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MASTER

TITLE: EFFECT OF REACTOR COOLANT PUMPS FOLLOWING A SMALL BREAK IN A PRESSURIZED WATER REACTOR

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EFFECT OF PUMP OPERATION FOLLOWING A SMALL BREAK IN A PRESSURIZED WATER REACTOR*

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ABSTRACT

Small-break loss-of-coolant accidents were calculated to help determine whether to trip the reactor-coolant pumps early in the accident when the reactor scrams or to delay the pump trip (pump trip times ranged from 450 s to no trip at all). Four-in.-diam (approximate) cold-leg breaks in Babcock & Wilcox (B&W) and Westinghouse (W) pressurized-water reactors were investigated using the Transient Reactor Analysis Code, TRAC-PD2. The results indicated that for a 4-in.-diam cold-leg break the optimum mode of pump operation is design dependent. In terms of primary system mass depletion, the case with no pump trip was preferable for the W plant, whereas an early pump trip was preferable for the B&W plant. When the pumps were not operating in the W plant, the loop seals plugged with liquid, leading to a pressure buildup in the upper plenum and, consequently, a high liquid flow through the break. The vent valves in the B&W plant mitigated the consequences of the loop seals plugging; the effect was enough to favor an early pump trip.

INTRODUCTION

The reactor-coolant pumps on/off issue has received considerable attention with the increased interest in small-break loss-of-coolant accidents (SBLOCA). Controversy arises because, whereas core cooling is enhanced by forced circulation, sustained pump operation could also cause a higher system mass loss than if the pumps were tripped.

Analyses based on best-estimate calculations with TRAC-PD2 (Ref. 1) were performed for two generic designs of pressurized-water reactors (PWR): Babcock & Wilcox (B&W) and Westinghouse (W). The TRAC B&W calculations indicated that for a 4-in.-diam cold-leg break more primary system mass was lost and core uncovery was greatest if the pump trip was delayed until 900 s. Several cases with different pump trip times were run for the W plant, indicating that it was best not to trip the pumps at all. The loop-seal behavior had an important effect on the response of the system when the pumps were not operating in the W plant.

Both B&W and W cases were run with full high-pressure injection (HPI) and full auxiliary feedwater (AFW) systems capacity available. The effect of reduced safeguards was assessed for the W plant. In this case only one half HPI and one half AFW capacity was available. Although a rod temperature increase did not occur in any

*Work performed [illegible]

of the full-safeguards cases, there was an increase in the reduced-safeguards cases. The temperature rise was greater in the pump-trip-at-reactor-scram case.

TRAC ANALYSIS OF B&W PLANTS

For a B&W lowered-loop plant, two small cold-leg-break transients were investigated; in one transient, the reactor-coolant pumps were tripped immediately after the reactor scrammed, whereas in the other, the pump trip was delayed until there was substantial voiding of the primary system. The 0.00697 m^2-break size (0.075 ft^2) was equivalent to 1.8% of the cold-leg flow area. Full capacity of the HPI and AFW systems was available.

The TRAC model of the B&W plant includes the vent valves between the upper plenum and downcomer. The valves can provide a flow path for steam to go directly from the upper plenum through the downcomer to the break in the cold leg once the pressure in the upper plenum exceeds the downcomer pressure. The model also includes the AFW sprayed into the steam space of the once-through steam generator. HPI and accumulator injections were modeled at their correct locations. Further details of the modeling are presented in the appendix.

The significant events of the two SBLOCAs are presented in Table I. Neither case had enough voids to cause cladding temperatures to exceed initial values. The primary system mass was lower in the pump-trip-at-900-s case (Fig. 1). The operating pumps maintained a higher pressure in the downcomer than in the upper plenum so that the vent valves did not open until the pumps were tripped. Consequently, the two-phase break flow contained more liquid while the pumps were in operation. From about 200 s until 900 s, the vapor fraction reaching the break was higher for the pump-trip-at-reactor-scram case because of the open vent valves, and therefore, the break mass flow was lower with the early pump trip. After 900 s, the vapor fraction of the fluid leaving the break was approximately the same in both cases.

