AP600 DESIGN CHANGE DESCRIPTION REPORT

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1.0 Introduction

This report provides a summary description of changes to the AP600 design described in the AP600 Standard Safety Analysis Report (SSAR), Revision 1 (Reference 7.1) and AP600 Probabilistic Risk Assessment (PRA), Revision 0 (Reference 7.2). The purposes of this document are to:

- Identify changes to the AP600 design.
- Provide an overview of the impacts resulting from the changes,
- Serve as interim documentation until licensing documentation is revised (e.g., SSAR and PRA).

This report cc[illegible] eleven design changes to the AP600.

Section 2 of this report provides a description of each of these changes, including the purpose of the change, impacts (if any) to safety analyses, PRA, SSAR documentation, and AP600 test program.

Section 3 of this report provides an evaluation of the cumulative impact of the automatic depressurization system (ADS) design changes on the AP600 PRA.

Section 4 of this report provides an evaluation of the cumulative impact of the design changes on the AP600 design basis safety analyses.

Section 5 of this report provides a discussion on ADS Systems tests to be performed at the VAPORE facility in Cassacia, Italy.

Section 6 of this report provides a discussion of ADS valve testing proposed for performance during the development, manufacture, installation, startup and operation of the ADS valves.

2.0 Description Of Design Changes

This section describes modifications to the AP600 design described in the AP600 Standard Safety Analysis Report (SSAR), Reference 7.1. Table 2-1 provides a summary of the design modifications described in this report. These modifications are individually discussed in the following subsections.

2.1 Passive Residual Heat Removal Heat Exchanger Actuation Logic (DM-01)

Description

This design modification involves a change to the actuation logic for the passive residual heat removal heat exchanger (PRHR HX). A signal to actuate the PRHR HX on a core makeup tank (CMT) actuation signal has been added. The signals that actuate the PRHR HX on a high pressurizer level or on a high steam generator level have been deleted. Table 2-2 shows the revised PRHR HX actuation signals.

Purpose of Modification

This design modification is introduced to reduce the potential for automatic depressurization system (ADS) actuation during a steam generator tube rupture (SGTR) event by increasing the margin between the minimum CMT level and the CMT level at which ADS actuation occurs.

This change, as well as changes DM-02 and DM-03 provide increased margin to pressurizer overfill during non-LOCA events on a best estimate basis. In the SSAR non-LOCA analysis, operator action is required to prevent the pressurizer from overfilling. The operator has more than one hour to take action. The combined impact of DM-01, DM-02 and DM-03 is that operator action is not required on a best estimate basis to prevent pressurizer overfill. Additional means to provide increased margin to overfill are currently being evaluated.

Safety Analysis Impact

One of the concerns for a SGTR event is the possibility of steam generator overfill. This overfill could potentially result in a significant increase in the offsite radiological consequences. Automatic protection and passive design features are incorporated into the AP600 to automatically terminate the break flow to prevent overfill during an SGTR. Actuation of the PRHR system, isolation of Chemical and Volume Control System (CVS) flow, and isolation of startup feedwater on high-2 steam generator (SG) level prevent overfill for an SGTR.

The design basis SGTR analysis presented in SSAR Section 15.6.3 includes an evaluation of potential for steam generator overfill. An analysis is performed to demonstrate that the AP600 design features prevent steam generator overfill, without taking credit for operator actions that can also prevent overfill. The limiting single failure for the overfill analysis is the failure of the startup feedwater control valve to throttle flow when nominal steam generator level is reached. Other conservative assumptions that maximize steam generator secondary volume (such as high initial steam generator secondary level, maximum initial RCS pressure, offsite power available, maximum CVS injected flow, maximum startup feedwater flow, and minimum startup feedwater delay time) are also assumed. The results of this analysis demonstrate that the AP600 protection system and passive system design features prevent steam generator overfill. The results also show that automatic actuation of the ADS does not occur. In these analyses the PRHR HX was actuated on a high SG level.

SGTR sensitivity studies performed subsequent to the SSAR analyses indicate that operator action to actuate the PRHR HX may be necessary to prevent ADS actuation in a case where CVS fails and the startup feedwater system (SFWS) works as designed (i.e., no CVS flow, SFWS throttles back to maintain steam generator level). In this case SG level does not increase to the high-2 level setpoint and actuate the PRHR HX.

PRHR HX actuation on CMT actuation addresses this event by actuating the PRHR HX independent of SG level. The high SG level actuation of the PRHR HX is no longer required because the CMTs and, therefore, the PRHR HX would already have been actuated. The high pressurizer level PRHR HX actuation signal is also not required because the CMTs and therefore the PRHR HX would have already been actuated.

The change to PRHR HX actuation logic also reduces the potential of pressurizer overfill following an inadvertent operation of the CMTs on a best estimate basis.

PRA Impact

The change to the PRHR HX actuation logic requires modification of the initiating signals in four of the PRHR fault trees (PRT, PRP, PRB, PRS).

The unavailability of the PRHR HX is dominated by failure of the control or the operator. As such, this change is not expected to have a significant impact on the PRA results.

SSAR Impact

This change impacts the following SSAR Chapters:

Chapter 6	Engineered Safety Features (see Appendix A-1 for markup of SSAR Section 6.3, Passive Core Cooling Systems)
Chapter 7	Instrumentation and Controls
Chapter 15	Accident Analyses
Chapter 16	Technical Specification

Test Program Impact

Core Makeup Tank Separate Effects Tests

This design change has no impact on these tests.

The CMT tests are separate effects tests designed to investigate thermal-hydraulic phenomena within the CMT over a wide range of pressures and temperatures. The PRHR actuation logic change affects event timing but not the nature of the phenomena investigated during these tests. This change does not affect the range of conditions required to be tested.

Integral Systems Tests at Oregon State University

This design change impacts control logic for these tests.

The OSU tests are integral systems tests to obtain thermal-hydraulic data for computer code validation and to investigate long tem cooling behavior. Modifications to the facility logic and control software must be performed to incorporate the specified change. The revised logic will be incorporated into the facility control systems at OSU. These changes will be implemented for all OSU matrix tests.

Integral Systems Tests at SPES-2 Test Facility

This design change impacts control logic for these tests.

The SPES-2 tests are integral systems tests to obtain thermal-hydraulic data for computer code validation and to investigate systems interactions during high pressure transients. The first matrix test at SPES-2, the 2 inch SBLOCA SSAR reference case, has been completed using the SSAR (Reference 7.1) actuation logic. The control logic change will be implemented for the remaining SPES-2 tests.

The results of this first matrix test are being analyzed with respect to the AP600 design changes now being implemented. In addition, pretest analyses of the remaining SPES-2 tests are being performed using the revised logic and setpoints.

ADS Phase B Tests

This design change will have no impact on the ADS Phase B tests.

The ADS Phase B tests are performed by depressurizing the test facility through the ADS valve piping package, downstream piping and sparger. The conditions of the test are determined by the facility supply tank initial conditions (supply tank pressure and temperature) and the position of the facility control valve, which determines the fluid conditions upstream of the ADS valve piping package (see Figure 5-1).

The PRHR actuation logic is not explicitly modeled in the test facility. Pre-test analysis will be performed for the facility to ensure that the test conditions during the facility blowdown are appropriate based on AP600 plant calculations. This will determine the test facility initial conditions and the position of the facility control valves during the test. This change does not affect the range of conditions required to be tested.

2.2 Pressurizer Heater Control Logic (DM-02)

Description

This design modification involves a change to the control logic for the pressurizer heaters. A change has been made to the pressurizer heater control logic to provide an automatic block of heater operation upon receipt of a CMT actuation signal. This logic will be implemented by taking isolated CMT actuation signals from the protection and monitoring system (PMS) to the plant control system (PLS). Redundancy is employed in the control system to trip redundant nonsafety-related breakers, three in series for each heater.

Purpose of Modification

This logic change provides a means for blocking the pressurizer heaters (in addition to a manual block) and reduces the potential for SG overfill and automatic ADS actuation for an SGTR accident. This logic change also reduces the potential for pressurizer safety valve actuation during Condition 2 events.

Safety Analysis Impact

The SSAR analysis (Reference 7.1) does not include continued pressurizer heater operation. Long term operation of the pressurizer heaters during certain non-LOCA events acts to retard the depressurization of the RCS following actuation of the CMTs. Without the automatic pressurizer heater block, it may be necessary for the operators to manually open the pressurizer heater breakers to prevent SG overfill or automatic ADS actuation for the design basis SGTR event. Continued operation of the pressurizer heaters does not adversely affect the response of the plant for several thousand seconds. As a result, the safety case is based on the operators opening breakers locally. Therefore, this control logic can be nonsafety-related since its function is to reduce unnecessary demands on the operator.

PRA Impact

This change does not affect the systems modeled in the PRA or their success criteria.

SSAR Impact

This change impacts the following SSAR Chapters:

Chapter 5	Reactor Coolant System and Connected Systems
Chapter 7	Instrumentation and Controls
Chapter 15	Accident Analyses

Test Program Impact

Core Makeup Tank Separate Effects Tests

This design change has no impact on these tests.

The CMT tests are separate effects tests designed to investigate thermal-hydraulic phenomena within the CMT over a wide range of pressures and temperatures. The pressurizer heater control logic change may affect event timings but not the nature of the phenomena investigated in this test facility. This change does not affect the range of conditions required to be tested.

Integral Systems Tests at Oregon State University

This design change impacts control logic for these tests.

The OSU tests are integral systems tests to obtain thermal-hydraulic data for computer code validation and to investigate long term cooling behavior. Modifications to the facility logic and control software must be performed to incorporate the specified change. The revised logic will be incorporated into the facility control systems at OSU. These changes will be implemented for all OSU matrix tests.

Integral Systems Tests at SPES-2 Test Facility

This design change impacts control logic for these tests.

The SPES-2 tests are integral systems tests to obtain thermal-hydraulic data for computer code validation and to investigate systems interactions during high pressure transients. The first matrix test at SPES-2, the 2-inch SBLOCA SSAR reference case, has been completed using the SSAR (Reference 7.1) actuation logic. The control logic change will be implemented for the remaining SPES-2 tests.

The results of this first matrix test are being analyzed with respect to the AP600 design changes now being implemented. In addition, pretest analyses of the remaining SPES-2 tests are being performed using the revised logic and setpoints.

ADS Phase B Tests

This design change will have no impact on the ADS Phase B test conditions.

The ADS Phase B tests are performed by depressurizing the test facility through the ADS valve piping package, downstream piping and sparger. The conditions of the test are determined by the facility supply tank initial conditions (supply tank pressure and temperature) and the position of the facility control valve, which determines the fluid conditions upstream of the ADS valve piping package (see Figure 5-1).

The pressurizer heater control logic is not explicitly modeled in the test facility. Pre-test analysis will be performed for the facility to ensure that the test conditions during the facility blowdown are appropriate based on the AP600 plant calculations. This will determine the test facility initial conditions and the position of the facility control valves during the test. This change does not affect the range of conditions required to be tested.

2.3 CVS Makeup Control Logic (DM-03)

Description

This design modification involves a change to the CVS control logic so that on a CMT actuation signal one CVS makeup is started on a 10 percent pressurizer level. That pump will automatically stop on a 20 percent pressurizer level. The control logic as described in the AP600 SSAR (Reference 7.1) calls for both CVS pumps to start on a CMT actuation signal. See Table 2-3 for a description of the CVS makeup control logic change. This change makes the control logic following CMT actuation similar to normal operation except that following CMT actuation the level setpoints for starting and stopping the CVS injection are changed.

Purpose of Modification

The change improves the expected operation of the plant during non-LOCA events. This change maintains the function to provide automatic CVS makeup to minimize the potential of automatic actuation of ADS. In addition, it reduces the chance of lifting a pressurizer safety valve or of overfilling the pressurizer for non-LOCA events. The operators are able to start the second CVS pump in case the CMT draindown is approaching ADS. Higher pressurizer level control setpoints will be used at zero power when the CMTs have not been actuated, such that operation of CVS makeup will not occur with the CMTs operating unless the pressurizer level drops to 10 percent. Note that since the CVS is a nonsafety-related system, its control logic is provided by the PLS. This change does not affect the acceptability of the Chapter 15 analyses.

Safety Analysis Impact

The AP600 safety analyses do not credit operation of the CVS. CVS operation is only assumed when such operation results in a more limiting transient. This change does not impact the AP600 design basis safety analyses presented in Chapter 15 of the SSAR.

PRA Impact

This change does not affect how the CVS is modeled in the PRA.

SSAR Impact

This change impacts the following SSAR Chapters:

Chapter 7	Instrumentation and Controls
Chapter 9	Auxiliary Systems

Test Program Impact

Core Makeup Tank Separate Effects Tests

This design change has no impact on these tests.

The CMT tests are separate effects tests designed to investigate thermal-hydraulic phenomena within the CMT over a wide range of pressures and temperatures. The CVS makeup control logic change does not affect the range of conditions required to be tested.

Integral Systems Tests at Oregon State University

This design change impacts control logic for these tests.

The OSU tests are integral systems tests to obtain thermal-hydraulic data for computer code validation and to investigate long term cooling behavior. Modifications to the facility logic and control software must be performed to incorporate the specified change. The revised logic will be incorporated into the facility control systems at OSU. These changes will be implemented for all OSU matrix tests.

Integral Systems Tests at SPES-2 Test Facility

This design change impacts control logic for these tests.

The SPES-2 tests are integral systems tests to obtain thermal-hydraulic data for computer code validation and to investigate systems interactions during high pressure transients. The first matrix test at SPES-2, the 2-inch SBLOCA SSAR reference case, has been completed using the SSAR (Reference 7.1) actuation logic. The control logic change will be implemented for the remaining SPES-2 tests.

The results of this first matrix test are being analyzed with respect to the AP600 design changes now being implemented. In addition, pretest analyses of the remaining SPES-2 tests are being performed using the revised logic and setpoints.

ADS Phase B Tests

This design change will have no impact on the ADS Phase B test conditions.

The ADS Phase B tests are performed by depressurizing the test facility through the ADS valve piping package, downstream piping and sparger. The conditions of the tests are determined by the facility supply tank initial conditions (supply tank pressure and temperature) and the position of the facility control valve, which determines the fluid conditions upstream of the ADS valve piping package (see Figure 5-1).

The CVS makeup control logic is not explicitly modeled in the test facility. Pre-test analysis will be performed for the facility to ensure that the test conditions during the facility blowdown are appropriate based on the AP600 plant calculations. This will determine the test facility initial conditions and the position of the facility control valves during the test. This change does not affect the range of conditions required to be tested.

2.4 ADS Stage 1 Actuation Setpoint (DM-04)

Description
This design modification involves a change to the CMT level setpoint that actuates ADS Stage 1. This setpoint has been changed from 1500 ft^3 to 1350 ft^3.

Purpose of Modification
This modification increases the margin to automatic ADS actuation. Sensitivity studies performed for small steam line breaks and SGTR events show that a minimum CMT volume of approximately 1590 ft^3 occurs under conditions of minimum CMT steam condensation. Additional margin to automatic ADS actuation is desirable. This margin is obtained by reducing the CMT level setpoint for ADS actuation to 1350 ft^3.

Safety Analysis Impact
Sensitivity studies were performed to verify that lowering this setpoint did not adversely affect LOCAs. The direct vessel injection (DVI) line break analysis, shown in Section 4.3, uses this setpoint and results in improved performance.

PRA Impact
This change does not affect the success criteria used in the PRA.

SSAR Impact
This change impacts the following SSAR Chapters:

Chapter 6	Engineered Safety Features (see Appendix A-1)
Chapter 7	Instrumentation and Controls
Chapter 15	Accident Analyses
Chapter 16	Technical Specifications

Test Program Impact

Core Makeup Tank Separate Effects Tests

This design change has minimal impact on these tests.

The CMT tests are separate effects tests designed to investigate thermal-hydraulic phenomena within the CMT over a wide range of pressures and temperatures. The ADS Stage 1 setpoint change may affect event timings but not the nature of the phenomena investigated in this test facility. This change does not affect the range of conditions required to be tested.

Integral Systems Tests at Oregon State University

This design change impacts control logic for these tests.

The OSU tests are integral systems tests to obtain thermal-hydraulic data for computer code validation and to investigate long tern cooling behavior. Modifications to the facility control software must be performed to incorporate the specified change. This change will be implemented for all OSU matrix tests.

Integral Systems Tests at SPES-2 Test Facility

This design change impacts control logic for these tests.

The SPES-2 tests are integral systems tests to obtain thermal-hydraulic data for computer code validation and to investigate systems interactions during high pressure transients. The first matrix test at SPES-2, the 2-inch SBLOCA SSAR reference case, has been completed using the SSAR (Reference 7.1) actuation logic. The control logic change will be implemented for the remaining SPES-2 tests.

The results of this first matrix test are being analyzed with respect to the AP600 design changes now being implemented. In addition, pretest analysis of the remaining SPES-2 tests are being performed using the revised setpoint.

ADS Phase B Tests

This design change will have no impact on ADS Phase B test conditions.

The ADS Phase B tests are performed by depressurizing the test facility through the ADS valve piping package, downstream piping and sparger. The conditions of the tests are determined by the facility supply tank initial conditions (supply tank pressure and temperature) and the position of the facility control valve, which determines the fluid conditions upstream of the ADS valve piping package (see Figure 5-1).

The ADS Stage 1 setpoint is not explicitly modeled in the test facility. Pre-test analyses will be performed for the facility to ensure that the test conditions during the facility blowdown are appropriate based on the AP600 plant calculations. This will determine the test facility initial conditions and the position of the facility control valves during the test.

2.5 ADS Second and Third Stage Actuation Logic (DM-05)

Description
This design modification involves a change to the actuation logic for ADS Stages 2 and 3. The actuation of ADS Stages 2 and 3 has been changed so that Stages 2 and 3 actuate on a time delay after actuation of the previous stage. The actuation logic of ADS Stages 1 and 4 remains based on CMT level.

Purpose of Modification
This change removes dependence on the CMT level measurement during the most rapid phase of RCS / CMT pressure reduction. This removes the reliance on the heated junction resistance temperature detector (RTD) level instruments during this phase of operation. When the CMT level approaches the fourth stage level setpoint, the RCS pressure is nearly constant and CMT level measurement is less challenging.

The specific timer settings are shown in Table 2-4.

Safety Analysis Impact
These setpoints are included in the DVI break and spurious ADS analysis shown in Section 4.3. In both accidents this logic has resulted in similar or improved overall plant performance.

PRA Impact
Actuation of ADS Stages 2 and 3 on timers, instead of level, will not affect the ADS reliability.

SSAR Impact
This change impacts the following SSAR Chapters:

Chapter 6	Engineered Safety Features (See Appendix A-1)
Chapter 7	Instrumentation and Controls
Chapter 15	Accident Analyses
Chapter 16	Technical Specifications

Test Program Impact

Core Makeup Tank Separate Effects Tests

This design change has minimal impact on these tests.

The CMT tests are separate effects tests designed to investigate thermal-hydraulic phenomena within the CMT over a wide range of pressures and temperatures. The changes to the second and third stage ADS setpoints may affect event timings but not the nature of the phenomena investigated in this test facility. This change does not affect the range of conditions required to be tested.

Integral Systems Tests at Oregon State University

This design change impacts control logic for these tests.

The OSU tests are integral systems tests to obtain thermal-hydraulic data for computer code validation and to investigate long term cooling behavior. Modifications to the facility control software must be performed to incorporate the specified change. This change will be implemented for all OSU matrix tests.

Integral Systems Tests at SPES-2 Test Facility

This design change impacts control logic for these tests.

The SPES-2 tests are integral systems tests to obtain thermal-hydraulic data for computer code validation and to investigate systems interactions during high pressure transients. The first matrix test at SPES-2, the 2-inch SBLOCA SSAR reference case, has been completed using the SSAR (Reference 7.1) actuation logic. The control logic change will be implemented for the remaining SPES-2 tests.

The results of this first matrix test are being analyzed with respect to the AP600 design changes now being implemented. In addition, pretest analyses of the remaining SPES-2 tests are being performed using the revised setpoints.

ADS Phase B Tests

This design change will have minimal impact on ADS Phase B test conditions.

The ADS Phase B tests are performed by depressurizing the test facility through the ADS valve piping package, downstream piping and sparger. The conditions of the tests are determined by the facility supply tank initial conditions (supply tank pressure and temperature) and the position of the facility control valve, which determines the fluid conditions upstream of the ADS valve piping package (see Figure 5-1).

The second and third stage ADS setpoints are not explicitly modeled in the test facility. Pre-test analysis will be performed for the facility to ensure that the test conditions during the facility blowdown are appropriate based on the AP600 plant calculations. This will determine the test facility initial conditions and the position of the facility control valves during the test. This change does not affect the range of conditions required to be tested.

2.6 Core Makeup Tank Inlet Diffuser (DM-06)

Description
This design modification involves the addition of a diffuser to the inlet of each of the CMTs. The diffuser is an extension of the inlet pipe with the same size and schedule. The bottom of the diffuser is plugged and holes are drilled into the side. This design forces the incoming steam flow to turn 90 degrees which effectively reduces the steam penetration into the CMT. Figure 2-2 shows the diffuser and describes the arrangement of the holes.

Purpose of Modification
Preliminary tests at the AP600 CMT Test Facility (Waltz Mill Test Site) have shown that rapid steam condensation can occur in the CMT during some modes of operation. The mode of operation that is affected is when the CMT contains cold water and the cold leg balance line supplies steam to the CMT. This mode only occurs during medium to large LOCAs. The impact of this rapid steam condensation is a period of reduced CMT injection flow and additional piping loads.

Safety Analysis Impact
Additional testing at the CMT test facility with a prototypic steam diffuser shows that a CMT inlet diffuser greatly reduces pressure fluctuations. The period of reduced CMT injection is also reduced. LOCA analysis has been performed to assess the sensitivity to this reduced injection. A DVI LOCA was analyzed with different volumes of the CMT heating up before drain down began. These volumes bound the maximum mixing volume. With a larger mixing volume, the calculations result in a lower RCS inventory. However, with the addition of the DVI venturi the DVI LOCA results were better than those contained in the SSAR; there is now no core uncovery. Refer to the LOCA analysis contained in Section 4.3.

PRA Impact
This change will have no impact on the PRA since it does not affect how the systems are modeled and has no effect on success criteria.

SSAR Impact
This change impacts the following SSAR Chapters:

Chapter 5	Reactor Coolant System and Connected Systems
Chapter 6	Engineered Safety Features
Chapter 15	Accident Analyses

Test Program Impact

Core Makeup Tank Separate Effects Tests
Integral Systems Tests at Oregon State University
Integral Systems Tests at SPES-2 Test Facility

This design change will have no impact on the above tests.

A CMT inlet diffuser has already been incorporated into each of these AP600 tests facilities. The rationale for sizing the diffuser at each facility was based on the functional requirement to prevent rapid steam condensation in the CMT during periods of high steam flow into cold water. Each facility diffuser design was based on the scaling parameters of the individual facility.

ADS Phase B Tests

This design change will have no impact on the ADS Phase B tests.

The CMT is not modeled in the tests. The ADS tests are not dependent on the CMT heatup or draindown rates. This change does not affect the range of conditions required to be tested.

2.7 Direct Vessel Injection Nozzle Venturi (DM-07)

Description

A venturi has been incorporated into the reactor vessel DVI nozzle. Figure 2-3 shows the venturi.

Purpose of Modification

This venturi reduces the severity of a DVI line LOCA by reducing the RCS blowdown rate.

Safety Analysis Impact

A DVI line break LOCA is a limiting design basis event because of the size of the DVI line and because the break location disables one half of the injection supplies. Incorporating a reduced ID venturi into the DVI nozzle effectively reduces the maximum DVI break size. The minimum ID throat of venturi is long enough to effectively choke the flow. There is no impact on CMT and accumulator injection flow because orifices in the discharges of these tanks can be adjusted to compensate for the venturi, therefore, the resistance from the CMT and accumulator to the RCS will be unchanged. There will be a small effect on the in-containment refueling water storage tank (IRWST) injection resistance.

PRA Impact

This change will have no impact on the PRA because this break is still a medium LOCA when both reactor and CMT flow are taken into consideration. This change also does not affect the success criteria used in the PRA.

SSAR Impact

This change impacts the following SSAR Chapters:

Chapter 4	Reactor
Chapter 15	Accident Analyses

Test Program Impact

Core Makeup Tank Separate Effects Tests

This design change will have no impact on these tests.

In the AP600, the addition of a venturi to the DVI nozzle is accommodated by maintaining the overall pressure drop in the DVI line by reducing the size of the orifice already designed into the line.

The drainline in the CMT tests simulates the DVI line returning to the reactor vessel. The test drainline is designed for a range of flowrates out of the CMT which bound the range of flowrates for all anticipated transients.

Integral Systems Tests at SPES-2 Test Facility

This design change will have a small impact on facility hardware for this test facility.

The SPES-2 tests are integral systems tests to obtain thermal-hydraulic data for computer code validation and to investigate systems interactions during high pressure transients. The DVI line in SPES-2 will be modified to include a scaled venturi prior to the performance of the DVI line break test. Since the overall pressure drop is the same for the other SBLOCAs, regardless of inclusion of the DVI venturi, the other tests do not require its installation.

The nozzle will be installed for the duration of the test program, which will include SGTR and MSLB matrix tests. A cold test to characterize the pressure drop in the DVI line will be performed during the installation of the venturi.

Integral Systems Tests at Oregon State University

This design change will have a small impact on facility hardware for this test facility.

The OSU tests are integral systems tests to obtain thermal-hydraulic data for computer code validation and to investigate long term cooling behavior. The DVI line in OSU will be modified to include a scaled venturi prior to the performance of the DVI line break test. Since the overall pressure drop is the same for the other SBLOCAs, regardless of inclusion of the DVI venturi, the other tests do not require its installation. The DVI venturi has a resistance that is less than 1 percent of the total injection line resistance for the CMT, accumulator or IRWST. A cold flow test to characterize the pressure drop of the DVI venturi will be performed.

ADS Phase B Tests

This design change will have no impact on these tests.

The installation of a venturi impacts the plant response to a DVI line break. The pretest analyses performed to establish the test conditions cover the range of expected transients and the test blowdowns are designed to provide fluid conditions over a wide range of conditions. This change does not affect the range of conditions required to be tested.

2.8 PRHR HX Inlet Motor Operated Valve Arrangement (DM-08)

Description
A motor-operated valve (MOV) is added in the inlet line to each PRHR HX. Two MOVs replace the two manual valves and a MOV in the common line. Figure 2-4 shows this arrangement.

Purpose of Modification
This change improves the reliability of the PRHR HX and reduces the chances of forced shutdowns due to a valve problem occurring during at-power in-service testing (IST) of the PRHR HX control valves.

With the valve arrangement described in the SSAR (Reference 7.1), the MOV in the common line is normally open with its power removed. Its power is removed to avoid the possibility of a spurious valve closure failure which would render the PRHR HX inoperable. The MOV provides remote isolation in case of excessive PRHR HX leakage. It will also be closed every three months during the at-power IST of the PRHR HX air operated valve (AOV).

If MOV problems occur during IST, there would be a limited time to fix them to avoid a Technical Specification violation. Such problems include the MOV failing to open or problems with MOV limit switches falsely indicating that the MOV has not opened. Either of these problems would force the operators to declare the PRHR HX inoperable. Because the MOV is located inside containment, access and repair is difficult.

With the revised design, the chance of this occurring is reduced, since both MOVs would have to experience a problem to render the system inoperable. With both HXs available, the power would not have to be removed to prevent a spurious failure from defeating the PRHR HXs. This makes IST and leakage isolation of the PRHR HXs easier for the operators since they would not have to restore power locally at the motor control center before operating the MOV.

In the unlikely case that one of the PRHR HXs is isolated because of a leak, the single operable HX would have one inlet MOV which would have its power removed. This change significantly improves the PRHR HX operability since isolating a PRHR HX is an unlikely situation.

Safety Analysis Impact
There is no impact on the safety analysis because this change has no impact on system performance during design basis event or on the limiting single failures for these events.

PRA Impact
There is no impact on the PRHR HX reliability because these valves are not modeled in the PRHR HX fault tree.

In the event of a PRHR HX tube rupture, the operator should isolate the PRHR HX. Failures in the control system or operator action dominate the unavailability. Having to close two MOVs instead of one will have a small impact on the core melt frequency (CMF).

SSAR Impact
This change impacts the following SSAR Chapters:

Chapter 3	Design of Structures, Components, Equipment and Systems
Chapter 6	Engineered Safety Features
Chapter 7	Instrumentation and Controls
Chapter 16	Technical Specifications

Test Program Impact

Core Makeup Tank Separate Effects Tests
Integral Systems Tests at Oregon State University
Integral Systems Tests at SPES-2 Test Facility
ADS Phase B Tests

This design change has no impact on the above test programs.

This design change improves the reliability of the PRHR HX by reducing the chance of forced shutdowns due to a valve problem that may occur during IST. This change does not impact the performance of the PRHR HX in the AP600 during a transient and has no impact on the above test programs.

2.9 ADS Valves (DM-09, DM-10, DM-11)

During several meeting with Westinghouse the Nuclear Regulatory Commission (NRC) has expressed concerns over the use of gate valves to initiate/control ADS flow. Utilities have also expressed similar concerns. Westinghouse held an industry review of the ADS design that focused on the type of valves used. This review concluded that there is uncertainty with the use of gate valves in the AP600 ADS, especially in Stages 2 and 3. It was recommended that the use of other valve types be considered.

Several changes are included to allow for the use of different types of ADS valves. The general approach is to define an ADS arrangement that can accommodate different valve types. The AP600 safety analysis will be performed using bounding ADS valve and system parameters. The LOCA analysis performed in support of these changes used bounding valve and system parameters. As shown in Section 4.3, these parameters result in similar or improved plant performance.

The SSAR ADS description and piping and instrumentation diagrams (P&ID) (see Appendices A-1 and A-2) show generic valve types. The ADS Phase B tests to support design certification have been restructured to provide system behavior data over a wide range of conditions, including flow areas and valve resistances, to provide bounding system parameters (see Section 5). The selection and qualification of ADS valves which meet the bounding system parameters (see Section 6) is not part of design certification. Figure 2-5 shows the revised ADS design and the limiting valve characteristics.

2.9.1 ADS Stage 1 Flow Capacity (DM-09)

Description
The minimum effective critical flow area for the ADS stage 1 valves is reduced (see Table 2-1).

Purpose of Modification
The reduced flow area provides flexibility in ADS Stage 1 valve selection.

Safety Analysis Impact
The impact of this change is included in the evaluation presented in Section 4.3 The results of analyses of the DVI break LOCA and spurious ADS analysis show similar or improved performance in comparison to the SSAR (Reference 7.1).

PRA Impact

There is no impact on the PRA because these valves are small relative to the other ADS valves such that ADS success criteria are not changed.

SSAR Impact

This change impacts the following SSAR Chapters:

Chapter 5	Reactor Coolant System and Connected Systems
Chapter 6	Engineered Safety Features (See Appendix A-1)
Chapter 15	Accident Analyses

Test Program Impact

Core Makeup Tank Separate Effects Tests

This design change will have only minimal impact on these tests.

The CMT tests are separate effects tests designed to investigate thermal-hydraulic phenomena within the CMT over a wide range of pressures and temperatures. This change may affect the timing but not the nature of the phenomena being investigated in this test facility. This change does not affect the range of conditions required to be tested.

Integral Systems Tests at SPES-2 Test Facility

This design change will have only a minor impact on these tests.

The SPES-2 tests are integral systems tests to obtain thermal-hydraulic data for computer code validation and to investigate systems interactions during high pressure transients. The change required for ADS Stage 1 is not significant. A single valve is used in each stage to initiate and control the blowdown and an orifice is used to represent the second valve. A new orifice will be sized and installed to represent the minimal valve area, or maximum pressure drop in each stage. This would represent the minimum flow capability of the ADS for stage 1.

Integral Systems Tests at Oregon State University

This change will have only minor impact on these tests.

The OSU tests are integral systems tests to obtain thermal-hydraulic data for computer code validation and to investigate long tern cooling behavior. The same changes described above for SPES-2 for ADS Stage 1 are required for OSU and are not significant. The minimum flow capability of the ADS Stage 1 will be simulated in the test facility.

ADS Phase B Tests

The Stage 1 ADS valve procured for this test is consistent with this change. As a result there is no impact on this test.

2.9.2 ADS Stage 2/3 Valves (DM-10)

Description

The minimum effective flow area for ADS Stage 2 and Stage 3 valves is reduced (see Table 2-1). This reduction in flow area is compensated for by an increase in the fourth Stage ADS flow capability, see subsection 2.9.3.

Purpose of Modification

This change allows the use of alternative valve types. Figure 2-5 shows two valve options being considered.

Safety Analysis Impact

The impact of this change is included in the evaluation presented in Section 4.3 The results of analyses of the DVI break LOCA and spurious ADS analysis show similar or improved performance in comparison to the SSAR (Reference 7.1).

PRA Impact

An evaluation of the impact of this change (Section 3) shows that this change, coupled with the revised ADS Stage 4 design, has only a small impact on the AP600 core melt frequency (CMF).

SSAR Impact

This change impacts the following SSAR Chapters:

Chapter 5	Reactor Coolant System and Connected Systems
Chapter 6	Engineered Safety Features (See Appendix A-1)
Chapter 15	Accident Analyses

Test Program Impact

Core Makeup Tank Separate Effects Tests

This design change will have only a minimal impact on these tests.

The CMT tests are separate effects tests designed to investigate thermal-hydraulic phenomena within the CMT over a wide range of pressures and temperatures. This change may affect the timing but not the nature of the phenomena being investigated in this test facility. This change does not affect the range of conditions required to be tested.

Integral Systems Tests at SPES-2 Test Facility

This design change will have only minor impact on these tests.

The SPES-2 tests are integral systems tests to obtain thermal-hydraulic data for computer code validation and to investigate systems interactions during high pressure transients. The change required for ADS Stages 2 and 3 is not significant. A single valve is used in each stage to initiate and control the blowdown and an orifice is used to represent the second valve. A new orifice will be sized and installed to represent the minimal valve area, or maximum pressure drop in each stage. This would represent the minimum flow capability of the ADS for Stages 2 and 3.

Integral Systems Tests at Oregon State University

This change will have only minor impact on these tests.

