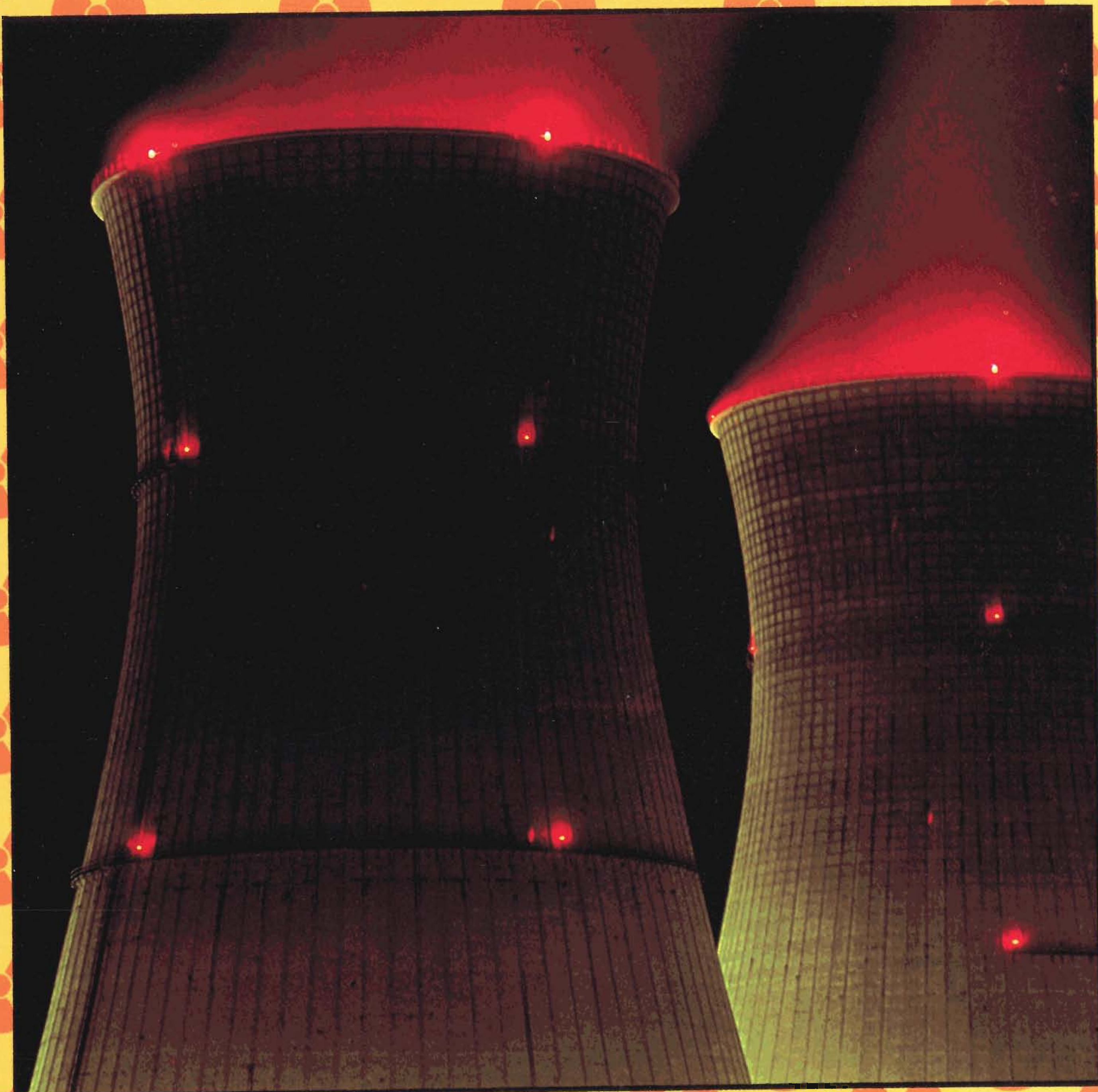
VOLUME 2, NUMBER 2

REACTOR SAFETY

Los Alamos Science

LOS ALAMOS NATIONAL LABORATORY



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Los Alamos Science

LOS ALAMOS NATIONAL LABORATORY

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dialogue

COMMENTS ON REACTOR SAFETY FROM LEADERS OF THE MANHATTAN PROJECT

In 1979, Los Alamos National Laboratory began interviewing scientists who were part of Project Y, the Los Alamos effort to produce the first atomic weapons. The interviews form the basis of the Laboratory's Historical Perspective Film Series. Excerpted here are comments from eight interviews, all apropos of the subject of this issue—reactor safety and the accident at Three Mile Island. The producer and interviewer for the films is the Laboratory's Mario Balibrera.

"No accident is beneficial. Three Mile Island was a tragedy. I think the report of the Kemeny Commission has been a very important contribution in trying to run down the real reasons for it. . .The equipment operated better than the people. If they'd left the plant alone, it appears that the accident probably never would have happened. . .No one was hurt and no one got an overdose of radiation, that is true, but it still was a tragedy in the sense that the people there had great psychological difficulties. For the next 10 or 20 years we are going to depend on nuclear energy in very considerable measure. . .I think we'll need the breeder, but not until after the turn of the century. . ."

Robert F. Bacher, Emeritus Professor of Physics
California Institute of Technology
Bacher directed the Experimental Physics Division and, later, the Bomb Physics Division of Project Y.

"Nuclear power is a necessity for all industrial countries. Safety, of course, is tremendously important, and we have to learn from our mistakes. In fact, I don't believe that one can improve safety without having some minor accidents because they will tell you what's wrong, what has to be improved. The Three Mile Island accident was very unfortunate, but we can learn a great deal from it. . .Operators have to be much better trained. . .The controls have to be changed. . .Edward Teller has some other very important suggestions that I think should be incorporated. . .As for the waste, there are many ways to dispose of it, and I am happy that in the last few years, this has received much more attention than it did before. . .Sweden has, in fact, adopted a very reasonable plan. . ."

Hans A. Bethe, John Wendell Anderson Professor of Physics
Cornell University
Nobel laureate Bethe was head of the Laboratory Theoretical Physics Division during the Manhattan Project.

"I'm not an activist, so in principle I'm not in favor of movements for anything. I think public opinion should be expressed through more knowledgeable channels than organized movements. On the other hand, I'm not a proponent of nuclear power. I think there are very real hazards in large-scale development of nuclear energy, the greatest being the matter of proliferation, which maybe we can't stop, anyhow, but I think it's so important that one should try to slow it down. I'm not antinuclear, but I'm not in favor of crash programs or rapid development. The hazards are real, and the thing has to be approached carefully. There are other hazards. . than proliferation. . .Anything man makes has failed sometimes. . .There's Murphy's Law, which hasn't been repealed yet, as far as I know. The other hazard is waste disposal; there doesn't seem to be a good solution to that yet, so I think there are several reasons why one should use some caution, much as we need power. . ."

Edwin M. McMillan, former Director
Lawrence Berkeley Laboratory
Nobel laureate McMillan was a prominent participant in planning and recruiting for the Los Alamos Project.

"I'm astonished that so much attention—to the point of conflict and mass demonstrations—has gone on around a few reactors when there are 30,000 bombs which nobody seems to want to talk about. ..I find that a curious disparity. . .Three Mile Island was sloppily done. . .I hope there are more severe and sensible licensing and operational procedures in the future. . .I think there will be. My own view is that (the reactors) should be operated on a Federal basis, like airports. . ."

Philip Morrison, Institute Professor
Massachusetts Institute of Technology
Morrison was a group leader in experimental physics at Los Alamos from 1944 to 1946.

"All I can say about the antinukes is that I'm flabbergasted because here you have a technology which has a marvelous (safety) record compared to any other technology, whether it's steam engines, railroads, or airplanes. . .It's the safest so far, but the public imagination has been caught by it. The remarkable thing is that in spite of the mistakes made at Three Mile Island, nothing happened, except that the company lost millions of dollars and there was a great show (that was) wonderful for television and the newspapers. My own feeling is that if people don't want nuclear energy, they don't have to have it. . .We can shut down the nuclear plants, and about 10 to 15 years from now, when we miss them, all will not be lost: we can buy plants from France, from Japan, Germany, England. . .It'll cost us more but look at the pleasure we'll have had in not having nuclear energy. . ."

I. I. Rabi, Emeritus University Professor
Columbia University
Nobel laureate Rabi sewed as a consultant to the Los Alamos portion of the Manhattan Project.

"One of the major problems of nuclear power is to get the public to understand the situation. My own idea is that we need some substitute source of power and that nuclear power is the only one we have at present that is accessible in a finite time and that one can do. I'm not enthusiastic about nuclear power, but on the other hand, I see absolutely absurd things—people willing to take risks that are a thousand times as big as the risks that nuclear power offers and they don't bat an eye about it; they are very happy about it. But if something has to do with radiation, then everything is unacceptable. . . Because you don't smell it, you don't see it, you don't taste it, so it is a bad thing. People should really be afraid and scared of atomic bombs, which are in the tens of thousands in the armament of the United States, Russia, and in sizeable numbers in many other countries. Now that is a really terrible danger for mankind, of major proportions, and to tell you the truth, to see people being afraid of a nuclear plant when they have ten thousand bombs around in all of these countries. . .It's a little strange. . ."

Emilio Segre, Emeritus Professor of Physics
University of California, Berkeley

Nobel laureate Segre' was a group leader at Los Alamos from 1943 to 1946 and was in charge of measuring the spontaneous fission of uranium and plutonium.

"Nuclear energy is not the whole answer to the energy question, but it's part of the answer if the developing world is to develop. Those who try to tell us that it is too dangerous don't know what they are talking about. They don't happen to know that the big regulated reactors have not cost a single human life. That's a better safety record than that of any other energy-producing industry. We need all of the energy sources if we don't want the Arabs to dominate our economy, and we don't want to be at the mercy of the Russians, when, as it easily may happen, the Russians gain influence and "Finlandize" the countries around the Persian Gulf. . ."

Edward Teller, Senior Research Fellow
Hoover Institution on War, Revolution and Peace

Teller has been and is engaged in advanced work on nuclear weapons and nuclear energy, including the critical period of the Manhattan Project.

"I'm happy that so many people are concerned (about the antinuclear movement). Right after the war I joined the group that set up the Association of Los Alamos Scientists, that then integrated with the Federation of American Scientists. Our first concern was to raise money and then get the attention of other people about this terrible nuclear threat. Now, although I'm pleased to see so many people so passionately interested, I'm a little disheartened at the level of our concern. They don't seem to know as much as I would like them to know and it seems to me that their criticism is, in many cases, hysterical and unthinking in terms of nuclear reactors and nuclear energy, which I believe we need. . .The dangers of that, compared to other dangers, seems to be emphasized out of all proportion. I suppose it was a good thing that Three Mile Island happened. . .It was magnified out of all proportion to what actually went on there, but it showed up psychological problems about the use of nuclear energy that are very real. . ."

Robert R. Wilson, former Director
Fermi National Accelerator Laboratory

Wilson was group leader in cyclotron reseach at the Laboratory in 1943.

ON PETER CARRUTHERS IN OUR LAST ISSUE

I just finished reading the interview of Peter Carruthers and I am all fired up. I was strongly moved by Mr. Carruthers' statements, for that most elementary of reasons—strong agreement. I have experienced a few of the many situations described by Mr. Carruthers, but I have not yet left the Cloistered Academic Halls. Like Carruthers, I see relatively few alternatives in Academia for a doer and a shaker, and the private turf syndrome appears to grow ever more compartmentalized year by year.

Nevertheless, I am still fighting the good fight against the institutionalized inertia, but as Mr. Carruthers has observed, the only significant satisfaction comes from the students. The only way I have ever achieved even a modicum of success in changing Academic Structure, involves that simplest of strategies—just do it. Propose it. Write it up. Make a motion. Prepare a new curriculum. Propose a new division. Most people, I believe, follow simple physical or metaphysical laws. They take the path of least resistance, they minimize their particular work function, they conserve energy. Consequently, by expending a little energy, by doing a little extra work, even Academia can be changed. I know. I have changed a little bit of it. The nagging question is—is it worth the effort? I dream of research and instead I write memos, or papers, or curriculums, or programs, or standards, or books. It is tine for the ego, except for the same nagging doubt. I usually just forget about that doubt, until I read something like Mr. Carruthers' interview, and then I start to think of focusing electron beams utilizing unstable interactions stabilized by adaptive control techniques. Or, I start to devise systems to locate stolen cars. Or, I just start to daydream. Please send me an application to Los Alamos. I'll put it in my future tile to await the next nagging doubt.

Prof. Richard Gray Costello, Chairman
Electrical Engineering Dept.
The Cooper Union
New York, NY

Your comments on articles appearing in Los Alamos Science are welcome. Please address c/o The Editor, Los Alamos Science, Los Alamos National Laboratory, Mail Stop 708, Los Alamos, NM 87545.

Inside This Issue

EDITOR'S NOTE

The risk of a reactor accident leading to a large radiation release has occupied the attention of safety analysts and the public for many years, It was also the risk that was raised during the accident at Three Mile Island. Several days after the accident began, the public was informed that a hydrogen bubble inside the reactor vessel might explode. The explosion never happened and the accident ended with no major radiation release, but the reactor itself is severely damaged and, as described in Boudreau's editorial, its recovery will be delayed for many years by political and economic obstacles.

The major damage and greatest danger occurred very early in the accident; however, this fact was not fully appreciated until later. At Los Alamos safety analysts watched a replay of the accident on a television screen, The replay was the product of a large and complex computer code that had taken several years to develop. It simulated the internal workings of the reactor system that had been hidden from the accident participants by numerous layers of concrete and steel. The severe situation during the early hours became clear.

During the actual accident, the operators overrode the automatic safety systems and for over three hours kept the flow of emergency cooling water to a minimum because they interpreted the abnormally high pressurizer water level shown by the control panel to mean that the reactor primary system was overfilled. Safety analysts saw something quite different on the computer-simulated replay. The pressurizer water level was high during this period because of steam in the system. Later, when the operators turned off the circulation pumps, the steam separated from the water and the water level in the reactor vessel fell below the top of the fuel rods. The uncovered rods began to overheat and soon became so hot that steam oxidized their protective cladding, This reaction produced large quantities of noncondensible hydrogen that impeded the flow of coolant for the remainder of the accident and eventually caused the concern about a hydrogen explosion. Meanwhile, at about three hours into the accident, the uncovered and damaged core was on its way to a meltdown. The meltdown never happened because some minutes later the operators turned up the flow of emergency cooling water and reflooded the core. It was a fairly close call.

In the calm of the aftermath, safety analysts were able to vary the accident scenario and watch what would have happened if the operators had allowed the automatic safety systems to operate as designed. The result would have been a minor accident that would never have made the five o'clock news.

The reactor designers and safety analysts were reassured by the final outcome, but the dramatic series of events clearly demonstrated that the human factor, largely ignored in safety analysis, can lead to more serious accidents than had been generally anticipated.

On the positive side, this fact has changed the direction of reactor safety research. Analysts will now be investigating a much broader range of problems, including the consequences of core meltdown and hydrogen production, fission product migration, and, most important, the interaction of human beings with a complex technology. Moreover, the successful computer analyses of the Three Mile Island accident have helped to convince the skeptics of the predictive capabilities of computer simulation. Advanced computer codes are no longer relegated to the domain of research. They are now being applied to very practical licensing and regulatory problems—the one of most interest to this editor being the prediction of accident signatures and the development of detailed instructions for accident management that, if used well, can reduce the risks from reactor accidents.

Most of the articles in this issue are objective descriptions of the technical achievements in code development and application, but the reader will no doubt detect an implicit pronuclear slant. Perhaps this is to be expected from the Laboratory that began the nuclear age and designed some of the very first reactors. The scientists here live and work around radioactive materials and reactor systems, and their immediate experience contradicts the public's fears. Moreover, their technical knowledge tells them that risks to the public and the environment from nuclear energy are much smaller than those from fossil fuels.

However, these technical experts are also acutely aware of the economic pressures and management realities that divide a commercial enterprise from a research project. As a government-sponsored research group kept at arm's length from industry, they retain a relative objectivity that allows them to see the problems clearly and to suggest technically sound solutions. Their influence should be welcomed by both the public and the industry.

Necia Grant Cooper

A PRIMER on REACTOR SAFETY ANALYSIS

by Michael G. Stevenson and James F. Jackson

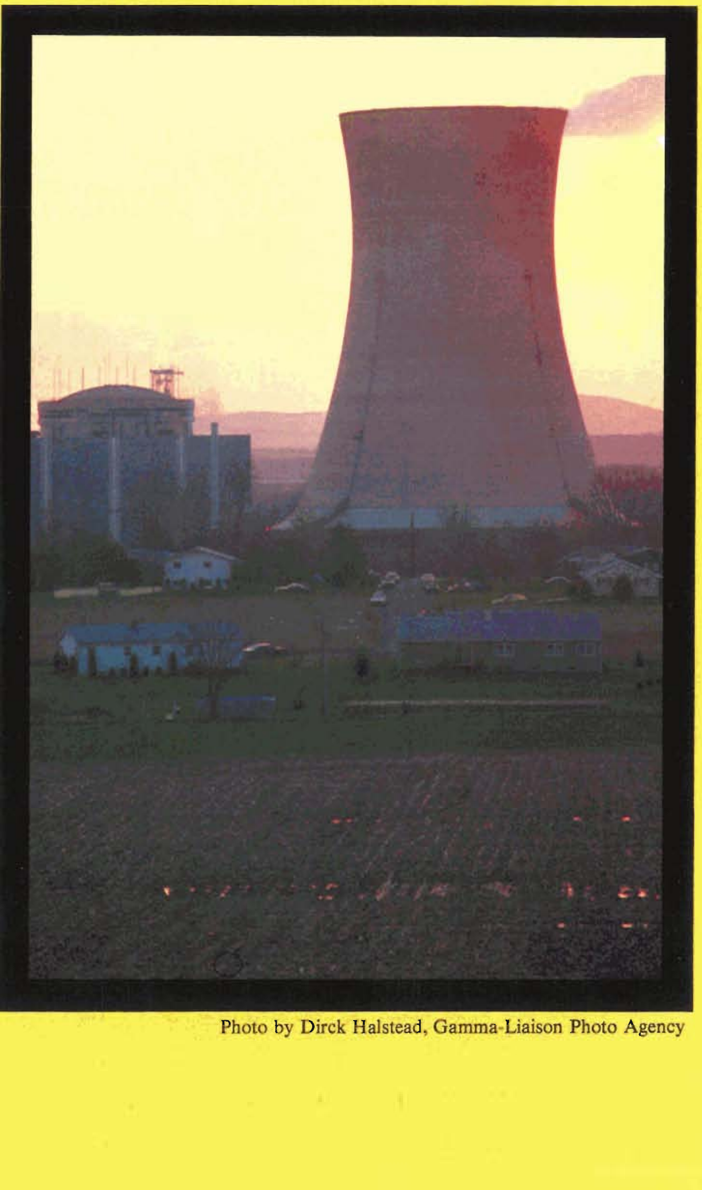

Photo by Dirck Halstead, Gamma-Liaison Photo Agency

How do we know that the emergency cooling system in a nuclear reactor will work in case of an accident? Although an automobile can be crash-tested to evaluate its safety performance, it is not practical to subject a full-scale nuclear power plant to severe accident conditions. Moreover, there are so many accident paths to be considered that the costs of full-scale experiments for all of them would be prohibitive. Therefore, the nuclear power industry relies more heavily on theoretical analysis of design and safety features than does any other high-technology industry.

Before the Three Mile Island accident, much of the safety analysis of commercial reactors focused on a hypothetical accident involving the rupture of a large pipe supplying cooling water to the reactor core. This design-basis loss-of-coolant accident was thought to be worse than any event that would ever happen. Water and steam would be expelled rapidly out the break (Fig. 1) and the core would be left temporarily uncovered and poorly cooled. Reactor designers and their critics disagreed as to whether or not the emergency core-cooling system would be able to inject water into the reactor core in time to prevent melting of the core and possible release of large quantities of radioactive material to the environment. To help settle this controversy, the Nuclear Regulatory Commission asked Los Alamos to develop a computer code that could realistically simulate the response of a reactor to this very unlikely event. The code, called TRAC, predicted that the emergency cooling system would reflood the core within two or three minutes after the break and that the core temperature would remain far below the melting point of the fuel. The code confirmed results of the less accurate, more conservative analysis methods that are the basis for reactor licensing. Thus, at the time of the Three Mile Island accident, most of the nuclear community believed that the probability of an accident involving core meltdown and major radiation release was so low that they should never have to deal with one.*

The events in Harrisburg, Pennsylvania have changed this perspective. Certainly, the careful design of reactor hardware was successful in preventing an astounding series of equipment malfunctions and misinterpretations by the reactor operators from developing into a serious threat to public safety. But on the

Despite their low probability, potential radiation releases from accidents involving core damage are formally considered in evaluating proposed sites for nuclear power plants.

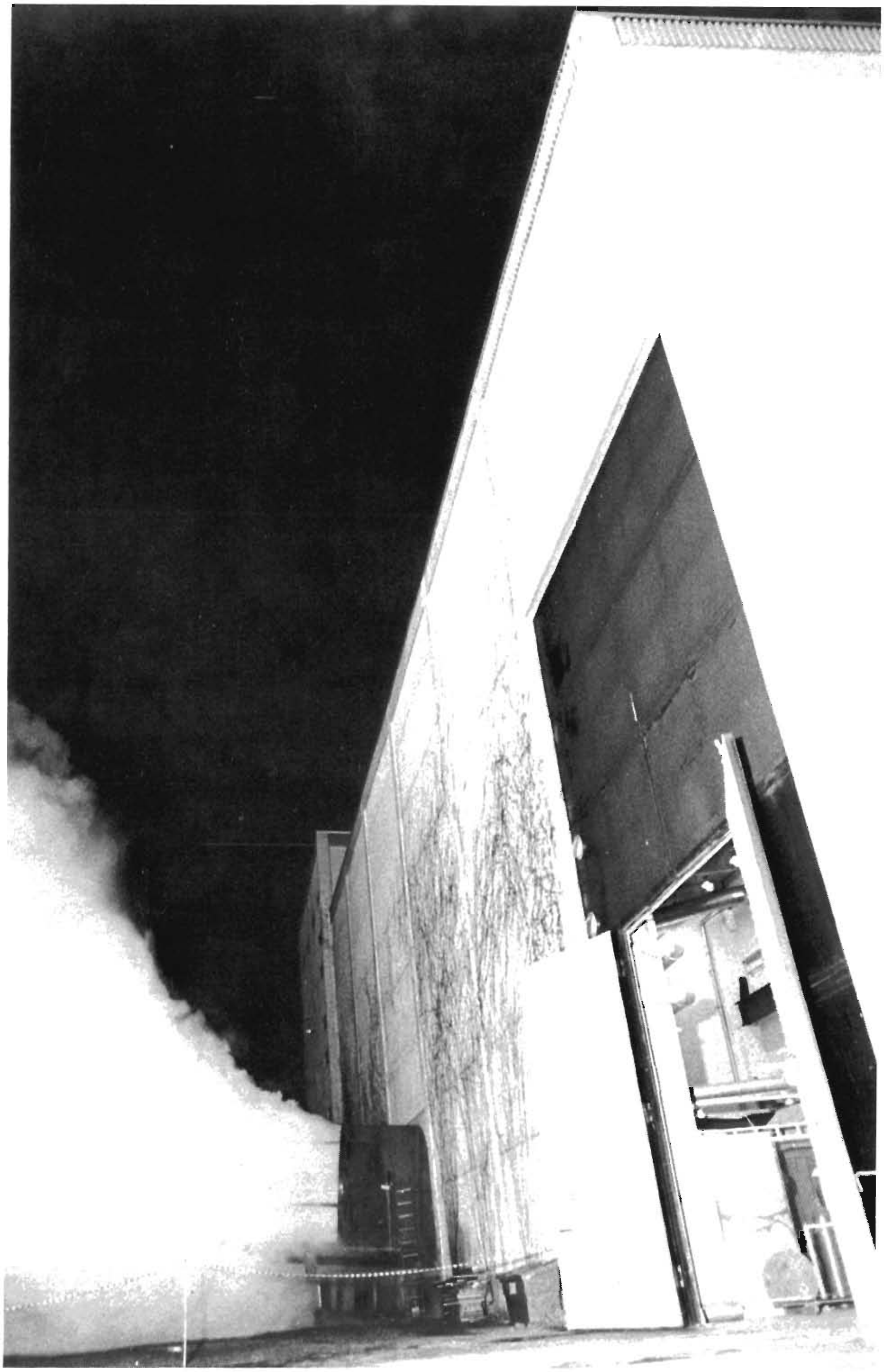

Fig. 1. Originally designed as a small nuclear power plant, the Marviken facility in Sweden has been converted to an experimental facility for studying the ejection of water from a ruptured pipe in a water-cooled reactor. Water in the reactor vessel is heated (with fossil fuel—the facility has no nuclear core) to a temperature and pressure typical of an operating reactor. The pipe break is simulated by opening a large valve at the bottom of the vessel and allowing the steam and water to be ejected into the building. Shown here is the front face of the building during a test. The huge jet of steam is being vented through a large pipe (several feet in diameter) installed in the side of the building. The scene inside the building must be awesome indeed. Data from these tests are being used in the development of accident-analysis codes at Los Alamos and elsewhere. (Photo courtesy of Studsvik Energiteknik AB.)

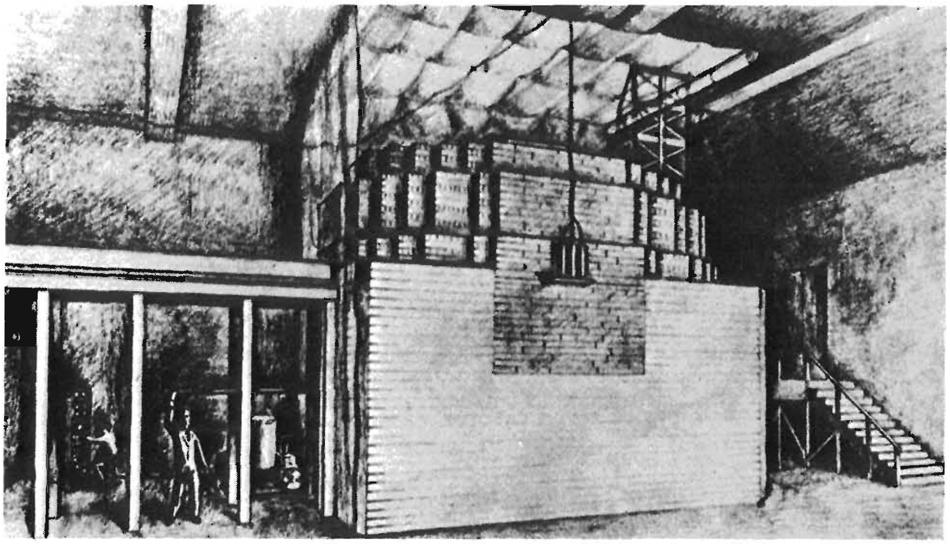

Fig. 2. An artist's sketch of the first nuclear reactor constructed in 1942 in a squash court under the west stands of Stagg Field, University of Chicago. It was made from about 40 tons of natural uranium and 385 tons of graphite. Note the manually operated control rod extending from the side of the "pile" and the large neutron detectors located at the upper part of the front face. The safety systems for this first atomic pile were especially simple. In addition to two sets of control rods, there was a rod called Zip that operated by gravity through weights and a pulley. In an emergency, or if the person holding the rope collapsed and let go, the rod would be drawn rapidly back into the pile. The back-up system was a "liquid-control squad" of three people standing on a platform over the pile ready to flood it with a neutron-absorbing salt solution.

morning of March 28, 1979, several hours after the Three Mile Island accident began, the reactor core was less than an hour away from meltdown. Melting of the fuel would not necessarily have resulted in a major radiation exposure of the public. However, the consequences of a possible meltdown are now being considered much more seriously in the licensing process.

The inquiries following Three Mile Island identified management problems rather than hardware problems as the main reason that a minor mechanical failure developed into a rather serious accident. The critical areas of operator training and human factors engineering had been underemphasized by the nuclear industry. The Nuclear Regulatory Commission had focused most of its attention on the licensing process, in which detailed safety analysis reports submitted by license applicants are reviewed with the help of technical experts and sophisticated computational tools. But the Commission was found not so well equipped to correct operating deficiencies in the 70 commercial light-water reactors now producing power in this country.

The philosophy guiding the Commission's work has begun to change and with it the work done for the Commission by the national laboratories. Accident analysis is still one of the major tasks, but its focus has shifted to accidents resulting from multiple malfunctions of plant components. The intent now is to simulate not only the automatic response of the system but also the consequences of human intervention. Out of such analyses, the Commission expects to get ideas for better feedback controls, to identify and catalog accident signatures so that operators can better tell what is going wrong, and to develop operator responses that will mitigate the consequences of system failures.

The Commission is also funding development of new computer codes to simulate accidents involving core melting and to trace the subsequent path of radioactive materials. And the laboratories are analyzing the capabilities of the containment systems that must prevent release of radiation should there ever be another serious accident. These activities will help implement the lessons learned at Three Mile Island.

Reactor Basics

Although a modern nuclear power plant is a very complex system designed to exacting specifications, a nuclear reactor, by itself, is a relatively simple device. In 1942 Enrico Fermi and his colleagues built a crude reactor on the first try (Fig. 2). By placing pieces of natural uranium in a stack of graphite blocks, they achieved a self-sustained and controlled nuclear fission chain reaction, and thereby demonstrated the potential for generating a large amount of usable energy.

The energy-producing process is nuclear fission, in which a nucleus absorbs a neutron and breaks apart into several fragments (Fig. 3). This process releases millions of times the amount of energy released in a typical chemical reaction and occurs readily in what are referred to as fissile isotopes. (Uranium-235 is the only naturally occurring fissile isotope; other examples are plutonium-239 and uranium-233.)

Practical application of fission as an energy source rests on another remarkable fact. Among the products of fission are additional neutrons that can themselves initiate fission of other nuclei and so begin a chain reaction. Sustaining this chain reaction has one basic requirement: a sufficiently large mass of fuel, what we call a critical mass. With less than this critical mass, too many neutrons escape from the fuel and the chain reaction stops.

Because thermal, or slowly moving, neutrons have a much higher probability of inducing fission in uranium-235 than do fast neutrons, most uranium-fueled reactors, including the first one, are designed to run on thermal neutrons. To slow the fast neutrons produced by the fission process to thermal energies, the fuel is surrounded by a "moderator" containing relatively light nuclei. The neutrons lose energy by collisions with these light nuclei. (In Fermi's reactor the graphite served as a neutron moderator.)

A single fission reaction typically produces two, or sometimes three, neutrons, but not all these are available to induce new fissions. Some are absorbed without inducing fission and some leak out of the core. To produce a stable power level in a reactor, the neutron population must be controlled so that on the average each fission causes only one additional fission. Gross control is achieved by moving control rods in and out of the core.* These control rods contain materials, such as boron or cadmium, that readily absorb neutrons (without undergoing fission) and thereby remove some of the neutrons from further participation in an ongoing chain reaction. Fail-safe systems are provided to insert control rods rapidly into the core and halt the chain reaction altogether under emergency conditions. This process is referred to as a reactor scram.

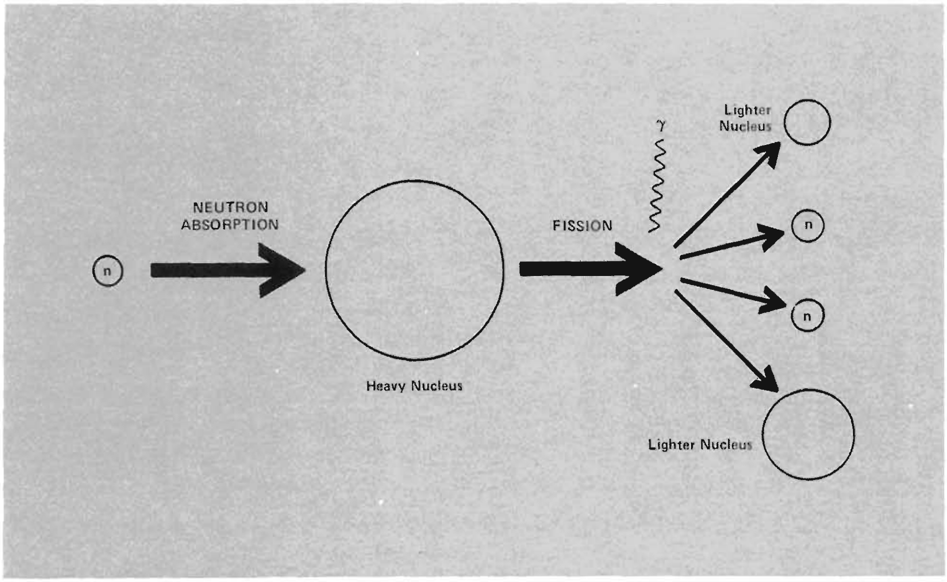

Fig. 3. The fission process. A heavy nucleus, such as uranium-235 or plutonium-239 absorbs a neutron (n) and breaks up into two lighter nuclei, two or sometimes three neutrons, and gamma rays. The lighter nuclei are usually radioactive.

Further control of a reactor arises from negative temperature-feedback effects that provide inherent stability. As the number of fissions increases, the resulting increased core temperature produces changes in material properties that tend to shut down the chain reaction. This self-regulation makes a well-designed reactor quite easy to control.

Most of the energy released by fission appears as kinetic energy of the lighter nuclei that are formed when the heavy nuclei split. These fission products collide with neighboring fuel nuclei and are slowed down within a very short distance. Their kinetic energy is converted to heat that transfers from the fuel to a liquid or gas coolant pumped through the reactor core. To prevent the core from overheating, the rate of heat transfer to the coolant must equal the rate of energy production in the core. The heat in the coolant can then be used to produce steam for electric power generation.

**Mechanical control of the neutron population is possible because of the delayed neutrons. For a discussion of neutronics, see "Breeder Reactor Safety—Modeling the Impossible" in this issue.*

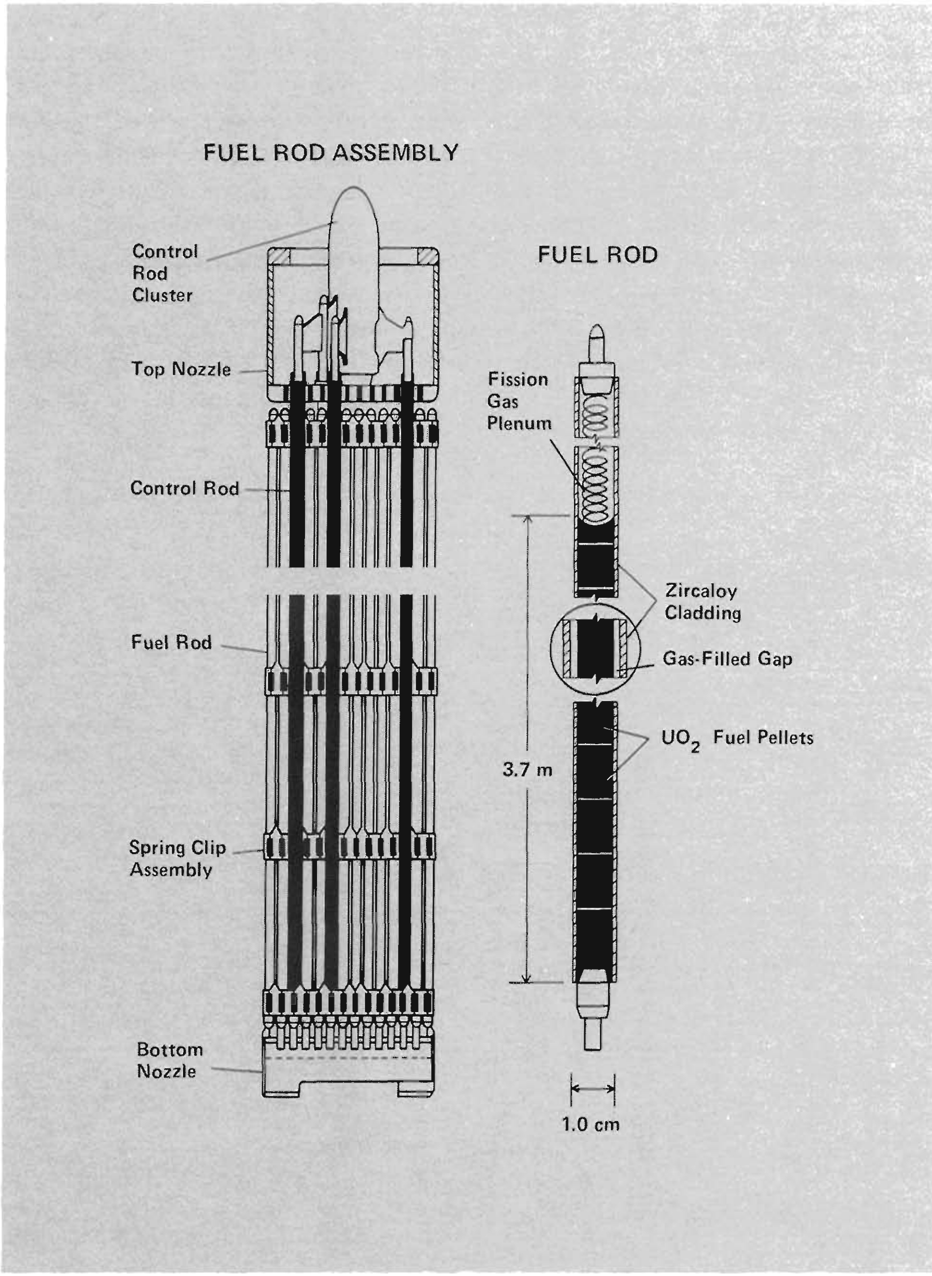

Fig. 4. Fuel rod and fuel-rod assembly for a pressurized-water reactor. Fuel rods are held in a square array by spring clip assemblies and by grid assemblies at the top and bottom. The structure is open permitting flow of coolant both horizontally and vertically. Control-rod guide tubes are interspersed among the fuel rods. Control-rod assemblies are lowered into the guide tubes to absorb neutrons and control the chain reaction. A typical core contains about 200 fuel-rod assemblies each containing about 200 fuel rods.

Commercial Light-Water Reactors

The goal of commercial reactor design is to build a plant that usually generates 1000 megawatts, or more, of electric power during normal operation and does not allow damage to the reactor core during all foreseeable circumstances. A typical reactor core is relatively small and could fit easily on a single railroad car. However, it contains enough fuel to produce 1000 megawatts electric for three years—the energy equivalent of 100,000 carloads of coal. To extract this amount of usable energy from a relatively small volume, a tremendous quantity of high-temperature water must be pumped through the core at a very high flow rate. In a typical pressurized-water reactor, 7500-horsepower pumps in each of two or four primary coolant loops move the water from the core to 21-meter-high (70-foot) steam generators.

Except for one gas-cooled reactor, all commercial nuclear power plants in the United States are light-water reactors; that is, they use ordinary "light" water to cool the core rather than the "heavy" water (D_20) used in some designs. The water also serves as a neutron moderator. Commercial light-water reactors are fueled with enriched uranium that contains 3% by weight of the fissile isotope uranium-235 as opposed to the 0.71% found in natural ores. The fuel is in the form of small ceramic pellets of uranium dioxide. To make a fuel rod, the fuel pellets are sealed in tubes about 4 meters (12 feet) long and not much wider than the diameter of a pencil. This protective cladding is fabricated from a special zirconium alloy (Zircaloy). About 40,000 fuel rods, held rigidly in place with special structures, make up the core of a light-water reactor. Examples of a fuel rod and a fuel-rod assembly are shown in Fig. 4.

Figure 5 shows a typical pressurized-water reactor, the most common type of light-water reactor. They are manufactured in the United States by Westinghouse Electric Corporation, Combustion Engineering, Inc., and Babcock & Wilcox. The diagram shows both the primary coolant loop, which transfers heat from the core to the steam generators, and the secondary coolant loop, which transports steam from the steam generators to drive the turbine-generators that produce electricity.

To preclude boiling and thereby maintain a high rate of heat transfer from the fuel rods to the coolant, the primary coolant water is pressurized to about 150 bars.* [The reactor vessel is fabricated from 25-centimeter-thick (10-inch) steel to withstand this high internal pressure.] The coolant is pumped down through an annular region surrounding the core (the downcomer) and up through the core where it is heated to about 590 kelvin (about 600° Fahrenheit). The heated water exits from the reactor vessel and flows through large steam generators where heat is transferred to water in the secondary loop. This water is at a lower pressure and rapidly boils. The steam then drives a turbine just as it does in any conventional power plant.

*1 bar = 10^5 pascals $\cong$ 1 atmosphere.

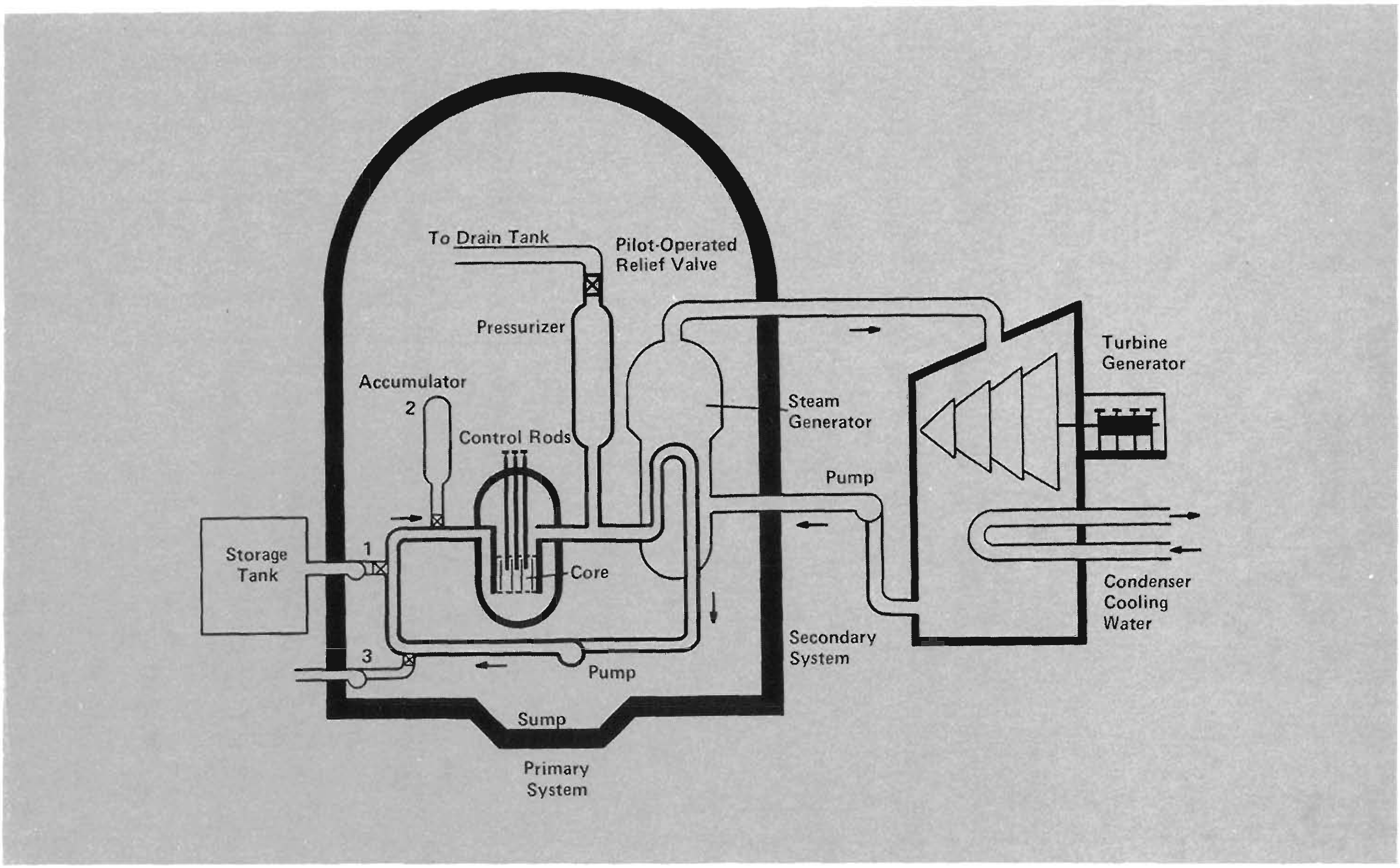

Fig. 5. A pressurized-water reactor showing a primary loop, a secondary loop, and the three subsystems (labeled 1, 2, and 3) of the emergency core-cooling system. In the primary system, water under high pressure (about 150 bars) is pumped by 7500-horsepower centrifugal pumps through a 12-meter-high (40-foot) reactor vessel to 21-meter-high (70-foot) steam generators.

Fig. 6. Control room of a commercial nuclear power plant. A myriad of lights, dials, and switches monitors and controls all the complex systems within the plant. (Photo courtesy of Florida Power & Light Company.)

Figure 5 also shows a typical emergency core-cooling system, which replaces water in the event of a leak in a primary coolant loop. Three separate subsystems are available depending on the pressure loss resulting from the leak. If the system pressure drops from the normal 150 bars to about 130 bars, a set of high-pressure pumps automatically inject water. (These pumps activated automatically during the early stages of the Three Mile Island accident but the operators turned them off because they misinterpreted what was occurring.) A further drop in pressure to about 40 bars (14 bars for Combustion Engineering plants) will cause the accumulator check valves to open automatically, allowing water from these large pressurized tanks to flow into the reactor vessel. Finally, at a pressure of about 14 bars, high-capacity, low-pressure pumps are activated that can supply large volumes of water. These pumps can ultimately obtain their water supply from a sump in the bottom of the reactor containment building where water would collect from any massive leaks.

During normal operation, the system pressure is regulated by the pressurizer, a large tank partly filled with water and connected to the primary system. To control the system pressure, steam in the upper part of this tank is heated with electric coils or condensed with cold-water sprays. The pilot-operated relief valve at the top of the pressurizer was the valve that stuck open and allowed a large amount of coolant to escape during the Three Mile Island accident.

The cooling, control, and in-depth safety systems, together with the balance-of-plant components, make a modern nuclear power plant a large and awesome construction. A plant has hundreds of valves, pumps, piping circuits, and instruments. The large control rooms are equipped with hundreds of instrument readout devices and system control switches (Fig. 6). It is believed that this complexity was a contributing factor to the difficulty the reactor operators had in quickly diagnosing the accident at Three Mile Island.*

The other type of commercial light-water reactor, the boiling-water reactor, is manufactured by General Electric Co. Rather than primary and secondary cooling loops, this reactor has one loop connecting the core to the turbine-generator. The cooling water is maintained at a low enough pressure (about 70 bars) to allow boiling in the reactor core. The steam is then piped directly to the turbine. Boiling-water reactors are also equipped with emergency core-cooling systems.

There are fewer boiling-water reactors than pressurized-water reactors in commercial operation. Because Los Alamos has not done extensive safety analysis of boiling-water reactors, they will not be discussed further.

*Report of the President's Commission on the Accident at Three Mile Island, The Need for Change: The Legacy of TMI *(U. S. Government Printing Office, Washington, D.C., 1979), p. 11.*

Two Trouble Spots—Fission Products and Decay Heat

Two of the most troublesome aspects of a reactor arise from the fact that the fission products are radioactive. First, of course, these radioactive materials must be isolated from the biosphere. Second, decay of the radioactive fission products is a heat source that cannot be turned off, even after the fission process has been shut down (Fig. 7). In a reactor that has been operating for some time, the power due to decay heat is a significant fraction (about 7%) of the total power. After shutdown, decay power decreases to about 1% in a few hours, but this 1% amounts to about 30 megawatts thermal in a large commercial reactor. Thus, to prevent damage to the core and possible release of radioactive materials, every power reactor must have provision for removal of decay heat under all foreseeable conditions.

During normal operation many fissions occur every second (about 10^{20} in a 1000-megawatt-electric reactor), and a spectrum of fission products results. Most fission products are neutron rich and unstable, and tend to decay by emission of beta particles and gamma rays.

The fission products are often characterized as gases, volatiles, or solids depending on their boiling temperatures. The gaseous products are mostly the inert gases xenon and krypton. Several isotopes of iodine are also produced and are an important potential radiological hazard. Some fission products, particularly noble metals with high boiling points, remain solid in the fuel pellets at normal operating temperatures and even at abnormally high temperatures during accident conditions.

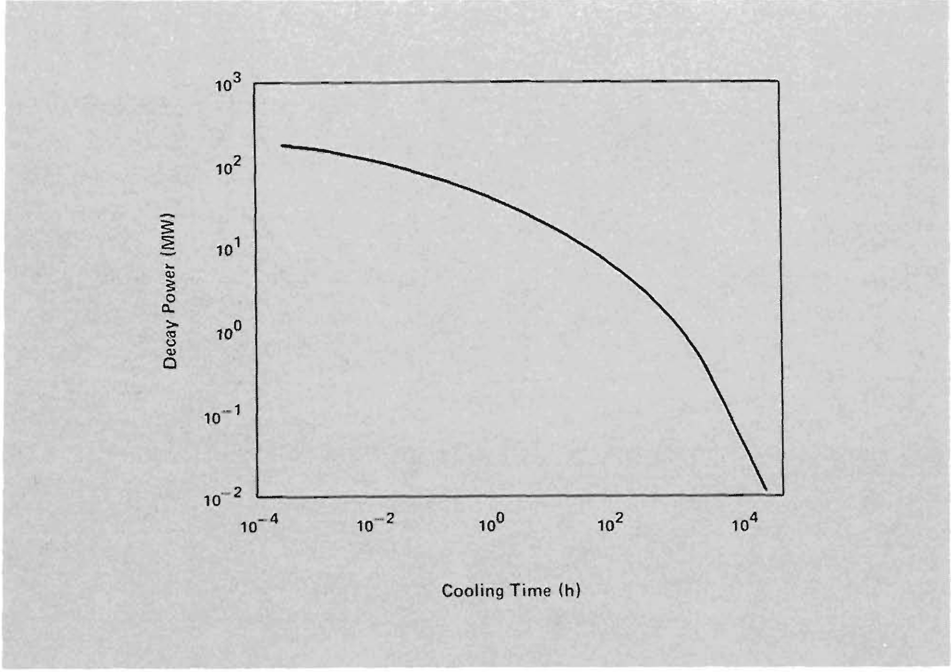

Fig. 7. A log-log plot of decay power as a function of time after reactor scram for the Three Mile Island Unit 2 reactor. This curve was calculated by the Laboratory's Nuclear Data Group. It depends on the reactor's power history and fuel and fission-product inventories and on details of the decay chains that fission products and transuranics follow as they spontaneously decay to more stable nuclear states.

Also contributing to the decay power and the potential danger posed by a reactor are "transuranics," elements beyond uranium in the periodic table. These are the result of neutron-induced reactions other than fission in fuel nuclei. The transuranics generally decay by emission of alpha particles and accompanying gamma rays.

Multiple Barriers — Design for Safety

As long as the fission products and transuranics remain confined, the impact of a reactor on operating personnel, the public, and the environment is very small. Four distinct barriers (Fig. 8) are designed to confine the radioactive materials: the ceramic (uranium dioxide) fuel pellets, the fuel-rod cladding, the primary

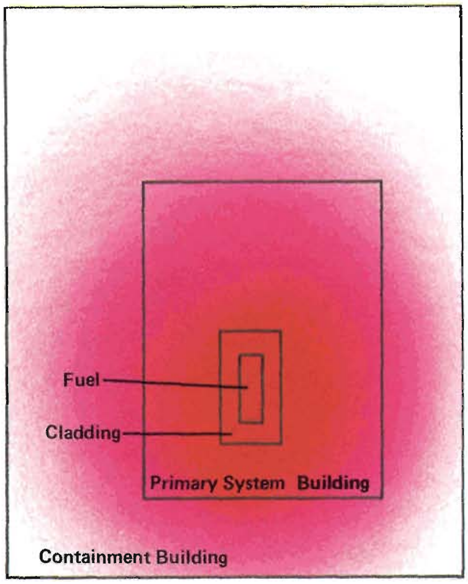

Fig. 8. The four barriers against release of radioactive materials in a pressurized-water reactor.

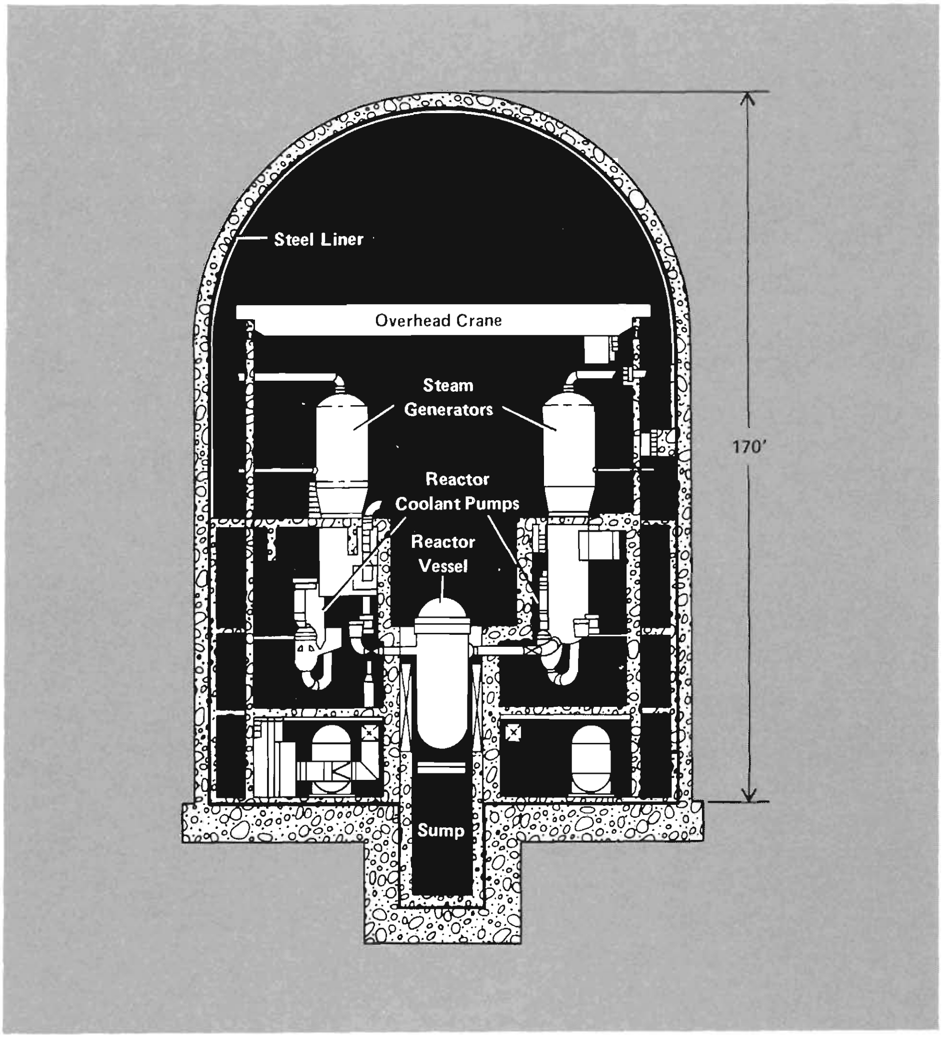

Fig. 9. Cross section of a typical containment building for a pressurized-water reactor. The concrete containment building houses the entire primary system, the pressure control system, ventilation equipment, and part of the emergency core-cooling system. The various components are encased in concrete and surrounded by a 0.63-centimeter-thick (0.25-inch) steel liner.

system boundary, and, finally, the containment building.

The uranium dioxide fuel pellets provide the first barrier against radiation release. Their exceptionally high melting point (3040 kelvin, or about 5010° Fahrenheit) and chemical stability prevent escape of nearly all fission products except in extreme accident conditions.

In normal operation, a small amount (about 1%) of the gaseous fission products do leak from the pellets but, under most conditions, are confined by the second barrier, the Zircaloy cladding surrounding the fuel pellets. If the core temperature rises during an accident, the cladding will generally fail before the fuel pellets melt, and this small fraction of gaseous products will escape to the primary coolant. During the Three Mile Island accident, the radiation problems were almost entirely due to gaseous fission products. (Evidently, there was little or no fuel-pellet melting.) It has been assumed that radioactive iodine would be one of the gaseous fission products released if the cladding were to fail. However, this assumption has been challenged by information from the Three Mile Island accident.*

The primary system boundary (see Fig. 5) is the third barrier preventing release of fission products. The reliability of this boundary is assured by the inherent strength of the thick vessel and piping and also by continual inspection of these components throughout the life of the plant. Nevertheless, spontaneous small and large breaks in this boundary are considered as possibilities for initiating loss-of-coolant accidents.

The reactor containment building is the fourth and final barrier to fission product release (Fig. 9). For light-water reactors, the containment generally consists of a steel liner surrounded by a 1.2-meter-thick (4-foot) structural concrete shell. This combination prevents leaks and can withstand a substantial internal overpressure, as well as external impacts caused by tornadoes, external explosions, or aircraft crashes. The containments are designed, conservatively, to stay intact during a worst-case loss-of-coolant accident, which would produce a building pressure of about 4 bars. This safety feature was important

**See "Good News about Iodine Releases" in this issue.*

in reducing consequences of the Three Mile Island accident, during which the containment withstood a pressure spike of about 2 bars. The pressure spike was evidently caused by rapid burning of hydrogen produced by oxidation of hot zirconium cladding.

But what is the maximum pressure that these strong containments can resist? To answer this question, Los Alamos and Sandia National Laboratories are carrying out a "Structural Margins to Failure" research program for the Nuclear Regulatory Commission. A later article summarizes some of the work in this area.*

Even with the containment intact, radiation can possibly be released through an indirect path. For example, at Three Mile Island, primary coolant water lost through the open pilot-operated relief valve eventually escaped to the containment and was pumped to storage tanks in an auxiliary building nearby. These tanks overflowed and led to small releases of gaseous fission products to the environment through the exhaust stack. To prevent such occurrences, all possible release paths and transport mechanisms, such as flowing water, must be considered.

Safety Analysis

The safety analyst's job is to determine, for any postulated accident, whether the maze of barriers stays intact and whether radioactive materials stay contained. But the maze is complex and changing during an accident. The locations and sizes of the barrier failures, the release paths, and the transport mechanisms all depend on temperature and pressure. The analyst must start from the beginning and predict the thermal and physical conditions throughout the entire accident.

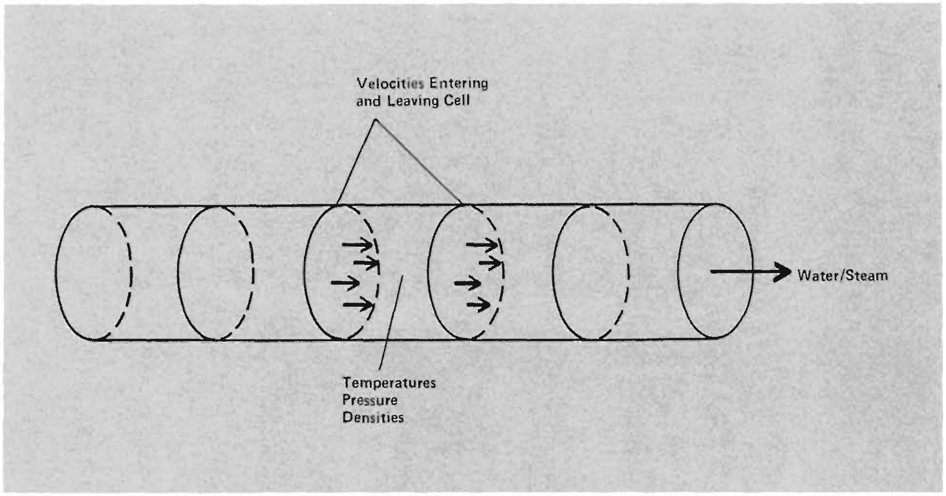

Fig. 10. Division of a coolant pipe into computational cells. Densities, pressures, and temperatures at the center of each cell are computed, as well as the velocities of the steam-water mixtures entering and leaving each cell.

The analysis usually requires a sophisticated computer model to simulate the energy and material flows throughout the system. Such models break down the system into many cells—small boxes of space—and audit the mass, temperature, and velocity of the materials in each cell. Figure 10 shows a typical cell structure for one component of a light-water reactor, a pipe.

The analysis begins with the reactor running smoothly at full power. Then something is assumed to go wrong—a pump fails or a pipe breaks—and the computer calculation follows the changes in water and steam flow rates and in system temperatures and pressures. Reactor scram and injection of emergency cooling water are also simulated as they would occur in the accident.

The computer model includes all or a large part of the complicated system of plant components. The analysis tracks in time the system's thermal hydraulics, including compressible two-phase steam-water flow—an engineering and computational problem of considerable difficulty.**

Energy Balance in the Reactor Core

The equations used in these computer codes assume conservation of mass, energy, and momentum for all the materials in each of the hundreds of cells in a typical calculation. Here we will discuss energy conservation to illustrate the factors influencing the core temperature. We start with an extremely simple model consisting of but one cell, the core as a whole.

*See "The Structural Integrity of Reactors" in this issue.

**See "Two-Phase Flow" in this issue.

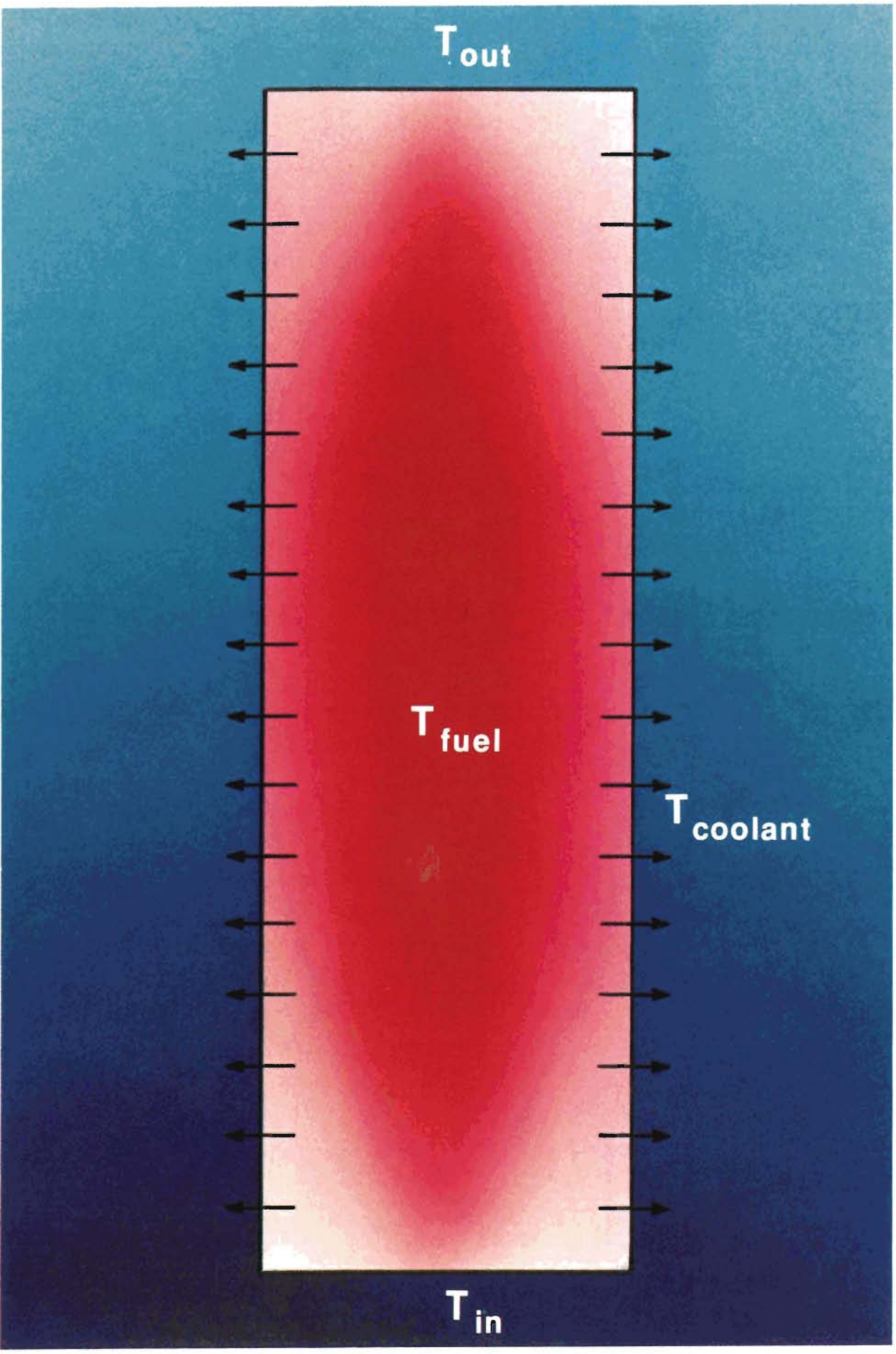

Fig. 11. In a two-cell representation of the core, coolant flowing past the nuclear fuel is heated at the rate $hA(T_{fuel} - T_{coolant})$, where h is the heat-transfer coefficient and A is the surface area of the fuel.

Energy released by nuclear processes (fission of the fuel and radioactive decay of the fission products) is produced in the core at a rate $Q_{nuclear}$. Conservation of energy says that this energy either is stored in the core at a rate Q_{core} or heats the coolant circulating through the core. The energy-conservation equation for our one cell model is thus

$$Q_{core} = Q_{nuclear} - 2Wc_p\,(T_{core} - T_{in}) \;, \tag{1}$$

where W is the mass flow rate of the coolant, c_p is the specific heat of the coolant, and T_{in} is the temperature of the incoming coolant. The core temperature, T_{core}, is assumed to be the average coolant temperature, that is, $\frac{1}{2}\,(T_{in} + T_{out})$, where T_{out} is the temperature of the outgoing coolant. (A more complete analysis would include the mass- and momentum-conservation equations needed to determine the coolant flow rate W. A more detailed model that "closed the loop" through the steam generator would provide a value for T_{in}.)

What can be learned from this simple energy-conservation equation? First, to maintain the core at a constant temperature, Q_{core}, which is proportional to dT_{core}/dt, must equal zero. Therefore, the nuclear heat production rate must be exactly balanced by the rate at which heat is removed by the flowing coolant. That is, $Q_{nuclear} = 2Wc_p\,(T_{core} - T_{in})$. Increases in $Q_{nuclear}$ associated with some normal operating procedures are countered by increasing the flow rate W (a usual maneuver) or by decreasing the inlet temperature T_{in}. The latter can be accomplished by removing more heat from the coolant in the steam generators.

An increase in Q_{core} can result from a decrease in the heat-removal rate. As a bounding example, suppose that all cooling of the core is suddenly lost while the reactor is scrammed, that is, when $Q_{nuclear}$ consists only of decay power Q_{decay}. Then, from Eq. 1, $Q_{core} = Q_{decay}$. For a typical light-water reactor core at decay power levels, we can estimate that the core temperature increases at a rate of about 0.5 to 1 kelvin (0.9 to 1.8° Fahrenheit) per second. At this rate, some tens of minutes are required for a completely uncooled core to heat to the fuel's melting point.

Assuming now that our model consists of two cells, fuel and coolant, we can illustrate the importance of the convective heat-transfer rate between them (Fig. 11). The rate of this transfer is the product of an overall heat-transfer coefficient h, the fuel surface A, and the difference between the average fuel and coolant temperatures, $T_{fuel} - T_{coolant}$. Again, energy balances provide equations for Q_{fuel} and $Q_{coolant}$.

$$Q_{fuel} = Q_{nuclear} - hA\ (T_{fuel} - T_{coolant}) \tag{2}$$

and

$$Q_{coolant} = hA\ (T_{fuel} - T_{coolant}) - 2Wc_p\,(T_{coolant} - T_{in}) \ . \tag{3}$$

Here again, W and T_{in} can be determined as indicated for Eq. 1.

Equation 2 illustrates the significance of "burnout" to balancing the rates of heating and cooling. (Burnout is the traditional term used in the boiler industry for situations where heat fluxes become so high that a boiler tube dries and melts, that is, burns out.) During normal operating transients in which $Q_{nuclear}$ increases, heat is transferred from fuel rods to coolant by the efficient processes of turbulent forced convection and nucleate boiling. In nucleate boiling, small vapor bubbles form rapidly on the surface and are swept away by the fast-flowing coolant. The heat-transfer coefficient is very large for this process, and heat fluxes across the cladding-coolant interface can be quite high even at low temperature differences. If, however, the heat flux exceeds a critical value, departure from nucleate boiling occurs, and the cladding surface becomes covered mostly by a film of steam. Because the heat-transfer coefficient between cladding and steam is very small, the rate of heat removal is low even for large temperature differences. Consequently, peak heat fluxes in an operating pressurized-water reactor are restricted to less than about 75% of the value at which departure from nucleate boiling occurs, and operational control systems are designed to maintain this condition during all normal power changes.

However, departure from nucleate boiling and even complete dryout of the fuel rods can occur under accident conditions such as the design-basis large-break loss-of-coolant accident mentioned earlier. The rapid loss of coolant would depressurize the primary system and cause vaporization of the remaining water and dryout of the fuel rods. Poorly cooled by the steam, the core would overheat were it not for the automatic activation of the emergency core-cooling system.

But how well do these systems actually work? To reach the lower plenum below the core, emergency coolant must flow down the downcomer against an upward flow of steam. Does most of the water flow around and out the break instead of down to the lower plenum? Once the lower plenum is filled, the core must be reflooded with water and the fuel rods quenched. Most people are familiar with the vigorous boiling-quenching process when a fire poker at, say, 530 kelvin (500° Fahrenheit), is inserted into a bucket of water. For a reactor, think of 40,000 pokers, 4 meters (12 feet) long, and at, say, 920 kelvin (1200° Fahrenheit) plunging into a 4.6-meter-diameter (15-foot) bucket of cold water. The cooling water initially entering the core would be almost instantly vaporized, much like water thrown into a hot skillet, and the huge amount of steam generated would tend to prevent more water from entering the core. How long does it take to reflood the core and quench the rods? Will the fuel rods get hot enough to fail before they are quenched?

Since it is impractical to perform a full-scale demonstration of the emergency core-cooling system under these extreme circumstances, the answers to these questions have had to come from theoretical analyses backed by numerous smaller-scale experiments.

Code Development for Light-Water Reactor Safety Analysis

In 1970 the Nuclear Regulatory Commission developed standards for assessing the adequacy of emergency core-cooling systems and codified them in Appendix K of Federal Regulation 10CFR50. Methods of analysis as well as performance criteria are included. For example, before a reactor can be licensed, the owner of a proposed facility must show through analysis based on an "evaluation model" that the peak clad-

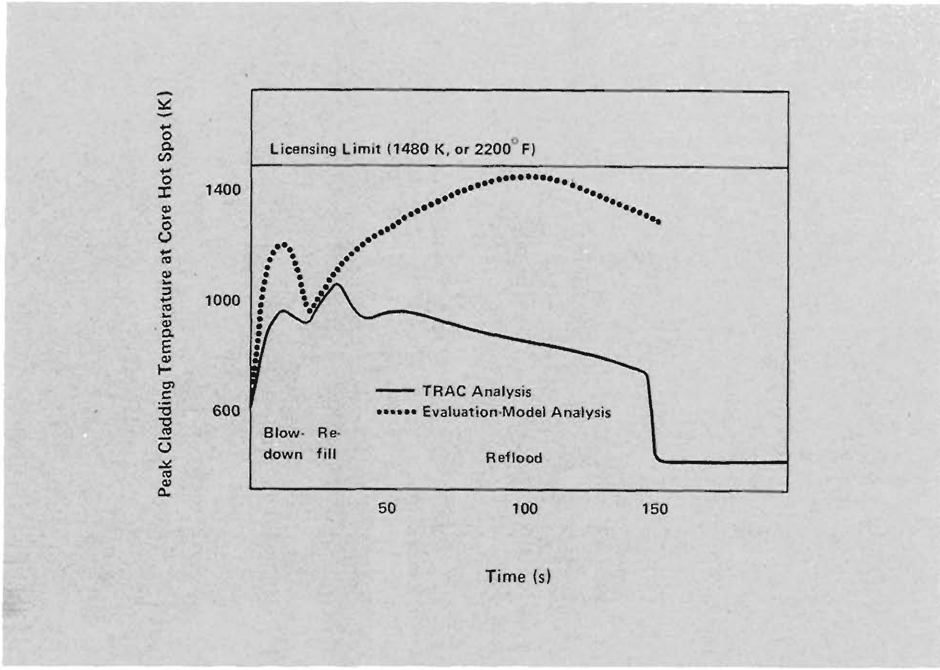

Fig. 12. Cladding temperature histories during a large-break loss-of-coolant accident in a typical four-loop pressurized-water reactor. One history (solid curve) is a TRAC analysis [J. R. Ireland and D. R. Liles, "A TRAC-PD2 Analysis of a Large-Break Loss-of-Coolant Accident in a Reference US PWR," Los Alamos Program technical note LA-2D/3D-TN-81-10 (March 1981)]; the other (dotted curve) is an evaluation-model, or conservative, analysis [G. W. Johnson, F. W. Childs, and J. M. Broughton, "A Comparision of 'Best-Estimate' and 'Evaluation Model' LOCA Calculations: The BE/EM Study," Idaho National Engineering Laboratory report PG-R-76-009 (December 1976)].

ding temperature would not exceed 1477 kelvin (2200° Fahrenheit) during the design-basis loss-of-coolant accident.* The evaluation model defined in Appendix K includes conservative assumptions, such as an unrealistically low heat transfer from the fuel rods to the coolant during the initial depressurization of the primary system. Despite these conservative assumptions, evaluation-model analyses were heavily criticized by scientists outside the industry. Many simplifications were required to perform the analyses, and, consequently, there was no assurance that the resulting predictions were, in fact, on the safe side. In 1974 an American Physical Society committee identified the thermal-hydraulics codes used for the analyses as the weakest link in the licensing process.**

It was to help counter this criticism that the research arm of the Nuclear Regulatory Commission began funding the Laboratory to develop TRAC, a state-of-the-art thermal-hydraulics code capable of simulating the complete design-basis loss-of-coolant accident sequence in one continuous calculation. Because this large system code was to cover an enormous range of thermal-hydraulic phenomena in a complete primary system, approximate models of the various phenomena had to be used. To aid and complement development of these models, the Commission also began funding more detailed analyses of individual reactor components and physical processes. Some of these analyses are described in a later article.***

Although TRAC was to include the most advanced numerical techniques available at the time, there was some skepticism about whether the code would work at all, much less provide realistic predictions in a reasonable computing time. But less than three years after development efforts began, it produced the first complete calculation of a large-break loss-of-coolant accident in about 30 hours on a CDC-7600. (Later versions of TRAC run much faster.) Figure 12 shows typical results for cladding temperatures during a large-break loss-of-coolant accident. The predicted peak cladding temperature (about 1030 kelvin, or 1400° Fahrenheit) is much lower than the limit set by the Nuclear Regulatory Commission, and we have

**This is the temperature above which the zirconium-steam reaction proceeds at a significant rate.*
***H. W. Lewis, Chairman, "Report to the American Physical Society by the Study Group on Light-Water Reactor Safety," Reviews of Modern Physics* 47, *Supplement No. 1 (1975).*
****See "Detailed Studies of Reactor Components" in this issue.*

considerable assurance from loss-of-coolant experiments that this temperature is correct. The emergency cooling process is turbulent and chaotic—but it works.

Both the models and methods of the TRAC code and its experimental verification are discussed in a later article.* Comparison of TRAC calculations with a large number of experiments shows generally good agreement and has led to improved models, particularly for heat transfer in the core. As a result, the code is now felt to be very reliable for predicting reactor response during large-break loss-of-coolant accidents.

TRAC's applicability to different types of accidents, such as long-duration transients involving small breaks and multiple failures, was tested in the aftermath of Three Mile Island.** TRAC analyses of that accident requested by investigative groups are in good agreement with available plant data and provided a basis for estimates of core damage by Laboratory personnel. Recent work on the code has concentrated on improving numerical efficiency and modeling for accidents of this type.

