ENCLOSURE 3

NATURAL CIRCULATION TEST PROGRAM

SAN ONOFRE NUCLEAR GENERATING STATION UNIT 2

Prepared for
Southern California Edison Company

SPECIAL NATURAL CIRCULATION TESTS
FOR NEAR - TERM OPERATING LICENSE FACILITIES

1. NATURAL CIRCULATION TEST
2. NATURAL CIRCULATION WITH SIMULATED LOSS OF OFFSITE AC POWER
3. NATURAL CIRCULATION WITH LOSS OF PRESSURIZER HEATERS
4. EFFECT OF STEAM GENERATOR SECONDARY SIDE ISOLATION ON NATURAL CIRCULATION
5. NATURAL CIRCULATION AT REDUCED PRESSURE
6. COOLDOWN CAPABILITY OF THE CHARGING AND LETDOWN SYSTEM
7. SIMULATED LOSS OF ALL ONSITE AND OFFSITE AC POWER
8. ESTABLISHMENT OF NATURAL CIRCULATION FROM STAGNANT CONDITIONS
9. FORCED CIRCULATION COOLDOWN
10. BORON MIXING AND COOLDOWN

RECOMMENDED TEST SEQUENCE FOR SONGS

LOW POWER TESTING

1. NATURAL CIRCULATION VERIFICATION

2. NATURAL CIRCULATION AT REDUCED PRESSURES

3. NATURAL CIRCULATION WITH REDUCTION OF HEAT REMOVAL CAPACITY

POWER ESCALATION TESTING

1. LOSS OF OFFSITE POWER

2. SIMULATED LOSS OF OFFSITE AND ONSITE AC POWER

3. NATURAL CIRCULATION WITH BORON MIXING AND COOLDOWN

COMPARISON OF RECOMMENDED SONGS TEST PROGRAM WITH NEAR-TERM PLANTS

	SEQUOYAH	NORTH ANNA	SALEM	DIABLO CYN (1)	McGUIRE	SONGS
TEST 1 (Natural Circulation Test)	X	X	X	X	X	X
TEST 2 (Natural Circulation with Simulated Loss of Offsite AC Power)	X	X	X	X	X	X (3)
TEST 3 (Natural Circulation with Loss of Pressurizer Heaters)	X	X	X	X	X	X
TEST 4 (Effect of Steam Generator Secondary Side Isolation on Natural Circulation)	X	X	X	X	X	X
TEST 5 (Natural Circulation at Reduced Pressure)	X	X	X	X	X	X
TEST 6 (Cooldown Capability of the Charging and Letdown System)	X	X	X	X		(2)
TEST 7 (Simulated Loss of all Onsite and Offsite AC Power)	X	X	X	X	X	X (3)
TEST 8 (Establishment of Natural Circulation from Stagnant Conditions)	X					
TEST 9 (Forced Circulation Cooldown)	X					
TEST 10 (Boron Mixing and Cooldown)	X	X (3)		X (3)		X (3)

NOTES: (1) With fuel load date uncertain, program may be modified prior to performance.

(2) Measured as part of the post-core Hot Functional Heat Loss Test.

(3) To be performed as part of power escalation program.

SAFETY EVALUATION

CONTENTS

ASSUMPTIONS USED FOR SIMULATION OF TEST A1

1. REACTOR POWER IS HELD CONSTANT FOR THE ENTIRE TRANSIENT.

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.

4. STEAM GENERATOR WATER LEVELS ARE MAINTAINED AT APPROXIMATELY 69%.

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.

