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RELAP5 Thermal-Hydraulic Analysis of the SNUPPS Pressurized Water Reactor

Prepared by C. M. Kullberg

Idaho National Engineering Laboratory
EG&G Idaho, Inc.

Prepared for
U.S. Nuclear Regulatory Commission

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Prepared by
C. M. Kullberg

Idaho National Engineering Laboratory
EG&G Idaho, Inc.
P. O. Box 1625
Idaho Falls, ID 83415

Prepared for
Division of Systems Research
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555
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ABSTRACT

Thermal-hydraulic analyses of five hypothetical accident scenarios were performed with the RELAP5 computer code for the Westinghouse Standardized Nuclear Unit Power Plant System pressurized water reactor. This work was sponsored by the US Nuclear Regulatory Commission and is being done in conjunction with future analysis work at the US Nuclear Regulatory Commission Technical Training Center in Chattanooga, TN. These accident scenarios were chosen to assess and benchmark the thermal-hydraulic capabilities of the Technical Training Center Standardized Nuclear Unit Power Plant System simulator to model abnormal transient conditions.

SUMMARY

U.S. Nuclear Regulatory Commission (NRC) rules now require all plants to have a plant specific simulator for operator training with the capability to model plant operation and transients in an environment closely resembling the plant control room. For many transients and accidents, current plant simulators may produce incorrect responses or be unable to model them. In an effort to study the capabilities of current simulators, the NRC initiated a project that will evaluate existing and upgraded plant simulators using the advanced thermal-hydraulic systems codes like RELAP5 and TRAC-BWR.

The Westinghouse Standardized Nuclear Unit Power Plant (SNUPPS) simulator, located at the NRC Technical Training Center (TTC) was modeled using RELAP5. The model, a four loop pressurized water reactor (PWR), contained detailed thermal-hydraulic representations of the pertinent PWR primary and secondary systems, including the feedwater train and steam lines. Detailed models of the key plant control systems were included.

The RELAP5 model was used to analyze five separate transients, selected to cover a wide range of possible thermal-hydraulic conditions that could occur in a reactor accident. The transients were: (a) loss of AC power, (b) small break Loss-of-Coolant-Accident with loss of AC power, (c) failed open pressurizer safety valve, (d) main steam line break with a steam generator tube rupture, and (e) loss of feedwater without scram.

In general, the calculated RELAP5 trends were reasonable for the scenarios studied in the analysis, and will provide a good basis for comparison with simulator data, based on the review by experienced operators and plant analysts. Some uncertainties in boundary conditions and modeling options have not been resolved and could affect simulator/RELAP5 comparisons.

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ACRONYMS

ANS	-	American Nuclear Society
ASG	-	Affected steam generator
BOL	-	Beginning of life
CVCS	-	Chemical and volume control system
DG	-	Diesel generator
ECC	-	Emergency core cooling
HPI	-	High pressure injection
INEL	-	Idaho National Engineering Laboratory
LOCA	-	Loss-of-coolant accident
LPI	-	Low pressure injection
MSIV	-	Main steam isolation valve
MSLB	-	Main steam line break
NRC	-	Nuclear Regulatory Commission
PLCS	-	Pressurizer level control system
PORV	-	Power operated relief valve
PPCS	-	Pressurizer pressure control system
PWR	-	Pressurized water reactor
RCP	-	Reactor coolant pump
SDCS	-	Steam dump control system
SGLCS	-	Steam generator level control system
SGTR	-	Steam generator tube rupture
SNUPPS	-	Standard Nuclear Unit Power Plant
TTC	-	Technical Training Center
USG	-	Unaffected steam generator

RELAP5 THERMAL-HYDRAULIC ANALYSIS OF THE SNUPPS PRESSURIZED WATER REACTOR

1. INTRODUCTION

In the eleven years since the accident at Three Mile Island, the need for effective reactor operator training has received significant attention in the nuclear industry. US Nuclear Regulatory Commission (NRC) rules now call for all plants to have a plant specific simulator for operator training with the capability to model plant operation and transients in an environment closely resembling the plant control room. In general, the simulators perform an outstanding job of simulating normal plant evolutions. However, for many transients and accidents, the current simulators may produce incorrect responses or be unable to model them. The learning experience provided the operators by such simulators maybe faulty or nonexistent for these situations.

In an effort to study the capabilities of current simulators, the NRC initiated a project that will evaluate the capabilities of existing and upgraded plant simulators using advanced thermal-hydraulic system codes (RELAP5, TRAC BWR). The simulators to be evaluated reside at the NRC Technical Training Center (TTC) in Chattanooga, TN. The TTC uses three resident simulators, representing Westinghouse, Babcock and Wilcox, and General Electric plants; in addition, they have use of a combustion engineering simulator at Windsor, CT. The project consists of creating advanced system code models of these plants, performing a series of transient calculations with the models, and comparing the code results with simulator results, both before and after scheduled simulator upgrades.

This report documents the RELAP5 transient analysis of the Westinghouse Standard Nuclear Unit Power Plant (SNUPPS) simulator. The five scenarios analyzed are presented in Table 1. This report will discuss only the code results; comparison with simulator data will be performed at a later time, but is not included in this report. Section 2 contains a description of the RELAP5 SNUPPS model used in the analysis. Sections 3 through 7 document the model changes, assumptions specific to each scenario, and the calculated results for the five scenarios. The conclusions drawn from the analyses are discussed in Section 8, with the references listed in Section 9.

Table 1. Summary of scenarios analyzed

Scenario	Initiating Event
1	Loss of AC Power (loss of off-site power with diesel generator failure)
2	Small break LOCA (initial 1000 gpm) with loss of AC Power
3	Failed open pressurizer safety-relief valve
4	Double-ended main steam line break with a steam generator tube rupture
5	Loss of feedwater pumps with temporary failure to scram

2. MODEL DESCRIPTION

This section summarizes the RELAP5 SNUPPS model used for the steady-state initialization and five transient simulations. The subsections describe the modeled thermal-hydraulic components, the control system model, and steady-state initialization. Calculations were performed using the RELAP5/MOD 2.5 computer code[1]. The models in this code have been extensively benchmarked and validated for a wide range of accident conditions and plant types. Information used to model the SNUPPS plant came from data collected at the TTC. Additional information was taken from a Westinghouse RESAR four loop plant model previously developed at the Idaho National Engineering Laboratory (INEL)[2,3]. This was done where direct simulator information was missing. RESAR numbers were used because of the close geometric similarities between the Westinghouse RESAR and SNUPPS configurations.

2.1 Thermal-Hydraulic Model

The RELAP5 model of the SNUPPS facility is a representation of all the major flow paths for both the primary and secondary systems. Also modeled were the primary and secondary power operated relief valves (PORVs) and safety valves. The emergency core cooling (ECC) system was included in modeling the primary side and the auxiliary feedwater system was included in the secondary side modeling. The model contained 277 volumes, 293 junctions, and 296 heat structures. A description of the primary and secondary systems are presented in the following sections. Table 2 summarizes the correspondence between the reactor system and the model components. Figures 1 through 6 illustrate the RELAP5 model nodalization scheme. In general temperature, pressure, and other calculated responses were not modeled with physical process instrumentation delay times that exist in an actual plant setting. This information was not available for the SNUPPS model.

Table 2. RELAP5 model nodalization numbering scheme

Component Number	Description
100-152	Loop A Primary System
100, 110	Hot Leg
107-1	Steam Generator Inlet Plenum
107-2 to 107-9	Steam Generator Tube Primaries
107-10	Steam Generator Outlet Plenum
120 Cold Leg Pump Suction	
125 Reactor Coolant Pump	
130,140,152	Cold Leg (pump discharge)
200-252 (Numbering Comparable to 100-152)	Loop B Primary System
300-352 (Numbering Comparable to 100-152)	Loop C Primary System
400-452 (Numbering Comparable to 100-152)	Loop D Primary System
619-647	Pressurizer
619, 620	Pressurizer Vessel
630 Pressurizer Surge Line	
644,645	PORV and Surge Tank
646,647	Safety and Surge Tank
635,637,639	Spray Lines
638,636	Spray Valves

Table 2. (continued)

Component Number	Description
131-161	Loop A Secondary System
131 Downcomer	
141 Boiler	
151 Separator	
161-1 to 161-4	Steam Dome
231-261 (Numbering Comparable to 131-161)	Loop B Secondary System
331-361 (Numbering Comparable to 131-161)	Loop C Secondary System
431-461 (Numbering Comparable to 131-161)	Loop D Secondary System
500-530	Reactor Vessel
500,502,504,506	Downcomer
508,512	Lower Plenum
514 Core	
516 Bypass	
518,520,522	Upper Plenum
524,526,528	Upper Head
530 Guide Tubes	
161-5 to 194	Loop A Feed and Steam Systems
161-5,171,173	Main Steam Line
172 MSIV	

Table 2. (continued)

Component Number	Description
174 Steam Line Check Valve	
177,178	PORV and atmosphere
175,176	Safety and atmosphere
181 Feedwater Control Valve	
184 Main Feedwater Isolation Valve	
182,187	Main Feedwater Line
189,191	Turbine Driven Auxiliary Feedwater
194,193	Motor Driven Auxiliary Feedwater
261-5 to 294	Loop B Feed and Steam Systems
(Numbering Comparable to 161-5 to 194)	
361-5 to 394	Loop C Feed and Steam Systems
(Numbering Comparable to 161-5 to 194)	
800-810	Common Steam System
800 Steam Header	
802	Main Steam Line
804,806	Turbine Stop Valve and Turbine Inlet
808,810	Steam Dump Valve and Condenser
822-878	Common Feedwater Systems
822 Condenser Hotwell	
824 Condensate Pump	
826,834	Main Feed Line
844 Low Pressure Bypass line	
830,840	Low Pressure heaters

Table 2. (continued)

Component	Description
850,852	Heater Drain System
854 Main Feedwater Header	
860 Main Feed Pump Suction	
861 Feedwater Pump	
862,867	Feed Pump Discharge
874 High Pressure Heater	
878 Main Feed Header	
700-771	ECC Systems
711,721,731,741	Loop, A,B,C, and D accumulators
712,722,732,742	Loop A,B,C and D ECC lines
714,724,734,744	Loop A,B, C and D HPI
715,725,735,745	Loop A,B, C and D LPI
771 Makeup and Letdown Injection	

2.1.1 **Primary System.**

The SNUPPS plant has four primary loops and each loop is represented in the RELAP5 model. These loops were designated as loops A, B, C, and D, respectively. Each modeled loop was composed of a hot leg, cold leg, pump suction, and U-tube steam generator section as shown in Figure 1. The pressurizer was attached to loop D and the pressurizer spray lines were connected to the cold legs of loops A and B. Attached to each cold leg was a low pressure injection (LPI) port, high pressure injection (HPI), and an accumulator with associated piping. The HPI and LPI were set up to inject one fourth of the total HPI and LPI into each loop. Also attached to the loop A cold leg was the chemical and volume control system (CVCS). Makeup and letdown functions were combined and represented by a single junction and a control system. Heat structures were added to each volume in the primary loop to represent the metal mass of the piping and the steam generator tubes. Heat structures were also used to model the pressurizer proportional and backup heaters. The reactor coolant pumps were modeled using RELAP5 pump components. Homologous curves, two-phase difference curves, and two-phase multiplier tables for head and torque from Westinghouse PWR pump data were used. There were 115 volumes associated with the primary loops.

Figure 2 shows the RELAP5 nodalization scheme for the SNUPPS vessel model. The downcomer, core bypass, lower plenum, core, upper plenum, and upper head were represented in the RELAP5 vessel model. The following vessel leakage paths were also modeled: (a) downcomer to upper plenum, (b) downcomer to upper head, (c) lower plenum to upper plenum core bypass, and (d) upper head to upper plenum via guide tube. Heat structures were modeled to simulate both the stored vessel energy and the reactor fuel rods. Decay heat was assumed to be at the American Nuclear Society (ANS) standard rate. There were 24 volumes associated with the vessel.

2.1.2 **Secondary System.**

The RELAP5 SNUPPS secondary system is shown in Figures 3 to 6. The Westinghouse model F steam generator secondary, shown in Figure 3, represents the major flow paths in the secondary side and includes the downcomer, boiler region, separator and dryer, and the steam dome. The steam generator secondary separators and steam dryers were lumped into a single hydrodynamic volume. Steam separation in the model thus took place at a single elevation rather than at two locations (separator and dryer), as in the actual steam generator hardware. Modeling experience has shown that the effect of this approximation to the flow field at the steam generator outlet would not be significant except for the main steam line break scenario. A flow restructure was modeled at the top of each steam generator steam dome. These restructures represented the actual existing venturi nozzles which limit the flow out of each steam generator in the event of a steam line break. Steam generator wide and narrow level signals based on differential level taps were also modeled using RELAP5 control variable inputs. The wide range level tap spanned almost the entire elevation from the steam generator, and the narrow range level spanned the upper region of steam generator.

The major hardware components of the steam line out to the turbine stop valves are shown in Figure 4. Each steam line connected to a common header and was modeled individually. Each line included a main steam isolation valve (MSIV), safety, and PORV valves. The steam dump, stop, and safety valves were modeled as single lumped valves with appropriate control logic to simulate the opening and closing of each individual valve in a particular bank.

The major hardware components of the feedwater system are presented in Figures 5 and 6. The feedwater system consisted of the condensate system, main feedwater system, and the auxiliary feedwater system. Included in the RELAP5 model were components to represent the feedwater heaters, condensate, heater drain, auxiliary, and feedwater pumps, feedwater isolation valves, feedwater control valves, check valves, and piping. The heater drain, condensate, and feedwater pumps were lumped in single RELAP5 pump components. These hardware components were also modeled with appropriate control variable logic to simulate responses to various transient conditions.

Several assumptions were made in the modeling of the auxiliary feedwater system. The motor (two pumps) and steam driven (one pump) auxiliary feed pumps were modeled to equally distribute liquid from the condensate storage tank to each steam generator loop. The auxiliary mass flow rate versus back pressure was assumed to match those from the RESAR model previously referenced. It was assumed that a 30 s delay time existed between the time the auxiliary feedwater system was activated and the time that auxiliary feedwater reached the steam generators. This delay time was the same used in other INEL Westinghouse plant models.

In general, the trip and control logic for various SNUPPS secondary hardware components were similar to other Westinghouse models developed at the INEL. One exception was the logic for the feedwater isolation valves. An additional trip was modeled to close these valves on a low steam generator level signal. This trip logic was based on information from a SNUPPS Westinghouse system description document found at the TTC.

There were 51 volumes representing the secondary side feedwater train, 68 volumes representing the steam generators, 25 control volumes representing the steam lines, turbine, and steam dump. Heat structures for the secondary system included the internal metal mass and primary to secondary heat slabs for each steam generator and piping for the feedwater and steam lines.

2.2 Control System Model

This section contains a summary and brief overview of the major control systems used in the SNUPPS RELAP5 model. Detailed information regarding relevant setpoints and time constants will not be mentioned due to the proprietary nature of most of this data. Set points were modeled from data provided by Westinghouse SNUPPS supporting literature including a general SNUPPS systems description document, and set point and limitations document provided to INEL by the TTC.

The primary control systems are described in Section 2.2.1 and the secondary control systems in Section 2.2.2. Because of the scope of this project, certain SNUPPS control systems were not modeled. In particular, the rod speed controller and turbine throttle valve control systems were not explicitly modeled. These control systems were assumed either not to be challenged or inoperative in the transients documented in this report.

2.2.1 Primary Control Systems.

The two key primary control systems used to model the SNUPPS transients were the pressurizer pressure control system (PPCS) and the pressurizer level control system (PLCS). The purpose of the PPCS control system was to maintain the desired primary system pressure. This function was performed using spray valves, relief valves, proportional heaters, and backup heaters. The purpose of the PLCS was to maintain the correct primary liquid inventory. This function was performed with the charging system.

The PPCS system compared a filtered pressurizer pressure reading with a set point pressure to calculate a pressure error. This error was processed into an appropriate signal to control the pressurizer heaters, spray valves, or relief valves. The PPCS system approximates an actual plant system with two exceptions. First, the spray valves did not maintain a minimum steady-state flow rate as in an actual plant. Secondly, the steady-state heater operation was different since actual environmental primary heat losses were not modeled. These above compromises were considered acceptable because of the nature of the transients analyzed in this report.

The PLCS functioned by comparing a specified level set point (calculated as a function of average primary coolant temperature) and a measured filtered level. The level signal was based on a filtered differential pressure tap measurement. The level error signal was used to control the charging flow to maintain primary coolant mass inventory. The level error signal was also used to actuate the backup heaters when the level error exceeded the high differential level setpoint. This heater response was designed to minimize possible pressure transients when excessive amounts of subcooled water entered the pressurizer. Pressurizer heater demand was blocked when the level became less than the low-level set point. The heaters were de-energized under these conditions to prevent damage to them.

2.2.2 Secondary Control Systems.

The two principal secondary side control systems modeled in the SNUPPS model were the plant trip steam dump control system (SDCS) and the steam generator level control system (SGLCS). The purpose of the plant trip SDCS was to remove stored energy from the primary system following a plant trip and bring the plant to equilibrium no-load conditions. Other operational modes for the SDCS were not challenged in the analysis and will not be discussed in this report. The purpose of the SGLCS was to maintain a proper steam generator liquid inventory. Proper inventory control ensured stable primary to secondary heat transfer as well as protecting the turbine from excessive moisture carryover.

Modulation of the steam dump valves was controlled by the SDCS by comparing the measured average primary coolant temperature and the set point no-load hot zero power temperature of 557 °F. Opening of the steam dump valves was blocked if there was not sufficient condenser vacuum, or if the primary system average temperature decreased below the minimum temperature set point.

The SGLCS used three input signals to regulate the feedwater control valves into each of the four steam generators. These signals were: (a) the steam generator narrow range level, (b) the feedwater flow rate measured down stream of the feedwater regulating valves, and (c) the steam flow rate measured at the steam generator outlet nozzles. The steam generator level signal was generated from a filtered narrow range differential pressure tap in each steam generator. Additionally, the steam generator reference level has been assumed to remain at a constant value of 50% which corresponds to plant full power conditions. For all the calculations, the SGLCS was assumed to be inactive once the auxiliary feedwater system was started.

2.3 Steady State Conditions

A steady state initialization was performed with the RELAP5 SNUPPS model. The comparisons with the simulator data, representing full power conditions, are presented in Table 3. All the numbers except the actual power magnitude were taken directly from steady-state simulator results supplied from the TTC. Other supporting documentation provided by the TTC indicates that 100% power conditions correspond to a nominal value of 3411 MW. However, the RELAP5 calculations could not generate the correct hot and cold leg temperatures unless the nominal power was reduced to 3343 MW. Differences in the modeled core bypass mass flow rate or the steady-state feedwater temperature between the SNUPPS simulator and RELAP5 models may contribute to the above differences in steady-state power levels. To compensate for the 2% discrepancy in power the current RELAP5 transient models have the core power decay curve adjusted upward by a factor of 1.02 to ensure that the total integrated decay power in the RELAP5 calculations would match those in the SNUPPS simulator calculations resulting from 3411 MW steady state operation.

Table 3. Comparison of the RELAP5 and simulator initial conditions

Plant Parameter	Simulator	RELAP5
Reactor Power (MW)	3411	3343
Primary Pressure (psia)	2252	2254
Pressurizer Level (%)	61.4	60.0
Primary Loop Flow (lb/s)	9499	9520
Average Hot Leg Temperature (°F)	618.4	618.9
Average Cold Leg Temperature (°F)	558.5	558.7
Secondary Steam Generator Average Pressure (psia)	1019	1022
Steam Generator Liquid Level (%)	50.	50.

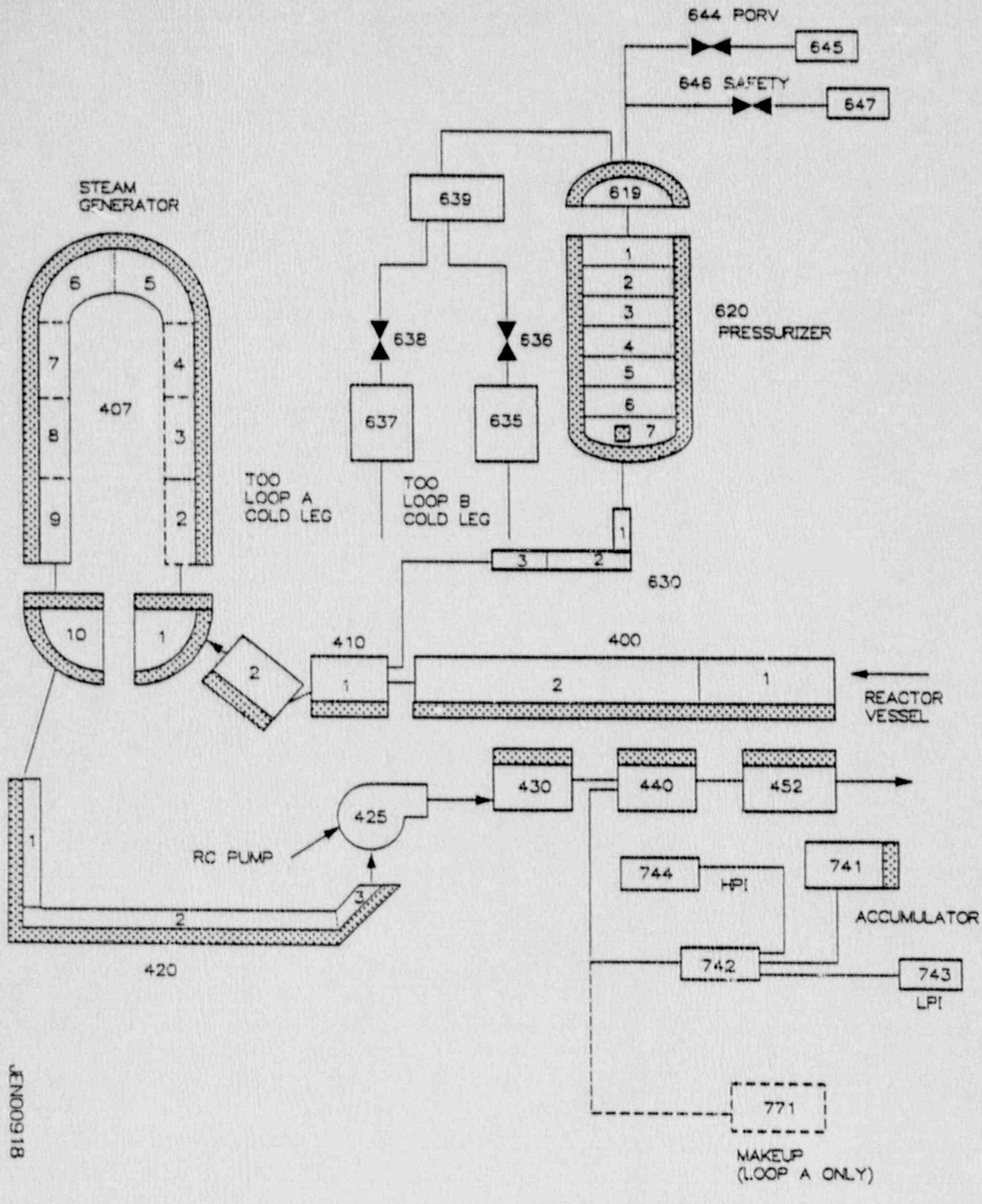

Figure 1. Nodalization of the SNUPPS primary coolant loops (Loop D shown).

