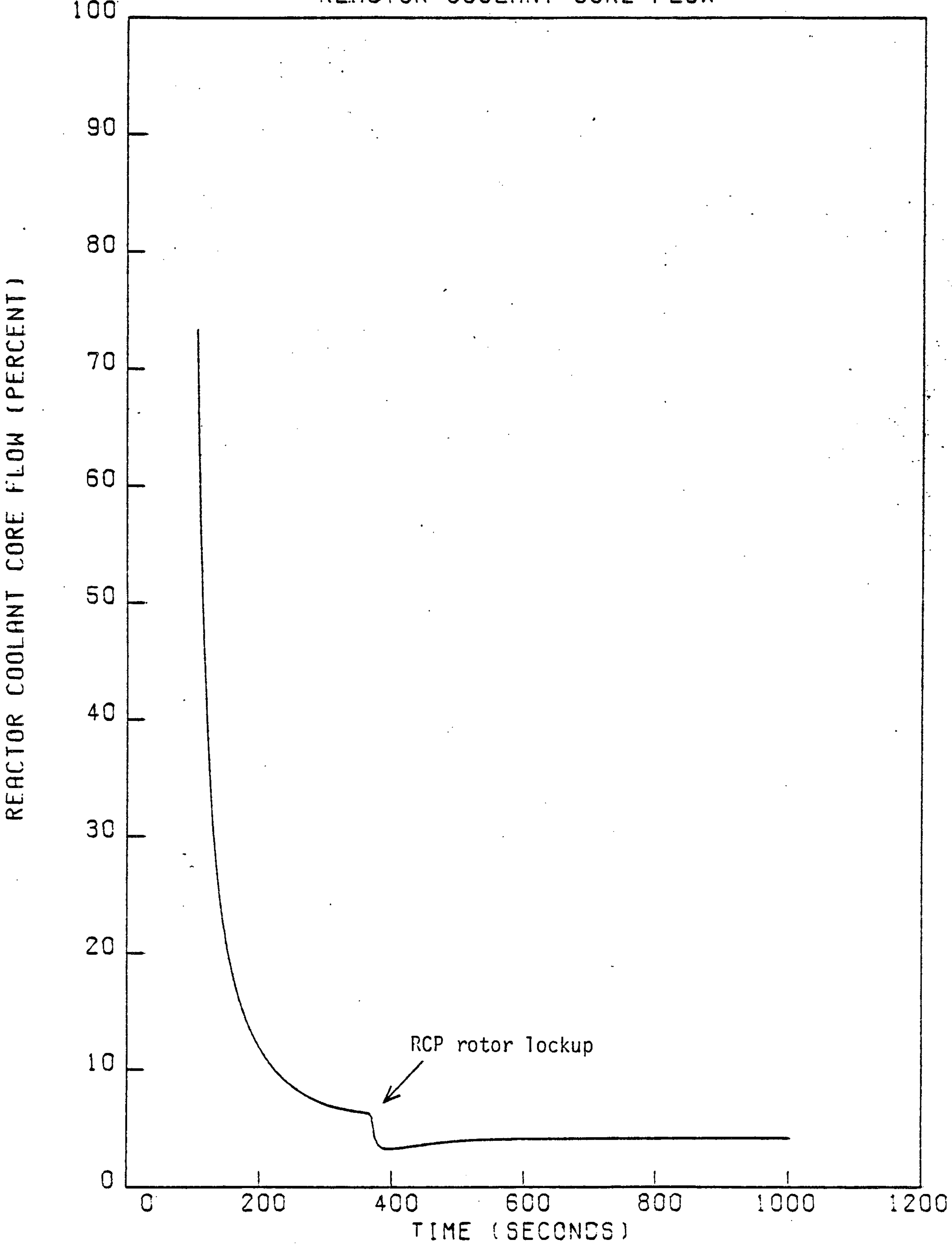


FIGURE 2-45
SONGS NAT CIRC AT REDUCED PRESSURE
NARROW RANGE CORE FLOW

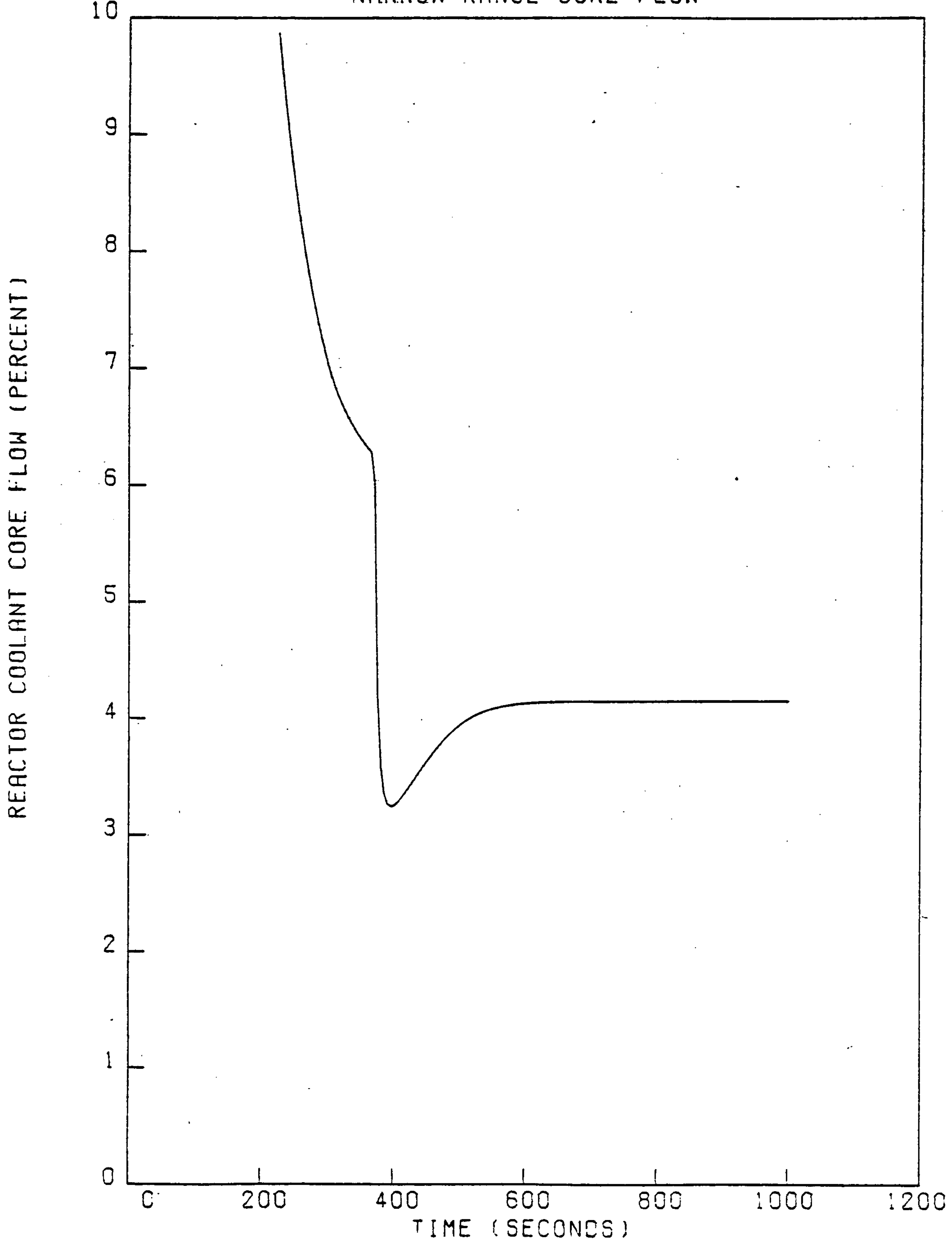


FIGURE 2-46
SONGS LOW PWR ASYMMETRIC NAT CIRC
NARROW RANGE CORE POWER

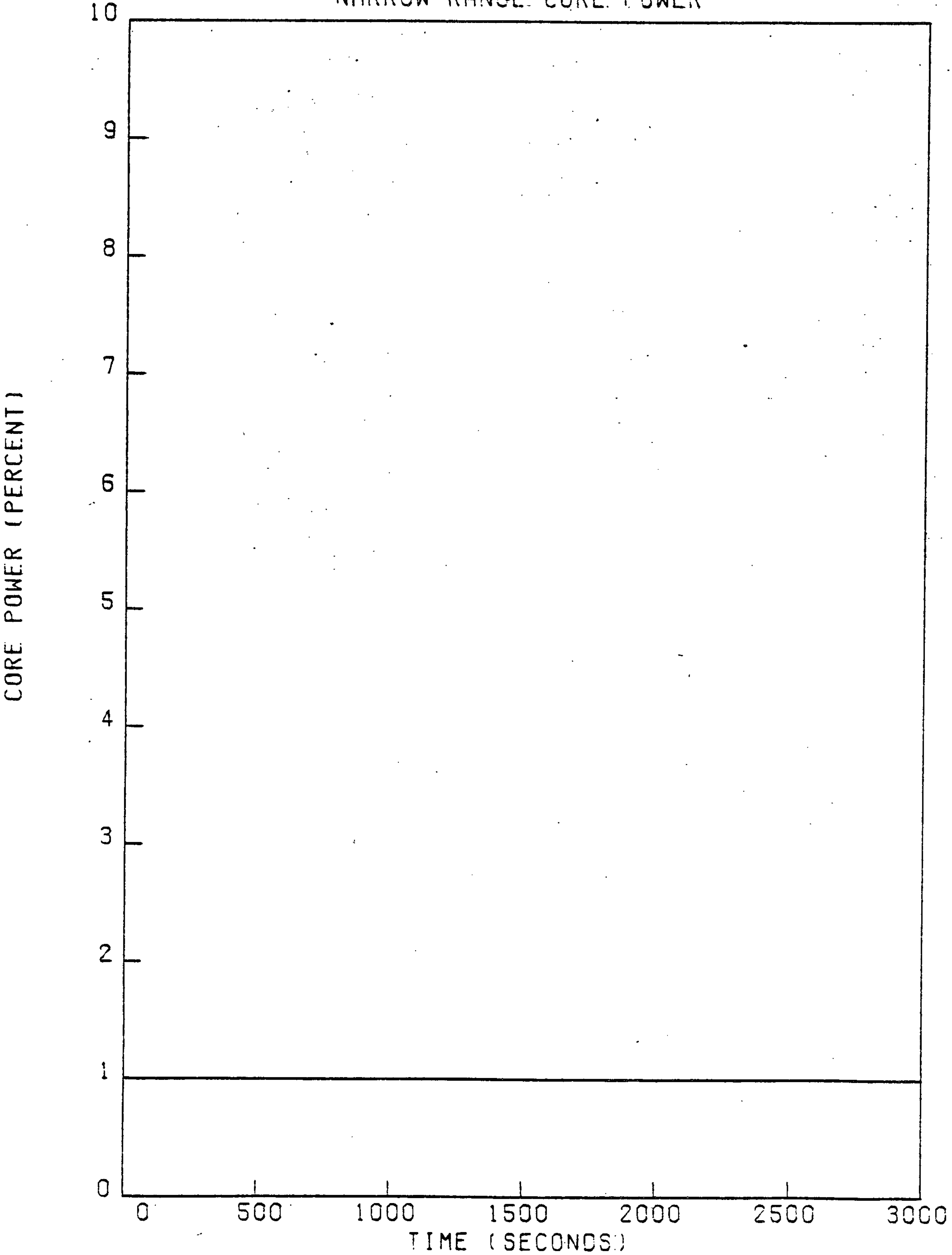


FIGURE 2-47
SONGS LOW PWR ASYMMETRIC NAT CIRC
S.G. A PRESSURE

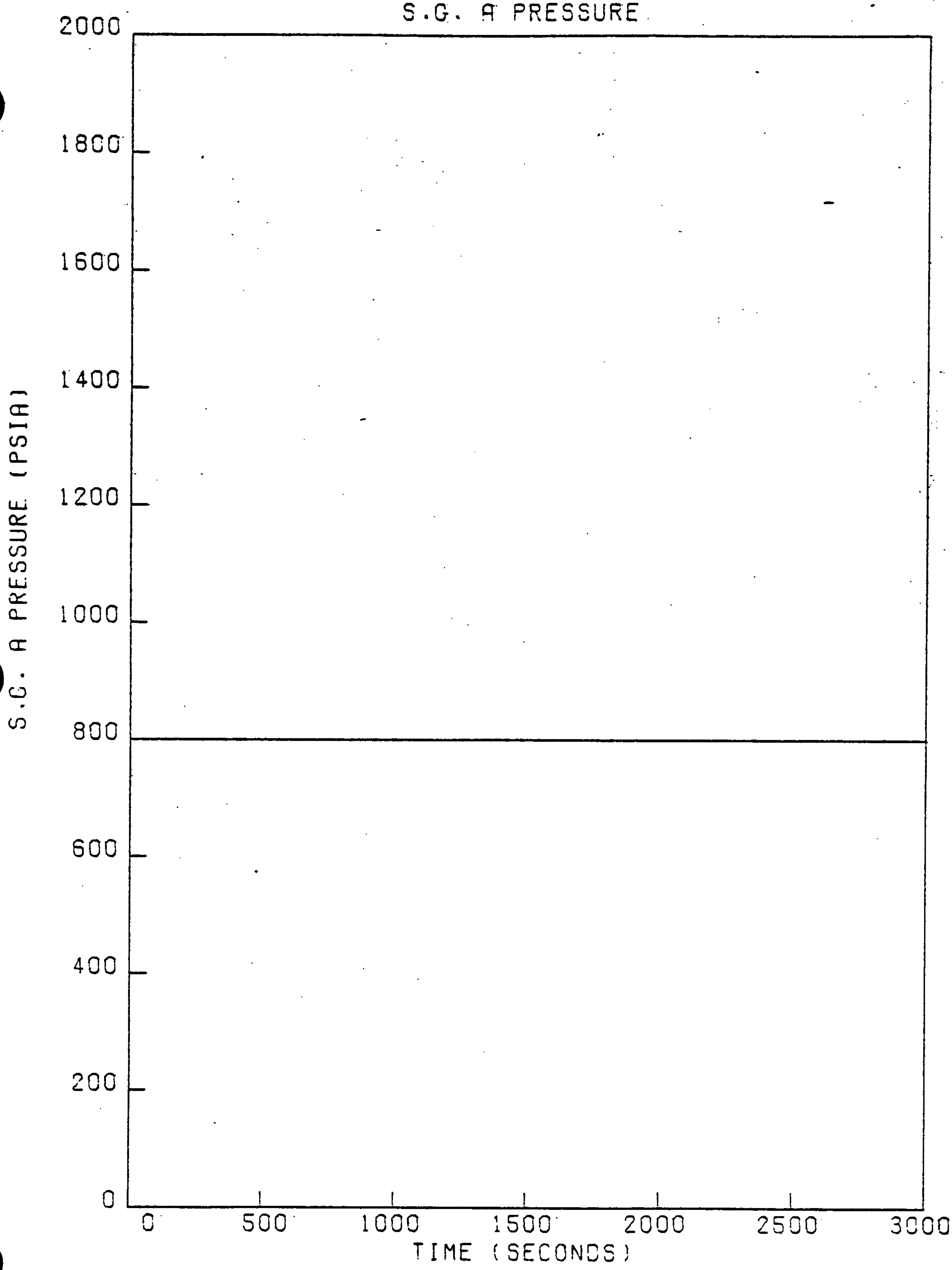


FIGURE 2-48
SONGS LOW PWR ASYMMETRIC NAT CIRC
S.G. B PRESSURE

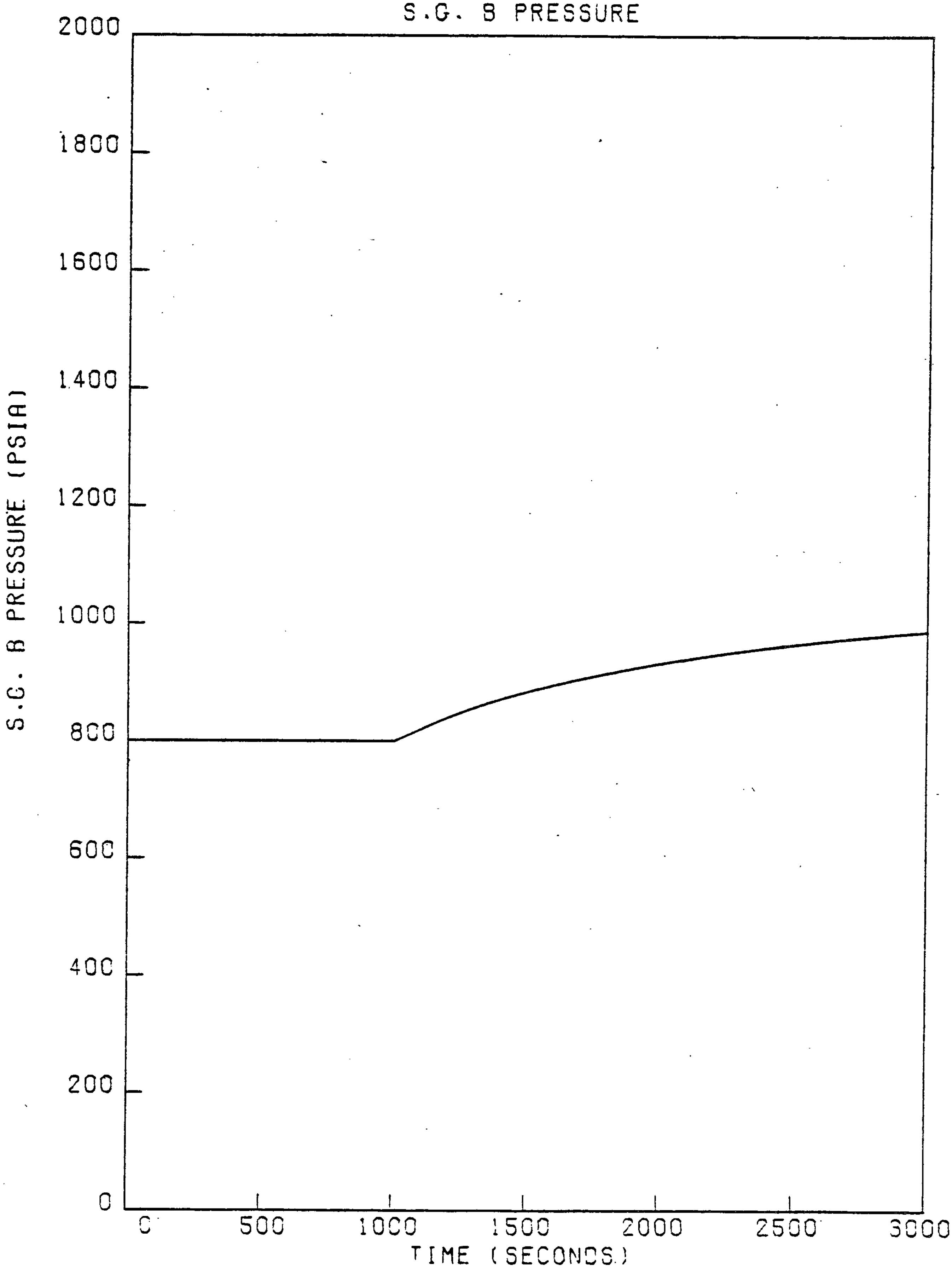


FIGURE 2-49
SONGS LOW PWR ASYMMETRIC NAT CIRC
PZR LEVEL

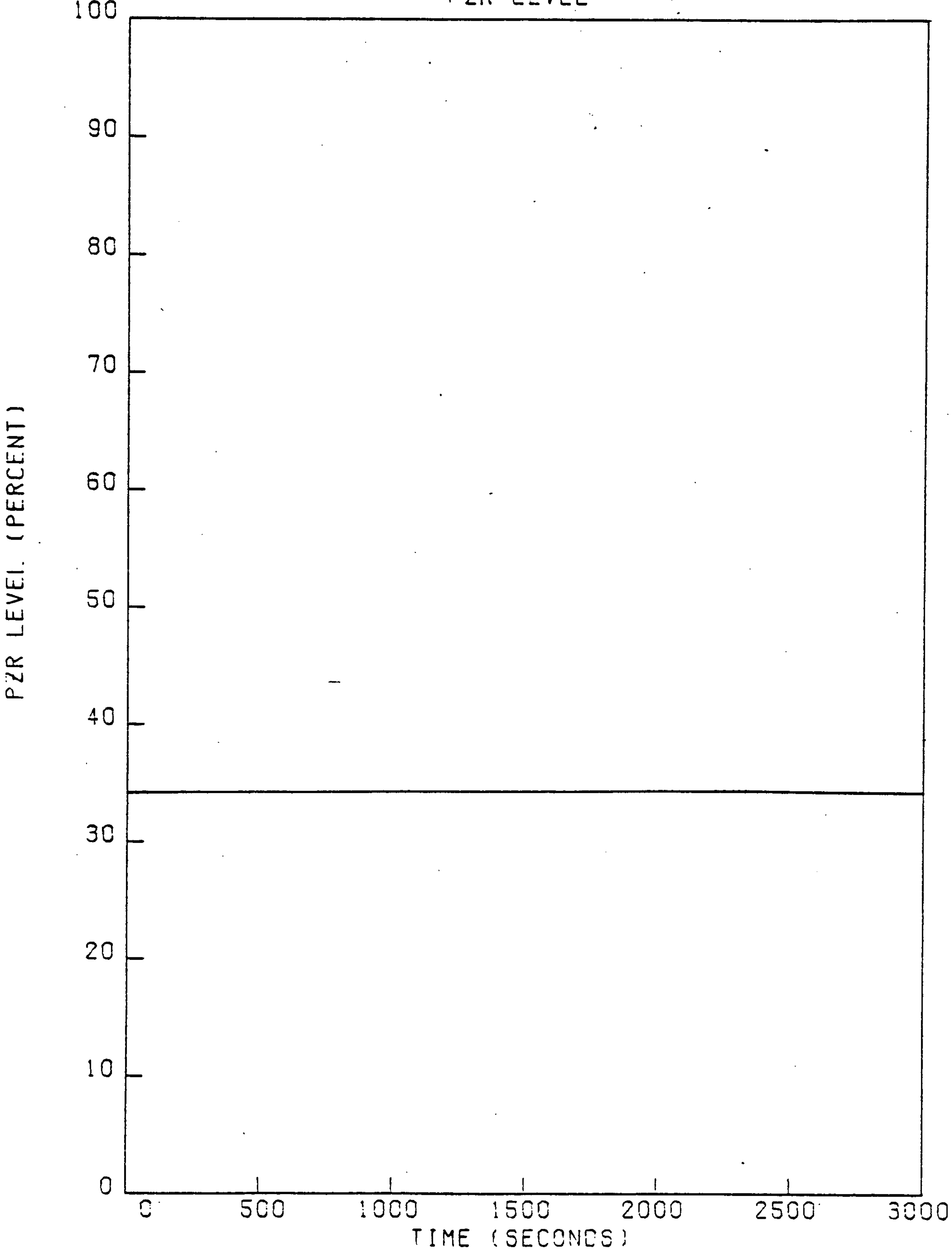


FIGURE 2-50
SONGS LOW PWR ASYMMETRIC NAT CIRC
LOOP A COOLANT TEMPERATURES

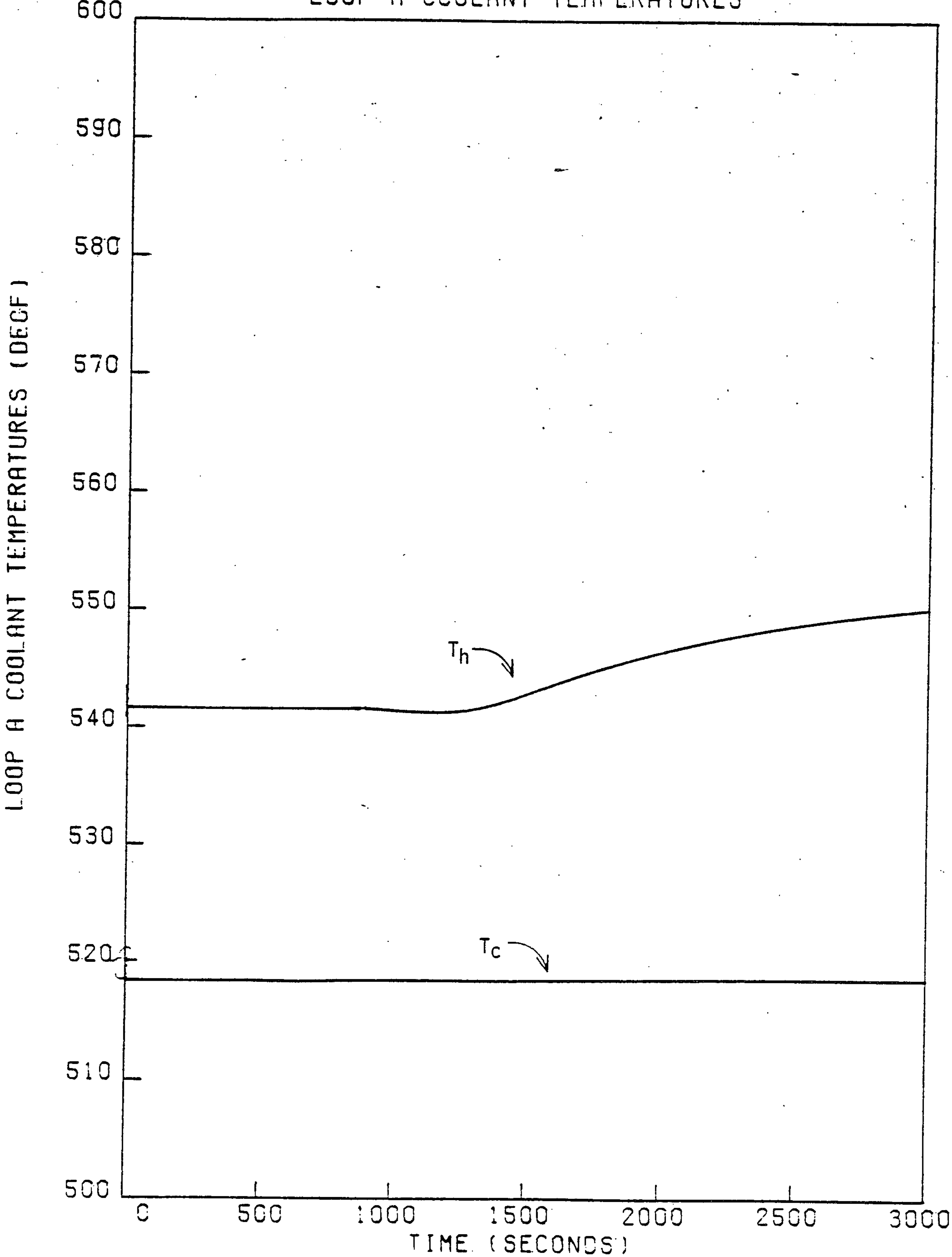


FIGURE 2-51
SONGS LOW PWR ASYMMETRIC NAT CIRC
LOOP B COOLANT TEMPERATURES

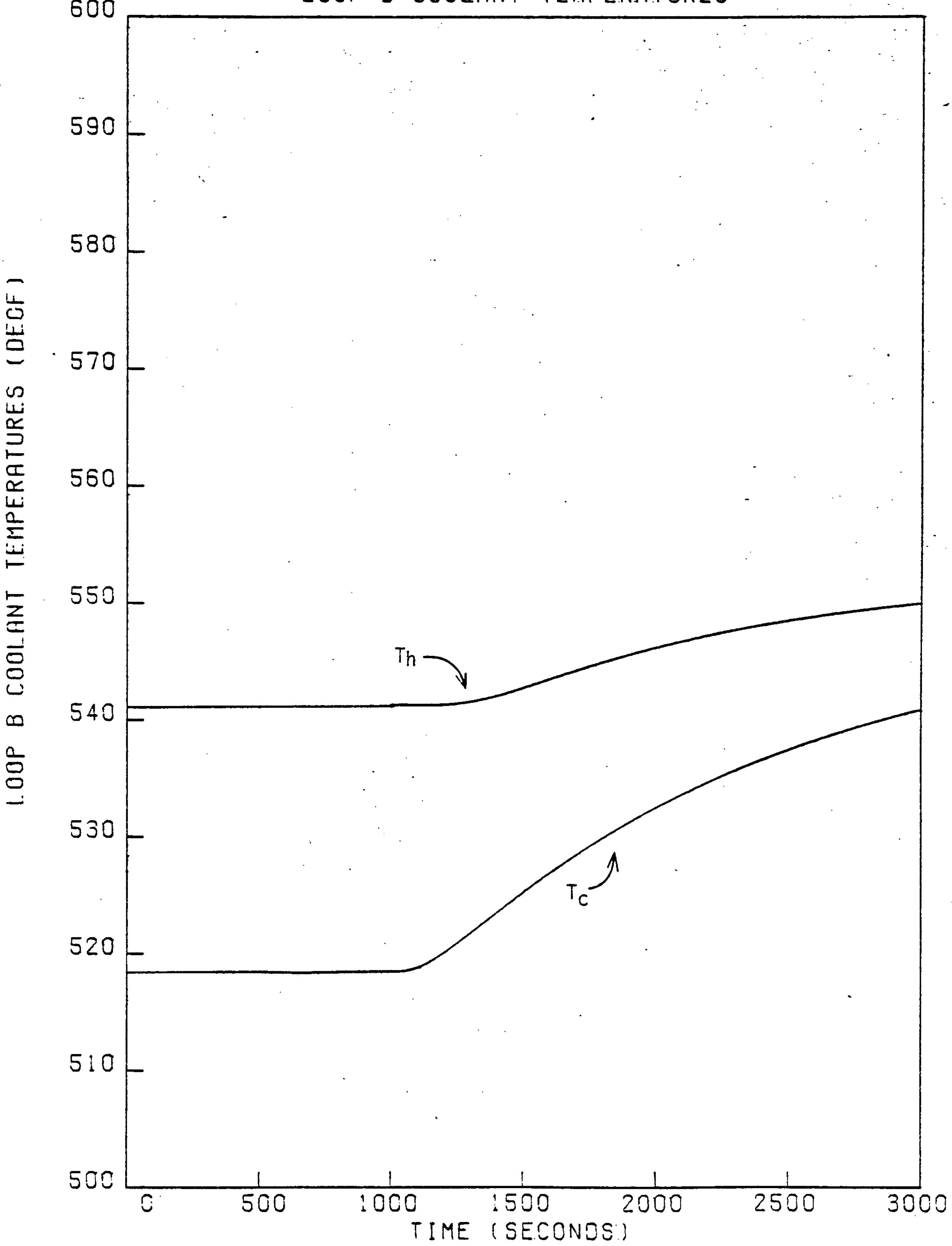


FIGURE 2-52
SONGS LOW PWR ASYMMETRIC NAT CIRC
PRESSURIZER PRESSURE

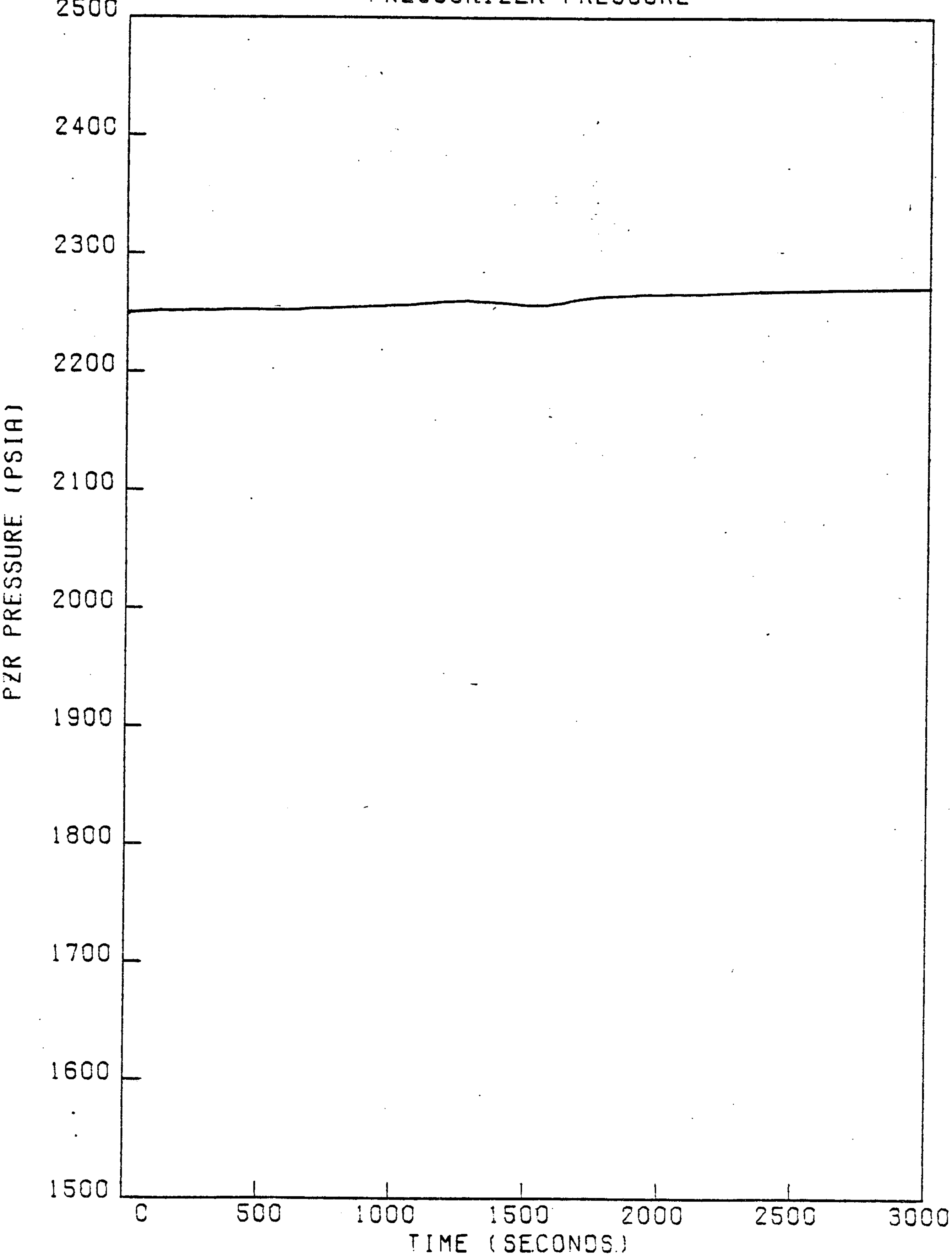


FIGURE 2-53
SONGS LOW PWR ASYMMETRIC NAT CIRC
NARROW RANGE CORE FLOW

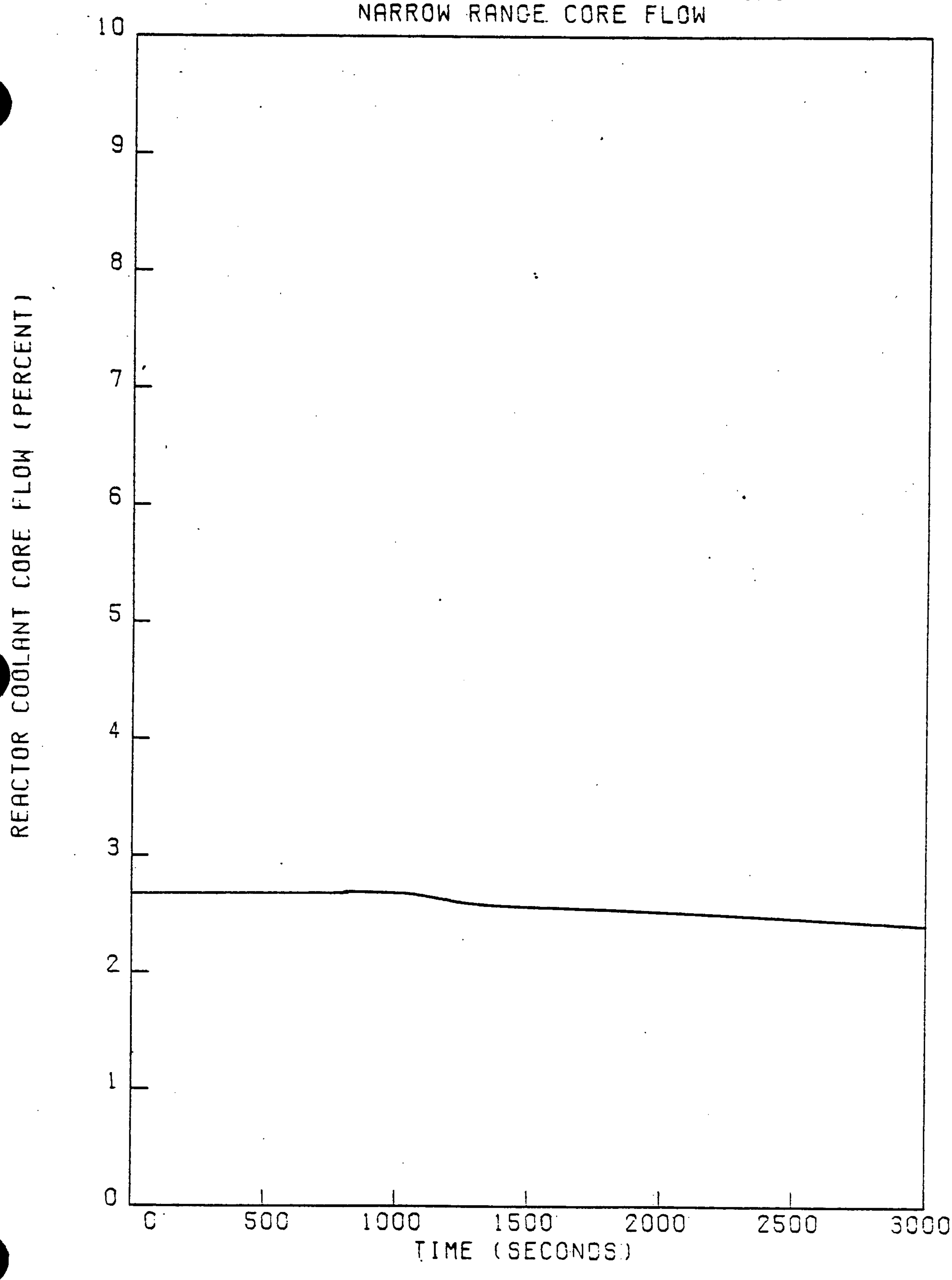


FIGURE 2-54
SONGS LOW PWR ASYMMETRIC NAT CIRC
CHARGING FLOW

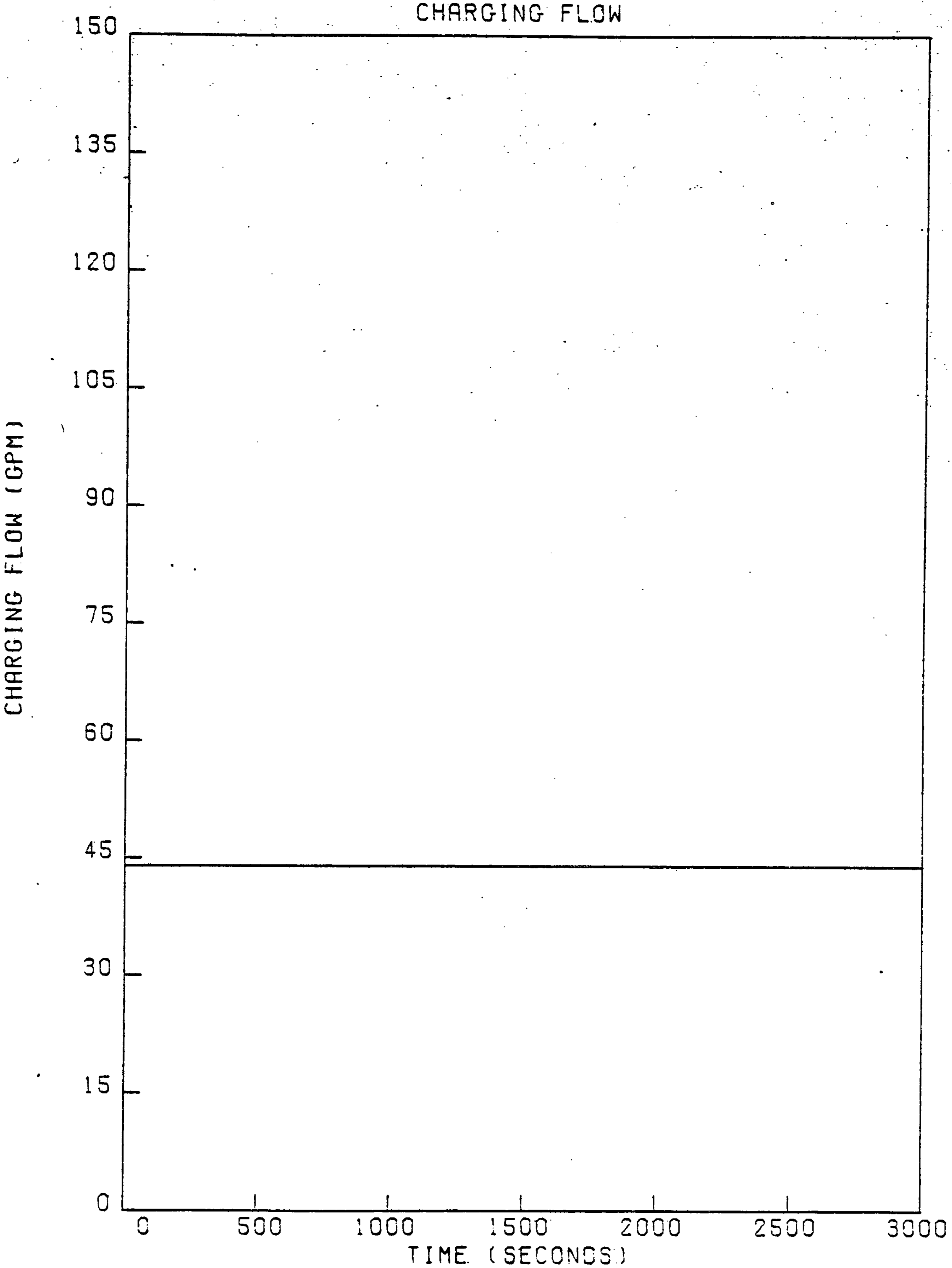


FIGURE 2-55
SONGS LOW PWR ASYMMETRIC NAT CIRC
LETDOWN FLOW

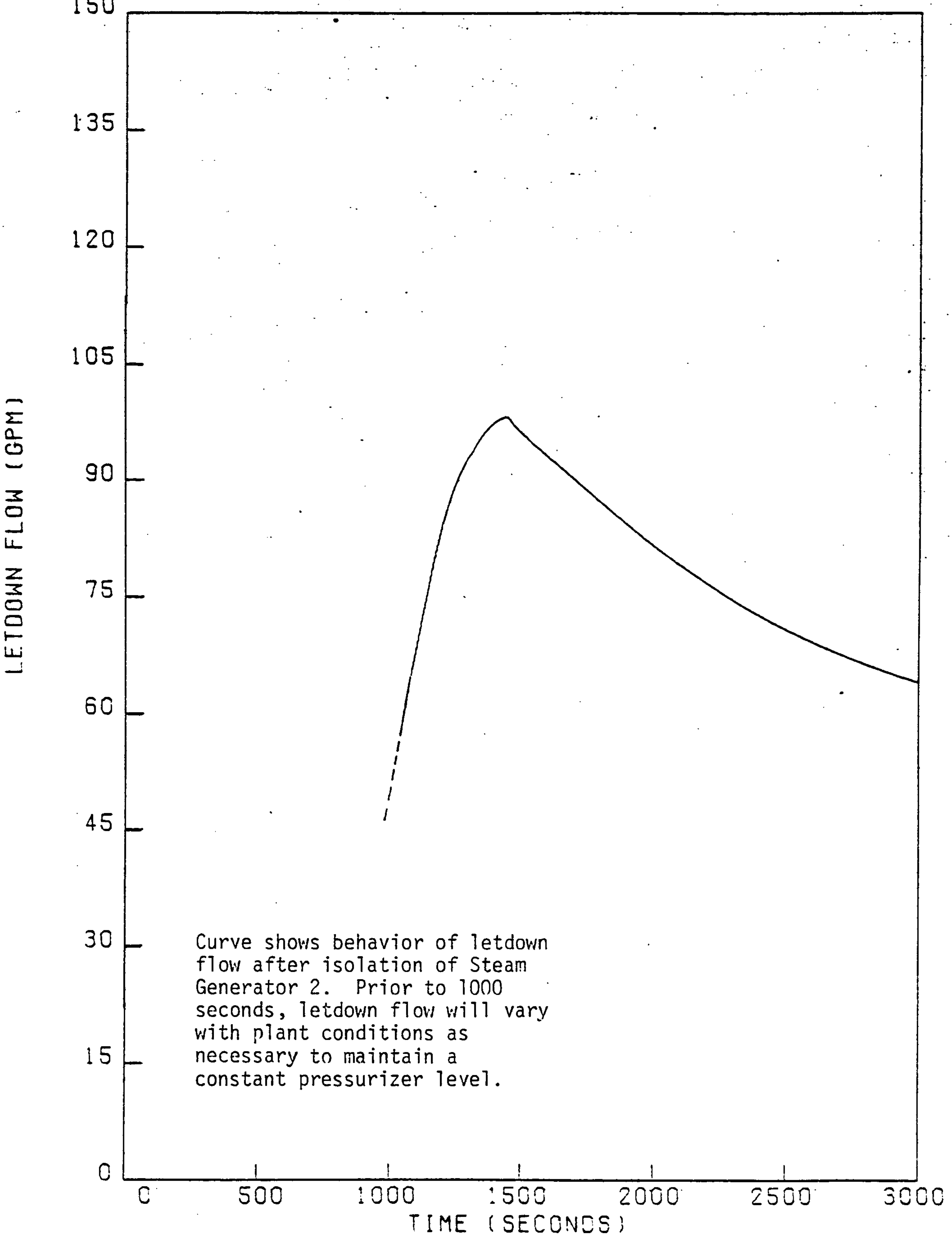


FIGURE 2-56
SONGS 20 PCT PWR LOSS OF AC
POST TRIP CORE DECAY HEAT

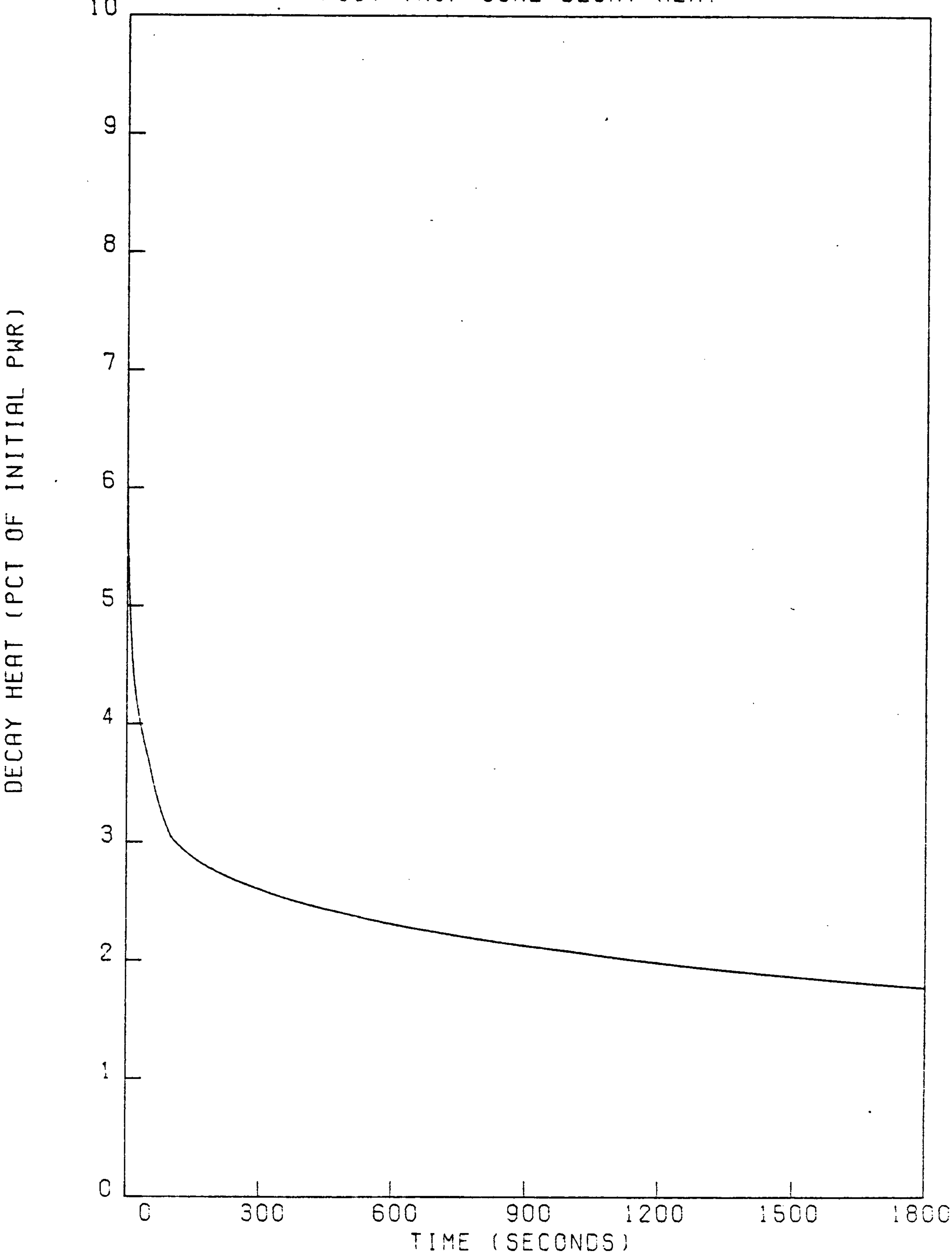


FIGURE 2-57
SONGS 20 PCT PWR LOSS OF AC
PRESSURIZER PRESSURE

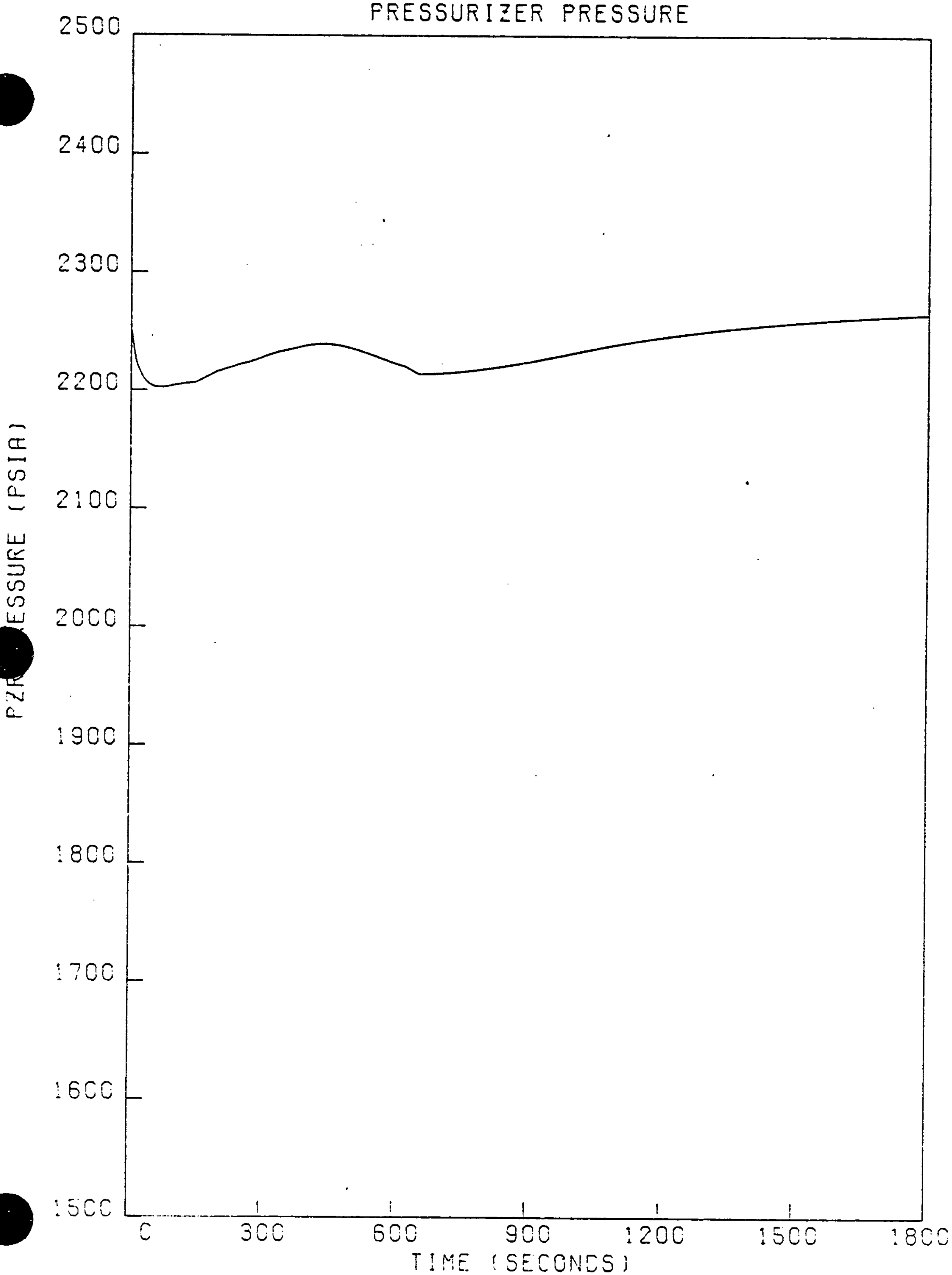


FIGURE 2-58
SONGS 20 PCT PWR LOSS OF AC
PZR LEVEL

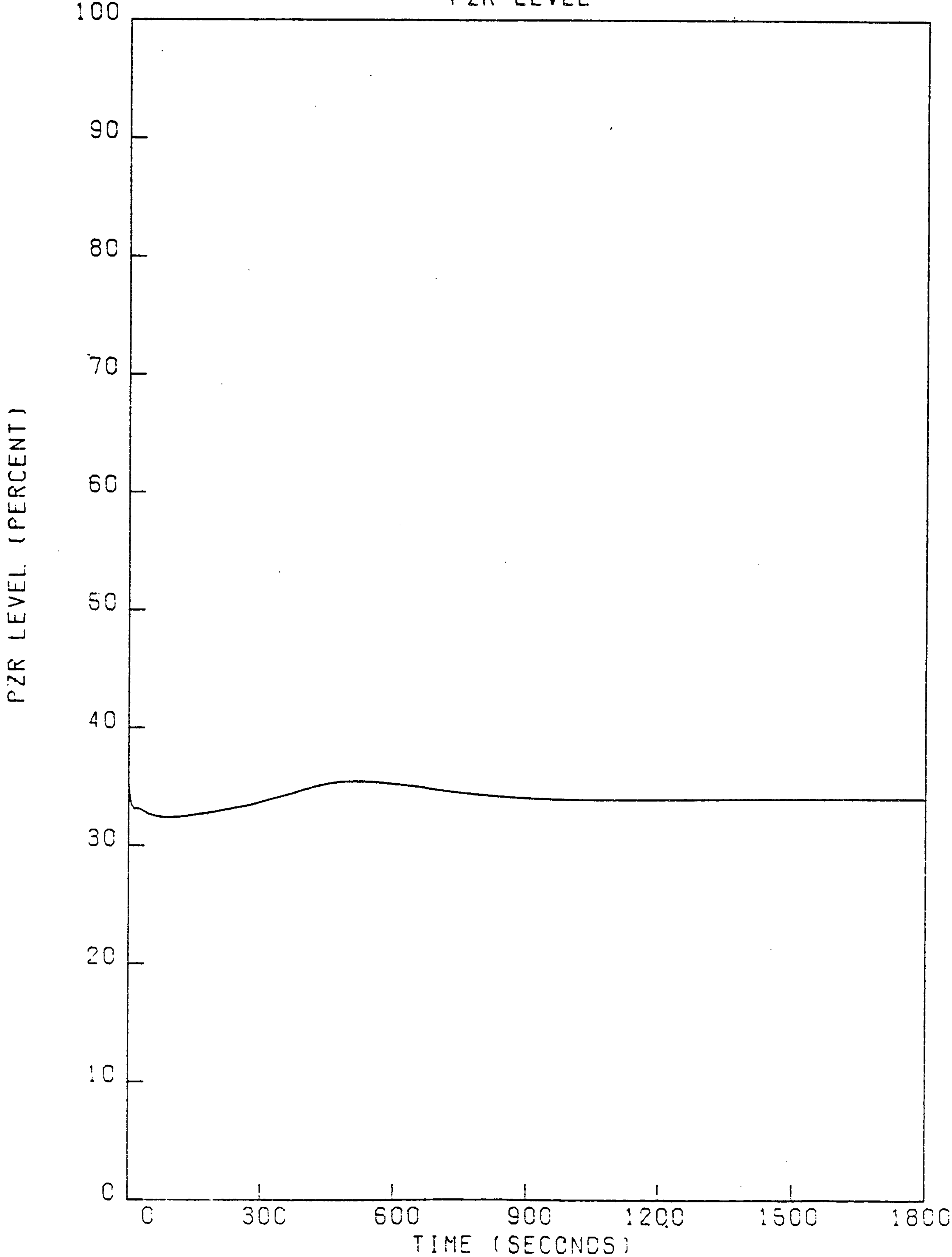


FIGURE 2-59
SONGS 20 PCT PWR LOSS OF AC
NORMALIZED RCS FLOW

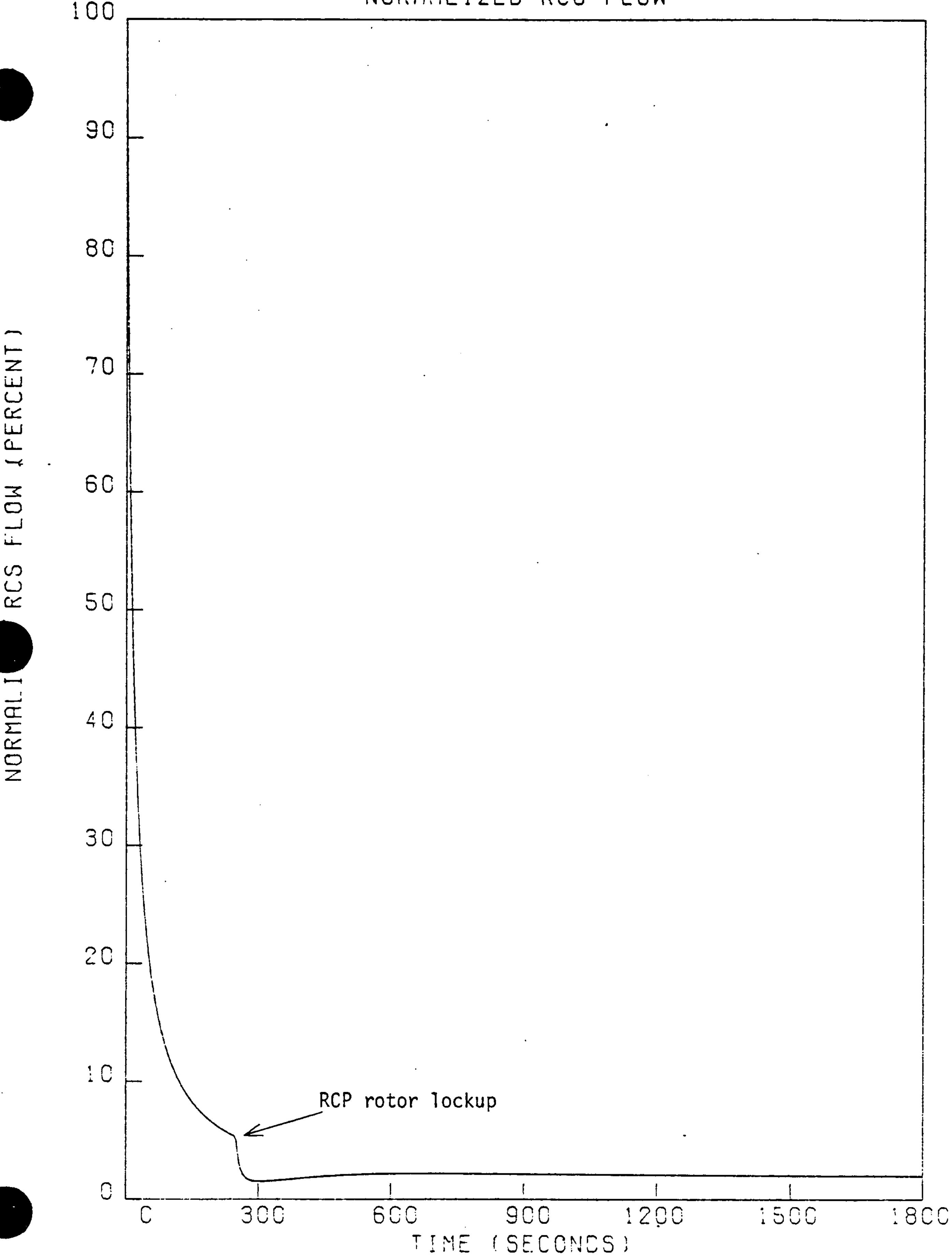


FIGURE 2-60
SONGS 20 PCT PWR LOSS OF AC
NORMALIZED RCS FLOW

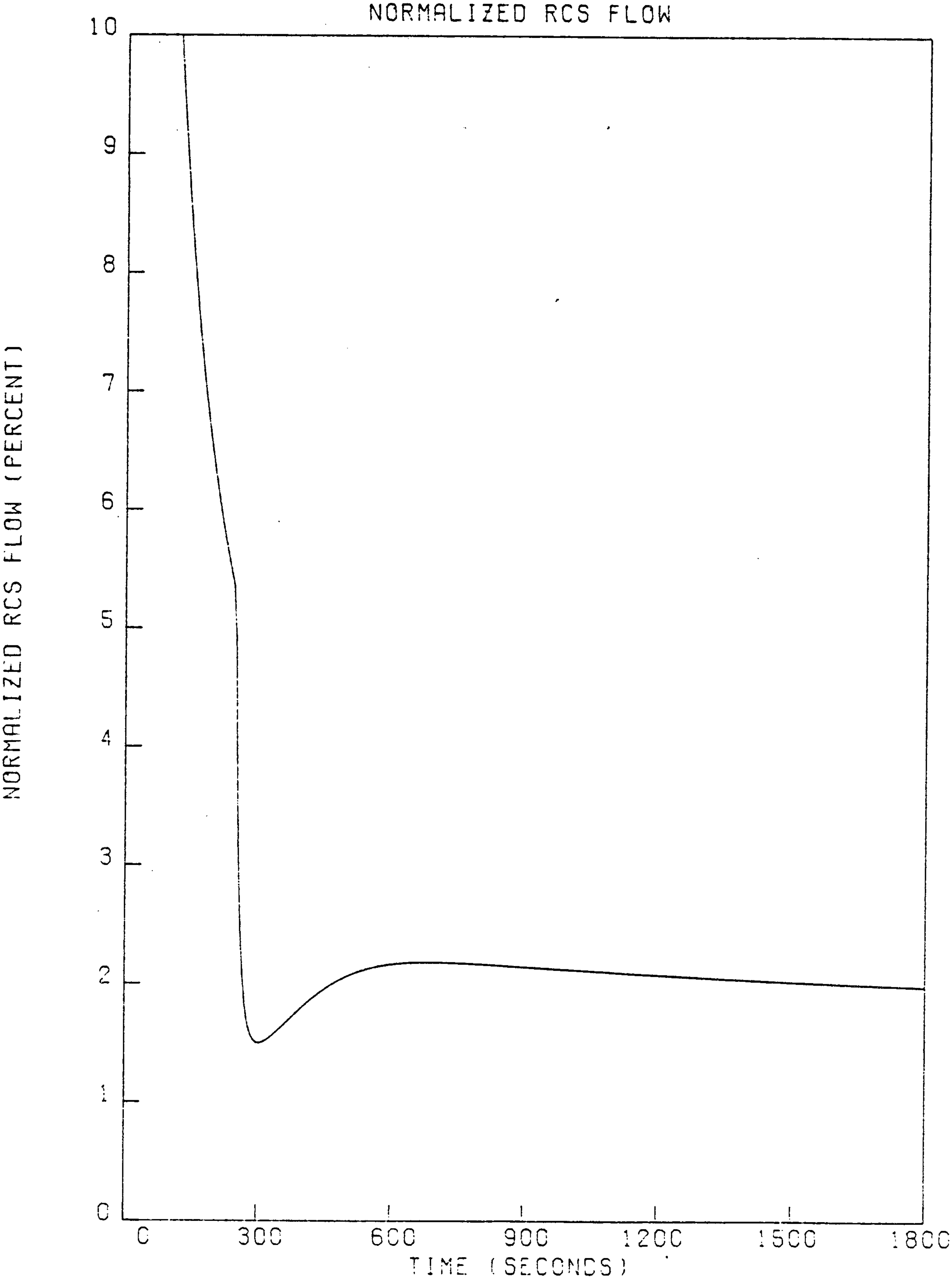


FIGURE 2-61
SONGS 20 PCT PWR LOSS OF AC
LOOP A COOLANT TEMPERATURES

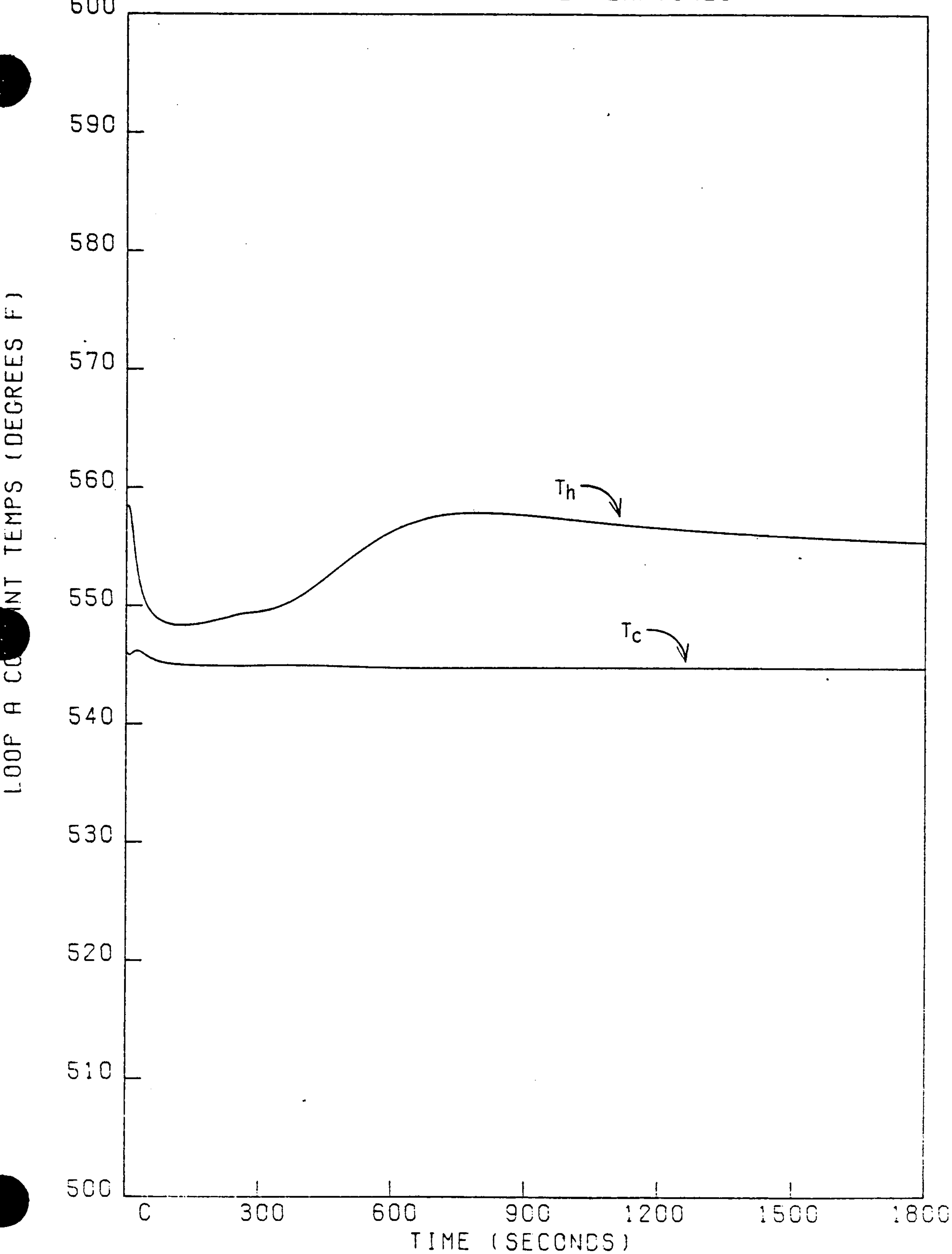


FIGURE 2-62
SONGS 20 PCT PWR LOSS OF AC
LOOP B COOLANT TEMPERATURES

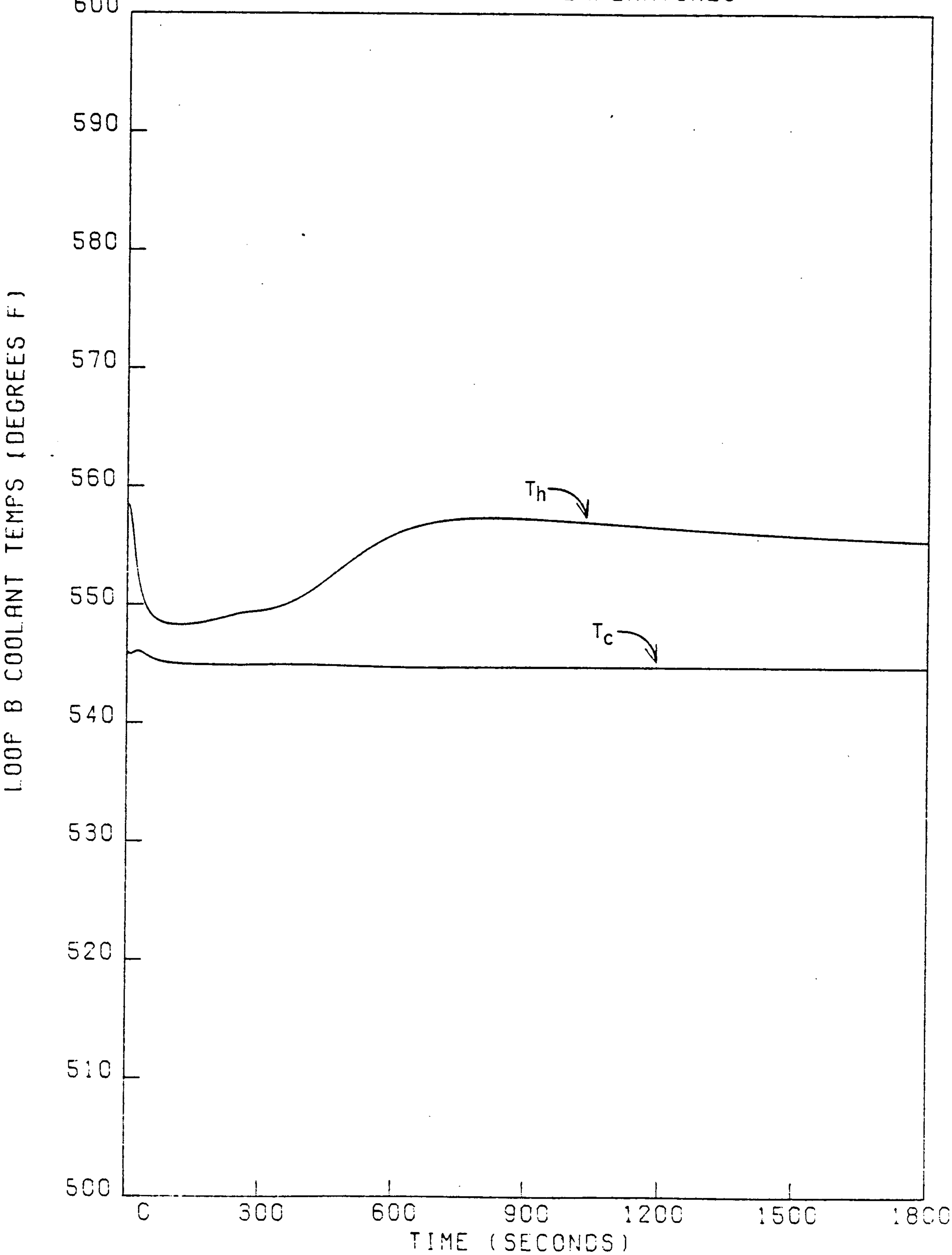


FIGURE 2-63
SONGS 20 PCT PWR LOSS OF AC
S.G. A PRESSURE

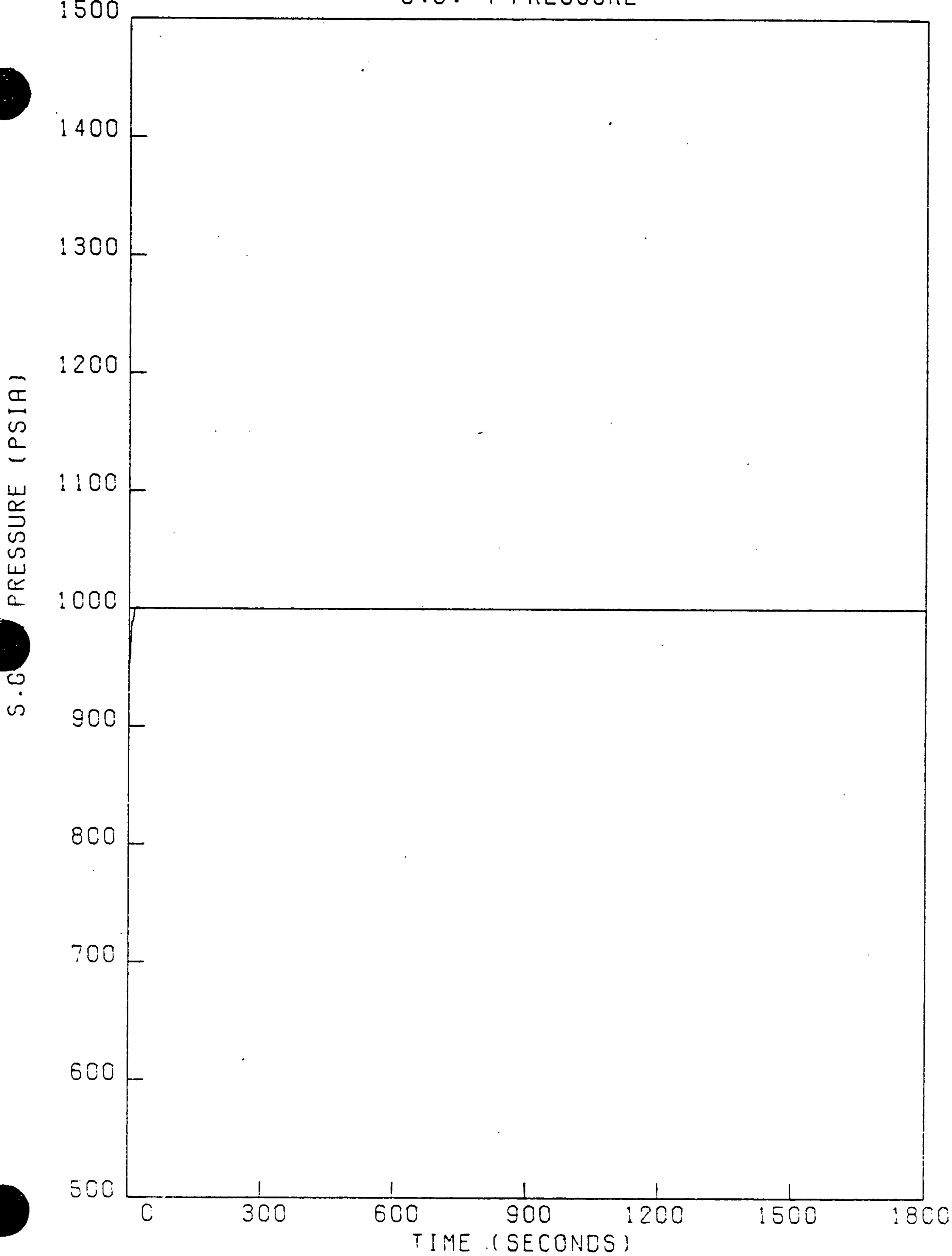


FIGURE 2-64
SONGS 20 PCT PWR LOSS OF AC
S.G. B PRESSURE

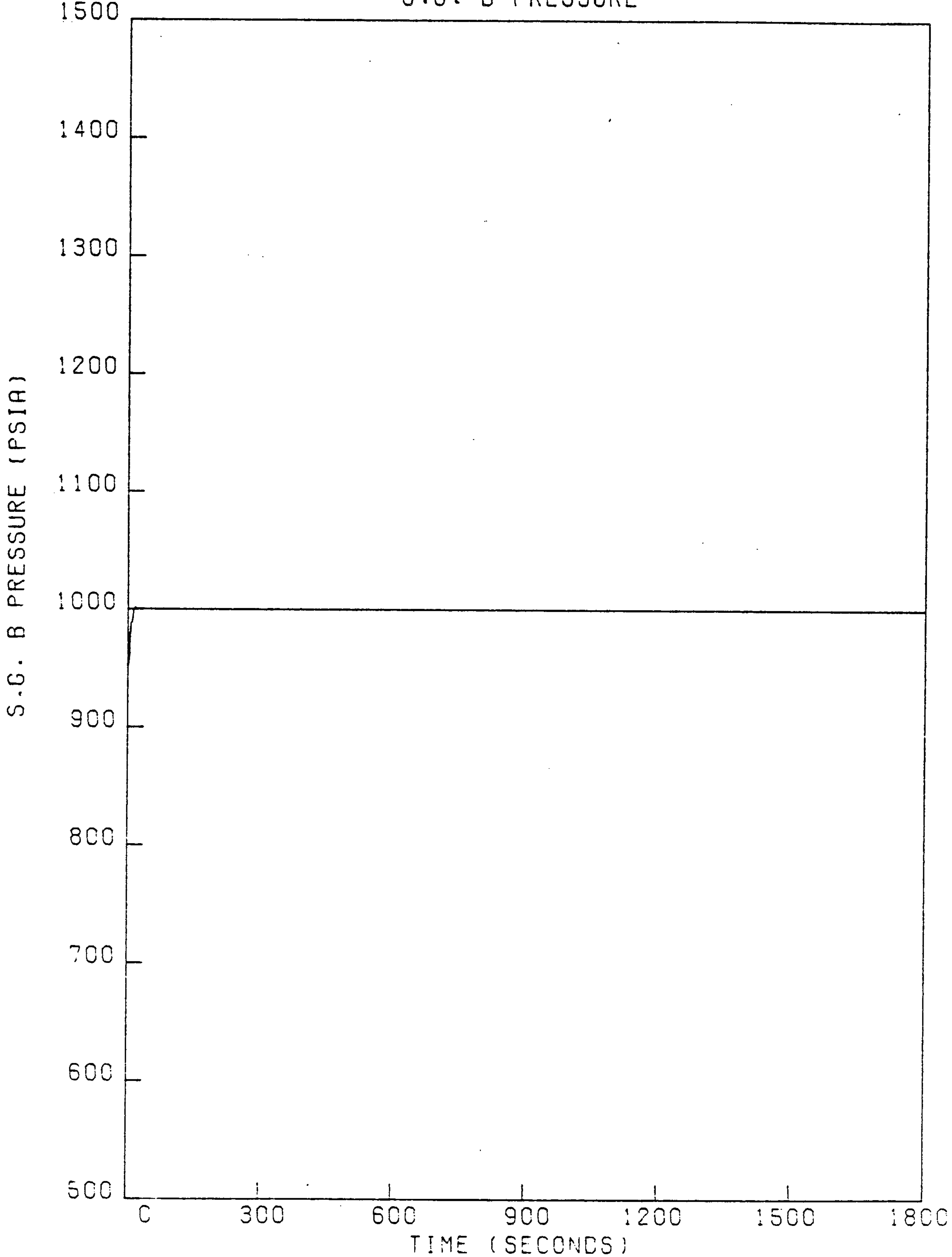


FIGURE 2-65
SONGS 20 PCT PWR LOSS OF AC
S.G. A NARROW RANGE LEVEL

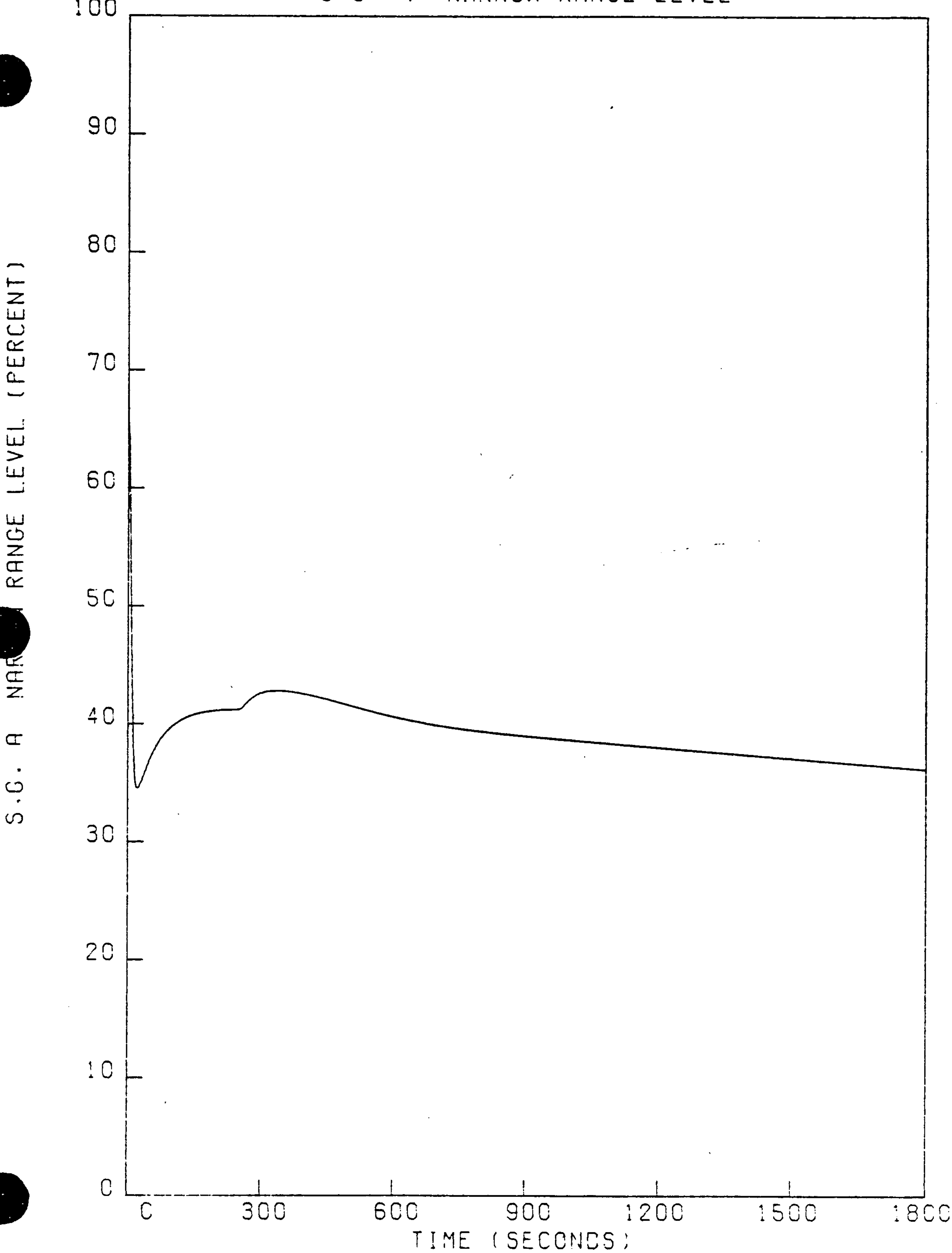


FIGURE 2-66
SONGS 20 PCT PWR LOSS OF AC
S.G. B NARROW RANGE LEVEL

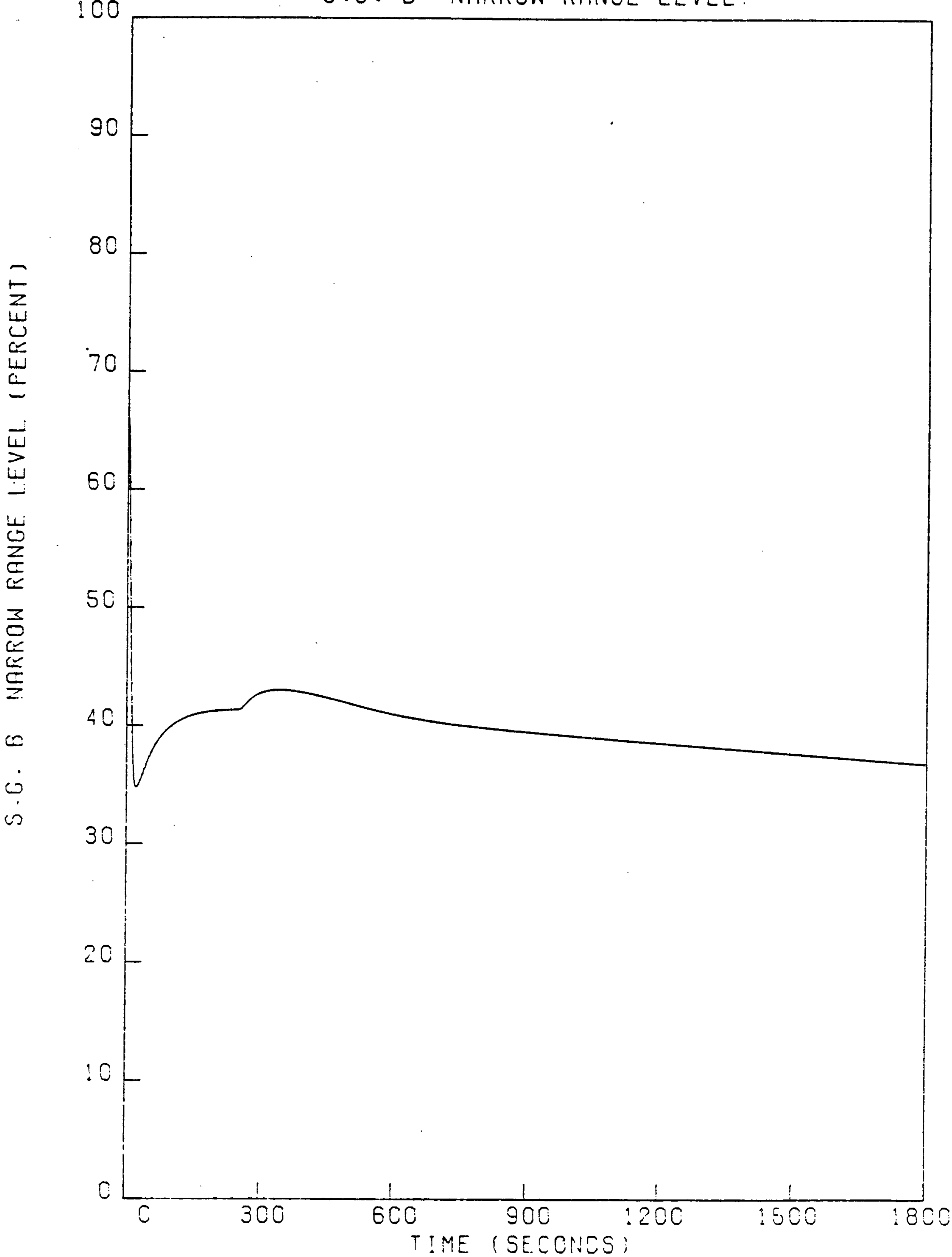


FIGURE 2-67
SONGS 20 PCT PWR LOSS OF AC
S.G. A STEAM FLOW

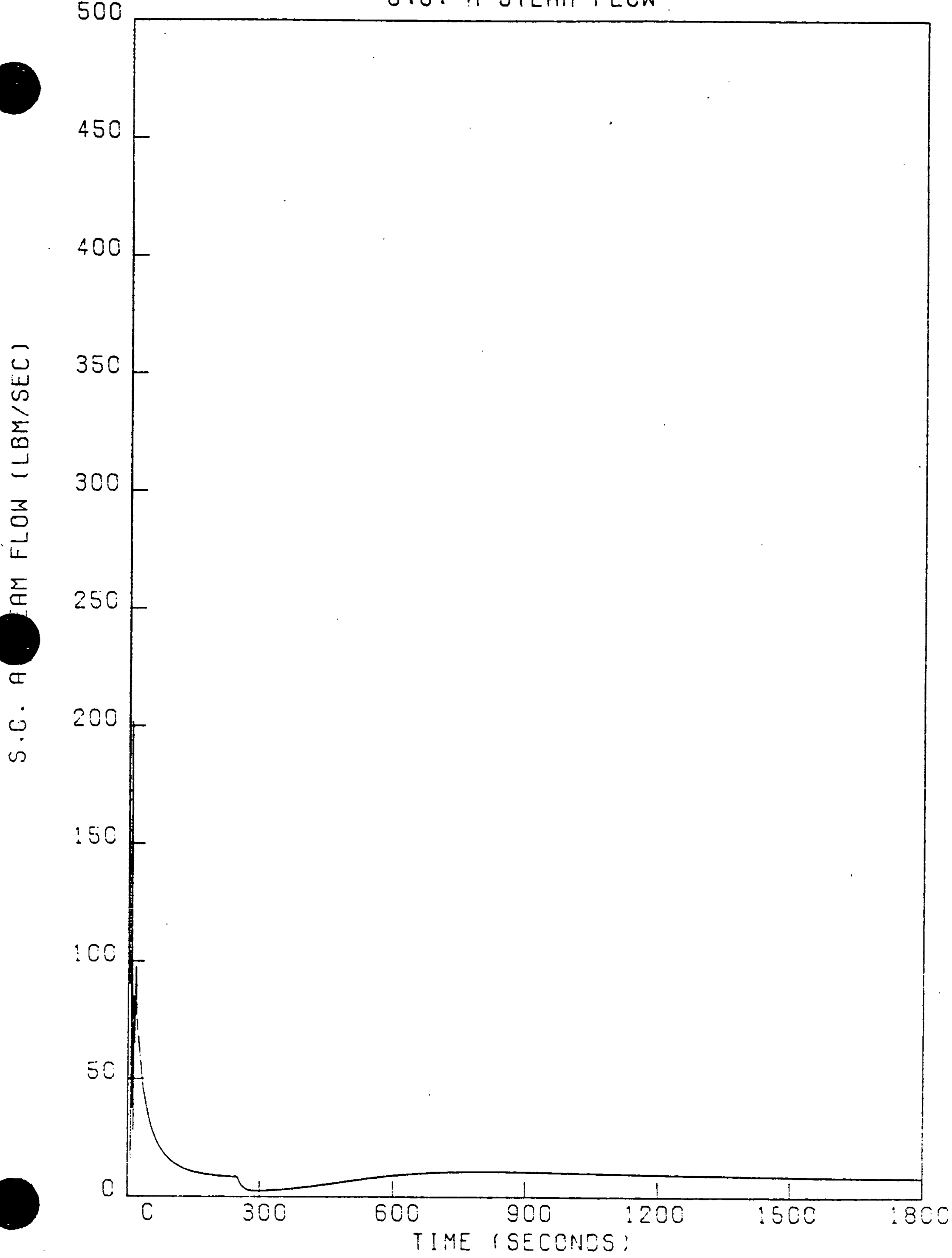


FIGURE 2-68
SONGS 20 PCT PWR LOSS OF AC
S.G. B STEAM FLOW

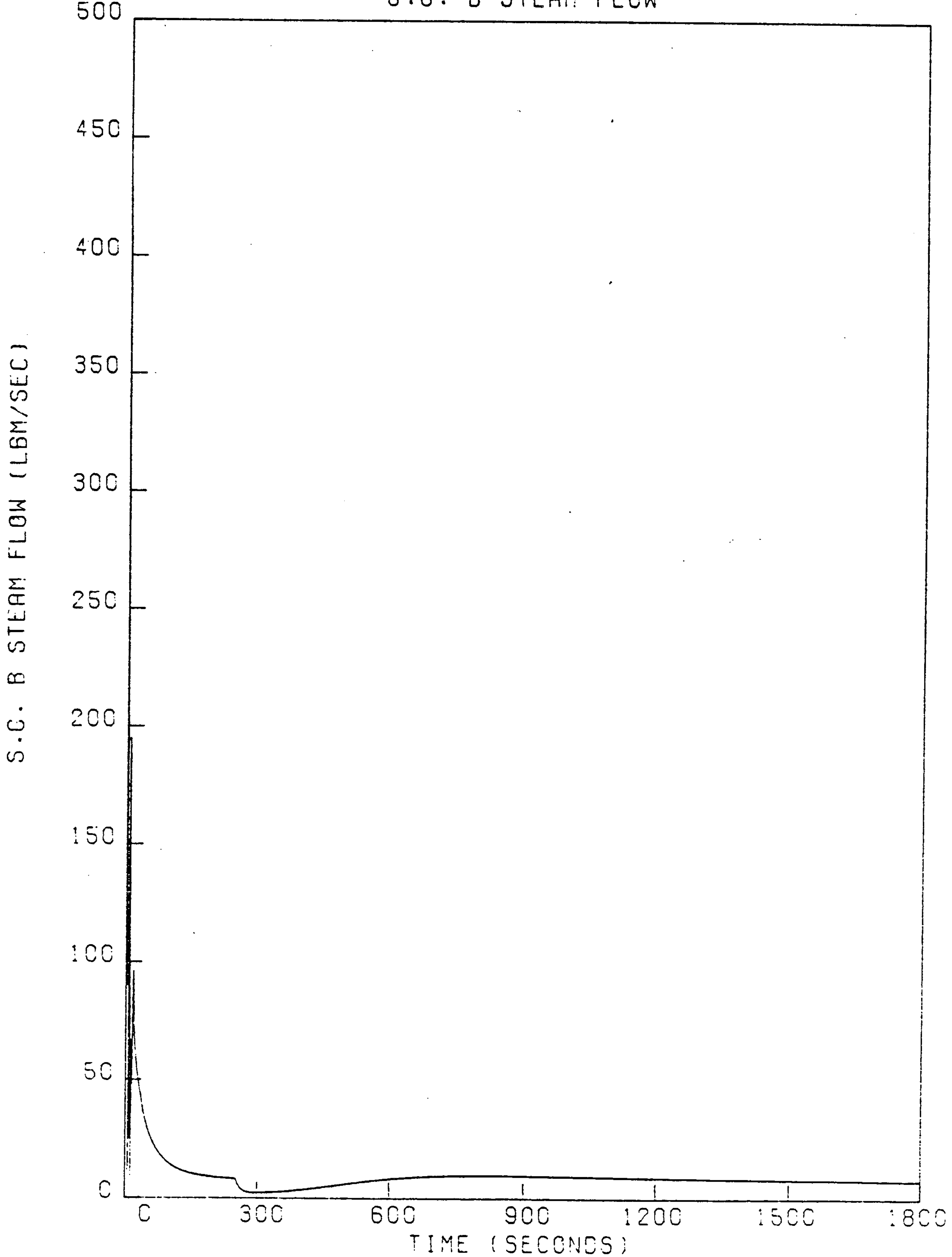


FIGURE 2-69
SONGS 20 PCT PWR LOSS OF AC
CHARGING FLOW

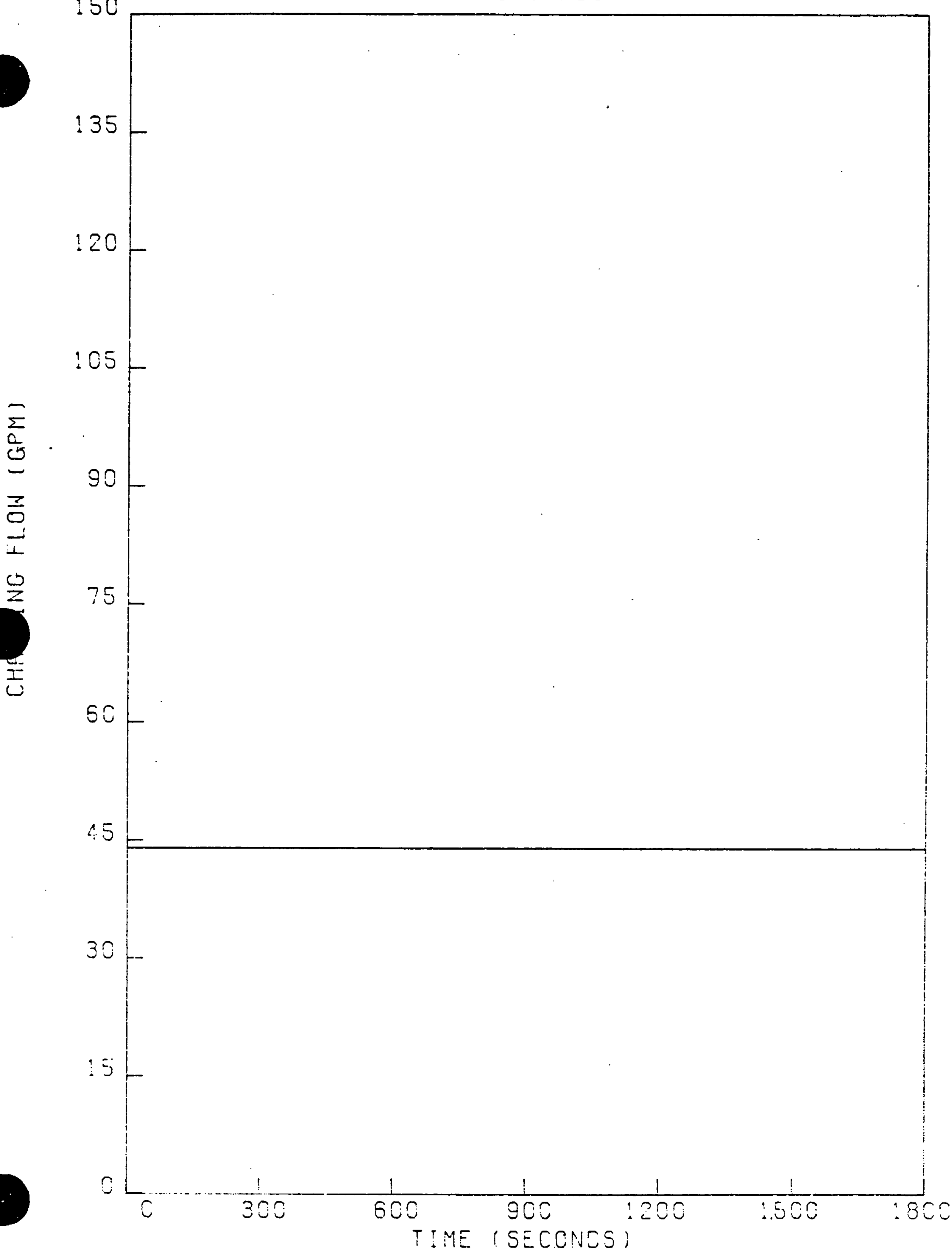


FIGURE 2-70
SONGS 20 PCT PWR LOSS OF AC
LETDOWN FLOW

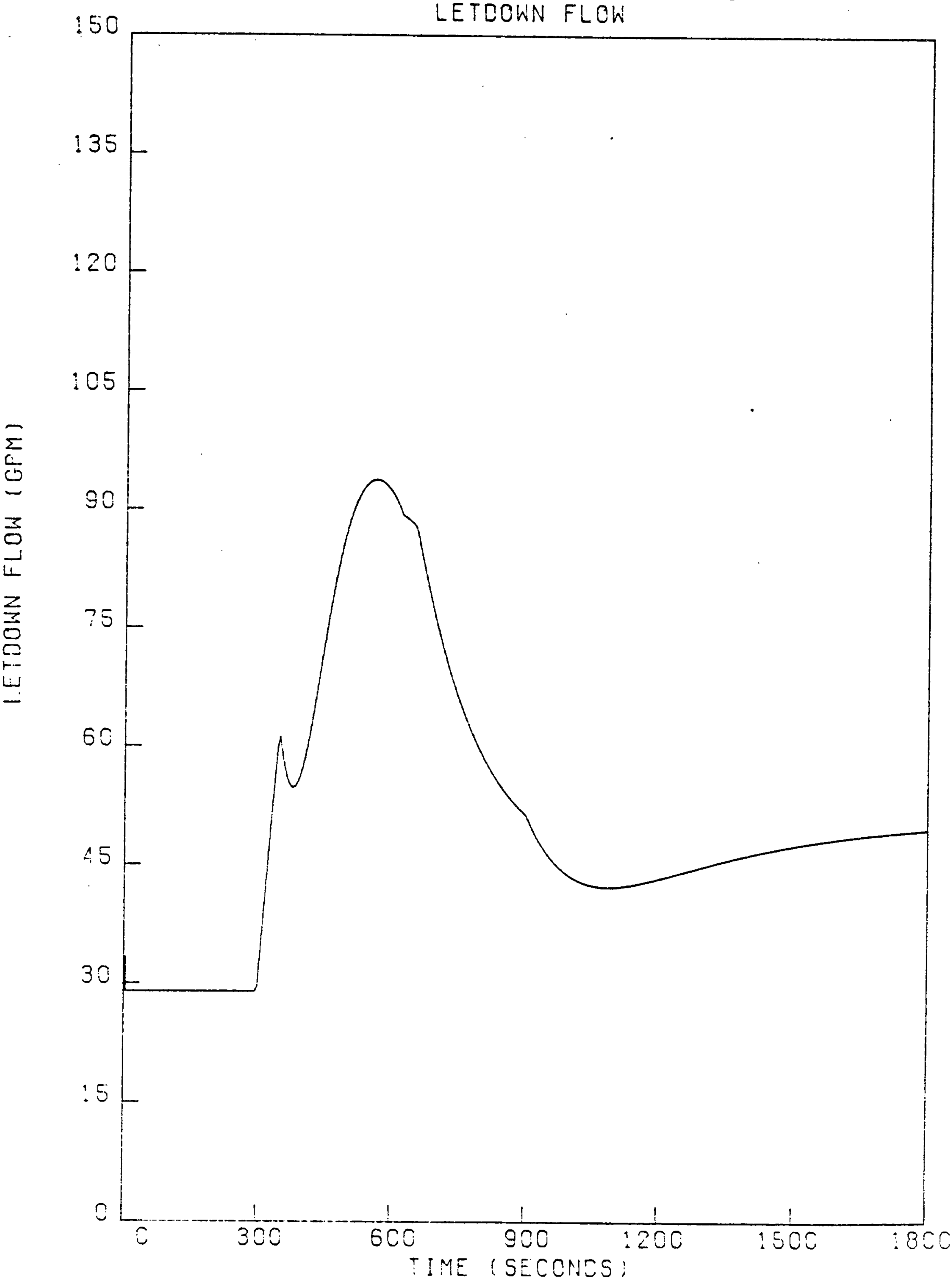


FIGURE 2-71
SONGS 20 PCT PWR LOSS OF AC
PZR HEATER OUTPUTS

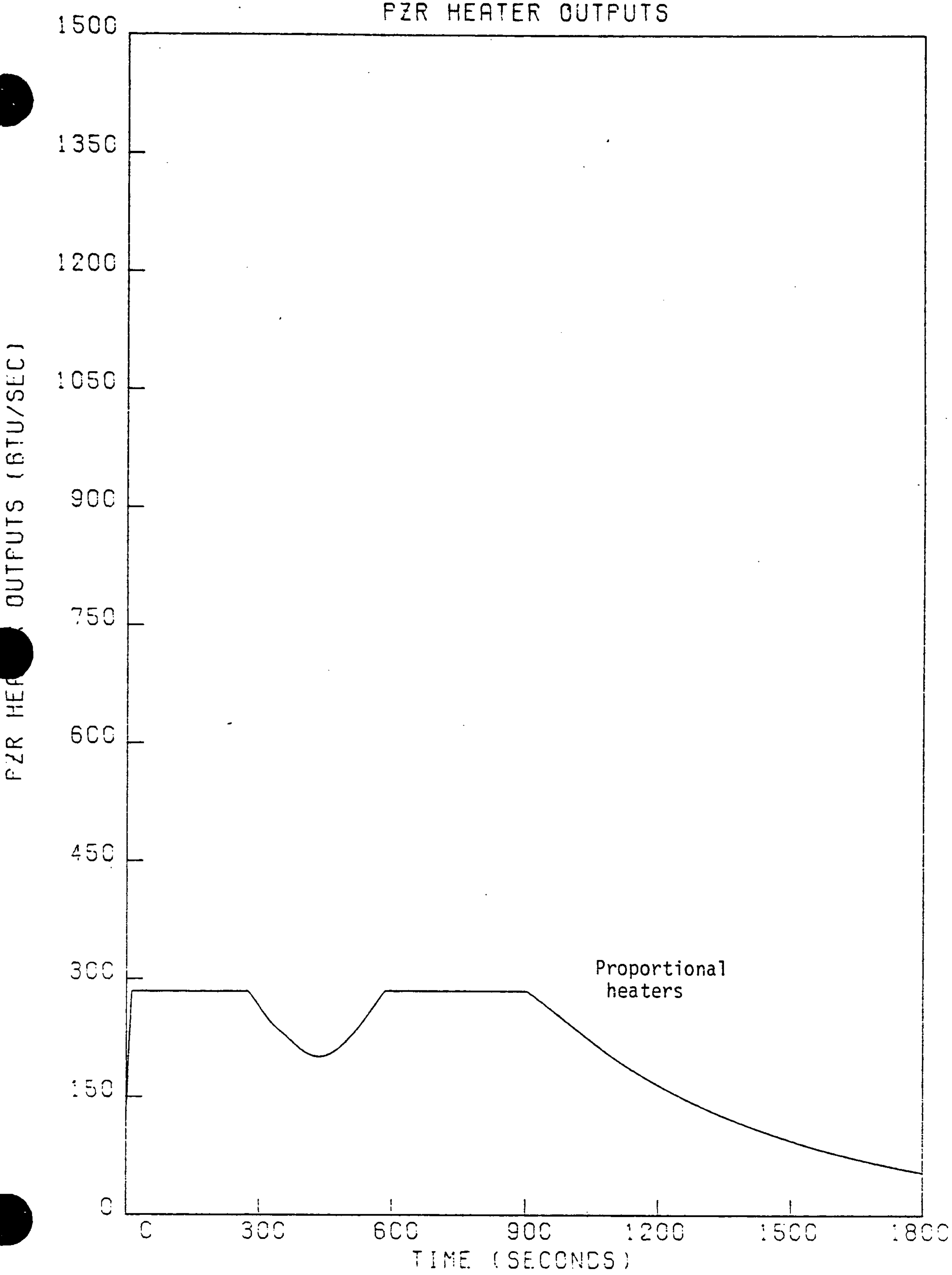


FIGURE 2-72
SONGS 80 PCT PWR LOSS OF FLOW
POST TRIP CORE DECAY HEAT

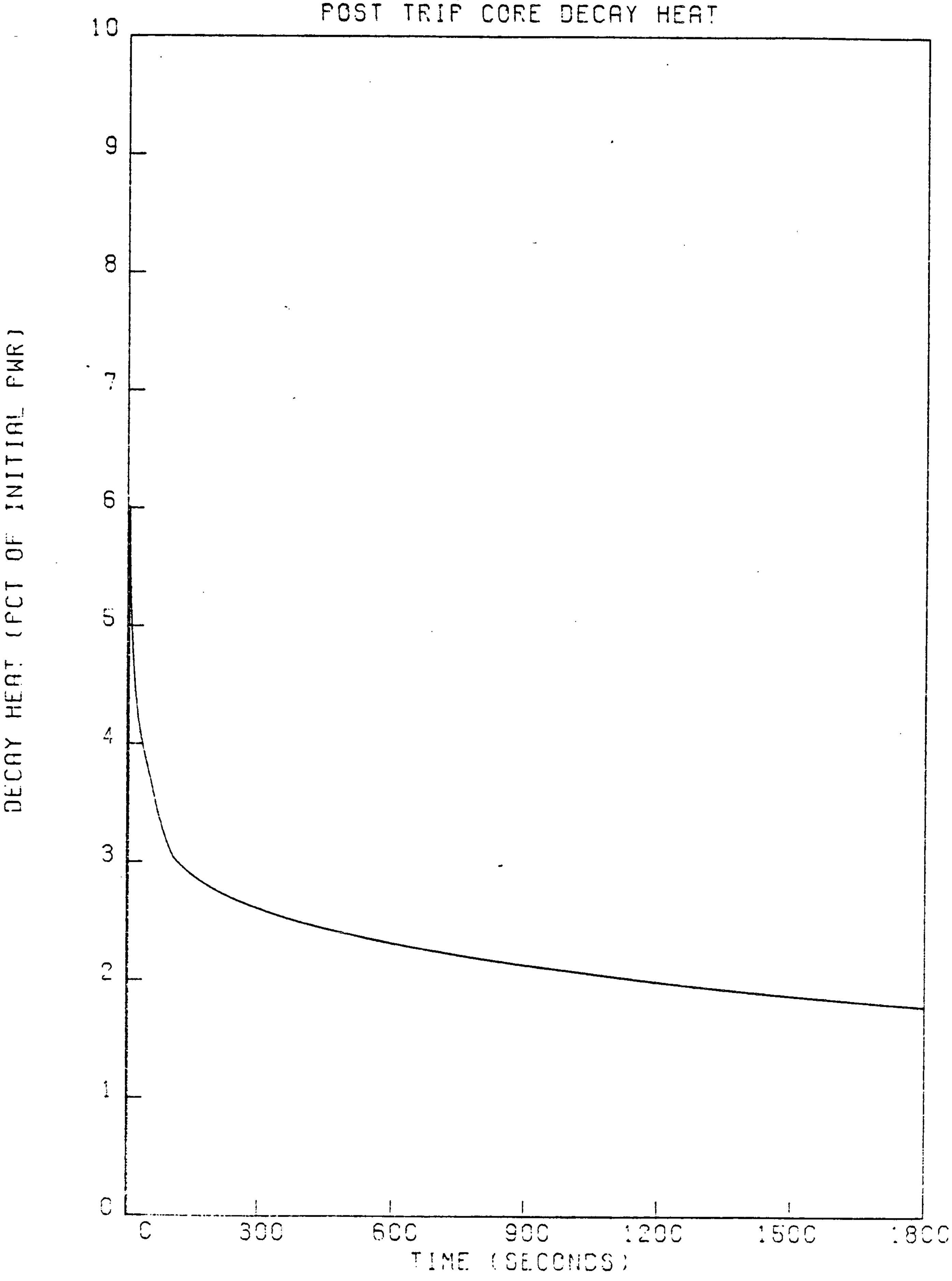


FIGURE 2-73
SONGS 80 PCT PWR LOSS OF FLOW
PRESSURIZER PRESSURE

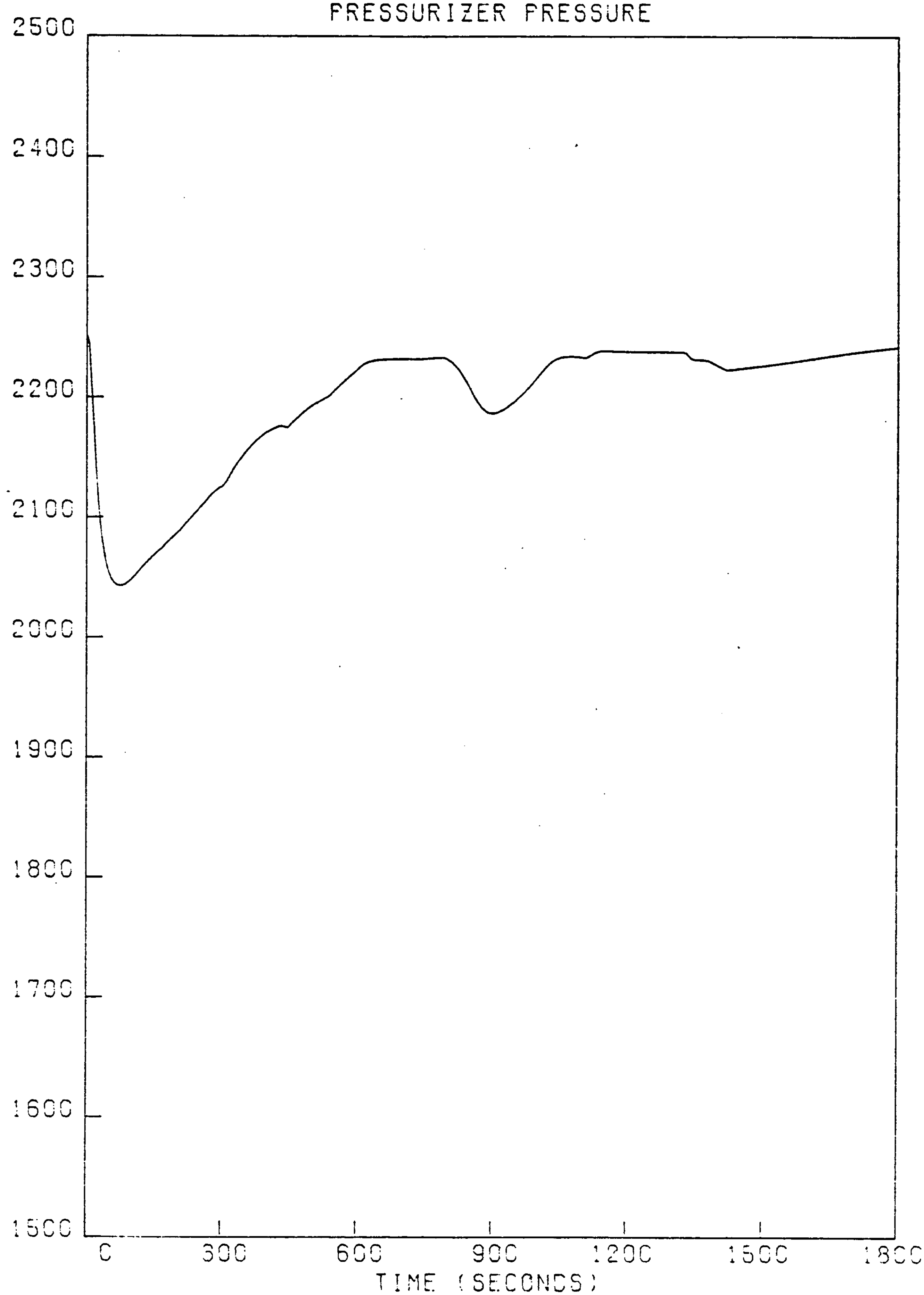


FIGURE 2-74
SONGS 80 PCT PWR LOSS OF FLOW
PZR LEVEL

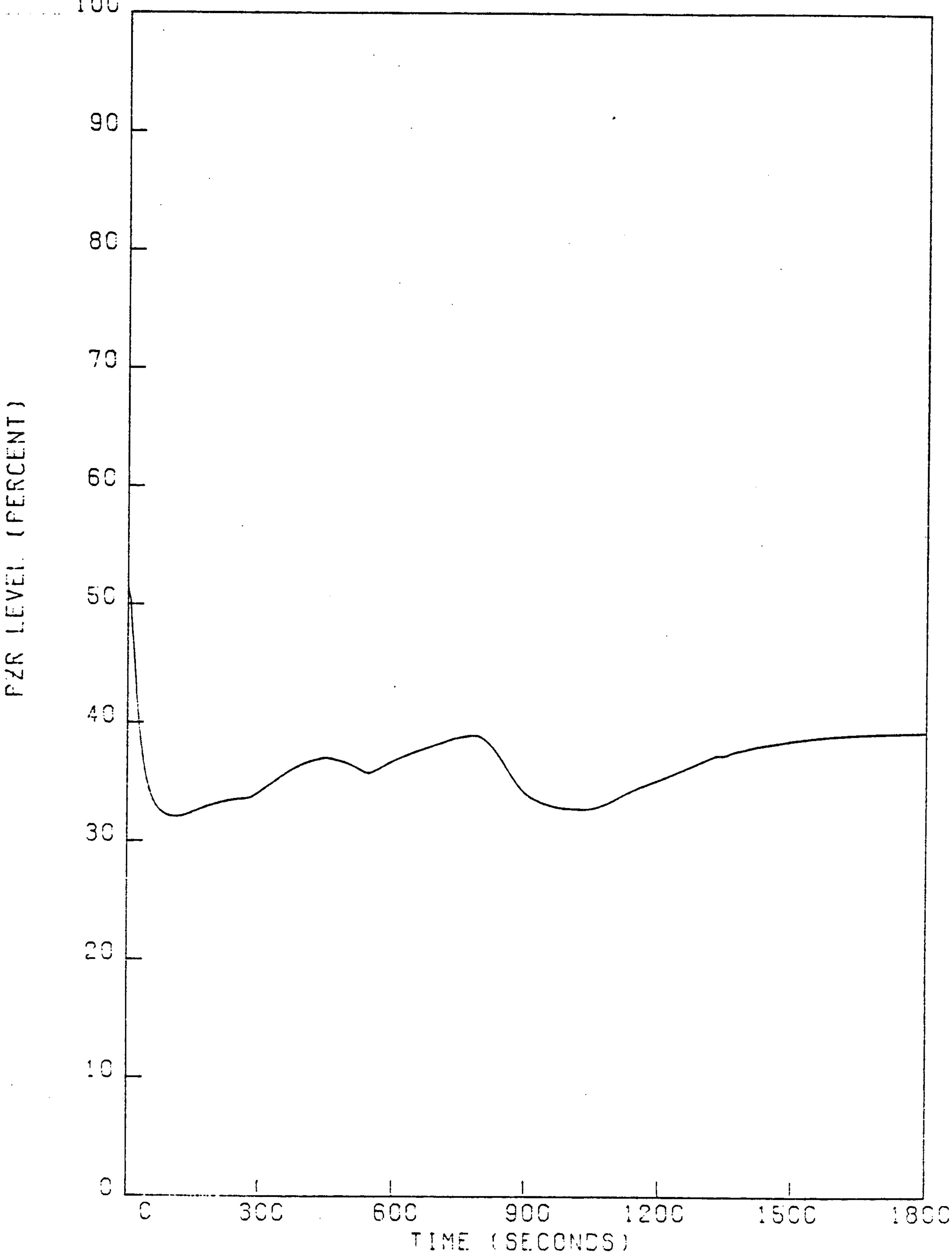


FIGURE 2-75
SONGS 80 PCT PWR LOSS OF FLOW
NORMALIZED RCS FLOW

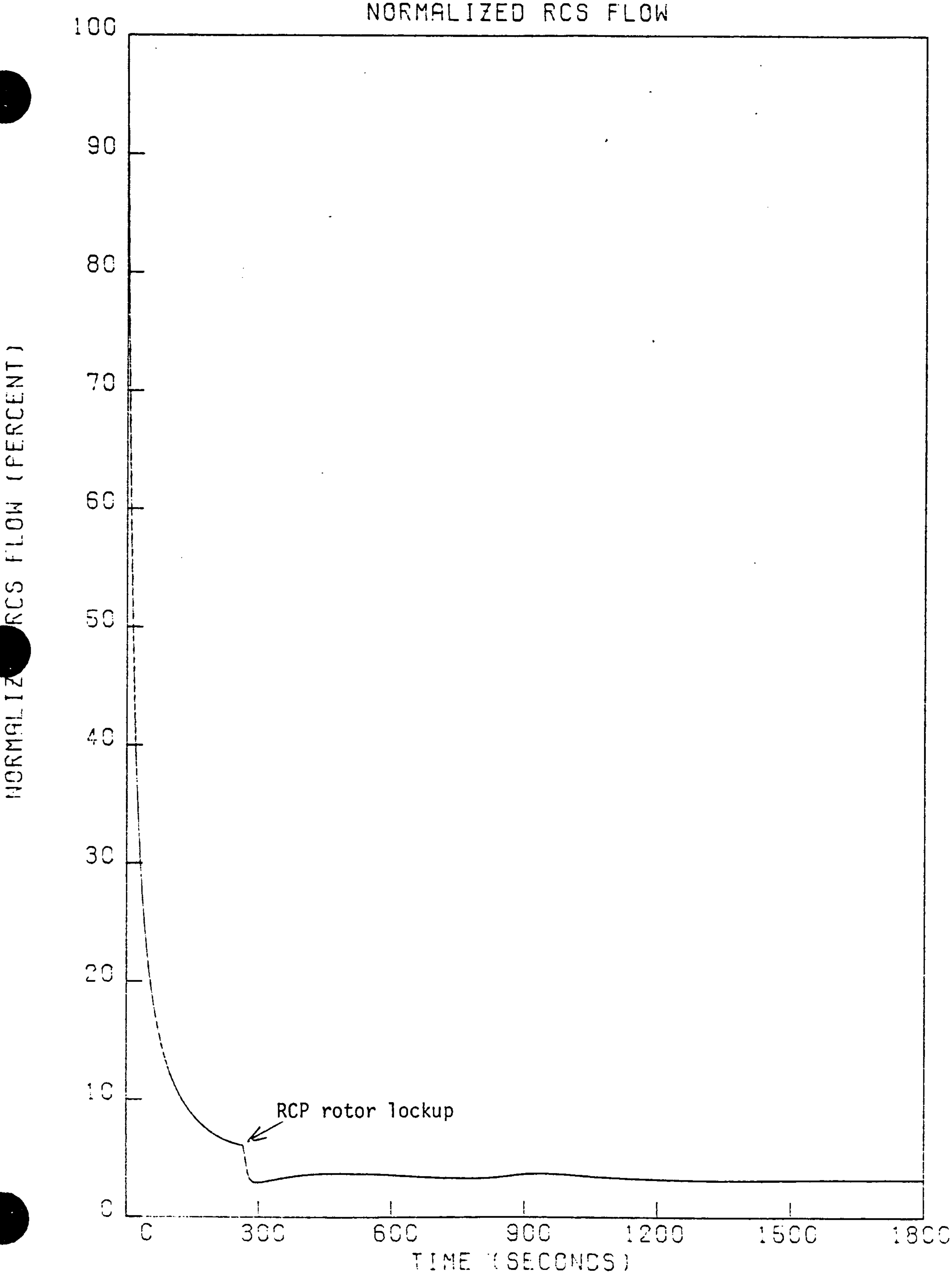


FIGURE 2-76
SONGS 80 PCT PWR LOSS OF FLOW
NORMALIZED RCS FLOW

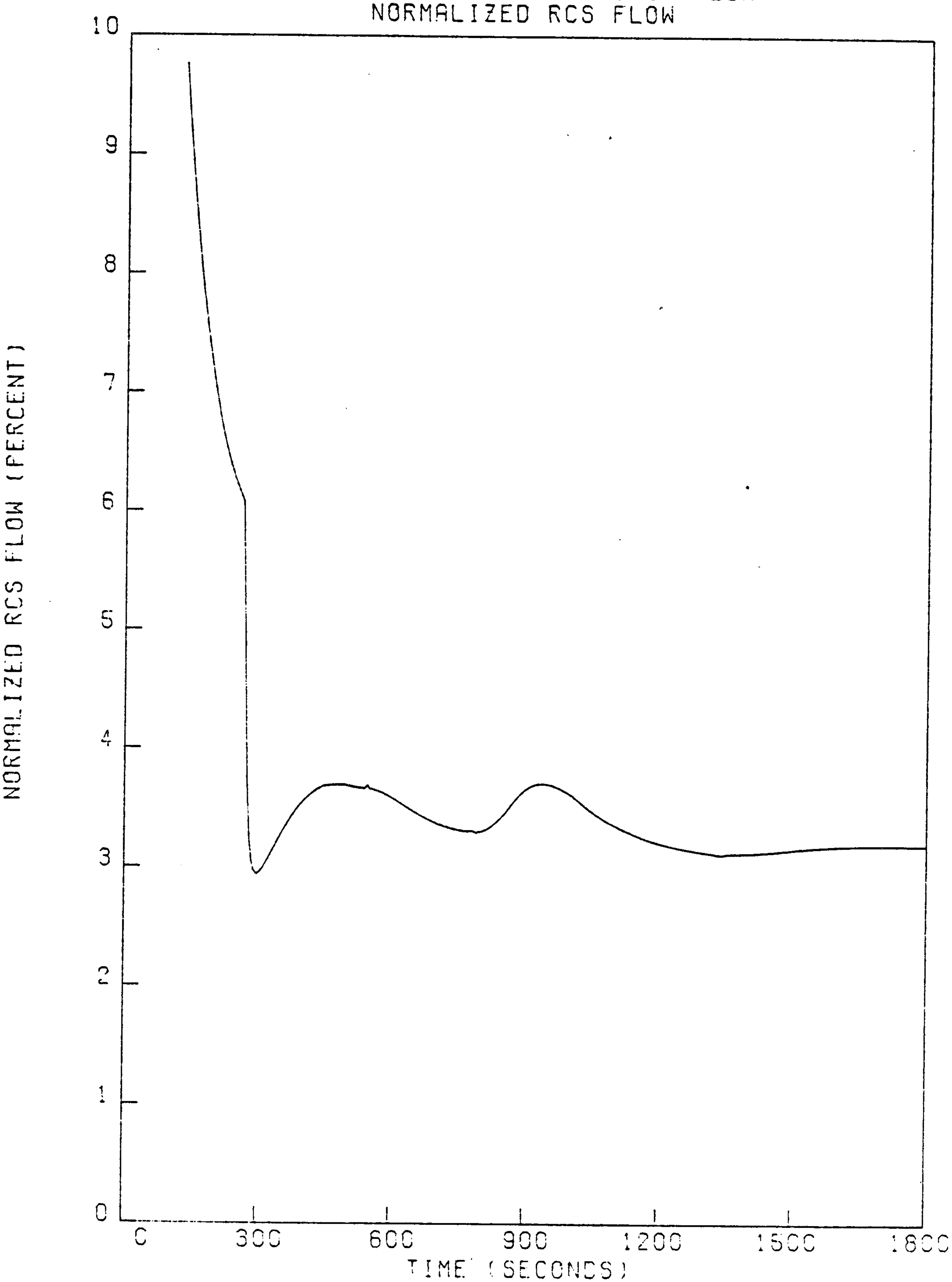


FIGURE 2-77
SONGS 80 PCT PWR LOSS OF FLOW
LOOP A COOLANT TEMPERATURES

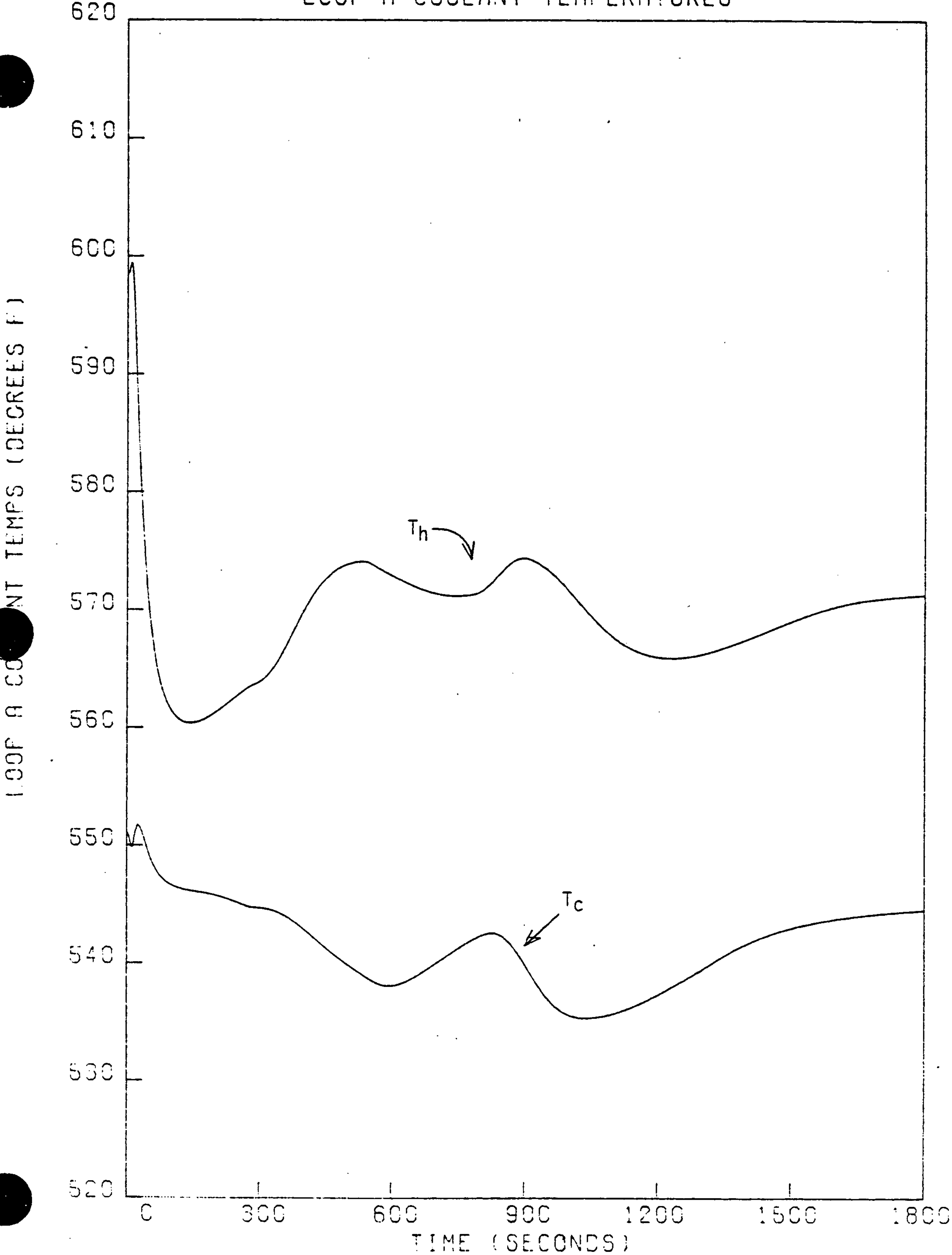


FIGURE 2-78
SONGS 80 PCT PWR LOSS OF FLOW
LOOP B COOLANT TEMPERATURES

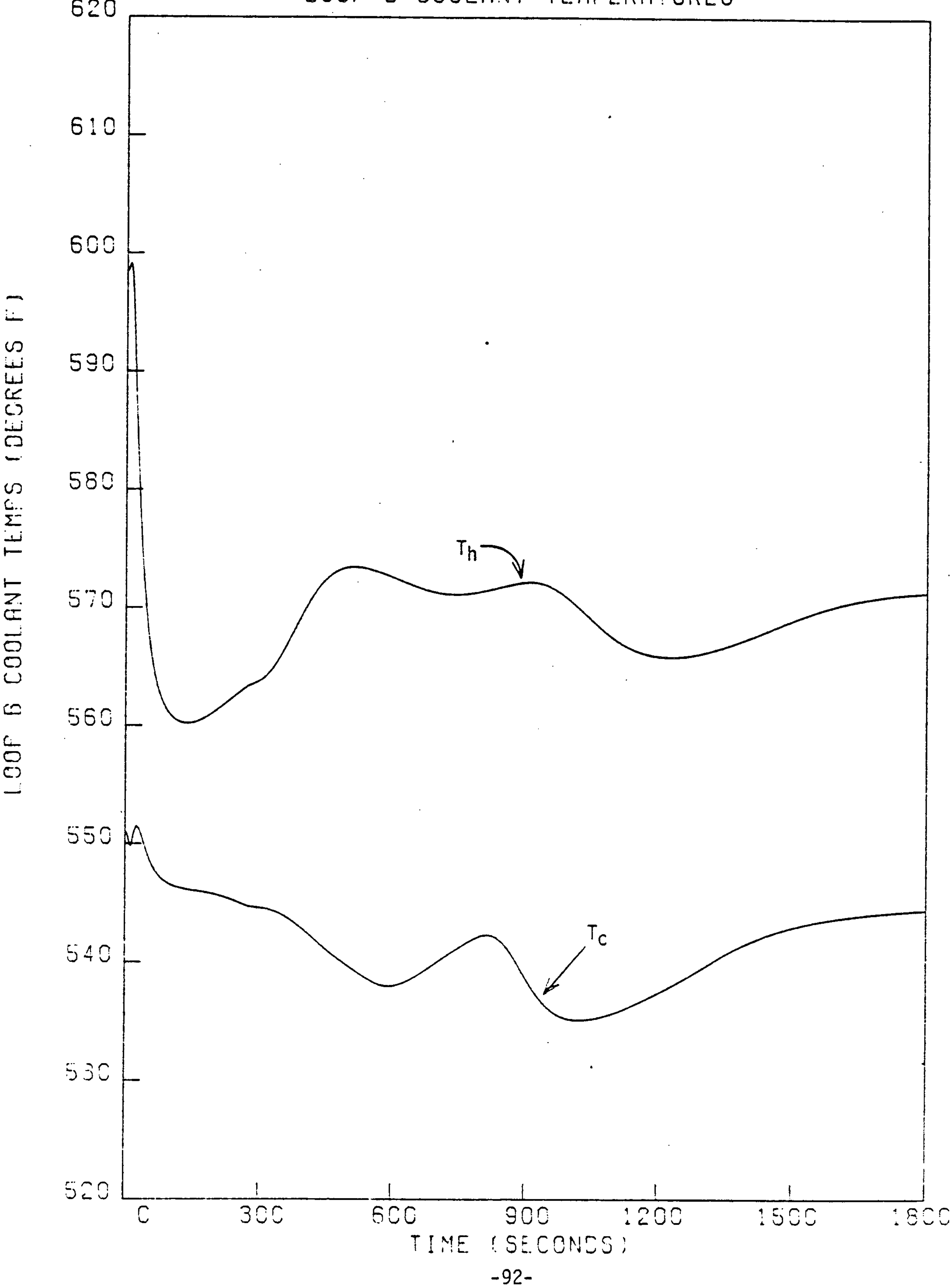


FIGURE 2-79
SONGS 80 PCT PWR LOSS OF FLOW
S.G. A PRESSURE

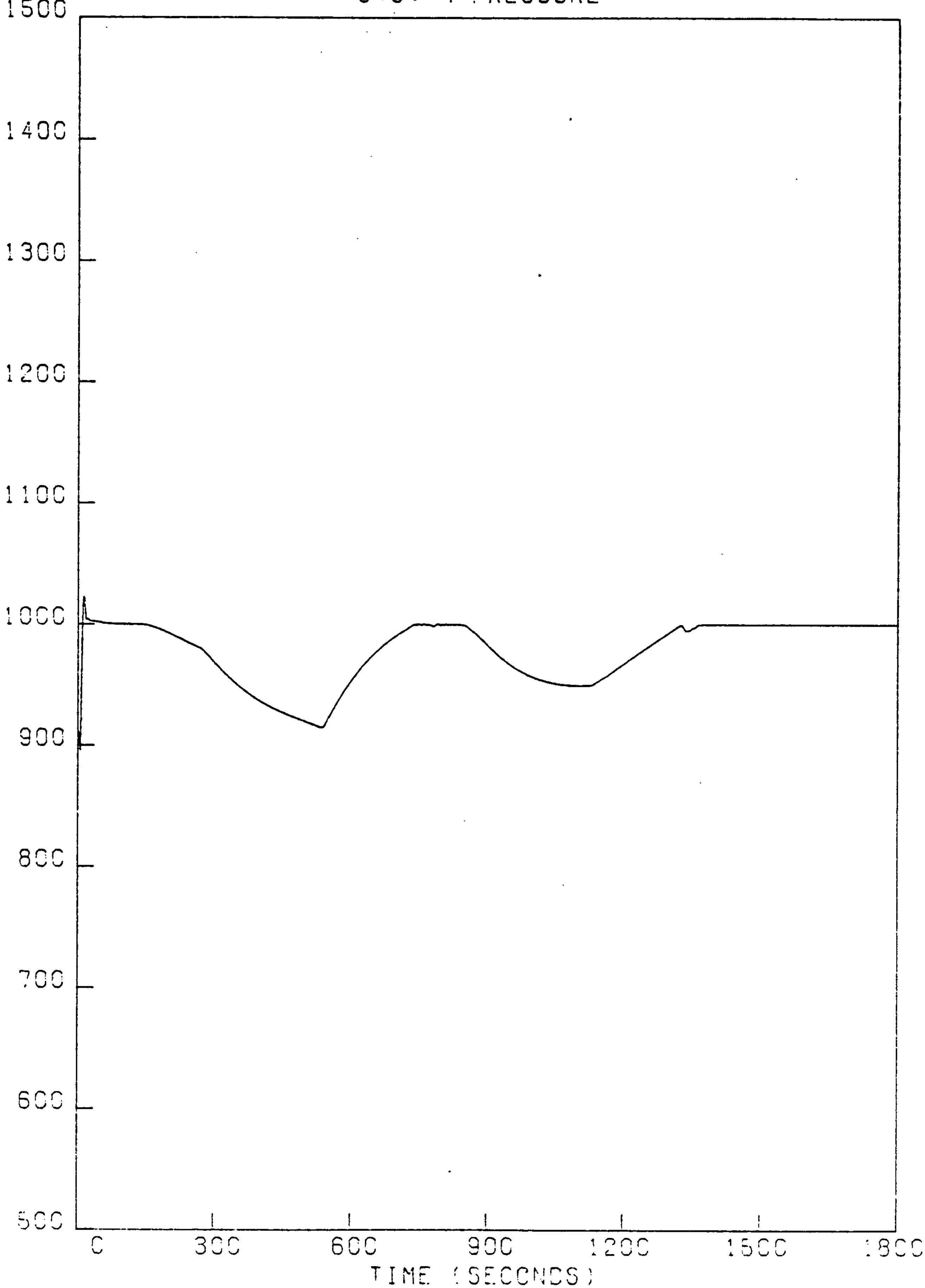


FIGURE 2-80
SONGS 80 PCT PWR LOSS OF FLOW
S.G. B PRESSURE

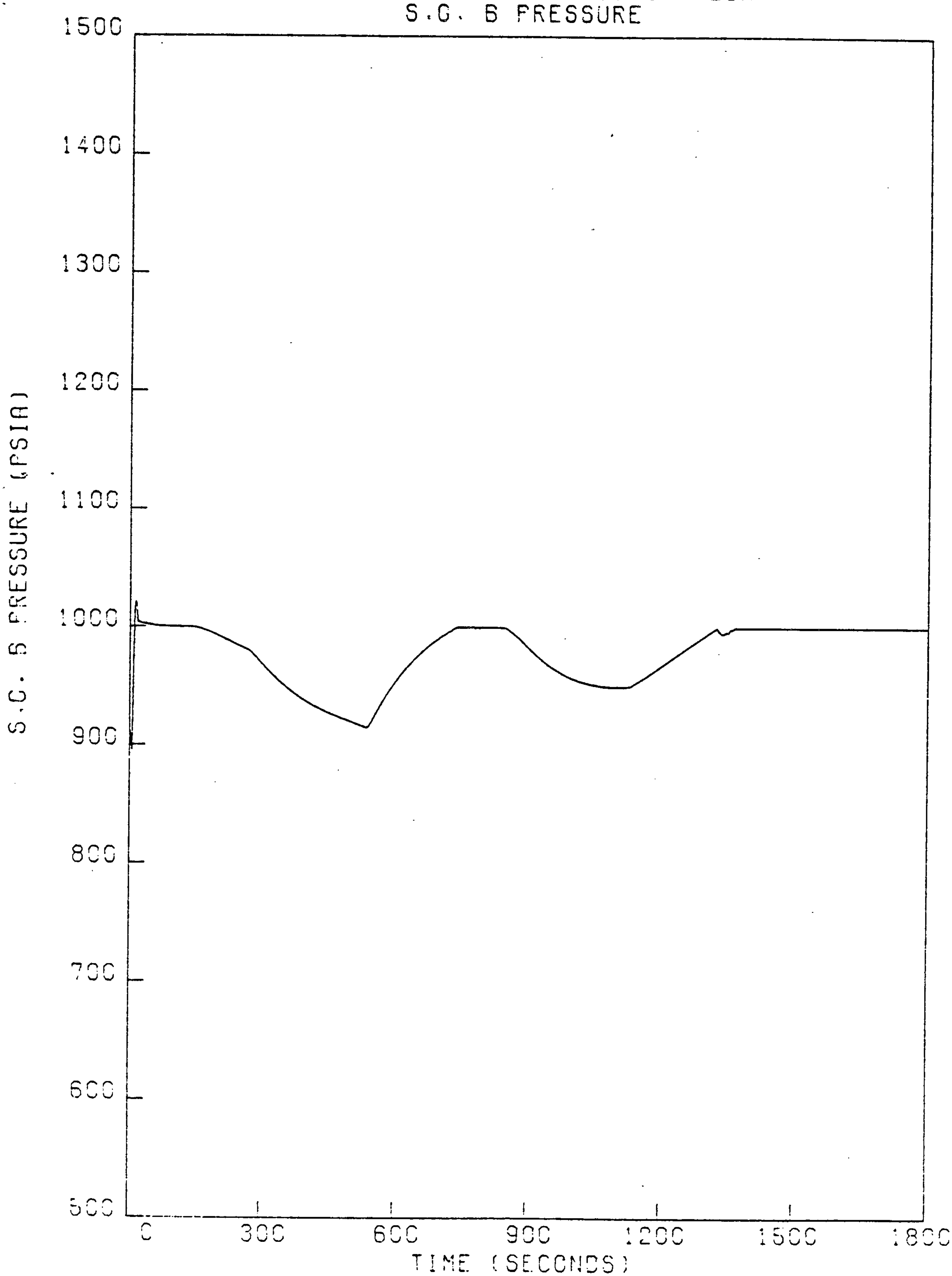


FIGURE 2-81
SONGS 80 PCT PWR LOSS OF FLOW
S.G. A NARROW RANGE LEVEL

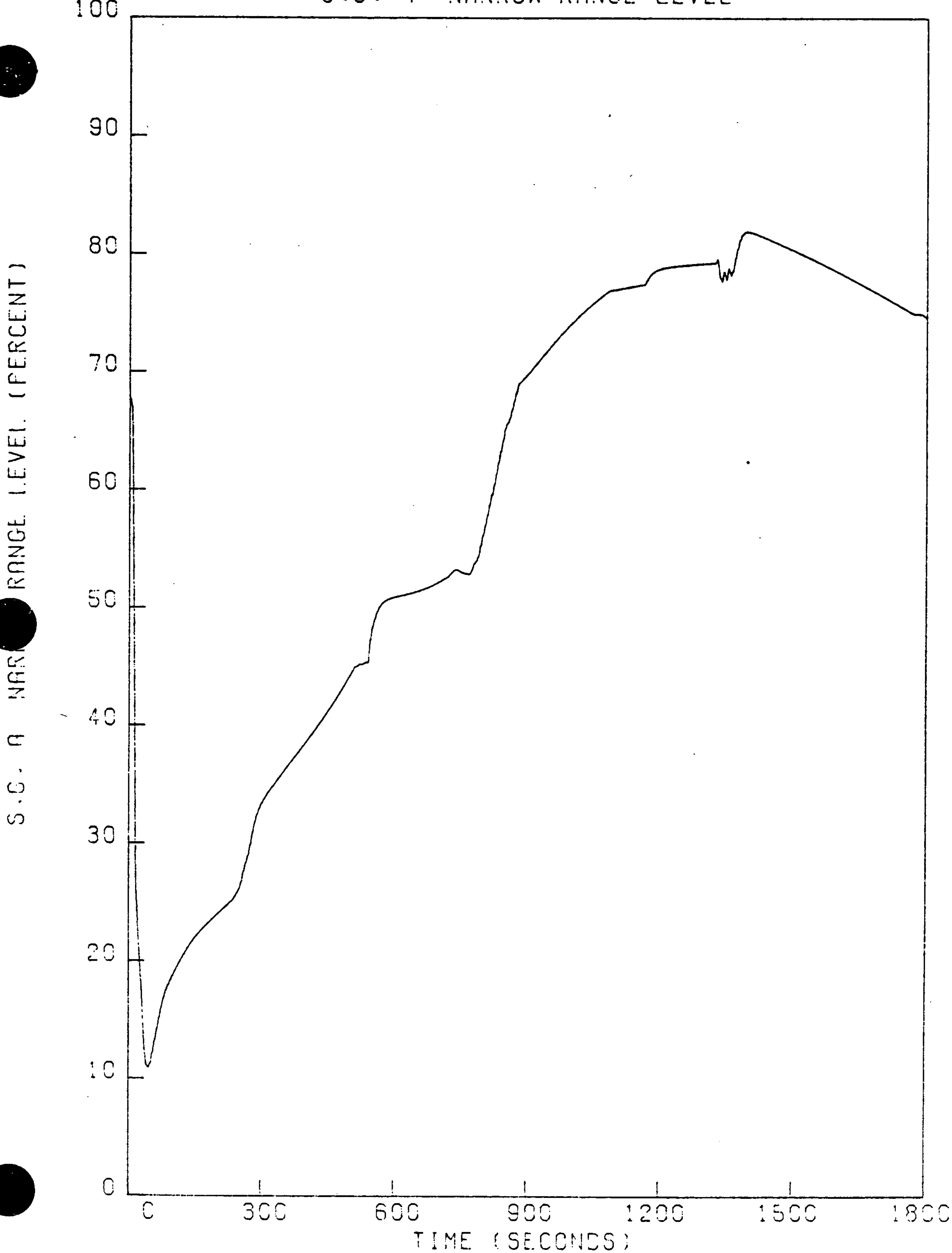


FIGURE 2-82
SONGS 80 PCT PWR LOSS OF FLOW
S.G. B NARROW RANGE LEVEL

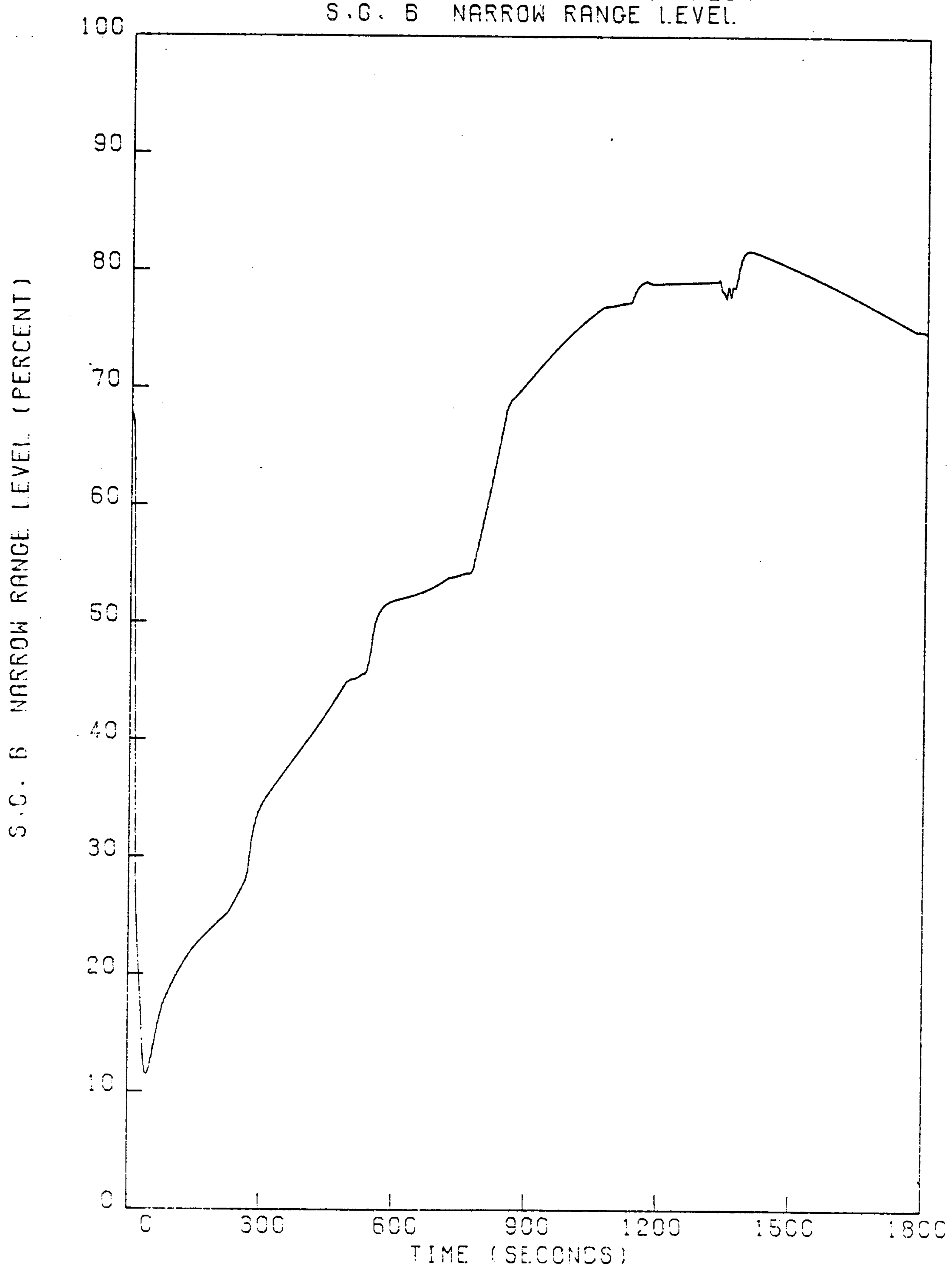


FIGURE 2-83
SONGS 80 PCT PWR LOSS OF FLOW
S.G. A STEAM FLOW

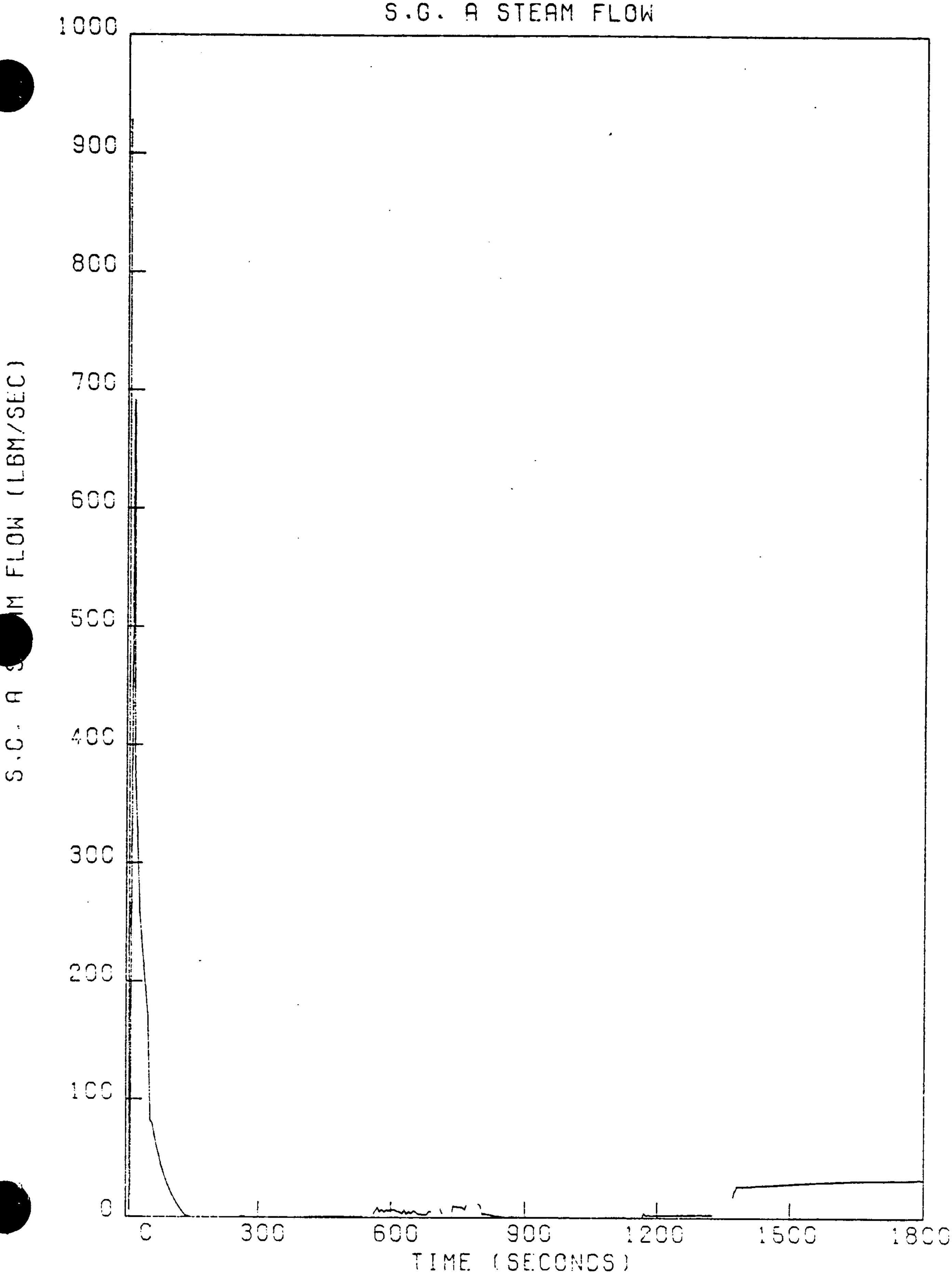


FIGURE 2-84
SONGS 80 PCT PWR LOSS OF FLOW
S.G. B STEAM FLOW

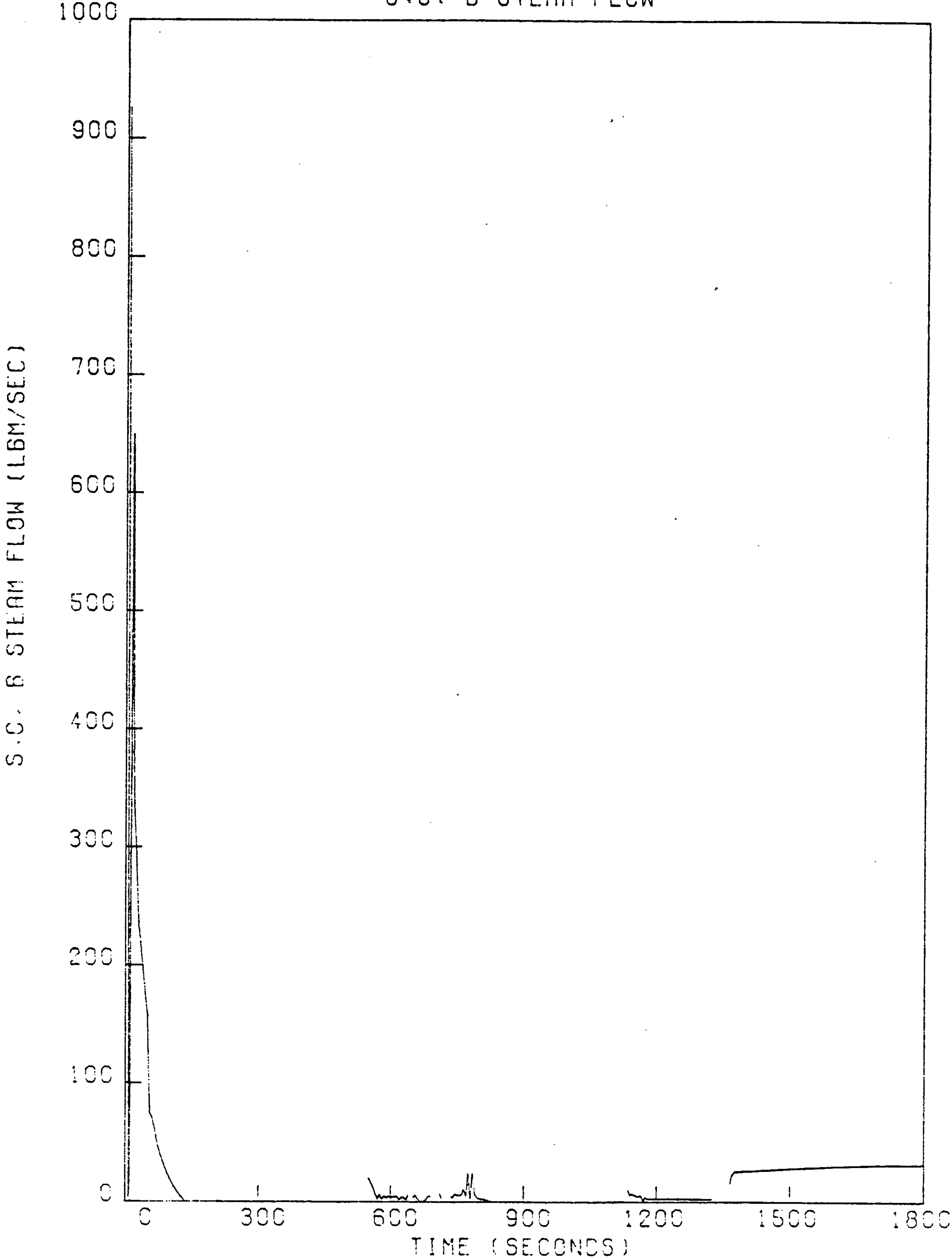


FIGURE 2-85
SONGS 80 PCT PWR LOSS OF FLOW
S.G. A MAIN FEEDWATER FLOW

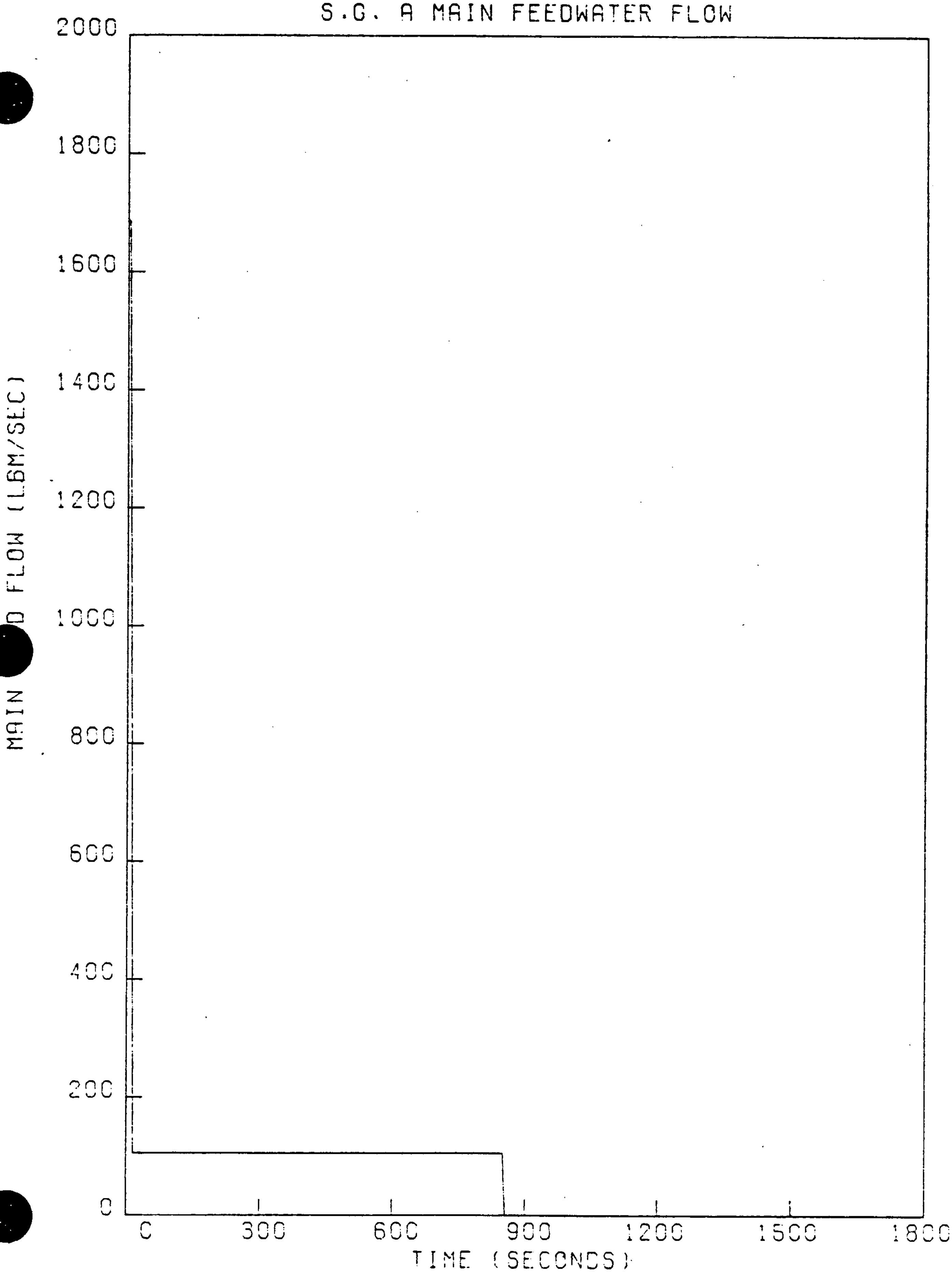


FIGURE 2-86
SONGS 80 PCT PWR LOSS OF FLOW
S.G. B MAIN FEEDWATER FLOW

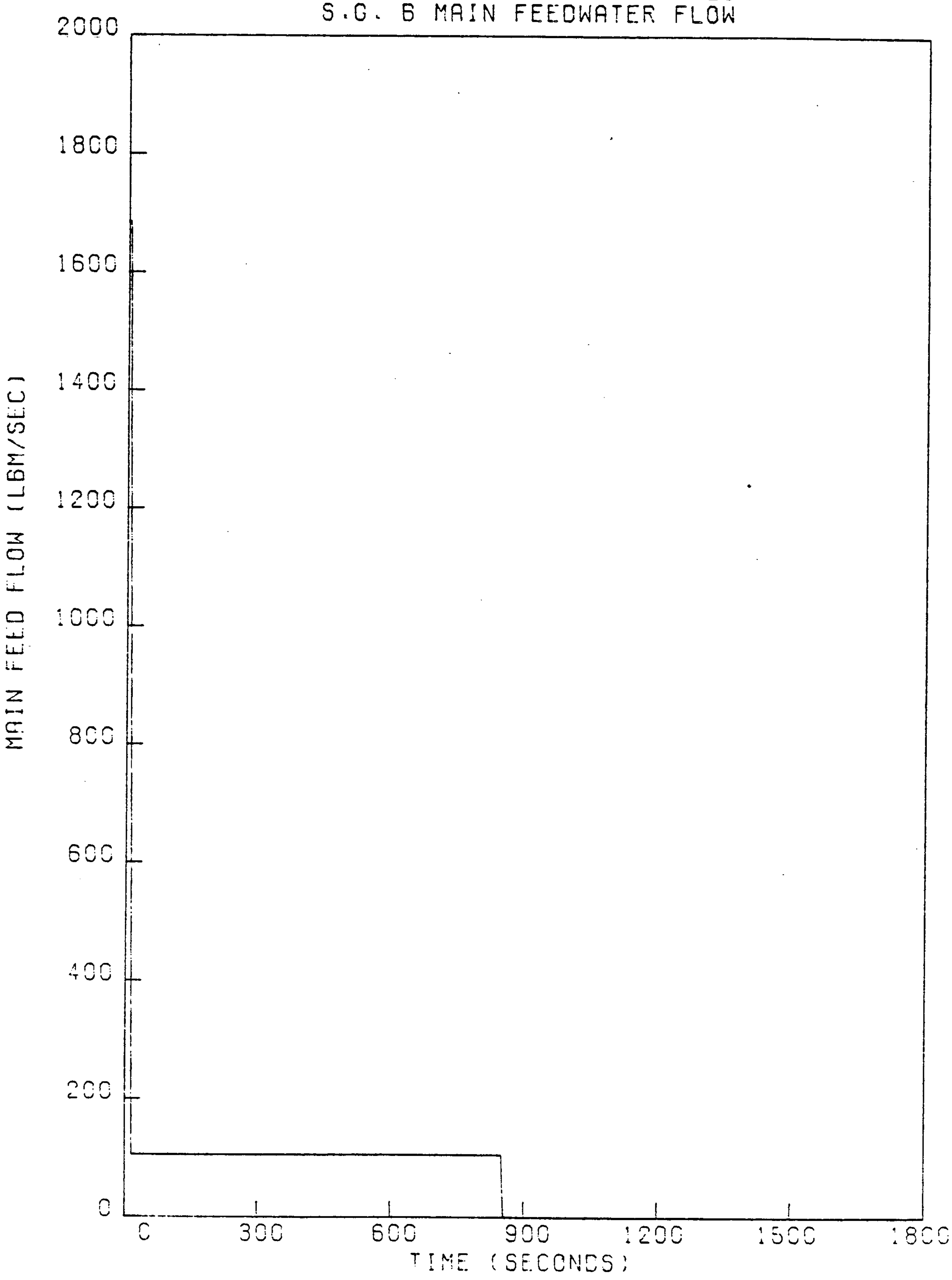


FIGURE 2-87
SONGS 80 PCT PWR LOSS OF FLOW
MAIN FEEDWATER TEMPERATURE

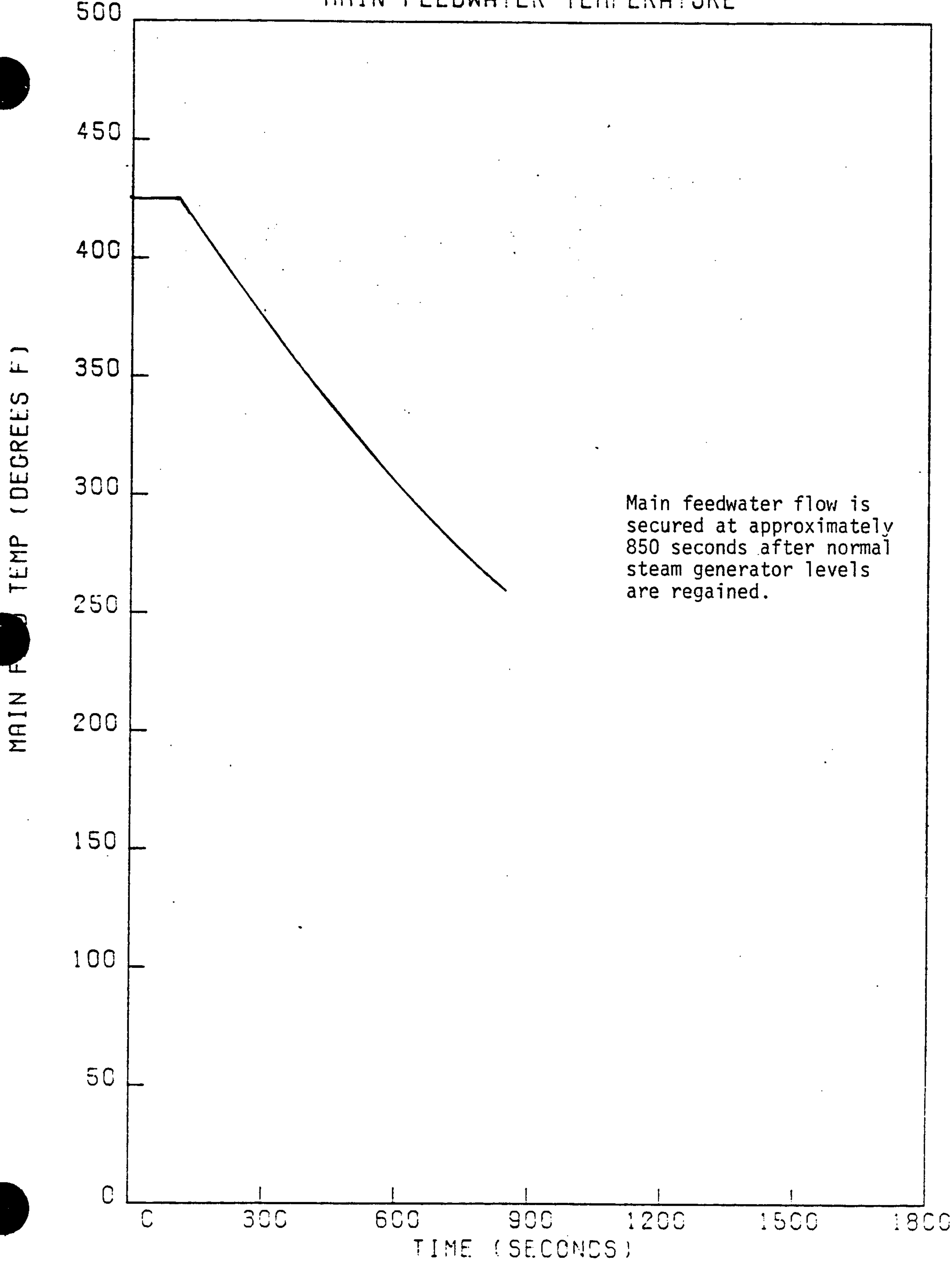


FIGURE 2-88
SONGS 80 PCT PWR LOSS OF FLOW
S.G. A AUX FEEDWATER FLOW

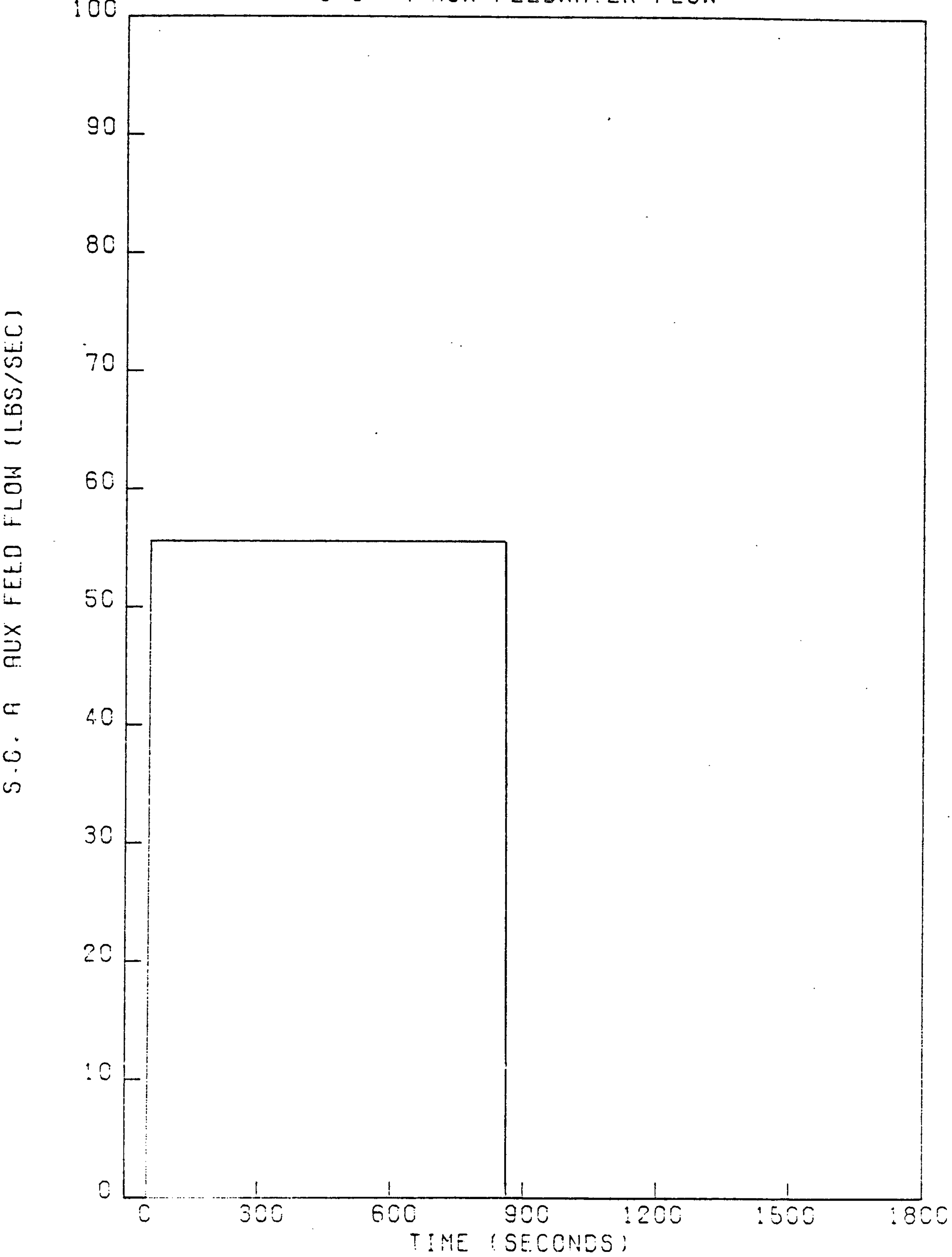


FIGURE 2-89
SONGS 80 PCT PWR LOSS OF FLOW
S.G. B AUX FEEDWATER FLOW

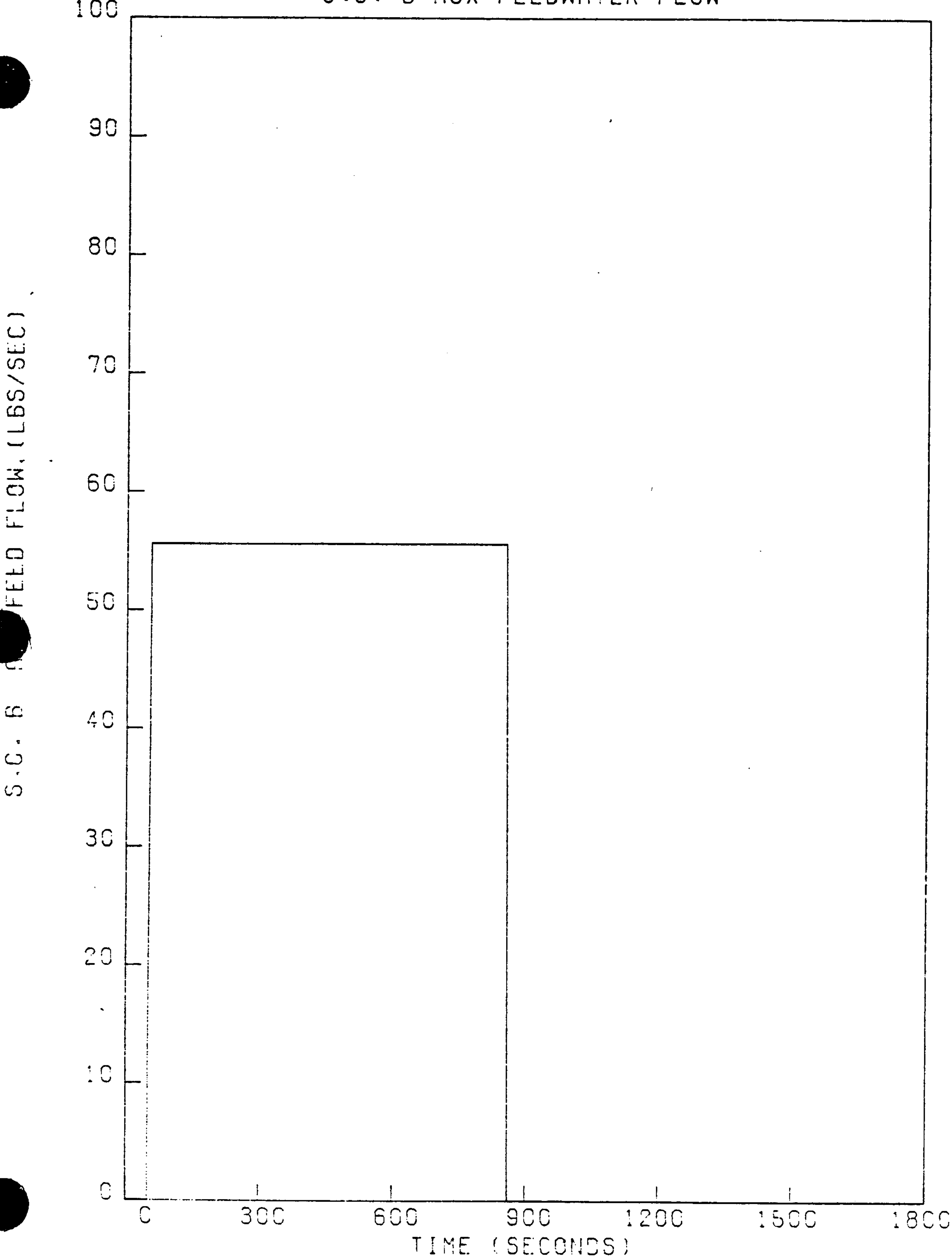


FIGURE 2-90
SONGS 80 PCT PWR LOSS OF FLOW
CHARGING FLOW

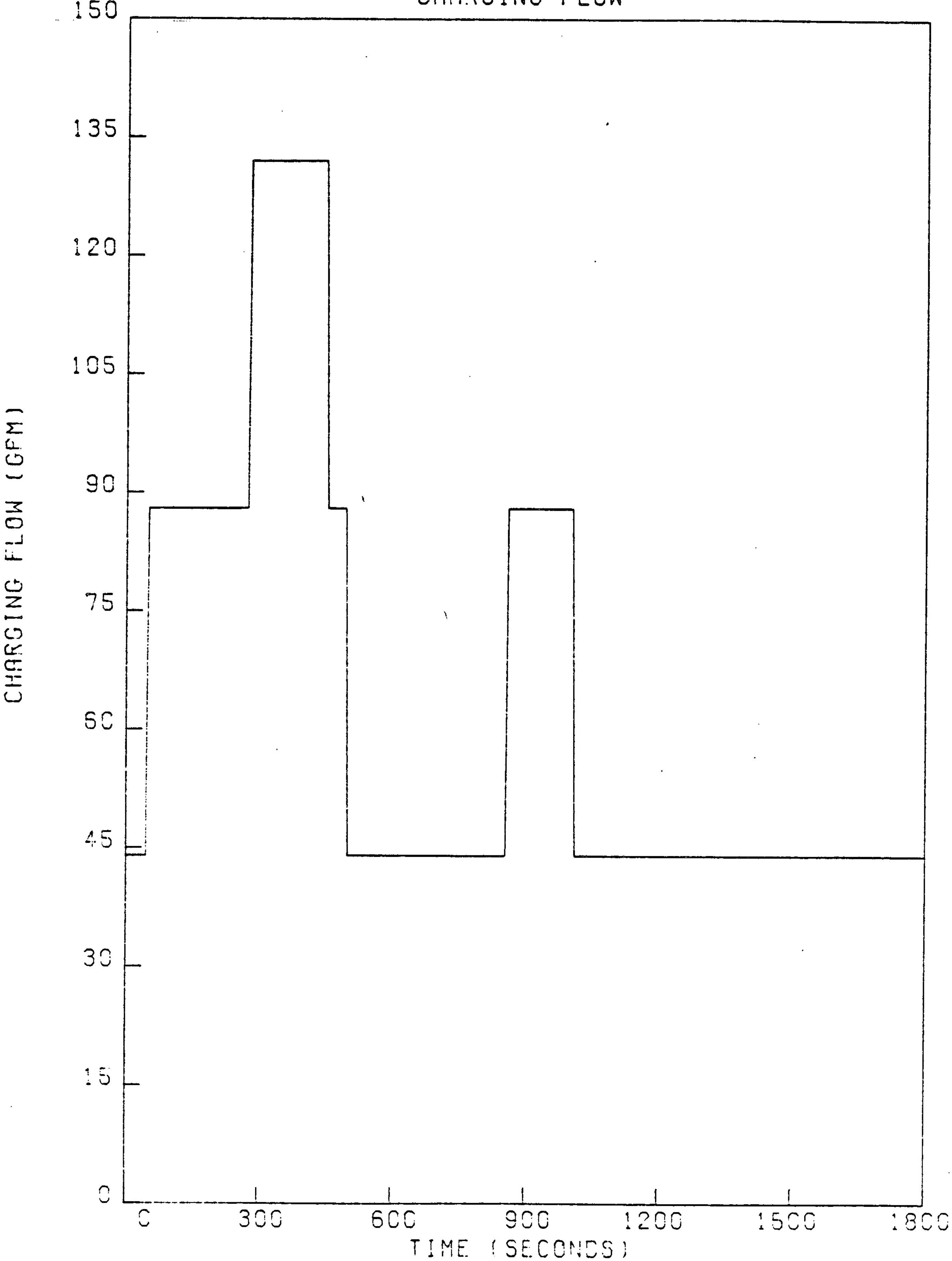


FIGURE 2-91
SONGS 80 PCT PWR LOSS OF FLOW
LETDOWN FLOW

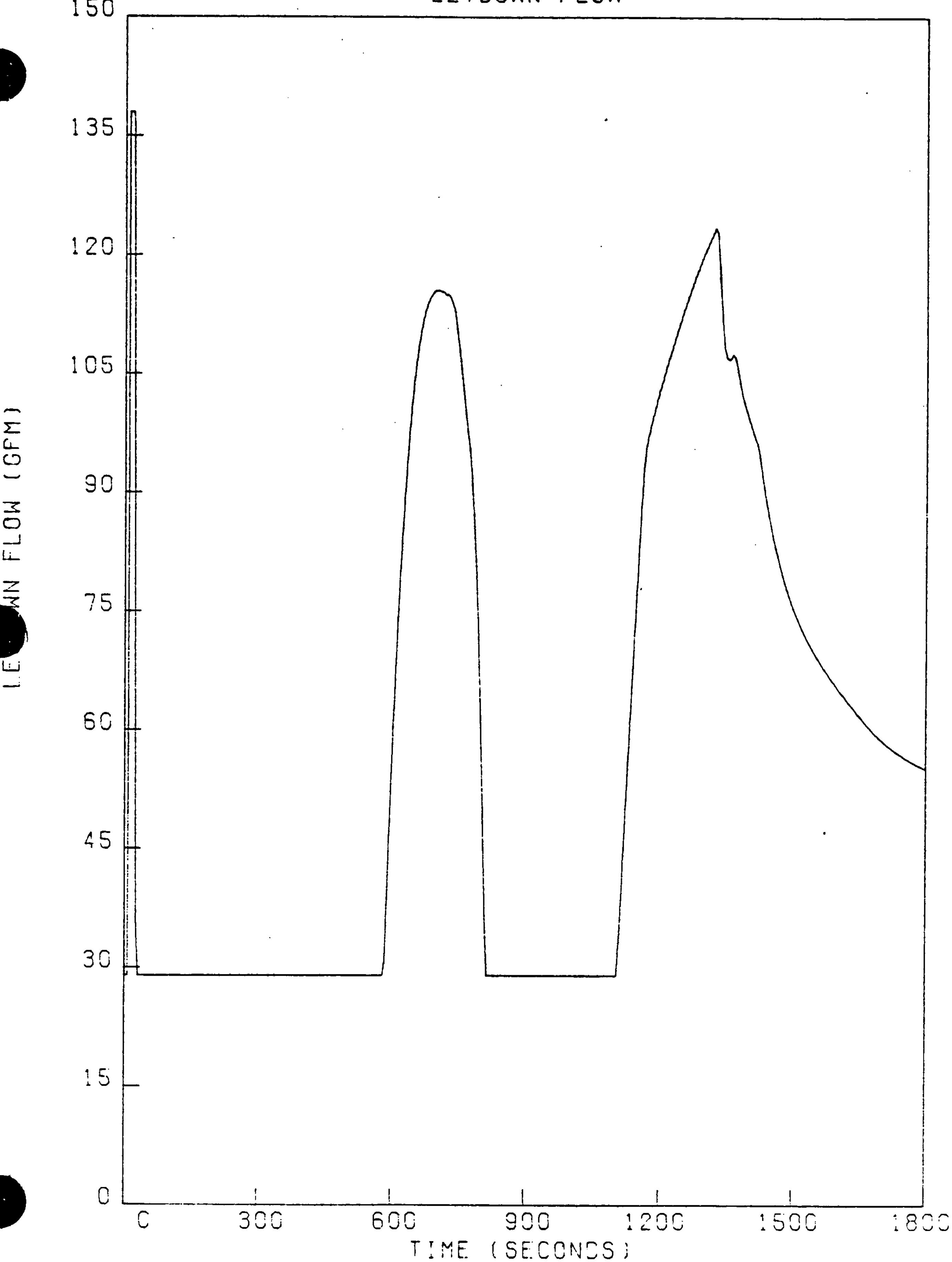
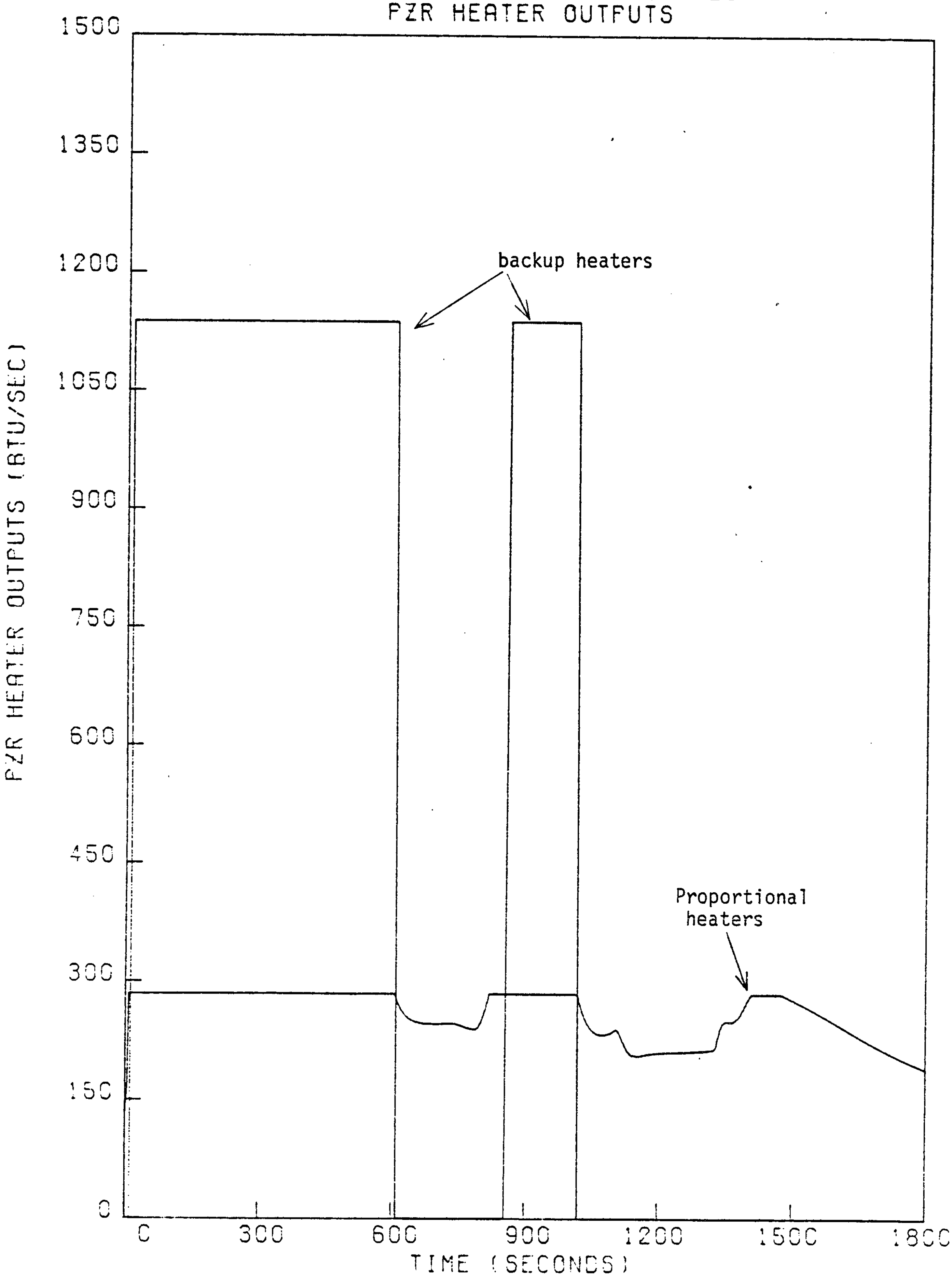


FIGURE 2-92
SONGS 80 PCT PWR LOSS OF FLOW
PZR HEATER OUTPUTS



3.0 SAFETY ANALYSIS

3.1 Introduction

3.1.1 Transient Analysis

