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TITLE TRAC INDEPENDENT ASSESSMENT FOR PWR ANALYSIS

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TRAC INDEPENDENT ASSESSMENT FOR PWR ANALYSIS*

by

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The Los Alamos National Laboratory is developing the Transient Reactor Analysis Code (TRAC) for application to pressurized-water reactors (PWRs). Several code versions have been released; each new version introduced improvements to existing models and numerics and added new models to extend the applications of the code. The first goal of the code was to analyze large-break loss-of-coolant accidents (LOCAs), and the TRAC-P1A and TRAC-PD2 codes primarily addressed the large-break LOCA. The TRAC-PF1 code contained major changes to the models, trips, and numerical methods. These modifications enhanced the computational speed of the code and improved its application to small-break LOCAs. The TRAC-PF1/MOD1 code added improved steam-generator modeling, a turbine component, and a control system together with modified constitutive relations to model the balance of plant on the secondary side and to extend the applications to non-LOCA transients.

During the past year we assessed TRAC-PD2, TRAC-PF1, and TRAC-PF1/MOD1. This work supports applications of the codes to large- and small-break LOCAs and non-LOCA transients. We used several experiments from the Loss-of-Fluid Test (LOFT) and Semiscale facilities.

We analyzed LOFT L2-3 and L2-5 with TRAC-PD2; both tests simulated large, double-ended cold-leg breaks. Test L2-3 operated the primary-coolant pumps at approximately constant speed, whereas Test L2-5 utilized an early pump trip and a very rapid pump coastdown. The code correctly calculated the hydraulic behavior in both tests. Figure 1 compares the calculated intact-loop hot-leg pressure to the data for Test L2-3; the comparison essentially is the same for Test L2-5. The code slightly overpredicted the pressure from ~5 s through the end of blowdown; as a result, the accumulator injected ~1 s late.

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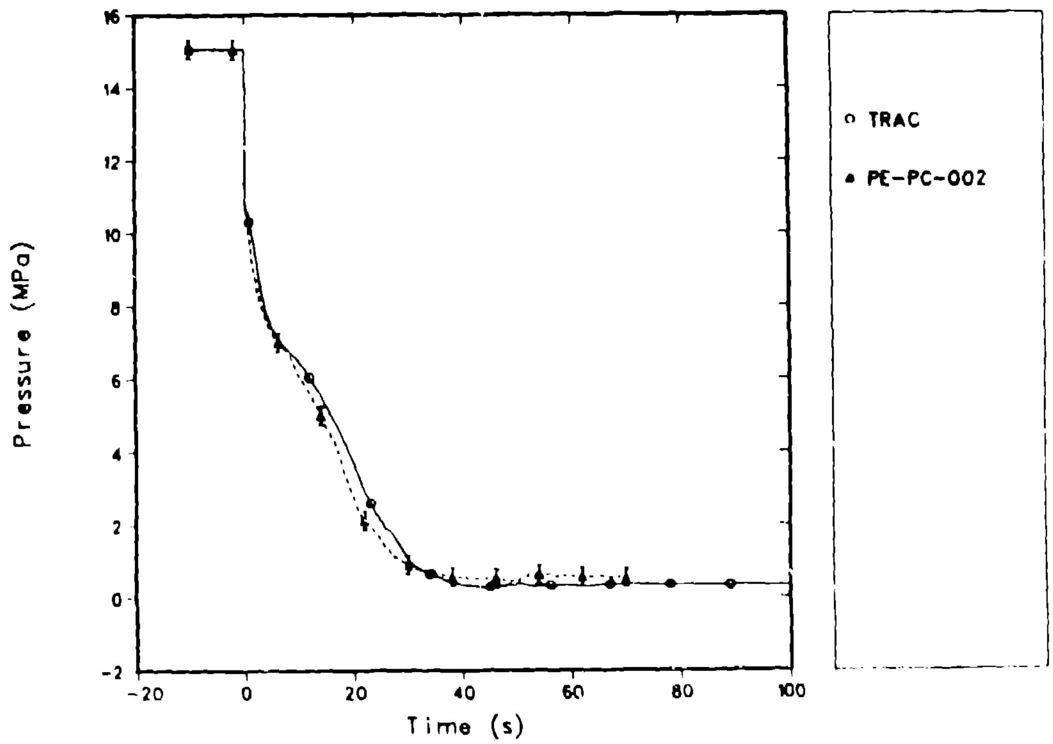

Fig. 1.
LOFT L2-3 calculated and measured intact-loop hot-leg pressures.

Figures 2 and 3 compare the calculated intact-loop cold-leg densities to the data for Tests L2-3 and L2-5, respectively. The L2-3 calculation used the interphase-condensation model as it was released in TRAC-PD2/MOD1; Fig. 2 indicates that the code correctly calculated the beginning of the condensation-induced density oscillations. The L2-5 calculation used a modified condensation model that reduced the condensation rate under certain flow conditions; we developed the updated model after large-break LOCA calculations for PWRs indicated that the original model resulted in high condensation rates. With Test L2-5, we assessed the modified model. As Fig. 3 indicates, the density oscillations start late. We reran the calculation with the released version of the condensation model, and the oscillations began at the correct time. This information affected the condensation model incorporated into TRAC-PF1/MOD1.