TABLE I
EVENTS IN B&W TRANSIENTS

	Event	Signal	Time (s) Trip at scram	Trip at 900 s
1.	Cold-leg break		0.0	0.0
2.	Reactor scram and main feedwater pump trip	13.1 MPa + 0.5 s	12.7	12.7
3.	Turbine stop valve closure	13.1 MPa + 2.5 s	14.7	14.7
4.	Auxiliary feedwater flow initiation	13.1 MPa + 40 s	52.2	52.2
5.	HPI initiation	10.1 MPa + 10 s	50.5	35.8
6.	Accumulator initiation	4.14 MPa	1060.0	1240.0

In the vessel, the core liquid volume fraction (Fig. 2) reached a minimum of about 0.8 when the loop flows stopped at about 250 s for the pump-trip-at-reactor-scram case. For the delayed pump trip, the minimum of 0.38 was reached at the time of the pump trip. In the upper level of the core, a maximum vapor fraction of 0.96 was reached briefly when the phases separated after the pump trip at 900 s. The vapor fraction in the bottom level of the upper plenum reached 1.0 for both cases. After 700 s for the pump-trip-at-reactor-scram case and 1000 s for the pump-trip-at-900-s case, the vent-valve flow was almost entirely vapor and the break flow decreased.

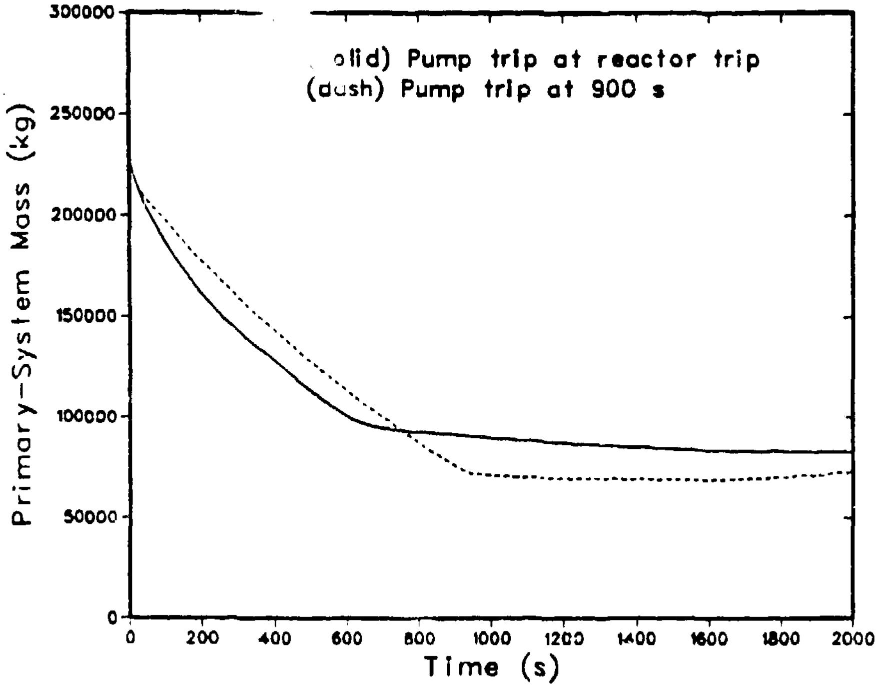

Fig. 1. B&W primary system mass.

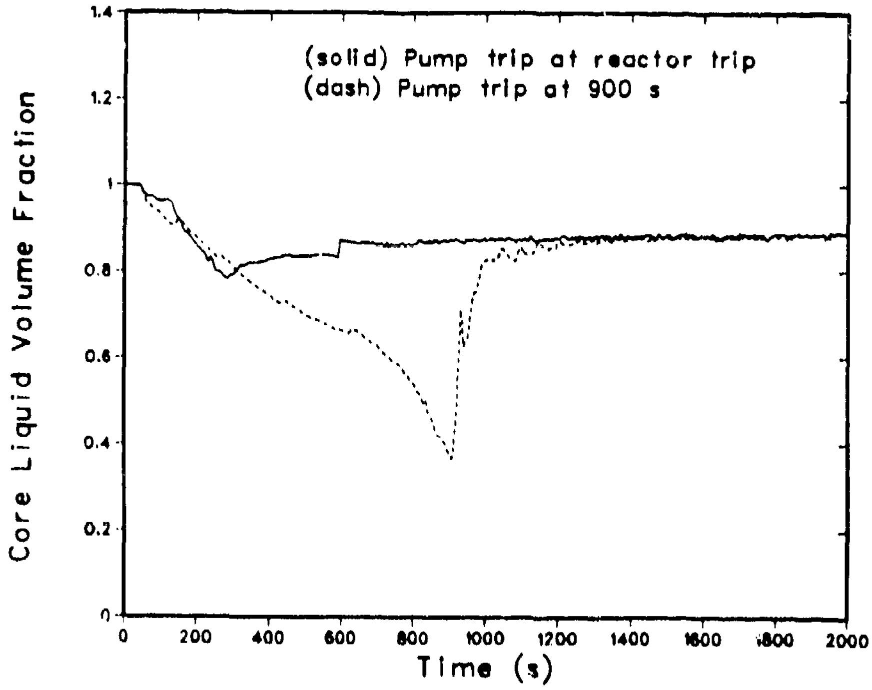

Fig. 2. Average core liquid volume fraction for B&W transients.

