ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATION

3.4.3 AUXILIARY FEEDWATER SYSTEM

APPLICABILITY: Applies to the auxiliary feedwater pumps and valves for MODEs 1, 2, and 3.

OBJECTIVE: To ensure the availability of auxiliary feedwater to remove decay heat from the core.

SPECIFICATION: Two trains of auxiliary feedwater, including associated pumps and valves, shall be OPERABLE.

ACTION:

A. With one Train of auxiliary feedwater inoperable, restore the inoperable train to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

B. With both Trains of auxiliary feedwater inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

BASIS: The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of offsite power.

Two auxiliary feedwater trains and the steam system relief valves provide core decay heat removal capability in the event of a sustained loss of off-site power. Either auxiliary feedwater train has the capability to satisfy decay heat removal requirements from the core, with a delivered flow of at least 185 gpm per train with three intact main feedwater lines and pressurized steam generators, 100 gpm per train with two intact main feedwater lines and pressurized steam generators, and 175 gpm per train with two intact main feedwater lines and depressurized steam generators.

AFW System Train A pumps and valves consist of AFW pumps G-10S and G-10 and associated valves, including flow control valves FCV-2300A, FCV-2300B, and FCV-2300C.

AFW System Train B pump and valves consist of AFW pump G-10W and associated valves, including flow control valves FCV-3300A, FCV-3300B, and FCV-3300C.

Attachment 3

AUXILIARY FEEDWATER SYSTEM
Accident ANALYSIS
FEEDWATER LINE BREAK
SAN ONOFRE UNIT 1

Auxiliary Feedwater System
Accident Analysis
San Onofre Unit 1

BACKGROUND

During review of the test results of the upgraded auxiliary feedwater system installed during the Cycle 10 refueling outage, Southern California Edison identified that, under certain circumstances, the AFW flow rate would exceed the waterhammer limit of 150 gpm per steam generator. This is contrary to the design and licensing basis of the AFW system.

In order to meet the design and licensing basis of the AFW system with respect to waterhammer, a permanent design change is to be implemented in the Cycle 11 refueling outage. This modification consists of resizing the three flow venturis in the AFW lines to each generator to limit the AFW flow to less than 150 gpm per steam generator under all post-trip or post-accident conditions.

The resizing of the AFWS venturis to meet waterhammer limits reduces the AFW flow rate available for Loss of Normal Feedwater (LONF) and Feedline Break (FLB) events. Hence, these events were reanalyzed as is discussed below with the reduced AFW flow rates to demonstrate that acceptance criteria are met.

The SONGS 1 current licensing basis consists of the following cases A-G (References 1 and 2).

UFSAR Case A - Partial Loss of Normal Feedwater at 100% with AFW flow of 185 gpm. This case was not reanalyzed since the resizing of the AFWS venturi does not affect AFW performance in this case.

UFSAR Case B - Complete Loss of Normal Feedwater at 100% power with AFW flow of 165 gpm. This case was not reanalyzed since the resizing of the AFWS venturi does not affect AFW performance in this case.

UFSAR Case C - Complete Loss of Normal Feedwater at 50% power with AFW flow of 185 gpm. This case was not reanalyzed since the resizing of the AFWS venturi does not affect AFW performance in this case.

UFSAR Case D - Main Feedwater Line Break Upstream of In-Containment Check Valves at 100% power with AFW flow of 125 gpm. This was reanalyzed using 100 gpm flow rate to the two intact steam generators.

UFSAR Case E - Main Feedwater Line Break Upstream of In-Containment Check Valves at 50% power with AFW flow of 125 gpm. This was reanalyzed using 100 gpm flow rate to the two intact steam generators.

UFSAR Case F - Main Feedwater Line Break Downstream of In-Containment Check Valves at 100% power with AFW flow of 250 gpm. This was reanalyzed using 175 gpm flow rate to the two intact steam generators.

UFSAR Case G - Main Feedwater Line Break Downstream of In-Containment Check Valves at 50% power with AFW flow of 250 gpm. This was reanalyzed using 175 gpm flow rate to the two intact steam generators.

UFSAR Cases A, B, and C are not reanalyzed, since they remain bounded for the replaced venturis. Only feedline breaks were reanalyzed.

UFSAR Cases D and E were reanalyzed to provide increased margin between minimum required AFW flow and the actual AFW flow.

UFSAR Cases F and G were reanalyzed since the reduced flow resulting from the replaced venturis was less than previously analyzed.

ACCEPTANCE CRITERIA

A feedline break event is considered an ANS 18.2 Condition IV incident. The Standard Review Plan (Reference 3) acceptance criteria for the Feedline Break event are as follows:

- Pressure in the RCS and Main Steam System should be maintained below 110% of the design values.

- Any fuel damage that may occur during the accident should be of a sufficiently limited extent so that the core will remain in place and geometrically intact with no loss of core cooling capability.

- Any activity release must be such that the calculated doses at the site boundary are well within the guidelines of 10CFR Part 100.

Westinghouse has adopted the following criteria, for purposes of interpreting the accident results of this Condition IV Accident:

- Maximum pressures do not exceed 110% of the design values.

- The core remains in place and geometrically intact with no loss of core cooling capability because the core remains covered with water.

- Any activity releases must be such that the calculated doses at the site boundary are well within the guidelines of 10CFR Part 100.

The Westinghouse acceptance criteria for a feedline break event have been extended for use in SONGS 1 FLB analysis. Calculations were performed to show that sufficient RCS mass is available to keep the core covered throughout the event. Thus the core remains in a coolable geometry.

METHOD OF ANALYSIS

The LOFTRAN code (Reference 4) was used to simulate the accidents. The assumptions applicable to all four cases are presented below. The assumptions specific to each case is presented separately. All assumptions, including initial conditions, were selected to maximize the consequences of the applicable accident.

General Assumptions

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

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

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

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

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

6. 1979 ANS 5.1 Decay Heat is modeled.

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

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

UFSAR Case D: Main Feedwater Line Break Upstream of In-Containment Check Valves at 100% power

Specific Assumptions

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

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

3. Initial pressurizer water level is 50% NRS.

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

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

6. **AFW is assumed to be manually actuated and the system manually aligned to deliver flow of 100 gpm split equally between the two intact steam generators 30 minutes after the initiation of the event (feedline break).**

7. **The steam flow/feedwater flow mismatch reactor trip is assumed available. Consistent with the Loss of Normal Feedwater (LONF) analysis of reference 1, reactor trip is assumed to occur 10 seconds after the feedline break.** Note the longer delay for reactor trip associated with the LONF **analysis is assumed since this scenario initially behaves** as **a complete loss of normal feedwater.**

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

Results and Conclusions

The results of the feedline break at full power located upstream of inside containment check valve accident are shown in figures 1 through 4. The time sequence of events is presented in table 2. Reactor trip is provided by the steam flow/feedwater flow mismatch signal. The results show that an AFW flow of 100 gpm initiated 30 minutes after the break is sufficient to remove core decay heat. Calculations of this case show that the core remained in a coolable geometry during this FLB scenario.The detailed calculations involved showing that the mass relieved through the pressurizer PORVs (between the time of initial relief through the PORVs and the time the PORVs reseat due to the heat removal capability of the AFW exceeding the core decay heat) was not sufficient to uncover the core. As such, the acceptance criteria for a FLB event the accident was met.

UFSAR Case E: Main Feedwater Line Break Upstream of In-Containment check Valves at 50% Power

Specific Assumptions

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

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

3. Initial pressurizer water level is 30.0% NRS.

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

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

6. AFW is assumed to be manually actuated and the system manually aligned to deliver flow of 100 gpm split equally between the two steam generators 15 minutes after the initiation of the event (feedline break).

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

Results and Conclusions

The results of the feedline break at 50% power located upstream of inside containment check valve accident are shown in figures 5 through 8. The time sequence of events is presented in table 3. Reactor trip is provided by high pressurizer water level (50% NRS) signal. The results show that an AFW flow of 100 gpm initiated 15 minutes after the break is sufficient to remove core decay heat. The reactor coolant system (RCS) remains subcooled and the pressurizer does not fill. As such, the acceptance criterion for a FLB event that the core remains in a coolable geometry during the accident was shown to be met.

UFSAR Case F: Main Feedwater Line Break Downstream of the In-Containment Check Valves at 100% Power.

Specific Assumptions

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

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

3. Initial pressurizer water level is 50% NRS.

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

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

6. AFW is assumed to be manually actuated and the system manually aligned to deliver flow of 175 gpm split equally between the two intact steam generators 20 minutes after the initiation of the event (feedline break).

7. The steam flow/feed flow mismatch reactor trip is assumed available. Reactor trip is assumed to occur 5 seconds after the feedline break.

Results and Conclusions

The results of the feedline break at full power located downstream of inside containment check valve accident are shown in figures 9 through 12. The time sequence of events is presented in table 4. Reactor trip is provided by the steam flow/feedwater flow mismatch signal. The results show that an AFW flow of 175 gpm initiated 20 minutes after the break is sufficient to remove core decay heat. Calculations of this case show that the core remained in a coolable geometry during this FLB scenario. The detailed calculations involved showing that the mass relieved through the pressurizer PORVs (between the time of initial relief through the PORVs and the time the PORVs reseat due to the heat removal capability of the AFW exceeding the core decay heat) was not sufficient to uncover the core. As such, the acceptance criterion for a FLB event that the core remains in a coolable geometry during the accident was met.

UFSAR Case G: Main feedwater Line Break Downstream of the In-Containment Check Valves at 50% Power

Specific Assumptions

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

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

3. Initial pressurizer water level is 30.0% NRS.

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

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

6. AFW is assumed to be manually actuated and the system manually aligned to deliver flow of 175 gpm split equally between the two intact steam generators 15 minutes after the initiation of the event (feedline break).

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

Results and Conclusions

The results of the feedline break at 50% power located upstream of inside containment check valve accident are shown in figures 13 through 16. The time sequence of events is presented in table 5. Reactor trip is provided by high pressurizer water level (50% NRS) signal. The results show that an AFW flow of 175 gpm initiated 15 minutes after the break is sufficient to remove core decay heat. The detailed calculations involved showing that the mass relieved through the pressurizer PORVs (between the time of initial relief through the PORVs and the time the PORVs reseat due to the heat removal capability of the AFW exceeding the core decay heat) was not sufficient to uncover the core. As such, the acceptance criterion for a FLB event that the core remains in a coolable geometry during the accident was met.

3702

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Mr. Don Hickman
NRR Project Manager
San Onofre Units 2 & 3
U. S. Nuclear Regulatory Commission
One White Flint North
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Rockville, Md 20852

CONCLUSIONS

The reanalysis of the Rupture of a Main Feedwater Pipe supports SONGS 1 operation with the reduced AFW flows presented in Table 1. In each case, except Case E, where the RCS remained subcooled, boiling occurred in the hot leg and reactor coolant was relieved through the pressurizer PORVs. However, the mass relieved through the PORVs was not sufficient to uncover the core and the core remained covered and in a coolable geometry at all times. Thus, all applicable acceptance criteria are shown to be met. In terms of radiological consequences following a feedline break, the four cases analyzed for this report are bounded by the radiological consequences accepted previously in the UFSAR.

