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VOLUME II Part 2

three mile island

A REPORT TO THE COMMISSIONERS AND TO THE PUBLIC

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NUCLEAR REGULATORY COMMISSION
SPECIAL INQUIRY GROUP

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Appendices

OUTLINE OF VOLUME II
PARTS 1, 2, AND 3

II. THE ACCIDENT AND ITS ANALYSIS

A SEQUENCE OF PHYSICAL EVENTS

The following narrative of the sequence of physical events in the accident at Three Mile Island is based on information from several sources: plant computer output, automatically recorded data, log entries, and operators' statements. For a more detailed description of this sequence of events, along with references to the original data sources, the reader should consult Appendix II.1.

In addition to factual material, the text contains many explanatory and interpretive statements. Inferences, interpretations, explanations, and opinions have been set off by brackets [ ]. In many cases, the physical data could support alternative interpretations. Wherever more than one interpretation of the data is possible, the choice of interpretations that is presented has been based on plausibility, normal practice, or consensus of experts. It should be understood, however, that this interpretation is not the only possible one. In many cases, it may never be possible to establish exactly what happened.

No references are provided in this section. All events described here have been referenced in the more detailed description of the sequence of events in Appendix II.1. In the text that follows, references are made to other sections of the report. More detailed explanations can be found in those sections.

Some events described in this section are also covered in Section II.E. The decision as to whether an event should be covered in both was made on the basis of the event's probable or possible effect on subsequent actions of the control room operators.

March 28, 1979—4:00 a.m.

At 4:00 a.m. on March 28, 1979, TMI-2 was operating at between 97% and 98% full power. The shift foreman and two auxiliary operators had been working in the auxiliary building on the No. 7 condensate polisher. Two licensed control room operators were on duty in the control room. The shift superintendent was in his office adjacent to the control room.

The condensate polishers use ion exchange resins for purification of the feedwater (Figure II-1). During operation, flow through the resin bed tends to compact the material into a rather solid mass. To transfer the resin beads to the resin regeneration system, it is necessary to break up this mass by blowing compressed air through it. [Apparently, during the process of air-fluffing, water entered an instrument air line through a check valve that had frozen in the open position.]

It has been postulated that the water in the air piping caused the polisher inlet or outlet valves, or both, to close.[1] [Problems with the valves in the

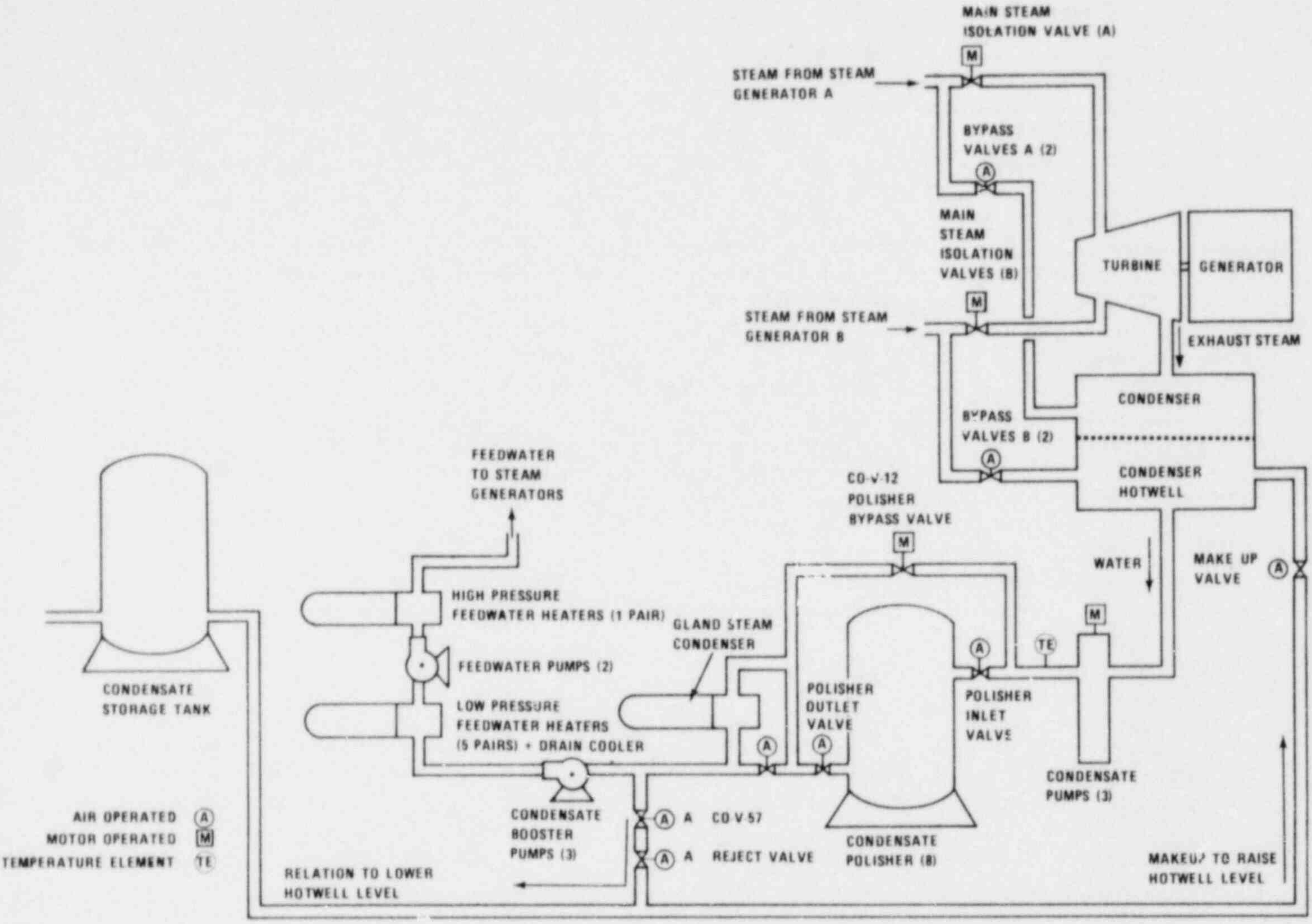


FIGURE II-1. The Condensate and Feedwater System

Under normal conditions, exhaust steam from the turbine is condensed in the condenser. Condensate (water) is pumped by the condensate pumps through the condensate polisher, where it is purified. The pressure is raised by the condensate booster pumps, and the temperature is increased in the low pressure feedwater heaters. The water is then pumped by the feedwater pumps through the high pressure feedwater heaters to the steam generators. The polisher bypass valve can be opened so that water flows directly from the condensate pumps to the condensate booster pumps. When the hotwell level is high, water flows through the reject valve to the condensate storage tank; this allows more water to leave the hotwell than is entering from the condenser. When the hotwell level is low, the makeup valve is opened and water is returned from the condensate water storage tank to the hotwell. The condenser and hotwell are always under vacuum. At the beginning of the accident, the inlet and outlet valves of the condensate polisher accidentally closed. The bypass valve would not open because of a control fault; this probably caused the condensate booster pumps to trip, followed by trip of one condensate pump and both steam generator feed pumps. A severe "water hammer" damaged the controls of valve CO-V57 and the reject valve so that condenser hotwell level could not be controlled.

There is no radioactivity associated with the condensate and feedwater systems nor are they unique to nuclear powerplants. The condensate and feedwater systems of fossil-fueled plants are very similar to those at TMI.

polisher system are discussed in Section II.C.1.d.] Closure of either the inlet or outlet valves would interrupt the flow of feedwater and cause the condensate pumps and condensate booster pumps to trip, that is, to be automatically shut down. Tripping of these pumps causes tripping of the main feedwater pumps, which in turn, causes tripping of the main turbine and electrical generator.

In the accident at TMI-2, it has not been definitely determined what caused the condensate pump to trip, although it is a reasonable inference that the operations on the polisher were somehow involved. It has been established that condensate pump 1A tripped and that both main feedwater pumps then tripped almost simultaneously. Approximately 1 second later, the turbine and generator tripped.

The three emergency feedwater pumps (two electric-driven and one steam-driven) started automatically within 1 second after the main feedwater pumps tripped. The purpose of the emergency feedwater pumps is to ensure a continuing supply of water to the steam generators (OTSG) when the main feedwater pumps are not working. Water from the emergency feedwater pumps is not normally delivered to the steam generators immediately after the main pumps cease to operate. The automatic valves (EF-V11A and EF-V11B, Figure II-2) will not open until two conditions have been met: (a) the emergency pumps are delivering their normal discharge pressure (at least 875 psig) and (b) the water level in the steam generators has sunk to 30 inches or less.

In addition to the automatic valves, there are block valves[2] (EF-V12A and EF-V12B) in the lines to the steam generators. These valves are required to be open while the plant is operating. At the time of the accident, however, the block valves were closed. The closed indication of these valves, which was shown on an indicator light in the control room, was not noticed by the operators.

The reactor is not automatically shut down when turbine trip occurs. [The desirability of an automatic shutdown feature is discussed in Section II.C.1.b.] The integrated control system (ICS)[3] decreases, but does not shut off, the reactor power. On loss of feedwater followed by turbine trip, the energy removed from the steam generators was less than the energy added by the reactor, and the pressure in the reactor coolant system (RCS) increased. The pressure increase began immediately.

To protect the RCS from excessive pressure, a pilot-operated relief valve (PORV) and two safety valves are provided. Three seconds after turbine trip, the pressure in the RCS had increased to the point (2255 psig) at which the PORV opened. The reactor was still delivering power, and pressure continued to rise, although not as rapidly. Eight seconds after the turbine trip, the pressure had reached the point (2355 psig) at which the reactor is automatically shut down.

During the time that these automatic functions were taking place, the operators took the following actions:

1. Checked the turbine throttle and governor valves for closure. (The operators found that one throttle valve meter did not show closure. Closure of the governor valves, which shuts off steam to the turbine, however, was shown.)
2. Switched the pressurizer from manual to automatic control. (The pressurizer had previously been manually controlled to equalize boron concentrations between the pressurizer and the reactor.)
3. Verified opening of the turbine bypass valves.
4. Set the generator circuit breakers in the locked-out position.
5. Manually tripped the turbine to make sure all trip functions operated.

Immediately after the reactor trip, the operators confirmed insertion of all control and safety rods. It was definitely known that the reactor was now shut down. Nuclear fission quickly stops when the control rods are inserted. The products of the fission reaction, however, are themselves radioactive and continue to decay after the reactor is shut down. The power produced in this radioactive decay is called decay heat. Immediately after shutdown, the decay heat is about 160 MW. Dropping very rapidly at first, the decay heat is approximately 33 MW about 1 hour after the reactor is shut down. Ten hours after shutdown, it is about 15 MW. After that, the decay heat decreases more slowly.

The reactor coolant expands when heated and contracts when cooled. The excess energy delivered by the reactor causes the coolant to expand until the reactor trips. After the reactor trips, the excess energy removed by the steam generators causes cooldown and contraction of the coolant. Volume changes in the coolant are reflected in changes of pressurizer level.

When the system is operating, water is continuously removed from the RCS via a drain called the letdown system, is purified, has boric acid added or removed, and is returned to the RCS through the makeup pumps (Figure II-3). In normal operation, makeup slightly exceeds letdown so that small losses from the system through the normal leakage are replaced. Before the accident, leakage was higher than usual, because a code safety valve, or possibly the PORV, was leaking. [Additional discus-

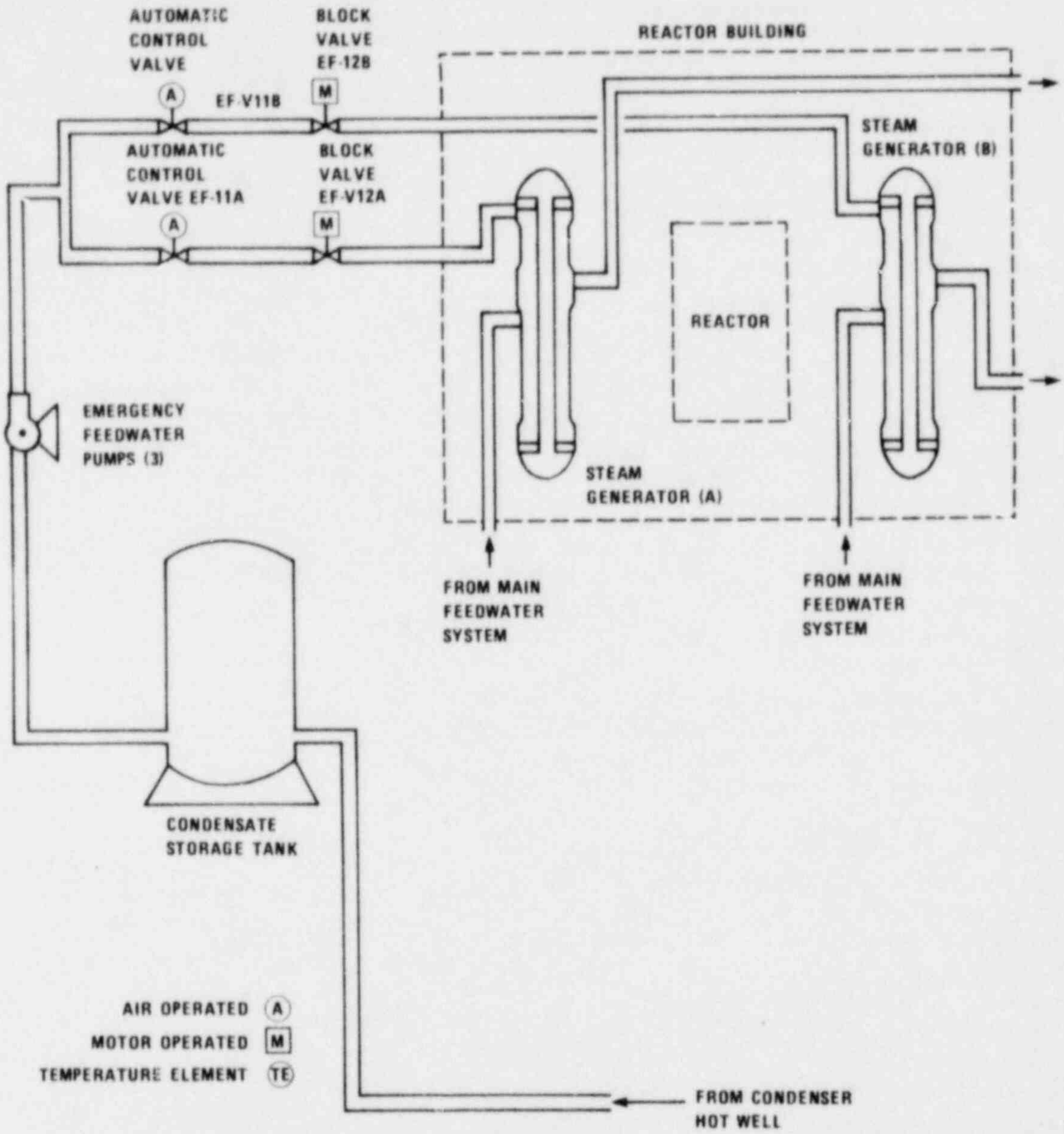


FIGURE II-2. The Emergency Feedwater System

Three emergency feedwater pumps (two electric, one steam driven) are started automatically on loss of the main feedwater pumps (Figure II-1). The emergency feedwater pumps take suction from the condensate storage tank; in an emergency, they can also take suction directly from river water. The automatic control valves will open (a) when the discharge pressure of the emergency feedwater pumps is high enough and (b) when the water level in the steam generators falls to 30 inches or less. Until both conditions are satisfied, the control valves remain closed. The block valves should have been open at all times; however, at the time of the accident these valves were closed. When the conditions were met, the control valves slowly opened, but no water was admitted to the steam generators because the block valves were still closed. About 8 minutes after the start of the accident, the operator discovered that the block valves were closed, and opened them. This admitted water to the steam generators.

The block valves can be operated by switches in the control room, by switches in the auxiliary building, and manually at the valves. It is not known from which point nor when the valves were closed.

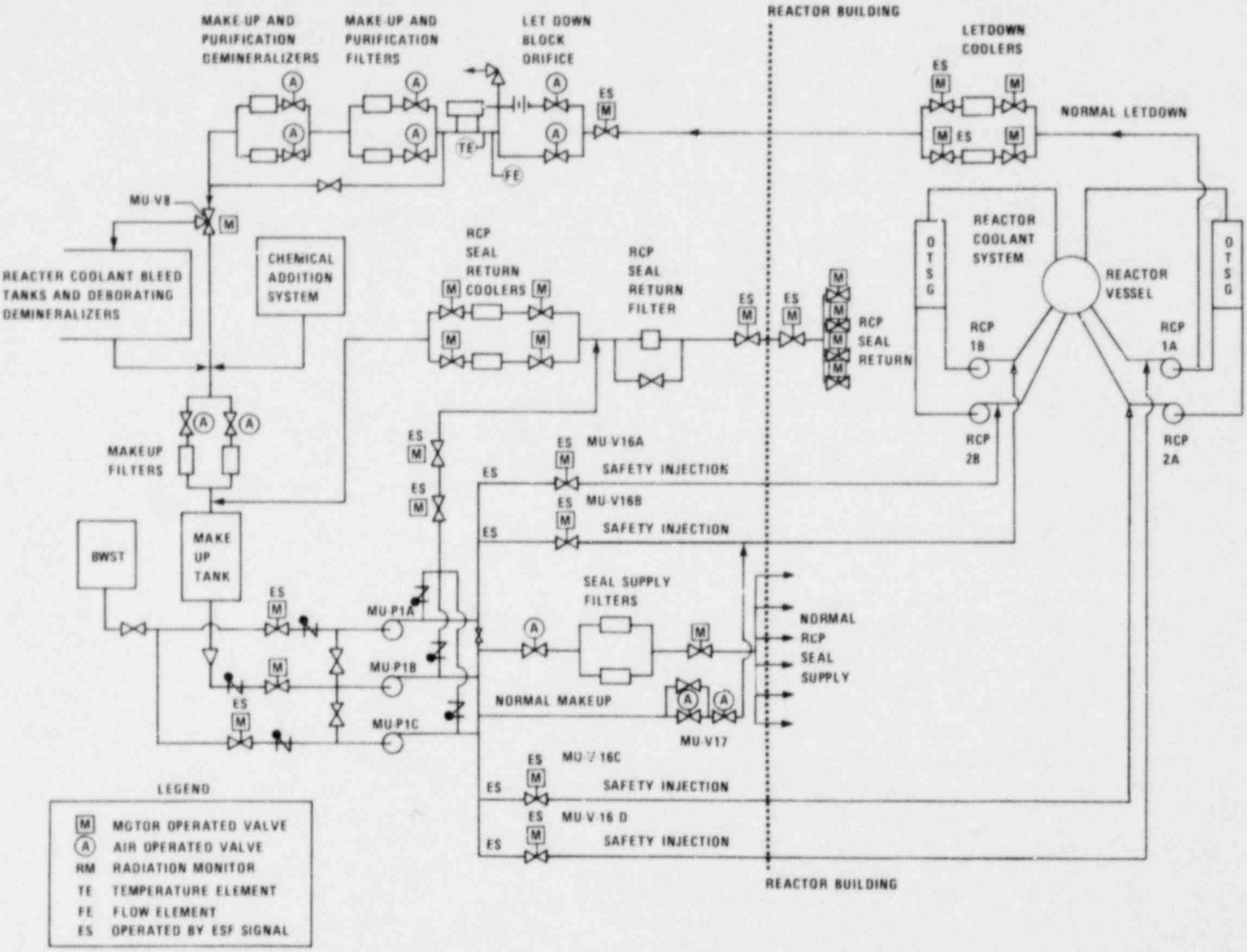


FIGURE II-3. The Makeup and Letdown System

During normal operation, water is removed from the reactor coolant system (RCS) near the suction of the reactor coolant pump (RCP-1A). The water removed is purified and cooled and can be sent either to the reactor coolant bleed tanks or to the makeup tank. At least one makeup pump is always operating; this supplies water to the reactor coolant pump seals. A small amount of the seal water leaks out and is returned to the makeup tank through the seal return system; the remainder enters the RCS. Any additional water required to maintain the correct inventory in the RCS is regulated by valve MU-V17 and enters the discharge line of RCP-1A. In the high pressure injection mode (when the engineered safeguards are actuated), two makeup pumps (normally 1A and 1C) take water directly from the borated water storage tank, and pump through valves MU-V16A, 16B, and 16D—which are wide open—to all four RCS cold legs. Letdown is stopped during engineered safeguards operation.

When the pressurizer is in the "automatic" mode, valve MU-V17 is controlled by pressurizer level.

sion of plant operation with leaking valves can be found in Section II.C.1.b.] Leakage from the PORV went to the reactor coolant drain tank (RCDT) where it was condensed and was then pumped to the reactor coolant bleed tanks. The buildup of water in the bleed tanks was then being transferred periodically to the makeup tank.

If not compensated for, the expected shrinkage of reactor coolant on cooldown could cause an excessive change of volume. To reduce the rate of volume change, therefore, letdown is stopped and makeup is increased. Thirteen seconds after the turbine trip, the operator stopped letdown. He also attempted unsuccessfully to start a second makeup pump. This operator has testified that the pump did not start because the switch was not held in position long enough. The pump was started later, however, by another operator. (The operation of the makeup pump is discussed further in Section II.C.1.c, in which it is concluded that only momentary switch contact is required to start the makeup pumps.)

Thirteen seconds after turbine trip, pressure had lowered to the point (2205 psig) at which the PORV is designed to close. An indicator light in the control room shows when the valve has been ordered to close—that is, when power to the valve opening solenoid is cut off—but does not show when the valve actually closes. It is now known that the valve did not, in fact, close as it was designed to do. The operators, however, had no direct means of knowing this.

Fifteen seconds after turbine trip, the pressurizer level reached a maximum of 255 inches (from an operating level of about 220 inches). Contraction of the coolant then caused a rapid drop in pressure, as was expected. [The operators expected to reduce the amount of contraction by adjusting makeup and letdown flows.] [By 28 seconds after turbine trip, the two conditions for admission of emergency feedwater to the steam generators had been met, and the automatic valves should have begun to open. Because the block valves were closed, of course no water could be admitted to the steam generators even with the automatic valves open. It appeared to the operator that the automatic valves were opening at an unusually slow rate, and the slow opening of these valves was initially attributed to the delay in feeding the steam generators.]

Thirty seconds after turbine trip, an alarm of high PORV outlet temperature was received on the alarm printer in the control room. This alarm was not printed out until several minutes later, because the alarm printer, which was receiving over 100 alarms per minute at the time, was overloaded. Such an alarm does appear on an annunciator; however, the annunciator is not readily visible from the normal operating location. [The high temperature alarm did not show that the valve was still open; the momentary opening known to have occurred previously, plus the known leakage, would have accounted for this alarm.]

By 30 seconds after turbine trip, the contraction of the reactor coolant had reduced the pressure in the RCS to the point (1940 psig) at which the reactor would have been tripped if it had not previously been tripped on high pressure.

By 40 seconds after reactor trip, both steam generators had boiled down to the low level alarm point. [This fact would not unduly concern the operators, given the apparently slow opening rate of the automatic emergency feedwater valves.][4]

A second operator now noticed that the second makeup pump had not started, and successfully started pump MU-P1B. He also opened the makeup throttling valve (MU-V16B, Figure II-3) to increase the amount of makeup flow. [This increased flow, along with reduced letdown, apparently overcame the coolant contraction.] Forty-eight seconds after turbine trip, the pressurizer level reached its minimum—158 inches—and then began to increase.

Meanwhile, the condenser hotwell (Figure II-1) was undergoing some expected level fluctuations, first dropping to 21.7 inches, then rising to normal. At 1 minute 13 seconds, the condensate level had reached the high level alarm point at 37.8 inches. [These initial fluctuations were not unexpected.] Unknown to the operators, however, an air line to the hotwell level controller was broken, apparently by a "water hammer" during the initial transient.[5] The operators were unable to regain control of hotwell level.

Very shortly thereafter, the temperature of the water in the RCDT had significantly increased. Unfortunately, the meter showing this temperature is in back of the main control panels and cannot be seen from the normal operating position. [Even if it had been noticed, this information might not have been interpreted as meaning that the PORV was still open. The RCDT liquid was already warm because of leakage and would have become hotter yet when the PORV opened in the initial transient.]

Two minutes after turbine trip, the RCS pressure had dropped to 1600 psig. At this pressure, the engineered safeguards (ES) automatically actuate. The ES system is designed so that when the RCS pressure drops to this level, makeup pumps MU-P1A and 1C will start (if not already operating), makeup pump MU-P1B will trip (if running), and the makeup valves will open to admit the full output of the pumps[6] into the RCS. At TMI-2, the ES system functioned smoothly. Makeup pump MU-P1A was running. When the RCS pressure dropped, makeup pump MU-P1C came on, makeup pump MU-P1B was tripped, and the throttling valves were opened wide.

[If the PORV had not been opened, it could now be expected that increased flow of makeup water into the system would accelerate the rate of rise of the pressurizer level (Figure II-4) and cause the RCS pressure to begin to climb again. Uncontrolled filling of the pressurizer might cause it to fill completely (pressurizer "solid"). Control of RCS pressure is lost with a solid pressurizer, and a very small temperature increase in the totally filled system could cause the pressure to rise to the point where the safety valves would open. If this were to happen, it is possible that the plant would have to be shut down. The safety valves might have to be repaired, because it is not unusual for safety valves to leak after being lifted. Operators are trained to avoid this situation. Operating procedures require them to switch to manual control and reduce makeup as soon as the pressurizer regains a normal level. (This practice is necessitated by a preexisting design deficiency discussed in Section II.C.1.c.)]

The operator bypassed the ES system and reduced the makeup flow, but the pressurizer level continued to increase rapidly. Pressure did not rise and even began to move slightly downward. The reason for the anomaly of rising pressurizer level and decreasing pressure was not recognized by the operators. Trained to avoid a solid pressurizer, they stopped makeup pump MU-P1C and increased letdown flow to its high limit, thereby temporarily arresting the rate of pressurizer level increase.

March 28, 1979—4:06 a.m.

[Loss of coolant through the PORV and excess of letdown over makeup accelerated the decline of RCS pressure. At the same time, very little heat was being removed by the steam generators. About 6 minutes after the turbine trip, the pressure had decreased to the point where some bulk boiling of the reactor coolant could have taken place. At about this same time, the pressurizer level came back on scale.]

[If the pressure dropped low enough for boiling to occur, control of the pressurizer level would have become more difficult. The open PORV would reduce the pressure in the pressurizer steam space. Steam forming elsewhere in the system would force more water through the surge line, raising the pressurizer level. If the RCS pressure rose so that the water was no longer saturated, the steam bubbles in other parts of the system would be condensed, and the pressurizer level would fall. In other words, the pressurizer level would be controlled by steam formation, as well as by the makeup and letdown system. At the same time, it would have been difficult to regain a bubble by using the heaters. The rate of energy loss through the PORV at the system pressure was many times greater than the energy added by the heaters.]

About 6 minutes after the turbine trip, unsuccessful attempts were made to restart the condensate pump CO-P1A and a condensate booster pump. The steam generators were completely dry and steam pressure was dropping rapidly. [Very little energy was being removed through the steam generators. Some energy was being removed by hot fluid flowing out the PORV, but this was not sufficient to prevent an increase in RCS temperature after the makeup flow was reduced.]

The relief valve on the RCDT was opening intermittently after approximately 3 ½ minutes. Operation of this valve allowed the tank to overflow into the reactor building sump. Operation of the relief valve was not noticed by the operators. RCDT parameters are displayed on panel 19a, which is located out of the operator's view. The level in the reactor building sump eventually got high enough to cause a sump pump to be automatically turned on.

[The flow of mixed water and steam out of the relief valve was filling the RCDT at a rate that may have been as high as 20 pounds per second.]

[The reactor building sump is normally pumped to the miscellaneous waste holdup tank. It appears that at the time of the accident, however, the reactor building sump pump was actually lined up to pump into the auxiliary building sump tank—which was already nearly full and had a broken rupture disk.[7] Overflow of the auxiliary building sump tank would cause overflow to go to the auxiliary building sump.]

March 28, 1979—4:08 a.m.

At 8 minutes after turbine trip, the operator discovered that the emergency feedwater block valves were closed and opened them. Opening these valves caused a rapid increase in steam pressure, which had previously dropped when the steam generators boiled dry, and a drop in RCS temperature. Steam generator level, however, did not recover noticeably for another 14 minutes. [The reason for the lag in recovery of the steam generator level is that emergency feedwater is sprayed directly onto the hot tubes and evaporates immediately (Figure II-5). Evaporation raises steam pressure, but

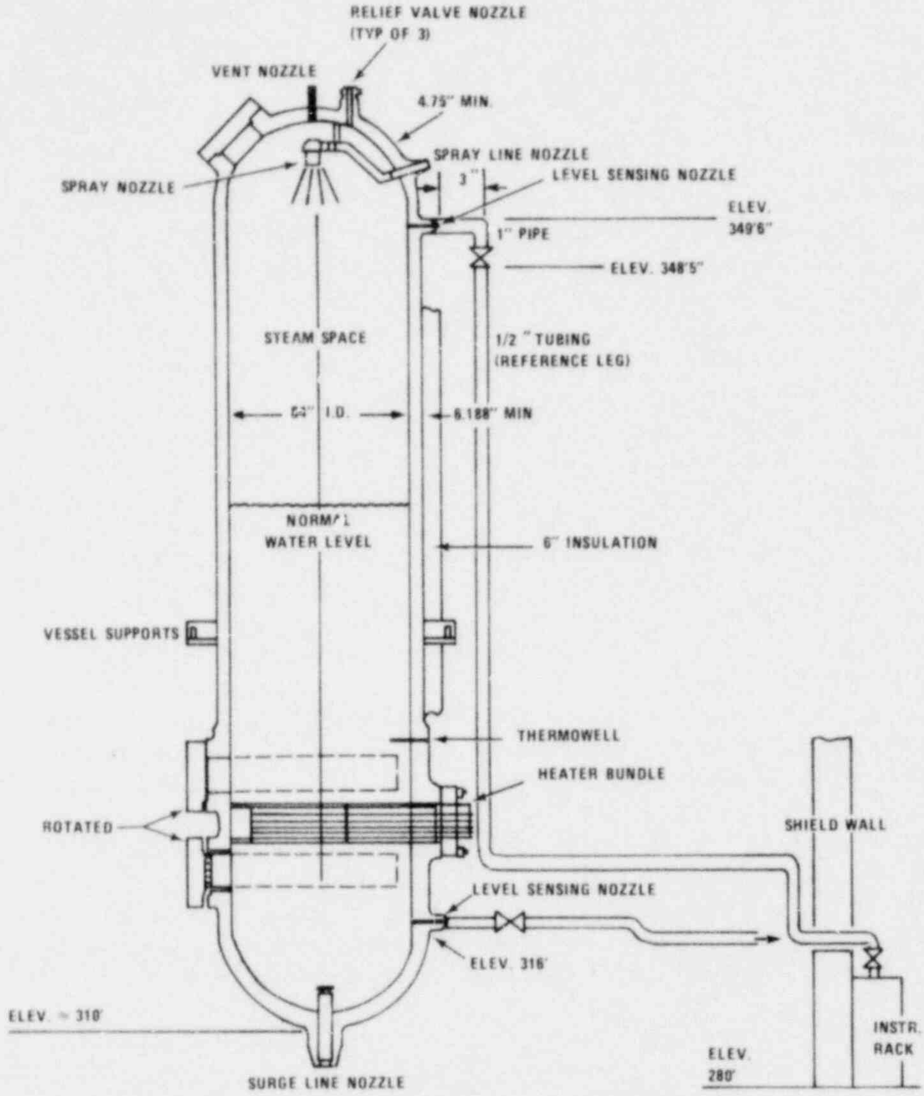


FIGURE II-4. The Pressurizer

The pressurizer controls the reactor coolant system pressure and water level. The pressurizer is connected to the hot leg through the pressurizer surge line. This line has a "V" bend (a "loop seal") so that any steam bubbles in the hot leg would not enter the pressurizer. The pressurizer is partly filled with water. The upper part contains steam. If the pressure drops, the pressurizer heaters are turned on. This raises the water temperature, which causes more steam to form and raises the pressure. If the pressure rises, the spray valve is opened and water from the cold leg (1A) is sprayed into the steam. This condenses some of the steam and reduces the pressure. The pressurizer spray depends on the operation of pump RC-P1A for its operation. The heaters and spray are usually operated automatically. However, they both can also be manually operated from the control room.

The water level in the pressurizer is measured by the level sensing systems. There are three independent level sensors. If the level drops, valve MU-V17 is opened to admit more makeup water. If the level rises, valve MU-V17 is closed. Valve MU-V17 can be operated either automatically, or manually from the control room.

The safety and relief valves and the vent valve are at the top of the pressurizer. One of these is the pilot-operated relief valve, which stuck open in the accident at TMI-2. The purpose of the safety and relief valves is to allow escape of steam if the pressure gets too high. The vent valve is used to bleed off air and other gases when the plant is being started up.

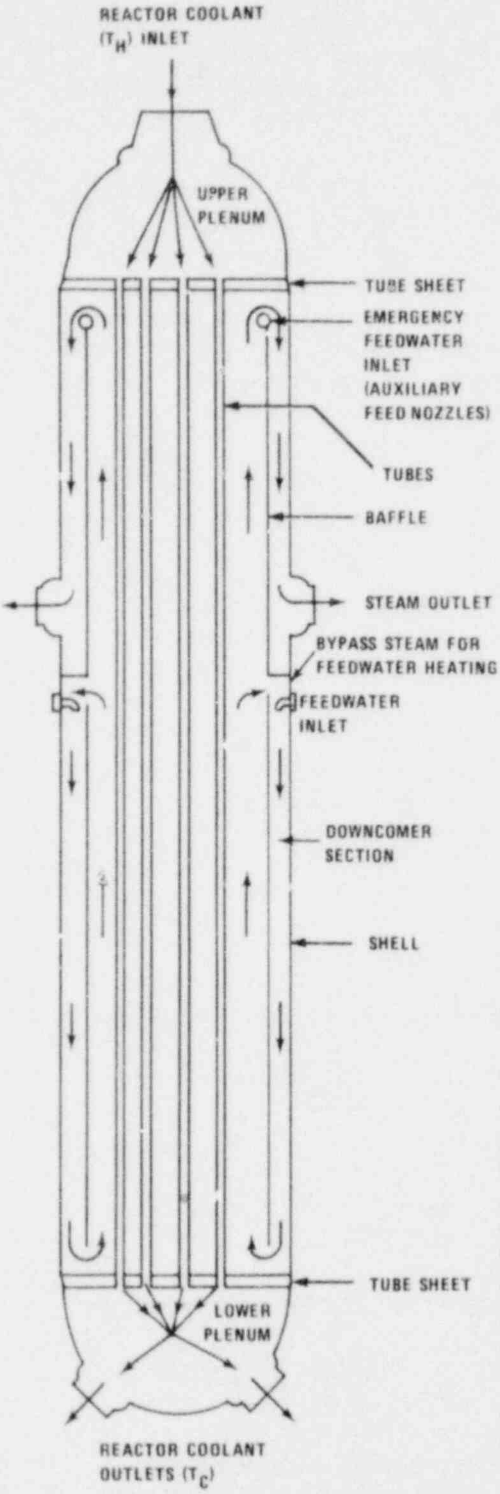


FIGURE II-5. The Once-Through Steam Generator

The water that has been heated in the reactor coolant system circulates through the tubes of the once-through steam generators (OTSGs). In normal operation, the feedwater is sprayed out of the feedwater nozzles into the downcomer. There is steam in the downcomer which raises the temperature of the already hot feedwater almost to the boiling point. The very hot feedwater collects around the tubes near the bottom of the OTSG. The reactor coolant system is maintained at a temperature above the boiling point of water at secondary pressures. Some of the heat is transferred to the feedwater, causing it to boil. The reactor coolant enters at the top, the hottest region. As the steam rises past the very hot tubes near the top, it becomes superheated.

Emergency feedwater is sprayed in through the emergency feedwater nozzles. This water, which is cold compared to normal feedwater, is sprayed directly onto the upper part of the tubes. This action cools the reactor coolant at the top of the tubes and causes it to contract, thereby increasing its density. Because of the increased density the coolant flows down through the tubes, even if the reactor coolant pumps are not operating. This is called natural circulation.

Even if the reactor coolant system is not full, some circulation can take place if the secondary side has a high water level. Steam filling the hot legs can condense in the steam generator as fast as it is being produced in the reactor. This is called reflux flow.

no water collects in the bottom until the tubes are cooled down.] About 7 or 8 minutes after the block valves were opened, sufficient heat had been removed from the system that the reactor coolant became cool enough so that little or no bulk boiling was taking place. [The voids (steam bubbles) in the system should have collapsed; their collapse would make the pressurizer level drop. That the pressurizer level dropped only about 30 inches when the RCS became subcooled shows that the steam bubble voids did not yet constitute a large fraction of the coolant volume.]

The opening of the RCDT relief valve was insufficient to keep the tank pressure from increasing. Fifteen minutes after turbine trip, the rupture disk on the RCDT broke as designed. The tank was now opened directly to the reactor building. [The pressure instrument on the tank actually measures the difference in pressure between the tank and the reactor building. An indication of the high rate of flow through the PORV is that the pressure measuring device indicates some pressure even after the rupture disk broke; i.e., the fluid was rushing in as fast as it could be discharged through the rupture disk opening. This high discharge to the reactor building suggests that a mixture of water and steam was coming out the PORV.] The alarm printer shows that a second sump pump started.[8]

At 19 minutes after turbine trip, the first of many radiation alarms was received from the reactor building air exhaust duct. [It is unlikely that any fuel had failed at this time. What probably happened was that violent boiling and temperature excursions had dislodged a lump of slightly radioactive material (crud) from the exterior of a fuel rod. It is also possible, although improbable, that the combination of reduced coolant pressure and higher than normal coolant temperatures could have allowed some minor cracks to appear in the fuel rod cladding. At any rate, there was some radioactivity in the coolant that came out of the PORV.]

The plant computer measures each parameter, temperature, pressure, level, etc., and then compares the reading for each to a preset alarm value. If the reading is found to exceed acceptable limits, a notation to that effect is typed out on the alarm printer. When the parameter is restored to acceptable limits, another notation is typed. The alarm printer records starting, stopping, or tripping of major equipment.

Operators can communicate directly with the computer through the utility typer. The utility typer can give an operator immediate information about selected parameters; e.g., whether the readings for these parameters are within normal limits. Certain combinations of parameters can be preprogrammed to be printed out in groups of data on request.

At the beginning of the accident, the computer alarm printout was synchronized with real time. The alarm printer can only type one line every 4 seconds, however, and during the accident, several alarms per second were occurring. Within a few minutes, the computer was far behind real time, and the alarms being printed were for events that had occurred several minutes earlier. The operators can bring the computer up to date, but only at the cost of clearing all alarms awaiting printout from memory. The computer was brought up to date during the course of the accident, and as a result, nearly 1½ hours of historical data have been lost. Also, when the computer alarm printer was brought up to date, real time information was available for only a few minutes, then the computer began to lag again. [Computer alarm data, therefore, was of very little value to the operators, although it has been useful in reconstructing the accident sequence.]

[Data of value to the operators were presented by meters, strip charts, multipoint recorders, status lights, and alarm annunciators. So many annunciators were lighted, however, that their value to the operators was probably diminished.] The annunciators for RCDT alarms, like the RCDT gauges, cannot be seen from the normal operating position.

March 28, 1979—4:25 a.m.

About 25 minutes after turbine trip, the operators received a computer printout of the PORV outlet temperatures. [The high temperature—285°F—was not perceived by the operators as evidence that the PORV was still open. When the PORV opened in the initial transient, the outlet pipe temperature would have increased even if the PORV had closed as designed. The operators supposed that the abnormally slow cooling of the outlet pipe was caused by the known leak in the relief or safety valves. Actually, sufficient evidence of the failure of the PORV to reclose was now available: the rapid rise in RCDT pressure and temperature, the fact that the rupture disk had blown, the rise in reactor building sump level (with operation of the sump pumps), and the continuing high PORV outlet temperature. The PORV outlet temperature was read again at 27 minutes after turbine trip. The evidence of an open valve, however, was not interpreted as such by the operators. Many of the instruments were behind the control panels, out of the immediate sight of the operators. It appears that at 30 to 40 minutes, the operators deliberately went behind the control panels to read the instruments, but then failed to recognize the significance of the readings.]

March 28, 1979—4:30 a.m.

At approximately 30 minutes, an auxiliary operator noticed that the suction line to condensate booster pump CO-P2B was leaking. [He believed the leak to have been caused by the "water hammer" at the time of the accident.] The pump was isolated by closing the suction valve.

Another auxiliary operator noticed that the reactor building sump pumps were on and that the meter showing the depth of water in the reactor building sump was at its high limit (6 feet). The background radiation in the auxiliary building had increased. (Although it was believed that the reactor building sump pumps were discharging to the miscellaneous waste holdup tank, the level in the holdup tank had not changed. On the orders of the control room operator, with the shift supervisor's concurrence, the operator shut off the sump pumps.)

The auxiliary operators, after considerable difficulty, manually opened the condensate polisher bypass valve. An air line to the condensate reject valve was found to be broken. [This broken air line was apparently the cause of operators' inability to control hotwell level.]

Operators were still encountering problems with the condensate system. They were also beginning to have problems with the reactor coolant pumps. [The operators now could have realized that what was occurring was not a normal turbine and reactor trip. They continued to be puzzled by the high pressurizer level and decreasing pressure, however, and no one took the time to investigate the RCDT gauges.]

[The reasons for the problems with the reactor coolant pumps were that steam bubble voids had formed throughout the system when the pressure was below the saturation pressure. The system pressure at the coolant pump inlets is required to be significantly above the saturation pressure. This requirement is called the net positive suction head (NPSH) requirement. If the NPSH requirement is not met, vapor bubbles will form in the lowest pressure regions on the suction side of the pumps. The formation of vapor bubbles, called cavitation, could cause severe pump vibration, which in turn could damage the seals and might even damage the attached piping. Operators ignored the NPSH requirement and left the reactor coolant pumps operating as long as possible. Had they not done this, more severe core damage could have occurred. As long as the pumps provided circulation, even of froth, the core was being cooled. As soon as all the pumps were stopped, circulation of coolant decreased drastically, because natural circulation was blocked by steam.] [Some circulation can be maintained by refluxing. In this type of flow, the water boils in the reactor vessel, and the steam flows through the hot legs, is condensed in the steam generators, and flows (as liquid water) back to the reactor vessel. For refluxing to occur, a spray of emergency feedwater must be hitting the tubes, or the water level on the secondary side of the steam generators must be higher than the water level on the primary side and the temperature significantly cooler. The level in steam generator A was low (about 30 inches). The steam pressure, hence the temperature, on the secondary side was not much lower than that on the primary side. Reflux circulation, therefore, would probably not have been effective.]

[Effective cooling might have been maintained if the steam generators had been filled to a high level and if the steam pressure had been kept significantly lower than the RCS pressure.]

[The voids in the system also caused the neutron detectors outside the core to read higher than expected. Normally, water in the downcomer annulus (Figure II-6), outside the core but inside the reactor vessel, shields the detectors. Because this water was now frothy, however, it was not shielding the detectors as well as usual. Not realizing that the apparent increase in neutrons reaching the detectors was caused by these voids, operators feared the possibility of a reactor restart. Although it can now be seen that their fears were unfounded, at the time they were one more source of distraction.]

The emergency diesel generators had been running unloaded ever since ES actuation. These diesels cannot be run unloaded for long without damage. They cannot be shut down from the control room, but must be locally tripped. Once the diesels are stopped, the fuel racks must be reset so the diesels can be automatically restarted. At 30 minutes after the turbine trip, the operator sent a man to the diesels to shut them down. The fuel racks, however, were not reset. Failure to reset these racks could have had serious consequences if offsite power had been subsequently lost, because radioactivity restricted access to the diesels. [This is discussed further in Section II.C.1.c.]

[Voiding throughout the system and the deteriorating performance of the reactor coolant pumps decreased the efficiency of the heat transfer through the steam generators. The rate of boiling was lower than usual, and operators found it difficult to keep the water level from creeping up. At 26 minutes, the steam driven emergency feedwater pump had stopped, and at 36 minutes, one of the electric pumps had stopped thereby throttling the flow of feedwater. At 50 minutes, operators were

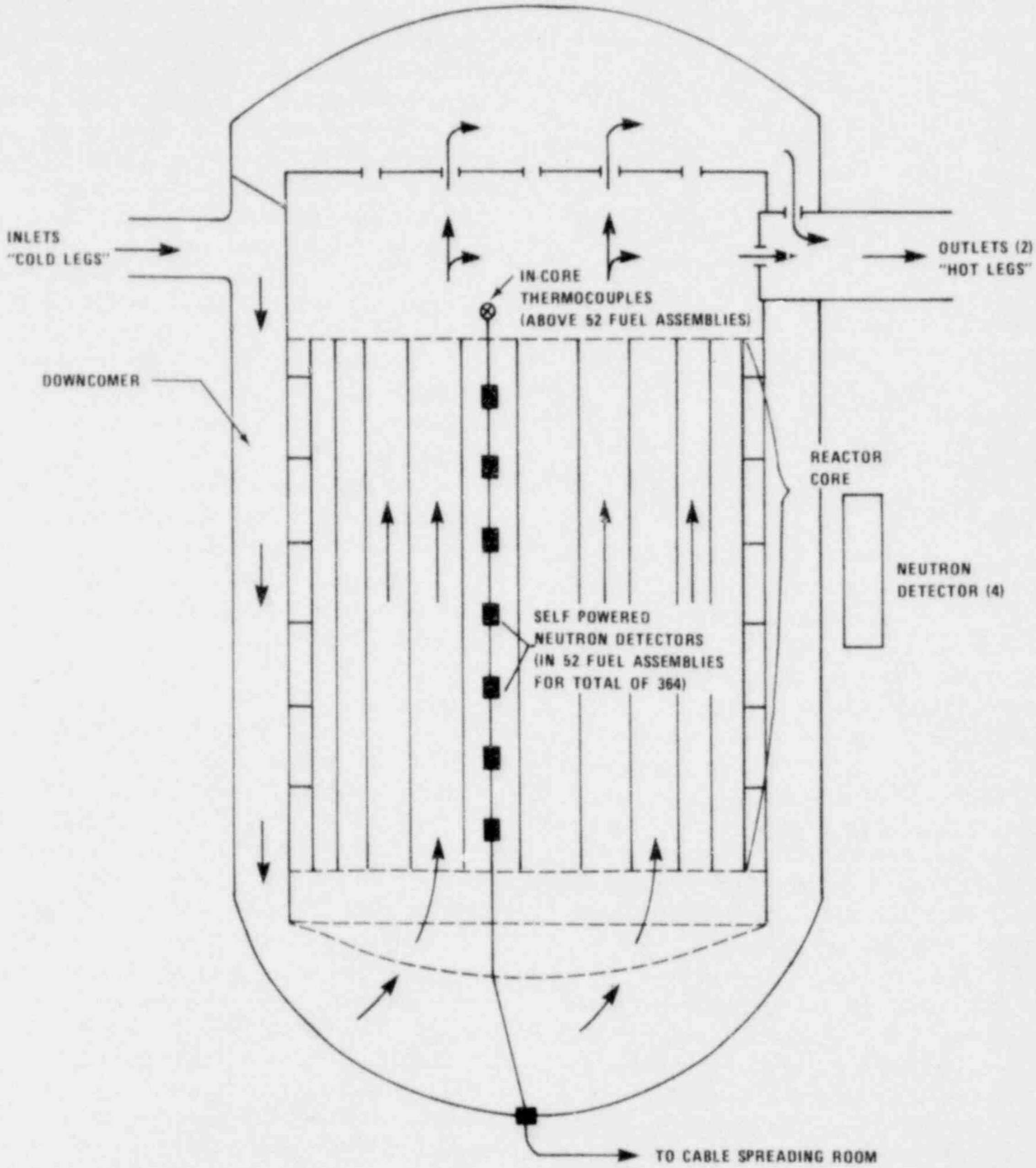


FIGURE II-6. The Reactor and Reactor Pressure Vessel

The reactor is contained in the reactor pressure vessel. Water is pumped in through the four cold legs (inlets), and flows down through the downcomer annulus. At the bottom of the vessel the flow is reversed and the water flows upward through the core. The temperature of the water is raised as it flows past the hotter fuel rods. Thermocouples–temperature measuring devices–are installed just above the fuel rods. These devices are not in contact with the rods and, therefore, measure the temperature of the fluid that has just left the core area. Water then flows out through the two hot legs (outlets) to the steam generators. Neutron detectors are located inside the core. In addition, there are two sets of detectors outside the reactor pressure vessel. The source range detectors read relatively low neutron levels. Before the upper limit of the source range is reached, the intermediate range detectors pick up and continue recording higher levels than the source range can read. No instruments are provided for reading the level of water in the reactor vessel.

still having trouble stabilizing the steam generator level, as well. While steam generator B was still filling, the level in A was decreasing.]

[The condition of the condensate system continued to deteriorate. Normally, the heat removed from the primary system via the steam generators is ejected to the atmosphere via the main condenser and cooling towers. The condensers must be maintained at a vacuum to operate efficiently, however, and condenser vacuum was gradually being lost. If condenser vacuum were to drop below acceptable levels, the condensate system would be automatically tripped and an uncontrolled dump of secondary steam to the atmosphere would occur (Figure II-7). To prevent loss of vacuum, operators deliberately shut down the condensate system 1 hour after the turbine trip and sought to maintain control over steam pressure by controlling the atmospheric steam dump.

March 28, 1979—5:00 a.m.

[At the end of the first hour, the situation with which the operators were confronted had severely deteriorated: pressurizer level was high and was only barely being held down, the reactor coolant pumps were still operating but with decreasing efficiency, the condensate system was no longer operable, the reactor building pressure and temperature were slowly increasing, the alarm computer lagged so badly that it was virtually useless, and radiation alarms were beginning to come on.]

At 1 hour 2 minutes, the alarm printer failed, and alarms were shifted to the utility printer for the next 11 minutes. Alarms from 1 hour 13 minutes to 2 hours 37 minutes are irretrievably lost.

At 1 hour 11 minutes, operators initiated reactor building cooling. Their action soon halted, and eventually reversed, the rise in reactor building temperature and pressure. [That this step was considered necessary by the operators suggests that they were aware of increasing temperature and pressure.]

[The increasing temperature and pressure should have been a good indication that a small-break LOCA was in progress. In fact, if the air cooling had not been initiated, the reactor building would probably have been isolated (sealed off) shortly after this time.]

March 28, 1979—5:13 a.m.

[The operation of the reactor coolant pumps was seriously impaired. High vibration, low flow, low amperage, and inability to meet NPSH requirements led the operators to start shutting down pumps.] At 1 hour 13 minutes, reactor coolant pump RC-P1A was stopped, and pump RC-P1B was stopped a few seconds later. [The reason for stopping pumps in the B loop is that power for the pressurizer spray comes from the A loop. The operators were hopeful of regaining control of pressurizer level and wanted to keep the pressurizer spray operable as long as possible.]

Shutting down two pumps reduced the flow of coolant through the reactor core. [Apparently, there was still enough mass flow in the steam/water mixture to provide cooling, but not as much cooling as that provided when a large volume of void-free water was circulating. There is no firm evidence of overheating at this time. The open valve was reducing the inventory of water in the RCS, though, and the pressure was getting lower. Water continued to boil to remove decay heat; this boiling increased the amount of steam in the system and further impeded circulation.]

A few minutes later, analysis of a sample of reactor coolant indicated a low boron concentration. [This finding, coupled with that of apparently increasing neutron levels, increased operators' fears of a reactor restart. As explained earlier, the supposed increase in neutron levels was spurious, appearing on the detector only because bubbles in the downcomer were allowing more neutrons to reach it. It is believed that the apparently low boron level was also spurious, that condensed steam diluted the sample. Neither explanation appears to have been considered at the time. The operators did apparently distrust the low boron concentration, and took steps to get a second sample.]

March 28, 1979—5:20 a.m.

At 1 hour 20 minutes, an operator had the computer print out the PORV and pressurizer safety valve outlet temperatures. The temperature of the PORV outlet was 283°F. The temperatures on the two safety valve outlets were 211°F and 219°F. [That there had been essentially no change in temperature in 55 minutes should have alerted the operators that the PORV valve had not closed; operators could have confirmed this by checking the RCDT and reactor building parameters or by closing the block valve to see if the outlet temperatures changed.]

Also at 1 hour 20 minutes, the letdown line radiation monitor began to increase. It increased steadily to the full-scale reading. [The increase in radioactivity cannot definitely be attributed to fuel failure. Certainly, it was not attributed to this at the time. The letdown monitor was notoriously sensitive, so that even minor changes in radioactivity would cause great variations in the reading.]

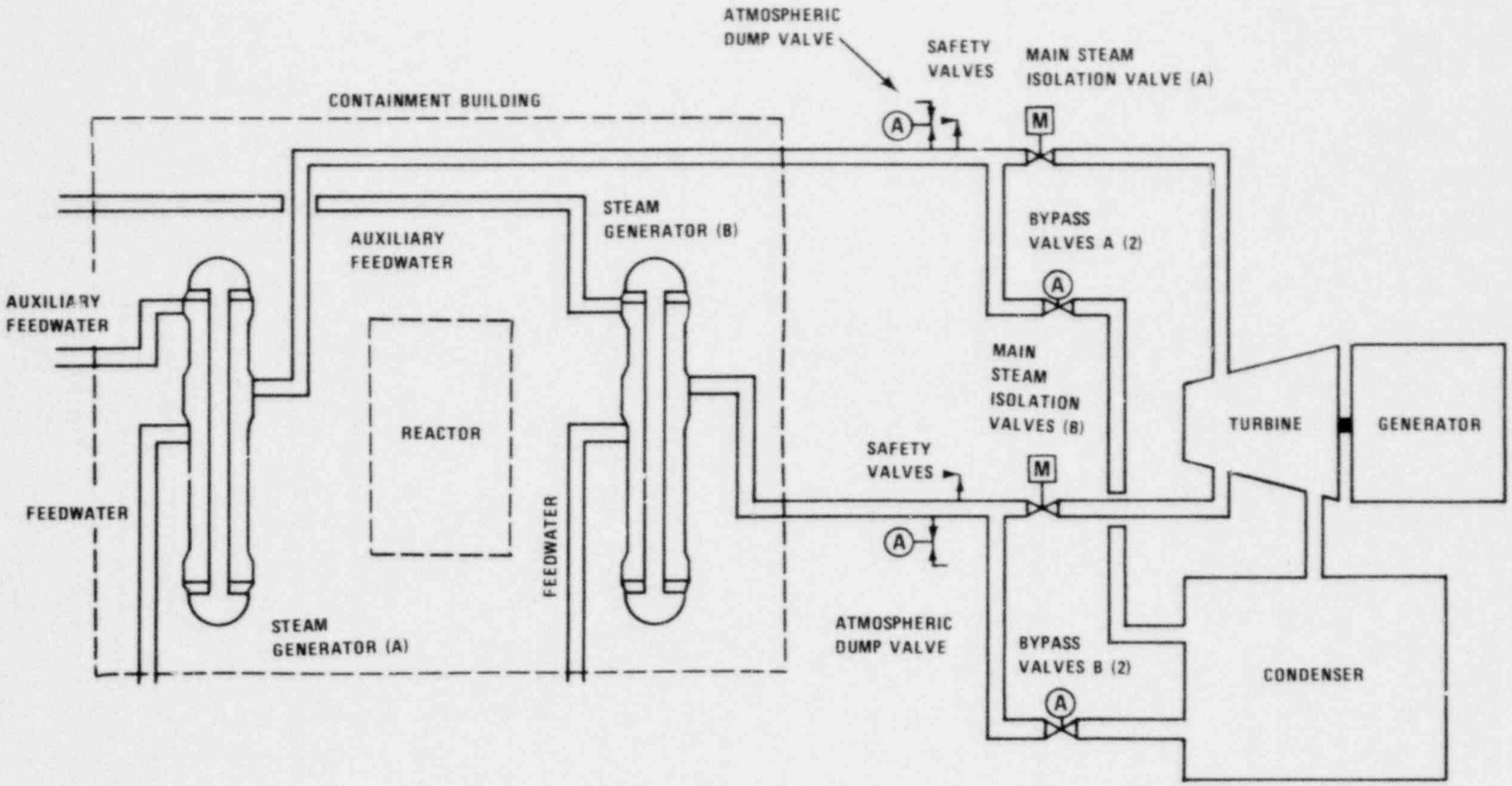


FIGURE II-7. Main Steam Lines and Dump Valves

Steam is delivered to the turbine in normal operation. When the turbine is tripped, the steam is preferably passed to the condenser via the bypass valves. If the condenser is not operating, steam can be released to the atmosphere through the atmospheric dump valves. Either the bypass valves or the dump valves can be automatically controlled to maintain steam pressure at a preset valve.

[The low steam pressure in steam generator B and the increase in reactor building pressure were believed to be caused by a leak from the steam generator.] At 1 hour 27 minutes, steam generator B was isolated (taken out of service). [With hindsight, it can be seen that the low pressure was simply caused by steam bubbles and a reduction of heat transfer in the B loop following stoppage of the pumps. A small change in building pressure was noted when the steam generator was isolated. The occurrence of this change at this time was probably coincidental.]

March 28, 1979—5:30 a.m.

At 1 hour 30 minutes, the apparent neutron level increased again. An RCS sample showed even lower boron concentration and increased radioactivity. [The activity was probably due to crud.]

The temperature of the RCS coolant in all primary system piping had been slowly increasing. Eventually, the primary side of steam generator A got hot enough so that more steam was produced on the secondary side, and the steam pressure began to rise. The increased steam production had two side effects: (1) the water level on the secondary side dropped and the steam generator boiled dry for the second time, and (2) the increased heat removal brought the RCS temperature down again.

[The efficiency of the reactor coolant pumps was still decreasing, and at 1 hour 37 minutes, the frothy mixture became too light to circulate. Separation of the froth would have sent the steam to the high parts of the system, while water collected in the low parts. An analogy is a kitchen blender with the bowl half full of water. With the blender at high speed, enough air bubbles are whipped into the water so that the bowl is full. If the speed drops, the air bubbles are lost and the lower half of the bowl is solidly filled with liquid water. This was reflected in the behavior of the neutron instrumentation. Apparently the downcomer, which had been previously filled with froth, now filled with water. The increased shielding stopped neutrons from reaching the detector and the apparent neutron level dropped by a factor of 30.]

Operators recognized that steam generator A was dry, and in an attempt to regain water level, they increased feedwater flow.

March 28, 1979—5:41 a.m.

At 1 hour 41 minutes, both remaining reactor coolant pumps (RC-P1A and 2A) were stopped because of increasing vibration and erratic flow. [The only heat transfer through the steam generators was now achieved by reflux flow (Figure II-5). This was inadequate for core cooling. It is now believed that the core was drying out. The operators were hoping to establish natural circulation in the primary system. Natural circulation was blocked by steam, and refluxing would be ineffective because the secondary temperature was nearly as high as the primary temperature.]

[The pressurizer is at a higher level than the reactor. It was assumed that the presence of water in the pressurizer meant that the core must be covered. Actually, because the PORV was open, pressure in the upper part of the pressurizer was reduced. The strong boiling that was occurring in the core, however, caused more steam to go into the upper part of the reactor vessel, and the pressure there was increased. The difference of pressure forced the water level higher in the pressurizer than in the reactor vessel.]

[Previous reports have alluded to a "loop seal," thus giving the false impression that the piping configuration alone somehow created this difference of level. Even with the loop configuration, to maintain a higher level in the pressurizer when the water in the pressurizer is saturated, a higher pressure is required in the reactor than in the pressurizer. If the pressures are equalized with the hot leg voided, the saturated pressurizer water level would drop to the level of the connection of the pressurizer surge line into the hot leg. Subcooled water could be maintained at a higher level. During most of the accident, the water in the pressurizer was slightly subcooled or saturated. During the time that the surge line was uncovered, the water in the pressurizer was subcooled. It was the combination of loop seal and temperature that kept the level high, rather than the loop seal alone.]

March 28, 1979—5:42 a.m.

At 1 hour 42 minutes, the decreasing level in the reactor vessel again reduced the shielding of the neutron instrumentation, and the apparent neutron count increased by about a factor of 100. Emergency boration was commenced to avert a restart. [Actually, a restart was impossible because of the partial emptying of the core, but no one recognized this. A further discussion of this topic is given in Section II.C.2.b.]

The hot-leg temperature now became decidedly higher than the cold-leg temperature. Superheated steam was present in the hot leg. [The superheating of the hot leg showed that a fair amount of the core was uncovered. It is impossible to superheat the hot leg without uncovering the core.]

Although none of the instrumentation directly indicates to the operators that the saturation temperature has been reached or exceeded, a copy of tables that show saturation temperatures as a function of pressure (the "steam tables") was available to them. [Apparently, however, operators did not draw the inference from the superheated hot leg concerning the core.]

[Up to this time, it might have been possible to salvage the situation without extensive core damage. If the PORV had been closed and full makeup flow had been instituted, it might have been possible to fill the system enough so that a reactor coolant pump could be restarted. As the uncovering of the core became more extensive, the opportunity to reverse the tide dwindled.]

[The upper part of the core was now uncovered. The steam rising past the fuel rods gave some cooling, but not nearly as much as when they were covered with water. The decay heat—about 26 MW—was higher than the heat removed, so the fuel temperature increased.]

[The fuel rods are clad with Zircaloy, an alloy of zirconium. Zirconium reacts with water to form zirconium dioxide and hydrogen. At operating temperatures, this reaction is extremely slow and does not represent a problem. At higher temperatures, however, the reaction goes faster. It is believed that the temperature of the fuel rods reached a point at which the reaction occurred rapidly, producing significant amounts of hydrogen. Furthermore, the reaction itself releases heat. Heat released from the reaction would have caused the cladding to become hotter, driving the reaction faster.]

[As long as the upper part of the system contained only steam, the bubble could be condensed (collapsed) by increasing the pressure or decreasing the temperature. However, with large amounts of hydrogen in the system, these measures would reduce the size of the bubble but could never collapse it. The accident could not now have been reversed by simply closing the PORV and increasing makeup.]

March 28, 1979—6:00 a.m.

At 2 hours into the accident, the pressure in loop A was 735 psig. At this pressure, the saturation temperature (the boiling point) is about 511°F. The loop A hot-leg temperature was actually 558°F—definitely superheated. Shortly after 2 hours, the narrow range hot-leg temperatures went offscale high, and cold-leg temperatures went offscale low. When this happened, the hot-leg temperature read constantly at the upper limit (620°F), and the cold-leg temperature read constantly at the lower limit (520°F).

The wide range temperature measurements were still available, although the narrow range temperatures can be read more accurately and the operators are in the habit of using them exclusively. One meter shows average temperature, which is actually an average of the narrow range indications. Average temperature shown at this time was 570°F, the average of the constant readings of 520°F and 620°F. [This steady average temperature evidently convinced the operators that the situation was static. The restricted range of these indicators and their influence on the accident are considered further in Section II.C.1.e.]

[The operators now knew that there was a problem. Natural circulation had not been established, and they had been forced to turn off the last RCP. Apparently, however, no one knew just what was wrong.]

At 2 hours 15 minutes, the reactor building air sample particulate radiation monitor went off scale. [This was the first of many radiation alarms that could definitely be attributed to gross fuel damage.]

At some time before this incident occurred, the core flood tanks had been valved off. If the pressure had dropped to the nominal nitrogen pressure in the core flood tanks (about 600 psig) with valves open, the tanks would have discharged water into the RCS. [The high pressurizer levels had convinced the operators that there was an adequate amount of water in the RCS, and it was thought to be completely unnecessary to allow the core flood tanks to operate. Sometime later, the core flood tank valves were reopened.]

March 28, 1979—6:18 a.m.

At 2 hours 18 minutes, the PORV valve outlet temperatures were again reviewed. A shift supervisor who had just come into the control room isolated the PORV valve by closing a block valve (RC-V2) in the same line. [Apparently, he did this to see whether it would have an effect on the anomaly of high pressurizer level and low system pressure.] The reactor building temperature and pressure immediately began to decrease and the pressure of the RCS increased. The shift supervisor who had closed the block valve immediately recognized that a leak had been stemmed. Others in the control room, however, were apparently slow in recognizing that the PORV had been leaking consistently for over 2¼ hours and that leakage of this valve had resulted in a small-break LOCA.

[Leakage through the PORV had now been stopped, but there was still no way to get rid of the decay heat, because there was virtually no circulation through the steam generators. The once-through steam generator (OTSG A) had 50% cold water, which would have been adequate if there had been circulation. The situation was in some ways worse than it was before the valve was closed. While the PORV was open, a considerable amount of energy, as well as mass, was being dumped into the reactor building.]

During this period of probable core damage, there was virtually no information on conditions in the core. Incore thermocouples (temperature measuring devices), which measure reactor coolant temperature at the exit from the core, could measure only up to 700°F. This limit is imposed by the signal conditioning and data logging equipment, not by the instruments themselves. When a temperature reading is off scale, the computer prints out question marks: "?????". The operators, however, cannot tell whether such an indication on the computer means that the readings are outside the scale limits, or whether there has been some other malfunction and the readings are simply not being taken correctly.

Many radiation monitors began to go offscale high. [This is an indication of severe core damage. The zirconium dioxide resulting from the same reaction that gives rise to the hydrogen is much more frangible than Zircaloy. The intense boiling could have caused shattering of much of this material; and the loss of cladding integrity, coupled with the high temperatures, could have allowed the more volatile radioactive substances in the fuel to escape into the reactor coolant.]

March 28, 1979—6:46 a.m.

At 2 hours 46 minutes, an unsuccessful attempt was made to start reactor coolant pump RC-P1A, and 2 minutes later, an equally unsuccessful attempt was made to start pump RC-P2A. At 2 hours 54 minutes, pump RC-P2B was started after operators bypassed some interlocks. This pump ran normally for a few seconds, then the flow dropped to zero and the pump ran at very high vibration levels; 19 minutes later it was stopped again.

At 2 hours 47 minutes, the computer-printed alarms were brought up to date. As previously explained, bringing the alarms up to date erases all alarms waiting for printout. The alarm summary was at this time 1 hour 34 minutes behind, so that alarms from 1 hour 13 minutes to 2 hours 47 minutes were irretrievably lost. The advantage gained was that operators were provided with current alarm data. Within a very short time, however, the computer was again hopelessly behind.

The problems with the condenser hotwell level control were finally solved at 2 hours 50 minutes. The broken air line to the reject valve was repaired, the valve now operated properly, and the condensate hotwell was pumped down to its normal level.

March 28, 1979—6:54 a.m.

At 2 hours 54 minutes, the pressurizer heaters tripped. Throughout the remainder of March 28, operators were plagued by difficulties in attempting to keep the pressurizer heaters in operation. The heaters are necessary for maintaining control of the pressurizer pressure, and the intermittent loss of the heaters was keenly felt. [It was believed at the time that the heaters were tripping because of the hot, humid atmosphere in the reactor building. The shift foreman went to the pressurizer heater control cabinet to check the circuit breakers. The circuit breakers were actually closed, but vent fans in the area had tripped because of high temperatures. The fans were restarted.

[The attempted starts of the reactor coolant pumps had not established circulation in the reactor coolant system. It appears, however, that a slug of water was forced into the downcomer by the momentary running of pump 2B. Flow meters indicated that about 1000 to 1100 cubic feet of water were moved in the 9 seconds of flow. This could have covered the core or could have flowed into the other RC pump cold legs that were nearly empty.]

[The flow of water resulted in a sudden drop in the indicated neutron levels, but rapid boiling soon reduced the water level and the levels rose again. The boiling also caused a rapid pressure rise and probably did considerable damage to the brittle oxidized cladding.]

Several high radiation alarms within the plant had now been received. At 2 hours 56 minutes, the shift supervisor declared a site emergency and began to notify local authorities. By now the control room was full of people, including Metropolitan Edison management and technical people. One estimate is that there were as many as 50 to 60 people present. Another report, however, says 18 to 20 people were in the control room. [Many of the actions taken were at the direction of the Metropolitan Edison emergency director. For simplicity, the term "operator" is used in this report to indicate actions taken from the control room, even though the operators themselves may not have been taking some actions on their own initiative.]

The letdown sample lines had now been reported to have an extremely high radiation level (600 r/h), and the auxiliary building was evacuated. An attempt was being made to secure another reactor coolant sample.

March 28, 1979—7:00 a.m.

[By 3 hours after the turbine trip, the situation appears in hindsight to have become quite grave. It should have been obvious that there was no circulation of reactor coolant. The abortive attempts to start reactor coolant pumps and the attempts to secure natural circulation by a high water level in the steam generator indicate that this was suspected at the time. Most incore thermocouples were reading off scale. The hot-leg temperatures were nearly 800°F. This superheating of the hot leg indicates both that the hot leg had virtually no liquid water in it and that at least the upper part of the core was dry. The many high radiation alarms indicate that extensive fuel damage had occurred.]

RCS pressure had been moving generally downward. There had been a slight recovery in pressure just before the last pumps were shut down. After the pumps were stopped, though, the pressure dropped rapidly from about 1140 psig to about 600 psig. Just before closure of the block valve, the pressure began to rise and when RC-P2B was turned on, the pressure rose rapidly from 1200 to 2200 psig.

At the same time, the pressurizer went off scale (above 400 inches). At this time, the loop B hot-leg temperature exceeded the scale limit of the wide range instrumentation (800°F).

At 3 hours, the condenser vacuum pump exhaust radiation monitor was showing increased radiation levels. A leak in steam generator B had been previously suspected, and the increased level of radiation seemed to confirm this. At 3 hours 4 minutes, the turbine bypass valves from steam generator B and the auxiliary feedwater valves to this generator were closed. This completely isolated the steam generator from the condensate system.

The external neutron instrumentation was showing an increase in apparent neutron levels. [This was an indication of the dropping water level in the reactor vessel.]

March 28, 1979—7:12 a.m.

At 3 hours 12 minutes, the PORV block valve was opened in an attempt to control RCS pressure. The opening of the valve caused a pressure spike in the RCDT, an increase in reactor building pressure, and an increase in the valve outlet temperature.

Reactor coolant pump RC-2B had been operating essentially without flow since being started at 2 hours 54 minutes. Because of low current, zero flow, and a high vibration level, the pump was shut down at 3 hours 13 minutes.

At 3 hours 20 minutes, the ES were manually initiated by the operator. This was quickly followed by a drop in pressurizer level. [The reason for actuation of the ES was the rapidly dropping RCS pressure.] (The ES would have actuated automatically at about the same time.) Makeup pump MU-P1C started and the makeup valves opened fully. RCS temperature dropped rapidly as the cold water flooded in. [It is believed that the sudden admission of cold water to the extremely hot core probably caused additional major damage to the core because of thermal shock. The external neutron indicators dropped suddenly, indicating a rapid change of level in the downcomer. The water added should have ensured that the coolant level was above the core height.]

Almost immediately, many radiation monitors registered alarms. The control building, except for the control room itself, was evacuated. [These radiation alarms are a good indication that severe core damage occurred. Apparently, the brittle oxidized cladding was shattered by the sudden admission of cold water, so that the fuel pellets were no longer held in their original position. This sudden rearrangement of the core may have permitted the volatile fission products to enter the coolant; these could later have streamed out of the open PORV into the reactor building.]

March 28, 1979—7:24 a.m.

At 3 hours 24 minutes, a general emergency was declared on the basis of the many radiation alarms.

The borated water storage tank (BWST) low level alarm was received at 3 hours 30 minutes. There were still 53 feet of water in the BWST. [That the level was falling, however, caused concern. Additional ES actuations could cause all the water in the BWST to be used up, and the highly radioactive water in the reactor building sump would have to be used for high pressure injection. The HPI pumping system would become radioactive, which could cause grave problems if repairs became necessary. There was thus an inclination to use ES as little as possible (high pressure injection water is taken from the BWST).] ES was reset and makeup pump MU-P1C was stopped.