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

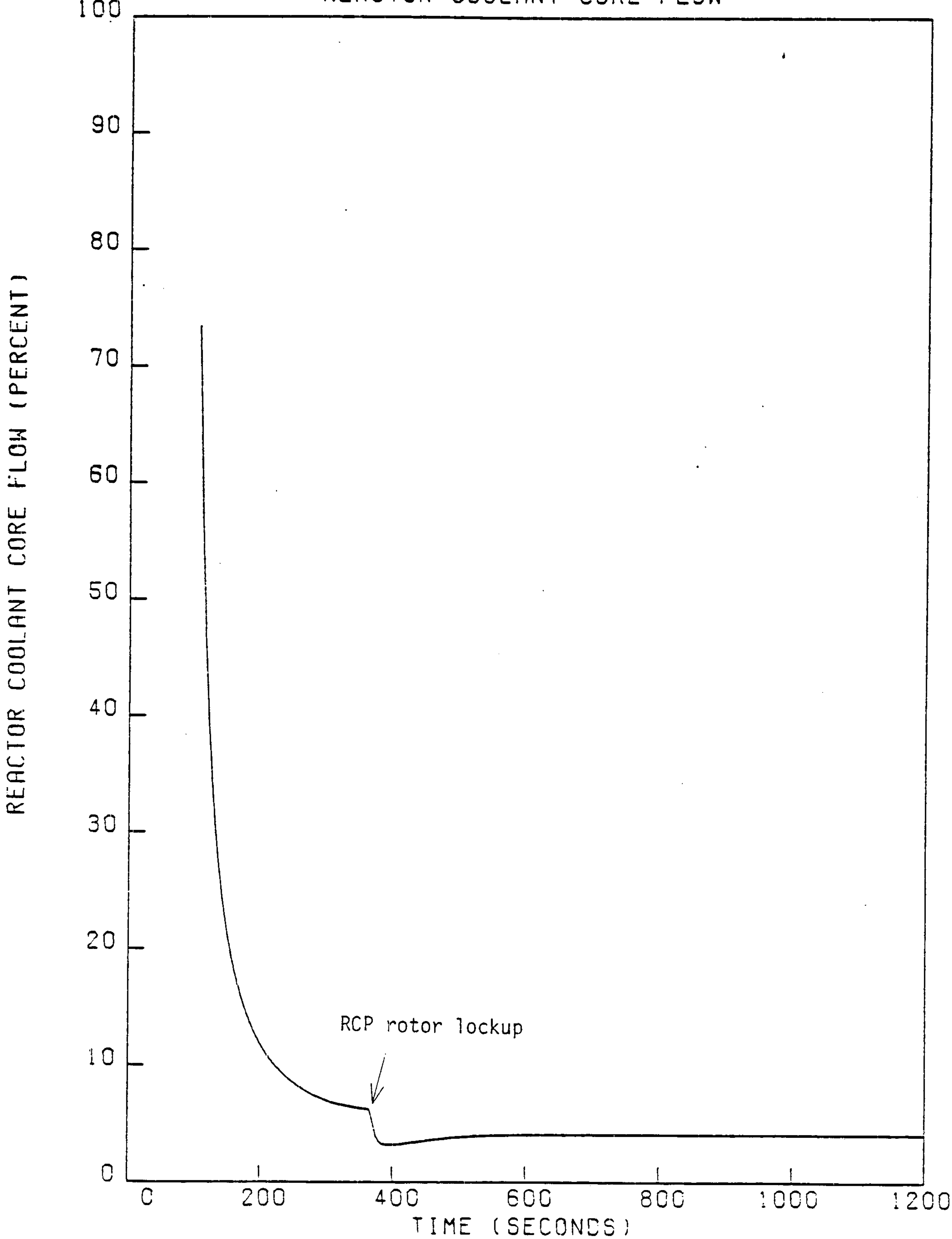


FIGURE 2-17
SONGS 3 PCT PWR NATURAL CIRCULATION
LOOP A COOLANT TEMPERATURES

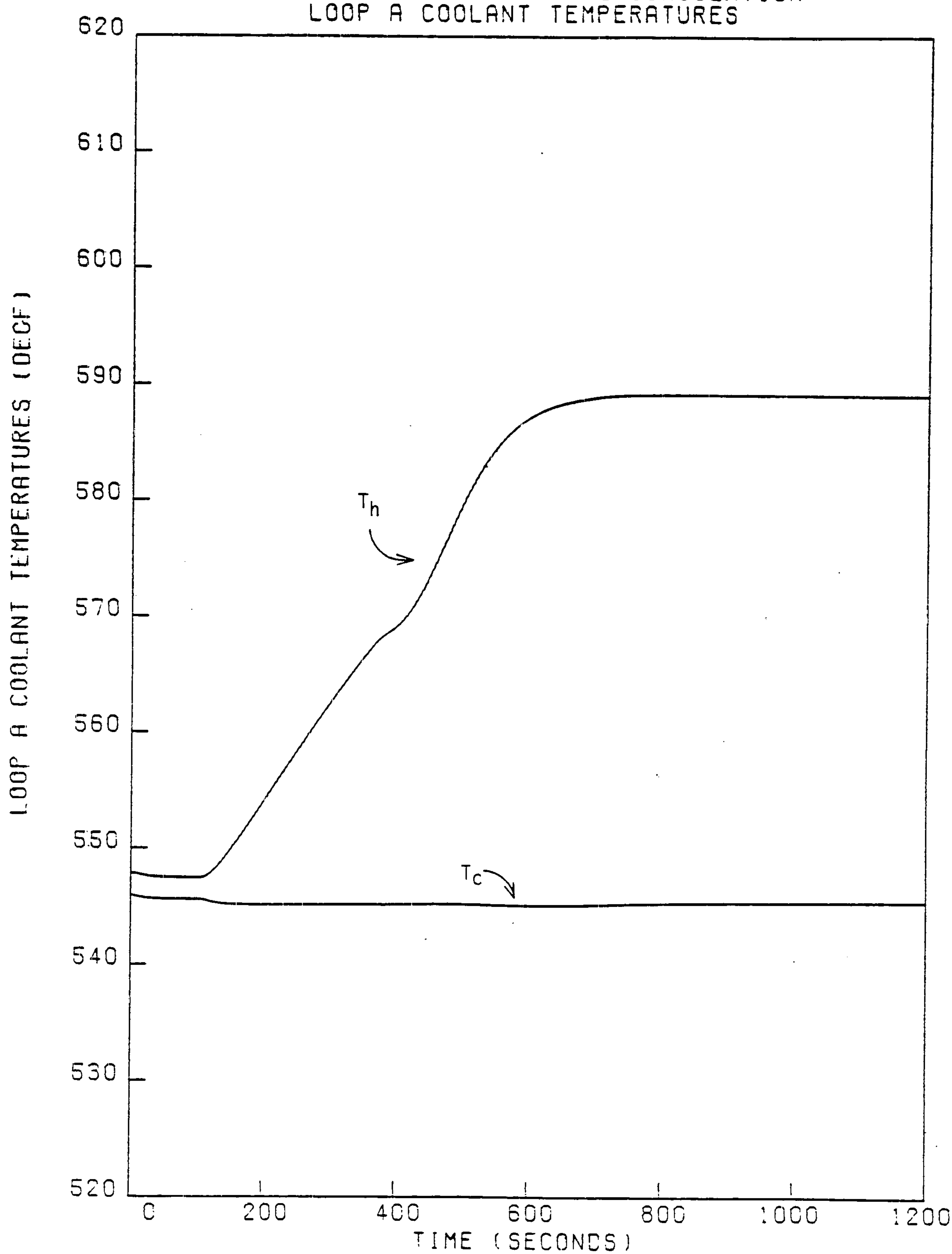


FIGURE 2-16
SONGS 3 PCT PWR NATURAL CIRCULATION
PZR LEVEL

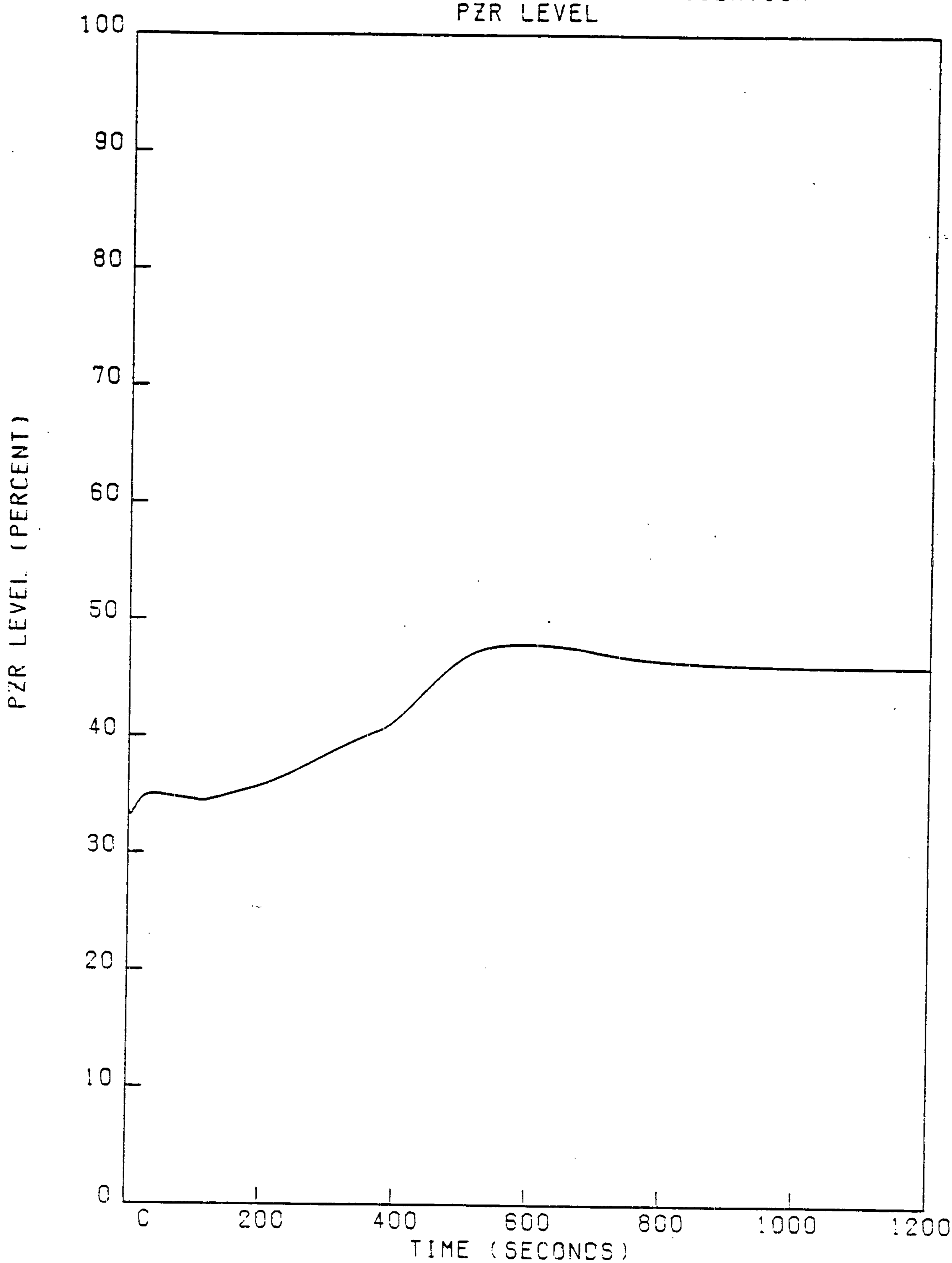


FIGURE 2-19
SONGS 3 PCT PWR NATURAL CIRCULATION
PRESSURIZER PRESSURE

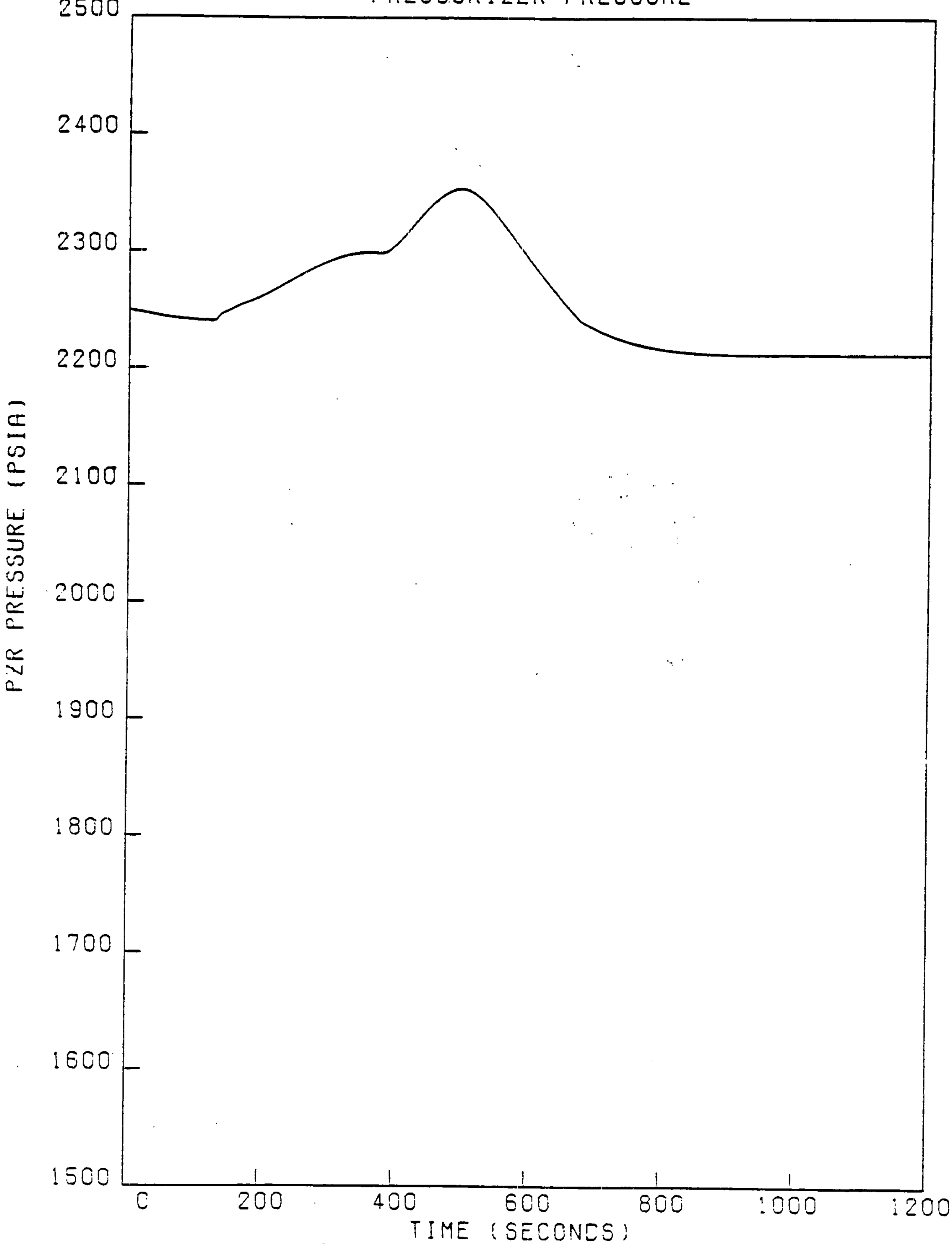


Table 2-1

Verification of Natural Circulation
Numerical Results(1)

Plant response vs power levels

Plant Parameter	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.

ASSUMPTIONS USED FOR SIMULATION OF TEST A2

1. REACTOR POWER IS HELD CONSTANT AT 3%.

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.