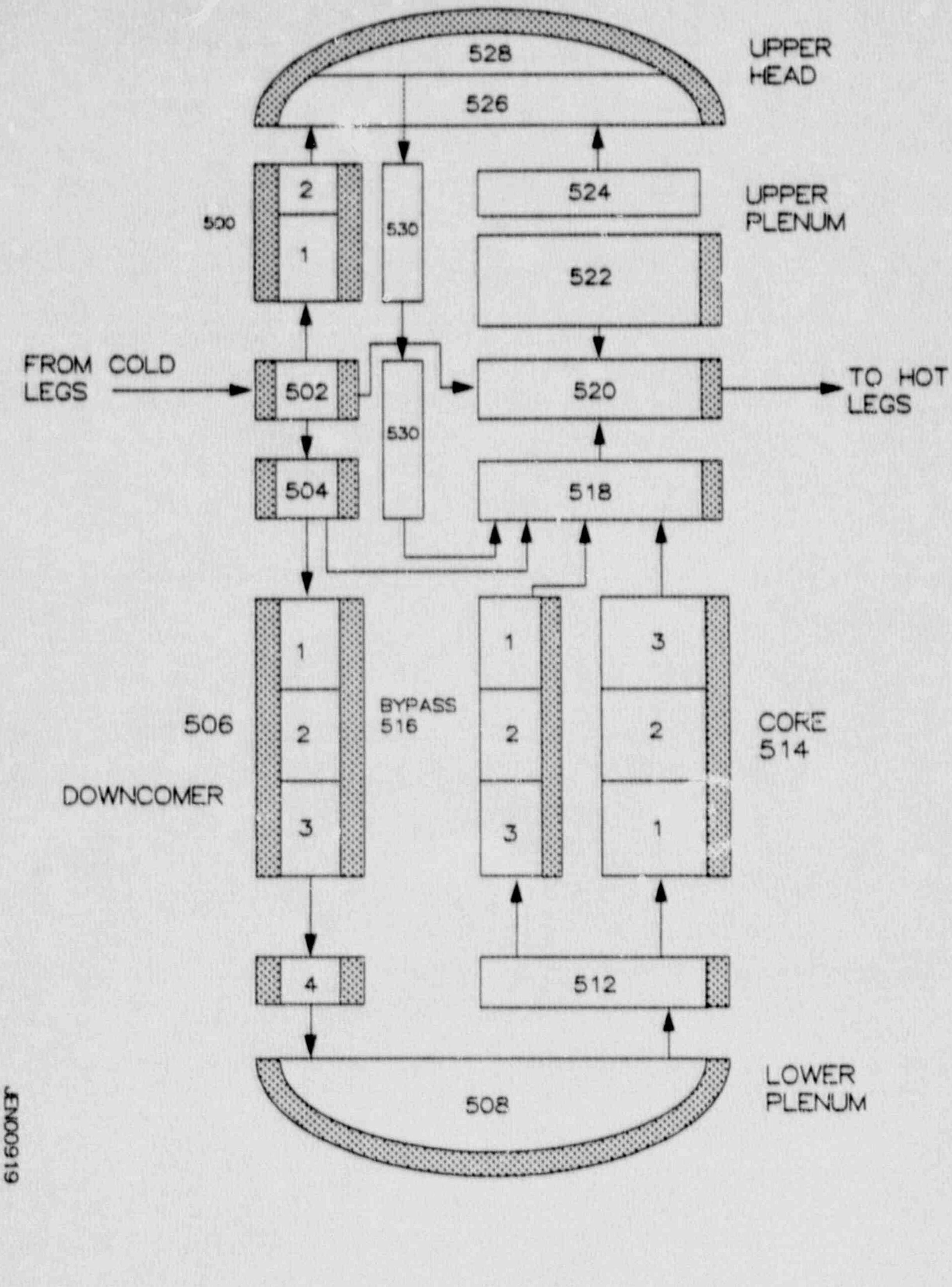

Figure 2. Nodalization of SNUPPS reactor vessel.

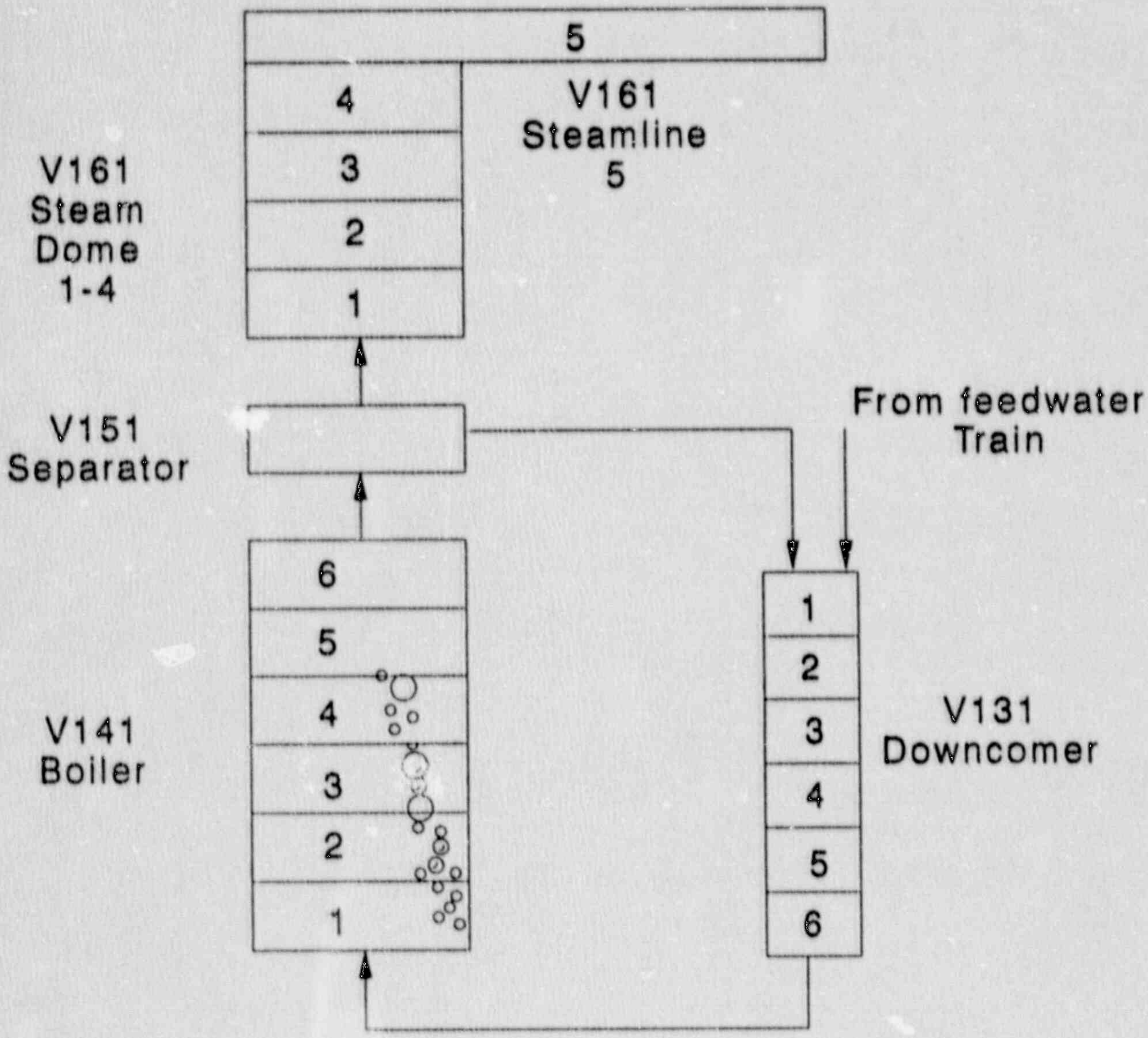

Figure 3. Nodalization of the SNUPPS Model F Steam Generator (Loop A shown).

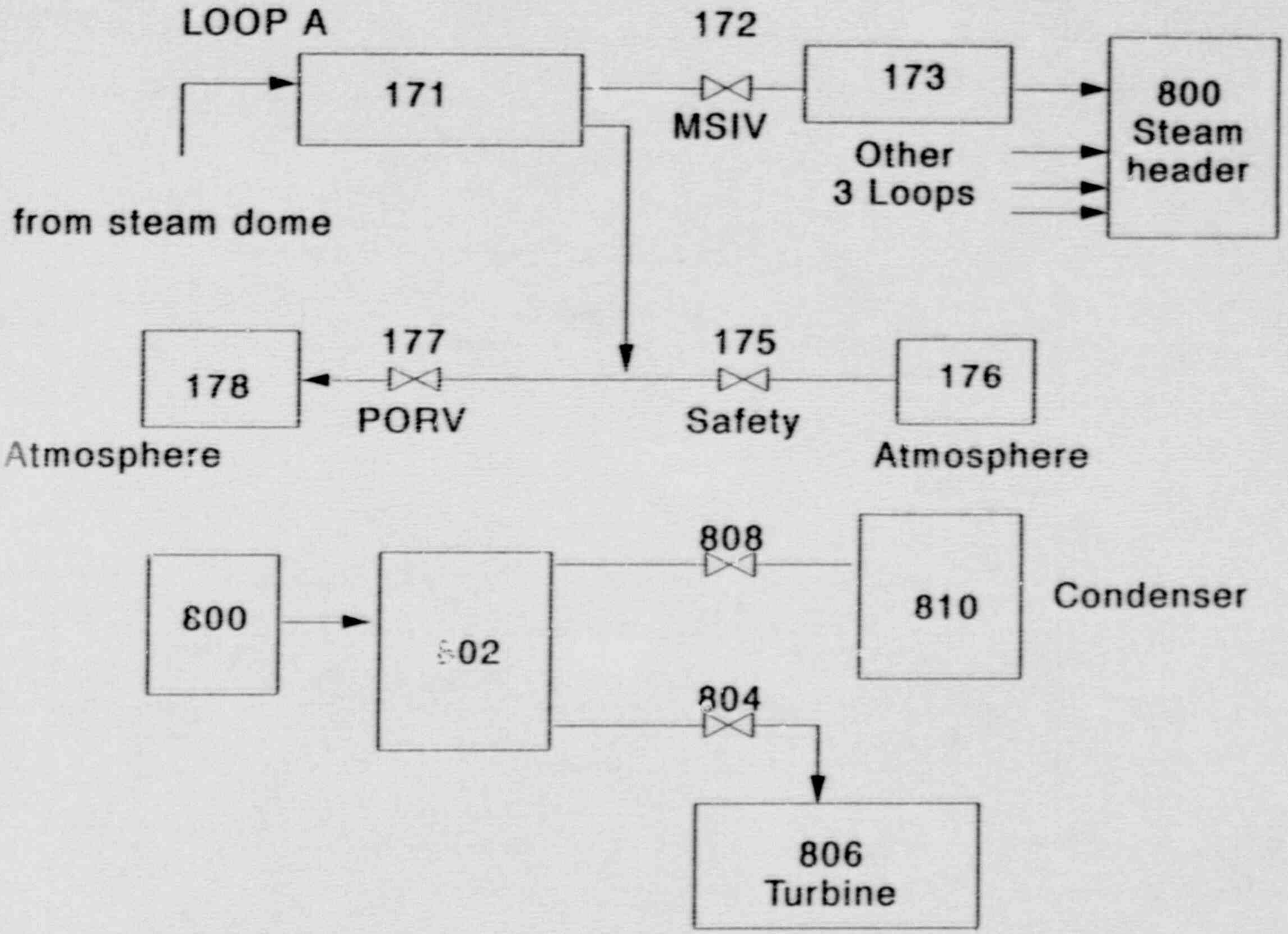

Figure 4. Nodalization of the SNUPPS Steam Line (Loop A shown).

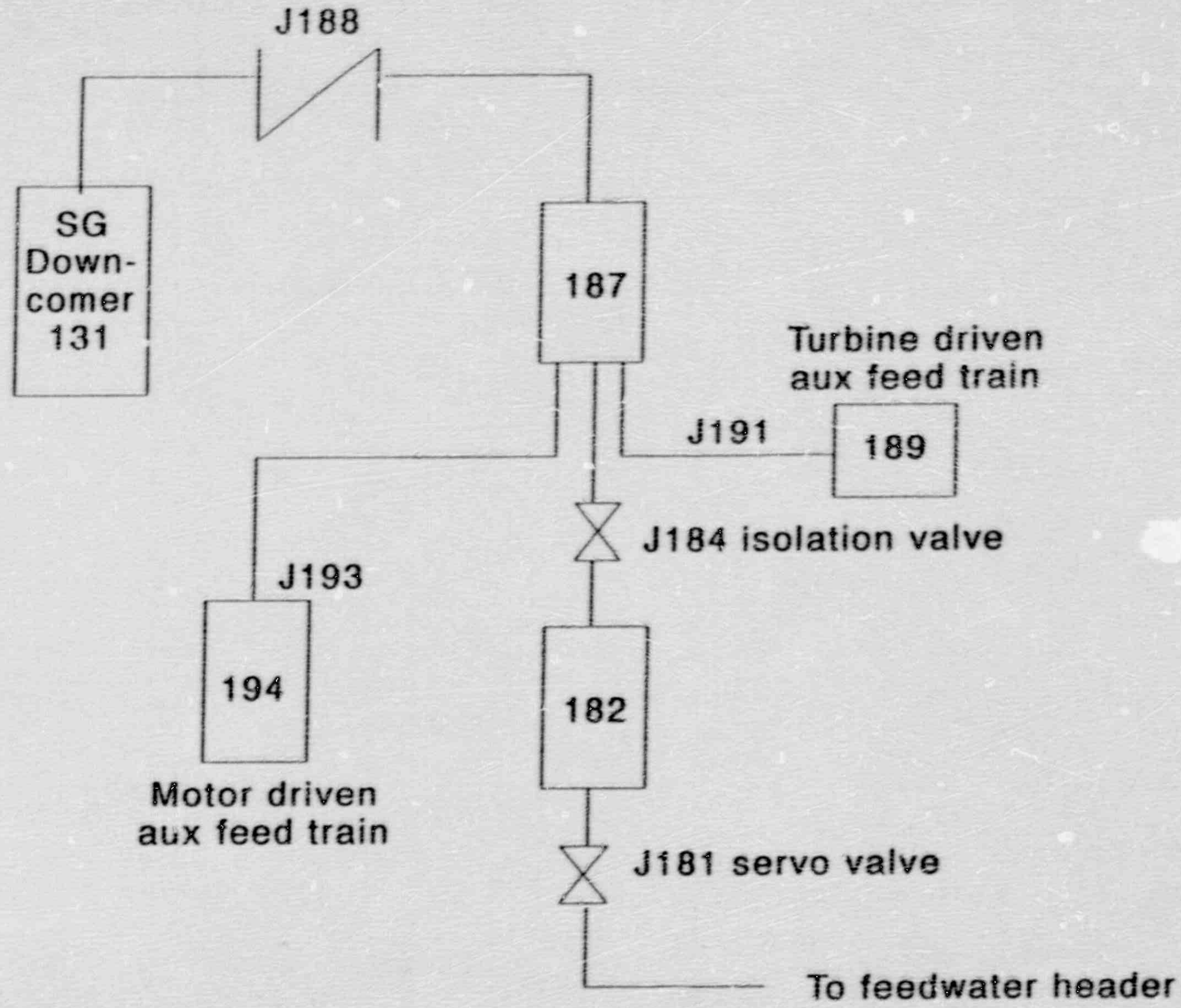

Figure 5. Nodalization of the SNUPPS Feed Train (Loop A shown).

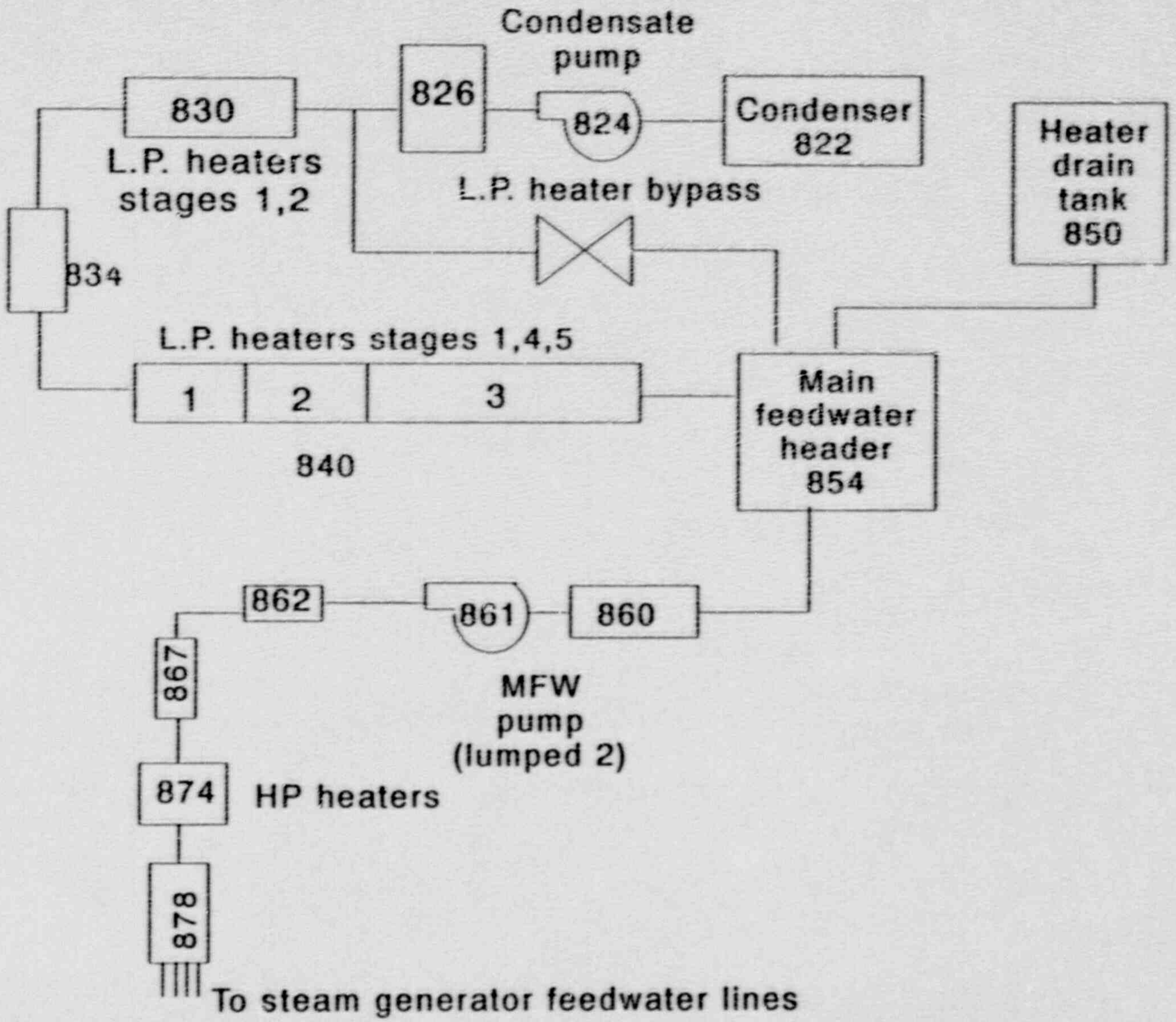

Figure 6. SNUPPS condensate train.

3. SCENARIO 1: LOSS OF AC POWER

The following section details the analysis of a loss of AC power in a SNUPPS plant initiated at full power. The subsections contain a description of the scenario, model changes and assumptions used in the calculation, and analysis of the results.

3.1 Scenario Description

The transient was defined as a loss-of-off-site-power accompanied with a failure of the diesel generators (DGs). Failure of the DGs made the ECC unavailable in the event of loss of primary coolant inventory. The loss-of-off-site-power resulted in tripping the reactor, tripping the turbine, isolating letdown and charging makeup, deactivating the pressurizer heaters, tripping the reactor coolant, feedwater, condensate, and heater drain pumps. No operator intervention was assumed during the transient.

3.2 Model Changes and Assumptions

The basic SNUPPS model used to perform the calculation is detailed in Section 2. The initial conditions assumed for the transient are documented in Section 2.3.

No nodalization changes were made to the SNUPPS RELAP5 model in simulating the loss of AC power scenario. It was assumed at the initiation of the transient that the feedwater isolation valves would ramp shut at a linear rate in 5 s (the actual feedwater valve closure signal would not necessarily be coincident with a loss of AC power). This modeling approximation was made in the absence of detailed information about the coast down characteristics of the SNUPPS feedwater train and is based on the modeling principles used in analysis of simulator plants. The primary pressurizer and secondary steam generator PORVs as well as the safety valves were assumed to be functional. After the turbine stop valves shut, the steam dump system was assumed to be unavailable because of loss of the condenser vacuum. Thus, the use of the atmospheric dump valves was needed to maintain primary to secondary cooling.

