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MEMORANDUM FOR: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

FROM: Robert B. Minogue, Director
Office of Nuclear Regulatory Research

SUBJECT: RESEARCH INFORMATION LETTER - 125 - TRAC-PD2 "AN ADVANCED BEST-ESTIMATE COMPUTER PROGRAM FOR PWR LOCA ANALYSIS"

125

I. INTRODUCTION

TRAC-PD2 is the second in a planned series of three detailed accident analysis codes for PWR's. The first version, TRAC-P1A, was transmitted to the Office of Nuclear Reactor Regulation in Research Information Letter number 92 (Ref. 1), which identifies the user needs. TRAC-PD2 is currently being applied to the analysis of a variety of accidents in full-scale LWR's, including large-break LOCA, small-break LOCA and operational transients. The improvements of TRAC-PD2 over TRAC-P1A are documented in Section II, the results section. The evaluation of the code is given in Section III, while its application to problems of interest to NRC is detailed in Section IV. The evolution and mission of the various TRAC-PWR revisions are shown in Table I.

II. RESULTS: IMPROVEMENTS OF TRAC-PD2 OVER TRAC-P1A

The PD2 version of TRAC (Ref. 2) has many improvements over the original P1A version (Ref. 3):

1. A new reflood algorithm has been added to the TRAC-PD2 code to better model the axial conduction and precursory cooling effects in the local region around bottom refill and falling-film quench fronts. The algorithm uses an intermediate axial temperature noding (specified by the user) and a moving fine mesh centered around the quench fronts. This latter mesh is moved in a manner that conserves energy. Integrated heat transfer rates are then used to couple the temperature field solution to the fluid dynamics calculation. The temperature field solution is a mixed technique, implicit in the radial direction across the fuel rod and explicit in the axial direction around the quench fronts.

2. In TRAC-P1A the momentum source term for the connections to the vessel had a sign error that tended to reverse the flow at pipe-vessel junctions. In PD2 the signs are corrected so that the fluid momentum in the connecting pipes is conserved.

3. In TRAC-P1A, the calculation of the wall friction pressure drop in cells containing area restrictions used the velocity at the minimum flow area and applied this velocity over the entire cell length. This over-estimated the frictional pressure drop, because the constricted flow area extends only for a small distance. The PD2 version of the code uses an averaged velocity to calculate the wall friction pressure drop and a local (orifice) loss to account for local flow restriction.

4. The condensation regime heat transfer model has been improved. The improved formulation is more realistic and alleviates the pressure spikes observed in the P1A calculations.

5. The solution strategy in the three-dimensional vessel component has been improved, thereby reducing the execution time and permitting tighter convergence criteria.

6. Conservation of mass is achieved in PD2; it was not achieved in P1A.

7. Improvements have been made in the wall-heat-transfer correlations, constitutive equations, metal properties evaluation, thermodynamic property evaluations, and water packing treatment.

8. A simple dynamic gap-conductance model has been included.

9. The programming of one-dimensional components has been simplified by using common subroutines wherever possible.

10. The types of boundary conditions that can be imposed at BREAKS and FILLS have been expanded to include more fluid properties, such as void fraction and fluid temperature.

11. Graphics output files are now produced that are compatible with the new graphics postprocessing programs, EXCON and TRAP.

12. A broader range of experimental results has been used to assess the code.

The EXCON and TRAP graphics postprocessors are significant improvements on the GRED and GRIT programs which were previously available. The improvements incldue:

(1)- standard FORTRAN programming is used throughout, with a replaceable, high-level interface to the DISSPLA graphics software package,

(2) enhanced selection and merging facilities for information from multiple TRAC runs,

(3) varied presentation formats, including three-dimensional perspective plots, dependent-variable correlation plots, spatial independent variables, data comparisons capabilities, and motion picture capabilities,

(4) rod temperature plots utilizing variable mesh data during reflood,

(5) interactive command language, and

(6) user-defined functional capabilities.

II. EVALUATION: DEVELOPMENTAL ASSESSMENT OF TRAC-PD2 AGAINST DATA

Tests selected for the developmental assessment of PD2 are listed in Table II. This set includes most of the experiments used for P1A developmental assessment plus additional integral, systems, and heat-transfer tests. The assessment set includes separate effects (tests involving basically only one plant component and one LOCA phase), system effects (coupled components up to entire loops, but only one LOCA phase), and integral effects (system tests covering more than one LOCA phase) over a wide range of scales. Results indicate that PD2 does a reasonable job for all of these tests (Refs. 4 & 5). Improvements observed over P1A are mostly in the reflood heat-transfer area. However, as a result of numerous other improvements in solution strategy, numerics, and constitutive relations, PD2 is a much more reliable and smoother-running code than P1A. Running time is the same or improved over P1A even though the reflood heat-transfer treatment is more complex.

To illustrate the performance of PD2, we have selected an integral test (S-06-3) in the Semiscale facility and an integral test (L2-2) in the LOFT facility. Test S-06-3 was a large-break LOCA test with accumulator and high- and low-pressure injection into the intact loop cold leg (Ref 6). There is good agreement between the calculated and measured mass flow rates on the vessel side of the break (Figure 1). In the intact loop, TRAC predicts the rapid decrease in mass flow rate due to two-phase degradation in the pump. As shown in Figure 2, TRAC tended to somewhat underpredict the peak clad temperature (PCT) but the overall comparisons were good except for the high-power rods at the top of the core.

Test L2-2, the first nuclear-powered test in the LOFT facility, was a large-break LOCA from an initial power of 25 MWt and an intact hot-leg temperature of 580 K. The calculated hydraulic response generally agrees very well with the data (Ref. 7). The primary discrepancy is an initial underprediction of the accumulator discharge rate which delays the start of refilling of the lower plenum. However, the core refill is predicted reasonably well and the PCT is close to the observed value. Figure 3 compares the break flow (vessel side of break) and shows good agreement except for the initial period of subcooled critical flow (first 10 s). The cause of the underprediction during the first 10 s is being studied at BNL as part of the independent assessment of TRAC-PD2.

Figure 4 shows typical results for the cladding temperature response for Test L2-2 at the core midplane for the central fuel bundle (high-power zone). The data shown are from three neighboring thermocouples. Other thermocouples in this same fuel bundle and at the same elevation show significantly different behavior so that the spread in the measurements is much larger than that shown in the figure. The TRAC-PD2 results shown are typical for all the rods in the central power zone except that the rods adjacent to the broken hot-leg do not experience the second dryout (this was also observed in some of the measurements). Both the calculation and data show a series of dryouts and rewets with the peak clad temperature occurring during blowdown. Comparisons at other elevations and in the intermediate- and low-power zones are similar to those shown in Figure 4.

IV. APPLICATION OF TRAC-PD2 TO FULL-SCALE LWR's

The primary mission of TRAC-PD2 is the analysis of large-break LOCA's in Pressurized Water Reactors. Enclosure 3 lists the variety of full-scale LWR analyses being performed with TRAC-PD2 at LANL. As can be seen, the code is being used for analysis of both large and small break LOCA's as well as operational transients.

DF Ross / for

Robert B. Minogue, Director
Office of Nuclear Regulatory Research

Enclosures:
1. "TRAC-PD2 An Adv. Best-Est. Prg. for PWR LOCA Analysis," NUREG/CR-2054, April 1981
2. J. C. Vigil, "TRAC-PD2 Dev. January 1981
3. Ltr., J. Ireland, LANL, to L. Shotkin, NRC, July 20, 1981

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T. Speis, NRR
J. Mazetis, NRR
B. Sheron, NRR
W. Hodges, NRR
G. Knighton, NRR
E. Throm, NRR
W. Jensen, NRR
J. Guttmann, NRR
N. Lauben, NRR
R. Audette, NRR

REFERENCES

1. Memorandum, R. Budnitz, RES to H. Denton, NRR, Research Information Letter No. 92, TRAC-P1A, June 18, 1980.

2. "TRAC-PD2, An Advanced Best-Estimate Computer Program for PWR LOCA Analysis," NUREG/CR-2054, April 1981.

3. "TRAC-P1A, An Advanced Best-Estimate Computer Program for PWR LOCA Analysis," NUREG/CR-0665, May 1979.

4. J. C. Vigil, "TRAC-PD2 Developmental Assessment Summary," Los Alamos National Laboratory informal report LA-UR-81-93, January 1981.

5. "TRAC-PD2 Developmental Assessment," Los Alamos National Laboratory report, to be issued in 1981.

6. B. L. Collins, M. L. Patton, Jr., K. E. Sackett, and K. Stanger, "Experiment Data Report for Semiscale MOD-1 Test S-06-3 (LOFT Counterpart Test)," EG&G Idaho, Inc. report NUREG/CR-0251, July 1978.

7. "Experiment Data Report for LOFT Power Ascension Test L2-2," Idaho National Engineering Laboratory report NUREG/CR-0492, 1979.

TABLE I

EVOLUTION AND MISSION OF TRAC-PWR VERSIONS

EVOLUTION OF ADDED CAPABILITIES		PRIMARY MISSION
DETAILED VERSION	FAST-RUNNING CAPABILITIES	
TRAC-P1A		LARGE-BREAK LOCA
TRAC-PD2		LARGE-BREAK LOCA
	TRAC-PF1	SMALL-BREAK LOCA
	TRAC-PF1/MOD1	OPERATIONAL TRANSIENTS
TRAC-PD3		ATWS/RIA

TABLE II

TRAC-PD2 DEVELOPMENTAL ASSESSMENT EXPERIMENTS

No.	Experiment	Scale	Thermal-Hydraulics Effects
1	Edwards Horizontal Pipe Blowdown (Standard Problem 1)	1/100[a]	One-dimensional separate effects during blowdown including critical flow, flashing, slip, and wall friction.
2	CISE Unheated Vertical Pipe Blowdown (Test 4)	1/1200[a]	Same as 1 plus pipe-wall heat transfer, flow area changes, and gravitational effects.
3	CISE Heated Vertical Pipe Blowdown (Test R)	1/1200[a]	Same as 2 plus critical heat flux (CHF).
4	Marviken Vessel Blowdown-Long Nozzle (Test 4)	1/1[b]	Same as 1 plus full-scale effects and delayed nucleation effects.
5	Marviken Vessel Blowdown-Short Nozzle (Test 24)	1/1[b]	Same as 4 plus nonequilibrium, two-dimensional nozzle flow.
6	THTF Blowdown Heat-Transfer Test 177	1/1[c]	Separate effects during blowdown including rod heat transfer with dryout and rewet.
7	Creare Downcomer tests (3) - Low ECC Subcooling	1/15[d]	Two-dimensional separate effects during refill including countercurrent flow, interfacial drag, and downcomer penetration.
8	Creare Downcomer tests (3) - High ECC Subcooling	1/15[d]	Same as 7 plus condensation effects.
9	FLECHT Forced Flooding Tests (PWR Tests 4831 and 17201, SEASET Test 4)	1/1[e]	One-dimensional separate effects during reflood including heat transfer, quench-front propagation, liquid entrainment, and carryover.
10	Bennett Vertical Tube CHF (Tests 5336, 5431, and 5442)	1/1[f]	One-dimensional pipe-wall steady-state heat transfer over the entire range of the boiling curve.
11	Semiscale Heated Blowdown Test S-02-8	1/2000[g]	Synergistic and systems effects during blowdown in a multiloop PWR simulator.
12	Semiscale Integral LOCA Test S-06-3	1/2000[g]	Integral effects during a complete LOCA in a multiloop PWR simulator.
13	Nonnuclear LOFT Blowdown (Test L1-4)	1/60[g]	Integral effects during isothermal blowdown and refill in a PWR simulator (nuclear core not in place).
14	Nuclear LOFT Integral LOCA (Test L2-2)	1/60[g]	Integral effects during a large-break LOCA in a scaled PWR.
15	CCTF Reflood Test C1-1	1/1[h]	Multidimensional and system effects during refill and reflood.

[a]Scale given is based on pipe flow area.

[b]Scale based on vessel and break pipe dimensions.

[c]Full-scale 7x7 array of electrically heated rods.

[d]Linear downcomer dimensions.

[e]Single bundle of ~ 100 electrically heated full-scale rods.

[f]Full-scale compared to fuel rod dimensions -- flow inside the tube is nonprototypic.

[g]Power and volume scaling.

[h]Full-height components; radius of electrically heated core is 1/5 scale.

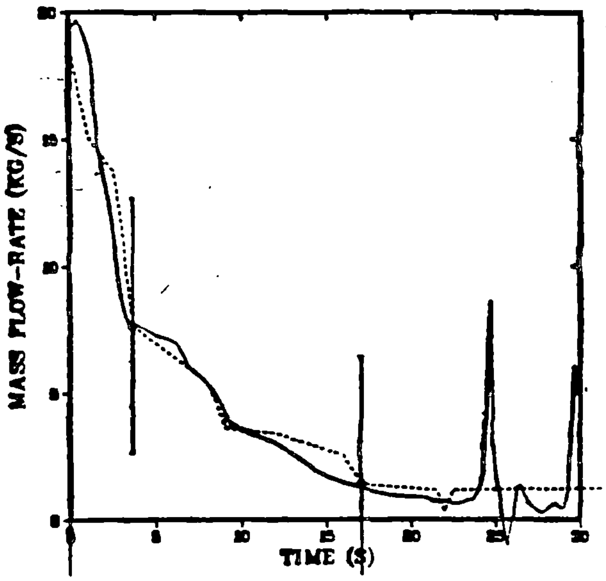

Fig. 1. Break flow (vessel side) for Semiscale LOCA test S-06-3 (solid = PD2, dash = data).

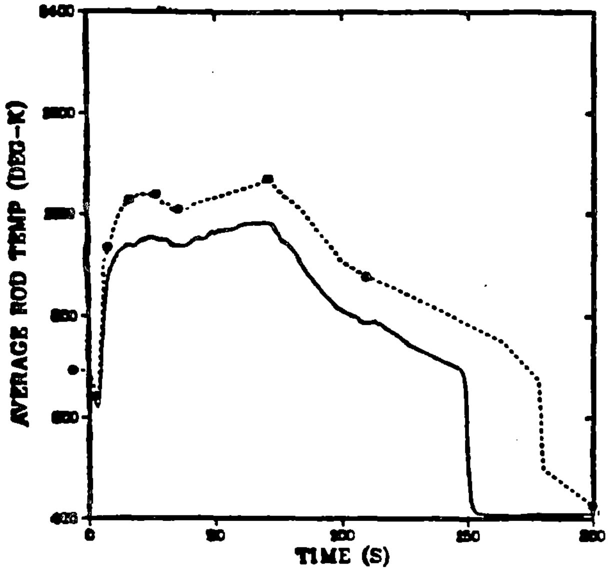

Fig. 2. PCT for Semiscale LOCA test S-06-3 (solid = PD2, dash = data).

