LA-UR -82-1939

CONF-820802--6

Los Alamos National Laboratory is operated by the University of California for the United States Department of Energy under contract W-7405-ENG-36.

MASTER

LA-UR--82-1939

DE82 019570

TITLE: POSTTEST ANALYSIS OF SEMISCALE LARGE-BREAK TEST S-06-3 USING TRAC-PF1

AUTHOR(S): B. E. Boyack

DISCLAIMER

SUBMITTED TO: International Meeting on Thermal Nuclear Reactor Safety
August 19-September 2, 1982
Americana-Congress Hotel
Chicago, IL

IS UNLIMITED

Los Alamos National Laboratory
Los Alamos, New Mexico 87545

FORM NO 836 R4
ST NO 2629 5/81

POSTTEST ANALYSIS OF SEMISCALE LARGE-BREAK TEST S-06-3 USING TRAC-PF1*

B. E. Boyack

Energy Division
Los Alamos National Laboratory
Los Alamos, New Mexico 87545

ABSTRACT

The Transient Reactor Analysis Code (TRAC) is an advanced systems code for light-water-reactor accident analysis. The code was developed originally to analyze large-break loss-of-coolant accidents (LOCAs) and running time was not a primary development criterion. TRAC-PF1 was developed because increased application of the code to long transients such as small-break LOCAs required a faster-running code version. Although developed for long transients, its performance on large-break transients is still important. This paper assesses the ability of TRAC-PF1 to predict large-break-LOCA Test S-06-3 conducted in the Semiscale Mod-1 facility.

INTRODUCTION

The Transient Reactor Analysis Code (TRAC) is an advanced systems code for light-water-reactor accident analysis. The original goal of the TRAC development effort was to provide a unified bench-mark systems code for the analysis of large-break loss-of-coolant accidents (LOCAs). Code adequacy was assessed by comparing predictions with an extensive experimental data base. As a bench-mark code, running time was not a primary concern for the relatively brief large-break-LOCA transients. However, recent emphasis has been placed on prediction of long transients such as small-break LOCAs. For such transients, a fast-running version, TRAC-PF1 [1], has been developed.

An experiment conducted in the Semiscale Mod-1 facility was selected as one element in the TRAC-PF1 developmental assessment program. Test S-06-3 [2] simulated the response of a pressurized water reactor (PWR) to a large-break LOCA. This paper reports the results of TRAC-PF1 predictions of Test S-06-3.

---

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

EXPERIMENTAL FACILITY

The Semiscale Mod-1 system [3] shown in Fig. 1 was a small-scale model of a four-loop PWR. The system included a pressure vessel with core simulator, upper and lower plenums, and downcomer; an intact loop with steam generator, pump, and pressurizer; a broken loop with simulated steam generator and simulated pump; coolant injection accumulators; high- and low-pressure coolant-injection pumps; and a pressure-suppression system with a suppression tank, header, and heated steam supply system. The Mod-1 core contained 40 rods, including 36 active rods that yielded a total core power of 1.004 MW.