This section presents the safety evaluation of design basis events presented in Chapter 15 of SONGS Units 2 and 3 Final Safety Analysis Report (Reference 1). The evaluations examine the operating region which exists during the natural circulation tests. This region includes conditions which are outside the bounds assumed in the Final Safety Analysis Report (FSAR). The impact of these differences on the transient analyses presented in the FSAR are discussed herein. Based on qualitative engineering judgement for most of the events the FSAR analyses results are shown to be more limiting and, thus, bounding. For those events for which it is not possible to make a qualitative judgment, quantitative evaluations were performed.

The events for which either qualitative or quantitative evaluations were performed are listed in Table 15.0-2 of Reference 1. Three different frequency categories considered in Chapter 15 of the FSAR were evaluated. Condition II events (moderate frequency) are evaluated in Section 3.2, Condition III events (infrequent) are evaluated in Section 3.3, and Condition IV events (limiting faults) are evaluated in Section 3.4.

3.1.2 Systems Operation

The systems operation during the course of the transients considered in this section is similar to that described in Section 15.0.2 of Reference 1 with the exception that certain systems will not be available for use during the natural circulation tests. In order to perform the three low-power tests, certain reactor protective system trips indicated in Table 15.0-3 of Reference 1 will be bypassed or

will be assigned different setpoint values. In addition, certain engineered safety features actuation setpoints were altered or bypassed. All exceptions to the FSAR technical specifications of Appendix A of San Onofre Unit 2 Operating License NPF-10 are noted in Section 5.0 of this report. The various operational and test termination criteria that the operator must follow to ensure the plant is operated safely at all times during the natural circulation tests are listed in Section 4.0.

3.1.3 Core and System Performance

3.1.3.1 NSSS Simulation

For those events requiring quantitative evaluations, the Nuclear Steam Supply System (NSSS) response to various initiating events was simulated using the CESEC digital computer program. As described in Reference 4, CESEC provides for the simulation of a Combustion Engineering NSSS. The program calculates the plant response for non-LOCA initiating events for a wide range of operating conditions including natural circulation. The code is a highly flexible analytical tool which models the major plant components for both the primary and secondary systems.

3.1.3.2 Thermal Margin Model

The determination of the occurrence of critical heat flux (CHF) for the low-power natural circulation tests requires methods which differ from the standard TORC/CE-1 methodology (References 5 and 6). Because the low flow rates present during natural circulation are well below the data range of the CE-1 CHF correlation, the Bowring mixed flow cluster dryout correlation (Reference 7) was employed to indicate the margin to CHF. The wide range of data used in the development of this correlation and its accuracy compared to the MacBeth (Reference 8) and

Barnett (Reference 9) correlations at low flow conditions were the main reasons for its selection. The parameter ranges of applicability for this correlation are:

Geometry*:	PTR, BWR, PWR
Pressure:	90 - 2250 psia
Mass Velocity:	0.04 - 3.0 x 10^6 lbm/ft^2-hr
Heated diameter:	0.3 - 1.4 inches
Channel length:	60 - 180 inches
Radial Peaking Factor:	1.0 - 1.32 (Intra-assembly peak)
Axial Peaking Factor:	1.0 - 1.38.

The Bowring correlation predicts CHF based on bundle average conditions and includes factors to compensate for non-uniform axial power distribution effects.

A single closed channel enthalpy rise calculation was used to determine the hot assembly coolant conditions. The hot assembly flow rate was assumed to be equal to the core average flow rate. For natural circulation conditions this is a conservative assumption since the flow redistribution in the core results in more flow being present in the hot assembly and less in the cooler regions. This redistribution of flow results in an essentially constant enthalpy rise in the core.

The approach to critical heat flux at the natural circulation flow rates is most likely to occur at very high qualities. At high qualities the moderator feedback reduces the power level in the upper

* PTR: Pressure Tube Reactor; BWR: Boiling Water Reactor; PWR: Pressurized Water Reactor.

region of the core. For conservatism, however, in the calculation of the critical heat flux ratio (CHFR) this effect was not considered.

After calculating the coolant conditions in the hot assembly, the critical heat flux was calculated at each axial location using the Bowring correlation. The CHFR was then determined by dividing the CHF by the local heat flux of the hottest pin in the assembly. Since the Bowring correlation is based on bundle average conditions, the use of the hot pin instead of the assembly average heat flux in calculating the CHFR is conservative. The same engineering factors employed in the FSAR analysis are also used to account for uncertainties in fuel fabrication in the calculation of the CHFR.

