TVA

SEQUOYAH

UNIT 1

MASTER PLAN

FOR THE

SPECIAL LOW-POWER TEST PROGRAM

8004140332

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PURPOSE OF PROGRAM

The purpose of the Sequoyah Nuclear Plant special low power test program was defined in December 1979 by S. David Freeman, Chairman of the Board of TVA, as an additional activity at the plant during the period of time after construction and before resumption of licensing nuclear plants for operation which could provide significant nuclear safety benefits to the nuclear industry.

The test program as submitted will provide meaningful technical information and will enhance operator training without posing safety hazards or economic risks. These tests will provide a demonstration of reactor operation in the natural circulation mode under both normal and certain degraded conditions.

SUMMARY

The special test program (STP) for the Sequoyah Nuclear Plant is described in this document. This master document was requested by the NRC staff to assist in their review of the STP. The document is divided into sections which cover:

(1) A general test program description
(2) The safety evaluation for each test
(3) The test schedule
(4) The startup test administrative procedures
(5) The test program procedures

These tests are designed primarily to provide for improvements in operator training and plant operations. Also, the test are in three technical information categories: natural circulation verification, boron mixing verification, and loss of power verification. These natural circulation tests will confirm that natural circulation can be obtained under expected and degraded plant conditions. It is believed that the natural circulation tests will significantly increase the technical data base concerning natural convection cooling physical parameters under a variety of complex conditions. The boron mixing and cooldown test will determine that the reactor coolant system can be uniformly borated during natural convection cooling. The simulated loss of all offsite and onsite alternating current power test will provide new data to confirm that pressurized water reactors can operate in the natural circulation mode under the degraded condition of loss of all ac power.

Section 1

INTRODUCTION

Section 1

INTRODUCTION

Background

TVA proposed to NRC, in a letter dated December 3, 1979, to Joseph Hendrie (NRC) from S. David Freeman, Chairman of the Board of TVA, that TVA pursue limited activities to the benefit of nuclear safety while TVA's Sequoyah Nuclear Plant awaited a full power license decision from the NRC. TVA's principle proposal was to conduct a special low power test program to be performed at power levels which would not allow the accumulation of significant radionuclide inventory in the core and which would not preclude implementation of physical changes to the plant resulting from the TMI-II "Lessons Learned." The NRC staff and the Advisory Committee on Reactor Safeguards (ACRS) have agreed that the special low power test program as outlined should be conducted. TVA has submitted draft test procedures in a letter dated January 18, 1980, from L. M. Mills (TVA) to L. S. Rubenstein (NRC) and has discussed the program with the NRC staff on January 26, 1980.

TVA finalized its safety evaluation and individual test procedures in its submittal of March 27, 1980, from L. M. Mills to L. S. Rubenstein. This document completes the TVA submission of material required by NRC for review and approval of the special test program. However, the document will be revised before, during, and after the program to provide a basis for NRC's and TVA's continuing review of the tests.

Test Program Summary

The test purpose and objectives are defined in the Standard Practice Procedure, SQA 109 (see section 4) and in each of the test program procedures (see section 5). SQA 109 ties the program together administratively and ensures that standard practice, including quality assurance, is followed during the development, conduct, and review of each test.

Normal operating procedures apply, unless otherwise specified, and are used throughout the special test program. Those necessary deviations are contained in the special test instruction. These test procedures have been reviewed relative to the normal operating procedures to eliminate possible ambiguities. Special emergency procedures are not required. If limits were to be exceeded, normal operating and training procedures would apply. Safety limits, setpoints, and limitations of operation have been changed (lowered or adjusted) to provide safety margin as required by the safety evaluation (section 2).

The latest revisions of each test procedure have specific criteria for operator manual initiation of reactor trip and safety injection. There are three specified limits for reactor trip and five specified limits for safety injection. The same limits and actions apply to all tests.

TVA has issued an administrative procedure (Sequoyah Nuclear Plant, Standard Practice SQA 109) which addresses the conduct of the special test program. This procedure includes the following information for the special test program: purpose and objectives, responsibility of organizations, preparation, review and approval of instructions, conduct of tests (including other applicable administrative procedures), and evaluation and approval of results.

This procedure also specifies that the shift engineer (SRO), assisted by the assistant shift engineer (SRO) and the licensed unit operators, is responsible for instituting immediate action in any situation to eliminate difficulties and to preclude violation of the operating license, technical specifications, or to avert possible injury to personnel or equipment damage.

The test engineer can stop a test when, in his opinion, conditions warrant such action. However, he has no authority over the licensed operator. If the test engineer stops a test, it is the licensed operator's responsibility to put the plant in an acceptable condition.

The special tests are scheduled as a part of the normal startup test program. The sequence in which the individual tests are performed is a part of the master sequencing instruction (S.U. 7.1) in the Sequoyah Startup Test Procedures.

Each individual test specifies the conditions to be established (including exceptions to technical specifications) and maintained. Also, as indicated above, manual operator actions are provided if certain limits are exceeded. All emergency procedures are applicable. Depending on the circumstances of the emergency, the appropriate emergency procedure will be used.

At the conclusion of each test there are specific steps in the procedure (see Appendix E) to reinstate all equipment according to the normal technical specification alignment unless another test, which requires the same change, is to immediately follow.

Test Program Description

The Sequoyah Nuclear Plant special low power test program consists of nine tests. The test program is in conjunction with, and in addition to, the normal startup test program. The test objectives and methods are clearly summarized in section 1. At the beginning of each test procedure (see section 5), a test description is provided. These pages are repeated in this section for the convenience of the reader. The test program sequence is as defined in S.U.-7.1 (see section 3). The weekly test program schedules will be available from the plant superintendent.

NATURAL CIRCULATION TEST

TEST DESCRIPTION

The test will be initiated by simultaneously tripping all reactor coolant pumps while at 3% power. The transient response will be monitored and establishment of natural circulation verified. Core exit thermocouples will be monitored to determine the core flow distribution. After stable conditions have been established, forced circulation will be reestablished.

TEST DESCRIPTION

This test is intended to provide a significant demonstration of reactor operation in the natural circulation mode under the degraded condition of loss of offsite AC power. The initial conditions for this test shall be as follows:

a. The reactor shall be at approximately 1% power. (Simulating reactor decay heat at hot standby following power operation).

b. All four reactor coolant pumps operating.

c. Auxiliary Feedwater System in service operating on offsite power.

d. Pressurizer Heaters in service controlling pressure.

e. Primary System at normal operating temperature and pressure.

This test will be conducted by simultaneously tripping all four reactor coolant pumps and initiating a blackout on the unit 1 6.9-kV shutdown boards which will result in a loss of motor-driven auxiliary feedwater pumps and pressurizer heaters. After the appropriate time delay, the diesel generators will energize the 6.9-kV shutdown boards and the motor-driven auxiliary feedwater pumps and pressurizer heaters will be reenergized. The establishment of natural circulation will be verified.

NATURAL CIRCULATION WITH LOSS OF PRESSURIZER HEATERS

TEST DESCRIPTION

The test will be initiated by tripping pressurizer heaters and reactor coolant pumps. Establishment of natural circulation will be verified and core exit thermocouples monitored to determine the core flow distribution. System pressure will be monitored to determine the rate of depressurization and, prior to reaching saturation, control of the saturation margin will be verified through the use of primary system charging flow and secondary system steam flow.

TEST DESCRIPTION

With natural circulation established at ~ 1% rated thermal power and reduced reactor coolant temperature, steam generators will be isolated sequentially to determine the effect on natural circulation conditions. Isolation of up to 2 steam generators will be tested if limitations permit. Steam generators will then be sequentially returned to service to verify that natural circulation can be reestablished.

TEST DESCRIPTION

The reactor coolant pumps are tripped with the reactor at 3% of rated power. The reactor coolant system is depressurized by turning off the pressurizer heaters and possibly turning on auxiliary sprays. Reactor power is reduced to 1.5% one hour after the pumps are tripped to approximate decay heat conditions. Saturation margin is monitored and increased charging and/or steam flow is used to maintain test limits.

TEST DESCRIPTION

The reactor coolant pumps are tripped with the reactor subcritical and the primary system at hot standby. All steam generators are isolated on the secondary side. Charging and letdown are increased to maximum capacity. After 30 minutes charging and letdown is reduced to minimum. The cooldown rates are observed for these maximum and minimum conditions.

SIMULATED LOSS OF ALL ONSITE AND OFFSITE AC POWER

Test Description

This test is intended to provide a significant demonstration of reactor operation in the natural circulation mode under the degraded condition of loss of all onsite and offsite AC power. For the purpose of plant and equipment safety, this total blackout condition will be simulated by the selective deenergizing of components and equipment.

TEST DESCRIPTION

With stagnant (no flow) conditions existing throughout the primary system, core power will be increased to simulate decay heat and steam generators will be utilized to establish a heat sink. Establishment of natural circulation will be verified by observing the response of the core exit thermocouples, hot leg wide range temperature indication, and cold leg wide range temperature indication. Core exit thermocouples will be monitored to access core flow distribution.

FORCED CIRCULATION COOLDOWN

Test Description

This test will generate a correction factor which will be applied to the excore detector outputs in order to compensate for PV downcomer shadowing during a cooldown from ~ 550°F to ~ 450°F.

The RCS will initially be ~ 3% power, in forced circulation. A cooldown via steam dumps will be initiated and continue until T_{avg} is approximately 450°F.

During the cooldown primary side calorimetrics will be performed, movable detector integral power calculations performed, and excore detector data obtained simultaneously.

Power should be maintained as constant as possible using the results of the primary side calorimetric and integral power calculations. Data reduction will be on a continuous basis.

After reaching ~ 450°F the plant will be allowed to heat up and additional data will be obtained.

Data reduction will average the cooldown and heatup data and generate an excore detector indicated power correction factor as a function of the average cold leg temperature.

TEST DESCRIPTION

This test will demonstrate that the Reactor Coolant System (RCS) can be uniformly borated approximately 100 ppm while in natural circulation. Boron samples will be taken continuously from the RCS (1 and 3) hot legs and the pressurizer to verify concentration uniformity.

This test will also demonstrate the capability to cool down the RCS on natural circulation using four steam generators. Cooldown will proceed until the RCS temperature is approximately 450°F.

Auxiliary sprays will be used to provide boron mixing between the PRZR and RCS. This will also demonstrate depressurization capability.

Section 2

SAFETY EVALUATION OF SPECIAL TEST PROGRAM

The proposed test program has been reviewed by the Westinghouse Electric Corporation and the Tennessee Valley Authority. This evaluation presents the results of that review.

1.0 DESCRIPTION OF TESTS

1.1 NATURAL CIRCULATION TEST (TEST 1)

Objective - To demonstrate the capability to remove decay heat by natural circulation.

Method - The reactor is at approximately 3% power and all Reactor Coolant Pumps (RCP's) are operating. All RCP's are tripped simultaneously with the establishment of natural circulation indicated by the core exit thermocouples and the wide range RTD's.

1.2 NATURAL CIRCULATION WITH SIMULATED LOSS OF OFFSITE AC POWER (TEST 2)

Objective - To demonstrate that following a loss of offsite AC power, natural circulation can be established and maintained while being powered from the emergency diesel generators.

Method - The reactor is at approximately 3% power and all RCP's are operating. All RCP's are tripped and a station blackout is simulated. AC power is returned by the diesel generators and natural circulation is verified.

1.3 NATURAL CIRCULATION WITH LOSS OF PRESSURIZER HEATERS (TEST 3)

Objective - To demonstrate the ability to maintain natural circulation and saturation margin with the loss of pressurizer heaters.

Method - Establish natural circulation as in Test 1 and turn off the pressurizer heaters at the main control board. Monitor the system pressures to determine; the saturation margin, the depressurization rate, and the effects of charging/letdown flow and steam generator pressure on the saturation margin.

1.4 EFFECT OF STEAM GENERATOR SECONDARY SIDE ISOLATION ON NATURAL CIRCULATION (TEST 4)

Objective - To determine the effects of steam generator secondary side isolation on natural circulation.

Method - Establish natural circulation conditions as in Test 1 but at 1% power. Isolate the feedwater and steam line for one steam generator and establish equilibrium. Repeat this for one more steam generator until two are isolated. Return the steam generators to service in reverse order.

1.5 NATURAL CIRCULATION AT REDUCED PRESSURE (TEST 5)

Objective - To demonstrate the ability to maintain natural circulation at reduced pressure and saturation margin. The accuracy of the saturation margin program of the plant computer will also be verified.

Method - The test method is the same as for Test 3, with the exception that the pressure decrease can be accelerated with the use of auxiliary pressurizer sprays. The saturation margin will be decreased to approximately 20°F.

1.6 COOLDOWN CAPABILITY OF THE CHARGING AND LETDOWN SYSTEM (TEST 6)

Objective - To determine the capability of the charging and letdown system to cool down the RCS with the steam generators isolated and one RCP operating.

Method - With the reactor shut down, trip three of the RCP's and isolate all four of the steam generators. Vary the charging and letdown flows and monitor the primary system temperatures to determine the heat removal capability.

1.7 SIMULATED LOSS OF ALL ONSITE AND OFFSITE AC POWER (TEST 7)

Objective - To demonstrate that following a loss of all onsite and offsite AC power the decay heat can be removed by natural circulation using the auxiliary feedwater system in the manual mode.

Method - The reactor is at approximately 1% power and all RCP's are running. All RCP's are tripped and a total station blackout is simulated. Instrument and lighting power is provided by the backup batteries since the diesels are shut down.

1.8 ESTABLISHMENT OF NATURAL CIRCULATION FROM STAGNANT CONDITIONS (TEST 8)

Objective - To demonstrate the establishment of natural circulation from stagnant (no flow) conditions in the primary system using reactor power to simulate decay heat.

Method - The reactor is critical in the HZP testing range and all RCP's are operating. Trip the RCP's and isolate the steam generators (feed and steam lines). When flow indicates zero, increase reactor power to approximately 1% and unisolate the steam generators. Natural circulation will be verified by observing the response of the hot leg temperatures and the core exit thermocouples.

1.9 FORCED CIRCULATION COOLDOWN (TEST 9A)

Objective - To determine the excore detector indicated power correction factor as a function of average cold leg temperature. This is required for Tests 4, 8, and 9B.

Method - The reactor is at approximately 3% power and all RCP's are running. The primary system is cooled down in $10^{\circ}F$ increments to $450^{\circ}F$ with calorimetric data and the indicated excore power obtained for correlation with T_{cold}. Data is also taken during the heatup to further verify the results.

1.10 BORON MIXING AND COOLDOWN (TEST 9B)

Objective - To demonstrate that the RCS can be uniformly borated while in natural circulation. Also demonstrate the capability to cool down the RCS in the natural circulation mode.

Method - Establish natural circulation at approximately 3% power and with relatively deep rod insertion. Borate the RCS by approximately 100 ppm through the normal boration path maintaining reactor power with rod withdrawal. Once the system is mixed, cool down the RCS maintaining constant reactor power.

2.0 IMPACT ON TECHNICAL SPECIFICATIONS

2.0 Evaluation of the proposed tests indicates that 13 technical specification requirements will be violated and will therefore require exceptions during the performance of the special test program. The matrix below lists the technical specifications that will not be met for each test.

		Test									
	Technical Specification	1	2	3	4	5	6	7	8	9A	9B
2.1.1	Core Safety Limits	X	X	X	X	X	X	X	X		X
2.2.1	Various Reactor Trips										
	Overtemperature ΔT	X	X	X	X	X		X	X		X
	Overpower ΔT	X	X	X	X	X		X	X		X
	Steam Generator Level	X	X	X	X	X	X	X	X	X	X
3.1.1.3	Moderator Temperature Coefficient				X				X	X	X
3.1.1.4	Minimum Temperature for Criticality				X				X	X	X
3.3.1	Various Reactor Trips										
	Overtemperature ΔT	X	X	X	X	X		X	X		X
	Overpower ΔT	X	X	X	X	X		X	X		X
	Steam Generator Level	X	X	X	X	X	X	X	X	X	X
3.3.2	Safety Injection - All automatic functions	X	X	X	X	X	X	X	X	X	X
3.4.4	Pressurizer			X		X		X			
3.5.1.2	Upper Head Injection System	X	X	X	X	X	X	X	X	X	X
3.7.1.2	Auxiliary Feedwater		X					X			
3.8.1.1	AC Power Sources		X					X			
3.8.2.1	AC Onsite Power Distribution System		X					X			
3.8.2.3	DC Distribution System		X					X			
3.10.3	Special Test Exceptions - Physics Tests				X					X	X

2.1 Each technical specification requiring an exception is listed below along with the reason for the exception and the basis for continued operation.

