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Summary of Thermal-Hydraulic Calculations for a Pressurized Water Reactor

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SUMMARY OF THERMAL-HYDRAULIC CALCULATIONS FOR A PRESSURIZED WATER REACTOR

by

John W. Bolstad and Roy A. Haarman

ABSTRACT

The results of two transients involving the loss of a steam generator in a single-pass, steam generator, pressurized water reactor have been analyzed using a state-of-the-art, thermal-hydraulic computer code. Computed results include the formation of a steam bubble in the core while the pressurizer is solid. Calculations show that continued injection of high pressure water would have stopped the scenario. These are similar to the happenings at Three Mile Island.

I. INTRODUCTION

One of the important aspects in the reviews of the physical security at nuclear power plants is the identification of targets and areas where the targets are located. The Nuclear Regulatory Commission (NRC) uses the term "vital area" to denote these target locations and defines a vital area as "any area which contains vital equipment within a structure, the walls, roof, and floor of which constitute physical barriers...," and vital equipment "means any equipment, system, device, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation...."[1] Further, a Type I vital area is an area "wherein successful sabotage can be accomplished by compromising or destroying the vital systems or components located within this area."[2]

A systematic approach is used to be certain that all fundamental sabotage-induced scenarios are considered. The approach uses fault trees and a Boolean algebra manipulation computer code that was developed at Sandia Laboratories, Albuquerque (SLA). The Los Alamos Scientific Laboratory (LASL) is applying the code to determine the Type I vital areas of each of the operating nuclear plants in the US.

The steam generator and associated feedwater systems have consistently been assessed as vital equipment associated with Type I vital areas. Most of the utilities with single-pass steam generators have questioned this position and have asserted that the reactor system can be adequately cooled with the high pressure injection (HPI) system operating in conjunction with safety relief valve actuation (the feed-and-bleed method).

However, no mechanistic calculation proving this assertion was provided by these utilities, but instead a simple energy balance was performed on the system HPI water. The calculation shows that if the HPI water could be evaporated in the core and the resulting steam discharged through the safety valves, then all of the decay heat could be removed in this manner a short time after a reactor scram. See the Appendix for a sample calculation of this type.

Another analysis bearing on this subject is given in Ref. 3. In this reference, an analysis of lightwater reactor response to a complete loss of ac/dc power with scram is discussed. This also appears to be a nonmechanistic calculation.

Because no mechanistic calculations were found to prove or disprove the feasibility of feed-and-bleed cooling, we performed many detailed thermal-hydraulic transient analyses to determine the effects of two scenarios resulting from reactor system transients initiated by the loss of the steam generator. These analyses were initiated on September 25, 1978, and completed on January 15, 1979. Thus they were not intended to simulate the Three Mile Island (TMI) incident, and indeed they differ in many respects; however, the similarities between the incidents analyzed here and the TMI incident are obvious.

A. General Description of Transients Analyzed

Both of the accident scenarios analyzed involve the loss of steam generators, although the second transient involves primarily a Loss of Coolant Accident (LOCA) condition. For the first transient analyzed we assumed the loss of all ac power coincident with the loss of the steam generator auxiliary feedwater system. We further assumed that after 10 min, the HPI pumps are available to inject into the primary system. Results were obtained for both the cases of one or two HPI pumps available. The intent of this analysis was to determine whether the reactor could be cooled by means of the HPI water and safety relief valve operation (the feed-and-bleed method) and, furthermore, how many HPI pumps would be necessary to provide sufficient cooling.

For the second transient analyzed we assumed the loss of all ac power coincident with the the loss of the steam generators. In this case we also assumed that the saboteur intentionally held open the electromagnetic relief valve on the pressurizer to create a LOCA condition. This valve was assumed to remain open for the duration of the transient. The intent of this analysis was to determine the period of time available to respond to the incident by turning on the HPI pumps. A parametric study was conducted assuming varying periods of time before HPI pump actuation.

B. Thermal-Hydraulic Reactor Transient Model

The basic tool used for the transient analyses was the computer code TRAC (Transient Reactor Analysis Code) developed at LASL. This code is a state-of-the-art, best-estimate code for accident analysis in pressurized water reactors (PWRs). The code features a three-dimensional (r,θ,z) treatment of the volume inside the pressure vessel with a two-phase nonequilibrium hydrodynamic model. The remaining components use a one-dimensional drift flux model to describe the thermal hydraulics. Details of the code as well as comparisons of calculative vs experimental results are given in Refs. 4-5.

C. Physical Model of the System

We developed a model of a PWR system to be used as a basis for the calculations described here. The resulting computational model is shown in Fig. 1. This model lumps the loop hot legs, cold legs, steam generators, and pumps into respective equivalent components. In deriving this model, the following quantities were preserved:

- fluid velocity through all components,
- elevation of components,
- fluid path length through components,
- pressure profile through all components, and
- stored heat in pipe walls.

The mass flow rate at any point in the model equals the total of the flow rates that exist in like components for the physical system.

The reactor vessel and internals are modeled in (r,θ,z) geometry using 24 cells. There are two azimuthal regions and two radial regions. There are

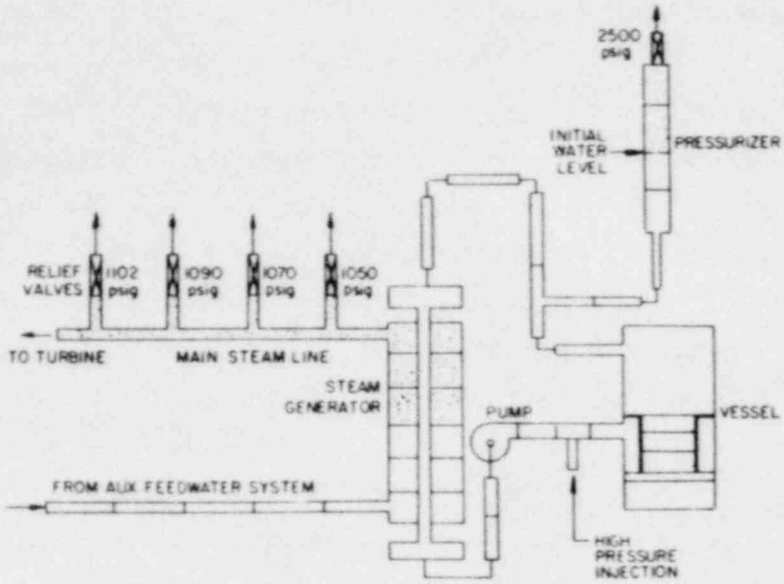

Fig. 1.
Reference noding diagram.

