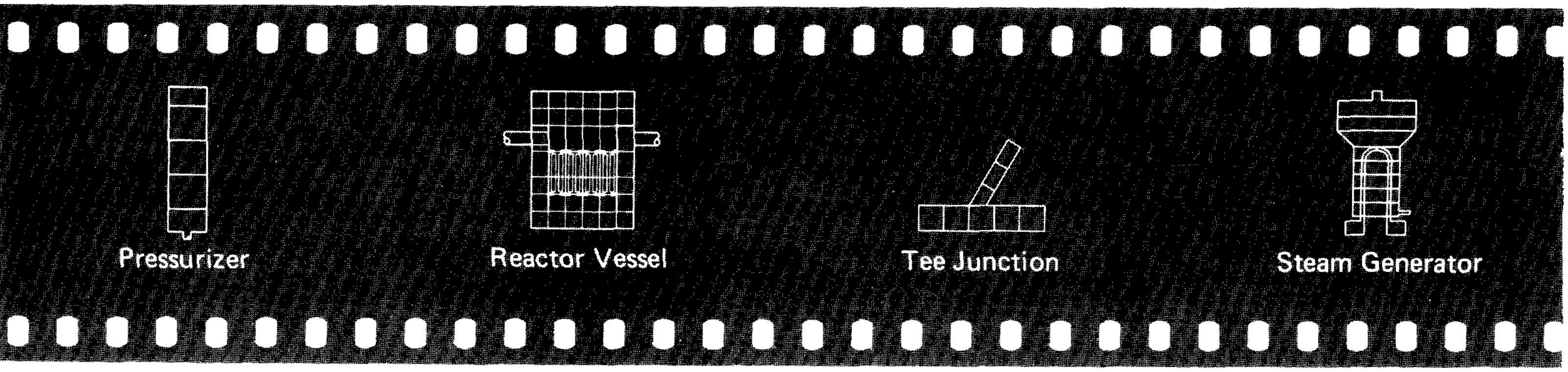


ACCIDENT SIMULATION WITH

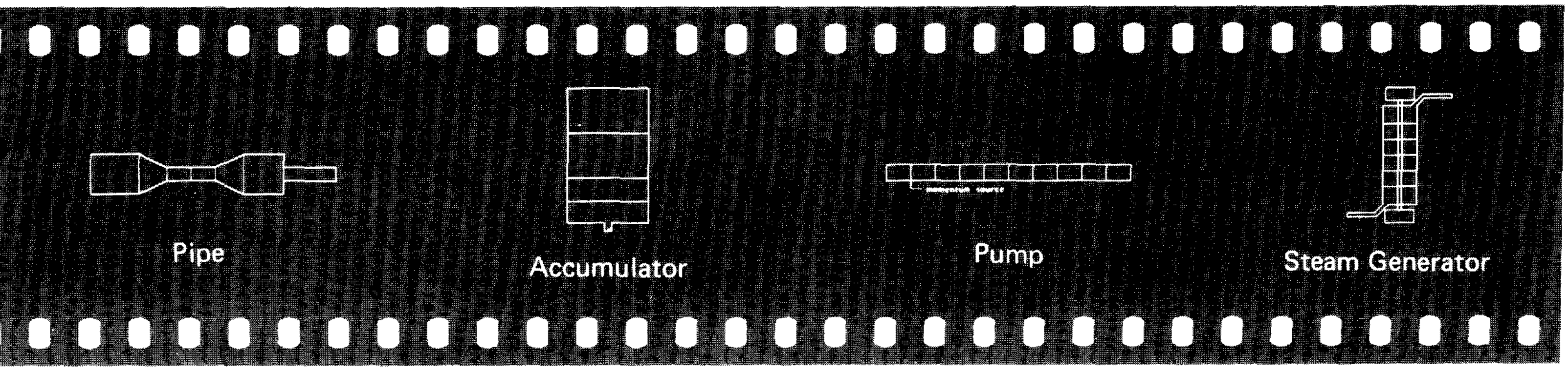
by John C. Vigil and Richard J. Pryor

An easy-to-use systems code can simulate the entire course of an accident in any light-water reactor system. Its predictive capabilities are being applied to current reactor-safety issues.

Imagine using an erector set to construct models of water-cooled reactors with any specified design. Imagine, too, that these are working models that can reproduce the behavior of full-scale reactors under accident as well as normal conditions. Such an erector set has been developed at Los Alamos and is available for use by researchers and engineers in the reactor community. Known as TRAC, for transient reactor analysis code, it consists of a large set of computer subprograms that can be put together to simulate the complex phenomena that may occur during any specified transient in any realistic reactor design. There are subprograms for the reactor components—the reactor core, the pipes, the pressurizer, the valves, the steam generators, the pumps, and the accumulators—and others for the physical processes—steam-water fluid dynamics, heat generation in the core, and heat transfer between the two phases of the coolant and between the coolant and the solid structures. When assembled into a large systems code and run on a high-speed computer, these subprograms simulate numerically the complete course of reactor transients, most notably the loss-of-coolant accident.

Los Alamos was asked to develop this versatile computer code to provide realistic predictions of reactor response to a large-break loss-of-coolant accident. The Laboratory began this task in early 1975, and less than three years later, TRAC became the first program to provide a continuous analysis of all phases of a loss-of-coolant accident in a full-scale four-loop pressurized-water reactor. Since then, other versions of TRAC have been developed with emphasis on either shorter running time or more detailed analysis. In addition, TRAC was the basis of a detailed version for boiling-water reactors developed at Idaho National Engineering Laboratory.

The accuracy of the most recent ver-



TRAC

sion of TRAC (TRAC-PD2) for large-break accident analysis has been extensively tested against small-scale experiments and integral tests at facilities such as LOFT and Semiscale. The first full-scale test of TRAC was its analysis of the first few hours (before the core was damaged) of the Three Mile Island accident. The results showed that the code is also applicable to small-break, multiple-failure accidents. Current applications of TRAC are in this area. To better handle these complex accidents, a new version of the code is being developed to include models for the turbine-generator and feedback controls. Numerical methods are also being improved to increase computing speed so that long-duration transients can be analyzed more efficiently.

TRAC and the Bounding Accident

Although extremely unlikely, the loss-of-coolant accident resulting from a large, double-ended break in the primary coolant system of a pressurized-water reactor (Fig. 1) has long been considered the bounding accident—the worst that could happen—and the accident against which the performance of emergency core-cooling systems is tested in the licensing process.

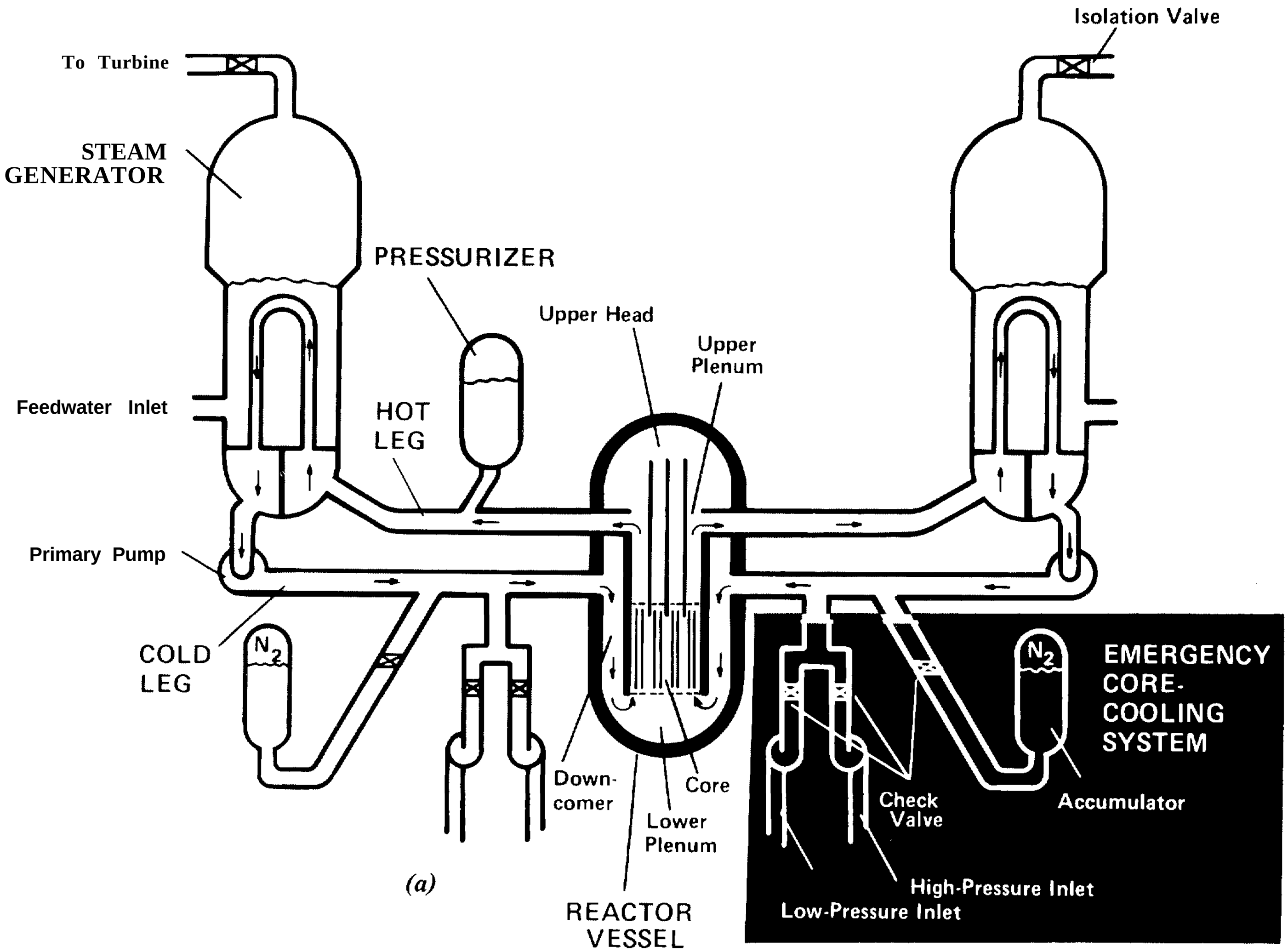
TRAC was designed specifically to simulate the large-break accident. Although this large systems code only approximates the intricate geometry of the plant and the physical processes that occur, it does simulate many complex phenomena that have been identified as important through small-scale experiments and more detailed computer studies of individual components.* Among these phenomena are critical flow, multidimensional effects, countercurrent fluid flow, fuel-rod quenching, and steam binding.

