A PRIMER on REACTOR SAFETY ANALYSIS

by Michael G. Stevenson and James F. Jackson

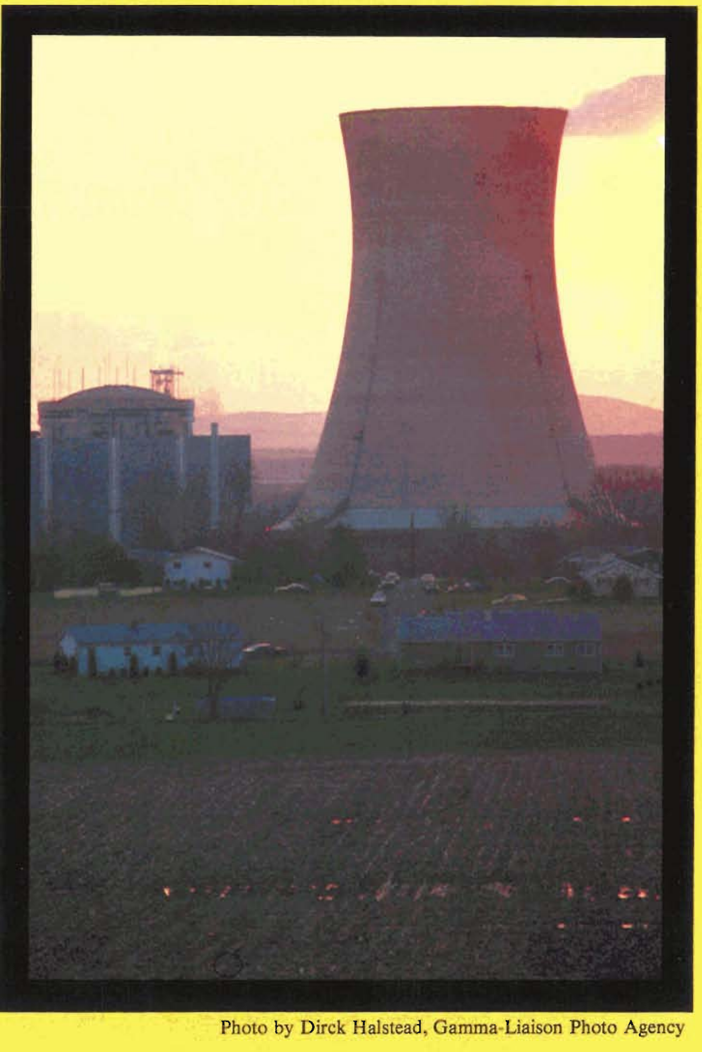

Photo by Dirck Halstead, Gamma-Liaison Photo Agency

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

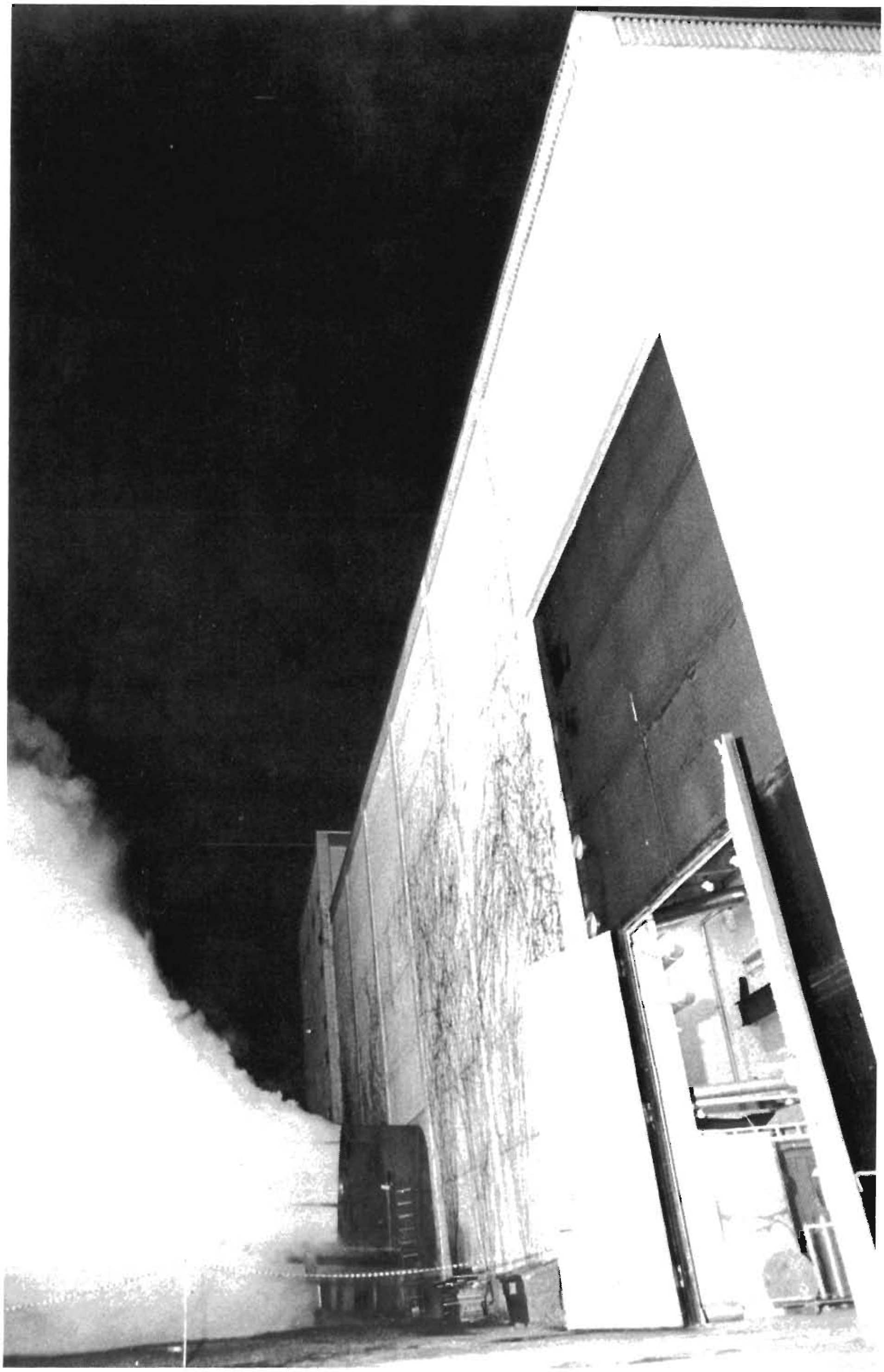

Fig. 1. Originally designed as a small nuclear power plant, the Marviken facility in Sweden has been converted to an experimental facility for studying the ejection of water from a ruptured pipe in a water-cooled reactor. Water in the reactor vessel is heated (with fossil fuel—the facility has no nuclear 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.)