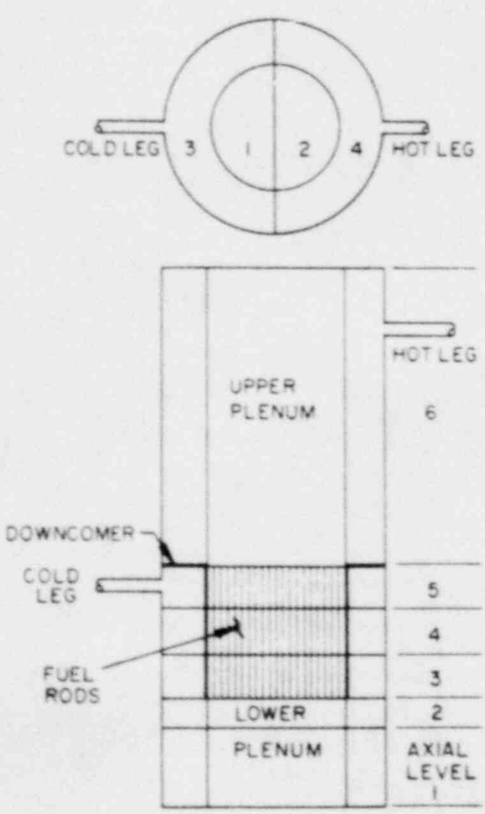

Fig. 2.
Details of the vessel nodes.

three axial regions in the core, two below the core (lower plenum), and one above the core (upper plenum). The detailed vessel nodes are shown in Fig. 2. The fuel rods are modeled using 9 radial nodes: six in the fuel and three in the cladding.

The pressurizer is modeled using four cells with the vapor-liquid interface between cells 2 and 3. The pressurizer heaters and sprayers were not modeled because details on the controller characteristics were not available.

The valve shown above the pressurizer in Fig. 1 is used to represent either the two pressurizer code safety valves for the feed-and-bleed transient or the pressurizer electromagnetic relief valve for the relief valve transient. The valves are modeled such that they pass the design capacity of saturated steam at the rated pressure. The capacities at other pressures and fluid conditions are calculated automatically by the code. Additional details on the design capacities of these valves are shown in Table I.

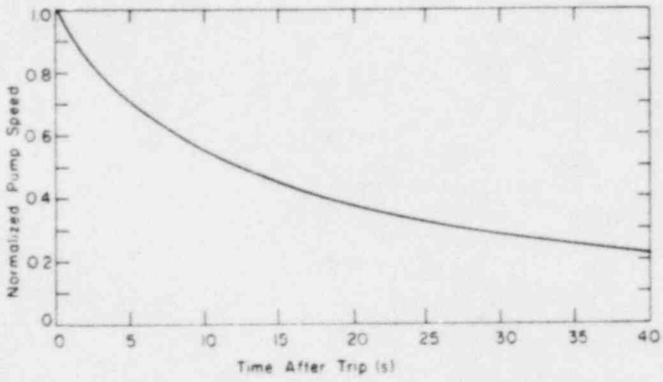

Fig. 3.
Main coolant pump coast-down characteristics.

The primary loop circulating pump shown in Fig. 1 is modeled to represent four pumps of the physical system. Thus the pump head is typical of any of the pumps in the physical system, whereas the pump mass flow rate represents the total mass flow rate through all the pumps. When a circulating pump trip occurs, the pump speed is assumed to coast down according to Fig. 3. The actual loop flow rate is calculated as a function of pump speed and head. The homologous head curves were input to provide the head-flow-speed relationship.

The HPI system injects into the cold legs at the location shown on Fig. 1. The HPI pump characteristic curve is one of the most important and sensitive parameters for the analyses performed here. The pump characteristic curve is

TABLE I

SAFETY AND RELIEF VALVE SETPOINTS AND RATED CAPACITIES

Valve Identification	Pressure (10^5 Pa)	Pressure (psig)	Rated Flow (Kg/s)	Rated Flow (10^6 lb/hr)
Pressurizer Safety Valve	173.4	2500.0	173.9	1.38
Pressurizer Electromagnetic Relief Valve	156.5	2255.0	14.1	0.112
Steam Generator Secondary Bank 1	73.4	1050.0	425.9	3.38
Steam Generator Secondary Bank 2	74.8	1070.0	425.9	3.38
Steam Generator Secondary Bank 3	76.2	1090.0	425.9	3.38
Steam Generator Secondary Bank 4	77.0	1102.5	425.9	3.38

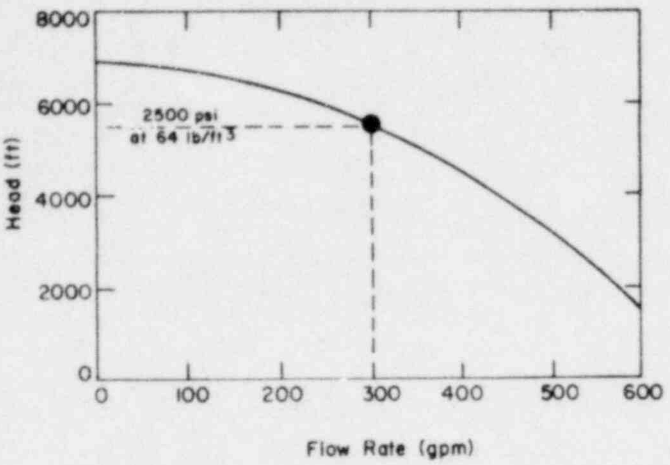

Fig. 4.
High-pressure injection pump characteristic curve.

shown on Fig. 4. There are actually three HPI pumps installed, and each of these has the capacity shown on Fig. 4. A distinguishing characteristic of this pump is that it is able to pump 0.019 m^3/s (300 gpm) at a developed head of 1.682×10^4 J/kg (5620 ft) (corresponding to a ΔP of 172.4×10^5 Pa (2500 psi) at a density of 1025 kg/m^3 (64 lb/ft^3)).

The steam generator modeled here represents two steam generators of the physical system. It is a once-through, vertical, straight-tube, tube-and-shell heat exchanger that produces superheated steam at the outlet. A sensitivity study has shown that the detailed model of the steam generator feedwater system and relief valves is most important for correct prediction of the primary system transient for the first few minutes after the loss of the steam generator. It is important to model correctly the complex transient interaction between the primary and secondary sides caused by the tube wall heat capacity and heat transfer in all regimes from forced convection to single-phase liquid through forced convection to superheated vapor. We have confidence in this model because a steady-state calculation yields the correct outlet steam temperature (superheated), boiling water height, and water inventor

The pipe leading to the steam generator secondary inlet consists of approximately 30.5 m (100 ft) of 6-in., schedule 120 pipe. This pipe is modeled because the feed-and-bleed analysis will assume a rupture in this line and the steam generator secondary side will blow down through this pipe.

A steam relief system is installed on each steam generator steam line to provide for heat removal and steam relief during periods when the main heat sinks are not available. Such a period occurs during a turbine trip on loss of vacuum or loss of electrical power to station auxiliaries. This system prevents operation of the steam generator safety valves during normal operating transients. We found it necessary to simulate quite closely this relief valve manifold because the transient results (for the first few minutes after the loss of the steam generator) were very sensitive to the behavior of the steam generator secondary side.

The model of the steam generator secondary side is shown on Fig. 1. The plant has a total of 18 steam relief valves (9 on each steam line); the 9 valves on each line are arranged into 4 banks, each with a different relief pressure setting. One valve in the model represents all of the relief valves at a given relief setting; therefore, four valves are shown on Fig. 1. All of the valves are modeled such that they pass the design capacity at their rated pressure. The code calculates the appropriate flow rates at higher pressures; at pressures below the relief pressure they will close. Table I summarizes the design flow rate and relief pressure setting for the steam generator relief valves as well as the pressurizer electromagnetic relief valve and pressurizer safety valves. The quantities in the table represent the total of the flow rates of all identical valves when a number of valves are represented by a single valve in the model. The transient results are found to be quite sensitive to the specific relief pressure setting and design flow rate.