The OSU tests are integral systems tests to obtain thermal-hydraulic data for computer code validation and to investigate long tern cooling behavior. The same changes described above for SPES-2 for ADS Stages 2 and 3 are required for OSU and are not significant. The minimum flow capability of the ADS for Stages 2 and 3 will be simulated in the test facility.

ADS Phase B Tests

This change significantly changes test requirements but has only minor impact on facility hardware.

The ability to accommodate a range of valve sizes and types has a major impact on the test data requirements for the ADS Phase B test program. The tests will provide data for computer code validation of the ADS system. The test matrix has been restructured to provide test data over a range of thermal-hydraulic conditions representing high flow resistance (minimum venting flow) and low flow resistance (maximum loads).

The test conditions are obtained by varying the system configuration and the system conditions upstream of the valve piping package. The system configuration is varied through the use of orifices and flow nozzles to represent valves with different flow characteristics. The fluid conditions are controlled by establishing the appropriate initial conditions in the test and then performing a controlled blowdown through each stage or stages of the ADS valve package. Data will be obtained for both saturated steam and steam/saturated water conditions.

Section 5 discusses the changes to these tests in more detail.

2.9.3 ADS Stage 4 Valves (DM-11)

Description
The ADS Stage four arrangement is changed as shown in Figure 2-4. The possible use of smaller valves in ADS Stages 2 and 3 requires more flow capacity in the fourth stage. Allowing for the use of squib valves limits the maximum line size.

The IST valves currently shown on the Passive Core Cooling System (PXS) P&ID (Appendix A-2) would be retained if air piston gate valves are used. However, if squib valves are used the IST valves would be deleted.

The fourth stage ADS valves are interlocked to preclude their opening at normal RCS operating pressures.

Purpose of Modification
This change allows the use of alternate valve types in ADS Stages 2 and 3 and ADS Stage 4. Figures 2-5 and 2-6 show valve options being considered.

Safety Analysis Impact
The impact of this change is included in the evaluation presented in Section 4.3 The results of analyses of the DVI break LOCA and spurious ADS analysis show similar or improved performance in comparison to the SSAR (Reference 7.1).

PRA Impact
An evaluation of the impact of this change (Section 3) shows that this change has only a small impact on the AP600 core melt frequency (CMF).

SSAR Impact
This change impacts the following SSAR Chapters:

Chapter 3	Design of Structures, Components, Equipment and Systems
Chapter 5	Reactor Coolant System and Connected Systems
Chapter 6	Engineered Safety Features
Chapter 7	Instrumentation and Controls
Chapter 15	Accident Analyses
Chapter 16	Technical Specifications

Test Program Impact

Core Makeup Tank Separate Effects Tests

This design change will have only minimal impact on these tests.

The CMT tests are separate effects tests designed to investigate thermal-hydraulic phenomena within the CMT over a wide range of pressures and temperatures. This change may affect the timing but not the nature of the phenomena being investigated in this test facility. This change does not affect the range of conditions required to be tested.

Integral Systems Tests at SPES-2 Test Facility

This design change will have only minor impact on these tests.

The SPES-2 tests are integral systems tests to obtain thermal-hydraulic data for computer code validation and to investigate systems interactions during high pressure transients. With this design change the fourth stage of ADS can now can discharge from both hot legs even when a single failure of a fourth stage valve is assumed. This is a direct result of the two flow paths in the revised design. SPES-2 will be reconfigured to allow both fourth stages to discharge into a single header and flow measurement system. The total flow out of both fourth stage ADS lines will be measured simultaneously.

Orifices installed in each ADS fourth stage discharge line will be sized to simulate the total pressure drop and flow area in each line. For a single failure of a fourth stage valve, an orifice representing the pressure drop of a single flow path will be installed.

Integral Systems Tests at Oregon State University

This design change will have only minor impact on these tests.

The OSU tests are integral systems tests to obtain thermal-hydraulic data for computer code validation and to investigate long term cooling behavior. The same changes described above for SPES-2 for ADS Stage 4 are required for OSU and are not significant. Headering of flow from both ADS Stage 4 lines into a single break measurement system will only occur for double ended break cases. Separate flow measurements from the fourth stage lines is possible for the single ended breaks.

ADS Phase B Tests

This design change has no impact on the ADS Phase B tests.

Table 2-1
Summary of AP600 Design Modifications

Mod. No.	Description	Current AP600 (SSAR, Rev. 1)	Revised AP600	Reason for Modification
DM-01	PRHR HX Actuation Logic	Actuation on High Pressurizer Pressure or Low Steam Generator Level	Actuation on CMT actuation	Reduces potential for ADS actuation following a SGTR event. Reduces potential for pressurizer overfill.
DM-02	Pressurizer Heater Control Logic	No automatic block	Automatic block on CMT actuation	Reduces potential for ADS actuation following a SGTR event. Reduces potential for pressurizer overfill.
DM-03	CVS Control Logic post CMT actuation	Start both pumps on CMT signal	Start/stop 1 pump on CMT signal at Pressurizer levels of 10/20%	Reduces potential for pressurizer overfill.
DM-04	ADS Stage 1 setpoint	CMT level, 1500 ft^3	CMT level, 1350 ft^3	Reduces potential for ADS actuation following SGTR/SLB events.
DM-05	ADS Stage 2/3 setpoints	CMT level, 1300/1000 ft^3	Stage 1 actuation plus timers	Removes reliance on CMT level instrumentation during rapid depressurization portion of ADS

Table 2-1
Summary of AP600 Design Modifications

Mod. No.	Description	Current AP600 (SSAR, Rev. 1)	Revised AP600	Reason for Modification
DM-06	CMT Inlet Diffuser	None	[ ]a,c	Reduces rate of steam condensation during medium to large LOCA events.
DM-07	DVI nozzle venturi	None	[ ]a,c	Reduces RCS blowdown rate for DVI LOCA
DM-08	PRHR HX Inlet Isolation	1 MOV, 2 manual valves	2 MOVs	Technical Specification improvement
DM-09	ADS Stage 1 Valve Effective Flow Area	[ ]a,c	[ ]a,c	ADS valve type flexibility
DM-10	ADS Stage 2/3 Valves	[ ]a,c	[ ]a,c	ADS valve type flexibility
DM-11	ADS Stage 4	[ ]a,c	[ ]a,c	ADS valve type flexibility

TABLE 2-2 REVISED PRHR HX ACTUATION SIGNALS

DESCRIPTION	SIGNAL	SETPOINT
PRHR HX Actuation (via PMS)	- Low SG narrow range level in any SG + low SFW flow after time delay	per SSAR (165 gpm, 60 sec)
	- Low SG WR level in any SG	per SSAR
	- CMT actuation	NA
	- ADS actuation	NA

TABLE 2-3 REVISED CVS MAKEUP CONTROL SIGNALS

DESCRIPTION	SIGNAL	SETPOINT
Normal RCS Makeup (1) (via PLS)	Low pressurizer level relative to programmed level starts makeup; higher level stops makeup	start - 0% (1) stop - 18% (1)
Post CMT Actuation RCS Makeup (2) (via PLS)	CMT actuation + low pressurizer level starts makeup, higher level stops makeup	start - 10% level stop - 20% level

Notes:

(1) One CVS pump starts with suction from boric acid and makeup water blended to match RCS boron concentration. Flow is controlled to a fixed flowrate, setpoint has not been selected but will be less than 100 gpm. Start / stop pressurizer levels (percent of level span) are relative to programmed level, approximately 23% / 40% at no power and 40% / 58% at full power.

(2) One CVS pump starts with suction from boric acid tank. Flowrate is not controlled; valve will be full open to provide maximum flow. Start / stop pressurizer levels are absolute values (percent of level span).

TABLE 2-4 REVISED ADS ACTUATION SIGNALS

DESCRIPTION	SIGNAL	SETPOINT
First Stage ADS		
- First Stage Actuation, Isolation Valve Actuation	- CMT actuation signal + Low-1 CMT level in either CMT	67% CMT volume
- First Stage Control Valve Actuation	- 1st Stage actuation + time delay	20 sec delay
Second Stage ADS		
- Second Stage Actuation, Isolation Valve Actuation	- 1st Stage actuation + time delay	60 sec delay
- Second Stage Control Valve Actuation	- 2nd Stage actuation + time delay	30 sec delay
Third Stage ADS		
- Third Stage Actuation, Isolation Valve Actuation	- 2nd Stage actuation + time delay	120 sec delay
- Third Stage Control Valve Actuation	- 3rd Stage actuation + time delay	30 sec delay
Fourth Stage ADS		
- Fourth Stage A Actuation, Isolation Valve Actuation	- 3rd Stage actuation + time delay + low-2 CMT level in either CMT	120 sec delay 20% CMT vol.
- Fourth Stage A Control Valve Actuation	- 4th Stage A actuation + time delay	30 sec delay
- Fourth Stage B Actuation, Isolation Valve Actuation	- 4th Stage A actuation + time delay	30 sec delay
- Fourth Stage B Control Valve Actuation	- 4th Stage B actuation + time delay	30 sec delay

FIGURE 2-1 CMT INLET DIFFUSER

a,c

FIGURE 2-2 DVI NOZZLE VENTURI

a,c

FIGURE 2-3 PRHR HX INLET ISOLATION

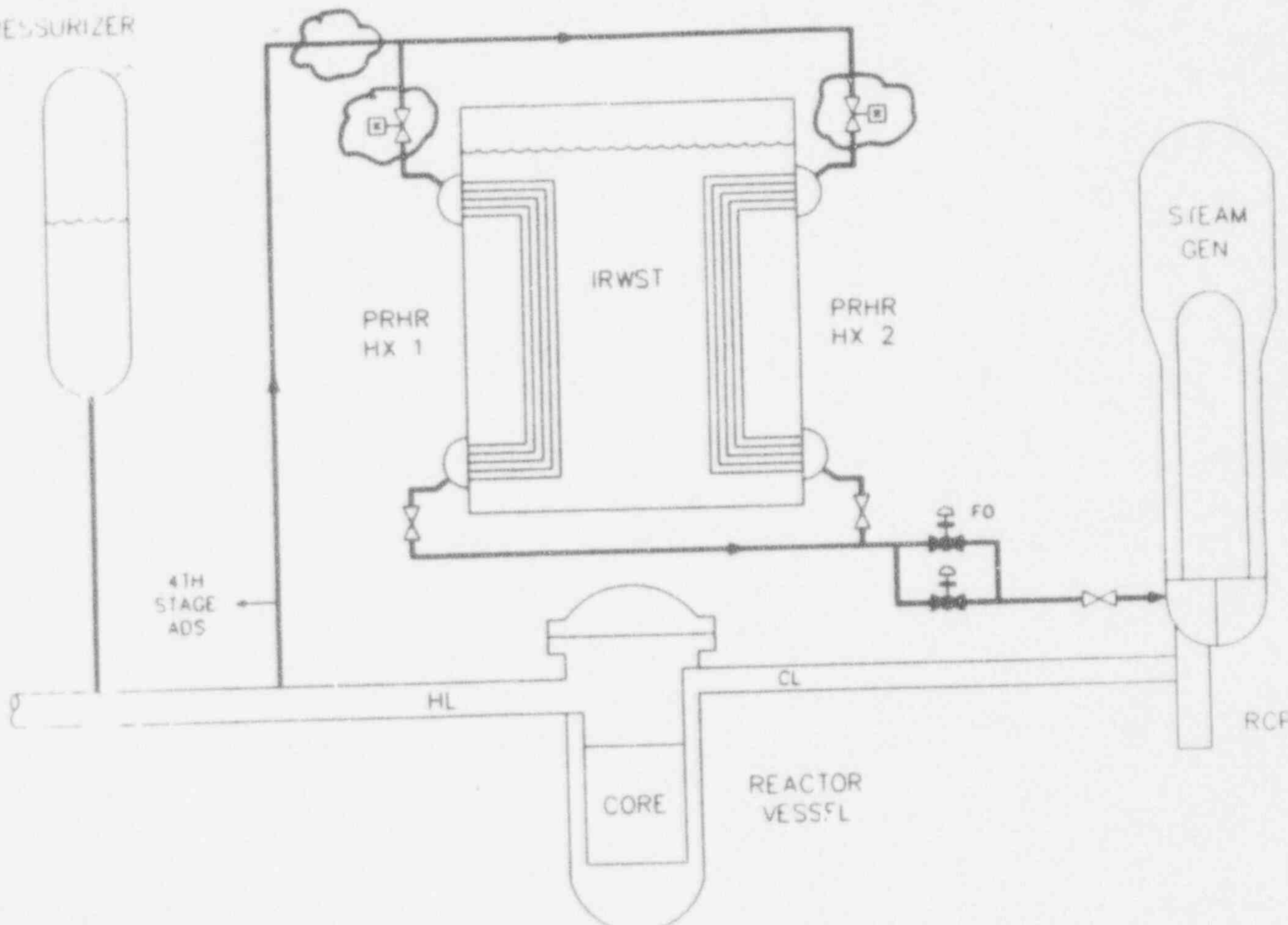


FIGURE 2-4 REVISED ADS DESIGN

a,c

FIGURE 2-5 ADS STAGE 1/2/3 EXAMPLES

a,c

FIGURE 2-6 ADS STAGE 4 EXAMPLES

a,c

3.0 PRA Evaluation of ADS Design Changes

The ADS model in the AP600 PRA (Reference 7.2) uses motor-operated gate valves in ADS Stages 1, 2 and 3 and air-operated gate valves in ADS Stage 4. The results of this PRA, specifically with respect to ADS reliability, were evaluated to determine the impact on core damage frequency (CDF) of ADS hardware modifications.

This evaluation started with inspection of the list of dominant accident sequence cutsets shown in Table F-7 of the AP600 PRA Report. Table F-7 reports the top 100 core damage cutsets and many cutsets, starting with cutset number 5, include failure of the ADS. However, in all of these top 100 core damage cutsets that include ADS failure, the ADS failure is a result of failure to signal the ADS to function, which includes both failure to automatically actuate and failure of the operator to actuate the ADS. The PRA computer output files were reviewed to assess more core damage cutsets to determine where ADS hardware failures first appear. The first ADS hardware failure cutsets appeared as a set of six, starting with number 135, each with a CDF of 2.15E-10. The next hardware associated failure of the ADS is number 175 with a core damage frequency of 1.52E-10. In the top 100 cutsets, ADS failure due to not generating an ADS actuation signal is involved in cutsets that contribute approximately 4.6E-8 to the core damage frequency .

This evaluation continued by performing sensitivity studies using the ADS fault trees that had the most influence on CDF. The four fault trees selected are ADS, ADA, ADN and ADC from the PRHR HX tube rupture event. These are the most important because the PRHR HX tube rupture event was the initiating event for 10 of the first 11 core damage sequences that involved ADS hardware failures. For this sensitivity study, new reference values for system reliability for each of the ADS fault trees were calculated to account for the addition of pressure switches that prevent Stage 4 ADS valves from opening at high RCS pressure. These fault trees were then evaluated for two different ADS configurations. These configurations are given in Table 3-1. The results of this study are also shown in Table 3-1.

On the basis of changes in unavailability, the various hardware design options evaluated have minimal effect on ADS reliability. As shown in the table, the maximum increase in fault tree unavailability, due to hardware changes, is less than a factor of two.

As indicated above, ADS hardware failures (6*2.15E-10+1.52E-10) contributed about 1.4E-9 of the total CDF while ADS actuation failures contributed about 4.6E-8. That is, hardware failures account for roughly three percent of the CDF cutsets involving the ADS. Even if the ADS hardware failure rate is doubled, the impact on the total AP600 CDF would be insignificant.

TABLE 3-1 ADS FAULT TREES UNAVAILABILITY

	SSAR	ADS Configuration #1	ADS Configuration #2
ADS Stages 2/3	MO gate	MO globe	MO globe
ADS Stage 4	AO gate	AO gate	Squib
Fault Tree ADN	2.77E-3	2.77E-3	2.77E-3
Fault Tree ADC	3.32E-3	3.32E-3	3.32E-3
Fault Tree ADS	2.91E-5	4.86E-5	4.93E-5
Fault Tree ADA	5.79E-4	6.09E-4	6.09E-4

4.0 Integrated Evaluation of Safety Analysis Impact

Chapter 15 of the SSAR (Reference 7.1) presents the design basis accident analyses performed for the AP600. Accidents are classified therein according to the type of event: increase in heat removal from the primary system, decrease in heat removal by the secondary system, etc. The passive safety system design changes affecting the CMTs and ADS impact some postulated accident events more than others. The SSAR Chapter 15 analyses were reviewed with respect to the design changes, and those which were determined to be significantly affected have been reanalyzed modeling the design changes pertinent to that particular event.

Assumptions made in the SSAR to perform analyses of the selected transients were reviewed in light of the design changes. The events determined to have the most significant impact from the design changes are steamline break, SGTR and LOCA. These events were reanalyzed and the results are presented in the following sections.

4.1 Steamline Break

Introduction

This section assesses the impact on the AP600 steamline break analysis of the design changes identified in Section 2. Four of the design changes have been identified as having potential impact on the safety analysis results for the steamline break accident. These are as follows:

<u>PRHR HX Actuation Signal</u>

> Delete the current signals that actuate the PRHR on high pressurizer or high steam generator level. Add a new signal that initiates PRHR operation on CMT actuation.

<u>ADS Stage 1 Actuation</u>

> Reduce the ADS Stage 1 actuation CMT level setpoint. The corresponding CMT volumes are 1500 and 1350 ft^3 for the old and new setpoints, respectively.

<u>CMT Inlet Diffuser</u>

> Install a CMT inlet diffuser where none previously existed.

DVI Nozzle Venturi

Install a DVI nozzle venturi where none previously existed.

In the discussion that follows, each of these changes is evaluated with regard to the steamline break accident.

PRHR HX Actuation Signal

The steamline break accident does not rely on either the high pressurizer or high steam generator level functions to initiate operation of the PRHR. For the limiting steamline break transients, any PRHR actuation would be predicted to occur on the low steam generator water level signal. The nature of the steamline break transient is such that, at least during the initial phases for the largest breaks, the water levels in both the pressurizer and the steam generators are decreasing. The pressurizer level falls because of the RCS cooldown produced by the steamline break accident. As the transient proceeds, the addition of borated coolant from the CMTs and/or the accumulators acts to restore water level in the pressurizer. The primary effect on steam generator water level is the loss of inventory through the break. For some small break cases, conservative analysis assumptions with regard to startup feedwater may actually produce a slight increase in inventory for the intact steam generator. The predicted steam generator water levels never approach high level setpoints.

In addition, since the conservative analysis assumptions for the core response steamline break accident maximize the severity of the RCS cooldown, the cases presented in AP600 SSAR Sections 15.1.4 and 15.1.5 model PRHR actuation at time zero. This eliminates any analytical sensitivity to a change in the signals that might actuate the PRHR at some point later in the transient. Therefore, the proposed changes in the signals that actuate the PRHR HX are bounded by the AP600 SSAR safety analysis for the steamline break accident.

ADS Stage 1 Actuation

The initiating signal for Stage 1 ADS is CMT water level. A basic design requirement for the AP600 is that the ADS not actuate for Condition 2 accidents. On this basis, it would be acceptable for the ADS to actuate during Condition 3 and 4 non-LOCA events. However, it is a design goal for the AP600 that ADS not actuate during any of the non-LOCA transients. The results for the limiting core response steamline break, as reported in Section 15.1.5 of the AP600 SSAR, indicate a relatively small reduction in CMT water volume during the transient. From an initial value of 2000 ft^3, the minimum CMT water volume predicted for the SSAR is 1890 ft^3, which provides significant margin to the stage 1 ADS actuation analysis setpoint of 1500 ft^3 that has been used. However, the assumptions used in the SSAR analysis did not maximize the potential for ADS actuation. As a result, supplemental analysis has been performed with the specific intent of addressing ADS actuation for the steamline break accident.

The results of this analysis, which are summarized in Table 4.1-1 and in Figures 4.1-1 through 4.1-14, demonstrate that for a limiting steamline break the minimum predicted CMT water volume is 1590 ft^3. This result indicates a 90 ft^3 margin to the SSAR Stage 1 ADS actuation setpoint (CMT volume of 1500 ft^3). The setpoint change described in Section 2 (CMT volume of 1350 ft^3) increases the margin to 240 ft^3. The significant reduction in minimum CMT water volume predicted for the current analysis relative to the referenced SSAR steamline break case is primarily caused by two model changes. This analysis assumes a 0.4 ft^2 break rather than the 1.4 ft^2 break in the SSAR analysis since sensitivity analyses show this to be more limiting with respect to minimum CMT level. Also the current analysis conservatively models condensation in the CMT that is only 10 percent of that applied in the SSAR case. Verification of the condensation model components will occur once the AP600 CMT test data is available.

In summary, the reduction in the Stage 1 ADS actuation setpoint to 1350 ft^3 helps to preclude ADS actuation during a limiting steamline break transient.

CMT Inlet Diffuser

The diffuser does not affect CMT draindown behavior when steam comes from the pressurizer; as is the case for the steamline break event.

DVI Nozzle Venturi

As described in Section 2.7, the installation of the DVI nozzle venturi should have no impact on the steamline break analysis for the AP600. The presence of this venturi is intended to mitigate the effects of a DVI line LOCA. The total resistance from the CMTs and accumulators to the RCS will be unchanged because of orifices in the discharges of these tanks. As a result, the steamline break results for the AP600 will be unaffected.

Conclusions

In summary, the AP600 design changes defined in Section 2.0 are acceptable with respect to the SSAR steamline break analysis for the AP600. Specific models and assumptions associated with the analysis for this accident will be verified against the AP600 tests.

TABLE 4.1-1

SEQUENCE OF EVENTS FOR STEAMLINE BREAK CASE
BREAK IN LOOP 2 & OFFSITE POWER AVAILABLE

EVENT	TIME (Seconds)
0.4 ft^2 double-ended rupture steamline break occurs	0
Reactor and turbine assumed tripped	0
PRHR heat exchanger assumed to actuate	0
Low steamline pressure SIS setpoint reached	2.5
Steamline isolation	14.5
Feedwater isolation	14.5
Reactor coolant pumps trip	17.5
CMT actuation	24.5
Startup feedwater isolated to all SGs	28.9
Accumulators actuate	1046
Faulted steam generator blowdown ends	~2600