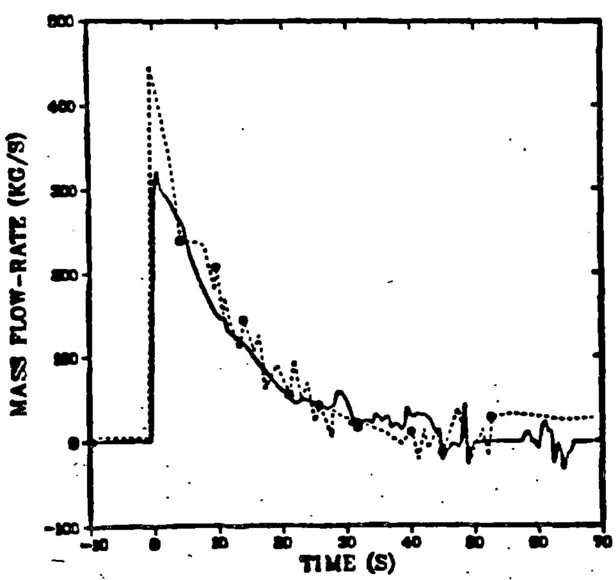

Fig. 3. Break flow (vessel side) for LOFT LOCA test L2-2 (solid = PD2, dash = data).

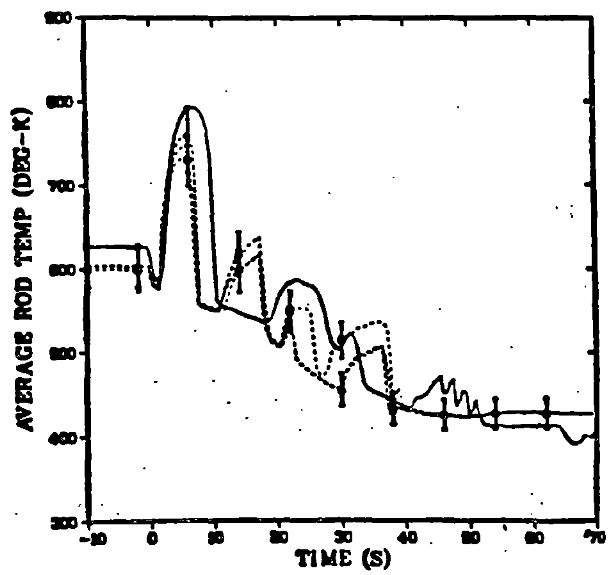

Fig. 4. PCT for LOFT LOCA test L2-2 (solid = PD2, dash = data).

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ENCLOSURE 2

TITLE: TRAC-PD2 Developmental Assessment Summary

AUTHOR(S): J. C. Vigil

SUBMITTED TO: Advisory Committee on Reactor Safeguards
Emergency Core Cooling (ECC) Subcommittee Meeting
Albuquerque, NM
January 14, 1981

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DEPARTMENT OF ENERGY
CONTRACT W-7405-ENG. 36

TRAC-PD2 DEVELOPMENTAL ASSESSMENT SUMMARY*

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ABSTRACT

TRAC-PD2 is the latest release version in a series of PWR system codes being developed at the Los Alamos National Laboratory. This paper presents a summary of developmental testing of TRAC-PD2 against separate-, system-, and integral-effects experiments covering a wide range of scales. The results show that TRAC-PD2 does a credible job overall and that significant improvements have been made over the previous version, TRAC-P1A.

I. INTRODUCTION

The Transient Reactor Analysis Code (TRAC) is an advanced best-estimate systems code for the analysis of loss-of-coolant accidents (LOCAs) and other thermal-hydraulic transients in light-water reactors (LWRs). This document provides a summary of the developmental assessment results for TRAC-PD2. TRAC-PD2 is the latest released version in a series[1-3] of pressurized water reactor (PWR) transient analysis computer programs being developed at the Los Alamos National Laboratory under the sponsorship of the Nuclear Regulatory Commission, Division of Reactor Safety Research. TRAC-PD2 differs from its predecessor (TRAC-P1A) primarily in its more detailed treatment of reflood heat transfer, accurate mass accounting for long transients, and overall increased reliability. It can be considered to be the first production version of TRAC.

Developmental assessment is the first stage of a two-stage testing process. It involves testing the code against a wide variety of thermal-hydraulic experimental data. The second phase, called independent assessment, begins after the code is publically released and is designed to test the

*This work performed under the auspices of the US Nuclear Regulatory Commission

predictive capability of the code when applied to new tests involving different scales and experimental configurations. Independent assessment of TRAC-PD2 is already under way and includes, for example, pretest predictions of the LOFT small-break tests. The code is also being applied to a variety of postulated transients (e.g., loss of feedwater) in full-scale power plants.

The experimental tests selected for developmental assessment of TRAC-PD2 are listed in Table I. This set includes most of the experiments used for TRAC-P1A developmental assessment[4] plus additional integral, systems, and heat-transfer tests. Note that the assessment set includes separate effects (tests involving basically only one plant component and one LOCA phase), system effects (coupled components up to entire loops but only one LOCA phase), and integral effects (system tests covering more than one LOCA phase) over a wide range of scales.

Brief summaries of the results for each developmental assessment calculation are given in Sec. II. The results were obtained with the same code version (TRAC-PD2) and are reported in detail in a separate document.[5] Conclusions that can be drawn from these results are summarized in Sec. III.

II. DATA COMPARISONS

A. Edwards Horizontal Pipe Blowdown Experiment

This experiment involved the depressurization of a straight horizontal insulated pipe (0.073 m ID by 4.1 m long) initially filled with stagnant subcooled water at approximately isothermal conditions. The flow-area scale of this experiment is approximately 1/100 compared to the cold-leg piping of a full-scale PWR. A glass rupture disk at one end of the pipe was broken to initiate the blowdown.

TRAC-PD2 calculations are in reasonable agreement with available experimental measurements[6] of fluid pressures and temperatures and with the single density measurement. An example of the typical agreement obtained is given in Fig. 1 which shows the pressure at gage station 1 (about 0.17 m from the broken end of the pipe). The PD2 results are very similar to those obtained with P1A but show slightly better agreement with the test data. Measurements

TABLE I. TRAC-PD2 DEVELOPMENTAL ASSESSMENT EXPERIMENTS

No.	Experiment	Scale	Thermal-Hydraulics Effects
1	Edwards Horizontal Pipe Blowdown (Standard Problem 1)	1/100[a]	One-dimensional separate effects during blowdown including critical flow, flashing, slip, and wall friction
2	CISE Unheated Vertical Pipe Blowdown (Test 4)	1/1200[a]	Same as 1 plus pipe-wall heat transfer, flow area changes, and gravitational effects
3	CISE Heated Vertical Pipe Blowdown (Test R)	1/1200[a]	Same as 2 plus critical heat flux (CHF)
4	Marviken Vessel Blowdown-Long Nozzle (Test 4)	1/1[b]	Same as 1 plus full-scale effects and delayed nucleation effects
5	Marviken Vessel Blowdown-Short Nozzle (Test 24)	1/1[b]	Same as 4 plus nonequilibrium, two-dimensional nozzle flow
6	THTF Blowdown Heat-Transfer Test 177	1/1[c]	Separate effects during blowdown including rod heat transfer with dryout and rewet
7	Creare Downcomer tests (3) - Low ECC Subcooling	1/15[d]	Two-dimensional separate effects during refill including counter-current flow, interfacial drag, and downcomer penetration
8	Creare Downcomer tests (3) - High ECC Subcooling	1/15[d]	Same as 7 plus condensation effects
9	FLECHT Forced Flooding Tests 4831 and 17201, SEASET Test 4	1/1[e]	One-dimensional separate effects during reflood including heat transfer, quench-front propagation, liquid entrainment, and carryover
10	Bennett Vertical Tube CHF (Tests 5336, 5431, and 5442)	1/1[f]	One-dimensional pipe-wall steady-state heat transfer over the entire range of the boiling curve
11	Semiscale Heated Blowdown Test S-02-8	1/3000[g]	Synergistic and systems effects during blowdown in a multiloop PWR simulator

TABLE I (con)

12	Semiscale Integral LOCA Test S-06-3	1/3000[g]	Integral effects during a complete LOCA in a multiloop PWR simulator
13	Nonnuclear LOFT Blowdown (Test L1-4)	1/60[g]	Integral effects during isothermal blowdown and refill in a PWR simulator (nuclear core not in place)
14	Nuclear LOFT Integral LOCA (Test L2-2)	1/60[g]	Integral effects during a large-break LOCA in a scaled PWR
15	CCTF Reflood Test C1-1	1/1[h]	Multidimensional and system effects during refill and reflood

a. Scale given is based on pipe flow area

b. Scale based on vessel and break pipe dimensions

c. Full-scale 7 x 7 array of electrically heated rods

d. Linear downcomer dimensions

e. Single bundle of ~100 electrically heated full-scale rods

f. Full scale compared to fuel rod dimensions - flow inside the tube is nonprototypic

g. Power/Volume scaling

h. Full-height components; radius of electrically heated core is 1/5 scale

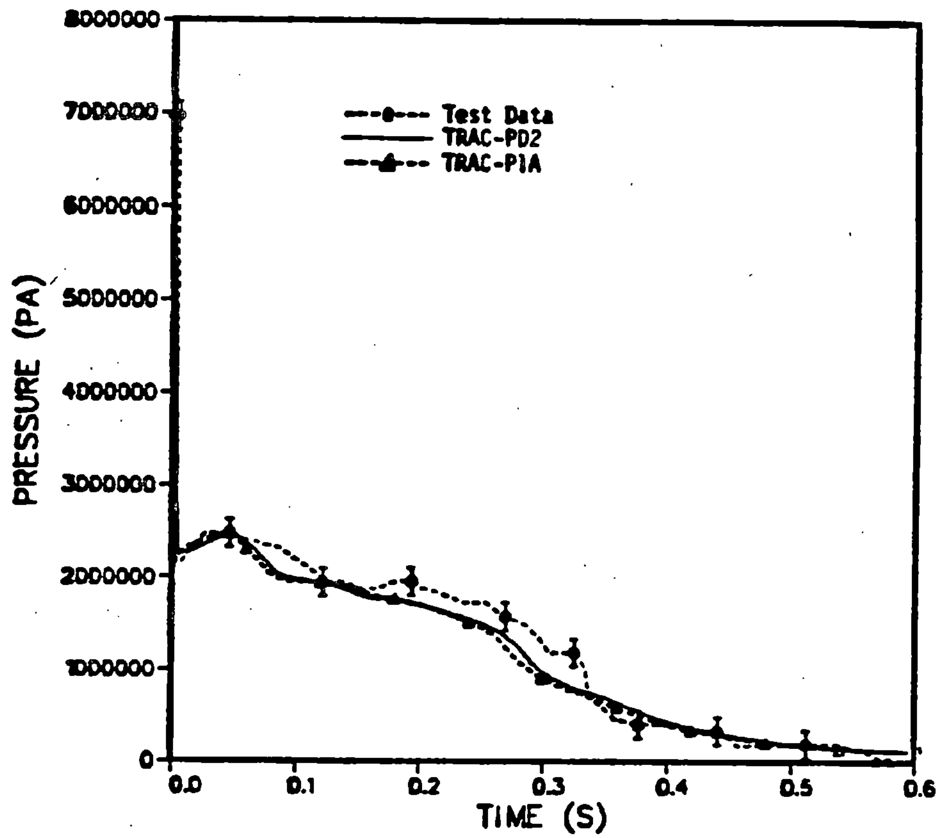

Fig. 1. Fluid pressure near the break for Edwards blowdown experiment.

of mass flow and pipe wall temperature were not made. In addition, there are experimental uncertainties in the initial temperature distribution, rupture disk dynamics, and the effect of residual disk fragments (about 13% of the pipe area) on the flow field. The calculated results are sensitive to these uncertainties and also to the wall friction factor correlation used. The TRAC model for this problem contains 46 fluid cells.

B. CISE Vertical Pipe Blowdown Experiments

In the CISE (Centro Informazoni Studi Esperienze) experiments,[7] subcooled water was circulated through a tubular test section consisting of a coiled insulated feeder (0.017 m ID by 9.9 m long), a straight vertical uninsulated pipe (0.021 m ID by 4.15 m long) whose walls could be electrically heated, and a coiled insulated riser (0.026 m ID by 10.0 m long). Compared to the cold leg of a PWR, the flow-area scale of this experiment is about 1/1200.

The blowdown was initiated by simultaneously closing two test section isolation valves and opening a discharge valve at the bottom of the feeder section. In the reference test (Test R) there was a power input of 110 kW to the heater section during the blowdown, whereas in Test 4 there was no power input.

TRAC-PD2 best-estimate calculations of the CISE tests are in good overall agreement with the measured data, including fluid pressure and temperature at several locations in the test section, pipe wall temperature in the heater section, and mass holdup measurements. Figures 2 and 3 are typical of the results obtained for these tests and show, respectively, the fluid pressure near the break and the pipe wall temperature near the top of the heater section for the heated test. TRAC-PD2 and P1A yield almost exactly the same hydraulic response. In the heated CISE test, the heater wall experiences dryout during the blowdown. TRAC-PD2 yields better results during the cooldown but the time to dryout is not as good as P1A. Because of the large length-to-diameter ratio, the calculated results are very sensitive to the wall friction factor correlation. Pipe wall stored energy also has a significant effect on the computed results. The TRAC models for the CISE problems contained 38 fluid cells.