One of the charateristics of natural circulation flow is that flow rate is dependent on the power level. For the proposed tests the power to flow relationship will be limited by one of the following limits on indicated temperature:

$$
\begin{array}{ll}
T_{hot}: & \leq 600^\circ F \\
T_{hot}: & \leq T_{sat} - 20^\circ F \\
\Delta T_{core}: & \leq 58^\circ F
\end{array}
$$

By using the most limiting of these criteria, along with appropriate uncertainties, and applying an energy balance to the core, a maximum allowable power can be determined as a function of pressure, inlet temperature and flow. During the natural circulation tests the actual power will be less than this maximum. Otherwise, one of the temperature limits indicated above will be exceeded and testing will be terminated. Figure 3-1 (p. 157) illustrates this maximum power to flow relationship for a pressure of 2300 psia and inlet temperature of 552°F.

The power at which the Bowring correlation yields a minimum CHFR of 1.26 as a function of flow at 2300 psia and 552°F inlet temperature is

also plotted in Figure 3-1. The 1.26 CHFR limit is based on a 95/95 one sided tolerance limit calculated from the test data reported in Reference 7.

The application of the Bowring correlation using the methods described above yields a conservative indication of CHF conditions. Therefore the actual thermal margin will be greater than that indicated by the Bowring correlation in Figure 3-1.

In conclusion, the limits on indicated temperature stated above will assure that adequate margin to CHF exists during the low-power natural circulation tests.

3.1.3.3 Initial Conditions

The ranges of values of the principal process variables considered in the safety evaluation of the natural circulation tests are listed in Table 3-1 (p. 154). In general, most of the initial conditions for the natural circulation tests form a subset of the conditions specified in Section 15.0.3.2 of the FSAR (Reference 1). Only the the pressurizer pressure for the low-power test at reduced pressures and the core inlet temperature and flow rate for both the low-power and the power ascension tests are outside the ranges of initial conditions specified in the FSAR.

The ranges of initial conditions employed in the safety analyses differ from the initial conditions specified in the natural circulation test procedures. This is because the safety analyses consider all operating conditions possible during the natural circulation tests and accounts for uncertainties. Consequently, the ranges of initial conditions presented in Table 3-1 conservatively envelop those specified in the test procedures.

3.1.4 Radiological Consequences

The assumptions, parameters and calculational methods used to determine the offsite doses that result from postulated accidents are described in Chapter 15 (Section 15.0.5 and Appendix 15B) of Reference 1. The radiological consequences were quantitatively analyzed only for the CEA ejection event described in Section 3.4.8. The radiological consequences of the other events are referenced to the more adverse Chapter 15 FSAR results for these accidents, as appropriate.

3.1.5 Summary of Safety Evaluation

A summary of the evaluation results for the Condition II, III, and IV events is provided in Table 3-2 (p. 155). In the analysis, the following assumptions in addition to the range of initial conditions listed in Table 3-1 were employed:

a) All normal reactor trip and engineered safety features actuation system functions operated per the FSAR analyses with the exception of the following: High logarithmic power level, low pressurizer pressure, low steam generator pressure, high local power density, low DNBR, low reactor coolant flow, safety injection, and emergency feedwater. Refer to Section 5.0 of this report for a detailed explanation of these exceptions.

b) The high logarithmic power level, high local power density, and low DNBR trip functions provided normal protection assumed in the FSAR during reactor startup. Specifically, the high logarithmic power level trip provided protection up to a power level of 10^{-4}%. At 10^{-4}% power the high local power density and low DNBR trip functions became operable. (These last two trips are provided by the core protection calculators (CPCs).) The CPCs provided protection from 10^{-4}% power until they were placed

in bypass at approximately 3% power just prior to reactor coolant pump trip during the low-power tests. Prior to placing the CPCs in bypass, the setpoint of the high linear power trip was reduced to 12% power (less instrument uncertainties).

c) The CEDM control system was in the "off" mode during Tests A1 through A3 except as required to position CEAs following reactor coolant pump trip and the establishment of natural circulation.

d) The CEDM control system was not used in the auto sequential or manual individual mode during Tests A1 through A3 following reactor coolant pump trip and the establishment of natural circulation.

e) The minimum core power during low-power testing was $\geq$ 0.1% to prevent operation under flow stagnation conditions and to ensure protection for CEA withdrawal events. (See Section 3.2.10.)

f) Prior to reactor coolant pump trip for the low-power natural circulation tests, reactor power was $\geq$ 0.1%.