The course of a large-break accident has three main phases: blowdown, during which the primary system depressurizes and the coolant flashes to steam; bypass/refill, during which emergency cooling water refills the lower plenum to the bottom of the fuel rods; and reflood, during which water refills the core and cools the fuel rods.

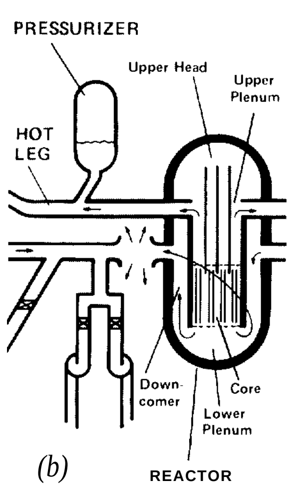
TRAC analyses of a standard four-loop pressurized-water reactor predict that, if all systems operate as designed, the fuel rods will be cooled within approximately three minutes and that no core damage will occur. These calculations also show that the NRC-specified assumptions are indeed conservative. For example, emergency cooling water will penetrate the lower plenum and reflood the core more rapidly than predicted by the licensing analyses.

Accident details and TRAC predictions outlined below will introduce the reader to the complex fluid-dynamics and heat-transfer problems that TRAC has addressed.

**See "Detailed Studies of Reactor Components" is this issue.*



(a)



(b)

Fig. 1. Coolant flow pattern through the primary system of a pressurized-water reactor (a) during normal operation when coolant flows down the downcomer and up through the core and (b) early in blowdown when coolant flows up the downcomer and out the broken cold leg.

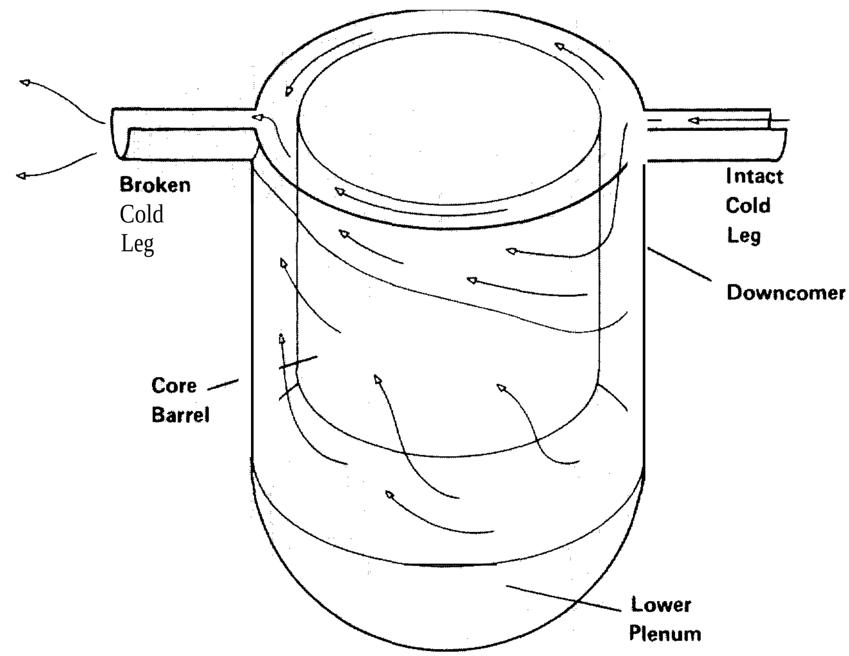


Fig. 2. ***Steam-water flows in the downcomer during bypass when emergency coolant swirls around the downcomer and out the cold-leg break.***

BLOWDOWN. Following a sudden, large break in a cold-leg pipe, the large pressure difference between the primary system (150 bars*) and the containment (-1 bar) forces water rapidly out the break (see Fig. 1). The rate at which water escapes is limited by the choking phenomenon, or critical flow. At first, the pressure is high enough that only subcooled water is discharged. Then, when the primary system pressure has fallen to the saturation pressure. the coolant flashes to steam and a two-phase mixture is discharged. Primary pump performance degrades drastically during this period.

During blowdown, all the water in the pressurizer, which maintains primary system pressure during normal operation, discharges into one of the hot legs.

The high-pressure injection system, consisting of low-flow-capacity pumps, turns on automatically early in blowdown and injects emergency coolant into the cold legs.

During all phases of the accident, the heat that may damage the core comes from two sources, reverse heat transfer in the steam generators and decay heat in the core. Reverse heat transfer occurs as the primary system pressure falls below that of the secondary system (-70 bars); the primary coolant is then heated by the secondary system. This accelerates "voiding," or coolant vaporization, in the core, a process that considerably reduces the efficiency of heat transfer from the fuel rods to the coolant. Although fission is halted automatically as the water in the core vaporizes (voiding has a very large and negative effect on the reactivity of the core), decay heat continues to be generated by fission products. The fuel rods dry and their temperature begins to rise, although some cooling is provided by the surrounding two-phase mixture.

For a large, double-ended break in a cold leg, TRAC predicts that blowdown lasts approximately 15 seconds. The calculations also show that it is during this phase of the accident that the fuel cladding reaches its maximum temperature, -950 kelvin. This temperature is considerably lower than the maximum (-1500 kelvin) allowed by the licensing guidelines.

During blowdown, some of the water in the lower plenum boils away or is swept out by high-velocity steam moving down through the core and up the downcomer to the broken cold leg. The amount of water remaining in the lower plenum determines the duration of the next phase of the accident.

BYPASS/REFILL. The second phase of the accident begins when the primary system pressure falls below that of the nitrogen in the accumulators (45 bars). Then, the check valves that normally isolate the accumulators from the primary system open, and expanding nitrogen forces water into the downcomer through the intact cold leg,

TRAC calculations show that, at first, water from the accumulator cannot reach the lower plenum. Instead, it is swept around the downcomer and out the broken cold leg (Fig, 2) by the countercurrent flow of steam. The steam is generated by flashing as the primary

*1 bar = 10^5 pascals ≅ 1 atmosphere.

system pressure falls and by boiling as heat is transferred from structural materials. Vapor flow toward the subcooled accumulator water increases as condensation decreases the local pressure. Water from the accumulator continues to bypass the lower plenum for approximately 10 seconds. Then, as the countercurrent steam velocities decrease, water begins to penetrate the lower plenum and refill begins.

During refill, multidimensional effects can occur in the downcomer with water flowing down one portion and steam moving up the diametrically opposite portion. Alternate "storage" and "dumping" of emergency coolant also takes place as the water's downward flow is held up periodically until a quantity collects that is sufficient to overcome the upward steam pressure. Refill lasts for about 10 seconds and ends when the water level in the lower plenum reaches the bottom of the fuel rods. To provide this realistic description of bypass/refill, TRAC uses a two-fluid thermal-hydraulics model and at least a two-dimensional representation of the downcomer geometry.

REFLOOD. Emergency core cooling culminates in the several minutes of reflood during which water refills the reactor vessel and quenches the fuel rods. The primary source of emergency coolant for reflood is water pumped into the cold legs by the low-pressure injection system. This system activates automatically when the primary system pressure falls below about 6 bars.