REFERENCES

1. Letter from SCE (Medford) to NRC (Lear), "Requests for Additional Information, SONGs 1", dated May 1, 1986.

2. Letter from SCE (Medford) to NRC, Engineered Safety Features Single Failure Analysis SONGs 1, dated November 20, 1987.

3. NUREG-0800, Revision 1, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants-LWR Edition," July 1981.

4. Burnett, T.W.T., et al., "LOFTRAN Code Description," WCAP-7907-A, April 1984.

TABLE 1

FEEDLINE BREAK ANALYSIS ASSUMPTIONS

UFSAR CASE DESCRIPTION	POWER LEVEL %	NOMINAL FULL POWER Tavg °F	INITIAL TRANSIENT Tavg °F	PRESSURIZER SPRAY	INITIAL PRESS. % NRS	REVISED AFW FLOW GPM	AFW START TIME MINUTES
Upstream Feedline Break Case D	103	575.15	579.15	NO	50	100	30
Upstream Feedline Break Case E	53	575.15	555.5	NO	30	100	15
Downstream Feedline Break Case F	103	575.15	579.15	NO	50	175	20
Downstream Feedline Break Case G	53	575.15	555.5	NO	30	175	15

TABLE 2

TIME SEQUENCE OF EVENTS FOR CASE D FLB

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

Event	Time, sec
Feedline Break between the 2 MFW check valves	10.
Reactor trip on steam flow/feed flow mismatch	20.
Rods begin to drop	20.
Pressurizer PORVs open (2200 psia)	1273.
AFW manually started of 100 gpm to 2 steam generators	1810.
Cold AFW reaches 2 steam generators	2420.
Pressurizer PORVs close	7930.
Heat removal of AFW is capable of removing core decay heat	8220.

TABLE 3

TIME SEQUENCE OF EVENTS FOR CASE E FLB

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

Event	Time, sec
Feedline Break between the 2 MFW check valves	10.
Pressurizer PORVs open (2200 psia)	36.
Reactor trip on high pressurizer water level (50% NRS)	160.
Rods begin to drop	162.
Pressurizer PORVs close	165.
Pressurizer PORVs open (2200 psia)	476.
AFW manually started of 100 gpm to 2 steam generators	910.
Cold AFW reaches 2 steam generators	1570.
Pressurizer PORVs close	1600.
Heat removal of AFW is capable of removing core decay heat	1660.

TABLE 4

TIME SEQUENCE OF EVENTS FOR CASE F FLB

Downstream FLB initiated at 103% power
with 175 gpm AFW initiated 20 minutes after the break

Event	Time, sec
Feedline Break downstream of MFW check valves inside containment	10.
Reactor trip on steam flow/feed flow mismatch	15.
Rods begin to drop	15.
Pressurizer PORVs open (2200 psia)	480.
AFW starts - 175 gpm to 2 steam generators	1215.
Cold AFW reaches 2 steam generators	1594.
Heat removal of AFW is capable of removing core decay heat (T_{avg} begins to drop)	1600.
Pressurizer PORVs close	1693.

TABLE 5

TIME SEQUENCE OF EVENTS FOR CASE G FLB

Downstream FLB initiated at 53% power
with 175 gpm AFW initiated 15 minutes after the break

Event	Time, sec
Feedline Break downstream of MFW check valves inside containment	10.
Pressurizer PORVs open (2200 psia)	76.
Reactor trip on high pressurizer pressure	84.
Rods begin to drop	86.
AFW starts - 175 gpm to 2 steam generators	910.
Heat removal of AFW is capable of removing core decay heat (T_{avg} begins to drop)	1270.
Cold AFW reaches 2 steam generators	1285.
Pressurizer PORVs close	1295.

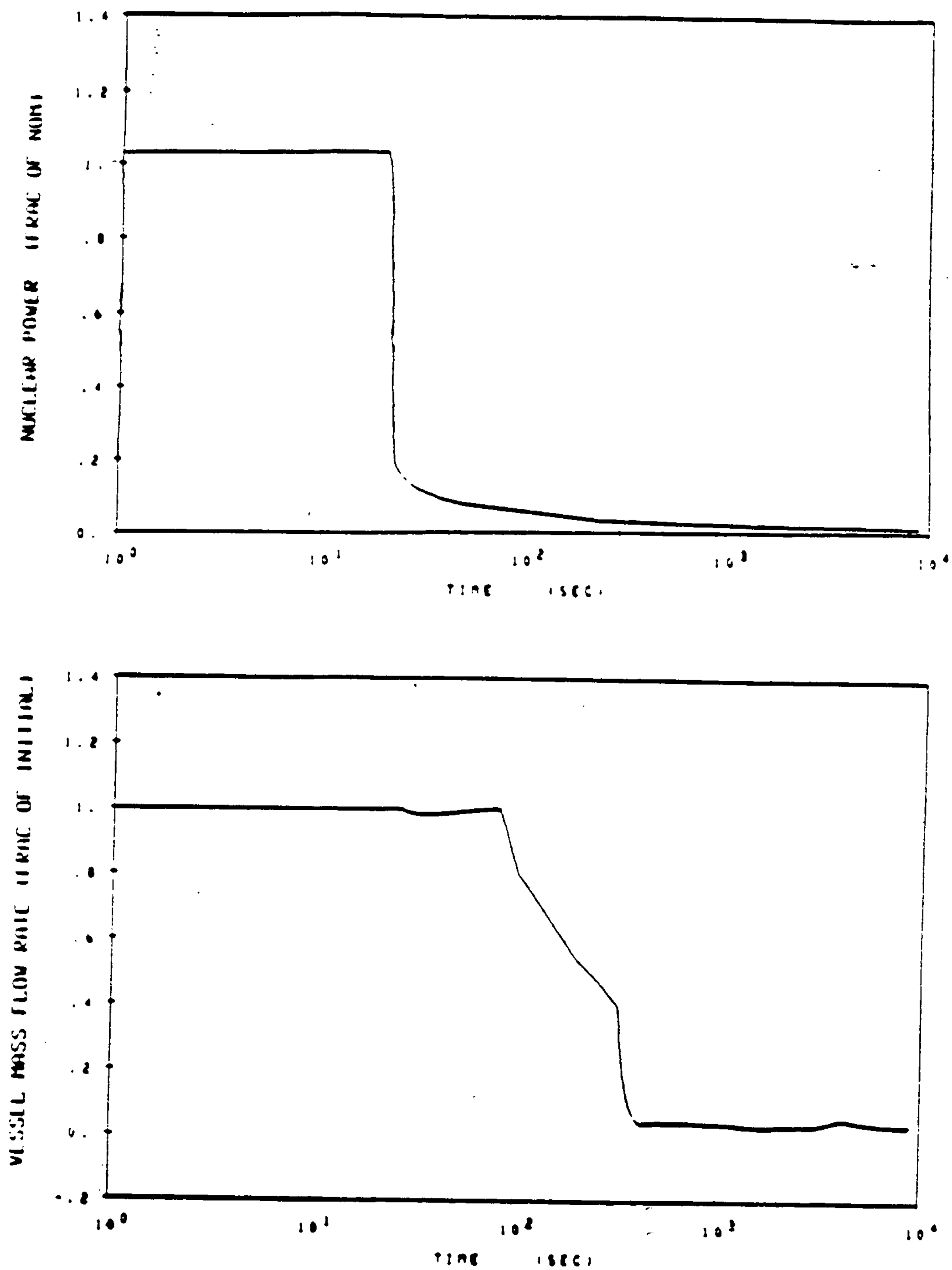


FIGURE 1 Nuclear Power and Vessel Flow vs. Time

Case D: SONGS 1 FLB Upstream of In-Containment Check Valves at 100% Power
AFW = 100 gpm to 2 steam generators at 30 minutes

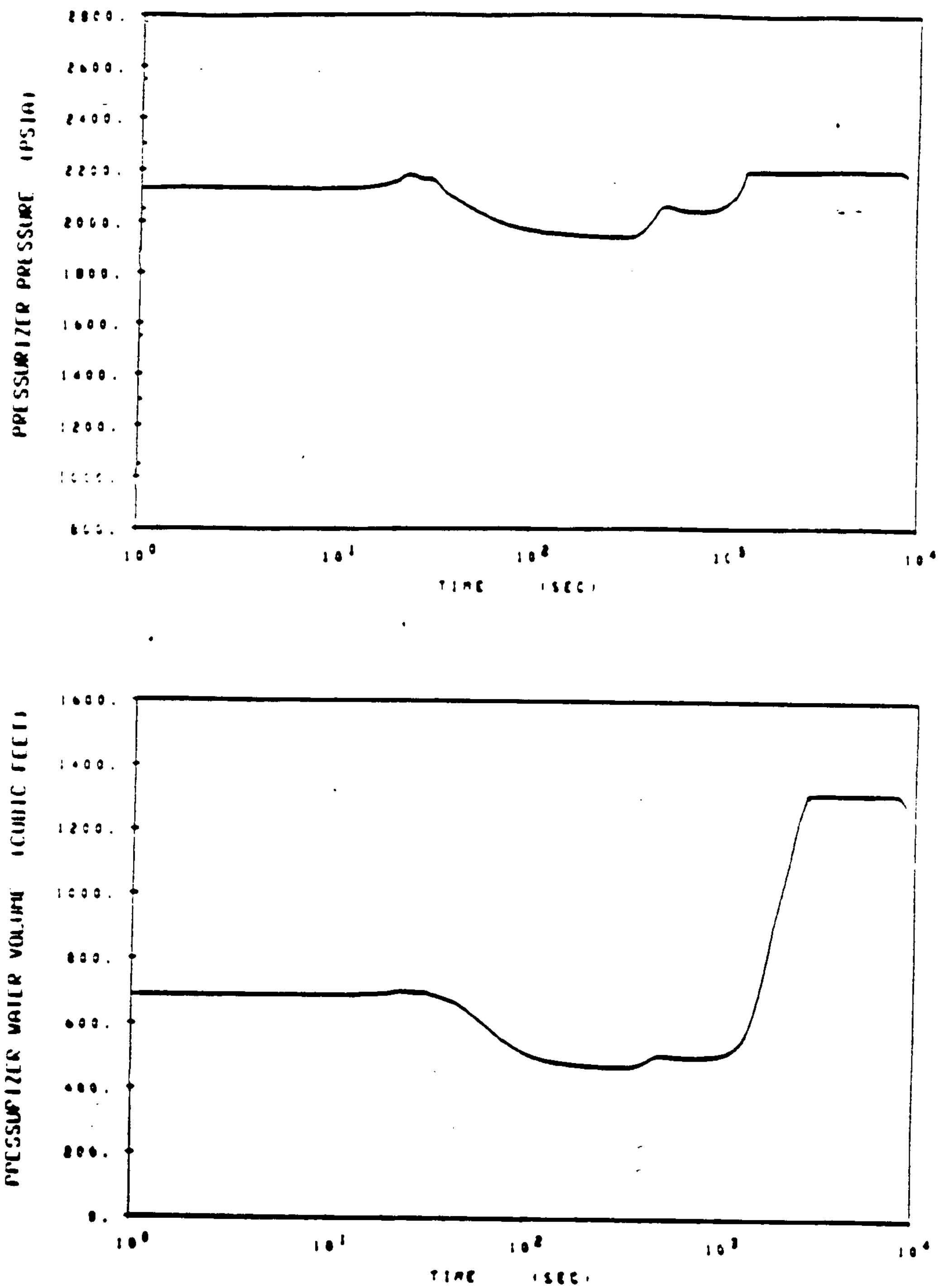


FIGURE 2 Pressurizer Pressure and Pressurizer Water Volume vs. Time

Case D: SONGS 1 FLB Upstream of In-Containment
Check Valves at 100% Power
AFW = 100 gpm to 2 steam generators at 30 minutes

