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Los Alamos PWR Decay-Heat-Removal Studies Summary Results and Conclusions

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Commission

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GLOSSARY

AFW	Auxiliary feedwater
ARV	Atmospheric relief valve
B&W	Babcock & Wilcox
BWST	Borated water storage tank
C-E	Combustion Engineering
CFT	Core flooding tank
ECC	Emergency core coolant
ESS	Engineered safeguards system
F1	Feed-and-bleed success criterion 1 failed
F2	Feed-and-bleed success criterion 2 failed
HP	High pressure
HPI	High-pressure injection
IP	Intermediate pressure
LOCA	Loss-of-coolant accident
LOFW	Loss of feedwater
LOSP	Loss of offsite power
LP	Low pressure
LPI	Low-pressure injection
MFLB	Main feedline break
MSLB	Main steamline break
NRC	Nuclear Regulatory Commission
PORV	Power-operated relief valve
PRT	Pressurizer relief tank
PWR	Pressurized water reactor

RCP	Reactor-coolant pump
RHR	Residual-heat removal
RWST	Refueling water storage tank
SG	Steam generator
SGSD	Steam-generator-secondary dryout
SGTR	Steam-generator tube rupture
SI	Safety injection
S1	Satisfies feed-and-bleed success criterion 1
S2	Satisfies feed-and-bleed success criterion 2
SRV	Safety relief valve
TAP	Task Action Plan
TRAC	Transient Reactor Analysis Code
TSV	Turbine stop valve
USI	Unresolved safety issue
W	Westinghouse

LOS ALAMOS PWR DECAY-HEAT-REMOVAL STUDIES
SUMMARY RESULTS AND CONCLUSIONS

by

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ABSTRACT

The adequacy of shutdown-decay-heat removal in pressurized water reactors (PWRs) is currently under investigation by the Nuclear Regulatory Commission. One part of this effort is the review of feed-and-bleed procedures that could be used if the normal cooling mode through the steam generators were unavailable. Feed-and-bleed cooling is effected by manually activating the high-pressure injection (HPI) system and opening the power-operated relief valves (PORVs) to release the core decay energy.

The feasibility of the feed-and-bleed concept as a diverse mode of heat removal has been evaluated at the Los Alamos National Laboratory. The TRAC-PF1 code has been used to predict the expected performances of the Oconee-1 and Calvert Cliffs-1 reactors of Babcock and Wilcox and Combustion Engineering, respectively, and the Zion-1 and H. B. Robinson-2 plants of Westinghouse. Feed and bleed was successfully applied in each of the four plants studied, provided it was initiated no later than the time of loss of secondary heat sink.

Feed and bleed was successfully applied in two of the plants, Oconee-1 and Zion-1, provided it was initiated no later than the time of primary system saturation. Feed and bleed in Calvert Cliffs-1 when initiated at the time of primary system saturation did result in core dryout; however, the core heatup was eventually terminated by coolant injection. Feed-and-bleed initiation at primary system saturation was not studied for H. B. Robinson-2.

Insights developed during the analyses of specific plant transients have been identified and documented. Based on the detailed results from the specific plant studies and the insights developed, feed-and-bleed feasibility statements for the four plants studied in detail are extended to larger groups of PWRs for which specific, detailed analyses have not been performed. These extension statements are largely based on inspection for similarity of HPI delivery characteristics and PORV relief capacities.

I. EXECUTIVE SUMMARY

The US Nuclear Regulatory Commission (NRC) has identified a number of nuclear-safety issues requiring further investigation. These have been designated as unresolved safety issues (USIs), and action plans have been prepared to resolve them. This paper describes one part of the effort to resolve USI A-45, Shutdown-Decay-Heat Removal.

Feed and bleed has been considered as one method of removing decay heat from pressurized water reactors (PWRs) following total loss of feedwater (LOFW). Feed and bleed is a procedure in which coolant is injected into the primary system by safety and/or non-safety grade systems (feed), absorbs the core-decay heat, and is released to the containment (bleed) through the power-operated relief valves (PORVs). Feed-and-bleed procedures are of interest because they would use equipment already existing in the plants. The specific steps taken in the feed-and-bleed procedure consist of (1) locking open the pressurizer PORVs, (2) initiating safety-injection (SI) flow, and (3) tripping the reactor-coolant pumps (RCPs).

This study had several objectives. The first was to evaluate the success or failure of feed and bleed for a limited number of PWRs. A detailed evaluation was performed for four plants. They were the Oconee-1 and Calvert Cliffs-1 plants of Babcock and Wilcox (B&W) and Combustion Engineering (C-E), respectively, and the Zion-1 and H. B. Robinson-2 plants of Westinghouse (W). Zion-1 and H. B. Robinson-2 are W four-loop and three-loop plants, respectively. A W two-loop plant was not studied in detail. Existing detailed models of each plant were adapted to study feed-and-bleed transients, and a general system thermal-hydraulics computer code was used to simulate in detail the performance of each plant following selected initiating events. The second study objective was to identify both plant-specific and generic insights about the feed-and-bleed procedure based on the detailed plant analyses performed. The final study objective was to extend the plant-specific conclusions to the broadest possible group of PWRs.

Summary results for our feed-and-bleed studies are presented in Table I. We determined the effectiveness of feed and bleed at two times during total LOFW transients at steam-generator-secondary dryout (SGSD) or loss of secondary heat sink and at primary-system saturation. The effectiveness of feed and bleed was determined either by calculation using detailed plant models (4 plants) or by extending the results using simple inspection (28 plants). Simple inspection applies to plants having characteristics similar to those for which detailed studies have been performed. Insights from the detailed studies are heavily weighted in the inspection process. Similar plants are assumed to perform in the same manner as the plants for which detailed calculations have been performed. Those plants judged too dissimilar are excluded from the process and no extension statements are made for those plants. No statements were provided for two W two-loop plants because they were too far removed from the calculated results to permit extension by simple inspection. Other techniques for extending the calculated results were examined but were not used because they were either ineffective or too costly. We have concluded the following about the feed-and-bleed procedure.

1. Decay heat removal in PWRs follows the loss of normal cooling mode through the steam generators.

2. The availability of high-pressure SI delivery capacity greatly enhances the effectiveness of the feed-and-bleed operation. Plants with only low-pressure or intermediate-pressure SI systems must initiate feed and bleed no later than the loss of secondary heat sink. Plants with high-pressure SI systems can successfully use feed and bleed until the time of primary system saturation.

3. PORV capacity becomes important during the transition from reactor trip to either residual-heat-removal (RHR) or low-pressure-injection (LPI) entry conditions if only safety-grade water supplies are considered. Plants with a single, small PORVs depressurize more slowly than plants with two large PORVs. Safety-grade water supplies may be consumed before RHR or LPI entry conditions are reached. However, recirculation of water from the reactor building sump may be available.

4. Simple inspection is a useful technique for extending the limited set of detailed plant-specific calculations to a broader set of plants. However, we are less confident in the accuracy of our conclusions reached by simple inspection than those based on detailed plant-specific calculations.

TABLE I

SUMMARY RESULTS

VENDOR	PLANT TYPE	CALCULATION	EXTENSION	SGSD	SATURATION
C-E	2 x 4 loop LP SI	Calvert Cliffs-1	Calvert Cliffs-2	Y	N
			Fort Calhoun-1	Y	N
			Maine Yankee	Y	Y
			Millstone-2	Y	N
			Palisades	Y	N
			St. Lucie-1	Y	N
			Ark. Nuclear One-2	NC	NC
B&W	2 x 4 loop HP SI	Oconee-1	Oconee-2, -3	Y	Y
			Ark. Nuclear One-1	Y	Y
			Crystal River-3	Y	Y
			Three Mile Island-1, -2	Y	Y
			Rancho Seco	Y	Y
W	4-loop HP SI	Zion-1	Zion-2	Y	Y
			DC Cook-1, -2	Y	Y
			Trojan	Y	Y
			Salem-1, -2	Y	Y
			Haddam Neck	Y	Y
	4-loop IP SI		South Texas-1, -2	Y	NC
	4-loop LP SI		Indian Point-2, -3	Y	NC
	3-loop HP SI		Summer	Y	Y
			Shearon Harris-1, -2	Y	Y
			Farley-1, -2	Y	Y
			Beaver Valley-1, -2	Y	Y
			North Anna-1, -2	Y	Y
			Surry-1, -2	Y	Y
	3-loop LP SI	Robinson-2	Turkey Point-3, -4	Y	NC
	2-loop IP SI		Prairie Island-1, -2	Y	NC
			Kewaunee	Y	NC
	2-loop LP SI		Ginna	NC	NC
			Point Beach-1, -2	NC	NC

Y = Yes
N = No
NC = No conclusion
LOSHS = loss-of-secondary heat sink

II. INTRODUCTION

The primary method for removal of decay heat from PWRs is through the steam generators to the secondary system using either the main feedwater or auxiliary-feedwater (AFW) systems. The probabilistic risk assessment reported in WASH 1400, later reliability studies, and related experience from the Three Mile Island Unit 2 accident have reaffirmed that the loss of capability to remove heat through the steam generators is a significant contributor to the possibility of core damage.

The NRC currently considers the adequacy of shutdown decay heat removal to be a USI (designated A-45). The purpose of Task Action Plan (TAP) A-45 (Ref. 1) is to "evaluate the adequacy of current licensing design requirements, to ensure that nuclear power plants do not pose an unacceptable risk because of failure to remove shutdown decay heat." A major part of TAP A-45 is concerned with the transition from reactor trip to a hot holding condition. Also of interest is the transition to hot shutdown and maintaining hot-shutdown conditions. Although a limited number of alternative means for removal of shutdown decay heat from PWRs is being examined by the NRC, this report focuses on activities at the Los Alamos National Laboratory to investigate the application of the "feed-and-bleed" concept as a diverse alternative method of removing decay heat that does not rely on the use of the steam generators. "Feed and bleed" refers to direct primary liquid injection through the SI or after the non-SI systems and a controlled (manual) depressurization through the PORVs on the pressurizer.

A. Success Criteria

Because there is interest in the plant transition from reactor trip to both a hot pressurized holding condition and to hot shutdown (entry conditions for either the RHR or LPI systems), we have identified criteria to measure the success or failure of a feed-and-bleed procedure in each case.

For transition from reactor trip to a hot, pressurized holding condition, we define the first success criterion as attainment of a stable primary system having the following three characteristics. First, the primary-system pressure is above the actuation pressures for both the LPI system and accumulators. Second, primary system and vessel mass inventories are stable or increasing. Third, fuel-rod cladding temperatures are near or below the primary saturation temperature with no departure from nucleate boiling (dryout). In the remainder of this report, we denote success or failure in transition from reactor trip to a hot, holding condition by S1 (satisfaction of the first success criterion) or F1 (failure of the first success criterion), respectively. This is a short-term success criterion in that further actions must be taken within a limited time to insure long-term success. Such actions include restoration of secondary-side cooling or initiation of suction from the containment sump combined with pumping the sump fluid to the high-pressure injection (HPI) system inlet pressure.

For transition from reactor trip to hot shutdown, we define a second success criterion as completion of a controlled primary-system depressurization and cooldown to achieve conditions permitting long-term cooling using either the RHR system or the LPI system taking suction from the containment sump. In the remainder of this report, we denote success or failure in transition from reactor trip to hot shutdown by S2 (satisfaction of the second success criterion) or F2 (failure of the second success criterion).

B. Report Objectives

The objectives of this report are as follows:

1. to predict and document the expected performance of specific PWRs subjected to a feed-and-bleed procedure following initiating events that result in a total loss of feedwater;

2. to develop and document both plant-specific and generic insights obtained from the specific plant analyses; and

3. to extend, where possible, the insights to establish the feasibility of feed and bleed on a generic basis for a larger group of PWRs for which specific, detailed analyses have not been performed.

Although stating the obvious, we believe it is important to emphasize that the results reported in fulfillment of each objective have different confidence levels. We have the highest confidence in the plant-specific results for which detailed calculations have been made. We have the least confidence in the extension statements. Our confidence in plant-specific and generic insights lies between the two extremes. We caution readers that the extension of plant-specific insights to other plants of the same vendor requires careful review of specific plant performance characteristics and operational limitations. We view our extension statements as providing a useful focus for initial review and dialogue, not as an end statement of fact. Specific plant operational characteristics will need to be carefully evaluated.

C. Technical Approach

The formal base for all the results included herein is a set of plant-specific calculations performed using the Transient Reactor Analysis Code (TRAC), versions PD2 (Ref. 2) and PF1 (Ref. 3). We have performed calculations for one or more plants of each US PWR vendor: B&W, C-E, and W. The specific plants for which TRAC-PF1 calculations were performed are Oconee-1 (B&W), Calvert Cliffs-1 (C-E), and Zion-1 and H. B. Robinson-2 (W). Audited plant models were available for each of the four plants and were used in this study. For each calculation we have assumed that the equipment, instrumentation, and procedures are available and function as specified. These assumptions may differ from actual conditions in the plants studied in detail.

We have sought to follow a consistent approach in analyzing the capability of specific plants to remove decay heat using a feed-and-bleed operation. The primary analysis tool used was the TRAC-PF1 computer code. A brief description of this code is provided in Appendix A. A common set of initiating events for loss-of-secondary cooling has been identified and investigated for each plant. These events consist of (1) an LOFW event induced by a loss of offsite power (LOSP), (2) a LOFW event, (3) a combined main-steamline break (MSLB) and LOFW, (4) a combined main-feedwater-line break (MFLB) and LOFW, and (5) a combined single-tube steam-generator tube rupture (SGTR) and LOFW event. For each event, three postulated transients were investigated, although not necessarily calculated. First, a base-line transient was determined for which there is no actuation of the SI or other non-SI system and no operator intervention. This transient, which leads to core dryout, establishes the timing of critical events such as steam-generator dryout, primary-system saturation, containment overpressure, and the start of core heatup. For the base-case transient, RCPs were left on throughout the transient, except for the LOSP event in which the RCPs tripped on LOSP. Current PWR operator guidelines specify leaving the RCPs on except for small break loss-of-coolant accidents (LOCAs).* The second transient evaluates plant thermal-hydraulic performance considering feed-mode operation after the SI signal, usually containment overpressure. We define "feed" as a limited mode of feed-and-bleed cooling for which emergency core coolant (ECC) injection is actuated and the PORV cycles as designed to control the primary-system pressure. The feed-injection flow is a function of primary-system pressure. The third transient evaluates the effectiveness of a feed-and-bleed procedure. Feed-and-bleed cooling is effected by starting ECC injection (feed) and by manually opening the PORVs on top of the pressurizer to reduce primary system pressure (bleed).

We have not performed a detailed TRAC calculation for each initiating event in the common transient set. Rather, we have chosen to conserve resources by eliminating specific calculations when we have sufficient information to make a reasoned conclusion.

III. OCONEE-1 INSIGHTS

Oconee-1 (Ref. 5) is a B&W PWR operated by the Duke Power Company. The design power of the reactor is 2568 MWt. The reactor-coolant system consists of the reactor vessel, two vertical once-through steam generators (SGs), four shaft-sealed RCPs, an electrically heated pressurizer, and interconnecting piping. The primary coolant system is arranged in two heat-transport loops, each with two RCPs and one steam generator. The reactor coolant is transported through piping connecting the reactor vessel to the steam generators and flows downward through the steam-generator tubes, transferring heat to the steam and water on the secondary-shell side of the steam generator. In each loop, the reactor coolant is returned through two lines, each containing a RCP, to the reactor vessel.

The steam generator is a once-through, vertical, straight-tube, tube-and-shell heat exchanger that produces superheated steam at constant pressure over the power range. Reactor coolant flows downward through the tubes and transfers heat to generate steam on the shell side. Main feedwater enters the tube bundle region via a downcomer annulus. Following a reactor trip, AFW is supplied to the steam generator through a sparger ring located at the top of the steam generator to assure natural circulation of the reactor coolant following the unlikely event of the loss of all RCPs. Oconee-1 has the following ECC system equipment: (a) the HPI system, of which a low flow portion is used in normal reactor operation, (b) the LPI system that operates for shutdown cooling, and (c) core-flooding tanks (CFTs), a passive system normally ready for operation. During normal reactor operation, the HPI system continuously recirculates reactor coolant for purification and for supply of seal water to the RCPs. The ECC HPI system is initiated at (a) a low primary pressure of 10.3 MPa (1500 psig) or (b) a reactor-building pressure of 0.028 MPa

(4 psig). Automatic actuation of the valves and pumps switches the system to the emergency operating mode to deliver water from the borated water storage tank (BWST) into the reactor vessel through the reactor coolant inlet lines. The ECC HPI system is intended primarily for small-break LOCAs.

The LPI system, designed to maintain core cooling for larger break sizes, operates independently of and in addition to the HPI system. Automatic actuation of the LPI system is initiated at (a) a primary-system pressure below 3.45 MPa (500 psig) or (b) a reactor-building pressure of 0.028 MPa (4 psig). LPI is accomplished through two separate flow paths, each including a pump and a heat exchanger and terminating at the reactor-vessel flooding nozzles located on opposite sides of the vessel. LPI flow is injected into the downcomer region. The initial operation of the LPI system involves pumping water from the BWST into the reactor vessel. When the BWST is ~94% empty, a low-water-level alarm is annunciated in the control room and the suction valve from the reactor-building emergency sump is manually opened, permitting recirculation of the spilled reactor coolant from the reactor-building emergency sump.

The core-flooding system provides core protection continuity for intermediate and large primary-system pipe failures by flooding the core when the primary pressure drops below 4.14 MPa (600 psig). The core-flooding system is self-contained, self-actuating, and passive in nature. The discharge pipe from each CFT is attached directly to a reactor-vessel core-flooding nozzle. Each core-flooding line at the outlet of the CFTs contains an electric motor-operated stop valve adjacent to the tank and two-line check valves in series. The stop valves at the core-flooding tank outlet are fully open during reactor power operation. Valve position indication is shown in the control room.

A schematic of the reactor-coolant system is presented in Fig. 1. The TRAC-PF1 input model of the Oconee-1 plant is described in Appendix B.I. For all nominal Oconee-1 calculations we assumed two HPI pumps. The SI delivery characteristics were based on best-estimate values obtained from the utility. One PORV was modeled for each calculation with the exception of one parametric case discussed in Sec. III.A.1. We used the 1973 American Nuclear Society decay heat curve.

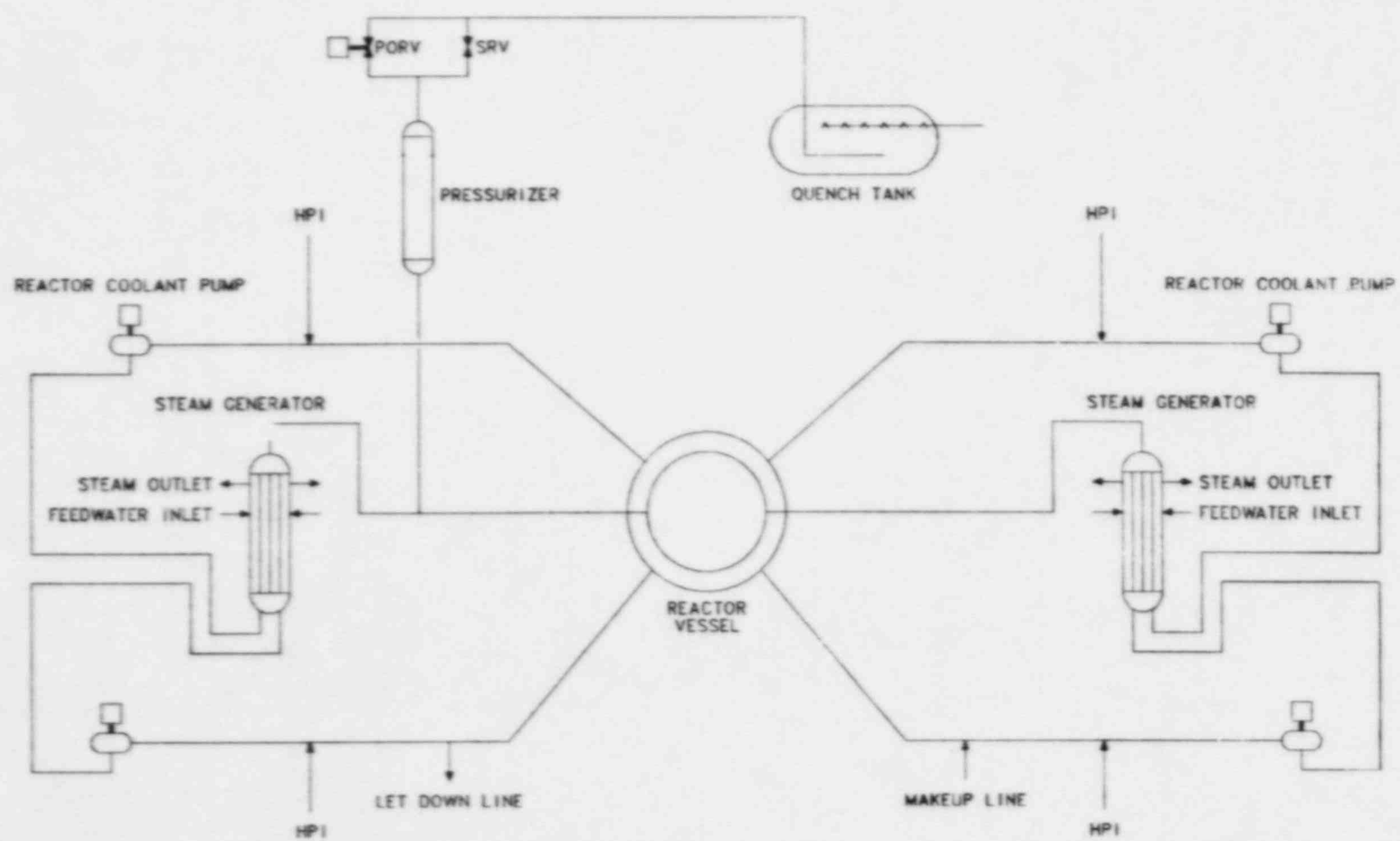

Fig. 1.
Oconee-1 reactor-coolant system.

A. Summary Results

We have prepared detailed reports of a series of studies that examined LOFW transients in the Oconee-1 plant.[6-9] In the following we present the summary results for loss-of-feedwater transients in the Oconee-1 plant. As previously discussed, there are five events considered. These are the LOSP-induced LOFW event, LOFW event, combined MSLB/LOFW event, combined SGTR/LOFW event, and combined MFLB/LOFW events. For each event, the base case, feed case, and feed-and-bleed case results will be discussed. The results of our Oconee-1 calculations are summarized in Fig. 2. Although success criterion S1 is written for a feed-and-bleed procedure, S1 often can be satisfied by a feed operation. Figure 2 also shows the period in which the feed or feed-and-bleed operation was initiated relative to key reactor events. The success criteria were satisfied for each non-base case calculated, the latest feed and feed-and-bleed procedures being initiated at the time of primary-system saturation.

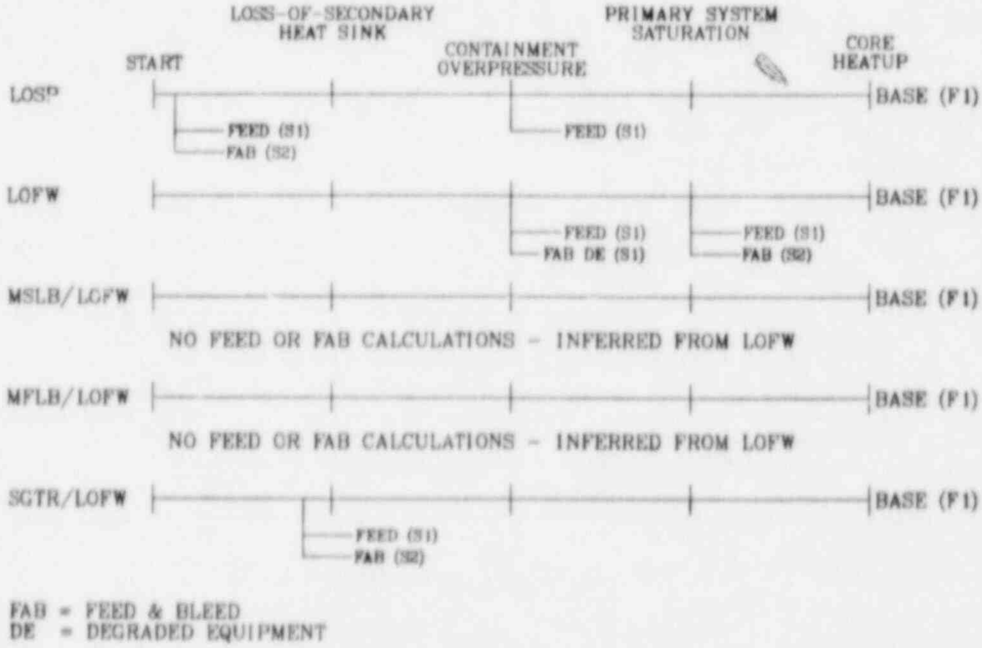

Fig. 2.
Oconee-1 success/failure chart.

1. LOSP-Induced LOFW Event. The LOFW transient was initiated by an LOSP event. It was assumed that the LOSP results in an immediate trip of the RCPs, gravity insertion of the control rods beginning at 0.5 s with complete insertion by 1.5 s, and immediate closure of the turbine stop valves (TSVs). The steam generator will dump via the safety relief valves (SRVs) or the turbine bypass valves if the main condenser cooling is not lost. It was assumed initially that feedwater flow dropped to zero instantly at the start of the transient and that no auxiliary feedwater was available. The effect of a finite coastdown of main feedwater and restoration of auxiliary feedwater was also examined.

a. Base Transient. The event sequence for the base transient in which the HPI fails to deliver water is given in Table II. Closure of the TSVs and interruption of main feedwater resulted in secondary-system pressurization to 7.23 MPa (1049 psig) and opening of the steam-line atmospheric relief valves (ARVs) at 2.1 s. Failure of the AFW system left the steam generators unable to

TABLE II

OCONEE-1
LOSP EVENT SEQUENCE FOR BASE-CASE TRANSIENT (NO HPI)

Time (s)		
No MFW	4.6 Full Power Equivalent Seconds of MFW	Event
0	0	Loss of offsite power Close TSVs RCPs coast down
0.5	0.5	Trip reactor
-	2.0	TSVs open
2.1	-	ARVs open
494	768	PORV opens
~70	~70	Minimum primary pressure
~600	~1000	SGs empty
950	1200	Pressurizer full
1654	1950	Containment overpressure
1800	2100	Voids in core
2200	2500	SRVs open
2500	2800	Tops of candy canes voided
3000	3300	Rapid heating of core
3400	3700	Core empty
3600	4700	End of calculation

remove decay heat by 70 s. This is shown in Figs. 3 and 4, which give the average pressure and liquid temperatures in the core region. Expansion of the primary coolant resulted in the PORV opening at 494 s, followed by filling the pressurizer steam space at 950 s with the PORV cycling on setpoint. Steam flowing from the primary system through the PORV pressurized the containment building to 0.128 MPa (4 psig) at 1654 s. This resulted in an ECC actuation signal at 1654 s. In this sequence, we assumed that the HPI system failed to deliver any water. Thus, water began boiling in the core region at 1800 s, and, as can be seen in Fig. 4, by 2200 s the core region and upper portions of the primary system were saturated. Saturated expansion of the primary coolant opened the SRVs. At 3000 s, the fuel-rod temperature began to increase rapidly (failed criterion 1, F1) as the core dried out (see Fig. 5) and at 3400 s the core was empty. The calculation was terminated at 3600 s.

To determine the effect of main-feedwater coastdown, an auxiliary calculation was made. The main feedwater in the auxiliary calculation decreased to zero in 16 s, using a realistic exponential decay that delivered an extra 3270 kg (7194 lb) of water to the steam generators, sufficient to remove 4.6 full-power seconds from the reactor. With the additional water being supplied

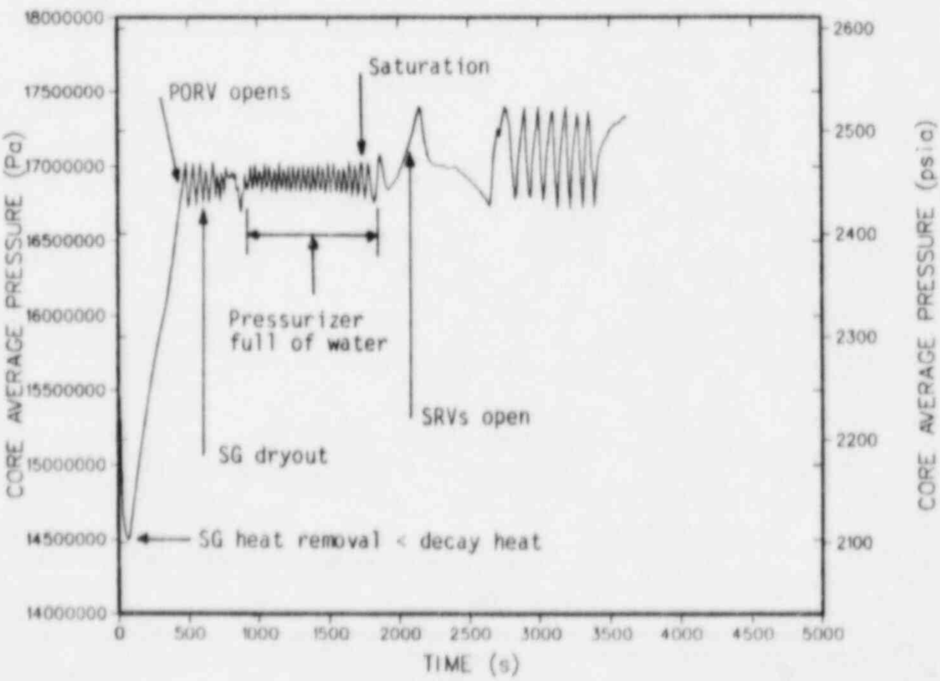

Fig. 3.
Primary-system pressure during the LOSP/LOFW event for base transient.

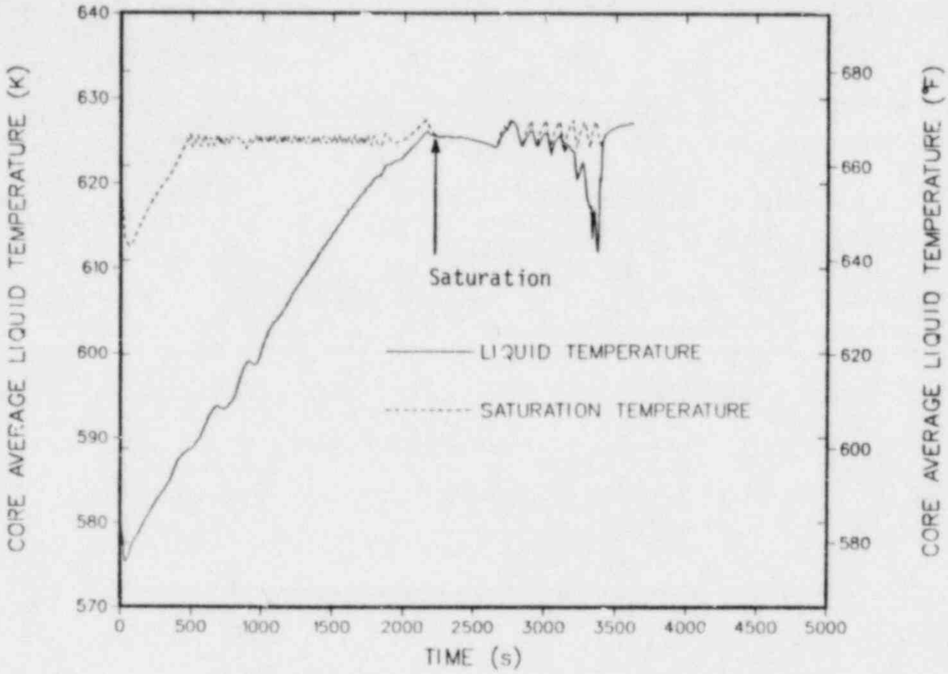

Fig. 4.
Core-liquid temperatures during the LOSP/LOFW event for base transient.

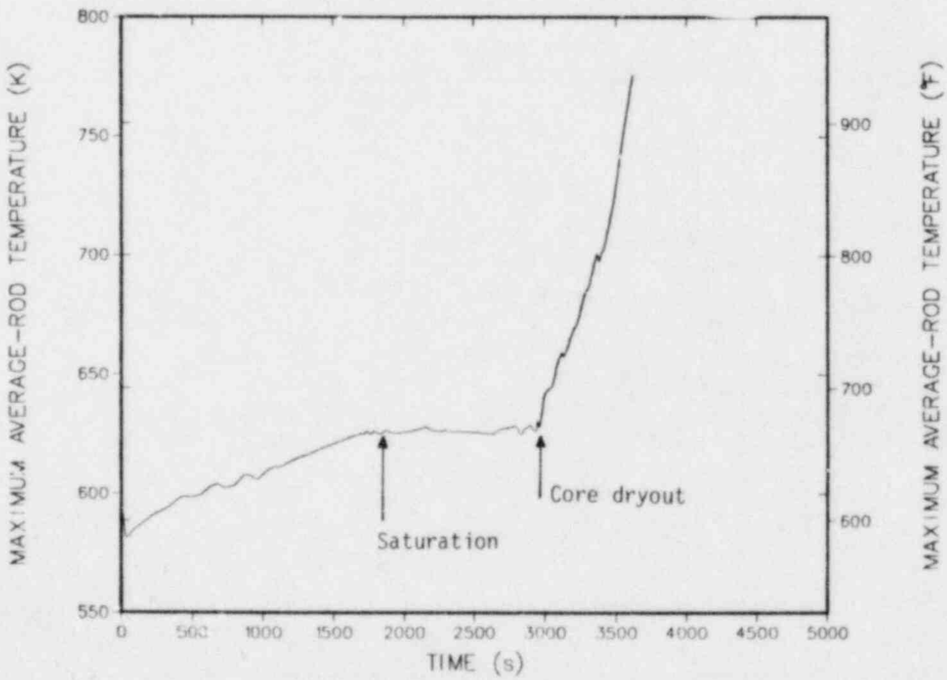

Fig. 5.
Cladding temperature during the LOSP/LOFW event for base transient.

to the steam generator, the primary-system heatup was delayed by approximately 300 s. This can be seen in Table I, which compares the base case (zero feedwater) to the feedwater-coastdown case. A 300-s delay in system heatup is consistent with the 4.6 full-power second energy-removal capability of the additional feedwater.

b. Feed Transient. The event sequence for the feed-only transient in which the HPI system automatically delivers water following containment overpressure is given in Table III. The sequence is the same as the base case LOSP-induced LOFW transient until the containment overpressure signal occurred at 1654 s. The system at 1654 s was subcooled and full. Two HPI pumps delivered sufficient water to avert saturation, replace water lost through the

TABLE III

OCONEE-1
LOSP EVENT SEQUENCE FOR FEED ACTUATED BY CONTAINMENT OVERPRESSURE SIGNAL

Time (s)		
No MFW	4.6 Full Power Equivalent Seconds of MFW[a]	Event
0	0	Loss of offsite power, close TSVs, RCPs coast down
-	16	Feedwater ceases
0.5	0.5	Trip reactor
2.1	-	ARVs open
494	768	PORV opens
~600	1000	SGs empty
950	1200	Pressurizer full
1654	1950	HPI actuated
5000		End of calculation, system full and subcooled

[a]Estimated based on Table I.

PORV, and cool the core. This can be seen in Fig. 6, which shows the core average liquid and saturation temperatures. At the end of the 5000-s calculation the primary system was at 606 K (632°F), subcooled by 20 K (36°F), and cooling at a rate of 0.00407 K/s (0.00733°F/s). Thus, it is evident that the Oconee-1 HPI system has sufficient capacity to satisfy the first success criterion (S1) in the feed mode. We also examined a feed sequence initiated at 120 s. The first success criterion, S1, was satisfied for this case also.

c. Feed-and-Bleed Transient. The event sequence for feed and bleed initiated by the operator is given in Table IV. The feed-and-bleed sequence was initiated at 120 s when the operator actuated the HPI system (two HPI pumps used) and opened the PORV. This is a very early operator intervention, but does represent the maximum effect of early feed-and-bleed initiation.

The pressure, as pictured in Fig. 7, decreased starting at 120 s until the pressurizer filled at 600 s. The pressure then increased until the cooling rate equaled the decay power at approximately 1200 s. At this point the pressure decreased, continuing over the next 3000 s until, at approximately 4500 s, it leveled off at 10.7 MPa (1550 psia). This behavior can be explained by

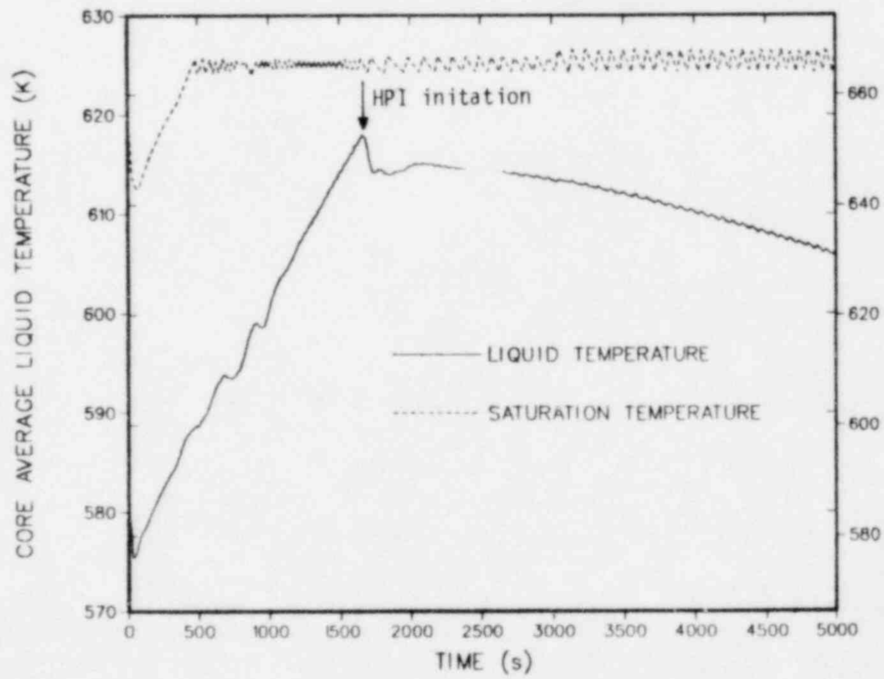

Fig. 6.
Core-liquid temperatures during the LOSP/LOFW event for feed initiated at containment overpressure.

TABLE IV

OCONEE-1
LOSP EVENT SEQUENCE FOR FEED AND BLEED INITIATED
AT 120 s WITH 2 HPI PUMPS

Time (s)	Event
0	Loss of offsite power, zero feedwater, RCPs coast down, TSVs closed
0.5	Trip reactor
2.1	ARVs open
120	Initiate feed and bleed
-	SGs empty
600	Pressurizer full
120	HPI actuated
-	Boiling in core
-	SRVs open
-	Tops of candy canes voided
16000	End of calculation
31000[a]	BWST empty

[a]Determined by the simple model.

variations in the PORV flow. The pressure in a liquid-full system is determined by the HPI pump characteristics. The pressure sought is that which achieves a balance of volumetric flow. According to the Burnell choked-flow model[10] that is used by TRAC-PF1, the mass flow out the PORV increases with liquid subcooling. At approximately 4500 s, the flow ceases to be choked and is therefore no longer dependent upon the subcooling. The PORV flow then depends only upon frictional losses through the valve. Further subcooling, therefore, resulted in no additional decrease in the pressure. As can be seen in Fig. 8, liquid in the hot leg is approximately 60 K (108°F) subcooled at 4500 s when the

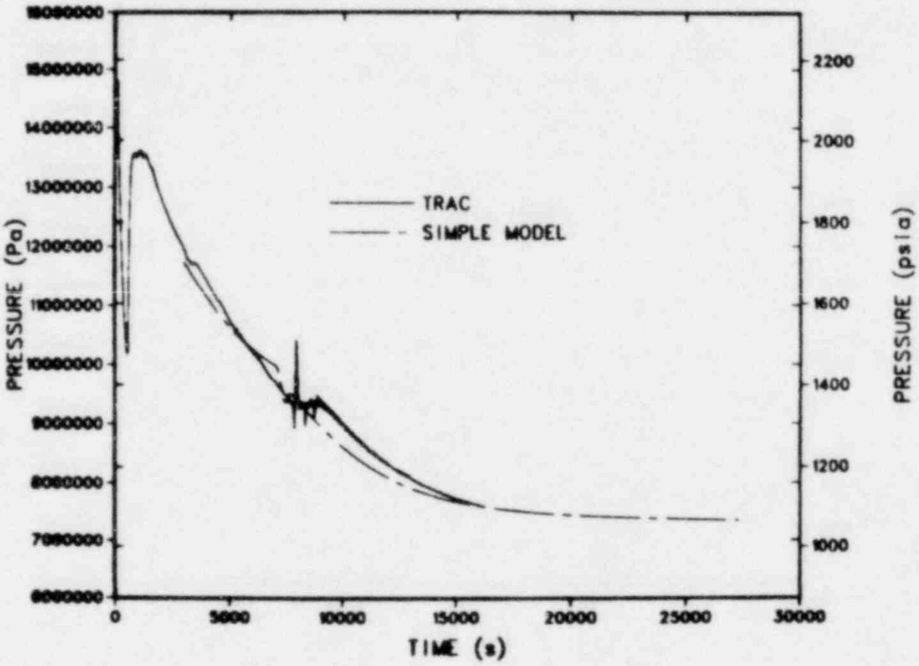

Fig. 7.
Primary pressure during the LOSP/LOFW event for feed and bleed initiated at 120 s.

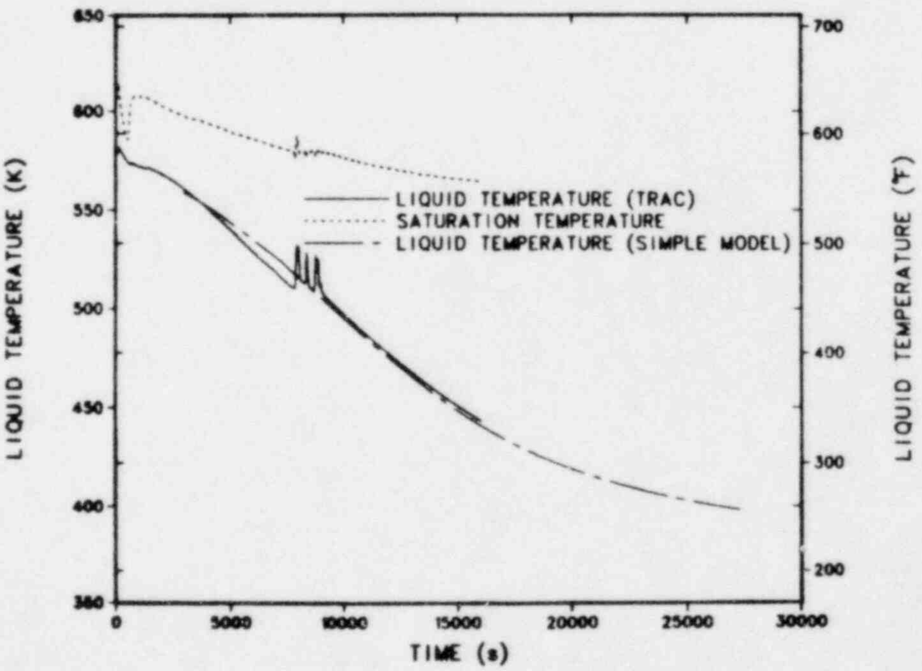

Fig. 8.
Hot-leg temperatures during the LOSP/LOFW event for feed and bleed initiated at 120 s.

TRAC-PF1 models predict that choking at the valve ceased. With the primary system at 10.7 MPa (1550 psia), continued HPI flow cools the primary system. This can also be seen in Fig. 8, which shows the cooling rate increasing at approximately 1200 s. Cooling and depressurization then continued to the end of the TRAC-PF1 calculation at 16000 s. To continue the simulation to the end of once-through cooling (depletion of the BWST), we used a fast-running simplified model. With this model adequate simulations of a liquid-full system can be performed and used to extrapolate TRAC-PF1 results. The results of the simple model are included in Figs. 6 and 8. It can be seen that the simple model matches the TRAC results quite well, certainly well enough for performing extrapolations. The simple model was run until the BWST was depleted at 27000 s (7.8 h). The operator must consider further action sometime before 7.8 h to establish a stable and permanent cooling mode. Thus, the feed-and-bleed procedure was effective in cooling the Oconee-1 reactor for 7.8 h. This represents satisfaction of the first success criterion S1.

The use of a feed-and-bleed operation with HPI throttling to cool and depressurize the reactor-coolant system to RHR design operating conditions was also examined. The maximum design pressure and temperature at which the RHR system can be operated are 2.5 MPa (350 psig) and 420 K (300^{o}F), respectively. To determine if these conditions can be obtained, a simulation of a controlled throttling of the HPI system was begun at 3600 s into the feed-and-bleed transient. B&W guidelines direct the operator to cool and depressurize the reactor at a rate such that both an adequate subcooling margin is maintained and pressurized-thermal-shock thresholds are not exceeded. Several calculations were made, and in each case it was not possible to cool and depressurize the reactor before depleting the BWST inventory. Upon receipt of a signal that BWST inventory depletion was near, the operator would switch to a recirculation mode drawing water from the containment sump and recirculating the water through the reactor using the LPI and HPI pumps. Thus, the depletion of BWST inventory mandates several valve realignments. Assuming the operator is able to effect these operations, a successful cooldown and depressurization to RHR-system operating conditions, S2, can be achieved provided the PORV relief capacity is doubled. The effect of an increased PORV relief capacity was also examined. Doubling the PORV relief capacity was sufficient to reach RHR cooling mode before depleting BWST inventory.

2. LOFW Event. The transients discussed in this section were initiated by a loss of main feedwater, with offsite power available. In the LOFW events, the RCPs remain on until manually tripped by the operators after HPI is initiated. The RCPs remain on throughout the base transient. It was assumed that the feedwater flow dropped to zero instantly at the start of the transient and that no auxiliary feedwater was available. The loss of feedwater resulted in an increase in the primary-system temperature and pressure. An overpressure reactor trip was assumed when the pressurizer pressure exceeded 16.0 MPa. This resulted in an insertion of -0.054 Δk control-rod reactivity in 1 s after a 0.5-s delay in closure of the TSVs. A summary event chart for the Oconee-1 LOFW calculations is presented in Fig. 9. Key reactor events and phenomena are presented.

a. Base Transient. The event sequence for the base transient in which the HPI system fails to deliver water is given in Table V. As can be seen in Fig. 10, the primary system pressure began to increase immediately following the loss of feedwater at the start of the transient. At 10.6 s the pressurizer pressure exceeded 16.0 MPa (2320 psia), resulting in a reactor-trip signal. The closure of the TSV caused the secondary pressure to increase to the steam-line ARV setpoint (7.23 MPa, 1050 psia) at 14 s. The primary pressure and temperature (Figs. 10 and 11) then decreased until 60 s, when the decay energy exceeded the energy removed by the steam generators and the pressure and temperature began to increase. Primary liquid expansion compressed the steam at the top of the pressurizer, and at 140 s the PORV opened. Steam flow from the PORV maintained the system pressure at approximately 17.0 MPa (2465 psia). At approximately 200 s, the steam generators were voided. At 400 s, primary liquid expansion filled the steam space in the pressurizer and liquid began to flow out of the PORV.