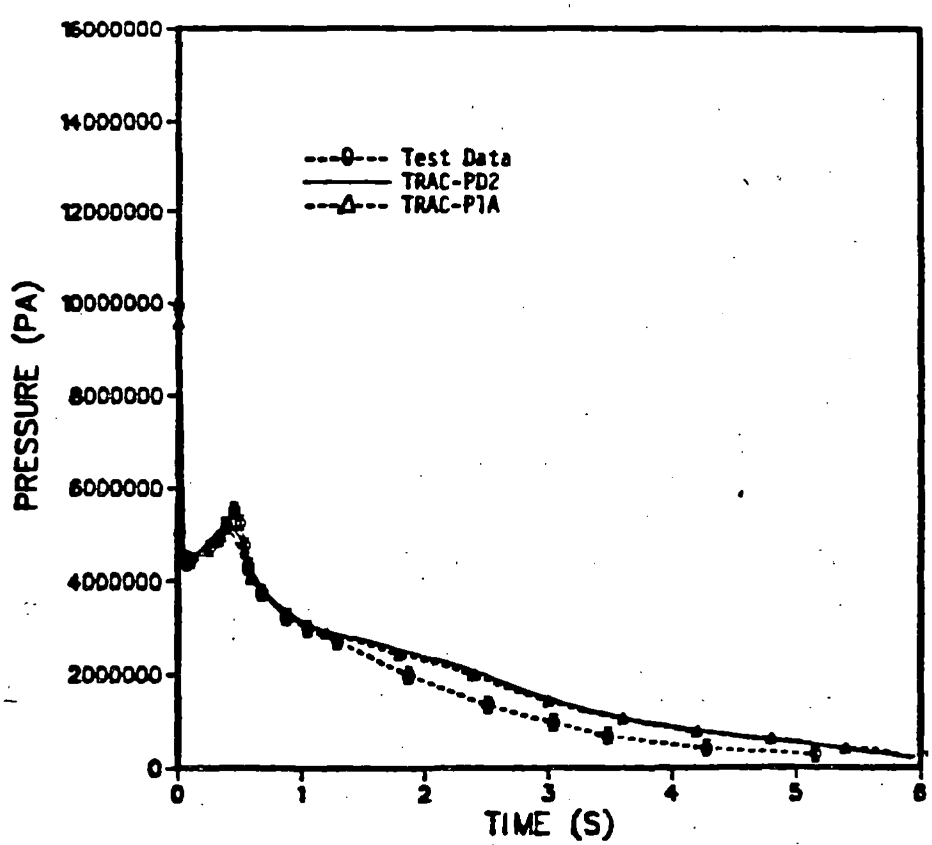

Fig. 2. Fluid pressure near the break for CISE heated blowdown test.

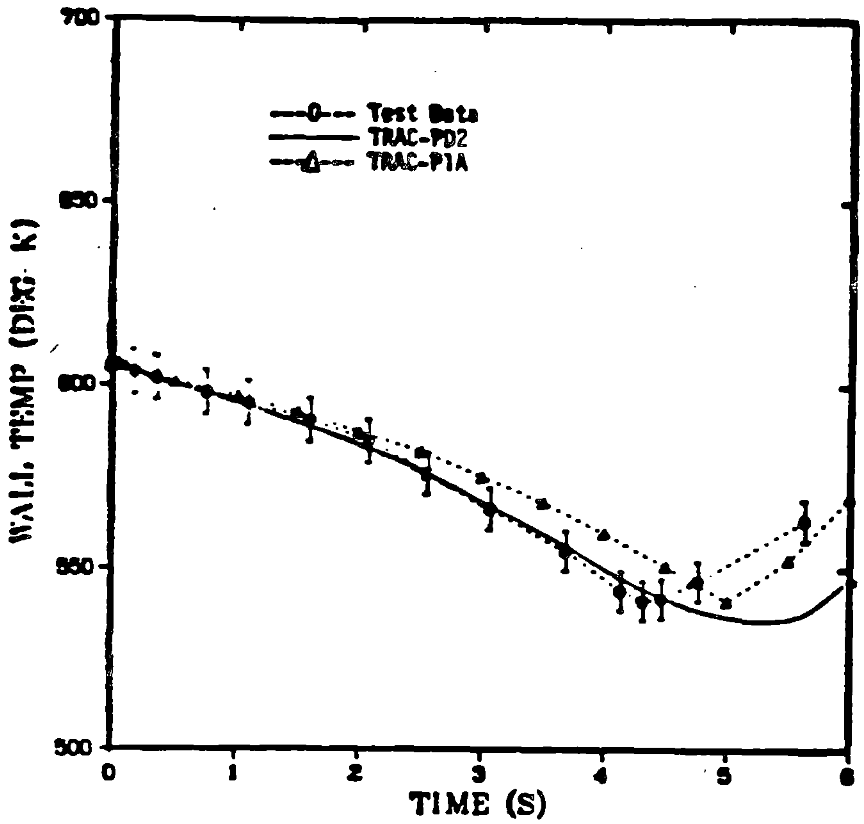

Fig. 3. Heater wall temperature for CISE heated blowdown test.

C. Marviken Full-Scale Vessel Blowdowns

The Marviken critical flow tests are designed to determine how well code models that were developed using small-scale experiments actually apply to full-scale systems. These tests involve the blowdown of a large (5.2 m ID by 21.5 m high) pressure vessel through a discharge pipe (0.75 m ID by 6.3 m long) which protrudes 0.74 m into the bottom of the vessel. A nozzle with a minimum diameter of 0.5 m was attached to the bottom of the discharge pipe. The nozzle length-to-diameter ratios were 3.1 and 0.33 for Tests 4 and 24, respectively. The blowdown is initiated by overpressurizing the gap between two rupture disks at the downstream end of the nozzle.

Little difference was observed between the PD2 and P1A results for these tests. For the long nozzle case (Test 4)[8], the TRAC results are in good overall agreement with pressure, temperature, and flow rate measurements. This is illustrated in Fig. 4 which compares the calculated flow rate with data derived from differential pressure and Pitot tube measurements. (The

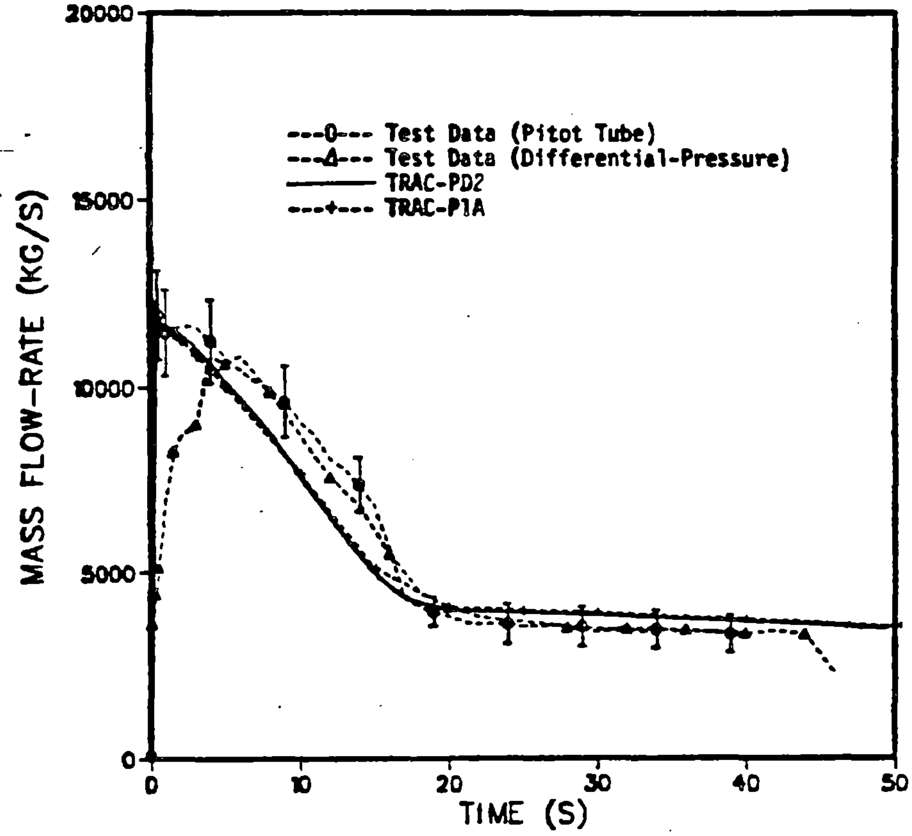

Fig. 4. Marviken Test 4 break mass flow rate.

data derived from differential pressure is valid only after ~ 5 s.) The agreement is not as good for the short nozzle case (Test 24)[9] as can be seen in Fig. 5. Nonequilibrium effects resulting from delayed nucleation, as well as two-dimensional flow effects, become increasingly important in the nozzle as L/D decreases. Neither of these effects is modeled in TRAC at this time. Delayed nucleation is enhanced in the Marviken tests by the use of deionized water. In an actual PWR, however, the coolant conditions are not likely to support this phenomenon. The operational procedure used prior to the blowdown resulted in rather complicated initial temperature distributions in the vessel; the calculated results are sensitive to these initial conditions. The TRAC model contained 60 fluid cells for Test 4 and 42 cells for Test 24.

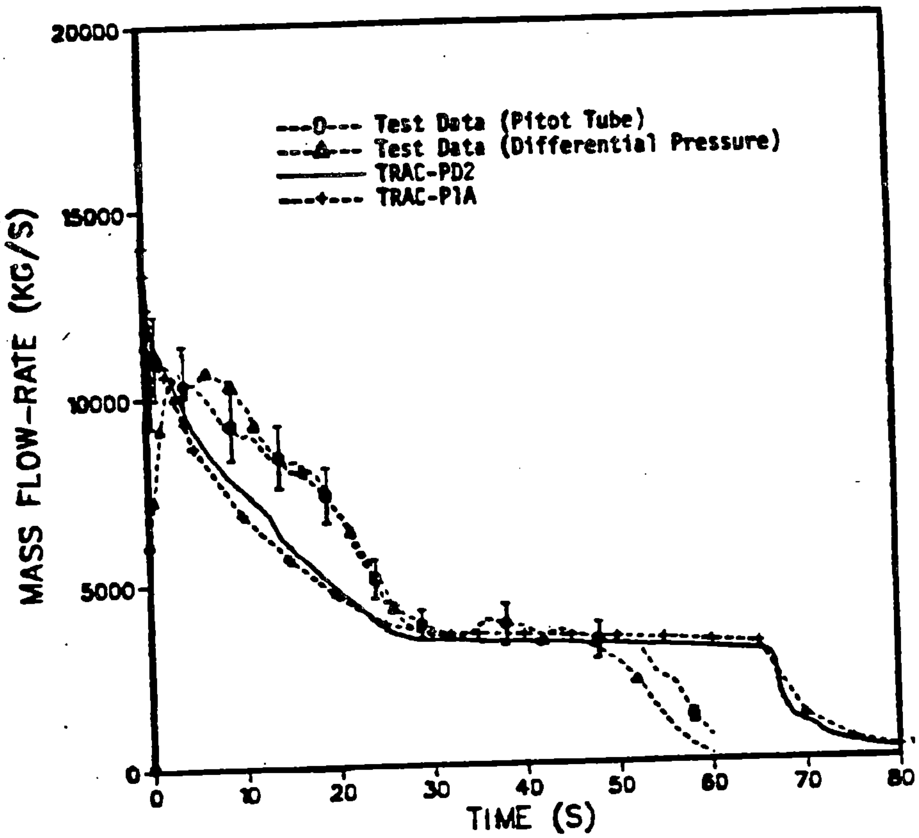

Fig. 5. Marviken Test 24 break mass flow rate.

D. THTF Blowdown Heat-Transfer Test 177

The Thermal-Hydraulic Test Facility (THTF) is a nonnuclear pressurized-water loop containing a 7x7 electrically heated rod bundle which simulates a PWR core. The full-scale fuel-pin simulators have a 3.66 m heated length and 0.011 m diameter. In Test 177 the blowdown is initiated by simultaneous opening of vessel inlet and outlet rupture disks. Power to the rods was decreased sharply during the first 4 s, returned to ~ 60% of full power at 5 s, and decayed to zero at 10 s. The test section was represented with a two-dimensional slab model containing 36 fluid cells.

The clad temperature response near the bottom of the core is shown in Fig. 6. Note that the measurements[10] show a wide range of responses for the rods at this elevation. The calculated average-rod response is generally within the spread in the measurements and includes the major features in the data.

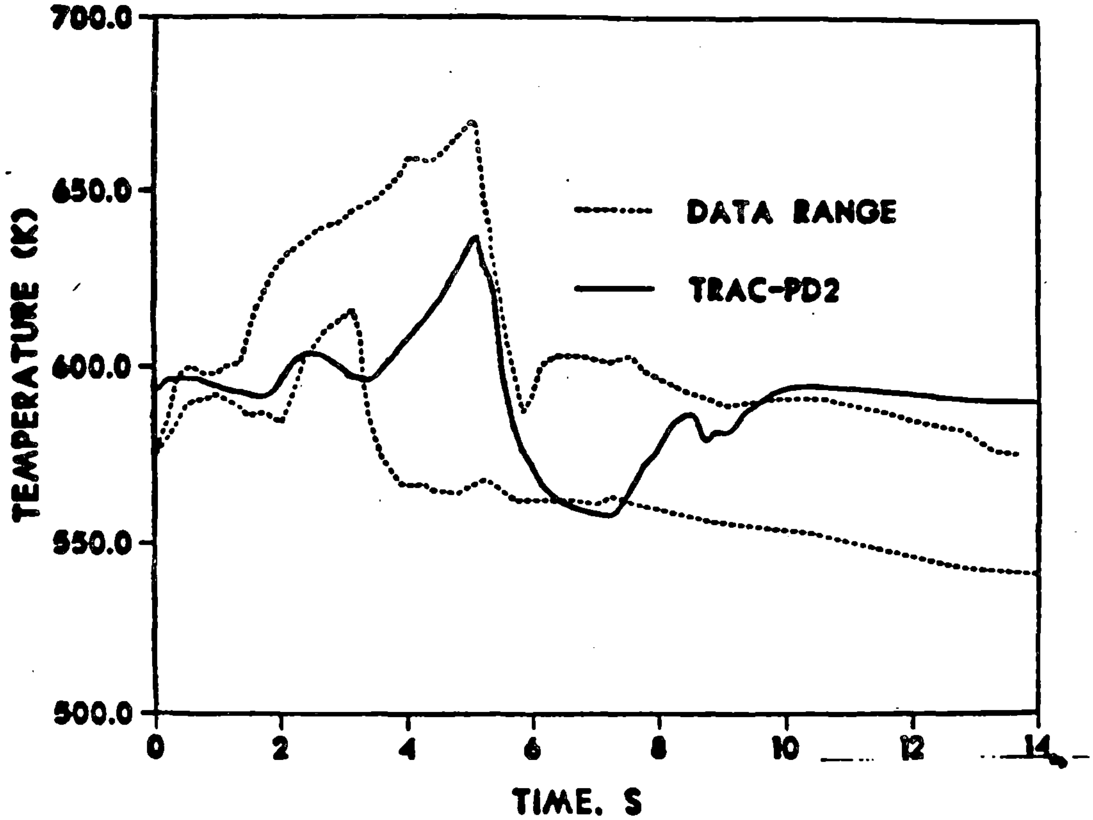

Fig. 6. Clad temperature response near the core bottom for THTF Test 177.

Similar agreement was obtained at the core midplane. However, near the top of the core the code failed to predict a rewet at about 5 s and consequently overpredicted the peak temperature at that elevation. This test was not calculated with P1A.