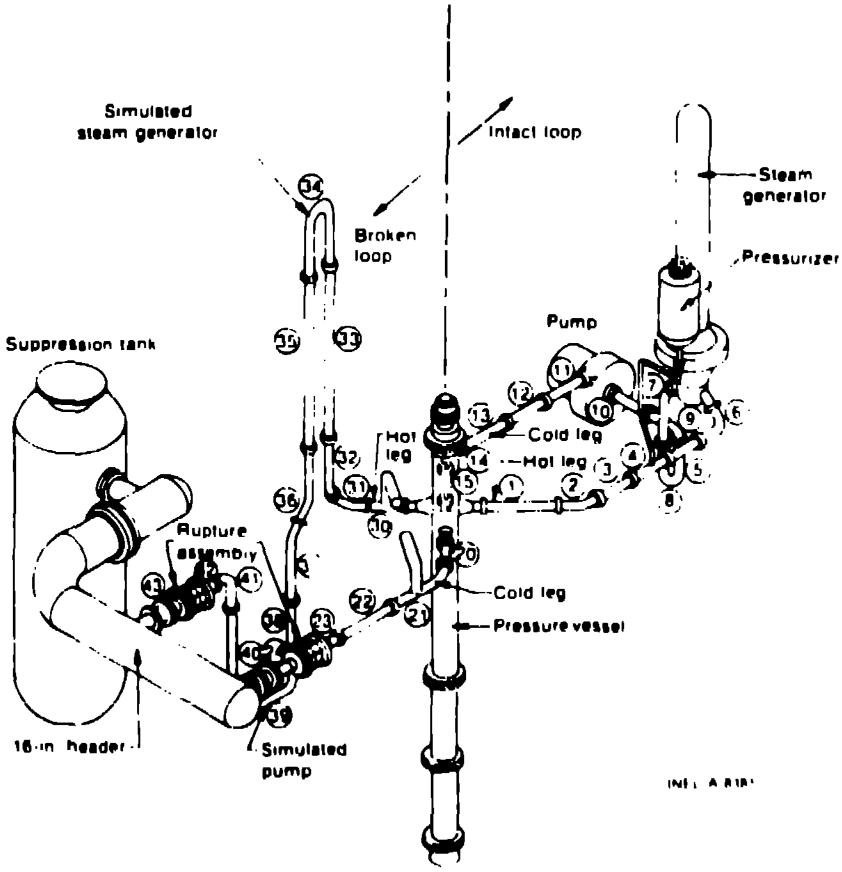

Fig. 1.
Semiscale Mod-1 system for cold-leg-break configuration - isometric.

TRAC MODEL

The AC model of the Semiscale Mod-1 facility (Fig. 2) generally corresponds to the hardware configuration. The two primary coolant loops, associated piping, and the test vessel are simulated. Although TRAC-PF1 can model a three-dimensional vessel, all system elements were modeled as one-dimensional components to assess their utility in large-break calculations. The elements used to develop the TRAC-PF1 input model are indentified in Table I. The model deviates from the hardware in the following respects:

1. the inlet annulus and downcomer that are inside the pressure vessel are modeled as one-dimensional elements outside the test vessel and

2. the facility containment system is not modeled directly but is represented by a break component with the containment pressure history specified.

The core is divided into eight vertical levels. An average- and a high-power rod are modeled at each core level. The hot- and cold-leg breaks are modeled as finely noded pipe components.

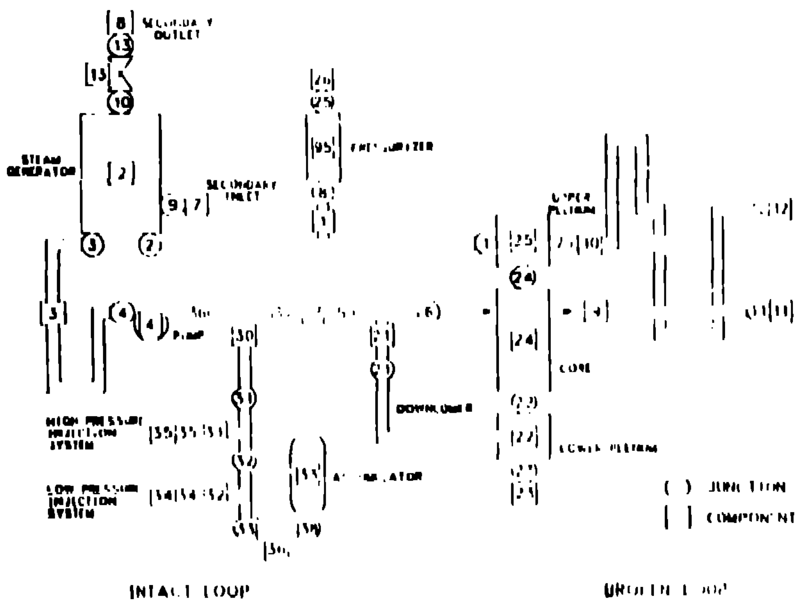

Fig. 2.
TRAC model of Semiscale Mod-1 facility.