At the same time, the PORV block valve was shut. Closing this valve, with pump MU-P1A still running, caused a rapid increase in pressurizer level.

March 28, 1979—7:35 a.m.

At 3 hours 35 minutes, it was noted that the auxiliary building basement was flooded. It will be recalled that the rupture disk on the auxiliary building sump tank had previously broken, so that much of the water pumped from the reactor building had wound up in the auxiliary building basement. High radiation readings were found in many areas of the auxiliary building.

The PORV block valve was reopened at 3 hours 41 minutes. Thirty seconds earlier, there was a sudden jump in the source range neutron detectors. [The jump may have been due either to water in the downcomer flashing into steam or to a disturbance of the core geometry. The change in the source range is believed to be due to an event internal to the core representing a change of geometry—unrelated to external events.]

At 3 hours 56 minutes, there was an ES actuation because of high reactor building pressure (the setpoint for actuation is 4 psig). When the ES actuated, the reactor building was automatically isolated. Isolation means that valves in all systems not absolutely essential for cooling the core are closed and the systems are shut down. Makeup pump MU-P1C started, and the intermediate closed cooling pumps were tripped automatically. The intermediate closed cooling pumps are needed for letdown and seal cooling, so the building isolation and ES were defeated 4 minutes after actuation and the pumps were restarted. [The delay in building isolation is discussed in Section II.C.1.c.]

March 28, 1979—8:00 a.m.

About 4 to 4 ½ hours into the accident, incore thermocouple temperature readings were taken off the computer; many registered question marks. Shortly after, at the request of the station superintendent, an instrumentation control engineer had several foremen and instrument technicians go to a room below the control room and take readings with a millivoltmeter on the wires from the thermocouples. The first few readings ranged from about 200°F to 2300°F. These were the only readings reported by the instrumentation control engineer to the station superintendent. Both have testified that they discounted or did not believe the accuracy of the high readings because they firmly believed the low readings to be inaccurate. In the meantime, the technicians read the rest of the thermocouples—a number of which were above 2000°F—and entered these readings in a computer book which was later placed on a control room console. The technicians then left the area when nonessential personnel were evacuated. [We have not developed evidence that their superiors were conscious of these additional readings on March 28.]

An attempt was made to start reactor coolant pump RC-P1A at 4 hours 18 minutes. Current and flow were monitored to see if the pump could be operated. The starting current was normal, but current quickly dropped to a low value and flow dropped to zero. This indicates that a slug of water may have been forced through, but the pump was not working continuously. It was stopped a minute later.

March 28, 1979—8:18 a.m.

Both makeup pumps (MU-P1A and 1C) were stopped at 4 hours 18 minutes. Two unsuccessful attempts were made to restart pump 1A. The control switch was then put in the "pull-to-lock" position. This completely defeated automatic starts of the pump. [The reasons for doing this were apparently the difficulties experienced in attempting to restart the pump, and a desire to avoid the possibility of having the pump come on if ES actuated. The pressurizer indicated full, and the operators were concerned about full high pressure injection flow coming on with an apparently "solid" system.]

[Actually, a very large part of the RCS was filled with steam and gas, and the system was far from being solid. This condition could have been recognized from the fact that the RCS hot legs were superheated. There was no danger of overpressurizing the RCS by high makeup flow.]

There was, in fact, another ES actuation at 4 hours 19 minutes. Decay heat pump DH-P1A started, and the intermediate closed cooling pump tripped, but makeup pump MU-P1A did not start. The ES actuation was immediately defeated and the intermediate closed cooling pump was restarted. Only one channel had been actuated, but the fact that one channel was defeated satisfied the "two out of three" logic which is required for ES actuation.

Makeup pump MU-P1B was started by the operator at 4 hours 22 minutes, and MU-P1C at 4 hours 27 minutes.

Problems in the condensate system were continuing. The condensers had been steadily losing vacuum. It was also necessary to maintain steam to the main turbine seals in order to operate the condenser at a vacuum. When main steam is not available, seal steam is provided by the oil-fired auxiliary boiler, which is shared by both TMI units. The auxiliary boiler broke down, so that seal steam could not be maintained, and it was necessary to shut down the condensate system completely.

March 28, 1979—8:31 a.m.

At 4 hours 31 minutes, the vacuum pumps were stopped and the condenser vacuum was broken. As a result, steam was now being dumped to the atmosphere. The letdown temperature alarmed high because of the frequent stoppages of the intermediate closed cooling pump. The high temperature alarm cleared at 4 hours 36 minutes.

[Only a small amount of heat could be removed by the steam generator because the upper part of the RCS was filled by a steam-gas mixture. This drastically cut flow on the primary side. The water level on the secondary side was rising because more water was coming in as feedwater than was leaving as steam. At 4 hours 42 minutes, emergency feedwater pump EF-P2A was stopped.]

March 28, 1979—9:00 a.m.

At 5 hours after turbine trip, the RCS pressure was reading 1266 to 1296 psig, the cold legs were subcooled, and the hot legs were superheated. Many radiation monitors were off scale. The containment dome monitor showed a very high reading of 6000 r/h. As it was apparent that conditions were far from satisfactory, the decision was made to repressurize. At 5 hours 18 minutes, the PORV block valve was closed.

At 5 hours 24 minutes, there was yet another ES actuation on high reactor building pressure. This was immediately defeated. Decay heat pump DH-P1A had already been stopped and put in the "pull-to-lock" position. The intermediate closed cooling pump tripped again, but was immediately restarted.

The diesel engines that operate the emergency generators had been stopped at 30 minutes after the turbine trip. These diesels provide an emergency electrical supply for the ES in the event of failure of the regular supply. During the past 5 hours, the diesels had been incapable of being rapidly started. If there had been an interruption in the power, someone would have had to go to the diesel generator area to start them. On the other hand, if the fuel racks were reset, the diesels would have restarted on every ES actuation. As previously explained, they cannot be run for long periods when unloaded, and someone would have had to go to the diesel generator area each time to reset them. Either way, someone would have had to pass through a high radiation area.

It was possible to reset the fuel racks at once, however, and then to leave the controls in position so that the diesels would not automatically start on ES actuation. In the event of a blackout, the diesels could have been immediately started from the control room, as soon as the operators realized that power was lost. Resetting the fuel racks was carried out at 5 hours 29 minutes.

March 28, 1979—9:43 a.m.

By 5 hours 43 minutes, the RCS was fully repressurized. The pressure was maintained between 2000 and 2200 psig by operation of the PORV block valve. When pressure got up to 2200 psig, the valve was opened and the pressure dropped. When the pressure got down to 2000 psig, the valve was closed and pressure increased. This control of the pressure was maintained for the next 1 ½ hours.

[It was supposed that the higher pressure might be able to collapse the bubble and allow natural circulation. In order to encourage natural circulation, operators raised the water level of steam generator A to 90% by using the condensate pump for feeding.]

March 28, 1979—10:17 a.m.

At 6 hours 17 minutes, control room personnel had to don respirators because of high radiation levels. These respirators made communications more difficult.

Auxiliary building fans were stopped at 6 hours because of the high radiation and so as not to spread radioactivity. The fans were restarted again at 6 hours 14 minutes.

A leak in steam generator B was suspected; this was the reason for isolating it previously. There was also some concern about the steam generator A. Steam from A was being released to the atmosphere, and any leak would have led to a release of radiation. An operator was dispatched to the roof with a meter that was held near the steam plume. This measurement confirmed that the steam being released was not contaminated.

An emergency feedwater pump (EF-P2A) was restarted at 7 hours 9 minutes to complete the filling of OTSG A. Filling was completed at 7 hours 30 minutes.

It became clear that even with a full steam generator and high pressure, natural circulation was not being established. The next plan was to depressurize sufficiently to inject water from the core flood tanks.

Each of the two core flood tanks holds 7900 gallons of borated water. The tanks are pressurized with nitrogen gas to 600 psig. During operation, the tanks are open to the reactor vessel, but backflow of water is prevented by check valves. If the RCS pressure drops below the pressure of the nitrogen gas, borated water will be injected directly into the reactor vessel.

When water is injected from the core flood tanks, expansion of the nitrogen gas causes its pressure to drop until it balances the RCS pressure. If the RCS pressure drops slightly below 600 psig, only a small amount of water will be injected. An amount of water approaching the full volume of the tanks will be injected into the reactor vessel only when the RCS pressure is much lower than 600 psig. The operators did not realize this and incorrectly believed that the small amount of water injected was indicating that the core was covered.

[Other reports have mentioned the existence of a loop seal between the core flood tanks and the reactor vessel. These reports give the unfortunate impression that the loop seal might somehow prevent water from flooding the core even if the RCS pressure is lower than the nitrogen gas pressure. Actually, this can only be true if the differential pressure is less than 5 to 10 psi. High pressure in the RCS in combination with the loop seal will always prevent large amounts of water from being injected.]

March 28, 1979—11:30 a.m.

At 7 hours 30 minutes, the PORV block valve and the pressurizer spray valve were opened, and the pressure began to drop. The operator defeated ES actuation at 7 hours 42 minutes, just before automatic actuation would have occurred.

At 8 hours 12 minutes, a core flood tank high level alarm was received, indicating a level of 13.32 feet. [This alarm indicates that the core flood tanks were taking water from the reactor coolant system, which means that the check valve must have been leaking slightly.] At 8 hours 40 minutes, the RCS pressure was down to the nominal pressure of the nitrogen gas (600 psig), and flow from the core flood tanks to the reactor vessel should have started. At 8 hours 55 minutes, the core flood tank level was down to 13.13 feet, indicating that a small amount of water went into the reactor vessel.

March 28, 1979—12:31 p.m.

[Evidently, operators intended to use the decay heat removal system if at all possible.] At 8 hours 31 minutes, operators started decay heat pumps DH-P1A and 1B in anticipation of getting pressure down to the level for which the decay heat removal system is designed (about 300 psig).

Up to this time, the atmospheric steam dump valve was open. Sometime between 8 hours 30 minutes and 9 hours 15 minutes, the atmospheric dump valve was closed on orders to the control room from Metropolitan Edison management, because of concern that this might be the source of small radioactivity levels being measured outside the plant.

The steaming rate was very low at this time, and closing the atmospheric dump valve did not make any noticeable change in steam pressure. On the basis of physical evidence alone, therefore, it is not possible to pin down the time of closure. A small increase in pressure and operating level that occurred at about 9 hours 45 minutes cannot be definitely attributed to closure of the atmospheric dump.

The condenser had already been shut down. The atmospheric dump was an alternate method of removing some heat from the steam generator. [The rate of heat removal was very low because there was virtually no circulation on the primary side. With closure of the dump valve, however, even this inadequate heat sink was lost. Energy removal via the open pressurizer relief valve and the letdown line kept the system from immediately heating up.]

The BWST level was still decreasing and there was increasing concern that the tank would run out. At 9 hours 8 minutes, suction from the BWST was stopped.

March 28, 1979—1:50 p.m.

[It became obvious that the RCS pressure could not be reduced to get the decay heat removal system in operation. Only a small amount of water was injected from the core flood tanks.] The PORV block valve was closed at 9 hours 15 minutes, and was thereafter reopened at intervals for short periods. At 9 hours 50 minutes, coincident with opening of the PORV, there was a very sudden spike of pressure and temperature in the reactor building. The building was isolated, and the ES actuated and building sprays came on. The setpoint for the building sprays to come on is 28 psig, so the pressure spike must have been at least that high. The strip chart shows a peak pressure of 28 psig.

It is now known that the pressure spike was due to hydrogen combustion in the reactor building. Evidence for this is the high pressure in the building (seen on three pressure-measuring instruments), the high temperature in the building (seen not only by the building temperature measuring device but also by the reactor coolant pump air intake alarm), and the depletion of the oxygen level in the building. [The lack of adequate equipment to control hydrogen concentration is discussed in Section II.C.1.c]

The building sprays quickly brought the pressure and temperatures down. At 6 minutes after actuation, the sprays were shut off from the control room because there appeared to be no need for them.

Initially, the spike was dismissed as some type of instrument malfunction. Shortly afterward, however, at least some supervisors concluded that for several independent instruments to have been affected in the same way, there must have been a pressure pulse. It was not until late Thursday night, however, that control room personnel became generally aware of the pressure spike's meaning. Its meaning became common knowledge among the management early Friday morning. [See Section II.C.2.a for a more detailed discussion of this issue.]

At about the same time, two 480-volt ac motor control centers (MCC-2-32A and 42A) tripped. The motor control centers (MCC) are in the auxiliary building; it is not certain that tripping was connected with the explosion. Two leakage closed cooling pumps (DC-P2A and DC-P2B) tripped at the same time; these pumps are the largest loads on the MCCs. The loss of these MCCs caused considerable inconvenience for later operation. Even though there was standby dc equipment available for some of the motors, the loss of the MCCs made direct control from the control room more difficult.

[Although it was impossible to get the pressure low enough for the decay heat system, it was supposed that there would still be some advantage in keeping the core flood tanks open to the reactor vessel.] Operators maintained the RCS pressure below 600 psi (down to a minimum of 410 psi) by periodically opening the PORV block valve.

The pressurizer at this time showed a full (greater than 400 inches) indication. [It is possible that the true level in the pressurizer could have been lower. An indication of 400 inches means that the temperature-compensated level inside the pressurizer-measuring leg equals the level in the reference leg. There is a possibility that the reference leg could have been less than full, although no information to substantiate or refute this hypothesis is available.]

March 28, 1979—2:28 p.m.

At about 9 hours 50 minutes, the loop A hot-leg temperatures came back on scale, went to a minimum of 460°F, and then climbed back up, reaching 590°F at about 11 hours 40 minutes. During this period, there were a number of short dips and rises superimposed on the general trend. [It is possible that the pressurizer at this time backed up into the hot leg.]

[An alternate hypothesis, which also ties in with other phenomena, is that steam reflux increased at this time. Steam flow across the top of the hot leg would have been blocked by hydrogen. Continued venting of the pressurizer may have removed enough hydrogen to allow steam to flow across the top of the hot leg and be condensed in the steam generator. This is shown by a simultaneous drop in steam generator level (water was boiled away), a jump in steam pressure, and a drop in RCS pressure.]

[The operators believed that they now had natural circulation established in the A loop. It was thought that the bubble in the A loop had disappeared. Actually, even well-developed refluxing would not give the heat sink needed to cool the system much further.] The RCS pressure hit a minimum of 420 psig and then began to increase again. [As the pressure dropped, boiling in the reactor vessel would have increased to the point at which the steam production exceeded condensation plus loss through the relief valve, resulting in another rise in pressure.]

At 10 hours 32 minutes, makeup pump MU-P1C was started. Makeup pump 1C was stopped again at 10 hours 36 minutes.

March 28, 1979—2:38 p.m.

At 10 hours 38 minutes, the hot-leg temperatures went off scale again. They came back on scale almost immediately, however, and thereafter continued to drop. Steam generator parameters indicate that there was a momentary drop in heat transfer, but that the steam generator quickly recovered and began to remove heat again. Note, however, that if the atmospheric dump valve is closed, as soon as some of the water in the steam generator secondary side has boiled and the rest of the water has heated up, the steam generator can no longer remove any more heat.

At 11 hours 6 minutes, the temperature of loop A suddenly increased. [This increase in temperature is an indication that the secondary side of the steam generator had become "heat soaked" and would no longer remove a significant amount of heat from the RCS.]

At 11 hours 10 minutes, personnel in the control room removed their respirators.

March 28, 1979—3:10 p.m.

At 11 hours 10 minutes, the pressurizer level indication dropped rapidly to 180 inches over an 18-minute period. The drop in pressurizer indication was more or less coincident with the increase in hot-leg temperature and in the A loop cold-leg temperatures. [There is a possibility (unsubstantiated) that pressurizer heaters had been turned on previously.] The pressurizer level stayed low for about 20 minutes, and then began to climb, eventually going off scale again. The operator had turned on makeup pump MU-P1C, and 20 000 gallons of water had been added from the BWST and makeup tank.

There was very little change in conditions until 13 hours after turbine trip. During the intervening time, the PORV block valve and makeup pump MU-P1C were operated several times in an effort to hold a constant pressure. The hot-leg temperature dropped again at about 12 hours 40 minutes, coincident with an increase in RCS pressure. [The pressure increase would cause some steam in the hot leg to condense; the condensation transfers heat to the secondary side and gives a modest increase in steam pressure.]

March 28, 1979—5:00 p.m.

At 13 hours after the turbine trip, the auxiliary boiler was brought back into operation. Steam for the turbine seals was now available and it was possible to hold a vacuum on the condenser. Two condenser vacuum pumps were started. [It was now expected that repressurization would collapse the bubble in the hot legs, and natural circulation could be achieved through OTSG A.] Repressurization began at about 13 hours 30 minutes. At this time, makeup was 425 gpm, using two makeup pumps. At 13 hours 45 minutes, following the resolution of a problem with the outlet valve, OTSG A began steaming to the condenser.

At 14 hours 39 minutes, valve MU-V16B began to close; at 14 hours 41 minutes, valve MU-V16C was throttled until the makeup flow was down to 105 gpm; and at 14 hours 43 minutes, makeup pump MU-P1C was stopped and valve MU-V16C was completely closed. The RCS pressure was then 2275 psig.

March 28, 1979—7:00 p.m.

At about 15 hours, many of the radiation monitors came back on scale. [It is not likely that the reduction in radiation levels was directly controlled by the repressurization. Closing the PORV block valve, however, did stop the radioactive coolant from getting out to the reactor building.]

[It was now believed that it might be possible to start a reactor coolant pump. There was some concern, however, as to whether a pump would operate. If there were voids in the system, sustained running would possibly damage the pump or blow out the seals. Therefore, the control room personnel decided to "bump" one of the pumps (run it for only a few seconds) and to observe current and flow while the pump was running.]

The loss of two MCCs meant that the ac oil lift pumps were out of service. It is not possible to start a reactor coolant pump unless the oil lift pump can be started. There is a standby dc oil lift pump, but it was necessary to send people to the auxiliary building to start it. This was done at 15 hours 15 minutes.

March 28, 1979—7:33 p.m.

At 15 hours 33 minutes, operators started reactor coolant pump RC-P1A by manually bypassing some of the inhibiting circuitry. The pump was run for 10 seconds, with normal amperage and flow. Dramatic results were seen immediately. RCS pressure and temperature instantly dropped, but began to rise again as soon as the pump was stopped. [Evidently, there was an immediate transfer of heat to the steam generator when the coolant circulated. There was also a rapid spike in the steam pressure and a drop in steam generator level.]

March 28, 1979—7:50 p.m.

After analysis of the results of the short term run of the reactor coolant pump, conditions looked so hopeful that operators decided to start the pump and to let it run if all continued to go well. At 15 hours 50 minutes, reactor coolant pump RC-P1A was restarted, and again all went well. Temperatures went down and stayed down, and a steady steaming rate was established.

[Reasonably stable conditions had now, for the first time, been established. New problems were to arise later, but they were less serious than those that had been handled up to this time.]

[Apparently, no one at this time realized that a bubble still existed in the RCS. What appears to have happened is that the starting of the reactor coolant pumps swept the remaining gas in the upper part of the system around with the water as discrete bubbles. The gas bubbles would tend to collect in the most quiescent part of the system—

the upper head of the reactor vessel. There is also a possibility of a dry "hot spot" within the core.]

[It is now believed that the gas was largely hydrogen. Hydrogen is slightly soluble in water, and its solubility is greater at high pressure. An attempt to depressurize the system would cause some of the dissolved hydrogen to effervesce out of the water, thereby increasing the amount of hydrogen in the bubble. (An analogy is a capped bottle of carbonated soft drink. When the cap is firmly seated, the pressure is high and the carbonated gas remains dissolved. If the cap is removed, however, the pressure quickly drops, and gas bubbles out of the liquid.) The effervescence of hydrogen out of the water would interfere with attempts to depressurize. As the pressure dropped, the bubble would grow in size and could interfere with circulation of the reactor coolant.]

[In addition to growing in size, the bubble and the dissolved gas would make it impossible to depressurize the RCS completely. The pressure is controlled by the size of the steam bubble in the upper part of the pressurizer. When this bubble contains only steam, spraying colder water into the top of the pressurizer shrinks the bubble and reduces the pressure. When the bubble contains a gas like hydrogen, however, spraying does not reduce the size of the bubble as much, so there is less control over the pressure.]

Another problem with reduced pressure occurred in the letdown system. As explained, gas comes out of solution when the pressure is reduced. The gas from the letdown water collected in the bleed tanks and makeup tank, increasing the pressure and making it necessary to vent the tanks often. The gas vented off, though, was not pure hydrogen—there were small amounts of radioactive materials as well. There was a limited space available for holding the gas released from the letdown flow. [See Figure II-8 for a schematic drawing of the gas venting system.]

[These two factors would make the reduction of pressure an extremely slow process that took several days to accomplish.]

[At 9:25 p.m. on March 28 (17 hours 25 minutes after turbine trip), it was apparent that the utility believed pressure could soon be reduced to a level at which the decay heat system could be used.]

Valve DH-V187 from the decay heat pump to the 1A cold leg was opened at that time. [The reason for opening this valve must have been the utility's intention to use the decay heat system shortly.]

Unfortunately, there was still no bubble in the pressurizer; the pressurizer was reading off scale.

The auxiliary building sump was full of contaminated water. The auxiliary building neutralizer tank (WDL-T8B) had been filled before the accident. At 9:29 p.m. the operators commenced pumping the contents of this tank to TMI-1, so that the auxiliary building sump contents could later be pumped into WDL-T8B. The transfer to TMI-1 was completed at 12:20 a.m. on March 29.

At 10:18 p.m. on March 28, it seemed that a bubble had been reestablished in the pressurizer. About 30 minutes later, however, the bubble was again lost and the pressurizer returned off scale.

At 10:34 p.m., letdown flow was lost. [It is believed likely that the letdown coolers became clogged with boric acid. Boric acid is more soluble in hot water than in cold water. The extensive boration during the accident might have caused a condition of saturation, so that when letdown water was cooled, boric acid precipitated out in the letdown coolers and filters.]

High pressure drop alarms, along with letdown flow alarms, began to come in shortly after midnight and continued through the early morning hours of March 29.

During these early morning hours, some radiation alarms also continued to be received. The auxiliary building and fuel handling ventilation was shut off between 12:55 a.m. and 2:10 a.m. Shutting down the ventilation caused radiation levels to increase in the control room; so from 2:11 a.m. to 3:15 a.m., control room personnel were required to wear respirators.

March 29, 1979—4:35 a.m.

At this time, the first of many ventings of the makeup tank MU-T1 was carried out. The waste gas decay tank vent header, to which the tank was being vented, was leaking into the auxiliary building.

At 4:43 a.m., the seal water temperature on reactor coolant pump RC-P2A alarmed high. The operator then got a printout of the seal water temperatures of all reactor coolant pumps. High temperatures were found on pumps RC-P1B, RC-P2A, and RC-P2B (all nonoperating).

Between 8:00 p.m. March 28, and 6:15 a.m. March 29, the pressure slowly decreased from 1300 to 945 psig. A pressurizer bubble had been definitely established at 4:00 a.m.; and by 6:15 a.m., the pressurizer level was down to 341 inches. During this period, the cold-leg temperature hovered between 230°F and 280°F.

At 6:30 a.m. March 29, the pressurizer was sprayed down. The results were an additional 40-

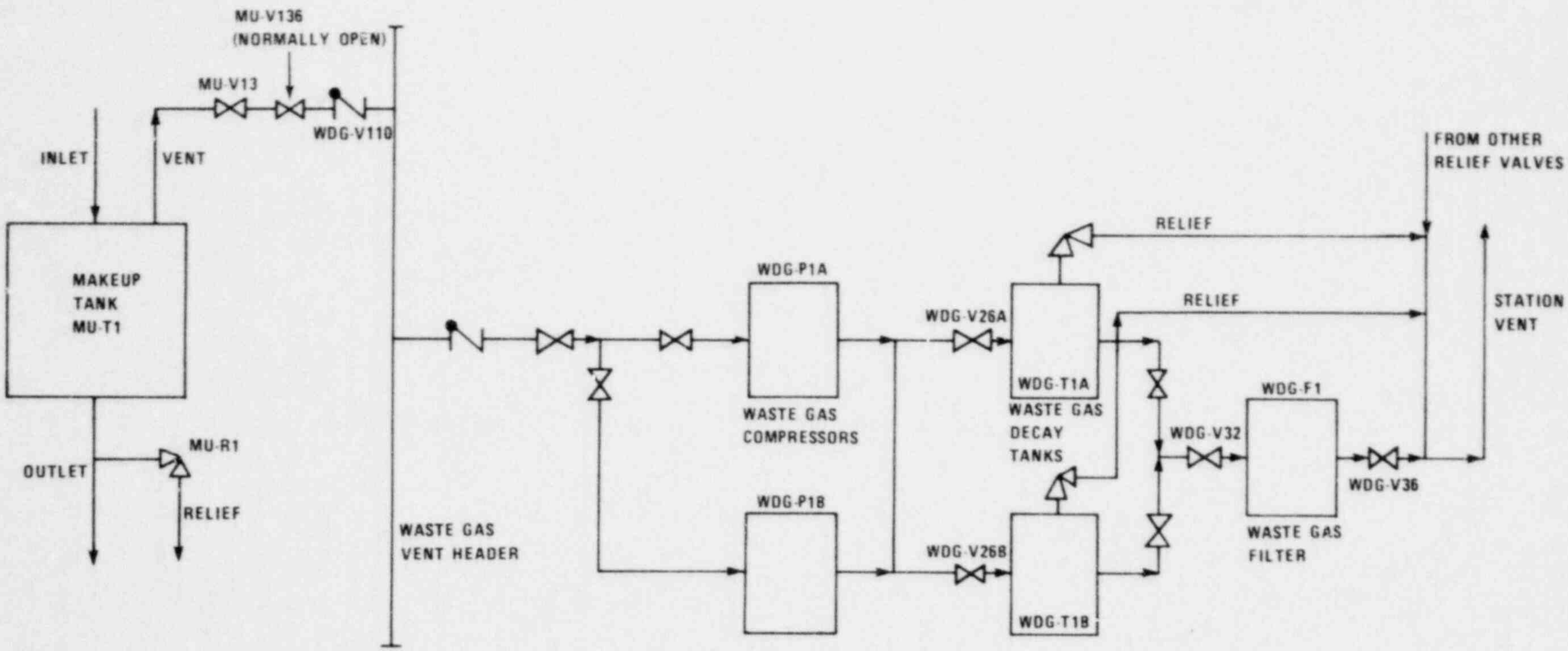


FIGURE II-8. Gas Venting System

Gases vented from the makeup tank are piped to the waste gas vent header in addition to gases from several other vents. Gas is pumped from the header to the waste gas decay tanks by the waste gas compressors. The gas is held in the decay tanks to allow decay of part of the radioactivity and is finally discharged to the atmosphere through the station vent (a full chimney) after being filtered in the waste gas filters.

psi drop in pressure and a 22-inch climb in the pressurizer level.

Letdown flow was reestablished at 6:31 a.m. Intermediate cooling temperature was increased, and apparently this increase raised the temperature at the coolers sufficiently to clear up the problem of boric acid fouling.

March 29, 1979—7:15 a.m.

At 7:14 a.m., the auxiliary building sump tank was pumped to the auxiliary building neutralizer tank WDL-T8B. The intention was to pump the auxiliary building sump to the auxiliary building sump tank.

At 7:16 a.m., the letdown flow was shifted to reactor coolant bleed tank (RCBT) B. It had been observed that when the makeup tank was vented, the radiation levels in the auxiliary building increased. [Apparently, this was because of the leak in the waste gas vent header.]

The contents of a second neutralizer tank (WDL-T8A) were pumped to TMI-1, beginning at 8:45 a.m. In addition to these contents, this tank contained preaccident water. It was destined to contain contaminated water from the auxiliary building sump.

March 29, 1979—12:40 p.m.

At 12:40 p.m., the sump pumps in the turbine building, control building, and control and service building were shut off. These pumps discharge to the industrial waste gas treatment system sump. The sump was completely filled and had overflowed to a settling pond. There was a leak from the pond (known as the "east dike drainage area") to the Susquehanna River.

March 29, 1979—1:15 p.m.

The industrial waste gas treatment system was started up at 1:15 p.m. in order to bring down the level of the overflowing sump and to eventually decrease the release of untreated water from the pond. The treated water from this system also discharges to the river. The treatment system was shut down again at 2:10 p.m. because of apparently high xenon levels in the discharge stream. It was later determined that the xenon reading was erroneous. Letdown was shifted from RCBT B to RCBT C at 2:58 p.m.

At 4:00 p.m., the auxiliary building sump tank was pumped to neutralizer tank WDL-T8A, and later the auxiliary building sump was pumped to the auxiliary building sump tank. After pumping out the sump, operators made an attempt at 7:00 p.m. to clean up the floor by washing it down underneath the plastic sheeting.

The industrial waste treatment system was restarted at 4:10 p.m. and was secured at 6:15 p.m., and had discharged a total of 25 000 gallons of treated waste.

March 29, 1979—8:20 p.m.

Degassing of the makeup tank MU-T1 continued to be a problem. One solution, which was tried at 8:20 p.m., was to degas the tank through the unit sample system, and back to the TMI-2 waste gas vent header. The attempt to do this was given up after 10 minutes of venting.

The next effort involved opening vent valve MU-V13 for 5 seconds to admit a small quantity of gas to the header. The purpose of admitting only a small amount of gas was to keep the header pressure down; it had already been noted that high pressure in the header made radiation levels rise in the auxiliary building. At the same time, the waste gas compressor was pumping out the vent.

Another attempted solution involved isolating all nitrogen venting to the vent header. The idea was to block all other discharges to the header to keep the header pressure down.

During the rest of the day, the makeup tank was cautiously vented again between 8:45 p.m. and 9:05 p.m., and again at 11:30 p.m., when the vent valve was bumped open at about 2-second intervals.

A significant increase in the fuel handling building exhaust gas monitor, from 300 mr/h to 1 r/h, was seen at 5:40 p.m. [It is assumed that this was connected with the venting, although it should be remembered that there were several other sources of contamination in the plant.]

One of the pressurizer level indicators failed at 9:14 p.m., but returned to service at 10:30 p.m. [This was not catastrophic, because there are three completely separate level sensors. Level indication is such a vital piece of information, though, that the loss of an indicator would be expected to cause concern.]

[The previous indications of a leak in steam generator B were now perceived to be false.] The steam pressure in OTSG B was holding steady at 25 psig, and the level was constant at 380 inches. Analysis of samples provided contradictory information concerning whether there had at some time been a leak.

March 29, 1979—12:00 p.m.

At the end of the day on March 29, the RCS pressure had risen slightly to 1105 psig, the temperature in the loop A cold leg was 325°F, and the pressurizer level was 325 inches. [It was now believed that there was a steam bubble in the reactor vessel. The presence of a bubble would also have caused difficulty in depressurizing. The presence of hydrogen could have been inferred at the time, however, from the difficulty caused by the outgassing of the makeup tank. This difficulty implied dissolved gas in the letdown stream.]

Difficulties with increasing gas pressure in the makeup tank took up much of the attention of the operators on March 30. It was noted early that the tank pressure was increasing even while the tank level was decreasing.

March 30, 1979—1:30 a.m.

At this time, the RCS temperature had dropped. The turbine bypass valves were closed slightly to raise the temperature by 8°F.

The makeup tank was vented to the waste gas decay tank (WGD-T18) from 1:50 a.m. to 2:15 a.m.

At 2:15 a.m. all sump pumps from the turbine building and control building area were shut off. One hour later, at 3:15 a.m., a temporary pump was used to pump the turbine building sump to the control building sump.

March 30, 1979—3:30 a.m.

Pressure in the makeup tank continued to increase. Because of the leak in the waste gas vent header, the venting of the tank was being controlled in an effort to keep the pressure in the header down. The tank was vented again at 3:30 a.m. At about the same time, more difficulty was being experienced in maintaining letdown flow. The valve between the letdown coolers and letdown block orifice (MU-V376) was being periodically cycled so that the pulsating flow might clear up the stoppage in the letdown system.

At 4:30 a.m., a filter system in the industrial waste treatment system was started. The waste treatment system was discharging to the river through the mechanical draft cooling tower blowdown line.

March 30, 1979—4:35 a.m.

The liquid pressure relief valve (MU-R1) on the makeup tank (MU-T1) opened at 4:35 a.m. because of the increasing gas pressure in the tank. The opening of this valve allowed the entire contents of the makeup tank to be discharged to the reactor coolant bleed holdup tanks. The level in the makeup tank dropped to zero, the outlet valve (MU-V12) from the makeup tank was shut, and the flow of water to the reactor coolant pump seals dropped to zero.

[It was realized that the high pressure in the makeup tank was leading to uncontrolled releases through the vent header. When the makeup tank dumped to the bleed holdup tanks, the relief valve on the latter would have lifted. Relief valves discharge directly to the stack, without treatment or holdup, so there was an uncontrolled puff release of radiation. Possibly, some water could also have entered the waste gas vent header.]

In order to obviate any later problems with the makeup tank, it was completely vented down via the waste gas vent header. [Some leakage in the vent header caused this venting to add to the contaminated gas being released.]

An attempt was made to pump from the reactor coolant bleed holdup tanks to the makeup tank; this attempt was unsuccessful because of the high pressure in the makeup tank (about 80 to 84 psig).

It was absolutely necessary to regain makeup flow in order to get seal water to the reactor coolant pumps. There was considerable concern about the low level in the BWST; therefore, the makeup pump suction had to be switched to the BWST.

The makeup tank was again vented to the waste gas vent header at 7:10 a.m. At 7:50 a.m., water was pumped from the reactor coolant bleed holdup tank to the makeup tank. Operators achieved some saving in makeup by stopping the flow of seal water to the nonoperating reactor coolant pumps RC-P2A, 1B, and 2B. At 8:15 a.m., they again aligned the makeup pump suction to the makeup tank.

March 30, 1979—9:40 a.m.

At 9:40 a.m., OTSG A was closed off for 7 minutes in order to heat the RCS to 280°F.

The pressurizer level was brought down at 10:45 a.m.; the intention was to eventually bring the level to 100 inches. At the same time, letdown was aligned to reactor coolant bleed holdup tank A. [Reducing the pressurizer level is usually a preliminary to depressurizing the RCS system.] The temperature of the A cold leg was 280°F, and the RCS pressure was 1043 psig.

March 30, 1979—12:20 p.m.

At 12:20 p.m., transfer of the contents of the miscellaneous waste holdup tank to TMI-1 was started. The transfer was completed at midnight.

An attempt was made to reduce radioactive gaseous discharges by venting the waste gas decay tank back to the reactor building. At 2:05 p.m., operators encountered difficulty in opening valve WGD-V30B to accomplish venting. They successfully opened the valve at 2:42 p.m.

March 30, 1979—4:00 p.m.

At 4:00 p.m., the pressurizer level was down to 215 inches. At the time the decision had been made to reduce the level, the pressurizer was at 390 inches.

[It was suspected at this time that a bubble still existed in the RCS. The bubble obviously could not have been steam, or it would long since have condensed given the low temperatures in the RCS. (A detailed discussion and evaluation of the formation and disappearance of the bubble will be found in Section II.C.2.)]

[If the mass or temperature of the reactor coolant are increased, the pressure will increase. If the RCS has very little steam or gas in it, there will be a rather large increase in pressure. If there is a large volume of steam or gas in the system, however, the pressure change corresponding to a change of mass and temperature will be cushioned. If a known change of liquid occurs, and the corresponding change of pressure is measured, it is possible to calculate the volume of gas in the system.]

[In order to calculate the gas volume precisely, it is necessary to know the change of liquid volume, the change of pressure, and the temperature fairly precisely. The pressure, volume, and temperature measuring devices of a nuclear powerplant are very rugged and reliable, but do not have laboratory precision; nor is such precision normally needed. Furthermore, the meters indicating the quantities are difficult to read exactly even if they were to indicate correctly. The difficulty in making precise measurements will make it difficult to calculate the gas volume with any great accuracy.]

[Another problem is that hydrogen is more soluble in water at high pressure. If the pressure in the RCS is increased, a hydrogen bubble would shrink; first, because it is being compressed, and second, because more hydrogen is dissolved in the water at the higher pressure.]

[To calculate the volume of the bubble at any time, letdown and makeup were aligned to the makeup tank. The level in the tank, along with the pressurizer level, was measured at the beginning and end of the experiment. Then the system pressure was changed by a known amount, and from this the volume of gas in the RCS could be calculated.]

[Some organizations computing the size of the bubble made corrections for the change in solubility of hydrogen and the different water densities in the makeup tank and the RCS, whereas others did not. However, even if a single method is used for calculations, two measurements made at slightly different times might give quite different results; first, because of the inherent imprecision of measurement, and second, because the bubble was actually shrinking.]

[Because the bubble was not pure hydrogen, but was really a mixture of steam and hydrogen, the results really ought to have been corrected by subtracting out the amount of steam. The results were the total amount of gas—whether hydrogen or water vapor. The real interest, however, was in the amount of hydrogen. Apparently, no one made this correction. Given the inexact nature of the measurements, it probably was not worthwhile. It should also be noted that the term bubble does not necessarily mean only a bubble in the top of the vessel. Any gas, anywhere in the RCS, would appear in the measurements.]

At 4:34 p.m., all the pressurizer heaters were turned off. This caused the pressurizer steam space to shrink. From measurements of both the shrinkage in volume and the decrease in pressure the size of the bubble could be calculated. The bubble was calculated to be about 366 cubic feet.

Calculations had been performed for previous times, whenever it appeared that a sufficient pressure change had taken place—the first calculation was made by Met Ed for 1:00 p.m. on March 29—but the experiment at 4:00 p.m. on March 30 seems to have been performed specifically to calculate the bubble size.

Problems with letdown flow were continuing. At 4:50 p.m., the letdown temperature alarmed high. Letdown flow was reestablished, and operators cleared the alarm in 5 minutes by opening the valve between the letdown coolers and the block orifice, MU-V376.

Only one reactor coolant pump was operating, RC-P1A. If this pump had failed, the plant would have been completely without an operable RCP. At 5:04 p.m., the oil pump for RC-P2A was started. Because there was a dc ground fault, however, the reactor coolant pump could not be started.

March 30, 1979—5:30 p.m.

There was considerable concern about the low level of the BWST, which was now down to 15.5 feet. At 5:30 p.m., a valve lineup was made so that clean, borated water would be pumped from the TMI-1 spent fuel tank to the TMI-2 surge tank and then to the TMI-2 BWST. Pumping of this water was started at 6:50 p.m.

Up to this time, degassing of the RCS had been accomplished principally by degassing the letdown water. This disadvantage of this method was that it was overpressurizing the makeup tank and contributing to radioactive releases via the leaky waste gas vent header. At 9:32 p.m. on March 30, the operators cautiously began "jogging" the pressurizer vent valve RC-V137. To keep the pressure up, they turned on the pressurizer heaters (three groups) at the same time. The effluent went to the reactor coolant drain tank and was condensed. The gas that came out was discharged through the RCDT rupture disk into the reactor building. The procedure was repeated at 10:17 p.m. and at 11:10 p.m.

By midnight on March 30, 1979, the total releases from the industrial waste treatment system had amounted to 72.56 millicuries. This was within the allowable limits. (Regulations allow the releases to be averaged over a year's time.)

March 31, 1979—2:05 a.m.

At 2:05 a.m. on March 31, a contact measurement on the reactor building equipment hatch gave a reading of 60 r/h. At the same time, contact readings on the waste gas decay tanks WDG-T1A and 1B gave 40 r/h. [These high readings do not mean that radioactive materials were being released at these locations. They do, however, indicate that intensely radioactive materials were contained in the reactor building and decay tanks.]

The pressurizer was vented to the reactor building from 1:45 a.m. and 3:15 a.m. Venting was then stopped while the hydrogen recombiner was placed in operation.

At 3:25 a.m. on March 31, the shift superintendent, shift foremen, and control room operators reviewed the emergency procedures for loss of the remaining reactor coolant pump. [This does not mean that the loss of the pump was expected; however, it was recognized that stoppage of the pumps could worsen the situation if not promptly countered.]

March 31, 1979—4:00 a.m.

At this time, exactly 72 hours after the accident began, loop A cold-leg temperature was 282°F, RCS pressure was 1060 psig, pressurizer level was 215 inches, and the level in the BWST was 18 feet. At this time, the calculated decay power was 7.4 MW, compared to 32.8 MW at 1 hour after turbine trip.

Pressures in the makeup tank MU-T1 were now decreasing. At 5:46 a.m., the pressure was down to 32 psig.

The reactor coolant temperatures had been gradually and slightly decreasing. At 5:48 a.m., the turbine bypass valves for OTSG A were closed slightly—from 47% open to 44% open—to arrest the cooldown.

March 31, 1979—7:53 a.m.

Venting of the pressurizer was started again at 7:53 a.m., even though the hydrogen recombiner was not yet operating. The RCS was vented at the following times during the day:

7:53–8:03 a.m.
8:28–8:46 a.m.
9:07–9:17 a.m.
9:35–9:57 a.m.
1:12–1:50 p.m.
2:25–3:00 p.m.
3:37–4:19 p.m.
4:56–5:37 p.m.
6:15–6:50 p.m.
7:50–8:34 p.m.
9:10–9:39 p.m.
10:21–11:52 p.m.

Release of hydrogen was accomplished with a minimum loss of coolant by cracking the vent valve open, while simultaneously using the pressurizer heaters and spray.

March 31, 1979—1:44 p.m.

Refilling of the BWST from TMI-1 was begun at 1:44 p.m. The method of filling was to use two sump pumps to pump from the TMI-1 spent fuel pool to the TMI-2 spent fuel surge tank, then to use the spent fuel cooling pump SF-P1A to pump the water intermittently to the TMI-2 BWST. This transfer was halted at 3:11 p.m. (at which time the BWST level was up to 26.5 feet) to allow the TMI-1 spent fuel pool to be refilled.

By 5:41 p.m., the pressure in the makeup tank MU-T1 had dropped to zero. The vent valve, MU-V13, was closed, and radiation levels in the vicinity of the vent header dropped. At 6:58 p.m., the valve was opened again to allow the makeup tank pressure to equalize, and the radiation readings increased. [The presence of radiation shows that the leak in the waste gas vent header was allowing gas to escape even at low pressure.]

The size of the bubble was calculated every few hours during the day by General Public Utilities (GPU) personnel, who used a simplified method of calculation that ignored many factors. To obtain an idea of the differences inherent in using different methods of calculation, one can compare the GPU calculation with the results of a method derived by Babcock & Wilcox (B&W) which included corrections for many of the items ignored by GPU. From data taken at 10:45 p.m., GPU calculated a bubble size of 894 cubic feet. The same data used in the B&W method gives a bubble size of 532 cubic feet when corrected to the same conditions.

[Even if one used the more exact method of calculating the bubble size, the inherent inaccuracy of reading and transcribing the required pressure and temperature measurements would make the estimation of bubble size subject to great uncertainty. Much of this uncertainty could have been eliminated if the measurements needed had always been put on the utility printer. Some of the readings can be reconstructed from the computer printouts, but some of the most important measurements are unaccountably missing.]

[In view of the inherent inaccuracy in the methods of bubble estimation, the differences in the methods used by different estimators, and the apparent casualness with which bubble data were acquired and recorded, it is understandable that there were large differences in public statements on bubble size.]

April 1, 1979—12:29 a.m.

The pressurizer was also vented on April 1, at about the same frequency as on March 31. Samples of the containment atmosphere were tested several times during the day. The hydrogen content remained about 2%, even with extensive venting.

At 12:29 a.m. on April 1, the turbine bypass valves, which had previously been closed slightly, were opened slightly. The purpose of opening these valves was to bring the RCS temperature down. Later, at 3:00 p.m. on April 1, the RCS pressure, which had exceeded 1000 psi, was reduced.

At 9:30 a.m., the contents of the miscellaneous liquid waste holdup tank WDL-T2 were transferred to TMI-1.

Pressures in the waste gas decay WDG-1A and 1B remained high. The pressure at 8:30 p.m. on April 1 was 86 psig.

April 1, 1979—10:00 a.m.

At this time, the auxiliary boiler was lost for 2 minutes. Although auxiliary steam is needed for condenser operation, the loss of the boiler for such a short time did not represent any threat to smooth cooldown.

A calculation of bubble size by GPU at 1:15 p.m. on April 2 showed shrinkage to 174 cubic feet. A calculation by the B&W method based on identical data showed no bubble in existence at this time. Calculations performed in the course of this investigation indicate that the bubble was gone the night of April 1 (see Section II.C.2.d.).

April 2, 1979—1:47 p.m.

The hydrogen recombiner was placed in service. The recombiner removes a steady flow of air from the reactor building, causes any hydrogen in it to recombine with oxygen, and returns the hydrogen-free air to the reactor building.

The hydrogen recombiner had first been started at 2:00 p.m. on March 30. Because there was a high radiation field in the neighborhood, however, operation of the recombiner was too hazardous for personnel. Heavy shielding was placed around the recombiner and its connections.

April 3, 1979—9:50 a.m.

The level in OTSG A was slowly raised to 97%. [Maintaining a high level on the secondary side of the steam generator would make it easier to ensure natural circulation if reactor coolant pump RC-P1A were lost, or if this were not possible, to start another pump.]

By midnight on April 3, the RCS temperature was 281°F, the pressure was 1050 psig, and all parties agreed that the bubble appeared to be gone. [It would still not be possible to depressurize completely. Hydrogen was still dissolved in the water, and reducing the pressure would have caused some to fizz out as gas, which would have reestablished the bubble.]

[Furthermore, there might have been some small discrete patches of hydrogen caught up in the internal structure of the reactor vessel. Reducing pressure could have caused these patches to expand and coalesce. The problems associated with hydrogen in the RCS, though, were now minor.]

[It should also be mentioned that it is now understood that there could not have been appreciable oxygen in the bubble; hence, an explosion would have been impossible. Even if there had been an explosion, though, it does not appear certain that the reactor vessel would necessarily have been damaged at all by it; and it appears highly unlikely that the vessel would have been damaged to the extent that there would have been a serious release of radioactive material.]

[The confusion stemmed from the known fact that water slowly decomposes into hydrogen and oxygen in the presence of radiation. What was apparently ignored by, or unknown to, some analysts is that when excess hydrogen is present, the reverse reaction (recombination) takes place at a much faster rate. Oxygen would thus be used up faster than it was formed, and no oxygen (other than minute traces) could ever appear in the bubble.]

[By midnight on April 3, the decay power was down to about 5 MW. This is a power density (spread over the entire reactor core in its original undamaged dimensions) of 2.9 watts per cubic inch. For comparison, a 60-watt light bulb produces about 6 watts per cubic inch.]

April 4, 1979 to April 7, 1979

Degassing continued throughout April 4 and 5. The pressurizer was periodically vented into the reactor building.

At 1:25 p.m. on April 6, reactor coolant pump RC-P1A tripped. Pump RC-P2A was successfully started about 2 minutes later.

At 8:00 p.m. on April 7, the RCS pressure was slowly lowered to 400 psig.

Stable conditions were established at 2:03 p.m. on April 27, 1979, when RC-P2A was stopped and natural circulation was established in both steam generators. (Steam generator B was later isolated, and adequate natural circulation was continued with steam generator A alone.) At that time, there were minor transients on some core thermocouples, which subsequently settled down.

The achievement of natural circulation ended the real emergency phase of the accident, but other problems have remained. The reactor building was heavily contaminated with about 5 million gallons of radioactive water, and the resulting waste disposal problem has not yet been solved. (There were even a few areas in the auxiliary building showing higher than normal radioactivity.) Perhaps most important, however, long term cooling of the badly damaged core will be necessary.

REFERENCES AND NOTES

[1] Postaccident tests have not established that water in the air lines would cause the valves to close. When the valves were inspected later, however, they were found to be closed.

[2] Block valves are provided in many systems to allow positive shutoff, especially where automatically operated or throttling valves are used. Throttling valves adjust the rate of flow of fluids and sometimes will not close absolutely tight. The block valve gives positive shutoff in case of leakage of the control or relief valve.

[3] The Integrated Control System (ICS), which controls reactor and turbine power, senses several system parameters and operates valves on the basis of these parameters.

[4] Many of the alarms indicate the approach of an unusual condition, rather than anything dangerous. These alarms are noted, but do not always call for immediate action.

[5] The failure of the condensate system did not impinge directly on the accident. Constant problems in the condensate system did distract operator attention, however, and may have led to additional confusion. See Section I.C.2. and I.C.10. for further discussion.

[6] Both pumps operating together will deliver between 850 and 1000 gpm, depending on system pressure.

[7] There was no direct indication that the RCDT rupture disk was broken. This could be inferred from the rapid drop in RCDT pressure and the sudden rise in building pressure.

[8] Reactor building sump pump operation is typed out on the alarm printer, but the printout of the alarm was delayed.

IMAGE EVALUATION
TEST TARGET (MT-3)

6″

MICROCOPY RESOLUTION TEST CHART

B RADIOLOGICAL RELEASES AND THEIR EFFECTS

1. INTRODUCTION AND BACKGROUND

The Three Mile Island Special Inquiry Group (SIG) investigated the radiological and health-related aspects of the March 28, 1979 accident at TMI-2. The principal objectives were to: (1) determine the immediate causes of and mechanisms for release of radioactive materials to the environment; (2) determine whether there were any direct sources of radiation outside the containment building during and subsequent to the accident; (3) determine the magnitude, sources, and duration of the releases of radioactive materials as well as any radiation leakage; (4) evaluate actions taken to mitigate releases and exposures; and (5) assess the radiological consequences of these releases and exposures to radiation on the health and safety of the exposed populations (both on site and off site).

To accomplish these objectives, the SIG evaluated the radiological and health-related conditions before, during, and after the accident. The inquiry examined the role of Met Ed, the role of the NRC in licensing and inspection, and the effort of the utility, industry, NRC, and other Federal and State agencies in response to the accident.

Our evaluation included reviews of Met Ed's Final Safety Analysis Report (FSAR)[1]; the NRC staff's Safety Evaluation Report (SER)[2]; its Final Environmental Statement (FES)[3]; pertinent design specifications and drawings of the Three Mile Island Station's gaseous and liquid radioactive waste treatment systems; Met Ed's radiation protection program and radiological instrumentation; the radiological monitoring data collected by the utility, NRC, and others who responded to the accident; and records and logs of the operation. These reviews were supplemented with site visits to obtain increased familiarity with the actual systems; interviews and depositions of site radiation protection personnel and consultants; and discussions with representatives of the various government agencies responding to the accident.

A chronology of significant radiological and radiation protection events is contained in Appendix II.6.

a. Principal Findings and Recommendations

We found numerous deficiencies in radiation protection practices and procedures, equipment, radwaste system, personnel training, and in the attitudes of both Met Ed and the NRC toward radiation protection and radiological health. These deficiencies are described in detail below. The principal findings and recommendations are as follows.

Findings

- There were numerous deficiencies related to radiation protection and radiological health; however, few, if any, of the deficiencies were causal factors in the TMI-2 accident.
- Even though the design bases of the radwaste systems were exceeded, the systems provided significant mitigation of the releases.
- The "defense-in-depth" concept, used in the regulatory process, was shown to be valid in mitigating the radiological consequences of the accident.
- The radiological consequences of the releases of radioactive material from TMI-2 into the environment are minimal at worst and may be nonexistent. Therefore, public concern regarding the effects of releases of radioactive materials from TMI-2 is not warranted.
- At Three Mile Island Station, a conflict existed between operations and radiation protection due to management's motivation toward production. As a result, radiation protection was perceived as a "necessary evil," and considered secondary to production.
- NRC failed to give sufficient attention to radiation protection and radiological health matters.
- The NRC review and inspection process in the area of radiation protection focused on conduct of normal power operation. Radiation protection in accident situations, such as existed at TMI, was not considered in the licensing review or inspection program.

Recommendations

- The role of radiation protection at commercial nuclear power reactors must be given greater emphasis by the Commission and licensee/applicants.
- The NRC must give additional emphasis to radiation protection and radiological health, and must change its organizational structure to improve management effectiveness to ensure that the agency's mandate "to protect the public health and safety" is fulfilled.
- Radiation protection programs at existing reactors should be reexamined to ascertain whether they are adequate to cope with normal and emergency conditions.
- The public must be fully informed of the manner by which nuclear powerplants are designed, licensed, and operated, and of the actual risks associated with radiation and radioactive materials.

b. Technical Background

The following sections summarize the technical aspects of production, retention, and release of radioactive materials in a nuclear powerplant. Those materials behave in accordance with their known physical and chemical characteristics. The radionuclides (radioactive atoms) released from the plant were primarily the noble gases and a small amount of radioiodine. The nonvolatile and water soluble materials were not released in any measurable quantities.

Radioactive Materials Produced by or in Nuclear Power Reactors—The primary source of energy in a nuclear reactor is the fission (breaking apart) of the nucleus of a uranium or plutonium atom. The production of radioactive materials is the natural consequence of the fission process. Additional energy (about 5%) is produced by the radiation emitted from these radioactive materials. This energy continues to be released after termination of the chain reaction (decay heat).

The radionuclides produced by the fission process are isotopes of elements found in nature. An isotope of an element has a different atomic mass but has the same chemical properties as another isotope of the same element. Therefore, the physical and chemical behavior of fission products and other radioactive materials produced in the reactor can be predicted. From this knowledge, the potential release, transport, and biological behavior of each fission product can be determined.

Fission Product Behavior—In a power reactor, there are several barriers to prevent the fission products from entering the working areas and the general environment. The ceramic fuel matrix in which the fission products are produced provides the first such barrier. Those elements that are volatile or gaseous at the operating temperature of the fuel are able to migrate through the ceramic fuel. However, the majority of the fission products produced is retained, either trapped or chemically bound. Examples of elements that are volatile, gaseous, or chemically unreactive with the fuel material are iodine, xenon, krypton, ruthenium, and cesium.

The second barrier to the release of the fission products is the fuel cladding. The ceramic fuel pellets are placed within thin walled tubes and sealed. Zircaloy was used for the fuel tubes in the reactors at Three Mile Island Station. There is a small gap between the fuel and the cladding in which the noble gases and other volatile nuclides collect and are contained.

The third barrier to the release of the fission product radionuclides is the reactor coolant. Many of the volatile fission products, the radioiodines and other radiohalogens, are soluble in the coolant in ionic (electrically charged) form. These materials can be removed by demineralizers such as those in the makeup and purification system of the reactor, or remain dissolved in the coolant. The majority of these radionuclides is contained within the primary coolant system. Other radionuclides such as the bariums, strontiums, and cesiums are also soluble in the coolant. However, the solubility of these radionuclides is dependent upon the pH of the coolant. As the pH of the primary coolant is increased (becomes more alkaline), their solubility decreases and they tend to precipitate or plate out. The noble gas radionuclides (kryptons and xenons) have very low solubility in the coolant, particularly at high temperatures and in the presence of other gases such as hydrogen, and evolve into a gas or vapor phase above the coolant or wherever the coolant is depressurized.

The fourth barrier to the release of fission products is the reactor pressure vessel and the piping of the primary coolant system, which are made of heavy walled steel. The fifth barrier is the containment building that houses the reactor. The containment building is designed to withstand overpressurization and external impacts and contain or delay fission product releases during an accident.

Release of Radionuclides into the Coolant during Normal Operations—If a defect in the fuel cladding develops, volatile fission products can be released into the coolant. NRC generally allows operation of a reactor with up to 1% of the fuel having a defect in its cladding.

In the absence of defective fuel elements, a small background concentration of fission products exists in the primary system. This background concentration results from the fissioning of trace quantities of uranium (termed tramp uranium) in or on the fuel cladding material.

In a reactor, many radioactive materials are produced in the primary system by capturing excess neutrons available from the fission process. In the fuel, several isotopes of plutonium are ultimately produced as the result of neutron capture by ^{238}U. Several "activation products" are produced as a result of neutron irradiation of water; e.g., ^{16}N, ^{13}N, and ^{18}F.

Minute amounts of material due to corrosion of the structure of the primary coolant system are carried by the water into the reactor core and activated—the resulting radioactive materials are called corrosion products; e.g., ^{60}Co, ^{58}Co, ^{59}Fe, and ^{54}Mn.

Radioactive Materials Released As a Result of the TMI-2 Accident—The radioactive materials released to the environment as a result of the TMI-2 accident were those that escaped from the damaged fuel and were transported in the coolant via the letdown line into the auxiliary building and then into the environment. The noble gases and radioiodines, because of their volatile nature and large concentration, were the primary radionuclides available for release from the auxiliary building.

Because the releases occurred primarily through a series of filters including charcoal filters designed to remove radioiodines, the released materials consisted primarily of the noble gas isotopes of krypton and xenon. Two krypton isotopes, 87 and 85, were not released in any significant quantities because of the short half-life of ^{87}Kr and the small amount of ^{85}Kr in the reactor core. It would be anticipated that ^{131}I, ^{133}I, and some ^{135}I would have been released from the plant due to their abundance and half-life. Several onsite and offsite measurements were made for both ^{131}I and ^{133}I, and these radionuclides were detected in some onsite samples on March 28.[4] Since the radioiodine releases were filtered and the primary radioiodine releases did not occur until several days later, the concentrations of the ^{133}I released to the environment were significantly reduced.

The principal release of radioactive noble gases occurred on the first day of the accident, March 28. The total quantity of released radioactive materials is estimated as 2.5 million Ci. (See Section II.B.2.f.) Table II-1 shows the calculated core inventory at the time of reactor shutdown, the estimated quantity released and the fractional contribution of each radionuclide to the total release.

After the first day, the ^{88}Kr and the ^{135}Xe concentrations were reduced by radioactive decay to less than detectable concentrations. All of the ^{133}I contained in the primary coolant released to the

TABLE II-1. Radionuclides released to the environment as a result of TMI-2 accident

Radionuclide	Half-life	Quantity in Core at Time of Shutdown (Curies)[5]	Estimated Quantity Released (Curies)	Estimated Fraction of Total Release
Kr-88	2.8 hours	6.92×10^7	3.75×10^5	0.15
Xe-133	5.2 days	1.42×10^8	1.58×10^6	0.63
Xe-133m	2.2 days	2.11×10^7	2.25×10^5	0.09
Xe-135	9.1 hours	3.31×10^7	3.0×10^5	0.12
Xe-135m	15.3 minutes	2.60×10^7	2.5×10^4	0.01
I-131	8.0 days	6.55×10^7	15	*

*On an estimated fractional basis of total nuclides released, iodine-131 was very small (about 15 curies as opposed to about 2.5 million curies of noble gases). See Section II.B.2.f.

auxiliary building eventually decayed to ^{133}Xe and ^{133m}Xe, which composed the major fraction of the radionuclides released from the plant.

2. RELEASE PATHWAYS AND MECHANISMS

The mechanism by which radioactive material left the TMI-2 core and the pathways for release to the environment are discussed in this section. This section also describes: (1) the radioactive waste treatment systems, designed to reduce the release of radioactive material to the environment during normal and accident situations; (2) additional mitigating actions taken by Met Ed subsequent to the accident; (3) calculations of quantities of radioactive materials released in gaseous and liquid effluents for various time periods after March 28; and (4) the postaccident radioactive waste at the Three Mile Island Station and the plans for its treatment.

a. Preaccident Background

In the FES[6,7] and the SER[8], the NRC staff concluded that the radioactive waste (radwaste) treatment systems at Three Mile Island Station were acceptable, based on conformance with Met Ed's designs, design criteria, and design bases to applicable NRC regulations and regulatory guides, as well as with staff technical positions and industry standards. The NRC staff also concluded that these systems satisfied the requirements of Appendix I to 10 C.F.R. 50, for maintaining releases "as low as reasonably achievable" (ALARA).

Gaseous Radwaste System — The gaseous radwaste system for TMI-2 processes gaseous wastes based on their origin and expected radioactivity levels. Figure II-9 shows the gaseous radwaste and ventilation systems. Filtration systems are included for the main condenser vacuum pump discharge, the turbine gland seal condenser discharge, the auxiliary building exhaust, the fuel handling building exhaust, and the reactor building purge unit. The auxiliary building and fuel handling building ventilation systems played a significant role in reducing the release of gaseous radioactive materials resulting from the March 28 accident. The reactor building purge system may play an important role in radioactive gaseous waste cleanup during recovery operations (Section II.B.2.h). The main condenser vacuum pump and turbine gland seal condenser discharges are normally released untreated but can be processed if the radioactivity in this effluent becomes high. These two systems did not contribute to or mitigate the March 28 releases from Three Mile Island Station.

The process gas system collects and stores radioactive gases stripped from the primary coolant in the letdown line, gases from the reactor building vent header, and vent gases from equipment. The low pressure vent header collects these gases and pipes them to one of two waste gas compressors, 40 standard cubic feet per minute (scfm), for compression and storage in the gas decay tanks. This storage allows radioactive decay prior to release to the environment. Releases are directed through a high efficiency particulate air (HEPA) filter to remove particulate material, and a carbon adsorber to remove gaseous radioiodine species.

The exhaust ventilation systems for the various buildings treat the exhaust air prior to release to the environment by particulate filters and carbon adsorbers.

The auxiliary building heating and ventilation system for TMI-2 is a once-through air flow system with no recirculation. Because the auxiliary building contains the makeup and purification system and the gaseous and liquid radwaste treatment systems, a small but measurable amount of radioactive material is expected to be present in the air in the

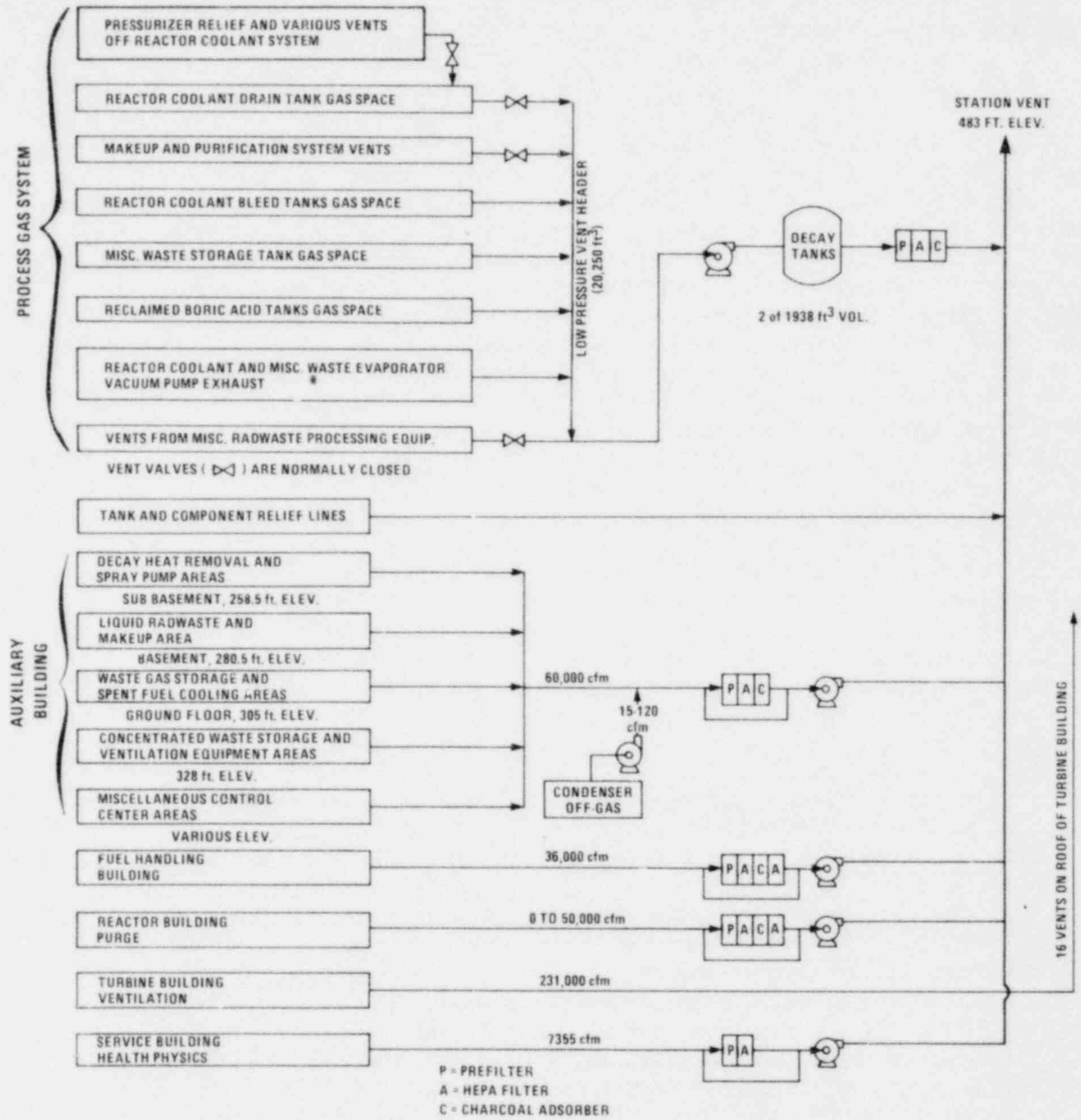


FIGURE II-9. Ventilation and Waste Gas System Release Pathways

The process gas system collects and stores the radioactive gases stripped from the primary coolant in the letdown line and also the gases from the reactor building vent header and vent gases from equipment. The low pressure vent header collects these gases and pipes them to one of two wa.te gas compressors (40 scfm) for compression and storage in the gas decay tanks. This storage allows radioactive decay before release to the environment. Releases are directed through a HEPA filter, to remove particulate material, and a carbon adsorber, to remove gaseous radioiodine species.

The exhaust ventilation systems for the various buildings treat the exhaust air before release to the environment by particulate filters and carbon adsorbers as indicated.

building because of normal component leakage. Accordingly, there is a cleanup system for the building exhaust that maintains the release of this radioactive material to the outdoor environment at a level that is as low as reasonably achievable (ALARA). There are two 30 000-cubic feet per minute (cfm) air filtration systems. Each consists of a prefilter, a high efficiency particulate air (HEPA) filter, a 2-inch-deep carbon adsorber, a second HEPA filter, and a fan. Each filter train is equipped with inlet and outlet dampers for isolation when changing filter components. All ventilation air from the auxiliary building is designed to be processed by these cleanup components at all times—there is no bypass line. The entire ventilation system is designed for continual use during normal operation of the reactor. It is not designed or intended for postaccident operation, and there are no technical specifications for balancing of ventilation flows or inplace testing of the exhaust air filtration components. The TMI-2 auxiliary building ventilation system is completely independent of the TMI-1 ventilation system.

The auxiliary building ventilation system underwent satisfactory functional and leak testing prior to startup,[9] although the bypass dampers were sealed. The sealing of the dampers routed all ventilation air through the cleanup components of the filter system which resulted in degradation of the filters over time due to the normal atmospheric contaminants. (See Section II.B.2.g for further discussion on the bypass dampers and the effects of sealing.) The auxiliary building filters are designed for normal ventilation purposes only and do not have any periodic inplace testing requirement.

The fuel handling building heating and ventilating system for TMI-2 is a once-through air flow system with no recirculation. There is a cleanup system on the exhaust for two reasons: (1) spent fuel is stored in the spent fuel pool, which releases small but measurable amounts of radioactive materials to the fuel handling building environment; and (2) it is possible that a fuel handling accident may release significant amounts of radioactive materials to the fuel handling building environment. It has two 18 000-cfm air filtration systems, each consisting of a prefilter, a HEPA filter, a 2-inch-deep carbon adsorber, a second HEPA filter, and a fan. Although a bypass line is installed around these components to prevent their degradation and to preserve them for postaccident situations, the filter systems had been manually valved into service prior to March 28. In fact, since the completion of acceptance testing in February 1978, all ventilation flow has been continuously routed through all the cleanup components.[10] Thus, we find that the carbon in the fuel handling building air exhaust filter system on March 28 was degraded.

The fuel handling building ventilation exhaust system is an engineered safety feature system designed to operate in a postaccident environment. TMI-2 technical specifications issued in February 1978 require periodic inplace testing of the exhaust units to verify that the systems are ready to perform after an accident. However, exemptions to pertinent sections of these technical specifications were granted until the first refueling outage for TMI-2, which has not occurred. The impact of these exemptions on releases of radioactive material subsequent to the March 28 accident is discussed below.

A basis for the exemptions was the NRC staff assumption that the ventilation systems were independent. However, the ventilation systems for TMI-1 and 2 are in direct communication. Accordingly, any gaseous radioactive material present in either spent fuel area will be exhausted via both fuel handling building ventilation units. In fact, all the design aspects of the radiation protection review of TMI-2 were characterized by the NRC staff as being independent of TMI-1.[11] We find that the review of the TMI-2 ventilation system did not consider interties with TMI-1.

All of the filtration systems and their initial charge of activated carbon were supplied by Mine Safety Appliances Company (MSA) of Pittsburgh, Pa. The carbon did not meet the specifications of, and its testing did not meet the recommendations of, NRC Regulatory Guide 1.52 (Revision 1, July 1976). The carbon did meet GPU specifications that required a removal efficiency of 99.95% for elemental iodine and 85% for methyl iodide when the new (unused) carbon was tested at a relative humidity of 90% and had a residence time of 0.25 seconds. The technical specifications for the fuel handling building exhaust filtration units require, in accordance with Regulatory Guide 1.52 (Revision 1, July 1976), a removal efficiency of 99.9% for elemental iodine and 99% for methyl iodide. As supplied, the MSA carbon (a coconut shell based carbon impregnated with stable iodine, as KI_3, to increase the efficiency for organic iodide removal) did not satisfy the applicable technical specification requirement of 99% for methyl iodide removal, but did satisfy the licensee specification of 85% (the actual test result was 96.97%). The NRC's Office of Nuclear Reactor Regulation agreed in February 1978 to allow the installation of this carbon until the first refueling outage for TMI-2.

The TMI-2 operating license allowed another technical specification exemption pertaining to the testing frequency for the installed carbon. Engineered safety feature air filtration systems are required by standard technical specifications[12] to have a representative sample of carbon removed every 720 hours of filter system operation and tested in a laboratory to verify that the radioiodine removal capability has not been seriously degraded. The operating license also exempted this section. The carbon did not meet the applicable requirements at installation in February 1978, and was not periodically tested to verify its condition. Thus, we find that the degraded carbon contributed to greater radioiodine releases than would have occurred had the carbon filters met all NRC requirements.

The fuel handling building exhaust system is the only engineered safety feature air filtration system at TMI-2 designed to prevent releases of radioiodine to the environment after an accident and, therefore, was the only air filtration exhaust system covered by the TMI-2 technical specifications. Met Ed did, however, install the same grade of carbon qualified to the same specification in all of the TMI-2 air filtration systems.

The filters and cleanup components for the fuel handling and auxiliary buildings were installed and tested in place in February 1978 and were not tested or inspected thereafter. Final painting and cleanup of these buildings between February and December 1978 generated significant amounts of fumes and aerosols that degraded the cleanup components. The components would most likely have been replaced had inplace testing occurred and shown degradation of the filters. We find that the design and testing of these filter systems did not permit the condition of the filters and leakage around the filters to be identified at any time from initial functional testing.

The lack of periodic inplace testing was due to (1) the technical specification exemptions on the fuel handling building filtration system, and (2) the lack of requirements for periodic inplace testing of the auxiliary building filtration system, because the filtration system is not considered to be an engineered safety feature system in the NRC licensing review process. Based upon postaccident determinations of filter carbon efficiencies, we find that radioiodine releases were higher than those releases might have been with NRC requirements for periodic inplace testing and carbon in the filter system. We find, also, that these radioiodine releases were higher by approximately a factor of 5, which is estimated from an analysis of expected removal efficiencies with inplace testing (95%) versus measured efficiencies (approximately 75% as shown in Table II-4).