The system model developed here has been developed under the criterion of minimum complexity (one loop analysis) while retaining the necessary features to allow an accurate prediction of the thermal-hydraulic effects present in the transients reported here.

II. ANALYSIS OF THE FEED-AND-BLEED TRANSIENT

A. Scenario for the Feed-and-Bleed Transient

The accident scenario for this transient was defined by the event tree analysis along with other conditions commonly assumed in vital area and sabotage analysis. The accident scenario we have analyzed is as follows.

1. Loss of all ac power.

   The loss of all ac power results in a turbine trip, loss of condenser, loss of steam generator feedwater, reactor trip, and trip of the primary loop circulating pumps.

2. Rupture of auxiliary steam generator feedwater system.

   This postulated assumption stems from the event tree analysis. We assume that the auxiliary feedwater line is severed approximately 30.5 m (100 ft) from the steam generator. The steam generator is allowed to blow down through this line.

3. Later availability of HPI pumps.

   Ten minutes after initiation of the transient, one or two HPI pumps are available to inject into the primary system.

Assumption 1 alone, loss of all ac power, is analyzed in the Safety Analysis Report for the facility. The reactor system is designed to handle this incident with no adverse effects. Assumption 1 along with assumption 2 implies the loss of the steam generators as the reactor system heat sink. An analysis of this situation is not covered in the Safety Analysis Report and therefore this analysis was undertaken. Assumption 3 is reasonable, but the time for HPI initiation is arbitrary. This time was proposed by others in defining the assumptions for this scenario. The following calculations show that manual HPI initiation must be assumed because, for this scenario, automatic initiation of the HPI system is not obtained by a low reactor pressure trip.

Following loss of all power, the main steam stop valves will close to protect the condenser. The steam generators will provide a heat sink for some time because of boiling of the secondary side coolant in them. This generated steam will exhaust to the atmosphere through the steam relief valves. At the same time, secondary side coolant will be lost through the rupture in the steam generator auxiliary feedwater line. The heat sink effect of the steam generators will be lost when they dry out. The primary coolant will then begin to heat up because of the reactor core decay heat. As the coolant heats up, it will expand, compress the pressurizer steam bubble, and raise the primary system pressure. When the HPI system is initiated, the pressurization rate will increase because of the increased water inventory. Eventually, the pressurizer safety valves will alternately open and reseat based on the system pressure changes because of the competing effects of water expansion (because of the sources and heating) and discharge through the safety valves. If sufficient water could be injected and penetrate the core, vaporize, and be discharged through the pressurizer safety valve as vapor, then the reactor could be cooled by the feed-and-bleed method as discussed in the appendix.
The question we are addressing here is whether the feed-and-bleed concept will mechanistically function as described previously and, if so, how much HPI water would be required to ensure its success.

B. Results for the Feed-and-Bleed Transient

1. Early Time Behavior (first 10 min)

During the first few minutes after initiation of the transient, the events occurring on the steam generator secondary side strongly influence the primary side response. The steam generator secondary side pressure is shown on Fig. 5. Within 4 s of the transient initiation, the steam pressure on the secondary side climbs from 63.7 x 10^5 Pa (910 psig) to almost 76.2 x 10^5 Pa (1090 psig). At this time the 1050 psig and 1070 psig relief valve banks are exhausting steam. At approximately 34 s, the 1070 psig relief valve bank reseats; the 1050 psig valves continue to relieve steam until 67 s. At this time all relief valves close and further pressure decay is due to the auxiliary feedwater line rupture.

The primary system pressure response is shown in Fig. 6. The pressurizer pressure rapidly climbs from 151.7 x 10^5 Pa (2185 psig) to 173.4 x 10^5 Pa (2500 psig) in 4 s. At this point the pressurizer safety valve lifts and maintains this pressure until 20 s. At this time the safety valve reseats and the primary system pressure decreases because of heat transfer to the steam generator (up to 45 s). After 45 s, heat transfer in the steam generator degrades, and the pressure in the primary system quickly increases once more to the safety valve setpoint. The long-term primary system pressure response will be discussed later. While the pressurizer safety valves are open, the pressure in

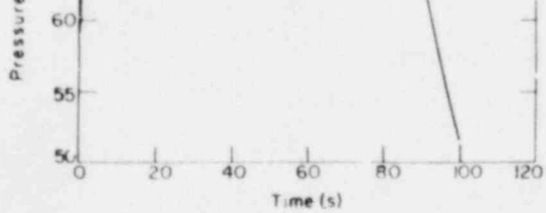

Fig. 5.
Short-time behavior of the steam generator secondary side pressure.

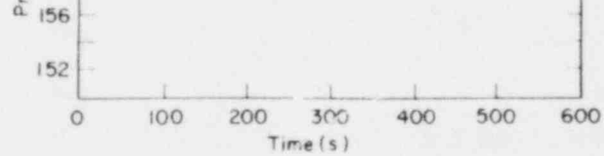

Fig. 6.
Pressurizer pressure.

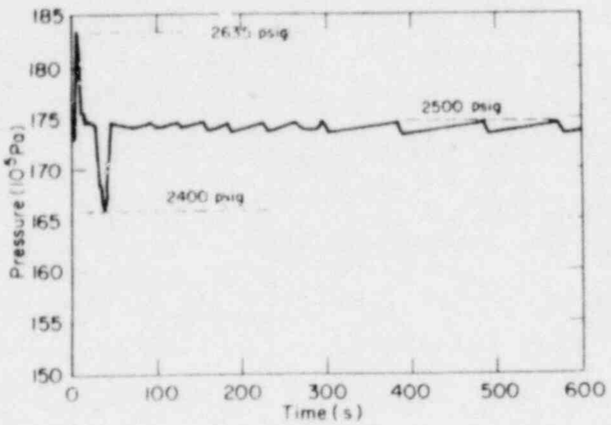

Fig. 7.
Vessel upper plenum pressure.

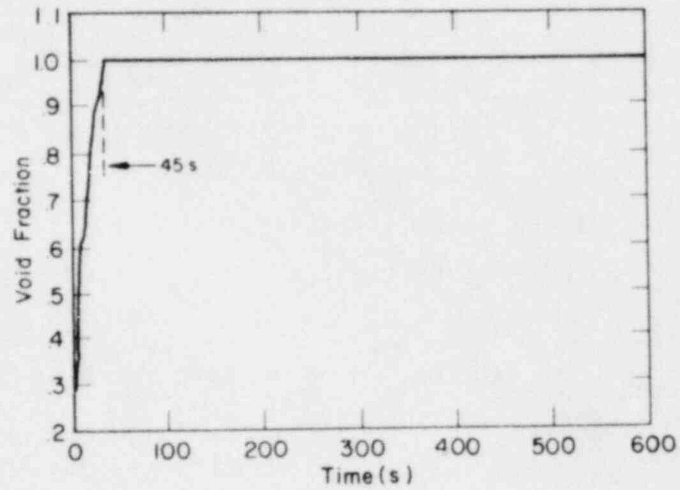

Fig. 8.
Steam generator secondary side void fraction.

the reactor vessel rises higher than that in the pressurizer and peaks out at approximately 182.7 x 10^5 Pa (2635 psig) at 7 s. This pressure response is shown in Fig. 7.