TRAC ANALYSIS OF W PLANTS

Several pump trip times were simulated with the TRAC best-estimate computer code for the W PWR. The five calculations were:

(1) pump trip at reactor scram - full HPI and AFW,
(2) pump trip at reactor scram - half HPI and AFW,
(3) pump trip at 450 s - half HPI and AFW,
(4) pump trip at 600 s - full HPI and AFW, and
(5) no pump trip - full HPI and AFW.

The break size was slightly larger (0.00811 m^2 or 0.087 ft^2) than for the B&W calculations, representing 2% of the cold-leg flow area. Modeling details are given in the appendix.

Table II lists the sequence of events for pump-trip-at-reactor-scram (full HPI and AFW) and the no-pump-trip (full HPI and AFW) cases. The pump-trip-at-reactor-scram case was the worst case for the W plant. The system mass was approximately equal for the pump-trip-at-reactor-scram case and the pump-trip-at-600-s case (full safeguards). However, the vapor fraction in the top level of the core increased to a much higher value in the pump-trip-at-reactor-scram case. The no-pump-trip case had considerably more primary system mass than any case when the pumps were tripped. When reduced safeguards were assumed, a significantly greater cladding temperature increase occurred with the pump-trip-at-reactor-scram case. No cladding temperature increase occurred for the full safeguards cases.

TABLE II
EVENTS IN W TRANSIENTS

	Event	Signal	Time (s) No pump trip	Time (s) Trip at scram
1.	Small break		0.0	0.0
2.	a. Reactor scram	13.1 MPa + 0.6 s	10.1	10.1
	b. HPI initiation			
	c. Main feedwater termination			
	d. Turbine stop valve closure on steam generator secondary			
3.	Upper plenum reached saturation		69.0	53.0
4.	AFW initiation	13.1 MPa + 60 s	69.5	69.5
5.	Accumulator initiation	4.14 MPa	1190.0	885.0
6.	Accumulator depletion		3165.0	2750.0
7.	LPI initiation	1.02 MPa	3220.0	3000.0

The loop-seal behavior accounted for the major difference initially in the system masses (Fig. 3) for the pump-trip-at-reactor-scram and no-pump-trip cases. The loop seals plugged with liquid between 200 s and 500 s, which resulted in high-density fluid exiting the break. Another major difference was less accumulator liquid lost out the break for the no-pump-trip case. With the pumps running, enough momentum was added to the incoming accumulator liquid to carry most of it past the break. The pumps maintained a two-phase mixture of uniform vapor fraction throughout the system, which allowed it to refill substantially; when the pumps were not operating, the fluid drained out the break. When the pumps tripped at 600 s, the loop seals plugged during the 700-900 s period. Again the plugging of the loop seals caused a large system mass loss, indicating it was best to leave the pumps operating indefinitely following a 4-in.-diam cold-leg break. The loop-seal plugging when the pump trip was delayed until 600 s did not cause any core uncovery, however, and thus, delaying the pump trip was better than immediately tripping the pumps.