The reactor building air purge has a capacity of 50 000 cubic feet per minute. Cleanup components in the system are a prefilter, a HEPA filter, a 2-inch-deep carbon adsorber, and a second HEPA filter. This filter system was not used in response to the March 28 accident, but may be of importance during recovery operations when, and if, the containment structure is purged (see Section II.B.2.h).

Liquid Radwaste System — The liquid radwaste treatment system for TMI-2 consists of equipment and instrumentation necessary to collect, process, monitor, and recycle or dispose of radioactive liquid wastes. The system is composed of three basic subsystems: the makeup and purification system, the miscellaneous waste system, and the industrial waste treatment system. Prior to treatment in the subsystems, wastes are segregated based on their origin, activity, and chemical composition. Treatment is on a batch basis, after which samples are analyzed to determine whether the waste is to be retained for further processing or discharged under controlled conditions to the Susquehanna River via the blowdown system of the mechanical draft cooling tower. There were no releases of liquid radwaste by this normal discharge path during or subsequent to the March 28 accident.

The makeup and purification system, as shown in Figure II-10, is used to maintain the quality and boron concentration of the primary coolant. A stream of the primary coolant, termed the letdown, is taken continuously from the reactor, treated, fed to the makeup tank, and ultimately returned to the reactor. The letdown can be held up in any of three reactor coolant bleed holdup tanks.

The liquid radwaste treatment system treats the liquid radwaste prior to discharge to the environment. The letdown stream is a designed pathway for primary coolant to enter the cleanup components in the auxiliary building.

The makeup tank, located downstream of the cleanup components, is designed to temporarily retain the treated letdown. The makeup tank contains a manually operated vent (MU-V-13) to allow any hydrogen overpressure to be vented. The standard operating procedures specify an operating pressure of between 10 and 20 pounds per square inch gauge (psig). The operator vents hydrogen if the pressure is high, or adds nitrogen if the pressure is low. Since radioactive gases may be present in the vent stream, the vent is connected to the vent header and the waste gas decay tanks. The makeup and purification system for TMI-2 is separated from the TMI-1 system. Liquid radwaste generated by operation of the makeup and purification system include the letdown (when the boron concentration is being lowered) and demineralizer regeneration wastes.

The miscellaneous waste treatment system treats the liquid radwaste collected in the containment and

TABLE II-2. Liquid releases from TMI-2

Release	March 28–31 1979	1st Quarter 1979 Jan. 1–Mar. 31	2nd Quarter 1979 April 1–June 30	Calendar Year 1978	Tech. Spec. Limit (Ci/yr)
I-131 (Curies)	0.11	0.11	0.13	0.0014	10**
Activation & Corrosion Products (Curies)	0.11*	0.20	0.13	0.39	10
Tritium (Curies)	0.55*	78	ND	38	NA

*Reported number is for TMI-1 and 2 combined.
**The total fission, activation, and corrosion products allowed to be released in 1 year is 10 curies.
NA = Not Applicable
ND = Not Determined or Measured

auxiliary building drains, laboratory and sampling drains, demineralizer resin and filter sluice water, deborating bed regenerants, and decontamination and other miscellaneous wastes. These streams are collected in holdup tanks, pumped through a filter to an evaporator, to a polishing demineralizer, and then stored in test tanks for recycling or discharge. The miscellaneous waste evaporator is shared by both TMI-1 and 2, and is physically located in TMI-1.

The industrial waste treatment system (IWTS) is not expected under normal conditions to contain liquids with any appreciable activity. Accordingly, it is not evaluated by the NRC staff in its review of the liquid radwaste treatment systems.[13] Figure II-11 shows that the sumps from the control and service building, the diesel generator building, the tendon gallery, and the turbine building are pumped to the industrial Waste Treatment Plant. Minimal treatment is provided by a filtration system before the wastes are discharged. The effluent flows through a radiation monitor; however, there is no automatic shutoff capability in the event of detection of levels exceeding technical specifications. A manually operated valve is installed to prevent any discharge of liquid.

The industrial waste treatment system is not expected to contain radioactive material, and is not reviewed as part of the liquid radwaste system.

b. Radwaste System Status at the Time of the Accident

In its review of the radwaste systems, the NRC staff calculated source terms for gaseous and liquid effluents and used these source terms to calculate the individual and population radiation doses expected to result from normal operations, including anticipated operational occurrences.[6,7] Expected releases of radioactive material during normal operation were calculated on a design basis of fission product leakage from 1% of the fuel.[14] Noble gas releases from normal operations were estimated to be 6700 Ci/yr, primarily ^{133}Xe from reactor building purges, and 0.01 Ci/yr of ^{131}I. The site (i.e., TMI-1 and 2 combined) is allowed by technical specification[15] to release as many as 220 000 Ci/yr of noble gases (when calculated on a ^{133}Xe dose equivalence basis) and 0.05 Ci/calendar quarter of ^{131}I.

Projected release rates of radioactive material in liquid effluents were approximately 0.24 Ci/yr, excluding tritium and dissolved gases. TMI-1 and 2 combined are allowed by technical specification[15] to release as many as 10 Ci/yr, excluding tritium and dissolved gases. The tritium release was estimated to be 550 Ci/yr.

On March 28, 1979, prior to 4:00 a.m., the TMI-2 liquid radwaste treatment system was operating normally. TMI-1 was returning to operations after a refueling outage, which generated liquid radwaste that required processing in order to continue startup. A spill of 20 000 gallons of contaminated water from the fuel transfer canal into the reactor building of TMI-1 near the end of the outage resulted in large volumes of low level liquid radwaste from decontamination operations. Because there is no *de minimis* level[16] below which low level liquid radwaste can be released untreated, this volume was being stored, which reduced the available liquid radwaste storage capacity at Three Mile Island Station on March 28.

Immediately prior to the accident, approximately 60% of the station's available liquid radwaste storage capacity (300 000 gallons per unit) was filled. Of particular importance, the auxiliary building sump was approximately 63% full, the auxiliary building sump tank (WDL-T-5) was approximately 76% full, the two contaminated-drains tanks (WDL-T-11A and 11B) were 77% and 24% full, respectively, and the three reactor coolant bleed holdup tanks,

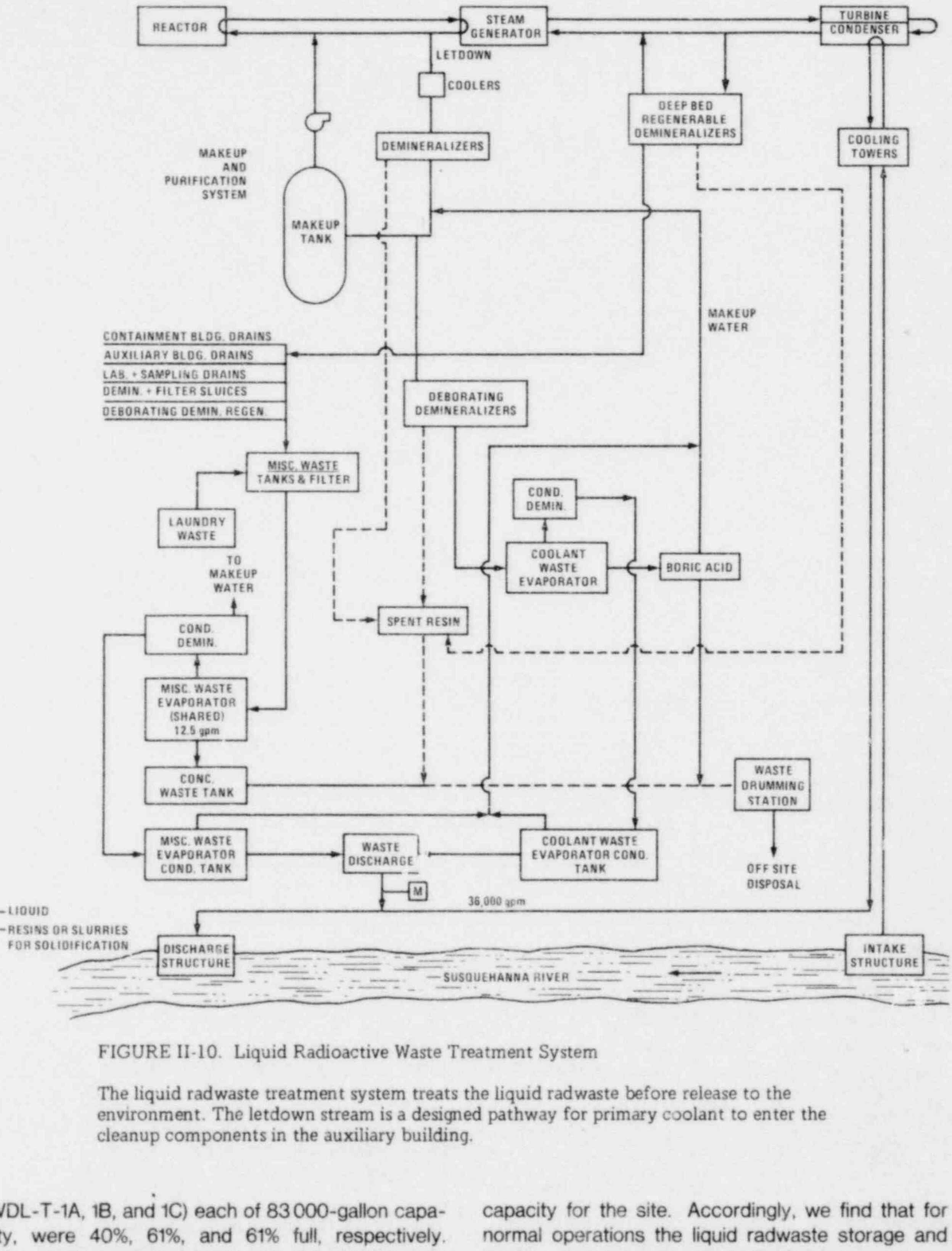


FIGURE II-10. Liquid Radioactive Waste Treatment System

The liquid radwaste treatment system treats the liquid radwaste before release to the environment. The letdown stream is a designed pathway for primary coolant to enter the cleanup components in the auxiliary building.

(WDL-T-1A, 1B, and 1C) each of 83 000-gallon capacity, were 40%, 61%, and 61% full, respectively. Although there was minimal input of liquid radwaste from TMI-2, 60% of the Three Mile Island Station's liquid radwaste tank capacity was not available on March 28. We attribute this to the lack of a *de minimis* release level and insufficient processing capacity for the site. Accordingly, we find that for normal operations the liquid radwaste storage and treatment system was marginal at best.

Prior to March 28, 1979, the gaseous radwaste system and the heating and ventilating systems had satisfactorily undergone numerous functional and acceptance tests.[17] However, a number of mainte-

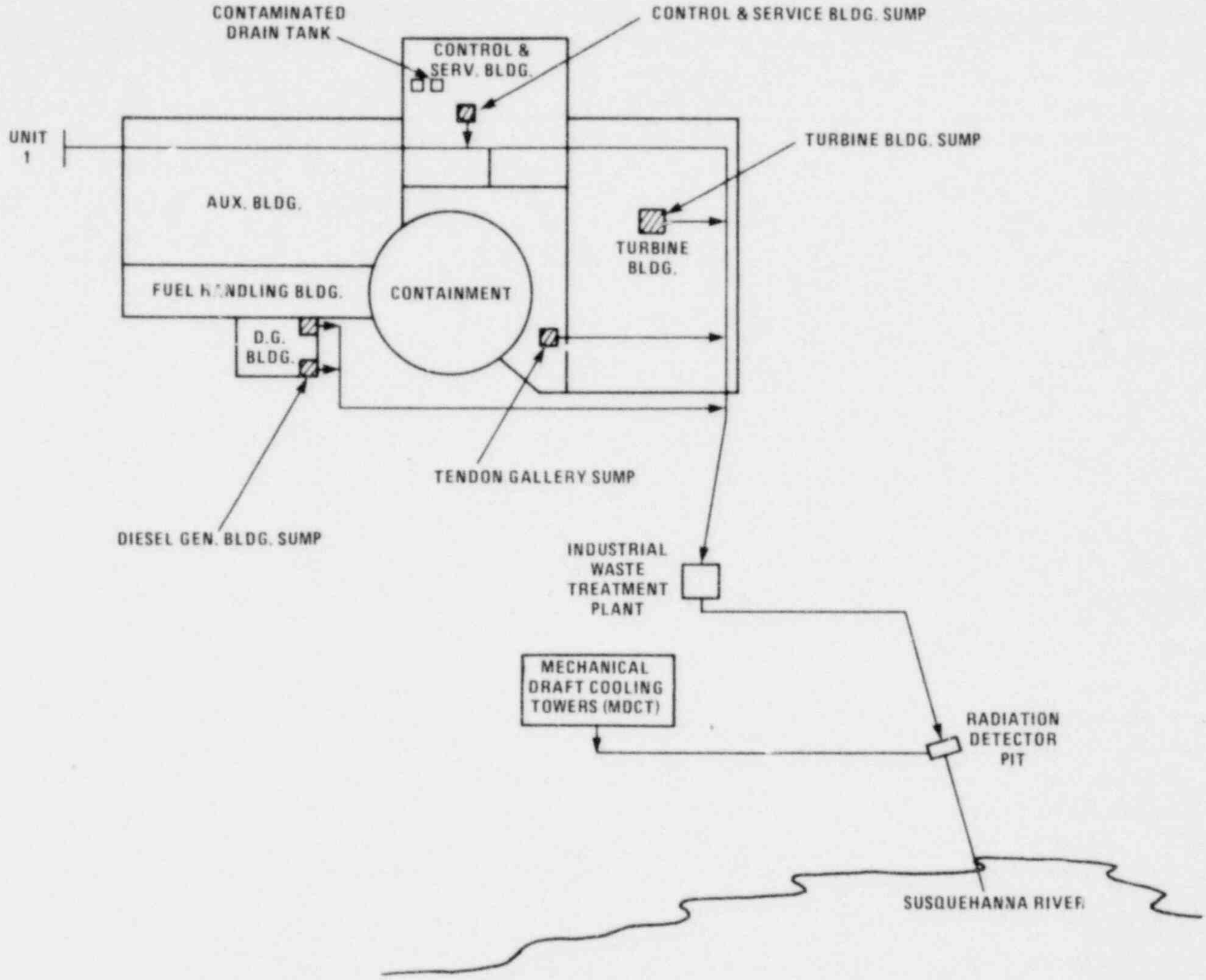


FIGURE II-11. Unit 2 Industrial Waste Treatment System

The industrial waste treatment system is not expected to contain radioactive material and is not reviewed as part of the liquid radwaste system.

nance work requests for the waste gas system were outstanding at the time of the accident. Both waste gas compressors (WDG-P-1A and 1B) needed service for various conditions (described in maintenance requests as "over pressurized," "makes loud noise," "no seal water level," "level control pump operation").[18] These compressors leaked during the March 28 incident. In addition, makeup tank vent valve MU-V-13 was suspected to be leaking.[19]

Operation of compressor A resulted in releases of gaseous radioactive materials to the auxiliary and fuel handling buildings with each venting of the makeup tank to the waste gas decay tanks. The radioactive noble gases in this leakage were not held up in the decay tanks and were released untreated to the environment. Compressor B, which was to be operated only in an emergency because it was considered to be in poor condition,[20] was not used until Thursday, March 29 and therefore, leaks in this compressor were not significant. We find that the leaks, particularly in compressor A, which led to the release of small amounts of radioactive material during normal operation, led to releases of radioactive material after core damage.

c. Liquid Release Pathways

The only release of radioactive materials in liquid effluents was via the industrial waste treatment system (IWTS) shown in Figure II-11. These releases were discharged to the Susquehanna River. Since radioactive material is not expected in its input streams, the IWTS is not designed to collect or process radioactive material. The IWTS is designed to collect sump liquid from the various buildings, provide minimal filtration, and discharge the sump water in the cooling tower blowdown.

Several times on March 28, Met Ed sampled the primary coolant and secondary system water from both steam generators to determine plant conditions. Because the sampling room is shared by both units and is located in TMI-1, the TMI-2 sample lines are several hundred feet long. This necessitates flushing and recirculation of each line for 45 minutes prior to sampling to obtain a representative sample. These actions resulted in significant amounts of highly radioactive liquid entering the contaminated-drains tanks in the control and service building (total capacity approximately 5000 gallons). The two tanks (already 77% and 24% full) received greater amounts of liquid than normally expected, and overflowed to the control and service building sumps, which were pumped to the IWTS for discharge because minimal liquid radwaste tank capacity was available. We find that the radwaste liquid storage capacity at the Three Mile Island Station was inadequate to cope with the emergency operations.

A second mechanism for release of liquid containing radioactive material was through the turbine building sump. Leaks between the primary coolant and secondary coolant, caused by steam generator B tube failures, contaminated the secondary side of TMI-2. Contaminated steam leaked from the turbine to the turbine building sump, and was then pumped to the IWTS.

From March 28 at 4:00 a.m. to March 30 at 12:00 midnight, approximately 265 000 gallons were released via the IWTS. Much of this volume consisted of preaccident water and Unit 1 water. It contained approximately 0.073 Ci of ^{131}I which was the only measured radionuclide. From the period March 28 through April 30, 0.23 Ci of ^{131}I, 0.24 Ci of all activation and corrosion products, and negligible amounts of tritium were released.[21] These releases, although above normal, did not approach any technical specification action limits (see Table II-2).

Discussions were held among Met Ed, the NRC, HEW, and various State agencies regarding termination of releases of liquid via the IWTS as early as Thursday, March 29. The purpose of the discussions was to verify that releases were within technical specification limits, and no liquid discharges were permitted for approximately 24 hours beginning at approximately 6:00 p.m. on March 29, to allow time to establish acceptable surveillance and monitoring activities. With an increase in sampling and analysis, discharges from the IWTS have continued, the overwhelming majority being blowdown from the mechanical draft cooling towers. Low activity water from TMI-1 that has been processed through a demineralizer cleanup system has also been discharged. However, none of the liquid radwaste in the TMI-2 auxiliary, fuel handling, and containment buildings has been released (see the discussion on Recovery Operations in Section II.B.2.h).

We find that the quantity of radioactive material in liquid effluents thus far released as a result of the March 28 accident at TMI-2 was not significant.

d. Transport of Radioactive Materials out of Containment

Following the turbine trip, the open pilot-operated relief valve (PORV) on the pressurizer permitted reactor coolant, at high temperature and pressure, to fill the reactor coolant drain tank. Fifteen minutes after the turbine trip, the reactor coolant drain tank rupture disc, which had a setpoint of 192 psig, failed and primary coolant flowed to the reactor building sumps. As a result, the reactor building sump pumps started automatically and transferred at most 8100 gallons to the auxiliary building sump tank. These pumps were manually turned off at 4:38 a.m.[22] Since the available capacity of the auxiliary building sump tank was only 700 gallons, liquid overflowed to the auxiliary building sump, which caused water to back up through the floor drains in both the auxiliary and fuel handling buildings.

This liquid did not contain large amounts of radioactive material because significant core damage did not occur until after 6:00 a.m. However, the liquid proved to be a means for highly contaminated reactor coolant to travel into areas of the auxiliary and fuel handling buildings as the accident progressed. A second, larger source of water that was not contaminated, but compounded the spread of radioactive material in the two buildings, was leakage from the four river water pumps (RR-P-1A, 1B, 1C and 1D) located on the 280-foot elevation of the auxiliary building. These pumps, which provide cooling water to plant components, leaked gallons per minute.

After core damage occurred, radioactive material was transported out of the reactor by the letdown line of the makeup and purification system. Because the letdown is a stream of primary coolant directly from the reactor, it contained significant amounts of radioactivity.

It was necessary to maintain some letdown flow to the makeup and purification system to ensure safe cooldown of the reactor between March 28 and April 2, 1979. As a result, leaks in the makeup and purification system (located in the auxiliary building), which release small amounts of radioactive material in normal operation, released large amounts of radioactive material during the accident, even though the letdown flow was reduced from its normal volumetric flow of 45 gallons per minute to about 20 gallons per minute. The letdown flow was, in fact, the major path for transferring radioactive material out of the reactor.

We find that leakage of radwaste system components, particularly in the makeup and purification system, which contained small amounts of radioactive material during normal operation, led to the most significant releases of radioactive material after core damage occurred. This source of liquid radioactivity was released to the auxiliary building and uncontaminated water spread over the floors of the auxiliary and fuel handling buildings.

e. Gaseous Release Pathways

The TMI-2 stack was the main release point for gaseous effluents. Numerous pathways to the stack existed for the release of radioactive gaseous effluents. The release pathways from the reactor to the auxiliary and fuel handling buildings are shown in Figure II-12. Figure II-13 shows the general arrangement of buildings at the site, and the TMI-2 stack.

The release of radioactive gases into the auxiliary and fuel handling building occurred by direct gas leakage and leakage of radioactive liquid from which radioactive gases evolved. Direct leaks of radioactive gas were the major source of radioactive gaseous releases.

Leaks in the vent header system and the waste gas decay system were the primary mechanisms for the direct release of gaseous radioactive material. The high pressure in the reactor coolant drain tank (up to 192 psig) prior to rupture disc failure led to a sequence of events that created a significant release pathway for gaseous radioactivity through the vent header.

The reactor coolant drain tank was connected to the vent header via two paths. Pressures in the reactor coolant drain tank prior to rupture disc failure pressurized the vent header. Before the rupture of the reactor coolant drain tank relief at 4:15 a.m., the radiation monitoring system detected activity that indicated that the waste gas vent header was leaking. Subsequent inspection has identified six leaks in the vent header system.[23]

The high pressure in the reactor coolant drain tank (up to 192 psig) prior to rupture disc failure led to a sequence of events that created a significant release pathway for gaseous radioactivity.

The vent line from the reactor coolant drain tank to the vent header was open on March 28, as indicated by the open status of valves WDL-V-126 and V-127.[24] The high pressures in the reactor coolant drain tank forced liquid (primary coolant) through the vent line to the vent header. The vent header relief valve (WDG-R-3) is set at 150 psig, so water under pressure caused leaks in the water drains. This water also damaged some of the 10 check valves located between the vent header and connected tanks (such as WDG-V-113 to the reactor coolant bleed holdup tanks, or WDG-V-153 to the reactor coolant evaporator). These check valves are designed to permit flow only from the component to the vent header and not in the opposite direction, but are known to operate inefficiently and fail easily.[25] Therefore, a significant pathway existed from the vent header to a number of tanks. The relief valves on these tanks, which were set at relatively low pressures (reactor coolant bleed holdup tank at 20 psig, reactor coolant evaporator at 10 psig), opened. Lifting of these relief valves resulted in untreated releases directly to the stack via the relief valve vent header (shown in Figure II-12). We find that the gaseous radwaste system design included "relief to atmosphere," which provided a path to the environment for untreated gas. We find, also, that the high reactor coolant drain tank pressures between 4:00 and 4:30 a.m. on March 28 damaged portions of the vent gas system and resulted in a gaseous release pathway to the vent header, through failed check valves to components with low-pressure relief valves. Once established, this release path was available whenever the vent header was used, such as in the venting of the makeup tank.

Problems with the waste gas system compressor have already been discussed. A postaccident examination of compressor B found a hole approximately the size of a quarter. The operation of the compressor at any pressure would be considered a significant release path.[23] However, compressor B was off line from March 28 until March 29.[26] In addition, the design of the waste gas system includes a pressure regulator (WDG-V-59) that limits the inlet pressure to the compressors to approximately 1 inch of water gauge. This prevented any high pressures in the vent header from reaching the

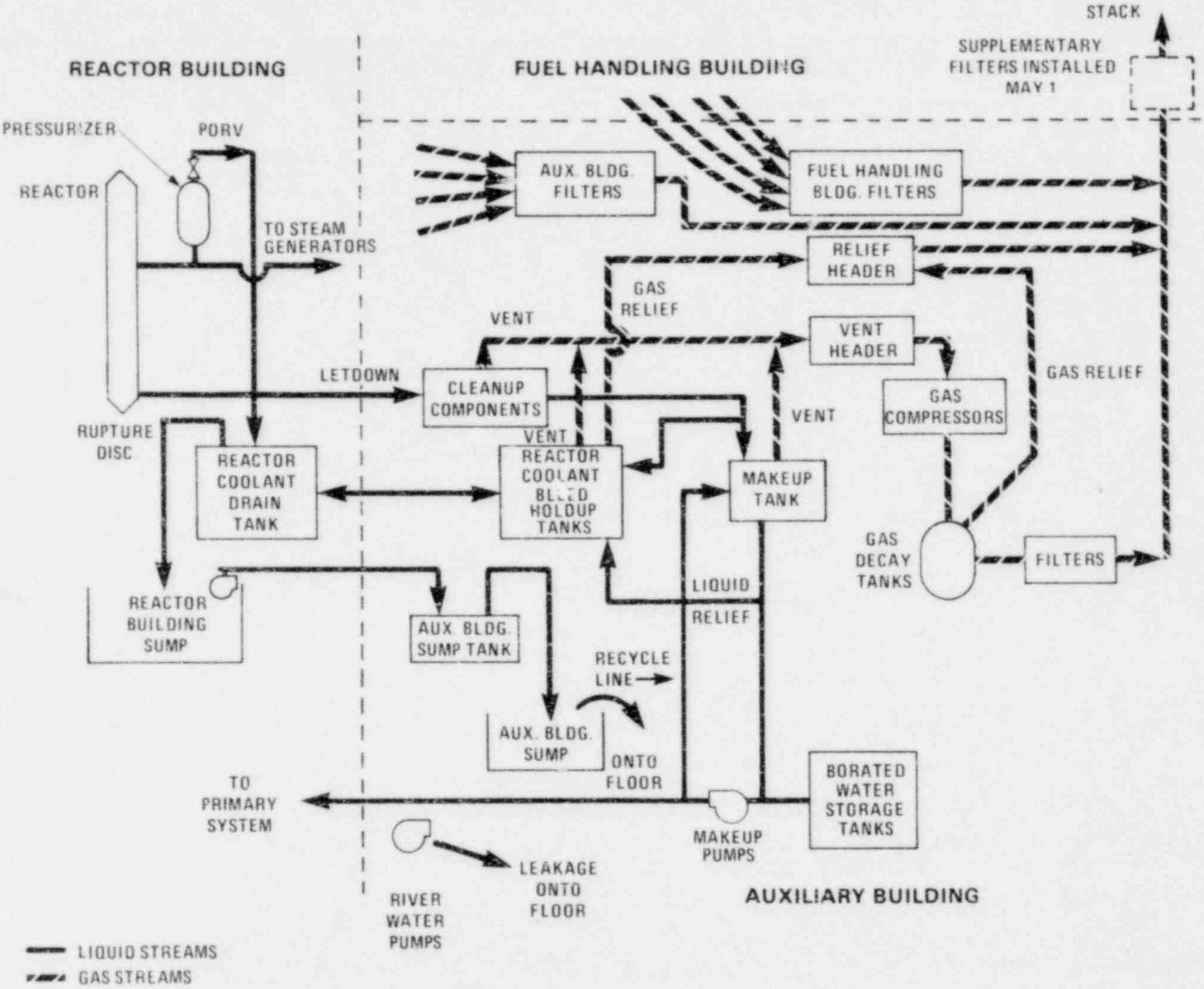


FIGURE II-12. Release Pathwa,

Continued operation of the letdown transferred primary coolant from the reactor to components in the auxiliary building. Pressure buildup in components due to degassing of the hydrogen and noble gases in the letdown system caused gaseous leakage to the auxiliary building and operation of relief and vent valves to release gaseous radioactivity to the auxiliary and fuel handling buildings and to the environment.

compressors. These two factors lessened the significance of the release pathway presented by the leaking waste gas system compressors.

Two minor gaseous leak paths existed—a failed rupture disc on the auxiliary building sump tank, and possible leakage of makeup tank vent valve MU-V-13. The sump tank rupture disc had failed prior to the accident, and any gaseous activity in the tank was released to the auxiliary building environment. This rupture disc has not been repaired. It has not been possible to verify whether leaks in MU-V-13 exist because of the high radiation levels in the area.

The radioactive noble gases and a small fraction of the iodines present in the water on the auxiliary building floors escaped into the building. This offgassing occurred primarily for the noble gases, because the iodines tend to remain in solution.[27] On Thursday morning, March 29, Met Ed recognized this pathway and attempted to minimize the releases of radioactive noble gases by placing sheets of polyethylene over the water. These pro-

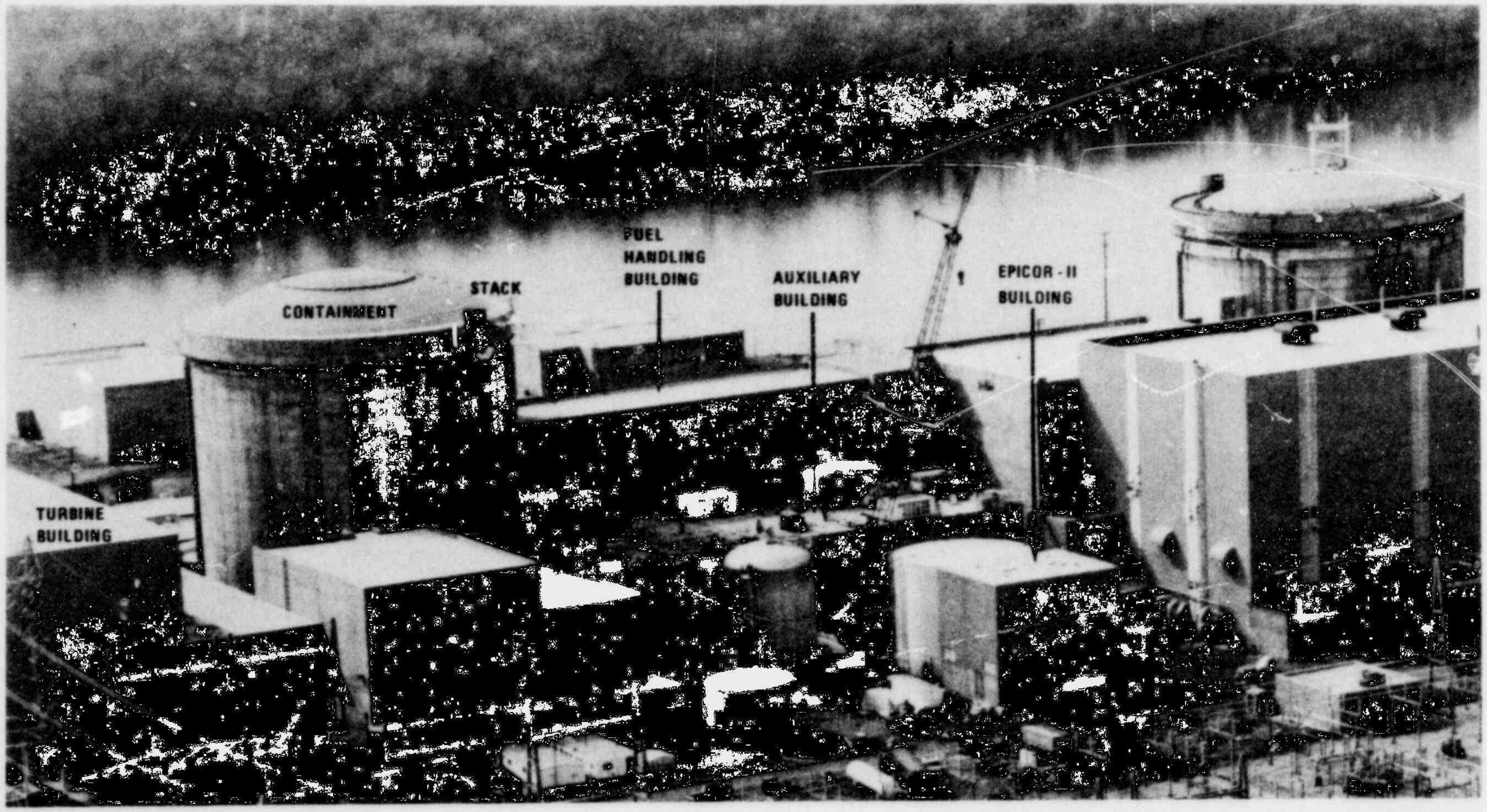


FIGURE II-13. General Building Arrangement tective efforts did not provide any substantive mitigation of releases because the sheets were not airtight.

Each time the makeup tank was vented, the radiation levels inside and outside the plant increased. The pathway for the gaseous activity from the makeup tank venting process and for other component vents is shown in Figure II-12. Cleanup components in the letdown and the makeup tank have manually operated vents that discharge to the vent header. During normal operation, vented gases are held up and filtered prior to release.

The makeup tank has a liquid relief to the reactor coolant bleed holdup tanks. The tank is designed to operate with approximately one-third of its volume as a gas space to allow gases from the cooled and depressurized primary coolant to evolve and be collected. Collection of noncondensible gases in the makeup tank caused a reduction in the letdown flow because of pressure buildup. This reduction of letdown flow became a concern in the early morning of March 29. As a result, manual ventings of the makeup tank to reduce pressure began at 4:35 a.m. on March 29. The venting process consisted of short bursts, with vent valve MU-V-13 being cycled open for short periods of time to minimize leakage of radioactive material. According to William Zewe, Shift Supervisor, venting of the makeup tank occurs only once every 2 or 3 months during normal operation to remove nonradioactive noncondensible gases and there is no standard operating procedure for venting the tank.[28] Nonetheless, on March 29, Met Ed wrote and approved operating procedures for the periodic venting of the makeup tank.[20]

The rate of pressure buildup in the makeup tank became too rapid to control with the cyclic opening of MU-V-13 during early Friday morning, March 30. The liquid relief on the makeup tank opened, allowing all of the contents in the tank to flow into the reactor coolant bleed holdup tanks. The makeup pumps (MU P-1A, 1B, and 1C) then switched suction to the borated water storage tank. This water bypassed the primary system and was recirculated to the makeup tank and to the reactor coolant bleed holdup tanks through the open liquid relief valve, thus depleting the supply of borated water.

It was crucial to reduce the pressure in the makeup tanks at this time for two reasons. First, the supply of borated water in the borated water storage tanks was being depleted. This supply was the only readily available source of borated water for continued boron control of the primary coolant. Second, the increase in pressure in the reactor coolant bleed holdup tanks through the open relief valve on the makeup tank increased the probability that the relief valves (20 psig setpoint, but pressures of greater than 30 psig were observed) on the bleed holdup tanks would open. The opening of the tanks would permit an uncontrolled release of gaseous radioactive material to the environment via the relief system.

A decision was made to vent the makeup tank continuously in an attempt to reduce pressure. During the morning of March 30, 1979, this action was suggested by Control Room Operator Craig Faust and all personnel present in the TMI-2 control room agreed.[29] At approximately 7:00 a.m. on March 30, MU-V-13 was opened. A caution tag was placed on the valve on March 31 at 11:15 p.m., stating, "Do not move this valve without Supt. or Shift permission (per J. Herbein)."

The opening of MU-V-13 at 7:10 a.m. on Friday, March 30 resulted in a momentary reading of 1200 mR/h, 130 feet above the TMI-2 stack. This reading was the event that apparently triggered the Friday evacuation recommendations. Leaving the valve open provided a continual pathway for gaseous radioactive material to enter the auxiliary building. Leaks in the vent header permitted the gases to enter the auxiliary and fuel handling buildings and be discharged through the stack. Since letdown flow is still being maintained, this release pathway still exists. However, all short-lived radionuclides in the reactor coolant have undergone significant decay since March 28, and releases of radioactive material from Three Mile Island Station are now negligible.

f. Source Terms for Releases of Radioactive Materials

Radiation monitor HP-R-219, located in the TMI-2 stack, is designed to measure the amount of radioactive material in the gaseous effluents of TMI-2. The monitor detects radioactive material in particulate form, radioiodine, and noble gas. The channels that detect noble gas went off scale before 8:00 a.m. on March 28, and consequently the recorded data are of little use in estimating a noble gas source term from the accident. Releases of radioactive material in particulate form were negligible because of the two banks of HEPA filters installed in the auxiliary and fuel handling building air filtration systems.

Radioiodine Source Term— Iodine releases have been calculated by analyses of the charcoal cartridges of HP-R-219. Beginning on March 28, these cartridges were periodically replaced. The cartridges that were removed were analyzed for their ^{131}I content. Table II-3 shows the results of the analyses through May 8. There were six time periods

TABLE II-3. I-131 releases

Time Period From	To	Curies I-131 Released for Time Period[1]	Cumulative Curies I-131[1] Released
0400 3/28	1900 3/28	0.22[2]	0.22
1900 3/28	1900 3/30	3.90	4.12
1900 3/30	2200 3/30	0.24[3]	4.36
2200 3/30	0600 4/1	0.31	4.67
0600 4/1	0315 4/3	1.57	6.24
0315 4/3	1905 4/3	0.13	6.37
1905 4/3	2232 4/3	0.09	6.46
2232 4/3	1830 4/5	1.15	7.61
1830 4/5	1516 4/6	0.03	7.64
1516 4/6	0600 4/7	0.36[3]	8.00
0600 4/7	0245 4/8	0.51	8.51
0245 4/8	0425 4/9	1.17	9.68
0425 4/9	0925 4/9	0.23[3]	9.91
0925 4/9	1608 4/10	0.05	9.96
1608 4/10	1840 4/11	0.12	10.08
1840 4/11	1920 4/11	0.01[3]	10.09
1920 4/11	2315 4/13	0.39	10.48
2315 4/13	1030 4/14	0.24[3]	10.72
1030 4/14	1915 4/14	0.19	10.91
1915 4/14	0522 4/15	0.24	11.15
0522 4/15	0804 4/15	0.08	11.23
0804 4/15	1802 4/15	0.51	11.74
1802 4/15	2140 4/15	0.09	11.83
2140 4/15	2346 4/15	0.05	11.88
2346 4/15	0408 4/16	0.10	11.98
0408 4/16	0758 4/16	0.08	12.06
0758 4/16	1156 4/16	0.07	12.13
1156 4/16	1550 4/16	0.05	12.18
1556 4/16	1810 4/16	0.09	12.27
1810 4/16	2356 4/16	0.13	12.40
2356 4/16	0402 4/17	0.04	12.44
0402 4/17	0835 4/17	0.05	12.49
0835 4/17	1226 4/17	0.03	12.52
1226 4/17	1634 4/17	0.03	12.55
1640 4/17	1946 4/17	0.06	12.61
1958 4/17	2357 4/17	0.07	12.68
2357 4/17	0405 4/18	0.08	12.76
0405 4/18	0550 4/18	0.05	12.81
0550 4/18	0800 4/18	0.05	12.86
0800 4/18	0945 4/18	0.02	12.88
0950 4/18	1200 4/18	0.01	12.89
1204 4/18	1647 4/18	0.03	12.92
1650 4/18	1823 4/18	0.01	12.93
1823 4/18	2347 4/18	0.07[3]	13.00
2347 4/18	0358 4/19	0.05	13.05
0358 4/19	0800 4/19	0.03	13.08
0803 4/19	1210 4/19	0.03	13.11
1212 4/19	1355 4/19	0.00	13.11
1355 4/19	1725 4/19	0.05	13.16
1728 4/19	2025 4/19	0.05	13.21
2025 4/19	0001 4/20	0.04	13.25
0001 4/20	0351 4/20	0.11	13.36
0351 4/20	0821 4/20	0.10	13.46
0821 4/20	1105 4/20	0.05	13.51
1105 4/20	1300 4/20	0.05	13.56
1300 4/20	1620 4/20	0.04	13.60
1620 4/20	2019 4/20	0.04	13.64
2023 4/20	2204 4/20	0.03	13.67
2249 4/20	0317 4/21	0.03	13.70
0320 4/21	0402 4/21	0.03	13.73
0404 4/21	0819 4/21	0.02	13.75
0819 4/21	1201 4/21	0.02	13.77

TABLE II-3. I-131 releases—Continued

Time Period From	To	Curies Iodine-131 Released for Time Period[1]	Cumulative Curies I-131[1] Released
1204 4/21	1624 4/21	0.02	13.79
1628 4/21	2017 4/21	0.02	13.81
2018 4/21	0103 4/22	0.03	13.84
0105 4/22	0441 4/22	0.02	13.86
0447 4/22	0804 4/22	0.02	13.88
0807 4/22	1229 4/22	0.02	13.90
1230 4/22	1621 4/22	0.03	13.93
1624 4/22	2024 4/22	0.04	13.97
2036 4/22	2130 4/22	0.00	13.97
2130 4/22	0004 4/23	0.03	14.00
0007 4/23	0406 4/23	0.03	14.03
0358 4/23	0758 4/23	0.02	14.05
0801 4/23	1201 4/23	0.02	14.07
1223 4/23	1614 4/23	0.05	14.12
1617 4/23	2010 4/23	0.01	14.13
2014 4/23	2156 4/23	0.01	14.14
2159 4/23	0015 4/24	0.01	14.15
0004 4/24	0404 4/24	0.02	14.17
0408 4/24	0637 4/24	0.01	14.18
0642 4/24	0813 4/24	0.01	14.19
0815 4/24	1215 4/24	0.01	14.20
1217 4/24	1600 4/24	0.01	14.21
1600 4/24	1955 4/24	0.02	14.23
1958 4/24	0001 4/25	0.01	14.24
0004 4/25	0512 4/25	0.01	14.25
0520 4/25	0658 4/25	0.00	14.25
0701 4/25	1200 4/25	0.01	14.26
1200 4/25	1555 4/25	0.01	14.27
1557 4/25	2010 4/25	0.01	14.28
2013 4/25	0013 4/26	0.01	14.29
0016 4/26	0357 4/26	0.01	14.30
0400 4/26	0802 4/26	0.00	14.30
0805 4/26	1220 4/26	0.01	14.31
1220 4/26	1558 4/26	0.00	14.31
1606 4/26	1913 4/26	0.01	14.32
1913 4/26	0006 4/27	0.01	14.33
0011 4/27	0038 4/28	0.03	14.36
0042 4/28	0830 4/28	0.00	14.36
0832 4/28	1625 4/28	0.01	14.37
1645 4/28	0025 4/29	0.01	14.38
0028 4/29	0008 4/30	0.05	14.43
0010 4/30	0010 5/1	0.04	14.47
0000 5/1	0000 5/2	0.04	14.51
0000 5/2	0000 5/3	0.01	14.52
0000 5/3	0000 5/4	0.01	14.53
0000 5/4	0000 5/5	0.01	14.54
0000 5/5	0000 5/6	0.01	14.55
0000 5/6	0000 5/7	0.01	14.56
0000 5/7	0000 5/8	neg.	14.56
0000 5/8	0000 5/9	neg.	14.56

[1]Source of Data: TDR-TMI-116, "Assessment of Offsite Radiation Doses from the Three Mile Island Unit 2 Accident," July 31, 1979.

[2]Based on auxiliary and fuel handling building release rates.

[3]Interpolated value from higher release rate for two surrounding time periods.

for which the charcoal sample cartridges were lost or not analyzed in a timely manner. In those instances, marked by an asterisk in Table II-3, interpolated values were obtained by assuming that the higher release rate for the time periods immediately preceding or following the period of interpolation existed for the period of interpolation. The operation of the filter systems was not disturbed during these periods, and this method of interpolation is therefore considered to be conservative. Releases of radioiodine after May 8 were negligible because of radioactive decay of the ^{131}I (8-day half-life) and the installation of higher efficiency filtration systems on the auxiliary and fuel handling building exhausts. We find that the calculated ^{131}I source term for the accident is approximately 15 Ci, with the time-dependent release rates as specified in Table II-3.

The ^{131}I source term of approximately 15 Ci is in close agreement with that calculated by Met Ed[30] by an environmental consultant for Met Ed[31] and in substantial agreement with a source term of approximately 27 Ci estimated by an air cleaning consultant [Nuclear Consulting Services, Inc. (NUCON) of Columbus, Ohio].[32] The technique employed by NUCON for calculating the iodine source term was a layering of the spent carbon trays, analysis for ^{131}I, then integration of the amount of ^{131}I over the bed depth, done independently for the auxiliary and fuel handling buildings. NUCON acknowledged that the estimate was high, because the fuel handling building carbon trays with the highest, rather than average, activity were analyzed. The activity in these trays was two to three times higher than the average. When this is considered, the source term calculated is in close agreement with those already discussed.

It is also possible to confirm the ^{131}I source terms by a calculation employing the amount of ^{131}I captured by the different carbon adsorbers and the measured efficiency of the carbon at various operating conditions. Data on the various iodine species (elemental iodine, methyl iodide, and hypoiodous acid) are available for an auxiliary building air sample taken on April 8, 1979.[33] The removal efficiency of the various carbon adsorbers for these species at 95% relative humidity has been determined.[32,34] The total iodine estimated to be released as a result of these calculations is 32 Ci (see Table II-4). This estimate is presented as an upper bound for iodine releases, since carbon becomes less efficient for iodine species (particularly methyl iodide) at relative humidities above 85%. The estimate is high by approximately a factor of 2 to 3 because the highest, not average, activity carbon cells from the fuel handling building were analyzed to determine filter efficiencies for the various species.

It should be noted that the highest relative humidity estimated to exist inside the auxiliary building during and subsequent to the accident is 80%. This estimate is based on measured outside-humidity conditions in Harrisburg at the time of the accident and assumes a depth of 3 inches of water throughout the floor of the 280-foot elevation in the auxiliary building.[35] This fact also suggests that the estimated ^{131}I release of 32 Ci is high.

Data are also available for the removal efficiency of the carbon installed at the time of the accident at an operating condition of 30% relative humidity.[32] This was the lowest probable relative humidity inside the auxiliary building after the accident, and will result in carbon performing at its greatest efficiency for the removal of iodine species. Table II-5 contains an estimate of the lower limit for the iodine source term of 17 Ci. Since the fuel handling building trays are considered high in curie content, by a factor of 2 or 3, this correction (using 2 to 1) yields an actual lower bound of 10 Ci as the iodine source term. We find that the iodine source term of 15 Ci as presented in Table II-3 is substantiated by a number of other independent analyses, and is a valid estimate of the quantity of ^{131}I released to the environment from the accident. Further, based on the data in Tables II-4 and II-5, we find that a nonengineered safety feature filter system designed for normal operation only, i.e., the auxiliary building exhaust ventilation filtration system, greatly reduced the quantity of radioiodine released to the environment.

Noble Gas Source Term—The quantity of radioactive noble gases released because of the accident was first estimated by a back-calculation based on environmental thermoluminescent dosimeters (TLDs). These TLDs provide the best estimate of the integrated radiation dose at a specific location, and can yield a source term when an isotopic spectrum and meteorological conditions are considered. This calculation, when done for various time periods and TLD locations, results in an accident source term of approximately 13 x 10^6 Ci of noble gas as ^{133}Xe.[36]

A second method of calculating a noble gas source term is also based on TLD values.[37,38] In this method, a set of trial release rates for each isotope is assumed, proportional to inplant area radiation monitors and a dose equivalence factor (average gamma energy per disintegration). With the release rates and measured onsite meteorological data for the time of the accident, gamma doses can

TABLE II-4. Calculated ^{131}I releases at 95% relative humidity

Filter System	Species*		Curies Captured	Carbon Efficiency	Curies Released
Auxiliary	I_2	35%	4.2	99.9	0
Building	CH_3I	40%	4.8	69.5	2.1
A Train	HOI	25%	3.0	99.8	0
Auxiliary	I_2	35%	5.1	99.8	0.01
Building	CH_3I	40%	4.8	56.0	4.6
B Train	HOI	25%	3.6	99.7	0.01
Fuel Handling	I_2	35%	12.8	97.2	0.37
Building	CH_3I	40%	14.7	75.6	4.7
A Train	HOI	25%	9.2	99.9	0.01
Fuel Handling	I_2	35%	16.9	98.5	0.26
Building	CH_3I	40%	19.3	49.1	20
B Train	HOI	25%	12.0	99.3	0.28
Total			112		32

*Assuming no other species present, and identical distributions in both buildings. Ignores 7.3×10^{-8} μCi/cc particulates. The concentrations of these species are 6.7×10^{-8} μCi/cc for elemental iodine, 7.9×10^{-8} μCi/cc for methyl iodide (CH_3I), and 4.8×10^{-8} μCi/cc for hypoiodous acid (HOI).

TABLE II-5. Calculated I-131 releases at TMI at 30% relative humidity

Filter System	Curies Captured	CH_3I Efficiency	Curies Released
Auxiliary Building A Train	12	91.2	1.2
Auxiliary Building B Train	14.6	88.8	1.8
Fuel Handling Building A Train	36.7	97.1	1.1
Fuel Handling Building B Train	48.3	78.7	13.0
Total	112		17

be computed for each onsite TLD monitor site for which exposures are available. A comparison of calculated and measured exposures tests the accuracy of the calculated release rates. When a best fit of calculated versus measured exposures is obtained by varying the release rates, for each time period with available data, the release rates are added to present a total source term of 10×10^6 Ci noble gases, with a radionuclide content as follows: 8.3×10^6 Ci ^{133}Xe; 1.5×10^6 Ci ^{135}Xe; 1.7×10^5 Ci ^{133m}Xe; 1.4×10^5 Ci ^{135m}Xe and 6.1×10^4 Ci ^{88}Kr.

The most reliable method of determining the releases of radioactive noble gases would be via direct measurement of releases from the TMI-2 stack. However, the radiation monitor (HP-R-219) installed on the stack for this purpose was designed for normal operation only, and went off scale at approximately 7:45 a.m. on March 28. Direct measurement of the releases thus was not possible, and attempts were made to calculate the noble gas releases by an indirect method.

Analysis of area radiation monitors showed that an external area gamma monitor (HP-R-3236), located on the 305-foot elevation of the auxiliary building, remained on scale for the duration of the accident. This monitor is located between the two reactor building purge air filtration units and is shielded from the operating air filtration units for the auxiliary and fuel handling buildings. The monitor was sensitive to changes in the radioactive release rates because it was close to the exhaust ductwork. A review of the readout from HP-R-3236 shows its highest reading (which was still on scale) after the accident to be 6.5 R/h (at 11:00 p.m. on March 28). A correlation between the reading on HP-R-3236 and HP-R-219, when both were on scale, permits estimation of these releases from the stack for the duration of the accident. The known flow rate in the stack allows the calculation of the integrated noble gas source term from the area gamma monitor (HP-R-3236). This calculation results in an estimate of 2.37×10^6 Ci of total noble gases being released because of the accident. This method of calculation was employed by the Presidential Commission on the Accident at Three Mile Island.[39]

An isotopic distribution can be obtained by calculating the core distribution at the time of reactor shutdown and decaying each isotope 3 hours to account for transit time to the building environments. This calculation assumes all noble gas isotopes are transported equivalently from the core to the building environments. The calculation yields an estimate of release of 1.51×10^6 Ci ^{133}Xe; 0.35×10^6 Ci ^{88}Kr; 0.28×10^6 Ci ^{135}Xe; 0.21×10^6 Ci ^{133m}Xe; and 0.01×10^6 Ci ^{135}Xe, which is consistent with the estimate made by the President's Commission. We find this noble gas source term to be the best estimate because it is based on an extrapolation of measured releases in proximity to the source and not TLD exposures at remote locations.

g. Mitigation of Releases of Radioactive Materials

The buildings and equipment at the Three Mile Island Station provided substantial mitigation of the release of radioactive material to the environment. The primary coolant absorbed significant quantities of radioactive material.[40] Normal activity of the primary coolant is approximately 1 μCi/cm^3 and was actually 0.4 μCi/cm^3 just prior to the accident.[41] During the accident, the activity increased to more than 20 000 μCi/cm^3. The containment structure prevented large quantities of radioactive material from being released. Six months after the accident, there were approximately 50 000 Ci of noble gases (^{85}Kr) and 600 000 gallons of contaminated water inside that structure. The piping and tanks in the auxiliary building also retained quantities of radioactive material. These provided either holdup, to allow short-lived radionuclides to decay, or containment of the radionuclides in a form that would allow treatment after the accident.

Additionally, a number of installed plant components were actively used after the accident to mitigate the release of radioactive materials. The most important of these were the ventilation systems and the exhaust air filters installed in the auxiliary and fuel handling buildings. As discussed in Section II.B.2.a, these systems are designed to filter all of the exhaust ventilation separately from these two buildings prior to release through the plant stack. The auxiliary building filtration system is designed for normal operation only, and would be expected to remove 99% of particulate material and 90% of radioiodines. The fuel handling building filtration system is designed for both normal operation and use after a fuel handling accident in the fuel handling building, and would be expected to remove 99% of particulate material and 95% of radioiodines for both conditions. The systems are not designed for noble gas holdup.

Both filter systems were in operation during the initial stages of the emergency. Attempts were made to reduce releases of radioactive materials by shutting off the exhaust fans a number of times between March 28 and 30. These stoppages, however, resulted in increased radiation levels inside the plant, including the control rooms. The ventilation systems have been in continuous operation since March 30, except for minor maintenance periods.

The cleanup components installed in the filter systems were built and purchased according to specifications that the NRC staff found acceptable. The specifications for the carbon (see Section II.B.2.a) were acceptable to the NRC staff when reviewed in 1975. Prior to the accident, these specifications were upgraded and the quality of the carbon used in TMI-2 would not be acceptable in 1979.[42] The HEPA filters satisfied Military Specification MIL-F-51068, "Filter, Particulate, High Efficiency, Fire-Resistant,"[43] which is the industry standard. The same quality HEPA filters were installed in both of the filtration systems even though only the fuel handling building system was designated as safety grade.

The original design of the filter systems included bypass dampers to allow ventilation air to bypass the cleanup components during periods of low radioactivity, thus prolonging the component life. However, testing of these dampers indicated that they leaked at a 15% rate.[44] The dampers were sealed.[45] After sealing, testing proved satisfactory but resulted in all air being directed through the cleanup components whenever the ventilation system was in operation. The dampers opened and shut sporadically during the accident, and we find that the dampers did not permit the filter system to operate as effectively as possible during the accident. The operation of these filter banks during the year after completion of acceptance testing, combined with their exposure to paint fumes, resulted in degraded carbon being in the filters at the time of the accident.

The condition of the filtration systems after the accident was determined by two methods. The first method involved analysis of building air samples taken upstream and downstream of the filters. The overall decontamination factor was 1.2.[33] As a result, a decision to changeout the carbon in the filter system was made. The changeout was completed in mid- to late April. The spent carbon was sent off site for laboratory analysis by an independent consulting corporation.[32] The analyses indicated that, of the two filter trains, the fuel handling building ventilation exhaust system removed more radioiodine than the auxiliary building system. The variablity in performance between these systems was due to (1) an imbalance of ventilation flows (ventilation system balancing was never required or performed); (2) a faulty inlet damper that would sporadically open and shut;[46] and (3) the location of the vent header in the auxiliary building, which results in the air around the header actually being ventilated by the fuel handling building system.

Samples of the carbon taken from trains A and B of the auxiliary building and trains A and B of the fuel handling building ventilation systems were tested in accordance with Regulatory Guide 1.52 (Revision 1) with a pre-equilibration of 16 hours at the stated relative humidity. Tables II-4 and II-5 show the results of these tests for removal efficiency of the carbon in place at the time of the accident. Removal efficiencies ranged from a low of approximately 49% for methyl iodide at 95% relative humidity in fuel handling building train B, to over 99.9% for elemental iodine at 95% relative humidity in auxiliary building train A. Table II-4 shows that a total of 112 Ci of ^{131}I (all species) was captured by the carbon in the four filter trains. The amount of radioiodine captured is compared to a release of approximately 13 Ci (see Table II-3) to the time of filter changeout. We find that the filter systems installed at the time of the accident provided a decontamination factor of 9.5 (equivalent to an efficiency of 89.5%) for all species of iodine.

The carbon installed at the time of the accident was also analyzed for water content and pH as a function of bed depth. Low pH values can be correlated to an exhausted carbon that has low removal efficiencies.[47] These values are tabulated in Table II-6, along with the activities determined to be on each layer of carbon. Values for moisture content are listed only for train A of the auxiliary building. The other samples were sent unsealed and absorbed moisture in transit, invalidating any determination of water content.

A comparison of the status of the carbon as determined by the two methods discussed above shows discrepancies. Inplace tests indicated the carbon was severely degraded, while after-the-fact laboratory testing showed that the carbon would still perform satisfactorily. Both methods have inherent weaknesses. Inplace air samples may not be representative and give only an instantaneous reading. Laboratory tests suffer from procedural problems (such as whether to pre-equilibrate the carbon to the stated relative humidity prior to test) and also from noble gas contamination of the carbon. We find that neither inplace testing of the filter systems nor the laboratory testing of the carbon was adequate to characterize the condition of the carbon after the accident.

Changeout of the carbon adsorbers in each filter system was accompanied by concurrent changing of all the HEPA filters in these components. These HEPAs were visually examined before changeout and were intact and in satisfactory condition, but were damaged during changeout of the carbon trays. Unfortunately, no used HEPA filters or sections of filter media were retained for analysis.

Twenty-seven percent of the iodine species[33] was in particulate form. In addition, the filter sys-

TABLE II-6. Analyses of carbon exposed during the accident

		Auxiliary Building				Fuel Handling Building			
		Train A		Train B		Train A		Train B	
Depth (inches)	H_2O %	pH	Activity μCi/gm	pH	Activity μCi/gm	pH	Activity μCi/gm	pH	Activity μCi/gm
First 0.5	2.53	4.3	10.1	3.4	15.4	4.1	34.5	ND*	76.1
Second 0.5	3.41	4.5	4.4	4.0	4.9	4.7	6.4	3.4	26.5
Third 0.5	2.58	5.9	2.5	4.3	3.1	4.5	0.9	3.9	18.5
Fourth 0.5	1.05	5.6	1.7	4.6	1.5	4.5	0.1	3.9	15.1

*ND=Not determined

terns contain two individual banks of HEPA filters (one upstream of the carbon and one downstream), each of which were acceptance tested to greater than 99.95% leak-tightness. Thus, we find that the HEPA filters removed essentially all of the particulates generated.

After completion of the postaccident changeout of all the cleanup components in both trains of the auxiliary and fuel handling building systems, inplace leak-testing was not performed to verify the leak-tightness of these systems. A visual inspection was considered sufficient because of (1) the necessity to return the filtration units to operation as soon as possible, (2) the lack of manpower, and (3) the potential for increased worker exposures. Although it is good engineering practice to leak-test filter systems in place after changeout, the decision to defer leak-testing of these filter systems was warranted. Because further releases through these filter systems have been negligible,[48] the performance of the systems has demonstrated their integrity.

The carbon used as replacement in the four filter systems was impregnated with either stable iodide, as KI_3, or a mixture of KI and triethylenediamine (TEDA). Problems in readily obtaining replacement cells were encountered because the TMI-2 cells are 40 inches long, rather than the standard industry length of 30 inches.[49] This discrepancy and possible problems arising from the use of 40-inch trays were reported to Burns and Roe on November 20, 1973,[50] but Burns and Roe required the 40-inch trays. Thus, a special size cell was needed for replacement and it was difficult to quickly obtain a sufficient number. The cells were refilled and reinserted into the systems. All trays were refilled with coimpregnated carbon, except for 79 trays in the auxiliary building train B filter system, which were refilled with carbon impregnated only with stable iodine, as KI_3. The use of coimpregnated carbon was desirable because it is better able to remove methyl iodide at high humidities than is carbon impregnated with stable iodine, as KI_3.

Appendix II.2 presents the available data on the carbon used as replacement as a function of time. Although test procedures conform to the recommendations of Regulatory Guide 1.52 (Revision 1), the data lack consistency. Two types of impregnated carbon were obtained from two sources (MSA and NUCON), and carbon sampling methods did not conform to industry standards.[51] The carbon samples removed for analysis were shipped in plastic bags, with incomplete data on cells sampled, location in bank, date obtained, and type of charcoal. There is also no means to ascertain whether the sample was properly mixed to assure homogeneity prior to shipment. In addition, different cells with different operating histories have been removed for sampling. This removal resulted in some of the carbon being tested that had been used to refill a test cell at the previous sampling, and not testing other carbon that had been in service since the changeout. These sampling problems resulted in nonrepresentative samples with results that may be neither reproducible nor valid. However, based on the remaining adsorptive capacity of the carbon after approximately 6 months of service of 83% to 99% (see Table 1 of Appendix II.2) and the negligible iodine releases after replacing the carbon, we find that the coimpregnated carbon has performed satisfactorily in reducing radioiodine releases to the environment.

When Met Ed realized the severity of the accident and the potential for release of significant quantities of radioiodine to the environment, it decided to obtain a supplementary filtration system to further mitigate radioiodine releases. The decision was made prior to the large influx of NRC personnel to the site on Friday, March 30. Met Ed decided to install four separate 30 000-cfm filter units on the roof of the auxiliary building. These units consist of heaters, prefilters, HEPA filters, a 2-inch-deep bed of KI_3-impregnated carbon, and a second bank of HEPA filters. The units were obtained in the first week in April from MSA, which had already shipped the units to Richland, Wash. for installation in the Washington Public Power Supply System's (WPPSS) Nuclear Units 1 and 4. The filter systems had not been installed in Washington, and were immediately air lifted to the Three Mile Island Station for installation. By mid-May, the filter systems had been installed on the roof of the auxiliary building to filter all of the ventilation air from the auxiliary and fuel handling buildings prior to release. The filter units are installed in series with the existing auxiliary and fuel handling building filters, and therefore all ventilation air has been filtered twice before release to the environment.

The TMI-2 stack was capped on May 20, ensuring that all ventilation exhaust flows were through the supplementary auxiliary building filtration system. An effluent monitor downstream of each filter train measures releases of iodine, particulates, and noble gases. Since May 20, three of the four filter systems have been on line at all times, and releases have been negligible.

The cleanup components installed in the supplementary auxiliary building filtration system were the components marked for use at the WPPSS units. The HEPA filters were specified to satisfy Military Specification MIL-F-51068D, which is the industry standard.[43] An inplace leak test was also performed on each bank as an acceptance test, and the results showed a minimum leak-tightness of 99.85%. It should be emphasized that all ventilation exhaust air was treated by four individual banks of HEPA filters after the installation of the supplementary auxiliary building filtration system: two banks in the filter systems inside the building, and two banks in the supplementary auxiliary building filtration systems installed on the roof.

The carbon installed in the supplementary auxiliary building filtration system satisfied the specification for the WPPSS units, and was certified as passing a laboratory test demonstrating the ability to remove at least 95% of methyl iodide when tested at 95% relative humidity and 212°F, and 99.9% of elemental iodine when tested at the same conditions, for each batch of carbon. Of the nine batches of carbon tested in March 1978 by MSA and certified as acceptable, the minimum methyl iodide removal efficiency was 96.28%, and the minimum elemental iodine removal efficiency was 99.87%.[44]

Attempts to evaluate the performance of the carbon in the supplementary auxiliary building filters as a function of time have been hampered due to the inability to obtain a representative adsorbent sample from the installed bank.[52] The available data are included in Appendix II.2 (Appendix Table II-3), and although the carbon shows degradation, it is still extremely effective in removing radioiodines (a minimum of 84.3% at 95% relative humidity and 99.4% at 30% relative humidity for methyl iodide removal) after 5 months of service. This can be attributed to the existing auxiliary and fuel handling building filters acting as guard beds and removing the bulk of the nonradioactive contaminants. We find that the use of the supplementary auxiliary building filtration systems has mitigated the releases of radioactive material.

Two other filtration systems were added to exhaust streams from TMI-2 in the first few weeks after the accident. These additional systems were not as significant in mitigating the releases as the supplementary auxiliary building filter systems. Both systems were supplied by American Air Filter Company in Louisville, Ky. The first was a small (less than 1000 cubic feet per minute) system installed on the exhaust of the radwaste chemical lab trailer outside of the TMI-2 turbine building wall. The cleanup components consisted of a HEPA filter and 2 inches of KI_3-impregnated carbon. The system was installed in early April, was put in operation after satisfactory leak-testing on May 2, 1979 (HEPA filter and carbon tray 99.99% leak-tight), and has not been retested due to its minimal impact on plant operation or releases.

The second system was a 1000-cfm filtration system installed on the condenser vacuum pump exhaust. This exhaust does not normally contain significant amounts of radioactive material and is not treated in a pressurized water reactor. However, it was determined following the accident that this exhaust was contaminating the auxiliary building; and the system was installed in early April, leak-tested on April 9, and put in operation. The system consists of a heater, an upstream bank of HEPA filters, two 2-inch-deep carbon adsorbers (KI_3-impregnated) in series, and a downstream bank of HEPA filters. Leak-testing proved acceptable (99.99% for both HEPA banks, and 99.98% for the one carbon bank tested.) The carbon was certified as removing 98.7% of methyl iodide when tested in the laboratory at 130°C and 95% relative humidity. The performance of this carbon has been followed as a function of exposure time, and the results are included in Appendix II.2. The same sampling and reproducibility problems exist for this system as for the auxiliary, fuel handling, and supplementary auxiliary building filter systems. Removal efficiency is still approximately 90% for methyl iodide (September 1979), and the carbon has not been changed to date.

In addition to the installation of supplementary filtration systems to assist in mitigating the release of radioactive particulates and iodine, attempts were made to reduce the impact of noncondensible gases and noble gases that were stripped out of the primary coolant in the letdown line of the makeup and purification system. These gases were overpressurizing the makeup tank and the vent header, and were resulting in increasing pressures in the waste gas decay tanks. Met Ed was aware of this situation on Wednesday, March 28, and began to install copper tubing from each waste gas decay tank and the makeup tank back into containment that day. Flame arresters and sampling ports were installed in the lines. The connection to containment was made through an existing hydrogen purge penetration (R-57/C). Since the containment structure has a large volume (approximately 2 million cubic feet) and is designed to withstand pressures of at least 50 pounds per square inch (psi), the decision to use the containment was based on sound technical judgment.

On Friday morning, March 30, the pressure in the waste gas decay tanks was approximately 80 psig, and there was concern that the setpoint of 120 psig on the relief valves would be achieved. If this occurred, the highly radioactive gases would be released through the relief valve vent header and would move directly to the stack. Attempts were made on Friday afternoon to transfer these gases back to containment. The first attempts showed leakage in the tubing, but after repairs further attempts were successful. The line installed from the makeup tank back to containment was completed on April 12, but no records have been found indicating that this line was ever used for transferring gases back to containment. Transferring gases back to containment via vent lines and the use of containment as a large waste gas decay tank proved to be extremely effective in allowing plant operations to continue by maintaining letdown flow.

Based on the high activities in the various radwaste system components after the accident, the overflow of liquid tanks, and the overpressurization of components due to the gaseous fission products, we find that the design bases of the radwaste systems were exceeded, and that a number of radwaste system modifications that assisted in mitigating the releases of radioactive materials to the environment were made after March 28, 1979. These included a supplementary auxiliary building air filtration system to filter all ventilation exhaust air from the auxiliary and fuel handling buildings, a condenser vacuum pump air filtration system, and vent lines from the makeup tank and waste gas decay tanks to transfer gases back into containment.

We find that the two filtration systems operating at the time of the accident to reduce releases of radioactive materials to the environment (auxiliary and fuel handling buildings), had identical safety grade cleanup components, and that the safety grade versus nonsafety grade designation was meaningless during the accident. Finally, we find that although the design bases of the radwaste systems were exceeded, the systems as operating at the time of the accident, and the additional actions taken, provided significant mitigation of the release of radioactive materials.

h. Recovery Operations

The recovery operation for TMI-2 includes treating gaseous and liquid radioactive materials that remain in various plant structures. Radioactive gases are primarily within the containment structure. Because of radioactive decay since March 28, ^{85}Kr (10.3 year half-life) is the only radionuclide with measurable activity. It is present in a concentration of approximately 0.78 μCi/cm^3, which for the 2.1 x 10^6 cubic feet containment volume equates to approximately 48 000 Ci. No definite plans have been established for treating the ^{85}Kr. Viable options include releasing the gas to the environment untreated during favorable meteorological conditions, holding up the krypton on a large (tens of thousands of pounds) bed of carbon that could be cooled to increase the adsorptive capacity, pressurizing the gas into tanks for storage, or cryogenically distilling the gas to remove the krypton. Atmospheric dilution under favorable meteorological conditions would result in atmospheric concentrations to levels below the maximum permissible concentrations in 10 C.F.R. Part 20 for unrestricted areas. This option is easiest to implement, and will not result in significant exposures to the public.

There are two types of liquid radwaste in TMI-2 components that need to be processed. The first is 600 000 gallons of highly radioactive liquid contained entirely within the containment structure. The radioactive composition of the liquid was last determined on August 28, 1979, as listed in Table II-7.[53] Plans for treatment have not been finalized, but two systems under consideration are a demineralizer system submerged in the TMI-2 fuel pool and an evaporation and solidification system which would require a new building to be constructed to house all the treatment components. Designs for any system built will need to consider the additional

TABLE II-7. Analysis of TMI-2 containment building water

Isotope	μCi/ml Activity*
Hydrogen-3 (Tritium)	1.0
Strontium (89 and 90)	45
Strontium-90	2.8
Zirconium-95	1.8×10^{-3}
Niobium-95	5.0×10^{-3}
Ruthenium-103	5.7×10^{-3}
Ruthenium-106	7.0×10^{-3}
Tin-113	5.0×10^{-4}
Antimony-125	1.5×10^{-2}
Tellurium-129	1.2×10^{-2}
Iodine-129	1.5×10^{-5}
Iodine-131	1.2×10^{-2}
Cesium-134	40
Cerium-134	5.6×10^{-3}
Cerium-137	2.5×10^{-2}
Cesium-137	1.8×10^{2}
Lanthanum-140	7.1×10^{-2}
Cerium-141	1.2×10^{-3}
Cerium-144	6.3×10^{-3}
Barium-1[illegible]	1.3×10^{-3}

*Average of three samples taken August 28, 1979.

3 million gallons of water expected to be generated as a result of decontamination.

The second type of radwaste that needs to be processed is intermediate level liquid (defined as having ^{131}I and ^{137}Cs concentrations greater than 1 μCi/ml but less than 100 μCi/ml) contained in various TMI-2 auxiliary building tanks. This radwaste resulted from (1) inventory existing prior to the accident, (2) contaminated water transferred from the reactor containment building sump to the auxiliary building during the early phases of the accident, (3) letdown from the reactor coolant system, and (4) normal continued leakage of system components. The significant radionuclide present is ^{137}Cs, with a half-life of 30 years. Approximately 280 000 gallons of intermediate level waste exists in the auxiliary building tanks, as indicated in Table II-8. The radioactive inventory in each tank as of June 15, 1979 is tabulated in Table II-9. For comparison purposes, normal primary coolant activity is expected to approximate 1 μCi/ml total for all radionuclides except tritium.

Shielded piping has been installed from tanks in the auxiliary building to the chemical cleaning building, located on the east side of the island between TMI-1 and 2. This building, originally intended for the cleaning of steam generators, now houses the processing system for the intermediate level liquid waste. This system, known as EPICOR-II, has been specifically designed and constructed for the purpose of processing the TMI-2 intermediate level liquid radwaste contained in the auxiliary building tanks. It consists of a prefilter/demineralizer designed to remove particulate radioactive wastes, cesium and other cationic radionuclides; a demineralizer for further removal of cationic radionuclides; another demineralizer for removal of both cationic and anionic (iodine) radionuclides; tanks; pumps; transfer piping; and instrumentation. After processing, the water is collected in the clean water receiving tank (133 000 gallon capacity) where it is sampled and analyzed. The results of this analysis will determine whether the treated water is transferred back to either TMI-1 or 2 for storage until ultimate disposal, or transferred to the off-specification water receiving/batch tank (95 000-gallon capacity) for reprocessing through EPICOR-II.

Changeout of the media in the prefilter/demineralizer and the demineralizers will be accomplished remotely. Cameras located in an adjacent structure will allow observation and control of the spent components during transport on an overhead monorail to a truck adjacent to the building. The components will be replaced on predetermined contact exposure rates, ranging from 3 R/h to 100 R/h for the various components. Approximately 50 changes of prefilter/demineralizers and demineralizers are expected for the processing of intermediate level TMI-2 liquid waste, based on ion-exchange capacity. This results in a total volume of 2500 cubic feet of spent resins. The casks will be temporarily stored on site, then the wastes solidified prior to offsite disposal in an approved facility.