Figure 8 shows that the steam generator boils dry at approximately 45 s. There is minimal heat transfer to the steam generator after this time. As shown in Fig. 9, the steam generator secondary side blows down to atmospheric pressure in about 5 min. This blowdown is caused by the rupture of the steam generator auxiliary feedwater line and is shown on Fig. 10.

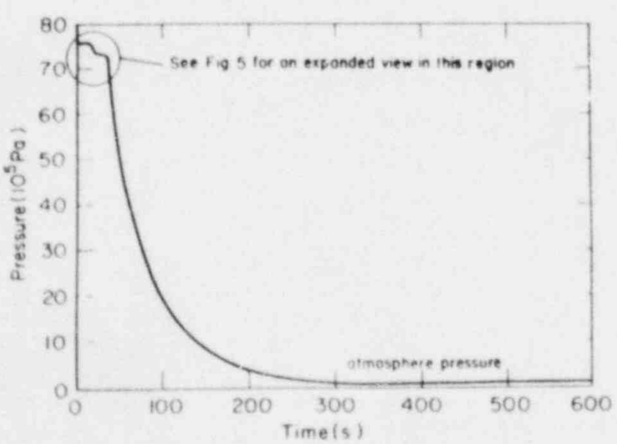

Fig. 9.
Steam generator secondary side pressure.

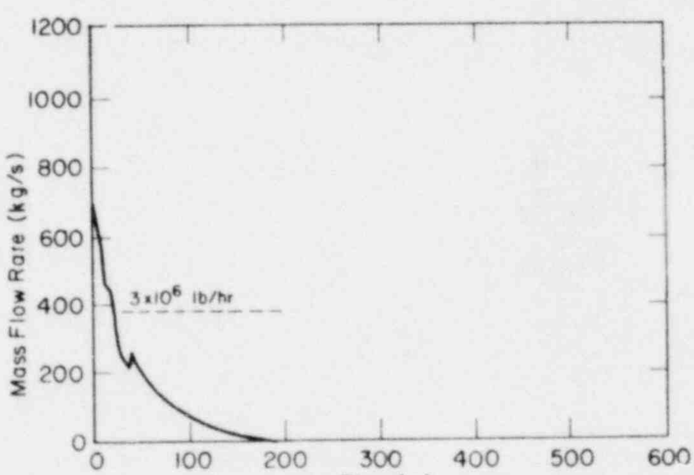

Fig. 10.
Auxiliary feedwater line mass flow rate.

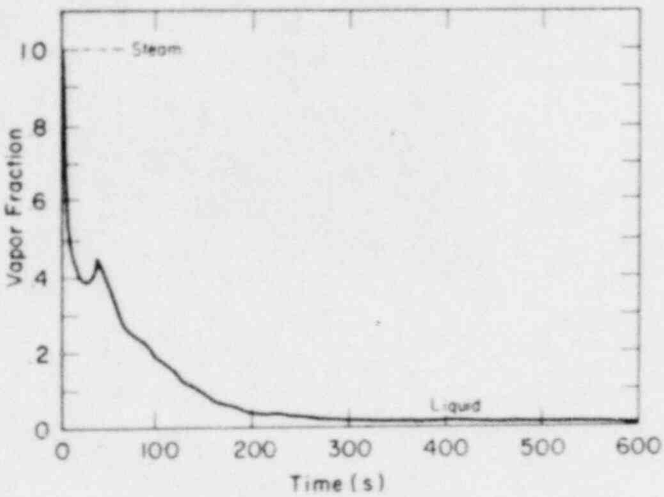

Fig. 11.
Pressurizer void fraction.

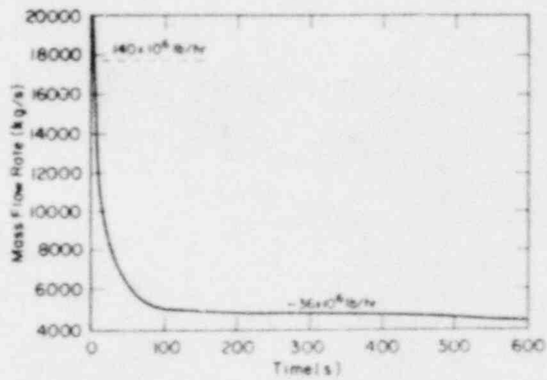

Fig. 12.
Core inlet mass flow rate.

After approximately 50 s, the pressurizer safety valves alternately open and reseat, exhausting mass from the top of the pressurizer. After about 5 min, the steam inventory in the pressurizer is exhausted as shown on Fig. 11. After this time, the pressurizer inventory and safety valve discharge is liquid rather than vapor.

During this time, the core mass flow rate is decreasing because of the main circulating pump trip and coast-down. However, natural circulation is established, and there is sufficient coolant available to remove the decay heat from the core. The flow coast-down of the primary coolant loop is shown on Fig. 12, and the heat-up of the primary coolant is shown on Fig. 13.

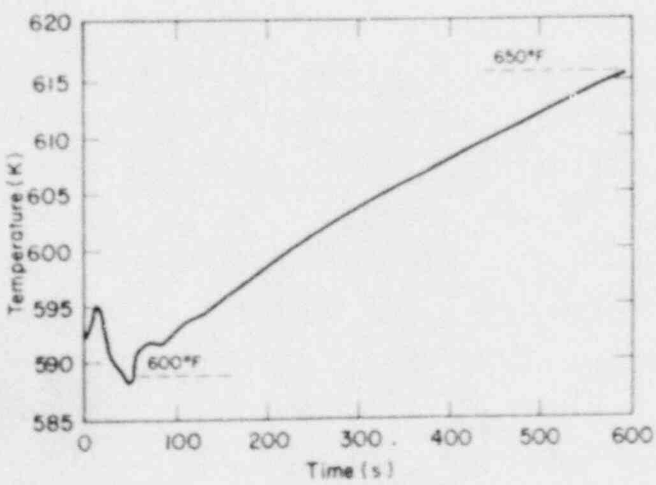

Fig. 13.
Core outlet temperature.

We now turn to a discussion of the long-term behavior of the system. As discussed earlier, the automatic low reactor pressure HPI trip (set at 111.3×10^5 Pa (1600 psig)) is not automatically set during this transient. (Fig. 6 shows

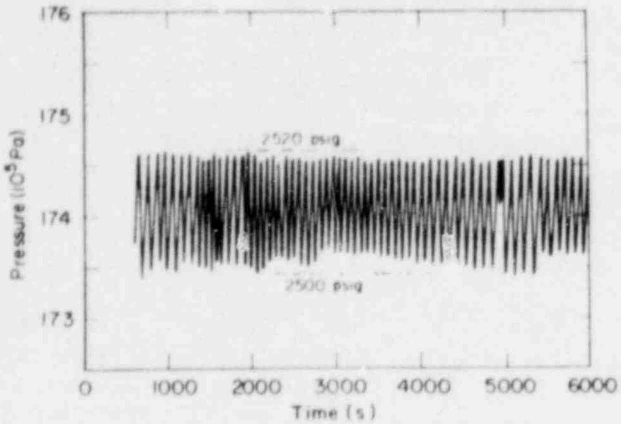

Fig. 14.
Upper plenum pressure.