Unless manual operator action was taken, the steam generator swirl vane and moisture dryers would eventually become flooded. Flooding of this steam generator region would terminate vapor production needed to drive the steam driven auxiliary feed water pumps. Vapor production would only be re-established after a period of draining and reheating of the steam generator boiler region. Because of the complex issues needed to be resolved in modeling this kind of vapor production cycle, it was decided that the blackout simulation should be terminated prior to flooding out the steam generators.

3.3 Calculated Results

Table 4 is a summary of the sequence of events that occurred in the loss of AC power transient (the calculated event times have been rounded off to the nearest second). The transient was characterized by an initial depressurization of the primary system and pressurization of the secondary system. Figure 7 displays the pressure response of the primary and secondary sides. The depressurization of the primary side from 2250 to 2000 psia was the consequence of primary system shrinkage due to the reactor scram and sudden reduction in thermal energy supplied to the primary coolant. Coincident with reduction in primary pressure was a drop in the hot leg temperatures (Figure 8) and pressurizer level (Figure 9). The calculated loop temperature responses were symmetrical. The reduction in the pressurizer level was caused by primary system shrinkage and the subsequent out surge of liquid from the pressurizer.

In contrast to the initial reduction in hot leg temperatures, the primary cold leg temperatures temporarily increased. This was caused by the closure of the turbine stop valves which brought the secondary side pressure from approximately 1023 to 1180 psia within the first 10 s. The subsequent rise in the secondary saturation temperature caused a temporary increase in the

Table 4. Scenario 1: sequence of events

Time (s)	Event
0.	Reactor tripped, reactor coolant pumps tripped, charging and letdown isolated, pressurizer heaters tripped, feedwater pumps tripped, feedwater isolation valves begin to close, turbine stop valves begin to close, turbine stop valves begin to close, steam driven auxiliary feedwater signal generated
1.	Turbine stop valves fully closed
5.	Steam generator PORVs open, feedwater isolation valves fully closed
20.	Temporary repressurizaton of primary side begins
300.	Primary loop flows complete transition to natural circulation conditions, primary system begins slow depressurization and cooldown
900.	Steam generator PORVs close, steam generator narrow range levels begin to increase
1800.	End of simulation

primary tube side temperatures. This increase was turned around after approximately 20 s when cooler hot leg liquid was convected around the primary loop. After this 20 s period the secondary pressures stabilized to approximately 1150 psia which matched the steam generator PORV opening set point.

The temporary repressurization of the primary system after about 50 s was the consequence of the reactor coolant pump (RCP) coastdown and loop flow reduction (Figure 10). The coastdown resulted in a reduction in primary to increase in the primary pressure. The hot leg temperature stabilized as the loop mass flow rates leveled out at around 300 s. The loss of forced flow from the RCPs and the transition to natural circulation resulted in the leveling out of the primary loop flows. During this period the primary system remained subcooled.

After approximately 300 s the primary loop flows stabilized and the primary system began a slow depressurization and cooldown due to the primary to secondary heat transfer. The steam driven auxiliary feedwater flow coupled with the discharge out of the steam generator PORVs was adequate for cooling the primary side. Because the cooling capacity of the auxiliary feedwater exceeded the core decay heat production, the liquid inventories of the steam generators began to increase at around 900 s. Figure 11 shows the responses of the narrow range liquid levels. By 1400 s the rate of refill had increased significantly. This was caused by a reduced mass flow rate and eventual closure of the steam generator PORVs due to the secondary pressure dropping below the 1150 psia set point. The mismatch between the auxiliary feedwater and PORV discharge flows resulted in an increase in the steam generator levels. The secondary pressure reduction was due to an increase in the liquid subcooling in the steam generator secondary convective cooling, which led to the hot legs re-heating and a downcomer region. This caused a reduction in the latent heat vapor production and an increase in sensible heating of the secondary coolant. The long term cooling trends implied that the primary system would eventually reach a state of thermal equilibrium with the steam generator secondary sides.

At 1800 s the general primary and secondary thermal-hydraulic trends were established and the simulation was terminated before the steam generator separators were flooded. In conclusion, the RELAP5 simulation indicated that: (a) the primary and secondary thermal-hydraulic responses of the four coolant loops were symmetrical, (b) the primary coolant remained at subcooled conditions up to 1800 s, (c) the calculated trends imply that the secondary and primary systems would eventually reach a state of thermal equilibrium.

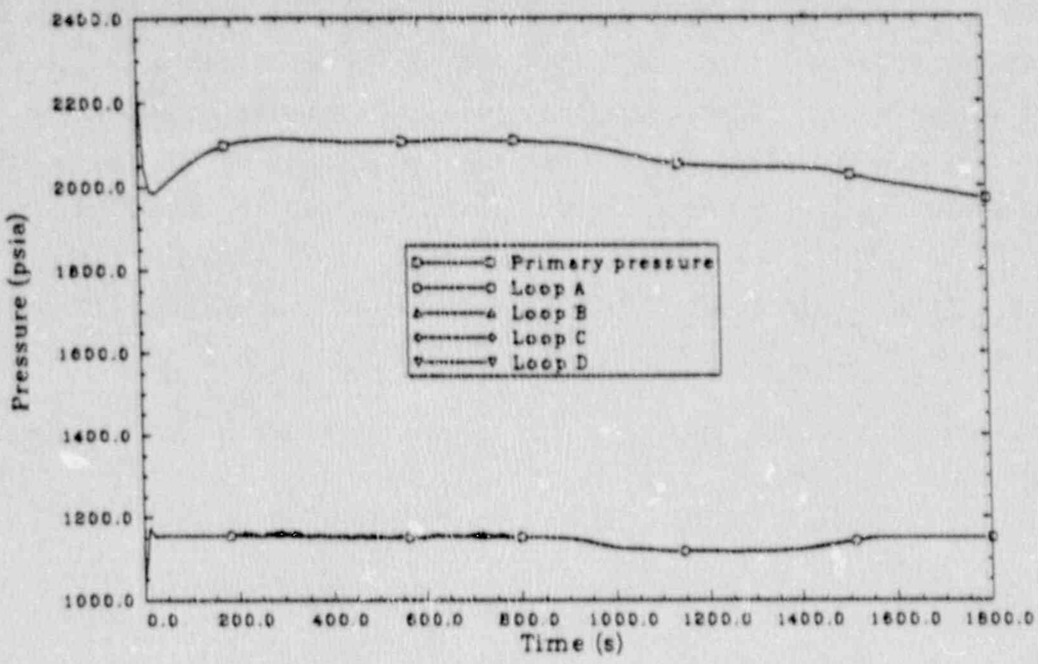

Figure 7. Scenario 1: primary and secondary pressures.

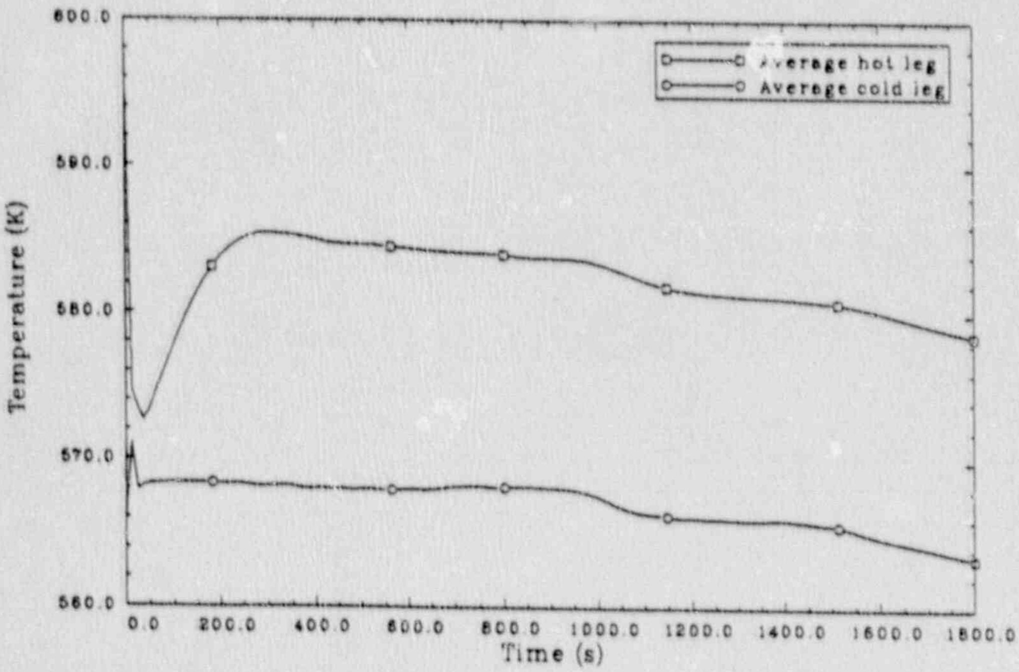

Figure 8. Scenario 1: primary hot and cold leg temperatures.

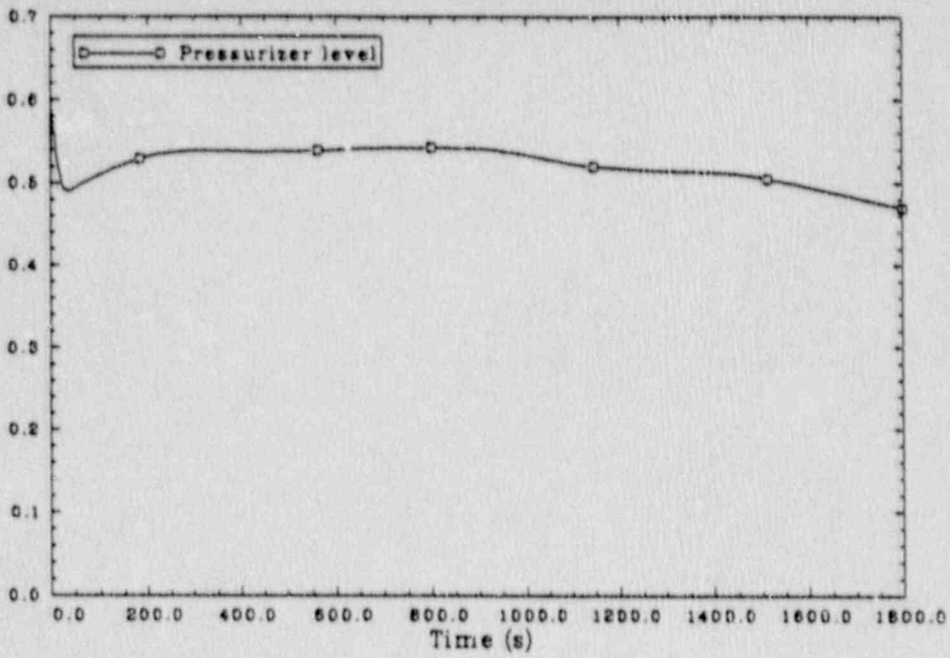

Figure 9. Scenario 1: pressurizer level.

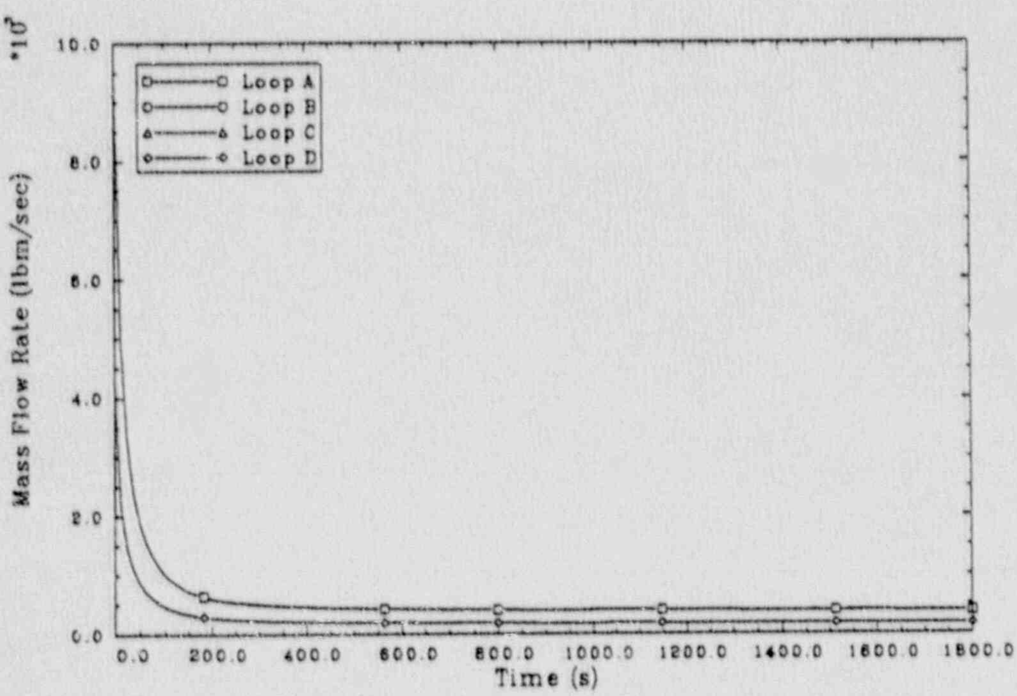

Figure 10. Scenario 1: primary hot leg mass flow rates.

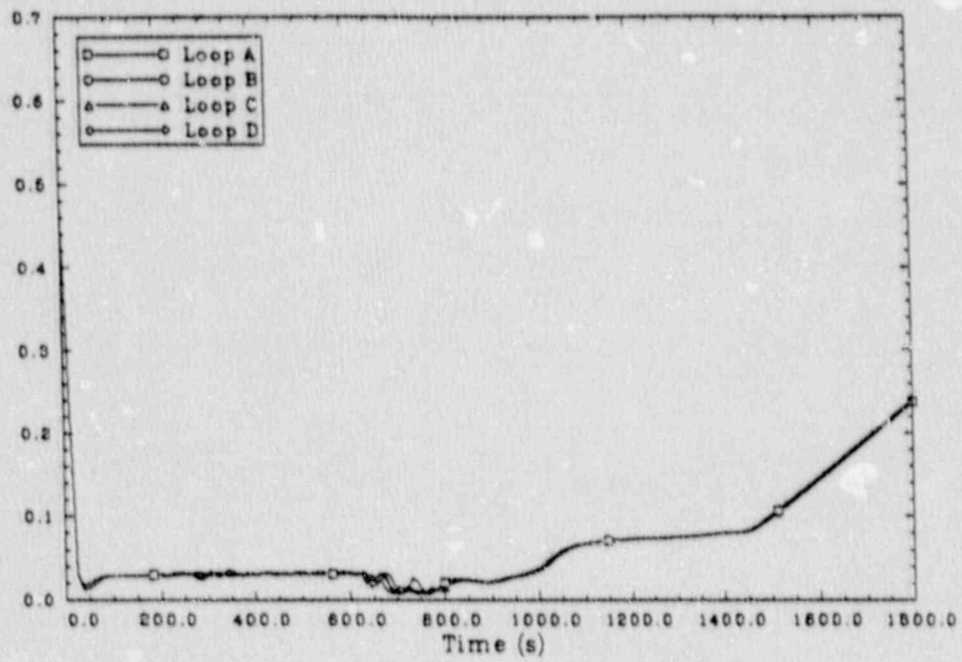

Figure 11. Scenario 1: steam generator narrow range levels.

4. SCENARIO 2: SMALL BREAK LOCA WITH LOSS OF AC POWER

The following section details the analysis of a cold leg small break LOCA coincident with loss of AC power in a SNUPPS plant. The transient was initiated at 100% power. The subsections contain a description of the scenario, model changes and assumptions used in the calculation and analysis of the results.

4.1 Scenario Description

The transient was defined as an initial 1000 gpm small break LOCA accompanied by loss of offsite power with an instantaneous failure of the DGs. This assumed failure made ECC unavailable and eventually led to core uncovering. The transient was initiated by opening a 1000 gpm break in loop D downstream of the RCP, tripping the reactor, tripping the turbine, isolating letdown and charging makeup, deactivating the pressurizer heaters, tripping the reactor coolant, feedwater, condensate, and heater drain pumps. No operator intervention was assumed during the transient.

4.2 Model Changes and Assumptions

The basic SNUPPS model used to perform the calculation is detailed in Section 2. The initial conditions assumed for the transient are documented in Section 2.3.

One nodalization change was made to the SNUPPS RELAP5 model in simulating this scenario. A break junction and time dependent volume were connected to the loop D cold leg volume 452 (see Figure 1) to simulate the 1000 gpm break. The break was modeled with the RELAP5 choked flow option so that the break mass flow varied with the cold leg upstream pressure and temperature conditions. The dimension of the break was sized at approximately one inch in diameter to yield the initial 1000 gpm flow. As in Scenario 1, it was assumed at the initiation of the transient that the feedwater isolation valves would ramp shut at a linear rate in 5 s after loss of AC power. This assumption was made in the absence of any detailed information about the coastdown characteristics of the feedwater train after a loss of AC power scenario. The primary pressurizer and secondary steam generator PORVs as well as the safety valves were assumed to be functional. After the turbine stop valves shut, the steam dump system was assumed to be unavailable because of loss of the condenser vacuum. Thus, the use of the atmospheric dump valves was needed to maintain secondary to primary cooling. Steam generator level control was assumed not to exist after the auxiliary feedwater system was initiated.

It was decided that the transient would be terminated when the core region began to void, and before the upper regions of the steam generators were flooded by auxiliary feedwater.

4.3 Calculated Results

Table 5 is a summary of the sequence of events that were calculated in the RELAP5 Scenario 2 simulation. The transient was initially characterized by a rapid depressurization of the primary system and pressurization of the secondary system to 1150 psia. Figure 12 shows the primary and secondary pressure responses. The initial depressurization and shrinkage of the primary system was the consequence of the simultaneous opening of the break and tripping the reactor. The initial rate of depressurization was reduced as the RCPs began to coast down and primary to secondary heat transfer was reduced. The loop flow coastdown rates were similar to the results in Scenario 1. The loop mass flow rates stabilized to natural circulation conditions at about 300 s. Figure 13 displays the hot leg mass flow rates for loops A-D.

Also the primary hot and cold leg initial temperature responses were similar to those calculated in Scenario 1. Figure 14 presents the average hot and cold leg temperature responses. The hot leg coolant temperatures initially decreased as a result of the reactor scram for about 50 s, then began increasing because of the reduced loop mass flow rates. Peaking of the hot leg temperatures at about 300 s was coincident with the transition to natural convection in the primary loops. The temporary increase in cold leg

Table 5. Scenario 2: sequence of events

Time (s)	Event
0.	Cold leg break opened, reactor tripped, reactor pumps tripped, charging and letdown isolated, pressurizer heaters tripped, feedwater pumps tripped, feedwater isolation valves begin to ramp shut, turbine stop valves begin to close, steam driven auxiliary feedwater signal generated
1.	Turbine stop valves closed.
5.	Steam generator PORVs open, feedwater isolation valves fully closed.
400.	Pressurizer empties, hot legs, and vessel upper plenum begin to void.
700.	Natural circulation mass flow rate increases due to progressive vessel voiding.
1600.	Temporary voiding at the break plane begins.
1800.	End of simulation.

temperatures was caused by the sudden increase in the secondary side pressure and saturation temperatures. This caused the cold leg temperatures to increase as liquid from the steam generator U-tube regions reached the cold legs. This trend lasted about 20 s until cooler liquid from the hot legs was convected around the loops.

By 400 s the pressurizer had emptied (Figure 15). The emptying of the pressurizer resulted in the voiding of other regions of the primary system as additional primary liquid exited out the break. The progressive voiding of the primary system did effect the primary depressurization rate and loop mass flow rates. The calculated vapor void fraction responses in the hot legs, as well as the vessel upper plenum and head, are shown in Figures 16 and 17. Voiding in the hot legs and upper plenum at approximately 500 s and voiding in the vessel upper head at 800 s caused reductions in the primary depressurization rate. Voiding in the vessel region induced a transition from single to two phase natural circulation conditions by approximately 700 s. The flow transition resulted in an increase in the magnitude of the loop flows. This was caused by an increase in the static differential pressure heads between the vessel and steam generator U-tube regions. The enhancement of natural circulation during the transition from single to two-phase natural circulation conditions has been experimentally measured in PWR subscale facilities.[4,5]

Primary system voiding also induced reductions in the break mass flow rate. The calculated inflections in the break flow response occurred at approximately 500 and 700 s (Figure 18) and coincided with the reductions in the primary system depressurization rate. Throughout the entire transient calculation, the loop natural circulation mass flow rates were significantly larger than the break flow rate (Figures 13 and 18). While the break acted as a perturbation to loop D mass flow and temperature responses, the primary and secondary responses differed little between loop D and loops A-C. In general, the break flow was at single phase liquid conditions until about 1600 s, when the break transitioned temporarily to two phase flow.

The steam generator level responses were nearly identical for all four loops. Figure 19 displays the narrow range level responses for loops A-D. Increases in the narrow range levels commenced after approximately 1000 s, which is coincident with the steam generator PORVs closure. The response in the broken loop indicated a slightly slower refill relative to steam generators A-C. This was due to the slightly larger mass flow rate in loop D and slightly higher secondary steam generation rate in that steam generator.

In the final stages of the simulation, the primary coolant hot and cold leg temperatures were becoming equal with the hot leg temperature at saturation conditions. There was a small differential temperature across the vessel which maintained loop natural circulation. The nearly isothermal conditions on the primary side were due to constant secondary heat sink conditions and low natural circulation loop flow rates. As the primary hot and cold leg temperatures began to converge, the primary and secondary systems also began to approach thermal equilibrium; the primary and secondary pressures converged at approximately 1550 s. Moreover, significant core voiding had occurred by this time and core liquid depletion had commenced. However, no vapor super heating or core temperature excursions had been calculated up to the time the transient was terminated.

A continuation of the transient beyond 1800 s would eventually result in the termination of primary loop flow natural circulation (caused by continued primary mass loss out the break), core boil off, fuel rod dry out, and subsequent cladding temperature excursions. However, simulating this stage of the transient should be deferred until more information is obtained about steam generator response during separator flooding. In conclusion, the simulation of scenario 2 was characterized by symmetrical primary and secondary side loop thermal-hydraulic responses. In addition, towards the end of the simulation the primary and secondary systems were approaching thermal equilibrium conditions.