The NRC published an environmental assessment of the operation of EPICOR-II on August 14, 1979, NUREG-0591, "Environmental Assessment Use of EPICOR-II at Three Mile Island, Unit 2." Numerous public comments were received and answered, and on October 16, 1979, an order was issued by the Commission to begin operation of EPICOR-II. The system began operation the week of October 22, 1979.

TABLE II-8. Radioactive water volumes in TMI-2 auxiliary building tanks

Tank	Volume (gallons)
Reactor Coolant Bleed Holdup Tank A	77 250
Reactor Coolant Bleed Holdup Tank B	77 250
Reactor Coolant Bleed Holdup Tank C	77 250
Neutralizer Tank A	8 780
Neutralizer Tank B	8 780
Miscellaneous Waste Holdup Tank, Auxiliary Building Sump and Sump Tank, Miscellaneous Sumps	13 500
Waste Evaporator Condensate Tanks, Contaminated Drain Tanks	16 200
TOTAL	279 000

i. Summary of Findings and Recommendations for Section II.B.2

Findings

We find that:

- although the design bases of Three Mile Island Station's radwaste systems were exceeded, the systems as operating at the time of the accident provided significant mitigation of the release of radioactive materials to the environment (Section II.B.2.g);
- for normal operations the liquid radwaste storage and treatment systems were marginal, at best, due to the lack of a *de minimis* level below which liquid radwaste can be discharged without treatment, and insufficient processing capacity (Section II.B.2.b);
- the radwaste liquid storage capacity was inadequate to cope with the emergency operations (Section II.B.2.c);
- the NRC review of TMI-2 design did not consider the impact of TMI-1 in certain areas such as ventilation systems (Section II.B.2.a);
- leakage of radwaste system components, particularly in the makeup and purification system, which contained small amounts of radioactive material during normal operation, led to the most significant releases of radioactive material after core damage occurred (Section II.B.2.d);
- due to lack of maintenance on the waste gas system, leaks existed, particularly in compressor A, which led to additional releases of radioactive material after core damage (Section II.B.2.b);
- high pressure damaged portions of the vent gas system, which resulted in a gaseous release pathway (Section II.B.2.e);
- the gaseous radwaste system design included "relief to atmosphere," which provided a path to the environment for untreated gas (Section II.B.2.e);
- Met Ed initiated modifications after the accident that helped to mitigate the releases; these modifications included the supplementary auxiliary building filter systems, and vent lines from the waste gas decay tanks back to containment (Section II.B.2.g);
- the quantity of radioactive material thus far released in liquid effluents as a result of the accident is not significant (Section II.B.2.c);
- the quantity of radioactive material released in gaseous effluents due to the accident consisted of 15 Ci of ^{131}I and 2.4 million Ci of noble gases (Sec. II.B.2.f);
- the carbon installed in the auxiliary and fuel handling building exhaust systems was in a degraded condition on March 28, and contributed to the radioiodine releases. The design and testing of the filters did not allow the condition of the filters or leakage around the filters to be determined. If carbon had been in place at the time of the accident that satisfied the technical specifications, radioiodine releases would have been lower by a factor of 5 (Section II.B.2.a);
- the auxiliary and fuel handling building exhaust filter systems installed at the time of the accident provided a decontamination factor of 9.5 (equivalent to an 89.5% efficiency) for all species of radioiodine (Section II.B.2.g);

TABLE II-9. Inventories of radioactive materials in auxiliary building tanks as of June 15, 1979 (μCi/cc)

Nuclide	Reactor Coolant Bleed Holdup Tank			Neutralizer Tank		Miscellaneous Waste Holdup Tank, Auxiliary Building Sump and Sump Tank, Miscellaneous Sumps	Evaporator Condensate Tanks, Contaminated Drain Tank
	A	B	C	A	B		
H-3	0.23	0.27	0.29	*	*	0.98	*
I-131	1.9	2.8	3.0	0.15	0.18	1.0	0.1
Cs-134	6.5	7.6	7.7	0.56	0.72	2.4	0.1
Cs-136	0.28	0.29	0.28	0.02	0.02	0.08	0.1
Cs-137	28	35	35	2.5	3.3	10.1	0.1
Ba-140	0.09	0.3	0.29	0.01	0.3	0.8	0.1

*No analysis performed

- HEPA filters installed in the ventilation exhaust systems removed essentially all of the particulates generated (Section II.B.2.g);
- for the two filtration systems operating at the time of the accident (auxiliary and fuel handling buildings) each had identical safety grade cleanup components, rendering the safety grade versus nonsafety grade designations of these systems meaningless (Section II.B.2.g);
- inplace testing and laboratory testing of carbon samples were inadequate to analyze the effectiveness of the ventilation exhaust filters during the first week of the accident (Section II.B.2.g); and
- replacement carbon in the various filter systems was impregnated with an amine (triethylenediamine), and this carbon was effective in reducing radioiodine releases (Section II.B.2.g).

Recommendations

Unless otherwise specified herein (Section II.B), the recommendation is applicable to the NRC and applicant(s)/licensee(s).

We recommend that:

- the design bases for radwaste and other related systems, such as the makeup and purification system, be reexamined to determine appropriate design criteria for the expected levels of activity and volumes that will be generated in both normal operation and accident situations;
- review of radwaste systems should include all related systems, such as the industrial waste treatment system, to ensure that potential releases (whether within the plant or to the environment) are treated;
- a *de minimis* level be established for low-level liquid radwaste, and any liquid at a nuclear power station containing less than this *de minimis* level of radioactive material be allowed to be released untreated;
- radwaste system components (with the potential for containing primary coolant or waste gas products) be periodically tested for leaks and any leaks exceeding a minimum acceptance level be repaired;
- consideration be given to locating systems such as the makeup and purification system in an isolating building (such as the reactor building);
- consideration be given to the installation of tie-lines from components outside containment having the potential to contain significant activity (e.g., makeup tank, waste gas decay tanks, reactor coolant bleed holdup tanks) back to containment for use during an accident;
- methods be developed for inplace testing of ventilation systems, such as continuous upstream/downstream sampling or inplace radioactive tracer testing, to ascertain overall filter system performance when needed;
- procedures be developed for the evaluation of spent carbons exposed to accident conditions and to consider the effect of high concentrations of noble gas and iodine;
- specific filtration systems be designated and designed for use only after an accident; separate filter systems be provided for normal operation;
- dampers around filter systems be eliminated or improved to minimize leakage;
- to increase the radioiodine removal capabilities, consideration be given to coimpregnating carbons with an amine such as triethylenediamine, and to use of deeper carbon beds.

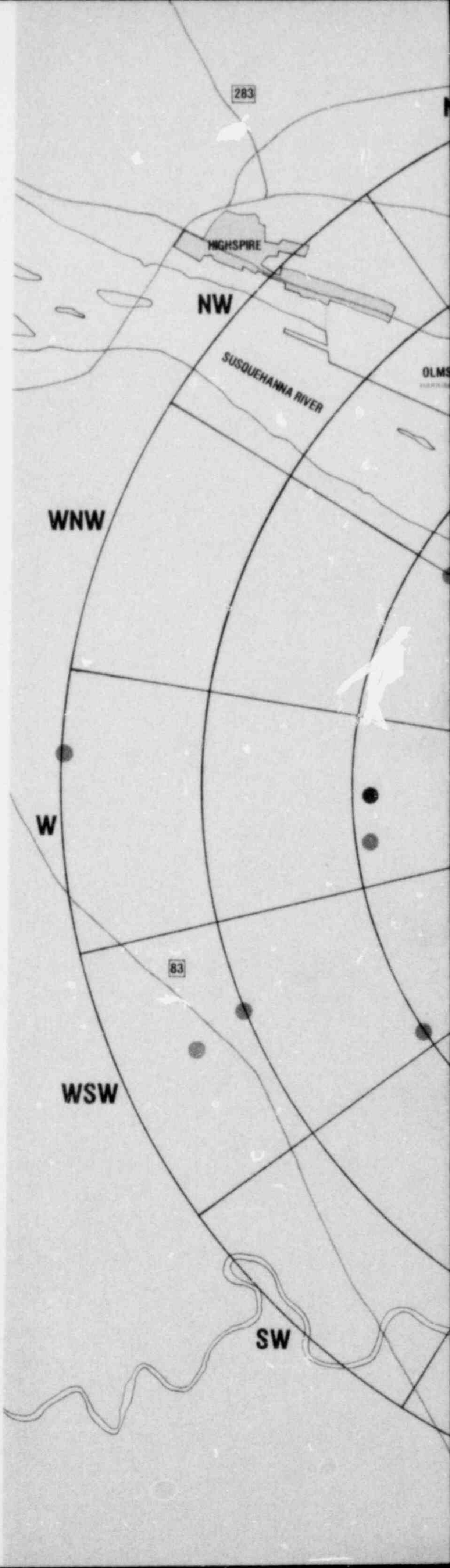
3. ENVIRONMENTAL MONITORING

The purpose of radiological monitoring at nuclear powerplants is to protect workers and the public by ensuring that exposure of workers on site to radiation and releases of radioactive materials off site are kept within the limitations of applicable Federal regulations and as low as reasonably achievable (ALARA).

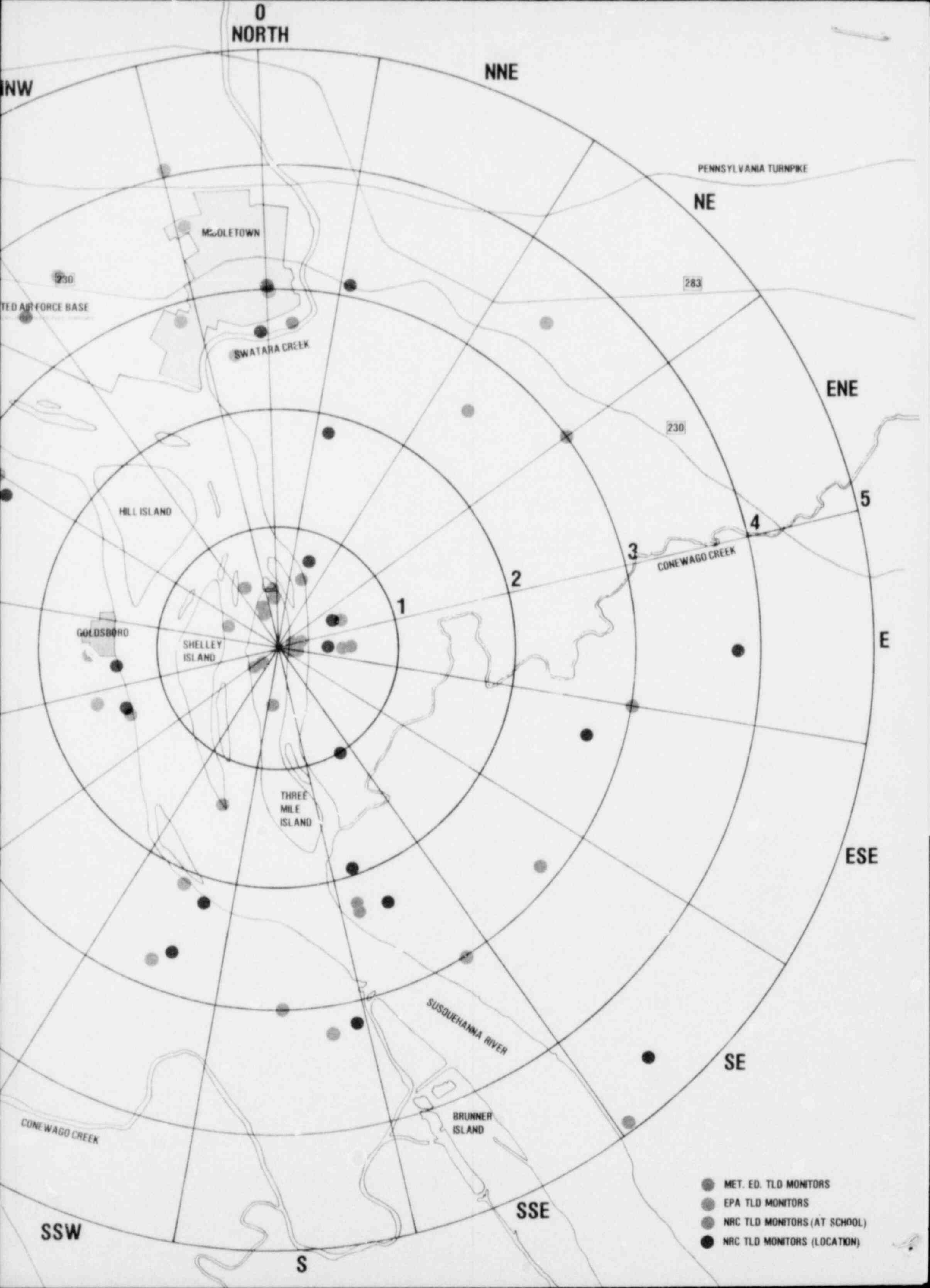
Radiation monitoring of onsite personnel is accomplished by means of dosimeters, such as thermoluminescent dosimeters (TLDs) and self-reading pocket ion chambers (pocket chambers). Area monitoring is performed using fixed, mobile, and portable radiation detection instruments. Concentrations of airborne radioactive materials are monitored using fixed and portable air sampling devices. Evaluation of internal contamination of personnel is accomplished by means of bioassays (urinalyses) and whole-body counting (WBC).

Onsite and offsite environmental radiation monitoring also uses TLDs. In addition, a program is in force to sample air, water, milk, vegetation, fish, and river sediments to assess the amount of radioactive materials deposited off site.

At Three Mile Island Station, it was necessary to increase onsite and offsite monitoring as a result of the accident. The large number of people on site, together with the increased chance of high radiation exposure after the accident, required greater emphasis on radiation safety, including additional dosimetry. Onsite monitoring is discussed in Section II.B.5. The prospect or fear of substantial offsite releases led to increased environmental monitoring, which is discussed below.



COLOR PLATE I. LOCATION OF DOSIMETRY SITES WITHIN A 5 MILE RADIUS OF TMI NUCLEAR STATION.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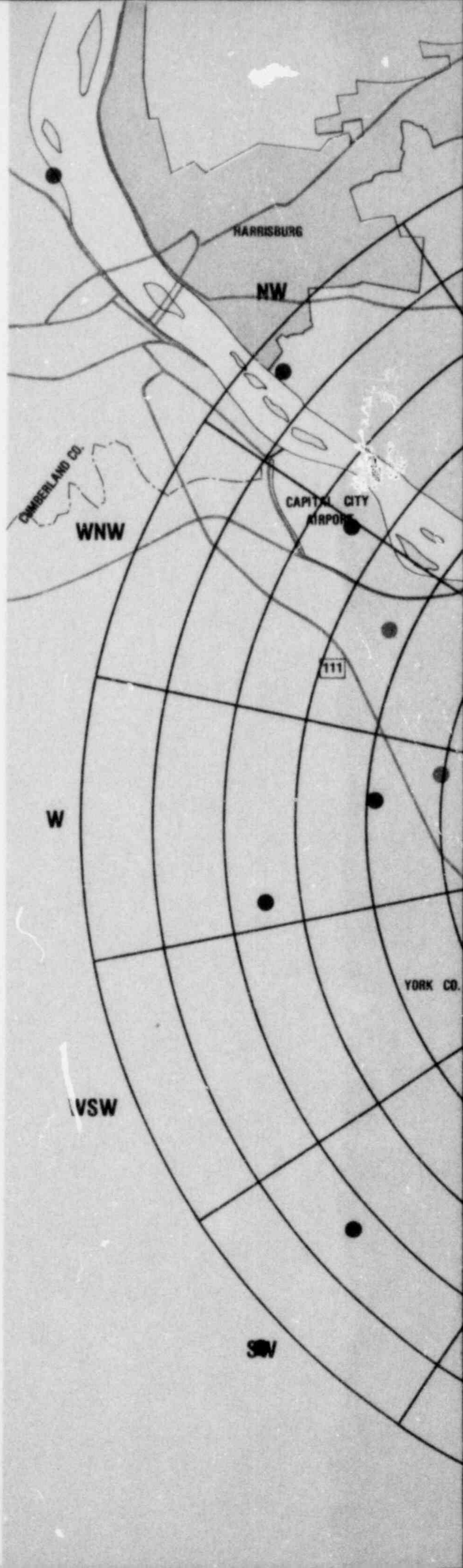
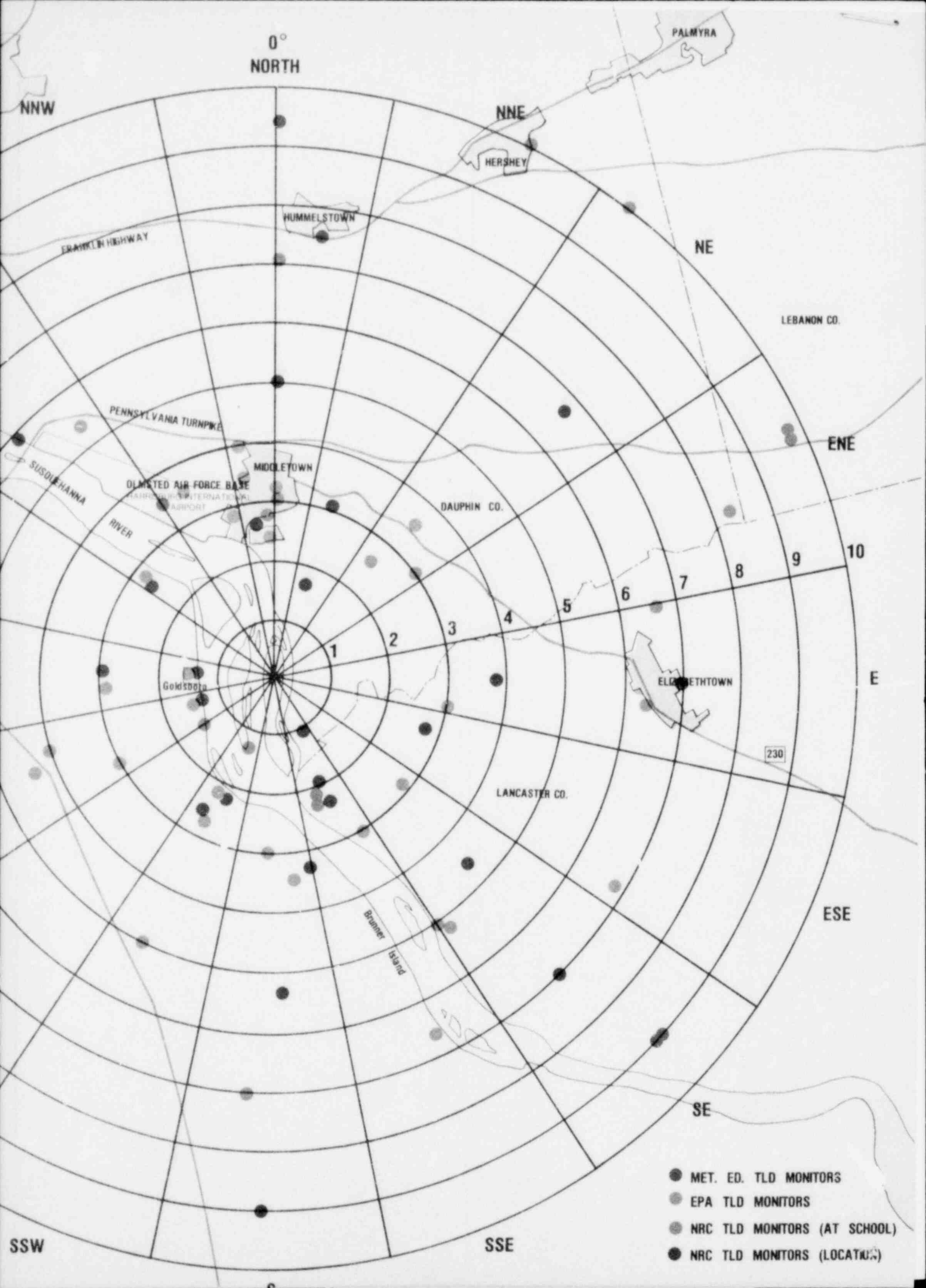
HILL ISLAND
CONEWAGO CREEK
GOLDSBORO
SHELLEY ISLAND
THREE MILE ISLAND
SUSQUEHANNA RIVER
BRUNNER ISLAND
MET. ED. TLD MONITORS
EPA TLD MONITORS
NRC TLD MONITORS (AT SCHOOL)
NRC TLD MONITORS (LOCATION)



COLOR PLATE II. MAP OF TMI 2 AREA



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MIDDLETOWN
OLMSTED AIR FORCE BASE
HARRISBURG INTERNATIONAL AIRPORT
DAUPHIN CO.
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NRC TLD MONITORS (AT SCHOOL)
NRC TLD MONITORS (LOCATION)

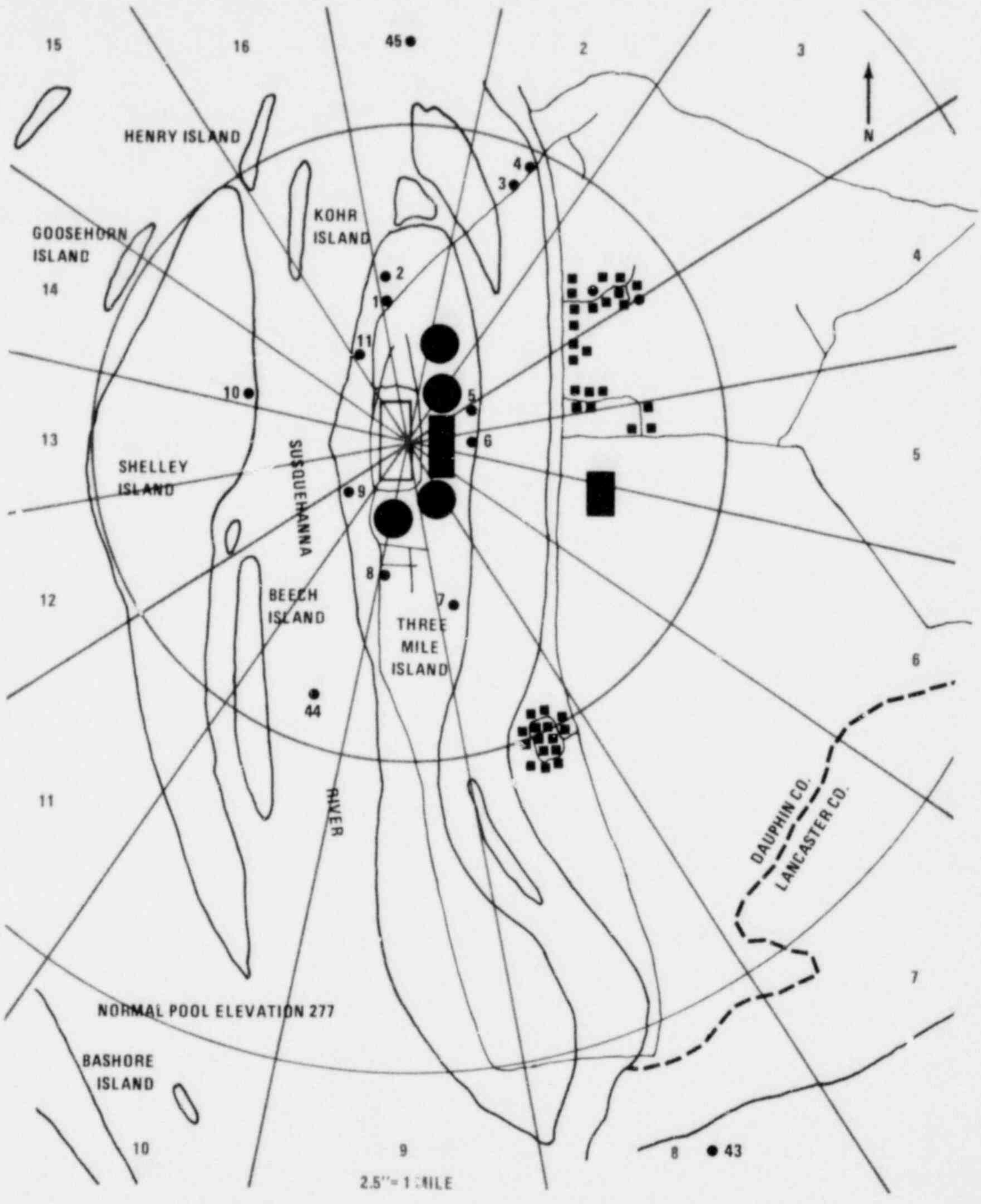


FIGURE II-14. Location of Onsite TLDs

technicians and later (primarily March 29 and 30), three individuals. Three teams were dispatched initially and up to six teams made surveys on March 29 and 30.[75]

Direct radiation measurements were performed with portable radiation survey instruments by the land-based and helicopter-based teams. Instruments used were generally the PIC-6A (an ion chamber-type instrument having a range from 0–1000 R/h), the RO-2 (an ion chamber-type instrument having a range from 0–5000 mR/h) and the E-520 (a GM-type instrument having a range from 0–2 R/h).[75]

These teams also collected short term air samples (particulate and iodine) for field determination of radioiodine concentrations (primarily on March 28).

a. Offsite Radiological Environmental Monitoring Program (Preaccident)

A Radiological Environmental Monitoring Program (REMP) for Three Mile Island Station has been conducted for Met Ed since June 1969 and is described in Appendix II.3. At the time of the accident, Teledyne Isotopes Corporation was responsible for the analytical portion of the REMP.[54]

The REMP consists of gaseous and liquid effluent monitoring, sampling of flora and fauna, soil, vegetation, and milk in the environs of the nuclear station to detect whether there are any plant effluents that might contribute to the exposure of the public. This program also is designed to detect if any long term buildup of radioactive material is occurring.[55]

In addition to the REMP, environmental radiological monitoring is performed at TMI using environmental TLDs.[56] The location of the onsite TLDs is described in Table II-10 and shown in Figure II-14. Met Ed also had an onsite monitoring program using personnel TLDs.

b. Augmented Radiological Monitoring Program (Postaccident, March 28 to April 15, 1979)

As part of the immediate response to the accident, radiological monitoring at and around TMI was augmented by Met Ed, other utilities, consultants and contractors, and Federal, State, and local agencies. From these sources came additional personnel, technical expertise, analytical laboratory capability, radiation survey instrumentation, environmental monitoring (including extensive offsite ground and airborne radiation surveys, and sampling of air, terrestrial, and water media), and an additional method to predict plume behavior. For example, immediate radiological monitoring expertise was provided by Porter-Gertz, Consultants.[57] Release and plume predictions were provided by Pickard, Lowe, and Garrick, Inc., meteorological consultants.[58] Dosimetry expertise, management, and personnel were provided by the Electric Boat Division of General Dynamics Corporation,[59] by Pennsylvania Power & Light Company,[60] and by the Naval Reactors Division of the Department of Energy (DOE).[61] Expertise in the maintenance and control of the varied portable radiation survey instruments that were used was provided by Electric Boat Division,[62] Georgia Power and Light Company,[63] and Naval Reactors.[64] Additional assistance provided by other sources, particularly from Government agencies, is discussed elsewhere.[65-74]

Metropolitan Edison— As an initial response to the accident, Met Ed performed offsite surveys around Three Mile Island. Teams were dispatched in the downwind direction to perform surveys at points that were inside the expected extent of the plume. Teams consisted initially of two radiation chemistry

TABLE II-10. Onsite TLD locations for operational radiological environmental monitoring program (REMP)[56]

Station Designation	Map Number (Fig. II-14)	Location
1S2	2	0.4 mile N of site, North Weather Station
2S2	3	0.7 mile NNE of site on light pole in middle of North Bridge
4S2	5	0.3 mile ENE of site on top of dike, East Fence
5S2	6	0.2 mile E of site on top of dike, East Fence
9S2	8	0.4 mile S of site at South Beach
11S1	9	0.1 mile SW of site west of Mechanical Draft Tower on dike
14S1	10	0.4 mile WNW of site at Shelly's Island picnic area
16S1	11	0.2 mile NNW of site at gate on fence on west side

These samples were later counted with a Ge(Li) system based in a mobile laboratory at the site.[75]

On March 29, the REMP was augmented by expanding the number of sampling locations and frequency.[76] Table II-11 describes the augmented REMP, also termed Emergency REMP.

Commonwealth of Pennsylvania— On the advice of the Pennsylvania Bureau of Radiation Protection (BRP), the Pennsylvania Agriculture Department sampled farm milkings on the evening of March 28 and the morning of March 29. This sampling program continued through mid-June.[77] The BRP performed ground surveys in the offsite area and collected and analyzed data.[77] The Bureau of Water Quality Management and BRP joined with the EPA to provide a water sampling and analysis program.[78,79] BRP placed portable air samplers around the plant area and at the observation center and analyzed the results.[78] BRP also placed liquid effluent monitors near or on the station discharges.[78]

Nuclear Regulatory Commission (NRC)—While in transit to Three Mile Island on March 28, the NRC Region I teams conducted limited radiation surveys. The results of these surveys were reported to Met Ed at the observation center.[80] In addition to these initial surveys, NRC teams performed ground monitoring surveys on the east side of the Susquehanna River for several weeks after March 28. NRC deployed TLDs at 37 offsite locations on March 31 and at an additional 10 locations on April 5. These TLDs were placed and read by RMC.[81] The locations of the NRC TLDs are listed in Table II-12 and shown in Color Plates I and II. NRC placed portable air samplers around the plant area and observation center and analyzed the results.[78] NRC also placed liquid effluent monitors near or on the station discharges.[79]

Department of Health, Education, and Welfare (HEW), Bureau of Radiological Health (BRH)— In response to the accident, BRH deployed TLDs around the site starting on the evening of March 31, 1979. The TLDs deployed, type TLD 100, had a minimum sensitivity of 10 to 20 mrem. A total of 173 dosimeter sites (237 dosimeter packages) had been set up by Monday afternoon, April 2, 1979[82] and are shown in Figure II-15.

The BRH dosimeter sites were distributed over a 20-mile radius (about 1200 square miles) centered at TMI-2. Within the 0- to 10-mile radius, the area was divided into 2 by 2-mile grids. The individual

TABLE II-11. Augmented or Emergency REMP[76]

Media	No. of Indicator Locations	No. of Background Locations	Sampling Frequency	Analyses[1]
Air particulates	5	3	Every 3 days[2]	Gross beta, gamma spectra
Air iodine	5	3	Every 3 days[2]	Radioiodine
Surface/drinking water	5[3]	2	Daily[4]	Gross beta, radioiodine
Effluent water	1	0	Daily[4]	Tritium, gamma spectra
Precipitation (rain water)	2	2	As available[5]	Gamma spectra
Fishes	1	1	Weekly	Gamma spectra, strontium
Aquatic plants	2	1	Weekly (if available)	Gamma spectra
Aquatic sediment	2	1	Weekly	Gamma spectra, strontium
Milk	4[6]	1	Daily	Radioiodine, gamma spectra
Vegetation	4	1	Monthly	Radioiodine, gamma spectra
Soil	4	1	Monthly	Gamma spectra
Misc. foodstuffs[7]	1	1	As available	Gamma spectra
TLD	15	5	Every 3 days[2]	Dose rate

[1] The listed analyses are performed on each sample and are in addition to those performed in the operational REMP.

[2] Sampling periods were from 3/29-3/31, 3/31-4/3, and every three days thereafter until 4/24/79. As of 4/24/79, samples are collected weekly.

[3] An indicator location was added on 4/22/79.

[4] Sampling was done on 3/29, 3/31, and daily thereafter.

[5] Precipitation was collected on 3/31, 4/5, and 4/27.

[6] Due to its use by newborn goats, milk is not always available from a goat farm.

[7] Includes poultry, beef, eggs, pork, and game, if available.

TABLE II-12. NRC TLD locations[81]

Station	Distance (miles)	Direction (degrees)	Sector	Description
N-1a	2.4	356	N	School (added 4/5/79)
N-1	2.6	358	N	Middletown
N-1c	3.0	0	N	School (added 4/5/79)
N-1e	3.5	349	N	School (added 4/5/79)
N-1f	4.0	351	N	School (added 4/5/79)
N-2	5.1	0	N	Clifton
N-3	7.4	6	N	Hummelstown
N-4	9.3	0	N	Union Deposit
N-5	12.6	3	N	–
NE-1	0.8	25	NNE	North Gate
NE-2	1.8	19	NNE	Geyers Church
NE-3	3.1	17	NNE	Township School
NE-3a	3.6	44	NE	School (added 4/5/79)
NE-4	6.7	47	NE	–
E-1	0.5	61	ENE	1200′ N of E-1a
E-5 (E-1a)	0.4	90	E	Residence
E-3	3.9	94	E	Newville
E-4	7.0	94	E	Elizabethtown
E-2	2.7	110	ESE	Unpopulated area
SE-4	4.6	137	SE	Highway 441
SE-4a	5.0	146	SE	School (added 4/5/79)
SE-5	7.0	135	SE	Bainbridge
SE-1	1.0	151	SSE	Unnamed community on Highway 441
SE-2	1.9	162	SSE	Falmouth
SE-3	2.3	160	SSE	Falmouth
S-1	3.2	169	S	York Haven
S-1a	3.35	173	S	School (added 4/5/79)
S-2	5.3	178	S	Conewago Hts
S-3	9.0	181	S	Emigsville
S-4	12.0	184	S	Woodland View

TABLE II-12. NRC TLD locations—Continued

Station	Distance (miles)	Direction (degrees)	Sector	Description
SW-1	2.2	200	SSW	Bashore Island
SW-2	2.6	203	SSW	Pleasant Grove
SW-3	8.3	225	SW	Zions View
SW-4	10.4	225	SW	Eastmont
W-2	1.3	252	WSW	Goldsboro
W-3a	4.4	247	WSW	School (added 4/5/79)
W-1	1.3	263	W	Goldsboro
W-3	2.9	270	W	Unnamed community
W-4	5.9	272	W	Lewisberry
W-5	7.4	262	W	Lewisberry
NW-1	2.6	303	WNW	Harrisburg Airport
NW-3	7.4	297	WNW	New Cumberland
NW-2	5.9	310	NW	Highspire
NW-4	9.6	306	NW	Harrisburg
NW-5	13.8	312	NW	Harrisburg
N-1b	2.75	346	NNW	School (added 4/5/79)
N-1d	3.5	333	NNW	School (added 4/5/79)

grid sectors were weighted by population, and sites were identified in the field on the following basis:

High Population Density—4 dosimeter sites
Medium Population Density—2 dosimeter sites
Low Population Density—1 dosimeter site

When possible, two dosimeter packages were placed at each site, one outside and one inside a building. In the 10- and 20-mile ring, only external sites were used.

The dosimeters were left in place until a small sample (19) was collected and replaced on April 10 for a preliminary evaluation. All dosimeters were collected on April 17 and 18 and replaced.[82]

HEW also carried out a limited bioassay program, performing urine analyses on 33 residents living near the plant. The samples were collected over 5 days (April 4 through 8).[83]

HEW collected milk, food, and water samples in the area around Three Mile Island to a distance of 30 miles. Raw milk was sampled from 29 locations, and included samples from both cows and goats.[83] The specific information as to animal location, sample type, supplier, feed, herd site, and recipient dairy is indicated in Table II-13. The source, azimuth, and distance from TMI for each food and milk sampling location are listed in Table II-14. HEW collected water samples from various points on the Susquehanna River, from taps in Harrisburg, Columbia, Harrisburg Airport, Port Deposit Water Treatment Facility, Conestoga, Middletown, and various locations in Maryland.[84]

Environmental Protection Agency (EPA)—EPA deployed its major response efforts from its Las Vegas, Nev. laboratory. EPA personnel arrived in the TMI area on March 31 and began an offsite environmental sampling effort. EPA also brought laboratory analysis capability and set up an analytical facility in Harrisburg. EPA was requested to coordinate all offsite Federal environmental monitoring for the long term efforts on April 13, 1979.[85]

From April 1 to April 3, EPA set up an offsite air sampling network. Thirty-one air sampling stations

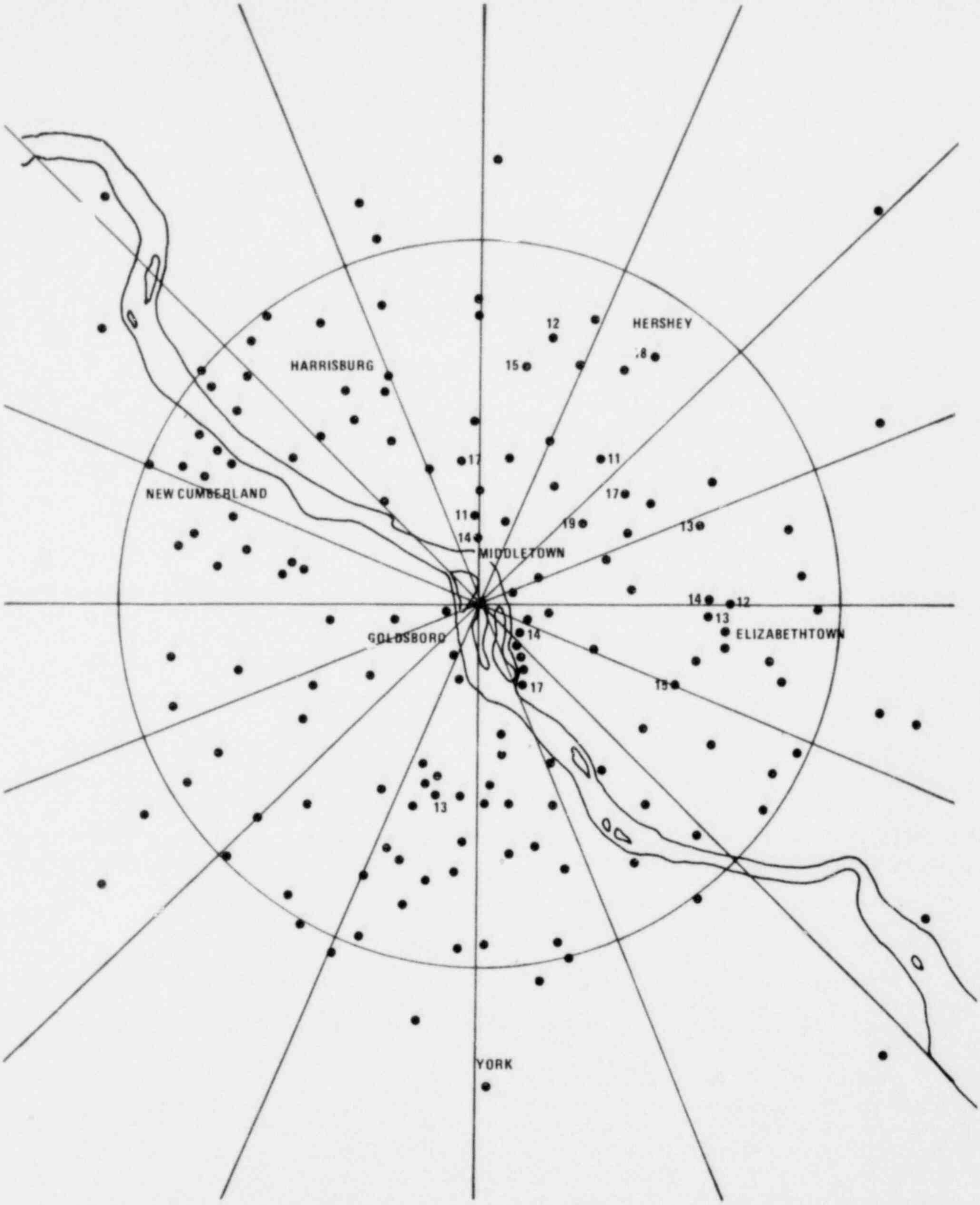


FIGURE II-15. Location of HEW Monitoring Sites ●
Detectable exposures are shown (mR).
Period of exposure: 3/31/79 to 4/18/79.

TABLE II-13. Raw milk sample program for HEW[86]

Milk Supplier	Product	Feeds	Animal Location	Dairy to Which Sold	Herd Size
Christian Becker	Raw Milk	Stored	Inside	Hershey Foods Hershey, PA	40
H. Risser Meadow Vista)	Raw Milk	Stored	Inside	Mt. Joy Farmer Corporation Mt. Joy, PA	200
Ken Glatfeller	Raw Milk	Stored	Inside & Dry Lot	Rutter Bros York, PA	125
J.R. Alwine	Raw Milk	Stored	Inside & Dry Lot	Mt. Joy Corp. Mt. Joy, PA	102
Jim Williams	Raw Milk	Stored	Inside & Grazed	Interstate Coop. S. Hampton, PA	108
Jeremiah Fisher	Raw Milk	Stored	Inside & Dry Lot	Interstate Coop. S. Hampton, PA	42
Clarence Lytle	Raw Milk	Stored	Inside & Dry Lot	Harrisburg Dairy Harrisburg, PA	102
Beshore Farms	Raw Milk	Stored	On Dry Lot	Rutter Bros. York, PA	82
Masonic Homes	Raw Milk	Stored	Under Roof	Harrisburg Dairy Harrisburg, PA	115
Jay Swope	Raw Milk	Stored	Under Roof	Lehigh Valley Allentown, PA	25
Leroy Hertzler	Raw Milk	Stored	Inside & Dry Lot	Rutter Bros. York, PA	27
Avalong	Raw Milk	Stored	In & Out	Own Processor	100
Bruce Zell	Raw Milk	Stored	Inside	Hershey Foods Hershey, PA	80
Myers Farms	Raw Milk	Stored	On Property	Hershey Foods Hershey, PA	35
Sunnyhill Farms	Raw Milk	Stored	Inside & Dry Lot	Own Processor	160
Timothy Tyson	Raw Milk	Stored	Under Roof	Mt. Joy Corp. Mt. Joy, PA Lehigh Valleys Allentown, PA	54
Paul Nolt	Raw Milk	Stored	Under Roof	Mt. Joy Corp. Mt. Joy, PA Lehigh Valleys Allentown, PA	39
H.E. Heindel	Raw Milk	Stored	In & Out	Maryland Coop.	138
Rutter Bros.	Raw Milk	Stored	Inside	Own Processor	60
Ashcombe Farm Dairy	Raw Milk	Stored	Dry Lot	Own Processor	200

TABLE II-13. Raw milk sample program for HEW—Continued

Milk Supplier	Product	Feeds	Animal Location	Dairy to Which Sold	Herd Size
Alton Hower	Raw Goat's Milk	Graze	Outside	Own Processor	3
Lloyd Sarver	Raw Goat's Milk	Stored	Inside	Own Processor	1
Dale Barshinger	Raw Milk	Stored	Pasture 3 hrs/day	Maryland Coop.	38
Doil L. Zirkle	Raw Milk	Stored	Pasture 3 hrs/day	Interstate Coop.	43
Evergreen Valley Farm	Raw Milk	Stored	Inside Dairy	Hershey Food Hershey, PA	42
Lester Hawthorne	Raw Milk	Stored	Inside	Penn Dairies Lancaster, PA	150
Menno Gruber	Raw Milk	Stored	Inside	Hershey Foods Hershey, PA	60
Bruce Taylor	Raw Milk	Stored	Inside	Rutter Bros. York, PA	50
Joseph Conley	Raw Milk	—	—	—	30

were established, with 12 stations located at a distance of 3 miles from the plant, at 30° spacing along the arc; 10 stations at 6 to 7 miles, located between the 3-mile stations; and 9 stations in populated locations more than 7 miles away at Bellaire, Manchester, Carlisle, Hummelstown, Campbelltown, York, Hershey, Lebanon, and Lancaster.[88]

Each station contained an air sampler of approximately 10-cfm capacity (400 m^3/day) with a glass fiber prefilter for particulate collection and a charcoal cartridge for radioiodine collection. Samples were changed on a daily basis and counted using a Ge(Li) detector.[89] The location of each air sampling station is shown in Table II-15.

At each EPA monitoring station, calcium fluoride TLDs consisting of three badges, each containing two chips, were placed. In addition, 50 people at these locations wore badges on a voluntary basis.[90]

Gamma exposure rate recorders were located at each air sampling station and three additional locations (Stations 031, 032, and 033). The recorder/monitors were deployed from March 31 to April 4 and were operated throughout the intensive phase. They contain a pressurized gas proportional detector with output to a strip chart recorder enclosed in an aluminum case. The strip chart from each recorder was collected daily.[91]

EPA conducted water sampling at locations on the Susquehanna River and in Chesapeake Bay. Drinking water from 21 surface supplies was also sampled. The drinking and surface water sampling effort was reduced on April 6 to include only major public drinking water sources on the Susquehanna River (Lancaster, Columbia, and Wrightsville). On April 8, the Wrightsville and Columbia stations were dropped and another station was set up on Brunner Island. Composite samples (24-hour) were collected daily from these sites.[92] Daily grab samples were collected on the liquid effluent discharges. A continuous ^{133}I monitor was also installed.[92]

EPA initiated milk sampling in the offsite area on April 5. A total of nine dairy farms were included in this effort.[94] Their locations are indicated in Table II-16.

Three special stations were established for radioactive noble gas sampling at stations 001, 006, and 014 (Table II-15). Air samples of at least 2/3 m^3 were collected over a 2- to 3-day period.[95]

TABLE II-14. Source locations for HEW food and milk sampling program[87]

Distance (miles)	Direction (degrees)	Name	Location
20	340	All Lebanon Bakery	Lebanon
16	315	Alton Hower	Enola
10	34	Arndt's Ice Cream	Hershey
12	225	Ashcombe Dairies	Dover
15	270	Ashcombe Farm Dairy	Mechanicsburg
12.5	166	Avalong Farms, Inc.	York
22	159	Bakers Homemade Bread	Red Lion
10	178	Bartons Bakery	Mt. Wolf
5	176	Beecher, Katherine Candies	Manchester
7	292	Bedshore Farms	New Cumberland
18	88	Bickel's Potato Chip Co., Inc.	Manheim
10	313	Brookwood Farms	Harrisburg
20.5	52	Brouse's Pastry Shop	Lebanon
7	182	Bruce Taylor	Manchester
4.5	4	Bruce Zell	Hummelstown
20	90	Bucker, Raymond Farm	Lititz
12	128	Byers Pastries	Marietta
1	125	Christian Becker	Elizabethtown
3	16	Clarence Lytle	Middletown
13	175	Cloverland	York
12	182	D. F. Stauffer Biscuit Company	York
6	195	Dale Barshinger Dy.	York
20	260	Dillsburg Grain & Milling	Dillsburg
14	50	Dol-Mar	Annville
5	207	Doll L. Zirkle Dy.	Manchester
20.5	51	Dunkin Donuts	Lebanon

TABLE II-14. Source locations for HEW food and milk sampling program—Continued

Distance (miles)	Direction (degrees)	Name	Location
8	10	Dutchland Farms, Inc.	Rheems
22	35	Eastern Milk	Jonestown
16	285	Eastern Milk	Mechanicsburg
16	103	Elmtree Acres	Mt. Joy
2.5	121	Evergreen Valley Farm	Elizabethtown
22	180	Farmer Boy	Glen Rock
15	41	Gingrich's Bakery	Campbelltown
22.5	87	Graybill's	Lititz
14	182	Green's Dairy, Inc.	York
9	19	H. B. Reese Candy Co.	Hershey
15	155	H. E. Heindel	York
4	135	H. Risser	Bainbridge
11	318	Harrisburg Dairies, Inc.	Harrisburg
3.1	335	Harrisburg Int'l Airport	Middletown
11.2	312	Harrisburg R. P.	Harrisburg
10	24	Hershey Chocolate Company	Hershey
5	163	Hilshire, Claire	Elizabethtown
12	110	I.R. Musser Poultry Farm, Inc.	Mt. Joy
2	81	J. R. Alwine	Middletown
12.5	35	Ja-Mar	Palmyra
3.5	132	Jay Swope	Elizabethtown
5	284	Jeremiah Fisher	Etters
11	50	Johanna	Palmyra
2.8	275	Joseph Conley	Etters
22	117	Kendig	Millersville
6	166	Ken Glatfeller	Mt. Wolf

TABLE II-14. Source locations for HEW food and milk sampling program—Continued

Distance (miles)	Direction (degrees)	Name	Location
22	165	Knaubs Cake and Deli House	Dallastown
1	33	Kraft, Inc.	Palmyra
7.5	163	Leroy Hertzler	Mt. Wolf
5	125	Lester Hawthorne	Elizabethtown
2	210	Lloyd Sarver	York Haven
7	97	Longenecker Hatchery, Inc.	Elizabethtown
3	355	Longenecker's Meats, Inc.	Middletown
22	117	Manorview	Millersville
6	98	Masonic Homes	Elizabethtown
10	25	Mazzoli's Ice Cream	Hershey
12	121	Mellinger's Poultry Farm	Mt. Joy
5	129	Menno Gruber	Bainbridge
20	165	Midway Super Thrift Market	Dallastown
30	335	Miller Bros.	Millersburg
18	168	Mrs. Smith's Pie Co.	York
11	87	Mt. Joy Corporation	Mt. Joy
10	288	Myers Farms	New Cumberland
10	178	Naylors Candies, Inc.	Mt. Wolf
17	281	Oak Grove Poultry Farm	Mechanicsburg
11	107	Paul Nolt	Mt. Joy
11	180	Peerless Farm Products	York
23.5	109	Penn Dairies, Inc.	Lancaster
12.9	181	Penn Dairies, Inc.	York
10.5	320	Penna Dutch Megs. Inc.	Harrisburg

TABLE II-14. Source locations for HEW food and milk sampling program—Continued

Distance (miles)	Direction (degrees)	Name	Location
13	175	Perrydell Farm	York
21	125	Queen Dairy Foods	Conestoga
20	335	R Own Dairy	Halifax
22.5	90	R.W. Sauder	Lititz
11	318	Reservoir	Harrisburg
5	355	Rose Enterprises, Inc.	Middletown
20	54	Royers Cake Box	Lebanon
12.5	184	Rutter Bros.	York
10	314	Sams Ice Cream, Inc.	Harrisburg
20	51	San Giorgio Macaroni, Inc.	Lebanon
15	286	Schenks Pastries	Mechanicsburg
22.5	47	Showerdale	Lebanon
6	89	Simon Candy Company	Elizabethtown
11	50	Smith's Modern Dairy	Palmyra
12	214	Smitties Soft Pretzel	Dover
12	105	Spanglers Flour	Mt. Joy
18	330	Speeces Dairy	Dauphin
20	193	Stump Acres	York
9	341	Sunnyhill Farms	Harrisburg
4.6	157	Susquehanna River	York Haven
11.6	129	Susquehanna River	Marietta
13.9	133	Susquehanna River	Wrightsville
2.4	169	Susquehanna River	Falmouth
22	156	Tastysnack, Inc.	Windsor
10.5	63	Timothy Tyson	Palmyra
5.5	344	Tom Williams	Middletown
10	315	Town & Country Pastry Shop	Harrisburg

TABLE II-14. Source locations for HEW food and milk sampling program—Continued

Distance (miles)	Direction (degrees)	Name	Location
6	100	Troutmans Dairy	Elizabethtown
21	125	Turkey Hill Dairy	Conestoga
14.1	125	Turkey Hill Mini Market	Columbia
3	355	Universal Flexible Packaging, Inc.	Unk
8	3	Verdelli Farms, Inc.	Hummelstown
10	314	Visaggios Bakery	Harrisburg
17	259	Wayne Feed Supply Storage	Dillsburg
14	50	Wengerts Dairy, Inc.	Lebanon

TABLE II-15. EPA air sampling and monitoring locations (intensive phase)[93]

Station	AZ	Distance (miles)	Location
001	290	6.2	Frogtown, Pa.—Robert Bean Gulf Station
002	320	5.2	*Highspire, Pa.—Highspire Fire Station No. 1
003	325	3.5	Meade Heights, Pa.—Harrisburg Intl Airport
004	350	3.0	*Middletown, Pa.—Elwood's Sunoco Station
005	040	2.6	Royaltown, Pa.—Londonderry Township Bldg.
006	055	3.0	Royaltown, Pa.—Blandine Hershberger residence
007	080	6.6	Elizabethtown, Pa.—Koser's Fruit Market
008	070	8.2	*Bellaire, Pa.—Robert Risser residence
009	100	3.0	Newville, Pa.—Brooks Farm, Earl Nissley residence
010	095	6.3	*Elizabethtown, Pa.—Arco Service Station
011	130	2.9	Falmouth, Pa.—Charles Brooks residence
012	120	6.9	Maytown, Pa.—Bassler's Church
013	150	3.0	Falmouth, Pa. - Dick Libhart residence
014	145	5.3	*Bainbridge, Pa.—Bainbridge Fire Company
015	155	6.6	Saginaw, Pa. - United Methodist Church

TABLE II-15. EPA air sampling and monitoring locations (intensive phase)—Continued

Station	AZ	Distance (miles)	Location
016	180	7.0	*Manchester, Pa. - Manchester Fire Department
C17	180	3.0	*York Haven, Pa. - York Haven Fire Station
018	220	2.5	Pleasant Grove, Pa. - George Ziegler residence
019	205	5.0	Strinestown, Pa. - Brenner Mobil Service Station
020	205	2.5	Woodside, Pa. - Zane Reeser residence
021	250	4.0	*Newberrytown, Pa. - Exxon Kwick Station
022	275	5.0	Yocumtown, Pa. - IML Freight Yard
023	265	2.9	Goldsboro, Pa. - Muellar residence
024	275	26	*Carlisle, Pa. - Union Fire Company No. 1
025	360	7	*Hummelstown, Pa. - Keffer's Exxon Service Station
026	025	10	*Hershey, Pa. - Arco Service Station
027	040	10	Campbelltown, Pa. - Gulf Service Station
028	055	20	*Lebanon, Pa. - Goodwill Fire Company
029	110	025	Lancaster, Pa. - Southern Manheim Fire Co.
030	180	13	*York, Pa. - Springetts Fire Co. No. 1
031	270	1.5	*Goldsboro, Pa. - Dusty Miller residence
032	255	1.5	Goldsboro, Pa. - Harold Bare residence
033	205	2.2	Pleasant Grove, Pa. - George Shaffer residence
034	305	2.7	Plainfield, Pa. - Polites residence
035	068	3.5	Royaltown, Pa. - George Hershberger residence

*Sampling located in indicated town. Other sampling stations are located near indicated towns.

EPA analyzed its environmental samples at its temporary laboratory in Harrisburg and its laboratories in Las Vegas, Nev., and Montgomery, Ala.[95]

Department of Energy (DOE)—DOE and its contractors, in accordance with the Interagency Radiological Assistance Plan, conducted a substantial environmental monitoring effort in response to the accident. The Commonwealth of Pennsylvania and the NRC asked formally for DOE assistance on the morning of March 28.[96]

The Radiological Assistance Team (RAT) from Brookhaven National Laboratory (BNL), and the Aerial Measurement System/Nuclear Emergency Search Team (AMS/NEST) from Andrews Air Force Base, Md., arrived by midafternoon on March 28. The RAT assisted the Commonwealth of Pennsylvania by taking vegetation, soil, and air samples; and by making direct radiation measurements off site. The AMS/NEST measured and characterized radiation levels in the plume created by plant discharges. These data were immediately provided to the Commonwealth of Pennsylvania and the NRC to assist in determining the hazard to the public.[97]

A local DOE command post was established on March 28 at the Capital City Airport in New Cumberland, Pa. Various contractors and branches of DOE augmented the radiological monitoring effort.

TABLE II-16. EPA milk sampling locations (intensive phase)[97]

1.	Milton Hershey Dairy #41, Hershey, Pa.
2.	Conewago Farms Dairy, Elizabethtown, Pa.
3.	Aungst Dairy, Rheems, Pa.
4.	A. W. Hoffer, Dairy, Middletown, Pa.
5.	Ruhl Dairy, Middletown, Pa.
6.	David Miller Dairy, Falmouth, Pa.
7.	Elmer Gruder Dairy, Falmouth, Pa.
8.	Leroy Herzler Dairy, Mount Wolf, Pa. 17347
9.	Beshore Farms Dairy, New Cumberland, Pa.

Their contributions are briefly described in Appendix II.4.

DOE monitoring activities included:[98]

- Aerial surveys using helicopters to locate and measure radiation, and to characterize airborne discharges from TMI.
- Meteorological forecasts and predictions of plume trajectories needed for guidance in radiation monitoring and evacuation planning.
- Installation of radio and telephone communications, including coordination with the AT&T Long Lines Command Center, for special NRC, Commonwealth of Pennsylvania, and DOE telephone requirements; staffing the command post, providing rapid telephone and radio communications of data and information between DOE field units, DOE Headquarters, NRC, and the Commonwealth of Pennsylvania.
- Collection of environmental soil, grass, surface water, and air samples taken in the paths of the discharge plumes as well as in the general surrounding area. The sampling procedures used were designed to optimize detection of any radionuclides which might be present.
- Gamma spectrum analysis of environmental samples to detect and identify the radionuclides present.
- Evaluation and analysis of radiation survey data.
- Coordination of shipping and arrangement for radiochemical analyses of reactor coolant and containment air samples.
- In situ measurement and characterization of radiation on the ground and in the air, in the path of airborne discharges from the TMI plant, as well as in the surrounding area.
- Processing, compilation, and analysis of all radiation data in response to a request from T. Gerusky, Director of the Pennsylvania Department of Environmental Resources.
- Documentary and scientific photography.

National Bureau of Standards (NBS)—NBS calibrated portable survey instruments and TLDs used during the accident. Since ^{133}Xe was the predominant radionuclide released, the portable survey instrumentation and TLDs used to monitor releases were not used to detect and measure the energies for which they were calibrated. Most instruments were calibrated with ^{137}Cs gamma rays (662 KeV), although ^{133}Xe emits gamma rays of considerably lower energy (81 KeV). With this wide energy difference, many survey instruments and TLDs overresponded (by factors of from 1.5 to 20).[99]

c. Summary of Results of Portable Survey Instrument and Aircraft Monitoring

A summary of significant survey data collected by Met Ed and other agencies during the period of March 28 to April 5, 1979, during which most of the releases of radioactive materials occurred, is presented in Tables II-17 and II-18. These data were taken directly from copies of survey forms or from reports and logbooks of the various agencies. Many of the data sources lack important information such as instrument type, open or closed shield, exact time, exact location, and the identification of the individual making the survey. In addition, most of the instruments were not calibrated for the radiation emitted by ^{133}Xe. These data, however, were all that were available for decisionmaking purposes at the time of the emergency. Data contained in Tables II-17 and II-18 and discussed below do not include the many measurements that did not detect any radiation above natural background.

March 28, 1979—The first onsite survey indication of a release of radioactive materials occurred at 10:00 a.m. when a 7-mR/h exposure rate was measured at the fence line at the east edge of the site. The first positive indication of an offsite release of radioactive materials was made approximately 0.5 to 1 mile east-northeast of the site at 11:00 a.m. Throughout the day, releases to the environment occurred. Exposure rates continued to vary, generally rising as releases occurred and quickly falling as the radioactive materials were dissipated. The

TABLE II-17. Summary of significant survey data on site March 28 to April 5, 1979

Date	Time	Location – Distance From Site	Elevation (Feet)	Exposure Rate (mR/h)	Type of Radiation	Agency Performing	Reference	Comments
0328	10:00 a.m.	GE-4;*Fence, east	Ground	7		Met Ed	100	First positive onsite reading
0328	3:00 p.m.	GE-2; North gate	Ground	70		Met Ed	101	
0328	5:00 p.m.	GE-10; Fence northwest	Ground	140		Met Ed	100	
0328	11:00 p.m.	GE-10	Ground	365	β & γ	Met Ed	100	Highest ground readings on site
0328	11:00 p.m.	GE-10	Ground	50	γ	Met Ed	100	Highest readings on site
0328	~6:00 p.m.	Over north gate	Helicopter	50		DOE	102	Highest airborne reading on that day
0329	5:00 a.m.	GE-9; Fence, west-northwest	Ground	150	β & γ	Met Ed	103	Highest ground reading on that day
0329	5:00 a.m.	GE-9; Fence, west-northwest	Ground	100	γ	Met Ed	103	Highest ground reading on that day
0329	2:10 p.m.	Above Unit 2 stack	15 over stack	3,000	β & γ	Met Ed	104	Highest reading during the accident
0329	2:10 p.m.	Above Unit 2 stack	15 over stack	400	γ	Met Ed	104	Highest reading during the accident
0330	8:00 a.m.	GE-7; Fence, south	Ground	30	β & γ	Met Ed	105	Venting of makeup tank
0330	8:00 a.m.	GE-7; Fence, south	Ground	9	γ	Met Ed	105	Venting of makeup tank
0330	8:00 a.m.	GE-8; Fence, south-west	Ground	25	β & γ	Met Ed	105	Venting of makeup tank
0330	8:00 a.m.	GE-8; Fence, south-west	Ground	8	γ	Met Ed	105	Venting of makeup tank
0330	8:02 a.m.	Above Unit 2 stack	130 over stack	1200	β & γ	Met Ed	106	Directly in the plume
0330	3:00 p.m.	GE-9; Fence, west-northwest	Ground	90	β & γ	Met Ed	105	
0330	3:00 p.m.	GE-9; Fence, west-northwest	Ground	9	γ	Met Ed	105	

*GE numbers refer to the fixed on island monitoring points.

0331	3:28 a.m.	GE-4; Fence, east	Ground	150	β & γ	Met Ed	107	Highest ground reading on that day
0331	3:28 a.m.	GE-4; Fence, east	Ground	20	γ	Met Ed	107	Highest ground reading on that day
0331	11:15 a.m.	Between GE-3 and GE-4; Fence, east-northeast	Ground	100	β & γ	Met Ed	108	
0331	11:15 a.m.	Between GE-3 and GE-4, Fence, east-northeast	Ground	35	γ	Met Ed	108	
0331	3:51 p.m.	500 kV Substation	Ground	12	β & γ	Met Ed	109	
0331	3:51 p.m.	500 kV Substation	Ground	3	γ	Met Ed	109	
0401	4:28 a.m.	GE-4 and GE-5, Fence, east and southeast	Ground	40	β & γ	Met Ed	110	Highest ground reading on that day
0401	4:28 a.m.	GE-4 and GE-5, Fence, east and southeast	Ground	20	γ	Met Ed	110	Highest ground reading on that day
0402	1:40 p.m.	GE-9, Fence, west-northwest	Ground	15	β & γ	Met Ed	111	Highest ground reading on that day
0402	1:40 p.m.	GE-9, Fence, west-northwest	Ground	7	γ	Met Ed	111	Highest ground reading on that day
0402	2:30 p.m.	Over the Unit 2 screen house	Helicopter	90-240	β & γ	Met Ed	112	Different altitudes. Measurements taken between 2:25 and 2:50 p.m.
0403	12:12 p.m.	GE-5; Fence, southeast	Ground	10	β & γ	Met Ed	113	Highest ground reading on that day
0403	12:12 p.m.	GE-5; Fence, southeast	Ground	1.9	γ	Met Ed	113	Highest ground reading on that day
0404	4:19 a.m.	East Side, between north and south gates	Ground	5.5	β & γ	Met Ed	114	Highest ground reading on that day
0405	1:04 p.m.	Fence, east	Ground	3.5	β & γ	Met Ed	115	Highest ground reading on that day
0405	1:04 p.m.	Fence, east	Ground	0.6	γ	Met Ed	116	Highest ground reading on that day

TABLE II-18. Summary of significant survey data off site March 28 to April 5, 1979

Date	Time	Location—Distance From Site (miles)	Elevation (feet)	Exposure Rate (mR/h)	Type of Radiation	Agency Performing	Reference	Comments
0328	11:00 a.m.	0.5-1 east-north-east	Ground	3		Met Ed	101	First positive offsite reading
0328	3:00 p.m.	0.5-1 east-northeast	Ground	20-50		Met Ed	101	
0328	6:05 p.m.	16 north	Helicopter	0.1-0.2		DOE	102	
0328	6:05 p.m.	7	Helicopter	1		DOE	102	In center of plume
0328	10:00 p.m.	2-3 northwest	Ground	12		Met Ed	101	
0329	6:00 a.m.	1-2 west	Ground	30	β & γ	Met Ed	116	Highest ground offsite reading
0329	6:00 a.m.	1-2 west	Ground	20	γ	Met Ed	116	Highest ground offsite reading
0330	9:00 a.m.	0.5-1.0 east-southeast	Ground	10	β & γ	Met Ed	117	
0330	9:00 a.m.	0.5-1.0 east-southeast	Ground	0.4	γ	Met Ed	117	
0330	9:00 a.m.	0.5-1.0 south-east	Ground	8	β & γ	Met Ed	117	
0330	9:00 a.m.	0.5-1.0 south-east	Ground	4.5	γ	Met Ed	117	
0330	11:53 a.m.	PA 441, Red Hill Farm Fruit Stand	Ground	5-6	β & γ	DOE	118	
0330	12:15 p.m.	Goldsboro	Ground	5		DOE	119	
03[illegible]	[illegible]:00 p.m.	1-2 west	Ground	6	β & γ	Met Ed	117	
0330	4:00 p.m.	1-2 west	Ground	1	γ	Med Ed	117	
0330	10:35 a.m.	PA 441 northeast	Ground	17	β & γ	Met Ed	108	Highest offsite reading that day
0330	10:35 a.m.	PA 441 northeast	Ground	4	β	Met Ed	108	Highest offsite reading that day
0331	2:39 p.m.	Gingrich Road, 1 east	Ground	7	β & γ	Met Ed	109	
0331	2:39 p.m.	Gingrich Road, 1 east	Ground	2	γ	Met Ed	109	

0331	9:03 p.m.	¼ east of Observation Center	1800 MSL*	19	γ	Met Ed	120	Highest airborne offsite reading that day
0331	12:00 p.m.	PA 441, ¼ east	Ground	7	β & γ	DOE	121	
0331	6:55 p.m.	New Cumberland	Ground	1.5	β & γ	DOE	121	
0401	4:32 a.m.	Over the 500 kV Substation	650 MSL	30	β & γ	Met Ed	110	
0401	4:35 a.m.	Observation Center	Ground	7.5	β & γ	Met Ed	110	Highest reading that day
0401	4:35 a.m.	Observation Center	Ground	1.0	γ	Met Ed	110	Highest reading that day
0401	6:51 a.m.	1-2 southeast	Ground	2.5	β & γ	Met Ed	110	
0401	6:51 a.m.	1-2 southeast	Ground	1.5	γ	Met Ed	110	
0401	12:45 p.m.	Falmouth Pike & PA 441	Ground	2.5	β & γ	DOE	122	
0401	12:45 p.m.	Falmouth Pike & PA 441	Ground	1.5	γ	DOE	122	
0402	1:44 p.m.	Goldsboro Square	Ground	1.5	β & γ	Met Ed	111	Highest offsite reading that day
0402	1:44 p.m.	Goldsboro Square	Ground	0.1	γ	Met Ed	111	
0402	11:15 p.m.	Goldsboro	Ground	0.5	γ	DOE	123	
0403	1:15 p.m.	PA 441, north	Ground	3.0		DOE	124	
0403	2:50 p.m.	0.4 east	Ground	1.0		DOE	125	
0404	4:43 a.m.	Above Goldsboro	450 MSL	1.4	β & γ	Met Ed	126	
0404	6:33 a.m.	Goldsboro	Ground	3	β & γ	Met Ed	127	Highest offsite reading that day
0404	6:33 a.m.	Goldsboro	Ground	0.03	γ	Met Ed	127	Highest offsite reading that day
0405	5:41 a.m.	0.5 east	650	1.8	β & γ	Met Ed	128	
0405	6:30 a.m.	2-3 east-northeast	Ground	0.4	β & γ	Met Ed	129	
0405	6:30 a.m.	2-3 east-northeast	Ground	0.08	γ	Met Ed	129	
0405	10:46 a.m.	0.2 south	Ground	1.9		DOE	130	

*MSL—mean sea level

highest exposure rate seen on site this day, was 365 mR/h ($\beta + \gamma$) or 50 mR/h (γ). The Commonwealth of Pennsylvania, together with the DOE and the NRC, observed levels of 1 to 10 mR/h ($\beta + \gamma$) in the offsite area during the first day. (Normally, the unit "Roentgen" (R or mR) should not be used to denote the exposure to β-radiation. We are using those units because most of the survey instruments indicate mR/h. The actual values of the potential $\beta + \gamma$ dose are highly uncertain because the instruments were not calibrated for the conditions of the exposure and the mixed radiation fields. The actual values are most likely less than the indicated readings.)

Airborne measurements were made in the plume by the DOE helicopter. These helicopter observations indicated that the plume could be detected out to a distance of 16 miles (0.1 to 0.2 mR/h) with a centerline passing from the plant north to Hummelstown. The plume was bounded on the east with a line to Hershey and on the west with a line to Rutherford Heights.

March 29, 1979—The highest ground exposure rate noted on this day was 150 mR/h ($\beta + \gamma$) and 100 mR/h (γ) on the fence line. The maximum offsite surface exposure rate observed was 30 mR/h ($\beta + \gamma$) and 10 mR/h (γ). The highest airborne exposure rate observed during the accident was 3000 mR/h ($\beta + \gamma$) and 400 mR/h (γ).

March 30, 1979—Releases resulting from venting of the makeup tank yielded onsite ground exposure rates of 30 mR/h ($\beta + \gamma$) and 9 mR/h (γ) at the fence due south of the plant. Exposure rates of 20 mR/h ($\beta + \gamma$) and 8 mR/h (γ) were observed at the same time at the fence line southwest of the plant. At 8:02 a.m., a helicopter measurement was taken directly in the plume. An air exposure rate of 1200 mR/h, the highest rate seen that day, was observed at an altitude of 600 feet mean sea level (MSL) (approximately 130 feet above the TMI-2 stack). At 9:00 a.m. offsite ground readings peaked at 10 mR/h ($\beta + \gamma$) and 0.4 mR/h (γ). These readings probably represented the effects of the venting of the makeup tank. At 3:00 p.m., the maximum onsite surface reading of the day was observed (90 mR/h ($\beta + \gamma$) and 9 mR/h (γ) at the fence west-northwest of the plant).

March 31, 1979—At 3:28 a.m., the highest onsite ground exposure rate of the day of 150 mR/h ($\beta + \gamma$) and 20 mR/h (γ) was observed at the fence line. The maximum offsite surface exposure rate was 17 mR/h ($\beta + \gamma$) and 4 mR/h (γ), at 10:35 a.m. on Pa 441, northeast of the B cooling tower. The Met Ed helicopter team observed the maximum airborne exposure rate of the day of 19 mR/h (γ) at 7:03 p.m. at an altitude of 1800 feet MSL, ¼ mile east of the observation center. DOE Bettis teams monitoring the offsite area observed a maximum of 7 mR/h ($\beta + \gamma$) ¼ mile east of the plant on Pa 441 at 12:20 p.m.

April 1, 1979—Maximum exposure rates were lower on April 1, 1979. The maximum onsite surface exposure rate was 40 mR/h ($\beta + \gamma$) and 10 mR/h (γ) measured at the fence line. The Met Ed helicopter team reported an exposure rate of 30 mR/h ($\beta + \gamma$) 650 feet MSL above the 500 kV substation at 4:32 a.m. These readings were the highest measured during the day.

April 2, 1979—Exposure rates on site and off site were considerably lower on April 2. The maximum onsite ground exposure rate was 15 mR/h ($\beta + \gamma$) and 7 mR/h (γ). The highest offsite reading was 1.5 mR/h ($\beta + \gamma$) and 0.1 mR/h (γ) in Goldsboro Square. The Met Ed helicopter team observed 90 to 240 mR/h ($\beta + \gamma$) over the TMI-2 screen house. DOE Bettis teams observed a maximum of 0.5 mR/h (γ) at the Pennsylvania Fish Commission boat access in Goldsboro.

April 3, 1979—On April 3, the maximum onsite ground exposure rate was 10 mR/h ($\beta + \gamma$) and 1.9 mR/h (γ), observed at the fence line. A DOE team observed the maximum offsite exposure rate of 3.0 mR/h on Pa 441, north of the plant.