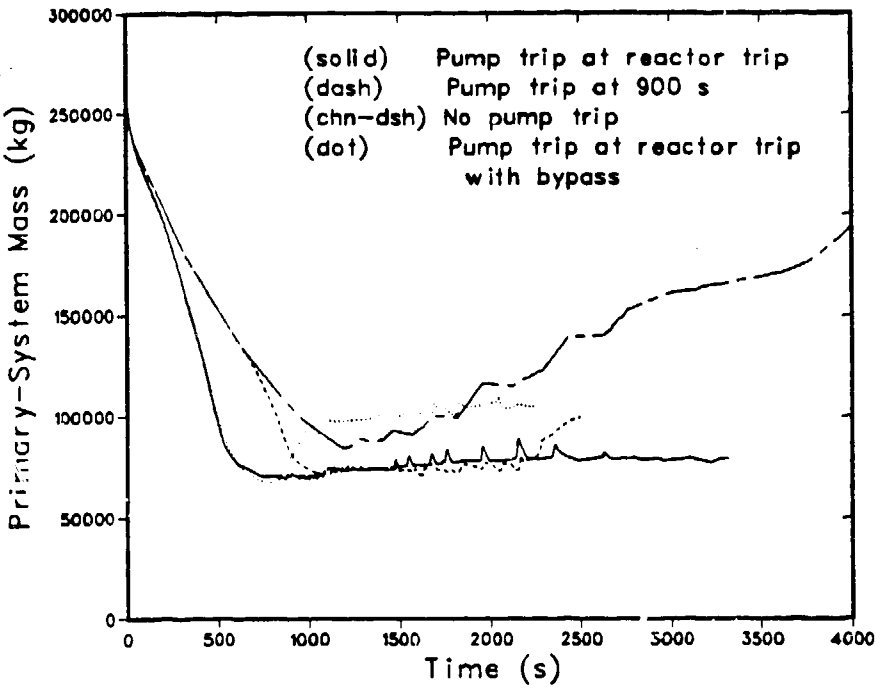

Fig. 3. W primary system mass.

Although the no-pump-trip case had significantly more core uncovery (Fig. 4), the forced flow kept the rods cooled even with considerably less primary system mass. Tripping the pumps at 600 s caused immediate phase separation and the core filled with liquid.

Explanation of Loop-Seal Plugging

A detailed discussion of loop seals is given because they had a large effect when the pumps were not operating. With the aid of Fig. 5, the loop seal behavior may be summarized as follows:

(1) loop seals fill with liquid;
(2) boiling in the core increases the pressure in the upper plenum;
(3) increased pressure exerts a force on the liquid level in the core and in the loop seal;
(4) both liquid levels decrease;
(5) when the level decreases to the bottom of the loop seal, the loop seal clears of liquid; and
(6) this clearing creates a vent path for the steam generated in the core and the core liquid level recovers.

Effect of Upper-Plenum-to-Downcomer Bypass

Because the loop-seal plugging strongly affected the primary system mass, study of the upper-plenum-to-downcomer bypass flow was warranted. This bypass could have a large effect after the pumps trip if sufficient steam was vented through the bypass to prevent the pressure buildup in the upper plenum while the loop seals were plugged. Primary-system mass would not be depleted as much as in the original pump-

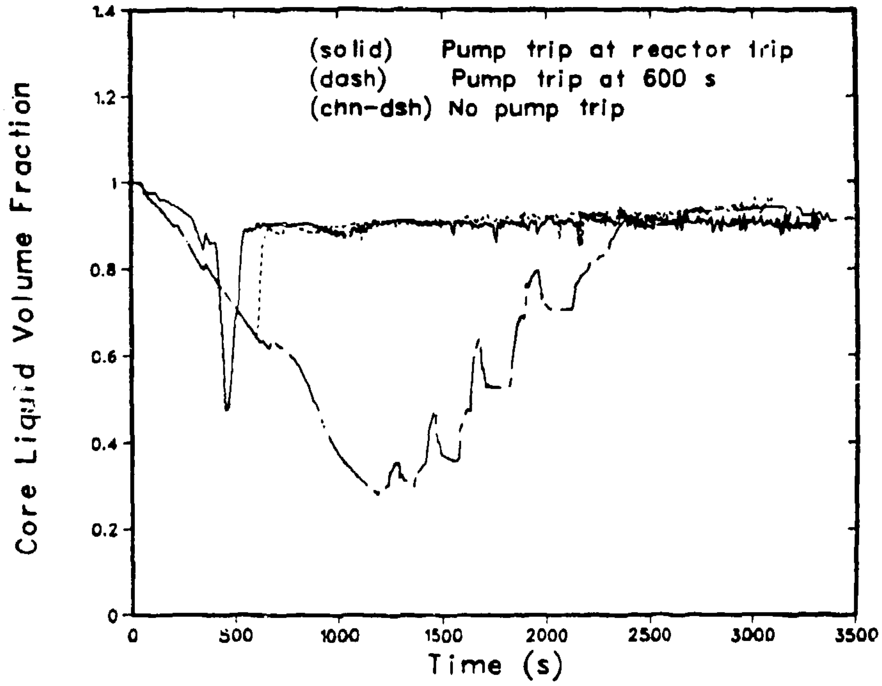

Fig. 4. Average core liquid volume fraction for W plant.