g) For Test A3, core power was approximately 1% and the initial steam generator secondary pressure was approximately 800 psia prior to isolation of Steam Generator 2.

h) Safety injection was available for operation in the event of a loss-of-coolant accident. The safety injection actuation signal based on a low pressurizer pressure was reset to lower values as required during Test A2.

i) The temperature limits of 3.1.3.2 above were maintained.

j) For Tests A1 through A3, all full length and part length CEAs were fully withdrawn with the exception of Group 6 which was $\geq$ 60 inches withdrawn.

k) All CEAs in a group were within $\pm$ 1.5 inches of the group average.

l) The CEAs were positioned per assumption "j" above prior to placing the CPCs in bypass.

The results summarized in Table 3-2 (p. 155) are categorized according to the following bases:

Category	Basis
1	Bounded by FSAR analysis results.
2	Evaluation shows SRP criteria are met.
3	Operator action is required for protection.
4	Probability of occurrence reduced to acceptable limit by restrictions on operating conditions.

Table 3-2 shows that the consequences of most of the events considered are bounded by the FSAR analysis results. For certain events (e.g., increase in main steam flow, startup of an inactive RCP, steam system piping failures, and CEA ejection), the evaluations performed demonstrate that the Standard Review Plan (SRP) acceptance criteria are met. Operator action is required for protection for the CEA misoperation event. For the inadvertent operation of the ECCS, loss of external load, and the turbine trip events, the probability of occurrence is reduced by restrictions on the operating conditions during the tests.

In summary, the following evaluations of the Condition II, III, and IV events indicate that automatic actions as well as manual actions

during the natural circulation tests will prevent violation of the SRP acceptance criteria of Reference 2.

3.2 Condition II - Moderate Frequency Events

3.2.1 Decrease in Feedwater Temperature

The consequences of any decrease in feedwater temperature event that might occur during the planned natural circulation tests would be no more adverse than the consequences of the increase in main steam flow event presented in Section 3.2.3. A large margin to critical heat flux (CHF) would be maintained throughout the event. Primary and secondary coolant system pressures would remain below design pressures. Offsite doses would remain within a small fraction of 10 CFR 100 guidelines.

During the low-power tests, A1 through A3, feedwater will be supplied to the steam generators using the auxiliary feedwater system. This system takes its suction on a condensate storage tank and supplies water which is normally maintained at a temperature of 70°F. Since preheating does not take place, no significant temperature change is likely given an appropriate malfunction in the auxiliary feedwater system.

The analyses presented in Section 3.2.3 covers all possible values of increase in main steam flow up to the maximum possible for that event. The most adverse consequences are found to occur at steam flow rates which are less than this maximum value. Therefore, the consequences of a decrease in auxiliary feedwater temperature cannot be more adverse than the consequences of the event presented in Section 3.2.3.

During the power ascension tests, the consequences of a decrease in feedwater temperature event will be no more adverse than the

consequences presented for this event in the FSAR. The rational for this judgement is the same as that provided in Section 3.2.3 for the increase in main steam flow event and applies to all the increased heat removal events.

3.2.2 Increase in Feedwater Flow

The consequences of any increase in feedwater flow event that might occur during the planned natural circulation tests would be no more adverse than the consequences of the increase in main steam flow event presented in Section 3.2.3. A large margin to critical heat flux (CHF) would be maintained throughout the event. Primary and secondary coolant system pressures would remain below design pressures. Offsite doses would remain within a small fraction of 10 CFR 100 guidelines.

During the three low-power tests, feedwater will be supplied by the auxiliary feedwater system. The total capacity of this system is approximately 5% of the normal full power feedwater flow rate, and therefore the potential cooldown due to the maximum possible increase in feedwater flow is less than the maximum cooldown obtained from the increase in main steam flow event. The consequences of this event would thus be no more adverse than the consequences presented in 3.2.3.

The analyses presented in Section 3.2.3 covers all possible values of increase in main steam flow up to the maximum possible for that event. Therefore, the consequences of an increase in feedwater flow cannot be more adverse than the consequences of the event presented in Section 3.2.3.

During the power ascension tests, the consequences of an increase in feedwater flow event will be no more adverse than the consequences presented in the FSAR. The rationale for this judgement is the same as that provided in Section 3.2.3 for the increase in main steam flow event and applies to all the increased heat removal events.