At the beginning of reflood, the fuel rods are relatively hot because heat transfer has not been very effective during most of blowdown and all of

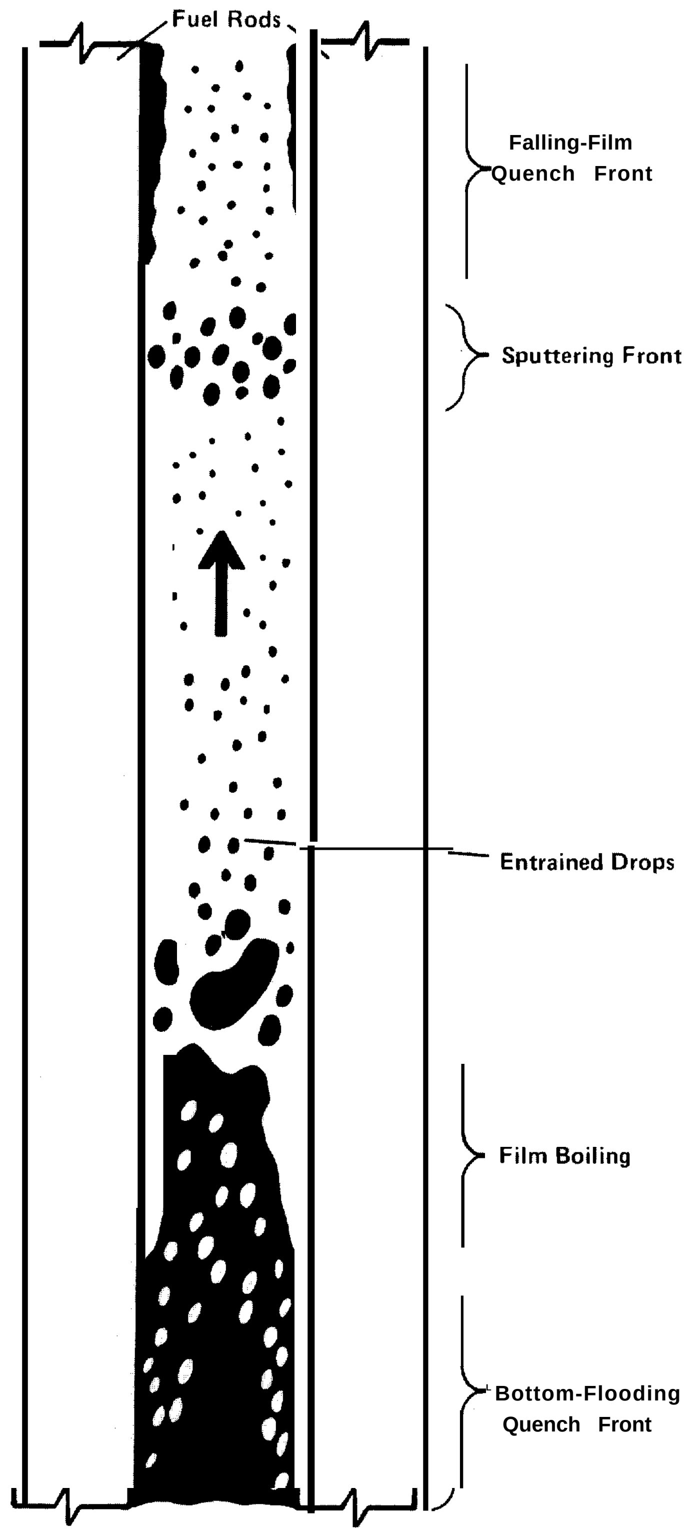


Fig. 3. During reflood, fuel rods are quenched from the bottom by water rising through the core and from the top by liquid films falling through the upper core-support plate.

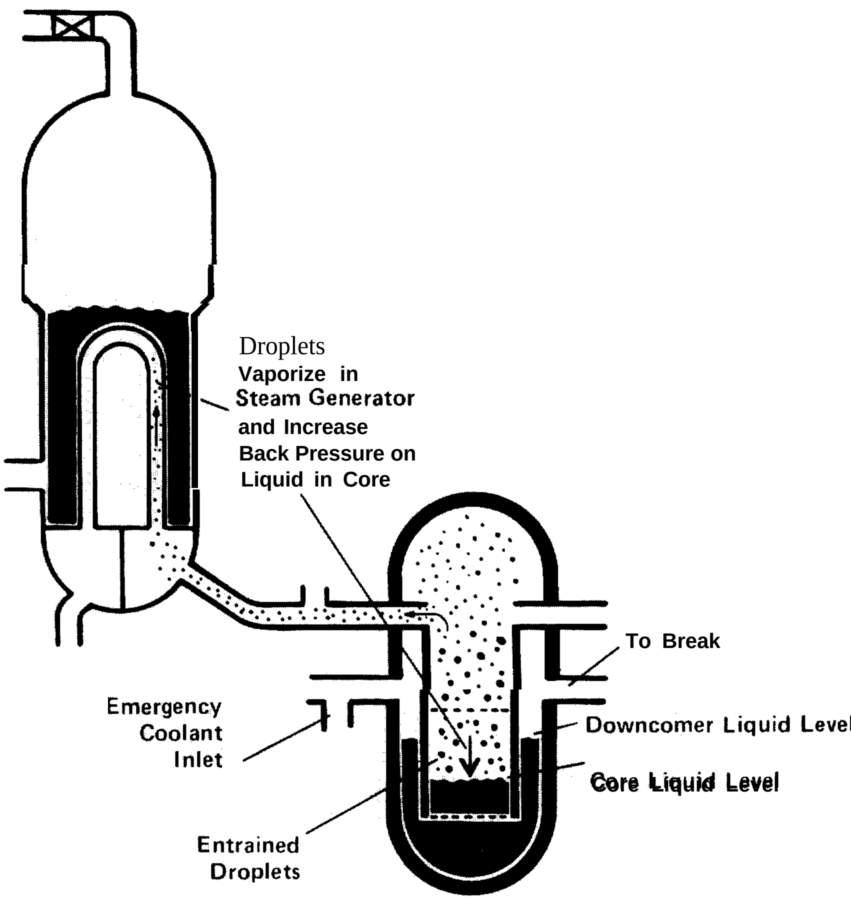


Fig.** 4, **Steam binding during reflood. The pressure created by vaporization of entrained droplets in the steam generator opposes the flow of emergency coolant to the core.

bypass/refill. Consequently, when water first covers the bottom of the fuel rods, it is unable to wet the cladding surface because heat transfer is predominantly by film boiling. Eventually, the cladding temperature falls below the minimum stable film-boiling temperature, the liquid wets the surface, the fuel rods cool by the efficient mechanism of nucleate boiling, and their temperature at that elevation drops sharply to near the water temperature, that is, the rods are quenched. Quenching progresses from bottom to top as the core is reflooded but, as explained below, some top-down quenching also occurs at the same time (Fig. 3).

The quenching process releases a large amount of heat to the reflooding water and causes steam to form. The steam carries water droplets upward as it rises between the fuel rods; these entrained droplets help to cool the rods at higher elevations. (This effect, as well as axial heat conduction in the rods from unquenched to quenched regions, is called precursory cooling.)

The entrained droplets are responsible for top-down quenching. As they rise through the upper plenum, they de-entrain, or deposit, on various structures and form a pool on the upper core-support plate. At first, water from the pool cannot flow down to quench the rods because steam is moving upward through the holes in the upper core-support plate. This phenomenon is similar to that occurring in the downcomer during bypass. At some point, however, water films penetrate the holes and begin to quench the fuel rods from the top. Top-down quenching by falling films takes place first at the core periphery where decay heat is lowest. Tracking of the quench fronts due to both bottom flooding and falling films was probably the most difficult technical problem we faced in modeling a large-break accident.

Reflooding, and hence quenching, can be retarded by the phenomenon known as steam binding (Fig. 4). The driving force for reflooding the core is the difference between the water levels in the downcomer and the core. This force can be counterbalanced by an increase in core pressure produced as entrained water droplets are carried to the steam generator and vaporized by reverse heat transfer.

As described later, separate-effects tests for the reflood phase indicate that the TRAC droplet-entrainment model may need improvement, but, in general, the treatment of the steam-water dynamics during reflood is in agreement with experiment.

Code Design and Computational Models

TRAC was to be a benchmark systems code for large-break accidents, but its flexible design makes it suitable for studying many types of transients. For example, TRAC-PD2 has been used successfully to analyze the first few hours of the Three Mile Island accident, small-break loss-of-coolant transients in the LOFT facility, and loss-of-feedwater scenarios in full-scale pressurized-water reactors. The fast-running version (TRAC-PF1) currently under development at Los Alamos is designed to address these transients more efficiently and accurately.

TRAC owes this enormous flexibility to its completely modular design. By joining the modules (subprograms) in a meaningful way, the user can simulate a wide range of phenomena, from a simple blowdown to a multiple-failure transient. The user need supply only the problem geometry and the boundary conditions.

Figure 5 shows the structure of TRAC. including component and functional subprograms. To specify the problem geometry, the user instructs the code to join component subprograms that correspond to specific reactor components. TRAC includes component subprograms sufficient to model primary loops in their entirety and secondary loops except for the turbine-generator and condenser, which can only be approximated. Also available are subprograms to model boundary conditions at breaks and fills.