FIGURE 4.1-1
NUCLEAR POWER TRANSIENT, STEAM SYSTEM PIPING FAILURE

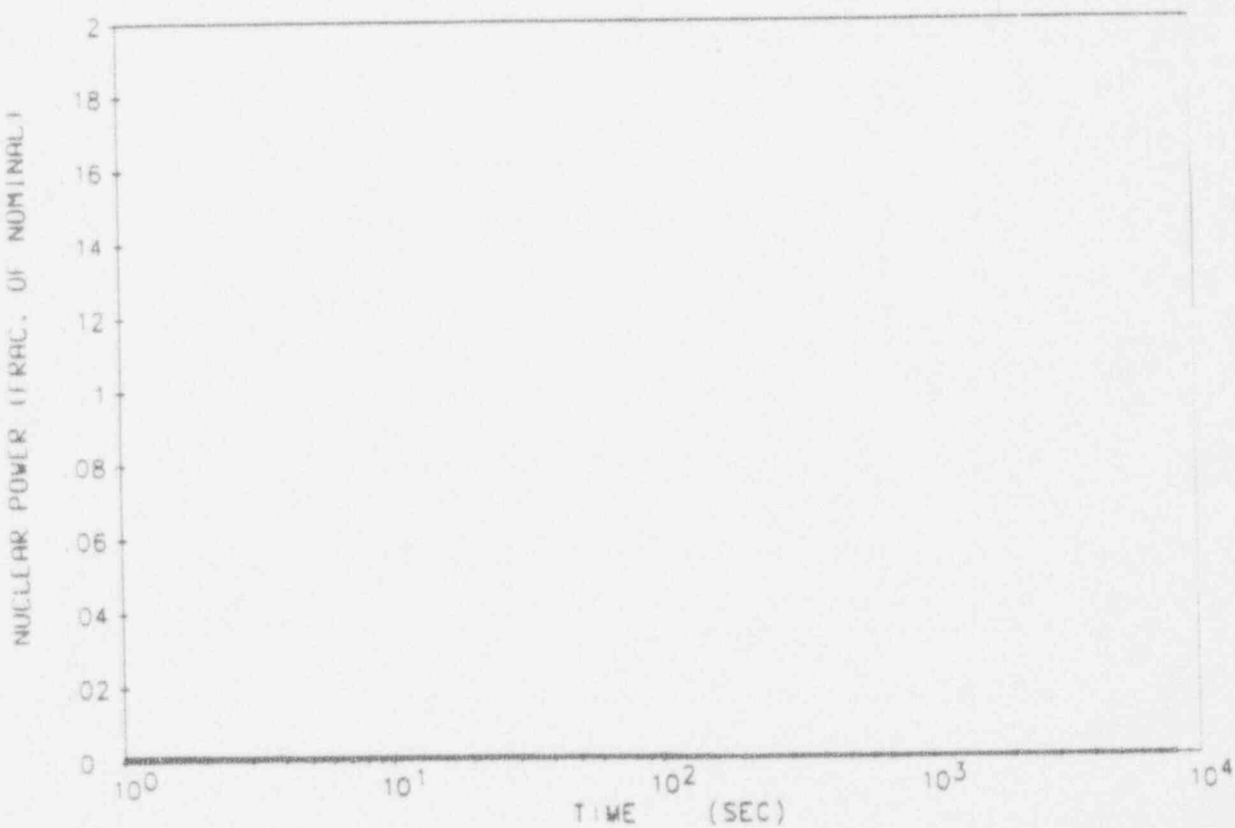


FIGURE 4.1-2
CORE HEAT FLUX TRANSIENT, STEAM SYSTEM PIPING FAILURE

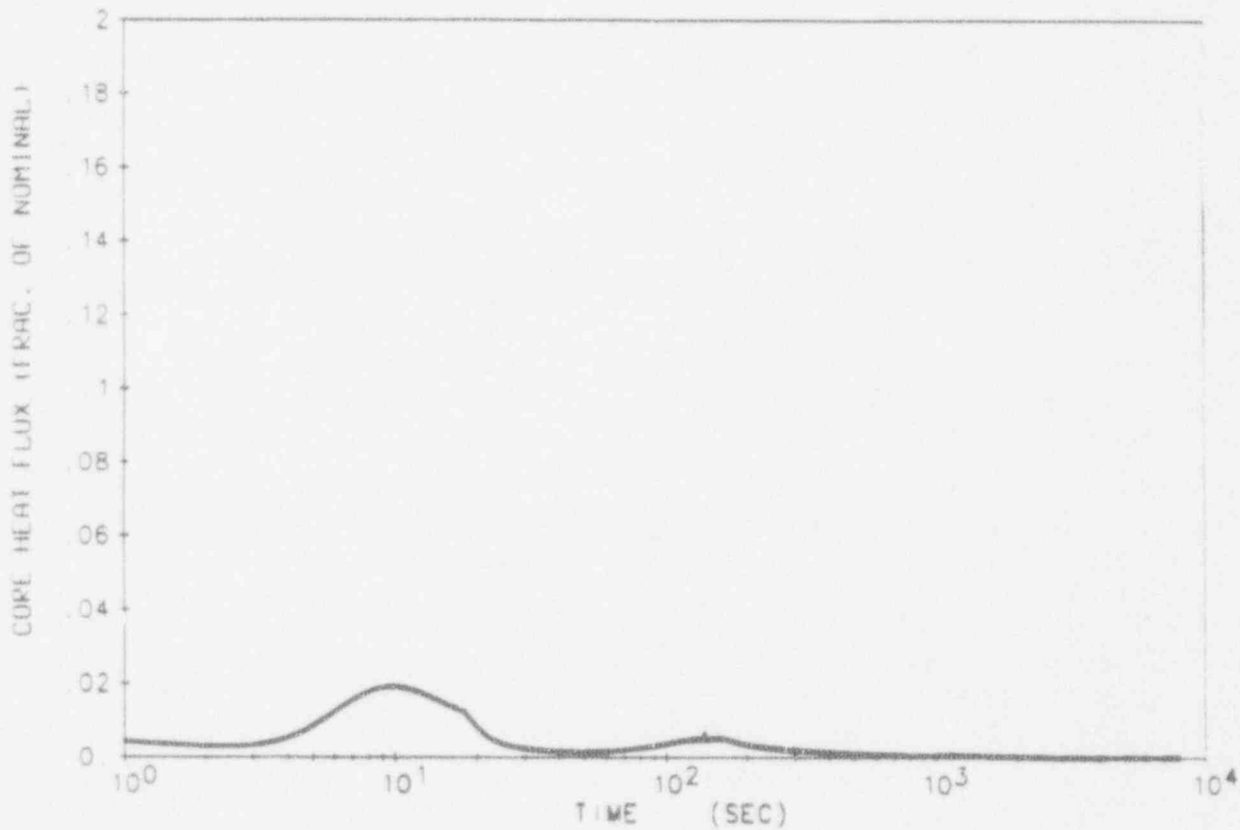


FIGURE 4.1-3
REACTIVITY TRANSIENT, STEAM SYSTEM PIPING FAILURE

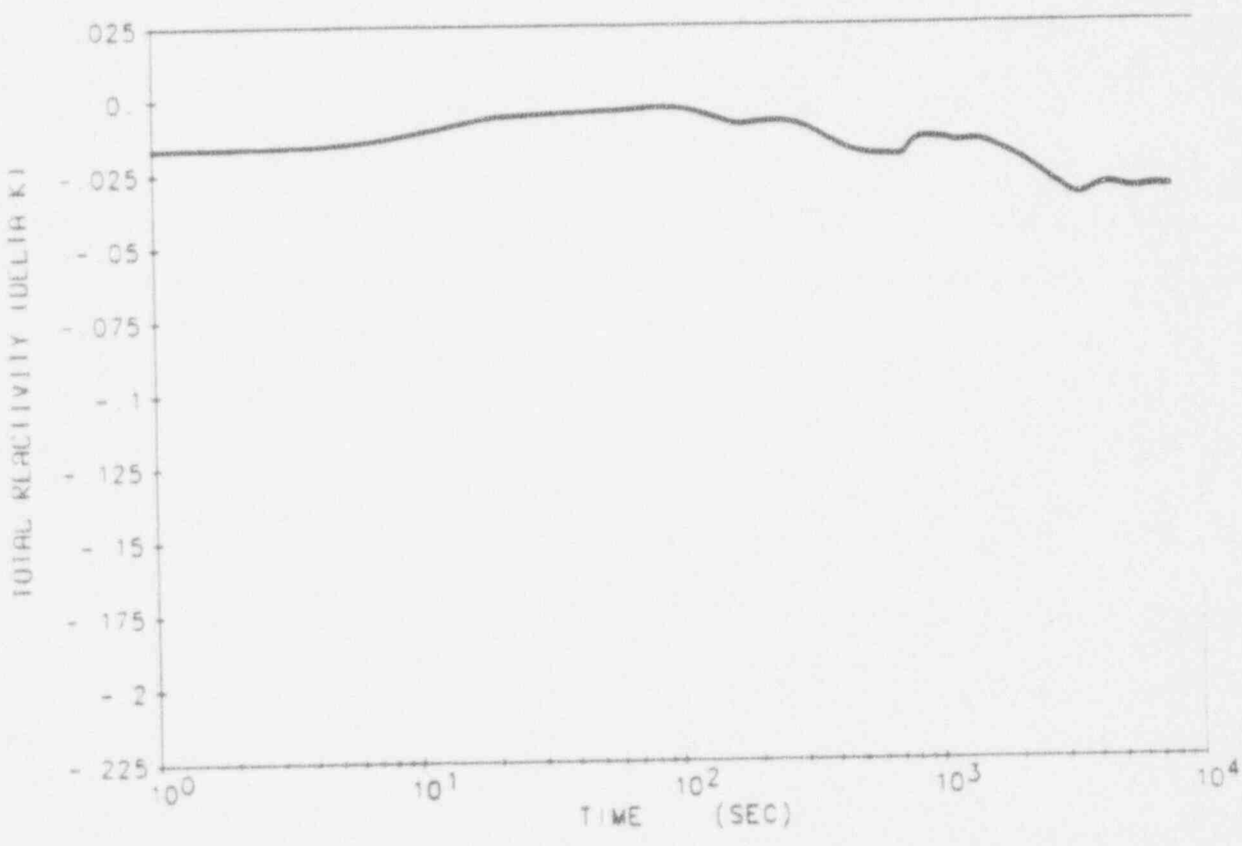


FIGURE 4.1-4
CORE AVERAGE TEMPERATURE TRANSIENT, STEAM SYSTEM PIPING FAILURE

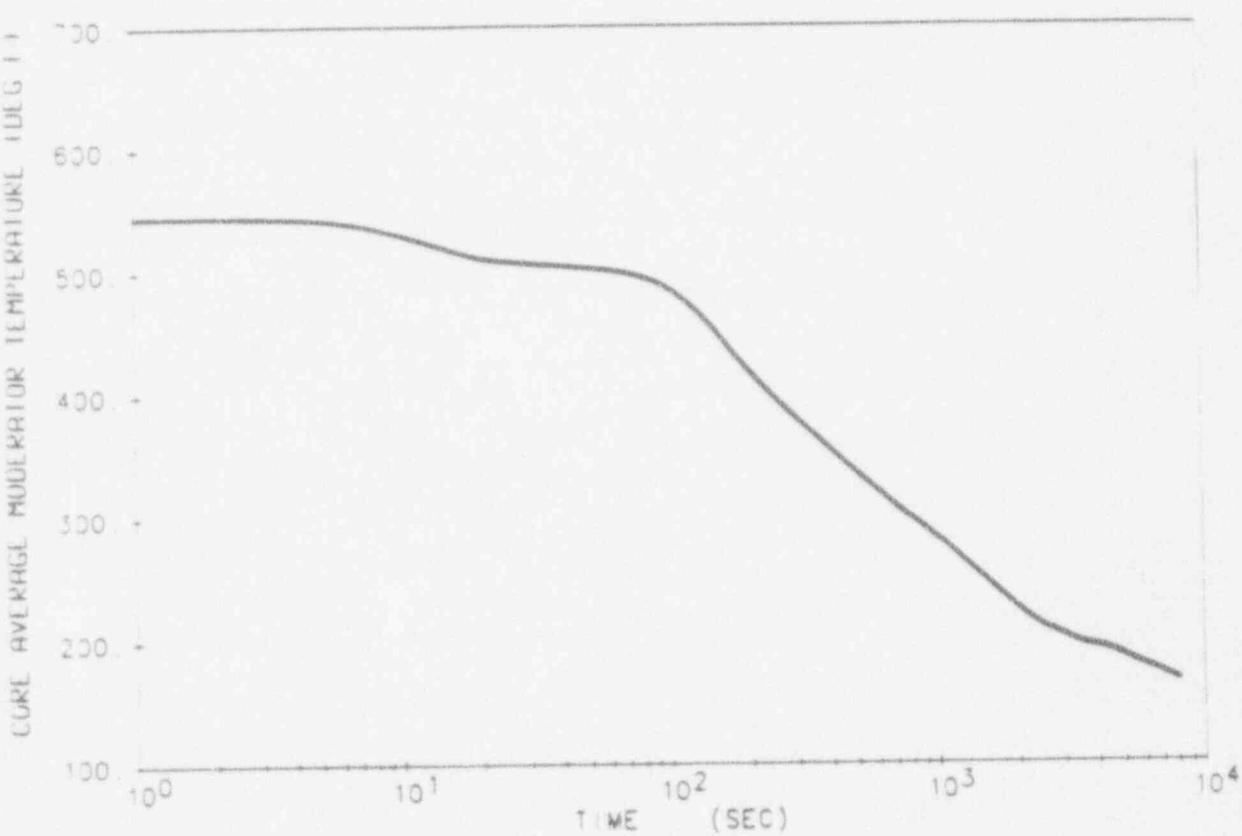


FIGURE 4.1-5
REACTOR VESSEL INLET TEMPERATURE, STEAM SYSTEM PIPING FAILURE

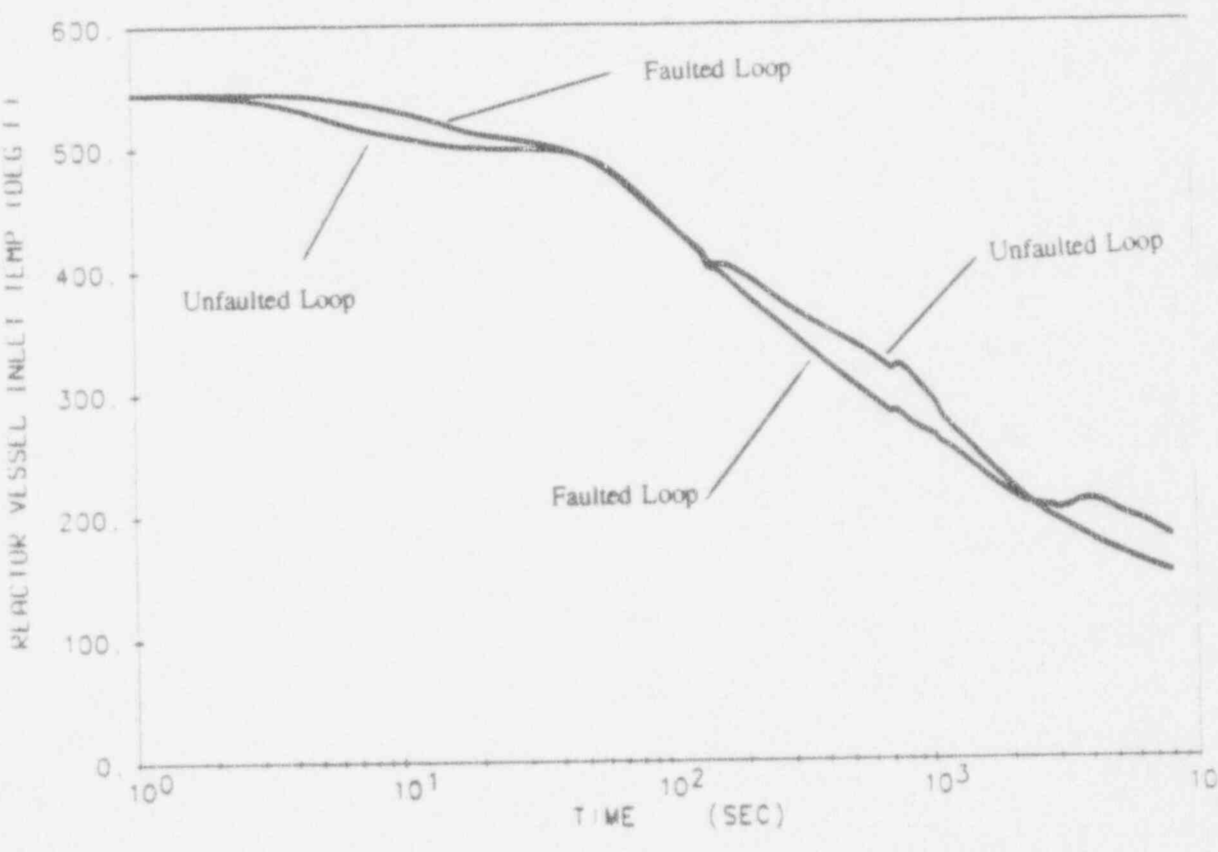


FIGURE 4.1-6
RCS PRESSURE TRANSIENT, STEAM SYSTEM PIPING FAILURE

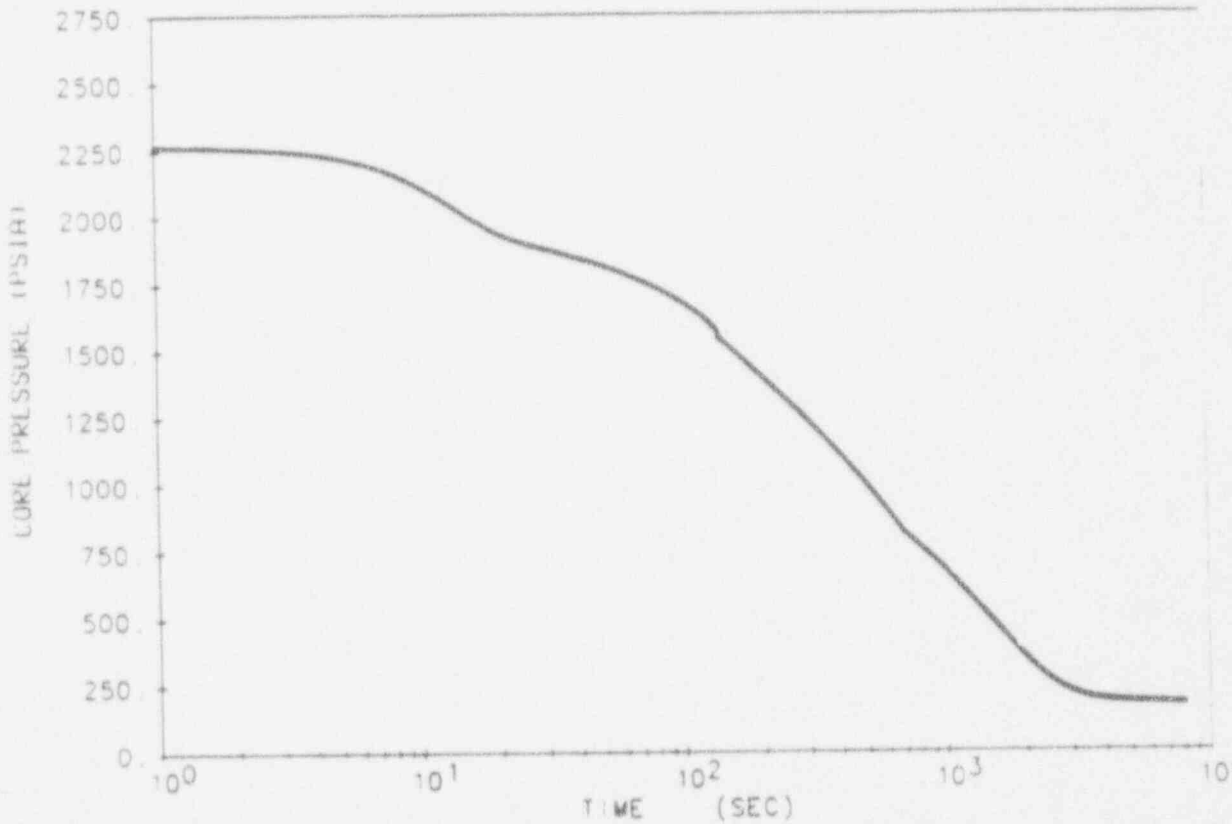


FIGURE 4.1-7
PRESSURIZER WATER VOLUME TRANSIENT, STEAM SYSTEM PIPING FAILURE

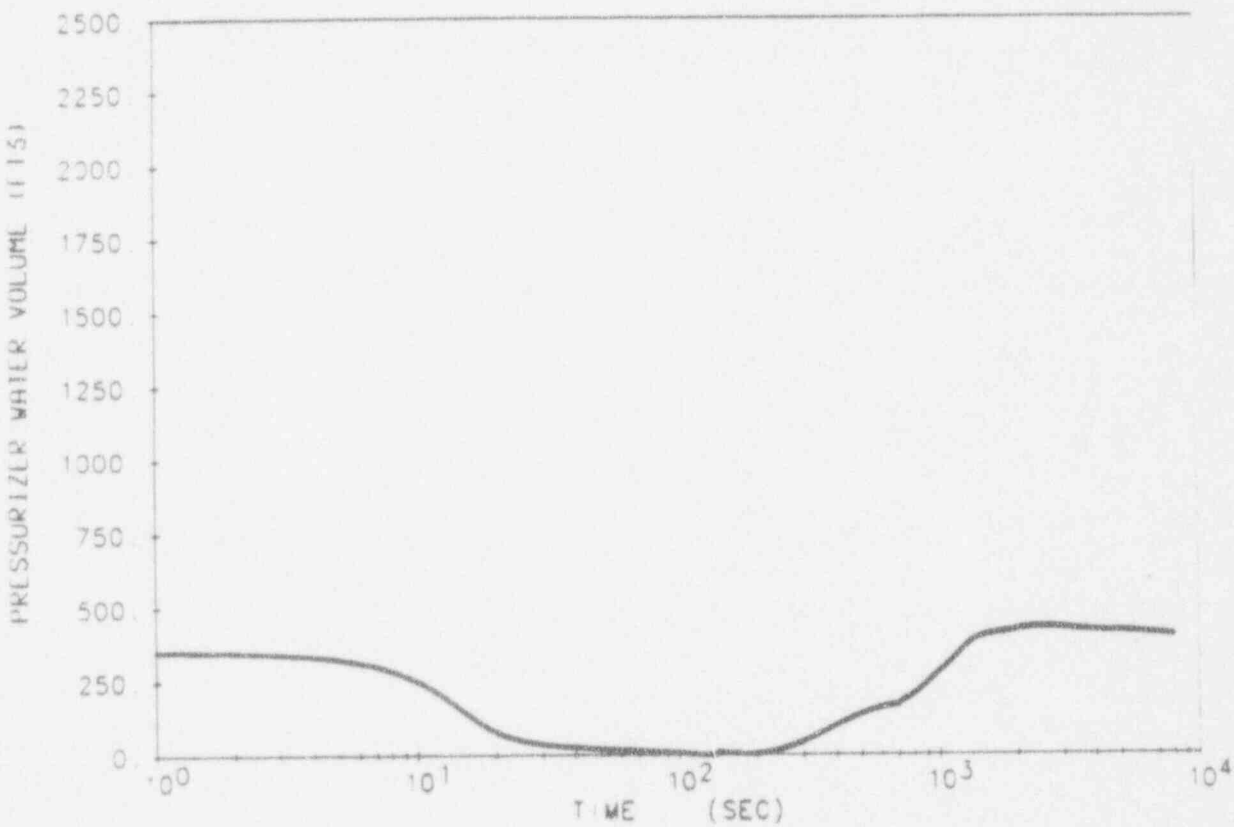


FIGURE 4.1-8
CORE FLOW TRANSIENT, STEAM SYSTEM PIPING FAILURE

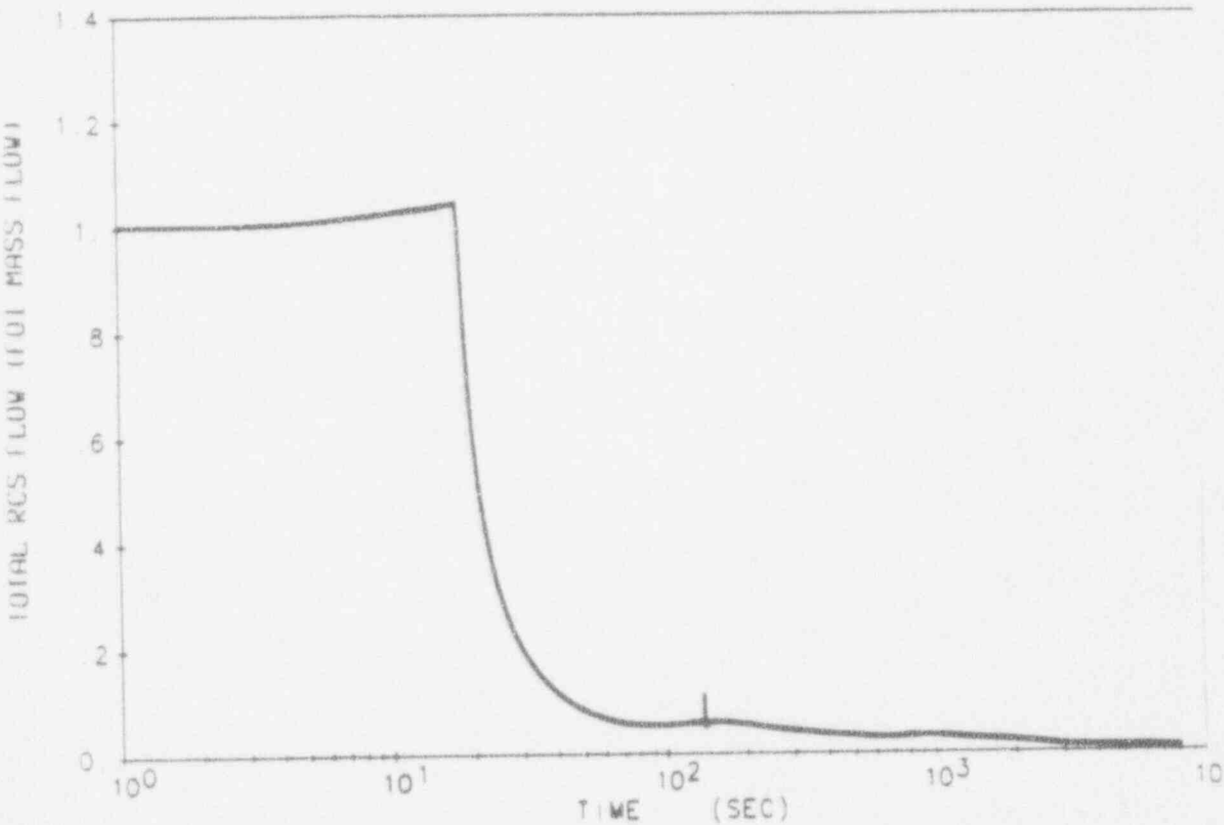


FIGURE 4.1-9
FEEDWATER FLOW TRANSIENT, STEAM SYSTEM PIPING FAILURE

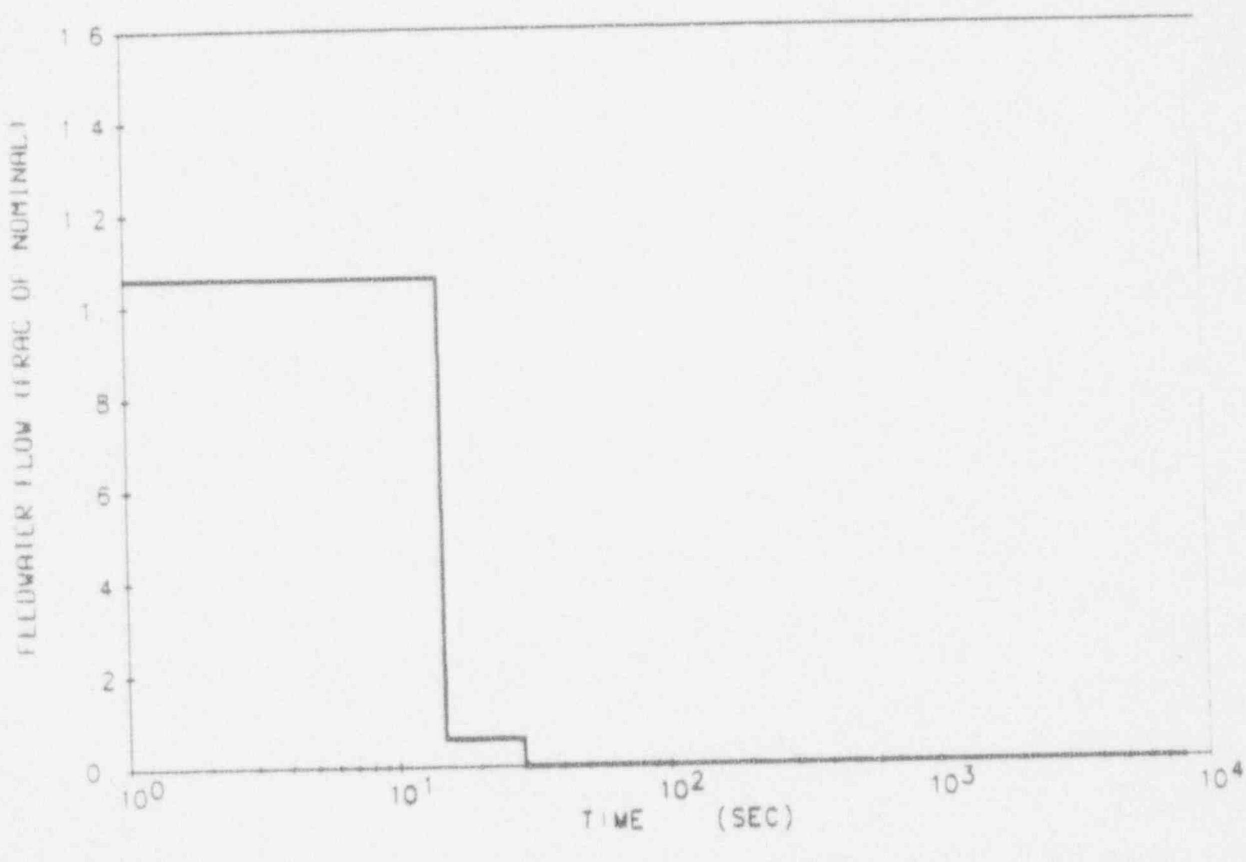


FIGURE 4.1-10
CORE BORON TRANSIENT, STEAM SYSTEM PIPING FAILURE

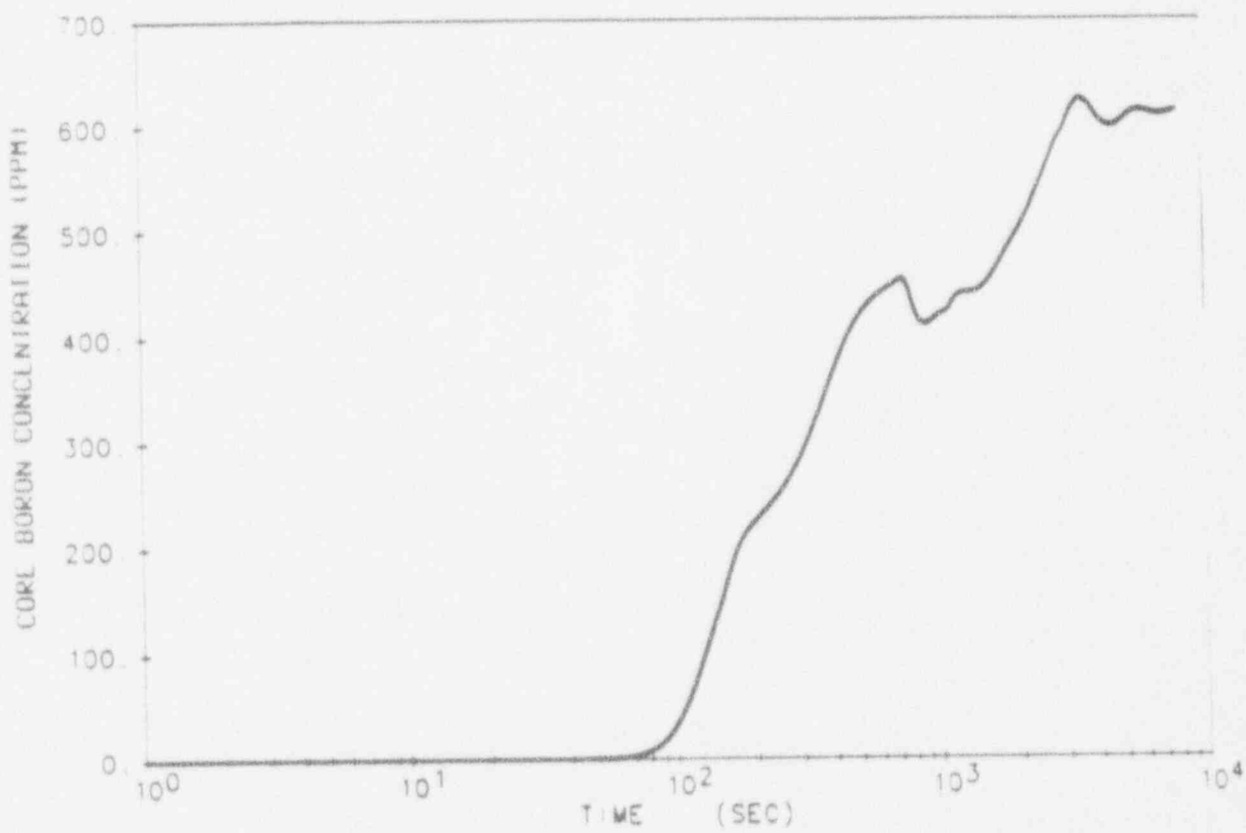


FIGURE 4.1-11
STEAM PRESSURE TRANSIENT, STEAM SYSTEM PIPING FAILURE

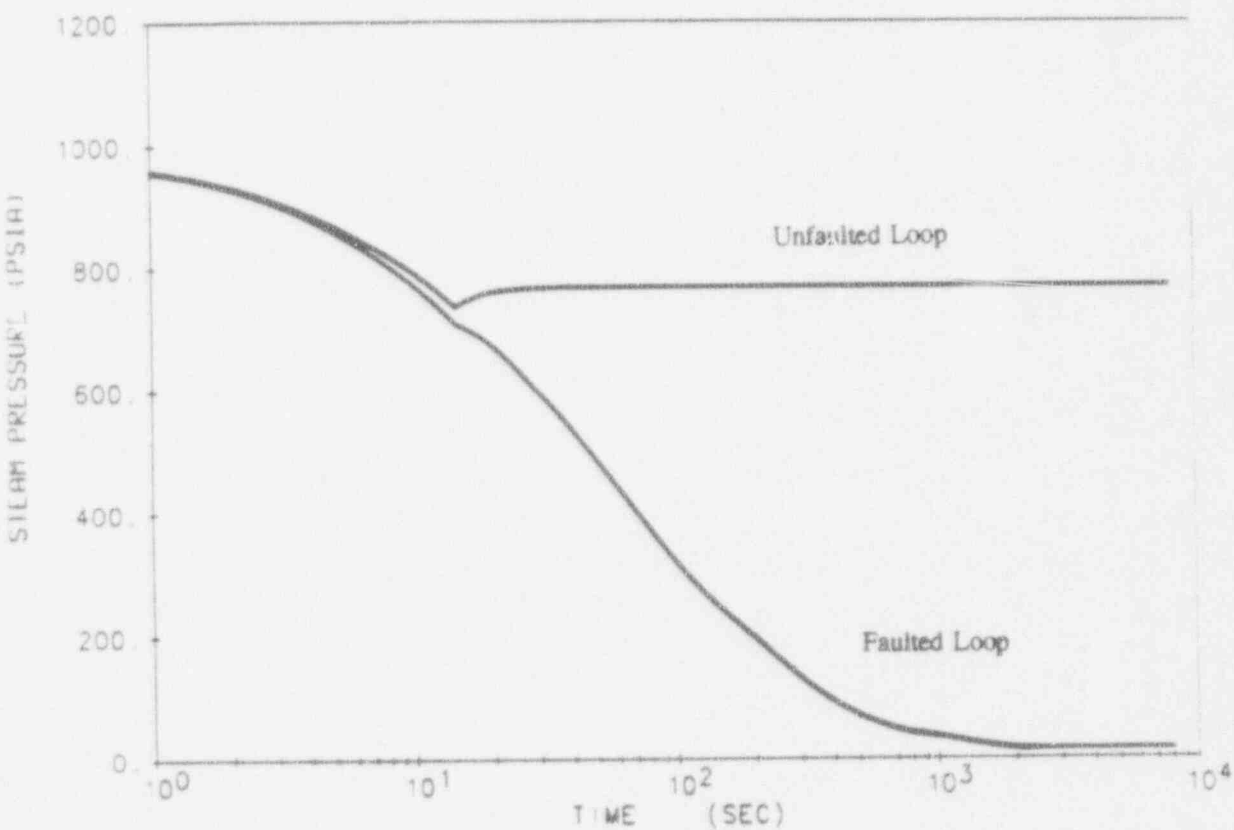


FIGURE 4.1-12
STEAM FLOW TRANSIENT, STEAM SYSTEM PIPING FAILURE

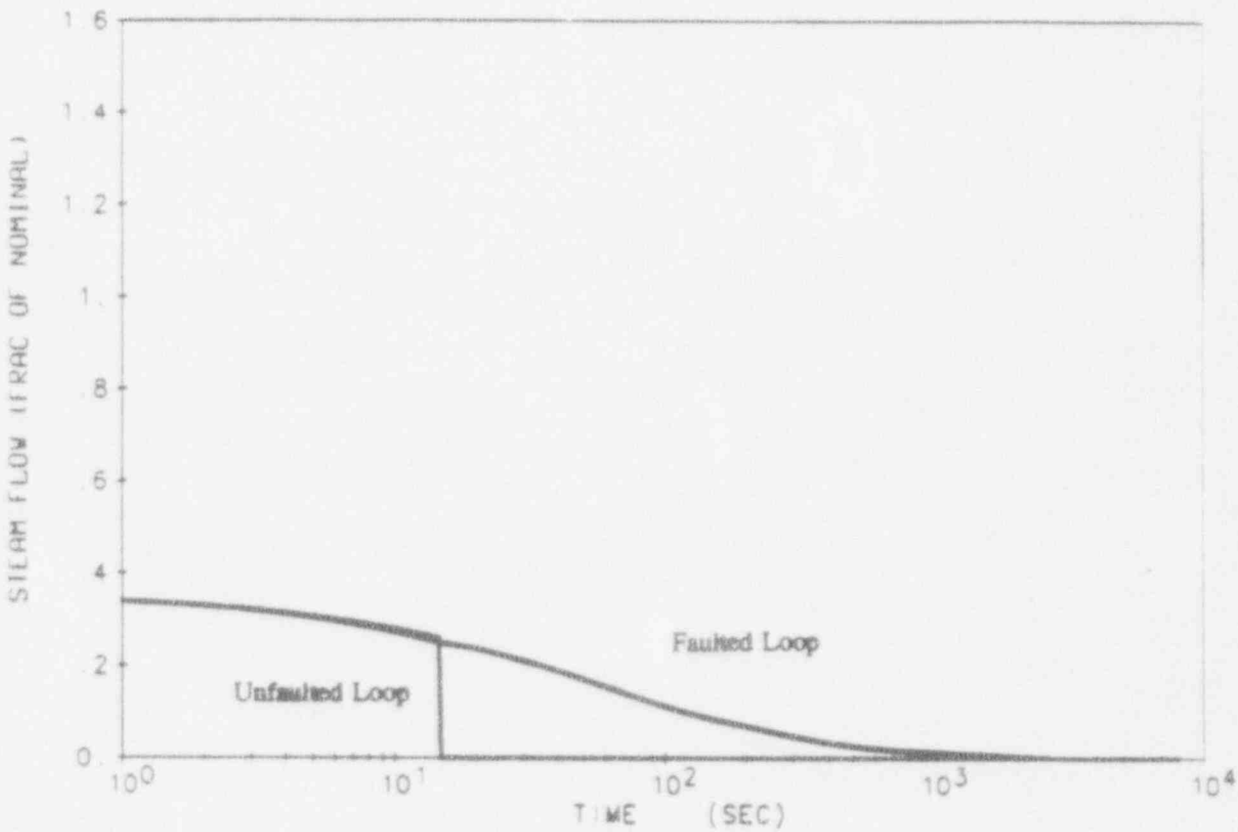


FIGURE 4.1-13
CORE MAKEUP TANK INJECTION FLOW, STEAM SYSTEM PIPING FAILURE

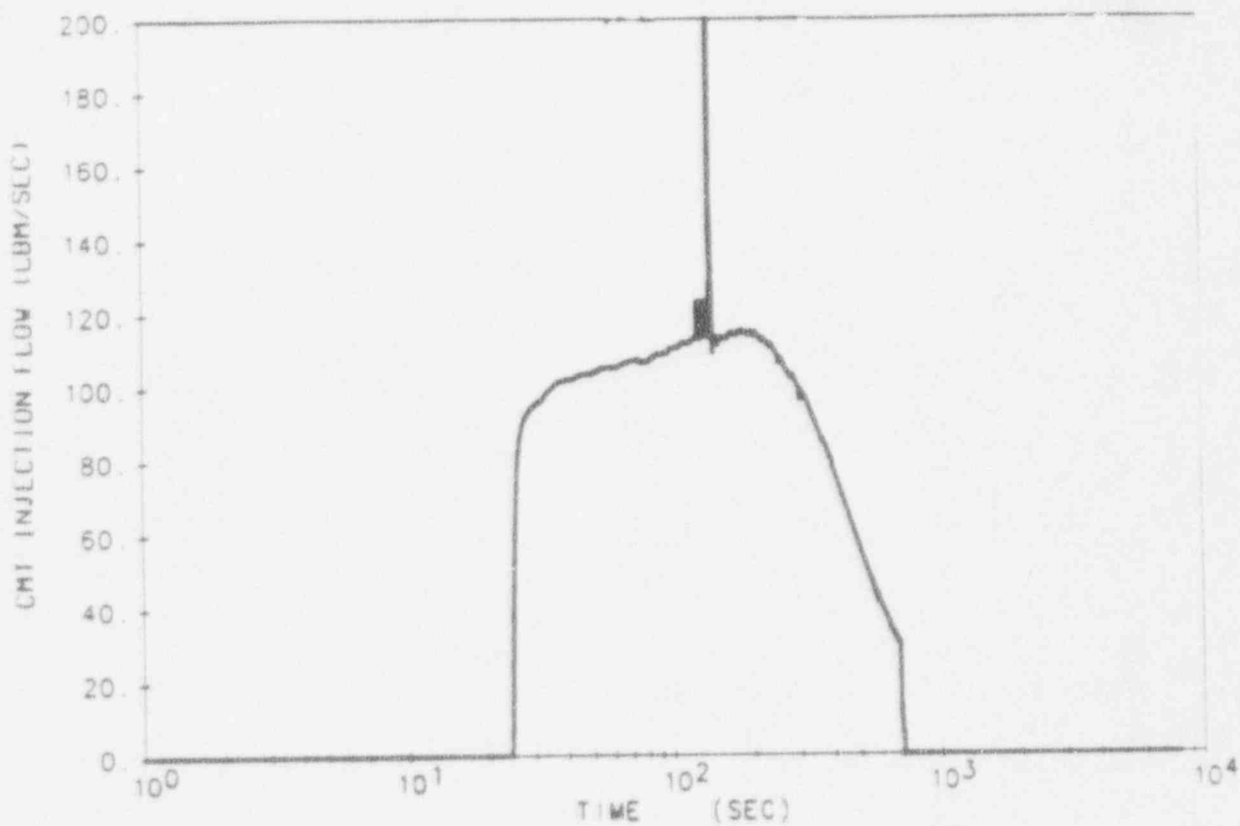
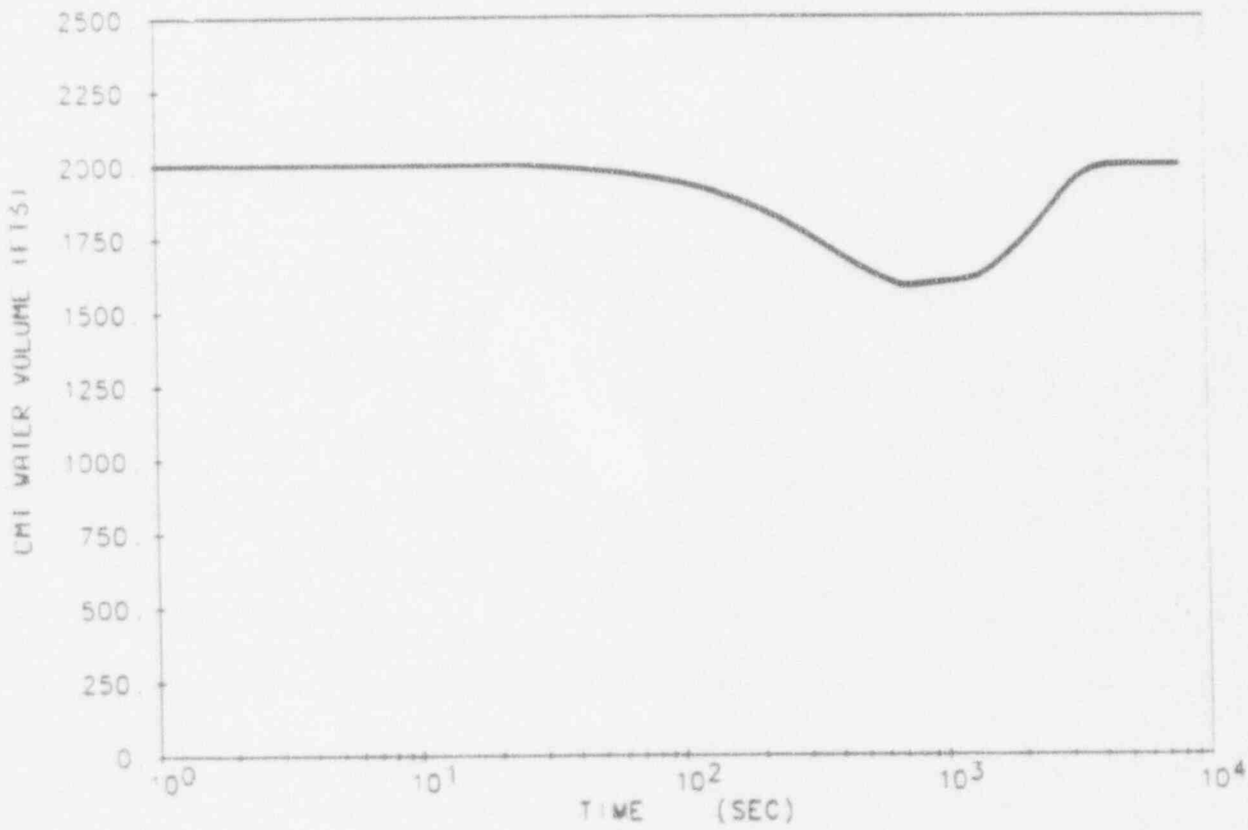


FIGURE 4.1-14
CORE MAKEUP TANK WATER VOLUME, STEAM SYSTEM PIPING FAILURE



4.2 Steam Generator Tube Rupture

Introduction

Section 15.6.3 of the AP600 SSAR presented the design basis SGTR event and the corresponding radiological dose releases for the AP600. The results show that the overfill protection logic and the passive system design features will initiate automatic actions that stabilize the RCS in a safe condition while preventing steam generator overfill and ADS actuation. The resulting radiological doses released to the environment for this accident are within the 10 CFR 100 limits.