2.1.1 Reactor Core Safety Limits (T.S. 2.1.1)

This specification restricts operation to within the nucleate boiling regime by limiting reactor coolant system average temperature as a function of reactor power and reactor coolant system pressure during 4 loop operation. In the natural circulation mode, this specification will not be met because the reactor coolant pumps will not be operating.

During performance of the special tests, core exit thermocouple temperature will be limited to 610°F, core average temperature will be limited to 578°F, reactor coolant loop delta T will be limited to 65°F, and the reactor will be manually tripped if the margin to saturation reaches 15°F. No boiling will be experienced in the core and the temperature limits of specification 2.1.1 will be met.

2.1.2 Reactor Trip System Instrumentation Setpoints (T.S. 2.2.1)

1. Overpower and Overtemperature ΔT

The overtemperature and overpower ΔT trip functions obtain a temperature input from sensors located in the RTD bypass loops. During natural circulation, flow through the bypass loops will be extremely low and the temperature indication from these loops will be highly suspect. To prevent inadvertent tripping of the plant when in the natural circulation mode, these trip functions will be bypassed. Operator action will be relied on to provide the protective function previously furnished by the ΔT trips. This will be accomplished by manual reactor trip or safety injection actuation when:

Pressurizer Level (trip) < 17% span or an unexplained decrease of more than 5% not concurrent with a T_{ave} change

Pressurizer Level (SI) < 10% span or an unexplained decrease of more than 10% not concurrent with a T_{ave} change

Pressurizer Pressure (SI) - Decrease by 200 psi or more in an unplanned or unexplained manner

Steam Generator Level (trip) < 5% narrow range span

Steam Generator Level (SI) < 0% narrow range span or equivalent wide range level

RCS subcooling (trip) $\leq$ 15°F T_{sat} margin

RCS subcooling (SI) $\leq$ 10°F T_{sat} margin

NIS Power Range, 2 channels (trip) > 10% rated thermal power

2. Steam Generator Water Level

This trip function ensures that there is a minimum volume of water in the steam generators above the tops

of the U-tubes to maintain a secondary side heat sink. At times during the test program it may be difficult to hold the 12% margin between the normal operating level and the low-level trip setpoint at 21% of span. The required setpoint (volume of water above the U-tubes) is dependent on the amount of core heat and the time required to establish adequate feed flow to the steam generators. With the plant limited to 5% rated thermal power or less and being at BOL on Cycle 1, there will be little or no decay heat present: the heat source will be the core operating at the limited power level. Tripping the reactor on any of the different operable trip functions or the operator action points (see section on Operational Safety Criteria) will ensure that this requirement will be met. Thus, we find that it is acceptable to lower the trip setpoint to 5% span for all of the special tests.

2.1.3 Moderator Temperature Coefficient (T.S. 3.1.1.3)

The moderator temperature coefficient is limited to 0pcm/^{o}F or more negative to ensure that the value of the coefficient remains within the limiting conditions assumed for this parameter in the plant accident and transient analyses. When performing tests during which the primary system is cooled, the coefficient will become slightly positive. However, the isothermal temperature coefficient is expected to remain negative or approximately zero. This coefficient can be interpreted as the sum of the moderator and doppler coefficients; with it at a zero or negative value, the impact from rapid heatups and cooldowns will be minimized. In addition, the effect of a small positive moderator temperature coefficient has been considered in the analyses performed for the test conditions.

2.1.4 Minimum Temperature for Criticality (T.S. 3.1.1.4)

The minimum temperature for criticality is limited to 541°F by specification 3.1.1.4 and 531°F by specification 3.10.3. To perform tests 4, 9A, and 9B, it is expected that the RCS average temperature will drop below 531°F. We have determined that operation with T_{avg} as low as 425°F is acceptable assuming that:

1. Control Bank D is inserted to no deeper than 100 steps withdrawn, and

2. Power range neutron flux low setpoint and intermediate range neutron flux reactor trip setpoints are reduced from 25% RTP to 7% RTP.

This will considerably reduce the consequences of possible transients by (1) reducing individual control rod worths (Bank D) on unplanned withdrawal, (2) reducing bank worth (Bank D) on unplanned withdrawal, (3) maximizing reactivity

insertion capability consistent with operational requirements, (4) limiting maximum power to a very low value on an unplanned power excursion, and (5) allowing the use of the "at power" reactor trips as backup trips rather than as primary trips.

2.1.5 Reactor Trip System Instrumentation (T.S. 3.3.1)

The overpower ΔT, overtemperature ΔT, and the steam generator water level trips will not meet the operability requirements of this specification. However, this specification can be excepted for the reasons previously noted under Reactor Trip System Instrumentation Setpoints (T.S. 2.2.1).

2.1.6 Engineered Safety Feature Actuation System Instrumentation (T.S. 3.3.2)

In order to prevent inadvertent safety injection during the performance of the special tests, the automatic injection functions will be blocked. The reactor trip, control room trip indications/alarms, and manual safety injection initiation will be operable. If the ESF instrumentation detects a condition which is over the trip setpoint, the safety injection signal will provide a reactor trip and control room indication/alarm. The automatic injection will be prevented by forcing the logic to see that the reactor trip breakers are open. If the operator determines that the actual situation requires injection, he will manually initiate safety injection using the control room hand switches. We believe that this mode of operation is acceptable for the short period of time these tests will be carried out based on the following:

1. Close observation of control room trip indications/alarms and rigid adherence to the action points specified in the section on Operational Safety Criteria will ensure manual safety injection actuation.

2. Little or no decay heat is present in the system, thus safety injection serves primarily as a pressurization function (shutdown margin capability is considerably more than 1.6% ΔK/K for control rods at or above the insertion limits).

Blocking these functions will allow the performance of these tests at low power, pressure, or temperature while close operator surveillance will ensure initiation of safety injection, if required.

2.1.7 Pressurizer (T.S. 3.4.4)

The pressurizer provides the means of maintaining pressure control for the plant. Normally this is accomplished through the use of pressurizer heaters and spray. In several tests the pressurizer heaters will be either turned off or rendered inoperable by loss of power. This mode of operation is acceptable in that pressure control will be maintained through the use of pressurizer level and charging/letdown flow.

2.1.8 Upper Head Injection Accumulators (T.S. 3.5.1.2)

During the performance of these tests the UHI system will be locked out. This is to prevent inadvertent actuation of the system. The UHI system provides borated water in the event of significant depressurization indicating a LOCA. It has been noted that with little or no decay heat at low power levels, this system provides little or no benefit in the event of a LOCA. Based on this knowledge, We have determined that it is acceptable to perform these special tests with the UHI system locked out.

2.1.9 Auxiliary Feedwater System (T.S. 3.7.1)

The auxiliary feedwater system will be rendered partially inoperable for two tests. The two tests simulate some form of loss of AC power, i.e., motor driven auxiliary feedwater pumps inoperable. Westinghouse has determined that this is acceptable for these two tests because of the little or no decay heat present allowing sufficient time (∿ 30 minutes) for operating personnel to rack in the pump power supplies and regain steam generator level.

2.1.10 Power Sources (T.S. 3.8.1.1, 3.8.2.1, and 3.8.2.3)

Technical Specification 3.8.1.1 requires two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution systems and requires that four separate and independent diesel generator sets be operable.

Technical Specification 3.8.2.1 requires that the 6900- and 480-volt shutdown boards and 120-volt AC vital instrument boards be operable and energized.

Technical Specification 3.8.2.3 requires that the DC vital battery channels be energized and operable. Each channel shall include one 125-volt DC board, one 125-volt DC battery bank, and a full capacity charger.

In test 2, the test switches on the 6900-volt shutdown boards logic relay panels 2A-A and 2B-B for diesels 2A-A and 2B-B are placed in the test positions. This action prevents the unit two diesels from responding to a blackout signal. The circuit breakers between the unit one unit boards and 6900-volt shutdown boards 1A-A and 1B-B are opened to simulate a loss of offsite power. If for some reason either of the diesels fail to start, power will be returned to the effected board by closing either the normal or alternate feeder breaker.

Test 7 simulates a loss of all onsite and offsite AC power by selectively de-energizing system components and equipment. As part of the test, the Class 1E distribution is aligned so that the only supply to the 125-volt vital AC instrument boards will be the 125-volt vital batteries. This will be done opening the circuit breaker between the chargers and the battery boards. The 480-volt supply to the inverters will also be opened. At this point, the loads supplied by the 120-volt vital AC instrument boards will be supplied only by the batteries. If a problem develops with the batteries, power may be restored to the battery boards by reclosing the circuit breakers from the chargers or by closing the breakers for the 480-volt AC supply to the vital inverters.

2.1.11 Special Test Exceptions - Physics Tests (T.S. 3.10.3)

This specification allows the minimum temperature for criticality to be as low as 531°F. Since it is expected that RCS T_{avg} will be taken as low as 450°F, this specification will be excepted. See Section 2.1.4 for basis of acceptability.

2.1.12 Technical Specifications Not Excepted

While not applicable at power levels below 5% RTP, the following technical specification limits can be expected to be exceeded:

1. Heat Flux Hot Channel Factor - $F_Q(Z)$ (T.S. 3.2.2)
   At low temperatures and flows $F_Q(Z)$ can be expected to be above normal for 5% RTP with RCP's running. However, at such a low power level no significant deviations in burnup or Xe peaks are expected.

2. RCS Flow Rate and R - ($F_{\Delta H}$) (T.S. 3.2.3)
   At low temperatures and flow $F_{\Delta H}$ can be expected to be higher than if pumps are running. However, no significant consequences for full power operation are expected.

3. Quadrant Power Tilt Ratio (T.S. 3.2.4)
   With no, one, or two pumps running and critical, core power distributions resulting in quadrant power tilt may form. At low power levels and for short periods of time, these tilts will not significantly influence core burnup.

4. DNB Parameters (T.S. 3.2.5)
   In the performance of several tests the plant will be depressurized below 2230 psia. At low operating power levels this depressurization is not significant as long as subcooling margin is maintained.

3.0 OPERATIONAL SAFETY CRITERIA

3.0 During the performance of these tests, the operator must meet the following set of criteria for operation.

3.1 For all tests

a.	Primary System Subcooling (T_{sat} Margin)	> 20°F
b.	Steam Generator Water Level	> 33% Narrow Range Span
c.	Pressurizer Water Level	
	(1) With RCP's Running	> 25% Span
	(2) Natural Circulation	≥ Value when RCP's tripped
d.	Loop ΔT	≤ 65°F
e.	T_{avg}	≤ 578°F
f.	Core Exit Temperature (highest)	≤ 610°F
g.	Power Range Neutron Flux Low Setpoint and Intermediate Range Neutron Flux Reactor Trip Setpoints	≤ 7% RTP
h.	Control Bank D	100 steps withdrawn or higher

3.2 Reactor trip and test termination must occur if any of the following conditions are met

a.	Primary System Subcooling (T_{sat} Margin)	≤ 15°F
b.	Steam Generator Water Level	< 5% Narrow Range Span
c.	NIS Power Range, 2 Channels	> 10% RTP
d.	Pressurizer Water Level	< 17% Span or an unexplained decrease of more than 5% not concurrent with a T_{avg} change
e.	Any Loop ΔT	> 65°F
f.	T_{avg}	> 578°F
g.	Core Exit Temperature (highest)	> 610°F
h.	Uncontrolled rod motion	

3.3 Safety injection must be manually initiated if any of the following conditions are met

a.	Primary System Subcooling (T_{sat} Margin)	≤ 10°F
b.	Steam Generator Water Level	< 0% Narrow Range Span or Equivalent Wide Range Level
c.	Containment Pressure	≥ 1.54 psig
d.	Pressurizer Water Level	< 10% Span or an unexplained decrease of more than 10% not concurrent with a T_{avg} change
e.	Pressurizer Pressure	Decreases by 200 psi or more in an unplanned or unexplained manner

Safety injection termination must be in accordance with the termination criteria set forth in the plant Emergency Operating Instructions.

3.4 These operating and function initiating conditions are selected to ensure that the base conditions for safe operation are met, i.e.,

a. Sufficient margin to saturation temperature at system pressure to ensure adequate core cooling,

b. Sufficient steam generator level to ensure an adequate secondary side heat sink,

c. Sufficient level in the pressurizer to ensure coverage of the heaters to maintain pressure control,

d. Sufficient control rod worth to ensure adequate shutdown margin and minimize impact of uncontrolled bank withdrawal, and

e. Limit maximum possible power level in the event of an uncontrolled power increase.

4.0 EVALUATION/ANALYSIS OF FSAR EVENTS

In this section the safety effects of those special test conditions which are outside the bounds of conditions assumed in the FSAR are evaluated. The interaction of these conditions with the transients analyses in the FSAR are discussed. For those transients for which it is not obvious that the FSAR is limiting, new analyses are provided.

4.1 Evaluation of Transients

The effect of the unusual operating conditions on the transients analyzed in the FSAR are evaluated.

4.1.1 CONDITION II - Incidents of Moderate Frequency

4.1.1.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition

Restriction of control rod operation to manual control and constant operator monitoring of rod position, nuclear power, and temperatures greatly reduce the likelihood of an uncontrolled RCCA withdrawal. Furthermore, during all but one test (9B) the total inserted control rod worth will be less than that required to go prompt critical. Operation without reactor coolant pumps, and in some cases with a positive moderator temperature reactivity coefficient, tend to make the consequences of RCCA withdrawal worse compared to the operating conditions assumed in the FSAR. For these reasons the operating procedures require that following any reactor trip at least one reactor coolant pump will be restarted and the reactor boron concentration will be such that it will not go critical with less than 100 steps withdrawal on D Bank. Thus, the only opportunity for a rod withdrawal from subcritical or zero power conditions without RCP flow is during test No. 8. An analysis of this event is presented in Section 4.2.1.

4.1.1.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

The same considerations discussed in Paragraph 4.2.1.1 apply here. In addition, the low operating power and the Power Range Neutron Flux Low and Intermediate Range Neutron Flux trip setpoints (7% RTP) act to mitigate this incident, while lack of the overtemperature ΔT trip removes some of the protection provided in the FSAR case. An analysis is discussed in Paragraph 4.2.2.

4.1.1.3 Rod Cluster Control Assembly Misalignment

The FSAR discussion concerning static RCCA misalignment applies to the test conditions. The consequences of a

dropped RCCA would be a decrease in power. Thus no increase in probability or severity of this incident is introduced by the test conditions.

4.1.1.4 Uncontrolled Boron Dilution

The consequences of, and operator action time requirements for, an uncontrolled boron dilution under the test conditions are bounded by those discussed in the FSAR. The fact that the control rods will never be inserted to the insertion limits, as well as the Power Range Neutron Flux Low Setpoint and the concurrent operator monitoring of reactor power, temperature and charging system operation, provides added protection.

4.1.1.5 Partial Loss of Forced Reactor Coolant Flow

Because of the low power limits the consequences of loss of reactor coolant pump power are trivial; they are bounded by normal operating conditions for these tests.

4.1.1.6 Startup of an Inactive reactor Coolant Loop

When at least one reactor coolant pump is operating, the power limit for these tests results in such small temperature differences in the reactor coolant system that startup of another loop cannot introduce a significant reactivity disturbance. In natural circulation operation, inadvertent startup of a pump would reduce the core water temperature and thus provide a change in reactivity and power. Because of the small moderator reactivity coefficient at beginning of life the power increase in the worst condition would be small and gradual and the flow to power ratio in the core would be increasing. The Power Range Neutron Flux Low Setpoint (7% RTP) reactor trip provides an upper bound on power. Because of the increase in flow-to-power ratio and because of the low setpoint on the reactor trip, DNB is precluded in this transient.

4.1.1.7 Loss of External Electrical Load and/or Turbine Trip

Because of the low power level, the disturbance caused by any loss of load is small. The FSAR case is bounding; the turbine generator will not be in operation during the special test program.

4.1.1.8 Loss of Normal Feedwater

Because of the low power level, the consequences of a loss of feedwater are bounded by the FSAR case. In the case of loss of all feedwater sources, if the reactor is not shutdown manually, it would be tripped on Low-Low Steam Generator Water Level. Ample time is available to reinstitute auxiliary feedwater sources.