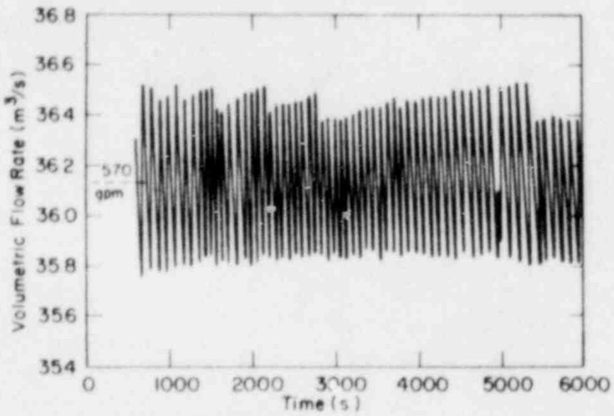

Fig. 15.
High-pressure injection flow rate.

that the minimum pressure experienced is above the trip set pressure.) The transient has been analyzed assuming that the HPI pumps are manually started at 10 min. Two cases have been analyzed:

- both HPI pumps available for service, and
- only one HPI pump available for service.

2. Long Term Behavior with Two HPI Pumps Operating

With the initiation of the HPI pumps, the safety valves open and reseat on a periodic basis. This process affects the primary system pressure as shown on Fig. 14. This change in the system pressure results in a variable HPI flow rate as shown in Fig. 15. This flow rate is a function of the system pressure according to the pump characteristic curve, Fig. 4.

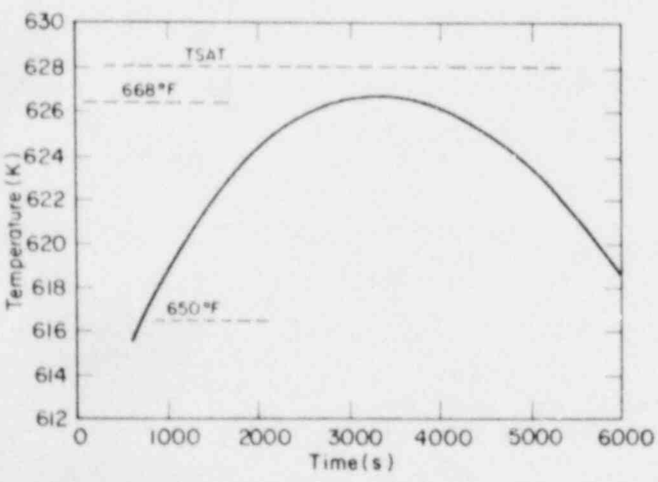

Fig. 16.
Core outlet temperature.

The continual injection of the cold HPI water and its circulation into the primary coolant loop results in the eventual cooling of the primary water being circulated. Approximately 1 h after transient initiation, the temperature excursion is turned around, and continual cooling of the primary system water takes place as shown on Fig. 16.

Fine details of this feed-and-bleed process are shown on Figs. 17 and 18. Figure 17 shows the pressurizer pressure responding to the safety valve action shown on Fig. 18 and the HPI coolant source shown on Fig. 15.

This calculation represents a successful application of the feed-and-bleed concept; however, the reader should note that the system did not perform in the normally defined manner. That is, instead of steam being produced in the core and released through the safety valves, the pressurizer went solid, and liquid was discharged rather than vapor.

3. Long Term Behavior with One HPI Pump Operating

With two HPI pumps operating, the primary coolant temperature stayed below the saturation temperature corresponding to the system pressure. With only one HPI pump operating, a completely different system response is calculated. With only one HPI pump operating, the primary coolant temperature reaches saturation temperature at 20 min as shown on Fig. 19. At this time, vapor generation starts in the core, and a steam bubble begins to form. The bubble grows, and the core midplane is uncovered at approximately 1 h. The rapid growth of the steam bubble is shown on Fig. 20.

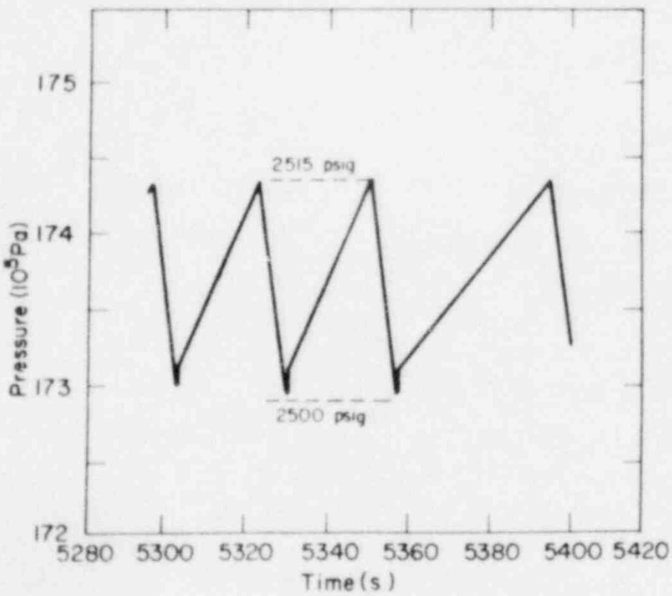

Fig. 17.
Pressurizer pressure.

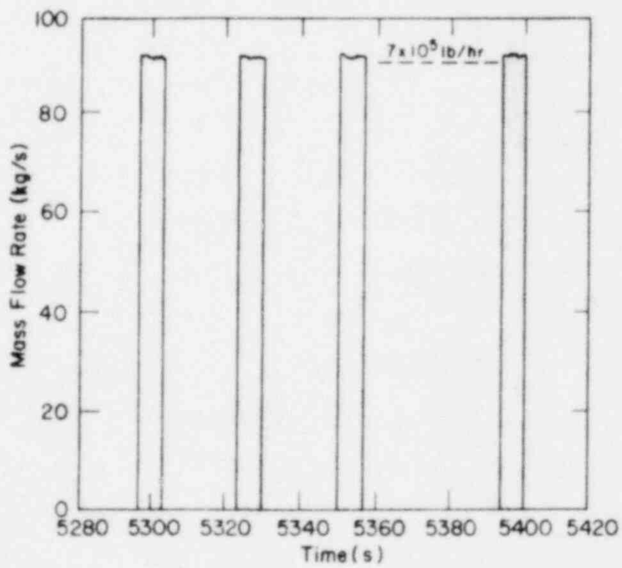

Fig. 18.
Pressurizer safety valve mass flow rate.