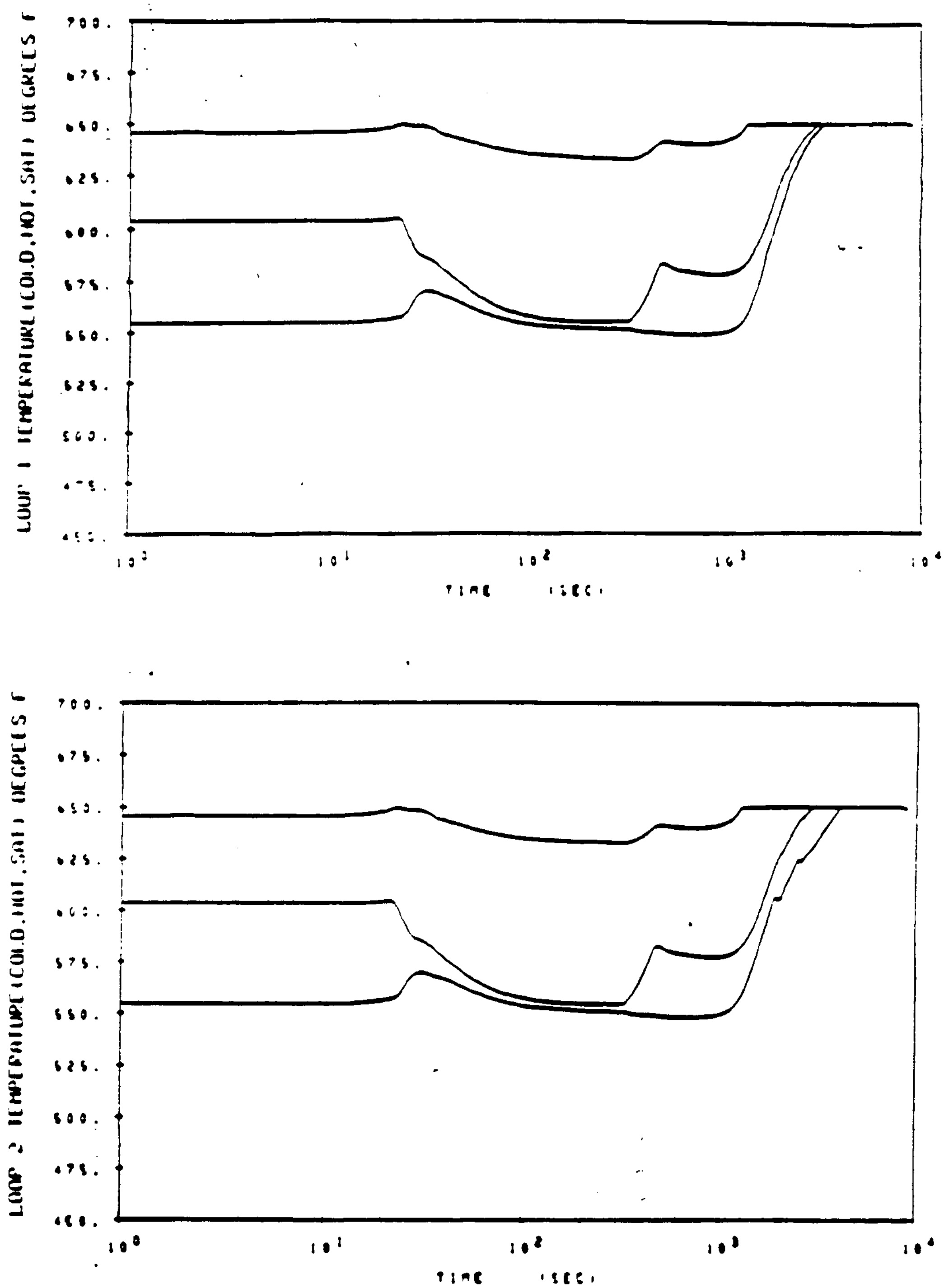


FIGURE 3 Faulted Loop Tsat, Thot, and Tcold vs. Time
Intact Loop Tsat, Thot, and Tcold vs. Time

Case D: SONGS 1 FLB Upstream of In-Containment Check Valves at 100% Power
AFW = 100 gpm to 2 steam generators at 30 minutes

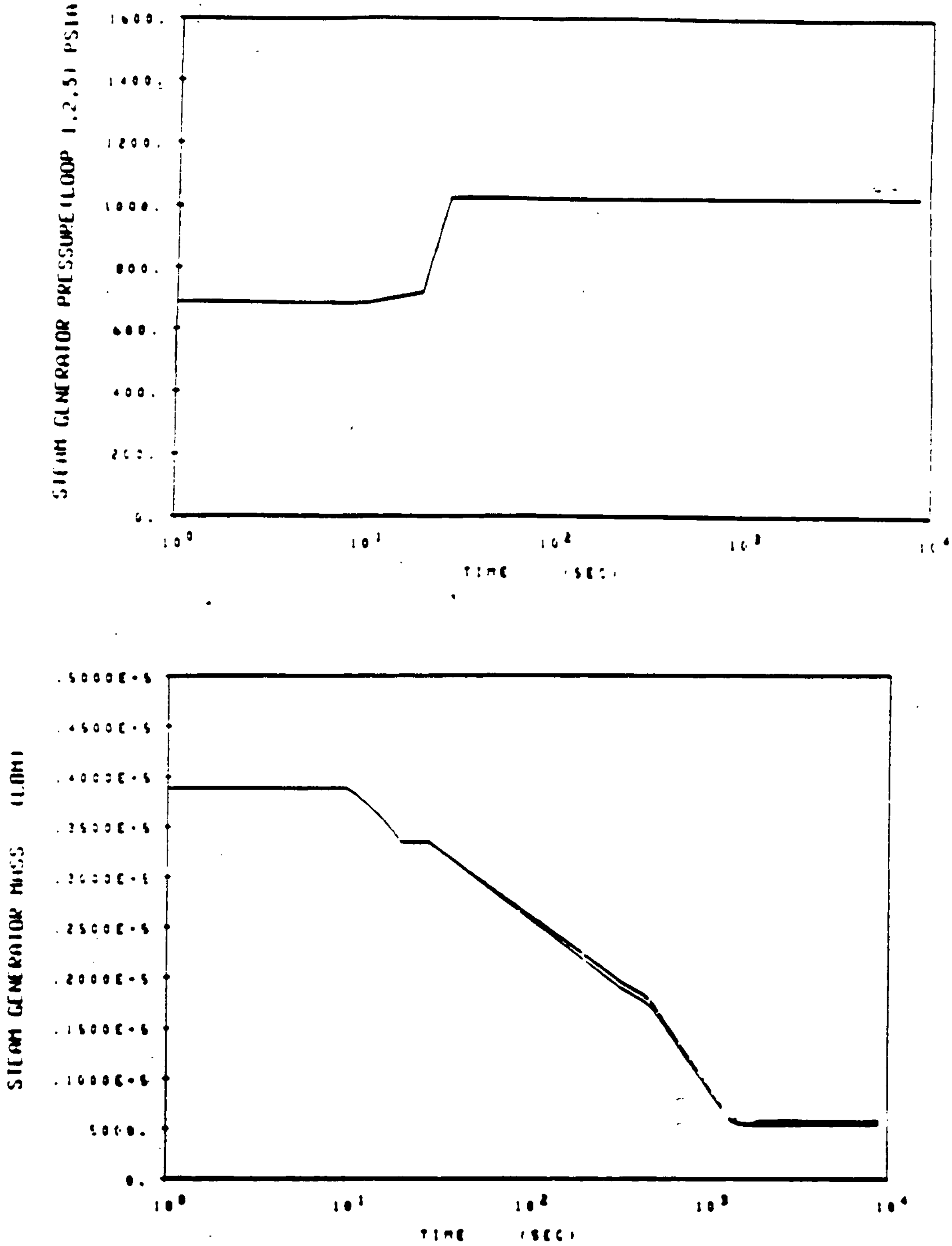


FIGURE 4 Steam Generator Pressure and Steam Generator Mass vs. Time

Case D: SONGS 1 FLB Upstream of In-Containment
Check Valves at 100% Power
AFW = 100 gpm to 2 steam generators at 30 minutes

