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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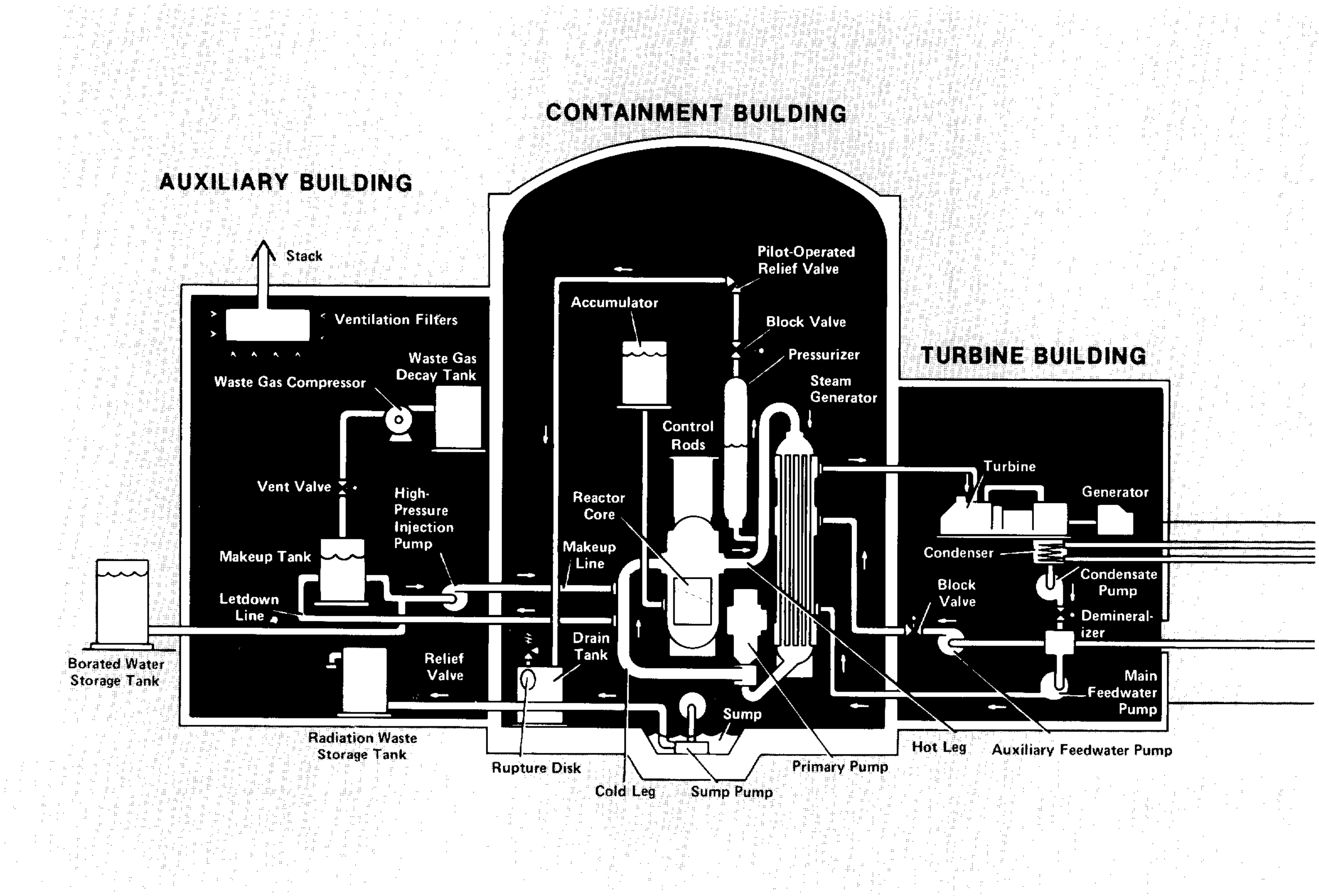
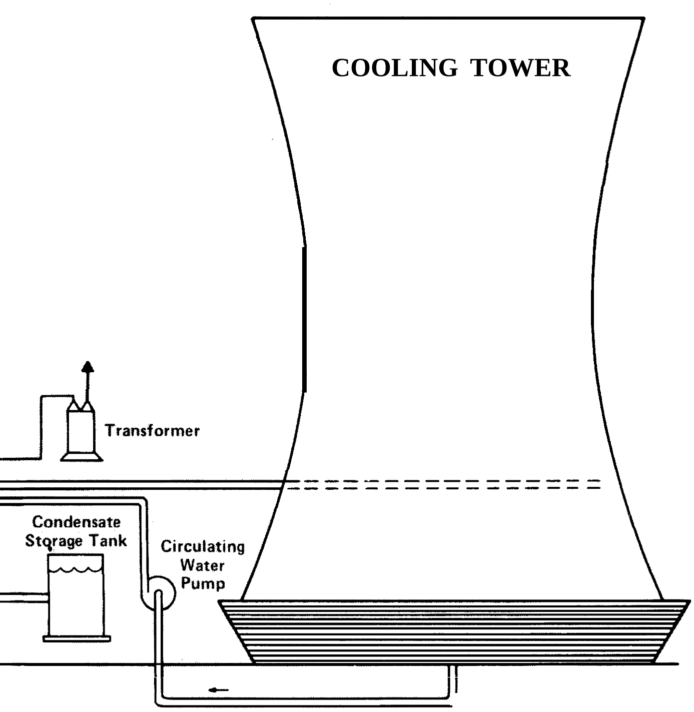


