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COMPARISON OF TRAC CALCULATIONS WITH EXPERIMENTAL DATA*

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ABSTRACT

TRAC is an advanced best-estimate computer code for analyzing postulated accidents in light water reactors. This paper gives a brief description of the code followed by comparisons of TRAC calculations with data from a variety of separate-effects, system-effects, and integral experiments. Based on these comparisons, the capabilities and limitations of the early versions of TRAC are evaluated.

1. INTRODUCTION

The Transient Reactor Analysis Code (TRAC) is being developed at the Los Alamos Scientific Laboratory (LASL) under the sponsorship of the Division of Reactor Safety Research, United States Nuclear Regulatory Commission. TRAC is an advanced best-estimate computer code for the analysis of accidents in light water reactors (LWRs).

The initial versions of TRAC are directed primarily toward large break loss-of-coolant-accidents (LOCAs) in pressurized water reactors (PWRs). The first documented version, TRAC-P1, was completed in March 1978. An improved version, TRAC-P1A, was released through the National Energy Software Center (Argonne National Laboratory) in March 1979.

The current development effort is proceeding along two paths. The first path is to continue the development of versions that employ very detailed thermal-hydraulic modeling. The next version to be released in this series is an improved (and faster running) version of TRAC-P1A. This new version will be designated as TRAC-PD2 and is scheduled for release in the spring of 1980. Future versions of TRAC will be designated as TRAC-xyz where x=P for PWR versions and B for boiling water reactor (BWR) versions; y=D for detailed versions and F for fast running versions; and z is a version designation number.

Even though TRAC-PD2 will be much faster running than TRAC-P1A, the sophisticated multidimensional hydrodynamics treatment will still require substantial running times for large problems. For some applications, such as parametric studies, much shorter running times are desirable. Fast running versions of TRAC are under development to satisfy this need. Although these

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versions will provide less detail (and involve more empiricism), they can be calibrated against the carefully assessed detailed versions for specific applications. The initial fast running PWR version, TRAC-PF1, is currently under development at LASL.

The development of a version of TRAC designed specifically to analyze accidents in BWRs was recently initiated at the Idaho National Engineering Laboratory (INEL) with some technical support from General Electric Company. The initial BWR version, TRAC-BD0, was completed in December 1979. Development of the initial release version, TRAC-BD1, is underway.

A key element in the TRAC effort is to establish the predictive capability of the released versions of the code. This is being done by a very careful assessment against a broad range of experimental data. This involves both blind pretest and posttest predictions and detailed posttest analyses. The main purpose of this paper is to summarize some of the recent experimental comparisons and give a brief evaluation of the current capabilities and limitations of TRAC.

The results presented here were nearly all obtained with TRAC-P1A. In some cases, similar calculations have been performed with prerelease versions of TRAC-PD2. Some of these results will be mentioned where they differ significantly from TRAC-P1A. This is done to give a more accurate picture of current TRAC capability.

2. TRAC CHARACTERISTICS

Detailed descriptions of TRAC can be found eleswhere.[1,2,3] A summary description is given in this section to provide the appropriate background for this paper.

TRAC can be characterized as an advanced, best-estimate LWR system analysis computer program. It incorporates state-of-the-art methods and models. The models are designed to yield realistic solutions as opposed to the conservative evaluation models used in licensing codes.[4] User-selected options and "tuning dials" are minimized in the basic fluid dynamics and heat transfer modeling. This approach places great demands on the basic thermal-hydraulic modeling because the code must determine local flow topology and supply appropriate constitutive relations. Thus, the development of accurate flow-regime-dependent constitutive relations is vital to the success of TRAC. An ultimate goal of the TRAC effort is to produce codes that have a demonstrated capability to adequately predict the results of a very broad range of experiments with no tuning of basic physical models from one test to another.

TRAC can be used to obtain steady-state solutions to provide self-consistent initial conditions for subsequent transient calculations. Both a steady-state and transient calculation can be performed in the same run if desired. An important characteristic of TRAC is the ability to address the entire LOCA (blowdown, bypass, refill, and reflood) in one continuous and consistent calculation. Trips can be specified to simulate protective system actions or operational procedures (e.g., the opening or closing of a valve).

TRAC is completely modular by component and by function. Component modules, which consist of subroutines or sets of subroutines, are available to model vessels (with associated internals), steam generators, pressurizers, etc. The user can construct a wide variety of configurations by joining together an arbitrary number of these components in a meaningful way. Thus, the user can solve problems ranging from a simple pipe blowdown to a multiloop PWR LOCA. Component modularity allows component models to be improved, modified, or added without disturbing the rest of the code.

Functional modules, which also consist of subroutines or sets of subroutines, are available for multidimensional two-fluid hydrodynamics, one-dimensional drift-flux hydrodynamics, thermodynamic and transport properties, wall heat transfer, etc. Functional modularity allows the code to be easily upgraded as improved correlations and experimental information become available.

A three-dimensional cylindrical (r-θ-z) or two-dimensional Cartesian (x-y) hydrodynamic calculation can be performed within the reactor vessel. Components outside the vessel are treated in one-dimensional geometry. The vessel module is used to model all regions inside the pressure vessel including the downcomer, lower plenum, core, upper plenum, and upper head. It is in these regions of the reactor system that multidimensional effects are most likely to be significant.

Two-phase flow in the various TRAC components is treated using nonhomogeneous, nonequilibrium models. A two-fluid six-equation model is used within the reactor vessel. These equations are based on the conservation of mass, momentum, and energy for the separate liquid and vapor fields. Supplementing these field equations are so-called constitutive relations (or closure equations) that specify (1) the transfer of mass, energy, and momentum between the liquid and vapor phases and (2) the interaction of these phases with the system structure. The nature of these interfacial transfers and interactions is dependent on flow topology and, therefore, a flow-regime-dependent constitutive equation package is included in TRAC. This package is continually being tested and upgraded as new information becomes available and as our understanding of two-phase flow improves.

The flow in the one-dimensional loop components is described by a five-equation drift-flux model. These equations are based on conservation of mass, energy, and momentum for the mixture and conservation of mass and energy for the vapor. Liquid and vapor velocities are not assumed to be equal, but are expressed in terms of a relative velocity that is dependent on flow topology.

Heat transfer models in TRAC include (1) conduction models to calculate temperature fields in structural materials and fuel rods and (2) convection models to provide heat transfer between structure and coolant. Heat transfer to the two-phase fluid is calculated using a generalized boiling curve constructed from a library of heat transfer correlations based on local surface and fluid conditions.

Conduction models are available for obtaining temperature fields in one-dimensional (cylindrical) pipe walls, lumped-parameter slabs, and one-dimensional (cylindrical) fuel rod geometries. Pipe wall conduction is used in the components outside the vessel, whereas, the slab and fuel rod conduction models are used in the vessel module. The fuel rod conduction analysis accounts for gap conductivity changes, metal-water reactions, and quenching phenomena. A fine-mesh axial renoding capability is available for fuel rods to allow more detailed modeling of reflood heat transfer and tracking of quench fronts due to bottom flooding and falling films. Quench fronts are advanced using an empirical velocity correlation.

The system of field and constitutive equations in TRAC is solved using efficient spatial finite-difference techniques. A semi-implicit time differencing technique is normally used in most components. This technique is subject to the Courant stability limitation that restricts the time step size in regions of high-speed flow. A fully implicit time differencing option is available for the fluid dynamics in most of the one-dimensional components. This option allows fine spatial resolution in regions of high velocity (e.g., in a nozzle) without restricting the time step size.

TRAC is designed to run on a CDC 7600 computer, but standard programming techniques are being used to ease its conversion to other computers. All storage arrays are dynamically allocated so that the only limit on problem size is the available core memory. A capacity of 60K words of small core memory and 220K words of large core memory is sufficient to handle most problems of interest.

3. TRAC ASSESSMENT

Comparisons of TRAC calculations with experimental data have been made for tests in a wide variety of experimental facilities as part of a broad code assessment and testing program. Two phases are involved in this program. Developmental assessment is the first phase and is performed prior to public release of a particular code version. It involves primarily posttest analyses and is closely coupled to code development since results can have an immediate impact on the code models. The primary objective of developmental assessment is to test the code models against available experimental results.

A code version is released for p lic use when developmental testing indicates that its performance objectives have been met. Independent assessment, the second assessment phase, begins at this point and mainly involves pretest and posttest predictions of tests in designated facilities using the publicly released (and frozen) versions of TRAC. The primary objective of this phase is to determine the predictive capability of TRAC when applied to new tests involving different scales and experimental configurations. In addition, results from this assessment phase provide guidance for future code versions.

