

Figure 1

Computer simulations, verified by tests on a variety of experimental arrangements, provide assurance that a reactor's emergency provisions would respond adequately to a loss-of-coolant accident.

Charles K. Leeper

The need to increase the nation's electrical energy resources during the coming decades will place heavy demands upon the nuclear power industry. Nuclear power plants, both old and new, must meet Atomic Energy Commission regulations, which include the requirement for an emergency-core-cooling system to control the damage that might result from a loss-of-coolant accident (a "LOCA"). In this article, we will discuss emergency provisions for such accidents, the development and verification of appropriate mathematical models for the processes involved, related experimental programs, and we will assess the progress of the AEC's program in these areas.

The reactor fuel, in the form of uranium oxide pellets, is encased in metal tubes—the "cladding." The resulting "rods" are bundled and assembled to form a cylindrical core. Should the supply of coolant to the core be interrupted for a sufficient period, the core would heat up and conceivably the cladding could rupture, releasing radioactive fission products. In an extreme case (continuing loss of coolant), a core melt-down could occur. This is the type of accident for which emergency-core-cooling is provided, and the problem is to develop methods to predict, quantitatively, what would happen to a particular reactor design during various phases of the emergency without having to incapacitate a full-size reactor to test the methods.

The AEC's water-reactor safety program emphasizes sound engineering, quality fabrication and procedurized operation as the way to achieve highly dependable operation from nuclear power plants. However, because the probability of system and component failures cannot be reduced to zero, the AEC program also emphasizes the provision of redundant safety features wherever further risk reduction appears

desirable. The loss-of-coolant accident is an example; early in the development of nuclear utility power plants, it was identified as one condition for which redundant safety features should be provided in the design of light-water-cooled plants. As a result, reactor manufacturers are required to provide an auxiliary system capable of supplying emergency cooling to the plant to control damage that could result from a hypothetical rapid loss of coolant. The supporting AEC program includes research on mechanisms of the plant response to the accident, development of mathematical models to permit computer simulation of plant behavior during the emergency, and the conduct of system tests to demonstrate the ability of mathematical models to simulate the response of an individual plant subjected to a hypothetical accident.

Water-cooled nuclear power plants may be classified into pressurized-water reactor and boiling-water reactor—PWR and BWR—types. A typical PWR plant schematic is shown in figure 1. The primary cooling system circulates water from the pumps (1) by way of the reactor-vessel downcomer (2) and the lower plenum (3) into the reactor core (4), and thence through the upper plenum (5) to the steam-generator primary-coolant tubes (6). From the steam generator, the cooled water returns to the suction of the coolant pumps (1). The hot water circulating through the steam-generator tubes is used to boil secondary-system water (at a lower pressure) in the steam generators to produce main turbine steam. The pressurizer (7) maintains the primary system pressure well above saturation. Figure 1 shows two primary coolant loops, but actual plants may have as many as four. In addition to these elements, the primary coolant system is provided with