Each component subprogram automatically accesses functional subprograms that compute the important physical processes occurring within the com-

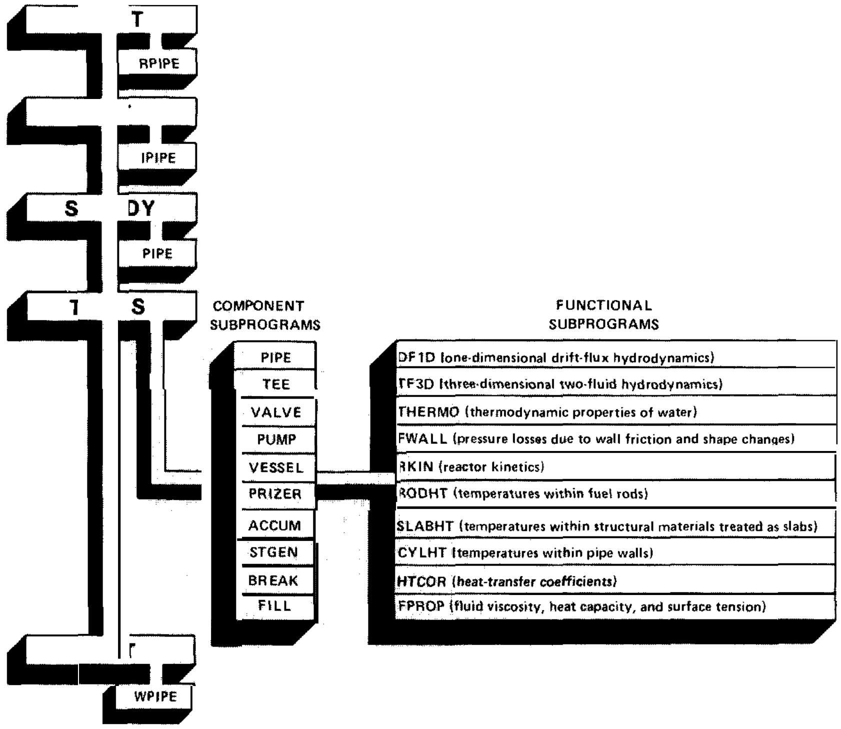


Fig. 5. TRAC is divided into five main subprograms, each of which handles a major aspect of the problem. INPUT accepts the user's description of the problem, INIT calculates quantities required for analysis that need not be supplied as input, STEADY calculates pretransient, or steady-state, conditions of the reactor, TRANS calculates the response of the reactor to the transient, and EDIT provides output. Within each of these main subprograms are subprograms that deal with particular reactor components. For all but TRANS, only the pipe component subprogram is shown; for TRANS, all the component and some important functional subprograms are listed. Each component subprogram accesses appropriate functional subprograms for relevant calculations.

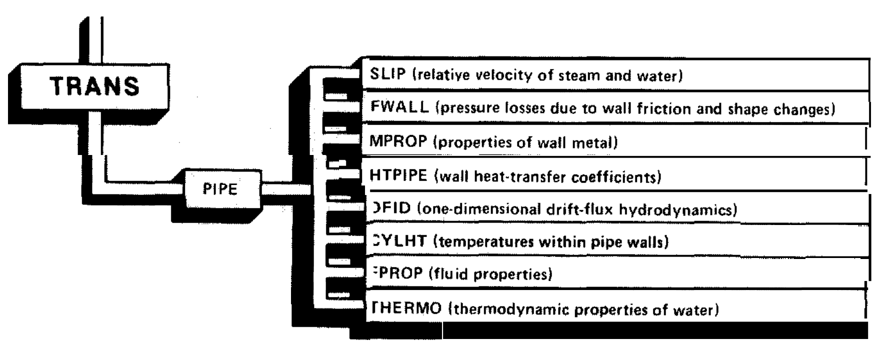


Fig. 6. Main functional subprograms accessed by PIPE to calculate *the fluid dynamics and heat transfer within a pipe.*

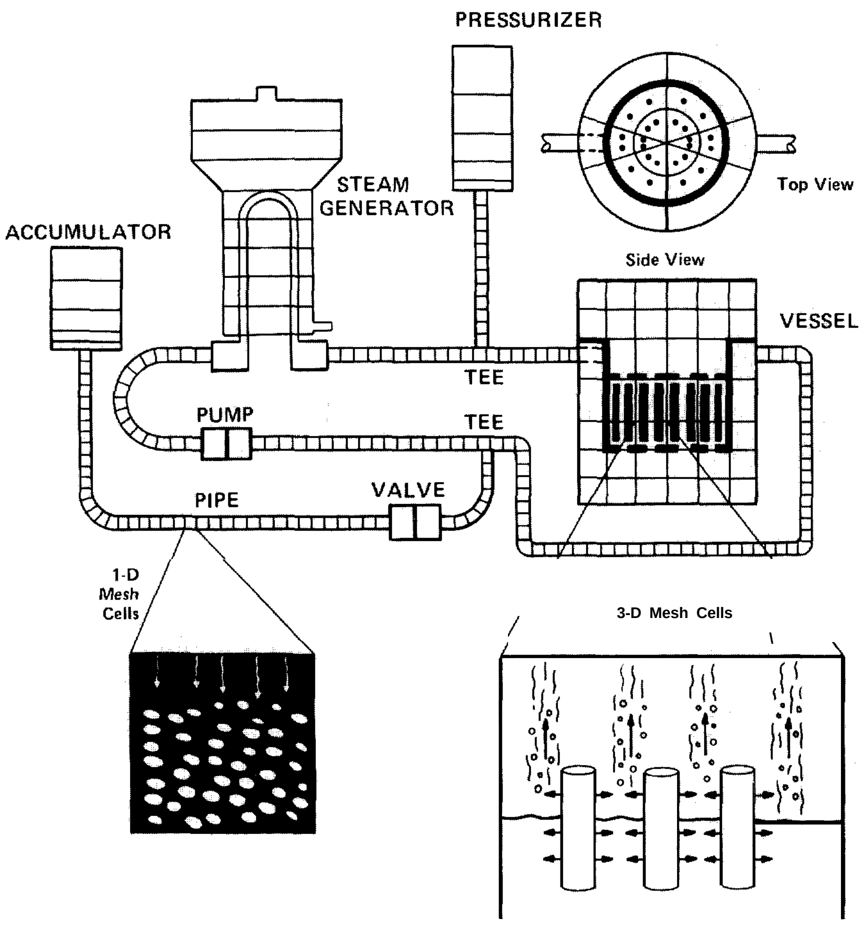


Fig. 7. Typical computational mesh for a vessel and a single coolant loop. In vessel cells, TRAC computes the nuclear heat and its transfer among fuel rods, flowing steam and water, and structural materials. In pipe cells, TRAC computes steam- water flow conditions and heat transfer between the two phases and pipe walls. Other reactor components are treated as variations on a pipe: a pump as a pipe with a momentum source; a valve as a pipe with a variable flow area; a pressurizer as a vertical pipe closed at one end with a heater/sprayer and a sharp steam-water interface; a steam generator as a pipe within another pipe; and an accumulator as a vertical pipe closed at one end with a sharp interlace between water and pressurized nitrogen.

ponent: steam-water fluid dynamics, heat transfer, and, in the vessel component, neutronics, or nuclear heat generation. For example, all component subprograms except that for the vessel access the same functional subprogram (DFID) to solve the one-dimensional fluid-dynamics equations. A pipe subprogram calls on other functional subprograms to obtain additional information required for solution of these equations, such as relative velocity of the two phases and heat-transfer coefficients between pipe walls and vapor or liquid (Fig. 6).

The reactor vessel and its internal structures (downcomer, core, upper and lower plena, and so on) are represented in three- or two-dimensional geometry at the user's choice. Components outside the vessel are represented in one-dimensional geometry. Figure 7 shows a vessel and a single coolant loop assembled into computational cells with TRAC component subprograms.

TWO-PHASE FLUID DYNAMICS. The TRAC approach to modeling the steam-water dynamics is described in the preceding article. A two-fluid model based on conservation of mass, momentum, and energy for the liquid and vapor permits treatment of nonhomogeneous and nonequilibrium two-phase flow. That is, the liquid and vapor phases can move with different velocities and can have different temperatures, a situation that occurs during emergency coolant injection when superheated vapor and subcooled water flow in opposite directions. Other less-advanced codes require that the two phases have the same velocity or that one phase be at the saturation temperature.

For lack of a real theory, the constitutive relations are approached empirically. These relations describe the exchange of mass. energy, and momentum between steam and water and between solid structures and steam-water coolant. The exchange rates depend on information not available from the two-fluid equations, namely, the flow regime in effect. Figure 8 shows the important flow regimes for upward flow through a vertical array of fuel rods. TRAC in-

