PSEG

Public Service Electric and Gas Company 80 Park Place Newark, N.J. 07101 Phone 201/430-7000

Ref. 80-09

August 7, 1980

Mr. Harold R. Denton
Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. Albert Schwencer, Acting Chief
Licensing Branch #3
Division of Licensing

Dear Mr. Denton:

REQUEST FOR AMENDMENT
FACILITY OPERATING LICENSE DPR-75
UNIT NO. 2
SALEM GENERATING STATION
DOCKET NO. 50-311

In accordance with the Atomic Energy Act of 1954, as amended and the regulations thereunder, we hereby transmit copies of our request for amendment and our analysis of the changes to Facility Operating License DPR-75 for Salem Generating Station, Unit No. 2.

This request consists of proposed changes to the Safety Technical Specifications (Appendix A) involving changes to temporary exempt certain Technical Specification requirements during the conduct of the Special Low Power Test Program. The Technical Specifications requiring exemption are listed in Table 3-1 of the Safety Evaluation which is included as an attachment to this request. Westinghouse has reviewed our program and procedures and we have incorporated their comments.

This change involves an issue which has acceptability for the issue clearly identified by an NRC position, and is deemed not to involve a significant hazards consideration. Therefore, it is determined to be a Class III amendment as defined by 10CFR 170.22 and a check for the amount of $4,000.00 is enclosed.

This submittal includes three (3) originals and forty (40) copies.

B021 5 3/40

Very truly yours,

Frank P Librizzi REB

Frank P. Librizzi
General Manager -
Electric Production

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Ref. LCR 80-09

U.S. NUCLEAR REGULATORY COMMISSION
DOCKET NO. 50-311

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
FACILITY OPERATING LICENSE NO. DPR-75
NO. 2 UNIT
SALEM GENERATING STATION

Public Service Electric and Gas Company hereby submits proposed changes to Facility Operating License No. DPR-75 for Salem Generating Station, Unit No. 2. This change request relates to Safety Technical Specifications (Appendix A) of the Operating License, and pertains to exemption from certain Technical Specification requirements during the conduct of the Special Low Power Test Program.

Respectfully submitted,

PUBLIC SERVICE ELECTRIC AND GAS COMPANY

BY: Frederick W. Schneider
FREDERICK W. SCHNEIDER
VICE PRESIDENT

Ref. LCR 80-09

STATE OF NEW JERSEY)
) SS:
COUNTY OF ESSEX)

FREDERICK W. SCHNEIDER being duly sworn according to law deposes and says:

I am a Vice President of Public Service Electric and Gas Company, and as such, I signed the request for change to FACILITY OPERATING LICENSE NO. DPR-75.

The matters set forth in said change request are true to the best of my knowledge, information, and belief.

Frederick W. Schneider
FREDERICK W. SCHNEIDER

Subscribed and sworn to before me
this 7th day of August, 1980.

Jeanne E. Macciocca
Notary Public of New Jersey

My commission expires on Oct. 1, 1983

EG 460 A

Public Service Electric and Gas Company 80 Park Place Newark, New Jersey 07101

Attached is our check in full payment of items described below.

Check No.

U S NUCLEAR REGULATORY COMMISSION

Date AUG 1 80 1941

Date	Statement of Remittance	Amount of Invoice .00*	Deductions .00*	Net Amount .00*
1980 AUG 7	REQUEST FOR AMENDMENT LICENSE DPR-75	4,000.00		4,000.00*

092213

Detach and retain this statement. Please refer to above date and check number in your correspondence to the Vice President and Comptroller.

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Public Service Electric and Gas Company Newark, New Jersey 07101

To Fidelity Union Trust Company, Newark, N.J.

No. 1941

PSE&G

Date AUG 1 80

Pay $4,000 Dollars and .00 Cents $4,000.00T

To the order of

U S NUCLEAR REGULATORY COMMISSION
WASHINGTON D C 20555

RJ Hornok
Assistant Treasurer

Authorized Signature

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SALEM UNIT 2

SPECIAL LOW-POWER TESTS

SAFETY EVALUATION

JULY 1980

6110A

1.0 INTRODUCTION AND SUMMARY

In an attempt to move the licensing process for Salem, PSE&G proposed to the NRC the same special tests to be performed at reactor power levels at or below 5% of Rated Thermal Power as TVA proposed for Sequoyah. These tests would demonstrate the plant's capability in several simulated degraded modes of operation and would provide opportunities for operator training. The basic mode of operation to be demonstrated is natural circulation with various portions of the plant equipment not operating, e.g., pressurizer heaters, loss of offsite power (simulated), loss of onsite AC power (simulated), and RCPs for plant cooldown.

Westinghouse has reviewed the proposed tests and has with the exception of TVA proposed tests 8 and 9 (startup from stagnant conditions and boron mixing and cooldown) has determined that with close operator surveillance of parameters and suitable operator action points in the event of significant deviation from test conditions, the tests as outlined in the Salem Special Test procedures are acceptable and can be performed with minimal risk. It is recognized that in order to perform these tests some automatic safety functions, reactor trips and safety injection, will be defeated. Westinghouse has determined a set of operator action points which should replace these automatic actuations. It is also recognized that several technical specification requirements will not be met while either preparing for or performing these tests. Again Westinghouse has determined that the low power levels and operator action will suffice during these time periods.

Westinghouse has reviewed the effect of the proposed test conditions on the incidents and faults which were discussed in the Accident Analysis section of the Salem Final Safety Analysis Report. In most cases, the FSAR discussion was found to bound the consequences of such events occurring under testing conditions. Consequences of an ejected RCCA have not been analyzed because of the low probabilities. For some incidents, because of the far-off-normal conditions, the analysis methods available have not shown that, with reliance on automatic protection system action alone, the FSAR analyses are bounding. In those cases reliance is placed on expeditious operator action. The operator action points as defined will provide protection for such events.

2.0 DESCRIPTION OF TESTS

2.1 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.

2.2 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 1% 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.

2.3 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 effect on saturation margin and the depressurization rate. Demonstrate the effects of charging/letdown flow and steam generator pressure on the saturation margin.

2.4 EFFECT OF STEAM GENERATOR SECONDARY SIDE ISOLATON 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 so that two are isolated and establish equilibrium. Return the steam generators to service in reverse order.

2.5 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 meter 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.

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

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

Method - With the reactor shutdown, 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.

2.7 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, including the emergency diesel generators, the decay heat can be removed by using the auxiliary feedwater system in the manual mode.

Method - The reactor is shut down and all RCP's are running. A total station blackout is simulated. Instrument and lighting power is provided by the backup batteries since the diesels are shutdown.

2.8 ESTABLISHMENT OF NATURAL CIRCULATION FROM STAGNANT CONDITIONS

Westinghouse does not believe that it is advisable to perform this test as noted in a letter from T. M. Anderson, Westinghouse, to H. Denton, NRC, NS-TMA-2242, April 29, 1980.

2.9 FORCED CIRCULATION COOLDOWN

This test is performed as preparation for the Boron Mixing and Cooldown Test. Since Westinghouse does not believe it is advisable to perform the Boron Mixing Test as defined using core heat, it is not necessary to perform the Forced Circulation Cooldown Test.

2.10 BORON MIXING AND COOLDOWN

Westinghouse does not believe that it is advisable to perform this test utilizing core heat as noted in NS-TMA-2242, T. M. Anderson, Westinghouse, to H. Denton, NRC.

3.0 IMPACT ON PLANT TECHNICAL SPECIFICATIONS

In the evaluation of the proposed tests Westinghouse has determined that approximately thirteen technical specifications will be violated, and thus require exceptions, during the performance of the tests. Table 3-1 lists the technical specifications that will require exceptions and the tests for which they will not be met. The following notes the reasons these specifications must be excepted and the basis for continued operation during the tests.

3.1 IMPACT SUMMARY

3.1.1 T.S. 2.1.1 REACTOR CORE SAFETY LIMITS

The core limits restrict RCS T_{avg} as a function of power, RCS pressure (pressurizer pressure) and loops operable. These limits provide protection by insuring that the plant is not operated at higher temperatures or lower pressures than those previously analyzed. The core limits in the Salem tech specs are for four loop operation. Obviously when in natural circulation with no RCP's running these limits would not be met. However, it should be noted that the tests will be performed with limits on core exit temperature ($< 610^{o}F$), T_{avg} ($\leq 578^{o}F$) and Loop ΔT ($< 65^{o}F$) such that no boiling will be experienced in the core and the limits of specification 2.1.1 for temperature will be met. The limits will not be met simply because less than four RCP's would be running.