Fig. 1. This diagram of the Three Mile Island Nuclear Power Station's Unit 2 reactor system* shows *one of its two primary coolant loops and all other system components important to the accident. The time sequence outlined below includes some system responses that are known only from later analyses.

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.
30min (04:30:37)	Auxiliary building storage tanks overflow. Some radioactive materials escape to the environment through the building's vent stack.
lh, 13 min (05:13:37)	Operators turn off primary pumps in B loop because the steam in the system causes them to vibrate excessively.
lh, 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 pi lot-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 pro vides 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 noncon-densible 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.

[*] *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.***

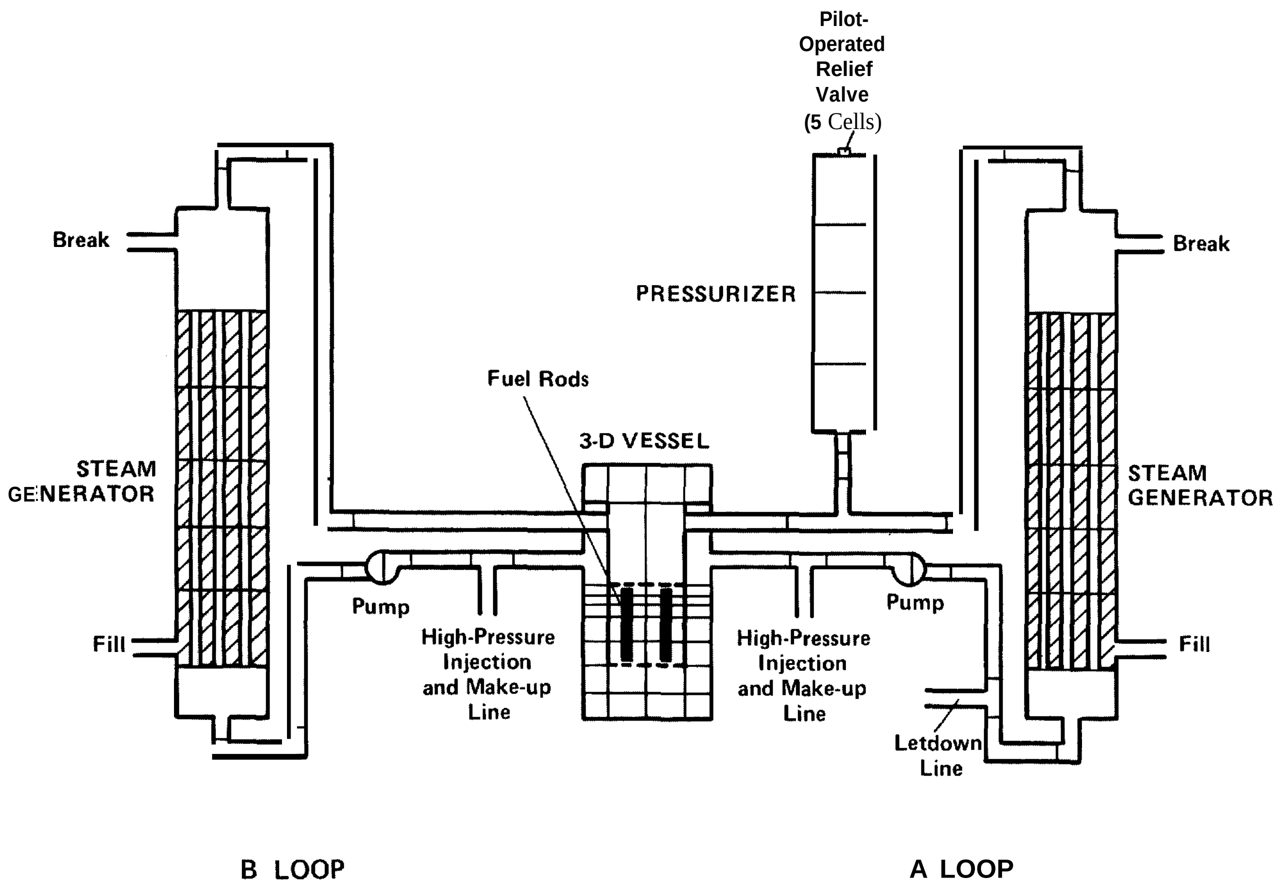

Fig. 2. Schematic of the TRAC computing mesh for the two primary coolant loops of Three Mile Island Unit 2 reactor. To reduce the number of cells, the mesh represents the two cold legs in each loop by a single cold leg. The reactor vessel mesh, divided into nine axial levels, includes four lumped fuel rods to model heat transfer between fuel rods and fluid. (The actual core contained 177 fuel-red 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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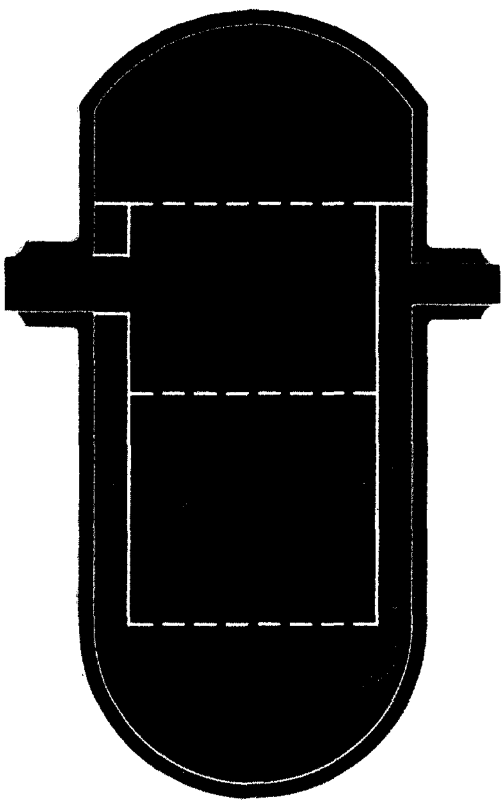
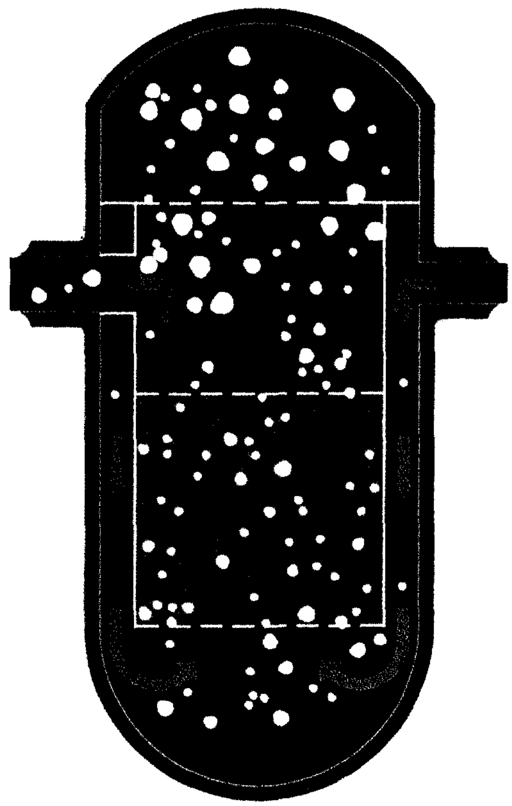
TRACANALYSIS OF THE THREE MILE ISLAND ACCIDENT