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

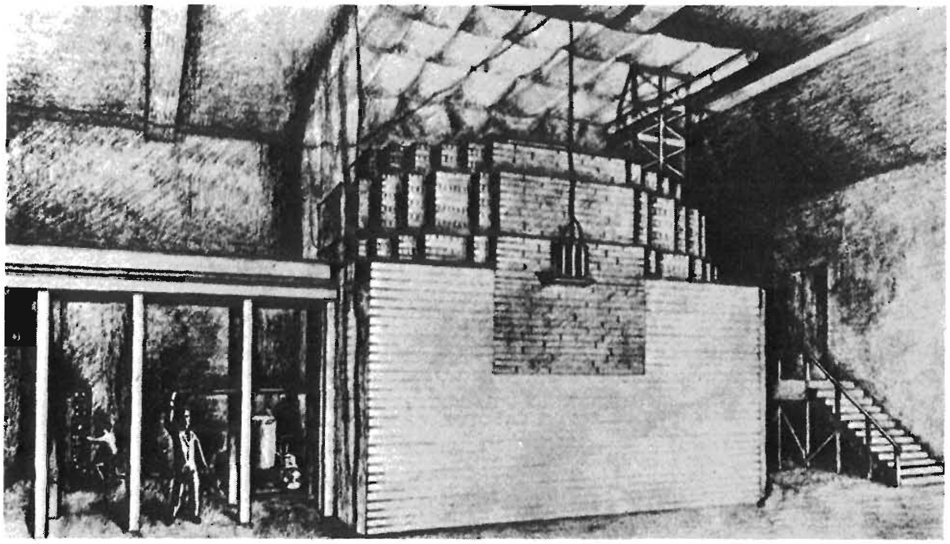

Fig. 2. An artist's sketch of the first nuclear reactor constructed in 1942 in a squash court under the west stands of Stagg Field, University of Chicago. It was made from about 40 tons of natural uranium and 385 tons of graphite. Note the manually operated control rod extending from the side of the "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.

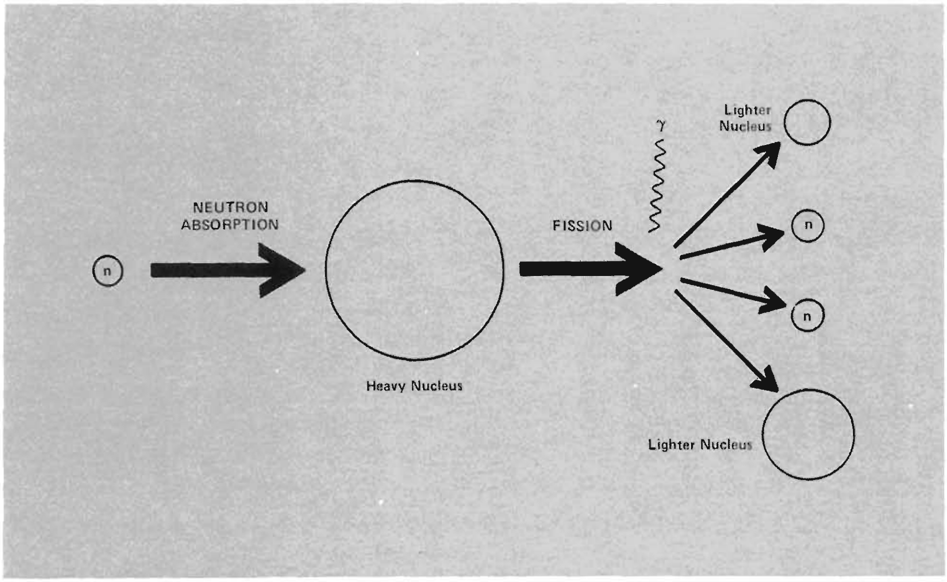

Fig. 3. The fission process. A heavy nucleus, such as uranium-235 or plutonium-239 absorbs a neutron (n) and breaks up into two lighter nuclei, two or sometimes three neutrons, and gamma rays. The lighter nuclei are usually radioactive.