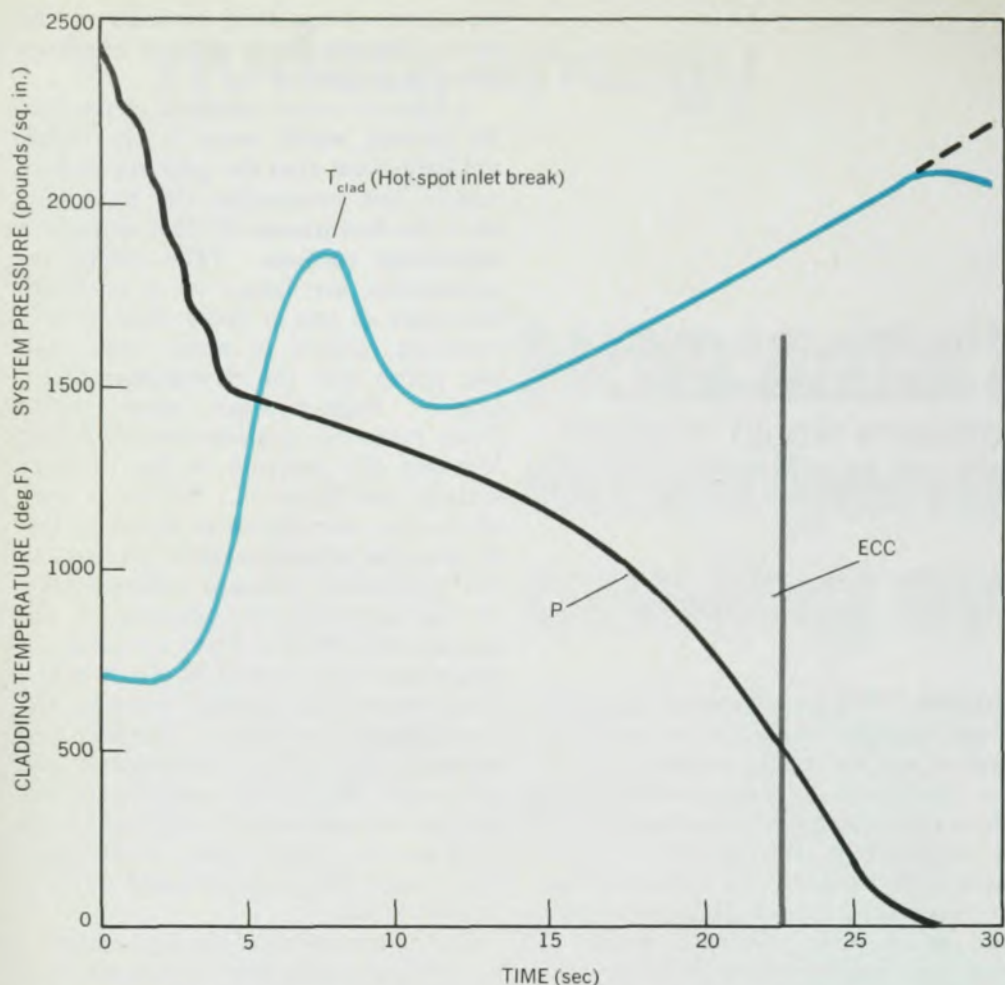
emergency-core-cooling systems in the form of accumulators (8) and auxiliary injection pumps (9).

A loss-of-coolant accident of particular interest would occur in the highly unlikely event that the primary coolant supply line connecting the pump (1) with the downcomer (2) had suffered a significant rupture. (The design requirements are based on a complete severance of one of these pipes with a resulting outflow of water from both the pump and the downcomer directions.) High-pressure water would flash into the containment building, lowering the pressure in the primary system (see figure 2). Within a very short time, enough water would be lost to drop the primary system pressure to the saturation pressure corresponding to the operating temperature of approximately 300°C. Then a mixture of steam and water would blow out of the break until the system reached the containment pressure. During this pressure fall, called "blowdown," the flow from the various portions of the primary system would consist of steam bubbles in pressurized water and, later, water droplets entrained in high-velocity steam.

The temperature of the fuel cladding in the reactor core during such an accident is a function of the heat generated, the amount of cooling afforded by the water and steam mixture that blows through the core on its way to the break, and the amount of cooling afforded by water from the accumulators and pumps. The reactor would shut down as soon as significant boiling occurred in the core, because of the reduction in the mass of water available to moderate neutrons. Thereafter, heat generation in the core would be limited to that from decay heat, gradually declining from an initial 5% of full power.

Figure 2 shows the temperature of the fuel cladding rising rapidly following the onset of the accident. The reason for the initial rise is steam blanketing of the fuel elements, the attendant large reduction in cooling, and the resulting radial equalization of the temperature within the cylindrical fuel element. Improvements in cooling, because of changes in core-flow conditions, lower the temperature of a short period, after which the excess of decay heat over available cooling leads to a steady rise in core temperature. Shortly before the system reaches containment pressure, accumulators start venting pressurized water into the supply lines and a portion of this water reaches the core to effect the needed cooling action. This reduces the core

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During a loss-of-coolant accident, the pressure in the primary cooling system would fall rapidly as high-pressure water flashes into the containment building. When the primary-system pressure has fallen to the saturation pressure at approximately 300°C (the operating temperature), a mixture of steam and water blows out of the break until the system reaches containment pressure. The temperature of the fuel cladding would rise rapidly at first, as steam blankets the fuel elements and reduces the cooling. Ensuing changes in the cooling conditions result in a brief period when the temperature falls, followed by a period of steady temperature rise. Emergency core cooling, which begins shortly before the system reaches containment pressure, reduces the temperature and brings the system under control; auxiliary pumps then provide long-term cooling.

Figure 2

cladding temperature—bringing the system under control. The auxiliary injection pumps then provide long-term cooling.