3.1.2 T.S. 2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip System provides protection from various transients and faulted conditions by tripping the plant when various process parameters exceed their analyzed values. When in natural circulation two trip functions will be rendered inoperable, Overtemperature ΔT and Overpower ΔT. There is a temperature input to these functions which originates from the RTD bypass loops. Due to the low flow conditions, 5% or less, the temperature indications from these loops will be highly suspect. To prevent the inadvertent tripping of the plant when in the natural circulation mode these functions will be bypassed. Their protection functions will be performed by the operator verifying that Pressurizer Pressure and Level, Steam Generator Level, and subcooling margin (T_{sat}) are above the operator action points for Reactor Trip and Safety Injection.

Steam Generator Level (Low and Low-Low) is the third trip function that can be affected. When at low power levels it is not uncommon for this function to be difficult to maintain above the trip setpoint. This function assures that there is some volume of water in the steam generators above the tops of the U-tubes to maintain a secondary side heat sink. The amount of water is based on the decay heat present in the core and to prevent dryout of the steam generators. With the plant limited to 5% RTP 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 will assure that this requirement will be met. Thus, Westinghouse finds that it is acceptable to lower the trip setpoints from 17% span and 25% span respective to 5% span for all of the special tests.

3.1.3 T.S. 3.1.1.4 MODERATOR TEMPERATURE COEFFICIENT

The Moderator Temperature Coefficient is limited to 0 pcm/^{o}F or more negative. When performing tests with the plant critical below 541^{o}F this coefficient may be slightly positive. However, it is expected that the Isothermal Temperature Coefficient will remain negative or approximately zero. The tests will be performed such that this is the case and thus minimizing any impact from rapid heatups or cooldowns. In addition, the effect of a small positive Moderator Temperature Coefficient has been considered in the accident analyses performed for the test conditions.

3.1.4 T.S. 3.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

The Minimum Temperature for Criticality is limited to 541°F by spec. 3.1.1.5 and 531°F by spec. 3.10.3. To perform test 4 it is expected that the RCS average temperature will drop below 531°F. Westinghouse has determined that operation with T_{avg} as low as 485°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 back-up trips rather than as primary trips.

3.1.5 T.S. 3.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

The reactor trips noted in Section 3.1.2 will not meet the operability requirements of spec. 3.3.1. Specification 3.3.1 can be excepted for the reasons noted in Section 3.1.2 of this evaluation.

3.1.6 T.S. 3.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

To prevent inadvertent Safety Injection and to allow performance of the special tests, all automatic Safety Injection functions will be blocked. Indication of partial Safety Injection logic trips and manual initiation will be operable, however, the automatic Safety Injection actuation functions will be made inoperable by forcing the logic to see that the reactor trip breakers are open. Westinghouse believes 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 the partial trip indication by the operator,

2. Rigid adherence to the operator action points as defined by Westinghouse, see Section 3.2.

3. 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 1.60% $\Delta K/K$ 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 and close operator surveillance will assure initiation of Safety Injection, if required, within a short time period.

3.1.7 T.S. 3.4.4 PRESSURIZER

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.

3.1.8 T.S. 3.7.1.2 AUXILIARY FEEDWATER SYSTEM

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 ($\sim$ 30 minutes) for operating personnel to rack in the pump power supplies and regain steam generator level.

3.1.9 T.S. 3.8.1.1, 3.8.2.1, 3.8.2.3 POWER SOURCES

These specifications are outside Westinghouse control, however it is acceptable to alter power source availability as long as manual Safety Injection is operable and safety related equipment will function when required.

3.1.10 T.S. 3.10.3 SPECIAL TEST EXCEPTIONS - PHYSICS TESTS

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 485°F this specification will be excepted. See Section 3.1.4 for basis of acceptability.

3.1.11 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. 3.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$
   At low temperatures and flows $F_Q(Z)$ can be expected to be above normal for 5% RTP with RCPs running. However at such a low power level no significant deviations in burnup or Xe peaks are expected.

2. 3.2.3 RCS FLOWRATE AND R - $(F_{\Delta H})$
   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. 3.2.4 QUADRANT POWER TILT RATIO
   With no, one, two, or three pumps running and critical, core power distributions resulting in quadrant power tilt may form. At low power levels and for short periods of times these tilts will not significantly influence core burn-up.

4. 3.2.5 DNB PARAMETERS

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.

3.1.12 SPECIAL TEST EXCEPTIONS

1. Special Test Exception Specification 3.10.3 allows limited exceptions for the following:

   3.1.1.4 Moderator Temperature Coefficient
   3.1.1.5 Minimum Temperature for Criticality
   3.1.3.1 Movable Control Assemblies
   3.1.3.5 Shutdown Rod Insertion Limits
   3.1.3.6 Control Rod Insertion Limits

2. Special Test Exception Specification 3.10.4 allows limited exception for 3.4.1.1 Reactor Coolant Loops - Normal Operation.

3.2 OPERATIONAL SAFETY CRITERIA

During the performance of these tests the operator must meet the following set of criteria for operation:

1. Maintain For All Tests

a) Primary System Sub-cooling (T_{sat} Margin)	> $20^{o}F$
b) Steam Generator Water Level	> 30% Narrow Range Span
c) Pressurizer Water Level	
(1) With RCPs running	> 22% Span
(2) Natural Circulation	≥ Value when RCPs tripped
d) Loop ΔT	≤ $65^{o}F$
e) T_{avg}	≤ $578^{o}F$
f) Core Exit Temperature (highest)	≤ $610^{o}F$
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

2. Reactor Trip and Test Termination must occur if any of the following conditions are met:

a) Primary System Sub-cooling (T_{sat} Margin)	≤ $15^{o}F$
b) Steam Generator Water Level	< 5% Narrow Range Span or Equivalent Wide Range Level
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	> $65^{o}F$
f) T_{avg}	> $578^{o}F$
g) Core Exit Temperature (highest)	> $610^{o}F$
i) Uncontrolled rod motion	
j) Control Bank D less than 100 steps withdrawn	

3. Safety Injection must be manually initiated if any of the following conditions are met:

a) Primary System Sub-cooling (T_{sat} Margin)	$\leq 10^{\circ}F$
b) Steam Generator Water Level	< 0% Narrow Range Span or Equivalent Wide Range Level
c) Containment Pressure	$\geq$ 4.7 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 must not be terminated until the Westinghouse criteria as defined in EOI:E-2, Loss of Secondary Coolant are met.

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

1. Sufficient margin to saturation temperature at system pressure to assure adequate core cooling (no boiling in the hot channel),

2. sufficient steam generator level to assure an adequate secondary side heat sink,

3. sufficient level in the pressurizer to assure coverage of the heaters to maintain pressure control

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

5. limit maximum possible power level in the event of an uncontrolled power increase.

TABLE 3-1

TECHNICAL SPECIFICATION IMPACT

	Technical Specification	Test 1	2	3	4	5	6	7
2.1.1	Core Safety Limits	X	X	X	X	X		
2.2.1	Various Reactor Trips							
	Overtemperature ΔT	X	X	X	X	X		X
	Overpower ΔT	X	X	X	X	X		X
	Steam Generator Level	X	X	X	X	X		X
3.1.1.4	Moderator Temperature Coefficient				X			
3.1.1.5	Minimum Temperature for Criticality				X			
3.3.1	Various Reactor Trips							
	Overtemperature ΔT	X	X	X	X	X		X
	Overpower ΔT	X	X	X	X	X		X
	Steam Generator Level	X	X	X	X	X		X
3.3.2	Safety Injection - All automatic functions	X	X	X	X	X		X
3.4.4	Pressurizer			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			

4.0 SAFETY EVALUATION

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 transient analyses in the FSAR are discussed.

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 - FAULTS 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 reduces the likelihood of an uncontrolled RCCA withdrawal. 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. An analysis of this event is presented in Section 4.2.1. For Test 7, this transient is bounded by the FSAR analysis, since all reactor coolant pumps are operating.

4.1.1.2 Uncontrolled Rod Control Cluster Assembly Bank Withdrawal at Power

The same considerations discussed in Paragraph 4.1.1.1 apply here. In addition, the low operating power and the Power Range Neutron Flux Low and Intermediate Range Neutron Flux trip setpoints 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 Control Cluster 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 constant 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; indeed 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 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 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.

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's Auxiliaries (Station Blackout)

Because of the low power level, the consequences of a loss of off-site 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 Neutron Flux Low setpoint trips, and close operator surveillance of feed flow, RCS temperatures, RCS pressure, and nuclear power. The case of excess heat removal due to feedwater system malfunctions 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 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 when core outlet subcooling reached approximately 10°F as an automatic backup 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 1925 psig 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 without Safety Injection. Thus the SAR analysis is bounding.

4.1.1.14 Spurious Operation 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 - INFREQUENT FAULTS

4.1.2.1 Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates Emergency Core Cooling

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 overheating on a short term basis. The system inventory and normal charging flow 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.

It must be noted that 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 resulting from the low power level and short operating history.

4.1.2.2 Minor Secondary System Pipe Breaks

The consequences of minor secondary system pipe breaks are within the bounds discussed in Paragraph 4.2.3.