The impact of the PXS design changes on the progression of a SGTR event has been evaluated. The following aspects of a SGTR event were included in this evaluation:

- offsite radiological dose analysis,
- margin to ADS actuation, and
- margin to SG overfill.

SGTR Evaluation

The identified design changes that impact the SGTR transient are:

1) initiation of the PRHR on a CMT actuation signal, rather than on a high-high steam generator level signal,
2) changes to the CVS control logic,
3) incorporation of a diffuser at the inlet to the CMT,
4) addition of a pressurizer heater shutoff on a CMT actuation signal, and
5) lower CMT level setpoint for ADS actuation.

Initiation of the PRHR on a CMT actuation signal, rather than on a high-high steam generator signal as presented in the SSAR, impacts the timing of the various phases of the transient, as well as the magnitude of break flow and steam release (the major factors used in determining the offsite radiological doses). The design basis SGTR analysis was reanalyzed with the PRHR initiated on a CMT actuation signal rather than on a high-high steam generator level signal. The system responses for this transient are shown in Figures 4.2-1 to 4.2-8. The figures parallel the figures in the SSAR, described in Section 15.6.3.2.1.3. The results of this analysis show that the overall progression of the transient is not changed significantly. This is demonstrated in Table 4.2-1, where the sequence of events for the SSAR transient (with PRHR initiated on high-high steam generator level) is compared to the same transient with the PRHR initiated on a CMT actuation signal, and in Figure 4.2-9, where break flow (the major indicator of the progression of the transient) is shown for the two transients. This analysis demonstrated a reduction in break flow and steam releases, the major contributors to the offsite radiological dose calculation. It is concluded that the change in PRHR initiation signal will result in reduced offsite radiological doses.

This reanalysis conservatively does not include the change to the CVS. The change to the CVS makeup program, if modeled, would provide a benefit via reduced primary to secondary break flow resulting from reduced injection from the CVS.

For the SSAR, SGTR sensitivity analyses were performed to demonstrate that the ruptured steam generator will not overfill. This conclusion has been verified for the identified design changes. There are several reasons for this conclusion. Most significantly, the modified CVS control logic and the revised PRHR actuation logic will reduce break flow throughout the transient and cause earlier initiation of the RCS cooldown and increased margin to overfill.

Separate sensitivity analyses have been performed to verify that with the identified changes a design basis SGTR would not result in ADS actuation. These analyses are performed with conservative assumptions that reduce margin to the low CMT level ADS actuation setpoint. The key assumptions include no CVS makeup, reduced CMT condensation, and the failure of the faulted steam generator power-operated relief valve (PORV) in the open position. Neglecting the CVS makeup flow tends to reduce the RCS pressure which increases the CMT injection. Reducing the CMT condensation will enhance CMT injection while reducing makeup flow into the CMT from the pressurizer connection line, thereby maximizing the level reduction in the CMT. For this study the nominal CMT condensation predicted by the LOFTTR2 code model has been reduced by a factor of 10 to provide assurance that the analysis is bounding. The failure of the faulted SG PORV results in further RCS depressurization, as the secondary pressure drops below the safety valve setpoint. The results show that the minimum CMT water volume predicted from the sensitivity analysis is well above the low CMT level ADS setpoint (1350 ft^3).

The addition of a pressurizer heater block, either automatically on a CMT actuation signal or manually, provides a benefit for SGTR offsite dose and overfill analyses by reducing heat addition to the pressurizer, thus allowing for quicker cooldown and depressurization.

Conclusion

From the results of the SGTR analyses it is concluded that the design changes will not result in steam generator overfill or ADS actuation, and that the offsite radiological doses for the limiting design basis SGTR reported in the SSAR continue to be bounding.

TABLE 4.2-1
SEQUENCE OF EVENTS
SGTR WITH PRHR INITIATED ON A CMT ACTUATION SIGNAL COMPARED TO
SSAR ANALYSIS WITH PRHR INITIATED ON HIGH SG LEVEL

Event	SSAR PRHR on High SG Level Time (seconds)	PRHR on CMT Actuation Signal Time (seconds)
Double ended SGTR	0	0
One CVS pump actuated and pressurizer heater turned on	0	0
Reactor trip on low pressurizer pressure	1074	1074
Main feedwater pumps assumed to trip and begin coastdown	1074	1074
Startup feedwater initiated (includes maximum delay)	1150	1150
CMT actuation signal on low-1 pressurizer pressure	1411	1411
Additional CVS pump initiated	1412	1412
CMT injection begins (includes maximum delay)	1433	1433
PRHR initiated on CMT actuation signal (includes maximum delay)	-------	1471
Startup feedwater to faulted SG throttled to maintain low SG narrow range level setpoint	2526	2472
Faulted SG PORV fails open when secondary level approximately reaches high-2 SG narrow range level setpoint	3706	3742
Faulted SG PORV block valve closes on low steam line pressure signal (includes valve delay tine)	4346	4000
Steam release from faulted and intact SG PORVs terminated	4347	4001
CVS and startup feedwater pumps isolated on high-2 SG narrow range level setpoint	5174	4576
PRHR initiated on high-2 SG narrow range level setpoint (includes valve opening delay time)	5234	-------
Break flow terminated and stable condition reached	10000	10000

FIGURE 4.2-1
PRESSURIZER WATER VOLUME FOR
SGTR ANALYSIS WITH PRHR INITIATED ON A CMT ACTUATION SIGNAL

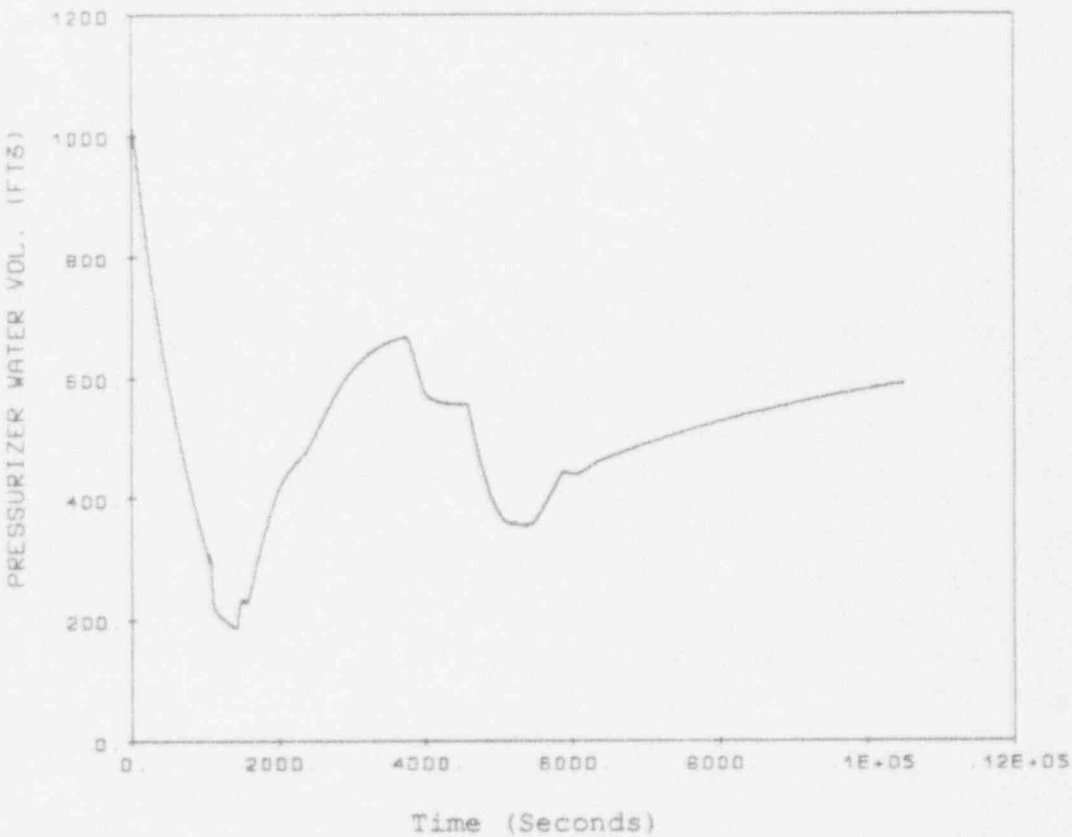


FIGURE 4.2-2
PRIMARY PRESSURE FOR
SGTR ANALYSIS WITH PRHR INITIATED ON A CMT ACTUATION SIGNAL

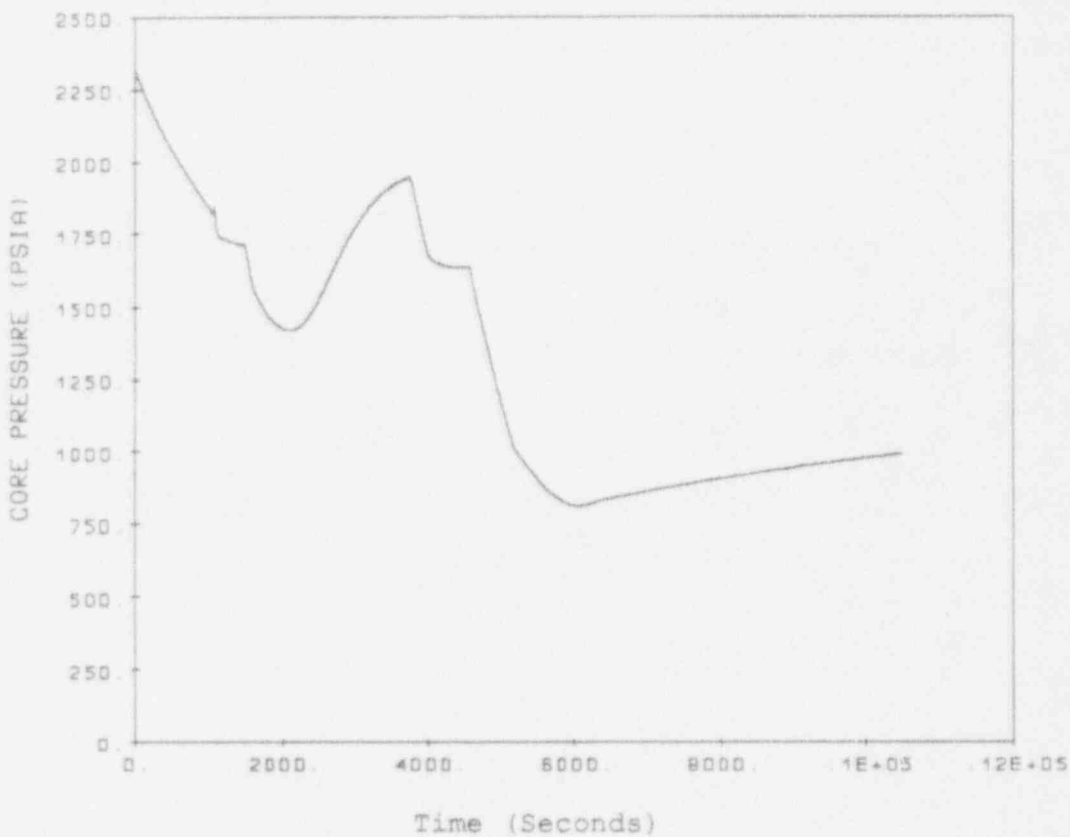


FIGURE 4.2-3
SECONDARY PRESSURE FOR
SGTR ANALYSIS WITH PRHR INITIATED ON A CMT ACTUATION SIGNAL

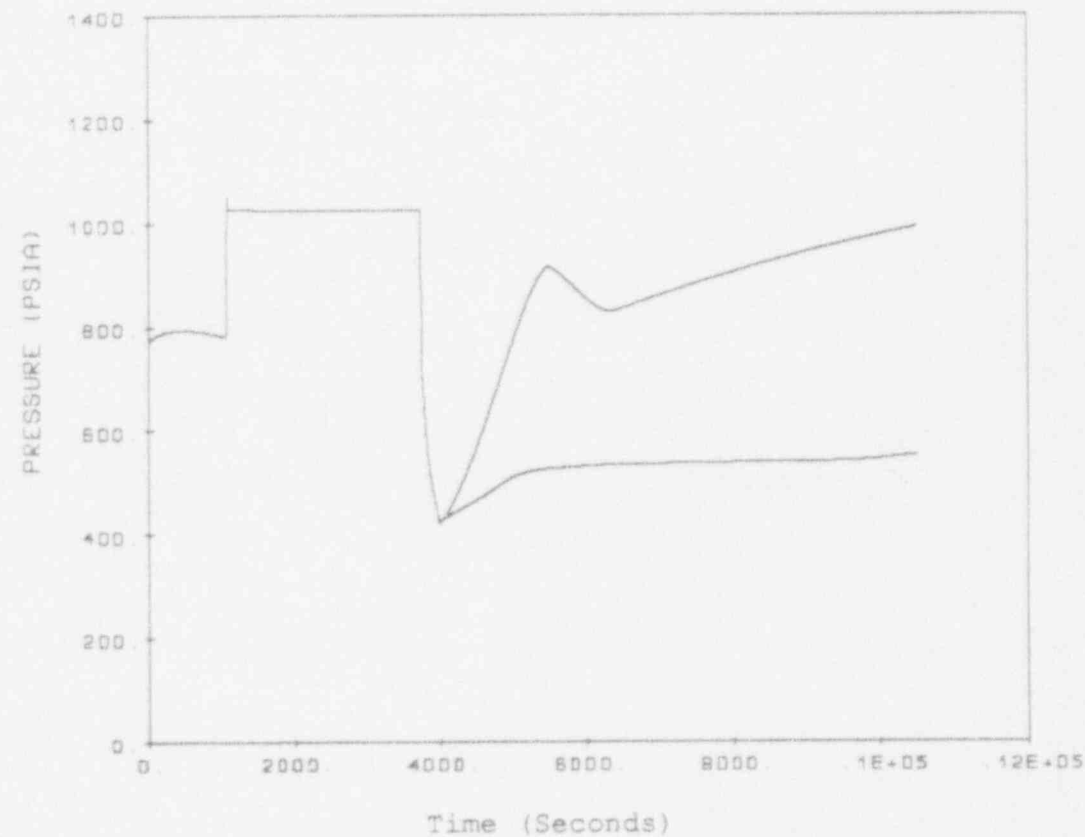


FIGURE 4.2-4
PRIMARY TO SECONDARY BREAK FLOW FOR
SGTR ANALYSIS WITH PRHR INITIATED ON A CMT ACTUATION SIGNAL

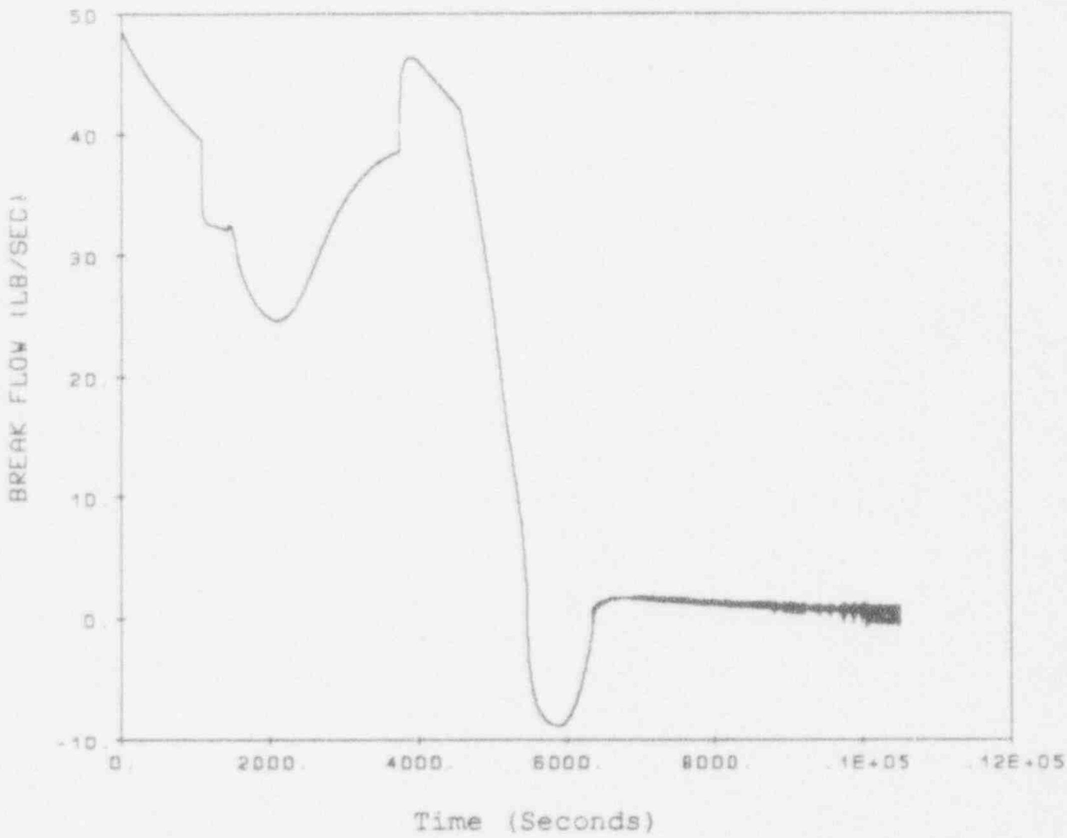


FIGURE 4.2-5
RUPTURED STEAM GENERATOR WATER VOLUME FOR
SGTR ANALYSIS WITH PRHR INITIATED ON A CMT ACTUATION SIGNAL

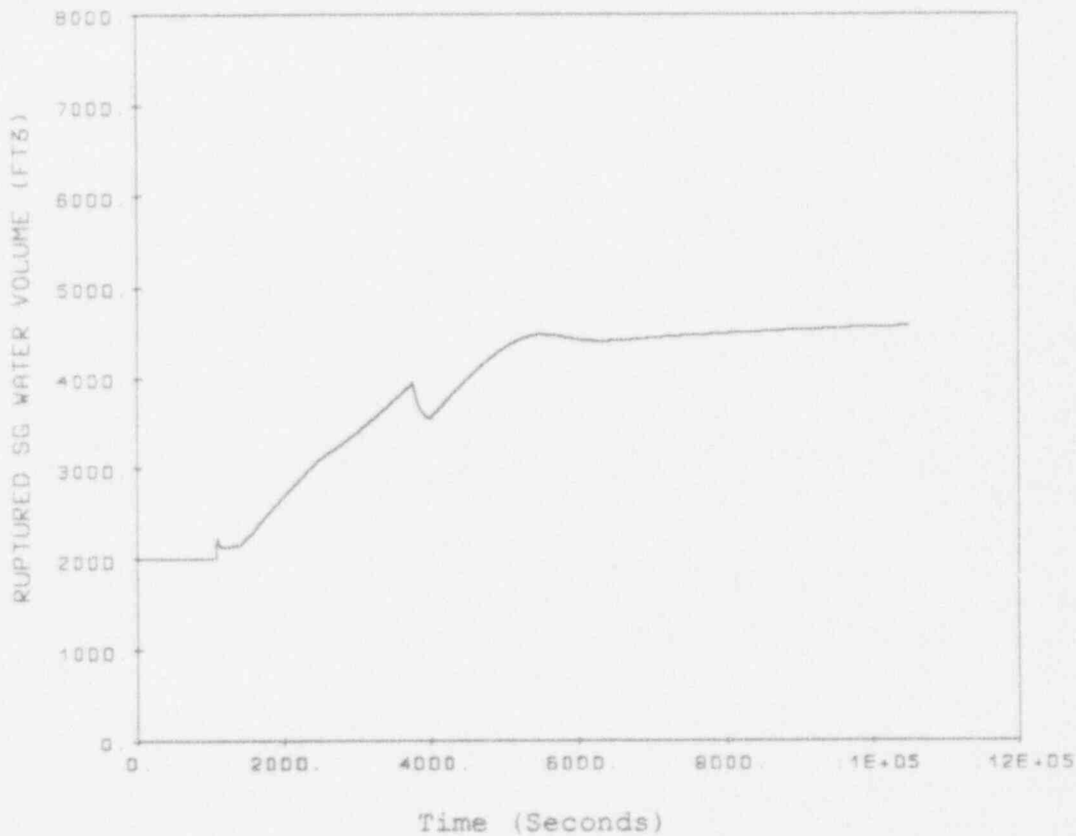


FIGURE 4.2-6
RUPTURED STEAM GENERATOR STEAM RELEASE FOR
SGTR ANALYSIS WITH PRHR INITIATED ON A CMT ACTUATION SIGNAL

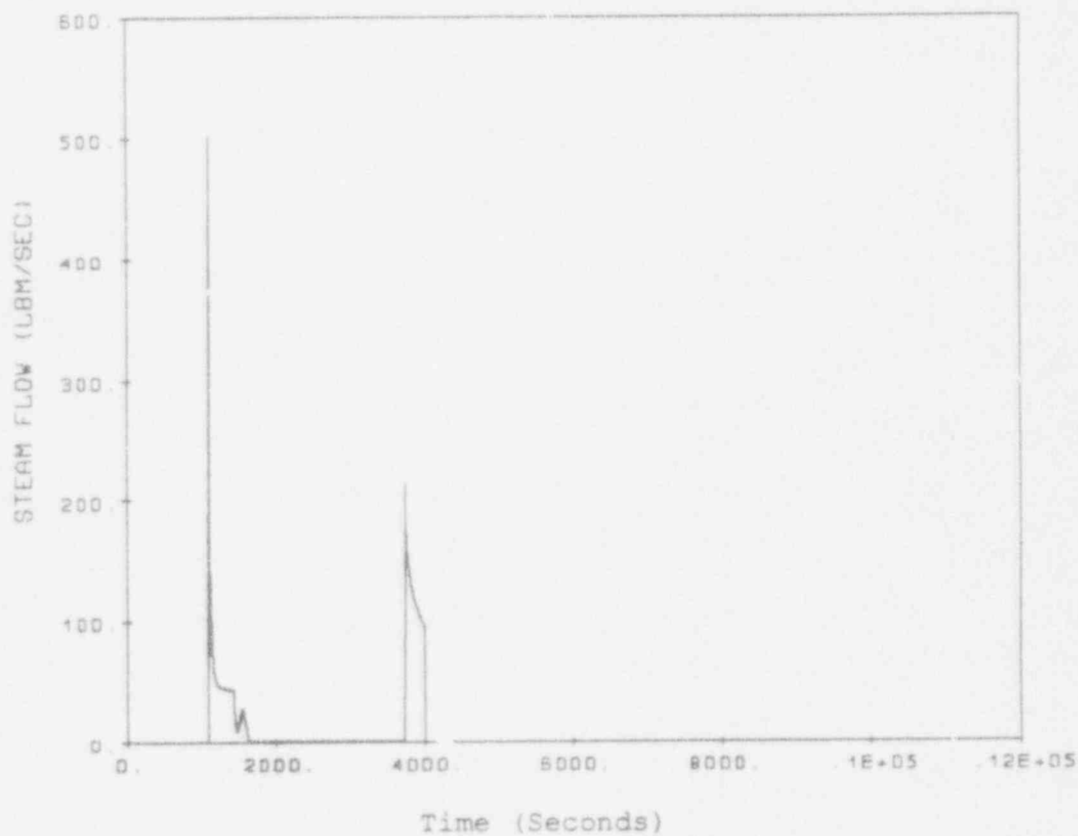


FIGURE 4.2-7
RUPTURED LOOP CVS AND CMT INJECTION FLOW FOR
SGTR ANALYSIS WITH PRHR INITIATED ON A CMT ACTUATION SIGNAL

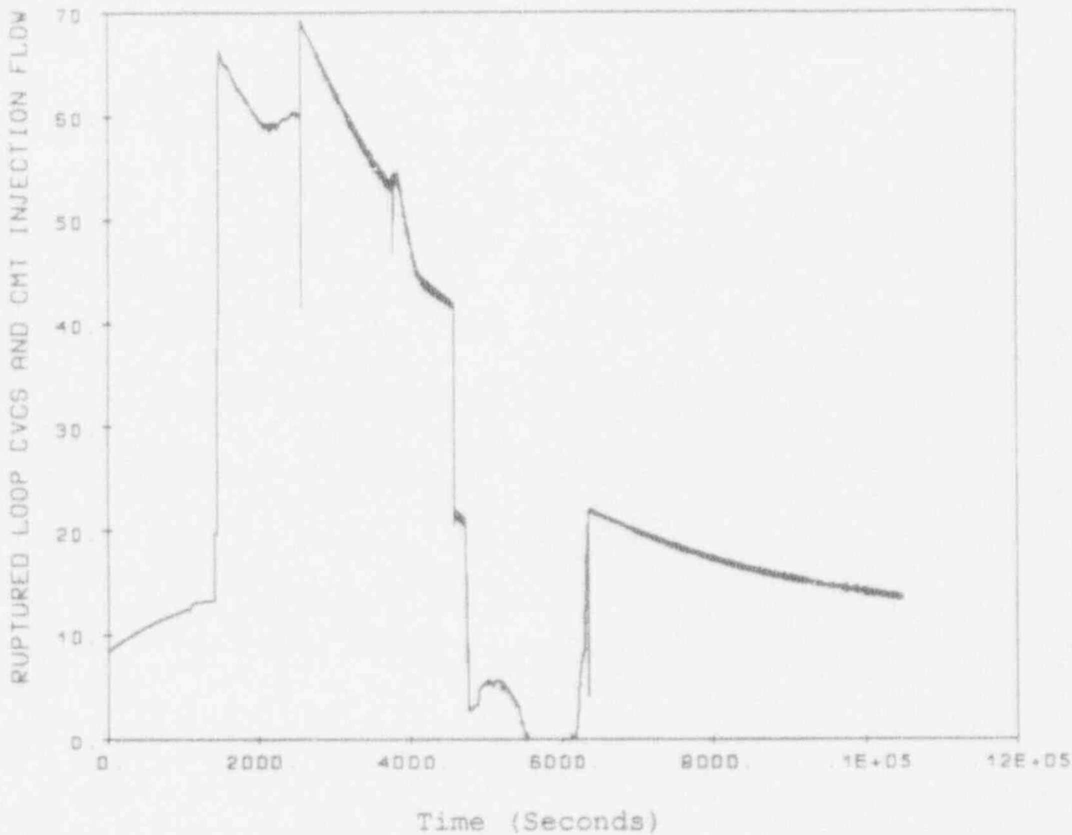


FIGURE 4.2-8
CMT WATER VOLUME FOR
SGTR ANALYSIS WITH PRHR INITIATED ON A CMT ACTUATION SIGNAL

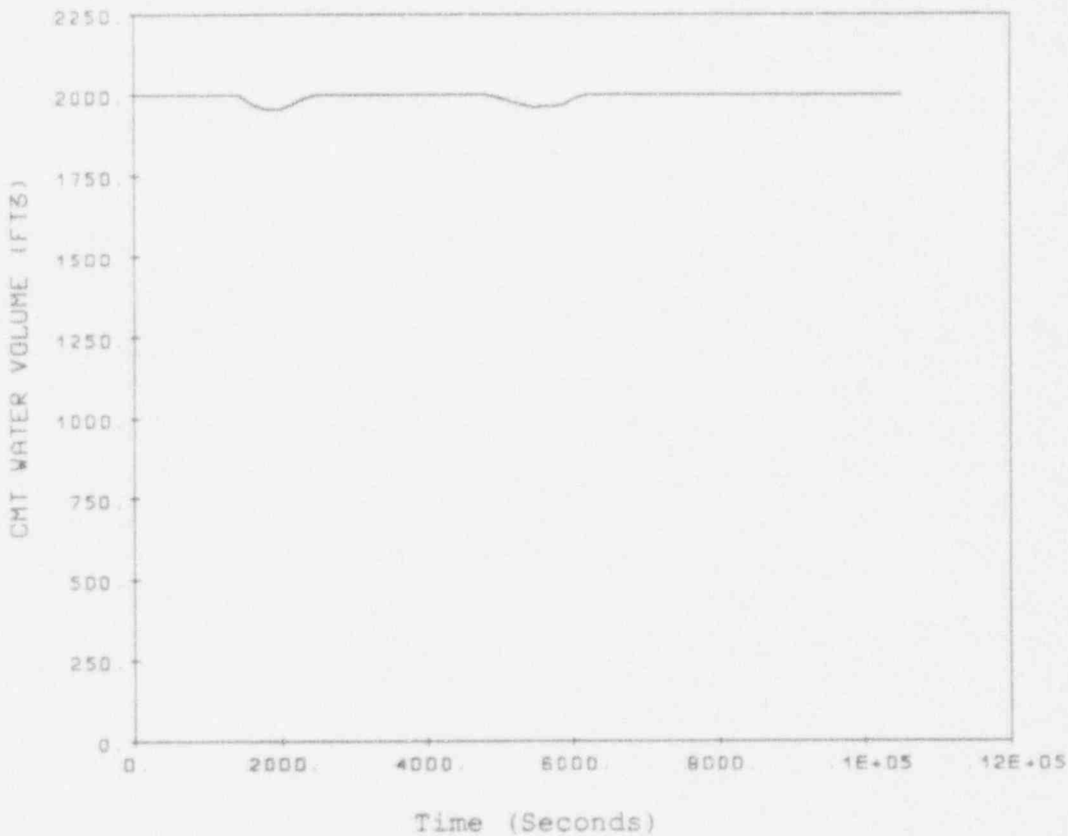
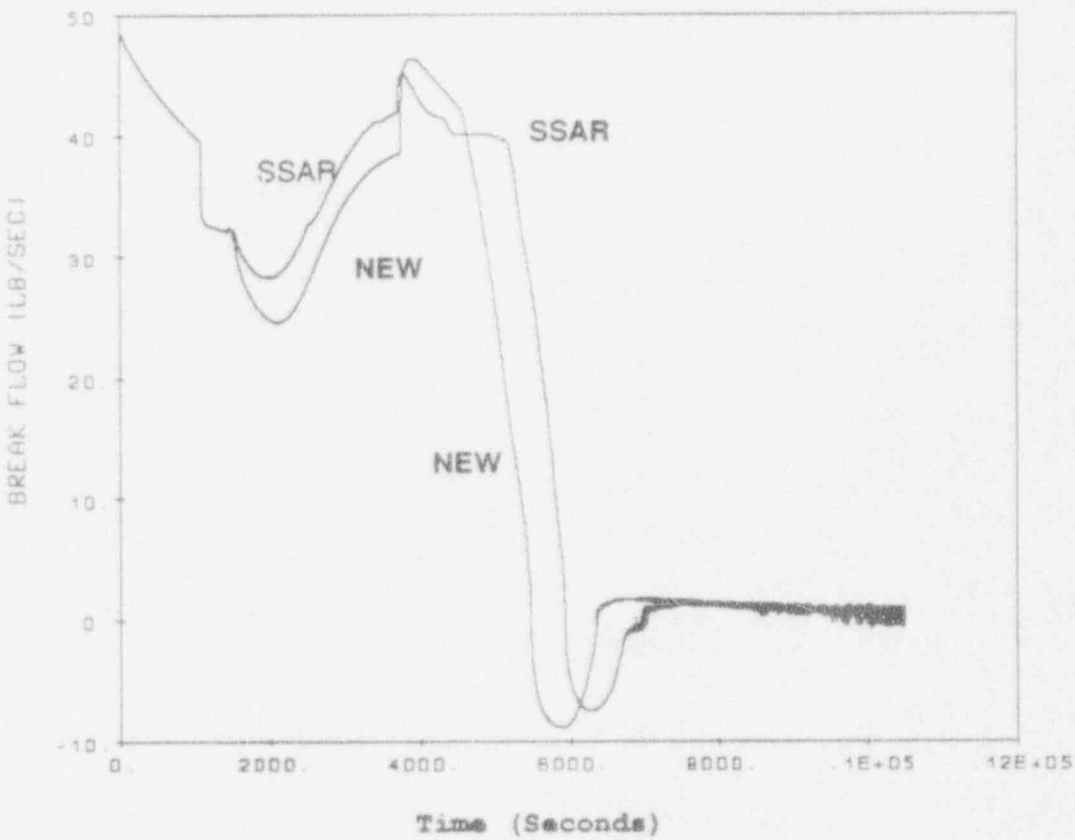


FIGURE 4.2-9
PRIMARY TO SECONDARY BREAK FLOW FOR
SGTR ANALYSIS WITH PRHR INITIATED ON A CMT ACTUATION SIGNAL
COMPARED TO THE SSAR ANALYSIS WITH PRHR INITIATED ON HIGH SG LEVEL



Westinghouse

4.3 Loss of Coolant Accidents

Introduction

The AP600 SSAR LOCA analyses are a spectrum of postulated break sizes ranging from a one-inch equivalent diameter break to double-ended hot leg (DEHLG) and cold leg (DECLG) guillotine breaks. The DECLG cases exhibit the limiting calculated peak cladding temperature (PCT) values. In these large break LOCA cases, minimal CMT injection occurs before PCT is calculated, and the ADS does not actuate until long after PCT occurrence. Thus, large break LOCA performance is unaffected by these design changes.