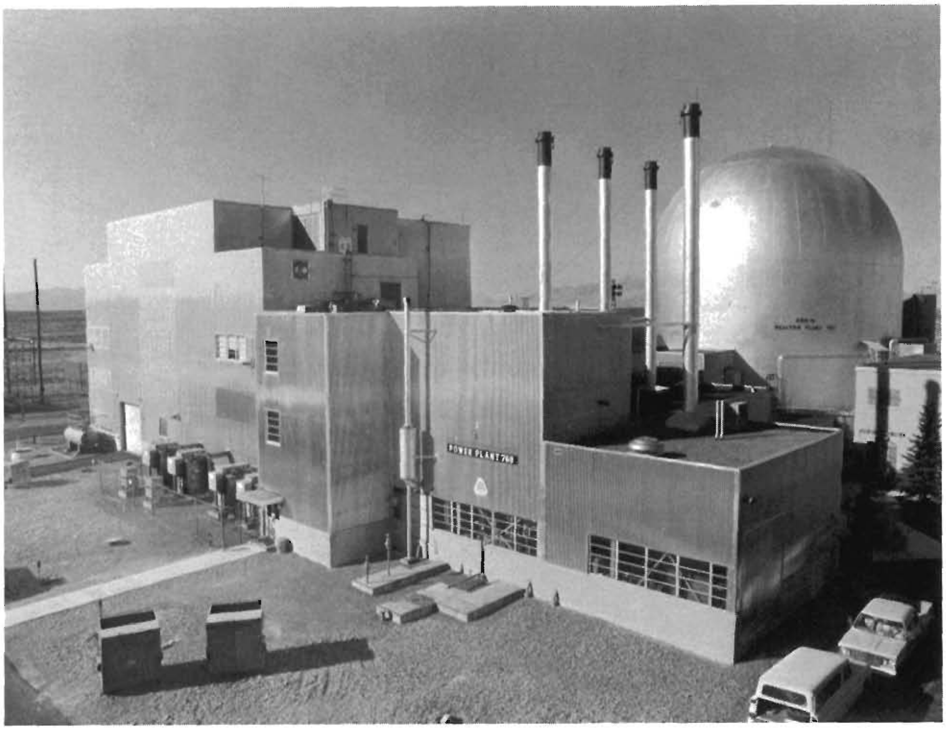

Fig. 13. The EBR-II reactor is located at Idaho National Engineering Laboratory near Idaho Falls, Idaho. This liquid-metal-cooled fast breeder reactor has operated successfully for over 15 years. The reactor produces up to 20 megawatts of electrical energy and has had an excellent history of reliable operation. A predecessor of EBR-II, a small reactor called EBR-I, produced the first nuclear-generated electricity in 1951. (Photo courtesy of Argonne National Laboratory.)

Fast Breeder Reactors

Light-water reactors, which run on thermal neutrons and fission of the fissile isotope uranium-235, utilize only a very small fraction of the energy potentially available from our uranium resources. Over 99% of natural uranium is uranium-238, a "fertile" isotope that can be converted into a fissile isotope, plutonium-239.

The fast breeder reactor is designed to carry out this nuclear alchemy. It not only produces power through a chain reaction based on fission of plutonium-239, but also uses the excess neutrons to convert uranium-238 into plutonium-239 through neutron absorption and subsequent beta decay:

$$^{238}\mathrm{U} + n \rightarrow {}^{239}\mathrm{U} \rightarrow {}^{239}\mathrm{Np} \rightarrow {}^{239}\mathrm{Pu} \ .$$

This conversion takes place in the reactor core, which contains both plutonium-239 and uranium-238, and in a blanket of uranium-238 that surrounds the core. To breed more fuel than it consumes, the breeder reactor must run on fast neutrons. Therefore, moderating materials, such as water, that slow down the fast neutrons created by fission are eliminated from the core region.

Fast breeder reactors can increase utilization of uranium resources by a factor of 50 over what can be achieved with light-water reactors. In fact, breeder reactors could supply all of our electrical energy needs for thousands of years.

*See "Accident Simulation with TRAC" in this issue.

**See "Three Mile Island and Multiple-Failure Accidents" in this issue.

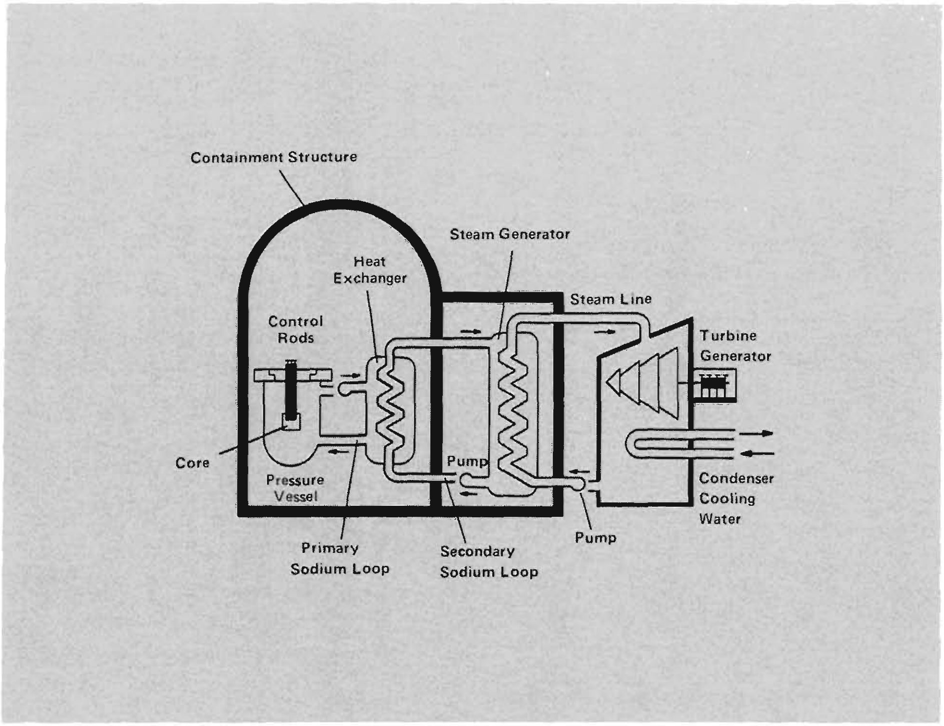

Fig. 14. A loop design for a liquid-metal-cooled fast breeder reactor showing the primary and secondary sodium cooling loops and the steam loop to the turbine-generators. The second sodium loop ensures that no radioactive sodium flows through the steam generators. The primary system is at near atmospheric pressure and therefore does not need a pressurizer. The reactor core contains more fissile fuel in a more compact configuration than does a light-water reactor.

Because of this high potential payoff, research on fast breeder reactors has been a high-priority effort in the United States for over 20 years. Interestingly, the first reactor-generated electricity came in 1951 from a small fast reactor prototype called the Experimental Breeder Reactor I (EBR-I). A second-generation reactor of this type, EBR-II, has successfully operated at Idaho National Engineering Laboratory for over 15 years (Fig. 13).

Liquid sodium is the primary coolant in liquid-metal-cooled fast breeder reactors, the most common design for fast breeder reactors. In one form, referred to as a loop design, the general component layout is similar to that in a pressurized-water reactor (Fig. 14). The proposed Clinch River Breeder Reactor is an example of this design. It has no pressurizer because the coolant is maintained at near atmospheric pressure, but it requires an extra set of heat exchangers to ensure that the sodium flowing through the steam generators is not radioactive. The steam generators must be very carefully designed, built, and maintained to minimize the chance for coolant leakage because sodium and water react violently on contact.

SAFETY ANALYSIS OF FAST BREEDER REACTORS. Liquid-metal-cooled fast breeder reactors have several safety advantages. The sodium coolant, which is at nearly atmospheric pressure, does not severely stress the primary system and would not be rapidly expelled from a break. Therefore, loss-of-coolant accidents are not a major concern. A complete loss of coolant can be made practically impossible by putting catch tanks around all major components. Emergency core-cooling systems are therefore not necessary. In addition, because the reactor operates at coolant temperatures well below the boiling point of sodium, transients involving departure from nucleate boiling are not a problem, provided the control systems operate correctly. Further, the sodium coolant has excellent capabilities for passive (without pumps) decay-heat removal when the reactor is scrammed.

Despite these apparent advantages, the breeder reactor has one major disadvantage. The core of a breeder, unlike that of a light-water reactor, is not in its most reactive configuration. If the control rods should fail to scram the reactor during certain potential accidents, some of the fuel may melt and reassemble in a configuration that would support a rapidly increasing fission rate. Fortunately, such energy-releasing excursions are inherently self-limiting. High temperatures and core expansion almost instantaneously cause sufficient nuclear feedbacks to reduce the fission rate. Nevertheless, a large amount of energy can be released in a very short time before these feedbacks take effect. Therefore, great care is taken to provide

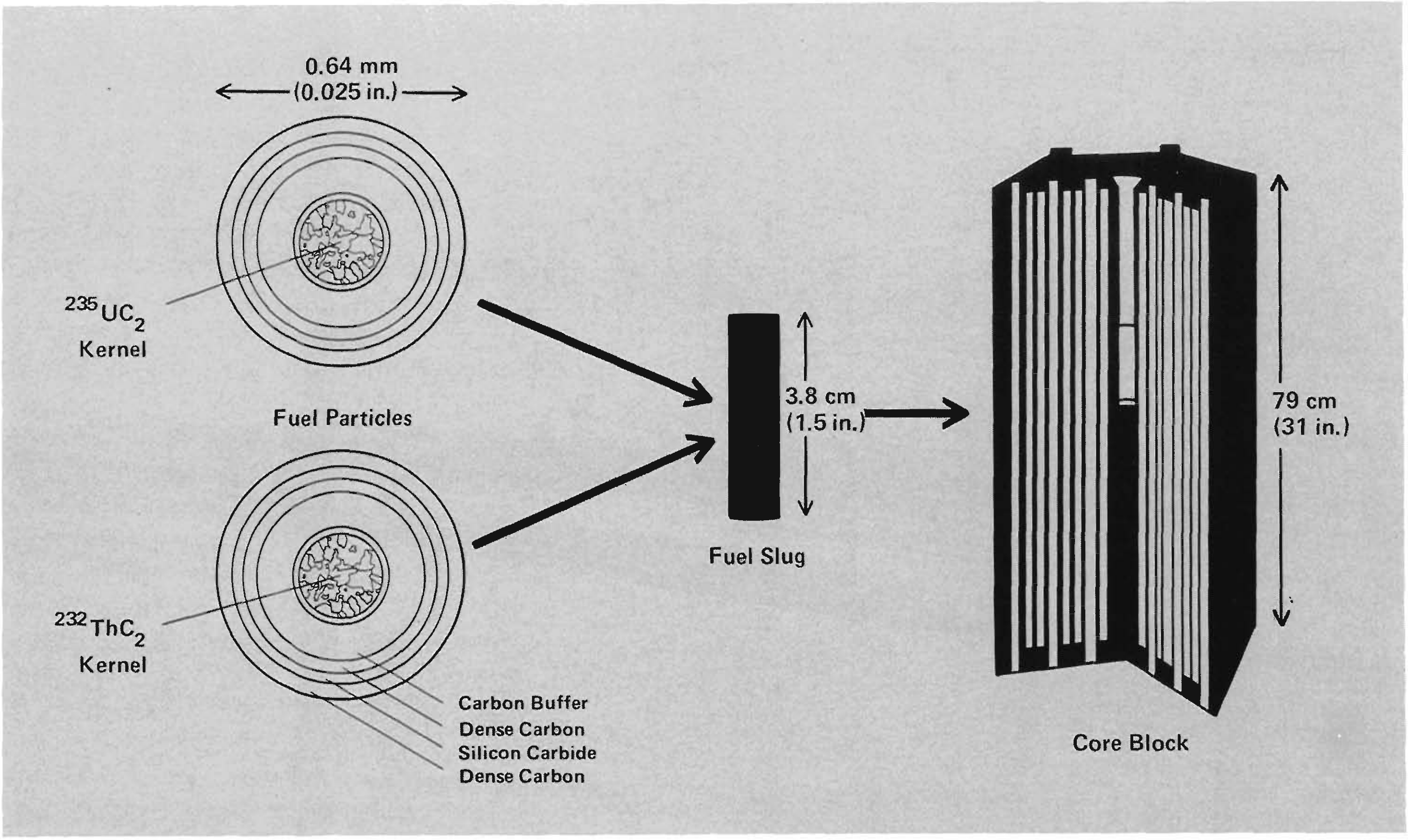

Fig. 15. Fuel for the high-temperature gas-cooled reactor is in the form of small particles containing a kernel of either fissile uranium-235 or fertile thorium-232, both as dicarbides. Typically, three barrier coatings plus an inner buffer zone encase the kernel and serve to contain the fission products. The particles are dispersed in a graphite matrix, which is formed into a fuel slug. The fuel slugs are inserted into holes drilled in a graphite core block; helium flows through other holes. The core contains seveal thousand core blocks, some of which can accommodate control rods.

diverse and redundant scram systems for breeder reactors.

Although a core-disruptive accident is extemely unlikely, it has received considerable attention as the worst possible accident—one that poses a threat to the containment. The Laboratory was asked to develop a computer code simulating this accident to determine its potential for damage. The result is SIMMER, a coupled neutronic-hydrodynamic computer code that is unique in being able to treat the complex interaction of solid, liquid, or vapor phases of fuel, steel cladding, and sodium coolant as they are affected by fission energy release.* The hydrodynamic treatment of interpenetrating materials and multiphase flow is based on methods developed at Los Alamos by Francis H. Harlow and his coworkers.

SIMMER analyses have been in good agreement with experiments involving isolated aspects of a simulated core-disruptive accident. Results for the accident as a whole indicate a much lower potential for damage than do earlier, more conservative analyses.

**See "Breeder Reactor Safety—Modeling the Impossible" in this issue.*

Gas-Cooled Reactors

Reactors that use a gas as the primary coolant have been under development for many years. Such reactors can operate at higher temperatures than water-cooled reactors because phase change (boiling) is not a constraint. The British have been particularly active in building gas-cooled reactors; the West Germans and Japanese also have a strong interest in this approach. Los Alamos developed considerable expertise on gas-cooled reactors through the Rover program, a program carried out be-

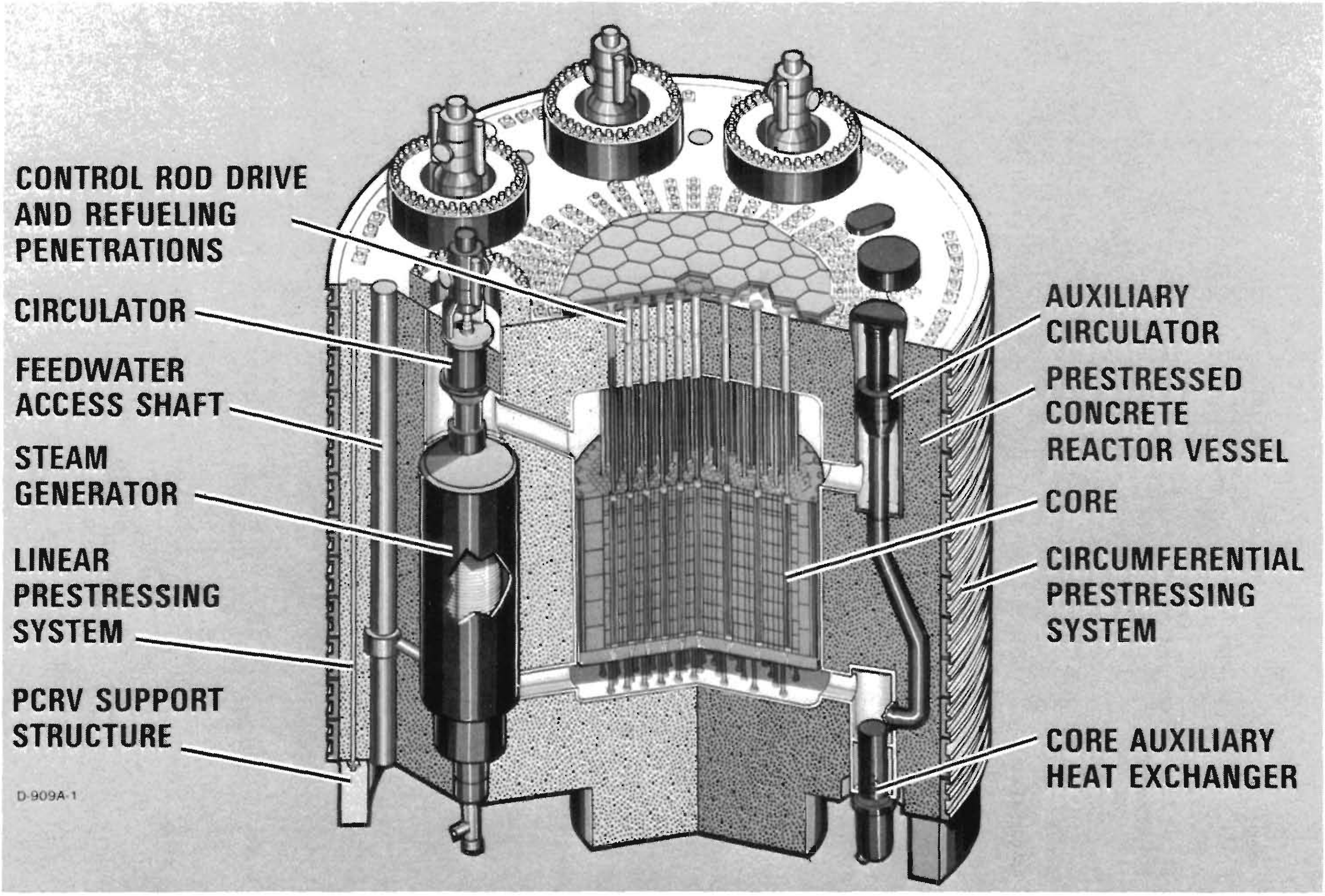

Fig. 16. A massive prestressed concrete vessel encloses the primary system of the Fort St. Vrain high-temperature gas-cooled reactor. Helium circulators force pressurized helium down through the core and up through steam generators. The auxiliary heat exchanger removes heat from the helium when the steam generators are out of service. (Diagram courtesy of General Atomic Company.)

tween 1955 and 1974 to develop a reactor-powered rocket engine. As part of this program, several gas-cooled reactors were developed and successfully ground-tested.

The current gas-cooled reactor program in the United States centers on the high-temperature gas-cooled reactor, a concept developed by General Atomic Company. The Fort St. Vrain reactor located near Denver, Colorado is the only commercial gas-cooled reactor in the United States. Although this type of reactor offers advantages in terms of efficiency and safety, it is a second-generation reactor technology that was caught in the nuclear power downturn before it could become well established commercially.

The core of a high-temperature gas-cooled reactor is very different from that of the reactors discussed above. The fuel (Fig. 15) is in the form of tiny beads. Special coatings around the beads contain the fission products generated during use. The beads are dispersed in a graphite binder and inserted into large graphite blocks. These blocks are locked together to form the core. The graphite also serves as the neutron moderator. The coolant is helium pressurized to as high as 72 bars in recent designs. A circulator forces the helium through thousands of holes drilled in the core blocks and through steam generators. Figure 16 shows a typical primary system and the monolith of prestressed concrete that encases the entire primary system. A network of axial and circumferential cables keeps the concrete vessel under constant compression.

The unique core design of the high-temperature gas-cooled reactor affords a degree of accident protection not possible in water-cooled reactors. Here the dispersed fuel produces a low energy density and the large amount of graphite provides an enormous heat sink. Even if the helium circulator is not operating, several hours worth of decay heat can be absorbed by the core before it heats to the point of damage. After a few hours of such heating, the fission products begin to diffuse from the fuel to the coolant channels, but slowly moving helium will transport them to colder regions of the primary system where most would be deposited. The graphite core can withstand extremely high temperature (about 3900 kelvin, or 6500°

Fahrenheit) before beginning to sublime rather than melt. Proponents of gas-cooled reactors describe them as more forgiving because they offer more time to take appropriate emergency measures than do light-water reactors.*

Los Alamos work on gas-cooled reactors included development of the Rover nuclear rocket engine based on an ultra-high-temperature reactor with a graphite core. Current gas-cooled reactor safety research at the Laboratory concentrates on investigation of structural dynamics and on analysis of possible accidents. The tool for accident analysis is the computer code CHAP, which resembles TRAC in its full-system analysis capabilities. Laboratory staff members also assist the Nuclear Regulatory Commission on safety issues related to the Fort St. Vrain reactor.

Safety Analysis at Los Alamos

We have emphasized the development of accident-simulation codes such as TRAC, SIMMER, and CHAP because Los Alamos is a leader in this field. These state-of-the art computer codes have made possible realistic analyses of accident consequences. We have built confidence in their predictive capabilities through extensive testing against experiments and are now applying these codes to actual safety problems. For example, one controversial issue facing the nuclear industry is whether or not the main coolant pumps should be turned off in the event of a small-break loss-of-coolant accident in a pressurized-water reactor. The results of our detailed calculations with TRAC will help provide the Nuclear Regulatory Commission with a technical basis for establishing operating guidelines.

See "The View from San Diego: Harold Agnew Speaks Out" in this issue.

Fig. 17. On the basis of the Laboratory's extensive research on respirators, Los Alamos personnel were requested to observe and evaluate the protection provided to workers involved in the cleanup at Three Mile Island. The Laboratory had tested most of the respirators in use there for effectiveness against inhalation of radionuclides, particularly iodine isotopes, and has developed techniques to assure their proper use. Here a respirator is being checked for leaks with a strong smelling solution known as banana oil. (Photo by Alan Hack.)

Another example will be the licensing of the Clinch River Breeder Reactor. This will involve calculating how strong the containment must be to withstand a core-disruptive accident. The SIMMER code will be used to help resolve this and other safety issues for the breeder reactor program.

Much of the code development work at Los Alamos is part of a broad program in reactor safety research sponsored by the Nuclear Regulatory Commission and carried out in large part by the national laboratories. Idaho National Engineering Laboratory performs most of the large-scale experiments, Sandia National Laboratories (Albuquerque) performs some experiments and a considerable amount of risk analysis, and Los Alamos leads in the development, verification, and application of advanced computer techniques. Other laboratories involved include Brookhaven National Laboratory, Argonne National Laboratory, Oak Ridge National Laboratory, and Battelle Memorial Institute's Columbus and Pacific Northwest Laboratories.

The Nuclear Regulatory Commission also relies on the national laboratories for technical assistance in reviewing license applications and investigating

TABLE I

REACTOR AND NUCLEAR FUEL CYCLE SAFETY RESEARCH PROGRAM AT LOS ALAMOS

Activity	Current Personnel Level	Lead Group(s)
RESEARCH FUNDED BY THE NUCLEAR REGULATORY COMMISSION		
Development, assessment, and application of the TRAC code for light-water reactors	33	Q-9, Q-7
Development, assessment, and application of the SIMMER code for liquid-metal-cooled fast breeder reactors	22	Q-7
Multinational (United States, West Germany, and Japan) reactor safety program	15	Q-8
Analytic and experimental studies of ventilation systems for nuclear facilities	9	WX-8
Nuclear safeguards studies	8	Q-4
Analytic and experimental studies of high-temperature gas-cooled reactors	7	Q-13, Q-9
Respirator studies	7	H-5
Development and application of codes for light-water reactor components	6	T-3
Analytic and experimental studies of structural margins for reactor containment buildings	6	Q-13
Studies of radionuclide transport in soil	5	LS-6
Risk and statistical analysis	3	S-DO
RESEARCH FUNDED BY THE DEPARTMENT OF ENERGY		
Liquid-metal-cooled fast breeder reactor safety studies	7	Q-7
Application of PINEX (pinhole experiment) imaging system to liquid-metal-cooled fast breeder reactor safety experiments	4	P-15
Statistical analysis of light-water reactor component failures	1	S-DO
TECHNICAL ASSISTANCE TO THE NUCLEAR REGULATORY COMMISSION IN ITS REACTOR LICENSING ACTIVITIES		
Identification of vital areas in nuclear power plants	6	WX-8
Reactor containment building analysis	5	Q-7, T-1, Q-13
Audits of small-break loss-of-coolant accident analyses	3	Q-7
Seismic reviews of reactor sites	2	G-2
Miscellaneous support	5	Q-7, WX-8

specific safety issues. The Commission's safety requirements summarized in Federal Regulation 10CFR50 serve as the basis for evaluating plant designs. All power reactors, research reactors, and fuel-cycle facilities in the private sector are covered by this regulation.

To comply with 10CFR50, a license applicant must submit documents showing that the proposed facility is safe and will not adversely affect the health of the public. These documents include complete descriptions of the reactor, the auxiliary systems, and the site, as well as detailed safety analyses.

Los Alamos has developed multidisciplinary teams to help the Commission in all phases of this technical review. These teams include structural, electrical, nuclear, and mechanical engineers, seismologists, and experts on radiation and its health effects.

Associated with these safety reviews, Los Alamos performs research and testing in cooperation with New Mexico State University to help establish standards for plant ventilation systems and reactor containment structures. The purpose of these efforts is to ensure the confinement of radioactive materials during all accidents, including those caused by fires, explosions, and tornadoes. Experimental facilities at both Los Alamos and the University are used in this research.

An outgrowth of this technical assistance work is our direct involvement in assessing the physical security plans at commercial nuclear power plants.* These assessments have included analyses of accident sequences that might be initiated by sabotage.

The Laboratory has other responsibilities in reactor safety, some of them rather different from those mentioned above. For example, our Industrial Hygiene Group conducts research on respirators for protecting workers from inhaled radionuclides. The expertise developed in this field has been called upon in the cleanup at Three Mile Island (Fig. 17).

Table I summarizes the Laboratory's research and technical assistance activities in reactor and nuclear fuel-cycle safety.

Conclusion

Our broad involvement in safety analysis has brought us in direct contact with the public, the nuclear industry, and the government regulatory agencies. We are asked many difficult questions about safety and invariably the correct answers are not simple. Careful technical analysis is essential to any safety evaluation. By and large our work on worst-case accidents has shown that nuclear power plants have large margins to protect against release of radioactive materials. Now we are applying our sophisticated analysis tools to model the consequences of multiple equipment failures and human intervention in less severe situations. The purpose is to give the operators effective strategies for minimizing the effects of system failures. We believe that the predictive capabilities we have developed over the last decade will help ensure the continued safe operation of our nation's nuclear power plants. ■

See "Keeping Reactors Safe from Sabotage" in this issue.

AUTHORS

Michael G. Stevenson is Deputy Leader of the Laboratory's Energy Division, whose activities center on reactor safety, nuclear safeguards, and energy technology. He is a recognized expert in the area of nuclear reactor safety. He received a Bachelor of Engineering Science and, in 1968, a Ph.D. in mechanical engineering from the University of Texas at Austin. In his 13 years of experience in fission reactor safety research at Babcock & Wilcox, Argonne National Laboratory, and, since 1974, at Los Alamos, he has worked on light-water-reactors, gas-cooled reactors, and liquid-metal fast breeder reactors. He is interested in all aspects of the nuclear fuel cycle, and during 1978 and 1979 was the United States representative to the International Nuclear Fuel Cycle Evaluation as a member of its subgroup on environmental, safeguards, and management aspects of fast breeders.

James F. Jackson is internationally recognized in the area of nuclear reactor safety analysis. He earned his Ph.D. in engineering from the University of California at Los Angeles in 1969, a Master of Science in nuclear engineering in 1962 from the Massachusetts Institute of Technology, and a Bachelor of Science in mechanical engineering from the University of Utah in 1961. As a senior research engineer at Atomics International, he worked on the design and safety evaluation of the SNAP-10A reactor that was successfully tested in orbital flight. He then moved to Argonne National Laboratory and became deeply involved in advanced computer methods for safety analysis of liquid-metal fast breeder reactors. After spending two years teaching nuclear engineering at Brigham Young University and consulting in the area of reactor safety, he joined the Los Alamos staff in 1976. He served the Energy Division in various positions of leadership and in 1980 was appointed the Deputy Associate Director for Nuclear Regulatory Commission Programs. In March 1981 he assumed his current position, Leader of the Energy Division. He is a member of the American Nuclear Society and has served on the Executive Committee of its Nuclear Reactor Safety Division. This spring he was the recipient of a Distinguished Performance Award from the Laboratory for his contributions to its reactor safety research effort.

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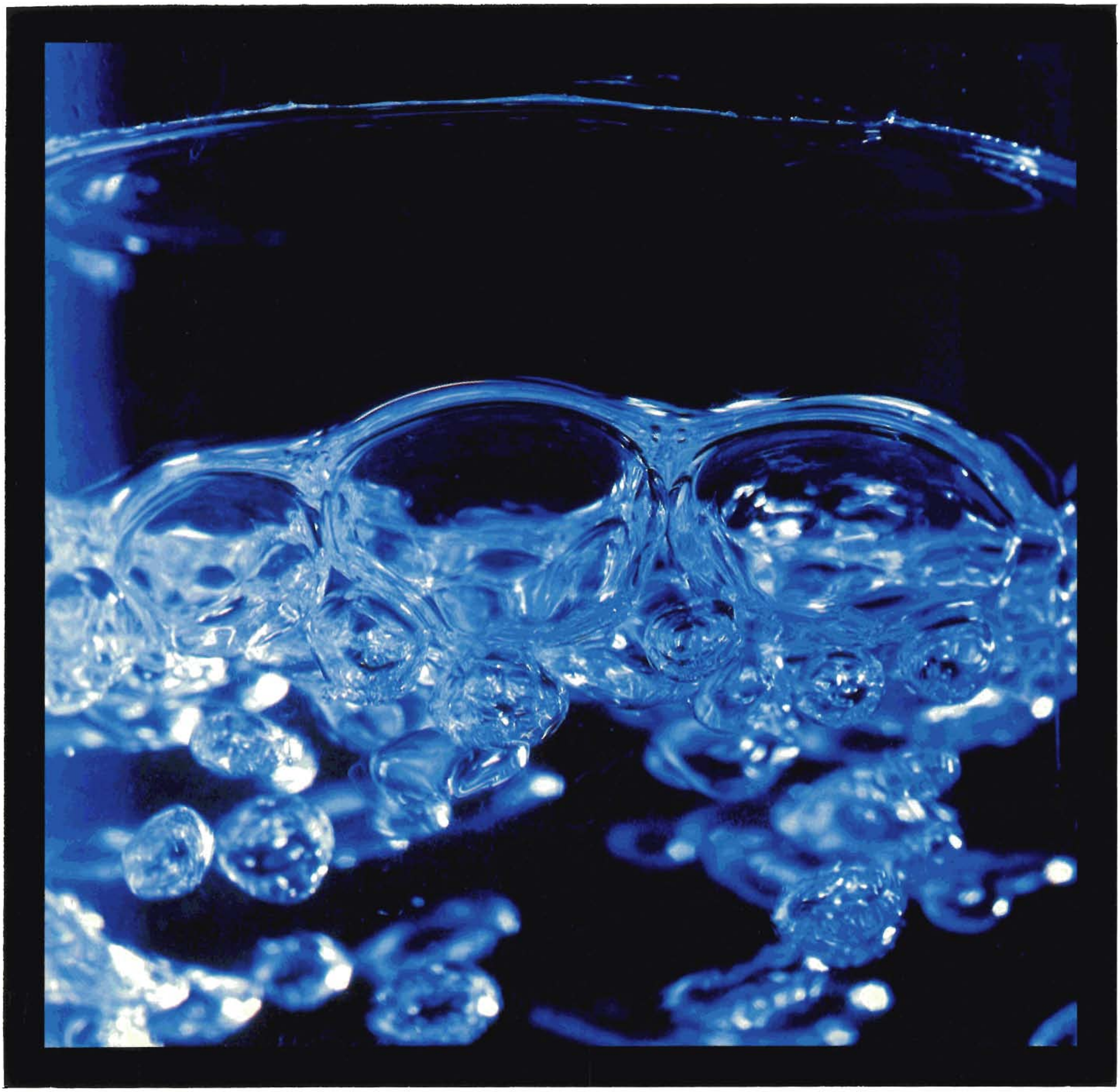
two phase flow

by Dennis R. Liles

Complex steam-water flows occur in a pot of boiling water just as they do in a pressurized-water reactor during a loss-of-coolant accident. Successful methods for analyzing these two-phase flows were first developed at Los Alamos National Laboratory.



Many natural and manmade situations provide examples of two-phase flow—bubbles rising in a carbonated drink, raindrops falling through the air, gasoline and air reacting in an automobile engine, water and steam circulating through a nuclear reactor. Common to all two-phase flows is the existence of discernible interfaces, or boundaries, that separate one phase from the other. Whether the flow involves two immiscible liquids, a liquid and a solid, a liquid and a vapor, or a solid and a vapor, the interfacial topology constantly changes as the phases interact, exchanging energy, momentum, and often mass. These interactions and changes in interfacial topology are the most difficult aspect of two-phase flow to model. Although little progress has been made in describing the detailed dynamics from first principles, macroscopic properties of two-phase flows can be determined satisfactorily from approximate models. Such models are essential for the safe and economic operation of a host of commercial systems—power generation, heating and cooling, material processing, and transport systems, to name a few.

Here we focus attention on the steam-water flows that may occur during transients in pressurized-water reactors, but the methods presented are applicable to liquid-solid and liquid-liquid flows as well. The Laboratory has been a leader in the development of sophisticated numerical techniques for analysis of multiphase flows and in the construction of computer codes based on these techniques. Applications of these codes are described in the four articles that follow. In this article, we discuss the basic principles incorporated in models for liquid-vapor flows and illustrate the numerical techniques for solving the resulting equations. The level of sophistication described here is typical of that in TRAC, the large systems code developed, at the request of the Nuclear Regulatory Commission, by Los Alamos for light-water reactor safety analysis.

Flow Regimes

Two-phase flows exhibit various flow regimes, or flow patterns, depending on the relative concentration of the two phases and the flow rate. A simple but generally adequate set of descriptive phrases for most of the important liquid-vapor flow regimes consists of bubble flow, slug flow, churn flow, annular flow, and droplet flow.

Bubble flow describes the flow of distinct, roughly spherical vapor regions surrounded by continuous liquid. The bubble diameter is generally considerably smaller than that of the container through which they flow. Bubble flow usually occurs at low vapor concentrations.

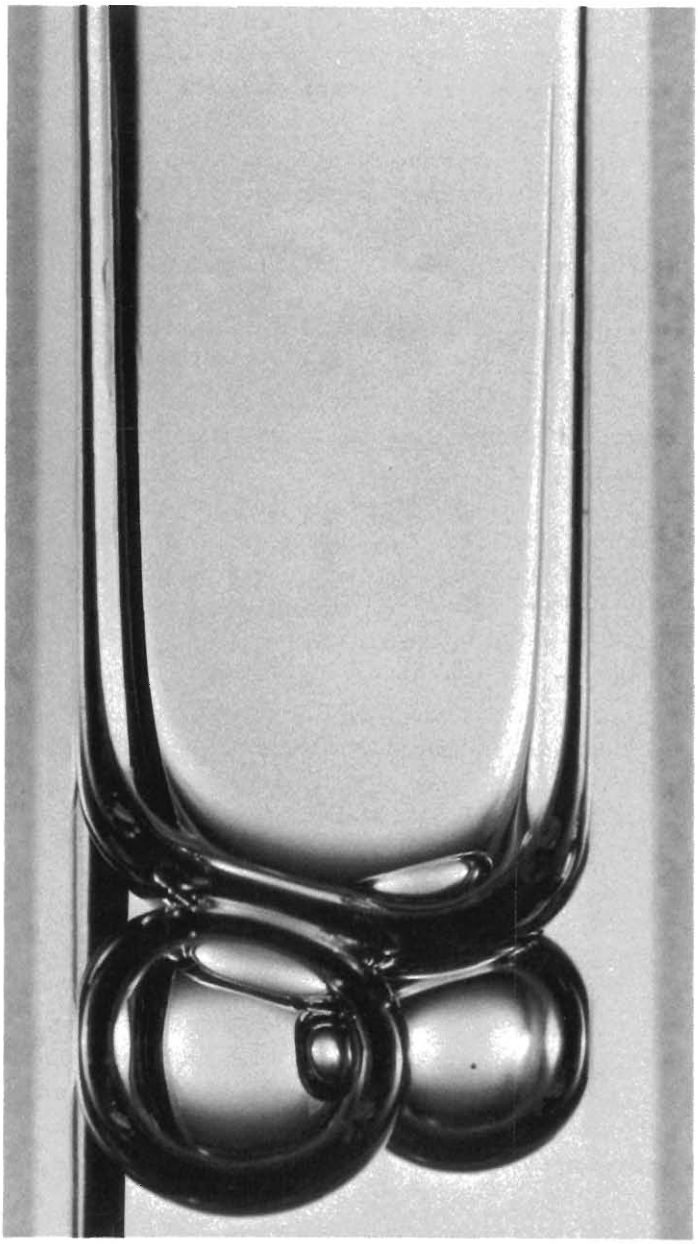


Fig. 1. Photograph of slug flow.

If the vapor and liquid are flowing through a pipe, bubbles may coalesce into long vapor regions that have almost the same diameter as the pipe (Fig. 1). This is called slug flow.

At moderate to high flow velocities and roughly equal concentrations of vapor and liquid, the flow pattern is often very irregular and chaotic. If the flow contains no distinct entities with spherical or, in a pipe, cylindrical symmetry, it is said to be churn flow.

At high vapor concentration, the liquid may exist as a thin film wetting the pipe wall (annular flow) or as small, roughly spherical droplets in the vapor stream (droplet flow). If both a thin film and droplets exist, the flow is described as annular-droplet flow.

All these regimes can be exhibited by liquid flowing vertically upward through a heated tube (Fig. 2). Each regime requires somewhat different modeling because the dominant interactions between liquid and vapor change their character from one regime to another.

Everyone has observed some of these flow patterns in the home. For example, as a pot of water is heated, small bubbles form on the hot bottom surface. These grow, detach, and rise to the surface, driven by their buoyancy and by liquid convection. When the bubbles reach the surface, they break and send tiny droplets upward in a visible mist. Interaction of the small waves resulting from the bubbles' collapse produces larger droplets. Initially, these accelerate upward from the surface but are too large to be carried very far by the rising steam, so they fall and splash back onto the liquid. Even this mundane situation is chaotic and complicated, and its simulation presents interesting problems.

Anyone who has attempted to drink liquid from an inverted pop bottle has experienced slug flow. The liquid exits as a series of chunks rather than a smooth stream, and the air that replaces the liquid enters the bottle as a series of vapor slugs. The same general formulation that describes bubbles rising in a pot can be used to describe the flow of liquid from the inverted bottle or the complex steam-water flows in a pressurized-water reactor.

Steam-Water Flows in Pressurized-Water Reactors

During normal operation of a pressurized-water reactor, water in the primary cooling system is at a pressure of about 150 bars (about 150 atmospheres) and a temperature of about 590 kelvin (about 600°F). The water, circulated by large centrifugal pumps, flows into the reactor vessel, down an annulus, up through the core where it is heated by the fuel rods, into an upper plenum, and out of the vessel. The hot

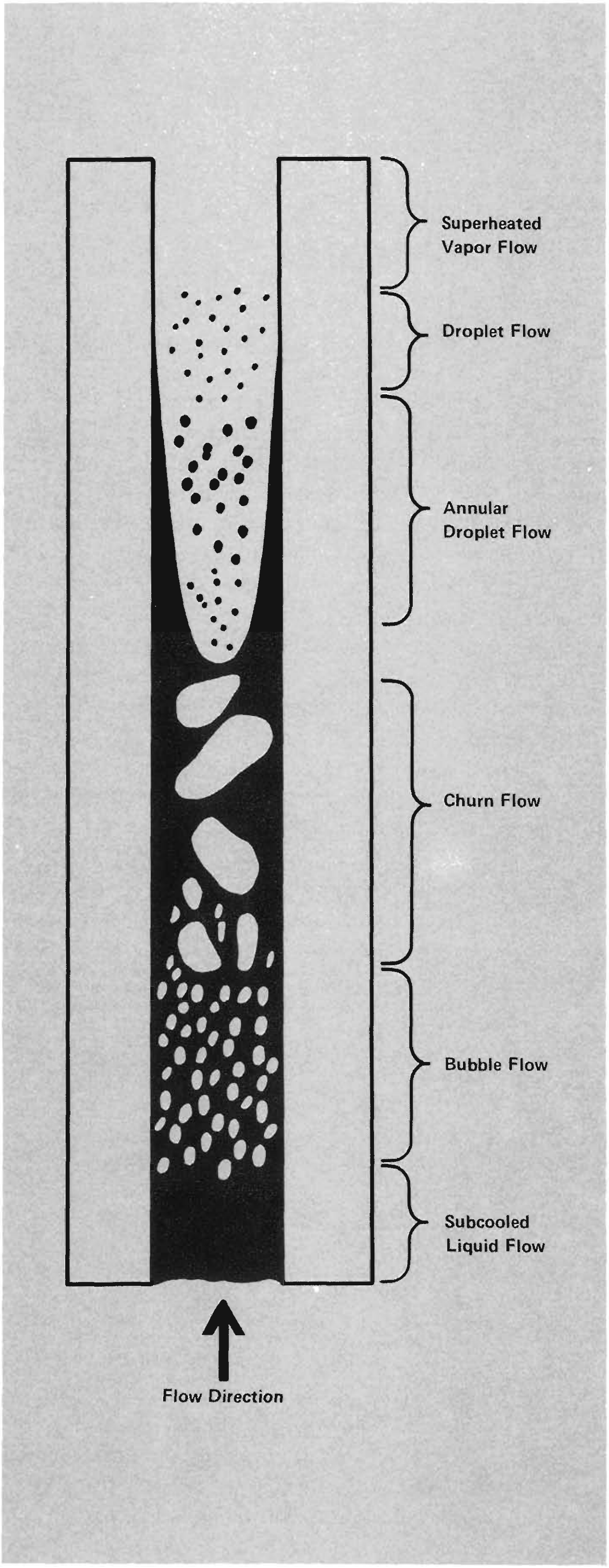


Fig. 2. Flow regimes exhibited by water flowing vertically upward through a heated tube at moderate to high flow velocities.

water, still liquid, flows from the reactor vessel through a heat exchanger, called the steam generator, where the energy is removed from the primary system. The cooler liquid returns to the pump and the process continues. The water on the secondary side of the steam generator is at a lower pressure and quickly boils. The steam powers a turbine that drives a generator, and is then recovered from the turbine, condensed in another heat exchanger, and returned to the secondary system pumps.

Now suppose that a pipe breaks in the primary system. The pressure drops and the superheated water flashes to steam. As the pressure drops further, emergency core-cooling systems are activated to prevent overheating of the core. These systems inject cold water into the pipes connected to the reactor vessel. Both vaporization and condensation may occur simultaneously in different reigons of the primary system and produce complex, turbulent steam-water flows. To follow the evolution of these flows and predict their effectiveness in cooling the core requires detailed models of the two-phase flow.

The Two-Fluid Model

Analysis of two-phase flow begins with the most general principles governing the behavior of all matter, namely, conservation of mass, momentum, and energy. These principles can be expressed mathematically at every point in space and time by local, instantaneous field equations. However, exact solution of these equations is almost impossible and very expensive, requiring the tracking of many convoluted liquid-vapor interfaces that change continuously in time. Instead, the usual procedure is to average the local, instantaneous equations in either time or space, or both. Although we lose information in the process, the resulting equations yield accurate solutions to a wide variety of practical problems so long as the averaged variables bear some resemblance to the actual situation, that is, so long as the flow is not too chaotic.

During the averaging, the two phases may be treated together to obtain averaged variables for a two-phase mixture; alternatively, treating each phase separately, we obtain averaged variables for both phases. The latter procedure yields the two-fluid model, which is a bit more general and useful. (The mixture model can be derived from the two-fluid model.)

A usual two-fluid model consists of six field equations: averaged mass, momentum, and energy equations for the liquid and another set of three for the vapor. For example, integrating across the cross section of our heated tube (Fig. 2) at some particular time, through regions with liquid and regions with vapor, we obtain area-averaged conservation equations for the liquid and the vapor. Or, integration over a small volume element provides volume-averaged equations. We could also integrate over a period of time at some particular location in the tube to obtain time-averaged conservation equations. Finally, additional variables are introduced into the averaged conservation equations, namely, the volume (or area) fraction of the vapor α_1 and of the liquid α_2 for a given region. Because the flowing material is either vapor or liquid, α_1 and α_2 are not independent. Rather $\alpha_1 + \alpha_2 = 1$.

Other procedures (for example, Boltzmann statistical averaging) may be followed to obtain usable field equations, but these are the most common. Fortunately, all the averaging techniques produce effectively identical sets of equations, at least one-dimensional equations.

The field equations are usually derived by assuming that the interface separating the phases has zero thickness and zero mass, and hence cannot store momentum or kinetic and thermal energy. To complete the field equations, mass, momentum, and energy fluxes of one phase must be connected across the interface to the corresponding fluxes of the other phase. With suitable simplifications, these connections are usually effected with "jump conditions."

We will illustrate application of the two-fluid model to vapor-liquid flow with the field equations for conservation of mass and the appropriate jump condition. For flow of a single phase in the absence of sources and sinks, conservation of mass is expressed as

$$\frac{\partial \rho}{\partial t} + \nabla \cdot \rho \vec{v} = 0 \ ,$$

where ρ is the density of the fluid and $\vec{v}$ is its velocity. For the case at hand, we need two mass-conservation equations, one for each phase, and must include the possibility of vaporization and condensation at the rates Γ_1 and Γ_2, respectively. We obtain the following mass-balance equations for vapor and liquid. (Averaging symbols have been omitted for simplicity.)

$$\frac{\partial (\alpha_1 \rho_1)}{\partial t} + \nabla \cdot (\alpha_1 \rho_1 \vec{v}_1) = \Gamma_1 \tag{1}$$

and

$$\frac{\partial\,(\alpha_2\rho_2)}{\partial t} + \nabla\cdot(\alpha_2\rho_2\,\vec{v}_2) = \Gamma_2\ . \qquad (2)$$

Conservation of mass implies that the jump condition at the interface is

$$\Gamma_1 + \Gamma_2 = 0\ .$$

That is, production of vapor at the interface depletes the liquid phase by an equal amount.

The field equations based on conservation of energy and momentum, although similar, are more complicated and are often formulated with additional simplifying assumptions. For example, we often ignore turbulent stresses in the momentum equation and turbulent work terms in the energy equation.

Constitutive Relations

The field equations are an expression only of conservation principles; they describe neither the thermodynamic properties of the materials involved nor the interactions between the phases and between each phase and the medium in which the flow occurs. Completion of the analysis requires "constitutive relations" that describe these properties and interactions.

For the steam-water flows that are of interest here, the constitutive relations that are the most difficult to specify properly are those describing the interactions between the phases. Consider, for example, the averaged equation for conservation of mass of the vapor phase (Eq. 1). Expressed in words, this equation simply states that, within a volume element, the temporal change in the vapor mass equals the rate of vapor production minus the exiting vapor flux.

However, this equation cannot be applied to a real problem until we have a constitutive relation that specifies the rate of vapor production.

A number of basic models have been used to determine this variable. Early vapor-liquid studies were often based on a thermodynamic equilibrium model. This model includes the assumption that when two phases coexist, both must be at the saturation temperature. Thermodynamic equilibrium is maintained in this model by balancing pressure changes with sufficient vaporization or condensation.

Although adequate to describe many situations, this model fails when the effects of thermodynamic disequilibrium are important. Such is the case, for example, in a reactor core during accident conditions. Droplets of water at temperatures close to saturation may be entrained by steam at a temperature much higher than saturation. To evaluate properly the cooling effects of the steam-droplet mixture on the fuel rods, the temperature of the droplets and of the steam must be considered separately.

INTERFACIAL MASS AND ENERGY EXCHANGE. In our studies of transient reactor behavior, we use a simple non-equilibrium phase-change model based on a thermal-energy jump condition at the vapor-liquid interface. At a region in space where both phases exist, we specify an energy balance between the phases at the interface (Fig. 3). Because we assume that the interface cannot store thermal energy, the net energy transferred to the interface by vapor and liquid must be used up by vaporization (or condensation). Thus the rate of vapor production Γ_1 is given by

$$\frac{q_1 + q_2}{\triangle H} = \Gamma_1\ . \qquad (3)$$

where q_1 and q_2 are the rates of heat transfer to the interface

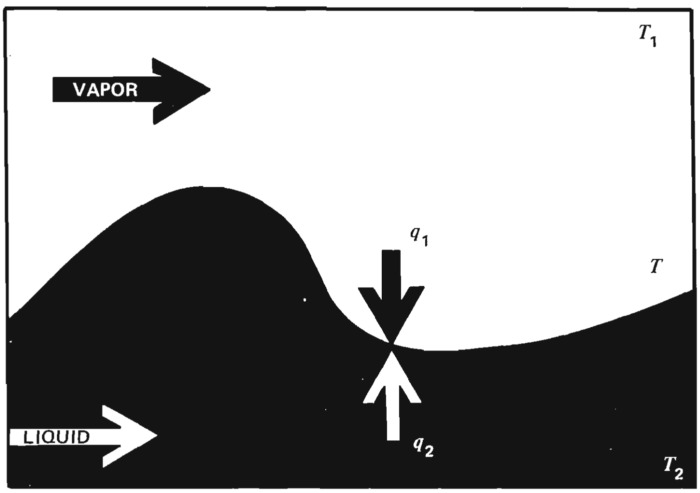


Fig. 3. Mass and thermal energy are exchanged between vapor and liquid through a massless interface. Because the interface cannot store thermal energy, the net energy transferred to the interface, $q_1 + q_2$*, must result in vaporization or condensation.*

from the vapor and the liquid, respectively, and ΔH is the enthalpy difference between the phases. If the interface is assumed to be at the local saturation temperature T_{sat}, then $\Delta H = L$, the heat of vaporization of the material at the local pressure.

Expressions for the interfacial heat transfers are obtained by assuming that each phase has an average temperature, denoted by T_1 and T_2, and by applying Newton's law of cooling.

$$q_1 = h_1A(T_1 - T_{sat})$$

and

$$q_2 = h_2A(T_2 - T_{sat}) \ .$$

The proportionality constants h_1 and h_2 are the heat-transfer coefficients between the interface and vapor and between the interface and liquid, respectively, and A is the interfacial area. Substituting these expressions into Eq. 3, we obtain Γ_1 as a function of h_1, h_2, and A.

$$\Gamma_1 = \frac{A}{L}\left[h_1(T_1 - T_{sat}) + h_2(T_2 - T_{sat})\right] \ . \tag{4}$$

The temperatures T_1 and T_2 can be calculated from the coupled field equations and may be regarded as known. However, the interfacial heat-transfer coefficients and the interfacial area depend on the interfacial topology, which is not specified in our averaged two-fluid model.

We usually obtain values for h_1, h_2, and A by first determining the local flow regime from a steady-state flow-regime map. Such a map relates observed flow regimes to local flow conditions, that is, to volume fraction of one or the other phase and to flow velocities of both phases. (These variables are available from the field equations.) Figure 4 shows a particularly simple flow-regime map based on observations of upward air-water flow in a vertical pipe. Having determined the local flow regime, we use empirical correlations appropriate to that regime to obtain values for h_1, h_2, and A. Although this technique cannot be fully justified from first principles, it is relatively simple and often supplies reasonable answers to complex problems.

Sometimes further information may be needed to use the customary empirical correlations for h_1, h_2, and A. For example, the flow-regime map may specify droplet flow, but a

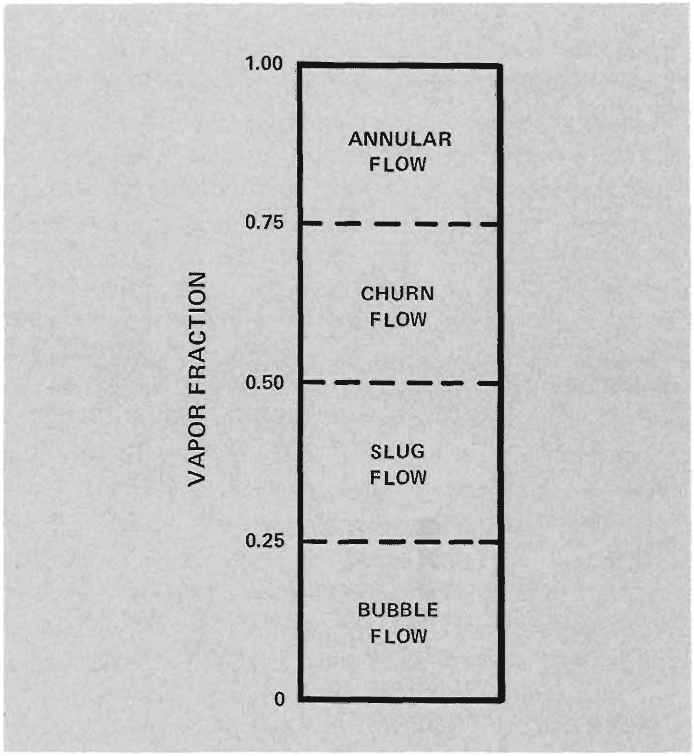


Fig. 4. Observed flow-regime map for upward air-water flow in a vertical pipe. The flow regime is independent of flow velocity and depends only on the vapor fraction. As the vapor fraction increases (and the number and size of the bubbles increase), collisions between the bubbles become more frequent, and they coalesce into slugs. At higher vapor fraction, vapor slugs cannot exist, and churn flow sets in. Finally, an annular-droplet flow occurs at very high vapor fraction.

mean droplet size is required. A local approximation based on a Weber number criterion is often used to specify an average droplet diameter d. This criterion is an expression of the idea that, for droplet flow to exist, disruptive forces (forces due to relative motion of droplets and vapor that tend to break up the droplets) and restoring forces (due to surface tension σ) must be in a certain ratio. Expressed mathematically,

$$\frac{\rho_1(v_1 - v_2)^2 d_{max}}{\sigma} = We \ ,$$

where d_{max} is the maximum droplet size and We, the Weber number, is some constant. We use $d \approx 0.5\ d_{max}$.

The Weber number criterion does not take into account the existence of a spectrum of droplet sizes, velocities, and cooling rates; in addition, it sometimes predicts nonphysical results. For example, consider subsonic droplet flow in a convergent-divergent nozzle. Applied to large drops as they enter the convergent section, the Weber number criterion gives a reasonably accurate estimate of their largest size as they break up.

However, further downstream in the region of slower flow the same criterion predicts coalescence that does not in fact occur (Fig. 5).

Simplifications similar to those delineated above are included in most current computer codes. Fortunately, for many problems of interest, accuracy of the interfacial terms need only be sufficient to provide reasonable overall results. However, work is progressing on replacing some of these approximations with additional differential equations for a characteristic length (or area) field to be convected around with the flow. These equations will provide a better history of droplet evolution and more realistic estimates of the interfacial interactions.

INTERFACIAL MOMENTUM EXCHANGE. We have discussed in some detail the development of constitutive relations that describe the interfacial exchange of mass (by the mechanism of phase change) and its relationship to the interfacial exchange of energy. Another important interfacial interaction that must be taken into account is exchange of momentum between the two phases. This exchange arises because, in general, the two phases do not travel at the same velocity. (Witness the upward flow of steam bubbles in a pot of heated water or of carbon dioxide bubbles in a newly opened pop bottle.) A full description of the interfacial momentum transfer requires consideration of various phenomena, including, among others, "added-mass" effects, Bassett forces, steady-state drag forces, and phase-change thrust effects. However, the customary procedure is to consider only the last two, which are the local forces that dominate most problems. Both are dependent on the local flow regime, and again, flow-regime maps and empirical correlations are invoked.

A source of error in most calculations should be pointed out. Averaging operators, which have not here been indicated explicitly, can be important in formulating models because the averaged equations include many quantities that are averages of products. But in most calculations, it is assumed, for example, that $\langle \rho v^2 \rangle = \langle \rho \rangle \langle v \rangle \langle v \rangle$, where $\langle$ and $\rangle$ denote a spatial average. This assumption is strictly valid only if the density and velocity are constant across the region where the averaging is effected. (Such errors can be corrected for if information about the density and velocity profiles is available.) We raise the point here because the error so introduced is larger for momentum fluxes (ρv^2 terms) than for mass fluxes (ρv terms).

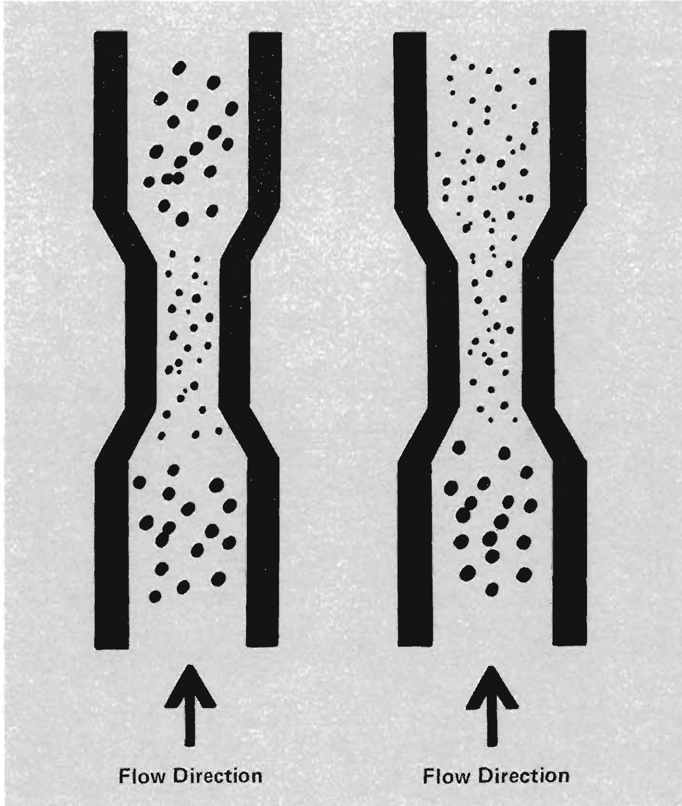


Fig. 5. A comparison of predicted and actual subsonic droplet flow through a convergent-divergent nozzle. (a) The Weber number criterion predicts that large drops break up into small drops in the convergent section and coalesce into large drops in the divergent section. (b) In reality, the small drops do not coalesce in the divergent section.

INTERACTIONS WITH CONTAINING MEDIUM. The interactions between each phase and the medium through which they flow (such as pipe walls and structures within a reactor vessel) are another set of necessary constitutive relations. Wall shear and wall heat transfer must be modeled with some accuracy to obtain realistic analyses of transient reactor response. Particular attention must be paid to modeling the extreme variation (by orders of magnitude) of heat transfer from the fuel rods as local flow conditions change. Correlating procedures using Newton's law of cooling are customary, but the resulting functions that specify the heat-transfer coefficients to the liquid and vapor are complicated and not always well supported by experimental data.

Numerical Solution Techniques

Even the simplified models for two-phase flow described above are fairly complicated. The two-fluid model includes six

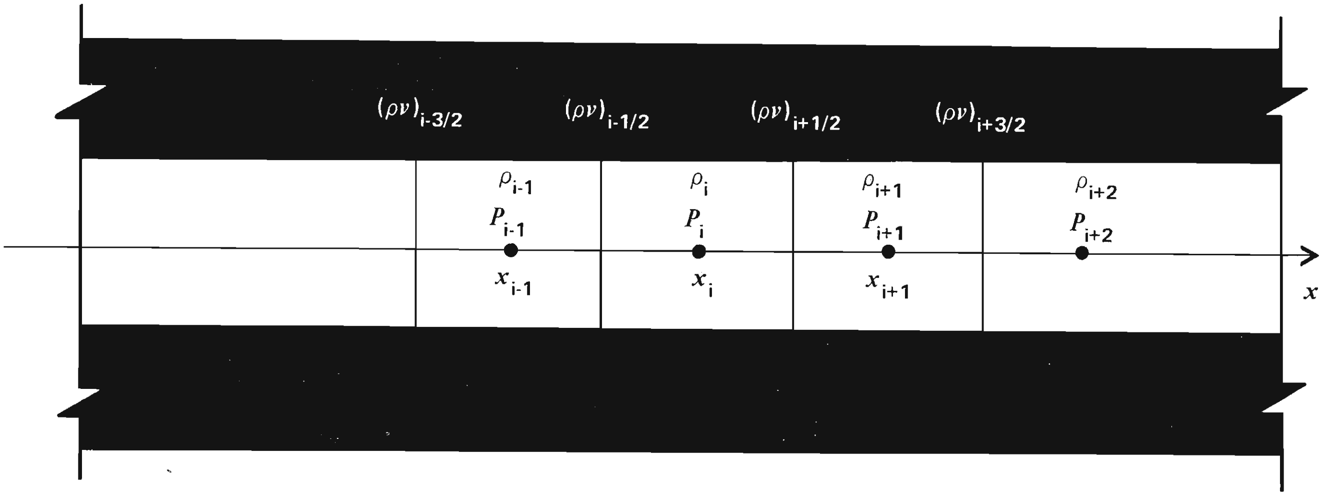


Fig. 6. To apply the method of finite differences to one-dimensional flow, the x-axis is divided into equal intervals with midpoints denoted by x_i. The variables ρ, P, and ρv are defined at the indicated locations.

coupled partial differential equations and numerous thermodynamic and constitutive equations. Solutions for specific problems are obtained by using numerical techniques and high-speed computers. The Laboratory's greatest contribution to analysis of two-phase flow is the development of numerical solution techniques and large-scale computer codes. Francis H. Harlow and his associates were among the first to compute these flows successfully with a two-fluid model.

The partial differential equations that represent the conservation laws cannot be solved directly with a (digital) computer. Instead, these equations must be approximated by algebraic equations. We will use the method of finite differences to solve a set of equations describing the flow of a single phase through a pipe. We assume that the flow can be described in sufficient detail in one dimension, along the pipe axis. The set consists of equations for conservation of mass and momentum and a thermodynamic equation of state.

$$\frac{\partial \rho}{\partial t} + \frac{\partial (\rho v)}{\partial x} = 0 \ , \tag{5}$$

$$\frac{\partial (\rho v)}{\partial t} + \frac{\partial (\rho v^2)}{\partial x} = -\frac{\partial P}{\partial x} \ , \tag{6}$$

and

$$\rho = \rho(P) \ ,$$

where ρ is the microscopic density, v is the velocity, and $\rho(P)$ is some (known) function of the pressure P.

For convenience, we divide the distance along the x-axis into equal finite intervals, or cells, of length Δx and denote the midpoints by x_i. The thermodynamic variables ρ and P are defined at the cell midpoints and the mass flux ρv at the cell edges (Fig. 6). We also divide the time coordinate into equal intervals of duration Δt with endpoints denoted by t_j. Superscripts and subscripts on the dependent variables indicate, respectively, time and location.

The temporal term of our mass-conservation equation (Eq. 5) may be approximated by

$$\frac{\partial \rho}{\partial t} = \frac{\rho_i^{j+1} - \rho_i^j}{\Delta t}$$

and the spatial term by

$$\frac{\partial (\rho v)}{\partial x} = \frac{(\rho v)_{i+1/2} - (\rho v)_{i-1/2}}{\Delta x} \ .$$

(Note that we have not yet specified the times for the spatial term. We shall address this issue below.) Our approximation for Eq. 5 in the cell bounded by $x_{i-1/2}$ and $x_{i+1/2}$ is thus given by

$$\frac{\rho_i^{j+1} - \rho_i^j}{\Delta t} = -\frac{(\rho v)_{i+1/2} - (\rho v)_{i-1/2}}{\Delta x} \ . \tag{8}$$

We could approximate our momentum-conservation equation (Eq. 6) over the same cell, obtaining

$$\frac{(\rho v)_i^{j+1} - (\rho v)_i^j}{\Delta t} + \frac{(\rho v^2)_{i+1/2} - (\rho v^2)_{i-1/2}}{\Delta x} = -\frac{P_{i+1/2} - P_{i-1/2}}{\Delta x} \ . \tag{9}$$

We choose instead to approximate it over the cell bounded by x_i and x_{i+1}, and obtain

$$\frac{(\rho v)_{i+1/2}^{j+1} - (\rho v)_{i+1/2}^j}{\Delta t} + \frac{(\rho v^2)_{i+1} - (\rho v^2)_i}{\Delta x} = -\frac{P_{i+1} - P_i}{\Delta x} \ . \tag{10}$$

(Again, we have not yet specified the times for the spatial terms.)

There are two reasons for choosing to "stagger" the mass and momentum cells. First, Eq. 9 specifies pressures at the cell edges rather than at the cell centers where we have defined them. Solving this problem involves use of pressures spanning three cells (P_{i+1} and P_{i-1}). In contrast, the pressures specified in Eq. 10 span only two cells, a situation that greatly improves the solvability of the system of linear equations (improves the diagonal dominance of the resulting matrices). Second, notice that mass flux is the dependent variable common to Eq. 8 and Eqs. 9 or 10. Both Eq. 8 and Eq. 10 specify this variable at cell edges, whereas Eq. 9 specifies it at cell centers. Therefore, mass flux values from Eq. 10 can be substituted directly into Eq. 8, a convenient situation.

The numerical analyst must select a numerical technique and a difference scheme, such as that represented by Eqs. 8 and 10, that exhibit accuracy and stability. The term accuracy means that, as Δt and Δx are made smaller and smaller, the numerical results are better and better approximations of the exact solution to the original differential equations. Stability means that the results show no unbounded growth of errors. Generally, stability depends on the choice of Δt and Δx.

We can decide how to time-difference the spatial terms in Eqs. 8 and 10 on the basis of stability criteria. Let us assume for the moment that we are using what is known as a fully explicit difference scheme, that is, all the spatial terms are specified at time t_j. It can be shown that our technique is then stable only if everywhere

$$\frac{(v + C)\Delta t}{\Delta x} < 1 \ , \tag{11}$$

where C is the local sound speed $(\partial P/\partial \rho)^{1/2}$ at constant entropy. We can develop a feel for why this is so by examining the consequences of violating the criterion. Then, for small velocities ($v \ll C$), $\Delta t > \Delta x/C$. For example, we will set Δt equal to $2\Delta x/C$. During a time interval of this duration, a small, narrow pressure pulse at x_i will travel a distance $2\Delta x$ (the wave speed of the pulse is C) to x_{i+2}. At the end of the time interval, at t_{j+1}, the mass flux at $x_{i+3/2}$, and hence certainly at $x_{i+1/2}$, should be affected. But Eq. 10, containing pressure values at t_j, does not reflect the influence of the pulse. In fact, calculated results based on time intervals violating the criterion of Eq. 11 will quickly show exponential growth of errors and become meaningless.

Because the sound speed is high for liquids, the stability criterion of Eq. 11 restricts us to quite short time intervals. We prefer to use instead a semi-implicit technique: in the momentum-conservation equation, to specify the pressures at t_{j+1} and the momentum fluxes at t_j, and, in the mass-conservation equation, to specify the mass fluxes at t_{j+1}. Through similar arguments based on mass transport, it can be shown that this semi-implicit technique will be stable only if $v\Delta t/\Delta x < 1$, a much less restrictive criterion.

Applying the semi-implicit difference scheme to Eqs. 8 and 10 and rearranging, we arrive at the following system of equations.

$$\rho_i^{j+1} = \rho_i^j - \frac{\Delta t}{\Delta x}\left[(\rho v)_{i+1/2}^{j+1} - (\rho v)_{i-1/2}^{j+1} \right] \tag{12}$$

and

$$(\rho v)_{i+1/2}^{j+1} = (\rho v)_{i+1/2}^{j} - \frac{\Delta t}{\Delta x}\left[(\rho v^2)_{i+1}^{j} - (\rho v^2)_i^j + P_{i+1}^{j+1} - P_i^{j+1} \right] \ . \tag{13}$$

A problem remains: we need values for the momentum fluxes. First, note that momentum fluxes can be calculated from mass fluxes, that is, $\rho v^2 = (\rho v)^2/\rho$. Then, we must decide what mass fluxes to use. Stability considerations demand that we use "upwind" mass fluxes. That is, if v_{i+1} is positive, we calculate $(\rho v^2)_{i+1}$ from the mass flux at $x_{i+1/2}$. If v_{i+1} is negative, the mass flux at $x_{i+3/2}$ is used.

An equation for ρ_i^{j+1} is obtained by substituting expressions for $(\rho v)_{i+1/2}^{j+1}$ and $(\rho v)_{i-1/2}^{j+1}$ (both provided by versions of Eq. 13 at $x_{i+1/2}$ and $x_{i-1/2}$) into Eq. 12. The reader so enthusiastic as to attempt the algebra will generate an equation for ρ_i^{j+1} in terms of known quantities (quantities at t_j) and the pressures $(P)_{i+1}^{j+1}$, P_i^{j+1}, and $(P)_{i-1}^{j+1}$. At this point, we linearize our equation of state.

$$\rho_i^{j+1} = \rho_i^j + \frac{d\rho}{dP}\left(P_i^{j+1} - P_i^j\right) \ , \tag{14}$$

where $d\rho/dP$ is obtained from Eq. 7. Combining Eq. 14 with our final equation for ρ_i^{j+1} results in an equation for pressure with a tridiagonal band structure in P_{i-1}, P_i, and P_{i+1}. Solution of this equation provides us with pressures at t_{j+1} and, hence, with densities and mass fluxes from the equation of state and Eq. 13, respectively. We have now advanced all variables from t_j to t_{j+1}. The process continues until the time boundary is reached.

Our sample problem is an example of an initial-value and a boundary-value problem. We must therefore somehow be provided with initial values of ρ, P, and ρv for all x and with boundary values for all t. For some numerical techniques, inclusion of boundary conditions can be a tricky matter; the requirement that we achieve closure for the linear equations often implies the need for more boundary conditions than are demanded by the original differential equations. Inclusion of boundary conditions in the finite-difference technique illustrated here is generally straightforward. Sufficient boundary conditions for single-phase flow through a pipe consist of the pressures external to the pipe at both ends and the density on the inlet side.

With considerably more tedious detail, the method of finite differences can be applied to the more complicated equations describing two-phase flow. Although it may seem nearly impossible, large computer codes that accurately portray all the complexities of a reactor transient can be constructed with this numerical technique and the models described above. TRAC is an outstanding example of such a code.

The challenges in producing a code like TRAC, which currently contains about 40,000 statements, are numerous; careful assessment of the models and methods is necessary. The results, however, are a tool for describing the complicated two-phase flows in reactors and for providing better estimates of reactor safety. ■

AUTHOR

Dennis R. Liles, Leader of the Code Development Group, has worked in the area of reactor safety since joining the Laboratory in 1974 and has been in charge of numerical solution techniques and models for TRAC. After receiving his Bachelor of Science in mechanical engineering from the Georgia Institute of Technology in 1968, he taught in the U. S. Army at Fort Bliss, Texas. He earned a Master of Science in mechanical engineering from the University of Texas at El Paso in 1971 and a Ph.D. in the same field from the Georgia Institute of Technology in 1974. His primary interests are in the physics of two-phase flow and the numerical solution of fluid-dynamics problems. His contributions in both these areas have enhanced the ability of analysts to predict the behavior of nuclear reactors. An active member of the American Nuclear Society, he currently serves on the program committee of its Thermal Hydraulic Group.

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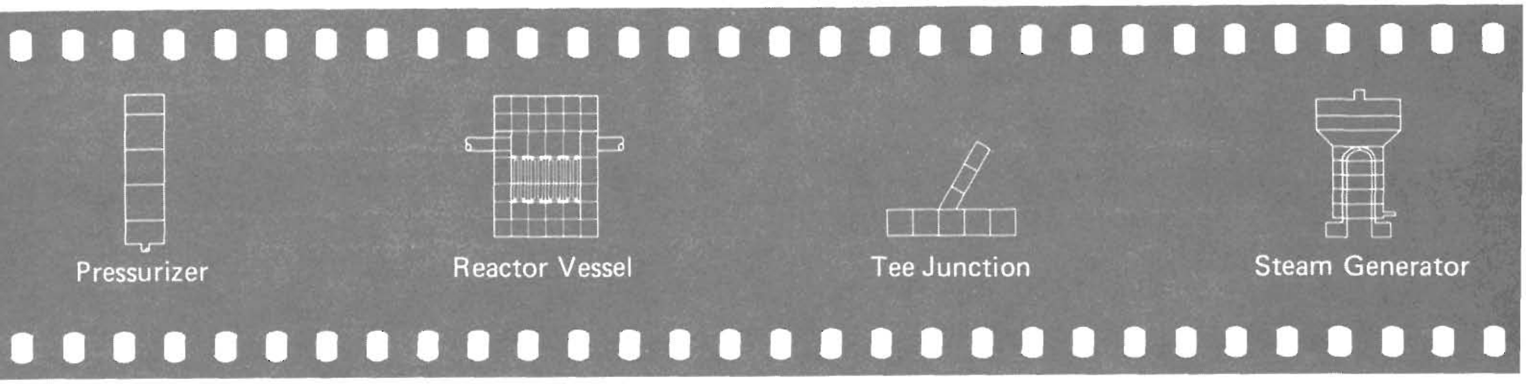


ACCIDENT SIMULATION WITH

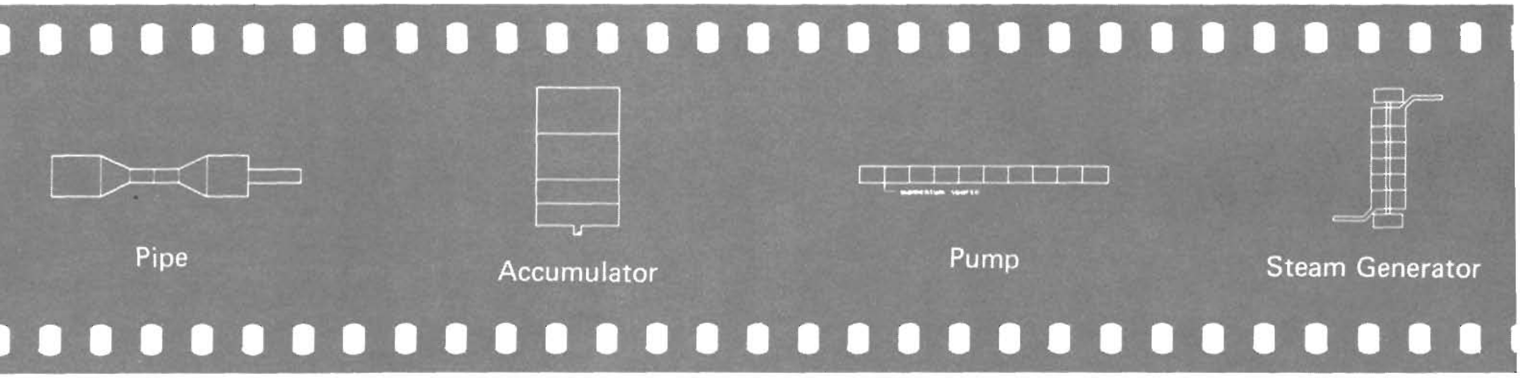
by John C. Vigil and Richard J. Pryor

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.

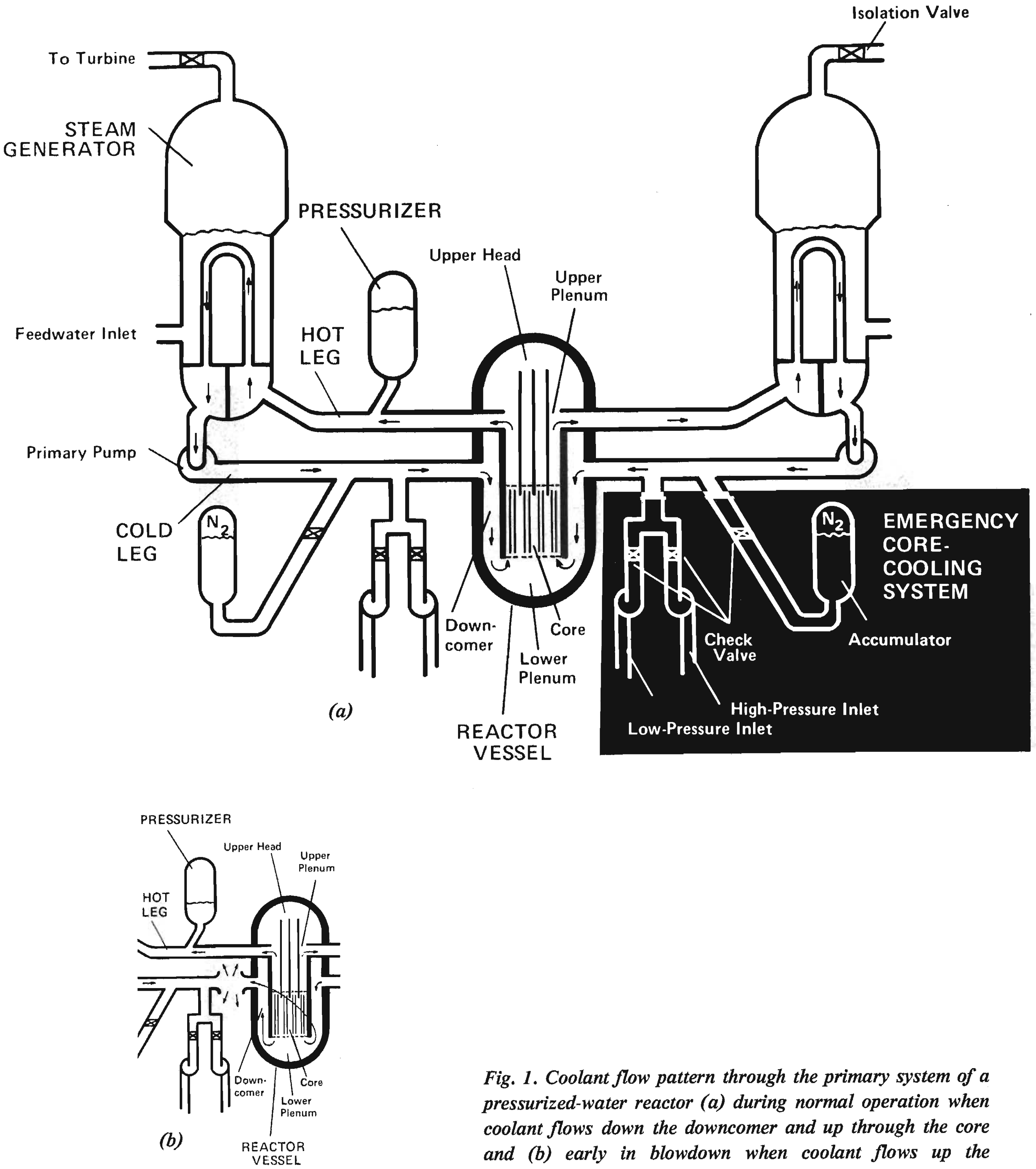


Fig. 1. Coolant flow pattern through the primary system of a pressurized-water reactor (a) during normal operation when coolant flows down the downcomer and up through the core and (b) early in blowdown when coolant flows up the downcomer and out the broken cold leg.