4.1.2.3 Single Rod Cluster Control Assembly Withdrawal at Power

The FSAR analysis shows that assuming limiting parameters for normal operation a maximum of 5 percent of the fuel rods could experience a DNBR of less than 1.3 following a single RCCA withdrawal. As the FSAR points out, no single electrical or mechanical failure in the control system could cause such an event. The probability of such an event happening during the test period is further reduced by the short duration of this period, by the restriction to manual control, and by the close operator surveillance of reactor power, rod operation, and hot leg temperature.

4.1.2.4 Other Infrequent Faults

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, as described in the FSAR, have been reviewed and found to bound the consequences of such events occurring during test operation.

4.1.3 CONDITION IV - LIMITING FAULTS

4.1.3.1 Major Reactor Coolant Pipe Ruptures (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 on a short term basis. During the large break event the system inventory and cold leg accumulators will have removed enough energy to have filled the reactor vessel to the bottom of the nozzles. Following the system depressurization there is enough water in the reactor vessel below the nozzles to keep the core covered for over one hour using conservative assumptions. This is sufficient time for the operator to manually initiate SI and align the system for long term cooling. At no time during this transient will the core be uncovered.

It must be noted that 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 resulting from the low power level and short operating history.

4.1.3.2 Major Secondary System Pipe Rupture

The small moderator temperature reactivity coefficient, close operator surveillance of pressurizer pressure, cold leg temperature, and reactor power, with specific initiation criteria for reactor trip; low trip setpoints on the Intermediate-Range and Power-Range Neutron Flux trips; Low Flow Mismatch setpoint for Reactor Trip and MSIV closure on High Steamline Flow in coincidence with Low Steam Pressure; and Low Pressurizer Pressure trip (S.I. initiation) assure a Reactor Trip without excessive reactor power following a cooldown transient caused by the secondary system. Following reactor trip, assuming the worst 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 small break LOCA transients, and the operator actions required to deal with this 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 the faulted steam generator.

2. Steam line isolation of the faulted steam generator.

3. Steam relief capability of at least one non-faulted steam generator.

4. RCS depressurization capability.

Evaluation of the PSE&G 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 Accidents

The FSAR analysis of fuel handling accidents is bounding.

4.1.3.6 Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

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. 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 probability of an RCCA ejection during ths brief period in the test sequence does not warrant such an analysis.

4.2 ANALYSIS OF TRANSIENTS

4.2.1 ANALYSIS OF RCCA BANK WITHDRAWAL FROM SUBCRITICAL CONDITION

An analysis was performed to bound the test transients. The methods and assumptions used in the FSAR, Section 15.2.1 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 485°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 65% of full power, as is shown in Figure 4.2.1. The peak clad temperature reached was under 1300°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 RCCA 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 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 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 water from the steam generator reached the core; thus, considering the close and constant surveillance during these tests, 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 one moderator reactivity coefficient and 5% initial power are shown in Figures 4.2.5 and 4.2.6.

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 555oF.

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 1oF 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 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 which affect all loops uniformly will actuate reactor trip and steamline isolation on Low Steamline Pressure signals in any two lines, because the required coincident High Steamline Flow signal is set to zero flow. Low Pressurizer Pressure and Power Range Neutron Flux low setpoint trips serve as further backups. An example is the double ended rupture of a main steamline downstream of the flow restrictors with all steamline isolation valves initially open. Figures 4.2.7 and 4.2.8 show the response to such an event, with an initial power of 5% and natural circulation. The Low Steamline Pressure trip occurs almost immediately. 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.

4.3 ADDITIONAL CONSIDERATIONS

In the great majority of cases it was concluded, either by reanalysis or by comparison with previously analyzed FSAR conditions, that fuel clad integrity would be maintained without need for operator mitigating action. For the LOCA or steambreak events, it was concluded that the operator would have more than ample time (> 1 hour) to respond by manual action, e.g., manually initiate safety injection, to preclude fuel damage.

Finally, in certain other cases, primarily associated with certain inadvertent RCCA withdrawal events, the postulated accident conditions were neither amenable to direct analysis nor credit for operator intervention. In particular, the postulated accident conditions were outside the bounds of accepted analysis techniques so that fuel damage was not precluded either by analysis or identified operator action. For these cases, the basis for acceptability was primarily associated with the low probability of an inadvertent rod withdrawal event during the limited duration of the special tests.

This section provides an additional assessment relative to the potential for and consequences of fuel failure for these "unanalyzed" accident conditions associated with certain rod withdrawal events. This assessment is partially based upon an attempt to bound certain effects which may exist for conditions removed from the range of direct model applicability. Additional information (attached) is provided for four areas:

1. Thermal margin associated with normal test conditions.

2. The potential for DNB during accident conditions.

3. The clad temperature response assuming that DNB occurs.

4. Radiological consequences associated with presumed gross fuel failure.

The conclusions of this assessment are as follows:

1. DNB is not expected for the limiting thermal condition associated with any RCCA withdrawal event.

2. Even assuming DNB, there should be adequate heat transfer to prevent clad overheating.

3. Fuel clad failure is not expected.

4. Even assuming 100% clad failure and other extreme conservatisms, the resulting offsite dose would be small.

4.3.1 DESIGN CONSIDERATIONS

Margin to hot channel boiling has been incorporated with all normal test conditions by establishing a lower bound requirement on the degree of reactor coolant subcooling. This test requirement assures that postulated accidents are initiated from a condition of excess thermal margin.

4.3.2 DNB CONSIDERATIONS

For certain cooldown transients, the conclusion that DNB is precluded was drawn based on use of the W-3 critical heat flux correlation. Although the analyses for the cooldown events discussed in section 4.2.3.2 result in mass velocity below the range of direct applicability of the correlation, the reactor heat flux was so low relative to the predicted critical heat flux that even a factor of 2 would not result in serious concern for DNB for this event.

For the non-cooldown transients the limiting conditions, with respect to DNB, are farther away from the W-3 range of applicability because the coolant temperature is higher and the power-to-flow ratio is larger.

Comparison of the W-3 DNB correlation to low flow DNB test data and correlations (references 1 and 2) indicate that it will conservatively predict critical heat flux at low pressure ($\sim$ 1000 psi) conditions with low coolant flow. Pool boiling critical heat flux values (reference 3) at these pressures are higher than those predicted by the low flow correlations. Further review of the data in reference 1 indicates that the critical heat flux at higher pressure is significantly lower than the above data at 1000 psi. The minimum critical heat flux of the data set is $.16 \times 10^6$ BTU/hr-ft^2 for a data point at 2200 psia at a mass velocity of $.2 \times 10^6$ lbm/hr-ft^2.

Since the exit quality for this data point was 64%, it is unlikely that the reactor would be able to maintain a heat flux of that level due to the nuclear feedback from voiding. The power distribution would tend to peak towards the bottom thus further reducing the local quality at the peak flux locations.

Also the pool boiling correlations in reference 3 show some decrease in critical heat flux above 1000 psia to the maximum pressure of applicability of 2000 psia. However extrapolation of the correlations to a value of zero critical heat flux at the critical pressure (3206.2 psia) would not result in lower critical heat fluxes than shown in the data set from reference 1. Since the core average heat flux at 10% of nominal power (highest expected power for heatup events) is only on the order of .02 x 106 BTU/hr-ft2 a large peaking factor would be required to put the reactor heat flux as high as the critical heat flux.

For the transients considered, the only ones that lead to significant off normal peaking factors are rod motion transients. The rod withdrawal from subcritical is a power burst concern. As such, it is expected that even if DNB occurred, the rod surface would rewet. For the rod bank withdrawal, the combination of maximum power and peaking factor would result in a peak power lower than the data referenced above. Given the lack of data, it is difficult to completely preclude DNB, although a prudent judgement indicates that it is indeed remote.

4.3.3 CLAD TEMPERATURE CONSIDERATIONS

Should DNB occur, the peak clad temperature reached would depend primarily on the local nuclear transient following DNB and on the behavior of the post-DNB heat transfer coefficient.

For a rapid power transient, as is illustrated by the SER analysis for RCCA bank withdrawal from a subcritical condition, the fuel temperature reactivity feedback and reactor trip on a nuclear flux signal would shut down the reactor before sufficient energy could be generated to cause a damaging rise in clad temperature. In that case, the maximum clad temperature calculated was under 1300oF even assuming an extremely low heat transfer coefficient (~ 2 BTU/hr-ft2-oF).

A possibly more limiting condition for RCCA withdrawal would be the case in which a power increase causes DNB but would either not result in reactor trip on high nuclear flux or the trip is delayed. In the former case, a steady state condition with hot spot DNB could be postulated. In this state the clad temperature could be calculated given only the total core power, local heat flux channel factor, heat transfer coefficient and saturation temperature.

