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GLOSSARY

AFWS - Auxiliary Feedwater System

ARV - Atmospheric Relief Valve

CCP - Centrifugal Charging Pump

ECC - Emergency Core Cooling

FSAR - Final Safety Analysis Report

LER - License Event Report

LOCA - Loss-of-Coolant Accident

MCP - Main Coolant Pump

MSIV - Main Steam Isolation Valve

MSL - Main Steam Line

NRC - Nuclear Regulatory Commission

PORV - Power-Operated Relief Valve

PRT - Pressure Relief Tank

PWR - Pressurized Water Reactor

RCP - Reactor Coolant Pumps

RHR - Residual Heat Removal

RSS - Reactor Safety Study

SASA - Severe Accident Sequence Analysis

SIP - Safety Injection Pump

SV - Safety Valve

TMI - Three Mile Island

TRAC - Transient Reactor Analysis Code

LOSS-OF-FEEDWATER TRANSIENTS FOR THE ZION-1 PRESSURIZED WATER REACTOR

by

N. S. DeMuth
D. Dobranich
R. J. Henninger

ABSTRACT

The response of the Westinghouse Zion-1 pressurized water reactor to transients initiated by loss of main feedwater with auxiliary feedwater unavailable was simulated using the Transient Reactor Analysis Code (TRAC). The normal response mode in which emergency systems perform as designed was first studied to identify critical equipment performance and operator actions necessary for normal recovery. Subsequent analyses were performed to determine the effects of additional equipment failures, such as valves sticking open, and delayed or degraded operation of emergency systems. Strategies were developed for operator actions not covered in existing emergency procedures and were tested using TRAC simulations to evaluate their effectiveness in preventing core uncovery.

I. INTRODUCTION

The accident at Three Mile Island (TMI) has focused attention on the potential for equipment/instrumentation failures and their consequences during anticipated transients. It is now recognized how easily normal recovery procedures can be rendered ineffective and how important operator responses are to controlling these types of accidents and to mitigating their severity. Median estimates of accident probabilities developed in the Reactor Safety Study[1] (WASH-1400), together with current schedules for reactor construction in the USA, indicate that one or more serious accidents involving radioactive releases may occur before the end of this century with current reactor designs and operating procedures.[2] In the aftermath of TMI, investigators pointed out the need for simulating a wide range of postulated transient or accident conditions including equipment failures and operator actions, for adding and upgrading instrumentation to monitor plant conditions during accidents, and for improving emergency procedures and training to assure proper operator response to various accident conditions.[3-5] The Nuclear Regulatory Commission (NRC), as part of its response to these needs, initiated the Severe Accident Sequence Analysis (SASA) program to further our understanding both of reactor accident phenomena and of the human-machine interface during a spectrum of accidents.

Los Alamos National Laboratory, Idaho National Engineering Laboratory (INEL), and Sandia National Laboratories (SNL) are investigating potentially severe accidents in pressurized water reactors (PWRs) for the SASA program. The initial studies of severe accidents up to core uncovery were divided between Los Alamos and INEL. The investigations at Los Alamos focused on transients involving loss of feedwater to the steam generators, and INEL studies considered accidents involving loss-of-off-site power.[6] Los Alamos and INEL studies currently focus on defining initiating events and on simulating numerically the accident progression up to core uncovery; investigators at SNL are determining the accident events from core uncovery through meltdown and containment failure. Investigations of other accident scenarios are under way, and later reports will describe these sequences.

The contribution to SASA by Los Alamos required delineation of potentially severe accident sequences at specific nuclear power plants and thermal-hydraulic simulations of the plant response to equipment failures and operator actions during the accident. These were accomplished by performing computer simulations using the Transient Reactor Analysis Code (TRAC-PD2), an advanced, best-estimate computer program for the analysis of accidents in light-water reactors.[7] TRAC simulates the behavior of a PWR subjected to abnormal and transient conditions, including saturation of the primary coolant and loss-of-coolant accidents (LOCAs). Much of the steam-side equipment including the steam generator secondary, main steam line (MSL) relief and safety valves, main steam isolation valves (MSIVs), and auxiliary feedwater system (AFWS) can be modeled. The version of TRAC-PD2 used in the SASA program was modified to permit simulation of the pressurizer relief and safety valves.

The first plant selected for study in the SASA program was the Zion-1 PWR, which is one of two nearly identical units operated by the Commonwealth Edison Company at its Zion Station in Illinois. Design of the nuclear steam supply system at these plants is similar to other four-loop Westinghouse PWRs. Design and operation information on the plant and its safety systems was obtained from the Final Safety Analysis Report,[8] abnormal and emergency operating procedures, and from discussions with operations personnel at the plant.

This report describes investigations of transients involving loss of feedwater to the steam generators. Loss of main feedwater, coupled with loss or unavailability of auxiliary feedwater, can lead to reactor coolant loss from the primary system relief and safety valves after the heat sink provided by the water initially present in the steam generator secondaries is depleted. Emphasis was placed on defining the event sequences and associated timing of automatic or operator-initiated actions to prevent core uncovery. The failure of equipment to perform as designed and operator actions that could aggravate or mitigate the severity of the accident also were considered. Strategies were developed for operator action not covered in existing emergency procedures and were tested using TRAC simulations to evaluate their effectiveness in preventing core uncovery. Our studies assessed the effects of delayed initiation and degraded performance of critical equipment (for example, the AFWS) in preventing core uncovery and the effects of uncertainties in plant operating variables on the accident sequence and severity.

II. INITIATING EVENTS AND NORMAL RECOVERY MODES

Loss-of-main-feedwater flow to the steam generators was assumed to be the initiator for transients in which auxiliary feedwater was unavailable. Interruption of main feedwater flow can be caused by several circumstances, such as loss-of-off-site power; tripping of the turbines; malfunctions in the feedwater flow control system; and mechanical failures in the pumps, valves, or piping. Failure of the AFWS to start and perform as designed can result from

1. closed AFWS pump discharge valves;
2. breaks in the main header;
3. failures in two pump loops while the third is disabled for maintenance or testing; and
4. lack of sufficient flow from the condensate storage tank and service water supplies, owing to plugged vents.

For transients initiated by loss-of-off-site power, unavailability of auxiliary feedwater can result from failure of the diesels to start and provide emergency power for the motor-driven pumps and motor-operated valves combined with failure of the turbine-driven pump.

For our analyses of the plant response to feedwater transients, the reactor was assumed to be operating at its rated power within safety and support system limits prescribed in the technical specifications. Transients resulting from loss-of-off-site power, including those initiated by voltage or frequency fluctuations, affect plant operation differently from those produced by other initiators in that the reactor coolant pumps (RCPs) trip automatically at the start of these transients, while for other initiators the RCPs continue operating until tripped by the operator.

Normal recovery from transients involving loss of main feedwater requires several automatic actions with the operator verifying that these actions have occurred. These actions include

1. reactor scram,
2. tripping main coolant pumps and starting diesels for loss-of-off-site power initiator,

3. starting turbine-driven AFWS pump, and

4. starting motor-driven AFWS pump.

After verifying that these actions have occurred, the operator is instructed to monitor the pressurizer level and pressure, start heaters and charging pumps as necessary to restore the pressurizer level to its normal range, and to throttle AFWS flow after the steam generator levels return to the narrow range. With these systems operating as designed, the operator can place the plant in a hot shutdown condition and subsequently proceed to cold shutdown, where the residual heat removal system (RHR) can be used for cooling.

If the AFWS fails to function on initial demand, the operator can dispatch control-room personnel to learn the reason for the malfunction and attempt to correct it manually. If this is unsuccessful and off-site power is available, the operator can align valves such that the condensate and condensate-booster pumps can be used to supply water from the condensate storage tank to the main feedwater pump suction; then by "bumping" the main feedwater pumps, the water inventory in the steam generator secondaries can be replenished temporarily. Another option available to the operator is actuating the emergency core cooling (ECC) system to prevent inadequate cooling of the core; this will be discussed later in this report. An evaluation of severe accident sequences initiated by loss-of-off-site power is given in Ref. 6.

III. COMPUTER MODEL DESCRIPTION

A system schematic of the TRAC-PD2 model for the Zion PWR is shown in Fig. 1. Information for this model of a four-loop Westinghouse plant was derived from the Final Safety Analysis Report (FSAR) for Zion-1 and from visits to the site. To improve calculational efficiency, three of the loops (A, C, and D) were modeled as one combined loop and the remaining loop (B), which contains the pressurizer, was modeled separately. At the top of the pressurizer are the components that model the primary pressure relief system, which includes power-operated relief valves (PORVs), safety valves (SVs), header, and pressure relief tank (PRT). Operation of the PORVs was simulated with a special valve model added to TRAC-PD2 for SASA applications. This model included an option to allow different opening and closing pressure

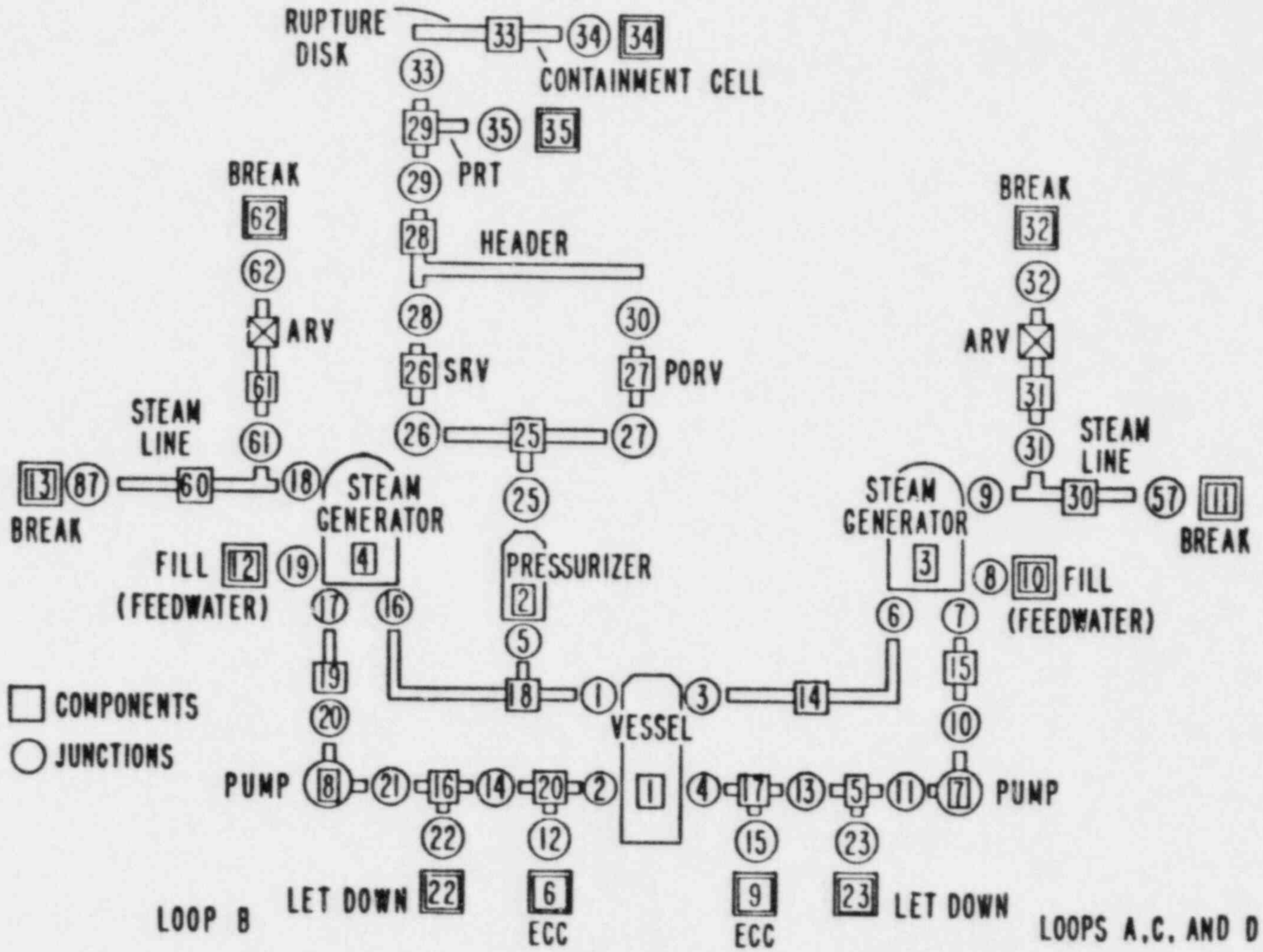

Fig. 1. TRAC system schematic for the Zion-1 PWR.

setpoints as well as different opening and closing stroke rates. The SVs were modeled as static check valves that open and close depending upon the pressure difference across the valve. The PORVs and SVs connect to a common header that leads to the pressure relief tank. A pathway from the pressure relief tank to the containment is provided by rupture disks that are designed to open when the pressure in the tank reaches 0.69 MPa, thus providing a connection from the relief valve discharge to the containment. The containment was modeled as a single TRAC cell with provisions for fan coolers, heat slabs, and containment sprays. Included in the primary loops are separate components for modeling ECC injection, primary coolant makeup and letdown, main coolant pumps, and U-tube steam generators. On the secondary loops, the feedwater pumps, steam lines and atmospheric relief valves (ARV), and SV are modeled.

The TRAC model for the Zion-1 plant contains 196 mesh cells, including 32 in the three-dimensional vessel component shown in Fig. 2. To improve computational efficiency for the lengthy feedwater transients, a much coarser noding scheme was employed than would be used for large-break loss-of-coolant analyses. However, accurate modeling of the performance of critical systems (for example, operation of PORVs, SVs, and ARVs and initiation of ECC flows on a containment overpressure signal) introduced additional complexities and substantially increased the noding requirements. The initial conditions for feedwater transients computed with TRAC are compared in Table I with information from the FSAR.[8] These comparisons indicate that the steady-state operation of the Zion-1 plant is being simulated accurately.

IV. RECOVERY FROM LOSS-OF-FEEDWATER TRANSIENTS

Thermal-hydraulic calculations were performed to simulate the response of the Zion-1 reactor to transients in which main feedwater flow was interrupted and the auxiliary system failed to supply feedwater on demand. The initiating event for this transient was assumed to be loss-of-off-site power, which trips the turbines, the reactor coolant pumps, the main feedwater pumps, and generates a signal to scram the reactor. For these calculations, a delay of 0.6 s was assumed between the time the scram signal was generated and the control rods started to fall, and a mid-range value for the shutdown margin of reactivity (3.5% $\delta k/k$) was used. Normal recovery from these transients would be effected by automatic actuation of the turbine-driven auxiliary feedwater

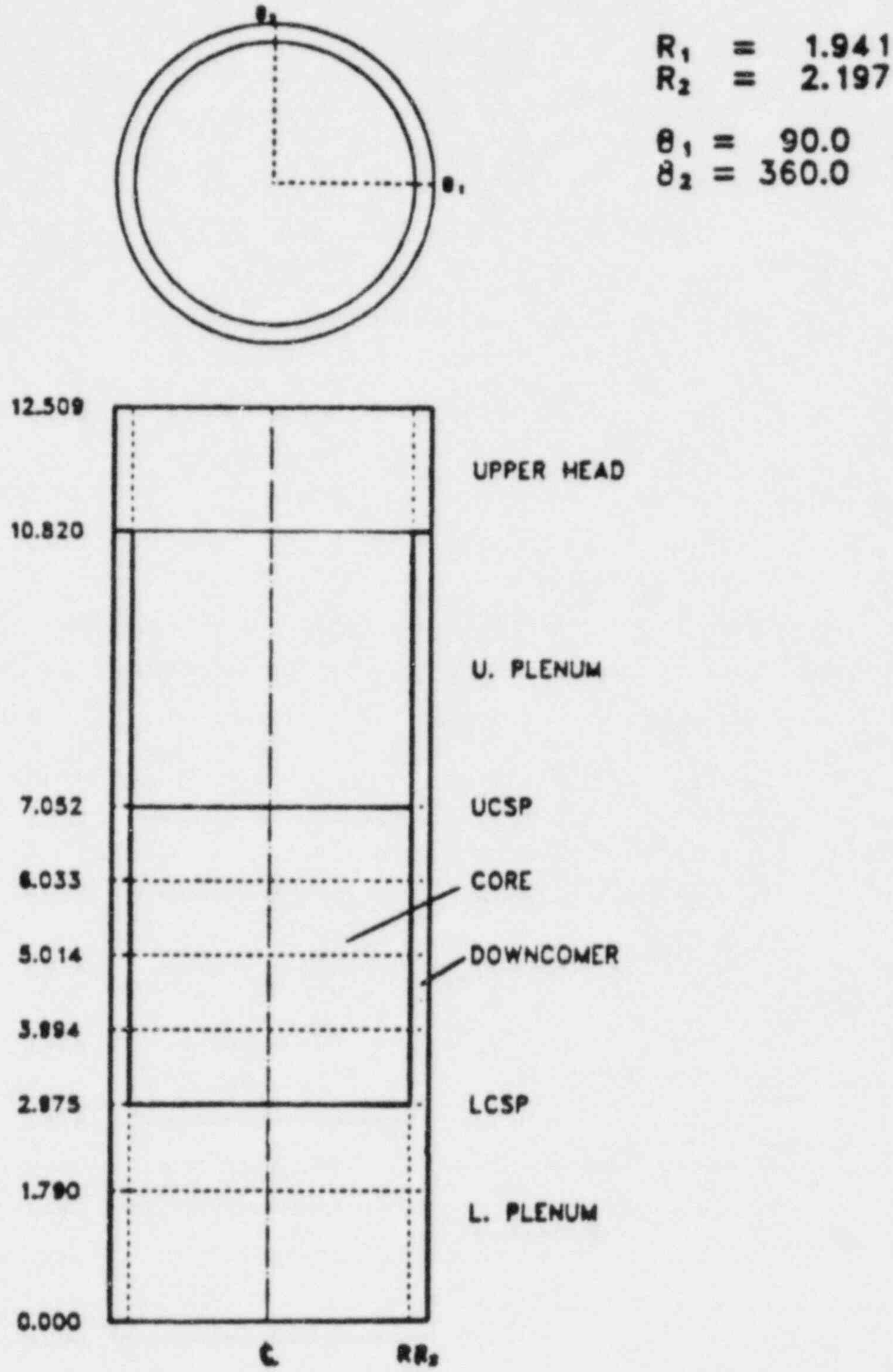

Fig. 2. TRAC model of reactor vessel for the Zion-1 PWR.

TABLE I

COMPARISON OF STEADY-STATE OPERATING CONDITIONS FOR THE ZION-1 REACTOR

Parameter	TRAC[a]	FSAR[b]
Vessel inlet temperature (K)	549.8	549.9
Vessel outlet temperature (K)	585.5	585.5
Coolant flow per loop (kg/s)	4223.	4252.
Pressure drops (Pa)		
Vessel	3.39×10^5	3.58×10^5
Steam generator	2.07×10^5	2.08×10^5
Loop total	6.41×10^5	6.31×10^5
Pressurizer		
Pressure (Pa)	1.54×10^7	1.55×10^7
Temperature (K)	618.0	618.0
Steam Generator Secondary		
Steam exit temperature (K)	537.1	536.7
Pressure (Pa)	5.00×10^6	4.96×10^6
Exit flow (kg/s)	437.6	441.0

[a] Ref. 7.
[b] Ref. 8.

pumps, which would begin to deliver water at about 15 s, and by sequencing onto the emergency power system of the motor-driven auxiliary feedwater pumps, which would begin to deliver after about 30 s. In the absence of operator intervention, failure of the AFWS to supply water to the steam generators would result in loss of the heat sink represented by the water inventory of the secondaries. The event sequence calculated for this transient is summarized in Table II, and the plant response characteristics are shown in Figs. 3 through 10.

TABLE II

EVENT SEQUENCE FOR RECOVERY FROM A LOSS-OF-FEEDWATER ACCIDENT[a]

Time (s)	Time (min)	Event
0.0		Loss of off-site power trips turbines, reactor cooling pumps, main feedwater pumps and generates reactor scram signal.
0.6		Control rods drop (1-s insertion time).
15		Turbine-driven pumps fail to deliver auxiliary feedwater.
30		Motor-driven pumps fail to deliver auxiliary feedwater.
60	1	Atmospheric relief valves on steam lines open.
3800	63	Steam generator secondaries empty of water.
4000	67	PORV opens (primary pressure = 16.1 MPa).
4800	80	Pressurizer solid; PRT rupture disks open.
5800	97	ECC tripped on high containment pressure (0.13 MPa).
6800	113	Pressurizer level begins to decrease.
7200	120	Primary coolant saturates, loss of natural circulation.
7300	122	Clad temperature reaches peak of 625 K.
7600	127	Upper plenum 90% empty, top of core begins to uncover.
7800	130	Recovery begins ($T_L < T_{sat}$), core 8% empty.

[a]Figs. 3 through 10.

The decay power of the reactor during this transient is shown in Fig. 3. After the reactor coolant pumps tripped and coasted down, natural circulation flows were established in the primary loops, as shown in Fig. 4. These flows

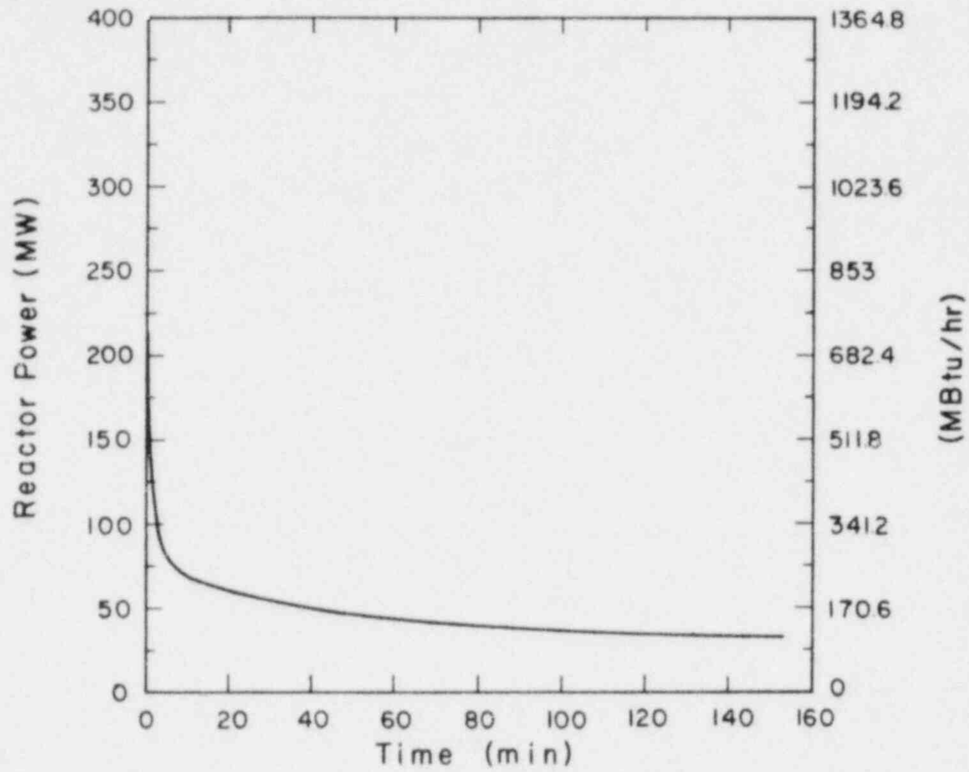

Fig. 3. Reactor power decay following scram.

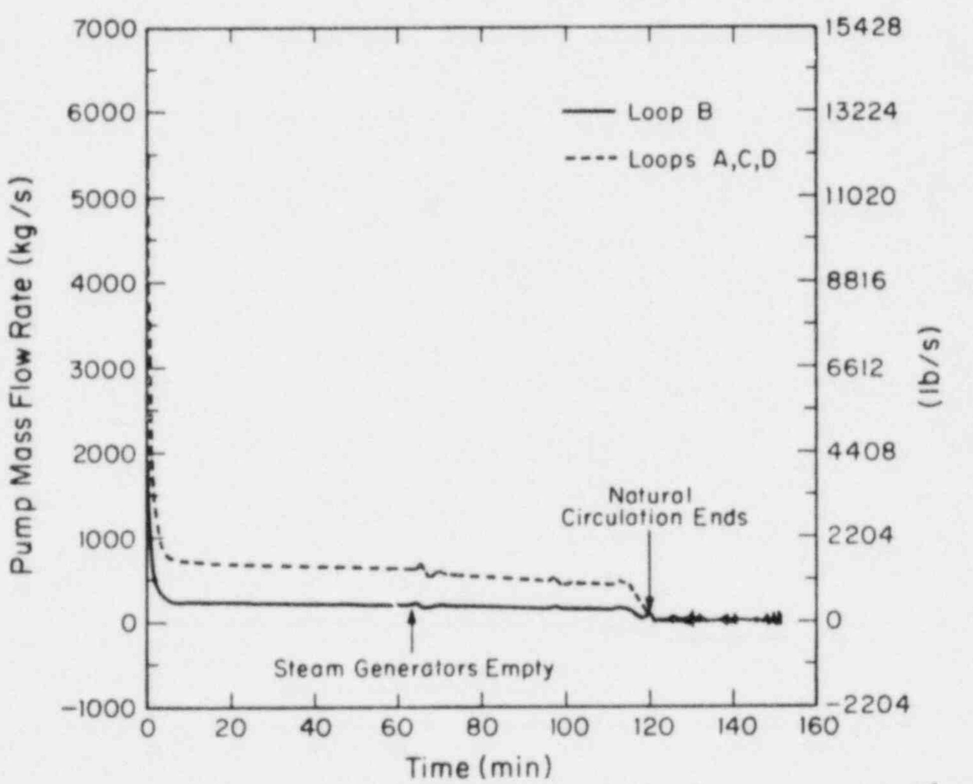

Fig. 4. Natural circulation established after pump coastdown ends at ~2 h.

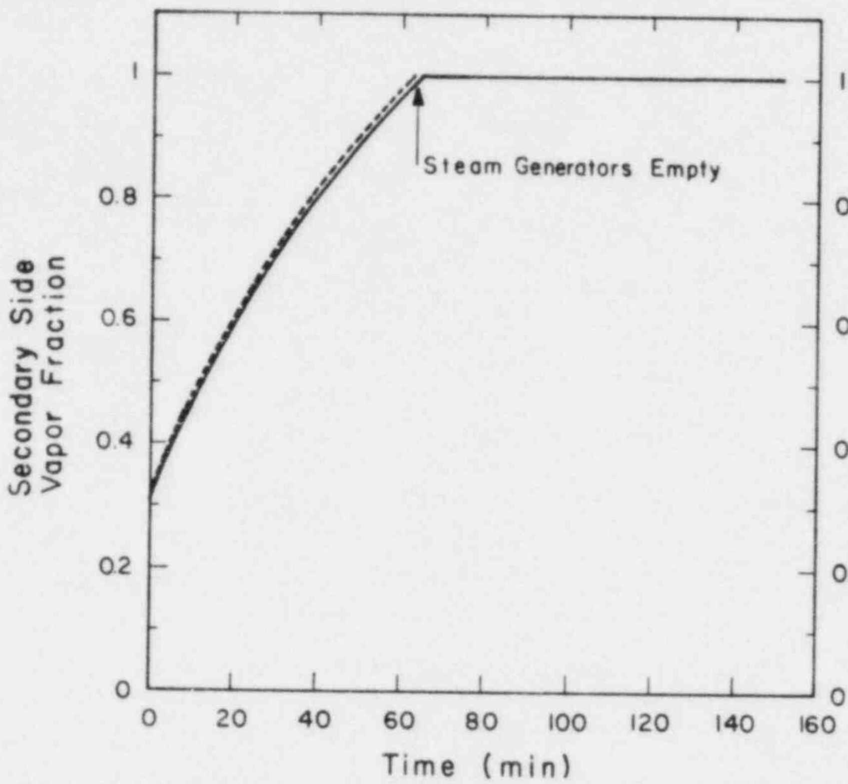

Fig. 5. Water inventories present in the steam generator secondaries are depleted after ~1 h.