4.1.1.9 Loss of Offsite Power to the Station Auxiliaries (Blackout)

Because of the low power level, the consequences of a loss of offsite power are bounded by the FSAR case.

4.1.1.10 Excessive Heat Removal Due to Feedwater System Malfunctions

The main feedwater control valves will not be used while the reactor is at power or near criticality on these tests. Thus, the potential water flow is restricted to the main feedwater bypass valve flow or auxiliary feedwater flow, about 15% of normal flow. The transient is further mitigated by the low operating power level, small moderator temperature reactivity coefficient, the low setpoints on the Intermediate and Power Range Nuclear Flux trips, and close operator surveillance of feed flow, RCS temperatures, RCS pressure, and nuclear power. The case of excess feedwater with very low reactor coolant flow is among the cooldown transients discussed in more detail in Section 4.2.3.

4.1.1.11 Excessive Load Increase Incident

The turbine will not be in use during the performance of these tests, and load control will be limited to operation of a single steam dump or steam relief valve. The small moderator temperature reactivity coefficient also reduces the consequences of this transient. Close operator surveillance of steam pressure, cold leg temperature, pressurizer pressure, and reactor power, with specific initiation criteria for manual reactor trip, protect against an unwanted reactor power increase. In addition, the low setpoints for Power-Range and Intermediate-Range Neutron Flux reactor trips (7% RTP) limit any power transient. In addition, modification of the high steamline flow setpoint allows a reactor trip on Low Steam Pressure only. Analyses are discussed in Section 4.2.3.

4.1.1.12 Accidental Depressurization of the Reactor Coolant System

Close operator surveillance of pressurizer pressure and of hot leg subcooling, with specific initiation points for manual reactor trip, provides protection against DNB in the event of an accidental depressurization of the RCS. In addition, automatic reactor trip caused by the Low Pressurizer Pressure Safety Injection signal would occur even before core outlet subcooling reached the 15°F minimum limit specified for manual trip. During test 3 and 5, when this trip is bypassed to allow deliberate operation at low pressure, the pressurizer PORV block valves will be closed to remove the major credible source of rapid inadvertent depressurization. (The Low Pressure trip is automatically reinstated when pressure goes above 1970 psi and the PORV block valves will be reopened at that time.)

4.1.1.13 Accidental Depressurization of the Main Steam System

The FSAR analysis for accidental steam system depressurization indicates that if the transient starts at hot shutdown conditions with the worst RCCA stuck out of the core, the negative reactivity introduced by Safety Injection prevents the core from going critical. Because of the small moderator temperature reactivity coefficient which will exist during the test period, the reactor would remain subcritical even if it were cooled to room temperature even without Safety Injection. Thus the SAR analysis is bounding.

4.1.1.14 Spurious Initiation of the Safety Injection System at Power

In order to reduce the possibility of unnecessary thermal fatigue cycling of the reactor coolant system components, the actuation of high head charging in the safety injection mode, and of the safety injection pumps, by any source except manual action will be disabled. Thus, the most likely sources of spurious Safety Injection, i.e., spurious or "spike" pressure or pressure-difference signals from the primary or secondary systems, have been eliminated.

4.1.2 CONDITION III - Incidents of Moderate Frequency

4.1.2.1 Loss of Reactor Coolant from Small Ruptured Pipes

A review of the plant small break loss of coolant accident behavior during the low power testing sequence indicates that without automatic Safety Injection the system inventory and normal charging flow will provide the short term cooling for the small break transient. A sample calculation for a 2-inch break shows that the core remains covered for at least 6000 seconds. This is sufficient time for the operator to manually initiate SI and align the system for long term cooling.

The magnitude of the resulting clad heatup transient during a LOCA event from these conditions is significantly reduced from the FSAR basis scenario due to the low decay heat and core stored energy which will exist at the time of these tests.

4.1.2.2 Minor Secondary System Pipe Breaks

The consequences of minor secondary system pipe breaks are within the bounds of the transients discussed in Paragraph 4.2.3 (Analysis of Cooldown Transients).

4.1.2.3 Single Rod Control Cluster Assembly Withdrawal at Power

The single rod control cluster assembly withdrawal at power under natural circulation flow was

not analyzed. The FSAR analysis shows that assuming limited parameters for normal operation with forced flow, a maximum of 5% of the fuel rods could experience a DNBR of less than 1.3 following a single RCC withdrawal. As pointed out in the FSAR, no single electrical or mechanical failure in the control system could cause such an event. In addition, the probability of this event happening during these tests is reduced by:

1. Close operator surveillance of control rod motion with instructions to trip the reactor if uncontrolled rod motion occurs.

2. Rod control is restricted to operator manual control therefore any rod operation will be under close operator surveillance and any rod motion which occurs without operator demand will be interpreted as uncontrolled rod motion.

3. Close operator surveillance of reactor power and hot leg temperature.

In summary, due to the extremely low probability of single rod control cluster assembly withdrawal at power during this test sequence, analysis of this event is not warranted.

4.1.2.4 Other Incidents of Moderate Frequency

The consequences of an inadvertent loading of a fuel assembly into an improper position, complete loss of forced reactor coolant flow, and waste gas decay tank rupture, have been reviewed. The consequences of each of these events occurring during test conditions are bound by those described in the FSAR.

4.1.3 CONDITION IV - Limiting Faults

4.1.3.1 Major Rupture of Pipes Containing Reactor Coolant Up to and Including Double Ended Rupture of the Largest Pipe in the Reactor Coolant System (Loss of Coolant Accident)

A review of the plant loss of coolant accident behavior during the low power testing sequence indicates that without automatic safety injection there is sufficient cooling water readily available to prevent the fuel rod cladding from over heating. During the large break event, the reactor coolant system (RCS) will begin to depressurize to the containment, removing stored energy and decreasing the RCS inventory. The cold leg accumulators will begin injecting into the RCS, replacing inventory. As

a result, the vessel is filled to the bottom of the nozzles (at no time is the core uncovered). At 100 seconds, there is enough water in the reactor vessel below the nozzles to keep the core covered for 1.7 hours. This is sufficient time for the operator to manually initiate SI and align the system for long term cooling.

The magnitude of the resulting clad heatup transient during a LOCA event from these conditions is significantly reduced from the FSAR basis scenario by the low decay heat and core stored energy existing at the time of the performance of these tests.

4.1.3.2 Major Secondary System Pipe Rupture

During a cooldown transient caused by the secondary system, protection against excessive reactor power is provided by the following: the small moderator temperature reactivity coefficient; close operator surveillance of pressurizer pressure, cold leg temperature and reactor power, with specific criteria for manual reactor trip and safety injection; low trip setpoints on the intermediate range and power range Neutron Flux trips; reactor trip and MSIV closure on low steam pressure; reactor trip on low pressurizer pressure. Following reactor trip, assuming the most reactive RCCA stuck out of the core, the reactor would remain subcritical even if it were cooled to room temperature. Transient analyses for a steam pipe rupture are provided in Section 4.2.3. The consequences of a main feedline rupture are bounded in the cooldown direction by the steam pipe rupture discussion. Because of the low operating power, the heatup aspects of a feedline rupture are bounded by the FSAR discussion.

4.1.3.3 Steam Generator Tube Rupture

The steam generator tube rupture event may be categorized by two distinct phases. The initial phase of the event is analogous to a small LOCA event. Prior to operator-controlled system depressurization, the steam generator tube rupture is a special class of the small break LOCA transients, and the operator actions required to deal with the situation during this phase are identical to those required for mitigation of a small LOCA. Hence, evaluation of the steam generator tube rupture during this phase is wholly covered by the safety evaluation of the small LOCA.

After the appropriate operator actions have taken place to deal with the initial LOCA phase of the event, the remainder of the steam generator tube rupture accident

mitigation would consist of those operator actions required to isolate the faulted steam generator, cooldown the RCS, and depressurize the RCS to equilibrate primary RCS pressure with the faulted steam generator secondary pressure. These actions require utilization of the following systems:

1. Auxiliary feedwater control to faulted steam generator.

2. Steam line isolation of the faulted steam generator.

3. Steam relief capability of at least one nonfaulted steam generator.

4. RCS depressurization capability.

Evaluation of the TVA special test procedures has verified that all of the above systems are immediately available for operator control from the control room. Therefore, it is concluded that the ability to mitigate the steam generator tube rupture event is not compromised by the modifications required for operation at 5% power during the proposed tests, and that the analyses performed for the SAR regarding this event remain bounding.

4.1.3.4 Single Reactor Coolant Pump Locked Rotor

Because of the low power level, the locking of a single reactor coolant pump rotor is inconsequential.

4.1.3.5 Fuel Handling Accident

The FSAR analysis of a fuel handling accident is bounding.

4.1.3.6 Rupture of a Control Rod Drive Mechanism

Except for a short period during test 9B, the control rod bank insertion will be so limited (i.e., only Bank D inserted, with at least 100 steps withdrawn) that the worth of an ejected rod will be substantially less than the delayed neutron fraction. Thus, the power rise following a control rod ejection would be relatively gradual and terminated by the Power Range and Intermediate Range Neutron Flux reactor trips. During test 9B, the control rods will be inserted further while the reactor coolant pumps are running. In this case, the bank insertion will be less than the insertion limit at normal no-load conditions, thus the ejected rod worth would be lower and the consequences of an ejected rod less severe than those described in the FSAR. In test 9B, immediately following RCP trip

and coastdown, the control rod banks will be withdrawn gradually as boron is added until they reach the condition of only Bank D in, withdrawn to at least 100 steps. While the core power transient and power distribution following an RCCA ejection at this time would be less severe than those shown in the FSAR, the result of combining these ameliorating effects with the effect of the natural circulation flow rate on clad-to-water heat transfer and RCS pressure have not been analyzed. The extremely low probablility of an RCCA ejection during the brief period in the test sequence does not warrant such an analysis.

4.2 Analysis of Transients

4.2.1 ANALYSIS OF TRANSIENTS

An analysis was performed to bound the transient for test 8, i.e., the only case in which the reactor will be near critical but not at power without reactor coolant pumps running. The methods and assumptions used in the FSAR, Section 15.2.1.1 were used with the following exceptions:

1. Reactor coolant flow was 0.1% of nominal.

2. Control rod incremental worth and total worth were upper bound values for the D bank initially 100 steps withdrawn.

3. Moderator temperature reactivity coefficient was an upper bound (positive) for any core average temperature at or above 425°F.

4. The lower bound for total delayed neutron fraction for the beginning of life for Cycle 1 was used.

5. Reactor trip was initiated at 10% of full power.

6. DNB was assumed to occur spontaneously at the hot spot, at the beginning of the transient.

The resulting nuclear power peaked at 44% of full power, as is shown in Figure 4.2.1. The peak clad temperature reached was under 1200°F, as is shown in Figure 4.2.2. No clad failure is expected as a result of this transient.

4.2.2 ANALYSIS OF RCC BANK WITHDRAWAL AT POWER

Analyses of RCCA bank withdrawal transients were performed for natural circulation conditions. The transients were assumed to start from steady-state operating conditions at either 1% or 5% of full power, and with either all steamline isolation valves open or two of those valves closed. A

range of reactivity insertion rates up to the maximum for two banks moving was assumed for cases with all steamlines open, and up to the maximum for one bank moving for the cases with two steamlines isolated. Both maximum and minimum bounds on reactivity feedback coefficients for beginning of life, Cycle 1, were investigated. In all cases, reactor trip was initiated at 10% nuclear power.

Reactor conditions at the time of maximum core heat flux are shown in Figures 4.2.3 and 4.2.4 as functions of the reactivity insertion rate for three, four-loop active cases. For high reactivity insertion rates, the minimum reactivity coefficient cases give the greatest heat flux after the trip setpoint is reached, and have the lowest coolant flow rate at the time of peak heat flux. For these cases, even the slowest reactivity insertion rates studied did not result in any increase in core inlet temperature at the time of peak heat flux. For maximum feedback cases, however, the transients for very low reactivity insertion rates go on for so long that the core inlet temperature finally increases before trip, i.e., after approximately one and one-half minutes of continuous withdrawal. Thus, the cases shown bound the worst cases.

4.2.3 ANALYSIS OF COOLDOWN TRANSIENTS

Cooldown transients include feedwater system malfunctions, excessive steam load increase, accidental depressurization of the main steam system, and minor and major secondary system pipe ruptures. Attention has been focused on the possibility and magnitude of core power transients resulting from such cooldowns before reactor trip would occur. (Following reactor trip, no cooldown event would return the reactor to a critical condition.)

During natural circulation operation, approximately one to two minutes would elapse following a secondary side event before cold primary water reached the core; thus, considering the close and constant surveillance during these tests, ample time would be available for the operator to respond to such an event. Analyses were also performed to determine the extent of protection provided by automatic protection systems under trip conditions.

4.2.3.1 Load Increases

A load increase or a small pipe break, equivalent to the opening of a single power-operated steam pressure relief valve, a dump valve, or a safety valve, would cause an increase of less than four percent in reactor power, with a corresponding increase in core flow with natural circulation, assuming the bounding negative

moderator temperature coefficient for the beginning of life, Cycle 1. Thus no automatic protection is required, and ample time is available to the operator to trip the reactor, isolate feedwater to the faulted steam generator, and isolate the break to the extent possible. Calculated results for the sudden opening of a single steam valve, assuming the most negative BOL Cycle 1, moderator reactivity coefficient are shown in Figures 4.2.5 and 4.2.6 (5% initial power, steam dump valve), and in Figures 4.2.7 and 4.2.8 (1% initial power, two steam generators isolated, pressure relief valve).

4.2.3.2 High Flux Protection

Reactor trip on high nuclear flux provides backup protection for larger pipe breaks or load increases. Analyses were performed to determine the worst core conditions that could prevail at the time of high-flux trip, independent of the cause. The following assumptions were used:

1. Upper-bound negative moderator isothermal temperature coefficient, vs. core average temperature, for beginning of life, Cycle 1.

2. Lower-bound fuel temperature - power reactivity coefficient.

3. Initial operation with core inlet temperature 555°F.

4. Initial powers of 0% and 5% of full power were analyzed.

5. Hot leg coolant at incipient boiling at the time of reactor trip. This results in some boiling in the reactor. The negative reactivity introduced by core boiling would effectively limit power; this negative reactivity was conservatively neglected.

6. Uniform core inlet temperature and flow.

7. Reactor trip equivalent to 10% of full power at the initial inlet temperature. The power as measured by the NIS is assumed to be diminished from the true power by 1% for each 1°F decrease in reactor inlet temperature, resulting in a true power of greater than 10% at the time of trip.

8. Core flow rate as a function of core power was conservatively assumed equal to the predicted flow under steady-state operating conditions.

Analyses of core conditions based on these assumptions indicate that the DNB criterion of the FSAR is met.

4.2.3.3 Secondary Pressure Trip Protection

Large steamline ruptures downstream of the steamline check valves will affect all loops uniformly since the lines enter a common header just upstream of the HP turbine. Since the high steamline flow setpoint is set to zero flow (that is, the bystable is in the tripped condition), reactor trip and MSIV isolation will be activated by two out of four low steamline pressure signals. Low pressurizer pressure and Power Range Neutron Flux trip setpoints serve as further backups. The response to a double ended main steamline rupture downstream of the check valves is shown in Figures 4.2.9 and 4.2.10. An initial power of 5% and natural circulation conditions are assumed. In the example shown, the main steamline isolation valve on loop one was assumed to fail to close. No power excursion resulted, and the reactor remained subcritical after the trip.

For large steamline ruptures inside of the steamline check valves, the normal protection is provided by the steamline delta-P trip and isolation. However, this signal has been disabled for these tests since the steamline pressure difference is expected to exceed the trip setpoint range. For this transient, the Power Range Neutron Flux trip serves as a backup to manual protection. An example case is shown in Figures 4.2.11 and 4.2.12. For this case, operation of 1% power was assumed with steam generators three and four isolated and with natural circulation. A double-ended rupture of the main steamline upstream of the steamline venturi was assumed. The transient was allowed to continue without manual trip until the Power Range Neutron Flux trip was reached at 104 seconds.