Figures 4 and 5 show the calculated cladding temperatures at 0.79-m elevation of the central fuel bundle compared to data for Tests L2-3 and L2-5, respectively. For Test L2-3, the code correctly calculated the general thermal behavior of the fuel cladding throughout the core, including the early rewets. The main discrepancy was the inability of the code to calculate the complete rewetting of the lower section of the central fuel bundle. This discrepancy is related to the combination of the film-boiling and the minimum film-boiling

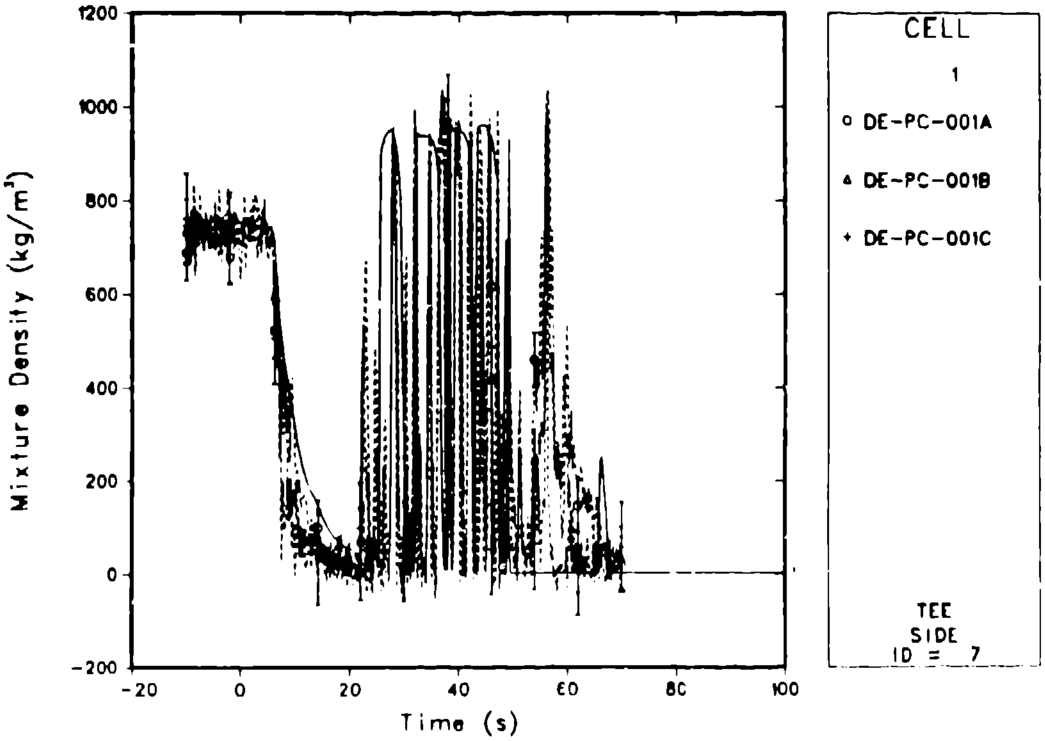

Fig. 2.
LOFT L2-3 calculated and measured intact-loop cold-leg densities.

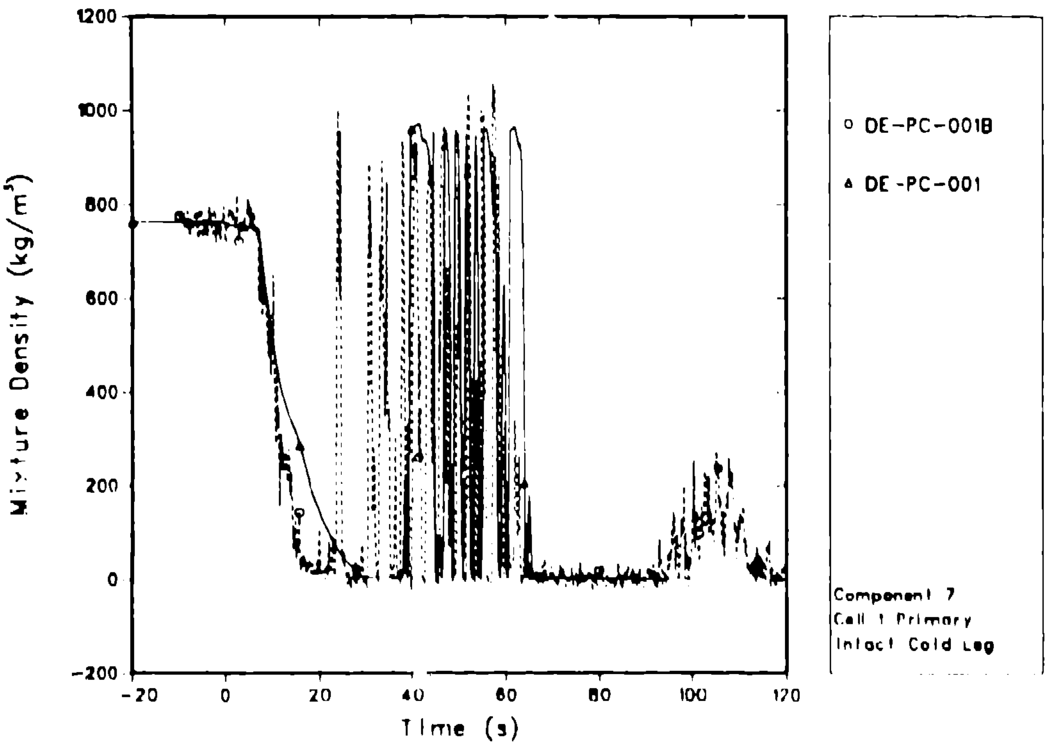

Fig. 3.
LOFT L2-5 calculated and measured intact-loop cold-leg densities.