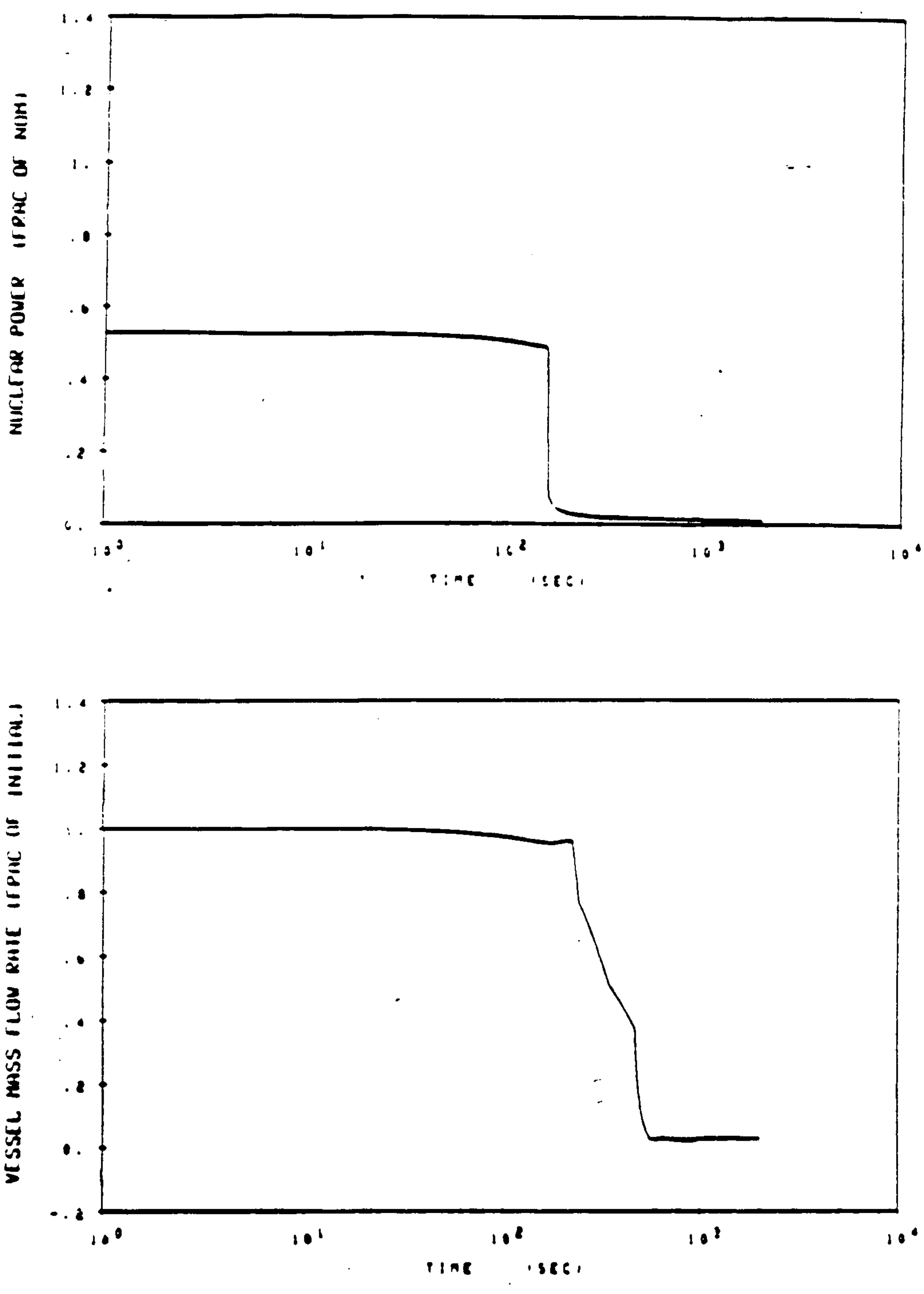
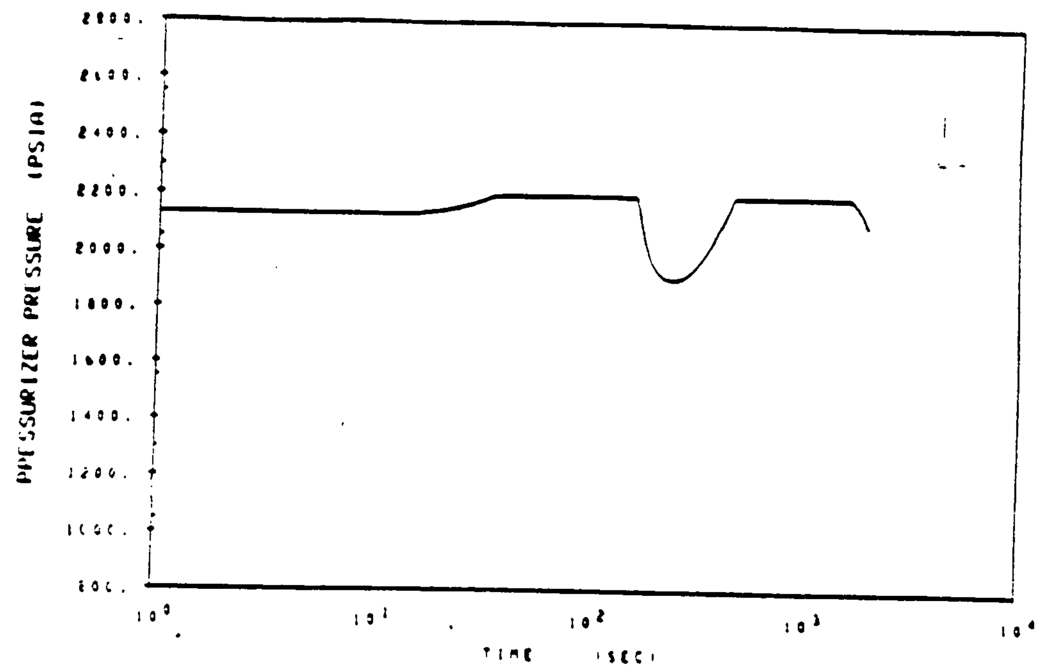


FIGURE 5 Nuclear Power and Vessel Flow vs. Time

Case E.: SONGS 1 FLB Upstream of In-Containment Check Valves at 50% Power
AFW = 100 gpm to 2 steam generators at 15 minutes



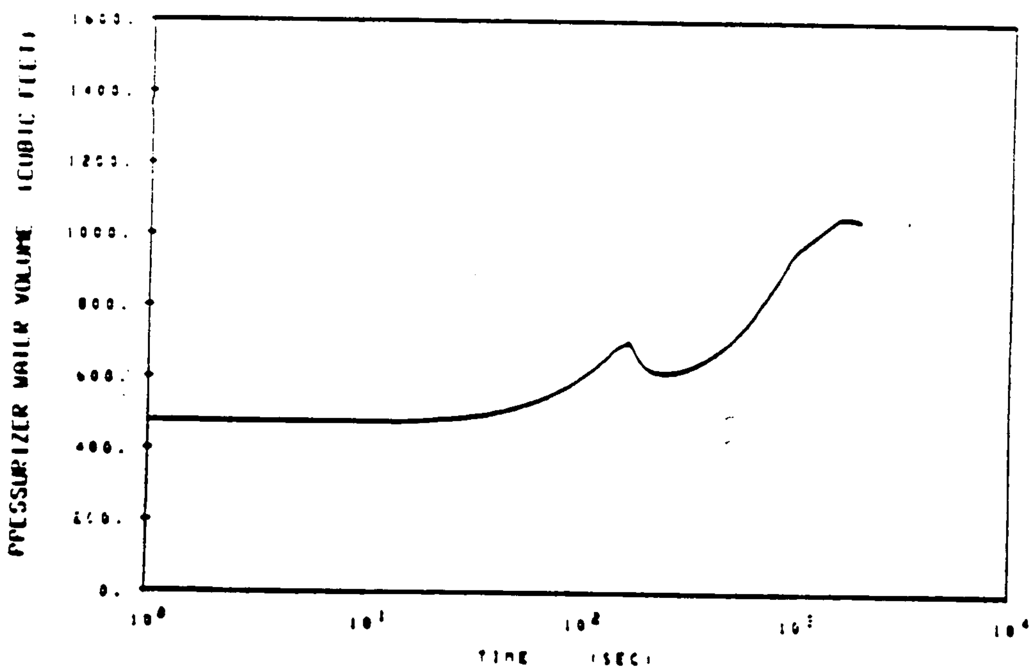
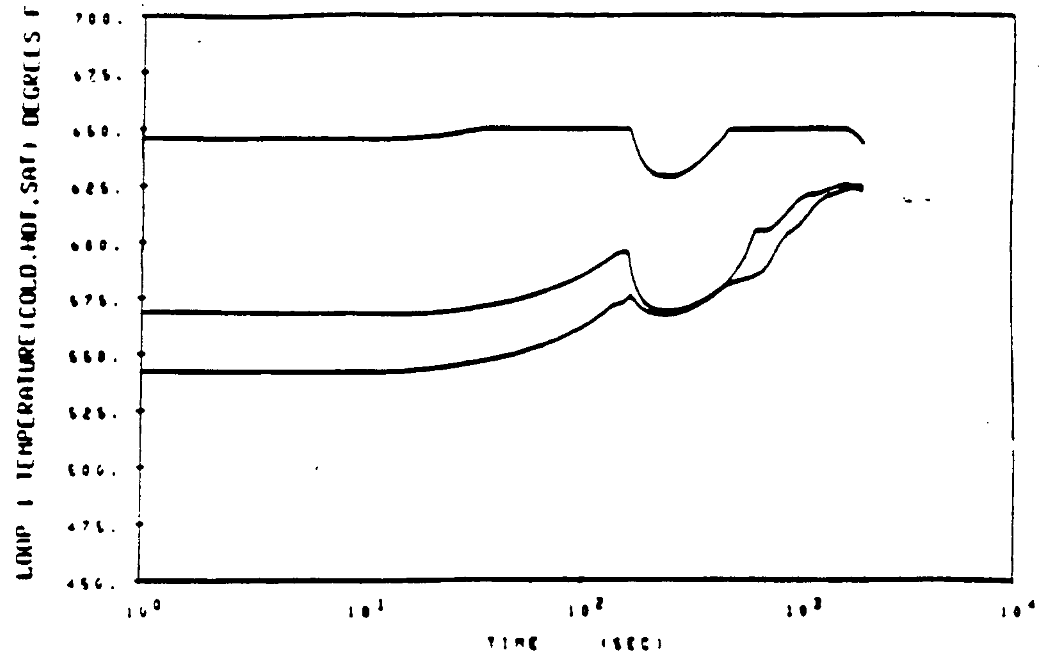


FIGURE 6 Pressurizer Pressure and Pressurizer Water Volume vs. Time

Case E: SONGS 1 FLB Upstream of In-Containment Check Valves at 50% Power
AFW = 100 gpm to 2 steam generators at 15 minutes



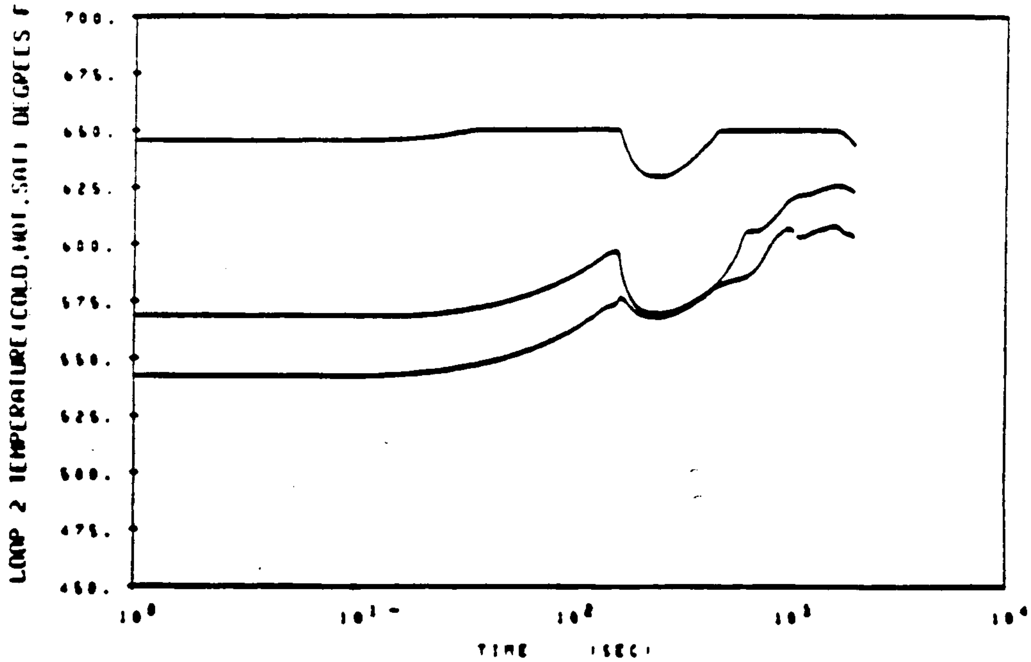


FIGURE 7 Faulted Loop Tsat, Thot, and Tcold vs. Time
Intact Loop Tsat, Thot, and Tcold vs. Time

Case E: SONGS 1 FLB Upstream of In-Containment Check Valves at 50% Power
AFW = 100 gpm to 2 steam generators at 15 minutes