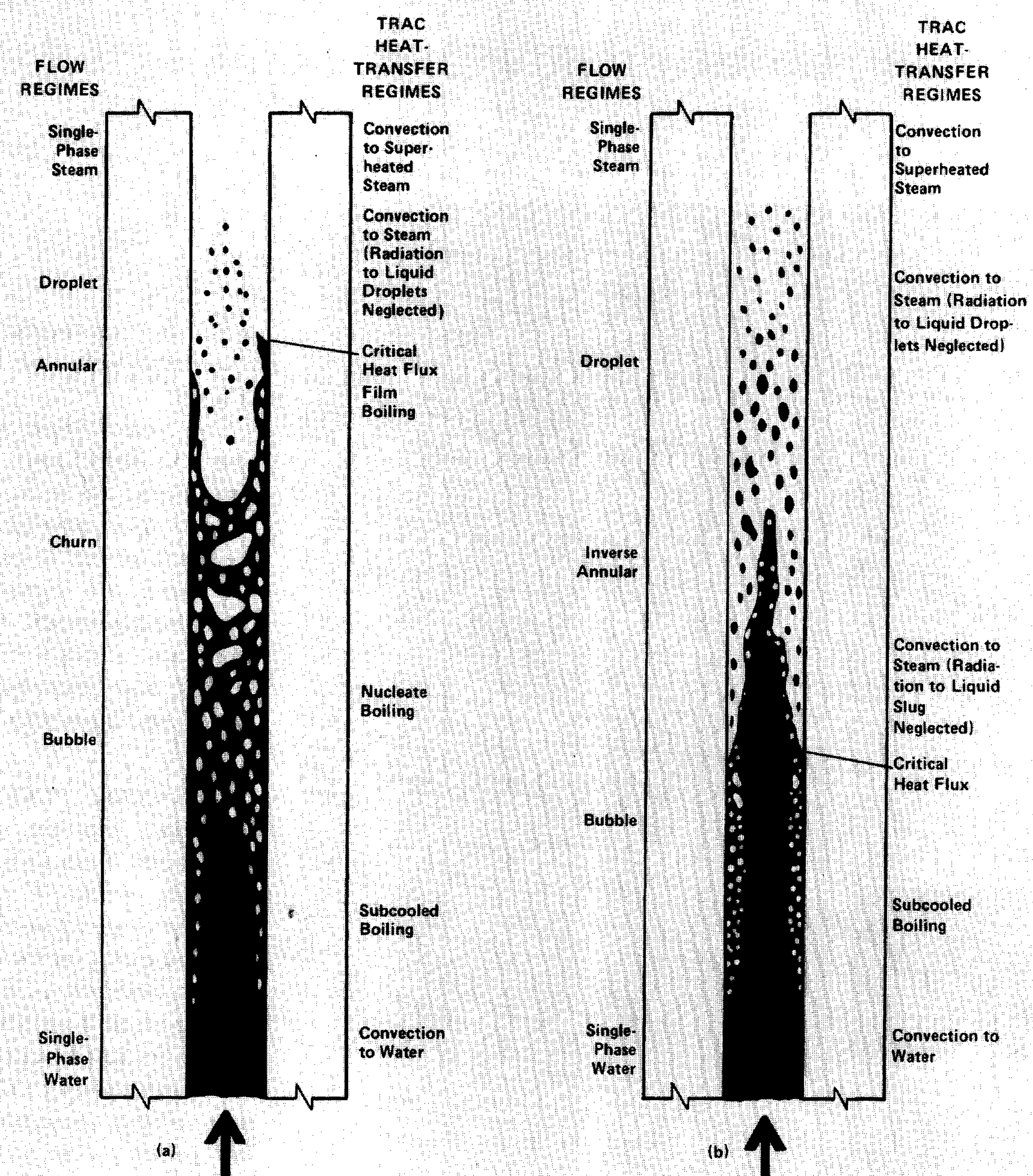


Fig. 8. Flow regimes and associated heat-transfer regimes for upward flow through a vertical array of fuel rods as (a) low and (b) high heat fluxes.

cludes an empirical flow-regime map that correlates calculated values of the vapor fraction and the mass flux with particular flow regimes. Once the flow regime is determined, TRAC computes the exchange rates from empirical algorithms. This method of handling the constitutive relations yields acceptable results in agreement with a wide variety of experiments, but further improvement of TRAC is expected mainly from increased knowledge in this area.

HEAT TRANSFER. The mechanism for transferring heat between coolant and structural materials or fuel rods also depends on the flow regime. Figure 8 also displays the heat-transfer regimes associated with each flow regime. TRAC includes models for the following heat-transfer mechanisms: convection to single-phase liquid, nucleate boiling, transition boiling, film boiling, convection to single-phase vapor, condensation, and liquid natural convection.

Temperatures of fuel rods and structural materials are calculated with heat-conduction models: a one-dimensional model for pipes; a one-dimensional lumped-parameter slab model for reactor vessel structures, such as downcomer walls and core-support plates; and a two-dimensional model for fuel rods.

The fuel-rod heat-conduction model simulates the effects of internal heat generation, quenching phenomena, zirconium-steam reactions, and changes in the size of the gap between cladding and fuel. The conduction model subprogram automatically divides the fuel rods into smaller cells during reflood calculations to provide finer detail for this phase of a transient. To track the quench front, the subprogram also uses dynamic indicators to rezone the rods into a super-fine mesh that can resolve the large axial temperature gradient at the front.

NUCLEAR POWER GENERATION. During a transient, power generation in the core changes with time. TRAC models these changes with two methods. One is simply the use of a power-versus-time table supplied as input by the user. The other is solution of the point-reactor kinetics equations that describe core power as a function of time, with total reactivity as the controlling parameter. Reactivity-feedback effects due to changes in core-average fuel temperature, coolant temperature, and coolant density are taken into account. Power from fission and fission-product decay is calculated with 6 delayed-neutron groups and 11 decay-heat groups.

The spatial distribution of power in the core and within the fuel rods is specified as input and remains fixed during the transient. This approximation is adequate for all loss-of-coolant transients because fission is halted by voiding of the core or scramming the reactor. However, for analysis of transients without scram, reactivity-insertion accidents, and some operational transients, changes in the spatial power distribution may be important and would require the use of a space- and time-dependent power generation model.

COMPUTATIONAL TECHNIQUES. The field and constitutive equations are solved by efficient spatial finite-difference techniques. Normally, a semi-implicit time-differencing technique is used for all calculations. This technique is subject to the Courant stability limitation that restricts the time-step size in regions of high-speed flow (for example, in a broken leg). Therefore, a fully implicit time-differencing option is also available for solution of the one-dimensional flow equations; this option permits tine spatial resolution in regions of high-speed flow without restricting the time-step size.

To improve convergence, the solution strategy for the vessel includes these techniques: direct matrix inversion (rather than iteration) for vessels with less than 80 cells; coarse-mesh rebalance for vessels with more than 80 cells; relinearization of the vessel equations to correct the assignment of a donor cell when the fluid velocity changes sign during a time step; and a time-step backup procedure when invalid temperatures, pressures, or void fractions are encountered.

A stability-enhancing two-step numerical method included in TRAC-PF1 removes the Courant time-step limitation and permits analysis of transients of long duration at real time or better. To further enhance stability, wall heat transfer is treated more implicitly in this version.

OUTPUT. TRAC produces an extraordinary amount of information during the course of a calculation. At each step and for each mesh cell, TRAC provides values for the following variables: fluid pressure, void fraction, temperatures and velocities of the two coolant phases (for vessel cells, the velocities are vector quantities), and temperatures of solid materials, such as the cladding. Other variables (for example, mass and momentum fluxes and fluid density) can be obtained from these basic variables. A versatile graphics package is available to help the user digest this information by producing movies and a wide variety of plots.

To determine the validity of TRAC results, they must be compared with experiment, but, unfortunately, velocities and temperatures of the two coolant phases cannot be measured accurately. Variables that can be measured directly and accurately include fluid pressure, mixture temperature, and metal temperatures. Indirect and less accurate measurements can be made of void fraction and steam-water mixture velocities. The number and location of variables measured are necessarily much smaller than those calculated; furthermore, in some cases, the measurement device can significantly perturb the variable being measured.

How Good is TRAC?

The end objective for TRAC is to provide a credible predictive tool for all light-water reactor transients. But can we rely on TRAC predictions of events that have never been measured in full-scale reactors? We believe the answer is yes. The code has been tested against many different experiments that span a wide range of scales, reactor components, and geometric arrangements and involve most of the important thermal-hydraulic phenomena expected in a full-scale power plant under normal and accident conditions.

The constitutive relations in TRAC are based on so-called model development experiments. These are usually small-

TABLE I

FACILITIES FOR TRAC ASSESSMENT

Facility	Operating Institution and Location	Scale	Phenomena and Phase of Accident Studied	Description[a]
Semiscale Mod-1	Idaho National Engineering Laboratory United States	Small	System effects during all accident phases	One active and one passive loop
Semiscale Mod-3	Idaho National Engineering Laboratory United States	Small	System effects during all accident phases	Full-height core, two active loops, and upper-head-injection capability
LOBI	Commission of the European Communities, Ispra Establishment Italy	Small	System effects during blowdown and bypass/refill	Two active loops and full-height core
FLECHT	Westinghouse Electric Corporation United States	Small	Separate effects during reflood	Single-bundle full-height core
FLECHT-SEASET	Westinghouse Electric Corporation United States	Small	Separate and system effects during reflood	Single-bundle full-height core and one coolant loop
THTF	Oak Ridge National Laboratory United States	Small	Heat transfer during blowdown	Single-bundle full-height core
Pipe Blowdown Tests	Centro Informazoni Studi Esperienze Italy Atomic Weapons Research Establishment United Kingdom	Small	Separate effects during blowdown	Pipe-wall-heating capability
Tube CHF Tests	Atomic Weapons Research Establishment United Kingdom	Small	Steady-state pipe wall heat transfer over entire range of boiling curve	Pipe-wall-heating capability
LOFT	Idaho National Engineering Laboratory United States	Intermediate	System effects during all accident phases	Nuclear core, one active and one passive loop
PKL	Gesellschaft fur Reaktorsicherheit m.b.H. West Germany	Intermediate	Separate effects during bypass/refill and reflood	340-rod full-height core and three coolant loops
CCTF	Japan Atomic Energy Research Institute Japan	Intermediate	Separate effects during bypass/refill and reflood	2000-rod full-height cylindrical core
Downcomer Tests	Creare, Inc. United States Battelle Columbus Laboratories United States	Intermediate	Separate effects during bypass/refill	Downcomer and lower plenum with external steam source
Marviken 111	Studsvik Energiteknik AB Sweden	Large	Critical flow during blowdown	Full-scale vessel
SCTF	Japan Atomic Energy Research Institute Japan	Large	Separate effects during bypass/refill and reflood	Full-scale (axial and radial) slab core
UPTF[b]	Gesellschaft fur Reaktorsicherheit m.b.H West Germany	Large	Separate effects during bypass/refill and reflood	Full-scale downcomer and upper plenum with internals

[a]Unless otherwise noted, nuclear processes are simulated by electric heating,
[b]Construction will begin soon on this facility; TRAC has been used for design analysis.