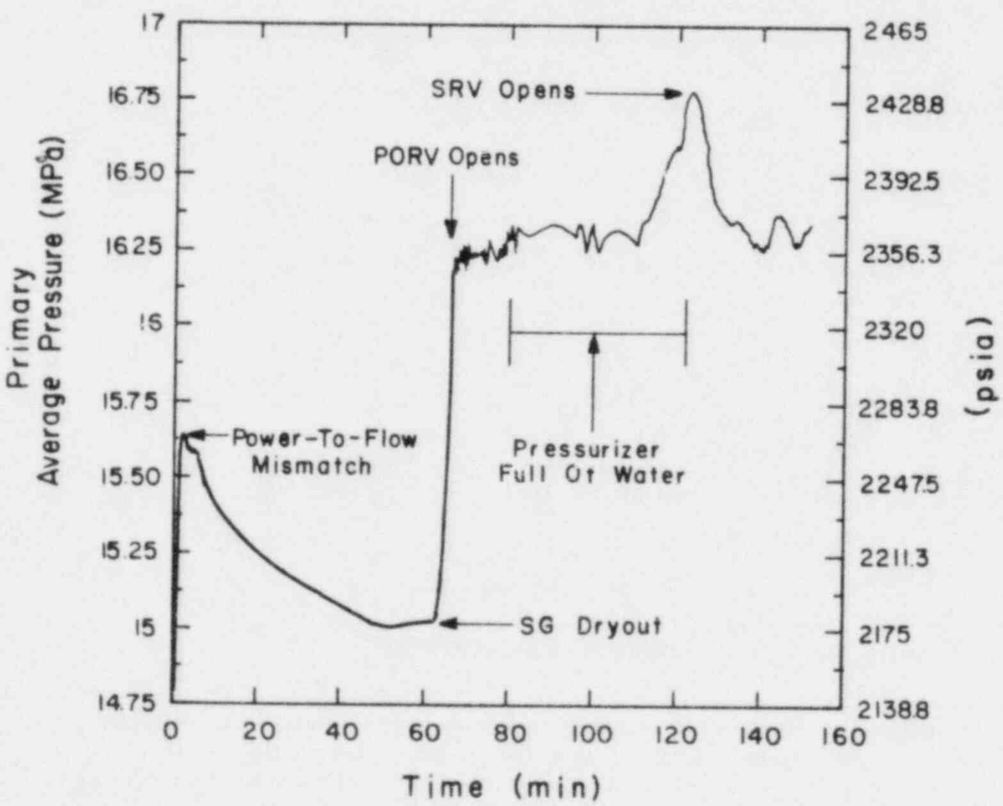

Fig. 6. Primary pressure during loss-of-feedwater transient.

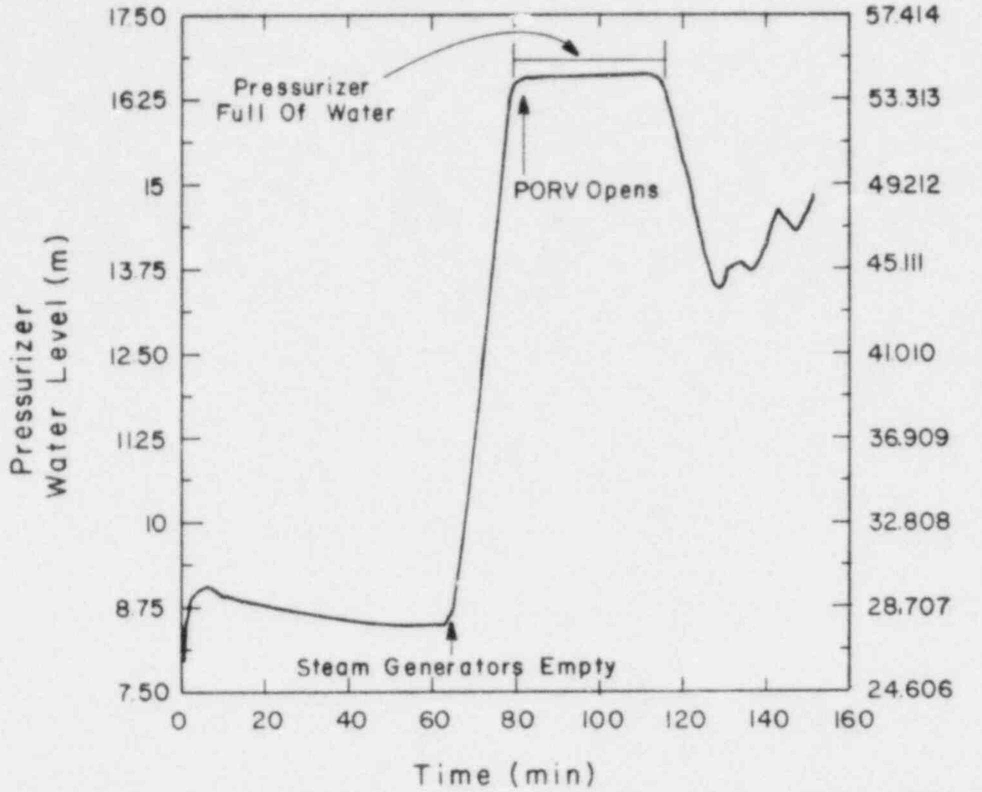

Fig. 7. Pressurizer fills with water after steam generators empty.

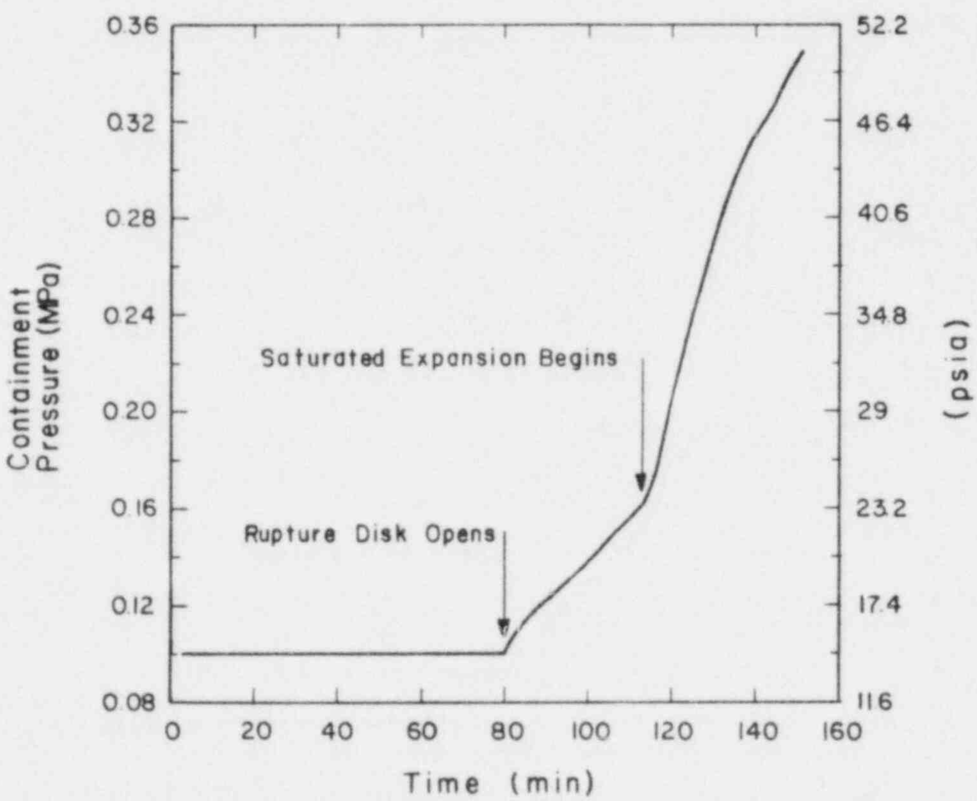

Fig. 8. Rupture disks on pressurizer relief tank open allowing primary coolant discharge into containment.

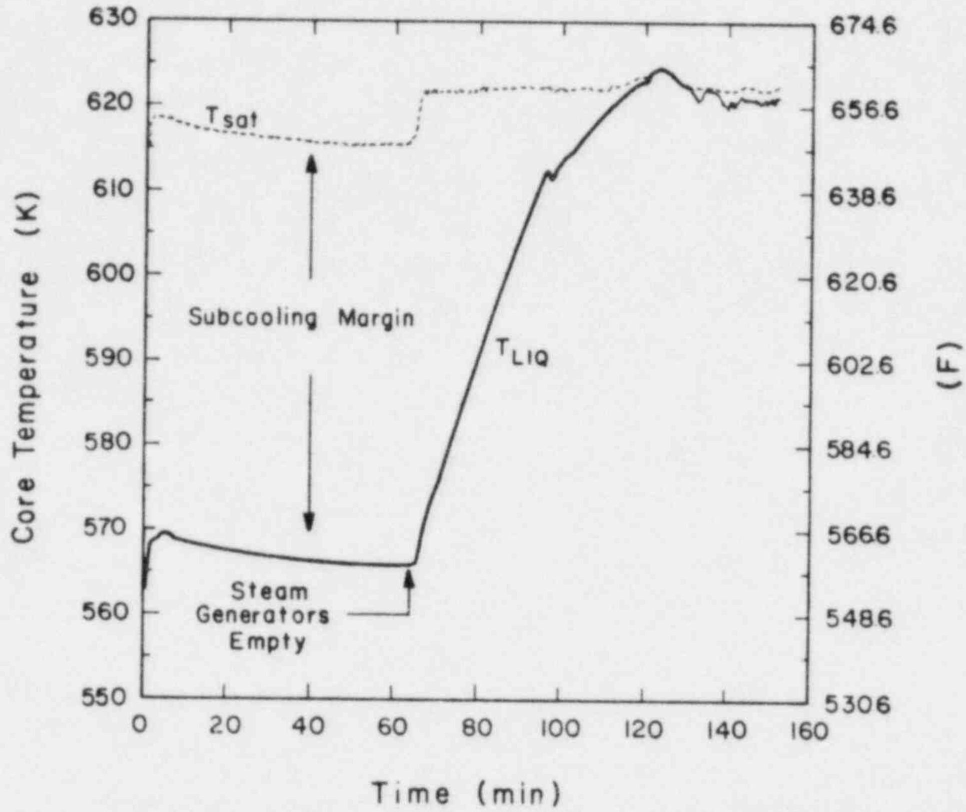

Fig. 9. Primary coolant reaches saturation conditions after ~2 h.

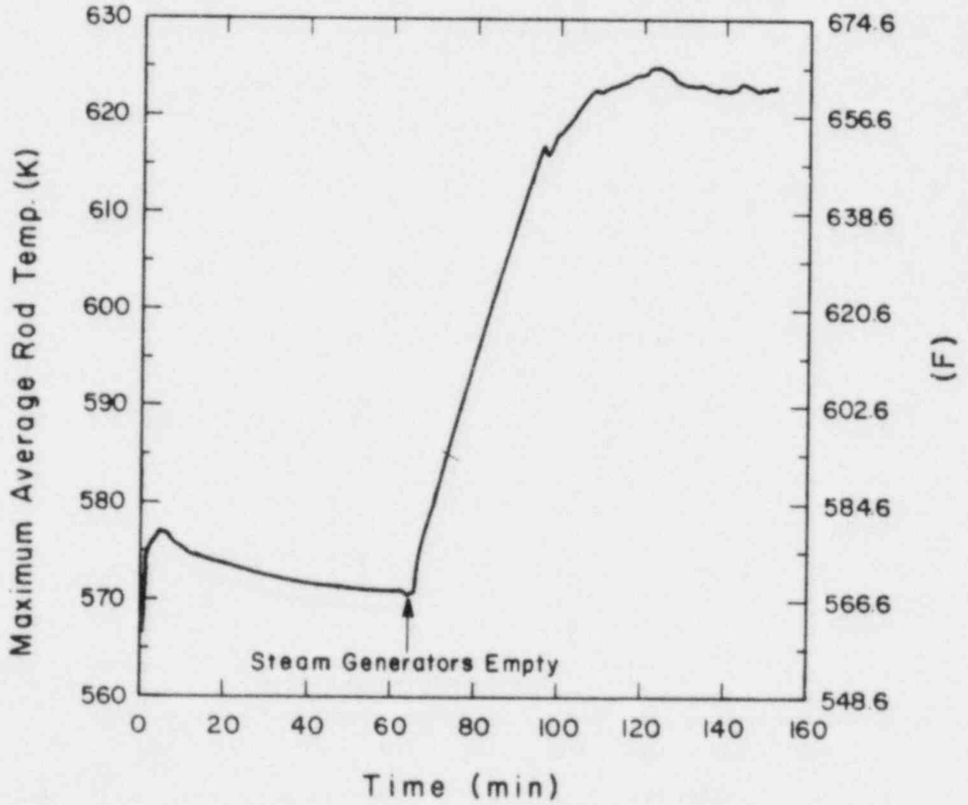

Fig. 10. Cladding heatup on fuel rods operating at average powers during loss-of-feedwater transient.

continued until the primary coolant reached saturation conditions and the upper portions of the primary voided. During the first ~1 h of the transient, decay heat was removed by boiling the water inventory present on the secondary side of the steam generators (Fig. 5), and the steam was exhausted through the atmospheric relief valves. After loss of the heat sink, represented by the water inventory of the steam generator secondaries, the primary pressure and temperature increased rapidly, causing the PORVs to open. The primary pressure shown in Fig. 6 remained constant near the PORV setpoint as long as the volumetric expansion rate was less than the relieving capacity of the PORVs. The heatup and expansion of the primary coolant caused the water level in the pressurizer to rise (Fig. 7) so that steam-water mixtures are exhausted through the PORVs. Shortly after the pressurizer filled with water, the rupture disks on the pressurizer relief tank opened, providing a pathway into the containment (Fig. 8), and the ECCS was initiated by a containment overpressure signal at ~97 min.

After the steam generators emptied, the temperatures of the fuel, coolant, and vessel internals increased, and decay heat then was removed by expanding the primary coolant through the PORVs. The open area of the PORVs was controlled by the expansion rate of the primary coolant and changes to maintain the pressure near its setpoint. When the primary reaches saturation (Fig. 9), the expansion rate changed from a subcooled to a saturated mode, the saturated expansion rate being greater by a factor of about 6. Once the PORVs reached their maximum open area, the pressure increased until the SV setpoint was attained. For the conditions of this transient, the fully open PORV area, together with the higher pressure, was sufficient to relieve the saturated expansion and to prevent continued opening of the safety valves. The system then continued boiling at pressures near the PORV setpoint. The ECC flow at pressures equal to or greater than the PORV setpoint was insufficient to remove the decay energy of the reactor without boiling the primary coolant. Depletion of the primary coolant continued until the decay power declined to a level where the boiling of the subcooled ECC water could provide the necessary cooling. This was calculated to occur before the core began uncovering so that the primary system reached a quasi-equilibrium condition and further heatup of fuel-rod cladding shown in Fig. 10 was arrested. As the decay power slowly decreased, steam voids in the primary condensed, and subcooling was recovered.

Reactor cooling can be maintained for several hours by injecting water through the high-head charging pumps until the refueling water storage tank is depleted. During this time the operators must re-establish secondary cooling and/or replenish the refueling water storage tank before proceeding to a cold shutdown condition.

V. INSTRUMENTATION RESPONSE AND OPERATOR ACTIONS

Loss-of-feedwater transients are indicated by changes in instrument readings and alarms occurring within a few seconds after the initiating event. Included among these will be indications that certain automatic actions have occurred (for example, reactor scram and starting of turbine-driven auxiliary feedwater). During the early portion of the transient, the operator must deduce the type of transient, locate the appropriate emergency procedure, and perform the operations listed therein. Initially the operators are required to verify that certain automatic actions have occurred, and then they are to control the primary temperature and pressure using auxiliary feedwater and makeup flows. Within ~1 min after the start of the transient, the operator should observe that no auxiliary feedwater is flowing to the steam generators and therefore should attempt to activate the pumps from the control room. Between 1 and 2 min, the operator will observe that the reactor has scrammed, the primary pressure is decreasing from the slightly elevated peak, the main coolant pumps are coasting down, and natural circulation flows (4 to 5% of full flow) are established.

Subsequent indications of no auxiliary feedwater flow, together with falling water levels in the steam generators should prompt the operator to dispatch personnel to examine the auxiliary feedwater pumps and piping. Between 20 and 30 min into the transient, the operator should have sufficient information to decide whether auxiliary feedwater flow can be restored before the water inventory in the steam generator secondaries is depleted. If the operator takes no action, dryout of the steam generators will occur, accompanied by increasing primary pressures and temperatures and a rising water level in the pressurizer. PORV operation will result in rising pressure and temperatures in the pressurizer relief line, followed by rising pressure and temperatures in the containment building. The reactor instrumentation also will indicate the gradual core heatup and loss of subcooling margin in the primary coolant. These indications could prompt the operator to intervene in the accident by initiating emergency core cooling before it is actuated by a containment overpressure signal at ~97 min into the accident.

About 2 h into the transient, temperature gauges indicate the loss of subcooling, and flow meters indicate the cessation of natural circulation. The high-head charging pumps may not sustain operation at high pressures and flows, so operator intervention in accident sequences of this type is necessary to prevent further core damage.

Restoration of secondary cooling before natural circulation ceases will enable the reactor to be brought to a hot shutdown condition. If off-site or emergency power is recovered, this can be accomplished by restoring auxiliary feedwater flow (for example, opening valves as was done at Three Mile Island - Unit 2) or by nonstandard techniques such as aligning condensate and condensate-booster pumps to draw water from the condensate storage tank while bumping the main feedwater pumps. Options for operator intervention using equipment and systems attached to the primary include

1. early initiation of ECC flow;
2. operation of the PORVs to reduce pressure, thereby increasing the flow from the centrifugal charging pumps and augmenting it with flow from the safety injection system pumps; and
3. blowdown of the steam generator secondaries to cool and depressurize the primary to a point where decay power can be removed by recirculating flow through the residual heat removal system.

The results of numerical simulations to determine the feasibility and limitations of these actions are discussed in Sec. VIII.

VI. EVENT TREES

Event trees identify the possible outcomes of initiating events, with individual accident sequences being depicted by the various paths in the event trees. The event tree shown in Fig. 11 illustrates several accident sequences initiated by a loss-of-off-site power transient, which trips the main feedwater pumps. Automatic insertion of the control rods and initiation

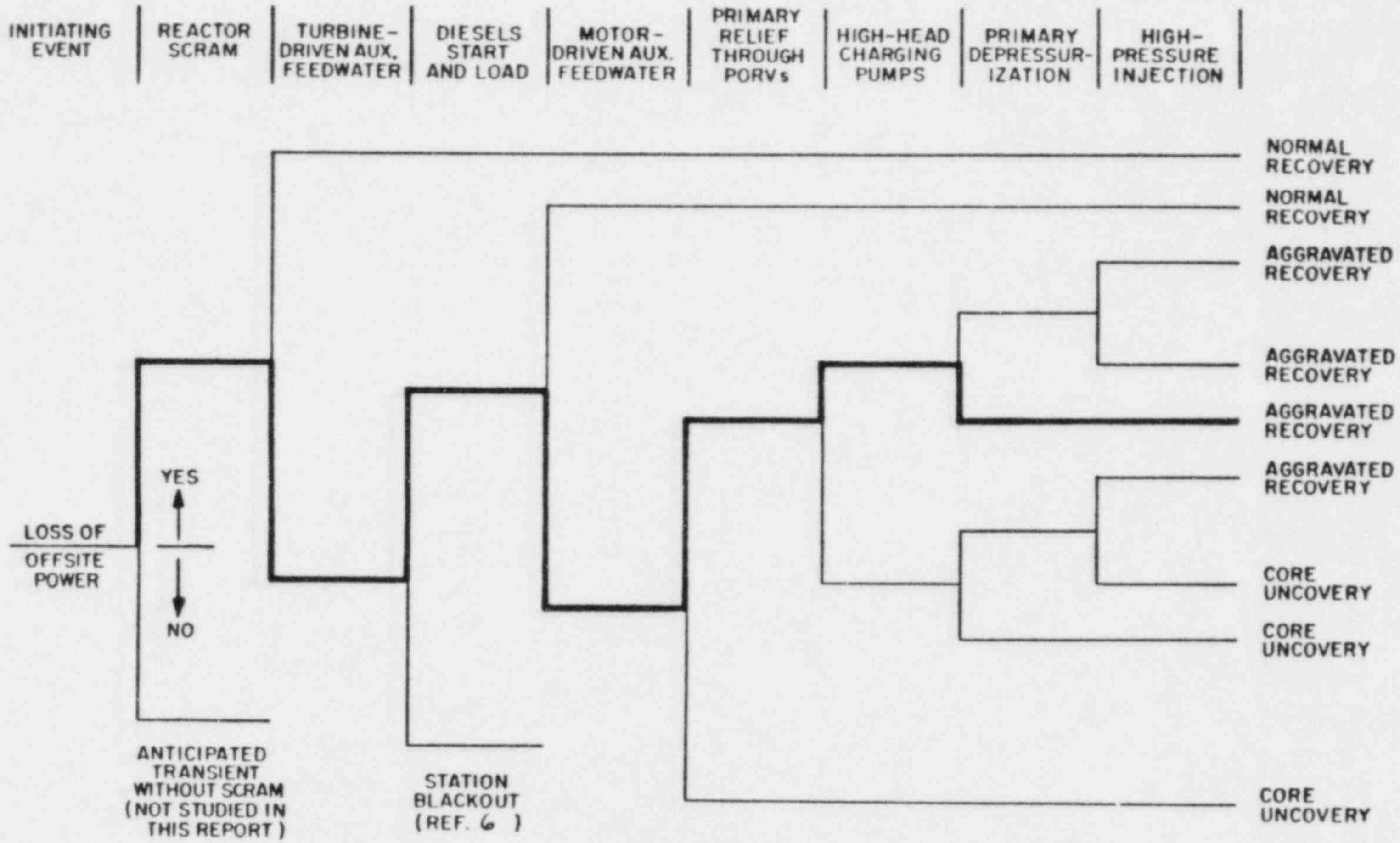

Fig. 11. Event tree for feedwater transients.

of auxiliary feedwater flow to the steam generators from either the turbine-driven or the motor-driven pumps will enable the plant to be placed in a stable condition. Accident sequences in which the reactor is not shut down by control rod insertion or boron injection are not considered in this study, and accidents involving failure of emergency power are discussed in Ref. 6.

The event sequence depicted by the wide line in Fig. 11 corresponds to the loss-of-feedwater accident discussed in Sec. IV. The outcome of this accident is a quasi-stable condition (denoted aggravated recovery), in which the reactor is cooled by water from the refueling water storage tank. To place the plant in a stable condition, the operators must restore secondary cooling or depressurize to the point where the reactor can be cooled by recirculating flow through the residual heat removal system. Failure of the PORVs (or the associated block valves) to open will result in the primary pressure increasing to the setpoint of the safety valves. At these pressures, the flow provided by the high-head charging pumps will be insufficient to cool the reactor, and the primary coolant inventory will be depleted, thus uncovering the core. Failure of the high-head charging pumps to deliver water at the pressure setpoint of the PORVs can also lead to core uncovery unless the operator acts to reduce the primary pressure facilitating cooling by the high-pressure injection system.

The event sequences for feedwater transients depicted in Fig. 11 indicate that a stable plant state can be reached with auxiliary feedwater provided by both motor-driven pumps or by the one turbine-driven pump. Total loss of secondary cooling can lead to core uncovery, if the PORVs fail to open or if the high-head charging pumps fail to deliver at the pressure setpoint of the PORVs. If the operator acts to depressurize the primary following failure of the charging pumps, the plant can be placed in a quasi-stable condition with cooling provided by the high-pressure injection system.

VII. FAILURES CAPABLE OF AGGRAVATING RECOVERY

Normal recovery from a loss-of-feedwater transient requires the following actions:

1. control rod insertion,

2. auxiliary feedwater supply (from either motor-driven or turbine-driven pumps),

3. secondary steam relief (either through steam dump or atmospheric relief valves),

4. primary coolant makeup, and

5. essential AC and DC power.

Equipment malfunctions that delay or degrade performance of these necessary functions will aggravate normal recovery and possibly lead to a severe accident. Equipment malfunctions during anticipated transients also may produce instrumentation responses different from those in emergency or abnormal operating procedures, thereby making early accident identification more difficult and adding uncertainty to proper operator response.

Several equipment failures in systems critical to normal recovery were considered in this study, and others were identified for further investigation and inclusion in a later report. The effect on plant response and on accident severity of the following malfunctions is described in this section.

1. Delayed or degraded supply of auxiliary feedwater to the steam generators occurs.

2. Atmospheric relief valve on one steam generator fails to reclose after opening.

3. One PORV fails to reclose after opening.

4. Charging pumps fail to supply emergency cooling at the PORV setpoint pressure.

Further investigation is planned to evaluate transients involving delayed scrams, steam generator tube ruptures, and loss of coolant from the steam space of the pressurizer.

A. Delayed and Degraded Auxiliary Feedwater Supply

Numerical simulations were performed to determine the minimum auxiliary feedwater flow necessary for recovery. The initiating event and automatic actions were the same as listed in Table II, except that the auxiliary feedwater flow was started early in the transient, and the flow was varied to find the minimum value that led to normal recovery with current operating procedures.

The Zion-1 plant has one turbine-driven and two motor-driven auxiliary feedwater pumps; each motor-driven pump has a capacity of 30 kg/s, and the turbine-driven pump is rated at 60 kg/s. The results of TRAC calculations, which used a flow corresponding to 15% of the total capacity of the auxiliary feedwater pumps, are shown in Figs. 12 and 13. This is the minimum AFWS flow necessary for recovery. The water inventory in the steam generators (Fig. 12) decreased steadily until ~6-1/4 h, at which time the amount of heat that could be removed by boiling the auxiliary feedwater at the pressure corresponding to the setpoint of the atmospheric relief valves equaled the decay heat produced in the reactor core. At ~4-3/4 h, the primary-to-secondary heat transfer was degraded by the low inventory so that the primary temperature and pressure (Fig. 13) began to rise; this rise was arrested at ~6-1/4 h, however, as the amount of heat generated equaled that removed by boiling the feedwater. Subsequently, the steam generator inventories continued to increase so that the plant could maintain hot shutdown conditions and proceed to cold shutdown.

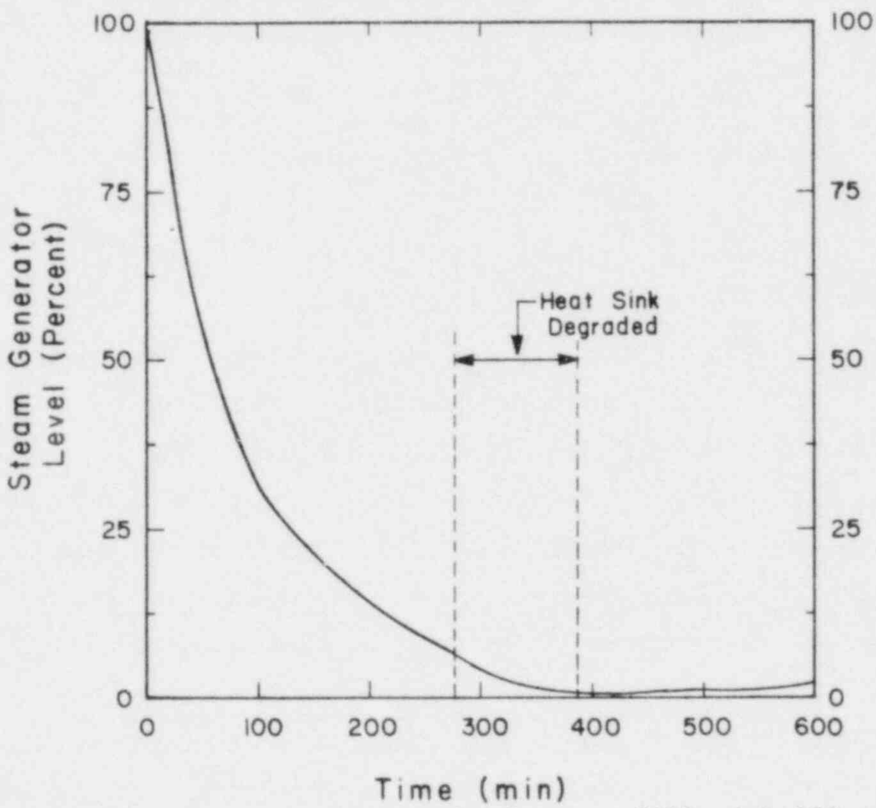

Fig. 12. Steam generators begin to refill with 15% total auxiliary feedwater flow.