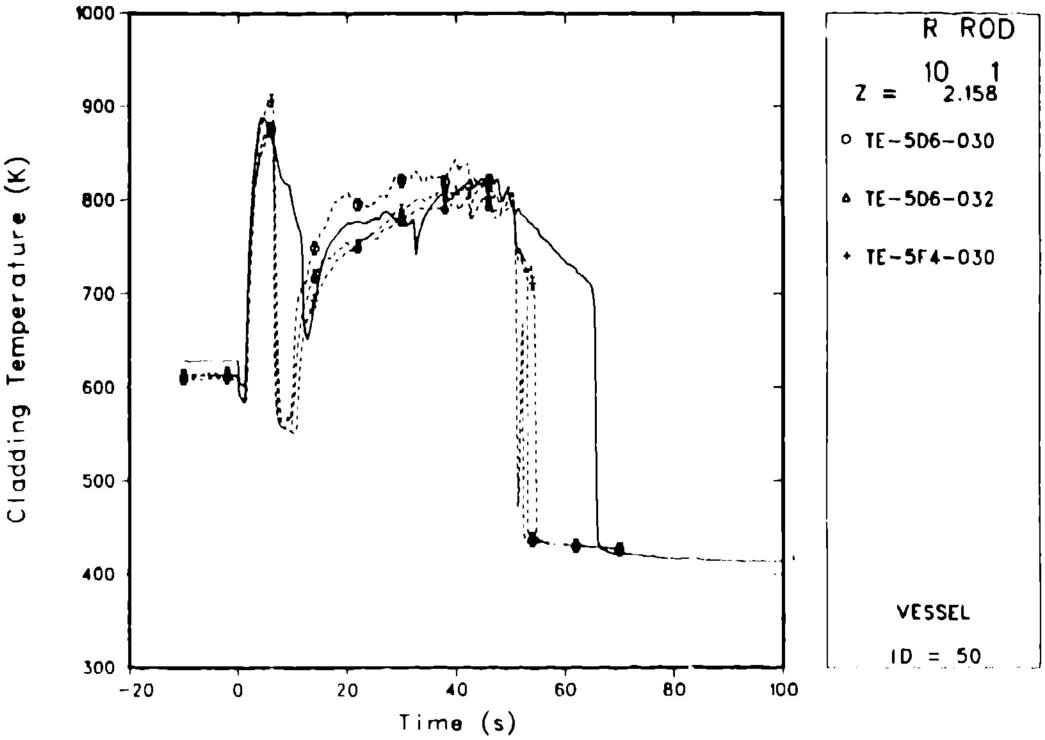

Fig. 4.
LOFT L2-3 calculated and measured cladding temperatures.

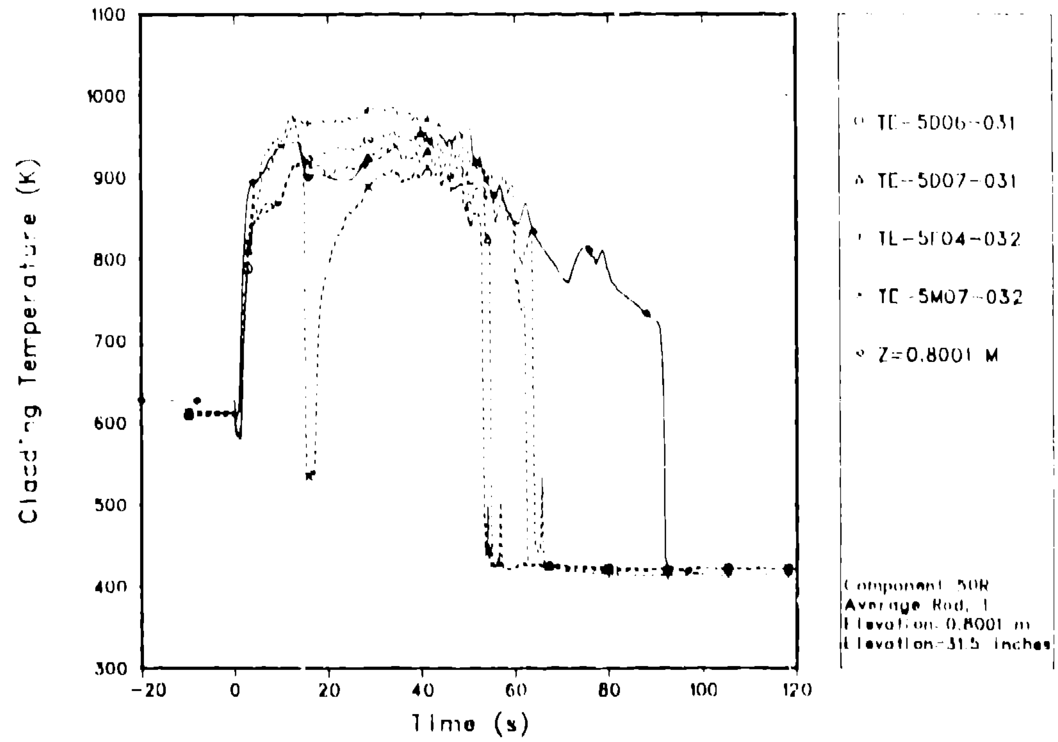

Fig. 5.
LOFT L2-5 calculated and measured cladding temperatures.

temperature correlations in the code that effectively prevent rewets from occurring at surface temperatures above ~700 K. Differences between the minimum film-boiling temperatures from the code and from the facility resulted in the calculated final quench being late.

The thermal response of the fuel cladding in Test L2-5 was very sensitive to the core hydraulics and to the power profiles in the core. As shown in Fig. 5, the code calculated the thermal response well at the 0.80-m elevation of the central fuel bundle. However, the code overpredicted the cladding temperatures at higher elevations and underpredicted the cladding temperatures at lower elevations; we attributed these differences to an incorrect axial power profile (which was later confirmed by the LOFT Program).

In summary, the calculations for both tests agreed qualitatively with the core flows that can be inferred from a number of instruments (no direct measurements of the core flows were made). For Test L2-3, the code correctly calculated the hydraulics leading to the early bottom-up rewet of the core. The code calculated the early top-down rewet of the top of the core that Test L2-5 exhibited; the top-down rewet was sensitive to the balance of flows into the core and to the axial power profile. During the analyses, we discovered a deficiency in the modeling associated with the emptying of the accumulator; we modified the code to alleviate the problem.