TABLE H

ASSESSMENT PHENOMENA IN PRESSURIZED-WATER REACTOR COMPONENTS

Component	Assessment Phenomena
Core	Conductive and convective heat transfer, dryout and rewetting, entrainment and de-entrainment, quench-front propagation, flow topology, multidimensional effects.
Upper plenum and head	Entrainment, de-entrainment, and re-entrainment, pool formation and flooding, emergency coolant injection, liquid inventory, multidimensional effects
Lower plenum	Voiding, sweepout, refill, heat transfer by mixing, condensation
Downcomer	Liquid bypass, penetration, and refill, condensation, wall heat transfer, multidimensional effects
Steam generator	Heat transfer, steam binding, pressure drop
Pump	Head and torque, friction, two-phase degradation
Pressurizer and accumulator	Depletion rate
Piping	Flow topology, wall heat transfer and friction, flow rate, condensation, critical flow

scale laboratory experiments that explore the basic physical processes associated with two-phase thermal hydraulics: bubble growth, vapor nucleation, interphase transfer of mass, momentum, and energy, flow regime variation, and so on. Such experiments are being performed at numerous institutions, including national laboratories, universities, and industrial research laboratories. Application of such information to full-scale reactors is yet incomplete.

Testing of TRAC itself is done by comparison with two basic types of experiments: separate-effects experiments designed to study a single phase of a loss-of-coolant accident or the response of a single reactor component and integral experiments that involve all the major components of the primary system during more than one phase of the transient. Some of the experimental facilities used to test TRAC are described in Table I. Table H lists the important phenomena associated with pressurized-water reactor components that are studied experimentally and then compared with TRAC predictions. The comparisons lead to new experiments and improved versions of the code.

TRAC-PD2, the latest version to be released to the reactor community, was tested against separate-effects and integral tests covering a wide range of scales and was found to do a credible job overall. To illustrate the code's performance at the time of release, we present results from a separate-effects test for the reflood phase, the most difficult phase of an accident to simulate.

REFLOOD TEST. FLECHT, the full-length emergency-cooling heat-transfer facility, was designed to study heat transfer, quench-front propagation, and droplet entrainment and de-entrainment during the reflood phase of a loss-of-coolant accident. FLECHT consists of a single fuel bundle containing approximately 100 full-length fuel rods mounted in a flow housing with upper and lower plenum regions (Fig. 9). The bundle and housing are electrically heated until the bundle is covered with saturated steam but the lower plenum is full of water. Reflood is initiated by injecting water into the lower plenum when the desired maximum rod temperature is reached. Electric heating is decreased during reflood to simulate decay heat, Figure 10 compares TRAC predictions and experimental values for the quench-front location as a function of time. (The quench front is the point at which the fuel-rod temperature has dropped rapidly to near that of the reflooding water.) Note that complete quenching occurred earlier than predicted by TRAC. This discrepancy is attributed to radiant heat transfer from the heated rods to the housing and to unheated rods, an effect not included in TRAC because it is unimportant for a full-scale pressurized-water reactor.

The mass of fluid exiting from the upper plenum region was also measured and is compared with calculated values in Fig. 11. The good agreement appears to indicate an acceptable entrainment model in TRAC. However, there is some evidence from these and other experiments that more de-entrainment in the upper plenum is needed to improve the calculated results for the top-down quench front.

INTEGRAL TESTS OF SMALL-BREAK ACCIDENTS. Following the release of TRAC-PD2, we have continued to test the code against integral experiments that involve all major components

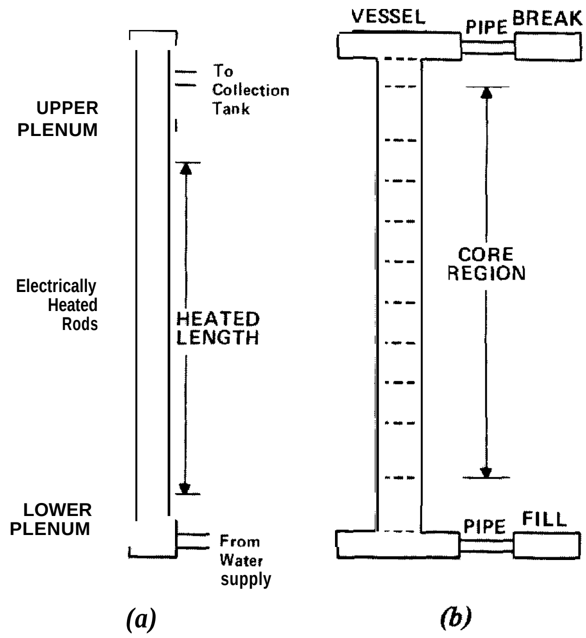


Fig. 9. Schematic diagram of (a) FLECHT and (b) its division into computational cells in the TRAC model. FLECHT'S simulation of a core consists of a single bundle of electrically heated, full-length rods in a 10 by 10 array. Because multidimensional effects were not the focus of the experiment, the vessel was treated as a slab (an option available in TRAC) and the two-fluid equations were formulated and solved in one dimension, along the vessel axis.

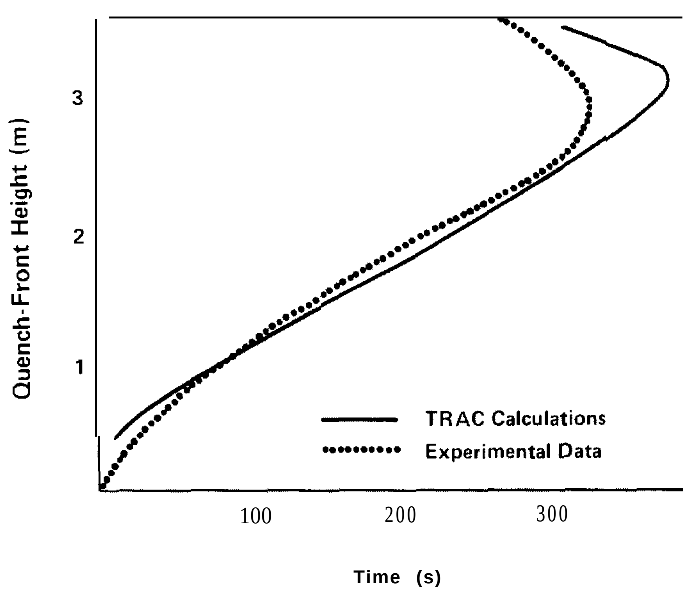


Fig. 10. Quench-front propagation during a reflood test at FLECHT.

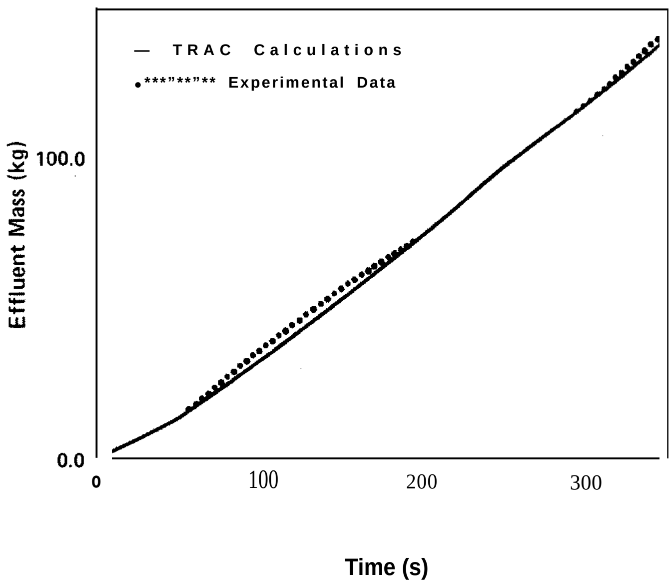


Fig. 11. Fluid mass exiting from the vessel during a FLECHT *reflood test.*

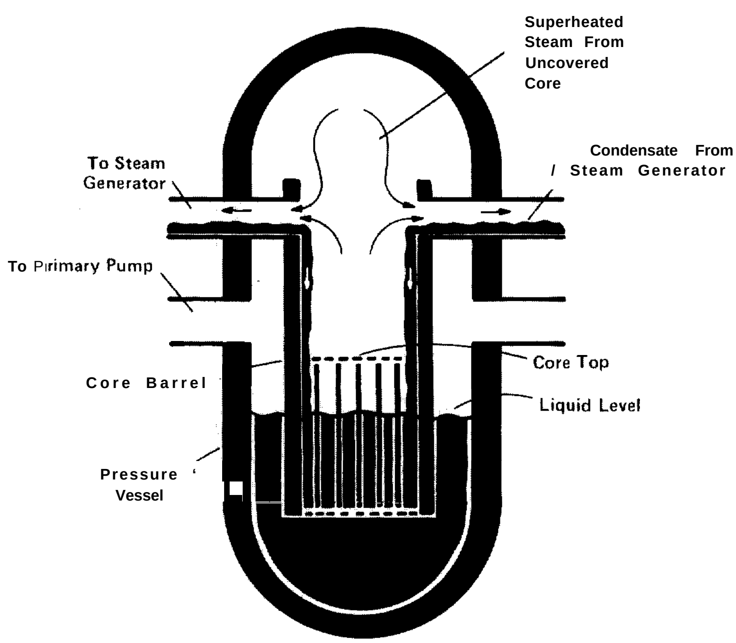


Fig. 12. Reflux cooling plays a role in cooling the core if the primary pumps are turned off and the core is partially uncovered. Superheated steam from the core condenses in the steam generator and flows back to the core along the hot legs.