April 4, 1979—Exposure rates were slightly higher on April 4. The maximum onsite ground exposure rate observed was 5.5 mR/h ($\beta + \gamma$). Maximum offsite airborne exposure rate of 1.4 mR/h ($\beta + \gamma$) was observed above Goldsboro at 450 feet of elevation. The maximum offsite ground exposure rate was 3 mR/h ($\beta + \gamma$) and 0.03 mR/h (γ) measured in Goldsboro.

April 5, 1979—Some releases of radioactive material continued on April 5. The maximum airborne exposure rate was 1.8 mR/h ($\beta + \gamma$), at 650 feet (MSL). The maximum offsite exposure rate observed by Met Ed teams was 0.4 mR/h ($\beta + \gamma$) and 0.08 mR/h (γ) 2 to 3 miles east-northeast to northeast of the site. The maximum onsite exposure rate was 3.5 mR/h ($\beta + \gamma$) and 0.6 mR/h (γ), east of TMI-2 at the fence line. The maximum offsite ground exposure rate observed by a DOE team was 1.9 mR/h at 0.2 miles south of the plant.

April 6, 1979—By April 6, offsite exposure rates had dropped almost to natural background levels. Some small onsite exposure rates were observed, and these will continue as recovery operations are carried out.

Conclusion—The exposure rates observed on site and off site as a result of the accident were low. The maximum airborne exposure rate reported at any time was 3000 mR/h ($\beta + \gamma$) and 400 mR/h (γ). This reading was made directly in the plume over the plant on the afternoon of March 29. The release quickly dissipated and exposure levels on the ground on site were orders of magnitude less. On March 30, an airborne exposure of 1200 mR/h ($\beta + \gamma$) was observed in the plume about 130 feet above the TMI-2 stack. Again, releases of radioactive material quickly dissipated and the exposure levels on the ground were orders of magnitude less.

During the period April 2 to April 13, the DOE Environmental Measurements Laboratory (EML) conducted offsite radiation exposure rate measurements at distances of 0.37 to 9.26 miles from the plant. The detectors deployed by the EML provided the most precise measurements of exposure rates off site. Of the 37 sites at which measurements were made, only three had exposure rate levels above background; the highest one was 1 mR/h, on April 3, 0.37 miles from the plant.

d. Summary of Radiological Environmental Sampling Results

In response to the accident, thousands of environmental samples were collected (and continue to be collected) by Met Ed, the Commonwealth of Pennsylvania, and the various agencies of the Federal Government. Samples were collected during the period of March 28 to April 16, from air, water, milk, vegetation, soil, and foodstuffs. Our review of these sampling results indicates that although several radionuclides (^{137}Cs, ^{89}Sr and ^{90}Sr, ^{133}Xe, and ^{131}I) were detected in some samples, only very low levels of radioiodines and radioxenons can be attributed to releases from the accident. The trace quantities of radiocesium and radiostrontium detected in a few samples are attributed to and consistent with residual global fallout from previously conducted nuclear weapons tests. This confirms that the releases from the TMI facility were limited to the noble gas radionuclides and a small quantity of radioiodines.

Air Samples—Releases were detected in the offsite area by sampling the air at ground level. For all samples taken from March 28 to April 12, when changing of the filters in TMI's process ventilation was initiated, the levels of ^{131}I detected off site were very low (a few picocuries per cubic meter (pCi/m^3) or less). The highest concentration observed during this period was 32 pCi/m^3.[131] The maximum permissible concentration (MPC) of ^{131}I in the air in an unrestricted area is 100 pCi/m^3.[132] Increased levels of radioiodines were detected after April 12, over a wide area close to the plant. These releases of radioiodine were attributed to the filter-changing operations in TMI-2.[133] Three samples obtained by NRC in the area immediately downwind of the plant during the 24-hour period ending at midnight on April 16, indicated ^{131}I levels of 110–120 pCi/m^3, the highest observed ^{131}I concentration off site.[134] At 12:27 a.m. on April 16, 1979, a sample taken at the gate of the 500-kV substation contained 88 pCi/m^3.[135]

EPA ground measurements of radioiodines in air around the site during this period were below detectable concentration levels. The maximum concentration that the EPA observed away from the site was 2.3 pCi/m^3, in a sample collected from 11:58 a.m. on April 15 to 9:15 a.m. on April 16, at the Charles Brooks residence in Falmouth, Pa. Most of the positive airborne concentrations observed by EPA during the April 12 to April 16 period were 1 pCi/m^3 or less.[136]

Particulate air samples taken in the area after the accident did not show any particulate radionuclides attributable to the accident at TMI. Isotopes of xenon, namely 131m, 133, 133m and 135, were the only radioactive gases detected.[125]

Milk Sampling Results—After the accident, small concentrations of ^{131}I were detected in a few samples of the hundreds of samples of milk taken. The milk was produced at several farms within 15 miles of the site. The highest radioiodine concentration was 41 pCi/l in a sample of goat's milk collected by Met Ed on March 30, 1.2 miles north of the site, along Pa Route 441.[137] The highest levels of radioactivity in cow's milk were detected by the FDA. These were 36 pCi/l of ^{131}I (originally reported to be 41 pCi/l) and 46 pCi/l of ^{137}Cs[138] (the ^{137}Cs was attributed to fallout from previous weapons testing). These values are well below the EPA protective action level for milk of 12 000 pCi/l of ^{131}I and 340 000 pCi/l of ^{137}Cs.[138] Traces of ^{89}Sr and ^{90}Sr were also detected in 12 of 694 milk samples collected by the FDA and were attributable to residual fallout from previous atmospheric nuclear testing.[139]

No ^{131}I was detected in the milk samples collected by the EPA, although a single sample indicated a trace (6.7 pCi/l) of ^{137}Cs. This trace was also attributed to residual global fallout.[140]

Surface/Drinking Water Sampling Results—Only three surface water samples of the many collected postaccident indicated any positive radioiodine results. The results of these samples, taken by Met Ed, were 0.4 pCi/l, 0.72 pCi/l, and 0.66 pCi/l.[141] The MPC for ^{131}I in water for unrestricted areas is 300 pCi/l.[142]

Effluent Water Sampling Results—The EPA collected samples of the effluent from the TMI outfalls. Xenon-133 was detected in only four samples of liquid effluents from TMI outfalls that were taken by the EPA:[143]

- 1200 pCi/l from Outfall 002 (12 inch) at 4:30 p.m. on April 4.
- 5100 pCi/l from Outfall Marker 112 (20 inch) at 4:40 p.m. on April 4.
- 110 pCi/l at Outfall 003 at 3:00 p.m. on April 10.
- 130 pCi/l at Outfall 003 at 10:33 a.m. on April 11.

Only one positive radioiodine sample was collected from the TMI oily waste sump. The result of this sample, which was taken by the EPA at 10:45 a.m. on April 12, was 740 pCi/l.[143]

Vegetation Sampling Results—During the period from March 28 to April 12, 1979, only two vegetation samples yielded positive ^{131}I results. The samples were collected by the DOE on April 3, 1979 (80 pCi/m^2, at 11:27 a.m., north of Red Hill Plaza) and on April 4, 1979 (260 pCi/m^2 at 5:00 p.m., at a point 3 miles north of pole No. T-761).[144,145]

During the period April 13 to 16, the DOE collected many grass samples. Iodine-131 was detected in eight samples. The highest level detected was 730 pCi/m^2 of ^{131}I obtained from a sample taken near the plant in an area beneath the plume.[146]

Conclusion—The low levels of radioiodines and traces of radioxenons collected in environmental samples taken from the area around Three Mile Island Station confirm that releases of radioactive material from the accident were not significant. All of the offsite analytical results were significantly below regulatory limits.

e. Summary of TLD Data

Various types of TLDs were deployed in the environs of Three Mile Island before, during, and after the accident to determine the radiation characteristics of the radioactive materials released. The types of TLDs used by each of the groups responding to the accident and pertinent information regarding the TLDs are summarized in Table II-19.

Because all of the TLDs used were different, each had unique energy response characteristics, and the materials included in the TLD package to make the TLD respond uniformly over a wide range of energies also were different. These differences, coupled with a lack of background history for many of the TLD locations that were used in response to the accident, made interpretation of data from these devices difficult.

Table II-20 contains the results of the Met Ed TLDs for the period December 27, 1978, through April 15, 1979. These TLDs were in place since December 1978 for the quarterly dose assessment in accordance with the REMP. These dosimeters were retrieved on March 29, to determine the offsite population dose. Replacement dosimeters were changed at 3-day intervals in accordance with the augmented REMP. The data in Table II-20 were corrected for background, resulting in the data shown in Table II-21 that are the net dose data attributable to the accident.

TABLE II-19. Summary of TLD types deployed at Three Mile Island station

Group	TLD Supplier	TLD Material	TLD Reader Used
Met Ed	Teledyne/Isotopes	$CaSO_4$:D_y	Teledyne Model 7300
Met Ed	RMC	$CaSO_4$:T_m	RMC UD-505A
NRC	RMC	$Li_2B_4O_7$:Cu +Ag/$CaSO_4$:T_m	UD 710
NRC	RMC	$CaSO_4$:T_m	UD 710
HEW	Harshaw	LiF	Harshaw-Atlas
EPA	Harshaw	CaF_2:D_y	Harshaw 2271

TABLE II-20. Met Ed Teledyne and RMC quality control dosimeter results for first quarter 1978 background rate and total exposures including background for the period December 27, 1978 to April 15, 1979[148]

Site Identification	1st Quarter 1978 Background Rate (mR/month)	Total Exposures Including Natural Background (mR)						
		12/27/78 to 3/29/79	3/29/79 to 3/31/79	3/31/79 to 4/03/79	4/03/79 to 4/06/79	4/06/79 to 4/09/79	4/09/79 to 4/12/79	4/12/79 to 4/15/79
1C1	4.10	20.1	3.2	1.4	0.5	0.5	0.6	0.3
7F1	6.57	24.1	1.1	0.5	0.9	1.0	0.7	0.5
15G1	5.13	18.4	1.9	-0.7	0.5	0.8	0.4	0.5
12B1	3.57	16.3	9.4	0.2	1.2	1.3	0.3	0.1
9G1	5.60	21.3	1.4	0.1	0.6	0.9	0.6	0.5
5A1	4.60	18.6	8.3	7.7	3.0	1.2	2.2	0.2
4A1	4.60	20.2	34.3	41.4	2.2	0.7	0.6	0.4
2S2	4.07	43.7	32.5	3.4	0.9	0.6	0.3	0.2
1S2	4.67	97.2	20.0	-0.1	0.6	1.4	0.4	0.2
16S1	6.40	1044.2	83.7	7.0	1.5	1.0	0.6	0.6
11S1	5.07	216.0	107.1	45.0	21.8	8.5	1.1	0.6
9S2	4.67	25.0	25.1	4.6	1.8	1.3	0.4	0.3
4S2	4.80	35.5	124.3	28.0	7.9	1.6	0.6	0.2
5S2	4.30	30.5	49.3	26.7	15.5	6.0	2.7	0.2
4G1	5.30	17.2	1.2	0.6	0.6	0.7	0.4	0.3
8C1	3.50	13.0	10.7	1.7	1.3	1.0	0.4	0.1
7G1	7.20	25.8	1.0	-0.5	0.8	1.1	0.7	0.4
1GA1	2.03	907.7, 453.4	45.1	1.7	0.9	0.7	0.4	0.2
14S1	2.17	131.2, 148.3	48.8	9.5	1.5	0.9	0.3	0.1
10B1	1.97	40.6, 36.6	14.9	0.4	1.1	0.8	0.6	0.4
7F1Q	6.15	23.3	0.8	1.5	0.9	1.0	1.0	0.9
15G1Q	4.70	17.6	1.1	0.8	0.7	0.6	0.6	0.7
5A1Q	4.57	16.1	5.4	5.2	2.0	1.3	1.8	0.6
1S20	5.71	95.7	15.3	1.3	0.8	0.9	0.8	0.7

TABLE II-20. Met Ed Teledyne and RMC quality control dosimeter results for first quarter 1978 background rate and total exposures including background for the period December 27, 1978 to April 15, 1979—Continued

Site Identification	1st Quarter 1978 Background Rate (mR/month)	Total Exposures Including Natural Background (mR)						
		12/27/78 to 3/29/79	3/29/79 to 3/31/79	3/31/79 to 4/03/79	4/03/79 to 4/06/79	4/06/79 to 4/09/79	4/09/79 to 4/12/79	4/12/79 to 4/15/79
16S1Q	3.93	929.4	61.5	5.6	1.3	0.9	0.8	0.9
11S1Q	5.35	168.5	75.7	35.2	14.2	5.5	1.0	0.9
4S2Q	4.91	31.4	71.4	21.3	4.7	1.0	1.0	0.7
5S2Q	4.32	27.7	36.6	21.2	11.5	4.7	2.2	0.9
4G1Q	4.94	17.7	0.6	1.4	0.7	0.8	0.8	0.7
8C1Q	4.07**	12.6	8.4	2.6	1.1	0.7	0.7	0.6

*At these three sites, two dosimeters were left in place for 6 months, thus two readings are available. This practice is followed because the sites are inaccessible during the normal quarterly exchange time (~January 1st).

**Second Quarter, 1978: First Quarter missing.

TABLE II-21. Net exposures, attributable to the accident, obtained from Met Ed Teledyne data[149]

Site designation	12/27/78 to 3/29/79	3/29/79 to 3/31/79	3/31/79 to 4/03/79	4/03/79 to 4/06/79	4/06/79 to 4/09/79	4/09/79 to 4/12/79	4/12/79 to 4/15/79
		Net Exposures Attributable to the Accident (mR)					
	X	X	X	X	X	X	X
1C1	6.5	2.0	0.7	0.1	0.1	0.1	-0.1
7F1	3.6	0.4	-0.1	0.2	0.2	0.0	-0.1
15G1	2.4	1.0	-0.8	0.0	0.1	-0.1	0.0
12B1	4.6	6.1	-0.1	0.6	0.6	0.0	-0.2
9G1	3.7	0.7	-0.3	0.0	0.2	0.0	0.0
5A1	4.0	5.3	4.8	1.7	0.5	1.2	-0.2
4A1	5.3	22.7	27.3	1.2	0.2	0.1	0.0
2S2	26.4	21.5	2.0	0.3	0.1	-0.1	-0.1
1S2	69.8	13.1	-0.4	0.1	0.6	0.0	-0.2
16S1	861.1	55.5	4.2	0.6	0.2	0.0	0.0
11S1	168.6	71.2	29.7	14.2	5.3	0.4	0.1
9S2	9.2	16.5	2.8	0.9	0.6	0.0	-0.1
4S2	17.6	82.7	18.4	5.0	0.8	0.1	-0.2
5S2	14.7	32.7	17.5	10.0	3.7	1.5	-0.1
4G1	1.0	0.6	0.1	0.1	0.1	-0.1	-0.1
8C1	2.0	7.0	0.9	0.6	0.4	0.0	-0.2
7G1	3.4	0.4	-0.8	0.1	0.3	0.0	-0.2
16A1	758.0	30.0	1.0	0.5	0.3	0.1	0.0
14S1	11.9	32.4	6.2	0.9	0.5	0.1	-0.1

TABLE II-21. Net exposures, attributable to the accident, obtained from Met Ed Teledyne data—Continued

Site designation	12/27/78 to 3/29/79	Net Exposures Attributable to the Accident (mR) 3/29/79 to 3/31/79	3/31/79 to 4/03/79	4/03/79 to 4/06/79	4/06/79 to 4/09/79	4/09/79 to 4/12/79	4/12/79 to 4/15/79
	X	X	X	X	X	X	X
10B1	27.4	9.8	0.1	0.6	0.4	0.3	0.1
7F1Q	4.7	0.4	0.9	0.3	0.4	0.4	0.3
15G1Q	3.4	0.8	0.3	0.2	0.1	0.1	0.2
5A1Q	2.3	5.1	4.7	1.5	0.8	1.3	0.1
1S2Q	78.4	14.9	0.7	0.2	0.3	0.2	0.2
16S1Q	917.5	61.3	5.2	0.9	0.5	0.4	0.5
11S1Q	152.3	75.3	34.7	13.7	4.9	0.5	0.4
4S2Q	16.5	71.1	20.8	4.2	0.6	0.5	0.2
5S2Q	14.6	36.3	20.8	11.1	4.3	1.8	0.4
4G1Q	2.8	0.3	0.9	0.2	0.3	0.3	0.2
8C1Q	0.3	8.1	2.2	0.7	0.2	0.2	0.2

Table II-22 contains daily data from the NRC TLDs for the period March 31 through April 7, 1979. These data were used by the Ad Hoc Interagency Dose Assessment Group. With the exception of the first day of NRC TLD data, the data for the exposure period of April 1 through May 1, 1979 were used by the President's Commission to determine the population dose for this period.[147] These results are discussed in more detail in Section II.B.4.a.

The TLD data indicate that the major offsite releases of radioactive materials occurred on the first day. The highest readings were obtained on site and at Kohr Island (see TLDs 16S1 and 16A1 in Table II-20). These readings indicate that the plume traveled to the north-northwest. The other high TLD readings (station 14S1) indicated that portions of the plume may have migrated to the west-northwest for short periods of time. With the exception of the Kohr Island dosimeter, all of the high readings were on site. The highest net TLD reading offsite location, about 2 miles to the southwest, was 27 mrem.

During the period of April 1 to April 3, 1979, only the Kohr Island dosimeter and the dosimeter located near the observation center indicated a dose in excess of 10 mrem. Higher readings exceeding 10 mrem were noted on site. During the period March 31 to April 3, the data indicate that no significant offsite releases occurred. Only four onsite readings exceeded 10 mrem, the highest being approximately 30 mrem.

f. Findings and Recommendations

We find that:

- Several organizations including the Federal Government responded to the accident and capably undertook the enormous task of environmental monitoring.
- The TLDs placed by Met Ed as part of its environmental radiation monitoring for routine operation provided adequate data to characterize the radiation levels in the environment attributable to the accident.
- Data from the supplementary TLDs placed in the environment by the NRC, the HEW, and the EPA following the accident were of limited use because of the different number and types of TLDs employed and the lack of information regarding background history and response characteristics of the TLDs.
- The Atmospheric Release Advisory Capability (ARAC), a computer system with the capability of predicting plume behavior and location, was a tool available for use in responding to the accident but was not effectively used within the NRC (see Appendix II.5).

We recommend that:

- The NRC reevaluate its requirements for environmental radiological monitoring to ensure that monitoring of released radioactive materials in both normal and accident conditions is at least as adequate as the environmental monitoring that occurred in response to the accident. This reevaluation should include:
  - — the location and number of TLDs permanently installed in the site environs;
  - — stations to monitor airborne (particulate, gaseous, and iodine) activity;
  - — the placement of fixed real-time instrumentation for monitoring radiation in site environs.

4. ESTIMATES OF DOSES AND POTENTIAL HEALTH CONSEQUENCES OF RELEASES OF RADIOACTIVE MATERIALS

Several independent studies using different analytical techniques have estimated the radiation exposure and resultant dose from the TMI-2 accident to the public. These studies have concluded, and we agree, that the adverse health consequences attributable to the population dose are minimal at worst.

Onsite occupational exposures during the accident were also relatively low. Only three exposures in excess of the NRC quarterly exposure limits were recorded despite high radiation fields in the auxiliary building. The adverse health consequences attributable to these exposures will be minimal at worst. The total collective occupational dose that will accrue as a result of this accident cannot be determined until recovery operations are complete.

a. Population Dose Assessment

Met Ed had TLDs in place on and around the site environs at the time of the accident (see Section II.B.3). Beginning on March 31, 1979, NRC placed additional TLDs around the site. The Met Ed and the NRC TLD data were used to assess population dose resulting from the accident.

The Environmental Protection Agency (EPA) and the Department of Health, Education, and Welfare

TABLE II-22. NRC TLD data-radiation exposures for periods from March 31 to April 7, 1979 (includes background)[150]

Station	3/31-4/1 mR	4/1-4/2 mR	4/2-4/3 mR	4/3-4/4 mR	4/4-4/[illegible] mR	4/5-4/6 mR	4/6-4/7 mR
N-1	1.0 ± 0.1	0.3	0.37 ± 0.08	0.32 ± 0.08	0.28 ± 0.08	0.32 ± 0.04	0.43 ± 0.05
N-2	(wet)	0.3	0.45 ± 0.05	0.40 ± 0.06	0.33 ± 0.08	0.48 ± 0.15	0.40 ± 0.05
N-3	1.2 ± 0.3	0.3	0.43 ± 0.05	0.32 ± 0.08	0.34 ± 0.09	0.47 ± 0.05	0.50 ± 0.11
N-4	1.0 ± 0.1	0.3	0.48 ± 0.08	0.33 ± 0.05	0.37 ± 0.05	0.42 ± 0.02	0.48 ± 0.10
N-5	(wet)	0.3	0.58 ± 0.08	0.37 ± 0.5	0.35 ± 0.05	0.48 ± 0.10	0.52 ± 0.08
NE-1	7.0 ± 2.1	0.2	0.45 ± 0.08	0.32 ± 0.04	0.45 ± 0.05	0.38 ± 0.04	0.45 ± 0.08
NE-2	(wet)	0.3	0.48 ± 0.09	0.37 ± 0.10	0.33 ± 0.08	0.47 ± 0.10	0.47 ± 0.12
NE-3	1.6 ± 0.5	0.3	0.42 ± 0.09	0.38 ± 0.08	0.37 ± 0.08	0.46 ± 0.05	0.45 ± 0.10
NE-4	2.1 ± 0.5	0.3	0.37 ± 0.05	0.38 ± 0.04	0.33 ± 0.05	0.40 ± 0.09	0.43 ± 0.05
E-1	25.0 ± 8.1	0.4	0.53 ± 0.1	0.32 ± 0.04	2.6 ± 0.60	0.50 ± 0.09	0.48 ± 0.08
E-5(E-1a)	8.4 ± 4.6	0.3	0.73 ± 0.2	0.38 ± 0.08	1.7 ± 0.45	1.2 ± 0.27	0.32 ± 0.04
E-2	4.3 ± 0.5	0.3	0.55 ± 0.7	0.55 ± 0.10	0.38 ± 0.08	0.45 ± 0.10	0.35 ± 0.08
E-3	2.1 ± 0.4	0.4	0.42 ± 0.1	0.40 ± 0.06	0.50 ± 0.06	0.48 ± 0.08	0.32 ± 0.08
E-4	2.5 ± 0.4	0.3	0.4 ± 0.1	0.35 ± 0.14	0.42 ± 0.19	0.43 ± 0[illegible]	0.22 ± 0.04
SE-1	10.1 ± 2.0	0.3	9.1 ± 1.6	0.43 ± 0.10	0.92 ± 0.19	0.40 ± 0.00	0.55 ± 0.06
SE-2	3.5 ± 0.5	0.3	4.4 ± 0.7	0.87 ± 0.16	0.38 ± 0.08	0.35 ± 0.05	0.25 ± 0.05
SE-3	2.3 ± 0.6	0.3	2.8 ± 0.7	0.57 ± 0.10	0.45 ± 0.05	0.40 ± 0.06	0.25 ± 0.05
SE-4	3.0 ± 0.4	0.3	2.1 ± 0.4	0.30 ± 0.06	0.53 ± 0.08	0.47 ± 0.08	0.25 ± 0.05
SE-5	2.5 ± 0.7	0.3	0.13 ± 0.1	0.42 ± 0.04	0.37 ± 0.08	0.62 ± 0.31	0.38 ± 0.13
S-1	1.6 ± 0.1	0.4	2.2 ± 0.4	1.1 ± 0.05	0.37 ± 0.05	0.35 ± 0.05	0.40 ± 0.00
S-2	1.0 ± 0.2	0.4	1.5 ± 0.2	0.52 ± 0.08	0.32 ± 0.10	0.35 ± 0.05	0.43 ± 0.08
S-3	1.2 ± 0.3	0.4	1.5 ± 0.3	0.47 ± 0.05	0.40 ± 0.06	0.40 ± 0.06	0.55 ± 0.10
S-4	1.2 ± 0.2	0.3	1.4 ± 0.2	0.33 ± 0.05	0.45 ± 0.10	0.55 ± 0.18	0.42 ± 0.08

SW-1	0.9 ± 0.1	0.8	1.2 ± 0.3	1.1 ± 0.18	0.37 ± 0.08	0.37 ± 0.10	0.45 ± 0.05
SW-2	0.9 ± 0.2	0.5	1.3 ± 0.3	0.37 ± 0.12	0.30 ± 0.09	0.43 ± 0.08	0.38 ± 0.08
SW-3	1.1 ± 0.3	0.4	0.78 ± 0.1	0.65 ± 0.10	0.45 ± 0.10	0.38 ± 0.08	0.42 ± 0.02
SW-4	0.9 ± 0.1	0.5	0.75 ± 0.1	0.62 ± 0.10	0.45 ± 0.14	0.50 ± 0.14	0.5C ± 0.09
W-1	3.0 ± 1.9	1.2	1.4 ± 0.24	1.7 ± 0.35	1.3 ± 0.29	0.57 ± 0.10	0.48 ± 0.08
W-2	0.9 ± 0.1	0.5	1 ± 0.1	0.62 ± 0.04	0.72 ± 0.04	0.37 ± 0.08	0.38 ± 0.08
W-3	1.1 ± 0.1	0.5	0.78 ± 0.2	1.1 ± 0.15	0.42 ± 0.08	0.38 ± 0.08	0.47 ± 0.08
W-4	1.0 ± 0.2	0.4	0.67 ± 0.1	0.42 ± 0.10	0.45 ± 0.14	0.45 ± 0.05	0.57 ± 0.08
W-5	1.2 ± 0.2	0.6	0.4 ± 0.15	0.65 ± 0.12	0.60 ± 0.13	0.40 ± 0.06	0.57 ± 0.14
NW-1	0.9 ± 0.2	1.7	1.3 ± 0.25	0.30 ± 0.06	0.38 ± 0.08	0.52 ± 12	0.53 ± 0.04
NW-2	1.2 ± 0.5	0.4	0.62 ± 0.08	0.40 ± 0.15	0.33 ± 0.05	0.35 ± 0.05	0.38 ± 0.08
NW-3	1.4 ± 0.7	0.8	0.63 ± 0.12	0.40 ± 0.25	0.38 ± 0.04	0.40 ± 0.09	0.42 ± 0.05
NW-4	5.5 ± 1.8	0.3	0.4 ± 0.06	0.30 ± 0.06	0.37 ± 0.08	0.32 ± 0.04	0.45 ± 0.10
NW-5	4.6 ± 2.0	0.4	0.42 ± 0.04	0.42 ± 0.21	0.32 ± 0.04	0.48 ± 0.08	0.45 ± 0.05
S-1a	Not in Service until 4/5/79					0.35 ± 0.05	0.43 ± 0.05
SE-4a	Not in Service until 4/5/79					0.33 ± 0.05	0.25 ± 0.05
W-3a	Not in Service until 4/5/79					0.65 ± 0.39	0.45 ± 0.10
NE-3a	Not in Service until 4/5/79					0.38 ± 0.08	0.57 ± 0.08
N-1a	Not in Service until 4/5/79					0.50 ± 0.19	0.47 ± 0.04
N-1b	Not in Service until 4/5/79					0.40 ± 0.06	0.50 ± 0.06
N-1c	Not in Service until 4/5/79					0.40 ± 0.09	0.45 ± 0.08
N-1d	Not in Service until 4/5/79					0.35 ± 0.05	0.50 ± 0.06
N-1e	Not in Service until 4/5/79					0.40 ± 0.06	0.44 ± 0.08
N-1f	Not in Service until 4/5/79					0.47 ± 0.15	0.37 ± 0.08

(HEW) also placed TLDs around the site. Because the limit of sensitivity of these dosimeters was about 10 mR, they did not provide data useful to dosimetric calculations. If significant quantities of radioactive material had been released after April 1, however, these dosimeters would have been of great value in determining the dose to the offsite population. Additional radiological monitoring in the environment by the Department of Energy (DOE), Met Ed, NRC, and the Commonwealth of Pennsylvania confirmed that radiation levels off site were quite low and remained so during the course of and subsequent to the accident (see Section II.B.3).

Ad Hoc Interagency Dose Assessment Group Study— The Ad Hoc Group[151] analyzed the TLD data available through April 7. The group determined that the most likely collective population dose as a result of the accident was 3300 person-rem for the period March 28 through April 7. The Ad Hoc Group estimated that the possible doses ranged from 1600 person-rem to 5300 person-rem. In developing these estimates, several simplifying assumptions were made. As a result, several factors known to reduce estimates of exposure were not taken into account, including: (1) shelter factor (the protection afforded to people remaining indoors), (2) population redistribution, (3) actual organ doses which are smaller than the air dose calculated from the net TLD exposure, and (4) overresponse of the dosimeters supplied by Teledyne Isotopes, Inc. In addition, a conservatively small value for background was subtracted.[151]

The highest value (5300 person-rem)[152] resulted from inclusion of data from NRC TLDs for the first day of their deployment, which yielded dose values higher than could be substantiated by other TLDs or by field or aerial measurements. The Ad Hoc Group believed that insufficient background subtraction could have been the cause.

Two other methods used to estimate the population dose were presented in the Ad Hoc Group's report. One method used standard meteorological dispersion calculations and an estimated source term to calculate the population dose. By this method, the population dose was estimated to be 2600 person-rem.[153] The other population dose estimate was based on radiation measurements made from DOE helicopters. This method resulted in a population dose estimate of 2000 person-rem.[154] A subsequent recalibration of the DOE instruments indicated that they were overresponding to the radiation emitted by ^{133}Xe, indicating that the initial DOE population dose estimate may be high.

Task Group on Health Physics and Dosimetry of the President's Commission — This Task Group estimated the offsite population dose by several methods. The primary estimate was based on the same TLD data analyzed by the Ad Hoc Group, plus certain additional data available after April 7. This Task Group concluded that the most probable population dose was 2800 person-rem,[155] without accounting for the shelter factor. With a shelter factor, the estimate of the population dose was 2000 person-rem.[156]

In arriving at its population dose estimates, the Task Group evaluated the energy-response characteristics of the TLDs, and the accuracy and precision of the measurements made. These factors were used to establish the bounds of population dose values from 600 person-rem to 6500 person-rem.

The Task Group determined that the first batch of TLDs deployed by NRC, which had been used by the Ad Hoc Group to derive its maximum estimate of population dose, was irradiated during storage and transit prior to deployment.[157] Because the contribution from this irradiation to the total dose could not be ascertained, these data were not included in the Task Group's dose assessment. Apparently, the use of a shielded shipping container and a control dosimeter was not considered either for the deployment or retrieval of the dosimeters. This situation should not have occurred and is not in accord with acceptable practice.

The Task Group used three computer models with different meteorological modeling and dispersion calculations, and a source term, to make additional population dose estimates.[158] The estimates are shown in Table II-23.

The Task Group concluded that the "most likely collective (population) dose," as determined by these methods was 500 person-rem. They also stated that even if the results were in error by as much as a factor of 10, the "highest likely collective dose" was 5000 person-rem; and the "lowest likely collective dose" was less than 50 person-rem.[159]

TABLE II-23. Population dose estimates using computer models

Computer Model	Population Dose (person-rem)
ADPIC	276
AIRDOS-EPA	390
TMIDOS	970

Other Collective Dose Estimate—Using an independent computer model for atmospheric dispersion and dosimetry, and an estimated source term considerably larger than that used by the Task Group of the President's Commission, Woodard[160] calculated the population dose to be about 3500 person-rem for the period from March 28 to April 30, although releases were effectively terminated by March 31. No corrections were made for occupancy or shielding. The uncertainties in this calculation were estimated to be within a factor of 2 depending upon whether the plume was elevated or not. The range is from a low value of 2098 person-rem to a high value of 6836 person-rem.[161]

TMI Special Inquiry Group—We analyzed the offsite population dose estimates of the studies discussed above. The estimates are summarized in Table II-24. The studies were independently performed with different methodologies, yet arrived at similar population dose estimates. Each of the dose estimates was comprehensive in its analyses of the potential pathways of the plume and the potential error sources in the data. The maximum population dose estimates indicate that the population dose could not have exceeded 5000 person-rem.

Based on our review of the population dose studies, we deemed it unnecessary to perform an additional independent analysis of the raw data. We find that the collective dose as determined by the TLDs is within the ranges estimated by the Ad Hoc Interagency Dose Group and the Task Group on Health Physics and Dosimetry of the President's Commission. Correcting for occupancy factors, shielding, and reductions in the population due to voluntary evacuation, the population dose is believed to be somewhere in the lower end of those ranges, or about 2000 person-rem.

TABLE II-24. Population dose estimates

Source	Population Dose (person-rem)
Ad Hoc Interagency Population Dose Assessment Group	3300
President's Commission, Task Group on Health Physics and Dosimetry	2800 2000*
Woodard (Pickard, Lowe, & Garrick)	3500
ADPIC	300
AIRDOS-EPA	400
TMIDOS	1000

*Includes shelter factor.

There are no data or methodologies available by which to establish the collective dose with any greater accuracy. Among the factors that contribute to the inability to improve the collective dose estimates are the uncertainties associated with individual TLD determinations at the level of doses measured, the sparcity of the data, and the influence of the many factors that contribute to additional exposures of the TLD for which correction factors cannot now be ascertained. However, the placement of the TLDs and the prevailing wind directions at the time of the accident indicate that the close-in TLDs properly measured the radiation emanating from the plume. Furthermore, because the health effects implications do not change in this range of population doses, it is not necessary to attempt to estimate the range of the population dose more accurately. We find that despite the uncertainties in the TLD data, the data were adequate to characterize the magnitude of the collective dose to the population.

Additional Offsite Dosimetry— The HEW Public Health Service[164] attempted to determine offsite exposure from photographic film present in stores in the TMI area during the first 3 days after the accident. The Public Health Service concluded that even if the fogging noted on the purchased films was attributed to radiation exposure, the total dose would be less than 5 mrad. Some of these films were from the Middletown, Pa. area, adding further evidence that the offsite population exposures were low, in agreement with the TLD readings.[162]

Met Ed deployed several of its personnel TLD badges around the site as an additional means of determining onsite doses. The data from these badges were compared to the data from the environmental dosimeters. These data were very erratic and the results ranged from a factor of 6 higher to a factor of 10 lower than the environmental monitoring TLD data. No correlation or explanation for these wide variations could be established, so the results could not be used in the population dose assessment.

b. Maximum Individual Offsite Dose

The maximum individual offsite dose would be received by a person near the plant in the path of the plume. Based on the TLD data, the maximum dose would be received by an individual located on the east bank of the Susquehanna River. The Ad Hoc Interagency Dose Assessment Group estimated this dose to be 83 mrem (expressed as less than 100 mrem).[163] The Health Physics and Dosimetry Task Group of the President's Commission estimated the dose to be between 20 and 70 mrem.[164] Its estimate included correction factors for occupancy and dosimeter overresponse and is in close agreement with the Ad Hoc Group estimate. Our review of the available data and analytical methodologies employed by both groups verified these estimates.

The highest actual individual offsite dose identified was received by an individual who was on Hill Island for short periods of time during the accident. The Ad Hoc Group calculated a most probable dose of 37 mrem[165] to this individual. The President's Commission estimate was about 50 mrem.[164] Our review of the available data and analytical methodologies used by both groups verified these estimates.

We find that the maximum offsite individual dose was less than 100 mrem.

c. Internal Dose Assessment

Radionuclides that enter the body result in a radiation dose to that individual. The dose is dependent upon many factors, the most significant of which are the degree of uptake, localization, the residence time of the radionuclide(s), and the type and energy of the emitted radiation(s). The routes of intake of radionuclides into the body are well known, and the environmental sampling program before and after the accident is designed to detect and measure radionuclide concentrations in the environment. When these concentrations have been determined, the resultant internal dose to members of the public can be estimated. As described in Section II.B.2.f, the only radionuclides released to the environment in measurable amounts, as a result of the TMI accident, were noble gases and, to a much lesser extent, radioiodines.

Noble gases, when inhaled, do not chemically react within the body, and the major fraction is promptly exhaled. A small amount of the noble gases passes into the blood, a small fraction of which is dissolved in body fat. Even this fraction has a relatively short residence time. Thus, the dose received from internal exposure to noble gas is very small in comparison to the external dose that would be received by a person in or near a cloud of noble gas.

Radioiodines behave physiologically in the same manner as stable iodine. The thyroid gland concentrates and uses iodine. Radioiodine entering the body is taken into the blood; a fraction (about 25%) is taken up by the thyroid gland and remains for a significant period of time.

The report by the Health Physics Task Group of the President's Commission presented internal dose assessments. Based on the maximum concentration measured, hypothetical maximum individual doses were calculated. Because of scarcity of positive data (the majority of the environmental samples yielded negative values, below minimum detectable limit), no population dose assessment from internal exposure was performed.

The Task Group estimated maximum internal doses to individuals offsite from the ^{131}I intake to be 6.9 mrem to the thyroid of a newborn child and 6.5 mrem to the thyroid of a 1-year-old child. On site, they estimated the maximum dose to an adult thyroid to be 53 mrem.[166] The Task Group also estimated maximum internal whole body dose from the other radionuclides, such as ^{133}Xe, to be 0.3 mrem and the lung dose to be 3 mrem. These estimates agree with those reported by the Ad Hoc Group.[167]

Further confirmation of the type of radionuclides released by TMI and the small internal population dose was provided by whole-body counting. Several hundred people residing in the environment of TMI underwent this procedure and all results were negative for radionuclides that could have been released during the accident. We find that the contribution of internal exposure to the population and individual dose was small compared to the dose from external irradiation.

d. Skin Dose Assessment

In case of an immersion in a plume of xenon-133 (the major radionuclide released), the skin dose from beta radiation could be up to four times higher than the whole-body gamma dose.[168] The maximum permissible dose to the skin, however, is six times that of the whole body.[169]

Points of plume touchdown and data from TLDs on integrated beta dose were not reported. In any case, any individual in the plume would have benefited from shielding afforded by clothing. For these reasons, the Health Physics Task Group did not quantitatively assess the skin dose from beta radiation.[168]

The health effects of skin exposure are considerably smaller than those from whole-body exposure. Thus, the possible additional skin exposure would not have any discernible effect. The Ad Hoc Group reached similar conclusions.[170]

e. Occupational Exposure

Met Ed reported three accident-related whole-body exposures in excess of the NRC quarterly limit of 3 rem. These doses were 3.9, 4.1, and 4.2 rem. In addition, two workers received overexposures to their hands. These doses have been calculated by the NRC at about 50 rem to skin of the forearm of one worker and about 150 rem to the fingers of the other[171]. The worker who received 150 rem to his fingers is the same individual who received a whole-body exposure of 4.2 rem. (On August 27, 1979, six workers received overexposures to the skin and extremities. The doses, as measured by TLDs, were up to 50 rads to the skin and between 40 and 150 rads to the extremities.)[172]

The potential for severe, additional overexposures existed during the first few days of the accident. Extremely high radiation fields, in excess of 1000 R/h, existed in the auxiliary building.[173] Moreover, unauthorized entries to the building were made in violation of station health physics procedures. Although a person could have been severely overexposed, there is no evidence that anyone was.

The total estimated occupational collective dose through June 30 was about 1000 person-rem.[174] Table II-25 shows the number of individuals monitored and the collective occupational doses received for the period March through September 1979.

Table II-26 shows the number of individuals who received whole-body doses in excess of 100 mrem during the period from March through September 1979. The data in this table were extracted from Met Ed's TLD personnel dosimetry report.

The collective dose received by the 1596 individuals receiving doses in excess of 100 mrem is approximately 800 person-rem. These data show that no individual has received a dose in excess of the allowable annual limit of 5000 mrem.[172] The average dose received by these 1596 individuals was 10% of that limit.

Table II-27 contains the dose accumulation rate for the seven individuals receiving more than 3000 mrem during that 7-month period. The table shows that most of the relatively high individual exposure occurred during the first month after the accident.

The collective occupational dose is smaller than that received by the surrounding population, although it will continue to rise during recovery operations. Moreover, the Health Physics and Dosimetry Task Group of the President's Commission concluded, after its review of the procedures and data regarding the occupational exposures resulting from the accident, that "the available data on occupational exposure at Three Mile Island must be treated with caution. It may be incomplete."[174] We agree with this conclusion.

We find that the accident at TMI-2 resulted in several exposures in excess of regulatory limits to plant personnel in the first few days following the accident. We find further that the collective occupational dose and the extent of overexposure is not large in relation to the radiation fields and contamination levels encountered during the accident, although the actual collective occupational dose is not precisely known.

f. Health Effects of Low Level Ionizing Radiation

The human health effects of ionizing radiation may be classified as: (1) acute somatic effects, (2) developmental or teratogenic effects, (3) late somatic effects, and (4) genetic effects.

Acute somatic effects involve various forms of radiation sickness occurring shortly (a few days or weeks) after whole-body doses of about 100 rad or more. Teratogenic effects involve various kinds of developmental abnormalities following irradiation *in utero*. Such effects have been observed in animals following doses as low as 5 rad[176] and in humans following doses exceeding 50 rad.[177] There is no evidence associating much smaller doses of radiation to developmental effects.[178,179]

The radiation exposures caused by the accident resulted in individual doses considerably smaller than those associated with acute and teratogenic effects. The most important effects of radiation on man which may be caused by low level radiation are those which may appear, or continue to appear, at long intervals of time after exposure in the individual irradiated (late somatic effects) or in his or her progeny (genetic effects). (As used in this report, "low level" or "low dose" refers to doses below individual occupational dose standards of 5000 mrem per year).

Late Somatic Effects — The most important late somatic effect of low doses of radiation is the increase of incidence of cancer. Most human studies on populations exposed to radiation (e.g., atomic bomb survivors in Hiroshima and Nagasaki, radium dial painters) indicate that radiation-induced life

TABLE II-25. Occupational dose March 1 to September 30, 1979 [175]

Month	Number of Dosimeters Distributed	Collective Dose (person-rem)
March	1131	334
April	4504	140
May	5282	350
June	2973	159
July	2500 (approx.)	63
August	2500 (approx.)	63
September	2472	36

TABLE II-26. Occupational doses in excess of 100 mrem March 1, 1979 to September 30, 1979

Dose Range (mrem)	100–250	251–500	501–750	751–1000	1001–2000	2001–3000	3001–4000	4001–5000	More than 5000
Number of Individuals	648	465	213	118	129	16	4	3	0

TABLE II-27. Dose accumulation rate for individuals receiving more than 3000 mrem from March 1, 1979 to September 30, 1979[175]

	Dose (mrem)						
Period	Indiv. A	Indiv. B	Indiv. C	Indiv. D	Indiv. E	Indiv. F	Indiv. G
03/01-03/31	4100	4120	1785	3575	2230	1785	2360
04/01-04/30	160	10	915	40	990	915	1335
05/01-06/30	15	30	45	220	100	45	180
07/01-09/30	30	15	395	70	345	395	210

shortening is largely due to increased cancer mortality.[180,181]

Radiation-induced cancer is detectable only in a statistical sense. A particular case cannot be attributed to radiation[182]. Human evidence for radiogenic cancer comes from epidemiological studies conducted on relatively large population groups exposed to doses much larger than those experienced by the population in the vicinity of the Three Mile Island Station. Numerous animal studies confirm the carcinogenic properties of radiation, but those studies also necessarily involved exposure to relatively large doses. Cancers induced by radiation are indistinguishable from those occurring from other causes. Radiogenic cancer thus can only be inferred on the basis of an excess above the expected natural incidence.

Theoretical considerations suggest that at any level of radiation, no matter how small, some carcinogenic potential exists. Thus far, nearly all human data rely on observations at high dose levels and high dose rates (doses generally greater than 50 rem and dose rates on the order of rads per minute) and the risk factors given in most scientific publications[183,184,185] are derived from these data. To quantitatively assess the health consequences of the incremental radiation exposure received by the population as a result of the TMI-2 accident, it is necessary to determine how the risk factors derived from relatively high doses and dose rates can be used in estimations of health effects resulting from doses of a few millirads to tens of millirads of low LET radiation. (LET, linear energy transfer, is the average amount of energy lost by particle per unit of track length; low LET radiation characteristics of beta rays (electrons), X-rays and gamma rays, are radiations to which the population in the vicinity of TMI was exposed.)

One way of determining radiation risk factors, which serves as the basis of current radiation exposure standards, is to assume that the effects observed at high doses from high dose rates can be directly and linearly extrapolated to low doses delivered at very much lower dose rates, and that there is no dose (or threshold) below which there is no health risk. Applying these assumptions results in a linear, nonthreshold, dose-rate independent, dose-effect relationship.

The majority of the scientific community considers that the linear, nonthreshold extrapolation represents the upper limit of effects at very low doses, and that the risk factors derived using such an extrapolation probably overestimate the actual risk.[186] This view is stated in relevant publications of the National Academy of Sciences (BEIR I and BEIR III)[185,187] and the United Nations (UNSCEAR 77).[188] Both BEIR I and BEIR III indicate that the actual risk could be appreciably smaller for low level irradiation, and even zero. However, they also indicate that, because of the greater killing of cells at high doses and high dose rates, extrapolations based on effects observed under such conditions may be postulated to underestimate the risks. In most cases, however, the linear hypothesis probably overestimates rather than underestimates the risk from low level, low LET radiation.

BEIR III further states that it is not known whether dose rates of gamma or X-radiation of around 100 mrad/year are detrimental to exposed people; any somatic effects would be indistinguishable from those occurring naturally or caused by other factors. The observed variations in incidence (from place to place and from year to year) are far greater than any likely effect of radiation delivered at such dose rates.

The 1977 UNSCEAR report is consistent with the view of BEIR I and BEIR III that the linear nonthreshold extrapolation describes the upper limit of risk. UNSCEAR concluded that at doses of a few rad, the estimates are likely to be too high and the actual rate might be substantially lower. UNSCEAR also states that the risk from irradiation due to radionuclides deposited within the body is not different from that from external radiation, provided that the absorbed dose to a given tissue is the same from both modes of irradiation. Thus, the risk from the total radiation dose received by the population is the same whether the dose is received from external exposure or from radioactive materials that might have been ingested or inhaled.

Upper limits of possible premature cancer deaths resulting from this accident can be estimated using the linear, nonthreshold dose-response relationship. However, in addition to dose response relationships, several other assumptions must be made in derivation of risk estimates. The ongoing human studies suffer from many imperfections: imprecise dose determination, limited number of subjects, and inability to control variables. Because these studies are not completed, many assumptions have to be made, including: (1) the duration of increased risk following irradiation, (2) latent period (time interval between irradiation and detection of effect), and (3) whether the risk following a given population dose should be expressed by some number of excess cancers, regardless of natural incidence (absolute risk), or as a fractional increase of the natural risk in a given population (relative risk). Because of the numerous assumptions that have to be made, the risk coefficients and risk estimation models published by various scientific organizations differ.[189]

The Radiation Health Effects Task Group of the President's Commission on the Accident at Three Mile Island applied risk factors and models published by various national and international risk assessment bodies, as discussed above, to estimates of doses received by the population as a result of the TMI-2 accident. Table II-28, taken from this Task Group's report, contains the ranges of projected numbers of lifetime excess cancer among the offsite population.[190] This table also shows the ranges of the estimated additional risk of developing cancer by the maximally exposed individuals in the vicinity of Three Mile Island Station. Our analysis yields the same values.

We find, therefore, that it is extremely unlikely that any individual will suffer discernible ill effects, during his or her lifetime, from radiation exposure associated with the TMI accident. The effects on the population as a whole, if any, will certainly be nonmeasurable and nondetectable.

TABLE II-28. Summary of various projected lifetime cancer numbers or risk estimates for whole-body external gamma radiation doses to offsite TMI population (within 50 miles)*[190]

Source of Estimates or Risk Factors	Projected Numbers of Cancers At 3000 Person Rem**			Cancer Risk Max. Exposed Person (approx. 70 mrem)**		
	Fatal	Non-Fatal	Total	Fatal	Non-Fatal	Total
Ad Hoc Group	0.6	0.6	1.2	$1.4/10^5$	$1.4/10^5$	$2.8/10^5$
EPA***						
General Pop.	0.3-1.6	0.3-1.6	0.6-3.3	--	--	--
Adults	0.24-0.5	0.24-0.5	0.5-1.0	$(0.7\text{-}1.4)/10^5$	$(0.7\text{-}1.4)/10^5$	$(1.4\text{-}2.8)/10^5$
Children < 10 yr.	0.06-1.2	0.06-1.2	0.12-2.4	$(0.7\text{-}14)/10^5$	$(0.7\text{-}14)/10^5$	$(1.4\text{-}28)/10^5$
Reactor Safety Study						
Upper Bound Model	0.3	0.3	0.6	$0.9/10^5$	$0.9/10^5$	$1.8/10^5$
Central Model	0.06	0.06	0.12	$0.17/10^5$	$0.17/10^5$	$0.34/10^5$
Lower Bound Model	0.0	0.0	0.0	0.0	0.0	0.0
UNSCEAR 1977	0.30	0.30	0.60	$0.7/10^5$	$0.7/10^5$	$1.4/10^5$
ICRP 1977	0.3	0.3	0.6	$0.7/10^5$	0.7^5	$1.4/10^5$

* Values obtained by applying projections or risk coefficients yielded by models in listed reports to TMI dose estimates used in this report.

** 3,000 person-rem 50% higher than most probable actual total collective dose, and 70 mrem the dose the maximally exposed individual estimated by HP&D Task Group.

*** Range for general population the sums of lower range values and upper range values for adults and children < 10 years. Extraordinarily high upper range values for children and general population due to inclusion of causally questionable association of high risk of childhood cancer with in utero diagnostic irradiation and to projection of the assumed high relative risk of radiogenic cancer in children (0-9 years) to the 50+ age group in the BEIR 1972 relative risk model used.

Genetic Effects—When cells are exposed to ionizing radiation, the chromosomes of the cell nuclei may be damaged by the production of gene mutations, involving alterations in the elementary units of heredity that are localized within the chromosomes or by the induction of changes in the structure or number of the chromosomes. When such changes are induced in the germ cells, they may be transmitted to descendants of the irradiated subject. This has been clearly established in experimental studies on short-lived animal species.

Although similar genetic changes may also be induced in humans, none has yet been demonstrated, perhaps because the effect is too small to detect with the data resources available or with present methods of observations. Direct human information is therefore limited.[177] Studies of Japanese children conceived after their parents were exposed to atom bomb radiation have not demonstrated an observable increase in genetic defects.[191] For lack of human data, estimates of the genetic risk to population from low dose and dose rates are based on linear extrapolation from low dose laboratory mouse data. The 1972 BEIR report estimated that spontaneous human mutation rates may be increased between 0.5 and 5.0% per rem of gonadal dose, which is equivalent to a mutation doubling dose of 20 to 200 rem.[192] (A doubling dose is that dose which doubles the frequency of any given effect.) The 1977 UNSCEAR Report provides similar estimates.[193] Although such risk values are difficult to translate into actual health effects, the 1972 BEIR report has estimated that a cumulative dose of 5 rem per generation might be expected in the United States to produce between 60 and 1000 genetically determined illnesses of various sorts per million live births.[49] This would represent a 0.1 to 1.6% increase over the expected incidence of 60 000 cases.

The estimates of genetic effects given in the draft BEIR III Report are not notably different from those cited above: 5 to 75 additional serious genetic disorders per million live births in the first generation following parental dose of 1/rem. Such a parental dose will, according to BEIR III estimates, result over all time (i.e., over many future generations) in a total increase of 60 to 1100 serious genetic disorders per million liveborn offspring.[195]

The ranges of risk estimates underscore the limited understanding of genetic effects of radiation on human population. But even the upper values of risk estimates are small compared to the current estimates of the existing incidence of serious human disorders of genetic origin—about 107 000 per million liveborn offspring.[196]

g. Radiation Doses Due To Natural Background and Medical Practice

In estimating the potential health impact of radiation doses received by the population in the vicinity of the Three Mile Island Station, it is useful to maintain a perspective by comparing these doses to radiation doses that the same population receives from other sources, mainly natural background and medical X-ray procedures. Mankind (and all other living things) has been exposed to ionizing radiation since the beginning of time. There are three primary sources of this natural exposure: (1) solar and galactic cosmic radiation, (2) very long-lived radioactive materials present in the earth's crust, and (3) radioactive materials produced by cosmic radiation in the atmosphere. Some of the naturally occurring radioactive materials are chemically indistinguishable from nonradioactive materials normally present in the human body and are therefore always present inside our bodies (e.g., potassium-40, carbon-14).

The average dose to the gonads and bone marrow of people living in areas of normal background radiation is shown in Table II-29. The average annual dose in the United States, shown in Table II-30, is not significantly different. The doses in these tables are averages. Natural background radiation varies widely; even large local variations are possible, as shown in Tables II-31 and II-32. People living at high altitudes or in areas of high external terrestrial radiation receive much higher doses.[197]

On the basis of a nationwide survey conducted by the U.S. Department of Health, Education, and Welfare, it is estimated that in 1970, out of a population of 200 million persons, 130 million had one or more X-ray examinations.[198] The most commonly performed procedures, radiographic chest examinations and dental examinations, result in a mean dose to total active bone marrow of about 10 mrad per examination. The annual per capita rate for each of these examinations is about 0.3. Some other examinations, although performed with lesser frequency, cause much higher mean marrow doses; e.g., upper GI series, 535 mrad; barium enema, 875 mrad; pelvimetry, 595 mrad. It is estimated that in 1970 the active marrow dose per each adult in the U.S. population from medical X-ray procedures was approxi-

TABLE II-29. Global annual per capita doses from normal exposure to natural sources of radiation (in mrad)[199]

Radiation Source	Gonads	Active Marrow
External Irradiation		
Cosmic rays	28	28
Terrestrial radiation	32	32
Internal Irradiation		
Potassium-40	15	27
Radon-222	0.2	0.3
Other Nuclides	2	4
ROUNDED TOTAL	78	92

TABLE II-30. Average annual doses from natural natural background radiation in the United States (in mrem)[200]

Radiation Source	Gonads	Active Marrow
Cosmic radiation	28	28
Cosmogenic radionuclides	0.7	0.7
External terrestrial	26	26
Radionuclides in body	27	24
ROUNDED TOTALS	82	79

TABLE II-31. Selected estimates of natural "background" radiation levels in the United States (annual dose rate [mrem/year])[201]

Location	Cosmic Radiation	Terrestrial Radiation	Internal Radiation	Total
Atlanta, Georgia	44.7	57.2	28	130
Denver, Colorado	74.9	89.7	28	193
HARRISBURG, PENNSYLVANIA	42.0	45.6	28	116
Las Vegas, Nevada	49.6	19.9	29	98
New York, New York	41.0	45.6	28	115
PENNSYLVANIA	42.6	36.2	28	107
Washington, D.C.	41.3	35.4	28	105
UNITED STATES (range)	40–160	0–120	28	70–310

TABLE II-32. Examples of differences in annual doses due to natural background variations[202]

Natural Background Variation	Estimated Difference in Annual Doses
Living in Denver, Colo. compared to Harrisburg, Pa.	+ 80 mrem/yr
Living in a brick house instead of a wood frame house	+ 14 mrem/yr
Added dose from potassium-40 due to being male instead of female (There is 25% less potassium in women than men.)	+ 4.8 mrem/yr

mately 100 mrad.[203] The genetically significant dose (GSD) from medical X-ray procedures is estimated at 20 mrem per person in 1970.[204] (GSD is the gonad dose from medical exposure that, if received by every member of the population, would be expected to produce the same total genetic effect on the population as the sum of the individual doses actually received.) This lower estimate is due to the fact that in most X-ray procedures the dose to the gonads is lower than the mean marrow dose, and in calculation of GSD, the dose to the gonads is weighted, based on the expected number of future children that the irradiated individual will have.

In addition to natural background and medical X-ray procedures, there are other sources of radiation exposure to the general population; e.g., diagnostic use of radiopharmaceuticals, consumer products containing radioactive material, and air travel. The contribution of these radiation sources to the total population dose is small compared to the dose due to natural background and medical X-ray procedures. The average dose, of 1.4 mrem, received by the approximately two million people as a result of the TMI-2 accident is less than 1% of the annual dose from both natural background and medical practice.

h. Cancer Incidence and Mortality in the United States

Cancer is the second leading cause of death in the United States, after heart disease. In 1976, there were 377 312 reported deaths in the U.S. from cancer, which corresponds to 175.8 cancer deaths per 100 000 people and accounts for 19.8% of all deaths.[205] The American Cancer Society estimated that in 1979 there would be 765 000 new cases of cancer in the United States and 395 000 people will die from it, which corresponds to the death rate of 180 per 100 000 people.[206] The estimated cancer death rate for the United States varies from 57 in Alaska to 250 in Florida (not adjusted for population age distribution). The estimated death rate in Pennsylvania is 208.[207] Based on this estimate, we calculate that among the more than two million people living within 50 miles of the Three Mile Island Station, there will be approximately 4000 cancer deaths per year unrelated to the accident.

The American Cancer Society estimates that, if the present rates continue, 25% of all people in the United States will eventually develop cancer and 15% will die from it.[208] Applying these approximate statistics to the population within 50 miles of the Three Mile Island Station indicates that approximately 325 000 people in that area would normally die of cancer.

The natural incidence of cancer varies considerably depending on the type and site of the cancer, age, sex, geographic location, dietary habits, environment, and other factors. Because cancers induced by radiation are indistinguishable from those occurring naturally, it is usually impossible to determine in cases of low level radiation exposure if this radiation was causative in induction of a few of the many thousand cancer cases normally expected in a given population.

i. Summary of Health Effects

Our analysis of the potential health effects resulting from radiation exposure due to the TMI-2 accident is in accord with the conclusion of the Radiation Health Effects Task Group of the President's Commission.[209] As a result of the radiation exposure to the offsite population within 50 miles of the TMI site, the projected incidence of fatal cancer is less than one; and fatal plus nonfatal cancers is less than 1.5, with zero not excluded. This projection is to be contrasted to the nearly 541 000 cancers (325 000 fatal and 216 000 nonfatal) expected in this population over its remaining lifetime that are not related to the TMI accident.

The additional lifetime fatal cancer risk to the individual receiving the maximum probable dose offsite (less than 100 mrem) is about 1 in 100 000. The additional risk of fatal cancer to an individual receiving the average offsite dose (1.4 mrem) is about 1 in 5 000 000. This risk is additive to the existing risk of fatal cancer of about one in seven. The risk of nonfatal cancer is about the same as the risk of fatal cancer, and the combined normal risk is about one in four.

The additional cancer risks due to internal irradiation and skin irradiation are very small compared to the above values and can be regarded as being included in the values presented above for whole-body gamma irradiation. Even if the cancer risks defined above were to be expressed, the resultant cancers would not be detectable among the population in the vicinity of TMI-2. (Note that zero additional incidence is not excluded.)

The whole-body external occupational exposure of 1000 person-rem has potential total cancer risk of less than 0.5 (zero not excluded). The risk to the maximally occupationally exposed individual (4.2 rems) is about 1.2 in 1000 for both fatal and nonfatal cancers.

The potential incidence of genetically related ill health is considerably smaller than that of producing a fatal or nonfatal cancer. This risk is estimated to be about 0.002 cases per year, and about one case per million live births for all future human existence. This contrasts with an estimated 3000 cases per year of genetically related ill health among the offspring of the population in the vicinity of Three Mile Island based on present birth rate (28 000 births per year), and not related to the accident.

In our view, the fact that there will be no, or very minimal, adverse health effects from the accident has not been understood by the public. We believe that the public misconception that the risks associated with this accident, and with radiation in general, are much greater than they are in fact is due to the failure to convey credible information regarding these risks in an understandable form. Thus, we believe that substantial efforts are necessary to educate the public to eliminate the apparent gap between "real" and "perceived" risks of radiation.

Summary of Findings

We find that:

- despite the uncertainties in the offsite TLD data, it was adequate to characterize the magnitude of the collective dose to the population (Section II.B.4.a);
- the collective dose as determined by the TLDs is within the ranges estimated by the Ad Hoc Interagency Dose Assessment Group and the Task Group of the President's Commission. Correcting for occupancy factors, shielding, and reductions in the population due to voluntary evacuation, the

population dose is believed to be somewhere in the lower end of those ranges, about 2000 person-rem.

- the maximum offsite individual dose was less than 100 mrem (Section II.B.4.b);
- the contribution of internal exposure to the population and individual dose was small compared to the dose from external irradiation (Section II.B.4.c);
- the accident resulted in several exposures in excess of regulatory limits to plant personnel in the first few days following the accident (Section II.B.4.e);
- the collective occupational dose and the extent of overexposure is not large in relation to the radiation fields and contamination levels encountered during the accident (Section II.B.4.e); and
- it is extremely unlikely that any individual will suffer discernible ill effects, during his or her lifetime, from radiation exposure associated with the TMI-2 accident. The effects on the population as a whole, if any, will certainly be nonmeasurable and nondetectable. (Section II.B.4.f).

5. RADIATION PROTECTION PROGRAM

The production of power by nuclear energy entails exposure to radiation of plant personnel, as well as a risk of exposure to the general public. The primary functions of a radiation protection, or health physics, program are to maintain those exposures below limits specified in applicable Federal and State regulations and as low as reasonably achievable (ALARA).

The potential for exposure to both onsite and offsite populations increases under non-normal conditions: when the plant is undergoing major maintenance or refueling, or accident conditions. Consequently, radiation protection functions assume greater importance during such conditions.

Exposure and resultant doses can be kept ALARA by proper engineering design, good work practices, monitoring, and preplanning of the tasks to be performed. A good radiation protection program requires a concerted effort and mutual understanding on the part of management, operations, and radiation protection personnel. The program also requires an adequate staff of well-trained individuals who are supplied with appropriate instrumentation and protective devices and who have the authority to control access to radiation areas. An effective program also includes continual training and refresher courses for all plant personnel, maintenance of equipment, personnel monitoring, and the maintenance of accurate exposure records.

Fulfillment of these radiation protection functions and goals, especially during normal power operations, entails a large amount of routine work; for example, the conduct of area radiation surveys; wipe testing for contamination control; collection and analyses of air and water samples; maintenance of access control to radiation areas; issuance and control of dosimetric devices; maintenance of dosimetry records; and calibration of instruments.

Radiation protection is frequently perceived as no more than a "meter reading" and sample collecting function. The management at Three Mile Island Station, as well as a large segment of the nuclear industry, had this misconception. Radiation protection was regarded as distinctly secondary in importance to power operations and a "necessary evil." The NRC similarly did not attach great importance to radiation protection.

The radiation protection program at Three Mile Island Station was seriously deficient. Many of its deficiencies were made evident by the accident, but they were, or should have been, known well before March 28, 1979. The Three Mile Island Station program, although apparently below average, was not significantly worse than radiation protection programs at other nuclear power stations. The NRC's regulation of radiation protection programs has similarly been inadequate.

a. The Regulatory Framework

The NRC has promulgated regulations regarding radiation protection programs in 10 C.F.R. Parts 19, 20, and 50. In addition, the NRC Regulatory Guides (particularly Series 8 Guides) and Standard Review Plans (particularly Chapters 12 and 13) provide guidance regarding radiation protection programs.[210] Industry standards are established by the American National Standards Institute (ANSI). Other guidance is provided by sources such as the International Commission on Radiological Protection (ICRP), the National Council on Radiation Protection and Measurement (NCRP), and the U.S. Bureau of Mines.

The technical specifications, a part of the operating license, require that a utility establish and maintain a radiation protection program that complies with 10 C.F.R. Part 20. The NRC's Office of Nuclear Reactor Regulation (NRR) approves the procedures that the utility initially establishes and any modification or amendment of them. The NRC's Office of Inspection and Enforcement (IE) reviews the operation of the radiation protection program.

The technical specifications for TMI-2 carry only minimal specific reference to the radiation protection program. Section 6.11 states:

> 6.11 RADIATION PROTECTION PROGRAM"
>
> Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 C.F.R. Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.[211]

The NRC staff reviewed Met Ed's radiation protection program proposed in Chapter 12 of the TMI-2 FSAR[212] and discussed the review in Section 12 of the Safety Evaluation Report (SER).[213] It appears that the NRC staff review of radiation protection programs, including Met Ed's, focused on their adequacy in the conduct of normal and anticipated operational occurrences.[214] "Anticipated operational occurrences" are defined by NUREG-0115 and NUREG-0117 as "unplanned releases of radioactive material from miscellaneous actions such as equipment failures, operator error, administrative error, that are not of consequence to be considered an accident." It was implicitly assumed by the NRC staff that the program and procedures developed for normal operation would readily extrapolate to abnormal conditions. The error in this assumption was demonstrated by the accident at TMI-2.

b. Implementation and Weaknesses of the Radiation Protection Program

The deficiencies in TMI's radiation protection program, as well as the weaknesses in NRC regulation and the radiation protection response to the accident are discussed below.

Design— Consideration of radiation protection is a central part of the design of a nuclear power station. Traditionally, consideration of design has been focused on providing shielding and radiation protection facilities adequate to support normal operations and anticipated operational occurrences. During the course of the accident, the plant's design bases were exceeded, resulting in serious radiation protection problems. As a consequence, the role of radiation protection in design will have to be increased.

Shielding— The NRC staff's shielding design review[215] concluded that expected exposure to operating personnel was consistent with the requirements of 10 C.F.R. Part 20 and the ALARA concept during normal operations and anticipated operational occurrences. Met Ed classified plant areas into radiation zones based on maximum design dose rates and expected frequency and duration of occupancy. It described the location, size, and shape of significant sources of radiation in the auxiliary and fuel handling buildings and the containment structure. Source term calculations were based on: (1) 2772-MW thermal power, (2) a failed fuel rate of 1%, and (3) an acceptable set of estimated leakage rates and partition factors. Pipes, demineralizers, tanks, evaporators, pumps, and sampling points containing radioactive materials were located in shielded areas or compartments, and Met Ed proposed to use labyrinths, shield valve galleries and penetrations, reach rods, remote valve actuation, and portable shielding to maintain exposures ALARA. The assumptions used in Met Ed's shielding calculations were considered conservative and acceptable to the NRC staff.

The NRC staff's review of TMI-2 considered shielding for the primary coolant sample lines within TMI-2 but did not consider shielding these lines when they passed into TMI-1 where the primary coolant sampling room that serves both units is located. The highly radioactive primary coolant resulting from the accident and the failed TMI-2 fuel produced high radiation fields in TMI-1, reducing accessibility in those areas through which the lines passed. We find that the design of TMI-2 and the NRC staff's review of this design did not adequately consider the relationship between TMI-1 and TMI-2.

Although the NRC staff's conclusions regarding the adequacy of shielding design were valid for normal operation, highly radioactive primary coolant and wastewater were contained in the piping and tanks during the accident and produced radiation levels higher than anticipated by the design bases. We find that the shielding was not adequate to cope with the accident.

Ventilation Systems—The NRC staff's review of the TMI-2 ventilation systems[216] concluded that th design would ensure that personnel were not exposed to normal or abnormal airborne concentrations exceeding those in 10 C.F.R. Part 20, and was consistent with the ALARA concept by: (1) maintaining air flow from areas of low radioactivity potential to areas of high radioactivity potential, (2) preventing recirculating air in the auxiliary and fuel handling buildings, (3) maintaining a negative pressure in the auxiliary and fuel handling buildings with respect to the atmosphere, and (4) periodically purging the containment structure with outside air through high efficiency particulate air and charcoal filters. Various other areas of the plant contained high efficiency particulate air and charcoal filters to minimize the buildup of airborne radioactivity, and the air filtration system in the control room was designed to limit radiation exposure to occupants consistent with General Design Criterion 19 of Appendix A to 10 C.F.R. Part 50.

The NRC staff's assessment of the ventilation systems of TMI-2 did not include a review of the engineering aspects of ventilation systems (e.g., fan capacity, duct size, and balance of system). Moreover, the NRC does not possess the expertise to assess the engineering adequacy of the ventilation systems.[217]

Operational experience with the ventilation system of TMI-2 during normal operation provided evidence of deficiencies in the system ventilation. The ventilation of the nuclear sampling room and hood was inadequate for the sampling of primary coolant during normal operation. Sampling resulted in releases of radioactive material out of the face of the hood and the alarming of the mobile airborne radiation monitor in the nuclear sampling room.[218] Station personnel were aware that the monitor's alarm was indicating inadequate ventilation of the sampling hood. During the accident, this ventilation system deficiency resulted in the release of radioactive gases from primary coolant samplings, which affected the accessibility of important areas.

We find that: (1) the ventilation system in the primary coolant sampling room was inadequate for both normal and emergency operations, and (2) the NRC staff's review of the ventilation system was inadequate.

Control Room Habitability—The NRC Staff's review of control room habitability systems[219] concluded that the TMI-2 design met General Design Criterion 19. The design used concrete shielding and high efficiency filter trains to ensure a habitable environment within the control room. In the event of a high radiation signal from the monitor located in the air intake structure, the control room supply was to be automatically shut off and the safety-grade filter system (including particulate filters and carbon adsorbers) was to go into operation. The system would process 15 620 cubic feet per minute (cfm) of control room air in a recirculating mode and would process up to 1500 cfm of filtered outside air to pressurize the control room. This mode of operation could also be manually initiated by the operator.

Until nearly 8:00 a.m. on March 28, 1979, the ventilation air to the control room was treated in its normal flow path by particulate filters only. At this time, a control room operator manually activated the recirculation filter system.[220] At almost the same time, this activation would have automatically occurred on high containment pressure (7:56 a.m.). In any event, the control room was in the recirculation ventilation mode from approximately 8:00 a.m., providing protection against iodines and particulates. However, due to poor meteorological conditions, noble gases released from the facility did infiltrate the control room.

Radiation Protection Facilities— Each unit was designed to have a counting laboratory. However, the TMI-2 laboratory was never made operational. Thus, TMI-2 shared the TMI-1 laboratory. The facilities in each unit also included a calibration room for monitoring instruments, a locker room for changing into protective clothing and respirators, and a personnel and equipment decontamination room. The NRC staff's review[221] concluded that these facilities were adequate.

During the accident, the counting laboratory was disabled at the most crucial time because of high background radiation that resulted from airborne radioactive releases into TMI-1 arising from sampling of the primary coolant. The decision to don respiratory protection in the control room, which hampered communications, resulted from the inability to quickly analyze control room air. The control room air intake monitor (HP-R-220) alarmed at 9:48 a.m. for particulates and at 10:10 a.m. for noble gases. Masks were donned by the control room personnel at 10:17 a.m.,[222] and were on until 3:10 p.m. Masks were put on for a second time at 2:11 a.m. on March 29, when the particulate channel of HP-R-220 alarmed, but the levels quickly decreased and the masks were removed at 3:15 a.m.[223] We find that the improper design of the ventilation system of the sampling room and that loss of the counting room were responsible for the conservative use of respirators in the control room, which led to a severe reduction in communications ability among control room personnel and added to the difficulty in coping with the emergency.

Because of its inability to analyze samples on site, Met Ed began shortly after the accident to send air samples to the Commonwealth of Pennsylvania's Bureau of Radiological Protection in Harrisburg for analysis. By 7:30 p.m. on March 28, the NRC Region I mobile laboratory, which had analytical capability, arrived at the site. A mobile laboratory from RMC arrived on March 29. Samples were also sent to the home laboratories of RMC and Teledyne Isotopes.

We find that the design of, and the NRC staff's review regarding, the sample counting room were inadequate because the shielding and location of that facility did not provide sufficient protection to maintain operability. We find further that Met Ed's implementation of the design was inadequate because there was only one operational facility to be shared by both units.

Decontamination Facilities— Prior to the accident, Met Ed planned for routine emergency decontamination of personnel and small equipment, tools, and instruments. Decontamination facilities were provided in the health physics area of each unit but contained supplies adequate only for the limited use expected during normal operation. These facilities were utilized during the earliest stages of the accident to decontaminate personnel. At 9:10 a.m. on March 28, the airborne radiation levels became too high in the TMI-1 health physics area and the use of these facilities was lost. We find that inplant personnel decontamination facilities were inadequate to cope with emergency conditions.