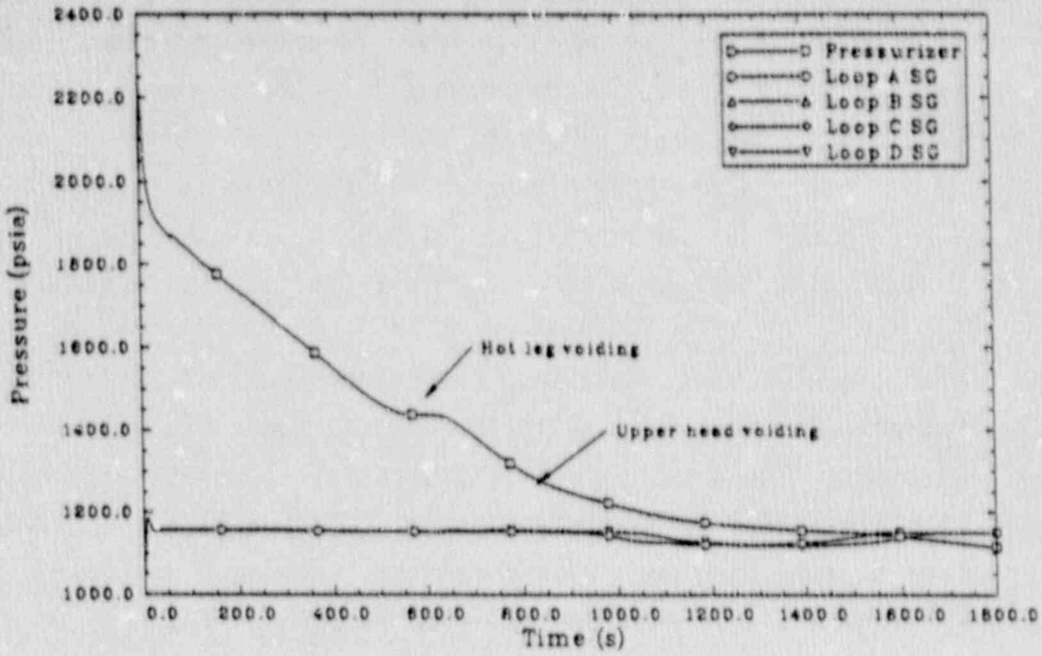

Figure 12. Scenario 2: primary and secondary pressures.

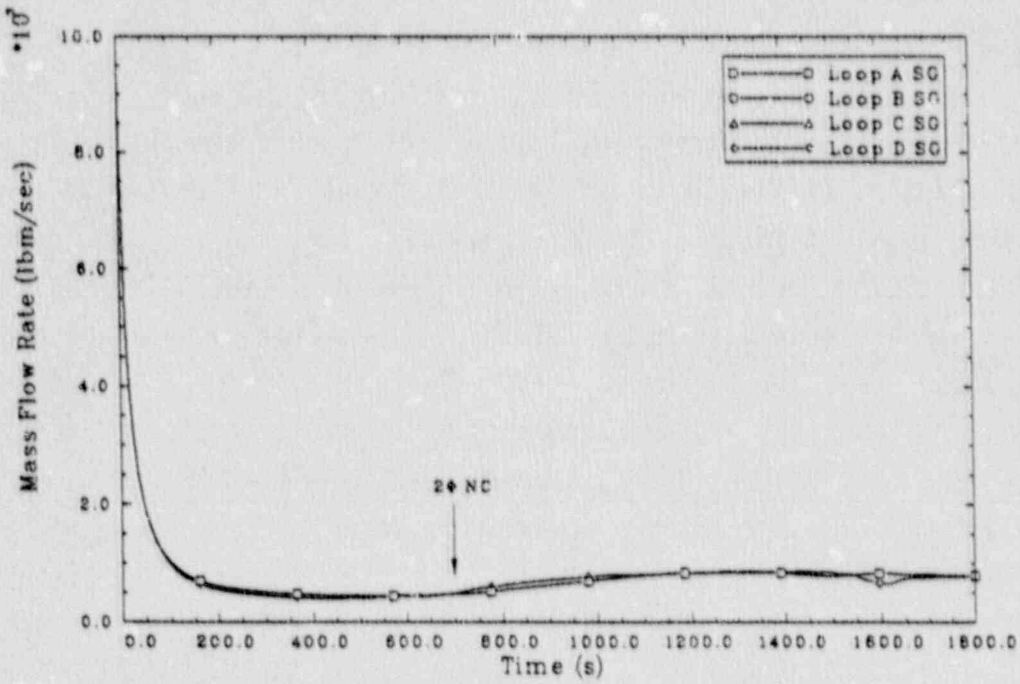

Figure 13. Scenario 2: primary hot leg mass flow rates.

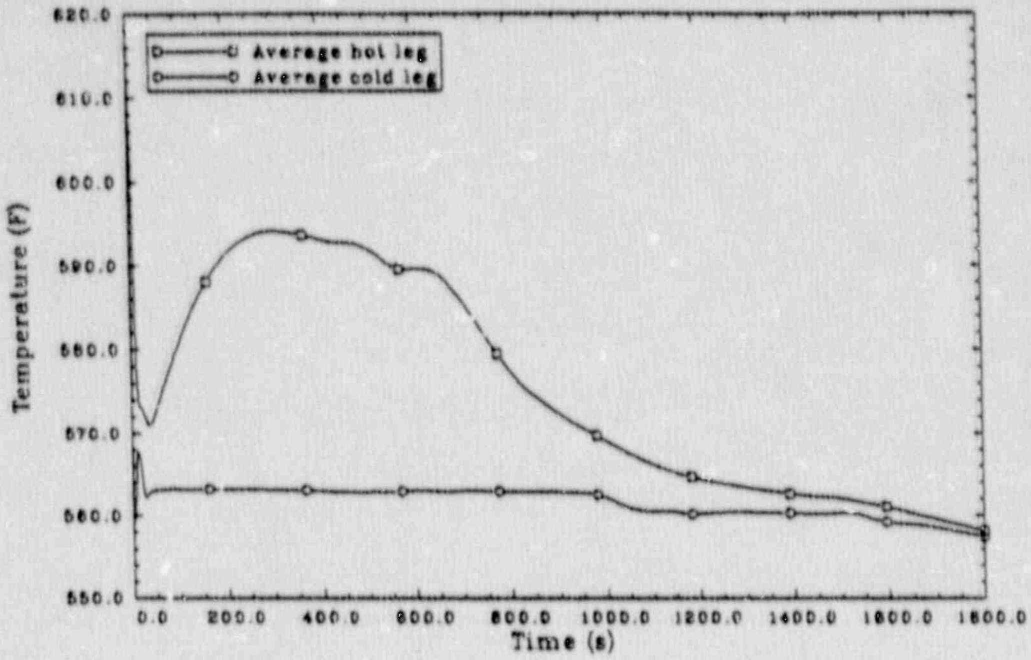

Figure 14. Scenario 2: average primary hot and cold leg temperatures.

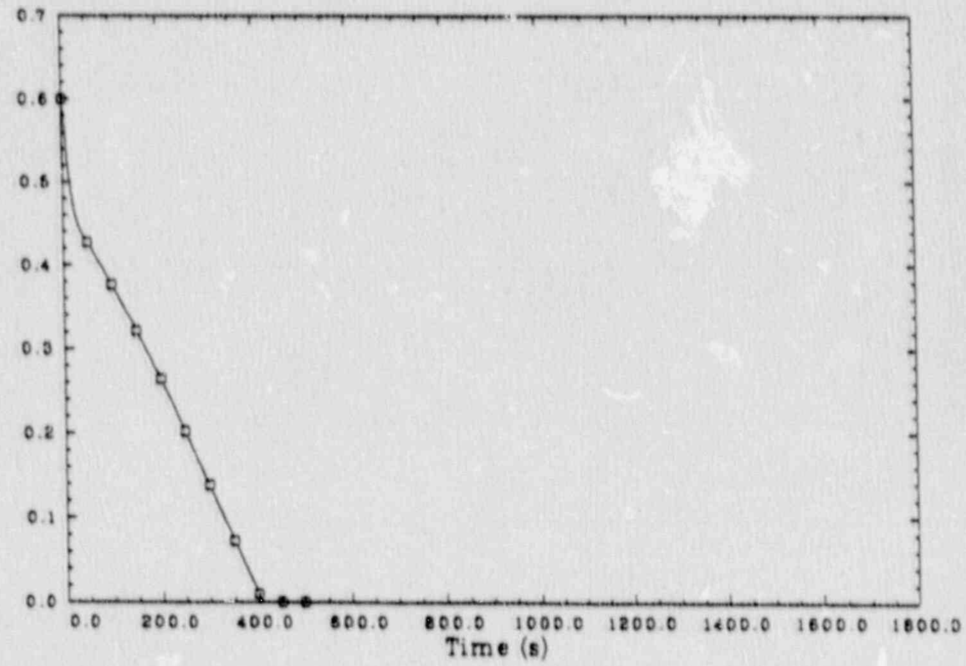

Figure 15. Scenario 2: pressurizer level.

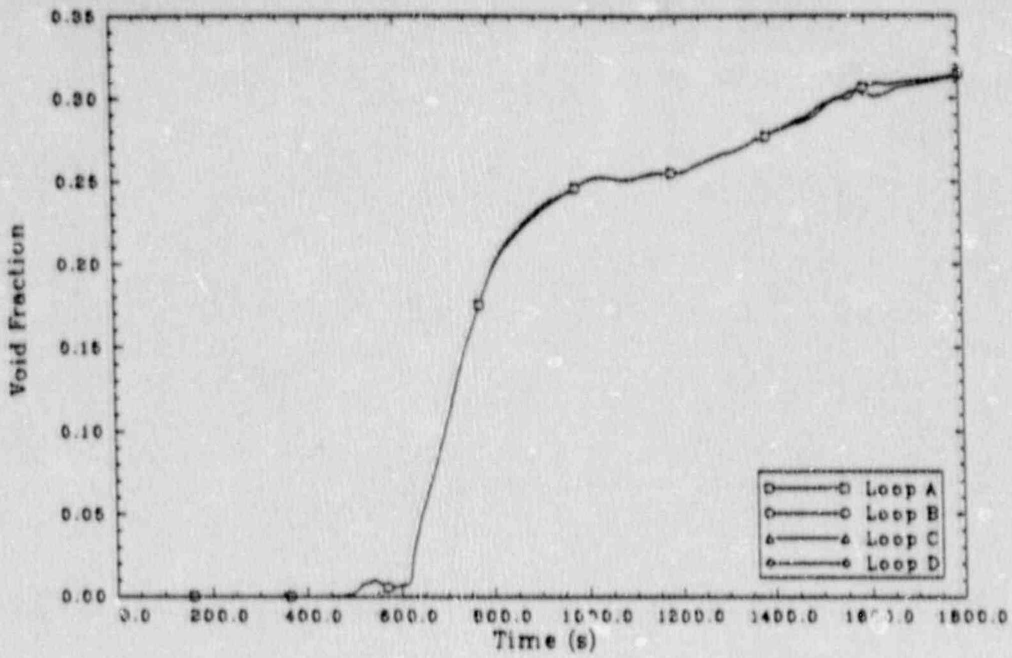

Figure 16. Scenario 2: primary hot leg void fractions.

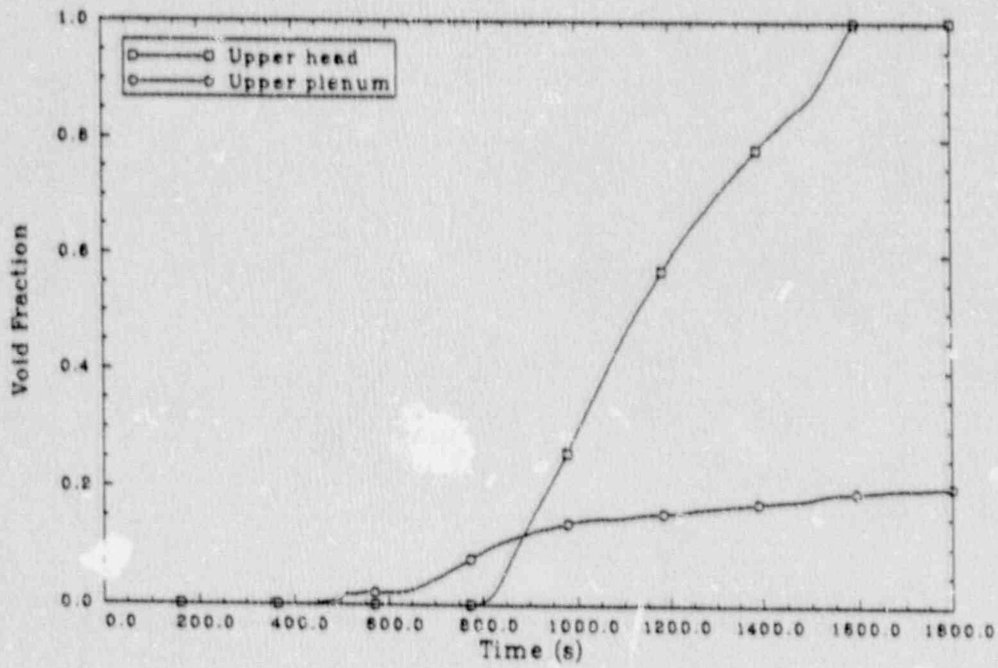

Figure 17. Scenario 2: upper plenum and upper head void fractions.

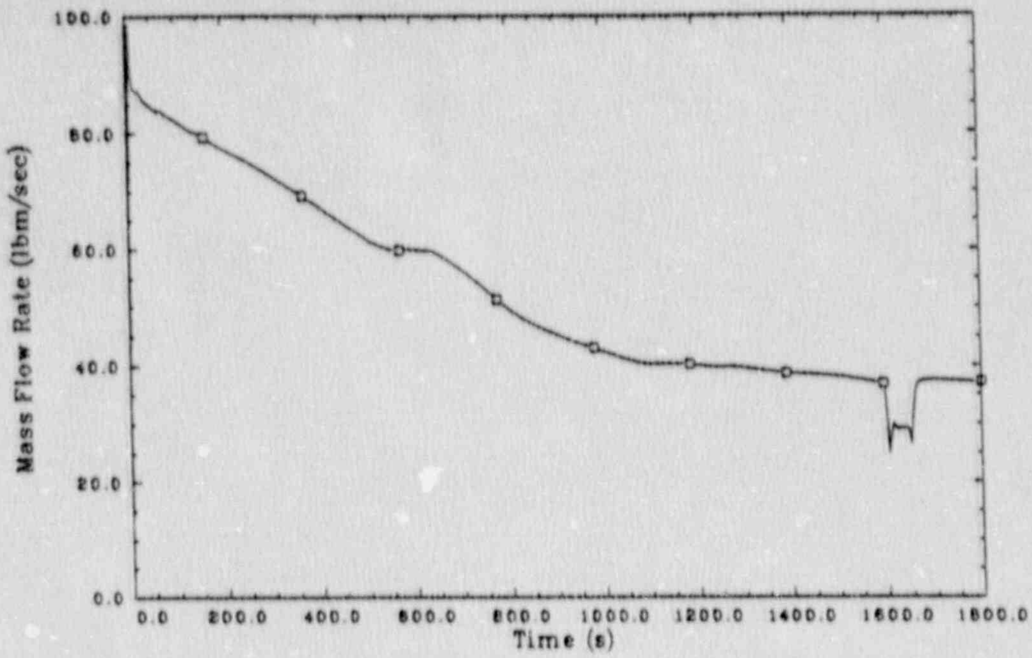

Figure 18. Scenario 2: loop D cold leg break mass flow rate.

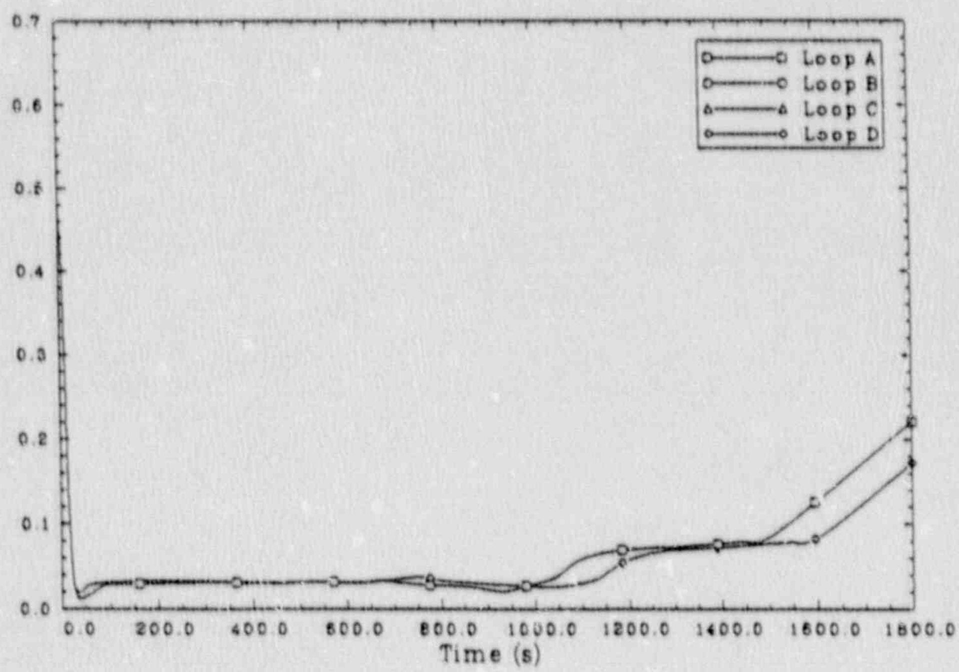

Figure 19. Scenario 2: steam generator narrow range levels.

5. SCENARIO 3: FAILED OPEN PRESSURIZER SAFETY VALVE

The following section details the analysis of a stuck open pressurizer safety valve simulation. The transient was initiated at 100% power. The subsections contain a description of the scenario, model changes and assumptions used in the calculation, and analysis of the results.

5.1 Scenario Description

This transient scenario was defined as a failed open pressurizer safety valve. The transient was initiated at full power by opening one pressurizer safety valve over a period of 100 s and remained locked open thereafter. The assumptions about the mode of valve failure are artificial and are designed to test the capabilities of the simulator rather than model a probable failure. No operator intervention was modeled during the transient.

5.2 Model Changes and Assumptions

The basic SNUPPS model used to perform the calculation is detailed in Section 2. The initial conditions assumed for the transient are documented in Section 2.3.

Since there was no operator intervention, the RCPs were assumed not to be manually tripped off. All other control systems were assumed to operate in their automatic modes.

The pressurizer model was modified in this transient since it was observed that the calculated insurge of subcooled liquid into the pressurizer region caused computational problems in calculating the pressure response. In order to deal with this problem, the RELAP5 volume equilibrium option was used in pressurizer control volumes 619 and 620 (see Figure 1). The use of the equilibrium option forced the liquid and vapor temperatures to be identical; its use is recommended when sudden shifts in phasic temperatures cause anomalous pressure spikes, as in this calculation. The final outcomes of the simulations with and without the equilibrium options were not significantly different.

5.3 Calculated Results

The sequence of events for this transient are presented in Table 6. The transient was initiated at the instant that one pressurizer safety began to fail open. The primary pressure began to immediately drop (Figure 20) as steam exited out the top of the pressurizer. Vapor discharging through the safety valve was displaced by liquid flowing from the surge line into the pressurizer resulting in an initial increase in the pressurizer normalized level (Figure 21). Other parameters in the system remained nearly constant until 60 s when a scram signal was generated. The reactor trip signal resulted from a low pressurizer pressure signal of 1915 psia. Coincident with the reactor trip was closure of the turbine isolation valves, and the opening of the steam dump valves.

Following the reactor trip, the primary pressure rapidly dropped, which resulted in the generation of a low-low 1864 psia pressurizer SI signal at 62 s. The SI signal resulted in tripping of the main feedwater pumps, actuation of motor auxiliary feedwater pumps, and the closure of feedwater isolation valves. By 67 s the feedwater isolation valves had closed and the turbine auxiliary feedwater trip signal was reached on a 2/4 steam generator low-low level reading. The closure of the turbine stop valves resulted in a pressure increase in secondary pressure from approximately 1020 psia to 1180 psia. This caused the steam generator PORVs to lift between 65 and 80 s.

At 60 s the rapid reduction in core power caused the primary system average temperature to drop which induced a subsequent shrinkage of the primary loop inventory. The primary liquid volumetric shrinkage corresponds to the drop in the pressurizer level (Figure 21). The loop A primary loop hot, cold leg, and loop average temperatures are presented in Figure 22. Temperature responses were the same in the other loops. Following the reactor trip the decrease in the loop average temperatures caused a temporary

Table 6. Scenario 3: sequence of events

Time (s)	Event
0.	Pressurizer safety valve begins to fail open
3.	Pressurizer proportional and backup heaters on
60.	Reactor tripped off on low pressure signal, turbine tripped, steam dump system activated
62.	SI signal generated on low pressurizer pressure signal and ECC activated, feedwater pumps tripped, motor auxiliary feedwater activation signal generated
67.	Feedwater isolation valves fully closed.
68.	Steam Turbine auxiliary feedwater signal generated on 2/4 low SG nr low-low level trip.
100.	Pressurizer safety valve fails full open.
225.	Pressurizer normalized level reaches 100% of full span.
300.	End of simulation.

pressurizer out surge that lasted until approximately 80 s. After the scram the cold leg temperatures increased slightly. This increase was the consequence of the secondary side pressure and saturation temperature increase after the turbine was tripped. By 70 s the total ECC mass flow rate had exceeded the break mass flow rate and by 90 s the ECC volumetric flow rate had exceeded the break volumetric rate. Figure 23 shows a comparison of the break mass flow rate out of the safety valve and the total ECC into the primary system.

At 100 s the safety valve had failed full open and the break flow began to decrease (Figure 23) with continued primary depressurization. At 220 s the pressurizer level indicated 100% of full span and the safety valve break transitioned from single phase steam to two-phase flow conditions. This transition caused an abrupt increase in the stuck open valve mass flow rate as liquid began to exit through the break plane. In the simulation steam voids were still present at the top of the pressurizer steam dome even though the normalized level indicates 100% of full span.