The boiling-water reactor has a somewhat different primary cooling system, but has many basic processes in common with the pressurized-water reactor during the loss-of-coolant accident. The flow schematic, figure 3, illustrates the primary system arrangement. In the BWR the core is cooled by primary water circulating from the jet pumps (1) to the lower plenum (2) through the core (3) and through the steam separators (4) to return to the jet pumps (1). Recirculation pumps (5) force a portion of the primary water at higher velocity through the jet pumps (1) to bring about the required primary water pumping action. The BWR "design-basis accident" consists of a break in one of the jet-pump recirculation lines downstream of the circulation pump (5). There are several jet pumps for each boiling-water reactor.

In the event of a loss-of-coolant acci-

dent in a BWR, emergency cooling is introduced through spray systems (6) mounted above the core. Cooling is initially effected by spray ricocheting through the bundles and by radiation from fuel rods to auxiliary core surfaces cooled by this spray. Later, excess spray floods the lower plenum and core to bring the temperature under control.

Model development and verification

The licensing of a nuclear plant requires, among other things, a demonstration by its manufacturer that the plant behavior during a loss-of-coolant accident is such that the fuel in the core can be maintained in a coolable geometry both during and after the emergency. Because an experimental blowdown of a full-scale plant is impractical, the manufacturer and the AEC use mathematical models of the plant and of the fuel to predict the plant and fuel performance during the hypothetical accident, and to determine how adequately the fuel is cooled by the plant blowdown process and by

the emergency-core-cooling action. The calculated cooling conditions are compared with AEC regulatory criteria to determine whether the plant is to be licensed or redesigned.

Development of such mathematical models requires a knowledge of all phenomena involved in the loss-of-coolant accident, the derivation of appropriate equations, and the generation of computer codes and plant models; then the models must be verified by showing from experimental data that they will predict the emergency performance of full-scale plants with acceptable error.

The verification procedure must examine the performance of the models in all of the components of the primary system for a range of scales and through the several accident phases. AEC and manufacturer tests are being planned, or are underway, at laboratory, pilot-plant, experimental nuclear plant and full-scale nuclear plant scales. Accident time phases include the blowdown, core-flooding and low-pressure injection-system sustained cooling phases. The phases of the blowdown and reflood processes are primarily determined by the state of the fluid (pressurized water, two-phase water, cold emergency-core cooling water, cold low-pressure water).

As selected experimental series are completed, the available data are applied to assess the capabilities of the mathematical models to predict the performance of full-scale nuclear plants. These assessments are then issued periodically to users of models so that they may not only be informed concerning the performance of models generated by the AEC, but can also apply the available data to assess the capabilities of the manufacturer's own models. Experimental programs to assess model capabilities include programs to determine fuel off-design capability, plant component characteristics, the nature of two-phase flow processes and characteristics of systems under various accident conditions.

Fuel experiments

The fuel consists of cylindrical uranium oxide pellets encased ("clad") in a zirconium alloy tube about 1 cm in diameter. Square arrays of these "rods" form the fuel elements used in the core. The fuel experimental programs include laboratory experiments to determine the reaction rate of fuel cladding with steam-water mixtures (as a function of temperature), the nature of deformations (ballooning) experienced by the fuel when the system pressure is decreased while fission-gas pressure within the fuel cladding is sustained, and experiments to determine the reaction of single fuel elements and fuel clusters to a variety of transient conditions.

With a new reactor called the "Power Burst Facility," we will be able to provide a time-varying neutron and/or coolant environment for test fuels in both single-rod and rod-cluster configurations. Fuel lengths (active) of up to three feet can be accommodated (plant reactor fuel is 12 feet long), and a wide range of operating conditions can be provided.

Emphasis in testing in the Power Burst Facility is on fuel behavior rather than on simulation of integral plant performance. Test results will provide data with which we can compare pre-test analyses of test sample behavior. Loss-of-coolant tests contemplated include: (1) tests in which the in-pile tube is "blown down" from typical power-reactor conditions and actual decay heat is generated by the rods subsequent to blowdown; and (2) tests initiated with the in-pile tube filled with steam (simulating conditions after blowdown is completed), in which the decay heat generation is simulated by nuclear operation of the driver core to produce low-level fission heat in the test fuel. Tests in category 1 would investigate the fuel behavior during the blowdown and in the post-blowdown fuel performance. In the second case, the ability to curtail heating by shutting down the test reactor will make it easier to investigate how the cladding fails.