4. STEAM GENERATOR WATER LEVELS ARE MAINTAINED AT APPROXIMATELY 69%.

5. PRESSURIZER LEVEL CONTROL IS IN AUTOMATIC.

6. PRESSURIZER PRESSURE IS CONTROLLED IN AUTOMATIC AT 1700 PSIA.

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

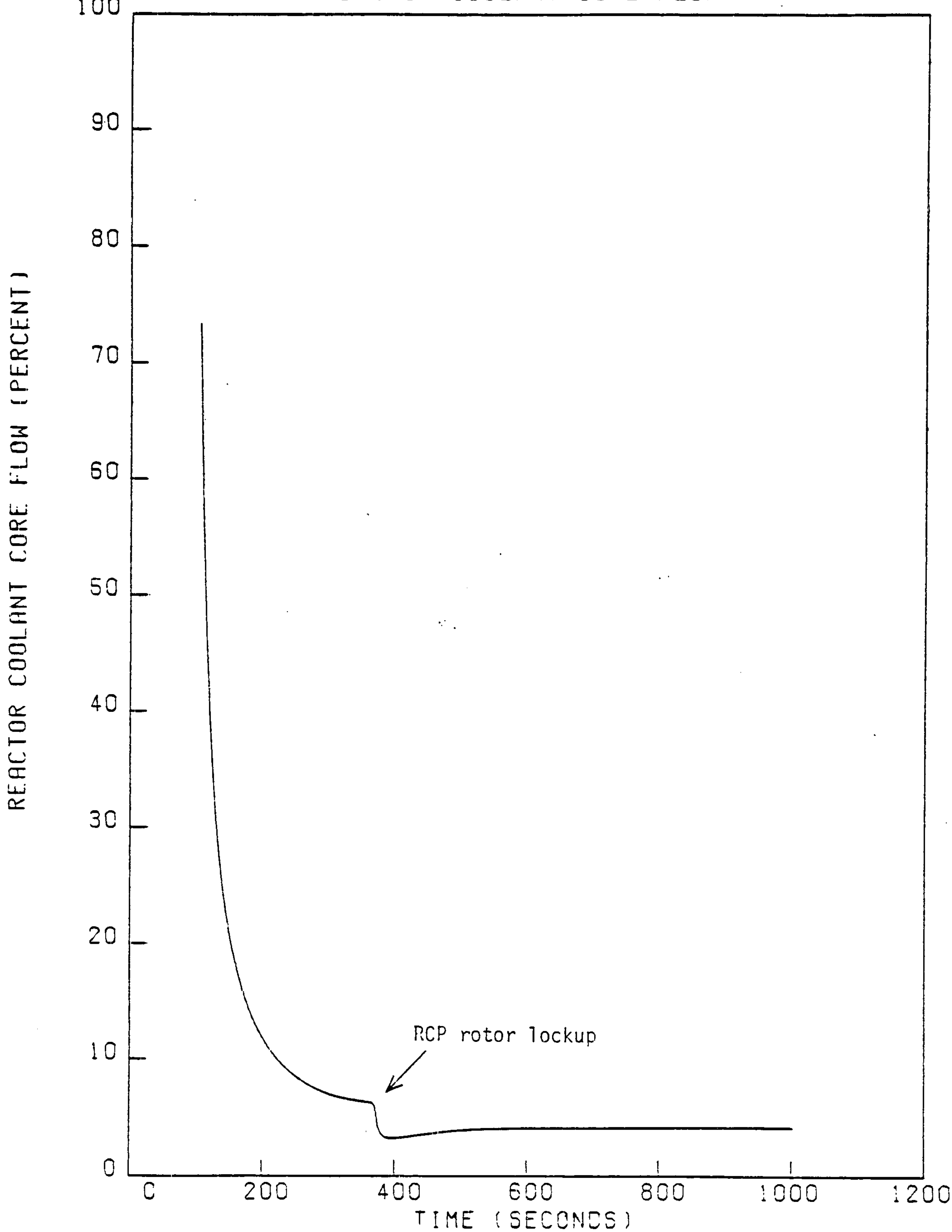


FIGURE 2-41
SONGS NAT CIRC AT REDUCED PRESSURE
LOOP A COOLANT TEMPERATURES

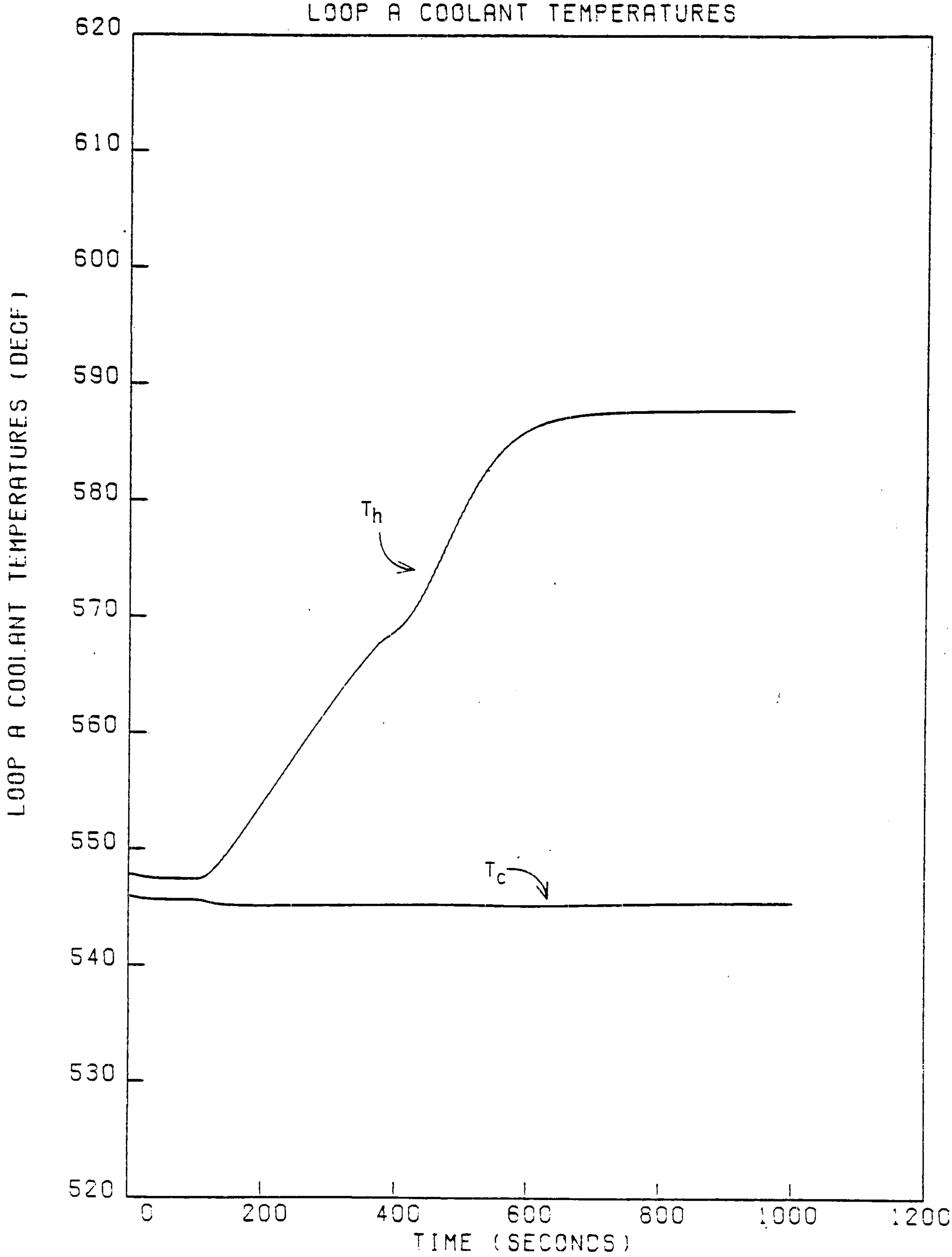


FIGURE 2-40
SONGS NAT CIRC AT REDUCED PRESSURE
PZR LEVEL

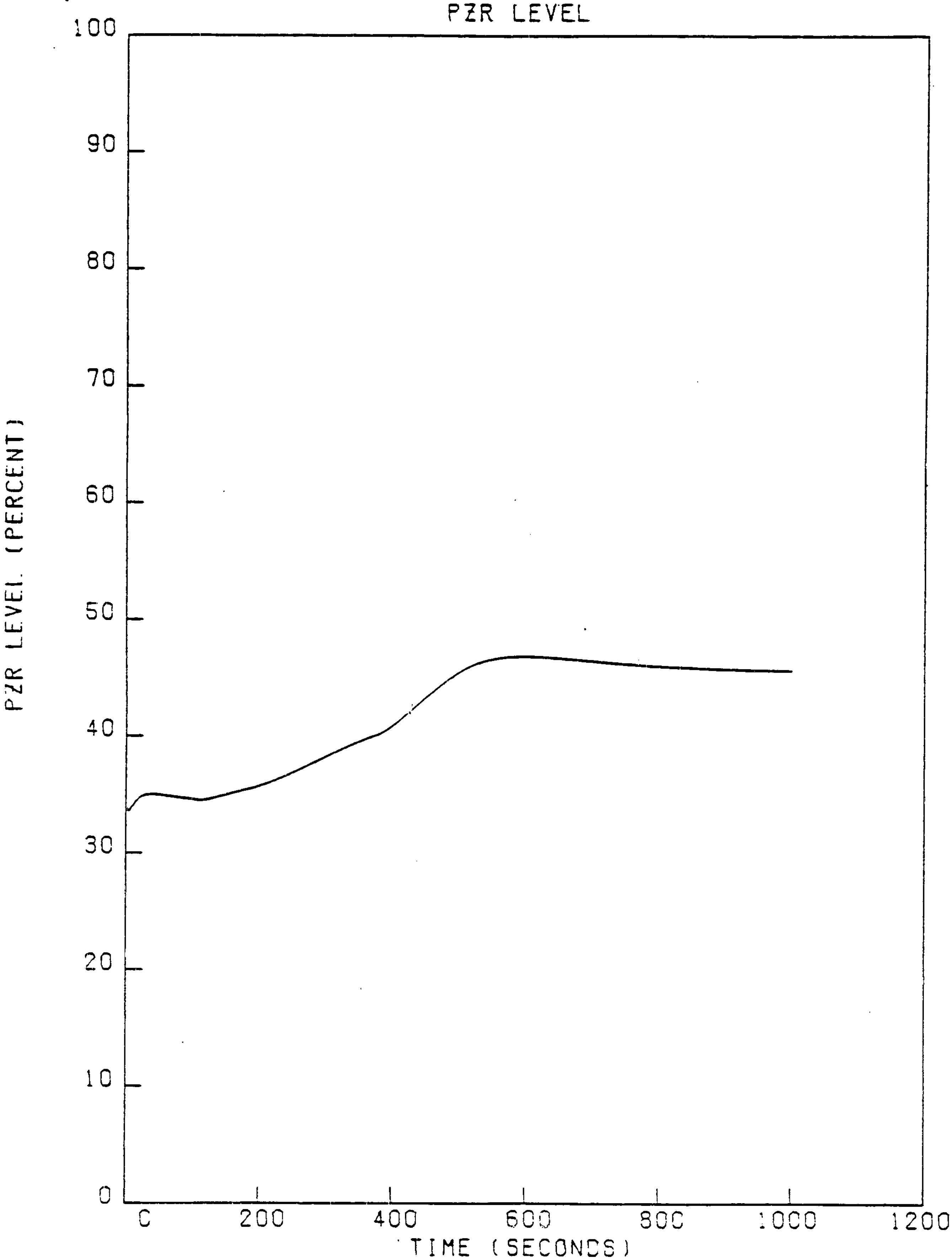
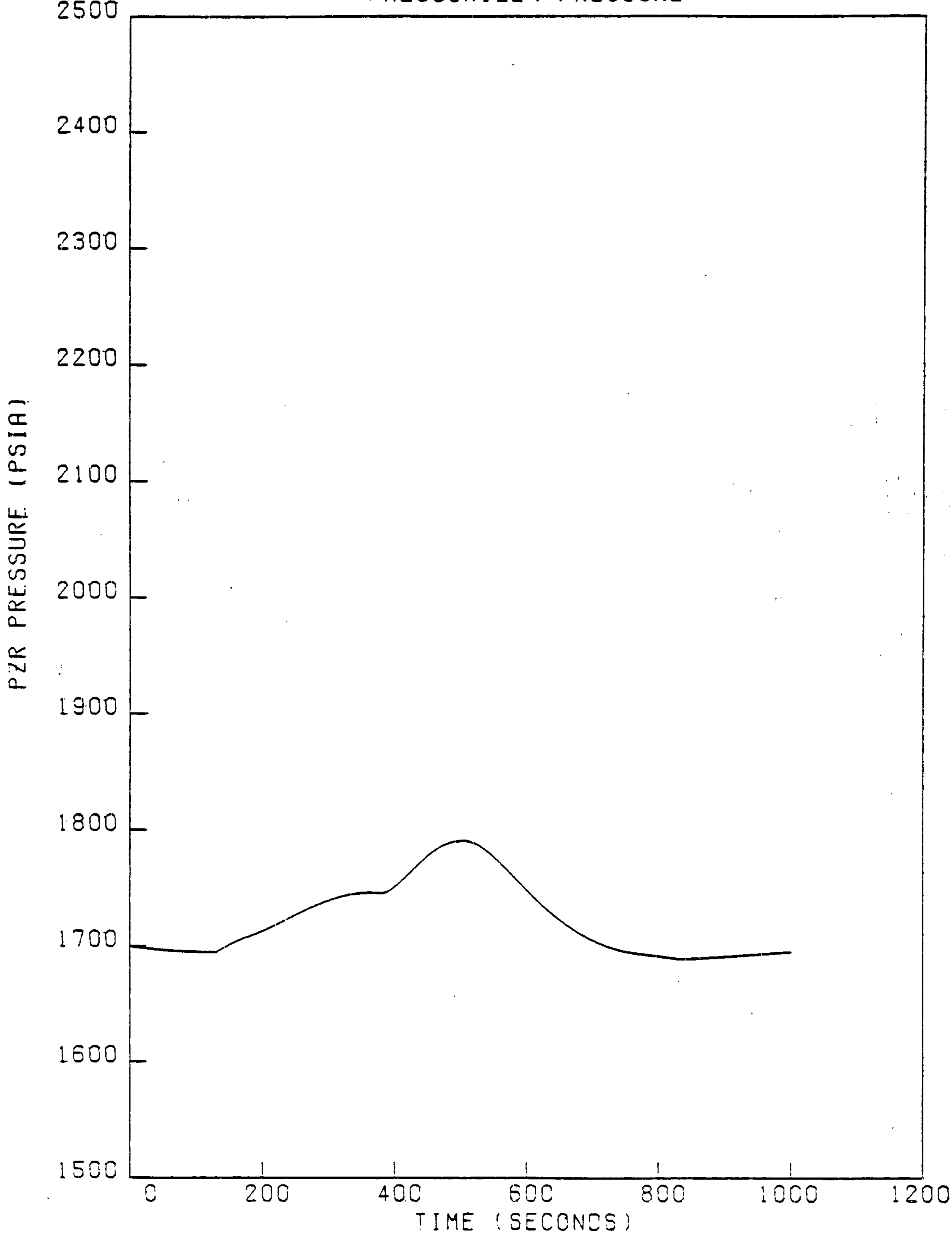


FIGURE 2-43
SONGS NAT CIRC AT REDUCED PRESSURE
PRESSURIZER PRESSURE



ASSUMPTIONS USED FOR SIMULATION OF TEST A3

1. REACTOR POWER IS HELD CONSTANT AT 1%.

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.