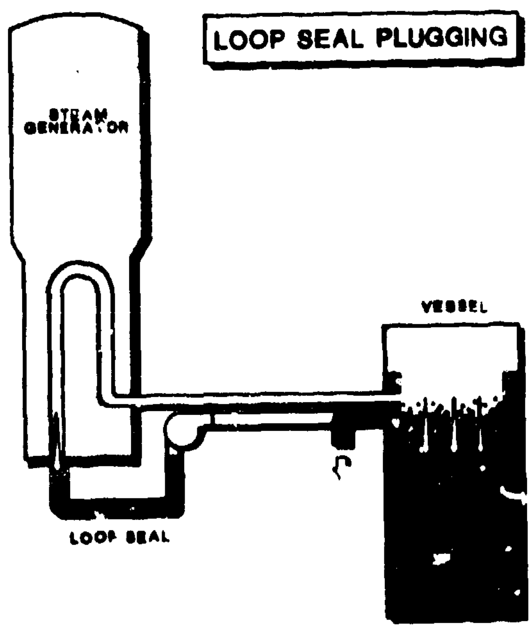

Fig. 5. Illustration of loop-seal plugging.

The bypass is present because the hot leg is not welded to the core-barrel assembly. To make repairs to the internals of the core-barrel assembly, the capability must exist to lift the assembly out of the vessel. Thus, the hot leg fits snugly against the core-barrel assembly but a bypass flow amounting to 1-2% of the loop flow passes directly from the downcomer to the upper plenum during steady-state operation.

An exact value for the bypass flow area was not known. In this calculation, the area was adjusted to give a bypass flow of 1.2% during steady-state operation. This corresponded to a gap width of 0.001 m (40 mil).

While the loop seals were plugged, only slight differences existed between the calculations with and without bypass. The bypass slightly alleviated the pressure buildup in the upper plenum so that less liquid was forced out the break from the downcomer. The flow through the bypass was 10 kg/s of two-phase mixture.

Surprisingly, after the loop seals cleared, significant differences were seen during accumulator injection. Without the bypass, the steam generated in the core and subsequent pressure increase in the upper plenum exerted a downward force on the fluid in the core, resulting in less liquid in the vessel. With the bypass modeled, the pressure in the downcomer and upper plenum equilibrated allowing the liquid level in the upper plenum to rise as high as the hot legs. This amounted to about 15 000 kg of additional mass in the vessel and could be important in keeping the core cooled.

Figure 3 compares the primary system mass for the other modes of pump operation. The primary system mass for the pump-trip-at-reactor-scram case with bypass is closer to the no-pump-trip case without bypass. Because the pumps were not operating, refill could still be expected to be as slow as in the pump-trip-at-reactor-scram case without bypass.

Effect of Reduced Safeguards

When licensing assumptions were used (one-half HPI and AFW), the rod temperatures increased significantly (Fig. 6). The primary system mass was lower in both reduced-safeguards cases because of the reduced HPI. The effect of the uncovery was greatest in the pump-trip-at-reactor-scram case because the top of the core was completely uncovered. With the pumps operating, liquid was circulated through the core. With full HPI, even though the pumps were not operating, there was enough liquid to cover the core.

SUMMARY

The difference in the results for the two generic designs of plants for the pump-trip-at-reactor-scram case is directly related to the presence of the loop seals and the vent valves. With the loop seals filled in the W plant, there is no vent path for the steam generated in the core. The upper-plenum-to-downcomer bypass provides little pressure relief. Thus, in the W plant, liquid is forced out the break as the pressure increases in the upper plenum (Fig. 7). In the B&W plant, the vent valves provide a path for the steam and a high-quality mixture flows out the break, even with the loop seals plugged (Fig. 8).

In conclusion, for a 4-in.-diam break, the time of the pump trip does not significantly affect the fuel rod temperatures (except when reduced safeguards were assumed). The presence of vent valves mitigates the consequences of loop-seal plugging. TRAC-PD2 calculations of B&W and W plants indicate that for a 4-in.-diam cold-leg break early pump trip is optimum for B&W plants and no pump trip is optimum for W plants.