In general, the loop temperature and flow responses were almost identical. Small differences did exist between loop D and loops A-C because of the pressurizer connection to loop D. The break flow out of loop D via the pressurizer only perturbed the temperatures and mass flow rates in that loop since the loop mass flow rates were significantly larger than the break flow. Figure 24 presents the calculated cold leg loop mass flow rates for loops A-D. Shown in Figure 25 are the vapor void fractions responses in the RCP volumes for loops A-D. By 180 s the RCPs had cavitated about the time the cold leg fluid temperatures reached saturation conditions. RCP voiding caused a net reduction in loop flow because of the modeled two-phase flow degradation curves used in the RELAP5 model. Also, the pump cavitation happened at about the time the primary coolant average loop temperatures reached the 557 °F no load set point. Reaching this set point also resulted in the closure of the steam dump valves.

Figures 26 and 27 show the calculated steam generator narrow and wide range level responses for loops A-D, respectively. The initial drop in levels was due to the mismatch between the vapor exiting out the top of the steam generators and the cessation of feedwater flow to the boiler regions. By 90 s auxiliary feedwater had entered the steam generator downcomer and boiler regions and the wide range levels began a slow recovery. Recovery in the narrow range levels did not begin until about 200 s.

By 300 s it was judged that the most important events of the simulation had been observed and the transient was terminated. From the simulation results it can be inferred in the absence of operator intervention, that the primary system would slowly cool down as injected ECC liquid replaced the saturated primary coolant going out the safety valve. Subcooled auxiliary feedwater would eventually cool down and flood the steam generators. Without operator intervention the auxiliary feedwater would eventually fill each of the steam generator dome cavities and enter the steam discharge lines.

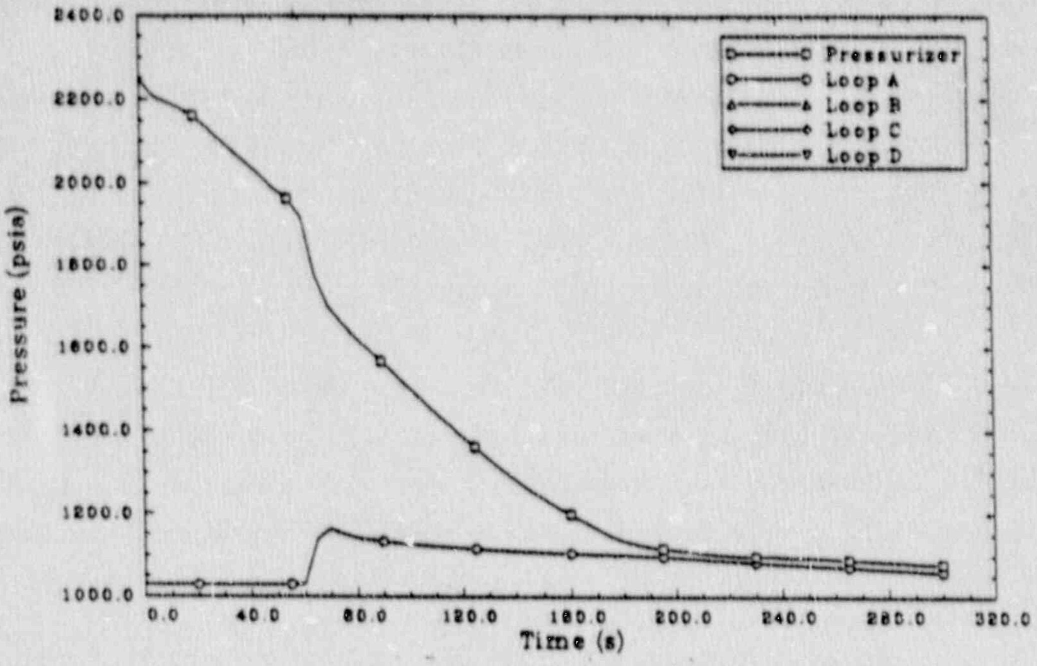

Figure 20. Scenario 3: primary and secondary pressures.

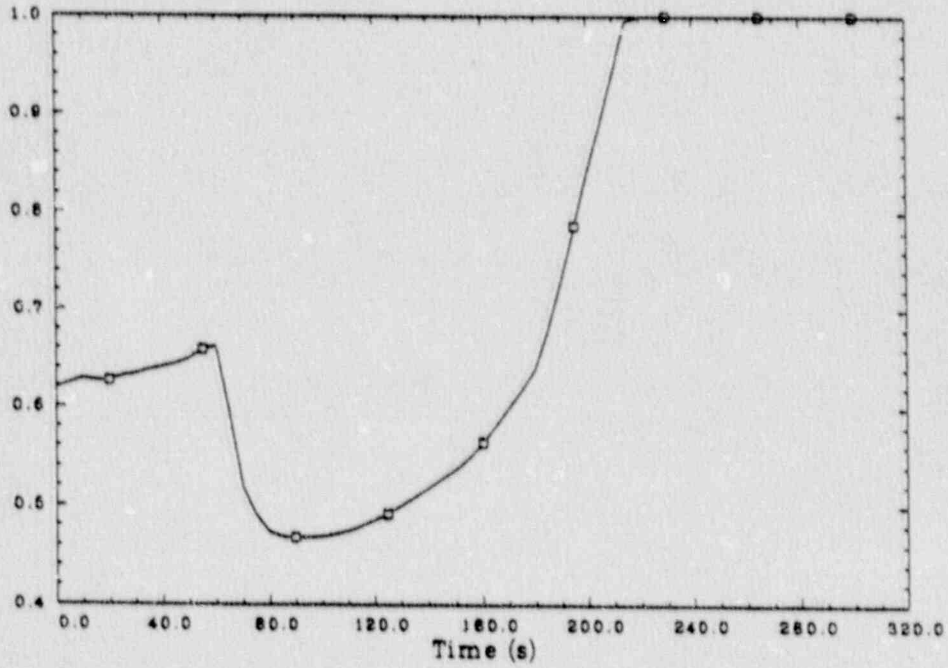

Figure 21. Scenario 3: pressurizer level.

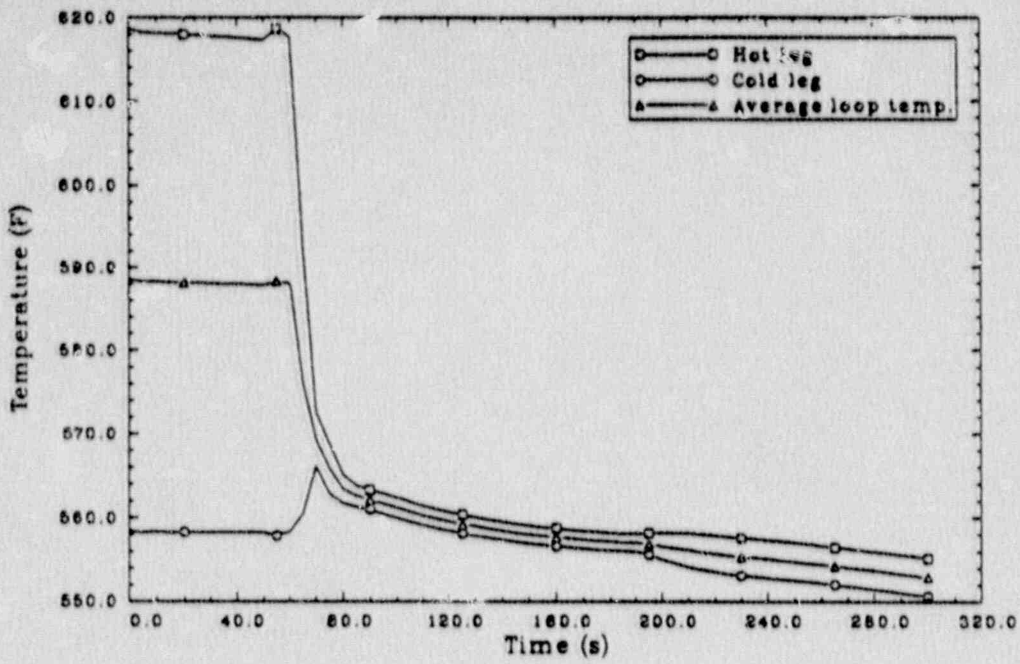

Figure 22. Scenario 3: primary hot, cold and loop average temperatures.

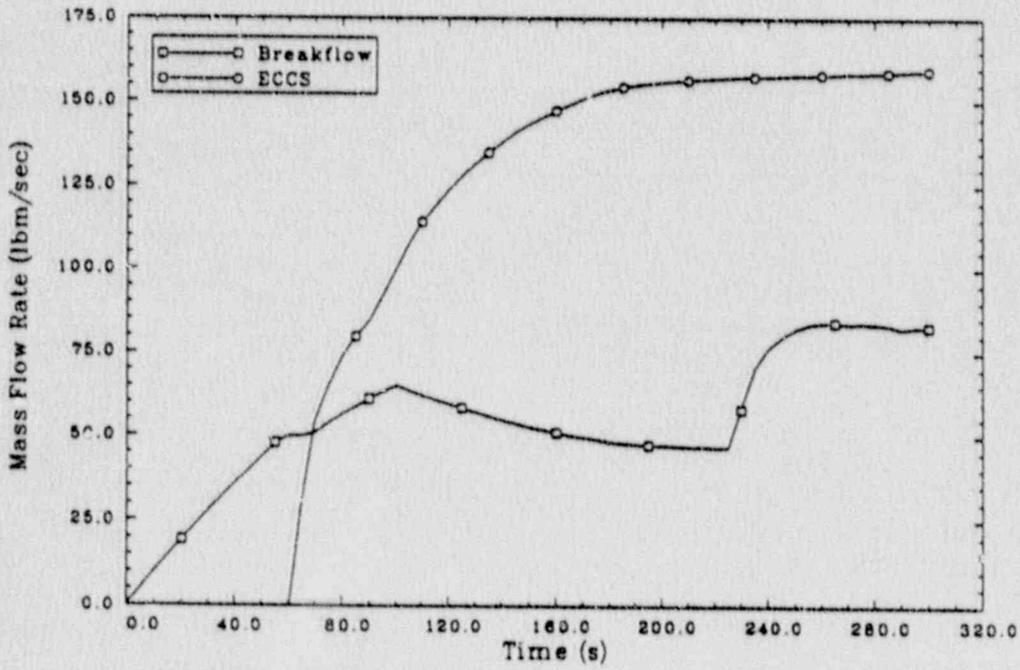

Figure 23. Scenario 3: break and total ECCS mass flow rates.

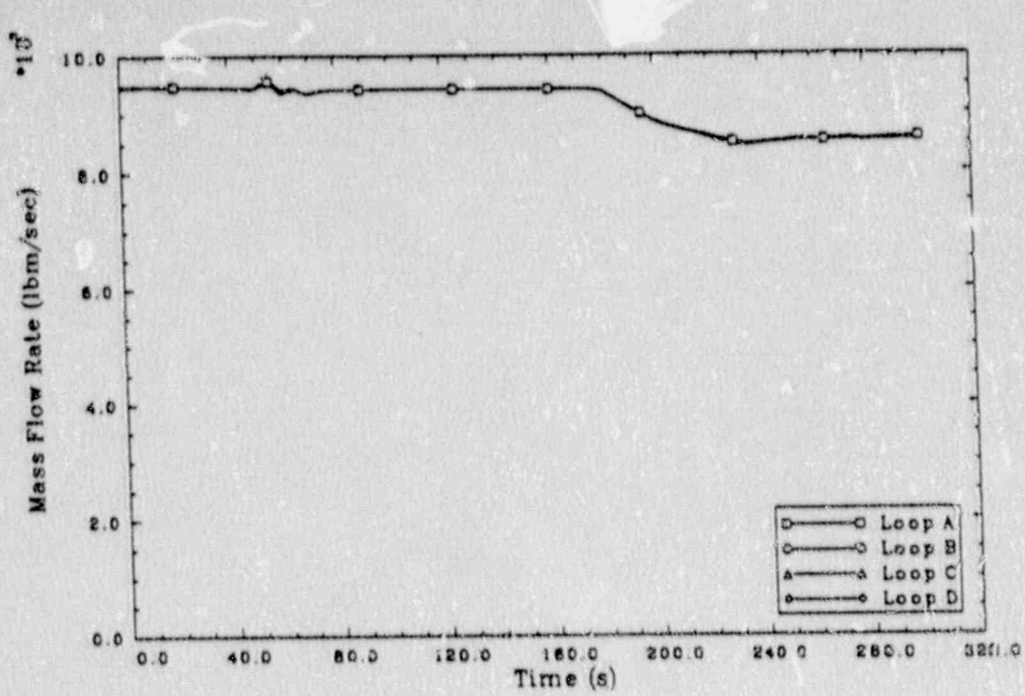

Figure 24. Scenario 3: primary cold leg mass flow rates.

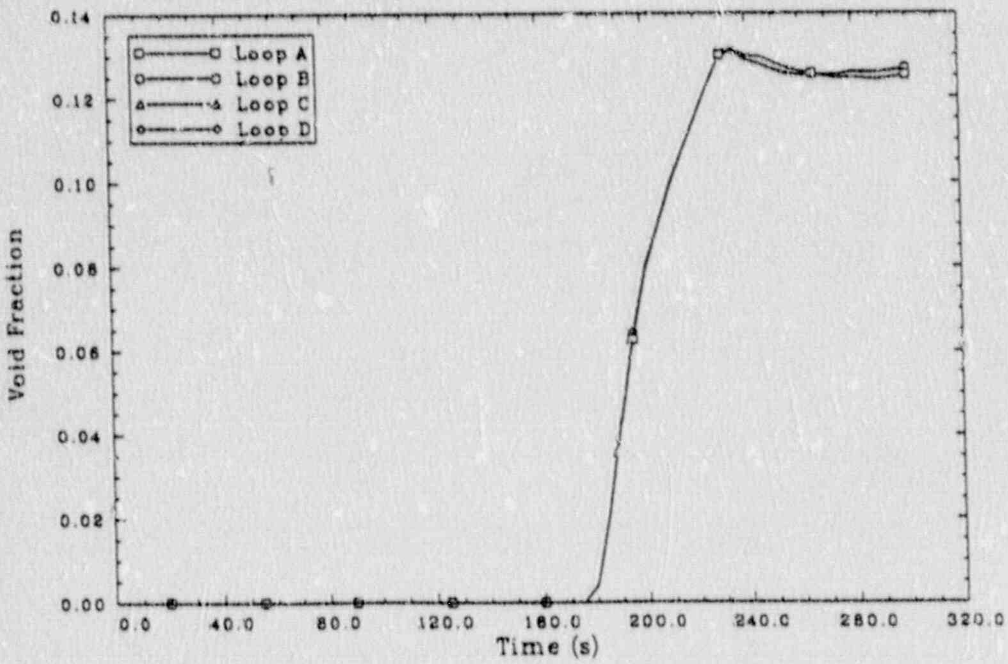

Figure 25. Scenario 3: loop reactor coolant pump vapor void fractions.

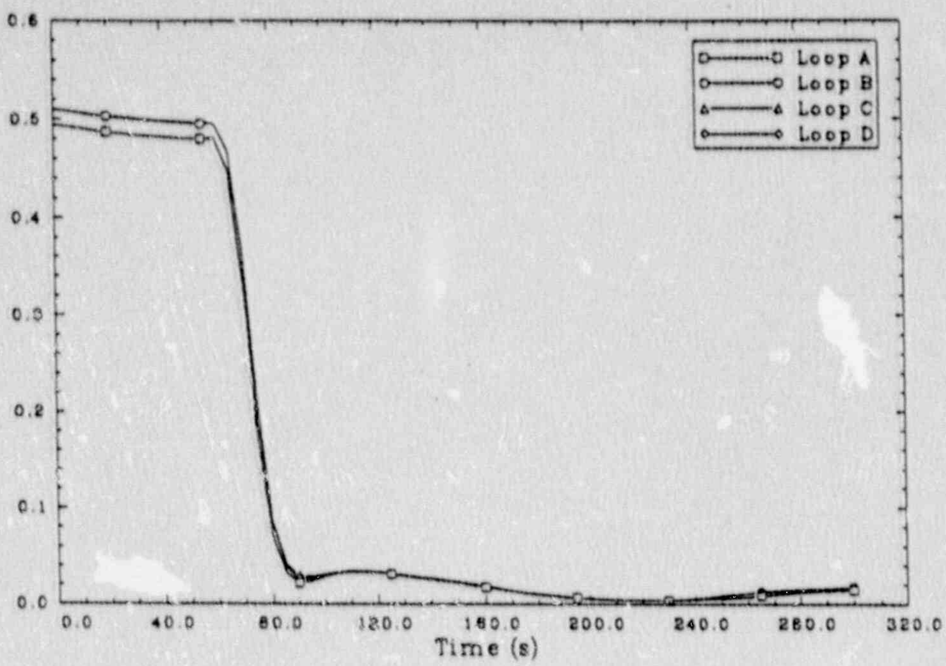

Figure 26. Scenario 3: steam generator narrow range levels.

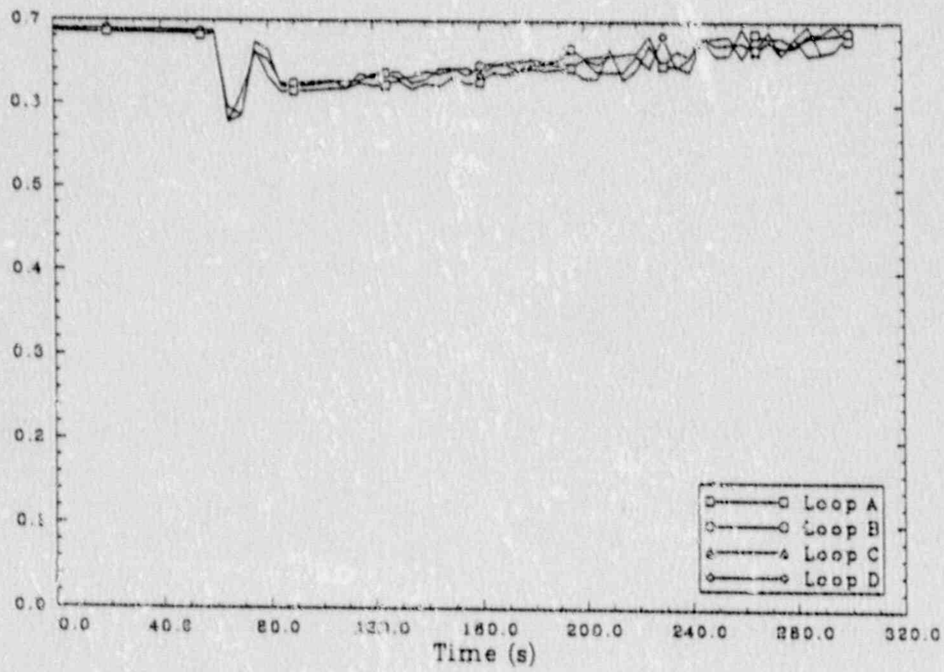

Figure 27. Scenario 3: steam generator wide range levels.

6. SCENARIO 4: MAIN STEAM LINE BREAK WITH STEAM GENERATOR TUBE RUPTURE

The following section details the analysis of a double ended guillotine rupture of a SNUPPS main steam line, upstream of the MSIVs, with the concurrent rupture of a single generator tube. The subsections contain a description of the scenario, model changes and assumptions used in the calculation, and analysis of the results.

6.1 Scenario Description

The transient was defined to be an instantaneous non-isolatable, double ended guillotine rupture of a main steam line break (MSLB) upstream of the MSIV with the reactor at full power. A concurrent steam generator tube rupture (SGTR) in loop A was also simulated. The tube rupture was assumed to be located at the top of the tube sheet on the inlet side. No operator actions were assumed during the course of the event.

6.2 Model Changes and Assumptions

The basic SNUPPS model was used to perform the calculation; this model is described in Section 2. The initial full power conditions assumed for the transient are described in Subsection 2.3.

The break in Steam Line A was modeled with the insertion of two RELAP5 valve components attached to volumes 161 and 171. These valve components were tripped open at the initiation of the transient. In addition, the RELAP5 junction component 170 was closed at the same time the double ended rupture was initiated; thus, isolating the steam header from the affected steam generator (ASG). These changes are shown in Figure 28.

Additional modeling modifications were implemented to simulate the tube rupture for the ASG. The original steam generator model simulated the primary inlet and outlet plena as part of a single pipe volume. For the SGTR event, these plena were separated from the U-tubes (volumes 111 and 112). A break valve was inserted to provide a path from the steam generator primary inlet plenum (volume 111) to the boiler section of the ASG. An additional U-tube channel was modeled with volume 109; this volume was identical to the nominal U-tube bundle (volume 107) except the flow area and volume were that of a single tube. One end of the new volume was connected with the ASG primary outlet plenum (volume 112); the other end was connected to the ASG boiler section with a valve. Both valves were assumed to open at transient initiation. These changes are shown in the nodalization diagram, Figure 29.

All control systems were assumed to operate in their automatic modes. No additional modifications were made to the steam generator separator model. For this particular scenario, uncertainties in the modeling of the separator region could potentially effect results in comparison to the simulator response.

6.3 Calculated Results

The sequence of events that occurred during the MSLB with SGTR transient is shown in Table 7. The transient was initiated by opening the two steam line A break valves and the two valves representing the SGTR event. At 0.08 s, an SI signal due to a low pressure signal in steam line A was generated. This signal initiated the reactor scram, turbine trip, main feedwater isolation, and motor auxiliary feedwater. By 5 s, feedwater to all four steam generators was terminated with the closure of the feedwater isolation valves. Also at 5 s the steam generator PORVs on the unaffected steam generators (USGs) lifted to decrease the pressure in these units; the PORVs closed at 10 s. At 8 s, narrow range levels on the USGs decreased to the low level alarms, sending an initiation signal to the steam turbine auxiliary feedwater. At 30 s, motor auxiliary feedwater began to feed into all four steam generators; at 38 s, the turbine auxiliary feedwater reached the steam generators. The pressurizer level dropped to 0% by 47 s. By 70 s the wide range level for ASG had reached 0%. and the blowdown process had ended. Coincident with the end of the blowdown in the ASG was an end to the primary system cooldown and depressurization. After this period the primary system began to stabilize

Table 7. Scenario 4: sequence of events

Time (s)	Event
0.08	Main steam line loop A ruptures, reactor scram and SI signal generated on low steam line pressure signal, motor auxiliary feedwater signal is generated, feedwater pumps trip, feedwater isolation and main steam isolation valves begin to ramp shut, turbine trips
8.	Turbine auxiliary feed signal generated on 2/4 low steam generator narrow range level signal
30.	Motor auxiliary feedwater begins feeding steam generators
38.	Turbine auxiliary feedwater begins feeding steam generators
47.	Pressurizer liquid level reaches 0%
70.	Loop A steam generator wide range level reaches 0%
105.	Pressurizer level begins to increase above 0%
120.	End of simulation

with the hot and cold leg temperatures in the unaffected loops converging at about 105 s.