4. WATER LEVEL IN STEAM GENERATOR 1 IS MAINTAINED AT APPROXIMATELY 69%.

5. PRESSURIZER PRESSURE AND LEVEL CONTROLS ARE IN AUTOMATIC.

6. HEAT LOSSES FROM THE RCS TO THE CONTAINMENT ARE NEGLIGIBLE.

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

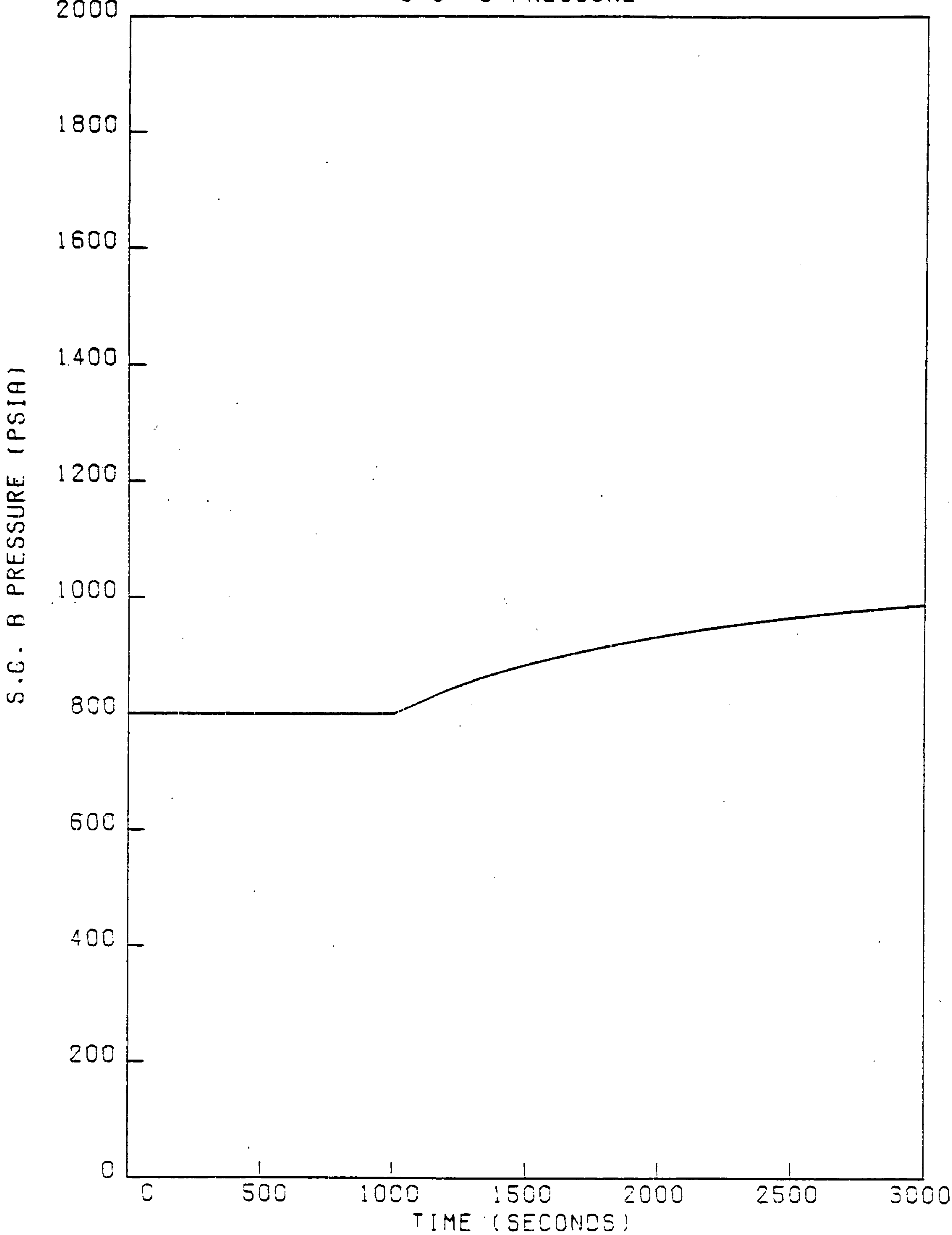


FIGURE 2-50
SONGS LOW PWR ASYMMETRIC NAT CIRC
LOOP A COOLANT TEMPERATURES

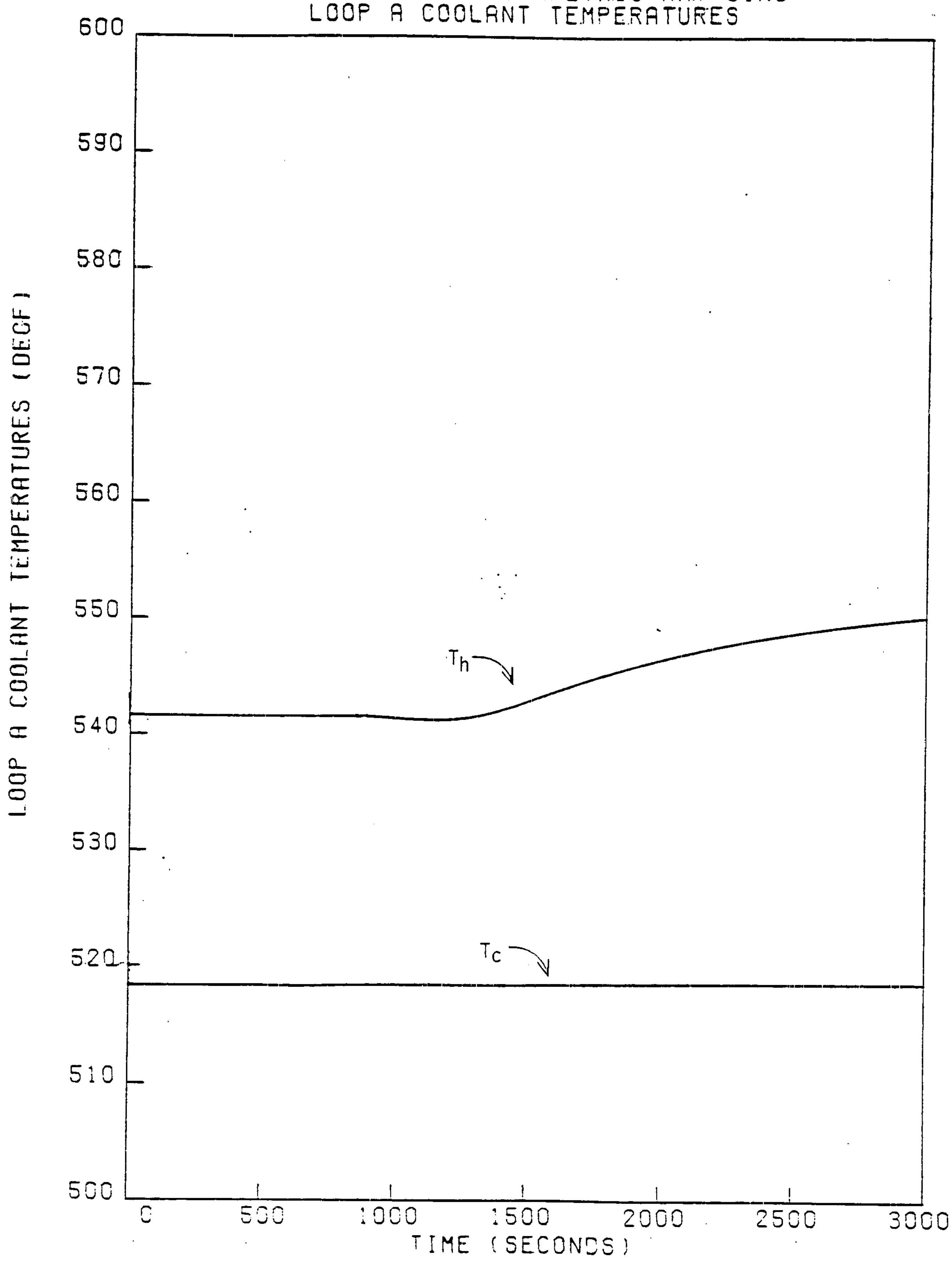


FIGURE 2-51