TABLE I
System Model Components
Transient Calculation

Component Number	Component Type	Description	Number of Fluid Cells
1	TEE	Intact-loop hot leg	6,5
2	STGEN	Intact-loop steam generator	22,12
3	PIPE	Intact-loop pump suction	6
4	PUMP	Intact loop pump	2
7	FILL	Intact-loop steam generator feedwater	1
8	BREAK	Intact-loop secondary-pressure set point	1
9	PIPE	Broken-loop cold leg	18
10	PIPE	Broken-loop hot leg	34
11	BREAK	Broken loop suppression-system set point	1
12	BREAK	Broken-loop suppression-system set point	1
13	VALVE	Intact-loop steam line	2
21	TEE	Downcomer inlet annulus and downcomer	1,7
22	TEE	Downcomer and lower vessel plenum	4,1
23	FILL	Bottom of vessel	1
24	CORE	Vessel core	8
25	TEE	Vessel upper plenum	1,1
26	FILL	Top of pressurizer	
30	TEE	Intact-loop cold leg	1,1
31	TEE	HPIS piping	2,1
32	TEE	LPIS piping	21
33	ACCUM	Intact-loop accumulator	3
34	FILL	LPIS boundary condition	1
35	FILL	HPIS boundary condition	1
36	VALVE	Accumulator valve	2
37	PIPE	Intact-loop cold leg	1
95	PRIZER	Pressurizer	12

ASSESSMENT RESULTS

Steady-state conditions immediately before blowdown initiation were calculated using TRAC-PF1. Table II lists the measured and calculated conditions at blowdown initiation. The largest calculated deviation was the intact-loop cold-leg volumetric flow that exceeded the measured value by ~4.5%. The higher flow rate resulted from matching the hot-leg temperature closely.

Figure 3 compares the calculated and measured system pressures. During the blowdown phase the calculated system pressure decayed more rapidly than the measured system pressure. The influence of a finely noded representation was investigated by modeling the breaks with a critical-flow model using coarse noding. The changes in calculated system pressure decay during blowdown were minor. The emergency core-cooling accumulators, which were tripped on system pressure, were activated at 16.07 s. This was ~2.5 s earlier than measured in the test.

The loop mass flows generally were well calculated by TRAC-PF1. Figure 4 compares the intact-loop mass flows through the pump. The calculated flow decayed ~1 s earlier than measured and became zero at 25 s. A small residual flow was measured throughout the blowdown phase of the test. The comparison between the calculated and measured broken-loop cold-leg entrance mass flows (Fig. 5) is excellent. The calculated and measured core inlet mass flows are compared in Fig. 6.

TABLE II

Test S-06-3 Initial Conditions

	Experiment	Calculation
Core power (MW)	1.004	1.004[a]
Intact-loop cold-leg fluid temperature (K)	563.	562.1
Hot-leg to cold-leg temperature differential (K)	34.1	34.1
Pressurizer pressure (MPa)	15.769	15.769[a]
Pressurizer liquid mass (kg)	9.09	9.27
Steam-generator feedwater temperature (K)	497.	497.[a]
Fluid temperature in broken loop on the pump side (K)	562.	562.[a]
Fluid temperature in broken loop on the vessel side (K)	591.	591.[a]
Intact-loop cold-leg flow (/s)	6.68	6.98

[a]Specified as input parameter to steady-state calculation.