In some tests, clusters of fuel that have one defective fuel rod will be used to determine whether cladding failure of a damaged rod can induce cladding failure in neighboring rods. The amount of fission gas pressure present in a fuel rod and the distortion of fuel pellets can both potentially influence the performance of the rod, both in ballooning and in heat-transfer mechanisms, so we plan to use some irradiated test fuels in the testing program. In parallel with these tests, other tests will be run with specially fabricated fuels in which the known major effects of irradiation have been simulated.

Flow process experiments

Another class of experiments is studying flow processes in the downcomer. During the blowdown, introduction of the emergency core cooling begins before the primary system is completely vented. Under these circumstances, we anticipate that the emergency cooling provided in the broken line will proceed directly out of the break and not provide cooling for the balance of the system. Coolant introduced in unbroken loops will proceed into the reactor vessel, and will then either fall through the downcomer to the lower plenum of the vessel or be diverted by the outflow of lower-plenum steam, thereby joining the flow out of the break. Because the coolant that is

lost at the break becomes unavailable for core cooling, this process must be properly described in the models. Experiments to investigate this phenomenon include a small-scale Plexiglass experiment, which provides an up-flow of air to simulate steam generation and a down-flow of water to simulate emergency-core-cooling flow in the reactor vessel downcomer. For various flow rates, steady-state data are obtained to relate the fraction of emergency-cooling liquid that bypasses the downcomer to the air up-flow parameters. In a similar experiment with a stainless-steel system, steam, used instead of air, allows us to assess the impact of condensation and vaporization effects on the bypass ratio.

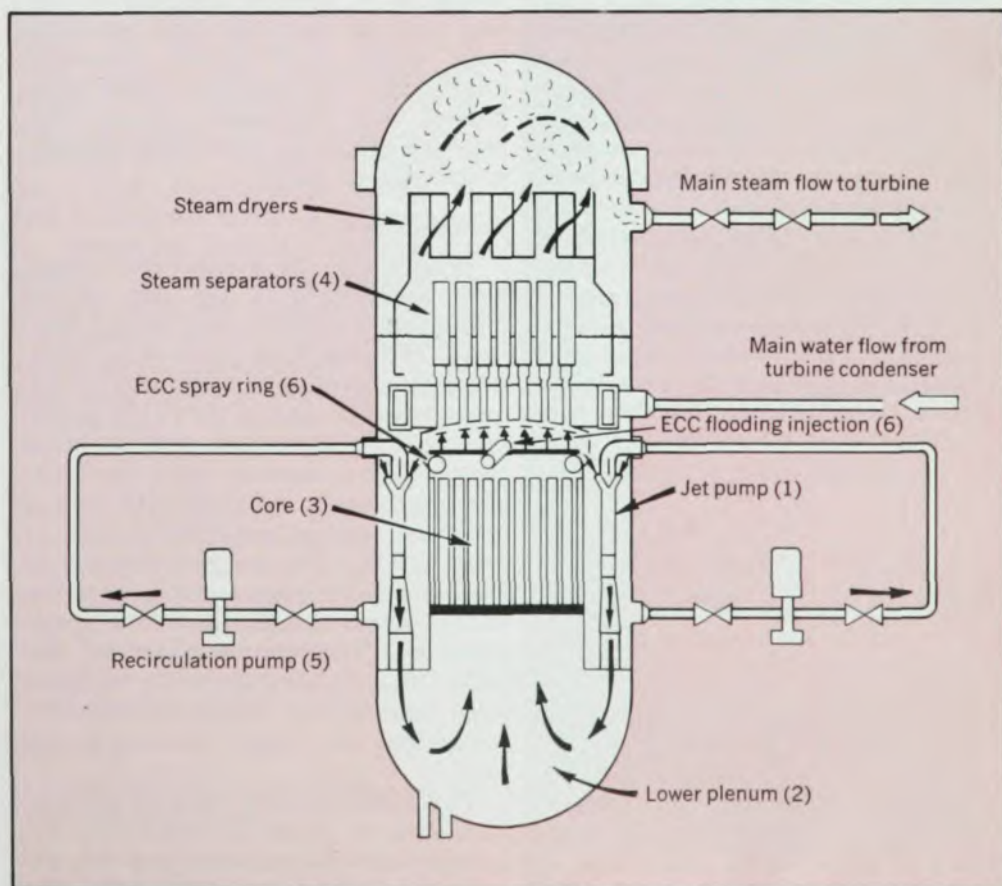
System effects can be studied in a pilot-plant scale with an essentially one-dimensional apparatus called the "semiscale" experiment. This experiment maintains system-volume ratios similar to those found in a full-scale plant, while operating at approximately 1 MW electrical power level. Its simplified "reactor" vessel has a multi-rod, electrically-heated core, together with a downcomer, primary coolant inlet and outlet and the necessary upper and lower plenum volumes. Flow from the outlet is conducted to a simulated steam generator, from which the flow returns to the vessel by way of a primary coolant pump.