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.

*1 bar = 10^5 pascals ≅ 1 atmosphere.

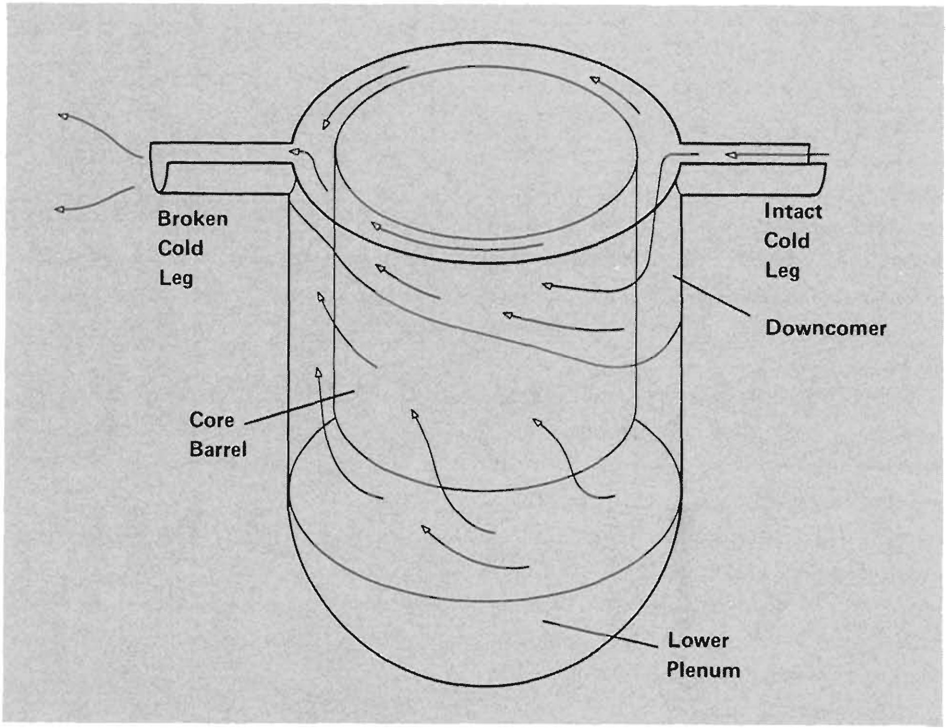


Fig. 2. Steam-water flows in the downcomer during bypass when emergency coolant swirls around the downcomer and out the cold-leg break.

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 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

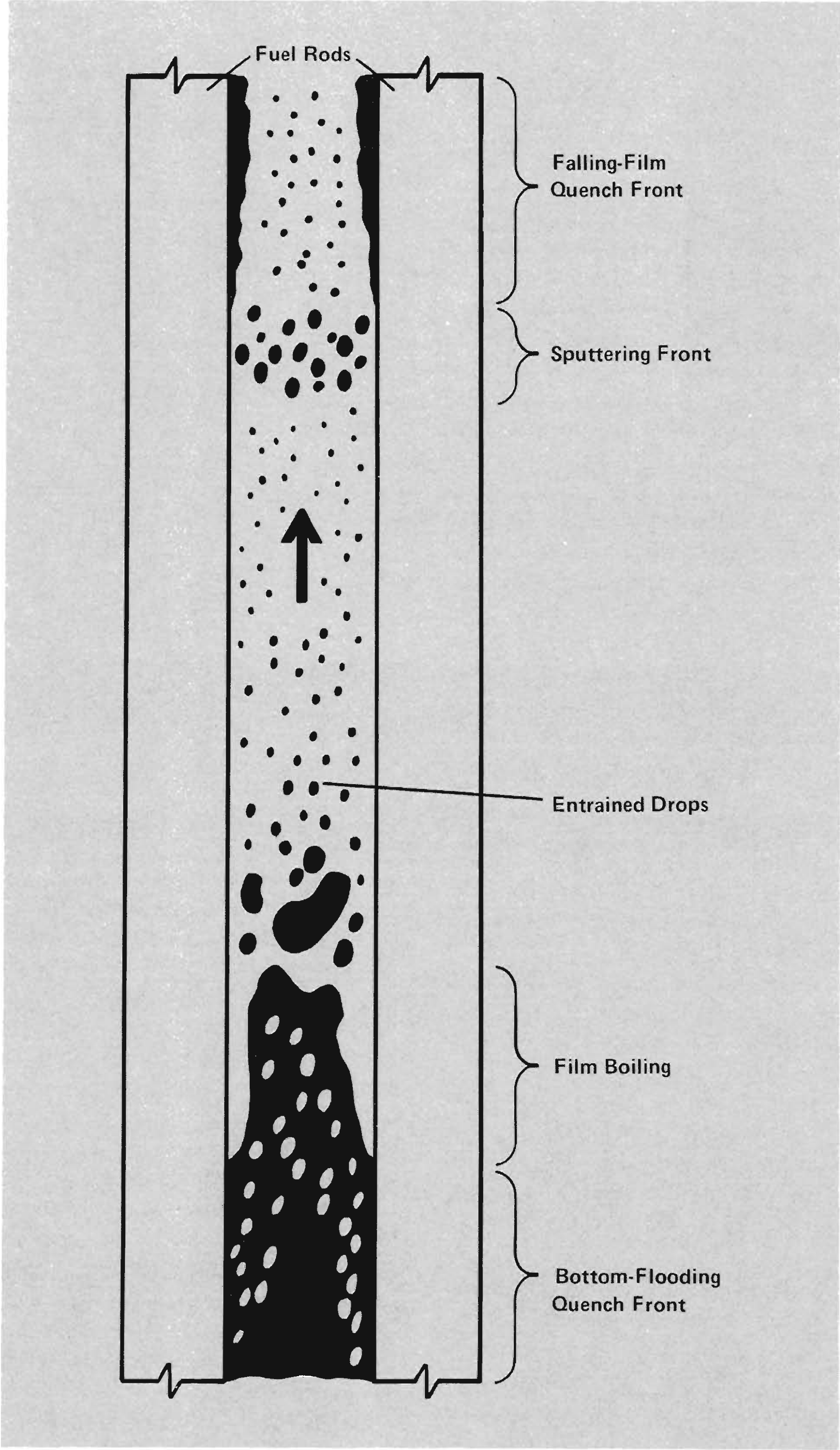


Fig. 3. During reflood, fuel rods are quenched from the bottom by water rising through the core and from the top by liquid films falling through the upper core-support plate.

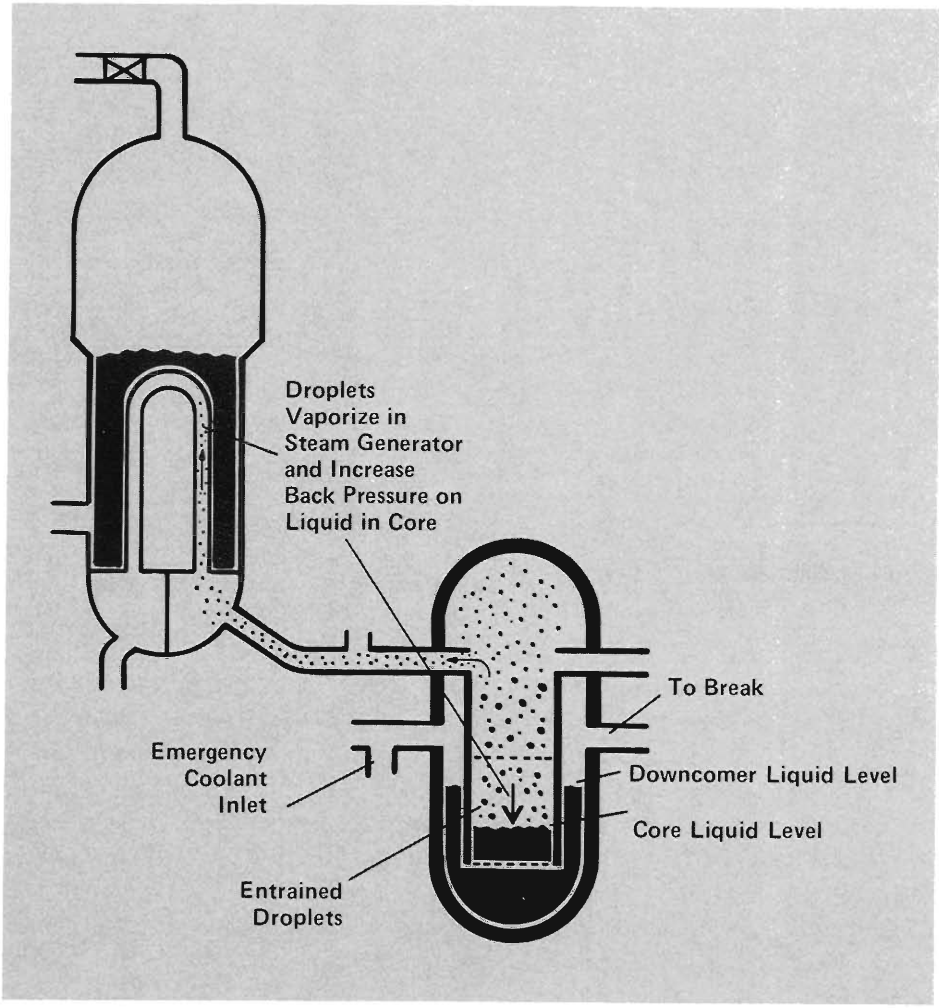


Fig. 4. Steam binding during reflood. The pressure created by vaporization of entrained droplets in the steam generator opposes the flow of emergency coolant to the core.

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-

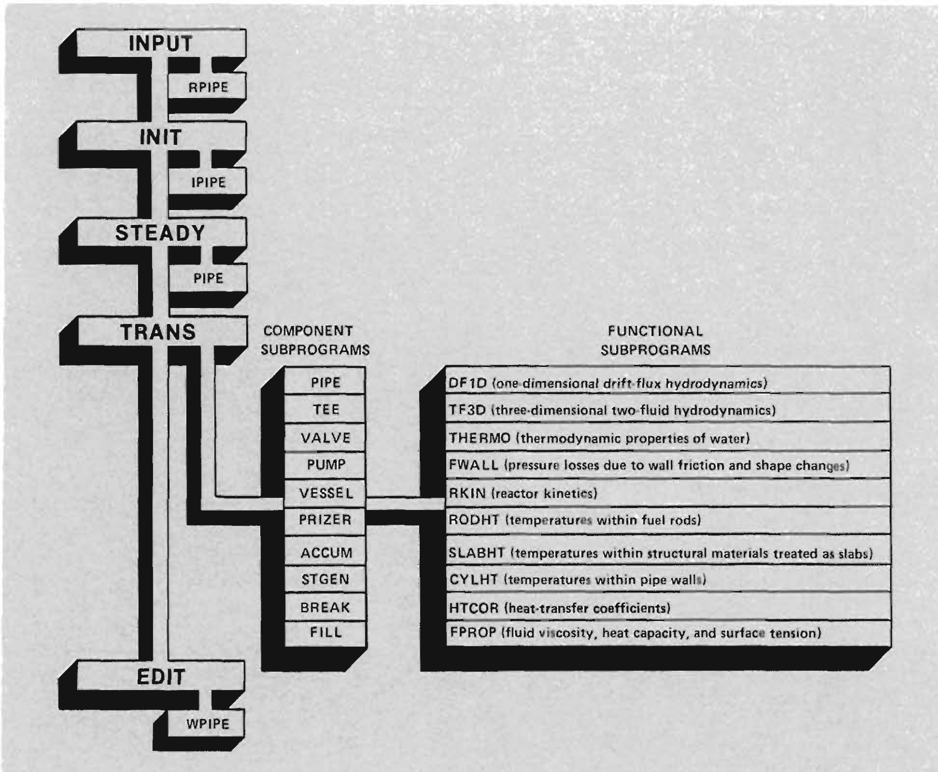


Fig. 5. TRAC is divided into five main subprograms, each of which handles a major aspect of the problem. INPUT accepts the user's description of the problem, INIT calculates quantities required for analysis that need not be supplied as input, STEADY calculates pretransient, or steady-state, conditions 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.

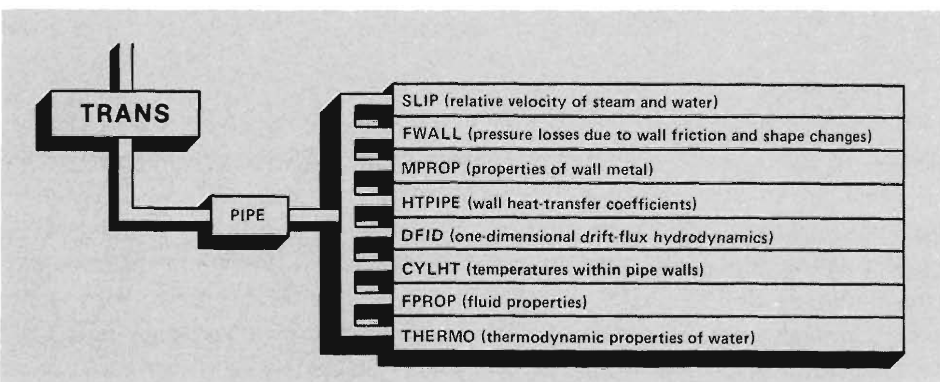


Fig. 6. Main functional subprograms accessed by PIPE to calculate the fluid dynamics and heat transfer within a pipe.

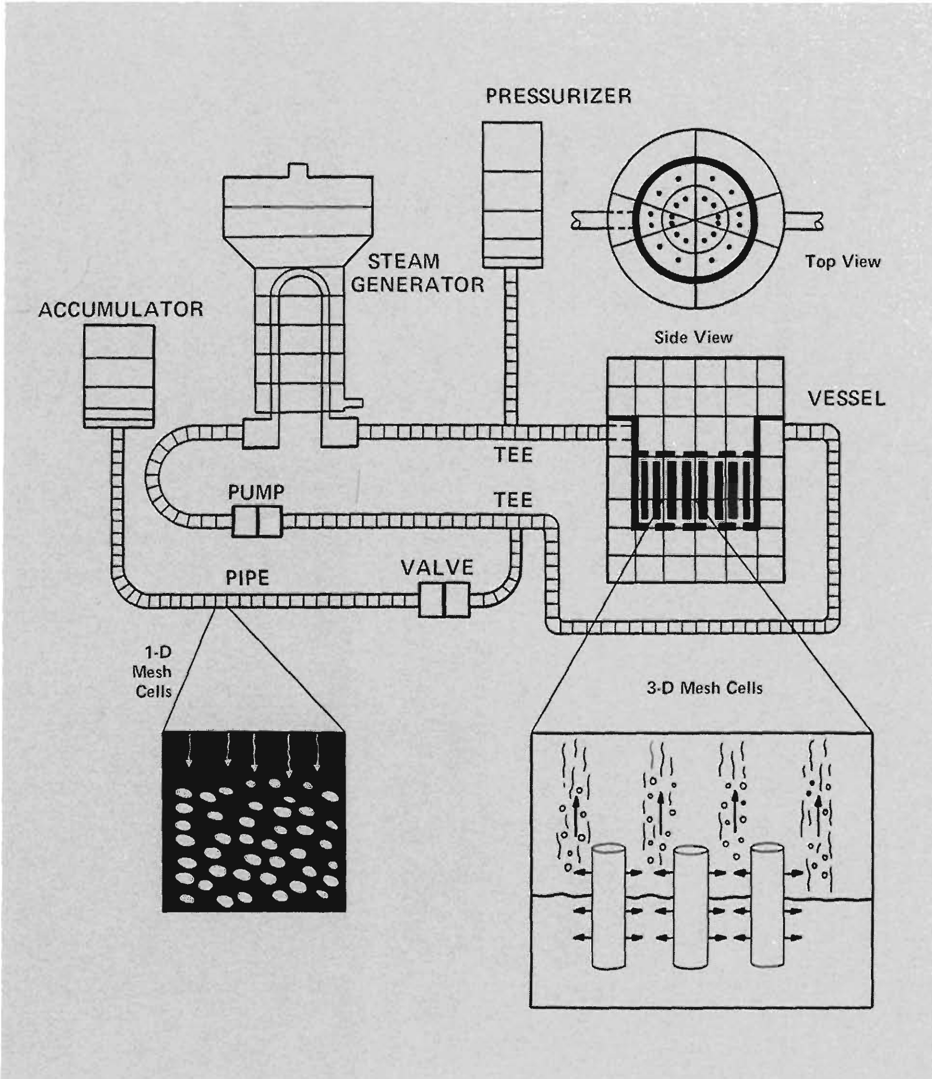


Fig. 7. Typical computational mesh for a vessel and a single coolant loop. In vessel cells, TRAC computes the nuclear heat and its transfer among fuel rods, flowing steam and water, and structural materials. In pipe cells, TRAC computes steam-water flow conditions and heat transfer between the two phases 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 interface 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 (DF1D) 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-

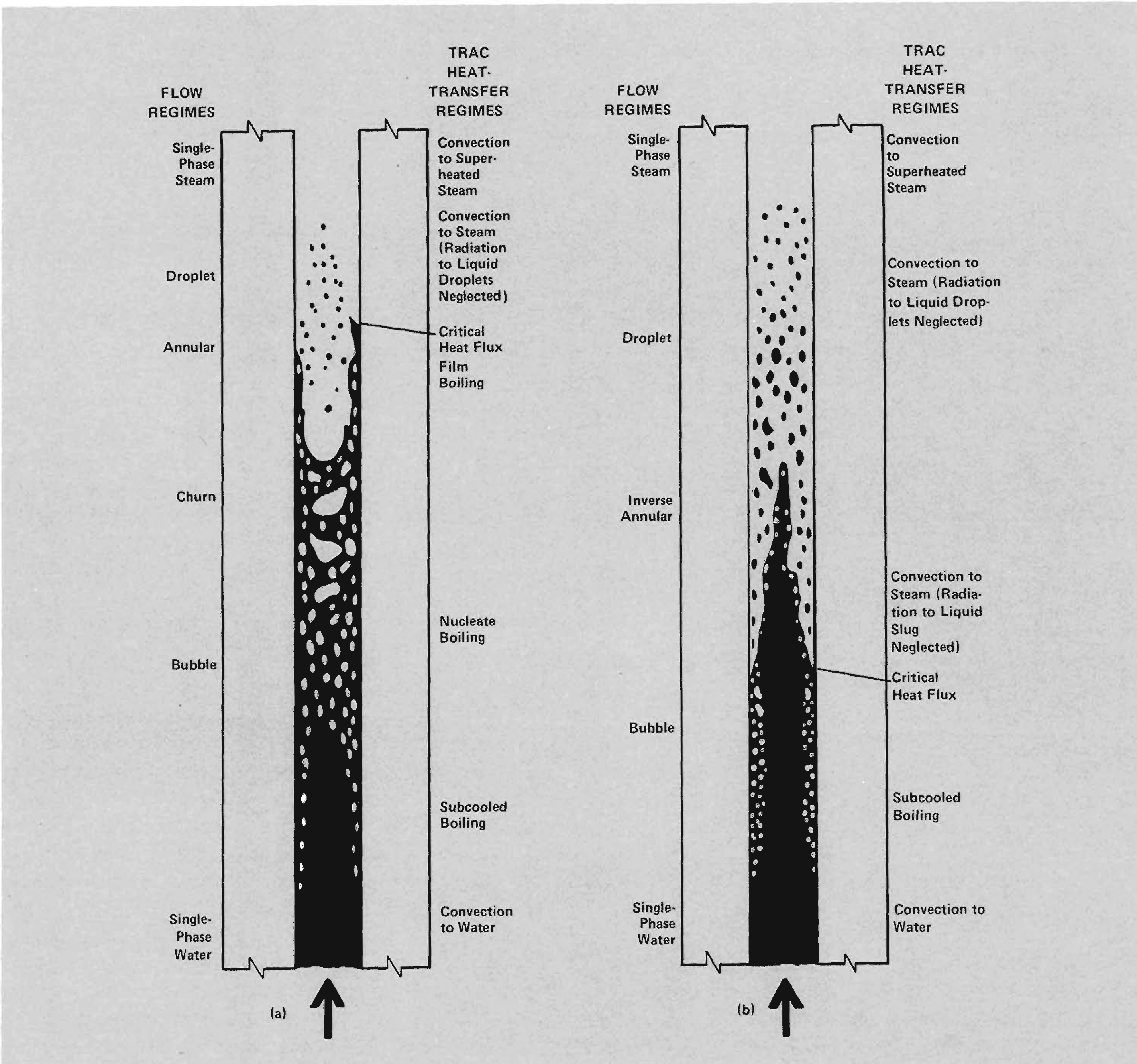


Fig. 8. Flow regimes and associated heat-transfer regimes for upward flow through a vertical array of fuel rods as (a) low and (b) high heat fluxes.

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 fine 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 für 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 III	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 II

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 II 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

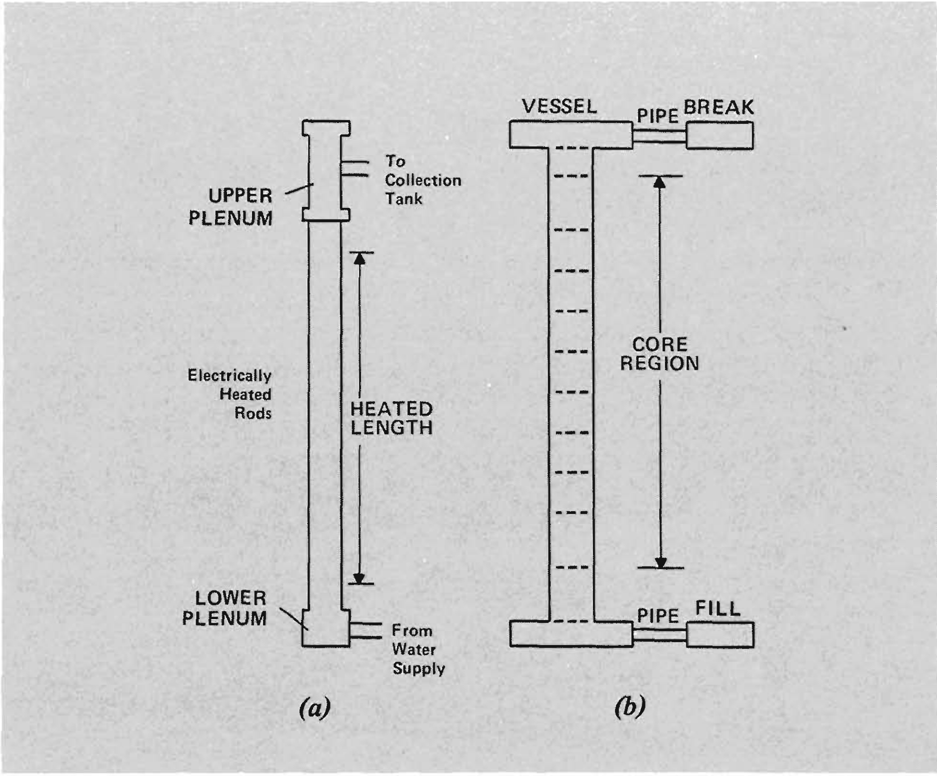


Fig. 9. Schematic diagram of (a) FLECHT and (b) its division into computational cells in the TRAC model. FLECHT's simulation of a core consists of a single bundle of electrically heated, full-length rods in a 10 by 10 array. Because multidimensional effects were not the focus of the experiment, the vessel 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.

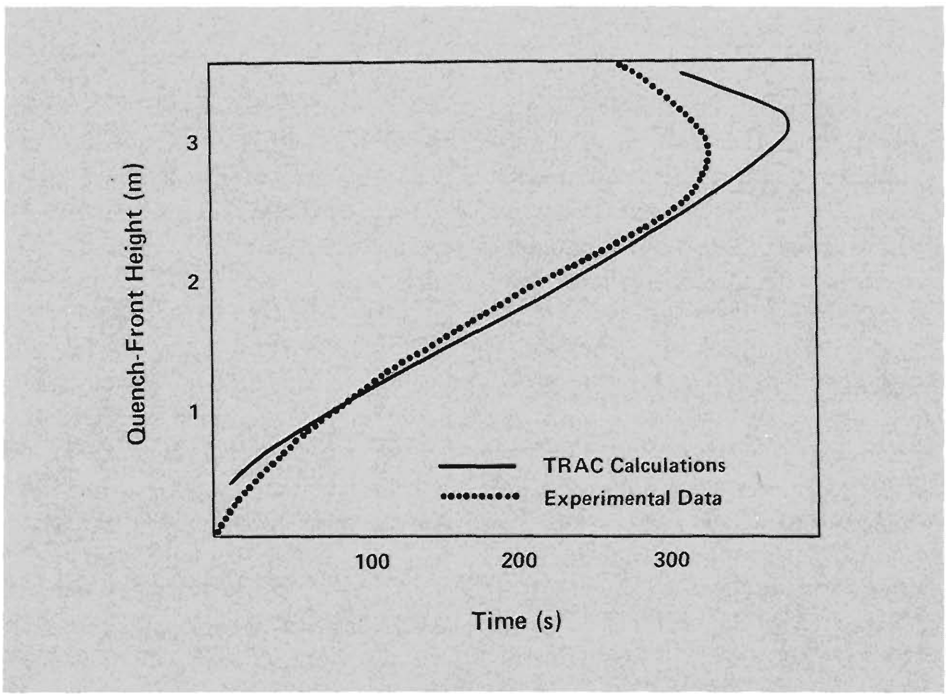


Fig. 10. Quench-front propagation during a reflood test at FLECHT.

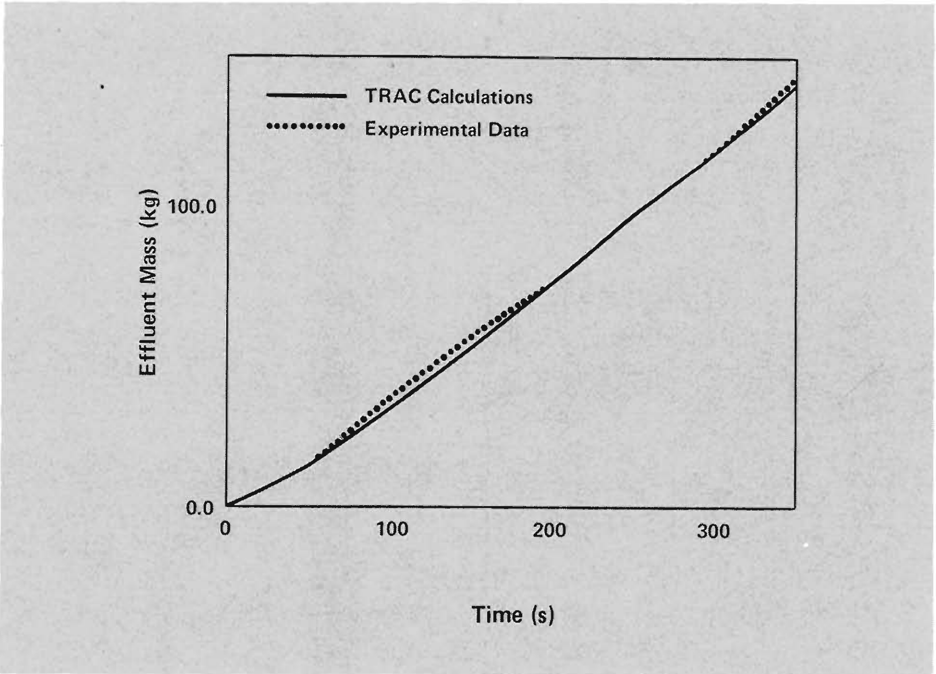


Fig. 11. Fluid mass exiting from the vessel during a FLECHT reflood test.

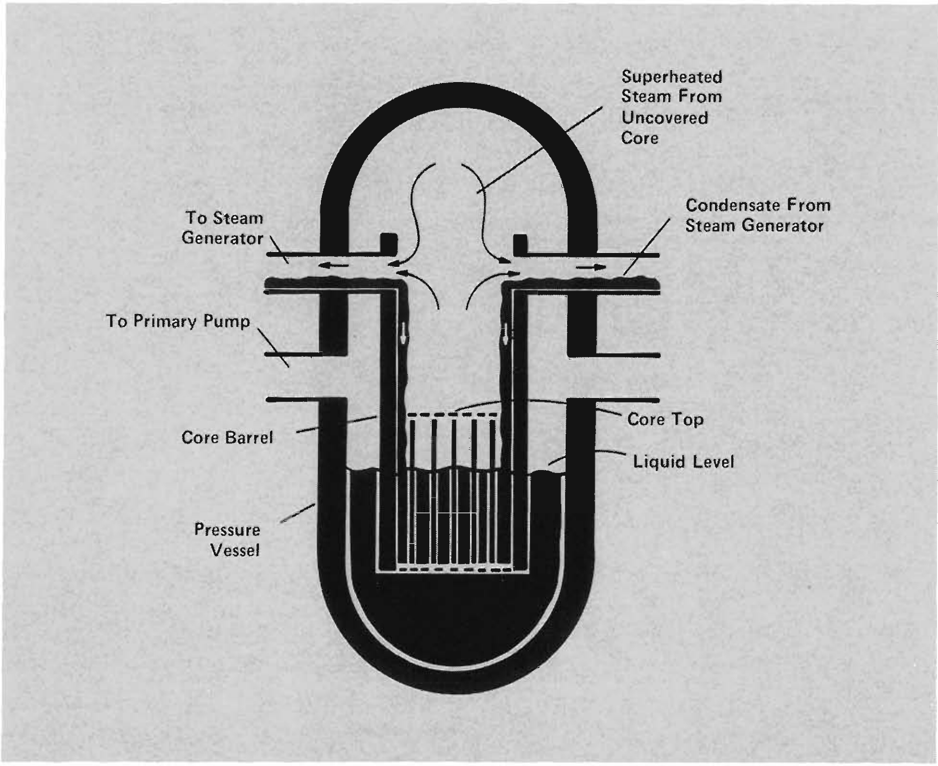


Fig. 12. Reflux cooling plays a role in cooling the core if the primary pumps are turned off and the core is partially uncovered. Superheated steam from the core condenses in the steam generator and flows back to the core along the hot legs.

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

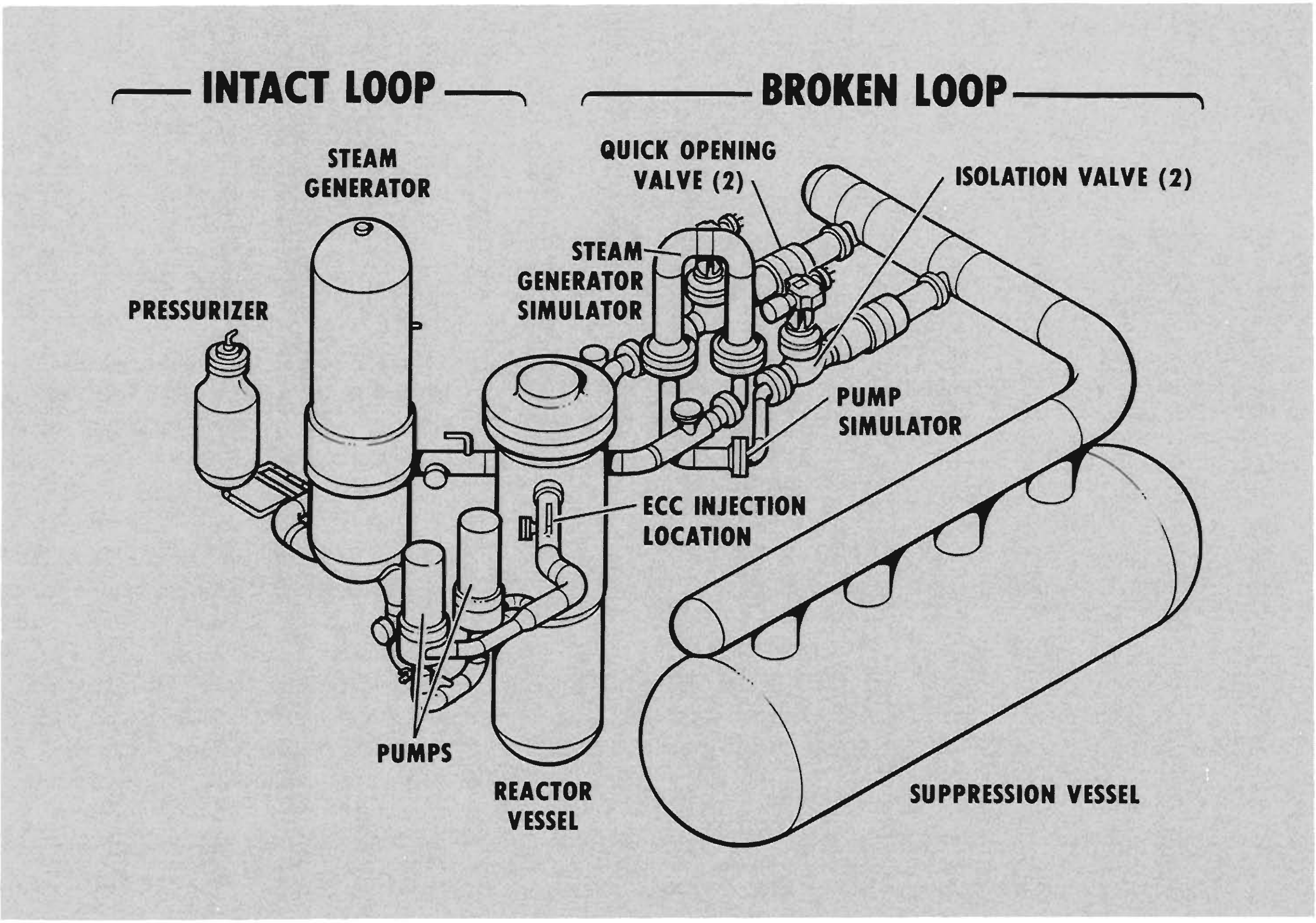


Fig. 13. Major components of LOFT, an intermediate-scale facility for integral loss-of-coolant tests. Volume, power, and flow and break areas are scaled at 1 to 60. LOFT is unusual in that it contains a real nuclear core rather than electric heaters. Breaks are simulated by the quick-opening valves.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

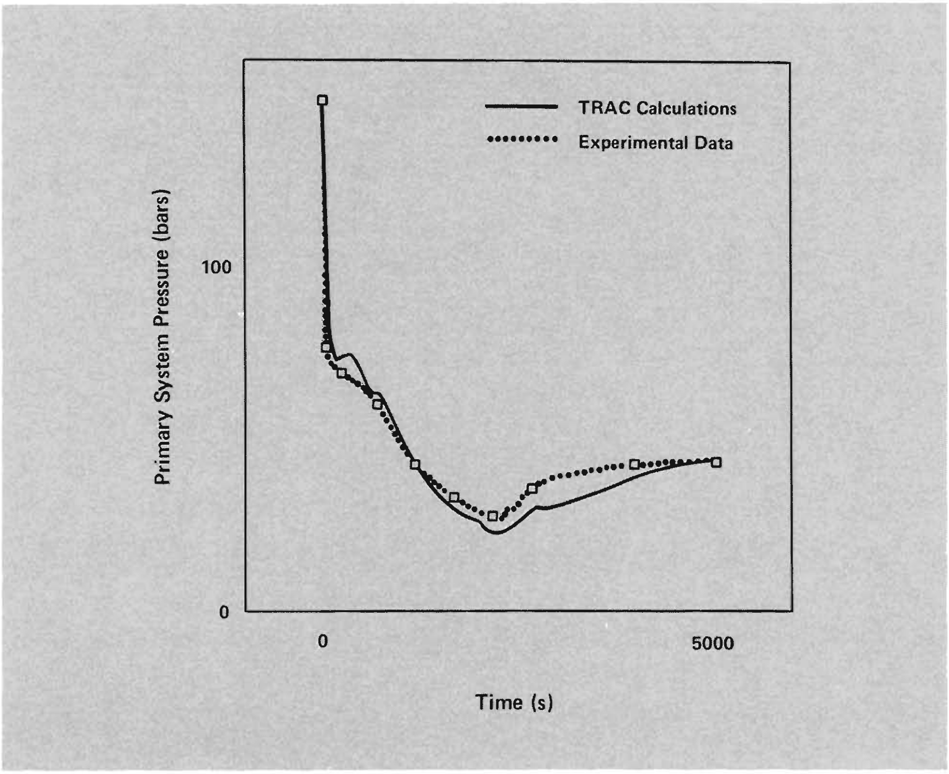


Fig. 14. Primary system pressures during a simulated small-break loss-of-coolant accident at LOFT with the primary pump turned off immediately.

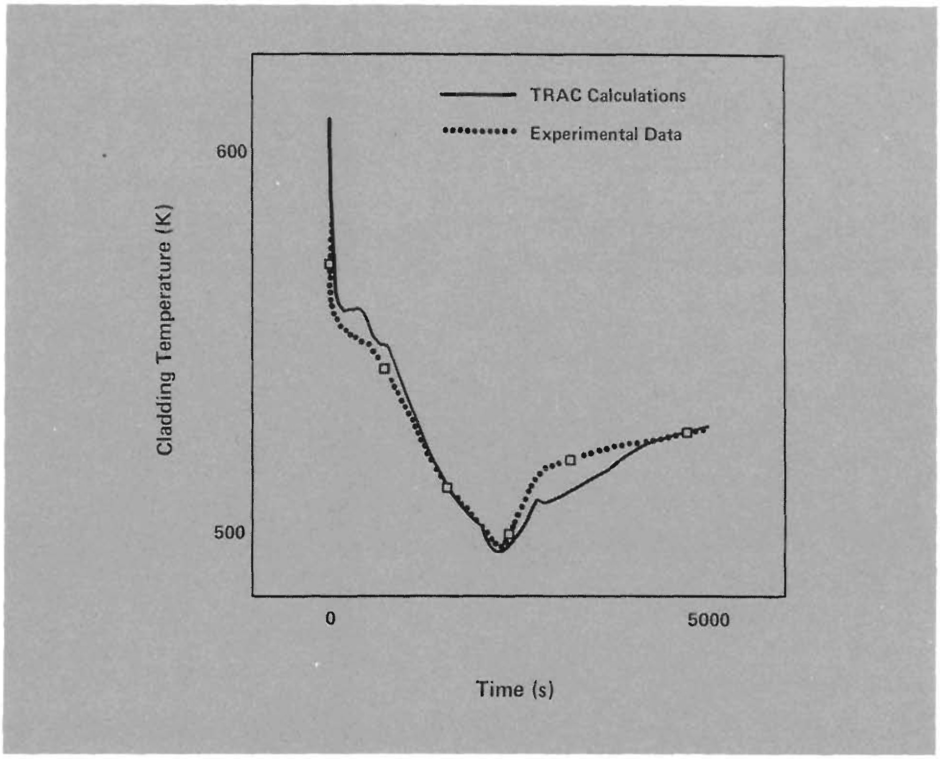


Fig. 15. Cladding temperatures during a simulated small-break loss-of-coolant accident at LOFT with the primary pump turned off immediately.

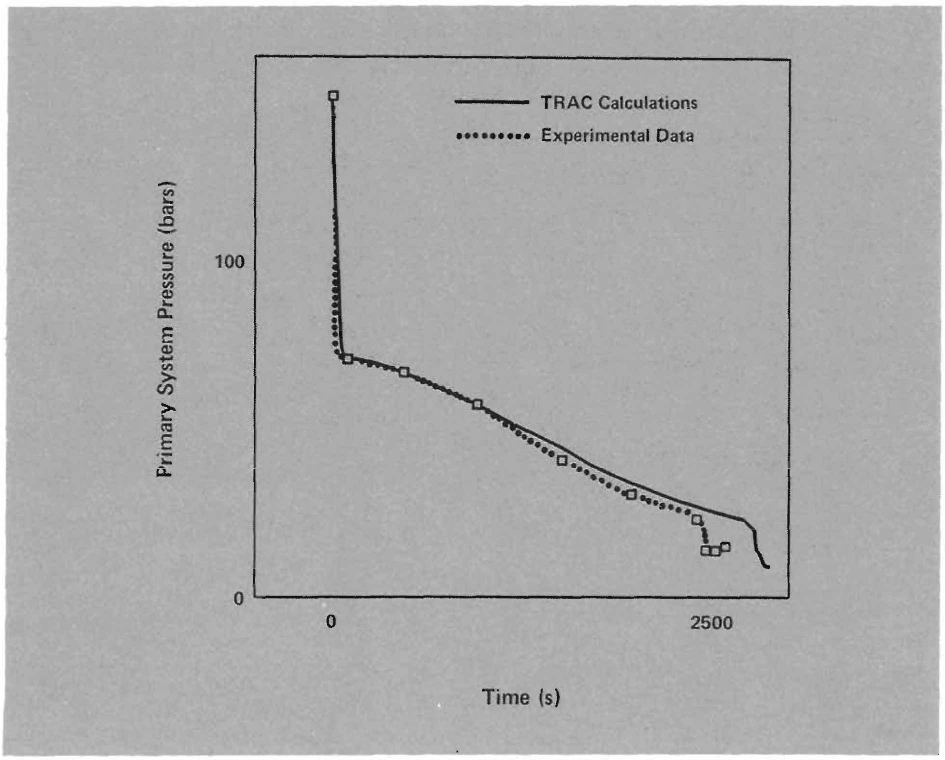


Fig. 16. Primary system pressures during a simulated small-break loss-of-coolant accident at LOFT with the primary pumps operating until about 2400 seconds.

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.

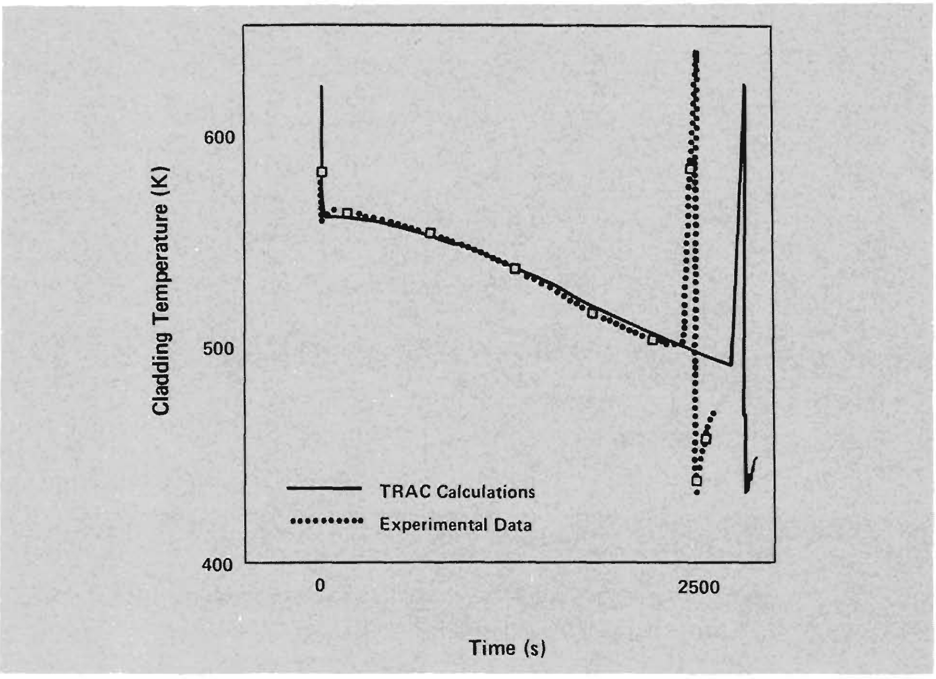


Fig. 17. Cladding temperatures during a simulated small-break loss-of-coolant accident at LOFT with the primary pumps operating until about 2400 seconds.

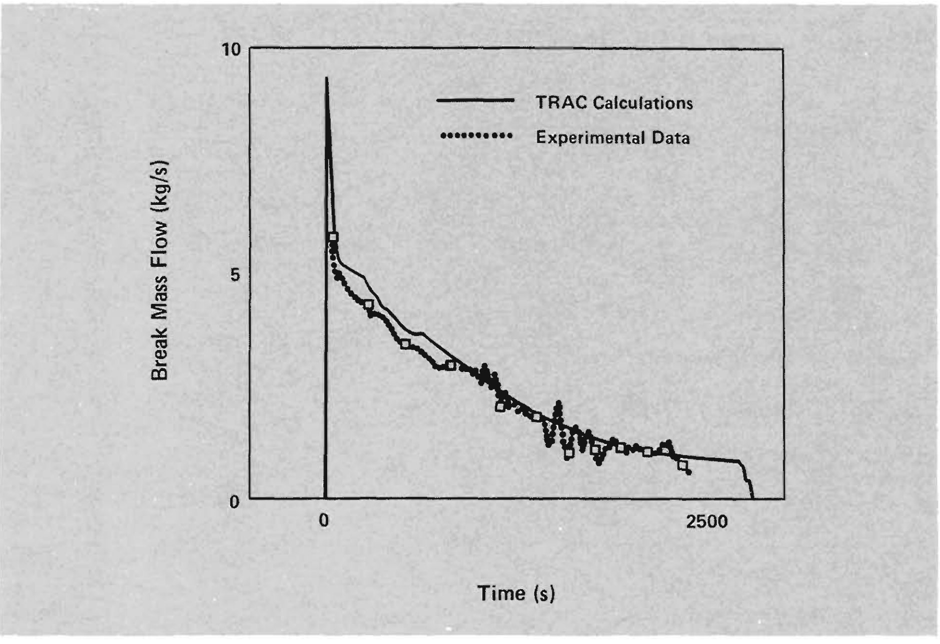
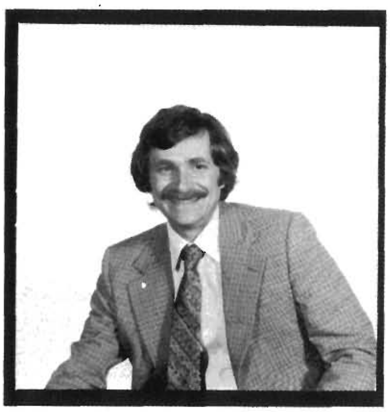


Fig. 18. Fluid mass exiting from a simulated small break in a cold leg of LOFT with the primary pumps operating until about 2400 seconds.

AUTHORS

John C. Vigil obtained his Bachelor of Science in physics in 1961 from the New Mexico Institute of Mining and Technology, where he graduated with highest honors; he earned his Ph.D. in nuclear engineering in 1966 from the University of New Mexico. He was awarded a Special Fellowship in Nuclear Science and Engineering by the Atomic Energy Commission in 1961 and was listed in *American Men of Science* in 1968 and in *Who's Who in the West* in 1970. He has been with the Laboratory since 1963 and has specialized in the development and application of large computer codes for reactor analysis. Before assuming his present position as Assistant to the Associate Director for Energy Programs, he was Assistant Energy Division Leader for Reactor Safety.



Richard J. Pryor joined the Laboratory in 1976 and is currently the Program Manager for Nuclear Reactor Programs. Before this assignment, he was Leader of the Code Development Group, which is responsible for development of TRAC. He received a B.S. in physics from Pennsylvania State University in 1965 and a Ph.D. in nuclear physics from the University of Pittsburgh in 1970. He is a member of the American Physical Society and the American Nuclear Society.

Further Reading

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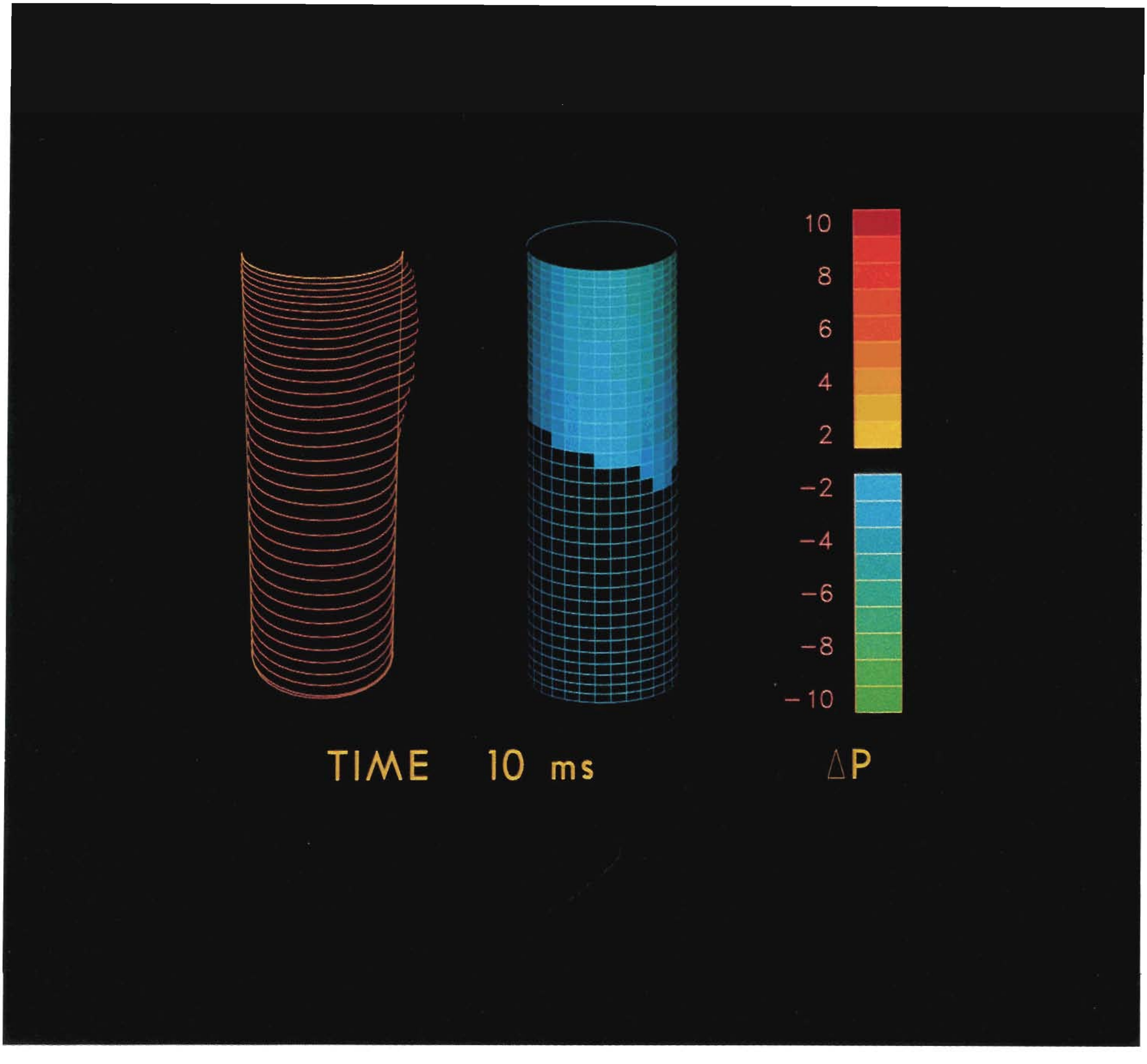
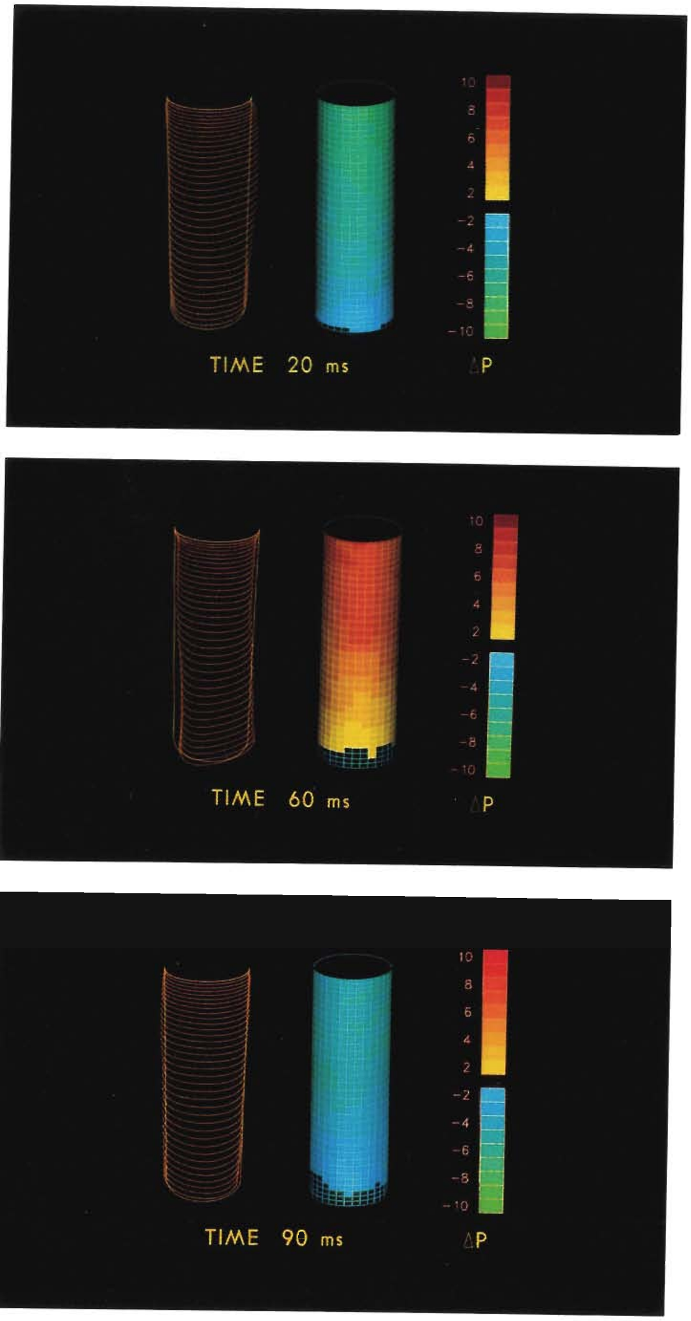
DETAILED STUDIES OF

Might the core barrel of a nuclear reactor suffer permanent deformation as a result of a break in an inlet pipe? The question has been examined with coupled fluid- and structural-dynamics computer codes. Shown here are four frames from a computer-generated movie by Rongriego depicting the calculated results. The left of each frame shows the deformation, which has been magnified 200 times for clarity. On the right of each frame color variations indicate the variation of pressure differences across the core barrel.



reactor components

by Anthony A. Amsden,
Bart J. Daly,
John K. Dienes, and
John R. Travis



Detailed numerical modeling of multiphase flows for reactor safety analyses has a relatively short history. It began in 1974 with the development of the KACHINA code by Francis H. Harlow and Anthony A. Amsden of the Los Alamos Fluid Dynamics Group. KACHINA was the first code to provide stable numerical solutions for multidimensional two-phase fluid dynamics. Since that time, research efforts in reactor safety analysis based on this method have been established at many installations throughout the world. The Los Alamos effort has continued with the ongoing development of multiphase computational techniques and, more recently, with the application of these techniques to practical problems in reactor safety.

The numerical methods developed at Los Alamos formed the basis for TRAC, the large systems code that computes thermal-hydraulic interactions throughout an entire reactor system during a simulated accident. The studies described here focus on separate effects—flow and thermal interactions in particular components of a reactor. Separate-effects studies are usually performed in much greater detail than is possible with a systems code. Unlike TRAC, which computes approximate one-dimensional interactions in all components except the reactor vessel, separate-effects codes compute very detailed interactions in two and three dimensions and may include the dynamic interaction of fluids with structures.

Often, multidimensional studies reveal important phenomena that do not appear in one-dimensional studies. An example discussed in detail below is critical two-phase flow through areas of restricted flow, such as a nozzle or a small break in a pipe. Attempts to develop one-dimensional correlations to describe these flows were not very successful. A correlation that worked well for one type of nozzle could not produce accurate predictions for a different type. A two-dimensional study indicated that the problem arose from a failure to account for certain geometric variations in the nozzles. After geometric effects were included in the correlation, it was applied successfully to a wide variety of nozzle configurations and sizes. This improved correlation can now be used with confidence in the systems codes.

The component codes do not generally employ empirical correlations for heat-transfer rates and other exchange functions. Instead, we develop thermal-hydraulic models from first principles and test their accuracy by comparing calculational results with experimental data. One notable example of such models is a set of constitutive relations for mass and momentum exchange in the mixing of steam and water by turbulent motion. These constitutive relations have been tested by extensive comparison with experimental data and then applied to practical reactor problems for which no experimental data are available.

Through this process of model development, comparision with experiment, and application to practical problems, we not only establish confidence in our own computational results, but we also demonstrate the capability of numerical methods for simulating complex multifluid and fluid-structure interactions. Our work has therefore helped to increase confidence in the results of analyses with large systems codes.

We will discuss in some detail our work on fluid-structure interactions and critical two-phase flow and describe briefly other efforts that illustrate the breadth of the modeling capabilities we have developed in this field.

Fluid-Structure Interactions

Pressurized-water reactors operate at relatively high pressure, typically about 150 bars (about 2250 pounds per square inch). Consequently, a sudden break of a large inlet or outlet pipe will produce strong depressurization waves that can create very high transient stresses in the reactor structure. Large-pipe breaks are not expected, even as a result of earthquakes, corrosion, or sudden changes in reactor power. However, reactor systems are designed so that, should one occur, the reactor itself would not be damaged and no significant amounts of radioactivity would be released. To determine the margins of safety under these extreme conditions, it is necessary to calculate in detail the dynamic interactions between the fluid and the structural components following a sudden break. Below we discuss interactions with two specific structures, the core barrel and the control-rod guide tubes (Fig. 1).

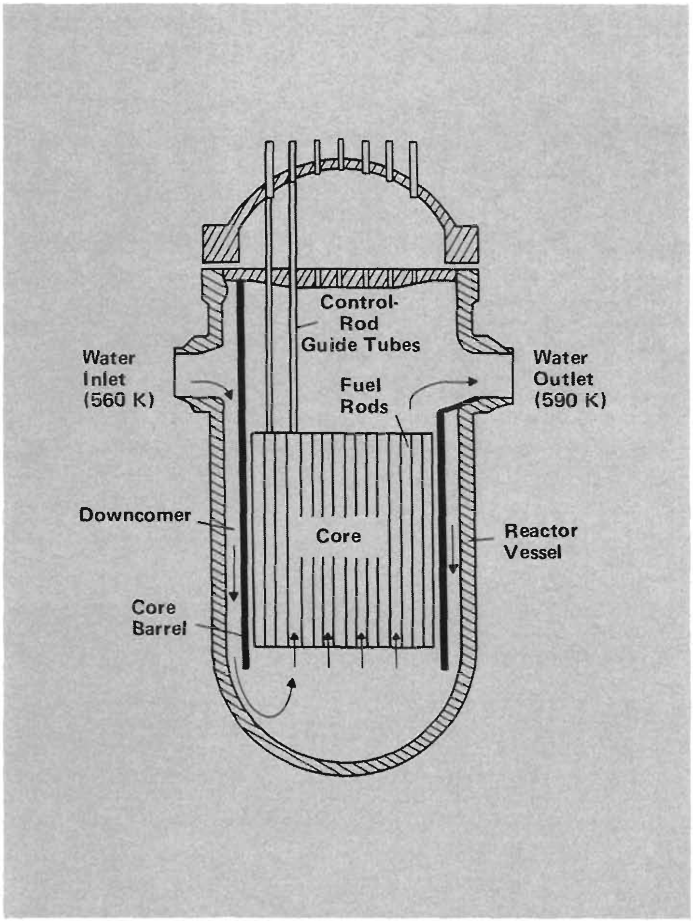


Fig. 1. Schematic diagram of the pressurized-water reactor components discussed in the text.

CORE BARREL RESPONSE TO INLET PIPE BREAK. A number of years ago, the Nuclear Research Center in Karlsruhe, West Germany (Kernforschungszentrum Karlsruhe) began a program to evaluate numerical models of fluid-structure interactions in reactors by comparing calculated results with experimental data. They were particularly interested in the response of the core barrel to a sudden, or guillotine, break in an inlet pipe. They made plans to perform experiments at Heissdampfreaktoranlage (HDR), an experimental facility near Frankfurt, West Germany, and, at the same time, asked a number of theoretical groups, including the Fluid Dynamics Group at Los Alamos, to predict the results of these experiments.

During normal operation, water enters the reactor vessel through an inlet pipe and flows down the downcomer and up through the core (see Fig. 1). The core is separated from the downcomer by a cylindrical steel shell, the core barrel. The core barrel serves a dual function: it holds the fuel rods rigidly in place and separates the cold incoming water from the hot water rising in the core.

Should an inlet pipe break, a depressurization, or rarefaction, wave will propagate into the downcomer at the speed of sound in the water, just under 1 meter per millisecond. As the wave propagates down the downcomer, it leaves a low-pressure region behind it. The resulting high pressure difference across the core barrel causes its outward displacement. In

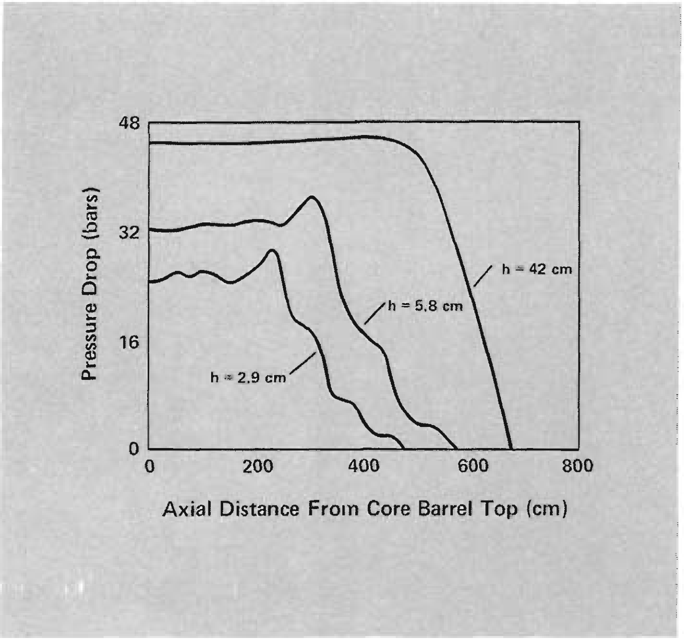


Fig. 2. Profiles of pressure drop in the downcomer showing the effect of wall thickness h *on the wave structure of the rarefaction.*

addition, a precursor wave propagates down the core barrel ahead of the main wave in the water (the speed of sound in steel is about 5 times greater than in water), but its effect is small. The motion of the core barrel generates acoustic waves in the water in the core, but their effect also is expected to be small.

These phenomena can be anticipated qualitatively, but five years ago when we undertook to quantify them, the available computer codes were inadequate to calculate the fluid pressure and the stresses in the core barrel. We needed three-dimensional codes for both the complex steam-water flow and the structural motion.

Los Alamos Calculations. To model the fluid motion we used K-FIX, a code for three-dimensional flow of two compressible phases. This code is based on the multifield technique of KACHINA and includes a fully implicit exchange of mass, momentum, and energy between the vapor and liquid phases. Phase transitions and interfacial heat transfer are coupled to the fluid dynamics in the pressure iteration. The method reduces to the Implicit Continuous-fluid Eulerian (ICE) technique for single-phase flow.

To model the core barrel motion, we developed a special-purpose code called FLX that solves the Timoshenko shell equations with an explicit finite-difference technique. (In the earliest work on this problem, the core barrel motion was represented by the classical theory of beams, but we rejected this approximation because, for example, it cannot account for local deformations of the core barrel, particularly where the cylindrical shell bulges toward the break. We also rejected the normal-mode description chosen by the theoretical group at Kernforschungszentrum Karlsruhe because it is difficult to formulate mathematically and cannot easily accommodate changes in the boundary conditions or modification to the structure.) Our finite-difference version of the shell equations is relatively straightforward and can be integrated numerically with the very fine time and spatial resolution needed to simulate the complex wave patterns generated by sudden loading.

The coupling of fluid dynamics and structural motion is accomplished in two parts. The fluid-dynamics code computes the pressure gradient acting on the core barrel and this pressure gradient is used in the structural code that solves the Timoshenko shell equations. The motion of the core barrel changes the width of the downcomer and, through this volume change, affects the fluid density. The fluid-dynamics code then incorporates the new density and computes the corresponding flow and pressure fields.

It is not necessary to use the same zoning or time steps in the two codes. In fact, we usually run the structural code with a time step less than a tenth of that used in the fluid-dynamics code because of the relatively high sound speed in the steel core barrel.

To illustrate how the stiffness of the core barrel affects the propagation of the depressurization wave in the downcomer, we present in Fig. 2 some calculations performed with the coupled code. Shown are the pressure profiles in the downcomer at one point in time for three different thicknesses. A 42-centimeter-thick core barrel acts as if it were rigid. With a thickness of 5.8 centimeters (typical of reactor geometries), the elastic motion of the core barrel produces significant oscillations in the pressure profile. The oscillations increase in amplitude as the wall thickness is decreased further. The calculations show that the pressure drop behind the depressurization wave and the wave speed both increase with increasing stiffness of the core barrel. We have derived an approximate analytic expression for the average wave speed $\bar{c}$.

$$\bar{c} = \frac{c_0}{1 + (\rho c_0^2 a^2 / Ehb)} \quad , \tag{1}$$

where c_0 is the speed of sound in the water, ρ is the density of the water, a and h are the radius and thickness, respectively, of the core barrel, b is the width of the downcomer, and E represents Young's modulus. This expression, like the calculation, shows the wave speed increasing with wall thickness.

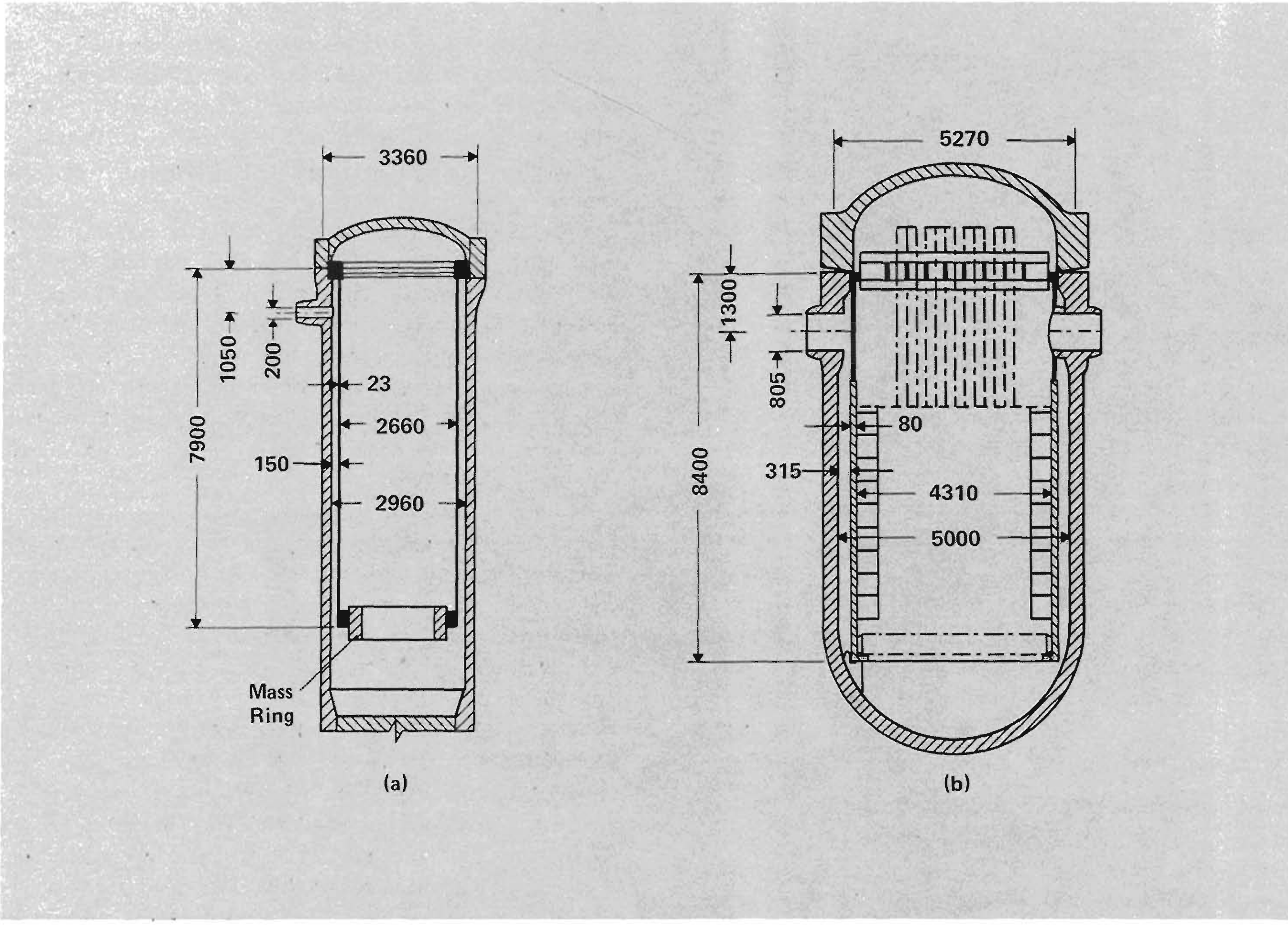


Fig. 3. Comparison of (a) HDR, the West German facility at which the depressurization experiments were performed, and (b) a typical pressurized-water reactor. The HDR mass ring simulates the mass of the fuel rods. Dimensions are in millimeters.

The HDR Experiment. In June 1980, the first of a series of experiments was carried out at HDR (Fig. 3). The fuel rods are simulated by a 10-metric-ton ring supported at the bottom of the core barrel. The height of the facility is typical of pressurized-water reactors, but its diameter is considerably smaller.

The response of the HDR core barrel to a guillotine break in a cold leg was monitored with about 75 instruments (pressure gauges, accelerometers, and strain gauges) that had been carefully selected and tested to operate at the temperature and pressure typical of a pressurized-water reactor. The initial temperature (540 kelvin) and pressure (108 bars) were supplied by electric heaters.

Before the experiment was carried out, six United States and West German groups calculated the response of the core barrel to a sudden break and submitted the pretest results to the Kernforschungszentrum Karlsruhe. The Los Alamos predictions for the pressure distribution and deflection of the core barrel at four times are shown in the opening figure. The core barrel undergoes transient oscillation but exhibits no per-

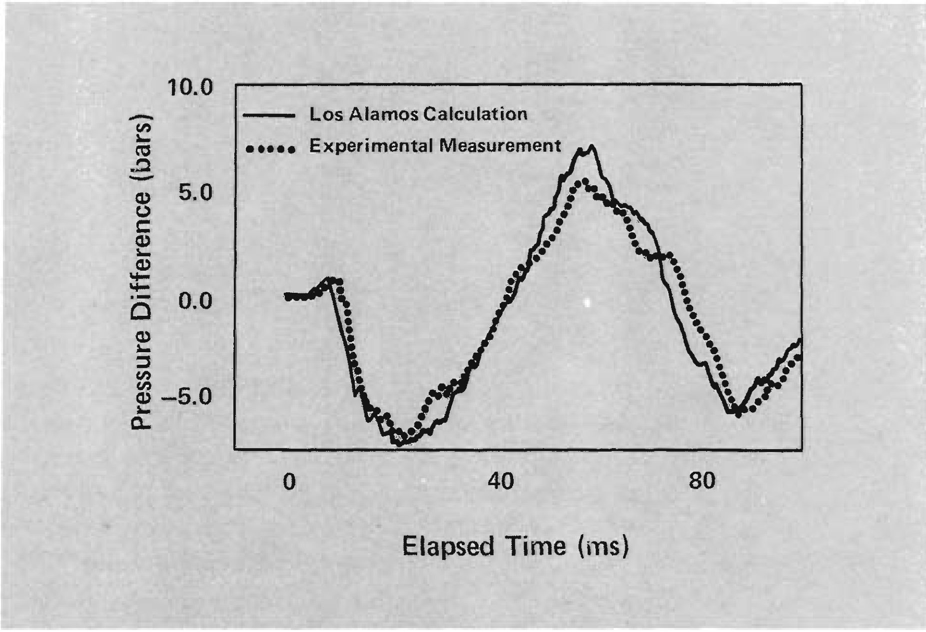


Fig. 4. Comparison of Los Alamos pretest prediction and HDR measurement of the pressure difference across the core barrel (at 330 cm below the inlet pipe) caused by a guillotine break in the inlet pipe.

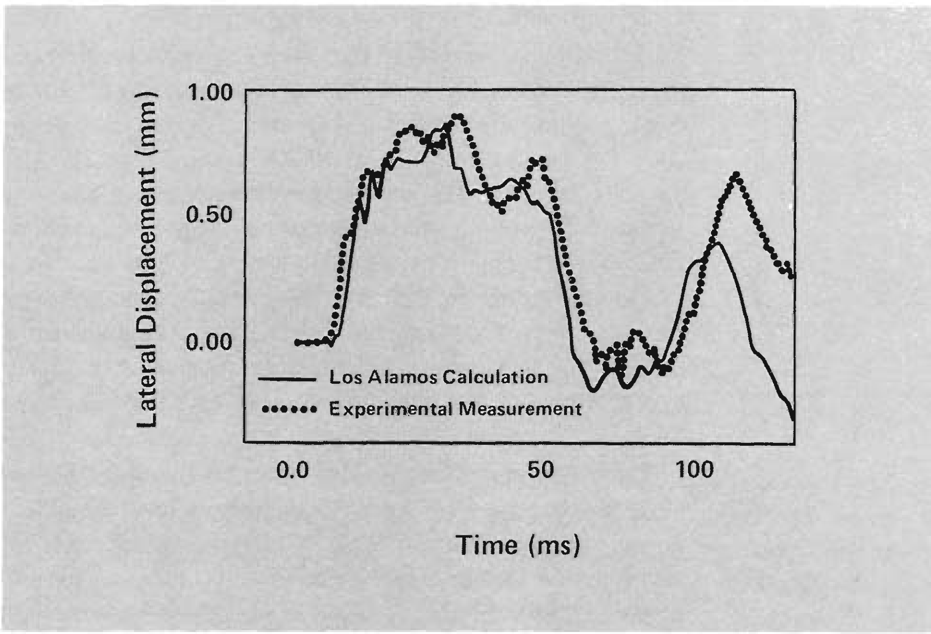


Fig. 5. Comparison of Los Alamos pretest prediction and HDR measurement of the displacement of the core barrel (at 330 cm below the inlet pipe) caused by a guillotine break in the inlet pipe.

manent deformation. Figures 4 and 5 show the good agreement between the Los Alamos calculations and the experimental data. This good agreement prompted a recent workshop on the HDR test to recommend that the Los Alamos coupled code be used for the official posttest calculation.

The good agreement between our calculations and the experiment is due in large part to the long lead time available to us. The coupled code evolved over a period of about five years, even though many of the techniques were already in hand at the beginning of the program. The level of accuracy finally achieved would not have been possible in a shorter time.

The structure and fluid flow in HDR are simplified compared to an operating reactor, but the comparison with experiment demonstrated that good predictions are possible. Commercial power reactors present many features in addition to those included in the HDR test, such as fuel rods, support plates, complex pipe breaks, and various flow restrictions. Accounting for these additional features will require a significant amount of additional work, but we feel that the capability for accurate calculations has been demonstrated.

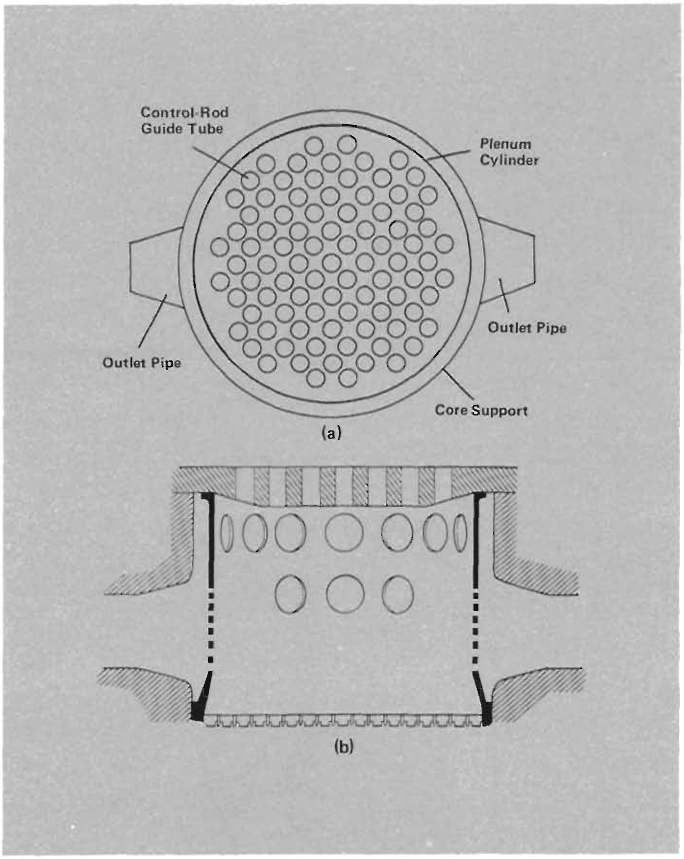


Fig. 6. Upper plenum of the pressurized-water reactor chosen for analysis of the effect of an outlet-pipe break on the displacement and possible deformation of the control-rod guide tubes. (We based our computer model on many of the features found in the West German Babcock-Brown Boveri reactor.) The top view (a) shows the location of the 89 control-rod guide tubes enclosed within the plenum cylinder, and the cross-sectional view (b) shows the large and small holes in the plenum cylinder through which water flows from the core to the outlet pipes. Fourteen large holes are distributed uniformly around the cylinder at a high level, and two diametrically opposite sets of three large holes are positioned at a lower level 90° from the outlet pipes. Two sets of 37 small holes are directly aligned with the outlet pipes. This hole arrangement, which forces much of the water to make a circuitous exit from the plenum to the gap between the plenum cylinder and the core support, affords the guide tubes some protection from the effects of a rarefaction wave resulting from sudden depressurization in an outlet pipe.