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.*

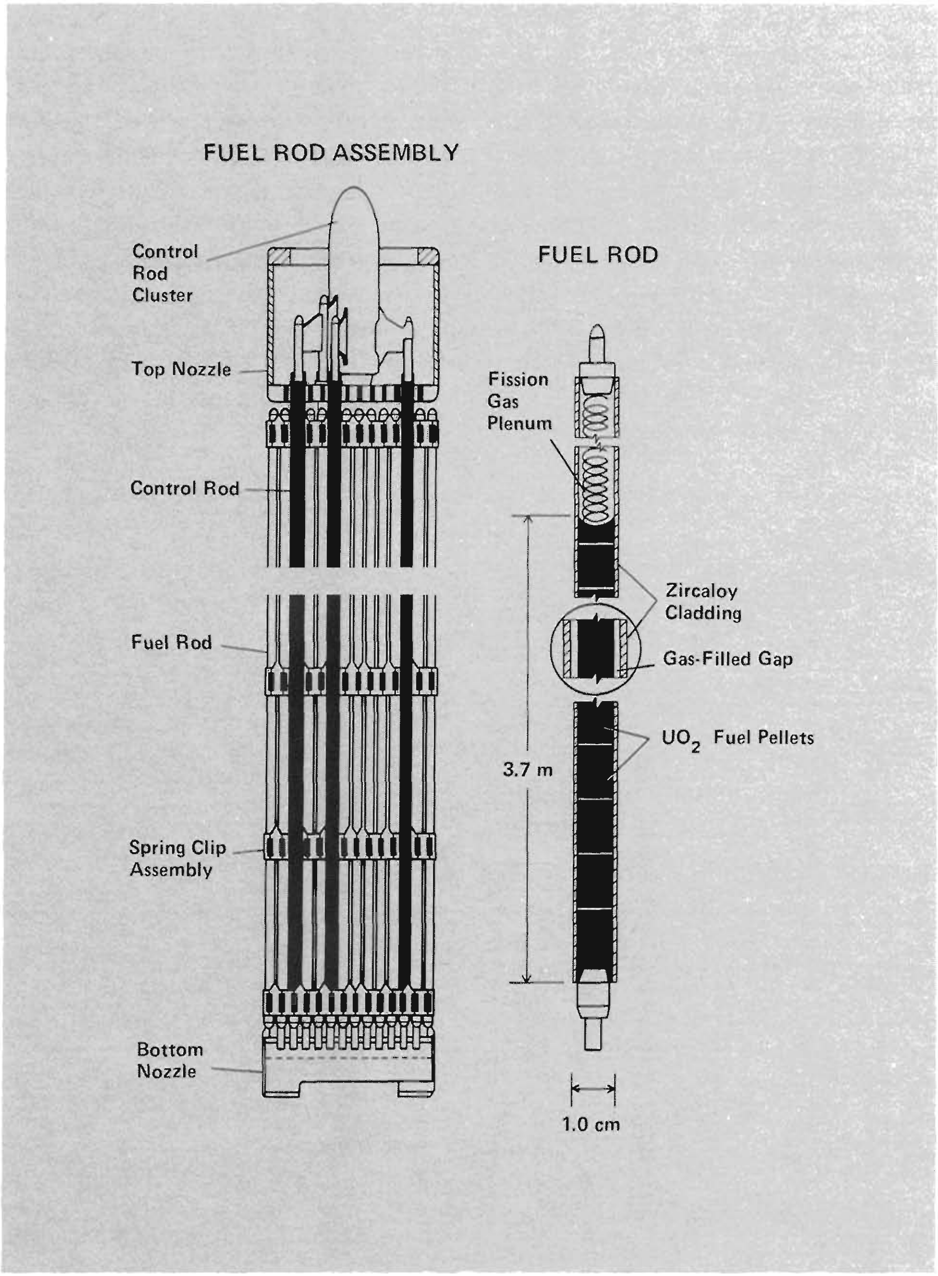

Fig. 4. Fuel rod and fuel-rod assembly for a pressurized-water reactor. Fuel rods are held in a square array by spring clip assemblies and by grid assemblies at the top and bottom. The structure is open permitting flow of coolant both horizontally and vertically. Control-rod guide tubes are interspersed 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_2O) 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 ≅ 1 atmosphere.