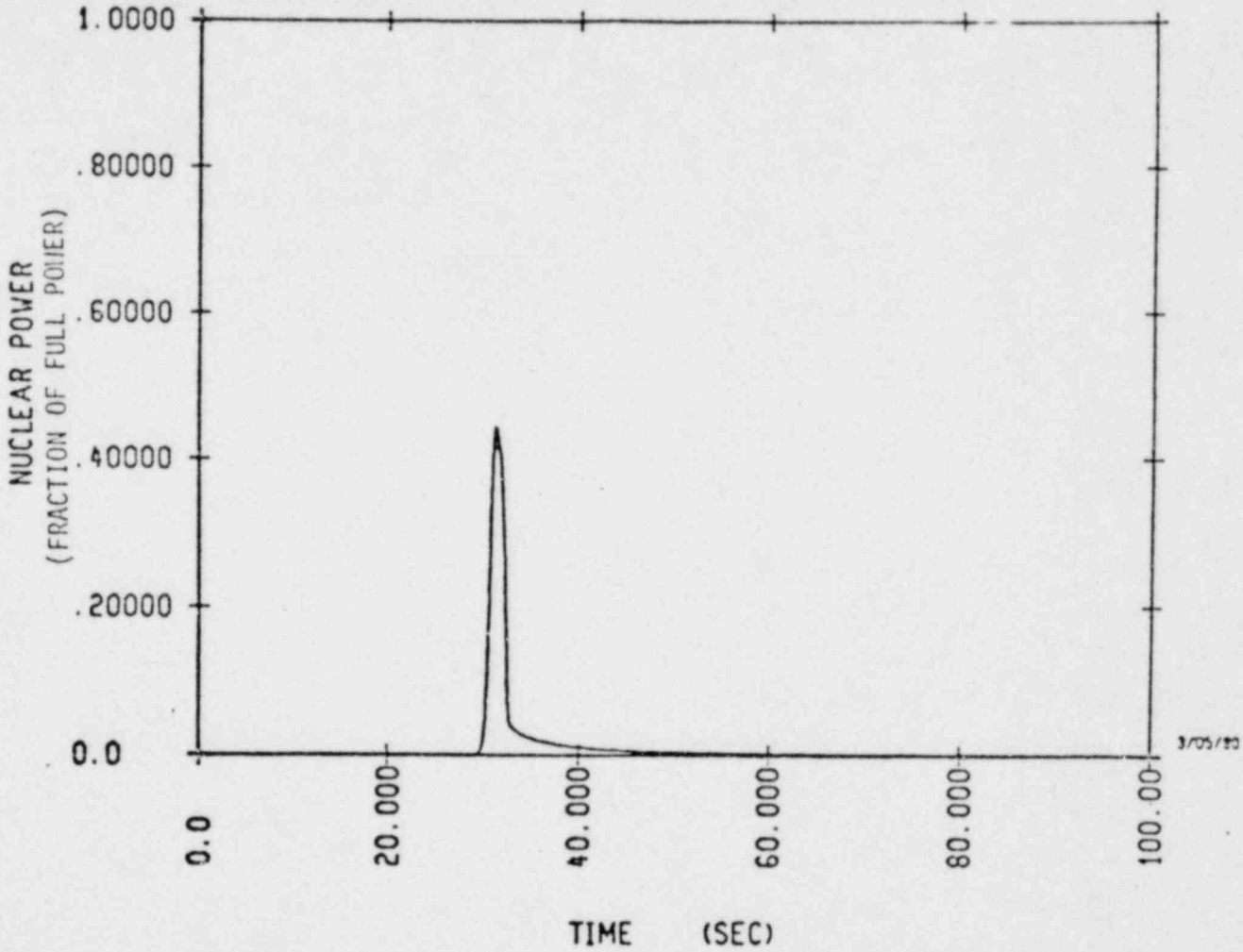

Figure 4.2.1 Uncontrolled Rod Bank Withdrawal from a Subcritical Condition, Neutron Flux vs Time

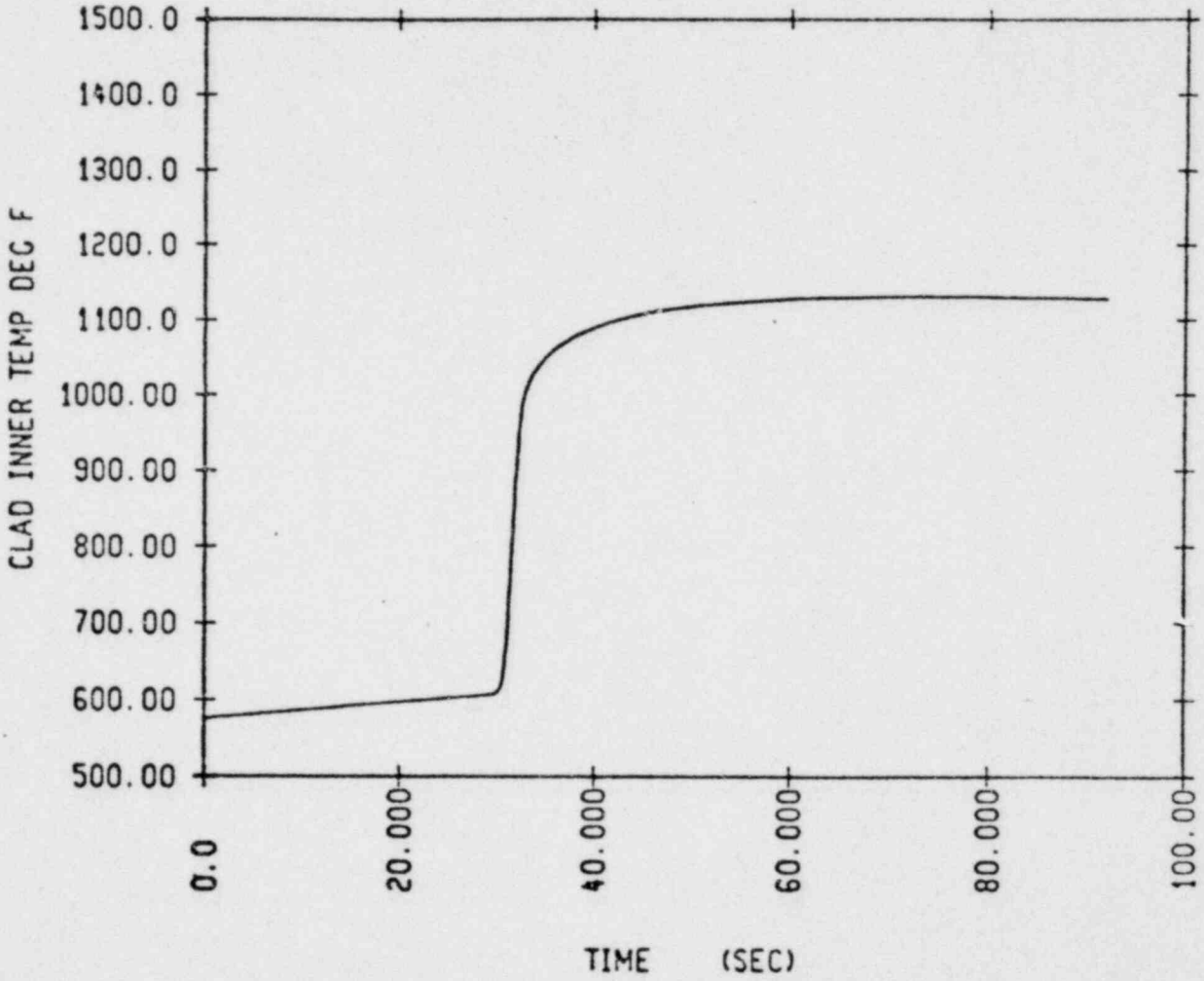

Figure 4.2.2 Uncontrolled Rod Bank Withdrawal from a Subcritical Condition, Hot Spot Clad Temperature vs Time, Assuming DNB at Time = 0.

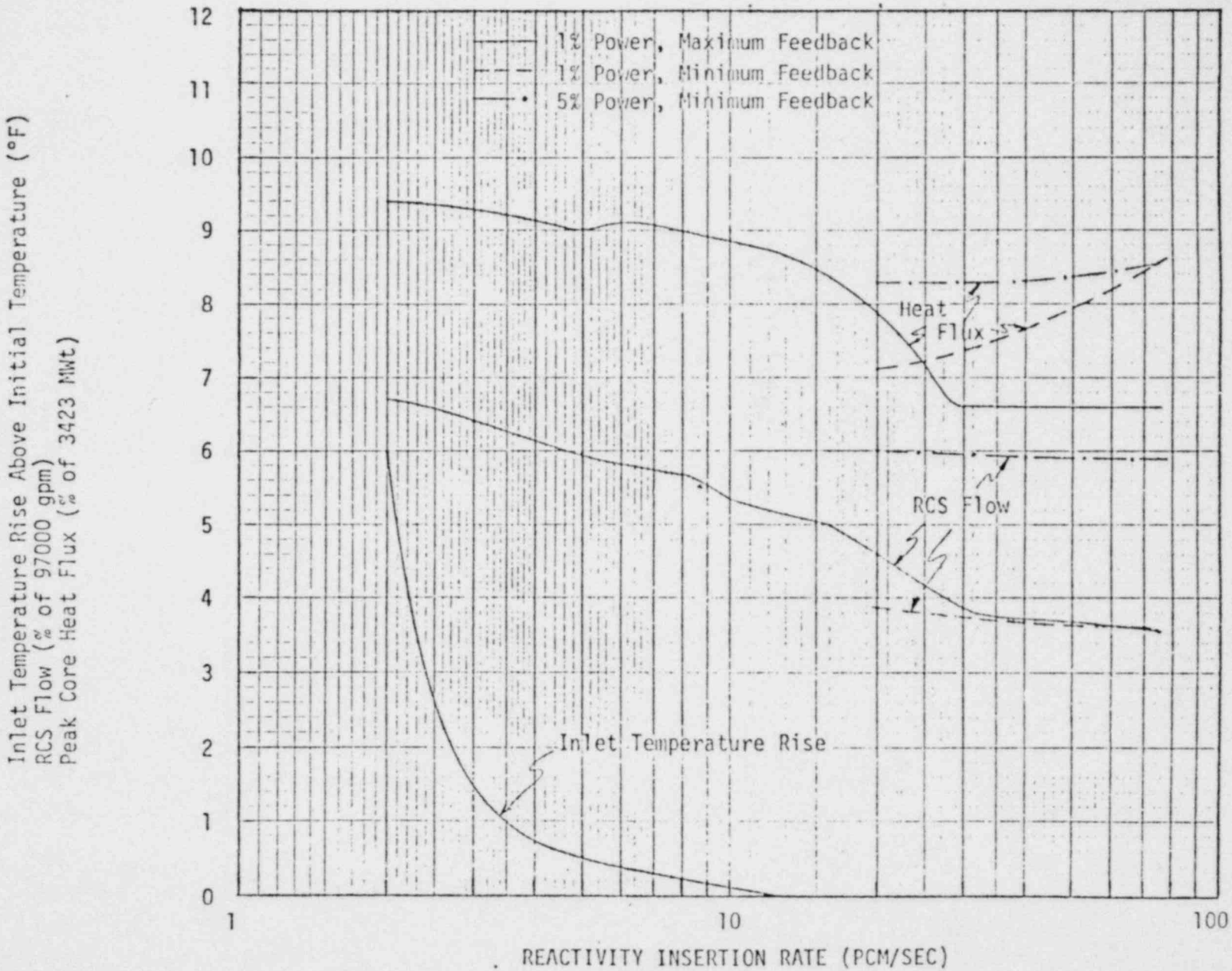

Figure 4-2.3 Uncontrolled Rod Bank Withdrawal at Power. Peak Heat Flux, Core Coolant Flow, and Increase in Core Inlet Temperature

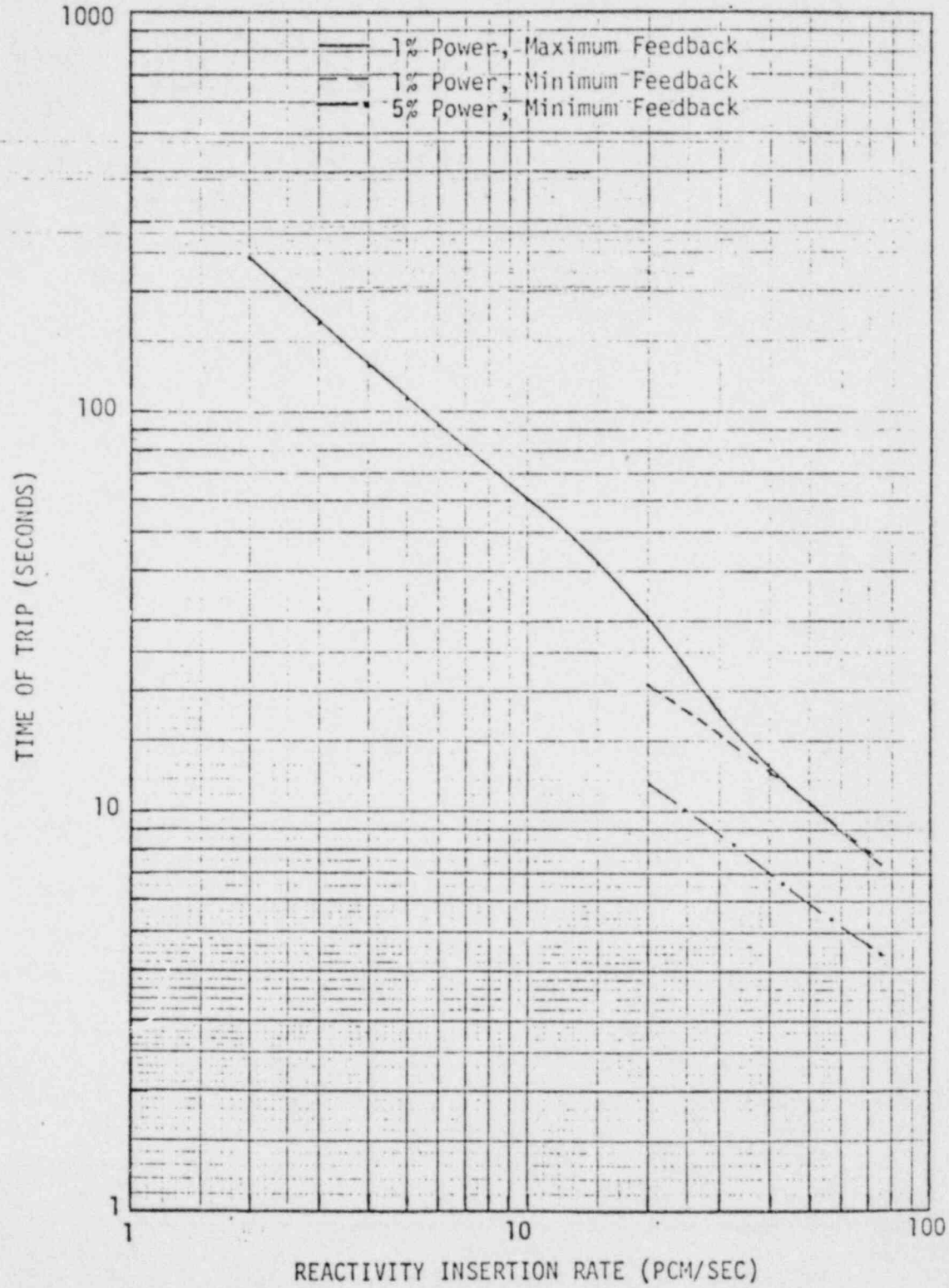

Figure 4.2.4 Uncontrolled Rod Bank Withdrawal at Power. Time of Reactor Trip vs Reactivity Insertion Rate