The core power is postulated to be essentially at the power which would cause a reactor trip on high Power Range Neutron Flux low setpoint. The trip setpoint is at 7% for these tests. To allow for calorimetric errors and normal system errors, trip is assumed to occur at 10% of rated thermal power (RTP), unless a large decrease in downcomer coolant temperature occurs during the test. In tests 3 and 5, depressurization to less than approximately 1450 psia could require temperature reduction, as is indicated in Figure 4.3.1; however, such low pressures are not expected.

Figure 4.3.2 shows the relationship of peak clad temperature, local heat transfer coefficient, and the product of heat flux hot channel factor (F_Q) times core power (fraction of RTP). For the event of an uncontrolled RCCA bank or single RCCA the upper bound of this heat flux product is approximately 0.34. Using this value, the heat transfer coefficient required to keep the peak clad temperature below 1800oF, the threshold of significant heat flux increases due to zirconium-water reaction, can be found from Figure 4.3.2.

Various film boiling heat transfer correlations have been reviewed to evaluate the heat transfer coefficient for post-DNB conditions. Although no correlations were found which cover the complete range of conditions being tested, some data exist which can be extrapolated to obtain representative heat transfer coefficients. The Westinghouse UHI film boiling correlation (reference 4), was developed at low flow conditions similar to those postulated for incidents occurring during the PSE&G tests. This correlation was extrapolated to the higher pressure conditions of the tests to obtain representative film boiling coefficients. This resulted in a heat transfer coefficient in excess of (100 BTU/hr-ft2-oF)a,c at 2200 psia and 5% flow with quality between 10-50%. Other film boiling heat transfer correlations, developed at higher pressures, were also examined. These correlations were extrapolated down to the lower flow conditions of the PSE&G tests as another approach to obtain representative film boiling coefficients. Using both the Mattson et al (reference 5) and the Tong (reference 6) film boiling correlations resulted in post-DNB heat transfer coefficients in excess of 150 BTU/hr-ft^2-°F at the conditions given above.

These results indicate that a clad temperature excursion resulting in fuel damage is not likely to occur even if DNB is assumed.

4.3.4 DOSE ANALYSIS CONSIDERATIONS

The dose analyses were performed for a hypothetical accident senario using conservative assumptions so as to determine an extreme upper bound on postulated accident consequences. The analysis assumed a reactor accident involving no pipe-break with a coincident loss of condenser vacuum. This accident scenario is representative of the Condition II type events analyzed in the FSAR. The bounding were assumptions made in the analysis which include:

170 Mwt (5% power)
1.0 dose-equivalent I-131 RCS activity (tech spec limit)
500 gpd steam generator leak in each SG (tech spec limit)
100% clad damage and gap activity release

10% iodine/noble gas in gap space
100 DF in steam generators
500 iodine spike factor over steady state
509,000 lb. atmospheric steam dump over 2 hours
1.7×10^{-3} sec/m^3 x/Q percentile value

The results of the analysis show that the two hour site boundary doses would be 5 rem thyroid, 0.9 rem total body and 0.4 rem to the skin.

The analysis of the accidents has incorporated some very conservative assumptions which goes beyond the normal degree of conservatism used in FSAR analyses. The most prominent of these assumptions and a brief description of the extreme conservatism includes:

1) Equilibrium radionuclide inventories established at 5% power. For iodines, this requires ~ 1 month of steady state operation at 5% uninterrupted.

2) Fuel clad gap inventories at 10% of core inventory, this is a time dependent, temperature dependent phenomona. At 5% power, very little diffusion to gap space is expected for the short test period.

3) 100% fuel rod clad damage.

4) Primary to secondary leakage to tech spec values. Since Salem is a new plant, no primary to secondary leakage is expected. If leakage were present, it would most likely slowly increase in steps up to tech spec levels.

5) Percentile meteorology, there is 95% probability of better diffusion characteristics and thus lower offsite doses.

For these reasons, in the unlikely event of a potential accident during the tests, the resulting dose is small, even assuming 100% clad damage and other extreme conservatisms.

4.3.5 OTHER CONCERNS

The LOCA analyses presented indicate that there are over 6,000 seconds for the operator to take action. This is more than sufficient time for the operator to take corrective action. Some transients were not analyzed or discussed in this supplement due to the combination of the low probability of the transient occurring and the very short time period of the special tests. This is true for the rod ejection accident. The combination of the low probability of occurring and the bounding dose evaluation for a condition II transient given here indicate that these events do not need to be analyzed. Similar dose calculations have been done for the steamline break accidents which results in somewhat higher doses than the condition II analysis. These dose results indicate that the fact that the NIS channels are not completely qualified does not alter the conclusion that the results are bounded.

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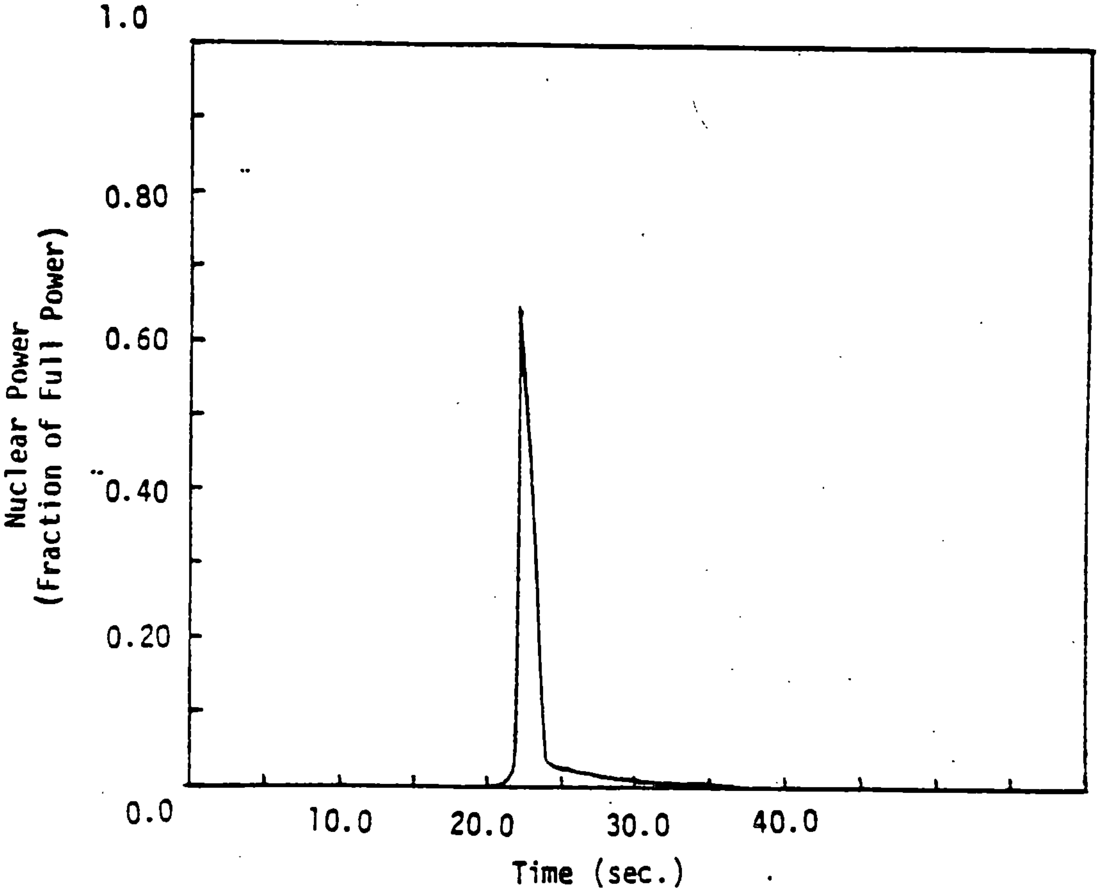

Figure 4.2.1 Uncontrolled Rod Bank Withdrawal from a Subcritical Condition, Neutron Flux vs. Time