3.2.3 Increase in Main Steam Flow

3.2.3.1 Introduction

The increase in heat removal by the steam generators as a result of increased main steam flow is defined as any unplanned increase in steam generator flow, other than a steam line break. Should an increase in main steam flow occur during the planned natural circulation tests, there would be no violation of the acceptance critertia set forth in the SRP (Reference 2). A large margin to critical heat flux (CHF) would be maintained throughout the event. Primary and secondary coolant system pressures would remain below design pressures. Offsite doses would remain within a small fraction of 10 CFR 100 guidelines.

An increase in main steam flow may be caused by one of the following incidents of moderate frequency:

A. Failure in the turbine bypass control system which would result in an opening of one or more of the bypass valves. The flowrate of each valve is approximately 11% of the full power turbine flowrate. There are four turbine bypass valves for a total of approximately 45% flow.

B. An inadvertent opening of an atmospheric dump valve or steam generator safety valve caused by operator error or failure within the valve itself. Each atmospheric dump and safety valve can release approximately 5% of the full power turbine flowrate. (This event is treated separately in Section 3.2.4. However, as discussed there, the consequences with respect to fuel performance and primary and secondary system pressures are included within those of the analyses presented here.)

The planned natural circulation tests are organized into two parts:

a) low-power tests, for which the reactor will be critical throughout the tests, and

b) power ascension tests, for which the reactor will trip at the beginning of the tests in response to the reactor coolant pump (RCP) coastdown, or operator action.

The conditions for the initial portion of the power ascension tests have already been included in the analyses presented in the FSAR (Reference 1). RCP coastdown concurrent with or during the course of the event was considered for each event or event with concurrent single failure. Should an increase in heat removal event occur after the beginning portion of the tests, i.e., after reactor trip, the consequences could be no worse than those presented in the FSAR since the reactor would be subcritical at the time of event initiation. The analyses presented in the FSAR demonstrate that no SRP criteria are violated for any excess heat removel event. Therefore, there would be no violation of SRP criteria should an increase in heat removal event occur during the power ascension tests.

A generic sequence of events and systems operations for any increase in main steam flow event which might occur during one of the low-power tests is presented in Sevtion 3.2.3.2.

The consequences of a postulated increase in heat removal event during any of the low-power tests have been evaluated with respect to SRP criteria. With the most adverse set of initial conditions, within the space of operating conditions permitted by the test procedure, and the most adverse valve opening, the evaluation shows that a large margin to CHF would be maintained throughout the event. This evaluation is presented in Section 3.2.3.3.

An explanation of why the primary and secondary system pressures would remain at, or below, design pressures during any increase in energy removal event is given in Section 3.2.3.4.

Offsite doses for the increase in steam flow event as a result of the opening of one or more of the turbine bypass valves will not be more adverse than those for the inadvertent opening of a steam generator atmospheric dump or safety valve presented in Section 3.2.4. Fuel performance for both events is the same. However, the event described in Section 3.2.4 results in a greater amount of steam release to the atmosphere. Therefore the reasons why the offsite dose for the increase in main steam flow event remain a small fraction of the 10 CFR 100 guidelines are presented in Section 3.2.4.

3.2.3.2 Sequence of Events and Systems Operation

While operating at steady-state, low power, natural circulation test conditions with steam flow via partially opened turbine bypass valves to the condenser, it is assumed that due to a spurious signal or operator error the opening of one or more of the turbine bypass valves is increased. The resulting increased main steam flow increases the rate of heat removal by the steam generators, causing cooldown of the RCS. Due to the negative moderator temperature coefficient, core power will increase. At the same time the increased temperature difference between the steam generator inlet and outlet on the primary side, caused by the increase in steam demand, and the increased temperature difference between core inlet and outlet, caused by the increased power, will cause an increase in the natural circulation flow. If the increase in valve opening is larger than some small specific test and initial condition dependent value a low steam generator pressure trip will limit the reactor power. If the increase in valve opening is smaller than the foregoing value the reactor power will be limited by the high reactor power trip or by the moderator and Doppler reactivity feedback. If the reactor is tripped