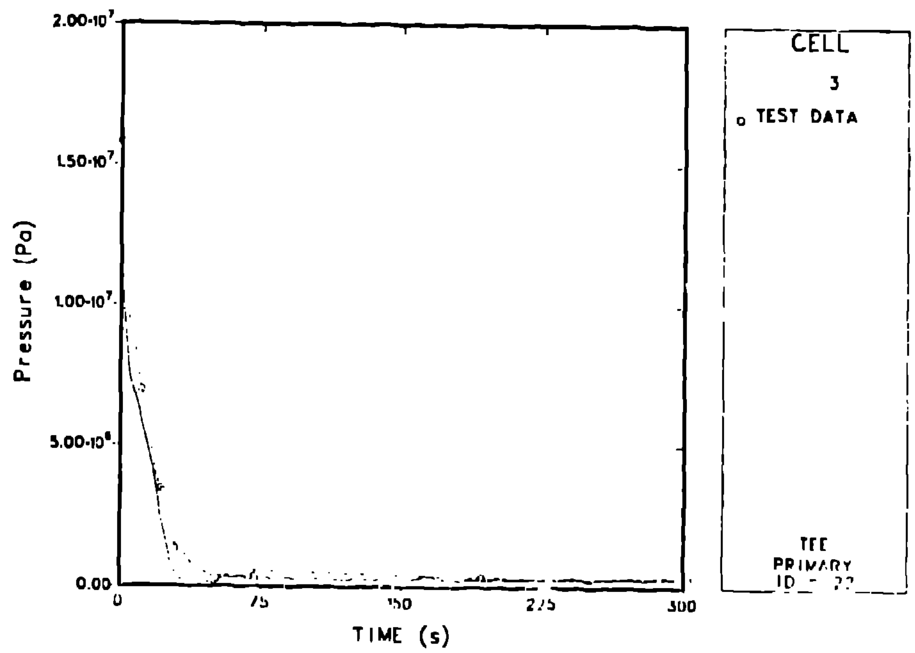

Fig. 3.
Calculated and measured system pressures.

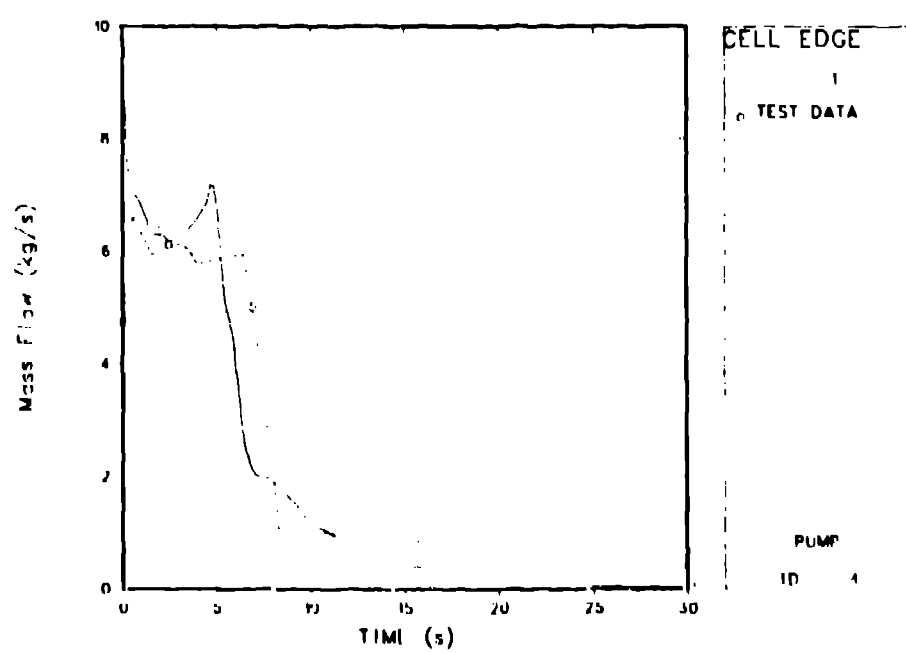

Fig. 4.
Calculated and measured intact-loop pump mars flows.