Although the mass flow was higher, the volumetric flow was insufficient to prevent further system pressurization and the SRVs opened briefly at 500 s. At 900 s, 17600 kg (38720 lb) of steam (enough to result in a containment overpressure signal) had flowed into the containment building, generating an SI signal. For this case, it was assumed that the HPI system failed to deliver any water. Continued heating of the primary liquid resulted in saturation of the entire top of the system and the core began to void. The entire system was saturated at approximately the same time because the RCPs were left running, keeping the system well mixed. Saturated expansion resulted in opening the

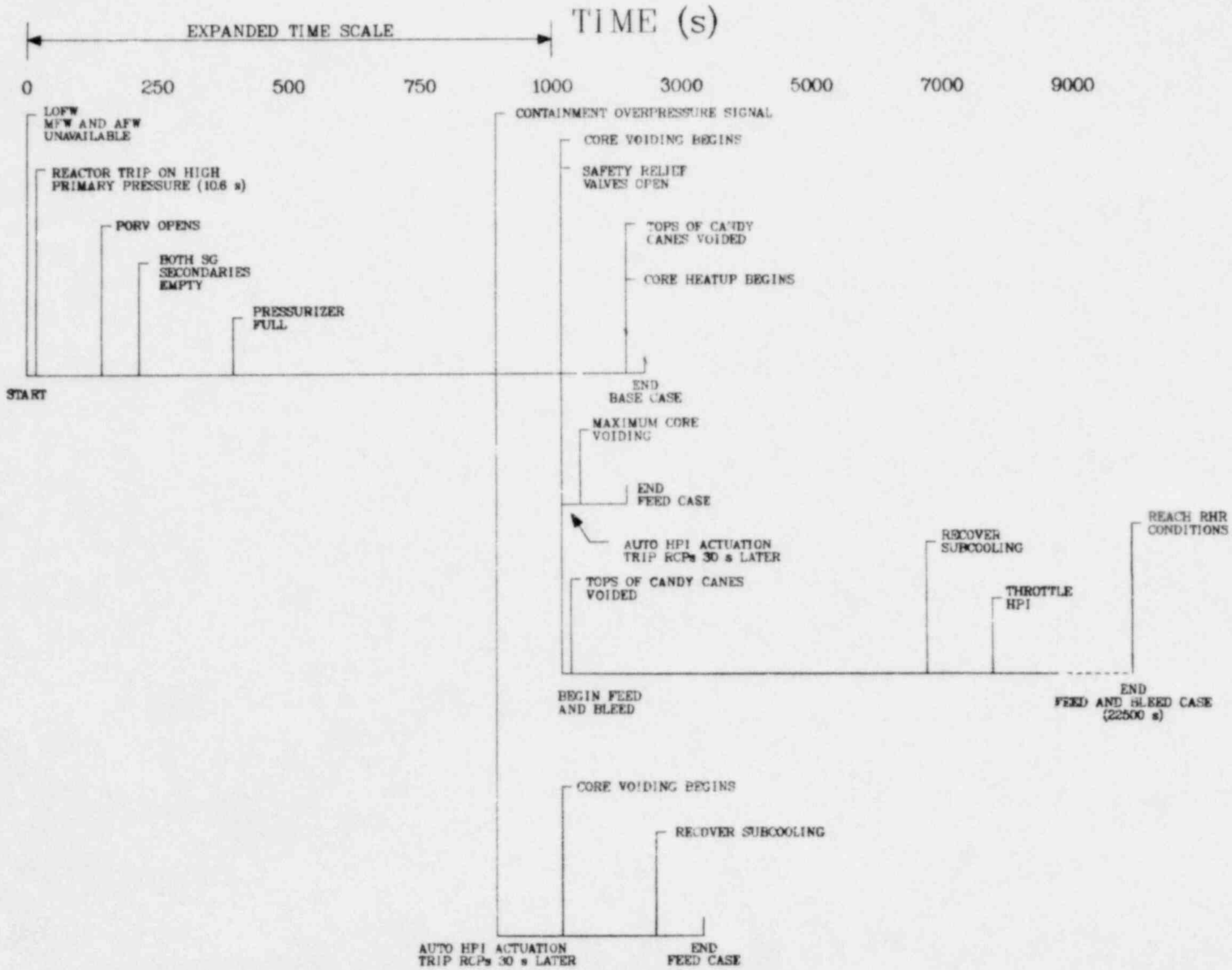

Fig. 9.
Oconee-1 LOFW event line chart (nominal equipment).

TABLE V

OCONEE-1
LOFW EVENT SEQUENCE FOR BASE-CASE TRANSIENT (NO HPI)

Time (s)		
No MFW	4.6 Full Power Equivalent Seconds of MFW[a]	Event
0	0	Zero feedwater
10.6	10.5	Trip reactor, close TSVs P > 16 MPa (2315 psia)
14	-	ARVs open
60		Decay power > steam generator removal
140	440	PORV opens
200	500	SGs empty
~400	700	Pressurizer full
500		SRVs open briefly
900	1200	Containment overpressure
1200	1500	Voids in core
1200		SRVs open
2200	2500	Tops of candy canes voided
2200		Rapid heating of core
2200		Core empty
2340		End of calculation

[a]Estimated based on Table I.

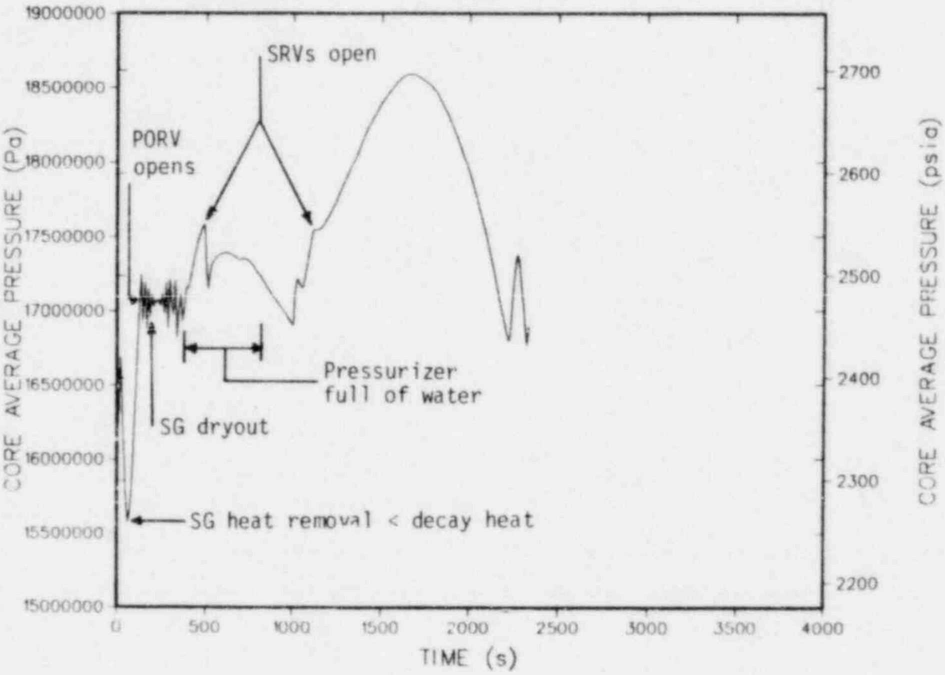

Fig. 10.
Primary pressure during the LOFW event for base transient.

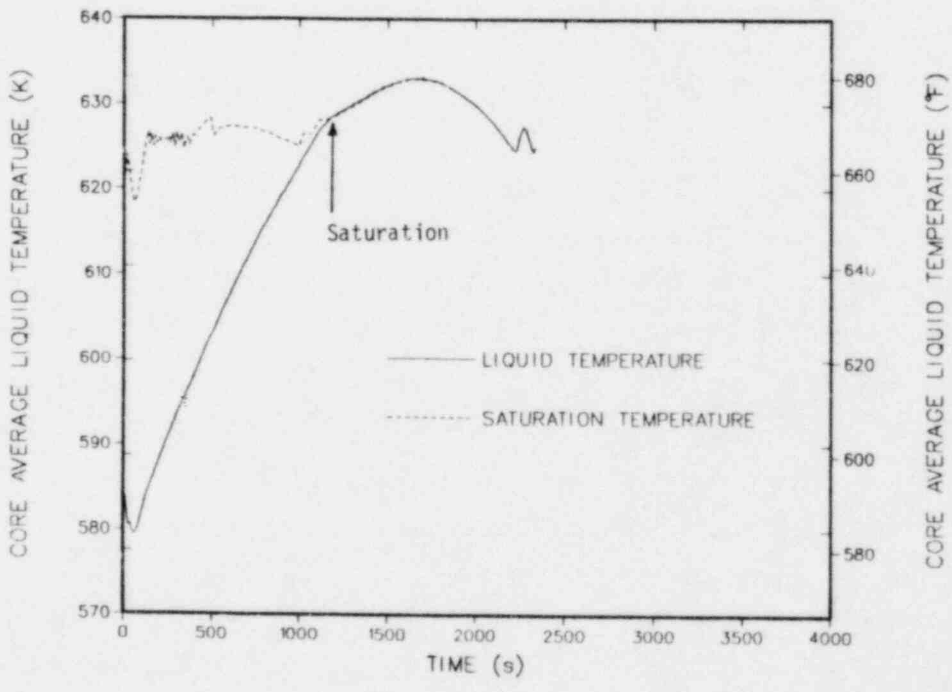

Fig. 11.
Primary temperature during the LOFW event for base transient.

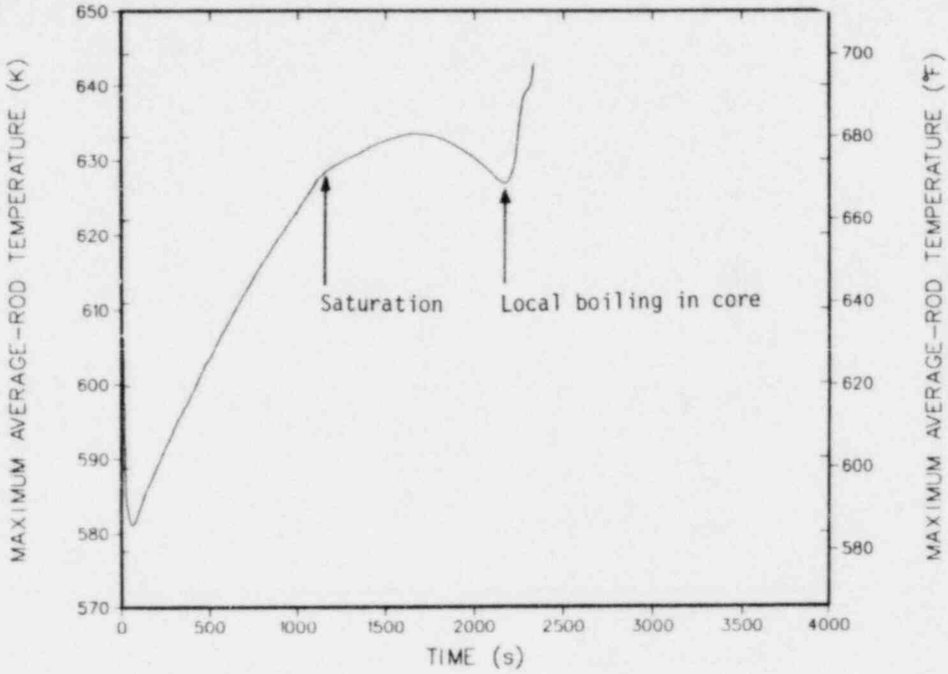

Fig. 12.
Cladding temperatures during the LOFW event for base transient.

SRVs. System voiding continued until 2200 s, when the core was empty and rapid heating of the fuel rods ensued (Fig. 12). Failure of the first success criterion, F1, is evident at the time the cladding departs from saturated fluid cooling.

b. Feed Transient. The event sequence for the feed transient in which the HPI system automatically delivers water following containment overpressure is given in Table VI. The sequence is identical to the base case until the SI signal is generated at 900 s. Additionally, it was assumed that the operator manually tripped the pumps at 930 s as specified by plant operating procedures. The liquid and saturation temperatures in the hot leg are shown in Fig. 13. Saturation of the higher temperature, higher elevation portions of the primary system was averted and the maximum liquid temperature occurred at approximately 1500 s. Although saturation of the entire system was averted, some boiling did occur beginning at 1200 s at the top of the core region. Thus, some vaporization of the HPI liquid was necessary for decay energy removal. Note that for the LOSP-induced LOFW events, no boiling was necessary for cooling and a hot-leg subcooling margin was maintained when the reactor was tripped immediately. The major integrated power difference between this case and the

TABLE VI

OCONEE-1
LOFW EVENT SEQUENCE FOR FEED COOLING ACTUATED AUTOMATICALLY

Time (s) No MFW	Time (s) 4.6 Full Power Equivalent	Event
0		Zero feedwater
10.6		Trip reactor, close TSVs P > 16 MPa (2315 psia)
14		ARVs open
60		Decay power > steam generator removal
140		PORV opens
200	500	SGs empty
~400	700	Pressurizer full
500	800	SRVs open briefly
900	1200	HPI actuated by containment overpressure
930	1230	Trip RCPs
1200-2900	1500	Voids in core
~2900		Recovery of subcooling
3490		End of calculation

LOSP-induced LOFW event was the extra 10-s period at full power until a trip signal occurred on system overpressure.

The system pressure, shown in Fig. 14, remained near the PORV setpoint pressure, with the PORV open, until approximately 2500 s. Except for the brief opening at 500 s, the SRVs were not necessary for system pressure control. At 2500 s, the PORV began to cycle open and shut to maintain the system pressure at approximately 17.0 MPa (2465 psia). Boiling in the core region ceased at 2900 s, and the system was cooled by sensible heat increase of the HPI liquid

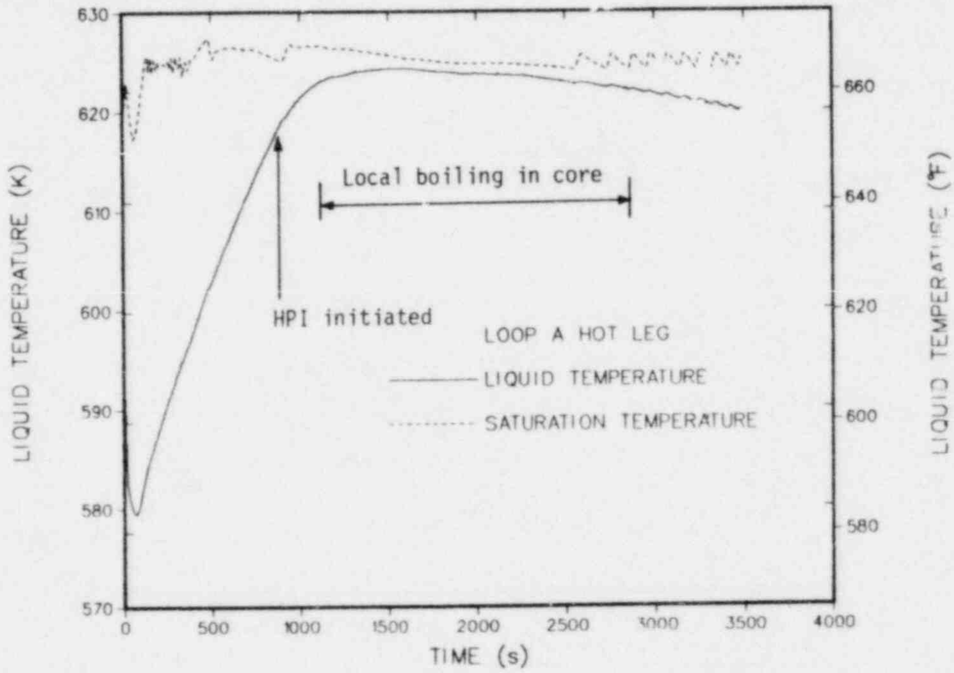

Fig. 13.
Primary hot-leg temperatures during the LOFW event for feed initiated at containment overpressure.

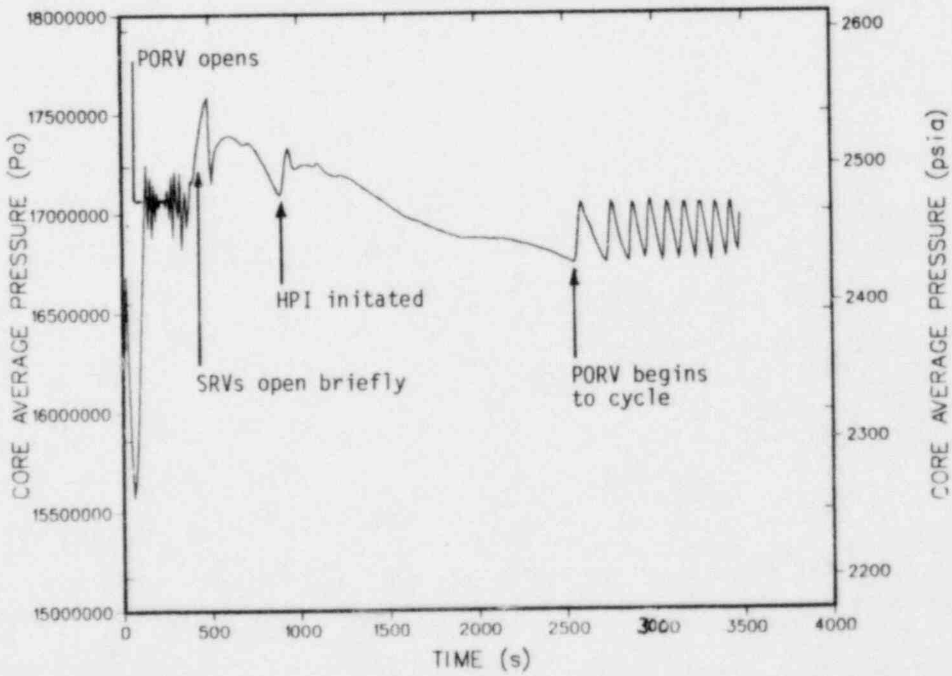

Fig. 14.
Primary pressure during the LOFW event for feed initiated at containment overpressure.

alone. Cooling of the liquid-full system began at this time. Given the rate of flow at the end of the calculation, once-through HPI water will last an additional 53000 s (15 h). The operator thus has 15 h to establish some other mode of cooling. For this transient the Oconee-1 HPI has sufficient capacity to cool the plant in the feed mode (S1).

c. Feed-and-Bleed Transient. The use of a feed-and-bleed operation following a LOFW event will now be discussed. The transient can be considered to be a transient for which the operator initiates a feed-and-bleed operation at 1200 s, the time of primary system saturation. Prior to 1200 s, the transient is identical to the base LOFW transient wherein the primary system has saturated and the SRVs have opened to relieve the primary system. The SRVs are challenged because Oconee-1 has only a single PORV and that has a small relief capacity relative to the other plants discussed in the remainder of the report. For this transient (initially a feed transient) we initiated the HPI at 1200 s. The RCPs were tripped 30 s later. However, the PORV was already open at the time and stayed open, except for several brief intervals, until the PORV was locked open at 2100 s. At 2100 s the conditions for S1 satisfaction of the first success criterion was established. The PORV behavior is shown in Fig. 15. Except for two brief periods between 1200 s and 2100 s, the PORV was fully open to meet the relief demands associated with system coolant expansion. Thus, this transient is considered to be a late application of feed and bleed (at the time of system saturation).

The event sequence for this case is given in Table VII. Following HPI actuation and tripping the RCPs, the phases separated and the tops of the loops voided by 1360 s. By 2000 s, the SRVs were no longer necessary to control system pressure and they stopped opening. At 2100 s, it was assumed that the operator latched the PORV open. This action made no immediate difference because the PORV was already fully open at 2100 s. As can be seen in Fig. 16, the pressure began to decrease at approximately 3000 s as a result of the open PORV. Depressurization continued and the core region liquid levels increased until the core was refilled at about 6900 s. At 7000 s, core cooling by water flow alone was sufficient and boiling ceased in the core region. This is also evidenced in Fig. 17, which shows the liquid becoming subcooled in the hot leg at 7000 s.

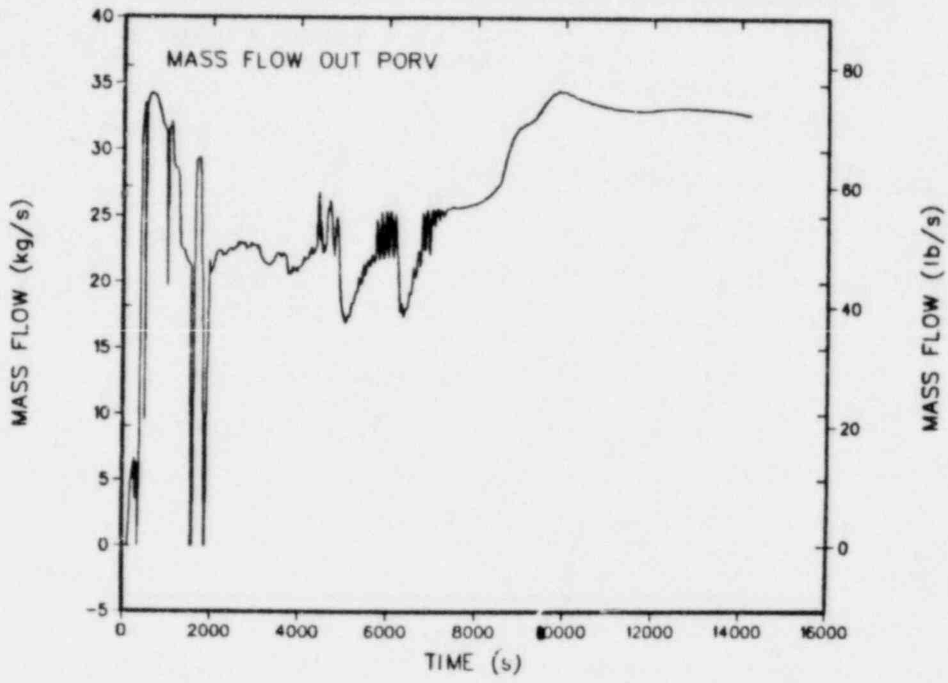

Fig. 15.
PORV mass flow rate during the LOFW event for feed and bleed at primary-system saturation.

For this calculation, it was assumed that when the subcooling in a hot leg reached 28 K (50°F), the operator began to throttle the HPI inflow to depressurize the system. This occurred at approximately 8100 s, as can be seen in Fig. 17. Throttling the HPI resulted in a steady depressurization cooling as shown in Figs. 16 and 17. The calculation was terminated at 14250 s.

The temperature and pressure were extrapolated to see if conditions would be obtained at which the RHR system could operate. The pressures and temperatures in Figs. 16 and 17 were linearly extrapolated to 25000 s, and the combined pressure-temperature trace is plotted on Fig. 18, which appears in the B&W operator guidelines.[11] It can be seen that the throttling technique employed maintained the pressure-temperature relationship required by the guidelines when no RCPs are running, as in this case. It can also be seen that RHR conditions can be achieved at approximately 22500 s (fulfillment of success criterion S2). There was sufficient water in the BWST to complete this cooling and depressurization process. As previously discussed, it was not possible to reach this condition on BWST inventory alone using a feed-and-bleed operation following a LOSP-induced LOFW event. The difference was that the system was

TABLE VII

OCONEE-1
LOFW EVENT SEQUENCE FOR FEED-AND-BLEED COOLING
WITH SYSTEM PARTIALLY VOIDED

Time (s)	Event
0	Zero feedwater
10.6	Trip reactor
14	ARVs open
60	Decay power > steam-generator removal
140	PORV opens
200	SGs empty
~400	Pressurizer full
500	SRVs open briefly
1200	HPI actuated
1200	Voids in core
1200	SRVs open
1230	Trip RCPs
1360	Tops of candy canes voided
2100	Hold PORV open
7000	Recovery of subcooling
8100	Begin to throttle HPI
14250	End of calculation
22500	Reach RHR conditions[a]

[a]Estimated by linear extrapolation.

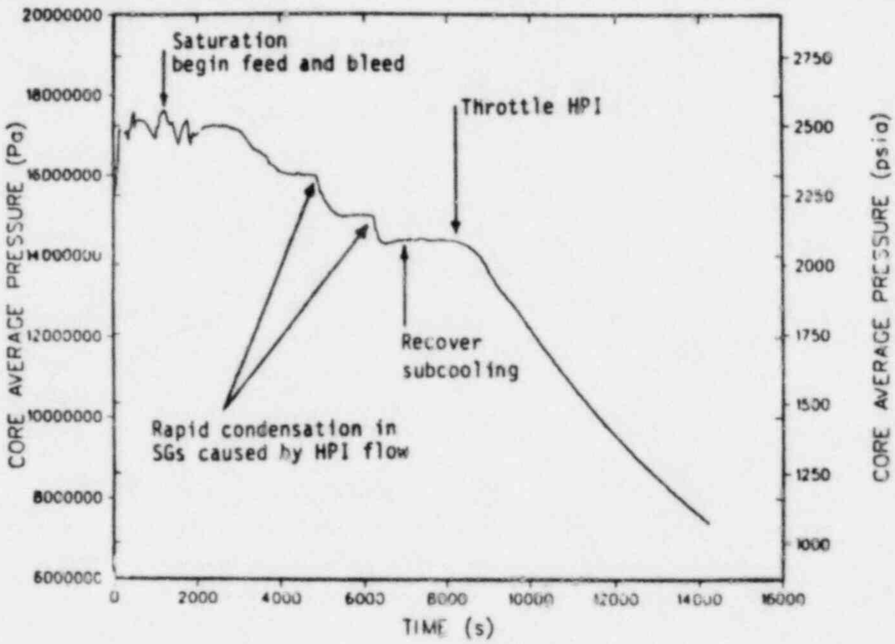

Fig. 16.
Primary pressure during the LOFW event for feed and bleed at primary-system saturation.

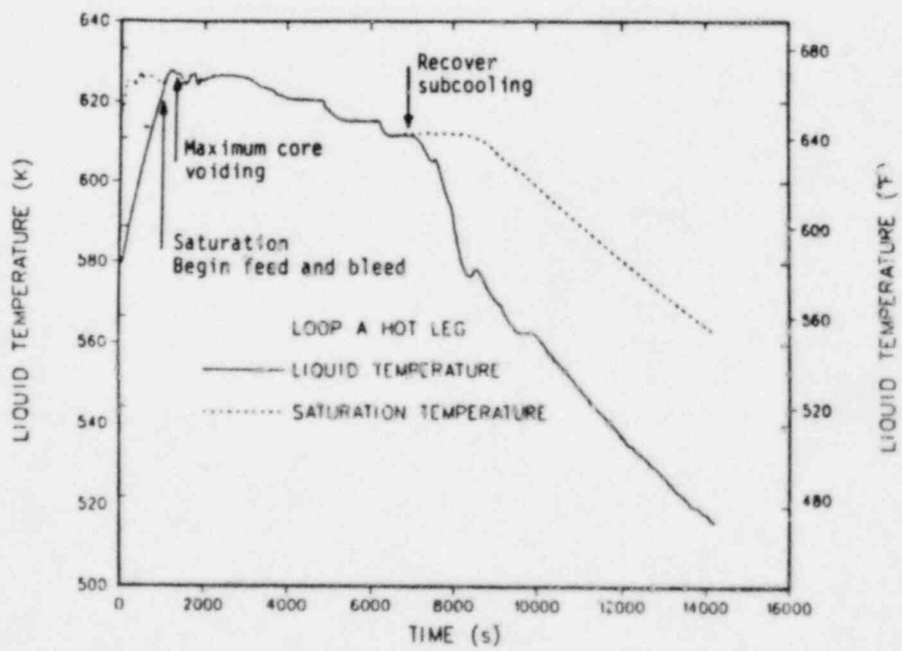

Fig. 17.
Hot-leg temperatures during the LOFW event for feed and bleed initiated at containment overpressure.

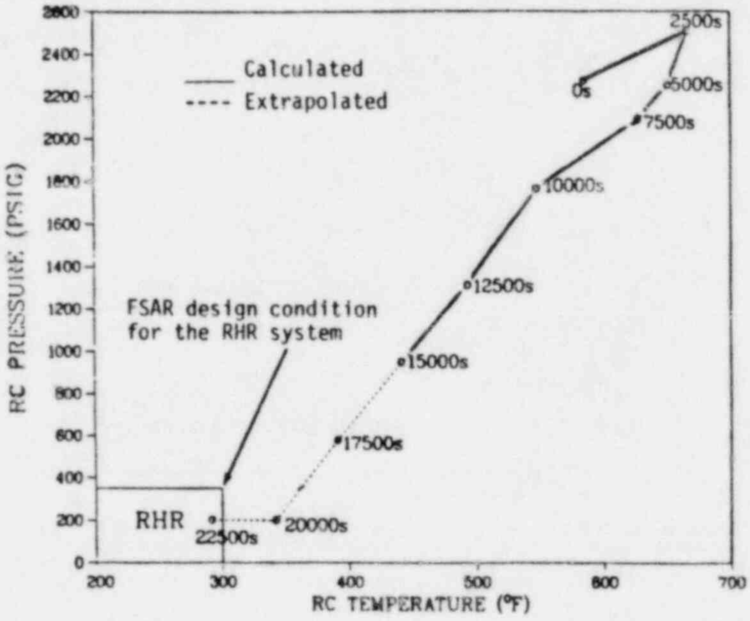

Fig. 18.
Depressurization cooldown map.

liquid-full and well mixed for the LOSP-induced LOFW event. This required a higher HPI flow to remove the energy from the water. In this study the top of the loops and steam generators were empty, decreasing the system heat capacity. The HPI flow required for a given decrease in temperature was sufficiently low that RHR system conditions were achievable.

One case of degraded equipment availability was studied for Oconee-1.[9] Two of the three in-plant HPI pumps are assumed for the nominal case. The degraded-equipment case considered the availability of only a single HPI pump following S1 signal generation following a containment overpressure signal at 900 s after the transient initiator. The first success criterion (S1) was satisfied at about 4300 s when the vessel mass inventory began to increase above its minimum. The cladding temperature stayed near or below saturation throughout the transient.

3. Combined MSLB/LOFW Event. To determine the effect of rapid secondary depressurization on primary heatup, it was postulated that the main steam line on loop A (see Fig. 8) suffers a double-ended break outside the containment and upstream of the main-steam-line isolation valve. Because the steam lines of both steam generators connect into a common header, both steam generators initially blow down to the atmosphere. Steam flow from the steam generators out the break results in overcooling of the primary system that in turn generates either a low-pressure (13.1 MPa, 1900 psia) or high-power (5% overpower) reactor trip that also closes the TSVs. This isolates the loop-B steam generator from the break. The reactor trip also trips the main feedwater which is assumed to coast down in 14 s adding 6500 kg (14318 lb) of feedwater to the steam generators. The reactor trip also generates a signal to begin AFW. However, AFW feedwater is assumed to fail so that the steam generators are left without feedwater.

The event sequence for the combined MSLB/LOFW transient is given in Table VIII. The double-ended steam-line break resulted in both steam generators blowing down. Steam flow out the break resulted in cooling and depressurization of the primary system. Cooling in the core region in turn resulted in a

TABLE VIII

OCONEE-1
MSLB/LOFW EVENT SEQUENCE FOR BASE-CASE TRANSIENT (NO HPI)

Time (s)	Event
0.0	Steam-line break in loop A
4.7	Trip reactor, close TSVs main feedwater coastdown
20	Broken-loop SG empty
33	Intact-loop TBV opens
200	Intact-loop SG empty
450	PORV opens
450	End of calculation

positive reactivity insertion and an overpower trip at 4.7 s when the power reached 105% of nominal power. The trip inserted the control rods, closed the TSVs (thus isolating the intact-loop steam generator) and allowed the main feedwater to coast down. Auxiliary feedwater, which should have been initiated at this time, failed and a LOFW resulted. By 20 s, the broken-loop steam generator was empty and the overcooling and resultant depressurization of the primary system stopped as can be seen in Fig. 19, which shows the primary-system pressure. By 33 s the intact-loop TBV opened as the isolated secondary pressure reached 7.05 MPa (1015 psia). At 200 s the intact-loop steam-generator secondary was empty. The primary system pressurized and heated until the PORV setpoint was reached at 450 s. Also shown in Fig. 19 is the condition for the base LOFW transient. It can be seen that the steam-line break that occurred before feedwater was lost merely compressed the time scale for the steam generators to empty. The overcooling and depressurization that initially resulted from the steam-line break was over by 20 s, and by 500 s the system was in the state similar to that of the simple LOFW transient. After approximately

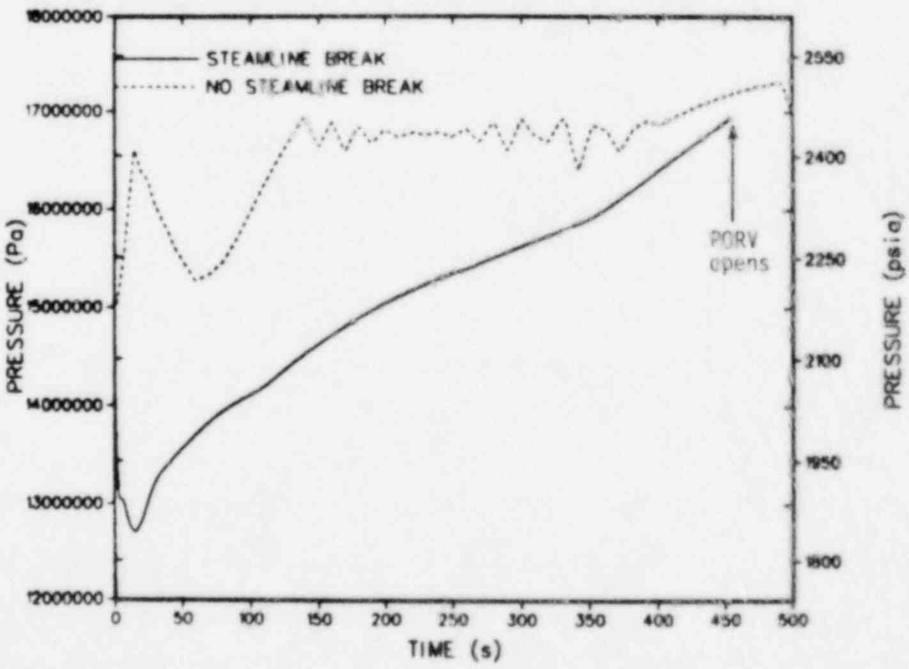

Fig. 19.
Primary pressure during the MSLB/LOFW event for base transient.

500 s, the feed-and-bleed conclusions for the LOFW event apply. The MSLB/LOFW transients for feed and for feed and bleed were not calculated.

4. Combined MFLB/LOFW Event. This event was not calculated for Oconee-1. However, it was calculated for Zion-1 and a discussion of the transient phenomena is presented in Sec. V.A.5. It is expected that the trends will be very similar to the combined MSLB/LOFW transient for Oconee-1. Following the MFLB, primary system overcooling will occur as the broken-loop steam generator blows down through the feed-line break. However, the overcooling will not be as severe as with the MSLB because a large fraction of the inventory will flash as it passes out the break. This liquid will not absorb energy from the primary. For the MSLB, nearly the entire liquid inventory flashes in the tube-bundle region, extracting energy from the primary. The time to steam-generator-secondary dryout is expected to be slightly earlier for the combined MFLB/LOFW transient when compared to the MSLB/LOFW transient. Timing of specific events can be estimated from the LOFW and combined MSLB/LOFW events.

5. Combined SGTR/LOFW Event. The initiating event was the rupture of a single steam-generator tube which started a primary-system depressurization. A reactor trip signal was generated when the primary system pressure reached the low-pressure setpoint. This caused the main feedwater pumps to trip and coast down and also closed the TSVs. It was assumed that AFW was not available and that the HPI system failed to deliver any water for the base case. The RCPs continued to operate throughout this transient.

a. Base Transient. The event sequence for the base transient is given in Table IX. Following the SGTR, the primary-system began to depressurize (Fig. 20). A reactor trip on low primary-system pressure occurred at 851 s. The main feedwater pumps tripped, and the AFW system failed to provide any water to the steam generators. The steam-generator secondaries dried out (1200 and 1400 s for the intact and damaged steam generators, respectively), and the primary began to heat up and expand. Hot-leg saturation is shown in Fig. 21. The PORV opened at 2460 s and the system was saturated at 2700 s. Without make-up water from the HPI system, the primary inventory was reduced by boiloff and the core was empty by 4000 s. Rapid core heating followed.

TABLE IX

OCONEE-1
EVENT SEQUENCE FOR SGTR/LOFW BASE-CASE TRANSIENT (NO HPI)

Time (s)	Event
0	1 SGTR
570	Heaters off (low pressurizer level)
851	Reactor trip ($p < 13.1$ MPa, 1900 psia), main feedwater coasts down
852	Close TSVs
853	Steam line ARVs open
891	ECC signal ($p < 11.4$ MPa, 1650 psia) but HPI fails
1200	Intact SG empty
1400	Damaged SG empty
2460	PORV opens
2700	System saturates
2750	Pressurizer full
4000	Core empty, rapid heating begins
4900	End of calculation

b. Feed Transient. We then examined feed operation for the SGTR/LOFW transient with automatic actuation of the HPI system on a low primary-system pressure of 11.4 MPa (1650 psia) at 891 s. The operator turned the RCPs off at 921 s after verifying that the HPI system was functioning. The automatic HPI initiation proved to be sufficient for core cooling. The primary system regained a substantial subcooling margin as shown in Fig 22. Primary-to-secondary flow was not large enough to prevent repressurization and the PORV opened at 1850 s. Because cooling was provided by HPI water flowing through the core and then out the PORV and the ruptured tube, the intact steam generator did not empty until approximately 3500 s. The damaged steam generator, on the other hand, was filling when the calculation was ended at

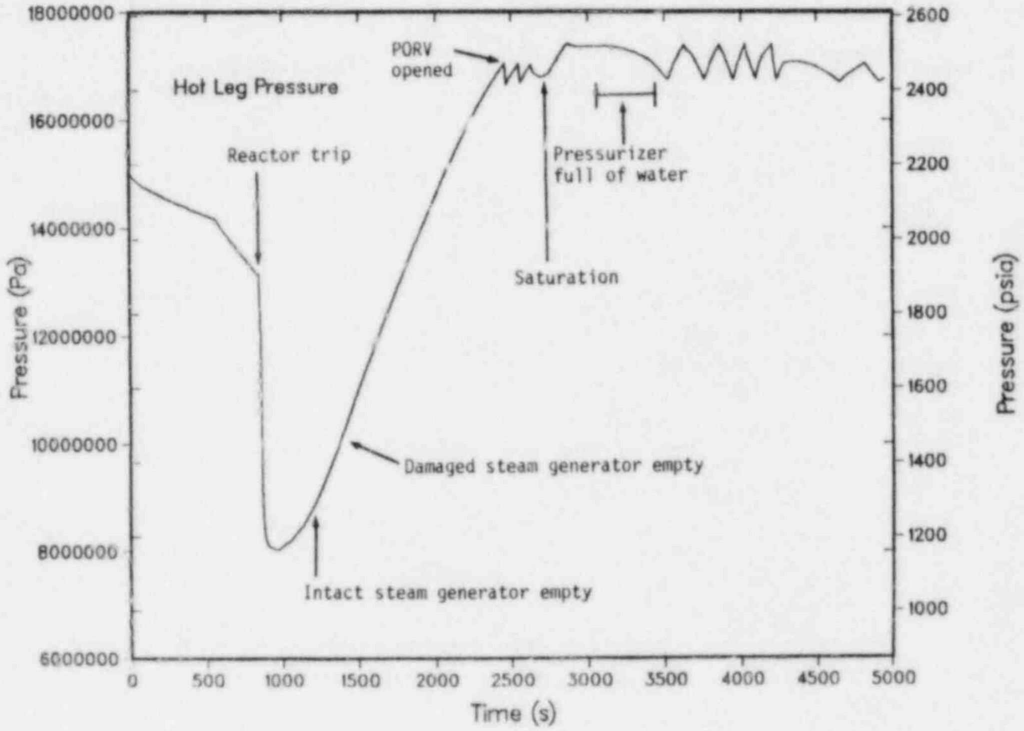

Fig. 20.
Primary-system pressure during the SGTR/LOFW event for base transient.

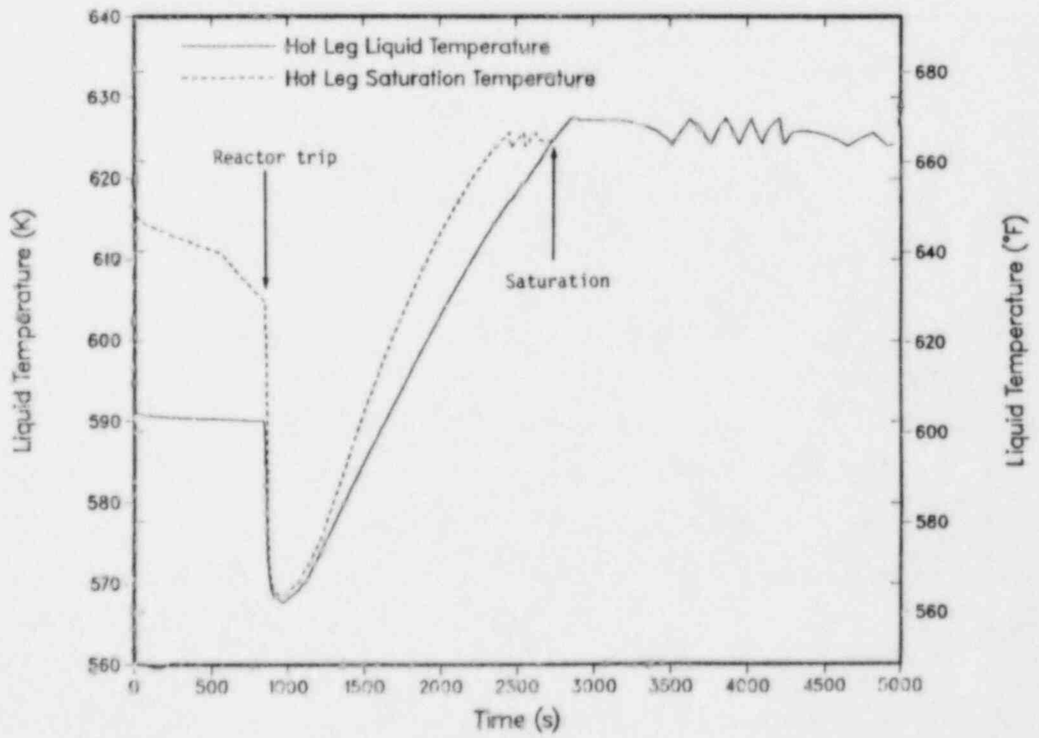

Fig. 21.
Hot-leg temperatures during the SGTR/LOFW event for base transient.

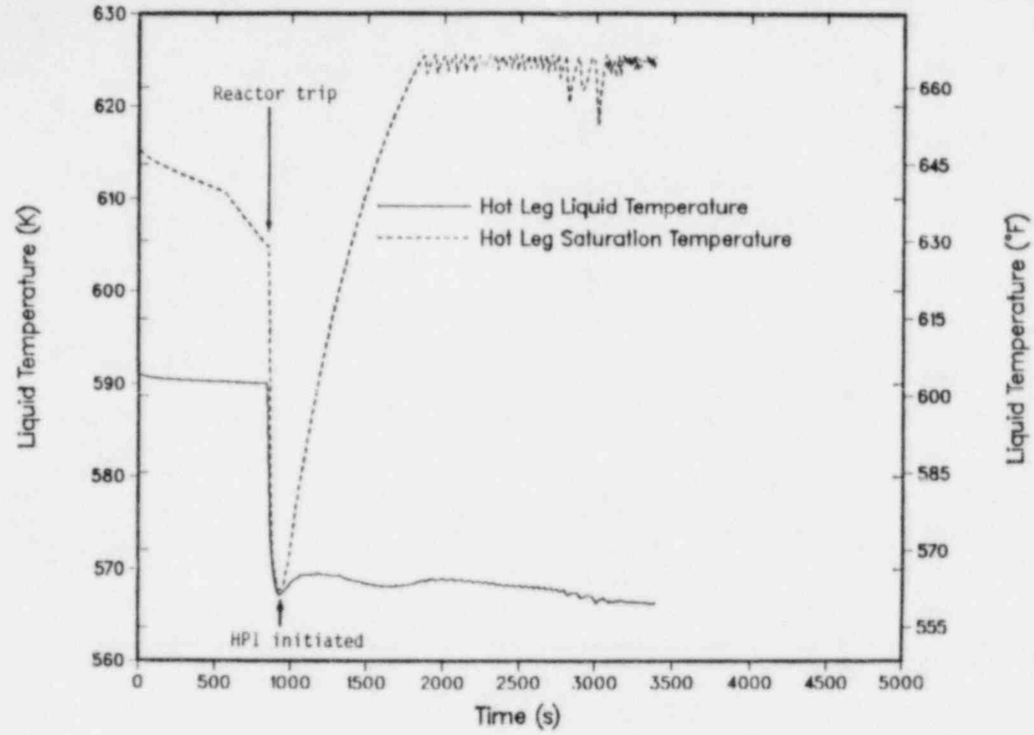

Fig. 22.
Hot-leg temperatures during the SGTR/LOFW event for feed initiated on low primary pressure (11.4 MPa).

3400 s. The damaged steam generator will be full by 7200 s. In the automatic response transient, core cooling was achieved but flow through the ruptured tube remained high. Additionally, the temperature of the primary water flowing through the ruptured tube was above the saturation temperature corresponding to the ARV setpoint pressure. The water thus flashed when it entered the steam generator secondary and kept the ARVs open, and primary fluid flowed into the atmosphere. Thus, we see success criterion 1 (S1) is satisfied but radiological release is certain.

c. Feed-and-Bleed Transient. To limit radiological releases, operator intervention is required to depressurize the primary system and limit the tube-rupture flow. To investigate this possibility, feed-and-bleed cooling was initiated by opening the PORV at 1200 s into the feed transient. The system response to opening the PORV is shown in Figs. 23 and 24, which display the pressure and liquid temperature, respectively. The pressure first decreased and the primary liquid expanded until the pressurizer was full at 1730 s. Reduced volumetric flow out the PORV with the pressurizer full resulted in a repressurization to 10.6 MPa (1537 psia). Subcooling was re-established and at 2270 s the primary liquid temperature was below the saturation temperature

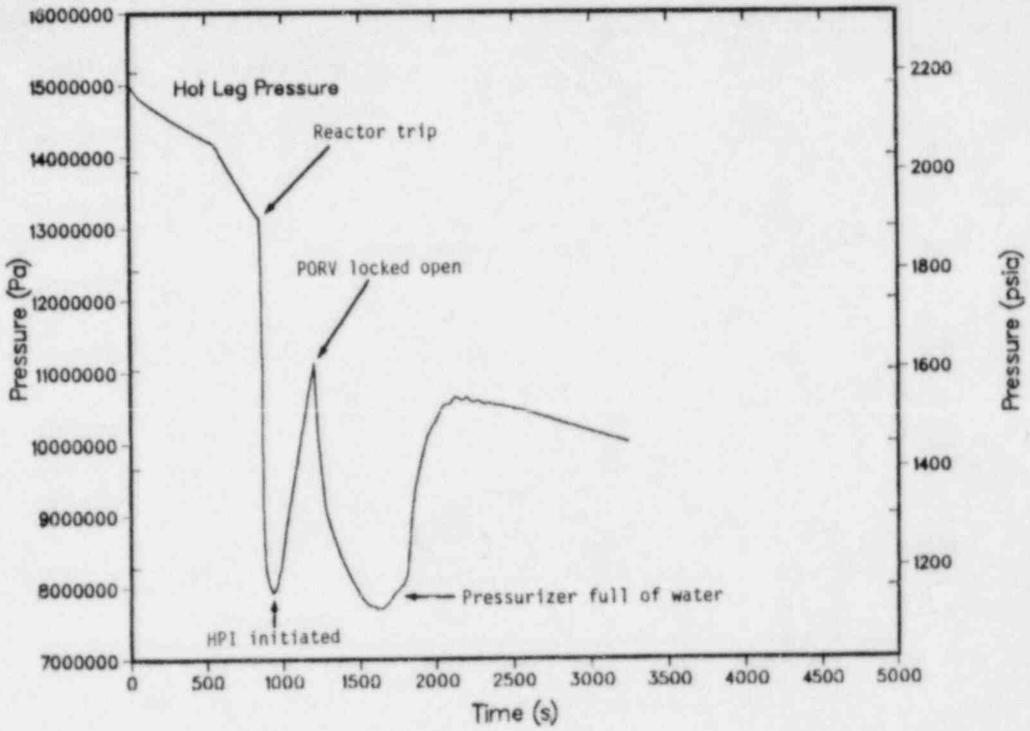

Fig. 23.
Hot-leg pressure during the SGTR/LOFW event for feed-and-bleed transient initiated at 1200 s.

