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COMPARISONS OF TRAC-PF1 CALCULATIONS WITH SEMISCALE MOD-3 SMALL-BREAK TESTS S-SB-P1 AND S-SB-P7*

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ABSTRACT

Semiscale Tests S-SB-P1 and S-SB-P7 conducted in the Semiscale Mod-3 facility at the Idaho National Engineering Laboratory are analyzed using the latest released version of the Transient Reactor Analysis Code (TRAC-PF1). The results are used to assess TRAC-PF1 predictions of thermal-hydraulic phenomena and the effects of break size and pump operation on system response during slow transients. Tests S-SB-P1 and S-SB-P7 simulated an equivalent pressurized-water-reactor (PWR) 2.5% communicative cold-leg break for early and late pump trips, respectively, with only high-pressure injection (HPI) into the cold legs. The parameters examined include break flow, primary-system pressure response, primary-system mass distribution, and core characteristics. For Test S-SB-P1 the experimental core uncovery began at ~800 s into the transient. The base-case calculation shows that the core was on the verge of uncovering after ~600 s, but no distinct core uncovery was predicted. However, when the break flow was increased by ~10% (significantly within the uncertainty of the experimental data), a core uncovery similar to that in the data was calculated. For Test S-SB-P7, the core uncovery was neither observed nor calculated.

*Work performed under the auspices of the US Nuclear Regulatory Commission.

I. INTRODUCTION

The Transient Reactor Analysis Code (TRAC) is an advanced best-estimate systems code for analyzing postulated accidents in light-water reactors. The latest released version of the code (TRAC-PF1)[1] provides this analysis capability for pressurized-water reactors (PWRs) and for a wide variety of thermal-hydraulic experimental facilities.

Semiscale Tests S-SB-P1 and S-SB-P7 (Ref. 2) were conducted in the Semiscale Mod-3 facility at the Idaho National Engineering Laboratory (INEL) to investigate the thermal-hydraulic phenomena resulting from a communicative small-break loss-of-coolant accident (LOCA) in a PWR. The primary factor differentiating the two tests was the operation of the primary coolant pumps. The resulting data are used to assess the analytical capability of TRAC-PF1. Of particular interest is the effect of primary coolant pump operation on the core thermal response. Effects associated with the emergency core coolant (ECC) injection, slab and rod heat transfer, and break flow model also are investigated.

II. SEMISCALE MOD-3 SYSTEM DESCRIPTION

The Semiscale Mod-3 system is a small-scale model of a four-loop PWR and includes an intact loop, a broken loop, an external downcomer assembly, and a pressure vessel. The intact loop includes a pressurizer, steam generator, and pump. The broken loop includes a steam generator, pump, and rupture valve assembly. The pressure vessel includes an upper head, an upper plenum, a 25-rod electrically heated core with thermocouples located 0.75 mm beneath the cladding surface, and a lower plenum. The external downcomer assembly includes an inlet annulus and downcomer pipe. Most system components have the same elevations as those in a full-sized PWR. The Semiscale Mod-3 system design description[3] contains additional details on the Mod-3 system.

III. TEST DESCRIPTION

Tests S-SB-P1 and S-SB-P7 simulated 2.5% cold-leg communicative breaks with pump coastdowns beginning early and late (3.4 s and 1099.7 s, respectively, after the pressurizer pressure reached 12.48 MPa). The simulated core had a flat radial power profile with three unpowered rods in a 5 × 5 matrix.

Core power decay, pump coastdowns, and steam generator valve actions were sequenced relative to a trip signal generated by a specified low pressure (12.48 MPa) in the pressurizer. The ECC was provided by the high-pressure injection system (HPIS) only. The accumulators in the intact and broken loops were valved out and the test was terminated before the system pressure fell below the normal low-pressure injection system (LPIS) set point.

The pressure suppression tank was bypassed for the test, and the break discharge was drained through a condensing system into a small catch tank. The catch tank inventory was measured before and after the test to obtain the total integrated break flow.

IV. TRAC MODEL

The TRAC input model for the Semiscale Mod-3 facility generally corresponds to the hardware configuration. Although TRAC-PF1 has the ability to model a three-dimensional vessel, all vessel elements are modeled using one-dimensional components to assess their utility and to save computation time. The TRAC-PF1 choked flow model is used to calculate the break flow. The input model consists of 42 components containing a total of 172 computational cells.