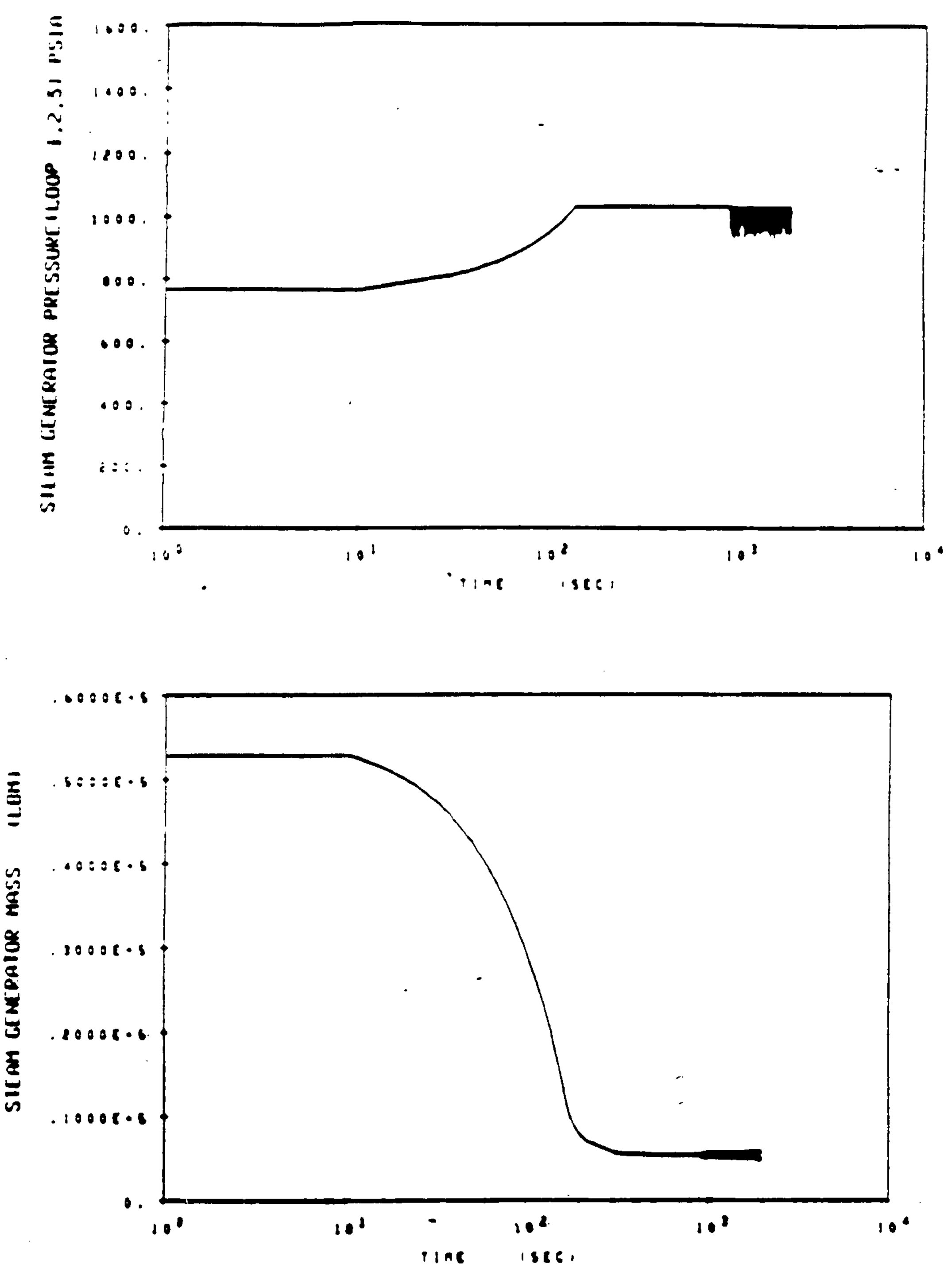
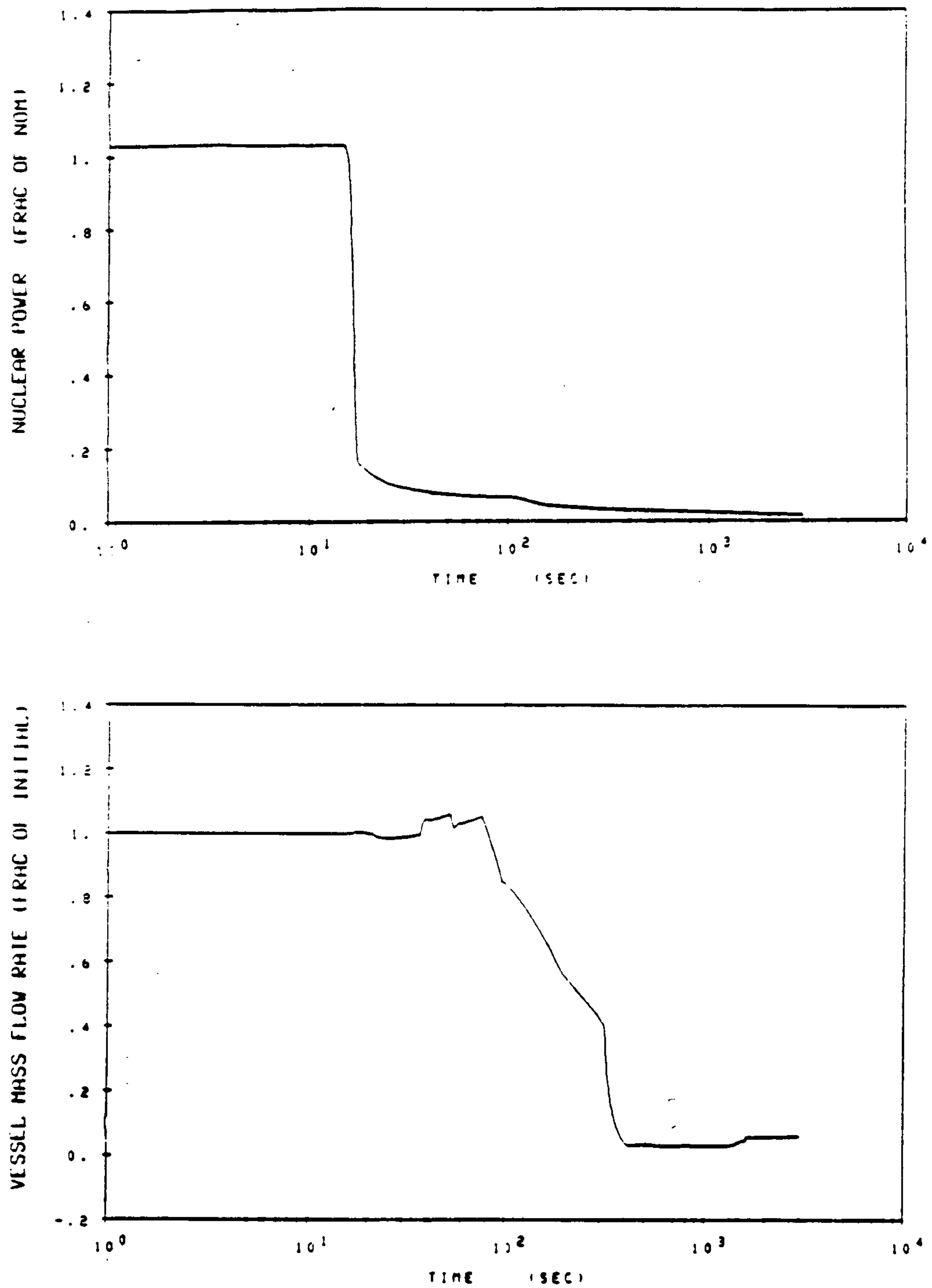
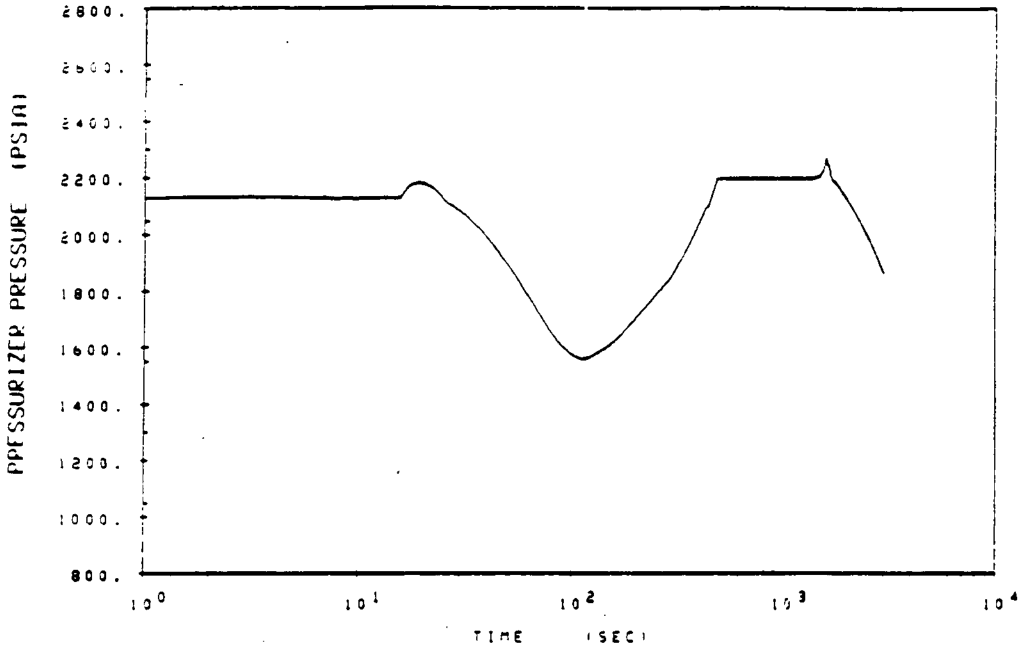


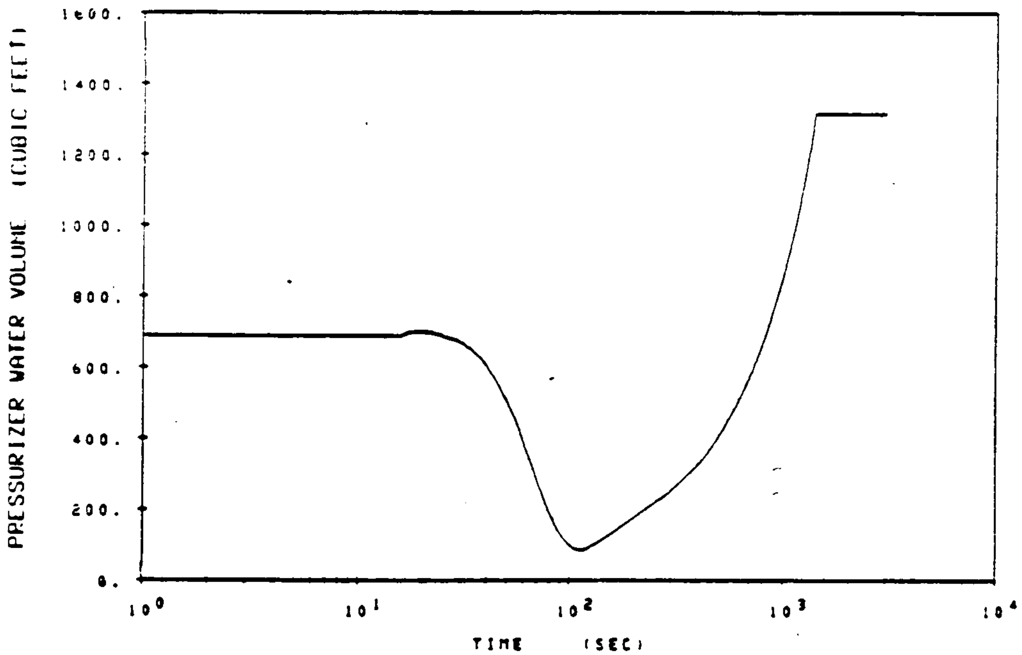
FIGURE 8 Steam Generator Pressure and Steam Generator Mass vs. Time