of the primary system during more than one phase of a transient. These experiments involve small-break and operational transients at Semiscale and LOFT, the loss-of-fluid test facility.

One focus of these studies is an issue that arose because of the Three Mile Island accident—the pumps on-pumps off issue. Is it better to turn off the primary pumps immediately after a small break or to leave them running?

Although leaving the pumps on may provide better cooling initially, this advantage may be outweighed in the long run by the greater loss of coolant.

But what mechanisms are available for cooling the core with the pumps off? If the core remains covered with water, natural circulation, or gravity-driven liquid convection, can provide sufficient cooling to remove the decay heat through the steam generators. And if the core becomes partially uncovered, reflux cooling (Fig. 12) comes into play. Superheated steam produced in the voided region of the core flows through the hot legs to the steam generators. There it condenses, and the water flows back along the hot legs to the vessel in a countercurrent stratified flow.

Two tests were performed at LOFT (Fig. 13) to investigate the effect of primary pump operation on the system's response to a small break in a cold leg. During one test, the coolant pumps were tripped immediately after initiation of blowdown; during the other, the pumps were left on until the primary system pressure fell from an initial pressure of 150 bars to 21.5 bars.

With the pumps off, the core remained covered during the entire test. Figures 14 and 15 compare TRAC-PD2 predictions and measured values of primary system

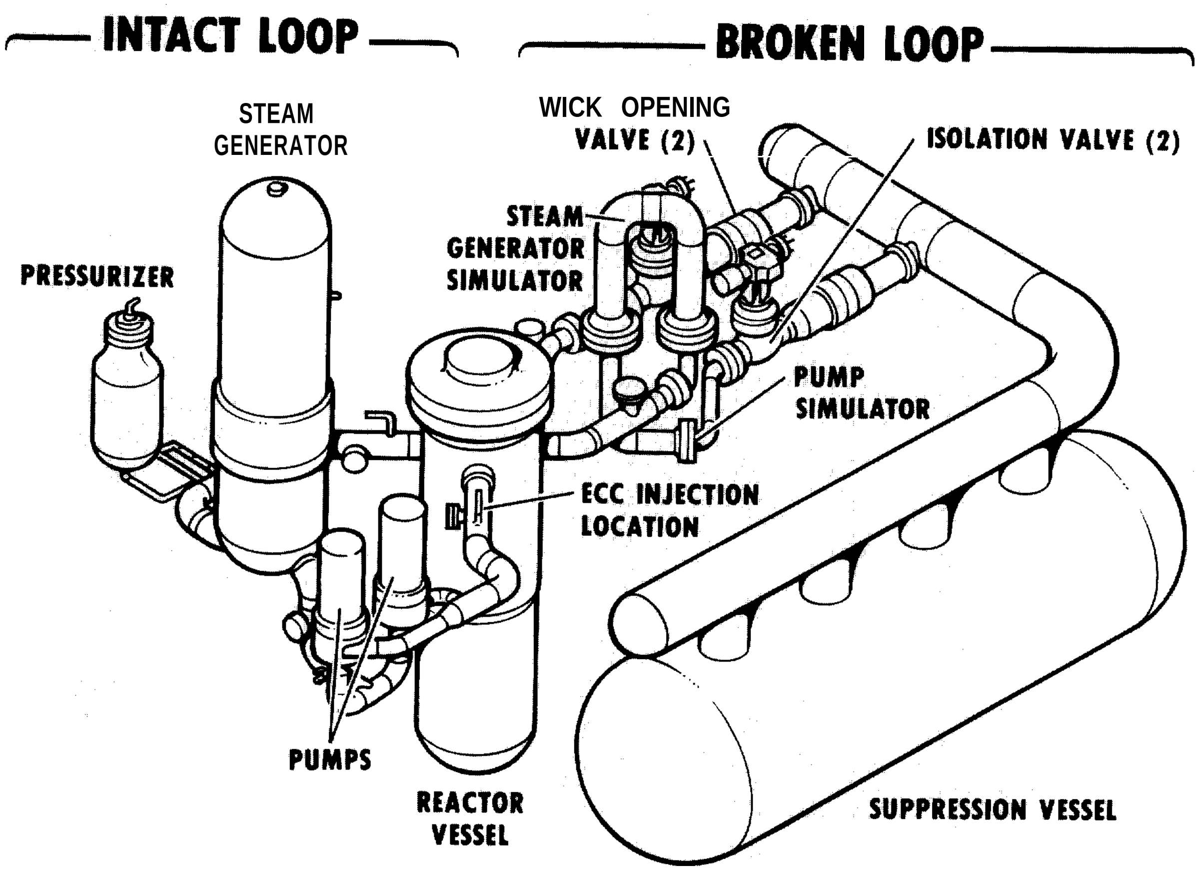


Fig. 13. Major components of LOFT, an intermediate-scale facility for integral loss-of-coolant tests. Volume, power, and flow and break areas are scaled at 1 to 60. LOFT is unusual in that it contains a real nuclear core rather than electric heaters. Breaks are simulated by the quick-opening valves. The suppression vessel collects the lost coolant and controls the back pressure on the vessel.

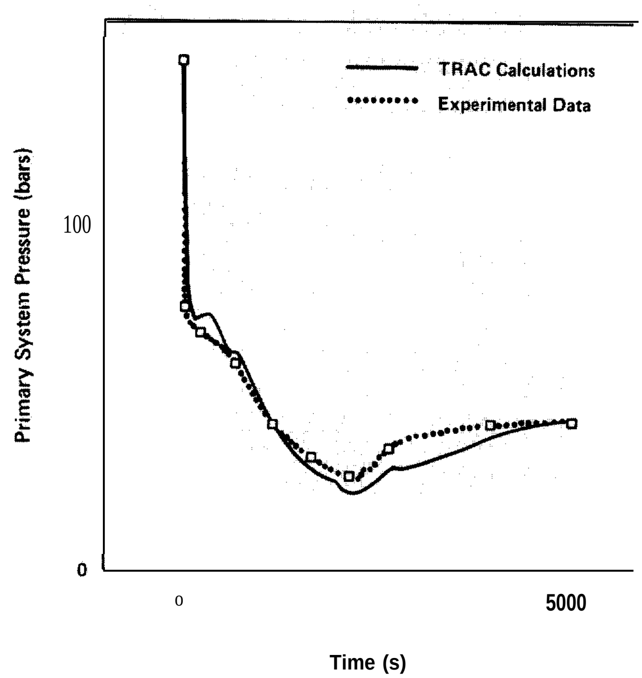


Fig. 14. Primary system pressures during a simulated small-break loss-of-coolant accident at LOFT with the primary pump turned off immediately.

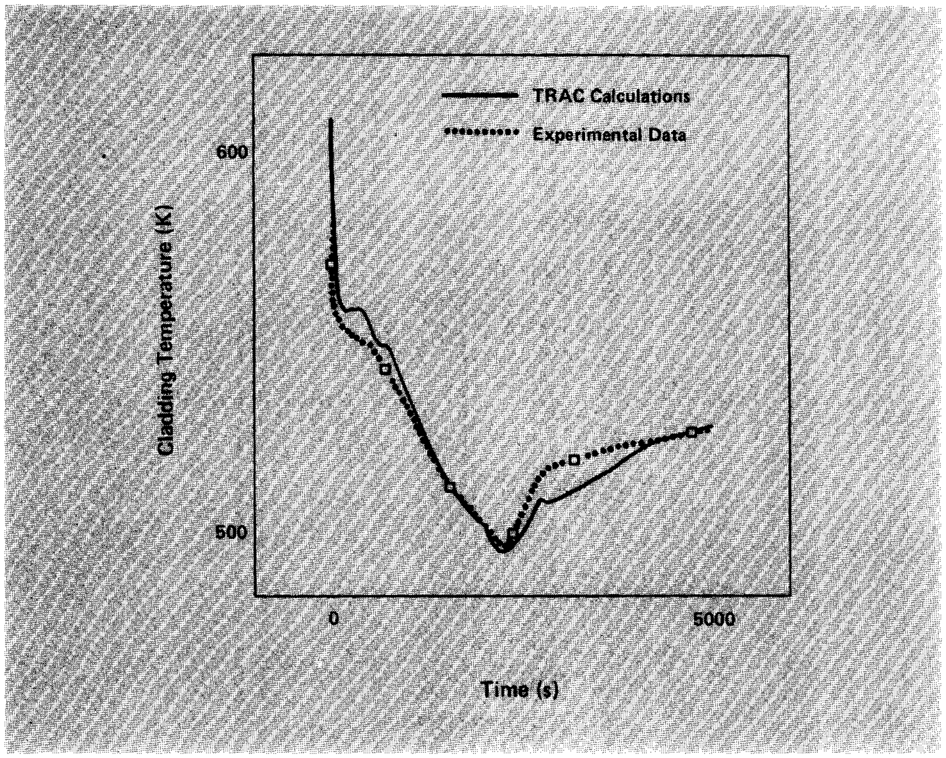


Fig. 15. Cladding temperatures during a simulated small-break loss-of-coolant accident at LOFT with the primary pump turned off immediately.