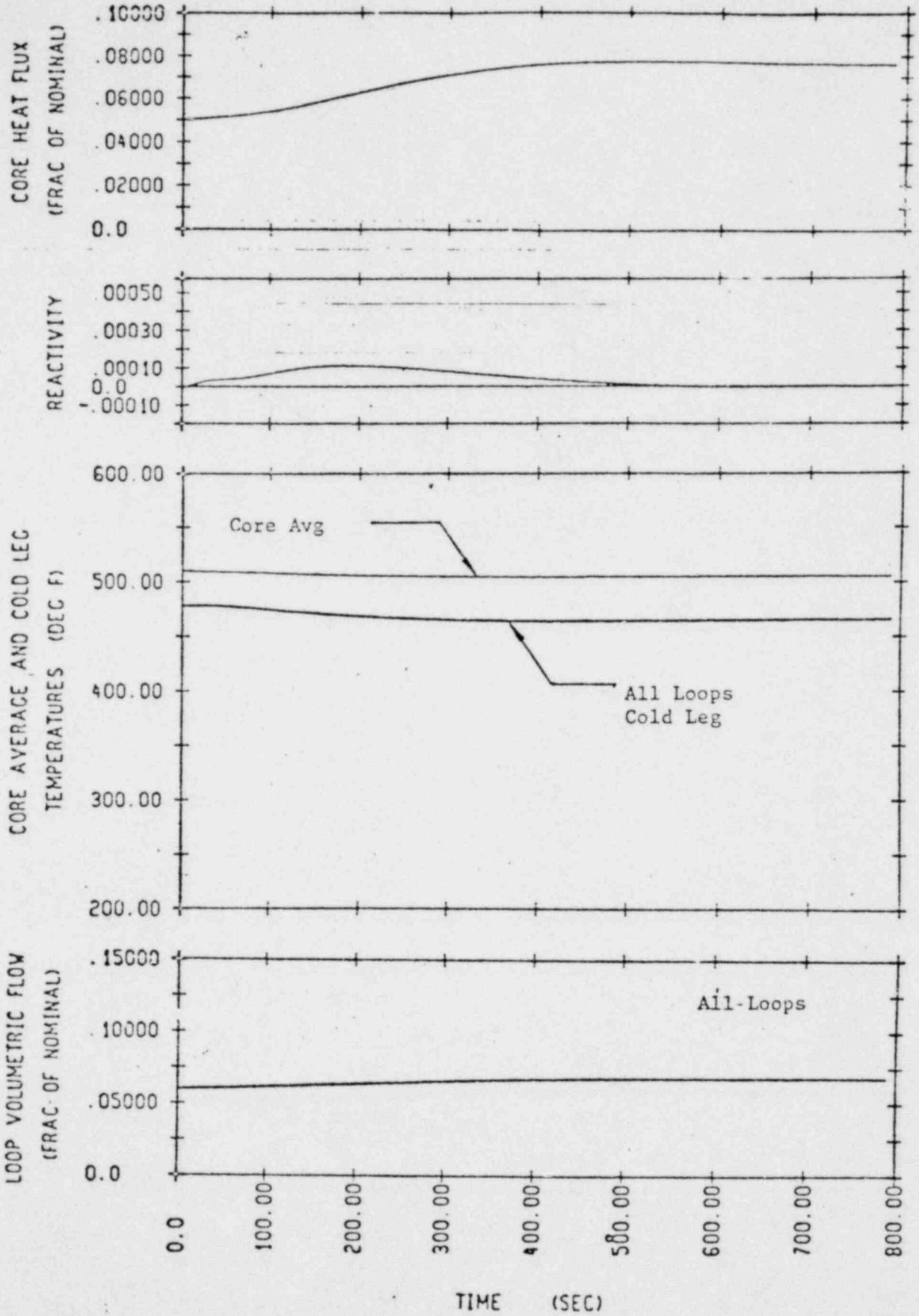

FIGURE 4.2.5 TRANSIENTS IN THE REACTOR CORE AND COOLANT LOOPS FOLLOWING THE OPENING OF A STEAM DUMP VALVE FROM 5% POWER, ALL LOOPS ACTIVE

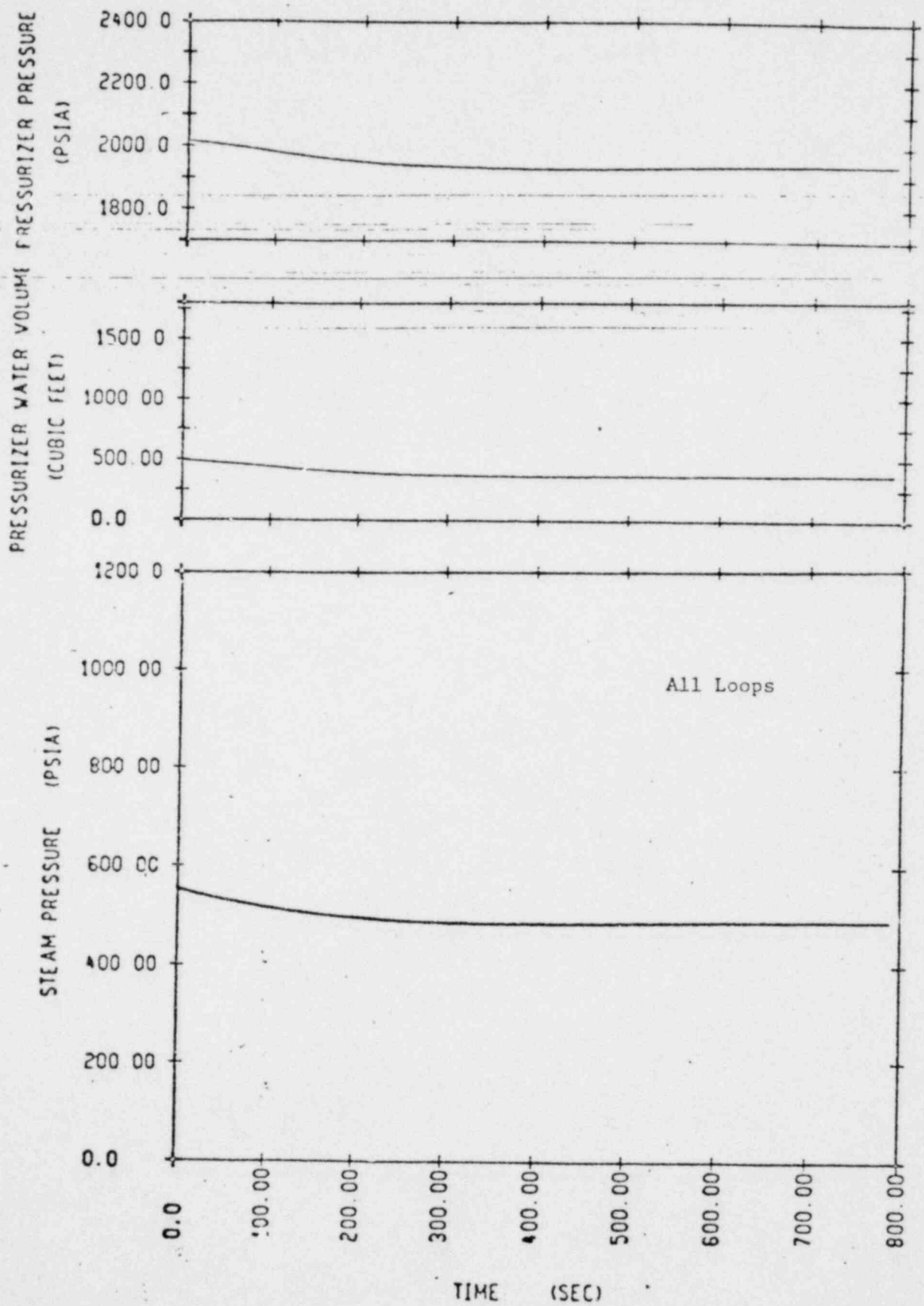

FIGURE 4.2.6 TRANSIENTS IN THE PRESSURIZER AND STEAM GENERATOR FOLLOWING THE OPENING OF A STEAM DUMP VALVE FROM 5% POWER, ALL LOOPS ACTIVE

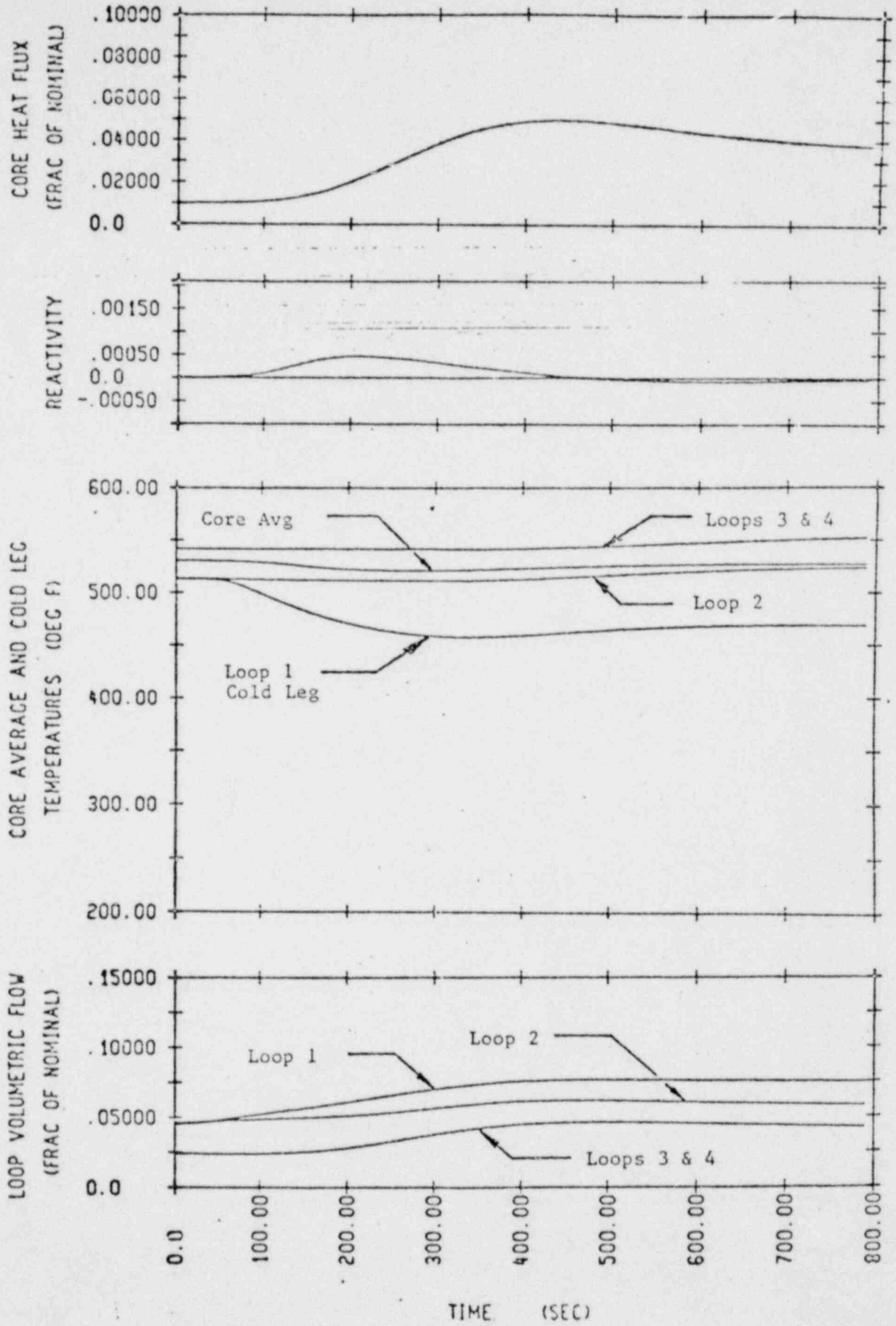

FIGURE 4.2.7 TRANSIENTS IN THE REACTOR CORE AND COOLANT LOOPS FOLLOWING THE OPENING OF A SINGLE STEAM RELIEF VALVE AT 1% POWER, TWO STEAM GENERATORS ISOLATED

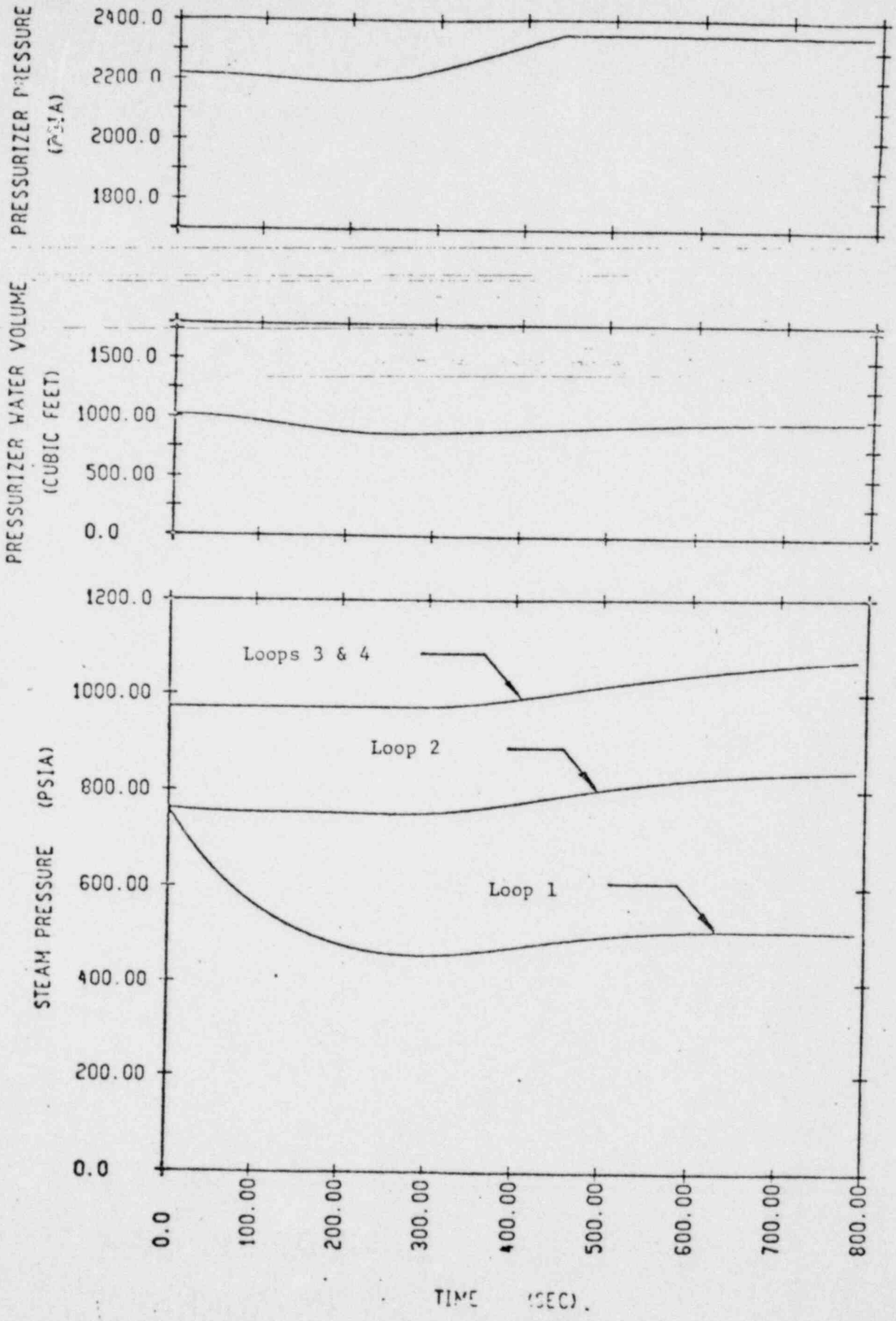

FIGURE 4.2.8 TRANSIENTS IN THE PRESSURIZER AND STEAM GENERATOR FOLLOWING THE OPENING OF A SINGLE STEAM RELIEF VALVE AT 1% POWER, TWO STEAM GENERATORS ISOLATED