We analyzed a series of Semiscale natural-circulation and reflux-cooling tests with TRAC-PF1. Figure 6 compares the calculated system pressure to the data for Semiscale Test S-NC-2. Although this test was described as a quasi-steady-state test, we modeled it as a transient because of the appearance of the data. We also modeled the effect of the external heat losses and the guard heaters instead of implementing an assumed adiabatic boundary condition. For this test, Figs. 7-9 compare the calculated and measured loop flows as a function of the primary-system inventory for core powers of 30 kW, 60 kW, and 100 kW, respectively. These three figures demonstrate that the code correctly calculated the magnitude of the natural-circulation flows, including the effects of the decreasing primary-system inventory. The code also calculated the correct inventory for the transition from natural circulation to reflux cooling.

We also analyzed Semiscale Test S-NC-6, during which a series of nitrogen injections were made to investigate the effect of a noncondensable gas on the reflux cooling. Figure 10 compares the calculated and measured system pressures for Test S-NC-6. Again, we modeled the test as a transient. As the figure indicates, the code overpredicted the system pressure after the initial drain to establish single-component reflux cooling. We traced the discrepancy to the

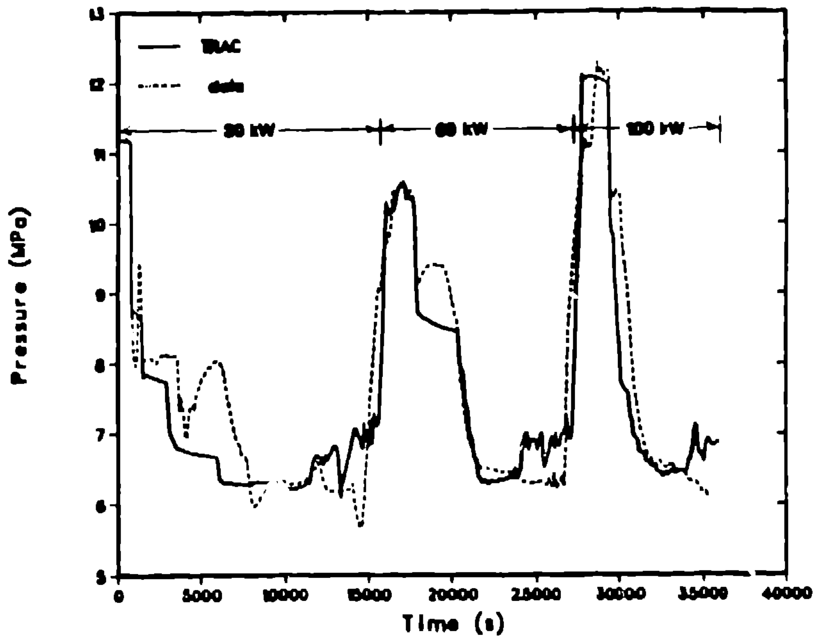

Fig. 6.
Semiscale S-NC-2 calculated and measured primary-system pressures.

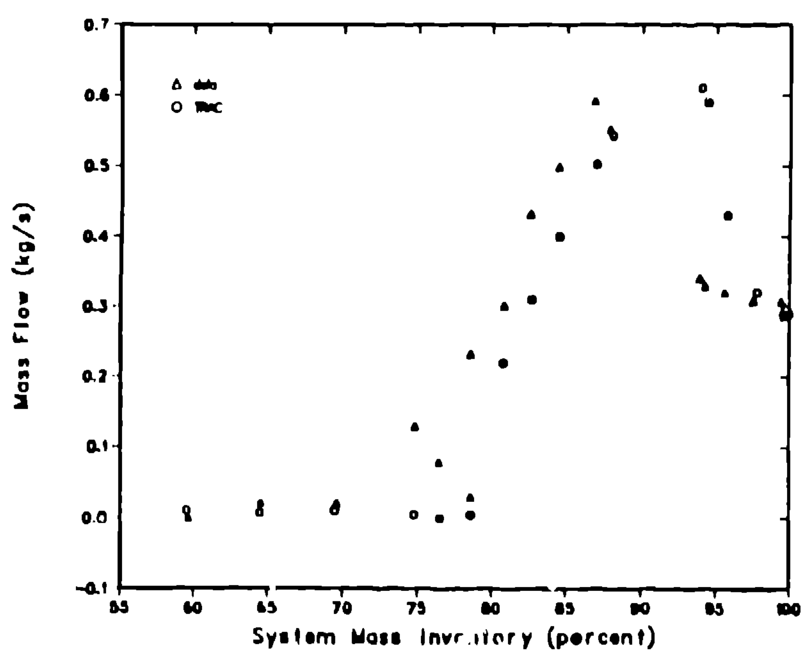

Fig. 7.
Semiscale S-NC-2 calculated and measured loop flows at 30-kW power.

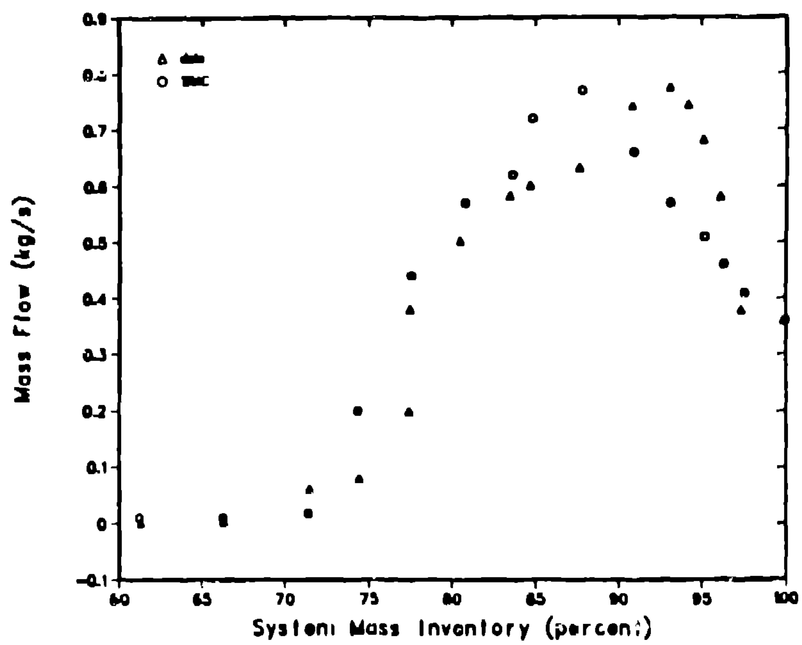

Fig. 8.
Semiscale S-NC-2 calculated and measured loop flows at 60-kW power.