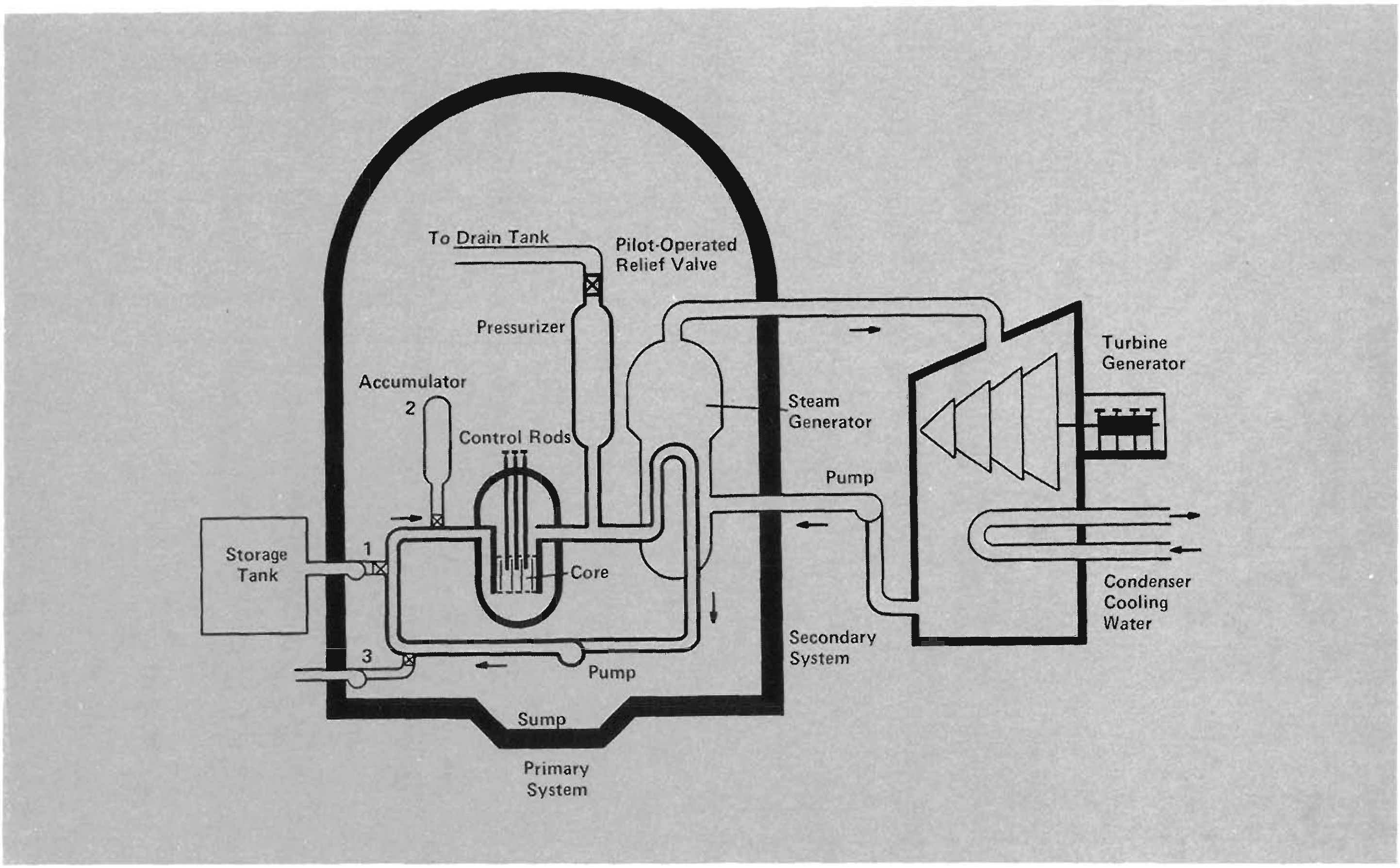

Fig. 5. A pressurized-water reactor showing a primary loop, a secondary loop, and the three subsystems (labeled 1, 2, and 3) of the emergency core-cooling system. In the primary system, water under high pressure (about 150 bars) is pumped by 7500-horsepower centrifugal pumps through a 12-meter-high (40-foot)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

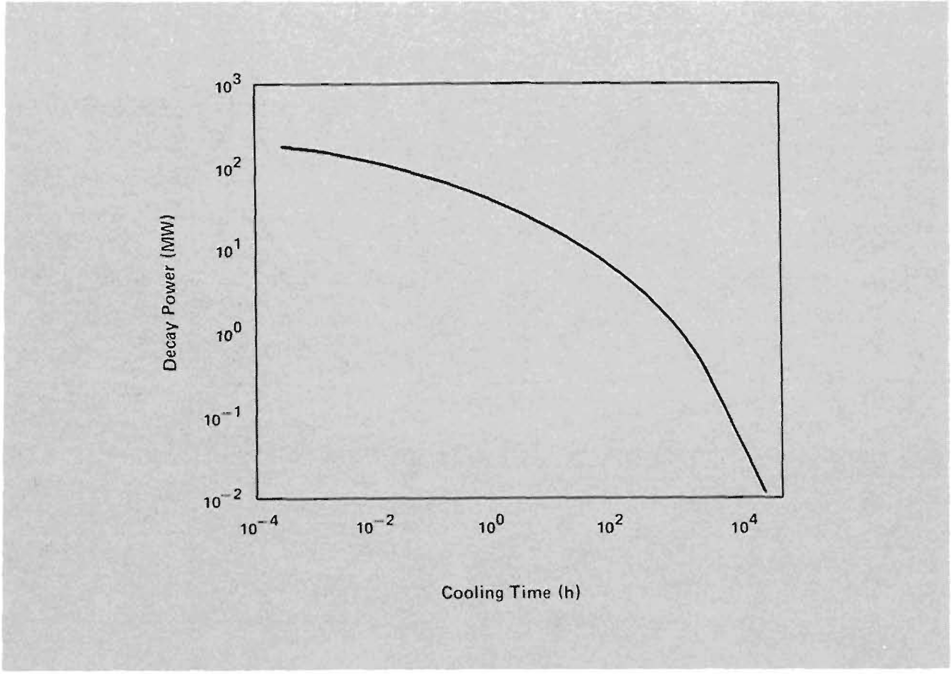

Fig. 7. A log-log plot of decay power as a function of time after reactor scram for the Three Mile Island Unit 2 reactor. This curve was calculated by the Laboratory's Nuclear Data Group. It depends on the reactor's power history and fuel and fission-product inventories and on details of the decay chains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

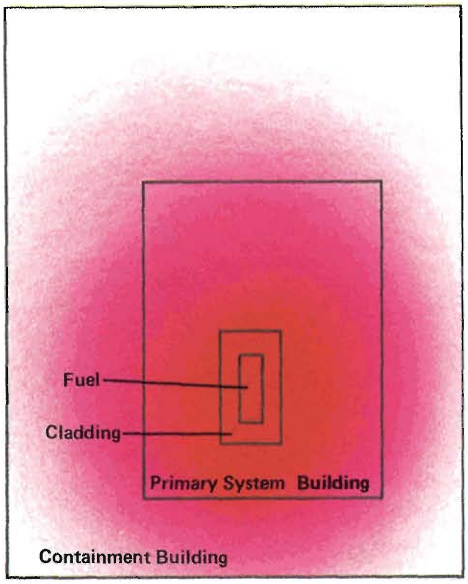

Fig. 8. The four barriers against release of radioactive materials in a pressurized-water reactor.