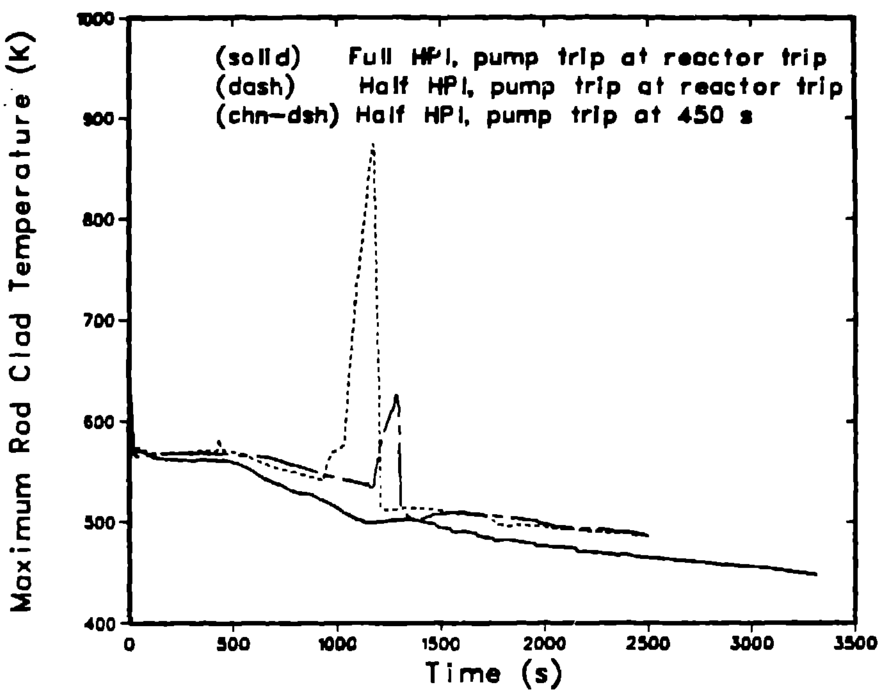

Fig. 6. Effect of reduced safeguards for W plant.

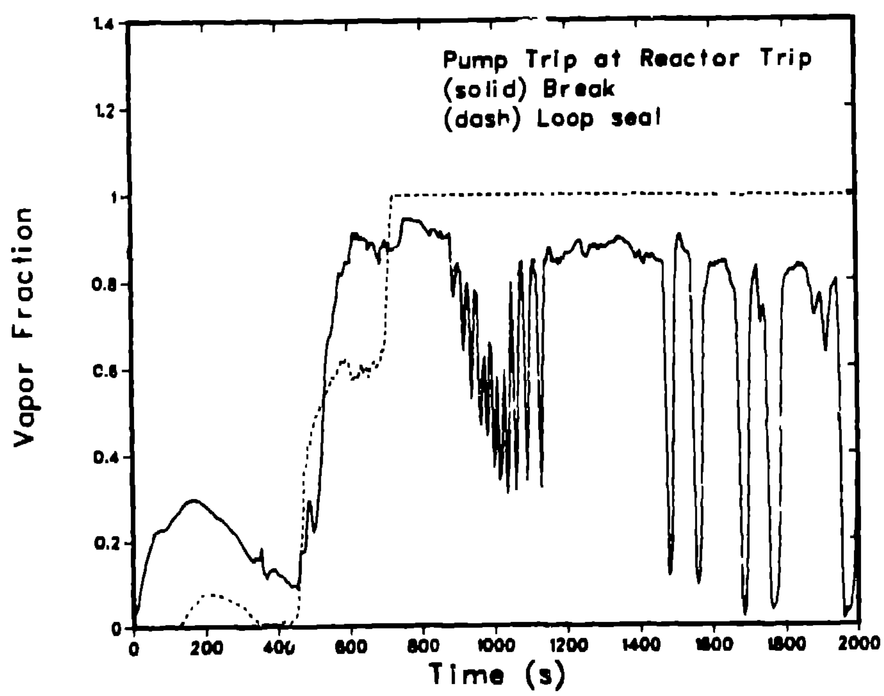

Fig. 7. Vapor fractions at break and loop seal for W plant.