Two of the small break LOCA cases were chosen for reanalysis to determine the impact of the passive safeguards systems design changes. LOCA analyses for postulated double-ended direct vessel injection line (DEDVI) break and inadvertent ADS actuation events have been performed to investigate the CMT and ADS design changes. The DEDVI and inadvertent ADS cases represent limiting small break LOCAs with respect to providing safety injection delivery to limit core uncovery and ADS depressurization capability to achieve IRWST injection, respectively. The NOTRUMP computer code (References 7.3 and 7.4), used in the SSAR analysis, was utilized. Only safety systems were modeled.

The NOTRUMP AP600 input was set for the SSAR analyses to comply with the standard Westinghouse Small Break LOCA Evaluation Model methodology (Reference 7.4). For better representation of the AP600, the following changes were made to the SSAR model:

1) The top node of the CMT was increased from 10 percent to 20 percent of the CMT volume. Since the larger top node is expected to bound the CMT inlet mixing region with a diffuser in place, the 20 percent top node volume is selected as appropriate and conservative.

2) The double-link horizontal stratified flow links are no longer used for the surge line connections. Rather, single links are adopted because they are more appropriate for the surge line flow path.

3) The design changes to safety-related passive safety features are modeled. The change to the CVS controls does not affect the SSAR LOCA analyses.

4) A multi-node PRHR representation of the heat exchanger (eight fluid nodes) is used. PRHR HX actuation occurs on a Safeguards ("S") signal. Standard condensation heat transfer correlations are applied when primary side steam condenses in the PRHR.

The DEDVI case exhibits core uncovery in the SSAR analysis. The actuation of ADS depressurizes the RCS to accumulator actuation pressure, thereby increasing safety injection flow. Therefore, the limiting single active failure for the DEDVI break is taken as failure of a set of one first stage and one third stage ADS valves to open on demand. For the inadvertent ADS case, reducing the venting capability of the RCS constitutes the limiting single failure. Thus, in the inadvertent ADS case, failure of one of the four fourth-stage ADS valves to open on demand is postulated. Furthermore, in both of the cases the fourth stage ADS vent area was arbitrarily reduced by 15 percent to provide a bounding calculation.

Double-ended Direct Vessel Injection Line Break (DEDVI) Results

This case models the double-ended rupture of the DVI line at the nozzle into the downcomer. The broken loop injection system (consisting of an accumulator, a CMT, and an IRWST delivery line) is modeled to spill completely out of the break. The injection line break evaluates the ability of the plant to recover from a moderately large break with only half of the total Emergency Core Cooling System (ECCS) capacity available. The venturi installed in the AP600 DVI nozzles provides a four-inch diameter flow restriction, which has been modeled in NOTRUMP. A discussion of the results follows, including comparisons with the SSAR DEDVI case in Table 4.3-1. The AP600 system depressurization and break flow rates are significantly lower than in the SSAR case due to the smaller break area from the vessel downcomer through the venturi.

The break is assumed to open instantaneously at 0 seconds. The accumulator on the broken loop starts to discharge via the DVI line to the containment. The subcooled discharge from the downcomer nozzle (Figure 4.3-1) through a four-inch diameter venturi causes a rapid RCS depressurization (Figure 4.3-2), and a reactor trip signal is generated at 7.2 seconds. The "S" signal is generated at 8.7 seconds and following a 1.2 second delay, the isolation valves on the CMT tank delivery and cold leg balance lines begin to open. The "S" signal also causes closure of the main feedwater isolation valves after a five second delay and trips the reactor coolant pumps after a 16.2 second delay. The opening of the PRHR isolation valve on an "S" signal starts the flow through the heat exchanger. The broken loop CMT discharges directly to the containment (Figure 4.3-3), and a small circulation flow provides injection from intact loop CMT (Figure 4.3-4).

As the pressure falls, the RCS fluid saturates and at about 23 seconds (vs. 6 seconds in the SSAR case) a mixture level forms in the upper plenum and falls to the hot leg elevation (Figure 4.3-5). The upper parts of the RCS start to drain (Figures 4.3-6 and 4.3-7), and a mixture level forms in the downcomer at about 115 seconds (vs. 20 seconds in the SSAR) (Figure 4.3-8) and falls to the elevation of the break. Two-phase discharge then occurs from the downcomer side of the break (Figure 4.3-1). A comparison of the venturi and SSAR DEDVI cases is provided in Table 4.3-1.

At about 85 seconds (vs. 20 seconds in the SSAR) the fluid at the top of the broken loop CMT saturates and a level forms and starts to fall (Figure 4.3-9). The first stage ADS setpoint is reached at 140 seconds, and after an appropriate delay, one first stage path is opened. The ensuing steam discharge from the top of the pressurizer (Figure 4.3-10) increases the RCS depressurization rate. Because this depressurization is beneficial in delivering safety injection water, the single active failure assumed for the DEDVI is the failure of a first and third stage ADS valve to open.

In the SSAR DEDVI case, with no venturi present and a much higher break flow, at about 60 seconds insufficient liquid remains in the core and upper plenum to sustain the mixture level. The mixture level, therefore, starts to collapse as the core dries out and the mixture level falls to a minimum at 100 seconds, uncovering the core. In the current "design change" case, the two second stage ADS valves begin to open at 234 seconds, following the timer delay between the actuation of the first two stages of the ADS. At 206 seconds the intact loop accumulator starts to inject into the downcomer (Figure 4.3-11) causing the mixture level in the downcomer to slowly rise (Figure 4.3-8); the mixture level in the upper plenum never falls below the hot leg elevation during the entire DEDVI transient. One third stage ADS valve opens at 354 seconds because of the time delay of 120 seconds for the actuation of this stage of the ADS. At 193 seconds the broken loop CMT level reaches the fourth stage ADS setpoint but the fourth stage ADS valves do not open until 474 seconds because the minimum time delay is 120 seconds between the actuation of the final two stages of the ADS. Two-phase discharge ensues through the fourth stage path (Figures 4.3-12 and 4.3-13). By 229 seconds the broken loop CMT empties (Figure 4.3-9).

At about 304 seconds the fluid at the top of the intact loop CMT saturates, and the mixture level in the tank starts to fall slowly (Figure 4.3-13). CMT and accumulator injection after 304 seconds causes the downcomer mixture level to rise slowly (Figure 4.3-8). The level in the upper plenum is maintained up to at least the hot leg elevation (Figure 4.3-5) throughout the break transient. After the accumulator empties at about 570 seconds stable, but decreasing, injection continues from the intact loop CMT as the RCS pressure declines slowly. At 1178 seconds, the intact loop CMT has not yet emptied (Figure 4.3-13), yet the RCS pressure has fallen to the point that IRWST injection begins. Stable injection from the tanks occurs at a rate of about 120 lb/s until the CMT empties. This flow is greater than the break and ADS flows, resulting in a slow rise in RCS inventory (Figure 4.3-14). The RCS inventory continues to increase when the IRWST provides the only injection. The minimum RCS mass inventory of 106,000 lbs is significantly greater than the corresponding SSAR DEDVI break value of 90,000 lbs.

Inadvertent Actuation of Automatic Depressurization System Results

An inadvertent ADS signal is spuriously generated and the first stage ADS valves open. The plant, which is operating at 100 percent power, is depressurized via the ADS alone. Only safety-related systems are assumed to operate in this analysis. The second and third stage ADS valves actuate based on the design time delays. At the 20 percent tank level, the fourth stage ADS valves, which are on the hot legs, receive signals to open. Three of the four fourth stage ADS paths are assumed to open; one of the paths fails to open as the assumed single active failure.

The scenario analyzed is the same inadvertent ADS actuation that is considered in the SSAR. The ADS stages 2 and 3 actuate according to the timer sequence. The sequence of events for the transient is given in a table of comparison with the SSAR case, Table 4.3-2.

The transient is initiated by the opening of the two first stage ADS paths. The total throat area of the valves is 9.2 square inches. Reactor trip, reactor coolant pump trip and safeguards signals are generated via the pressurizer low pressure signals with appropriate delays. Upon generation of the reactor trip signal the main steam isolation valves begin to close, after a 2 second delay. Five seconds after an "S" signal the main feedwater isolation valves begin to close. The opening of the ADS valves and the reduction in core power due to reactor trip causes the primary pressure to fall rapidly (Figure 4.3-15). Flow of fluid toward the open ADS paths causes the pressurizer to fill by about 30 seconds (Figure 4.3-16), and the ADS flow becomes two-phase. A level begins to form in the upper plenum and drops to the hot leg elevation (Figure 4.3-17) at about 100 seconds. The safeguards signal opens the valves isolating the CMTs and injection of cold water begins (Figure 4.3-18); the PRHR is also actuated by the "S" signal. The mixture level in the CMTs is constant until about 440 seconds, then the tanks begin to drain (Figure 4.3-19). The reactor coolant pumps begin to coastdown due to an automatic trip signal following a 16.2 second delay. The revised design, by actuating ADS Stages 2 and 3 via timers, has an impact on this postulated event. As the table indicates, second stage ADS actuation at 70 seconds accelerates the transient; the accumulators now empty earlier than in the SSAR analysis because the earlier ADS flow causes the primary pressure to fall rapidly (Figure 4.3-15). At about 200 seconds following the ADS actuation, enough mass has been discharged that a mixture level forms in the downcomer (Figure 4.3-20). CMT injection flow is diminished by accumulator flow; emptying of the accumulator results in increased CMT injection flow. The mixture level in the CMT falls steadily after about 600 seconds.

The levels in the CMTs eventually reach the fourth stage ADS setpoint. Vent paths opened from the hot legs begin discharging fluid. The increased depressurization reduces the flow from ADS Stages 1, 2 and 3 (Figures 4.3-21 and 22). The single active failure assumed is that one of the four fourth stage ADS valves fails to open, maximizing the resistance to depressurizing the RCS to achieve IRWST injection.

The reduced flow through ADS Stages 1-3 allows the pressurizer level to fall, and these stages begin to discharge only steam after 1900 seconds. By 1967 seconds the CMTs are empty and delivery ceases, and at 2176 seconds the RCS pressure has fallen enough to allow gravity drain from the IRWST to begin (Figure 4.3-23). The calculation was stopped with a quasi-steady state condition existing in the RCS with the IRWST delivery exceeding the ADS flows (which are removing the decay heat) and the RCS inventory slowly rising. Core uncovery does not occur, the upper plenum mixture level remaining well above the core elevation throughout (Figure 4.3-17). In the SSAR case, nine minutes elapsed after CMT empty time before IRWST injection could begin; during this time the core mixture level fell below the hot leg bottom elevation.

In the revised ADS design case, core mixture level remains within the hot leg boundary. Once the ADS valves become fully effective by virtue of venting steam alone rather than a two-phase mixture (Figures 4.3-24 and 25), IRWST injection begins within a matter of seconds. The minimum RCS mass inventory in this revised ADS design case remains above (Figure 4.3-26) the SSAR case minimum inventory value of 100,000 lbs mass throughout.

The inadvertent opening of the ADS transient analysis confirms that the minimum venting area of AP600 capably depressurizes the RCS to the IRWST delivery pressure. The ADS area is sufficient to depressurize the RCS even assuming the failure of one fourth stage ADS valve to open. Appendix K decay heat is applied, which conservatively over estimates the core steam generation rate. Even under these limiting conditions, IRWST injection is obtained readily, and the core remains covered such that no cladding heatup occurs.

Conclusions

The two limiting small break LOCA cases from the AP600 SSAR have been reanalyzed with NOTRUMP to investigate the ADS/CMT design changes. With the venturi modeled in the vessel injection line, no core uncovery is predicted for the DEDVI break case. The inadvertent ADS actuation case depressurizes the primary system much more rapidly than does the SSAR case, due to the use of timers for the second and third stage actuation. The increased capacity of the ADS fourth stage valves allows IRWST injection to be achieved more readily than in the SSAR inadvertent ADS case. Overall, these cases demonstrate that the CMT/ADS design changes improve plant performance with respect to the SSAR analysis for the identified bounding small break LOCA events.

TABLE 4.3-1 DEDVI BREAK SEQUENCE OF EVENTS TABLE

Case	SSAR	New design with venturi
Break open	0.0 seconds	0.0 seconds
Reactor trip signal	3.5 seconds	7.2 seconds
"S" signal	4.2 seconds	8.65 seconds
Reactor coolant pumps start to coast down	20.4 seconds	24.85 seconds
ADS stage 1 flow starts	75 seconds	164.2 seconds
Accumulator injection starts	117 seconds	206 seconds
ADS stage 2 flow starts	135 seconds	234.2 seconds
ADS stage 3 flow starts	255 seconds	354.2 seconds
ADS stage 4 flow starts	375 seconds	474.2 seconds
IRWST injection starts	1839 seconds	1178.7 seconds

TABLE 4.3-2 INADVERTENT ADS ACTUATION

Case	SSAR	New Design
ADS stage 1 flow starts	0.0 seconds	0.0 seconds
Reactor trip signal	28.5 seconds	25.6 seconds
"S"signal	33.5 seconds	28.97 seconds
Reactor coolant pumps start to coast down	49.7 seconds	45.2 seconds
Accumulator injection starts	633 seconds	204 seconds
PRHR Actuation	726 seconds	30.2 seconds
ADS stage 2 flow starts	916 seconds	70 seconds
Accumulator empty	1325 seconds	565 seconds
ADS stage 3 flow starts	1390 seconds	190 seconds
ADS stage 4 flow starts	1902 seconds	1617 seconds
Core make up tank empty	2260 seconds	1967 seconds
IRWST injection starts	2792 seconds	2176 seconds