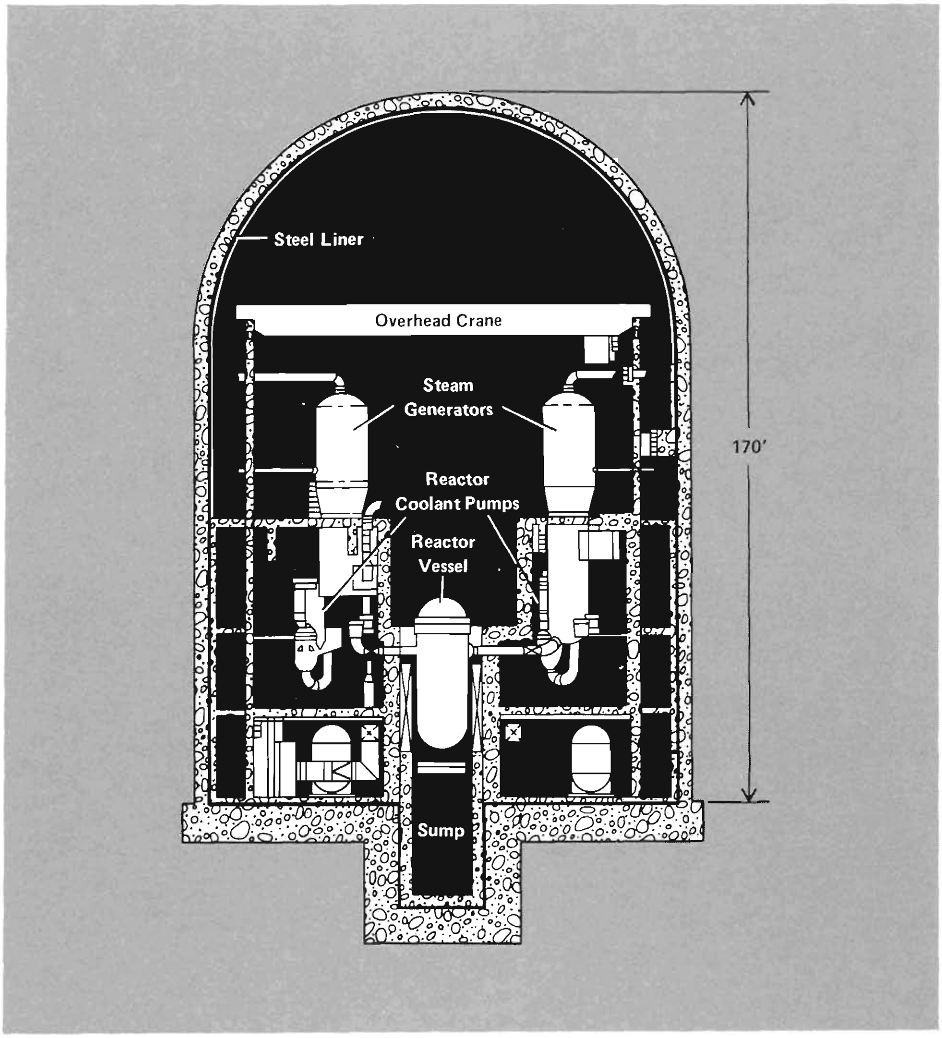

Fig. 9. Cross section of a typical containment building for a pressurized-water reactor. The concrete containment building houses the entire primary system, the pressure control system, ventilation equipment, and part of the emergency core-cooling system. The various components are encased in concrete 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.

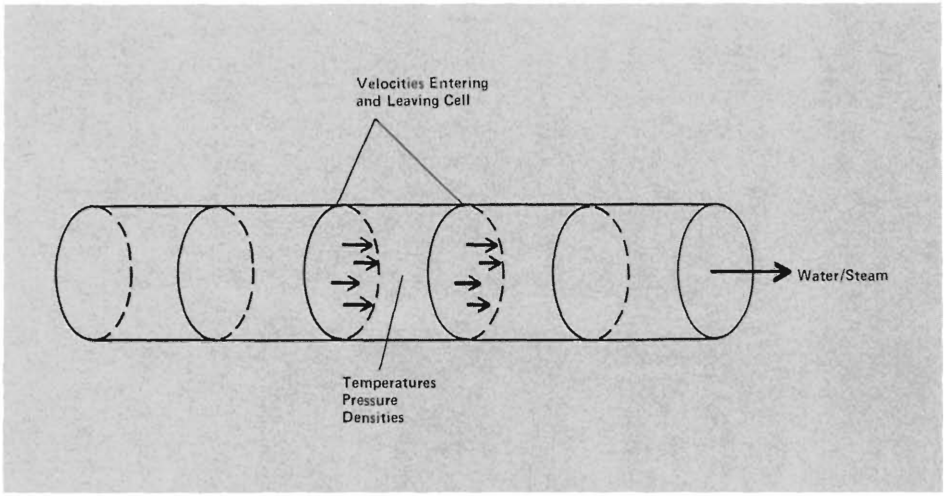

Fig. 10. Division of a coolant pipe into computational cells. Densities, pressures, and temperatures at the center of each cell are computed, as well as the velocities of the steam-water mixtures entering and leaving each cell.

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.