Case E: SONGS 1 FLB Upstream of In-Containment
Check Valves at 50% Power
AFW = 100 gpm to 2 steam generators at 15 minutes

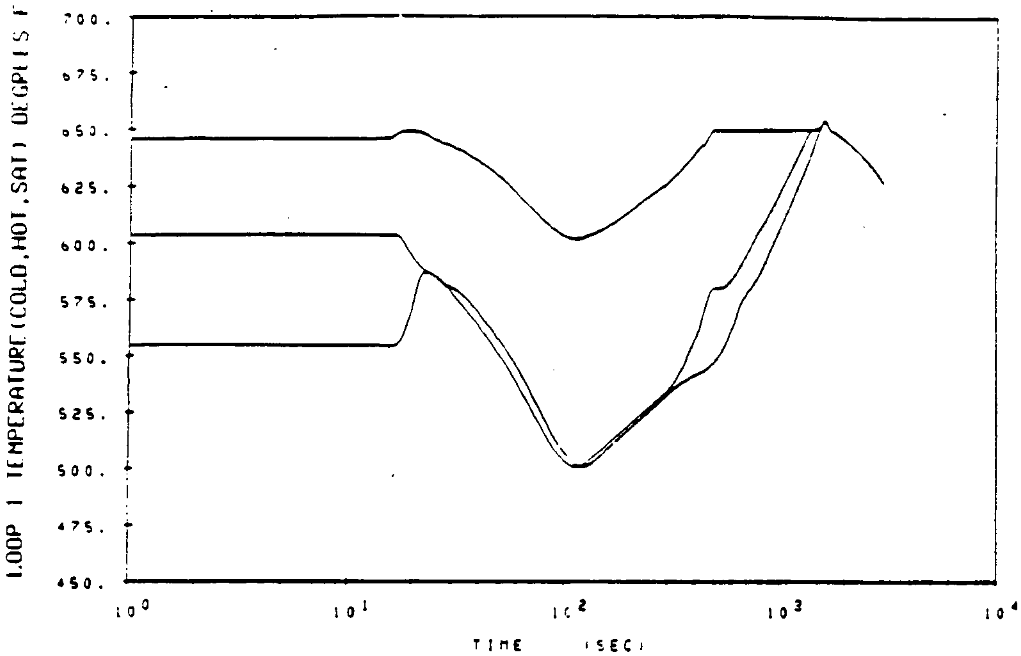


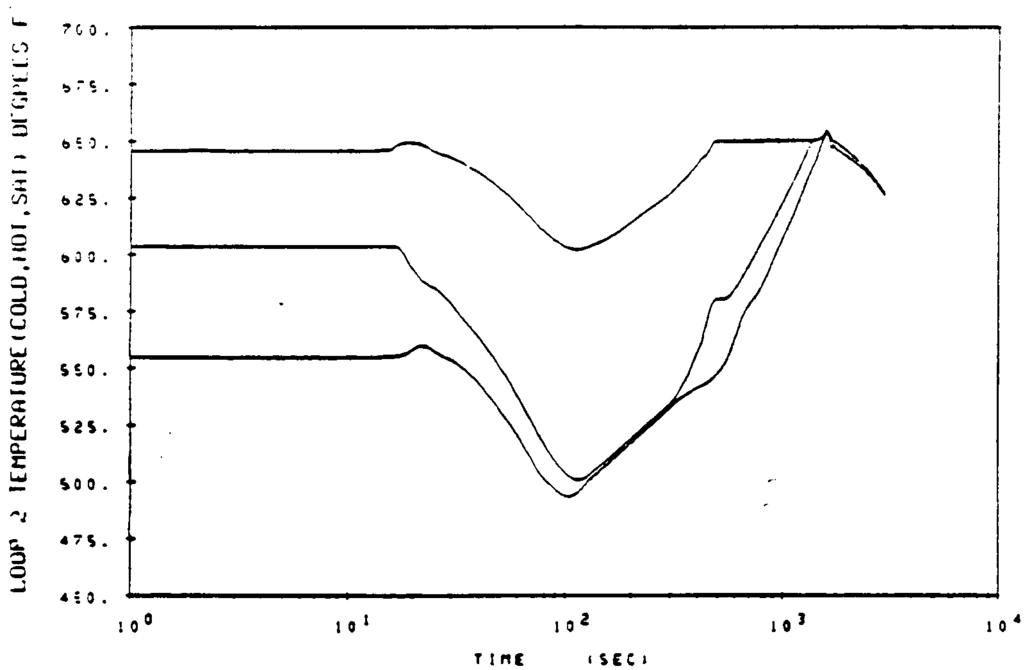
San Onofre Nuclear Generating Station Unit 1 - Feedline Break	
Case F - Downstream Break 103% power 175gpm AFW at 20 minutes	Figure 9 Nuclear Power & Vessel Flow vs. Time





San Onofre Nuclear Generating Station Unit 1 - Feedline Break	
Case F - Downstream Break 103% power 175gpm AFW at 20 minutes	Figure 10 Pressurizer Pressure & Water Volume vs. Time





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<tr><td>Case E - Downstream Break
103% power
175gpm AFW at 20 minutes</td><td>Figure 11
Faulted & Intact Loop Temperatures vs. Time</td></tr>
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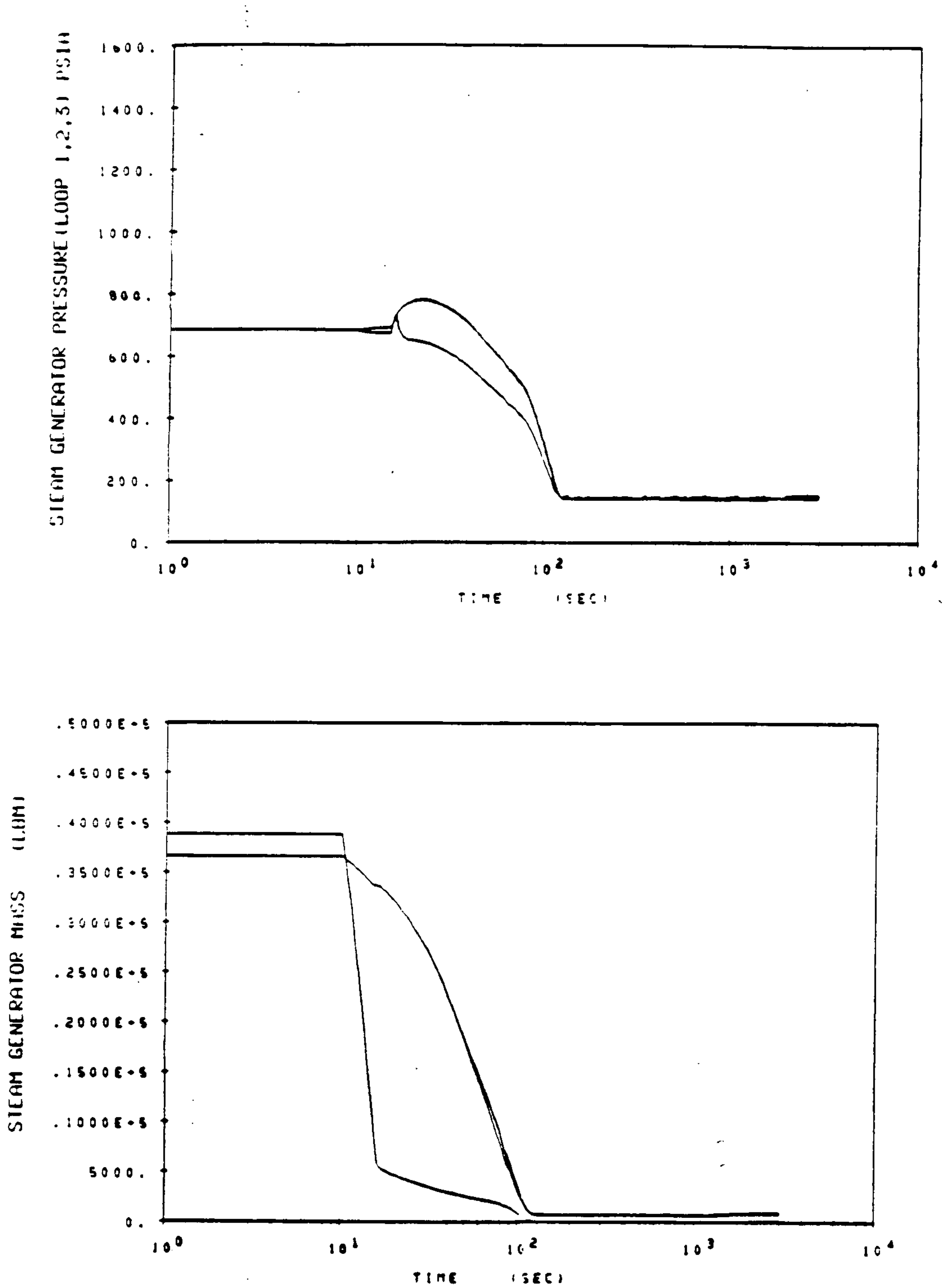