CONTROL-ROD GUIDE TUBE RESPONSE TO OUTLET PIPE BREAK. Another fluid-structure interaction of importance is the effect of an outlet-pipe break on the control-rod guide tubes in the upper plenum of a pressurized-water reactor (see Fig. 1). Would the guide tubes be deformed to such an extent that the control rods could not be lowered to shut down the reactor?

Analysis of this interaction involves the complex geometry of the upper plenum and the response of many guide tubes. The upper plenum of the reactor we analyzed (Fig. 6) included an arrangement of small and large holes in the plenum cylinder. This hole arrangement, which forces much of the flow to follow a circuitous path through the upper plenum, posed a particularly difficult modeling problem.

A three-dimensional fluid-dynamics code with considerable flexibility was required for the analysis. We chose SALE-3D, an implicit, three-dimensional, Arbitrary Lagrangian-Eulerian code that allows calculations in all flow-speed regimes. Written for the Cray computer, this code is particularly applicable to flows in highly complex geometries. It not only allows nonuniform zoning and curved boundaries, but, because it takes advantage of the high processing speed and large storage capacity of the Cray computer, it can also model geometric details with an accuracy never before practical. SALE-3D is used in tandem with a structural-response code that determines the guide-tube dynamics.

The fluid-dynamics computing mesh for the upper plenum (Fig. 7) is generated by distorting a Cartesian block of cells 52 across, 26 deep, and 10 high. The mesh approximates the circular cross section of each guide tube by 4 cells that form an octagon. The computing technique of SALE-3D permits the mesh to move with the fluid in a Lagrangian fashion, remain fixed in an Eulerian manner, or move in some arbitrarily specified way to provide a continuous rezoning capability.

Because early calculations had indicated that the elastic limit was likely to be exceeded for a number of the guide tubes, the goal of our analysis was an accurate assessment of the plastic response and resulting deformation. For this purpose, we subdivided each tube segment in the structural model into a set of 20 equal angular elements and used a sublayer model within each element to represent strain hardening. The structural-response code calculates the stresses, strains, deflections, and velocities in the horizontal plane for every segment of each guide tube, and also makes a record of those elements undergoing maximum stress and strain.

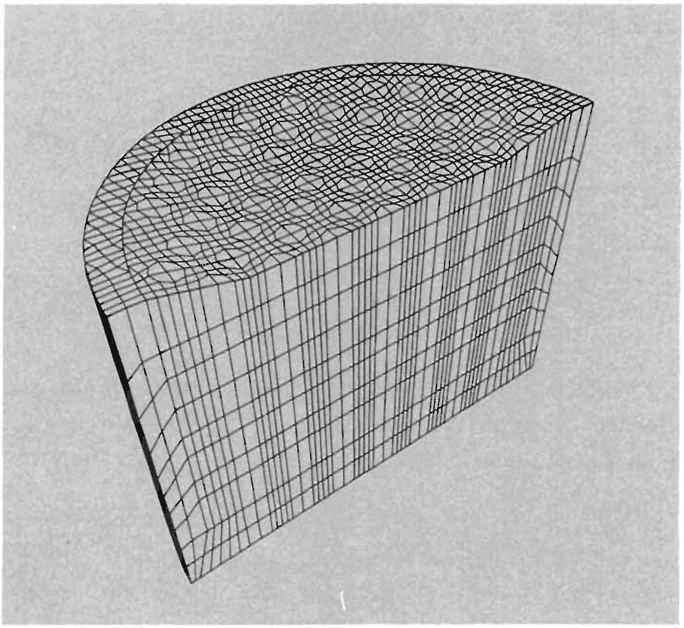


Fig. 7. Perspective view of the fluid-dynamics computing mesh for the upper plenum, which is created from a block of cells 52 cells wide, 26 cells deep, and 10 cells high. The front of the mesh lies on the symmetry plane that cuts through the midlines of the outlet pipes.

The transient calculation begins with a series of pressure changes at one of the two outlet pipes to simulate the depressurization resulting from a break. SALE-3D calculates the lateral forces on each guide tube at ten elevations, and these forces are then used to determine the guide-tube dynamics. The structural-response code returns the velocities for each elevation of each guide tube to SALE-3D. The guide-tube velocities are then applied as a boundary condition on the fluid flow. This interaction fully couples the fluid and structural dynamics. A plot of the velocity vectors in the fluid during depressurization is shown in Fig. 8. As a result of the fluid acceleration following depressurization, the transient speeds are nearly two and one-half times the steady-state values.

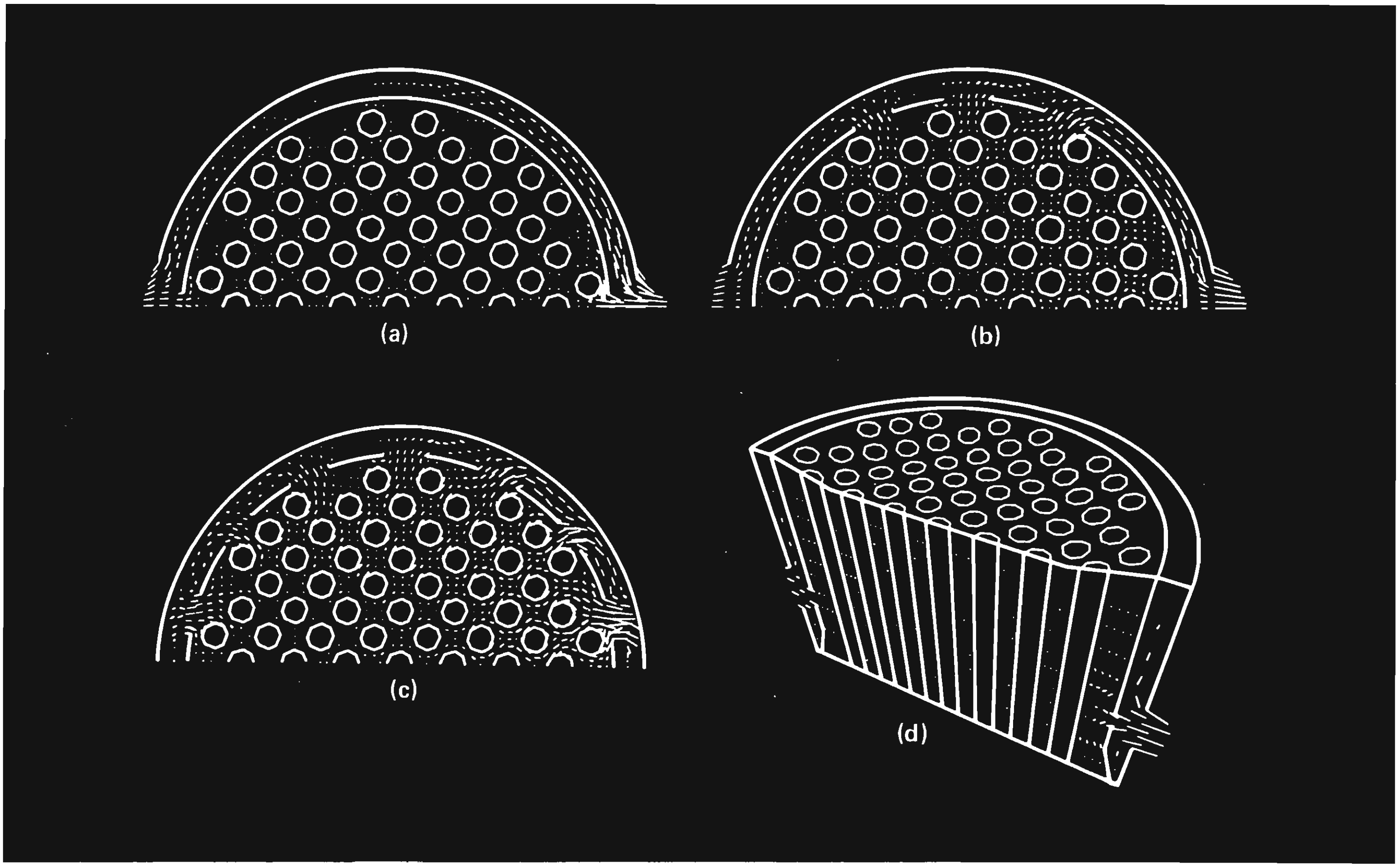


Fig. 8. Calculated velocity vectors showing the flow of water during a depressurization resulting from a break in the outlet pipe on the right. Shown are the vectors at (a) a horizontal level near the bottom of the outlet pipes, (b) at the horizontal level of the sets of three large holes in the plenum cylinder, (c) at the horizontal level of the sets of seven large holes in the plenum cylinder, and (d) across the vertical symmetry plane.

In this analysis, the water is assumed to persist in the liquid state throughout the early, most violent stages of depressurization. This assumption is based on the hypothesis that the initial pressure drop is the largest, and that it brings the fluid nearly to saturation. Subsequent pressure drops to levels low enough to initiate steam formation will be milder. Thus the greatest potential for damage occurs during the single-phase flow.

For damage assessment, it is the structural deformation that serves as the principal result from our analysis. The time histories of stresses and strains experienced by the guide tubes indicate that a break of an outlet pipe may cause significant plastic deformation in a number of the guide tubes.

These preliminary SALE-3D results demonstrate a new capability in modeling complex reactor flows. A more comprehensive program of study and comparisons with a variety of experimental data will be required to verify the code.

NEW CODES FOR FLUID-STRUCTURE CALCULATIONS. Other postulated breaks in a pressurized-water reactor can cause fluid-structure interactions. For example, a break in a steam generator would involve the barrier between the primary and secondary coolant circuits. Analysis of breaks in this structure is in a very primitive stage, but we anticipate that increasingly sophisticated methods will be developed to address such problems.

The analyses described above are carried out with general-purpose codes adapted to specific accidents. The ease of modifying these relatively simple codes permits the users to apply ingenuity and insight in solving particular problems. From a practical point of view, fluid-structure interactions are too complicated and too varied to be analyzed with a single general-purpose code. A more useful computational tool would be a family of versatile fluid and structural codes that can be coupled in various combinations, and indeed this is the direction in which investigators are moving.

Critical Flows in Two-Phase Systems

One of the most important phenomena determining the duration of the depressurization, or blowdown, phase of a large-break loss-of-coolant accident is the rate at which coolant exits from the broken pipe. We know from observation that the flow out the break reaches a maximum value independent of the pressure difference between the inside and the outside of the pipe break, provided that the pressure difference is greater than a critical value. This limiting flow phenomenon is called critical, or choked, flow. It is well understood for single-phase compressible fluids, but, at the time we began our study, thermodynamic models and one-dimensional fluid-dynamic calculations of two-phase critical flow often did not accurately predict the observed data. Calculated values of critical flow velocities were usually too large and had to be multiplied by empirically determined factors known as break-flow multipliers to achieve agreement with measured values. Our studies, based on a two-dimensional theory, show that nozzle geometry and non-equilibrium effects must be included to predict the critical flow velocity accurately.

When a single-phase compressible fluid flows through a nozzle, the critical flow velocity equals the speed of sound at the nozzle throat. The physical explanation is simple: When the fluid is moving with the speed of sound, a downstream pressure disturbance propagates upstream as fast as the fluid is moving downstream, so the net propagation of the disturbance is zero. Therefore, under critical flow conditions, the nozzle throat acts as a barrier to any downstream pressure changes. The limiting flow velocity can be altered only by changing the conditions upstream of the throat.

A vapor-liquid mixture, which is also a compressible fluid, exhibits a similar but much more complicated phenomenon. The critical flow velocity is still the sonic velocity at the throat, but the sonic velocity is affected by vaporization along the accelerating flow path, by the spatial distributions of the liquid and the vapor, and by nonequilibrium effects that occur when the liquid phase superheats because of rapid depressurization. The sonic velocity in a homogeneous two-phase mixture can be far less than the sonic velocity in either of the separate single-phase components. This reduction is attributed to the vapor's acting as a weak spring coupled to the large liquid masses.

TWO-DIMENSIONAL FLUID EQUATIONS. For the relatively high flow rates characteristic of critical flows, we assume a homogeneous modei for a steam-water mixture in which both phases move at equal velocities and are at the same temperature. (Because this temperature is not necessarily the saturation temperature of the mixture, the homogeneous model does not imply thermal equilibrium.) In a two-dimensional flow region of variable thickness, the equations governing the two-phase mixture density ρ, the velocity components u and v, and the internal energy I are

$$\frac{\partial \rho}{\partial t}+\frac{1}{A}\left(\frac{\partial A\rho u}{\partial x}+\frac{\partial A\rho v}{\partial z}\right)=0\ , \tag{2}$$

$$\frac{\partial u}{\partial t}+u\frac{\partial u}{\partial x}+v\frac{\partial u}{\partial z}=-\frac{1}{\rho}\frac{\partial p}{\partial x}\ , \tag{3}$$

$$\frac{\partial v}{\partial t}+u\frac{\partial v}{\partial x}+v\frac{\partial v}{\partial z}=-\frac{1}{\rho}\frac{\partial p}{\partial z}\ , \tag{4}$$

and

$$\frac{\partial \rho I}{\partial t}+\frac{1}{A}\left(\frac{\partial A\rho I u}{\partial x}+\frac{\partial A\rho I v}{\partial z}\right)=-\frac{p}{A}\left(\frac{\partial A u}{\partial x}+\frac{\partial A v}{\partial z}\right)\ , \tag{5}$$

where p is the fluid pressure and A is the thickness of the flow region normal to x and z. These equations are derived from the three-dimensional equations by integrating the latter over the thickness A in the third direction and assuming negligible variation of all dependent variables in this direction. Effects of gravity and wall friction are neglected because of the high flow velocity and relatively short nozzle length.

To compute two-dimensional flow in cylindrical (r,z) geometry, the thickness A and the coordinate x are identified with the radial coordinate r. To compute one-dimensional axial flow in a variable area pipe, A is set proportional to the cross-sectional area of the pipe, u is held identically zero, and all x-derivatives are omitted from the equations. This latter situation arises automatically in the numerical solutions when the finite-difference mesh is defined to be only one cell wide in the x-direction.

Equations 2-5 must be supplemented by an equation for the macroscopic vapor density $\alpha\rho_v$, where ρ_v is the microscopic, or thermodynamic, vapor density:

$$\frac{\partial \alpha\rho_v}{\partial t}+\frac{1}{A}\left(\frac{\partial A\alpha\rho_v u}{\partial x}+\frac{\partial A\alpha\rho_v v}{\partial z}\right)=\Gamma \tag{6}$$

Here, Γ is the rate of production of vapor mass per unit volume.

The form of the phase-change model embodied in Γ is crucial if nonequilibrium effects are to be predicted correctly. Although we can only approximate the microphysical phenomena involved in vaporization, we have been able to develop a nonequilibrium phase-change model that works well for predicting critical flows. This model is discussed later.

EQUILIBRIUM TWO-DIMENSIONAL CALCULATIONS OF CRITICAL FLOW RATES. Using the homogeneous two-phase mixture model described above, we calculated the critical flow rate for a blowdown experiment at the Semiscale test facility. Semiscale is a small-scale version of a pressurized-water reactor primary system for studying loss-of-coolant accidents resulting from the break of a large cooling pipe. In the experiment that we analyzed the pipe break was simulated by a nozzle known as the Henry nozzle (Fig. 9). We used the conditions measured a short distance upstream from the nozzle

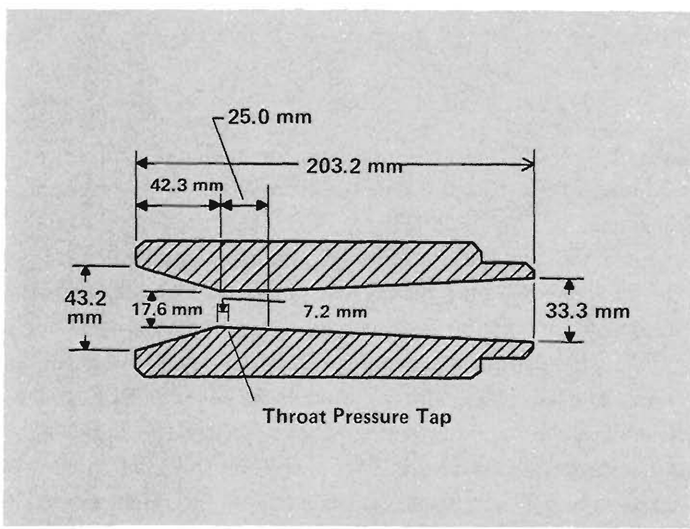


Fig. 9. Design of the Henry nozzle used at the Semiscale facility to simulate pipe breaks. Flow is from left to right.

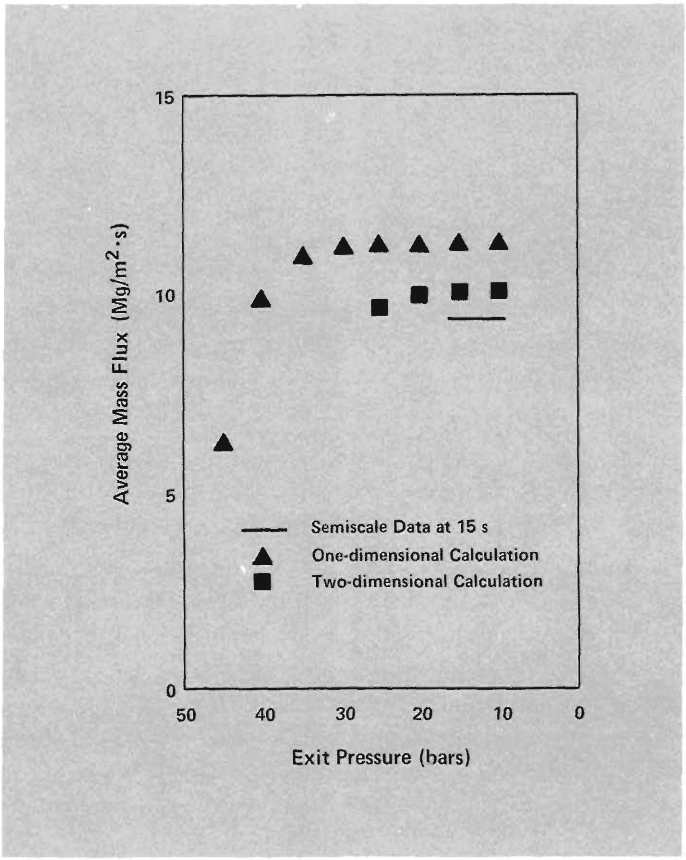


Fig. 10. Measured and calculated average mass fluxes as a function of nozzle exit pressure at 15 seconds into blowdown test S-02-4 at Semiscale.

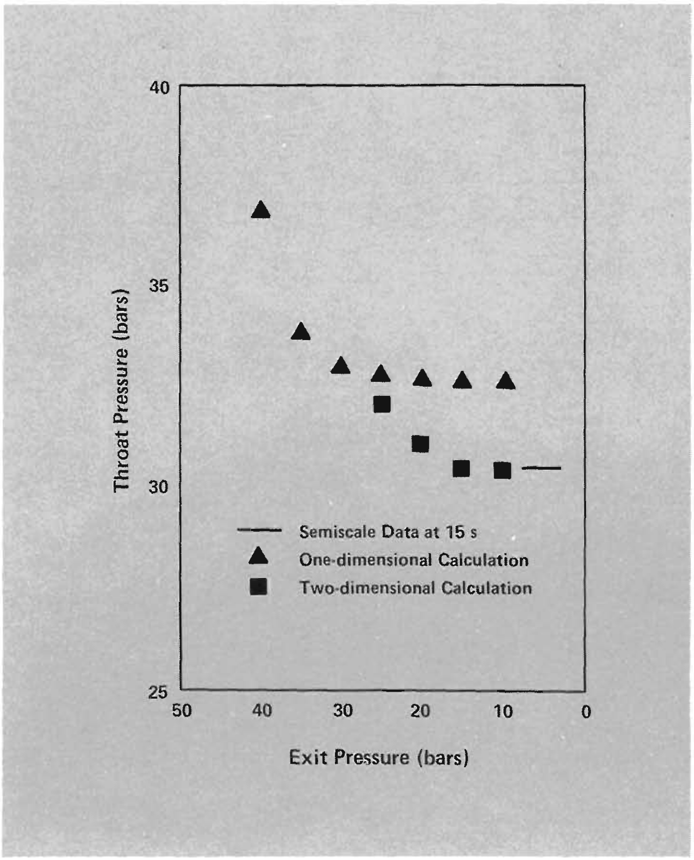


Fig. 11. Measured and calculated pressures at nozzle throat entrance as a function of nozzle exit pressure at 15 seconds into blowdown test S-02-4 at Semiscale.

entrance as boundary conditions for our calculations and solved the fluid equations in the immediate neighborhood of the nozzle.

Our initial calculations involved determining the critical flow rate 15 seconds after blowdown began. At 15 seconds, the vapor volume fraction is fairly large and the flow rate is likely to be independent of the vapor production rate, so we assumed an equilibrium phase-change model. In other words, Γ was chosen large enough to maintain the vapor and the liquid at the saturation temperature for each value of the local pressure. The boundary conditions upstream of the Henry nozzle entrance were 48 bars for the pressure, 534 kelvin for the temperature, and 56 kilograms per cubic meter for the mixture density.

We varied the pressure at the nozzle exit between 45 and 10 bars. For selected pressures in this interval, the computations were carried out until the flow reached a steady state, typically at 8 milliseconds after starting the flow from rest. The computed average mass flux and throat pressure are shown in Figs. 10 and 11. Figure 10 indicates that the flow reaches a limiting value as the exit pressure is reduced. The computed critical flow value is in good agreement with the measurements without the use of a break-flow multiplier or any other adjustment. The corresponding one-dimensional calculations also exhibit a critical flow as the exit pressure is reduced, but the computed mass flux must be multiplied by 0.833 to agree with the data.

To understand the nature of the two-dimensional effects, we

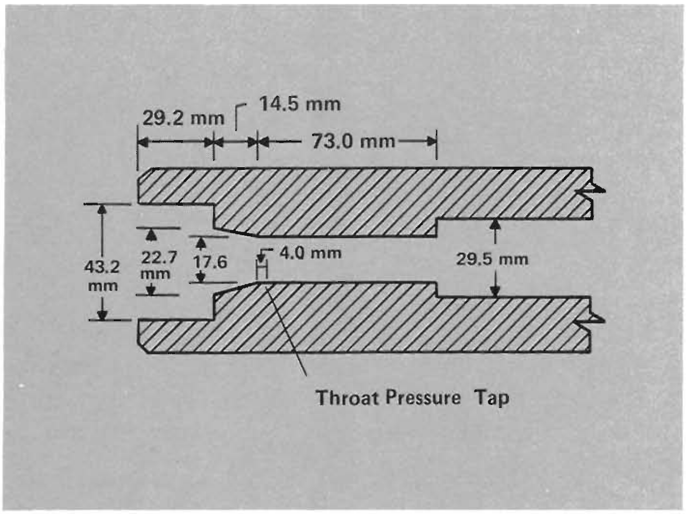


Fig. 12. Design of the nozzle normally used at the LOFT facility to simulate pipe breaks. The LOFT nozzle has a more abrupt throat entrance than does the Henry nozzle. Flow is from left to right.

carried out one- and two-dimensional calculations for a similar experiment in which the Henry nozzle was replaced by the nozzle design used at the LOFT (loss-of-fluid test) facility. Although the abrupt entrance to the throat of the LOFT nozzle (Fig. 12) would seem more likely to exhibit two-dimensional effects than the tapered entrance to the Henry nozzle throat, our one-dimensional results for the LOFT nozzle need only a small correction to agree with the two-dimensional calculation.

We studied the effect of entrance geometry further with a two-dimensional calculation for a Henry nozzle modified so that the entrance to the throat was abrupt rather than tapered. This change in geometry produced only a small change in the mass flow rate and the throat pressure.

Next we investigated the effect of varying the ratio of throat length to throat diameter for the general geometric configuration of the LOFT nozzle. Figure 13 shows the break-flow multipliers required to reach agreement between one- and two-dimensional calculations for various ratios. If the throat length is short relative to its diameter, two-dimensional effects are large. But for ratios greater than about 5, two-dimensional effects are no longer important and the exit flow can be described by a one-dimensional calculation.

A detailed look at the velocity profiles explains this effect.

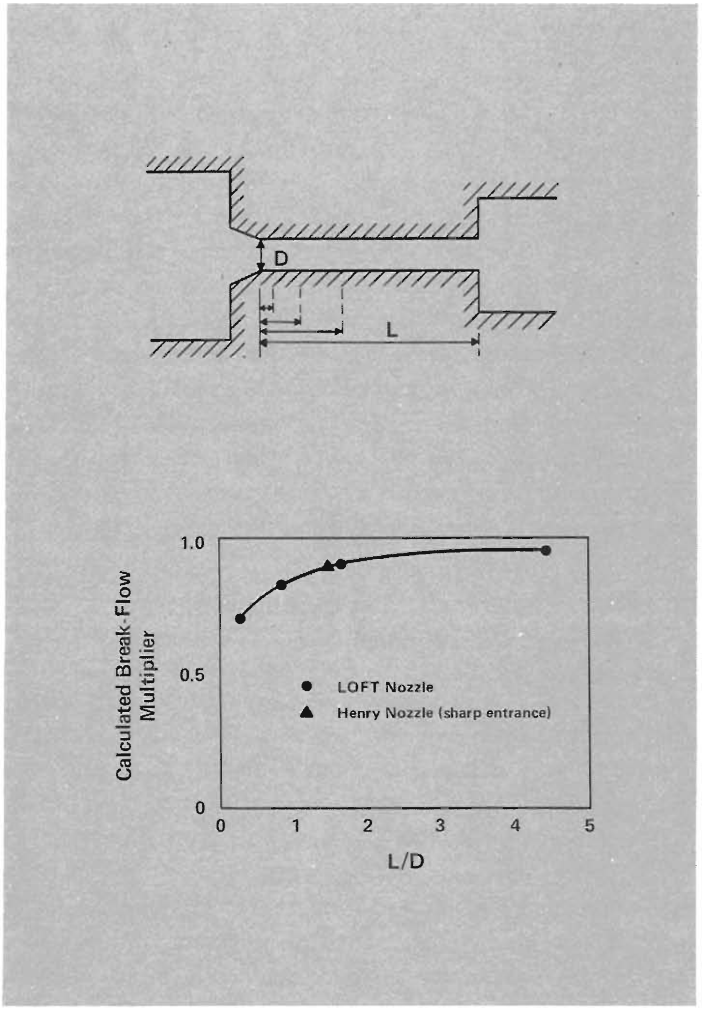


Fig. 13. Effect of the ratio of throat length L *to throat diameter* D *on the calculated break-flow multiplier. Different* L/D *ratios for the LOFT nozzle were obtained by varying* L *as indicated.*

At the throat entrance the radial velocity components are negative and, accordingly, accelerate the central axial velocities. Therefore, a strong radial velocity gradient develops in the entrance region. At a short distance downstream, the radial velocity components become positive and transfer momentum rapidly outward from the center. Here, approximate one-dimensional velocity distributions develop. However, if the throat length is too short for the flow to develop a one-dimensional velocity profile, the one-dimensional models will require a break-flow multiplier to agree with observed data.

EFFECTS OF NONEQUILIBRIUM PHASE CHANGE. The calculations presented so far have corresponded to homogeneous equilibrium phase change. To assess the relative importance of nonequilibrium phase change, we calculated the mass flow rates at the nozzle exit during the first 20 seconds of blowdown using two phase-change models, the equilibrium model described above and a model in which the phase change is zero. Figure 14 shows the calculated values and experimental data for the Henry nozzle. The values were obtained by multiplying the results of a one-dimensional calculation by the calculated break-flow multiplier for the Henry nozzle.

During the first 3 seconds of blowdown the fluid entering the nozzle is single-phase liquid. Its temperature is initially 28 kelvin below the saturation temperature, but, as the pressure decreases, the fluid rapidly reaches the saturation point and becomes superheated. The fact that the data lie between the calculated extremes indicates that nonequilibrium phase change occurs during these first few seconds.

After 3 seconds, when a two-phase mixture enters the nozzle, the calculation with equilibrium phase change agrees with the data. Finally, after 10 seconds when the mixture entering the nozzle is mostly steam, the calculated mass flow rates for both vaporization models coincide with each other and agree with the data. The flow rate is independent of the vapor production rate and is solely determined by the upstream conditions.

To calculate the nonequilibrium effects during the first 3 seconds, we need a detailed model of nonequilibrium vaporization. In a stationary environment, depressurization would lead to vapor production and bubble growth with the growth rate controlled by heat conduction to the bubble surface according to the relation

$$\frac{dr}{dt} = \left(\frac{6}{\pi}\right)\left(\frac{1}{r}\right)\left(\frac{\rho_l}{\rho_v}\right)^2 \alpha_l \left[\frac{C_l\,(T_l - T_{sat})}{L}\right]^2 , \qquad (7)$$

where r is the bubble radius, ρ_l is the microscopic liquid density, α_l is the liquid thermal diffusivity, C_l is the liquid specific heat, T_l is the bulk liquid temperature, T_{sat} is the saturation temperature, and L is the heat of vaporization. During the depressurization and acceleration of the fluid through a converging nozzle, the bubble growth rate varies because T_{sat} and ρ_v depend on the pressure and T_l decreases as heat is used to vaporize the liquid. The instantaneous bubble radius thus depends on the entire bubble history.

The vapor volume fraction α is related to r and N, the number of bubbles per unit of mixture, by

$$\alpha = N\left(\frac{4}{3}\pi r^3\right) . \qquad (8)$$

Combining Eqs. 7 and 8 we derive the following expression for Γ.

$$\Gamma = \rho_v \frac{\partial \alpha}{\partial t} = \rho_l \left(\frac{18}{\pi}\right)\left(\frac{\alpha}{r^2}\right)\left(\frac{\rho_l}{\rho_v}\right) \alpha_l \left[\frac{C_l\,(T_l - T_{sat})}{L}\right]^2 . \qquad (9)$$

For application to the highly dynamic environment of a critical flow, we retain the form of Eq. 9 but choose a liquid thermal diffusivity and bubble radius that reflect the combined

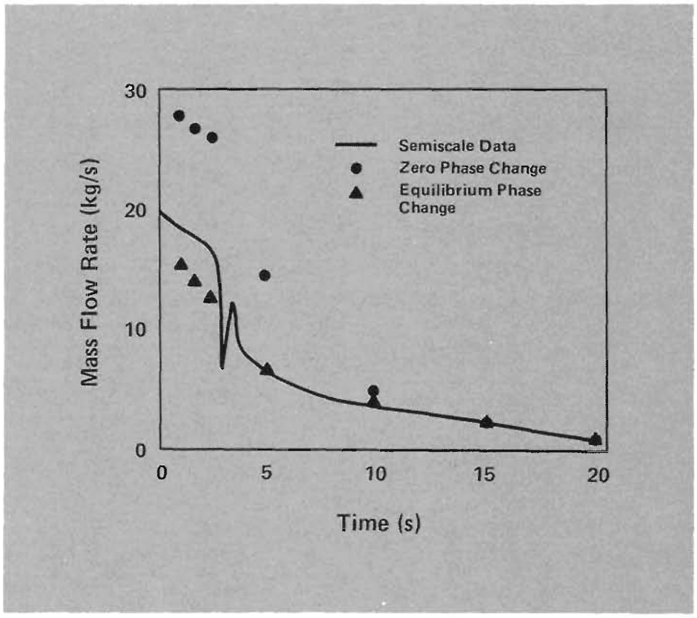


Fig. 14. Measured and calculated mass flow rates during the first 20 seconds of blowdown test S-02-4 at Semiscale. The calculations are based on two phase-change models, an equilibrium model and a model in which the phase change is zero.

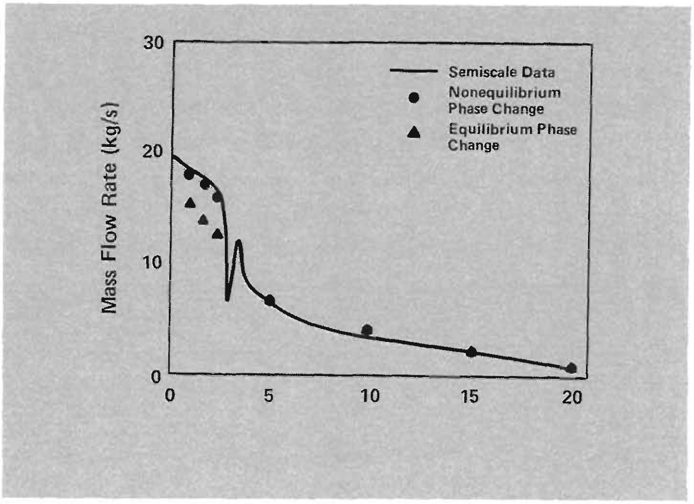


Fig. 15. Measured and calculated mass flow rates during the first 20 seconds of blowdown test S-02-4. The calculations are based on nonequilibrium and equilibrium phase-change models.

effects of relative motion and turbulence. These modifications allow the model to approach the correct limit in a quiescent environment.

In general, there is a spectrum of bubble radii, but we choose the critical radius for bubble breakup to characterize this spectrum. We determine an initial bubble radius by specifying initial values of N and α. The bubbles grow according to Eq. 7 with α_l replaced by $\bar{\alpha}$, a liquid thermal diffusivity enhanced by relative motion and turbulence. Consequently, the bubbles grow faster than the conduction-controlled rate. The bubbles continue to grow until they reach a critical size, determined by a Weber number criterion, and then begin to break up. The Weber number characterizes the competition between the dynamic forces that lead to bubble breakup and the restoring force of surface tension. From this point on, the typical bubble radius is taken as the critical radius and the specified initial number of bubbles no longer plays a role.

The critical radius for bubble breakup is given by

$$r_{\text{critical}} = \frac{2.3\sigma}{\bar{\nu}^2 \; (\rho_l^{\,2} \; \rho_v)^{1/3}} , \tag{10}$$

where σ is the surface tension and $\bar{\nu}$ is the relative speed between the bubble and the surrounding fluid. To include the contribution of local turbulent fluctuations in the liquid to the relative speed we write $\bar{\nu}$ as

$$\bar{\nu} = \beta \nu_l \quad , \tag{11}$$

where ν_l is the liquid speed and β is a function of vapor fraction. We choose values of β consistent with observed turbulent velocity fluctuations, which are generally less than 10 per cent of the mean flow velocity. Toward the middle vapor-fraction range, β increases because of increased turbulent mixing from the higher shear flow associated with thinning liquid sheets. The increase in β may also result from an increase in mean relative velocity.

The enhanced liquid thermal diffusivity $\bar{\alpha}$ that replaces α_l in Eq. 9 is

$$\bar{\alpha} = \alpha_l + Br\bar{\nu} \; , \tag{12}$$

where B is an empirically determined dimensionless constant. The value of $B = 0.1$ matches the flow rate data for the Semiscale tests. The range of applicability of this value can only be accurately established after extensive data comparisons.

In Fig. 15, the nonequilibrium results for the mass flow rate during blowdown are compared with the data for the Henry nozzle from Fig. 14. The nonequilibrium results agree very well with the measured mass flow rate during the entire period of blowdown. However, at early times the calculated throat pressures (Fig. 16) are higher than the measured wall pressures at the throat entrance. This difference is probably caused by a rarefaction region in the proximity of the corner that is not modeled by the one-dimensional calculation.

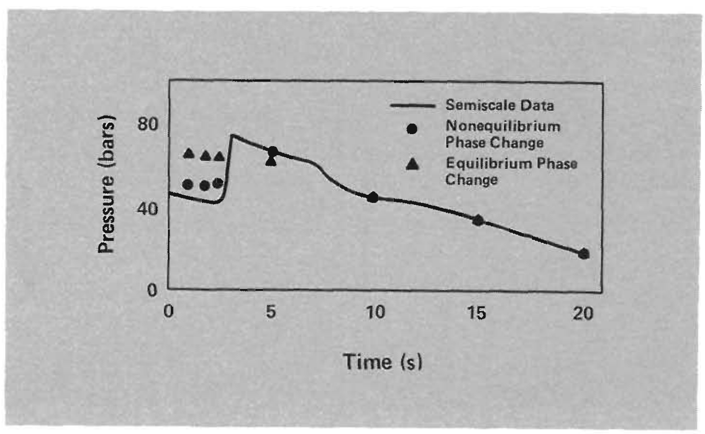


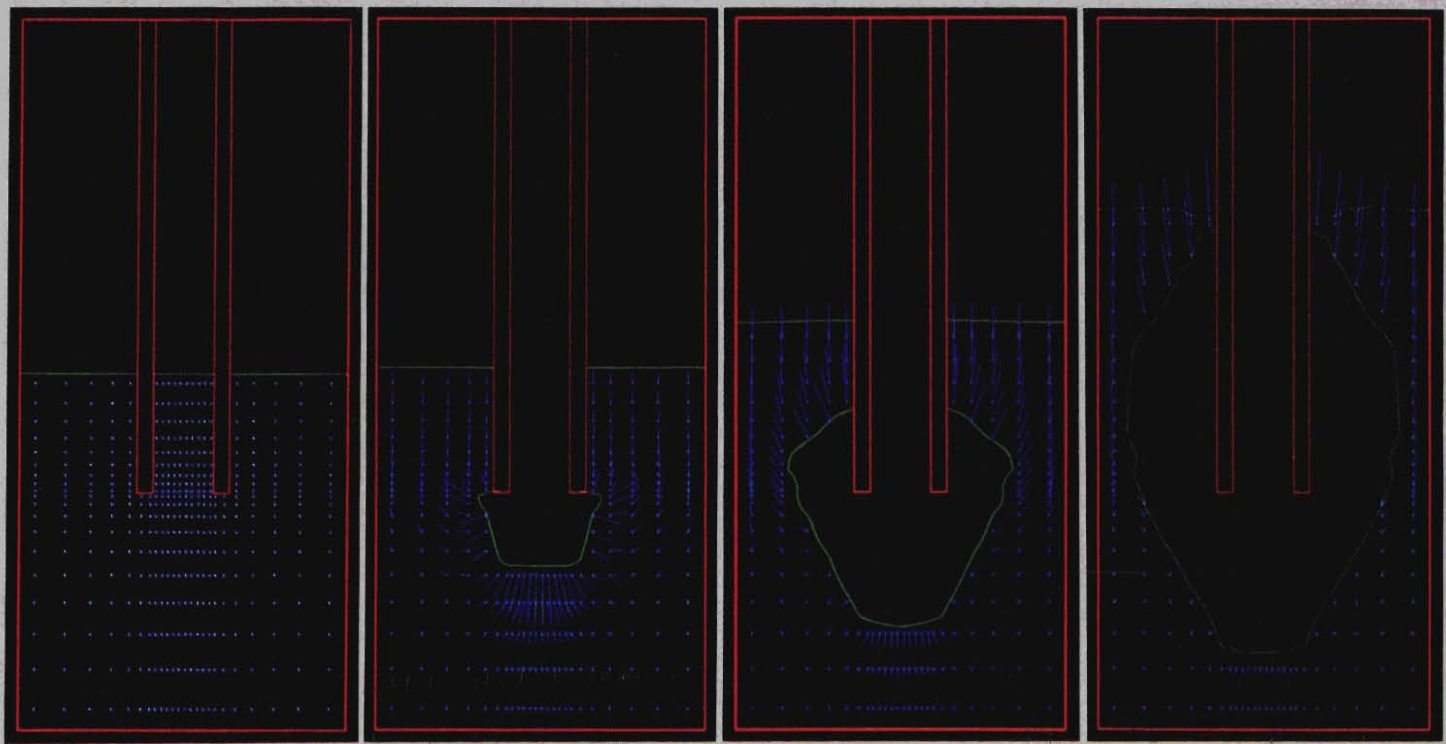
Fig. 16 Measured and calculated pressures at entrance to nozzle throat during the first 20 seconds of blowdown test S-02-4 at Semiscale. The calculations are based on non-equilibrium and equilibrium phase-change models.

In addition to these small-scale tests, the nonequilibrium model has been tested against data obtained from the full-scale critical flow project at the Marviken facility in Sweden, from the low-pressure MOBY DICK loop at the Nuclear Studies Center in Grenoble, France, and from the low-pressure critical flow loop at Brookhaven National Laboratory. These tests involved fluid pressures from about 90 bars down to slightly greater than 1 bar. Pipe diameters ranged from 75 centimeters down to a few centimeters. We encountered no scaling problems in going from small- to full-scale geometries because the nonequilibrium model is based on local flow and thermodynamic conditions.

This study has proved to be an important contribution in predicting two-phase homogeneous critical flows through

SUPPRESSION-POOL DYNAMICS IN BOILING WATER REACTORS

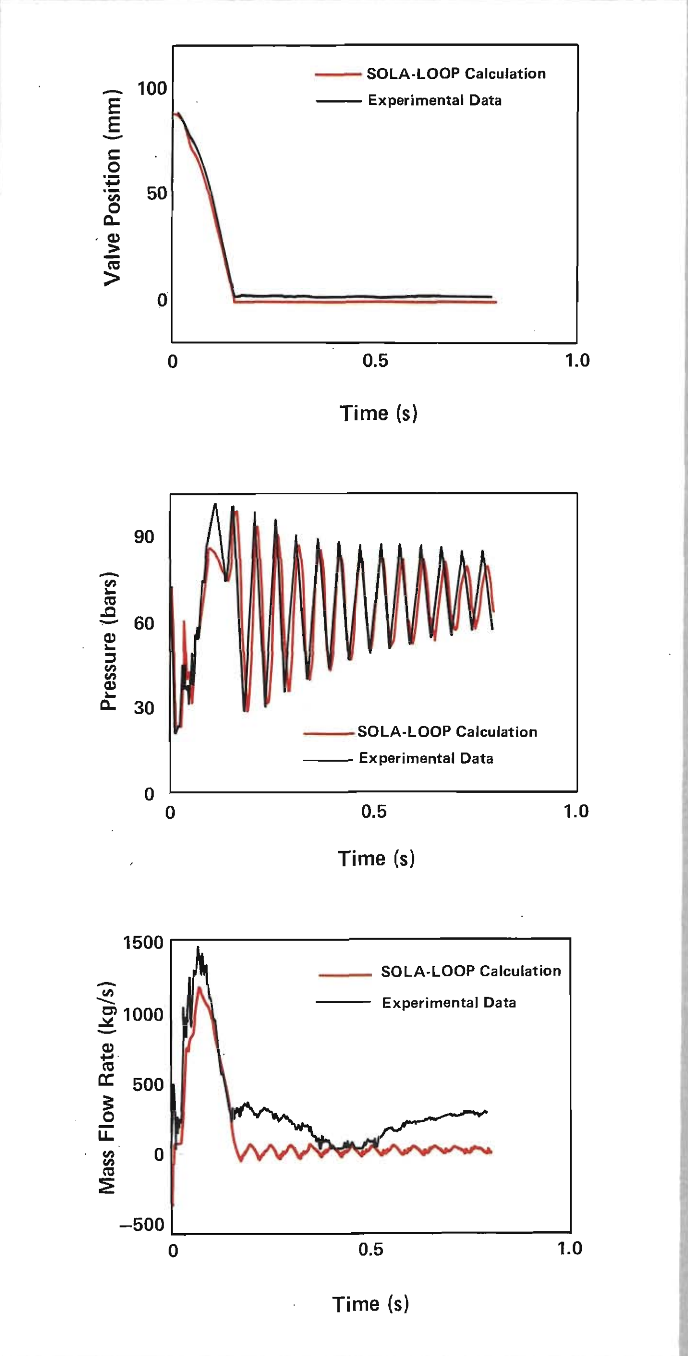
The suppression pool of a boiling-water reactor system is designed to condense steam that might fill the containment building following the break of a coolant pipe. The vents leading into the pool must be cleared of air before the steam can enter the pool. To study this process, we used a small-scale model consisting of a cylindrical container with a single axisymmetric vent entering into the water pool. The hydrodynamic phenomena associated with vent clearing were simulated with the SOLA-VOF code. Typical computed free-surface configurations and velocity vector fields are shown. Water velocities are indicated in blue, the water/gas interface in green, and the gas as black voids. The final figure in the sequence shows the high-pressure gas from the vent about to break through the water surface.



nozzles. We have shown that two-dimensional geometric effects not accounted for in one-dimensional calculations reduce the critical flow rates and therefore extend the duration of blowdown. We have also shown that nonequilibrium effects reduce the duration of blowdown because they increase the sound speed and therefore the critical flow rates.

The problems discussed above are only a small sampling of the component studies carried out in the Fluid Dynamics Group. To illustrate the broad applicability of the numerical techniques developed for reactor safety, four other studies are described briefly in the accompanying illustrations. One concerns liquid-metal fast breeder reactors, and the others deal with pressurized-water reactors. ■

THE WATER-HAMMER EFFECT



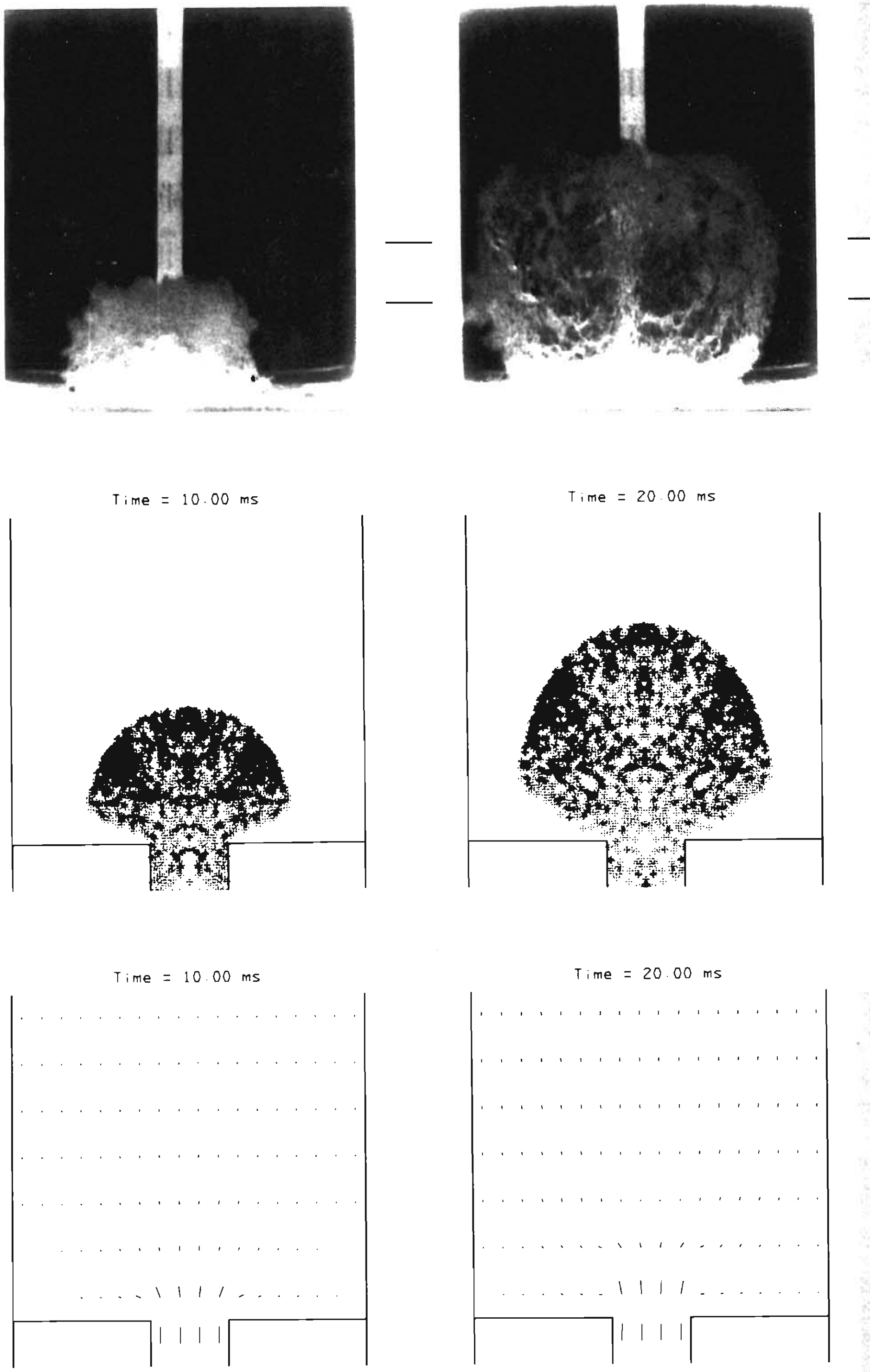
A few commercial pressurized-water reactors are equipped with valves in each primary loop that close automatically and rapidly if a break should occur in the loop. The function of these loop isolation valves is to limit the loss of coolant and, hence, the escape of radioactive materials to the containment building.

The rapid closing of the valves can, however, produce large pressure oscillations within the pipe, the so-called water-hammer effect. The forces produced by the break itself (pressure-release waves) and by the water-hammer effect can lead to considerable stresses on the valve, pipe, and supports.

We calculated the flow in the pipe network of the West German Speisewasserrückschlagventils (SRV) 350 blowdown experiments with SOLA-LOOP, a nonequilibrium, drift-flux code for calculating two-phase flow in pipe networks. The model for the dynamic behavior of the valve is given by $m(d^2x/dt^2) = \sum F$, *where* x *is the valve position and the forces* F *are the viscous drag, the pressure force, the damping force, the external actuating force and/or spring force, and the gravitational force. SOLA-LOOP supplies time-dependent fluid velocities, densities, and pressures to the valve model, and in return, the valve model calculates a time-dependent valve position that is used to determine the resistance to flow through the valve.*

Shown here are measured and calculated time histories for the valve position (top), the pressure just upstream of the valve (middle), and the mass flow rate just upstream of the valve (bottom). The changes in slope of the valve position curve at 75 millimeters (0.03 second) and at 45 millimeters (0.1 second) correspond to pressure spikes during the valve-closing portion of the experiment ($\leq$*0.15 second). After the valve closes, the water-hammer effect (1/4 period pressure waves) are set up in the isolated pipe between the closed valve and the reactor vessel. These waves decay in time, but their large amplitude may create damaging stresses in certain structural components. The mass flow rate increases as the blowdown commences and then decreases as the closing valve creates flow resistance. Although the measured valve position indicates that the valve is closed, the measured mass flow-rate data indicate some leakage through the valve between 0.15 and 0.4 second and a rebounding of the valve head at approximately 0.5 second.*

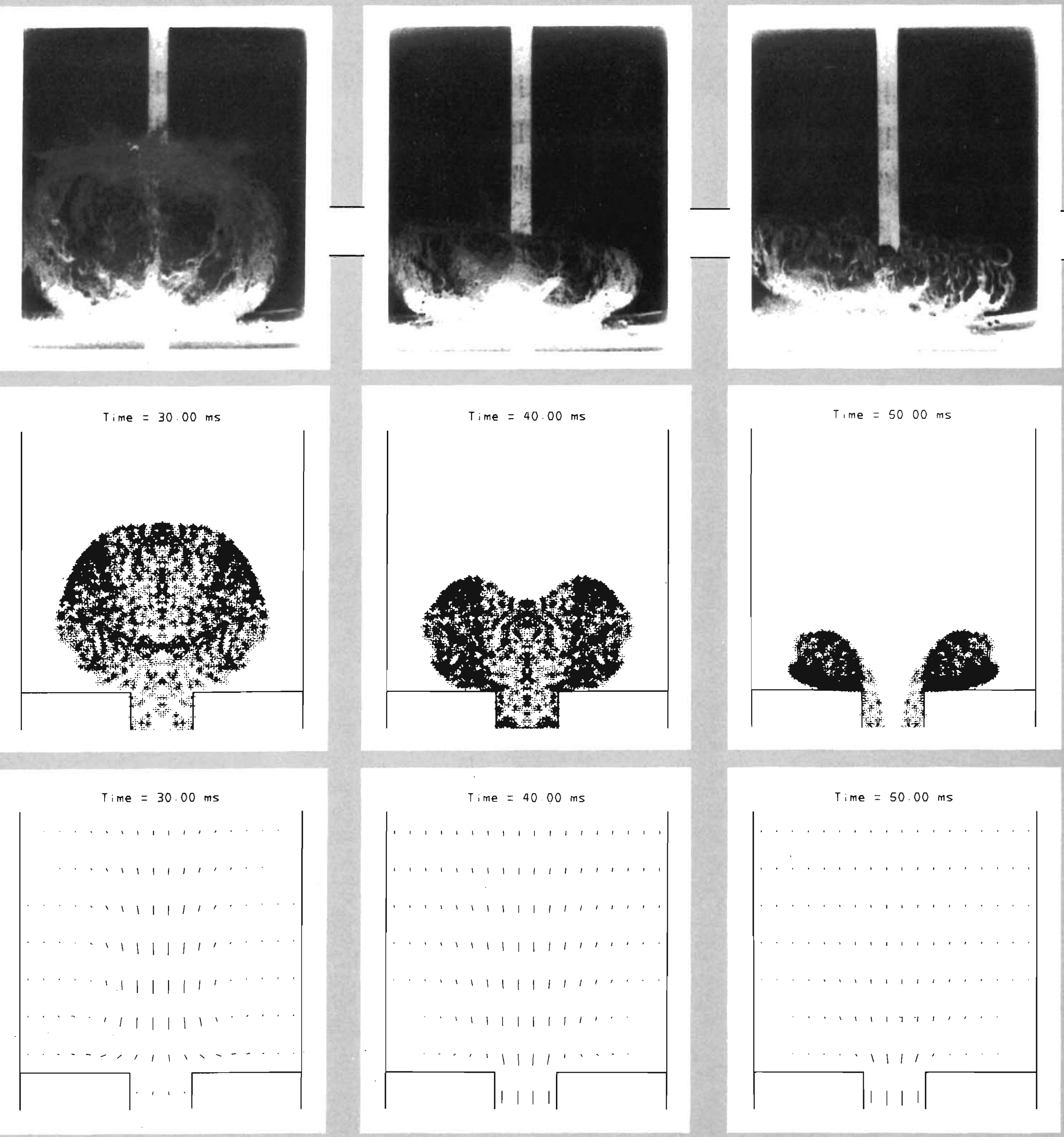
CORE-BUBBLE DYNAMICS FOR BREEDER REACTOR ACCIDENT ANALYSIS



Safety analysis of the liquid-metal fast breeder reactor currently focuses on hypothetical core-disruptive accidents. In one hypothetical scenario, a sudden increase in fuel temperature due to a power burst causes the fuel to melt and vaporize at very high pressures. The expanding bubble of fuel vapor can then vent through the upper core structure into the sodium pool that covers the core region.

The dynamics and energy yield of this bubble ejection must be determined before we can assess the mechanical work that would be done by the sodium pool on the vessel head. The dynamics of the bubble expansion have been simulated by experiments in which a chamber filled with high-pressure air is ruptured and the resulting high-pressure jet of air expands into a water-filled chamber above. Shown here are frames (courtesy of Argonne National Laboratory) from a high-speed motion picture of one experiment and corresponding plots of marker particle configuration and velocity vector field calculated at Los Alamos.

As a diaphragm to the high-pressure chamber is ruptured, the surge of air increases the pressure in the water-filled



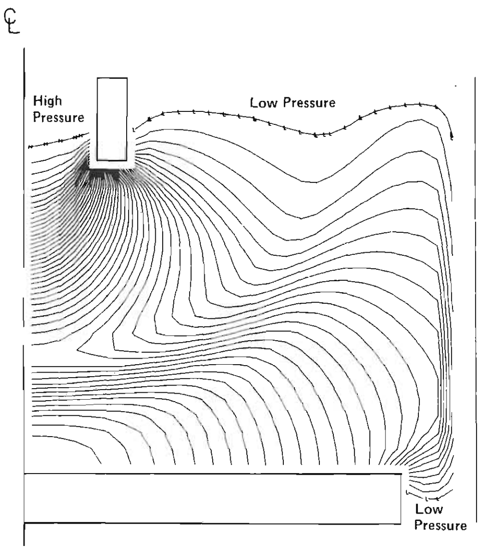
hamber. The momentum imparted to the water leads to an verexpansion of the air and a subsequent drop in bubble ressure. The pressure continues to decrease until 28 milli-econds when the bubble reaches its maximum volume. The ubble begins to collapse into the toroidal shape shown at 40 nd 50 milliseconds. During the collapse the bubble pressure ncreases from the downward-directed momentum of the water. Beyond 50 milliseconds the pressure tends to equilibrate, and he bubble breaks up under the action of turbulence and uoyancy.

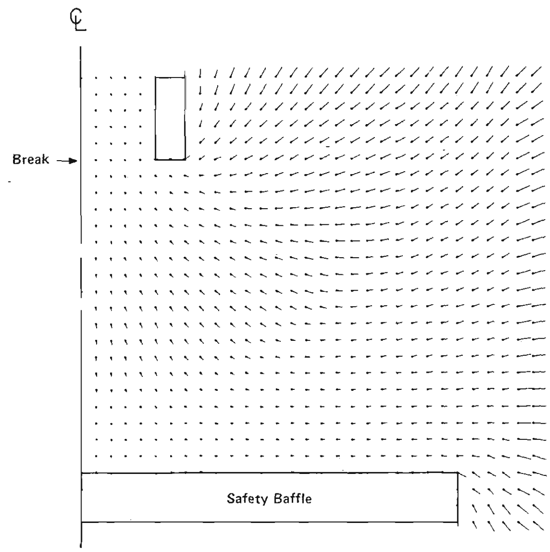
The velocity vector field shows a spherical distribution at 10 and 20 milliseconds. At later times, the bubble collapse is evidenced clearly by the reversal of the velocity vectors and the secondary-flow vortex pattern set up between the centerline and outside boundary. The vortex becomes smaller until at 50 milliseconds it is isolated in a corner with most of the velocity vectors directed toward the lower chamber opening.

STEAM-WATER JET IMPINGEMENT ON REACTOR STRUCTURES

Other Studies

Nuclear power plant structures and components must be designed to withstand the impingement of a steam-water jet released from a hypothetical broken coolant pipe. We performed a two-phase calculation in cylindrical symmetry to simulate this interaction. Shown here are the velocity vector plot (top) and pressure field plot (bottom) for a high-pressure jet of hot water issuing from a break and impinging on a safety baffle. The velocity vector plot shows that the flow accelerates as it leaves the pipe. The acceleration is caused by flashing of the water to steam as the pressure drops below saturation pressure. The pressure contours show the pressure dropping rapidly as the flow exits from the pipe and then rising sharply near the center of the baffle. Proper design of safety baffles and pipe restraints requires accurate computation of this pressure field.





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Acknowledgment

Other members of the Fluid Dynamics Group have played major roles in the studies described here. The authors would particularly like to acknowledge the contributions of Ronald N. Griego, Francis H. Harlow, Cyril W. Hirt, Bryan A. Kashiwa, Billy D. Nichols, William C. Rivard, Hans M. Ruppel, Leland R. Stein, and Martin D. Torrey.

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Bart J. Daly earned his Bachelor of Arts in geology and mathematics from the University of Wyoming in 1950 and his Master of Arts in mathematics from Arizona State University in 1960. In the intervening years he worked in the field of geophysical oil exploration. He has been in the Fluid Dynamics Group since joining the Laboratory in 1960. He has developed and applied numerical methods for fluid-dynamics studies, particularly in the areas of hydrodynamic instability, turbulence modeling, multifluid exchange processes, and reactor safety research. Currently, he is the Fluid Dynamics Group's project leader for Nuclear Regulatory Commission programs.

John K. Dienes received a Bachelor of Arts in mathematics from Pomona College and a Ph.D. in applied mechanics from the California Institute of Technology, where he was a Gillette Fellow. He worked in the aerospace industry on aeroelastic and control problems until 1963, when he joined General Atomic Company to participate in the Orion Program, a system for interplanetary nuclear-pulse propulsion. There he became acquainted with the fluid-dynamics codes developed at Los Alamos. Since joining the Laboratory's Fluid Dynamics Group in 1975, he has been interested in the application of these codes to a variety of problems, including fluid-structure and fuel-coolant interactions in nuclear reactors. He has also been involved with impact and explosive cratering, particularly in the development of constitutive laws for the deformation and fragmentation of solids at extreme pressures and strains, and has published over a hundred articles and reports in this field.

John R. Travis earned his Bachelor of Science with honors from the University of Wyoming in 1965 and received his Ph.D. from Purdue University in 1971. During his graduate studies, he was the recipient of an Advanced Studies Program Fellowship at the National Center for Atmospheric Research in Boulder, Colorado, where he conducted the research for his doctoral dissertation in computational fluid dynamics. From 1971 to 1973 he was with Argonne National Laboratory, where he was involved in the development of the fast reactor safety analysis code SAS. He joined the Laboratory's Chemistry-Materials Science Division as a Staff Member in 1973, where he developed a numerical model for predicting the redistribution within the atmospheric boundary layer of particulate contaminants from soil surfaces. In 1974 he joined the Theoretical Division's Fluid Dynamics Group to develop computational fluid-dynamics models for investigating reactor safety problems of specific interest to the Nuclear Regulatory Commission.

THREE MILE ISLAND
and Multiple Failure Accidents

Photo by Dirck Halstead, Gamma-Liaison Photo Agency by John R. Ireland, James H. Scott, and William R. Stratton

The events at Three Mile Island beginning on March 28, 1979 caught everyone by surprise, including the safety analysts at the national laboratories. A serious accident involving damage to the reactor core was totally unanticipated. The general confusion during the crisis was evident to everyone and the need for better operator training and emergency planning has been well publicized. But the attempt by research scientists to help during the accident and their subsequent efforts to determine what had happened and to help prevent such accidents in the future are less well known.

The accident (Fig. 1) began during attempts to unclog a pipe leading from the demineralizer in a secondary loop of the reactor. A combination of malfunctioning valves in the demineralizer and blocked valves in a backup safety system stopped the flow of feedwater to the steam generators. The turbine tripped automatically, and the reactor scrammed shortly thereafter. With no heat removal through the steam generators, the primary system pressure rose. The pilot-operated relief valve opened to reduce the pressure, and, unbeknownst to the operators, it stuck in the open position and remained in that state, undetected, for about 150 minutes. During that time, the resulting loss of coolant and pressure decrease in the primary system caused a buildup of steam. Then, when the primary pumps were turned off, steam separated from the coolant and continued to build up in the reactor vessel until it surrounded the upper part of the core. Because slowly moving steam is a poor coolant, the core temperature rose and the cladding around the fuel began to fail. The loss of coolant was finally halted, but the damage continued until about 200 minutes into the accident when the emergency cooling pumps were turned on at full throttle and reflooded the core. At that time the core was severely damaged, and the primary system contained large quantities of steam and hydrogen that impaired the flow of coolant through the core. The operators realized that the core may have been uncovered, and throughout the first day they struggled to establish stable conditions.

The seriousness of what had occurred was not generally realized until late the next day when a pressure spike on the monitor printout from the previous day gave evidence that hydrogen had burned inside the containment building. Evidently, severe overheating of the core had caused the cladding to react with steam and produce large amounts of hydrogen, some of which escaped to the containment through the open relief valve. The discovery led to the frightening, but perhaps unwarranted, concern about a possible hydrogen explosion in the reactor vessel.

At this point the research division of the Nuclear Regulatory Commission began calling the national laboratories, including Los Alamos.

The scientists were as unprepared as the immediate participants to handle the ongoing crisis. There were no sophisticated computer

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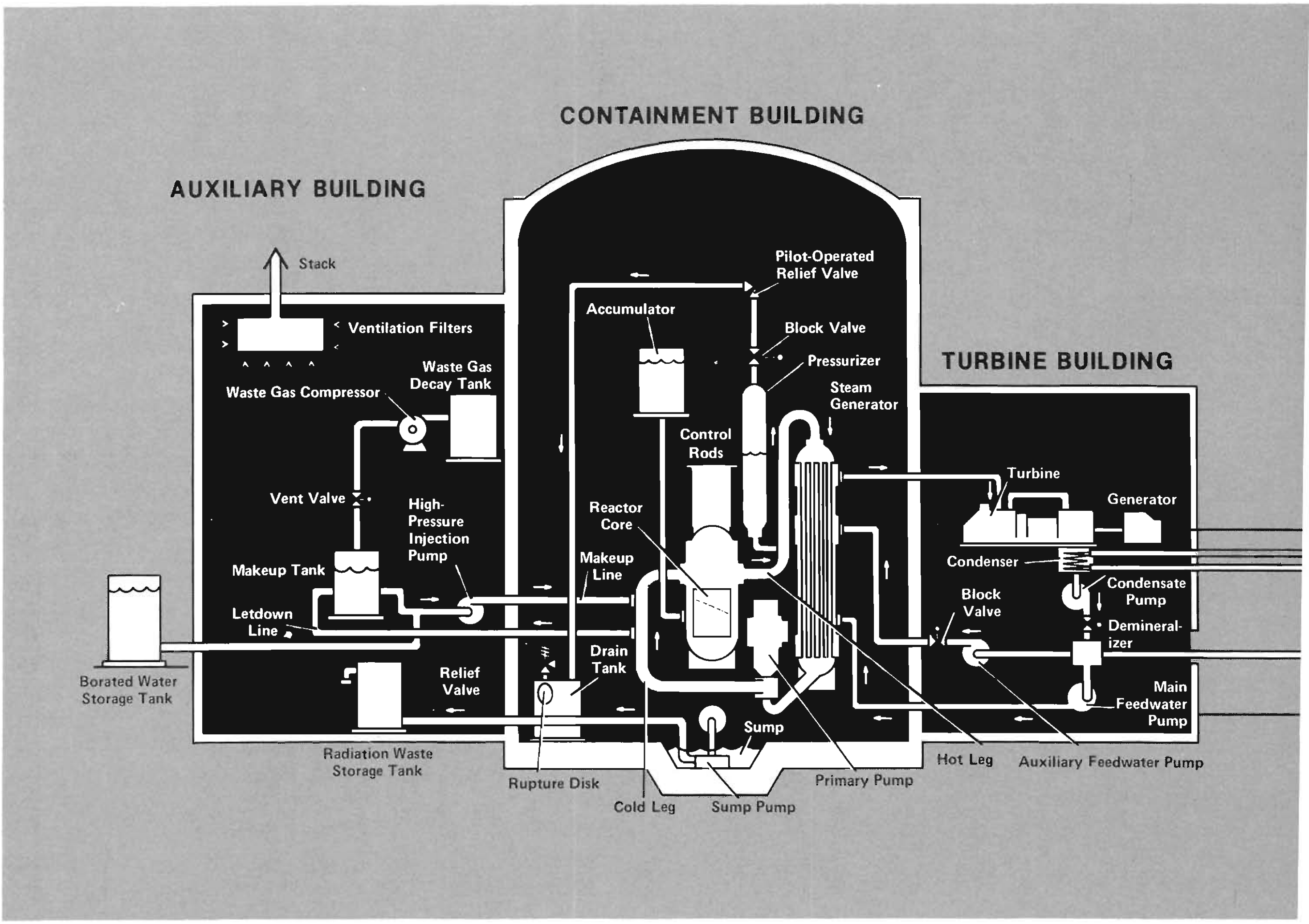
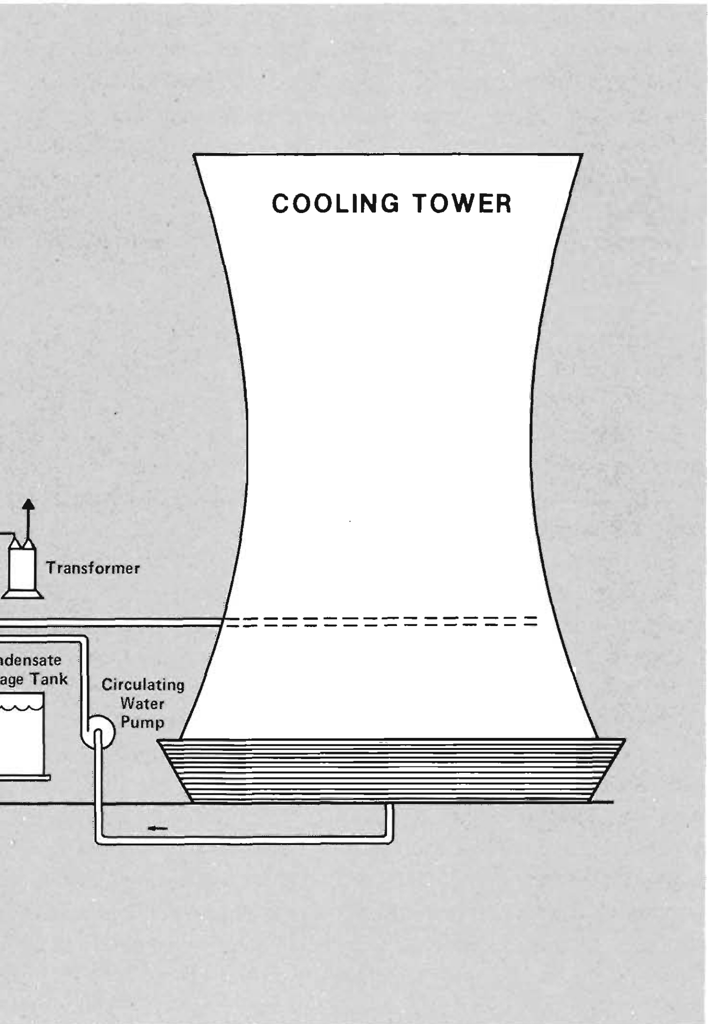


Fig. 1. This diagram of the Three Mile Island Nuclear Power Station's Unit 2 reactor system shows one of its two primary coolant loops and all other system components important to the accident. The time sequence outlined below includes some system responses that are known only from later analyses.

ELAPSED TIME[a]	SYSTEM RESPONSE or OPERATOR ACTION
0 (04:00:37)	Feedwater pumps trip. Turbine trips automatically. Auxiliary feedwater pumps activate, but valves in this line are closed. Primary system pressure increases as heat exchange in the steam generator decreases.
6 s (04:00:43)	Pilot-operated relief valve on the pressurizer opens to relieve excess pressure. Vented steam flows to the drain tank in the containment building.
10 s (04:00:47)	Reactor scrams automatically because of high pressure signal, and nuclear heat generation decreases to decay heat only. Primary system pressure decreases.
16 s (04:00:53)	Pilot-operated relief valve fails to reclose although operators receive information to the contrary. Coolant escapes through the stuck-open valve to the drain tank.
2 min (04:02:37)	Pressure falls to point where high-pressure injection system activates automatically to compensate for coolant loss through the stuck-open valve.
4 min (04:04:37)	Pressure-relief valve on drain tank opens. Some coolant, which is (as usual) very slightly radioactive, escapes from the tank to the containment, collects in the sump, and is pumped to storage tanks in the auxiliary building.



5 min (04:05:37)	High water level in the pressurizer leads operators to throttle high-pressure injection system and drain water through the letdown line. After this time, emergency coolant flow is insufficient to balance the losses through the pilot-operated relief valve and the letdown line.
6 min (04:06:37)	Primary system pressure falls to point at which the coolant beings to boil.
8 min (04:08:37)	Operators open closed valves in auxiliary feedwater line, but coolant loss, pressure decrease, and steam formation continue. Operators are at a loss to understand what is going on.
15 min (04:15:37)	Drain tank ruptures and more coolant escapes to the containment and is pumped to the auxiliary building.
30 min (04:30:37)	Auxiliary building storage tanks overflow. Some radioactive materials escape to the environment through the building's vent stack.
1h, 13 min (05:13:37)	Operators turn off primary pumps in B loop because the steam in the system causes them to vibrate excessively.
1h, 40 min (05:40:37)	Operators turn off primary pumps in A loop for the same reason. With no forced circulation, steam and water separate in the core. Cooled only by steam along some portion of their length, the fuel rods begin to heat.
2h, 20 min (06:20:37)	Operators close a block valve upsteam of the pilot-operated relief valve. Although this actions halts the loss of coolant, it also halts the cooling provided by steam escaping from the pressurizer. The fuel rods heat more rapidly, and eventually cladding and steam react and produce hydrogen. Cladding failure and structural damage to the core begin to occur.
2h, 54 min (06:54:37)	Operators restart a primary pump but turn it off after 19 minutes because it is not running properly.
3h, 12 min (07:12:37)	Pressurizer block valve is opened and then closed 5 minutes later. Steam flow out the block valve provides some core cooling.
3h, 20 min (07:20:37)	Operators increase high-pressure injection flow for a few minutes. This action probably covers the core with water, but coolant flow is impeded by steam and hydrogen in the primary system and by the core's altered configuration. To collapse the steam bubbles, operators alternately inject water through the high-pressure injection line and vent excess pressure through the pilot-operated valve. These "feed and bleed" maneuvers are hampered by the noncondensible hydrogen.
8h, 20 min (12:20:37)	Operators note a pressure spike on a graph of the pressure within the containment building, but do not recognize the spike as evidence of a hydrogen burn in the containment.
15h, 50 min (19:50:37)	Operators activate a primary pump and achieve forced circulation. The system reaches a relatively stable condition, but it is not until almost a month later that "cold shutdown" is effected.

[a]*The TRAC analysis used the times given here, which may differ from those given in other reports of the accident.*

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tools to model an accident involving core damage. When Los Alamos was asked to estimate the extent of damage to the core and the amount of hydrogen that might have been produced, the scientists had to resort to hand calculations. They were also asked to use TRAC to estimate the amount of water that had been lost from the primary system, but without detailed specifications of the Three Mile Island plant, computer calculations were no better than rough estimates.

Although Los Alamos scientists and others around the country responded with the urgency required by the situation, it is clear that their help had little impact on the course of the accident. It was the operators and engineers at the site who, through skillful manipulation of the cooling systems, reduced the steam and hydrogen bubbles in the primary system and brought the reactor into a stable cooling mode with no major radiation release.* They and the in-depth safety systems must be given the credit for bringing the accident to a close with no injuries to the public.

It appears that accidents must be managed by people at the site who are familiar with the plant and the details of the immediate situation. The role of the laboratories is to work on preventive measures so that when something does go wrong there is a storehouse of knowledge that can guide the management of the accident.

TRAC and TMI

The first job after the Three Mile Island accident was to understand what had happened and why. Many Laboratory personnel lent their technical expertise to the investigations that followed,** but the most substantial contribution was a detailed calculation of the conditions inside the reactor during the early stages of the accident. Los Alamos had the only computational tool available to model the thermal hydraulics of the accident in a realistic fashion, the state-of-the-art systems code known as TRAC.

Because the current version of TRAC (TRAC-P1A) did not include the effects of altered core geometry or of noncondensible gases (such as hydrogen), the Laboratory was asked by the President's Commission on the Accident at Three Mile Island for an analysis covering only the initial 3 hours of the accident before substantial core damage occurred. Los Alamos was also asked for an estimate of the total core damage up to 3.5 hours based on calculated temperatures and pressures and for analyses of postulated accident variations to determine the impact of operator actions on the course of events. This information was submitted to the President's Commission and to the Nuclear Regulatory Commission's Special Inquiry Group in September 1979.

The Los Alamos calculations were the first calculations of the accident and also the first test of TRAC on a full-scale system. These early results have not changed substantially over the last two years and agree to a large extent with later independent analyses.

It is generally agreed that the severity of the Three Mile Island accident was due in large part to inappropriate operator actions and inadequate emergency operating procedures. For the purpose of analysis, however, it may be characterized dispassionately as a small-break loss-of-coolant accident with degraded emergency coolant injection.

Analysis of such a transient with TRAC posed only one difficulty. TRAC was specifically designed for analysis of design-basis loss-of-coolant accidents that last, not several hours, but several minutes. For analysis of short-duration transients, a reactor system is divided into a large number (about 750) of fairly small computational cells. To ensure stability and accuracy of the sophisticated numerical methods included in TRAC, small time steps (about 5 milliseconds) must accompany small computational cell lengths. But small time steps would imply unreasonably long computing times for analysis of a 3-hour transient. Therefore, the TRAC analysis of the Three Mile accident was based on a model of the Unit 2 reactor (Fig. 2) consisting of less than 100 cells.*** It was not certain beforehand whether this small number of cells would yield acceptable results. However, the model was judged adequate on the basis of a TRAC steady-state calculation that produced results in good agreement with plant data. These results were used as initial conditions for the transient calculation.