V. RESULTS

A. Test S-SB-P1 (Early Pump Trip)

The initial conditions and specified test parameters used in Semiscale Test S-SB-P1 are listed in Table I. The TRAC steady-state calculation closely approximates the actual test conditions. Table II lists the main sequence of events during the transient for the test and the calculation, which are again in good agreement.

Figure 1 shows experimental and calculated system pressure histories. During the first 1000 s of the transient, the pressure is overpredicted by an average of ~10%. At least a part of this pressure overprediction is the result of the lower break flow prediction (although the transient break flow data are not available, ~8% underprediction in the integrated break flow is estimated from the catch tank measurement). Also, during the first 1000 s of the transient, the pressure is sensitive to the system heat loss to the surroundings that has considerable uncertainty.*

The density comparisons in the loops (not presented) show, in general, good comparisons with the data with an average discrepancy of ~100 kg/m^3. Thus, TRAC-PF1 satisfactorily calculates the liquid mass distributions in the loops for Test S-SB-P1. The calculated liquid mass in the vessel, therefore, should be very close to that in the data. However, the cladding temperature comparisons show that core dryout is observed near the top whereas the prediction does not show any such tendency. However, a void fraction of $\gtrsim$0.7 is calculated near the top of the core when it is supposed to uncover, which indicates that the core is on the verge of uncovering. The primary reason for the failure to calculate the core uncovery is the lower break flow prediction.

To investigate the effect of break flow (which is underpredicted by ~8%) on the core thermal response, a sensitivity run was made by artificially increasing the break area to achieve a more accurate break flow calculation. As a result, the break flow in this run is actually overpredicted by ~2%. Clad temperatures in the upper part of the core for this run are compared in Fig. 2. The comparison is excellent with the core dryout predicted at the right time. The clad temperatures at lower elevations also are found to be in good agreement with those in the data with no core dryout predicted at these locations as indicated by the data.

*A primary-system steady-state heat loss of 125 kW was modeled in TRAC. The actual loss is estimated to be between 80-180 kW [Semiscale Review Group Meeting, presentation by A. G. Stephens (August 18, 1981)].

TABLE I

TEST S-SB-P1 INITIAL CONDITIONS

Parameter	Actual	Calculated
Core power (MW)[a]	1.96	1.96[b]
Pressurizer		
Pressure (MPa)	15.58	15.58[b]
Liquid volume (m^3)	0.0215	0.0215[b]
Intact loop		
Mass flow (kg/s)	8.21	8.22
Cold-leg temperature (K)	550.3	550.6
Hot-leg temperature (K)	584.7	583.5
Broken loop		
Mass flow (kg/s)	2.65	2.65
Cold-leg temperature (K)	550.6	551.1
Hot-leg temperature (K)	582.6	583.5
Pump speeds (rad/s)		
Intact loop	253.	266.
Broken loop	1285.	1643.
Steam generator secondaries		
Intact loop		
Pressure (MPa)	5.42	5.00
Temperature (K)	542.2	537.0
Water mass (kg)	132.3	133.9
Feedwater temperature (K)	487.8	487.8[b]
Broken loop		
Pressure (MPa)	5.24	5.03
Temperature (K)	540.0	537.4
Water mass (kg)	266.3	325.8
Feedwater temperature (K)	487.8	487.8[b]

[a]Flat radial profile.

[b]Specified as input parameter.

TABLE II

TEST S-SB-P1 SEQUENCE OF EVENTS

Event	Actual Time (s)	Calculated Time (s)
Transient initiated by opening break	0.0	0.0
Pressurizer pressure reached 12.48 MPa	17.2	19.3
Steam valve on broken-loop steam generator started to close	17.2	19.3
Steam valve on intact-loop steam generator started to close	17.2	19.3
Steam valve on broken-loop steam generator fully closed	18.8	21.3
Core power decay started	20.6	22.8
Pump coastdowns started	20.6	22.8
Steam valve on intact-loop steam generator fully closed	21.2	23.3
Broken-loop steam generator feedwater valve started to close	25.6	27.7
Intact-loop steam generator feedwater valve started to close	25.6	27.7
Broken-loop steam generator feedwater valve fully closed	not recorded	31.7[a]
Intact-loop steam generator feedwater valve fully closed	not recorded	31.7[a]
HPIS injection started	45.6	48.2
Auxiliary feedwater started[b]	80.6	82.7
Auxiliary feedwater shut off[b]	570.6	572.7
Transient terminated	1670.6	1670.6

[a]For modeling purposes, the valve closing time was estimated from the differential pressure reading across the orifice in the feedwater line.