SONGS LOW PWR ASYMMETRIC NAT CIRC
LOOP B COOLANT TEMPERATURES

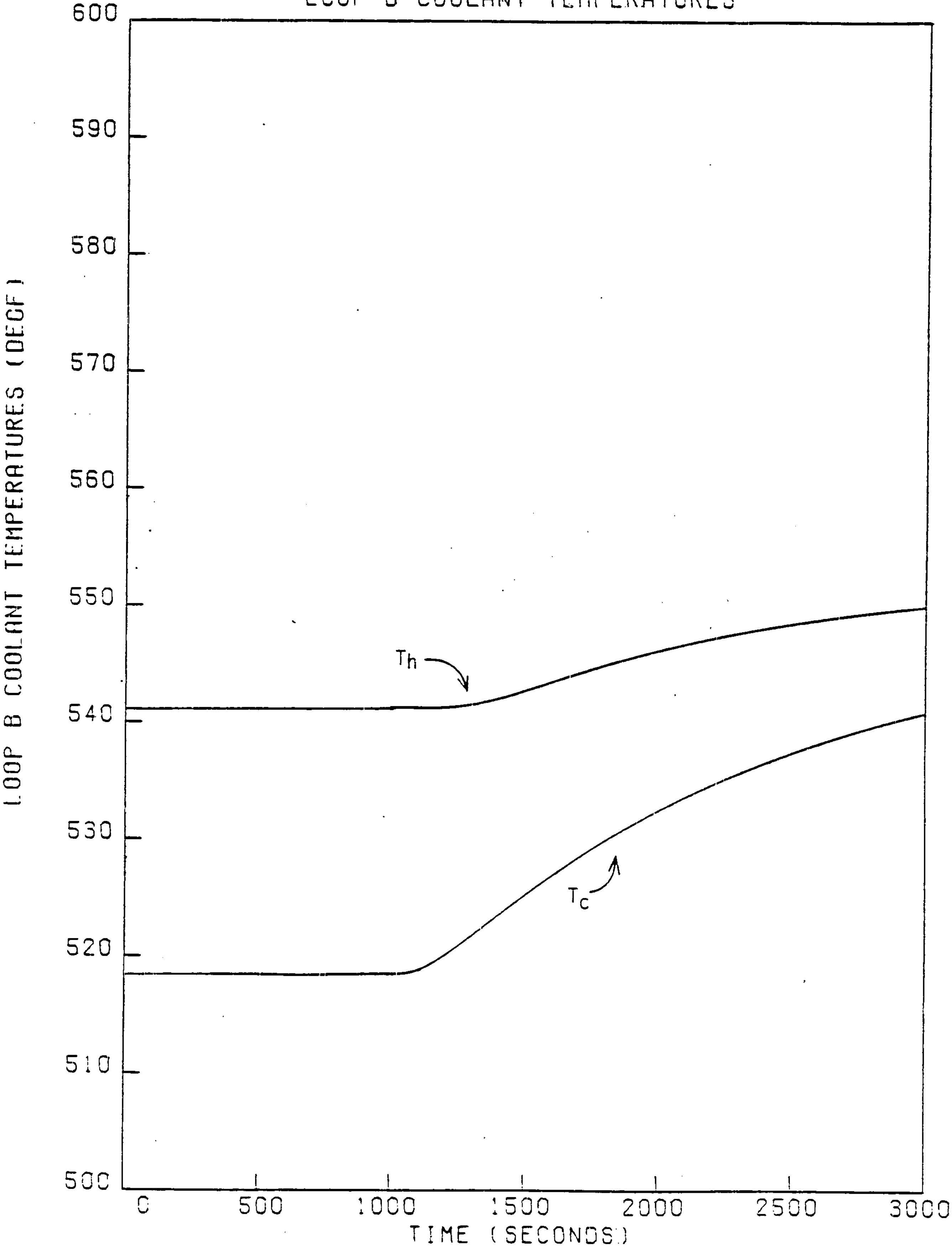
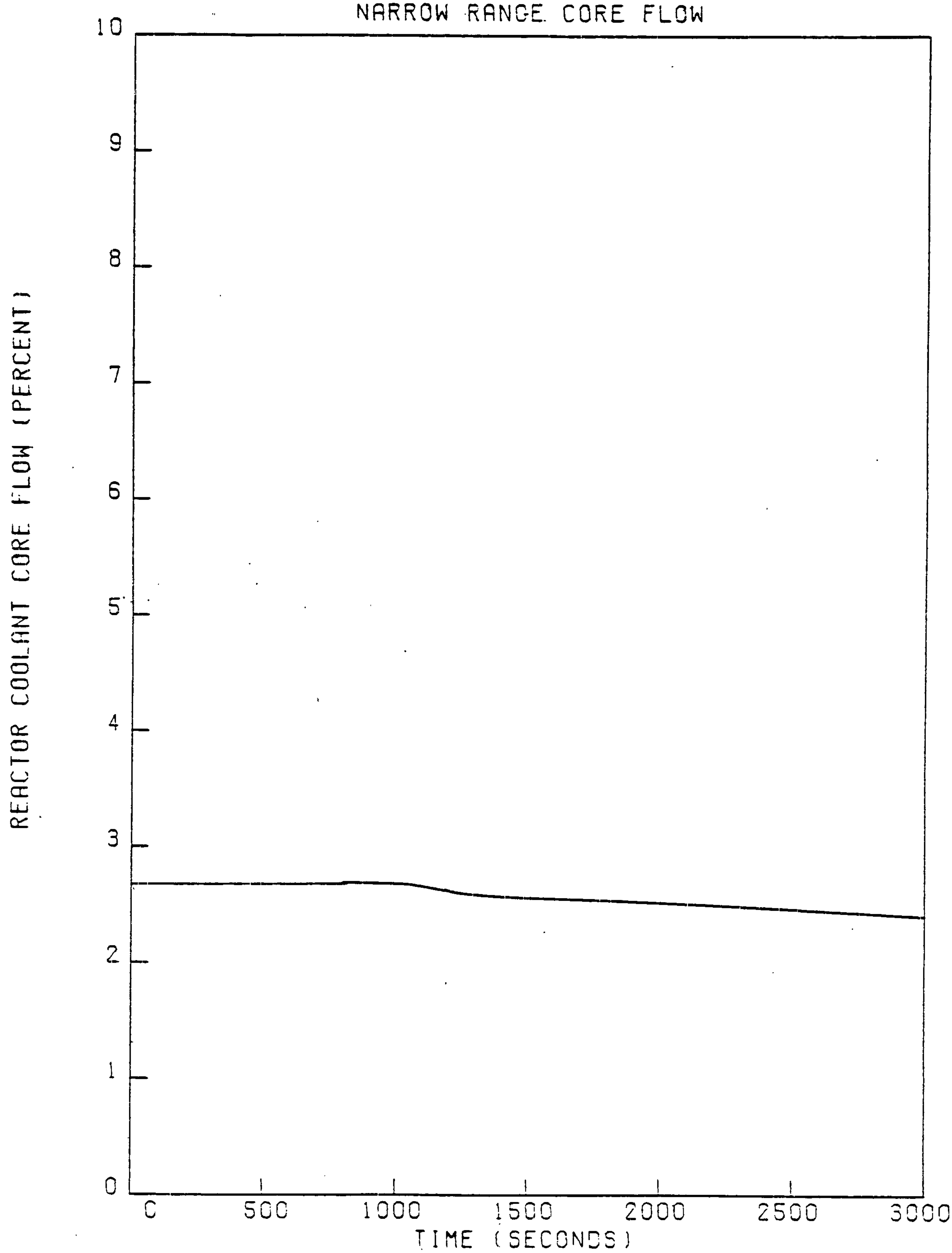


FIGURE 2-53
SONGS LOW PWR ASYMMETRIC NAT CIRC
NARROW RANGE CORE FLOW



ASSUMPTIONS USED FOR SIMULATION OF TESTS B1 & B2

1. REACTOR COOLANT SYSTEM IS INITIALLY STABLE AT 20% POWER.

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 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.