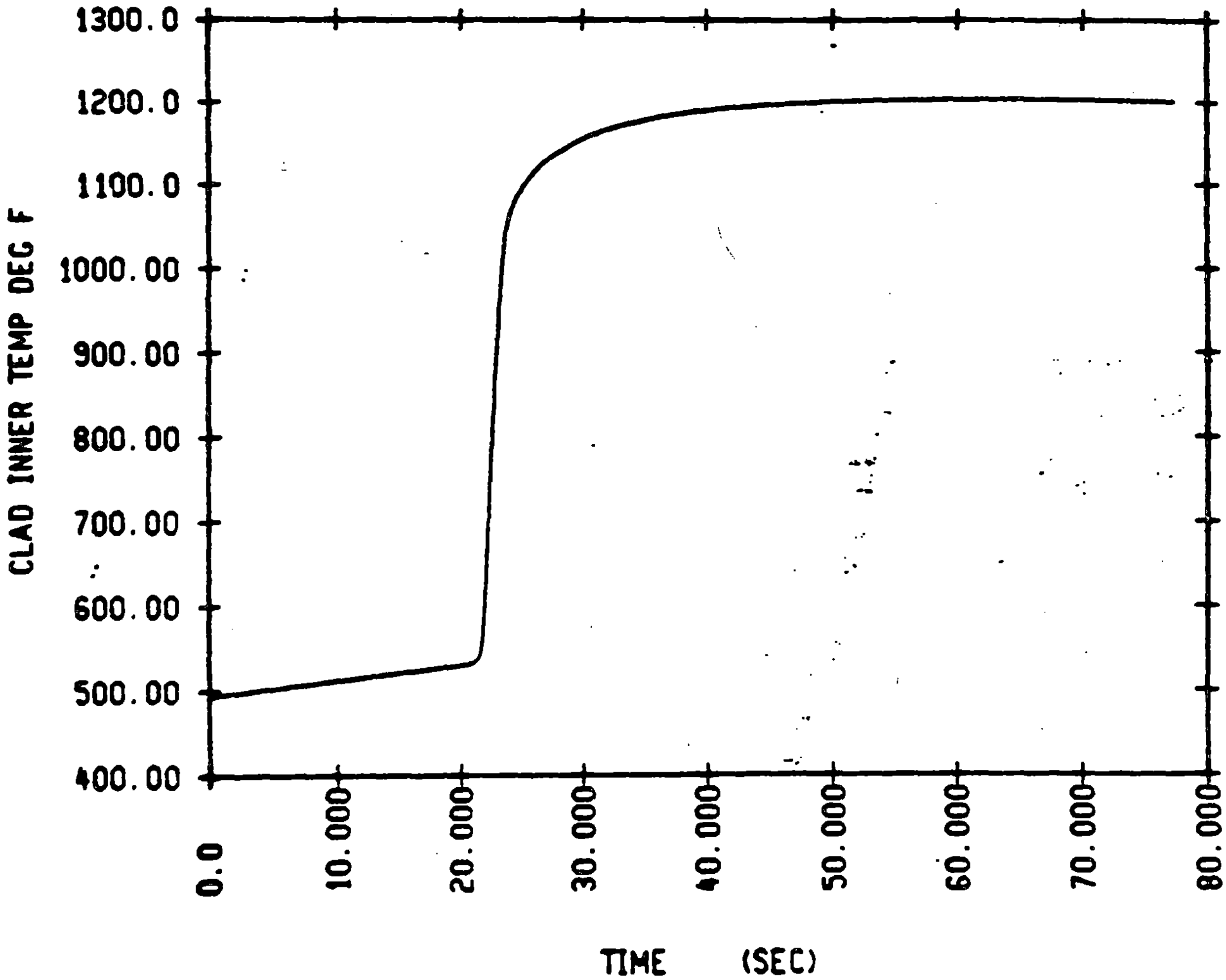

Figure 4.2.2 Uncontrolled Rod Bank Withdrawal from a Subcritical Condition, Hot Spot Clad Temperature vs Time, Assuming DNB at Time = 0.

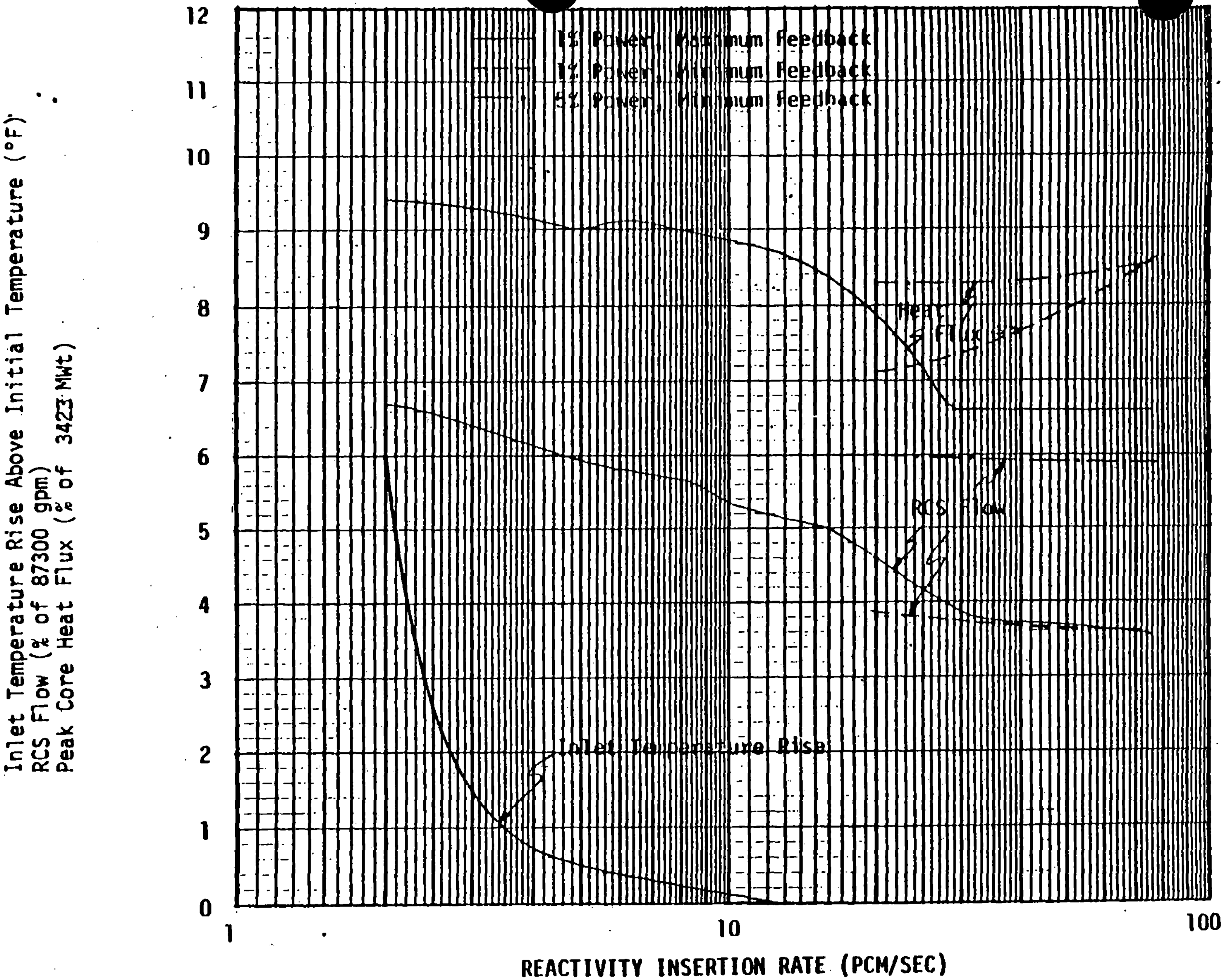

Figure 4-2.3 Uncontrolled Rod Bank Withdrawal at Power. Peak Heat Flux, Core Coolant Flow, and Increase in Core Inlet Temperature