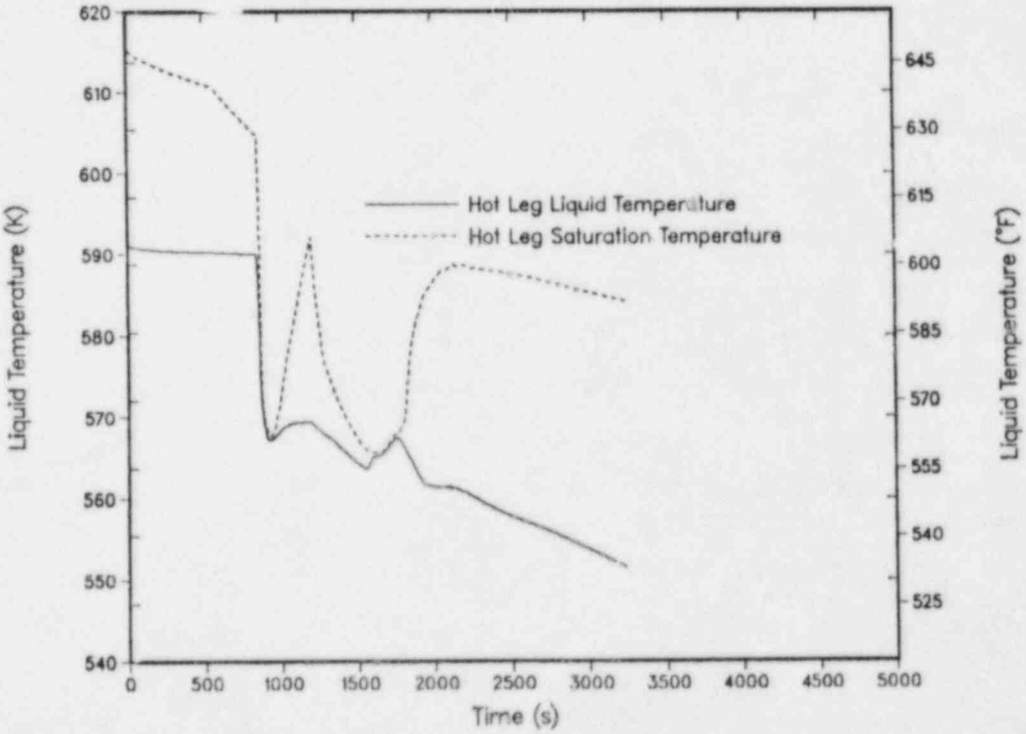

Fig. 24.
Hot-leg temperatures during the SGTR/LOFW event for feed-and-bleed transient initiated at 1200 s.

corresponding to the ARV setpoint pressure. The ARVs closed, stopping flow to the atmosphere. The primary pressure remained above the secondary pressure and tube-rupture flow continued to fill the affected steam generator. The calculation was ended at 3250 s, with further operator action required to stop primary-to-secondary leakage. To stop this leakage, the operator must reduce the primary-system pressure by throttling the HPI flow while being careful to keep the system cool enough so that the ARVs remain closed. A throttled-feed-with-bleed procedure would be effective in providing stable cooling of the reactor. Success criterion 2 (S2) was satisfied by the feed-and-bleed operation.

B. Summary Insights and Conclusions

We have examined a number of loss-of-feedwater transients either singly or combined with other failures for the Oconee-1 plant. Three features of the plant have had the most significant effect on the accident signature. The first of these is the small steam-generator-secondary inventory characteristic of a once-through steam generator. The total inventory of about 35000 kg (77092 lb) is from 3 to 5 times smaller than the inventory of U-tube steam generators for equivalent plant sizes. The primary effect of steam-generator-secondary inventory is on event timing and, therefore, on operator reaction times. For the base LOSP case, steam-generator-secondary dryout occurs at approximately 600 s. Primary-system heatup begins earlier as the rate of heat removal through the steam generator degrades as steam-generator secondary inventory is depleted and the decay power cannot be removed.

Steam-generator-secondary dryout occurs even earlier in the LOFW transient, primarily because the reactor trip occurs later for this event. Event timing is important as it influences the ability of the operator to identify the failure(s) and initiate the proper recovery activities. We did not have available at the time our calculations were performed either plant-specific (Oconee-1) or B&W guidelines that specified when a feed-and-bleed procedure should be initiated and the procedures to be followed. However, we have found that a feed-and-bleed procedure can be successfully initiated as late as the time of primary-system saturation following steam-generator-secondary dryout.

The second feature of importance is the delivery capability of the Oconee-1 HPI system, which has the largest flow delivery rate at the PORV setpoint of the four plants for which detailed calculations have been performed. We find that the HPI system delivers sufficient coolant to cool the reactor in the feed mode even after significant primary system voiding has occurred. If the nominal HPI system (2 pumps) is actuated before containment overpressure, the primary system remains liquid-full and subcooled.

We did not perform a calculation to evaluate a feed-and-bleed procedure with nominal equipment beginning when the containment overpressure signal is generated. Although lacking procedures to direct the operator to begin a feed-and-bleed procedure at the time of containment overpressure, we believe that the operator would have sufficient time, information, and cause to initiate feed-and-bleed by the time the containment overpressure signal was received. We did evaluate a feed-and-bleed procedure with degraded equipment availability (one of two available HPI pumps) beginning when the containment overpressure signal is generated. We found that the plant could successfuly transition to a hot, holding condition (S1).

We have obtained a number of insights while analyzing the plant-specific calculations. One of these insights is that a plant operating in the feed mode that satisfies the criteria for successful feed-and-bleed during transition from reactor trip to hot standby (S1) will also satisfy the same success criteria in the feed-and-bleed mode. We base this insight on the following observations of PORV and HPI performance characteristics. First, the PORV is either full open or cycling during the successful feed-mode transient. Therefore, if the PORV is manually latched open, the primary-system pressure will eventually fall. Second, the HPI system will respond by increasing the flow, thereby enhancing cooling. Although the system pressure and therefore maximum cladding temperatures set by the coolant saturation temperature will be lower, the criterion for successful feed-and-bleed during the transition from reactor trip to hot shutdown will be satisfied.

We performed one Oconee-1 feed-mode calculation with nominal equipment availability at system saturation and the onset of boiling in the core. For the LOFW transient, the containment overpressure signal is generated at about 900 s, and if the HPI is not started, voiding begins in the core by 1200 s. For this parametric study, HPI was initiated in the feed mode at 1200 s. The criterion for a successful feed-and-bleed operation (S1) was satisfied while operating in the feed mode.

The LOFW feed-mode calculation was also used to provide a check on our conclusion that a plant satisfying the success criterion for transition from reactor trip to a hot holding condition in the feed mode will also satisfy the criterion in feed-and-bleed mode. During the feed-mode transient, the small Oconee-1 single PORV was unable to discharge sufficient energy to keep the system pressure below the SRV setpoint. Thus, the PORV was fully open (the equivalent of latched open) until 2500 s when the PORV began to cycle. We restarted the feed-mode calculation at 2100 s with the PORV latched open, thereby generating the equivalent of feed-and-bleed calculation beginning at 1200 s when the HPI was initiated. The feed-and-bleed procedure was successful (S1) with subcooling recovered at about 7000 s.

We also examined the transition from reactor trip to hot shutdown using the same LOFW transient. Beginning at 8000 s the HPI was throttled to provide a controlled depressurization and cooldown. By 22500 s, the primary had been cooled and depressurized to RHR-system operating conditions (S2) using only the inventory of the BWST. Early initiation of a feed-and-bleed procedure shortly after reactor trip (120 s) for a LOSP has also been investigated. In this case we studied the transition directly from reactor trip to hot shutdown. With HPI throttling it was possible to cool and depressurize the reactor in a controlled manner. However, it was not possible to reach RHR-system operating conditions before exhausting the inventory of the BWST at about 50000 s. To be successful, additional actions would be needed to replenish the inventory of the BWST or initiate LPI suction from the containment sump and pump-up to the HPI-system inlet pressure.

The third important feature affecting feed-and-bleed is the PORV relief capacity. Because the limited Oconee PORV relief capacity was the key factor in limiting the rate of cooldown and depressurization, we repeated the LOSP transient just described. The model was altered to include a PORV with double the relief capacity of the existing PORV. The larger PORV flow resulted in a rapid depressurization to the RHR operating pressure. Throttling was then discontinued and the primary cooled to RHR-system operating conditions prior to depleting the inventory of the BWST inventory.

Transients initiated by a break on the secondary side combined with an LOFW were also examined. The combined MSLB/LOFW and MFLB/LOFW transients accelerated the time to steam-generator-secondary dryout. We find that the secondary-side break leaves the plant in a state similar to the simple LOFW transient after 500 s. We believe, therefore, that the conclusions for the simple LOFW transient may also be applied to the combined transients.

Transients initiated by an SGTR combined with an LOFW were investigated. Both success criteria 1 and 2 can be satisfied by feed-and-bleed operations. However, the combined SGTR/LOFW event is complicated by the necessity to terminate the primary-to-secondary flow with its potential for radiological releases to the environment. The small PORV relief capacity limits the ability to depressurize the primary while still cooling the reactor core.

We have reached the following conclusions about feed-and-bleed procedures in the Oconee-1 plant.

1. A feed-and-bleed procedure can be used successfully to cause the transition of the plant from reactor trip to a hot holding condition. If initiated before or at the time the containment overpressure signal is generated, the system can be stabilized in a liquid-full and subcooled state. The plant can be successfully transitioned to a hot holding condition even if only one of the two normally available HPI pumps is used.

2. A feed-and-bleed procedure can be used successfully to effect transition of the plant from reactor trip to a hot holding condition if initiated within 300 s of containment overpressure. However, some boiling will occur in the core. Feed and bleed may prevent core damage if initiated at times greater than 300 s after containment overpressure, but significant core voiding would occur. Additional calculations would be needed to determine the maximum hot rod temperatures under voided core conditions and subsequent HPI refilling.

3. A feed-and-bleed procedure can also be used successfully to cause the plant's transition to hot shutdown. However, this conclusion assumes that the capability exists to replenish either the inventory of the BWST or the operation of the LPI system, taking suction from the containment sump and delivery at the HPI-system inlet.

4. Use of feed and bleed in the transition to hot shutdown could be simplified with increased PORV-relief capacity. With sufficient capacity (at least double the existing PORV capacity), the plant can be depressurized and cooled using only the inventory of the BWST.

This may be desirable if the operations to use the LPI or replenish the BWST inventory are unreliable.

IV. CALVERT CLIFFS-1 INSIGHTS

Calvert Cliffs-1 (Ref. 12) is a C-E PWR operated by Baltimore Gas and Electric Company. Design output of the reactor is 2700 MWt. The reactor-coolant system consists of two closed heat-transfer loops. An electrically heated pressurizer is connected to one loop hot leg. The secondary-coolant system is designed to produce steam at a pressure of 5.9 MPa (850 psia). Overpressure protection is provided by PORVs and spring-loaded SRVs connected to the pressurizer. SRV and PORV discharge is released under water in the quench tank where the steam discharge is condensed. The two steam generators are vertical shell and U-tube units, each of which produces 2.558×10^6 kg/hr (5.635×10^6 lb/hr) of steam. Steam is generated in the shell side and flows upward through moisture separators. Steam outlet moisture content is less than 0.2%. The reactor coolant is circulated by four vertical, electric-motor-driven, single-bottom-suction, single-stage, horizontal-discharge centrifugal pumps.

The Calvert Cliffs-1 SI system includes HPI and LPI capability as well as charging flow and SI tanks (accumulators). Three positive displacement charging pumps (nonsafety grade) deliver a total SI flow of 8.3 kg/s (18 lbm/s), independent of primary system pressure. The HPI centrifugal pumps have a shutoff head of 8.8 MPa (1275 psia). Above this pressure, only charging flow is possible. Four SI tanks are provided, each connected to one of the four reactor inlet lines. Each tank has a volume of 56.6 m^3 (2000 ft^3) of borated water at refueling concentration and 28.3 m^3 (1000 ft^3) of nitrogen at 1.38 MPa (200 psig). In the event of a large LOCA, the borated water is forced into the primary system by the expansion of the nitrogen. The water from three tanks adequately cools the entire core. Borated water is also injected into the primary system by two LPI and three HPI pumps taking suction from the refueling water storage tank (RWST). For reliability, the design capacity from the combined operation of one high-pressure and one low-pressure pump provides adequate injection flow for any LOCA; in the event of a design-basis accident, at least one high-pressure and one low-pressure pump will receive power from the emergency power sources if normal power is lost and one of the emergency diesel-generators is assumed to fail. Upon depletion of the RWST supply, the high-pressure pump suctions automatically transfer to the containment sump and the low-pressure pumps are shut down. The high-pressure pump has sufficient capacity to cool the core adequately at the start of recirculation.

A schematic of the reactor-coolant system is presented in Fig. 25. For all Calvert Cliffs-1 calculations we assumed three charging pumps and two HPI pumps. The SI delivery characteristics for all pumps were based on best-estimate values obtained from the utility. Two PORVs were modeled for each calculation. We used the 1973 American Nuclear Society decay-heat curve. The TRAC-PF1 input model of the Calvert Cliffs-1 plant is described in Appendix B.II.

A. Summary Results

We have prepared detailed reports of a series of studies that examined loss-of-feedwater transients in the Calvert Cliffs-1 plant.[13-15] Also included are summary results for two LOFW feed-and-bleed calculations for which a detailed report is to be prepared in the future. In the following we present the summary results for LOFW transients in the Calvert Cliffs-1 plant. Again, we report on the same five events in the common set as reported for Oconee-1. The results of our Calvert Cliffs-1 calculations are summarized in Fig. 26. Although success criterion S1 is written for a feed-and-bleed procedure, S1 can often be satisfied by a feed operation. Figure 26 also shows the period in which the feed or feed-and-bleed operation was initiated relative to key reactor events. It appears that feed and bleed cannot be safely initiated later than the time the secondary heat sink is lost.

1. LOSP-Induced LOFW Event. The LOFW transient was initiated by an LOSP event. It was assumed that an LOSP results in an immediate trip of the RCPs, a reactor trip, a turbine trip and closure of the TSVs. It was also assumed that main feedwater flow dropped to zero instantly at the start of the transient and that no AFW was available. The event sequence for the base-case LOSP transient without SI system operation is given in Table X. The reactor and turbine trip automatically on LOSP, and the RCPs, main feedwater, and steam dump/bypass systems are disabled because the pumps and valves used in these systems do not operate without offsite power. Normally, AFW delivery to the steam generators would commence within about two minutes following LOSP because the AFW pumps are driven with emergency power supplies. However, the objective of the base case requires failure of the AFW.

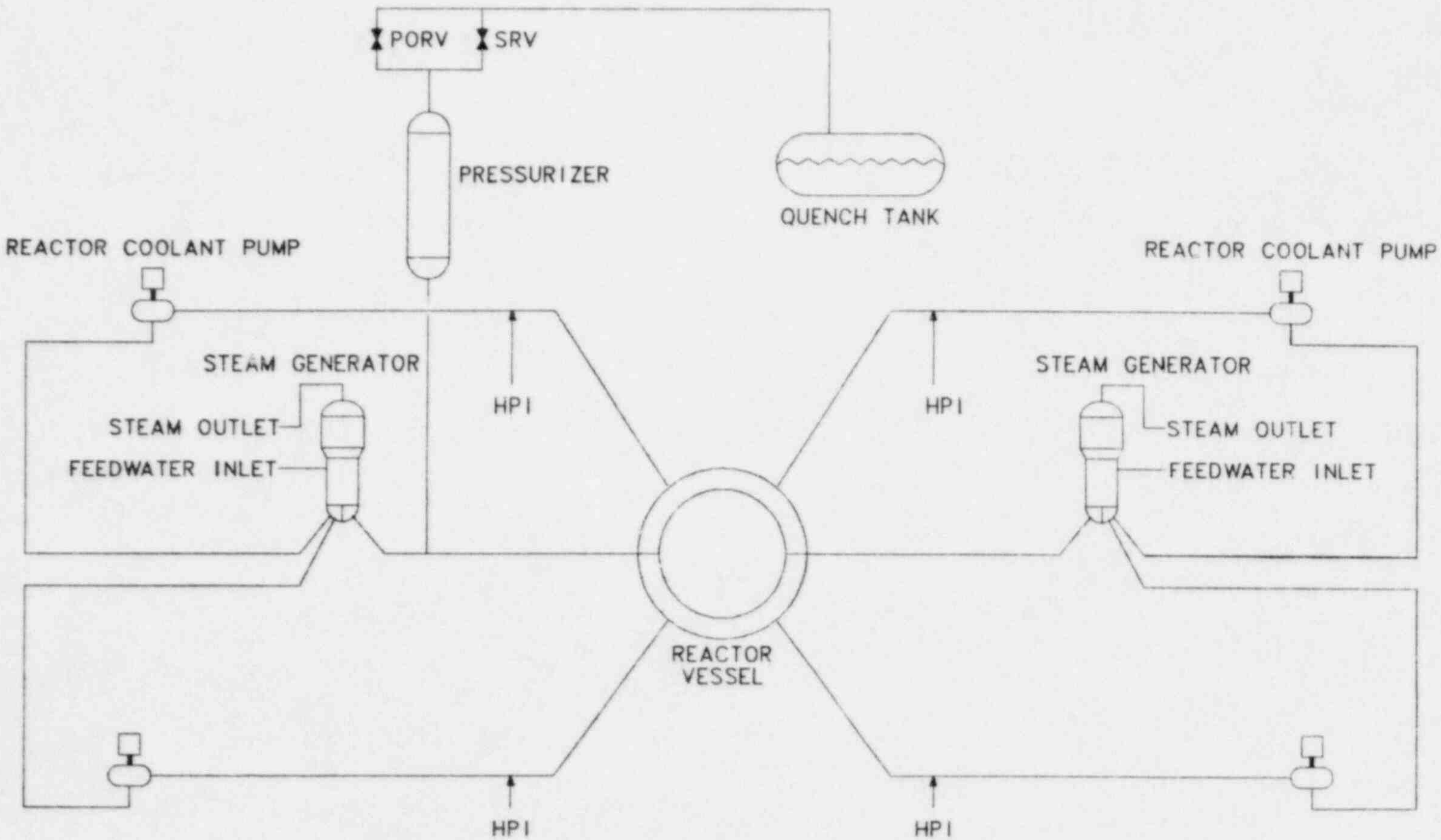

Fig. 25.
Calvert Cliffs-1 reactor-coolant system.

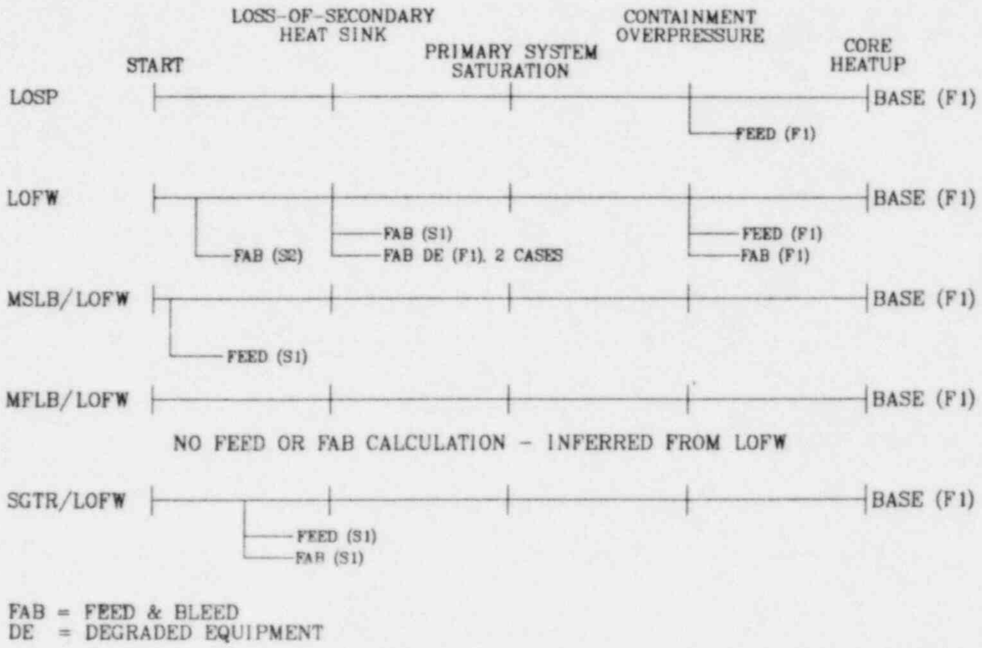

Fig. 26.
Calvert Cliffs-1 success/failure chart.

a. Base Transient. The signature of the base transient is similar to that of the Oconee-1 plant for the same event. However, the event timing is stretched out because of the greater steam-generator-secondary inventory of the U-tube steam generators in the Calvert Cliffs-1 plant. Following the reactor and turbine trips, the secondary-side SRVs opened at 5.0 s. A prolonged period followed during which the steam-generator-secondary inventory was depleted by boiling.

By 3500 s both steam generators were dried out, and the primary-system temperature began to increase rapidly as shown in Fig. 27. The fluid expansion associated with increasing temperature began refilling the pressurizer at this time, and the resulting compression of vapor in the pressurizer increased the primary pressure to the PORV setpoint of 15.7 MPa (2400 psia) by 3807 s (Fig. 28). By 4100 s the pressure in the pressurizer quench tank reached 0.791 MPa (100 psig), and the rupture disk ruptured. Thereafter, the fluid that was vented from the PORVs raised the containment pressure.

TABLE X

CALVERT CLIFFS-1
LOSP EVENT SEQUENCE FOR BASE-CASE TRANSIENT (NO HPI)

Time (s)	Event
0.0	Initiating event caused reactor trip, turbine trip, RCP trip, failure of the main and auxiliary feedwater systems and the steam dump/bypass system
5.0	SRVs on steam lines open on high pressure, 6.895 MPa (1000 psia)
3500	Both steam-generator secondaries are empty
3807	PORVs open on high pressure, 15.7 MPa (2400 psia)
4100	Pressurizer quench tank vents
5000	Pressurizer solid
6100	Hot legs saturate and loops stagnate
6300	SI actuation signal on high containment pressure, 0.129 MPa (4 psig); SI system disabled
7000	Pressurizer level begins to decrease
8000	Core heatup begins
8600	Core 90% voided

By 5000 s the pressurizer was solid, and the mass discharge rate from the PORVs increased as the steam was replaced by liquid. By 6100 s the primary temperature reached saturation, and the PORV discharge rate increased abruptly. Voiding in the steam-generator U-tubes associated with saturation of the primary caused the loop flows to stagnate, but heat transfer to the steam generators had ceased much earlier as a result of steam-generator dryout. By 6300 s the containment pressure had increased to a 0.129 MPa (4 psig), and the containment overpressure signal was initiated. By 7000 s the pressurizer level began decreasing, and the pressurizer SRVs momentarily lifted. Voiding of the vessel and core increased dramatically at 7800 s and rapid core heatup (F1) ensued shortly afterwards as shown by Fig. 29.

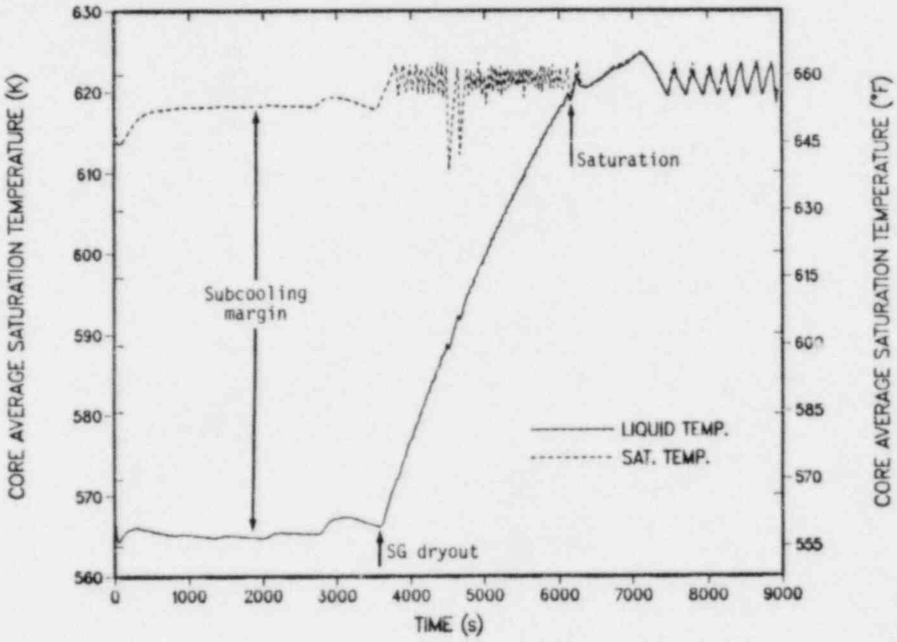

Fig. 27.
Core-liquid temperatures during the LOSP/LOFW event for base transient.

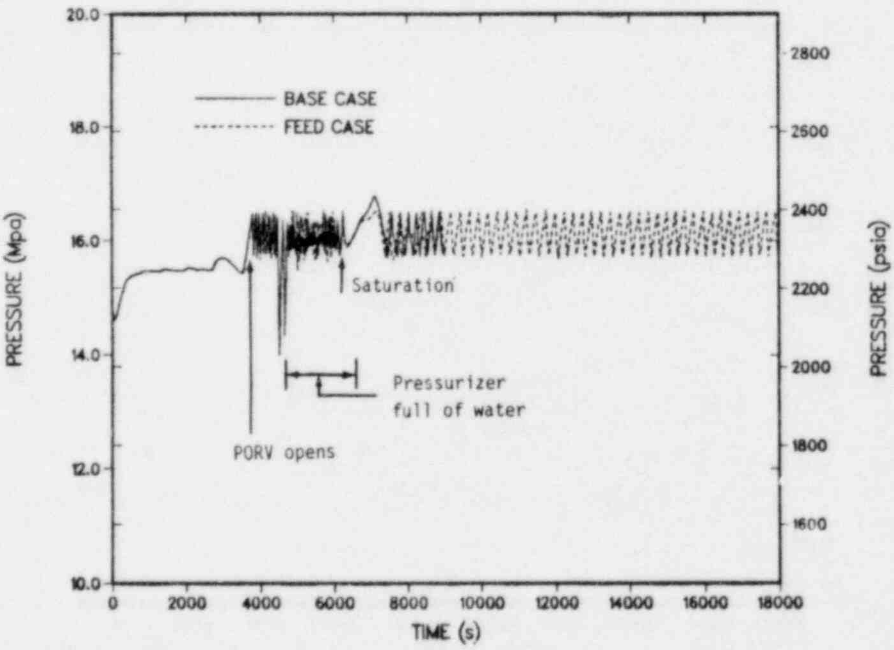

Fig. 28.
Primary pressure during the LOSP/LOFW event for base transient.

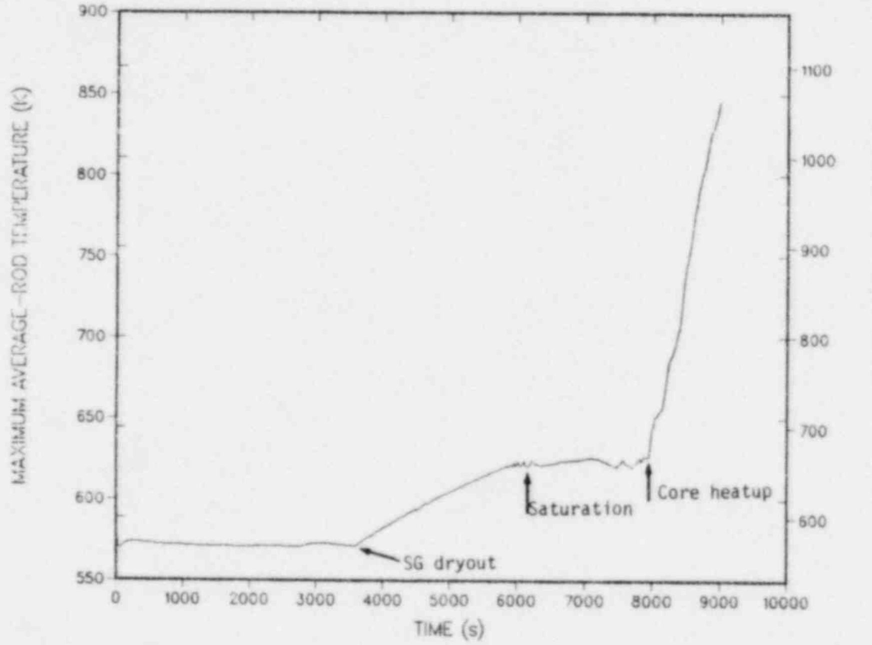

Fig. 29.
Cladding temperature during the LOSP/LOFW event for base transient.

b. Feed Transient. The feed case was performed by restarting the base case after enabling the SI system to deliver charging flow at 286 K (55°F) to the cold legs. The restart was taken at 6300 s, the estimated time of containment overpressure. Figure 30 shows that even the small amount of SI system charging flow available at the PORV setpoint, 8.3 kg/s (18 lbm/s), was very effective in delaying loss of cooling in the core. Although the SI system flow was not immediately successful in halting core voiding, which eventually reached 50%, the gradual penetration of the cooling effect into the core region combined with decreasing decay power finally arrested the depletion of the core liquid. The calculation was terminated at 18000 s because conditions then appeared to be stable and recovery without serious damage was expected. The decay power at 18000 s is ~20 MW, whereas the power necessary to vaporize all of the incoming SI system flow is ~21 MW; hence, a gradual refilling of the primary system is inevitable (assuming an adequate SI system supply). Figure 31 compares the maximum average fuel-rod temperature for the base and feed cases. The effectiveness of the feed in maintaining core cooling is evident.

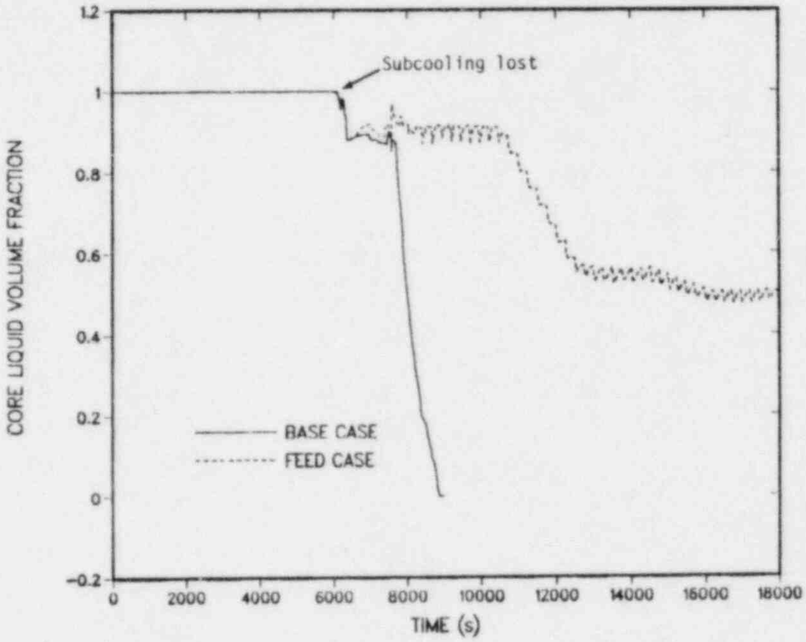

Fig. 30.
Core-liquid volume fraction during the LOSP/LOFW event for feed initiated at containment overpressure.

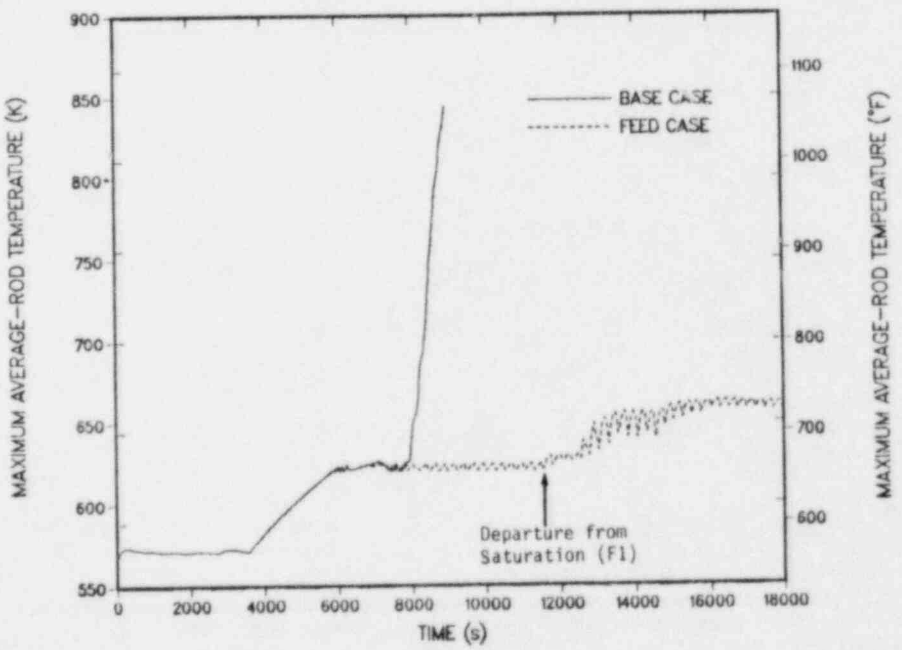

Fig. 31.
Cladding temperature comparison between base and feed transients during the LOSP/LOFW event.

While this case was successful in arresting total core uncovery, it was not successful in limiting the fuel-rod cladding temperature at or below saturation temperature (F1). The predicted maximum cladding temperature was 660 K (729°F), whereas the peak saturation temperature was 620 K (657°F).

c. Feed and Bleed. A feed-and-bleed case was not run in this study. However, Fig. 30 indicates that extensive voiding occurs in the core during feed cooling. As the charging flow was insufficient to prevent significant core voiding, the effect of locking open the PORV to depressurize would result in increased core voiding. The integrated PORV mass-flow rate for the base and feed cases is shown in Fig. 32. When the pressure dropped below the HPI shutoff head (8.8 MPa, 1275 psia), the core would begin refilling. However, core cladding temperatures would exceed saturation temperatures resulting in a failure to meet success criteria. Because the SI system flow rate increases rapidly below 8.8 MPa (1275 psia), a feed-and-bleed strategy initiated earlier, using the PORVs to lower the pressure below 1275 psia, would be effective. This conclusion was verified by running a feed-and-bleed calculation for the LOFW case reported in the next section.

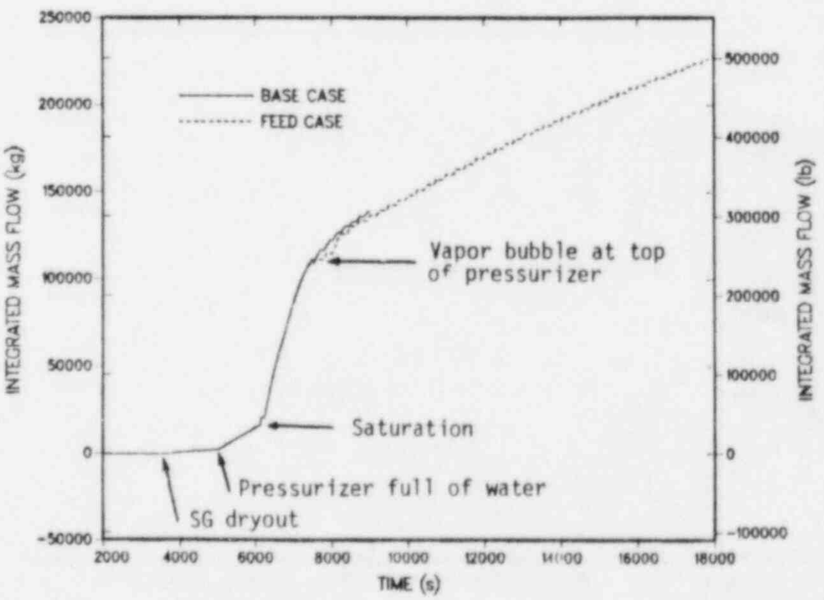

Fig. 32.
Integrated PORV mass flow rate during the LOFW event for the base and feed transient.

2. LOFW Event. The transients discussed in this section were initiated by a loss of main feedwater. It was assumed that the feedwater flow dropped to zero instantly at the start of the transient and that no auxiliary feedwater was available. The event sequence for the base LOFW transient in which the SI system fails to deliver water is given in Table XI. The primary difference between the LOFW transient described herein and the LOSP transient is in the operation of the RCPs. During an LOSP transient, the RCPs do not operate so forced circulation is lost. However, this effect is not as important as the energy added to the primary system when the RCPs are operating. Consequently, compared to the LOSP transient event sequence, the timing of events in the LOFW transient event sequence will be accelerated. A summary event chart for the

TABLE XI

CALVERT CLIFFS-1
LOFW EVENT SEQUENCE FOR BASE-CASE TRANSIENT (NO HPI)

Time (s)	Event
0.0	Initiating event caused failure of the main and auxiliary feedwater systems
22.6	Reactor and turbine trip on low level in the SGs, -1.27 m (-50 in.); trips generate a "quick-open" signal for the ARVs and TBVs
1250	Both SGs secondaries are empty
1680	PORVs open on high pressure, 15.7 MPa (2400 psia)
1800	Pressurizer quench tank ruptures
2300	Pressurizer solid
2900	Hot legs saturated and loops stagnate
3000	SI actuation signal on high containment pressure, 0.129 MPa (4 psig); SI system disabled
3400	Pressurizer level begins to decrease
4000	Core 90% voided
4200	Core heat-up begins

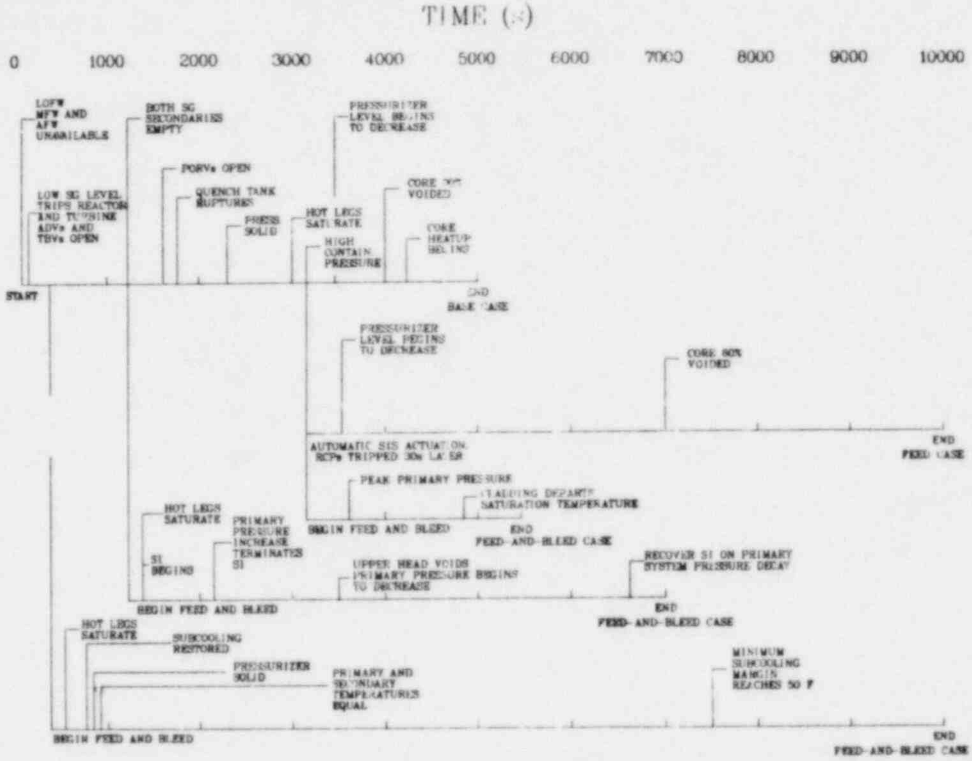

Fig. 33.
Calvert Cliffs-1 LOFW event line (nominal equipment).

Calvert Cliffs-1 LOFW calculations with nominal equipment availability is presented in Fig. 33. Key reactor events and phenomena are displayed.

a. <u>Base Transient</u>. The base transient was initiated by blocking the main feedwater flow and disabling the AFW at 0.0 s. Normally, AFW delivery to the steam generators would commence within two minutes following LOFW; however, the objective of this investigation requires failure of the AFW for the base case. The RCPs were assumed to operate throughout the base case.

By 22.6 s, the LOFW caused the water level in the steam generators to drop 1.27 m (50 in.) below its normal level and a reactor-trip signal was generated automatically. The reactor-trip signal simultaneously tripped the turbine, and the combined reactor/turbine trip generated a "quick-open" signal for the ARVs (called "atmospheric dump valves" in C-E plants) and TBVs because the reactor power was in excess of 69% of full power. The ARVs and TBVs together are capable of dumping up to 45% of the steam produced at full power. The ARVs are regulated to maintain the average reactor temperature between 551 K (532°F) and 565 K (557°F). The TBVs are regulated by the same signal unless the steam pressure exceeds 6.2 MPa (900 psia), in which case they are regulated to limit steam pressure.

The primary-system pressure history during the base LOFW transient is shown in Fig. 34. The primary system underwent a marked contraction because of overcooling before 100 s. By 1250 s both steam generators dried out and the primary-system temperature began to increase rapidly as shown in Fig. 35. The fluid expansion associated with increasing temperature began refilling the pressurizer at this time, and the resulting compression of the vapor in the pressurizer increased the pressure to the PORV setpoint of 15.7 MPa (2400 psia) by 1680 s (Fig. 34). By 1800 s the pressure in the pressurizer quench tank reached 0.791 MPa (100 psig), and the rupture disk ruptured. Thereafter, the fluid vented from the PORVs raised the containment pressure.

By 2300 s the pressurizer was solid, and the mass discharge rate from the PORVs increased as the steam was replaced by liquid. By 2900 s the primary temperature reached saturation (Fig. 35), and the PORV discharge rate increased abruptly. By 3000 s the containment pressure had increased to 0.129 MPa (4 psig) and SI signal occurred. However, the SI system was not activated

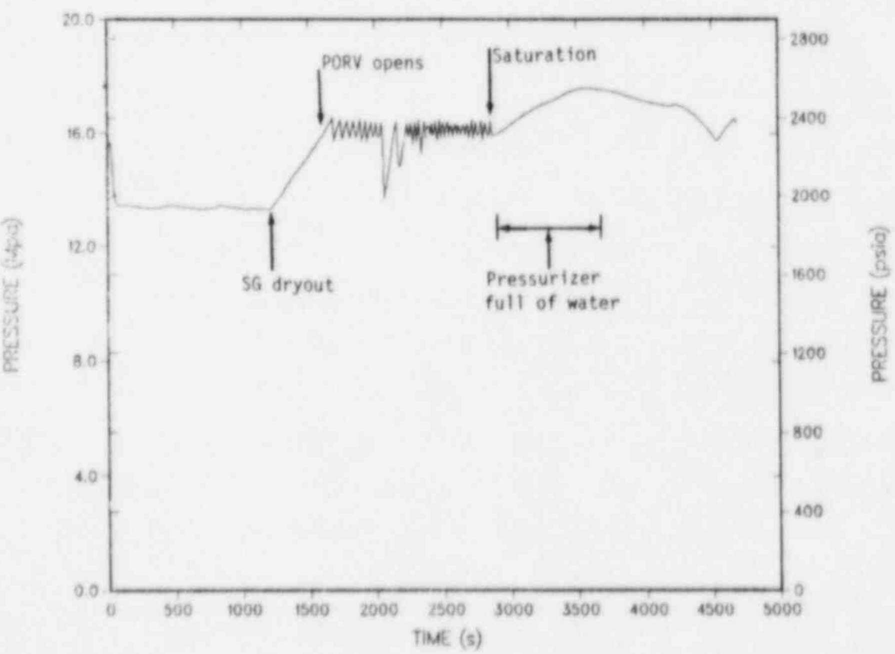

Fig. 34.
Primary pressure during the LOFW event for base transient.

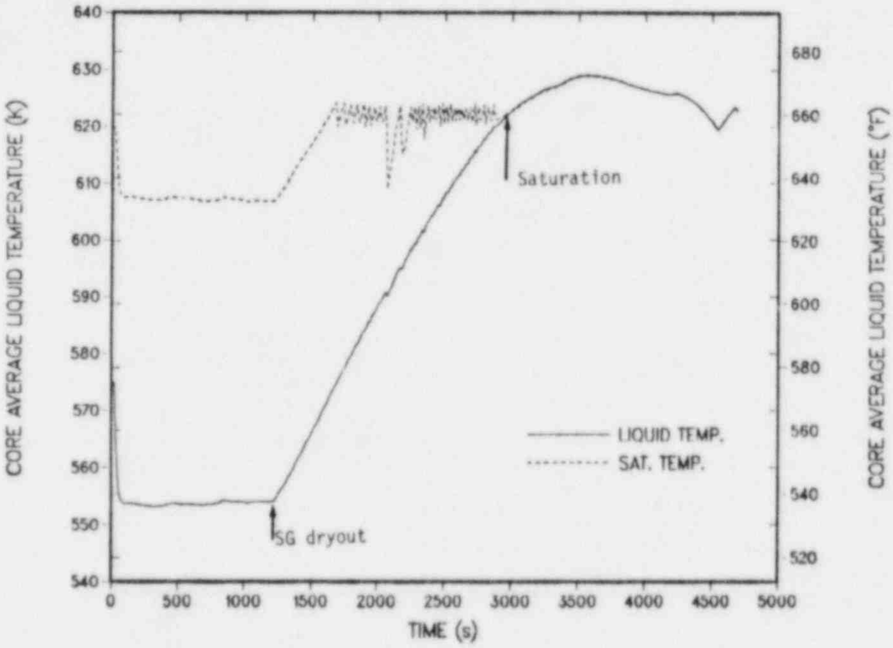

Fig. 35.
Core-liquid temperatures during the LOFW event for base transient.

during the base-case calculation. By 3200 s the pressurizer pressure reached 16.735 MPa (2427 psia), and the pressurizer safety valves lifted. By 3400 s the pressurizer level began decreasing. As the pressurizer level decreased, steam replaced the liquid being discharged through the PORVs, and this caused the pressure to decrease (Fig. 34) because the higher enthalpy of steam allows a given heat rejection rate to be achieved with less driving pressure. By 4000 s the core was 90% voided, and rapid core heatup ensued shortly afterwards (F1), as shown by Fig. 36.

b. Feed Transient. The feed case was performed by restarting the base case after enabling the SI system to deliver charging flow at 286 K (55°F) to the cold legs. The transient is identical to the base case until 3000 s, the time at which the SI actuation signal is estimated to occur. The RCPs were tripped 30 s after the SI signal in accordance with guidelines established following the Three Mile Island incident. The loop flows stagnated almost immediately after the RCPs were tripped. Figures 37 and 38 show that, although the charging flow delayed core heatup, it is not clear that damage to the fuel would not occur. The void fraction in Fig. 37 reached 60%, and the cladding

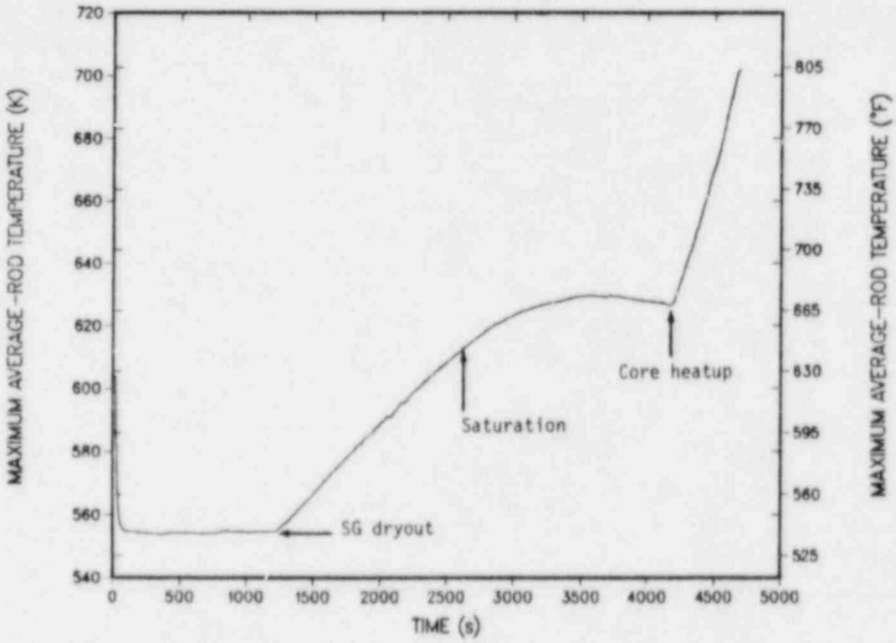

Fig. 36.
Cladding temperatures during the LOFW event for base transient.

temperature in Fig. 38 underwent two rapid increases characteristic of degraded heat transfer. Thus, the first success criterion is failed (F1). The PORV discharge flow at 10000 s is remaining relatively constant at about 12 kg/s (~26 lbm/s); consequently, the charging flow of 8.3 kg/s (18 lbm/s) is insufficient to replace the fluid being discharged through the PORVs, and it is unlikely that the trends in Figs. 37 and 38 would change. The calculation was terminated at 10000 s when it was concluded that recovery would not occur without serious core damage.

c. Feed and Bleed. Three feed-and-bleed cases were examined by varying the time of initiation. The first feed-and-bleed case was performed by restarting the base case after opening the PORVs and enabling the SI system to deliver charging and HPI flow at 286 K (55°F) to the cold legs. The restart began at 300 s because it was estimated that it would take an operator about five minutes to attempt a series of operations to restore feedwater or AFW before he would finally resort to feed-and-bleed cooling.* The RCPs were

*Information supplied by Tom Franz, NRC (September 1983).