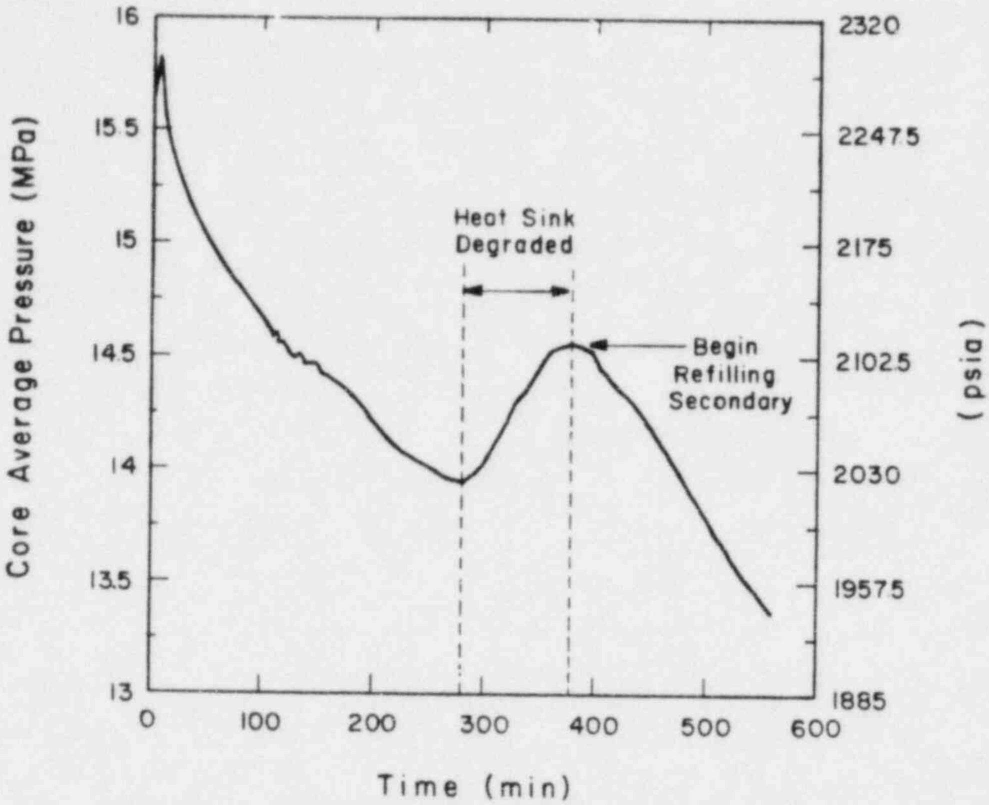

Fig. 13. Pressure increases owing to degraded primary/secondary heat transfer.

Improper alignment of valves and/or other malfunctions could result in auxiliary feedwater to only one of the four steam generators. Calculations were performed that used the minimum auxiliary feedwater flow required to cool the plant to define the plant response characteristics when only one steam generator received feedwater. The results of these calculations, which are shown in Figs. 14 and 15, indicate that the three steam generators not receiving feedwater were depleted about 1-1/2 h into the transient. At this time, the primary coolant temperatures increased, accompanied by a redistribution of the natural circulation flows between the cooled and uncooled loops. Flow through the cooled loop increased while flow through the other loops decreased. Although the peak hot-leg temperatures were ~15 K greater than calculated for degraded auxiliary feedwater flow distributed equally among the steam generators, the plant could be cooled by natural circulation with degraded auxiliary feedwater flow supplied to only one steam generator.

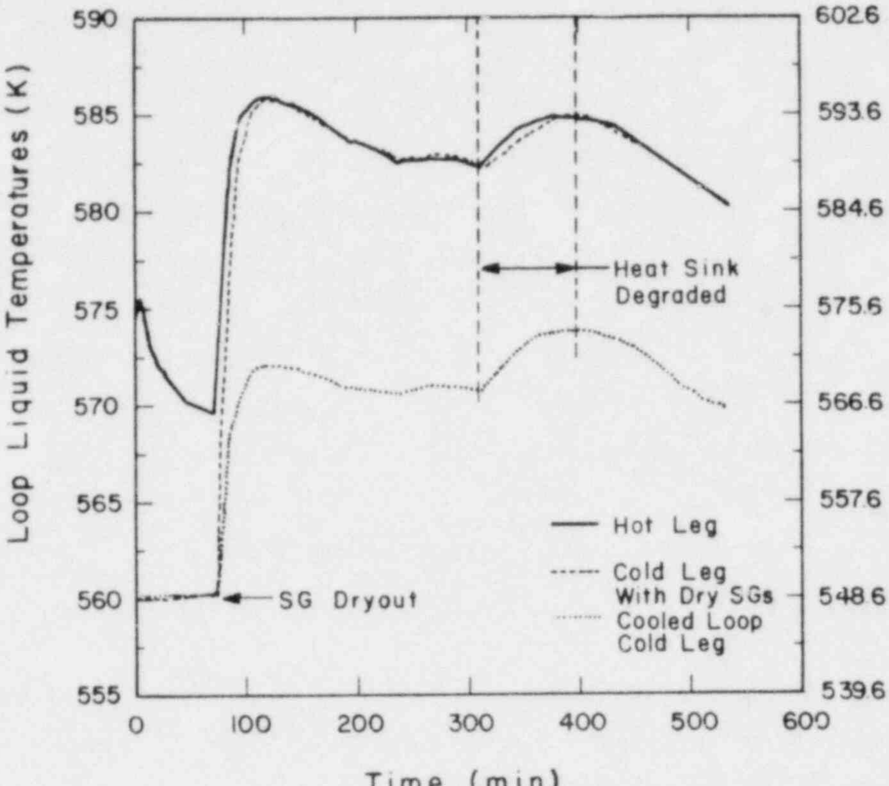

Fig. 14. Temperatures in hot and cold legs increase with only one steam generator being cooled by auxiliary feedwater.

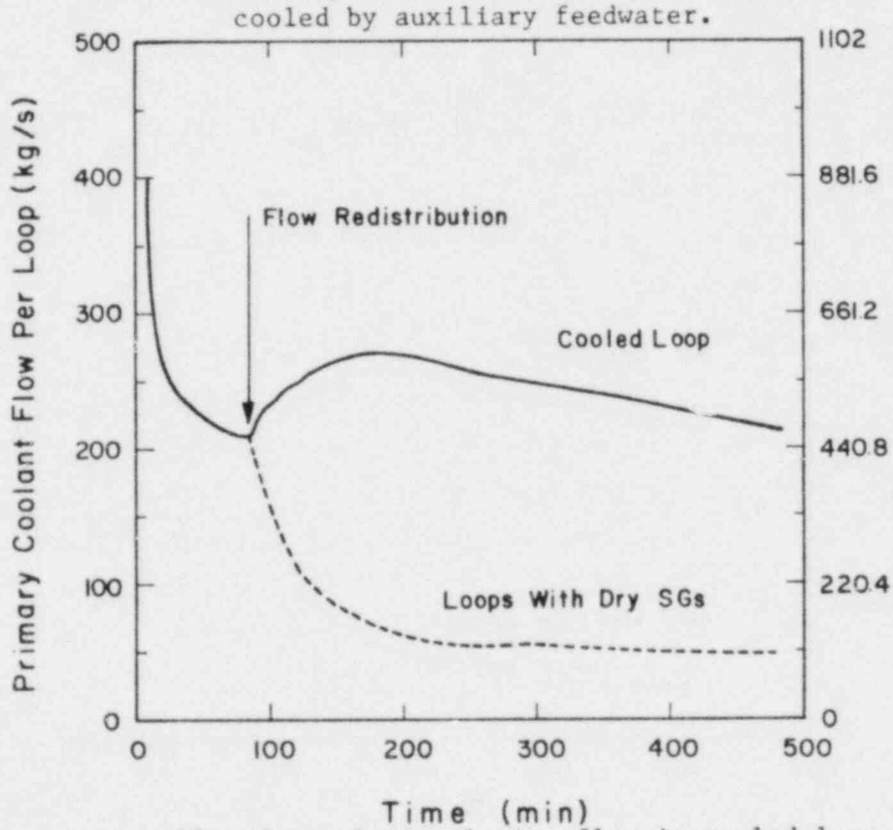

Fig. 15. Natural circulation flow in cooled loop increases after dryout of three steam generators.

Recovery of auxiliary feedwater flow after natural circulation ceased (about 2 h into the transient) was ineffective in cooling the plant owing to the high steam fraction in the upper portions of the primary loops. The effectiveness of introducing full auxiliary feedwater flow just before natural circulation ceased was demonstrated, and the plant response was calculated. Rapid depressurization of the primary (Fig. 16) followed the start of auxiliary feedwater flow to the steam generators. This was accompanied by water falling from the pressurizer and refilling the upper plenum, which had a rapidly increasing vapor fraction (Figs. 17 and 18). The effect on the primary flow rate of the pressurizer emptying to refill the upper plenum is shown in Fig. 19, which also indicates the recovery of natural circulation about 20 min after secondary cooling water was supplied.

B. Failure of One Atmospheric Relief Valve

The effect on plant response of an uncontrolled blowdown of one steam generator through a faulty atmospheric relief valve was investigated to determine whether such an event would affect the timing of critical events in

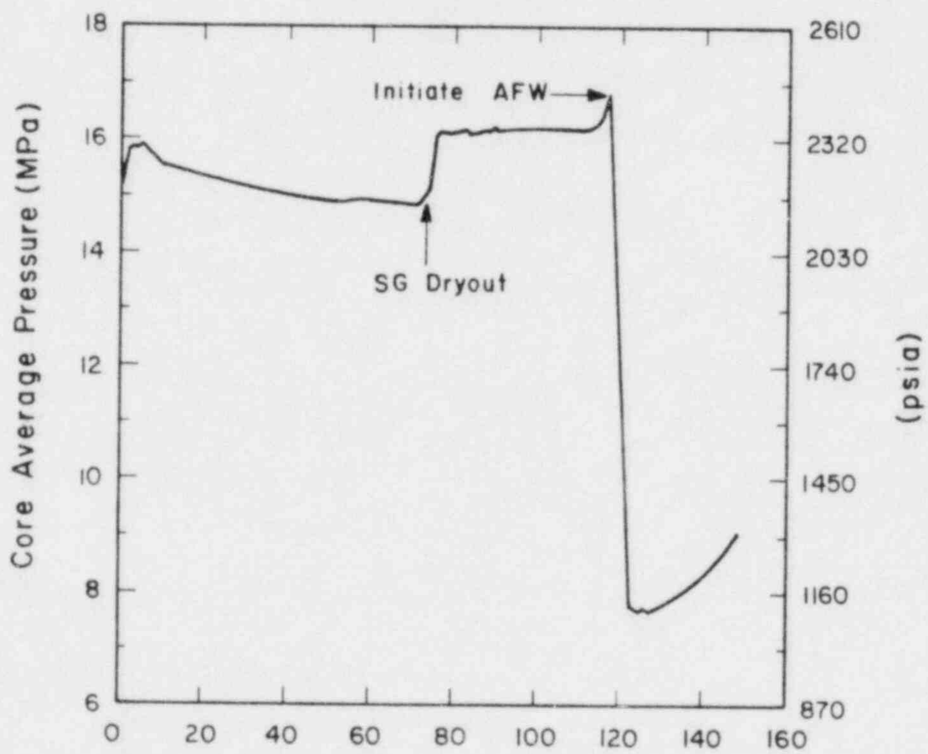

Fig. 16. Delayed introduction of auxiliary feedwater produces rapid depressurization of the primary.

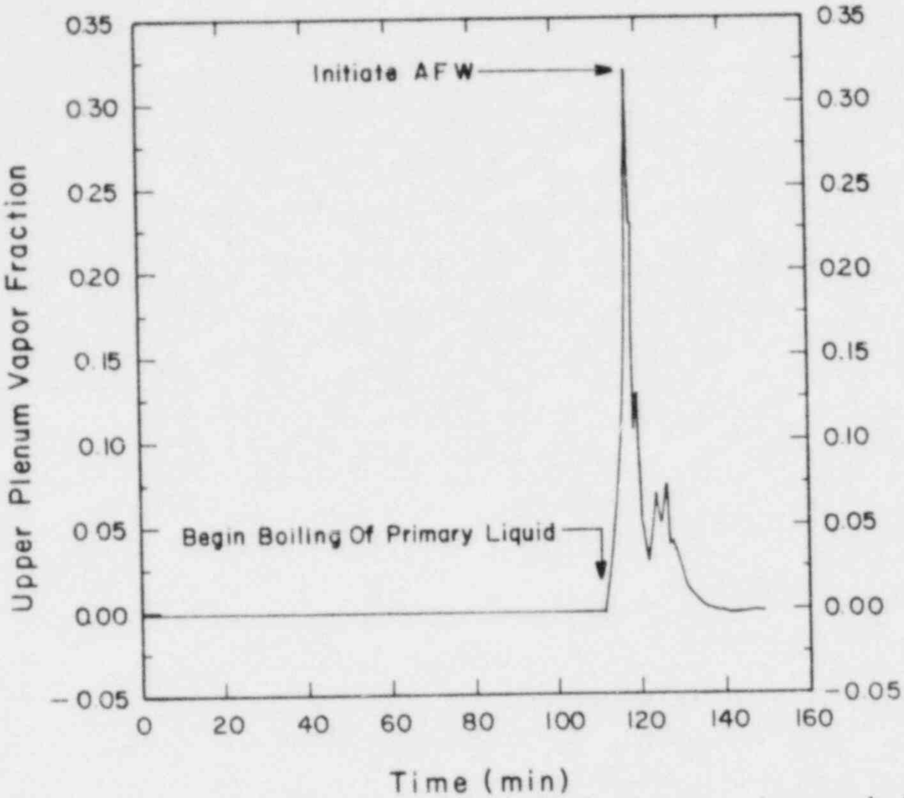

Fig. 17. Restoration of auxiliary feedwater minutes before natural circulation flows cease arrests voiding of upper plenum.

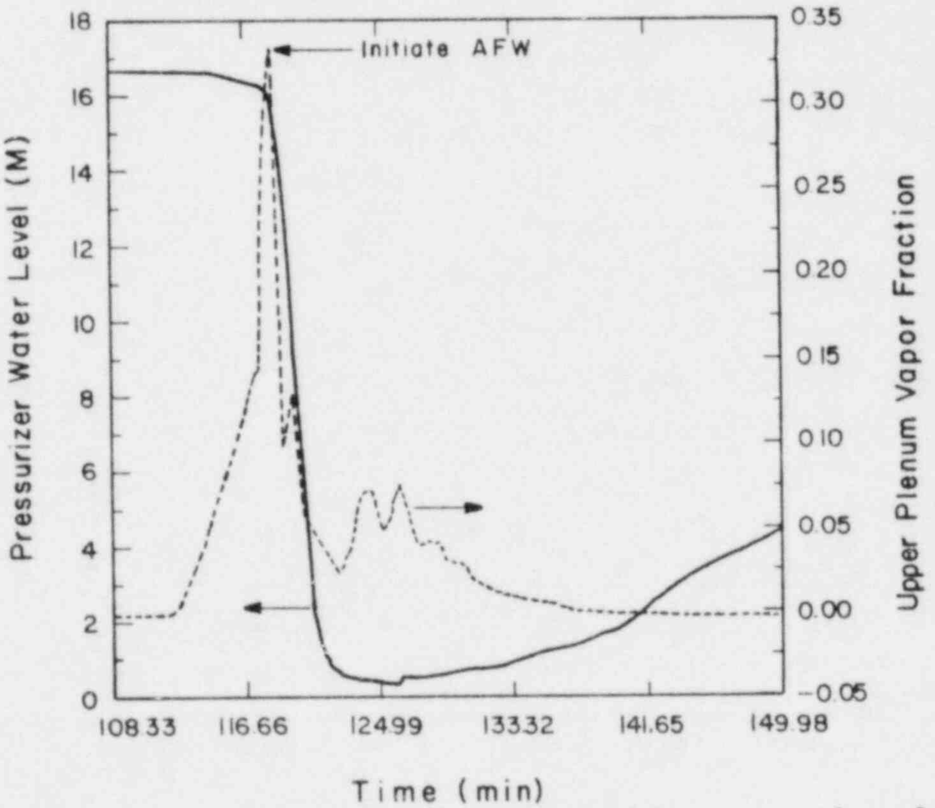

Fig. 18. Rapid depressurization caused by restoration of auxiliary feedwater causes pressurizer to empty refilling the upper plenum.

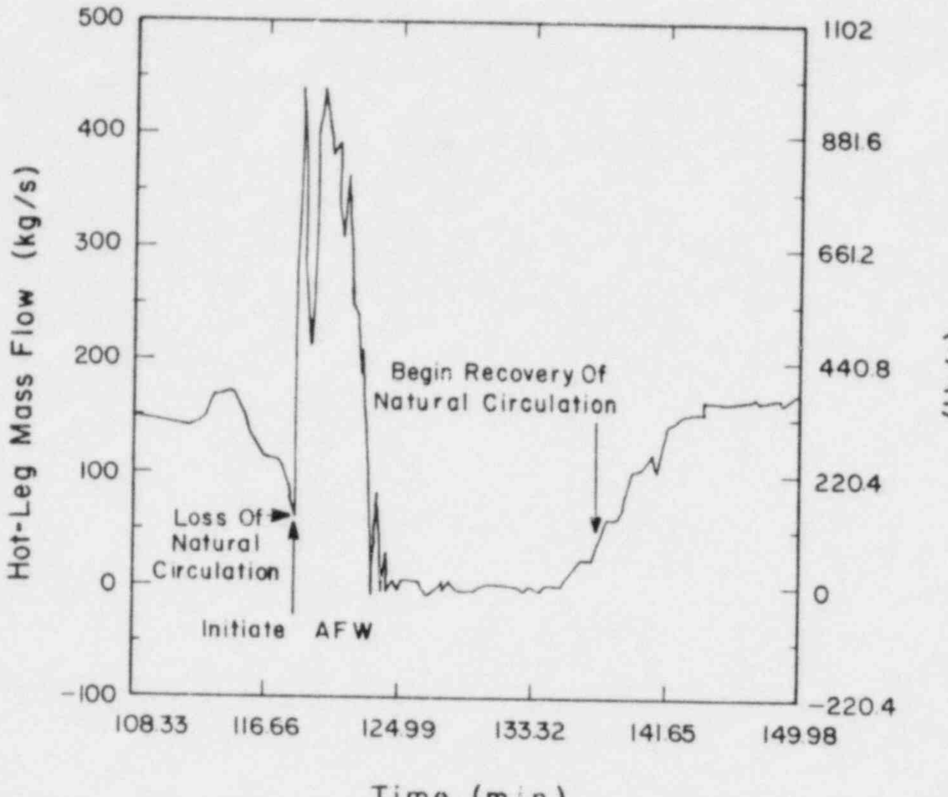

Fig. 19. Primary flow rate shows pressurizer flooding vessel and recovery of natural circulation cooling.

the accident and consequently make recovery more difficult. Calculations in which the relief valve on one steam generator was assumed to fail fully open after receiving the initial signal to open indicated that the affected steam generator blows down in ~10 min, as shown in Fig. 20. This produced decreasing primary coolant temperatures and pressures owing to overcooling during the blowdown. After this steam generator emptied, primary temperatures and pressures rose to nearly normal levels with the primary cooled by natural circulation in the remaining three steam generators. The blowdown of one steam generator is seen in Fig. 21 to have little effect on the time to deplete the secondary inventories of the other three.

C. Failure of One PORV to Reclose After Opening

In the accident sequence described in Sec. IV, the PORVs opened to relieve primary pressure shortly after the steam generator water inventory was depleted. Failure of one of these valves to reclose after opening provides a discharge path for primary coolant during the feedwater transient, thus altering the accident characteristics and instrumentation responses. (Failure to recognize the characteristics of a stuck-open relief valve and subsequent

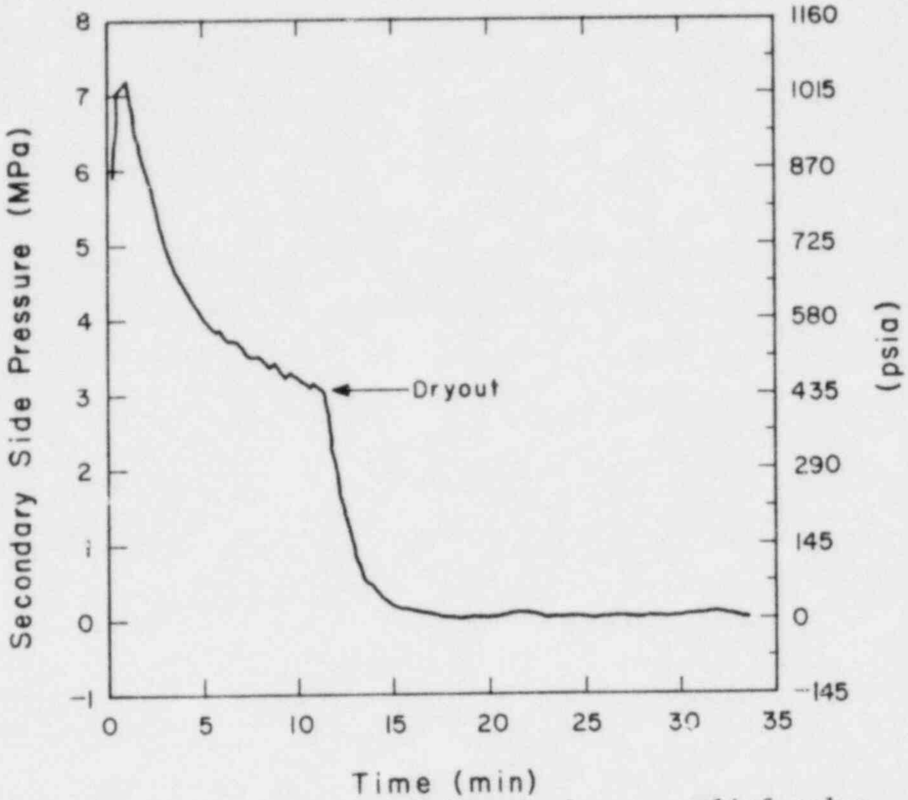

Fig. 20. Steam generator with stuck-open relief valve blows down in about 10 min.

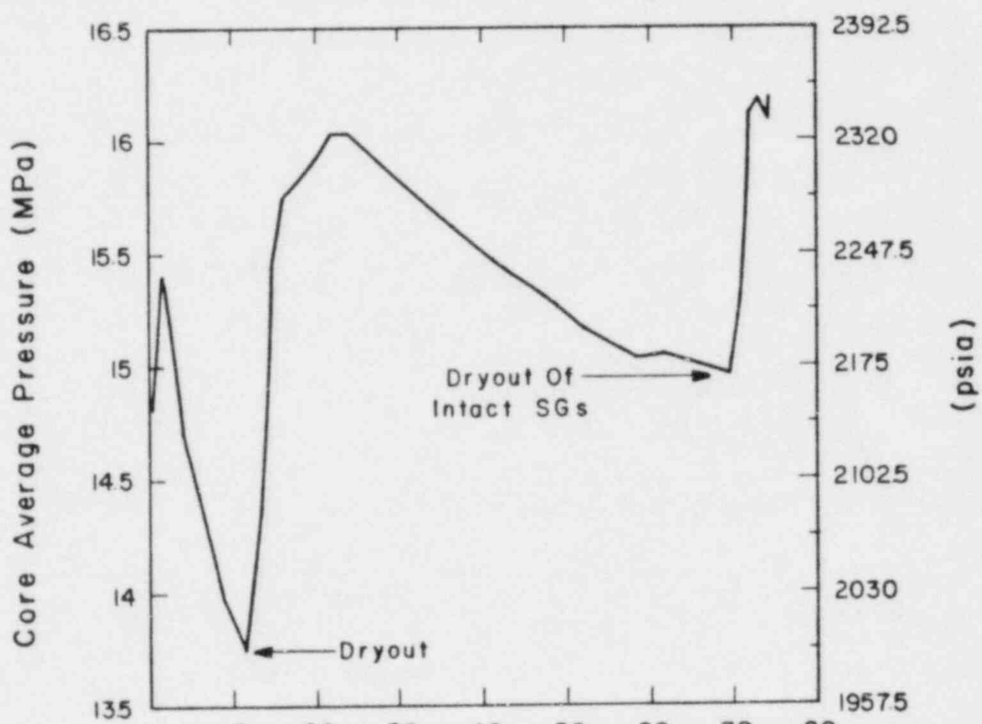

Fig. 21. Blowdown of one steam generator at 10 min has little effect on the dryout time of the other three.

inappropriate intervention in the transient by the operators led to core damage at TMI-2.) Numerical calculations were performed to define the accident signature and evaluate the effectiveness of emergency core cooling to prevent core uncovery.

The automatic plant responses and characteristics of this accident through the time that the PORVs opened (~1.1 h) were identical to those presented in Sec. IV. For this accident scenario, however, one of the two PORVs was assumed to remain fully open, so that the primary pressure began to decrease. The calculated event sequence for this accident is summarized in Table III, and the thermal-hydraulic characteristics are shown in Figs. 22 through 29.

Steam escaping through the PORVs reduced the pressure below the close setpoint, but one PORV remained open and caused the primary pressure to drop below 9 MPa, as shown in Fig. 22. A low pressurizer pressure signal (11.7 MPa) would be obtained, but the coincident low pressurizer level indication required for automatic actuation of emergency core cooling would not occur, so that the ECC System flow would not start at this time.*

The rupture disk on the pressurizer relief tank opened at ~68 min venting steam to the containment. Saturation conditions in the primary coolant were reached after ~72 min as shown in Fig. 23, and emergency core cooling flow was initiated on a containment overpressurization signal at ~79 min. The effect of introducing cold emergency cooling water to reduce the cold-leg temperatures is seen in Fig. 24. The maximum cladding temperature of the average power fuel rods began leveling after about 90 min as shown in Fig. 25. The vessel liquid inventory was at its lowest level at ~80 min and recovered slowly until the calculation terminated at 106 min. Figure 27 shows the cessation of natural circulation flows just before emergency cooling was initiated, at which point the flow rate into the primary (Fig. 28) was balanced by the steam flow out the PORV, so that the decay heat was removed without loss of the primary-system inventory. After about 92 min, the primary pressure and temperature stabilized. The emergency core cooling flow of

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*After the accident at TMI-2, actuation of the ECC System was changed so that the coincident low pressurizer level signal is not required.

TABLE III

EVENT SEQUENCE FOR LOSS-OF-FEEDWATER TRANSIENT WITH ONE PORV STUCK OPEN[a]

Time (s)	Time (min)	Event
0.0		Loss of off-site power trips turbines, reactor cooling pumps, main feedwater pumps, and generates reactor scram signal.
0.6		Control rods drop.
15		Turbine-driven pumps fail to deliver auxiliary feedwater.
30		Motor-driven pumps fail to deliver auxiliary feedwater.
60	1	Atmospheric relief valves or steam lines open.
3800	63	Steam generator secondaries empty of water.
4000	66	PORV setpoint pressure reached; both PORVs open, and one stays open.
4070	68	Low pressurizer pressure signal (< 11.7 MPa), no action.
4100	68	Pressurizer relief tank disk ruptures.
4300	72	Primary system saturates.
4730	79	ECC tripped on high containment pressure (0.13 MPa).
6400	107	End of calculation, system recovering.

[a]Figs. 22 through 29.

~36 kg/s through the vessel and out the PORV was capable of removing 44 MW, while the decay power at this time was 36 MW. Thus, subcooling, followed by refilling and cooldown of the primary system, would continue as long as water could be supplied from the refueling water storage tank.

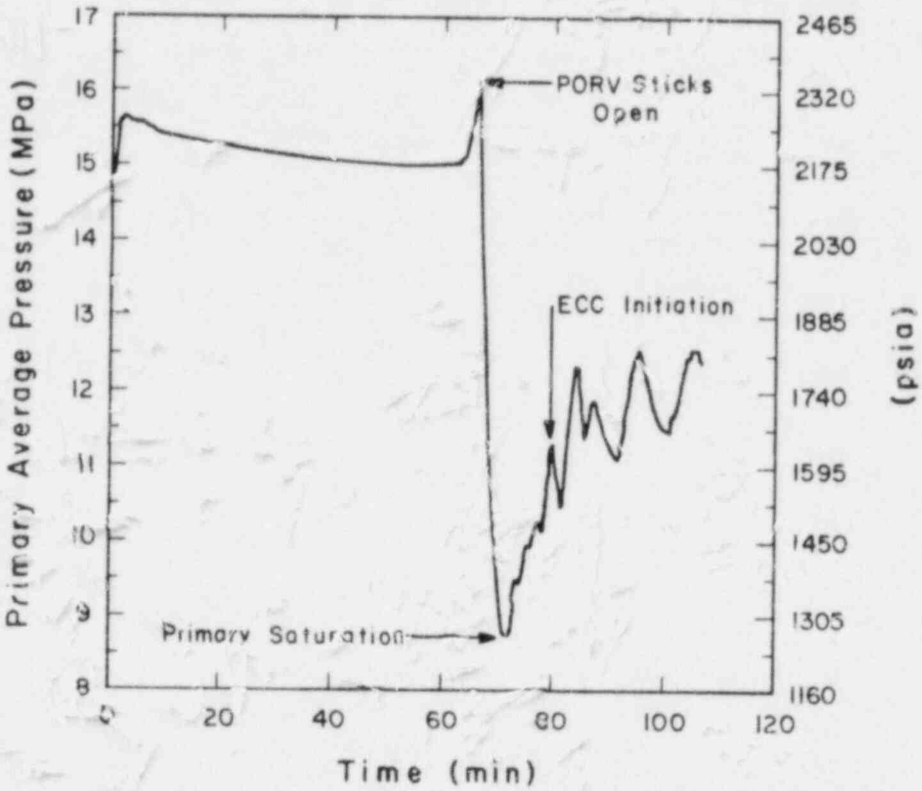

Fig. 22. Primary pressure decreases rapidly after one PORV fails fully open.