The RELAP5 simulation was terminated at 120 s with the ASG nearly empty. The liquid inventory in ASG had stabilized with break flow (mass loss out the steam generator) being balanced with the auxiliary feedwater and tube rupture mass flows (mass flows into the steam generator). The primary system pressure was slowly increasing as a consequence of ECC injection. The decision to terminate the calculation was based on the judgement that the most severe phase of this transient had occurred.

Figure 30 shows the secondary pressure responses. The blowdown of the ASG was characterized by a period when the boiler region had not dried out and primary to secondary heat transfer was significant, and a later period when the steam generator region had dried out and primary to secondary heat transfer was degraded. Up to 70 s the blow down of ASG resulted in a corresponding depressurization of the primary system (Figure 31). At 70 s the depressurization rate increased. This increase was due to the emptying of the ASG and degradation of the heat transfer from the primary system. The reduction in primary to secondary heat transfer also stopped the depressurization of the primary system. By 90 s the mass flow into and out of the ASG had reached an approximate balance and the pressure had stabilized to about 50 psia.

The USGs showed an initial pressure increase due to the closing of the MSIVs, and a decrease due to the opening of the PORVs. After the PORVs closed, the USG pressures continued to decrease, following the primary cooldown.

The steam generator narrow and wide range levels are shown in Figures 32 and 33. The USG narrow range levels showed the effects of feedwater isolation as well as the steam release through the PORVs. In the ASG there was a temporary level increase before the narrow range level dropped. This increase was due to the separator model as well as the dynamic pressure head across the narrow range level taps. The level decrease in the USG generated a low level signal that activated the steam turbine auxiliary feedwater pump. The wide range levels for the USGs show the initial decrease in level followed by the stabilization after auxiliary feedwater is introduced. The ASG wide range level showed some spikes due to the separator modeling in the steam generator, but decreased to zero by 70 s.

Figures 34 and 35 present the steam line and tube rupture break mass flow rates. Several spikes were observed in the steam line break flow during periods of filling of the steam generator separator volume. During these periods, liquid was convected from the separator to the break plane in 'slugs', which caused sudden oscillations in the break flow rate. Eventually the break flow unchoked after the ASG boiler region dried out. The dryout of the ASG boiler region caused its pressure to decrease to 50 psia and the break to unchoke. The SGTR flows, shown in Figure 35, present the difference in flows from the tube up-flow (junction 797) and downflow sides (junction 796). The hydraulic resistance in the downflow path is significantly larger than the upflow path. As a consequence the up flow mass flow rate is significantly larger than the downflow side.

During the initial 70 s blowdown phase of the ASG, the primary coolant underwent significant shrinkage due to the thermal contraction as well as to mass loss out the SGTR break. The combined effect caused the pressurizer level to drop to 0% at 47 s (Figure 36). Although the pressurizer emptied, no significant voiding occurred elsewhere in the primary system. As seen in the Figure 36, the pressurizer level indication was starting to recover at about 105 s. This refill was the consequence of ECC volumetric flow rate exceeding the SGTR volumetric flow rate. The calculated ECC loop mass flow rates are presented in Figure 37.

Finally, the effects of the MSLB on primary coolant temperatures are shown in Figures 38 to 41. In the affected loop, both hot and cold leg temperatures fell due to the drop in power from the reactor and the uncontrolled cooldown through the ASG. At 70 s when the ASG emptied, the cold leg temperature began to recover as less heat was removed. In the unaffected loops, the USGs began acting as heat sources after they were isolated. This produced cold leg temperatures higher than hot leg temperatures. Once the cooldown ended (when the ASG emptied), the unaffected loop temperatures became virtually equal.

In conclusion, the MSLB with SGTR transient was characterized as an uncontrolled cooldown of the primary system, continuing until the broken steam generator was empty. After the ASG emptied, the primary system pressure began to slowly increase as the transient changed to an ECC-break flow 'feed and bleed' mode. In this mode the primary system began to slowly repressurize since the ECC mass flow rate was slightly larger than the SGTR flow rate.

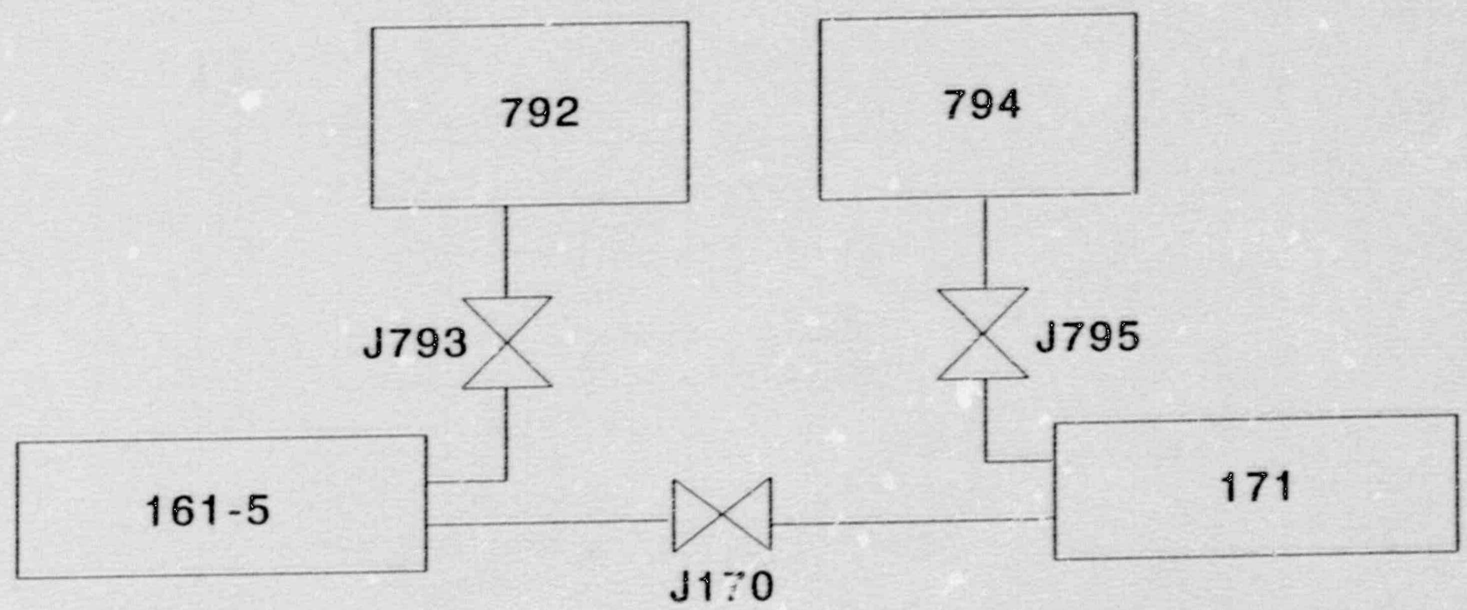

Figure 28. Scenario 4: nodalization changes for loop A steam line break.

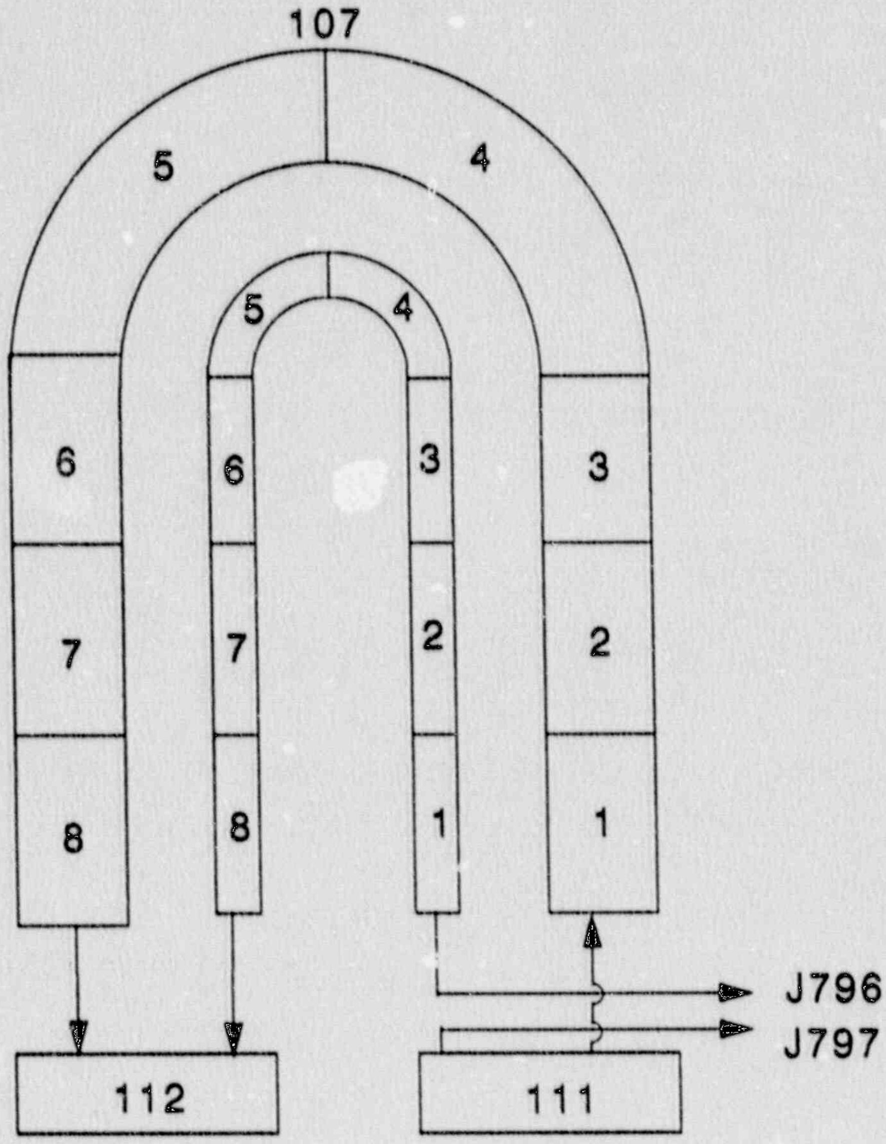

Figure 29. Scenario 4: nodalization changes for loop A ste:m generator t:be rupture.

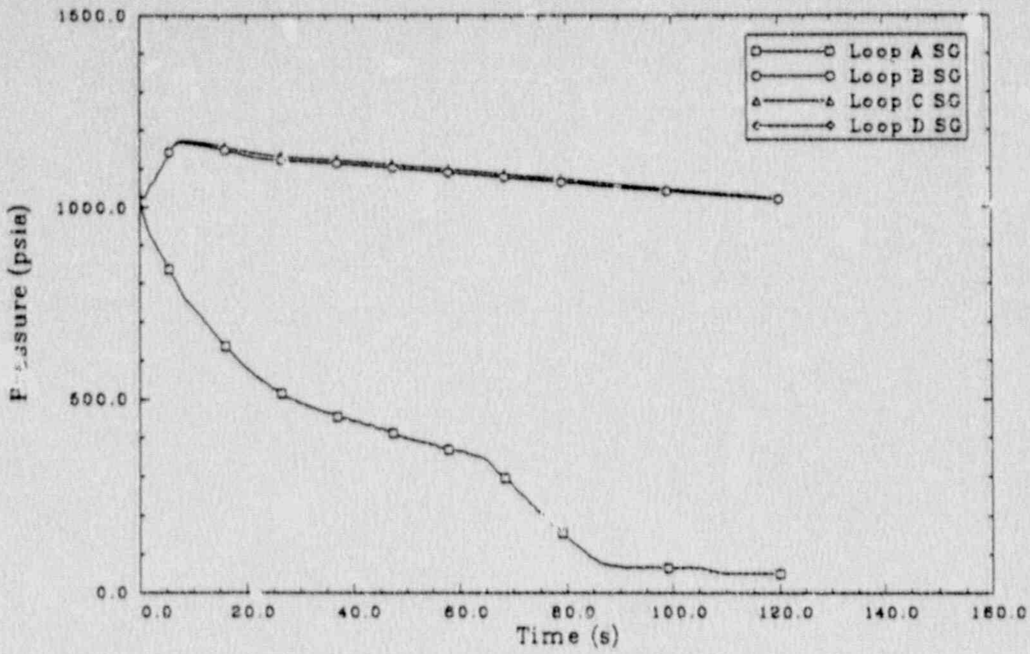

Figure 30. Scenario 4: steam generator pressures.

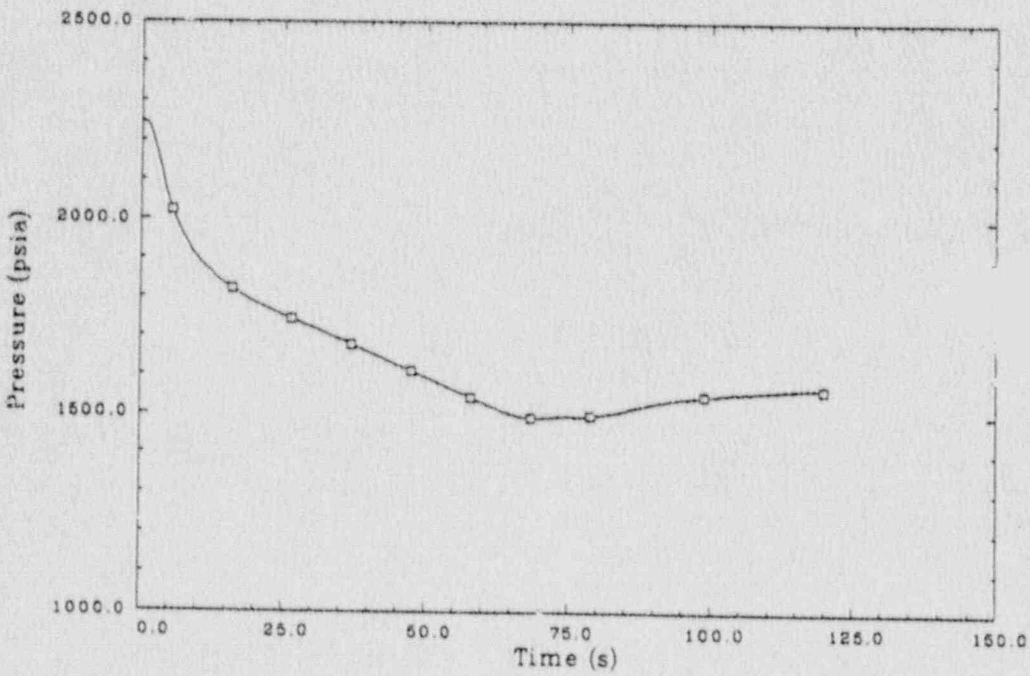

Figure 31. Scenario 4: pressurizer pressure.

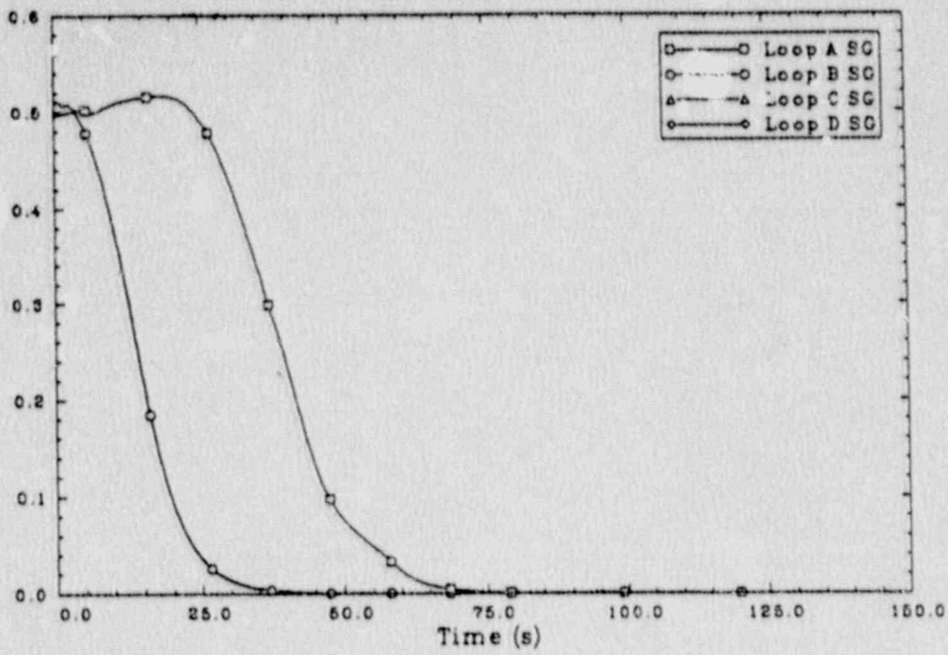

Figure 32. Scenario 4: steam generator narrow range levels.

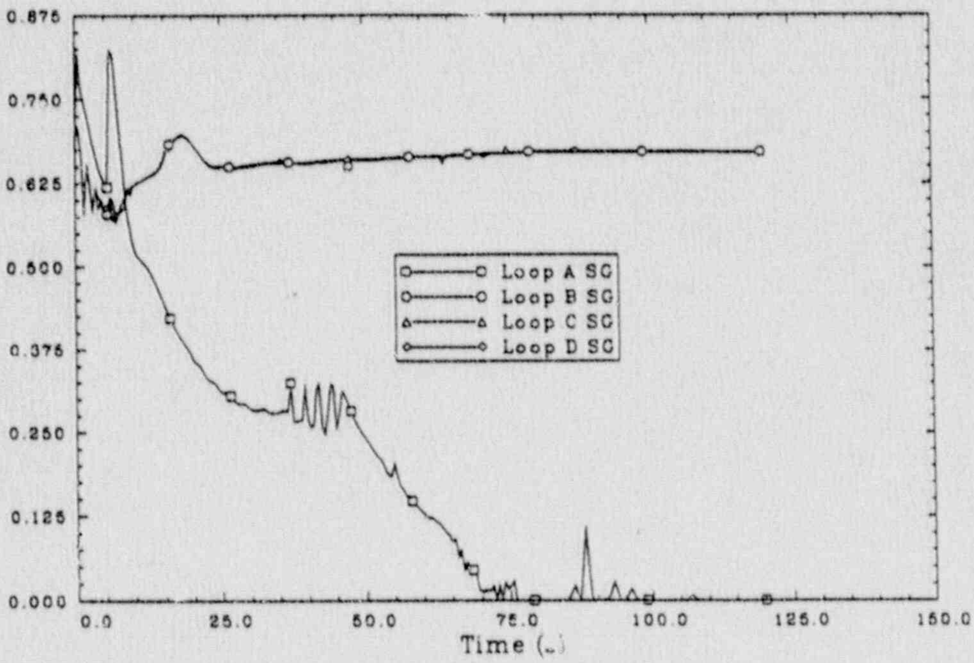

Figure 33. Scenario 4: steam generator wide range levels.

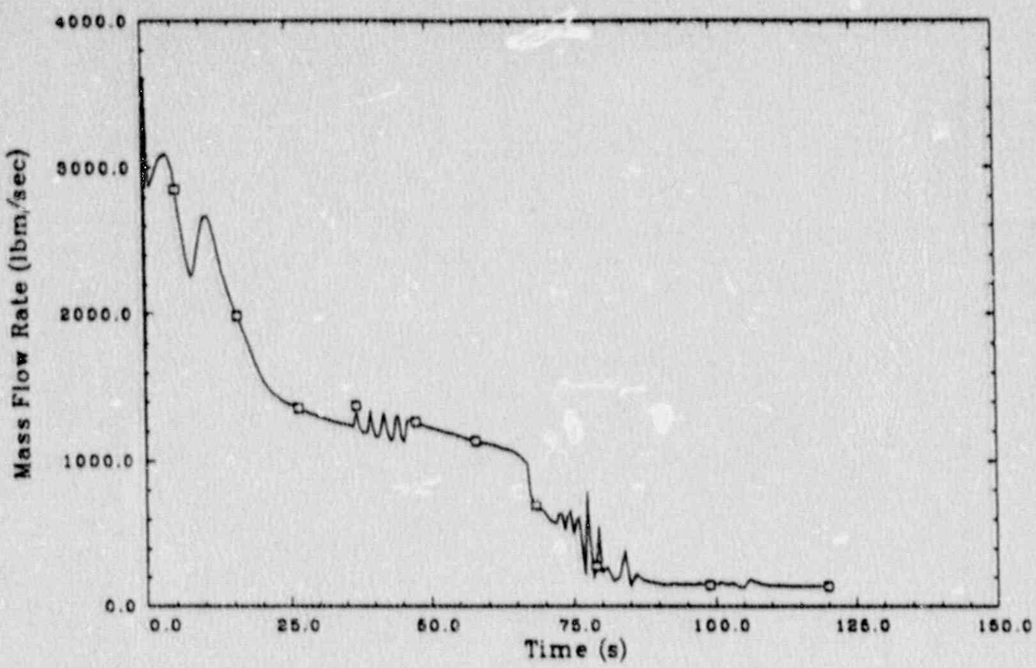

Figure 34. Scenario 4: steam line break mass flow rate.