E. Creare Quasi-Static Downcomer Experiments

The primary purpose of the Creare 1/15-scale downcomer experiments[11] was to study the effect of countercurrent steam flow rate, emergency core coolant (ECC) subcooling, and downcomer wall superheat on the delivery of ECC from the downcomer to the lower plenum. The apparatus consisted of a vessel with downcomer, lower plenum, four cold-leg ports, four simulated hot-leg penetrations, and a steam injection port at the top. In the quasi-static experiments, a steady steam flow is established up the downcomer, and water is then injected at a constant flow rate into three of the cold-leg ports; the fourth cold-leg port simulates the broken cold leg. After an initial transient period, the steam and water flows reach a quasi-steady state in which some of the injected water is bypassed and the remainder penetrates into the lower plenum. Data from these experiments are used to generate flooding curves which specify the amount of water delivered to the lower plenum as a function of the reverse steam flow rate. The steam flow rate is varied to encompass the range from complete delivery to complete bypass.

The TRAC-PD2 model for the Creare experiments consists of a three-dimensional vessel containing 112 fluid cells and one-dimensional piping connections for the injection and break ports. The calculational procedure closely parallels the experimental procedure. Results of the Creare calculations are in good overall agreement with experimental data for a wide range of ECC injection rates and subcoolings. This is illustrated in Fig. 7 for a low-

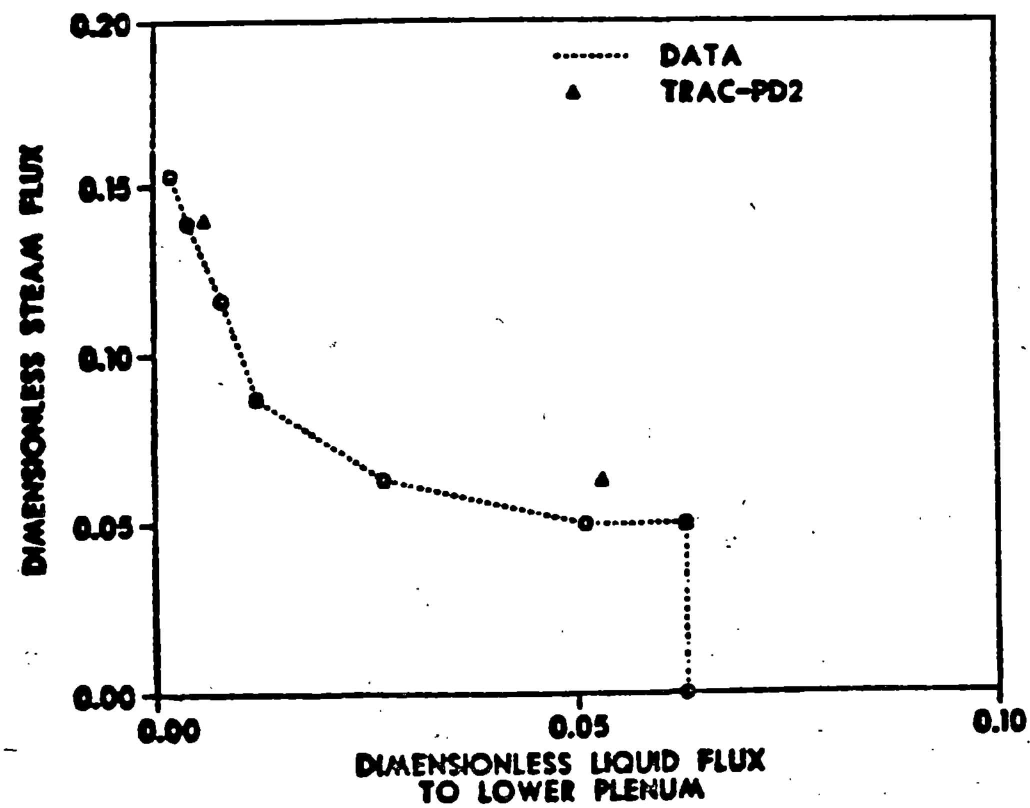

Fig. 7. Flooding curve for Creare low-subcooling 30 gpm tests.

subcooling case. The complete bypass and complete delivery points on the flooding curves are predicted well by TRAC-PD2 for both low- and high-subcooling cases; the intermediate delivery points are not predicted as well. These results are very similar to those obtained with TRAC-P1A.

F. FLECHT Forced-Flooding Experiments

The Full-Length Emergency-Cooling Heat-Transfer (FLECHT) Facility was designed to provide separate effects data for evaluation of reflood heat transfer and quench-front propagation models. FLECHT consists of a full-length fuel bundle containing approximately 100 electrically heated rods mounted in a flow housing with upper and lower plenum regions. Prior to a reflood test, core test section and housing are preheated by applying power to the rods and auxiliary housing heaters. With the lower plenum full of water and the test section containing only saturated steam, ECC injection into the lower plenum is initiated when the desired maximum rod temperature is reached during the preheating period. The power is decreased during the reflood period to model reactor decay power.

Two FLECHT tests and one FLECHT-SEASET test, representing a range of conditions, were chosen for inclusion in the TRAC-PD2 assessment set. These tests emphasize low flooding rates since these were not predicted well by TRAC-P1A. FLECHT Test 4831 used an axial cosine power distribution and a flooding rate of 1.5 in/s, a skewed axial power profile and flooding rate of 6 in/s were used in FLECHT Test 17201, and SEASET Test 4 used highly subcooled ECC at 1.0 in/s. The single-channel geometry of these experiments is represented well by the slab vessel option. In fact, a one-dimensional representation was obtained by using only one cell per axial level. Nine of the 12 vessel levels were used to represent the core region with each of these core levels subdivided into 5 fine-mesh axial intervals for the reflood heat-transfer calculation. Conduction in the heater rod was represented with 8 radial nodes and super-fine axial noding determined dynamically by the fine-mesh rezoning method incorporated in TRAC-PD2.

Comparisons of calculated and measured clad temperature response and carryover rates for these tests indicate that the PD2 reflood and entrainment models are significantly improved over P1A. An example of this is shown in

Fig. 8 which compares the midplane clad temperature for Test 4831.[12] In general the PD2-calculated peak clad temperatures (PCT) at various elevations are about the same as those obtained with P1A but the times to PCT are improved considerably. PD2 generally quenches late whereas P1A quenched too soon with the effect increasing with elevation. Radiation heat transfer from the heated rods to the bundle housing and unpowered rods is not included in the TRAC model but is estimated by the experimenters[12] to account for 25-30% of the heat transfer in some cases. An approximate radiation heat-transfer model was used in a special version of PD2 to estimate the impact of this effect on the quench time. Results, illustrated in Fig. 9 at the 10-ft elevation for Test 17201,[13] indicate that radiative heat transfer can account for most of the late quenching in PD2 (P1A quench times become worse when this effect is included).

In most cases PD2 predicted a quench temperature close to the test data, indicating a good minimum stable film boiling temperature model. Time-averaged effluent mass flow rates calculated with PD2 agree very well with measurements for both SET tests. This is illustrated in Fig. 10 for Test 4831. Overall the PD2 results indicate acceptable reflood and entrainment models for these forced flooding tests. Results may be further improved by use of a separate droplet field which may be needed to obtain the correct axial void distribution above the liquid pool. Calculation of a gravity reflood FLECHT-SET test is in progress to help resolve this question.

G. Bennett Steady-State CHF Experiments

The Bennett experiments[14] consisted of steady-state critical heat flux (CHF) tests in heated vertical tubes for various coolant mass flow rates and inlet subcoolings, wall heat fluxes, and heated tube lengths. Three typical tests (runs number 5336, 5431, and 5442), which cover a range in these parameters, were selected for inclusion in the PD2 developmental assessment set. The inside diameter and heated length of the tube are representative of a full-scale fuel rod. However, the flow inside the tube is not prototypic of a reactor fuel rod where the coolant flow is over the outside surface.

In general, the high flow-rate results with PD2 are in better agreement with the data than was the case with P1A. This is illustrated in Fig. 11

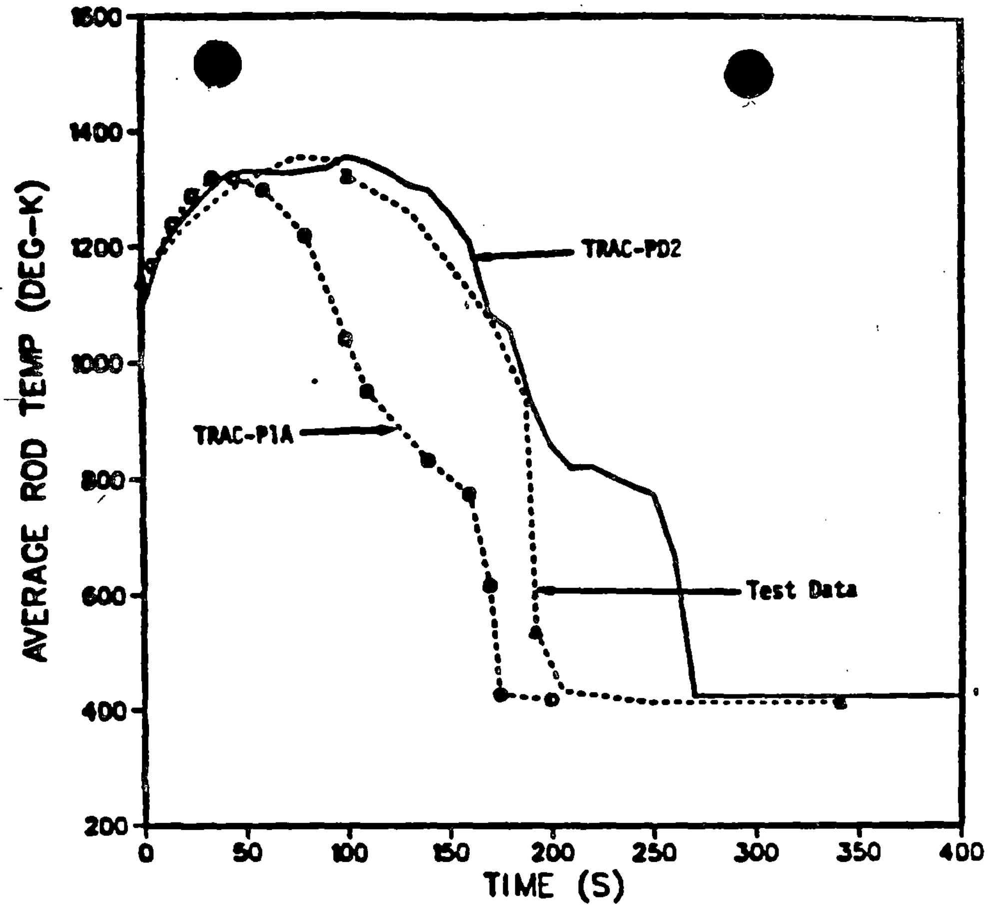

Fig. 8. FLECHT Test 4831 six-foot elevation rod clad temperature.

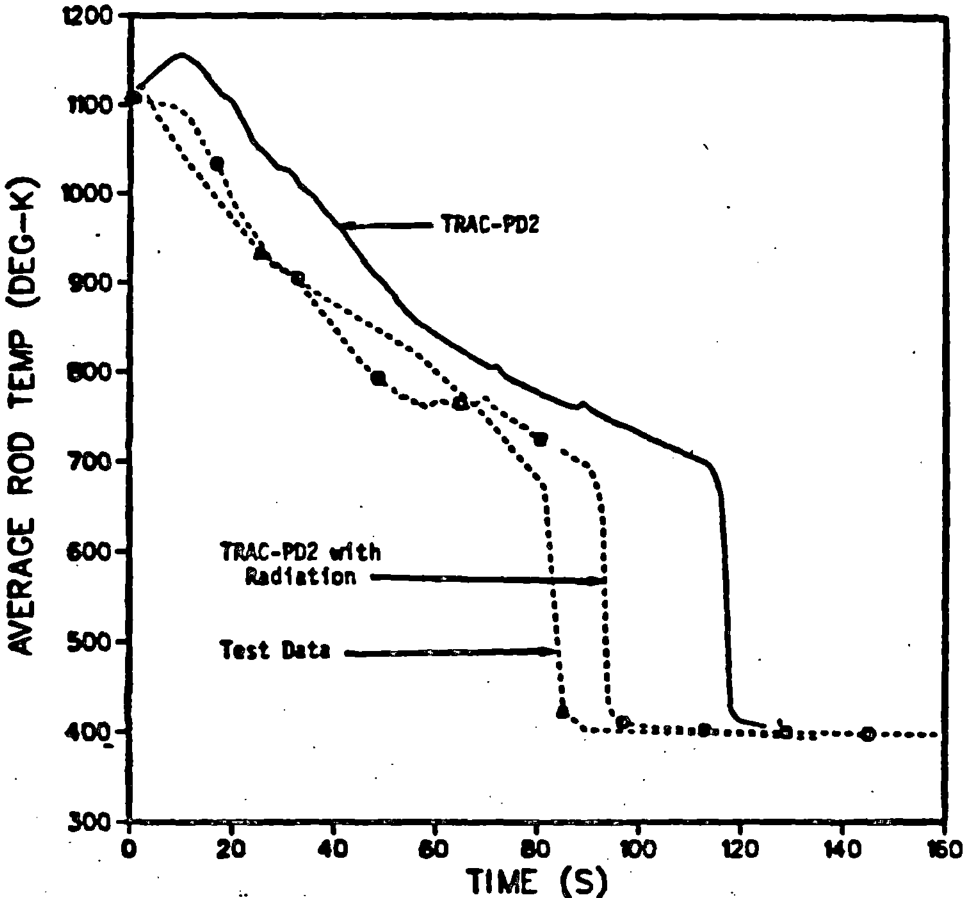

Fig. 9. FLECHT Test 17201 ten-foot elevation rod clad temperatures.