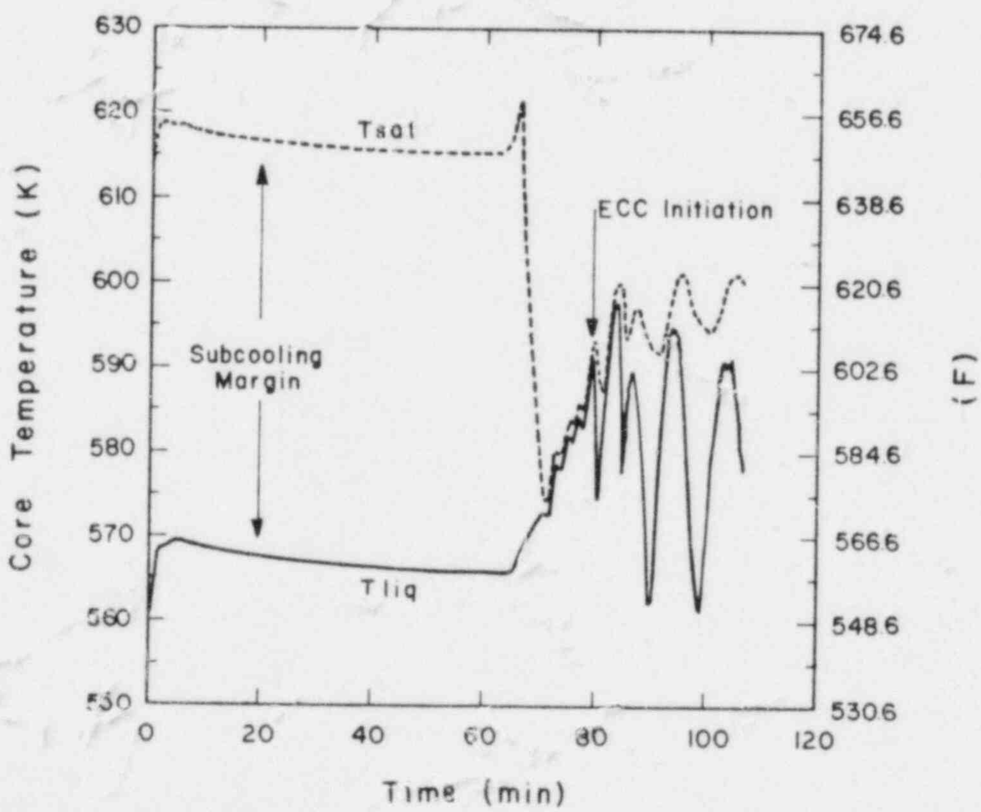

Fig. 23 Subcooling margin is lost before ECC flow is initiated on containment overpressure.

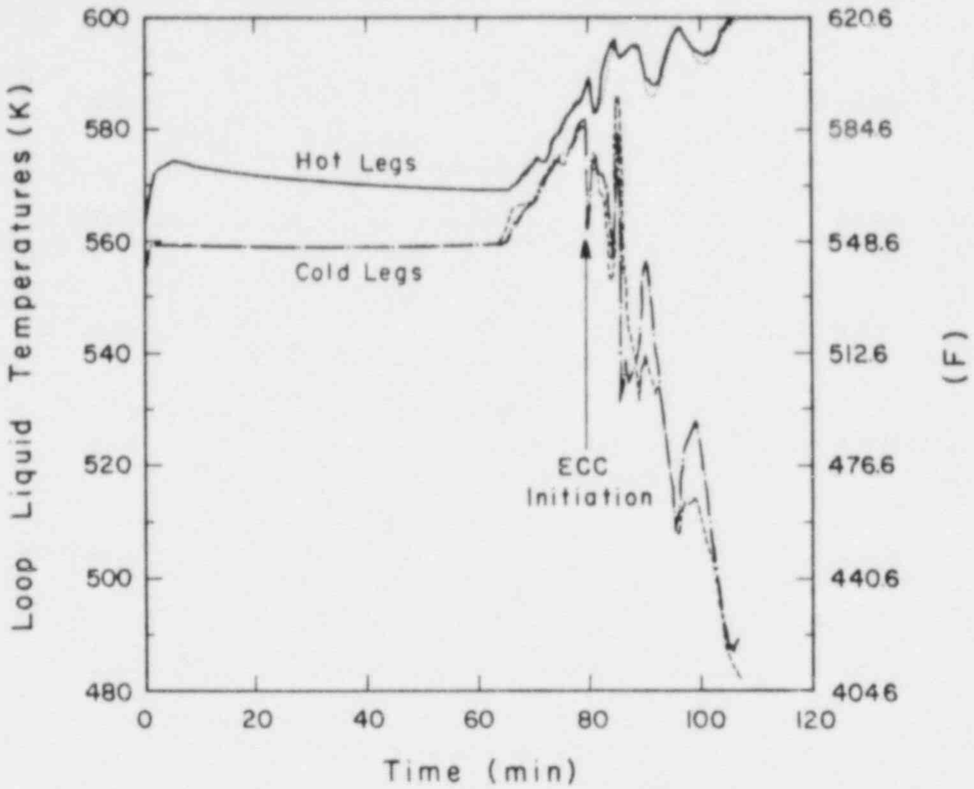

Fig. 24. Temperature difference between hot and cold legs increases after primary saturation and initiation of ECC flow.

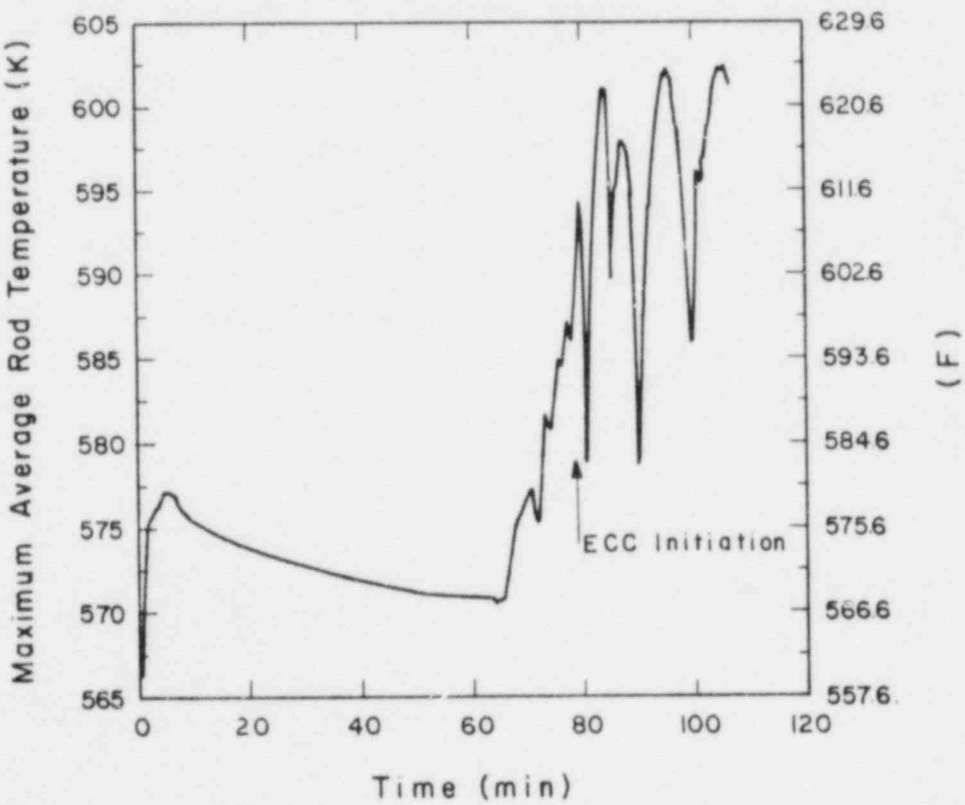

Fig. 25. Cladding temperature increases after steam generator dryout are limited to ~600 K by ECC flow.

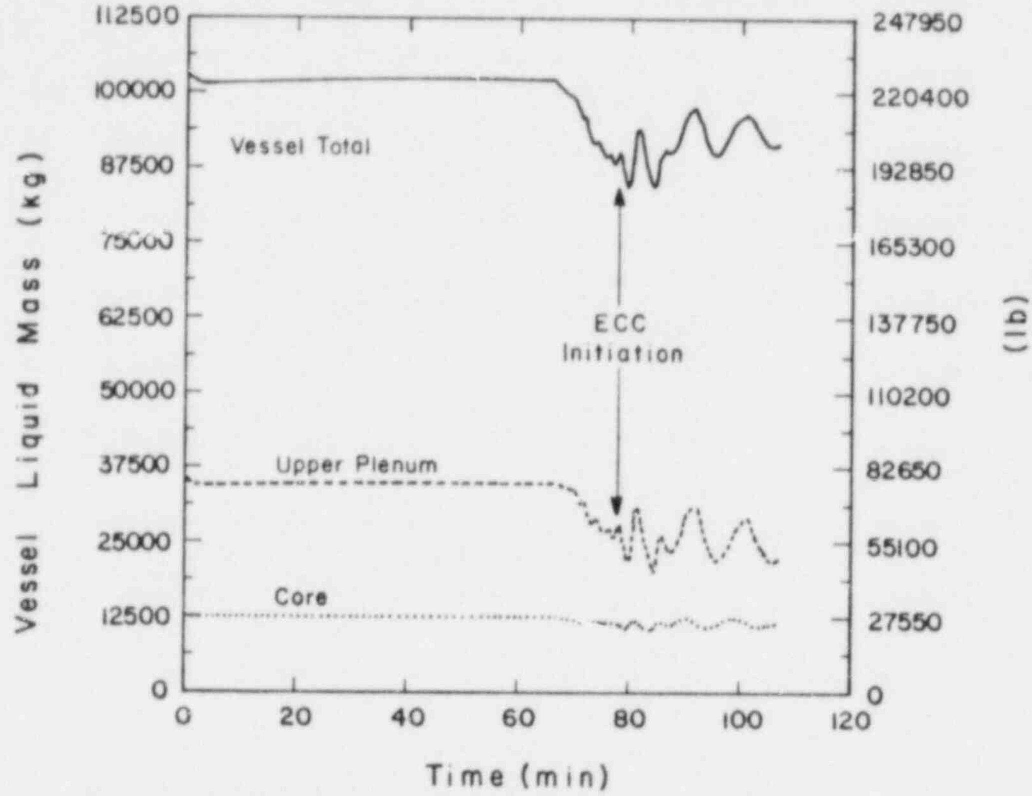

Fig. 26. Initiation of ECC System flow arrests voiding of the reactor vessel.

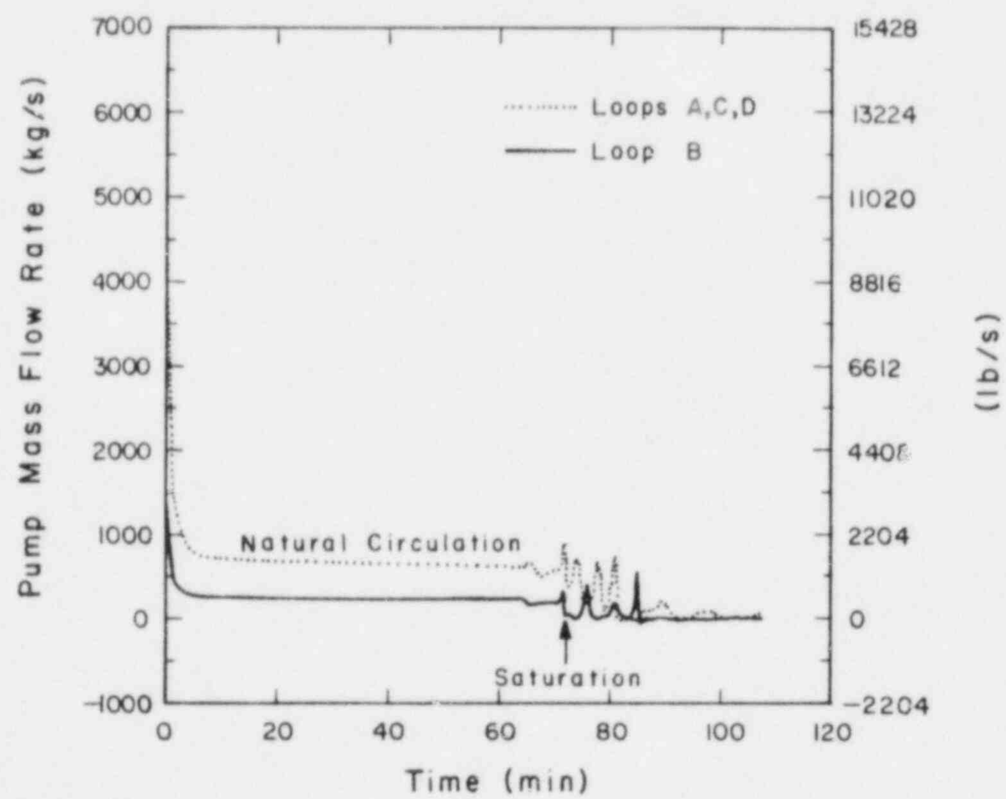

Fig. 27. Natural circulation flows stop shortly after primary saturation occurs.

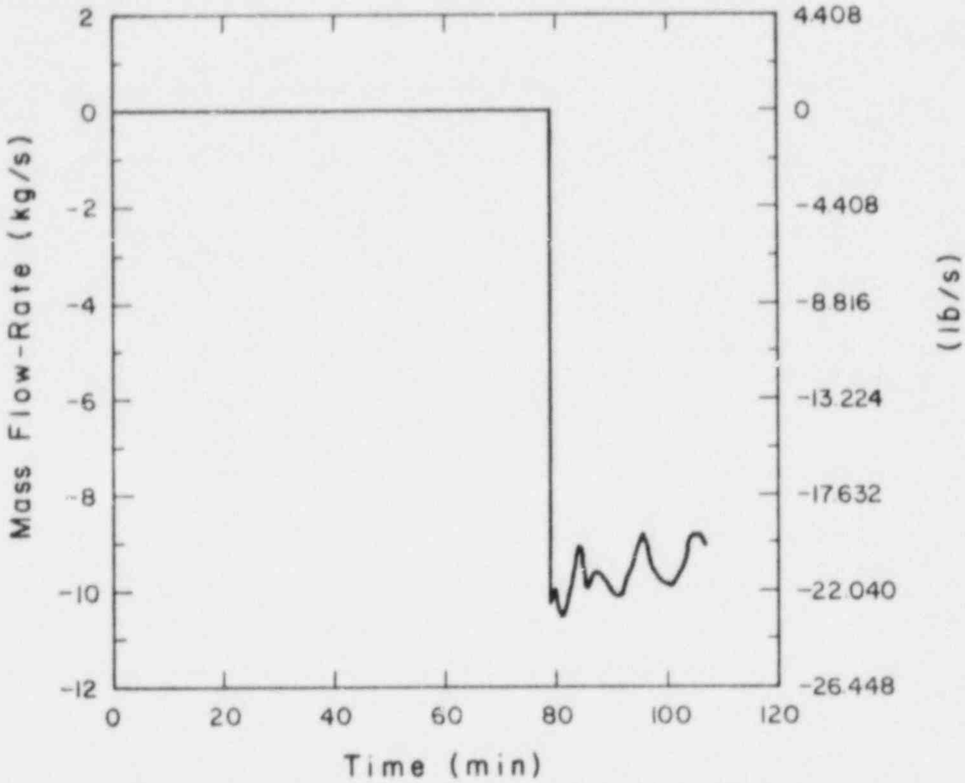

Fig. 28. ECC System flow into each loop.

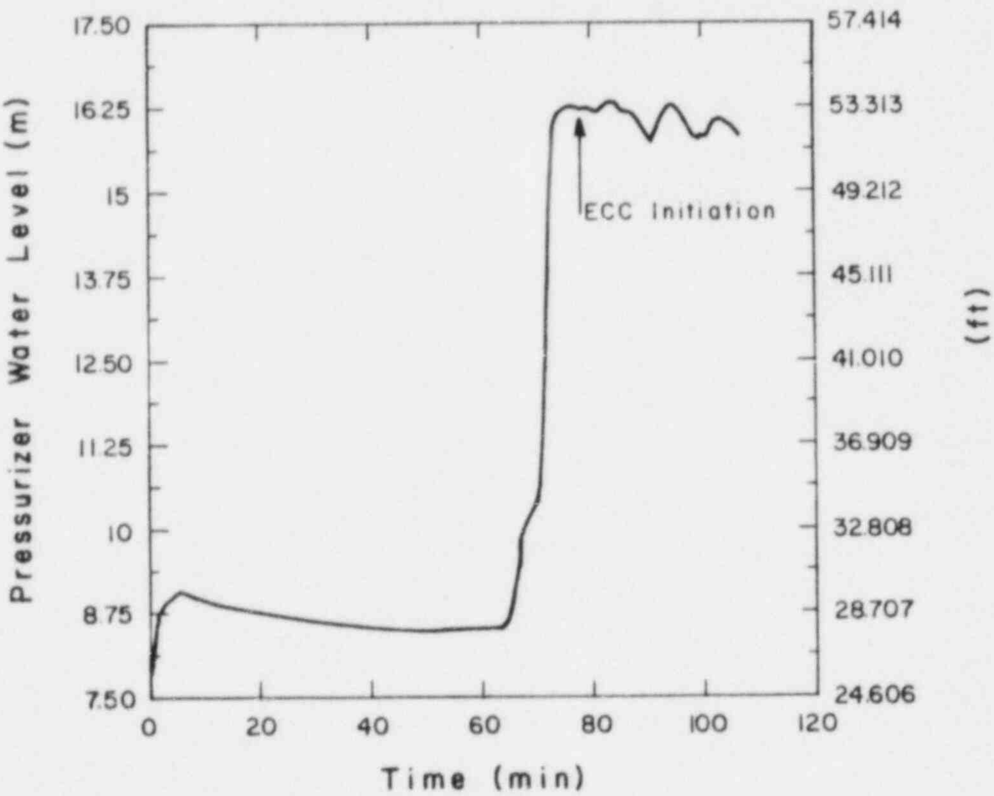

Fig. 29. Pressurizer water level for loss-of-feedwater transient with one PORV stuck open.

D. Failure of Emergency Core Cooling System

A sequence leading to core uncovery and subsequent core damage will occur if the charging pumps fail to deliver emergency cooling at the PORV setpoint pressure. The events in this accident scenario are given in Table IV. This sequence is identical to that discussed in Sec. IV until ~97 min, at which time the ECC failed to perform on demand. Primary coolant saturation (Fig. 30) occurred approximately 12 min earlier (108 min), and by 117 min the core began to uncover (Fig. 31). The water level in the pressurizer is

TABLE IV

EVENT SEQUENCE FOR LOSS-OF-FEEDWATER TRANSIENT WITH FAILURE OF EMERGENCY CORE COOLING[a]

Time (s)	Time (min)	Event
0.0		Loss of off-site power trips turbines, reactor cooling pumps, main feedwater pumps, and generates reactor scram signal.
0.6		Control rods drop.
3800	63	Steam generator secondaries empty of water.
4000	67	PORV opens (primary pressure = 16.1 MPa).
4800	80	Pressurizer solid, PRT rupture disks open.
6400	107	Pressurizer level begins to decrease.
6500	108	Primary coolant saturates, loss of natural circulation.
7000	117	Upper plenum 90% empty, top of core begins to uncover.
7300	122	Upper plenum empty, core 20% empty.
8600	143	Core empty.

[a]Figs. 30 through 32.

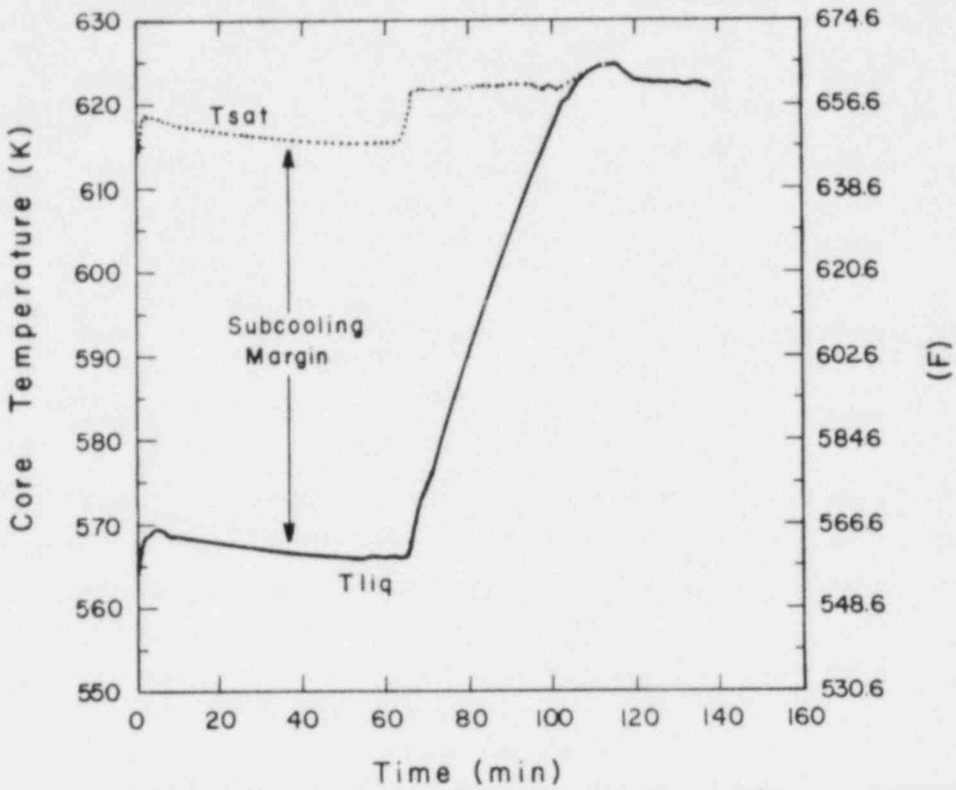

Fig. 30. With failure of ECC System, primary coolant saturates after 108 min.

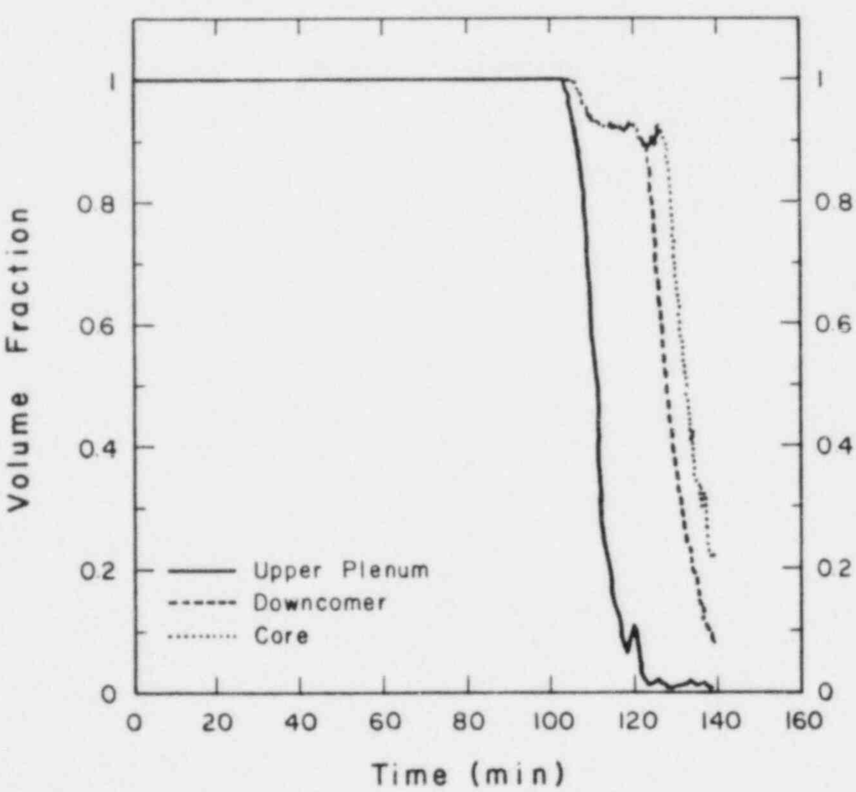

Fig. 31. Liquid volume fractions show vessel empties when ECC is not available.

calculated as a collapsed water level by TRAC. Bubble formation, therefore, gives an indication of a decreased water level at the time of system saturation. At 143 min, the core emptied and the clad rapidly heated (Fig. 32). This defines the maximum time available to effect recovery from loss-of-feedwater accidents before degraded core cooling mechanisms become an important contributor.

VIII. POTENTIAL OPERATOR STRATEGIES FOR ACCIDENT MANAGEMENT

A. Early Manual Initiation of Emergency Core Cooling

In this and following sections, calculations that simulate various options for operator intervention are discussed. The loss-of-feedwater initiator, tripping of the main coolant pumps at the start of the transient, and tripping the reactor 0.6 s into the transient are the same as the nominal sequence. In this section, we discuss two TRAC calculations of operator-initiated ECC flows early in the transient. In the first calculation (Case A), the ECC system was initiated at the start of the transient, but due

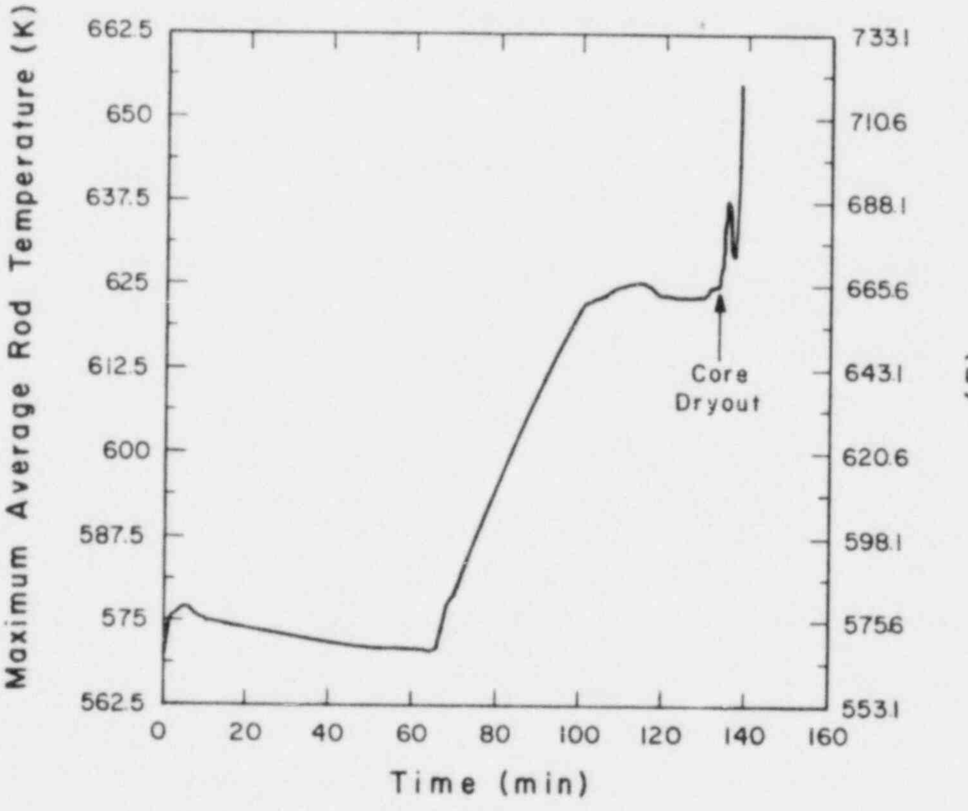

Fig. 32. Cladding temperature increases rapidly as core region empties at 140 min.

to a malfunction, it was able to deliver water only at half the nominal rate. In the second calculation (Case B), the ECC was initiated at 10 min, and delivered water at the nominal rate. These calculations were made to determine the adequacy of half the ECC flow and to evaluate the effect on recovery of early ECC initiation. Of particular interest is whether primary coolant saturation could be averted.

The events of the two TRAC calculations are given in Tables V and VI. The important system parameters for these calculations as plotted in Figs. 33 through 48. are similar. Following the loss-of-feedwater initiator, primary system pressure increased. As is shown in Figs. 33 and 34, pressure increased until the PORV setpoint was reached at 2 s and 10.8 min,

TABLE V

EVENT SEQUENCE FOR LOSS-OF-FEEDWATER TRANSIENT WITH ECC AT HALF CAPACITY INITIATED AT START OF TRANSIENT[a]

Time (s)	Time (min)	Event
0.0		Loss of feedwater, main coolant pumps (MCPs) tripped, ECC at half capacity initiated manually.
0.6		Reactor trip.
2		PORV setpoint pressure (16.1 MPa).
2500	42	Pressurizer solid.
4600	77	Steam generator secondary dryout.
8300	138	Primary system saturates.
12000	200	End of calculation, ECC insufficient to prevent core dryout.
21000	350	Estimated time of core dryout.