FIGURE 4.3-1
BREAK LIQUID FLOW, DEDVI BREAK

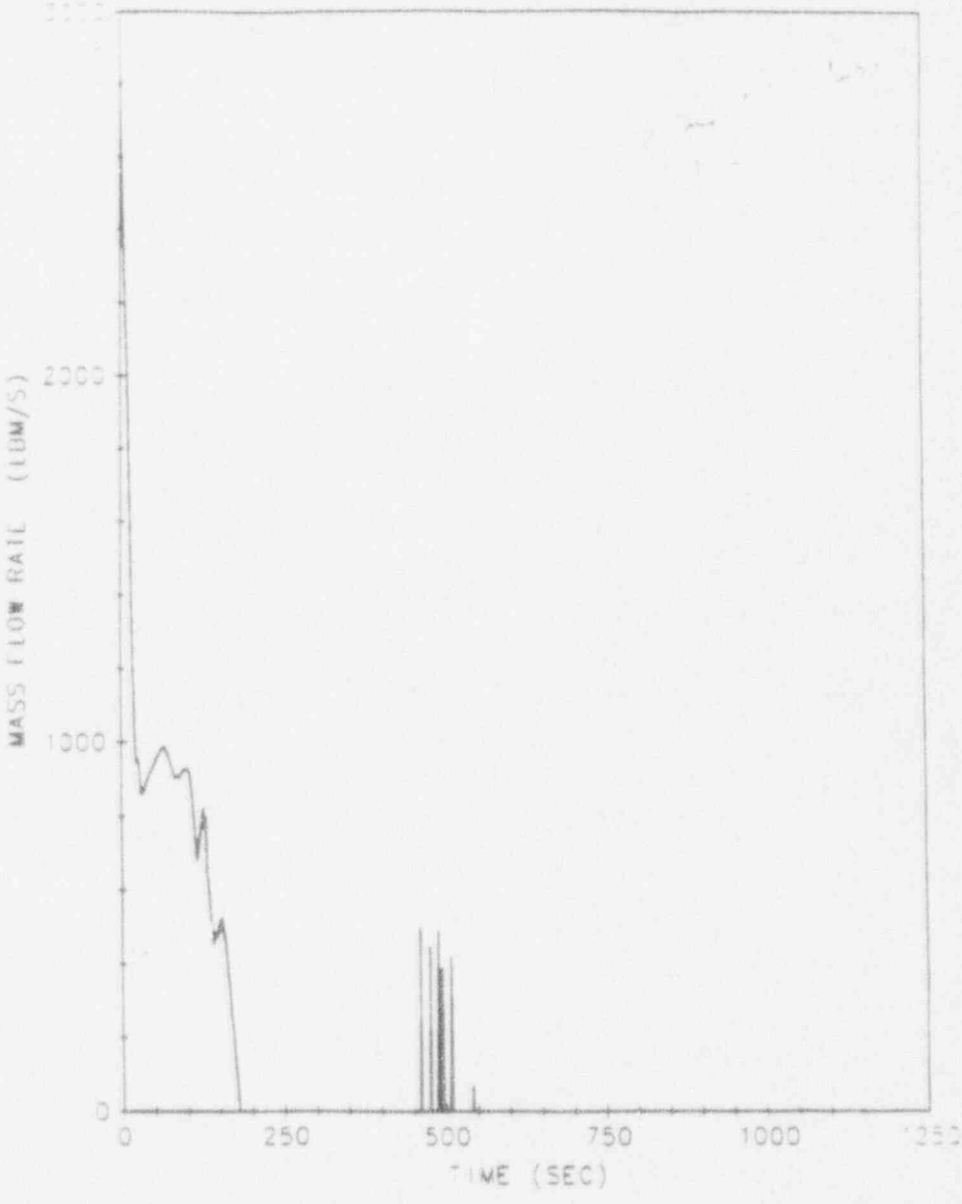


FIGURE 4.3-2
UPPER PLENUM PRESSURE, DEDVI BREAK

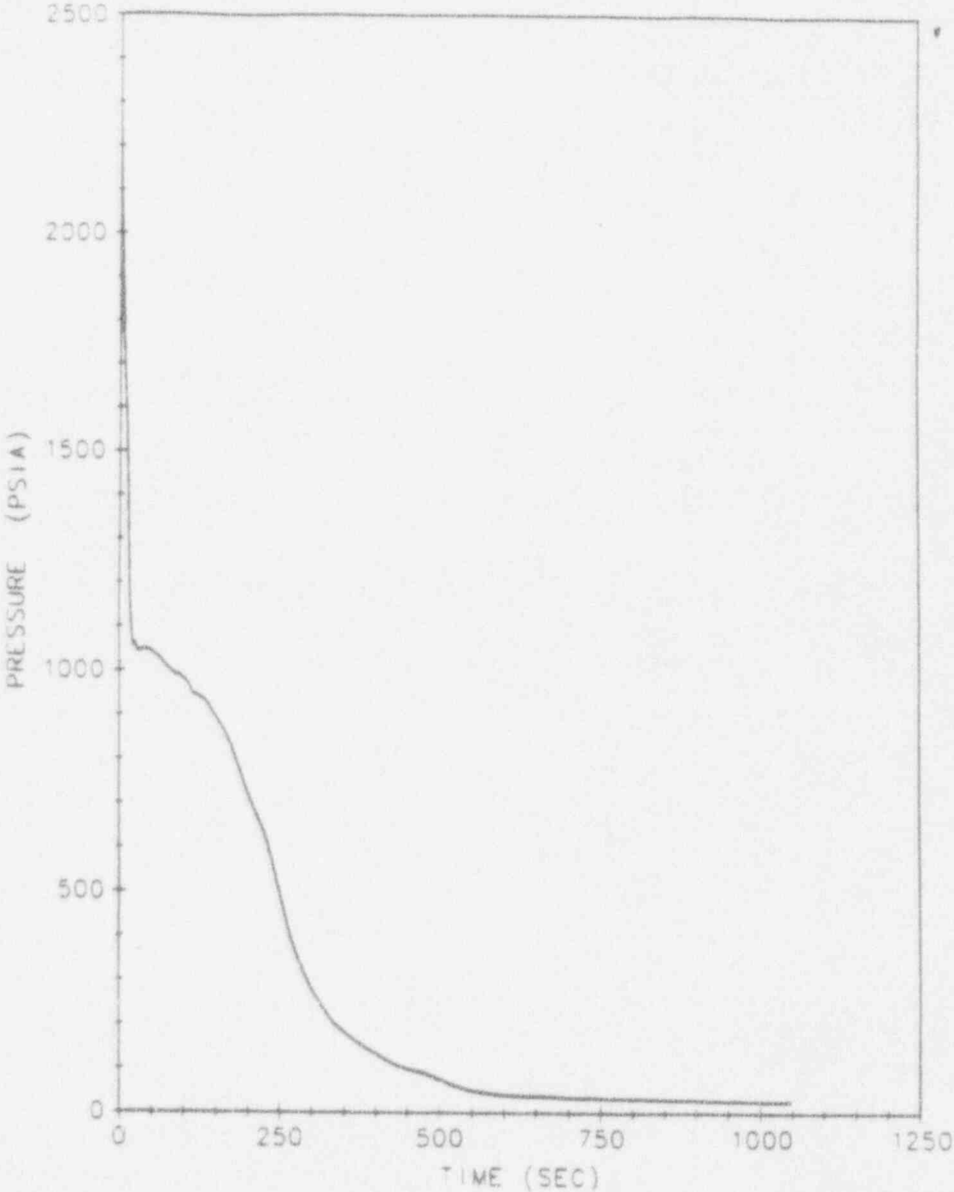


FIGURE 4.3-3
LOOP 1, CMT TO DVI FLOW, DEDVI BREAK

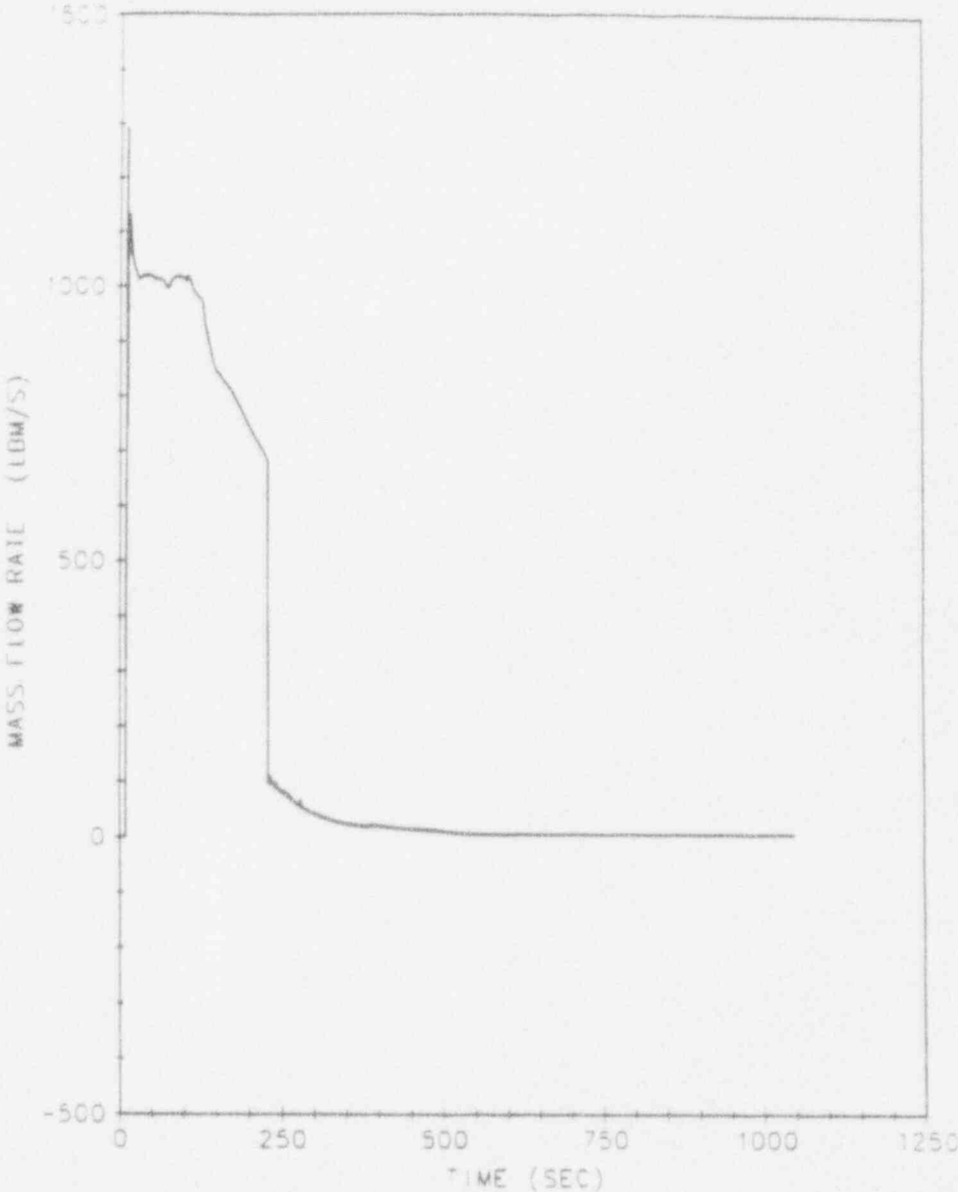


FIGURE 4.3-4
LOOP 2, CMT TO DVI FLOW, DEDVI BREAK

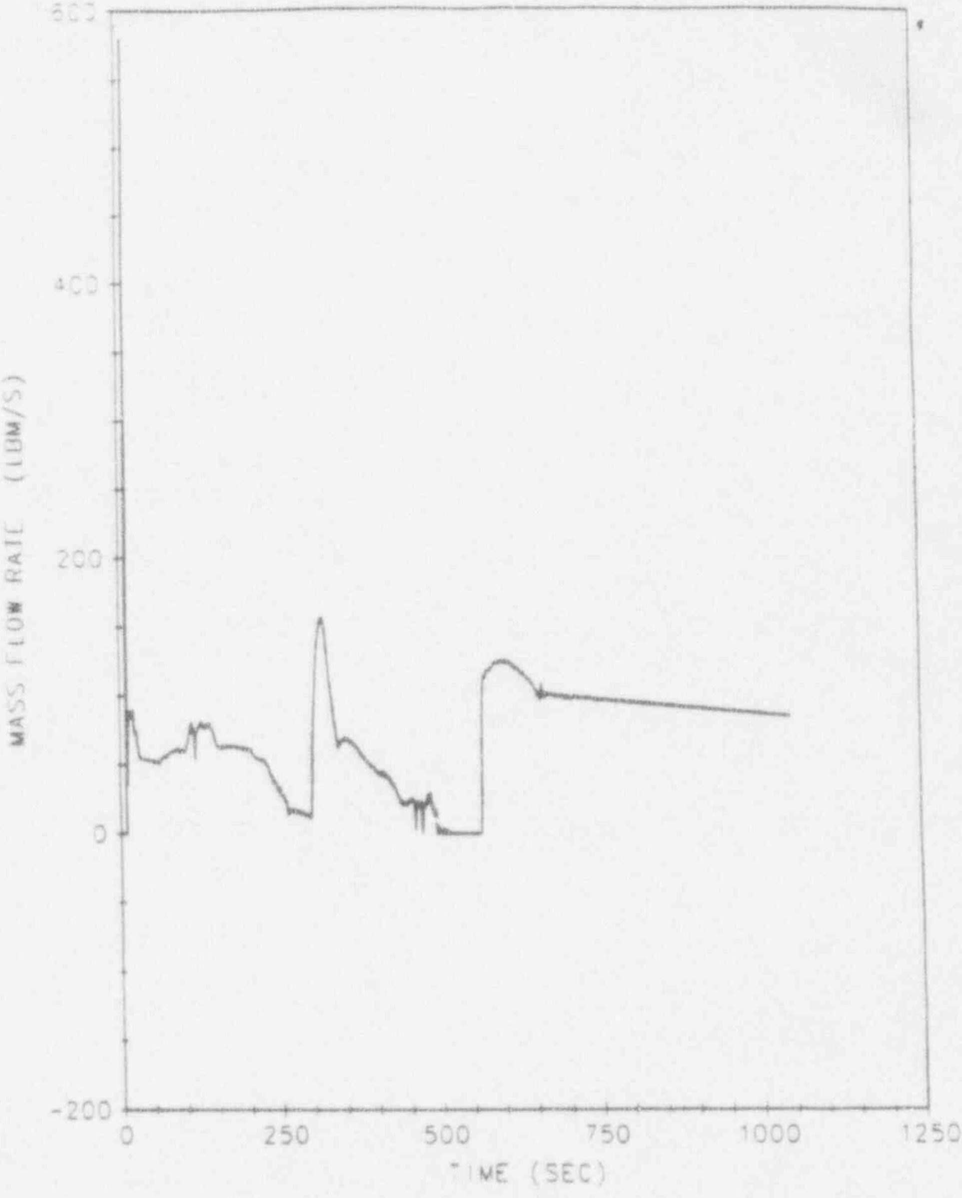


FIGURE 4.3-5
CORE STACK MIXTURE LEVEL, DEDVI BREAK

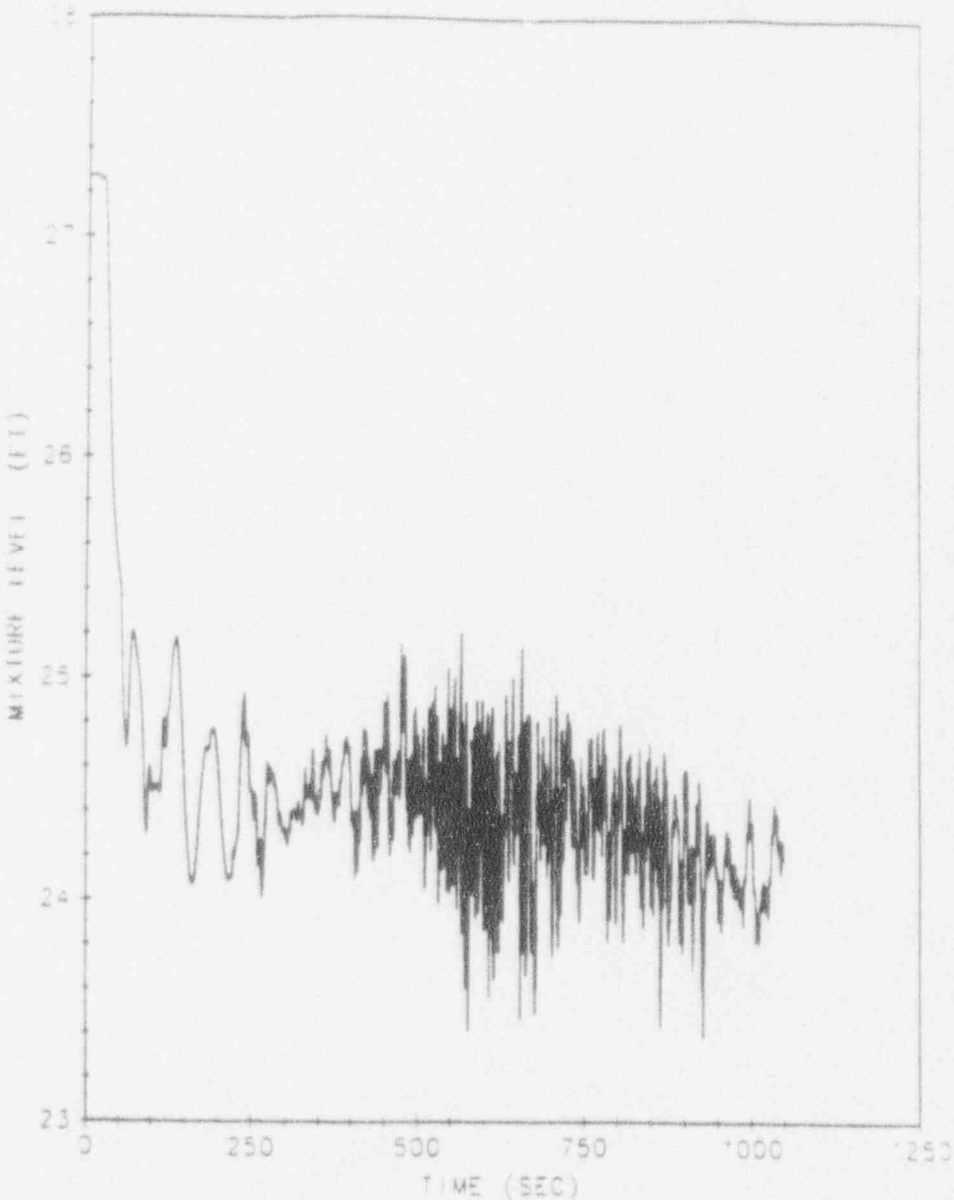


FIGURE 4.3-6
UPPER HEAD MIXTURE LEVEL, DEDVI BREAK

FIGURE 4.3-7
LOOP 1 SG DOWNFLOW SIDE, DEDVI BREAK

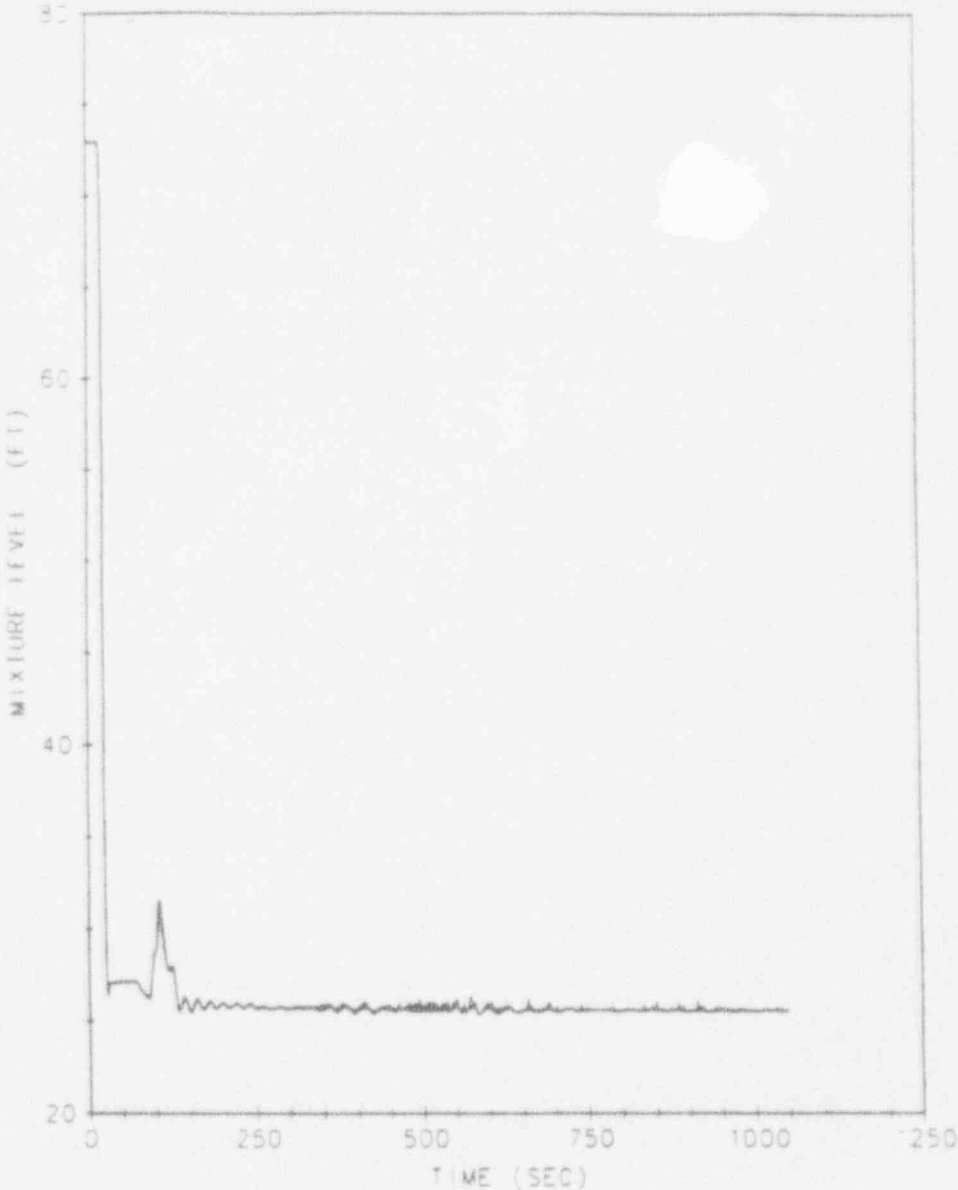


FIGURE 4.3-8
DOWNCOMER MIXTURE LEVEL, DEDVI BREAK

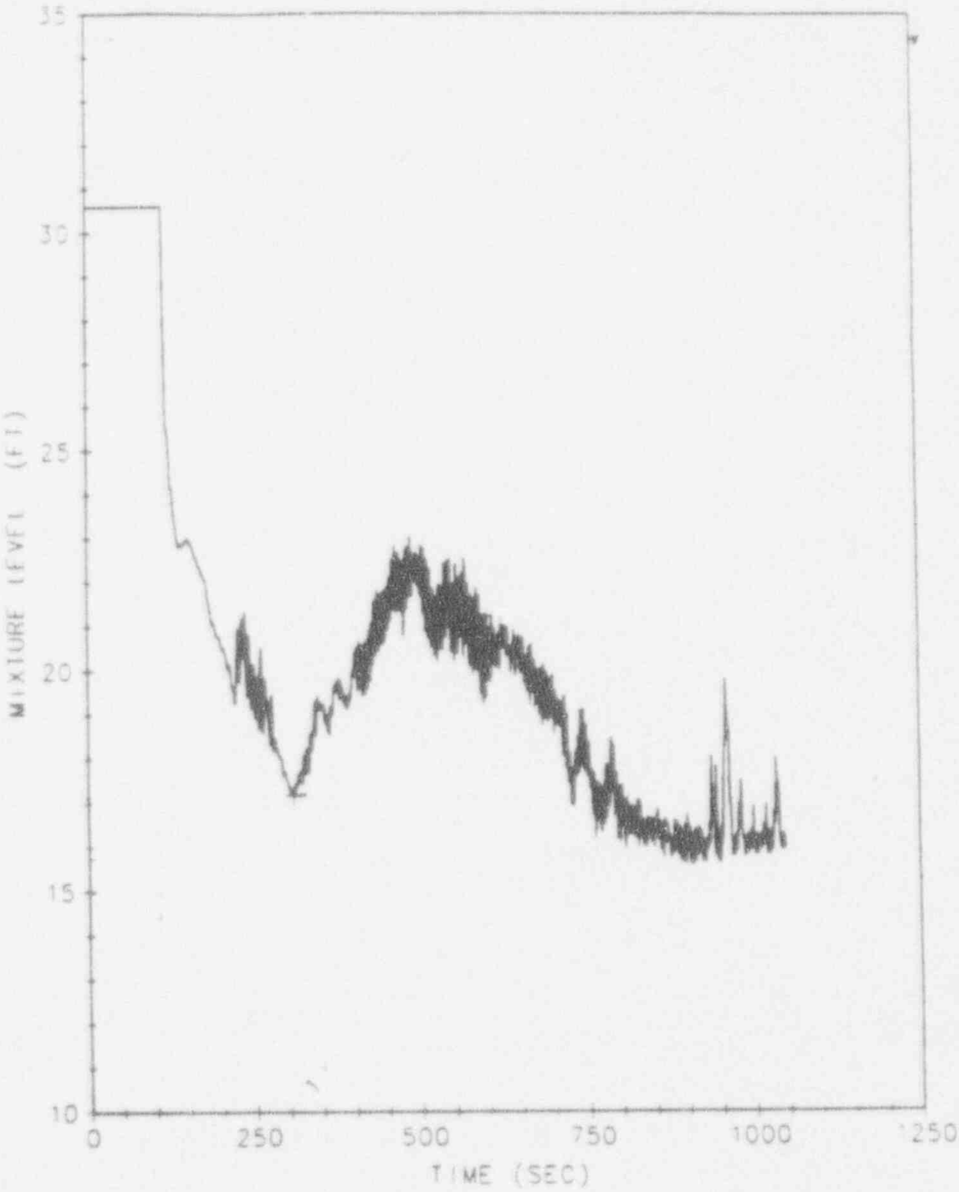


FIGURE 4.3-9
LOOP 1 CMT MIXTURE LEVEL, DEDVI BREAK

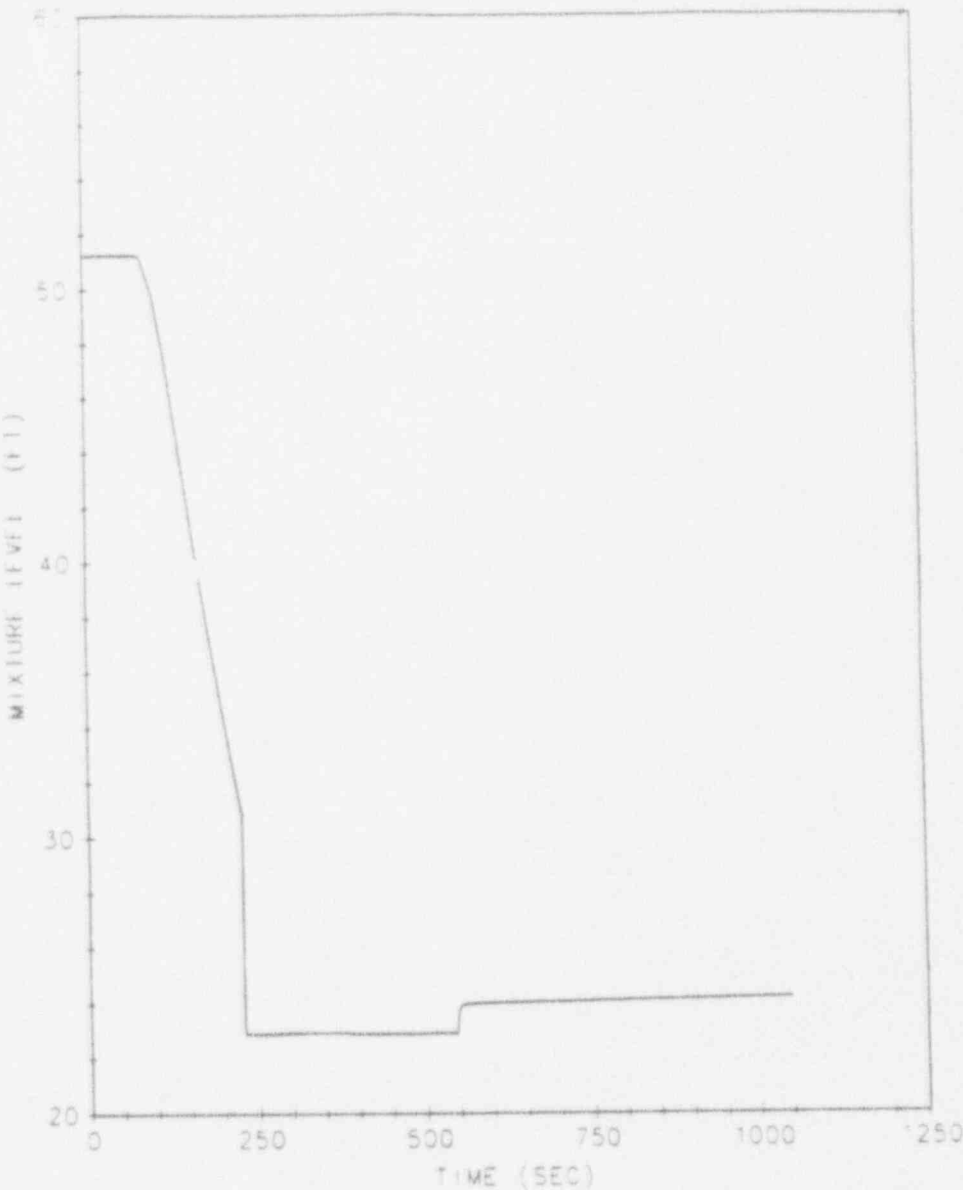


FIGURE 4.3-10
ADS TRAINS 1 TO 3 VAPOR FLOW, DEDVI BREAK

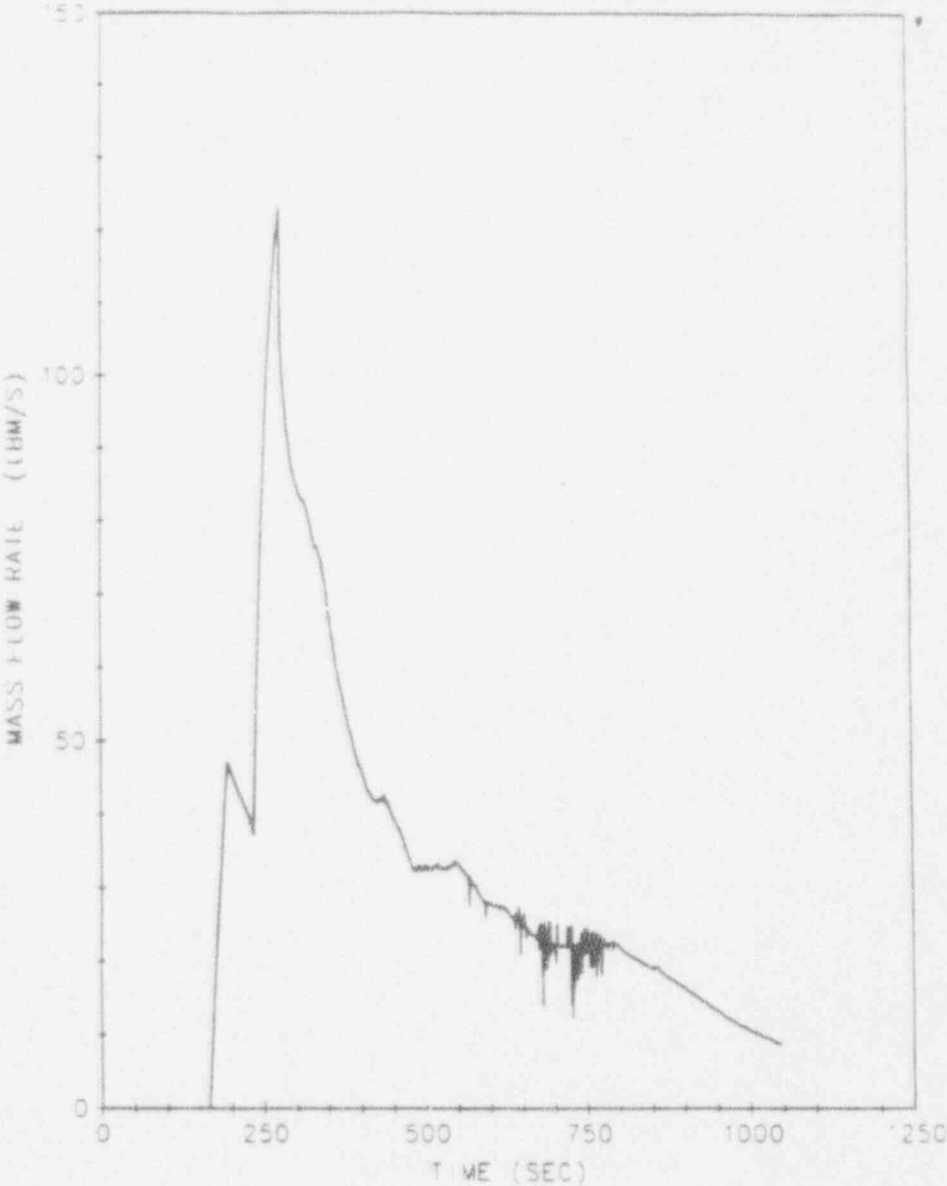


FIGURE 4.3-11
LOOP 2 ACCUMULATOR MASS FLOW RATE, DEDVI BREAK

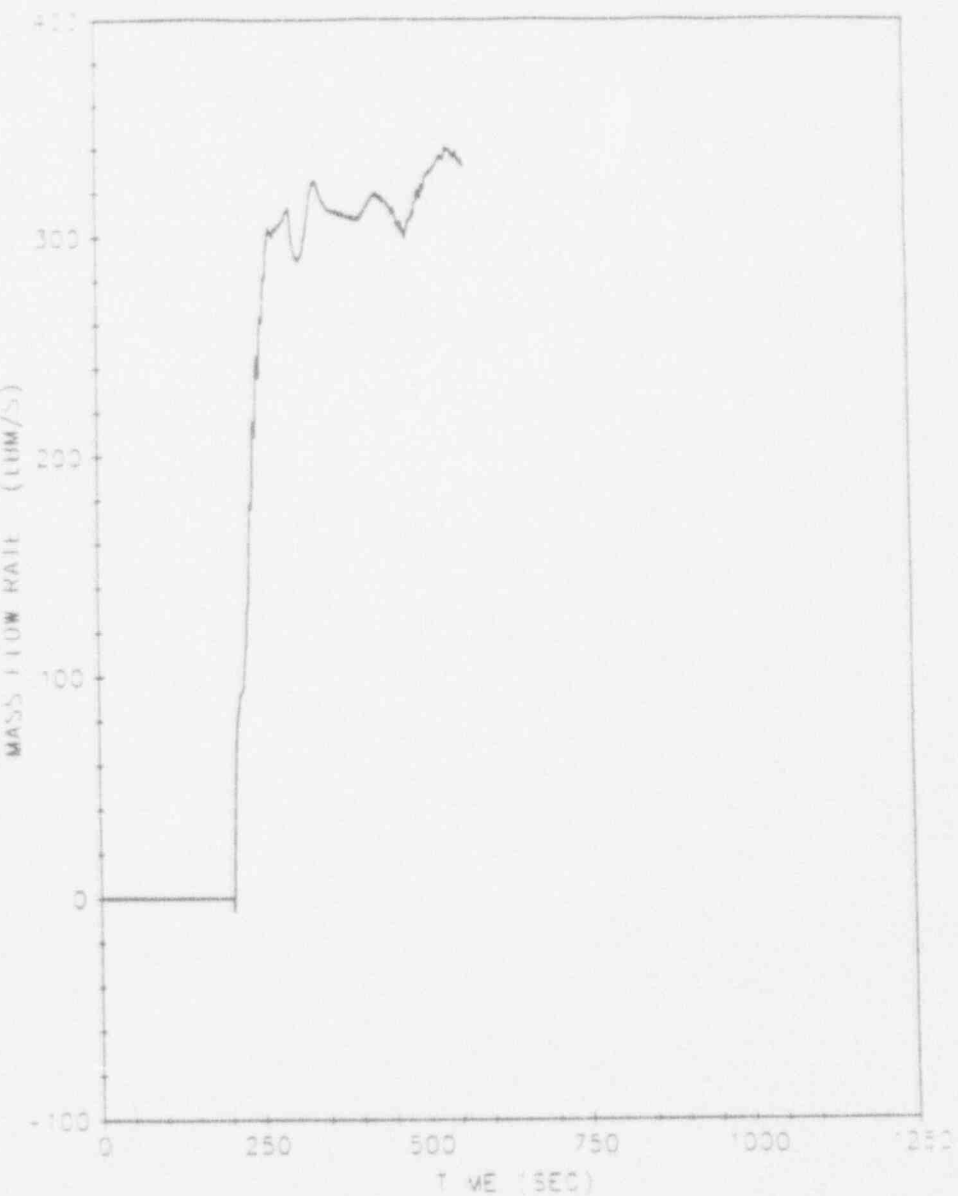


FIGURE 4.3-12
ADS 4 LIQUID FLOW RATE, DEDVI BREAK

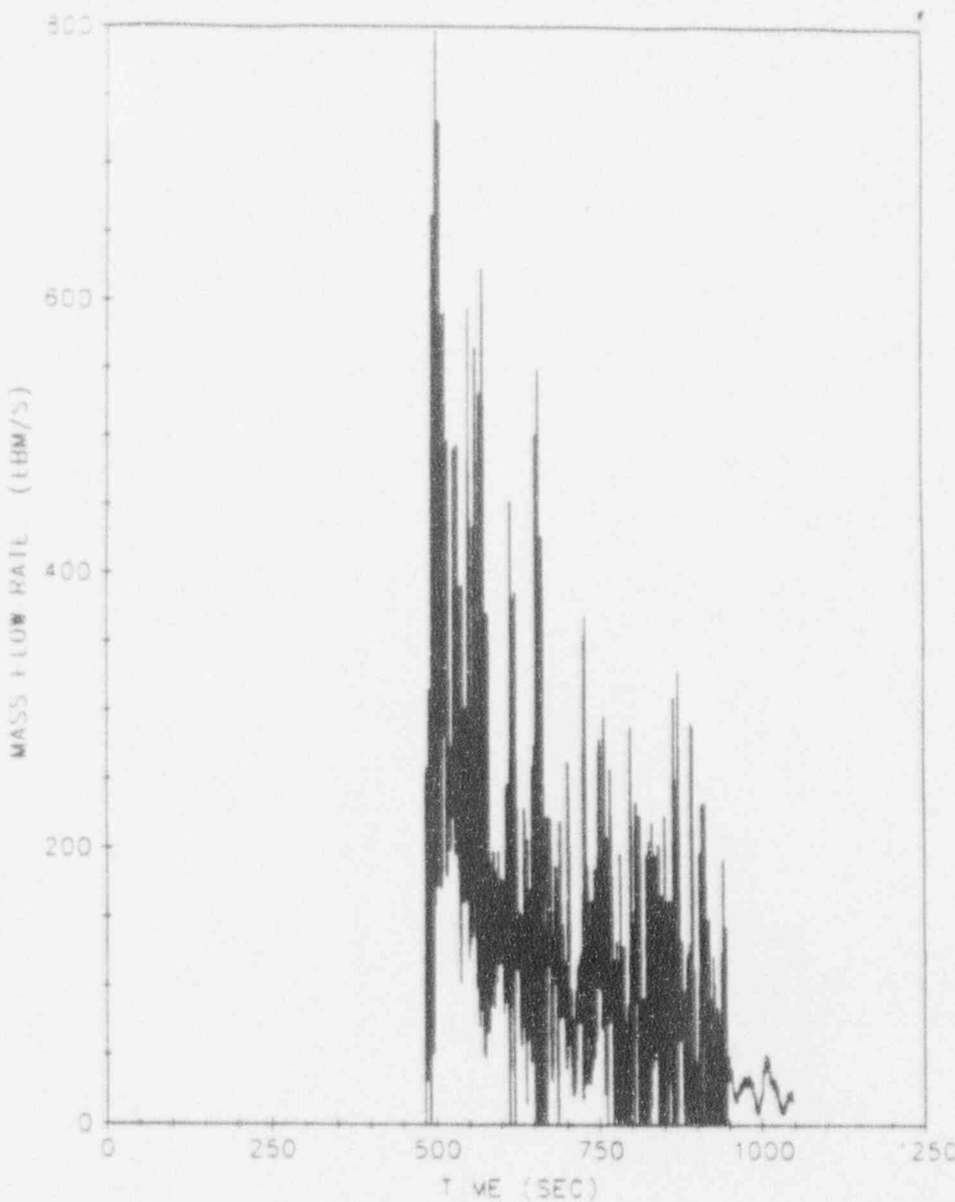


FIGURE 4.3-13
ADS 4 VAPOR FLOW RATE, DEDVI BREAK

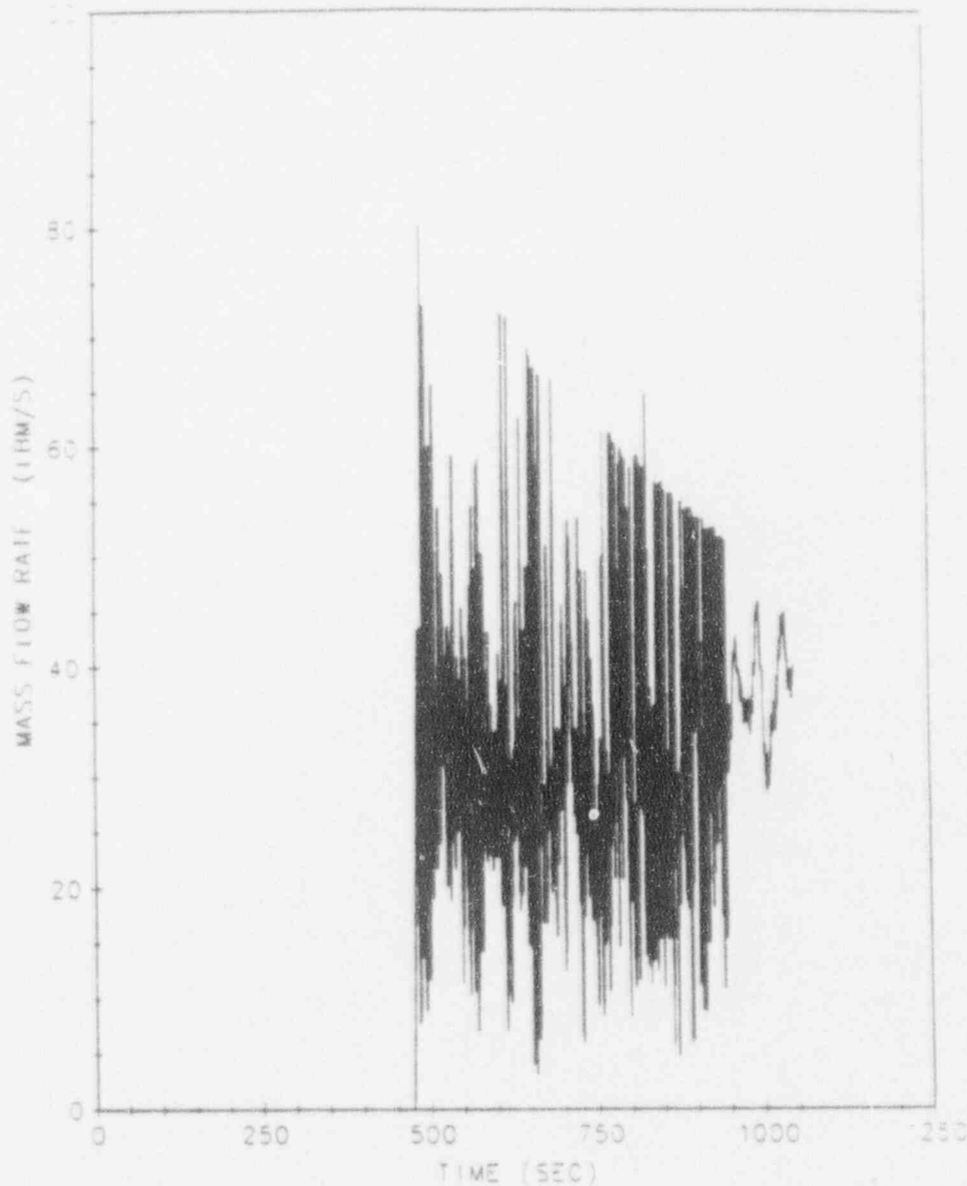


FIGURE 4.3-14
PRIMARY MASS INVENTORY, DEDVI BREAK

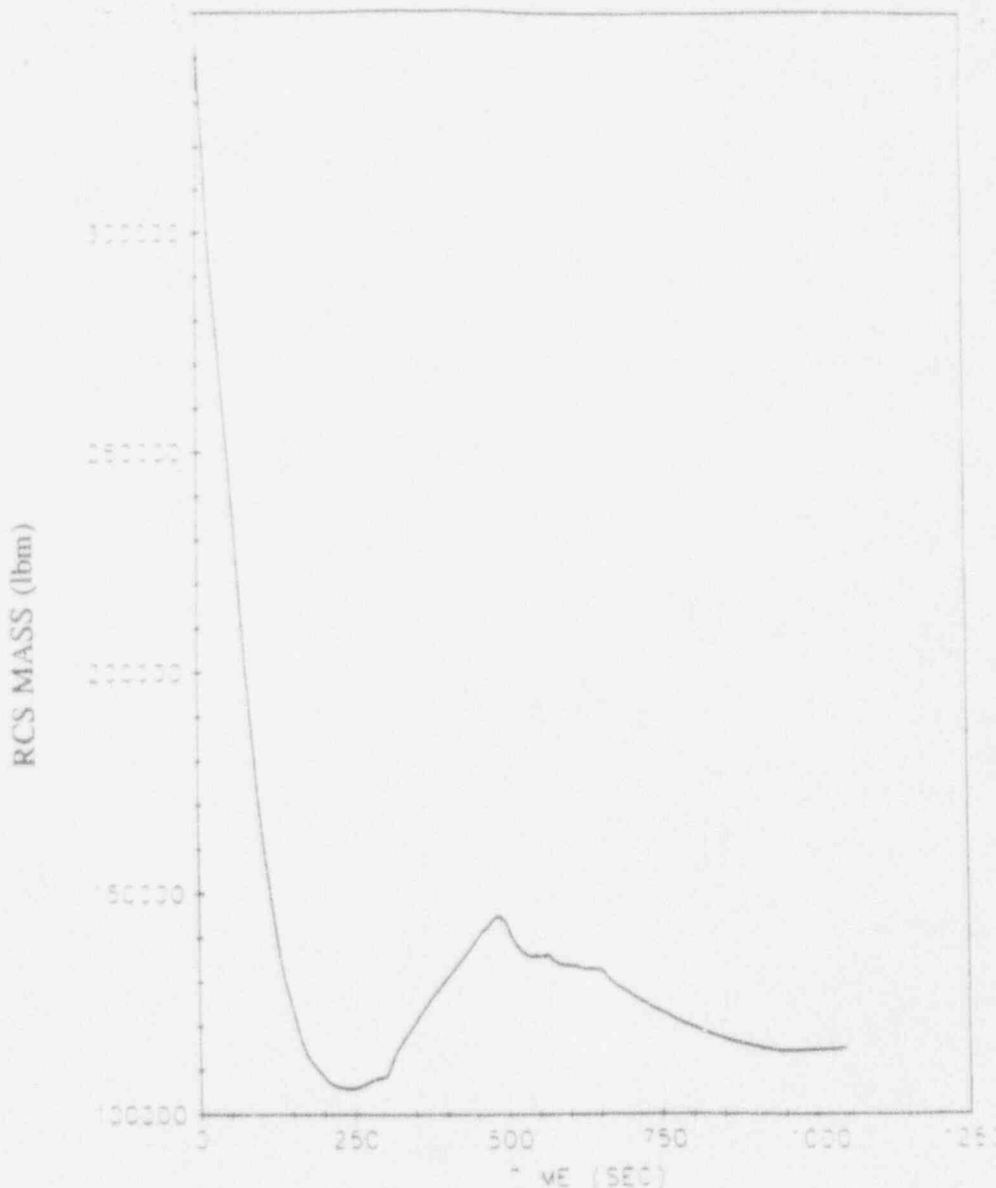


FIGURE 4.3-15
DOWNCOMER PRESSURE, INADVERTENT ADS ACTUATION

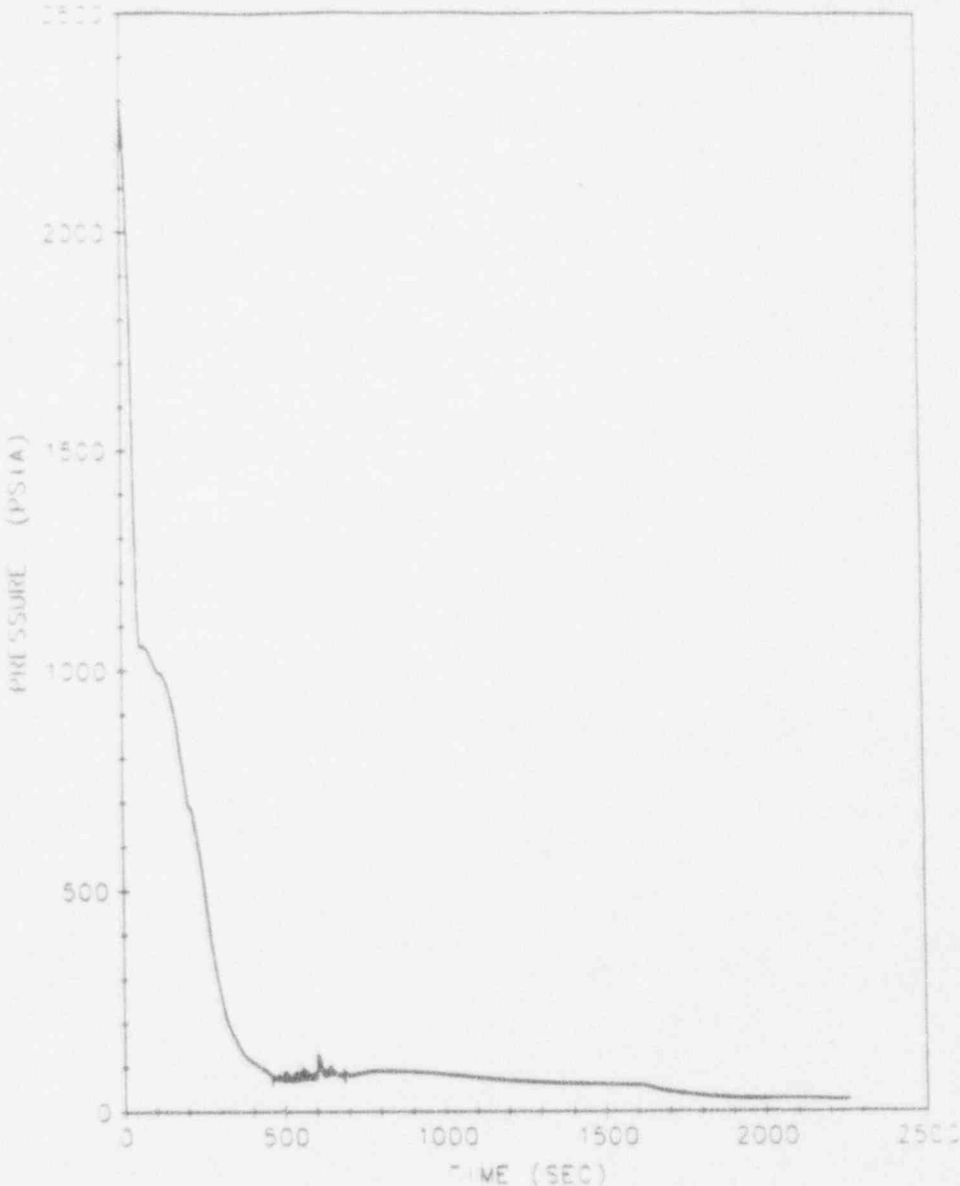


FIGURE 4.3-16
PRESSURIZER MIXTURE LEVEL, INADVERTENT ADS ACTUATION

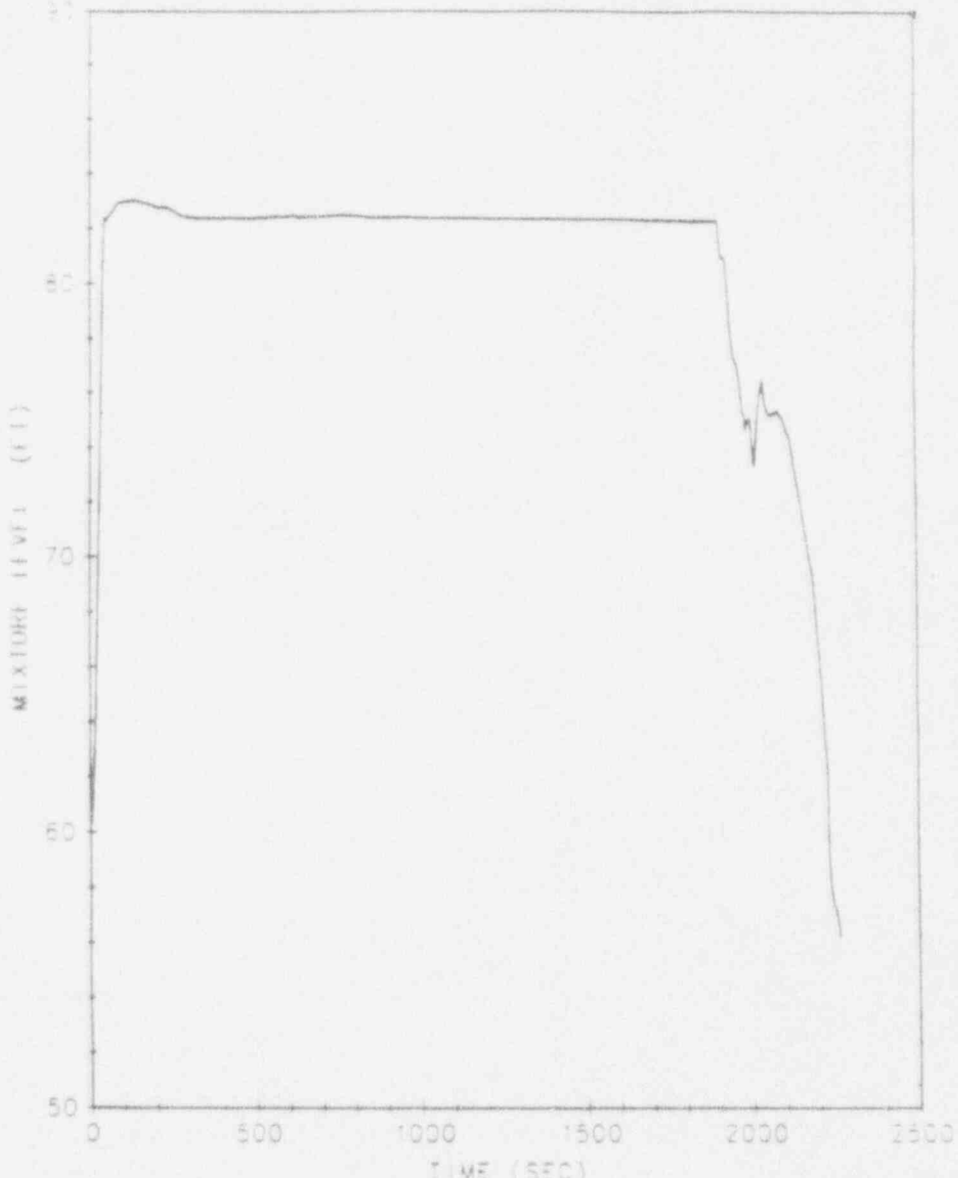


FIGURE 4.3-17
CORE STACK MIXTURE LEVEL, INADVERTENT ADS ACTUATION

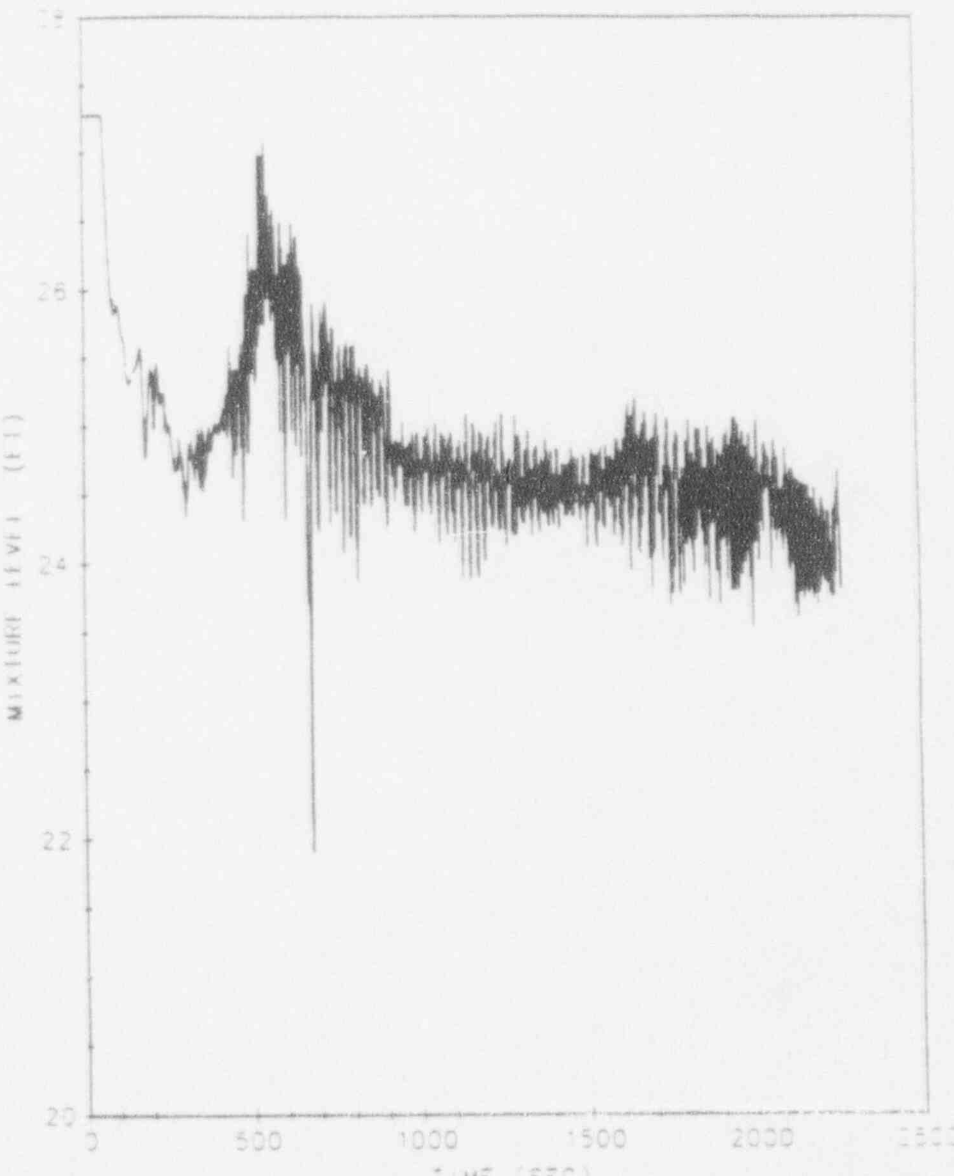


FIGURE 4.3-18
LOOP 1 CMT TO DVI FLOW, INADVERTENT ADS ACTUATION

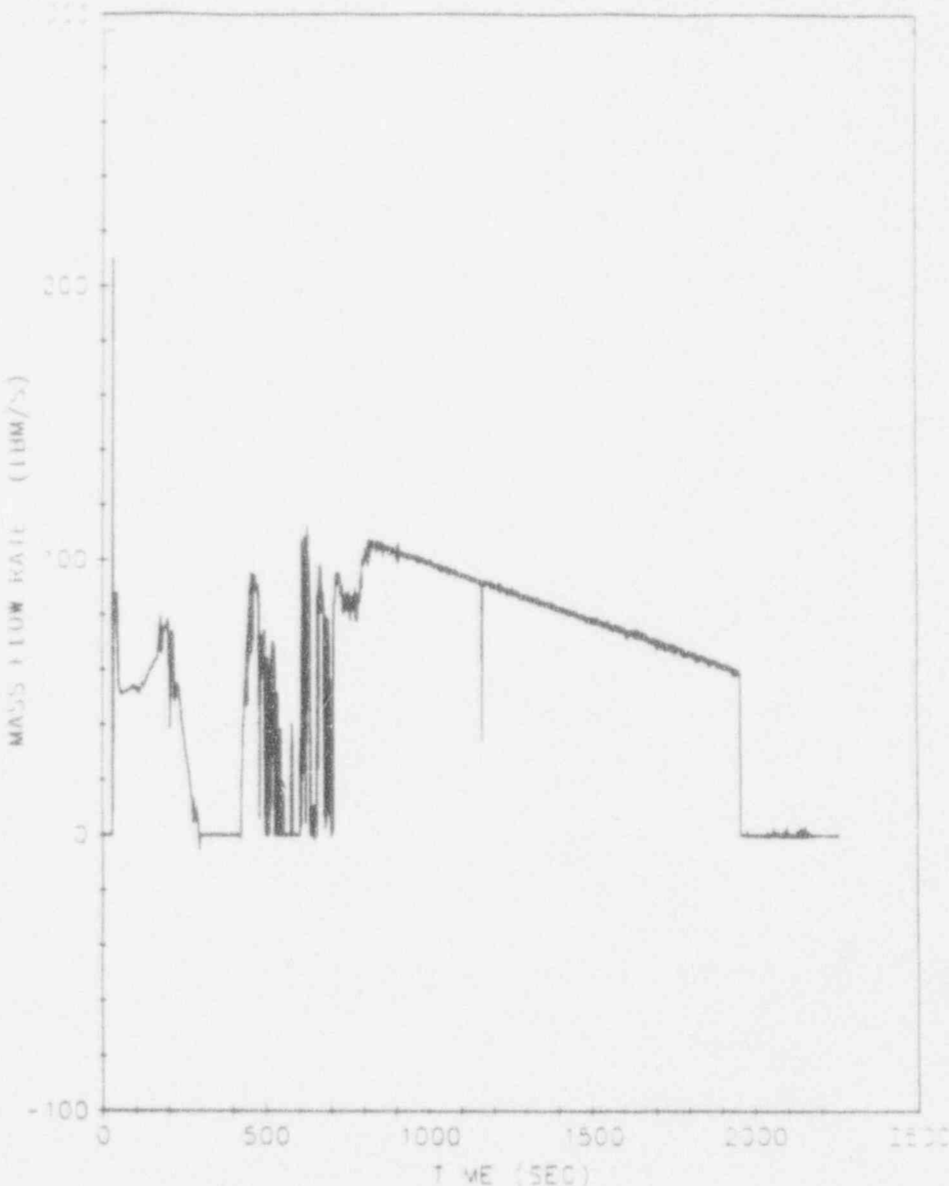


FIGURE 4.3-19
LOOP 1 CMT MIXTURE LEVEL, INADVERTENT ADS ACTUATION

FIGURE 4.3-20
DOWNCOMER MIXTURE LEVEL, INADVERTENT ADS ACTUATION

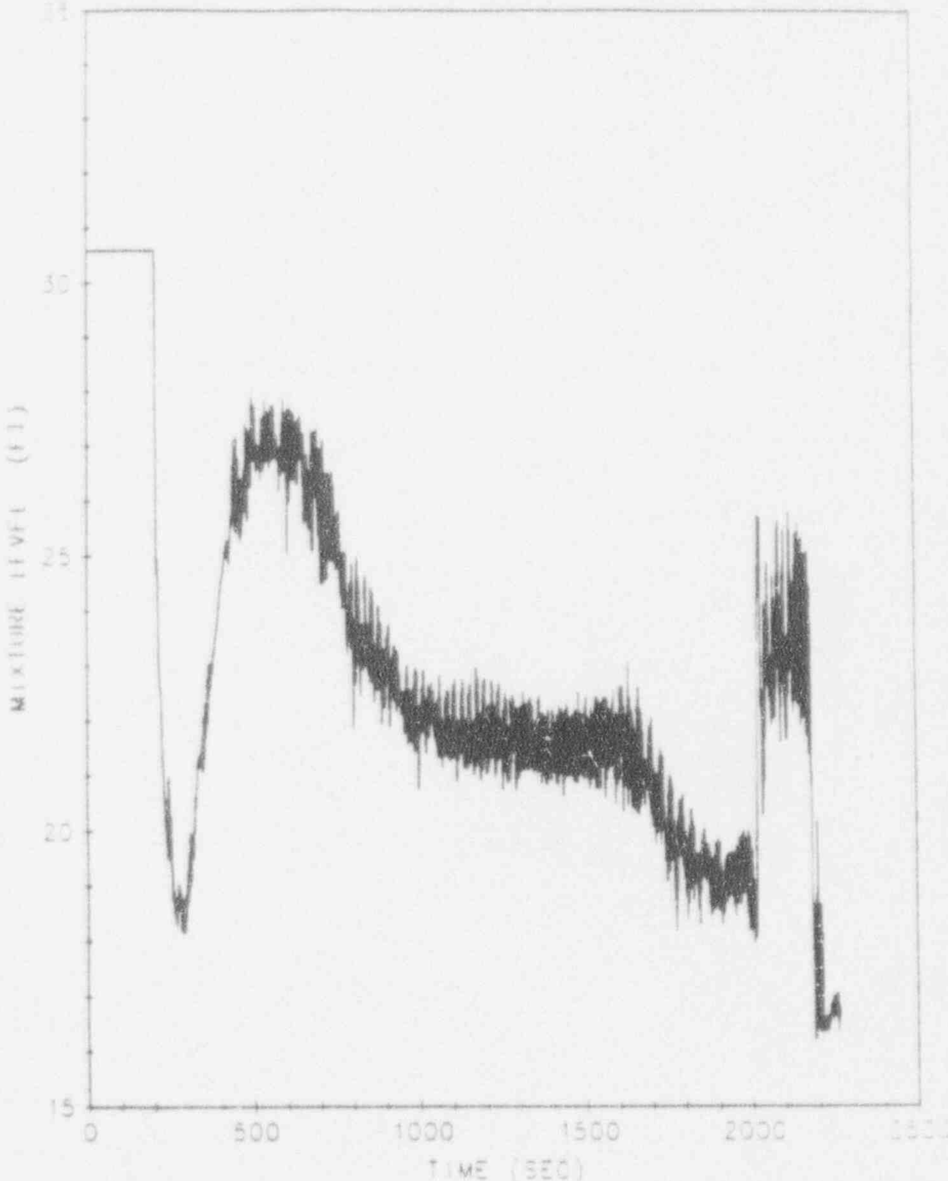


FIGURE 4.3-21
ADS TRAINS 1 TO 3 LIQUID FLOW RATE, INADVERTENT ADS ACTUATION

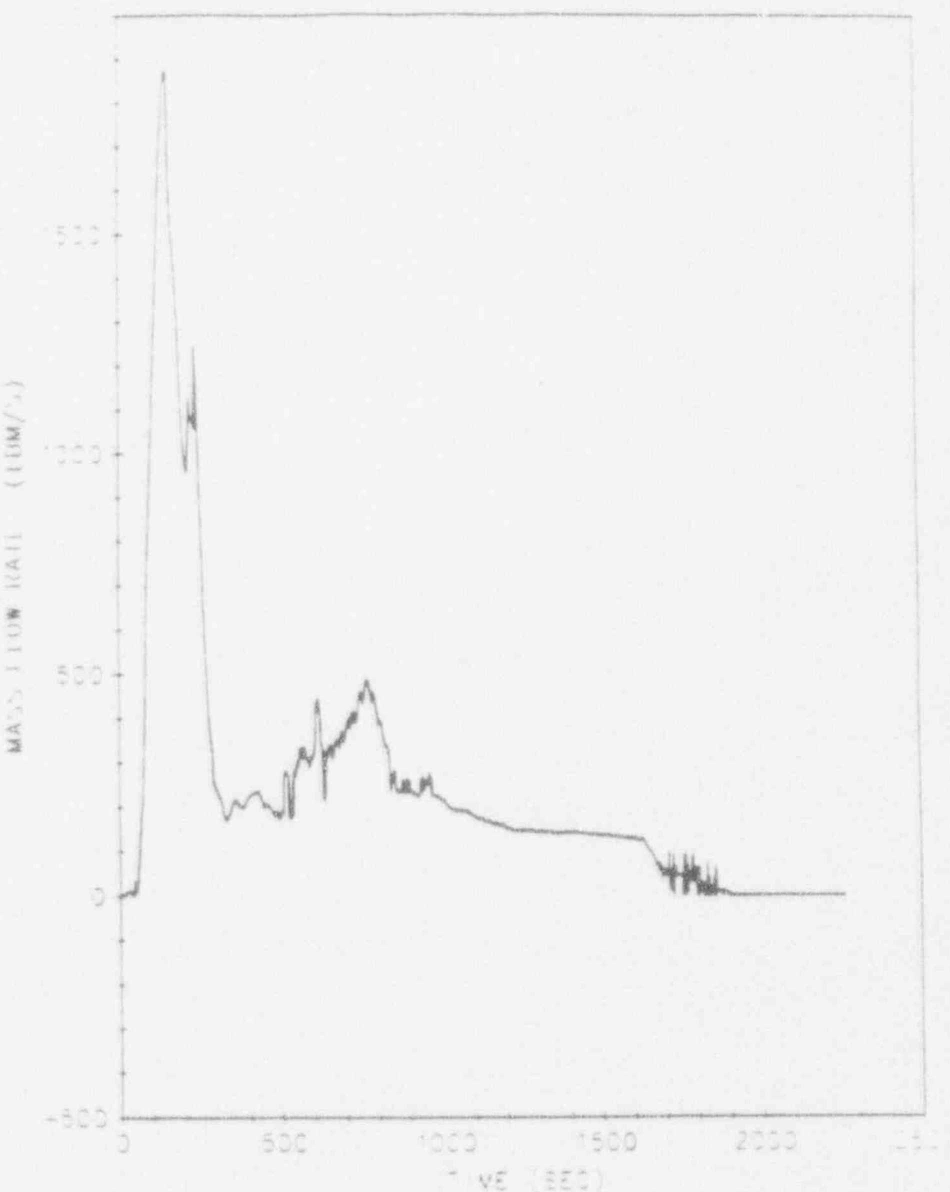


FIGURE 4.3-22
ADS TRAINS 1 TO 3 VAPOR FLOW RATE, INADVERTENT ADS ACTUATION

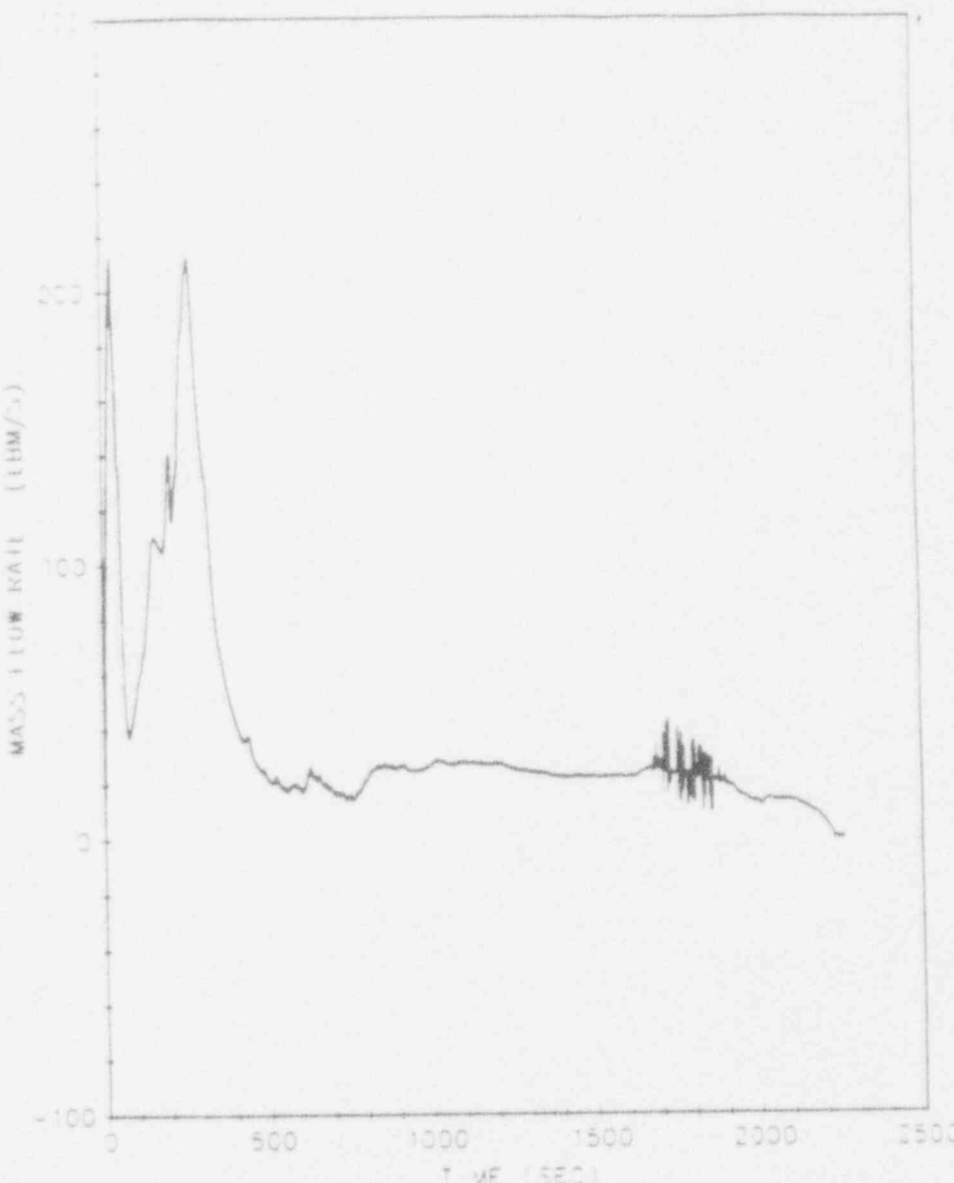


FIGURE 4.3-23
LOOP 1 IRWST INJECTION FLOW, INADVERTENT ADS ACTUATION

FIGURE 4.3-24
ADS 4 LIQUID FLOW RATE, INADVERTENT ADS ACTUATION

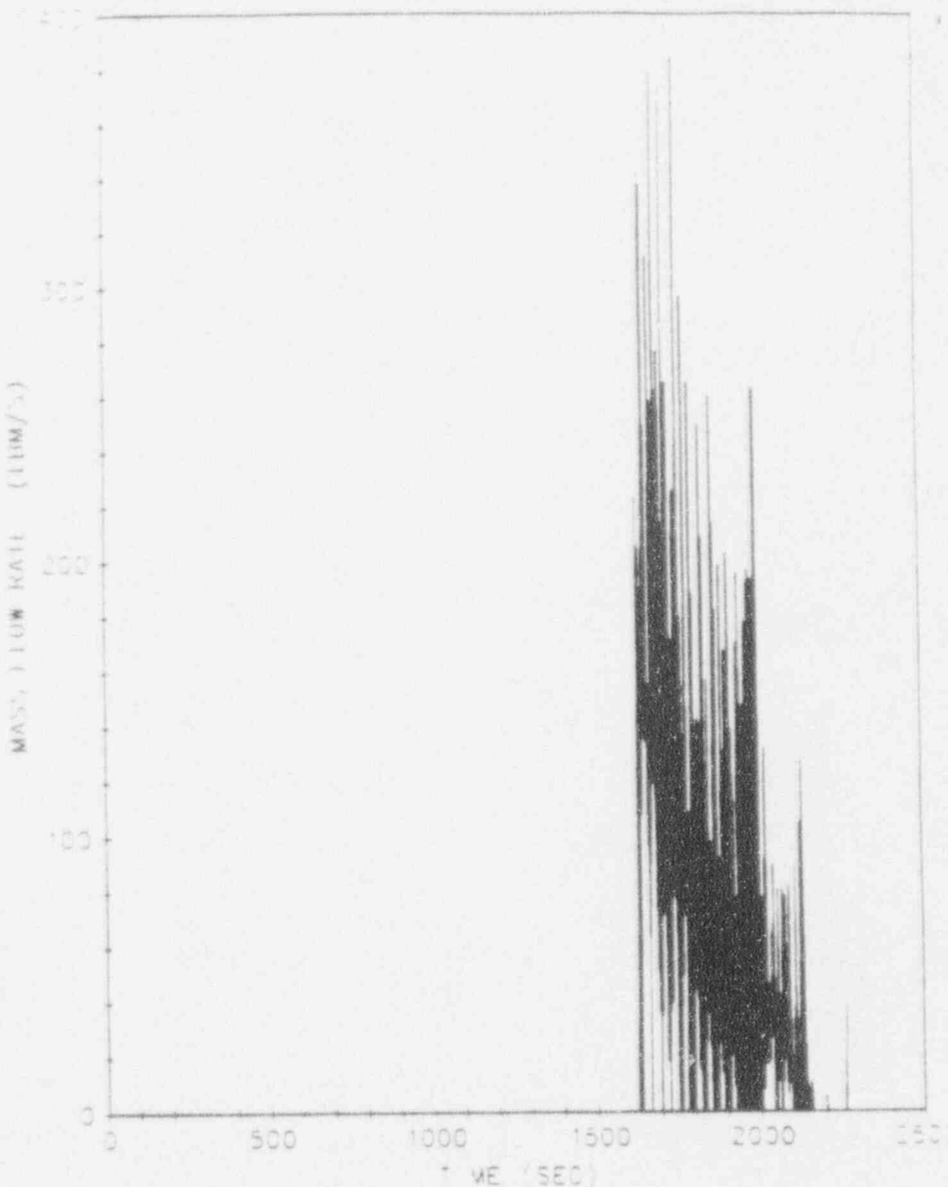


FIGURE 4.3-25
ADS 4 VAPOR FLOW RATE, INADVERTENT ADS ACTUATION

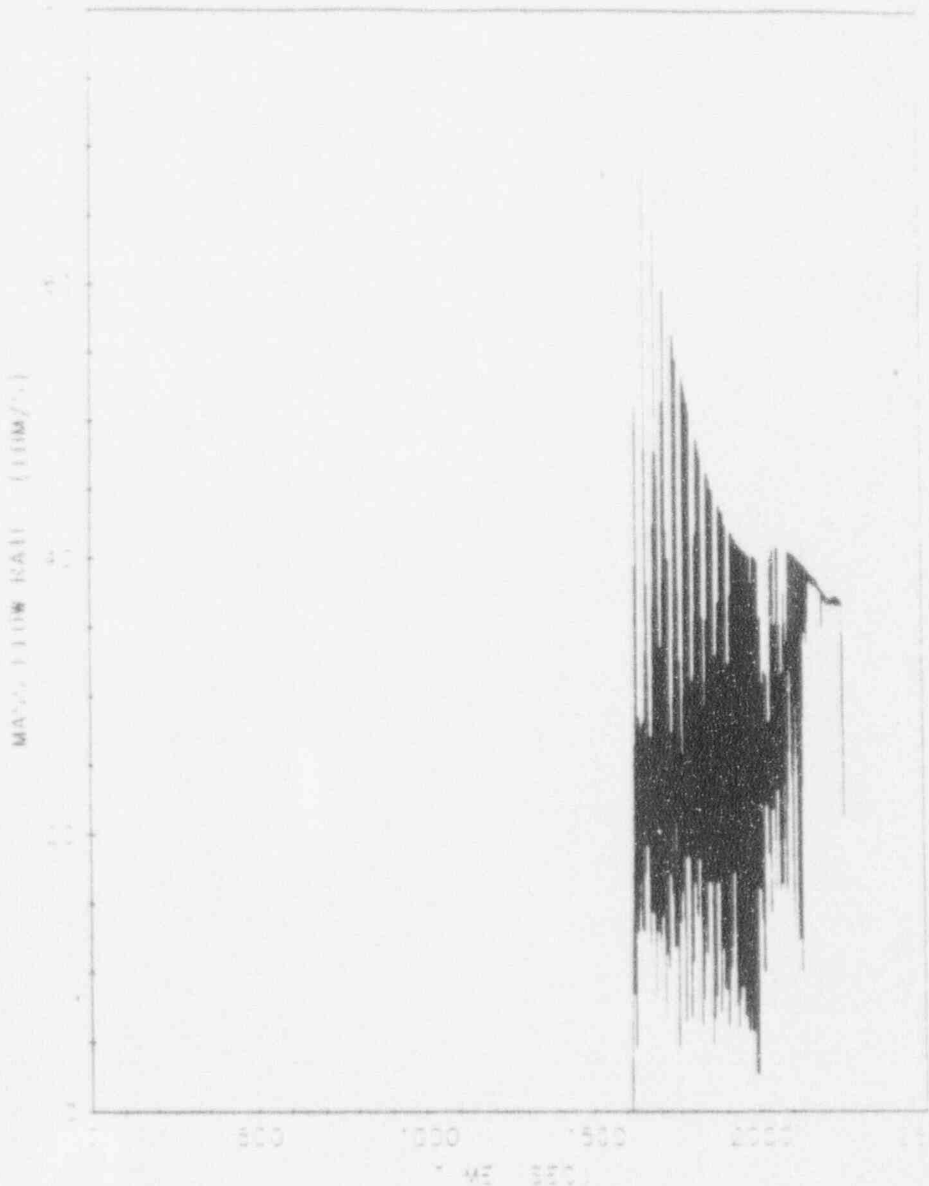
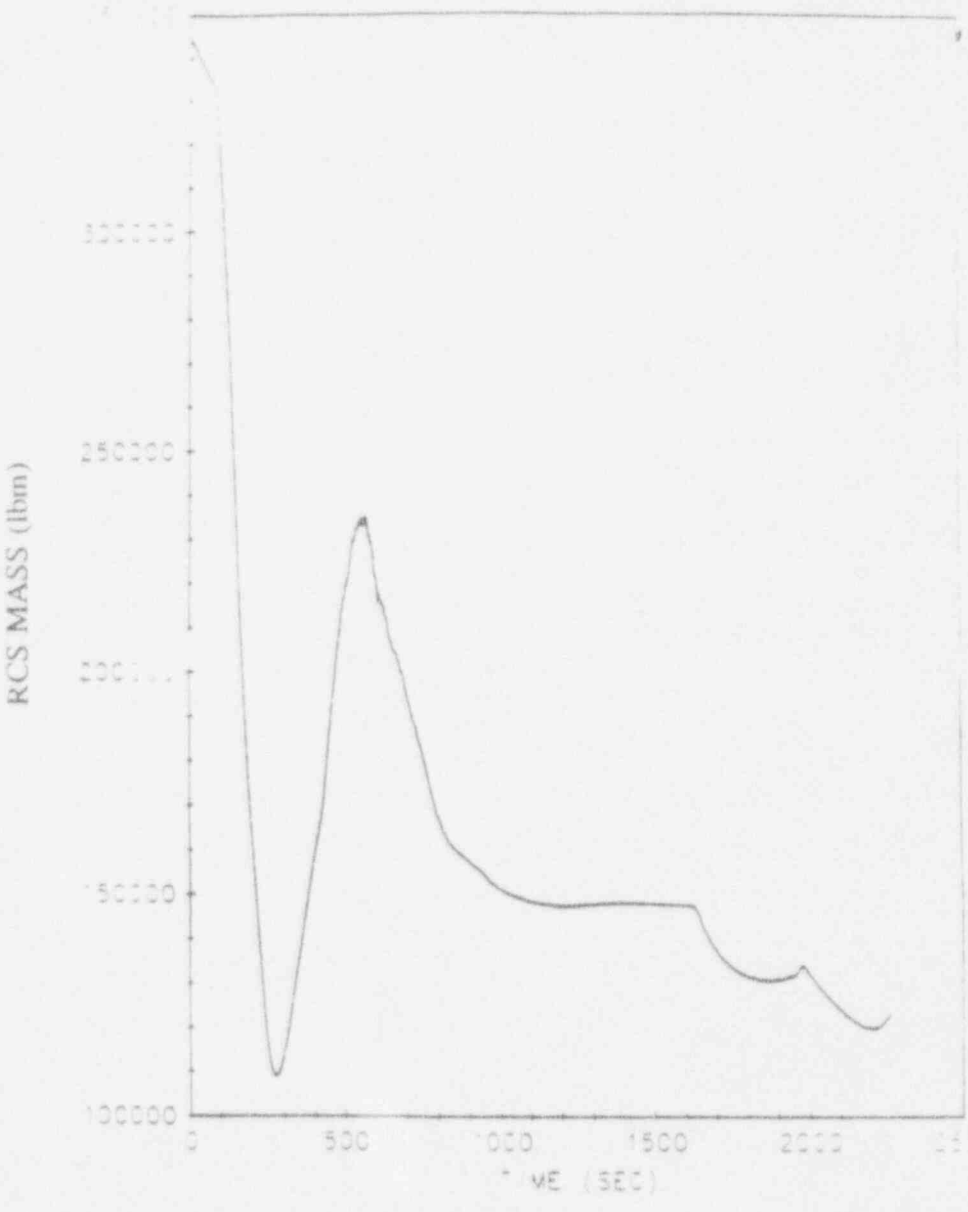


FIGURE 4.3-26
PRIMARY MASS INVENTORY, INADVERTENT ADS ACTUATION



5.0 Test Plans for Automatic Depressurization System (ADS Phase B)

This section discusses test plans for system tests to be performed for the AP600 Automatic Depressurization System (ADS) at the VAPORE facility in Cassacia, Italy. These tests, designated as ADS Phase B, have been restructured to accommodate the ADS design changes discussed in Section 2.9.

5.1 Test Objectives