The terms pretest prediction, posttest prediction, and posttest analyses have the following meaning as used in this paper. Pretest predictions refer to calculations performed prior to performance of the test. They are also called double-blind predictions because neither the actual initial and boundary conditions nor the transient test results are known. Anticipated initial boundary conditions are used in a pretest prediction. Posttest predictions are also called single-blind calculations because the actual initial and boundary conditions are known, but the transient test results are not available. Posttest analyses refer to calculations performed after all test results are available.

The comparisons presented in this paper represent only a sampling of the developmental and independent assessment calculations performed thus far. The results are separated into separate-, systems-, and integral-effects tests. TRAC has also been applied to transients in full-scale PWRs and to planned large-scale reflood facilities in Germany and Japan. These applications activities, including calculations of the Three-Mile-Island accident, are providing information on the ability of TRAC to properly handle the effects of scale.

4. SEPARATE-EFFECTS TESTS

Separate-effects tests basically involve only one LWR component (or part of a component in the case of the vessel) and one phase of a LOCA. Examples of TRAC comparisons with these types of tests are given in this section. Included in these comparisons are posttest analyses of a blowdown of a large-scale vessel, refill of a 1/15-scale vessel with counter-current steam-water flow in the downcomer, and forced reflooding of a full-length electrically heated rod bundle. More detailed results of these posttest analyses can be found in Ref. 3.

4.1 Marviken Full-Scale Critical Flow Test 4

The Marviken full-scale critical flow tests[5] can be used to assess the ability of computer codes to predict large pressure vessel blowdowns. Four major components are included: a pressure vessel originally designed to be part of the Marviken BWR nuclear power plant, a discharge pipe, a test nozzle with the minimum flow area in the system, and a rupture disk assembly. For Test 4, the nozzle had a minimum diameter of 0.509 m with a length/diameter ratio of 3.

Before the test, deionized water partially filled the vessel and was heated by taking water from the bottom of the vessel out through an electric heater and adding it back into the steam dome at the top of the vessel. This procedure produced a rather complicated initial temperature distribution in the vessel. Saturated steam filled the vessel region above the initial water level, and the water at the nozzle inlet had a substantial amount of subcooling (about 60 K). The test was initiated by release of the rupture disks and terminated after about 48 s by closing a ball valve in the discharge pipe.

The TRAC model of Marviken Test 4 includes four components. A zero velocity FILL models the vessel upper boundary; a semi-implicit PIPE models the vessel above the 2.6-m level, including the maximum diameter region and the top cupola; a fully implicit PIPE models the lower part of the vessel, discharge pipe, nozzle, and rupture disc assembly; and a BREAK component provides a pressure boundary condition at the rupture disk assembly lower boundary. Fifteen fluid cells were used in the semi-implicit pipe and 45 in the fully implicit pipe. It should be noted that there is no empirical break model in TRAC-P1A and that the break flows are, therefore, calculated directly from the basic flow equations.

TRAC results were compared with the Marviken blowdown flow rate and the pressures and temperatures at several locations. Figure 1 shows the TRAC mass flow rate compared with the flow rate derived from velocity (pitot-static) and differential pressure measurements. TRAC results agree very closely with the initial peak, somewhat underpredict the subcooled part of the blowdown, and agree very well with the saturated portion of the blowdown (20-45 s).

The pressure near the vessel top is shown in Fig. 2. All calculated vessel pressures are very close to the experimental results after the first few seconds. During this early period, the data show a dip probably due to delayed nucleation in the deionized water that TRAC-P1A does not model. The agreement of the calculated fluid temperatures with measurements at various locations in the vessel is similar to that for the pressure.

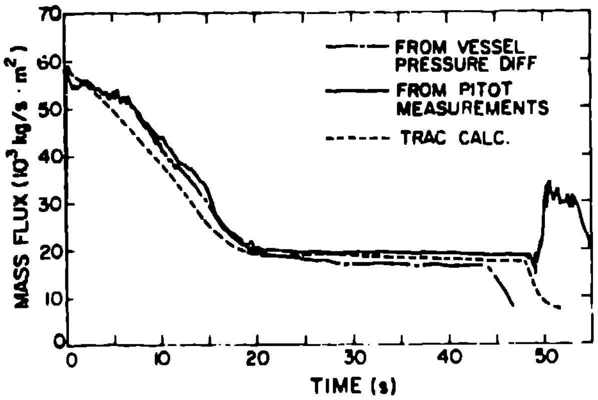

Fig. 1. Mass flux for Marviken blowdown Test 4.

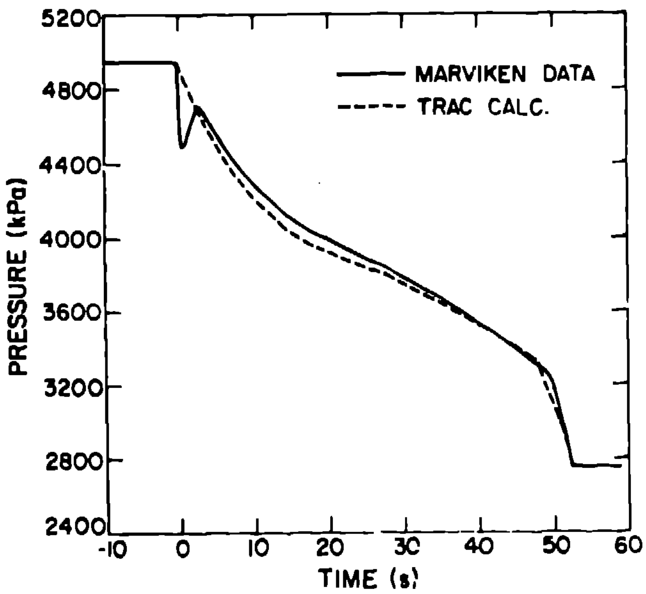

Fig. 2. Pressure near top of vessel for Marviken Test 4.

The good agreement between the calculated and measured results for Marviken Test 4 indicates that TRAC properly treats scale effects in one-dimensional critical-flow configurations. This calculation also identified the need for a delayed-nucleation model in TRAC. Such a model is now being added.

Calculations of six other Marviken tests, including two blind posttest predictions, indicated that two-dimensional flow effects in the nozzle become important as the nozzle length/diameter (L/D) ratio decreases. This effect was particularly noticeable in Test 24 where L/D=0.3. This indicated that a critical flow model may be required in certain situations to provide adequate 1-D modeling. Various models are now being investigated for possible inclusion in TRAC.

4.2 Creare Quasi-Static Countercurrent Flow Experiments

The Creare countercurrent flow experiments[6] investigated the effects of countercurrent steam flow rate, downcomer wall superheat, and ECC subcooling on the delivery of ECC water from the downcomer to the lower plenum. The basic component of the Creare test facility is a 1/15-scale (linear dimension), multiloop, cylindrical model of a PWR downcomer region. The Creare vessel can be arranged in at least six different geometrical configurations. The configuration used in the tests analyzed here is the so-called "base-line" configuration having a 0.0127-m (0.5-in.) downcomer gap and a "deep plenum" geometry. The vessel has four cold legs oriented 90° to each other. Three of the cold legs are assumed to be "intact" and are connected to ECC injection lines. A single "broken" cold leg connects to the pressure suppression tank. There are also four hot legs; however in the tests presently being considered, the hot legs are closed off.

The test procedures for the quasi-static countercurrent flow tests are as follows. Steam is injected at the top of the vessel, flows down the center of the vessel into the lower plenum, up the downcomer, and out the broken cold leg. After reaching the steady steam flow rate, water is injected simultaneously and equally into the three intact cold legs at a constant preset flow rate. After a short transient period, the plenum normally begins to fill. The test is run until the lower plenum is full or until the filling rate can be determined from strip chart records. A complete penetration curve is composed of a set of tests at a given liquid injection rate and liquid temperature with the steam flow rate varied over a range such that water delivery ranges from complete delivery to complete bypass.

The Creare vessel was modeled using 7 axial levels with each level subdivided into 2 radial and 8 azimuthal zones for a total of 112 mesh cells. The calculational procedure parallels that of the Creare experimental procedure. A steady-state calculation is performed to establish a constant reverse steam flow and lower plenum pressure. The intact cold legs are isolated and the broken cold-leg back pressure is selected to give the correct lower plenum pressure. This assures the correct liquid subcooling when the ECC is injected. This steady-state calculation is run until J^*_{gc} (dimensionless reverse steam flow rate, see Ref. 6) reaches a constant value. The transient calculation is then started from the steady-state initial conditions, but with the inlet conditions in the three intact cold legs specified to give the correct ECC injection flow and temperature.