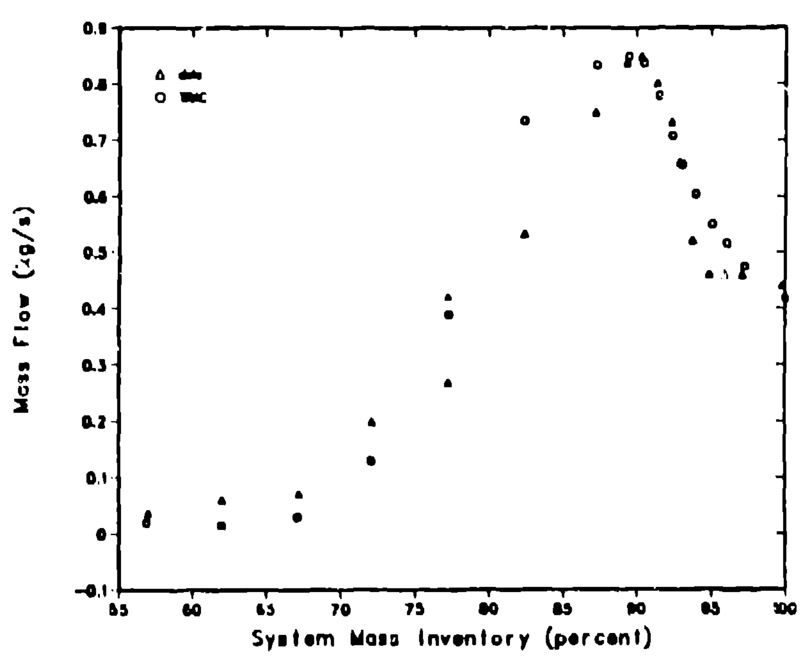

Fig. 9.
Semiscale S-NC-2 calculated and measured loop flows at 100-kW power.

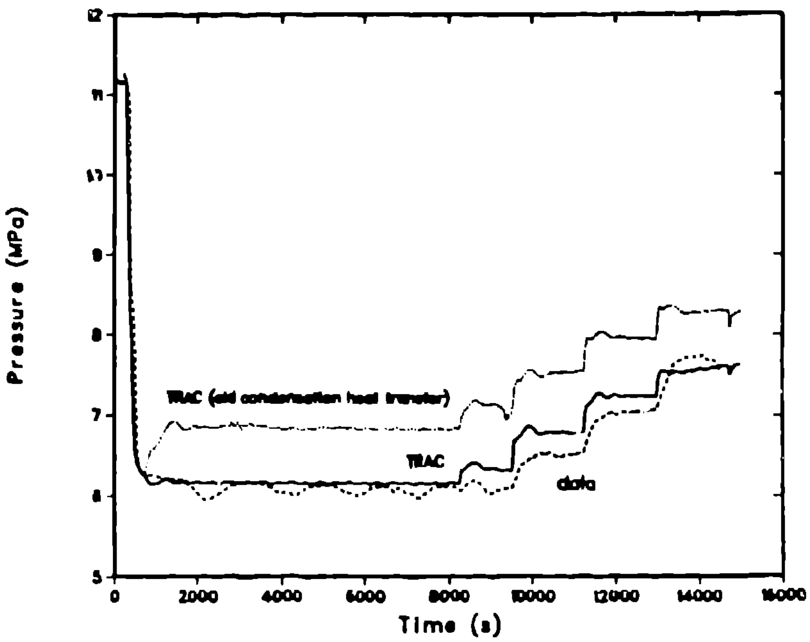

Fig. 10.
Semiscale S-NC-6 calculated and measured primary-system pressures.

wall-condensation heat-transfer correlation in the released version of TRAC-PF1. The use of the TRAC-PF1/MOD1 wall-condensation heat-transfer correlation increased the heat-transfer coefficients, and significantly improved the comparison.

The nitrogen injections caused step increases in the pressure after ~8000 s. The injections occurred in the hot leg just upstream of the steam generator; the nitrogen flowed with the vapor until the vapor condensed, leaving behind the nitrogen. As the nitrogen occupied more and more of the steam-generator tubes, there was less surface area available for heat transfer. The pressure increases resulted from increased primary-to-secondary fluid-temperature differences required to remove the heat from the core. The code correctly calculated this behavior.

These analyses demonstrated the capability of the code to calculate single- and two-phase natural circulation, reflux cooling, and the transition between natural circulation and reflux cooling. During the analyses (but not as a result of the analyses), we discovered several errors in the horizontal stratified-flow logic; therefore, we also tested and validated in TRAC-PF1 the modified logic appearing in TRAC-PF1/MOD1. We demonstrated that the wall-condensation heat transfer in TRAC-PF1 was low and showed that the modified correlation in TRAC-PF1/MOD1 was adequate for these analyses.

Our TRAC-PF1 calculations of LOFT L9-1/L3-3 and L6-7/L9-2, which simulated respectively a loss-of-feedwater transient and the cooling phase of the Arkansas Nuclear One Unit 2 turbine-trip transient (each test had compounding additional failures), demonstrated that the code could be applied successfully to non-LOCA transients. Figure 11 compares the calculated and measured pressurizer pressures for Test L9-1/L3-3. During the first ~1000 s, the pressurizer sprays controlled the system pressure; from ~1200-3300 s, the power-operated relief valve (PORV) controlled the pressure. At ~3300 s, the PORV was latched open to initiate the L3-3 portion of the transient. The calculation of the L3-3 portion of the transient was hampered by uncertainties in the PORV flow characteristics and in the steam-generator heat transfer (the secondary-side inventory distribution).