[a]Figs. 33, 35, 37, 39, and 41.

TABLE VI

EVENT SEQUENCE FOR LOSS-OF-FEEDWATER TRANSIENT WITH FULL ECC INITIATED AT 10 min[a]

Time (s)	Time (min)	Event
0.0		Loss of feedwater, MCPs tripped.
0.6		Reactor trip.
600	10	ECC initiated manually.
650	10.8	PORV setpoint pressure (16.1 MPa).
2000	33.3	Pressurizer solid.
5700	95.0	Steam generator secondary dryout.
11500	191.6	End of calculation, system saturated, recovery will occur.
~15000	~250	Decay power at a level that is removable by ECC flow.

[a]Figs.34, 36, 38, 40 and 42.

respectively, for Cases A and B. The initiation of ECC resulted in the pressure increase seen in the figures. Figures 35 and 36, which display the pressurizer water level for the two calculations, show that ECC flow filled the primary system at 42 and 33 min, respectively. With the primary system completely filled with water, liquid was discharged through the PORVs. Decay power could not be removed by ECC flow alone. The steam generator secondary water thus boiled off until dryout occurred at 77 and 95 min, respectively. This loss of heat sink resulted in an increase in primary system temperatures depicted by Figs. 37 and 38. In both cases, the temperature increased to the saturation temperature corresponding to the PORV setpont pressure (622 K). The rate of temperature increase was lower in Case B because ECC flow was double that of Case A, and power had decayed to a lower level at the time of steam generator secondary dryout. When saturation occurred, the primary

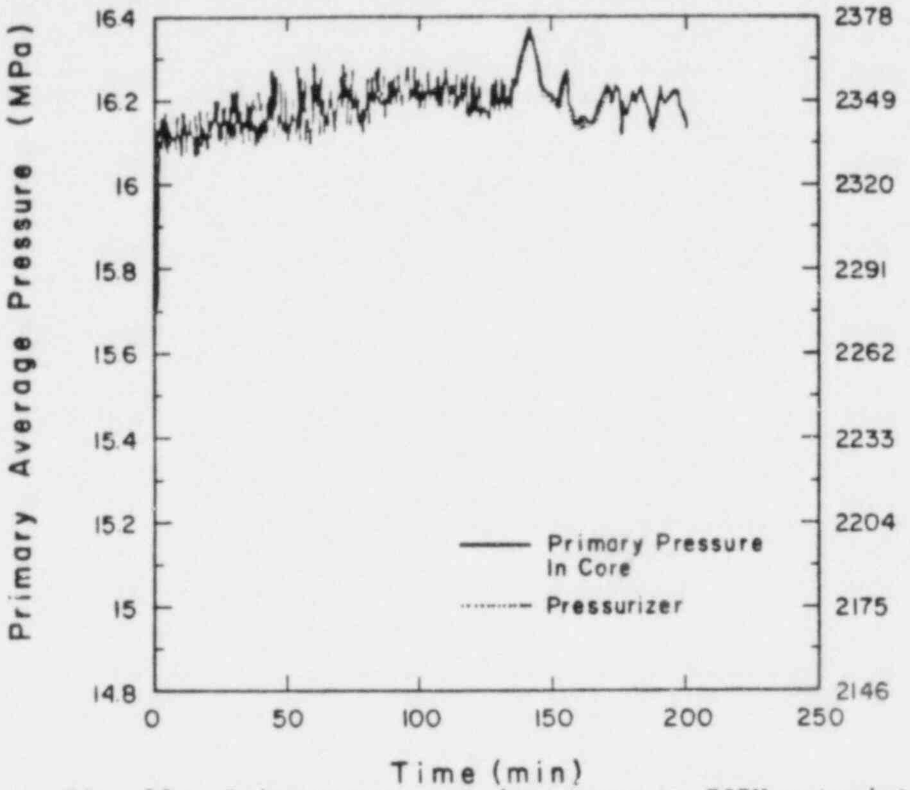

Fig. 33. Primary pressure increases to PORV setpoint and remains there when ECC System is initiated at start of transient.

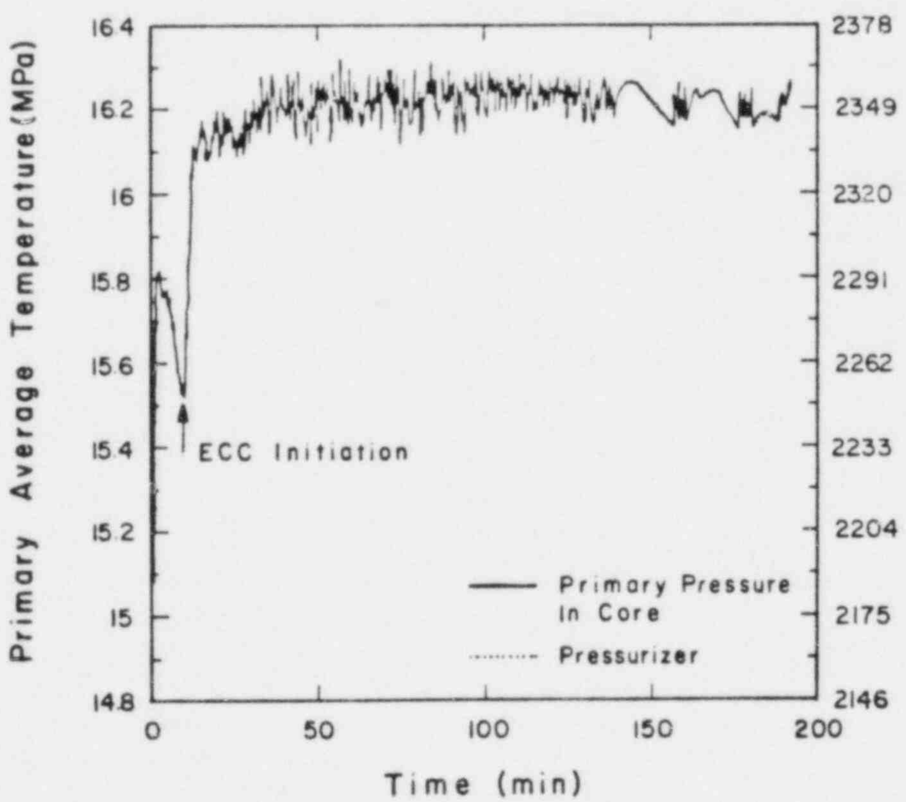

Fig. 34. Primary pressure increases to PORV setpoint when ECC System is initiated at 10 min.

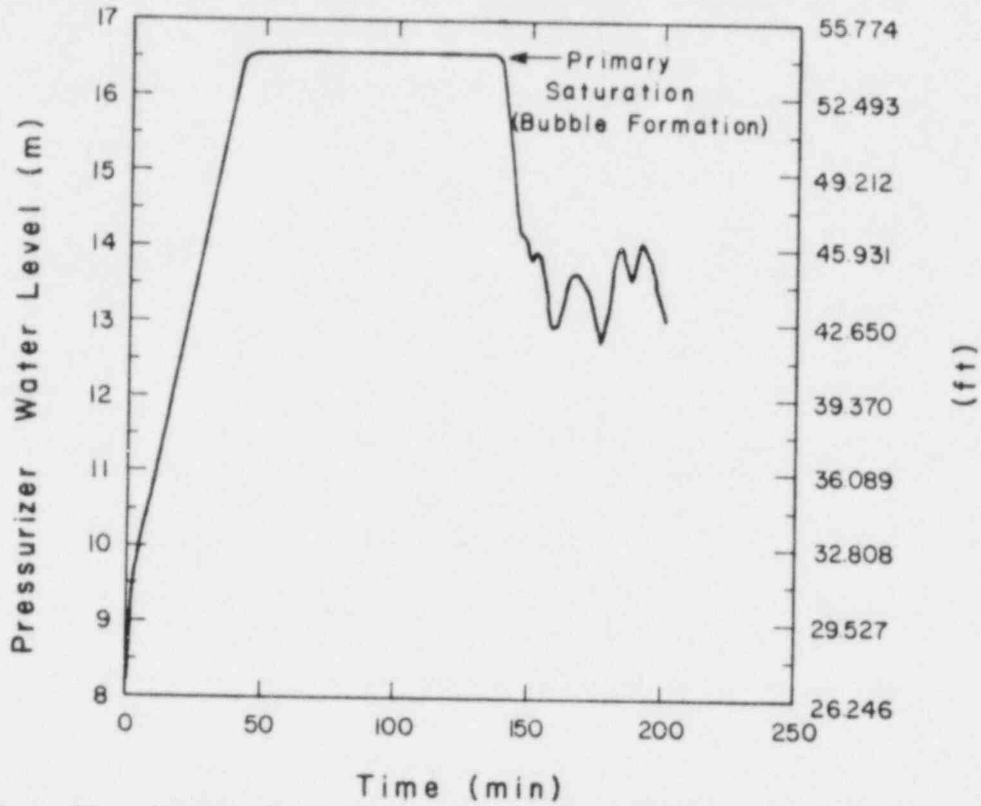

Fig. 35. Pressurizer fills when ECC is initiated at start of transient; drop occurs when primary saturates at 138 min.

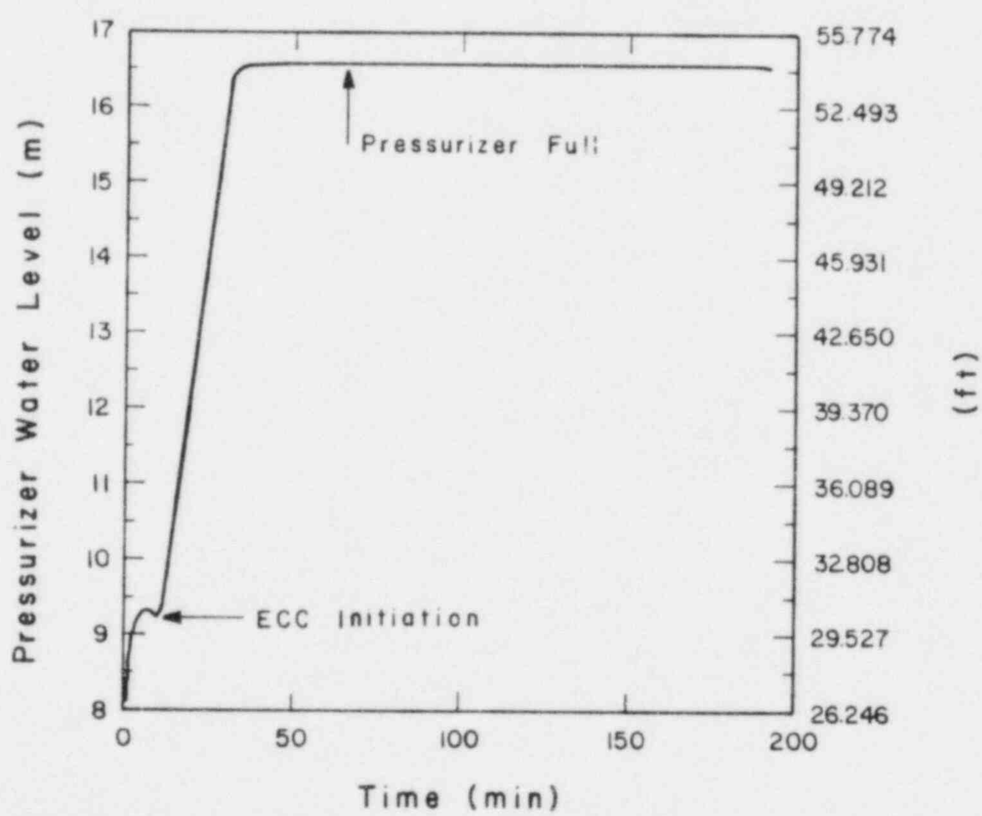

Fig. 36. Pressurizer fills when ECC system is initiated at 10 min.

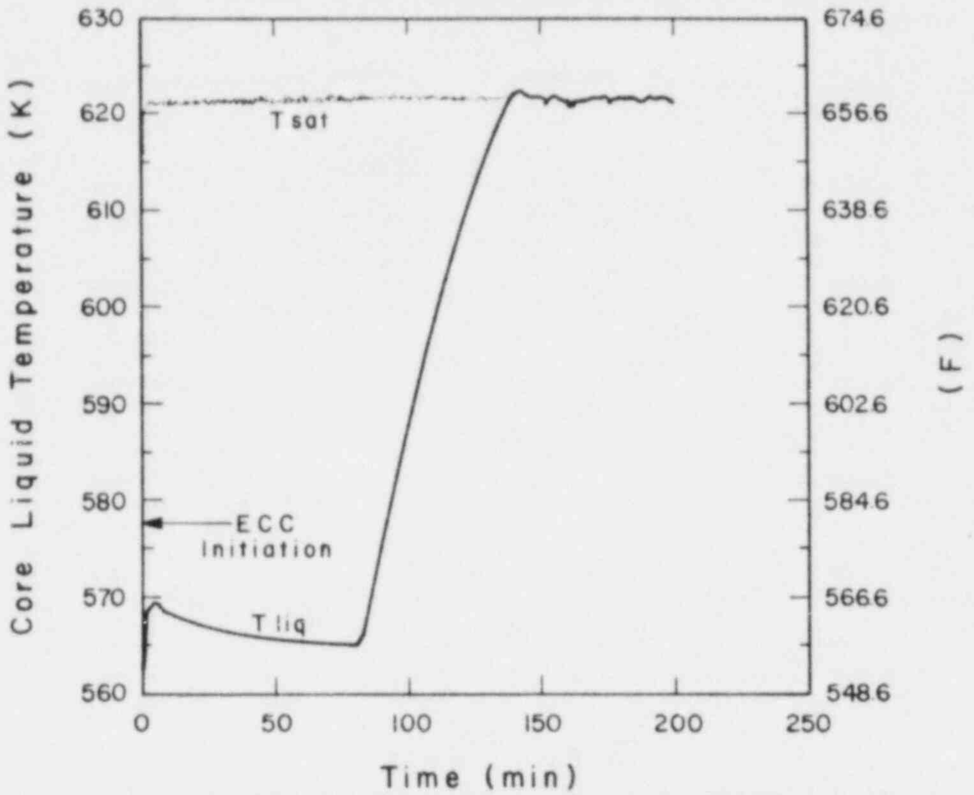

Fig. 37. Liquid temperature in the core increases to saturation after steam generator dryout at 77 min.

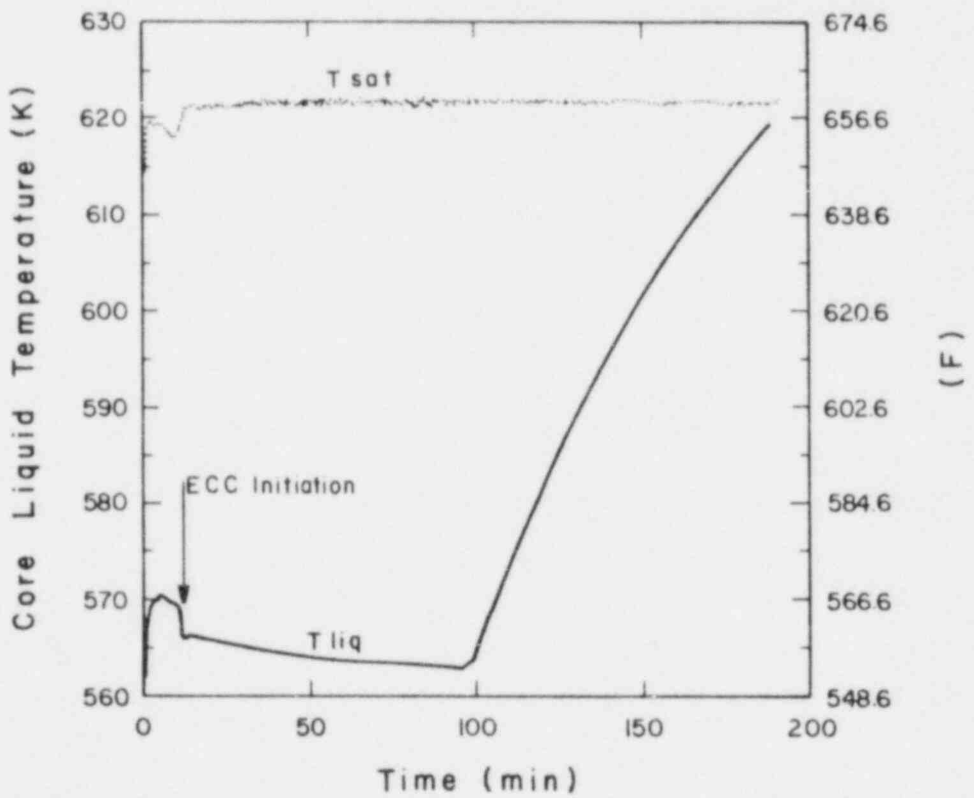

Fig. 38. Liquid temperature approaches saturation at 195 min with ECC system initiated at 10 min.

coolant expanded more rapidly. This was manifested in the increased PORV flow rates at 138 min and 191 min in Figs. 39 and 40. This flow was sufficient to accommodate the expansion. At the end of Case A, ECC flow through the vessel and out the PORVs removed 15.7 MW, including both the increase in sensible heat and boiling of ECC water. The decay power was 30 MW. The remaining 14.3 MW was removed by boiling away the primary coolant inventory. The liquid inventory in the vessel for Case A is given in Fig. 41. Flow through the PORVs was 22.6 kg/s, thus 15.9 kg/s was lost from the system. Approximately 22 h is required for the power to decay sufficiently to be removed by ECC injection. During that period, the core would boil dry and another temperature excursion would follow. Thus, recovery could not occur with the ECC at half capacity, even if initiated at the start of the transient. At the end of the Case B calculation, primary system saturation had occurred; ECC flow of 12.5 kg/s was capable of removing 29.4 MW of the 30.5-MW decay power produced. The condition of the system in Case B was similar to that of the nominal scenario described in Sec. IV. Primary system inventory boiling (see Fig. 42) was necessary until the decay power decreased to 29.4 MW. As is indicated in Table VI, this occurred at approximately 250 min. Therefore, while the system will recover, saturation could not be avoided even with ECC initiation early in the transient.

B. Primary Depressurization Using PORVs

If the ECC flow were started before 100 min, the core would have remained covered and recovery would be possible. Primary fluid boiling would be necessary in this situation for approximately 2-1/2 h, but the primary inventory would be sufficient to prevent core uncovery. A delay of more than 100 min (but less than 133 min) in initiating ECC flow would not prevent core uncovery, but it would be sufficient to allow recovery without core damage. Vapor generation would adequately cool the core during the partial uncovery. If the ECC were unavailable before approximately 133 min, the core would uncover sufficiently to allow heatup of the clad and subsequent damage unless additional action were taken. If feedwater were recovered before the primary loops emptied, a rapid depressurization of the primary would occur with subsequent reflooding of the primary from the pressurizer and with continuation of natural circulation. After the loops emptied, however, feedwater recovery would be ineffective. Therefore, initiation of secondary

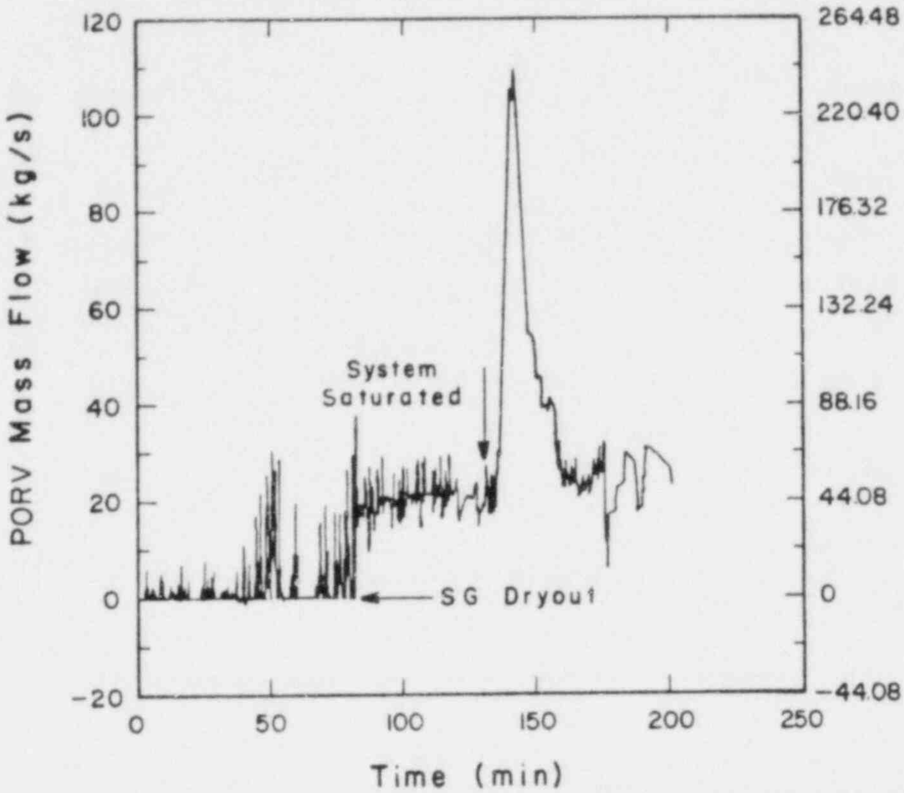

Fig. 39. Flow out the PORV increases due to system expansion after steam generator drys out at 77 min.

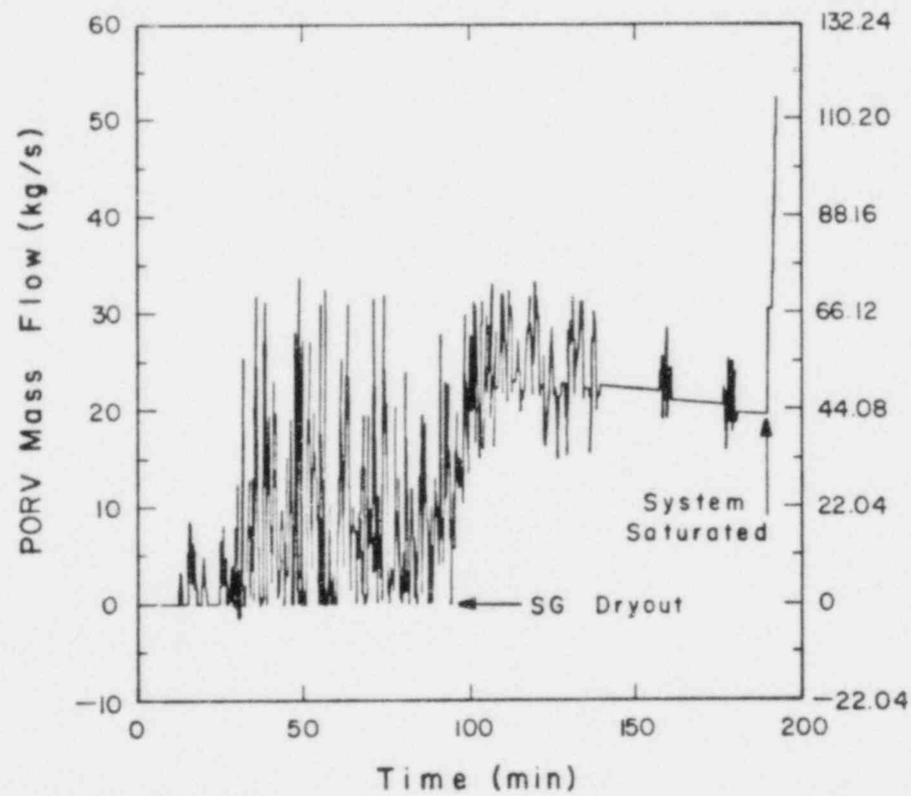

Fig. 40. Flow out PORV increases due to system expansion after steam generator dryout at 95 min.

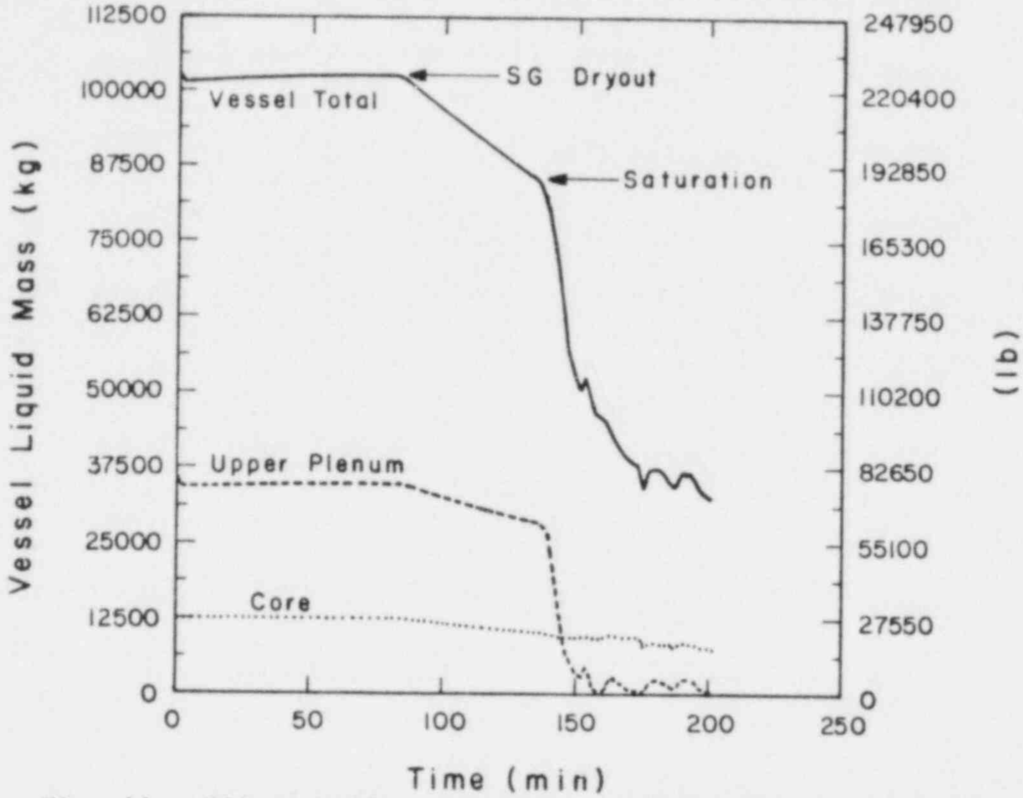

Fig. 41. ECC at half capacity is insufficient to maintain vessel inventory.

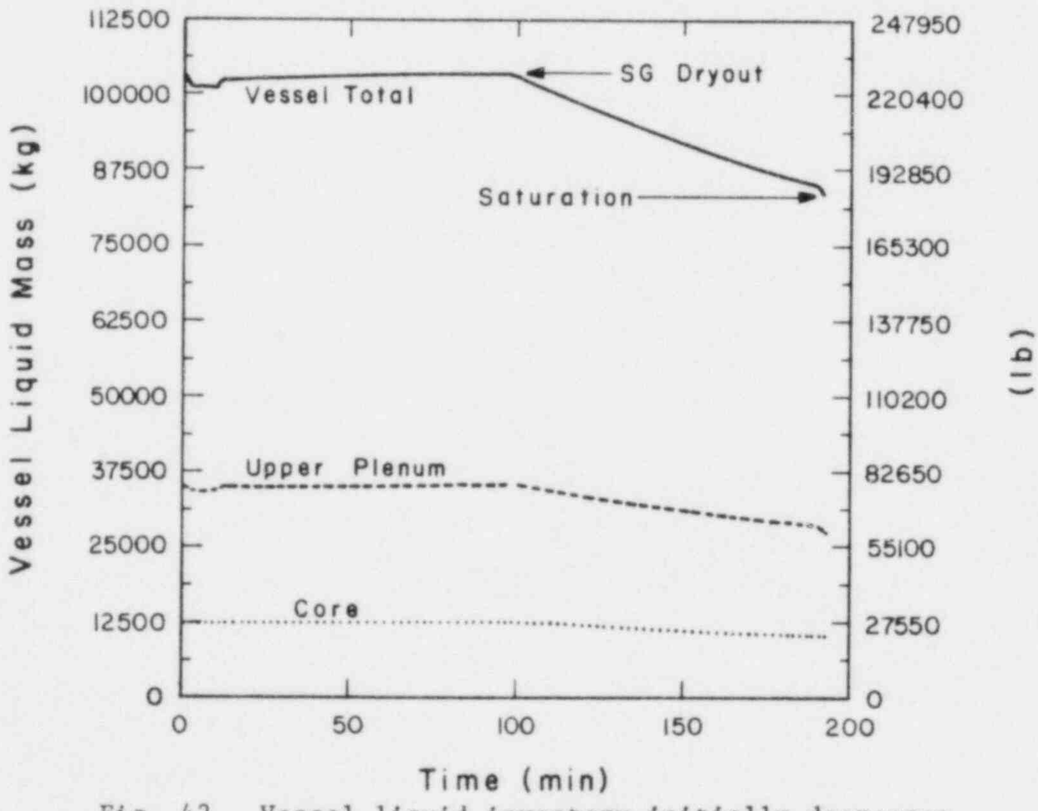

Fig. 42. Vessel liquid inventory initially decreases with ECC initiated at 10 min.

cooling before loop voiding (around 117 min) or full ECC injection before 133 min is necessary to prevent core overheating.