[b]Refers to the power to auxiliary feedwater pumps. The actual flow histories are based on the liquid levels in steam generator secondaries.

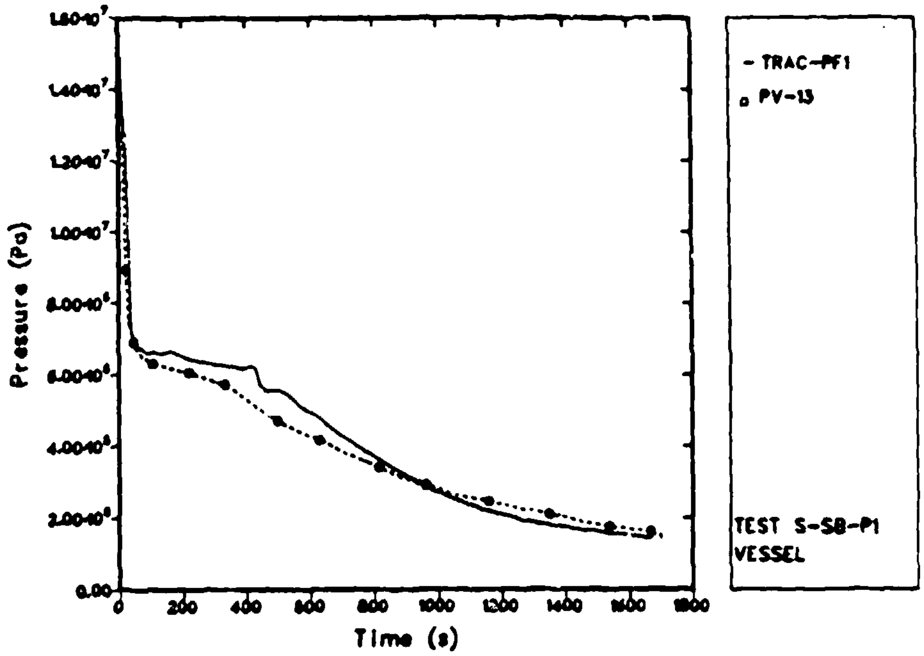

Fig. 1.
Upper-plenum pressures for Semiscale Test S-SB-P1.