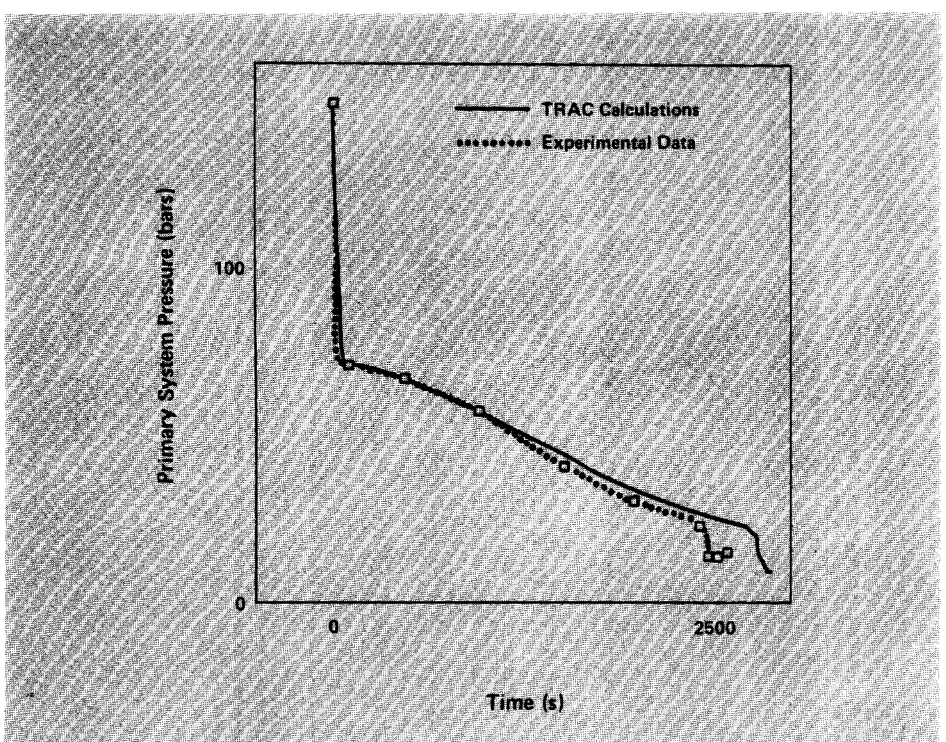


Fig. 16. Primary system pressures during a simulated small-break loss-of-coolant accident at LOFT with the primary pumps operating until about 2400 seconds.

pressure and cladding temperature. The initial rapid pressure decrease (Fig. 14) corresponds to the subcooled portion of blowdown; boiling and flashing account for the slower decrease later. At approximately 2300 seconds, the break is isolated (closed), and the pressure begins to increase and stabilizes at about 5000 seconds. At this point, the heat removed by natural circulation balances the decay heat. The cladding temperature (Fig. 15) follows the saturation temperature of the fluid and stabilizes at about 90 kelvin below the initial temperature.

During the pumps-on test, the core is cooled satisfactorily by the two-phase mixture until the pumps are turned off at about 2400 seconds. Note that during this period, the pressure and cladding temperature histories (Figs. 16 and 17) are very similar to those for the pumps-off test. The mass flow out the break (Fig. 18) is, of course, greater with the pumps on. When the pumps are tripped, steam and water separate and the upper portion of the core is uncovered. This results in a rapid rise in cladding temperature. (A similar situation occurred during the Three Mile Island accident when the primary pumps were turned off by the operators). When the cladding temperature reached 590 kelvin, the test was terminated by injecting emergency coolant from the accumulator. Because TRAC slightly underpredicted the rate of primary system pressure decrease, it also predicted that the pump trip and resulting temperature excursion occurred later (see Fig. 17). Otherwise, the calculated and measured histories are in excellent agreement.

These studies are continuing and the new faster-running version of TRAC (TRAC-PF1) should be able to simulate these long (several hours) transients more accurately and economically. It will include models of stratified countercurrent flow and feedback controls, improved models of flow at a break, and a more detailed representation of fluid flow and heat transfer in the steam generator. These phenomena play a larger role in small-break accidents than in large-break accidents.

A new-generation reactor analysis code is also under development at Los Alamos. This code will address severe accidents for which core melting and relocation of core materials must be taken into account. TRAC'S ability to treat the entire primary system and the ability of SIMMER* to treat core meltdown will be used extensively in this new effort.

Conclusion

In summary, results thus far indicate that the basic modeling and numerical framework in TRAC are fundamentally sound. Model improvements have been identified and will be incorporated into the next code version. Current applications of TRAC include its use to analyze transients in full-scale pressurized-water reactors as part of a multinational research program on refill and reflood in large-scale facilities. We are applying it to studies of multiple-failure accidents in an attempt to identify accident signatures and operator actions for accident mitigation.** We are also using TRAC to resolve safety issues and licensing questions of interest to the Nuclear Regulatory Commission and to evaluate reactor design changes. The code has only recently reached maturity and we expect it to have a major impact in all these areas in the coming years. ■

*SIMMER is a computer program for fast-reactor analysis developed by the Laboratory. See "Breeder Reactor Safety-Modeling the Impossible" in this issue.
**See "TMI and Multiple-Failure Accidents" in this issue.

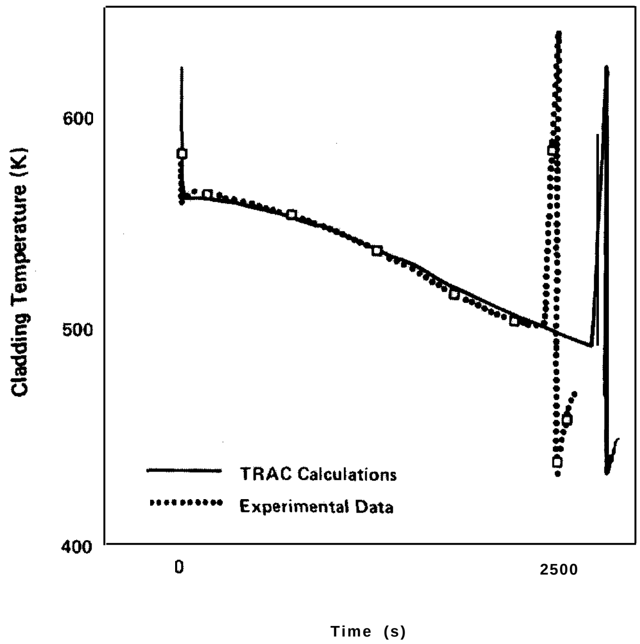


Fig. 17. Cladding temperatures during a simulated small-break loss-of-coolant accident at LOFT with the primary pumps operating until about 2400 seconds.

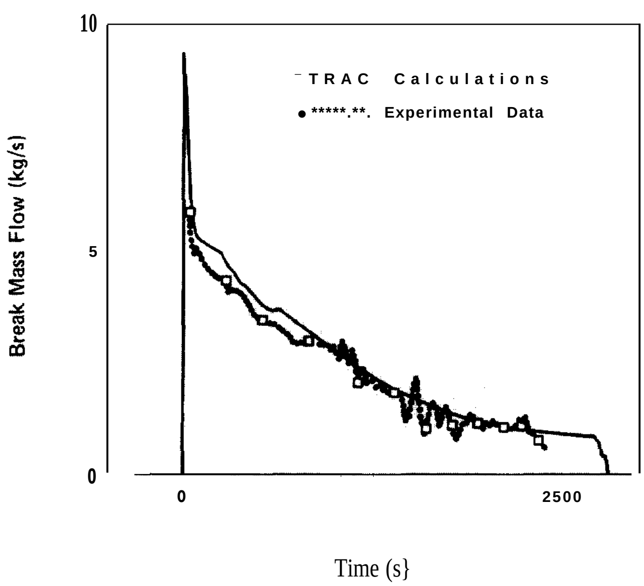


Fig. 18. Fluid mass exiting from a simulated small break in a cold leg of LOFT with the primary pumps operating until about 2400 seconds.

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Further Reading

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