Other input to the transient calculation included a sequence of events (initiated by operators or by plant controls) and boundary conditions specifying the variation during the transient of reactor power, primary pump speed, high-pressure injection flow, steam generator feedwater flow, and back pressures on the pilot-operated relief valve and the steam generator lines. Because the available plant

**The Department of Energy Emergency Response Teams made an accurate measurement of the escaped fission products on the afternoon of the first day. The total radiation released during the accident resulted in an average exposure of 1 millirem to persons living within 50 miles of the plant and 6.5 millirems to persons within 10 miles. The sidebar "Good News about Iodine Releases" discusses some important findings about radiation releases during the accident.*

***See sidebar "Los Alamos Assistance to TMI Investigations."*

****Even so, about 15-20 hours on a CDC-7600 computer were required for analysis of the accident and a total of about 200 hours for analyses of both the accident and its postulated variations.*

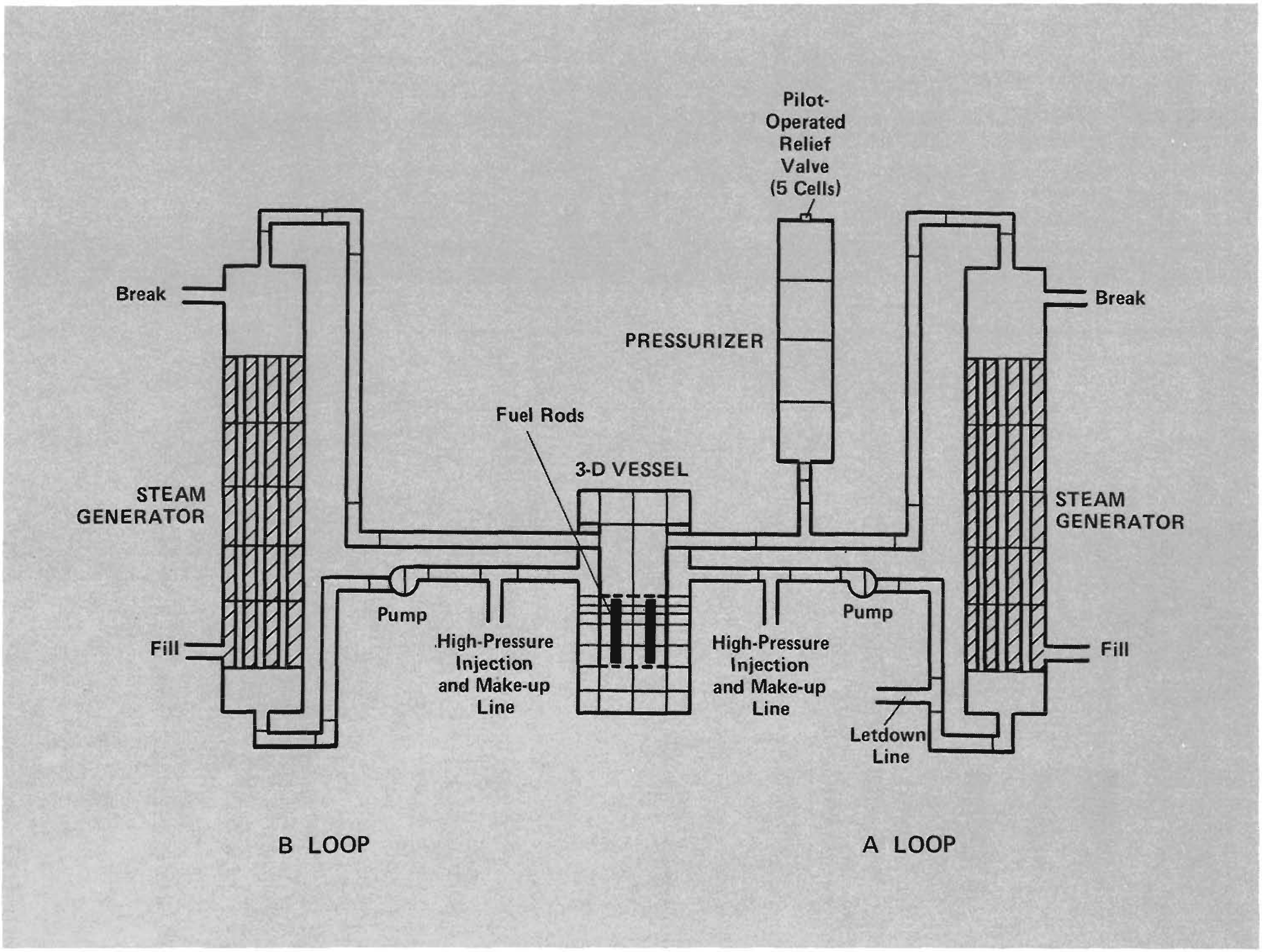

Fig. 2. Schematic of the TRAC computing mesh for the two primary coolant loops of Three Mile Island Unit 2 reactor. To reduce the number of cells, the mesh represents the two cold legs in each loop by a single cold leg. The reactor vessel mesh, divided into nine axial levels, includes four lumped fuel rods to model heat transfer between fuel rods and fluid. (The actual core contained 177 fuel-rod assemblies, each with 208 fuel rods.) The flow through the pilot-operated relief valve and the upper part of the pressurizer was calculated by using very fine noding and the fully implicit hydrodynamics option. Known system conditions were used as boundary conditions for the once-through steam generators. The high-pressure injection and letdown lines were modeled as positive and negative flow boundary conditions, respectively. Neither the accumulators nor the action of heaters and sprayers in the pressurizer were modeled.

data were incomplete, reasonable assumptions had to be made for a number of variables, including the flow-rate histories for the high-pressure injection and letdown systems. (Water is removed from the primary system through the letdown system for purification or to reduce the primary system pressure or the pressurizer water level.)

Results of the transient calculation are displayed in the sidebar "TRAC Analysis of the Three Mile Island Accident." Calculated values for the primary system pressure, primary coolant temperature, and pressurizer water level agree well with the available plant data and are helpful in reconstructing the course of the accident.

This good agreement lends high credibility to the TRAC-calculated fuel-rod temperatures. These values were important for estimating core damage and were not available from plant data because the thermocouples for the fuel rods covered only the range of temperatures expected during normal operation. The calculated fuel-rod temperatures indicate that core voiding (the buildup of steam in the core) began at about 100 minutes into the accident—when the last of the primary pumps were turned off and forced circulation stopped.

The TRAC-calculated core liquid levels also show that core voiding began at this time. In addition, they indicate that only about the lower quarter of the core was water-covered at approximately 3 hours. (As is well known, the absence of instrumentation to measure liquid levels in the core was a major factor leading to escalation of the accident.)

The graph of core liquid levels also shows the results of an analysis by the Nuclear Safety Analysis Center, an arm of the Electric Power Research Institute. Using data from neutron monitors in the containment building, this group calculated the level of a steam-water mixture. The calculated mixture level is higher than the collapsed liquid level from the TRAC analysis, as it should be, and the curves exhibit similar trends. The consistency between the two quite different analyses gives further confidence in the TRAC results.

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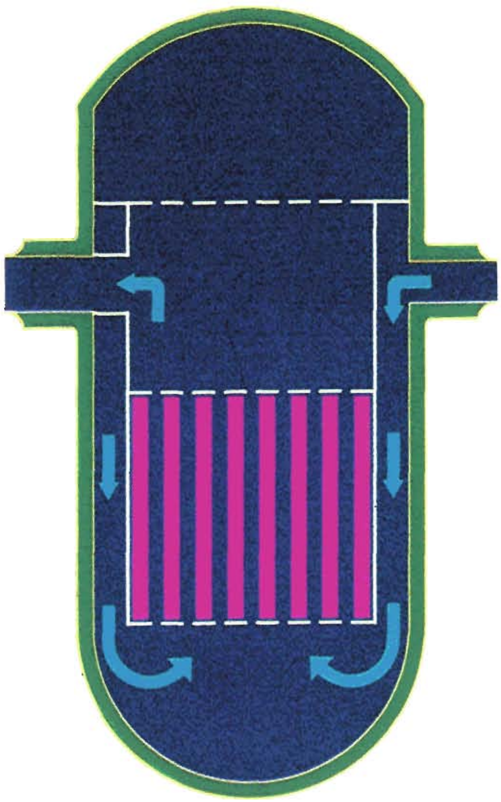
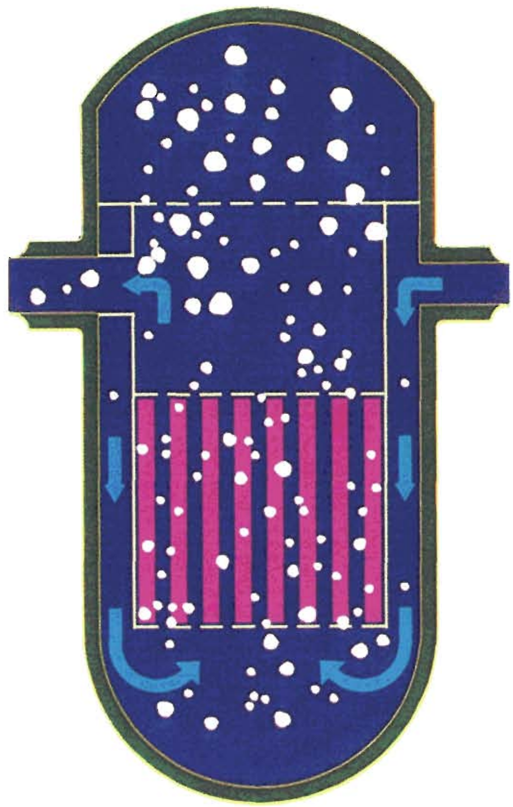
TRAC ANALYSIS OF THE THREE MILE ISLAND ACCIDENT

Sidebar 1:



t = 0
Normal operation

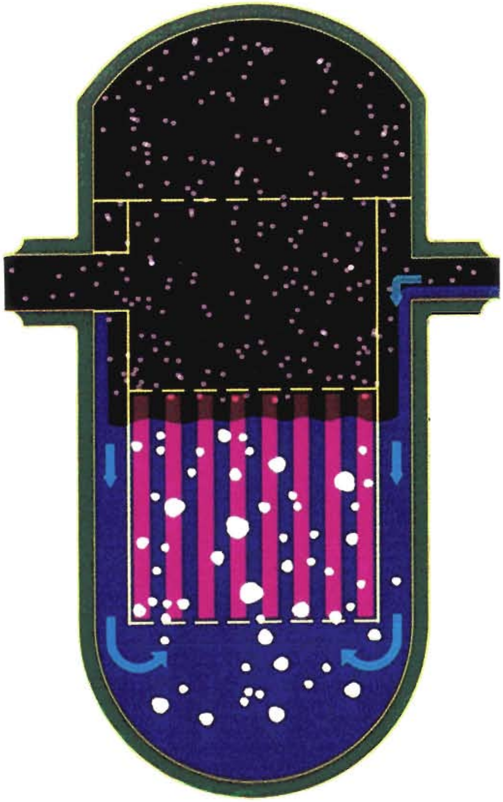


t = 70 min
Equilibrium conditions in core
Cooling by boiling and forced convection

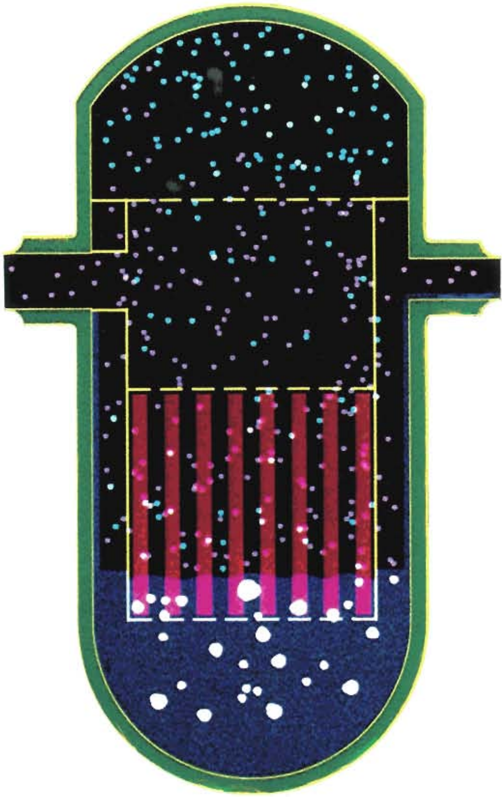
There were no eyewitnesses to the scene within the Unit 2 reactor—and the instrument readings gave only an incomplete and misleading picture. However, from the TRAC analysis we have been able to reconstruct an accurate account of the conditions inside the primary system during the first three hours of the accident.

After steady-state initial conditions for the system were established, the transient calculation was initiated by stopping the flow of feedwater to the steam generators. The primary system pressure then rises above the normal operating range and the pilot-operated relief valve opens. The pressure continues to rise until about 10 seconds when automatic scramming of the reactor causes the pressure to drop. Because the valve fails to close, the system pressure continues to decrease until the steam generators dry out at about 2 minutes. With no heat removal through the steam generators, the pressure begins to rise again until about 8 minutes when auxiliary feedwater is supplied to the steam generators. With the valve still open, the enhanced heat transfer in the steam generators causes the system pressure to decrease further. Finally, an equilibrium state is attained in which decay heat produced in the core is balanced by the energy removal in the steam generators and through the open valve.

During the equilibrium period, which lasts from about 15 to 75 minutes, the primary system loses coolant continuously through



t = 102 min
Primary pumps off
Steam and water separate
Cooling by natural convection
Core temperature rises



t = 180 min
Vessel head filled with hydrogen from cladding oxidation
Upper 75% of core uncovered

the open valve and the letdown system, and the flow through the valve is stable at about 20 kilograms per second.* The low system pressure permits boiling in the core, which provides enough cooling to offset the coolant losses and maintain a stable system pressure and low core temperatures.

The transient calculation includes the assumption that the letdown flow was greater than the high-pressure injection flow by about 2.7 kilograms per second between 10 and 140 minutes.

This stability ends after all of the primary pumps are tripped (the B-loop pumps were tripped at 73 minutes and the A-loop pumps at 100 minutes). From this point on, the system operates in a natural circulation mode, and energy removal through the steam generators is less efficient than in the forced convection mode (pumps on). Without forced circulation, steam and water in the primary system separate. The core becomes partially uncovered and the fuel rods begin a temperature "excursion."

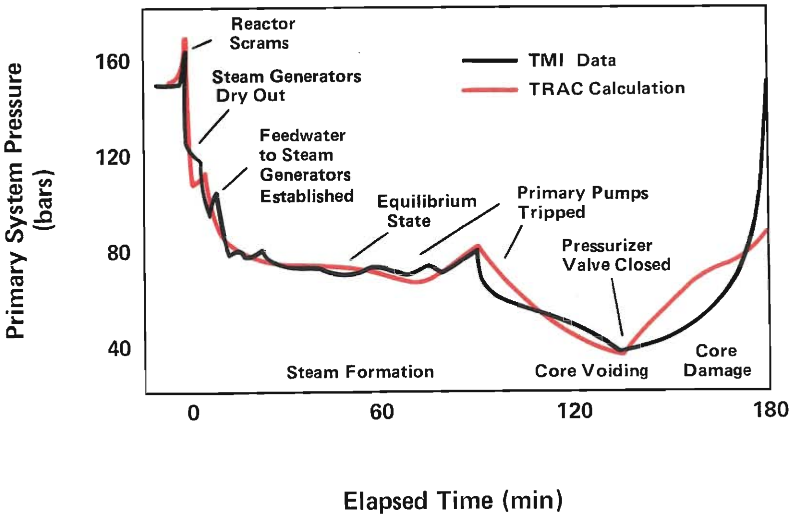
From about 120 to 140 minutes boiling continues, the core water level drops, and the rods heat at roughly 0.25 kelvin per second. With coolant still flowing out the open valve and the letdown line and with steam moving through the core at the rate of 0.5 meter per second, the heat-transfer coefficients between the fuel rods and the steam are slightly higher than those for natural convection. By 140 minutes, the loops are essentially void (steam-filled) and water remains only in the pump suction legs (loop seals). The water

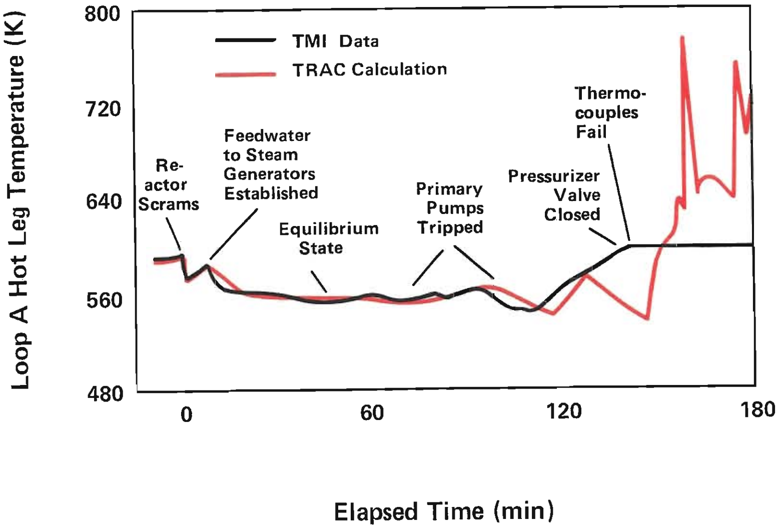
Sidebar 1: Continued

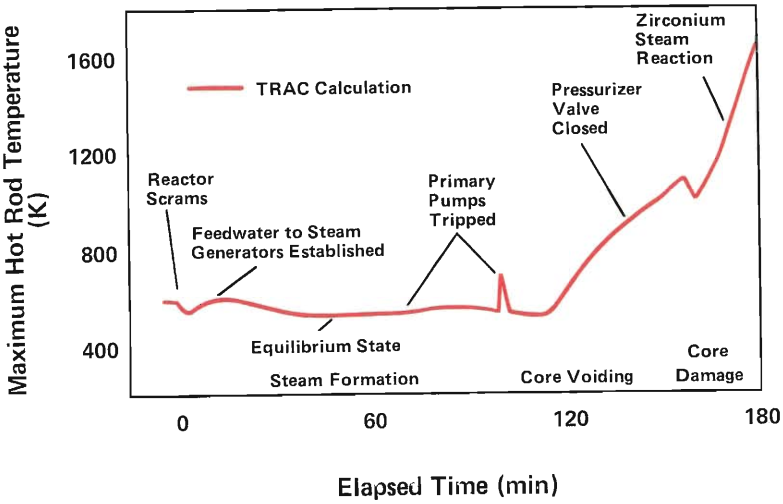
level in the pressurizer drops because of increases in letdown flow rates and decreases in system pressure that cause the water in the pressurizer to flash to steam.

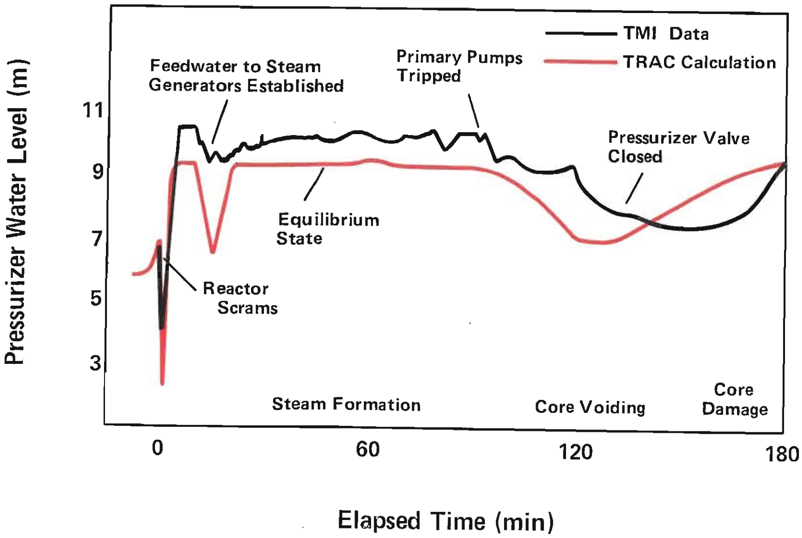
Closing the block valve in series with the pilot-operated relief valve at 140 minutes causes the steam flow in the core to stagnate. Steam can no longer escape through the valve and water in the loop seals prevents any flow through the loops. Without natural circulation, the system begins to pressurize and continues to pressurize for the remainder of the calculation. Vapor velocities through the core are generally less than 0.1 meter per second and the heat-transfer coefficients are very low (on the order of 50 watts per square meter per second, representative of natural convection to superheated steam). The steam begins to superheat and the rod temperatures continue to increase, except for a brief temperature drop at 160 minutes. This temperature decrease is caused by boiling in the lower core cells, which enhances the vapor velocities for a brief period. When these cells become void, the vapor velocities decrease and the rods again heat. The rods continue to heat at a slightly higher rate than before until the temperature reaches about 1300 kelvin and the zirconium-steam reaction begins to provide a significant additional heat source. Then the core temperatures increase at about 1 kelvin per second. The calculation was stopped when the rod temperatures exceeded 1650 kelvin because at that point the core modeling was no longer realistic.

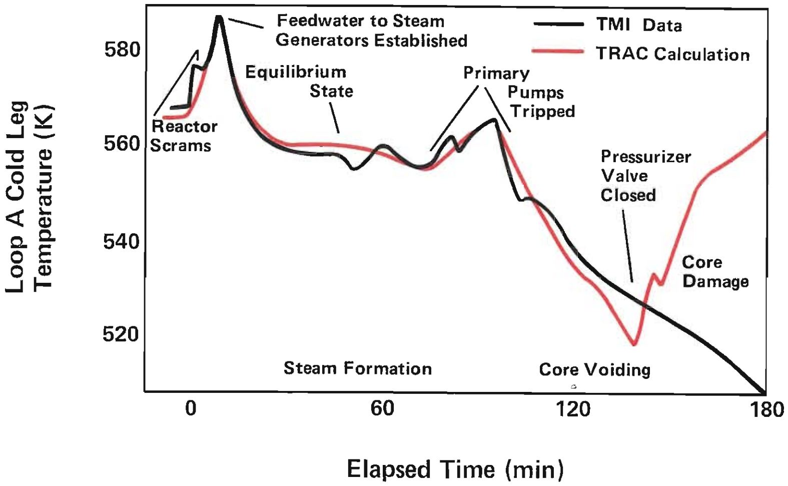
At approximately 3 hours, the top 75 per cent of the core is uncovered. The fuel-rod temperatures remain relatively low in the lower core region because some water is still available for cooling. The pressurizer water level is increasing both in the TRAC calculation and in the plant data. The pressurizer never empties because steam produced in the core condenses in the bottom of the pressurizer and countercurrent flow limiting at the pressurizer inlet prevents the downward flow of water against the upward flow of steam. ■

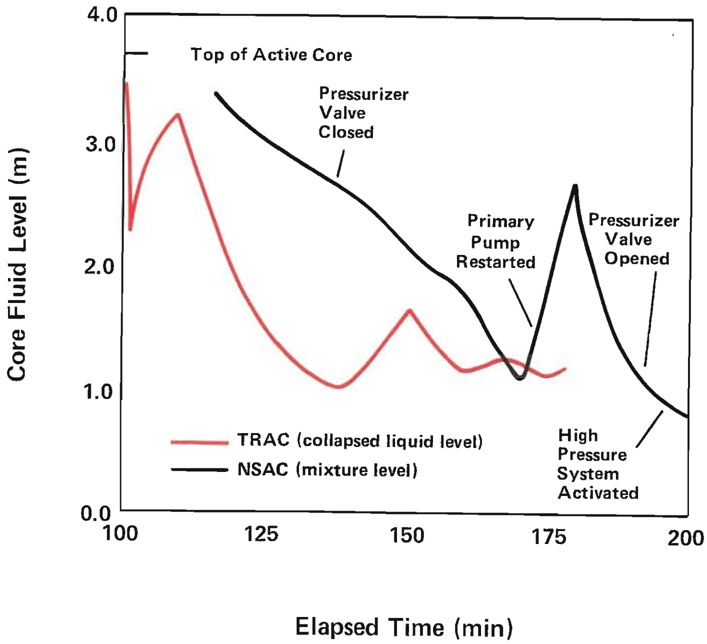







TMI Data
TRAC Calculation
Pressurizer Water Level (m)
11
9
7
5
3
Feedwater to Steam Generators Established
Primary Pumps Tripped
Pressurizer Valve Closed
Equilibrium State
Reactor Scrams
Steam Formation
Core Voiding
Core Damage
0
60
120
180
Elapsed Time (min)


Feedwater to Steam Generators Established
TMI Data
TRAC Calculation
Loop A Cold Leg Temperature (K)
580
560
540
520
Equilibrium State
Primary Pumps Tripped
Reactor Scrams
Pressurizer Valve Closed
Core Damage
Steam Formation
Core Voiding
0
60
120
180
Elapsed Time (min)


4.0
3.0
2.0
1.0
0.0
Core Fluid Level (m)
Top of Active Core
Pressurizer Valve Closed
Primary Pump Restarted
Pressurizer Valve Opened
High Pressure System Activated
TRAC (collapsed liquid level)
NSAC (mixture level)
100
125
150
175
200
Elapsed Time (min)

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Core Damage Estimates

Not until the head is lifted from the reactor vessel at Unit 2 will the state of the core be known with any certainty. Was the core uncovered more than once? Did any of the fuel melt or only the cladding? No one knows for sure.

Present estimates suggest that most of the core damage took place during the first uncovering of the core and the subsequent reflood and quenching of the fuel rods, that is between 100 and 210 minutes into the accident. Here we will discuss the Laboratory's damage estimates, which were based on TRAC-calculated primary system pressures and fuel-rod temperatures up to 180 minutes and on extrapolated values thereafter.

It is expected that the low primary system pressures and elevated fuel-rod temperatures during core uncovering caused the Zircaloy fuel-rod cladding first to balloon, then to rupture, and finally to oxidize.

The cladding would balloon, or increase in diameter, because of the pressure difference between the gas inside the fuel rods and the steam outside. We estimated that the cladding ballooned to the extent that neighboring fuel rods came in contact with each other and coolant flow was impeded. However, ballooning probably had little effect on the time and extent of fuel-rod rupture.

The next stage of damage, rupture of the cladding, would lead directly to release of gaseous fission products to the primary coolant. We estimated that cladding in the upper 15 per cent of an average fuel rod ruptured at about 153 minutes into the accident. Thereafter, fuel-rod temperatures continued to increase, so it is probable that almost all the fuel rods eventually ruptured. These estimates are consistent with observed increases in radiation levels in the containment dome between 153 and 159 minutes and between 193 and 197 minutes. These estimates agree also with other analyses.*

Another effect caused by high fuel-rod temperatures is oxidation of the cladding by steam, an exothermic reaction that would increase the temperatures even more. TRAC-calculated cladding temperatures indicate that substantial oxidation took place at fractional axial core heights from 0.6 to 0.9, or along about 1 meter (3.3 feet) of the upper third of the 3.7-meter (12-foot) fuel rods. The maximum amount of hydrogen that could have been generated by oxidation of the outer surface of the cladding is 130 kilograms (287 pounds), enough to fill the reactor vessel's upper head plus part of the upper plenum.

The zirconium oxide formed by oxidation is a glass-like substance that cracks when subjected to rapid temperature changes. Therefore, when the core was reflooded with water at about 200 minutes, the rapid temperature change undoubtedly fractured some of the oxidized cladding. Thereafter, exposed hot fuel pellets, which are even more brittle than the cladding, probably fragmented also.

Extrapolated values for fuel-rod temperatures indicate that some of the cladding actually melted. This molten material may have been retained within the oxide sheath until temperatures reached 2300 kelvin (3600° Fahrenheit) and, if so, it probably dissolved some of the uranium dioxide fuel. When the core was reflooded, the molten material resolidified as a zirconium/uranium dioxide eutectic and probably formed partial blockages in the affected fuel-rod assemblies.

Figure 3 summarizes the Laboratory's estimates of maximum core damage for the period ending at 210 minutes. These estimates, along with guidelines for examining the damaged core when the reactor vessel is finally opened, were sent in December 1979 to L. E. Hochreiter of the TMI Examination Planning Group 7.2 for the Joint DOE/EPRI/NRC/GPU Technical Working Group.

Analyses of Accident Variations

The analyses and estimates discussed above deal with the actual happenings at Three Mile Island. The President's Commission also requested TRAC analyses for postulated variations of the accident to determine the impact of various events on the accident's severity.** Three variations were analyzed: no delay in auxiliary feedwater supply to the steam generators; a longer delay (60 minutes into the accident rather than 8 minutes) in auxiliary feedwater supply; and full-capacity operation of the high-pressure injection pumps at all times after the system pressure reached the setpoint for their automatic activation.

The analyses indicate that the availability or unavailability of the auxiliary feedwater supply had little effect on the ultimate course of the accident. However, the effect of throttling the high-pressure injection pumps was considerable. The analysis indicates that no core damage would have occurred with the pumps operating as designed. These conclusions are of importance for future considerations of reactor design, operation, and instrumentation.

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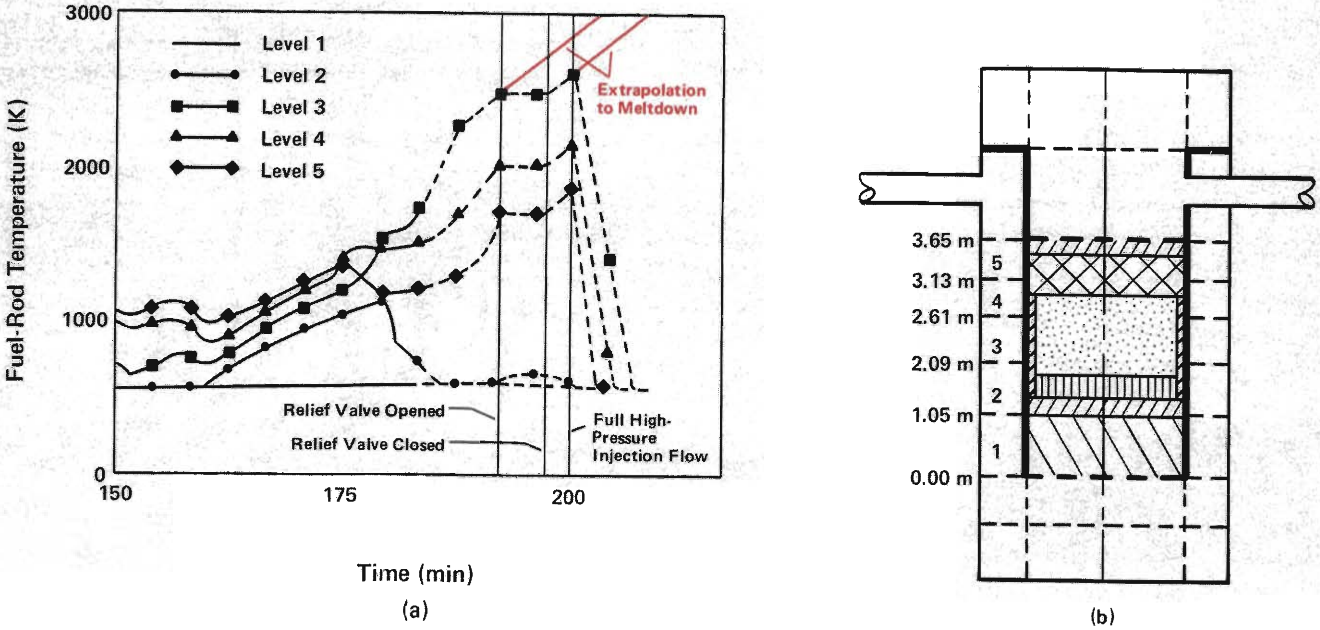
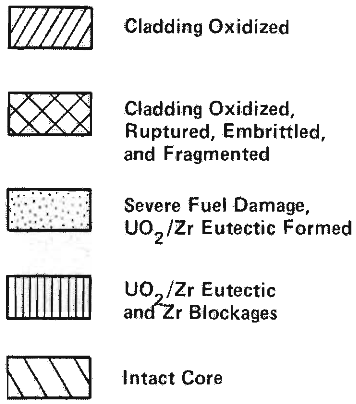
M. L. Picklesimer, "Bounding Estimates of Damage to Zircaloy Fuel Rod Cladding in the TMI-2 Core at Three Hours After the Start of the Accident, March 28, 1979," Nuclear Regulatory Commission memorandum (June 20, 1979) and K. H. Ardron and D. G. Cain, "TMI-2 Accident Core Heat-Up Analysis," Nuclear Safety Analysis Center report NSAC-24 (January 1981).

***The possible effects on the containment of core damage even more severe that that which occurred are discussed in the sidebar "What If The Core Melted?"*





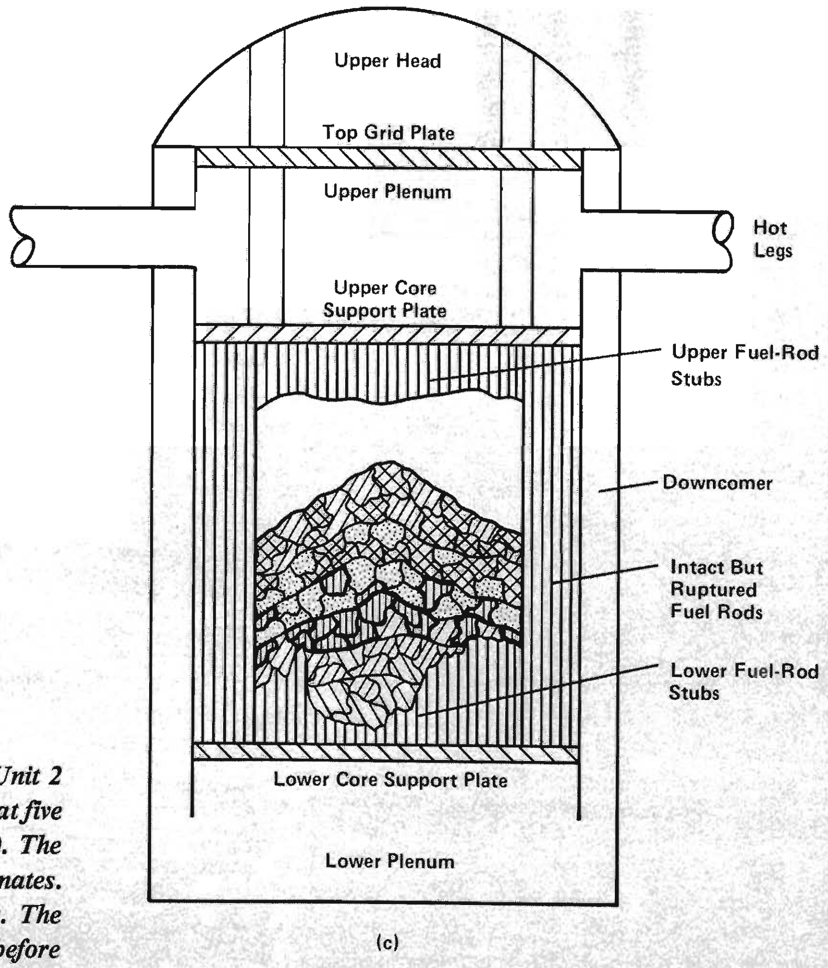


Fig. 3. Estimates of damage to the Three Mile Island Unit 2 reactor core. (a) TRAC-calculated fuel rod temperatures at five core levels defined by the TRAC computing mesh in (b). The fuel-rod temperatures are the basis for core damage estimates. (b) Estimated condition of core materials at 3.5 hours. The materials are shown in the TRAC computing mesh before slumping to the lower core region. (c) Artist's conception of the present appearance of the core.

WHAT IF THE CORE MELTED?

Sidebar 2:

The consequences of severe core damage may impose stresses on the containment building greater than heretofore envisioned. This last barrier to escape of radioactive fission products remained intact during the accident at the Three Mile Island Unit 2 reactor, but the President's Commission investigated its response to variations of the accident involving even worse core damage. The assumption of core damage, including fuel melting, presents several possibilities for breaching the structural integrity of the containment: a hydrogen explosion, a steam explosion, and interaction between molten fuel and the containment's concrete base. On the whole, the Commission's findings were reassuring, but further studies of the effects of severe core damage on containment integrity are continuing.

HYDROGEN EXPLOSION. During the Three Mile Island accident, reaction of steam and zirconium in the fuel-rod cladding produced a significant amount of hydrogen. Burning of some of this hydrogen in the containment created a pressure spike of about 2 bars, which is well below the design limit (about 4 bars) of the containment. The Commission considered the response of the containment to burning or detonation of the maximum amount of hydrogen, that is, the amount produced by reaction of all available zirconium. They concluded that burning of the hydrogen would not overstress the containment and, with less certainty, that detonation would impose a maximum load on the containment close to but below its structural limit.

Because the Three Mile Island containment building is stronger than some, these conclusions are not applicable to all light-water reactors. However, the problem of hydrogen detonation could be solved by installation of ignitors in the containment to prevent accumulation of much more than a burnable mixture of hydrogen and air. Such ignitors are being installed at the Sequoyah reactor, part of the Tennessee Valley Authority electrical system.

STEAM EXPLOSION. The term "steam explosion" refers to the violent (but nonchemical) interaction between hot molten metal and water. Such explosions have been observed in the metal and paper industries. They are accompanied by forceful discharge of water (and sometimes metal) from the zone of interaction. In some instances, surrounding structures have been damaged.

If molten fuel should fall into water remaining in the reactor vessel, a steam explosion could occur and damage the vessel and the containment by two mechanisms. One is generation of a high-pressure shock wave, as in a chemical explosion. But a steam explosion differs from a chemical explosion in two important respects: the peak pressure is lower by orders of magnitude and the risetime of the pressure pulse is considerably longer. Several studies indicate that a steam explosion would not cause vessel failure by this mechanism, and hence would not damage the containment.

The other mechanism involves the upward acceleration of a water and/or a fuel slug by expanding steam. Given sufficient energy, the slug could dislodge some portion of the upper vessel, which in turn could crash into the containment. This scenario requires simultaneous contact of sufficient quantities of molten fuel and water and in addition, highly efficient transfer of heat between fuel and water. It is considered very unlikely that either of these requirements can be satisfied.

The Commission's conclusion that a steam explosion would not cause failure of the containment is the same as that reached by a Swedish scientific committee in 1980 and is applicable to all light-water reactors.

FUEL-CONCRETE INTERACTIONS. If massive core melting is assumed, failure of the vessel is likely and would lead to deposition of debris, consisting of molten fuel and structural materials, on the concrete base of the containment. Estimates of the time required for penetration of the base range from a minimum of 3 days to a maximum of infinity. Solidification of the debris, which is estimated to occur within 1 to 2 days, would slow but not halt erosion of the concrete and would reduce mobility of the fission products.

If penetration of the base should occur, interactions with the underlying bedrock are not significantly different from those with concrete; the site's geology would influence the ultimate fate of the fission products.

Gaseous products of the fuel-concrete interactions are predicted to overpressurize the containment only under extreme conditions, such as lack of containment sprays or decay-heat-removal capability. In addition, the hydrogen produced is not predicted to cause failure. ■

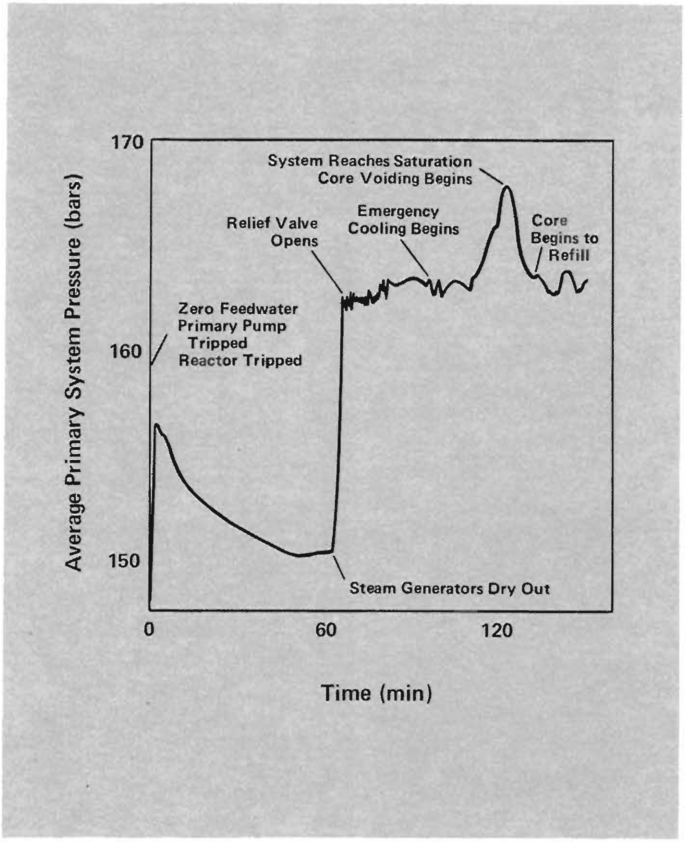


Fig. 4. TRAC-calculated primary system pressure at Zion Unit 1 during a postulated loss-of-feedwater accident.

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After Three Mile Island

It is now clear that a combination of several failures, each perhaps minor compared to the break of a large pipe, can lead to core damage and the possible release of radioactive materials. But if the previous focus of reactor safety research is now judged to have been too narrow, the new focus seems at first hopelessly diffuse. Is it necessary to analyze all possible multiple-failure accidents at every nuclear power plant?

Fortunately, this modern analogue of cleaning the Augean stables has not proved to be necessary. The multitude of possibilities can be reduced to a manageable number of accident types, such as loss-of-feedwater accidents or failure-to-scram accidents. The Nuclear Regulatory Commission is funding studies of these accidents types through its Severe Accident Sequence Analysis Program. Participants in the program are Los Alamos National Laboratory, Oak Ridge National Laboratory, Sandia National Laboratories, and Idaho National Engineering Laboratory. The programmatic research is divided into two areas covering accident aspects before and after core damage, the so-called front and back ends. Research at Los Alamos concentrates on the front end.

Our goal is to determine, for each nuclear power plant, what accidents can occur, how to diagnose them, and what operator actions or engineered safety features may terminate an accident or mitigate its consequences.

We use the technique of fault-tree analysis to enumerate accident types. The several hundred to several thousand fault trees presented by a particular nuclear power plant are condensed, sometimes with the help of a computer program, to a few tens of similar trees. For example, failure of the eight emergency diesel generators at Browns Ferry are represented by eight separate but similar fault trees, but these may be collapsed into one fault tree representing loss of onsite emergency power.

We identify the similar trees as the accident types that must be considered at that plant. For example, at one of the plants studied, the following accident types are possible.

- Station blackout—loss of all onsite and offsite power.
- Interfacing system loss of coolant—loss of coolant through an interface between high- and low-pressure systems, such as through a ruptured steam generator tube.
- Loss of feedwater—loss of all main and auxiliary feedwater to steam generators.
- Pressurizer valve loss of coolant—loss of coolant due to malfunction of one or more of the pressurizer valves.
- Small-break loss of coolant—a break in the primary system that does not lead to rapid loss of coolant or to rapid depressurization.
- Large-break loss of coolant—a break in the primary system that leads to rapid loss of coolant and to rapid depressurization.
- Loss of residual heat removal—loss of the ability to remove decay heat during the transition from hot to cold shutdown.
- Failure to scram—failure of the control system to effect halt of fission on demand.

For each identified accident type, we learn how the plant responds from TRAC analyses. We first compute the consequences of the initiating failure(s) in the absence of operator intervention. Then we perform further analyses, including various postulated operator actions. These analyses use a computer model of the plant that is sufficiently detailed to represent all unique design features and emergency safety systems. From the results we hope to answer questions such as the following. Does the sequence of system responses during the accident present a recognizable signature? What system responses are critical to core damage? Can these critical responses be slowed or averted? What system components are needed to terminate the accident? What information should be available to operators for accident diagnosis and management?

As an example, consider a hypothetical loss-of-feedwater accident initiated by a loss of offsite power at Zion Unit 1, a four-loop pressurized-water reactor. We assume that the reactor has scrammed automatically and that there is no forced circulation because the primary pumps have tripped.

Below, we outline the significant features of this transient in the absence of operator intervention. The TRAC-calculated primary system pressure history is shown in Fig. 4.

GOOD NEWS ABOUT IODINE RELEASES

Sidebar 3:

One of the elements vital to the proper functioning of the human body is iodine. This element, in trace quantities, is essential for the synthesis of metabolism-regulating hormones by the thyroid gland. To produce these hormones as needed, the thyroid gland selectively absorbs iodide ions from the blood, accumulating and storing 25 to 45% of the body's normal intake of iodine. The thyroid gland is thus particularly susceptible to damage by radioactive iodine isotopes, should these be available to the body.

Such isotopes are present among the fission products within the fuel rods of a reactor, and the possibility of their escape to the atmosphere from damaged fuel rods has dominated considerations of reactor accidents and the design of safety systems. The isotope iodine-131 is of particular concern because of its relatively high fission yield (2.77%) and significantly long half-life (8.07 days).

In 1957 an accident at the Windscale reactor in Cumberland, England resulted in escape to the atmosphere of more than 20,000 curies of iodine-131 and a maximum radiation dose to the public (observed in the thyroid glands of several children) of 5 to 15 rads. Despite its rather minimal public consequences, this accident may have had a determining influence on the assumptions adopted for regulatory purposes in the early 1960s by the Atomic Energy Commission and later by the Nuclear Regulatory Commission. It is assumed that 25% of the core inventory of iodine would be distributed as volatile species within the containment as a result of the rupture of a major coolant pipe and, should the containment be breached, would escape to the atmosphere without diminution. A similar fate is assumed for the total core inventory of inert gases, such as xenon-133 and krypton-85, but these chemically inert materials pose a considerably lesser danger to human health.

Information obtained during the accident at the Three Mile Island Unit 2 reactor indicates that, in the case of iodine, these assumptions should be regarded, not as a conservatism, but as an error. Measurements of both xenon-133 and iodine-131 showed that, although the core inventories of both isotopes were roughly comparable (154 million curies of xenon-133 and 64 million curies of iodine-131), the quantity of iodine that escaped to the atmosphere (13 to 18 curies) was less than that of xenon (2.4 to 13 million curies) by a factor of 10^5 to 10^6.

In a letter of August 14, 1980 to the Nuclear Regulatory Commission, A. P. Malinauskas and D. O. Campbell of Oak Ridge National Laboratory and W. R. Stratton of Los Alamos National Laboratory have proposed an explanation for this great disparity.* They suggest that iodine exits from damaged fuel rods predominantly as cesium iodide (CsI) rather than as volatile species such as molecular iodine (I_2). The reducing environment of a water-cooled reactor during a loss-of-coolant accident sustains this chemical state and also converts other iodine species, should they be present, to iodide ions. The escaped CsI will readily

- At 63 minutes, the primary system pressure rises because the steam generators have dried out and no longer remove heat from the primary coolant.
- At about 66 minutes, the relief valve on the pressurizer opens and begins to discharge steam.
- By 80 minutes, water begins to flow through the relief valve because the increased temperature in the primary system has caused the coolant to expand. The pressure remains fairly constant, but the temperature continues to increase.
- At 96 minutes, the emergency core-cooling system is actuated by a containment overpressure signal.
- At 120 minutes, the coolant in the primary system is saturated. The coolant begins to boil, the upper part of the vessel voids, the primary system pressure rises, and safety valves on the pressurizer open briefly.
- By about 130 minutes the partially voided core has begun to refill; thus, the system is recovering.

This calculation shows that the automatic safety systems would bring the reactor to quasi-stable conditions without any intervention. However, actions by the operators can prevent core voiding or reduce the severity of the accident. Below we list some conclusions based on TRAC analyses regarding successful management of the accident.

1. If, within the first hour, the operators notice a drop in the water level of the steam generators and are able to restore at least 30 per cent of the auxiliary feedwater supply, no voiding will occur in the primary system and the core will be adequately cooled.
2. If auxiliary feedwater cannot be restored, the operators can prevent boiling only by initiating the complex sequence of manipulations known as feed-and-bleed cooling near the beginning of the transient. This cooling technique consists of alternately injecting emergency coolant with the high-pressure

condense on available metal surfaces at temperatures at or below 673-773 kelvin (750-930° Fahrenheit) and will enter into solution as cesium and iodide ions upon encounter with water or condensing steam. This situation will persist in the absence of an oxidizing atmosphere. Thus the amount that could escape to the atmosphere from a water-cooled reactor would be considerably lower than has been assumed.

In contrast, during the accident at the air-cooled and graphite-moderated Windscale reactor, metallic fuel and (probably) graphite were burning—clearly an environment favorable to oxidation of CsI to Cs_2O and I_2.

In further support of their hypothesis, the scientists cite the following observations.

- Iodine and cesium escape at the same time from leaking fuel rods in pressurized-water reactors during normal power transients. This behavior is completely different from that of the inert gases.
- Of those compounds that could be formed by iodine within fuel rods of water-cooled reactors, CsI is thermodynamically the most stable. Further, because the fission yield of cesium is larger than that of iodine by a factor of 10 to 11, cesium is always available in great excess for reaction with iodine.
- Used fuel rods have been made to fail in experiments simulating accident conditions in water-cooled reactors, and the iodine released has been recovered predominantly as CsI rather than as I_2.
- The chemistry of iodine is such that, if water is accessible, iodine species such as CsI react with the water so that the iodine concentration in the gas phase is very much smaller than its concentration in the water.
- An investigation, still continuing, of incidents involving fuel-rod damage at other water-cooled reactors indicates that, as at Three Mile Island, much smaller amounts of iodine escaped to the atmosphere than has been assumed.

This hypothesis must be strengthened by information about the fundamental chemistry—under the conditions within a reactor—of cesium and iodine and of fission products in general. In response to this issue, the Nuclear Regulatory Commission and the Department of Energy have sponsored studies to pinpoint those areas of research that should be pursued.

If further study confirms that cesium and iodine behave in the manner proposed, many criteria for reactor safety must be reevaluated and the reactor systems for fission-product control must be reexamined. In addition, and most importantly, the public could then be assured that the danger posed by even a very severe reactor accident may be significantly lower than previously estimated. ■

**H. J. Kouts of Brookhaven National Laboratory has independently developed a similar hypothesis about the behavior of cesium and iodine.*

injection system and venting steam through the pressurizer relief valve. However, if the containment has been isolated automatically by an overpressure signal due to vented coolant, use of feed-and-bleed cooling is severely restricted because the compressed air that operates the relief valve cannot be replenished.

3. After the steam generators dry out, the operators will see increases in the pressurizer water level and in the primary system pressure and temperature. They should respond by initiating feed-and-bleed cooling. If the containment is not isolated and feed-and-bleed cooling begins between 1 and 2 hours, some core voiding will occur but the system will recover much faster than it would otherwise.

4. If the primary pumps were not tripped at the start of the accident, leaving them running until the emergency core-cooling system actuates automatically will prolong the accident slightly but will not materially alter its ultimate course.

This particular accident and all related accidents, such as loss of feedwater with stuck-open pressurizer relief valve (the Three Mile Island accident) or a loss-of-feedwater with stuck-open atmospheric relief valve, have very characteristic signatures that can help the operators to diagnose the situation. Not all multiple-failure accidents have such characteristic signatures, and in some cases additional instrumentation may be needed for proper identification.

The SASA program is currently focused on accident sequences at large two- and four-loop pressurized-water reactors. The emphasis at the Laboratory is on plant-specific accident delineation, early accident recognition, early accident management, and definition of critical times and actions. By improving the operational safety of reactors, the severity of multiple-failure accidents, and thus the risk to public health, can be reduced. ■

LOS ALAMOS ASSISTANCE TO THREE MILE ISLAND INVESTIGATIONS

Sidebar 4:

Los Alamos National Laboratory was a source of considerable technical assistance to groups investigating the Three Mile Island accident. These groups called on Laboratory staff for direct participation in the investigations and for relevant information. Providing this assistance was a satisfying experience for those involved. Needless to say, the efforts mentioned below were supported by those of many other Laboratory personnel.

William R. Stratton was a member of the Technical Assessment Task Force of the President's Commission on the Accident at Three Mile Island (also known as the Kemeny Commission). In addition to his investigative and advisory duties, Stratton was principal author of "Technical Staff Analysis Report on Alternative Event Sequences," an assessment of the consequences of postulated variations of the accident.

Five Laboratory scientists served as consultants to the Technical Assessment Task Force. One of these, Beverly Washburn, had been the licensing project manager for the Three Mile Island Unit 2 plant while on loan to the Nuclear Regulatory Commission from 1973 to 1975. His familiarity with many of the details of the plant proved valuable. He was author of the staff reports "Radiation Releases and Venting of Tanks Friday Morning, March 30, 1979" and "The Evacuation Recommendations on Friday Morning, March 30, 1979." He assisted in preparation and review of other staff reports and participated in some of the staff depositions.

Three other consultants, John R. Ireland, Walter L. Kirchner, and Peter K. Mast, were authors of "Fuel Damage Estimates with the Transient Reactor Analysis Code (TRAC)." Robert D. Burns, also a consultant, and Kirchner were among the authors of "Consequences of a Hypothetical Fuel Melting Accident at TMI-2," "Potential for Damage to Reactor Vessel or Containment Due to Steam Explosions Associated with Fuel Melting Accidents," and "Penetration of the Concrete Basemat." Burns was among the authors of "Fission Products Within the Reactor Containment Building as a Consequence of the Hypothetical Fuel Melting Accident." (All these reports are included in "Technical Staff Analysis Report on Alternative Event Sequences.") Burns was also sole author of "Technical Staff Analysis Report on WASH 1400—Reactor Safety Study," a review of the relationship between the accident probabilities and risk estimates of that study and the Three Mile Island accident.

At the request of the Commission, John R. Ireland, Peter K. Mast, Thomas R. Wehner, Paul B. Bleiweis, Walter L. Kirchner, and Michael G. Stevenson submitted TRAC analyses of Unit 2's response for the first 3 hours of the accident and estimates based on these analyses of core damage and hydrogen production. They also supplied TRAC analyses of Unit 2's response to postulated variations of the accident sequence. This information was used extensively by the Commission staff in preparation of "Technical Staff Analysis Report on Alternative Event Sequences" and by staff of the Nuclear Regulatory Commission Special Inquiry Group in preparation of a section of "Three Mile Island: A Report to the Commissioners and to the Public" (the Rogovin report). The information has also been published as "Preliminary Calculations Related to the Accident at Three Mile Island" [Los Alamos Scientific Laboratory report LA-8273-MS (March 1980)].

Donald G. Rose provided information to the Commission about the response of the pressure vessel to a hydrogen explosion and of the containment building to a steam explosion; he also prepared the staff report "Pre- and Post-Accident Security Status at Three Mile Island."

Eddie R. Claiborne, Richard L. Cubitt, Roy A. Haarman, and John L. Rand supplied the Commission with the study entitled "Three Mile Island Sabotage Analyses."

Talmadge R. England and William B. Wilson used the Laboratory-developed computer program CINDER to furnish the Commission with information about Unit 2's post-accident decay power. This information has been published as "TMI-2 Decay Power: LASL Fission-Product and Actinide Decay Power Calculations for the President's Commission on the Accident at Three Mile Island" [Los Alamos Scientific Laboratory report LA-8041-MS, Revised (March 1980)].

John W. Bolstad and Roy A. Haarman submitted TRAC analyses of postulated reactor transients quite similar to the Three Mile Island accident. These analyses, which had been completed before the accident as part of a sabotage study, provided the Commission with a better understanding of some aspects of the accident. They have since been published as "Summary of Thermal-Hydraulic Calculations for a Pressurized Water Reactor" [Los Alamos Scientific Laboratory report LA-8361-MS (May 1980)].

Jay E. Boudreau was a Task Group Leader of the Three Mile Island Special Investigation carried out by the Subcommittee on Nuclear Regulation for the Committee on Environment and Public Works of the U. S. Senate. He was author of "Recovery at Three Mile Island" in "Nuclear Accident and Recovery at Three Mile Island," which reports the findings of the Special Investigation. In addition, he was principal author of a study for the Subcommittee of two industry-sponsored groups involved in reactor safety entitled "Review of the Nuclear Safety Analysis Center and the Institute for Nuclear Power Operations." ■

James H. Scott was born in Norton, Virginia in 1942. He earned a Bachelor of Science in physics from Virginia Polytechnic Institute in 1964 and a Master of Science in nuclear engineering from the University of Virginia in 1971. He worked as an accident analyst for General Electric Company, Babcock & Wilcox, and the Hanford Engineering Development Laboratory before coming to Los Alamos in 1975. At the time the multiple-failure accident analysis work was initiated he was Leader of the Accident Analysis Group. He is currently a Program Manager in the Nuclear Programs Office.

AUTHORS

John R. Ireland, a native of Hereford, Texas, was born in 1951. He earned his Bachelor of Science in mechanical engineering from New Mexico State University in 1974. He then went to work at the Nuclear Energy Division of General Electric Company in San Jose, California, where he specialized in safety analysis of boiling-water and liquid-metal fast breeder reactors. He joined the Laboratory after obtaining his Master of Science in mechanical engineering from the University of California at Berkeley in 1977. He is currently Project Leader for TRAC applications in the Safety Analysis Group. His expertise in the field of reactor safety was employed extensively by the Nuclear Regulatory Commission, by Senate subcommittees, and by the President's Commission during and immediately after the Three Mile Island accident. John's analysis of the situation is quite far-reaching: "The lessons we learned at TMI are many. First, nuclear facilities have large safety margins, even when mechanical problems and operator errors complicate operation; second, specialists like myself must work harder not only to anticipate and analyze accident situations but to communicate our findings; and third, we must reinforce our liaison between research organizations, the Nuclear Regulatory Commission, and public utilities."

William R. Stratton earned his A. B. and Ph.D. in physics and mathematics at the University of Minnesota in 1947 and 1952, respectively. He joined the Laboratory staff in 1952 and worked in theoretical weapons design. Later he became involved in theoretical studies of criticality safety and dynamic behavior of supercritical systems. Stratton has been involved in a wide spectrum of reactor safety studies and has been cited for outstanding contributions to the national power reactor program. He was a leader in the Laboratory's 17-year Rover Program and was involved in the design and analysis of the Kiwi-TNT experiment, which established an experimental baseline for theoretical prediction of reactor excursions. Stratton was the United States representative to the Cadarache Laboratory in France from 1965 to 1966 and served as a member of the Advisory Committee on Reactor Safeguards from 1966 to 1975. He was a member of the American team of experts that evaluated the hazards presented by the Russian spacecraft that crashed over Canada, and he was called to the technical team that advised the President's Commission on the Accident at Three Mile Island. He is the author of more than 50 publications, most in the area of reactors and reactor safety.

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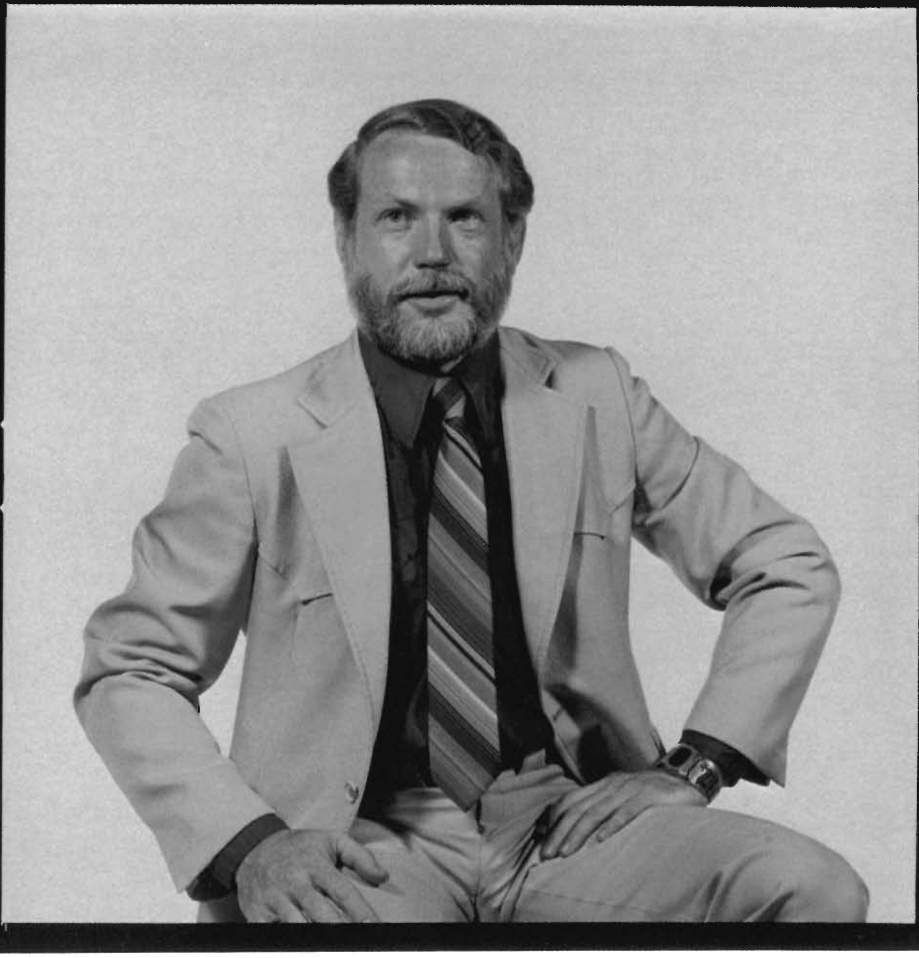
EDITORIAL BY JAY BOUDREAU

THREE MILE ISLAND: *Aftermath and Impact*



Jay E. Boudreau is Deputy Associate Director for Nuclear Programs at Los Alamos National Laboratory. After the accident at Three Mile Island, he served as a Task Group Leader of the Three Mile Island Special Investigation carried out by the Subcommittee on Nuclear Regulation for the Committee on Environment and Public Works of the U. S. Senate. He began his career at the Laboratory in 1973 as a Staff Member of the Transport and Reactor Theory Group and has since then been Alternate Group Leader and Group Leader for Fast Reactor Safety and also Program Manager for Nuclear Regulatory Commission research. He earned his Bachelor of Science in 1967, his Master of Science in 1969, and his Ph.D. in Engineering in 1972, all from the University of California at Los Angeles. In 1980 he was awarded a Certificate of Appreciation by the U. S. Senate. He is a member of the American Nuclear Society.

Los Alamos has been deeply involved in analyzing what happened at Three Mile Island and in developing the technical knowledge that will help prevent further accidents of this kind. But there are equally complex and challenging nontechnical questions about nuclear reactor accidents that our society must now resolve.

The United States has been using nuclear-generated electricity continuously since 1957 when the first commercial plant went on-line in Shippingport, Pennsylvania. Today 70 operational plants are producing 11 per cent of our electricity. Yet for all our familiarity with nuclear power, we cannot agree on what to do about the disabled Metropolitan Edison Company power plant on Three Mile Island.

For over two years the damaged reactor core of Unit 2, miles of radiation-contaminated wire and pipe, and 700,000 gallons of radioactive water have remained on the Susquehanna River island while Metropolitan Edison, citizen groups, courts, and state and federal agencies have argued about responsibilities and cleanup procedures. And right next door sits Unit 1, fully intact, but unable to operate because of a continuing dispute between Metropolitan Edison and the Nuclear Regulatory Commission.

How do we now recover from such an accident? And how do we reassure ourselves that nuclear energy, despite the accident and the general confusion surrounding it, is still a credible component of our energy supply system?

Before considering the serious economic and political impact of the Three Mile Island accident, we should look at the kind of physical damage done, at the amount of material involved, and at the cleanup technology required.

During the accident the core of the Unit 2 reactor was badly damaged, and 700,000 gallons of radioactive water were released into the containment building through a relief valve that was stuck open. The building's atmosphere was contaminated by 45,000 curies of krypton-85, a radioactive gas. It is likely that many of the control rods were melted, that the zirconium fuel-rod cladding was oxidized, embrittled, and shattered by thermal shocks, and that some of the ceramic fuel was also shattered into small fragments.

While the condition of the core will not be known until the reactor vessel head is removed, analysis indicates there will be a significant amount of fuel debris on grid spacers within the core; some of the fuel debris may even have been pumped into the steam generators. If there are large deposits of fuel debris on the steam generator tube sheets, the latter will have to be disassembled, increasing the radiation dose that cleanup workers will receive.

Because the condition of the control rods is not known and because fuel relocation and compaction could lead to increased generation of neutrons somewhere in the system, the concentration of boron (a neutron absorber) in the cooling water has been increased from its normal level of 1000 parts per million (ppm) to 3500 ppm to prevent criticality in the most reactive configurations possible. This concentration of boron will have to be maintained at all times, including during core removal.

Core removal will be the most challenging job from a technical standpoint and may take a whole year to complete. Because the vessel head penetration conduits have been damaged and because entanglement with core debris is likely, removing the head will require special care. The exact techniques that will be used are not yet decided, but the procedure will probably involve several steps: those conduits not damaged will be removed, optical devices will be inserted to view the underside of the head, and then special cutting tools will be devised to sever the entangled conduits.

Once the core head is removed, similar techniques will be used to take apart the core, which is now probably made up of particulate debris, resolidified material, and intact fuel-pin stubs. This mess will have to be cut apart and the core removed in sections. Although techniques developed in cleaning up other reactor accidents will be available, on-the-spot tool design will be required. Remote optical devices will be used to observe the character of the environment, mockups will be built, and the newly designed tools will be tested before attempts are made to section the core.

After the core is sectioned, its pieces and debris will be encased under water in transfer casks. These casks will then be moved through the spent-fuel transfer tubes and stored in the plant's spent-fuel pool until a decision is made about shipment off the site.

The health and safety of the public will not be in danger

THREE MILE ISLAND: *Aftermath and Impact*

EDITORIAL

during the cleanup operation, but the estimated 1000 cleanup workers will have to work in relatively high radiation fields for routinely long periods of time. Even so, these hazards can be dealt with through strict health physics and safety procedures. Thus both the technical and health problems appear tractable. The real problem lies elsewhere.

Cleaning up the reactor has severely taxed our regulatory, political, and industrial institutions. Neither industry nor the Nuclear Regulatory Commission was truly prepared for the accident or for its aftermath. In the early days of nuclear development, power reactors were owned by the government, and cleaning up accidents was the responsibility of the Atomic Energy Commission. Public involvement was minimal, and cleanups were usually quick and relatively inexpensive. In the mid-1960s the federal government gave the responsibility of reactor ownership and operation to private industry. Then in 1974 Congress established a new agency to license and regulate the nuclear power industry—the Nuclear Regulatory Commission. With licensing as its main concern and with no experience in commercial reactor cleanup, it is not surprising that the Nuclear Regulatory Commission dealt initially with the Three Mile Island cleanup in a series of ad hoc reactions.

There has been a long sequence of delays. The week after the accident Metropolitan Edison began design of a processing system (EPICOR-II) to treat contaminated water. It planned to discharge the processed water (which it claimed would meet state and federal radiation standards) into the Susquehanna River beginning in May 1979. The plan was never carried out. The city of Lancaster and the Susquehanna Valley Alliance both went to court to prevent discharge of the treated water. The Nuclear Regulatory Commission first responded by prohibiting any water processing without its permission. Later that year, after an environmental assessment, the Commission allowed the utility to begin processing the water but still prohibited its discharge.

Even these first steps opened the way for legal and political debates. The Nuclear Regulatory Commission was challenged as to whether it was acting legally in separating water decontamination from the rest of the cleanup operation and in basing its approval on an environmental assessment rather than on a fully developed environmental impact statement. Critics maintained that the environmental assessment process applied only to reactor licensing, not to an action as potentially significant as the Three Mile Island cleanup. It was argued that segmenting the cleanup was illegal, and in response to the protests, an environmental impact analysis was performed, including full public participation.

Failure to hold public hearings before venting krypton from the containment building resulted in the Sholly case. After a protracted environmental assessment and safety review, the Nuclear Regulatory Commission had concluded that venting would not involve any significant hazard to the public. The Commission's action was challenged. In 1980 the United States Court of Appeals in the District of Columbia found that the Commission had violated the law by allowing the krypton to be vented without first holding public hearings. Appeal of this case is now under consideration by the Supreme Court.

The Nuclear Regulatory Commission issued its Final Programmatic Environmental Impact statement for the overall cleanup of the Three Mile Island plant in March 1981, after an extended public comment period. The study presented alternatives ranging from full cleanup of the damaged reactor to no action other than continuing to maintain it in its present condition. The report concluded that full cleanup should proceed as quickly as possibly to reduce the potential for uncontrolled releases of radioactive material to the environment. The report also concluded that existing methods were adequate or could be suitably modified to perform virtually all of the necessary operations without exceeding accepted environmental limits. In April 1981, 24 months after the accident, the Commissioners issued a policy statement urging Metropolitan Edison to accelerate the pace of the cleanup.

The two-year delay from the time of the accident to the final report of the Nuclear Regulatory Commission is not the only problem Metropolitan Edison has in dealing with its damaged reactor. The company faces drastically inflated recovery costs and the need for about a thousand professionals and skilled laborers to perform the cleanup job. The utility has hired the Bechtel Corporation to prepare two comprehensive cleanup plans, but virtually none of the steps recommended in these plans can be implemented without Nuclear

Regulatory Commission approval and without rate relief. The rate-setting commissions, however, are reluctant to lay the full financial burden of cleanup on the rate payers, because cleanup costs have continued to escalate. Thus, Metropolitan Edison and its holding corporation, General Public Utilities, which together employ eleven thousand persons and serve four million customers, have been forecasting serious cash flow problems.

Metropolitan Edison has been losing about $1 million per month for the last two quarters, a condition without precedent in the electric utility industry. The market value of the holding company's stock fell from $18 per share before the accident to $4.50 per share in the spring of 1981, and no dividends had been paid on its common stock for the two previous quarters. The company has laid off 200 employees involved in customer service and 500 contractor employees, many of whom were engaged in routine maintenance. The cutback restricts the number of possible new residential hookups and increases the likelihood of protracted power outages (perhaps up to a week) after severe storms.

Metropolitan Edison continues to operate with a very limited margin between borrowing requirements and credit availability. If a majority of the banks decide that the company's revenues are not sufficient to assure its financial credit, or if the utility's future seems cloudy enough that the banks feel they may not be paid, the banks may call their loans. The company has no access to other lines of credit. Its revolving credit agreement with the banks is set for review in October 1981.

If Metropolitan Edison enters receivership, it is not clear what may happen to its remaining assets. In receivership the courts would decide how any available money would be spent, and there is no way to predict how much of the revenue would be awarded by the court to clean up Three Mile Island. Would cleanup costs be covered before bond credit was paid off? Would a bankruptcy trustee have the freedom unilaterally to change rates instead of having set by Public Utility Commissions? We do not know.

Meanwhile, Metropolitan Edison's Unit 1, which was down for refueling at the time of the accident, remains idle. To ensure a high level of safety and to increase public confidence, the Nuclear Regulatory Commission in July 1979 refused to allow Unit 1 to come back on-line. However, revenues from the operation of Unit 1 could go a long way toward improving Metropolitan Edison's standing in the financial community. This, in turn, might allow the utility to borrow some of the money to clean up Unit 2. (A further problem stems from the fact that Unit 1 was removed from the rate base in April 1980. This means, under current Public Utility Commission restrictions, that no income from Unit 1 can be used for the cleanup operation. Hence, the mechanism for eventual repayment of loans for the cleanup is uncertain.)

This last July the Governor of Pennsylvania submitted a proposal for financing the cleanup of Three Mile Island. The proposal calls for all involved parties to share the expense, and it includes the restart of Unit 1. While the proposal is viewed as an important first step, follow-up in securing agreements still remains. Before anything else can happen, the Nuclear Regulatory Commission must approve the restart of Unit 1. The Atomic Safety and Licensing Board has concluded its public hearings and, as predicted, reported favorably to the Commission in September. However, restart could be delayed until winter because the Nuclear Regulatory Commission has stated its intent to review the Board's decision before giving final approval.

Metropolitan Edison is not the only utility affected by the aftermath of Three Mile Island. Our failure to resolve the impasse has cast a shadow on the whole nuclear power industry. Soaring inflation, new and often contradictory federal and state regulations, and public intervention are all major factors in halting growth of nuclear power. No new plant orders have been placed since 1978, and more than 50 plants have been cancelled in the past five years. From 18 to 21 additional plant cancellations are expected in 1981.

A Department of Energy study (DOE/RG-0036, July 1980) reports on the national impact of protracted licensing delays of nuclear power plants. (Some of these delays were caused by safety adjustments made after the Three Mile Island accident.) The study recommends that

"...Every effort should be made to maintain the

current schedule for construction and licensing of these commercial units which are scheduled for operation by the end of 1985:

> *Diablo Canyon Units 1 and 2, California*
> *San Onofre 2, California*
> *La Salle County Units 1 and 2, Illinois*
> *Farley 2, Alabama*
> *McGuire 1, North Carolina*
> *Summer 1, South Carolina*
> *Sequoyah 2 and Watts Bar 1, Tennessee."*

The study notes that failure to complete the nuclear units scheduled for operation by 1985 will result in the use of an additional 700 million barrels of oil. If no new nuclear units are added by 1985, then electric power reserve margins will be unsatisfactory throughout the entire midsection of the nation, from Michigan to Texas. Moreover, even if such plans are implemented, the northwest power pool will probably have an energy supply shortage by 1985.

The outlook for licensing is not good. Only three uncontested nuclear plants have been licensed by the Nuclear Regulatory Commission since the moratorium precipitated by Three Mile Island. This brings the total number of commercial operating nuclear reactors to 72. There are now 96 plants in some stage of licensing, but many of these may be delayed or cancelled.

The pending decision by the Supreme Court in the Sholly case will be important to the future of the nuclear power industry. Should the Court find that each and every amendment to an operating license must be subject to public review, the way would be open for thousands of hearings every year. The number of hearings would constitute a de facto shutdown of the nuclear industry.

As mentioned, increased complexity in the licensing process for nuclear plants is part of the aftermath of Three Mile Island. For example, after the President's Commission investigated the accident, the Carter Administration added two new partners to the licensing process—the state governors and the Federal Emergency Management Agency. Every nuclear plant must now have an emergency response plan approved by the governor of the state and also by the federal agency. These additional steps, desirable though they may be, are bound to delay licensing of nuclear plants.

Utilities are critical of such additional steps, and they are also critical of the uncertainty they face as a result of the proliferating, and largely uncoordinated, regulations. It is not surprising, therefore, that utilities are not placing orders for new nuclear plants. At a time when projected shortfalls of electricity are being well documented, the Nuclear Regulatory Commission predicts that only one construction permit application will be submitted between 1981 and 1986.

Moreover, such licensing problems are not restricted to nuclear plants; it is increasingly difficult to bring coal-fired generating plants into service. The difficulty stems from increasing government interest in and regulation of the problems associated with burning coal; for example, the build-up of carbon dioxide in the atmosphere, acid rain, land destruction, and transportation. Capital requirements for coal plants are already large (about 2/3 those for nuclear plants), and the expensive regulatory restrictions now added to the basic costs make utilities reluctant to commit themselves to build new coal plants.

Safety and environmental regulations are not the only factors restricting construction of new nuclear power plants. Inflation and tight money are also involved. As recently as 10 years ago, utilities routinely funded a large portion of a nuclear construction project internally. Today, with only about 20 per cent of such funds generated internally, the companies must compete in the open market for money to fund the remainder. Furthermore, it is becoming increasingly difficult for utilities to compete for capital in the open market. One reason is the high investment risk associated with costly plants that will not be producing electricity for 10 to 14 years. Another reason is the general wariness about a nuclear investment after the experience of Metropolitan Edison in recovering from the Three Mile Island accident. In addition, the reluctance of public utilities commissions to allow rate increases on the grounds that they fuel inflation makes it difficult for utilities to be sure of an adequate return on investments. This situation is reflected in the market for utility bonds. In the past two years Standard and Poors has cut 37 bond ratings of electric utilities while raising only 6. And utility stocks now average only 3/4 of their book value. Yet despite their poor standing, the utilities will have to find some $600 billion in investment capital before the year 2000 just to keep up with the 2 to 3 per cent a year increase in demand estimated by the Department of Energy. This sum is four times more than the total capital invested by the utilities up to 1980.

As we have seen, the handling of the Three Mile Island accident weighs heavily on the nuclear power industry and on all electric power generation. Industry and government have still not resolved the key questions raised: Who pays for cleanup? How do we ensure safety and environmental protection without regulating the industry to death? The questions that we have so far resolved about the accident are technical ones. We know what happened and why it happened, and we know how to clean up the mess (though not what to do with the waste products). And we have some very good ideas about how to avoid similar accidents in the future.

Questions about the Three Mile Island accident are not the only elements in the electric power equation. Lack of investment capital, inflated construction costs, and uncertainties about the rate structure are also important. Yet none of these elements need be obstacles unless we want them to be.

The administration can streamline the procedure for licensing nuclear plants. It can reduce the number of steps and the number of agencies involved and can speed up the review process.

The Department of Energy can work more closely with state utility commissions and taxing authorities to find ways to encourage plant construction. For example, easing state taxes during plant construction would reduce construction costs and lower consumer rates.

At the same time, the industry should be given greater incentive to phase out and retire expensive, oil-fired plants. The current practice of simply passing increased oil costs to customers does not provide that incentive. The utility companies and the public utility commissions need to make a joint commitment to solve the problem.

Congress could establish an insurance pool to help companies like Metropolitan Edison recover from nuclear accidents. The industry's Edison Electric Institute has already pledged a contribution to the cleanup of Three Mile Island provided there are matching government and Public Utility Commission funds. Congressional action is all that is lacking.

Finally, the public as a whole and particularly citizen groups interested in clean air, resource conservation, and alternative energy sources must become involved in the solution. As a nation, we can no longer afford the luxury of opposing technological answers to social problems, especially the problem of energy. We must all accept the responsibility for weighing alternatives and arriving at practical conclusions. We must accept both the costs and the risks of our choices. We must, in fact, commit ourselves to the society we live in and to the technology that society has developed. This means both enjoying the benefits and paying for the mistakes.