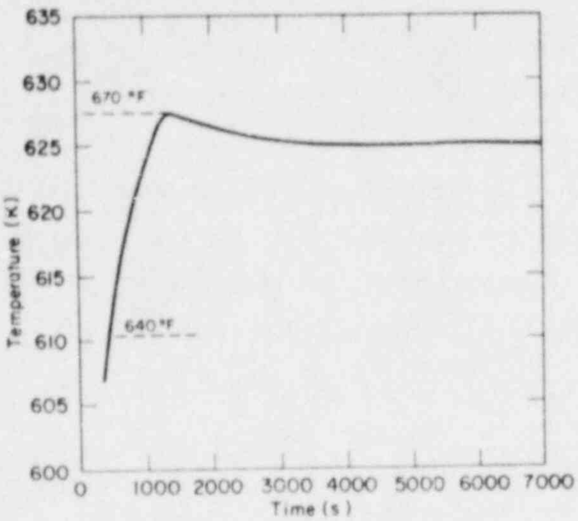

Fig. 19.
Core outlet temperature.

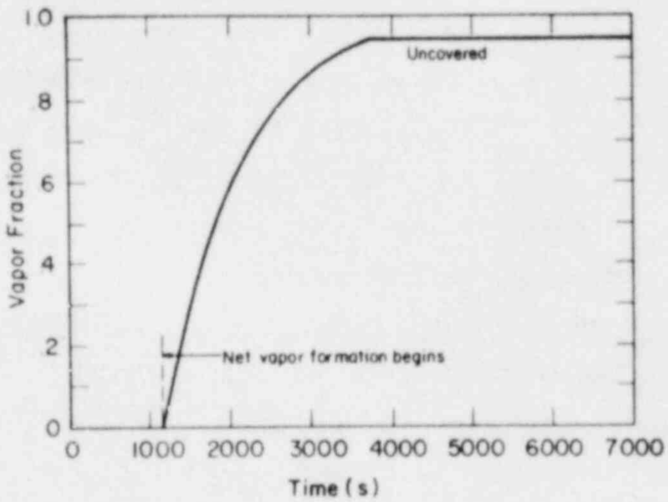

Fig. 20.
Core void fraction.

The vapor bubble expands into the upper plenum and the loop hot legs pushing the reactor coolant water ahead of it. During the time interval 20-25 min, a large amount of coolant inventory is discharged through the safety valves. During th· time interval 20 min to 1 h, so much primary coolant water is lost that there appears to be no chance to recover the core in a reasonable amount of time. This effect is shown in Fig. 21 in which the total vessel liquid inventory is plot⁺⁺·d.

III. ANALYSIS OF THE RELIEF VALVE TRANSIENT

A. Scenario for the Relief Valve Transient

The accident scenario for this transient was defined by the event tree analysis along with other conditions commonly assumed in vital area and sabotage analyses. The accident scenarios we have analyzed are as follows.

1. Loss of all ac power.
   The loss of all ac power results in a turbine trip, loss of condenser, loss of steam generator feedwater, reactor trip, and trip of the primary loop circulating pump.
2. No auxiliary feedwater available.
   For this case we have not assumed a rupture of the steam generator auxiliary feedwater line. For this transient, the less severe assumption of no auxiliary feedwater is used; this could be brought about by any one of a number of events.

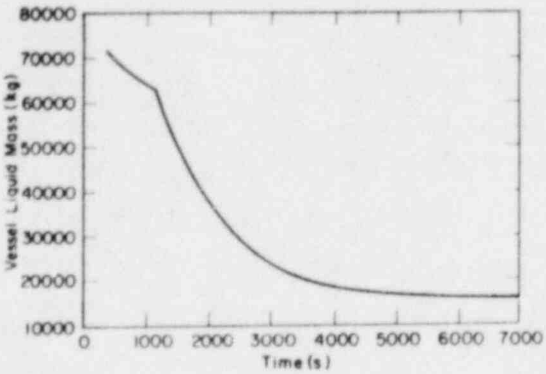

Fig. 21.
Total mass contained in reactor vessel.

3. Later HPI pump actuation.

At some time after transient initiation, both high pressure injection pumps are actuated.

Following loss of all power, the isolation valves on the steam lines will close to protect the condenser. The steam generators will provide a heat sink for some time (≈ 5 min) because of boiling of the secondary side coolant in them. This generated steam will exhaust to the atmosphere through the steam relief valves. The heat sink effect of the steam generators will be lost when they dry out. The primary system pressurization rate will increase as the primary coolant heats up.

Meanwhile the pressure relief valve is exhausting steam from the top of the pressurizer. Primary system water inventory is being lost, and the remaining coolant is heating up. When the saturation temperature is reached, a steam bubble starts forming in the core and expands. The fuel cladding temperature heat-up rate increases.

At some later time, we assume a manual initiation of the HPI pumps. If this HPI water can penetrate the core, the temperature excursion could be turned around and the fuel rods quenched. The first question we are addressing here is whether the core can be quenched before excessive fuel damage is encountered.

We want to determine whether the core can be quenched by prompt initiation of the HPI system and how soon after the transient initiation the HPI pumps must be actuated to ensure negligible fuel damage.

B. Results for the Electromagnetic Relief Valve Transient

During the first few minutes after initiation of the transient, the primary side pressures are strongly influenced by the events occurring on the steam generator secondary side. The steam generator secondary side pressure is shown on Fig. 22. Within 4 s of the transient initiation, the steam pressure on the

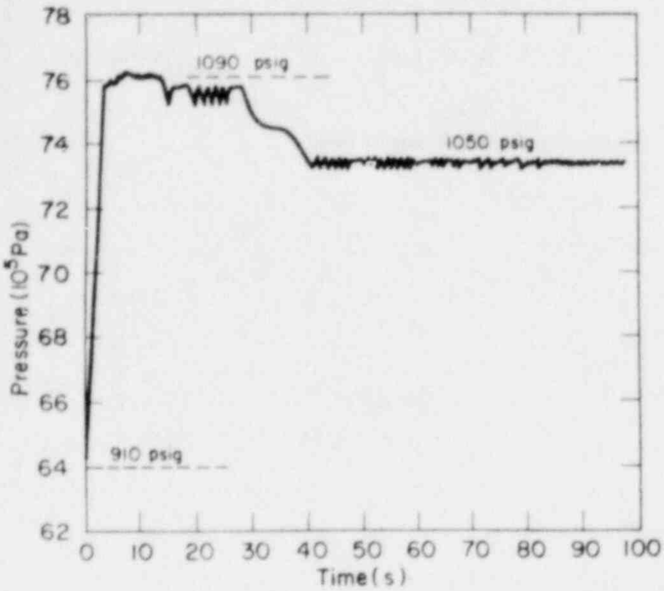

Fig. 22.
Steam generator secondary side pressure.