Sidebar 1:



t = 0
Normal operation

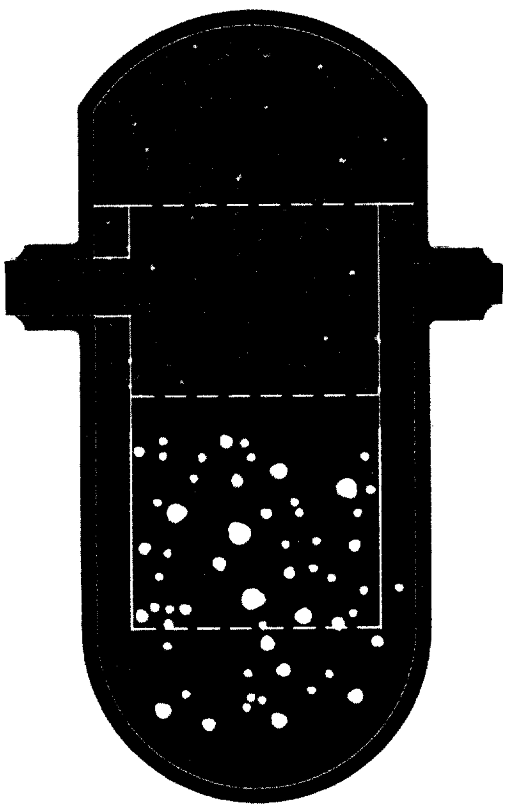


t = 70 **min**
Equilibrium conditions in core
Cooling by boiling and forced convection

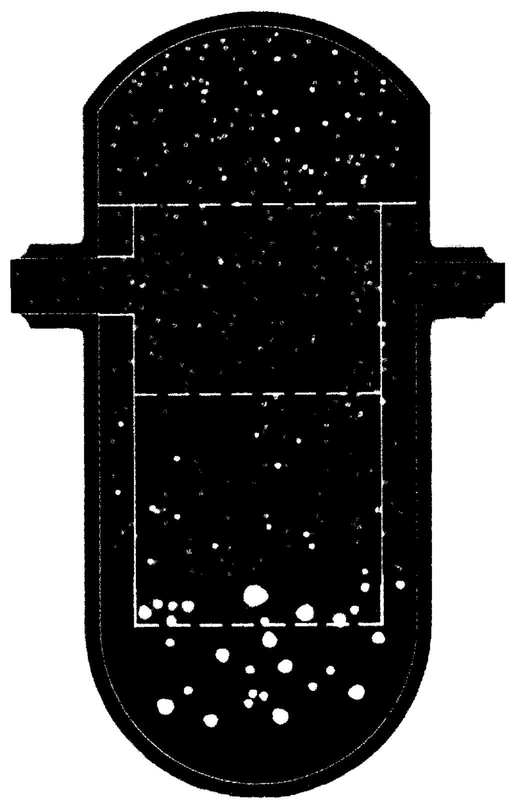
Them were no eyewitnesses to the scene within the Unit 2 reactor—and the instrument readings gave only an incomplete and misleading picture. However, from the TRAC analysis we have been able to reconstruct an accurate account of the conditions inside the primary system during the first three hours of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

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

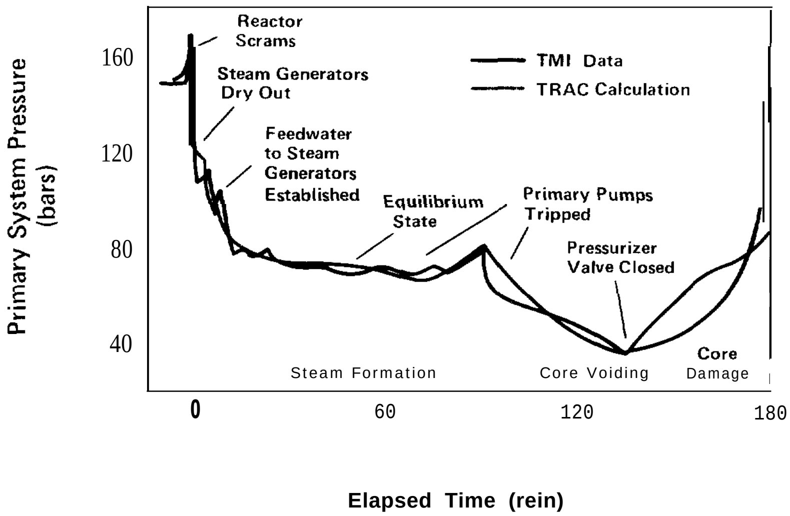
The transient calculation includes the assumption that the letdown flow was greater than the high-pressure injection flow by about 2.7 kilograms per second between 10 and 140 minutes.