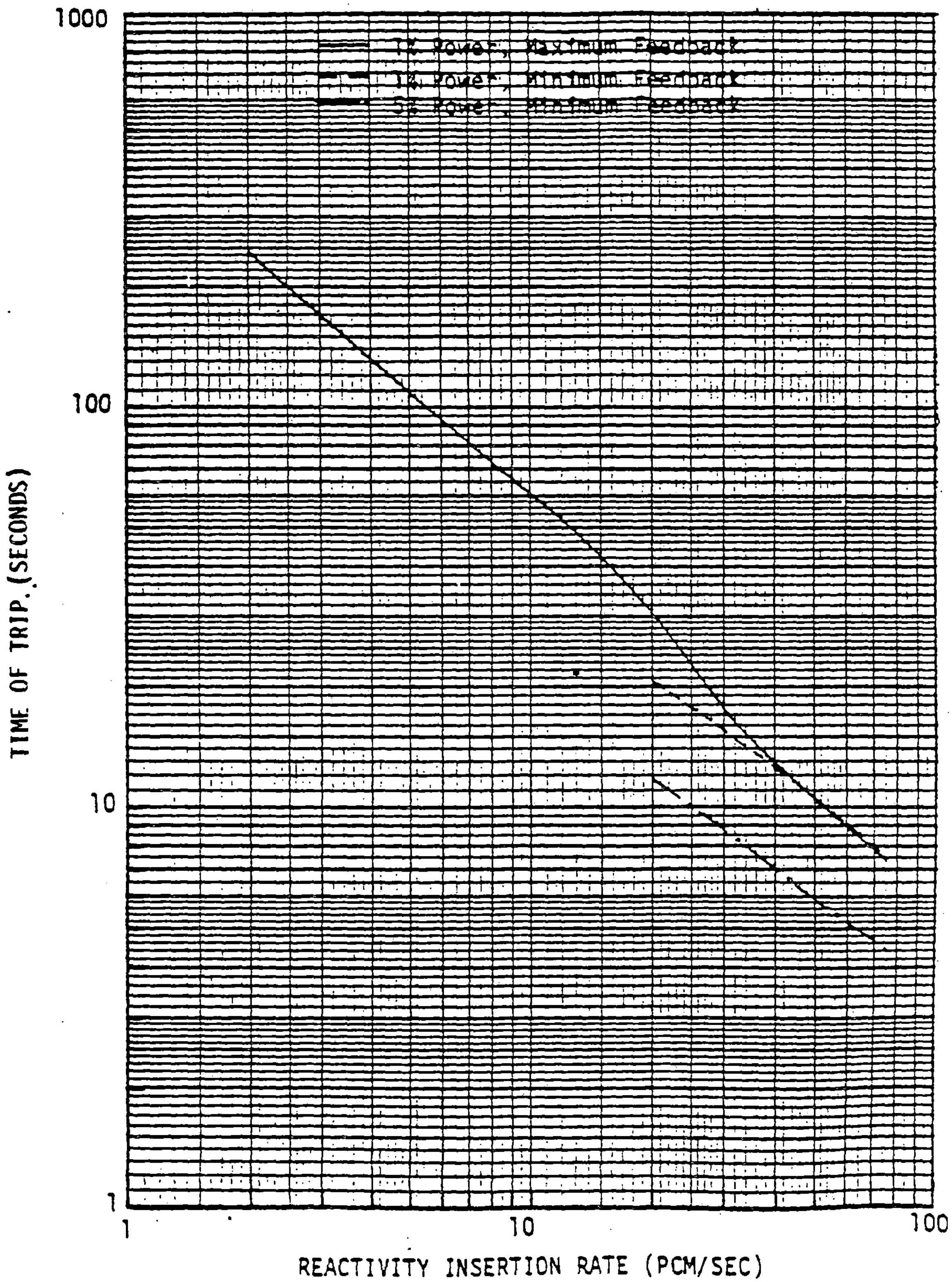

Figure 4.2.4 Uncontrolled Rod Bank Withdrawal at Power. Time of Reactor Trip vs Reactivity Insertion Rate

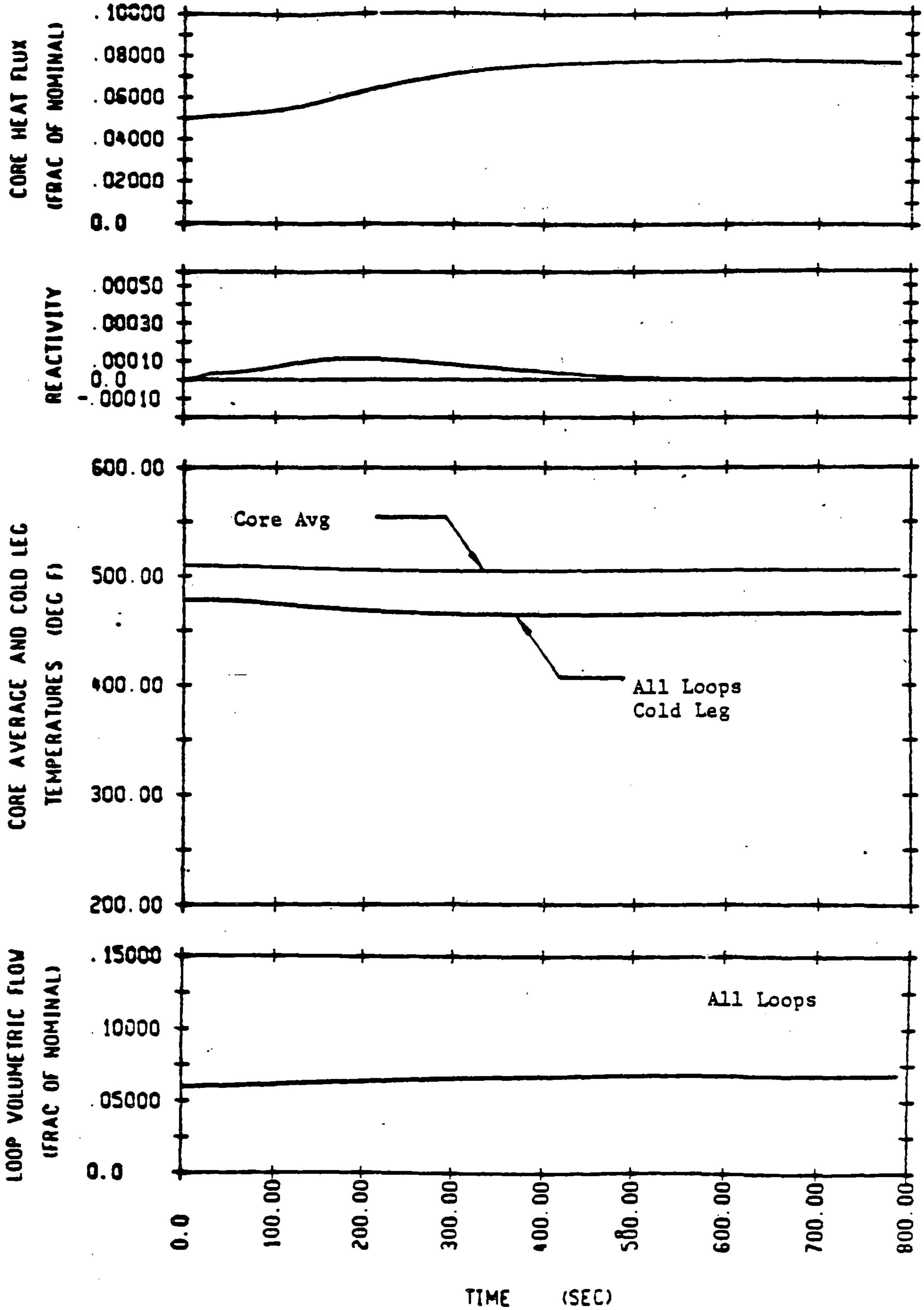

FIGURE 4.2.5 TRANSIENTS IN THE REACTOR CORE AND COOLANT LOOPS FOLLOWING THE OPENING OF A STEAM DUMP VALVE FROM 5% POWER, ALL LOOPS ACTIVE