The transient values of J^*_{gc} and J^*_{fd} (dimensionless water flow rate delivered to the lower plenum) are calculated at the bottom of the downcomer, and the collapsed liquid level in the lower plenum is calculated based on the liquid fractions in that region. The plotted values of J^*_{gc} and J^*_{fd} for each calculation are determined as follows. The value of J^*_{gc} is the initial steady-state value. This variable undergoes an initial transient following ECC injection and may not return to the full value due to steam condensation. The calculated value of J^*_{fd} is determined from the average lower plenum filling rate as is the case with the experimental results.

The Creare quasi-static countercurrent flow experiments covered a wide range of ECC flow rates and subcoolings. Four TRAC calculations were made to generate two complete penetration (or flooding) curves. These two curves are for the following ECC flow rates and injection temperatures: (a) 1.86×10^{-3} m^3/s and 373 K (30 gpm and 212°F) and (b) 3.78×10^{-3} m^3/s and 339 K (60 gpm at 150°F). The reactor scale injection flow rate is 3.78×10^{-3} m^3/s (60 gpm). The first case has very low subcooling and, therefore, the only effect that can produce bypass is the interfacial drag between the steam and the liquid. Figure 3 compares the results of the low subcooling case. Near the complete dumping location at $J^*_{gc} = 0.043$, the calculated J^*_{fd} is equal to 0.047, which is in excellent agreement with the measured value of 0.051. At a high steam flow rate, $J^*_{gc} = 0.14$, there is almost complete bypass of the injected liquid. At this steam flow rate, TRAC also predicts nearly complete bypass. The calculated J^*_{fd} is equal to 0.005, while the measured value is 0.004.

The second case has significant subcooling, and interfacial heat transfer now becomes significant in determining the quantity of liquid delivered. The penetration curve becomes much flatter, which means that the system tends to operate in either a complete bypass or complete delivery mode. Operation in the intermediate delivery/bypass range is thus experimentally difficult to achieve as the change in steam flow rate required to cause a transition from complete delivery to complete bypass is very small. Figure 4 compares the

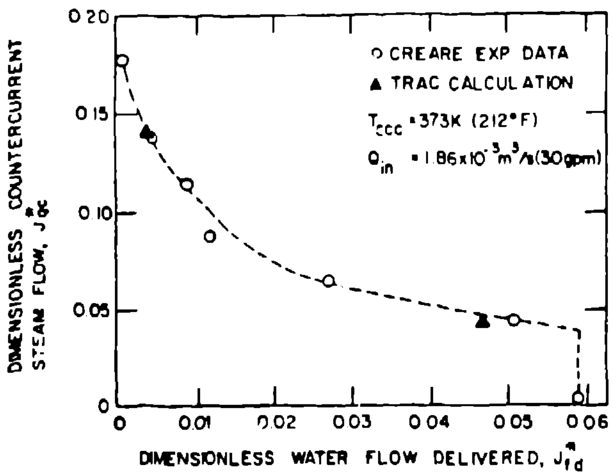

Fig. 3. Flooding curve for Creare low subcooling tests.

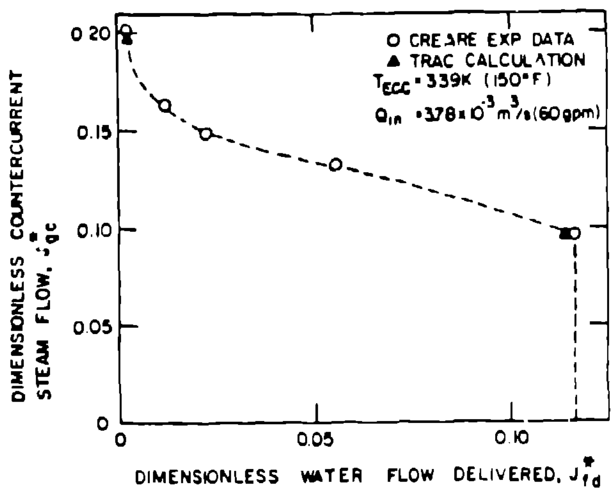

Fig. 4. Flooding curve for Creare high subcooling tests.

results for the high subcooling case. The complete dumping location at J^*_{gc} = 0.10 is again in excellent agreement. The calculation shows that almost all of the injected liquid is delivered to the lower plenum. At a steam flow rate of J^*_{gc} = 0.20, essentially all of the liquid is bypassed in both the experiment and in the TRAC calculation. Thus, the critical end points for this relatively high subcooling case are predicted quite well.

These calculations serve as "code-testing" for the interfacial momentum and heat transfer constitutive relationships in the three-dimensional VESSEL module. The comparisons between experimental data and TRAC calculations were in very good overall agreement. This indicates that TRAC is capable of satisfactorily predicting the bypass and penetration of ECC in annular downcomer geometries at this scale of experiment. Results of LOFT Test L1-4 (see Sec. 5.2) indicate that TRAC also accurately predicts bypass on a larger scale facility; however, further comparisons at even larger scale are needed.

4.3 FLECHT Forced Flooding Tests

The Full Length Emergency Cooling Heat Transfer (FLECHT) program is a series of reflood heat transfer simulation experiments designed to yield separate effects experimental data for use in evaluating heat transfer performance of emergency core cooling systems in pressurized water reactors. The FLECHT tests to date can be separated into the following four categories: The early high flooding rate tests,[7,8] systems effects tests (FLECHT-SET),[9-11] low flooding rate tests,[12-14] and skewed power profile tests.[15-16] The high and low flooding rate tests are of particular interest since these were performed with forced flooding injection, which minimized system effects, and are the simplest to evaluate for code model testing.

Except for the more recent skewed power tests, the experiments were performed with 100 full-scale, electrically heated, nuclear fuel-rod simulators in a square duct housing. The axial power profile of a nuclear rod was approximated by a step-wise variation in the number of heater wire coils per unit length. The total power during the experiment was programmed to follow the ANS power decay curve, plus 20%, normalized to an assumed delay time until the start of reflood (usually 30 s).

The test procedure was as follows. The lower section of the flow housing was filled with water to the bottom of the heated rod length. Power to the rods and housing was applied and maintained until the desired initial rod cladding temperatures were attained. Flooding at the specified rate was then initiated and simultaneously, the power was ramped on the desired decay curve. Temperatures and related fluid conditions were recorded until the bundle was completely quenched.

The single-channel geometry of these experiments lends itself very well to the use of the slab vessel option in TRAC. As a matter of fact, a one-dimensional representation was obtained by using only one cell per axial level. The base case model contained 9 axial levels in the core with each of these levels containing 5 fine-mesh axial intervals for the reflood heat transfer calculation. Conduction in the electrically heated rod was represented with 8 radial nodes. Initial conditions were set at measured input values for rod, housing wall, and fluid temperatures. Since sufficient experimental detail is not available to determine all necessary input values precisely, some interpolation or estimation of initial temperatures was performed.

Test conditions for the three cases presented here are given in Table I, and a summary of the calculated and measured results is given in Table II. TRAC-P1A predicts the maximum temperature (and hence the temperature rise) quite well for all three tests. For the high flooding rate case (Test 03541), the calculated turnaround time and quench time also agree very well with the data. This is not the case, however, for the low flooding rate tests where the code predicts early turnaround and quenching. The early turnaround times are believed to result from excess vapor generation calculated in the lower region of the rod bundle. Underprediction of carryover rates results in a rapid refill of the core region and accounts for early quenching.

TRAC tends to overpredict the heat transfer coefficient for the upper regions of the rod bundle from the turnaround point to the quench time. This resulted in wall temperatures that were lower than the experimental data. Consequently, the quench temperatures were also low as compared to the data.

The comparisons of the TRAC predictions and the FLECHT reflood data were used to evaluate the two-phase flow and reflood heat transfer models in the vessel component. It was apparent that the reflood heat transfer models in TRAC-P1A are less than satisfactory for predicting the low reflood rate phenomena observed in the FLECHT forced flooding experiments. Specifically, the quench front propagation, liquid entrainment, and transition and film boiling heat transfer models were items that required further development. These areas, along with difficulties encountered under gravity-flow and systems-response conditions, are being improved in the next code version (TRAC-PD2).