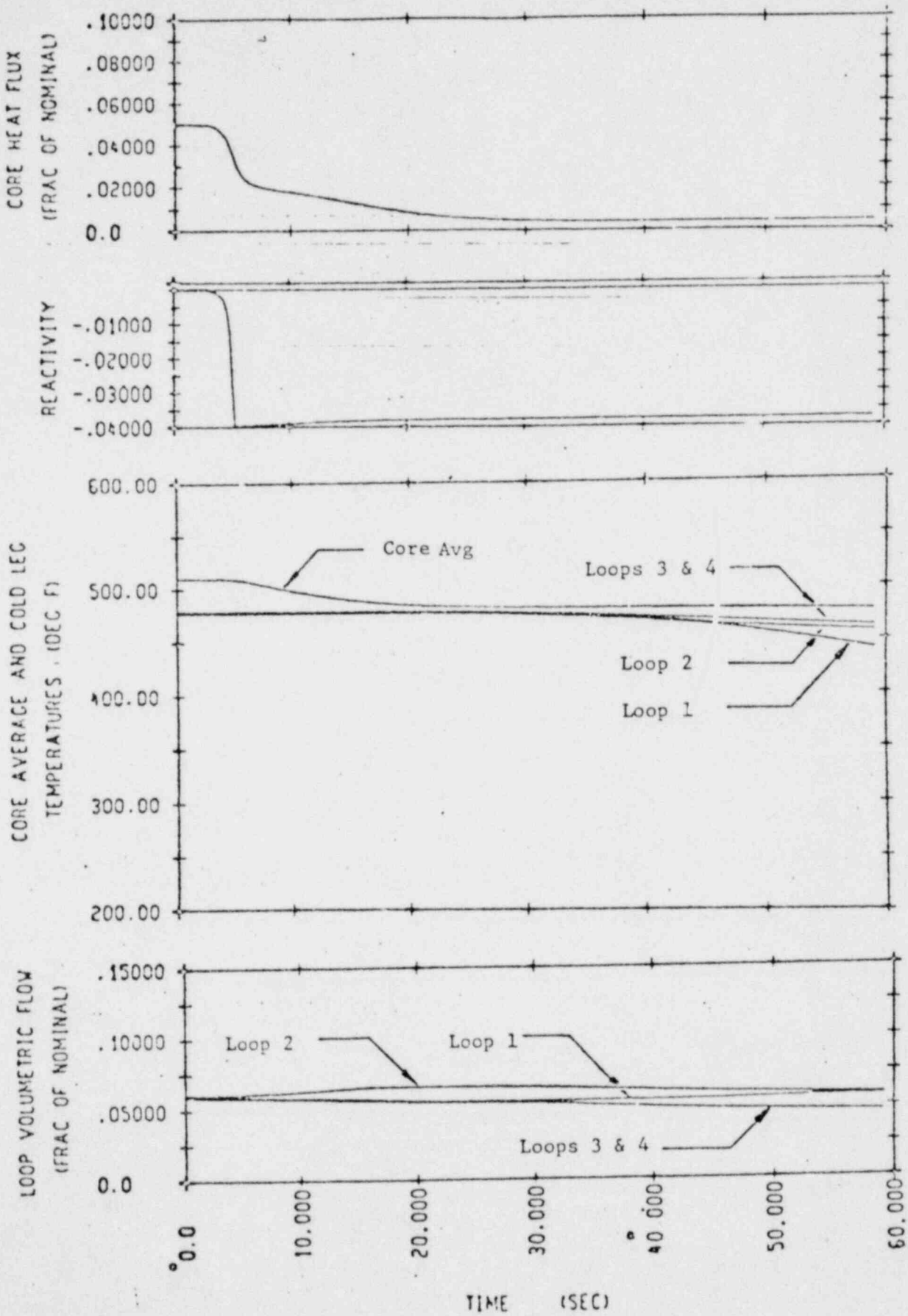

FIGURE 4.2.9 TRANSIENTS IN THE REACTOR CORE AND COOLANT LOOPS FOLLOWING A DOUBLE ENDED RUPTURE OF A MAIN STEAMLINE DOWNSTREAM OF THE STEAMLINE ISOLATION AND CHECK VALVES AT 5% POWER, ALL LOOPS ACTIVE

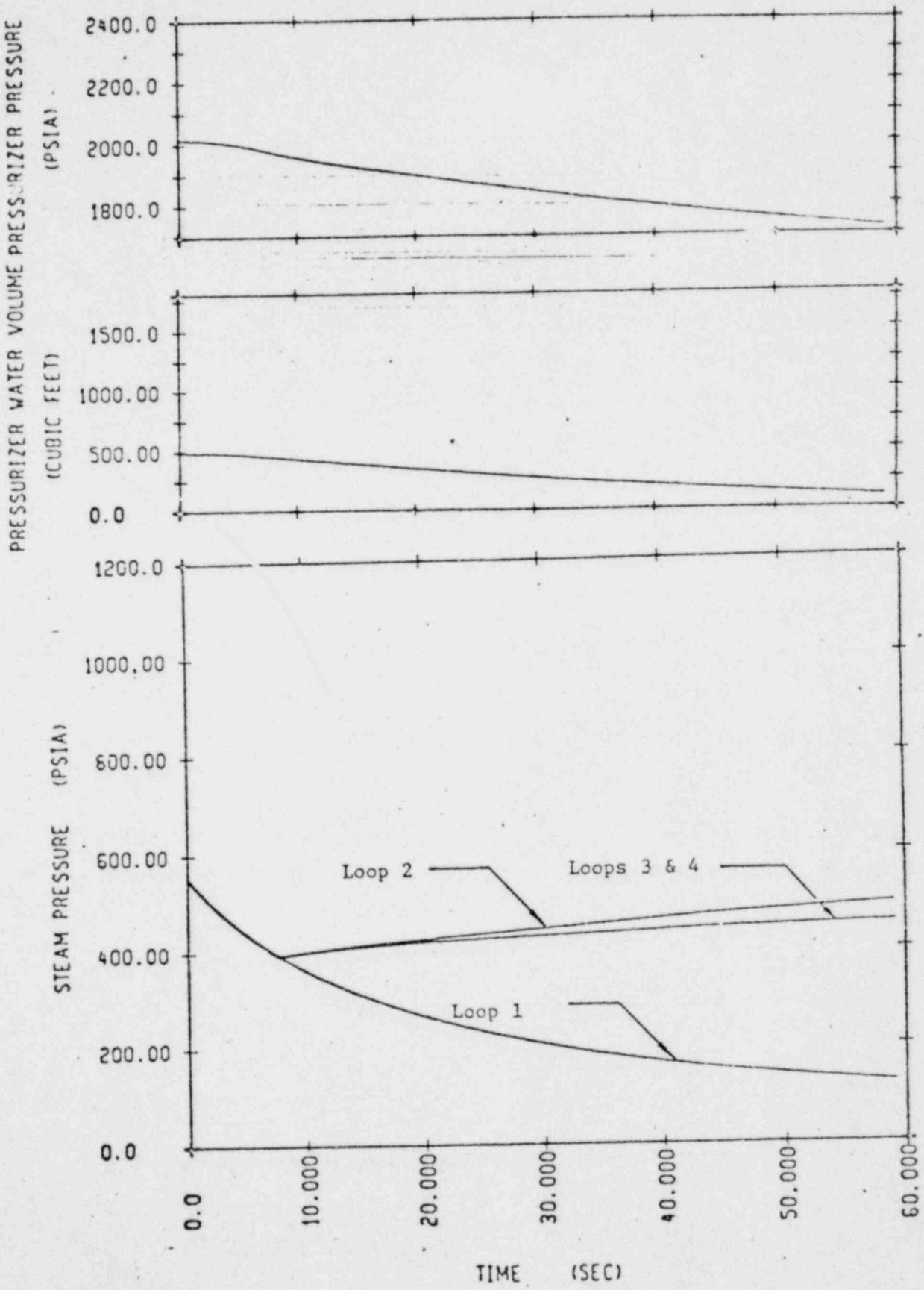

FIGURE 4.2.10 TRANSIENTS IN THE PRESSURIZER AND STEAM GENERATOR FOLLOWING A DOUBLE ENDED RUPTURE OF A MAIN STEAMLINE DOWNSTREAM OF THE STEAMLINE ISOLATION AND CHECK VALVES AT 5% POWER, ALL LOOPS ACTIVE

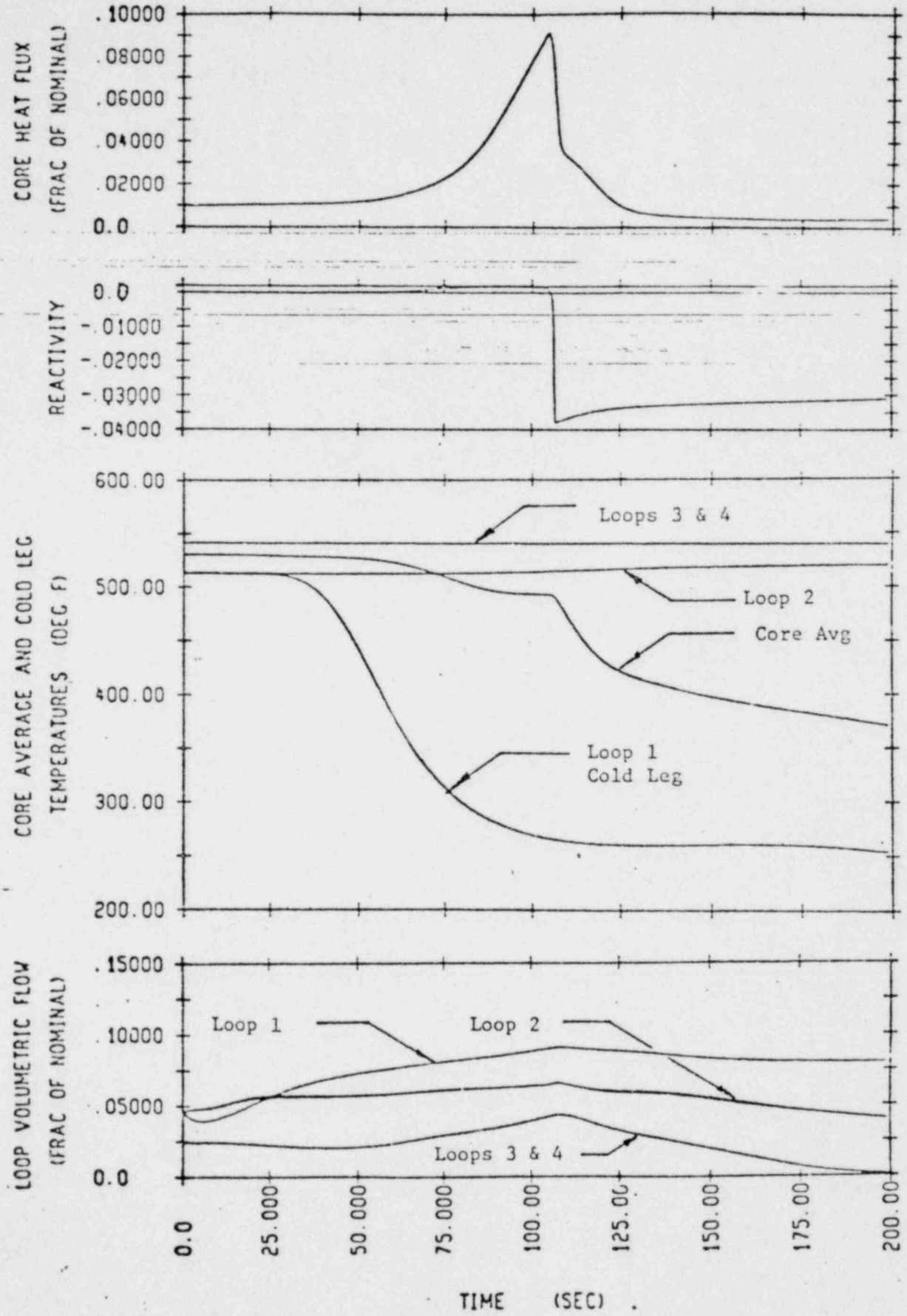

FIGURE 4.2.11 TRANSIENTS IN THE REACTOR CORE AND COOLANT LOOPS FOLLOWING A DOUBLE ENDED RUPTURE OF A MAIN STEAMLINE UPSTREAM OF THE STEAMLINE VENTURI AT 1% POWER, TWO LOOPS ISOLATED

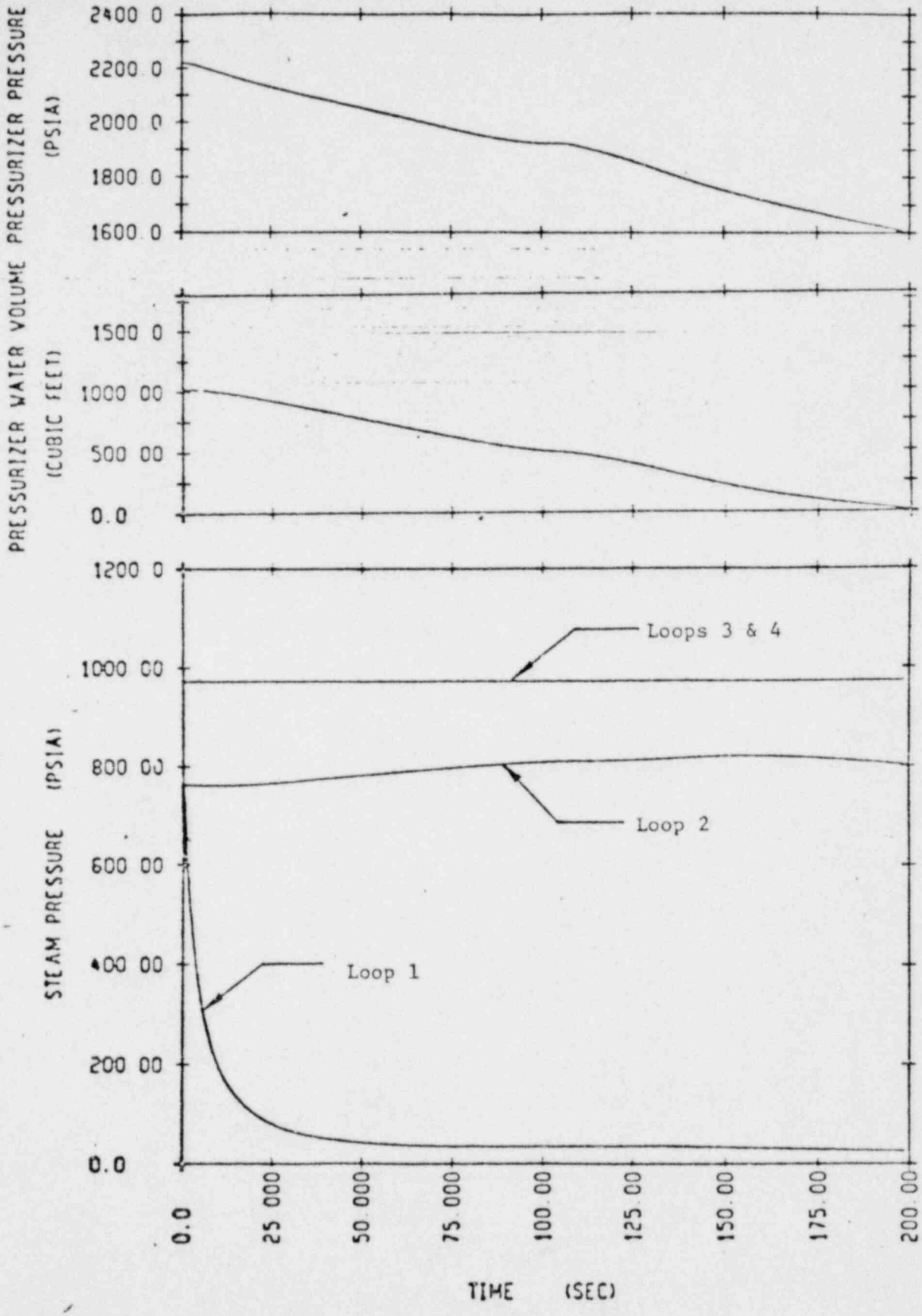

FIGURE 4.2.12 TRANSIENTS IN THE PRESSURIZER AND STEAM GENERATOR FOLLOWING A DOUBLE ENDED RUPTURE OF A MAIN STEAMLINE UPSTREAM OF THE STEAMLINE VENTURI AT 1% POWER, TWO LOOPS ISOLATED

Section 3

SPECIAL TEST PROGRAM SCHEDULE SU-7.1

TABLE 7

SEQUENCE OF STEPS FOR NATURAL CIRCULATION TESTING

NOTE: This sequence can be modified by the test director.

Special Test #	Test Name	Shift Performance	Expected Duration (hr)	General Comments
1	Natural Circulation	Group 1_____/_____ Group 2_____/_____ Group 3_____/_____ Group 4_____/_____ Group 5_____/_____ (Test Director)	4 4 4 4 4	Special Test #1 should be run once for each of the five groups The test director signs off each group upon completion of each test.
9A	Forced Circulation Cooldown	_____/_____	12 hrs.	Special Test #9A is a data gathering test. As such it will only be performed once.
9B	Boron Mixing And Cooldown	Group 1_____/_____ Group 2_____/_____ Group 3_____/_____ Group 4_____/_____ Group 5_____/_____	8 hrs. 8 hrs.	Special Test #9B should be run twice. All five operations groups must participate to the extent that each performs enough at a natural circulation cooldown to understand the problems envolved and techniques required for a controlled cooldown rate.