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

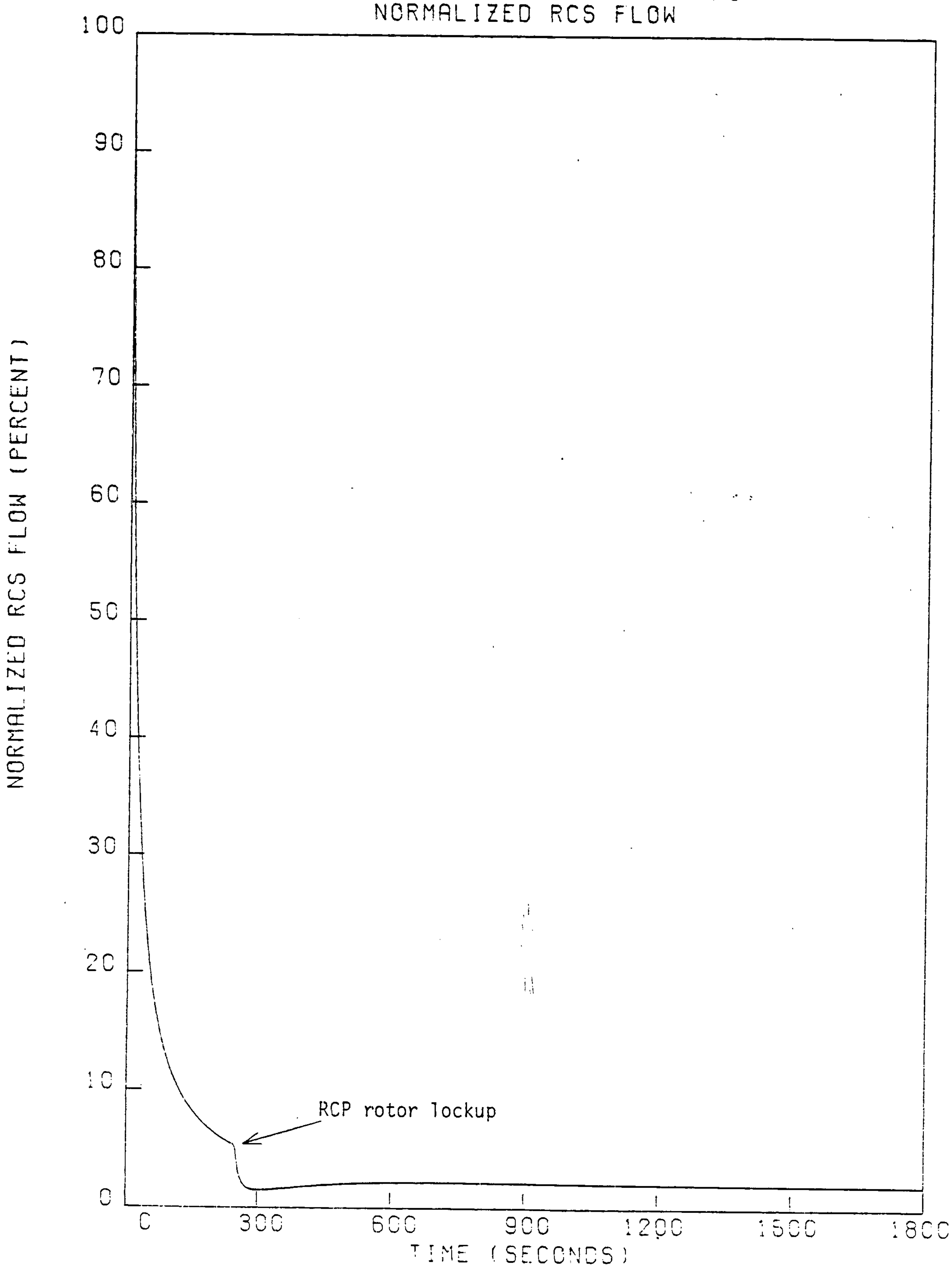


FIGURE 2-57
SONGS 20 PCT PWR LOSS OF AC
PRESSURIZER PRESSURE

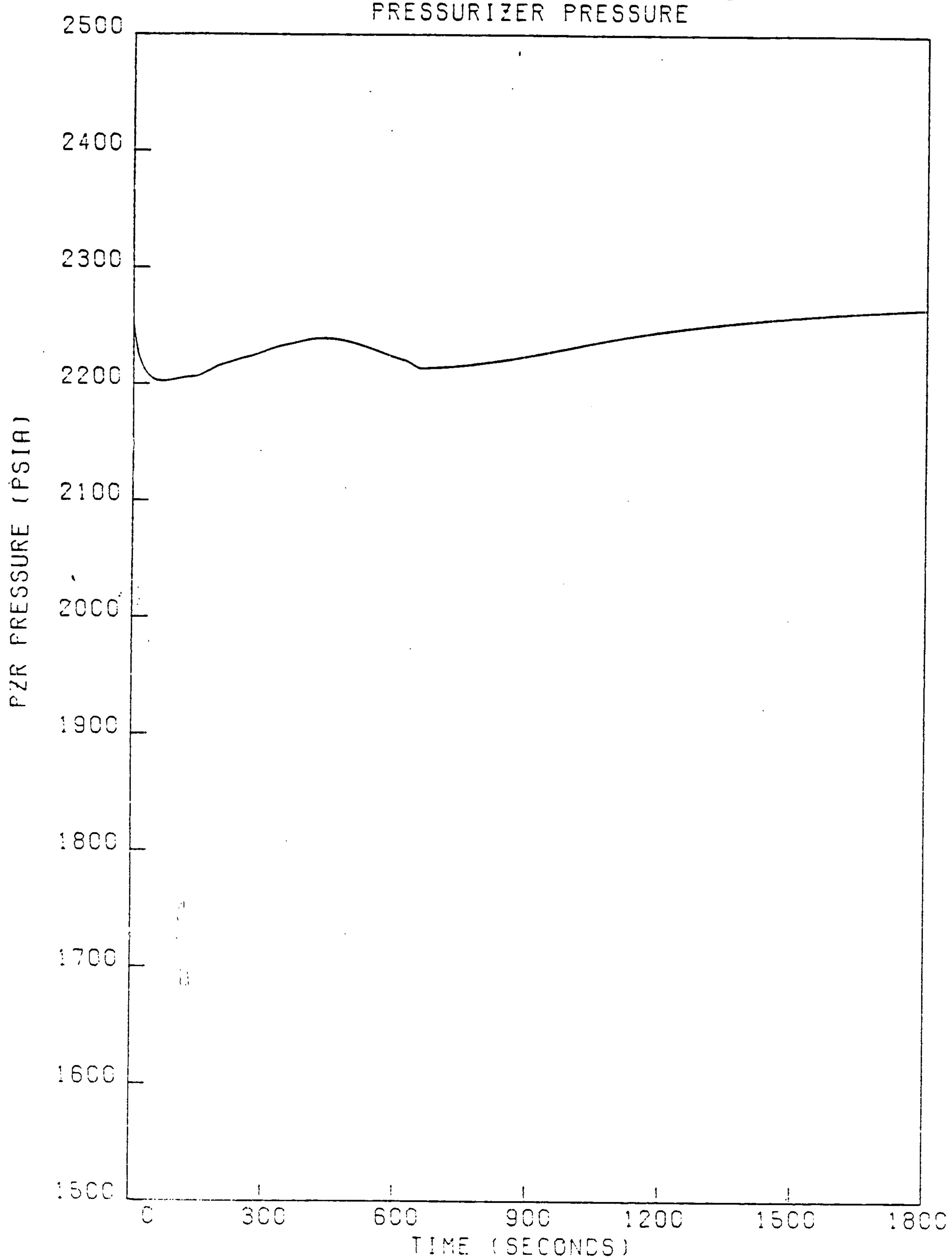


FIGURE 2-58
SONGS 20 PCT PWR LOSS OF AC
PZR LEVEL

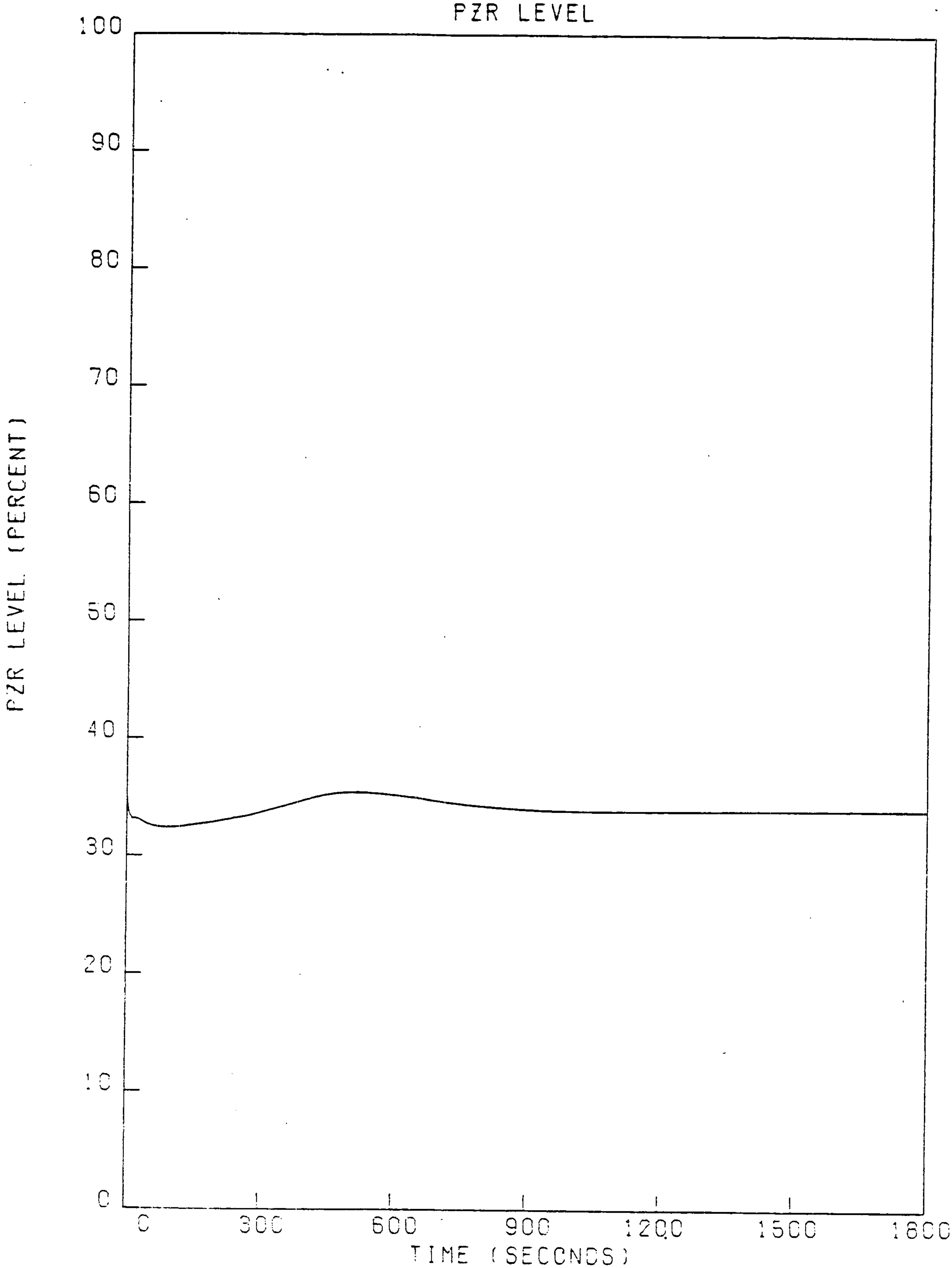


FIGURE 2-61
SONGS 20 PCT PWR LOSS OF AC
LOOP A COOLANT TEMPERATURES

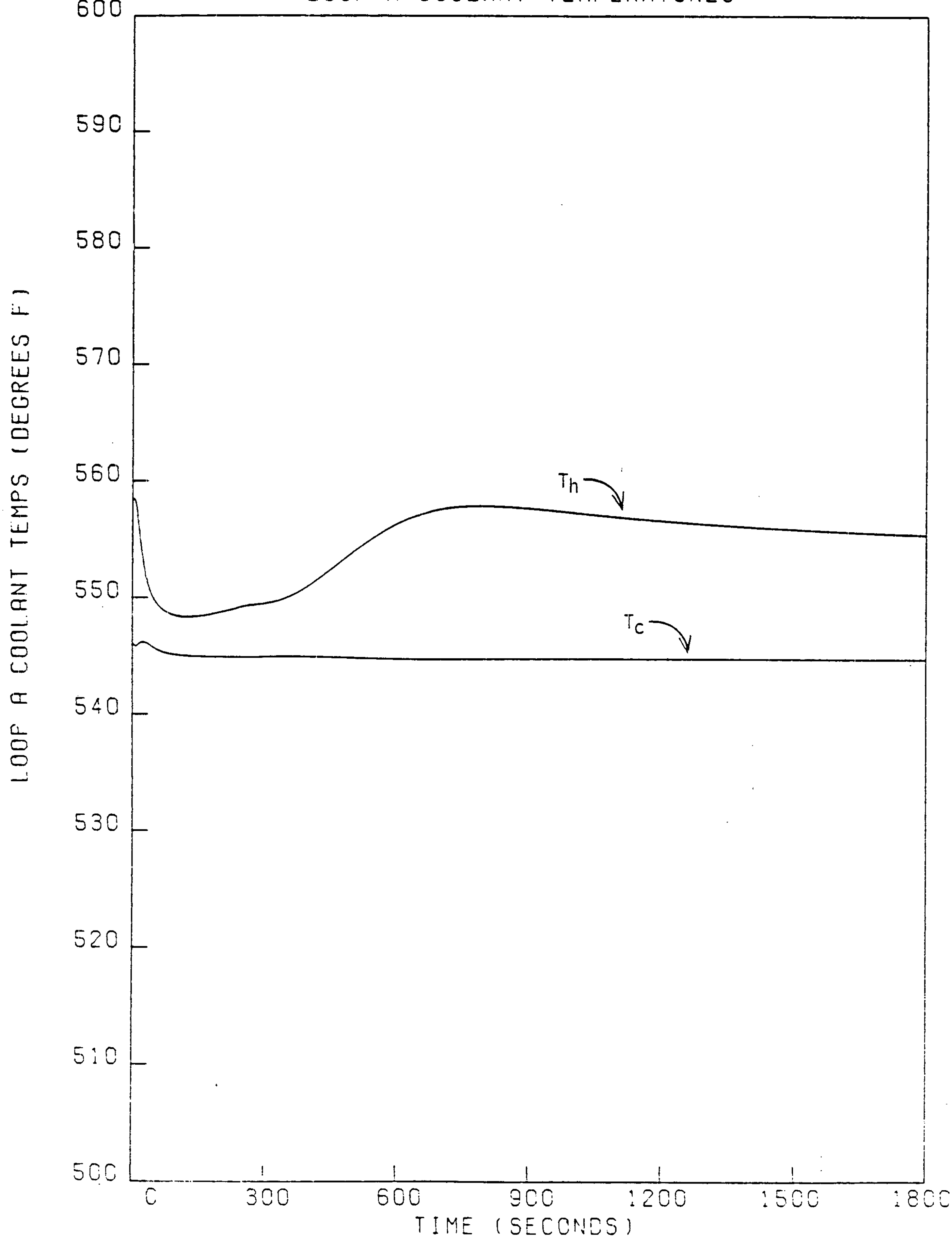
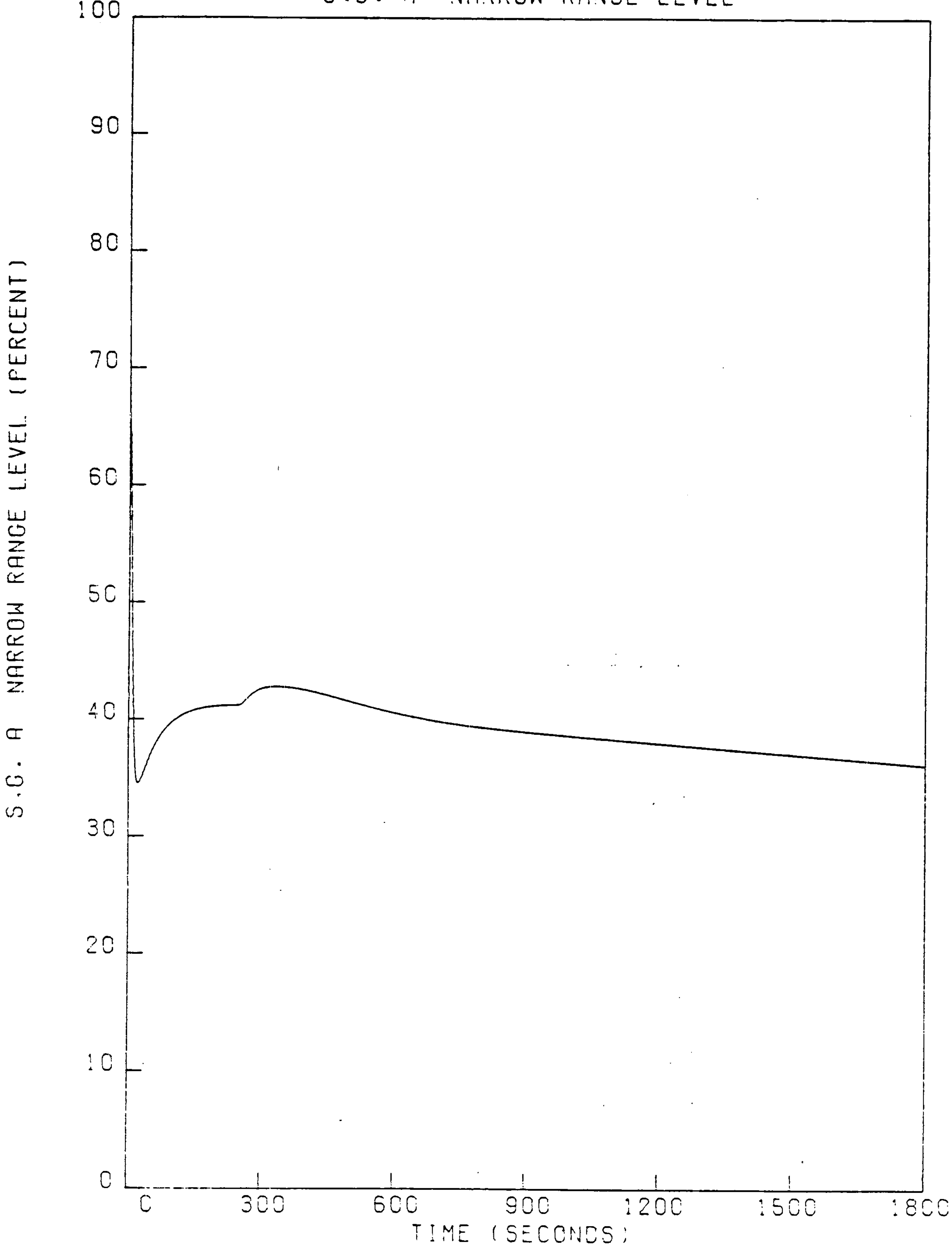


FIGURE 2-65
SONGS 20 PCT PWR LOSS OF AC
S.G. A NARROW RANGE LEVEL



TEST METHOD

80% Total Loss of Flow / Natural Circulation - Test B3

I. Reactor critical at 80 ± 0.5% indicated power.

II. All plant safety related and auxiliary loads powered from external sources.

III. Simultaneously trip all four RCPs.

IV. Boron mixing verification.

V. Natural circulation cooldown.

- Plant cooldown at elevated pressures.
- Isolate letdown and depressurize.
- Monitor for void formation.

SAN ONOFRE UNIT 2 N.C. TEST PROGRAM

SAFETY ANALYSIS

1. OBJECTIVE
2. MAJOR ASSUMPTIONS
3. INITIAL CONDITIONS
4. SAFETY ANALYSIS SECTION FORMAT
5. SUMMARY OF RESULTS
6. CONCLUSION

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SAFETY ANALYSIS

OBJECTIVE

TO DEMONSTRATE THAT THE NATURAL CIRCULATION TESTS DO NOT INVOLVE AN UNRESOLVED SAFETY QUESTION AS DEFINED IN 10 CFR 50.59 BASED ON EVALUATIONS OF CHAPTER 15 FSAR EVENTS FOR THE OPERATING SPACE WHICH EXISTS DURING THE TESTS.