on a low steam generator pressure signal the power increase before trip will be small due to a large time delay between decrease in steam generator temperature and decrease in core temperature under natural circulation conditions. Therefore, the maximum reactor power is obtained for the largest valve opening which does not result in a low steam generator pressure trip. Due to the long loop times the operator has adequate time to trip the reactor prior to any automatic actuations. However, the foregoing assumes that the reactor operator does not take action to limit the power increase until after the maximum power is reached. Thereafter, acting upon a variety of available indications of parameters outside the limits set for the test (neutron power high, hot leg temperature high, and core delta T high), the operator trips the reactor and closes the main steam isolation valves. He then brings the plant to a stable condition, controlling steam flow with the main steam isolation bypass valves or the steam generator atmospheric dumps.

3.2.3.3 Evaluation of Consequences with Respect to Fuel Performance

The consequences of any increase in heat removal event which might occur during the low-power natural circulation tests may be evaluated in the power versus flow space presented in Figure 3-2. (p. 158)

During steady-state, natural circulation flow is, to first-order, proportional to NSSS power to the one-third power. A variety of natural circulation test results from C-E plants (References 10 and 11) indicate that at 1% power the natural circulation reactor vessel flow will be greater than 1000 lbm/sec. Thus the functional relationship, percent power equals the cube of the quantity reactor vessel flow in lbm/sec divided by 1000, or $Q = (W/1000)^3$ bounds the steady-state powers that can exist with any steady-state natural circulation flow rate. This bound is shown on Figure 3-2. Also shown on this figure are the bounds for acceptable fuel performance and for test operation, as in Figure 3-1, and the bound imposed by the high reactor power trip.

For the natural circulation tests the loop times are large due to the low flow rates. As a result, transient changes in flow occur in approximately the same manner as changes in flow between one steady-state power level and another. Further, all increase in heat removal events result in a decrease in "cold side" (i.e., that part of the flow path from the approximate midpoint of the steam generator tubes to the core mid-plane) temperatures before any change in core power. Since the driving head for natural circulation flow comes from the difference in density between the cold side and the hot side of the coolant flow path, increases in flow will precede increases in core power for increase in heat removal events. Thus, any increase in heat removal event which is initiated on or to the right of the $Q = (W/1000)^3$ curve of Figure 3-2 will have a trajectory which falls to the right of this curve. Postulated increase in steam flow events during each of the low-power tests have been simulated using the CESEC computer code (Reference 4). In all cases the entire event trajectories fall to the right of the $Q = (W/1000)^3$ curve. Therefore, the $Q = (W/1000)^3$ curve forms an upper bound for core power at any reactor vessel flow rate for increase in heat removal events.

Additional bounds exist in the power versus flow space of Figure 3-2. As indicated in Section 3.2.3.2, the maximum reactor power will be limited by a low steam generator pressure trip, a high reactor power trip, Doppler and moderator reactivity feedbacks, and operator action. Of these only the bounds introduced by high reactor power trip and operator action are shown in Figure 3-2. However, the other bounds will actually limit core power before the high reactor power trip in most, if not all, cases.

The value for the high reactor power trip used in Figure 3-2 includes both uncertainties and the maximum decalibration due to coolant shadowing. Coolant shadowing decalibration is caused by an increased density in the reactor vessel downcomer during cooldown transients.

The steam generator low pressure trip setpoint corresponds to a temperature no more than 65°F lower than the initial steam generator secondary side temperature. Therefore, the maximum possible amount of coolant shadowing decalibration that can occur prior to reactor trip is that amount corresponding to a 65°F decrease in reactor vessel downcomer temperature.

In conclusion, the bounds imposed by the $Q = (W/1000)^3$ curve and the high reactor power trip shown in Figure 3-2 preclude any approach to CHF during increase in heat removal event. This conclusion is further strengthened by consideration of the bounds imposed by the low steam generator pressure trip, Doppler and moderator feedbacks and operator action. Finally, it should be noted that the fuel performance bound in Figure 3-2 is based on a maximum pressure and temperature. As either of these parameters decrease, the fuel performance bound would move further to the left. A large margin to CHF would be maintained throughout any increase in steam flow event.