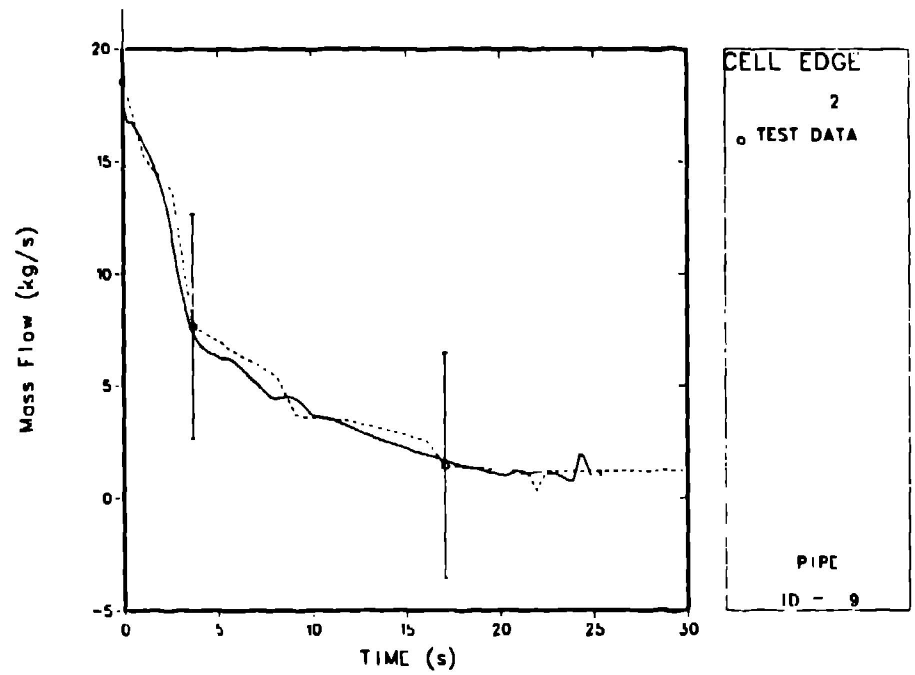

Fig. 5.
Calculated and measured broken-loop cold-leg entrance mass flows.

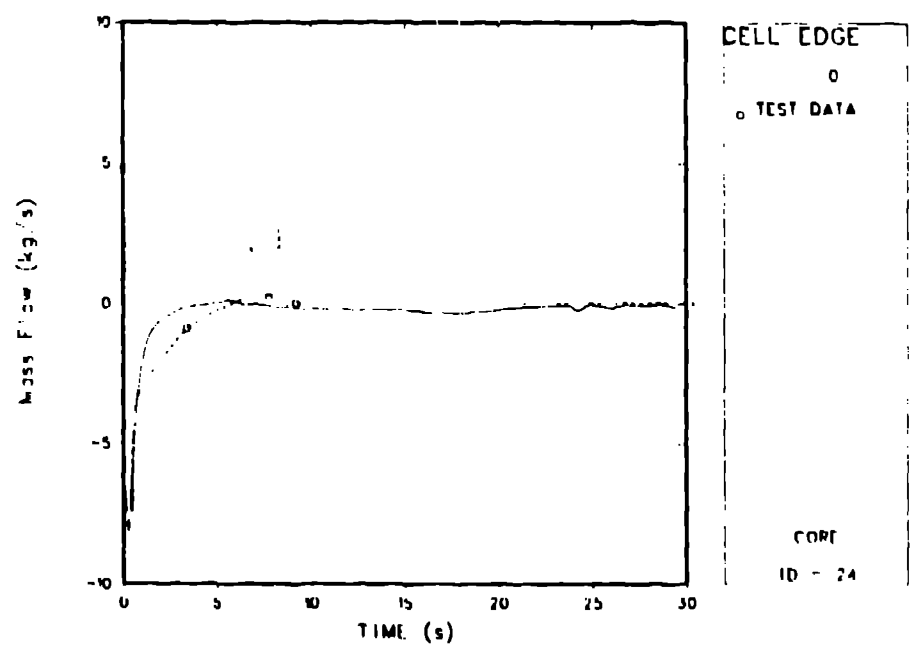

Fig. 6.
Calculated and measured core inlet mass flows.

Within the first second following blowdown initiation, a sharp flow reversal at the core inlet was measured and calculated. The core inlet flow approached zero at ~6 s; the calculated flow decayed more rapidly than the measured. A surge of fluid upward into the core was measured but not calculated between 7 and 10 s.

Figure 7 compares the rod cladding temperature histories at a level near the middle of the heated core. TRAC-PF1 overpredicted the maximum cladding temperature by 125 K. The measured rate of cladding temperature increase during the first 10 s of the blowdown was reduced by the upward flow surge into the core between 7 and 10 s. The excess temperature rise calculated by TRAC-PF1 was a direct consequence of the failure to predict the brief surge flow through the core. A quench was not calculated at 300 s, whereas the measured quench occurred at ~180 s. After reviewing the predicted quenching temperature at all core levels, it was concluded that the wall temperature used in TRAC-PF1 to define the boundary between the transition and film boiling regimes in the boiling curve may be too low. A sensitivity study was conducted by varying the expression that defines the homogeneous nucleation temperature used in calculating the transition wall temperatures. For the study, the homogeneous nucleation temperature was specified at a constant, the saturation temperature at the critical pressure (647.3 K). The predicted time of cladding quenching (Fig. 7) improved markedly, which suggests that further study of the correlation for the homogeneous nucleation temperature is needed.