TABLE I

FLECHT EXPERIMENTAL TEST CONDITIONS FOR TRAC CALCULATIONS

Test Number	Pressure (MPa)	Inlet Fluid Temperature (K)	Flooding Rate (m/s)	Peak Power (kW/m)
03541	0.39	337.6	0.25	4.07
04831	0.28	324.8	0.04	3.12
02414	0.28	327.1	0.02	2.76

TABLE II

SUMMARY OF CALCULATED AND MEASURED RESULTS
AT THE FLECHT BUNDLE MIDHEIGHT

	Test 03541		Test 04831		Test 0[illegible]14	
	Exp.	TRAC	Exp.	TRAC	Exp.	TRAC
Initial Temperature (K)	1143	1144	1144	1144	1144	1144
Max. Temperature (K)	1193	1190	1333	1333	1453	1449
Temperature Rise (K)	50	46	189	189	308	305
Turnaround Time (s)	8	6	74	40	96	80
Quench Time (s)	71	72	219	170	345	210

A completely new reflood treatment that explicitly treats axial heat conduction near the quench front has been implemented. Tests to date are showing significantly improved results.

5. SYSTEMS-EFFECTS TESTS

Systems effects tests involve several coupled LWR components up to and including all the major primary system components arranged in a closed-loop configuration. These tests address only a portion of the LOCA. Unlike integral effects tests, however, systems effects tests omit some major feature of the transient. Examples are omission of ECC injection during blowdown or core heating during refill. To illustrate TRAC comparisons with these types of tests, we have chosen a heated blowdown test without ECC injection in the Semiscale Mod-1 facility and an unheated blowdown test with ECC injection in the LOFT facility. Further details of these posttest analyses are given in Ref. 3.

5.1 Semiscale Mod-1 Heated Blowdown Test S-02-8

The Semiscale Mod-1 test apparatus (Ref. 17) is an improved version of the Isothermal Semiscale system. In the Mod-1 system, nuclear heating is simulated by a core comprised of 39 electrically heated rods with both the power and volume scaled to a typical PWR in a ratio of approximately 1 to 3000. The test apparatus consists of a pressure vessel with simulated reactor internals; an intact loop with active steam generator, pump, and pressurizer; a broken loop with simulated steam generator, a simulated pump, and pipe rupture assemblies; and a pressure suppression system with header, auxiliary steam supply and suppression tanks.

Test S-02-8 was a simulation of a double offset shear (200%) cold-leg break. It differed somewhat from other Mod-1 tests in that the resistance of the simulated pump was reduced by a factor of about 4 below the more typical value. Prior to the test, the system was brought to a steady-state condition and blowdown was initiated by breaking the two rupture disks.

The TRAC model of Test S-02-8 contains every component modeled by TRAC, except an accumulator. The model contains a total of 111 fluid cells in one-dimensional components and 152 fluid cells in the three-dimensional vessel component. The initial conditions calculated with TRAC for use at the start of the blowdown are compared with the experimental data in Table III.

TABLE III

COMPARISON OF CALCULATED AND MEASURED INITIAL CONDITIONS
FOR SEMISCALE HEATED BLOWDOWN TEST S-02-8

Parameter	Units	Test Data	TRAC
Core Power*	MW	1.59	1.59
Intact loop cold leg fluid temperature	K	556.5	553.7
Hot to cold leg temperature differential	K	37.8	39.7
Pressurizer pressure	KPa	15600.0	15596.0
Pump mass flow rate	kg/s	7.35	7.38
Pump speed	Rad/s	295.3	296.0
Pump Pressure	KPa	283.0	268.0

*Input to TRAC-P1A

Differences are generally due to inconsistencies in the test data. None of these inconsistencies are felt to seriously affect the results of the transient analysis.

The comparison of TRAC-predicted lower plenum pressure with Test S-02-8 data[18] presented in Fig. 5 indicates that TRAC does a good job of predicting system performance. The slight underprediction of pressure beginning at 11 to 12 s is probably due to prediction of less superheat in the upper part of the core than was present in the actual test.

The most important variable that a LOCA analysis code calculates is the maximum cladding temperature. Figure 6 presents a comparison of the TRAC predictions of this variable with a band of temperatures that includes all of the heater rod cladding thermocouples in the highest power step in the Semiscale system. With the exception of a slightly advanced time to Departure from Nucleate Boiling (DNB), TRAC does an excellent job of predicting the cladding temperature response in the high power zone.

A meaningful comparison of predicted and test-derived core inlet mass flow rate is limited to the first 6 s after rupture due to the dead band in the core inlet turbine flow meter. This comparison is shown in Fig. 7, and indicates that TRAC predicts the magnitude of the immediate core flow reversal well, but predicts the core flow to return to a positive direction about 1 s before the test data.

Figure 8 shows that TRAC does an excellent job of predicting the hot-leg break mass flow rate. The small increase in the test data between 10 and 15 s is due to a slug of higher density fluid coming from the intact hot leg. TRAC calculations of pressurizer surge line flow (Fig. 9) and pump flow rate (Fig. 10) also agree well with test data, demonstrating that TRAC does an excellent job of predicting intact loop fluid flow rates.

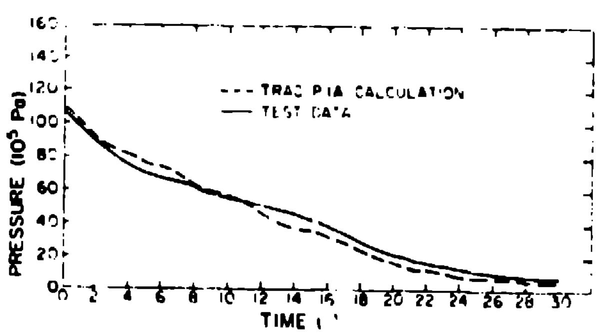

Fig. 5. Lower plenum pressure for Semiscale Test S-02-8.

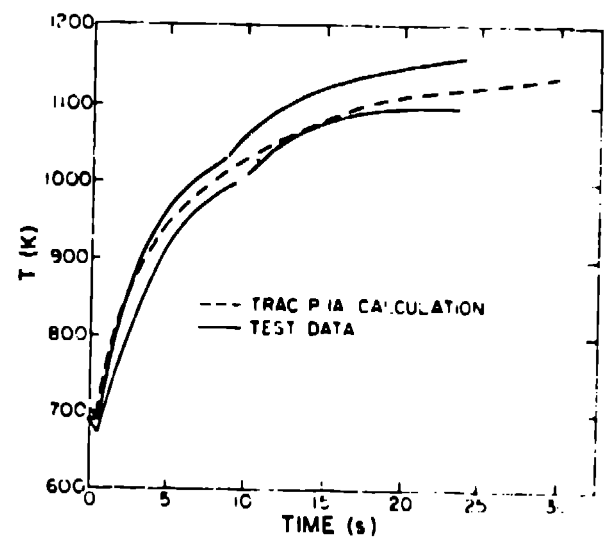

Fig. 6. Cladding temperature in high power zone for Semiscale Test S-02-8.

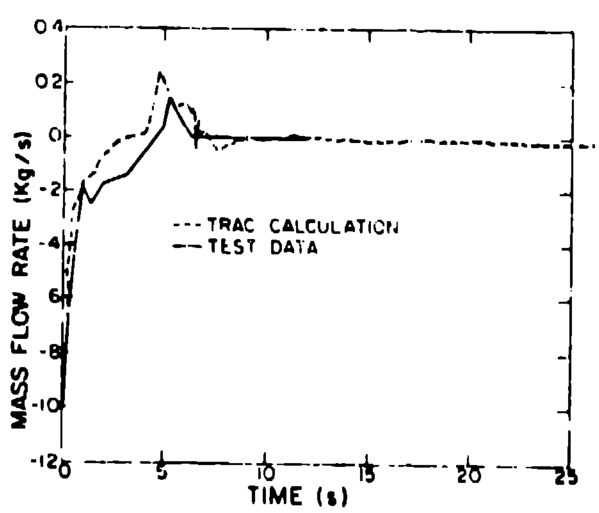

Fig. 7. Mass flow rate at core inlet for Semiscale Test S-02-8.

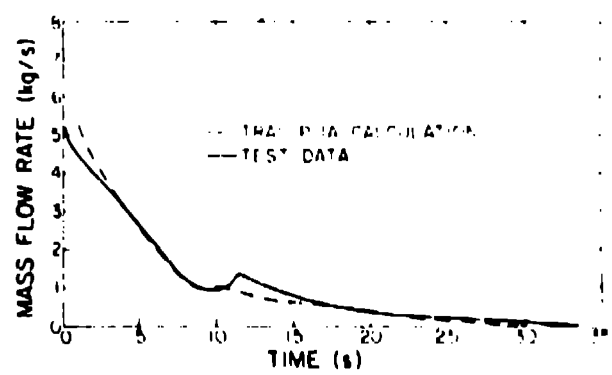

Fig. 8. Hot-leg break mass flow rate for Semiscale Test S-02-8.