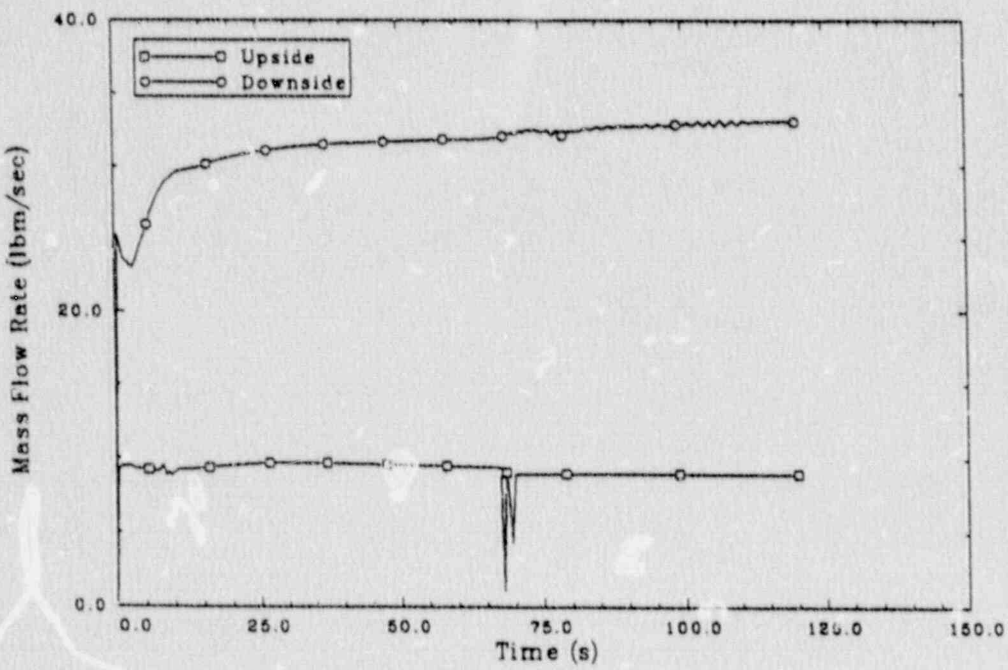

Figure 35. Scenario 4: steam generator tube rupture mass flow rates.

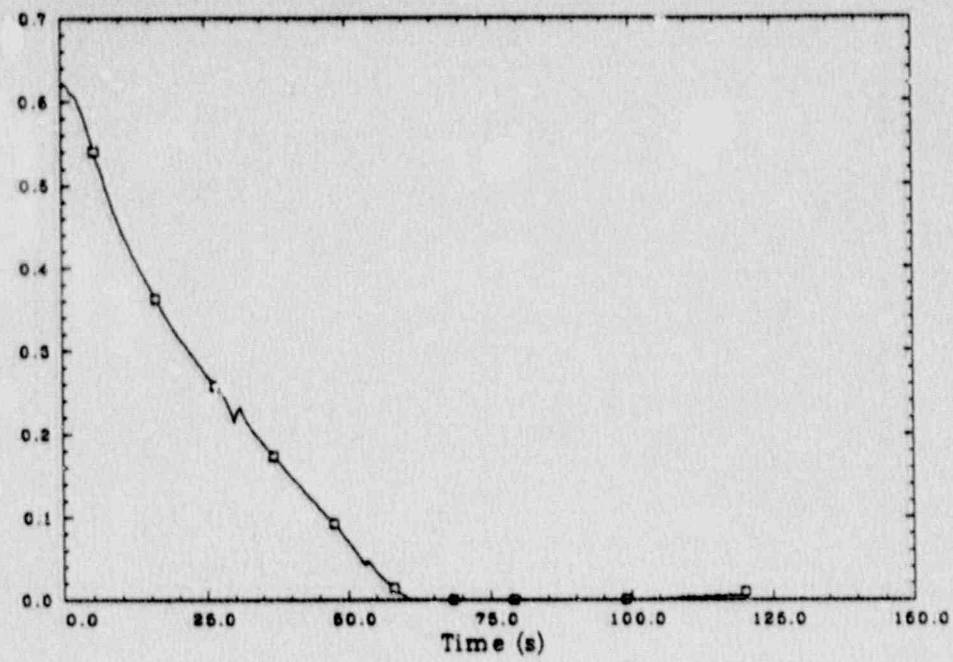

Figure 36. Scenario 4: pressurizer level.

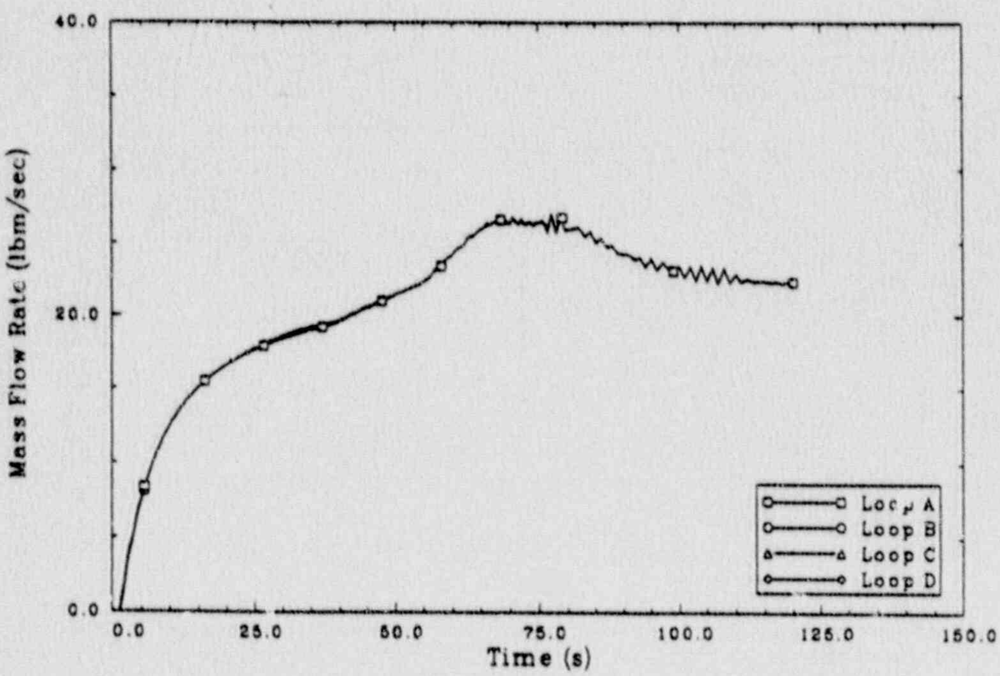

Figure 37. Scenario 4: ECCS mass flow rates.

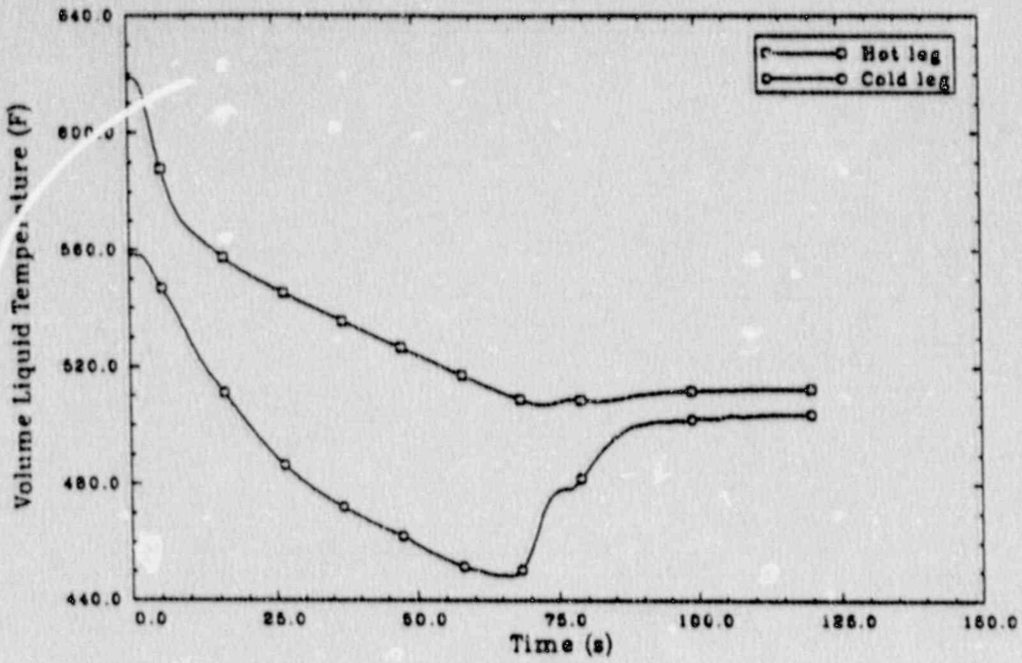

Figure 38. Scenario 4: loop A hot and cold leg temperatures.

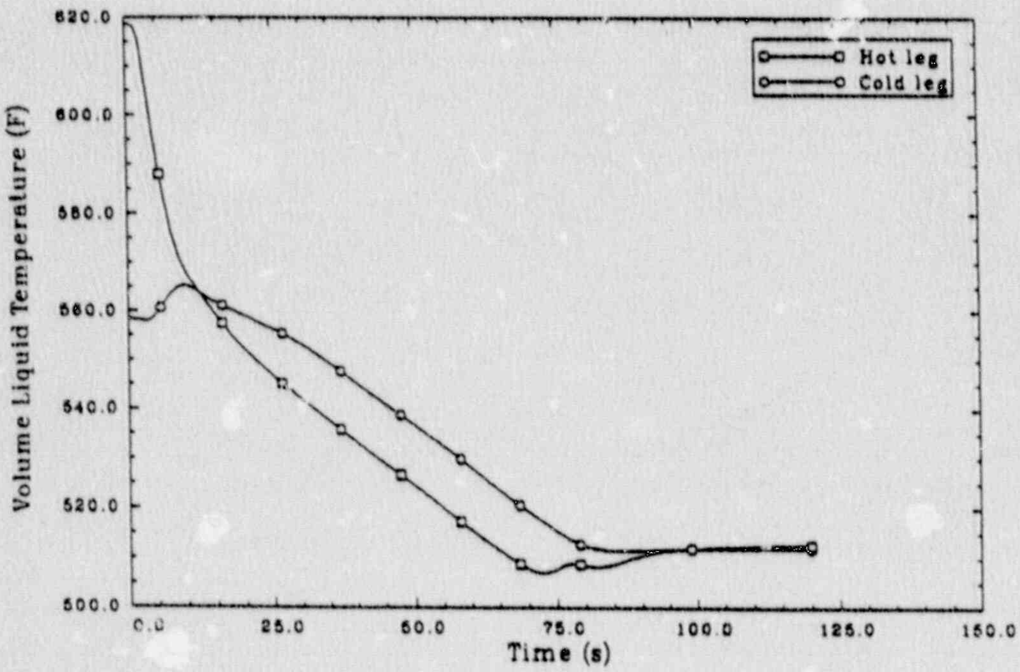

Figure 39. Scenario 4: loop B hot and cold leg temperatures.

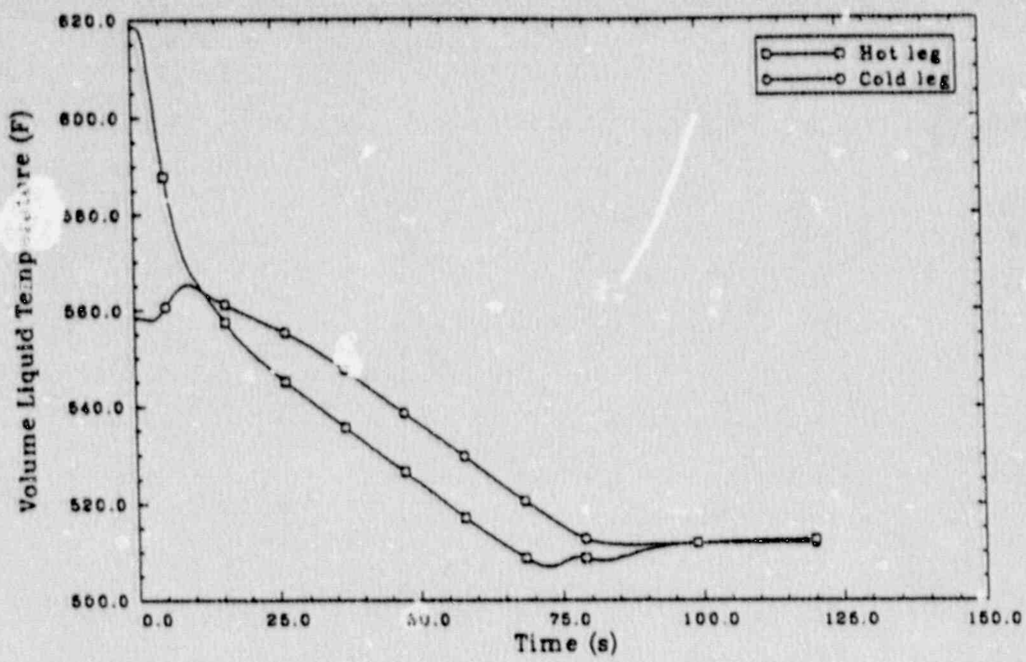

Figure 40. Scenario 4: loop C hot and cold leg temperatures.

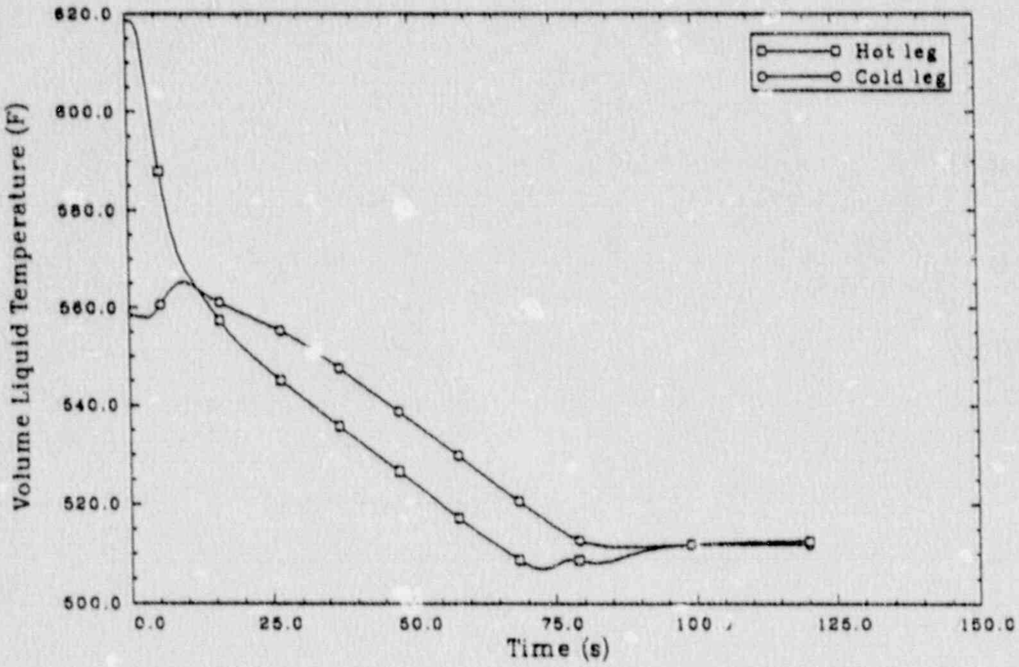

Figure 41. Scenario 4: loop D hot and cold leg temperatures.

7. SCENARIO 5: LOSS OF FEEDWATER WITHOUT SCRAM

The following section details the analysis of loss of feedwater accident with a delayed reactor scram. The transient was initiated from full power. The subsections contain a description of the scenario, model changes and assumptions used in the calculation, and analysis of the results.

7.1 Scenario Description

The transient was a complete loss of feedwater with an additional malfunction which prevented the automatic scram signal from tripping the reactor. The loss of feedwater was defined to be the simultaneous loss of all power to the feedwater pumps, heater drain pumps, and condensate pumps in the secondary feedwater train. Operator intervention was simulated with a RELAP5 time dependent trip to scram the reactor. This trip was delayed until after the automatic trip signal was generated in the simulation.

7.2 Model Changes and Assumptions

The basic SNUPPS model used to perform the calculation is detailed in Section 2. The initial conditions assumed for the transient are documented in Section 2.3.

Because of the delayed scram signal assumed, a reactor kinetics model was used for this simulation. This was done to ensure that after the loss of the secondary heat sink, the attendant increase in the moderator and fuel temperatures would result in a correctly simulated power response. Beginning of life (BOL) moderator and fuel temperature reactivity coefficients were used. These reactivity coefficients constitute a prime uncertainty in this scenario. It was assumed that the failure to scram was accompanied by a lockup of the rod speed controller, so that the control rods contributed zero reactivity in the RELAP5 simulation. All other modeled automatic control systems were assumed to work correctly before and after the generation of the automatic scram signal.

7.3 Calculated Results

The sequence of events that occurred in the loss of feedwater transient are presented in Table 8. The initiating event (loss of the feedwater train) generated an immediate motor auxiliary feedwater signal. At 22 s a feedwater isolation signal was generated by a steam generator low level signal. By 27 s a reactor trip signal had been generated by a steam generator low-low level narrow range signal, but failed to automatically scram reactor. However, the turbine stop valves did close. The degraded heat transfer between the primary and secondary sides resulted in the pressurizer PORV and safety valves opening at 35 and 102 s respectively. By 102 s the pressurizer level was at 100% of full span and the steam generators were less than 5% of their wide range full span. At 105 s assumed operator intervention was modeled by tripping the reactor. The primary pressure peaked at 108 s, reaching approximately 3000 psia. After the primary system reached its peak pressure, the plant transitioned to a depressurization and cooling mode. By 118 s the pressurizer PORVs and safety valves had closed. Continued plant cooldown and depressurization resulted in a SI signal being generated on low steam line pressure at 162 s. By 180 s primary cooling was sufficient to cause the pressurizer level to drop below 100% of full span.

At 270 s the simulation was terminated. The primary system was in a continued cooldown mode but the average loop temperatures had not reached the 557 °F no load average loop setpoint. Liquid recovery in the steam generator secondary sides had not yet commenced since the steam dump valves were still open. This resulted in the vaporization of most of the injected auxiliary feedwater that entered the steam generator boiler regions. The auxiliary feedwater provided an adequate heat sink for the removal of the decay heat. The decision to terminate the calculation was based on the conclusion that the most severe phase of the transient was over.

Figure 42 presents the calculated primary and secondary pressures. Before the automatic cram signal was generated, the pressure responses were

Table 8. Scenario 5: sequence of events

Time (s)	Event
0.	Feed train pumps trip off and motor auxiliary feedwater signal generated
22.	Feedwater isolation valves begin to close on single steam generator low level signal
23.	Turbine auxiliary feedwater signal generated on a 2/4 steam generator low level signal
27.	Scram signal generated on steam generator low-low level signal, reactor fails to trip, turbine tripped, steam dump valves open
35.	Pressurizer PORVs and steam generator PORVs modulate open
40.	Steam Generator safety valves modulate open
90.	Steam generator safety valves close and dryout calculated in steam generator boiler region
102.	Pressurizer Safety valves modulate open at 2500 psia, pressurizer liquid level reaches 100% of full span
105.	Reactor manually tripped, steam Generator PORVs close
108.	Pressurizer pressure reaches maximum value of 3000 psia
114.	Pressurizer Safety valves close
118.	Pressurizer PORVs close

Table 8. (continued)

Time (s)	Event
162.	SI signal generate on low steam line pressure
180.	Pressurizer level drops below 100% of full span
270.	End of simulation

fairly constant up to 27 s.The automatic scram signal was generated on a steam generator narrow range low-low signal. At this time the turbine was tripped and the steam dump valves opened. This transition from energy removal by the turbine to removal by the steam dump system significantly reduced the rate of primary to secondary heat transfer. The reduction in heat transfer resulted in an increase in the primary loop temperatures. Presented in Figure 43 are the cold, hot, and loop averaged temperatures for loop A. The trends in loops B-D were essentially the same but are not shown. The temperatures were characterized by a rapid increase in the time interval between 30 and 50 s. This period was followed by another period from 50-90 s where the temperatures had leveled out.

By 90 s the primary pressure and temperatures again began to sharply increase. This was a consequence of dryout in the steam generator secondaries and another drop in primary to secondary heat transfer. These rapid increases where not reversed until after the reactor was tripped. The primary pressure peaked at 108 s, reaching a maximum value of 3000 psia, and thereafter began to decrease. After the reactor scram the loop temperatures also began to drop.

Presented in Figure 44 is the calculated pressurizer level response. The pressurizer level response followed the primary loop temperature trends. Prior to the generation of the automatic scram signal the pressurizer level was relatively stable. Following the automatic reactor trip signal and degraded heat sink conditions, there was an initial rapid level rise during the period from 30 to 60 s. This was followed by a brief stabilization period from approximately 60 to 90 s. During this time interval the primary to secondary heat transfer was relatively stable. Beginning at 90 s the dryout of the steam generator secondaries induced a sudden increase in the primary fluid heatup rate and another rapid insurge of liquid into the pressurizer and level increase.

By 100 s the pressurizer level was at 100% span with the pressurizer pressure above the 2500 psia safety valve lift setpoint. The plant was tripped manually at 105 s to ensure that unrealistic primary pressures would not be calculated. The opening of the pressurizer PORVs and safety valves caused some asymmetrical behavior in loop flow responses in the early stages of the transient. During periods of flow out these valves the loop D mass flow responses differed from the flows in loops A-C. Presented in Figure 45 are the mass flow rates for loops A-D. After the PORV and safety valves closed the loop flow responses differed little between loop D and loops A-C.

Figure 46 presents the simulated core power. Prior to the manual reactor scram the increasing coolant temperatures resulted in a calculated core power reduction of approximately 16%. This power reduction was the consequence of both doppler and moderator feedback calculated in the RELAP5 kinetics model.

By 90 s dry out and super-heating in the steam generator boiler regions had begun as liquid levels fell in the steam generators (Figures 47 and 48). Figure 49 shows the fluid, steam, and saturation temperatures for the loop A steam generator at the top of the boiler. The temperature responses for the steam generators in loops B-D were similar. Once the dryout stage was reached there was a subsequent reduction in the secondary vapor generation rate. This reduction led to a depressurization of the secondaries. The depressurization was a consequence of the steam dump valves remaining open while secondary vapor generation had been terminated. By 130 s the steam generator wide range level readings had dropped to zero. After 160 s fluctuations in the wide range levels were calculated. These fluctuations were due to perturbations in the differential pressure readings as auxiliary feedwater entered the downcomer and flashed into vapor.

Despite the secondary dryout conditions, the primary to secondary heat transfer was still adequate to cool and depressurize the primary system. This was because virtually all of the auxiliary feedwater entering the steam generator boiler region vaporized. The vaporization was the result of core heat decay removal as well as the opening of the steam dump valves which tended to blow the steam generators down. By 162 s the secondary pressure had dropped to 600 psia which triggered the ECC system. Presented in Figure 50 is the total ECC mass flow rate.