Figure 12 compares the calculated and measured pressurizer pressures for LOFT L6-7/L9-2. The calculated results compared well with the data. The major difference involved the natural-circulation flow established following pump trip and coastdown; the code calculated a flow that was approximately twice the measured flow. Although increasing the locked-rotor resistance in the pump

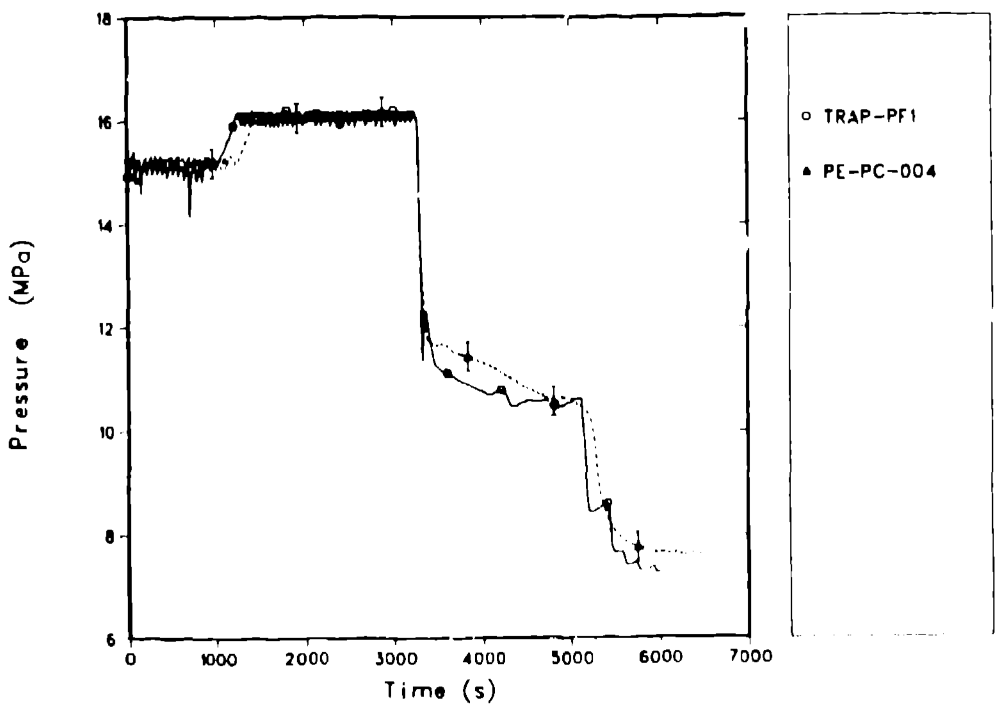

Fig. 11.
LOFT L9-1/L3-3 calculated and measured pressurizer pressures.

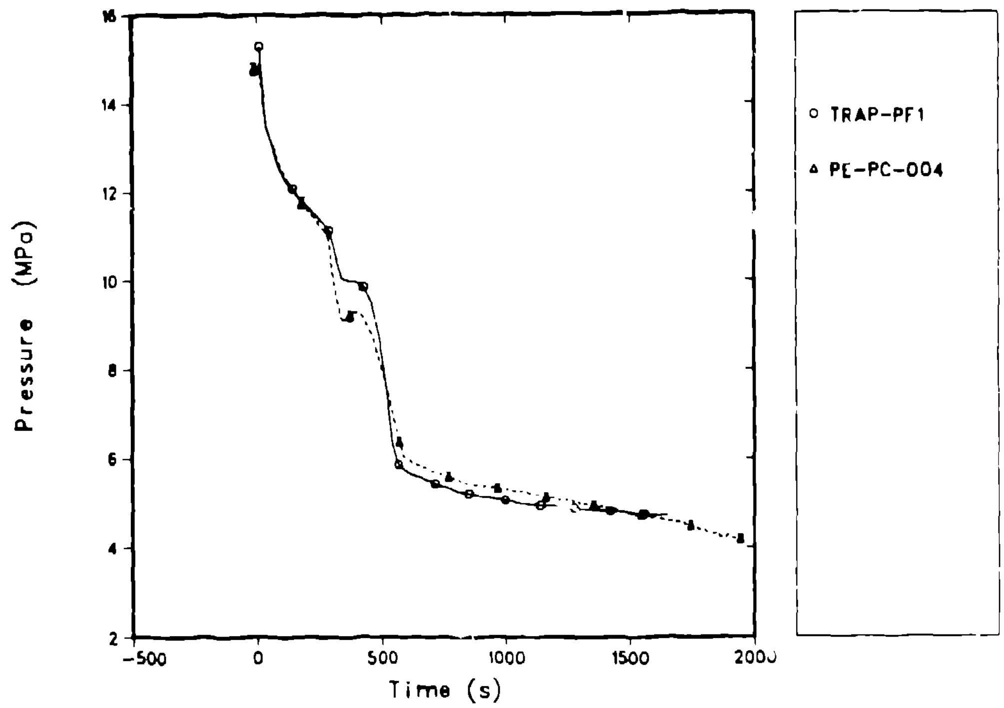

Fig. 12.
LOFT L6-7/L9-2 calculated and measured pressurizer pressures.

model could bring the flows into agreement, we believe that the measured flow is affected by thermal stratification in the downcomer reducing the driving head for natural circulation.

Except for low wall-condensation heat transfer, we did not discover any significant problems with code models through the analyses of these two LOFT tests. However, the analyses did demonstrate the necessity to represent all flow paths, including leakage paths that had been ignored in previous analyses, and all the structural mass and heat-transfer surfaces. This increased detail in the facility model is required to obtain the correct energy inventory and distribution, both of which can impact non-LOCA transients. These results point to the need for additional generality in the steam-generator component (provided in TRAC-PF1/MOD1), in the heat slabs, and for a plenum-type component with more connections than currently allowed with a tee component.