In an earlier study [4], predictions of Test S-06-3 made with a previous TRAC version, TRAC-PD2, were assessed. In general, the predictions were quite similar. However, TRAC-PD2 calculated the core heatup and cladding quenching phenomena better. A comparison of running times of the code versions revealed an even more significant feature. The CDC-7600 central-processor-unit (CPU) time required to calculate the first 200 s of the Test-S-06-3 transient using TRAC-PF1 was 3211 s; using TRAC-PD2, 19020 s. Thus, TRAC-PF1 is approximately a factor of 6 faster in calculating large-break transient Test S-06-3.

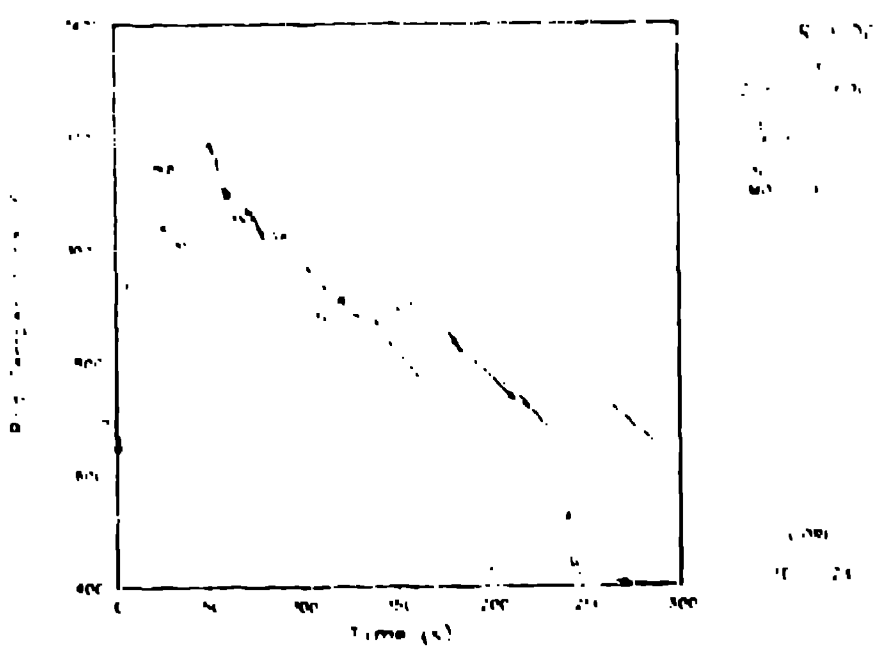

Fig. 7.
Calculated and measured cladding temperatures at mid core.

CONCLUSIONS

A posttest study of Semiscale large-break-LOCA Test S-06-3 has been completed to assess TRAC-PF1 predictions of these transients. In general, TRAC-PF1 predicted the transient well. During the blowdown phase the system pressure was underpredicted. Loop mass flows were predicted well. Quenching of the rod cladding was predicted to occur much later than measured. A parametric study was conducted and the minimum stable film-boiling temperature was increased. A marked improvement in the predicted time of quenching resulted. The results showed that TRAC-PF1 calculated the transient about as well as TRAC-PD2. However, TRAC-PF1 calculated the transient in ~17% of the time. Thus, TRAC-PF1 offers significant improvement in calculation time not only for small-break but also for large-break transients.

REFERENCES

1. J. H. MAHAFFY, D. R. LILES, and T. F. BOTT, "TRAC Methods and Models," Proceedings of the American Nuclear Society Specialists Conference on Small Break Loss-of-Coolant Accident Analyses in LWRs, Monterey, California (August 25-27, 1981), Report WS-81-201, Electric Power Research Institute report (1981).

2. B. L. COLLINS, et al., "Experimental Data Report for Semiscale Mod-1 Test S-06-3 (LOFT Counterpart Test)," Report NUREG/CR-0251, EG&G Idaho, Inc. (July 1978).

3. E. M. FELDMAN and D. J. OLSON, "Semiscale Mod-1 Program and System Description for the Blowdown Heat Transfer Tests (Test Series 2)," Report ANCR-1230, EG&G Idaho, Inc. (August 1975).

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