Summary of Findings and Recommendations

We find that:

- the design of TMI-2 and the NRC staff's review of the design did not adequately consider the relationship between TMI-1 and TMI-2;
- the shielding was not adequate to cope with the accident;
- the ventilation system in the sampling room was inadequate for both normal and emergency operations;
- the NRC staff's assessment of the ventilation systems of TMI-2 did not include a review of the engineering aspects of ventilation systems;
- the NRC does not possess the expertise to assess the engineering adequacy of the ventilation systems;
- the improper design of the sampling room and the loss of the counting room were responsible for the conservative use of respiratory equipment in the control room, which led to a severe reduction in communications among control room personnel;
- the design of, and NRC staff's review regarding, the sample counting room were inadequate because the shielding and location of that facility did not provide sufficient protection to maintain operability, and there was only one operational counting room shared by both units; and
- inplant personnel decontamination facilities were inadequate to cope with emergency conditions.

We recommend, therefore, that licensees, in their design, and the NRC, in its review, ensure that adequate consideration be given to radiation protection matters, particularly:

- shielding, including shielding of primary coolant sampling lines;
- ventilation systems;
- counting room location and shielding;
- inplant personnel decontamination facilities; and
- the relationship between two or more units at the same site.

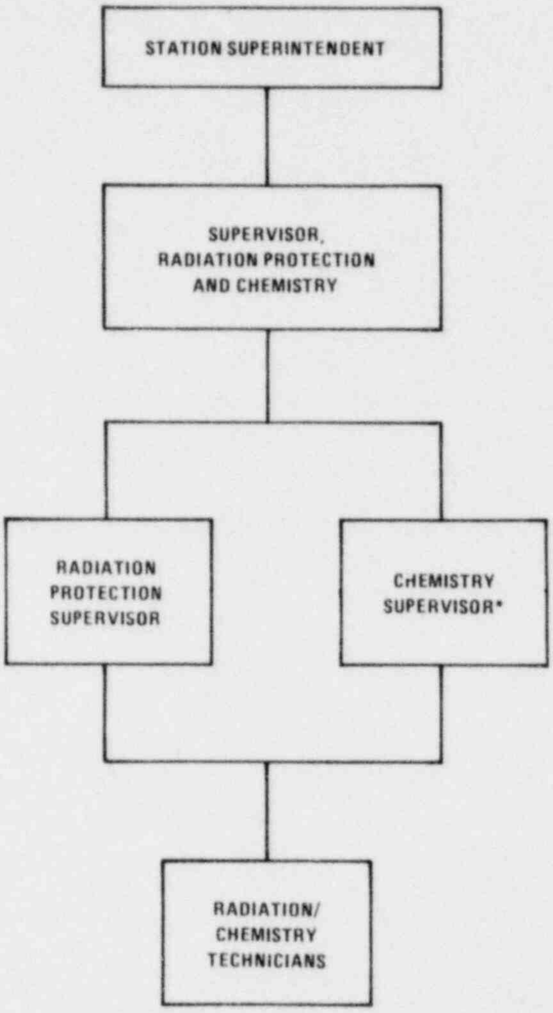
Management and Organization

The management and organization of a radiation protection program should provide effective leadership and supervision of the program during normal operation and emergency situations. The management and organization of the radiation protection program failed to fill this role because of an inadequate organizational structure, personnel who were not qualified for the positions they held, inadequate communications at all levels, and, most significantly, upper management's attitude that radiation protection was less important than production.

During the accident, the emergency organization underwent several changes because of realignments of functions and relocations of key individuals. This added to the preexisting communications inadequacies and resulted in the loss of control over the radiation protection program during the first several days of the accident.

Preaccident Organization— The organization of the radiation protection staff was approved by the NRC staff and is shown in Figure II-16. This organization has the Supervisor of Radiation Protection and Chemistry, Richard Dubiel, reporting directly to the station superintendent, the senior plant official. However, a different organization (Figure II-17) was prescribed by Met Ed at the time of the accident, which added another level of management between Dubiel and the senior plant official. Moreover, Dubiel actually reported to the unit superintendents.[224] Dubiel's theoretical and actual reporting responsibilities were inconsistent with the technical specifications.[225]

Under the Met Ed organization as approved by the NRC, the radiation protection-chemistry technicians (R–CTs) performed the dual functions of radiation protection and chemistry and reported along two different chains of command. The dual function of technicians is common in the nuclear power industry.[226] The inefficiencies of this system detracted from the implementation of an effective radiation protection program. One R–CT characterized this as "the biggest, most devastating hindrance to our



* REMAINED UNFILLED SINCE ISSUANCE OF THE TECHNICAL SPECIFICATIONS. THE DUTIES AND RESPONSIBILITIES OF THIS POSITION WERE BEING FULFILLED BY THE SUPERVISOR, RADIATION PROTECTION AND CHEMISTRY AND THE CHEMISTRY FOREMEN.

FIGURE II-16. Organization of the Radiation Protection and Chemistry Staff as Indicated in Technical Specifications, Figure 6.2-1.

department."[227] We find that the dual functions of radiation protection and chemistry that the R-CTs performed were detrimental to the efficiency of the radiation protection program.

Qualifications— Minimum qualification requirements for the radiation protection staff are contained in the FSAR, Section 13, and Technical Specification 6.3. The latter states:

> Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions except for the Supervisor of Radiation Protection and Chemistry, who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.[228,229]

Met Ed's position of supervisor of radiation protection and chemistry corresponds with the position of Regulatory Guide 1.8's radiation protection manager (RPM). The guide states that the RPM should be an experienced professional in applied radiation protection at nuclear facilities dealing with radiation protection problems and programs similar to those at nuclear power stations. The guide further indicates that he should be familiar with the design features and operations of nuclear power stations that affect the potential for exposures of persons to radiation; he should have the technical competence to establish radiation protection programs; and he should have the supervisory capability to direct the work of professionals, technicians, and journeymen required to implement the radiation protection programs.

The guide states that the RPM should have a bachelor's degree or the equivalent in a science or engineering subject, including some formal training in radiation protection, and should have at least 5 years of professional experience in applied radiation protection. At least 3 years of the RPM's professional experience should be in applied radiation protection work in a nuclear facility dealing with radiological problems similar to those encountered in nuclear power stations, preferably in an actual nuclear power station.

At the time of the accident, Dubiel, Supervisor of Radiation Protection and Chemistry, had 6½ years of power station or applicable industrial experience, a bachelor's degree in physics, and a master's degree in nuclear engineering.[230] He thus satisfied the requirements of the FSAR and Regulatory Guide 1.8.

The FSAR states that the radiation protection supervisor shall have a minimum of 5 years of experience in radiation protection at a nuclear facility. He should have a minimum of 2 years of related technical training.[231]

Thomas Mulleavy, the Radiation Protection Supervisor, had a total of 19 years of power station or applicable industrial experience and met the minimum radiation protection experience criteria by a wide margin. However, there is no evidence to indicate that he met the minimum criteria for technical training.[232]

Dubiel's time and attention were spread much too thinly because: (1) he was acting (in conjunction with the chemistry foremen) as chemistry supervisor, (2) he had the additional burden and responsibilities of running the radiation protection program for a two-unit station, and (3) he had too many people reporting to him.[233] As a result, we find that Dubiel did not function as the RPM as defined in

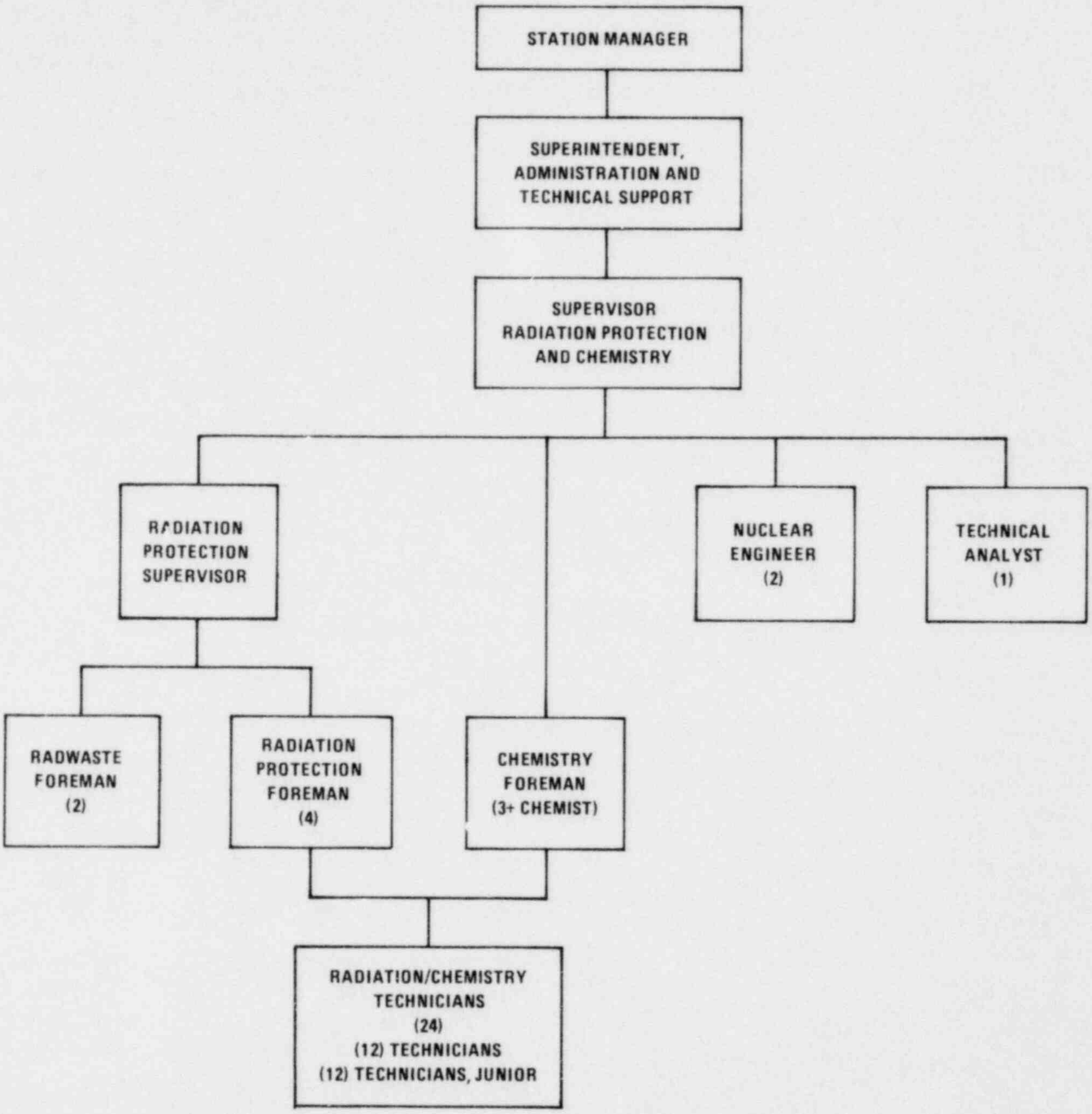


FIGURE II-17. Actual Organization of TMINS Radiation Protection and Chemistry Staff Before March 28, 1979 (NUREG-0600, P.II-1-2)

Regulatory Guide 1.8, and that the role fell, by default, to Mulleavy, who was not qualified for this position.

The FSAR states that the chemistry supervisor shall have a minimum of 5 years' experience in power station chemistry and water treatment, of which a minimum of 1 year shall be in radiochemistry. He shall have a minimum of 2 years of related training.[234] This position was not filled, but the functions and responsibilities of the position were assumed by Dubiel in conjunction with the chemistry foremen. We find that Dubiel was not qualified for this position.[230]

The positions of radiation protection foreman and chemistry foreman were established by Met Ed. We find that the positions are not identified in the technical specifications or the FSAR and that the NRC does not require minimum qualifications for these positions.

According to the FSAR, each radiation protection technician should have a minimum of 2 years' experience in radiation protection or closely related areas. He should have a thorough knowledge of the design and operation of all types of radiation monitoring and analytical instrumentation in the station.[235] Six of the 24 R-CTs did not have 2 years of working experience in their specialty.[236]

In addition, the R-CTs were transferred between functional areas and between units, with assignments seldom lasting more than 1 week. This removed them from any particular duty area for about 6 weeks, and provided little incentive for them to become highly proficient in a certain area.[237] As a result, R-CTs did not receive a thorough knowledge of the design and operation of the radiation detection instrumentation. We find that R-CTs did not develop or maintain adequate job skills and did not meet FSAR requirements.

Communications — Serious communications gaps existed between every level of the radiation protection organization. This was a substantial problem that had existed for some time before the accident, as noted in an audit of the program conducted for Met Ed.[238] We find that the communication problem contributed significantly to the deficiencies noted in the radiation protection program.

Management Attitudes—A typical conflict in the nuclear industry existed between the operations staff, which was production-oriented, and the radiation protection staff, which was service-oriented.[239,240] Station management, as well as the operations staff, viewed radiation protection as a "necessary evil."[241-243] The conflict, and management's attitude, are discussed in detail in Section II B.5.c.

The attitude of management and operations seriously impaired the effectiveness of the radiation protection program and was reflected in:

- violation of station health physics procedures by operations staff;[244]
- poor enforcement of station health physics procedures;[244,245]
- low morale of radiation protection staff;[245-248]
- failure of radiation protection management to support R-CTs;[249]
- low priority in maintenance and repair of portable radiation survey instrumentation;[250,251]
- waivers of requirements of procedures;[252] and
- lack of support (financial and personnel) of radiation protection department.[251,253]

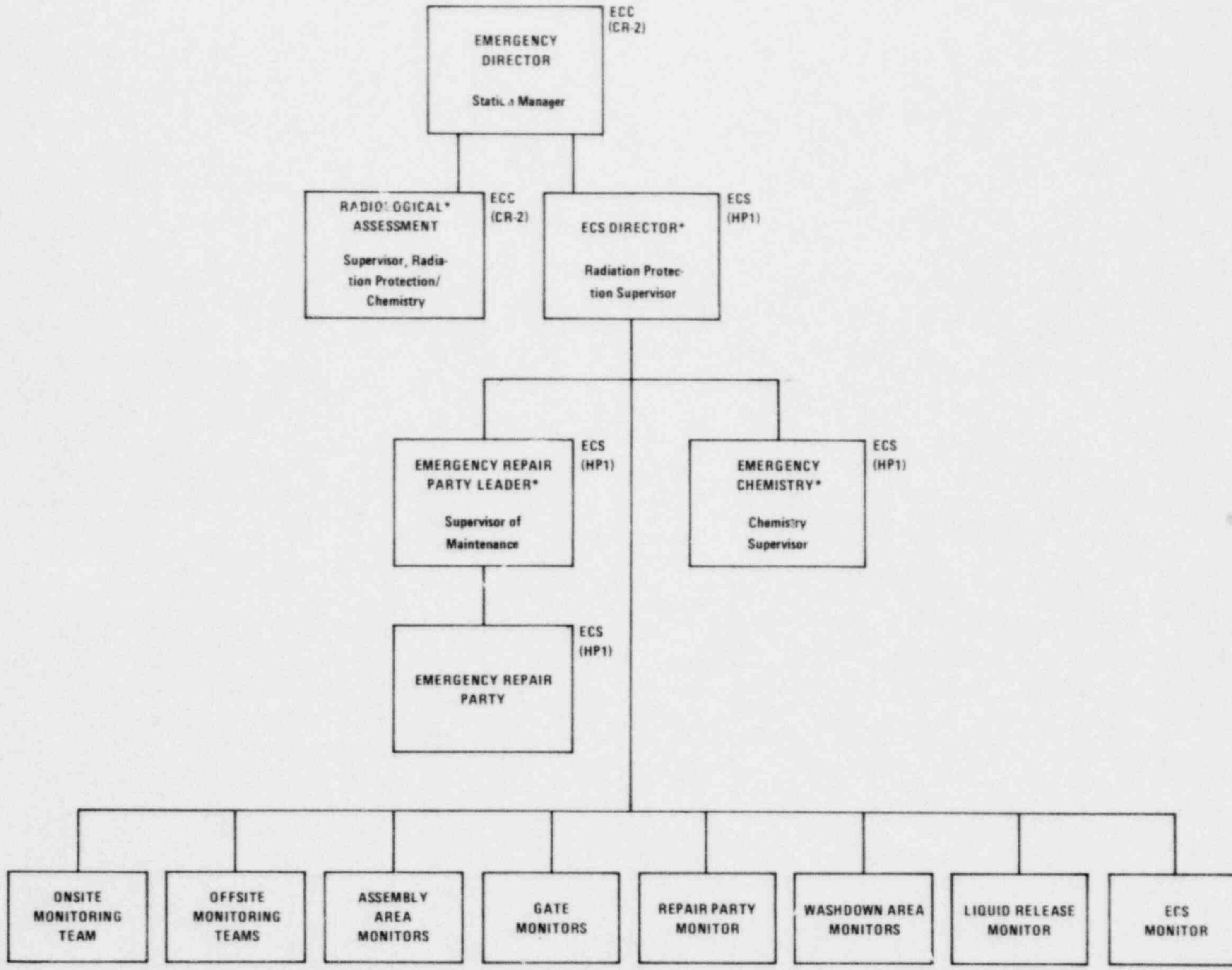
We find that the conflict between operations and radiation protection was serious, existed at all levels of the plant's organization, and contributed in a major way to the deficiencies noted in the radiation protection program.

Organization and Management During the Accident— A radiation protection organization for emergencies, different from the everyday organization, was prescribed by TMI's emergency plan. The emergency organization changed several times during the accident—first, back to the everyday organization, and then, due to forced relocations of the ECS and agreements between Dubiel and Mulleavy, several more times to different variations. With each change, some control of radiation protection functions was lost. The communications gap that existed during normal operations was compounded during the emergency and contributed to confusion that existed at the foreman and R-CT level. As a result, radiation protection control during the accident was very poor, manifesting itself in unrestricted and uncontrolled access to high radiation areas and other lapses from good radiation protection practice.

Initially, the emergency radiation protection response organization approximated that indicated in the emergency plan (see Figure II-18). Dubiel served as Radiological Assessment Advisor to the Emergency Director. Control of the emergency radiation protection organization was from the emergency director through Mulleavy who served as the ECS Director. In accordance with this organization, Mulleavy should have controlled all of the emergency repair, chemistry, monitoring (inplant and offsite) and washdown area activities.

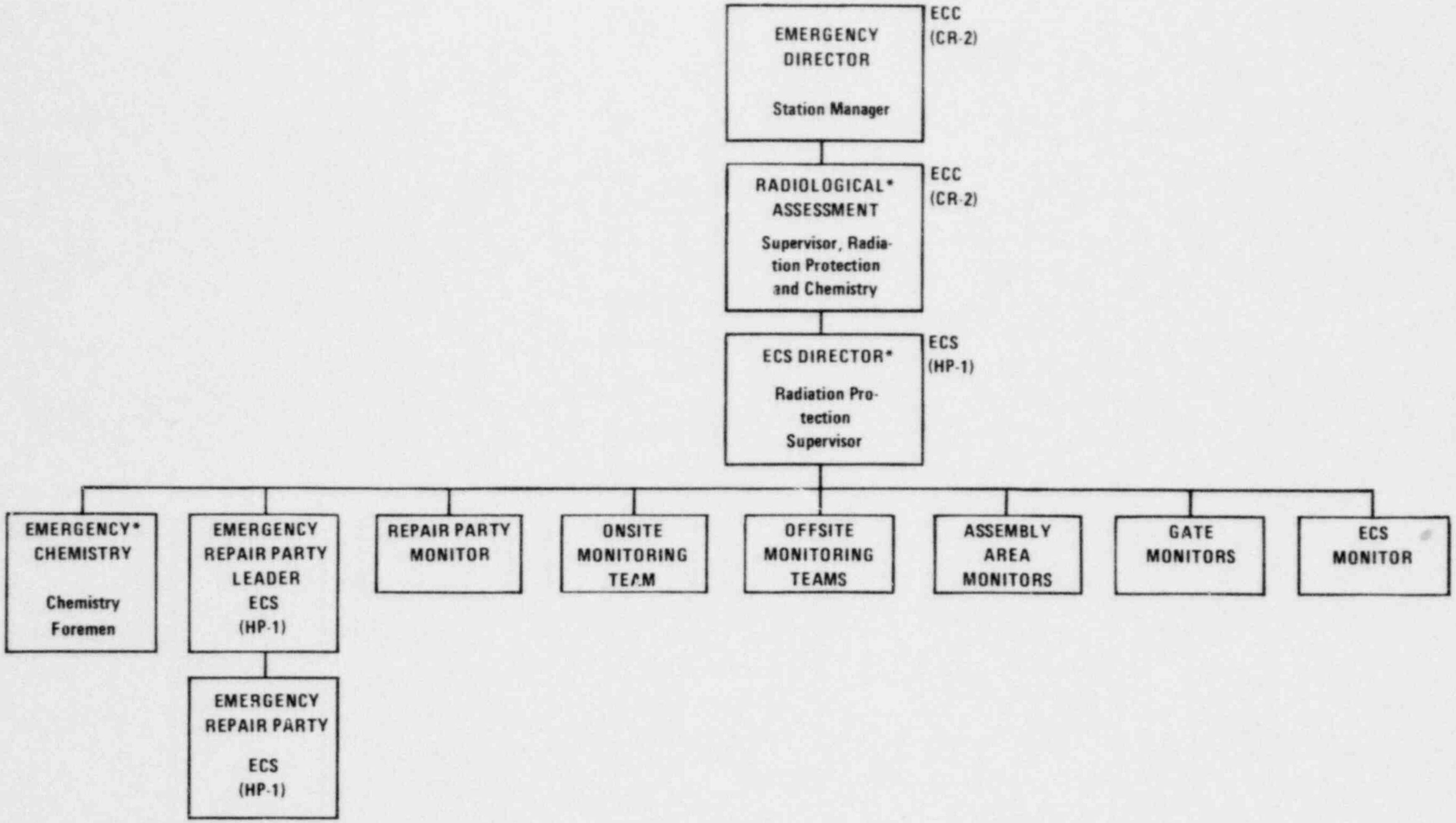
At approximately 7:30 a.m. on March 28, 1979, the emergency radiation protection organization assumed a new form that had been developed during emergency drills and was more consistent with the normal organization of the radiation protection organization. This second organizational structure is shown in Figure II-19. Mulleavy now reported to Dubiel, who was his normal supervisor, although all radiation protection functions would continue to be performed through Mulleavy at the ECS in the TMI-1 health physics laboratory area. Because of airborne radioactivity at that location, the ECS was relocated at about 9:00 a.m. to the alternative ECS in the TMI-2 control room. This relocation impaired effective control of the inplant radiation protection and repair party monitors.

Shortly after establishment of the alternative ECS, Dubiel and Mulleavy met and determined that Mulleavy would maintain control of onsite and offsite



ECC – EMERGENCY CONTROL CENTER, UNIT 2 CONTROL ROOM
ECS – EMERGENCY CONTROL STATION, UNIT 1 CHEMISTRY/HEALTH PHYSICS LAB AREA
* – FUNCTIONAL TITLE ADDED FOR CLARITY, LICENSEE'S PLAN LISTS NORMAL DUTY TITLES
SOURCE: NUREG – 0600, FIGURE II-2-2

FIGURE II-18. "Normal" Emergency Organizatior for Radiation Protection Functions



ECC – EMERGENCY CONTROL CENTER, UNIT 2 CONTROL ROOM

ECS – EMERGENCY CONTROL STATION, UNIT 1 CHEMISTRY/HEALTH PHYSICS LAB AREA

* – FUNCTIONAL TITLE ADDED

SOURCE: NUREG – 0600, FIG. II-2-3

FIGURE II-19. Emergency Organization for Radiation Protection in Effect on March 28, 1979, 7:30 a.m.-9:00 a.m.

monitoring activities and Dubiel would assume direct control of all inplant radiation protection functions.[254] Dubiel determined this action was necessary because the emergency plan did not contain provisions for dealing with the radiological conditions that then existed in the plant. This third organization is indicated in Figure II-20. At about 10:30 a.m., the alternate ECS was relocated to the TMI-1 control room, where Mulleavy assumed a new function of Onsite and Offsite Monitoring Director. The fourth emergency organization is shown in Figure II-21.

Neither Dubiel nor Mulleavy nor anyone else actually provided effective supervision of radiation protection activities during the first several days. Consequently, the personnel they were supposed to supervise often acted on their own. For example, at about 11:30 a.m., a washdown area to serve as a monitoring and decontamination point for individuals evacuating the island[254] was established at the 500-kV substation at the direction of Peter Velez, a radiation protection foreman. This action was taken without the knowledge of Mulleavy or Dubiel.[255]

The inplant radiation protection and repair party monitoring function was not suitably under Dubiel's control until the morning of March 29, 1979. Activities, including entries into the auxiliary building, occurred without his knowledge, although he believed that he had control of this function and that he was fully aware of all entries into the auxiliary building and other radiation protection activities.[256] As Michael Janouski, a Senior R-CT summarized, the radiation protection personnel acted on "instinct," with no direction. "It was like we did not have a boss."[257]

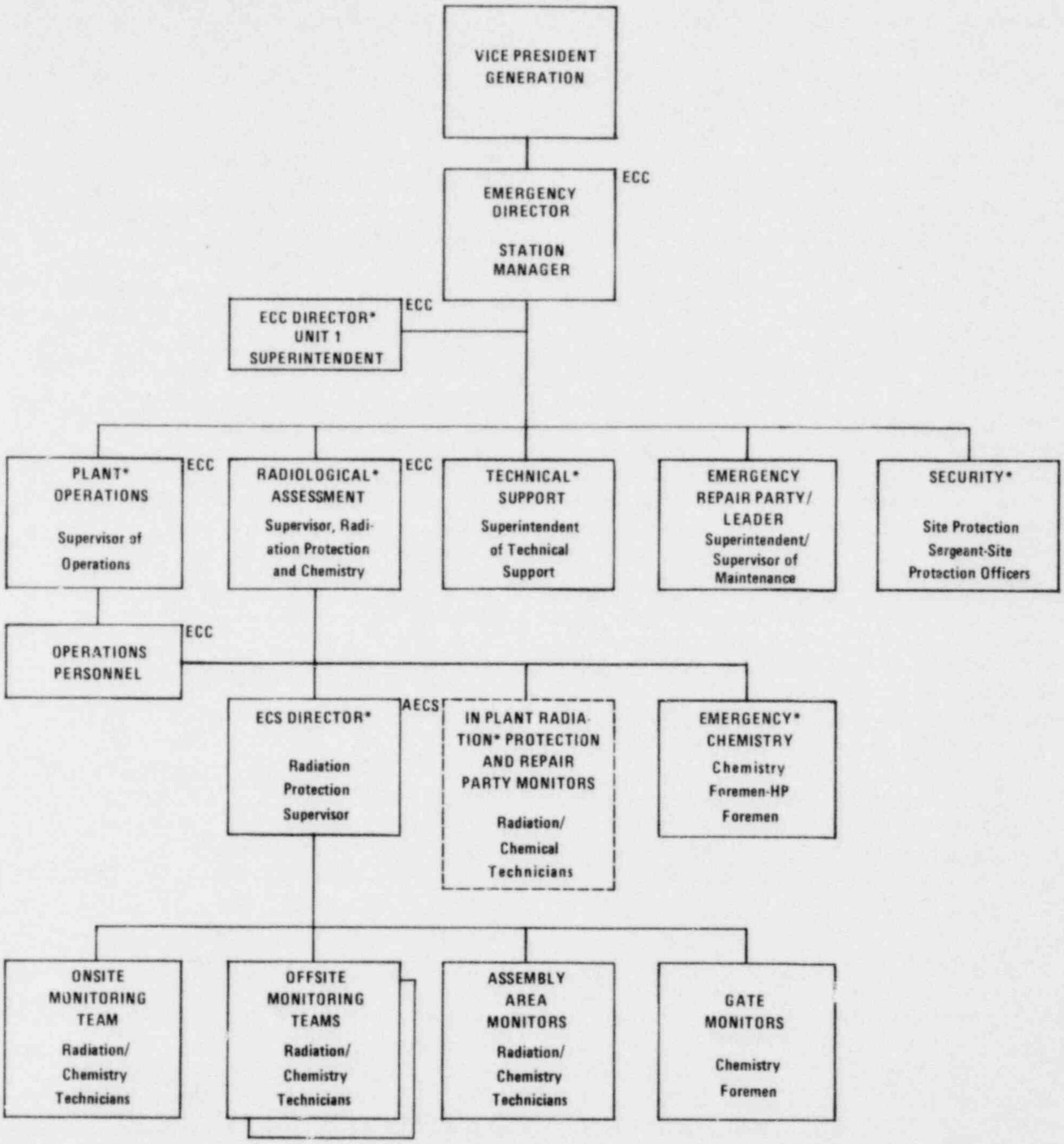
There was, thus, a lack of management control of the inplant radiation protection function early in the accident. During the first several days of the accident, the radiation protection program was seriously compromised: (1) the radiation work permit (RWP) procedure was not used, (2) no logs of entries into the auxiliary building were maintained, (3) no accumulation system of pocket dosimeter dose measurements was utilized for entries into high radiation areas, (4) records of dose rate surveys in the auxiliary building were made at times, but were not maintained, (5) surveys of personnel contamination were made, but in only one instance were records maintained, (6) high radiation areas were not controlled in accordance with Technical Specification 6.12, and (7) no positive control over entries into the auxiliary building was maintained. We find the compromise of the radiation protection program during the accident was unjustified and contributed to the potential for unnecessary and hazardous exposure of plant personnel to radiation.

The reduced emergency radiation protection program was in effect until approximately midnight on March 30, 1979, when the RWP procedure was finally reestablished after continued urging of the NRC inspectors.[258,259] By this time, however, many uncontrolled entries had been made into the auxiliary building without adequate direction or knowledge of previous entries; adequate communications between plant personnel; or adequate radiation protection instrumentation, personnel dosimetry, and radiation protection. These entries occurred while radiation levels in excess of 1000 R/h existed in the auxiliary building. In view of these conditions, the potential for serious radiation injuries to employees making entries into the auxiliary building existed during this time; although, as discussed in Section II.B.4, only three overexposures above regulatory limits were reported. (See Section II.B.4 for further discussion of this point.) (Descriptions characterizing the events and unauthorized entries into the auxiliary building and overexposures experienced during the first 3 days of the accident are contained in Section II.3 of NUREG-0600.)

Summary of Findings and Recommendations

We find that:

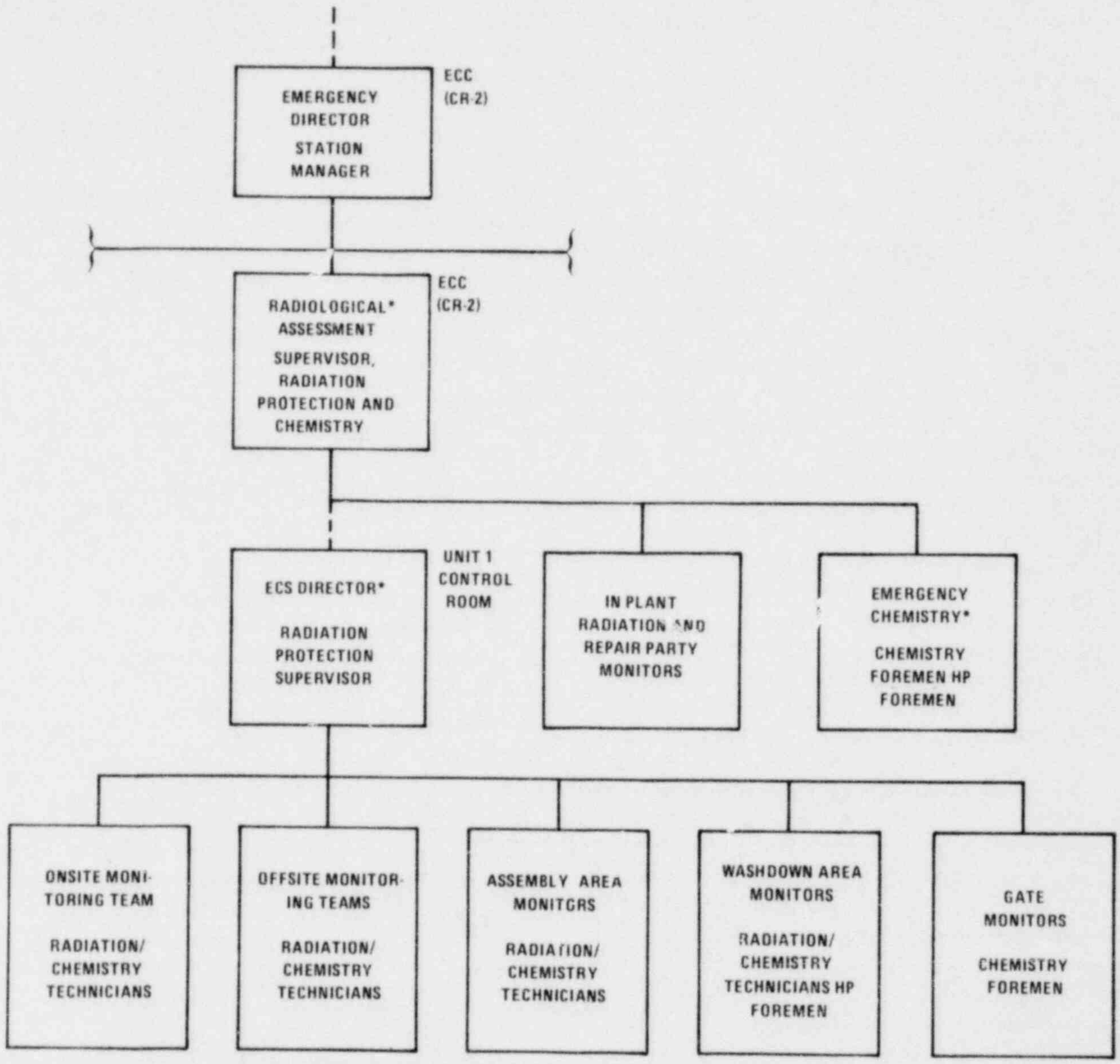
- the performance of the dual functions of radiation protection and chemistry by the R-CTs is detrimental to the efficiency of the radiation protection program;
- Dubiel, the Supervisor of Radiation Protection and Chemistry, did not function as the Radiation Protection Manager (as indicated in Regulatory Guide 1.8), and the role fell, by default, to Mulleavy, who was not qualified for this position;
- Dubiel was not qualified for the position of Chemistry Supervisor;
- the positions of radiation protection foreman and chemistry foreman were not included in the FSAR or the technical specifications and the NRC did not require minimum qualifications for these positions;
- the R-CTs did not develop or maintain adequate job skills and thus did not meet FSAR requirements;
- a serious communications gap existed at all levels of the radiation protection organization and contributed significantly to the deficiencies noted in the radiation protection program;
- a serious conflict existed between operations and radiation protection staffs at all levels of the station's organization and contributed in a major way to the deficiencies noted in the radiation protection program; and



ECC – EMERGENCY CONTROL CENTER – UNIT 2 CONTROL ROOM
AECS – ALTERNATE EMERGENCY CONTROL STATION – UNIT 2 CONTROL ROOM
* – FUNCTIONAL TITLE ADDED

SOURCE: NUREG-0600, FIGURE II-2-4

FIGURE II-20. Emergency Organization for Radiation Protection in Effect on March 28, 1979, 9:00 a.m.-11:00 a.m.



SOURCE: NUREG-0600

ECC

*FUNCTIONAL TITLE ADDED

FIGURE II-21. Emergency Radiation Protection Organization in Effect on March 28, 1979, 11:00 a.m., to March 30, 1979, 12:00 p.m.

- the compromise of the radiation protection program during the accident was unjustified and contributed to the potential for unnecessary and hazardous exposure to radiation.

We recommend that:

- the functions of radiation protection and chemistry be separated and that technicians not be required to perform in both roles;
- the duties of a radiation protection manager should be clearly specified and performed by a qualified individual;
- the NRC require minimum qualifications for the positions of radiation protection foreman and chemistry foreman;
- the technical specifications be amended to include the positions of radiation protection foreman and chemistry foreman;

- technicians be given training adequate to meet FSAR requirements and to develop and maintain adequate job skills; and
- Met Ed take appropriate steps to eliminate the serious communications problems in the radiation protection organization.

Radiation Protection Procedures

The radiation protection program for normal operations at TMI is specified in Station Administrative Procedure 1003, Radiation Protection Manual (Rev. 12, 12/13/77). It is supplemented by station health physics procedures (HPP) that specify the limits; criteria; responsibilities; equipment usage; instrument issue, use, control, and calibration; area posting and control; facilities; dosimetry; and training and emergency procedures. Individual HPPs may apply to the station as a whole or to each unit. The procedures are developed by the radiation protection department and are reviewed by the respective unit plant operating review committee (PORC). If the procedures pertain to both units, then each unit PORC reviews the procedure. Before June 1978, station HPPs were approved by the station superintendent, and since that time the respective unit superintendents have approved the procedures.

Met Ed radiation protection procedures for normal operations were generally adequate but their emergency procedures were inadequate. Generally, during the first several days of the accident, Met Ed did not comply with either the radiation protection procedures for normal operations or the procedures for emergency situations. Those procedures that were used during the accident did not provide adequate radiation protection and contributed to unnecessary personnel exposures.

Radiation Emergency Procedures 1670.1 through 1670.15 are intended to provide guidance during emergencies. However, they do not adequately address inplant radiation hazards or the role of the radiation protection staff during an emergency response. This lack of necessary guidance to cope with emergencies contributed to serious deficiencies in the radiation protection program during the accident, for example:

- the organization of the radiation protection staff deviated from that specified in the emergency plan;
- access control to the auxiliary building was lost, resulting in uncontrolled entries into high radiation areas without proper protection;[260] and
- the radiation protection program was compromised and essential procedures regarding such matters as radiation work permits, completion of survey forms, and maintenance of logs were not complied with during the first few days, resulting in the loss of important data that could have been used for briefing teams prior to entries into high radiation areas, observing trends to detect changes in plant status and ensuring continuity between personnel shift changes.

Met Ed had no written procedures for personnel decontamination on site. The matter was left to the discretion of the radiation protection supervisor.[261] Procedures to be used by Hershey Medical Center in cases of medical emergencies involving personnel contamination were set forth in the emergency plan as station HPP 1670.10. However, personnel contamination on site without any medical problem is more probable than contamination that requires medical attention. Onsite decontamination can be more timely and more effective, but it needs to be performed properly and by knowledgeable personnel using approved procedures. We find that Three Mile Island Station did not have adequate procedures for onsite personnel decontamination.

The onsite personnel decontamination facilities were used early in the accident. At 9:10 a.m., the airborne radiation levels became too high in the TMI-1 health physics area and the use of these facilities was lost. The TMI-2 decontamination area also was lost very early due to high radiation levels.

The island was evacuated of nonessential personnel at 11:10 a.m. on March 28, and a personnel release point was established at the north security gate. The south gate was closed. To prevent crowding at the north gate, a personnel release point was established at the 500-kV substation. Personnel and vehicles leaving the island were directed to the substation for monitoring and decontamination, but no controls were established to ensure that all vehicles and personnel went there.

There had been no prior preparations for using the 500-kV substation as an alternative release point. A limited amount of equipment consisting of paper coveralls, plastic boots, rags, and some contamination-monitoring portable survey instruments (RM-14-HP-210 probe) was taken to the substation. It was first thought that no water supply was available at this location; however, a water supply was located later on the first day. No shower facilities or appropriate wastewater holdup capability were available and little decontamination was actually performed.

The personnel contamination detected at the substation consisted primarily of xenon that tended to be adsorbed on clothing, particularly polyesters, and on hair. Personnel were frisked, and if count rates in excess of 200 cpm were detected,[262] their clothing was exchanged for paper coveralls. If excessive contamination remained, the individuals were detained until the xenon could dissipate, usually a matter of minutes to several hours. Levels of up to 10 000 cpm were observed.

With few exceptions, the personnel performing frisking of personnel and vehicles were inexperienced, and without any written procedures to follow, their surveys were undisciplined.[263] No thyroid surveys were made; no records or followups of any personnel decontamination were maintained; and no bioassay samples were taken.[263,264]

On March 29, an R-CT established a decontamination area in the men's room of the observation center. Because there were no written procedures, radioactive wastewater was flushed down the sink, contaminated clothing was "laying around," and no logs or records were maintained.[265]

Within the plant, personnel decontamination also went unsupervised and was performed by the individuals themselves, primarily by showering, and in some cases, decontamination was performed in unauthorized facilities. No documentation of personnel contamination was maintained.

Met Ed did have emergency procedures for offsite vehicle and equipment decontamination; however, they were not employed during this emergency. Decontamination of offsite vehicles and equipment was to be performed at specific locations on Route 441, north and south of the plant.[260] These preplanned areas were not used in the emergency response. Equipment, supplies, facilities, and radioactive wastewater holdup capabilities were not available at these locations. During the emergency, vehicle monitoring was performed at the 500-kV substation. Decontamination consisted of keeping the vehicles parked until the radioactive noble gases decayed and dissipated.

Finally, there were no emergency procedures for collection of primary coolant samples that contained significant quantities of radioactive materials. Therefore, ad hoc procedures were developed for the initial sampling without adequate consideration of the high exposure rates. As a result, the initial primary coolant sample was taken without appropriate dosimetry and instrumentation, and overexposures of some personnel were experienced.[266]

Summary of Findings and Recommendations

We find that:

- the emergency plan radiation protection procedures did not adequately address the role of the radiation protection staff or adequately account for inplant radiation hazards during the emergency response; and
- there were no procedures for onsite personnel decontamination.

We recommend that:

- emergency plans provide for radiation protection staff response to inplant radiation hazards; and
- radiation protection procedures should be followed during emergencies, and appropriate documentation should be maintained.

Training

The level of training in radiation protection that nuclear powerplant personnel need varies in relationship to the degree of association each person has with radiation work and radiation areas. Basic training may be adequate for nonradiation workers, for temporary personnel working outside restricted areas, and for workers who will be on site for only a few days. A higher level of training is necessary for people working in radiation areas and in the control room. Radiation protection technicians obviously should be comprehensively trained. In addition to the normal complement of workers at a power station, others such as the local fire, police, medical, and civil defense groups need training. Another group of individuals who need to be trained or whose current knowledge of radiation protection should be ensured, are contract health physics technicians, commonly referred to as "rent-a-techs."

The TMI-2 FSAR, Section 12, and Technical Specification 6.4 require that a training and retraining program for the unit staff be conducted and that such training meet or exceed the requirements of Section 5.5 of ANSI N18.1-1971. The NRC staff found this to be acceptable.[267]

The training department at Three Mile Island Station had no substantive responsibility for radiation protection training.[268] The responsibility for the development and implementation of the radiation protection training and retraining program was vested in Richard Dubiel, the Supervisor of Radiation Protection and Chemistry. He, in turn, delegated the training coordination and maintenance of training records to Peter Velez, a Radiation Protection Foreman. Radiation protection training was provided in three categories—nonradiation protection personnel, radiation protection personnel, and emergency response personnel from surrounding communities. The course titles and target groups of the program are shown in Table II-33.

Radiation protection personnel were dissatisfied with the quality and extent of the radiation protec-

TABLE II-33. Radiation protection training program Three Mile Island Station[259]

Program Title	Personnel Receiving Training
1. Basic I	Temporary personnel on site less than 1 day.
2. Basic II (1 hour)	Temporary personnel working outside restricted areas.
3. Basic III (3 hours)	Permanent personnel working outside restricted areas. Temporary personnel in restricted areas for more than 1 day.
4. Intermediate I (3 hours)	All radiation workers. All personnel under radiation work permits (RWP).
5. Intermediate II (8 hours)	Maintenance personnel, engineers, supervisors, others requiring radiation work permit clearance.
6. Advanced Radiation Protection (2 weeks)	Auxiliary operators, control room operators, senior licensed operators.
7. Comprehensive Radiation Protection (3 months)	Radiation/chemistry technicians.
8. General Employee Training (No time specified)	Selected temporary personnel (all permanent personnel once per year).
9. Training for local fire, police and civil defense departments.	

tion training they had received in the past few years.[270,271] For example, some R-CTs requested training in the use of SAM-2, an instrument intended for emergency radiation monitoring, but did not receive such training. These individuals later learned that their training records indicated that they did receive the training. When the R-CTs brought this discrepancy to the management's attention, they were given training on the responsibilities of radiological monitoring teams, but not on the use of SAM-2.[272] The R-CTs' continued lack of familiarity with the SAM-2 became apparent during the emergency, when the R-CTs could not properly operate the instrument, lost confidence in it, and then abandoned it.

Dubiel and Mulleavy acknowledged that there was no formal retraining program for themselves or their foremen. Dubiel further indicated that little time was available to give the training and little time was available to R-CTs to be formally trained because of a heavy workload and insufficient staffing.[273] The R-CTs were working on a 6-week rotational schedule, in which the sixth week was to be set aside for training-retraining. However, understaffing precluded any significant technical training for at least 1½ years before the accident, and the training that was to be given in the sixth week was abandoned in an attempt to keep up with vital health physics functions.[274]

In addition, notification of the R-CTs of changes in station procedures, one of the ANSI N18.1-1971 retraining requirements, was ineffective. Notification of procedural changes was made to the radiation protection staff by placing a note on the bulletin board indicating a change. There was no formal mechanism for ascertaining whether the R-CTs read the change.[272,275]

Radiation protection coverage during refueling, maintenance, and other outages required the use of "rent-a-techs" to supplement the station radiation protection staff. "Rent-a-techs" are not recognized by the NRC as a group separate from regular licensee employees. Because "rent-a-techs" are, by definition, temporary employees, they often do not have the same level of knowledge of the plant and station procedures as the permanent staff. Although the NRC regulations do not clearly so require, "rent-a-techs" should be subject to the same training and qualification requirements for the positions and functions that they fill.

As part of the training conducted by Met Ed, paragraph 6.1.2 of the emergency plan requires the conduct of an annual site or general emergency drill. Station Health Physics Procedure 1670.9, "Emergency Training and Emergency Exercise," provides for planning, conducting and documenting drills. During calendar year 1978, Met Ed conducted seven radiation emergency drills that satisfied the site or general emergency drill requirements. These seven drills were conducted between October 23 and November 8, 1978, and were, in effect, dress rehearsals for the one that would be observed by the NRC. The November 8 annual drill was observed by the NRC. No other drills which satisfied the TMI emergency plan requirements were conducted during 1978.

Although spacing these drills throughout the year may have been more effective, in retrospect, it was perhaps fortuitous that an intensive set of drills was conducted so near the time of the accident. Participation in the drills had a constructive effect on the conduct of the emergency response. For example, the relocation of the Emergency Control Station, a drill scenario, occurred during the accident. The relocation went more smoothly than it probably would have if there had been no drills. At the end of each drill, a formal critique of the drill was held for all participants, observers, and staff. However, participation in the drill critiques by all participants was not required. Even though overtime was authorized for R-CTs to attend the critiques, many did not attend.[276-278] We believe that participation of all personnel in drill critiques is necessary for proper evaluation of the plant's performance.

We find that there was no adequate radiation protection training or retraining program in effect prior to the accident, that radiation protection functions were performed, on occasion, by personnel not adequately trained in radiation protection, and that the NRC has never clearly specified the training and qualification requirements of "rent-a-techs" in the health physics area.

The NRC inspected the radiological aspects of Met Ed's program a number of times during the past 2 years, but we found no evidence that the inspections detected the deteriorated condition of the radiological training program. The IE's inspections for radiation protection training are usually limited to an audit of the training records and course outlines, and do not include any attempt to determine the competence of the instructors or the trainees.[279] Part of this can be attributed to the fact that NRC has never established standards for the evaluation of radiation protection personnel or training programs. Although the training records indicated that the R-CTs were given at least the minimum number of hours required, the R-CTs performance during the accident demonstrated the failure of the training program.

We find that IE's inspection procedures for radiation protection training programs are inadequate because the results of the programs, i.e., the expertise of the students, are not evaluated; and that the NRC has not established standards for the evaluation of the training or retraining of the radiation protection personnel.

We find also that there is need to consider the feasibility and advisability of licensing or certifying radiation protection personnel. We note that licensing or certification of all radiation protection personnel at commercial nuclear power reactors has been suggested to deal with this situation. Currently, the NRC has before it a petition, PRM 20-13, submitted on January 24, 1979, which calls for NRC certification of all health physics personnel.

Summary of Findings and Recommendations

We find that:

- there was no adequate radiation protection training or retraining program in effect prior to the accident;
- the IE's inspection procedures for radiation protection training program are inadequate because the results of the programs, i.e., the expertise of the students, are not evaluated;
- the NRC has not established standards for the evaluation of the training or retraining of the radiation protection personnel;
- radiation protection functions were performed on occasions by personnel not adequately trained in radiation protection;
- the NRC has never clearly specified training and qualification requirements of "rent-a-techs" in the health physics area; and
- there is a need to consider the feasibility and advisability of licensing or certifying radiation protection personnel.

We recommend that:

- the NRC require the implementation of an adequate radiation protection training program by establishing standards for licensee radiation protection programs and for competency of licensee radiation protection personnel;
- the NRC inspect for actual competence of the trainees and trainers in addition to auditing records of training; the NRC and the licensee review radiation protection staffing and organization

to assure that radiation protection functions are fulfilled by adequately trained personnel;

- the NRC develop guidance regarding the specific use and training of "rent-a-techs" at licensed facilities;
- the NRC appoint a group of experts to examine the feasibility and advisability of licensing or certifying radiation protection personnel at commercial nuclear power reactors. The report of this group should be submitted within 6 months of its appointment. Options to be examined include:
  - — licensing by the NRC.
  - — licensing by a Government agency other than the NRC.
  - — requiring certification by the American Board of Health Physics (ABHP) for specified functions (e.g., radiation protection manager).
  - — requiring certification by another nongovernmental body; and
- the NRC defer action on a petition (PRM 20-13) presently pending before the Commission, which requests that radiation protection personnel at all levels in licensed activities be certified by the Commission until the aforementioned study is completed.

Inplant Monitoring and Instrumentation

The inplant area monitoring program is described in the FSARs for TMI-1 and TMI-2 [280,281] and specified in the health physics procedures for each unit. The NRC staff reviewed this program and found the area monitoring system to be acceptable.[282]

Fixed Area Gamma Radiation Monitoring System—The area gamma radiation monitoring system was designed to function separately for each unit, with readouts in each control room. The system provides operators with indications and records of radiation levels at each monitored point. It provides both audible and visual alarms in the control room and local audible and visual alarms at those monitors located in areas where high radiation levels may constitute a hazard. Each channel consists of a detector located at a predesignated, fixed location, a local indicator, a power supply, a control room readout module with alarms and alarm setpoint adjustment. Each channel is recordable by one point of a multipoint recorder.[283] During the accident, the fixed area gamma radiation monitoring systems for each unit were used.

The area gamma monitoring system for TMI-1 consists of 15 channels equipped with ionization chamber detectors as shown in Table II-34. All channels have a range of 0.1 mR/h to 10 R/h, except the TMI-1 reactor building dome monitor (RM-G8) which has a range of 1 R/h to 1×10^6 R/h.[283] All TMI-1 area gamma monitors were operational at the time of the accident.[284]

The area gamma monitoring system for TMI-2 consists of 21 monitors as shown in Table II-35. All TMI-2 area monitors, with the exception of the reactor building dome area monitor (HP-R-214), are

TABLE II-34. Area radiation monitors, TMI-1[285]

Channel Tag. No.	Location: Area	Location: Building
RM-G1	Control Room	Reactor
RM-G2	Radiochem Lab, Elev. 305 ft	Auxiliary
RM-G3	Sampling Room, Elev. 325 ft	Auxiliary
RM-G4	Hot Machine Shop	Auxiliary
RM-G5	Reactor Bldg. Personnel Access	Reactor
RM-G6	Refueling Bridge 1	Fuel Handling
RM-G7	Refueling Bridge 2	Fuel Handling
RM-G8	Reactor Bldg. High Range	Reactor
RM-G9	Fuel Handling Bridge	Fuel Handling
RM-G10	Bldg. Entrance, Elev. 305 ft	Auxiliary
RM-G11	Near Waste Tank, Elev. 305 ft	Auxiliary
RM-G12	Drumming Area	Auxiliary
RM-G13	Building Entrance, Elev. 281 ft	Auxiliary
RM-G14	Near Waste Evap.	Auxiliary
RM-G15	Heat Exchange Vault, Elev. 271 ft	Auxiliary

equipped with Geiger–Muller (G–M) tube type detectors and have a range of 0.1 to 10^4 mR/h. The reactor building dome monitor is an ionization chamber with a range of 1 to 10^6 R/h[287] and is contained in a 2-inch-thick lead shield,[218] having an approximately 1/8-inch hole through it.[288]

After the accident, the containment atmosphere was severely contaminated with radionuclides. It is possible that some of the contaminants penetrated the hole and deposited on the surfaces of the detector. Therefore, we find that the readings obtained on HP-R-214 could not be considered reliable indicators of the radiation fields within the containment structure.

Fixed Atmospheric Air Monitoring System— Each reactor has a separate fixed atmospheric monitoring system. There are 10 monitors in TMI-1 and 15 monitors in TMI-2.

The atmospheric monitors for TMI-1 are described in Table II-36. At the time of the accident in TMI-2, the TMI-1 radiochemistry laboratory and nuclear sampling room air monitor (RM-A12) and the TMI-1 control room air intake monitor (RM-A1) were inoperable, and had been inoperable since April 22, 1977 and April 18, 1978, respectively.[218]

The atmospheric monitors for TMI-2 (described in Table II-37) use isokinetic sample probes and have a particulate filter, a charcoal cartridge for iodine detection, and a detector in a shielded volume for gas monitoring.[290] At the time of the accident, the TMI-2 waste gas decay tank 1A gas monitor (WDG-R-1485) was inoperable, and had been inoperable since February 16, 1979.[218]

As a result of the accident, radioactive materials were released into the atmosphere of the auxiliary building. The resulting radiation levels exceeded the response capabilities of many of the atmospheric monitors. The noble gas channel of the stack monitor (HP-R-219) went off scale before 8:00 a.m. on March 28, eliminating the only direct means of assessing the quantities and rate of release of radioactive material from the plant. This information was vital to the evaluation of offsite consequences.

TABLE II-35. Area radiation monitors, TMI-2[286]

Channel Tag. No.	Location	
	Area	Building
HP-R-213	Incore Instrm. Panel Area	Reactor
HP-R-214	Reactor Building Dome, Elev. 418 ft	Reactor
HP-R-215	Fuel Handling Bridge	Fuel Handling
HP-R-218	Waste Disposal Storage Area	Fuel Handling
HP-R-231	Aux. Bldg. Sump Tank Filter Room Elev. 280 ft	Auxiliary
HP-R-232	Access Corr. Elev. 305 ft	Auxiliary
HP-R-233	Access Corr. Elev. 305 ft	Auxiliary
HP-R-234	Access Corr. Elev. 305 ft	Control & Service
HP-R-3236	Reactor Building Purge Unit Area Elev. 328 ft	Auxiliary
HP-R-3238	Aux. Bldg. Exhaust Unit Area	Auxiliary
HP-R-3240	Fuel Bldg. Exhaust Unit Area	Auxiliary
HP-R-201	Control Room	Control
HP-R-202	Cable Room	Control
HP-R-204	Reactor Building Emerg. Cooling Booster Pump Area	Auxiliary
HP-R-205	Reactor Coolant Evap. Control Panel Area	Auxiliary
HP-R-206	Makeup Tank Area	Auxiliary
HP-R-207	Intermediate Cooling Pump Area	Auxiliary
HP-R-209	Fuel Handling Bridge North	Reactor
HP-R-210	Fuel Handling Bridge South	Reactor
HP-R-211	Personnel Access Hatch	Reactor

TABLE II-36. Atmospheric monitoring system, TMI-1[289]

Channel	Location	Type of Monitor
RM-A1	Control tower intake	Particulate Gas Iodine
RM-A2	Reactor Bldg. Air Sample Line	Particulate Gas Iodine
RM-A4	Fuel Handling Bldg. Exhaust Ventilation Ducts	Particulate Gas Iodine
RM-A6	Aux. Bldg. Exhaust Ventilation Ducts	Particulate Gas Iodine
RM-A5	Condenser Vacuum Pump Exhaust	Gas Iodine
RM-A7	Waste Gas Discharge	Gas Iodine
RM-A8	Aux. & Fuel Handling Bldg. Exhaust	Particulate Gas Iodine
RM-A9	Reactor Bldg. Exhaust	Particulate Gas Iodine
RM-A12	Radiological Lab Monitor-movable	Particulate Gas Iodine
RM-A13	Spent Fuel Area Monitor-mobile	Particulate Gas Iodine

The absence of this information resulted in the use of indirect means of evaluation that were untimely and inaccurate.

The NRC staff's assessment of the adequacy of the atmospheric monitoring capability of TMI-2 did not consider the air contamination levels that could result from the degree of core damage experienced in the accident. We find that atmospheric monitors were inadequate to measure the quantities of radioactive materials released, that critical information was lost as a result, and that the NRC staff's SER assessment of the proposed atmospheric monitoring system for TMI-2 was inadequate because it did not consider the monitor response ranges in the presence of high radiation background.

Liquid Effluent Monitoring System—Each unit has a liquid effluent monitoring system. The indicators, alarms and recorders are located in the control room of each unit.

The liquid effluent monitoring system for TMI-1 consists of nine monitors. Five of these monitors are used for monitoring closed cooling loops that act as barriers against release of radioactive materials to the river. The primary coolant letdown is monitored to detect defects in fuel cladding.[292] The wastewater and the miscellaneous sump discharge monitors are located prior to the dilution point.[292] Should a preset level be reached, the wastewater monitor will automatically close the discharge valve.[292] The plant effluent discharge monitor was not in service during the accident and had been inoperable since March 13, 1979. The backup monitor for the plant effluent discharge had been out of service since April 22, 1977.[218]

TMI-2 liquid effluent monitors are equipped with sodium iodide (NaI) scintillation detectors and are listed in Table II-38. MU-R-720 primary coolant letdown (failed fuel monitor) monitors reactor coolant letdown upstream of the purification demineralizers. The output from the detector in this monitor is fed to two channels, one measuring the gross gamma activity and the other measuring ^{135}I activity.

One instrument in the spent-fuel cooling circuit

TABLE II-37. Atmospheric monitoring system, TMI-2[291]

Channel	Location	Type of Monitor
HP-R-219	Station Vent	Particulate Iodine Gas
HP-R-221A	Fuel Handling Exhaust Duct Upstream of Filter	Particulate
HP-R-221B	Fuel Handling Exhaust Duct Downstream of Filter	Particulate
HP-R-229	Hydrogen Purge	
HP-R-225	Reactor Bldg. Purge Air Exhaust Ducts A&B	Iodine
HP-R-222	Aux. Bldg. Purge Air Exhaust Upstream of Filter	
HP-R-228	Aux. Bldg. Purge Air Exhaust Downstream of Filter	
HP-R-227	Reactor Bldg. Air Sample	Gas
HP-R-220	Control Room	Particulate
HP-R-224	Movable Monitor	Iodine
HDG-R-1480	Waste Gas Discharge Duct	Gas
WDG-R-1485	Waste Gas Tank WDG-T-1A Discharge	Gas
WDG-R-1486	Waste Gas Tank WDG-T-1B Discharge	Gas
VA-R-748	Condenser Vacuum Pump Discharge	Gas

(SF-R-3402) continuously monitors radioactive materials released in the spent-fuel storage pool. The monitor is an offline sampler. The detection of radiation indicates possible leakage of radioactive materials from stored spent fuel.[294]

One radiation monitor (WDL-R-1311) continuously measures the radioactivity level in the plant discharge line at a point upstream from the discharge dilution point for the mechanical-draft cooling tower. Should the preset radioactivity level be reached, the monitor will initiate closing of liquid radwaste discharge valves and stop the evaporator condensate pumps. In addition, an electrical interlock is provided that precludes the simultaneous discharge of liquid waste from TMI-1 and TMI-2. One additional radiation detector continuously monitors the common plant effluent from TMI-1 and TMI-2 to the river. This monitor is a backup to WDL-R-1311.[294]

Portable Radiation Survey Instruments—The portable radiation survey instruments for TMI-2 were described by type but not by number in the FSAR.

TABLE II-38. Liquid effluent and process radiation monitors, TMI-2[293]

Channel	Location
MU-R-720	Primary Coolant Letdown
IC-R-1091	Intermediate Closed Cooling Water
IC-R-1092	Intermediate Closed Cooling Water (Letdown Cooler)
IC-R-1093	Intermediate Closed Cooling Water (outside of Reactor Bldg.)
WDL-R-1311	Liquid Waste Effluent from Unit 2
DC-R-3399	Decay Heat Closed Cooling Water
DC-R-3400	Decay Heat Closed Cooling Water
NS-R-3401	Nuclear Service Closed Cooling Water
SF-R-3402	Spent Fuel Cooling Water

The NRC staff concluded in the SER that the TMI-2 portable radiation survey instrumentation was acceptable.[295]

Portable radiation survey instruments are maintained under the responsibility of radiation protection personnel. Met Ed's program requires an inventory of instruments for measuring alpha, beta, gamma, and neutron radiation. The inventory must be adequate to allow for periodic calibration, maintenance, and repair. The portable radiation survey instruments available at TMI immediately before the accident are listed in Table II-39. It is doubtful that the number of instruments in inventory would have been adequate to support normal operations, even if all of them had been operational. As Table II-39 indicates, less than 50% of the instruments were operational at the time of the accident. There were no instruments available to detect neutrons. We find that the number of portable radiation survey instruments that were available at the time of the accident was grossly inadequate to support normal, and certainly not emergency, operation.

Existence of a backlog of portable radiation survey instruments in the TMI repair shop awaiting repair, parts, or calibration was a common occurrence stemming from the low priority the radiation protection program received.[299] In addition, excessive damage to instruments occurred through employee neglect, carelessness, and absence of accountability. For example, one portable survey instrument had been inadvertently crushed by the radwaste trash compactor.[300] The outage at TMI-1 that immediately preceded the accident further depleted the inventory of available instruments and exacerbated the problem. We find that the management of the portable radiation survey instrumentation program at Three Mile Island Station was inadequate.

Met Ed relied upon the station's normal complement of portable radiation survey instruments to support an emergency. For this reason, only four emergency kits were maintained for postaccident monitoring. Each of the kits contained a SAM-2 intended to measure ^{131}I. The SAM-2 in one kit was out of service at the time of the accident and the

TABLE II-39. Portable radiation detection survey instruments at Three Mile Island[296-298]

Instrument	Radiation Detected	Type	Range	Available/Inventory At Time of Accident
Eberline E-520	Beta Gamma	GM	0–2000 mR/h	6/14
Eberline PAC-4S	Alpha	Scint	0–2×10^6 cpm	0/2
Eberline Teletector #6112	Beta Gamma	GM	0–1000 R/h	4/16
Eberline PBR-4	Neutrons	BF_3-PC	0–5000 mrem/h	0/2
Eberline PIC-6A	Beta Gamma	IC	0–1000 R/h	4/14
Eberline RO-2	Beta Gamma	IC	0–5 R/h	5/12
Eberline RM-14 with HP-210 probe	Beta Gamma	GM	0–5×10^4 cpm	18/18
Victoreen 808 Vamp	Gamma	GM	0–100 mR/h	0/5

GM = Geiger Muller
Scint = Scintillator
IC = Ionization Chamber
BF_3-PC = Boron Trifloride Proportional Counter in Cadmium/Polyethylene Ball

SAM-2 in another kit was issued even though it failed its calibration check.[301] In any event, the SAM-2 was a poor choice for field use by technicians to measure ^{131}I, particularly in the presence of noble gas.

In short, there was not a sufficient supply of instruments to perform personnel monitoring adequately and to conduct onsite and offsite radiation surveys in response to the accident. Large numbers of portable survey instruments of all types had to be provided by vendors, other utilities, contractors, and Government agencies to augment the available inventory.

We find Met Ed's reliance on the routine inventory of radiation protection survey instruments for emergency response left a serious gap in its ability to support the necessary monitoring functions during the initial phases of the emergency. We recognize that augmentation of portable survey instruments is inevitable in responding to accidents of the nature and duration of the TMI-2 accident. In addition, we find that the number of emergency kits and the suitability of the instruments therein was inadequate.

Radiation Counting Laboratory Instrumentation—Because the TMI-2 Ge-Li gamma spectrometer and the liquid scintillation detector system were never placed in operation and the TMI-1 counting laboratory could not be used in the early stages of the accident, replacement analytical services were necessary.

Portable Air Sampling Equipment—In addition to the fixed and mobile atmospheric monitors, Met Ed also utilized several portable air samplers. On March 28, 1979, 21 of the 24 air samplers in inventory were operational.[302]

Summary of Findings and Recommendations

We find that:

- the NRC staff's SER assessment of the proposed atmospheric monitoring system for TMI-2 was inadequate because it did not consider the monitor response ranges that would be necessary in the presence of high levels of radioactive materials;
- readings obtained from the TMI-2 dome monitor (HP-R-214) during the accident could not be considered reliable indicators of the radiation fields within containment;
- the atmospheric monitors were inadequate to measure the quantities of radioactive materials released and as a result, critical information was lost;
- the number of portable radiation survey instruments that were available at the time of the accident was grossly inadequate to support normal, and certainly not emergency, operations;
- the management of the portable radiation survey instrumentation program during normal operations was inadequate;
- Met Ed's reliance on routine inventory of radiation protection survey instruments for emergency response left a serious gap in its ability to support its emergency response; and
- the number of emergency kits and the suitability of the instruments therein was inadequate.

We recommend that:

- the NRC reassess the requirements for inplant fixed radiation monitoring instruments to ensure that the instruments will be adequate for the radiation and contamination levels that could be expected to exist during an accident; and
- the NRC evaluate and specify requirements for type, quality, and quantity of operational portable radiation survey instruments for both normal and accident conditions.

Respiratory Protection and Protective Clothing

Respiratory protection is required by 10 C.F.R. Part 20.103. However, the regulations, the FSAR, and the technical specifications do not specify the type, performance, or quantity of respiratory protection devices to be maintained on site.

As of February 1979, Met Ed had the following protective devices available on site:[303]

Self-contained breathing device, routine work: 44
Self-contained breathing device,
emergency egress: 6
Backup air supply bottles for self-contained
breathing devices, routine work: 15
Full-face respirator with particulate filter
cartridge: 150
Half-face respirator with particulate filter: 25

We believe the inventory was adequate for normal operations.

The only air supply refill capability at Three Mile Island Station for the self-contained devices was available in TMI-1. Due to high levels of airborne radioactive materials, this capability could not be used during the initial phases of the emergency because the quality of the intake air could not be ascertained.[304]

The maintenance, inspection, and decontamination procedures for respiratory devices are discussed in station procedures HPP-1616. No one person is accountable for these important aspects of the respiratory protection program because of the 6-week rotation schedule. Because respirators are delicate, their care, maintenance, and decontamination (both biological and radiological) after use is important. This control could be more efficient if individuals were assigned to these tasks permanently. We find that control of the respiratory protection program was inadequate during normal operation because responsibility was shared among all the R-CTs as part of the maintenance and inspection functions.

During the first several days of the accident, the issuance, maintenance, inspection, and decontamination of respiratory protection devices went uncontrolled because of lack of accountability and facilities and because of the large number of personnel requiring respiratory protection. Some respirators were reissued without decontamination[305] and some were decontaminated with improper materials, causing subsequent users to become ill.[306] We find that control of the respiratory protection program during the first few days of the accident was inadequate.

Because Met Ed relied upon its normal complement of respiratory devices for emergency use, the respiratory protection program at Three Mile Island Station had to be augmented during the emergency. The need for additional equipment, including Scott Air Pacs, was quickly realized and efforts were initiated to obtain the necessary support, which was promptly provided by industry, vendors, and local fire departments. Iodine adsorption canisters were not available on site before the emergency. A perceived need for these canisters resulted because of the loss of analytical capability, and they were obtained from outside sources. Because there is no approved iodine canister, the canisters obtained for Three Mile Island Station received temporary approval from the NRC for limited use.[307,308] We find that the onsite inventory of respiratory protective devices was insufficient to support a prolonged response to an emergency.

Protective clothing available included shoe covers and head covers (hoods and surgeon caps); gloves (cotton, plastic, rubber); coveralls; laboratory coats; plastic or rubber suits; and face shields. Adequate supplies of protective clothing were on hand during normal operations and for the initial phases of the emergency. These supplies were substantively augmented during the accident. Although the protective clothing was available, some individuals made entries into contaminated areas without wearing hoods.[309-311]

Summary of Findings and Recommendations

We find that:

- control of the respiratory protection program was inadequate during normal operation because responsibility was shared among all the R-CTs as part of the maintenance and inspection functions;
- control of the respiratory protection program during the first few days of the accident was inadequate; and
- the onsite inventory of respiratory protection devices was insufficient to support a prolonged response to an emergency.

We recommend that:

- the NRC require that the responsibility for the respiratory protection program be vested in a single individual and that technicians be permanently assigned to perform the tasks of inspection, maintenance, and decontamination of respiratory protection equipment;
- the NRC specify the minimum number of functional respiratory protection devices required by type and size for both normal operations and for emergencies.

Personnel Dosimetry

Personnel dosimetry is used to assess the efficacy of maintaining the external and internal exposures received by the plant workers ALARA. Personnel dosimetry is required by 10 C.F.R. Part 20.202.

The personnel dosimetry program used at the Three Mile Island Station is specified in the station health physics procedures. This program is generally described in the FSAR[312] and was evaluated by the NRC staff in its review of the operating license application. The NRC staff found the proposed dosimetry program acceptable and indicated so in the SER.[313] The NRC regulations do not specify minimum standards for management of personnel dosimetry programs.

Personnel dosimetry at the Three Mile Island Station is the responsibility of the radiation protection department and was conducted with the use of TLDs and pocket chambers for determining whole-body exposures. Extremity monitoring was conducted by taping a TLD to the appropriate extremity. Internal dosimetry was based on urinalysis and whole-body counting (WBC).[314]

Preaccident Personnel Dosimetry—The TLD system is operated in accordance with Station Health Physics Procedure 1642, "Operation and Calibration of the Thermoluminescent Dosimetry System," Revision 1, September 28, 1977.

TLDs are issued to all personnel at the station who enter the controlled area. During normal operations they are processed monthly, or more often if exposures are suspected. Harshaw, Inc., provides the dosimetry system, which uses a lithium fluoride (LiF) two-chip dosimeter and Model 2271 reader. In addition to TLDs, self-reading dosimeters (pocket ionization chambers) are issued to individuals as required. Each individual is instructed in the necessity of reading the self-reading dosimeter at frequent intervals while in radiation areas.[312]

Whole-body counts are taken if the nature of exposure or suspected exposure is such that internal contamination is possible. The WBCs also assist in assessing the adequacy of the station radiation protection control practices.

The TMI-2 radiation protection procedures require that all contractor personnel provide a baseline urine sample prior to entry to a controlled area. Urinalyses are also performed on contractor personnel upon completion of a specific job. The radiation protection supervisor has the option of ordering additional bioassay analyses and/or whole-body counts on any personnel should the need arise.[312] Both WBC services and urinalyses were contracted to offsite vendors.

No specific individual had been assigned responsibility for control of external personnel dosimetry at Three Mile Island Station.[315,316] Each R-CT performed this function as a routinely assigned task. Because of the weekly rotation of assignments, a specific R-CT might be expected to read the TLDs for 1 week, twice a year. The expertise gained by a R-CT during his twice-yearly limited duty assignment at the TLD reader was diluted to the point that none of the R-CTs was familiar enough with the equipment and procedures to understand all aspects of the system.[316]

The execution of an acceptable personnel dosimetry program requires specialized training and constant attention to the details of the system. At least two audits recognized this requirement.[315,316] Both audits recommended that a qualified individual be assigned the sole responsibility for the dosimetry program. No action was taken on these recommendations until after the accident, when a specialist was hired to supervise the dosimetry program.