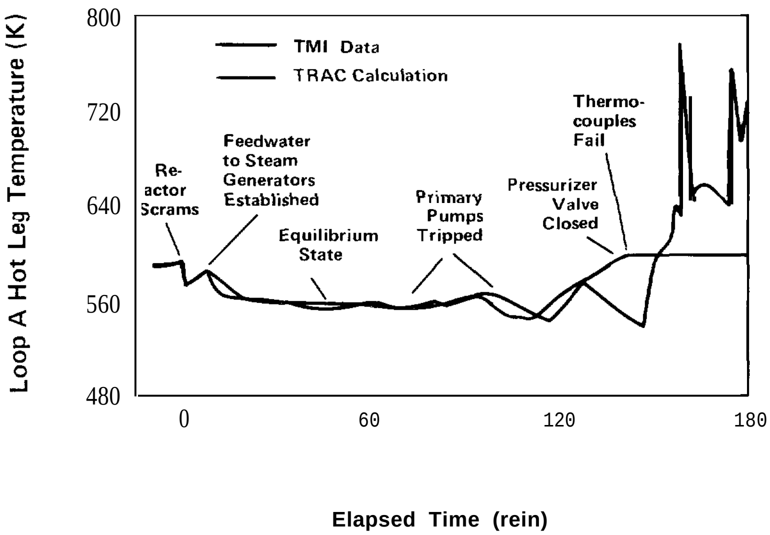
Sidebar 1: Continued

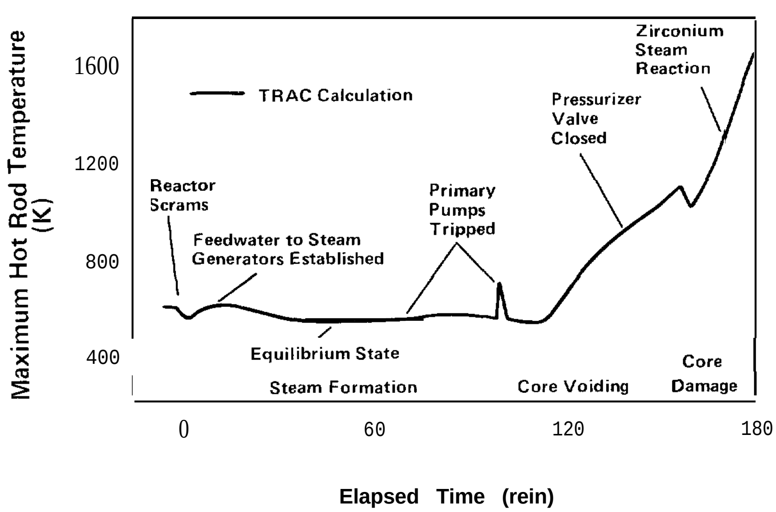
level in the pressurizer drops because of increases in letdown flow rates and decreases in system pressure that cause the water in the pressurizer to flash to steam.

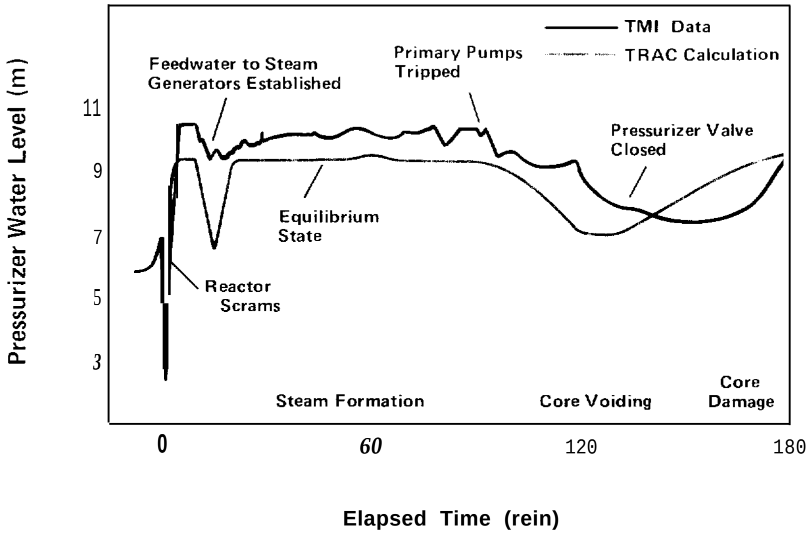
Closing the block valve in series with the pilot-operated relief valve at 140 minutes causes the steam flow in the core to stagnate. Steam can no longer escape through the valve and water in the loop seals prevents any flow through the loops. Without natural circulation, the system begins to pressurize 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 be gins 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.