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TABLE 7

SEQUENCE OF STEPS FOR NATURAL CIRCULATION TESTING

NOTE: This sequence can be modified by the test director.

Special Test #	Test Name	Shift Performance	Expected Duration (hr)	General Comments
8	Establish NC from Stagnant Conditions	Group1_____/____ Group2_____/____ Group3_____/____ Group4_____/____ Group5_____/____ (Test Director)	8 8 8 8 8	Special Test 8 should be run twice. All five operations groups must participate in the stagnant start transient.
3	Natural Circulation With Loss of Pressurizer Heaters	Group1_____/____ Group2_____/____ Group3_____/____ Group4_____/____ Group5_____/____ (Test Director)	Both Test 8 8 8	Special Test #3 and #5 should be performed simultaniously. The test director may assign sub directors each responsible for a test. The test director is responsible for coordinati the test.
5	Natural Circulation at Reduced Pressure	Group1_____/____ Group2_____/____ Group3_____/____ Group4_____/____ Group5_____/____ (Test Director)	8 8	
4	Steam Generator Isolation in NC*	Group1_____/____ Group2_____/____ Group3_____/____ Group4_____/____ Group5_____/____ (Test Director)	 12 12	Special Test #4 should be run twice. All five Operations Groups must participate to the extent that each performs a steam generator isolation and reestablishment of natural circulation following the return of the S/G to service. *Possible SI

*Added Entire Page

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TABLE 7

SEQUENCE OF STEPS FOR NATURAL CIRCULATION TESTING

NOTE: This sequence can be modified by the test director.

Special Test #	Test Name	Shift Performance	Expected Duration (hr)	General Comments
2	Natural Circulation With Simulated Loss of Offsite AC Power	Group 1_____/_____ Group 2_____/_____ Group 3_____/_____ Group 4_____/_____ Group 5_____/_____ (Test Director)	12 12 8	Special Test #2 should be run three times & ST #7 should be run once. All five Operations Groups must participate to the extent that each observes the unit response upon test initiation and performs some portion of the stabization and recovery operation.
7	Simulated Loss of All Onsite and Offsite AC Power	Group 1_____/_____ Group 2_____/_____ Group 3_____/_____ Group 4_____/_____ Group 5_____/_____ (Test Director)	12	ST #7 NOTE: After each shift performs the required stabilization function in manual return the component to auto then back to manual. In this way each crew establishes manual control.
6	Cooldown Capability if the Charging and Letdown	Group 1_____/_____ Group 2_____/_____ Group 3_____/_____ Group 4_____/_____ Group 5_____/_____	4 4 4 4 4	Special Test #6 can be conducted at various times through but the special test schedule in which hot shutdown conditions exist. All five operating groups should complete the test prior to concluding special test schedule.

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Step No.	Sub Step No.	OPERATION	Test Proc No.	INITIAL CONDITIONS								CHANGE IN INITIAL CONDITIONS			COMMENTS	INITIAL/DAT
					RCC BANK POSITION											
					SD		CONTROL									
				% FP	A,B	C,D	A	B	C	D		ITEM	CHANGE	METHOD		
37	-	Connect One NIS Channel To Reactivity Computer	TI-25		0	0	0	0	0	0		N/A	N/A	N/A		
38	-	Verify criticality requirements met	SU-7.1												See table 6	
39	1	Achieve initial criticality, determine power range for zero power Physics tests	7.2 7.2				0	0	0	0		RCC Bank Position	Norm Seq. D to 160	RCC Control System	Plot ICRR data during approach to criticality	
	2		7.2		228	228	228	228	228	160		RCS Boron Concentration	Dilute to criticality	CVCS	Plot ICRR Data During approach to criticality	
	3	Start of formal zero power Physics tests: Reactivity Computer check.	7.2	HZP	228	228	228	228	228	160		Power	AR within zero pwr. test range	Bank D w/drawal	Minimum of 3 Reactor period measurements.	
39a		Measurement of Upper Head Temperature	8.5.7	HZP	228	228	228	228	228	160		N/A	N/A	N/A	This test should be run in parallel with HZP	
40		Plant Measurements Baseline and Operational Data	1.0	HZP	228	228	228	228	228	160		N/A	N/A	N/A	Radiation survey, effluent monitoring program, chemical-radiochemical analyses	

Step No.	Sub Step No.	OPERATION	Test Proc No.	INITIAL CONDITIONS							CHANGE IN INITIAL CONDITIONS			COMMENTS	INITIAL/DAT
					RCC BANK POSITION										
					SD		CONTROL								
				% FP	A,B	C,D	A	B	C	D	ITEM	CHANGE	METHOD		
41	1	Configuration, Adjustment and bank D differential and integral worth measurment	7.5	HZP						160	Bank D Position	To 200 steps	RCS Boron Addition		
	2	Boron endpoint measurement, and isotherm temp. coefficient	7.3.1	HZP						200	Bank D Position	To 228 steps	w/drawl D	Boron Endpoint	
			7.3.1	HZP						200	RCS Temp.	± 5°F	Steam dump	Temp. Coefficient	
	4	Incore M/D flux (all rods out configuration)	7.3.2	HZP						228				Map taken at HZP using O Power Flux Mapping Unit	
	5	Low power NIS calibration	8.5.3	HZP						228				To be performed at the first power escalation to 3-5%.	

Step No.	Sub Step No.	OPERATION	Test Proc No.	INITIAL CONDITIONS							CHANGE IN INITIAL CONDITIONS			COMMENTS	INITIAL/D
					RCC BANK POSITION SD CONTROL										
				% RTP	A,B	C,D	A	B	C	D	ITEM	CHANGE	METHOD		
41	6	Bank D differential and integral worth measurement	7.4	HZP						228	Bank D Position	To 0 Steps	RCS Boron Dilution	Compensate overshoot with Bank C insertion	
	7	Boron Endpoint measurement	7.3.1	HZP					0	20	Bank D Position	20 to 0 steps	Insert Bank D	Boron Endpoint	
			7.3.1	HZP					0	20	RCS Temp	±5°F	Steam Dump	Temp. coefficient	
	8	Measurement of bank worth by rod swap method	1.3	HZP					0	20	All Bank Position	0 Step	Exchange with Bank D	This is required for base line on the reduced startup program for Unit 2.	
	9	Incore M/D flux map (Bank D at 0, all other Banks out)	7.3.2	HZP					228	0				Map taken at HZP with 0 power Flux Mapping Unit	
	10														

Step No.	Sub Step No.	OPERATION	Test Proc No.	INITIAL CONDITIONS							CHANGE IN INITIAL CONDITIONS			COMMENTS	INITIAL/DA
					RCC BANK POSITION SD CONTROL										
				Q NFP	A,B	C,D	A	B	C	D	ITEM	CHANGE	METHOD		
41	11	Bank C differential and integral worth measurement	7.4	HZP					228	0	Bank C Position	To 85-$\frac{+0}{-5}$ Steps	RCS Boron Dilution	Bank B should be $\geq$ 210 steps	
	11A	Pseudo rod ejection at hot zero power	7.6	HZP					85	0	RCCA	To 228 Steps	RCS Boration	HZP pseudo rod ejection Rodworth and Boron End point flux map before and after	
	11B	Bank C differential and integral worth measurement	7.4	HZP					85	0	Bank C Position	To 0 Steps	RCS Boron Dilution	Compensate overshoot with Bank B insertion	
	12	Boron endpoint measurement isothermal temperature coefficient measurement	7.3.1	HZP				200	0	0	Bank B Position	200 to 228 to 200 Steps	w/draw B	Boron End point.	
			7.3.1	HZP				200	0	0	RCS TEMP	$\pm 5^{o}F$	Steam Dump	Temperature Coef.	
	13	Bank B differential and integral worth measurements	7.4	HZP				200	0	0	Bank B Position	To 0 Steps	RCS Boron Dilution	Compensate overshoot with Bank A insertion.	
	14	Boron endpoint measurement, isothermal temperature coefficient measurements	7.3.1	HZP			200	0	0	0	Bank A Position	200 to 228 to 200 Steps	w/draw A	Boron End point	

Step No.	Sub Step No.	OPERATION	Test Proc No.	INITIAL CONDITIONS							CHANGE IN INITIAL CONDITIONS			COMMENTS	INITIAL/DA
					RCC BANK POSITION SD CONTROL										
				G MFP	A,B	C,D	A	B	C	D	ITEM	CHANGE	METHOD		
			7.3.1	HZP			200	0	0	0	RCS Temp.	±5°F	Steam Dump	Temp. Coefficient	
41	15	Bank A differential and Integral Worth Measurements	7.4	HZP			200	0	0	0	Bank A Position	To 0 Steps	RCS Boron Dilution	Compensate Overshoot with S/D D Insertion	
	16	Boron Endpoint Measurement, Isothermal temp. Coefficient measurement	7.3.1	HZP		228 200	0	0	0	0	S/D D Position	200 to 228 to 200 Steps	W/draw S/D D	Boron Endpoint	
			7.3.1	HZP		228 200	0	0	0	0	RCS temp.	±5°F	Steam Dump	Temp. Coefficient	
	17	Minimum shutdown Verification	7.7	HZP			0	0	0	0	S/D D Position	AR	RCS Boron Dilution		
	17a	Boron Endpoint Measurement	7.3.1	HZP		228 0	0	0	0	0	S/D D Position	To 0 Steps	Insert S/D D	Boron Endpoint	
	17b	Boron Endpoint Measurement	7.3.1	HZP		0	0	0	0	0	S/D C Position	To 0 Steps	Insert S/D C	Boron Endpoint	
	18	RCC Control Bank Differential and Integral worth Measurements during normal bank w/drawal sequence	7.5	HZP	228	228	0	0	0	0	Control Banks Position	A,B,C to 228 steps D to 100 steps	RCS Boron Addition	Normal Overlap	

Step No.	Sub Step No.	OPERATION	Test Proc No.	INITIAL CONDITIONS							CHANGE IN INITIAL CONDITIONS			COMMENTS	INITIAL/DATE
					RCC BANK POSITIONS										
					SD		CONTROL								
				Q %FP	A,B	C,D	A	B	C	D	ITEM	CHANGE	METHOD		
	19	Natural Circulation Special Testing	1 2 3 4 5 6 9A 7 9B 8											See Table 7	
2		Verify requirements Prior to proceding to 30% Power Plateau	7.1				228	228	228	AR				See Table 6	
3a		Calibration of Steam Feedwater flow Instrumentation	W-11.7 (A)	HZP						AR	N/A	N/A	N/A	Obtain Zero Power Data	
3b		Thermal expansion	TWA-23B<1.0%									AR			

Section 4

SPECIAL LOW POWER TEST PROGRAM ADMINISTRATIVE
PROCEDURE STANDARD PRACTICE SQA-109

SPECIAL LOW-POWER TEST PROGRAM

1.0 Purpose

The purpose of this standard practice is to establish the special low-power test program, which is to be conducted in conjunction with the low-power physics test program. It describes the program, defines the responsibilities for administration and for test preparation, review and approval, and describes restraints which should be used.

2.0 Objectives

The special tests listed in Attachment 1 will be conducted prior to exceeding a core power level greater than five percent of rated core thermal power. These tests are intended to provide a demonstration of reactor operation in the natural circulation mode under both normal and certain degraded conditions.

These tests will provide significant operator experience and training with the plant in the natural circulation mode. Certain tests will be repeated such that all licensed personnel will participate in the establishment and recognition of natural circulation conditions. The simulator may be used to provide portions of this participation after it has been compared to actual plant response. Results from these tests will be used to verify the simulator models and training techniques used on the Sequoyah simulator.

3.0 Responsibility and Organization

The Division of Nuclear Power (NUC PR) is responsible for the start-up test program. It has the responsibility of coordinating all activities of licensed units including onsite liaison with regulatory organizations and the safe and efficient conduct of operations.

3.1 Plant

3.1.1 Plant Superintendent

The plant superintendent is responsible for the final approval of the test instructions; for revisions thereto, and for the approval and issuance of supplementary instructions as necessary; planning, scheduling, and management of tests for safe plant operations; and the satisfactory completion of all tests, collection of data and field evaluation.

3.0 Responsibility and Organization (Cont.)

3.1.2 Assistant Plant Superintendent

The assistant plant superintendent is responsible to the superintendent for day-to-day operating units and in the superintendent's absence is responsible for assuring that all tests and operations are in compliance with the technical specifications.

3.1.3 Operations Supervisor

The operations supervisor is responsible for the safe operation of plant equipment in accordance with the operating license, technical specifications and approved instructions. He issues the necessary daily instructions to the shift engineers for operating plant equipment during the startup program. The operations supervisor will maintain records documenting licensed operator participation in all special tests or special test demonstrations on the Sequoyah simulator.

3.1.4 Shift Engineers

The shift engineer is in direct control of plant operations. He is assisted by assistant shift engineers, unit operators, assistant unit operators, and other plant support personnel as required. In addition, the shift engineer has control over the actions of other personnel while they are involved with plant systems or components. He has the responsibility for instituting immediate action in any given situation to eliminate difficulties or remove equipment from service to preclude violation of the operation licenses, technical specifications or to avert possible injury to personnel or equipment damage.

3.1.5 Power Plant Results Supervisor

The results supervisor is responsible for providing adequate engineers for test coverage and for the collection of tests data. While testing is in progress, a test engineer will be assigned to each shift as a technical advisor and test coordinator. This test engineer can stop a test when in his opinion, conditions warrant such actions. The results supervisor is also responsible for detailed test scheduling and for field review of test data prior to it being sent to the Plant Operations Review Committee.

3.0 Responsibility and Organization (Cont.)

3.1.6 Special Test Group Coordinator

The Special Test Group will consist of: a coordinator from the Reactor Engineering Branch and technical representatives from the Operations Section, the Results Section, and the Preoperational Test Section of the Sequoyah Nuclear Plant and the Nuclear Engineering Branch, and the Division of Engineering Design.

The Special Test Group Coordinator is responsible for the draft and final special startup test instructions. He is responsible for the coordination of test instruction review with the Division of Engineering Design, the Westinghouse Electric Corporation, and the Nuclear Regulatory Commission.

The Special Test Group will identify any necessary changes to the plant technical specifications any required deviations from the Precautions, Limitations, and Setpoints documents.

The Special Test Group will provide technical assistance for the conduct of all special tests, and will perform the field analysts of all special tests data. The group will coordinate the review of all special test data by the Division of Engineering Design, Westinghouse Electric Corporation, and the Nuclear Regulatory Commission.

The Special Test Group will coordinate any other activities relating to the test program with organizations outside the plant. This coordination will include the participation of and the dissemination of results to such organizations.

The Special Test coordinator is functionally supervised by the plant superintendent and all group activities are subject to the plant superintendent's review and approval.

3.1.7 Maintenance Supervisor

The maintenance supervisor is responsible for equipment maintenance during the startup program. He reviews startup tests scheduled to assure that equipment scheduled for test or required in support of testing is in a state of readiness.

3.0 Responsibility and Organization (Cont.)

3.2 Division of Nuclear Power

3.2.1 Director, Division of Nuclear Power

The Director, Division of Nuclear Power, is responsible for coordinating all division activities affecting operation, maintenance, and engineering in nuclear plants. He shall provide overall administrative guidance in the conduct of the startup test program. The Assistant Director (Operations) is directly responsible for all operations activities and the Assistant Director (Engineering and Maintenance) is directly responsible for engineering and maintenance activities.

3.2.2 Reactor Engineering Branch

The Reactor Engineering Branch shall be responsible for providing a safety evaluation per N-OQAM Part II, Section 4.6 for each test in the special low-power test program. As required by N-OQAM Part II, Section 4.6, this evaluation will include an unreviewed safety question determination (10 CFR 50.59). The branch shall obtain assistance as required from any other organizations in performing this safety evaluation.

The Reactor Engineering Branch shall also provide central office observers on each day shift during the special test program.

The duties and responsibilities of these observers are as follows:

Be cognizant of the scope and intent of the special test program.

Be cognizant of the test being conducted.