If the PORVs were held fully open, depressurization of the primary would begin (Fig. 43) causing a substantial increase in the ECC flow (Fig. 44). This large ECC flow could cool the core and prevent core damage.

An operator action that seems contrary to recovery is that of creating an opening in the primary pressure boundary. This action might be necessary if the pressurizer water level (Fig. 45) were to drop below its steady-state value (approximately 10 m) concurrent with high system pressure before ECC initiation. After that time, recovery by any other means would be ineffective in preventing core damage. To verify these findings, we calculated system response to holding the PORVs open when they reached the fully open position at 127 min. The events for this case are shown in Table VII. Recovery began at 130 min (Figs. 46 and 47) as indicated by the vessel filling and the decreasing clad temperature.

C. Bleed-and-Feed Scenarios

In the previous sections, the ability to maintain hot shutdown conditions using the ECC system was demonstrated. The centrifugal charging pumps (CCPs) were assumed to deliver water against the pressure determined by relief-valve setpoints. The analyses showed that CCP flow rates at relief-valve setpoint pressures were insufficient to prevent steam generator secondary dryout and primary coolant saturation. Recovery was possible for Zion-1 in this "feed-and-bleed" mode if both CCPs and their associated piping and valves functioned as designed. Many plants lack the high-pressure delivery capacity to prevent boiling in the core region. In this section, a different strategy termed "bleed-and-feed" is discussed. In this mode, relief valves are operated to reduce system pressure and thus increase ECC output.

The TRAC model is similar to that used in the previous sections. The only difference is that the ECC model used here includes only the safety injection pumps (SIP). This part of the ECC system would deliver water at a system pressure of 10.7 MPa. This model was used to generalize the results to PWRs other than Zion-1 and to evaluate the situation at Zion if the CCPs were unavailable. Four bleed-and-feed scenarios, consisting of increasingly complex levels of operator intervention, were analyzed using TRAC. In the first scenario (Case A), the PORVs were opened and the SIPs were initiated

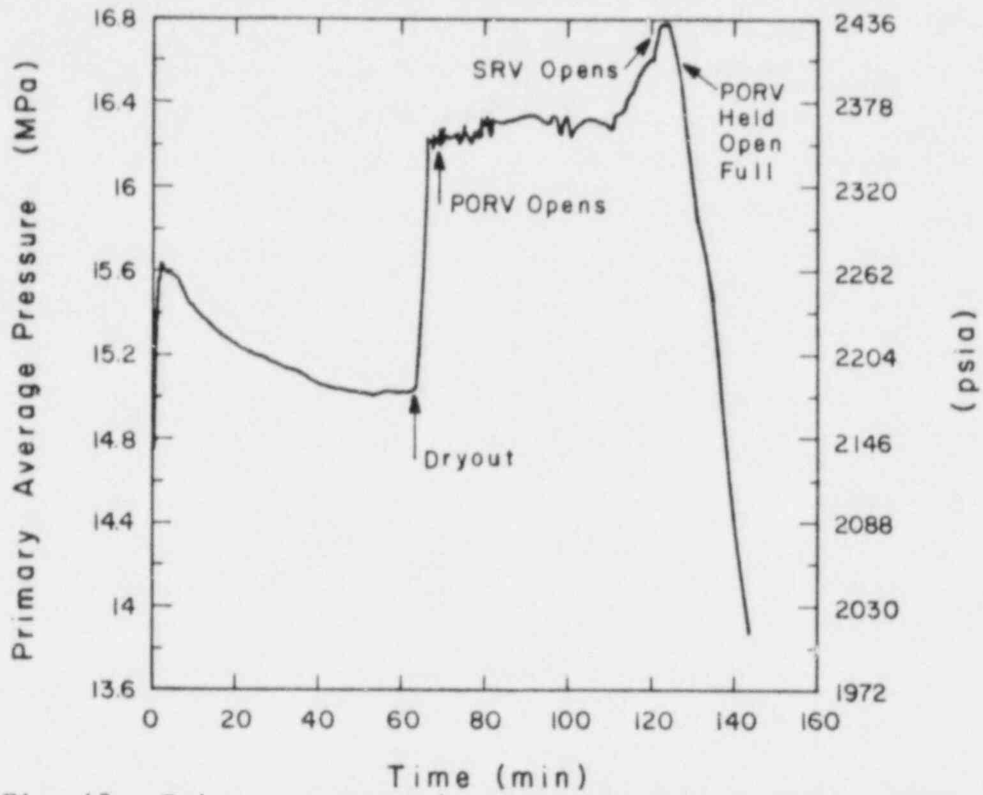

Fig. 43. Primary pressure increases to PORV setpoint, then decreases when PORVs are held open.

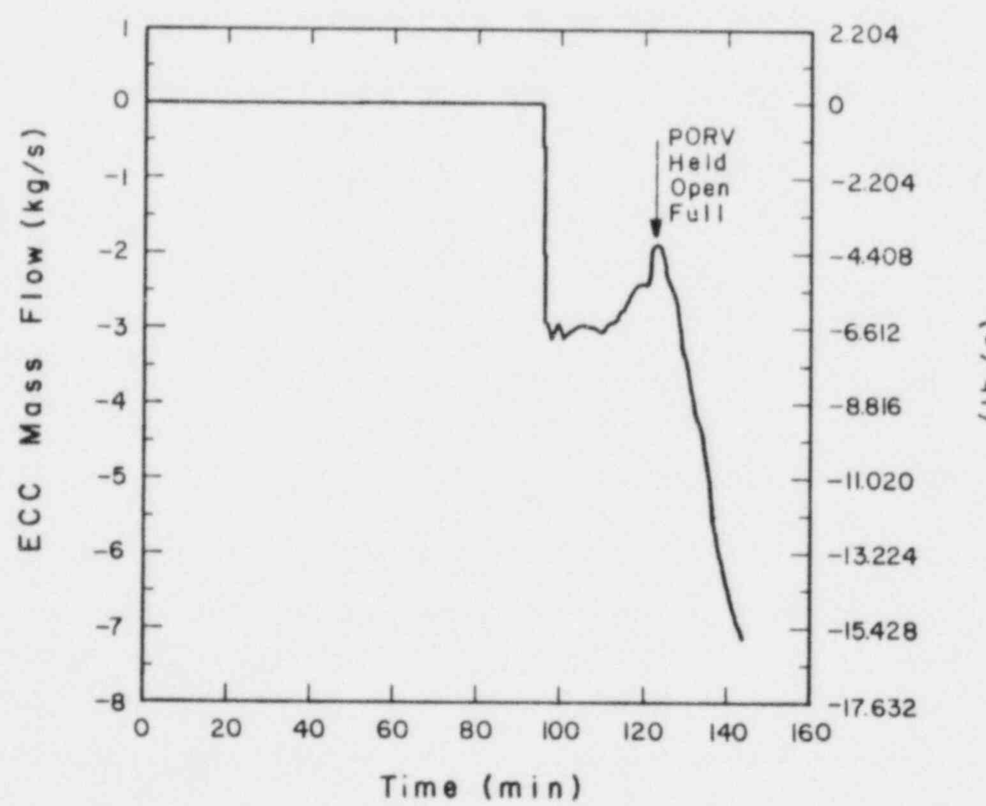

Fig. 44. ECC flow into each loop increases when PORVs are held open at 126 min.

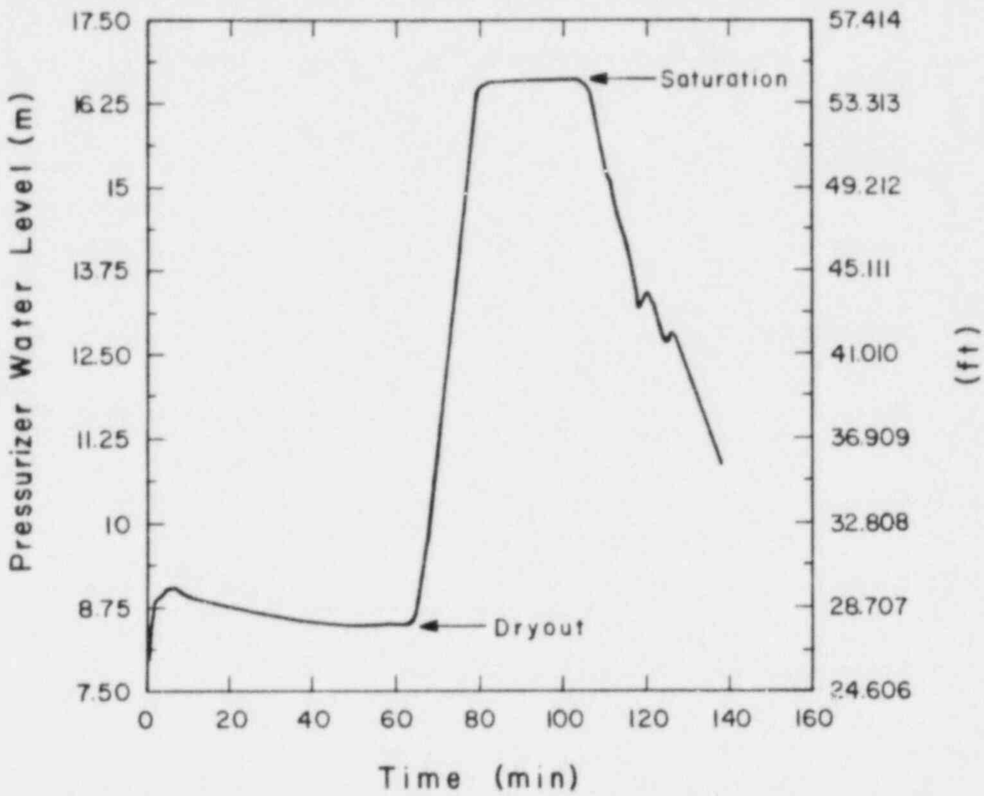

Fig. 45. Pressurizer water level increases after steam generators dry out, then decreases following saturation.

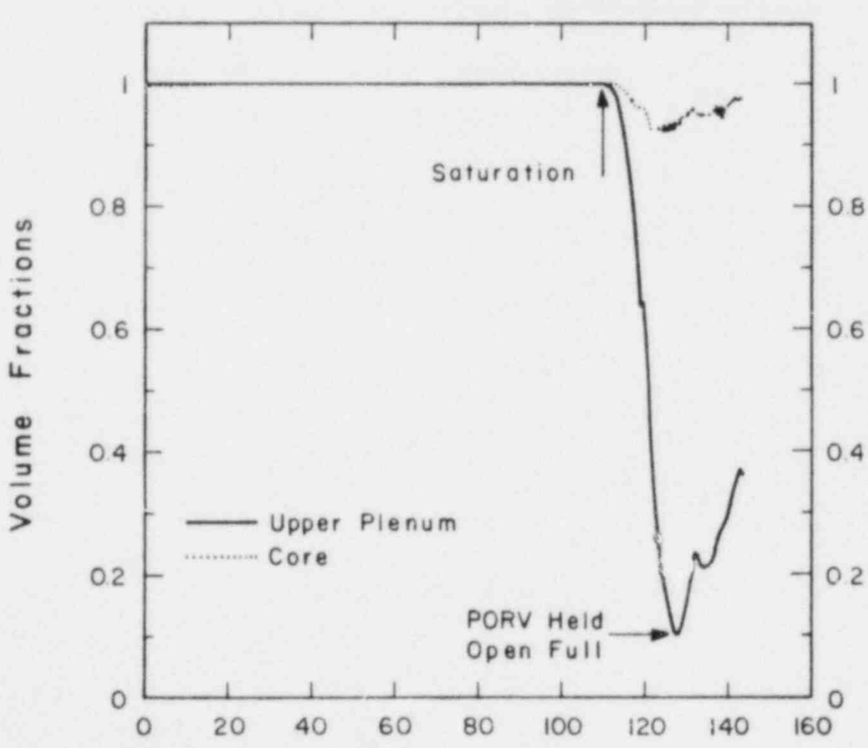

Fig. 46. Liquid fractions show vessel refilling when PORV is held fully open.

TABLE VII

EVENT SEQUENCE FOR LOSS-OF-FEEDWATER TRANSIENT WITH PORVs HELD FULLY OPEN[a]

Time (s)	Time (min)	Event
0.0		Loss of feedwater, MCPs tripped.
0.6		Reactor scram.
3800	63	Steam generator secondary side dryout.
4000	66	PORV opens, system pressure = 16.1 MPa.
4800	80	Pressurizer solid, PRT rupture disks open.
5800	97	ECC tripped on high containment pressure (0.13 MPa) (Initial ECC flow = 12.8 kg/s).
6800	133	Pressurizer level begins to decrease.
7200	120	System saturates (p = 16.8 MPa), loss of natural circulation.
7300	122	Peak clad temperature of 625 K reached.
7600	127	Upper plenum 90% empty, top of core begins to uncover, PORVs reach full open and are held open.
7800	130	Recovery begins, primary pressure dropping rapidly.

[a]Figs. 43 through 47.

10 min after the start of the transient. The event sequence is given in Table VIII; the important thermal-hydraulic parameters are plotted in Figs. 48 through 53. The system behavior until 10 min was identical to the nominal scenario described in the previous section. Upon opening the PORVs at 10 min, the pressure dropped rapidly to approximately 8 MPa (see Fig. 48). When the pressure dropped below 10.7 MPa at about 11.7 min, the SIPs began to deliver water (Fig. 49). The pressure plateau that was reached was determined by the

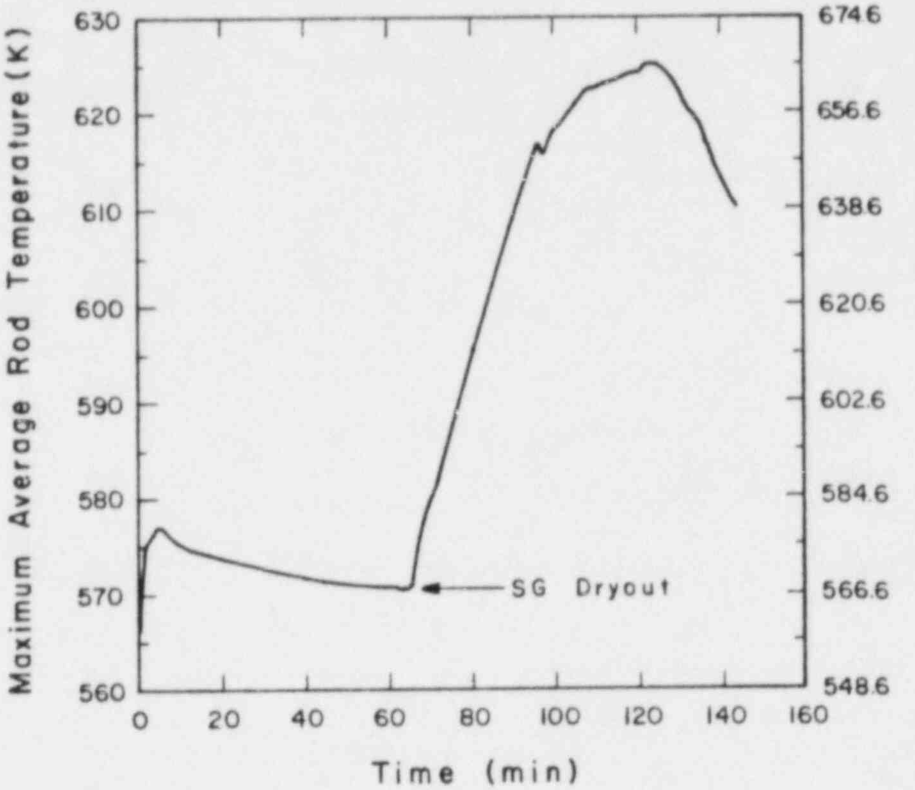

Fig. 47. Cladding temperatures decrease with PORVs fully open.

saturation conditions in the secondary loops, which in turn were determined by the ARV setpoint pressures (Fig. 50). Reduction of the primary system pressure resulted in saturation at 13.3 min and the primary steam-water mixture began to expand. The PORV openings were large enough so that flow through these valves prevented a large increase in primary system pressure. As the void fraction in the upper plenum and hot legs increased, flow through the cold leg upstream of the ECC inlet stopped (see Figs. 51 and 52). At 38.3 min, the minimum vessel inventory was reached; the core at 38.3 min was about 20% empty (Fig. 51). From 38.3 min to the end of the calculation, the coolant and cladding temperatures decreased slowly. At 50 min, the steam generators stopped voiding (Fig. 53) and retained approximately 32% of the original water inventory. From 60 min, the steam generators, along with the primary system, were cooled by the ECC flow. At the end of the calculation for Case A, the core void fraction was 5%, and the system was cooling and refilling. The source of ECC water (the refueling water storage tank) would be depleted in approximately 11 h maintaining the flow rate calculated at 83 min. By that time, some alternate form of cooling would be required. The

TABLE VIII

EVENT SEQUENCE FOR BLEED-AND-FEED CASE A

(OPEN PORVs AND INITIATE ECC AT 10 min)[a]

Time (s)	Time (min)	Event
0		Loss of feedwater, main coolant pumps tripped.
0.6		Reactor scram.
600	10	Open PORVs, initiate SIPs manually.
700	11.7	SIPs begin to deliver, primary pressure 10.7 MPa.
800	13.3	Primary saturates.
2300	38.3	Minimum vessel inventory, core 20% voided.
3000	50.0	Steam generator secondaries stop voiding.
5000	83.3	End of calculation, system recovering.

[a]Figs. 48 through 53.

reactor cannot be cooled by recirculating ECC water because the system pressure will be too high for the residual heat removal system pumps to operate. Containment recirculation can be accomplished only by aligning the charging pumps with the RHR pumps. Thus, further action is required to bring the plant to a stable condition.

In Case A, the primary system pressure decrease was limited by the saturation condition in the secondary system. In Case B, the atmospheric relief valves were opened at 60 min to lower the temperature and pressure on the secondary side; this cooled and lowered the pressure of the primary. The event sequence, which is the same as that of Case A to 60 min, is given in Table IX. The important system parameters are plotted in Figs. 54 through 58. The intent of this calculation is to determine if, by this strategy, the primary system pressure can be lowered to the RHR system

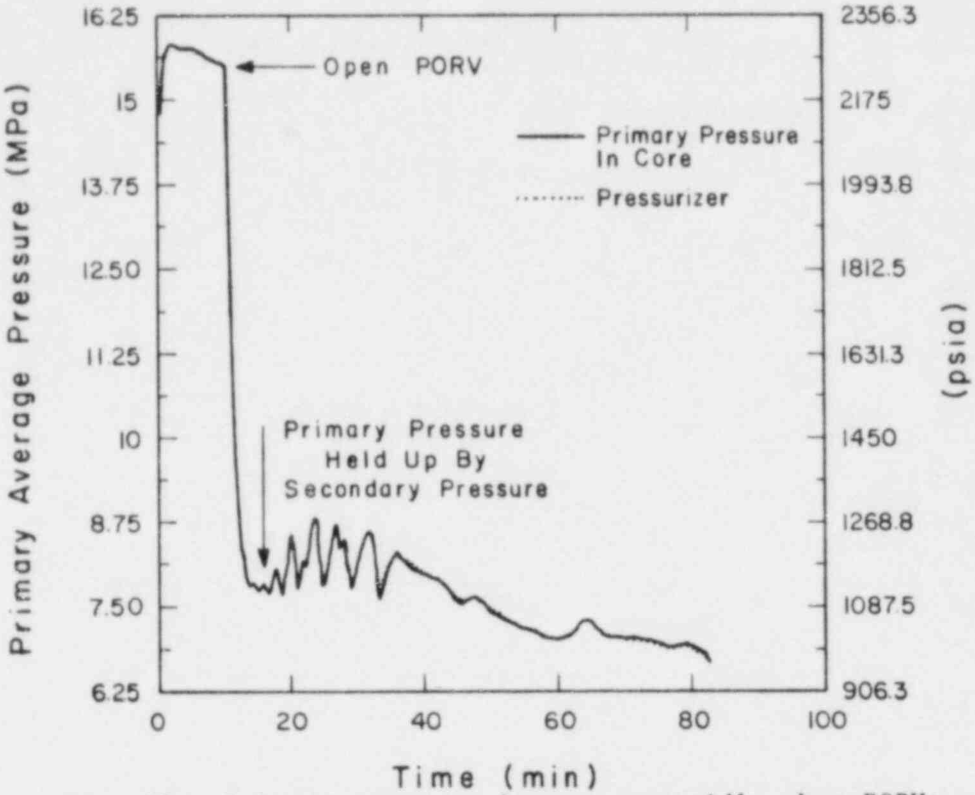

Fig. 48. Primary pressure decreases rapidly when PORVs are opened at 10 min.

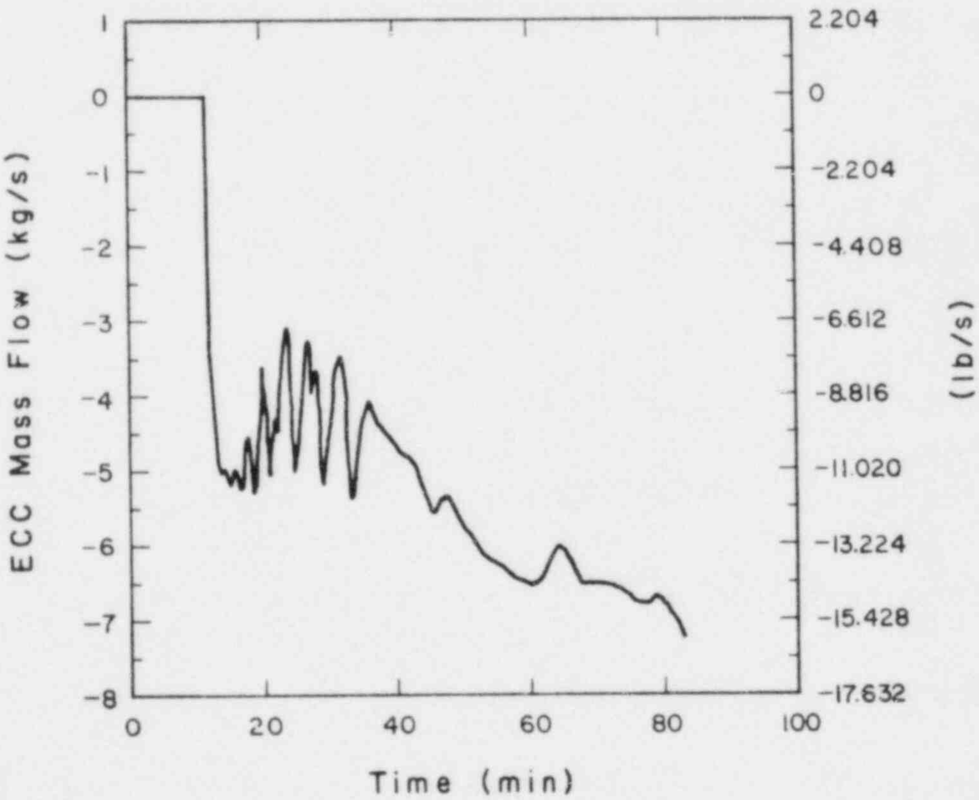

Fig. 49. ECC flow into each loop increases slowly as primary pressure decreases.

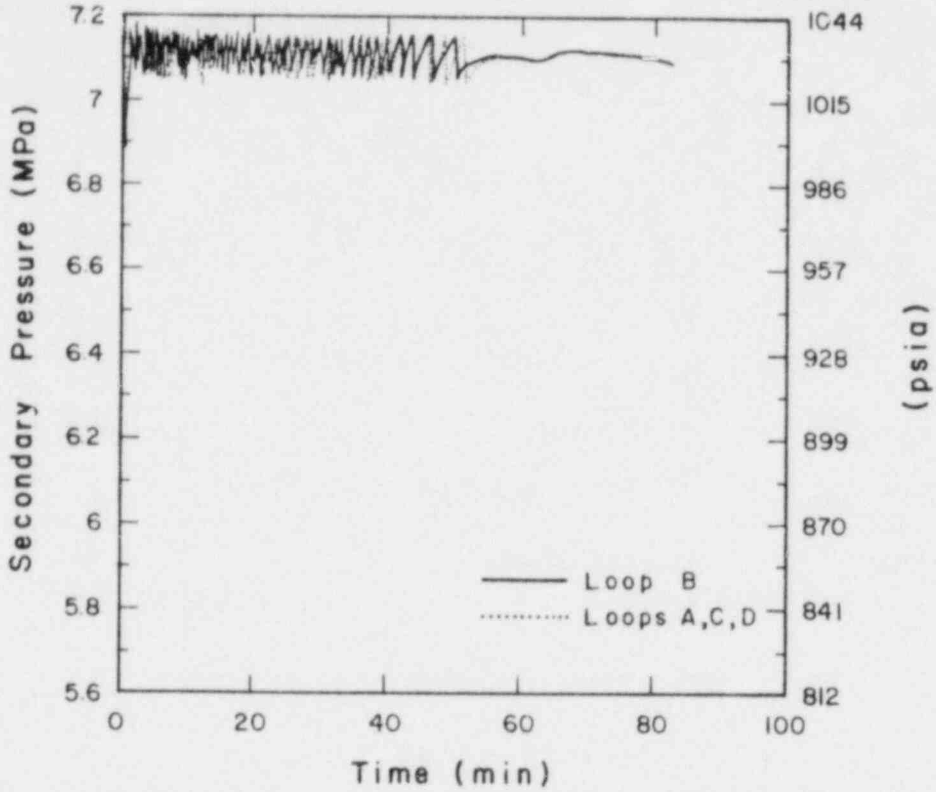

Fig. 50. Secondary pressure increases to ARV setpoint and remains there throughout transient.

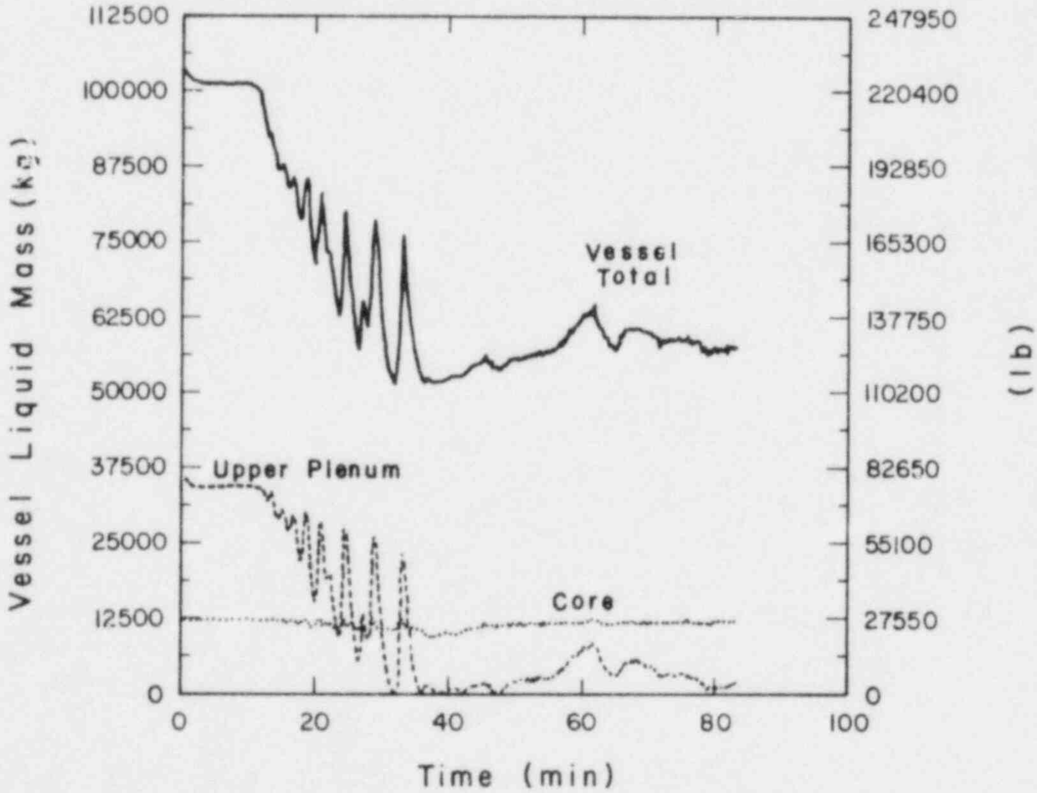

Fig. 51. Vessel inventory shows refilling.