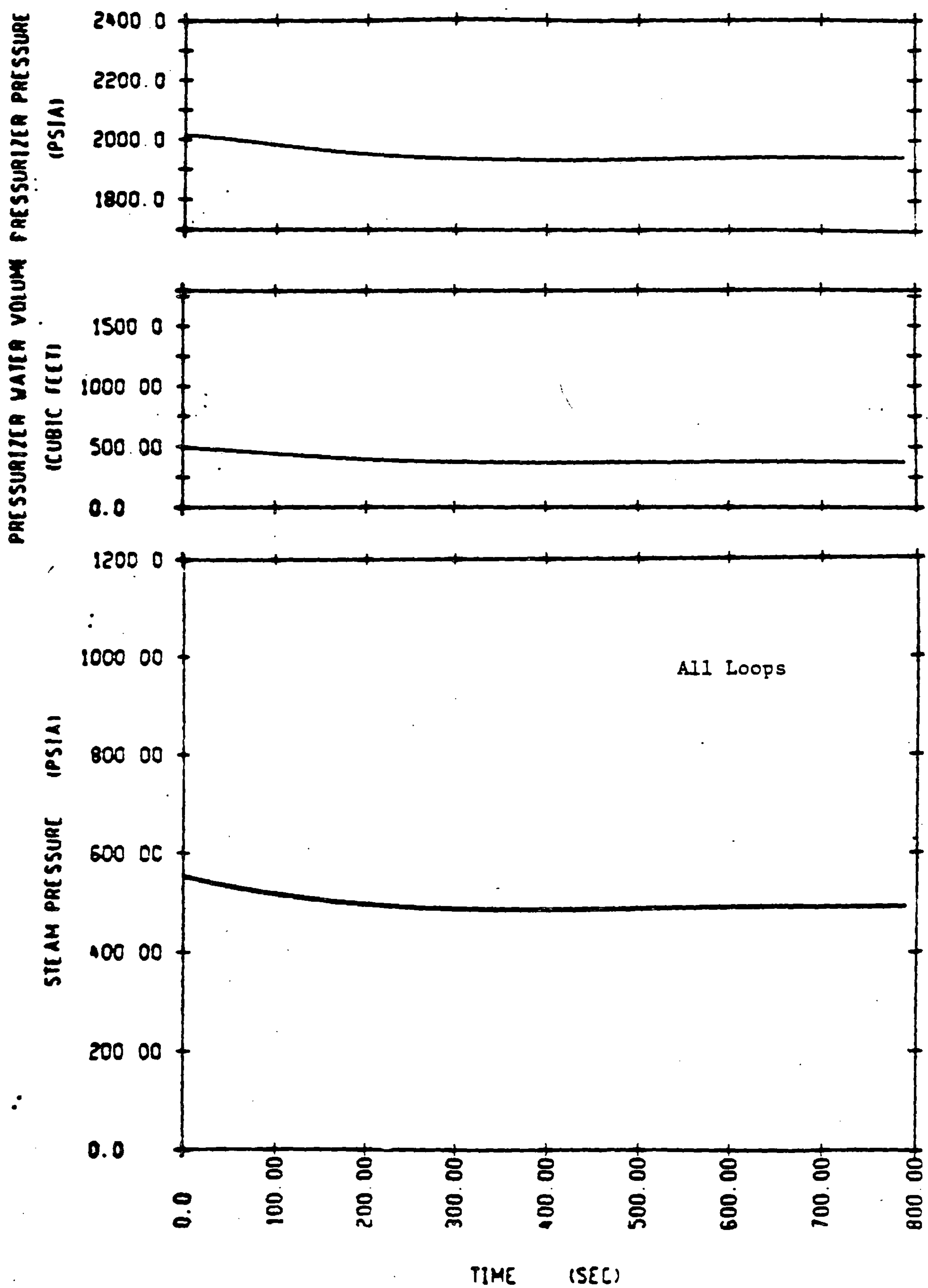

FIGURE 4.2.6 TRANSIENTS IN THE PRESSURIZER AND STEAM GENERATOR FOLLOWING THE OPENING OF A STEAM DUMP VALVE FROM 5% POWER, ALL LOOPS ACTIVE

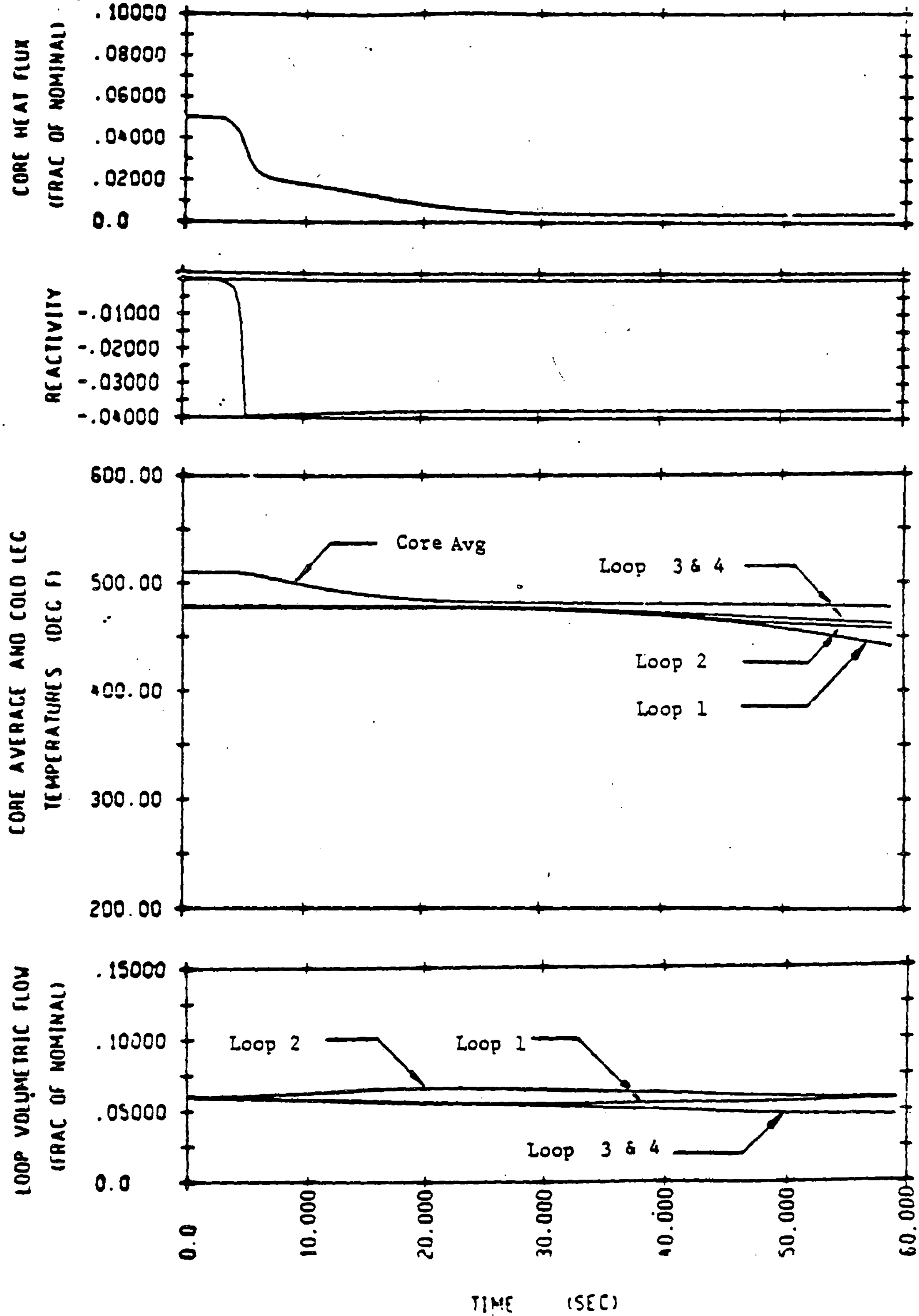

FIGURE 4.2.7 TRANSIENTS IN THE REACTOR CORE AND COOLANT LOOPS FOLLOWING A DOUBLE ENDED RUPTURE OF A MAIN STEAM-LINE DOWNSTREAM OF THE STEAMLINE ISOLATION

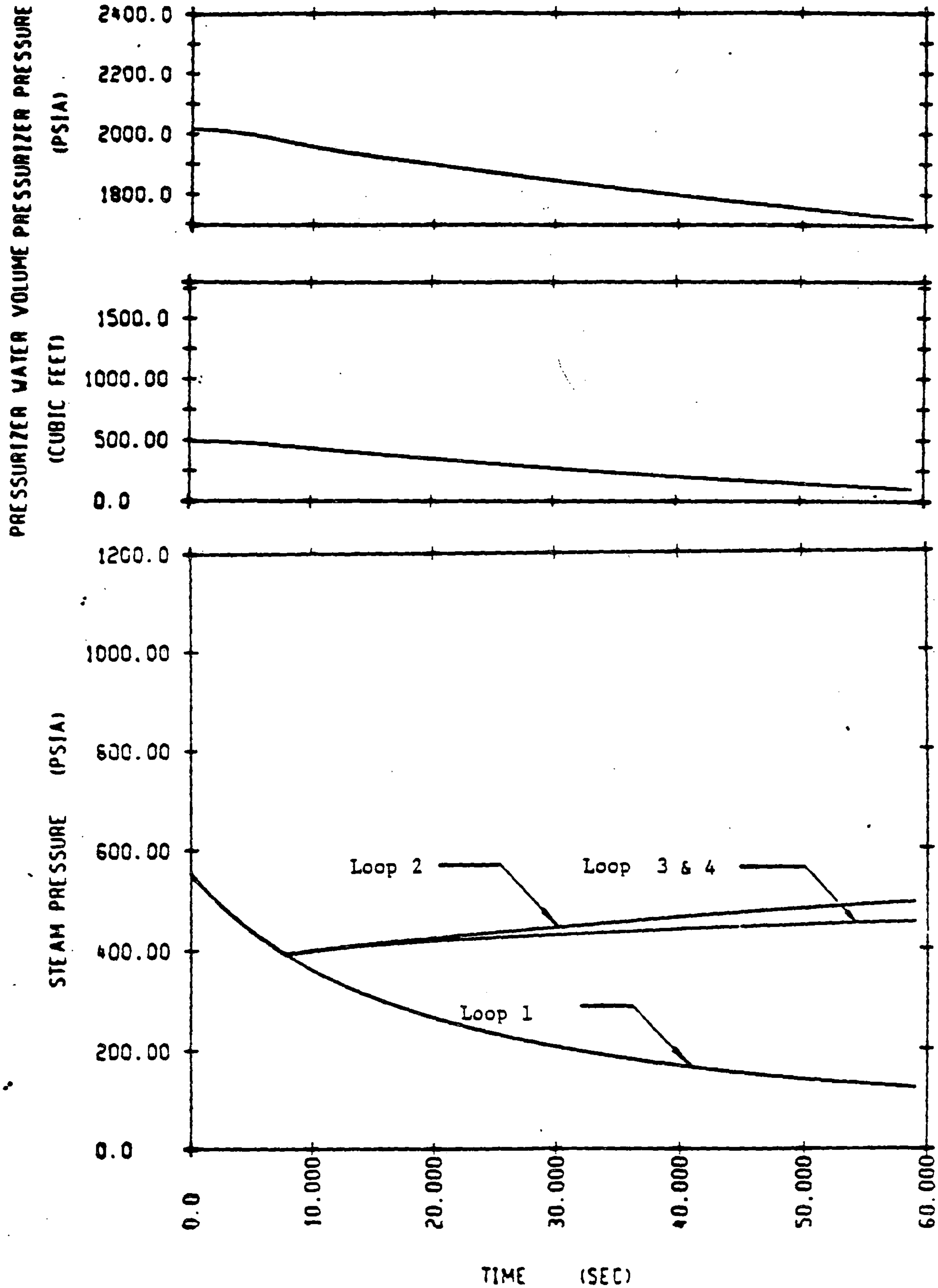

FIGURE 4.2.8 TRANSIENTS IN THE PRESSURIZER AND STEAM GENERATOR FOLLOWING A DOUBLE ENDED RUPTURE OF A MAIN STEAMLINE DOWNSTREAM OF THE STEAMLINE ISOLATION