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MAJOR ASSUMPTIONS FOR SAFETY ANALYSIS

A. TO SATISFY TEST CRITERIA

- CERTAIN REACTOR TRIPS AND ESFAS FUNCTIONS BYPASSED OR ALTERED FOR LOW POWER TESTS
- CORE POWER ~1% AND INITIAL SG PRESSURE ~800 PSIA FOR TEST A3
- SI AVAILABLE DURING ALL TESTS, WITH THE LOW PRESSURIZER PRESSURE SETPOINT FOR SIAS RESET TO LOWER VALUES FOR TEST A2
- TEMPERATURE LIMITS ON HOT LEG, COLD LEG, AND CORE ΔT MAINTAINED
- FULL AND PART LENGTH CEAS FULLY WITHDRAWN, EXCEPT FOR GROUP 6 WHICH WAS $\geq$ 60" WITHDRAWN FOR LOW POWER TESTS

B. TO SATISFY SRP ACCEPTANCE CRITERIA

- HIGH LINEAR POWER TRIP REDUCED TO 12% FOR LOW POWER TESTS
- NORMAL PROTECTION ASSUMED IN FSAR ANALYSIS VIA REACTOR TRIP FUNCTIONS AVAILABLE DURING REACTOR STARTUP
  - HIGH LOGARITHMIC POWER TRIP UP TO 10^{-4}%
  - HIGH LOCAL POWER DENSITY AND LOW DNBR TRIPS FROM 10^{-4}% TO ~3%
- CEDMCS IN "OFF" MODE EXCEPT TO POSITION CEA
- MINIMUM CORE POWER $\geq$ 0.1% FOR LOW POWER TESTS

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INITIAL CONDITIONS EMPLOYED IN THE SAFETY ANALYSIS (1)

	RANGE		
PARAMETER	TESTS A1, A2, & A3	TEST B1&B2	TEST B3
REACTOR POWER (%)	0.1 - 7(2)	20	80
T_C (°F) (3)	498 - 552	498 - 549	498 - 554
T_H (°F)	≤ 602	≤ 602	≤ 602
ΔT_{CORE} (°F)	≤ 60	≤ 60	≤ 60
SUBCOOLED MARGIN (°F)	≥ 18	≥ 48	≥ 48
PZR. PRESSURE (PSIA) (3)	1650 - 2300	≤ 2300	≤ 2300
PZR. LEVEL (%)	31 - 54	31 - 35	31 - 56
SG PRESSURE (PSIA)	550 - 1025	≤ 1000	≤ 1000
SG LEVEL (%)	69 ± 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.

(3) VALUES OF THESE PARAMETERS AND CORE FLOW RATE ARE OUTSIDE THE RANGE OF CONDITIONS ASSUMED IN THE FSAR.

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TOPICAL REPORT CEN-201 (S)

SAFETY ANALYSIS SECTION FORMAT

SECTION 3.1	INTRODUCTION
SECTION 3.2	CONDITION II - MODERATE FREQUENCY EVENTS
SECTION 3.3	CONDITION III - INFREQUENT EVENTS
SECTION 3.4	CONDITION IV - LIMITING FAULTS

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SUMMARY OF SAFETY ANALYSIS RESULTS

1. QUANTITATIVE EVALUATIONS SHOW SRP ACCEPTANCE CRITERIA ARE SATISFIED

   - INCREASE IN MAIN STEAM FLOW
   - STARTUP OF AN INACTIVE RCP
   - STEAM SYSTEM PIPING FAILURES
   - CEA WITHDRAWAL AT POWER
   - CEA EJECTION
   - LOSS OF FEEDWATER FLOW

2. OPERATOR ACTION REQUIRED

   - CEA MISOPERATION

3. PROBABILITY OF OCCURRENCE REDUCED

   - INADVERTENT OPERATION OF THE ECCS
   - LOSS OF EXTERNAL LOAD
   - TURBINE TRIP

4. BOUNDED BY FSAR ANALYSIS RESULTS

   - ALL OTHER EVENTS

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SAFETY ANALYSIS

CONCLUSION

THE PROPOSED TESTS DO NOT INVOLVE AN UNRESOLVED SAFETY QUESTION SINCE SRP ACCEPTANCE CRITERIA ARE SATISFIED FOR DESIGN BASIS EVENTS UNDER NATURAL CIRCULATION TEST CONDITIONS.

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.

TEST TERMINATION CRITERIA

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.	--

TEST TERMINATION CRITERIA

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.	--

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	≤ 20°F

Table 5-1

TECHNICAL SPECIFICATION IMPACT

Tech. Spec. Number	Title	A1	A2	A3	B1&B2	B3
		Test				
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	
3.4.1	REACTOR COOLANT LOOPS AND COOLANT CIRCULATION	X	X	X		
3.4.1.2	HOT STANDBY	X	X	X	X	X

ENCLOSURE A
NATURAL CIRCULATION TEST AND
SUPPORTING ANALYSES REQUIRED
IN BRANCH TECHNICAL POSITION RSB 5-1

The functional requirements of Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System," are as follows:

"Functional Requirements

The system(s) which can be used to take the reactor from normal operating conditions to cold shutdown* shall satisfy the functional requirements listed below.

1. The design shall be such that the reactor can be taken from normal operating conditions to cold shutdown* using only safety-grade systems. These systems shall satisfy General Design Criteria 1 through 5.

2. The system(s) shall have suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system function can be accomplished assuming a single failure.

3. The system(s) shall be capable of being operated from the control room with either only onsite or only offsite power available. In demonstrating that the system can perform its function assuming a single failure, limited operator action outside of the control room would be considered acceptable if suitably justified.

*Processes involved in cooldown are heat removal, depressurization, flow circulation, and reactivity control. The cold shutdown condition, as described in the Standard Technical Specs refers to a subcritical reactor with a reactor coolant temperature no greater than 200 °F for a PWR and 212°F for a BWR.

4. The system(s) shall be capable of bringing the reactor to a cold shutdown condition, with only offsite or onsite power available, within a reasonable period of time following shutdown, assuming the most limiting single failure."

For PWRs, the loss of offsite power results in reactor scram and loss of the main condenser, the main feedwater pumps and the RCS pumps. The plant can be brought to a hot standby condition with feedwater supplied by the auxiliary feedwater pumps taking suction from the condensate storage tank and with steam discharge to the atmosphere via the atmospheric dump valves or steam generator safety valves. In accordance with the Standard Review Plan prior to incorporation of RSB 5-1, the auxiliary feedwater system, condensate storage tank and safety valves meet safety grade requirements. The residual heat removal system (RHRS), which was also required to be a safety grade system prior to incorporation of RSB 5-1, is used to take the reactor to cold shutdown conditions. However, the RHRS cannot be operated in this mode until the RCS pressure and temperature have been reduced to about 400 psig and 350°F*, respectively. Hence, the major new impact of the functional requirements of RSB 5-1 on PWR plant design involves the transition region from hot standby conditions down to the conditions permitting operation of the RHRS.

There are four basic functions involved during plant operation in this transition region when the plant is being taken to cold shutdown. These are (a) boration of the RCS to the cold shutdown concentration, (b) circulation of the reactor coolant to promote mixing and uniform cooldown, (c) removal of stored and decay heat to reduce the RCS temperature from about 545°F to 350°F and (d) depressurization of the RCS from about 2200 to 400 psig. With

*For B&W plants this temperature is 305°F

loss of offsite power, these functions must be accomplished with the RCS in a natural circulation condition.

In addition, the requirement that the design be such that the plant could be taken to cold shutdown using only safety-grade systems has an impact on the systems and procedures used while the plant is operating in this transition region. For example, the letdown line in the chemical and volume control system has air-operated valves which are supplied by an air compressor which is not seismic Category 1. If the air supply is lost, boration of the RCS under natural circulation conditions and without letdown would be required.

There is insufficient information available to permit reliable estimates of the times required to achieve adequate mixing of borated water under natural circulation conditions and using the particular systems and procedures specified by the applicant to meet the functional requirements of RSB 5-1. In additon, there may be other factors involving, for example, vessel stress limits in the upper head region and steam bubble formation which could limit cooldown or depressurization rates. Some of the effects which should be considered in setting up the tests and supporting analyses are as follows:

A. Boration

1. stratification leading to delay in mixing time
2. regional non-uniformity of boron concentration (e.g., pressurizer versus loops, idle versus active loops, upper head region versus loops)

3. availability of letdown and auxiliary pressurizer spray
4. boration versus RCS temperature adequate to maintain margin for shutdown during cooldown

B. Circulation

1. effect of isolated steam generator
2. need for circulation promoted by heat removal at steam generator to cool loops after RHRS in operation since RHRS tends to cool only reactor

C. Heat Removal

1. thermal stress in vessel upper head region could limit cooldown rates
2. effect of isolated steam generator which acts as heat source during cooldown.

D. Depressurization

1. steam bubble formation in upper heat region which could limit depressurization rate
2. steam bubble formation on primary side of steam generators (particularly isolated steam generator) which could limit depressurization rate

In view of the uncertainties involved in cooldown and depressurization under natural circulation conditions, RSB 5-1 requires that tests for PWRs, with supporting analysis, be conducted to (a) confirm that adequate mixing of borated water added prior to or during cooldown can be achieved under natural circulation conditions and permit estimation of the time required to achieve such mixing, and (b) confirm that the cooldown under natural circulation conditions can be achieved within the limits specified in the emergency operating procedures.

The first goal of the test is to demonstrate that adequate mixing and cooldown during operation in the transition region can be obtained using the procedures, systems and equipment which have been determined to meet the functional requirements of RSB 5-1. Hence, any exceptions to these procedures, systems and equipment should be identified and justified by the applicant, and approved by the staff, on the basis that they do not have a significant effect on the achievement of this test goal. The second test goal is to obtain information which can be used to prepare emergency operating procedures and to determine the adequacy of sizing of the seismic Category 1 condensate storage tank.

Prior to the test, a report should be submitted to the staff which defines the test goals, gives the technical bases for the test and justifies the acceptance criteria. The report should include calculations of natural circulation flow rates and loop transit times, estimates of expected boron concentrations, and should consider the effects of instrument errors, sample line transient times and instrument response times on interpretation of the test results. This report would serve to justify the proposed operating conditions and procedures to be used in the test and would be reviewed by the staff in conjunction with the proposed test procedures.

Within ninety days after completion of the test, a report on the test results should be submitted to the staff. This report should describe the results of the test, the interpretation of the test results, and the use of the results in estimating cooldown times, preparation of operating procedures and sizing of the condensate storage tank.

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