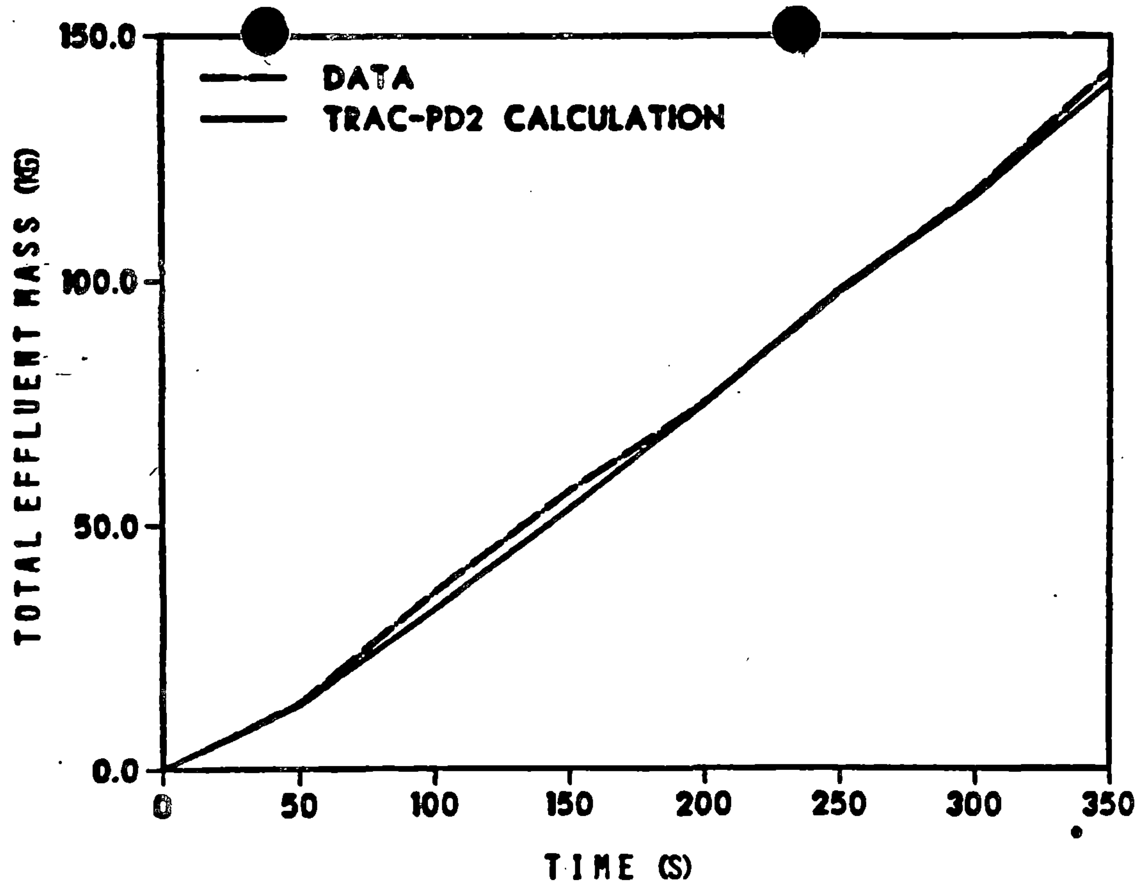

Fig. 10. FLECHT Test 4831 total effluent mass history.

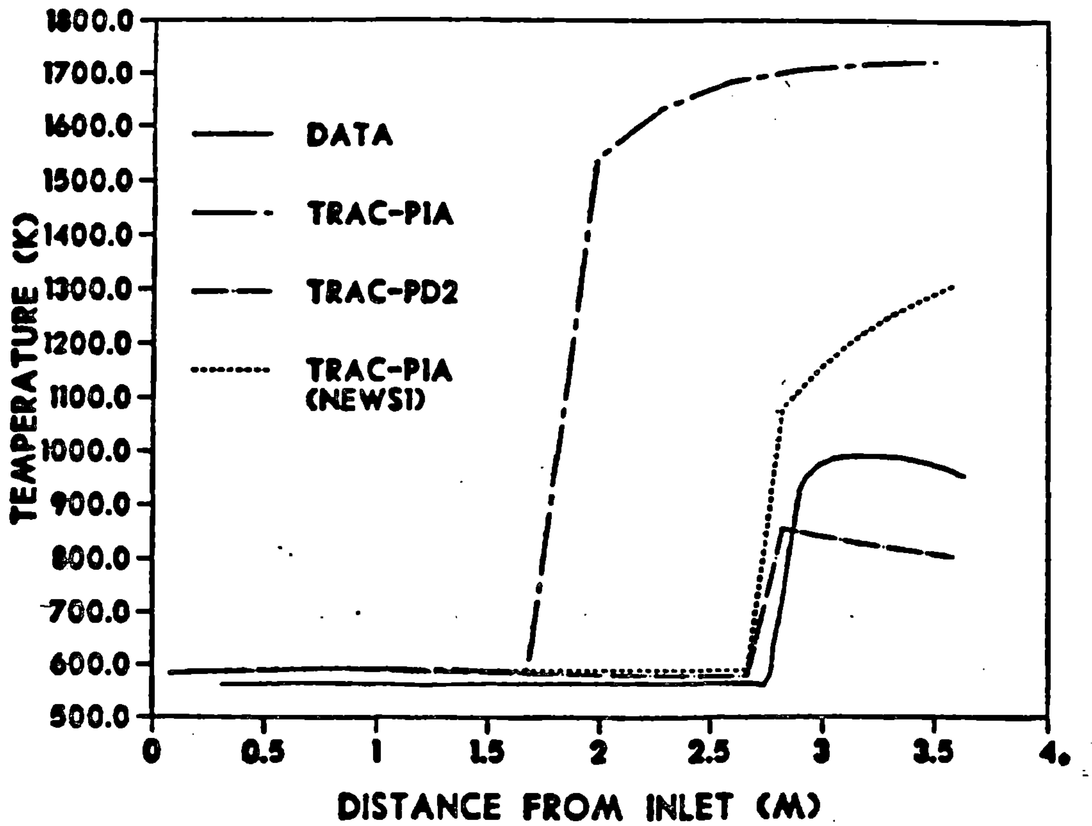

Fig. 11. TRAC comparison with data for Bennett run number 5442.

which shows the results for run 5442. The results labelled TRAC-P1A (NEWS1) were obtained with several error corrections to P1A described in the first TRAC newsletter. One of these errors resulted in dryout too low in the tube with P1A. We have therefore used the corrected P1A version for comparison with PD2. Note in Fig. 11 that the peak temperature and post-CHF behavior are predicted much better with PD2. For the low flow-rate case (5431), the CHF point with PD2 is not as good as was the case with P1A (NEWS1). This is due to a change in the boiling curve interpolation region at the high end of the void fraction range. This change was required to avoid steep heat-transfer coefficient gradients which were causing problems in other calculations. The Bennett low flow-rate results are very sensitive to this change.

H. Semiscale System Blowdown and Integral Tests

The Semiscale Mod-1 system consisted of a pressure vessel with internals; an intact loop with a steam generator, pump, and pressurizer; a blowdown loop with a simulated steam generator, simulated pump, and two rupture assemblies; and a pressure-suppression tank. The vessel contained a 0.011-m downcomer gap and 39 electrically heated rods which could be programmed to simulate the surface heat flux of a nuclear rod. The volume (or power) scale factor between this apparatus and a large PWR is about 1/3000.

1. Test S-02-8

Test S-02-8 consisted of a 200% double-ended cold-leg break without ECC injection but with a programmed power decay curve to simulate decay heat in a nuclear core.[15] The blowdown was initiated from a steady-state temperature distribution in the core and loops at a power level of 1.6 MW. The TRAC model of Test S-02-8 contains a total of 172 fluid cells, including 52 cells in the three-dimensional vessel model.

Calculated steady-state initial conditions and transient results agree well with the measurements of system variables. Typical transient results are illustrated by the lower plenum pressure (Fig. 12), mass flow rate in the intact loop (Fig. 13), and midplane clad temperature (Fig. 14). In general,

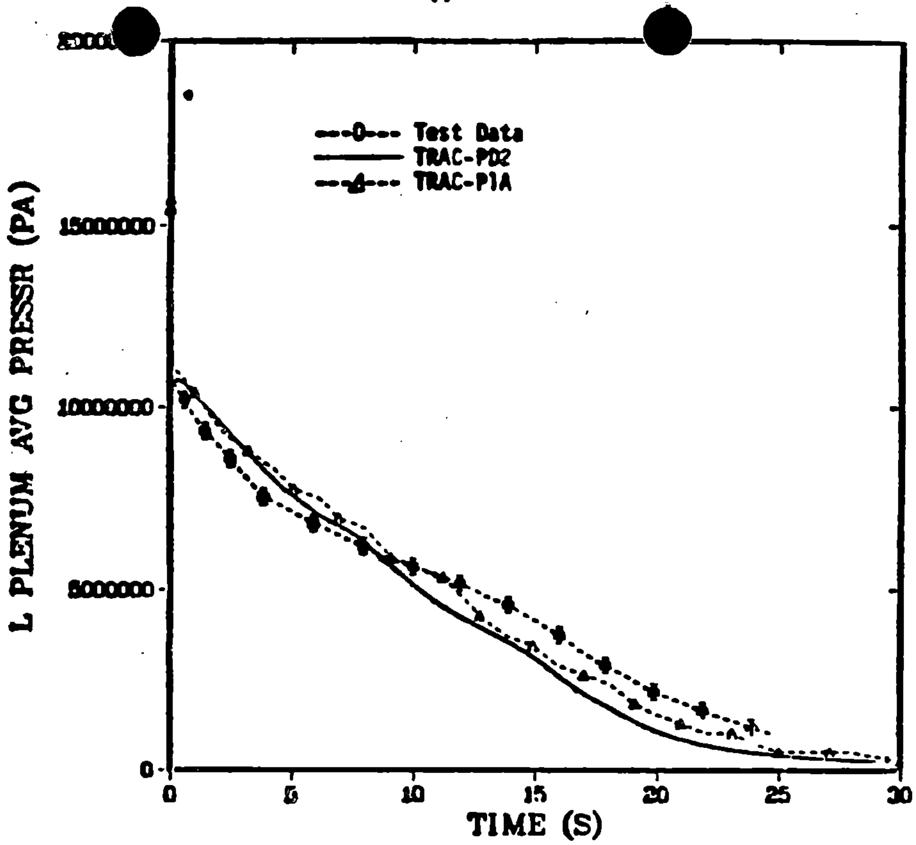

Fig. 12. Lower plenum pressure for Semiscale heated blowdown test S-02-8.

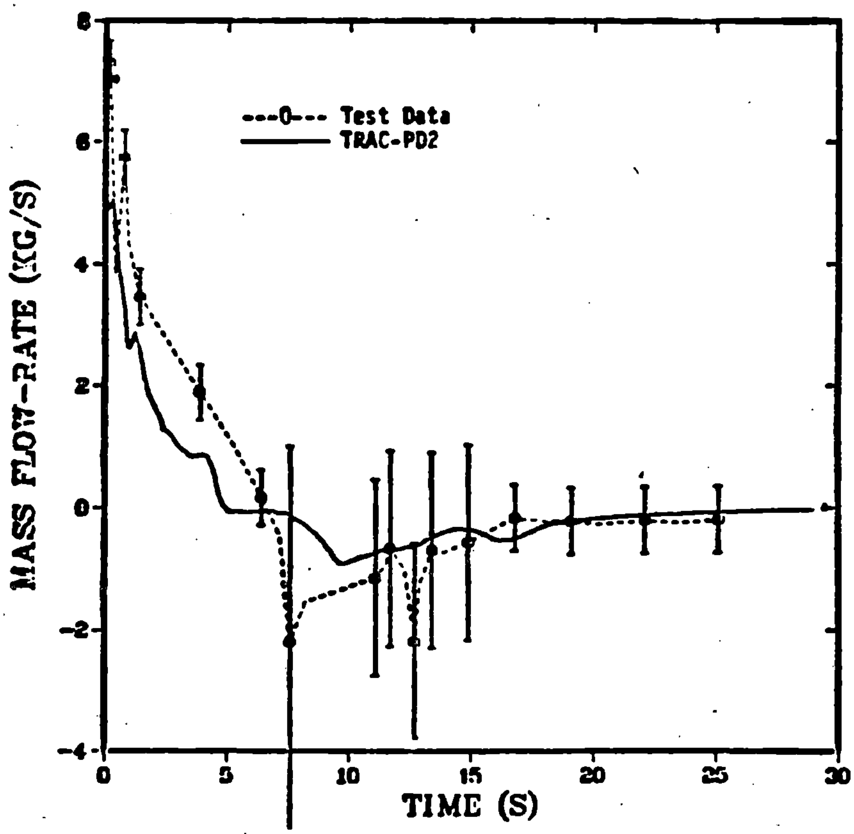

Fig. 13. Intact loop (cold leg) mass flow rate for Semiscale heated blowdown test S-02-8

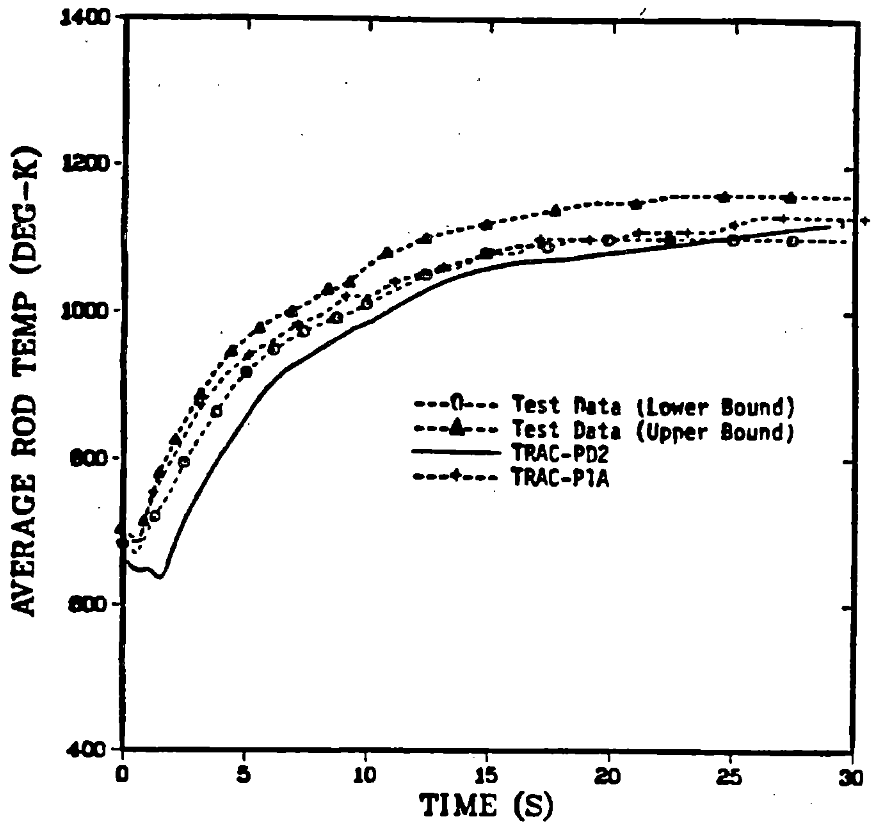

Fig. 14. Cladding temperature response at core midplane (high-power zone) for Semiscale heated blowdown test S-02-8.

both PD2 and P1A predict the system blowdown response very well with PD2 giving slightly better overall agreement with the data.