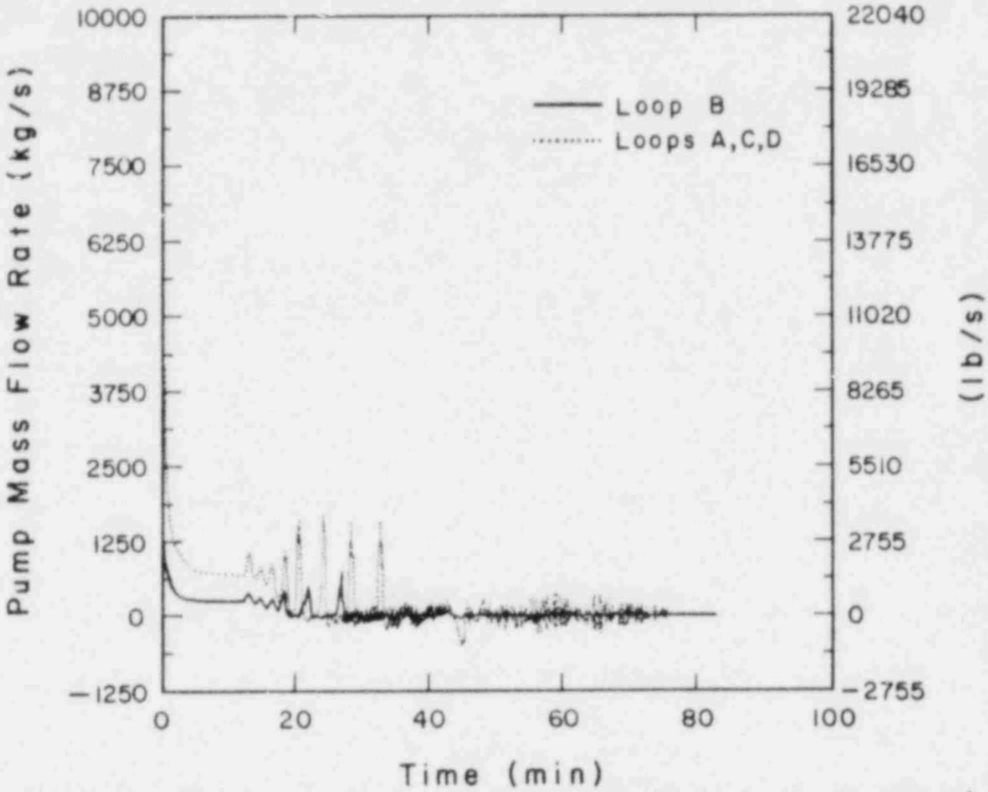

Fig. 52. Mass flow rate through pumps shows loss of natural circulation after saturation at 13 min.

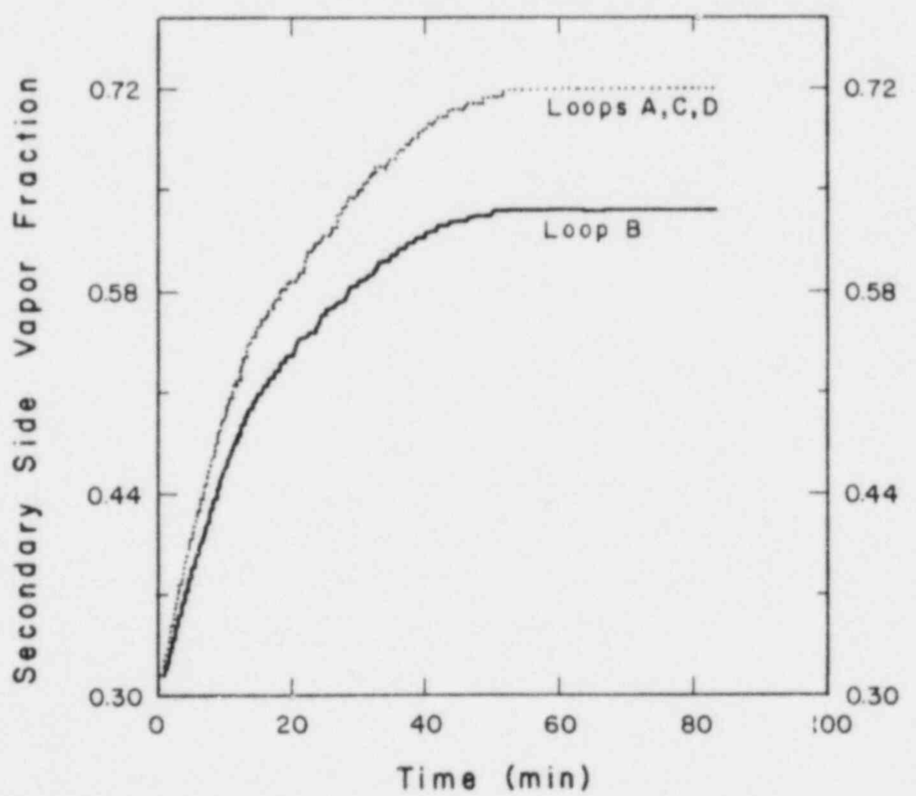

Fig. 53. Steam generator voiding ceases at 50 min as primary is cooled by ECC flow.

TABLE IX

EVENT SEQUENCE FOR BLEED-AND-FEED CASE B[a]

(CASE A + OPEN ARVs AT 60 min)

Time (s)	Time (min)	Event
0		Loss of feedwater, MCPs tripped.
0.6		Reactor trip.
600	10	Open PORVs, initiate SIPs manually.
700	11.7	SIPs begin to deliver, primary pressure 10.7 MPa.
800	13.3	Primary saturates.
2300	38.3	Minimum vessel inventory, core 20% voided.
3000	50.0	Steam generator secondaries stop voiding.
3600	60.0	Open ARVs manually to depressurize secondary.
4100	68.3	Close ARVs and PORVs to prevent steam generator secondary dryout and to pressurize the primary.
5800	96.7	End of calculation, primary subcooling, ~6% of steam generator secondaries inventory remains, system is recovering.

[a]Figs. 54 through 58.

operating pressure (3.0 MPa). Figure 54 shows the effect on the secondary system pressure of opening the ARVs at 60 min. The secondary pressure dropped to 1.5 MPa by 68 min. The effect of this lowered secondary temperature and pressure upon primary temperatures and pressure is seen in Figs. 55 and 56. The primary pressure dropped to 2.4 MPa (Fig. 55). The coolant temperatures decreased by 60 K over the same span (Fig. 56). The lowered secondary system temperature increased the primary-to-secondary heat transfer and cooled the primary. At 68.3 min, the ARVs were closed to prevent dryout of the steam generator secondaries. As can be seen in Fig. 57, closing the ARVs stopped

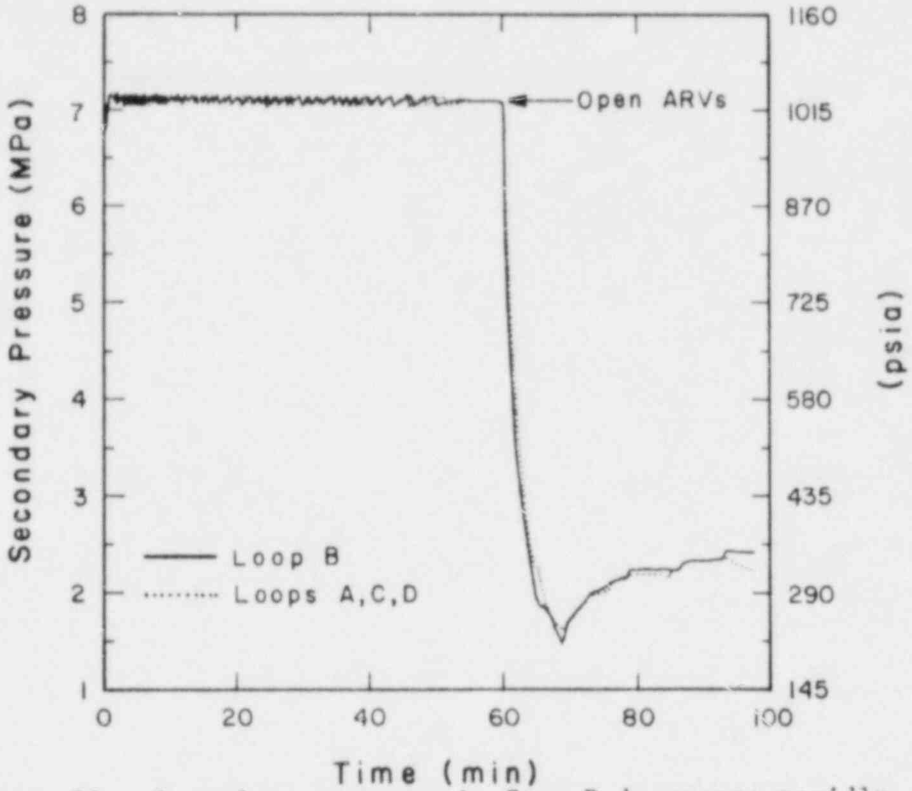

Fig. 54. Secondary pressure in Case B decreases rapidly when ARVs are opened at 60 min.

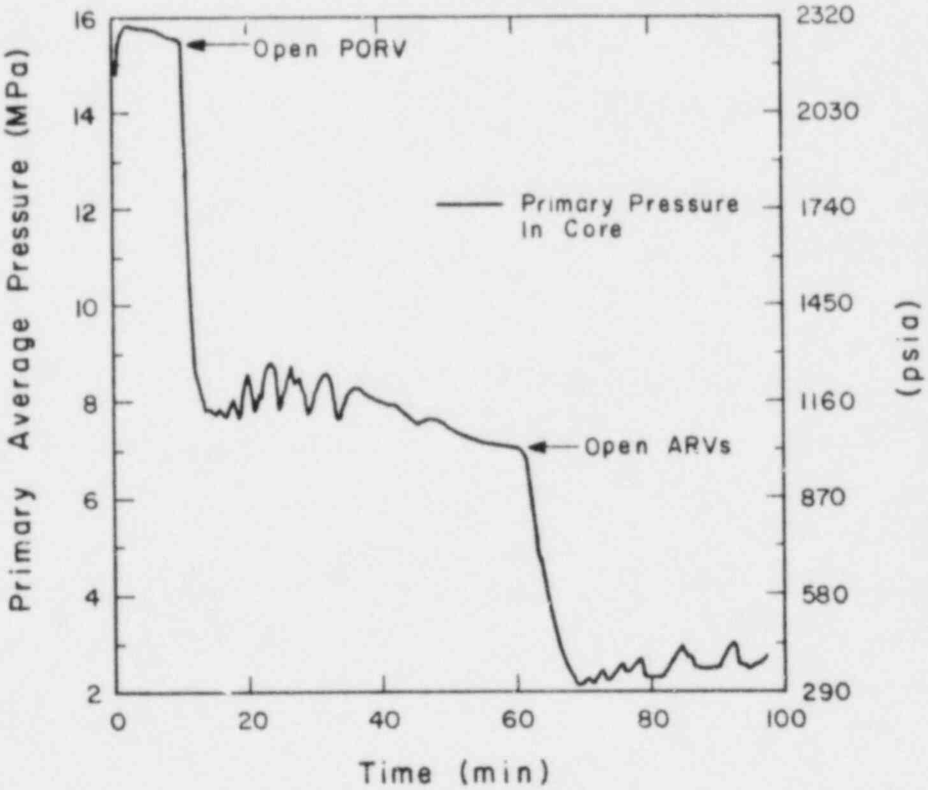

Fig. 55. Primary pressure in Case B decreases when ARVs are opened at 60 min.

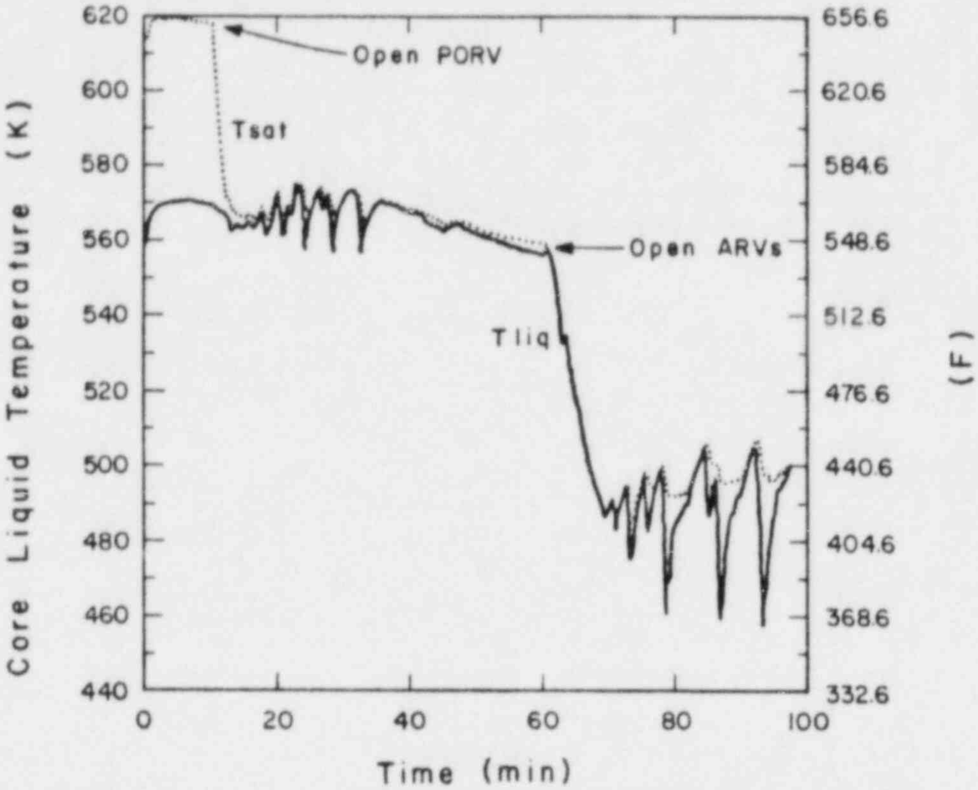

Fig. 56. Core liquid temperature in Case B decreases when ARVs are opened at 60 min.

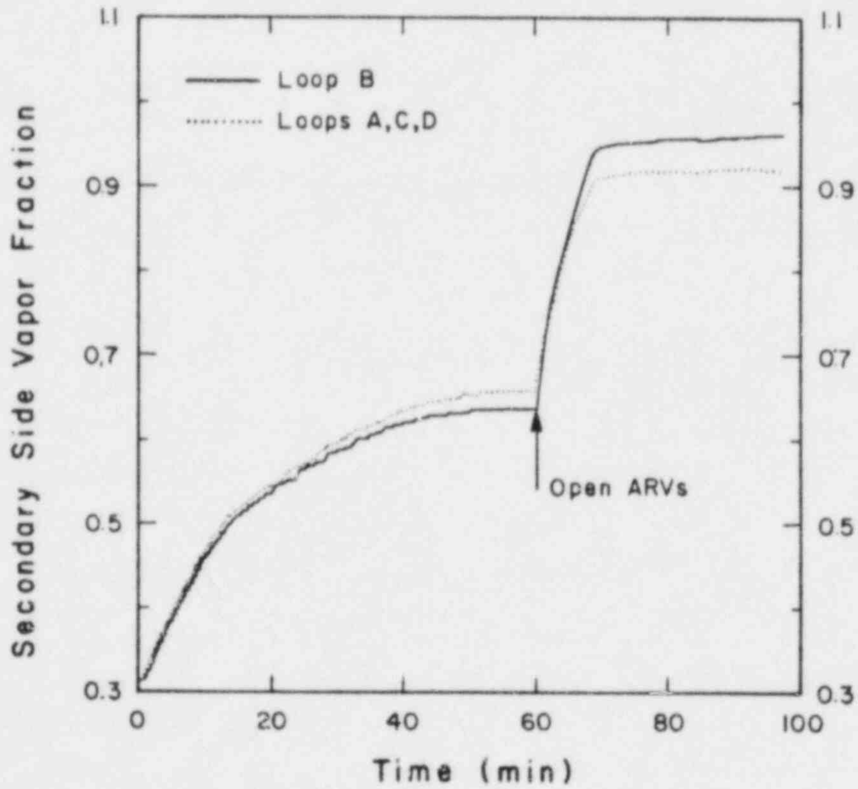

Fig. 57. Small amount of water remains in steam generators at the end of Case-B calculation.

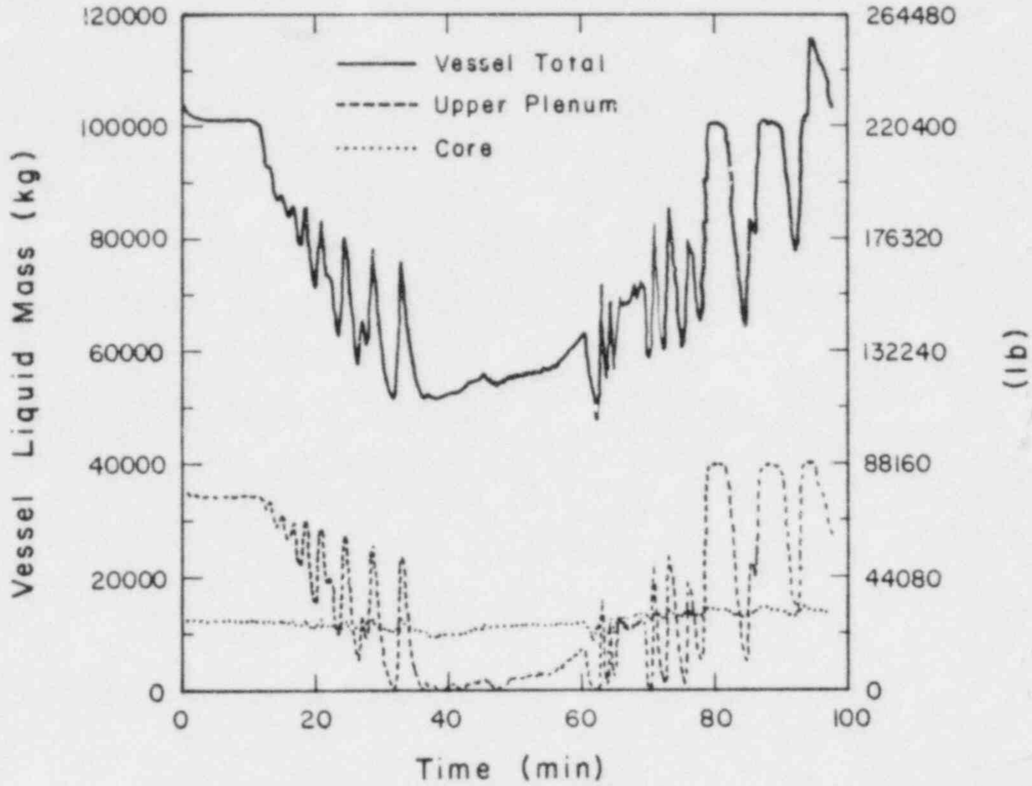

Fig. 58. Vessel inventory indicates refilling of primary as ECC flow increases.

the voiding with approximately 6% of the original water inventory remaining. In addition to closing the ARVs, the PORVs were closed to allow the primary system to pressurize, subcool, and refill (see Fig. 58). At the end of the calculation, the primary pressure was such that the RHR system could operate in a recirculation mode to cool the reactor. At the ECC flow at 96.7 min, it was calculated the supply of once-through ECC water would be depleted in 7 h. A recirculation cooling mode using the RHR system could be initiated after ~1.5 h, if the RHR system were capable of removing 40 MW.

The third bleed-and-feed scenario (Case C) is similar to Case B except for timing. In Case C, the PORVs were opened and the ECC system was initiated at 20 min. The events for Case C are given in Table X. Figures 59 and 60 give the primary system pressure and steam generator secondary void fraction for this case. As with Cases A and B, the pressure dropped to 8.0 MPa within 2 min of opening of the PORVs. At 23.3 min, the primary system was saturated at 8.0 MPa. At 25 min, the ARVs were opened to see if boiling the remaining water in the steam generators could reduce the primary pressure to 3.0 MPa. It can be seen in the figures that the primary pressure was reduced to 2.4 MPa

TABLE X

EVENT SEQUENCE FOR BLEED-AND-FEED CASE C[a]

(OPEN PORVs AND INITIATE ECC AT 20 min; OPEN ARVs AT 25 min)

Time (s)	Time (min)	Event
0		Loss of feedwater, main coolant pumps tripped.
0.6		Reactor trip.
1200	20.0	Open PORVs, initiate ECC manually.
1300	21.7	SIPs begin to deliver, primary pressure at 10.7 MPa.
1400	23.3	Primary saturates.
1500	25.0	Open ARVs manually.
2000	33.3	End of calculation, system will recover.

[a]Figs. 59 and 60.

at the time that the steam generators were nearly empty. Thus, 20 min is the limit for initiating the above actions and still retaining enough water in the steam generator to reduce the primary pressure below 3.0 MPa. At the end of the calculation, the ECC flow, which was the same as that of Case B, could remove 39 MW by increasing its sensible heat and an additional 100 MW if it were vaporized. The reactor power was 52 MW at this time. Thus, cooling of the primary system and recovery in a manner similar to Case B is possible in this scenario.

In the fourth and most complicated (from an operational viewpoint) scenario, the ARVs and PORVs were operated to maintain the primary subcooling and to prevent steam generator dryout. The events are given in Table XI. The primary system pressure, the core average and saturation temperatures, and the steam generator secondary void fractions are given in Figs. 61 and 62, respectively. At 10 min, the ARVs were opened 80%, and the ECC was initiated. The open ARVs decreased the secondary temperature; the primary-to-secondary heat transfer increased, resulting in a decrease in the primary temperature and pressure.* At 18 min, the primary pressure and temperature were reduced to 12.4 MPa and 559 K respectively (Figs. 61 and 62). This corresponded to a subcooling of 41 K, so the PORVs were opened to reduce the pressure further to decrease the subcooling margin, and to increase ECC flow. When the primary pressure decreased to 8.0 MPa, the PORVs were operated to maintain a 24 K subcooling margin. The step decreases in pressure allowed the subcooling margin to be maintained. By following this stepwise pressure-reduction strategy, the pressure was reduced to 3.1 MPa at 33.3 min. The steam

*In an auxiliary calculation, the ARVs were opened fully at 10 min to see how much the primary temperature and pressure could be reduced by this mechanism alone. By 25 min, the steam generators were voided, the primary pressure had been reduced to 7.6 MPa and the liquid was 60 K subcooled. Thus, some additional action is required to reduce the primary system pressure further.

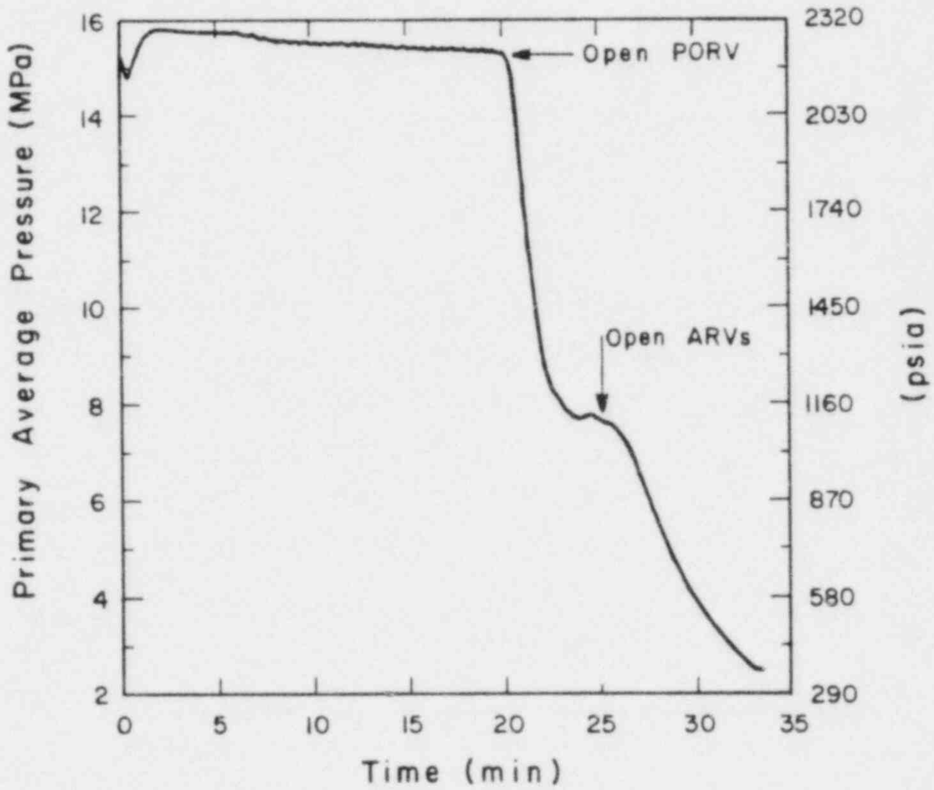

Fig. 59. Primary pressure in Case C decreases when PORVs are opened at 20 min and ARVs are opened at 25 min.

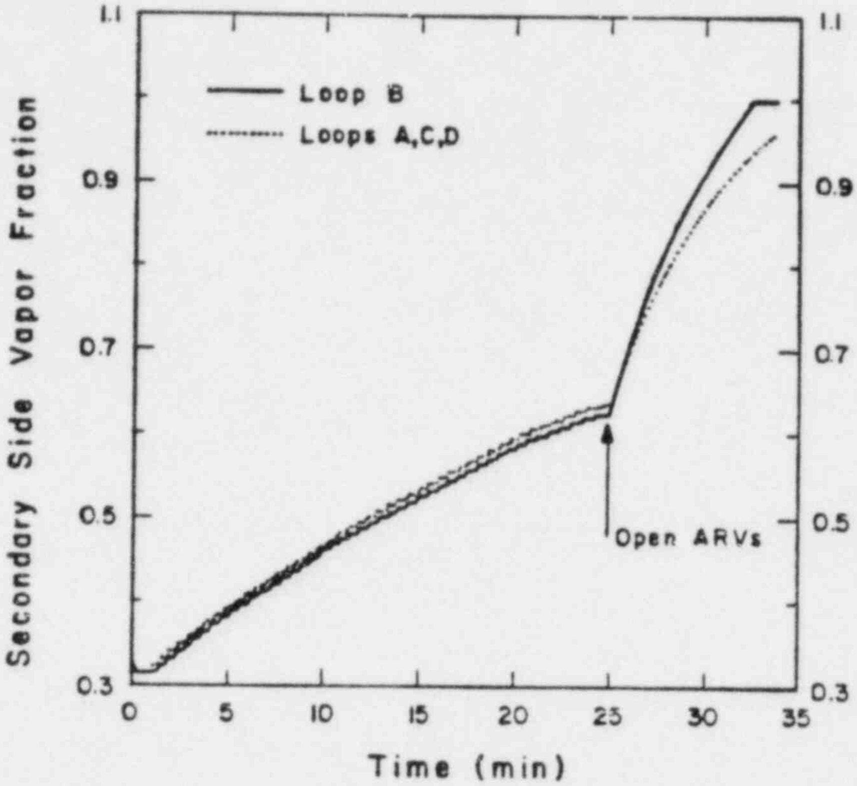

Fig. 60. Steam generators empty after ARVS are opened at 25 min.

generators were nearly empty, and the core region was 2 K subcooled. Heating of ECC liquid (without vaporization) was capable of removing 39 MW of the 52-MW decay power produced by the core. Further action, therefore, is required to maintain subcooling and to prevent heatup of the primary system. As in the above scenarios, recovery and cooldown of the system would be possible if the RHR system were capable of removing the core decay heat. Because the RHR delivery pressure was reached at an earlier time (33 min), decay power in this case is 55 MW.

In a second auxiliary calculation related to the above scenario, it was assumed that the full ECC system (CCPs + SIPs) was available. Using the same stepwise pressure-reduction strategy, the primary pressure was 3.0 MPa after 58 min. With increased ECC flow from the CCPs, a larger subcooling margin (15 K) was maintained. It also was possible to retain a small fraction (2%) of the original steam generator secondary water inventory. Also, full ECC flow at 3.0 MPa was sufficient to remove decay heat and cool the reactor.

TABLE XI

EVENT SEQUENCE FOR BLEED-AND-FEED CASE D[a]

(OPTIMAL VALVE OPERATION TO RECOVER WITH PRIMARY SUBCOOLED)

Time (s)	Time (min)	Event
0		Loss of feedwater, main coolant pumps tripped.
0.6		Reactor scram.
600	10	Open ARVs 80%, initiate SIPs.
1080	18	Reduce PORV setpoint pressure to 8 MPa, open ARVs fully.
1120	18.7	SIPs begin to deliver, primary pressure 10.7 MPa.
1440	24	Reduce PORV setpoint pressure to 5 MPa.
1710	28.5	Reduce PORV setpoint pressure to 2.9 MPa, close ARVs.
2000	33.3	System subcooled at ~3.1 MPa but further action required to maintain subcooling.