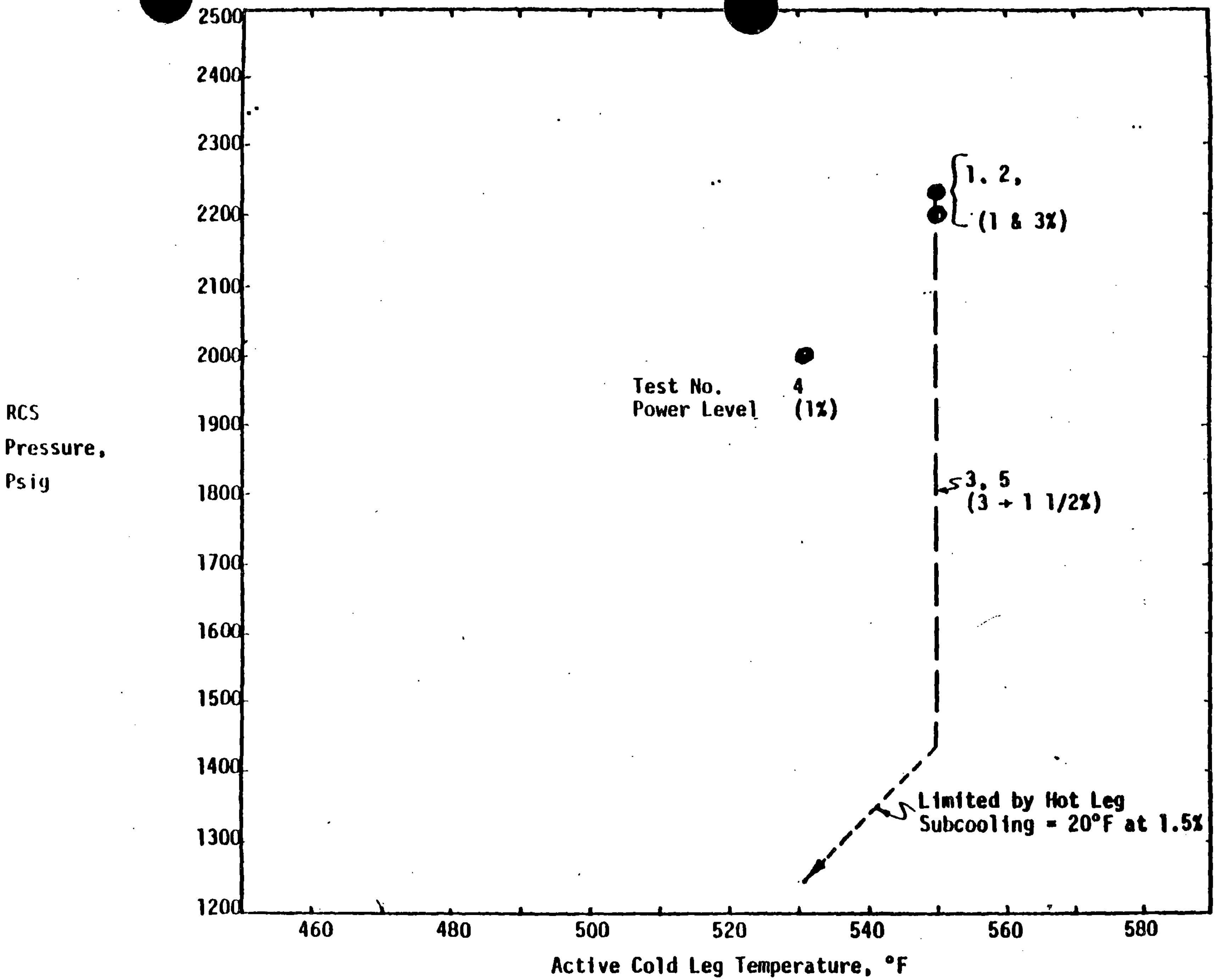

FIGURE 4.3.1 NATURAL CIRCULATION TEST CONDITIONS

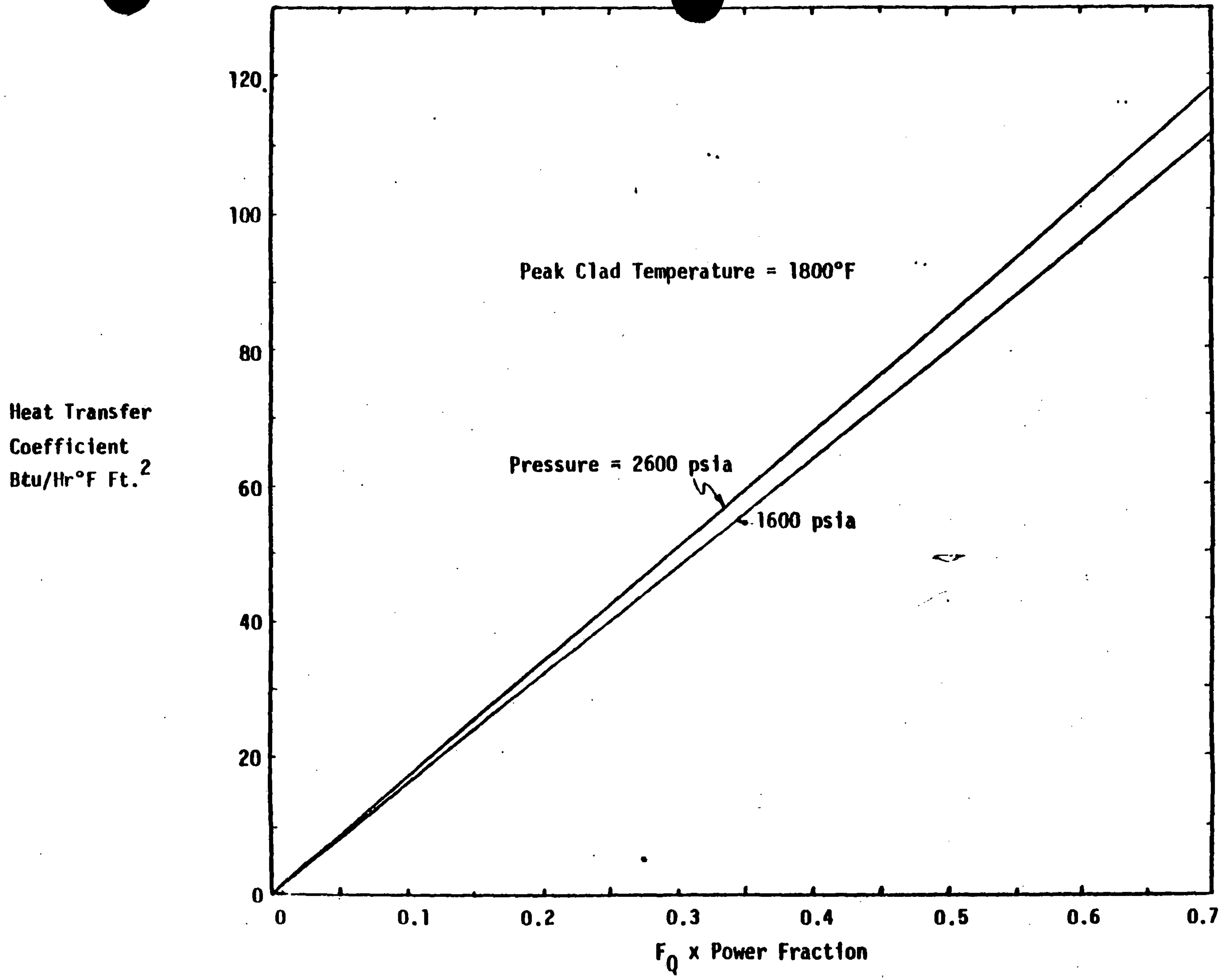

FIGURE 4.3.2 HEAT TRANSFER COEFFICIENT VS. HEAT FLUX FOR CLAD TEMPERATURE OF 1800°F