2. Test S-06-3

Test S-06-3 was a large-break integral LOCA test with accumulator and high and low pressure injection into the intact loop cold leg.[16] The system configuration for this test was otherwise essentially the same as for Test S-02-8. As illustrated in Figs. 15-17, the PD2 results for this test are in overall good agreement with the data (this case was not calculated with P1A). The agreement between the calculated and measured mass flow rates on the vessel side of the break (Fig. 15) is remarkable given the uncertainties in the test data. In the intact loop (Fig. 16), TRAC predicts the rapid decrease in mass flow rate due to two-phase degradation in the pump. As shown in Fig. 17, TRAC tended to somewhat underpredict peak cladding temperatures but the overall comparisons were good except for the high-power rods at the top of the core.

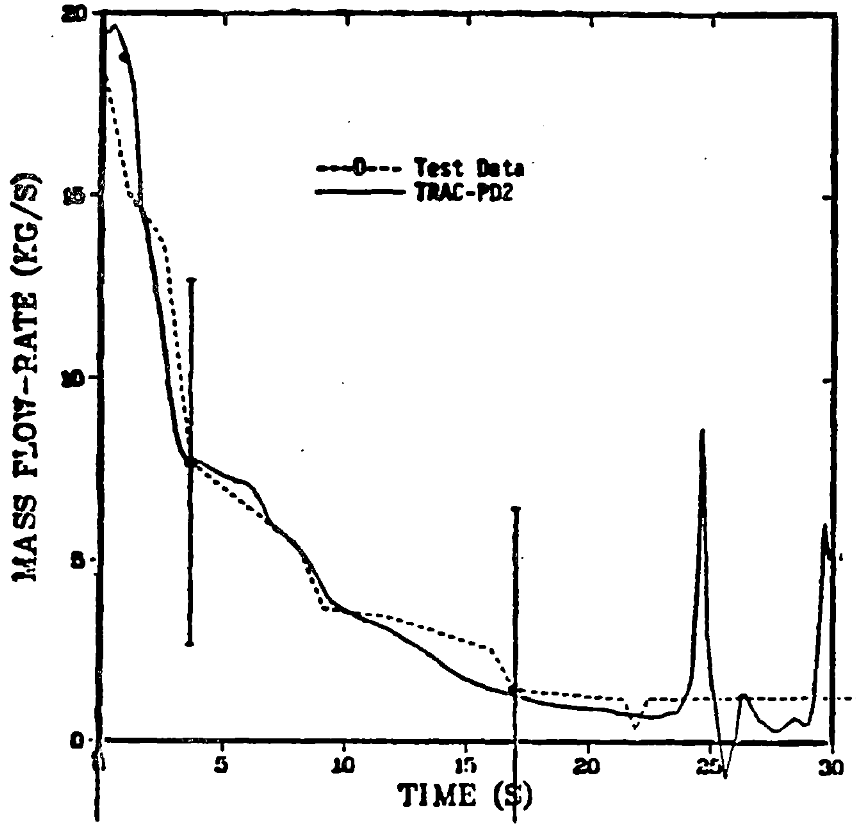

Fig. 15. Break mass flow rate (vessel side) for Semiscale integral test S-06-3.

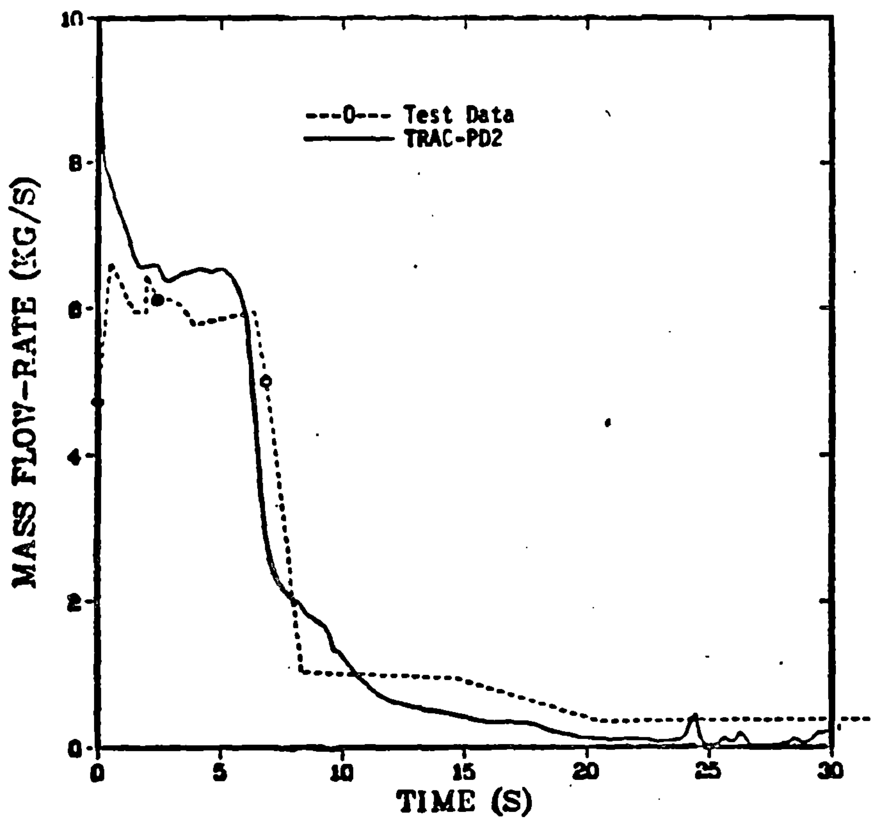

Fig. 16. Intact loop pump mass flow rate for Semiscale integral test S-06-3.

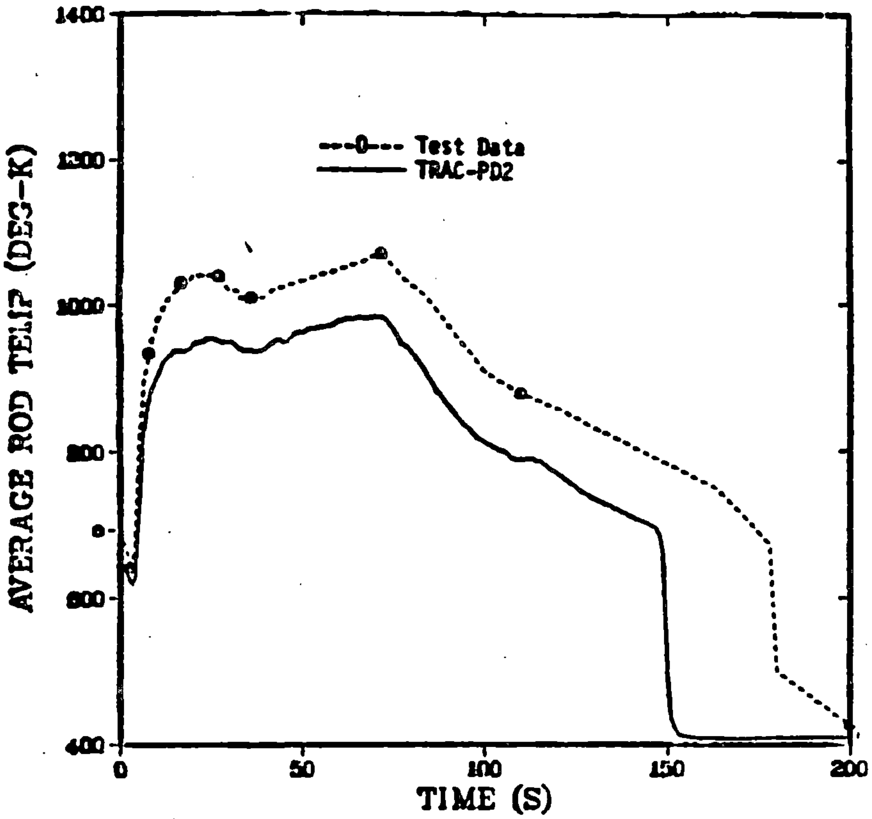

Fig. 17. Hot-rod clad temperature response at core midplane for Semiscale integral test S-06-3.

I. LOFT Integral Tests L1-4 and L2-2

The Loss-of-Fluid Test (LOFT) Facility is a scale model of a large PWR with volume (or power) scaling of 1/60; flow and break areas are scaled using the same ratio. LOFT consists of a pressure vessel with internals; an intact loop with a pressurizer, steam generator, and two pumps; a blowdown loop with a simulated steam generator, simulated pump, and two quick-opening valves; a pressure suppression system; and an emergency core cooling system (ECCS) in the intact loop containing an accumulator, high-pressure injection system (HPIS), and low-pressure injection system (LPIS). The pressure vessel contains upper and lower plena, a downcomer, and a core support barrel. For Test L1-4, which was performed prior to installation of the nuclear core, a hydraulic core simulator represented the flow resistance of the fuel-rod bundles. Test L2-2 was conducted with the nuclear core in place.

1. Test L1-4

Test L1-4[17] was a 200% double-ended cold-leg break starting from initial isothermal temperature, pressure, and flow rate of 552 K, 15.75 MPa, and 268 kg/s, respectively. The purpose of this test was to obtain information on HPIS and LPIS injection and to obtain data for evaluating downcomer bypass and ECC mixing. Since there was no core in this test, the reflood phase of a LOCA was not simulated. The test was modeled with 28 TRAC components containing a total of 205 fluid cells (72 cells in the three-dimensional vessel model).

Calculated initial steady-state conditions are close to the experimental values. Calculated transient results are also in good overall agreement with the measurements including mass flow rates, temperatures, densities, and pressures throughout the system. Some typical examples are shown in Figs. 18-20. The pressure comparison in the intact loop cold leg is shown in Fig. 18. TRAC slightly overpredicts the pressure during the initial part of the transient and underpredicts at the end. This is probably due to the lumped parameter heat-slab model (used to model structural-material heat transfer) releasing too much heat early and not enough towards the end of the transient. Refill of the vessel (Fig. 19) is somewhat delayed in the calculation but the general features agree well with the measurements. The measured liquid mass is too low during the initial part of the transient because the conductivity probes extend only to the top of the downcomer. As illustrated for the break mass flow rate in Fig. 20, the PD2 and P1A (NEWS1) results for this test are very similar. Note that ECC bypass is seen in both the calculation and test data after about 40 s.

The generally good agreement between TRAC-PD2 and the test results indicate that TRAC provides a good representation of integral effects in LOFT during the blowdown and refill phases of a LOCA. A shortcoming that has been identified is the inability of the lumped parameter heat-slab model to represent accurately the time history of energy addition to the fluid from structural materials in the vessel. A distributed-slab model has been developed and will be included in a future modification to PD2.

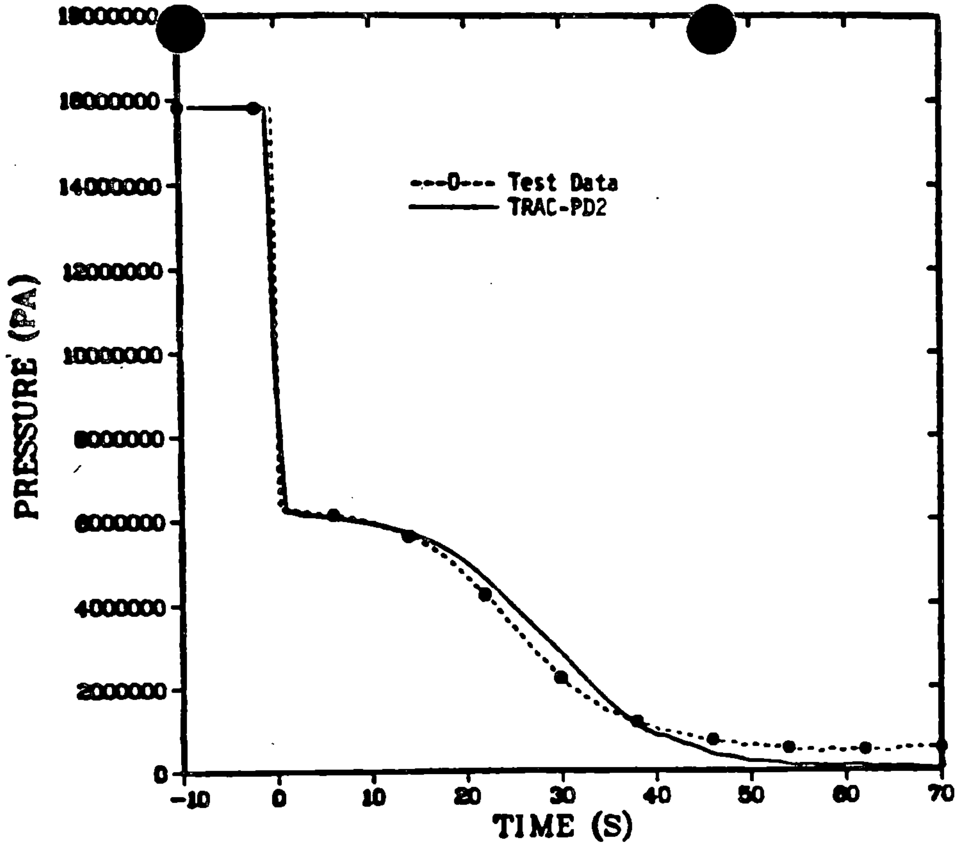

Fig. 18. LOFT Test L1-4 intact loop cold leg pressure.

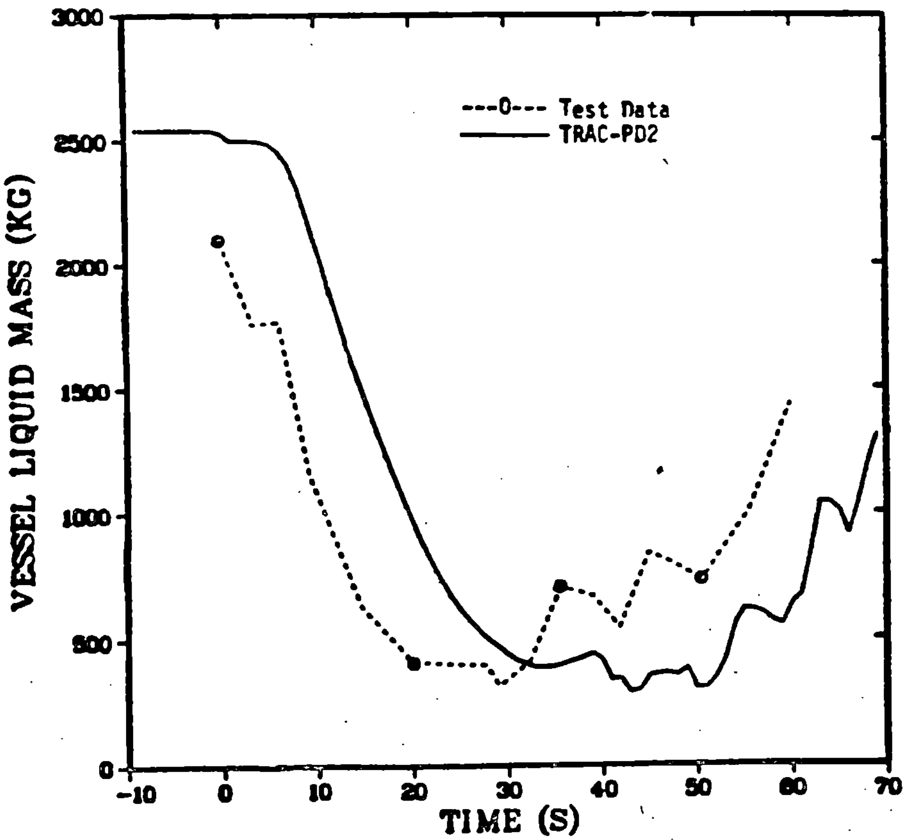

Fig. 19. LOFT Test L1-4 vessel liquid mass.