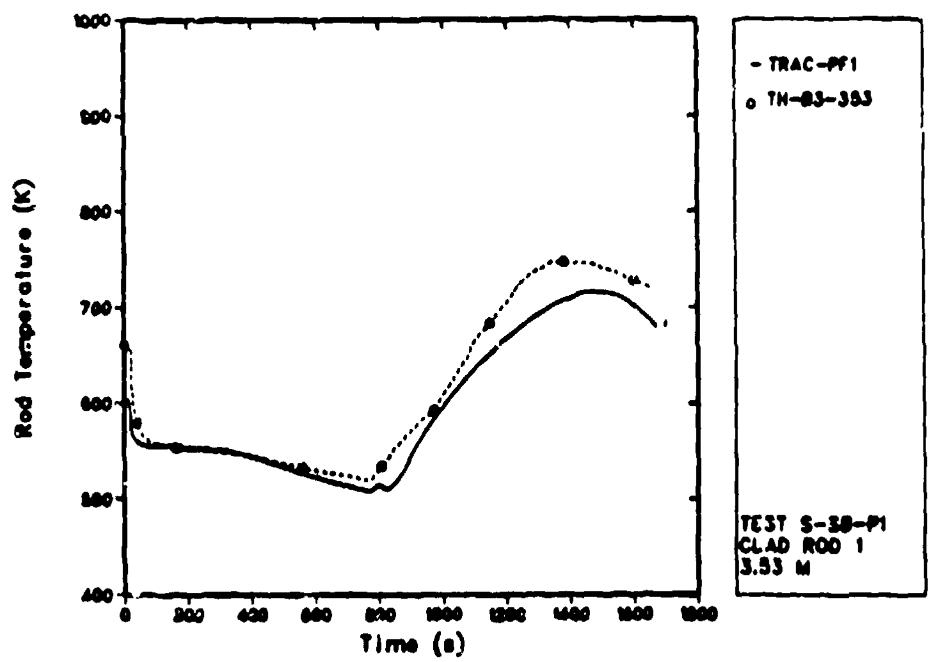

Fig. 2.
Comparison of the clad temperatures at 3.53-m elevation for Semiscale Test S-SB-P1 between the experimental data and the TRAC-PF1 calculation with increased break flow.

The CDC 7600 central-processor-unit (CPU) time required to run a 1671-s system transient was 2860 s at an average 0.37-s time step. The running time to simulate the same length of transient using TRAC-PD2 (Ref. 4) was 22 136 s.

B. Test S-SB-P7 (Late Pump Trip)

For all practical purposes, the initial and boundary conditions for Tests S-SB-P1 and S-SB-P7 were the same with the exception of different pump coastdown times. Thus, Tables I and II also apply to Test S-SB-P7 with exception of the pump trips which occurred at ~1117 s in both the experiment and the calculation.

Experimental and calculated break flows are presented in Fig. 3. The mass flow is overpredicted between 300 and 1000 s of the transient because of a higher density prediction upstream of the break during this time. However, the overprediction in the break flow may not be as large as it appears in Fig. 3 because the instrument reading after 500 s lies mostly in the dead band range. The measured mass flow uncertainty, therefore, is expected to be much larger than depicted in Fig. 3. A better estimate of the error in the calculated break flow is made by comparing the integrated flows with the catch tank measurements. Such a comparison shows that the flow is underpredicted by an average of 5% for the first 814.6 s and overpredicted by an average of 29% during the rest of the transient, with an average overprediction of only 4% for the entire transient. This suggests that the actual flow during the first 300 s of the transient must have been significantly larger than indicated by

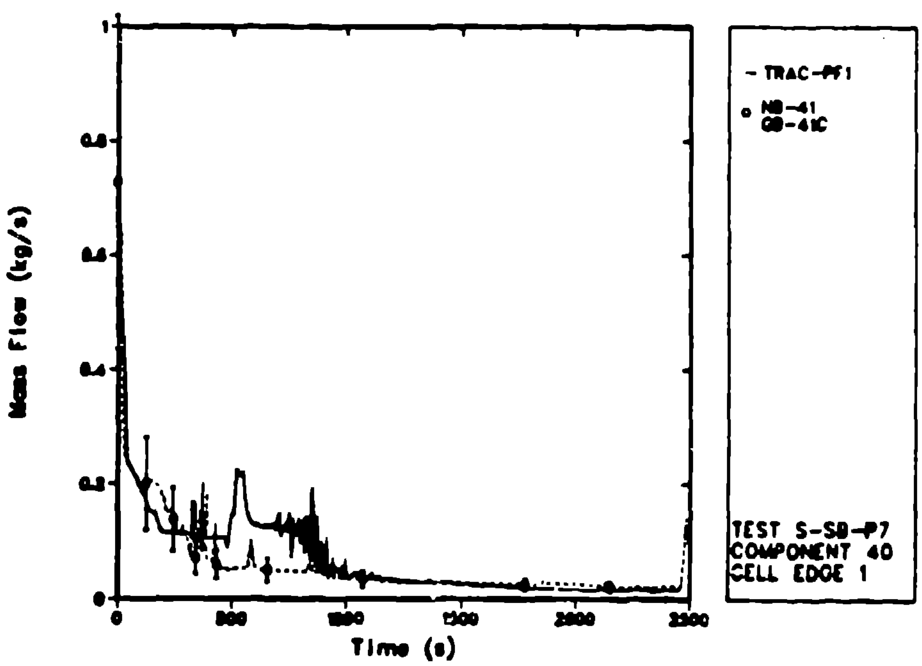

Fig. 3.
Break flows for Semiscale Test S-SB-P7.

the measurement. These comparisons clearly point to the large uncertainty in the experimental data plotted in Fig. 3.