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<tr><td>Case F - Downstream Break
103% power
175gpm AFW at 20 minutes</td><td>Figure 12
Steam Generator Pressure & Mass vs. Time</td></tr>
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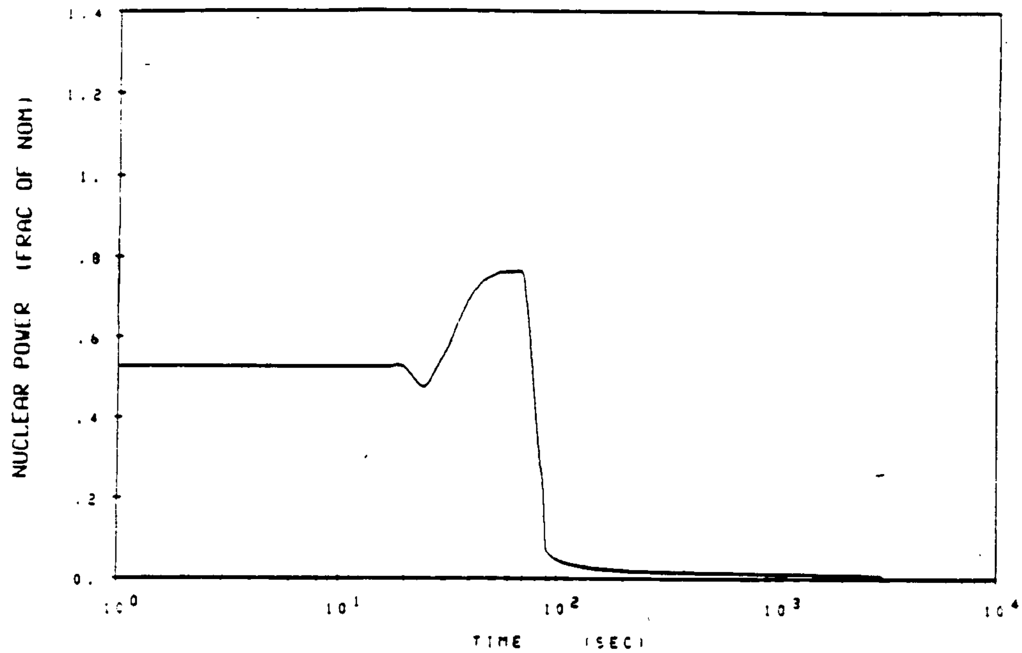
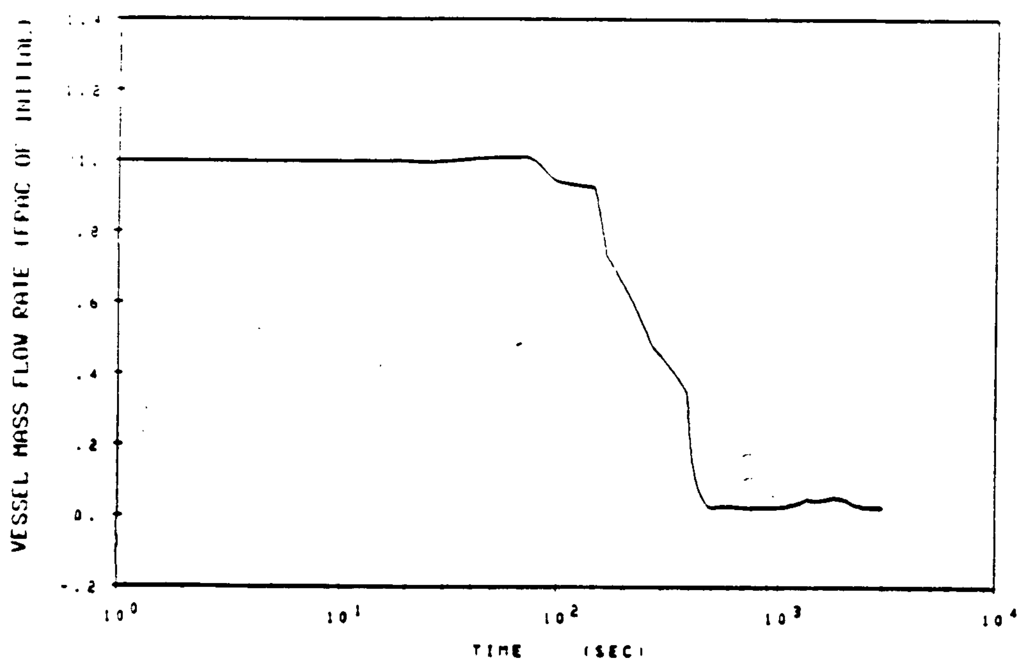
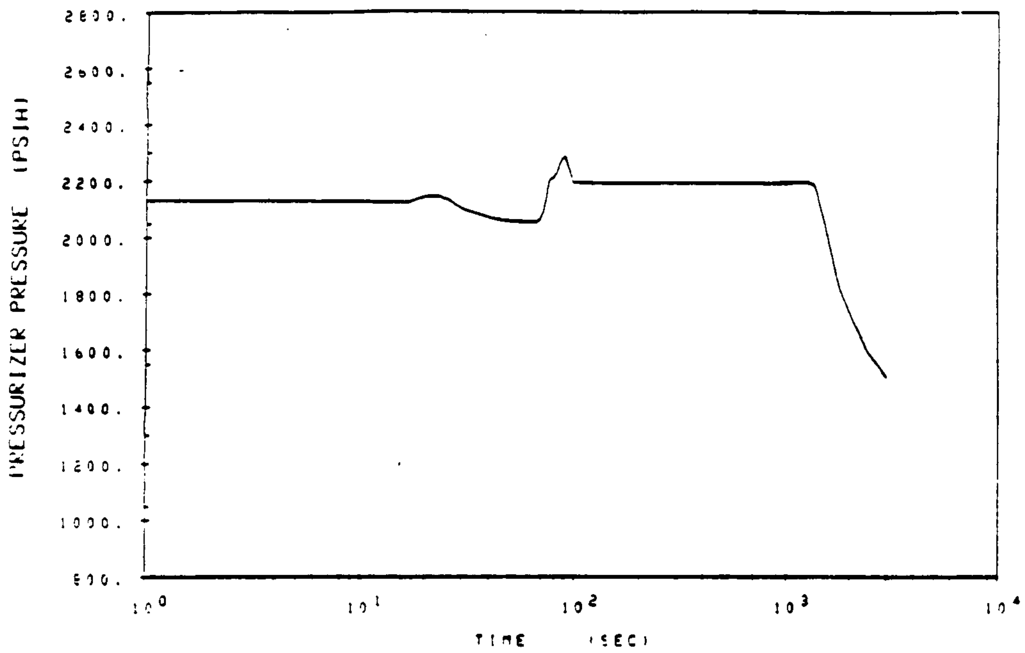
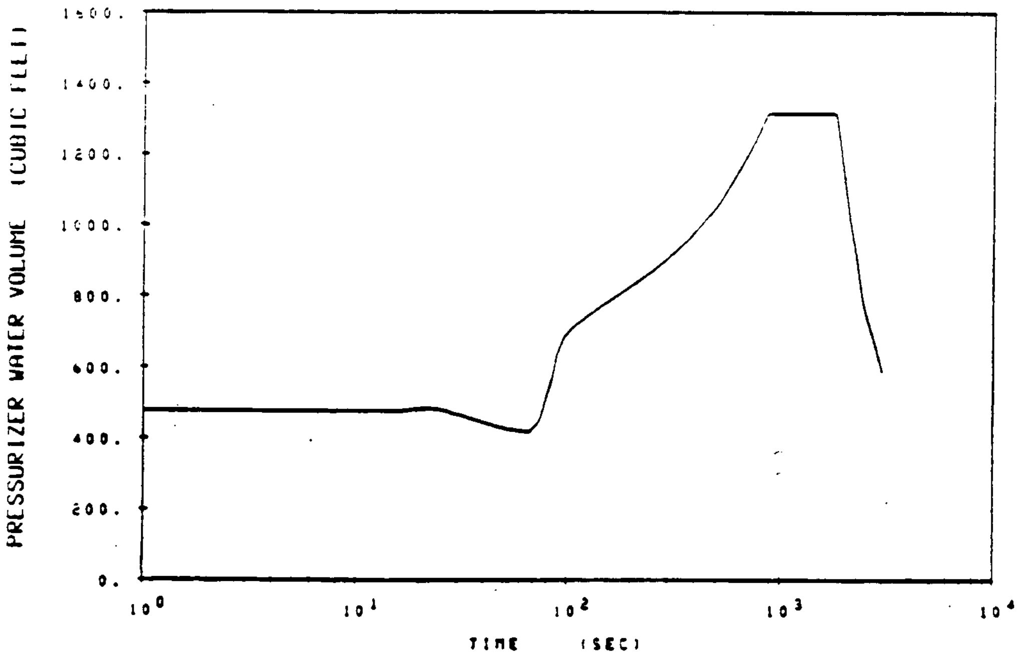


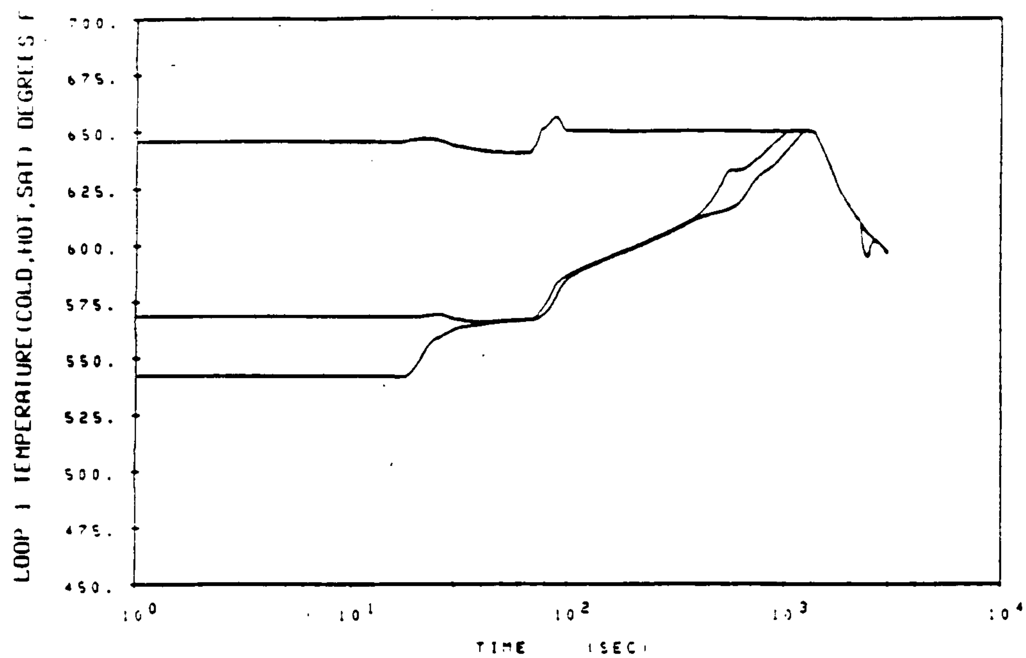


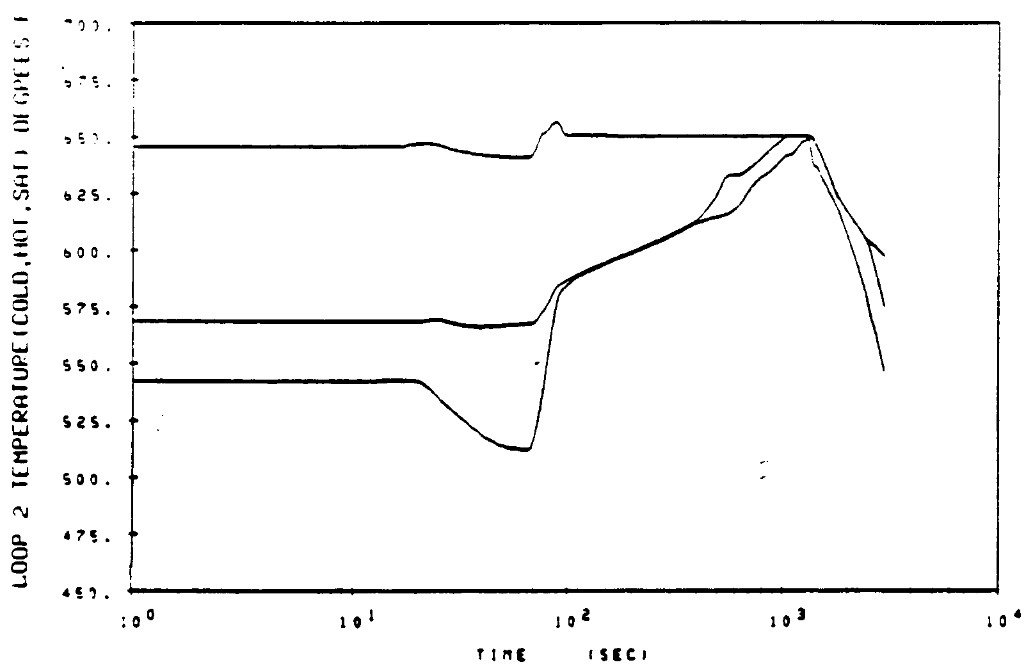
San Onofre Nuclear Generating Station Unit 1 - Feedline Break	
Case G - Downstream Break 53% power 175gpm AFW at 15 minutes	Figure 13 Nuclear Power & Vessel Flow vs. Time