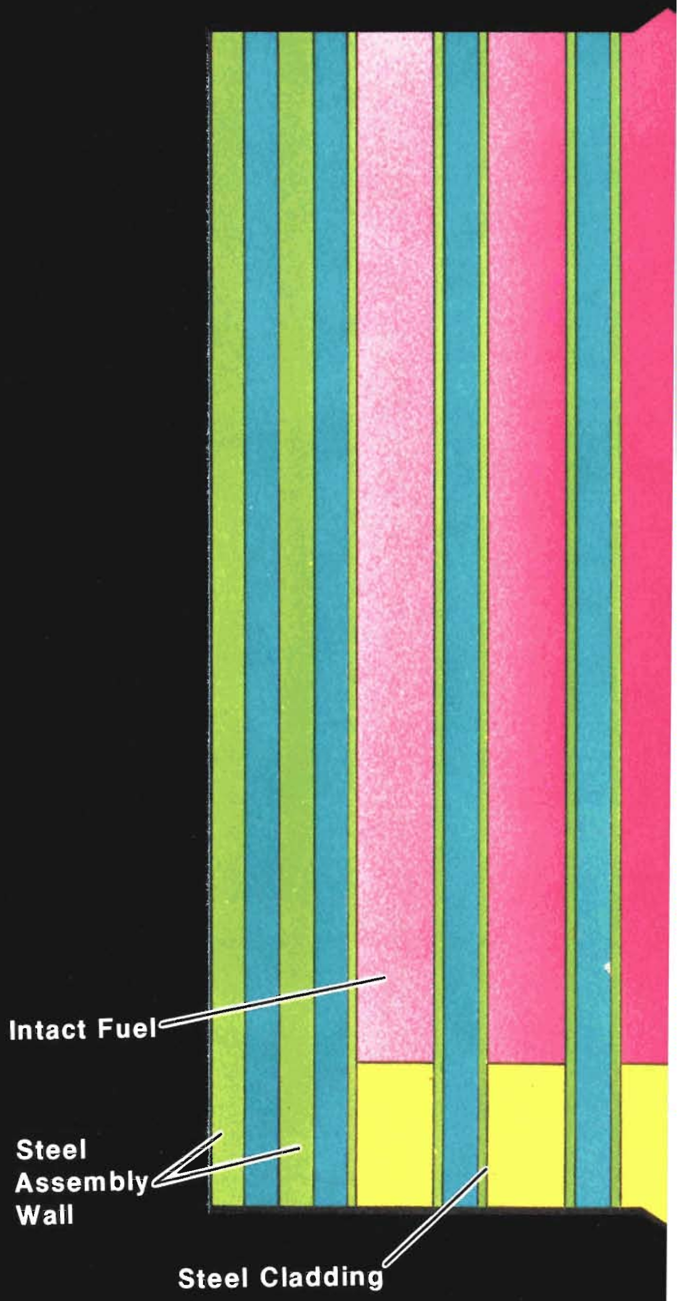
What happened at Three Mile Island is at once vindication and indictment. Even when safety systems were overridden, the accident was largely contained. No one was hurt—that is the vindication of our engineering. But the mess is still there, and we cannot seem to clean it up—that is the indictment of our complex political and socioeconomic system. Unwillingness to assume responsibility is not a problem of the federal government or of the state government or of the nuclear industry or of the engineering profession or of any other group. It is a collective problem, and we must recognize it as such. Our state and federal governments have the authority and the resources to deal with Three Mile Island. The commitment and the will to use this capability must soon be brought to bear. The nation's best interests are not being served by continued delay. ■

The author obtained information for this article from the Office of Management and Budget, General Public Utilities, the Nuclear Regulatory Commission, the Department of Energy, the Securities and Exchange Commission, the General Accounting Office, the United States Senate, the United States House of Representatives, the Atomic Industrial Forum, the Edison Electric Institute, the Electric Power Research Institute, the State of Pennsylvania, and numerous other state, federal, and private entities, as well as individuals.

BREEDER REACTOR SAFETY-MODELING THE IMPOSSIBLE

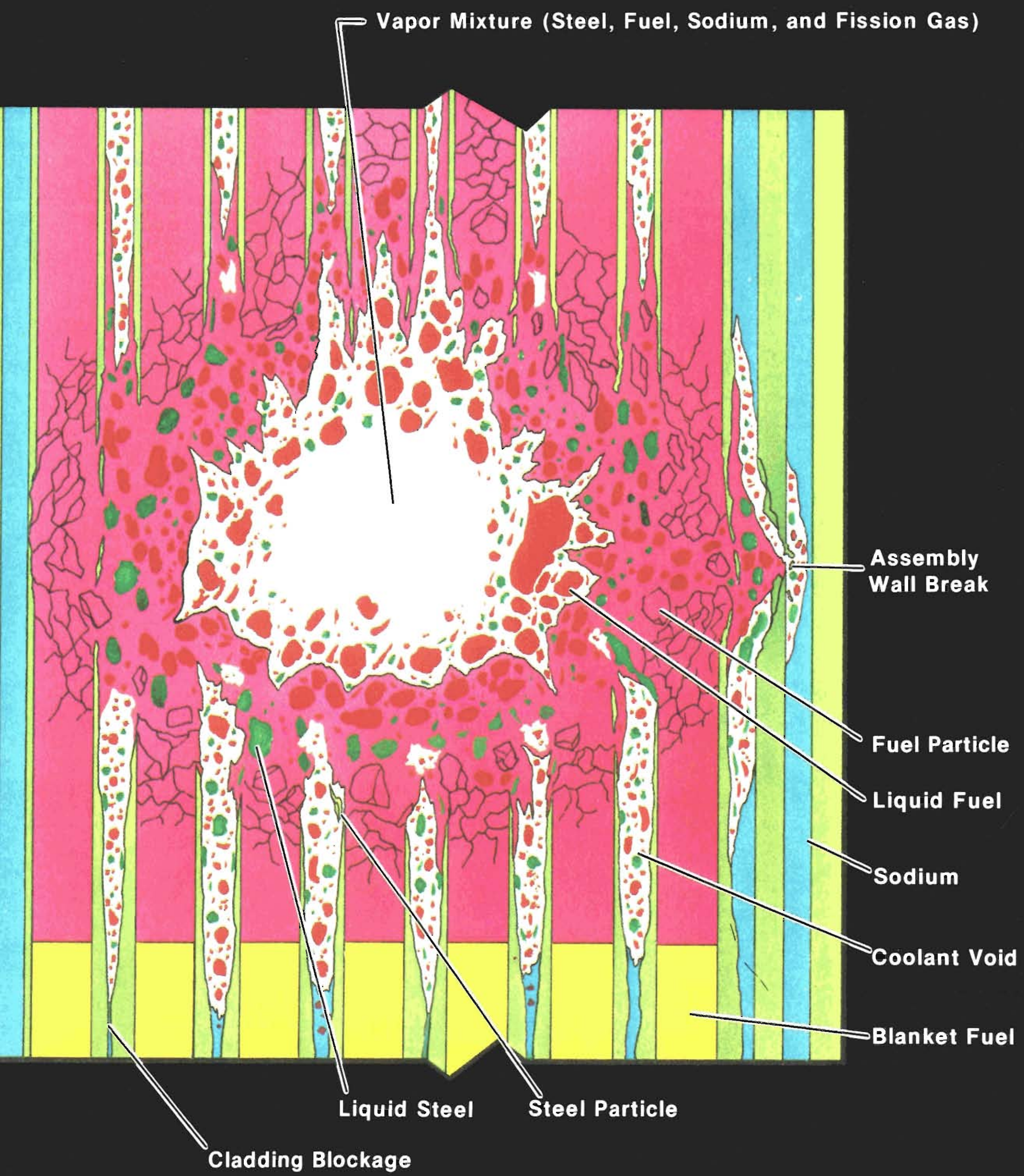
by Charles R. Bell

SIMMER, a computer code developed at Los Alamos National Laboratory, was designed to model an incredibly complex event—the disintegrating of the core of a liquid-metal fast breeder reactor during a hypothetical meltdown accident. Analyses with SIMMER are leading to a more realistic assessment of the safety of this type of reactor.



Is the breeder reactor an alchemist's dream come true? The ancient alchemists attempted to transform base metals of low value into noble metals of high value; the breeder transforms abundant isotopes of low value into rare but easily fissionable isotopes of high value. But the alchemists tried to achieve their transformations with chemical reactions. They were not aware that nuclear reactions, prompted by the collisions of neutrons and nuclei, could cause the hoped-for transformations.

All nuclear reactors depend on the generation of neutrons during the fission process to maintain a chain reaction, but a breeder is designed to produce an overabundance of neutrons. These excess



Artist's conception of disintegrating fuel pins in the fuel assembly of a liquid-metal fast breeder during a severe accident.

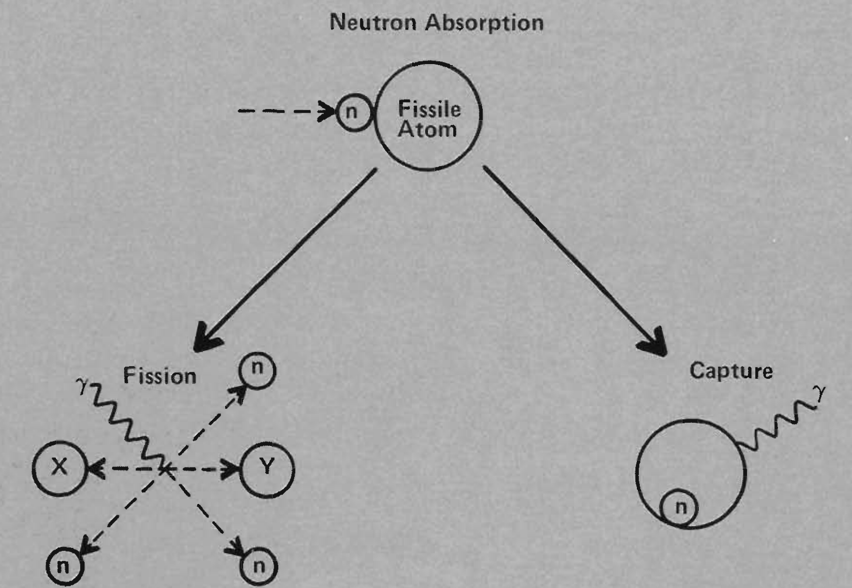
neutrons perform the alchemist's magic as they convert "fertile" fuel incapable of sustaining a chain reaction into "fissile" fuel at a rate faster than needed to replace the original fuel. Hence, in a breeder, not only is energy generated, but excess fuel is "bred."

But can the transformation process be adequately controlled? Physicists and engineers have long recognized that during certain hypothetical, low-probability accidents a burst of neutron production could generate a damaging surge of energy and pressure resulting in a potentially large radiation release to the environment.

Thus, a strong attack on these hypothetical accidents has been mounted, both to prevent them and to understand their consequences. An important analytical tool necessary for this understanding was developed over the last six years by the Energy and the Theoretical divisions at Los Alamos National Laboratory. The tool, SIMMER,* is a complex computer code that examines the dynamics of a typical breeder core when disrupted during hypothetical accidents. The core being studied is that of a liquid-metal fast breeder reactor, the breeder of primary focus in the United States since the 1960s. The code couples an accounting of neutron population and power generation with a fluid-dynamic calculation of the behavior of all core materials. Initially considered an almost impossible problem, the development of this code is leading to a more realistic assessment of the safety of liquid-metal fast breeders based on detailed knowledge concerning the temperatures, pressures, energies, and mass flows in a disrupted core.

The conclusions so far? The pressure and temperature surges expected in hypothetical accidents appear to be much less than previously estimated. As a result, management of this sophisticated alchemy

*SIMMER is an acronym for S_n, implicit, multifield, multicomponent, Eulerian, recriticality.



absorptions = fissions + captures

$$N = N_f + N_c$$

$$\text{neutron yield per absorbed neutron} = \left(\begin{array}{c}\#\text{ of neutrons}\\ \text{released per fission}\end{array}\right) \times \left(\begin{array}{c}\text{fraction of absorptions}\\ \text{that cause fissions}\end{array}\right)$$

$$\eta = \nu \times \left(\frac{N_f}{N_f + N_c}\right)$$

Fig. 1. When neutrons are absorbed by fissile atoms, a fraction of these cause fissions releasing other neutrons, but the remainder are captured. The neutron-yield parameter η accounts for this fraction as well as for the average number of neutrons released per fission event.

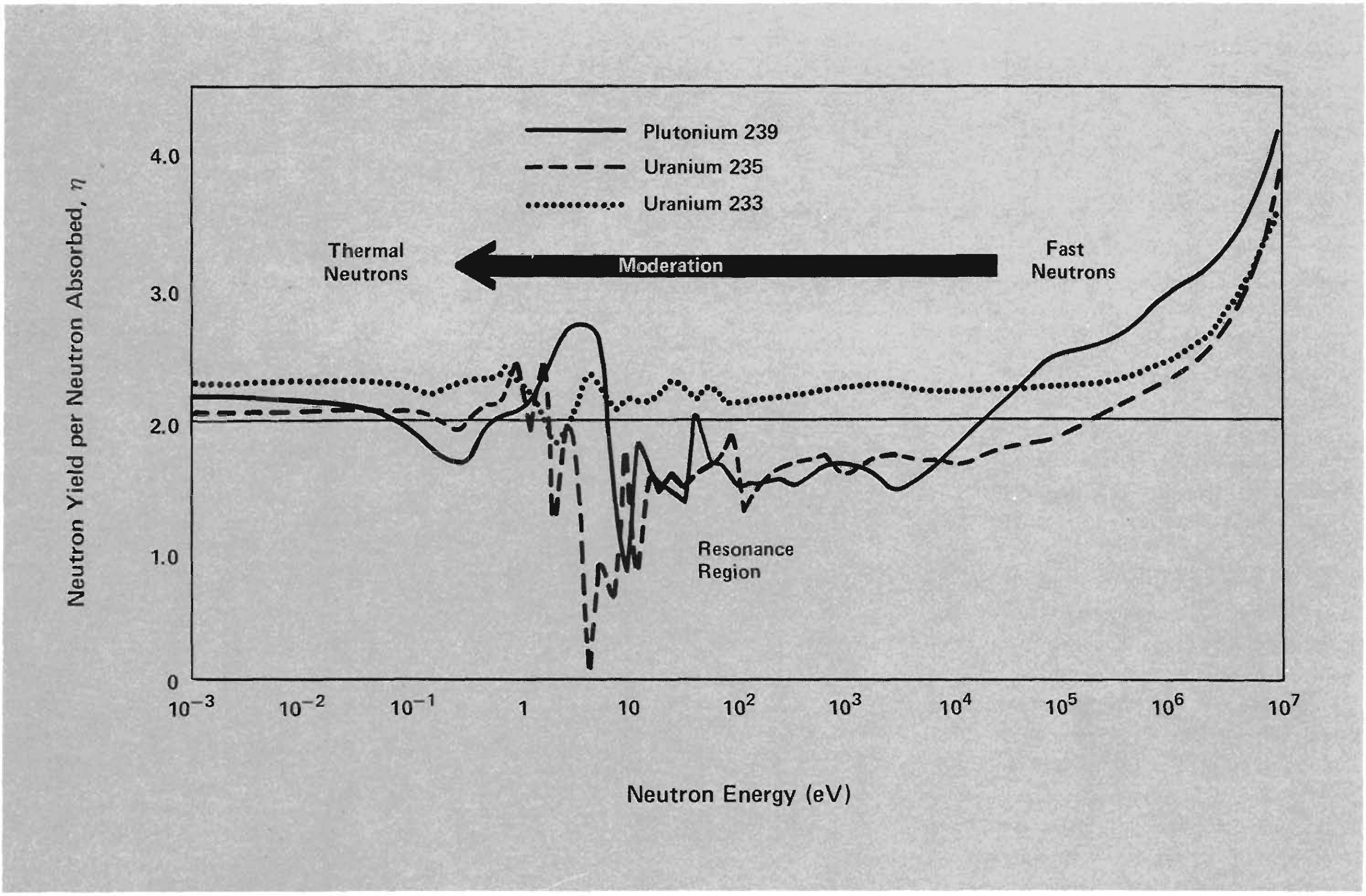


Fig. 2. The neutron yield per neutron absorbed, η, as a function of neutron energy. Fast neutrons have an η large enough to achieve breeding: that is, η is significantly larger than the "break-even" value of 2. Plutonium-239 has the highest η for fast neutrons, but uranium-233, the fissile isotope in the thorium-uranium breeding cycle, has η larger than 2 over most of the neutron energy range. (Figure adapted from ERDA-1541, Vol. 1, p. II-77, June 1976.)

may be safer than imagined.

The Liquid-Metal Fast Breeder

NEUTRONICS. The key to a breeder is the neutron. The generation of energy, the control of the reactor, the breeding of new fuel, and particular safety problems unique to breeders are all related to the reactor's neutronics. For example, the pioneers in nuclear physics early recognized the potential for breeding from the fact that each nuclear fission releases two or three neutrons. Since one of these neutrons is necessary to continue the chain reaction, then one or two "byproduct" neutrons remain. Some of these byproduct neutrons can be utilized through transmutation processes to produce fissile isotopes from fertile isotopes. [The two main processes contemplated for breeders are the "uranium-plutonium cycle" ($^{238}U + n \xrightarrow{\gamma} {}^{239}U \xrightarrow{\beta} {}^{239}Np \xrightarrow{\beta} {}^{239}Pu$) and the "thorium-uranium cycle" ($^{232}Th + n \xrightarrow{\gamma} {}^{233}Th \xrightarrow{\beta} {}^{233}Pa \xrightarrow{\beta} {}^{233}U$). In both cycles the first isotope is fertile, the last is fissile.] If, on the average, one byproduct neutron per fission is used to provide replacement fuel, the remaining neutrons are free to breed excess fuel.

Of course, there are a variety of fates possible for a neutron wandering free in a reactor core. Some neutrons are lost to other processes: for example, absorption by nonfuel constituents or leakage from the core. Other neutrons are captured by the fissile-fuel atoms, but are only absorbed; there is no fissioning and no further release of neutrons. As a result, efficient breeder design, like efficient financing, attempts to maximize the return yield for each investment, that is, attempts to maximize the number of neutrons liberated for each absorbed neutron. This concept and the parameter η representing the neutron yield per absorbed neutron are illustrated in Fig. 1. For a nonbreeder reactor if η is enough larger than 1 to account for neutron losses, a chain reaction can be maintained. However, for a breeder η should be significantly larger than 2.

Figure 2 is a plot of η versus neutron energy for three fissile isotopes and shows the most promising region for breeding to be that of "fast" or high-energy neutrons. The parameter η rises here because the ratio of fissions to captures for absorbed neutrons is rising with neutron energy; the more energy the neutron brings to the interaction, the more likely the fission event. The figure also shows that, with respect to neutron yield, plutonium-239 is the preferred choice for breeder fuel.

Neutrons released in the fission process are already fast. Commercial light-water reactors reduce these energies, with the water acting as moderator, to the thermal region where there is high total absorption of neutrons by the fuel. But to take advantage of a large η, the neutrons must remain fast, so breeders are purposefully designed with little moderation.

How is this population of neutrons and the ensuing chain reaction controlled? The neutrons resulting directly from the fission process are called "prompt" neutrons (Fig. 3). In liquid-metal fast breeders, the prompt neutrons have an average "lifetime" of about 10^{-7} second between their birth and their death due to absorption or escape from the core. Since neutron population growth is exponential, a slight overpopulation of prompt neutrons will grow very rapidly to a new, larger population. For example, a neutron population only 0.1 per cent in excess of the stable, "critical" population will take slightly longer than 10^{-4} second to triple. This is much too fast for realistic control of the reactor.*

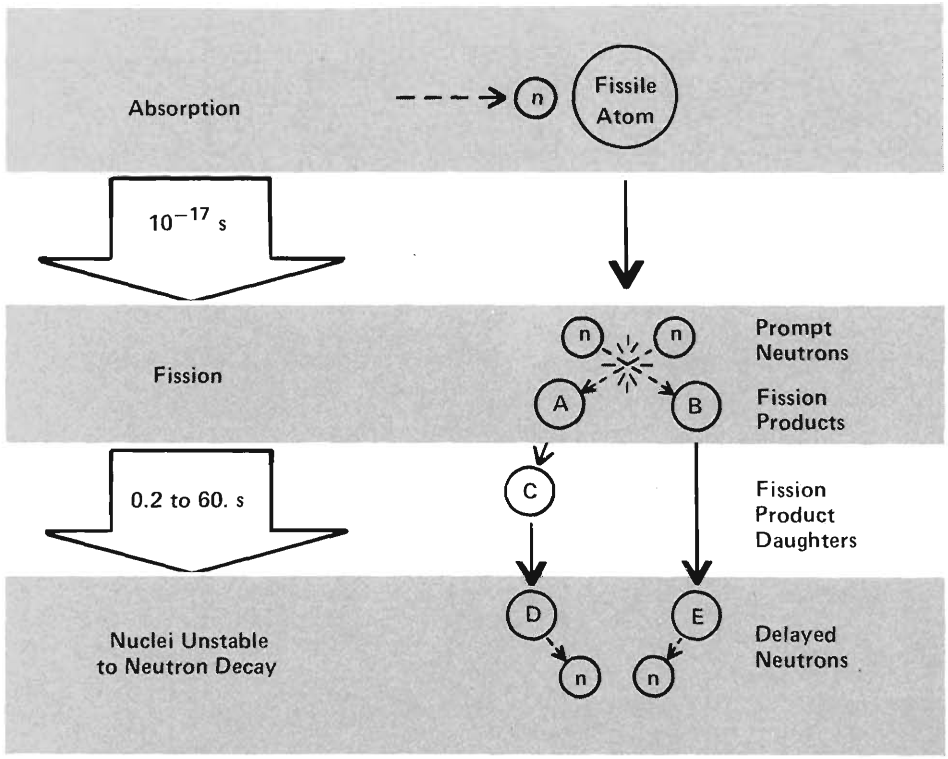


Fig. 3. Prompt neutrons are released during fission almost immediately after absorption of the neutron triggering the event. Delayed neutrons are released considerably later whan a fission product or a fission-product daughter is produced that is unstable to neutron decay. Other radiations have been eliminated for clarity.

Fortunately, "delayed" neutrons are also part of the total neutron population. These neutrons are released in the decay of some of the unstable nuclei produced by the original fission event and are thus born on the order of seconds after that event. If the neutron population is always kept underpopulated, or subcritical, with respect to prompt neutrons, delayed neutrons can be used either to complete a stable, balanced population or to provide the slight imbalance needed to alter the population. The growth to a new stable population will then be slowed considerably and mechanical control of the process becomes feasible.

The Clementine reactor, designed and built in 1945 at Los Alamos, successfully demonstrated the feasibility of delayed neutron control of a fast-neutron system. This very small reactor had a cylindrical core 0.15 meter in diameter and 0.14 meter in length, 35 plutonium metal fuel rods, and used mercury as a coolant. It produced 25 kilowatts at full power.

CORE. The special need of the breeder to generate and harbor neutrons efficiently dictates a core design that differs in several key respects from the thermal-neutron reactor core. First, breeder cores are made compact to minimize the masses of nonfuel materials such as stainless steel and coolant. These materials decrease the average yield by moderating neutron energies and the neutron availability by absorption.

Second, the fuel is highly enriched. While it is true that fast neutrons absorbed by fissile atoms result in high neutron yields (large η), the absorption cross section, that is, the probability for those absorptions to occur in the first place, is several orders of magnitude lower than for thermal neutrons. The solution to this problem is to increase the number of targets by increasing the density of fissile atoms. Thus, a typical breeder fuel is enriched to between 20 and 30 per cent in the fissile isotope compared to 3 per cent in light-water reactor fuel. A further advantage to high fissile-fuel density is that neutron mean free paths are kept small compared to the size of the core, thus

**The same is true of commerical reactors using thermal neutrons (see Fig. 2), except average lifetime is about 10^{-4} second and the approximate tripling time is 0.1 second.*

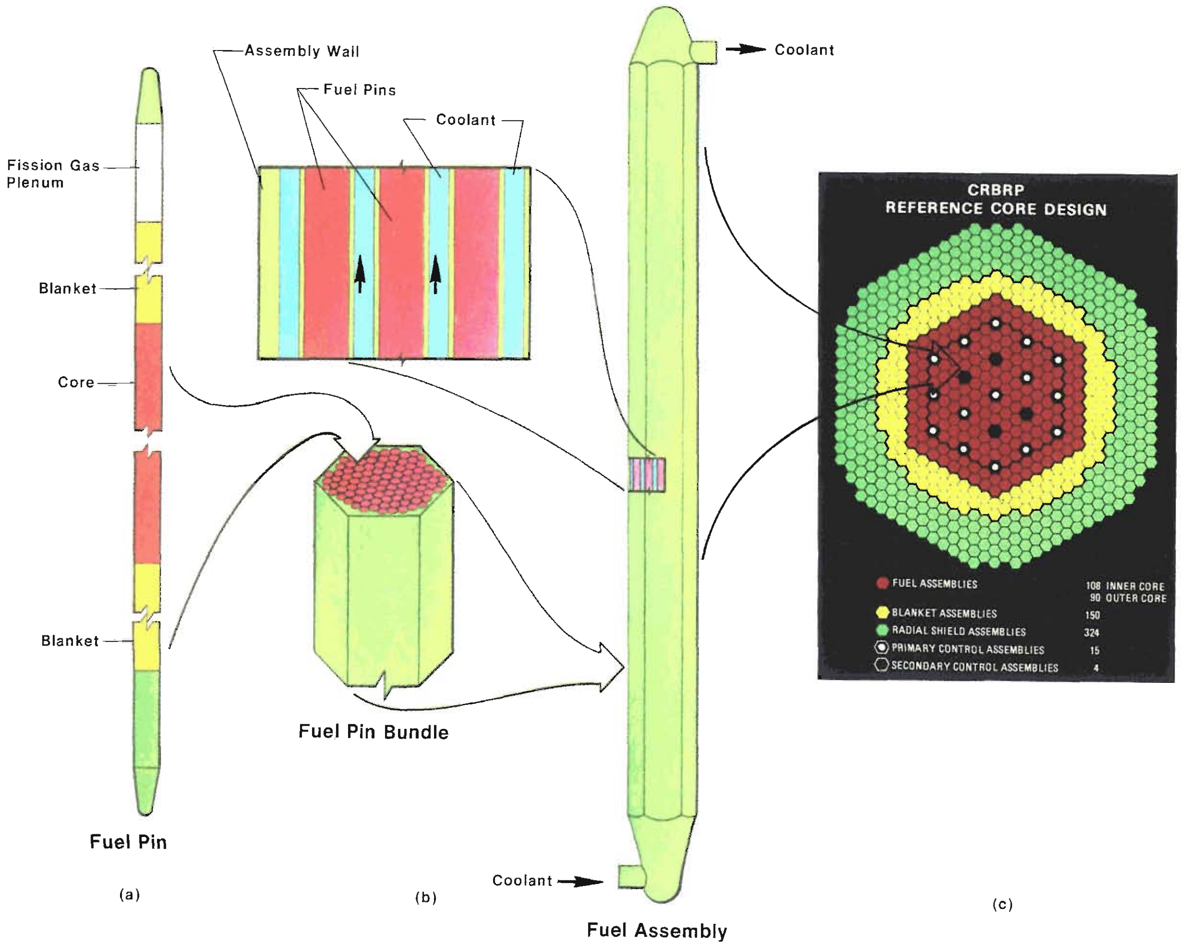


Fig. 4. (a) Fuel pins are long (2.8 meters), narrow (6 millimeters in diameter), and covered with a stainless-steel cladding (0.4 millimeter thick). There are three distinct regions of fuel in each pin with blanket fuel (yellow) placed above and below active-core fuel (red). A hollow region at the top serves as a fission gas plenum. (b) Two hundred seventeen fuel pins are bundled inside each stainless-steel fuel assembly. Liquid sodium coolant (arrows) flows up through the fuel assembly next to the fuel pins. (c) Fuel assemblies (red) are arranged into a core array along with blanket fuel assemblies (yellow) and control assemblies (white for primary, black for secondary). Radiation-shield assemblies (green) surround the array.

reducing neutron leakage.

The combined result of a compact core and high fuel density is high power density, typically about 300 megawatts per cubic meter. This necessitates a coolant system with good heat-transfer properties: for example, liquid-metal coolant flowing over large surface areas of the fuel.

Finally, a breeder core contains both an active core that supports the chain reaction and a blanket of fertile fuel that completely surrounds this core and captures escaping neutrons.

The manner in which these features are brought together for the proposed Clinch River Breeder Reactor, chosen for construction near Oak Ridge, Tennessee, is shown in Fig. 4. To create the necessary heat-transfer area, the fuel is encapsulated in thousands of small-diameter (6-millimeter) pins clad with stainless steel. Each of these pins is long (2.8 meters) and consists of several sections. The center section contains the active-core fuel, typically an oxide of the fissile isotope plutonium-239 mixed with the oxide of the fertile isotope uranium-238. Above and below the core section are blanket sections of fertile fuel, usually "depleted" uranium (99.8 per cent uranium-238 and 0.2 per cent uranium-235). A hollow section at the top of the fuel pin serves as a fission-gas plenum that collects the gaseous fission products diffusing out of the fuel below.

Between 200 and 300 of these fuel pins are bundled into a hexagonal array within a fuel assembly. Liquid sodium coolant enters the assembly at the bottom, flows upward among the fuel pins, then out the top.

Fuel assemblies are themselves positioned in arrays to form the active core. Blanket assemblies of fertile fuel are arranged about the active core to complete the blanket horizontally. Control rods are included among the fuel assemblies, and shield assemblies encircle the entire array to block escaping radiation. In some core designs, blanket assemblies are placed within the active core to enhance breeding.

The main features of the reactor vessel for the Clinch River Breeder are shown in Fig. 5. The vessel is approximately 6 meters in diameter and 17 meters in length. The active core, 1.8 meters in diameter and 1 meter in length, requires a fissile inventory of approximately 2 metric tons.* Two types of control rods with completely independent actuation systems assure reactor shutdown when demanded either by the automatic response of the control system or by the operators. Since the coolant is liquid sodium, which readily burns in air, the vessel must be carefully sealed and given an inert atmosphere. The closure head is, therefore, a complex structure of thermal shields and rotating, eccentric plugs that allow remote-control refueling operations. The sodium enters at the bottom (arrows), moves up through the fuel assemblies, mixes in the upper sodium pool, and finally flows out of the vessel to heat exchangers.

Despite high chemical reactivity with air and water, liquid sodium has several properties that make it an excellent coolant for fast breeders. First, it is a relatively poor moderator.** Second, its thermal conductivity allows for rapid transfer of heat within the high-power-density core and so eliminates concerns about local boiling, surface dryout, and fuel overheating. Third, the metal is a liquid over a large temperature range. Thus, the operating temperature can be high enough to achieve high thermal efficiencies of about 40 per cent, yet is still 300 kelvin below the boiling point. Finally, the system operates at atmospheric pressure.

The characteristics of low operating pressure and the capability of large heat removal prior to coolant boiling have led many engineers to argue that the liquid-metal fast breeder is generically safer than the light-water reactor. If control and safety systems function as intended, the possibility of a core meltdown should be no higher, and perhaps even less likely, than for the light-water reactor.

The Severe Accident

REACTIVITY CHANGES. Despite those generic safety features, certain aspects of the fast breeder raise a special safety threat in many people's minds. These aspects center, once again, on the neutron.

Unlike the light-water reactor, the core of a liquid-metal fast breeder is not in its most reactive configuration. Typically, there are several critical masses of fuel present in the breeder core. This fact, in conjunction with the core's high power density, led people to conjecture accidents in which the fuel melted and slumped into a supercritical configuration. Any similar slumping and compaction of fuel in a light-water reactor would eliminate coolant, which is important as a moderator. With the elimination of coolant, the population of thermal neutrons driving the chain reaction in the light-water reactor would decrease. In contrast, elimination of coolant in a breeder would actually increase the neutron population.

This concern about meltdown led to an emphasis on hypothetical core-disruptive accidents for liquid-metal fast breeders. Why hypothetical? Accidents become severe when there is a sustained inability to remove heat from the fuel at a rate commensurate with its generation. For this to happen, two major systems must fail. For example, if the reactor control system fails to control the power level, the safety system will scram or shut down the reactor. But if both systems fail, then a major thermal or heatup excursion can occur. Likewise, a severe acci-

**A commercial light-water reactor of the same power capacity (1000 megawatts thermal) requires only 1 metric ton of fissile material.*

***The hydrogen atoms of water have a mass equal to that of the neutron and so can absorb a large portion of the neutron's energy in a single collision; the heavier sodium atoms (atomic mass = 23) cannot.*

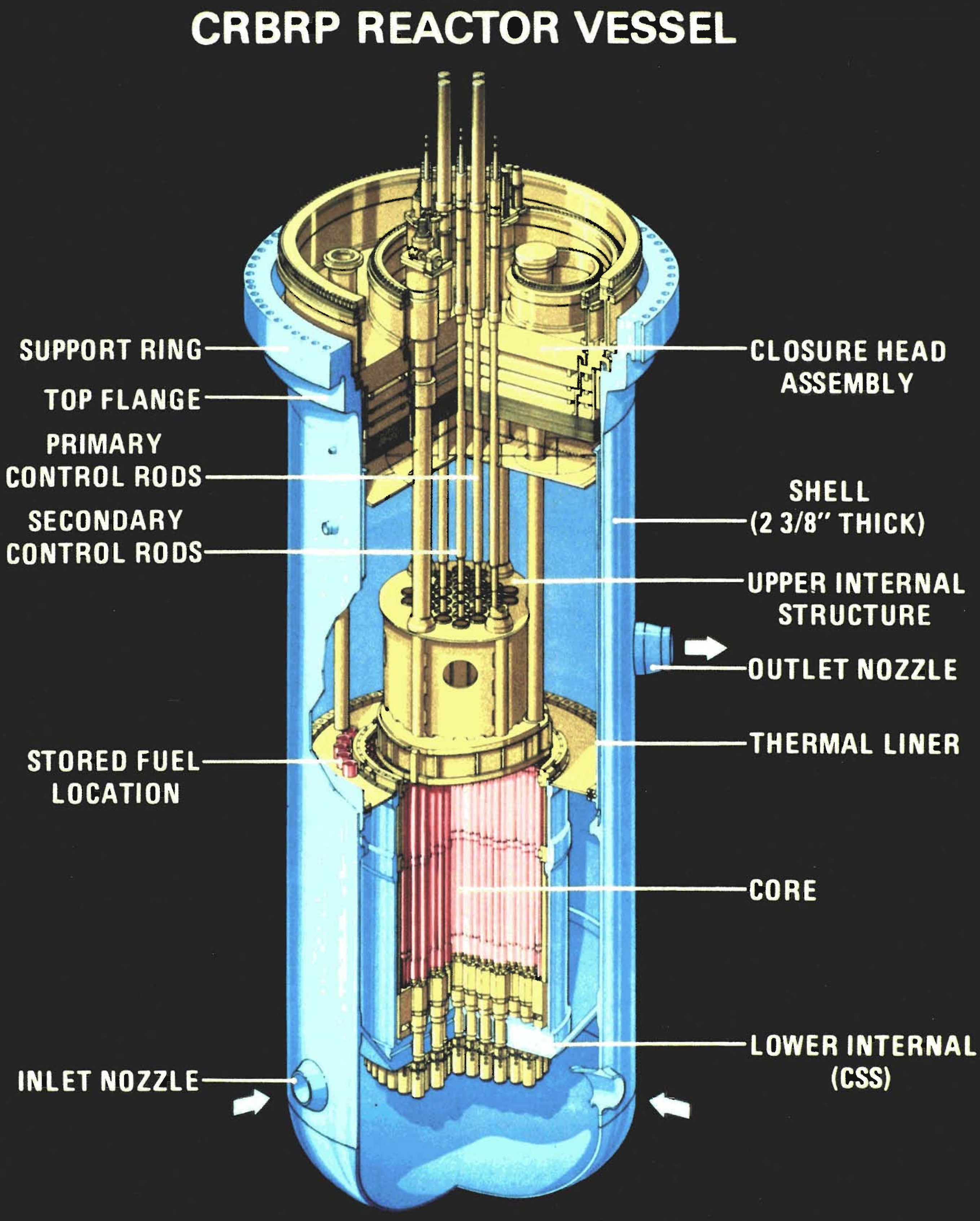


Fig. 5. The reactor vessel for the proposed Clinch River Breeder Reactor plant (CRBRP). Illustrated are the array of fuel assemblies, the associated control mechanisms, the core-support structure (CSS), and the primary containment vessel. Liquid sodium coolant enters at the bottom, flows up through the core and fuel assemblies, and out the vessel to the heat exchangers. (Courtesy CRBRP Project Office, Oak Ridge, TN 37830.)

dent will happen when heat transport from the core deteriorates due to pump failures or extreme pipe breakage, but again, only when coupled with failure of the safety system to scram. Other possibilities are associated with malfunctions of the redundant heat removal systems.

The system failures described can, in some cases, result in disruptive feedback: the failure causes heatup, which can melt fuel; if the fuel collapses, it creates excess neutrons and a power increase which accelerates the heatup. Central to this feedback are changes in the neutron balance called "reactivity." Positive reactivity involves the creation of excess neutrons, which, of course, increases the disruptive feedback; negative reactivity would moderate or decrease the feedback.

If the generation of excess neutrons is large enough, the controlled balance maintained by the delayed neutrons will be destroyed and there will be a sudden and very rapid rise in power. The reactor has gone "super-prompt-critical," a potentially dangerous condition that, in severe cases, will terminate only after the high pressures of hot, vaporizing core materials have dispersed the fuel.

Unfortunately, the liquid-metal fast breeder core designs that are most desirable in terms of performance and economics also have an undesirable domination of positive reactivity during severe heatup transients. For instance, core structural materials and coolant both absorb neutrons and moderate neutron energies. Loss of these materials contributes to positive reactivity both by increasing available neutrons and by causing a shift to higher neutron energies and, thus, higher neutron yields (Fig. 2). This happens if the liquid sodium boils or otherwise voids around the fuel pins, or if the stainless-steel cladding melts and flows from the core.

Fortunately, the positive reactivity is moderated by several inherent negative reactivities. The release of fission-product gases and the thermal expansion of core materials tend to disperse the fuel. Specific core design features can be incorporated that aid dispersal during a core-disruptive accident. Also, more neutrons are absorbed in the fertile fuel as temperature increases (the Doppler effect*).

AN ACCIDENT SCENARIO. The analysis of a postulated accident is designed to determine, with reasonable confidence, the degree of any positive reactivity, its transient behavior, and its potential for damage. Details of one accident scenario—failure of all primary sodium coolant pumps and complete failure of the reactor shutdown system—will provide perspective on the complexity involved in this chore.

First, there is gradual loss of heat transport from the core as coolant circulation slows to a stop. Different fuel assemblies have different power levels and flow rates, so they heat up at different rates. The Doppler effect and thermal expansions of sodium, fuel, and steel help keep reactivity feedbacks and power changes small. Soon, though, sodium boiling and voiding in fuel assemblies with the highest ratio of power to coolant flow lead to a net positive reactivity effect and accelerated heatup; then voiding begins in other fuel assemblies.

During this phase (the first 10 to 30 seconds) the dominant phenomena are the transfer of heat between materials, boiling and voiding of the coolant, and condensation of sodium vapor on cold structures at the ends of the various assemblies. Neutronics remains relatively smooth throughout the core. The analysis of this phase is simplified by the constant geometry of the core and the quasistatic character of the phenomena.

In the second phase, heat removal is highly degraded because the coolant has voided. Melting of fuel and cladding occurs on the order of a second after sodium voids are formed in a particular fuel assembly. The resulting complex situation is shown schematically in the opening figure of the article. The melted fuel pins in the central portion of the active core form a two-phase column of liquid fuel, liquid steel, fuel fragments, fission gas, fuel vapor, and steel vapor. The liquid components are immiscible, with greatly different melting points, viscosities, and thermal conductivities. The gaseous phase has been generated primarily near the midplane where the fission power was initially highest. The pressure from the gas and vapors expels liquids and fragments to the ends of the active core and into the colder axial blankets.

Analysis of this phase of the accident thus requires multiphase, multicomponent fluid dynamics and variable core geometry. Moreover, characteristics of the fluids vary greatly depending on initial conditions, time into the accident, and status of individual fuel assemblies.

The fission process causes the fuel to have a higher temperature than the other materials within the fluid mixture. Thus, a bewildering variety of heat-transfer processes and associated phase changes are possible. Some of the possible phase changes are illustrated in Fig. 6. All these processes are transient and have feedback effects on fluid displacement that, in turn, produce feedback effects on neutronics, reactor power, and fuel temperatures. The feedbacks are not only strong, but also highly nonlinear due to exponential relationships between liquid temperatures and vapor pressures and between material displacements and power. In fact, neutronic response changes from a relatively smooth core-wide distribution to one with local distortions and sudden peaks.

Although this complex, chaotic behavior occurs within individual fuel assemblies, the

**Intermediate-energy neutrons are strongly absorbed at specific energies in the resonance region between thermal and fast neutrons (Fig. 2). As the temperature increases, these absorptions broaden, allowing absorption of a larger range of energies and, therefore, a larger fraction of the limited numbers of these neutrons.*

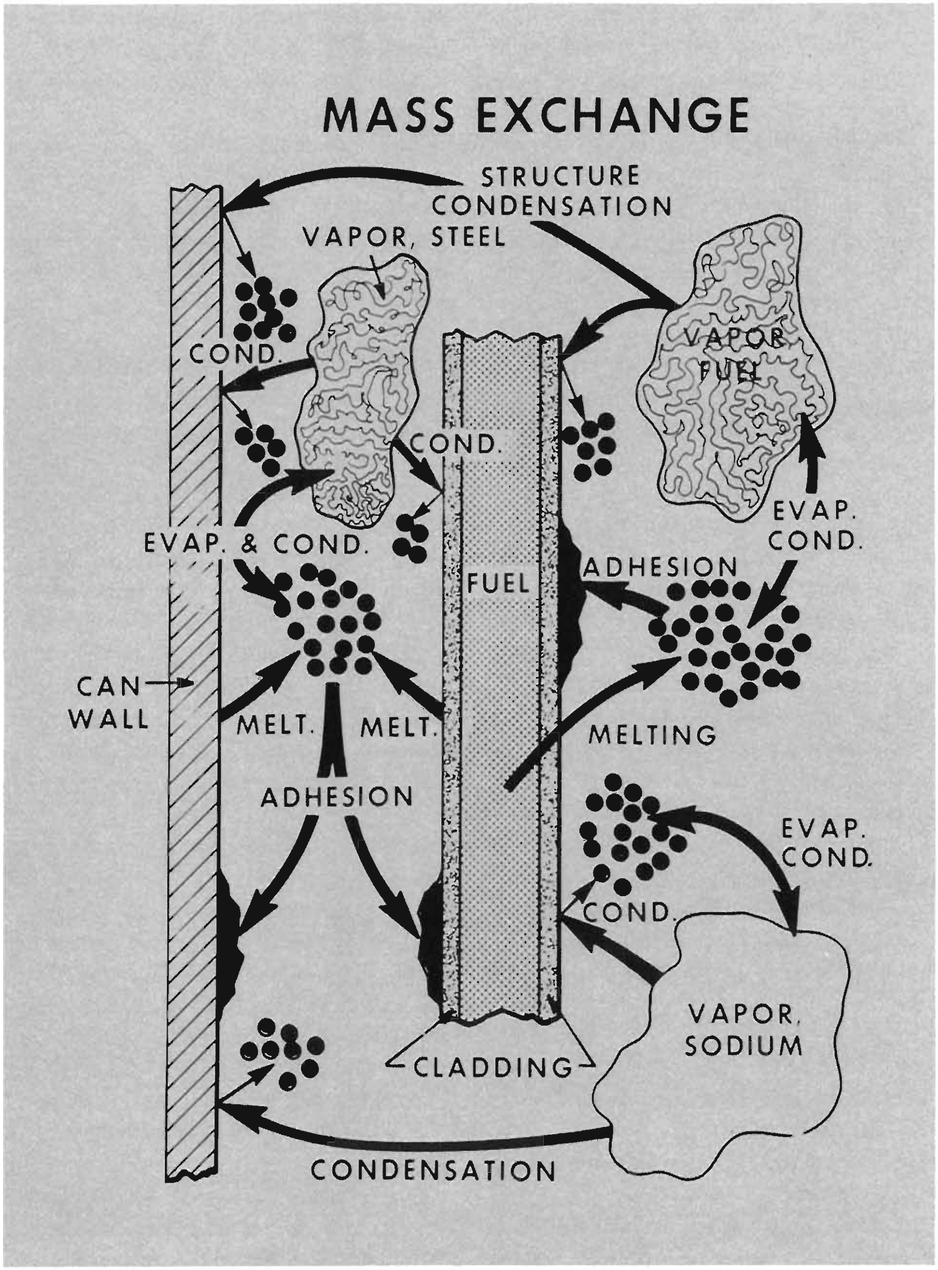


Fig. 6. Typical phase changes possible during a severe accident.

overall disruption process must be viewed on a core-wide basis. Reactor neutronics and power are dependent on the behavior in all the assemblies, and the common boundary conditions prevailing at the inlet and outlet sodium plena produce a core-wide fluid-dynamic coupling.

The third phase, designated the "transition phase," occurs when the fuel-assembly duct walls begin to melt. Now the flow field becomes multidimensional on a core-wide scale. Local fluid dynamics are no more complex than in the previous phase, but the extent of material relocation within the core may be much greater. This results in a difficult neutronics problem because of large voided and compacted regions where neutron mean free paths are no longer small compared to the dimensions of the medium. The large voids introduce strong directional effects, known as neutron streaming, which are important to this phase of the accident.

Neutronic coupling in the transition phase is potentially stronger because disrupted fuel is no longer in discrete assemblies; thus larger masses of fuel are capable of concurrent motion. Changes in reactivity can be both large and rapid. At this point two different outcomes are possible depending on such things as the reactor design or the reactor state at the beginning of the accident. First, if a large fraction of the original fuel has managed to remain within the active core region, a super-prompt-critical excursion can occur that heats the fuel in milliseconds to high temperatures and pressures. The fuel in the core, in essence, blows apart. While the dispersal of the fuel terminates the neutronic excursion, the pressure surge poses a direct mechanical threat and the possibility of breached containment.

On the other hand, if the fuel inventory has been reduced to about half the original amount by gradual leakage, or if large quantities of blanket materials have diluted the fuel, a severe power excursion will probably not occur. Thus, fluid-neutronic coupling becomes weak and of little further importance. The threat along this path is a potential meltthrough at the bottom of the reactor vessel from decay heat sometime within a few days.

SAFETY DESIGN. From this brief description of a severe accident, it is possible to understand why many knowledgeable scientists and engineers have considered detailed mechanistic analyses of these accidents to be an impossible problem. An appropriate description of the thermodynamic, fluid-dynamic, thermal, and neutronic behavior has been judged beyond the state of the art. As a result, the hypothetical core-disruptive accident has, until recently, been dealt with in two ways.

First, engineers have attempted to design reliable systems with very low probabilities of even entering the severe accident regime. Second, the complexity of the problem has been side-stepped by basing designs on highly conservative bounding estimates of the "damage potential." This is the potential for neutronically heated core materials to produce high pressures damaging to containment structures and is typically based on the assumption of isentropic (reversible and adiabatic) expansion of the fuel. This approach worked well for small breeders such as the second experimental breeder developed by Argonne, EBR-II, which had a thermal power rating of 62 megawatts.

However, for the large breeders being considered today, the bounding approach places difficult if not impossible demands on design. For a hypothetical core-disruptive accident of a given energy-density level, damage potential increases approximately in proportion to reactor size, whereas the ability of reasonable designs to absorb damage scales weakly, or even inversely, with reactor size. As a result, if proof of containment for these accidents is necessary and it is tied to the demands of simple bounding analyses, an impasse may be reached in the licensing process.

SIMMER. Seven years ago, a team of engineers and scientists at Los Alamos decided to tackle this demanding problem. With the support of the Nuclear Regulatory Commission and the Department of Energy and with the use of the computer resources at Los Alamos, work was begun on the SIMMER code in 1974.

The approach adopted was to develop a generalized numerical framework based on the conservation laws of mass, energy, and momentum constrained by the initial and boundary conditions. Models representing the "physics" and consistent with physical laws or state-of-the-art understanding were to be inserted into this framework. This approach permits maximum flexibility for description of the physical interactions among materials, many of which were not well known at the time the work was begun. It also allows modeling sensitivities and uncertainties to be assessed in the interactive context of overall accident simulation. Thus, the impact of imperfections in knowledge can be established in a realistic manner.

EULERIAN FRAMEWORK. The "bookkeeping" needed to follow specific particles of material moving about and mixing together during a core-disruptive accident would be unbelievably complex. As a result, an Eulerian numerical approach was chosen for the fluid dynamics. This approach follows the evolution of material parameters at fixed points in space. The reactor core is divided by a mesh into a collection of cells, and the densities and energies of the material components are calculated at each cell as a function of time. This technique introduces some undesirable smearing of the transported entities within and between cells, but the approximation is considered acceptable.

While the fluid dynamics of core disruption are three-dimensional, considerable symmetry usually exists in the circumferential direction. Accordingly, a two-dimensional, cylindrical approximation is normally used in SIMMER, and each cell is specified by a radial and an axial coordinate.

MULTIPLE FIELDS. Because of the variety of material phases and components, certain simplifications must be incorporated into the fluid dynamics. The first simplification is to treat all materials having the same approximate velocity as a "field." Thus, one momentum equation at most needs to be assigned to each field. SIMMER uses three fields: structure, liquid, and vapor (Table I). The structure field includes all materials that are stationary in space. Besides the expected structural materials, fission gas trapped inside fuel pins and resolidified materials are components of this field. The liquid field includes all materials that flow: normal coolant, melted solids, and solid particles moving with these liquids. The vapor field, the gases, generates the pressure distribution that drives the motion of the liquid field as well as itself.

MULTIPLE COMPONENTS. Each field has a large number of material components, the densities and energies of which must be specified in each Eulerian cell. However, some of these densities and energies can be combined. For example, all gases in the vapor field have short thermal equilibrium times and so all gases in a given cell are given the temperature and energy of the mixture. On the other hand, since the location of fertile and fissile fuel must be considered individually for neutronic purposes, the density is specified separately for both in all three fields.

CONSERVATION LAWS. The Eulerian numerical framework with its three fields and multiple components must be formulated with conservation laws for momentum, mass, and energy as the foundation. There are only two momentum equations, one each for the liquid and vapor fields, but none for the structure field since it is assumed stationary. There are 23 mass equations and 12 energy equations, one for each density or energy component in each field.

The equations are in terms of densities, velocities, pressures, and internal energies, but they also include many source and sink terms to account for transfer of heat, mass, and momentum between fields. The mass equations account for vaporization, condensation, freezing, and melting; the momentum equations account for gravity and drag forces; the energy equations account for heat exchange and the various heats generated by phase changes, neutronics, viscous dissipa-

TABLE I.
FIELDS AND COMPONENTS OF SIMMER FLUID DYNAMICS

Field	Energy Components	Density Components	Geometry and Models
Structure (stationary)	Fuel pins Cladding steel Assembly wall steel Fuel crust Control pins	Fertile fuel Fissile fuel Cladding steel Assembly wall steel Fertile fuel crust Fissile fuel crust Contol pins Intergranular fission gas Intragranular fission gas	Fuel assembly geometry Order and thickness of layers on fuel pins Exterior surfaces exposed to liquid field Flow-channel diameters
Liquid (mobile)	Liquid sodium Melted fuel Melted steel Melted control pins Fuel particles Steel particles	Liquid sodium Melted fertile fuel Melted fissile fuel Melted steel Melted control pins Fertile fuel particles Fissile fuel particles Steel particles	Dispersed droplets in continuous vapor field Solid particle size Liquid droplet size from: surface tension internal pressure fluid dynamic forces (drag) coalescence model (collisional)
Vapor (mobile)	Vapor mixture	Fertile fuel vapor Fissile fuel vapor Steel vapor Sodium vapor Control pin vapor Released fission gas	Homogeneous mix Local vapor volume fraction

tion, and pressure-volume work.

EXCHANGE FUNCTIONS. The essential physics is introduced through the source and sink terms in the form of exchange functions. These functions give SIMMER its flexibility because each represents the modeling of a specific, independent physical process. Improvements in the understanding of specific phenomena are reflected as improvements in modeling and, hence, in the exchange functions that are inserted into the conservation equations. Currently, the modeling area receives considerable emphasis through both analysis and experiments.

One modeling area of particular concern is the manner in which materials break up. For instance, what distribution of particle sizes will result when a fuel pin begins to melt and crumble? What fraction will be liquid droplets and what fraction solid particles? Another area of concern is the physical properties of materials at high temperatures.

A typical modeling study starts with an attempt to describe the phenomenon by using standard engineering functions. In these equations appear various coefficients or exponents that correlate forces or material properties; for example, a Reynold's number may be used that relates dynamic pressure and viscous stress for vapor flow over a spherical particle. Many of these coefficients have parameters that depend on the specific geometry of the phenomenon being modeled: particle sizes, flow-channel diameters, heat-transfer contact areas. Thus, an important step in the analysis is a realistic description of the physical configuration.

Next, attempts are made to "benchmark" the model by comparing its predictions with experiments dealing directly with the phenomenon. Some of these experiments are done at Los Alamos, but most are carried out at other laboratories. SIMMER is run using only the exchange functions, the geometry, and those parts of the code necessary to the particular experiment being simulated. At this point, the model can be "fine tuned" by adjusting coefficients until the desired agreement between calculated and experimental results is achieved.

As implied by this discussion, the Eulerian mesh imposes a certain geometrical scale on the problem, but the exchange functions allow for effects due to a finer scale. Each function depends on local cell conditions, including the predicted characteristics of the microscale geometry.

The properties of the microscale geometry in the structure field are based on a typical

TABLE II

TRANSFER PROCESSES INCLUDED IN SIMMER

Source[a]	Process	Sink or Final State[a]
MOMENTUM TRANSFER		
Cladding Assembly wall Fuel or control material	Breakup	Liquid or vapor field
Vapor field	Pressure change	Liquid field
Cladding Assembly wall Fuel or control material	Melting (or freezing)	Liquid field
Liquid field	Vaporization	Vapor field
Steel vapor Fuel or control material vapor	Condensation	Liquid field
——	Viscous coupling transverse to flow	——
——	Liquid-vapor, drag-driven slip flow	——
MASS TRANSFER		
Liquids (fuel, steel, sodium, control material)	Vaporization	Vapor field
Fission gas trapped in fuel	Release	Vapor field
Vapors (fuel, steel, sodium, control material)	Condensing out onto structures	Liquid field
Fuel vapor	Condensation	Liquid fuel
Steel vapor	Condensation	Liquid steel
Sodium vapor	Condensation	Liquid sodium
Control material vapor	Condensation	Liquid control material
Solids (fuel, cladding assembly walls, control material, fuel particles, steel particles, fuel crust)	Melting	Liquid fuel Steel Control material
Liquids (fuel, steel)	Freezing out	Assembly walls
Liquid steel	Freezing out	Cladding
Liquid control material	Freezing out	Control material
Liquid fuel	Freezing	Fuel particles
Liquid steel	Freezing	Steel particles
Solids (fuel, fuel crust)	Breakup	Fuel particles
Solids (cladding, assembly walls)	Breakup	Steel particles

ENERGY TRANSFER

Fuel pins	Heat transfer	Cladding Liquids (fuel, sodium, steel, control material) Vapor field
Fuel crust	Heat transfer	Assembly walls
Fuel particles	Heat transfer	Liquids (steel, sodium, control material)
Cladding	Heat transfer	Control material Liquids (sodium, control material) Vapor field
Assembly walls	Heat transfer	Liquids (sodium, control material) Vapor field
Steel particles	Heat transfer	Liquids (sodium, control material)
Liquid steel	Heat transfer	Solids (cladding, assembly walls, steel particles, control material) Liquids (sodium, control material) Vapor field
Liquid fuel	Heat transfer	Solids (cladding, assembly walls, fuel particles, steel particles) Liquids (sodium, control material, steel) Vapor field
Liquid sodium	Heat transfer	Control material Liquid control material Vapor field
Liquids (fuel, steel, sodium, control material	Vaporization (or condensation)	Vapor field
Fuel	Melting	Liquid fuel
Steel particles	Melting (or freezing)	Liquid steel
Liquid fuel	Freezing	Fuel particles
Liquids (fuel, steel)	Freezing out	Assembly walls
Liquid steel	Freezing out	Cladding
Cladding Assembly wall Fuel or control material	Viscous dissipation during momentum transfer	Liquid field Vapor field
Liquid field	Viscous dissipation during momentum transfer	Vapor field
—	Viscous dissipation during momentum transfer by transverse viscous coupling	—

[a]Source and sink may reverse roles depending on local temperatures.

fuel assembly. Included are flow-channel diameters, the properties of surfaces interacting with the flow of the liquid and vapor fields, and the thicknesses and order of layered materials of the fuel pins, assembly walls, and solidified material.

The liquid field is generally viewed as dispersed immiscible droplets in a continuous vapor field. This concept is extrapolated to simulate situations where the liquid is continuous. Solid particle sizes are specified directly, but liquid-droplet diameters are computed by balancing local fluid-dynamic and surface-tension forces and by tabulating all local coalescenses.

All transfer processes expected to have any significant strength are included in SIMMER as exchange functions. These are listed in Table II. The sheer number of these processes illustrates the high degree of interactivity among physical processes attempted in SIMMER and required of any reliable mechanistic approach.

Also important is the ability of the code to deal with a wide range of transient phenomena in many environments. During an accident, stationary materials become mobile, flow obstructions such as duct walls are removed, and normal flow channels become blocked. This alteration of normal core geometry is included through exchange functions related to structure melting, structure disintegration by local thermal and pressure loads, and freezing of liquids on structures. Also, functions representing microscale geometry provide areas and distributions of particles and droplets for heat transfer, phase change, and coalescence phenomena calculations.

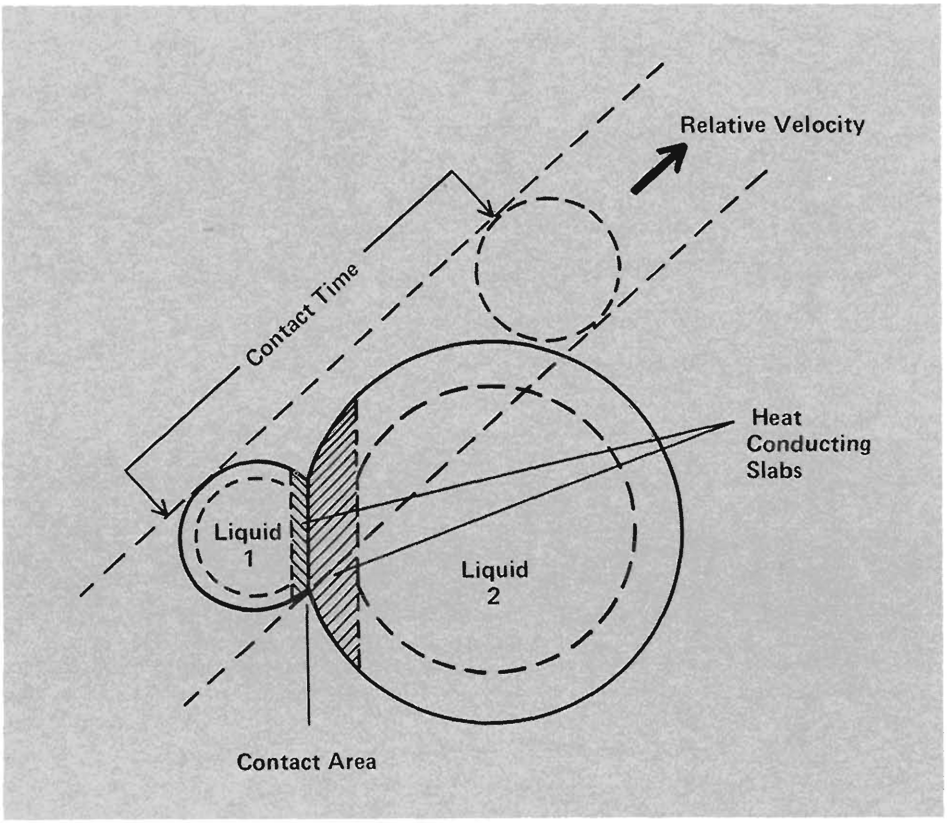


Fig. 7. Droplet-droplet collision and associated heat-transfer parameters.

A DROPLET HEAT-TRANSFER MODEL. As an example of the level of detail incorporated in the models used in SIMMER, the liquid-liquid heat-transfer model will be outlined here. This model assumes that each droplet sweeps out a volume determined by its radius and velocity and then collides with other droplets in that volume. The average fluctuating velocity used may differ from the liquid-field velocity or from component to component because of acceleration differences due to droplet sizes and densities. A collision rate between various liquid components is calculated for each mesh cell.

Figure 7 illustrates three other parameters used in the heat-transfer calculation. The collisional contact area is based on the cross-sectional area of the smaller droplet, but a correction factor is included to approximately account for relative velocity and angle of impact. The heat-transfer rate per unit area is assumed to be quasistatic and is based on conduction between the two slabs of material next to the contact surface. Each slab thickness is taken as 20 per cent of the droplet's radius; this short conduction length is due to the fact that most of the heat capacity in a sphere is effectively near the outer surface. Contact time is estimated from the time the respective droplet volumes intersect.

The heat-transfer rate per unit volume is the product of contact area, time of contact, heat-transfer rate per unit area, collisional rate, and the temperature difference between droplets. The final equation is in terms of cell parameters: the radius and temperature of each droplet, the liquid volume fraction, and the heat conductivities of both components.

EQUATION OF STATE. The multicomponent equation of state that completes the coupling between the fields can take one of two forms in SIMMER. An analytic equation-of-state approach requires direct input of material properties such as specific heats, heats of fusion and vaporization, and vapor pressures. The equations then relate the

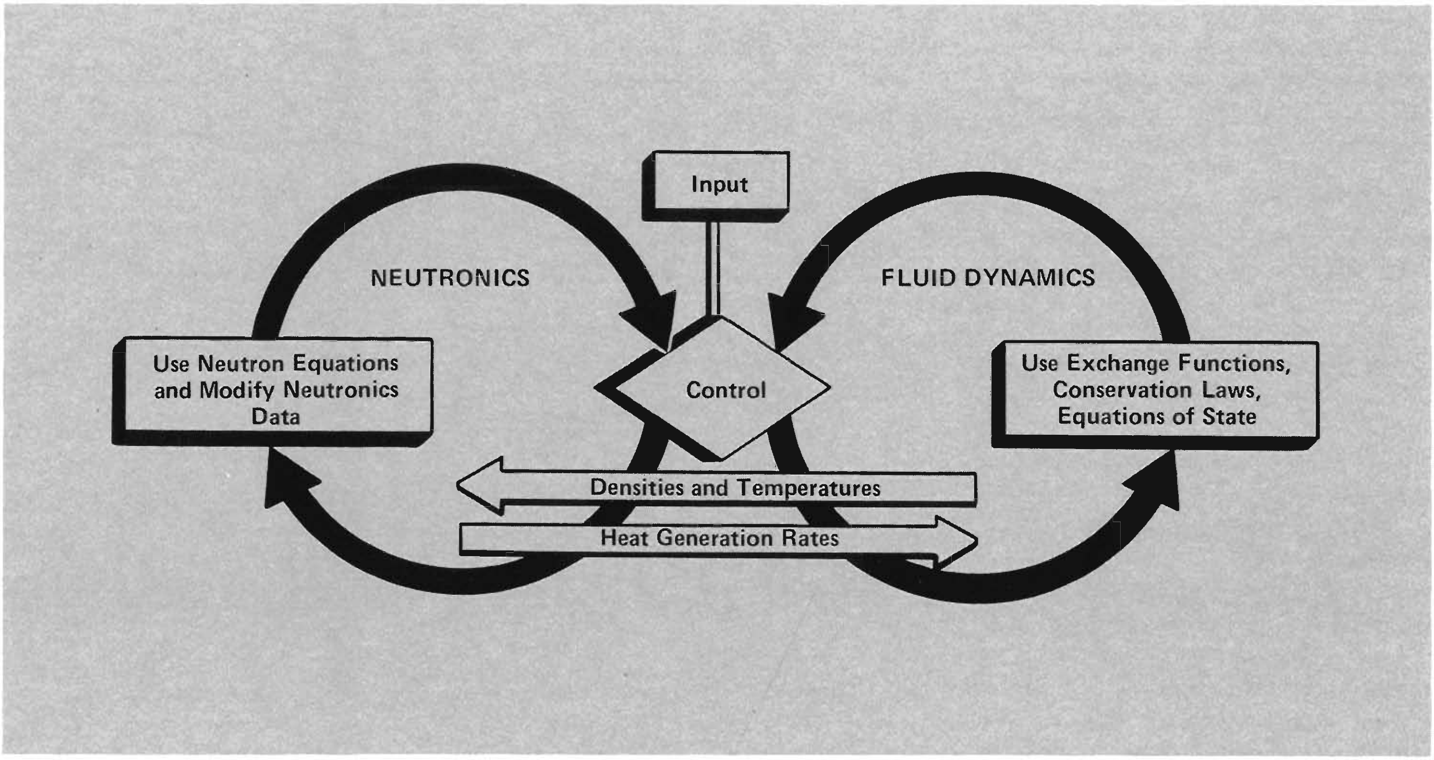


Fig. 8. Schematic of the coupling between the neutronics and the fluid-dynamics calculational loops in SIMMER.

microscopic density and internal energy of the components in each field to the pressure and temperatures. The reference energy states of all materials are synchronized to a temperature of 0 kelvin. Also, the vapor materials are assumed to obey the simple concepts of ideal gas mixtures.

The other approach is a tabular equation of state that enables data from the Los Alamos equation-of-state library, Sesame, to be used. This library provides a wider and more realistic data base than can be obtained with an analytic approach, but the interfacing of the conservation equations and exchange functions with the tables has not yet been fully completed.

NEUTRONICS. Coupled with the fluid dynamics analysis is a calculation of the neutronic and power response of the disrupting core. Thus, as shown in Fig. 8, SIMMER consists of two interacting calculational loops: one for fluid dynamics and one for neutronics.

The mesh structure used for the fluid dynamics calculations is also the basis for the neutronics calculations. However, regions where neutronic effects are known to be negligible can be eliminated. Also, specified regions can be further subdivided when needed to obtain realistic representations of neutronic spatial effects.

The neutronics and fluid dynamics equations are not solved for the same time step, but are required to agree at certain times. This approach is what permits the separation of the two loops. The interaction between loops occurs with the transfer of such key quantities as material temperatures, densities, and heat generation rates.

Three levels of sophistication are provided in SIMMER for the neutronics loop. The simplest approach assumes a uniform neutron distribution in space and an invariant energy spectrum during the transient. Reactivity feedbacks are derived from overall reactivity coefficients and, for each material, a reactivity effect per unit mass. This approach is known as "point kinetics" since it treats the reactor as a single point in space. Because of its simplicity, it appears in SIMMER merely as a step in the fluid dynamics loop.

As the core becomes more disorganized, neutronic characteristics change markedly, requiring an approach that incorporates space, time, and spectral effects. In some cases, a diffusion treatment of neutron transport in space can be used that represents an intermediate level of sophistication and has the advantage of numerical economy. In general, however, an approach using the full neutron transport or neutron conservation equation is necessary. This approach includes the dependence of the neutron distribution on space, time, neutron energy, and neutron direction. The solution of this equation is formidable and has received much attention from reactor physicists for a number of years. The techniques used in SIMMER are state of the art.

Severe Accident Analysis

The combination of the features discussed above resulted in a flexible code capable of simulating experiments, performing controlled examinations of isolated phenomena, and analyzing severe accident behavior in a variety of reactor configurations. As a result, an interactive approach involving applications and testing was implemented early in the development of the code with very fruitful results.

The first major application came in 1977 when SIMMER I, the first version, was still in a developmental stage. This application involved an investigation of the hypothetical accident already described in which the core suffers a super-prompt-critical excursion producing a state of high thermal energy (fuel temperatures on the order of 4000 to 6000 kelvin and a maximum vapor pressure of approximately 25 megapascals). As previously noted, the isentropic expansion analysis of this event indicates that the pool of liquid sodium should surge up against the reactor head, generating large pressures and structural deformations, and could thus pose a threat to primary containment.

In 1977 the SIMMER I capabilities were adequate for a more realistic analysis of this expansion process. These capabilities included the modeling of a multitude of heat-transfer processes, structural constraints on fluid flows, and mass-transfer effects, all in a transient context. To everyone's surprise, the calculated mechanical energy was only about 5 per cent of that from the isentropic expansion for identical initial conditions.

The impacts of this finding, if substantiated, are fourfold. First, the damage potential of severe accidents was previously being estimated in an overly conservative fashion. Second, the ability of reactor containment systems to withstand severe accidents and thereby protect the public may be substantially greater than thought. Third, the extent to which the highly complex parts of the severe accidents are resolved could be relaxed by allowing some difficult phenomena to retain large uncertainties. Finally, a phenomenological signature of the expansion was established against which concepts and estimates could be tested.

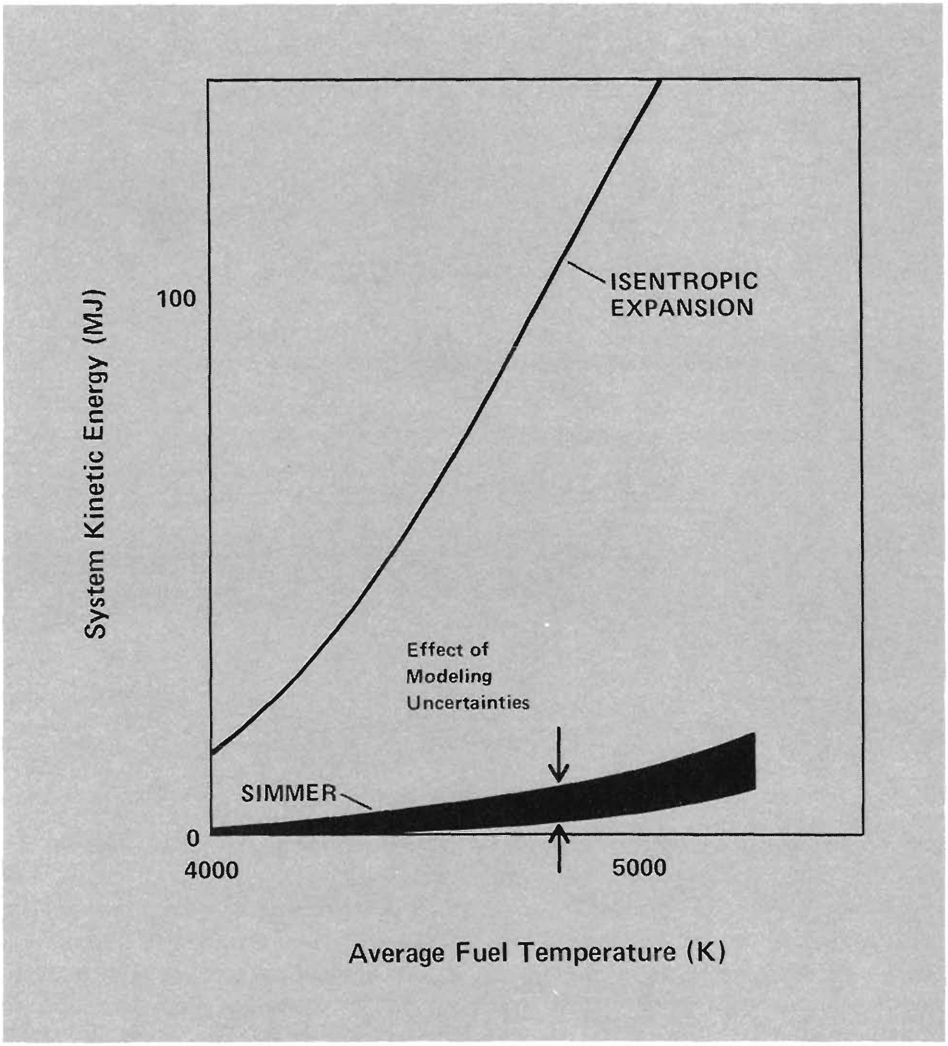


Fig. 9. Reactor damage potential (fluid kinetic energy) versus accident severity (initial fuel temperature) calculated by SIMMER and by assuming an isentropic expansion of the fuel. The band represents the effect of SIMMER modeling uncertainties.

Follow-up investigations with an improved and expanded version of the code, SIMMER II, revealed the reduction in mechanical energy to be due to interactions among the various aspects of the expansion process. For example, the pin structure in the upper core produces a strong throttling effect that prevents rapid discharge of core material into the sodium pool as well as redirecting heat flow so that not all the available energy

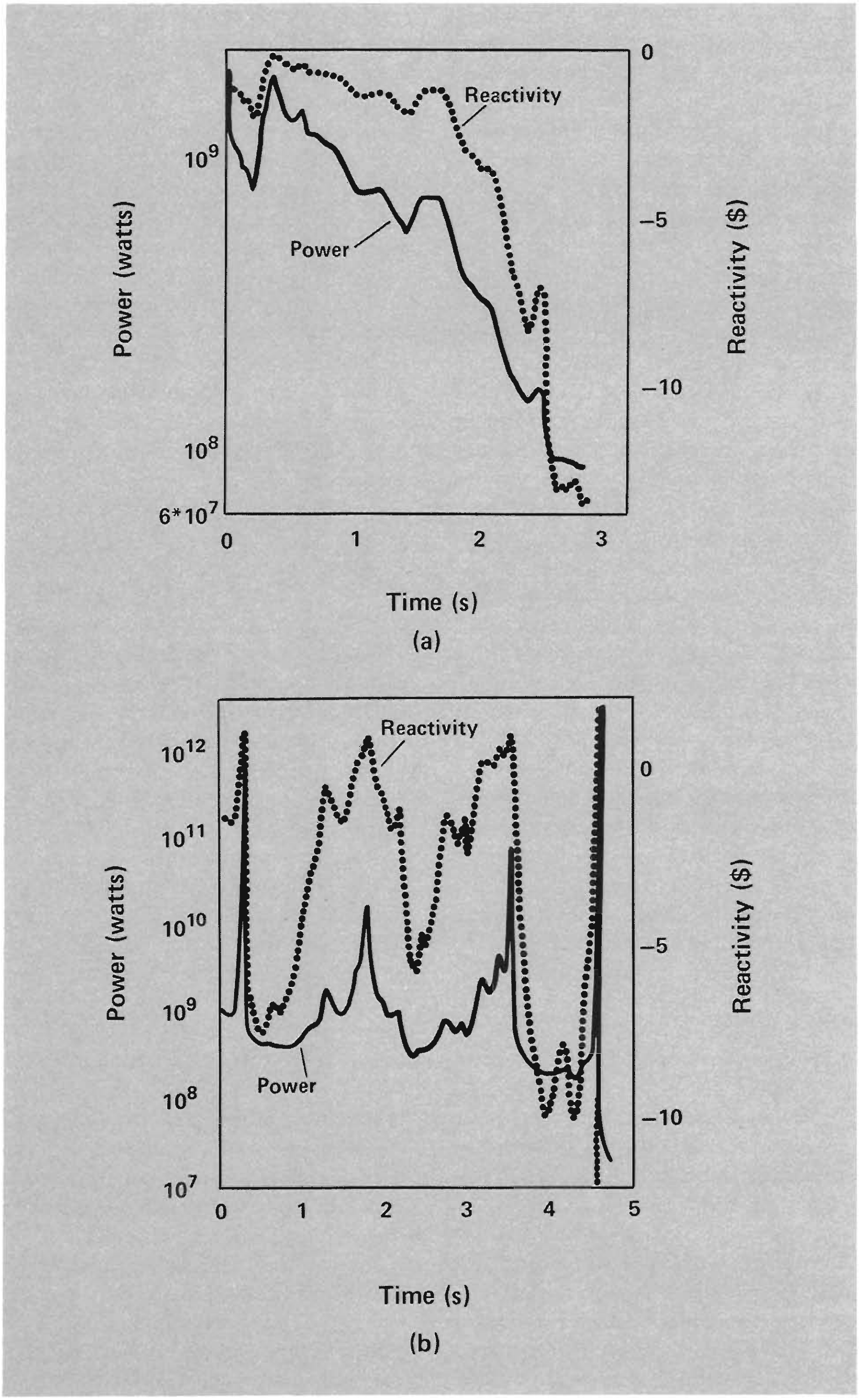


Fig. 10. Two examples of the power and reactivity response calculated by SIMMER for a liquid-metal fast breeder during a hypothetical accident. Initial conditions are identical except in a) large amounts of fuel are able to escape through the axial blanket sections while in b) the channels within the axial blanket sections of the fuel pins are clogged and only slight amounts of fuel escape. (The unit of reactivity is the dollar; one dollar is the reactivity change between delayed-critical and prompt-critical conditions.)

is delivered to the sodium. Nonuniform expansion, resulting from interaction with core structure and from preferential expansion of the hottest fuel, also leads to a considerable drop in the effective expansion pressures.

The investigations also showed that the reduction was highly insensitive to modeling uncertainties. This result is summarized in Fig. 9, where the calculated fluid kinetic energy due to the expansion is plotted against accident severity in terms of the initial fuel temperature at the start of the expansion. The band on the figure represents the combined interactive influence of modeling uncertainties; it was determined by assigning estimated uncertainties to 25 exchange processes important to the accident, randomly selecting values within these uncertainty ranges, and then repeatedly redoing the SIMMER calculations. The result shows that known modeling inadequacies can be tolerated without compromising the gratifying reduction in damage potential.