With this experiment we can deter-

mine the ability of the computer codes to simulate the LOCA behavior of system components when coupled by appropriate flow passages. Reactor plants generally have multiple steam-generator loops; this experiment has been designed with only two loops, one of which is broken to induce blowdown. The other, unbroken, loop simulates up to three loops of a four-loop system. Breaks are simulated with remotely-actuated burst diaphragms. The semi-scale experiment is assembled in several forms to permit us to test a variety of pump and steam generator simulators and actual units. By the use of substitution tests, we plan to determine the relative influence of component simulation fidelity on the response of the system.

Experimental nuclear plant

The degree to which the mathematical models predict the performance of large-scale systems and components during a loss-of-coolant accident is to be determined using the LOFT ("Loss of Fluid Test"), which consists of a nuclear reactor provided with one full steam-generator loop and one loop with a simulated steam generator and pump. The system break is provided between the simulated pump and the downcomer of the reactor vessel. A fast-acting valve is programmed to simulate the characteristics of a variety of breaks, the effluent from the



Boiling-water reactor, shown schematically. The primary cooling system consists of items 1 through 5. Spray systems (6) provide emergency core cooling in the event of a loss-of-coolant accident downstream from the recirculating pump.

Figure 3

valve being conducted to a suppression tank that simulates containment pressure response. This experiment will provide 55 MW of nuclear heat generation during powered operation; it will show many three-dimensional effects anticipated in a full-scale plant, and will permit tests of increasing severity.

Core reflood experiments

For certain phases of the system response, the models for parts of the system are best treated in a more empirical fashion. One of these phases is the process of reflooding the core following blowdown of the system. As the pressure of the system reaches containment pressure, flow out of the break ceases, and emergency cooling liquid flowing through the unbroken coolant inlet pipes falls through the downcomer into the lower plenum. When the lower plenum is full, the cooling liquid begins to rise in the reactor core elements. The elements are by now quite hot, so the reflooding process produces sputtering, flying droplets, and, possibly, surging of the two-phase regime that develops. The motion of the liquid and generated steam results in a droplet-laden steam-flow out of the top of the core, thus providing improved cooling for the core. The flow of emergency cooling liquid into the core occurs when the height of the water in the downcomer exceeds that in the core itself. This height differential is offset by the momentum change in the coolant and in the steam as it rises through the core. An additional resistance to the flow is provided by the steam generator and pump through which the generated steam must pass on its way to the break.

Experiments to determine the heat-transfer characteristics of the core under reflooding conditions are called the "FLECHT" (Full Length Emergency Cooling Heat Transfer) experiments. For these experiments, conducted with electrically heated fuel rods, we have provided constant flow of emergency core cooling into the core inlet; lately, we have repeated them with water-leg pressure differentials rather than controlled liquid flow rates. Water spattering, droplet entrainment in the steam and radiation effects are observed during the rod quenching process.

Other experiments

Other experimental activities include tests of pumps and downcomers in scales ranging from pilot-plant scales to full-plant scale. In the smaller scales, tests will utilize flow with both liquid and mixed water-steam compositions, so that we can determine the impedance characteristics of pumps and the counterflow characteristics of downcomers under both single-phase

Table 1. Experimental Scales of ECC Verification Activities

	Pump characteristics	Reactor-core re-flood flow and heat transfer	ECC by-pass and steam-water mixing	Reactor-core blow-down flow and heat transfer	Fuel-element characteristics	System interactions
Full-scale pressurized-water reactors	■					■
Loss-of-fluid test (LOFT)	●	●	●	●	■	●
Pilot-plant scale tests	●	●	●	●	●	●
Laboratory tests		●	●	●	●	

●—major data contribution ■—limited data contribution

and two-phase flow conditions. Because the process of verification consists of taking data from a range of experiment scales and using these data to project the systematic and random-error characteristics of both the hardware and the computer models, it is important to include data for a full-scale utility system. Although it is impractical to subject a plant to a complete loss-of-coolant test, the typical plant does experience some transients in the course of normal operation. Our computer models can examine the reaction of a plant to a broad range of one- or two-phase flow transients, so it becomes desirable to collect data on plant-operating transients and some experimental transients (during start-up testing) in order to establish that the models can, in fact, describe events in these types of plant transient operation.