Pocket chambers for each individual entering a controlled area were issued to the foreman in charge of the ongoing work and the foreman provided them to the individuals. Under these circumstances, individuals were not accountable for their pocket chambers. As a result, during the TMI-1 refueling outage preceding the accident, several hundred pocket chambers were lost over a 3-month period. Upon leaving the TMI-1 controlled area, the individual workers read their dosimeters, informed the R-CT (controlling access) of the reading, and then proceeded to another location where they were supposed to turn in the pocket chambers. Often the workers did not follow this procedure and kept the dosimeters. Because there was no personal accountability for each pocket chamber, there was no way to recover them and the data regarding their performance were lost. After the loss of approximately 600 pocket chambers, the procedures were changed. No personal accountability program resulted from this change, and another 200 pocket chambers were lost over the remaining 2 months of the outage. The lack of control over the pocket chambers was attributed to insufficient personnel and inadequate funding.[317] However, the loss of 800 pocket chambers, each costing an average of $50, would have made it well worthwhile to have had a control mechanism.

We find that the management and implementation of the external personnel dosimetry program (TLDs and pocket chambers) were inadequate during normal operation. We find, further, that the NRC has not adequately addressed standards for management and control of a personnel dosimetry program.

Personnel Dosimetry During the Accident—On the afternoon of March 29, the TLD reader and support equipment were moved from their normal location on site to the observation center because of a significant increase in the onsite background activity. No record is available to verify that proper calibration had been performed prior to placement of this equipment into operation.

The TLD system was operated until March 31 by an R-CT who had received only 2 hours of on-the-job training in the use of the TLD system on June 6, 1977. He had not operated the equipment in about a year and a half. The operator did not have a copy of the operating or documenting procedures. He performed the job for approximately 48 continuous hours with little or no assistance. His work included the reading of all TLD badges that were turned in and the preparation (zeroing) of the badges to be issued in April.[318]

On March 31, the TLD reader was returned to the site. At this time, Harshaw provided an additional reader and assigned two engineers to assist with its installation and operation. The second reader was operational on April 1.[319]

Additional personnel to operate and manage the personnel dosimetry program during the emergency response were provided by the Electric Boat Division of General Dynamics Corporation and other contractors. With the heavy influx of augmenting personnel, large quantities of additional TLD badges and self-reading dosimeters were rushed to the site by the various suppliers. Extremity badges, which were not available on site prior to the accident, were also provided and were used to monitor personnel who were performing functions with potential for high extremity exposures.

During the early days of the emergency, entries were made into high radiation areas and areas with unknown exposure rates within the auxiliary building. These entries were made without the use of pocket chambers or with the wrong type (low range vs. high range) of pocket chambers. Pocket chambers that went off scale during these entries were ignored. No system was used to assess and record the cumulative individual exposures determined by pocket chambers. We find that the external personnel dosimetry program during the accident was inadequate.

Two whole-body counters were on site for the TMI-1 refueling (from Helgeson Nuclear Inc. and from Radiation Management Corporation). Additional technicians were also provided to assist in whole-body counting of plant personnel who were, or may have been, contaminated because of the accident.

Individuals with gross skin contamination resulting from activities such as sampling primary coolant and maintenance were sent to the WBCs for counts. In some instances these individuals were so contaminated that the radiation emanating from them saturated the WBC equipment.[320] There was no mechanism in effect to ensure that individuals who were directed to obtain WBCs ever did so. Referral records were not kept. We find that the use of whole-body counting for internal personnel dosimetry during the emergency was inadequate.

Summary of Findings and Recommendations

We find that:

- the management and implementation of the external personnel dosimetry program (TLDs and pocket chambers) were inadequate during normal operation;
- the NRC has not adequately established standards for management and control of a personnel dosimetry program;
- the performance of the external personnel dosimetry program during the accident was inadequate; and
- the use of whole-body counting for internal personnel dosimetry during the emergency was inadequate.

We recommend that:

- Met Ed establish an improved system for control, issuance, and recovery of personnel dosimeters;
- Met Ed ensure that their personnel dosimetry program is managed and implemented by competent personnel; and
- the NRC require licensees to ensure that adequate personnel dosimetry services, including sufficient staff, be available and that personnel dosimetry records, evaluations, and referrals for bioassay be maintained during emergencies.

c. The Responsibility of the Utility and the NRC

The deficiencies in the radiation protection program at Three Mile Island were pervasive and serious. The utility was aware of the deficiencies in the program before the accident, but its efforts to improve the program were slow and weak. The NRC was, or should have been, aware of the deficiencies but took no, or trivial, action to remedy the problems.

There appear to have been two reasons why an inadequate radiation protection program existed. First, the attitude of both Met Ed and the NRC was that the program was of secondary importance and, accordingly, warranted much less attention than the operational aspects and hardware. Second, both Met Ed and the NRC shared the assumption that an accident like TMI-2 would not occur because engineered safety features incorporated in the design would prevent or mitigate any serious accident. Under anticipated conditions, they believed the existing radiation protection program would provide sufficient protection. The accident has shown that the attitude of Met Ed and the NRC was not proper and that radiation protection must be given higher priority that is on an equal level with operations.

Audits by the Utility

Met Ed's radiation protection program had been audited before the accident by Met Ed's quality assurance department, by GPU, and by outside consultants. These audits identified numerous, and serious, deficiencies in the program and made recommendations for correcting them. The management did not take timely corrective actions to implement these recommendations.

Internal Audits—Met Ed's internal audits were conducted according to a specific audit plan based on procedures, regulatory guides, and applicable regulations, and contained a checklist of specific attributes to be audited within the subject area of interest. Upon completion of the audit, a report was issued to the station manager, the supervisor of radiation protection and chemistry, and others. The report established time limits for satisfactory completion of any required corrective action. The internal audit program provided for a mechanism to enforce the deadline for corrective action. Extensions of the deadlines, however, were routinely requested and granted, and the mechanism became ineffective.

Corrective actions, even on trivial matters, took months for completion. For example, an audit of the respiratory program was conducted on March 16, 1978. The audit report was issued on April 28, 1978, with seven findings requiring corrective action.[321] None of these findings should have taken very long to resolve; yet the earliest completion date for any item was September 5, 1978. Two items remained outstanding at the time of the accident, almost 1 year after completion of the audit.

> *78-07-5* Maintenance records are not kept to provide knowledge of service time for respirators, common failure modes of particular respirator types, and personnel complaints on respirator design.
>
> *78-07-6* The protection factors in HPP-1616 are inconsistent with the values specified in Regulatory Guide 8.15.

Items of this nature should have been easily resolved since most required simple procedural changes or a letter from the station manager. They should have also been found during IE radiation protection inspections. However, these deficiencies were not reported by IE, even though it made four inspections between January 1978 and March 1979. Action on finding 78-07-6 was not completed at the time of the accident. This finding was considered by Met Ed as "an item of noncompliance with Regulatory Guide 8.15."[321]

GPU Audit of Three Mile Island Radiation Protection Program— In June 1977, GPU and Pickard, Lowe and Garrick, Inc., a consultant, conducted an audit of the radiation protection program at Met Ed's request.[322]

Their report noted the following deficiencies:

1. "The high frequency rotation of technicians between chemistry and health physics activities is probably inefficient."
2. "There is a problem with combining the chemistry and health physics functions. Chemistry is closely related to the reliabilities of plant operation, whereas *Health Physics is more of a conscience function*." (Emphasis added)
3. "The Health Physics department does not review everything that goes through PORC."
4. "The present 24-man technical staff is probably marginal for routine operation of Unit 1 and 2 (not counting outage considerations), and is probably slightly (1 or 2 men) inadequate for Unit 2 startup, but could be satisfied by overtime."
5. "Unit 2 has already made an impact on Health Physics/Chemistry technical activities and overtime is currently required even with 24 technicians."
6. "Closer supervision of Health Physics/Chemistry technicians by foremen is desirable."
7. "TMI is one of a growing number of plants performing inhouse external dosimetry (TLD). This dosimetry service, which is a repetitive analysis using specialized equipment and requires substantial data processing, would probably be better performed by one person."

Appropriate recommendations were made to correct deficiencies.[315] However, little was done to rectify the situation, as is shown in a subsequent audit conducted for Met Ed by the NUS 2 years later.

The NUS Audit of the Radiation Protection Program— At Met Ed's request another audit was conducted from February 26 to March 2, 1979 by the NUS Corporation.[323] The audit was intended to provide an overview of the health physics program and not an indepth review. Even on the basis of this limited review, the NUS was highly, and correctly, critical of the program. Among the findings made by the NUS were the following:

1. The present organization precludes the adequate performance of some critical health physics functions. The basic problem appears to be that the health physics organization has not been properly upgraded to meet current demands. These demands include implementing the myriad of regulatory requirements that have evolved during the past few years and providing the health physics coverage necessary for a two-unit operation.[324]

2. The combination of health physics and chemistry groups is generally ineffective and has resulted in serious problems at the technician level. The scope of work is so extensive that the technicians are not properly qualified to perform all of their assigned duties.[325]
3. The assignments of the 24 health physics-chemistry technicians are rotated on a 6-shift schedule and technicians periodically perform all tasks. The shift for "other station duties" was at one time designated as a training shift that no longer exists. Because assignments are for 1 week only, there is little incentive for the technician to become highly proficient in the various tasks within that area. The results are that many tasks are done in a superficial manner, some are performed incorrectly, and some are not done.[326]

   > The severity of the above situation is magnified for some jobs which are performed on a monthly basis. With the existing rotation system and the vast number of tasks involved, a given technician may not perform one of these monthly jobs more than once every two years. *A vital function in this category...is reading and documenting the results of personnel radiation dosimeters.* (Emphasis Supplied.)

4. The supervisor of radiation protection and chemistry has too many people reporting to him because the position of chemistry supervisor remains unfilled.[327]
5. A major cause of inadequate health physics-chemistry technician staffing is the misuse of these personnel in doing the menial tasks of tool, equipment, and respirator decontamination.[328]
6. Technicians were doing a great deal of work which should be performed by clerks. "C served evidence was the misfiling of nea, one-half (about 500–600) of the completed Radiation Work Permits for the year 1978 ... The only individual who is qualified and/or available to compute, format, keypunch and list the exposure data is also the only actual clerical person in the Radiation Protection/Chemistry Department."[329]
7. Because of the lack of rent-a-techs for the current outage, on-the-job health physics coverage, which is required for inexperienced workers and is normally performed by rent-a-techs, is grossly inadequate.[330]
8. "The inadequacies in training of the Health Physics/Chemistry technicians are readily apparent ... their actions are by rote ... when confronted by only slightly off-normal situations, they often lack sufficient understanding of their job to confidently take the appropriate action ... [and] also appear to have insufficient knowledge of plant systems, including the radiological considerations that would apply if the system were opened." "A serious impact of the inadequate technician training is lack of confidence, not only on the part of the technicians themselves, but also by their foremen and supervisors, as well as other station personnel."[331]
9. "The overriding of decisions made by health physics personnel has become a routine occurrence at TMI. Decisions made by Health Physics technicians on the back shifts are frequently overridden by the Operations Shift Foreman."[332]
10. "Activities which may involve considerable changes in radiological conditions are frequently conducted by operations personnel, without notification of health physics."[333]
11. "A definite communications gap is apparent ... between the Radiation Protection/Chemistry Supervisor and the Health Physics Supervisor. Another gap appears to exist between the Health Physics Supervisor and the Health Physics Foremen, and yet another between the foremen and the technicians."[334]
12. "No effective method is employed to ensure that all the technicians are aware of procedure changes."[335]
13. "Personnel dosimetry is one of the weakest areas within the TMI health physics program."[336] "A major reason for the weakness of the TMI personnel dosimetry program is that no individual is assigned to conduct proper reviews of the records."[337]
14. "Both the frequency and locations at which routine air samples are taken appear to be inadequate."[338]

Problems were also noted in radiation surveys and contaminated tool control. The report made appropriate recommendations to correct the discrepancies.

The NUS report was issued on March 20, 1979. Thus, Met Ed did not have the opportunity to thoroughly consider the report prior to the accident, although Met Ed was well aware of the deficiencies in the radiation protection program. Ironically, a meeting among Dubiel, Station Manager Gary Miller, and others to discuss the report had been scheduled for the morning of March 28.[339]

The findings of the NUS report, which we verified particularly through depositions of Met Ed personnel, reflect the continued inadequacy of the radiation protection program over the past several years.[340]

Examination by the NRC

The serious deficiencies in the radiation protection program at TMI raise questions as to the adequacy of the NRC process for licensing and inspecting radiation protection programs at commercial nuclear power reactors.

Inspections of nuclear powerplants are conducted by IE at all phases of plant existence, from the initial management meeting before construction to the closeout inspection and survey when the facility is decommissioned. Areas of concern and emphasis change depending upon the specific stage of plant construction or operation. The inspections at the preoperational stage involve detailed evaluations of the applicant's program. Operational inspections, which are performed annually, shift emphasis to confirmation of adequacy of the radiation protection program by review of records, documentation, and procedures.

Inspections during refueling emphasize compliance with FSAR commitments, technical specification requirements, the need for special procedures, and assessment of advance planning. These inspections occur approximately every other year. The respiratory protection and access control programs also receive particular attention during refueling outage inspections.

We reviewed the IE Inspection Reports for TMI-2 for the period January 1978 to March 1979. Of the 44 inspections made during that time, only four were made specifically for radiation protection: (1) January 5–6, and 26–27, 1978; (2) May 5 and August 9, 1978; (3) October 6, 10–12, 17, and 19, 1978; and (4) February 13–15, 24–25, 28, and March 12, 1979. No items of noncompliance were found during the first two inspections. During the other two inspections, items of noncompliance were reported regarding the posting of high radiation areas, conduct of timely surveys, records maintenance of effluent sampling, high radiation areas without adequate instruments for continuous indication of dose rate, and failure to adhere to certain procedures.

During an inspection for operations conducted on May 10–12, 1978, two radiation protection related items of noncompliance were indicated: (1) failure to perform airlock surveillance, and (2) failure of an individual to monitor himself upon leaving a controlled area. Also, an environmental inspection on April 17–21, 1978 indicated three areas of noncompliance: (1) radiation levels in excess of regulatory limits in an unrestricted area, (2) failure to sample and analyze air particulates and iodines, and (3) failure to meet analytical sensitivity for ^{89}Sr in drinking water.[341]

The IE inspections before the accident did not reveal the serious deficiencies in the radiation protection program discussed above. Only as a result of its investigation of the TMI-2 accident did IE identify these deficiencies. Although many of the deficiencies noted by the IE investigators after the accident resulted in an issuance of notice of violation and imposition of civil penalties,[342] many deficiencies could not be cited as violations because of the vagueness of regulatory requirements. For example:

- Qualifications—"The review and interviews with the technicians indicated five of the twelve radiation chemistry technicians did not appear to have 1 year of related technical training in chemistry or radiation protection. Nine of the twelve radiation chemistry technician Juniors did not have 1 year of related technical training. No apparent item of noncompliance was identified since the term 'should' as used in ANSI N18.1-1971, Section 2.2.1., denotes a recommendation, not a requirement."[343]
- Portable Instrument Availability—"Many additional survey instruments and air samplers were necessary to support the in-plant and environmental monitoring after the accident .... No clear regulatory requirement or licensee commitment established minimum inventories for portable survey instruments at this facility."[344]
- Personnel Dosimetry—"No regulatory requirements or license commitments establish minimum standards for management of personnel dosimetry systems."[345]

The vagueness of the radiation protection requirements raises questions as to the adequacy of the NRC process for review and licensing of radiation protection programs at commercial nuclear power reactors. The proposed radiation protection programs are submitted for NRC staff review at the CP stage and the OL stage. The radiation protection section of the radiological assessment branch in the Office of Nuclear Reactor Regulation conducts the staff review of the proposed program. The review is conducted in accordance with Chapter 12 of the Standard Review Plan to determine compliance with the requirements of 10 C.F.R. Part 20 and conformance to applicable regulatory guides. The review is essentially a paper review of the program; it does not include a specific evaluation of the people, the equipment, or the facility, but is focused on whether the applicant's-licensee's program contains a consideration of these elements in their programs. The conduct of the NRC review of the Three Mile Island Station radiation protection program is discussed in detail in Sections II.B.5.a and II.B.5.b.

An implicit assumption in limiting the review of radiation protection programs is that the total safety review process ensures that engineered safety features would mitigate the consequences of serious accidents. As a result, the focus of the radiation protection review has been predominantly on normal operations and anticipated operational occurrences (i.e., major maintenance and refueling outages).[346]

We find that the NRC review and inspection process in the area of radiation protection focused on conduct of normal power operation. Radiation protection in accident situations, such as existed at TMI-2, were not adequately considered in the licensing review or inspection program.

The Attitude of Met Ed Management

Met Ed's management at TMI did not accord the radiation protection program the importance that the accident has indicated is necessary. Management was "operations oriented,"[347] and its predominant concern was "to keep the plant running."[348] Radiation protection always took a "back seat."[349] Management perceived radiation protection as a "necessary evil"—controls that stood "in the way, many times, of production," but had to be applied "in order to comply with the current regulations."[350] Richard Dubiel, Supervisor of Radiation Protection and Chemistry, agreed with this assessment.[351]

> Q. I have gained the impression from the testimony of Messrs. Janouski, Velez and Mulleavy, that they believe ... that health physics is something of a stepchild in comparison, let's say, to operations ... someone who is getting smaller portions, someone who is not treated as well.
>
> A. Yes, I think that is a fair assessment, understanding that operations are the moneymakers, so to speak. They are the ones who are going to keep the plant operating.

The low priority given to radiation protection was reflected in the Met Ed organizational structure. Dubiel, the highest ranking member of the radiation protection department, reported as a practical matter to the unit superintendents.[352] Thus, radiation protection was literally placed under the direction of operations. It is not surprising that the "stop-work" authority of the radiation protection department was rarely exercised because attempts by the department to exercise authority were regularly overruled by operations personnel.[353]

The attitude toward radiation protection was also manifested by a variety of decisions that involved allocation of money or manpower. For example, an extraordinary number of instruments needed for the radiation protection program were not operational at the time of the accident. A significant factor contributing to this problem was that repair of instruments needed for plant operations was given higher priority. Health physics instruments were repaired "when we can get to it,"[347] often months after the instrument became inoperable.

Management's attitude toward radiation protection developed in part because of its view that it was not necessary to give priority to radiation protection to deal with normal operations and accidents that could be reasonably anticipated. Regardless of whether management's assumptions of what could be anticipated were reasonable, they have been shown by events at TMI-2 to be incorrect.

We find that a conflict existed between operations and radiation protection due to management's motivation toward production. As a result, radiation protection was perceived as a "necessary evil," and its importance was subordinated to production.

The Attitude of NRC

The NRC similarly treats radiation protection as secondary in importance to production. It appears that there are few persons within upper and middle management (above the assistant director level) of the agency with adequate training in, knowledge of, and sensitivity to radiation protection and radiation health. The NRC safety reviews of commercial nuclear power reactors have been hardware oriented. The focus of those reviews has been on equipment and engineered safeguard features to mitigate and safeguard against accidents. As a result, the belief that "accidents can't happen" has colored the agency's approach to radiation protection. As discussed previously, the focus for evaluating and inspecting radiation protection programs has been in assessing the adequacy of these programs to support normal operations and anticipated operational occurrences. We find that the attitude that radiation protection was of secondary importance was held by the NRC.

In addition, the NRC traditionally emphasizes the offsite effects of accidents. The consequences of accidents analyzed for siting purposes are used as the planning base for development of emergency plans and for implementing procedures. The analyses of these accidents address only the offsite-site boundary consequences. Therefore, the emergency plans and the implementing procedures that have evolved comprehensively address offsite related response actions, and actions related to inplant consequences and response are, essentially, completely ignored.

We find that the configuration of emergency organizations, the scope and content of emergency procedures, the design of the facility, and the equipping of emergency facilities do not adequately consider inplant consequences of accidents.

A Change in Attitude?

The accident has engendered a substantial amount of activity by both Met Ed and the NRC to improve the radiation protection program at TMI.

Met Ed—Significant changes introduced by Met Ed have involved the respiratory protection program, the management of the dosimetry program, and the control of access to high radiation areas.[354] The apparent change in attitude is best illustrated by the response of David Limroth, the Supervisor of Administrative and Technical Support, to the following question:[355]

> Q. To what do you attribute the increase in visibility and authority of the health physics program?
>
> A. This whole operation out here today is one hundred percent contingent on a sound health physics program faced with the problems which we have now that the reactor has been brought down to a quiescent state, if you will. . . . We're faced with a massive clean-up effort with relatively unknown challenge in the reactor containment building.
>
> Health physics or radiation protection is the keystone to the success of this operation.

The NRC—The IE investigation revealed many weaknesses in the TMI radiation protection program and resulted in a Notice of Violation and imposition of civil penalties on Met Ed. This Notice of Violation contained numerous radiation protection violations.[356] The transmittal letter indicated that:

> These noncompliances demonstrate serious weaknesses in your ability to maintain an effective health physics program . . .

The NRR Lessons Learned Task Force Report-Short Term Recommendations (NUREG-0578) contained a number of recommendations aimed at improving the overall radiological protection at reactor facilities. For example:

- Recommendation 2.1.6.a—Integrity of systems outside containment likely to contain radioactive materials
- Recommendation 2.1.6.b—Design review of plant shielding of spaces for postaccident operations
- Recommendation 2.1.8.a—Improved postaccident sampling capability
- Recommendation 2.1.8.b—Improved range of radiation monitors
- Recommendation 2.1.8.c—Improved inplant iodine instrumentation.

The NRR Lessons Learned Task Force Final Report (NUREG-0585) also made recommendations regarding training (Recommendation 1.6) and emergency procedures (Recommendation 4) which, when implemented, will improve the effectiveness of radiation protection programs. The IE Lessons Learned Report (NUREG-0616) made nearly 100 recommendations to improve the NRC's inspection and enforcement process, many of which will significantly improve radiation protection programs. On October 17, 1979, Harold Denton, Director of the Office of Nuclear Reactor Regulation, wrote to Met Ed to inform them of the establishment of a special panel of health physics experts to review their program.

All of these actions suggest that there has been a change in attitude toward radiation protection by Met Ed and the NRC. It remains to be determined whether their apparent change in attitude is real and will continue or if they will lapse into treating radiation protection as a necessary evil after they are no longer under the intense scrutiny that has followed the accident.

Met Ed's Radiation Protection Program Compared to Other Commercial Nuclear Power Reactor Programs

We did not examine the radiation protection programs at other commercial nuclear power reactors. However, the scope and nature of the deficiencies noted in the program (Section II.B.5.a through II.B.5.c) raise questions as to whether similar deficiencies might exist at other reactor facilities.

We have explored this avenue of inquiry in a limited manner via informal meetings with the NRR and the IE radiation protection personnel. On September 25 and 26, 1979, meetings were held with senior radiation protection personnel from each IE regional office. The purpose of the meeting was to ascertain, if possible, some comparison of the Three Mile Island Station radiation protection program to those of other commercial facilities. We also met with the NRR radiation protection personnel on October 19, 1979. The purpose of this meeting was to ascertain, if possible, whether the deficiencies noted in Three Mile Island Station's radiation protection program were indicative of generic deficiencies in the NRC's licensing process. Transcripts of these meetings were taken.[357,358] Various elements of the licensee's radiation protection programs were discussed at these meetings including management, procedures, training, personnel dosimetry, personnel exposure and contamination experience, instrumentation (portable and fixed), contamination control, emergency planning, and environmental monitoring.

Based upon these meetings, we believe that the radiation protection program at TMI, although below average, was not significantly worse than those at other commercial reactor facilities. These discussions also support the following findings discussed in this section:

- Many regulatory requirements for radiation protection are not clearly specified in the regulations and technical specifications.
- The basis for review of licensee radiation protection programs has been focused on normal operations and anticipated occurrences.
- The radiation protection programs at operating commercial nuclear power reactors should be reexamined to ensure that the lessons learned from TMI-2 are appropriately reflected in them.

Summary of Findings and Recommendations

We find that:

- a conflict existed between operations and radiation protection because of management's motivation toward production. As a result, radiation protection was perceived "as a necessary evil" and its importance was subordinated to production;
- many of the deficiencies in the Three Mile Island Station's radiation protection program existed prior to March 28, 1979 and went undetected by the routine IE inspections;
- many of the deficiencies in the radiation protection program, even if detected by routine IE inspections, were not covered specifically by regulatory requirements and thus hampered IE from requiring any corrective action;
- the NRC review and inspection process in the area of radiation protection focused on conduct of normal power operation. Radiation protection in accident situations, such as existed at TMI-2, were not adequately considered in the licensing review or inspection program; and
- the attitude that radiation protection was of secondary importance was held by the NRC.

We recommend that:

- the radiation protection function at commercial nuclear powerplants should be independent of operations and should be elevated to equal importance with production;
- the NRC should give greater emphasis in its licensing review and inspection processes to radiation protection. The NRC should reassess the radiation protection programs at commercial nuclear power reactors;
- the NRC should give additional emphasis to radiation protection and radiological health in accordance with the agency's mandate "to protect the public health and safety;"
- the NRC should develop a regulatory base to ensure that inplant radiological conditions resulting from an accident are considered in the planning of emergency procedures.

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C PLANT BEHAVIOR AND CORE DAMAGE

1. DEFICIENCIES IN THE PLANT AND THEIR INFLUENCE ON THE ACCIDENT

a. Introduction and Summary

At the time of the TMI-2 accident, certain conditions existed within the plant that, in retrospect, have been suggested as deficiencies contributing to the accident or preventing its prompt termination. The results of our evaluation of the suggested deficiencies are compiled in this section; findings regarding the significance of each possible deficiency and recommendations resulting from these findings are also included.

Approximately 30 items are grouped into three general sections according to particular plant systems. Those possible deficiencies dealing with aspects of the primary reactor coolant system are included in Section II.C.1.b, engineered safety features in Section II.C.1.c, secondary coolant system in Section II.C.1.d. Additionally, a discussion of the qualification and use of instrumentation in the accident is included in Section II.C.1.e.

General Recommendations

In a number of instances, the findings regarding specific deficiencies are symptomatic of problems of a more general nature; that is, a number of specific deficiencies can be considered the result of a single, more fundamental cause. In the sections below, the specific deficiency notes this connection, when appropriate. However, because of their importance, two general concerns merit special attention here.

General Recommendation 1. Definition and Consideration of Design Basis Events and Accidents

The design basis for nuclear powerplants has been developed and implemented with a widely held judgment that this basis encompassed those events of primary importance in protecting the safety of the public. Because of this judgment, balance in the consideration of different types of accidents swung markedly toward one type (the large break loss-of-coolant accident), while other types received relatively little (if any) attention. Prior reports and events notwithstanding, this lack of balance remained steadfast until the events of March 28.

Some of the deficiencies discussed below point to the lack of balance present in the regulatory process prior to the TMI-2 accident. Issues such as the sensitivity of the Babcock & Wilcox (B&W) nuclear steam supply system, the design of the pressurizer and related equipment such as the pilot-operated relief valve (PORV), the radiological design of vital equipment, the isolation characteristics of the reactor building, and the actuation and control of the emergency feedwater system, among others, are indicative of this problem.

For specific issues such as those mentioned above, specific recommendations are derived. These recommendations indicate the need for reexamination of such issues as the frequency of use of the PORV in B&W plants, qualification of the reactor coolant system pressure control system, the importance of steam generators and related equipment during certain accident conditions, and the capability for hydrogen removal from the reactor building.

Addressing and resolving such relatively narrow issues individually could be expected to improve the safety of nuclear powerplants to some extent. However, such a piecemeal approach would not resolve the more fundamental cause of these concerns and, as such, would not provide the magnitude of safety improvement that the TMI-2 accident indicates is needed. The achievement of the latter goal requires a new balance in the regulatory process. A systematic, integrated approach to the examination of a variety of accidents and the interrelationships within and among these accidents is clearly a necessity.

The results of the evaluations of possible deficiencies lead us to the general recommendation that: Reconsideration of the required "design basis" for nuclear power plants should be initiated immediately. Among the areas that must be reconsidered are:

- the level of safety that must be achieved by the plants;
- the types of accidents for which the plants are designed (such as small and large loss-of-coolant accidents (LOCAs), total loss of ac power, loss of main and auxiliary feedwater, anticipated transients without scram (ATWS);
- the method by which the "design basis" is established (i.e., by the single failure criterion, qualitative and quantitative risk assessment);
- the criteria for the determination of "safety-grade" equipment, including both the determination of the equipment to be so qualified and the actual standards to which this equipment should be qualified; and
- the magnitude of the accident, including but not limited to the severity of fuel damage and core disruption, the magnitude of release of radioactive material, and the magnitude of hydrogen generation.

General Recommendation 2. Use of Human Factors Principles and Disciplines in the Design and Operation of Nuclear Powerplants

The lack of balance in the regulatory process has resulted in a second, equally significant shortcoming of which particular suggested deficiencies discussed below are indicative. Issues such as the capability for bypass of engineered safety feature equipment and the sensitivity of the B&W nuclear steam supply system suggest the lack of proper evaluation of human factors during the design, licensing, and operation of TMI-2. The extent to which human factors considerations have been included in the design and operation of nuclear powerplants would, of course, be expected to vary from plant to plant. However, the paucity of such considerations in the particular case of TMI-2 clearly indicates that proper consideration has by no means been ensured. The conclusions reached in this section regarding the lack of proper human factors evaluations are in agreement with those of Section II.E, which specifically deals with human factors considerations in the design and operation of TMI-2. For this reason, the general conclusion of Section II.E is repeated here:

Thus, we conclude that the integration of human factors principles and disciplines into all facets of the design, construction, operation, maintenance, testing, and regulation of nuclear powerplants will significantly improve nuclear safety.

b. Possible Deficiencies Related to the Primary System

Sensitivity of the B&W Nuclear Steam Supply System

Since the TMI-2 accident, the vulnerability of the B&W nuclear steam supply system relative to other pressurized water reactor designs has been the subject of considerable discussion because of the differences in the operational responsiveness of the designs. A number of features of the B&W design have been suggested as contributors to this apparently greater vulnerability. These features are discussed individually in the sections below; the final section then integrates the individual evaluations into overall conclusions on the design's vulnerability.

Pressurizer Size

The pressurizer is a steel cylinder with hemispheres welded on either end that is attached to the reactor coolant system by a pipe, as shown in Figure II-22.[1] The purpose of the pressurizer is to maintain system pressure and to absorb system volume changes during transients. Heaters near the bottom of the pressurizer heat the water so that a steam bubble is maintained in the top of the vessel. This bubble serves as a cushion. The cushion can be enlarged by additional heating, forcing water out of the pressurizer and back into the reactor coolant system and thus increasing system pressure. By cooling the pressurizer water, the bubble is shrunk, decreasing system pressure.

The pressurizer also has a level indicator showing the level of water in the pressurizer; that is, the level of the boundary between the water and the bubble. Operators commonly use the pressurizer water level indicator to tell them about water level in the entire primary system; if there is some level indication in the pressurizer, the rest of the system should be full of coolant under normal circumstances; if the pressurizer level disappears (goes to zero), there may be no certain way of telling how much water is in the system or even if the reactor core is covered with coolant water.

Pressurizer level can respond in a number of ways during transient conditions (such as reactor trips and accidents). During the initial phase of the TMI-2 accident for example, the level first moved upward, then downward, and then upward again. The first upward movement was in response to the "bottling up" of heat in the reactor. As temperature climbed in the reactor, the water expanded and increased the level in the pressurizer. The level then moved downward, when the reactor "scrammed" and reduced the generation of heat by over 90%, thus causing the reactor coolant to shrink, and temperature and pressure to reduce sharply. When the operators observed the decreasing level, they responded immediately by stopping the normal "letdown" flow of water out of the reactor and increasing the "makeup" flow of water into the reactor coolant system. The level moved upward again (as the operators expected), but then something highly unusual happened. The level did not stop going up, but continued until it indicated to the operators that the pressurizer was completely full of water. It was at this point that operators throttled the high pressure injection system (which had come on automatically) in the belief that less, not more, water needed to be added to the primary system. Although they did not realize it, the stuck-open relief valve in the top of the pressurizer was permitting coolant to flow through the pressurizer and out of the system.

The first potential concern evaluated here is that the volume of the TMI-2 pressurizer might be relatively small compared to other pressurized water reactors of comparable power. The specific concern in this instance is that a smaller volume would result in greater changes in level in the TMI-2 pressurizer for any particular transient in the reactor. Because the rapid rise in pressurizer level early in the TMI-2 accident apparently contributed to the confusion experienced by the operating crew, we must consider the possibility that a relatively small pressurizer volume was a design deficiency contributing to this accident.

An examination has been made of the volume of the pressurizers of a number of nuclear plants, the details of which may be seen in Table II-40. This examination indicates that the pressurizer volume in TMI-2 is comparable to that in other plants.

A number of operational events have occurred in B&W plants involving loss of pressurizer level indication in both the high and low directions. These events, which are shown in Table II-41, may be construed to imply that the pressurizer volume is insufficient to accommodate certain transient events. However, consideration of the causes of these operational events suggests that the pressurizer volume is not directly the problem. Rather, it appears that the plant sensitivity to secondary side transients is the basic problem; that is, a sensitivity to the amount of heat removal through the steam generators and to the rapidity of the changes in heat removal capability during transient events. This sensitivity is discussed in more detail below.

Steam Generator Secondary Side Coolant Inventory

The design of the B&W pressurized water reactor (PWR) includes steam generators that are considerably smaller (in terms of secondary side water inventory) than Westinghouse or Combustion Engineering PWRs, as may be seen in Table II-42. In the event of an interruption of feedwater to the steam generators, as occurred at the beginning of the TMI-2 accident, the smaller size in the B&W design results in a more rapid drying out of the secondary side; this then results in a more rapid

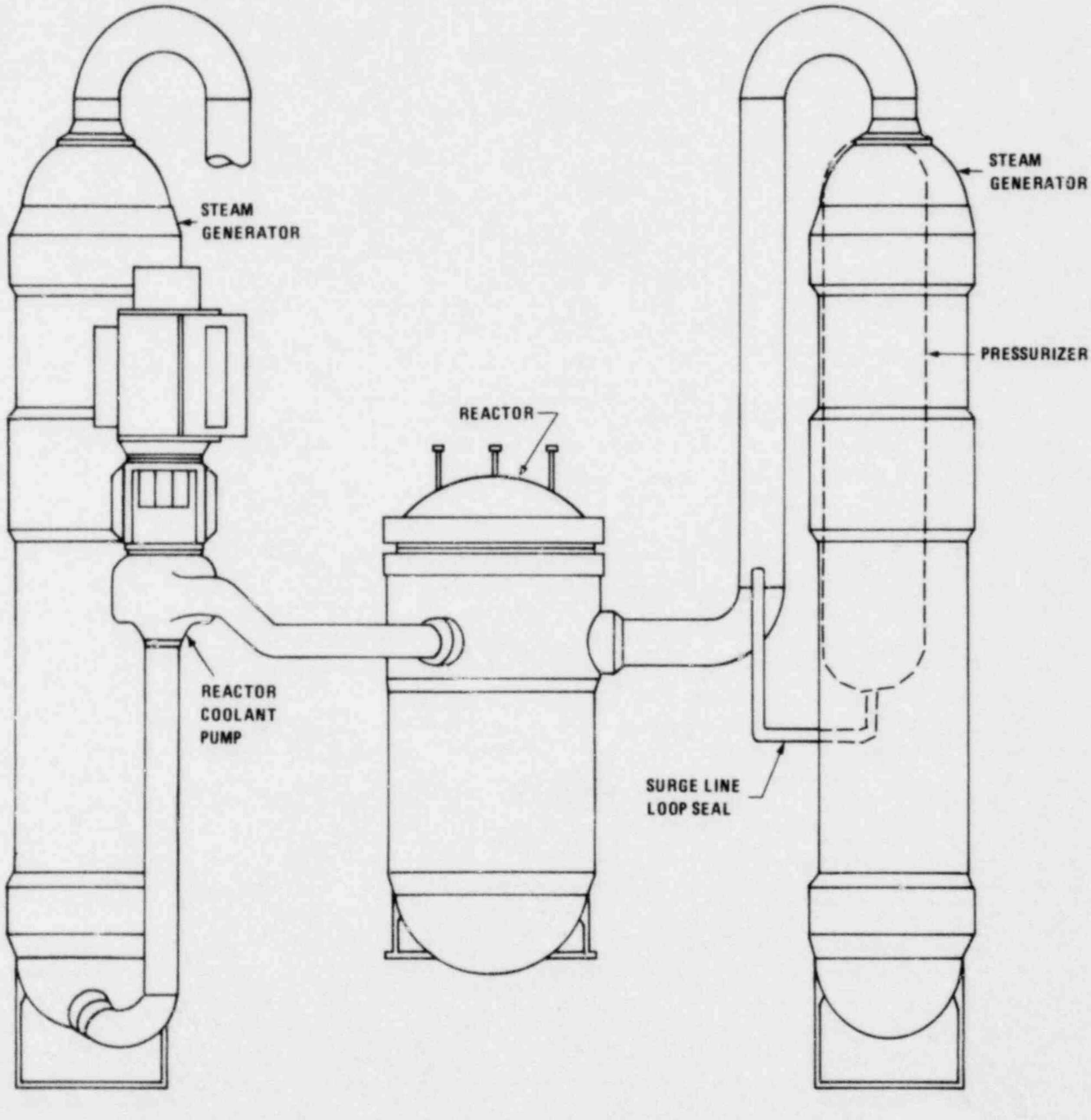


FIGURE II-22. Pressurizer Surge Line Loop Seal Arrangement

TABLE II-40. Pressurizer sizing

Plant	Vendor	Thermal Power (MW)	Pressurizer Volume (ft^3)	Ratio: Pressurizer Volume (ft^3) to Thermal Power (MW)
TMI-2	B&W	2772[2]	1500[3]	0.54
Oconee	B&W	2568[4]	1500[5]	0.58
San Onofre 2&3	CE	3390[6]	1500[7]	0.44
St. Lucie	CE	2560[8]	1500[9]	0.58
Surry	W	2441[10]	1300[11]	0.53
Sequoyah	W	3411[12]	1800[13]	0.53

TABLE II-41. Instances of loss of pressurizer level in B&W plants[14,15]

Plant	Date	Direction of Level Losses		Cause
		High	Low	
Davis Besse	11/29/77		X	Loss of ac power
Davis Besse	9/14/77	X		Loss of feedwater—stuck-open PORV
Rancho Seco	3/20/78		X	Electrical malfunction—ICS
TMI-2	3/29/78	X		Bus failure—stuck-open PORV
TMI-2	4/23/78		X	Main steam safety valves fail open
TMI-2	11/07/78		X	Loss of feedwater
TMI-2	3/28/79	X		Loss of feedwater—stuck-open PORV

TABLE II-42. Steam generator secondary side coolant inventories for various pressurized water reactors

Plant	Designer	Power (MW)	Total Steam Generator Secondary Side Coolant Inventory (Approximate)
TMI-2	B&W	2772[2]	110 000 lb[16]
Oconee	B&W	2568[4]	110 000 lb[17]
Calvert Cliffs	CE	2560[18]	430 000 lb[19]
St. Lucie	CE	2560[8]	440 000 lb[20]
Surry	W	2441[10]	261 000 lb[21]
Sequoyah	W	3411[12]	376 000 lb[22]

loss of heat removal from the primary coolant, causing its temperature and pressure to increase more quickly.

A second design feature of the B&W steam generators also contributes to the more rapid heatup of the primary coolant. In these steam generators, a significant fraction of the area for heat transfer is used in increasing the quality of the exiting steam by film boiling heat transfer. Thus lower portions of the steam generator tubes are experiencing nucleate boiling while upper regions are experiencing film boiling. Heat transfer coefficients in a region of film boiling are significantly less than those in a region of nucleate boiling, so that a rapid change in water level results in a rapid change in the mechanism of heat transfer and a correspondingly rapid change in the amount of heat removal actually accomplished. Such rapid changes can occur when secondary side water levels are affected by transients such as the loss of the main feedwater pumps experienced at the beginning of the TMI-2 accident. As the water level decreases, the uncovered tubes experience a change from nucleate to film boiling, reducing rapidly the heat transfer in that region. This then reduces the total heat removal achieved from the primary coolant.

The fast response of the B&W steam generators is a favorable feature in the context of plant generation of electricity. However, in the context of reactor safety, the faster response to abnormal transients (than experienced in other PWR designs) requires more rapid attention and intervention by the operating crew and/or automatic controls to prevent or minimize the effects on the reactor coolant system. Ed Frederick, a control room operator who was manipulating the makeup and high pressure injection controls in the TMI-2 control room during the initial stages of the TMI-2 accident, has testified:

> Specifically on the pressurizer, you often find yourself working very hard to maintain yourself within those limits, even on a simple reactor trip. It will take several manual actions to maintain, for instance, the minimum 100-inch figure for keeping the heaters covered. Much of the reactor trip procedure is devoted to pressurizer level control, so I can't really think of anywhere that we purposely ignore this or try to exceed it and/or let it be exceeded because they are so important to the plant pressure control.
>
> Q: So you obviously ... are concerned with pressurizer level not going down:
>
> A: Right.[23]

Because of this need for rapid operator action, we consider the B&W design to be fundamentally more susceptible to human errors than other PWR designs.

Use of the Pilot-Operated Relief Valve (PORV)

In the B&W pressurized water reactor design, the PORV at the top of the pressurizer is used routinely during transient events. When a transient event causes the reactor coolant system pressure to increase, the PORV is designed to open in an attempt to compensate for the increase. Because of this design feature, the PORV is used about five times a year in each B&W plant (see Table II-43). In contrast, the Combustion Engineering (CE) and Westinghouse PWR designs do not routinely use the pressurizer PORV. Data shown in Table II-43 indicate that the frequency of use of the PORV is significantly less (a factor of 10 or more) than in the B&W design.

Data assembled in Table II-43 indicate the frequency of PORV *failures to reclose* for plants designed by each of the PWR vendors. For the B&W-designed plants, the frequency of experiencing a stuck-open PORV is estimated to be about 0.1 to 0.3 per reactor year, depending on the inclusion or exclusion of events occurring while the plants were not in power operation. In plants designed by Combustion Engineering, only one instance of a PORV failing to reclose has been discovered; this event occurred while the plant (Palisades) was at hot shutdown conditions.[26,28] If one assumes that

TABLE II-43. Operating experience with PORV

Vendor	Operating Experience (Reactor Years as of 3/28/79)	Number of PORV Openings	Frequency of PORV Opening (per Reactor Year)	Number of PORV Failures to Reclose			Frequency of PORV* Failure to Reclose		
				Power Operation	Nonpower Operation	Total	Power Operation	Nonpower Operation	Total
B&W	33[24]	150[25]	5	3[14]	5[28]	8	0.1	0.2	0.3
CE	35[24]	4[26]	0.1	0[26]	1[26]	1	–	0.03	0.03
W	141[27]	43[27]	0.3	1[29]	0[27]	1	0.007	–	0.007

*Per reactor year.

this event is relevant, then the frequency of such events in CE plants is estimated to be about 0.03 per reactor year. In plants designed by Westinghouse, one instance of a PORV failing to reclose has been noted. This event occurred in the NOK-1 plant in Beznau, Switzerland,[29] while the plant was in power operation. As Table II-43 indicates, the frequency of experiencing a PORV failing to reclose in Westinghouse plants is about 0.007 per reactor year.

Data collected in the reactor safety study[30] on the probability of a pipe break comparable to the size of a PORV opening indicates that this probability has a median value of about 0.001 per reactor year. Because this value is less than that estimated for the failure of a PORV to reclose, one can conclude that such PORV failures can be major contributors to the likelihood of PWRs experiencing small-break loss-of-coolant accidents. The data in Table II-43 indicate that B&W plants are particularly susceptible to this problem.

Since B&W plants appear to be particularly vulnerable, the question arises regarding the reason. A comparison of the ratio of the number of PORVs failing to reclose during power operation to the number of PORV openings shows that this ratio is essentially the same for Westinghouse and B&W plants (no data exist in this case for CE plants). The greater vulnerability of B&W plants is due to significantly greater use of (or demand upon) the PORV. It is the particular design and operational characteristics of the B&W plants that, by requiring more frequent reliance on the PORV, result in the significantly greater susceptibility of these plants to small loss-of-coolant accidents as the result of the sticking open of this valve.

Unlike a small loss-of-coolant accident resulting from a pipe break, such an accident resulting from a stuck-open PORV can be mitigated by use of the PORV block valve. Thus, operator intervention to close the block valve reduces, in effect, the likelihood of experiencing a serious, prolonged loss of coolant. The experience in TMI-2, however, suggests that operator intervention cannot be overly relied upon.

In summary, because of the frequent use in B&W plants of the pressurizer PORV in mitigating various normal transients, the likelihood of experiencing a stuck-open valve is significantly higher in B&W plants than in other PWRs. Because a stuck-open pressurizer PORV can be equivalent in consequence to a small loss-of-coolant accident (LOCA) and can occur more frequently than other types of small LOCAs, we conclude that prior to the TMI-2 accident the likelihood of a small LOCA (as the result of a stuck-open PORV) was significantly greater in B&W plants than in Combustion Engineering and Westinghouse plants. (Actions taken since the accident by NRC and the B&W utility owners have reduced this frequency significantly.)

Lack of Anticipatory Reactor Trip

At the time of the accident at TMI-2 (like the incidents at other B&W plants) there were no provisions to cause the reactor to shut down automatically in response to a total loss of feedwater or a turbine trip. Instead, the integrated control system (ICS) was designed so that reactor power would automatically be run back in the expectation that the pressure increase caused by the loss of heat removal through the feedwater system would be mitigated by the opening of the PORV. The designers intended, through this combination of reduction in power and operation of the PORV, to keep the primary system pressure below the "high pressure" safety limit at which the reactor automatically tripped off, thus avoiding undesirable downtime.

In contrast, had the reactor automatically scrammed by an anticipatory trip when the turbine tripped, there would have been a sharp decrease in the amount of heat being added to the primary system, and the pressure might not have increased enough to cause the PORV to open, thus preventing the accident. An anticipatory reactor trip following turbine trip requires a turbine steam stop valve closure or generator breaker open signal to the reactor protection system for a near simultaneous reactor trip. The anticipatory trip prevents, in most instances, the opening of the PORV and negates control rod runback, which is a feature of the B&W ICS.

Automatic reactor trip under these conditions was not required by NRC regulations. Some other vendors—GE and Westinghouse—voluntarily provided for these "anticipatory trips" in their designs.[31] The NRC management had decided that such control systems[32] and anticipatory trips fell outside of the scope of the NRC staff review; therefore, the staff had never performed a safety analysis to determine the significance of anticipatory reactor trips in dealing with various kinds of abnormal events in the plant.

Of particular interest in this section is the effect of anticipatory trip on the overall responsiveness of the TMI-2 plant. The influence of the lack of such a feature is to decrease the time available to the operating crew to cope with the event. The delay of reactor trip causes the input of a significant amount of energy into the reactor coolant system above that which would have been input had the reactor been tripped immediately. In transient events where the normal cooling path is interrupted (e.g., when the main feedwater pumps and turbine are tripped), this additional energy input can substantially change the steam generator dry-out time and affect the reactor coolant system pressure and temperature. The overall effect of the delay in reactor trip is, thus, a decrease in the time in which the operating crew has to perform necessary actions. Because human errors become increasingly likely as the time to perform actions decreases, the lack of an anticipatory trip in those B&W plants not having an immediate trip upon turbine trip (like TMI-2) may be translated into an increased susceptibility of these plants to human errors.

In an overall sense, the sensitivity of B&W designed plants to transient events causes these plants to be more vulnerable to accidents than other pressurized water reactor designs. The method of reactor coolant system pressure control during transient events places reliance on a relief valve with a known propensity for failing in an open position; this design feature results in a greater likelihood of experiencing a small loss-of-coolant accident through this valve. Other features of the plant design require the operating crew to make more hurried judgments in response to the initiating event and reduce the time available to take corrective measures. These features make the B&W design less "forgiving" to errors by the operating crew.

Findings—Because of particular design features and operational characteristics of B&W plants and the resulting plant sensitivity to transient events, these plants have a significantly higher likelihood of experiencing a small loss-of-coolant accident as a result of a stuck-open PORV than other pressurized water reactors and, further, have an increased susceptibility to human errors during plant transients than other pressurized water reactors. The combination of these two aspects in the B&W plants has contributed significantly to the accident at Three Mile Island.

Recommendations—Methods should be developed and implemented that reduce the frequency of use of the PORV in B&W plants. Methods to reduce this frequent use implemented since March 28, 1979 (i.e., anticipatory reactor trip, PORV setpoint increase) provide a temporary, but not necessarily optimum, solution to this concern. Alternate methods of providing equivalent or greater degrees of protection that also reflect a systematic consideration of a spectrum of transient events should be analyzed. Implementation of the best available solution should then be undertaken.

The analysis of methods to reduce the frequency of PORV use should be undertaken as one part of a systematic evaluation of the potential safety implications of the sensitivity of B&W plants to transient events. This more general evaluation should consider, for example, the implications of loss of pressurizer level indication (in the low and high directions) and of frequency of actuation of engineered safety features, as well as the frequency of PORV use noted above.

The failure to recognize this particular concern before the TMI-2 accident can be attributed to the more general lack of adequate consideration of transient-initiated accidents. The findings and recommendations discussed here are indicative of the lack of balance exhibited in the licensing process. The analysis and resulting plant modifications recommended here should therefore be considered and pursued with general recommendation 1 in Section II.C.1.a clearly in mind. Although this concern appears to be more significant for B&W plants, similar evaluations should be made of the other LWR designs.

Because of the greater sensitivity of B&W plants to human errors during transient events, proper human factors evaluation of the human factors in surveillance procedures, emergency procedures, and systems and other equipment design should particularly be emphasized as such programs are initiated within the licensing and regulation process. The overall development and implementation of human factors programs are the subject of general recommendation 2; the specific findings and recommendations discussed here should thus be considered as support to, and taken in the context of, this general recommendation.

Pressurizer and Pressure Control System Design Features

Design of the Pilot-Operated Relief Valve

Two aspects of the design of the pressurizer PORV will be discussed in this section. The first is the capability of the PORV to pass mixtures of steam and water. The second aspect is the possibility that the discharge piping arrangement from the PORV in TMI-2 may have been the cause of the valve remaining open when it was supposed to close. These are discussed separately below.

Capability of the PORV to Pass Two-Phase Flow

The first issue of interest here is the capability of the PORV to pass a two-phase mixture of liquid water and steam. The possibility arises that, upon the complete filling of the pressurizer, the two-phase flow through the valve may have caused sufficient damage to prevent any further operation. Investigation of this possibility was pursued with both the manufacturer of the valve (Dresser Industries) and with B&W. Dresser indicates that the PORV of the type in TMI-2 has not been qualified for discharging two-phase or liquid water flow.[33] However, statements by a B&W staff member indicate that, as part of their analysis of the ATWS issue, the capability for water discharge through the relief and safety valves was evaluated. The conclusion of this B&W evaluation was that, although these valves were not qualified for water discharge, this discharge would not lead to "unacceptable damage."[34] Thus, considerable uncertainty remains as to the capability of the PORV to pass two-phase or liquid water flow.

Findings—The capability of the PORV to discharge two-phase or water flow appears to be sufficiently uncertain to merit additional consideration.

Recommendations—Additional analysis and testing of the capability of PORVs to discharge two-phase or liquid water flow should be required to establish this capability definitively. This recommendation supports the recommendation made by the Lessons Learned Task Force (short term recommendation 2.1.2).[35]

Effect of PORV Discharge Line Piping Arrangement on Reclosure Capability

The piping arrangement for the discharge from the PORV pilot valve has also been studied to determine whether backpressure forces in that line prevented closure of the PORV. This possibility has been pursued with Dresser Industries,[33] and the results of this inquiry indicate that forces in the particular pilot valve discharge line installed in TMI-2 would not be sufficient to hold open the pilot valve. Since one would expect to experience such a failure the first (and each) time the PORV was used, the lack of previous failures at TMI-2 resulting from such backpressures would tend to support the conclusions reached by Dresser.

Findings—The particular arrangement of the PORV pilot valve discharge line in TMI-2 does not appear to be the cause of the failure of the PORV to reclose.

Reactor Coolant Pressure Control

Reactor coolant pressure is automatically controlled by pressurizer (1) electric heaters, (2) the spray valve, and (3) the power-operated relief valve. (For detailed discussion of the operation of this control system see Refs. 36 and 37.) The pressure control system is not classified by the NRC as a system important to safety in their review of the FSAR;[36] therefore, in the analyses the failure of the pilot-operated relief valve (PORV) to close was not considered to cause unacceptable consequences in a transient mitigation sequence.[36,38] However, failures in the PORV and later in the electric heaters, at about 3 hours into the accident, limited the ability of this control system to maintain system pressure above saturation at occasions when the operators judged it necessary to increase system pressure to retain the plant in a safe condition.[39,40] Pressurizer spray also became unavailable for pressure control when the forced reactor coolant circulation was interrupted after the reactor coolant pumps were stopped[41] by the operators. The operators at TMI seemed to have forgotten that pressurizer spray capability cannot be maintained after the reactor coolant pumps are stopped because they attempted to spray after the pumps were stopped.

Electric power supply for the pressure control system is provided by the offsite power source, and interruption of this power source would have made the pressure control system unavailable indefinitely and the PORV block valve (located between the pressurizer and the PORV) unable to close at the operator's command. (See Section I.B.1 of this report for additional discussion.)

Findings—B&W, Met Ed, and NRC failed to acknowledge the safety significance of the reactor coolant pressure control system. There was a lack of failure mode and effects analyses (FMEA) of this and other control systems. Although NRC experts recognized that such control systems were important to safety, it remained NRC policy to exclude these control systems from safety review. (For additional information regarding staff evaluation of control systems, see Refs. 32, 42, 43, 44, and 45.)

Recommendations—The categorization of the pressure control system as "nonsafety" should be reevaluated. If this system is deemed important to safety, as we believe, it should be designed to safety criteria, and at minimum, automatic closure of the block valve by system pressure should be considered to limit the need for operator intervention. Furthermore, to retain availability of the pressure control system in the event of loss of the offsite power sources, electrical interconnections should be made to permit the supply of power to the pressure control system from the onsite ac power sources. As also recommended by the Lessons Learned Task Force,[46] FMEA of all control systems should be conducted immediately for all plants.

Use of Pressurizer Level Instrumentation

An unexpectedly high water level in the pressurizer persisted during the early events of the accident as a result of swelling of the overheated primary coolant, perhaps from the unavailability of emergency feedwater within the expected time and the formation of steam voids in the core. The expected response in pressurizer level indication, following initial events in the TMI-2-type accident, is a rapid decrease in level when the emergency feedwater system is immediately available. However, the steam void formation and, to some extent, the lack of the emergency feedwater caused the primary coolant to expand, leading to a high pressurizer level; this implied that the primary system was filling to a solid condition. This condition was not understood by the operators, who used their approved written procedures[47] and intervened to interrupt high pressure injection (HPI) and normal makeup and also increased the letdown from the system.

Level indication was provided by three physically independent level transmitters, two of which failed later during the accident, causing the use of alternate indirect methods to ensure continued level indication. This level indication remained important for the continued assessment of the primary coolant system pressure.

During an event similar to TMI-2 at a foreign reactor of a different vendor, actuation of safety injection was never initiated automatically, as was required by abnormal system conditions that existed during the event.[29] Automatic actuation was dependent on the simultaneous decrease of pressure and level in the pressurizer. Since pressure and level did not simultaneously decrease, as in TMI-2, the coincidence permissives were not satisfied. Operators at the foreign reactor recognized the design deficiency and manually actuated safety injection.

Failure to attribute safety significance correctly to pressurizer level, even following some telling incidents,[29,48] allowed routine operational judgments to dictate reactor coolant system performance.

Findings—It appears that thermohydraulic bounding analyses, from which most of the principal instrumentation and control for reactor protection is derived, lack accuracy in predicting some system variations. The unexpectedly high-level indication at TMI-2 and earlier during an incident at a foreign reactor indicate the need for reassessment of some thermohydraulic models of accident sequences.

Recommendations—Bounding thermohydraulic analyses should be reevaluated to determine their accuracy in predicting system variations.

Automatic reactor protection actions should be derived, to the degree possible, from independent process variables.

Automatic actions through coincidence of independent process variables should be limited, to the degree possible, for nonreactor protection functions.

Pressurizer level instruments should be designed to criteria applied for instrumentation systems important to safety, and emphasis should be placed on achieving diversity in the measured parameters.

Surge Line Loop Seals

Another concern with the B&W pressurizer design is that it includes a "loop seal" in the pressurizer surge line (see Figure II-22). We have studied the possibility that this loop seal contributed to the artificially high pressurizer levels indicated to the operating crew and, in this way, contributed to the throttling of the high pressure injection system and the resulting core uncovering and fuel damage.

In the first 1 to 2 hours of the accident, a number of effects were influencing the pressurizer level, causing it to go off scale in the high direction and remain there. Among these influences were the stuck-open PORV, the high initial flow rates from the makeup pumps, the increase in coolant volume because of heating, and perhaps the flashing in the pressurizer reference leg. Analysis since the time of the accident suggests that, of all the effects noted above, high coolant flow rates from the makeup pumps were the most significant contributor to the initial increase of the level off scale.[49]

The particular influence of the surge line arrangement in the early hours of the accident is not so easily discernable. Analysis done by Westinghouse since the TMI-2 accident on a small-break loss-of-coolant accident in the pressurizer steam space (which includes a stuck-open PORV) indicates that Westinghouse-designed plants would experience a similar increase in pressurizer level resulting from a stuck-open PORV.[50] Because Westinghouse plants have a vertical surge line (i.e., no loop seal), it would appear that the loop seal arrangement is not an important influence in causing increasing pressurizer level during a break in the steam space.

During the time that water levels in the reactor coolant system (RCS) were below the connection of the pressurizer surge line to the hot leg, the loop seal arrangement had a more pronounced effect on the artificially high pressurizer level. With a loop seal arrangement it is possible for a steam volume in the hot leg to support a column of water in the pressurizer if the system pressure is greater than the saturation pressure of the pressurizer liquid. Thus, during the time period in which the RCS coolant level was below the surge line (including the time period in which core exposure and damage occurred), the loop seal arrangement of the pressurizer surge line was a significant contributor to the artificially high pressurizer level seen and used by the operating staff. The disjunction between water level in the pressurizer and water level in the core region is most readily apparent (in hindsight) in this time period.

Findings— Apparently the increase in pressurizer level off scale in the first few minutes was due primarily to increased flow into the reactor coolant system from the makeup pumps and not to the particular design of the surge line loop seal.

During the time of core uncovering and damage, the arrangement of the surge line loop seal was an important contributor to the artificially high pressurizer level seen by the operating crew.

Recommendations— We believe that a more direct method of indicating water level in the reactor core is needed to complement the potentially misleading pressurizer level instrumentation.

Reactor Coolant Pump Control

Several times during the course of the accident, forced circulation of the primary coolant was attempted to ensure decay heat removal from the primary. This was needed because natural circulation was inhibited because of noncondensible gases and steam in the primary coolant system. Reactor coolant pump operability was required to regain forced circulation.

During various periods later in the accident, reactor coolant pumps were removed from service because conditions in the primary system exceeded those allowable for continued pump operation.[51,52,53] Hence, the desired forced circulation was interrupted for extended periods of time.

Several times operators were unsuccessful in their attempts to restart reactor coolant pumps because various permissives in the start circuit of the pump controls were violated.[54] The pumps were started, however, when operators physically bypassed permissives. The ac electrical power supply to oil lift pumps for the reactor coolant pumps was lost when two motor control centers were inadvertently tripped immediately following the pressure surge of 28 psig in the containment just before 2:00 p.m. on the first day. This trip violated a reactor coolant pump permissive (oil lift pump running) in the coolant pump start circuit. Operators, however, manually bypassed this permissive and started the reactor coolant pumps with oil lift provided by other pumps powered from a dc power supply. The operators were able quickly to recognize the correct permissives for bypassing because difficulties experienced before the accident with the same permissives required similar actions to be taken.

Operation of the reactor coolant pumps at certain times mitigated the accident (see Section II.D) of this report). However, reassessments by the NRC and vendors following the accident have revealed that an immediate trip of the reactor coolant pumps during some small-break LOCAs are required to limit the loss of inventory through the break.[55] Because of differences in the results of calculations done for the Special Inquiry Group (see Section II.D) it is not possible to conclude that this would have been the appropriate action at TMI.

Findings— The operators were required to have a very intimate knowledge of complex control permissives to complete circuits and place the pumps back into service. Up-to-date control logics were not made available to the operators to ensure accurate knowledge of the controls.

The desire to minimize loss-of-coolant inventory has resulted in the requirement for immediate trip of reactor coolant pumps following any incident to ensure that, in the event of a small-break LOCA, loss through the break will not result in exposure of the core.

Recommendations—If the immediate trip of reactor coolant pumps remains a requirement for mitigation of accidents, automatic features should be installed to ensure their immediate trip. If, however, the pumps are required to operate during any part of an accident, their power supply and control systems should be designed to the criteria applied for systems important to safety.

We believe that the requirement for an immediate reactor coolant pump trip is a temporary fix, and we recommend that an immediate reevaluation be made to ensure that the emergency core cooling system (ECCS) is designed with sufficient capacity to preclude uncovering of the core when the reactor coolant pumps continue to run during any accident.

Control logics for all complex systems and components should be made available to the operators to ensure their continued familiarity with all control permissives and inhibits.

Inhibitions to Natural Circulation

Throughout the first day of the accident attempts were made by the operating crew to induce natural circulation cooling in the reactor coolant system. During the interval when the reactor coolant pumps were not providing forced circulation cooling (i.e., from about 5:40 a.m. to about 7:50 p.m.), it was judged that this mode of cooling the core was highly desirable, however, attempts to induce it were apparently unsuccessful until about 4:00 to 5:00 p.m., when some natural circulation may have been achieved. In this section we evaluate the contribution of the plant design to the inhibition to natural circulation under abnormal circumstances.

A distinction should be made concerning how "natural circulation" cooling is being defined and used in this discussion. For the purposes of this section, the term "natural circulation" is used in the narrow context of single-phase natural circulation cooling; that is, cooling by the flow of only liquid water through the core and steam generators. An alternate method of cooling by steam generation in the core will be referred to as the "reflux boiling" mode. In this reflux type of cooling, heat removal from the fuel is achieved by boiling water in the core, which then flows as steam to the steam generators and condenses. The accumulation of liquid water in the bottom of the steam generators, with subsequent flow back into the lower part of the vessel ("refluxing"), replenishes the supply of water in the core.

Before entering into a discussion of the capability for natural circulation cooling in B&W plants during abnormal circumstances, some discussion of this capability during normal circumstances is useful. Since the TMI-2 accident, the capabilities of B&W plants in such situations have been questioned. Based on operating experience where natural circulation cooling was achieved and on specific natural circulation cooling tests in B&W plants,[56] it appears that the capability for such a cooling mode under normal circumstances is adequate.

In the TMI-2 accident, the capability for natural circulation cooling was initially lost within minutes after the turbine reactor trip, caused by the initial depressurization of the reactor coolant system (RCS) and the resulting flashing of RCS water into steam. When the last reactor coolant pumps were tripped at 5:40 a.m., the steam in the RCS collected at the various high points of the system: the upper head of the reactor vessel and the upper sections of the hot legs (the "candy canes"). The presence of steam in the hot legs, in concert with the large coolant mass loss out the PORV, prevented natural circulation cooling at that time and for some time afterward.

Very soon after the reactor coolant pump trip at 5:40 a.m., the core began to be uncovered as a result of the continued coolant mass loss out the PORV. For at least the next hour, the core was partially uncovered and fuel temperatures rose very high, causing the generation of hydrogen from the metal-water reaction. As this hydrogen was being produced, it too was rising into the high points of the reactor coolant system. Thus, from approximately 6:00 or 7:00 a.m. until about 5:00 or 6:00 p.m. the inhibition of natural circulation already resulting from steam was compounded by the presence of noncondensible hydrogen. Because of these two substances, attempts during this time to induce natural circulation by repressurization or to reinstitute forced flow by starting a reactor coolant pump were unsuccessful.

Consideration of coolant levels and other factors suggest that, upon tripping of the last reactor coolant pumps at about 5:40 a.m., the reflux boiling method of cooling was also inadequate. After the time of the pump trip, water levels are estimated to have settled to roughly the top of the core (see Section II.C.2). At this time the B steam generator was isolated, secondary side water levels were relatively low, the A steam generator water level being increased to about 21 feet, through the use of one emergency feedwater pump.[57,58] Also, the stuck-open PORV had remained as yet undiscovered, so that coolant loss from the RCS continued. Under these circumstances some steam condensation in the A steam generator may have been occurring. However, as the water level in the RCS continued to fall and the horizontal sections of the cold legs drained, the capability for continued reflux boiling decreased.

With additional uncovering of the core, hydrogen generated by the Zircaloy-steam reaction rose into the hot legs, inhibiting steam flow to the steam generators. This "binding" reduces the heat transfer capability, and the possibility of core cooling by reflux boiling is further decreased.

It thus appears that within 30 minutes after the tripping of the last reactor coolant pumps, the effectiveness of the reflux boiling mode of cooling was essentially lost. This loss can be attributed to the continued coolant loss out the PORV and the inability to use the heat removal capability of the steam generators effectively.

The eventual (apparent) restoration of some reflux boiling capability and the restarting of a reactor coolant pump some time later appear to be attributable to substantial refilling of the RCS and the escape of some of the steam-hydrogen mixture from the loop A hot leg. This escape appears to have been due primarily to the depressurization of the RCS beginning at about 11:40 a.m. This decrease in pressure allowed the mixture of steam and gas to expand to the point that it could flow into the pressurizer through the surge line and then out the PORV into the reactor building. The reduction in the amount of blockage in the loop A hot leg then apparently allowed sufficient flow to move through the hot leg to provide some cooling. This reduction also may have made possible the reactor coolant pump restart at 7:50 p.m.

It becomes apparent from the above discussion that the RCS hot legs were a primary source of the blockage that prevented natural circulation. Since these are high points in the system, this is not unexpected; similar behavior would be expected in the U-tube region of the hot legs in Westinghouse and Combustion Engineering pressurized water reactors. However, because the hot-leg high points in these PWRs are within the steam generators where feedwater can be used directly to condense and "unblock" steam pockets (without noncondensible gas), the problem of steam blockage is not as serious a concern as in B&W plants.

The presence of hydrogen or other noncondensible gases in the steam pockets in the hot legs of any PWR makes the restoration of natural circulation or reflux-boiling cooling more difficult. However, the relatively larger volume of the B&W hot-leg design than of the U-tube arrangement in other PWRs suggests that the amount of noncondensible gas required to block natural circulation or reflux boiling initially may be somewhat larger in B&W plants. However, because this larger volume can also retain a greater amount of gas, the ability to sweep gas out once it accumulates (by use of the reactor coolant pumps) may be less in the B&W design.

It appears, therefore, that the B&W PWR design is somewhat more vulnerable to loss of both natural circulation and reflux boiling capability during abnormal circumstances. This relatively greater vulnerability is due to the design of the hot legs that makes steam or the combination of steam and noncondensible gas more difficult to remove once trapped. Because of this, the concept of remote venting capability to be discussed below should be of greater use for B&W plants.

An additional insight gained during this evaluation merits some discussion here. As one considers the various problems encountered in achieving natural circulation or reflux boiling (along with the more detailed discussion of the events of March 28 in Section II.C.2), it becomes apparent that the capability of the steam generators as a heat removal mechanism was not well utilized or not well understood by the operating crew on that day.

The capability for heat removal through the steam generators when forced flow is not occurring in the RCS is dependent on a number of parameters. Among the more important are relative water levels on the primary and secondary sides of the tubes and the secondary side pressure.[59,60] The actions of the crew during the first 16 hours of the accident suggest that the maintenance of high water levels on the steam generator secondary side (or the continued flow of emergency feedwater onto the tubes) did not receive attention appropriate to its importance. Further, depressurization of the secondary side to improve its heat removal capability was apparently not attempted during the accident. The less-than-complete use of the steam generators suggests the need for a better awareness on the part of operating crews of the importance of this heat removal mechanism during transient-initiated and small-break accidents.

We note that, since the TMI-2 accident, B&W has instructed its plant owners (of lowered-loop plants) to require that steam generator level be raised to high levels (95% of the operating range) after RCP trip in a small-break LOCA.[61] This action is a good first step toward utilization of the steam generator heat removal capability. Additional guidance in this area to provide both an adequate use and an understanding of this capability seems to be warranted.

Findings—Under normal circumstances, the capability for natural circulation cooling in B&W plants appears to be adequate.

Under abnormal circumstances, the ability to restore natural circulation cooling (once lost) appears to be somewhat more difficult in B&W plants than in other pressurized water reactors.

The importance of the steam generators and their application as a mechanism for heat removal during transient-initiated and small-break accidents like that at TMI-2 does not appear to have been adequately understood.

Recommendations— We believe that no specific recommendations concerning inhibitions to natural circulation cooling are necessary here; however, the recommendations listed below concerning the use of remotely operable vents at the RCS high points are also germane to the natural circulation cooling concern.

The importance of appropriate use of the steam generators as a heat removal mechanism during transient-initiated and small-break accidents should be a matter of careful discussion among the regulatory, vendor, and utility staffs.

The apparent lack of understanding of the importance and capability of the steam generators can be considered symptomatic of the larger concern of the lack of balance in the regulatory process discussed in Section II.C.1.a. Thus, this specific finding and recommendation should be considered as supportive to, and taken in the broader context of, the need for a reevaluation of the design basis established for nuclear plants, as discussed in general recommendation 1 in Section II.C.1.a.

Lack of Remote Vent Capability at the Reactor Coolant System High Points

During the first 5 days of the accident, two significant concerns arose because of the trapping of steam and noncondensible gases (hydrogen, xenon, krypton) in the various high points of the reactor coolant system (RCS). As discussed above, the presence of these substances in the RCS hot legs inhibited attempts to restore natural circulation cooling and impaired the accident recovery during the first day. During the subsequent 4 days, the presence of a hydrogen "bubble" in the upper head of the reactor vessel was a major concern.

It was known at that time that manual vent valves were located at both the tops of the hot legs and the top of the reactor vessel. However, these valves required local operation; because of the radiation environment in the reactor building, it was not possible to go to the valves and open them.

In the sense that the accident recovery process was hampered by the lack of remotely operable vents at high points of the RCS, the B&W plant design (as well as other reactor designs) may be considered deficient. The addition of remotely operable valves, or the modification of currently installed manual vents appears to be a desirable change.

It should be noted that the addition of remotely operable valves would not be without some negative safety implications. Such valves provide additional possible paths for losses of coolant from the RCS because of the inadvertent opening of a vent valve due to equipment failure or human error or to the intentional, malicious opening by a person. Thus, the addition of these vents increases to some extent the likelihood of a loss of coolant from the RCS. Care should be taken in the design of such a vent system to minimize the possible effects of equipment failures and human interactions.

Findings—The lack of a remotely operable vent at the reactor coolant system high points significantly impeded the recovery from the TMI-2 accident.

Recommendations—We believe that the capability to remotely vent the high points in the RCS of light water reactors is an important feature that should be provided. Because certain failures in such vents could lead to a loss of coolant from the RCS, due consideration of this possibility should be one aspect of the design requirements. Measures to reduce the likelihood of unintentional (or malicious) use of these valves also merits consideration.

Leaks in the Reactor Coolant System

Before the accident, the pressurizer relief valve was apparently leaking into the reactor coolant drain tank (RCDT) at approximately 6 gallons per minute. This continuous leakage caused the boron concentration to continuously increase in the pressurizer and the relief valve exhaust to continuously indicate approximately 180°-200°F (the normal is 130°F). (See Refs. 62 and 63 for additional details.)