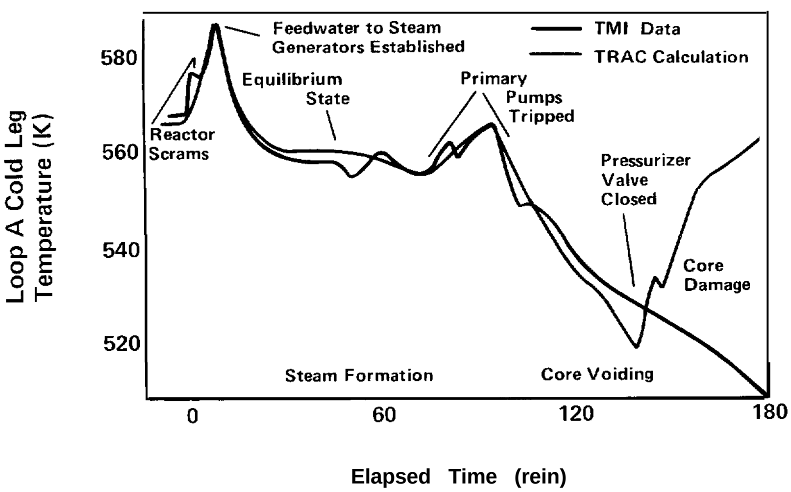
At approximately 3 hours, the top 75 per cent of the core is uncovered. The fuel-rod temperatures remain relatively low in the lower core region because some water is still available for cooling, The pressurizer water level is increasing both in the TRAC calculation and in the plant data. The pressurizer 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. ■

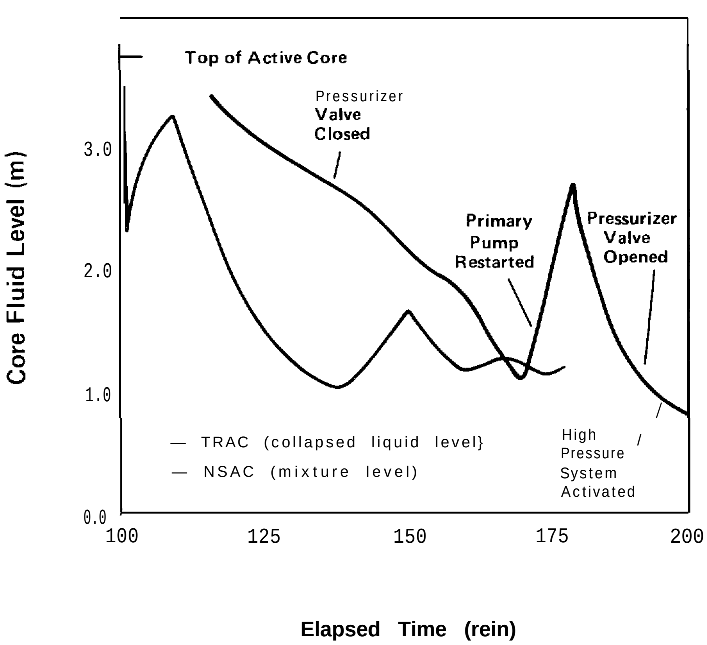







TMI Data
TRAC Calculation
Feedwater to Steam Generators Established
Primary Pumps Tripped
Pressurizer Valve Closed
Equilibrium State
Reactor Scrams
Steam Formation
Core Voiding
Core Damage
Pressurizer Water Level (m)
11
9
7
5
3
0
60
120
180
Elapsed Time (rein)