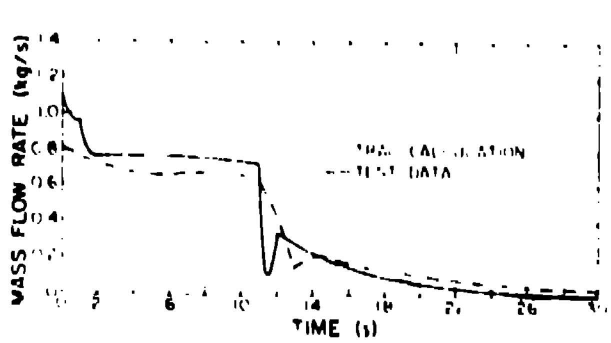

Fig. 9. Pressurizer surge line mass flow rate for Semiscale Test S-02-8.

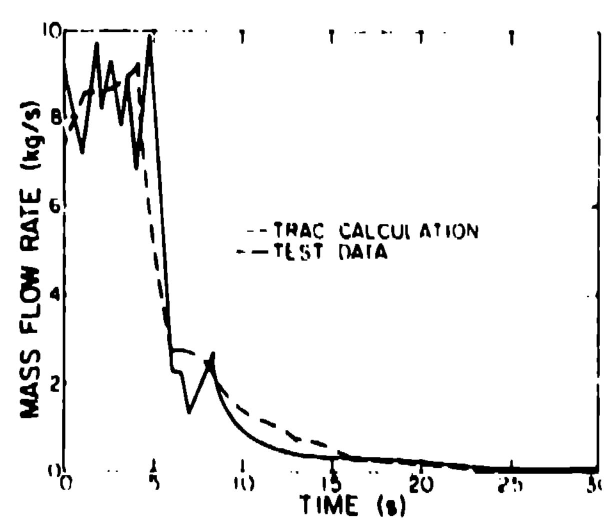

Fig. 10. Intact loop pump inlet mass flow rate for Semiscale Test S-02-8.

In summary, TRAC posttest analyses of Test S-02-8 are generally in very good agreement with test results and illustrate the ability of the code to accurately predict the thermal-hydraulic response of a PWR-type system during blowdown but without ECC injection.

5.2 LOFT Nonnuclear Isothermal Blowdown Test L1-4

The Loss of Fluid Test Facility (LOFT)[19] is a scale model of a large pressurized water reactor (LPWR). The volume scaling ratio between the LOFT system and the LPWR is approximately 1:60; flow and break areas are also scaled using the same ratio. The LOFT L1-4 system consists of a pressure vessel; an intact loop with a pressurizer, steam generator and two pumps; a blowdown loop with a simulated steam generator, a simulated pump and two quick-opening valves; and a pressure suppression system.

The pressure vessel contains a hydraulic core simulator, upper and lower plena, a downcomer, and a core support barrel. The blowdown loop is a volume-scaled representation of one loop of a four-loop LPWR. The simulated steam generator and pump consist of piping containing many orifice plates to achieve the desired hydraulic resistance. The intact loop has a volume approximately three times larger than the blowdown loop and represents three intact loops of a four-loop LPWR. This loop has a u-tube steam generator, two centrifugal pumps, and a pressurizer. The pressure suppression system simulates the large containment volume and back pressure of the LPWR and contains the blowdown effluent.

Test L1-4, a 200% double-ended cold-leg break with cold-leg ECC injection, was performed to provide information on delayed High Pressure Injection System (HPIS) and Low Pressure Injection System (LPIS) cold-leg injection, obtain data for evaluating downcomer bypass and mixing of the ECC with the primary coolant, and provide thermal-hydraulic data for comparison with test predictions and other experimental data for code assessment purposes. Prior to the blowdown, the primary system was brought to its initial temperature, pressure, and flow rate of 552 K, 15.75 MPa, and 268.4 kg/s, respectively, using the work-energy addition of the primary coolant pumps. The pressurizer heaters were de-energized and the blowdown was initiated by opening the two quick-opening valves. Electrical power to the primary system motor generator was terminated within 1 s after blowdown initiation, which allowed the pumps to coast down under the influence of the flywheels and the fluid dynamic forces on the pumps. ECC injection was directed to the intact cold leg during blowdown. Injection from an accumulator was initiated at a system low pressure trip of 4.24 MPa (absolute pressure). The HPIS pump was preset to inject at 1.085 x 10^{-3} m^3/s and to initiate at 24 s after blowdown. The LPIS pump was adjusted to initiate no sooner than 35.5 s after the initiation of blowdown, and its flow rate varies from 0 to 0.01 m^3/s, depending on the system pressure.

The LOFT system contains a variety of components interconnected in series and parallel branches. Furthermore, this system is complicated by area changes and orifice plates. TRAC models the blowdown and intact loops with one-dimensional components. The reactor vessel component is modeled with the three-dimensional VESSEL module. The TRAC model uses 26 components with 27 junctions and a total of 215 fluid cells. The reactor vessel component is divided into 9 axial, 2 radial, and 4 azimuthal segments for a total of 72 fluid cells. The upper and lower plena contain 4 and 8 fluid cells, respectively. The TRAC representation of the ECC system consists of an accumulator connected to a series of two tees. These tees are connected to two FILL modules specifying the HPIS and LPIS flows.

The TRAC calculation for this problem was performed in two stages. First, steady-state initial conditions were obtained by running a transient calculation (i.e., use of the transient option in TRAC) starting from an initial zero flow rate, a uniform pressure and temperature, and with the two quick-opening valves closed. As shown in Table IV, all the calculated steady-state parameters are within 3% of the measured[30] values. The blowdown portion of the calculation is performed by restarting from the dump file obtained from the steady-state calculation and activating the two quick-opening valves.

Figures 11 and 12 compare mass flow rates per system volume from the blowdown legs. The measured mass flow rate from the simulated pump side of the break (Fig. 11) is approximately constant in the interval 3-10 s. This is probably because the pressure losses associated with the many orifice plates in the simulated steam generator and pump result in an approximately constant upstream choking pressure. No attempt was made to model in detail these numerous orifice plates. The mass flow rate from the vessel side of the break (Fig. 12) decreases monotonically because the pressure losses in this leg are not sufficient to maintain a choking upstream pressure. Oscillations in the measured mass flow rate beginning at approximately 25 s are due to ECC bypass. TRAC predicts these oscillations in mass flow rate and fluid density very well.

Figure 13 compares the flow rate per system volume in the intact cold leg. The sharp initial increase and decrease in this variable at early times is due to the initial reactor vessel decompression. Oscillations in the measured flow rate at the intact cold leg beginning at approximately 23 s are due to the ECC injection. TRAC predicts the initial sudden increase and decrease as well as the later oscillations in the flow rate.

A comparison of the measured and calculated reactor vessel liquid mass is given in Fig. 14. TRAC predicts the reactor vessel liquid mass quite well, including the time to refill and the subsequent oscillations in the reactor vessel liquid mass due to slugging ECC delivery. The early time (0-20 s) rapid depletion of mass in the data is thought to be due to the lack of water level instrumentation within the core. Thus, the core water level is assumed to be equal to the water level measured by the downcomer instrument stalks.

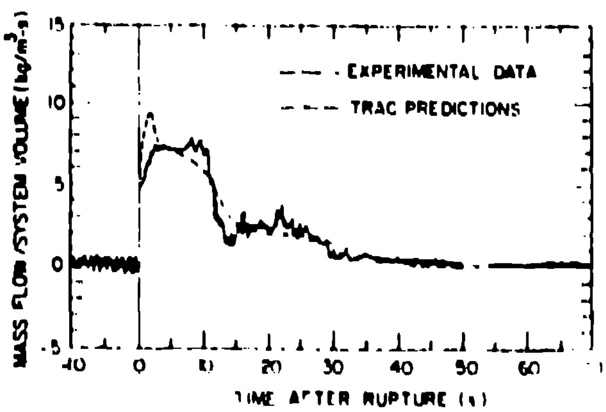

Fig. 11. Mass flow rate in the broken hot leg for LOFT Test L1-4.