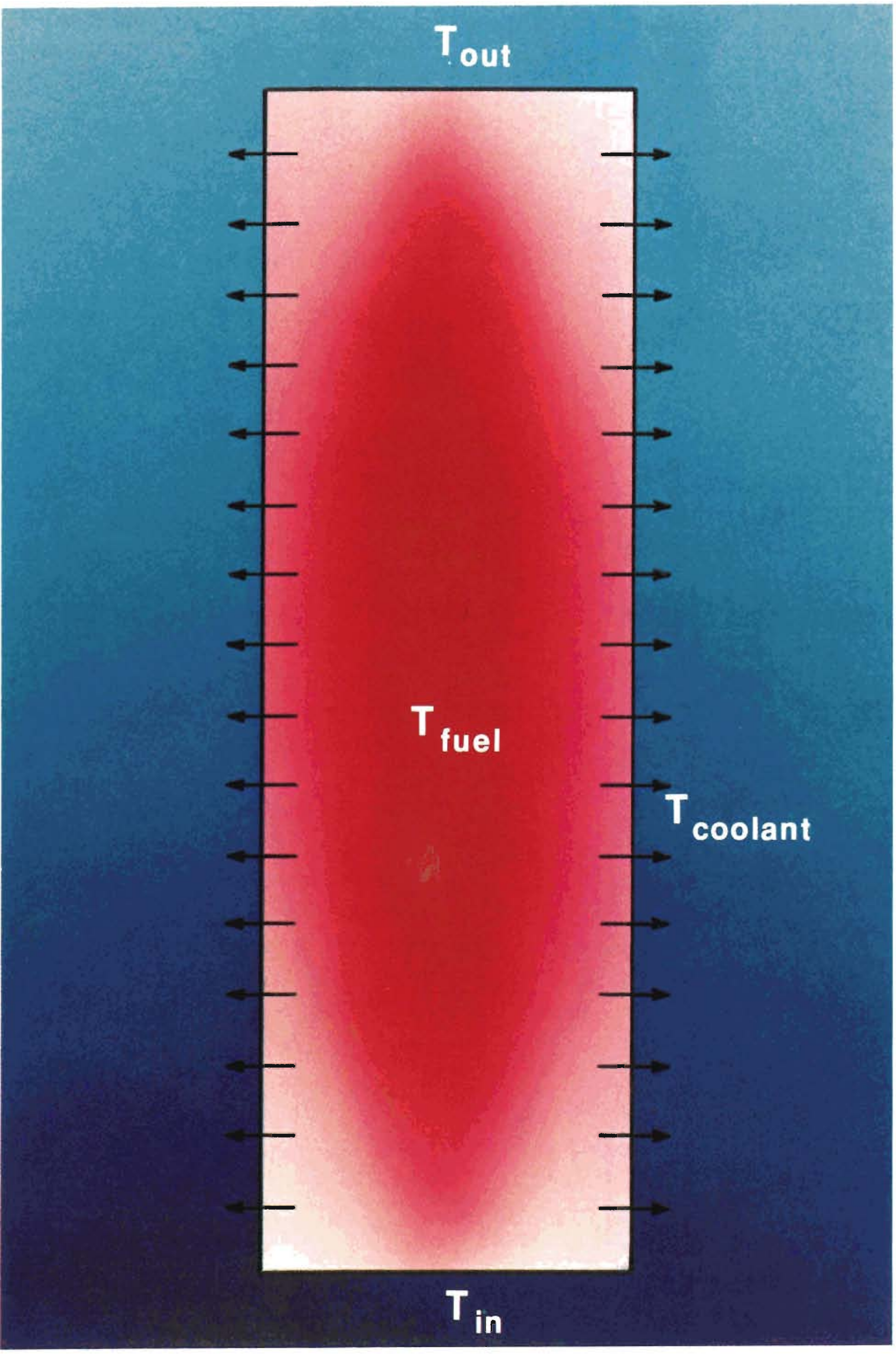

Fig. 11. In a two-cell representation of the core, coolant flowing past the nuclear fuel is heated at the rate $hA(T_{fuel} - T_{coolant})$, where h is the heat-transfer coefficient and A is the surface area of the fuel.

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, $Q_{\text{core}} = Q_{\text{decay}}$. 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, $T_{\text{fuel}} - T_{\text{coolant}}$. Again, energy balances provide equations for Q_{fuel} and Q_{coolant}.

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

and

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

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-

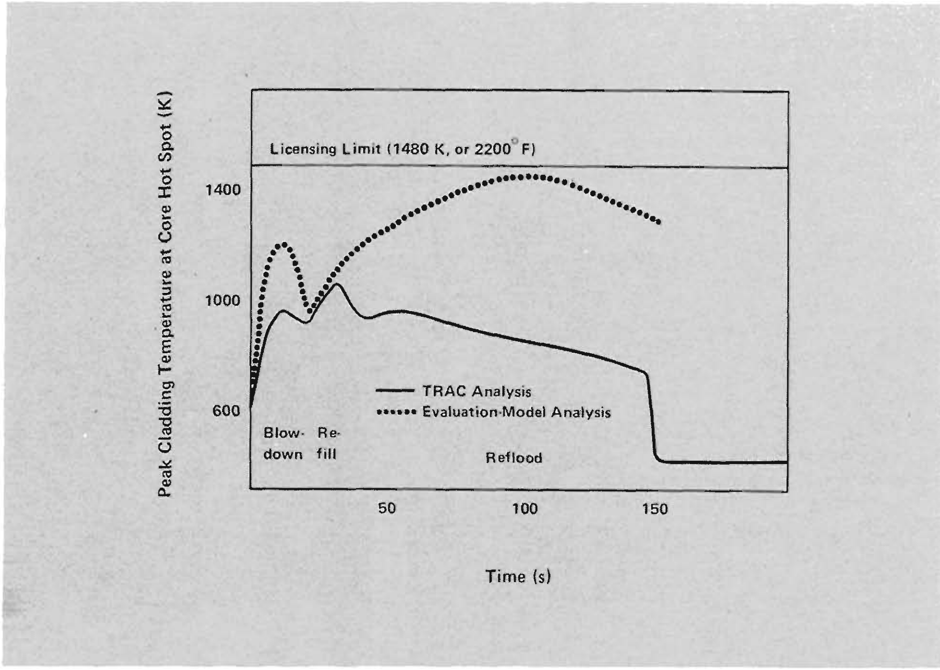

Fig. 12. Cladding temperature histories during a large-break loss-of-coolant accident in a typical four-loop pressurized-water reactor. One history (solid curve) is a TRAC analysis [J. R. Ireland and D. R. Liles, "A TRAC-PD2 Analysis of a Large-Break Loss-of-Coolant Accident in a Reference US PWR," Los 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.