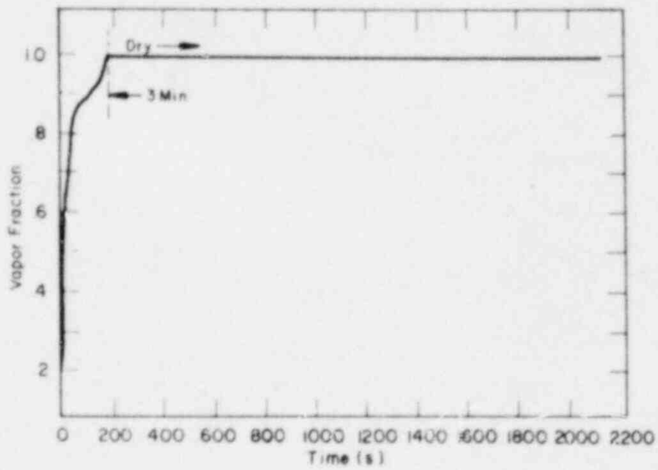

Fig. 23.
Steam generator secondary side void fraction.

secondary side climbs from 63.7 x 10^5 Pa (910 psig) to 76.2 x 10^5 Pa (1090 psig). At this time, the 1050-, 1070-, and 1090-psig relief valves are exhausting steam. At 12 s, the 1090-psig relief valve bank reseats, and at 28 s the 1070-psig relief valve bank reseats. From this point on, the 1050-psig relief valve bank maintains the steam generator secondary side pressure at approximately 1050 psig. The steam generator boils dry in approximately 3 min as shown on Fig. 23.

On the primary side, the pressurizer electromagnetic relief valve is discharging mass as shown on Fig. 24. The variation in the mass flow rate shown on the figure is due to the changing pressurizer pressure and void fraction. The figure shows that after the pressurizer goes solid (approximately 10 min), the relief valve discharges about twice its rated capacity. This is because the pressurizer is discharging liquid rather than saturated steam. The pressurizer void fraction is shown on Fig. 25. The figure indicates that the pressurizer goes solid at about 10 min. Of course, this is due to the discharge of steam and two-phase mixture through the pressurizer relief valve and the subsequent replacement of this mixture with relatively cold water from the hot leg.

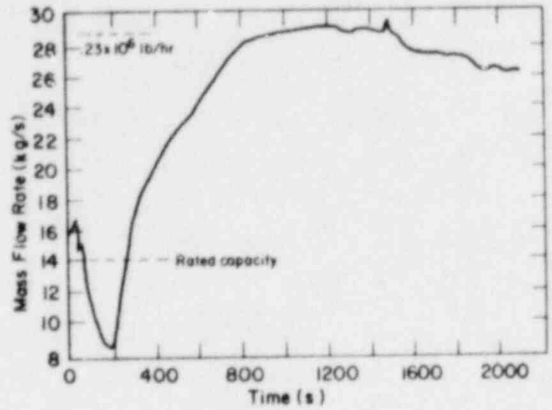

Fig. 24.
Mass flow rate through the pressurizer electromagnetic relief valve.

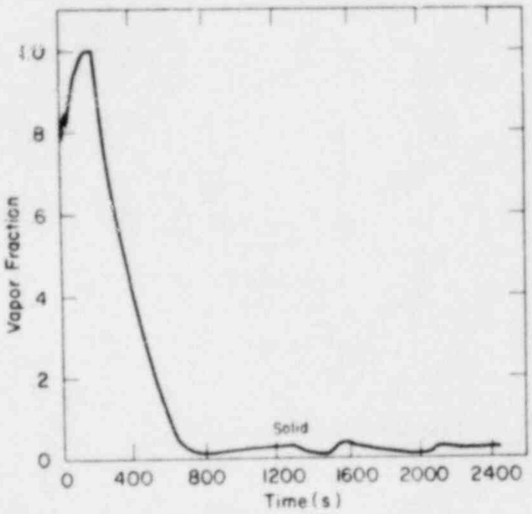

Fig. 25.
Pressurizer void fraction.

The results discussed above are relatively insensitive to the actual time when the HPI pumps are manually activated. To examine more closely the physical events that occur during this transient, it will be necessary to become more specific. We now will discuss results specific to one particular case typical of others examined. Specifically, we will now consider the physical events occurring if the HPI pumps are activated at 20 min after the transient initiation.

The primary coolant temperature is increasing after initiation of the transient because of the decay heat source. Ten minutes after the transient starts, the core starts to form a steam bubble as shown in Fig. 26. The bubble increases in size, and the core centerline is uncovered at 23 min., although HPI was initiated at 20 min. Thus we conclude that the core centerline may become uncovered if HPI injection is not initiated before 20 min. This situation does not imply core damage, however; the fuel rod heat-up will be discussed later.

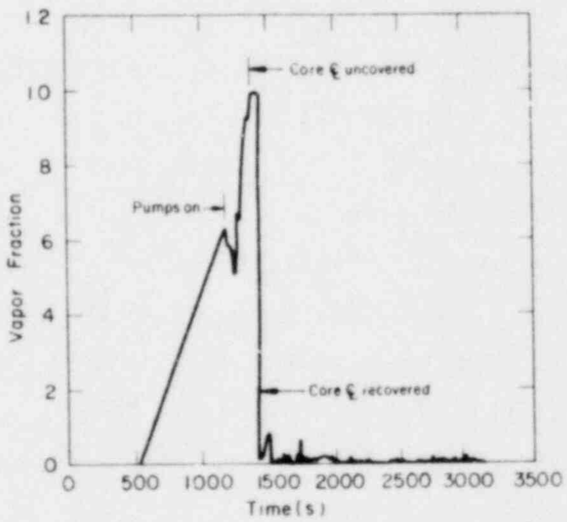

Fig. 26.
Vapor fraction at the core centerline for HPI initiation at 20 min.

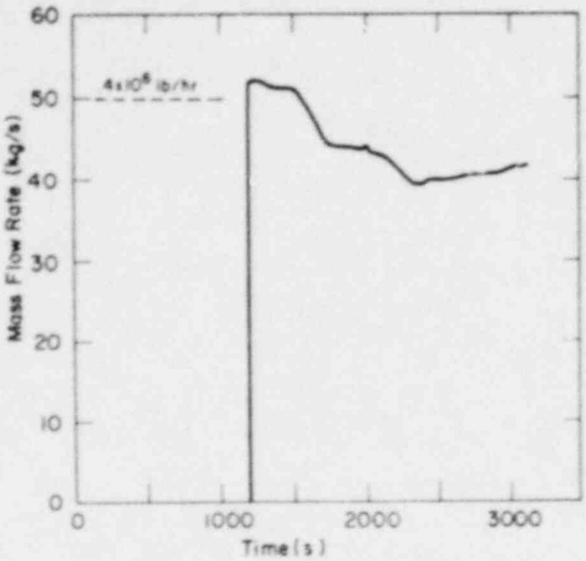

Fig. 27.
High-pressure injection mass flow rate for HPI initiation at 20 min.

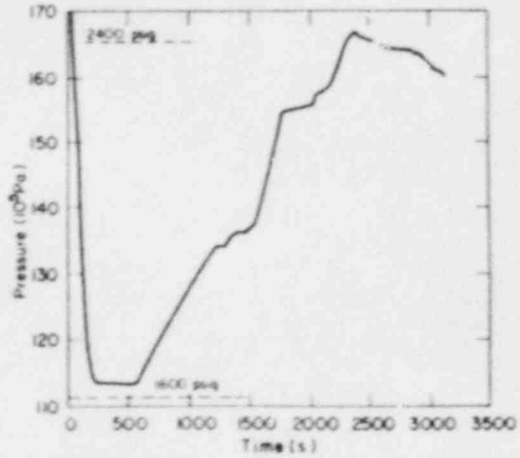

Fig. 28.
Primary system pressure for HPI injection at 20 min.