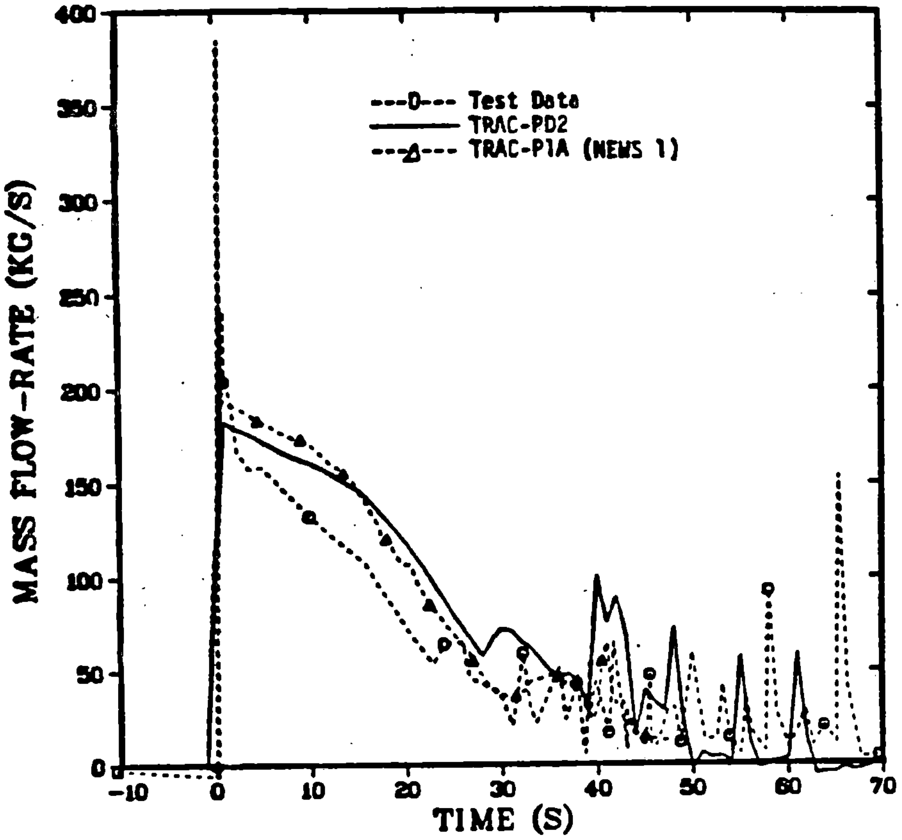

Fig. 20. LOFT Test L1-4 break mass flow rate at vessel side of break.

2. Test L2-2

Test L2-2[18] was the first nuclear-powered test conducted in the LOFT facility. Tests in the L2 series are 200% double-ended cold-leg break LOCAs performed at gradually increasing power levels to determine the nuclear core and system integral response during all phases of a LOCA. Test L2-2 was conducted at 50% power (25 MWt) and an intact hot leg temperature of 580 K (coolant temperature rise from core inlet to core outlet is 23 K). The LOFT nuclear core contains 1300 fuel rods which are full-scale in the radial dimension and approximately half-scale in length. Components other than the core were the same as for Test L1-4 except for some additional instrumentation. Except for the vessel noding, the TRAC-PD2 model for Test L2-2 is similar to that used for Test L1-4.

The calculated hydraulic response generally agrees very well with the data as shown in Table II which lists the major events for the test. The primary

TABLE II

CHRONOLOGY OF TEST L2-2

Event	Time (s) TRAC-PD2	Data
Blowdown Initiated	0	0
End of Subcooled Blowdown	0.1	.07
Time of Maximum Clad Temperature*	7	4
HPIS Initiated	12	12
Pressurizer Empty	15	15
Accumulator Flow Initiated	17	18
LPIS Initiated	29	29
Lower Plenum Filled with Liquid	50	35
Saturated Blowdown Ended	45	44
Core Filled with Liquid	60	55

*Maximum Clad Temperature (K) 795 780

discrepancy is a lower accumulator discharge rate in the calculation which delays refilling of the lower plenum. However, the time to core refill is predicted reasonably well and the peak clad temperature is close to the observed value.

Results for the intact loop are illustrated in Fig. 21 which shows the pressure in the hot leg. The tendency seen in L1-4 to overpredict the pressure early and underpredict late is also present in the L2-2 results. Figure 22 compares the break flow (vessel side of break) and shows good agreement except for the initial period of subcooled critical flow (first 10 s). The underprediction during the first 10s is probably due to lack of a delayed nucleation model in TRAC.

Figure 23 shows typical results for the cladding temperature response at the core midplane for the central fuel bundle (high power zone). The data shown are from three neighboring thermocouples. Other thermocouples in this

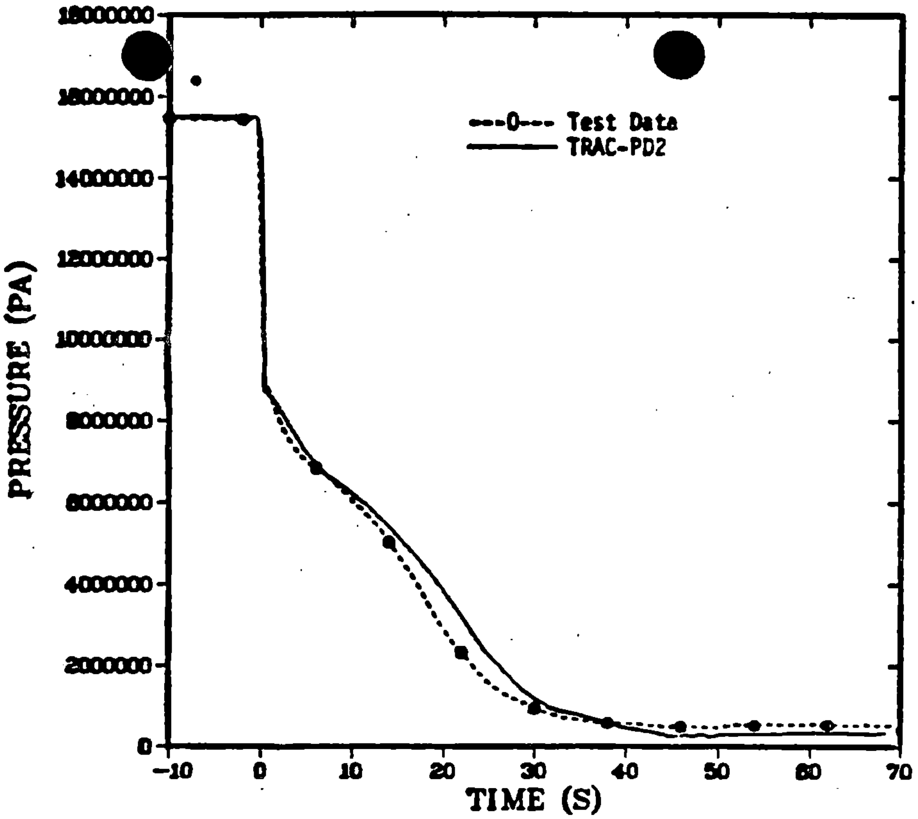

Fig. 21. Intact-loop hot-leg pressure for LOFT integral test L2-2.

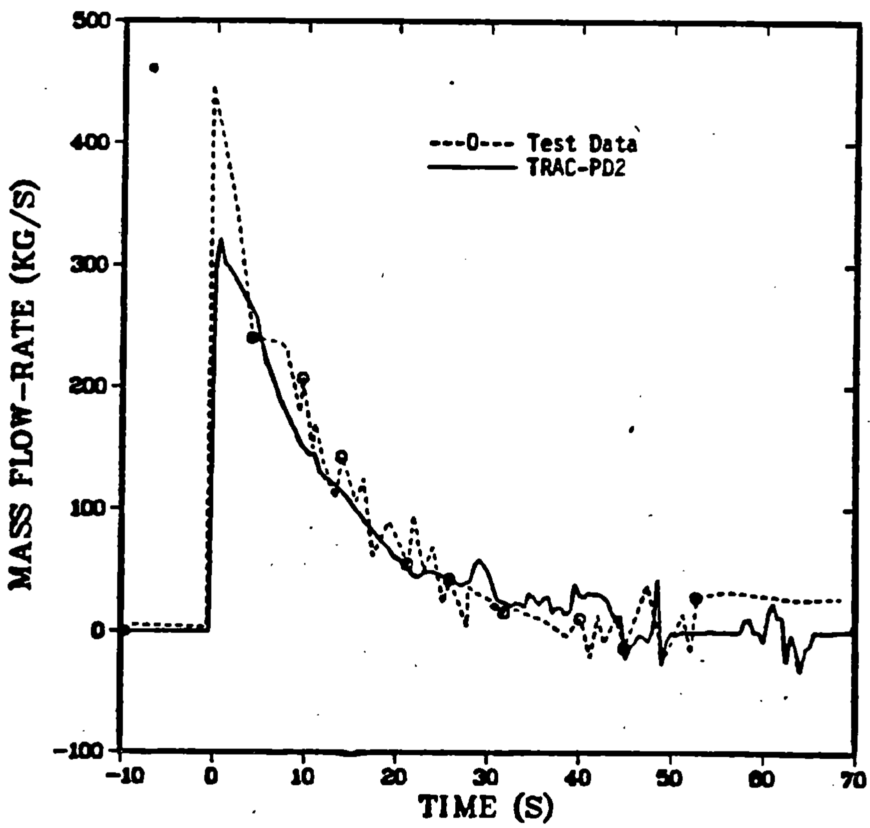

Fig. 22. Break flow (vessel side) for LOFT integral test L2-2.

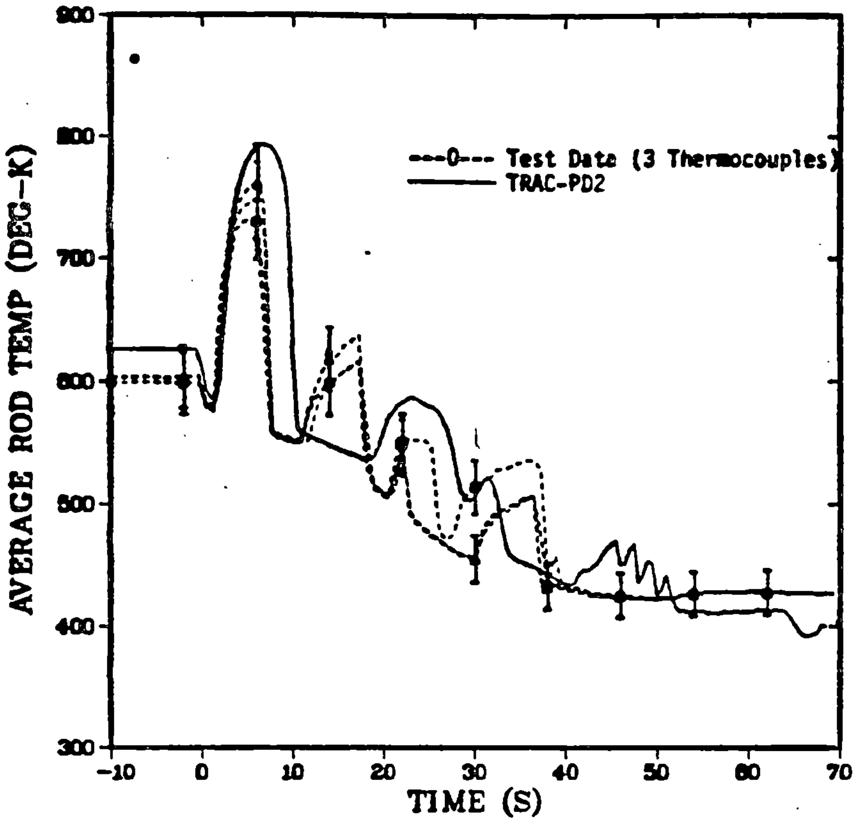

Fig. 23. Midplane clad temperature in high-power zone for LOFT integral test L2-2.

same fuel bundle and at the same elevation show significantly different behavior so that the spread in the measurements is much larger than that shown in the figure. The TRAC-PD2 results shown are typical for all the rods in the central power zone except that the rods adjacent to the broken hot leg do not experience the second dryout (this was also observed in some of the measurements). Both the calculation and data show a series of dryouts and rewets with the peak clad temperature occurring during blowdown. Comparisons at other elevations and in the intermediate- and low-power zones are similar to those shown in Fig. 23. Unlike P1A, PD2 predicts the core rewet that occurs at about 10 s without the use of a special rewet correlation. The peak clad temperature was found to be sensitive to the initial fuel conductivity and gap width.

J. CCTF Reflood Test C1-1 (Run 010)

The Cylindrical Core Test Facility (CCTF) is a large-scale experimental test facility designed to provide data on multidimensional and PWR system effects during the refill and reflood phases of a LOCA. It models a 4-loop PWR with each loop containing an active U-tube steam generator and a pump simulator. Full-scale elevations are used but the radial dimensions in the pressure vessel are 1/5-scale. The core region consists of 2000 electrically heated rods arranged in 8x8 square arrays for a total of 32 assemblies.

Test C1-1[19] was a cold-leg ECC injection reflood test simulating a 200% double-ended cold-leg break in a full-scale PWR. This test also can be characterized as a gravity reflood test because the flooding rate is determined by the downcomer head and the core back-pressure. The TRAC model of CCTF includes 18 components with the three intact loops combined into a single loop and the broken loop modeled separately. Relatively coarse noding (~ 100 total cells with 44 in the vessel) was found to be adequate for this test. The calculational procedure parallels the test operating procedure.

A comparison of the calculated and measured sequence of events is shown in Table III. A comparison of the calculated and measured clad temperature response of the core midplane is shown in Fig. 24. Downcomer liquid level oscillations (period ~ 100 s) are calculated after the maximum level is reached at 150 s. The data also show oscillations but with smaller amplitude and longer period (~ 200 s). As is the case for the downcomer, TRAC calculates the correct average response of the core liquid level but exhibits more oscillatory behavior than the data.