The overall objective of the ADS Phase B tests is to collect thermal hydraulic performance data on the ADS that will be used to verify the analytical models used in design basis safety analyses. Specifically, the tests will be used to obtain fluid temperature, pressure and pressure drop data over a range of mass flow rates through a full scale simulation of the AP600 ADS. The tests will also be used to obtain additional sparger and quench tank loading data over a range of flow conditions.

5.2 Test Loop Configuration

Figure 5-1 shows the test loop configuration for the Phase B tests. The test facility provides a full scale simulation of the ADS System, including:

- ADS valve piping package,
- downstream piping, and
- sparger.

The ADS valve piping package for the tests is shown in Figure 5-2. As shown in this figure and in Table 5-1, a flanged spool piece will be installed at one of the two valve locations in each ADS stage to allow different valve types and sizes to be simulated.

5.3 ADS Phase B Tests

Tests will be performed to represent high flow resistance (minimum venting flow) and low flow resistance (maximum loads). These conditions will be obtained by varying the system configuration and by varying the system conditions at ADS valve piping package manifold. The system configuration will be varied by changing the flow path (i.e., stage 1 only, stage 1 and 2, etc.) and by changing the flow area of each flow path. The system conditions at the ADS valve piping package manifold (e.g., pressure and fluid quality) will be controlled via initial conditions of the supply tank and via the system control valve.

Three types of tests will be performed as part of the ADS Phase B tests: 1) saturated steam blowdowns, 2) steam and saturated water blowdowns for minimum ADS venting, and 3) Steam and Saturated Water Blowdowns for Maximum Loads.

Saturated Steam Blowdowns

These tests will use steam from the top of the supply tank and will cover a range of pressures for each ADS flow path. Only maximum flow resistance will be simulated for these tests since minimum flow resistance information was previously obtained from the Phase A portion of the ADS tests.

Steam and Saturated Water Blowdowns for Minimum ADS Venting

These tests will use saturated water from the bottom of the supply tank and will cover a range of pressure and two-phase conditions for each ADS flow path. Maximum flow resistance will be simulated. The facility supply valve (12 inch gate) will be positioned to obtain a range of qualities entering the valve piping package.

Steam and Saturated Water Blowdowns for Maximum Loads

These tests will use saturated water from the bottom of the supply tank and will cover a range of two phase flow and pressure conditions for each ADS flow path. Minimum flow resistance will be simulated to provide sparger loading data. The facility supply valve (12 inch gate) will be fully opened to obtain maximum mass flow entering the valve piping package. Tests will be performed at different quench tank temperatures to obtain loads over a range of conditions.

Table 5-2 shows the ADS Phase B test matrix.