[a]Figs. 61 and 62.

In summary, we have found that recovery with bleed-and-feed is possible. In addition, we have found that several strategies are available to reduce the system pressure to the RHR delivery pressure. If the RHR system were capable of removing 40 to 50 MW (depending upon the strategy used), recovery and cooldown of the system could be facilitated.

D. Symptom-Oriented Recovery Procedure

Symptom-oriented procedure guidelines detail the operator actions prescribed for recovery from severe transients. The effectiveness of these guidelines for a loss-of-feedwater transient was investigated assuming operator actions were initiated minutes before secondary dryout. These actions involved opening both PORVs and initiating ECC with both the high-pressure injection and charging pumps. Primary depressurization that

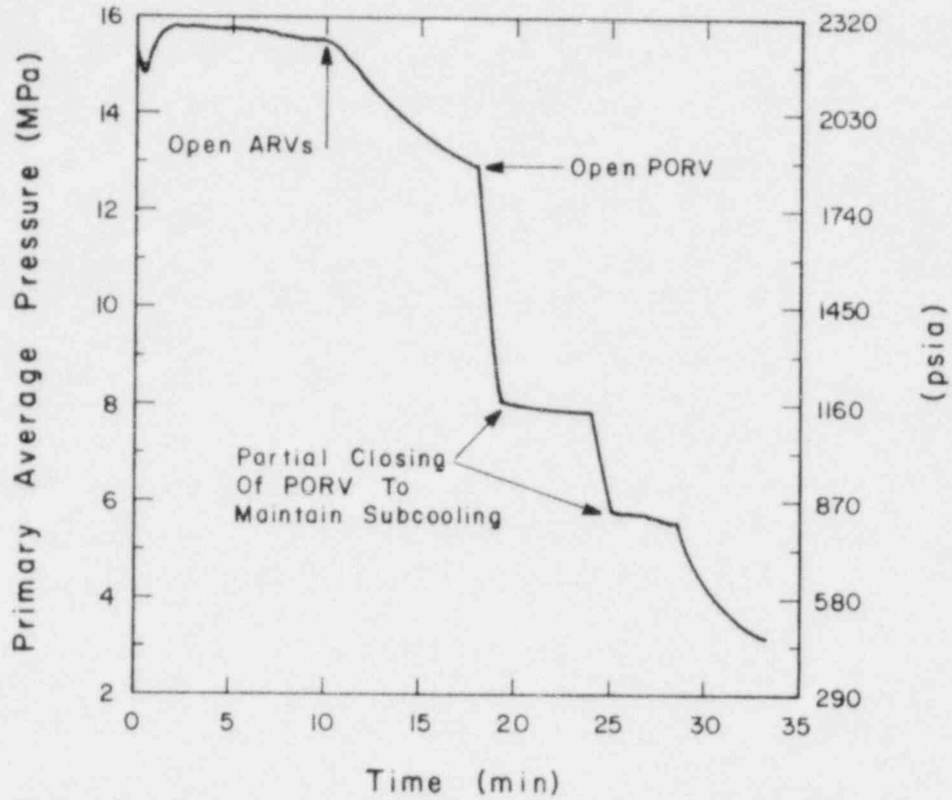

Fig. 61. Primary pressure decreases in stepwise fashion as PORV setpoint pressures are reduced.

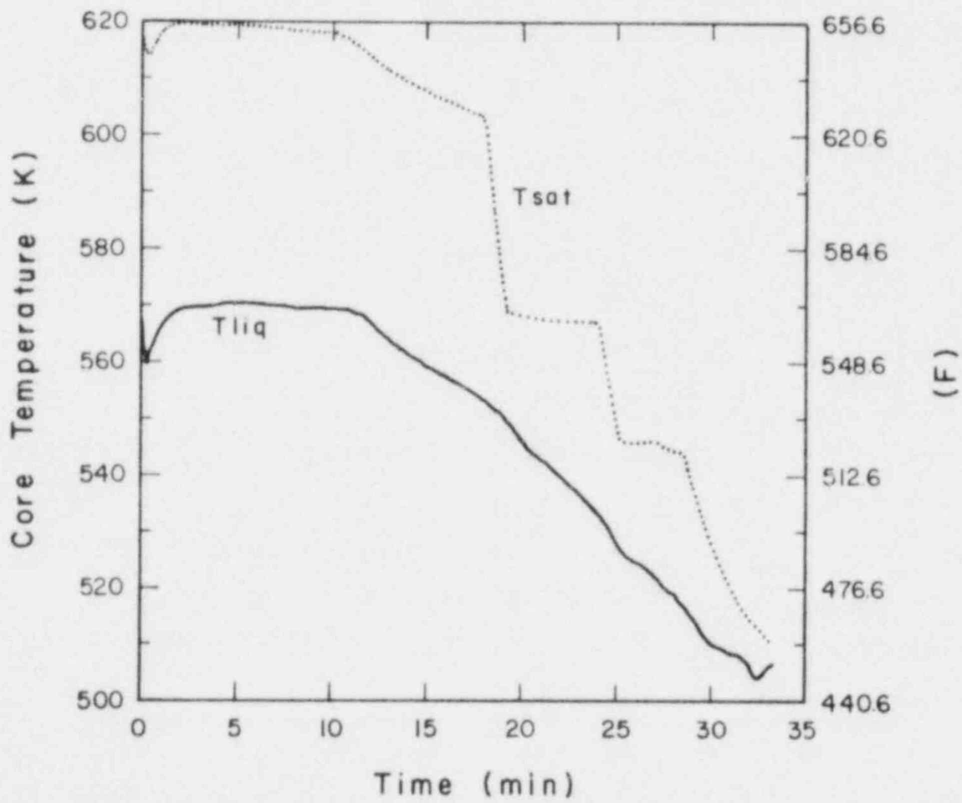

Fig. 62. Primary liquid remains subcooled in core region.

results from opening both PORVs enhanced the ECC flow and provided core cooling in a once-through mode with water from the refueling water storage tank (RWST) being discharged into the containment. Fan coolers were modeled in the containment as pressure dependent negative heat sources. Heat removal from three fan coolers was found to be sufficient to maintain containment pressures below the setpoint for actuating sprays. The RWST was empty approximately 5.5 h after ECC initiation, at which time the operators would begin ECC recirculation from the containment sump. Assuming all the fan coolers were unavailable, the containment pressure rose to 0.262 MPa (approximately 1.0 h after ECC initiation), and the containment sprays were initiated. In this situation, in which ECC and containment spray water is being drawn from the RWST, the RWST emptied about 1.5 h after ECC initiation.

The guidelines for this type of transient, therefore, provide protection to the core through once-through cooling and afford the operators sufficient time to begin ECC recirculation, even with the loss of containment fan coolers. Additional operator action [throttling the high pressure injection (HPI) flow] can delay the time before RWST depletion. With the PORVs fully open, the primary depressurizes to the secondary side atmospheric relief valve setpoint (7.0 MPa). At this pressure, the HPI flow is approximately 75 kg/s, which is twice the flow necessary to remove the decay energy. To preserve the RWST inventory, the operators should throttle the HPI flow to maintain a low subcooling margin.

IX. PLANTS THAT HAVE EXPERIENCED UNAVAILABILITY OF AUXILIARY FEEDWATER

Precursors to severe loss-of-feedwater transients have occurred, although prolonged AFWS unavailability following main feedwater trip has not been observed. Precursor events are defined here as those involving either total auxiliary feedwater unavailability during a test of the system or partial AFWS unavailability during a feedwater transient. Such events are recorded in licensee event reports (LERs) to the NRC, which are routinely cataloged and analyzed in Nuclear Safety, in reports to Congress on abnormal occurrences, and in Nuclear Power Experience, (an indexed bibliography of reactor events). Table XII contains a list of some relevant precursor events.

X. FREQUENCY AND PROBABILITY ANALYSIS

Fault-tree analyses of possible failure modes in the AFWS were performed as part of the Reactor Safety Study (RSS).[1] Many features of the AFWS for the plant analyzed in the RSS are similar to those at Zion-1 so that results from the RSS can be adapted and used in determining the likelihood of failures in the AFWS at Zion. The reduced fault tree for failure of the AFWS to perform on demand for the first 8 h following one of three initiating events is shown in Fig. 63. Transients involving loss of main feedwater are considered in the

TABLE XII

PRECURSOR EVENTS TO FEEDWATER TRANSIENTS INVOLVING EMERGENCY FEEDWATER UNAVAILABILITY

Reactor	Date	Description
ANO-2	4-80	Loss-of-off-site power transient involved lost AFWS pump suction on both pumps for 15 min.
ANO-1	6-80	Following reactor trip, the steam-driven AFW pump tripped on overspeed; the motor-driven AFWS pump functioned properly.
Trojan	2-76	Faulty relay prevented starting of both AFW pumps during a test.
TMI-2	3-79	AFWS valved out; discovered following main feedwater trip.
Millstone-2	3-80	AFWS pump packing failed following main feedwater trip; remaining AFWS pump functioned properly.
Zion-1	8-76	Two of three AFWS pumps failed to start following a reactor trip.
Haddam Neck	7-76	Both AFWS pump turbines failed to start during a test because of vapor binding.
Ginna	12-73	Both AFWS pumps were discovered to be inoperable because of air in the suction line.
Kewaunee	11-75	Resin in suction strainers reduced flow from all three AFWS pumps during a test.

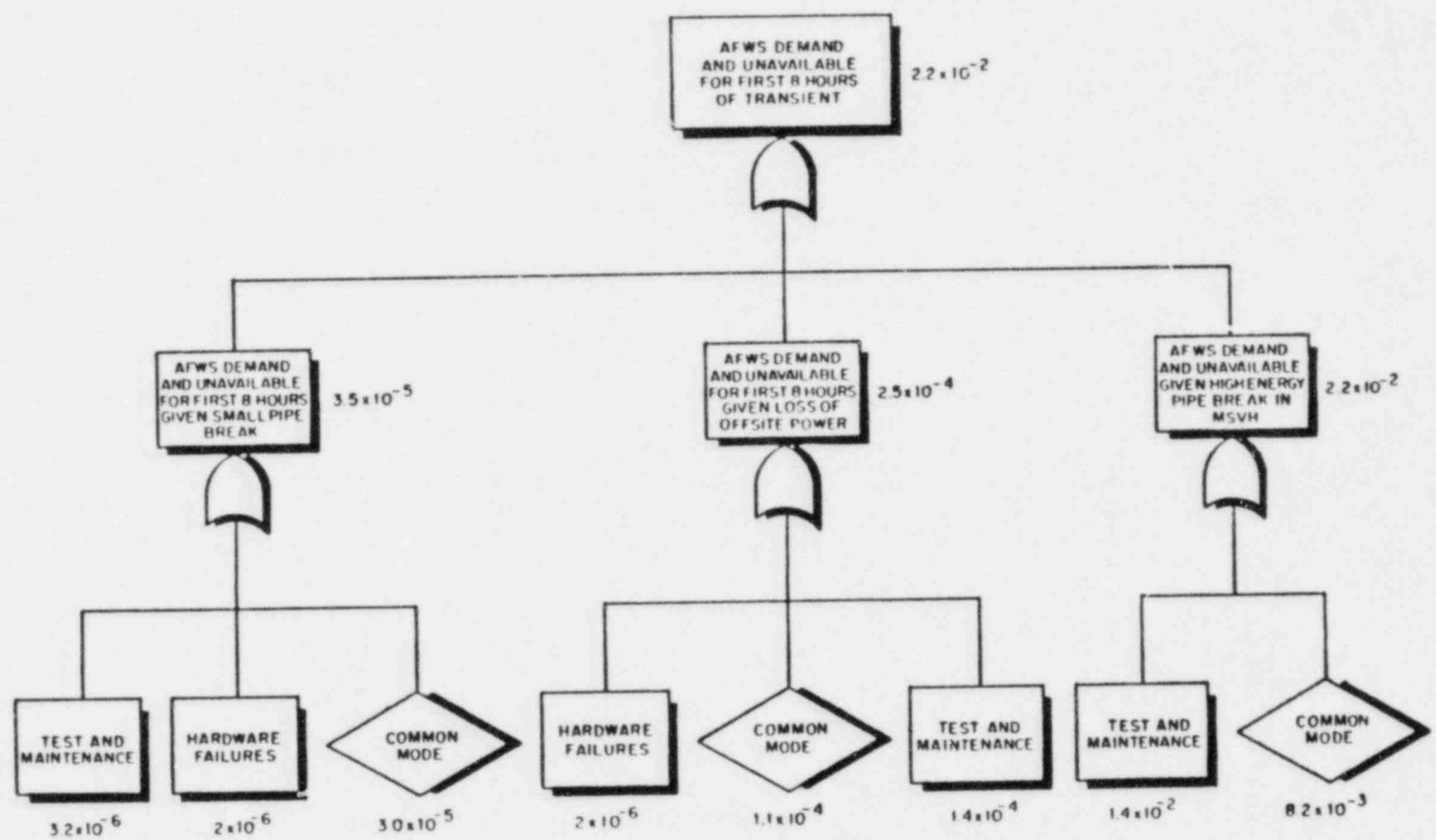

Fig. 63. Reduced fault tree for failure of AFWS to perform on demand (adapted from Ref. 1).

RSS for small pipe-break initiators as well as loss-of-off-site-power initiators.

The major contribution to the failure probability of the AFWS for the plant studied in the RSS arises from a pipe break (main steam or feedwater) inside the main steam valve house. Such an event could result in disabling both the motor-driven and the turbine-driven auxiliary feedwater pumps. At Zion, these pumps are located in the same building, and the motor-driven pumps are separated from the turbine-driven pump by a wall ~2 m high. Failure of the AFWS to perform on demand following loss-of-off-site power, in which the emergency power system functions as designed, was found to be more probable in the RSS than failure of the AFWS coupled with loss of main feedwater (small pipe break). Our investigations of accidents involving AFWS unavailability have focused on loss-of-off-site-power initiators and have considered other loss-of-feedwater initiators.

XI. CONCLUSIONS

Failure of the auxiliary feedwater system during transients initiated by loss-of-off-site power or loss of main feedwater can prevent the operators from placing the plant in a stable condition. Recovery of secondary cooling by auxiliary feedwater (or main feedwater) ultimately will be necessary for the plant to reach cold shutdown conditions. Results of TRAC calculations have shown that a stable plant state can be reached given any of the following:

1. 15% of rated auxiliary feedwater flow initiated at the start of the transient and supplied to one or all four steam generators;
2. introduction of auxiliary feedwater before natural circulation ceases (~2 h into the transient);
3. failure of atmospheric relief valve on one steam generator to reclose after opening followed by introduction of auxiliary feedwater before natural circulation ceases.

Should the plant experience a total loss of secondary cooling, the reactor can be cooled temporarily with the charging pumps, taking water from the refueling water storage tank and injecting it into the cold legs. The decay power generated may be removed for several hours (until the RWST is depleted) by boiling and discharge of this water through the PORVs. Failure of the PORVs to open at pressures less than or equal to their setpoint values or failure of the charging pumps to deliver their rated flows at the PORV setpoint pressure will exacerbate the accident so that the operators must take action to reduce primary pressure thereby increasing the ECC flows. Failure of one PORV to reclose after initial opening results in primary coolant saturation and ECCS initiation occurring earlier in the transient than calculated for nominal operation of the PORVs. Although the plant can maintain a quasi-stable condition with one PORV stuck open and with once-through cooling provided by the charging pumps, the time before natural circulation flow ceases is shortened from ~2 h with nominal PORV operation to ~1-1/4 h with a stuck-open PORV. These accident sequences and the resulting end states for the plant are summarized in Fig. 64.

Potential operator strategies for intervening in accidents involving loss of secondary cooling could involve manual actuation of ECC flow, depressurization by opening the PORVs, and use of the PORVs and ARVs to reduce the primary pressure and cool the plant. Early manual initiation of ECC flow prolonged the time before natural circulation ceased but did not prevent the primary coolant from reaching saturation conditions. Depressurization using the PORVs produced a substantial increase in the ECC flow from the charging and high-pressure injection systems; this increased flow cooled the core and began refilling the reactor vessel. For accidents involving loss of secondary cooling and failure of the high-head charging pumps (or for PWRs that do not have charging pumps capable of supplying water at the PORV setpoint pressure), reducing the primary pressure by opening the PORVs allows the plant to be cooled by the safety injection system.

Strategies for depressurizing while simultaneously cooling the primary were tested to determine whether the reactor could reach a stable condition with cooling provided by recirculating water through the residual heat removal system. Results of these analyses indicated that discharge through the ARVs of about one-half the nominal inventory of the steam generators would be required to cool the primary and allow depressurization to the maximum RHR operating pressure (3.0 MPa). Thus, the operator must commit to such a strategy within ~20 min. Further analyses indicated that a small margin of primary subcooling could be maintained while cooling and depressurizing to the

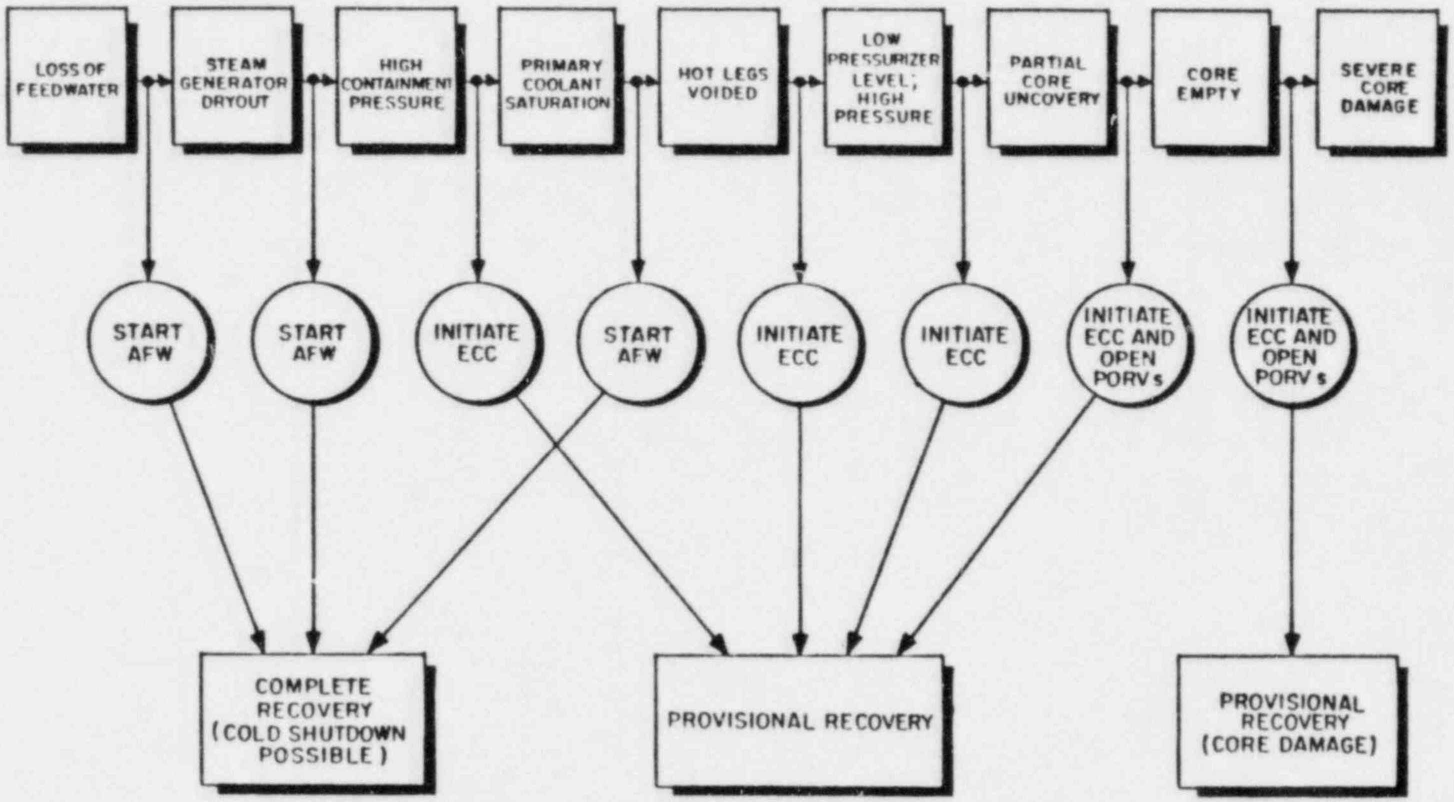

Fig. 64. Summary of accident sequences and states for feedwater transients at the Zion-1 plant.

RHR operating pressure, if the operator started within 10 min and operated the PORVs manually to maintain the subcooling margin. Because of uncertainties in plant operating parameters, the short time (~10 min) required to commit to such a strategy, and the detailed information required by the operator during depressurization, the effectiveness of this strategy cannot be assured from the analyses.

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APPENDIX

SENSITIVITY OF RESULTS TO UNCERTAINTIES

The numerical simulation of severe accidents requires models of plant characteristics and thermal-hydraulic phenomena. Information on the plant condition at the start of the transient and on the performance capabilities of equipment actuated during the transient is used with these models to predict the accident progression. Uncertainties in models, which approximate the physical characteristics and thermal-hydraulic behavior, and information on plant features and operating conditions can lead to uncertainties in the predicted results. For these results to be meaningful, the effects of these uncertainties must be recognized.

Uncertainties can be grouped into three categories: (1) plant state (input information) uncertainties, (2) model uncertainties, and (3) phenomenological uncertainties. Uncertainties in the plant state include both the initial conditions for the transient (for example, operating power level, power history, control-rod reactivity, makeup and letdown flows, and steam generator inventories) and the performance characteristics of equipment actuated by the plant response (for example, high-pressure injection flows, pressurizer heater/sprayer capacities, and pressure/temperature setpoints). Plant state uncertainties can be reduced or bounded by probabilistic treatment of the initial conditions and by developing better input information on equipment performance.

Model uncertainties result from inaccuracies in the mathematical representation of physical features (for example, operation of valves and pumps) and from approximations or simplifying assumptions made by the analyst to improve calculational efficiency. The reduction of modeling uncertainties requires more detailed experimental data on which the development of more accurate models can be based and more detailed representations of plant systems using available models. Phenomenological uncertainties result from incomplete information or mathematical formulations describing the phenomena (for example, flow of steam/water mixtures through steam relief valves). Reductions in phenomenological uncertainties would require both new experimental data and development of models based on improved mathematical formulations.

I. PLANT STATE UNCERTAINTIES

The plant state characteristics used as initial conditions for feedwater transients can vary over a wide range during normal operation between refuelings and unplanned shutdowns. The uncertainties in plant state will be both random and systematic. For example, the decay power levels will depend on the power history of the reactor up to the time of the accident, and this could affect both the timing and sequence of subsequent events. To evaluate the sensitivity of TRAC results to uncertainties in initial conditions, several calculations were performed in which initial plant operating parameters were varied over normal or assumed ranges, and the effect on the time to steam generator dryout was used to measure the sensitivity of results to uncertainties in initial conditions. The uncertainties considered in these calculations, their nominal values, and ranges are given in Table A-I.

TABLE A-I

UNCERTAINTIES IN INITIAL CONDITIONS FOR FEEDWATER TRANSIENTS

UNCERTAINTY	NOMINAL VALUE	RANGE
Scram signal delay	0.6 s	0.5 to 2.0 s
Shutdown margin ($\delta k/k$)	3.5%	1.0 to 5.0%
Initial power level	3238 MW	0.95 to 1.05
Power decay	ANS 5.1 (1971)	0.8 to 1.2
Steam generator inventory	43000 kg	0.9 to 1.1
Flow coastdown	FSAR (Ref. 8)	0.9 to 1.1
PORV setpoint	16.1 MPa	16.0 to 16.2 MPa

Sensitivity analyses of calculated results for the uncertainties and ranges listed in Table A-I have shown that

1. the time for steam-generator dryout can vary between 45 and 75 min;

2. calculated dryout times are most sensitive to uncertainties in decay power;

3. dryout times are also correlated but to a lesser extent, with variations in scram signal delay, shutdown margin, initial power level, and steam generator inventories; and

4. variations in flow coastdown, PORV setpoints, and model uncertainties considered in these sensitivity analyses do not affect the calculated results.

The use of sensitivity analyses to evaluate the effects of uncertainties yields an expected range of results (for example, the time to steam generator dryout). Because these variations in initial conditions occur randomly during plant operation, alignment of the parameters at the limits of their ranges could produce calculated results that are overly conservative (or optimistic). Further, uncertainties that fall outside the normal ranges should be considered as additional equipment failures (for example, a prolonged delay in the scram signal for a loss-of-off-site-power initiating event can be considered as an anticipated transient without scram).

II. MODEL UNCERTAINTIES

Early in this study of feedwater transients, the need for more detailed models of relief and safety valve operation was recognized, and improved models were developed and included in TRAC-PD2. The relief valves located on the pressurizer are designed to prevent the primary pressure from exceeding the setpoint pressure (16.1 MPa). Relief-valve models used in TRAC open and close depending on the pressure upstream of the valve, and these models allow multiple pressure setpoints as well as different rates for opening and closing. Safety valves are static check valves whose opening and closing depends upon the pressure gradient across the valve. A valve option developed for TRAC allowed the operation of four safety valves with different setpoints to be modeled as a single valve with multiple pressure-gradient setpoints. By specifying different setpoints for opening and closing, a more accurate representation of the physical system was possible; and numerical instabilities produced by unrealistic fluctuations (flutter) of these valves were eliminated. Although model uncertainties associated with operation of relief and safety valves were reduced by development of more detailed models, the lack of experimental data on stroke rates and drift of pressure and pressure-gradient setpoints introduces other uncertainties. Variations in stroke rates and setpoints over reasonable ranges were considered as part of the sensitivity studies discussed in Sec. I and were found to have little or no effect on the accident signatures.

The component model for pumps in TRAC allows the detailed specification of both steady-state and transient characteristics. However, the unavailability of experimental information on the transient performance and on the pump performance with two-phase (steam/water) mixtures introduces some uncertainty into the calculations. Because the pumps were tripped early in the accident, the effect of these uncertainties would be most apparent in flow coastdown to natural circulation. The effects of uncertainties in pump coastdown characteristics were considered as part of the sensitivity studies and found to have no appreciable effect on the accident.

The application of TRAC to lengthy feedwater transients in which coolant escaping from the primary is vented to the containment after filling the pressurizer relief tank has pointed out the need for an improved model of the containment. For several transients in this study, steam released to the containment node actuated engineered safety features, such as the emergency core cooling system, on containment overpressure. However, improved models are needed to describe more accurately steam condensation in the presence of air in the containment building and to account for heat losses through insulation to the containment atmosphere. (Models that account for the presence of noncondensible gases are included in TRAC-PF1, which was released in August 1981.) Heat losses through insulation to the environment of the containment are estimated to be approximately 1.5 MW. These losses, which are insignificant compared to the decay heat generated in brief transients following shutdown from full power, are about 5% of the decay energy produced after 2 h. For the accidents analyzed in this study, the effect of insulation losses would be to increase slightly the times associated with major events, such as steam generator dryout and primary coolant saturation. If insulation losses were neglected, it would not alter the accident sequence or consequences.

TRAC has been assessed extensively against experimental data for application to loss-of-coolant accidents. Results of these assessments are contained in a report, "TRAC-PD2, Independent Assessment - 1981," Los Alamos National Laboratory report (to be published), together with a discussion of uncertainties in the modeling of thermal-hydraulic phenomena.

III. PHENOMENOLOGICAL UNCERTAINTIES

Uncertainties in calculated flow rates for two-phase (steam/water) mixtures through the relief valves could affect the progression and severity of loss-of-feedwater transients. No experimental data were available on the capacity of these valves for relieving water or steam-water mixtures, so the potential magnitude of these uncertainties cannot be quantified. These uncertainties are important for calculations in which the pressurizer fills with water causing the PORVs to fully open. If the relief rates for two-phase mixtures are much lower than calculated, the primary pressure will increase thereby reducing the coolant flow provided by the charging pumps and by opening the safety valves. For transients in which the pressurizer does not fill with water, the calculations are not as sensitive to this uncertainty, as the steam flow through the relief valves corresponds to their rated capacity.

Another phenomenological uncertainty is associated with the ability of the steam generators to cool the reactor after the primary loops have emptied of water. TRAC calculations have indicated that recovery of feedwater after the primary loops empty will be ineffective in cooling the core and in preventing fuel damage. However, laboratory experiments at the PKL test facility in Germany have indicated that steam condensing in the steam generators and flowing back through the hot legs into the vessel can remove decay heat effectively as long as the core remains covered. Additional experimental data together with improved models for counter-current flow are required to quantify this uncertainty.

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