Current status

Progress on the experiments ranges from completion in the case of the Plexiglass, FLECHT, and certain semiscale and two-phase pump tests, to essential completion of fabrication of apparatus in the case of the LOFT experiment. We anticipate that a broad range of experimental data for blowdown processes, coupled with reflood processes, will be available as early as next spring. We are also considering requesting the instrumenting of a few utility plants with minimal flow-, pressure- and temperature-measuring devices, to provide an operating transient data base against which the computer codes may be further verified at full scale.

The Table indicates the principal process areas in which the verification activities will be conducted and the experimental scales from which data may be generated. As shown in the Table, pump characteristics will be determined for pilot and LOFT experi-

ments; the latter are provided with two-phase flow measuring devices in the downcomer, core-inlet and core-outlet regions. In addition, we are conducting large-scale experiments (approaching the scale of full reactor vessels) for selected flow processes. Note that the fuel-element characteristics will be determined primarily from laboratory- and pilot-scale experiments, the latter being provided by the Power Burst Facility. Data from the LOFT experiment will contribute to fuel knowledge in the sense that they will indicate regions in which fuel damage does not occur.

Error considerations

Current models are designed with conservative initial and accident conditions to ensure that the plants themselves will have adequate emergency core cooling. Computer codes now under development will calculate either best estimate or conservative values for pressures and temperatures at various points in the system. Data from the test program will then determine the statistical variations in component performance and the systematic and random errors in the mathematical modeling. A principal challenge of the verification activity will be to combine systematic and random error data obtained from several scales of experiments (under a variety of conditions) to project the systematic and random errors to be expected in computer model predictions of full-scale plant performance.

Perturbation studies with plant models indicate that the standard deviation for calculated clad temperatures is considerably greater than that for calculated pressures. Because the integrated time-temperature history of the element cladding is a determining factor in the amount of reaction between the fuel cladding and the steam environment, it is particularly impor-

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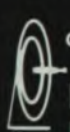
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tant to know the systematic and random errors for the temperature.

Conclusions

I feel that the following program areas deserve particular attention:

► Emphasis on fuel-behavior experiments to find how the fuel will behave during a loss-of-coolant accident and to determine cooling limits for fuel integrity.

► Development of the logic for verification of mathematical models, in order to provide statistically-based confidence in the predictions of plant performances during emergency core cooling.

► Continuing experimental study of two-phase processes in the downcomer, lower plenum, and break regions.

► A plant-level data collection program to extend the data base used in the verification process.

► Characterization of the fuel environment in terms of a weighted fuel cladding temperature-time integral, to express correctly the degree of reaction that can be anticipated between the cladding and the steam environment.

► Wide dissemination of data collected in the course of the experimental program, so that all code developers can assess the performance of their codes.

There is a direct relationship between the quantity of applicable data generated and the confidence that can be placed in the safety provisions of a particular plant. As one increases his conservatism in the assessing of the plant's emergency-cooling performance, the limiting power density at which a plant can be operated decreases. Clearly, excessive conservatism could result in the derating of a particular plant. Careful balancing of the several experimental efforts, to yield confidence levels appropriate for the individual process models, is, therefore, an essential ingredient in optimizing the power capacity of the nation's nuclear plants. This balance in experimental efforts should be a key result of a carefully executed verification program.

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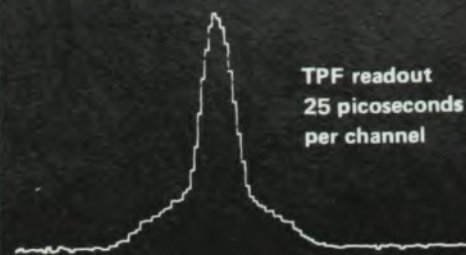
The nation's water reactor safety program is supported by the Atomic Energy Commission and by the industry; participants include national laboratories, private laboratories, the reactor manufacturers, the National Reactor Testing Station, the research and development arm of the AEC and the Regulatory Directorate of the AEC.

Reference

As a general reference to the matters discussed in this article, see:

"The Safety of Nuclear Power Reactors (Light Water Cooled) and Related Facilities," prepared by the staff of AEC in response to a request by the chairman of the Joint Committee on Atomic Energy, WASH-1250 (December 1972). □

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