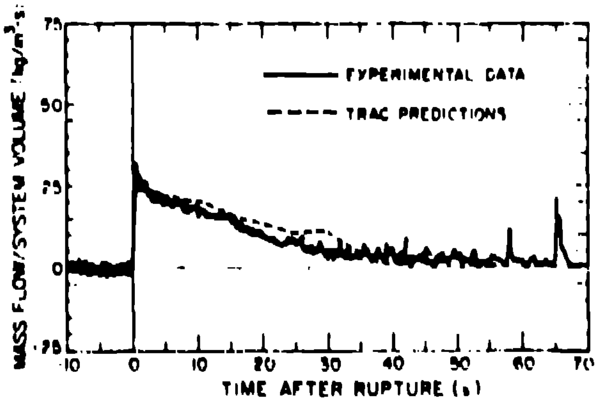

Fig. 12. Mass flow rate in the broken cold leg for LOFT Test L1-4.

TABLE IV

COMPARISON OF CALCULATED AND MEASURED INITIAL CONDITIONS FOR LOFT TEST L1-4

Parameters	Calculated	Measured
Loop Mass Flow Rate (kg/s)	261.6	268.4
Pressurizer Pressure (MPa)	15.7	15.75
Pressurizer Water Mass (kg)	419.2	418.8
Pressurizer Water Level (m)	1.17	1.16
Steam Generator Primary Side Pressure (MPa)	15.70	15.75
Steam Generator Primary Side Inlet/Outlet Temperature (K)	552.3/553.0	554.0/552.0
Steam Generator Secondary Side Temperature (K)	552.0	552.0
Steam Generator Secondary Side Pressure (MPa)	6.66	6.65
Core Inlet/Outlet Temperature (K)	552.7/552.6	552.0/554.0
Total System Water Volume at 552 K and 15.75 MPa (m^3)	7.72	7.72
Differential Pressure in Intact Loop Across Primary Pumps 1 and 2 (MPa)	0.139	0.140

LOFT Test L1-4 provided an opportunity to test the ability of TRAC to predict systems effects during the blowdown and refill stages of a LOCA. With the exception that the core simulator did not contain any heat-generating rods, all the TRAC components needed to analyze a full-scale LWR LOCA were exercised in this problem. The good agreement obtained between calculated and measured results indicates that TRAC-P1A provides a good representation of systems effects during blowdown and refill for a facility whose scale is intermediate between Semiscale and a full-scale PWR. In particular, these results indicate that the effects resulting from cold leg ECC injection and bypass during an unheated blowdown/refill transient are properly represented by the physical models and correlations in TRAC.

6. INTEGRAL-EFFECTS TESTS

Integral-effects tests involve all the major components and phenomena that are expected during one or more phases of a LOCA. Because LOFT is the only nuclear-heated integral-effects test facility available, we have chosen two LOFT nuclear tests for comparison with TRAC results. The first test (L2-3) is a double-ended cold-leg large-break LOCA with ECC injection. The second test (L3-1) is similar, except that it is a single-ended small-break LOCA. Calculated results for both tests were obtained from pretest predictions[21,22] using TRAC-P1A.

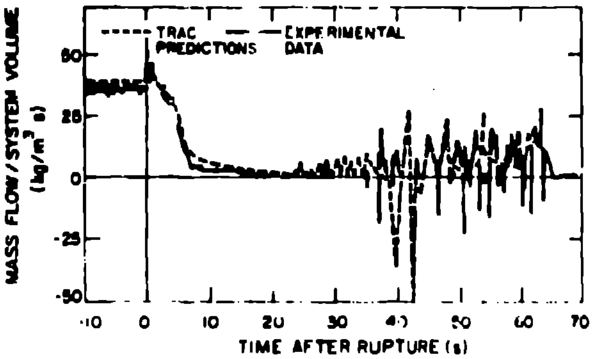

Fig. 13. Mass flow rate in the intact cold leg for LOFT Test L1-4.

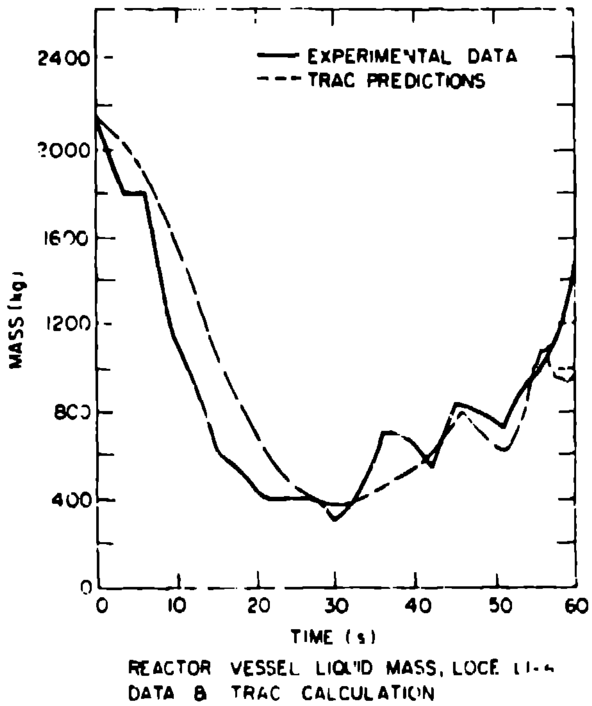

Fig. 14. Reactor vessel liquid mass for LOFT Test L1-4.

6.1 LOFT Large-Break Nuclear Test L2-3

The nuclear LOFT facility is the same as described previously for non-nuclear test L1-4, except that the nuclear core is in place. Tests in the L2 series[23] are loss-of-coolant experiments conducted at gradually increasing initial power levels to determine the nuclear core and integral systems response during all phases of a LOCA. Experiments in this series simulate a 200% double-ended offset shear in the cold leg with ECC injection. Test L2-3 was conducted from an initial power level of 37 MWt (75% of full power), 15.1 MPa system pressure, 593 K hot-leg temperature, and 200 kg/s core flow rate.

The TRAC model for L2-3 contains 27 components with a total of 322 fluid mesh cells. The entire reactor vessel is modeled using the three-dimensional, two-fluid VESSEL module, while all other components are modeled using one-dimensional modules. There are a total of 12 axial levels in the vessel, including 5 axial levels within the core region. Each level contains 12 fluid cells within the core radius and 4 fluid cells within each downcomer level. Thus, there are a total of 192 fluid cells within the vessel, including 60 within the core itself. The reflood fine mesh is initiated 10 s after accumulator injection is started. There are 5 uniform fine-mesh intervals for each axial level, giving a total of 25 fine cells.

The initial system thermal and hydraulic conditions for the pretest calculation of LOFT Test L2-3 were obtained using the steady-state option of TRAC-P1A. Calculated initial system conditions are compared in Table V with the specified test conditions. The actual test initial conditions were as specified within the uncertainty of the measurements. Good agreement was obtained between the calculated and specified initial conditions.

TABLE V

STEADY-STATE INITIAL CONDITIONS FOR LOFT TEST L2-3

Parameter	TRAC-P1A	LOFT Initial Conditions
Core Power (MW)*	37.2	37.2
Max Heat Generation Rate (kW/m)*	39.4	39.4
Hot Leg Temp (K)	591.4	591.5 $\pm$ 1.1
Core T (K)	35.6	35.8
Intact Loop Flow (kg/s)	185.4	187.7
System Pressure (Pa)	1.50×10^7	1.50×10^7

*Input to TRAC-P1A.

Pretest calculated results obtained with TRAC-P1A are compared with the data[24] in Figs. 15 to 18. The upper plenum pressure is shown in Fig. 15. Although the agreement is generally quite good, the calculation depressurized more slowly than the data and resulted in delayed ECC injection. Figure 16 shows the broken loop cold-leg mass flow. The early underprediction of the break flow is consistent with the overprediction of system pressure. The broken cold leg was predicted to void less rapidly than the data and showed more evidence of ECC bypass (sustained high density).

The calculated velocities in the intact loop demonstrated trends similar to the data; however, the magnitudes were not always in good agreement. The intact loop hot- and cold-leg densities generally demonstrated good agreement.

Figure 17 compares the calculated and measured cladding temperatures in the central ring for elevations between 0.533 and 0.762 m (the high-power zone). The data show an early dryout and rewet followed by a second dryout. TRAC-P1A predicted only the early dryout and, therefore, overpredicted the peak cladding temperature. The time to final quench was also overpredicted.