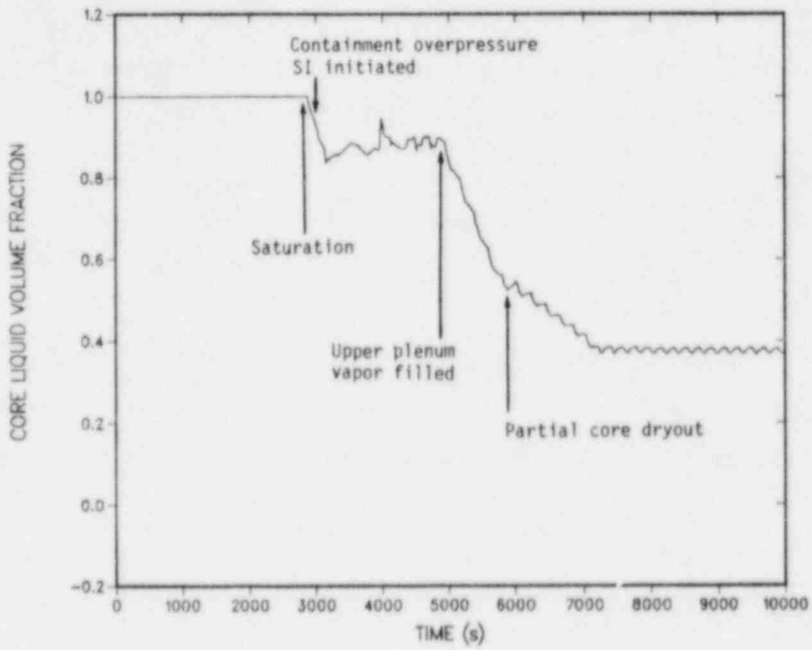

Fig. 37.
Core-liquid volume fraction during the LOFW event for feed initiated on containment overpressure.

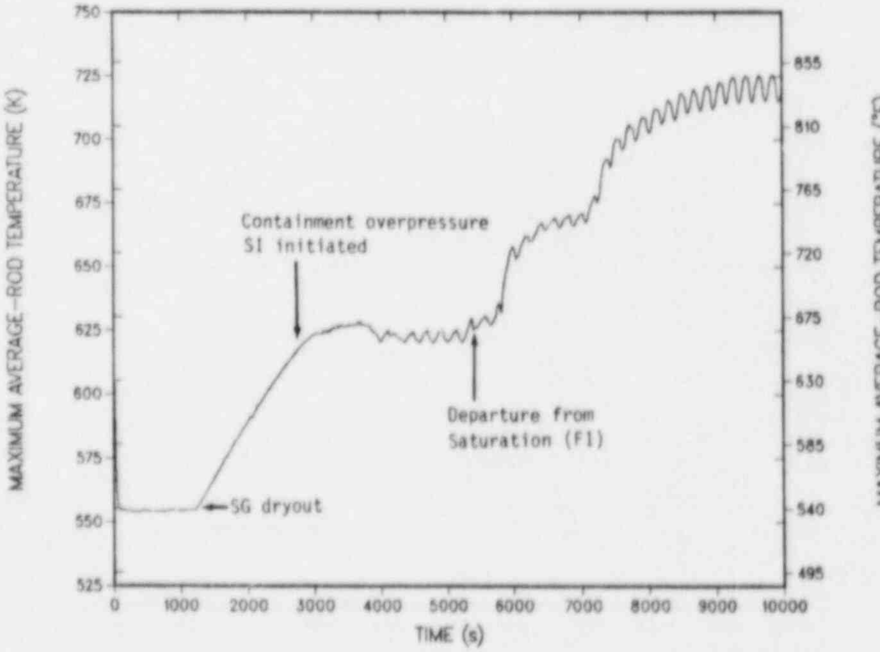

Fig. 38.
Cladding temperature during the LOFW event for feed initiated at containment overpressure.

tripped 30 s after the SI system began delivery in accordance with guidelines established following the Three Mile Island incident.

The primary pressure response to the feed-and-bleed operation is shown in Fig. 39. By 450 s the primary pressure had fallen to ~6 MPa (900 psia), and the hot legs were saturated (Fig. 40). The HPI flow reached ~50 kg/s (110 lbm/s) when the hot legs saturated, and by 800 s it restored subcooling to the hot legs. By 850 s, the SI system had refilled the primary, and the pressurizer was solid, and by 900 s the cold SI system flow had cooled the primary to the secondary temperature. After 900 s the steam generators actually became an additional heat burden on the SI system because blow-down cooling of the secondary ceased when the ARVs and TBVs automatically closed. The ARVs and TBVs are programmed to close when the average reactor temperature drops below 551 K (532^{o}F) and the steam pressure is below 6.2 MPa (900 psia). In a normal shutdown (without loss of feedwater), the secondary is depressurized through the TBVs to cool the primary.

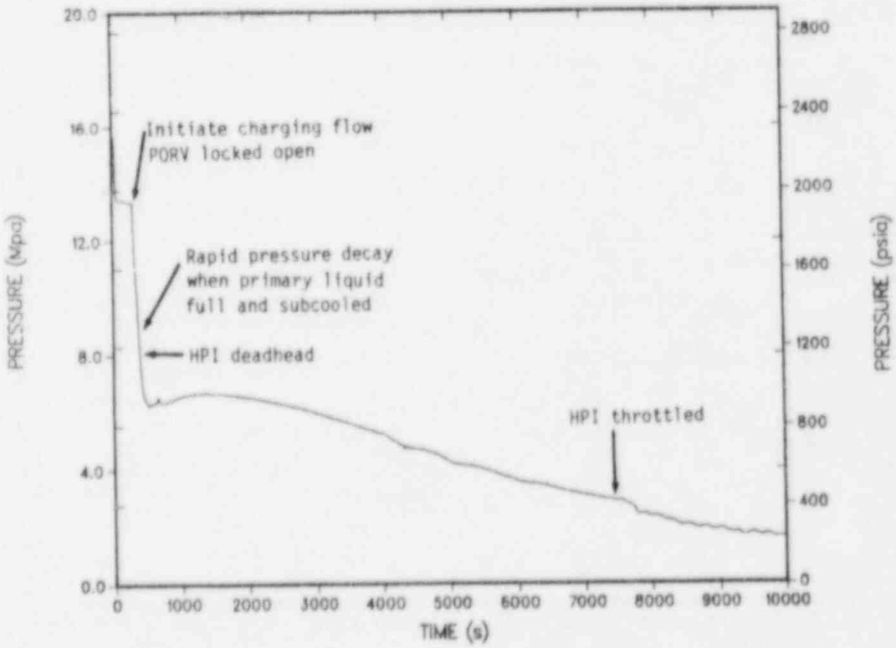

Fig. 39.
Primary pressure during the LOFW event for feed and bleed initiated at 300 s.

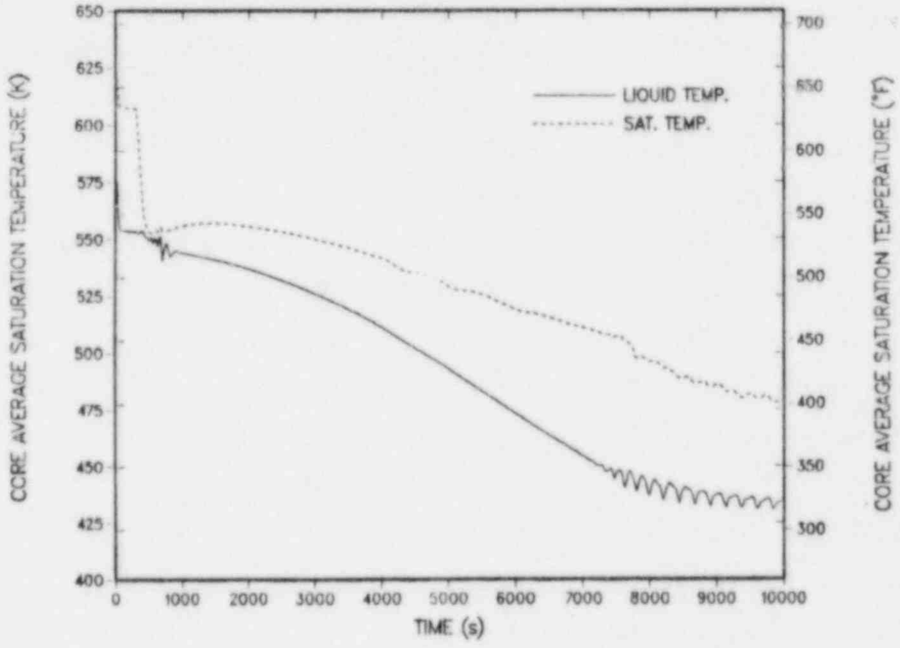

Fig. 40.
Core-liquid temperature during the LOFW event for feed and bleed initiated at 300 s.

The HPI flow continued to increase gradually with decreasing primary pressure until ~7500 s. By 7500 s the minimum subcooling margin in the hot and cold legs had reached 28 K (50°F), and thereafter the HPI flow was automatically throttled to maintain this minimum subcooling margin to reduce the risk of pressurized thermal shock. By 10000 s the primary pressures and temperatures had decreased to ~1.7 MPa (250 psia) and 435 K (325°F), respectively. The calculation was terminated at this point because it was clear that the pressure and temperature would have continued to decrease until the shutdown heat exchangers could be used to cool the plant to cold shutdown, thereby satisfying success criterion 2 (S2). Shutdown cooling can be initiated when the primary pressure and temperature are below ~1.9 MPa (270 psia) and 422 K (300°F), respectively. The SI system initially draws water from the RWST, which has a capacity of ~1400000 kg (3000000 lbm). During the 10000 s this calculation was run, the SI system used ~600000 kg (1300000 lbm). Furthermore, even if the RWST supply was exhausted, the SI system would automatically generate a recirculation actuation signal to switch the suction of the HPI pumps from the RWST to the containment sump.

The second feed-and-bleed operation was initiated at the time the secondary heat sink was lost (SGSD). The primary system pressure for this transient is shown in Fig. 41. Because the primary system is liquid-full and subcooled when the PORV is opened (see Fig. 41), the pressure rapidly drops to the saturation pressure corresponding to the hottest fluid in the primary. This effect is evident in the plot of core-rod temperature vs saturation temperature (Fig. 42). The initial pressure decline is sufficiently large that the HPI flow starts (Fig. 43). However, the combined charging and HPI flow is not sufficient to remove the decay heat following the loss of the secondary heat sink, and the primary begins to repressurize. By about 2150 s the pressure has increased enough that the HPI cutoff head is reached and HPI flow terminates. By about 3650 s voiding in the upper plenum has reached the level of the hot-leg piping attachments to the vessel. This terminates a pressurizer-like effect that supported the primary pressure and the primary pressure begins to decline. By 5800 s the primary pressure decays below the HPI cutoff heat and HPI flow is restored. With declining primary pressure and increasing HPI flow, the vessel

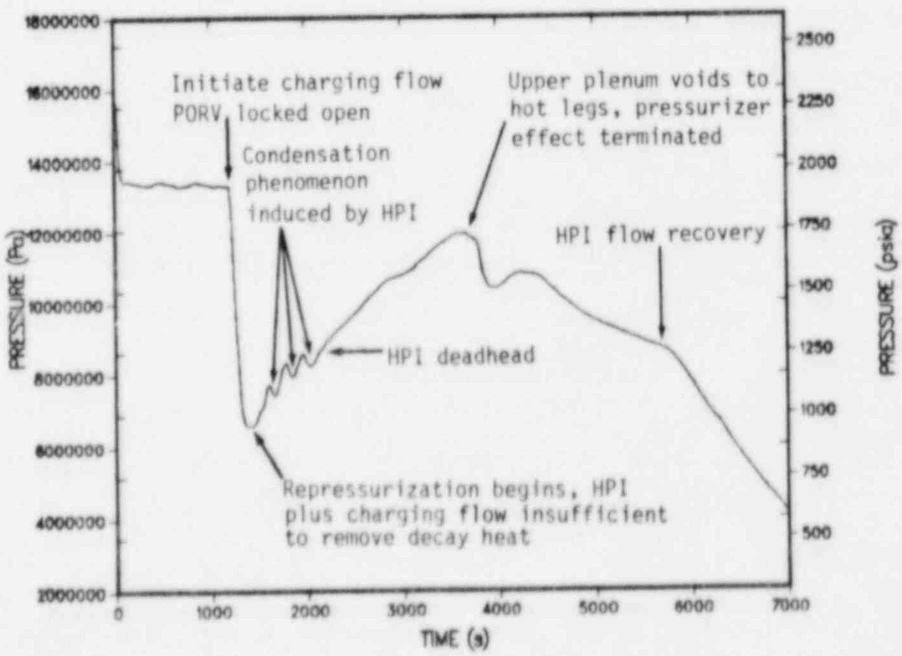

Fig. 41.
Pressurizer pressure during the LOFW event for feed and bleed initiated at SGSD.

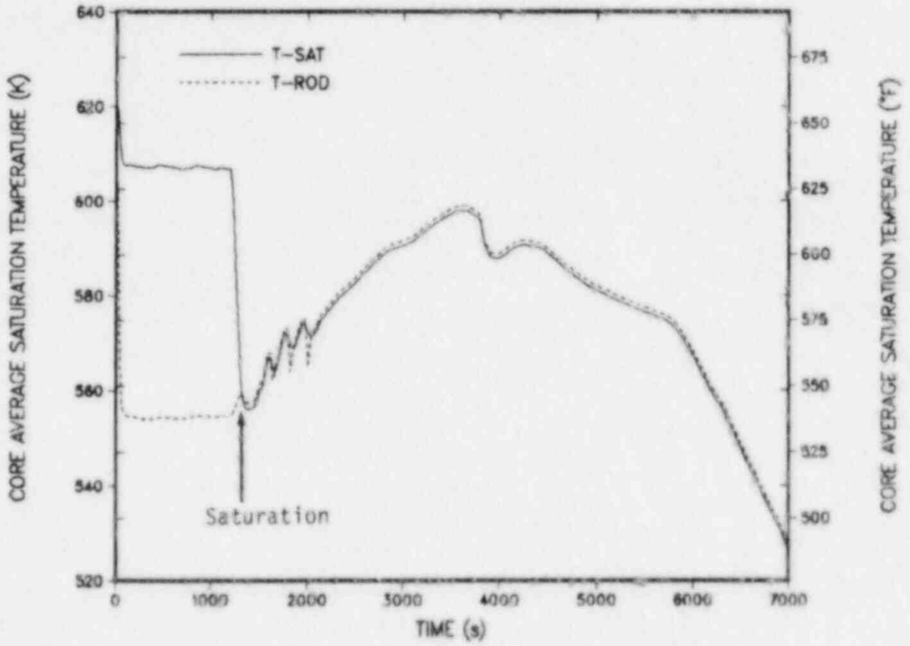

Fig. 42.
Cladding temperature during the LOFW event for feed and bleed initiated at SGSD.

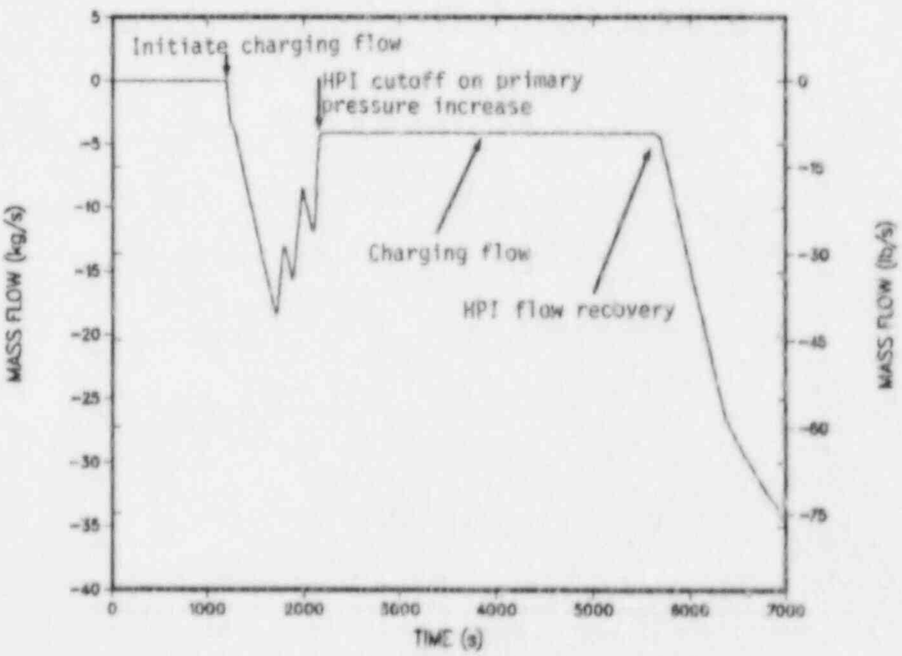

Fig. 43.
Safety injection flow rate during the LOFW event with feed and bleed initiated at SGSD.

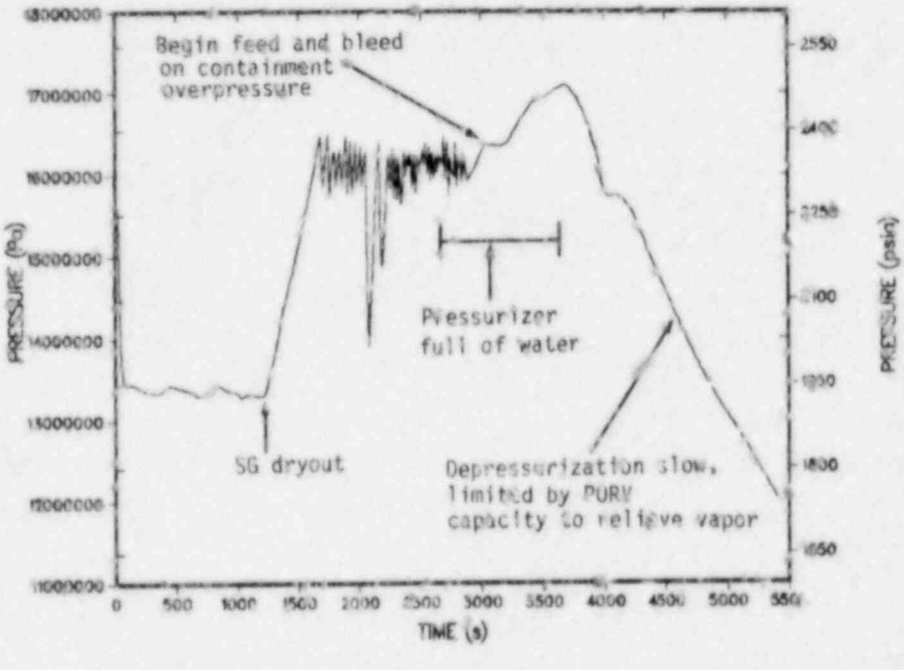

Fig. 44.
Pressurizer pressure during the LOFW event with feed and bleed initiated at containment overpressure.

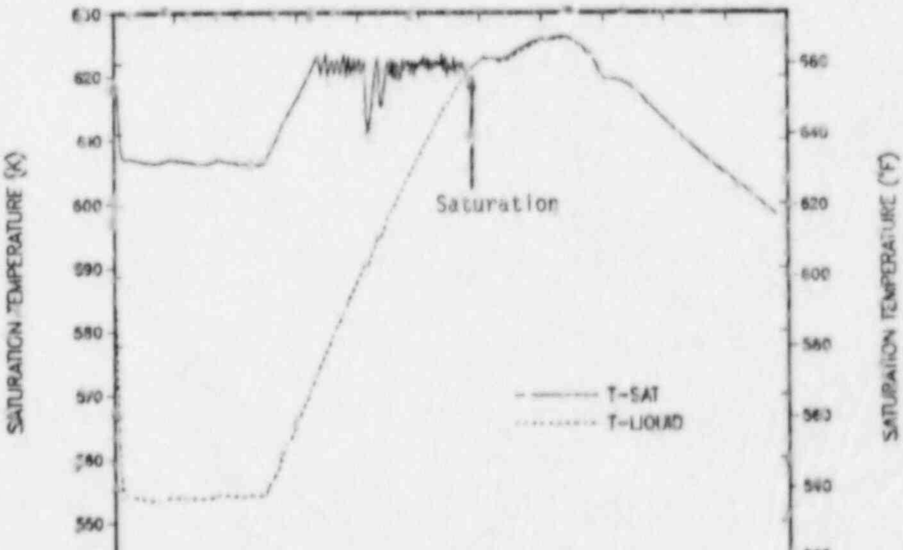

Fig. 45.
Hot-leg temperatures during the LOFW event with feed and bleed initiated at containment overpressure.

and primary mass inventory is increasing and success criterion 1 is satisfied (S1).

The third feed-and-bleed operation was initiated at the time the containment overpressure signal was generated. As seen in Fig. 33, the primary saturates before containment overpressure in Calvert Cliffs-1. The primary-system pressure for this transient is shown in Fig. 44. The transient is identical to the base LOFW transient until 3000 s when feed and bleed begins. Shortly before the containment overpressure signal is generated, the primary system saturates at 2900 s (Fig. 45). The PORV is liquid full by about 2300 s and remains liquid full until about 3500 s. During this time the volumetric relief out the PORVs is small and the primary pressurizes. However, by 3500 s the pressurizer liquid level begins to decrease, the volumetric flow out the PORVs increases and the primary pressure begins to decrease. However, because the primary is saturated, the rate of primary depressurization is slow and the pressure remains above the HPI system cutoff head. Without the HPI flow, the charging flow is insufficient to prevent core voiding and the cladding temperature departing from saturation (see Figs. 46 and 47). Success criterion 1 is not satisfied for this case (F1).

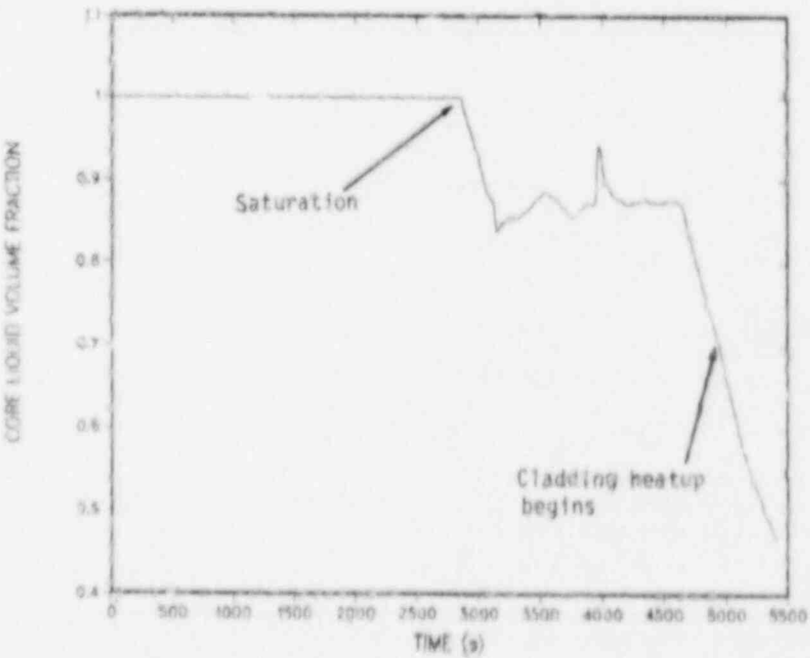

Fig. 46.
Core-liquid volume fraction during the LOFW event for feed and bleed initiated at containment overpressure.

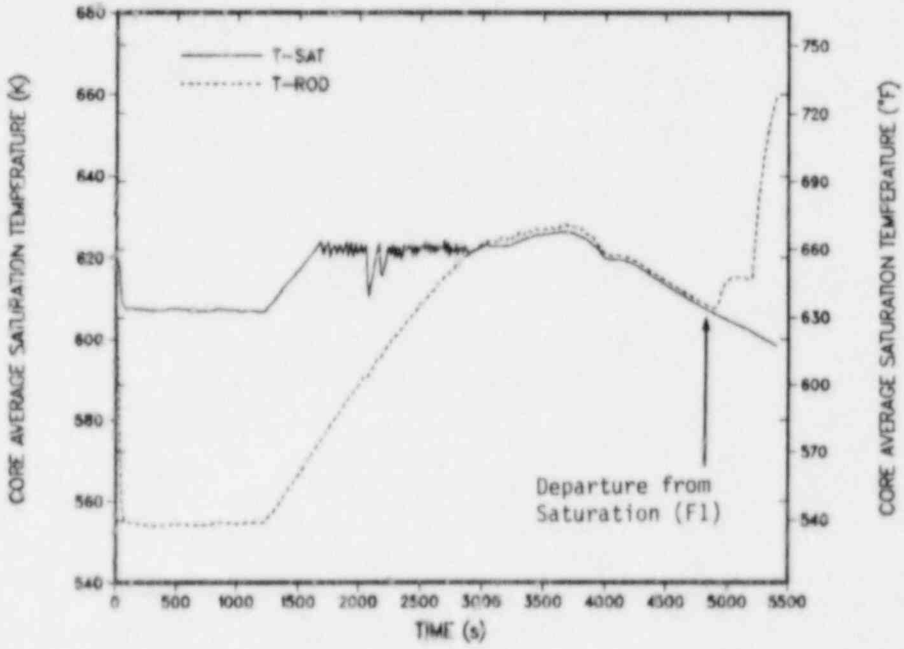

Fig. 47.
Cladding temperature during the LOFW event with feed and bleed initiated at containment overpressure.

Several cases of degraded equipment availability have been studied for Calvert Cliffs-1. The cases reported are for feed and bleed initiated at SGSD with less than nominal equipment availability. We briefly discuss two of these cases here. The first case considered the availability of only one HPI pump; the nominal number of HPI pumps is two. A core dryout was calculated at about 4700 s during the transition to a hot holding condition (F1). The pressure history for this transient is similar to the nominal case at SGSD as shown in Fig. 41. There were, however, two important differences. First, a reduced amount of HPI coolant was delivered to the primary during the initial primary pressure decay prior to repressurization. Second, the primary pressure did not decrease rapidly enough following creation of a vapor path to the PORV to restore HPI before core dryout occurred. The second case considered the availability of only a single PORV; the nominal number of PORVs is two.[15] Although this case was calculated to 16000 s, a core dryout was not observed. The entire transient is slowed relative to the nominal case because the loss of primary coolant is slowed with only a single PORV available. However, the rate of primary pressure decrease following creation of a vapor path to the PORV is also slowed. Thus, at 16000 s the primary pressure had not decreased below the pressure at which the HPI would be restored. Based on the extrapolated rates of primary coolant loss and primary pressure decrease, we believe a core dryout will occur during the transition to a hot holding condition (F1). Therefore, we see that a degradation of either PORV relief or HPI delivery capacity will result in a failure to reach a hot holding condition, whereas nominal PORV and HPI capacity at SGSD are sufficient to permit a successful transition to a hot holding condition.

3. Combined MSLB/LOFW Event. To determine the effect of rapid secondary depressurization on primary heatup, it was postulated that the main steam line on loop A suffers a double-ended break outside the containment and upstream of the main-steam-line isolation valve. Because the steam lines of both steam generators connect into a common header, both steam generators initially blow down to the atmosphere. Steam flow from the steam generators out the break results in an initial overcooling of the primary system.

a. Base Transient. The event sequence for the base-case transient is summarized in Table XII. An SI signal on steam-line pressure differential is generated following the steam-line break at 14.5 s and initiates a reactor trip. At 20.1 s, a steam generator isolation signal on low steam-generator pressure is initiated. This generates a main feedwater isolation signal and the main steam isolation valves begin to close. Normal recovery from the transient would be effected by automatic actuation of the turbine-driven auxiliary feedwater pumps within a minute. However, both auxiliary feedwater systems are assumed to fail. The broken-loop steam-generator blowdown is completed by 200 s. The PORV does not open until the primary-system pressure increases following dryout of the intact-loop steam generator. The ARVs on the intact steam-generator steam lines open at 900 s, and the secondaries of the intact steam generators empty of water by 2600 s. The transient calculation was terminated at 2600 s because the remaining course of the transient is nearly identical to the LOSP event following dryout of the intact-loop steam generators.

The plant response characteristics for the base case are shown in Figs. 48 through 52. The primary-system pressure during the combined MSLB/LOFW base transient is shown in Fig. 48. The primary is overcooled during the first 200 s as the broken-loop steam generator blows down. This overcooling results in a rapid primary liquid contraction, causing the pressurizer to void and a vapor bubble to form in the upper head of the reactor. This effect can be seen in the plot of the vessel upper-head void fraction, Fig. 49. The ARVs on the intact-loop steam generators open at 920 s, with the SRVs opening at 1140 s

TABLE XII

CALVERT CLIFFS-1
MSLB/LOFW EVENT SEQUENCE FOR BASE-CASE TRANSIENT (NO HPI)

Time (s)	Event
0.0	Main-steam-line break
14.5	SG isolation signal on low steam generator
14.5	Reactor trip
20.1	Begin feedwater isolation
20.1	Main-steam-isolation valves begin to close
27.6	Turbine-driven pumps fail to deliver auxiliary feedwater
30	Motor-driven pumps fail to deliver auxiliary feedwater
200	Broken-loop steam-generator blowdown completed
920	ARVs on intact steam lines open
2250	Intact-loop SG secondaries empty of water
2650	Calculation terminated

beginning a period of boiling the secondary inventory until dryout occurs at about 2500 s. While the steam-generator-secondary liquid inventory is being boiled off, all the reactor decay heat is rejected to the secondary and the primary remains at about 7.5 MPa. After 2300 s the heat transfer between the primary and secondary degrades as the secondary liquid inventory decreases and the primary again begins to repressurize to the PORV setpoint. The maximum core average rod cladding temperatures are shown in Fig. 50. The core saturation and liquid temperatures are shown in Fig. 51. Subcooling begins to decrease after 2500 s when the intact-loop steam generator dries out and primary-system heatup begins.

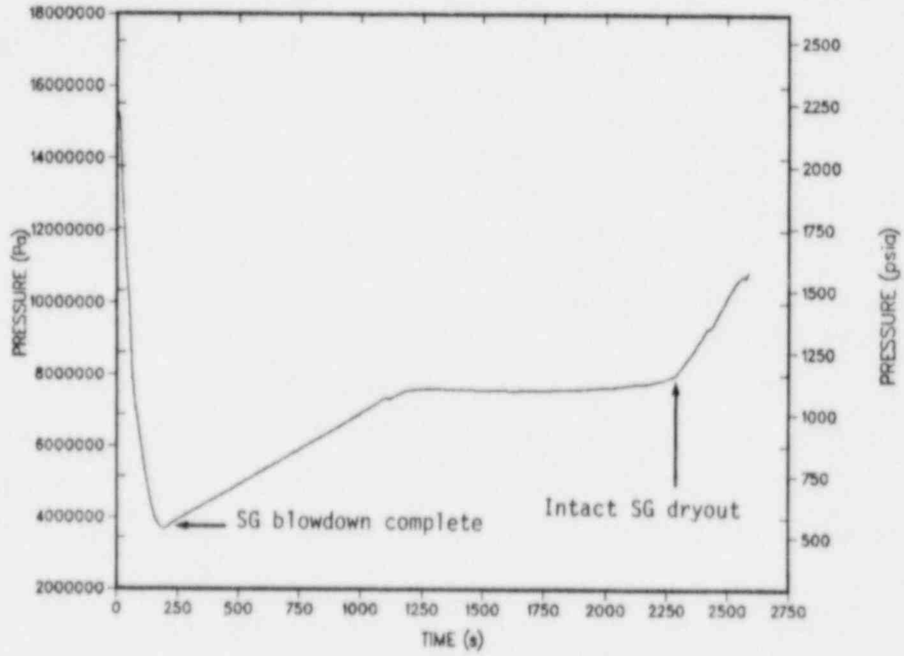

Fig. 48.
Primary pressure during the MSLB/LOFW event for base transient.

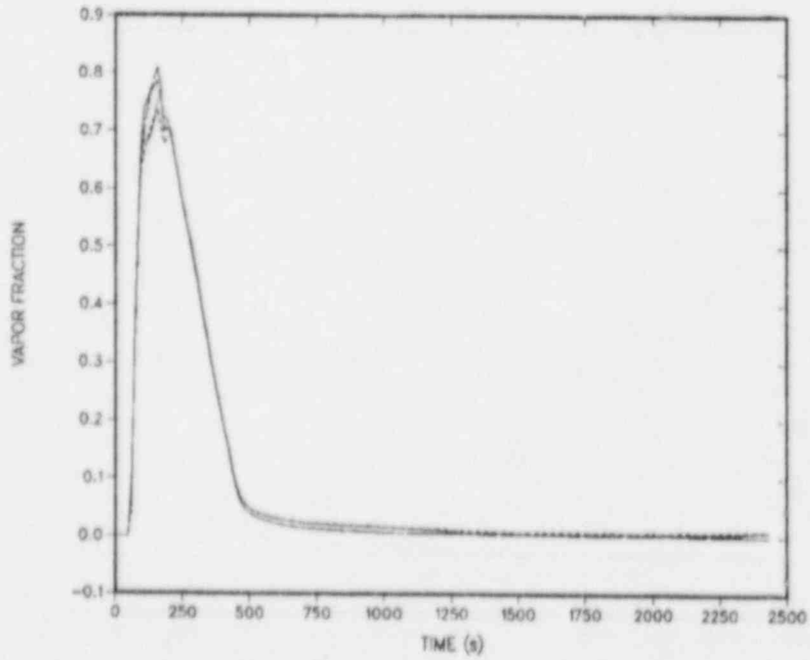

Fig. 49.
Upper-head void fraction during the MSLB/LOFW event for base transient.

The rapid secondary-side depressurization of the broken-loop steam generator is shown in Fig. 52. The intact-loop-secondary pressure increase, from about 25 to 35 s of the transient, is caused by isolating the broken-loop steam generator. The primary-side overcooling is so rapid that the intact-loop steam-generator heat transfer reverses for about 500 s. The maximum heat addition to the primary occurs just before the broken-loop steam-generator blowdown is completed at about 200 s. After about 500 s, primary-to-secondary heat transfer is re-established, and the intact-loop secondary pressure increases until the ADV setpoint is reached at 880 s.

Intact steam-generator-secondary dryout for the combined MSLB/LOFW transient occurs about 2500 s. At this time, heatup of the liquid-full primary is beginning. For the simple LOFW transient, the steam generators boil dry at 1250 s, and heatup of the liquid-full primary is just beginning. From this point on, the two transients are nearly identical even though there are significant differences in the transient prior to this time. It is concluded that the LOFW transient and parametric studies initiated subsequent to

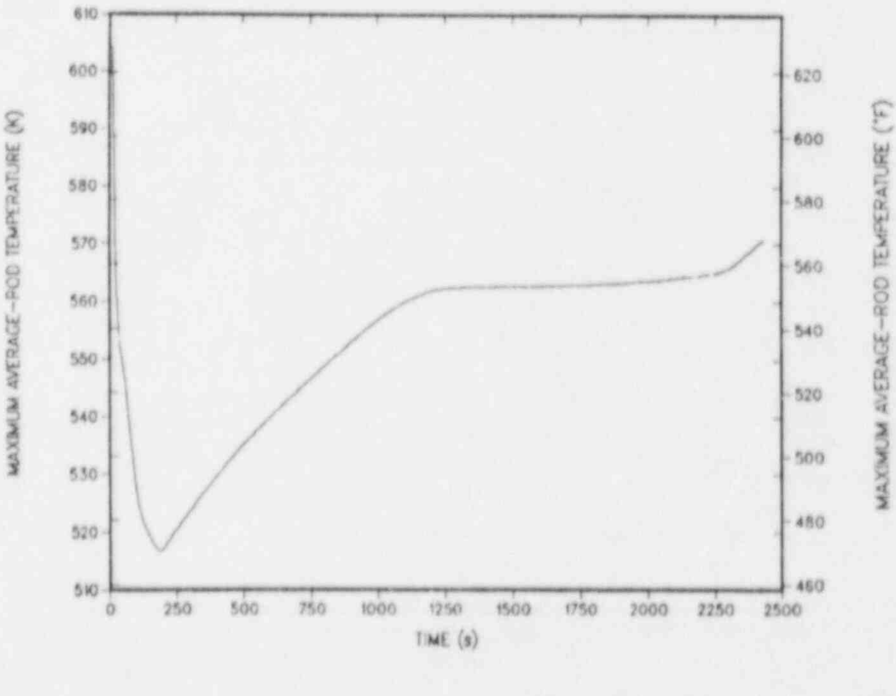

Fig. 50.
Cladding temperatures during the MSLB/LOFW event for base transient.

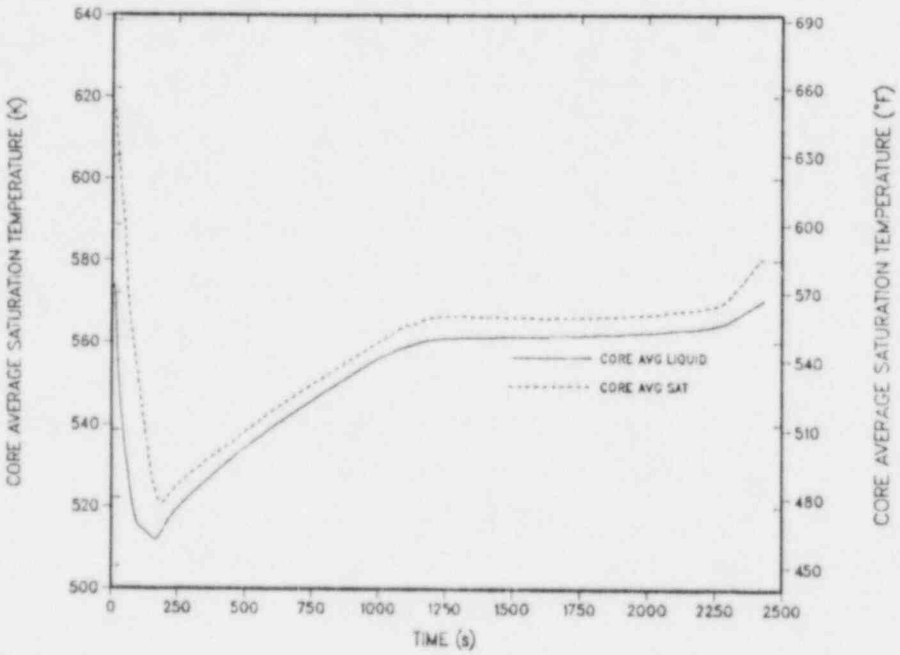

Fig. 51.
Core-liquid temperatures during the MSLB/LOFW event for base transient.

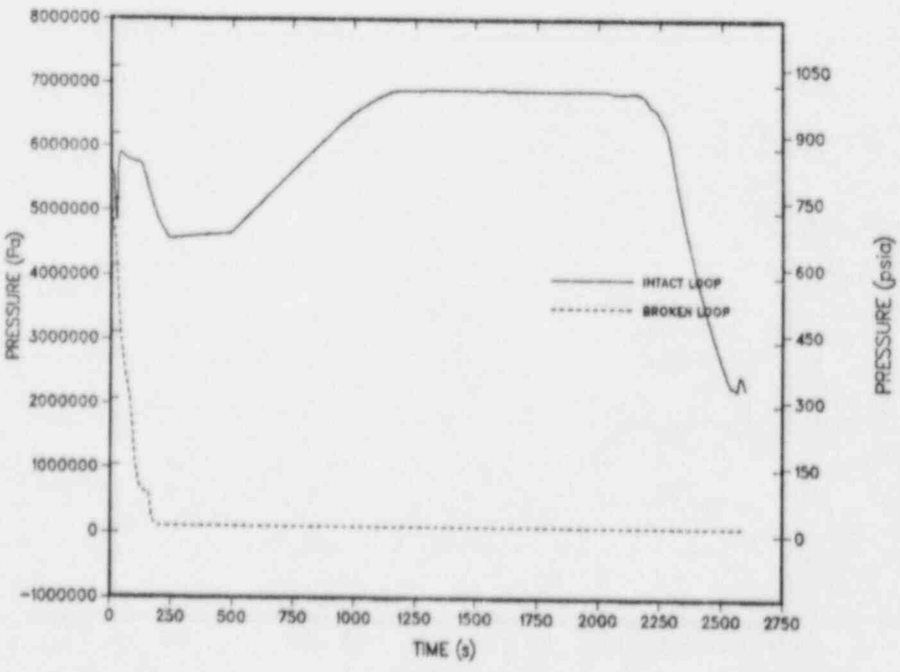

Fig. 52.
Steam-generator pressures during the MSLB/LOFW event for base transient.

steam-generator-secondary dryout (1250 s) can be used directly for the combined MSLB/LOFW transient.

b. Feed Transient. Because of the rapid initial overcooling of the primary system from the rapid steam-generator blowdown, automatic SI is initiated very early in the transient (31 s). The signal is generated on low pressurizer pressure (12 MPa). The sequence of events up to this point is identical to the base case transient.

The broken-loop steam-generator blowdown is completed by 200 s. The primary system begins to repressurize and the PORV setpoint is reached at about 1350 s. The PORV continues to cycle at setpoint and the pressurizer becomes solid at about 2000 s. Core subcooling increases throughout this transient and at 2000 s, the subcooling margin has stabilized at about 60 K ($108^{o}F$). Figures 53 through 56 present the pressurizer pressure, level, core subcooling margin and vessel upper-head void fraction for this transient.

The calculation was terminated at 2600 s as reactor conditions were stable. The early HPI actuation assured a successful feed mode (S1).

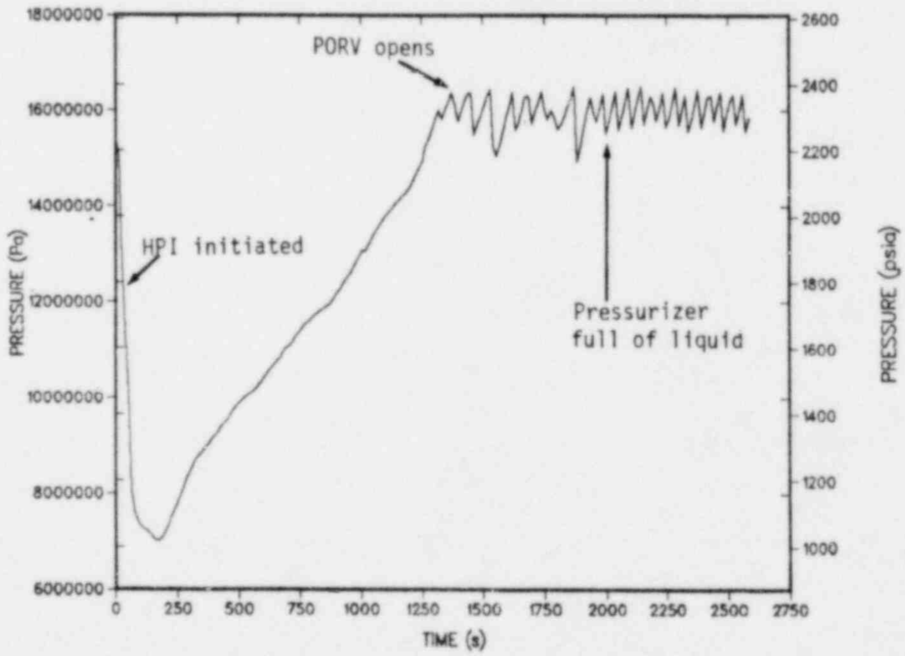

Fig. 53.
Pressurizer pressure during the MSLB/LOFW event for feed initiated on low pressurizer pressure at 31 s.

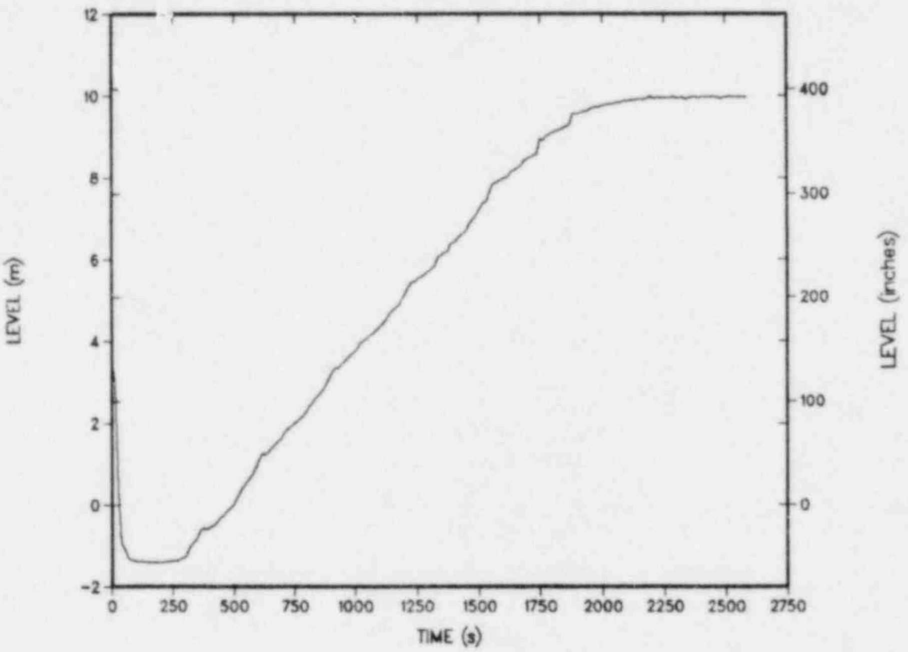

Fig. 54.
Pressurizer level during the MSLB/LOFW event with feed initiated on low pressurizer level at 31 s.

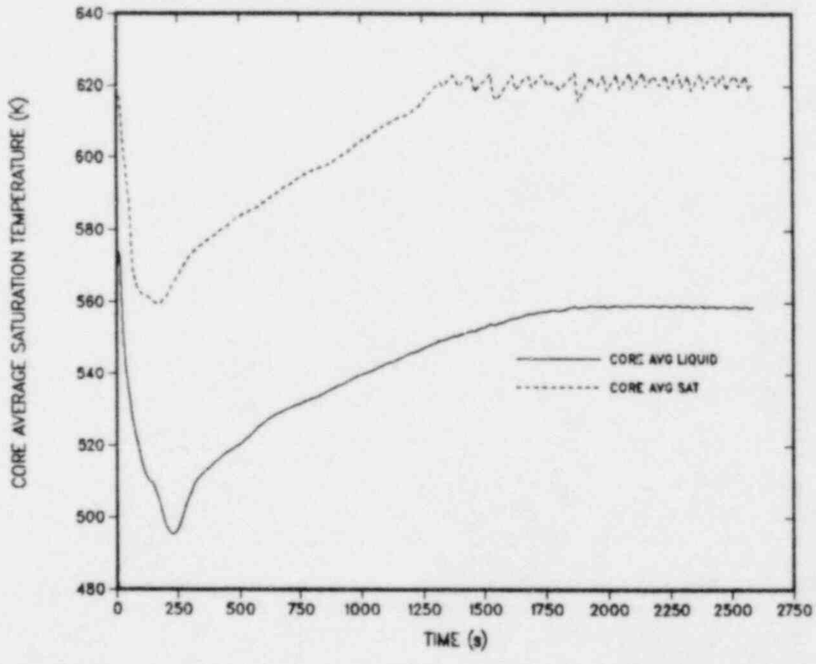

Fig. 55.
Core average temperatures during the MSLB/LOFW event with feed initiated on low pressurizer level at 31 s.

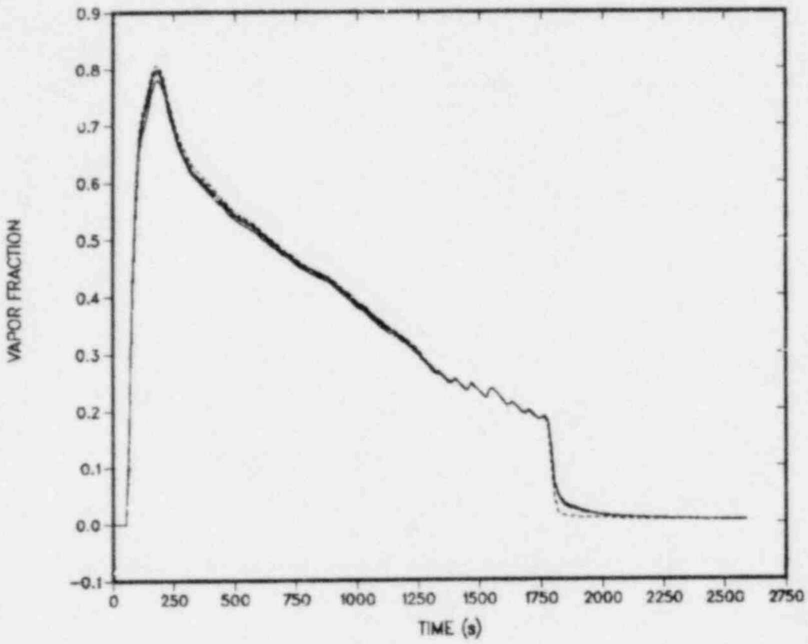

Fig. 56.
Core upper-head void fraction during the MSLB/LOFW event with feed initiated on low pressurizer pressure level at 31 s.

c. Feed-and-Bleed Transient. The feed-and-bleed transient was not run as the success of the feed transient assured a successful feed and bleed (S1).