As a result of this early work, there is presently an international interest in SIMMER and its capabilities. Major laboratories and government agencies in the United Kingdom, West Germany, Italy, and Japan, as well as industrial contractors in the United States, are actively utilizing SIMMER.

The second major application of SIMMER was initiated in 1979. This was the first attempt to simulate an actual core meltdown transient on a whole-core scale with coupled space-time neutronics and comprehensive thermal-fluid dynamics. The results of two calculations are shown in Fig. 10. The initial conditions and modeling assumptions are identical for both calculations except that the result shown in Fig. 10a assumed easy fuel transport through the axial blankets and thereby considerable fuel escape from the active core regions while the result shown in Fig. 10b assumed rapid freezing of the fuel in the escape paths and thereby slight fuel escape. This approach illustrates the method of dealing with modeling uncertainties by varying one key aspect and so bracketing the phenomenon of interest between two extremes.

Comparison of the transient power for these two extremes indicates the crucial effect of fuel removal. When fuel can escape easily, power dwindles to low levels within 3 seconds and at no time does it get much higher than 10^9 watts. However, when fuel escape is clogged, the result is continued power excursions with peaks of 10^{12} watts as late as 5 seconds into the accident.

This second application of SIMMER provided major insights into the characteristics of core disruption from a transient point of view and called into question qualitative views derived from steady-state perceptions. This latter point of view suggested that a few generic physical processes control the accident behavior, such as a material boilup in the core that permanently disperses the fuel and prevents recriticality. The transient context indicated that these processes may not be continuously operative; for example, even after boilup the fuel could flow back together into another critical mass.

Since full-scale, severe accident experiments are not considered desirable or feasible, there is a need to test the predictive capability of SIMMER through application to a wide spectrum of specialized experiments. One example of the various experiments simulated by SIMMER II has to do with the previously mentioned freezing and plugging phenomenon that may occur during a severe accident. Experiments conducted at Argonne National Laboratory examined this phenomenon, using a hot, molten thermite injected into the channel between steel-clad blanket fuel pins. The molten thermite contains molybdenum, which represents melted stainless steel, and uranium oxides, which represent melted fuel. The full experiment simulates the pressurized flow of melted fuel up into the blanket region of the core. The code successfully simulated the degree of transport of thermite into the channels, the plugging phenomenon, and the amount of destruction of the pins for a variety of initial conditions.

Conclusions.

Because of these SIMMER analyses, the necessary conditions for avoiding severe accidents are becoming discernible. As a result, research directions are being reassessed and the discussion of these accidents is turning from broad opinions to factual support of specific models.

The analytical results obtained so far have encouraged us to feel it is possible to unfold the accident path in a way that adequately represents reality. The fact that development of mechanistic computer codes has been supported by various parties in the breeder community for a number of years indicates a strong desire by the industry to also face that reality.

Moreover, since it is not reality, but rather the uncertainty surrounding a vague possibility, that leads to greater fear in the minds of people, it is necessary to continue refining the analytic tools and extending the experimental base for the study of severe accidents in the fast breeder. Only then can informed decisions be made concerning the safety of this potentially very beneficial modern alchemy. ■

AUTHOR

Charles R. Bell has been a staff member at Los Alamos since 1975. He received his Bachelor of Science degree in mechanical engineering from the University of Cincinnati in 1965 and his Ph.D. in nuclear engineering from the Massachusetts Institute of Technology in 1970. From 1970 to 1975 he worked for Atomics International, assessing severe core-disruptive accidents on breeder reactors. Also, he led the effort to design, develop, test, and apply a system analysis capability to investigate tube breaks in large sodium-water steam generators. At Los Alamos he has been working to develop and apply advanced techniques for detailed assessment of severe breeder reactor accidents. He has played a major role in establishing new perspectives on accident characteristics and in integrating these new perspectives into national and international research and development programs. He is a member of the American Nuclear Society.

Further Reading

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The American Breeder Reactor

COMMENT BY JAY BOUDREAU

From its inception the breeder reactor has been described as a self-fueling energy machine, the answer to our energy needs in the coming century. The United States started up the world's first breeder reactor in 1951 and followed with an operational pilot plant in 1963, the 20-megawatt-electric (MWe) Experimental Breeder Reactor II (EBR II). In 1969 we completed the initial design for an intermediate size (300-MWe) breeder reactor to be built at Clinch River, Tennessee, as the major step toward a commercially viable power generation system. In 1971 President Nixon established the liquid-metal fast breeder reactor (LMFBR) as the nation's highest priority research and development effort. Yet today, ten years later, we have slipped from our world preeminence in breeder technology, and the direction and very future of the breeder development program in the United States is now uncertain.

Meanwhile, the French, the British, and the Russians proceeded with their own original plans: the 250-MWe Phenix, the 250-MWe Prototype Fast Reactor (PFR), and the 350-MWe Bystrye Neitrony (BN-350) all came on--line about 1974. The French and the Russians have continued their programs. The Russian 600-MWe Bystrye Neitrony (BN-600) came on-line in 1979; their 1600-MWe Bystrye Neitrony (BN-1600) is scheduled for 1986. The 1200-MWe French Super-Phenix is scheduled for completion in 1984. Today it is France who leads the world in breeder reactor technology.

The slowdown and all-but demise of the American breeder reactor program have resulted partly from uranium fuel costs, partly from breeder reactor development costs, and partly from the politics of nonproliferation of nuclear weapons. Just the cost of producing plutonium fuel from a breeder is high enough to keep utility companies in the United States from being seriously interested in the breeder at this time. For example, the current market price of mined, processed uranium fuel (yellowcake), which is $25 per pound, would have to increase to nearly $165 per pound in today's dollars before the breeder would be financially competitive with the light-water reactor. cost equivalence might take seventy-five to one hundred years unless crises arise in fossil fuels and imported oil. Consequently, today there is little market pressure to maintain the impetus of the breeder programs begun in the Nixon era.

The cost of development and construction is another problem in the breeder reactor program. Currently, the capital cost of the breeder is significantly higher than that of the light-water reactor. Constructing a 1000-MWe light-water reactor would cost about $1.7 billion, while a fast breeder reactor system of comparable power could cost $3.4 billion. Because of the excessive cost, United States utility companies are reluctant to undertake the purchase of a breeder system without government subsidies. In France, where there is a shortage of domestic energy resources, breeder construction is subsidized as a matter of government policy. However, in this country opposition to such a policy has come from both political parties. For example, in a 1977 letter to the House of Representatives, Michigan Congressman David Stockman, who is now President Reagan's Budget Director, denounced the Clinch River project as "totally incompatible with our free market approach to energy policy."

The politics of nuclear weapons proliferation is still another issue in the breeder program. Because of grave international concern about proliferation, President Carter in April 1977 called for indefinite deferral of construction of commercial breeder reactors. Breeders are considered hazardous because they produce rather large quantities of plutonium, the basic material of nuclear weapons. However, separation of plutonium from breeder reactor fuel requires sophisticated reprocessing, and fabrication of nuclear weapons requires still further technology. The International Fuel Cycle Evaluation report published in 1979 found that the breeder fuel cycle poses no greater threat of international weapons proliferation than does the light-water reactor system. In 1981 an Iraqi reactor was destroyed by Israel in order to stop development of nuclear weapons in Iraq; that reactor was not a breeder but a light-water type. In the United States the proliferation-inspired moratorium has locked up our breeder program for four years; now, under a new administration, we are just beginning to take another look at the program. Whether the breeder development should simply pick up where it left off is open to question.

On the technical side the case for completing the 350-MWe Clinch River breeder reactor as a demonstration, power-producing facility is debatable. The reason for building commercial plants of gradually increasing size is to learn about scaling. It is not possible, for example, to ensure successful construction and operation of a 1000-MWe

Program Gets a Second Chance

COMMENT

breeder facility by extrapolating from our experience with the 20-MWe Experimental Breeder Reactor II. Thus, Clinch River was to be a steppingstone,

On the other hand, even during the Carter deferral of commercialization, funding for building and testing Clinch River's large components continued. The Department of Energy constructed large facilities specifically designed to test pumps, heat exchangers, and other parts. By now the components have been largely tested, and most of what will be learned from building Clinch River will be how the components behave together in an operating plant. While this is valuable knowledge, its direct applicability is diminished by the fact that present plans for the next generation facility, familiarly known as Son of Clinch, do not retain the same design features. Therefore, from a developmental point of view, the Clinch River project would increase in value if its design were modified to include these new features.

An even more important step toward commercialization would be construction of a larger, more advanced developmental plant. The concept for this larger plant is another result of the moratorium on breeder commercialization. When President Carter stopped the Clinch River project in 1977, he instituted a four-year conceptual design study to evaluate a variety of breeder designs that might minimize the threat of proliferation and enhance overall breeder effectiveness. The report, just issued to Congress last March, outlines a modern, streamlined, large developmental plant with state-of-the-art features not available when the Clinch River breeder was designed. This large developmental plant has more advanced pumps, heat exchangers, steam generators, and cooling loops. Its design is close to what we now envision for commercial plants. However, technical considerations alone do not dictate the timing for such a plant.

There are two questions. How can we finance the large developmental plant? And how should its construction tie in with the now revived Clinch River project? The large developmental plant is in the 1000-MWe range and, as mentioned before, would cost \$3.4 billion in 1981 dollars. One option would be to begin the large plant at some time in the future when expenditures for Clinch River have declined, But if too much time elapses, companies will not be able to afford to keep the existing cadre of experienced designers and reactor manufacturers on the payroll. The team will disband, and the price this country will pay is the lead time necessary to reassemble the team. No one really knows how many years this would take or what it would cost.

So far, \$1 billion in tax dollars has been spent on Clinch River; the total cost estimate in current dollars is \$3.0 to \$3.2 billion. The original cost estimate was \$700 million. An associated reprocessing facility is also planned, and its development could cost as much as another \$1 billion. To build the large plant concurrently would require an innovative financing scheme. One suggestion is a Congressionally chartered corporation composed of personnel from government, from national laboratories, and from industries and utilities. The corporation would be empowered to enter the private capital market to seek funding. The government could provide a loan guarantee and could possibly pay the interest on the loan. The term of the loan would commence during plant construction and would terminate when the loan had been repaid from revenue generated by power sales. Revenue from continued operation would be used to repay Treasury for its contribution of interest monies. Such a scheme could reduce to \$800 million the government investment in the large plant.

Two decades ago the breeder reactor was an experimental technology with great promise for solving future energy problems. The United States was a world leader in that technology. Now, when energy is a very immediate problem for most of the world, breeder reactors in France and in the Soviet Union are beginning to fulfill their promise. But we are still in the developmental stage; we have lost our sense of urgency about breeders; and our entire nuclear industry is just beginning to recover from the aftereffects of the Three Mile Island accident.

Our country's energy future is not at all secure; another series of crises over imported oil and new demands and higher prices for uranium could make the breeder reactor very attractive thirty years from now. From our own experience and from watching the European efforts, we know that development and commercial plant production take twenty years or more. We also know that the costs of development and construction are rising rapidly. Much of the preliminary testing of breeder reactor components is done. Now the moratorium is over. It seems a good time to go forward either with a revised intermediate project at Clinch River or with a new large developmental plant—or with both. ■


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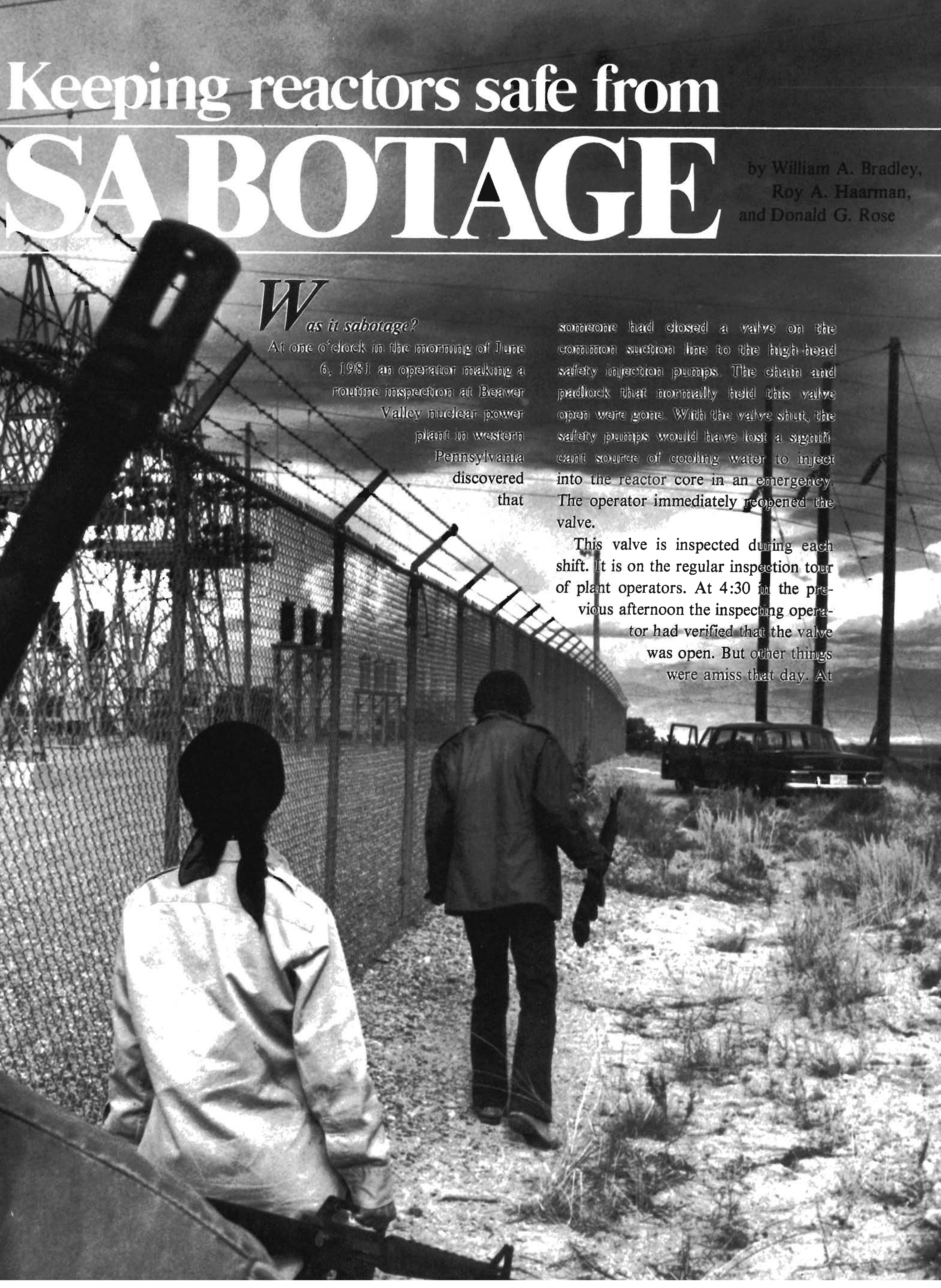
Keeping reactors safe from SABOTAGE

by William A. Bradley,
Roy A. Haarman,
and Donald G. Rose



Was it sabotage?
At one o'clock in the morning of June 6, 1981 an operator making a routine inspection at Beaver Valley nuclear power plant in western Pennsylvania discovered that someone had closed a valve on the common suction line to the high-head safety injection pumps. The chain and padlock that normally held this valve open were gone. With the valve shut, the safety pumps would have lost a significant source of cooling water to inject into the reactor core in an emergency. The operator immediately reopened the valve.

This valve is inspected during each shift. It is on the regular inspection tour of plant operators. At 4:30 in the previous afternoon the inspecting operator had verified that the valve was open. But other things were amiss that day. At nine o'clock in the morning operators found that locks and chains had been removed from other valves on three auxiliary feedwater pumps. The valves, however, were all in the normal, open position. But neither these locks and chains nor the ones for the suction-line valve could be found.

Duquesne Light Company, licensee of the Beaver Valley nuclear plant, immediately isolated the plant's vital areas and stepped up security. Operators began checking key equipment every two to four hours. And the Pittsburgh office of the Federal Bureau of Investigation began looking for the culprit.

The valve incident at Beaver Valley is over. Whatever the actual cause was, there was no effect on the plant. The inspection system functioned as intended. The power plant continues to operate. But this successful detection of tampering and protection of plant vital areas has significance far beyond Beaver Valley.

Concern for Security

Protecting American nuclear power plants from internal sabotage and external attack has long been a major concern of the United States Nuclear Regulatory Commission. Studies performed in the early seventies indicated that nuclear power plants were not attractive targets for terrorism and that their construction was highly resistant to damage, yet there were conditions under which the radioactive containment features could be sabotaged. This conclusion prompted the Nuclear Regulatory Commission, in February 1977, to publish a revised section to the Code of Federal Regulations, Title 10 Part 73.55.

The new requirements were aimed specifically at countering any form of sabotage that could release radioactive material and thereby create a hazard for the general public. But implementation of the new law required reviewing and upgrading the security plans for more than 70 nuclear power plants each with unique nuclear and secondary systems and unique geographic and demographic environments. (There is no "standard nuclear plant" in the United States. Although a single manufacturer may provide the basic reactor system for a group of plants, the remainder of each plant is a composite provided by various contractors.)

The Nuclear Regulatory Commission recognized the complexity of the security review project from the very beginning and the Commission called upon Los Alamos for engineering support even before final adoption of the new regulations. Eight teams were formed to analyze individual plans for physical security. Each team had one Los Alamos engineer for mechanical systems and one for electrical systems and two Nuclear Regulatory Commission personnel. Over a period of 18 months, beginning in February 1977, these teams visited every operating commercial power reactor in the United States at least once and many several times. What these teams learned from site visits and from security plans provided by the licensees was analyzed to determine how well each plant fulfilled the requirements of the new security rule. When deficiencies were found, the licensees were required to correct them.

Early in 1978 the Nuclear Regulatory Commission organized three additional teams of two Los Alamos engineers each, with support from Science and Engineering Associates of Albuquerque, to pinpoint potential sabotage targets at all nuclear power plants in the country and thus identify exactly what was the vital equipment that needed to be protected. These teams have visited 50 of the 70 nuclear reactors in the U.S. and their work is still underway.

Altogether the review process has had a profound effect upon the planning for security at nuclear power plants, especially in defining what we are trying to protect, what kinds of threats we face, and how we can realize the largest return for our investment in nuclear plant security. The review process also has implications for nuclear plant safety.

Protecting Property or People?

Before designing a physical security plan, two basic questions need to be answered. First, what is to be protected? The Nuclear Regulatory Commission answered this very simply: in this case, protection is not for the power plant but for the health and safety of the public. The purpose is to prevent "radiological sabotage." Radiological sabotage is defined in terms of a maximum radiation level established in the Federal regulations for siting nuclear power plants. It is any deliberate act that causes a radiation release sufficient to provide a dose of more than 300 rem to the thyroid or 25 rem to the whole body of a person who remains at the edge of the plant exclusion area for 2 hours after the release.

The decision to protect against a radioactive material release rather than to protect the entire power plant is

Keeping reactors safe from SABOTAGE

conceptually important because it allows the plant area and the analysis of physical security to be divided into two parts. The large area containing all components of the nuclear power plant is commonly called the protected area, and it is at the boundary of this area where physical security measures start. An intruder may get into the protected area and inflict damage to plant systems that interrupts normal operation, yet his actions here do not cause a radioactive release. The security analysis of the protected area concerns mainly the response of a guard force to a detected intrusion.

Within the general protected area are specific areas that are vital to radiological security; disabling equipment or systems in these vital areas could either directly cause a radiological release or prevent mitigation of a threatened release caused by damage elsewhere. Typical vital equipment includes the reactor containment, the main reactor controls, and the pumps, piping, and valves essential for reactor cooling. Analysis of the plant involves identifying vital equipment, pinpointing the actual location of that equipment at the plant site, and predicting the response of the reactor to sabotage of that equipment.

A second question is equally important to the design of a physical security plan. What is the threat? The answer to this question is not easy. Real sabotage threats might range in size from a single person to a large paramilitary force. Motivations might include the illusions of the individual terrorist as well as the grand mission of an antinuclear movement. Methods might include direct external attack as well as covert operations by persons inside the plant. Before 1974, the postulated external threat to a nuclear plant was generally considered to be of the lone bank-robber type. However, because of the growing concern about terrorists, the regulations issued in February 1977 by the Nuclear Regulatory Commission defined the design-basis threat to nuclear power plants to be a small group of dedicated, well-equipped, and well-trained attackers with or without assistance from a person inside the plant.

The Commission's definition of the threat put plant security in another light. Ordinarily, nuclear power plants would seem to be very difficult sabotage targets. The plant components and structures are large and strong and have many redundant control, safety, and shutdown systems. Redundancy in the design comes from the "single-failure" concept; under this concept we assume that accidental single failures may occur in any component in a system and, therefore, we must have backup components. However, we now realize that a well-trained, knowledgeable team of terrorists could circumvent this inherent safety feature by deliberately causing multiple failures in a selected system. Such a postulated threat, of course, introduces further complexity into the system analysis. But it is this same kind of common-mode failure where a single event precipitates a simultaneous multiple failure of some key system that has been highlighted by the Three Mile Island accident.

Requirements for the Protected Area

Postulate a team of saboteurs trying to enter a power plant's protected area and reach a vital area. How are they detected? How do the guards know whether the alarm is real? Where is this team going and how strong are they? How should the guard force be deployed to intercept them? What type of armaments will best counter this threat? Which would be more effective in delaying this threat until the police arrive—stronger doors at vital areas or a larger guard force?

As these questions illustrate, a nuclear power plant security system has many elements: physical barriers, detection devices, alarm systems, communication systems, guard training, guard force levels, and armaments. The engineering teams found that the combination of security elements and their interactions were unique to each plant. The main task for the Los Alamos engineers was to assist personnel from the Nuclear Regulatory Commission in comparing and evaluating the technical aspects of each plant's security system with the Commission's published requirements for security.

Then, since all the individual components of a physical security system must function together, the teams postulated intrusion scenarios in the protected areas to see if the plant guard force could respond in time to prevent the saboteurs from gaining access to a vital area.

Here are examples of some of the security elements and interrelationships that needed to be considered by the Nuclear Regulatory Commission and the engineering teams. For the simulated attack shown in Fig. 1, at point A the attackers breach the protected area barrier, usually an 8-foot cyclone fence topped with three strands of barbed wire. How fast can they do this? The times needed to breach many types of barriers with a variety of mechanical and explosive tools have been determined by repeated experiments at Sandia. In this case a cyclone fence is not a very effective barrier and can be climbed or penetrated in seconds. Even though vibration sensors or other detection systems may be on the fence, its major purpose is simply to define and limit the boundary of the protected area.

In this example, more effective protective elements are just inside the fence. Here, sighting along a level area kept clear of herbage, is an intrusion-detection device, perhaps a microwave system combined with electric-field, infrared, or seismic detectors. When the attackers breach the protected area, this system signals two alarm stations with visual and audible alarms.

The alarm signals the guards, but is the penetration real? Also sighting along this cleared area are a number of closed-circuit television cameras. A view of the penetrated section of the fence is displayed automatically so guards can determine whether the alarm is real. If so, the station guards call out the response force and initiate other necessary actions, such as notifying outside law-enforcement agencies.

Several questions are addressed in this part of the security review. Is this area lighted well enough? Should a closed-circuit camera be placed to view this door? Are the guard patrols frequent and random enough in this area to keep the probability low that the attackers will reach point B while the guards are at other distant locations in the plant site? How long will it take a well-equipped team to penetrate the barrier at B?

Once inside the building, the attackers attempt to move to point C, breach this door and reach the vital component inside. Should the barrier at C be strengthened? What is the measure of the reliability of the guard force communications system used to cope with this situation? Does guard force training permit an efficient, coordinated attempt to prevent the saboteurs from reaching their objective?

One suggested analytic method for answering these questions was a computer code developed by Sandia National Laboratory for the Nuclear Regulatory Commission; it is called EASI (for estimate of adversary sequence interruption). In this code the properties of the protective systems, such as the efficiency of the intrusion-detection system, the reliability of communications, and the time for the guard force to respond, are balanced against the time it takes the attacker to penetrate the various barriers and perform his sabotage. The comparision produces an estimate of the probability that the response force can intercept the attackers before they can do their mischief.

Although the basic EASI calculations are relatively simple, the large number of different elements makes the task ideal for computer analysis. By manipulating the variables of attack and response, the teams could evaluate tradeoffs and determine which would give the greatest protection for the money invested. One version of the code runs on hand-held

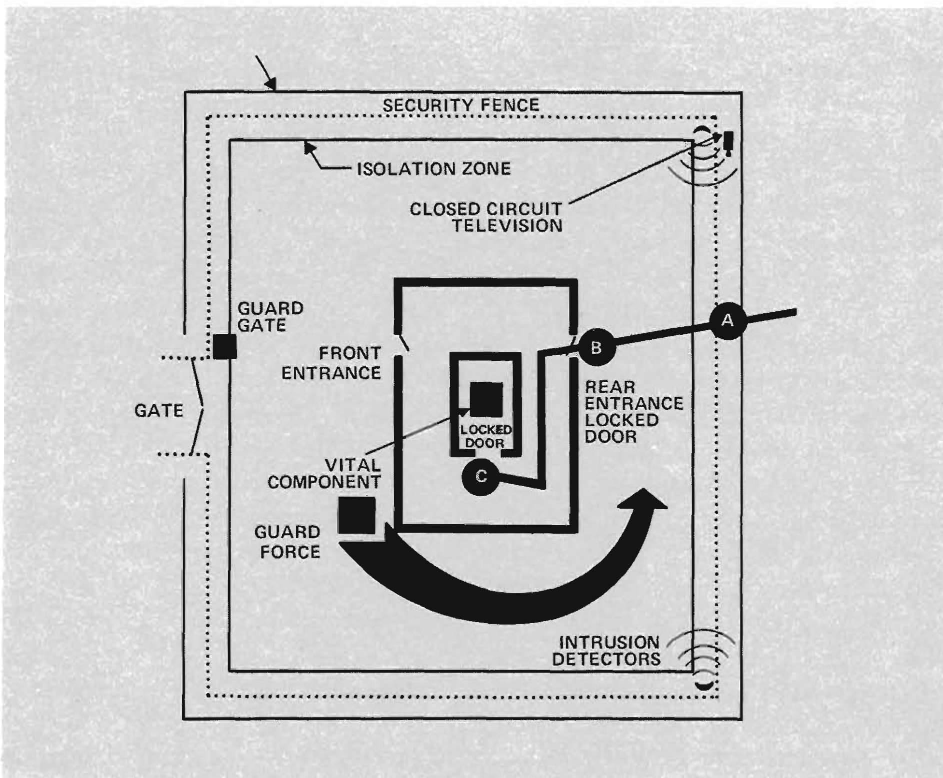


Fig. 1. Action sequence for a hypothetical sabotage attack.

computers; thus teams could evaluate facilities in the field and licensees could analyze their own plants.

The data needed for the EASI analysis—times for barrier penetration, distance traversed, guard response, and the reliability of communication and detection systems—cannot have exact values because they all have statistical fluctuations. Thus, the method can only provide a percentage estimate of guard success in interrupting hypothetical attackers. However, the method is ideal for evaluating the relative worth of several protective systems or the proposed improvements for a given system.

This type of analysis is illustrated in Fig. 2, a three-dimensional plot used to analyze one aspect of a protective system: the interruption probability versus the guard-force response time and the time to breach door B of Fig. 1. Point I toward the lower front corner represents a long guard response time (12 minutes), a short time for the saboteurs to breach door B (4 minutes), and thus, a low probability of interruption (5%). If this plot represented an actual data point for a plant, a Nuclear Regulatory Commission reviewer would note a physical security problem. The plant owner, looking at this same plot, could correct the defect by either shortening guard response time or increasing barrier strength at point B. In this particular case, he might decide that it would be more cost effective to strengthen the door and raise the breach time to 16 minutes (point II), thereby increasing the probability of interruption from 5% to nearly 90%. Whatever modification the owner makes, the Commission reviewer will be satisfied when the probability of interruption is high enough.

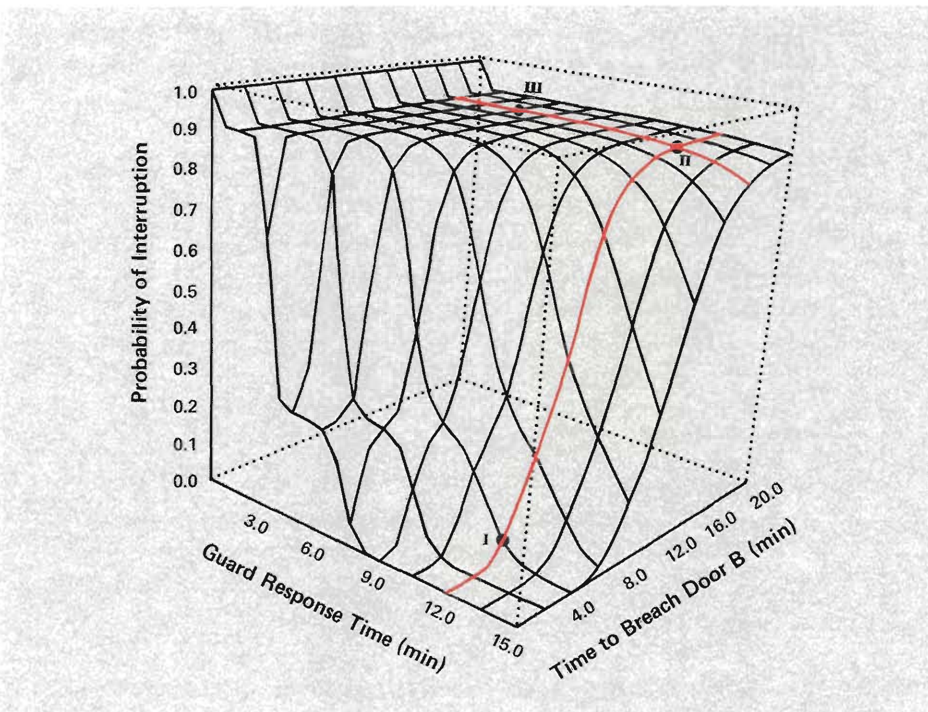


Fig. 2. Probability of interrupting hypothetical saboteurs as a function of guard response time and the time required to breach a locked rear door, as calculated with EASI.

Intrusion games can be played many times for each plant and the interruption probability can be plotted as a function of virtually any variable. Such analyses have allowed numerical assessment of complicated physical security problems. The three-dimensional aspect of these EASI plots is especially helpful in revealing either steep or flat regions on the probability surface. A steep region will cause dramatic increases in the probability of interruption for small improvements (such as from point I to point II), whereas a flat region (such as from point II to point III) shows where further improvements may not be cost effective.

One important aspect of physical plant security not directly covered in our scenario is protection of the plant from the inside man, a plant employee in any position of responsibility or even a visitor. Three protection methods have been suggested to plant owners: limit access to vital areas; prevent anyone from being in a vital area alone (the two-man rule); and allow only cleared persons into vital areas. An example of the two-man rule as protection from an inside saboteur is the use of two alarm stations; since the stations have identical alarms and controls, the guard in either station can monitor the other.

A number of other measures protect against a potential saboteur, who may be either a visitor or an employee. Access to the protected area is through a single gate where all persons are identified and checked for contraband. The last door from the entry guard station to the protected area can only be opened by a guard behind a rifle-proof barrier, and another guard observes this operation to prevent an inside man from letting a collaborator in. Similar precautions are followed for entry of vehicles. In fact, all packages in delivery vehicles must be identified, the shipment administratively verified, and the packages off-loaded at a special receiving area near the perimeter of the protected area.

Defining Vital Areas

Suppose a team of saboteurs gains entrance to the plant despite the protective measures. Or suppose a saboteur is already in the plant as an insider. Which components would the saboteurs go after? Where are they located? If the sabotage attempt succeeds, will the crippled reactor release a significant amount of radioactive material? To answer these kinds of questions, the engineering teams had to start by locating potential targets, the plant's vital equipment. To assure complete protection, all vital equipment must be so designated. However, the designation of noncritical areas as vital would add unnecessarily to plant costs and the burden of the plant security force. Such unnecessary designations could also add to safety problems.

The Nuclear Regulatory Commission has defined two levels of vital areas. A Type 1 vital area is a single location where a saboteur could cause successful radiological sabotage (for example, the nuclear reactor containment building). A Type 2 vital area contains equipment insufficient in itself to achieve a suc-

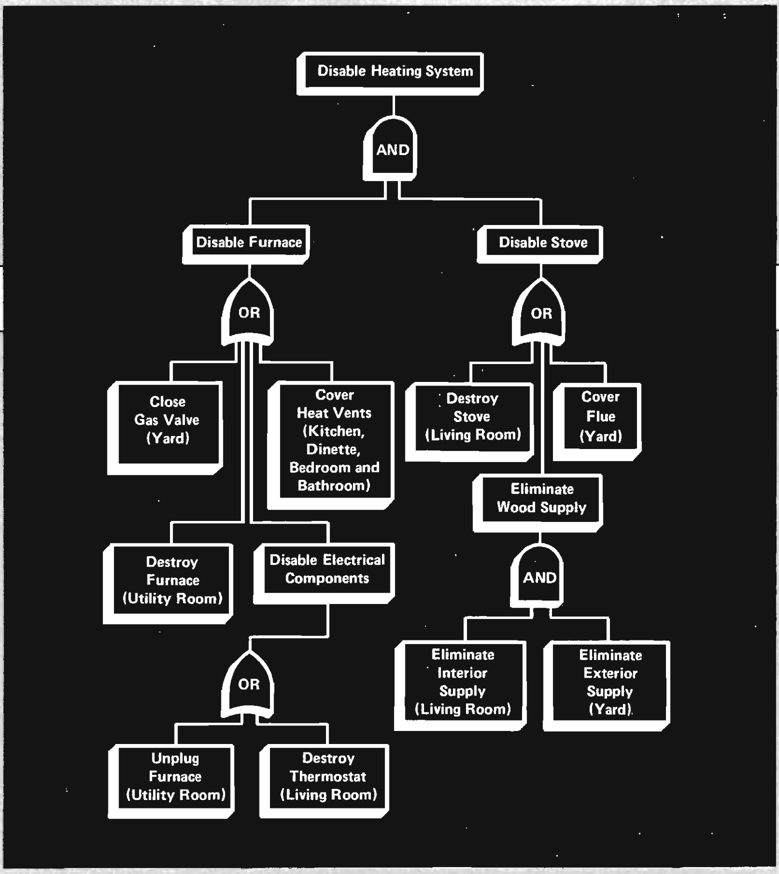
Sidebar: A FAULT TREE FOR HOUSEHOLD SABOTAGE

Consider an imaginary saboteur intent on disabling the heating system of a certain residence. First, of course, she gathers information about the system's components and learns that the house is equipped with a forced-air gas furnace in the utility room, a main gas valve in the yard, a thermostat in the living room, heat vents in the kitchen, dinette, bedroom, and bathroom, a wood-burning stove in the living room, and wood supplies in the living room and yard.

Because this particular saboteur has a rather analytic mind, she uses the following method to select a course of action. First, she draws a fault tree to show the possible paths to the goal. She uses the "and" symbol () to indicate actions all of which are required to produce the desired effect and the "or" symbol () to indicate actions each of which is sufficient in itself. Then, she ponders—analyzes—this fault tree and compiles a list of the various location scenarios, as she calls them, at which actions must occur to accomplish the crime. She also lists the various event scenarios, or necessary actions, associated with each location scenario.

The saboteur may now select a location scenario that seems most advantageous. Being sensible, she rejects those location scenarios requiring her presence in all or nearly all rooms of the house. A decision among the other possibilities will be made on the basis of her personal tastes and abilities.

Turning the tables on our imaginary saboteur, scientists at the Laboratory have applied this technique to one aspect of foiling sabotage at nuclear power plants—identification of "vital areas," those places or combinations of places at which radiological sabotage could be accomplished. Based on site-specific information, a fault tree for a particular plant is developed and analyzed with a computer program developed at Sandia National Laboratories. The program rejects those location scenarios requiring actions in an excessive number of places and provides a list of more credible location scenarios and associated event scenarios. These location scenarios may then be classified by the Nuclear Regulatory Commission as vital areas requiring implementation of various security measures. ■



LOCATION SCENARIO	EVENT SCENARIO
Yard	Close gas valve and cover flue
Living room	Destroy thermostat and destroy stove
Utility room and living room	Destroy furnace and destroy stove Unplug furnace and destroy stove
Utility room and yard	Destroy furnace and cover flue Unplug furnace and cover flue
Living room and yard	Destroy thermostat and cover flue Destroy thermostat and eliminate interior and exterior wood supplies Destroy stove and close gas valve Eliminate interior and exterior wood supplies and close gas valve
Utility room, living room, and yard	Destroy furnace and eliminate interior and exterior wood supplies Unplug furnace and eliminate interior and exterior wood supplies
Kitchen, dinette, bedroom, bathroom, and living room	Cover heat vents and destroy stove
Kitchen, dinette, bedroom, bathroom, and yard	Cover heat vents and cover flue
Kitchen, dinette, bedroom, bathroom, living room, and yard	Cover heat vents and eliminate interior and exterior wood supplies

cessful sabotage; a saboteur would have to commit destructive acts in more than one Type 2 area to cause a radioactive release.

But how can we determine what areas should be designated Type I vital areas? Before the vital area reviews, the auxiliary feedwater system in a pressurized-water reactor was considered critical enough to designate the locations of its components as Type I vital areas. However, some utility operators questioned this designation. They maintained that even with the loss of the auxiliary feedwater system, the reactor could still be cooled using the high-pressure injection pumps. Team engineers analyzed the problem using TRAC, the thermal-hydraulic computer code developed at Los Alamos. TRAC is discussed at length in the article "Accident Simulation With TRAC." Results of the study verified the operators' position; proper use of the high-pressure injection system could control the particular Babcock & Wilcox reactor studied and, thus, the locations of the auxiliary feedwater system were not necessarily Type I areas.

Since each American nuclear plant has a unique design, the engineering teams had to analyze each plant separately to locate its vital equipment. The key element in this analysis was another Sandia computer code* that acts as a "bookkeeper." Using this code for keeping his records, the engineer can develop and solve fault trees in applications involving a large number of event paths. In this application the scenarios available to a saboteur involved thousands of possi-

*R. B. Worrell, "Set Equation Transformation System (SETS)," Sandia Laboratories report SLA-73-0028A (July 1973).

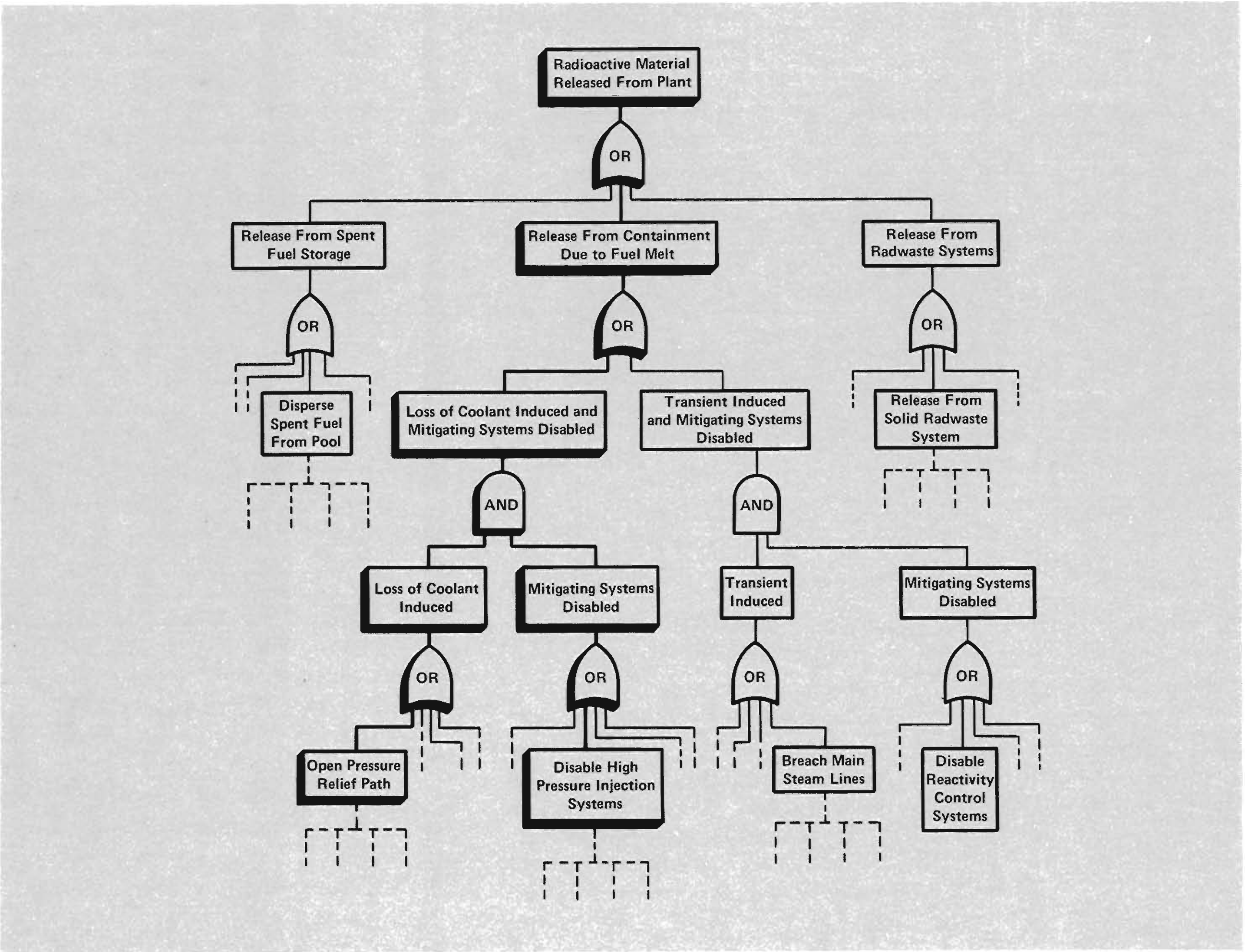


Fig. 3. Portion of a hypothetical sabotage fault tree for a light-water reactor.

ble event paths. When printed out in full, such a fault tree can be over thirty feet long. What we see here (Fig. 3) is a small part of a generic sabotage fault tree for a light-water reactor.

The engineers' first step in the analytic process was to review each power plant's Final Safety Analysis Report to familiarize themselves with various plant details. The next step was to visit the plant to discuss operating procedures with plant engineers and operators. The purpose of these discussions was to gain insight into the ways a saboteur could initiate a radiological event and then disable the safety systems that could control or mitigate that event.

These visits focused on all loss-of-coolant possibilities and included examination of all water systems connecting to the reactor primary coolant system. Systems mitigating against this type of sabotage-induced event include the emergency core-cooling system, reactivity-control systems, and post-accident heat-removal systems. The engineers also reviewed sabotage scenarios that lead to transient incidents such as loss of off-site power or breaching of the main steam lines, and identified the reactivity-

Keeping reactors safe from SABOTAGE

control and heat-removal systems necessary to control the transients.

In one hypothetical sequence, the saboteur opened the valve on the pressurizer in an attempt to induce a loss-of-coolant accident. This initiating event is represented in our fault tree analysis (Fig. 3) by the box in the lower left corner labelled "open pressure relief path." However, the reactor could still be controlled by the high-pressure injection system; in other words, this mitigating system would also have to be disabled if the sabotage is to be successful. Thus our sample fault tree leads from the appropriate two lower left boxes upward to an "and" gate. This means that both events must happen before the core uncovers and the threat of a fuel melt becomes real.

Including the reactor containment, there are three general areas where enough radioactive material might be found to cause a serious release; the other two are the spent-fuel storage pool and the radioactive waste treatment systems. Generally, the storage pool would be a significant source of radioactive material for some length of time after spent fuel assemblies were placed in it. The actual number of days this pool would be a threat depends on reactor core size, the stored fuel's power history, site meteorology, and the type of pool building; this length of time was calculated for each plant.

The study did not overlook theft of fissionable materials as another form of possible sabotage, but such theft was considered unprofitable on two accounts. First, the nuclear fuel used in light-water nuclear power plants is of such low enrichment that it cannot be used directly to construct nuclear explosives. Further, once the reactor is operating, the fuel is highly radioactive and cannot be handled without special equipment. A person attempting the theft of this fuel would quite likely receive a lethal dose of radiation. The liquid, gas, or solid radioactive waste contained in the waste treatment system also was considered in the analysis, but usually there would not be enough material in the system for it to be of real sabotage concern.

The major source of concern and potential for radiological release is in the reactor itself. If the saboteur can cause the fuel to melt significantly and cause the containment boundaries (fuel cladding, primary containment system, and containment building) to be breached or circumvented, then he can achieve successful sabotage. A direct breaching of the containment structure would be a difficult task because the walls are typically 4 to 5 feet of steel-reinforced concrete; however, there are other approaches the saboteur could envision to cause a radiological release that would be less difficult than breaching the reactor containment building.

Benefits of the Study

This review of plant security in the American nuclear power industry has given the Nuclear Regulatory Commission a sound, analytic basis for implementing its new security regulations. The interactions between systems were discussed with plant engineers and operators and verified by reference to the safety reports, emergency operating procedures, and various analyses done by equipment vendors, national laboratories, and the Regulatory Commission.

The reviewing process gave plant operators an insight into the analytic techniques used by Los Alamos team members and an appreciation for the value of these techniques. Many of the licensees were skeptical about the credibility of outside inspection teams until they saw that the analyses were simplifying rather than complicating their security operations.

Beyond the problems of plant security, the study has shown the potential of using TRAC to identify safety problems not detected by conventional safety analyses. The scenario that parallelled the Three Mile Island accident (see accompaying note "A Strange Coincidence") could as well have been undertaken in a safety analysis instead of the security analysis. The computer code does not distinguish between the loss of a nuclear plant component from sabotage and the loss of that same component from an accident. The value of this tool has been demonstrated and it is now available to the Nuclear Regulatory Commission for both security and safety evaluations.

Finally, in its role as an energy research laboratory, Los Alamos National Laboratory has also benefited from participation in this program to identify vital areas and to assist the Nuclear Regulatory Commission in implementing security regulations. Los Alamos engineers are gaining component-level familiarity with all nuclear power plants in the United States. Discussions of study results with plant engineers have helped in validating and refining analytic techniques. And the overall effort has demonstrated another application of the Laboratory's technological capabilities. ■

Sidebar: **A STRANGE COINCIDENCE**

During the vital areas study, Los Alamos nuclear engineers performed a series of thermal-hydraulic transient analyses to determine the effect of two sabotage scenarios involving loss of the steam-generating function in a nuclear power plant.* A computer-based event tree had flagged the auxiliary feedwater system as Type I vital equipment because its destruction in conjunction with certain other acts might cause dangerous overheating of the reactor core. Some utilities questioned this designation. They maintained that even if the steam generator were out, water pumped into the nuclear core by the high-pressure injection system and then released as steam through the safety valves would remove the decay heat. But no one had made the mechanistic calculation that would prove or disprove the feasibility of this feed-and-bleed cooling. Hence the Los Alamos study. The results indeed supported the views of the utility operators concerning certain reactors systems; the high-pressure water injection system could take over and thus the auxiliary feedwater system would not be Type I equipment. By a strange coincidence, the scenarios also foreshadowed many of the key events of the Three Mile Island accident.

One scenario in this computer study postulated a loss of all ac power, which resulted in a number of events including the sudden shutdown of the turbines and the reactor and a loss of the steam generator's heat-withdrawing properties. These events were duplicated at Three Mile Island by the initial accident sequence. The scenario assumed that the relief valve on the pressurizer was opened. This was the valve that accidently stuck open during the Three Mile Island accident. The study then examined how the reactor would behave if no auxiliary feedwater were available and the high-pressure injection pumps were not turned on for various time periods. The operators at Three Mile Island, believing their pressurizer vessel to be filling completely with water, or "going solid," sharply reduced flow from the high-pressure injection pumps. A solid pressurizer would indicate too much water in the primary coolant system and risk loss of pressure control. In actuality the open valve acted as a leak (small loss-of-coolant accident) and the primary system was losing coolant. So the actions taken by the operators to counter the apparent solid pressurizer (cutting back on high-pressure injection) actually aggravated a relatively minor loss-of-coolant situation. This led to the creation of voids in the primary system and ultimately to the uncovering of the reactor core. This was the major cause of the reactor fuel damage at Three Mile Island. The misinterpretation by the operators about what was actually happening to their reactor hinged on the phenomenon of a solid, liquid-filled pressurizer coincident with a reactor core that was being uncovered. The response predicted by the Los Alamos computer analysis included the formation of a steam bubble in the reactor core that increased in size and uncovered the core centerline in 23 minutes. The close parallel between the hypothetical sabotage and the real accident demonstrates vividly the importance of detailed, computer-aided analysis in the evaluation of both the security and the safety of nuclear power plants. ■

**J. W. Bolstad and R. A. Haarman, "Summary of Thermal-Hydraulic Calculations for a Pressurized Water Reactor," Los Alamos Scientific Laboratory report LA-8361-MS (May 1980).*

Acknowledgments

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Commendation by Nuclear Regulatory Commission

The following members of the Los Alamos technical staff assisted the Nuclear Regulatory Commission in the security analysis of nuclear power plants: Richard J. Bohl, William A. Bradley, Donald F. Cameron, Walter S. Chamberlin, Eddie R. Claiborne, Ronald L. Cubitt, Richard D. Foster, Paul M. Giles, Roy A. Haarman, Walter D. Hatch, James O. Johnson, Jerry J. Koelling, Richard W. Leep, Charges A. Linder, Joseph W. Neudecker, Jr., Alden T. Oyer, John L. Rand, Donald G. Rose, and Dean H. Whitaker. The Commission praised the men for "their dedication in spending considerable time away from Los Alamos to visit nuclear power facility sites and in devoting greater than normal effort in completing review assignments in a timely manner. . ."

Further Reading

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Authors Donald G. Rose (left), Roy A. Haarman, and William A. Bradley.

THE STRUCTURAL INTEGRITY OF REACTORS

by Charles A. Anderson and Joel G. Bennett

During the TMI accident, the last line of defense against a major release of radioactive fission products was the reactor containment building. This barrier functioned as designed and survived a 2-bar pressure spike from a sudden hydrogen burn inside the building. But how large a pressure spike was possible before the containment would have failed, releasing radioactive material? Was there, in fact, a margin of safety beyond the approximately 4-bar design limit?

These questions emphasize the fact that while much attention is being focused on the role of the reactor core during a nuclear

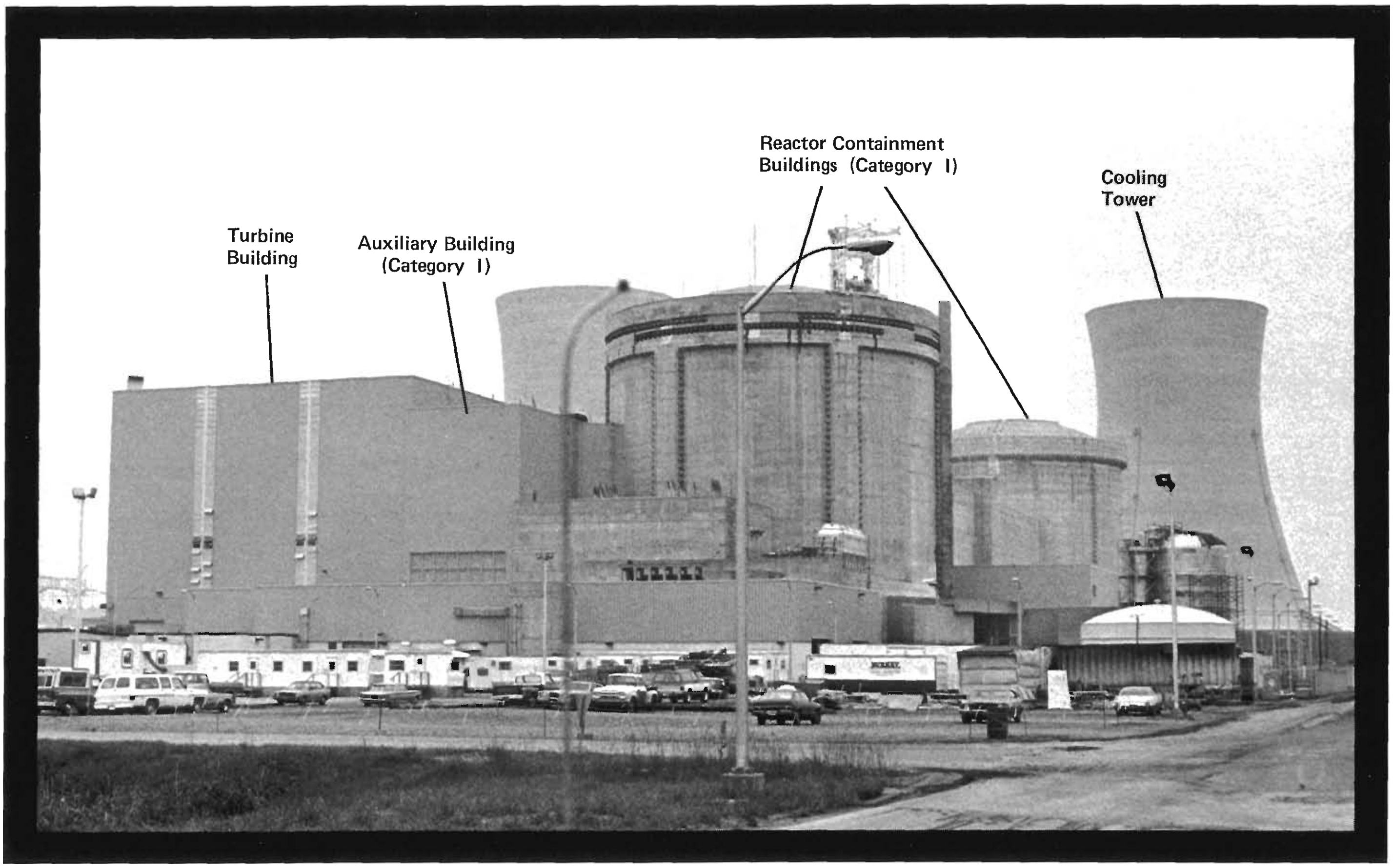

Fig. 1. Building categories at Three Mile Island Nuclear Generating Station. Category I buildings are vital to the prevention of the release of radioactive material during an accident or a natural disaster such as an earthquake. Except for the reactor containment building, these structures are usually box-shaped, are made with reinforced concrete, and include steel columns and beams where deemed necessary. The turbine building and the cooling tower are examples of buildings that are not Category I.

accident, there are structural components in the power plant that must be relied on to protect the public against the consequences of such an accident. Moreover, these structures must also provide protection of sensitive plant equipment during natural disasters such as earthquakes and tornados. As a result, the proper design of these structures must take into account a wide variety of loads and failure modes.

SAFE DESIGN. Any structure that can initiate an accident sequence if it fails or that must remain functional during an accident to prevent release of radioactive material is called a Category I structure. In a typical nuclear power plant (Fig. 1) the building housing the reactor core and the control building are both Category I. Auxiliary and equipment buildings are considered Category I if they include vital equipment such as backup diesel generators, safety valves, the spent-fuel pit, or fuel handling and radioactive waste facilities. A turbine building is usually not a Category I structure, although its potential impact on adjacent Category I structures must be considered.

Safe design of Category I structures is the responsibility of an architect-engineer under contract to the electrical power utility. Crucial to his work are design-basis loads. Certain of these, such as earthquake and tornado-born missile loadings, are site specific, while others, such as pipe-break loadings, are plant specific. The architect-engineer sizes the plant structural members both to withstand various combinations of these design-basis loads and to transmit only acceptable loads to sensitive plant equipment.

The Nuclear Regulatory Commission insures proper design by requiring the architect-engineer to adhere meticulously to certain design-procedure rules. These include the Commission's regulatory guides and Standard Review Plan, as well as the pressure vessel and piping codes of the American Society of Mechanical Engineers (ASME), and the construction codes of the American Concrete Institute. Also, certain of the Category I structures critical to the safety of the nuclear plant are tested before the plant is permitted to operate. Thus, the containment building is subject to a static internal pressurization of 15 per cent over the design-basis pressure.

MARGIN TO FAILURE. The design of Category I structures is inherently conservative since it is based on, among other things, restricting loads to the linear elastic region of material behavior. Thus, a typical structure stressed by design-basis loads will behave elastically and return to its original configuration upon unloading (Fig. 2). It is well known, however, that there is a large additional capacity beyond elastic behavior for resisting applied loads. This capacity can be used to ameliorate the consequences of accidents that load the structure beyond the expected design-basis loads. Design procedures using this reserve capacity are allowed in Europe, but not, at present, in the United States.

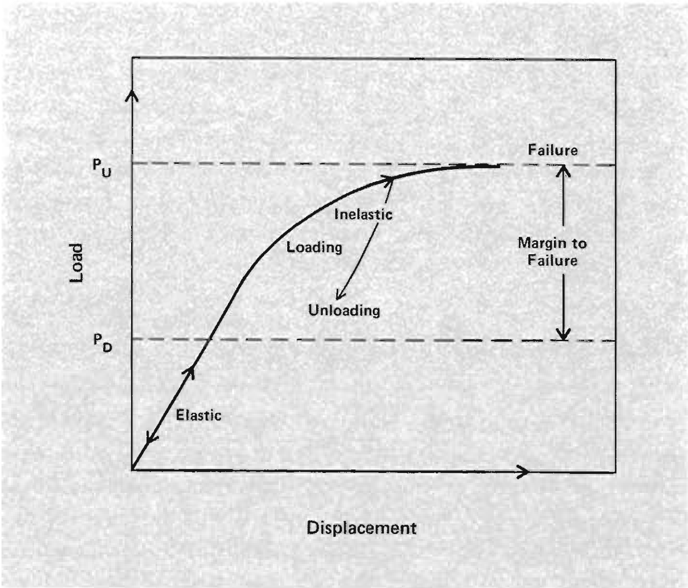

Fig. 2. Representative load-displacement relationship for Category I structural elements. In the elastic region the element will unload by returning to its original shape. However, in the inelastic region loading is large enough to cause a permanent net displacement such as in buckling or crushing. The design-basis load, P_D, is in the linear region while the ultimate load before failure, P_U, includes the reserve capacity due to inelastic deformation. The quantity $P_U - P_D$ is proportional to the margin to failure defined in the text.

A useful measure of this reserve is the margin to failure, defined as

$$\frac{P_U}{P_D} - 1 \; .$$

The variable P_D is the design-basis load and P_U is the ultimate load on the structure before failure, including the inelastic reserve capacity.

Why is it important to assess the margin to failure? The Diablo Canyon nuclear power plant, sited in an earthquake-prone area, is one example. A fault was found near the plant after it had already been constructed, so the potential seismic loads are greater than those for which the plant was designed. Knowledge of the structural margin to failure under earthquake loadings would greatly help now in relicensing the plant under revised seismic criteria. Also, knowledge of the ultimate load capacity for the Three Mile Island reactor containment building would have done much to allay the concern about the possible rupture of that containment by a hydrogen explosion.

But there are difficulties in determining the margin to failure. The behavior of Category I structures near P_U is strongly nonlinear and is often characterized by cracking and crushing of concrete, yielding of metals, buckling of metal shells, and slippage at support points. Thus a realistic treatment of this behavior will necessarily involve mathematically sophisticated analyses using computers followed by careful experimental verifications.

Los Alamos Program

Los Alamos and Sandia National Laboratories, under the sponsorship of the Nuclear Regulatory Commission, are carrying out a research program to develop methods that determine ultimate load capacities. The Los Alamos program is studying the failure of two types of Category I structural systems: concrete box-type structures where heavy shear walls provide resistance to earthquake ground motion, and steel containment vessels which could fail by buckling. Specific program tasks are 1) to develop analytical or numerical models for the behavior of these structures near ultimate load, 2) to verify these models with experiments on scaled structural systems, and 3) to propose amendments to code rules or the Nuclear Regulatory Commission's licensing requirements. Participants in the program include Laboratory contractors (e.g., the Earthquake Engineering Research Center at Berkeley, where we plan to carry out seismic testing) and an advisory committee of persons from universities and relevant industries to help plan the program and review the results.

BUCKLING OF STEEL CYLINDERS. As an example of how the Los Alamos program is working, the recent study of the buckling of thin-walled steel cylinders with large penetrations will be outlined.

Figure 3 shows the large concrete and steel containment building used for certain light-water reactors. The building is the last line of defense in the event of an accidental break in the reactor pressure vessel or its associated coolant system. One example of a penetration is shown: the entry for personnel and equipment which, during normal operation of the reactor, would be closed and sealed. Other sealed penetrations exist for pipes and cables. The ice condenser (an ice-filled device included to condense steam during an accident) is shown as an example of a large mass of equipment attached to the steel containment shell.

Buckling of the inner steel vessel in this structure may occur during several types of accidents. For example, if a coolant pipe, carrying water at about 150 bars, suffers a large break inside the building, a high-pressure jet will be directed against the steel cylinder.

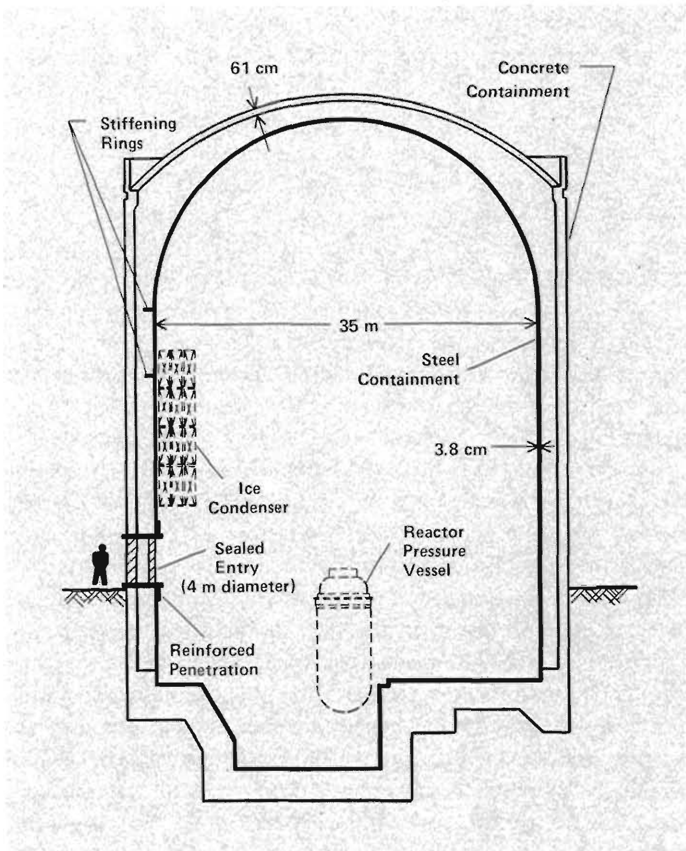

Fig. 3. One type of steel and concrete containment for the main reactor building. A large reinforced penetration for personnel and equipment access is shown; other smaller penetrations would be included for pipes and cables. The heavy ice condenser attached to the wall is an example of a source of asymmetrical loading that could lead to buckling during an accident. The reactor pressure vessel and its associated coolant system constitute another containment barrier within the containment building.

Or during an earthquake the large masses attached to the cylinder and the shifting of the structure relative to the large pipes penetrating the vessel will result in buckling stresses. Also, during a steam explosion, the ice condenser may create sharp temperature and pressure gradients that result in asymmetrical loading of the shell. If buckling occurs, radioactive material can be released in at least two ways: through any punctures that result from the impact of the displaced shell against adjacent structures and through any broken seals around penetrations.

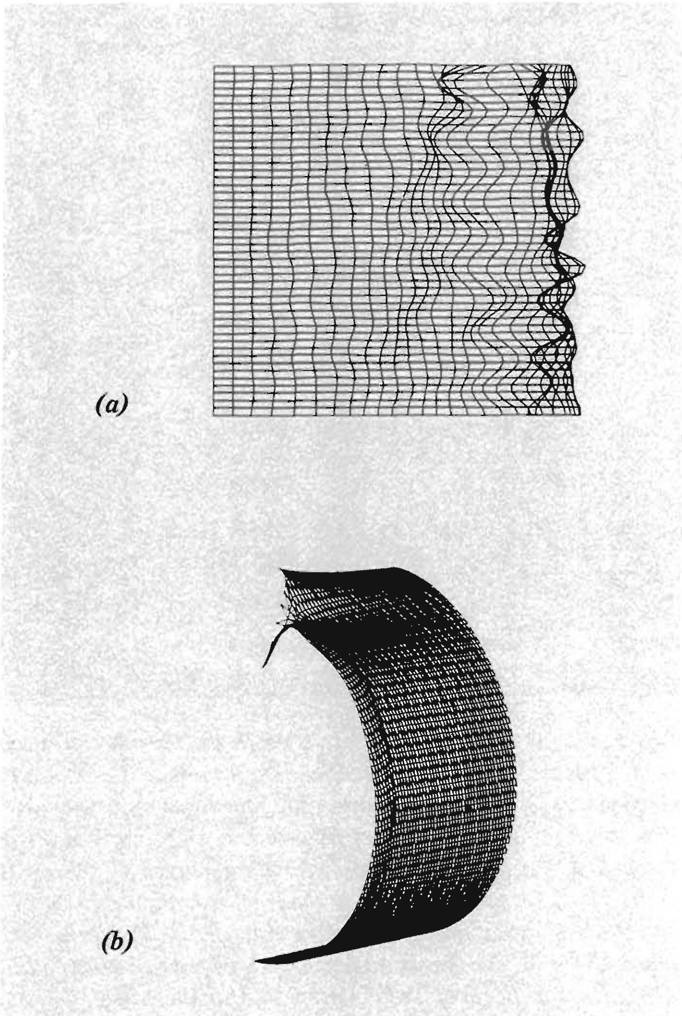

Fig. 4. Two examples of the computer-generated buckled shapes of steel cylinders. Part (a) is a side view of an unpenetrated cylinder and shows the wave pattern typical of a buckling failure. In part (b), the computer has rotated a section of a penetrated cylinder to reveal the buckling that occurs close to the hole at the top left edge of the mesh.

The entry for personnel and equipment constitutes the largest penetration (about 4 meters in diameter) of the containment shell. An important question is how this penetration affects the buckling stability of the shell. The ASME code rules specify the amount of reinforcing needed around the penetration to keep it from affecting the ultimate load capability when failure is by plastic flow of steel. Such material flow is the type of failure normally encountered in thick-walled steel boilers and pressure vessels and is, thus, the type of failure originally dealt with in the code. However, reactor contaiment vessels are thin walled and subject to buckling. Will the same code rules specifying the amount of reinforcement work for both failure types?

COMPUTER ANALYSIS. Los Alamos engineers first performed an analytical study of this problem using a three-dimensional, finite-element buckling code. This computer code calculates in model structures the stresses caused by external loads and can show how the stresses are affected by changes in geometry. Thus, a long, narrow, perfectly straight column under axial loading (a compressive force on both ends of the column) has a certain load-carrying capacity determined by the material's capability to withstand stress. If, however, the column is slightly bowed, the ultimate load-carrying capacity is reduced dramatically.

To analyze this type of problem, the model structure is divided into a large number of cells, equations of elastic equilibrium are formulated numerically for each cell, and the equations are solved by the code for the given loads. The equations include both a linear term representing small elastic deflections and a nonlinear term representing the effect of large deflections on the stresses in the structure. It is through this last term that the buckling behavior is incorporated into the analysis and the margin to failure determined.

The analytical study of containment vessels attempted to identify the buckled shapes at failure of unpenetrated, penetrated, and penetrated-reinforced cylinders when subjected to axial loading. The calculations also simulated the imperfections in both geometry and end loading that naturally occur in steel cylinders; that is, a typical fabricated cylinder will not be perfectly round and will not have a perfectly constant height for the end loading to bear down upon. Examples of computer-generated buckled shapes are shown in Fig. 4.

The analysis showed that the penetration significantly lowered the ultimate buckling load. Also, while imperfections in roundness had only a small influence on this load, the buckling capacity of the cylinder was very sensitive to height imperfections and, therefore, the distribution of the applied end load. Finally, the calculations showed that reinforcing the penetration according to the ASME code would raise the buckling load, but not to the value for the unpenetrated cylinder.

SCALED EXPERIMENTS. A comprehensive series of experiments was carried out to verify the analytical results. Steel cylinders simulating containment shells were fabricated to one-sixtieth actual size. A number of these cylinders were left unpenetrated; others were fabricated with a scaled penetration and then reinforced to various amounts according to the ASME code rules; none were stiffened by rings as is normal for containment vessels. The cylinders were checked for roundness, end parallelism, variation in wall thickness, and other fabrication imperfections, and then were instrumented with strain gages. Measured imperfections were similar in scaled magnitude to those measured in actual reactor containment shells. After careful shimming between the cylinder and testing machine to help approach uniform end loading, the cylinder was loaded to failure. Figure 5 shows one of the penetrated and reinforced cylinders after testing.

These experiments clearly showed that fabrication imperfections dominated the buckling failure of steel cylinders. For example, unpenetrated cylinders buckled at a load considerably lower than the value predicted for perfect cylinders. Cylinders with penetrations but no reinforcement failed at essentially the same load as unpenetrated cylinders; that is, the effect of the hole was apparently too small to cause buckling before the shell failed from imperfections. Imperfections are, thus, felt to be the main reason for the considerable scatter in the data for the steel cylinders shown in Fig. 6 (dots). For comparison, data (triangles) are also plotted from a study of a reusable Mylar shell. Because of the high quality of the Mylar cylinder, these data show little scatter as the buckling load increases with reinforcement. In both cases, the amount of reinforcement is expressed as a percentage of that recommended in the ASME code for reinforced penetrations.

Since the computer analysis indicated the ultimate buckling load to be highly sensitive to the distribution of the applied end load, the data were examined with this idea in mind. Strain gage records from the experiments were used to determine a parameter, A, measuring the degree of asymmetrical loading with respect to the position of the hole. When the load at which the first buckling occurs is plotted versus this parameter, the expected correlation becomes apparent (Fig. 7). If A is greater than 1, the hole is overloaded with respect to the average load on the cylinder, and this leads to the predicted lower buckling loads. When A is less than 1, the opposite effect occurs.

The data, viewed in this light, supported the analytical conclusion that reinforcing the penetration in the manner prescribed by the ASME code would increase the buckling load, but not back to the unpenetrated value. More importantly, this description of the buckling study reveals the importance of the interplay of analysis and experiment in revealing key parameters and their effect on the ultimate failure load. As it turns out, a part of the ASME code accounts for fabrication imperfections in a manner that agrees with the results of the buckling tests; it is this part of the code that insures a margin to failure for the buckling of steel containment vessels when the normal imperfections of these vessels dominate the failure.

Experiments are now underway to investigate the buckling behavior of ring-stiffened scale models of reactor containment shells for loadings that could occur under accident conditions. These experiments will be used to benchmark the computer codes being proposed to predict the ultimate load-carrying capacity of containment shells. Other experiments will investigate the behavior of reinforced concrete shear walls at ultimate load. Information from all these experiments will be used by the Nuclear Regulatory Commission to help establish the margin to failure for Category I structures subjected to severe accident loads. ■

Fig. 5. One of the steel cylinders used in the simulated tests of containment shell buckling. The central hole is a scaled representation of the penetration for personnel and equipment access. In this case, the penetration has been reinforced inside the cylinder to 33 per cent of that recommended by the ASME code. Considerable buckling is evident around the hole. The small wired devices attached to the cylinder on both sides and above the hole are strain gages. Other gages are attached on the sides and back of the cylinder and at the same positions inside.

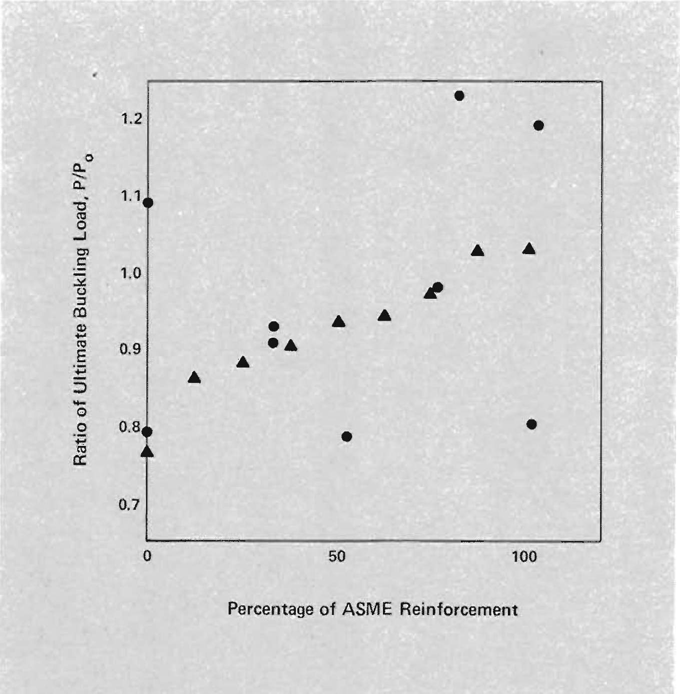

Fig. 6. The dependence of buckling load on the amount of reinforcement (expressed as a percentage of that recommended by the ASME *code). The load ratio used here,* P/P_0*, is the ratio of the ultimate buckling load for a penetrated-reinforced cylinder (P) to that for an unpenetrated cylinder* (P_0)*. Thus, a value of one for* P/P_0 *means the penetrated-reinforced cylinder was as strong as the unpenetrated cylinder. The triangles are from a buckling study of a reusable, high-quality cylinder and thus show little scatter as increased reinforcement raises the buckling load back to the value for the unpenetrated cylinder. The large scatter in the Los Alamos steel cylinder data (dots) is felt to be due largely to the variation of fabrication imperfections from cylinder to cylinder.*

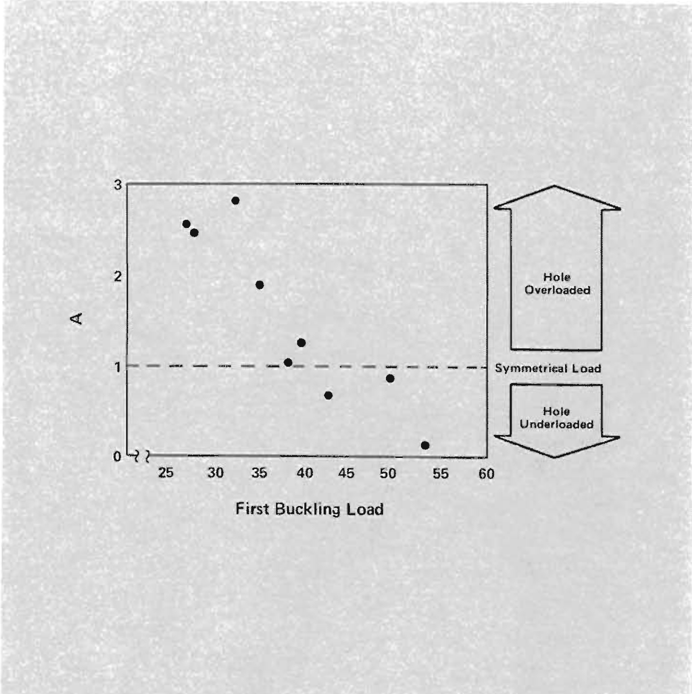

Fig. 7. The effect of asymmetrical loading on the magnitude of the buckling load. The parameter A is a measure of the degree of asymmetrical loading with respect to the position of the hole. When A is greater than 1, the hole was overloaded with respect to the average load on the cylinder and buckling occurred at lower loads. When A is less than 1, the hole was underloaded and buckling occurred at higher loads. The correlation shown here demonstrates that load asymmetry resulting from height imperfections accounts for much of the experimental scatter in steel cylinder buckling loads.