We also analyzed the Crystal River Unit 3 transient of February 26, 1980, with the TRAC-PF1 code. A loss of feedwater drove this transient. Figure 13 compares the calculated and measured primary-system pressures. The two oscillations in the calculated pressure resulted from the filling of individual cells in the hot-leg piping. Figure 14 compares the calculated and measured upper-plenum liquid temperatures, and clearly shows the effect of the calculated

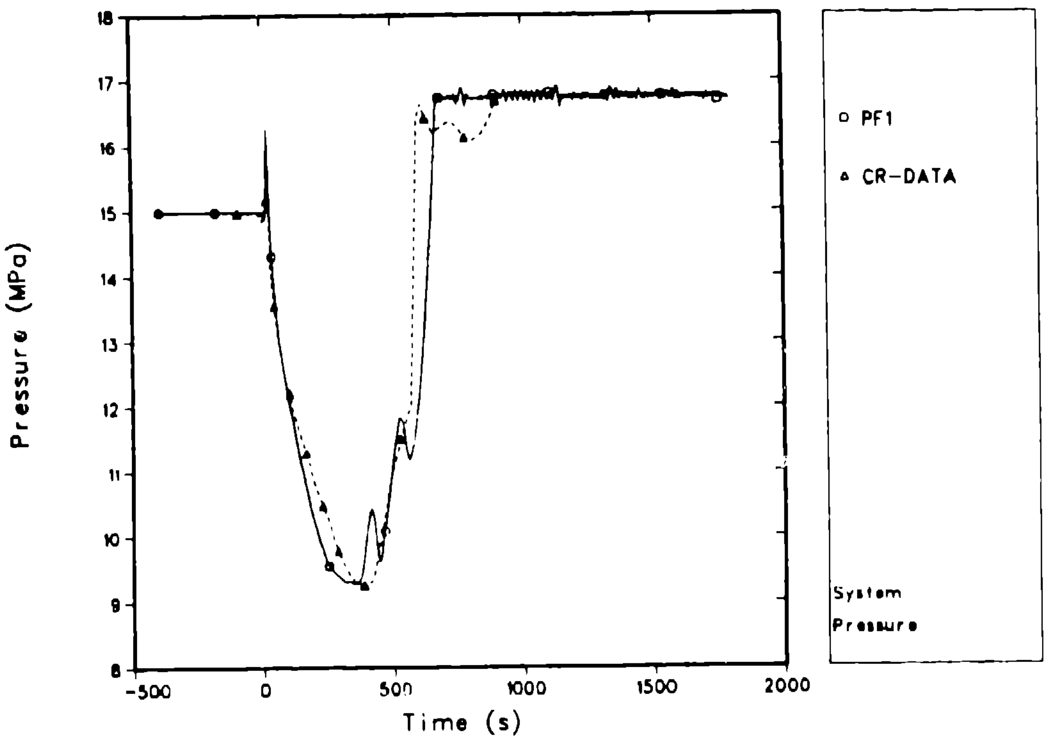

Fig. 13.
Crystal River calculated and measured primary-system pressures.

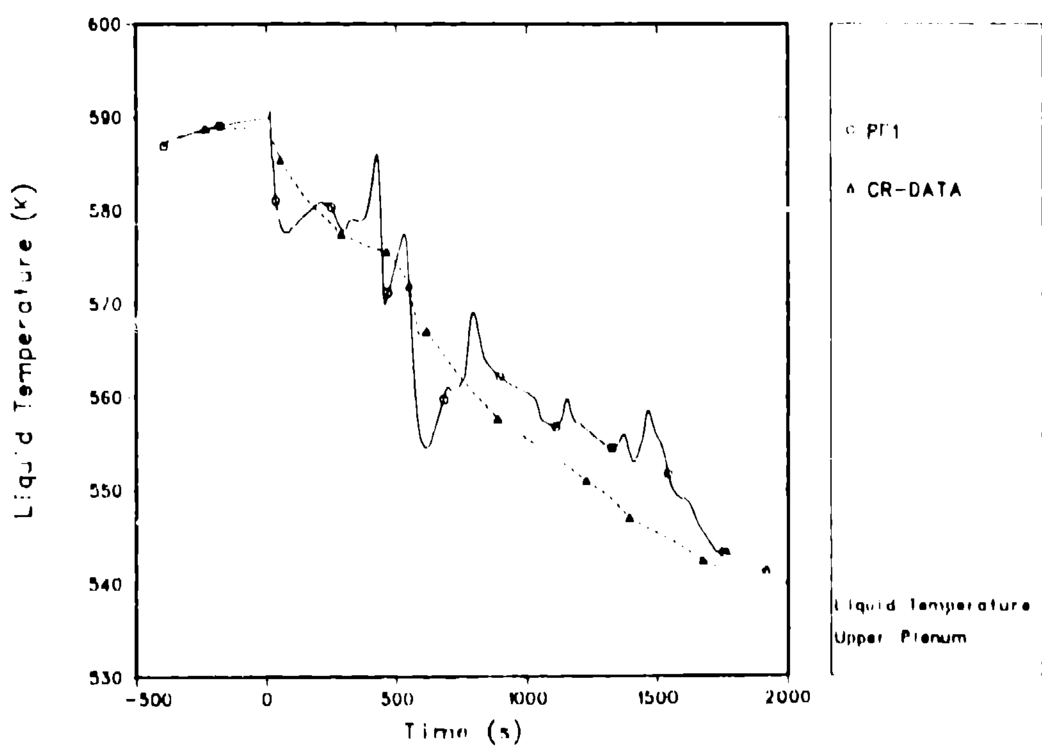

Fig. 14.
Crystal River calculated and measured upper-plenum liquid temperatures.

pressure oscillations. The early data are from alarm trips and probably should not be connected. Figures 15 and 16 compare the calculated and measured flows in the Loop A and Loop B cold legs, respectively. The code correctly calculated the stagnation of Loop A caused by a bubble forming in the Loop A hot leg. This analysis demonstrated that the code can be applied successfully to real transients in commercial power plants.