3.2.3.4 Evaluation of Consequences with Respect to Over-Pressure

Increase in main steam flow events are initially depressurization events. After reaching a new equilibrium power level for valve openings too small to cause depressurization of the secondary side to the low steam generator pressure trip setpoint, some repressurization can occur if a high reactor power trip does not occur or if no operator action is taken. Even if operator action is not taken and no reactor trip occurs earlier, a reactor trip will occur on high pressurizer pressure at the high pressurizer pressure setpoint. The auxiliary feedwater system and the primary and secondary system safety valves are designed to relieve well in excess of the energy available from decay heat after operation at full power, while maintaining primary and secondary pressures at, or below, design pressure. During the planned natural circulation tests decay heat levels will be more than an order of magnitude lower than the values for which these

systems are designed. Therefore, primary and secondary pressures will remain below design pressures during any increase in heat removal events that might occur during these tests.

3.2.3.5 Conclusion

Should an increase in main steam flow event occur during the planned natural circulation tests a large margin to CHF would exist throughout the event, primary and secondary system pressures would remain below design pressures, and offsite doses would remain within a small fraction of 10 CFR 100 guidelines.

3.2.4 Inadvertent Opening of a Steam Generator Atmospheric Dump Valve

With respect to fuel performance and primary and secondary system pressures, the consequences of any inadvertent opening of a steam generator atmospheric dump or safety valve event that might occur during the planned natural circulation tests would be no more adverse than those consequences for the increased main steam flow event presented in Section 3.2.3. A large margin to CHF would be maintained throughout the event. Primary and secondary coolant system pressures would remain below design pressures. With respect to radiological releases, the consequences of this event would be no more adverse than the consequences presented for this event in the FSAR (Reference 1). Offsite doses would remain within a small fraction of 10 CFR 100 guidelines.

An atmospheric dump valve (ADV) may be inadvertently opened by the operator or may open due to an active failure in the control system. A steam generator (SG) safety valve may be opened only as a result of a passive valve failure. The consequences of the inadvertent opening of the ADV will bound the consequences of the inadvertent opening of the SG safety valve, since they both relieve steam at the same flowrate (approximately 5% of full power turbine flowrate). The

potential for cooldown due to the opening of the ADV is less than the maximum cooldown attainable from an increase in main steam flow event (Section 3.2.3).

During the power ascension tests, the consequences of an inadvertent opening of a steam generator atmospheric dump valve or safety valve event will be no more adverse than the consequences presented in the FSAR. The rationale for this judgement is the same as that provided in Section 3.2.3 for the increase in main steam flow event and applies to all the increased heat removal events.

The discussion presented in Section 3.2.3 covers all possible cases of increase in main steam flow during the low-power tests up to the maximum possible for that event. Therefore, the consequences with respect to fuel performance and primary and secondary system pressures of an inadvertent opening of a steam generator atmospheric dump or safety valve cannot be more adverse than the consequences of the event presented in Section 3.2.3.

Any radiological releases that might occur during an inadvertent opening of a steam generator atmospheric dump or safety valve during the low-power natural circulation tests would be less than for the corresponding event in the FSAR (Reference 1) since:

a. No fuel failure is predicted for either event;

b. Primary and secondary system activities would be substantially lower for the time in life during which the tests will be conducted than the values assumed for the analyses presented in the FSAR; and

c. Due to the low power levels at which the tests will be conducted, less decay heat would have to be removed by possible steam release to the atmosphere during any post-event cooldown than the amount of decay heat that was assumed in the FSAR analyses.

3.2.5 Loss of External Load

The loss of external load need not be considered during the low-power and power ascension natural circulation tests. Normal turbine-generator operation will not take place when natural circulation conditions are established.

3.2.6 Turbine Trip

The turbine trip event need not be considered during the low-power and power ascension natural circulation tests. Normal turbine-generator operation will not take place when natural circulation conditions are established.

3.2.7 Loss of Condenser Vacuum

The loss of condenser vacuum (LOCV) event may occur due to a failure of the circulating water system to supply cooling water, failure of the main condenser evacuation system to remove non-condensible gases, or excessive in-leakage of air through the gland packing.

Upon the LOCV the turbine bypass valves close and are blocked from opening. For the power ascension tests, the main feedwater pumps, if in operation, trip on pump turbine driver exhaust low vacuum. The loss of subcooled feedwater flow and steam flow result in a gradual heatup and pressurization of the steam generators (SGs) and the reactor coolant system (RCS).

For the power ascension tests, the transient progresses much more slowly than the LOCV described in Section 15.2.1.3 of the FSAR (Reference 1) since the power mismatch after reactor trip is limited to the amount of decay heat present. The operator responds to the increasing hot leg temperature, pressurizer pressure, and secondary pressure by terminating the test and restoring forced reactor coolant

flow. The RCS heat removal is accomplished by the operator by manually controlling the SG atmospheric dump valves (ADVs) and the auxiliary feedwater system. In the absence of operator action, the secondary pressure slowly rises to the main steam safety valve (MSSV) opening setpoint. The MSSVs may cycle open and shut, and maintain the plant in a quasi-steady state. The primary safety valves and the MSSVs provide overpressure protection for the primary and secondary systems, respectively, and the auxiliary feedwater system automatically restores the SG levels.

For the LOCV event during the power ascension tests, the primary and secondary pressures will remain well within 110% of the design pressure. In response to indications identified in the test procedures, prompt operator action will terminate the test, and restore forced reactor coolant flow. These indications would imply the existence of impending inadequate core cooling due to a disruption of natural circulation. However, quantitative evaluations performed have indicated that no disruption of natural circulation occurs for this event.

The sequence of events, system responses, and event consequences following the LOCV event occurring during the low-power natural circulation tests are the same as those described above for the power ascension tests with the following exceptions:

1) There will be no loss of feedwater following LOCV since feedwater is supplied by the auxiliary feedwater system during the low power tests. (Note that this statement also holds true for the loss of offsite power test after the main feedwater is shutoff.)

2) The operator will respond to the increasing hot leg temperature, pressurizer pressure, and secondary pressure by tripping the reactor in addition to terminating the test and restoring forced flow.

3.2.8 Loss of Normal AC Power

The loss of normal AC power may result from either a loss of the external grid or a loss of the onside AC distribution system. Under the natural circulation conditions present during the low-power and power ascension tests, the loss of AC event is no more limiting than the loss of condenser vacuum event described above.

3.2.9 Partial Loss of Forced Reactor Coolant Flow

This event is bounded by the total loss of forced reactor coolant flow described in Section 3.3.10.

3.2.10 Uncontrolled CEA Withdrawal from a Subcritical or Low Power Condition

This event is of most concern during operation at power levels in the range of 10^{-4}%. As discussed in Section 15.4.1.1 of the FSAR (Reference 1), at power above 10^{-4}% reactivity feedback mechanisms prevail and provide a dampening effect on the severity of the transient. Power levels in the range of 10^{-4}% power during the natural circulation tests occur only during startup when the reactor coolant pumps are operating. The consequences of an uncontrolled CEA withdrawal from a low-power condition during the power ascension tests is bounded by the current FSAR discussion. During the low-power tests, reactor power will be $\geq$ 0.1% whenever the plant is in natural circulation. The FSAR analysis bounds the consequences of this event occurring during the startup portion of the low-power tests prior to the establishment of natural circulation. The CEA withdrawal occuring at power levels $\geq$ 0.1% is discussed in Section 3.2.11.

3.2.11 Uncontrolled CEA Withdrawal at Power

During the low-power tests, Bank 6 can be inserted up to 60%. No other banks can be inserted. The low-power test procedures dictate

that the Control Element Drive Mechanism Control System (CEDMCS) will be in the "off" mode unless the operator is intentionally moving rods. When the CEDMCS is in the "off" mode, it does not respond to any external rod motion signals. Only in the unlikely event of internal logic failures could rod motion occur. The procedures also dictate that close operator surveillance will be maintained regarding both CEA position and power level. The combination of these two factors render a CEA withdrawal during the short time period of the low-power tests extremely unlikely. The operator will be instructed to trip the plant if a CEA withdrawal occurs. Furthermore, the high linear power trip will be set to occur at or before 12% actual core power to provide additional protection.

In the unlikely event that a CEA withdrawal did occur, the following is a discussion of the impact of the transient on thermal margin. During a CEA withdrawal under natural circulation conditions, the flow increases as the power increases. The maximum power would be approximately 12% due to the high power trip. The amount of flow relative to the power level is the primary thermal-hydraulic condition affecting thermal margin to CHF. The flow at a power level of 12% is calculated to be sufficient during a CEA withdrawal to provide adequate thermal margin to CHF.

During the power ascension tests, the core protection calculators (CPC) will be operating. The CPCs will provide protection during this time period to prevent violation of the specified acceptable fuel design limit (SAFDL). The CEA withdrawal under these conditions is bounded by the FSAR analysis.

3.2.12 CEA Misoperation

The events in this category are as follows: single CEA withdrawal; single full length CEA drop; full length subgroup drop; single part length CEA drop; part length subgroup drop.

During the low-power tests, the single CEA withdrawal will be protected against in the same manner as the subgroup or group withdrawal. (Refer to Sections 3.2.10 and 3.2.11 for a discussion of the rationale involved.) Since the part length rods will not be inserted during the low-power tests, all the rod drop events result in an initial decrease in core power. Initial Doppler and moderator reactivity feedback will tend to mitigate the decrease in power. The local power will not increase above its pre-drop value in any area of the core until additional moderator reactivity feedback occurs. Additional moderator reactivity feedback will occur when the colder coolant cycles back to the core inlet plenum and begins rising through the core. Due to the natural circulation flowrate of the tests, the coolant will take at least four (4) minutes, and probably more, to complete the cycle. Constant operator monitoring of the rod position will be required by the test procedures. Therefore, adequate time is available for the operator to react to the rod drop prior to any local power increase. The procedures require the operator to terminate testing in the event of abnormal rod motion, including rod drop and rod ejection.

During the power ascension tests, the core protection calculators (CPC) will be operating. A CEA misoperation could possibly occur only during the time period prior to reactor trip. The CPCs provide protection during this time period to prevent violation of the SAFDL. Furthermore, the initial condition space considered in the FSAR analysis includes the initial conditions specified for the power ascension tests. Thus, for the power ascension tests, the CEA withdrawal and drop events are bounded by the FSAR analysis.

3.2.13 CVCS Malfunction (Inadvertent Boron Dilution)

During the low-power and power ascension natural circulation tests, the consequences of a chemical and volume control system (CVCS) malfunction that results in an unplanned boron dilution are bounded by

the consequences presented in Section 15.4.1.4 of the FSAR (Reference 1). For these tests, the maximum possible dilution rate is no more than the rate assumed in the FSAR analysis. In addition to the boron dilution alarm, protection is provided by the high linear reactor power trip (at 12% actual core power) for the low-power tests and constant monitoring of key parameters for both the low-power and power ascension tests.

3.2.14 Startup of an Inactive Reactor Coolant System Pump

During the low-power and power ascension natural circulation tests, the inadvertent startup of an inactive reactor coolant system pump results in a decrease in core coolant temperature. The decreasing temperature may cause power to increase due to a negative moderator temperature coefficient (MTC). However, the change in power would be small because of the small MTC which exists at the beginning of core life. Furthermore, for the low-power tests the power level maintained is very low ($\leq$ 5% indicated power) and should power increase beyond the 12% value for any reason, a reactor trip would result. For the power ascension tests, natural circulation conditions are established subsequent to reactor trip. Therefore, for both the low-power and power ascension tests, a large margin to critical heat flux (CHF) will be maintained due to the increasing flow to power ratio.

3.2.15 CVCS Malfunction (Increase in RCS Inventory)

During the low-power and power ascension natural circulation tests one charging pump is in operation, and the other two are in standby. A malfunction in the CVCS may result in the startup of a second charging pump producing an unplanned increase in RCS inventory.

During the natural circulation tests, the consequences of a CVCS malfunction that produces an unplanned increase in reactor coolant

inventory are bounded by those discussed in Section 15.5.1.1 of the FSAR (Reference 1). This is due to the initial pressurizer level being lower than for the FSAR case, allowing more time for the operator to detect the event. Also, the constant operator monitoring of pressurizer level provides additional assurance that the event consequences will be mitigated.

3.2.16 Inadvertent Operation of the ECCS

The reactor coolant system pressure will be maintained above the shut-off head of the safety injection pumps and the opening pressure of the safety injection tanks throughout the duration of the natural circulation tests. Constant operator monitoring during these tests assures the RCS pressure being maintained above the safety injection pump shutoff head. For the low-power natural circulation test at reduced pressures, the safety injection actuation setpoint is manually reset before RCS pressure is reduced below the safety injection actuation setpoint. Therefore, during the natural circulation tests, inadvertent operation of the emergency cooling system (ECCS) will not be possible.

3.3 Condition III - Infrequent Events

3.3.1 Decrease in Feedwater Temperature with a Concurrent Single Failure

The event combinations of decrease in auxiliary feedwater temperatures with each possible single failure of an active component have been evaluated for the low-power tests. Since none of the single failures considered in the FSAR alter the cooldown rate, the transient behavior of the RCS will not be altered by combining a single failure with the decrease in auxiliary feedwater temperature event. Therefore, the consequences of the event in combination with a single failure during the low-power tests are no more adverse than the event consequences described in Section 3.2.1.

During the power ascension tests, the consequences of a decrease in feedwater temperature with a concurrent single failure of an active component will be no more adverse than the consequences presented in the FSAR. The rationale for this judgement is the same as that provided in Section 3.2.3 for the increase in main steam flow event and applies to all the increased heat removal events.

3.3.2 Increase in Feedwater Flow with a Concurrent Single Failure

The event combinations of increase in auxiliary feedwater flow with each possible single failure of an active component have been evaluated for the low-power tests. Since none of the single failures considered in the FSAR alter the cooldown rate, the transient behavior of the RCS will not be altered by combining a single failure with the increase in auxiliary feedwater flow event. Therefore, the consequences of the event in combination with a single failure are no more adverse than the event consequences described in Section 3.2.2.

During the power ascension tests, the consequences of an increase in feedwater flow with a concurrent single failure of an active component will be no more adverse than the consequences presented in the FSAR. The rationale for this judgement is the same as that provided in Section 3.2.3 for the increase in main steam flow event and applies to all the increased heat removal events.

3.3.3 Increase in Main Steam Flow with a Concurrent Single Failure

The event combinations of increase in main steam flow with each possible single failure of an active component have been evaluated for the low-power tests. Since none of the single failures considered in the FSAR alter the cooldown rate, the transient behavior of the RCS will not be altered by combining a single failure with the increase in main steam flow event. Therefore, the consequences of the event in combination with a single failure are no more adverse than the event consequences described in Section 3.2.3.

During the power ascension tests, the consequences of an increase in main steam flow with a concurrent single failure of an active component will be no more adverse than the consequences presented in the FSAR. The rationale for this judgement is the same as that provided in Section 3.2.3 for the increase in main steam flow event and applies to all the increased heat removal events.

3.3.4 Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with Concurrent Single Failure

The event combinations of inadvertent opening of a steam generator atmospheric dump or safety valve with each possible single failure of an active component have been evaluated for the low-power tests. Since none of the single failures considered in the FSAR alter the cooldown rate, the transient behavior of the RCS will not be altered by combining a single failure with the inadvertent opening of a steam generator atmospheric dump or safety valve event. Therefore, the consequences of the event in combination with a single failure are no more adverse than the event consequences described in Section 3.2.4.

During the power ascension tests, the consequences of an inadvertent opening of a steam generator atmospheric dump valve with a concurrent single failure of an active component will be no more adverse than the consequences presented in the FSAR. The rationale for this judgement is the same as that provided in Section 3.2.3 for the increase in main steam flow event and applies to all the increased heat removal events.

3.3.5 Loss of External Load with a Single Failure

The loss of external load need not be considered during the low-power and power ascension natural circulation tests. Normal turbine-generator operation will not take place when natural circulation conditions are established.

3.3.6 Turbine Trip with a Single Failure

The turbine trip event with a single failure need not be considered during the low-power and power ascension natural circulation tests. Normal turbine-generator operation will not take place when natural circulation conditions are established.

3.3.7 Loss of Condenser Vacuum with a Single Failure

The single failures considered in the FSAR (Reference 1) for the decreased heat removal events are as follows:

1. Loss of all AC power on turbine trip.
2. Failure of pressurizer level measurement channel associated with the pressurizer level control system (PLCS).
3. Failure of one auxiliary feedwater (AFW) pump.
4. Failure of the steam bypass control system (SBCS).

The loss of all AC power on turbine trip need not be considered during the low-power and power ascension tests since normal turbine-generator operation will not take place when natural circulation conditions are established.

A failure of the pressurizer level measurement channel associated with the pressurizer level control system (PLCS) need not be considered since the operator will be continuously monitoring the pressurizer level during the tests. The operator will manually operate the PLCS, if it is not able to automatically maintain the desired pressurizer level.

The failure of one AFW pump does not result in a total loss of feedwater flow, even though the feedwater flow to both steam generators may be temporarily interrupted. However, through operator or automatic action, auxiliary feedwater flow to the steam generators will be restored using the unaffected auxiliary feedwater pump or pumps.

If the initiating event is a loss of auxiliary feedwater event, and the single failure is a failure of the electric-driven auxiliary feedwater pump to start, then there is a potential for loss of two of the three AFW pumps. In that case, the operator would have to rely on one electric-driven AFW pump for heat removal. Only one of the SGs will be supplied with feedwater, and the water level in the other SG will be slowly decreasing. However, if the reactor is already tripped (for power ascension tests), or subsequent to a reactor trip (for low-power tests), the small amount of decay heat can be satisfactorily removed by one SG.

If the initiating event is a loss of main feedwater flow, and the single failure is the failure of one AFW pump to start, then the event will be less limiting than above, since both SGs can be fed by the remaining AFW pumps.

A failure in the SBCS resulting in an inadvertent opening of the turbine bypass valves is bounded by the analyses of the steam line breaks. For all of the decreased heat removal events, the SBCS is assumed to be unavailable, and consequently, a failure in the SBCS resulting in the closure of the turbine bypass valves is not relevant for the decreased heat removal events.

3.3.8 Loss of Normal AC Power with a Single Failure

The discussion of single failures described in Section 3.3.7 applies to the Loss of Normal AC Power event with a single failure.

3.3.9 Loss of Feedwater Flow

The loss of feedwater flow is defined to be the reduction of feedwater flow without a corresponding reduction in steam flow. A loss of feedwater flow may result from the loss of feedwater pumps or condensate pumps, or the closure of feedwater control or isolation valves.

During the natural circulation period of the power ascension tests, the feedwater is provided to the SGs either by the main feedwater system or the auxiliary feedwater system depending on the test. For these tests, a decrease in feedwater flow occurring during the post trip, natural circulation period results in a gradual depletion of SG secondary inventory. The operator promptly responds to the decreasing SG level(s) by restoring feedwater flow using either the electric-driven AFW pump or the turbine-driven AFW pump and terminating the test. There is no significant effect on RCS parameters, and hence no effect on the fuel and pressure boundary integrity.

During the low-power natural circulation tests feedwater is assumed to be provided by the AFW system under manual control. Under these conditions, the loss of feedwater flow will result in the decrease in feedwater flow to both steam generators. The operator will respond to the decreasing SG level by restoring feedwater flow to both steam generators using the electric-driven AFW pumps and by terminating the test by tripping the reactor. In the absence of operator action the reactor would trip on low SG level, and an emergency feedwater signal would start the electric-driven auxiliary feedwater pumps and restore feedwater flow to both steam generators. Although the original decrease in feedwater flow may have resulted from a failure within the auxiliary feedwater system, redundancy in the design of the auxiliary feedwater system assures that adequate feedwater flow for post-trip heat removal can be provided.

The decrease in feedwater flow event results in negligible effect on the RCS response and, hence, has a negligible effect on the RCS pressure boundary and fuel integrity.

3.3.10 Total Loss of Forced Reactor Coolant Flow

The total loss of forced reactor coolant flow would result in natural circulation. Natural circulation conditions will provide adequate

core cooling during the tests. For all the tests, the reactor coolant pumps are initially running and the power is initially less than 100%. A total loss of forced reactor coolant flow can only occur prior to the establishment of natural circulation flow. The results of this event during the tests will be no more adverse than what is presented in Section 15.3.2.1 of the FSAR (Reference 1).

3.3.11 Partial Loss of Forced Reactor Coolant Flow with a Concurrent Single Failure

No single failure could be identified which significantly affects the results of the partial loss of forced reactor coolant flow addressed in Section 3.2.9.

3.3.12 CVCS Malfunction with a Concurrent Single Failure

The single failures that impact the results of a CVCS malfunction are (1) startup of the third charging pump and (2) failure of the letdown system. Both result in a more rapid increase in the reactor coolant system (RCS) inventory.

During the low-power and power ascension natural circulation tests, the consequences of a CVCS malfunction with a concurrent single failure are bounded by those discussed in the Section 15.5.2.1 of the FSAR (Reference 1). This is due to the initial pressurizer level being lower than the FSAR case, allowing more time for the operator to detect the event. Also, the constant operator monitoring of pressurizer level during the tests provides additional assurance that the event consequences will be mitigated.

3.4 Condition IV - Limiting Faults

3.4.1 Steam System Piping Failures

Steam system piping failures are defined as breaks in the main steam system and are referred to as steam line breaks (SLBs). Should an SLB occur during the planned natural circulation tests the primary reactor coolant system would be maintained in a safe status and there would be no hazard to the public. A large margin to CHF would be maintained throughout the event. There would be no post-trip return to criticality. Primary and secondary coolant system pressures would remain below design pressures. Offsite doses would remain within a small fraction of 10 CFR 100 guidelines.

The planned natural circulation tests are organized into two parts:

a) The low-power tests, for which the reactor will be critical throughout the tests, and
b) the power ascension tests, for which the reactor will trip at the beginning of the tests in response to the reactor coolant pump (RCP) coastdown or operator action.

The conditions for the initial portion of the power ascension tests have already been included in the analyses presented in the FSAR (Reference 1). The consequences of RCP coastdown concurrent with or during the course of the event were evaluated for SLBs. Should an SLB occur after the beginning portion of the tests, i.e., after reactor trip, the consequences would be no worse than those presented in the FSAR since the reactor would be subcritical at the time of event initiation. The analyses presented in the FSAR demonstrate that no SRP criteria are violated for any SLB event. Therefore, there would be no violation of SRP criteria should an SLB event occur during the power ascension tests.

The consequences of steam line breaks for the low-power tests have been evaluated with respect to criteria on over-pressure, fuel performance, and radiological releases.

Steam line breaks are initially depressurization events. During the portion of the transient after steam generator dryout and before operator action, some repressurization can occur due to safety injection pump flow, decay heat addition, and heat transfer from the hotter walls and structure of the reactor coolant system (RCS). However, the auxiliary feedwater system and the primary and secondary system safety valves are designed to relieve in excess of the energy available from these sources while maintaining primary and secondary pressure at, or below, design pressure. Further, during the low-power tests decay heat levels will be more than an order of magnitude lower than the design value.

Degradation in fuel performance can occur during SLB initiated events either during the portion of the transient prior to and during reactor trip (henceforth referred to as the pre-trip portion) or during the post-trip return-to-criticality or approach-to-criticality portion of the transient (henceforth referred to as the post-trip portion).

The potential for degradation in fuel performance during the pre-trip portion of the transient is, in general, maximum for a SLB of less than the maximum possible break area. Very large breaks result in early reactor trip on low steam generator pressure. It has been found that the break area which maximizes the potential for pre-trip degradation in fuel performance for the low-power tests is less than the effective area of the turbine bypass valves. Therefore, the results of the SLB event which maximizes pre-trip degradation in fuel performance are identical to the results of the increased main steam flow event which maximizes pre-test degradation in fuel performance (Section 3.2.3). A large margin to CHF would be maintained throughout the event. (Note that this evaluation was performed with the assumption of the highest worth CEA stuck in the fully withdrawn position).

Post-trip degradation in fuel performance can only occur if there is a post-trip return-to-power. Due to the relatively small magnitude of the (negative) moderator temperature coefficient of reactivity at beginning of core one conditions, the core would remain at least 1.5% $\Delta\rho$ subcritical even if it were to be cooled down to 212°F following a SLB event during the low-power tests. (The determination of this amount of sub-criticality assumes the CEA of the greatest worth stuck in the fully withdrawn position, a 10% increase in the slope of the moderator reactivity versus coolant temperature function, a 10% decrease in the net CEA worth, a 15% increase in the slope of Doppler reactivity versus fuel temperature function, and does not credit the effect of safety injection boron.) If a steam line break were to occur upstream of the MSIV for the operating steam generator during Test A3, Verification of Natural Circulation with Reduced Heat Removal Capacity, it would be necessary for the operator to terminate auxiliary feedwater flow to the affected steam generator to prevent the eventual cooling of the RCS below 212 °F. However, even if no safety injection boration were to occur, the operator would have at least 20 minutes in which to terminate auxiliary feedwater flow before there would be any return-to-cruticality. Further, based upon the FSAR analyses, sufficient safety injection boration would occur during such an overcooling event to preclude return-to-criticality for any amount of overcooling. Therefore, there would be no post-trip return-to-power and, consequently, no post-trip degradation in fuel performance for any SLB event which might occur during the planned tests.

Since there would be no degradation in fuel performance during any portion of an SLB initiated event for the planned natural circulation tests, the radiological consequences would not be greater than those calculated for the limiting dose SLB event presented in Table 15.1-19 of the FSAR (Reference 1). The doses presented in this table are small fractions of the guideline values of 10 CFR 100. For the low-power natural circulation tests, the primary and secondary system

activities would be substantially lower than those assumed in the FSAR. Therefore the offsite doses would be even smaller fractions of the 10 CFR 100 guideline values than those presented in Table 15.1-19 of the FSAR.

In summary, should an SLB occur during the planned natural circulation tests there would be no violation of criteria for over-pressure, fuel performance, or radiological releases.

3.4.2 Loss of Feedwater Flow with a Single Failure

The discussion of single failures provided in Section 3.3.7 applies to the loss of feedwater flow with a single failure.

3.4.3 Feedwater Line Break

A feedwater line break (FWLB) may occur due to a pipe failure in the main or auxiliary feedwater system. The FWLB may produce a total loss of feedwater flow and a blowdown of one steam generator. The result of the SG blowdown can either be an RCS heatup or cooldown, depending on the break size and the enthalpy of the blowdown fluid. The feedwater line breaks resulting in an RCS cooldown are bounded by steam line break analyses and are not considered in this discussion.

A FWLB not resulting in the blowdown of a SG, e.g., a break upstream of a reverse flow check valve, degenerates into a loss of feedwater flow event. The consequences of this event are described in Section 3.3.9.

For breaks which result in a heatup of the RCS and the blowdown of one SG, the FWLB event described in Section 15.2.3.1 of the FSAR is more severe than a FWLB occurring during the low-power and power ascension natural circulation tests. Under natural circulation conditions, only the lower portion of the U-tube region is effectively transferring

heat from the primary to the secondary fluid. Thus, for the low-power tests, the SG inventory drops below the low SG level reactor trip setpoint before any significant decrease in heat removal capability occurs. Therefore, the mismatch in power generation and heat removal at the secondary side is much more severe for the FSAR analysis than for a FWLB during the low-power tests. For the power ascension tests, the reactor has already been tripped and only small amounts of decay heat need to be removed. Additionally, as indicated earlier, under natural circulation conditions, only the lower portion of U-tube region is effectively transferring heat from the primary to secondary fluid, and thus, degradation in primary to secondary heat transfer will not occur for a longer time in comparison to the case presented in the FSAR analysis. Therefore, the mismatch in power generation and heat removal will be more severe for the FSAR analysis than for a FWLB during the power ascension tests, also.

In conclusion, primary and secondary pressures will remain well within 110% of the design pressure during the feedwater line break event for the low-power and power ascension tests. Operator action is specified to terminate the testing in response to indications identified in the test procedures that would imply the existence of impending inadequate core cooling due to disruption of natural circulation. However, quantitative evaluations performed have indicated that no disruption of natural circulation occurs during the FWLB event for the low-power and power ascension tests.

3.4.4 Reactor Coolant Pump Shaft Seizure

For the natural circulation tests, this event is meaningful only before coastdown of the reactor coolant pumps (RCPs). After coastdown of the RCPs, the fluid momentum is insufficient to move the pump impellers. Therefore, the pump impellers will remain stationary once natural circulation conditions are established. However, should this

event be hypothetically assumed to occur after natural circulation conditions are established, no decrease in core flow would result, and the safety of the plant would not be compromised.

Prior to coastdown of the RCPs, greater initial thermal margin exists than was assumed in the FSAR Chapter 15 analysis. This is principally due to the fact that all the tests are performed at less than 100 percent power. Thus, the results of a reactor coolant pump shaft seizure are bounded by the analysis in Section 15.3.3.1 of the FSAR.

3.4.5 Single Reactor Coolant Pump Sheared Shaft

As indicated in Section 3.4.4, this event is meaningful only before coastdown of the RCPs. Prior to coastdown of the RCPs, greater initial thermal margin exists than was assumed in the analysis presented in Section 15.3.3.2 of the FSAR (Reference 1). This is mainly due to the fact that all the tests are performed at less than 100% power. Thus, the results of a single reactor coolant pump sheared shaft event are bounded by the analysis in Chapter 15 of the FSAR.

3.4.6 Complete Loss of Forced Reactor Coolant Flow with a Concurrent Single Failure

No single failure could be identified which significantly affects the results of the total loss of forced reactor coolant flow described in Section 3.3.10.

3.4.7 Inadvertent Loading of a Fuel Assembly in an Improper Position

During the low-power and power ascension natural circulation tests, the consequences of the inadvertent loading of a fuel assembly in an improper position is bounded by the consequences presented in Section 15.4.3.1 of the FSAR. The power level experienced during these tests

is lower than that assumed in the FSAR analyses. As a result, the impact of any increase in radial power peaking due to the misloading would be less severe than that indicated in the FSAR. Additionally, constant monitoring of neutron power during the tests provides assurance that the event consequences will be mitigated.

3.4.8 CEA Ejection

The CEA Ejection (CEAE) event is postulated to occur due to a mechanical failure of the CEA drive mechanism (CEDM) or the CEDM nozzles such that system coolant pressure ejects the CEA and its drive shaft to the fully withdrawn condition. This is a low probablity event so that the likelihood of this event during the short time period of natural circulation testing is considerably reduced from that normally associated with the event.

The testing covered by this evaluation is separable into two distinct sections for consideration of the CEAE: The power ascension tests and low-power critical operation tests. The power ascension tests will begin with the reactor operating normally at the specified power and with the CEA insertion restricted by the technical specifications. Therefore, for the initial part of these tests, the FSAR analyses will bound any possible CEAE event. Natural circulation conditions are established for these tests subsequent to reactor trip due to either operator action, or a CPC trip (see Section 2.1). If any CEA fails to insert upon reactor trip, the test will be terminated by operator action. Thus, for the remainder of the test period, the plant will be fully rodded and substantially subcritical. A CEAE event from this condition would not introduce sufficient reactivity to approach criticality so that there would be no power excursion. The effect of the loss of primary coolant during the event is covered by the discussions of Section 3.4.11.

The low-power natural circulation tests will be conducted with the plant operating at or near critical conditions but within a reduced operating space. Of particular interest to the CEAE event analysis is the restriction of the CEA position to no more than 60% insertion of the lead bank. This constrains the possible reactivity insertion to substantially less than delayed neutron fraction. (The BOC zero power data of Tables 15.4-24 and 15.4-26 of the FSAR give the maximum lead bank single CEA worth as 0.292% $\Delta\rho$* and the delayed neutron fraction as 0.007234.)

The constraint on ejected worth will prevent the high power spike characteristic of limiting low-power CEAE events. Instead, the power takes a prompt jump of 3% to 4% of rated thermal power from its initial level of 5% of rated power and then increases with a period characteristic of delayed neutron kinetics. That is, the rate of power increase would be sufficiently slow as to preclude substantial overshoot between receipt of a trip signal and shutdown. This increase will be terminated by either a manual trip, or a high linear power at or below 12% rated power.

In comparison to the FSAR analysis (Section 15.4.3 of Chapter 15), the smaller power increase will result in a smaller insurge of fluid to the pressurizer, and a smaller primary-to-secondary heat imbalance. Consequently, the peak RCS pressure will be lower than that presented in the FSAR for the CEA ejection event.

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*This value is for a single CEA from the lead bank (Location C) and is a conservative estimate for these test conditions since 1) the data of Table 15.4-26 of the FSAR is for the full length ejection from a heavily rodded condition, and 2) during the low-power tests it would only be possible to have an ejection from a lightly rodded partially inserted condition.

Evaluations of margin to critical heat flux (CHF) have been performed for the operating conditions that are allowed for the low-power tests. In particular, an estimate at the limits of allowed pressure and temperature, and at 12% of rated power and 3% of rated flow (stable flow is expected to be higher) indicated that substantial margin to CHF would be available. Since this is approximately the power at which the reactor will trip during a CEAE event, CHF would be unlikely except, possibly, in the region immediately surrounding the ejected CEA. It is judged to be highly unlikely that this region would be extensive enough that as much as 9% of the fuel pin would experience CHF. Thus the FSAR CEAE analysis, for which 9% of the fuel pins were predicted to experience CHF, bounds the consequences of a CEAE event during these tests. Note that, although C-E's methodology does not equate CHF with cladding failure, the pins that experience CHF are assumed to experience cladding failure in the FSAR analysis for calculation of radiological release in accordance with Regulatory Guide 1.77.

Even though the number of pins that are assumed to fail are bounded by the FSAR case, the low fission product inventory due to the low burnup at the time of the low-power tests would allow substantially higher assumed fuel failures without violation of the 10 CFR 100 limits. In fact, it has been estimated that 100% of the fuel pins could fail without violating the dose guidelines.

The substantially lower peak power reached during this event relative to the full flow CEAE events coupled with the similar coolant temperatures implies substantially lower deposited enthalpies and fuel temperatures. Neitner of these parameters will approach the limiting values described for the full flow CEAE in Section 15.4.3.2 of the FSAR (Reference 1).

3.4.9 Primary Sample or Instrument Line Break

A primary sample or instrument line break represents a violation of the primary system piping outside the containment building. Therefore, the line break chosen for analysis in Section 15.6.3.1 of the FSAR (Reference 1) is a double-ended break of a letdown line outside containment. Since the letdown line penetrates the containment building, the major concern with a break in this line outside containment is the offsite accident dose. The offsite accident dose is dependent on two parameters, namely, the amount of fluid leaked out through the break and the activity associated with this fluid.

The amount of fluid leaked out through the break is primarily dependent on the RCS pressure. During the low-power and power ascension natural circulation tests the RCS pressure is maintained at or below 2300 psia. This value is about 100 psi lower than that was employed in the FSAR analysis for this event. Consequently, the integrated leakage through the break will be lower than that is presented in the FSAR. Additionally, since the operators are instructed to constantly monitor NSSS conditions during the natural circulation tests, any rapid reduction in RCS pressure due to the break will be readily noticed by the operators and, in response, mitigating actions will be taken. Clearly, the operator actions will lead to even lower integrated RCS leakage.

During the low-power and power ascension natural circulation tests, the primary coolant activity is expected to be considerably lower than the technical specification limit employed in the FSAR analysis. This is because of the very low burnup of the fuel.