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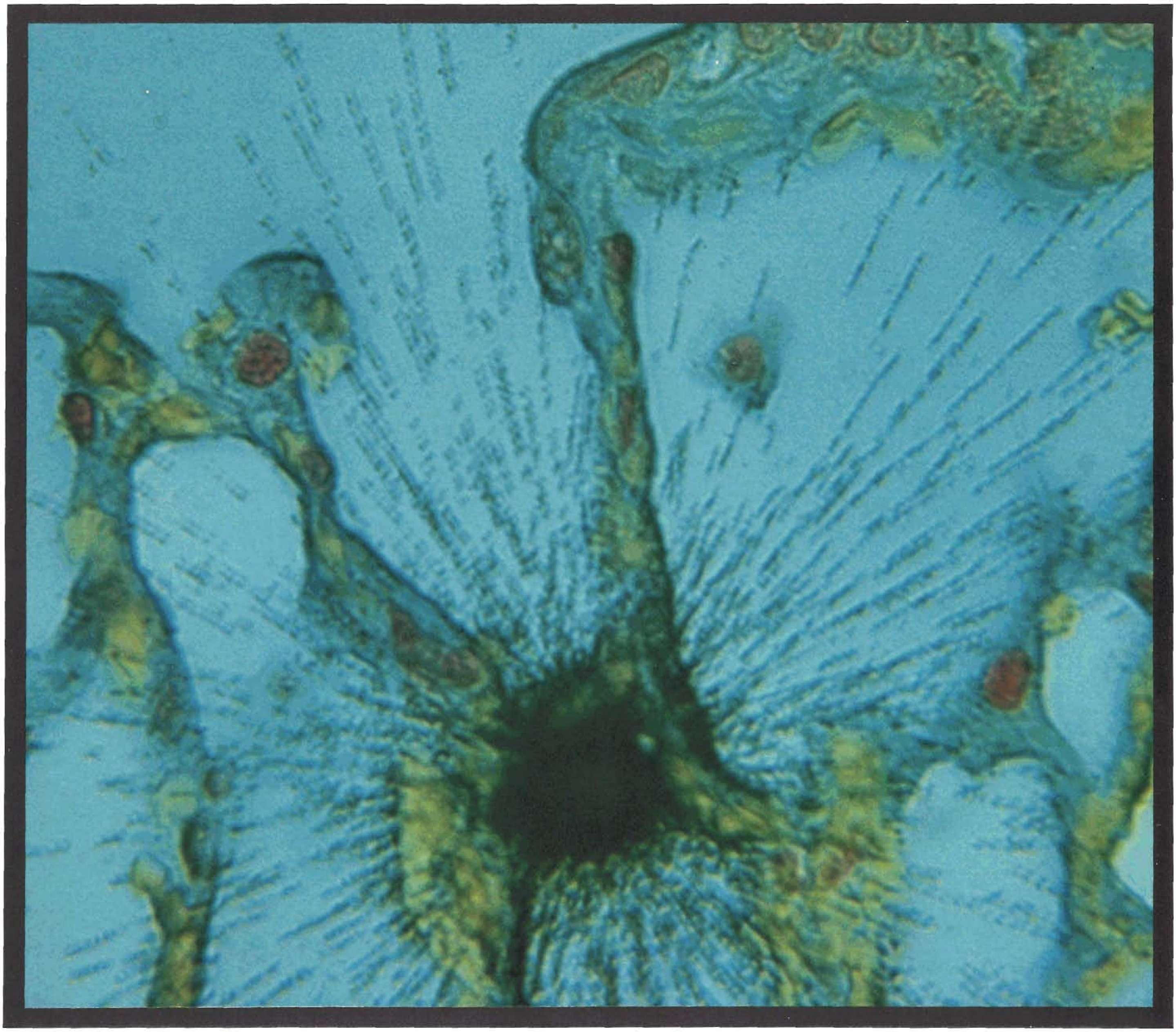


Autoradiograph of radioactive microsphere embedded in hamster lung tissue. Ceramic microspheres slightly larger than red blood cells and containing less than 1% plutonium-239 by weight lodge in pulmonary capillaries when injected into the jugular vein. The streaks emanating from the microsphere, forming an "alpha star," are alpha-particle ionization tracks in the film emulsion. Cell nuclei, here stained red-brown, contain DNA and are potential sites for initiation of radiation-induced cancer tumors, but there is no evidence of abnormalities in this tissue section despite an 18-month exposure to radiation from the microsphere. This example is from research by the Laboratory's Life Sciences Division on the role of internally deposited radionuclides in pulmonary diseases, including cancer.[1] *(Photo by David M. Smith and James R. Prine)*

LOW-LEVEL RADIATION How harmful is it?

by Roger C. Eckhardt

"Last Friday, the Holsteins on the Lytle Farm started acting kind of touchy, lining up side by side at the fence and staring south. That was two days after the Three Mile Island nuclear power plant, five miles due south as the cow stares, started generating fear instead of electricity." —A journalist for *The New York Times.*

"If these cows start leaving town on their own, I'm getting out of here too."—Clarence Lytle 2nd, partner on the Lytle Farm.

"I've been working with this for ten years, and I have a pretty thorough familiarization. I'm not saying I'm brave. If you understand, your mind is at ease."—Edward Houser, Three Mile Island chemistry foreman and the worker who received the highest dose on the day of the accident.

"I don't know about that stuff, that nuclear. Sounds to me so powerful man can't tame it right."—72-year-old resident of Yocumtown, Pennsylvania.

"The amount of radiation that escaped was no threat to the people in the area. . .the radiation outside the plant was far less than that produced by diagnostic x rays."—Officials of the Nuclear Regulatory Commission.

"I don't think they're telling us the whole truth. They won't come out and say, 'Yes, everything is all right.' "—Resident of Highspire, Pennsylvania.

"Any dose is unsafe because there is no lower threshold for radiation."—George Wald, Nobel laureate and Emeritus Professor of Biology at Harvard University.

These reactions* to the accident at Three Mile Island make clear the fear and confusion regarding the potential radiation hazard from nuclear power plants. There are those who fear mutant babies and glowing cows and who oppose nuclear energy and its invisible radiation dangers no matter what safeguards are instituted. Others argue that nuclear energy can be rendered free of radiation hazards, but only at the expense of a nuclear police state. Still others feel that nuclear power is a pollution-free, benign source of energy, and the only viable solution to our nation's energy crisis.

Contributing to the fear and confusion is a range of scientific opinion about the long-term effects of low doses of ionizing radiation. There is no doubt that high doses have deadly results for man: a single dose of 600 rems of gamma radiation would likely result in death within a month to a majority of the exposed population.[2] For doses 100 or 1000 times less, which are relevant to radiation workers and the general public, respectively, the effect believed to be most important is an increased risk of cancer. But the extent of the risk is a subject of controversy, and estimates differ by as much as a factor of 100. For example, included in the most recent and most respected report on this subject,[3] familiarly known as BEIR III, are dissenting statements by two members of the preparing Committee. One member characterizes the published risk estimates as too low, and the other as too high.

The controversy has its basis in one simple fact. There are no unambiguous data on the incidence of effects at the low doses received by workers in the nuclear or medical industries, and the lack of data at doses characteristic of the general public is even more complete. To develop a reasonable model or make accurate predictions, scientists need data bearing directly on the phenomenon being considered; otherwise, the models are only educated guesses subject to further modification and the predictions are only extrapolations. This is the situation with the biological effects of low-level ionizing radiation.

The most widely accepted estimates for the effects of low-level radiation are based on extrapolation of data on survivors of the Nagasaki and Hiroshima bombings. These survivors experienced a single, moderate to high exposure (10 to 400 rads mean dose to the tissue). In the absence of a real theory, the correct technique for extrapolation to lower doses is unknown, and many factors, such as dose rate, are not considered in the data analysis. The data base itself is now being questioned because the relative amounts of gamma rays and neutrons released in the explosions may have been different than assumed.[4-6]

Many animal data are being gathered, but their relevance is

**All quotations are from issues of* The New York Times *during the week following the Three Mile Island accident. © 1979 by the New York Times Company. Reprinted by permission.*

unknown. A dose accumulated over 30 years in humans cannot be duplicated in animals that live only several years. Also, how valid are extrapolations from animal to man when significant differences between radiation-induced effects in laboratory animals of different species are frequently observed?

Ideally, epidemiological studies of humans exposed to the doses, dose rates, and types of radiation of most concern should be the basis for risk estimates. Such data are not only difficult to acquire, but also include the effects of other causative agents, such as chemical carcinogens, natural background radiation, other manmade radiation sources, and even particular social and psychological habits.

Can a quantitative range be placed on the scientific uncertainty that results from these problems? Figure 1 depicts the currently expected number of deaths due to cancer among a million people in the United States and, also, two different estimates of excess cancer deaths resulting from an additional exposure to the population of one rad of x or gamma rays per person. One estimate represents those published in BEIR III and the other, greater by an order of magnitude, represents the typical range of scientific uncertainty. The fact that the estimated excess cancers from a 1-rad dose cannot be shown on the same scale as the expected deaths illustrates the difficulty in detecting the effects of such exposures, much less of doses down to millirads. The figure also illustrates that the range of scientific uncertainty is much more circumscribed than the range of opinion among the general public.

Uncertainty about the hazards of low-level radiation is well-grounded and will persist, possibly indefinitely. Here we will attempt to answer some of the questions about ionizing radiation and discuss the rationale behind radiation protection standards. Perhaps the perspective we present will allay exaggerated fears. Although it may be true that no radiation dose is absolutely safe, in fact, the risk from doses comparable to those received by the public in the vicinity of the Three Mile Island accident is so low as to be undetectable.

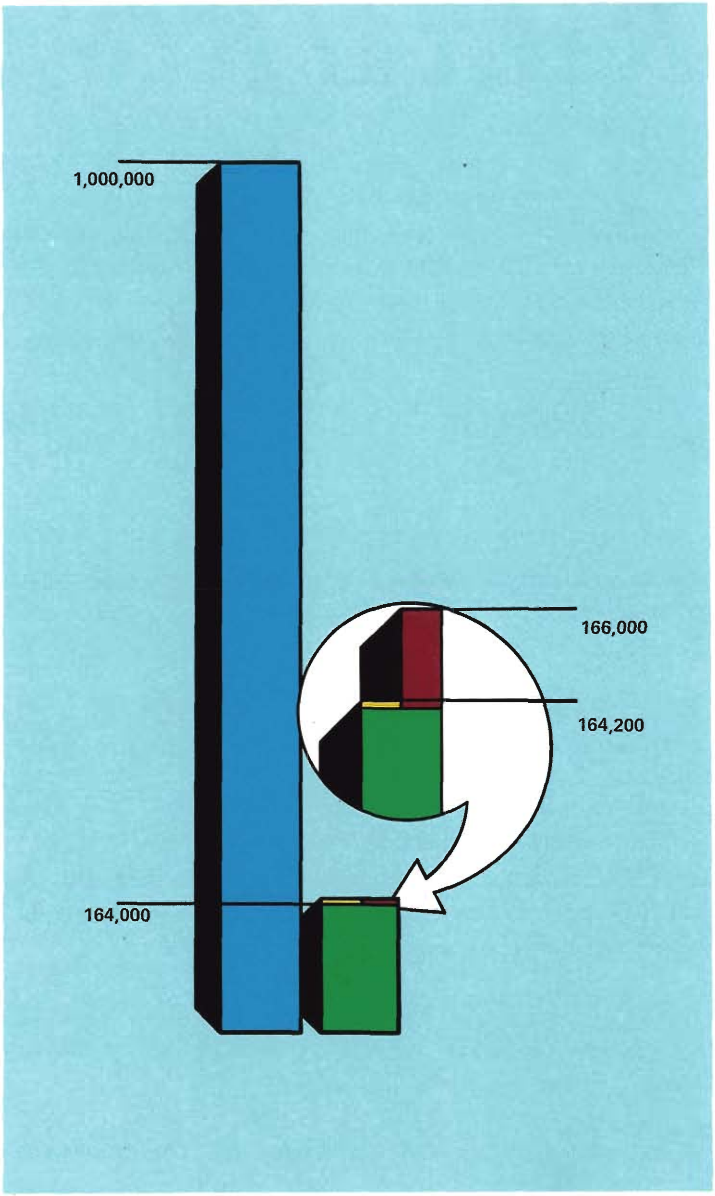


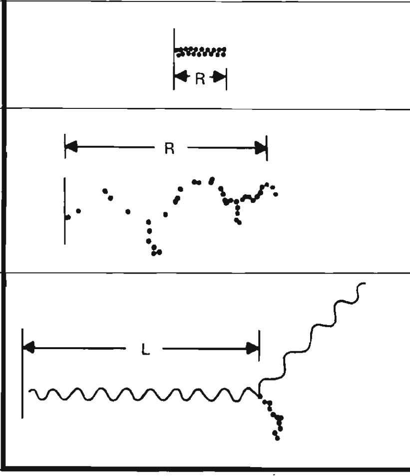
Fig. 1. Among a representative population in the United States of 1,000,000 (blue), the currently expected number of deaths due to all forms of cancer (green) is 164,000. The number of excess cancer deaths resulting from an additional 1-rad exposure of the population to x or gamma radiation (yellow) is, according to BEIR III, approximately 200. Also shown (red) is the number of deaths if the risk estimates are greater than those of BEIR III by an order of magnitude, a variation typical of current scientific uncertainty.

What are the Natural and Manmade Sources of Ionizing Radiation?

Natural background radiation has always been and still remains the greatest contributor of ionizing radiation to mankind. There are two main sources of this radiation. One is

Sidebar 1:

IONIZING RADIATION



Radiation	Track Shape and Density (not to scale)	Radionuclide	Energy (MeV)	R or L (mm)
Alpha Particle		^{239}Pu	5.2	0.046 (average)
Beta Particle		^{14}C	$\leqslant 0.15$	$\leqslant 0.3$ mm
		^{90}Sr	$\leqslant 2.2$	$\leqslant 11.0$ mm
Gamma Ray		^{125}I	0.035	33 (average)
		^{60}Co	1.3	160 (average)

Characteristics of ionizing radiation from typical radionuclides. The dots in each track represent ionizations. For alpha and beta particles, R is the range in water; for gamma rays, L is the distance in water to the initial ionizing interaction.

The term ionizing radiation refers to electromagnetic radiation and particles with enough energy to cause ionization of the atoms or molecules of the irradiated material. Alpha particles, beta particles, and x rays are forms of ionizing radiation, but the ultraviolet light reaching the surface of the earth and microwave radiation are not. Of most concern is ionizing radiation with energies of millions of electron volts, including gamma rays and the particles ejected during the decay of such radionuclides as uranium-238 and strontium-90.

Charged particles, such as alpha and beta particles, cause ionization by direct Coulomb interaction with the irradiated material. Electromagnetic radiation, such as x and gamma rays, or uncharged particles, such as neutrons, generate secondary charged particles through absorption of electromagnetic energy or direct collisions. The secondary particles then ionize the irradiated material.

In a living cell, the sudden passage of the intense electric field of these particles disrupts the delicate orientation of water and protein molecules and generates organic free radicals, which react with enzymes, chromosomes, and other molecules necessary to the cell's life processes.

The critical element for understanding the interaction of ionizing radiation and matter is energy deposition. The amount of energy deposited, or the "absorbed dose," is measured in rads. In biological matter, however, different types of radiation can deposit the same total energy but produce different amounts of damage. For example, alpha particles, which produce high ionization densities along their paths, cause more cancer than do x or gamma rays. The unit used to quantify the degree of damage is the rem. The rem is the dose in rads times a quality factor appropriate to the type of radiation. ■

COMMON RADIATION UNITS

Unit	Measured Quantity	Definition
Curie	Radioactivity of source	1 curie = 3.7×10^{10} decays per second
Roentgen	Ionization produced by radiation (defined for x and gamma rays only)	1 roentgen produces 1 electrostatic unit of charge in 1 cubic centimeter of air at standard temperature and pressure
Rad	Energy deposited in matter by ionizing radiation	1 rad = 0.01 joules per kilogram of irradiated material
Rem	Energy deposited times a quality factor representing biological damage	rems = Q $\times$ rads Q = 1 for x and gamma rays Q = 10 for alpha rays

cosmic radiation produced by collisions of high-energy particles impinging continuously on the earth's atmosphere. The atmosphere serves as a shield, but a fraction of the radiation reaches the earth's surface and results in whole-body irradiation of the population. The thinner atmospheric shield present at higher altitudes and during airplane flights results in doses larger than those at sea level. Table I lists dose estimates for this and other radiation sources and notes the body portion exposed.

The other source of background radiation is naturally occurring radionuclides. These radionuclides surround us in the environment, particularly in the soil, and reside in our body after being ingested in air, food, and water. An individual's annual dose from terrestrial sources outside the body depends on the amounts of elements such as uranium, thorium, or potassium in the soil and can vary by an order of magnitude. The main contributor of internal beta and gamma radiation from ingested radionuclides is potassium-40, a radioactive isotope of an element vital to life. Another radionuclide currently of concern is radon. This element can diffuse out of brick, concrete, stone, soil, and water and build up in tightly sealed, energy-efficient homes.

To this pervasive background radiation must be added the manmade sources of ionizing radiation. One of the most significant of these is the medical use of x rays. Of comparable significance in 1963 was the radioactive fallout from atmospheric weapons testing. This source, however, has since declined markedly. Other sources include research activities

TABLE I

SOURCES OF IONIZING RADIATION[a]

Source	Body Portion Exposed	Average Annual Exposure (mrem/yr)
NATURAL		
Cosmic radiation		
Sea level	whole body	26
1.8 km (6000 ft)	whole body	52
External radionuclides		
Atlantic and Gulf coasts	whole body	15-35
Colorado Plateau	whole body	75-140
Internal radionuclides		
Potassium-40	gonads	19
Potassium-40	bone marrow	15
Uranium isotopes	gonads	12
Total Natural Radiation	whole body	100
	gonads	80
	bone marrow	80
	lungs	180-530
MANMADE		
Medical use of x rays	gonads	20
Atmospheric weapons testing		
1963	whole body	13
1980	whole body	4.4
Nuclear operations	whole body	<1
Cigarette smoking	localized points in lungs	≤8000

[a]All dose estimates are from Ref. 3, pp. 37-69.

and a wide range of consumer and industrial products, such as television, luminous watch and clock dials, airport x-ray devices, smoke detectors, static eliminators, tobacco products, fossil fuels, and building materials. These last collectively add only slightly to the average dose.

In light of public response to ionizing radiation, the last two sources listed in Table I are of particular interest. The average annual dose of an individual in the United States resulting from nuclear operations is estimated to be less than 1 millirem per year. In contrast, a cigarette smoker may be burdening the surface of his bronchial tract at highly localized points with up to 8000 millirems per year.

By keeping these doses due to natural and manmade sources in mind, the doses resulting from the Three Mile Island accident[7] can be put in reasonable perspective. The radionuclides released during the accident resulted in an average estimated dose of 1.4 millirems to the approximately 2,000,000 people living in the vicinity of the plant. This whole-body dose is lower than the typical bone-marrow dose of 10 millirems per chest X ray and is more than an order of magnitude lower than the average annual whole-body dose of 26 millirems from cosmic radiation at sea level. In the extreme case of an unclothed individual standing outdoors, 24 hours a day for 6 days, across the river from the plant in the path of the prevailing winds, the total dose received has been calculated to be below 100 millirems, that is, below the total whole-body dose due to natural background radiation. The highest exposures resulting from the accident were to several of the plant personnel who received doses of approximately 4 rems. These doses are the only potentially significant ones, being in excess of the quarterly limit of 3 rems allowed for radiation workers by the Nuclear Regulatory Commission.

What Biological Effects of Low-Level Ionizing Radiation Are of Most Concern?

The biological effects of primary concern are not the drastic and immediate effects of high doses but the more subtle late effects, such as cancer and gene mutation, that may result from prolonged or sporadic exposure at low levels. These effects are classified as genetic or somatic. Somatic effects, of which cancer is the most important, are experienced directly by those exposed, whereas genetic effects are experienced by their descendants. Genetic effects involve damage specifically to the germ cells in the gonads, whereas somatic effects involve a wide range of body cells.

Only the radiation dose received by the gonads of future parents during their reproductive span is of genetic significance. The average gonadal dose of manmade radiation to an individual in the United States is approximately 30 to 40 millirems per year. During a 30-year human reproductive span, this dose rate produces an additional genetically significant dose of roughly 1 rem. BEIR III estimates the increase in genetic disorders due to continued exposure of many generations at this level to range from 60 to 1100 disorders per million liveborn.[8] This estimate should be compared to the current incidence of 107,000 genetically related disorders per million liveborn.

Twenty years ago, genetic effects were believed to be far more important than somatic effects. However, this conclusion was drawn from animal experiments in which the dose was delivered at high rates. Further studies have shown that lower dose rates, such as those characteristic of occupational exposure, are less effective at inducing genetic effects. Also, estimates of the cancer induction rate have increased as the study populations age and more slowly developing cancers appear. The net result is that cancer is now considered to be the most important late effect of exposure to radiation.

Although members of the BEIR Committee disagreed about the risk of radiation-induced cancer, there were many points concerning this effect on which the Committee members were in complete accord. Some of the more important of these accepted points are listed below.

- The latent period of cancer (the time between exposure and the appearance of cancer) may be long—years or even decades.
- Nearly all tissues and organs of the human body are susceptible to radiation-induced cancer, but sensitivity to the induction of cancer varies considerably from site to site.
- Leukemia was at one time thought to be the principal type of radiation-induced cancer; however, solid cancers, such as lung, breast, and thyroid cancers, are the more numerous result.
- Age, both at irradiation and diagnosis, is a major factor in cancer risk; for example, a very high risk of leukemia was found in atomic-bomb survivors irradiated in the first years of life, and the highest risk of radiation-

induced breast cancer in women occurs for exposures in their second decade of life.

- Because of the greater incidence of breast and thyroid cancer in women, the total radiation-induced cancer risk for women is greater than for men.
- There is an increasing recognition that certain human genotypes are more susceptible than others to cancer after exposure to radiation (and other carcinogens), but the role of susceptibility in cancer induction is not yet well understood.
- There is evidence that the dose rate may change the radiation effect per unit dose, but the information currently available is insufficient to be used meaningfully when estimating the risk of cancer induction in man.

Although controversy surrounds the BEIR III risk estimates for radiation-induced cancer, we quote two of the estimates eventually published in that report.[9] A single whole-body dose of 10 rads of x or gamma radiation to a million persons is estimated to result in about 800 to 2200 deaths in excess of the normally expected 164,000 cancer deaths. A continuous lifetime exposure of 1 rad per year of this same type of radiation would result in 4800 to 12,000 excess deaths. It is not yet clear how the new information about the type of radiation released at Hiroshima and Nagasaki will affect these estimates.

How Are the Effects at Low Doses Estimated From the Known Effects at High Doses?

The problems inherent in quantifying the relationship between cancer incidence and ionizing radiation are numerous. To begin with, cancer is actually a group of diseases, and a particular site-specific cancer usually affects less than one person in a thousand each year. In addition, all available data indicate that the increase in incidence caused by radiation is small. We are therefore faced with the problem of detecting a small increase in an already low incidence.

Further, because radiation-induced cancers are indistinguishable from those due to other mechanisms, it is not possible to determine whether a given cancer was caused by radiation or would have occurred even in the absence of exposure. Therefore, evidence for cancer induction by radiation rests on a comparison of site-specific cancer incidence in an exposed group with the incidence in a similar unexposed, or control, group. Unfortunately, the sizes of the groups needed to detect a small absolute cancer excess become extremely large at low doses.

For example, let us assume that an excess cancer incidence is detectable with a particular statistical certainty in an exposed group of 1000 at a dose of 100 rads. Further assume that the excess incidence per rad is the same at all doses. Then, to obtain the same statistical certainty requires an exposed group of 100,000 at a dose of 10 rads and an exposed group of 10,000,000 at a dose of 1 rad. And, of course, similar numbers of people are required for the unexposed groups. Continuation of this reasoning should make it readily apparent why one cannot detect effects of doses in the range of millirads.

As mentioned above, studies of the Hiroshima and Nagasaki survivors have provided the largest data set pertaining to radiation exposure and cancer. Nearly 24,000 persons received doses estimated to be 10 rads or more.[10] To date, statistically significant excesses of various types of cancer have been established for such doses: first leukemia,[11-14] then thyroid cancer,[15] and now lung and breast tumors.[16] For other types of cancer, these studies may provide statistically significant correlations between excess cancer incidence and dose down to about 10 rads.

Other groups examined for radiation-induced cancer include medical patients given x-ray treatments, uranium miners, radium dial painters, radiologists, and nuclear workers. These groups are small and, in addition, have posed difficulties in obtaining correct dose estimates and matched control groups.

As a result, cancer incidence at low doses can generally only be estimated by extrapolating data at higher doses (Fig. 2). The linear, no-threshold hypothesis is the simplest approach to extrapolation. Here it is assumed that there is no threshold dose below which the effect does not occur and that the incidence is directly proportional to the dose. This method of extrapolation has been adopted by Government agencies until conclusive evidence for use of a more appropriate technique is presented.

Another method of extrapolation is to assume a "linear-quadratic" relationship between incidence and dose. Here the incidence is very nearly proportional to dose at low doses, but at high doses the incidence increases more rapidly, namely as the square of the dose. Applied to the same data in the high-dose region, a linear-quadratic extrapolation necessarily predicts lower risks at low doses than does a linear extrapolation. Likewise, a quadratic relationship with no linear term would predict even lower risks.

The BEIR Committee attempted to decide among the linear, linear-quadratic, and quadratic extrapolation techniques for the atomic-bomb data by applying statistical goodness-of-fit tests. They concluded that, in this respect, no one extrapolation technique was more satisfactory. Ultimately they chose to base their risk estimates for cancer on linear-quadratic extrapolation. A possible model for such a relationship attributes the linear term to cancer-inducing lesions, say in the form of broken DNA molecules, generated within a single ionizing track and therefore linearly dependent on dose. The quadratic term accounts for lesions formed through interactions between ionizing tracks, which are thus quadratically dependent on dose.

Another extrapolation method produces higher risk estimates at low doses than does linear extrapolation. Such a relationship may result from the existence of susceptible groups in the population who are harmed at much lower doses

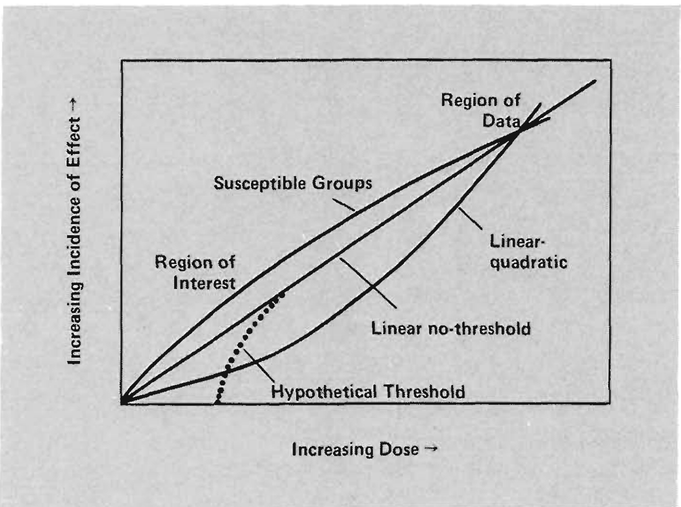


Fig. 2. Experimental data on the incidence of radiation-induced effects are available only at doses higher than those of primary concern. These data are extrapolated to low doses by various techniques. Scientific opinion currently favors linear, no-threshold or linear-quadratic extrapolation for radiation-induced cancer. The susceptible-groups curve illustrates the principle of representing a susceptible population with a higher extrapolation curve.

than are the majority. For instance, there is evidence of greater risk of radiation-induced thyroid cancer in Jewish children than in other ethnic groups.[17] Because the size of these groups is currently believed to be small, this extrapolation technique is not widely used.

How is Low-Level Radiation Separated From Other Factors as the Determining Cause of an Effect?

Regardless of the extrapolation technique chosen, the epidemiologist must carefully assess the influence on the data themselves of many confounding and interactive factors. An especially important factor is the nature of the radiation exposure. Type of radiation, dose rate, dose, exposed organs, available shielding, and specific radionuclides involved—all influence the conclusions and should be accurately determined. For example, studies of the effects due to early medical x-ray treatments may require the rejuvenation and operation of old x-ray equipment to estimate the doses received by the patients.

Personal factors include the subject's size, race, genetic makeup, education, and smoking habits; there is evidence that stress can increase susceptibility to disease, including cancer. Age at time of exposure has already been mentioned as a well-established determinant for cancer risk. Similarly, the altitude and soil composition of the subject's habitat and the subject's occupational experience and exposure to carcinogenic chemicals play important roles.

The long latent period of cancer makes identification of cases and accurate quantification of their radiation exposures extremely difficult. The exposed population must be followed essentially through complete lifetimes, or the risks of late-developing cancers will be seriously underestimated. In fact, one of the first forms of cancer to be associated with radiation, leukemia, was identified primarily because it has a relatively short latent period, occurring as soon as 2 to 5 years after intense radiation exposure.[11]

An epidemiological study[18] of workers at the Hanford Works in Richland, Washington, well illustrates the problems that these factors may cause. (Valid risk estimates derived from studies of workers such as these are extremely important because the exposed group is subject to the highly fractionated, low-dose exposures of most relevance for establishing occupational radiation protection standards.) The investigators reported statistically significant associations between cumulative radiation-badge dose and excess mortality from cancers of many types, but particularly cancers of lung, pancreas, and bone marrow. Their estimates were markedly higher than those obtained from studies of acute, high-dose exposures.

Subsequent studies of the data revealed that the original analysis had not dealt adequately with certain of the confounding and interactive factors, such as age at dose and the demographic difference between exposed and nonexposed workers. After accounting for the neglected factors as best as possible, investigators found significant associations between dose and only two types of cancer, namely, multiple myeloma (a cancer of the bone marrow) and pancreatic cancer.[19]

The risk estimates for these two cancers were still high and implied an improbably large role for background radiation as the cause of the diseases among the general population. On the other hand, if the number of excess cancers of these two types had been low enough to yield reasonable risk estimates, the conventional requirements for statistical significance would not have been satisfied. This quandary is attributed to the limited sample size and low individual radiation doses of the Hanford workers.

To establish valid relationships between dose and effect, more extensive studies are obviously necessary. Since 1976, the Epidemiology Group of Los Alamos National Laboratory has been investigating the effects of plutonium on human health. This study began as a long-term clinical follow-up of the Manhattan Project plutonium workers[20] and was later expanded to a mortality study of 241 plutonium workers.[21] Neither of these efforts demonstrated a relationship between plutonium exposure and adverse health effects. These populations are included in a larger-scale epidemiological study of the approximately 100,000 past and present employees at 6 Department of Energy facilities. This study focuses on the incidence of and mortality due to cancer and other diseases among plutonium workers. Surveillance will continue through 1990 and will comprise a lifetime follow-up for many of the more heavily exposed early workers. Studies of populations residing in the vicinity of the same facilities are also underway.

At present, the mammoth amounts of data needed to establish the existence or nonexistence of excess diseases are being collected. The data include age, sex, ethnicity, chemical and medical x-ray exposures, smoking and other personal habits, and the dosimetry records for each employee. If excesses are demonstrated for the more heavily exposed workers, more data on important confounding factors and risk variables will be collected. Preliminary results are expected soon.

Concurrent with this study, the Laboratory is conducting a nationwide investigation of the deposition and distribution of plutonium and other transuranic elements in human tissue. Plutonium concentrations in the general population due to radioactive fallout are being determined from analyses of autopsy specimens provided by participating hospitals at various locations throughout the United States. In cooperation with the U. S. Transuranium Registry at Hanford, the Laboratory is also amassing data about plutonium concentrations in former nuclear workers, again by analysis of autopsy specimens.

It is hoped that these studies will avoid many of the problems of earlier epidemiological studies and will document the presence or absence of health effects due to plutonium deposition in the occupationally exposed.

How Have the Standards for Exposure to Ionizing Radiation Developed?

At the start of the Manhattan Project, only three radiation-exposure standards existed, all for occupational exposures. Radiation injury to radium dial painters from inhaled or ingested radioactive luminous compounds resulted in the establishment of limiting standards for radon in workroom air, 10^{-11} curies per liter, and for radium fixed in the body, 0.1 micrograms. Extensive occupational exposures to x rays led to the establishent of a limit of 0.1 roentgen per day for external x or gamma radiation. These standards were essentially tolerance doses based on observations of exposed individuals; their acceptance implied the existence of a threshold dose below which no effects occurred.

The years following World War II saw a rapid increase in exposures to a greater variety of radiation types. The National Committee on Radiation Protection (now the National Council on Radiation Protection and Measurements) was organized to examine the complex problems developing in radiation protection.[22] In the ensuing years, standards became more detailed as knowledge of the effects of radiation accumulated. By 1956, genetic hazard was considered the principal limitation on radiation exposure. Also, all exposures were considered cumulative since there appeared to be no cellular

Sidebar 2:

BADGES THAT GLOW

As a research institution, the Laboratory faces a greater variety of radiation exposure situations than do many employers, so demonstration of compliance with current radiation protection standards is not simple. Feeling that the older film badge was inadequate, the Health Division here designed a versatile thermoluminescent dosimeter badge (using Harshaw Chemical Company components) as the primary tool for monitoring radiation doses received by employees. The dosimeter badge can detect a dose as low as 0.01 rem and thus is more than sufficiently sensitive to prove compliance with the current standards. In fact, the badges show a background dose of about 0.4 millirem per day in agreement with the expected background at Los Alamos from cosmic radiation and radionuclides in soil and building materials.

A thermoluminescent dosimeter consists of a lithium fluoride material that absorbs and stores energy when exposed to ionizing radiation. The material has been doped with suitable impurities; free electrons released by the ionizing radiation become trapped at impurity sites where they may remain stored for months or even years at room temperature. However, when the material is heated, the trapped electrons "thermoluminesce" and release energy as visible light. The amount of light released can be measured and is proportional to the radiation dose. In addition, if the material is enriched rather than depleted in ^{6}Li, it becomes much more sensitive to neutron radiation.

The badge includes three neutron-insensitive dosimeters, each covered by a different filter that allows passage of radiation with particular characteristics. A fourth dosimeter contains the neutron-sensitive material.

The measured responses (light outputs) of the four dosimeters provide the following information.

- The "penetrating" dose equivalent to that received about 1 centimeter into the body. This dose is due to gamma rays and high-energy x rays.
- The "nonpenetrating" dose equivalent to that received about 0.007 centimeter into the body. This dose is due to beta particles and lower-energy x rays.
- The neutron dose (to be accurate this reading must be supplemented with a knowledge of the source and any moderating materials).

A Los Alamos employee wearing the Laboratory's thermoluminescent dosimeter badge clipped to his collar. The dosimeter card that holds the four thermoluminescent chips inside each badge is shown on the right. The card is removed and "read" for absorbed dose each month.

A computer program has been written that, using the measured responses, can distinguish between the low-energy x rays and beta particles of the nonpenetrating dose, estimate the dose due to beta particles only, and determine the fraction of beta particles in a mixture of gamma rays and beta particles. Moreover, the badge acts as a crude spectrometer estimating the energy of low-energy x rays and the effective energy of a mixture of low-energy x rays and gamma rays. This is necessary since correction factors must be used to calculate doses due to photons below 100 kilo electron volts in energy.

Recently, dosimeter badges submitted by 60 different processors were judged according to a standard developed by the Health Physics Society Standards Committee. Performance was measured in eight categories of radiation type and energy; each radiation category was divided into several dose-range intervals. Only the Laboratory's thermoluminescent dosimeter badge performed satisfactorily in all eight categories. ■

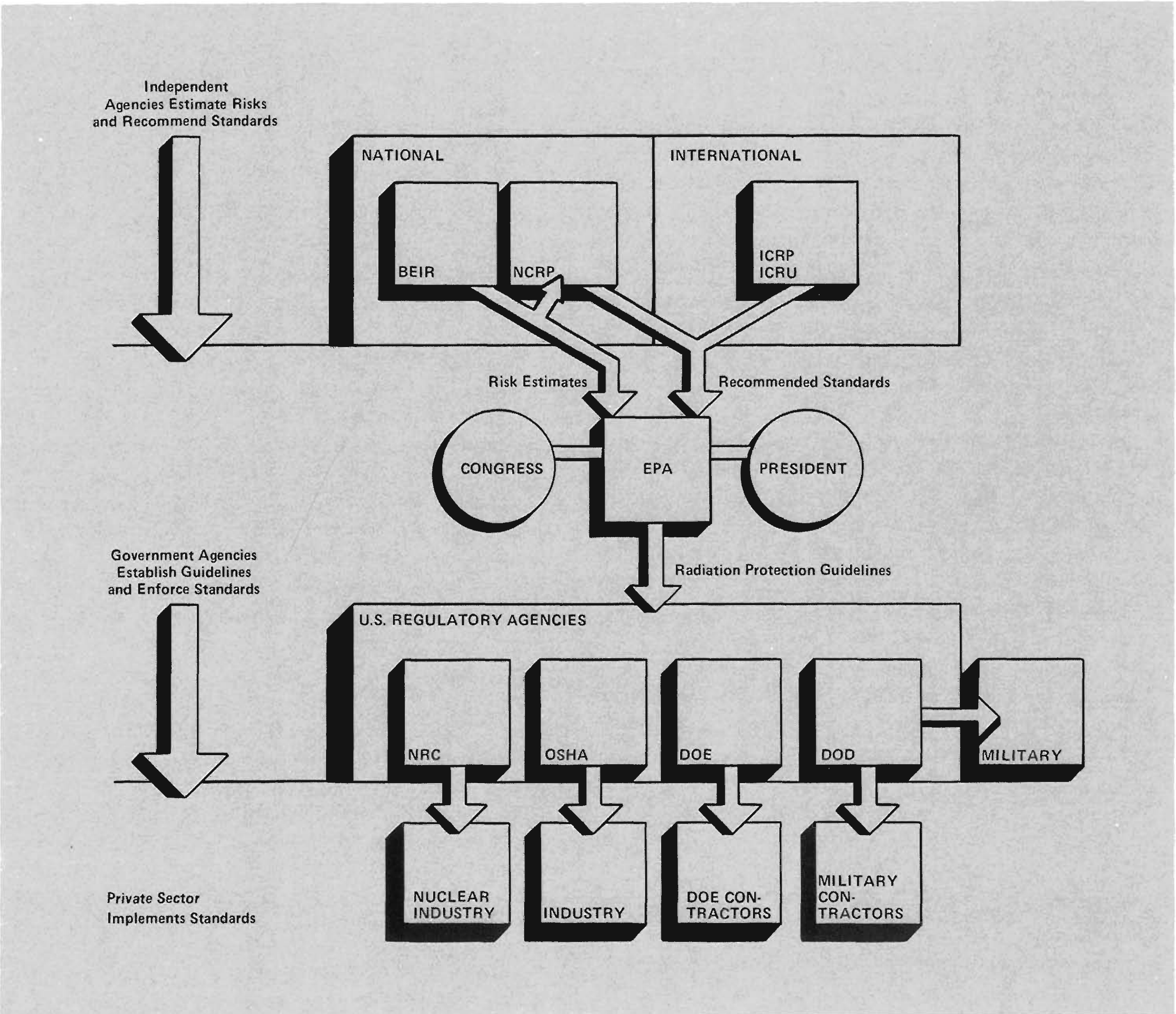


Fig. 3. The Environmental Protection Agency (EPA) is currently the focal point for development of radiation protection standards in the United States, being charged by Executive Order to advise the President and all Federal agencies on radiation matters affecting health. Other agencies involved include BEIR, the Committee on the Biological Effects of Ionizing Radiations established by the Congressionally chartered National Academy of Sciences; NCRP, the National Council on Radiation Protection and Measurements chartered by Congress; ICRP, the International Commission on Radiological Protection; ICRU, the International Commission on Radiation Units and Measurements; NRC, the Nuclear Regulatory Commission; OSHA, the Occupational Safety and Health Administration; DOE, the Department of Energy; and DOD, the Department of Defense.

recovery of genetic damage. Accordingly the Committee recommended a standard for occupational exposure of 5 rems per year and a standard for the general public of 0.5 rem per year. In recognition of the essentially linear relationship between dose and genetic damage down to zero dose, the Committee discarded the idea of a threshold dose and proposed a principle called "as low as practicable" or, in recent times, "as low as reasonably achievable." This principle states that radiation exposure must be avoided if unnecessary and should be kept as far below the standard as possible in light of social and economic considerations. Thus, present radiation standards consist of two parts: the exposure limit that is not to be exceeded, and the instruction to keep the actual exposure as low as reasonably achievable.

Acceptance of the no-threshold concept, which implies that any amount of radiation has some chance of causing harm, produces a dilemma about setting standards. One solution, used by both the International Commission on Radiological Protection[23] and the National Council on Radiation Protection and Measurements,[24] is to base standards on the concept of "acceptable risk." Application of the acceptable-risk concept will always be somewhat arbitrary, based as it is on decisions and judgments that take into account the benefits resulting from an activity as well as the risks.

Several points about radiation standards should be mentioned. First, a standard by no means represents a sharp dividing line between safety and disaster. But the tendency of much of the public to so regard a standard often results in concern, and sometimes panic, when even minor accidents occur.

Another point is the concern that standards may be set on the basis of ability to detect so that improved instrument sensitivity leads to lowered standards matching the new level of detection. However, the as-low-as-practicable regulations of the Nuclear Regulatory Commission for the general public are set at a level where direct measurement is not possible. Instead, proof of compliance is provided by calculations of radionuclide dispersion through the environment.

Finally, the standards recommended by the National Council on Radiation Protection and Measurements have no force in law and must be translated into legislated guidelines and standards by a number of Federal and state agencies (Fig. 3). Most importantly, the Environmental Protection Agency sets standards for all Federal agencies and the Nuclear Regulatory Commission issues regulations that are binding on all its licensees, that is, the nuclear industry.

An example of cooperative interaction between the groups that recommend, legislate, and administer the standards is their solution in 1956 to the problem of occasional occupational exposures above the 5-rems-per-year limit. Various averaging schemes were rejected by the lawyers and regulators who would be required to deal with such schemes. However, discussions among the groups led to the concept of age proration whereby a worker's cumulative exposure is related to his age N and is limited quantitatively by $5(N - 18)$ rems. Within this cumulative limit, Federal guidelines permit doses up to 3 rems per quarter or 12 rems per year. These guidelines allow a certain flexibility in the assignment of occupational exposures. For example, a worker's previous exposure history may permit performance during a year of several tasks requiring doses close to the quarterly limit of 3 rems. It should be noted that an Environmental Protection Agency survey showed that in 1975 99% of all radiation workers surveyed received an annual dose of less than 2.5 rems, and 0.15% a dose exceeding 5 rems.[25]

In January 1981 the Environmental Protection Agency proposed new guidelines for occupational exposures.[25] Included are changes in the requirements for the small number of workers who regularly receive large doses, recommendations for injested or inhaled radionuclides, weighting factors for nonuniform exposures of the body, and several alternative recommendations concerning pregnant women and exposures of the fetus. These proposals are currently under debate, but their passage appears uncertain. It is felt by many that the proposed guidelines pose technical difficulties and will not achieve significant reductions in actual occupational exposures.

Conclusions

The controversy over the hazards of low-level radiation is based on our inability to measure the risks directly. As epidemiological studies evolve that better eliminate confounding factors, more accurate risk estimates will be possible. In the meantime, standards are set by balancing risk estimates based on the best current scientific data against social and economic considerations.

The controversy will surely continue until definitive evidence for the effects of low-level radiation can be given, probably by unraveling the mysteries surrounding cancer and its causes. ■

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The View from SAN DIEGO

INTERVIEW BY BARB MULKIN

HAROLD AGNEW SPEAKS OUT

Harold Agnew is famous for his casual dress—corduroy pants and jacket and Southwestern bolo tie. Today, having joined the ranks of corporation executives, the former Los Alamos director presides over an interview with Los Alamos Science *in different attire. His corner office, as president of General Atomic Company in La Jolla, California, commands an impressive view of palm trees, lush foliage, and bright flower beds. The massive desk groans under a load of paper, as did his more austere working quarters in Los Alamos, and he is not loath to admit that the paper mill still gives him fits.*

Agnew retired as director of Los Alamos Scientific Laboratory in March 1979, after nine years at the helm. He had come to Los Alamos in the spring of 1943 as a junior physicist after working with Enrico Fermi on the first successful fission chain reaction at the University of Chicago. He has been associated with the nation's nuclear programs since 1942, when he joined the Metallurgical Laboratory of the Army Manhattan Engineer District. Agnew left Los Alamos in 1946 to earn his doctorate in physics under Fermi at Chicago. He returned in 1949 and rose steadily through the ranks until he became director.

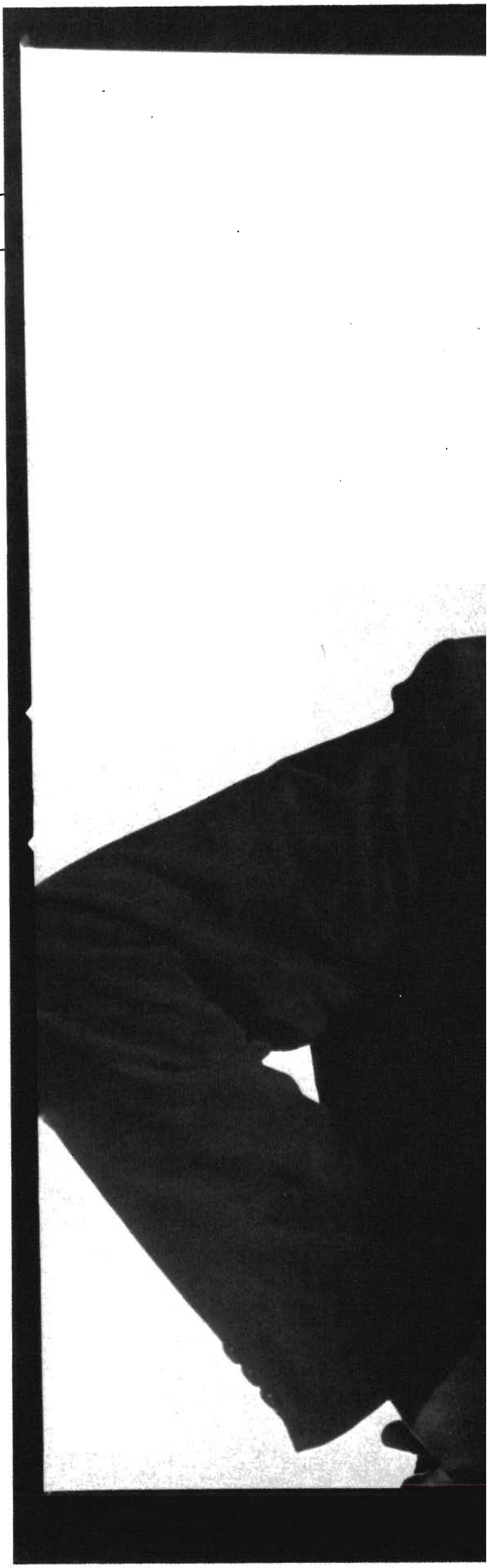
Under Agnew's tutelage, the Laboratory, founded in wartime expediency and continued as the nation's prime weapons research and development facility, became a highly diversified, multiprogram laboratory receiving funding from several Federal agencies. Agnew supervised an enormous growth period, pushed to maintain the Laboratory's academic excellence, restored its leadership as designer-supplier of all major strategic nuclear weapons systems, and redirected it to an energy-oriented mandate that complements and supports the continuing weapons research.

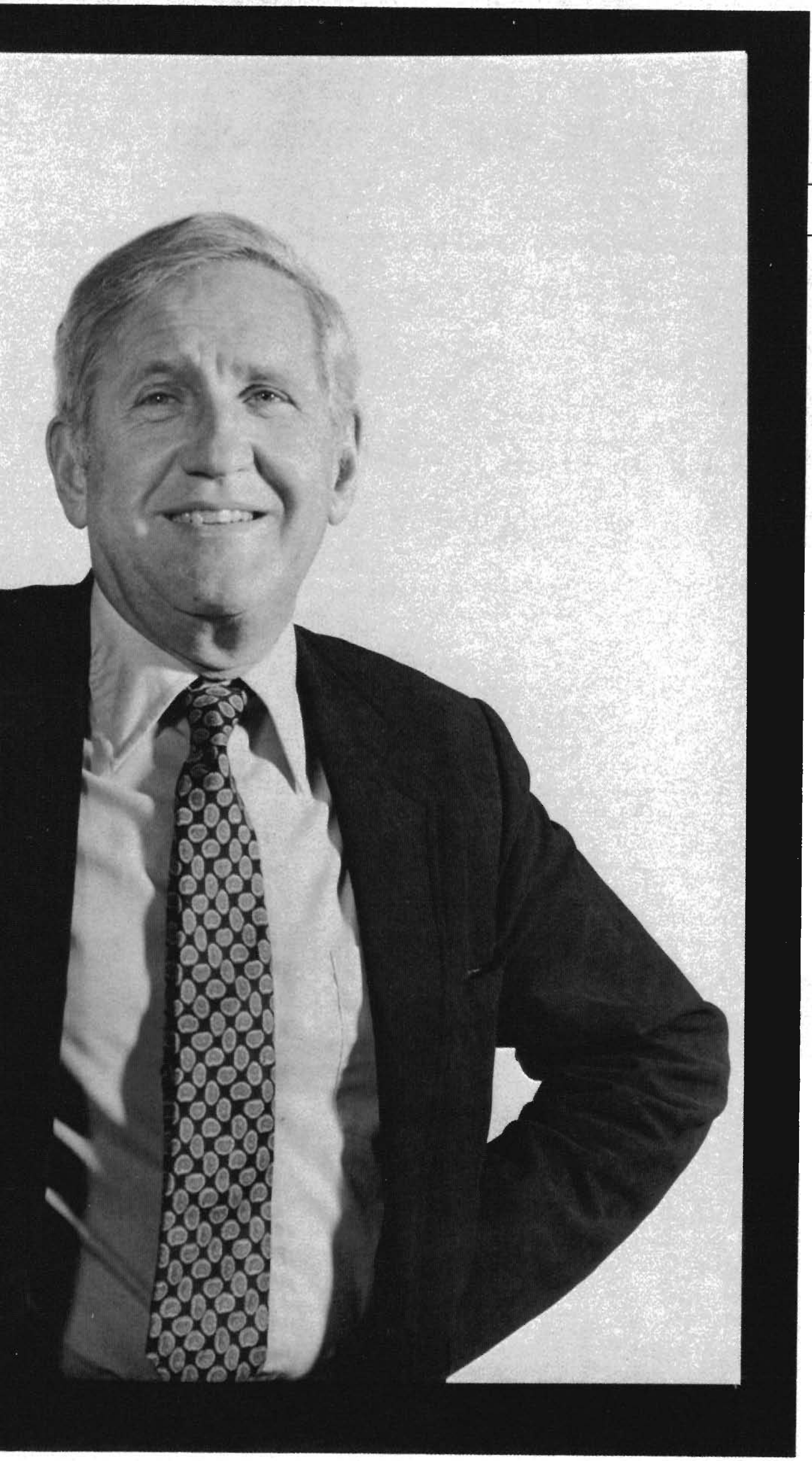
Agnew's efforts did not go unrecognized. On specific matters he served as an adviser to two United States presidents. He was the recipient of the Fermi Award, the highest given by the Department of Energy, and the E. O. Lawrence and National Aeronautic and Space Administration Awards for his contributions to the nation's nuclear programs and to NASA's space epic, the Apollo Program.

An outspoken proponent of a strong deterrent for defense and the benefits of nuclear power, he now heads the only United States company with an operating high-temperature gas-cooled reactor, the Public Service Company of Colorado's Fort St. Vrain plant at Platteville, Colorado.

Still an admitted hawk, Agnew now pushes just as vociferously for acceptance of the new gas-cooled reactor technology, and in inimitable style holds forth on this and other subjects with wit, incisiveness, and clarity.

You leave an interview like this one with the feeling that whenever Harold Agnew retires, it will be to New Mexico, where, in whatever capacity, he will continue to speak out, as he would put it, on a "whole gaggle of issues."





The View from SAN DIEGO

INTERVIEW

I would urge, though, that if at all possible, the national lab people get out into industry for a year or two, so they can appreciate what it's like on the other side of the fence.

SCIENCE: *You retired from the Laboratory, but obviously, you haven't retired. Why did you leave Los Alamos?*

AGNEW: I was there 35 years and director from 1970 to 1979. Nobody should run the course that long. A director doesn't remain vigorous beyond 10 years—after that you can't do what you should. You can't rattle cages. Also, there was the matter of remuneration, although I think the University of California has done something about that since I left.

SCIENCE: *What problems did you face in your new position?*

AGNEW: The difficult part, really, is coming into a place and not knowing the people. You don't know who the con artists are, who are the real pros. I sort of grew up in Los Alamos. I started as a technician, and all of us knew where the strengths were, whom to believe, who were very brilliant but sometimes got off the track. In a new place, you don't really know whom you can rely on, and it takes a while to learn.

SCIENCE: *What are the major differences between running a national laboratory and a commercial company; are there advantages of one over the other?*

AGNEW: One thing that's really quite different is that as one of the national labs, you're "part of the family" while industry is not. We may have had problems in the national laboratories with budget cuts and so forth, but you find that private industry is the first to get the axe when it comes to Federal funding—you really are not part of the family, and so you're viewed quite differently. The feeling is clear that government can get more from the labs because the labs don't get a fee, but the labs get their buildings free, their land free, whereas in private industry, the company has to provide everything. Also, the labs, in the old days, at least, had great freedom to use their funds to start new ideas. We could take money from one project and put it into another. This great flexibility was due to the enlightened management of defense programs by the Division of Military Applications—they understood the situation. In industry, you simply can't do this; you must budget in advance, and although you have a reserve for contingencies, it is nowhere near the amount of money that I had at Los Alamos. Also, in industry you don't have the flexibility of personnel—that is an ideal situation. I think these are the reasons we did so well at Los Alamos over the years.

SCIENCE: *Do you think this situation has changed? Had it in fact started to change before you left the Laboratory?*

AGNEW: It is harder now, and in fact it got harder for me because of the attitude in Washington (D.C.) and the increasing regulations. Maybe the whole thing will be easier again, under the Reagan Administration. I think Don's (Laboratory Director Donald Kerr) experience in Washington before he became director will help him to get things done. I believe he's just what Los Alamos needed.

SCIENCE: *Do you feel that as president of General Atomic Company you have less influence, especially on policy, than you had as director of Los Alamos?*

AGNEW: Well, certainly I think we're viewed as more suspect, presumably because there's something in it for us, some sort of material gain. It's clearly known that the people in the national labs aren't going to make something out of it for themselves. I would urge, though, that if at all possible, the national lab people get out into industry for a year or two, so they can appreciate what it's like on the other side of the fence. Also, I think this would promote cooperation of government and industry on the international scene. Look how the Japanese work—government is a partner with industry in Japan. But I experienced a real adversary relationship between our own government and American industry, especially under the Carter Administration. I hope this changes under Reagan, but there are many problems. There are so many antitrust laws, conflict of interest laws, that it will be very difficult to attain a partnership status.

SCIENCE: *Did you feel when you were at Los Alamos that you had a direct influence on national policy, especially the defense posture?*

AGNEW: No question about it. I met with President Carter for almost two hours on the (Comprehensive) Test Ban Treaty, through (Energy Secretary) Schlesinger's intervention, together with Livermore's Roger Batzel. We influenced Carter with facts so that he did not introduce the CTB, which we subsequently learned he had planned to do. There's no question in my mind that Roger and I turned Carter around because we incurred so many enemies from the other side! It was obvious we had had an impact.

SCIENCE: *Were there other incidents such as this one?*

AGNEW: Yes. I had a hand with President Kennedy in the permissive action link. This was the matter of electronic locks on our warheads so that if they fell into the wrong hands they could not be used.

is! I think Reagan is a very prudent individual, and so are many of those associated with him. I feel the defense policy will not be imprudent or provocative, but I think it will satisfy, to some degree, those who are much to the right.

SCIENCE: *Do you know Mr. Reagan?*

AGNEW: No, but although I've never been

the core, you can still cool the system with air. We're finding another thing that's extremely important—that radiation exposure to plant personnel in water systems is becoming a nightmare.

SCIENCE: *The situation on rems per man year is better in gas-cooled systems?*

AGNEW: Yes. In the Fort St. Vrain gas-

SCIENCE: *Other people could not launch them?*

AGNEW: Oh, they could launch them, but they couldn't fire them. Some of us realized that it was imperative that we install these locks, but the military fought the issue.

SCIENCE: *Why?*

AGNEW: Because they wanted full control over the warheads. At that time they considered the concept an insult to their integrity and felt that it would deny personal control to them, but it was a matter of national security, and it had to be done. We instituted the program, and after I talked to Kennedy, I think my influence was felt. There again, that's an advantage of being associated with a national lab. Everyone knows you have nothing personal to gain, so you have good credibility.

SCIENCE: *What kind of incident was the permissive action link designed to avert?*

AGNEW: Well, suppose two countries set to, and we had our weapons in both countries. They could take them over, but if they did, they couldn't fire them. That's the type of thing we were trying to guard against.

SCIENCE: *You've been at General Atomic for more than two years. From this remove, do you see the role of the national laboratories changing?*

AGNEW: No. I think they are still centers of excellence, and I think the biggest fault of Carter's Administration was that they never really appreciated the strengths and contributions of the national labs. They didn't recognize what the national labs could do, say, for the energy program. They fumbled and bumbled and brought in other organizations to do the job—I won't call them "beltway bandits," but fly-by-night outfits—when in fact the government could have used the labs at much less expense and with much better results, much better continuity.

SCIENCE: *What do you think will happen to our defense programs under Reagan? There are those who think he's a hawk. . .*

AGNEW: I'm a lot more of a hawk than he

associated with him, I think he's decisive, that he makes good decisions the way President Kennedy did, and that such decisions are based on good, solid advice given to him by professionals in the field.

SCIENCE: *What about the future of nuclear power under the Reagan Administration, especially after Three Mile Island. The problem was contained, but hasn't the biggest effect been the enormous public impact?*

AGNEW: Yes, and it really has been enormous. It gave fuel to those who were, for whatever reason, opposed to nuclear energy, but I think in the long run, the effect has been to settle the public safety issue. As Edward Teller says, he was the only one who was injured—he had a heart attack! I think Three Mile Island showed one thing, and I was heartened because this is something I've been preaching ever since I came to General Atomic: the issue is not public safety, for that has been settled. The issue is this: if we are going to maintain public utilities, using money invested by private citizens, then we're going to have to do more to protect the equity of these individuals. *That's* what Three Mile Island showed—that you can lose equity in a plant, and that it is more at risk in water-cooled reactors than it need be. It's been shown clearly that gas-cooled reactors have better thermodynamic efficiency, better uranium fuel utilization, and in fact, that they are much more *forgiving!*

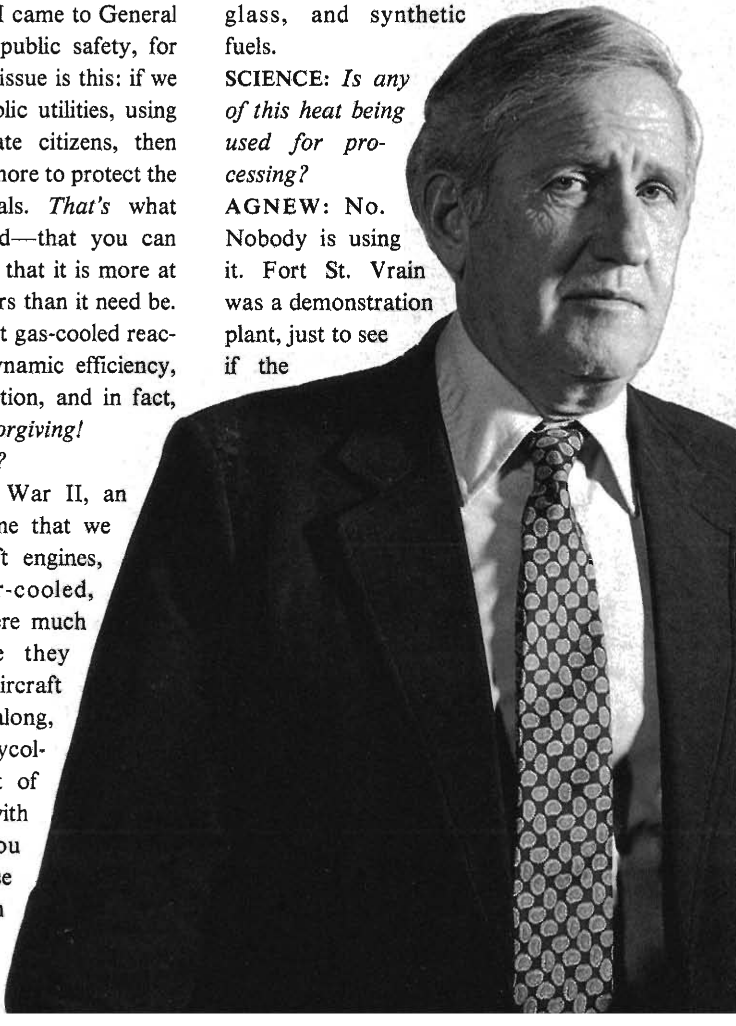
SCIENCE: *More forgiving?*

AGNEW: Yes, In World War II, an English friend reminded me that we had two kinds of aircraft engines, glycol-cooled and air-cooled, and that the air-cooled were much more forgiving because they could take abuse—antiaircraft fire, shrapnel—and limp along, but if you cut a line on a glycol-cooled engine, you're out of luck! It's the same way with gas-cooled reactors. You can't lose cooling because even if you lose pressure in

cooled reactor, the coolant, helium, does not become radioactive. In our last refueling there was essentially no exposure to people. We took out a circulator that had been in the reactor for over a billion kilowatt hours and within one week, we could do hands-on maintenance. I think that's something that would relieve the public's worries about nuclear power. We cannot continue to ignore gas-cooled systems—and they have been ignored. The Japanese are particularly interested in them for process heat for industry. General Atomic has a license arrangement with Japan right now for them to use high-temperature gas-cooled reactors for such heat. Our Fort St. Vrain reactor has 1350°F outlet helium temperatures available right now, and this is going to be awfully important for industry in processing cement, glass, and synthetic fuels.

SCIENCE: *Is any of this heat being used for processing?*

AGNEW: No. Nobody is using it. Fort St. Vrain was a demonstration plant, just to see if the



technology would work. We're getting a lot of black names because the plant is running consistently only at about 70 per cent rated power, but when you consider that it *was* a demonstration plant, and that the Germans started their pebble-bed (gas-cooled) reactor two years before General Atomic started Fort St. Vrain, and they still don't expect it to become operational for three or four more years, then St. Vrain is quite remarkable. It's already delivered a couple of billion kilowatt hours.

SCIENCE: *With this record, then, why is there an apparent bias toward water systems?*

AGNEW: Because the utilities are paranoid on this subject. They and the manufacturers have a pipeline of orders for water-cooled systems and just won't face up to the fact that we ought to bring in this new product line. It's like the car dealer who has 1980 models left when the 1981s come in. It's pretty difficult to sell the old models while you're touting the new ones as the super-dupers of the world, so they won't put them on the lot. There has been little improvement in water-reactor technology since the first water-cooled reactor climbed out of the ocean from the submarine program, like an amphibian. The bias toward water reactors is there because of the tooling, the commitment on orders, the worry of rescrambling to get in line on new technology. I don't know how to change this, except to say that the water-cooled reactors have been great, but have served their purpose, and now it's time to go on.

SCIENCE: *Do you think the new technology will make nuclear power more acceptable?*

AGNEW: We have to bring in the new technology if we are to make it acceptable.

SCIENCE: *Did you feel this strongly about gas-cooled reactors when you were at Los Alamos?*

AGNEW: I didn't know anything about them, then. Nobody knows much about them really, and that's the biggest part of the problem.

SCIENCE: *Why is that?*

AGNEW: I guess because we have the only reactor and General Atomic is a very small company.

SCIENCE: *Didn't General Atomic at one time have orders for about 10 of these reactors?*

AGNEW: Yes, but in order to get going, you have to estimate you'll lose money on the first orders because of paying for the tooling and so on that goes with the first-of-a-kind cost. Two things happened: the market fell back and some orders were cancelled. The company knew it could not break even financially on the remaining orders, so it cancelled them.

A decision by a nation to develop nuclear weapons will be independent of its decision to reprocess spent nuclear fuel.

SCIENCE: *Then the orders were not cancelled out-of-hand by the utilities?*

AGNEW: Not quite: the utilities cancelled six orders and with only four in hand, GA thought it prudent to cancel these. In hindsight, they should have tried to renegotiate the contract for one. It cost the company a boodle. In fact, it cost them close to a billion dollars! Imagine that, a billion dollars! The owners won't put any more money into this, and I don't blame them, so that's where we are right now.

SCIENCE: *General Atomic owns half of the nuclear fuel processing plant at Barnwell, South Carolina?*

AGNEW: The Barnwell plant is another example of a vacillating government policy that paralyzed the nuclear industry. Because General Atomic was led to believe that reprocessing, with proper safeguards, would be encouraged in the private sector, GA and Allied Safeguards spent close to half a billion dollars to build the Barnwell plant—a beautiful plant—and then Mr. Carter decided it was not in the interest of national policy to reprocess and so refused to license Barnwell.

SCIENCE: *Well, the plant's still there. What do you see for it in the future, under Reagan?*

AGNEW: I don't know, but to me it seems criminal not to reprocess and consolidate the spent reactor fuel. I think by not doing so, we are being absolutely hypocritical! The rest of the world certainly sees us this way because, of course, the United States is reprocessing for the military at another facility. Presumably we're not going to reprocess commercial spent fuel so that we can set an example for the rest of the world. They consider us hypocritical, and they've laughed at us and have gone their own way and are reprocessing. A recent *Financial Times* article points out that the United Kingdom has reprocessed spent fuel from its Magnox reactors and has supplied plutonium to our weapons program; in return they get tritium and enriched uranium for their weapons program. A decision by a nation to develop nuclear weapons will be independent of its decision to reprocess spent nuclear fuel.

SCIENCE: *What has this situation done to us in terms of slipping technology and loss of leadership?*

AGNEW: We've lost the technical and moral leadership anyway, and the rest of the world is doing what is prudent for them. It amazes me that people talk about problems with storing long-lived transuranics, when it fact, if you reprocess, you recover them, plus the uranium, which you can then put back into the reactor. The rest of the materials, after a couple of hundred years, have the same level of radioactivity as the uranium you took out of the earth in the first place. I've talked to (California) Governor Brown about this, but he simply doesn't understand, any more than most of the public; his eyes just glaze over.

SCIENCE: *Is anyone with a scientific or technical background advising Brown on such matters?*

AGNEW: I don't know. The fact is there are some people who have certain beliefs—I won't call them religious beliefs—but still, these beliefs are like a religion.

SCIENCE: *Fanatical beliefs, perhaps?*

AGNEW: Exactly. Fanatical. I remember during World War II when I was with Luis Alvarez, we had an associate who read the Bible all the time. Luis used to attack this guy, in a very friendly way, and he would say: "How can you believe in this stuff? How can you believe in miracles?" We used to have long discussions about the miracles, like the Red Sea dividing. Finally, this man, who was a scientist, said: "Luis, either you believe or you don't believe. It's that simple. It has nothing to do with anything concrete—you just believe!"

SCIENCE: *You're saying that belief has nothing to do with facts?*

AGNEW: Right, and I draw this analogy to the antinuclear people. What they believe has nothing to do with the facts. I used to argue with Fermi, who was always very interested in cosmology—the big bang theory and all that. I would tell him I was mystified because when I looked at this theory, I couldn't figure out where all the original *dirt* came from. He would laugh, and we would talk about protons, and I would say: "O.K. Tell me where the protons came from, then." Fermi couldn't, of course. Nevertheless, they had to come from somewhere. The question is, where? I don't care if you start with energy or with mass, but you have to start with something, or it simply can't be. Yet it is there, and it really can't be—guess I'm getting old!

SCIENCE: *Do you think Governor Brown's views on nuclear power are influenced, perhaps, by the threat of earthquakes in California?*

AGNEW: Well, I don't know, but I do know that if we had gas-cooled reactors instead of water systems, we would have a lesser problem because we wouldn't be faced with millions of gallons of radioactive coolant spewing out and soaking into the ground.

SCIENCE: *But what if a quake caused gross structural damage to a gas-cooled reactor—wouldn't that be bad?*

AGNEW: You'd ruin the reactor, perhaps, but you wouldn't have the major decontamination problems. You remember in the old Rover* days, we blew up a reactor—we had Roman candles—but then we went out into the desert and picked up the hunks. The decontamination was a piece of cake!

SCIENCE: *The Rover reactor used solid fuel. Is that the key—the fuel—in this case?*

AGNEW: Yes, and the HTGR uses the basic Rover fuel—coated particles—and that's what impresses me. Ted Taylor (a thoughtful critic of our nuclear posture) called a while back and said: "You know, HTGRs are great for use in troubled parts of the world, the Middle East and so on, because you can't get into a mess with them. You can handle the decontamination because of the type of fuel and solid moderator used. And single-phase coolant has tremendous advantages." There's no question that in a catastrophe, like an earthquake or a conventional war, you might lose equity in a plant but you wouldn't make a mess which would be comparable to that from water-cooled systems.

SCIENCE: *Even if this is true isn't it too late to change the public's opinion about nuclear power?*

. . .if we don't bring in new technology, I think the nuclear power industry is doomed to extinction.

. . .in the old Rover days, we blew up a (gas-cooled) reactor—but then we went out into the desert and picked up the hunks. The decontamination was a piece of cake!

**Project Rover was a joint Atomic Energy Commission-National Aeronautic and Space Administration program to design and build a nuclear-powered rocket for interplanetary space missions. The rocket engines were powered by ultra-high-temperature gas-cooled reactors that used fuel particles similar to those in the Public Service Co. of Colorado's Fort St. Vrain plant.*

The View from SAN DIEGO

AGNEW: I think we can. Other countries are preaching that nuclear power is a good thing, but we have been remiss. We must begin explaining to schoolchildren facts about the atom—facts and not emotion. You see, people still don't realize that radiation doesn't creep and crawl!

SCIENCE: *Meanwhile, though, we have our current crop of reactors. Are you suggesting...*

AGNEW: No, we need them still, and they will serve their purpose. I think the things industry has done—put in improved management systems, reporting systems, and components, as well as better trained personnel—has helped the situation. But let's face it, we're still in our infancy in nuclear power. The present generation of reactors has served its purpose, and somewhere in the world the new gas-cooled systems will be adopted. If England and France had had sufficient helium for cooling, they would have stayed with these reactors, but instead, they had carbon dioxide and that is not very good, neutronically or as a coolant, and it also has temperature limitations. They also had metal-clad fuel, which limited temperature. With our Triso-coated fuel particles, the system is very good. You must remember that about a third of the electric generating capability in the United States belongs to the Gas Cooled Reactor Associates,* and they are saying to the government: "Look, we know nuclear power is cheaper, more reliable, and less insulting to the environment, but we need your help. This technology is too expensive today for any one utility company to pursue." These companies have teamed up to push gas-cooled reactors, but they don't want to jeopardize their present systems. However, if we don't bring in new technology, I think the nuclear power industry is doomed to extinction.

SCIENCE: *How many member utilities are there, and how much of the present nuclear generating capability do they have?*

AGNEW: About 30 to 40 companies and they own about 40 per cent of the nuclear generating capacity in the U.S.

SCIENCE: *They have a common bond; do they differ on technological approach?*

AGNEW: No, they don't. Here is another vacillation of the Carter Administration—early in that Administration, the Office of Nuclear Energy said if the utilities were in favor of new technology they should form an association and then they would be included in program management. As soon as they formed the GCRA, the Administrtion zeroed out the budget for the HTGR! We had to go to Congress to get it restored. We had absolutely no support from the government or from the Department of Energy.

SCIENCE: *What do you think Reagan will do about the Department of Energy? Is it too unwieldy?*

AGNEW: Clearly, something has to be done about it. I believe the (synthetic) fuels have to be pulled out, and they will be, under the Synthetic Fuels Act. I can anticipate that someday the defense programs will be pulled, but whether they'll go to the Defense Department or to a separate organization, I don't know. Either way, I believe it's inevitable.

SCIENCE: *If defense programs are pulled, will this be an improvement?*

AGNEW: It's too bad in a way. I think the talent is really in Los Alamos, Sandia, and Livermore (National Laboratories), in the context of what Energy's real job is. The labs understand projects, how to get them on line, on schedule, and within budget. Look at the Los Alamos plutonium facility, and Sandia's electron-beam facility. Or check on what's been done in the weapons program—when they had a commitment for a new weapons system, then by God the system was built because Herm Roser's DOE Albuquerque Operations Office really understands how to run production facilities. But then, this has led to a problem—there's resentment in the rest of DOE at how well managed the defense side has been. As a result, they don't want these people involved, for it would show up the rest of them.

SCIENCE: *If you were in a position to influence the new Administration, what would you do about the Department of Energy?*

AGNEW: I'd focus on special projects. The space program was an outstanding success because it had a specific objective. There are no ultimate goals right now in DOE, no focusing at all, except, perhaps, in the synthetic fuels program, and that may be on the skids.

SCIENCE: *The syn fuels goal must have*

INTERVIEW

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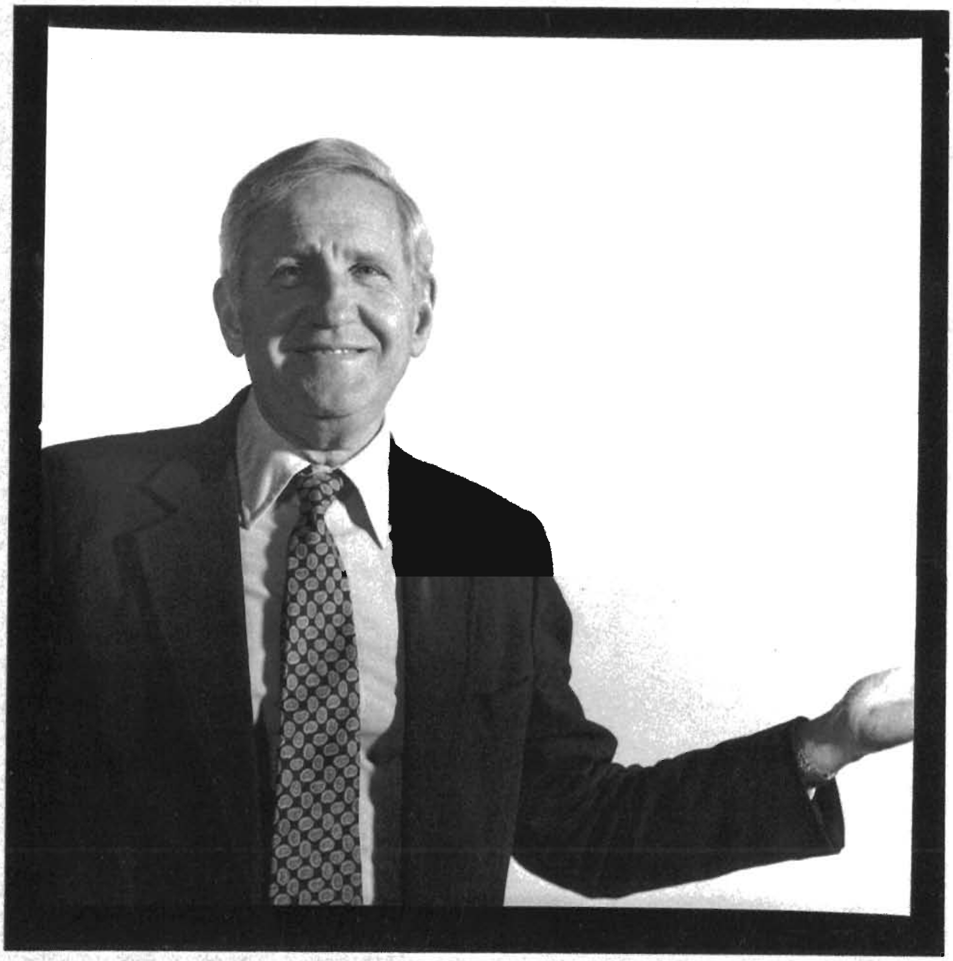
**A consortium of utility companies interested in developing HTGR generating capability.*

been started by the Carter Administration, then?
AGNEW: Yes, but it was really done by the Congress.
SCIENCE: *You've often been quoted as saying that a terrible communications gap exists between those in science—and especially those in nuclear science—and the public. Is this still true?*
AGNEW: We've always had a communications gap, especially in research and development, but it can be wiped out. The Russians are doing it, by making it mandatory that elementary and secondary schools stress a strong background in science, mathematics, and related technologies. It's an absolute requirement; they don't graduate if they can't hack it. Under our system today, everybody passes, no matter what subjects they take. This philosophy is having a tremendous impact, and will have even more of an impact down the road, for the era of learning on the job, of not needing a background in science or technology, is passing rapidly.
SCIENCE: *Do you think this problem extends through the ranks?*
AGNEW: Well, yes, because as we move into more automation with computers and other machines that are almost hands-off, the people running the machines cannot just poke here and there or pound with a hammer to fix things. They will have to have a background in science—physics, chemistry, whatever—to work the machines from the floor.
SCIENCE: *You're comparing the educational philosophy of the United States with that of a totalitarian society. Are you suggesting. . .*
AGNEW: I'm saying it can be done here. If school boards were worth a damn—and most of them aren't—they would insist that the curriculum go this way. I was listening to Buckley (conservative commentator) the other night, and he commented that if Einstein were alive today he couldn't teach in our public schools because he lacks courses in education!

When we were in New Mexico, years ago, my wife Beverly was serving on the State Board of Education and I was serving in the New Mexico State Senate. We tried to get a provision in the state code that if a person had a degree—an advanced degree—that person could teach in the field of that degree without having to be certified in the History of Education, the Psychology of the Child—all of those nonsense courses required to get a teaching certificate. Let's face it, either a person can teach or he or she can't. We've gone the wrong way in our educational system. It's terrible that many qualified people aren't teaching because they refuse to take those ludicrous courses and as a result cannot obtain a certificate.
SCIENCE: *When you left Los Alamos in 1979 you said you had a three-year contract with General Atomic and that you consider Los Alamos home. Are you looking toward a change in the future?*
AGNEW: In a way I am. I do know I don't plan to drop dead working! Bernd Matthias's passing really shook me and Beverly up—hell's bells, he was a contemporary. I want to do some art. That sounds weird, maybe, but I was pretty good at carving. I used to do silver work. I want some time to do some of these things, and Beverly is still improving her tennis—maybe she'll be in the next Miss Clairol Tournament! ■

It's terrible that many qualified people aren't teaching because they refuse to take those ludicrous courses and as a result cannot obtain a certificate.



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