We are analyzing Semiscale Test S-UT-8, a small-break LOCA simulation, with TRAC-PF1/MOD1. These analyses are continuing; however, the preliminary results indicate that this code provides significant improvements over TRAC-PF1 in critical-flow modeling, certain constitutive relations, the calculation of the primary-system inventory distribution, and the flexibility of the input.

The ongoing assessment effort at Los Alamos indicates that the overall quality of the code improves as new code versions are released. Although the work continues to indicate needed improvements in the code, the TRAC series of codes currently provides a very flexible analysis tool for treating a wide variety of transients pertinent to PWRs.

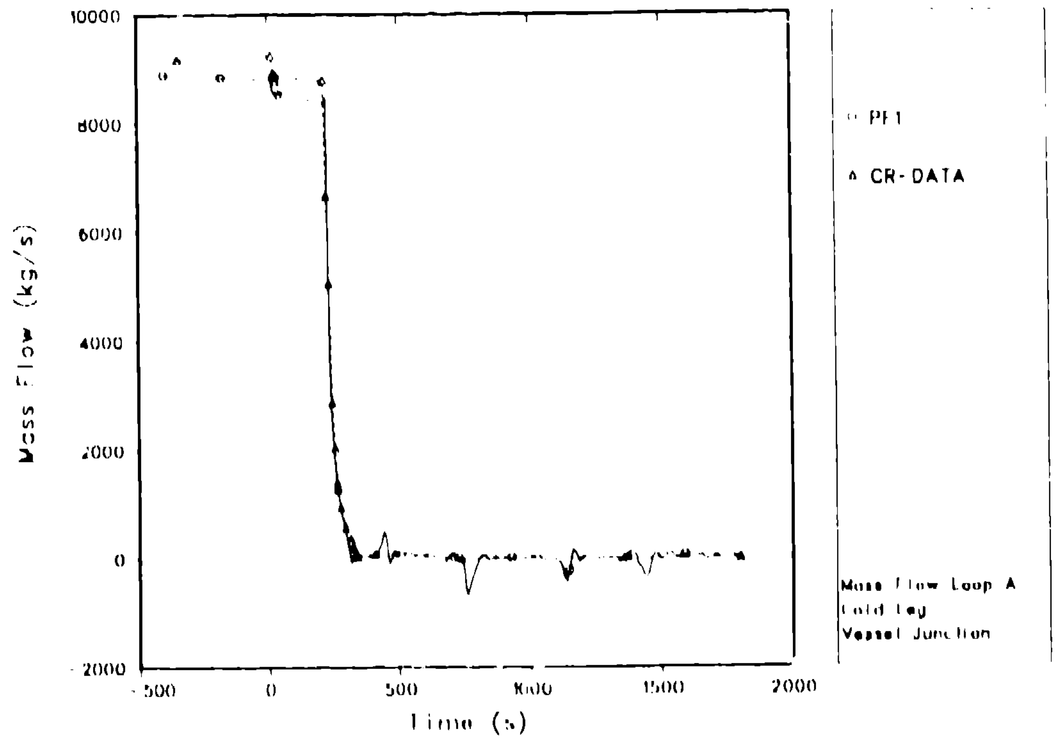

Fig. 15.
Crystal River calculated and measured Loop A cold-leg flows.

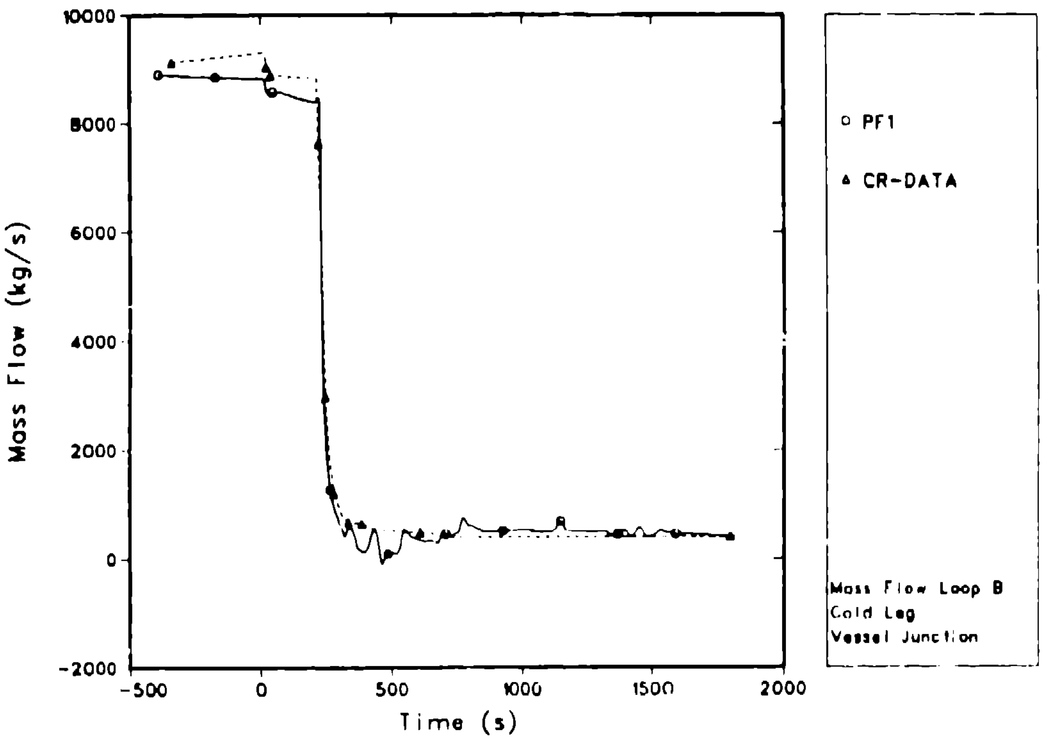

Fig. 16.
Crystal River calculated and measured Loop B cold-leg flows.