Figure 4 shows experimental and predicted system pressure histories. The pressure is slightly overpredicted during the first 1000 s and underpredicted during the rest of the transient. The discrepancy in the pressure calculation is caused primarily by the inaccuracy in the break flow calculation, which is underpredicted during the first one third of the transient and overpredicted during the rest of the transient. The pressure also is sensitive to the system heat loss, as mentioned earlier.

The calculated density comparisons (not shown), in general, are in good agreement with the data with the exception that during the first 1000 s of the transient the calculated density decays do not occur as rapidly as those in the experiment. This is primarily the result of the lower break flow prediction during this time. The calculated liquid distribution in the system, therefore, should be approximately the same as that in the experiment.

For Test S-SB-P7 core uncovery is neither observed nor calculated. Thus, the cladding temperatures (not presented) at various elevations in the core are slightly above saturation temperature in both the calculation and the experiment.

It took 5052 s of CPU time on a CDC 7600 to simulate a 2465-s system transient at an average 0.29-s time step. The running time to simulate the same length of transient using TRAC-PD2 (Ref. 4) was 42 839 s.

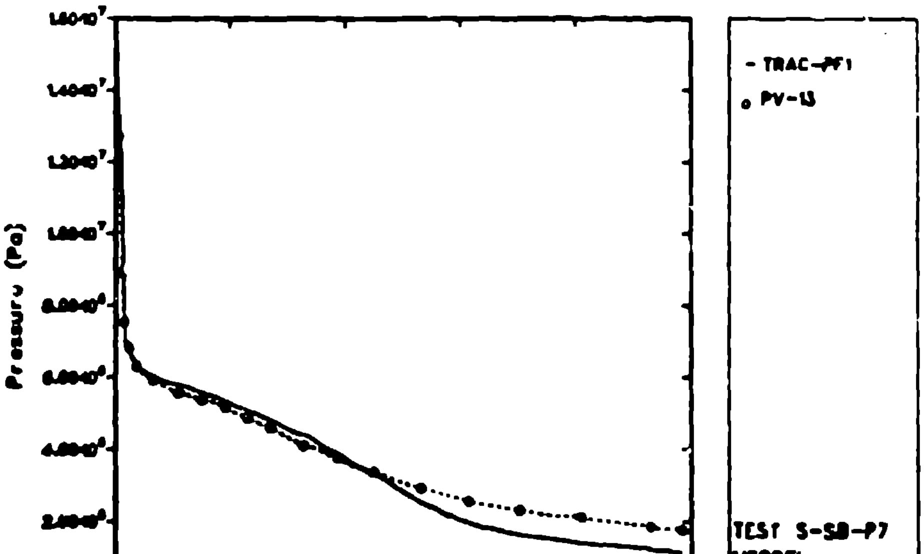

Fig. 4.
Upper-plenum pressures for Semiscale Test S-SB-P7.

VI. CONCLUSION

TRAC-PF1 provides a reasonable small-break modeling capability for predicting slow-transient thermal-hydraulic phenomena during a cold-leg break. Most comparisons between TRAC-PF1 results and experimental data generally predict correct trends. This conclusion was made by comparing the break flows, system pressures, primary side fluid densities, and clad temperatures.

TRAC-PF1 predicts the break flow well within the uncertainty of the measurement. However, more accurate measurement of the transient break flow is highly desirable because some inconsistencies in the transient break flow and the catch tank measurements have been found.

In both the experiment and the calculation, Test S-SB-P1 with early pump trip was found to be more severe with respect to core thermal response than Test S-SB-P7 with late pump trip.

In conclusion, TRAC-PF1 appears able to predict most of the thermal-hydraulic phenomena resulting from early and late pump-trip small-break LOCAs within the confines of the uncertainty in the boundary conditions. In general, quantitatively good break flows, system pressures, liquid mass distributions, and core thermal response have been calculated. No TRAC-PF1 modeling deficiencies were found. However, if more accurate measurement of the break flow could be achieved, it would be desirable to improve the TRAC choking model.

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