The HPI mass flow rate is shown on Fig. 27. A comparison of this flow rate with the mass flow rate being discharged through the relief valve (Fig. 24) shows that the injection flow rate is greater than that being lost through the relief valve. Thus we would expect that the core may be recovered. Indeed, Fig. 26 shows that the core centerline is recovered at 25 min. Thus, there is a time delay of 5 min from HPI initiation to core recovery; this delay time is expected to be greater if HPI is delayed longer.

The primary system pressure response to these events is shown on Fig. 28. Initially, the system pressure spikes to just below the pressurizer safety valve setpoint (173.4 x 10^5 Pa (2500 psig)). The pressure then rapidly decreases for 3 min until the steam generator boils dry. The pressure then remains relatively constant until the core starts voiding at 10 min. The pressure then rapidly increases as the steam bubble grows. The pressure history after HPI initiation is governed by steam generation or condensation, mass injection into the system, and the relief valve discharge rate. The net result of these competing factors is summed up by the amount of liquid in the vessel; this is shown on Fig. 29. The figure indicates the decrease in mass because of liquid heatup and expansion up to 10 min, the decrease in mass because of the steam bubble expansion from 10-20 min, and the subsequent recovery of the core.

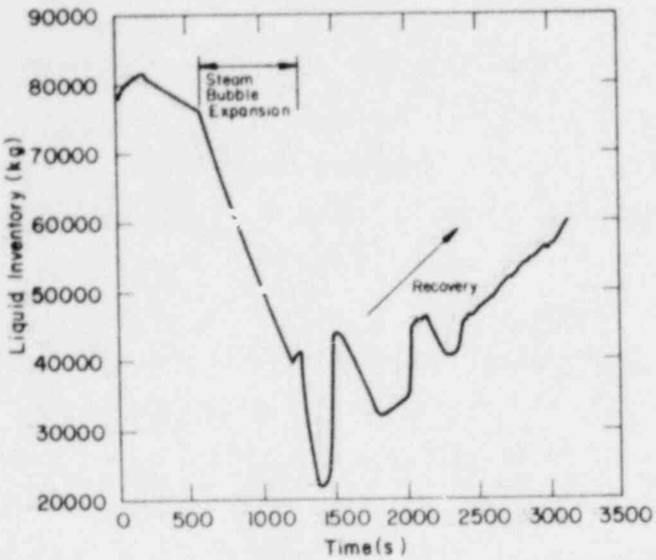

Fig. 29.
Total vessel liquid inventory for HPI injection at 20 min.

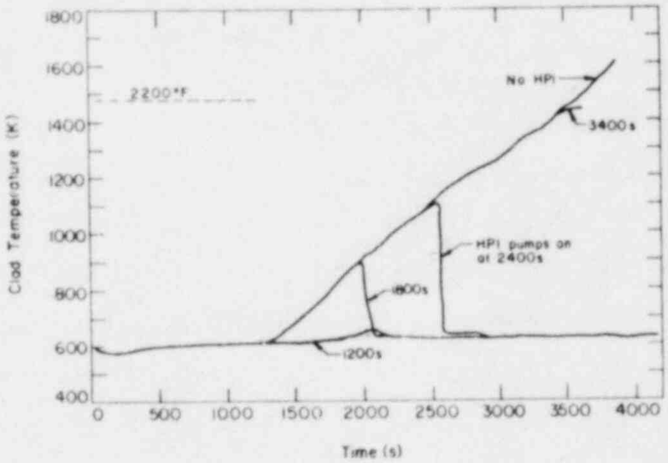

Fig. 30.
Hot spot cladding temperature for different time delays before HPI injection for the relief valve transient.

One other point is worthy of discussion in light of the Three Mile Island accident. That is, the pressurizer is solid liquid from 10 min on (Fig. 25), and at the same time the core may be partially or completely voided. This is a common conclusion to all of the relief valve transients we have examined.

We have determined that the core may become uncovered relatively soon after initiation of this transient, but we have not concluded that core damage results. A number of cases have been studied with HPI injection delayed up to 56.7 min after transient initiation. These results are summarized on Fig. 30 where the core hot-spot clad temperature is shown as a function of time after transient with HPI injection starting at 20, 30, 40, and 56.7 min after loss of the steam generator. The figure indicates that the transient can be turned around even if actuation of the HPI system is delayed for an extended time.

IV. SUMMARY AND CONCLUSIONS

We have analyzed two transients defined by a computer-based event tree to determine the Type I vital areas of nuclear power plants. These analyses were performed with the most applicable state-of-the-art code available at the time and a fairly detailed plant model. Both transients involved the loss of the steam generator, and there is a striking similarity between some of the phenomena reported here with the accident at Three Mile Island. Therefore this work is now being reported so that it may be used to better understand the Three Mile Island incident. These calculations are not intended to simulate the Three Mile Island incident; they were completed several months before the Three Mile Island accident occurred.

Calculations for the feed-and-bleed concept indicate that this mode of reactor cooling could be feasible if a sufficient amount of water could be delivered at a high head (that is, the safety valve setpoint). This exact flow rate has not been determined here but has been bracketed by a calculation indicating success and another indicating failure of the method.

Results for the relief valve transients discussed here show that it is possible to maintain the core in a stable state if enough HPI injection water is available and this water is delivered to the system in a reasonable amount of time.

Caution should be exercised in applying these results to other plants because we have found these results to be very sensitive to the detailed performance characteristics of particular equipment installed.

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APPENDIX
FEED-AND-BLEED CONCEPT

The feed-and-bleed concept may be demonstrated by a nonmechanistic calculation. The question is as follows: How much heat can be removed from a system by injection of HPI water?

Let us assume that the water delivered by the HPI system can be heated and completely vaporized at the pressurizer safety valve pressure by the decay heat. For example, if the HPI pump is delivering 0.01893 m^3/s (300 gpm) at 310.9 K (100°F), then

$$Q = 0.01893\ m^3/s\ (300\ gpm)$$

For $T = 310.9$ K (100°F) and $P = 173.4 \times 10^5$ Pa (2500 psia),

$$\rho = 994.9\ kg/m^3\ (62.11\ lb/ft^3),$$

$$h = 1.734 \times 10^5\ J/kg\ (74.61\ Btu/lb),\ \text{and}$$

$$h_g = 2.537\ MJ/kg\ (1091.4\ Btu/lb)\ \text{(saturated vapor at 2500 psia).}$$

The mass flow rate is then

$$\dot{m} = \rho Q = 994.9 \times 0.01893 = 18.9\ kg/s\ (1.5 \times 10^5\ lb/hr).$$

The heat absorbed to vaporize this water is then

$$H = \dot{m}(h_g - h) = 18.9\ (2.537 \times 10^6 - 1.734 \times 10^5) = 44.5\ MJ/s$$
$$(152 \times 10^6\ Btu/hr).$$

$$H = 44.5\ MW.$$

This amount of heat is roughly equal to that produced by the decay heat of a 2772 MW thermal reactor at 50 min after scram.

This calculation demonstrates that if core cooling could be accomplished in this manner, then a clad temperature rise could be turned around after 50 min.

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