Be familiar with the operation of a PWR-type reactor and

Provide daily status reports to the Assistant Director of Nuclear Power (Operations).

The Reactor Engineering Branch shall provide all tests results from the special startup test program to the Nuclear Training Center for evaluation for simulator models and training techniques used on the Sequoyah simulator. This data will be transmitted in a timely fashion such that once verified the simulator can be used for operator participation and training in various conditions of natural circulation.

3.0 Responsibility and Organization (Cont.)

3.3 Division of Engineering Design

The Division of Engineering Design shall review all special startup test instructions and results. System design engineers shall be provided as required during startup testing. This division shall review and approve any proposed plant modifications and replacements which involve design changes.

The Division of Engineering Design is also responsible for evaluating the special test program for any potential violation of equipment warranty conditions. They will advise the plant superintendent (through the Special Test Group) of any such potential equipment warranty violations.

3.4 Nuclear Fuels Branch

The Nuclear Fuels Branch is responsible for evaluating the special test program for any potential violations of fuel warranty conditions. They will advise (through the Special Test Group) the plant superintendent of any such potential warranty violations.

4.0 Preparation Review and Approval of Special Startup Test Instructions

4 1 Preparation

Draft special test instructions are prepared by the Special Test Group. These draft procedures are reviewed by the Division of Engineering Design and the Westinghouse Electric Corporation. Following the receipt of informal comments by these organizations, the draft test procedures will be sent to the Nuclear Regulatory Commission for comments.

Final startup test instructions will be prepared by the Special Test Group using the draft procedures and appropriate comments as a guide.

4.2 Test Instruction Content

The special test instructions shall include the following:

4.2.1 Test Description

The test description includes what the test consists of and how it is to be conducted. The purpose of the section is to serve as a guide reference for planning and scheduling.

4.0 Preparation Review and Approval of Special Startup Test Instructions (Cont.)

4.2.2 Objectives

The objectives include all purposes for conducting the test.

4.2.3 Prerequisites

All prerequisites which must be completed before a test is started must be listed in this section. Prerequisites also include any exceptions to the Standard Technical Specifications or the Precautions Limitations and Setpoints documents as issued for the Sequoyah Nuclear Plant. These prerequisites will also include a master checklist.

4.2.4 Precautions, Limitations, and Actions

List special precautions and limitations necessary for the protection of personnel and equipment and actions to be taken in case of anomalies. These actions shall include system parameter limits upon which manual operator action such as reactor trip or safety injection is initiated.

4.2.5 Special Test Equipment

List all special test equipment that is not normally installed.

4.2.6 Instructions

The instructions include the detail steps necessary to accomplish the test.

The power level at which the test is to be performed and a method for determining the power level throughout the test shall be included.

The special test instruction shall also contain the expected primary and secondary response to various test conditions and actions to be followed in conducting the tests.

Operational steps in the instructions necessary to establish a test condition should contain a blank space for signoff of completion of the instruction. Steps at which data is to be collected should contain no blank, but should be stated at the appropriate step in the instruction. When tests are to be conducted in segments or at different times, this should be clearly indicated in the detailed instructions and data sheets.

4.0 Preparation Review and Approval of Special Startup Test Instructions (Cont.)

4.2.6 Instructions (Cont.)

Instructions for returning the system to normal conditions should be included.

4.2.7 Data Sheets

Data sheets and the methods for data analysts should be included after the detailed instructions. These data sheets shall include signature spaces for the person taking the data and a reviewer.

4.2.8 Acceptance Criteria

Specific acceptance criteria for each segment of the test shall be listed.

4.2.9 Appendix A - References

All references used in preparing the instruction should be listed.

4.2.10 Appendix B - Test Deficiencies

Any test deficiencies encountered during conduct of the test should be recorded. Each deficiency shall be assigned a unique number, and each deficiency shall be recorded on a separate test deficiency sheet Appendix of each special startup test instruction.

4.3 Review and Approval

Draft special test instructions are reviewed by the Division of Engineering Design and the Westinghouse Electric Corporation. Following the receipt of informal comments by these organizations, the draft test procedures will be sent to the Nuclear Regulatory Commission for comments.

Final startup test instructions will be prepared by the special test group using the draft procedures as a guide. These final test instructions will be formally transmitted to the Division of Engineering Design and the Westinghouse Electric Corporation for their review. Transmittal letters for these reviews are in Attachments 2 and 3 respectively. If requested, the final test instructions will be sent to the Nuclear Regulatory Commission for their review. The test instructions will be available for onsite review by the Nuclear Regulatory Commission.

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4.0 Preparation Review and Approval of Special Startup Test Instructions (Cont.)

4.3 Review and Approval (Cont.)

The final test instructions shall be transmitted to the Director, Division of Nuclear Power for review and approval per N-OQAM Part II, Section 4.6. This review shall include an unreviewed safety question determination (10 CFR 50.59). The test instructions shall be transmitted with the cover sheet in Attachment 10.

The Plant Operations Review Committee (PORC) shall review the startup test instructions and recommend approval when it is satisfied that the test can be conducted safely without jeopardizing personnel or equipment. The plant superintendent shall approve the startup test instructions prior to their use.

5.0 Revisions to Startup Test Instructions

All startup test revisions will be handled in accordance with AI-4, Section 4.0.

6.0 Approval to Perform a Startup Test

The plant superintendent must authorize the performance of a startup test. This authorization will be indicated on the cover sheet of the "Work Copy". (See Attachment 9).

7.0 Conduct of the Startup Test Program

7.1 Plant operations shall be conducted by qualified supervisors and operators, and when required, they shall have the senior operator and operator licenses.

7.2 A qualified engineer shall be assigned to the control room to provide technical assistance during all special tests.

7.3 Westinghouse shall provide onsite assistance for the conduct of the Special Test Program.

7.4 Test data shall be maintained in accordance with procedures established in SQA44, Section 7.4.

7.5 A narrative log book shall be maintained by the test engineers as discussed in SQA44, Section 7.5.

8.0 Restraints During Conduct of the Special Startup Program

8.1 Approval for all plant technical specification changes required for the conduct of these special tests must be received from NRC.

8.0 Restraints During Conduct of the Special Startup Program (Cont.)

8.2 All deviations from the Precautions Limitations and Setpoints document must be identified and safety evaluations provided prior to these deviations being performed.

8.3 At any time that operations involve undue risk to personnel or equipment or, if continued, would violate the technical specification, operations will be suspended or modified until an evaluation is made by the PORC.

8.4 The plant superintendent shall ensure that strict adherence to the technical specifications is maintained at all times during the startup test program.

9.0 Recording and Correction of Deficiencies Discovered During the Special Startup Program

Deficiencies are recorded and handled as dicussed in SQA44, Section 9.0.

10.0 Special Test Results, Evaluations, and Approvals

10.1 The special test program must be completed and all special tests must be approved prior to increasing core power above 5 percent of rated core thermal power. The test instructions shall identify those acceptance criteria which must be met prior to proceeding above 5-percent power.

10.2 Upon completion of each test, a test evaluation report shall be prepared which addresses the acceptance criteria and all test deficiencies. The test evaluation report shall include a recommendation for the resolution of the deficiencies and disposition of the test. This report shall be prepared by the Special Test Group.

10.3 The test evaluation report along with the test instruction, will compose a complete test package. Attachment 4 contains the test package format and Attachment 5 is a test package cover sheet.

10.4 The test package will be reviewed by the Special Test Coordinator, the Power Plant Results Supervisor and the Plant Operations Review Committee. The test report shall be sent to the plant superintendent for his approval.

10.5 The special test packages shall also be reviewed by the Division of Engineering Design, the Westinghouse Electric Corporation, and the Nuclear Regulatory Commission. The Special Test Group will coordinate this review. Transmittal letters are included as Attachments 6, 7, and 8 respectively.

12.0 Special Startup Test Report

12.1 A summary report of the special test program shall be provided. This report shall include a description of the measured values for each test and a comparison of these values with predicted values. Any corrective actions that were required to obtain satisfactory operation shall also be described.

12.2 The summary report will also give comparisons of special test data and data obtained from the Sequoyah simulator. The report will also discuss any necessary changes in the Sequoyah simulator.

12.3 The report will discuss operator participation in each test and subsequent demonstrations of each test on the Sequoyah simulator.

Power Plant Superintendent

General Revision

Attachment 1

SPECIAL TEST INSTRUCTIONS

ST-1 Natural Circulation Verification

ST-2 Natural Circulation with Simulated Loss of Offsite Power

ST-3 Natural Circulation with Loss of Pressurizer Heaters

ST-4 The Effect of Steam Generator Isolation (Secondary Side) on Natural Circulation

ST-5 Natural Circulation at Reduced Pressure

ST-6 Determination of Cooldown Capability of the Charging and Letdown System

ST-7 Simulated Loss of All Onsite and Offsite AC Power

ST-8 Establishment of Natural Circulation from Stagnant Conditions

ST-9 Boron Mixing and Cooldown Under Natural Circulation Conditions

Attachment 2

George F. Dillworth, Chief, Nuclear Engineering Branch, W10C126 C-K

J. G. Dewease, Assistant Director of Nuclear Power (Operations), 727 EB-C

SEQUOYAH NUCLEAR PLANT - SPECIAL TEST PROGRAM - TEST INSTRUCTION APPROVAL

Th special test instructions listed below are enclosed for your review and approval. Your approval of these instructions is required prior to submission of the instructions to the Plant Operating Review Committee and subsequent final approval by the power plant superintendent.

Special Test Instruction

Attachments: 3
cc: ARMS PP, 823 EB-C
J. M. Ballentine, Sequoyah Nuclear Plant
Special Test Instruction(s)

Attachment 3

Sequoyah Nuclear Plant
P. O. Box 2000
Daisy, Tennessee 37319

Mike Siano
Westinghouse Electric Corporation
P. O. Box
Pittsburgh, Pennsylvania

Dear Mr. Siano:

SEQUOYAH NUCLEAR PLANT - SPECIAL TEST PROGRAM - REVIEW OF TEST INSTRUCTIONS

The special test instructions listed below are enclosed for your review. We would appreciate your review of these test instructions with particular emphasis on scope of testing, acceptance criteria, instrumentation, procedure and data evaluation.

Special Test Instruction

Accordingly, your expeditious review of this material will be appreciated.

Very truly yours,

J. M. Ballentine
Power Plant Superintendent

Enclosures: 2
cc: ARMS PP, 823 EB-C
J. G. Dewease, 727 EB-C

Attachment 4

TEST PACKAGE FORMAT

The Test Evaluation Report, the Startup Test procedure and all related data will makeup the complete Test Package. The Test Package should adhere to the following format:

TEST PACKAGE COVER SHEET

This cover sheet should be similar to that in Figure 1.

ABSTRACT

The test abstract shall give a brief description of the test, the test results and indicate if the test was successful. Any major deficiencies shall also be briefly mentioned.

TEST EVALUATION REPORT

I. Acceptance Criteria and Test Results

This section should contain a discussion of the test results in regard to the acceptance criteria. Any other pertinent test results should be presented in this section.

II. Test Deficiencies and Resolution of Deficiencies

This section should address all test deficiencies and include recommendations for the resolution of the deficiencies.

III. Disposition of the Test

This section should contain recommendations on whether or not the test is acceptable as is or if further consideration of the test results is warranted.

STARTUP TEST AND RELATED DATA

This portion should include the startup test procedure, including temporary changes, and all required data from TI's, etc.

The complete Test Package will go to the Results Supervisor for review, to the PORC for review, and then to the plant superintendent for approval.

Attachment 5

SEQUOYAH NUCLEAR PLANT

SPECIAL STARTUP TEST PACKAGE

Unit _____

ST _____

NAME: _____

SU-7.1 Sequence Number _____ Reactor Power _____

Test Package Preparer _____ / _____
Date

Special Test Coordinator Review _____ / _____
Date

Results Supervisor Review _____ / _____
Date

PORC Review _____
Date

Plant Superintendent Approval _____ / _____
Date

Attachment 6

George F. Dillworth, Chief, Nuclear Engineering Branch, W10C126 C-K

J. G. Dewease, Assistant Director of Nuclear Power (Operations), 727 EB-C

SEQUOYAH NUCLEAR PLANT - SPECIAL STARTUP TEST PROGRAM - REVIEW OF TEST RESULTS

The special test data packages listed below are provided for your review. Please provide your comments and accuracy of data reduction and the conclusions reached for each of these tests.

Special Test Data Package

Please provide us with the results of your review of this material at the earliest possible date.

Attachment 7

Sequoyah Nuclear Plant
P. O. Box 2000
Daisy, Tennessee 37319

Mike Siano
Westinghouse Electric Corporation
P. O. Box
Pittsburgh, Pennsylvania

Dear Mr. Siano:

SEQUOYAH NUCLEAR PLANT - SPECIAL TEST PROGRAM - REVIEW OF TEST RESULTS

The special test data packages listed below are provided for your review. Please provide your comments and accuracy of data reduction and the conclusions reached for each of these tests.

Special Test Data Package

Please provide us with the results of your review of this material at the earliest possible date.

Very truly yours,

J. M. Ballentine
Power Plant Superintendent

Attachment 8

L. M. Mills, Manager, Nuclear Regulation and Safety, 400 CST2-C

J. G. Dewease, Assistant Director of Nuclear Power (Operations), 727 EB-C

SEQUOYAH NUCLEAR PLANT - SPECIAL STARTUP TEST PROGRAM - NRC REVIEW OF TEST RESULTS

Please forward the following special test data packages to the NRC for their review.

Special Test Data Package

Any comments from the NRC on this material should be forwarded to me at the earliest possible date.

Attachment 9

I- _____

Title _____

Prepared By _____ Date _____

Reviewed By:

Plant Operations Review Committee _____
Date

Test Approval

_____ _____
Plant Superintendent Date

Authorization To Perform Test

_____ _____
Plant Superintendent Date

Certification Of Test Completion And Results

This test has been conducted in accordance with this procedure, and the systems and equipment have met the requirements contained herein. Exceptions, if any, are listed in the appendix sheet.

_____ _____
Special Test Coordinator Date

_____ _____
Power Plant Results Supervisor Date

Plant Operations Review Committee _____
Date

_____ _____
Plant Superintendent Date

_____ Plant Master File
_____ Superintendent
_____ Assistant Superintendent
_____ Maintenance Supervisor
_____ Results Supervisor
_____ Operations Supervisor
_____ Quality Assurance Supervisor
_____ Health Physicist
_____ Administrative Supervisor
_____ Chemical Laboratory
_____ Instrument Shop
_____ Shift Engineer's Office
_____ Unit Control Room
_____ Health Physics Laboratory
_____ Chief Storekeeper
_____ Mechanical Engineer
_____ Reactor Engineer
_____ Chemical Engineer
_____ Instrument Engineer
_____ Chief, Nuclear Generation Branch
_____ Assistant Maintenance Supervisor
_____ DPP Central Office File
_____ Training Center Coordinator
_____ Superintendent, Watts Bar Nuclear Plant
_____
_____

Rev. No.	Date	Revised Pages

The last page of this instruction is number __

ATTACHMENT 10

STEAR COVER SHEET

Special Test, Experiment, or Activity Request (STEAR) of a Licensed Unit	P PROD Designation STEAR:
Date: Originated By: Administrative Unit of Originator: Nuclear Plant & Unit: Plant System or Feature:	Reference Documents:

Description of and Reason for the Special Test, Experiment, or Activity:

	Yes	No	Initials	Date
Does the STEAR affect any CSSC item?				
Is the STEAR safety related?				

Safety Review for the STEAR Which Affects Items of the CSSC and are Safety Related.

Required safety reviews and safety analyses (10 CFR 50.59) have been completed for:

A. Unreviewed safety question

B. Technical specification limits

EN DES Signoff	PEB or PPMB Signoff	P PROD Signoff

Final Approval for Performing Special Test or Experiment:

Plant Operations Review Committee		Plant Superintendent	
Initials	Date	Initials	Date

Section 5

SPECIAL LOW POWER TEST PROGRAM
ADMINISTRATIVE PROCEDURES

Final test procedures, SQNP Special Tests, were submitted by L. M. Mills' letter to L. S. Rubenstein dated March 27, 1980. No substantive revisions to these procedures have been made as of April 9, 1980. As substantive revisions are made available for distribution, amended pages will be sent to the document holder. Substantive changes are any revisions except typographical corrections, syntax corrections, or proposals for changes.