Due to lower integrated leakage through the break and lower primary coolant activity in relation to those employed in the FSAR analyses, the offsite accident doses for the letdown line break outside

containment will be considerably lower than those presented in the FSAR. Thus, the FSAR results bounds the consequences of this event during the natural circulation tests.

3.4.10 Steam Generator Tube Rupture

In Section 15.6.3.2.2 of the FSAR (Reference 1), a steam generator tube rupture (SGTR) event with a loss of AC power is presented. The radiological consequences of this event are more limiting than a SGTR event without a loss of AC power. The radiological consequences of the event are dependent on the amount of primary-to-secondary leakage, the steam releases through the main steam safety valves (MSSVs), and the primary and secondary system activity levels.

The primary-to-secondary leakage is primarily dependent on the RCS pressure. During the low-power and power ascension natural circulation tests, the RCS pressure is maintained at or below 2300 psia which is about 100 psi lower than the initial RCS pressure assumed in the FSAR analyses of the steam generator tube rupture event. Consequently, the integrated primary-to-secondary leakage will be lower than that is indicated in the FSAR analysis. Additionally, since operators are constantly monitoring NSSS conditions during the natural circulation tests, rapidly decreasing RCS pressure (due to primary-to-secondary leakage) and high radiation alarms at the secondary side will alert the operators to take mitigating actions sooner than assumed in the FSAR analysis. Again, the primary-to-secondary leakage will be reduced, as a result.

During a SGTR event with a loss of AC power the MSSVs may open due to the unvailability of the condenser. The amount of MSSV steam release will depend on the initial core power level and whether or not auxiliary feedwater flow is available during the event. For the low-power tests, the core power level is maintained at less than 5% indicated power and auxiliary feedwater is used throughout the tests.

Therefore, loss of AC power will not impact supply of feedwater to the steam generators. Since the FSAR analysis assumed an initial power level of 102% and a delay for the actuation of auxiliary feedwater subsequent to loss of AC power, the amount of MSSV release will be substantially smaller for the low-power tests than that presented in the FSAR for the steam generator tube rupture event with a loss of AC power.

Two different power ascension tests are planned for the natural circulation test program. For both tests, the main feedwater is used prior to reactor trip. For the loss of offsite power test, the auxiliary feedwater is employed subsequent to reactor trip and loss of power. Additionally, for this test the core power level is considerably lower ($\sim$ 20%) than that employed in the FSAR analysis of the SGTR event. Therefore, the MSSV steam release for a SGTR event during this test will be substantially smaller than that presented in the FSAR.

For the other power ascension test, namely, the 80% power with loss of flow test, the main feedwater is used throughout the test. Should a loss of AC power occur during a SGTR event for this test, there may be a delay in the actuation of the auxiliary feedwater. However, constant operator monitoring during the test will assure prompt actuation of auxiliary feedwater. Additionally, the initial power level for this test is lower ($\sim$ 80%) than that employed in the FSAR analysis. Therefore, the MSSV release for a SGTR event will be lower than that presented in the FSAR analysis for this power ascension test also.

During the natural circulation tests, the primary and secondary system activities will be considerably lower than the technical specification limits employed in the FSAR analyses. These limits are based on long term full power operation with 1% failed fuel.

The low-power natural circulation test to be performed with one steam generator isolated is a special case. If the rupture occurs in the isolated steam generator, then there is only minimal danger of activity release to the atmosphere, and the operator, based on rapidly decreasing RCS pressure and high radiation alarms in the secondary side, can take necessary mitigating actions. If rupture occurs in the unisolated steam generator, then the operator will first return the isolated steam generator to service, and then isolate the ruptured steam generator. The observations made above concerning primary-to-secondary leakage, MSSV releases, and primary and secondary activity levels hold true for this scenario also.

Due to lower primary-to-secondary leakage, lower MSSV releases, and lower primary and secondary system activities in relation to those presented in the FSAR analyses, the offsite accident doses for the steam generator tube rupture event with a loss of AC power will be considerably lower than those presented in the FSAR. Thus, the FSAR results bound the consequences of this event during the natural circulation tests.

3.4.11 Loss-Of-Coolant Accident

In Sections 6.3.3 and 15.6.3.3 of the FSAR (Reference 1) the emergency core cooling system (ECCS) performance was demonstrated to be in compliance with the Acceptance Criteria of 10 CFR 50.46. Both large and small break loss-of-coolant accident (LOCA) transients were analyzed using the C-E Evaluation Models in conformance with Appendix K to 10 CFR 50. These analyses were performed assuming 102% of full power and 120% of the decay heat produced by infinite operation at full power. The low-power tests will be performed with an indicated power level no greater than 5% of full power. The low pressurizer pressure (LPP) setpoints for reactor trip and SIAS will be in the automatic mode and will not require operator action for reactor trip and safety injection actuation during a postulated LOCA. For the low-

power natural circulation test at reduced pressures (see Section 2.1.2) the LPP setpoint will be lowered as the primary system pressure is reduced, being always available in the event of a LOCA. For the low-power test for which one steam generator will be isolated (see Section 2.1.3), the other steam generator would be available as a heat sink and is more than sufficient to remove the small amount of decay heat generated by a core. The low power and reduced decay heat represent operating conditions bounded by the FSAR analyses. Therefore, a LOCA during the low power tests would be less severe than the LOCA transients analyzed in the FSAR. The FSAR analyses are conservative for application to these low-power tests.

For the power ascension tests the reactor may be operated at a power level as high as 80% of full power. This power level is still below the power level assumed in the FSAR LOCA analyses mentioned above. As in the low-power tests, the LPP setpoint for safety injection actuation will always be available. For the power ascension test which simulates a total loss of AC power (see Section 2.1.5) both diesel generators will be running and power will be available to the safety injection system. In the FSAR analyses, loss of offsite power is assumed and the ECCS is shown to be capable of cooling the core even if only one diesel generator is running. For the power ascension test in which the system pressure is reduced to the shutdown cooling entry pressure (see Section 2.1.5), the LPP setpoint will be reset as necessary as the system pressure is reduced, but will be available to generate an SIAS automatically. Therefore, the power ascension tests represent much less severe operating conditions than assumed in the FSAR analyses. The FSAR analyses are conservative for application to the power ascension tests.

3.4.12 Waste Gas System Leak or Failure

The consequences of this initiating event for the conditions of 105% design core power level (3560 MWt) operation with 1% failed fuel for

an extended period of time has been analyzed in Section 15.7.3.1 of FSAR Chapter 15 (Reference 1). Since the plant will be operated at lower power levels for shorter intervals of time, the inventory of noble gases and iodines and the associated activities will be lower than those considered in the FSAR analysis. Therefore, the consequences of this initiating event during the natural circulation tests are bounded by the results provided in Section 15.7.3.1.

3.4.13 Radioactive Waste System Leak or Failure

The analysis presentd in Section 15.7.3.2 of the FSAR (Reference 1) for this event assumed a radwaste system activity based on 1% failed fuel and full power operation for an extended period of time. During the natural circulation tests, since the plant will be operated at lower power levels for shorter intervals of time, the radwaste system activity will be considerably lower. Therefore, the consequences of this initiating event during the natural circulation tests are bounded by the results presented in Section 15.7.3.2 of FSAR Chapter 15.

3.4.14 Postulated Radioactive Releases Due To Liquid Tank Failures

The discussion in Section 15.7.3.3 of the FSAR (Reference 1) is applicable for this accident.

3.4.15 Design Basis Fuel Handling Accidents

Fuel handling accidents are relevant during refueling operations. Furthermore, the analysis of the design basis fuel handling accident is presented in Section 15.7.3.4 of the FSAR (Reference 1). Therefore, consequences of a fuel handling accident need not be considered for the planned natural circulation tests.

3.4.16 Spent Fuel Cask Drop Accidents

The discussion in Section 15.7.3.5 of the FSAR (Reference 1) is applicable for this accident.

3.4.17 Anticipated Transients Without Scram (ATWS)

The discussion in Section 15.8 of the FSAR (Reference 1) is applicable for these transients.

Table 3-1

Initial Conditions Employed
in the Safety Evaluation(1)

Parameter	Range: Tests A1, A2, & A3	Range: Test B1&B2	Range: Test B3
Reactor power (%)	0.1 - 7(2)	~ 20	~80
T_c (°F)	498 - 552	498 - 549	498 - 554
T_h (°F)	$\leq$ 602	$\leq$ 602	$\leq$ 602
ΔT_{core} (°F)	$\leq$ 60	$\leq$ 60	$\leq$ 60
Subcooled margin (°F)	$\geq$ 18	$\geq$ 48	$\geq$ 48
Pzr. pressure (psia)	1650 - 2300	$\leq$ 2300	$\leq$ 2300
Pzr. level (%)	31 - 54	31 - 35	31 - 56
SG pressure (psia)	550 - 1025	$\leq$ 1000	$\leq$ 1000
SG level (%)	69 $\pm$ 5	< 74	< 74

(1) The ranges of initial conditions used in the safety evaluation envelop those specified in the test procedures.

(2) A maximum power level of 5.5% was employed in the CEA withdrawal and ejection events.

TABLE 3-2

SUMMARY OF SAFETY EVALUATION(1)

EVENT	A1	A2	A3	B1&B2	B3
Condition II					
Decrease in feedwater temperature	2	2	2	1	1
Increase in feedwater flow	2	2	2	1	1
Increase in main steam flow	2	2	2	1	1
Inadvertent opening of an ADV	1,2	1,2	1,2	1	1
Loss of external load	4	4	4	4	4
Turbine trip	4	4	4	4	4
Loss of condenser vacuum	1,2	1,2	1,2	1,2	1,2
Loss of normal AC power	1,2	1,2	1,2	1,2	1,2
Partial loss of flow	1	1	1	1	1
CEA withdrawal at low power	1	1	1	1	1
CEA withdrawal at power	2	2	2	1	1
CEA misoperation	3,2	3,2	3,2	1	1
CVCS malfunction (boron dilution)	1	1	1	1	1
Startup of inactive RCP	2	2	2	2	2
CVCS malfunction (RCS inventory)	1	1	1	1	1
Inadvertent operation of ECCS	4	4	4	4	4
Condition III					
Decrease in feedwater temp. with SF	2	2	2	1	1
Increase in feedwater temp. with SF	2	2	2	1	1
Increase in Main steam flow with SF	2	2	2	1	1
Inadvertent opening of ADV with SF	1,2	1,2	1,2	1	1
Loss of external load with SF	4	4	4	4	4
Turbine trip with SF	4	4	4	4	4
Loss of condenser vacuum with SF	1,2	1,2	1,2	1,2	1,2
Loss of normal AC Power with SF	1,2	1,2	1,2	1,2	1,2
Loss of feedwater flow	2	2	2	2	2
Total loss of RCS flow	1	1	1	1	1
Partial loss of RCS flow with SF	1	1	1	1	1
CVCS malfunction with SF	1	1	1	1	1

TABLE 3-2 (Cont'd)

SUMMARY OF SAFETY EVALUATION

EVENT	TEST A1	A2	A3	B1&B2	B3
Condition IV					
Steam system piping failures	1,2	1,2	1,2	1	1
Loss of feedwater flow with SF	2	2	2	2	2
Feedwater line break	1,2	1,2	1,2	1,2	1,2
Reactor coolant pump shaft seizure	1	1	1	1	1
Single RCP sheared shaft	1	1	1	1	1
Total loss of RCS flow with SF	1	1	1	1	1
Improper loading of fuel assembly	1	1	1	1	1
CEA ejection	1,2	1,2	1,2	1,2	1,2
Primay sample or instr. line break	1	1	1	1	1
SG tube rupture	1	1	1	1	1
Loss-of-coolant accident	1	1	1	1	1
Waste gas system leak or failure	1	1	1	1	1
Radwaste system leak or failure	1	1	1	1	1
Liquid tank failures	1	1	1	1	1
Fuel handling accidents	1	1	1	1	1
Spent fuel cask drop	1	1	1	1	1
ATWS	1	1	1	1	1

(1) Key to Table 3-2:

1 - Bounded by FSAR analysis results.

2 - Evaluation shows SRP criteria are met.

3 - Operator action is required for protection.

4 - Probability of occurrence reduced to acceptable limit by restrictions on operating conditions.

Figure 3-1

Fuel Performance and Test Operation
Domains in the Reactor Power vs Vessel Flow Space

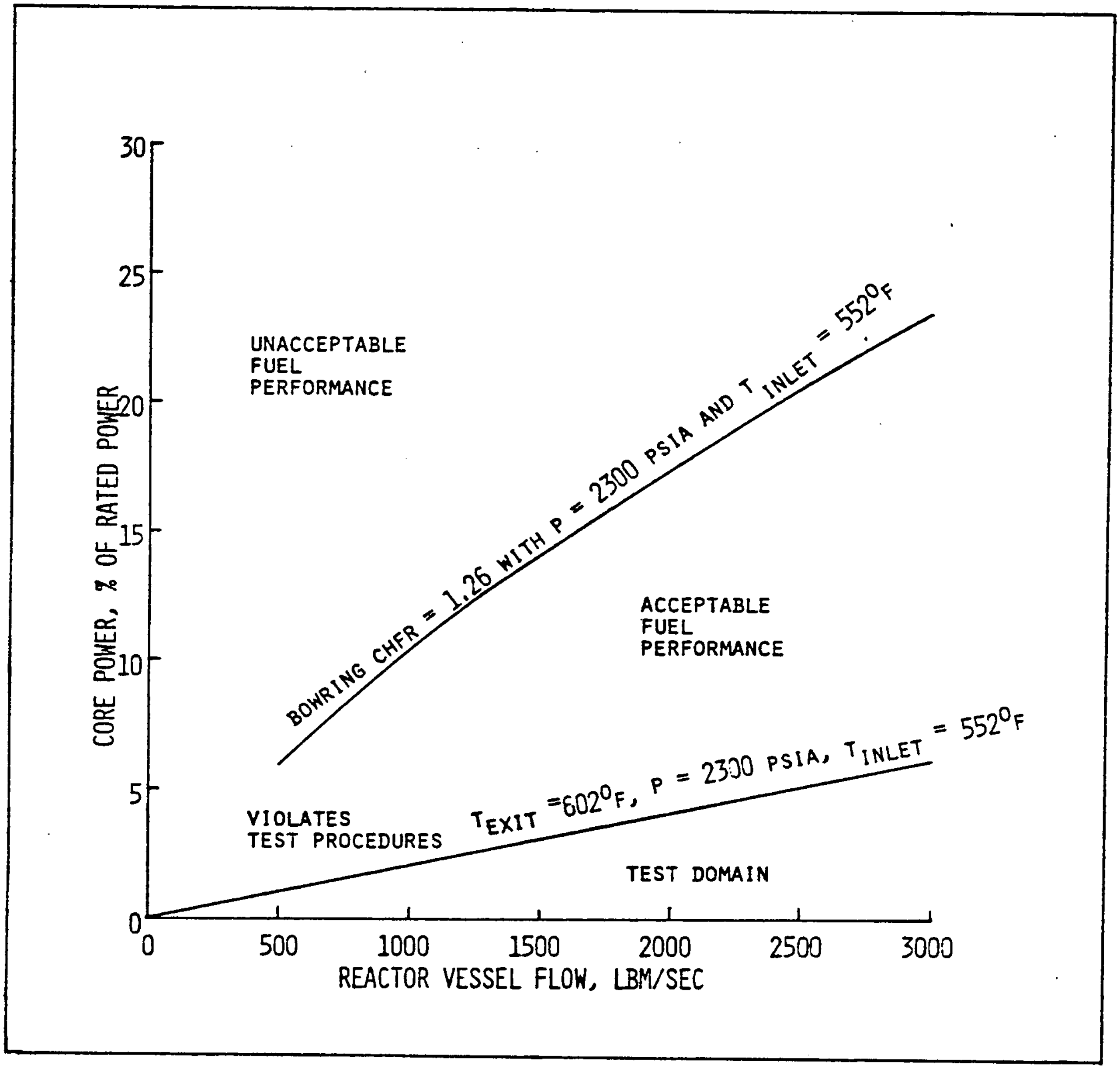
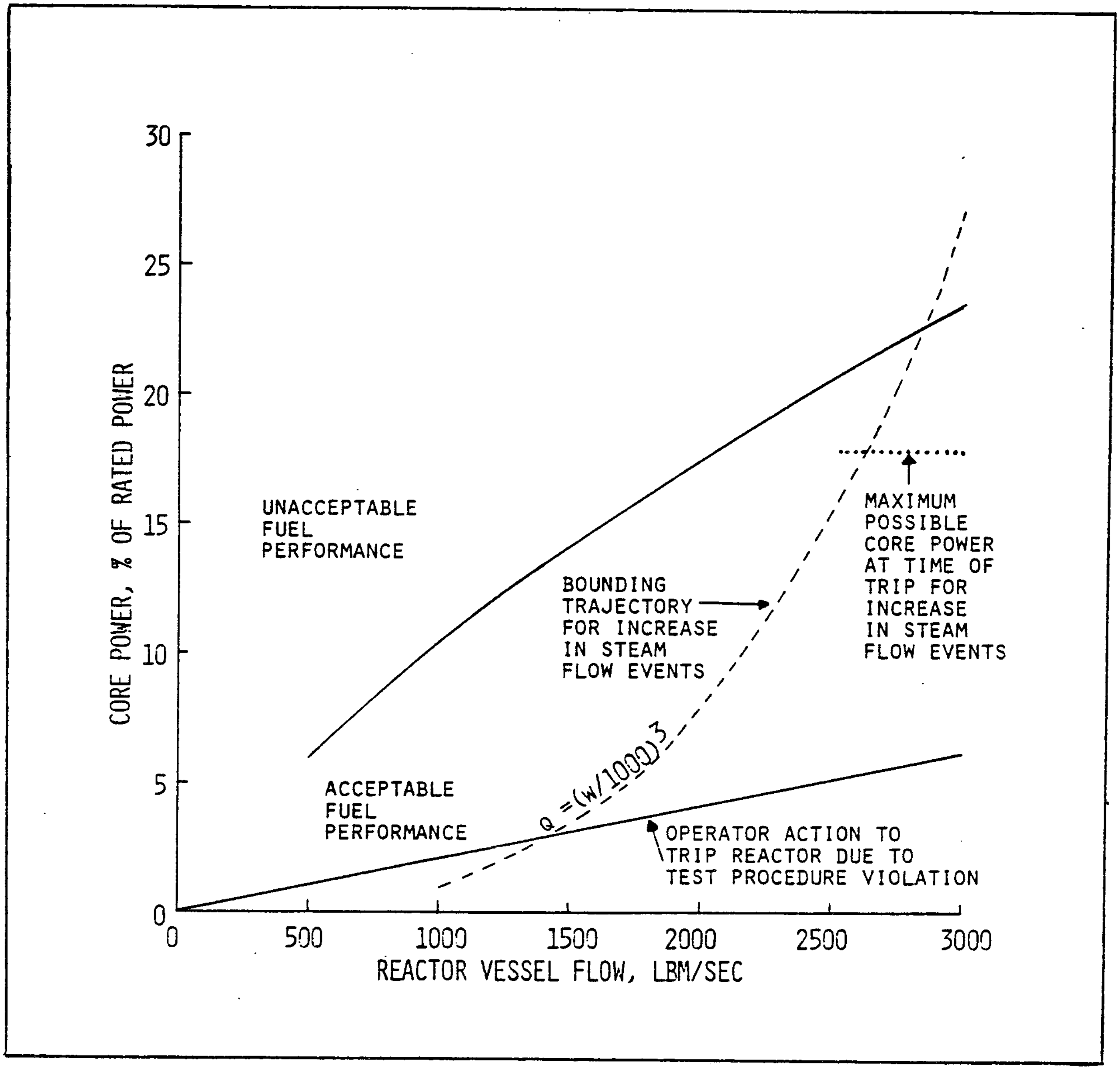


Figure 3-2

Bounds for Increase in Heat Removal
Events in the Reactor Power vs Vessel Flow Space



4.0 OPERATIONAL AND TEST TERMINATION CRITERIA

Test termination criteria for each of the natural circulation tests are listed in 4.1 below. (All numbers represent indicated values.) These criteria represent specific points where manual operator action will be taken to terminate testing in order to prevent operation outside the bounds assumed in the safety analysis of Section 3.0 of this report. Operational criteria for each of the natural circulation tests are listed in 4.2 below. (All numbers represent indicated values.) The various operational criteria represent an envelope within which the operator should try to maintain the plant. While operation outside the envelope defined by the operational criteria is not specifically prohibited, appropriate action should be taken to regain this envelope if necessary. Compliance with the operational and test termination criteria will ensure the following minimum conditions for safe operation are met: 1) Sufficient subcooled margin exists in the reactor coolant system to ensure adequate core cooling and prevent void formation. 2) Sufficient water level in each steam generator exists to ensure an adequate heat removal capability. 3) Sufficient pressurizer level exists to ensure adequate pressure control can be maintained. 4) Sufficient rod worth is available to ensure an adequate shutdown margin. 5) Adequate margin to critical heat flux exists in the core at all times during natural circulation. 6) The possibility of uncontrolled rod motion is minimized. 7) The maximum power level is limited in the event of an uncontrolled reactivity addition.

4.1 Test Termination Criteria

4.1.1 During Tests A1, A2, and A3 trip the reactor and terminate testing if any of the following conditions occur:

a. RCS loop subcooled margin $\leq 20°F$

b. Any loop ΔT $> 58°F$

c.	T_{avg}	> 578°F
d.	Core exit temperature (highest T_h)	> 600°F
e.	Reactor power	> 5%
f.	Sustained reactor startup rate	$\geq$ 1.5 DPM
g.	T_c	< 500°F or > 550°F
h.	Any uncontrolled rod motion (Including rod drop or rod ejection.)	--
i.	RCS pressure can not be adequately controlled.	--

4.1.2 During Tests B1&B2, terminate testing if any of the following conditions occur:

a.	Any abnormalities occur beyond the scope of the current test.	--
b.	Proper steam generator levels can not be maintained during simulated loss of all AC power.	--
c.	RCS pressure can not be adequately controlled during simulated loss of all AC power.	--

4.1.3 During Test B3, terminate testing if any of the following conditions occur:

a.	T_c (except during cooldown)	< 500°F

b. T_h $> 600°F$

c. RCS loop subcooled margin $\leq 50°F$

4.2 Operational Criteria

4.2.1 The following set of operational criteria should be met during the performance of Tests A1, A2, and A3:

a. RCS loop subcooled margin $> 20°F$

b. Steam generator water level $> 60\%$

c. Pressurizer water level

(1) With RCPs running $33 \pm 2\%$

(2) Natural circulation mode $\geq$ Value when RCPs tripped

d. Loop ΔT $\leq 58°F$.

e. T_{avg} $\leq 578°F$.

f. T_c $510° \leq T_c \leq 550°F$

g. High linear power trip (Value given is actual power. Exact setpoint will be lower due to instrument accuracy.) $\leq 12\%$

h. Core exit temperature (highest T_h) $\leq 600°F$

i. CEA Group 6 (All other full and part length CEAs to fully withdrawn.) $\geq 60''$ withdrawn

j.	Reactor power:	
	(1) Maximum transient value	$\leq$ 5%
	(2) Minimum value	$\geq$ 0.5%
	(3) Nominal steady-state value (except as otherwise specified)	1% to 3%
k.	Sustained reactor startup rate	$\leq$ 1 DPM
l.	Steam generator pressure	< 1070 psia
m.	All CEAs in a group should be within $\pm$ 1.5 inches of the group average.	--
n.	Use of the CEDM control system in auto sequential or manual individual mode is prohibited following RCP trip and establishment of natural circulation.	--
o.	Maintain the CEDM control system in the "off" mode except as required to position Group 6 CEAs only following RCP trip and establishment of natural circulation.	--
p.	Do not start a RCP when in natural circulation without first manually tripping the reactor.	--
q.	Maintain steam generator levels as steady as possible.	--
r.	If T_C falls below 510°F, adjust feedwater flow and/or steam flow as necessary to stop the cooldown.	--

4.2.2 The following set of operational criteria, in addition to 4.2.1 above, should be met during the performance of Test A2:

a.	Maximum pressurizer cooldown rate	$\leq$ 200°F in any one hour period
b.	Do not allow RCS pressure to decrease below 1750 psia.	--
c.	Monitor for RCS void formation.	--

4.2.3 The following set of operational criteria, in addition to 4.2.1 above, should be met during the performance of Test A3:

a.	Initial steady-state power level prior to steam generator isolation	~1%
b.	Initial steam generator pressures prior to steam generator isolation	~800 psia

4.2.4 The following set of operational criteria should be met during the performance of Tests B1&B2:

a.	Ensure that both emergency diesels start and operate properly.	--
b.	Maintain normal steam generator water level.	--

4.2.5 The following set of operational criteria should be met during the performance of Test B3:

a.	RCS loop subcooled margin	> 50°F.
b.	Core exit temperature (highest T_h)	$\leq$ 600°F.

c. If an automatic reactor trip does not occur within four (4) seconds following the RCP trip, manually trip the reactor. --

d. If T_c falls below 510°F, except during plant cooldown, adjust feedwater flow and/or steam flow as necessary to control the temperature decrease. --

e. Boron concentration of any makeup water added to the plant during cooldown must be the same or greater than the RCS boron concentration to prevent dilution. --

f. Maintain normal steam generator water level. --

g. Maintain normal pressurizer level prior to depressurization. --

h. Limit plant cooldown rate to 100°F in any one (1) hour period with RCS temperature greater than 200°F. --

i. Limit the pressurizer cooldown rate to 200°F in any one (1) hour period. --

j. Monitor for RCS void formation. --

5.0 IMPACT ON PLANT TECHNICAL SPECIFICATIONS

Based upon the results of the safety analysis presented in Section 3.0 above, it was determined that eight technical specifications will require exceptions to allow for the performance of these tests. Table 5-1 (p. 171) lists the technical specifications that require exceptions along with the tests for which these exceptions are needed. The following paragraphs note the reasons for the exceptions and the basis for allowing the tests to be performed.

5.1 Reactor Core Safety Limits (Tech. Spec. 2.1.1)

Reactor core safety limits restrict fuel operation to within the nucleate boiling regime and maintain the dynamically adjusted peak linear heat rate of the fuel at or less than 21 kw/ft. These limits prevent overheating of the fuel cladding, which could result in cladding penetration by fission products, and prevent centerline melting in any fuel rod. Reactor core safety limits were established based upon the assumption that all four reactor coolant pumps are in operation. Under conditions of natural circulation, i.e., no operating reactor coolant pumps, these limits will require exception to allow for the performance of the low-power tests, A1 through A3. During these tests, however, the following operational limits will be observed: Core exit temperature $\leq$ 600°F, core inlet temperature $\geq$ 500°F and $\leq$ 550°F, loop $\Delta T \leq$ 58°F, and margin to saturation > 20°F. These operational limits were selected to prevent abnormal core thermal conditions from developing and will ensure that an adequate margin to critical heat flux is maintained at all times during the low-power tests. Additionally, Tests A1 through A3 will be performed at low power levels such that the peak linear heat rate will remain well below the 21 kw/ft limit.

5.2 Reactor Trip Setpoints (Tech. Spec. 2.2.1)

The reactor protective instrumentation provides protection against core and reactor coolant system damage by tripping the reactor when any of its various trip setpoints are exceeded. In order to use a critical reactor to simulate decay heat under conditions of natural circulation, certain reactor trips must be altered or bypassed. The following trips will be affected and thus require exceptions: High logarithmic power level, low pressurizer pressure, low steam generator pressure, high local power density, low departure from nucleate boiling ratio (DNBR), and low reactor coolant flow.

The plant is provided with four independent core protection calculators (CPCs), one in each protection channel, which compute local power density and DNBR based upon inputs from various plant parameters. During natural circulation these calculators will generate a low DNBR trip due to a low reactor coolant flow condition and therefore must be placed in bypass. The plant is also provided with a 10^{-4}% bistable which normally prevents the CPCs from being manually bypassed above 10^{-4}% power. In order to perform the low-power tests, the setpoint of this bistable will first be raised from its normal value to a value $\geq$ 10%. This action will then enable the core protection calculators to be placed in bypass. This condition is acceptable as discussed in the applicable portions of the safety analysis section of this report.

The high logarithmic power level trip will be increased from its normal value to 100% rated thermal power in order to prevent an inadvertent reactor trip during the performance of Tests A1 through A3. This action is necessary since test conditions prevent the high logarithmic power level trip from being manually bypassed. Specifically, the high logarithmic power level trip can normally be bypassed only at power levels of greater than 10^{-4}%. This feature is associated with the 10^{-4}% bistable. Since the setpoint of this

bistable will be increased, as discussed in the last paragraph, the high logarithmic power level trip cannot be manually bypassed during the low-power tests and therefore must be reset to the higher value. Automatic plant protection against inadvertent reactivity additions will still be available, however, by lowering the setpoint of the high linear power trip. This setpoint will be adjusted such that an automatic trip occurs at a maximum acceptable power level of 12% based upon the safety analysis in Section 3.0.

The low pressurizer pressure trip setpoint will be manually reset to lower values as required during the performance of Test A2. This action is necessary to prevent a reactor trip while demonstrating natural circulation at reduced pressures. By observing the temperature limits outlined in 5.1 above, adequate heat transfer conditions in the core will be maintained at all times. In addition, operation with the low pressurizer pressure trip setpoint reset during this test has been considered and determined to be within the bounds of the safety analysis of Section 3.0.

If required due to reactor coolant system temperature decreases, the steam generator low pressure trip setpoint will be manually reset to lower values during the three low-power tests. This action may be necessary to prevent a reactor trip since the lower limit on T_c for the tests is 500°F. Operation with the steam generator low pressure trip setpoint reset during the three low-power tests has been considered and determined to be within the bounds of the safety analysis of Section 3.0.

The final reactor trip that must be bypassed is the low reactor coolant flow trip. The input for this trip is generated from the differential pressure across the primary side of the steam generators and ensures protection in the event of a reactor coolant pump sheared shaft accident and a steam line break inside containment. Since

the reactor coolant pumps will be secured during the low-power tests, protection in the event of a pump sheared shaft accident is not required. The steam line break inside containment event is bounded by the discussion in 3.4.1 of the safety analysis of Section 3.0.

5.3 Minimum Temperature for Criticality (Tech. Spec. 3.1.1.4)

The minimum temperature for criticality specification ensures that the reactor will not be made critical with a T_{avg} of less than 520°F. This limitation is required to ensure that the moderator temperature coefficient is within its analyzed temperature range, the protective instrumentation is within its normal operating range, the pressurizer is capable of being in an operable status with a steam bubble, and the reactor pressure vessel is above its minimum RT_{NDT} temperature. During the performance of the three low-power natural circulation tests, T_{avg} could fall below this minimum value. Combustion Engineering has determined that this condition is acceptable and that safe operation of the plant can be ensured provided that prompt operator action takes place as follows: If T_c falls below 510°F, adjust feedwater and/or steam flow as necessary to stop the cooldown; trip the reactor and terminate testing if T_c falls below 500°F.

5.4 Reactor Protective Instrumentation (Tech. Spec. 3.3.1)

As noted in 5.2 above, certain reactor trips must be altered or bypassed in order to perform the three low-power natural circulation tests. The technical specification covering the reactor protective instrumentation can be excepted for the reasons previously noted.

5.5 Engineered Safety Features Actuation System Instrumentation. (Tech. Spec. 3.3.2)

In order to perform Tests A1 through A3, the low steam generator pressure setpoint will be manually reset to lower values if

required. In addition, the low pressurizer pressure setpoint will be manually reset to lower values as required during Test A2. These actions are necessary to prevent certain engineered safety features actuation signals from interrupting the low-power tests. Combustion Engineering has determined that this condition is acceptable since it will allow for the orderly performance of the low-power tests while still permitting the various automatic engineered safety features systems to function if needed.

During Test A3, Natural Circulation with Reduced Heat Removal Capacity, the emergency feedwater actuation signal based upon a high steam generator differential pressure (SG2 > SG1) will be bypassed. For the short duration of the test this condition is acceptable since the auxiliary feedwater system will be maintained in manual and the automatic functions will be performed, if required, by the operator.

5.6 Pressurizer (Tech. Spec. 3.4.3)

Pressure control in the reactor coolant system is normally maintained by the pressurizer through the use of heaters and spray. During Tests A2 and B2, however, the pressurizer will be rendered partially inoperable by intentionally securing heaters or by a loss of electrical power. This mode of operation is acceptable since pressure control will be maintained through the use of pressurizer level and charging/letdown flow.

5.7 Steam Generators (Tech. Spec. 3.4.4)

For the performance of Test A3, Natural Circulation with Reduced Heat Removal Capacity, Steam Generator 2 will be isolated by securing feedwater flow and shutting its associated main steam isolation valve. While this action does not specifically violate the requirements of the technical specification, it does render the steam

generator inoperable as a heat removal source. This condition is acceptable since the operating steam generator has sufficient flow and heat removal capacity, as demonstrated by the pretest computer simulation presented in 2.2.3 above, to safely remove all heat generated by the reactor.

5.8 Auxiliary Feedwater System (Tech. Spec. 3.7.1.2)

During the performance of two of the tests the auxiliary feedwater system will be rendered partially inoperable. In Test A3 feedwater flow to Steam Generator 2 will be isolated, and in Test B2 power to the motor-driven auxiliary feedwater pumps will be secured. This condition is acceptable while performing Test A3 since, as noted in 5.7 above, sufficient heat removal capacity is available from the operating steam generator to safely remove all heat generated by the reactor. This condition is also acceptable while performing Test B2 since the turbine-driven auxiliary feedwater pump will be available to supply water to both steam generators.

Table 5-1

TECHNICAL SPECIFICATION IMPACT

Tech. Spec. Number	Title	Test A1	A2	A3	B1&B2	B3
2.1.1	Reactor Core Safety Limits	X	X	X		
2.2.1	Reactor Trip Setpoints					
	(1) High logarithmic power level	X	X	X		
	(2) Low pressurizer pressure		X			
	(3) Low steam generator pressure	X	X	X		
	(4) High local power density	X	X	X		
	(5) Low DNBR	X	X	X		
	(6) Low reactor coolant flow	X	X	X		
3.1.1.4	Minimum Temperature for Criticality	X	X	X		
3.3.1	Reactor Protective Instrumentation					
	(1) High logarithmic power level	X	X	X		
	(2) High local power density	X	X	X		
	(3) Low DNBR	X	X	X		
	(4) Core protection calculators	X	X	X		
	(5) Low reactor coolant flow	X	X	X		

Table 5-1 (cont.)

TECHNICAL SPECIFICATION IMPACT

Tech. Spec. Number	Title	Test A1	A2	A3	B1&B2	B3
3.3.2	Engineered Safety Features Actuation System Instrumentation					
	(1) Safety injection (SIAS)		X			
	(2) Main steam isolation (MSIS)	X	X	X		
	(3) Containment cooling (CCAS)		X			
	(4) Emergency feedwater (EFAS)	X	X	X		
3.4.3	Pressurizer		X		X	
3.4.4	Steam Generators			X		
3.7.1.2	Auxiliary Feedwater System			X	X	

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