4. Combined MFLB/LOFW Event. This event was not calculated for Calvert Cliffs-1. However, it was calculated for Zion-1 and a discussion of the transient phenomena is presented in Sec. V.A.4. It is expected that the timing will be very similar to the combined MSLB/LOFW transient for Oconee-1. Following the MFLB, primary system overcooling will occur as the broken-loop steam generator blows down through the feed-line break. However, the overcooling will not be as severe as with the MSLB because a large fraction of the inventory will flash as it passes out the break. This liquid will not absorb energy from the primary. For the MSLB, nearly the entire liquid inventory flashes in the tube-bundle region, extracting energy from the primary. The time to steam-generator-secondary dryout is expected to be slightly earlier for the combined MFLB/LOFW transient when compared to the MSLB/LOFW transient. Timing of specific events can be estimated from the LOFW and combined MSLB/LOFW events.

5. Combined SGTR/LOFW Event. The initiating event was the rupture of a single steam-generator tube. This caused a rapid depressurization of the primary system. A reactor-trip signal was generated when the primary-system pressure decreased to the low-pressure setpoint (14.48 MPa, 2100 psia). The main feedwater supply was terminated and the turbines were bypassed concurrent with the reactor trip. As a part of the problem definition, the AFW system was unavailable.

a. Base Transient. The event sequence for the base transient is given in Table XIII. Reactor trip, caused by low primary-system pressure, occurred at 170 s. The intact and damaged steam generators dried out at 2139 s and 2536 s, respectively. The RCPs continued to operate throughout the base case. The primary system began to heat up and expand after the steam generators dried out. The primary system repressurized to the PORV setpoint (16.55 MPa, 2400 psia). Because makeup water from the HPI system was not provided, the core would have

TABLE XIII

CALVERT CLIFFS-1
COMBINED SGTR AND LOFW EVENT SEQUENCE FOR
BASE-CASE TRANSIENT (NO HPI)

Time (s)	Event
0.0	Double-ended single SGTR
50	Pressurizer backup heaters activated
169.7	Reactor trip caused by low primary-system pressure, feedwater pumps tripped, turbine trip
170.2	TBV opens
177	Pressurizer full of vapor
2139	Intact-steam-generator secondary full of vapor
2536	Damaged-steam-generator secondary full of vapor
2537	Core heatup begins (0.0224 K/s, 145°F/h)
2591	Steam-generator isolation signaled by low secondary pressure
3323	Pressurizer refill starts
5117	Hot legs saturate
5391	End of calculation

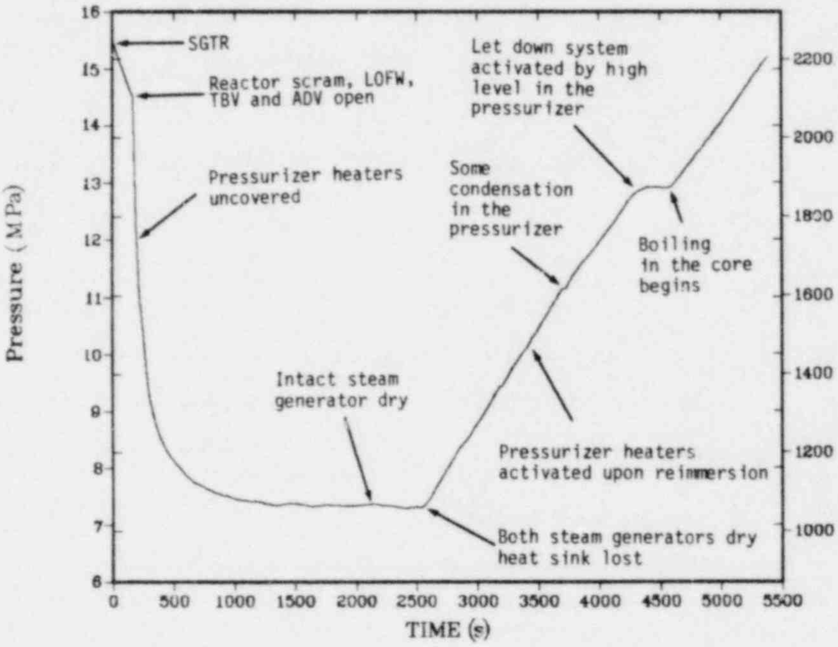

Fig. 57.
Pressurizer pressure during the SGTR/LOFW event for base transient.

eventually filled with vapor and an uncontrolled core heatup would have begun (F1). Fig. 57 illustrates the system pressure for the base case.

b. Feed Transient. If normal operation of the SI system is assumed, the HPI is initiated on a low primary-system pressure (12.1 MPa, 1755 psia). The actual injection process begins at 8.85 MPa (1284 psia). The operator turned the RCPs off 30 s after verifying the functioning of the HPI system. Natural circulation was established in the core and both loops. The SI initiation was not sufficient for core cooling because of the low delivery rate at higher pressures. The low-head limitation of the HPI caused a series of pressure and temperature oscillations in the primary and secondary systems. The HPI pressurized the primary to 8.85 MPa (1284 psia); the tube-rupture flow increased; system pressure decreased below the HPI limit; HPI flow increased; and so on. Therefore, the HPI flow was not sufficient to cool the core. The intact-steam-generator secondary dried out at 3174 s (53 min). The damaged-steam-generator was boiling off its inventory at a much slower rate because of the primary-to-secondary leakage. Figure 58 illustrates the system pressure for this transient.

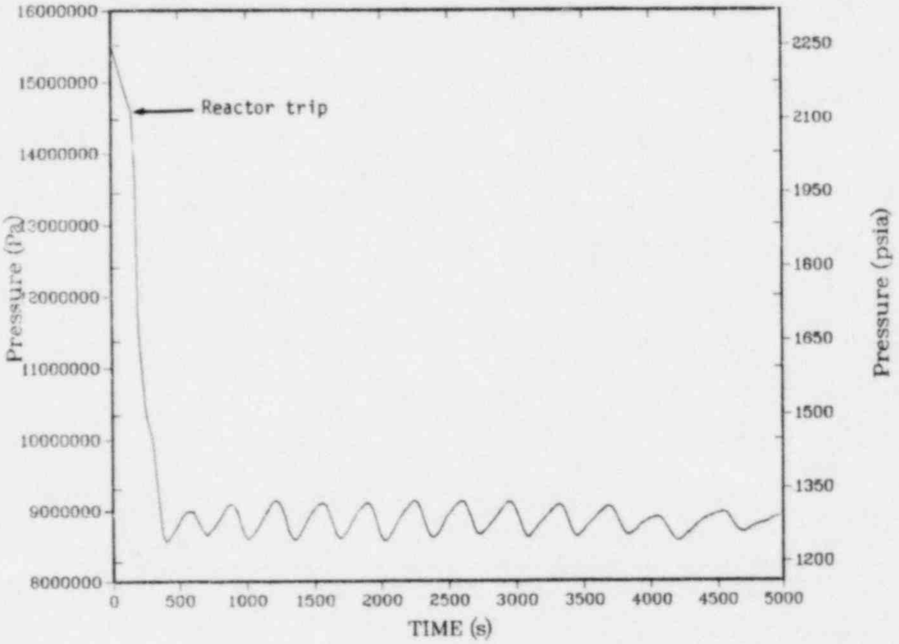

Fig. 58.
Primary pressure during the SGTR/LOFW event for feed initiated at low pressure (12.1 MPa).

c. Feed-and-Bleed Transient. The objectives of the operator in a SGTR/LOFW accident scenario are twofold: first, to maintain core cooling; and second, to mitigate primary-to-secondary flow and to reduce the environmental radiological releases. To limit the release, operator intervention is required to depressurize the primary system and stop the tube-rupture flow. This implies a feed-and-bleed procedure for which the operator opens the PORV after verifying a SGTR, loss of main and auxiliary feedwater, and SI actuation. This occurred approximately 6 min after reactor trip or 9 min after SGTR (at 550 s). The primary system rapidly depressurized to about 6.2 MPa (897 psia) and then stabilized at 6.7 MPa (967 psia). The feed-and-bleed-mode cooling decreased the primary-to-secondary leakage considerably and established a core-cooling rate of 3.67 K/s (23.8^{o}F/h). The tube-rupture flow was reduced to 4.1 kg/s (3.3×10^4 lbm/h). The initial tube-rupture flow was 20 kg/s (1.6×10^5 lbm/h). Figure 59 illustrates the primary-system pressure. The feed-and-bleed procedure was effective in providing stable cooling of the reactor (51).

B. Summary Insights and Conclusions

The feed and feed-and-bleed transients investigated for Calvert Cliffs are uniquely distinguished by the PORV and SI system characteristics. The PORV relief capacity is smaller relative to Zion, but larger than Oconee. In the base-case LOSP and LOFW transients, the primary pressure continued to increase for a short period after the PORV and SRVs opened. This characteristic was observed in Oconee. At normal operating pressure, the SI system has limited injection capability relative to the two other plants. Above the HPI shutoff head of 8.8 MPa (1275 psia), only the charging flow (8.3 Kg/s, 18 lbm/s) from the constant displacement pumps is available.

Upon loss of secondary cooling, for feed or feed and bleed to be effective, it must be initiated early in the transient and with nominal PORV and HPI availability. Our investigations have shown that in order to bound all transients, SI must be initiated by SGSD with nominal equipment availability to meet success criteria (S1). If SI is not manually initiated, the automatic initiation at containment overpressure may be effective in preventing gross fuel damage, but significant (greater than 50%) core voiding was observed to persist

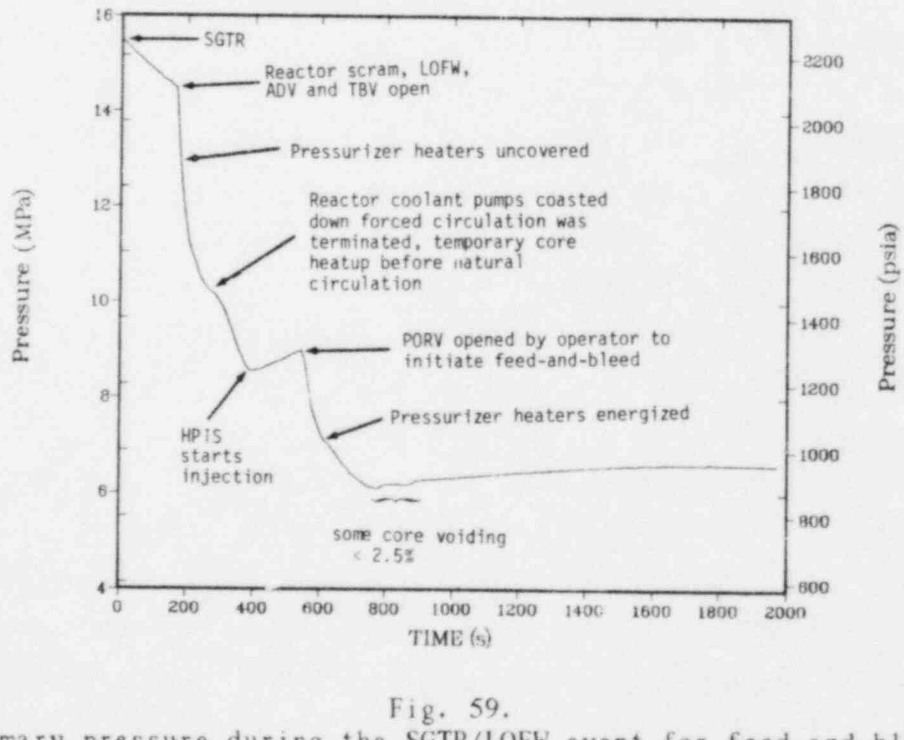

Fig. 59.
Primary pressure during the SGTR/LOFW event for feed and bleed initiated at 9 min.

for an extended period (see Figs. 30 and 37). The actual extent of fuel/cladding damage must be based on hot-channel factors and other considerations and is beyond the scope of this report. If nominal equipment is not available (either both PORVs or two of three HPI pumps), we found that the feed-and-bleed procedure fails if initiated at SGSD.

Based on operator guidelines, early manual initiation of SI (SGSD) can be expected. The earliest steam-generator dryout time was 1200 s for the LOFW event. This would allow 20 min for an operator either to re-establish secondary cooling or actuate SI. This large secondary heat capacity is typical of the U-tube-type steam-generator plants. In contrast, Oconee, with the once-through straight-tube steam generator, showed steam-generator dryout times of less than 10 min, using realistic trip and feedwater coastdown times for the LOFW event.

If feed and bleed is initiated prior to SGSD, hot-shutdown conditions can be achieved successfully (S2). To prevent primary overcooling, the HPI may require throttling later in the transient to prevent exceeding thermal-shock limits.

If feed is initiated after SGSD, the PORV should not be locked open until a subcooling margin has been established. In this case, initiating depressurization with a partially voided core would aggravate core conditions as the limited charging flow at high pressure (above 8.8 MPa, 1275 psia) cannot keep up with the PORV exit-flow rate. Feed and bleed after steam-generator dryout should not be attempted until a prior feed mode has established subcooled core conditions.

V. ZION-1 INSIGHTS

Zion-1 (Ref. 16) is a 3250-MWt W four-loop PWR operated by Commonwealth Edison. The reactor-coolant system consists of the reactor vessel, four SGs, four RCPs, the pressurizer, and the piping connecting these components. The SGs are vertical-shell and U-tube heat exchangers with integral moisture-separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the hemispherical bottom head of the steam generator. Steam is generated on the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel. Steam dryers are employed to increase the steam quality to a minimum of 99.75% (0.25% moisture). The moisture separator recirculation flow mixes with feedwater as it passes through the annulus formed by the shell and the tube-bundle wrapper. The

RCP is a vertical, single-stage, centrifugal, shaft-seal pump designed to pump large volumes of main coolant at high temperatures and pressures. The pressurizer connects to one of the four primary loops. Electrical heaters are installed through the bottom head whereas the spray nozzle, relief and safety valve connections are located in the top head of the pressurizer.

The SI system includes HPI and LPI capability as well as accumulators. Zion-1 is a high-pressure SI plant:* two safety-grade centrifugal charging pumps deliver a total of 20.6 kg/s (45.4 lb/s) at the PORV setpoint, and two SI pumps provide additional safety-grade coolant flow at intermediate pressures. Four accumulators are provided, each connected to one of the cold legs. Shutdown cooling can be initiated when the primary pressure and temperature are below 3.04 MPa (440 psig) and 450 K (350^{o}F), respectively. During the injection mode, the centrifugal charging pumps take suction from the RWST. The discharge from the pumps initially sweeps the concentrated boric acid in the boron injection tank into the reactor-coolant system. The charging pumps and safety injection pumps are commonly referred to as "high-head pumps" and the residual-heat removal pumps as "low-head pumps." Likewise, the term "high-head injection" is used to denote charging pump and safety injection pump injection and "low-head injection" refers to RHR pump injection. The safety injection pumps also take suction from the RWST and deliver borated water to the cold legs of the RCS. The safety injection pumps begin to deliver water to the RCS after the pressure has fallen below the pump-shutoff head. The RHR pumps take suction from the RWST and deliver borated water to the reactor-coolant system. These pumps begin to deliver water to the RCS only after the pressure has fallen below the pump shutoff head.

A schematic of the reactor-coolant system is presented in Fig. 60. The TRAC-PF1 input model of the Zion-1 plant is described in Appendix B.III. For all Zion-1 calculations we assumed the nominal equipment included two each

*Westinghouse plants with SI shut-off pressures greater than the PORV setpoints are considered high-pressure SI plants: Zion-1 is such a plant. Westinghouse plants with shut-off pressures less than 10.6 MPa (1540 psia) are classified as low-pressure SI plants. Plants with SI shut-off pressures greater than 10.6 MPa but less than the PORV setpoint are classified as intermediate-pressure SI plants. High-pressure SI systems do not have charging pumps other than in the SI system, whereas LP and IP SI plants have separate high-pressure charging pumps that are part of the chemical and volume-control system (non-safety grade) rather than the SI system.

charging pumps and pumps. The SI delivery characteristics for these "high-head pumps," were obtained from the Zion Final Safety Analysis Report.[17] Two PORVs were modeled for each nominal calculation. We used the 1973 American Nuclear Society decay-heat curve.

A. Summary Results

We have prepared detailed reports of a series of studies that examine LOFW transients in the Zion-1 plant.[18-25] In this report, we summarize these results. Again, we report on the same five events in the common set as reported for Oconee-1 and Calvert Cliffs-1. We also include in this report summary results for a base LOFW transient for which the reactor trip is as specified by Westinghouse in Ref. 26. In addition, the results of a feed-and-bleed procedure initiated at primary-system saturation are included. These results will not be reported elsewhere. The results of our Zion-1 calculations are summarized in Fig. 61. In addition to showing the success or failure of each study, the period in which the feed or feed-and-bleed operation was initiated relative to key reactor events is shown. The success criteria were satisfied for each non-base case calculated having an initiation time no later than when the containment overpressure signal is generated.

1. LOSP-Induced LOFW Event. The LOFW transient was initiated by an LOSP event. It was assumed that an LOSP results in an immediate trip of the RCPs, a reactor trip, a turbine trip, and closure of the TSVs. It was also assumed that feedwater flow drops to zero instantly at the start of the transient and that no auxiliary feedwater is available.

a. Base Transient. The event sequence for the base case LOSP transient without SI system operation is given in Table XIV. Normal recovery from this transient would be effected by automatic actuation of the turbine-driven AFW pumps, which would begin to deliver 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. Because AFW is not available, the steam-generator ARVs open at 95 s. The steam-generator inventory is depleted by boiling and at about 4170 s the steam-generator secondaries dry out. With steam-generator-secondary dryout the primary-system heat sink is lost, and the primary system begins to heat up and expand. The primary-system pressurization to the PORV setpoint at 4200 s is shown in Fig. 62. Saturation of the primary system hot legs at 6650 s is shown in Fig. 63. The uncontrolled heatup of the reactor core begins at about 8300 s as shown in Fig. 64 (F1).

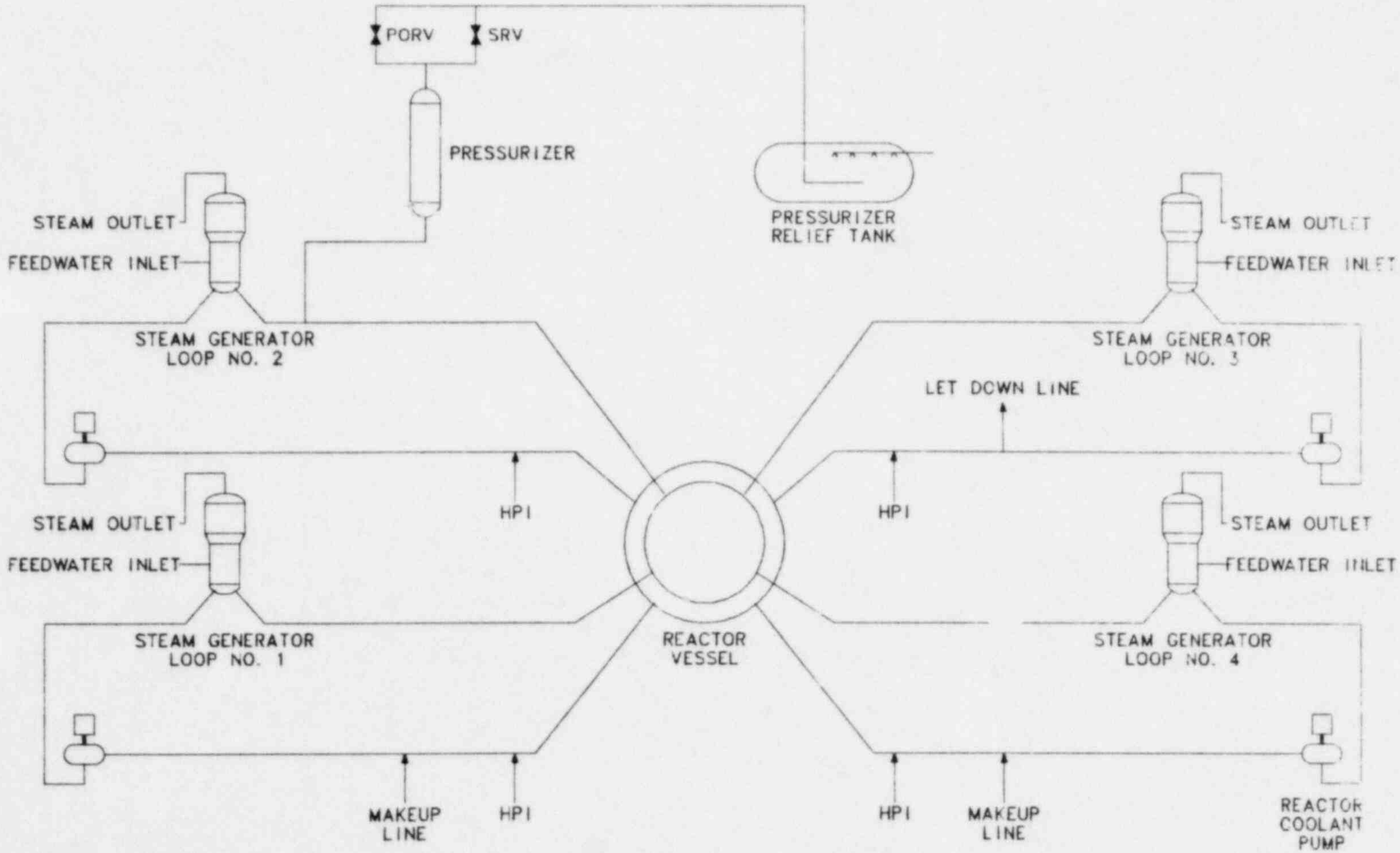

Fig. 60.
Zion-1 reactor-coolant system.

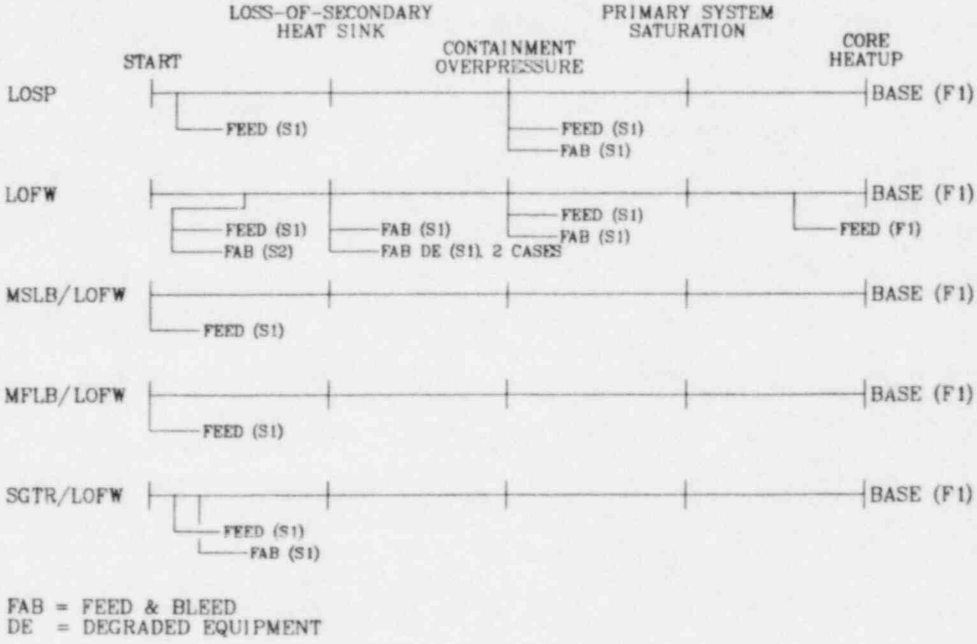

Fig. 61.
Zion-1 success/failure chart.

The base case LOFW event was calculated using TRAC-PF1 and compared to an extensive series of Zion-1 LOSP calculations prepared with TRAC-PD2 (Ref. 18). It was concluded that the TRAC-PF1 and TRAC-PD2 calculations were in essential agreement when calculating the base LOSP-induced LOFW transient. Therefore, the summary results for the TRAC-PD2-calculated feed and feed-and-bleed calculations will be included as part of this study.

b. Feed Transient. The event sequence for the feed case is presented in Table XV. The HPI injection is initiated at 5800 s on high containment pressure. This is about 600 s earlier than the time of high containment pressure calculated with TRAC-PF1. The earlier TRAC-PD2 calculated time is attributed to differences in modeling feedwater flow termination and liquid carryover out of the steam-generator secondary. The TRAC-PF1 times are believed to be more accurate and should be used for reference. The primary-system pressure response to the feed operation is shown in Fig. 65 from Ref. 18. Saturation of the primary coolant at 7200 s is shown in Fig. 66. The ECC flow in the feed mode is insufficient to remove the decay energy of the reactor without boiling the primary coolant. Depletion of the primary coolant continued

TABLE XIV

ZION-1
LOSP EVENT SEQUENCE FOR BASE-CASE TRANSIENT (NO HPI)

Time (s)	Event
0.0	LOSP trips turbines, RCPs main feedwater pumps and generates reactor trip signal
0.6	Control rods drop (1-s insertion time)
15	Turbine-driven pumps fail to deliver AFW
30	Motor-driven pumps fail to deliver AFW
95	ARVs on steam lines open
4170	SG secondaries empty of water
4200	PORV opens
5175	Pressurizer solid
5370	PRT rupture disks open
6460	High containment pressure (0.13 MPa)
6650	Primary-system hot legs saturate
6810	Pressurizer level begins to decrease
6920	Loss of natural circulation
7600	Upper plenum 90% vapor-filled
8300	Core heatup begins

until the decay power declined to a level where the boiling of the subcooled ECC water could provide the necessary cooling. This occurred before the core began uncovering as the primary system reached a quasi-equilibrium condition and the cladding temperature stabilized as shown in Fig. 67 (F1).

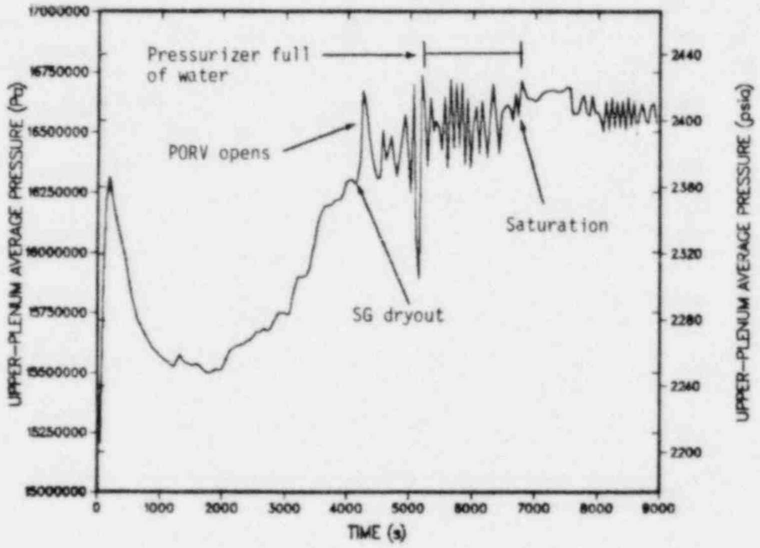

Fig. 62.
Primary pressure during the LOSP event for base transient.

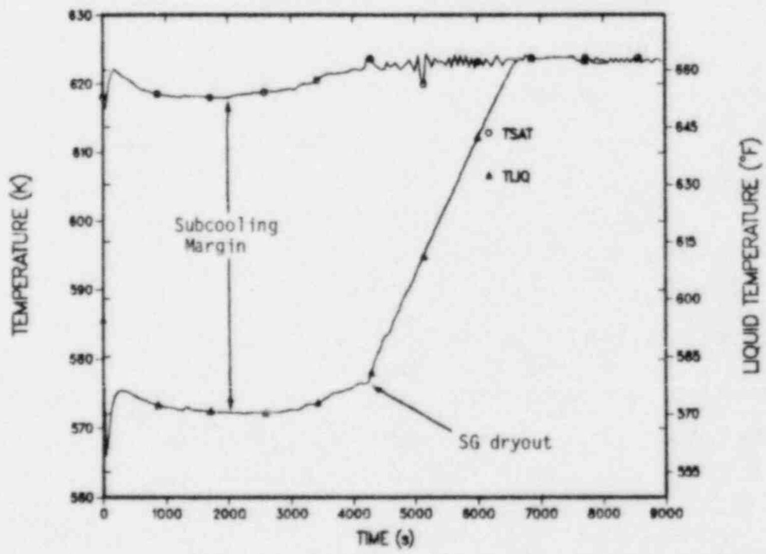

Fig. 63.
Primary hot-leg temperatures during the LOSP event for base transient.

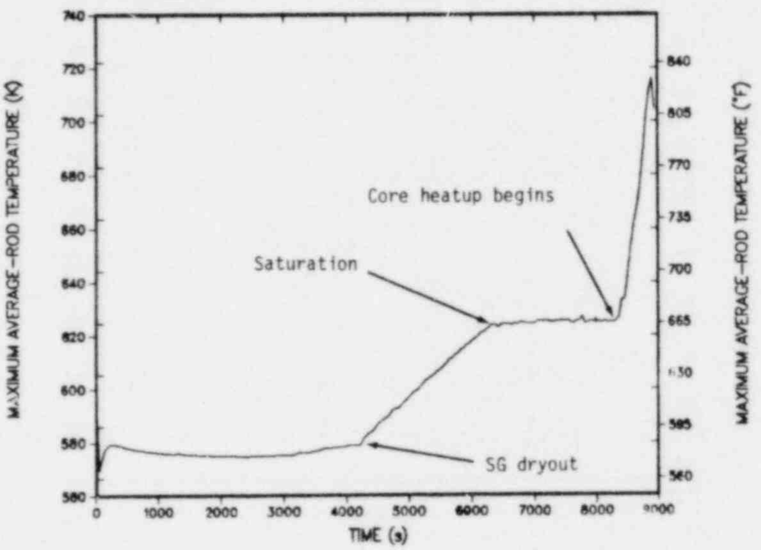

Fig. 64.
Cladding temperatures during the LOSP event for base transient.

c. Feed-and-Bleed Transient. The use of a feed-and-bleed procedure following a LOSP-induced LOFW transient was also reported in Ref. 18. The event sequence for this transient is presented in Table XVI. The PORVs were locked open at 7620 s. The primary pressure trace is presented in Fig. 68. Recovery begins shortly thereafter as indicated by the vessel filling (Fig. 69) and decreasing cladding temperature (Fig. 70). This transient shows that the Zion-1 SI system has sufficient capacity to recover core cooling late in the LOSP-induced LOFW transient (S1).

2. LOFW Event. The initiator for the base sequence is an LOFW event. There is a loss of all primary and auxiliary feedwater, and a loss of HPI. It was assumed that the reactor trip occurred at time zero for the majority of the calculations. However, the effect of a reactor trip, as specified by Westinghouse (Ref. 26), on event timing was also determined. The RCPs remain on throughout the transient whereas for the LOSP transient the RCPs are tripped at time zero. The effect of continued RCP operation is seen in the accelerated dryout of the steam-generator secondaries that was caused by the additional energy added to the system (15.84 MW) by the RCPs. Subsequent events follow the same pattern. The uncontrolled heatup of the core during the LOFW event occurs 1100 s earlier than in the LOSP event. The accident signatures for the LOFW and

TABLE XV

ZION-1
LOSP EVENT SEQUENCES FOR FEED ACTUATED BY
CONTAINMENT OVERPRESSURE SIGNAL

Time (s)	Event
0.0	LOSP trips turbines, reactor-cooling pumps, main feedwater pumps and generates reactor trip signal
0.6	Control rods drop (1-s insertion time)
15	Turbine-driven pumps fail to deliver AFW
30	Motor-driven pumps fail to deliver AFW
60	ARVs on steam lines open
3800	SG secondaries empty of water
4000	PORV opens (primary pressure = 16.1 MPa)
4800	Pressurizer solid; PRT rupture disk open
5800	ECC tripped on high containment pressure (0.13 MPa)
6800	Pressurizer level begins to decrease
7200	Primary coolant saturates, loss of natural circulation
7300	Cladding temperature reaches peak of 625 K
7600	Upper plenum 90% empty, top of core begins to uncover
7800	Recovery begins ($T_L < T_{sat}$), core 8% empty

LOSP transients are similar except for the time scale. A summary event chart for the Zion-1 LOFW calculations is presented in Fig. 71. Key reactor events and phenomena are shown and are described next.

a. Base Transient. The event sequence for the base case is presented in Table XVII. The primary-system pressure during an LOFW transient is shown in Fig. 72. The PORV open setpoint is 16.6 MPa (2408 psig), and the SRV setpoint is 17.2 MPa (2495 psig). The first opening of the PORV is at 3110 s, and first opening of the SRV is at 5140 s. With the RCPs running, the PORV relief rate following primary-system saturation is not sufficient to prevent a further increase in primary pressure to the SRV setpoint. For the LOSP transient, the

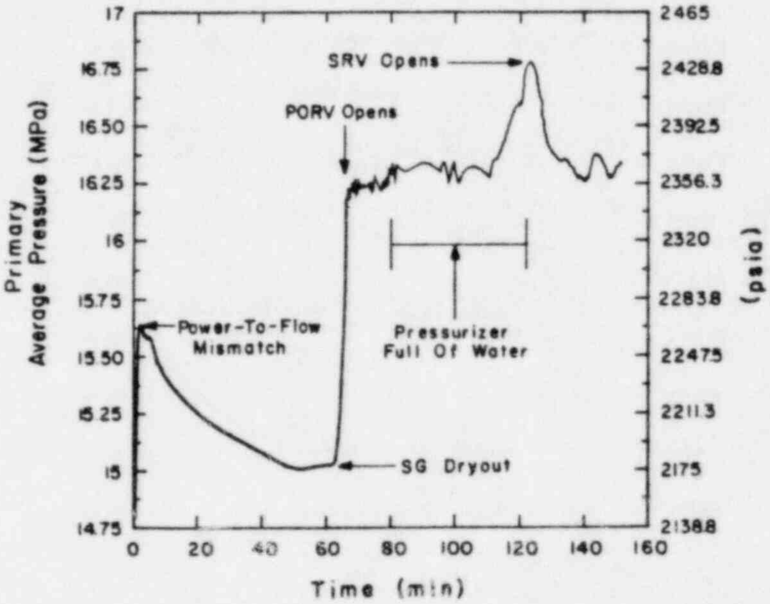

Fig. 65.
Primary pressure during the LOSP event for feed initiated on high containment overpressure.

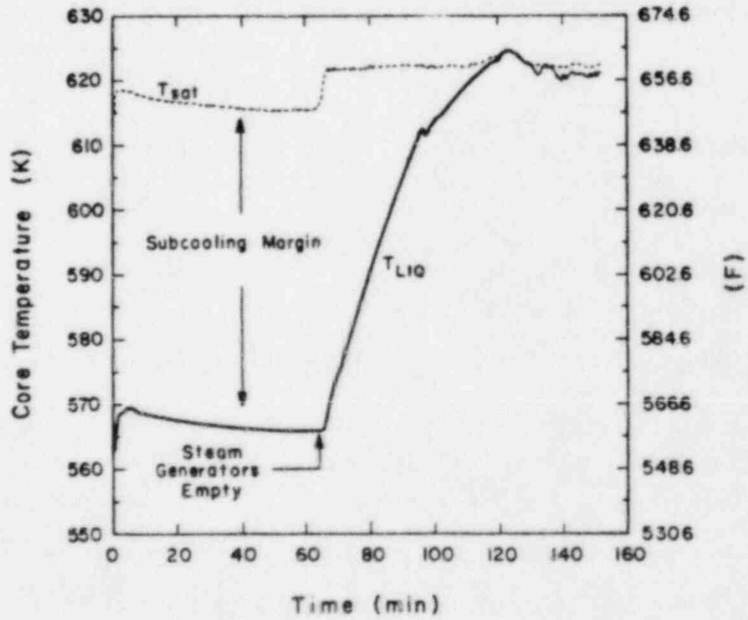

Fig. 66.
Primary-coolant temperature during the LOSP event with feed initiated on high containment overpressure.

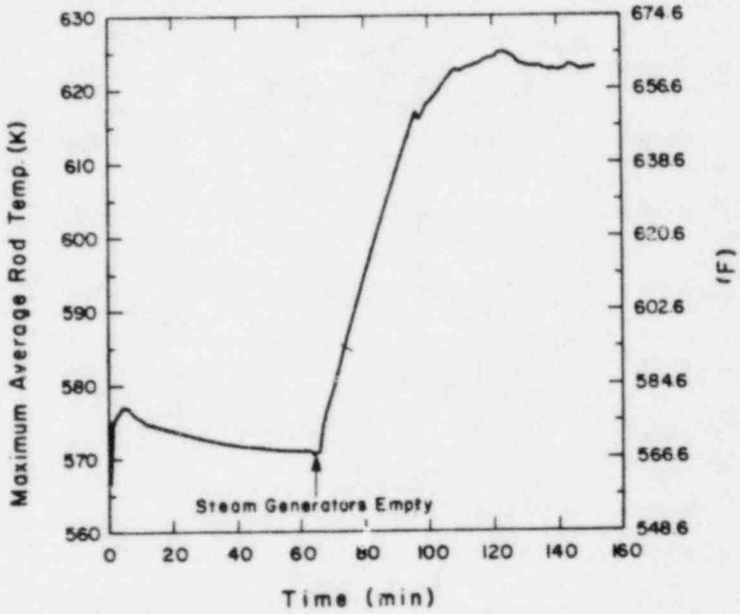

Fig. 67.
Cladding temperatures during the LOSP event for feed initiated on high containment overpressure.

TABLE XVI

ZION-1
LOSP EVENT SEQUENCE WITH FEED AND BLEED

Time (s)	Event
0	Loss of feedwater, main coolant pumps tripped.
0.6	Reactor trip.
3800	Steam generator secondary-side dryout.
4000	PORVs open.
5800	HPI initiated on high containment pressure.
7200	System saturates, loss of natural circulation.
7620	PORVs held open.
7800	Recovery begins, primary pressure dropping rapidly.

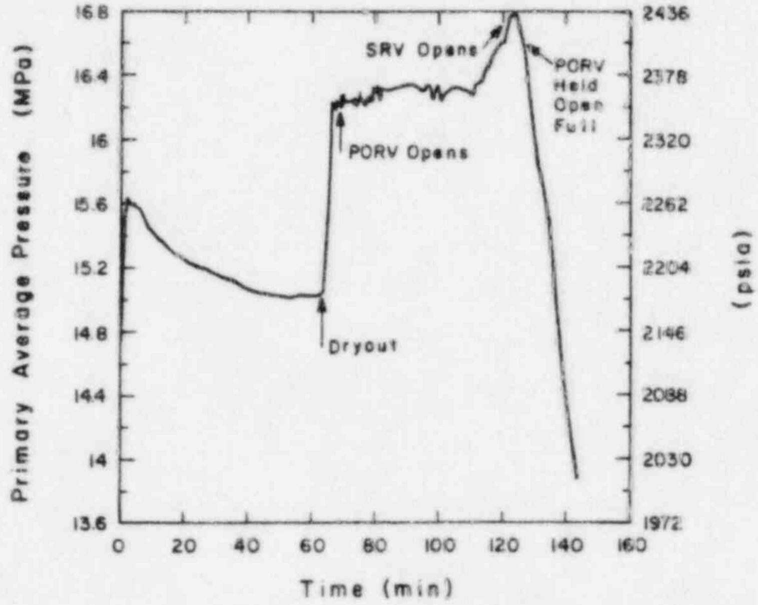

Fig. 68.
Primary pressure during the LOSP event for feed-and-bleed case.

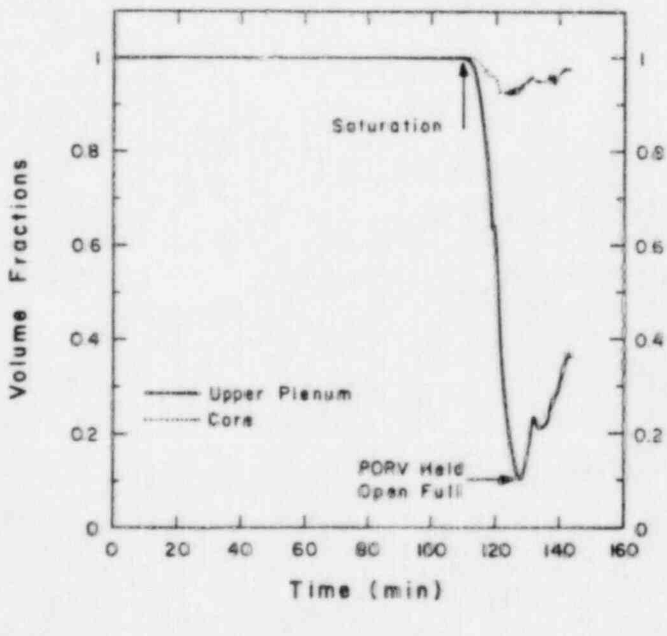

Fig. 69.
Core liquid volume during the LOSP for feed-and-bleed case.

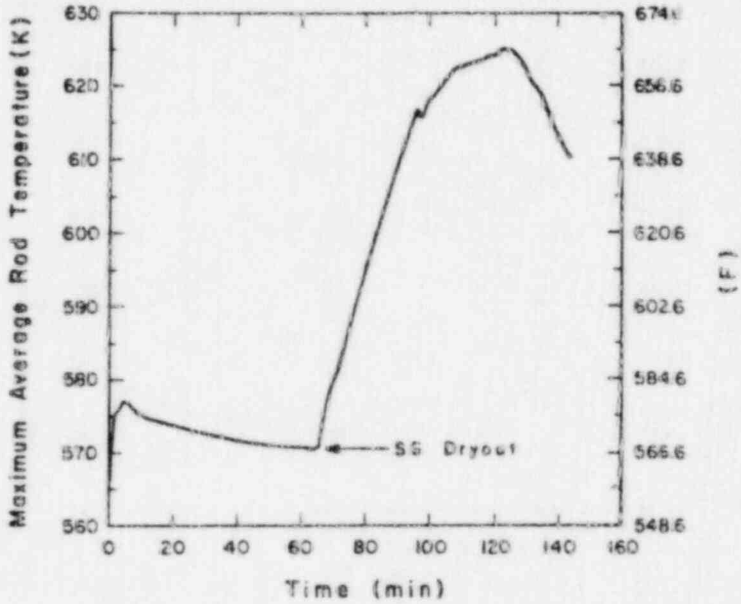

Fig. 70.
Cladding temperatures during the LOSP event for feed-and-bleed transient.

SRV pressure setpoint was not reached. The total integrated mass flow through the pressure relief train is shown in Fig. 73. The PORV opens at 3110 s, but because the primary-system mass loss is small, the pressurizer water level increases until the pressurizer is filled with liquid at 3840 s. A second and larger increase in primary-system mass loss occurs at about 4875 when the primary-system hot legs saturate (Fig. 74). The maximum cladding temperature is presented in Fig. 75. The cladding heats to near saturation during the period 3100 s to 4875 s as the subcooling decreases. The rapid core heatup begins at 6280 s (F1) as the core becomes completely filled with steam. We re-examined the base LOFW transient to determine the effect of a W-calculated trip time on event timing. We believe the trip will occur on a combined low steam-generator level and a steamline-feedwater flow mismatch signal. Several different trip times were calculated. Because we had insufficient information to decide which, if any, of the results were correct, we selected the Westinghouse[26] trip time (16 s) for comparison with our zero trip results. With the reactor at power for 16 s following LOFW, the steam-generator secondary dryout occurred earlier as expected. Using the W trip time, the steam generators dried out at 2450 s, whereas dryout for the trip at time zero occurred at 3080 s.

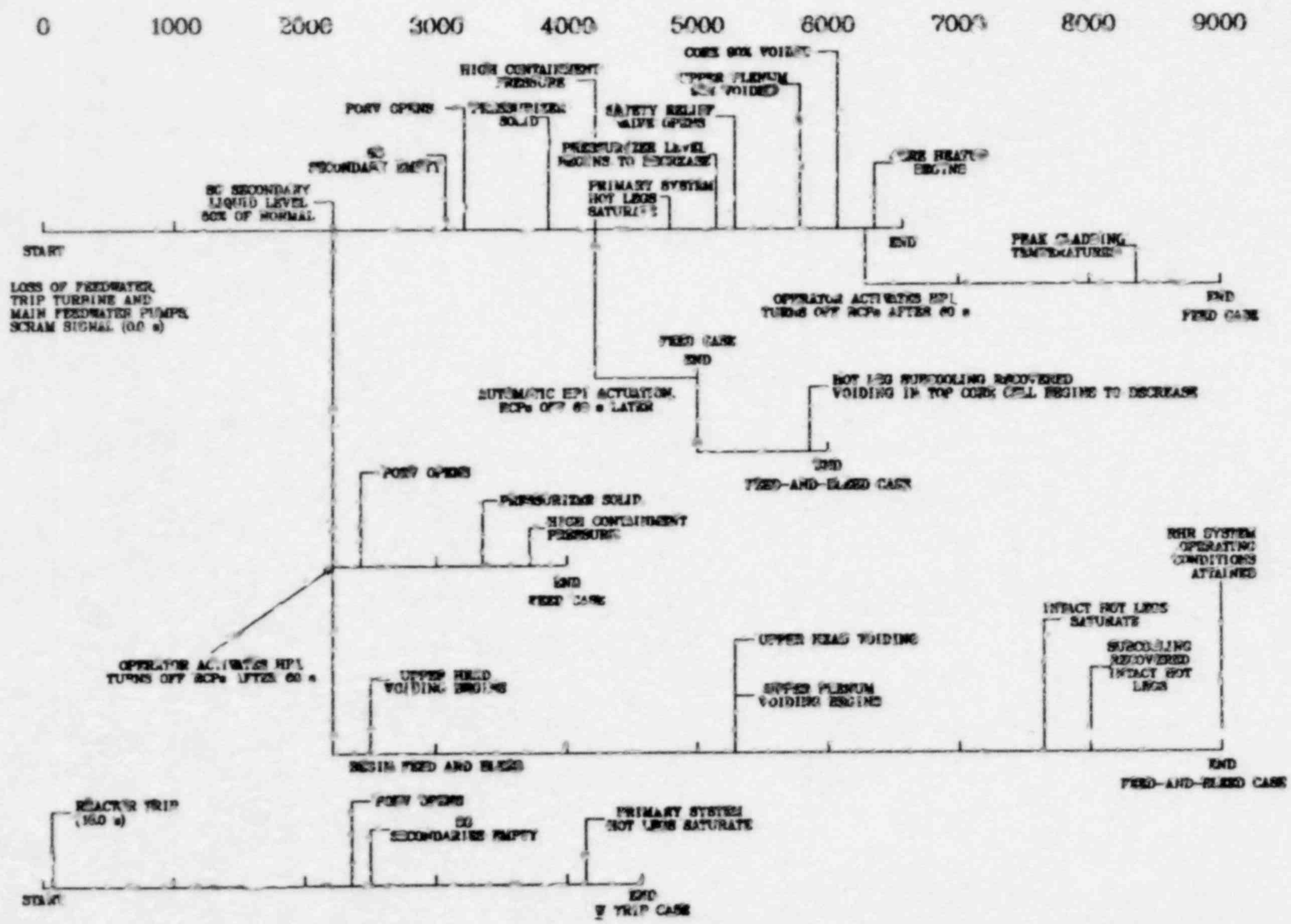

Fig. 71.
Zion-1 LOFW event line (nominal equipment).