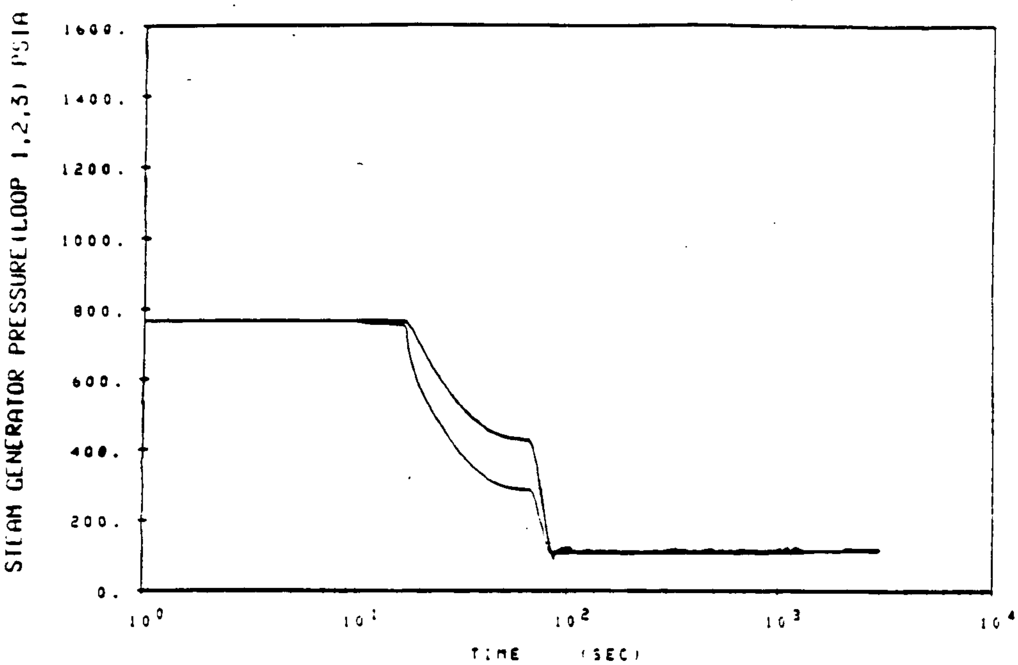


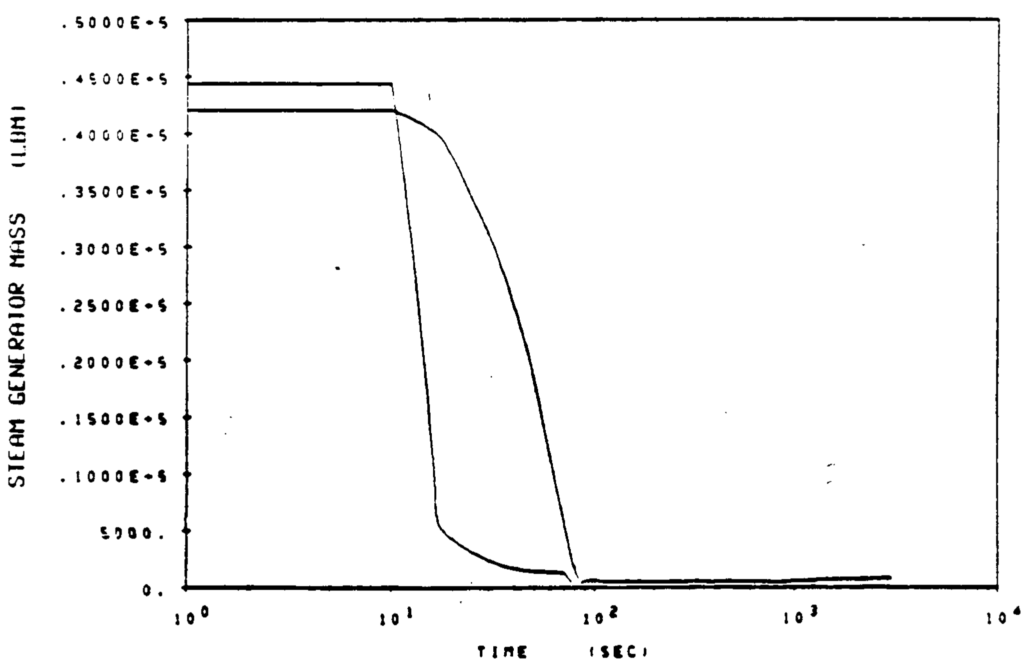
San Onofre Nuclear Generating Station Unit 1 - Feedline Break	
Case G - Downstream Break 53% power 175gpm AFW at 15 minutes	Figure 14 Pressurizer Pressure & Water Volume vs. Time





San Onofre Nuclear Generating Station Unit 1 - Feedline Break	
Case G - Downstream Break 53% power 175gpm AFW at 15 minutes	Figure 15 Faulted & Intact Loop Temperatures vs. Time





San Onofre Nuclear Generating Station Unit 1 - Feedline Break	
Case G - Downstream Break 53% power 175gpm AFW at 15 minutes	Figure 16 Steam Generator Pressure & Mass vs. Time

Attachment 4

AUXILIARY FEEDWATER SYSTEM
FLOW TEST
SAN ONOFRE UNIT 1

Attachment 4

Auxiliary Feedwater System
Flow test
San Onofre Unit 1

PURPOSE:

Testing will be performed in MODE 1 on the Auxiliary Feedwater System, Train A to provide flow rate data for the steam driven AFW pump, G-10, acting alone and in combination with the motor driven AFW pump, G-10S. The test will be performed to; 1), Verify that the system performance with AFW pump G-10 alone, and with pump G-10 plus G-10S, meets or exceeds the minimum analyzed flow requirements for these pump combinations; and 2), Verify the new reduced flow venturis deliver less than the design maximum 150 gpm to each steam generator. Analyses have been completed (Attachment 3) which show that acceptable AFW flowrates to the steam generators are achieved for all AFWS design basis events. MODE 5 testing of the Train A motor driven AFW pump, G-10S, and the Train B motor driven AFW pump, G-10W, will be completed prior to MODE 4 entry to verify that the auxiliary feedwater flow rates provided by these pumps and the modified piping system meets or exceed the analyzed flow requirements.

DESCRIPTION:

This test will be performed in MODE 1, with reactor power level at less than 25% of full power (nominally at 15% to 20% of full power). The control rods will be automatically controlled and the turbine generator will be on-line and connected to the switchyard. The main feedwater flow control valves are expected to be in automatic, but may be manually controlled. The 15% to 20% power level is optimal in that the reactor is in a stable condition, and the automatic control systems are adjusted to respond to plant perturbations resulting from the initiation, changes in , and termination of auxiliary feedwater flow. Since data for the Train A steam driven pump alone is required, the steam driven pump will be placed in automatic, and the motor driven pump will in the manual position. This will allow the steam driven pump to start when the AFWS is manually initiated from the control room. The motor driven pump will then be started by placing the pump control in automatic to obtain combined pump flow rates.

Train B of the AFWS will be in the automatic mode, enabling it to respond to a valid AFW actuation signal. If a valid AFW actuation occurs during the test, the Train B pump will start, interlocks will close the Train A discharge valves, and the breaker to the Train A motor driven pump will open, as designed. All other engineered safety systems will be in their normal, operable configurations during this test.

ENGINEERING EVALUATION:

The AFW system Train A test in MODE 1 below 25% power was assessed to determine the transient RCS response. Initial conditions include the plant at 25% power with the turbine generator on-line and connected to the switchyard, and the control rods in automatic. The MFW flow control valves would either be automatically or manually controlled. Per SONGS 1 Drawing No. 56793, Heat Balance Diagram at 112,548 KW Gross, MFW temperature would be about 308°F and flow rate would be approximately 3250 gpm. Startup of AFW Train A (AFW pumps G-10 and G-10S) would simultaneously increase feedwater flow and decrease feedwater temperature. The MFW flow control valves would be controlled to compensate for the increase in feedwater flow. Assuming MFW flow at approximately 3000 gpm and 308°F at 25% power, AFW flow at approximately 300 gpm; and 60°F, the effective (mixed) feedwater temperature is 283°F, or a reduction in feedwater enthalpy of 25 BTU/lb. A decrease in feedwater temperature causes a decrease in the temperature in the reactor coolant, resulting in an increase in reactor power due to the negative moderator temperature coefficient, and a decrease in the RCS and steam generator pressures. Without control system action, the reactor would reach equilibrium at a higher power level.

The consequences of the AFW system test are bounded by the excess feedwater event analyzed in the SONGS 1 UFSAR. The UFSAR feedwater event resulted in an RCS temperature cooldown rate of less than 1°F per minute. Based on a comparison of the feedwater enthalpy changes, the RCS cooldown rate due to the addition of AFW (25 BTU/lb) would be less than or equal to the UFSAR feedwater event cooldown rate . At an initial test power level less than 25%, a somewhat greater reduction in enthalpy would result, but the cooldown rate would remain bounded by the 1°F per minute UFSAR cooldown rate. The Unit 1 UFSAR, Section 7.4, excess feedwater event resulted in a power increase of less than 5% per minute assuming an end of life moderator temperature coefficient of -3.5×10^{-4} delta k/°F. At beginning of life, the moderator temperature coefficient is estimated at -1.0×10^{-4} delta k/°F, so that the power increase for the AFW test would be significantly less. Control rod motion would restore the primary average temperature. If the reactor control system were unable to maintain plant conditions within the protection limits during the accident, the overpower or variable low pressure protection (floor value) will cause a reactor trip. However, the power transients are expected to be very small, and a reactor trip is not anticipated. (RCS decrease for the UFSAR analysis is less than 20 psi.)

The acceptability of consequences for the limiting design basis accidents (feedline breaks, loss of normal feedwater) at a power level of 25% was also evaluated. AFW flow requirements for FWLB-

U and LONF events at 50% power are 100 gpm and 185 gpm respectively. The Train A minimum required AFW flow of 175 gpm for the FWLB-D will be verified with pump G-10S alone during testing to be completed in MODE 5. At 25% power, minimum AFW flow requirements would be less than at 50% power. These minimum AFW flow requirements are approximately 53% and 62% for LONF and FWLB-U respectively, of the predicted combined AFW flow from Train A AFW pumps G-10 and G-10S (i.e. 190 gpm total flow for FWLB-U and 300 gpm total flow for LONF). Hence AFW flow requirements for design basis events at 25% power are satisfied with approximately one-half the predicted AFW system flows.

CONCLUSION:

All safety systems, including the auxiliary feedwater system, will be operable during the performance of this test. As discussed in the Engineering Evaluation above, the transient system responses were evaluated and are bounded by analyses. The probability of a turbine trip is not increased as a result of this test. Manual control of main feedwater flow (based on steam generator level) at approximately 15% power or less is part of normal plant operation.