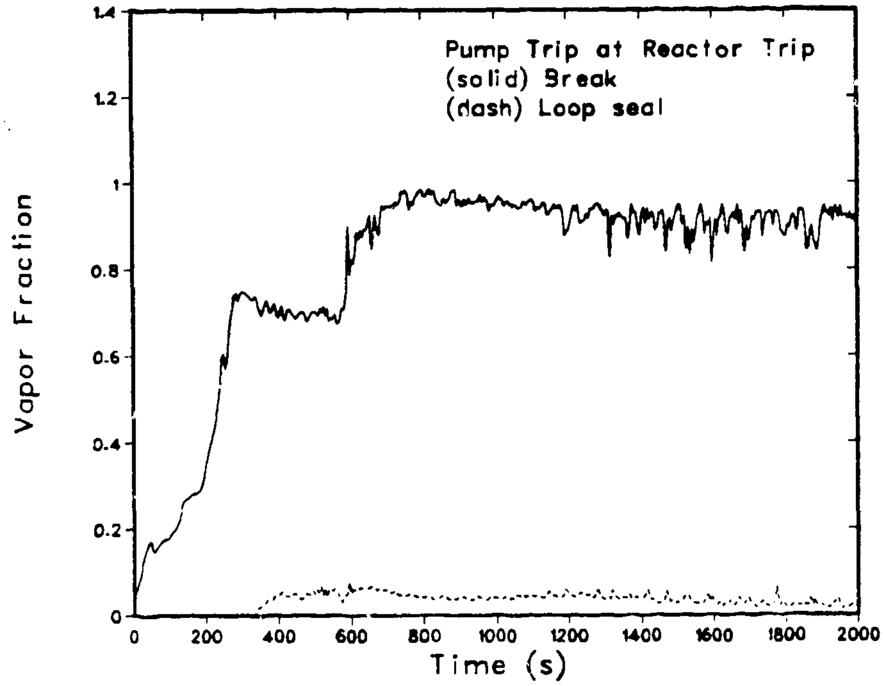

Fig. 8. Vapor fractions at break and loop seal for B&W plant.

REFERENCE

1. "TRAC-PD2: An Advanced Best-Estimate Computer Program for Pressurized Water Reactor Loss-of-Coolant Accident Analysis," Los Alamos National Laboratory report LA-8709-MS, NUREG/CR-2054 (April 1981).

APPENDIX

B&W Modeling Details

Figure A1 shows the TRAC noding diagram for the B&W lowered-loop model used for both the pump-trip-at-reactor-scram and the pump-trip-at-900-s transients. Loop A represents the loop with the cold-leg break. That loop includes the hot leg with the pressurizer connection, the steam generator, and two cold legs (one intact and one with the break). Each loop-A cold leg includes a loop seal, a pump, and an HPI connection. The reactor-coolant pumps are modeled using the Loss-Of-Fluid Test Facility (LOFT) pump characteristics built into TRAC but scaled with Three-Mile-Island-2 pump data. The break is located in one loop-A cold leg between the HPI connection and the vessel. Loop B represents the unbroken loop. It is similar to loop A except that there is no break or pressurizer and the cold legs are combined to increase calculational efficiency.

The vessel was modeled using four azimuthal segments, two radial segments, and nine axial levels. The nine levels include two in the lower plenum, four active core levels, two levels in the upper plenum to permit the vent valves to be above the hot- and cold-leg connections in case of water level changes, and an upper head. The accumulator and LPI connections are at the top level of the upper plenum.

The secondary side of each steam generator was attached to the main feedwater inlet, the auxiliary feedwater inlet, and a long pipe to the steam outlet with a side connection to a safety valve that vents to the atmosphere.