TABLE XVII

ZION-1
LOFW EVENT FOR BASE-CASE TRANSIENT (NO HPI)

Time (s)	Event
0.0	Initiating event caused turbine trip main feed-water pump trip and trip signal generation
NA	RCPs trip
0.6	Control rods drop (1-s injection time)
15	Turbine-driven pumps fail to deliver AFW
30	Motor-driven pumps fail to deliver AFW
91	ARVs on steam lines open
3080	SG secondaries empty of water
3110	PORV opens
3840	Pressurizer solid
3910	Pressurizer relief tank (PRT) rupture disks open
4095	High containment pressure
4875	Primary-system hot legs saturate
5125	Pressurizer level begins to decrease
5140	SRVs open
5875	Upper plenum vapor volume fraction 90%
6070	Core vapor volume fraction 90%
6280	Core heatup begins

b. Feed Transients. We examined several feed transients. The first was automatic actuation of the HPI on generation of a containment overpressure signal at 4100 s. The RCPs were tripped 40 s later. The calculation was stopped at 5000 s because we believe that information from Ref. 18 can be used to establish the sequence of events and the end result of this transient. The appropriate Ref. 18 sequence is an LOSP with failure of the auxiliary feedwater

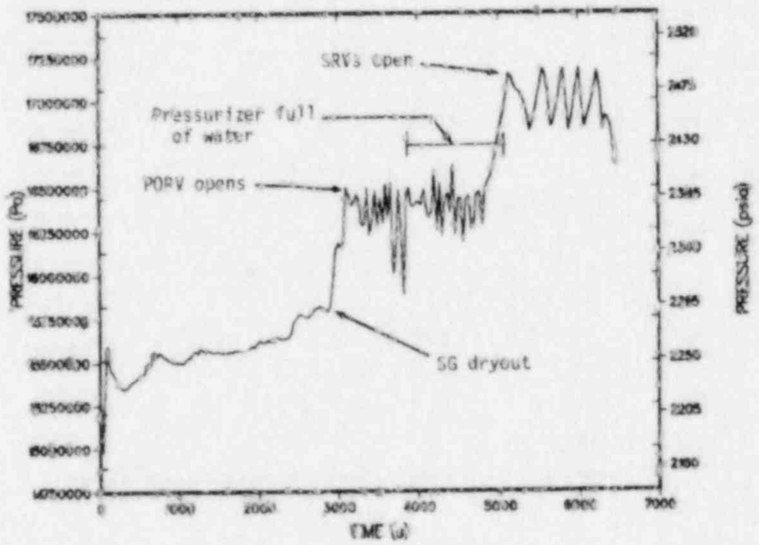

Fig. 72.
Primary pressure during the LOFW event for base transient.

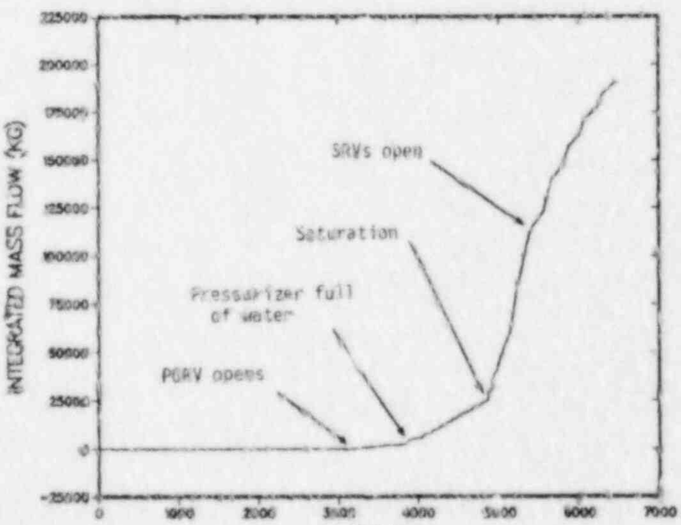

Fig. 73.
Pressurizer relief flow during LOFW event for base transient.

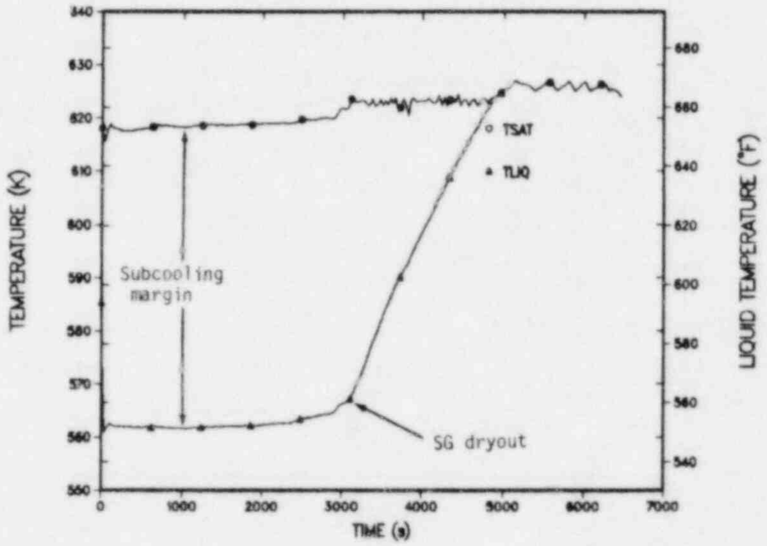

Fig. 74.
Hot-leg temperatures during the LOFW event for base transient.

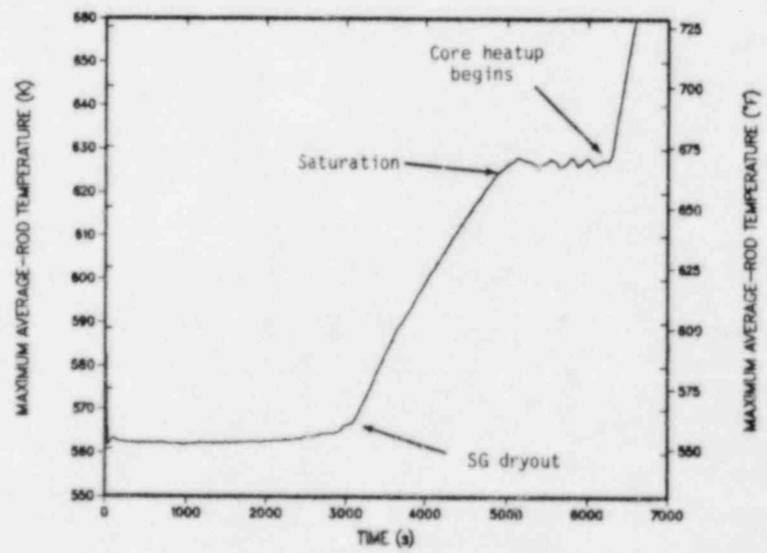

Fig. 75.
Cladding temperatures during the LOFW event for base-case transient.

and HPI actuation on containment overpressure. The containment overpressure signal for the LOSP transient occurs at 5800 s, or 1700 s later than the LOFW transient. The delay is the result of tripping the RCPs at the start of the transient, thereby reducing by 16 MW the energy input to the primary. The anticipated sequence includes primary-system saturation, loss of natural circulation, partial voiding of the core, refill of the core, and eventual recovery of subcooling.

Two transients were run to simulate options for operator intervention in the base LOFW transient using a feed-mode operation. The objective of the first study was to determine the latest time the operator could activate the HPI system and avoid core damage (defined as 1000 K (1341°F), the temperature at which cladding balloons). It was determined that if the operator intervenes before the start of rapid core heatup (time $\leq$ 6280 s), the cladding temperature remains below the damage limit. However, the first success criterion is not satisfied (F1). Figure 76 shows the maximum cladding temperature for the base transient and for operator actuation of the HPI at 6230 s. The RCPs were tripped 60 s after HPI actuation. The peak cladding temperature of 720 K (837°F) was reached at 8285 s, after which the core cooled to the system saturation temperature of 625 K (666°F). The net vessel mass flow is presented in Fig. 77. Subsequent to HPI initiation at 6230 s, the net vessel mass flow becomes positive as the lower plenum and core regions refill. The recovery of subcooling in the primary-system cold legs is shown in Fig. 78. Recovery of subcooling in the hot legs would follow vessel refill, but the calculation was terminated prior to this time.

The second operator-intervention study examined an early operator actuation of the HPI at about 2260 s; the RCPs were tripped 60 s later. The operator actions were assumed to occur at the time the steam-generator-secondary liquid level decreased to 50% of normal. Selection of the time was arbitrary, and the present operator guidelines do not instruct the operator to take such an early action. HPI injection into the liquid-full primary rapidly pressurizes the primary to the PORV setpoint. One consequence of earlier HPI initiation is a reduction in the rate of steam-generator-secondary inventory depletion. It is estimated the steam-generator dryout would occur at 4300 s, which is 1200 s later than the base case. Natural circulation is established in the liquid-full system after the RCPs are tripped. It is believed that the system will remain liquid full and that natural circulation will continue so long as HPI continues.

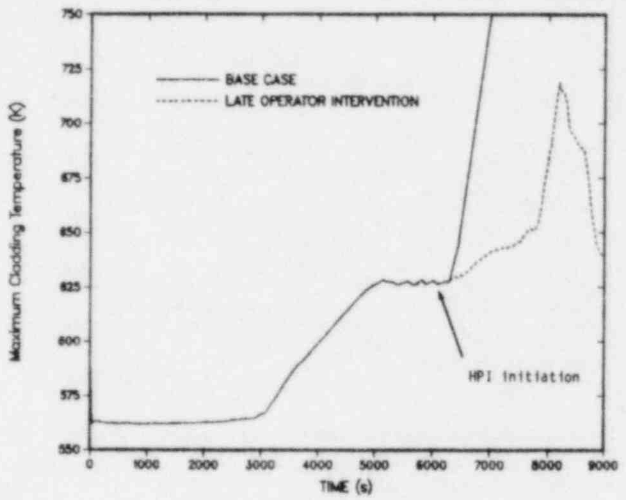

Fig. 76.
Cladding temperatures during the LOFW event for feed initiated at rapid core heatup.

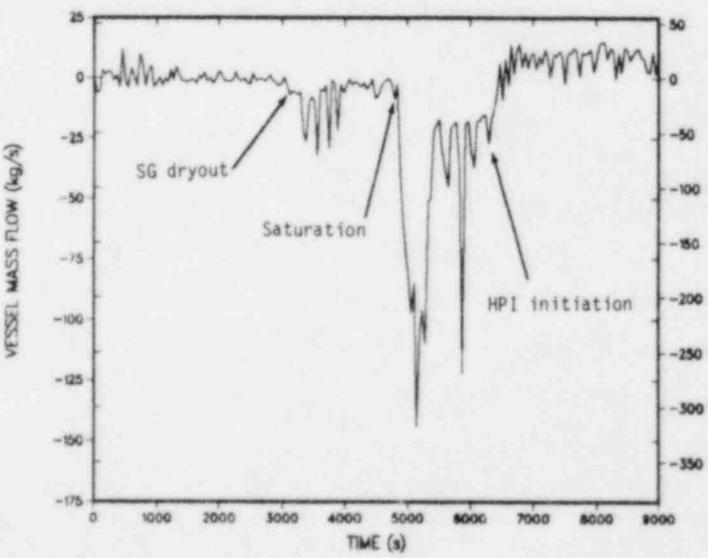

Fig. 77.
Net vessel mass flow during the LOFW event for feed initiated at rapid core heatup.

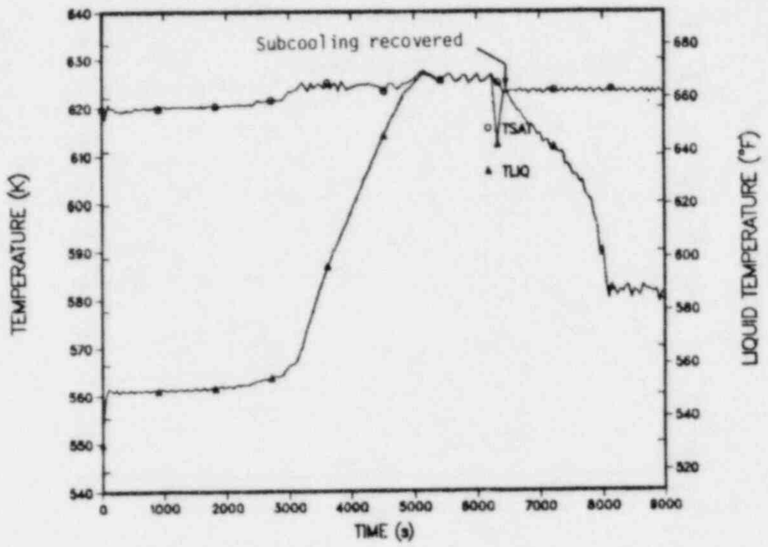

Fig. 78.
Cold-leg temperatures during the LOFW event for feed initiated at rapid core heatup.

The calculation was ended at 4000 s because the course of the remaining transient can be established using the event sequence for an LOFW with full ECC initiated at 10 min.[18] Following steam-generator-secondary dryout, the primary will slowly heat to the saturation temperature. At that time the HPI flow will be capable of removing nearly all the core decay power. Some primary-system boiling will occur until the core decay power decreases below the HPI heat-removal capability (15000 s) and the first success criterion (S1) is satisfied.

c. Feed-and-Bleed Transients. The use of a feed-and-bleed operation following an LOFW event has been examined for several cases. The feed-and-bleed procedure was initiated when the operator perceived that steam-generator-secondary cooling is or will be lost. We examined cases before, at, and after SGSD. The operator guidelines[27] specify that feed-and-bleed cooling be quickly established by starting HPI and verifying operation, establishing a bleed path by opening all PORVs, and stopping all RCPs. We studied three feed-and-bleed cases with nominal equipment availability. In the first case feed cooling is initiated at the time the containment overpressure signal is generated (4095 s). The PORVs cycle to maintain the system pressure at the PORVs setpoints, with the system liquid full and subcooled. However, the subcooling margin continues to decrease (Fig. 79), and at 5000 s the operator is assumed to initiate feed and bleed by locking open the PORVs. The primary pressure immediately decreases to the saturation pressure corresponding to the hottest liquid in the primary, as shown in Fig. 80. Boiling begins in the core and continues for about 550 s until subcooling is recovered at about 5600 s as shown in Fig. 79. Thus it can be seen that feed-and-bleed cooling is successful if initiated by the time the containment overpressure signal is generated (S1).

In the second feed-and-bleed case, we considered the initiation of feed and bleed at SGSD (3000 s). The operator was assumed to actuate the HPI at this time, tripping the RCPs 60 s later and opening the PORVs an additional 30 s later. As might be expected because feed and bleed was successful when initiated at a later time (primary system saturation), feed and bleed was also successful when initiated at SGSD. Throughout the transient, the cladding temperature stayed near or below saturation. The vessel liquid inventory reached a minimum near 5000 s and increased thereafter. Thus, for feed and bleed initiated with nominal equipment availability at SGSD, the success

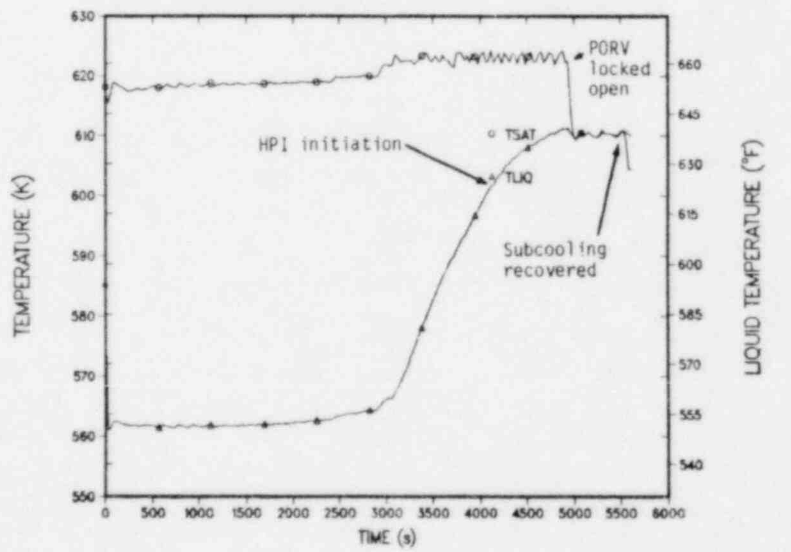

Fig. 79.
Hot-leg temperatures during the LOFW event for feed and bleed initiated at containment overpressure.

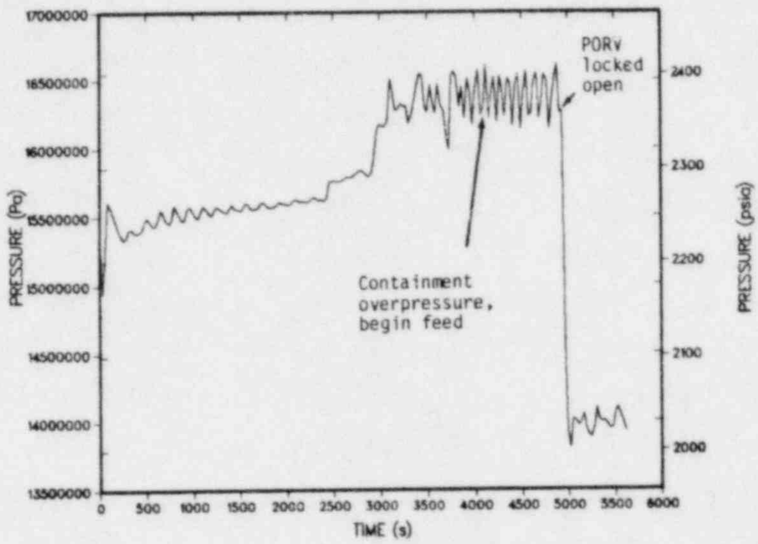

Fig. 80.
Primary pressure during the LOFW event for feed and bleed initiated at containment overpressure.

criterion for transition from reactor trip to a hot, holding condition is satisfied (S1).

In the third feed-and-bleed case we studied the early use of feed and bleed with the intent to cool and depressurize the reactor to RHR-system operating conditions for long-term cooling. This condition corresponds to a reactor-coolant system pressure and temperature of 3.04 MPa (440 psig) and 450 K (350°F), respectively.[17] The operator is assumed to initiate the feed-and-bleed procedure when the steam-generator-secondary liquid level drops to 50% of the normal steady-state value at about 2260 s. The primary-system pressure during the LOFW transient is shown in Fig. 81. The system pressure decreases rapidly after all PORVs are locked open and approaches the saturation pressure (8 MPa) of the hottest fluid in the primary. The upper-head region has very low flows and is only slightly cooled by the HPI flows. Therefore, primary-coolant saturation first occurs in the vessel upper head and is followed by a period of boiling that completely fills the upper head with vapor by 5200 s.

At 5200 s boiling begins in the upper level of the upper plenum. Vapor flows preferentially to loop B (see Fig. B.III.1). This flow asymmetry appears reasonable because the open PORV in loop B acts as a flow sink. The greater

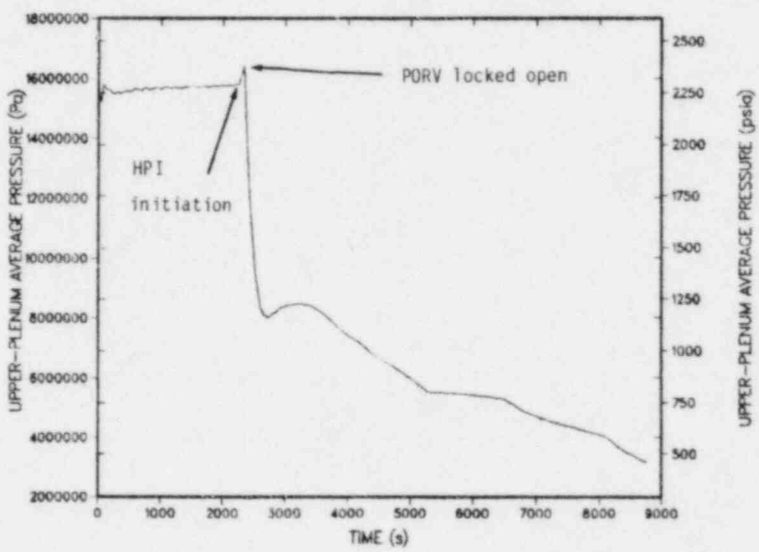

Fig. 81.
Primary pressure during the LOFW event for feed initiated at low (50%) level in the steam-generator secondary.

density difference across loop B increases the natural-circulation flow through the pressurizer loop (loop B) and starves the combined loops (loop ACD).

The loop-B and loop-ACD subcooling is shown in Figs. 82 and 83, respectively. Primary-coolant loop ACD hot-leg temperatures during the LOFW event for feed and bleed initiated at low (50%) level in the steam-generator secondary. After the PORVs are opened, the saturation temperature in loop B falls rapidly with system pressure. However, the HPI flow begins to reduce the loop-B hot-leg temperature, and saturation is not reached. A simple controller was added to the Zion-1 model to throttle HPI and prevent excessive subcooling [> 50 K (90°F)]. The loop-ACD subcooling (Fig. 83) displays the same behavior until 5200 s. The reduced natural circulation results in hot-leg saturation occurring at 7500 s. As the void fraction increases, the natural circulation flow through loop ACD increases and subcooling is re-established. At the end of the calculated transient, the primary system pressure is 3.0 MPa (435 psig) and the loop-B and loop-ACD hot-leg temperatures are 460 K (369°F) and 475 K (396°F) respectively. Thus, the RHR initiation pressure condition has been attained and

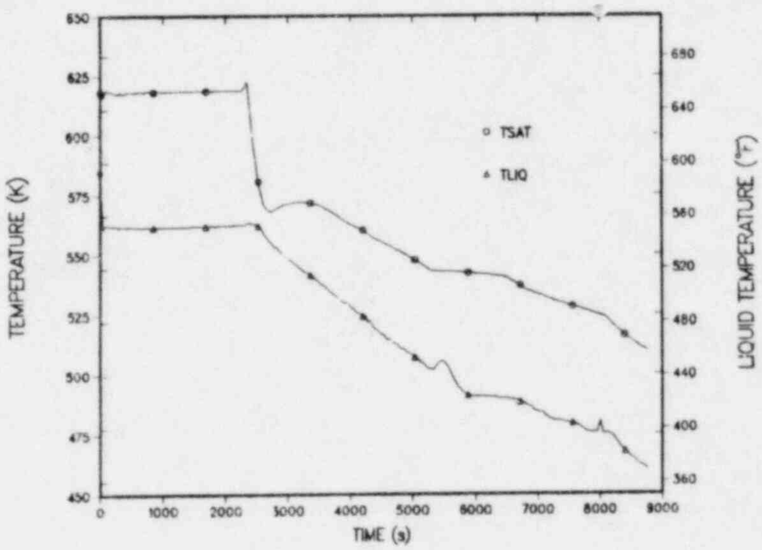

Fig. 82.
Pressurizer-coolant loop B hot-leg temperatures during the LOFW event for feed and bleed initiated at low (50%) level in the steam-generator secondary.

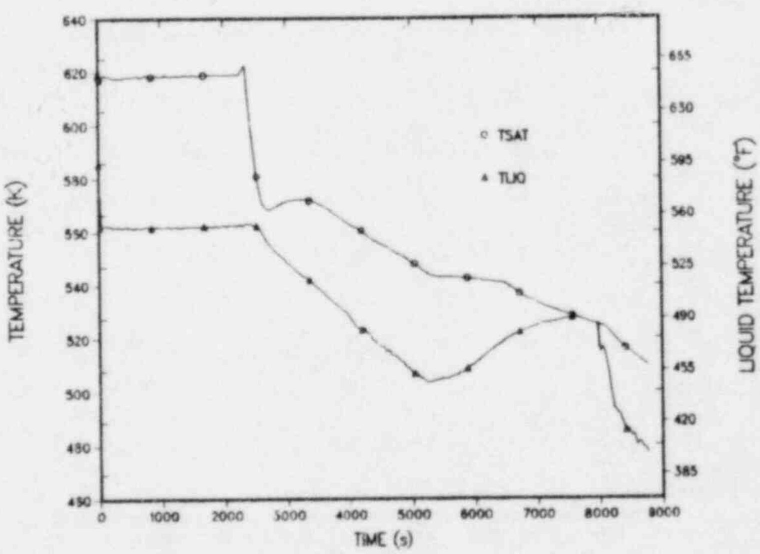

Fig. 83.
Primary-coolant loop ACD hot-leg temperatures during the LOWF event for feed and bleed initiated at low (50%) level in the steam-generator secondary.

a limited additional cooldown of 10-25 K (18-45°F) is needed to reach the temperature condition (S2).

The stated design bases for RHR system startup 3.04 MPa (441 psig) and 450 K (350°F) are predicated on the time after reactor shutdown being greater than 4 h. However, the shutdown transient has been calculated only to 2.5 h. It seems clear that the feed-and-bleed procedure will permit a successful transfer to the RHR system at 4 h if there is an adequate supply of water until that time. The capacity of the RWST from which HPI water is drawn is 1.325×10^6 kg (2.92×10^6 lb). At the end of the calculated transient (9000 s), about 430000 kg (948000 lb) has been drawn from the RWST. It is estimated that an additional 351000 kg (774000 lb) or a total of 781000 kg (1.72×10^6 lb) of water will be injected into the primary from the RWST to reach the 4-h RHR-system transfer time. This leaves at least a 2-h reserve of HPI water in the RWST.

The maximum average rod cladding temperature for the calculated transient is shown in Fig. 84. During the feed-and-bleed procedure, the core temperature cools continuously except for a brief temperature rise of about 10 K (18°F)

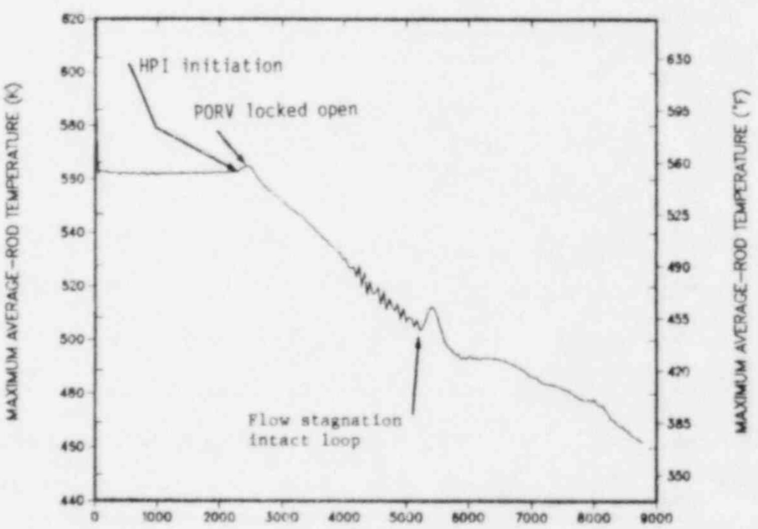

Fig. 84.
Cladding temperatures during the LOFW event for feed and bleed initiated at low (50%) level in the steam-generator secondary.

beginning at 5200 s. The temperature increase is caused by a temporary slowdown in the coolant flow through the core as the loop-flow switch (a decrease in the loop-ACD hot-leg flow and an increase in the loop-B hot-leg flow) occurs.

Two cases of degraded equipment availability have been studied for Zion-1. The cases reported are for feed and bleed initiated at SGSD with less than nominal equipment availability. The first case considered the availability of only one HPI pump; the nominal number of HPI pumps is two. There were several notable features of the transient relative to the nominal case of feed and bleed initiated at SGSD. Hot-leg saturation occurred shortly after initiation of the feed-and-bleed procedure. In the nominal case, the primary remained subcooled throughout the transient. The vessel liquid inventory began to increase late in the transient (about 6500 s) compared to the nominal case in which the vessel liquid inventory began to increase at 5000 s. However, there was a successful transition from reactor trip to a hot holding condition when feed and bleed was initiated at SGSD with only one HPI pump (S1). The second case considered the availability of only a single PORV; the nominal number of PORVs is two. The primary system remained subcooled throughout the transient with the minimum vessel liquid inventory attained at about 3500 s. Thus there was also a successful transition from reactor trip to a hot holding condition when feed and bleed was initiated at SGSD with only a single PORV available (S1).

3. Combined MSLB/LOFW Event. The base event sequence investigated is a combined MSLB/LOFW event. The main-steam-line break is located upstream of the main-steam-line stop-and-check valves and outside of containment. Only the affected steam generator blows down because the isolation valves on the intact steam generators prevent backflow to the break. The break size is 100% of the main-steam-line-pipe cross-sectional area. For the concurrent LOFW transient, the main turbine is not available for electric power generation, and there is a loss of all primary and auxiliary feedwater and a loss of HPI.

a. Base Transient. The event sequence for the base-case transient is summarized in Table XVIII. An engineered safeguards system (ESS) signal on steam-line pressure differential is generated immediately following the steam-line break at 0.2 s. After a 0.6-s delay, the reactor trip is initiated, main feedwater isolation begins, and the main-steam isolation valves begin to close. The AFW systems are assumed to fail. The broken-loop steam-generator blowdown is completed by 100 s. The PORV opens for a short time at 525 s and does not reopen until the primary-system pressure increases after the dryout of the

TABLE XVIII

ZION-1
EVENT SEQUENCE FOR COMBINED MSLB/LOFW EVENT BASE TRANSIENT

Time (s)	Event
0.0	Main-steam-line break
0.2	ESS signal on steam-line pressure differential
0.8	Reactor trip
0.8	Begin feedwater isolation
0.8	Main-steam-isolation valves begin to close
15	Turbine-driven pumps fail to deliver AFW
30	Motor-driven pumps fail to deliver AFW
100	Broken-loop SG blowdown completed
525	First PORV opening
540	ARVs relief valves on intact steam lines open
3120	Intact-loop steam-generator secondaries empty of water
3160	PORV reopens
3500	Calculation terminated

intact-loop steam generator. The ARVs on the intact steam-generator steam lines open at 540 s, and the secondaries of the intact steam generators empty of water by 3120 s. The transient calculation was terminated at 3500 s because the remaining course of the transient is nearly identical to the LOFW transient following dryout of the intact-loop steam generators.

The primary-system pressure during the combined MSLB/LOFW transient is shown in Fig. 85. The primary is overcooled during the first 100 s as the broken-loop steam generator blows down. Between 100 and 540 s the primary repressurizes because the isolated intact-loop steam generators are not capable of absorbing the decay heat. At about 525 s the PORV cycles open and shut over a 4-s period. The ARVs on the intact-loop steam generators open at about 540 s,

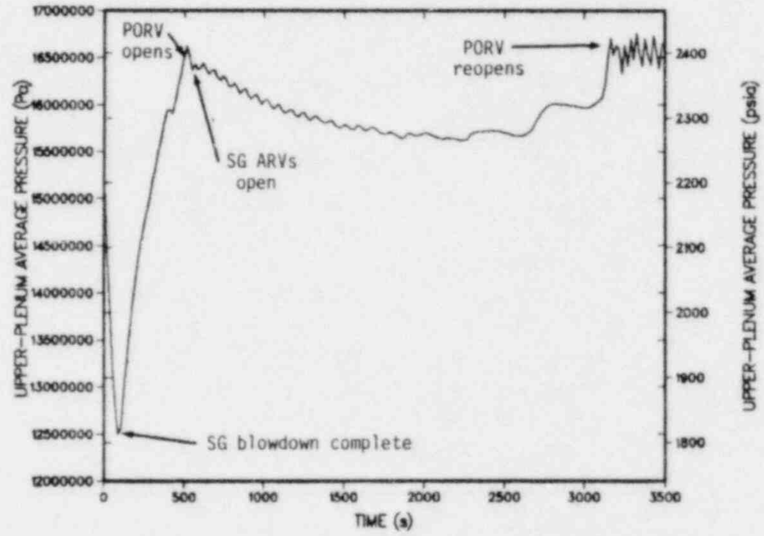

Fig. 85.
Primary pressure during the MSLB/LOFW event for base transient.

beginning a period of boiling the secondary inventory until dryout occurs at about 3120 s. While the steam-generator-secondary liquid inventory is being boiled off, all the reactor decay heat is rejected to the secondary and the primary begins a slow depressurization until about 2300 s. After 2300 s the heat transfer between the primary and secondary degrades as the secondary liquid inventory decreases and the primary again begins to repressurize to the PORV setpoint.

Intact steam-generator-secondary dryout for the combined MSLB/LOFW transient occurs about 3120 s. At this time, all four steam-generator-secondaries are dry (the broken loop by blowdown and the three intact loops by boiling), and heatup of the liquid-full primary is just beginning. For the simple LOFW transient, all four steam generators boil dry at 3080 s, and heatup of the liquid-full primary is just beginning. Thus at about 3100 s, the two transients are nearly identical even though there are significant differences in the transient prior to this time. It is concluded that the LOFW transient and parametric studies initiated subsequent to steam-generator-secondary dryout (3100 s) can be directly used for the combined MSLB/LOFW transient.

b. Feed Transient. An ESS signal is initiated if the steam-line pressure in any loop is about 0.7 MPa (100 psig) lower than that in the other loops. This signal also initiates a reactor trip and a sequence leading to HPI. A transient was run to examine automatic HPI actuation following an MSLB/LOFW accident. The HPI was initiated at 0.8 s and the RCPs were tripped 30 s later. The primary-system pressure for the parametric case is shown in Fig. 86. Again, the overcooling caused by the broken-loop steam-generator blowdown is evident. However, the HPI is initiated early and pressurizes the primary to the PORV setpoint by about 260 s. The sharp pressure drop at about 1200 s occurs because of rapid condensation as the last cell in the pressurizer fills with subcooled water. TRAC-PF1 appears to overpredict this condensation resulting in the sharp pressure drop. The primary-system temperatures are stable after 1250 s and continue to be stable until dryout of the intact-loop steam-generator secondaries. Because a large fraction of the reactor decay heat is being removed through the PORV, the boil-off of steam-generator-secondary inventory is slowed considerably compared to the base case. The hot-leg saturation and liquid temperatures are shown in Fig. 87. It can be seen that subcooling is

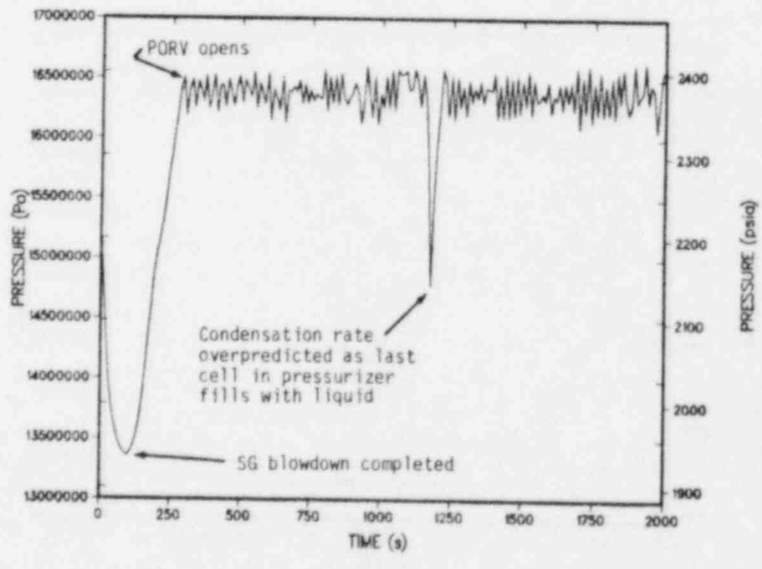

Fig. 86.
Primary pressure during the MSLB/LOFW event for feed initiated at 0.8 s.

being maintained and even increasing slightly. The feed mode of cooling is effective in maintaining the reactor in a stable state, and no change is expected until the intact-loop steam generators boil dry (estimated to be about 8000 s). After steam-generator dryout, a new long-term stable state with the system subcooled at the PORV setpoint pressure will be established.

c. Feed-and-Bleed Transient. We did not examine a feed-and-bleed operation for the MSLB/LOFW event. Because the feed mode was effective in cooling the reactor, we concluded that the feed-and-bleed mode would also be effective.

4. Combined SGTR/LOFW Event. The initiating event is the rupture of a single steam-generator tube that starts primary-system depressurization. A reactor-trip signal is generated when the primary-system pressure reaches the low-pressure setpoint. This in turn terminates the main-feedwater supply and closes the main-steam isolation valves. It is assumed that the AFW and HPI are unavailable for the base case. The RCPs continue to operate throughout this accident sequence.

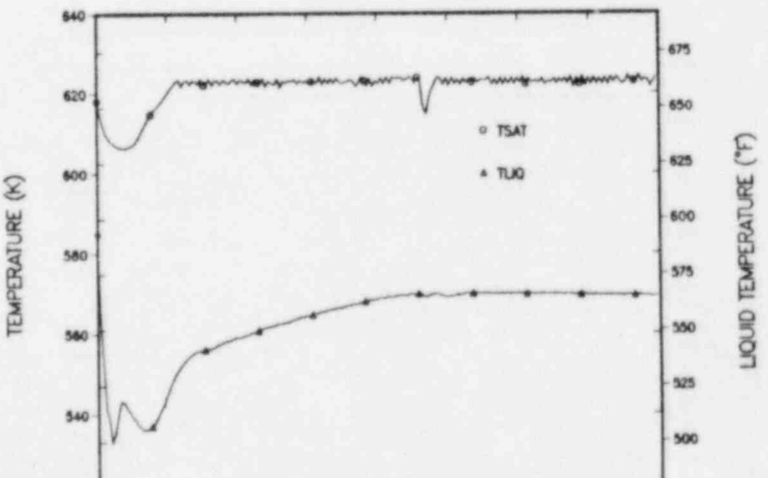

Fig. 87.
Hot-leg temperatures during the MSLB/LOFW event for feed initiated at 0.8 s.

TABLE XIX

ZION-1
EVENT SEQUENCE FOR STEAM-GENERATOR-TUBE-RUPTURE/LOFW EVENT
BASE TRANSIENT

Time (s)	Event
0.0	SG tube rupture
440	Pressurizer heaters uncovered
484	Reactor trip caused by low primary-system pressure, main-steam isolation valves close, feedwater pumps trip
549	Intact SG ARV first opens
554	Damaged SG ARV first opens
844	Pressurizer empty
3462	Pressurizer refill starts
5406	Primary-system hot legs saturate
5727	Intact SG secondary completely dry
5800	Core heatup begins
6300	Pressurizer heaters covered
7000	Intact ARV stops cycling
7500	Damaged-steam-generator secondary full of vapor

a. Base Transient. The event sequence for the base transients is given in Table XIX. Following the SGTR, the primary system begins to depressurize (Fig. 88). A reactor trip on low primary-system pressure occurs at 489 s. The main feedwater pumps trip, and there is a failure to provide AFW. The steam-generator secondaries dry out (5727 s and 7500 s for the intact and damaged steam generators, respectively), and the primary begins to heat up and expand. The saturation of the pressurizer leg is shown in Fig. 89. Because makeup water from the HPI system was not provided, the core would have eventually filled with vapor and an uncontrolled core heatup would have begun.

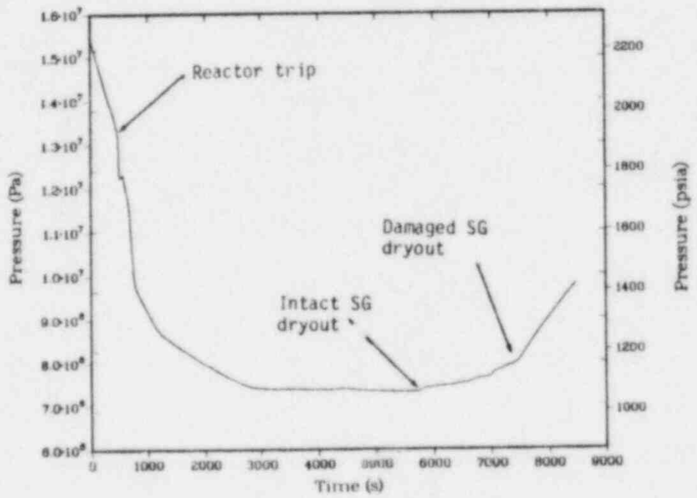

Fig. 88.
Primary pressure during SGTR/LOFW for base transient.

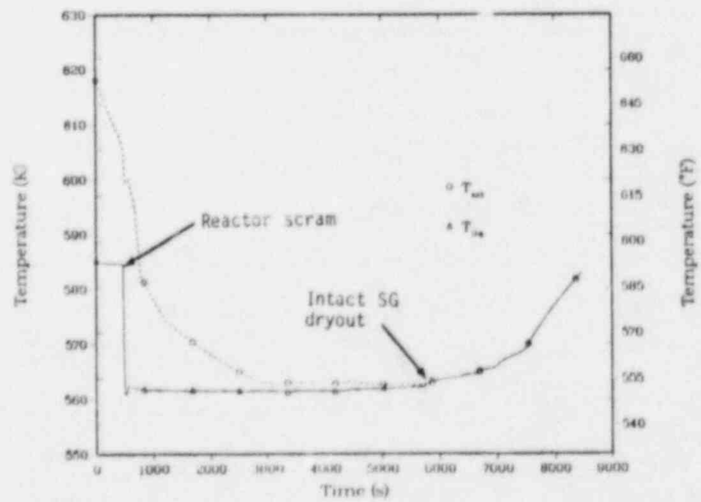

Fig. 89.
Pressurizer hot-leg temperature during SGTR/LOFW event for base transient.

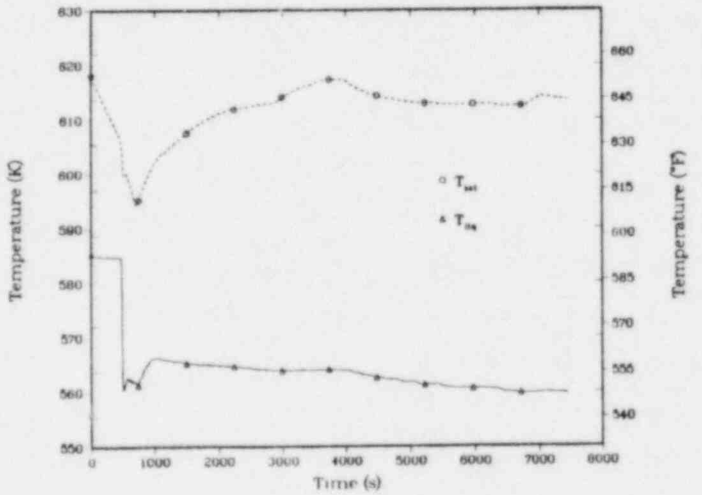

Fig. 90.
Hot-leg temperatures during SGTR/LOFW event for feed initiated at low primary pressure (11.72 MPa).

b. Feed Transient. We examined feed operation for the SGTR/LOFW transient with automatic actuation of the HPI on a low primary-system pressure of 11.72 MPa (1700 psi) at about 683 s. The operator turned the RCPs off 60 s after verifying the functioning of the HPI system. The 60-s delay time was arbitrarily chosen, as the delay time for the other feed transients was chosen to be 30 s. Natural circulation was established in the core and in both loops. The automatic HPI initiation proved to be sufficient for core cooling. The primary coolant maintained its subcooling as shown in Fig. 90. The HPI repressurized the primary system, which enhanced the mass flow out the ruptured tube. The primary-to-secondary mass flow was large enough to depressurize the primary; however, the HPI counteracted this by increasing flow, and ultimately an equilibrium pressure and mass flow level were reached. The PORV pressure setpoint was never reached. Hence, the only primary-coolant loss was through the damaged-steam-generator ARV. In the automatic response transient, core cooling was achieved by HPI flow only and the intact steam-generator secondaries did not dry out.

c. Feed-and-Bleed Transient. The major concern during the SGTR transient is the loss of primary coolant to the atmosphere and the consequential radiological releases. To limit the release, operator intervention is required to depressurize the primary system and stop the tube-rupture flow. In this calculation we studied a feed-and-bleed procedure for which the operator opened the PORV after verifying an SGTR, loss of main and auxiliary feedwater, and HPI actuation. This occurred approximately 360 s after reactor trip or 840 s after SGTR. The system rapidly depressurized to about 7 MPa (1015 psi), as illustrated in Fig. 91, and the pressurizer was immediately filled with liquid. The rapid depressurization caused an overcooling of the primary system that reduced the primary-to-secondary heat transfer in the intact steam generator to zero. The lower pressure resulted in saturation of the hot legs and the vessel. Between 1000 s and 2500 s boiling and voiding continued in the hot legs and the vessel. The core-heat removal was achieved primarily by the HPI flow. Therefore, the steam-generator secondaries did not dry out. The flow out the ruptured tube was reversed after the primary-side pressure decreased below the

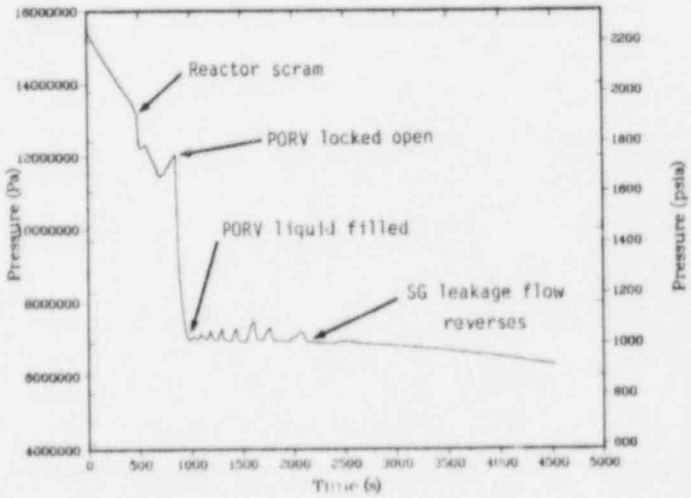

Fig. 91.
Primary pressure during SGTR/LOFW event for feed and bleed initiated at 850 s.

secondary-side pressure at about 2000 s. The feed-and-bleed procedure was effective in providing a stable cooling of the reactor.

5. Combined MFLB/LOFW Event. The event sequence for the base transient is summarized in Table XX. A reactor trip was assumed to occur at about 1.4 s. It is believed that during a real MFLB/LOFW transient the reactor trip would be caused by the steam-generator low-water-level mismatch tha occurs if the narrow-range steam-generator water level falls below the setpoint (25%) in coincidence with a concurrent steam-flow feed-flow mismatch. Because a detailed description of the Zion-1 narrow-range sensors was not available, the low-water-

TABLE XX

ZION-1
EVENT SEQUENCE FOR COMBINED MFLB/LOFW EVENT
BASE TRANSIENT

Time (s)	Event
0.0	Main-feed-line break
1.4	SG low-water-level mismatch signal (estimated)
2.0	Reactor trip
2.0	Begin feedwater isolation
2.0	Main-steam-isolation valves begin to close
15	Turbine-driven pumps fail to deliver AFW
30	Motor-driven pumps fail to deliver AFW
60	Broken-loop SG blowdown completed
174	ARVs on intact steam lines open
2910	Intact-loop SG secondaries empty of water
2962	PORV opens
3500	Calculation terminated

level trip was estimated after examining the early portion of the broken-loop steam-generator-inventory loss following the MFLB event.

a. Base Transient. After a 0.6-s delay, the reactor trip is initiated, main feedwater isolation begins, and the main-steam isolation valves begin to close. The AFW systems are assumed to fail. The broken-loop steam-generator blowdown is completed by 60 s. The ARVs on the intact steam-generator steam lines open at 177 s, and the secondaries of the intact steam generators empty of water by 2910 s. Following steam-generator-secondary dryout, the primary system begins to heat up and expand. The primary-system pressure increases to the PORV setpoint at 2962 s. The transient calculation was terminated at 3500 s because, following dryout of the intact-loop steam generator, the remaining course of the transient is nearly identical to that of the LOFW transient. The primary-system pressure during the combined MFLB/LOFW transient is shown in comparison with the MSLB/LOFW transient in Fig. 92. For the MFLB/LOFW transient, the primary is overcooled during the first 75 s as the broken-loop steam generator blows down. The overcooling is more pronounced for the combined MSLB/LOFW transient. The greater overcooling is caused by the manner in which the blowdown occurs. For the MSLB/LOFW transient, the steam-generator-secondary inventory flashes at the

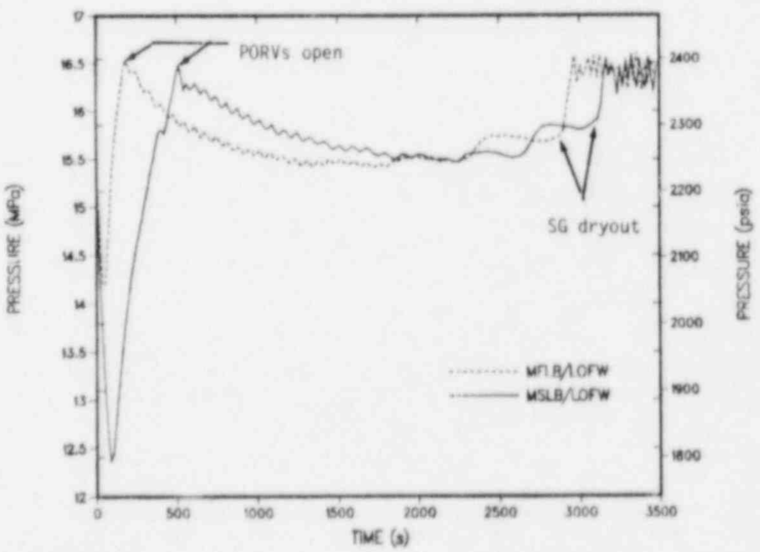

Fig. 92.
Primary-pressure comparison between MSLB/LOFW and MFLB/LOFW for base transients.

liquid-vapor interface in the tube-bundle region and then passes out the break. As the flashing process occurs within the steam generator, the energy required for the process is obtained from the primary. Thus, nearly the entire energy absorption capacity of the broken-loop steam generator is used for cooling the primary. This is not the case for the combined MFLB/LOFW transient. Although a fraction of the steam-generator-secondary inventory flashes at the liquid-vapor interface, the bulk of the liquid flows backward to the tubesheet, upward through the downcomer, and out the break, flashing as it exits without removing energy from the primary.