TRAC previously had demonstrated an inability to calculate the hot rod dryout-rewet phenomena in LOFT for Test L2-2. It was suspected that the minimum film boiling correlation in TRAC-P1A might be at fault since it was based on low pressure data. The minimum film boiling correlation of Iloeje[25] was subsequently tried and found to give much better results for Test L2-2. Because of this, a blind pretest prediction of Test L2-3 was also made using the Iloeje correlation in TRAC-P1A. The results of this calculation are also shown in Fig. 17 for the highest power region in the core. For this calculation, the code predicted the initial dryout and rewet and the sustained dryout. The calculation ended just before the rewet time in the data

The calculated cladding temperatures for the lower power rods at the core outer periphery were in quite good agreement with the data for both pretest predictions (with and without the Iloeje correlation). An example of this is shown in Fig. 18.

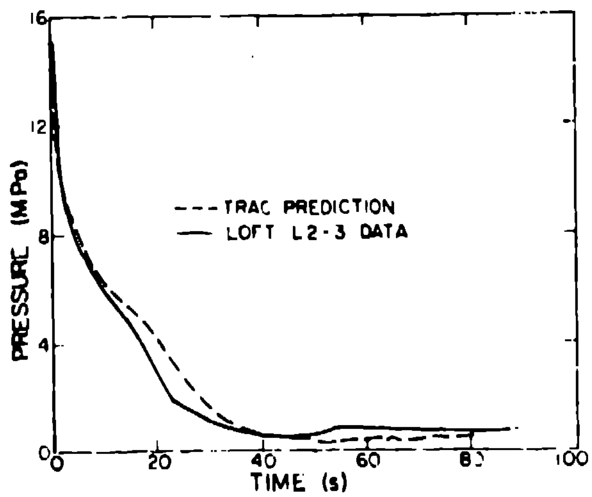

Fig. 15. Upper plenum pressure for LOFT Test L2-3.

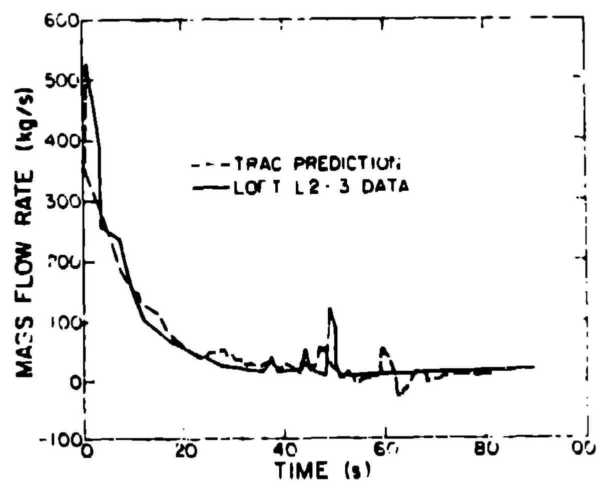

Fig. 16. Broken loop cold-leg mass flow rate for LOFT Test L2-3.

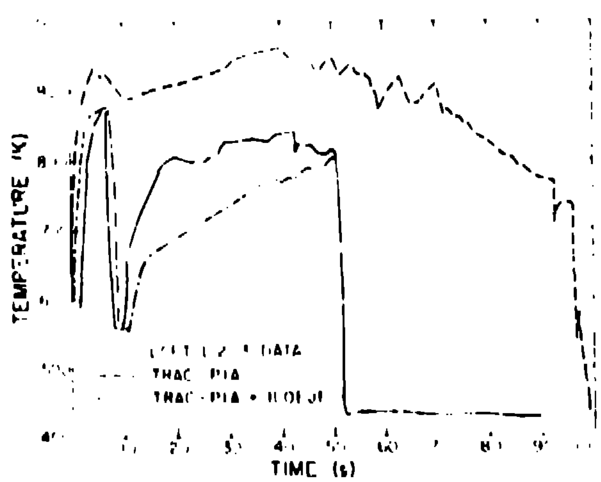

Fig. 17. TRAC pretest predictions of cladding temperatures at core midplane for LOFT Test L2-3.

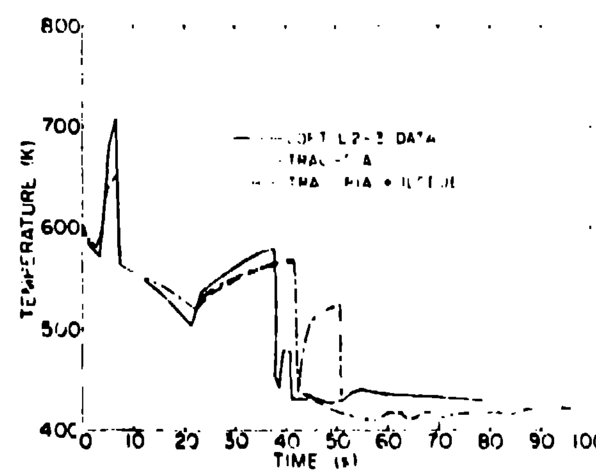

Fig. 18. TRAC pretest predictions of cladding temperatures at outer core periphery for LOFT Test L2-3.

Difficulties in predicting the dryout behavior in parts of the core are related to the underprediction of the broken cold-leg flow during the first 5 s of the transient. Underprediction of the cold-leg break flow is due to the fact that the vapor generation model does not account for the effect of delayed nucleation. Briefly stated, when the fluid cells in the break nozzle depressurize to saturation, immediate vapor generation occurred and the critical flow characteristics changed from nonequilibrium/subcooled to near equilibrium/saturation conditions, resulting in underprediction of critical flow. The problem was not as severe in the broken loop hot leg because of the higher flow resistance and temperature.

The calculated behavior in the intact loop was qualitatively good. The detailed comparisons might improve with a better calculation of the break flow and core thermal response. The comparison to the core thermal response indicated difficulties in calculating both dryout and rewet. The results with the Iloeje correlation indicated that an improved minimum film boiling correlation is necessary to calculate the rewet phenomena in LOFT. Before any one correlation is accepted for general use, its ability to predict data from other facilities must be demonstrated. Further testing against experimental data is underway.

6.2 LOFT Small-Break Nuclear Test L3-1

Following the Three-Mile Island (TMI) accident, the planned small-break tests (L3 series) in LOFT were moved ahead and the remaining large-break tests (L2 series) were postponed. Test L3-1 was a single-ended small break in the cold leg with ECC injection. The break orifice was sized so that the break flow is greater than the ECC flow and the system will depressurize. Test L3-1 was to be conducted from steady-state conditions of 50 MWt initial power and primary coolant loop flow of 478.8 kg/s. The actual conditions differed slightly from these values. ECC consisted of the HPIS, the accumulator, and the LPIS injecting into the intact loop cold leg. The broken loop hot-leg isolation valve was closed. The break simulator orifice installed in the broken loop cold leg had a 1.6-cm i.d. The test was initiated by manually tripping the reactor. When the control rods reached bottom, the pumps were tripped, and the quick opening blowdown valve in the broken loop cold leg was opened.

The TRAC model for Test L3-1 used similar components to that for L2-3, except that fewer mesh cells were used to speed up the calculation for the much larger transient time of Test L3-1. The L3-1 model contained 24 separate components involving 124 fluid cells (36 in the vessel).

The power decay curve used in the calculation was based on the proposed 1977 ANS 5.1 standard since the information on the 40 hours operation decay curve for LOFT was unavailable. This difference results in significant over-prediction of decay power late in the transient (approaching 21% at 1500 s). The pump coastdown was specified based on L3-0 data.[26]

The initial steady-state conditions were again calculated with the generalized steady-state option in TRAC using the specified power of 50 MWt. A comparison of the calculated initial conditions is made in Table VI with the specified and actual test conditions. Note that the actual initial conditions deviated somewhat from those specified; however, the calculated initial conditions agree within the uncertainty of the measurements for most of the variables compared.

At the beginning of the transient calculation, a trip was set to initiate core power decay and pump coastdown. HPIS was initiated on a trip signal based on the intact loop hot leg pressure. The accumulator injection was passively initiated by a check valve opening in response to falling system pressure. The calculation was completed through 1500 s. At the end of the calculation, the LPIS had not come on.

Comparisons shown here are with data from the L3-1 quick-look report.[27] More detailed comparisons with the data in the L3-1 full data report[28] are in progress. Figure 19 shows the comparison of the calculated hot-leg pressure and the data. The pressure was overpredicted for the first 1 200 s. The calculated pressure rise from 150-250 s was related to the loss of natural circulation cooling, and the core, while not drying out, began to heat slowly. The pressure rise was terminated by clearing the intact loop seal and venting steam from the hot-leg side to the break. Experimentally, the loop seal did not clear. Apparently, leakage through the reflood assist bypass valves, which was not modeled in the calculation, was sufficient to release steam from the hot-leg side without forcing the loop seal to clear.