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


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

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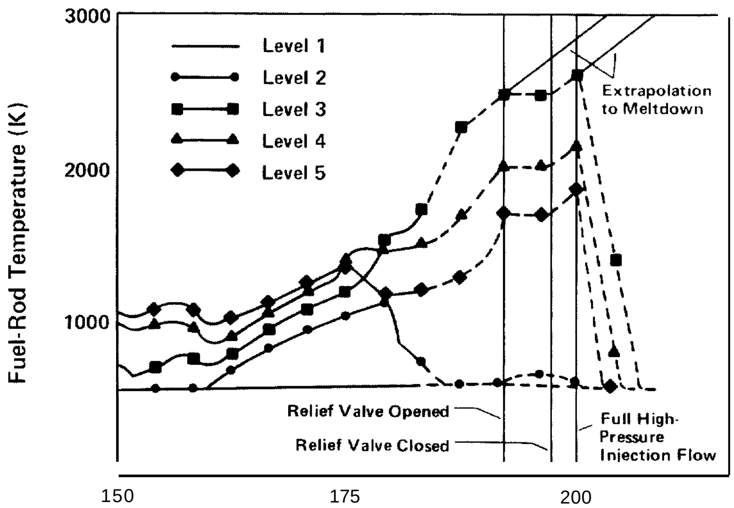
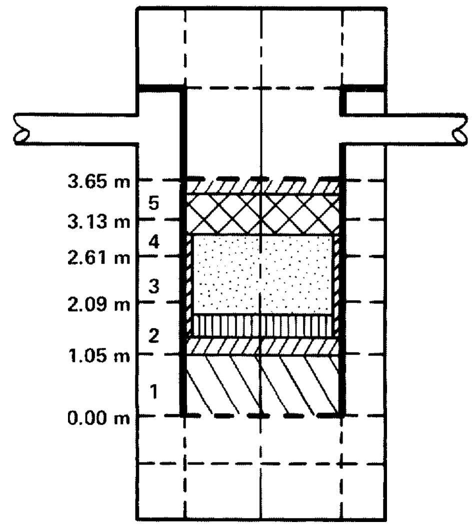
**M. L. Picklesimer, "Bounding Estimates of Damage to Zircaloy Fuel Rod Cladding in the TMI-2 Core at Three Hours After the Start of the Accident, March 28, 1979, " Nuclear Regulatory Commission memorandum (June 20, 1979) and K. H. Ardron and D. G. Cain, "TMI-2 Accident Core Heat-up Analysis, " Nuclear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

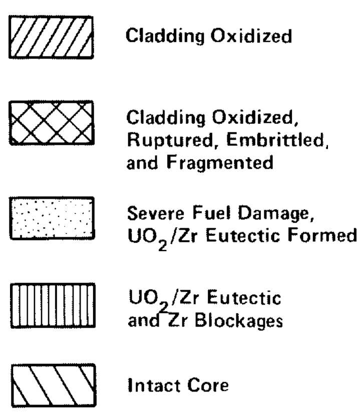
***The possible effects on the containment of core damage even more severe that that which occurred are discussed in the sidebar "What If The Core Melted?"*

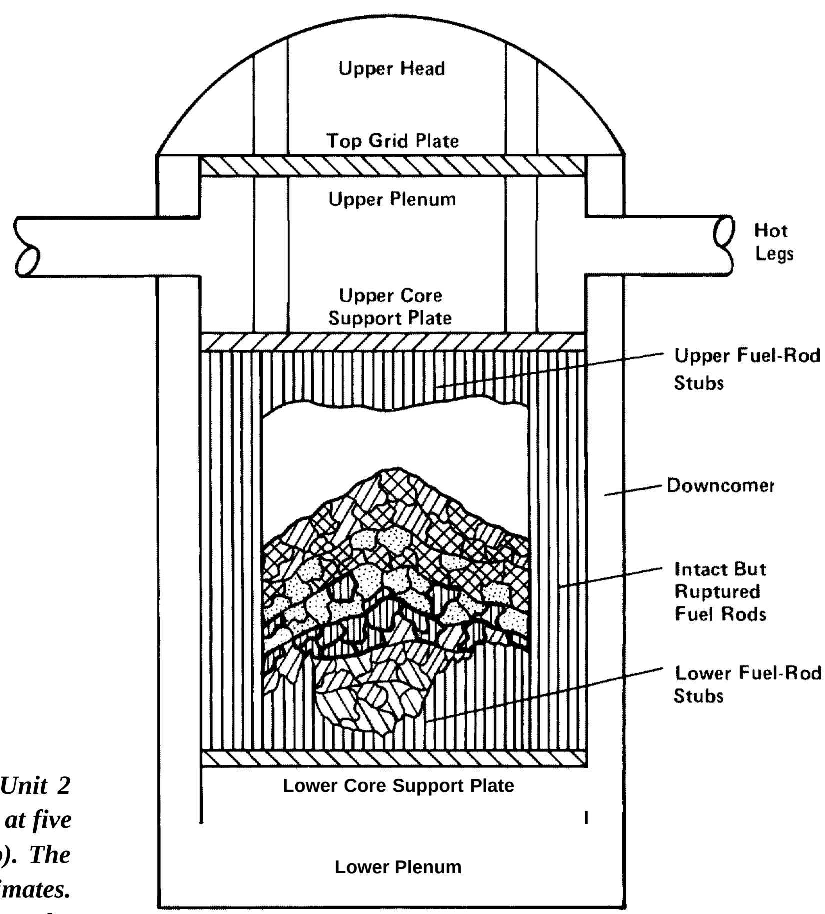


(a)