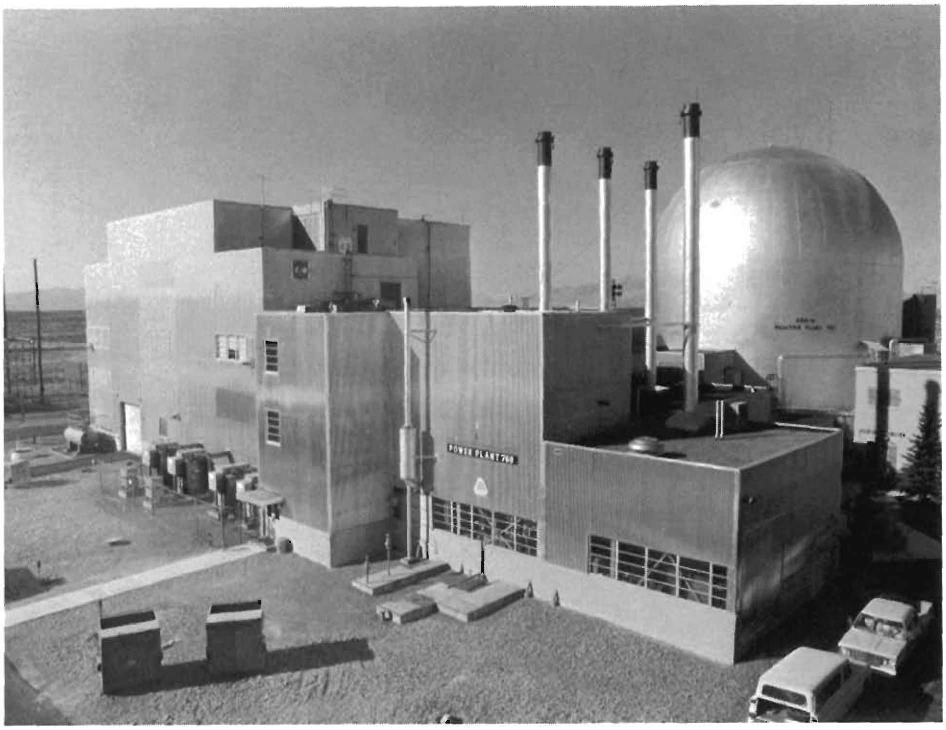

Fig. 13. The EBR-II reactor is located at Idaho National Engineering Laboratory near Idaho Falls, Idaho. This liquid-metal-cooled fast breeder reactor has operated successfully for over 15 years. The reactor produces up to 20 megawatts of electrical energy and has had an excellent history of reliable 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.

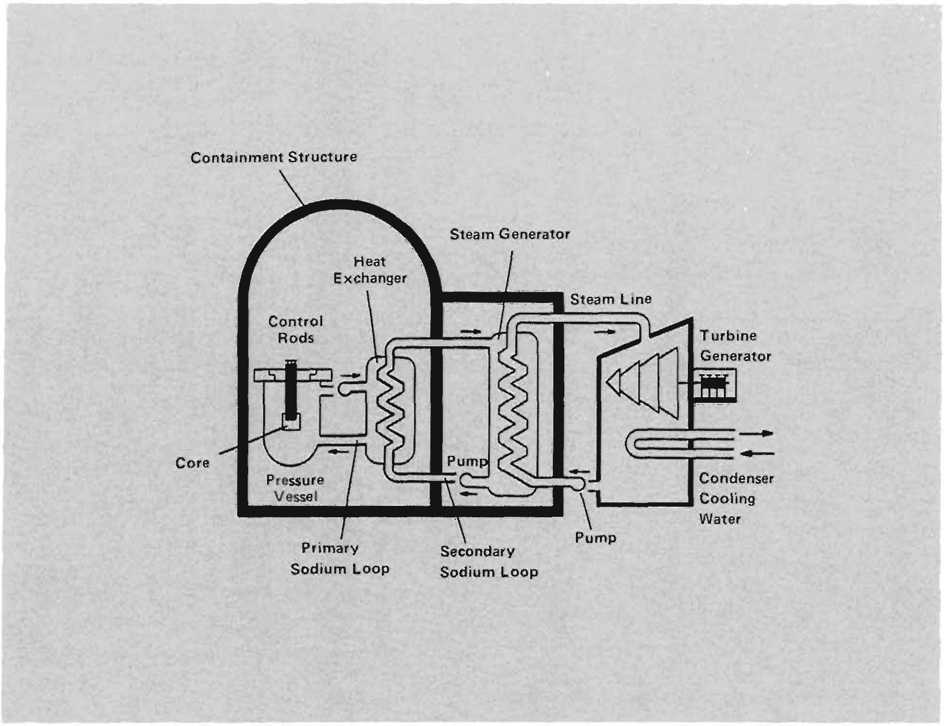

Fig. 14. A loop design for a liquid-metal-cooled fast breeder reactor showing the primary and secondary sodium cooling loops and the steam loop to the turbine-generators. The second sodium loop ensures that no radioactive sodium flows through the steam generators. The primary system is at near atmospheric 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