Intact-loop steam-generator-secondary dryout for the combined MFLB/LOFW transient occurs about 2910 s. At this time all four steam-generator-secondaries are dry (the broken loop by blowdown and the three intact loops by boiling), and heatup of the liquid-full primary is just beginning. For the simple LOFW transient, all four steam generators boil dry at 3080 s, and the heatup of the liquid-full primary is just beginning. Thus with a time shift at about 170 s added to the timing of events after steam-generator-secondary dryout for the MFLB/LOFW transient, the two transients are nearly identical even though there are significant differences in the transient prior to this time. It is concluded that the LOFW transient and parametric studies initiated subsequent to steam-generator-secondary dryout (2900 s) can be used with the indicated time shift for the combined MFLB/LOFW transient.

b. Feed and Feed-and-Bleed Transients. Because the combined MFLB/LOFW and MSLB/LOFW transients are similar, no transient was run to examine automatic high-pressure injection. An ESS trip leading to HPI actuation would have occurred at about 7.5 s following a high steamline-differential-pressure signal. This is close in timing to the automatic HPI actuation case examined for the MSLB/LOFW transient (HPI on at 0.8 s). For that transient it was concluded that the feed of HPI liquid to the primary (pressure at PORV setpoint of 16.5 MPa) is clearly effective in maintaining the reactor in a stable state and no change is expected until the intact-loop steam generators boil dry (estimated to be about 8000 s). The same conclusion is drawn for the MFLB/LOFW transient.

B. Summary Insights and Conclusions

As with the other plants examined, three plant features have the greatest effect on accident signature and on our insights and conclusions. These three features are steam-generator-secondary inventory, HPI delivery capacity, and PORV relief capacity. The total inventory of the Zion-1 steam-generator secondary is about 174000 kg (383000 lb). This is the largest inventory per MWt of the plants studied in detail. It is about 400% larger than the Oconee-1 reactor. For the base LOSP transient, steam-generator-secondary dryout occurs at about 4170 s. However, steam-generator-secondary dryout occurs much earlier for the LOFW event because the trip for this event follows the accident initiator by about one minute. During this interval the reactor continues at full power and much of the steam-generator secondary inventory is boiled off.

The W owners group Emergency Response Guidelines direct the operator to attempt to restore feedwater to the steam generators until primary-system pressurization and coolant heatup begin. Once the primary-system pressure and temperature begin to increase following loss of the secondary heat sink, the operator is directed to quickly establish once-through cooling using a feed-and-bleed procedure. This time can be as late as 4170 s (LOSP) or as early as 2400 s (LOFW using W trip time).

The Zion-1 HPI system delivers sufficient flow at both the PORV setpoint (feed mode) and lower pressures (feed and bleed) to permit successful control of loss-of-feedwater transients. We have studied feed-mode cooling initiated early in the transient (primary system liquid full and subcooled) and late in the transient (after primary-system saturation and voiding). We found that the first success criterion for feed and bleed was satisfied in the feed mode if feed was initiated by the time of primary-system saturation. We also found that the HPI-system delivery rate at the PORV setpoint was sufficiently large that core damage (defined as cladding temperature less than 1000 K, about the temperature at which the cladding begins to balloon) was prevented even if feed was initiated with primary system voiding in excess of 50%.

We have previously noted that a plant satisfying the first success criterion in the feed mode will also successfully satisfy the criterion using a feed-and-bleed procedure initiated at the same time. To again verify this insight we calculated several feed-and-bleed transients. For the LOFW transient we examined feed-and-bleed procedures initiated at the time the steam-generator-liquid levels dropped to 50% of normal and at the time a containment overpressure signal would be generated. These two times are about 1000 s before and 1000 s after the operator would begin to feed and bleed following the emergency response guidelines. In each case the primary is liquid-full and subcooled, although there is less subcooling when the procedure begins at the time of containment overpressure. In each case the first criterion for a successful feed-and-bleed operation is satisfied (S1). The first success criterion was also satisfied for feed and bleed initiated at SGSD with less than nominal equipment availability, a single HPI pump, and a single PORV.

The Zion-1 PORV relief capacity is sufficiently large that the primary pressure drops rapidly after the PORVs are latched open. The corresponding reduction in saturation pressure eliminates subcooling in the primary, and voiding occurs in the upper portion of the core when feed and bleed is initiated at the time of containment overpressure. However, the core remains cooled at all times during the procedure.

We examined the transition from reactor trip to hot shutdown from an LOFW event using a feed-and-bleed procedure initiated at the time the steam-generator-secondary liquid level decreased to 50% of normal. Unlike Oconee-1, the PORV relief capacity was sufficient to permit depressurization and cooldown to RHR-system design conditions using only the inventory of the RWST. Thus the second success criterion transition from reactor trip to cold shutdown is satisfied in the Zion-1 plant (S2).

Transients initiated by a break on the secondary side combined with an LOFW were also examined. The early accident signature of the combined MSLB/LOFW and MFLB/LOFW transients was not similar to that of the simple LOFW transient as primary-system overcooling occurs during the steam-generator blowdown. However, after loss of secondary heat sink, the combined transient is nearly the same as for the LOFW event. Event timing to loss of secondary heat sink is accelerated by about 200 s for the MSLB/LOFW event. The conclusions reached for the base transient also apply to the feed and the feed-and-bleed transients with some acceleration of the event timing.

We also examined a combined SGTR/LOFW event. Base, feed-mode, and feed-and-bleed transients were all calculated. With respect to the first feed-and-bleed criterion, we found that the plant could be depressurized and cooled successfully. In addition, the feed-and-bleed procedure terminated the tube-rupture flow, within 100 s of latching open the PORVs. Thus, the procedure offers the additional benefit of terminating the flow and the release of radionuclides to the environment.

We have reached the following conclusions about feed and bleed in the Zion-1 plant.

1. A feed-mode procedure can be used at the plant to cause the transition from reactor trip to a hot holding condition if initiated no later than the time of containment overpressure.

2. A feed-and-bleed procedure also produces the transition from reactor trip to a hot holding condition, even if the feed and bleed is initiated as late as the time of containment overpressure signal. However, some voiding will occur in the primary system because the primary rapidly depressurizes to saturation conditions. A transition to a hot holding condition can also be produced with degraded equipment availability if initiated at SGSD.

3. A feed-and-bleed procedure can also be used successfully to cause the plant transition to hot shutdown. There is sufficient HPI delivery and PORV relief capacity to cool and depressurize the plant using only the inventory of the RWST.

4. The early signatures of the combined MSLB/LOFW and MFLB/LOFW are dominated by overcooling of the primary. However, event timing prior to loss-of-secondary heat sink is only slightly accelerated (MSLB/LOFW) or mildly accelerated (MFLB/LOFW) as compared to the LOFW event. The early signature of the combined SGTR/LOFW is characterized by a primary-system depressurization. Feed and bleed is also effective for each of the combined transients.

VI. H. B. ROBINSON-2 INSIGHTS

H. B. Robinson[28] is a 2300-MWt three-loop W plant that is located near Hartsville, South Carolina, and is operated by Carolina Power and Light. The reactor-coolant system consists of the reactor vessel, three SGs, three RCPs, the pressurizer, and the piping connecting these components. The SGS are vertical shell and U-tube units. Steam separators are used to keep the moisture content below 1%. The RCPs are vertical, single-stage pumps. The pressurizer connects to one of the four primary loops. Electrical heaters are installed through the bottom head, whereas the spray nozzle, relief, and safety valve connections are located in the tophead of the pressurizer.

H. B. Robinson-2 is a low-pressure SI plant. The three SI pumps have a shutoff head of 10.1 MPa (1470 psia). The SI system also has two low-head RHR pumps. These pumps take suction from the RWST. The RHR pumps can be realigned to take suction from the containment sump after water has been expended from the RWST. The RHR system can be started when the system pressure is reduced below 1.03 MPa (150 psia) and the core decaypower is at or below the rated capacity of the RHR heat exchangers.

Three non-safety grade centrifual charging pumps are capable of delivering 11.3 kg/s (25.8 lb/s) at the PORV setpoint.

A schematic of the reactor coolant system is presented in Fig. 93. The TRAC-PF1 input model of the H. B. Robinson plant is described in Appendix B.IV. For the single H. B. Robinson-2 calculation performed, we assumed the nominal equipment included two SI pumps and two PORVs. We used the 1973 American Nuclear Society decay-heat curve.

A. Summary Results

As reported in Section V, we have performed an extensive set of calculations for a W four-loop plant that has high-pressure SI capability, Zion-1. We found that results for three-and two-loop W plants with low- or intermediate-pressure SI systems were limited. To determine if feed and bleed can be successfully applied to such plants, we calculated a single transient for the H. B. Robinson-2 plant.[29] We assumed no high-pressure centrifugal (charging) flow for the entire transient. This was done to assist us by providing information to make the simple-inspection statements reported in Sec. VII. The transient calculation was terminated at 4970 s when it was evident that cooldown to RHR conditions could be achieved (S2). The transient selected was a LOFW event with feed and bleed initiated at SGSD. The event sequence for this transient is shown in Fig. 94.

We assumed loss of both main and auxiliary feedwater at 1 s. The reactor tripped at 52 s on a 15% low-level signal in the steam generator. This is not the initial trip signal, and so a trip on 15% low-level signal represents a delayed trip. Steam generator dryout occurred at 840 s, and the feed-and-bleed procedures were initiated at 960 s; these consisted of opening the PORVs, tripping the RCPs, and starting the SI pumps. The primary pressure (Fig. 95) quickly decreased from the normal operating pressure to 8.2 MPa (1175 psia), and then slowly increased from decay-heat addition to about 11.6 MPa (1680 psia) at 2250 s. At 2250 s, the upper plenum and hot legs voided sufficiently for the

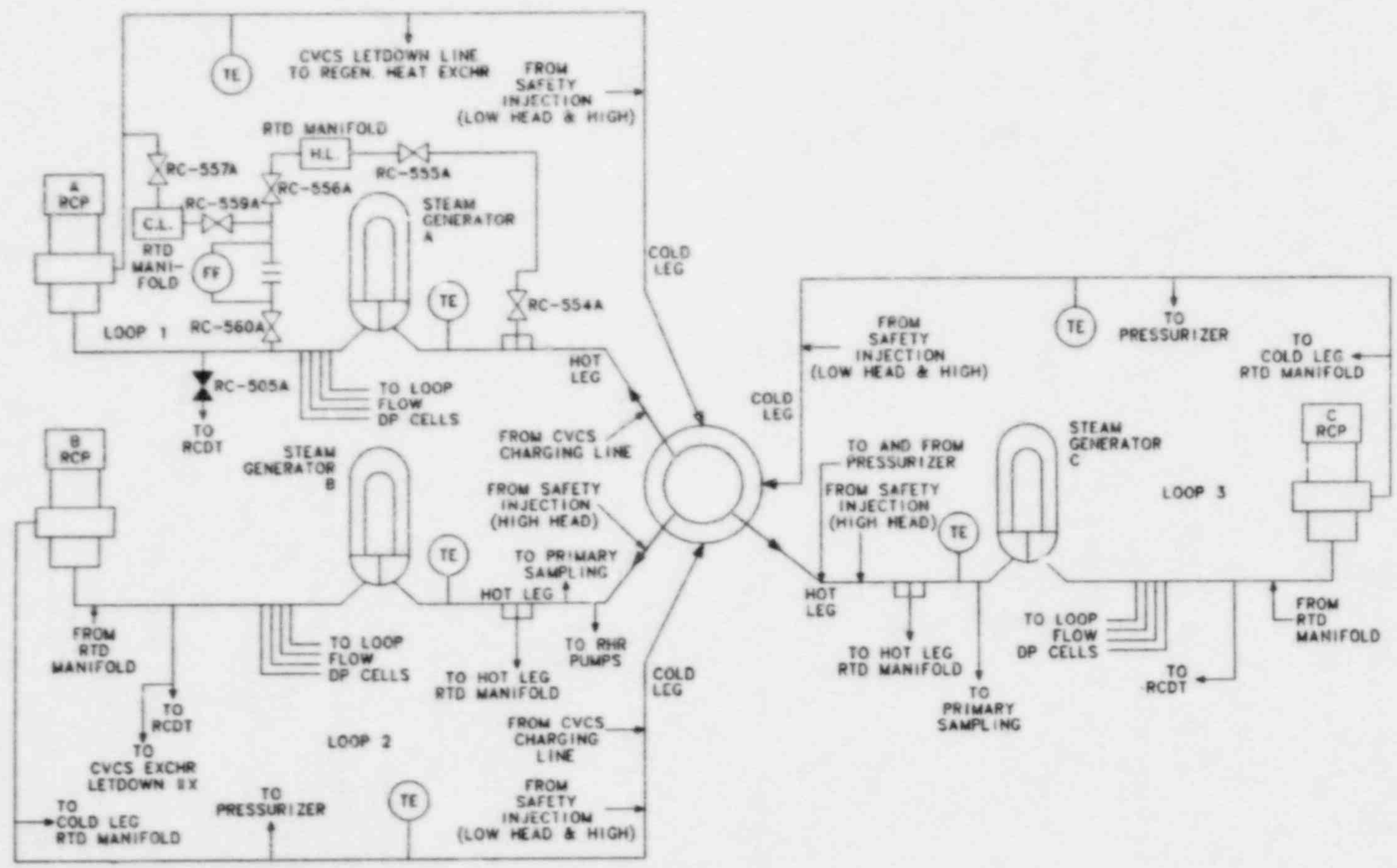

Fig. 93.
H. B. Robinson-2 reactor-cooled system.

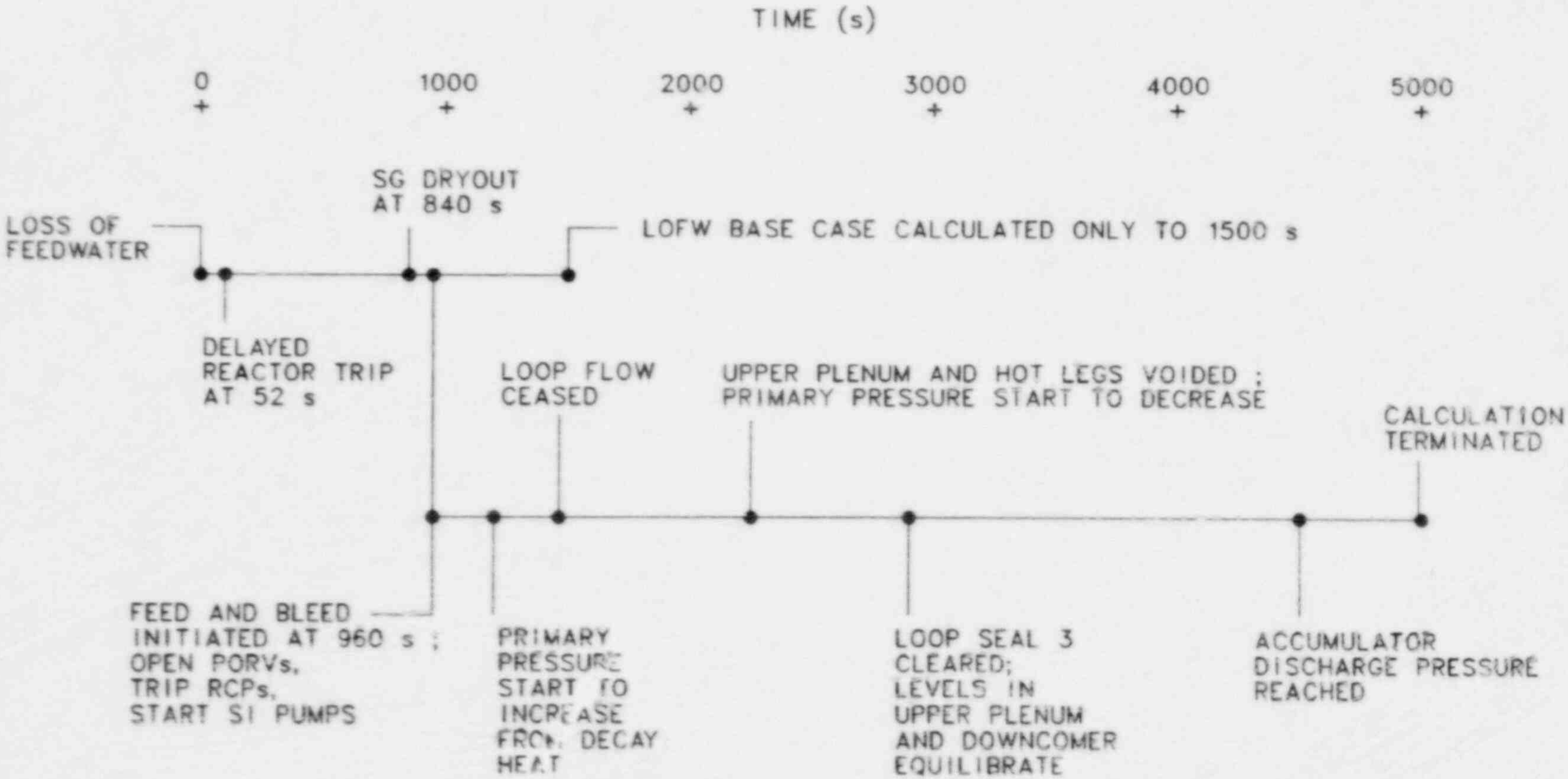

Fig. 94.
Transient event sequence for feed and bleed initiated at SGSD.

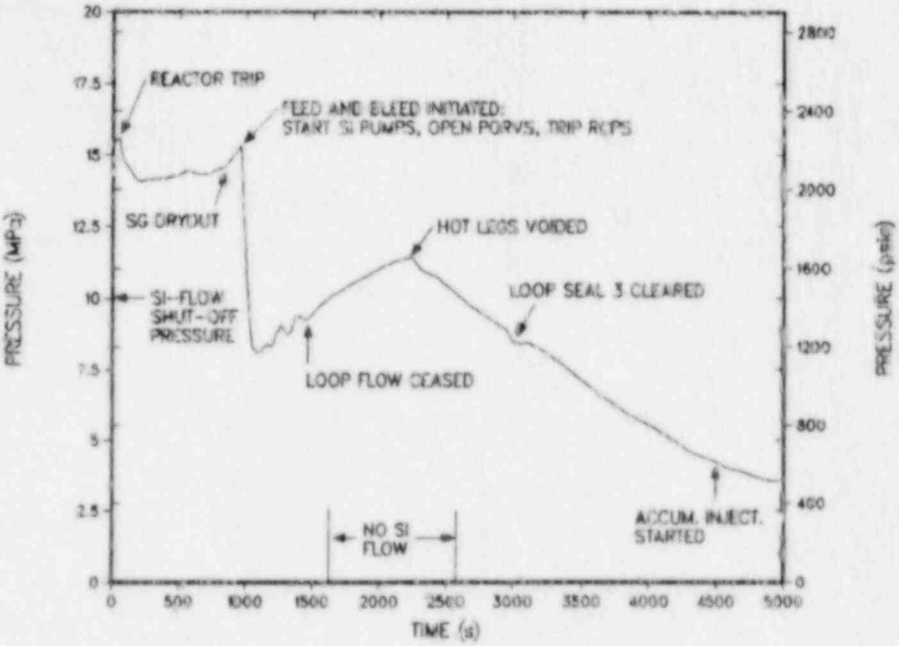

Fig. 95.
Primary pressure during the LOFW event with feed and bleed initiated at SGSD.

upper plenum vapor to escape through the PORVs. This high volumetric vapor flow provided pressure relief, and the primary pressure decreased thereafter, reaching the accumulator discharge pressure at about 4500 s. The SI flow did not start until 1030 s because of the low-head SI and stopped when the primary pressure increased at about the SI pump shutoff pressure of 10.2 MPa (1476 psia) betwen 1650 and 2580 s. After 2580 s, the pressure stayed below the SI shutoff pressure, and the SI flow continued uninterrupted thereafter. An extrapolation of the primary pressure indicates that the RHR operating pressure of 1.03 MPa (150 psia) would be reached at between 6000 and 7000 s.

The hot-leg coolant temperatures remained at or near saturation after the RCP trip. Subcooling in the legs was not maintained because of the low-head SI pumps, and it is likely that the hot legs would remain saturated until RHR conditions are reached. Cold-leg coolant remained subcooled because of the very low cold-leg flows and the resulting accumulation of low-temperature SI flow. The minimum vessel liquid inventory was reached at 3500 s and increased thereafter. Fuel cladding temperatures stayed near or below saturation with no indication of rod heatup as shown in Fig. 96.

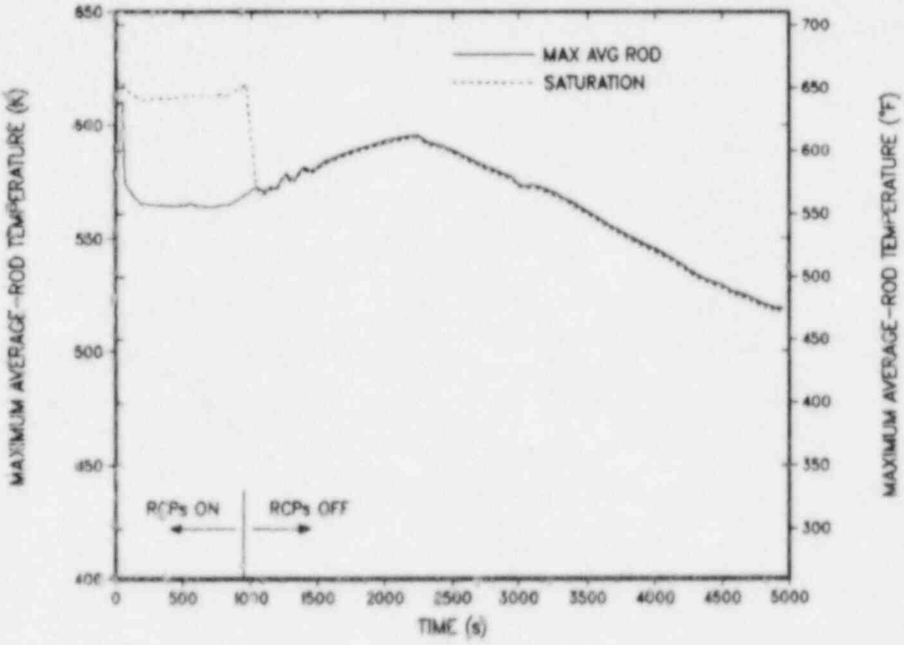

Fig. 96.
Cladding temperature during a LOFW event with feed and bleed initiated at SGSD.

We conclude that a complete LOFW with a delayed reactor trip can be cooled to RHR pressures with primary-side feed and bleed provided it is initiated before or at the time of SGSD. However, we found that any delay between LOFW and the reactor trip significantly hastened SGSD and reduced the time that operators have to detect and diagnose an accident and take appropriate action.

B. Summary Insights and Conclusions

Because only a single transient calculation was performed, feed and bleed initiated at SGSD, the summary insights are of necessity limited. However, we concluded that three-loop plants with either intermediate- or low-pressure SI capability could be cooled to RHR-entry conditions using feed and bleed (52). We assumed the charging pumps that deliver a limited coolant flow (11.3 kg/s, 25.8 lb/s) were inactive. Thus, the performance of the plant should be improved with operation of the charging pumps. Based on the similarity between the performance of Calvert Cliffs-2 and H. B. Robinson during a feed-and-bleed procedure, we believe that H. B. Robinson could not feed and bleed successfully as late as the time of primary system saturation. However, no calculation was done to verify this belief and so we reach no conclusions regarding feed and bleed initiated at primary-system saturation.

VII. COMPARATIVE EVALUATION

During the course of our extensive LOFW studies for the Oconee-1, Calvert Cliffs-1, Zion-1, and H. B. Robinson-2 reactors, we have determined that a limited number of plant features are most important in defining accident signatures and the outcome of recovery techniques. The recovery techniques examined were feed only (the ECC systems inject coolant at the PORV setpoint) and feed and bleed (the PORV is locked open and the ECC systems inject coolant at an increased rate because system pressure is decreased). The primary plant feature determining event timing for the primary heatup rate is the reactor trip time and the total steam-generator-secondary inventory. The primary plant features determining the timing and success of feed and feed-and-bleed cooling operations are the PORV capacity and the ECC system flow characteristics. These significant plant parameters are tabulated in Table XXI for the plants studied in detail.

Comparing the base-case transients, Oconee had the shortest heatup time, which was two to three times faster than that of Calvert Cliffs or Zion. As expected, the base transients showed that the LOSP event had a significantly

TABLE XXI

PLANT SIZING

	Oconee-1	Calvert Cliffs-1	Zion-1	H. B. Robinson-2
Steady-state power (MWt)	2584	2700	3250	2300
Total SG secondary inventory (kg)/(lb)	35000 77093	124700 274670	173840 382907	126906 279529
Number of PORVs	One	Two	Two	Two
Total rated PORV capacity (kg/s)(lb/s)	12.8 26.2	38.7 85.2	53.0 116.7	53.0 116.7
Total ECC flow (kg/s)/(lb/s) at PORV setpoint	27.2 59.9	8.3 18.3	15.6 34.4	11.3 25.8

slower heatup time than the LOFW event. In the LOFW event, the reactor does not trip immediately and the reactor coolant pumps continue to operate. These two effects increase heatup rates. In all base transients, the MSLB and the MFLB resulted in an initial primary overcooling. This resulted in a positive reactivity insertion and pressurizer liquid level decrease in all three plants. At Calvert Cliffs, the primary-system-pressure drop was sufficient to actuate the SI system.

Oconee-1, Calvert Cliffs-1, and Zion-1 were successful in the feed mode in stabilizing reactor conditions; however, the timing window of success was different. For both Oconee-1 and Zion-1, the feed mode was successful (S1) if initiated as late as primary system saturation. The high-head flow-injection capability of the SI systems in these plants makes this possible. However, for Calvert Cliffs, with its high-head low-flow SI capability, feed cooling must be established by SGSD. Although this requires that feed be initiated earlier (at steam-generator dryout vs at loss of core subcooling), the actual time delay is at least 20 min from the start of the initiating transient (LOSP, LOFW, MSLB, MFLB). This is at about the same time that the loss of core subcooling occurs in Oconee-1 for the LOFW event. For Zion-1, with the greater steam-generator inventory, the loss of core subcooling would occur after 30 min for the LOFW event. This is based on a consideration of delayed reactor trip times of up to a minute from the initial loss of feedwater.

The feed-and-bleed mode can be used successfully in all instances in which the feed mode has been determined to be successful: for Oconee-1 and Zion-1, up to the time of primary system saturation; for Calvert Cliffs-1 and H. B. Robinsion-2, up to the time of SGSD. Initiating feed or feed and bleed at or prior to these times will assure meeting success criterion S1. If feed is initiated later than the above times, significant core voiding may have occurred already, with the possibility of fuel/cladding damage. If feed is initiated later than the above times, feed and bleed should not be initiated until core subcooling has been reestablished. Locking open the PORV under partially voided core conditions could hinder core water-level recovery.

VIII. EXTENSION OF PLANT-SPECIFIC INSIGHTS

Detailed thermal-hydraulic studies have been performed for at least one plant of each US PWR vendor to determine if feed and bleed is a viable procedure for cooling a reactor following a complete LOFW initiator. Although the viabilility of the feed-and-bleed operation has been determined for four specific plants, the NRC is desirous of identifying all plants for which feed and bleed can be applied successfully. Clearly, this is an ambitious undertaking. At least four approaches have been identified for meeting this objective. In order of increasing cost and effort, they are (1) simple inspection, (2) enhanced inspection, (3) use of simplified plant-specific models, and (4) use of detailed models for each plant.

The first approach, simple inspection, applies to plants having characteristics similar to those for which detailed studies have been performed. Insights from the detailed studies are heavily weighted in the inspection process. Similar plants are assumed to perform in the same manner as the plants for which detailed calculations have been performed. Those plants judged too dissimilar are excluded from the process, and no extension statements are made for those plants. This procedure is not difficult and requires little additional effort to complete. Of the four approaches, we have least confidence in this one.

The second procedure we call enhanced inspection. It contains all the elements of simple inspection but includes limited plant-specific calculations. The inspection process is enhanced by constructing plant-specific feed-and-bleed operating maps.[10] Such maps are convenient tools for displaying mass and energy balances. The concept of the operating map can be explained by reference to an idealized map (Fig. 97). The feed-and-bleed success region is defined by two pressure boundaries. The lower pressure bound is identified by the intersection of core-decay power input and PORV energy removal. This bound is the lowest pressure at which the PORV heat removal balances the core-decay heat input. Because only a single core power is embodied in the map, a steady-state snapshot of the mass and energy balances is implied. The upper pressure bound is the highest pressure at which the injected coolant can completely compensate for the PORV outflow. A feed-and-bleed success region exists where both the mass-balance and the energy-balance relationships are mutually satisfied.

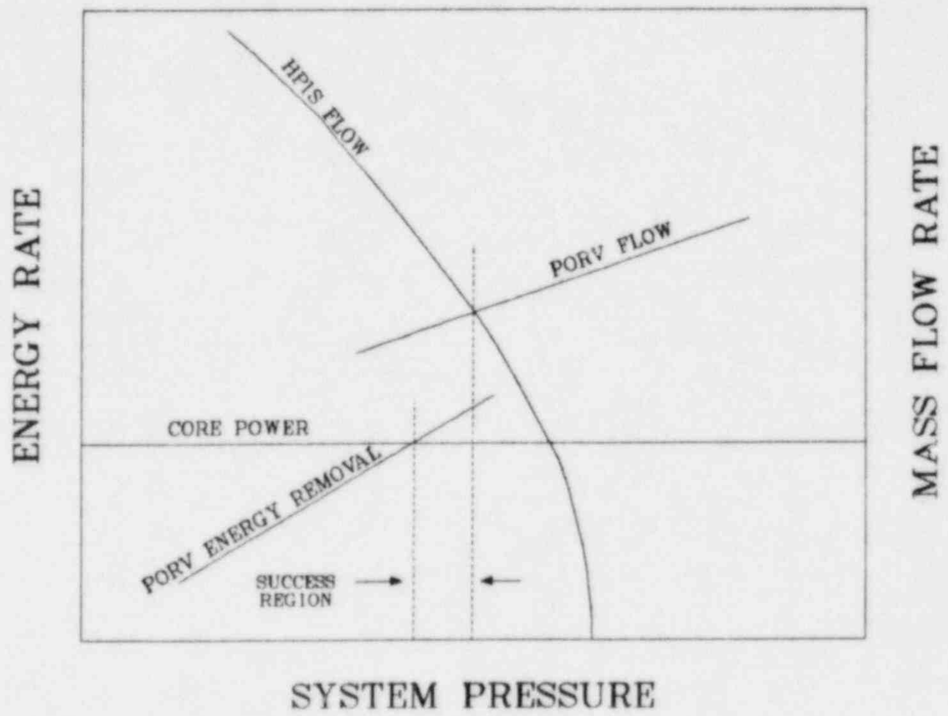

Fig. 97.
Idealized feed-and-bleed operating map.

Figure 97 was generated assuming saturated vapor at the PORV. However, calculated transients show a wide variation in fluid conditions during the course of a feed-and-bleed procedure. For this reason, a more complex map structure is required to define the success region completely. This is accomplished by adding additional saturated and subcooled PORV flow-vs-pressure curves. An additional modification to the longer map can be used; the energy rate ordinate is eliminated by determining an equivalent mass flow required to remove core-decay heat. The resultant basic feed-and-bleed operating map structure based on Calvert Cliffs-1 PORV and HPI characteristics is presented in Fig. 98. There is a range of success regions. The upper pressure bound is defined as previously discussed. The lower pressure bound is a succession of pressures at which the PORV outflow removes the core-decay heat given different fluid conditions.

Finally, we superimpose on the plant-specific feed-and-bleed operating map an important trace: the TRAC-PF1-calculated PORV flow-vs-pressure for the feed-and-bleed transient of interest. A completed map with superimposed calculated trace is shown in Fig. 99. The transient selected has feed and bleed initiated at the time of containment overpressure (2900 s) following a LOFW

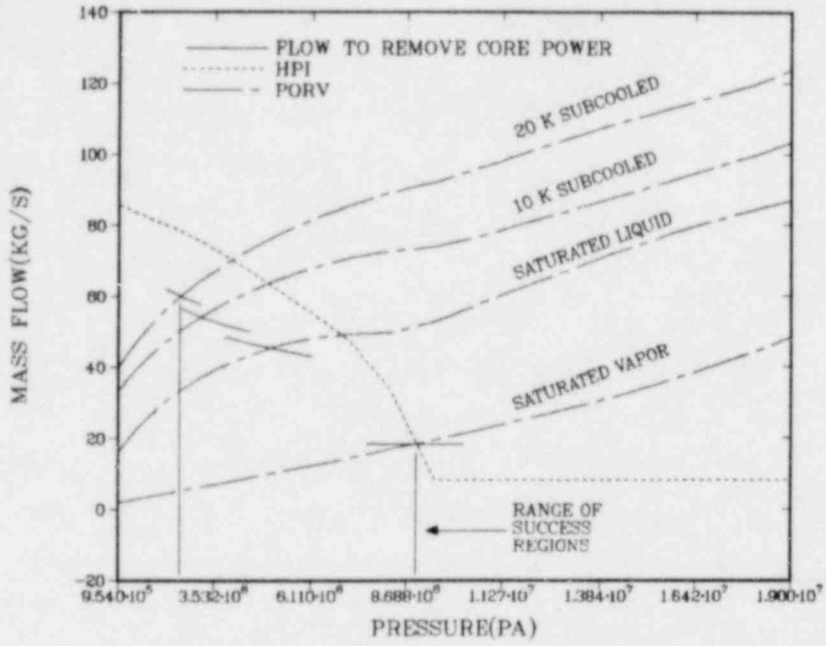

Fig. 98.
Generalized feed-and-bleed operating map for Calvert Cliffs-1.

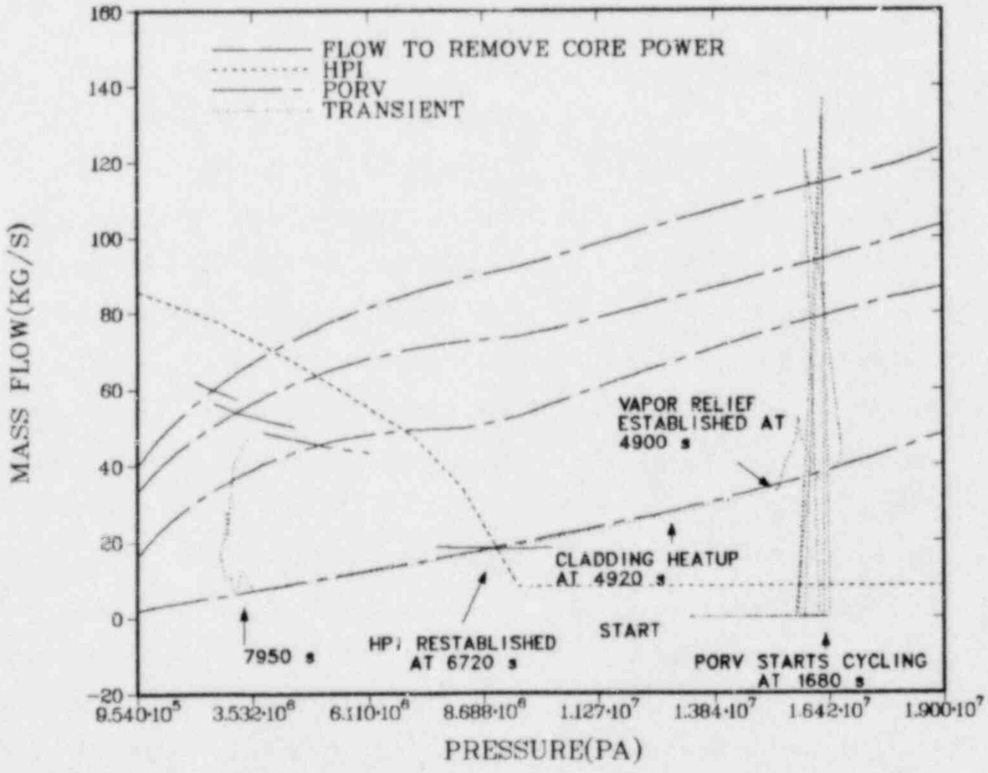

Fig. 99.
Generalized feed-and-bleed operating map for Calvert Cliff-1 with superimposed calculated transient.

transient. This transient was previously discussed in Sec. IV.A.2.C. Although the calculated trace starts outside the success region, it eventually enters at 6700 s and remains until the end of the calculated transient at 8830 s. As previously discussed in Sec. IV.A.2.c, this feed-and-bleed transient failure because of a core dryout occurred at 4920 s, and cladding temperature rapidly increased above saturation. Thus, we see that the existence of a feed-and-bleed success region does not guarantee that feed and bleed will be successful and that the map may not be used as a tool to predict the success or failure of feed and bleed. The example just discussed shows that plants can experience a transient in which the final state resides within the success region but feed and bleed has failed. The primary shortcoming of the maps has been found to be their steady-state basis. Clearly, transient phenomena determine the outcome. Alternative approaches for enhanced inspection have not been identified.

The third approach uses all the inspection information but emphasizes the development and use of simplified plant-specific models that are inexpensive but will capture the dominant phenomena of feed and bleed. However, such models would still be manpower-intensive and may require an extensive data base to ensure that the plants are properly modeled. The fourth approach is that taken for the four plant-specific studies conducted thus far. Detailed plant-specific models are developed and plant performance simulated using a detailed systems analysis code such as TRAC-PF1 to perform transient calculations. Although we have the most confidence in the results produced using this approach, the costs are prohibitive.

Within the time and funding constraints of the USI A-45 program, only the first two approaches were investigated. As previously discussed, the techniques that we had identified for enhanced inspection were inadequate. Therefore, we relied on simple inspection for our extension statements. While realizing the inherent limitations of this method, we believe that the resultant extension statements adequately characterize the ability of given plants to successfully feed and bleed. The process of extension from our insights about plants for which we performed detailed calculations to a broader class of plants is illustrated in Table XXII.

We have selected six C-E plants for our example: Arkansas Nuclear One-2, Calvert Cliffs-1 and -2, Fort Calhoun-1, St. Lucie-1, and Maine Yankee. Calvert Cliffs-1 is the reference plant for which detailed calculations have been performed. Calvert Cliffs-2 is nearly identical to Calvert Cliffs-1; therefore, using the simple inspection approach we would expect it to perform similarly during feed and bleed. Arkansas Nuclear One-2 is not equipped with PORVs; however, it does have a vent valve. We were unable to determine the vent valve relief capacity; therefore, we make no extension statements for this plant. Fort Calhoun-1 has a core thermal rating slightly over half the Calvert Cliffs-1 value. The PORV relief capacity/MWt is greater than that of Calvert Cliffs-1, the shutoff head is higher, the HPI delivery rate is greater, and the charging delivery rate is also larger. Because Fort Calhoun-1 either meets or exceeds the Calvert Cliffs-1 parameters identified as important to feed and bleed, the simple inspection approach leads us to expect that feed and bleed will be successful at Fort Calhoun-1 under similar circumstances; e.g., if

TABLE XXII

EXAMPLE OF SIMPLE INSPECTION PROCESS FOR C-E PLANTS

	Arkansas Nuclear One-2	Calvert Cliffs-1, -2	Fort Calhoun-1	St. Lucie-1	Maine Yankee
PORV Capacity (lb/hr/MWt)	Vent Valve [10] (Capacity Unknown)	56.7	69.7	59.8	57.0
HPI Shutoff Head (psi)	1517	1257	1387	1257	2471
gpm/MWt at 1000 psig	0.18	0.16	0.19	0.17	0.27
gpm/MWt at1600 1600 psig	0	0	0	0	0.21
Charging Capacity (gpm/MWt)	0.05	0.05	0.08	0.05	0.17

initiated no later than the loss-of-secondary heat sink. St. Lucie-1 is also similar to Calvert Cliffs-1; thus we assume, on the basis of simple inspection, that it will perform similarly during feed and bleed. The last plant in Table XXII, Maine Yankee, is quite different from the reference plant. In fact, this plant more closely resembles W and B&W plants that have high-pressure SI systems. Although not evidenced in Table XXII, a comparison of Maine Yankee with such plants by the simple inspection approach leads us to state that Maine Yankee can feed and bleed successfully as late as the time of primary-system saturation. In contrast, feed and bleed must be initiated no later than SGSD in Calvert Cliffs-2, Fort Calhoun-1, and St. Lucie-1, as in the Calvert Cliffs-1 reference plant, to be successful.

A. B&W-Designed Plants

A comparison of key characteristics of B&W plants relative to loss-of-feedwater transients is presented in Table XXIII, which is based on data from Ref. 31. The key parameters of interest are the PORV capacity and the HPI delivery capacity. All the operating B&W plants listed, except Davis-Besse, are lowered-loop plants. Davis-Besse is a raised-loop plant, meaning the steam-generator inlet nozzle is only slightly below the hot-leg center line at the vessel exit. In the lowered-loop plants, the steam-generator inlet nozzle is located about 30 ft below the hot-leg centerline at the vessel exit.

We made no extension statements for Davis-Besse. In addition to the raised-loop design, the HPI characteristics differ markedly from those of other B&W operating plants. We believe and recommend that a model of the Davis-Besse plant should be developed and used to analyze the feed-and-bleed procedure for Davis-Besse.

Although they are not identical, the Oconee plants, Arkansas Nuclear One-1, Crystal River-3, Three Mile Island-1, Three Mile Island-2, and Rancho Seco are similar in PORV capacity and HPI-delivery capacities. We therefore believe that feed-mode cooling will be effective in these plants. We also believe that feed and bleed can be used in each of these plants if appropriate written procedures are made available and if equipment and instrumentation are available and permit the initiation and control of the feed-and-bleed operation.

TABLE XXIII

COMPARISON OF KEY PLANT CHARACTERISTICS OF B&W PLANTS RELATIVE TO LOSS-OF-FEEDWATER TRANSIENTS[31]

Plant Parameter	Oconee 1,2,3	ANO-1	Crystal River-3	TMI-1	TMI-2	Rancho Seco	Davis-Besse
Core Thermal Power MWt/# of Loops	2568/2	2568/2	2452/2	2535/2	2772/2	2772/2	2772/2
PORV Capacity lb/hr/mwt / Set point, psi (to be reviewed per IE 79-05B)	41.7/2255	38.9/2255	40.8/2250	38.9/2255	40.4/2255	40.4/2255	40.4/2255
Number of PORVs	1	1	1	1	1	1	1
HPI Shutoff Head ft/psi	7000/3034	7000/3034	6500/2817	6500/2817	6500/2817	7000/3034	4000 MV 6500/1734 MV 2817
gpm at 1000 psia	500 ea	500 ea	500 ea	500 ea[a]	500 ea	500 ea	700 ea
gpm at 1600 psig	450 ea	450 ea	450 ea	450 ea	450 ea	450 ea	200 ea
Steam-generator time to dryout, full-power min	0.48	0.48	0.50	0.48	0.45	0.45	0.45

[a]Ref. 31 shows 350 GMP but this appears to be an error.

B. Combustion Engineering-Designed Plants

A comparison of key plant characteristics of C-E plants relative to loss-of-feedwater transients is presented in Table XXIV based on data from Ref. 32. All the operating C-E plants listed, except Arkansas Nuclear One-2, are equipped with PORVs. Arkansas Nuclear One-2 is equipped with a vent valve, but its relief capacity is not known to us. If sufficiently large, Arkansas Nuclear One-2 should also be able to feed and bleed at SGSD.

Although they are not identical, the Calvert Cliffs-1 and -2, Millstone-2, Palisades, and St. Lucie-1 are similar in PORV capacity and HPI-delivery capacities. We showed for Calvert Cliffs-1 that feed-mode cooling, if initiated at the time of containment overpressure following either an LOSP-induced LOFW event or an LOFW event, was not adequate but that feed-and-bleed cooling was effective in cooling the reactor for LOFW events if initiated no later than the time of steam-generator-secondary dryout. The Maine Yankee SI characteristics are very different from those of the other C-E plants. The SI characteristics are similar to those of the B&W plants. We therefore conclude that feed-and-bleed cooling will be successful if initiated no later than primary-system saturation. We believe that feed-and-bleed can be used in each of these plants if appropriate written procedures are made available and if the needed equipment and instrumentation are available and permit the initiation and control of the feed-and-bleed operation.

The core thermal power of Fort Calhoun-1 is 1420 MWt, or 55% of the Calvert Cliffs-1 core thermal power. The Fort Calhoun-1 HPI delivery at 6.89 MPa (1000 psig) is 70% of the Calvert Cliffs-1 HPI-delivery rate. The Fort Calhoun-1 PORV capacity is 33% larger than that of Calvert Cliffs-1. We believe that feed and bleed can be used in Fort Calhoun if appropriate written procedures are made available and if the needed equipment and instrumentation are available and permit the initiation and control of the feed-and-bleed operation.

Maine Yankee differs significantly from the other C-E plants. The primary-coolant system consists of three loops, each having a dedicated steam generator and RCP. In addition, the charging pumps at Maine Yankee provide for high head emergency core cooling in a manner similar to that of Zion-1. We believe that feed mode cooling will be effective in Maine Yankee. We also believe that feed-and-bleed can be used in Maine Yankee if appropriate written procedures are made available and if the needed equipment and instrumentation

TABLE XXIV

COMPARISON OF KEY PLANT CHARACTERISTICS OF C-E PLANTS RELATIVE TO LOSS-OF-FEEDWATER TRANSIENTS[32]

Plant Parameter	ANO-2	Calvert Cliffs-1&-2	Fort Calhoun-1	Maine Yankee	Milestone-2	Palisades	St. Lucie-1
Core Thermal Power Mwt/# of Loops	2815/2	2570/2	1420/2	2630/3	2560/2	2530/2	2560/2
PORV Capacity lb/hr/Mwt / setpoint, psi	NONE	56.7/2385	69.7/2392	57.0/2385	59.8/2380	60.5/2385	59.8/2385
Number of PORVs	None	2	2	2	2	2	2
HPI Shutoff Head ft/psi	3500/1517	2900/1257	3200/1387	5700/2471	2800/1213	2900/1257	2900/1257
gpm @ 1000 psig	500	400	280	715	475	400	425
gpm @ 1600 psig	0	0	0	550	0	0	0
Positive displacement charging pump capacity, gpm	128	132	120	450	132	133	132
Steam generator time to dryout, min	14	16	16	14	15	16	16

are available and permit the initiation and control of the feed-and-bleed operation.

C. Westinghouse-Designed Plants

A comparison of key plant characteristics of W plants relative to loss-of-feedwater transients is presented in Table XXV based on data from Ref. 33. There are significant differences among the W plants; we will consider them according to the number of primary loops and whether they are HP, IP, or LP SI plants. Extension statements are based on comparisons to Zion-1, which is a four-loop HP SI plant and H. B. Robinson-2, which is a three-loop LP SI plant.

1. Four-Loop Plants. Zion-1 and -2, D. C. Cook-1 and -2, Trojan, and Salem-1 are all HP SI plants that have similar PORV and HPI capacities. We showed that Zion-1 can be cooled using feed or feed-and-bleed procedures following loss-of-feedwater events. We believe that the four-loop W plants similar to Zion-1 can also be cooled using feed or feed-and-bleed procedures if each of the plants has appropriate written procedures and the needed equipment and instrumentation are available and permit the initiation and control of the operations. Haddam Neck is a smaller four-loop plant, but it has larger PORV and HPI capacities. We believe feed-and-bleed cooling will be effective in Haddam Neck if appropriate written procedures are available and the needed equipment and instrumentation are available and permit the initiation and control of the feed-and-bleed operation.

South Texas-1 and -2 are IP SI plants, and Indian Point-1 and -2 are LP SI plants. We base our extension statements for these four plants on our H. B. Robinson detailed calculation. The PORV and HPI capacities (on a per-loop basis) for these plants are similar to those of H. B. Robinson; the primary difference is that the plants have four primary loops while H. B. Robinson has three primary loops. Because no attempt was made to evaluate the ability of H. B. Robinson-2 to feed and bleed at the time of primary-system saturation, we have drawn no conclusion regarding the ability of these four plants to feed and bleed at the time of primary-system saturation.