At 270 s the simulation was terminated. It was concluded that: the most severe phase of the simulation was over; the primary system would eventually transition to no-load zero power conditions without any complications. In conclusion, the loss of feedwater transient with an automatic scram failure was characterized by two phases. In the first phase the primary system was subject to an uncontrolled heat up and pressurization prior to tripping the reactor. In the second phase the primary system was characterized by a cool down and depressurization with the primary loop temperature approaching the no-load 557 °F loop average set point.

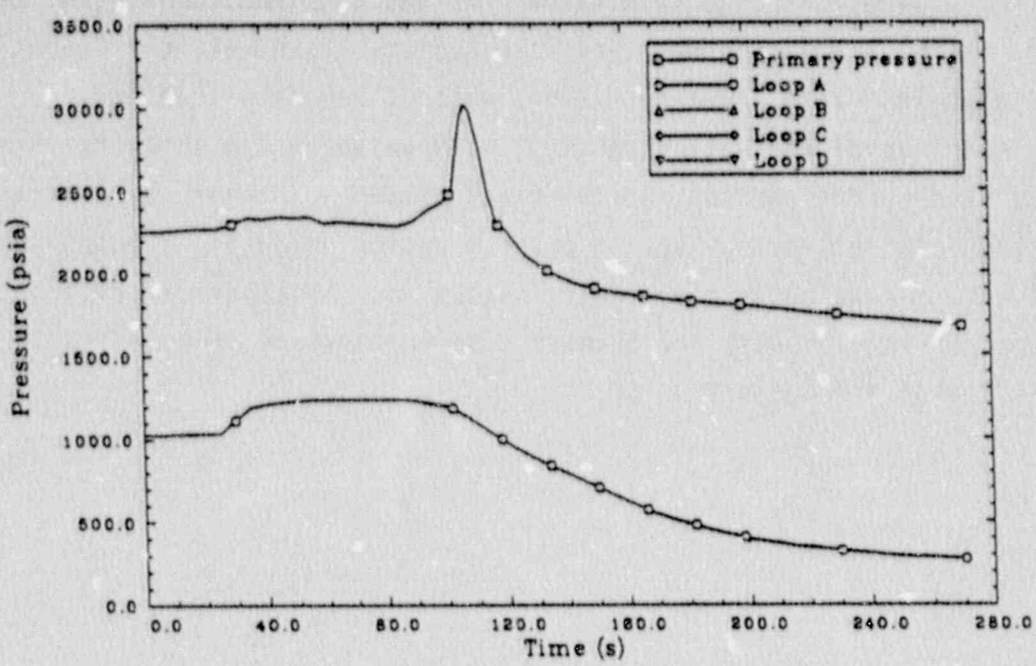

Figure 42. Scenario 5: primary and secondary pressures.

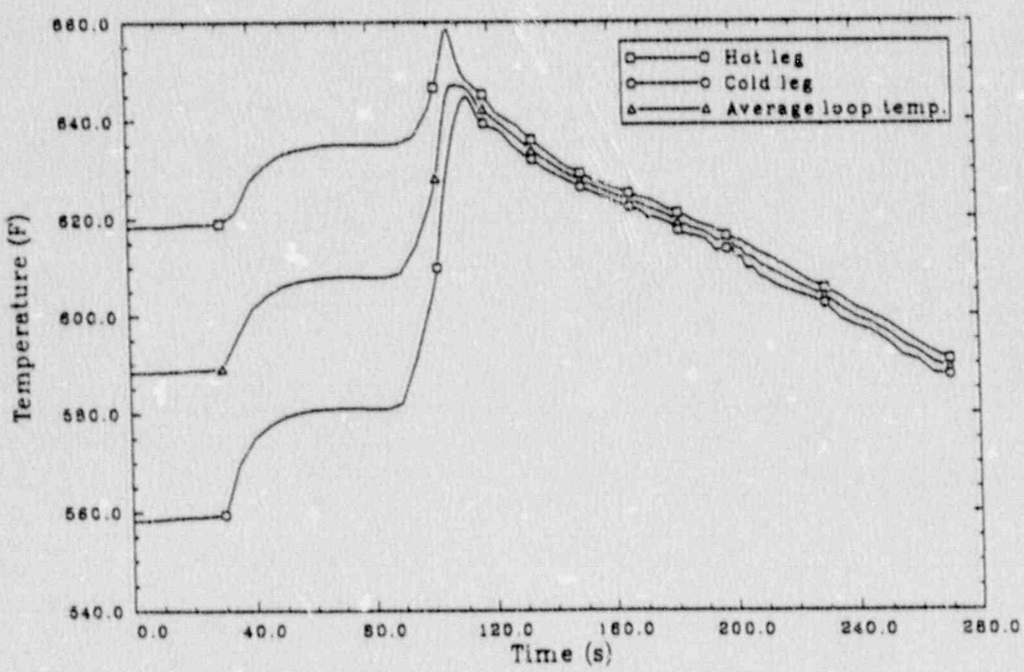

Figure 43. Scenario 5: primary loop hot, cold, and average temperatures.

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IMAGE EVALUATION TEST TARGET (MT-3)

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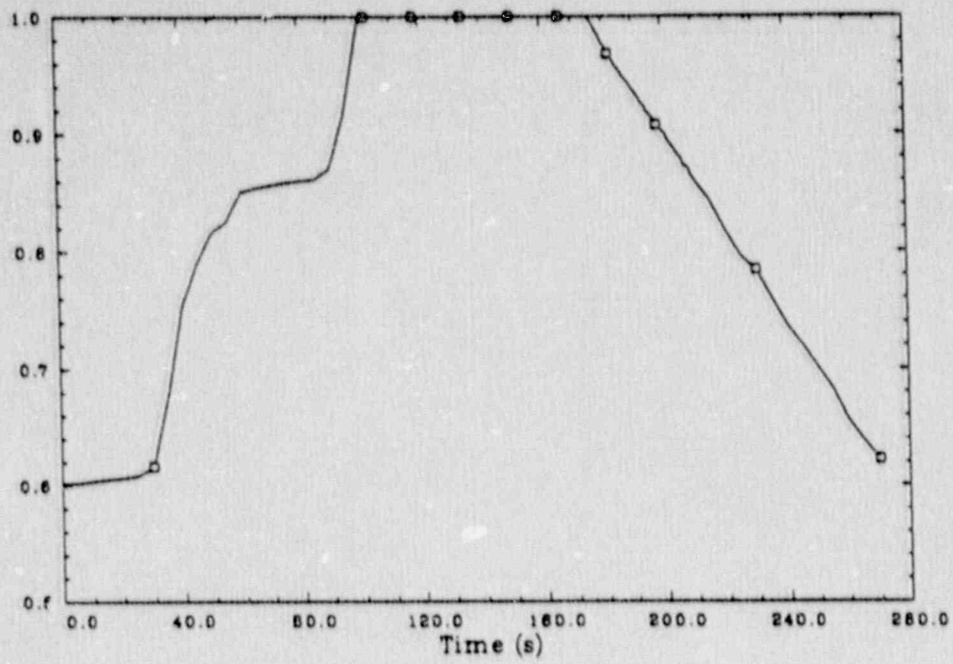

Figure 44. Scenario 5: pressurizer level.

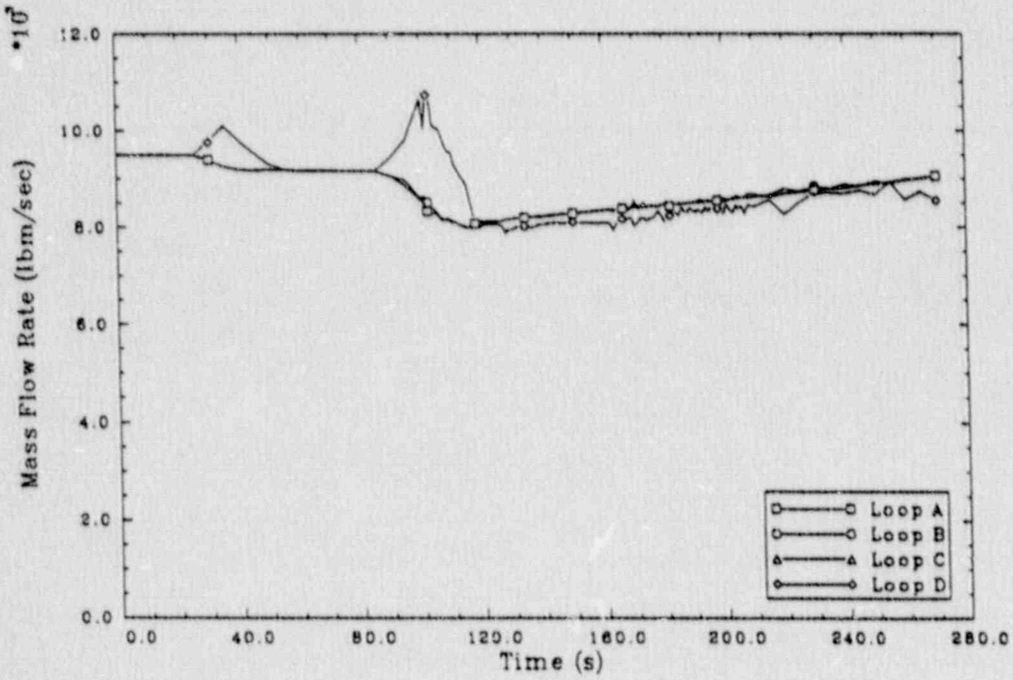

Figure 45. Scenario 5: primary leg mass flow rates.

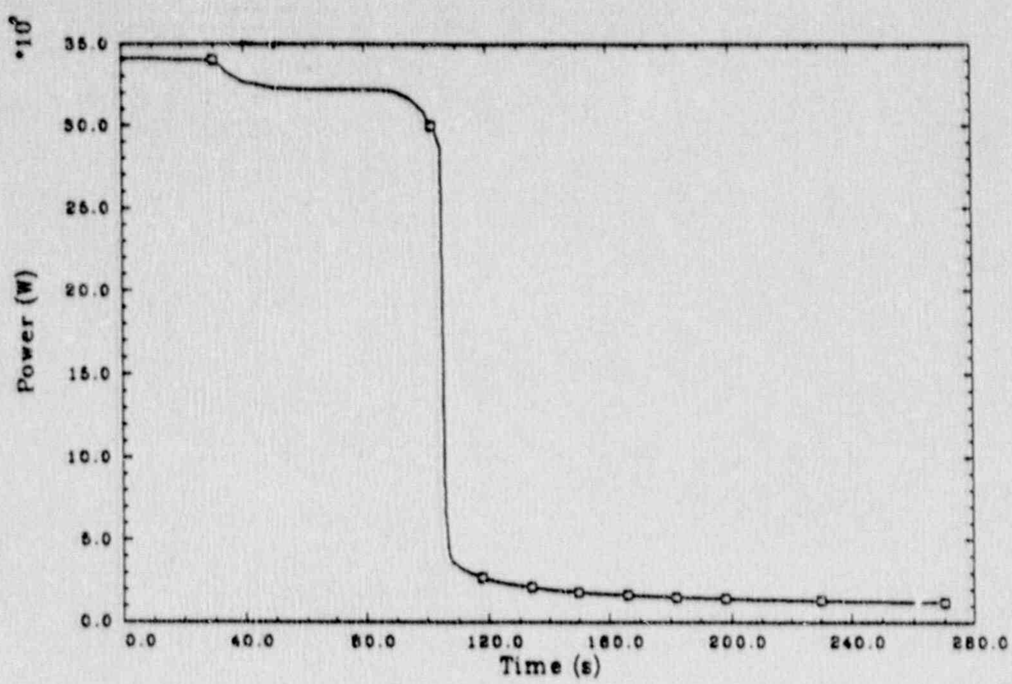

Figure 46. Scenario 5: reactor power.

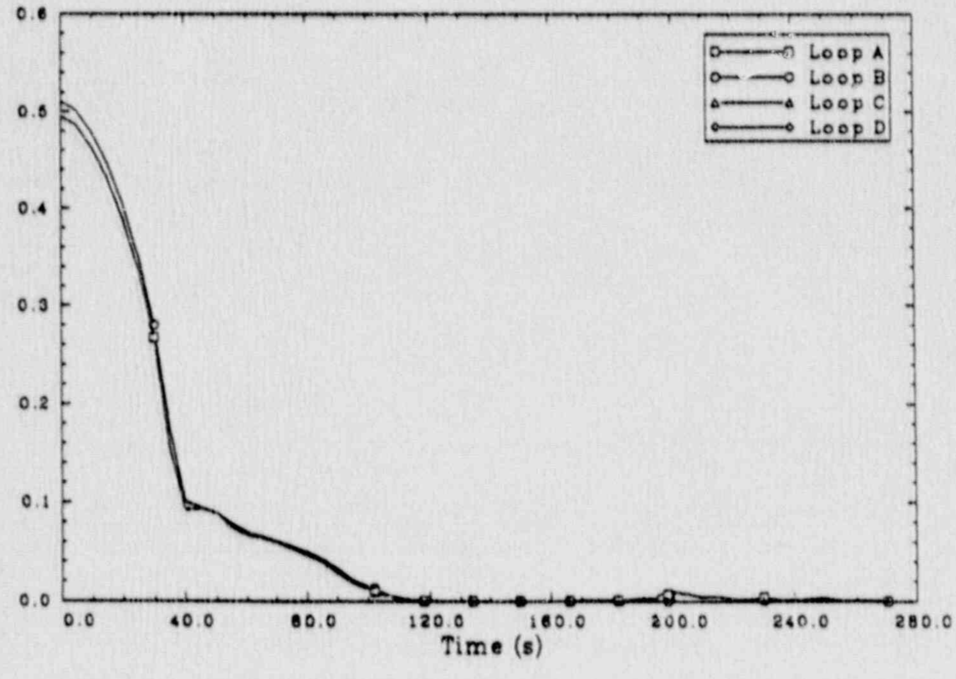

Figure 47. Scenario 5: steam generator narrow range levels.

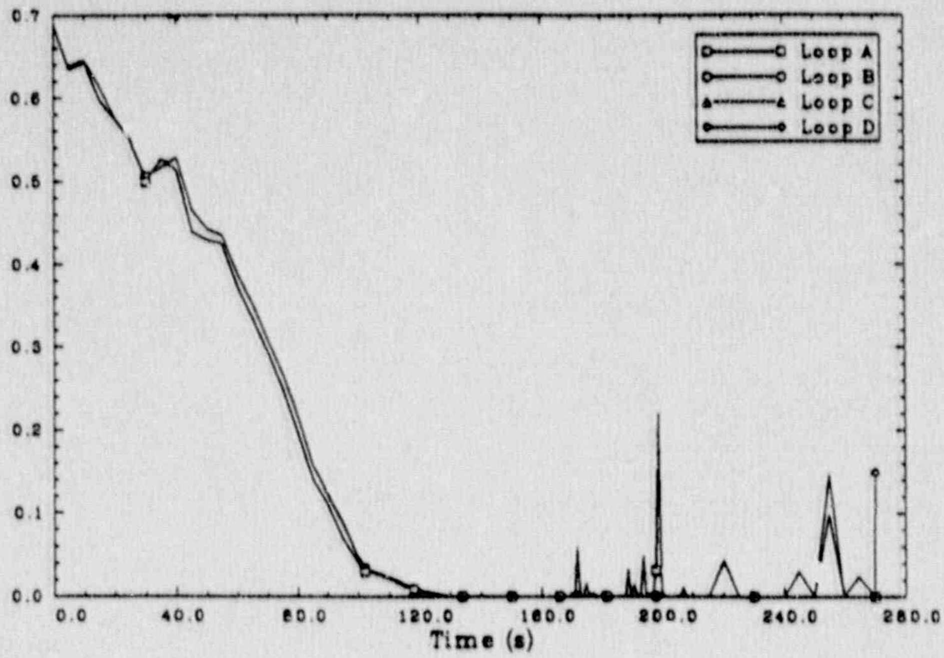

Figure 48. Scenario 5: steam generator wide range levels.

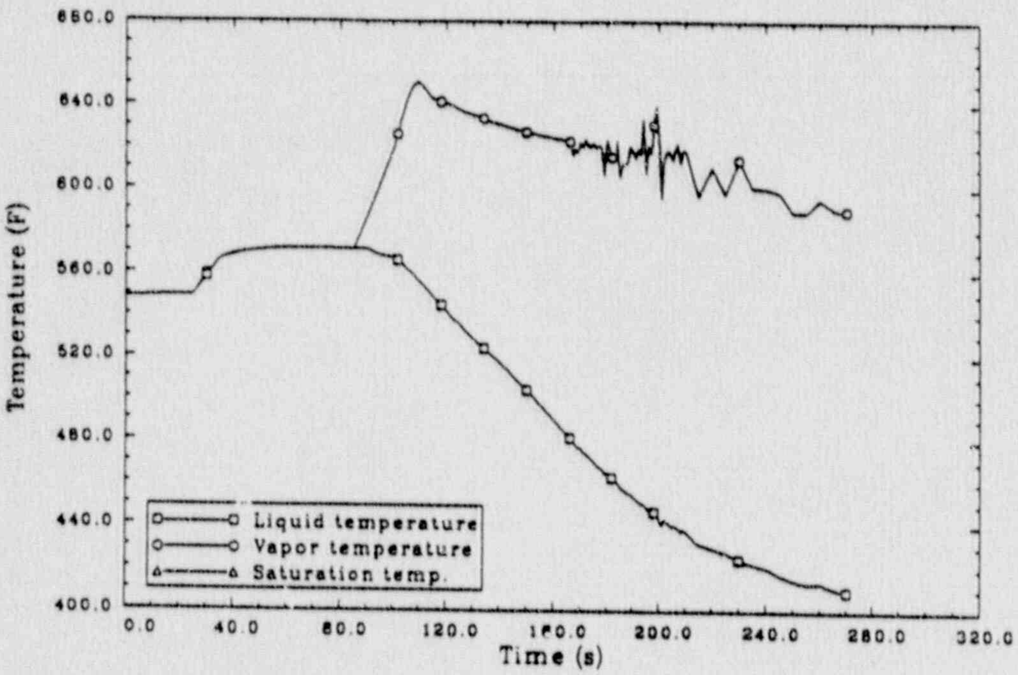

Figure 49. Scenario 5: loop A steam generator boiler region liquid, vapor, and saturation temperatures.

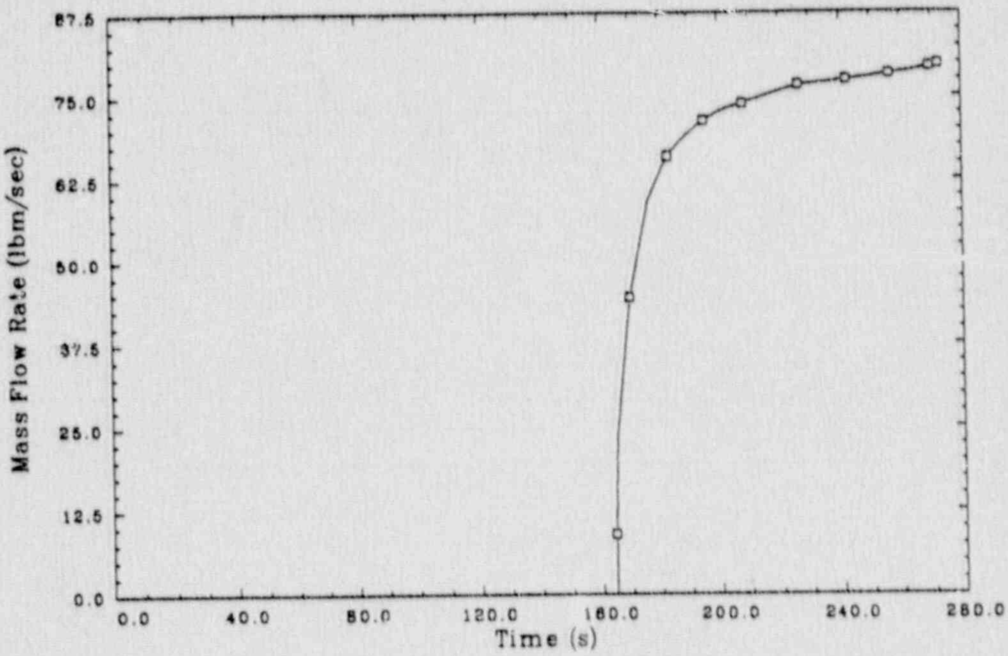

Figure 50. Scenario 5: total ECCS mass flow rate.

8. CONCLUSIONS AND RECOMMENDATIONS

Analyses of five SNUPPS scenarios were performed with the RELAP5 computer code. The purpose for doing these calculations was to benchmark the TTC SNUPPS simulator with RELAP5 to determine how well the simulator can model a wide range of hypothetical accidents and identify where simulator software technology can be improved. Computational information presented in this report is a sample of a much more detailed data base calculated by the RELAP5 code. Additional data for these simulations is stored on magnetic tape and maintained at the INEL. This data will be used for future TTC simulator/RELAP5 benchmark comparisons. The conclusions of this report include the following remarks.

1. In general, the calculated RELAP5 trends were reasonable for the scenarios in this report and will provide a valid basis for comparison with simulator data. This conclusion was based on extensive review of the scenario data by experienced operators and plant analysts at INEL.

2. Uncertainties relative to boundary conditions, delay times, and instrumentation process delay times still have not been resolved and could potentially effect simulator/RELAP5 results. The comparison with simulator data will not only be used to assess simulator performance and the code's capability to model the more mechanistic phenomena of plant behavior.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

Thermal-hydraulic analyses of five hypothetical accident scenarios were performed with the RELAP5 computer code for the Westinghouse Standardized Nuclear Unit Power Plant System pressurized water reactor. This work was sponsored by the U.S. Nuclear Regulatory Commission and is being done in conjunction with future analysis work at the U.S. Nuclear Regulatory Commission Technical Training Center in Chattanooga, TN. These accident scenarios were chosen to assess and benchmark the thermal hydraulic capabilities of the Technical Training Center Standardized Nuclear Unit Power Plant System simulator to model abnormal transient conditions.

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