Overall the data comparisons for this test show that TRAC-PD2 can calculate system effects during gravity reflood and that significant improvements have been made over TRAC-P1A with regard to quench behavior, carryover, pressure oscillations, mass conservation, and running time. There still remain residual problems with regard to carryover and precooling at the higher core elevations. Some of these problems may be associated with spacer-grid effects (e.g. droplet breakup) which are not modeled with the present TRAC model.

TABLE III

CCTF TEST C1-1

CHRONOLOGY OF EVENTS

Event	Time (s) Data	TRAC
Core Power On	0.0	0.0
Accumulator Injection Initiated (Hot-Rod Temperature Trip)	54.0	57.7
Beginning of Core Recovery	66.0	69.7
LPIS Injection Initiated	75.0	78.7
Max. Water Level Reached in Downcomer	150	150
Peak Clad Temperature Reached (TRAC = 1000 K, data = 1050 K)	150	190
Avg. Rod Quenched Through Midplane	375	450

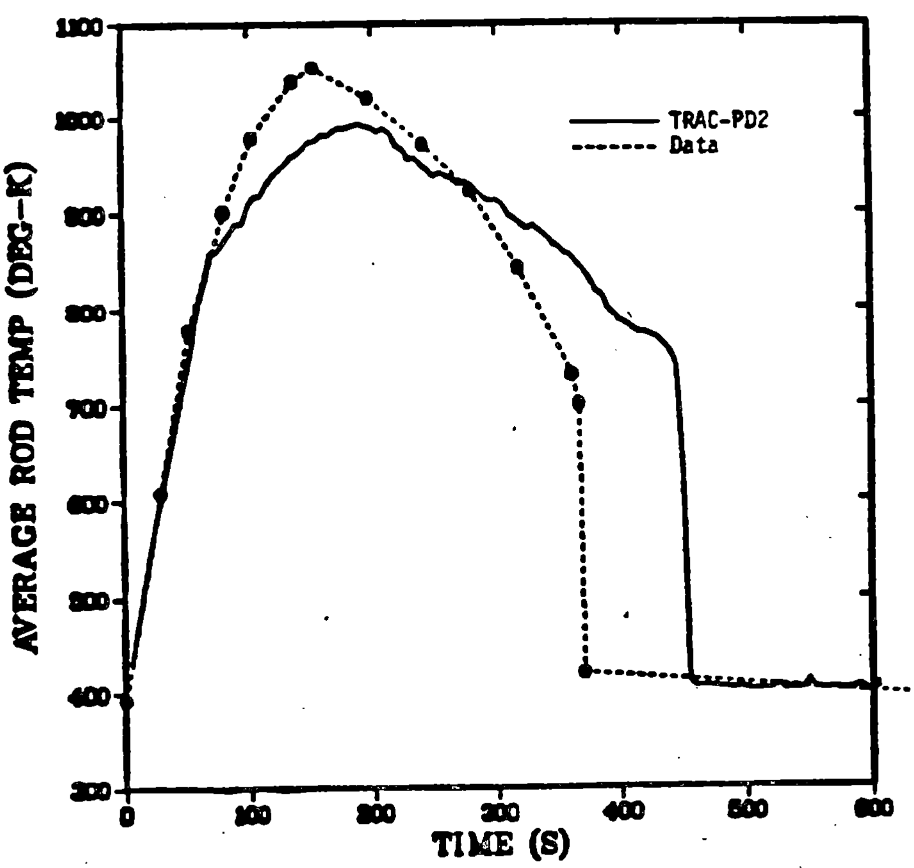

Fig. 24. CCTF Test C1-1 cladding temperature response at core midplane.

III. CONCLUSIONS

Results obtained for the developmental assessment tests indicate that TRAC-PD2 does a credible job for tests in many different experimental facilities involving separate, system, and integral effects over a wide range of scales. Improvements over TRAC-P1A are mostly in the reflood heat-transfer area where a more sophisticated and mechanistic model was implemented. As a result of this and numerous other improvements in solution strategy, numerics, and constitutive relations, PD2 is much more reliable and smoother-running than P1A. For example, PD2 can run long transients (e.g. small-break LOCAs) with very good mass conservation. Running time is the same or improved over P1A even though the reflood heat-transfer treatment is more complex.

Several areas requiring improvement have been identified during assessment and application of PD2. Some of these improvements (e.g. distributed slab model) are already available as modifications to PD2 and other improvements (e.g. critical flow model for small breaks) have been or will be implemented in the version (PF1) currently under development.

ACKNOWLEDGMENTS

The TRAC-PD2 developmental assessment calculations and analyses summarized in this report were performed by the following Los Alamos National Laboratory personnel: J. S. Gilbert, D. A. Mandell, J. K. Meier, T. Bott, J. M. Sicilian, and J. R. Ireland.

REFERENCES

1. "TRAC-P1: An Advanced Best-Estimate Computer Program for PWR LOCA Analysis. Volume I: Methods, Models, User Information, and Programming Details," Los Alamos Scientific Laboratory report LA-7279-MS, Vol. I (NUREG/CR-0063) (June 1978).

2. "TRAC-P1A: An Advanced Best-Estimate Computer Program for PWR LOCA Analysis," Los Alamos Scientific Laboratory report LA-7777-MS (NUREG/CR-0665) (May 1979).

3. TRAC-PD2: An Advanced Best-Estimate Computer Program for Pressurized Water Reactor Loss-of-Coolant Accident Analysis," Los Alamos National Laboratory report (in preparation).

4. J. C. Vigil and K. A. Williams (Compilers), "TRAC-P1A Developmental Assessment," Los Alamos Scientific Laboratory report LA-8056-MS (NUREG/CR-1059) (October 1979).

5. "TRAC-PD2 Developmental Assessment," Los Alamos National Laboratory report (in preparation).

6. A. R. Edwards and T. P. O'Brien, "Studies of Phenomena Connected with the Depressurization of Water Reactors," J. British Nuycl. E. Soc., 9, 125 (April 1970).

7. A. Premoli and W. T. Hancox, "An Experimental Investigation of Subcooled Blowdown with Heat Addition," Submission to Committee on Safety of Nuclear Installations, Specialists Meeting on Transient Two-Phase Flow, Toronto, Ontario (August 1976).

8. L. Ericson, L. Gros d'Aillon, D. Hall, J. Ravnsborg, O. Sandervag, and H. Akesson, "The Marviken Full-Scale Critical Flow Tests Interim Report; Results from Test 4," Marviken draft interim report MXC-204 (May 1978).

9. L. Ericson, L. Gros d'Aillon, K. Kilpi, O. Sandervag, R. Schultz, and J. Vidarsson, "The Marviken Full-Scale Critical Flow Tests Interim Report; Results from Test 24," Marviken draft interim report MXC-224 (May 1979).

10. C. B. Mullins, C. R. Hyman, W. G. Craddick, T. M. Anklam, and R. A. Hedrick, "PWR Blowdown Heat Transfer Separate-Effects Program Data Evaluation Report--THTF Test Series II," Oak Ridge National Laboratory report ORNL/NUREG-53 (NUREG/CR-0539) (April 1979).

11. C. J. Crowley, J. A. Block, and C. N. Cary, "Downcomer Effects in a 1/15-Scale PWR Geometry: Experimental Data Report," Creare, Inc. report NUREG-0281 (May 1977).

12. E. R. Rosal, et al., "FLECHT Low Flooding Rate Cosine Test Series Data Report," Westinghouse Electric Co. report WCAP-8651 (1975).

13. E. R. Rosal, et al., "FLECHT Low Flooding Rate Skewed Test Series Data Report," Westinghouse Electric Co. report WCAP-9108 (May 1977).

14. A. W. Bennett, G. F. Hewitt, H. A. Kearsey, and R. K. F. Keeys, "Heat Transfer to Steam-Water Mixtures Flowing in Uniformly Heated Tubes in which the Critical Heat Flux Has Been Exceeded," Atomic Energy Research Establishment report AERE-R5373 (1967).

15. H. S. Crapo and K. E. Sackett, "Experimental Data Report for Semiscale Mod-1 Test S-02-8 (Blowdown Heat Transfer Test)," Aerojet Nuclear Company report ANCR-NUREG-1238 (August 1976).

16. B. L. Collins, M. L. Patton, Jr., K. E. Sackett, and K. Stanger, "Experiment Data Report for Semiscale Mod-1 Test S-06-3 (LOFT Counterpart Test)," EG&G Idaho, Inc. report NUREG/CR-0251 (July 1978).

17. D. L. Batt, "Experimental Data Report for LOFT Nonnuclear Test L1-4," Idaho National Engineering Laboratory report TREE-NUREG-1084 (July 1977).

18. "Experiment Data Report for LOFT Power Ascension Test L2-2," Idaho National Engineering Laboratory report NUREG/CR-0492 (1979).

19. "Quick-Look Report on Large Scale Reflood Test-1, CCTF Test C1-1 (Run 010)," Japan Atomic Energy Research Institute report 8453 (August 1979).

ENCLOSURE 3

Los Alamos

Los Alamos National Laboratory
Los Alamos,New Mexico 87545

Q-7-81-121
MS-556

July 20, 1981

Dr. Louis M. Shotkin
Division of Accident Evaluation
Mail Stop 1130SS
US Nuclear Regulatory Commission
Washington, DC 20555

Dear Lou:

Table I provides a list of full-scale reactor analyses performed at Los Alamos using TRAC-PD2 that you requested. Los Alamos technical note report reference numbers and other correspondence describing the results of the calculations are listed in the last column of Table I. Table II lists the titles of these technical notes detailing results of some of these calculations. The calculations that are in progress are scheduled to be completed this fiscal year (FY 81).

At the end of this fiscal year, I will update these Tables and again forward them to you. Please let me know if you have any further requests or comments regarding this letter or the enclosures.

Sincerely,

John R. Ireland

John R. Ireland
Project Leader
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Table I

TRAC-PD2 Full-Scale LWR Analyses at Los Alamos

Plant	NSSS Vendor	Plant Type	Transient(s) Analyzed*	Date Completed	Documentation Reference Nos. **
Zion-1	W	4-Loop 15 x 15 fuel assemblies	LOFW-Nominal plus additional failures: eg. ECC unavailability, PORV stuck open, open ARVs, etc.	February 1981	LA-SASA-TN-81-1 LA-SASA-TN-81-2 LA-SASA-TN-81-4
Zion-1	W	4-Loop 15 x 15 fuel assemblies	MSLB, SGTR with primary system HLB or CLB	April 1981	LA-TCA-TN-81-1 LA-SASA-TN-81-3
Zion-1	W	4-Loop 15 x 15 fuel assemblies	200% double-ended cold-leg break	March 1981	LA-2D/3D-TN-81-10
Zion-1	W	4-Loop 15 x 15 fuel assemblies	SBLOCA-Cold leg	April 1981	LA-SBTA-TN-81-1
OCONEE-1***	B&W	Lowered Loop	MSLB, SBLOCA	June 1981	Q-7-81-R109, Letter J. Ireland to L. Shotkin dated May 26, 1981 - preliminary results. (Technical note documentation in progress)
Davis-Besse	B&W	Raised Loop	SBLOCA-Cold leg	In progress	---

Table I - Continued

TMI-1***	B&W	Lowered Loop	MSLB+SGTR-1 tube, 5 tube, 10 tube	In progress	---
TMI-1	B&W	Lowered Loop	MSLB-overfill transients-SGTR, PORV failure, pump seal failures	In progress	---
Arkansas*** Nuclear One-1	B&W	Lowered Loop	LOFW-various scenarios	In progress	---
Zion-1	W	4-Loop 15 x 15 fuel assemblies	SBLOCA-4 in. CLB pumps on/ pumps off issue	In progress	---
TMI-2	B&W	Lowered Loop	SBLOCA-4 in. CLB pumps on/ pumps off issue	In progress	---
Midland-1***	B&W	Lowered Loop	Overcooling accidents, LOFW, SBLOCA	In progress	---
Crystal River***	B&W	Lowered Loop	Crystal River accident	In progress	---

*LOFW Loss Of Feedwater
ARV Secondary Side Atmospheric Relief Valve
PORV Power Operated Relief Valve
MSLB Mail Steam Line Break
SGTR Steam Generator Tube Rupture
HLB Hot Leg Break
CLB Cold Leg Break
SBLOCA Small Break Loss-Of-Coolant Accident

**Los Alamos documents describing calculations are in Program Technical Note format - Table II lists titles of published Technical Notes.

***All B&W TRAC input decks are based on the TMI-2 plant with appropriate changes eg. power level, flows, etc.

Table II

Los Alamos Technical Notes

1. D. E. Lamkin, C. E. Watson, and D. Dobranich, "TRAC Analysis Of Coincident Main Steam-Line Break, Steam Generator Tube Rupture, And a Small Primary-Coolant Piping Break", Los Alamos National Laboratory TRAC Calculational Assistance Program Technical Note, LA-TCA-TN-81-1, April 1981.

2. D. Dobranich, "Loss-Of-Feedwater Transients For The Zion-1 PWR", Los Alamos National Laboratory SASA Program Technical Note, LA-SASA-TN-81-1, February 1981.

3. R. J. Henninger, "TRAC Calculations For Zion-1 Loss-Of-Feedwater With One PORV Stuck Open", Los Alamos National Laboratory SASA Program Technical Note, LA-SASA-TN-81-2, February 1981.

4. D. Dobranich, "Steam Generator Tube Rupture With ECCS Unavailable", Los Alamos National Laboratory SASA Program Technical Note, LA-SASA-TN-81-3, February 1981.

5. R. J. Henninger, "TRAC Calculation For Zion-1 Transient Without Scram With One PORV Stuck Open", Los Alamos National Laboratory SASA Program Technical Note, LA-SASA-TN-81-4, March 1981.

6. J. F. Lime, G. J. E. Willcutt, Jr., "TRAC-PD2 Calculation Of A Cold-Leg Small Break In A Westinghouse Four-Loop Pressurized Water Reactor", Los Alamos National Laboratory Small Break and Transient Audit Program Technical Note, LA-SBTA-TN-81-1, April 1981.

7. J. R. Ireland, D. Liles, "A TRAC-PD2 Analysis of a Large-Break Loss-of-Coolant Accident in a Reference US PWR", Los Alamos National Laboratory 2D/3D Program Technical Note, LA-2D/3D-TN-81-10, March 1981.

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OFFICE	RES:DAE:AMB	RES:DAE:AMB	RES:DAE:DD	RES:DAE:D	RES:DD	RES:D	
SURNAME	Shotkin:md LS	Fabic	Kelber	Bassett	Ross	Minogue	
DATE	7/16/81 7/29/81	9/9/81	9/10/81	9/10/81	9/1/81	9/11/81	