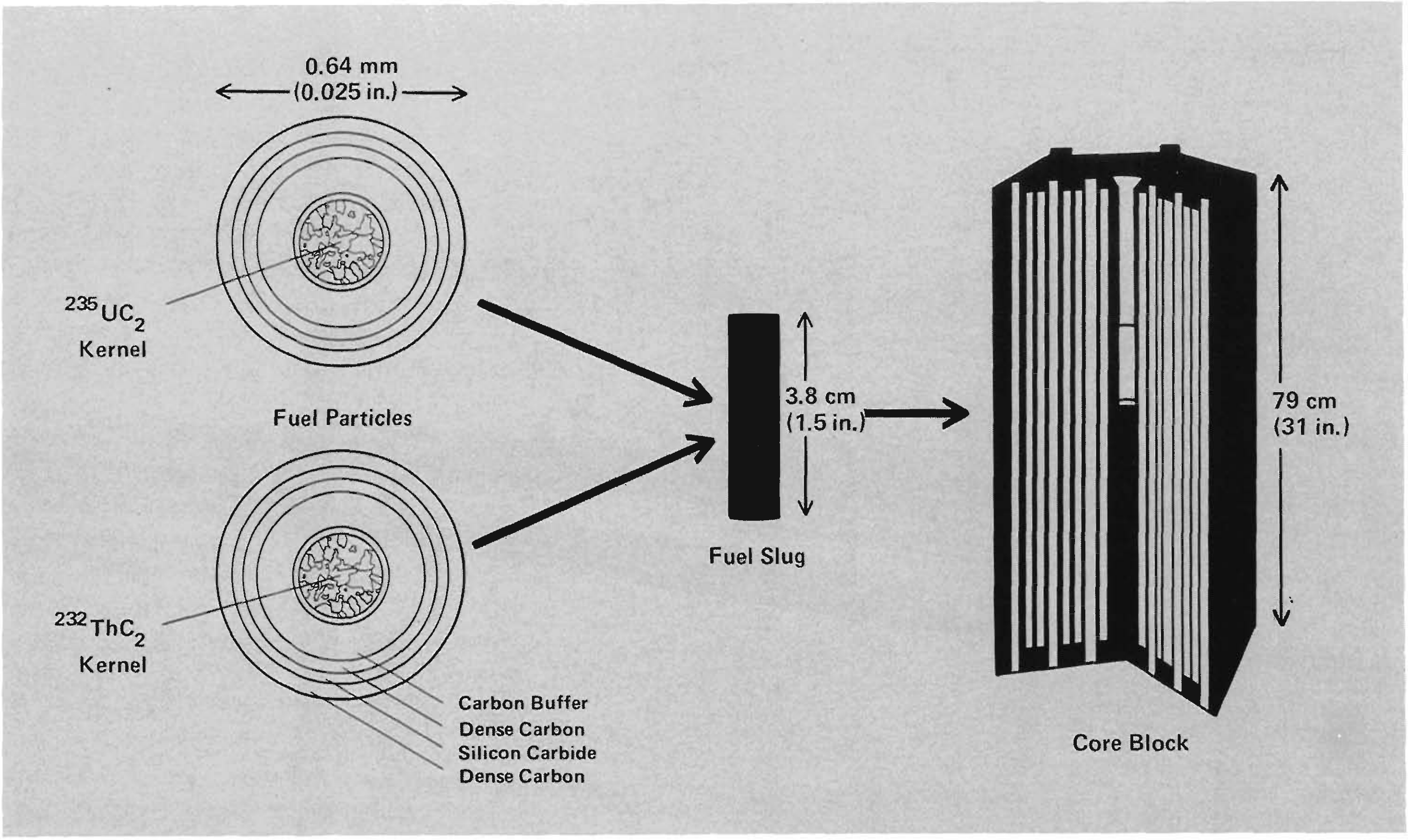

Fig. 15. Fuel for the high-temperature gas-cooled reactor is in the form of small particles containing a kernel of either fissile uranium-235 or fertile thorium-232, both as dicarbides. Typically, three barrier coatings plus an inner buffer zone encase the kernel and serve to contain the fission products. 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-

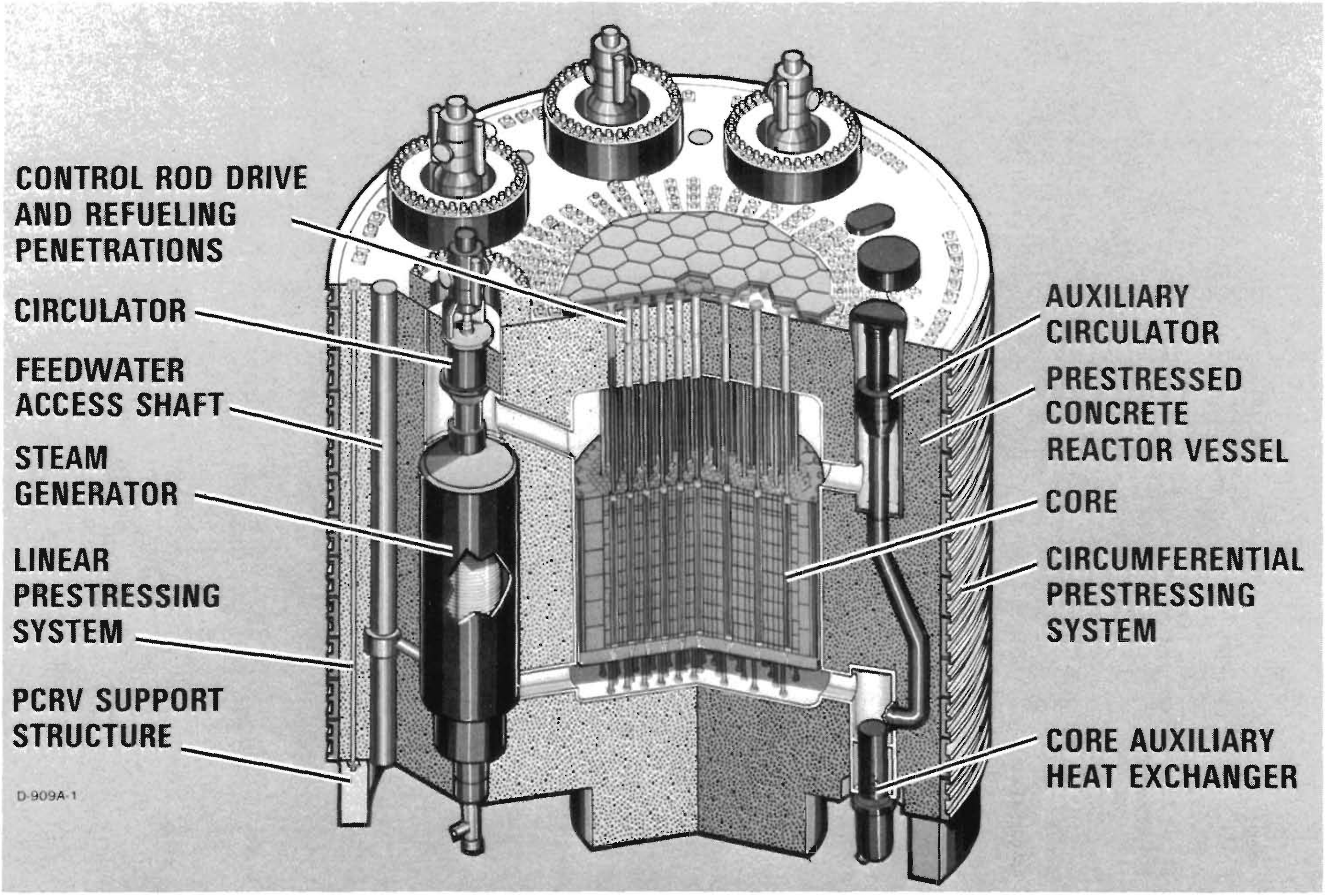

Fig. 16. A massive prestressed concrete vessel encloses the primary system of the Fort St. Vrain high-temperature gas-cooled reactor. Helium circulators force pressurized helium down through the core and up through steam generators. The auxiliary heat exchanger removes heat from the helium when the steam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

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

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.)

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

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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