TABLE 5-1 VALVE REPRESENTATION IN THE ADS VALVE PACKAGE (ADS PHASE B TESTS)

a,c

TABLE 5-2 ADS PHASE B TEST MATRIX (preliminary)

Blowdown Fluid	ADS Simulation	AP600 Pressure Simulated	Comments
Saturated Steam from top of supply tank	· Stage 1 open	~ 2250 to 400 psig	Maximum flow resistance simulated. Obtain 1ϕ T/H data
	· Stages 1 and 2 open	~ 800 to 100 psig	
	· Stages 1 and 3 open	~ 500 to 50 psig	
	· Stages 1,2, and 3 open	~ 500 to 50 psig	
Saturated water from bottom of supply tank	· Stage 1 open	~ 2250 to 400 psig	Maximum flow resistance simulated. 12 inch gate valve positioned to obtain a range of 2ϕ T/H data
	· Stages 1 and 2 open	~ 1200 to 100 psig	
	· Stages 1 and 3 open	~ 500 to 50 psig	
	· Stages 1,2 and 3 open	~ 500 to 50 psig	
	· Stage 2 open (inadvertent opening at full power)	~ 2235 psig	
Saturated water from bottom of supply tank	· Stages 1,2 and 3 open (1)	~ 500 to 50 psig	Minimum flow resistance simulated. Maximum flow/minimum quality for max loads on sparger and quench tank
	· Stages 1,2 and 3 open	~ 500 to 50 psig	
	· Stages 1 and 2 open (1)	~1200 to 100 psig	
	· Stage 2 open (1)	~ 2235 psig	

(1) - Quench tank water temperature initially at 212°F

FIGURE 5-1 ADS PHASE B TEST FACILITY SCHEMATIC

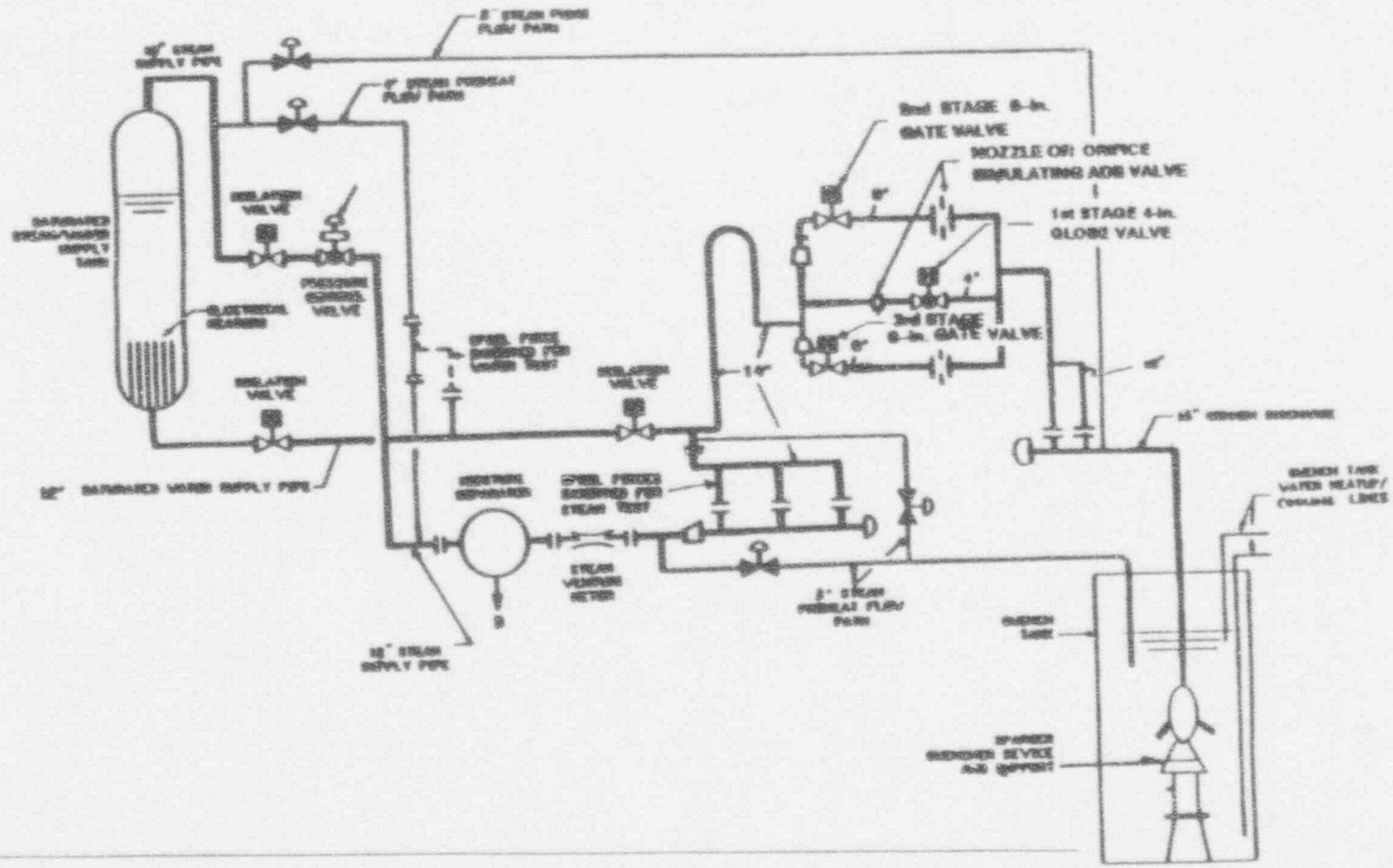


FIGURE 5-2 ADS PHASE B TEST PIPING/VALVE PACKAGE CONFIGURATION

a, c

6.0 ADS Valve Tests

There will be a number of tests performed outside of the Design Certification process during the development, manufacture, installation and operation of the ADS valves. These tests include:

- Testing to support ADS valve type selection and to support valve qualification testing
- Valve qualification testing
- Production testing
- Pre-operational testing
- In-service testing

Each of these tests is further discussed in the following sections.

6.1 ADS Valve Type Selection Testing

For this testing, full sized prototypic Stage 1/2/3 ADS valves will be installed in a ADS valve package piping simulation. Different valve designs will be tested for the ADS Stage 2 and 3 function, to provide a basis for performance comparisons. Testing will be performed over a range of flow conditions that bounds the actual ADS operation.

The overall objective of this testing is to characterize valve performance in this application, as an aid to the final selection of a valve type. This information is not required for design certification. Data obtained on valve performance will include the measurement of valve operator mechanical and electrical performance; required thrust to unseat, open, close, and seat the valves. The valve seats and disks will be visually examined after test runs to observe and document valve wear/damage.

These tests will provide input to the valve specification including fluid conditions, flow, temperature/pressure, dp conditions and IST conditions. They will also provide input to the EQ testing to determine the limiting test conditions (flow, fluid, temperature/pressure, dP).

6.2 Valve Qualification Testing

Valve qualification will occur after vendor selection and is a three step process of analytical qualification, functional testing of the valve/operator assembly, and IEEE qualification of the operator.

6.2.1 Analytical Qualification

Prior to qualification testing, vendors will produce calculations to confirm that their designs are acceptable for the ADS application. These calculations include:

- An ASME Class 1 design report to verify valve integrity under the design conditions in the equipment specification, including specified nozzle loads and design transients

- A seismic analysis to verify operability for the specified seismic accelerations at the maximum operating load

- A weak link analysis to provide the maximum loads the valve components can withstand for both test and operating conditions

- Operator sizing calculations using both the manufacturer's and EPRI's methodology (when applicable)

6.2.2 Functional Testing

Based on acceptable analytical qualification, valves will be manufactured for testing. The valve/actuator assembly to be tested will be identical to that used in the plant with respect to configuration, materials and dimensions. Prior to assembly, each test valve will be dimensionally inspected in accordance with EPRI guidelines. Critical dimensions will be recorded, along with any special features that are critical to operation. Testing is performed with instrumentation to measure stem thrust, torque, switch actuation, travel, motor speed, motor temperature, accelerations and fluid temperature, pressure and differential pressure. The test conditions will be determined with input from the type selection testing (Section 6.1).

6.2.3 Operator Qualification

The operator will require a separate qualification based on IEEE-382 which will include cyclic aging, vibration aging, seismic testing and environmental aging including LOCA/HELB.

Supplemental testing may be required to address separate issues related to the electric actuators in relation to effects of operating time on motor temperature and speed, motor temperature effect on motor output torque and motor speed effect on torque capability.

6.2.4 Valve Design Control

An auditable link will be developed between the tested valves and those subsequently shipped to plants. Production valves will be purchased to the same equipment specification requirements as the tested valves. Dimensional inspection of critical parts will be performed and recorded to verify that they are within required tolerances. The critical dimensions will be determined based on a review of the test results and design configuration by the valve manufacturer and Westinghouse.

Any modifications to the original design must be reviewed, evaluated and approved by Westinghouse prior to implementation by the manufacturer.

6.3 Production Testing

Each valve will be subjected to production testing including a hydrostatic shell test of the body/bonnet, hydrostatic disc test, leakage tests on the seat, backseat and packing, and a functional test. The latter is performed at nominal and reduced voltage, without flow but with the valve closed and at the design pressure differential. Finally, a cyclic test is performed for [TBD] cycles with no pressure or flow.

6.4 Pre-Operational Testing

Static baseline testing will be performed prior to startup, to verify that the valve is set up and functioning correctly. Following the static test, the stage 1, 2 and 3 valves will be subjected to individual valve tests with low flow and with design differential pressure (initial). Finally, an integrated blowdown test will be performed on the first AP600 from intermediate RCS pressure and temperature which will actuate ADS Stages 1, 2, 3 and 4.

6.5 In-service Testing

An in-service stroke time test will be performed every [TBD] months at zero differential pressure, ambient temperature. A pre-refueling test will be performed at low flow (a pressure differential between 400 and 1200 psi, and a temperature of 300 degrees). If squib valves are used for the 4th stage they would not be subject to these tests but would have IST requirements, such as periodically actuating their charges outside the ADS valves.

Every 5 years, verification testing will be performed on valve set-up and operator capabilities, in accordance with GL-89-10.

Post-maintenance testing of the operator and valve will be required whenever changes are made that may affect the required operating loads or operator output.

7.0 References

7.1 Document No. GWGL021, "Westinghouse AP600 Standard Safety Analysis Report", Revision 1, 1/13/94.

7.2 Document No. GWGL022, "Westinghouse AP600 Probabilistic Risk Assessment", Revision 0, 6/26/92.

7.3 Meyer, P.E., "NOTRUMP - A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A, (Proprietary) and WCAP-10080-A (Nonproprietary), August 1985.

7.4 Lee, N., Rupprecht, S.D., Schwartz, W.R., and Tauche, W.D., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-A (Nonproprietary), August 1985.

APPENDIX A-1

MARKUP OF AP600 SSAR SECTION 6.3 - PASSIVE CORE COOLING SYSTEMS

6.3 PASSIVE CORE COOLING SYSTEM

The primary function of the passive core cooling system is to provide emergency core cooling following postulated design basis events. To accomplish this primary function, the passive core cooling system is designed to perform the following functions:

- Emergency core decay heat removal

  Provide core decay heat removal during transients, accidents, or whenever the normal heat removal paths are lost. This heat removal function is available at reactor coolant system conditions including shutdowns and refuelings.

- Reactor coolant system emergency makeup and boration

  Provide reactor coolant system makeup and boration during transients or accidents when the normal reactor coolant system makeup supply from the chemical and volume control system is unavailable or is insufficient.

- Safety injection

  Provide safety injection to the reactor coolant system to provide adequate core cooling for the complete range of loss of coolant accidents, up to and including the double-ended rupture of the largest primary loop reactor coolant system piping.

- Containment sump pH control

  Provide for chemical addition to the containment sump during post-accident conditions to establish floodup chemistry conditions that support radionuclide retention with high radioactivity in containment and to prevent corrosion of containment equipment during long-term floodup conditions.

The passive core cooling system is designed to operate without the use of active equipment such as pumps and ac power sources. The passive core cooling system depends on reliable passive components and processes such as gravity injection and expansion of compressed gases. The passive core cooling system does require a one-time alignment of valves upon actuation of the specific components.

6.3.1 Design Basis

The passive core cooling system is designed to perform its safety-related functions based on the following considerations:

- It has component redundancy to provide confidence that its safety-related functions are performed, even in the unlikely event of the most limiting single failure occurring coincident with postulated design basis events.

- Components are designed and fabricated according to industry standard quality groups commensurate with its intended safety-related functions.

- It is tested and inspected at appropriate intervals, as defined by the ASME Code, Section XI, and by technical specifications.

- It performs its intended safety-related functions following events such as fire, internal missiles or pipe breaks.

- It is protected from the effects of external events such as earthquakes, tornadoes, and floods.

- It is designed to be sufficiently reliable, considering redundancy and diversity, to support the plant core melt frequency and significant release frequency goals.

6.3.1.1 Safety Design Basis

The passive core cooling system is designed to provide emergency core cooling during events involving increases and decreases in secondary side heat removal and decreases in reactor coolant system inventory. Subsection 6.3.3 provides a description of the design basis events. The performance criteria are provided in Subsection 6.3.1 and also described in Chapter 15, under the respective event sections.

6.3.1.1.1 Emergency Core Decay Heat Removal

For postulated nonLOCA events, where a loss of capability to remove core decay heat via the steam generators occurs, the passive core cooling system is designed to perform the following functions:

- The passive residual heat removal heat exchangers automatically actuate to provide reactor coolant system cooling and to prevent water relief through the pressurizer safety valves.

- The passive residual heat removal heat exchangers are capable of automatically removing core decay heat following such an event, assuming the steam generated in the in-containment refueling water storage tank is condensed on the containment vessel and returned by gravity via the in-containment refueling water storage tank condensate return gutter. The passive heat removal heat exchangers should provide decay heat removal for at least 72 hours if no condensate is recovered.

- The passive residual heat removal heat exchangers, in conjunction with the passive containment cooling system, are designed to remove decay heat for an indefinite time in a closed-loop mode of operation. In addition, the passive residual heat removal heat exchangers are designed to cool the reactor coolant system to 400°F in 72 hours, with or without reactor coolant pumps operating. This allows the reactor coolant system to be depressurized and the stress in the reactor coolant system and connecting pipe to be reduced to low levels. This also allows plant conditions to be established for initiation of normal residual heat removal system operation.

- During a steam generator tube rupture event, the passive residual heat removal heat exchangers remove core decay heat and sufficiently reduce reactor coolant system temperature and pressure, equalizing with steam generator pressure and terminating break flow, without overfilling the steam generator or actuating the ADS.

6.3.1.1.2 Reactor Coolant System Emergency Makeup and Boration

For postulated nonLOCA events, sufficient core makeup water inventory is automatically provided to keep the core covered and to allow for decay heat removal. In addition, this makeup prevents actuation of the automatic depressurization system for a significant time.

For postulated events resulting in an inadvertent cooldown of the reactor coolant system, such as a steam line break, sufficient borated water shall be automatically provided to makeup for reactor coolant system shrinkage. The borated water also counteracts the reactivity increase caused by the resulting system cooldown.

For a Condition II steam line break described in Chapter 15, return to power is acceptable if there is no core damage. For this event, the automatic depressurization system is not actuated to avoid the unnecessary loss of an activity barrier.

For a large steam line break, the peak return to power is limited so that the offsite dose limits are satisfied. Following either of these events, the reactor coolant system is automatically brought to a subcritical condition, consistent with the passive containment cooling capabilities.

For safe shutdown, the passive core cooling system is designed to supply sufficient boron to the reactor coolant system to maintain the technical specification shutdown margin for cold, post-depressurization conditions, with the most reactive rod fully withdrawn from the core. The automatic depressurization system is not expected to actuate for these events.

6.3.1.1.3 Safety Injection

The passive core cooling system provides sufficient water to the reactor coolant system to mitigate the effects of a loss of coolant accident. In the event of a large loss of coolant accident, up to and including the rupture of a hot or cold leg pipe, where essentially all of the reactor coolant volume is initially displaced, the passive core cooling system rapidly refills the reactor vessel, refloods the core, and continuously removes the core decay heat. A large break is a rupture with a total cross-sectional area equal to or greater than one square foot. Although the criteria for mechanistic pipe break are used to limit the size of pipe rupture considered in the design and evaluation of piping systems, as described in Subsection 3.6.3, such criteria are not used in the design of the passive core cooling system.

Sufficient water is provided to the reactor vessel following a postulated loss of coolant accident so that the performance criteria for emergency core cooling systems, described in Chapter 15, are satisfied.

The automatic depressurization system valves, provided as part of the reactor coolant system, are designed so that together with the passive core cooling system they:

- Satisfy the small loss of coolant accident performance requirements
- Provide effective core cooling for loss of coolant accidents from when the passive core cooling system is actuated through the long-term cooling mode.

6.3.1.1.4 Safe Shutdown

The functional requirements for the passive core cooling system specify that the plant be brought to a stable condition using the passive residual heat removal heat exchangers for events not involving a loss of coolant. For these events, the passive core cooling system, in conjunction with the passive containment cooling system, has the capability to establish safe shutdown conditions, cooling the reactor coolant system to about 400°F in 72 hours, with or without the reactor coolant pumps operating.

The core makeup tanks automatically provide injection to the reactor coolant system as the temperature decreases and pressurizer level decreases, actuating the core makeup tanks. The passive core cooling system can maintain stable plant conditions for a long time in this mode of operation, without ADS actuation. For example, with a RCS leak rate of 1 gpm, stable plant conditions can be maintained for about 120 hours before ADS would occur.

For loss of coolant accidents and for postulated events where ac power sources are lost, or when the core makeup tank levels reach the automatic depressurization system actuation setpoint, the automatic depressurization system initiates. This results in injection from the accumulators and subsequently from the in-containment refueling water storage tank, once the reactor coolant system is nearly depressurized. For these conditions, the reactor coolant system depressurizes to saturated conditions at about 240°F within 24 hours. The passive core cooling system can maintain this safe shutdown condition indefinitely for the plant.

The basis used to define the passive core cooling system functional requirements are derived from Section 7.4 of the Standard Review Plan. The functional requirements are met over the range of anticipated events and single failure assumptions. The primary function of the passive core cooling system during a safe shutdown using only safety-related equipment is to provide a means for boration, injection, and core cooling. Details of the safe shutdown design bases are presented in Subsection 5.4.7 and Section 7.4.

6.3.1.1.5 Containment Sump pH Control

The passive core cooling system is capable of maintaining the desired post-accident pH conditions in the recirculation sump water after containment floodup. The pH adjustment tank is capable of maintaining containment sump pH within a range of 7.0 to 9.5, to enhance radionuclide retention in the containment sump and to prevent stress corrosion cracking of containment components during long-term containment floodup.

6.3.1.1.6 Reliability Requirements

The passive core cooling system design satisfies a variety of reliability requirements, including redundancy (such as for components, power supplies, actuation signals, and instrumentation), equipment testing to confirm operability, procurement of qualified components, and provisions for periodic maintenance. In addition, the design provides protection in a number of areas including:

- Single active and passive component failures
- Spurious failures
- Physical damage from fires, flooding, missiles, pipe whip, and accident loads
- Environmental conditions such as high-temperature steam and containment floodup.

Subsection 6.3.1.2 includes specific nonsafety-related design requirements that help to confirm satisfactory system reliability.

6.3.1.2 Power Generation Design Basis

The passive core cooling system is designed to be sufficiently reliable to support the probabilistic risk analysis goals for core damage frequency and severe release frequency. In assessing the reliability for probabilistic risk analysis purposes, best estimate analysis is used for both the passive core cooling system performance and for plant response.

In the event of a small loss of coolant accident, the passive core cooling system limits the increase in peak clad temperature and core uncovery with design basis assumptions. For pipe ruptures of less than eight-inch nominal diameter size, the passive core cooling system is designed to prevent core uncovery with best estimate assumptions.

The passive residual heat removal heat exchangers and the in-containment refueling water storage tank are designed to delay significant steam release to the containment for at least one hour.

Two 100 percent capacity passive residual heat removal heat exchangers are included in the passive core cooling system. They provide sufficient redundancy so that an individual heat exchanger can be isolated in the event of tube leakage and plant operation can continue without having to shut down to repair the leaking heat exchanger.

The frequency of automatic depressurization system actuation is limited to a low probability to reduce safety risks and to minimize plant outages. Equipment is located so that it is not flooded or it is designed so that it is not damaged by the flooding. Major plant equipment is designed for multiple occurrences without damage.

The pH control equipment is designed to minimize the potential for system leakage during standby operation, and also to minimize the potential for and the impact of an inadvertent actuation.

The passive core cooling system is capable of supporting the required testing and maintenance, including capabilities to isolate and drain equipment.

6.3.2 System Design

The passive core cooling system is a seismic Category I, safety-related system. It consists of two core makeup tanks, two accumulators, the in-containment refueling water storage tank, two passive residual heat removal heat exchangers, the pH adjustment tank, two ADS spargers and associated piping, valves, instrumentation, and other related equipment. The automatic depressurization system valves, which are part of the reactor coolant system, also provide important passive core cooling functions.

The passive core cooling system is designed to provide adequate core cooling in the event of design basis events. The redundant onsite safety-related dc power sources provide power such that protection is provided for a loss of ac power sources, coincident with an event, assuming a single failure has occurred.

6.3.2.1 Schematic Piping and Instrumentation Diagrams

Figures 6.3-1 through 6.3-4 show the piping and instrumentation drawings of the passive core cooling system. Process flow diagrams are shown in Figures 6.3-5 through 6.3-7. Tables 6.3-1 through 6.3-3 provide a summary of the expected fluid conditions for the various locations shown on the process flow diagrams, for the specific plant conditions identified -- safety injection, decay heat removal, and sump pH control.

The passive core cooling system is designed to supply the core cooling flow rates to the reactor coolant system specified in Chapter 15 for the accident analyses. The accident analyses flow rates and heat removal rates are calculated by assuming a range of component parameters, including best estimate and conservatively high and low values.

The passive core cooling system design is based on the six major components, listed in Subsection 6.3.2.2, that function together in various different combinations to support the four basic passive core cooling system functions:

- Emergency decay heat removal
- Emergency reactor makeup/boration
- Safety injection
- Containment sump pH control.

6.3.2.1.1 Emergency Core Decay Heat Removal at High Pressure and Temperature Conditions

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.2.1.2 Reactor Coolant System Emergency Makeup and Boration

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.2.1.3 Safety Injection During Loss of Coolant Accidents

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.2.1.4 Containment Sump pH Control

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.2.1.5 Passive Core Cooling System Actuation

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.2.2 Equipment and Component Descriptions

[Westinghouse Proprietary]
[Provided under separate cover]

AP600

6.3.2.2.1 Core Makeup Tanks

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.2.2.2 Accumulators

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.2.2.3 In-Containment Refueling Water Storage Tank

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.2.2.4 pH Adjustment Tank

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.2.2.5 Passive Residual Heat Removal Heat Exchangers

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.2.2.6 Depressurization Spargers

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.2.2.7 Valves

Design features used to minimize leakage for valves in the passive core cooling system include:

- Where possible, packless valves are used.
- Other valves which are normally open, except check valves and those which perform control function, are provided with back seats to limit stem leakage.
- Stem leakage collection is used for valves larger than two inches in diameter.

6.3.2.2.7.1 Manual Globe, Gate, and Check Valves

Gate valves have backseats and external screw and yoke assemblies.

Globe valves, are full-ported with external screw and yoke construction.

Check valves are spring-loaded lift piston types for sizes 2 inches and smaller, and swing-type for sizes 2.5 inches and larger. Stainless steel check valves have no penetration welds other than the inlet, outlet, and bonnet. The check valve hinge is serviced through the bonnet.

The stem packing and gasket of the stainless steel manual globe and gate valves are similar to those described in Subsection 6.3.2.2.7.3 for motor-operated valves. Carbon steel manual valves are employed to pass nonradioactive fluid only and, therefore, do not contain the double packing and seal weld provisions.

6.3.2.2.7.2 Manual Valves

Where manual valves are locked open, administratively controlled, and provided with redundant status in the main control room, bypass and inoperable status indication is provided according to Regulatory Guide 1.47. Compliance with the recommendations of Regulatory Guide 1.47 is provided in Subsection 1.9.1.

Manual valves are generally used as maintenance isolation valves or throttling valves. When used for these functions they are under administrative control, which requires them to be locked in the correct position. They are located so that no single valve can isolate redundant passive core cooling system equipment or they are provided with alarms in the main control room to indicate mispositioning.

To help preclude the possibility of passive core cooling system degradation due to valve mispositioning, line connections such as vent and drain lines, test connections, pressure points, flow element test points, flush connections, local sample points, and bypass lines are provided with double isolation or sealed barriers. The isolation is provided by one of the following methods:

- Two valves in a series
- A single valve with a screwed cap or blind flange
- A single locked-closed valve
- A blind flange.

6.3.2.2.7.3 Motor-Operated Valves

The most common AP600 valve body design for motor-operated valves is a gate valve. Industry issues will be considered in the design of these valves. The motor operators are conservatively sized, considering the frictional component of the hydraulic unbalance on the valve disc, the disc face friction, and the packing box friction. Some special purpose valves have slow operators without motor brakes. For motor-operated valves, the valve disc is guided throughout the full disc travel to prevent cocking and to provide ease of gate movement. The seating surfaces are hard-faced to prevent galling and to reduce wear.

Where a gasket is employed for the body to bonnet joint, it is a fully trapped, controlled compression, spiral wound asbestos (or a qualified asbestos substitute) gasket with provisions for seal welding. The valve stuffing boxes are designed with a lantern ring leak-off connection.

6.3.2.2.7.4 Motor-Operated Valve Controls

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.2.2.7.5 Accumulator Motor-Operated Valve Control

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.2.2.7.6 Automatic Depressurization Valves

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.2.2.7.7 Low Differential Pressure Opening Check Valves

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.2.2.7.8 Accumulator Check Valves

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.2.2.7.9 Relief Valves

Relief valves are installed for passive core cooling system accumulators and the pH adjustment tank to protect the tanks from overpressure.

The passive core cooling system piping is reviewed to identify those lengths of piping that are isolated by normally closed valves and that do not have pressure relief protection in the piping section between the valves.

These piping sections include:

- Portions of in-containment passive core cooling system test lines that are not passive core cooling system accident mitigation flow paths and are not needed to achieve safe shutdown

- Piping vents, drains, and test connections that typically have two closed valves or one closed valve and a blind flange

- Check valve test lines with sections isolated by two normally closed valves.

The piping vents, drains, test connections, and check valve lines have design pressure/temperature conditions compatible with the process piping to which they connect. Therefore, valve leakage does not overpressurize the isolated piping sections and pressure relief provisions are not required.

6.3.2.3 Applicable Codes and Classifications

Sections 5.2 and 3.2 list the equipment ASME Code and seismic classification for the passive core cooling system. Most of the piping and components of the passive core cooling system within containment are AP600 Equipment Class A, B, or C and are designed to meet seismic Category I requirements. Some system piping and components that do not perform safety-related functions are nonsafety-related.

The requirements for the control, actuation, and Class 1E power supplies are presented in Chapters 7 and 8.

6.3.2.4 Material Specifications and Compatibility

Materials used for engineered safety feature components are given in Section 6.1. Materials for passive core cooling system components are selected to meet the applicable material requirements of the codes in Section 5.2, as well as the following additional requirements:

- Parts of components in contact with borated water are fabricated of, or clad with, austenitic stainless steel or an equivalent corrosion-resistant material.
- Internal parts of components in contact with containment emergency sump solution during recirculation are fabricated of austenitic stainless steel or an equivalent corrosion resistant material.
- Valve seating surfaces are hard-faced to prevent failure and to reduce wear. Zero or low cobalt hard facing is used where performance is acceptable.
- Valve stem materials are selected for their corrosion resistance, high-tensile properties, and their resistance to surface scoring by the packing.

Section 6.1 summarizes the materials used for passive core cooling system components.

6.3.2.5 System Reliability

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6.3.2.5.1 Response to Active Failure

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.2.5.2 Response to Passive Failure

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.2.5.3 Lag Times

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.2.5.4 Potential Boron Precipitation

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.2.5.5 Safe Shutdown

During a safe shutdown, the passive core cooling system provides redundancy for boration, makeup, and heat removal functions. Details of the safe shutdown design are described in Section 7.4.

6.3.2.6 Protection Provisions

The measures taken to protect the system from damage that might result from various events are described in other sections, as listed below.

- Protection from dynamic effects is presented in Section 3.6.
- Protection from missiles is presented in Section 3.5.
- Protection from seismic damage is presented in Sections 3.7, 3.8, 3.9, and 3.10.
- Protection from fire is presented Subsection 9.5.1.
- Environmental qualification of equipment is presented in Section 3.11.
- Thermal stresses on the reactor coolant system are presented in Section 5.2.

6.3.2.7 Provisions for Performance Testing

[Westinghouse Proprietary]
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6.3.2.8 Manual Actions

The passive core cooling system is automatically actuated for those events as presented in Subsection 6.3.3. Following actuation, the passive core cooling system continues to operate in the injection mode until the transition to recirculation initiates automatically following containment floodup.

Although the passive core cooling system operates automatically, operator actions would be beneficial, in some cases, in reducing the consequences of an event. For example, in a steam generator tube rupture with no operator action, the protection and safety monitoring system automatically terminates the leak, prevents steam generator overfill, and limits the offsite doses. However, the operator can initiate actions, similar to those taken in current plants, to identify and isolate the faulted steam generator, cool down and depressurize the reactor coolant system to terminate the break flow to the steam generator, and stabilize plant conditions.

Section 7.5 describes the post-accident monitoring instrumentation available to the operator in the main control room following an event.

6.3.3 Performance Evaluation

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.3.1 Increase in Heat Removal by the Secondary System

A number of events that could result in an increase in heat removal from the reactor coolant system by the secondary system have been postulated. For each event, consideration has been given to operation of nonsafety-related systems that could affect the event results. The operation of the startup feedwater system and the chemical and volume control system makeup pumps can affect these events. Analyses of these events, both with and without these nonsafety-related systems operating, are presented in Section 15.1. For those events resulting in passive core cooling system actuation, the following summarizes passive core cooling system performance:

6.3.3.1.1 Inadvertent Opening of a Steam Generator Relief or Safety Valve

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.3.1.2 Steam System Pipe Failure

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.3.2 Decrease in Heat Removal by the Secondary System

A number of events have been postulated that could result in a decrease in heat removal from the reactor coolant system by the secondary system. For each event, consideration has been given to operation of nonsafety-related systems that could affect the consequences of an event. The operation of the startup feedwater system and the chemical and volume control system makeup pumps can affect these events. Analyses of these events, both with and without these nonsafety-related systems operating, are presented in Section 15.2. For those events resulting in passive core cooling system actuation, the following summarizes passive core cooling system performance:

6.3.3.2.1 Loss of Main Feedwater

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.3.2.2 Feedwater System Pipe Failure

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.3.3 Decrease in Reactor Coolant System Inventory

A number of events have been postulated that could result in a decrease in reactor coolant system inventory. For each event, consideration has been given to operation of nonsafety-related systems that could affect the consequences of the event. The operation of the startup feedwater system and the chemical and volume control system makeup pumps can affect these events. Analyses of these events, both with and without these nonsafety-related systems operating, are presented in Section 15.6. For those events which result in passive core cooling system actuation, the following summarizes passive core cooling system performance:

6.3.3.3.1 Steam Generator Tube Rupture

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.3.3.2 Loss of Coolant Accident

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.3.3.3 Passive Residual Heat Removal Heat Exchanger Tube Rupture

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.3.4 Shutdown Events

The passive core cooling system components are available whenever the reactor is critical and when reactor coolant energy is sufficiently high to require passive safety injection. During low-temperature physics testing, the core decay heat levels are low and there is a negligible amount of stored energy in the reactor coolant. Therefore, an event comparable in severity to events occurring at operating conditions is not possible and passive core cooling system equipment is not required. The possibility of a loss of coolant accident during plant startup and shutdown has been considered.

During shutdown conditions, some of the passive core cooling system equipment is isolated. In addition, since the normal residual heat removal system is not a safety-related system, its loss is considered.

As a result, gravity injection is automatically actuated when required during shutdown conditions prior to refueling cavity floodup, as discussed in Subsection 6.3.3. The operator can also manually actuate other passive core cooling system equipment, such as the passive residual heat removal heat exchangers, if required for accident mitigation during shutdown conditions when the equipment does not automatically actuate.

6.3.3.4.1 Loss of Startup Feedwater

During Hot Standby, Cooldowns, and Heatups

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.3.4.2 Loss of Normal Residual Heat Removal Cooling With The Reactor Coolant System Pressure Boundary Intact

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.3.4.3 Loss of Normal Residual Heat Removal Cooling During Midloop Operation

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.3.4.4 Loss of Normal Residual Heat Removal Cooling During Refueling

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.4 Post-72 Hour Actions

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.5 Limits on System Parameters

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.6 Inspection and Testing Requirements

6.3.6.1 Preoperational Testing

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.6.2 In-Service Testing and Inspection

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.7 Instrumentation Requirements

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.7.1 Pressure Indication

6.3.7.1.1 Accumulator Pressure

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.7.1.2 Passive Core Cooling System Test Header Pressure Indication

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.7.2 Temperature Indication

6.3.7.2.1 Core Makeup Tank Inlet and Outlet Line Temperature

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.7.2.2 Passive Residual Heat Removal Heat Exchanger Inlet and Outlet Line Temperature

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.7.2.3 In-Containment Refueling Water Storage Tank Temperature

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.7.3 Flow Indication

6.3.7.3.1 Passive Core Cooling System Test Header Flow Indication

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.7.3.2 Passive Residual Heat Removal Heat Exchanger Outlet Flow

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.7.4 Level Indication

6.3.7.4.1 Core Makeup Tank Level

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.7.4.2 Accumulator Level

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.7.4.3 In-Containment Refueling Water Storage Tank Level

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.7.4.4 Containment Sump Level

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.7.5 Containment Radiation Level

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.7.6.1 Valve Position Indication

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.7.6.2 Valve Position Control

6.3.7.6.2.1 Passive Residual Heat Removal Heat Exchanger Outlet Valve Position Control

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.7.7 Automatic Depressurization System Actuation at 24 Hours

[Westinghouse Proprietary]
[Provided under separate cover]

6.3.8 References

[Westinghouse Proprietary]
[Provided under separate cover]

APPENDIX A-2

AP600 PXS AND RCS P&ID

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