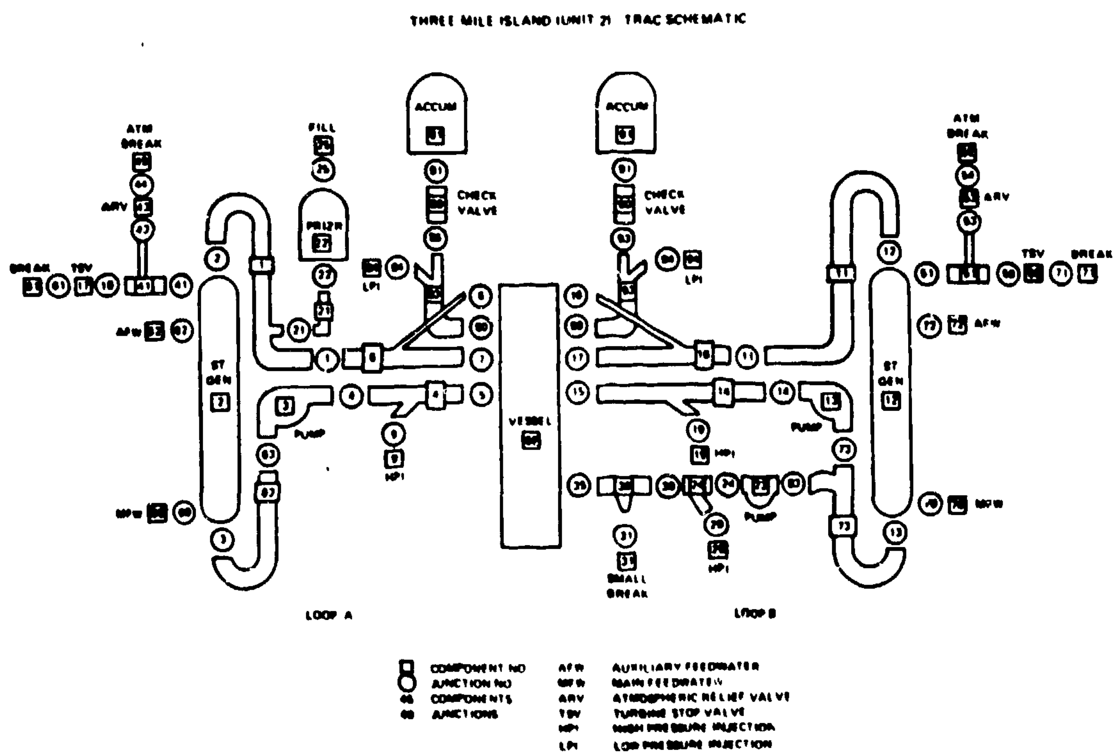

Fig. A1. B&W system schematic.

W Modeling Details

A Westinghouse four-loop PWR is modeled in this study. The system schematic is given in Fig. A2. The vessel consists of eight axial levels: two in the lower plenum, four in the core, one for the upper plenum, and one representing the upper head. The vessel is divided into two radial rings and azimuthally into a one-quarter segment for the broken-loop connections and a three-quarters segment for the intact-loops connections. The intact loops have been combined into one loop, which contains the pressurizer. The break is located downstream from the pump between the HPI inlet and the vessel.

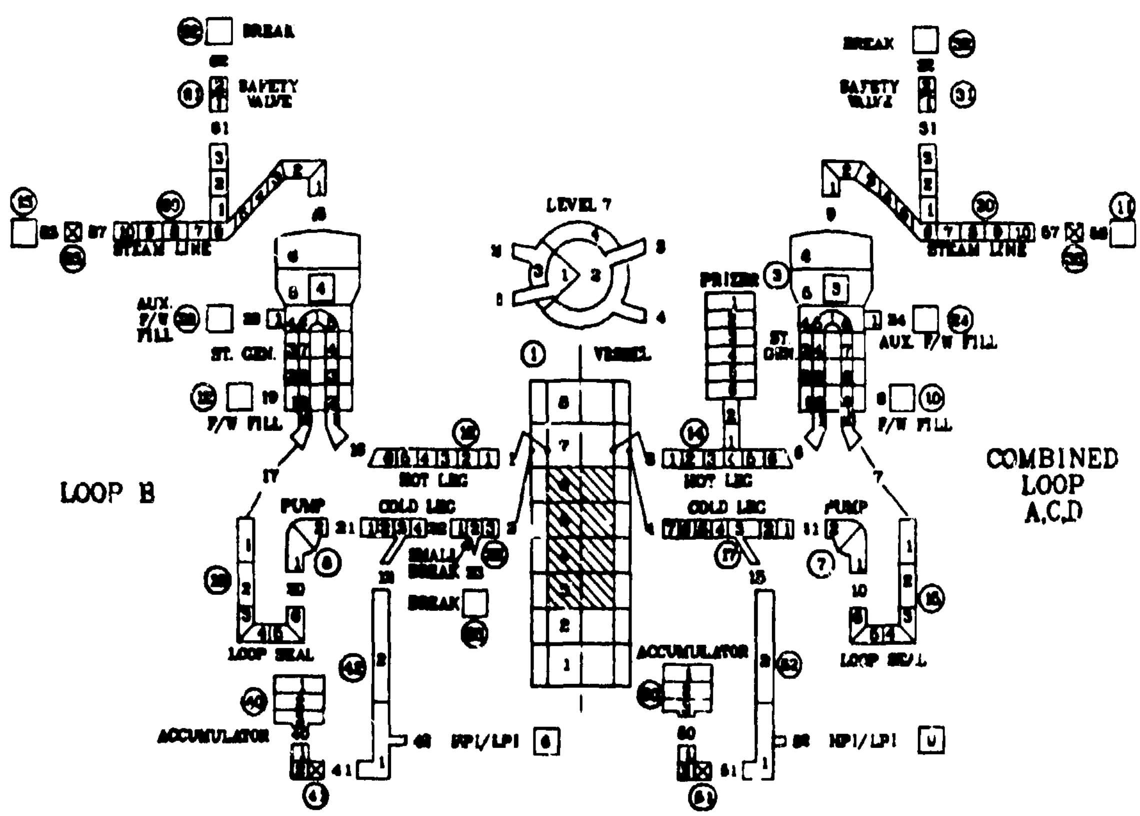

Fig. A2. W system schematic.