(b)





(c)

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 bask for core damage estimates. (b) Estimated condition of core materials at 35 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 igniters in the containment to prevent accumulation of much more than a burnable mixture of hydrogen and air. Such igniters 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. ■

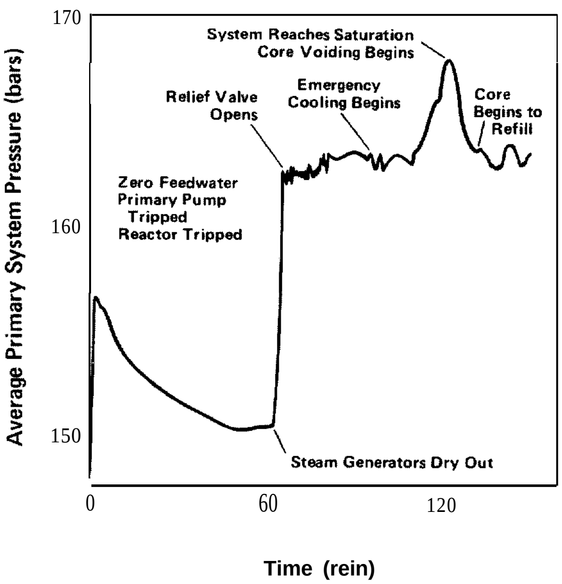


Fig. 4. TRAC-calculated primary system pressure at Zion Unit I during a postulated loss-of-feedwater accident.

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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 (12). 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 IL Stratton was a member of the Technical Assessment Task Force of the president's Commission on the Accident at Three Mile Island (aIso 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)].

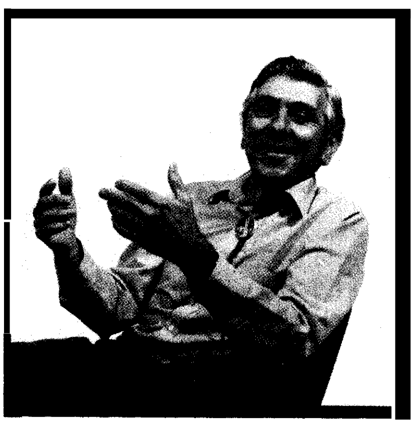
John W. Bolstad and Roy A. Haarman submitted TRAC analyses of postulated reactor transients quite similar to the Three Mile Island accident. These analyses, which had been completed before the accident as part of a sabotage study, provided the Commission with a better understanding of some aspects of 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 TM I 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.

Further Reading

"Three Mile Island Nuclear Station, Unit 2," License Application, FSAR, Volumes 1-10, Metropolitan Edison Company docket 50320-73 through 50320-82 (April 1974).

"Investigation into the March 28, 1979 Three Mile Island Accident by Office of Inspection and Enforcement," United States Nuclear Regulatory Commission report NUREG-0600 (July 1979).

The President's Commission On the Accident at Three Mile Island, *Report of the President Commission On the Accident at Three Mile Island, The Need for Change: The Legacy of TMI* (October 1979).

"Three Mile Island: A Report to the Commissioners and to the Public," Vols. I and II, Nuclear Regulatory Commission Special Inquiry Group report NUREG/CR-1250 (1980).

United States Senate Committee on Environment and Public Works report, "Nuclear Accident and Recovery at Three Mile Island" (June 1980).

J. R. Ireland, "Three Mile Island System Thermal-Hydraulic Analysis Using TRAC," Proceedings of the American Nuclear Society/European Nuclear Society Topical Meeting, Thermal Reactor Safety, Knoxville, Tennessee, April 6-9, 1980 (Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1980) Vol. I, pp. 574-581. Available from National Technical Information Service as CONF-800403.

W. L. Kirchner, J. R. Ireland, and P. K. Mast, "Fuel Damage Estimates for the TMI-2 Reactor," Proceedings of the American Nuclear Society/European Nuclear Society Topical Meeting, Thermal Reactor Safety, Knoxville, Tennessee, April 6-9, 1980 (Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1980), Vol. I, pp. 590-596. Available from National Technical Information Service as CONF-800403.

P. K. Mast, T. R. Wehner, and J. R. Ireland, "Analysis of Early Core Damage at Three Mile Island," Proceedings of the American Nuclear Society/European Nuclear Society Topical Meeting, Thermal Reactor Safety, Knoxville, Tennessee, April 6-9, 1980 (Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1980), Vol. I, pp. 582-589. Available from National Technical Information Service as CONF-800403.