The overprediction of pressure was the direct result of underpredicting the cold leg break flow (Fig. 20). (The first five data points were shown as discrete points in the quick-look report.) The subcooled critical flow was significantly underpredicted, and then from 50-400 s, the calculated flow leveled out on a plateau of much longer duration than can be inferred from the data. The calculated and measured cladding temperatures essentially followed saturation with no sustained dryouts.

As discussed earlier, the main reason for underpredicting the cold leg break flow was the lack of a delayed nucleation model in TRAC-P1A. In addition, it was found that small amounts of void were being incorrectly convected into the nozzle even though the water was significantly subcooled.

TABLE VI

LOFT TEST L3-1 INITIAL CONDITIONS

Parameter	EOS*	TRAC	Actual
Power (MW)	50.0	50.0	48.9 ± 1.0
Intact loop flow (kg/s)	470.8	481.4	484.0 ± 6.3
Intact loop hot leg pressure (MPa)	14.95	14.95	14.85 ± 0.04
Pressurizer liquid volume (m^3)	0.634	0.634	0.620 ± 0.008
Intact loop cold leg temperature (K)	556.7	556.5	554.0 ± 3.0
Intact loop hot leg temperature (K)		576.0	574.0 ± 1.0
Pump speed (both) (rad/s)		324.0	323.0 (Pump 1)
Steam generator secondary			328.0 (Pump 2)
Pressure (MPa)		4.89	5.43 ± 0.11

*Experiment Operating Specification.

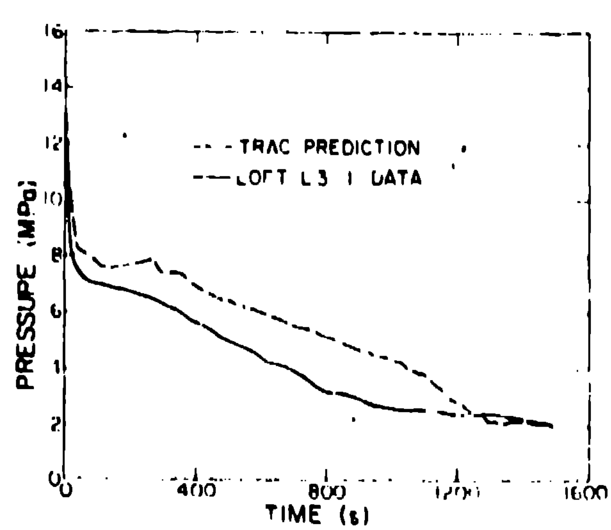

Fig. 19. Comparisons of the pretest prediction of hot-leg pressure to the data (PE-PC-5) for LOFT Test L3-1.

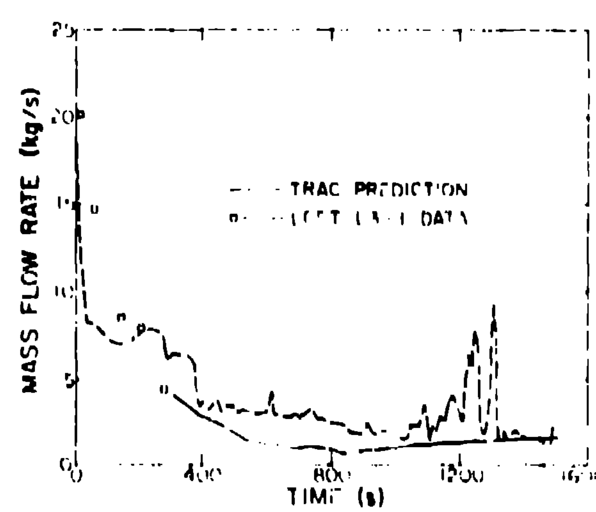

Fig. 20. Comparison of the pretest prediction of the broken-loop break flow with the data for LOFT Test L3-1.

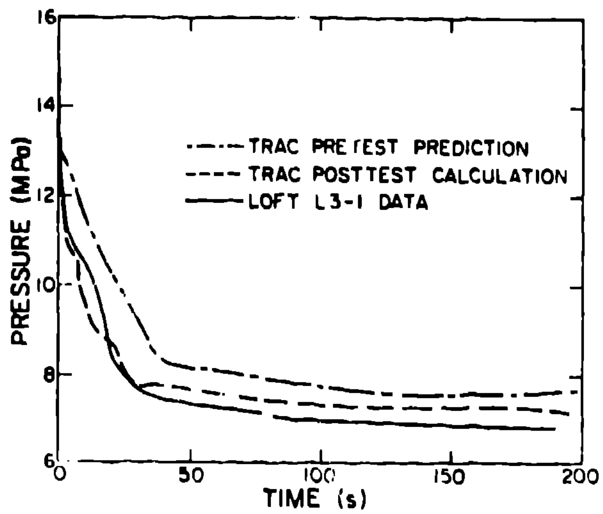

Fig. 21. Comparison of posttest and pretest calculated intact loop cold-leg pressure to LOFT Test L3-1 data (PE-PC-1).

A posttest calculation was performed with these two deficiencies corrected. Comparisons of the posttest and pretest calculated intact loop cold-leg pressures to the data are shown in Fig. 21. While the pretest calculation overpredicted the pressure throughout the 200 s shown, the posttest calculation is in better agreement with the data. The posttest calculation pressure comparison further supports the conclusions drawn relative to the break mass flow comparisons.

In addition to difficulties in calculating the subcooled break flow, other factors affecting the calculated pressure response include:

1. Use of the 1977 ANS decay curve (instead of the LOFT 40-hour operation decay curve) overestimates the decay heat.

2. Heat losses to the environment were not modeled, and these could be significant for this long-term transient.

3. Bypass flow through the reflood assist line was not modeled.

Code problems identified in the L3-1 analyses are being rectified in the next code version (TRAC-PD2). Input-related problems are currently being studied.

SUMMARY AND CONCLUSIONS

The results presented in this paper are only a sampling of the experimental assessment comparisons performed to date for the TRAC code. They are representative of the broader assessment set, however, and can serve as a basis for evaluating the capabilities of the current version of TRAC and establishing areas of needed improvement. In general, the comparisons have been very encouraging. They have indicated that the basic modeling and numerical framework being developed in TRAC is fundamentally sound. The comparisons have identified several areas where specific models needed to be improved, however. These improvements are being implemented in later versions of the code.

Experience to date has shown that TRAC generally does an adequate job of predicting blowdown and refill behavior. In particular, the multidimensional two-fluid treatment of the vessel downcomer flow seems to do an excellent job of predicting bypass behavior. The one-dimensional treatment of loop component flow also appears to do a good job, including the adequate prediction of most break-flow situations (without the use of an empirical break flow model). As pointed out in the paper, however, several improvements in the constitutive relationships are being implemented to alleviate specific difficulties, such as the underprediction of subcooled break flows.

The modeling of reflood behavior in the early version of TRAC has been less satisfactory. Although the approach of using a quench front velocity correlation did a reasonable job of predicting high flooding rate FLECHT experiments, the comparisons with data were less satisfactory for low flooding rate tests. This was partly due to an inadequate treatment of liquid carryover and, consequently, rod precooling. An entirely new reflood treatment is being implemented into TRAC-PD2 along with improvements in the entrainment modeling.

An important conclusion to be drawn from the early experimental assessment of TRAC is whether or not it is feasible to develop a code that can predict a broad range of experiments without allowing numerous user-selected options and parameter variations from one test to the next. As mentioned in the introduction, this is believed to be an essential element in providing confidence that a code has predictive capability for situations where direct experimental data does not exist (e.g., an actual reactor under accident conditions). Our experience to date indicates that this is indeed possible, although quite difficult. The demands on the fluid dynamics modeling are enormous since the flow-regime-dependent constitutive equation package must adequately recognize and model a broad range of constantly changing two-phase flow conditions.

The experimental assessment procedure becomes very tedious for this type of approach. Whenever significant changes are made to the modeling, the impact on the entire experimental assessment set must be evaluated. This becomes very time consuming as the number of experiments encompassed in the set becomes large. Nevertheless, experimental comparisons of the type presented in this paper are demonstrating that the basic approach is viable. Further, we believe that as the assessment set is successfully expanded to encompass all available data (including planned full-scale separate effect tests), a solid technical basis for predicting full-scale LWR behavior will have been provided.

ACKNOWLEDGEMENTS

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