2. Three-Loop Plants. We have based our three-loop extension statements on either the Zion-1 or the H. B. Robinson-2 calculations as appropriate. Summer, Shearon Harris-1 and -2, Farley-1 and -2, Beaver Valley-1 and -2, North Anna-1 and -2, and Surrey-1 and -2 are three-loop HP plants. We believe that these plants, when compared to Zion-1 on a per-loop basis, have sufficient PORV and HPI capacities to cool the reactor successfully using a feed-and-bleed

TABLE XXV

COMPARISON OF KEY PLANT CHARACTERISTICS OF W PLANTS RELATIVE TO LOSS-OF-FEEDWATER TRANSIENTS[26]

Plant Parameter	H Farley-1	H.B Robinson-2	Zion-1 &-2	Haddam Neck	Indian Point-2	Indian Point-3	Indian Valley-1
Core Thermal Power MWt/# of Loops	2652/3	2200/3	3250/4	1825/4	2758/4	3025/4	2660/3
PORV Capacity lb/hr/MWt / setpoint, psi	79.2/2335	95.5/2335	64.6/2335	115.1/2270	78.7/2335	78.7/2335	79.9/2335
Number of PORVs	2	2	2	2	2	2	3
HPI Shutoff Head ft/psi	6000/2600	3300/1430	6000/2600	6800/2948	3500/1517	3500/1517	5850/2536
gpm @ 1000 psig	750	350	490	~800	485	485	495
gpm @ 1600 psig	550	0	380	575	0	0	380
gpm @ PORV setpoint	250/325	N/A	165/230	325/380	N/A	N/A	175/275
Positive displacement charging pump capacity, gpm	-	231	-	-	294	294	-
Steam generator time to dryout, minutes	37.3	32.0	45.8	22.0	31.5	29.2	37.3

TABLE XXV (cont)

COMPARISON OF KEY OPERATING CHARACTERISTICS OF W PLANTS RELATIVE TO LOSS-OF-FEEDWATER TRANSIENTS

Plant Parameter	Turkey Point-3&-4	D.C. Cook-1	D.C. Cook-2	Prairie Is.-1&-2	Trojan	Salem-1	Ginna
Core Thermal Power MWt/# of Loops	2208/3	3250/4	3400/4	1650/2	3411/4	3338/4	1520/2
PORV Capacity lb/hr/MWt / setpoint, psi	95.1/2335	64.6/2335	61.8/2335	108.5/2335	61.6/2350	63/2350	117.8/2335
Number of PORVs	2	3	3	2	2	2	2
HPI Shutoff Head ft/psi	3500/1517	5800/2514	5800/2514	5000/2168	6000/2600	6168/2670	3426/1485
gpm @ 1000 psig	410	560	560	760	495	490	285
gpm @ 1600 psig	0	400	400	600	380	380	0
gpm @ PORV setpoint	N/A	170/225	170/225	N/A	180/230	150/210	N/A
Positive displacement charging pump capacity, gpm	231	-	-	180	-	-	180
Steam generator time to dryout, min	31.3	45.8	42.1	24.7	39.3	43.1	29.0

TABLE XXV (cont)

COMPARISON OF KEY OPERATING CHARACTERISTICS OF W PLANTS RELATIVE TO LOSS-OF-FEEDWATER TRANSIENTS

Plant Parameter	San Onofre-1	Surry-1&-2	North Anna-1	Point Beach-1&-2	Kewannee	Yankee Rowe
Core Thermal Power MWt/# of Loops	1347/3	2449/3	2775/3	1518/2	1650/2	600/4
PORV Capacity lb/hr/MWt / Set point, psi	80/2190	86/2335	76/2335	117.9/2335	106/2335	118/2400
Number of PORVs	2	2	2	2	2	1
HPI Shutoff Head ft/psi	6000/2600	6000/2600	5900/2550	3550/1539	5000/2167	1950/844
gpm @ 1000 psig	570	520	650	900	750	0
gpm @ 1600 psig	460	420	550	0	500	0
gpm @ PORV setpoint	165/300	255/250	250/325	N/A	N/A	N/A
Positive displacement charging pump capacity, gpm	-	-	-	180	180	99
Steam generator time to dryout, min	22.5	43.1	37.3	40.0	44.2	51.6

procedure. We again note that the appropriate written procedure must be available to permit the initiation and control of the feed-and-bleed procedure. Turkey Point-3 and -4 are three-loop LP plants that are similar to H. B. Robinson-2. By simple inspection, we conclude that these plants can also feed and bleed successfully as late as SGSD. We draw no conclusion regarding the success or failure of feed and bleed initiated at the time of primary-system saturation.

3. Two-Loop Plants. Prairie Island-1 and -2 and Kawaunee are two-loop IP SI plants. As the number of loops decreases, we are less confident in our ability to make extension statements on the basis of simple inspection. However, we conclude that these three plants, when compared to H. B. Robinson-2, have sufficient PORV and HPI capacities to feed and bleed successfully as late as SGSD. We note in doing so that these plants have IP SI capability. We do not feel confident in making extension statements for the two-loop LP SI plants Ginna and Point Beach-1 and -2.

IX. CONCLUSIONS

An alternative means for removal of shutdown decay heat from PWRs has been investigated. Feed and bleed, a diverse alternative method of removing decay heat that does not rely on use of the steam generators, includes the delivery of SI and charging flows to the primary and a controlled (manual) depressurization of the primary using the pressurizer PORVs. Before stating the conclusions of this study, we emphasize that our studies assume that the equipment used to effect a feed-and-bleed procedure is available and operable throughout the event. In a real sense, therefore, the study is idealized and each plant must be examined in detail to determine if the required equipment, instrumentation, and procedures are in place to permit the use of feed and bleed.

We have reached the following general conclusions about the feed and bleed procedure.

1. Feed and bleed is a potentially useful alternative method of decay heat removal in PWRs following the loss of normal cooling mode through the steam generators. The method relies on the existence of primary-system PORVs to provide a pathway for the release of core decay heat and sufficient SI capacity to deliver coolant to the primary.

2. The availability of HP SI delivery capacity greatly enhances the reliability of the feed-and-bleed operation. Plants with only LP or IP SI systems must initiate feed and bleed no later than the loss-of-secondary heat sink. Plants with HP SI systems can successfully use feed and bleed until the time of primary-system saturation.

3. PORV capacity becomes important during the transition from reactor trip to either RHR or LPI entry conditions if only safety-grade water supplies are considered. Plants with a single small PORV depressurize more slowly than plants with two larger PORVs. Safety-grade water supplies may be consumed before RHR or LPI entry conditions are reached. However, piggy-back operation of the LPI and HPI systems could be used for maintaining core cooling.

4. Simple inspection is a useful technique for extending the limited set of detailed plant-specific calculations to a broader set of plants. However, we are less confident of the accuracy of conclusions reached by simple inspection than of the accuracy of conclusions based on detailed plant-specific calculations.

The detailed plant calculations for Oconee-1, Calvert Cliffs-1, Zion-1, and H. B. Robinson-2 have helped us to develop insights about feed and bleed. We have reached the following additional and more specific conclusions that provide the foundation for our simple-inspection extension statements.

1. Feed and bleed was successfully applied in each of the four plants studied in detail provided it was initiated no later than the time of SGSD. For LOSP-induced LOFW events and LOFW initiating events, the accident signatures for each plant studied in detail (Oconee-1, Calvert Cliffs-1, and Zion-1) were similar, although timing varied. For the MFLB and MSLB combined with LOFW, the early event signature was dominated by the overcooling associated with steam-generator-secondary blowdown. After recovery from overcooling, the accident signature was similar to those of the LOSP and LOFW events.

2. Feed and bleed can be started after steam-generator-secondary dryout in Oconee-1 and Zion-1 and satisfy the criteria for successful feed and bleed. For these two plants, feed and bleed was successful if initiated no later that the time of primary system saturation. Feed and bleed was not successful in the Calvert Cliffs-1 plant if initiated at primary system saturation. We did not calculate this transient for H. B. Robinson-2. We did not determine the latest time at which feed and bleed could be successful.

3. The primary factor permitting late feed and bleed in Oconee-1 was the large flow delivered by the HPI system. The flow rates were sufficiently large that the small Oconee-1 PORV relief capacity was not a dominant factor. In fact, the success criterion for transition from reactor trip to a hot holding condition could be satisfied using a feed-only procedure.

4. Zion-1 could also satisfy the criterion for transition from reactor trip to a hot holding condition using a feed-only procedure. However, the large PORV relief capacity in Zion-1 was found to be important in aiding the transition from reactor trip to hot shutdown, which requires both cooldown and depressurization of the primary. The small PORV relief capacity in Oconee-1 slowed the transition from reactor trip to hot shutdown. It was not always possible to complete the transition using only the inventory of the BWST. Switching to recirculation-mode cooling by taking suction from the containment sump would be required.

5. Calvert Cliffs could not successfully effect the transition from reactor trip to hot standby using only the feed delivered by the charging pumps at the PORV setpoint if initiated when the containment overpressure signal is generated (shortly after primary saturation). After saturation, the PORVs could not depressurize the primary below the HPI cutoff head before cladding heatup began. Feed and bleed was successful if started earlier while the primary was liquid full and subcooled.

Our findings on feed and bleed, both directly by detailed investigation and extended by simple inspection, are summarized in Table XXVI.

TABLE XXVI

SUMMARY RESULTS

VENDOR	PLANT TYPE	CALCULATION	EXTENSION	SGSD	SATURATION
C-E	2 x 4 loop LP SI	Calvert Cliffs-1	Calvert Cliffs-2	Y	N
			Fort Calhoun-1	Y	N
			Maine Yankee	Y	Y
			Millstone-2	Y	N
			Palisades	Y	N
			St. Lucie-1	Y	N
			Ark. Nuclear One-2	NC	NC
B&W	2 x 4 loop HP SI	Oconee-1	Oconee-2,-3	Y	Y
			Ark. Nuclear One-1	Y	Y
			Crystal River-3	Y	Y
			Three Mile Island-1,-2	Y	Y
			Rancho Seco	Y	Y
W	4-loop HP SI	Zion-1	Zion-2	Y	Y
			DC Cook-1,-2	Y	Y
			Trojan	Y	Y
			Salem-1,-2	Y	Y
			Haddam Neck	Y	Y
	4-loop IP SI		South Texas-1,-2	Y	NC
	4-loop LP SI		Indian Point-2,-3	Y	NC
	3-loop HP SI		Summer	Y	Y
			Shearon Harris-1,-2	Y	Y
			Farley-1,-2	Y	Y
			Beaver Valley-1,-2	Y	Y
			North Anna-1,-2	Y	Y
			Surry-1,-2	Y	Y
	3-loop LP SI	Robinson-2	Turkey Point-3,-4	Y	NC
	2-loop IP SI		Prairie Island-1,-2	Y	NC
			Kewaunee	Y	NC
	2-loop LP SI		Ginna	NC	NC
			Point Beach-1,-2	NC	NC

Y = Yes
N = No
NC = No conclusion
LOSHS = Loss-of-secondary heat sink

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APPENDIX A

TRAC DESCRIPTION

The Transient Reactor Analysis Code (TRAC) is being developed at the Los Alamos National Laboratory under the sponsorship of the US Nuclear Regulatory Commission to provide advanced best-estimate predictions of postulated accidents in light-water reactors. The TRAC-PF1 code provides this capability for PWRs and for many thermal-hydraulic experimental facilities. Some distinguishing characteristics of TRAC-PF1 are summarized in the following. Within restrictions imposed by computer running times, attempts are being made to incorporate state-of-the-art technology in two-phase thermal hydraulics.

A. Variable-Dimensional Fluid Dynamics

A full three-dimensional (r, θ, z) flow calculation can be used within the reactor vessel; the flow within the loop components is treated one-dimensionally. This allows an accurate calculation of the complex multidimensional flow patterns inside the reactor vessel that are important during accidents. For example, phenomena such as ECC downcomer penetration during blowdown, multidimensional plenum and core flow effects, and upper-plenum pool formation and core penetration during reflood can be treated directly. However, a one-dimensional vessel model may be constructed that allows transients to be calculated very quickly because the usual time-step restrictions are removed by the special stabilizing numerical treatment.

B. Nonhomogeneous, Nonequilibrium Modeling

A full two-fluid (six-equation) hydrodynamics model describes the steam-water flow, thereby allowing important phenomena such as counter-current flow to be treated explicitly. A stratified-flow regime has been added to the one-dimensional hydrodynamics, and a seventh field equation (mass balance) describes a noncondensable gas field.

C. Flow-Regime-Dependent Constitutive Equation Package

The thermal-hydraulic equations describe both the transfer of mass, energy, and momentum between the steam-water phases and the interaction of these phases with the system structure. Because these interactions are dependent on the flow topology, a flow-regime-dependent constitutive equation package has been incorporated into the code. Although this package undoubtedly will be improved in future code versions, assessment calculations performed to date indicate that many flow conditions can be handled adequately with this package.

D. Comprehensive Heat-Transfer Capability

The TRAC-PF1 program provides detailed heat-transfer analyses both for the vessel and for the loop components. Included is a two-dimensional (r,z) treatment of fuel-rod heat conduction with dynamic fine-mesh reasoning to resolve both bottom flood and falling-film quench fronts. The heat transfer from the fuel rods and from other system structures is calculated using flow-regime-dependent heat-transfer coefficients obtained from a generalized boiling curve based on local conditions.

E. Consistent Analysis of Entire Accident Sequence

An important TRAC feature is its ability to address entire accident sequences, including computation of initial conditions, in a consistent and continuous calculation. For example, the code models the blowdown, refill, and reflood phases of a LOCA. In addition, steady-state solutions provide self-consistent initial conditions for subsequent transient calculations. Both steady-state and transient calculations can be performed in the same run, if desired. This modeling eliminates the need for calculations by different codes to analyze a given accident.

F. Component and Functional Modularity

The TRAC program is completely modular. The components in a calculation are specified through input data; available components allow the user to model virtually any PWR design or experimental configuration. This gives TRAC great versatility in application to varied problems. It also allows component modules to be improved, modified, or added without disturbing the remainder of the code. TRAC component modules currently include accumulators, pipes, pressurizers, pumps, steam generators, tees, valves, and vessels with associated internals (downcomer, lower plenum, core, upper plenum, etc.).

The TRAC program also is modular by function; that is, major aspects of the calculations are performed in separate modules. For example, the basic one-dimensional hydrodynamics solution algorithm, the wall-temperature field solution algorithm, heat-transfer coefficient selection, and other functions are performed in separate sets of routines that are accessed by all component modules. This modularity allows the code to be upgraded readily as improved correlations and experimental information become available.

APPENDIX B

PLANT MODELS

I. OCONEE-1

Oconee-1 is a Babcock-and-Wilcox lowered-loop pressurized water reactor and consists of a vessel, two once-through steam generators, and two hot legs and four cold legs, all of which are included in the TRAC model. Reactivity feedback from fuel and moderator temperature is included in the vessel model. The TRAC noding diagram for this model is shown in Fig. B.1. Information for this model was obtained from the Final Safety Analysis Reports.

The two cold legs on the "B" side are modeled as one combined cold leg for computational efficiency. Also modeled on the primary side are the main coolant pumps, loop seals, surge line, pressurizer, emergency core-cooling injection [including high-pressure injection and core-flooding tanks (accumulators)], hot-leg candy canes, and upper plenum vent valves. The secondary side of the model includes the steam lines, main feedwater, and auxiliary feedwater with steam-generator level control. The ARVs, TSVs and turbine-bypass valves are modeled by a mass flow versus pressure boundary condition. The primary-system relief train was modeled as a TEE component that was connected to the PORV and SRVs.

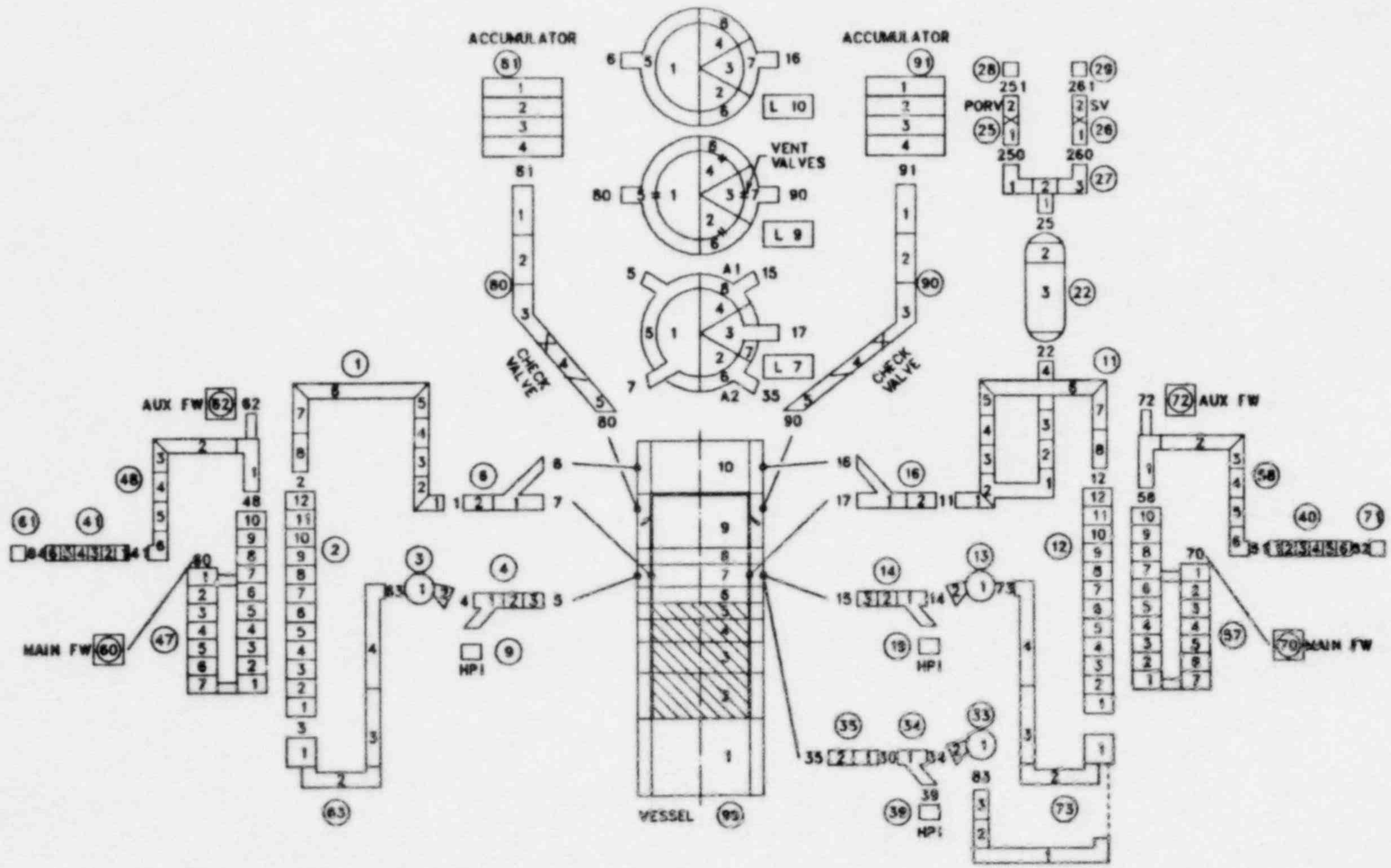


Fig. B.1.
TRAC noding of Oconee-1 PWR with combined B-loop cold legs and 4-θ vessel.

II. CALVERT CLIFFS-1

A system schematic of the TRAC-PF1 model for the Calvert Cliffs-1 PWR is shown in Fig. B.2. Information for this model of a two-loop four-RCP Combustion Engineering plant was derived from several sources. These include:

- the Updated Final Safety Analysis Report (FSAR),
- Bechtel drawings,
- drawings and verbal communications from Combustion Engineering,
- verbal communications from Baltimore Gas & Electric, and
- engineering analyses and computer input from Science Applications Inc.

To reduce the computing time, the two cold legs in each loop were lumped together into a single cold leg with twice the volume and flow area of the actual cold legs. The pressurizer quench tank was simulated with a pressure boundary condition at the outlets of the PORV and SRV.

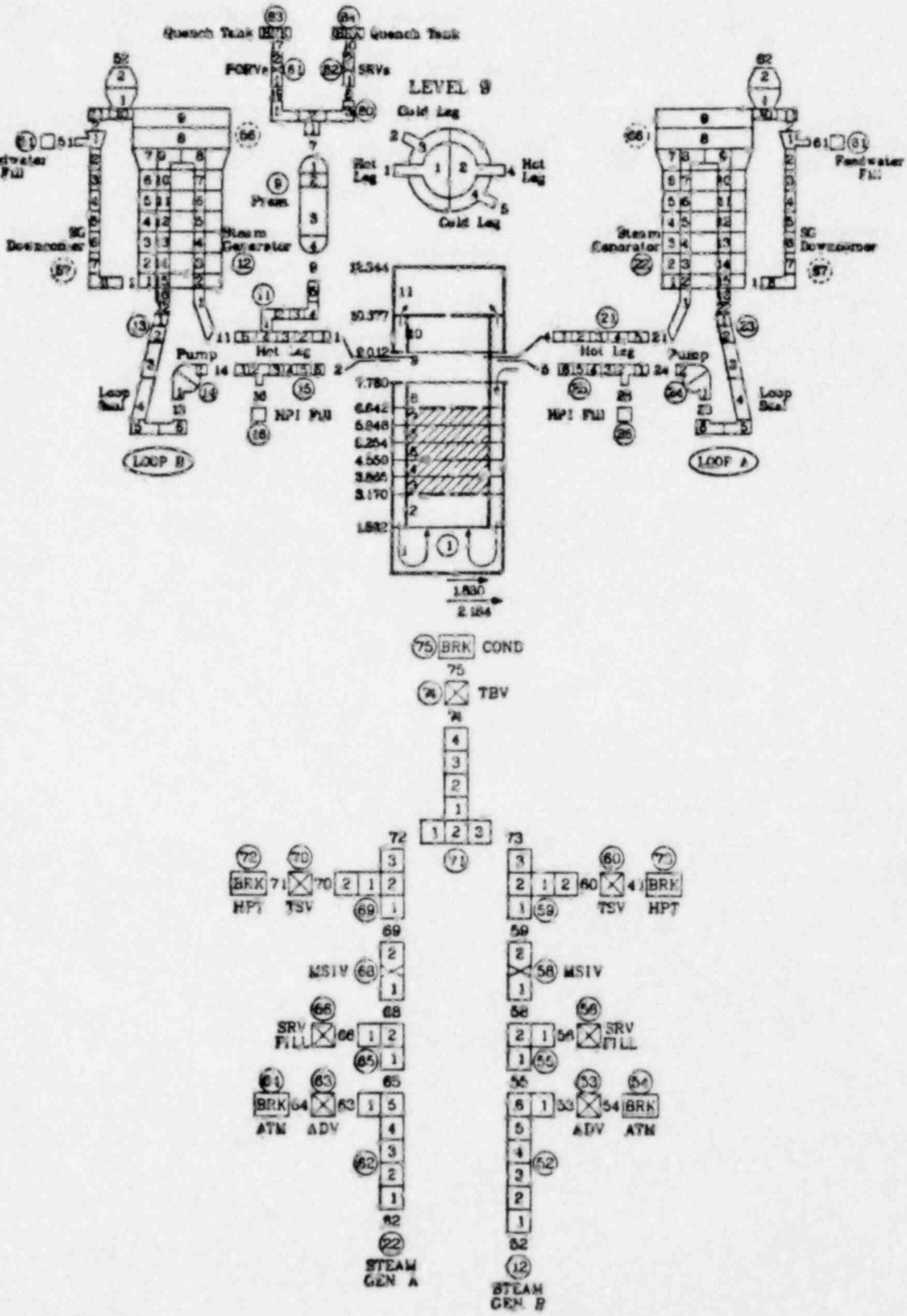


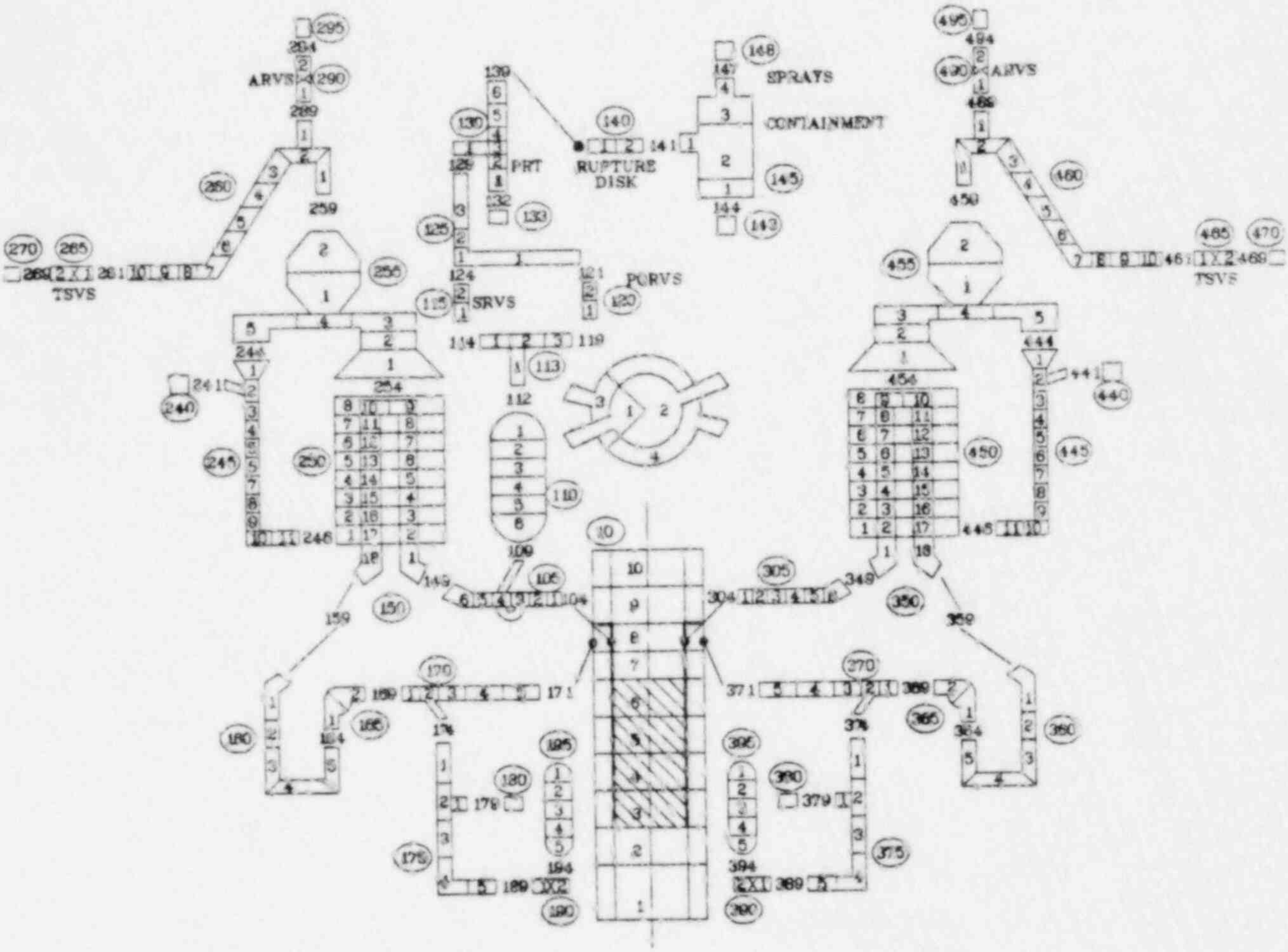
Fig. B.2.
TRAC noding of Calvert Cliffs-1 with combined cold legs.

III. ZION-1

A system schematic of the audited TRAC-PF1 model for the Zion-1 PWR is shown in Fig. B.3. Information for this model of a four-loop Westinghouse plant was derived from several sources. These include:

- the FSAR,
- Westinghouse drawings and proprietary data,
- Sargent and Lundy drawings,
- other vendor drawings,
- Generic evaluation of "Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," NUREG-0611 (January 1980), and
- visits to the reactor 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 the PORVs, SRVs, header, and PRT. Both the PORVs and SRVs were modeled as static check valves that open and close depending upon the pressure difference across the valve. The PORVs and SRVs connect to a common header that leads to the PRT. A pathway from the PRT to the containment is provided by rupture disks that are designed to open when the pressure in the tank reaches 0.687 MPa (99.7 psig), thus providing a connection from the PORV and SRV discharges to the containment. Included in the primary loops are separate components for modeling ECC injection, primary coolant makeup, RCPs, and U-tube steam generators. On the secondary loops, the feedwater sources, steam lines, and ARVs are modeled. This TRAC model for the Zion-1 plant consists of 42 components and 265 cells of which 40 are in the three-dimensional vessel.



LOOP-B
100 Loop Primary
200 Loop Secondary
200-249 FW
250-299 Steam

LOOP-A,C,D
300 Loop Primary
400 Loop Secondary
400-449 FW
450-499 Steam

Fig. B.3.
TRAC system schematic for Zion-1 PWR.

IV. H. B. ROBINSON-2

An audited input model for the primary system of the H. B. Robinson-2 PWR is shown in Figs. B.4 through B.8. Information for this model of a three-loop W plant was derived from several sources. These include:

- The FSAR,
- W drawings,
- Carolina Power and Light supplied data, and
- Idaho National Engineering Laboratory.

The three primary loops were each modeled individually, and loop three was chosen to be the pressurizer. The PORV and SRV modeling at the top of the pressurizer is shown in Fig. B.7. Both the PORVs and SRVs were modeled as check valves that open and close depending upon the pressure difference across the valve. Included in each primary loop are separate components for modeling ECC injection, primary coolant makeup, the RCPs, and the U-tube steam generators. The vessel is modeled in three dimensions.

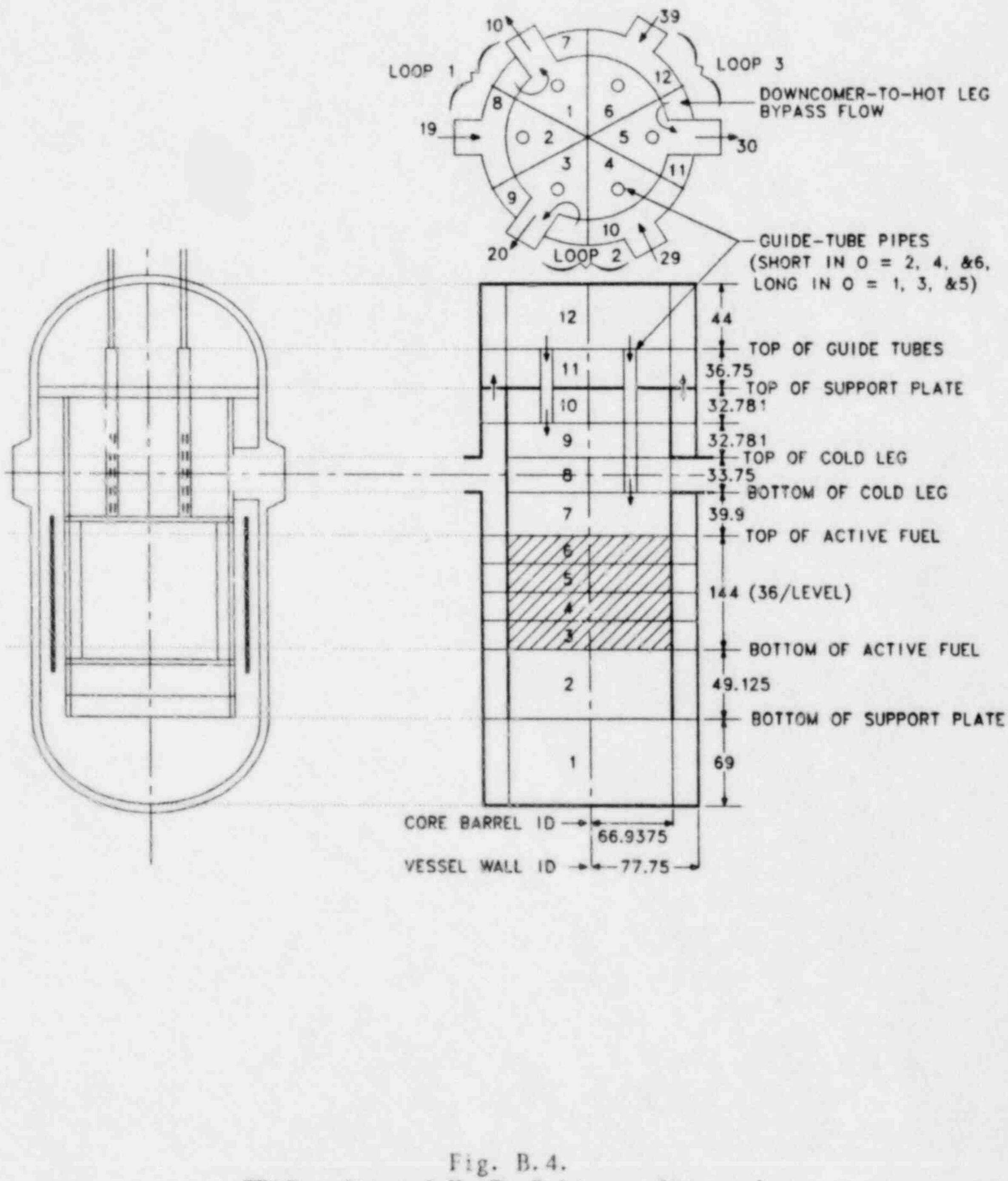


Fig. B.4.
TRAC noding of H. B. Robinson-2 vessel.

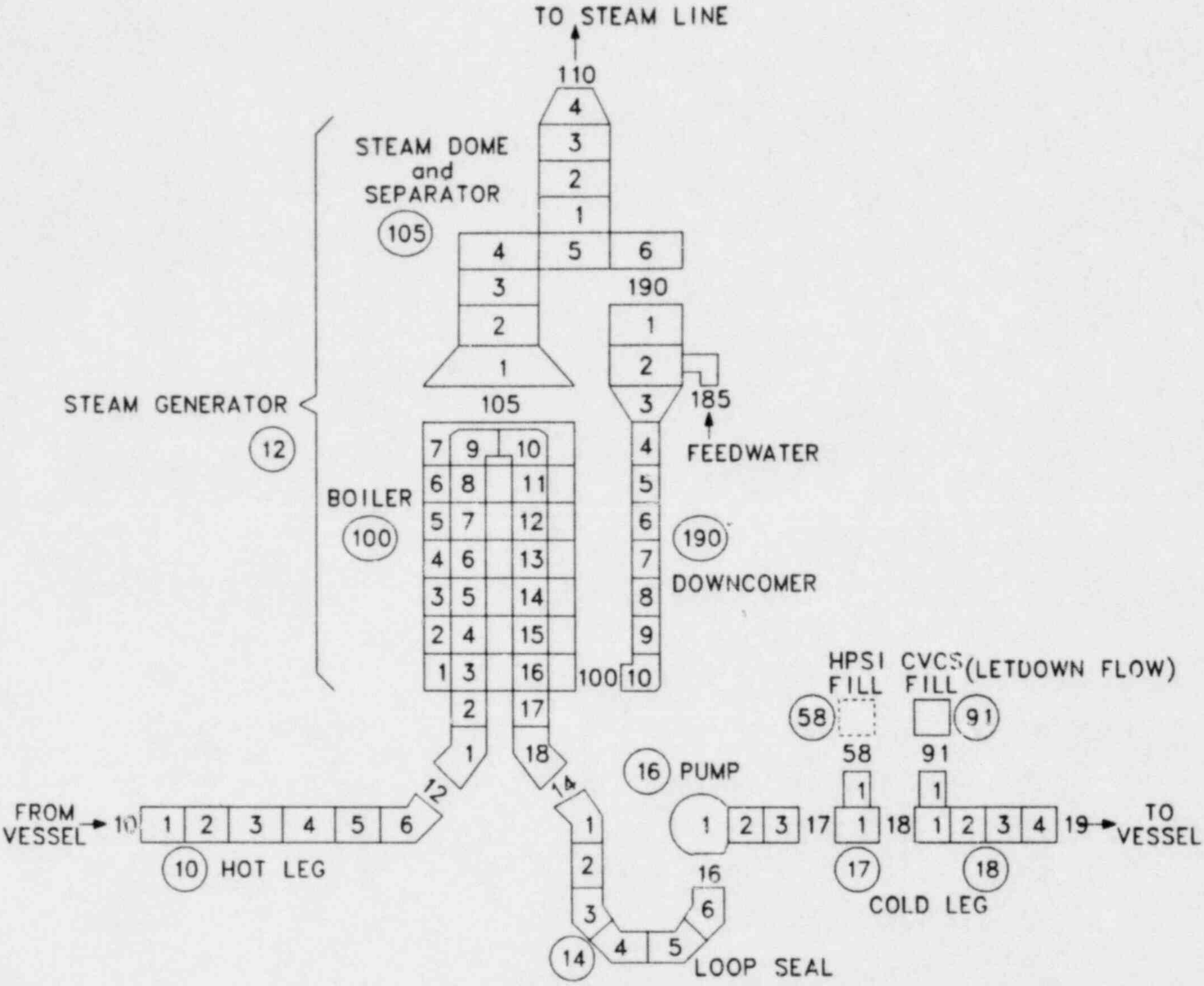


Fig. B.5.
Trac noding of H. B. Robinson-2 Loop 1.

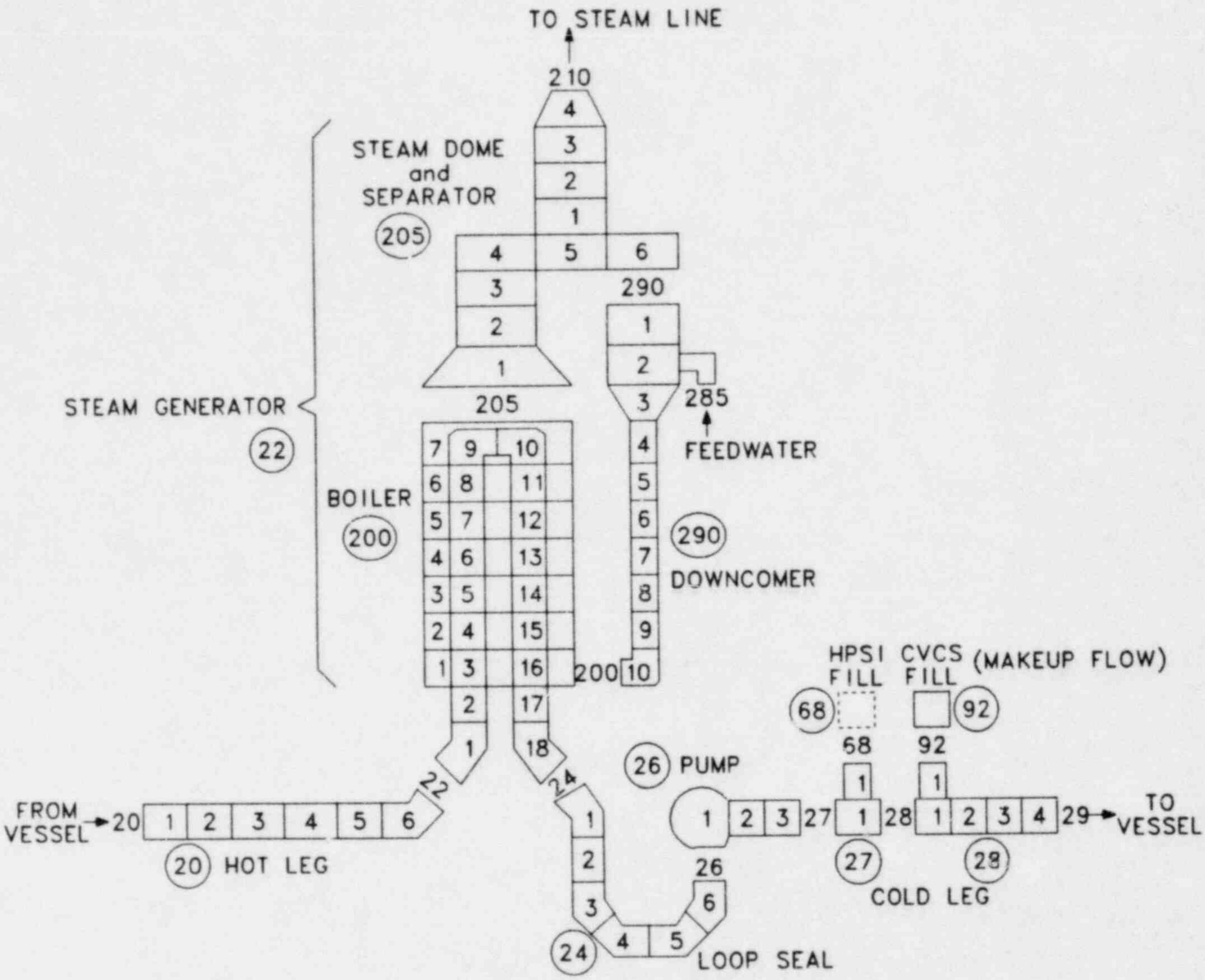


Fig. B.6.
TRAC noding of H. B. Robinson-2 Loop 2.

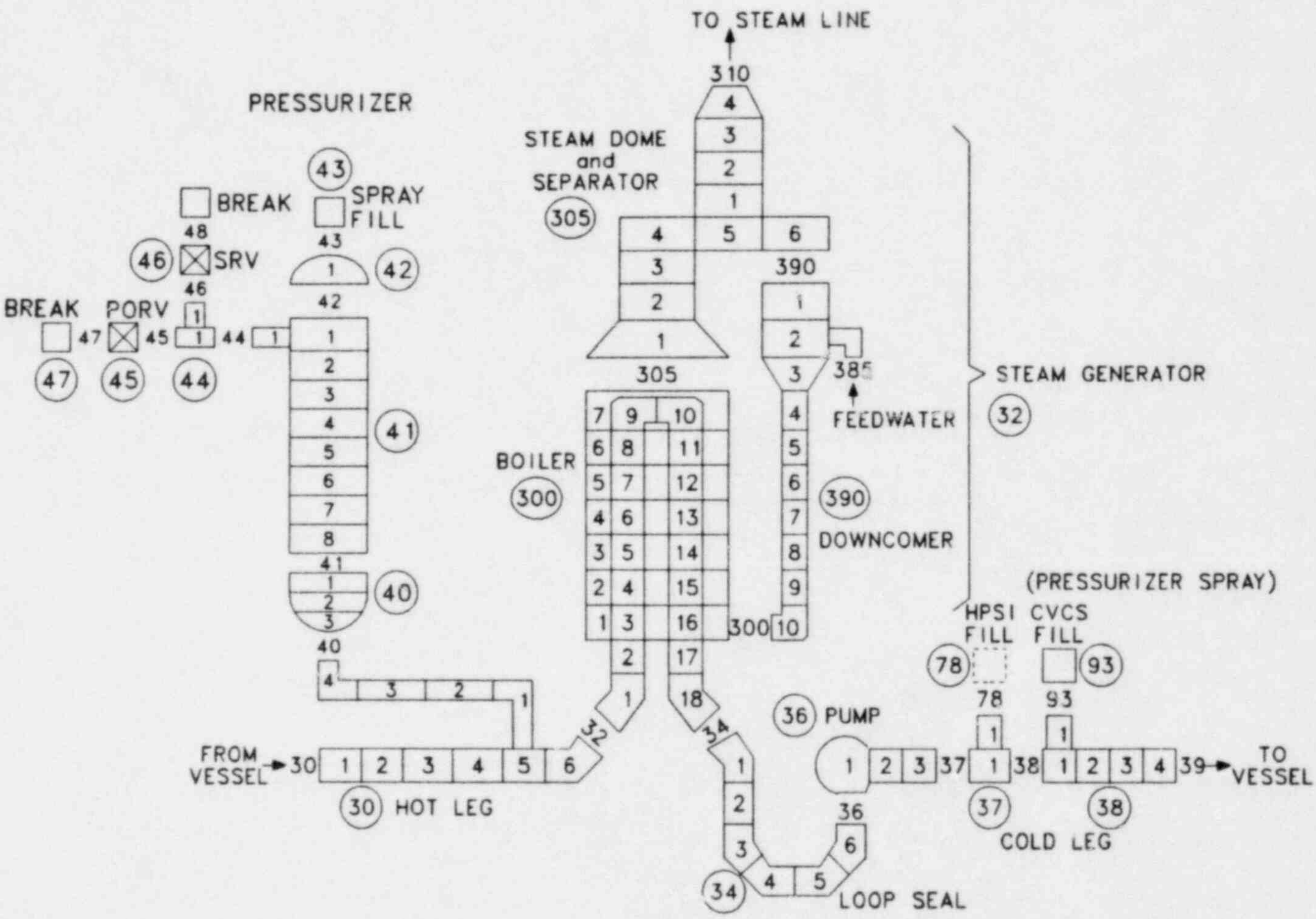


Fig. B.7.
TRAC noding of H. B. Robinson-2 Loop 3.

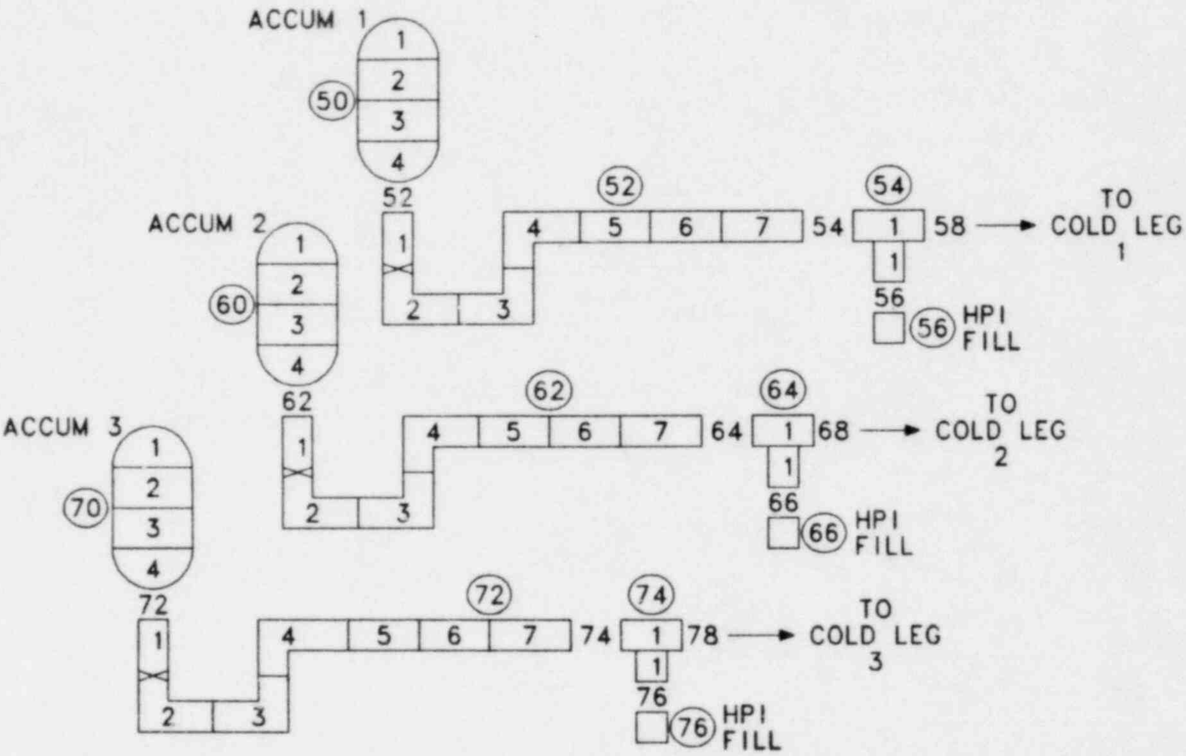


Fig. B.8.
TRAC noding of H. B. Robinson-2 SI system.

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The adequacy of shutdown-decay-heat removal in pressurized water reactors (PWRs) is currently under investigation by the Nuclear Regulatory Commission. One part of this effort is the review of feed-and-bleed procedures that could be used if the normal cooling mode through the steam generators were unavailable. Feed-and-bleed cooling is effected by manually activating the high-pressure injection (HPI) system and opening the power-operated relief valves (PORVs) to release the core decay energy.

The feasibility of the feed-and-bleed concept as a diverse mode of heat removal has been evaluated at the Los Alamos National Laboratory. The TRAC-PF1 code has been used to predict the expected performances of the Oconee-1 and Calvert Cliffs-1 reactors of Babcock and Wilcox and Combustion Engineering, respectively, and the Zion-1 and H. B. Robinson-2 plants of Westinghouse. Feed and bleed was successfully applied in each of the four plants studied, provided it was initiated no later than the time of loss of secondary heat sink.

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