Approximately 2600 gallons of water were transferred each shift (8 hours) from the RCDT to the makeup tank (MUT) via the RCDT up to the shift on which the accident occurred. During the first 4-½ hours of the shift on which the accident occurred, 1800 gallons were transferred, indicating a substantial increase in the leak rate to approximately 3600 gallons per shift on March 28, 1979.

Since plant startup, there had been leaks detected in the waste gas system, and plant documents indicate that some efforts had been made to determine the source of the leaks. Some of the identified problems apparently were not corrected prior to the accident, which caused releases to be larger than they normally would have been. Makeup tank vent valves had been suspected of leaking prior to the accident.[64]

In violation of technical specification requirements, the licensee was operating the facility, at least during the March 22-28, 1979 period, with a leak rate in excess of 1.0 gallon per minute. The operators became used to operating the plant with the excess leakage and were unable to recognize more serious leakage without larger deviations in plant parameters.

Findings— The PORV had been leaking at least since October 1978, and the discharge line temperature had been in the range of 180°-200°F.

The licensee had operated the facility in violation of technical specifications with a leak rate in excess of 1.0 gallon per minute.

The plant continued to operate with known leakage and excessive temperatures at the PORV; therefore, the operators were desensitized and unable to recognize the failure of the PORV to close after the primary system pressure was reduced.

c. Possible Deficiencies Related to the Engineered Safety Features

Reactor Building Isolation

To ensure that radiation from contaminated gases and liquids was contained within the reactor building (RB), isolation of certain piping systems was actuated by the high RB pressure (4 psig) safety features actuation signal (SFAS), which was reached approximately 4 hours after the start of the accident.

The design that provided RB isolation for TMI-2 on high pressure alone is based on postulated bounding large LOCA analyses that assume rapid increase in reactor building pressure before radiation releases resulting from the postulated fuel damage. (See Lessons Learned Task Force report[65] for additional discussion.) Other operating plants include reactor building isolation on high radiation or safety features actuation on low reactor coolant pressure. The course of the TMI-2 accident shows that the postulated sequence is invalid as the design requirement for RB isolation.

For a considerable time before isolation, radiation was released to the auxiliary building during the accident by reactor coolant letdown, reactor coolant drain tank vent, and reactor coolant pump seal injection return. Following isolation, the plant operators manually defeated the isolation signal from the reactor coolant letdown and the reactor coolant pump seal injection to place both systems, letdown and seal injection, back in operation because they judged both systems necessary for the recovery of the plant. This action further contributed to radiation releases in the auxiliary building.

Findings—The deficiency of the RB isolation system appears to be associated with (1) the lack of direct measurement of all important parameters (e.g., radiation), (2) inadequate LOCA analyses (e.g., small break) to determine accurate setpoint values of all important parameters, and (3) inadequate hardware and operating procedures that permit resetting of isolation signals and the reactivation of selected components and systems.[66]

The operator actions to defeat isolation manually demonstrate the need for reconsideration of which systems should be immediately isolated and which should be selectively isolated.

Recommendations— Transient and LOCA reanalyses should be performed to confirm important parameters for actuation of reactor building isolation, to the degree possible, from direct measurements of such parameters.

Reevaluation should be made to determine the need for removal of isolation from any component and system during an accident mitigation sequence. (See Ref. 67 for additional recommendations.)

Reactor Building Hydrogen Concentration Control

Approximately 10 hours following the initial opening of the PORV, hydrogen in the reactor building reached flammability concentration. The primary source of the hydrogen is attributed to the zirconium-water reaction in the reactor core, when the core overheated as a result of its prolonged uncovery.

The lack of an automatic hydrogen recombination system allowed the hydrogen to accumulate and then ignite, creating a pressure surge of about 28 psig in the reactor building. The building at TMI-2 is designed to withstand pressures in excess of 60 psig,[68] although some reactor buildings can only withstand pressures of about 12 psig.

The regulatory criteria, applied to TMI-2, required provisions for hydrogen recombination systems to deal with slow (several days) postaccident generation of hydrogen, following a LOCA, from (1) about 1% of clad metal-water reaction, (2) corrosion of materials inside the reactor building, and (3) radiolytic decomposition of water.[69] The primary source of hydrogen would in this case be from corrosion of materials inside the reactor building and not from the clad metal-water reaction that was the major source at TMI-2.

The provisions at TMI-2 called for postaccident installation and operation of external hydrogen recombiners. These recombiners would be hooked up at the 36-inch reactor building penetrations that were used for normal reactor building purging. However, such recombiners are not capable of preventing the rapid increase in pressure (28 psig) attributed to hydrogen ignition at TMI-2. (See Section I.B.1 of this report for additional discussion.)

Findings—The design basis applied to TMI-2 resulted in an inadequate hydrogen recombination system.

Recommendations—It appears necessary to determine more accurately the principal sources of hydrogen generation for the implementation of an appropriate hydrogen recombination system, or consideration should be given to containment designs that would not require hydrogen recombination systems.

Inadequacy of Shielding and Leakage Control of Engineered Safety Features

During the course of the accident and the postaccident recovery, significant problems arose relating to high radiation fields in the auxiliary building. These problems influenced decisions being made at the time concerning access to and work done in the auxiliary building and the method by which the RCS would be cooled down. Thus, the contamination in the auxiliary building suggests possible plant deficiencies. In this section, possible deficiencies in the radiological design of core cooling and other safety-related equipment are considered.

Pressurized water reactors typically have two complements of core cooling equipment used for normal and emergency situations: the decay heat removal (DHR) system (for normal shutdown and long term cooling) and the emergency core cooling system (for accident cooling). The DHR system provides core cooling at relatively low RCS pressures by drawing coolant from one core outlet pipe (a "hot leg"), passing it through the DHR pumps and heat exchangers in the auxiliary building, and injecting it back into the reactor vessel via either the core inlet piping (a "cold leg") or directly into the vessel downcomer.

The emergency core cooling systems in PWRs are designed first to draw coolant from an uncontaminated water supply such as the borated water storage tank (BWST) at TMI-2. Upon depletion of this tank, supply lines are switched to use water in the reactor building sump. This water is then drawn into the decay heat removal pumps and pumped either back into the RCS (if RCS pressure is sufficiently low) or to the suction of the high pressure injection pumps, with subsequent flow back into the RCS. The containment spray system uses a similar method of supplying water, drawing first from the BWST and subsequently from the reactor building sump.

Early on the first day of the TMI-2 accident the water in the RCS and the water collecting in the reactor building began to be contaminated with radioactive material being released from the damaged fuel. It soon became evident that the contaminated water could cause significant radiological problems in the auxiliary building if circulated through previously uncontaminated equipment (e.g., the decay heat removal pumps) and areas. For this reason, a method of core cooldown was chosen that would minimize the likelihood of drawing radioactive water into previously uncontaminated areas. Thus, in effect, the two options for core cooling that would have been expected to be used following an accident (i.e., emergency core cooling recirculation and long term heat removal by the DHR system) were considered highly undesirable.

The design basis radiological hazard for the DHR and emergency core cooling equipment and areas is described in Chapters 12 and 15 of the TMI-2 FSAR. Chapter 12 established the design basis upon which shielding is provided for certain components of the emergency core cooling system (ECCS). That vital equipment, which is part of the makeup and purification system or the decay heat removal system, is shielded to compensate for the assumed radioactivity levels in the reactor coolant resulting from *normal* operation of the plant. Apparently, other vital equipment that is not normally used during plant operation (e.g., in the containment spray system) is not required to have even this amount of shielding.[70] Contamination of these systems by highly radioactive coolant was apparently not contemplated within the established design basis.

A thorough evaluation should be performed to determine adequate response requirements for automatic or manual reinitiation of engineered safety features following inadvertent loss of power supply (e.g., offsite power) during a critical transient or accident mitigation sequence. (See Refs. 78, 79, 82, 83, 84, 85, and 86 for additional discussion on potential failure modes following inadvertent loss of power.)

High-Pressure Injection Control

Throughout the course of the TMI-2 accident, high-pressure injection pumps (1A and 1C) were either inadvertently tripping or were unable to start by automatic or manual commands.[87,88,89]

Failure to keep high-pressure injection (HPI) pumps operating has been attributed to control component deficiencies and to undesirable operator actions. Control switches were placed in pull-to-lock off position whenever the operator deemed it necessary to take pumps out of service. Pull-to-lock is an off-normal position prohibited by technical specifications during plant operation or accident mitigation.[90] At this off-normal position, automatic commands cannot start equipment whenever required by system conditions.

The inadvertent or deliberate placement of control switches in the pull-to-lock off position caused pumps to be inoperable when high-pressure injection was called upon (SFAS) by abnormal system conditions later in the accident sequence.

The lack of automatic override features to remove the pumps from the off-normal position or to alarm when pumps are not alined for safety injection is a deficiency that may have confused the operators regarding operability of the pumps, when the pumps would not start automatically, and the operators were unaware of the pumps' placement in the off-normal position.[91]

Other unsuccessful attempts to start pumps automatically or manually appear to have been attributed to contact bounce of latching relays[92] or degraded power supply.[93,94]

Findings—We find that, as the operators continued to be guided by pressurizer level to determine primary coolant inventory, manual actions were taken to control what was perceived, as excess coolant injection from automatic actuation of high-pressure injection pumps. The operators, at times, in anticipation of an automatic signal, placed pumps in off-normal position, thus removing them from the automatic controls.

At other times the operators, following failures in manual attempts to start certain pumps, placed these pumps in the off-normal position to prevent the pumps from starting automatically after they had commenced other pumps in operation manually.

The inability to start pumps manually is attributed to intermittent failures in latching relays, degraded power supply to the control circuit, or the operator's not completing control switch action.

Recommendations—The engineered safety features actuation signals (SFAS) should automatically remove components and systems important to safety from off-normal position and place them back to normal alinement for safety actuation. If it can be shown, however, that immediate realinement to normal is not required, the off-normal position should be indicated with an alarm to alert the operators to the system unavailability.

Control circuit components should be designed and periodically tested at expected degraded power supply conditions to ensure that they are capable of performing their intended function.

Core Barrel Vent Valves

The B&W design for a pressurized water reactor includes core barrel vent valves. These valves are installed in the upper region of the reactor vessel, and under certain conditions, the valves permit flow from the region above the core into the downcomer region. The vent valves were installed in B&W reactors to mitigate potential problems from the phenomenon of "steam binding" during a large loss-of-coolant accident. Steam binding is an effect postulated to occur in accidents in which high steam pressure above the core impedes the refilling of the core region with coolant. The vent valves are designed to relieve this pressure and thus assist in the refilling of the core region. During certain parts of the TMI-2 accident, conditions appear to have been correct for the vent valves to have opened. In this circumstance, water and steam that would have otherwise traveled into the steam generators and been cooled would be returned into the downcomer and subsequently to the inlet of the core. Thus the heat removal capability of the steam generators may have been compromised.

Efforts to resolve the importance of this issue were undertaken by the Special Inquiry Group. We have found no clear evidence to support the suggestion that significant harmful effects resulted from the presence and operation of these valves; however, analysis to resolve the issue has not, in the time available, provided conclusive answers.

Findings— Contamination of ESF and DHR equipment by radioactive coolant appears not to have been considered part of the design basis for this equipment.

Recommendations— The capability for postaccident radiation shielding and leakage control for vital equipment using potentially radioactive reactor building sump water and for long term cooling equipment (i.e. the DHR system) using potentially radioactive RCS water should be examined and, where necessary, improved in all LWRs. Accessibility to surrounding areas and equipment by plant personnel during accident mitigation and recovery should be a primary consideration in this regard. This recommendation complements a similar recommendation made by the Lessons Learned Task Force (short term recommendation 2.1.6).[71]

The lack of adequate shielding in the TMI-2 accident is indicative of the lack of consideration of accidents that could result in significant core damage. As such, the findings and recommendations related to this specific deficiency are indicative of the more general need for reconsideration of the design basis by which nuclear plants are licensed, as discussed in general recommendation 1 in the introduction to this section.

High-Pressure Injection (HPI) Bypass

One of the crucial contributors to the accident was the interruption early in the accident of HPI flow and its subsequent throttling and the increase in letdown flow from the primary, by the operators, as the accident progressed.[72,73,74] Those actions became possible only after reset (bypass) of the safety features actuation signal (SFAS), because without reset the continued presence of the SFAS would automatically reinstate equipment inservice immediately following interruption by the operators.

Emergency procedures at TMI for a number of abnormal conditions including small-break LOCAs require the immediate reset of the SFAS because of the deficiency in the ability of the HPI pumps, decay heat pumps and reactor building spray pumps to withstand runout—a condition that can cause damage to pumps from excessive vibrations. Operators are also instructed early in the LOCA procedures to prevent the primary system from filling solid by interrupting makeup flow to the reactor coolant systems. A solid reactor would be subject to overpressure transients that the operators were instructed to avoid. Such operator interventions, however, are not in compliance with the NRC stated regulatory position that credit for operator action is only given if such actions are taken 10 minutes or more after initiation of the accident signal.[75,76] This regulatory position implies that adequate design features should be in place to control automatically the mitigation of the accident for at least 10 minutes without operator intervention.

The LOCA emergency procedures for TMI-2 further instruct the operator to reactuate manually reactor building isolation and cooling following reestablishment of electric power supply in the event the offsite power sources were lost during the accident after the SFAS was reset. This instruction was placed in the procedures because it was recognized that loss of power removes the SFAS that actuated certain systems, and with return of power the SFAS must be manually reinstated. However, the instruction erroneously assumes that isolation and cooling, which is only a part of SFAS, includes safety injection. (See Section I.B.1 of this report for the historical perspective on this issue.)

If loss of offsite power had occurred at TMI-2, the emergency procedures would have been inadequate to ensure a delayed reinitiation of important safety features. The vulnerability to loss of required safety function following SFAS reset continues to exist to a varying degree in many operating plants. The NRC staff has erroneously testified before the licensing board for TMI-2[77] that the issue of safety injection reset is not applicable to TMI-2.

The NRC and B&W have failed to act on the repeated warnings from their own staff and the recommendations of the Advisory Committee on Reactor Safeguards (ACRS) to carry out their respective regulatory responsibilities to resolve the issue of reset.[78,79,80] A survey conducted by the NRC Office of Inspection and Enforcement[81] erroneously reported that adequate procedures are in place in all operating reactors including TMI-2 to cover all necessary operator actions before and after SFAS reset.

Findings— Deficiencies in operating plants continue to require operator intervention (early SFAS reset and manual control) to ensure adequate emergency core coolant injection or to prevent damage to safety components and systems. This may be improper in some circumstances.

Recommendations— Engineered safety feature systems and components should be designed to be capable, to the extent possible, of performing their intended function without operator intervention for at least 10 minutes following a real safety feature actuation signal initiation.

Findings—Resolution of the issue of the effect of the core barrel vent valves on the course of the TMI-2 accident has not been possible.

Recommendations—We recommend that an explicit assessment of the effects of the core barrel vent valves be included as part of the small-break loss-of-coolant accident analyses begun since the TMI-2 accident.

The question of the effect of these valves during the TMI-2 accident can to some extent be attributed to the lack of balance in the regulatory process discussed in the introduction of this section (Section II.C.1.a). That is, these valves were installed to compensate for a particular concern in a large loss-of-coolant accident, apparently without clear consideration of the possible effects during other types of accidents. The resolution of this issue should thus be undertaken in the context of a more systematic and integrated approach to accident analysis and general recommendation 1 discussed in this section's introduction. We believe that explicit consideration of this concern in the transient and small-break accident analysis expected to result from this more general reconsideration is also necessary.

Lack of Hot-Leg Injection Capability

There exists a capability in some pressurized water reactors (i.e., some of those designed by Westinghouse) to inject emergency core cooling water directly into the reactor coolant system hot legs in addition to the cold legs. The TMI-2 plant, like all B&W plants, does not have such a capability. We examine here whether, in situations such as that at TMI-2 where uncovery of the core occurs, a capability to pump water into the hot legs and directly onto the top of the core may be of significant benefit.

The capability for hot-leg injection to cool the top of the uncovered fuel is dependent on a number of factors. First, the area covered by the in-rushing water is dependent on the flow rate from the emergency core cooling system, so that fuel near the center of the core may not experience much additional cooling. Second, flashing of the water would be expected as it contacts the hot fuel; this steam generation, in concert with other steam generation from lower core regions, may entrain some liquid and carry it back into the hot legs or through the core barrel vent valves. This then could result in less overall cooling of the fuel, compared to cold-leg injection of equal amounts of coolant. As such, it is not readily apparent that the availability and use of a hot-leg injection capability would have enhanced the cooldown of the fuel in this accident.

Findings—It is not readily apparent that the lack of hot-leg injection capability in B&W plants significantly affected the course of the TMI-2 accident.

Adequacy of Debris Protection for the Reactor Building Sump

During the course of the accident, the gradual depletion of the primary water supply for the ECCS (the borated water storage tank) raised the possibility that ECC recirculation from the reactor building sump would be necessary. In the consideration of this, two concerns relating to the desirability of using the sump water arose. The first concern was the possibility that debris might have entered the sump that could then be drawn into the ECC equipment and cause damage. The second concern was radioactive contamination of the sump water. Because this water would have been drawn out into the ECC equipment in the auxiliary building, undesirable additional contamination of that building would have occurred. This latter concern is discussed separately in a previous section on the inadequacy of shielding and leakage control of engineered safety features.

The reactor building sump design was considered in the licensing of TMI-2 to be part of the engineered safety features systems and as such was discussed in Chapter 6 of the Final Safety Analysis Report (FSAR).[95] The FSAR specifically addresses sump debris elimination and indicates that the sump is completely enclosed in screens that minimize the likelihood of debris entry into the sump. Thus, for the conditions experienced during the TMI-2 accident, it appears that debris blockage of the reactor building sump should not have been a significant concern.

Findings— The reactor building sump design appears to have been adequate to protect vital equipment from debris damage in the event of sump water use in the recirculation mode of emergency core cooling.

Diesel Generator Lockout

The emergency diesel generators started automatically by the safety features actuation signal (SFAS) about 2 minutes into the accident. These diesels provide an alternate onsite power supply to equipment important to safety in the event of loss of the offsite power sources.

Shortly after their start, the diesel generators were turned off by the operators, as instructed by procedures.[96,97] This was done after it was established that offsite power was not lost and the diesels were running unloaded. The diesels at TMI-2 are not designed for prolonged operation unloaded because the exhaust system can be damaged from excessive carbon deposits. Unloaded operation is only permitted for 30 minutes. Therefore, following SFAS reset the diesels were turned off. Diesel generators of a different design can run unloaded at sufficient length of time without damage from excessive carbon deposits.

To prevent subsequent restarts of the diesel generators following reinitiation of SFAS, the operators defeated the automatic starting capability by shutting off the fuel at the fuel injectors in the diesel rooms.

Shutting off the fuel to the diesels left the plant vulnerable to total loss of ac power supply in the event of loss of offsite power. The diesels could have been made available at a later time, however, if it was recognized in time that the fuel was shut off. However, operator interviews have revealed that the principal operating staff was not aware at all times that the fuel was shut off.[97]

At a later time (9:30 a.m.) when the station electrical engineer arrived at the site, he instructed the operating staff to reset the diesel fuel racks and control the diesels at the control room. Control switches used to place the diesels out of service during maintenance were placed on manual control at that time.[98]

The inability of the diesel generators to run unloaded was acknowledged and accepted by the NRC staff for the TMI-2 and other plants currently operating. Acceptance was based on the postulate that the need for the diesels would only occur simultaneously with an accident or transient. This postulate has been contested by individual NRC experts and the ACRS.[86,99,100] The ACRS since 1976 has requested a generic resolution for this issue. However, the NRC staff has not acted and has not included this issue for resolution in any of their generic issues submitted to Congress.

Our review of the emergency procedures on LOCA[101] has revealed that instructions to the operator to reinitiate safety injection manually after SFAS reset and following loss of offsite power would not have resulted in the appropriate safety features actuation for safety injection. In recognition of the potential for loss of offsite power the instructions in the procedures call for manual reinitiation of reactor building isolation and coolant actuation. This actuation, however, is independent of the safety injection initiation of the SFAS, and therefore, if offsite power was lost following SFAS reset, injection systems would not have functioned properly.

Resetting of the SFAS gives the operator an opportunity to take manual control of components and systems that have actuated automatically by the SFAS. The actuated components normally seal the actuation by their individual controls, and therefore, removal of the actuating signal (reset) would not affect their actuated status. However, loss of power to these components will drop them from the actuated status, and restoration of power will not return them unless the SFAS is present.

Findings— There is no evidence of any formal analysis by the NRC or the licensees and their suppliers of the consequences of interruption of engineered safety features at any time during a transient or accident mitigation sequence. The procedures for manual reinitiation do not take into account the consequences of the interruption prior to the manual reinitiation.

The deficiency in the diesel generators to run unloaded without damage resulted in an insufficient redundancy in power supply during a crucial period of the accident.

Recommendations— Analysis should be performed to determine the consequences of inadvertent interruption of engineered safety features from loss of power at any time during a transient or accident mitigation sequence.

If the analysis shows that interruption of engineered safety features is unacceptable for any interval of time before automatic restoration of power from another source (e.g., diesel), consideration should be made for (1) simultaneous paralleling of offsite with onsite power supplies by SFAS, (2) simultaneous paralleling of offsite with only one train of onsite power supply by SFAS, or (3) either enhancing or removing available offsite power from the engineered safety features during a transient or accident mitigation sequence.

Decay Heat Removal System Not Designed for Operating Pressures

The decay heat removal (DHR) system in pressurized water reactors is designed for use during a normal plant shutdown rather than during accident situations. It is designed for use after the plant has been cooled down and depressurized by other systems (e.g., the emergency feedwater system) to relatively low temperatures and pressures. After this is accomplished, the DHR system is initiated to provide the long term cooling of the reactor core.

About 7 hours into the accident an attempt was made to depressurize the reactor coolant system from high pressures (about 2100 psia) to pressures at which the DHR system could have been used (about 300 psia). It was believed by the operating crew that the use of the DHR pumps, which have a much higher pumping capacity than the makeup pumps, would more quickly reduce the temperatures seen in the reactor coolant system.[102] However, the pressures in the RCS could not be decreased sufficiently low to use these pumps.

We have studied whether the relatively low design pressure of the DHR system is a plant deficiency that was detrimental to the recovery from this accident. In one sense the low design pressure is a deficiency, in that it did not permit use of the DHR pumps at the time period discussed above. In another sense the low design pressure of the DHR system is not a deficiency. For accidents such as that at TMI-2, where reactor coolant system pressures remain high, another cooling system with the capability to operate at high pressures is designed and installed, this being the high pressure injection (HPI) system. A DHR system designed for high pressures thus may be considered a backup system to the HPI system.

In the TMI-2 accident the high pressure injection system was automatically actuated and began to operate as designed a number of times. Subsequent crew actions reducing the flow from the HPI system greatly compromised the capability of the system and were the direct cause of the damage to the core. The apparent need for the DHR system (as perceived by the TMI-2 crew) is thus predicated on their prior actions that compromised the capability of the high pressure injection system.

An additional point to be made deals with the possible effects if the RCS pressure had dropped sufficiently low to allow DHR system operation. First, indications available to the operating crew during this time period on hot-leg conditions suggested that the legs were filled with superheated steam. Because the DHR pumps draw coolant from one of the hot legs, superheated steam or a steam-water mixture might have been drawn into the pumps, with uncertain consequences. Further, RCS coolant was highly radioactive by this time, meaning that contamination of the DHR system and surrounding areas in the auxiliary building would also result. Thus, a switch to using the DHR system during this time period may have worsened the situation rather than improved it.

A decay heat removal system designed for operating pressures thus may be thought of as additional equipment redundant to the high pressure injection system. This additional redundancy of equipment has the potential for somewhat improving the reliability of the high pressure cooling function. However, it seems likely that operator actions to compromise one system, as was the case with the HPI system of TMI-2, could also compromise any additional equipment. It is therefore not readily apparent that the lack of a decay heat removal system designed for operating pressures is a significant deficiency contributing to the accident at TMI-2.

Findings—A decay heat removal system designed for operating pressures would in essence be additional equipment redundant to the high pressure injection system. It is not clearly evident that the presence of such a system would have significantly altered the course of the TMI-2 accident.

d. Possible Deficiencies Related to the Secondary Coolant System

Emergency Feedwater Actuation and Control

Loss of main feedwater, which initiated the accident, resulted in the actuation of the emergency feedwater system—the emergency feedwater pumps were performing at full pressure within 40 seconds. However, because the discharge block valves were closed, feedwater did not enter the steam generators until 8 minutes into the accident after the block valves were manually opened. The steam generators automatically rose to a design level of 34 inches for recovery from a loss of feedwater transient as opposed to 32 feet (75% full) that the B&W analysis postulates for small-break accidents. (See Ref. 103 for additional details regarding the B&W analysis for small-break LOCAs.)

By the time the block valves were opened, the steam generators had boiled dry, the PORV had failed in the open position, and high pressure injection actuation had been initiated. Hence, a small-break LOCA was in progress and the emergency feedwater system should have supplied water to the steam generators to raise the level to 32 feet—the level required for successful mitigation of small-break LOCAs. The emergency procedures for TMI-2[104] did not include instructions for steam generator level requirements for the mitigation of small-break LOCAs. Additional studies, however, following the accident have resulted in revised procedures that include specific level requirements for small-break LOCAs.[105,106,107] It should be recognized, however, that the emergency feedwater enters the steam generator at the 32-foot level at TMI-2 and sprays down the tubes to the liquid level, thereby providing high level cooling whenever the steam generator is being fed.

According to B&W[108] high emergency feedwater level control is significant for the mitigation of small-break LOCAs. The analysis presented to the NRC by B&W in topical report BAW-10075A, Rev.1, was based on a 32-foot emergency feedwater level. This level, however, and its significance to mitigation of accidents were not reported to the NRC and were not included in the TMI-2 small-break LOCA emergency procedures.[109]

Operator interviews have indicated that the steam generator level at TMI-2 during emergencies is supposed to be 21 feet.[110] It is uncertain, however, whether the course of the accident would have been altered even if the 21-foot level was automatically reached, because an analysis does not appear to have been made by B&W for the 21-foot level and the reactor coolant pumps running.[111]

The 21-foot emergency water level in the steam generators that the operators thought was proper might have been reached during the accident if high pressure injection actuation had been coincident with loss of offsite power (reactor coolant pumps tripped).[112] However, because offsite power was not lost at TMI-2, the integrated control system (ICS) controlled the steam generator level at only 34 inches because the ICS did not recognize the incident as a small-break LOCA.

A design feature that controls steam generator level at 34 inches during feedwater transients appears to have been desirable to maintain pressurizer level indication by limiting shrinkage in the primary coolant. The need for dual level setpoint in the steam generator had become apparent in another B&W operating plant in the past. B&W did not inform its customers or the NRC of the deficiency in the control system to recognize small-break LOCAs with reactor coolant pumps running.[113]

The deficiency of the system design to recognize properly the steam generator level requirement of 32 feet may have contributed to the high pressurizer level indication, which the small-break LOCA emergency procedures do not predict would occur. Emergency procedures for small-break LOCAs predict low pressurizer level. Hence, the operators did not apply the small-break LOCA procedures and continued to throttle high pressure injection to prevent the primary system from filling solid.

Findings—Surveillance performed on the emergency feedwater system on March 26, 1979, resulted in the closure of the block valves (EF-V12A and B). The surveillance procedure allowed the simultaneous closure of the block valves when testing emergency feedwater pump operability. Such closure was required because of the known deficiency in the emergency feedwater level control valves (EF-V11A and B) in preventing leakage to the steam generators whenever the pumps were tested.[114,115]

The emergency feedwater system was not designed to respond properly to a small-break LOCA; that is, to fill the steam generators to the emergency level required for successful mitigation of the accident.

Met Ed and B&W failed to integrate emergency feedwater response with the proper accident analysis. (See Section I. B.1 for the historical perspective on this issue.)

Recommendations—Surveillance procedures should not permit the simultaneous defeat of redundant systems important to safety.

The emergency feedwater system should be designed, at minimum, with a diverse and redundant automatic SFAS actuation of pumps, discharge valve alinement and emergency steam generator level. This automatic actuation should be independent of the ICS.

Condensate Polisher

The condensate polisher removes impurities from the turbine steam condensate by means of deionizing resin. The polisher is part of the station condensate feedwater system that supplied water to the steam generators of the nuclear steam supply system.

There are eight polisher units, with any seven in operation at one time. The eighth is free to have the resin bed regenerated. Regeneration consists of removing a polisher from service, transferring the resin bed to a regeneration skid, regenerating, and returning the resins to the polisher. Each polisher is equipped with an air-operated inlet, inlet bypass, and outlet valve.

The condensate feedwater system is not considered safety-related equipment based on the fact that the total loss of normal feedwater is an analyzed accident.[116] Therefore, this equipment was not inspected as rigorously by Met Ed as it would have been if it had been classified safety related.

The condensate polisher was originally designed for the Oyster Creek No. 2 plant, which never materialized, and was transferred to the Three Mile Island facility. Early in the fabrication process, a design change was incorporated to have the inlet and outlet valves of each polisher unit fail in the "as-is" position on loss of air or power.[117] The equipment for this design change was installed and the essential component calibrations and electrical tests were performed in preparation for the functional test.[118,119] The equipment functional tests failed to verify the "fail as-is" feature.[120] Apparently, subsequent to the calibration and electrical tests, the control wiring to the solenoid valves that effect the "fail as-is" feature were disconnected.[121] In addition, at the time of the accident, these solenoid valves had a manual override feature that was actuated, so that even if the wires had been connected, the "fail as-is" feature would have been bypassed. Further investigation did not produce evidence to indicate that this was an authorized modification of the equipment. We can only speculate why this feature was disarmed. Some possibilities are that the actual design was unworkable, that there was improper installation, that the previously mentioned tests were never performed, or that there were incorrect operating procedures, incompetent operators, or incompetent maintenance.

The disarming of this feature could possibly have contributed to the initiation of the loss-of-feedwater transient that ultimately resulted in the accident. This will be discussed in detail below.

On February 19, 1977, it was identified by the General Public Utilities Startup Group that the transfer of resins could not be accomplished satisfactorily without the injection of service air to disperse the resins. Consequently, a ¾-inch diameter pipe was installed, connecting the service air header to each individual polisher unit.[122] Here again, this minor alteration may have played a part in the initiation of the feedwater transient and will be discussed later.

Either the design or the installation (or both) of at least the electrical systems in the condensate polisher were of questionable quality. From August 30, 1976 to October 6, 1977, there were 28 electrical work requests issued against the condensate polisher.[123] It is possible that the disconnected solenoid valve control wires occurred at this time.

The final acceptance of the condensate polisher occurred on November 17, 1977.[124] On October 19, 1977, just before this acceptance, water was noted in the service and instrument air systems.[125] This water caused, directly or indirectly, the outlet valves on the condensate polishers to close and resulted in a loss of feedwater. Fortunately, the facility was not at power and no adverse effects were noted. However, the author of the report stated that, "If this would have happened while at power, the unit would have been placed in a severe transient condition . . . ." This resulted in a recommendation by Metropolitan Edison that the Architect-Engineer, Burns and Roe, consider installing an automatic valve to bypass the condensate polisher on high differential pressure or low flow conditions. This recommendation was rejected on November 17, 1977.[126]

Again, on May 12, 1978, water was inadvertently introduced into the service and instrument air system. The operator felt that the water resulted from the failure to close the individual air valve on one polisher unit before it was returned to service.[127] Two memoranda were written on May 15 and 16, 1978. One recommended installation of an automatic bypass around the polishers and isolation of the instrument air from the service air system.[128] The second memorandum endorsed the first and directed immediate action to be taken.[129] No evidence of the directed actions could be found.

The facility experienced a trip from 90% power on November 3, 1978, because of a loss of feedwater transient. The master power switch to the condensate polisher control panel was inadvertently de-energized by a technician. This caused the outlet valves on the condensate polishers to close again.[130] If the "fail as-is" feature had been properly installed, this trip should not have occurred. The loss of power caused valve position control solenoids to dump the pneumatic signal air. If the "fail as-is" feature had been armed, it would have blocked this loss of signal, freezing the inlet and outlet valves in position.

A change in the operating procedure was initiated on January 25, 1979, in an attempt to control the valve positions—in essence, to treat the symptom rather than the cause. The change directed that local air switches for the inlet, inlet bypass, and outlet valves be placed in the manual-open position.[131]

Finally, on March 27, 1979, at 4:00 p.m., 12 hours before the start of the major accident, a resin transfer from the No. 7 polisher was started. The operator noted in his log book at 11:00 p.m. (the shift change), "Relieved shift resin clogged."[132] There are no more significant entries until April 1, 1979, because at approximately 4:00 a.m. on March 28, 1979, the condensate polisher discharge valves closed once more, unexpectedly, initiating the accident.

Metropolitan Edison, the NRC, the President's Commission, and this Special Inquiry Group have not been able to establish conclusively the exact cause of these valves closing. The following is a possible account based on the current facts as reviewed by this special inquiry.

During the early postaccident days it was felt that the condensate polisher service air connection allowed water to flow from the condensate polisher units, back through the service air system, through the service-instrument air cross-section, and out through the instrument air system to the condensate polisher control panel.[133] The amount of water required to flood these systems can range from 3000 to 6000 gallons. For example, there are several air receivers with volumes as follows:

Instrument Air Receivers

IA-T-1A	57 ft^3	426 gallons
IA-T-1B	57 ft^3	426 gallons

Service Air Receivers

SA-T-1A	96 ft^3	718 gallons
SA-T-1B	96 ft^3	718 gallons
SA-T-1C	96 ft^3	718 gallons
SA-T-2	235 ft^3	1757 gallons
SA-T-3	235 ft^3	1757 gallons
		6520 gallons

The 6520 gallons of water does not include the volume of the piping and the fact that there are automatic water drains throughout the two systems. Even if one assumes that the tanks never become full, it is reasonable to postulate that several thousand gallons of water are required to cause water to be seen at the condensate polisher control panel. If one assumes a 10-gallon per minute flow rate (which is high) through a ¾-inch diameter pipe, it would take 5 hours to fill 50% of the 6000-gallon capacity air receivers. If one assumes a more realistic 5-gallon per minute flow rate, it would take 10 hours to fill 50%. It is interesting to note that the No. 4 condensate polisher was put into service 12 hours and 40 minutes before the accident.[132] If an operator inadvertently had left the service air valve open on No. 4 polisher and returned it to service, it could have been the source of the large volume of water required to create the transient. It is very doubtful that the intermittent opening and closing of the service air valve on the No. 7 polisher to unclog resins could have been the source of all that water. The only other explanation is that during the transfer of resins from the No. 7 polisher the service air valve and the transfer water valve were left open simultaneously for the same extended periods.

It has been postulated that water alone caused the valves to close. In view of the following analysis, it is more plausible to hypothesize that the water caused a partial or total loss of instrument air. This is supported by several other facts. The first is that the operator experienced difficulty in getting the air operated emergency feedwater valves EF-11A and EF-11B open immediately after the reactor trip.[134] Second, it would not be expected that water interfering with a pneumatic valve operator would cause all valves to close simultaneously. Third, the condensate polisher differential pressure recorder, which is air operated, went to zero differential.[135] It is known that at least one condensate pump continued to run for a considerable time after the transient.[136] Based on this, the differential pressure would have attained some value above zero. Therefore, it appears that the instrument lost operating air pressure.

It appears now that a single component, such as the instrument air dryer, was affected by the water that caused a loss of instrument air, thus causing the condensate polisher valves to close, the differential pressure recorder to give false indication, and the emergency feedwater valves not to respond.

Lack of Automatic Bypass on the Demineralizer – Polisher

The initial loss of main feedwater at the start of the accident before the reactor trip has been attributed to resin clogging the condensate polishers, which resulted in the closure of the polisher outlet valves.[137] Bypass valves, COV-12, around the polishers are manually controlled from the control room, and therefore the initial transient probably could not have been prevented because it is unlikely that the operators could have acted quickly enough to have prevented reactor trip and the subsequent high pressure injection. Automatic actuation of the bypass valves with isolation of inlet or outlet valves at the polisher could have maintained main feedwater flow and have prevented the PORV from opening. Operators have indicated that automatic bypass valves at TMI-1 have prevented similar transients from occurring.[138]

Efforts to open the bypass valve from the control room failed because the valve had previously been jammed in the closed position,[139,140] making the motor operator unable to unseat the valve. The motor breaker was tripped by the torque limiting switches (that protect the motor) whose settings were exceeded. A description of the functional performance of the feedwater system is included in the plant FSAR[141] and the EPRI report.[142]

Instrument Air System

The loss of the main feedwater pumps, which initiated the turbine trip followed by a reactor trip, has been attributed to the presence of water in the instrument air system that caused the condensate polisher air operated outlet valves to close.[143,144] It is postulated that water at 100 psig in the condensate polisher entered the service air system, which is at 80-100 psig, through a failed-open check valve. Station service air used to free blockage in the resin transfer line is cross-connected with the instrument air system. Inadequate capacity in the instrument air system caused the licensee to cross-connect the service air to the instrument air as a normal mode of operation of the two systems. The mode of operation for air supply on the day of the accident was the cross-connected system.

The Met Ed crew had installed air dryers at various points in the instrument air system to prevent the accumulation of moisture. In particular, an air-water separator was installed in the condensate polisher instrument air line in series with two pressure regulators. This arrangement processed all air to the condensate valve controls and instruments located on the condensate polisher local control panel.

"Met Ed has performed tests on the condensate polisher instrument air system subsequent" to the accident and has indicated that upon isolation of "instrument air from the condensate system, the condensate outlet valves for each polisher tank go closed." However, the tests also "indicated that introduction of water into the air system did not affect the polisher outlet valves, in that the air-water separator functioned properly."[145]

Findings— We have been unable to establish conclusively the exact cause of the valve closing that led to the failure of the condensate polisher system, which initiated the TMI-2 accident, although a reasonable scenario has been developed.

In addition to the condensate-feedwater system, the instrument air system, which supplies motive power for the emergency feedwater control valves, was compromised by the cross-connected operation with the service air system. Only selected components of the instrument air system are quality group classified.

There was ample evidence that the condensate polisher was not trustworthy and was capable of inducing "analyzed accidents." The warnings of the operators were not heeded and the NRC inspectors were apparently not charged with the responsibility of identifying the problem. Because the condensate-feedwater system was not safety related it was beyond the purview of the NRC.

The inadequate capacity of air systems resulted in a compromise of the independence of an instrument air system from a process system whose use resulted in the disturbance to the plant. Although safety design criteria for instrument air systems postulate the loss of air supply to cause systems important to safety to be placed in a fail-safe mode, the failure mode of control or process systems is generally not known. Hence, limiting the interactions between control and safety systems could minimize plant disturbances.

Recommendations— The distinction between "safety" and "nonsafety" related systems should be replaced by a graded scale of significance.

It is understandable that certain systems and components should not be considered safety related. However, some mechanism must be established to control peripheral systems, such as the condensate-feedwater, that can initiate transients that challenge the reactor's protection systems.

System designs should consider implementation of piping configurations that can permit periodic testing of valves at system conditions (e.g., differential pressure, temperature, etc.) expected during emergencies. Proper torque switch settings could be verified by comparison of the power-torque delivered to the valve assembly during a test with the maximum setting of the torque switches for valve motor trip.

Interconnections of control, process, and safety systems should be limited unless suitable isolation can be provided to ensure that failures in the control or process systems do not cause unacceptable plant disturbances.

Condenser Hotwell Control

Following the initial turbine trip and closure of the main steam isolation valves, steam release to the main condenser continued through the turbine bypass valves.[146] However, in the course of the accident, the hotwell level control valve controller failed in the low-level setting and caused the hotwell to be flooded from the condensate storage tank. The failure of the level control valve controller caused the hotwell makeup valve to remain open, allowing condensate storage tank water to flood the hotwell and interrupt steam release to the condenser.

Subsequent to the flooding of the hotwell, the operators attempted to reduce the level by discharging the hotwell to the condensate storage tank through a condensate pump.[147,148] However, failure of the hotwell level reject valve did not permit the discharge until about 3 hours into the accident and after the reject valve was manually opened. For general discussion of the hotwell control, see the EPRI report.[149]

Following recovery of hotwell level, the condenser vacuum started to decrease and eventually was lost. Condenser vacuum is also a required function to maintain the ability to release steam to the condenser. Loss of vacuum resulted from loss of the auxiliary boiler that provides sealing steam for the interface between the turbine shell and the main shaft.

The important decay heat removal through the secondary was interrupted (at about 9 hours) when the main turbine condenser was lost (because of failures in the hotwell level control and condenser vacuum), and the atmospheric dump system was ordered stopped. For the period of time that the secondary system heat removal was lost, the operators maintained primary pressure control by releasing primary coolant inventory through the PORV block valve.

Findings—It appears that the importance for removal of decay heat through the secondary was not well recognized by the operating staff throughout the accident.

Recommendations — An assessment should be made to determine the extent to which the secondary heat removal systems should be designed to ensure their continued availability during postulated transient and accident conditions.

If the condenser steam dump or the atmospheric dump systems are required to maintain the plant in a safe condition for a range of transients or accidents, as a minimum, the controls and power supply for these systems should be designed according to criteria for systems important to safety.

e. Environmental Qualifications and Use of Instrumentation and Plant Data

The reliability and the accuracy of the information that was available to the operators during the TMI accident and to the investigators after the accident have been the subject of much discussion. In this section an attempt is made to document the environmental qualifications of the instrumentation and to summarize the uncertainties in the data recorded at TMI during the accident.

Environmental Qualifications of Instrumentation

Instrumentation within the TMI reactor building, if part of the reactor protection system (RPS) or safety features actuation system (SFAS), was required to survive and function under the following environmental conditions:[150,151]

Normal Conditions	*Postaccident Conditions*
40–120°F, atmospheric pressure, 40–70% relative humidity, and 25 mR/h	286°F, 51.3 psig, 100% humidity, and total 2×10^4 roentgens (24-hour operability)

Cables were generally qualified according to more stringent environmental requirements. From what we know, there is no reason to believe that the TMI accident environment should have damaged the RPS and SFAS systems in the first day of the accident.

Other instrumentation was classified according to whether it was or was not required for safety. The former category included instrumentation required for accident monitoring and for safe shutdown. Some of the instrumentation in each subclass was also contained in the RPS and SFAS and, therefore, was qualified according to the higher environmental conditions. Accident monitoring instrumentation (Table II-44) was also designed to operate in the postaccident environment. However, instrumentation required for safe shutdown (Table II-45) was not required to be qualified to these conditions, unless it also formed part of the RPS or SFAS.

The most vital data in accident situations are from accident monitoring instrumentation. It is clear, too, that systems and controls designed for safe shutdown are also vital for postaccident management. In addition, there is a clear need for instrumentation to enable the plant to be maintained in a stable, safe condition after shutdown.

The least severe qualifications were required of instrumentation that was thought to be "not required for safety." This category included such systems as automatic reactor coolant pressure control, pressurizer temperature measurement, automatic pressurizer level control, the integrated control system, and the control rod drive control system.

Limits of Operability

The ranges of operability of instrumentation systems (the maximum ranges of the transducers) are shown in Table II-46. The ranges of indication available to the operators are shown in Table II-47. Small excursions past the limits of operability should not damage instrumentation systems. However, ex-

TABLE II-44. Accident monitoring instrumentation

End-Point Parameters	Recognize Accident Condition	Functioning of Mitigating Equipment	Follow Course of Accident/ Transient	Required for LOCA		
				Large	Small	Transient
ESF busses energized		X		X	X	X
Pressurizer Level	X		X			X
SG Press.	X		X			X
RC Press. (wide range)	X	X	X	X	X	X
RC System Flow	X					X
Containment Press.	X	X	X	X	X	X
Emer. Feed. Press.		X				X
Containment Isolation		X		X	X	X
Area Rad. Monitor & Grab Sampling	X		X	X	X	
RC Temp. hot/cold	X		X		X	X
DH Cooler Outlet Temperature			X	X	X	X
DH Pump Suction Temp.			X	X	X	X
HPI Flow		X	X	X	X	X
LPI Flow		X	X	X	X	X
BWST Switch-over Valves		X	X	X	X	
Feed Latch (valve indication)	X	X				X
H_2 Content (grab sample)			X	X	X	
SG Level (Startup & Operate Range)			X		X	X

TABLE II-44. Accident monitoring instrumentation—Continued

End-Point Parameters	Recognize Accident Condition	Functioning of Mitigating Equipment	Follow Course of Accident/ Transient	Required for LOCA Large	Required for LOCA Small	Required for Transient
Reactor Bldg. Spray Pump Flow		X		X	X	X
Pressurizer Electromatic Relief		X				X

TABLE II-45. Systems required for safe shutdown

Control Rod Drive Control System
Makeup Pump Control
Letdown Line Isolation Valve Control
BWST Suction Valve Control
EFW Control
Pressurizer Spray Valve Control
Electromatic Relief Valve Control
Decay Heat Removal System Controls
Nuclear Services Closed Cooling Water System
Nuclear Services River Water System
Supporting Systems (Electrical, Air, etc.)

cursion past the indicating limits means that the information is not available to the operators.

Acceptability of Plant Data

The acceptability of data depends on a number of factors, some of which are subjective and difficult to quantify. Sensors, signal conditioning equipment, data display devices, and data recording devices are all subject to some inherent error. In addition to the error in equipment that is nominally in good working order, there is a problem of reliability; that is, some instruments break down. Finally, there is the question of utility; vitally needed data might be accepted and used even if the accuracy and reliability cannot be guaranteed.

Accuracy of the TMI Data

Required frequency of calibration and accuracies of classes of data are specified in the Final Safety Analysis Report (FSAR),[150] the technical specifications,[152] and the surveillance procedures.[153] It should be expected that instrumentation in good repair will always fall within the accuracy limits shown in Table II-48.

Errors found at the most recent calibration of selected instruments are given in Table II-49. It can be seen from this table that the requirements of the FSAR were met, at least immediately after calibration. When several components are cascaded, the overall error is approximately the algebraic sum of the error of individual components. For example, if a sensor, a bridge network, a compensator, and a recorder each have errors of 0.5%, the overall error when cascaded is approximately 2% (2.015%, exactly).

Two sources of error are not covered either by specifications or calibrations: reading error and chart timing error. Reading error of charts or meters is governed by the width of the recording or indicating band and by the fineness of graduations. A chart on which the recording is spread out over a wide band can obviously be read with greater accuracy than one on which the reading is tightly crowded into a narrow space. Likewise, finely graduated charts or meters can be read with higher accuracy than coarsely graduated charts. However, coarse graduations can often be more easily read quickly.

As a rule of thumb, it is estimated that reasonable reading accuracy to one-half the finest graduation is possible; however, on a few very finely graduated charts, accuracy is considered reasonable only to the finest whole graduation. Table II-50 shows achievable reading accuracy for a number of strip charts. It will be seen that each channel should be

TABLE II-46. System ranges

Item	System Desig.	Ind. Type	Range
Reactor Coolant Temperature	RC-5A-TE2 (Cold Leg)	Indicator	50–650°F
	RC-5A-TE-4 (Cold Leg)	Indicator	50–650°F
	RC-5B-TE3 (Cold Leg)	Indicator	50–650°F
	RC-5B-TE4 (Cold Leg)	Indicator	50–650°F
	RC-15A-TE1 (Hot Leg)	Recorder	0–800°F
	RC-15A-TE2 (Cold Leg)	Recorder	0–800°F
	RC-15A-TE3 (Cold Leg)	Recorder	0–800°F
	RC-15B-TE1 (Hot Leg)	Recorder	0–800°F
	RC-15B-TE2 (Cold Leg)	Recorder	0–800°F
	RC-15B-TE3 (Cold Leg)	Recorder	0–800°F
Reactor Coolant Pressure (SFAS Input)	RC-3A-PT3	Recorder	0–2500 psig
	RC-3A-PT4	Indicator	0–2500 psig
	RC-3B-PT3	Indicator	0–2500 psig
	RC-2-TE2	Indicator	0–700°F
Level	RC-1-LT2	Recorder	0–400 in
	RC-1-LT3	Recorder	0–400 in
Pressurizer Temperature	RC-2-TE1	Indicator	0–700°F
	RC-2-TE2	Indicator	0–700°F
OTSG A Level	SP-1A-LT1	Indicator	0–600 in
	SP-1A-LT2	Recorder	96–388 in
	SP-1A-LT3	Recorder	96–388 in
	SP-1A-LT4	Indicator	0–250 in
	SP-1A-LT5	Indicator	0–250 in
OTSG B Level	SP-1B-LT1	Indicator	0–600 in
	SP-1B-LT2	Recorder	96–388 in
	SP-1B-LT3	Recorder	96–388 in
	SP-1B-LT4	Indicator	0–250 in
	SP-1B-LT5	Indicator	0–250 in

TABLE II-47. Information readouts available to the operator for monitoring conditions in the unit

Measured Parameters	Total No. Reqd. Ch.	No. of Ch. Available[1]	No. of Sensors in a Channel	Types of Readouts	No. of Readouts[2]	Indicator Range	Indicator Accuracy[3]	Indicator Location	Purpose or Usage
Source Range Neutron Level	1	2	2	B,F	3	10^{-1} to 10^{+6} cps	±3	A,B,D	A
Source Range Startup Rate	1	2	2	A,F	3	-1 to 10 dpm	±3	A,B,D	A
Intermediate Range Neutron Level	1	2	2	B,F	3	10^{-11} to 10^{-3} amp	±3	A,B,D	A,B
Intermediate Range Startup Rate	1	2	2	A,F	3	-1 to 10 dpm	±3	A,B,D	A
Power Range Neutron Level	3[2,4]	4	4	A,F	3	0 to 125% FP	±2	A,B,D	A(B)
Power Range Neutron Level Imbalance	3[2,4]	4	4	A,F	3	-62.5 to 62.5% FP	±2	A,B,D	A(B)
RC Loop Outlet Temp.	2(1/Loop)	6(3/Loop)	3/Loop	A,E,F	4/Loop	520°-620°F	±2	B,C,D	B
RC Unit Outlet Temp.	—[5]	—[5]	—[5]	E	1	520°-620°F	±2	B	
RC Loop Inlet Temp. (Narrow Range)	2(1/Loop)	4(2/Loop)	4/Loop	A,E,F	4/Loop	520°-620°F	±2	B,D	B
RC Loop Inlet Temp. (Wide Range)	2(1/Loop)	4(2/Loop)	2/Loop	A,F	2/Loop	50°-650°F	±2	B,D	B
RC Unit T_c	—[5]	—[5]	—[5]	A	1	520°-620°F	±2	B	
RC Loop Avg. Temp.	—[5]	—[5]	—[5]	A	1/Loop	520°-620°F	±2	B	
RC Unit Avg. Temp.	—[5]	—[5]	—[5]	E	1	520°-620°F	±2	B	
RC Loop Temp. Diff.	—[5]	—[5]	—[5]	A	1/Loop	0-70°F	+2	B	
RC Unit ΔT_c	—[5]	—[5]	—[5]	A	1	-10°-10°F	±2	B	
RC Loop Pressure (Wide)	1	1	1	A,E	2	0-2500 psig	±2	A,B,C	B
RC Loop Pressure (Narrow)				A,E,F		1700-2500 psig		A,B,D	
Pressurizer Level	1	3	3	A,E,F	3	0-400 in H_2O	±2	A,B,C,D	B

Pressurizer Temp.	1	2	2	A,F	2	0–700°F	±2	B,D	B
RC Loop Flow	2(1/Loop)	2(1/Loop)	2/Loop	A,F	2/Loop	0–90x10^6 lb/h	±3	A,B,D	B
RC Total Flow	—[5]	—[5]	—[5]	E	1	0–180x10^6 lb/h	±3	A,B,E	
Steam Gen. Full Range Level	2(1/Loop)	2(1/Loop)	1/Loop	A,F	2/Loop	0–600 in H_2O	±2	B,D	B
Steam Gen. Startup Range Level	2(1/Loop)	2(1/Loop)	2/Loop	A,F	3/Loop	0–250 in H_2O	±2	A,B,C,D	B
Steam Gen. Operate Range Level	2(1/Loop)	2(1/Loop)	2/Loop	E,F	2/Loop	0–100%	±2	B,D	B
Emergency FW Status	2(1/Loop)	1/Loop	1/Loop	C,F	2/Loop	–	–	B,D	
Emergency FW Press.	2(1/Pump)	2(1/Pump)	1/Pump	A	1/Pump	0–100%	±2	B,E	B
Containment Pressure (RPS)	2	4	1	A,E,F	3	0–100 psig	±1	B	
(SFAS)	2	3				(–5 psig to –10 psig)			
Containment Isolation Status	4	4	1/Valve	C	1/Valve	–	–	B	
Containment Temp.	—[5]	—[5]	20	A	1	0–300°F	±2	B	
Steam Gen. Outlet Press.	2(1/Loop)	4(2/Loop)	2/Loop	A,E,F	3/Loop	0–1200 psig	±2	A,B,C,D	B
Steam Temperature	2(1/Loop)	4(2/Loop)	2/Loop	A,F	2/Loop	100°–650°F	±2	B,D	B
Startup FW Flow	2(1/Loop)	2(1/Loop)	1/Loop	A,E,F	3/Loop	0–1.5x10^6 lb/h	±2	B,D	A
Main FW Flow	2(1/Loop)	4(2/Loop)	2/Loop	A,E,F	3/Loop	0–6.5x10^6 lb/h	±2	B,D	B
Feedwater Temperature	2(1/Loop)	4(2/Loop)	2/Loop	A,F	2/Loop	0–500°F	±2	B,D	B
Nuclear Services River Water Pump Discharge Pressure	1/Pump	4(1/Pump)	1/Pump	A,F	3(1)/Pump	0–100 psig	±1	B,D,E	B
N.S. River Water Pump-Motor Amps.	1/Pump	4(1/Pump)	3/Channel	A,F	12(Total)	0–100 amps	±1	B,D,E	B
N.S. River Water Hdr. Temperature	1/Hdr	2(1/Hdr)	1/Hdr	A,F	3(1)/Hdr	20°–220°F	±1	B	B
N.S. Cooler Outlet Temperature	1/Cooler	2(1/Cooler)	1/Cooler	A,E,F	3(1)/Cooler	20°–220°F	±1	B,D,E	

TABLE II-47. Information readouts available to the operator for monitoring conditions in the unit—Continued

Measured Parameters	Total No. Reqd. Ch.	No. of Ch. Available[1]	No. of Sensors in a Channel	Types of Readouts	No. of Readouts[2]	Indicator Range	Indicator Accuracy[3]	Indicator Location	Purpose or Usage
Decay Heat Closed System Service Cooler River Water Outlet Temp.	1/Cooler	2(1/Cooler)	1/Cooler	A,E,F	3(1)/Cooler	20°-220°F	±1	B,D,E	B
Nuclear Services River Water Pump Disch. Hdr. Pressure	1/Hdr	2(1/Hdr)	2/Hdr	A,F	3(1)/Hdr	0-100 psig	±1	B,D,E	B
Decay Heat Service Cooler Cooling Water Inlet Temperature	2(1/Cooler)	1/Cooler	1/Cooler	A,F	3(1)/Cooler	20°-220°F	±1	B,D,E	
Decay Heat Service Cooler Cooling Water Outlet Temperature	1/Cooler	1/Cooler	1/Cooler	A	1/Cooler	20°-220°F	±1	B	B
Decay Heat Closed Cooling System Disch. Pressure	2(1/Pump)	4(2/Pump)	1/Pump	A	2(1)/Pump	0-100 psig	±1	B,E	B
Decay Heat Closed Cooling Surge Tank Level	1/Tank	2(1/Tank)	1/Tank	A	2(1)	0-5 ft 6 in	±1	B,E	B
Nuclear Services Closed Cooling Pump Suction Hdr. Pressure	1	1	1	A,F	2	0-30 psig	±1	B,D	B
Nuclear Services Closed Cooling Pump Disch. Hdr. Pressure	1	1	1	A,F	2	0-100 psig	±1	B,D	B
Nuclear Services Closed Cooling Service Coolers Inlet Temperature	1	1	1	A,F	2	20°-150°F	±1	B,D	B

Nuclear Services Closed Cooling Service Coolers Outlet Temperature	1/Cooler	2(1/Cooler)	1/Cooler	F	2(1)/Cooler	20° - 150°F	±1	D,E	B
Nuclear Services Closed Surge Tank Level	1	1	1	A	2(1)	0-8 ft 0 in	±1	B,E	B
Core Flooding Tank Level	2/Tank	4(2/Tank)	1/Channel	A,F	2/Channel	0-14 ft 0 in	±2	B,D	
Core Flooding Tank Pressure	4(2/Tank)	4(2/Tank)	1/Channel	A	1/Channel	0-800 psig	±2	B	A
Makeup Pump Suction Hdr. Pressure	1	1	1	A	1	0-100 psig	±1	B	B
High Pressure Injection Flow	1/Loop	4(1/Loop)	1/Loop	A	1/Loop	0-600 gpm	±2	B	B
Decay Heat Removal Reactor Outlet Temp.	1/Loop	2(1/Loop)	1/Loop	A	1/Loop	0-350°F	±2	B	B
Decay Heat Removal Pump Discharge Pressure	1/Pump	1/Pump	1/Pump	A	1/Pump	0-600 psig	±2	E	B
Decay Heat Removal Flow	1/Loop	1/Loop	1/Loop	A	1/Loop	0-5000 gpm	±3	B	B
Borated Water Storage Tank Temperature	1	1	1	A	1	0-200°F	±2	B	
Borated Water Storage Tank Level	2	2	1	A,F	3	0-56 ft 0 in	±2	B,D	
Sodium Hydroxide Storage Tank Level	1	1	1	A	2(1)	0-50 ft 0 in	±2	B,E	
Sodium Hydroxide Storage Tank Temp.	1	1	1	A	1	0-200°F	±2	B	
Decay Heat Removal System Cooler Outlet Temperature	1/Cooler	2(1/Cooler)	1/Cooler	A	1/Cooler	0-300°F	±2	B	B
Spent Fuel Cooling Pump Discharge Pressure	1/Pump	2(1/Pump)	1/Pump	A	2(1)/Pump	0-160 psig	±2	B,E	B

TABLE II-47. Information readouts available to the operator for monitoring conditions in the unit—Continued

Measured Parameters	Total No. Reqd. Ch.	No. of Ch. Available[1]	No. of Sensors in a Channel	Types of Readouts	No. of Readouts[2]	Indicator Range	Indicator Accuracy[3]	Indicator Location	Purpose or Usage
Spent Fuel Water Cooler Outlet Temperature	1/Cooler	1/Cooler	1/Cooler	A,F	3(1)/Cooler	0–250°F	±2	B,D,E	B
Spent Fuel Pool Temp.	2	1	1/Channel	A,F	2(1)	0–200°F	±2	B,D,E	
Spent Fuel Surge Tank Level	1	1	1	A	1	0–40 in	±2	B	B
Borated Water Pump Discharge Pressure	1	1	1	A	2(1)	0–160 psig	±2	B,E	B
Spent Fuel Cooling Flow to Demineralizer	1	1	1	A	2(1)	0–250 gpm	±2	B,E	B
Area Gamma Monitors	20 Monitors	20(1/Monitor)	1/Monitor	B,E	–[7]	0.1 to 10^4 mr/h	±2 of set point	A,B	B
Reactor Building Dome Monitor	1	1	1	B,E	3	10^3 to 10^8 mr/h	±2 of set point	A,B	B
Atmosphere Monitors (Particulate, Iodine and Gas)	12[6]	12(1/Monitor)	1/Monitor	B,E	–[7]	10^1 to 10^6 counts per minute	±2 of set point	A,B	B
Gas Monitor	4	4(1/Monitor)	1/Monitor	B,E	–[7]	10^1 to 10^6 counts per minute	±2 of set point	A,B	B
Liquid Monitor	10	1/Monitor	1/Monitor	B,E	–[7]	10^1 to 10^6 counts per minute	±2 of set point	A,B	B
Failed Fuel Detector (Gamma and Liquid)	1	1	1	B,E	3	10^1 to 10^6 counts per minute	±2 of set point	A,B	B

Legends: *Type of Readout*
- A—Linear Scale Indicator
- B—Log Scale Indicator
- C—Indicator Light
- D—Digital Indicator
- E—Recorder
- F—Plant Computer Output

Indicator Locations
- A—System Cabinets
- B—Control Room
- C—Local Auxiliary Panels
- D—Plant Computer Printout
- E—Local

Purpose or Usage
- Blank—Information only
- A—Total number of channels required for unit startup according to Tech. Specs.
- B—Total number of channels considered to be essential for safe, normal operation.

[1] Number of transmitters that are fed by the sensors providing the signal to the instrument siting.
[2] Number in parenthesis indicates number of local indicators with no electrical channel.
[3] Accuracy at a percent of full measure.
[4] Assumes one channel in bypass.
[5] Two or more signals combined to produce indicated parameter.
[6] Includes two portable monitors.
[7] Multiple readouts (more than 3).

TABLE II-48. Accuracy required by FSAR

Parameter	Range	Accuracy, % of Range	Accuracy, in Units
RC Outlet Temp. NR*	520-620°F	±2	±2°
RC Inlet Temp. NR*	520-620°F	±2	±2°
RC Inlet Temp. WR*	50-650°F	±2	±12°
Loop Δ T	0-70°F	±2	±1.4°
Loop Press. WR	0-2500 psig	±2	±50 psi
Loop Press. NR	1700-2500 psig	±2	±16 psi
Pressurizer Level	0-400 in	±2	±8 in
Loop Flow	0-90x10^6 lb/h	±3	±2.7x10^6 lb/h
Startup Range	0-250 in	±2	±5 in
Operate Range	0-100%	±2	±2%
RB Press.	0-100 psig	±1	±1 psi
RB Temp.	0-300°F	±2	±6°
St. Gen. Press.	0-1200 psig	±2	±24 psi
Steam Temp.	100-650°F	±2	±11°
HPI Flow	0-600 gpm	±2	±12 gpm
BWST Level	0-56 ft	±2	±1.12 ft

*RPS temperature loops must be accurate to ±1%.

readable to an accuracy at least up to the specified instrument accuracy.

Chart timing error should be easy to assess. It ought to be possible to read to 0.1-inch accuracy; at the most common chart speeds (2 inches per hour and 1 inch per hour) the reading error would not exceed 3 to 6 minutes. However, the following improper practices were found at TMI-2:

- Time of day was not accurately or clearly marked.
- Charts were translated without new markings.
- Chart speed did not match the speed written on the chart.
- There were insufficient fiducial time markings.
- Chart speed obviously changed during recording.

Because of these improper practices, the only way that timing can be read with any confidence on these charts is to locate two known events and measure the distance between them. Even this gives no assurance that the chart has not been tampered with between events. Time can be established on a few charts with an accuracy of 3 minutes. However, as a general rule, 12 minutes, or even greater variations, must be considered representative.

However, the reactimeter data are much more reliable. There is no possibility of an amplitude error other than the instrument channel error, and time can be matched to within a few seconds. Therefore, in attempting to match reactimeter data to strip-chart data, disagreements have been resolved in favor of the reactimeter.

Reliability of the TMI Data

Data channels that had given trouble in the past undoubtedly would be viewed as less reliable than those that had operated without difficulty. From a sample of 45 incidents reported in the TMI-2 "In-

TABLE II-49. Errors at most recent calibration of selected instruments

Inst. No.	System	Type	Cal. Date	Error Full Range
RC3A-PRI	RCS Pressure, N.R.	Recorder	6/30/78	0
RC5A-TI2	RCS Inlet Temp.	Indicator	3/30/78	0
RC4A-TI2	RCS Outlet Temp.	Indicator	4/13/78	0.5%
RC4B-TI2	RCS Outlet Temp.	Indicator	4/13/78	0
RC5A-TTI	RCS Inlet Temp.	Bridge	5/22/78	0
RC5A-TT2	RCS Inlet Temp.	Bridge	3/12/78	0.03%
RC2-TEI	Pressurizer Temp.	Sensor	3/3/77	0.04%
RC2-TTI	Pressurizer Temp.	Bridge	3/6/78	2.0%
RC1-LTI	Pressurizer Level	Transmitter	3/25/78	0.05%
RC1-LI1	Pressurizer Level	Indicator	8/9/76	0
RC1-LT2	Pressurizer Level	Transmitter	12/29/78	0.14%
RC1-LT3	Pressurizer Level	Transmitter	9/23/78	0
RC1-LR	Pressurizer Level	Recorder	12/22/77	1.5%
RC2-TT2	Pressurizer Temp.	Bridge	5/4/77	0.71%
RC9-TE	Pressurizer Inlet Temp.	Sensor	3/3/77	0.09%
RC10-TE1	Pressurizer Relief Outl. Temp.	Sensor	3/3/77	0.01%
AH-YMTR-5017	RB Temperatures	Recorder	3/23/79	0.15%
AH-TE-5012	RB Temperature, RCDT Area	Sensor	11/19/77	0.1%
AH-TE-5022	RB Temperature, 330 ft Elev.	Sensor	12/19/77	0.17%
BS-PR-1412	RB Pressure	Recorder	11/22/76	0
BS-PR-4388	RB Pressure	Recorder	11/22/76	0
DH3-LT2	BWST Level	Transmitter	2/17/79	0.22%
DH3-LI1	BWST Level	Indicator	8/4/77	1.79%
DH3-LT2	BWST Level	Transmitter	12/10/77	0.26%
DH3-LI2	BWST Level	Indicator	1/30/76	0.18%
SP6A-PI1	Steam Gen. Press., Loop A	Indicator	9/9/78	0
SP6B-PI1	Steam Gen. Press., Loop B	Indicator	9/9/78	0
SP6A-PI2	Steam Gen. Press., Loop A	Indicator	4/13/78	0
SP6B-PI2	Steam Gen. Press., Loop B	Indicator	4/13/78	0.4%
MS-TE-1097	OTSG A Outlet Temp.	Sensor	5/10/78	0.5%
MS-TT-1097	OTSG A Outlet Temp.	Transmitter	9/5/78	0.05%
SP1A-LT2	OTSG A Oper. Level	Transmitter	11/7/78	0.30%
SP1A-LAMI	OTSG A Oper. Level	Compensator	4/11/78	0.08%
SP1A-LR	OTSG A Oper. Level	Recorder	1/14/79	0
SP1A-LT4	OTSG A Startup Level	Transmitter	11/9/78	0.12%
SP1A-LT5	OTSG A Startup Level	Transmitter	11/9/78	0.10%
SP1A-LT1	OTSG A Full range	Transmitter	11/8/78	0.18%
SP1A-LI1	OTSG A Full range	Indicator	6/29/76	0.67%

strument Out of Service Log," 42% were alarms, 33% were radiation monitors, 13% were temperature channels, 4% were pressure channels, and the remainder were equally divided among level, flow, and electrical channels. It is probable that alarms and radiation monitors would be perceived as less reliable than other data. There were somewhat more problems with temperature channels than with some others, although this is unlikely to be significant, given the small size of the data sample.

Past operation of the data channels cannot give much information on the actual, as opposed to perceived, reliability. Conditions during the accident (e.g., temperature, humidity, and radiation) were much more challenging than at any time in the history of the plant. For example, the peak temperature

TABLE II-50. Estimated recorder reading accuracy

Parameter	Range	Est'd Rdg Accuracy	Req'd Instrument Accuracy
RCS Temp.	0–800°F	5°	16°
Steam Gen. Temp	0–800°F	5°	16°
RCS Unit Tave.	520–620°F	1°	2°
RCS Unit Outlet Temp.	520–620°F	1°	2°
RCS Press. (WR)	0–2500 psig	25 psi	50 psi
RCS Press. (NR)	1700–2500 psig	5 psi	16 psi
React. Bldg Press. (NR)	-5– +10 psig	0.2 psi[1]	0.15 psi
React. Bldg. Press. (WR)	0–100 psig	1 psi[1]	1 psi
React. Bldg. Temp.	0–200°F	1°	6°
Steam Press.	600–1200 psig	5 psi	24 psi
Pressurizer Level	0–400 in	2.5 in	8 in
Steam Gen. Level	0–100%	1%	2%
Makeup Tank Level	0–100%	1%	2%
SRM and IRM	8 decades	0.1 decade[2]	[3]
Rad. Monitors	5 decades	0.1 decade[2]	[4]
RCS Flow	0–110x10^{6}lb/h	1x10^{6}lb/h	5.4x10^{6}lb/h

[1]Chart alternates between wide and narrow range. Reading of each trace is different when not in its own range.
[2]Log scale—accuracy varies. This is an estimated average.
[3]3% of full range.
[4]2% of setpoint; varies with instrument.

measured on the incore thermocouples (2580°F) was near to the liquidus temperature of the Inconel sheaths (2600°F). As a result, melting of junctions and rewelding of false junctions is a distinct possibility. Further voiding of the pressurizer reference leg because of evolution of dissolved hydrogen may have occurred.

Degradation of insulation due to high temperature, humidity, and radiation in the reactor building could have caused false readings. Whether environmental extremes caused misperformance of instrumentation can only be a matter of conjecture. Even if a channel is found to be inoperative in a postmortem examination, it is not usually possible to determine when the failure occurred.

Perceived reliability is, of course, lower for out-of-range channels. Furthermore, the plant computer uses the same symbol for data out of range as for bad data, and it has generally not been possible to determine whether out-of-range data are correctly indicated without access to additional information.

High reliability can be ascribed to data confirmed from an independent source. Redundant reactimeter and strip-chart data generally tend to confirm each other, although the low accuracy and poor legibility of some of the strip charts make comparison difficult. Reactor coolant system (RCS) pressure and temperature data appear to be particularly well confirmed, while PORV block valve opening and closing times cannot be unequivocally confirmed. Estimates of data reliability are given in Table II-51.

The most vital information pertains to core water inventory. Because they lacked this information, the operators depended on an inappropriate substitute: pressurizer level, and this dependence on pressurizer level readings actually caused incorrect actions

TABLE II-51. Estimated data reliability

Data	Primary Data Source	Confirmatory Sources	Reliability Ranking
RCS Pressure	Reactimeter	Strip charts, utility printer	Good
RCS Temp.	Reactimeter	Strip charts, utility printer	Good
Press. Level	Reactimeter	Strip charts, utility printer	Good
Press. Temp.	Utility Printer	—	Good
OTSG Level	Reactimeter	Strip charts	Good
OTSG Press.	Reactimeter	Strip charts	Fair
EFW Flow	OTSG Level Change	None	Very Poor
MU Flow	Operator Recollection	BWST Level	Poor
PORV Block Valve Opening	Operator Recollection	Tailpipe temp., RB Press. and Temp.	Poor
BWST Level	Logs	None	Fair
Core Temps.	Incore T/C's (alarm printer)	One set of manually read voltages	Poor
Pump Start and Stop	Alarm printer	Operator recollection	Good

to be taken. Similarly, the lack of emergency feedwater flow indication caused the operators to seize on a set of substitutes—discharge pressure, "eleven-valve" opening, and steam generator level. This set of substitutes did, however, eventually lead to the correct conclusion, but only after a considerable delay.

Nearly as important as the lack of some needed data was a confusing excess of unnecessary information. As an example, one of the factors leading to the alarm printer falling behind was the great number of alarms caused by feedwater heaters. These alarms were not germane to the situation, and suppression of them would have helped clear the computer for more useful tasks.

Utility of Data for Historical Reconstruction

For a reconstruction of the accident sequence, additional data would have been useful. This is especially true when trying to understand the motivation for actions taken, where a voice recording of operator discussions would probably have been helpful.

The improper practices concerning strip-chart marking have hindered reconstruction of the accident sequence. Training in the importance of correct marking and stricter administrative control should ensure better marking practices. Also, some additional consideration should be given to the importance of historical reconstruction when selecting the channels to be recorded.

Accident reconstruction would also have been aided by more complete data recording on tape. The reactimeter data were quite helpful, but would have been even more useful if the entire range of each channel had been recorded and if the data channels had been specifically selected for accident analysis. Postmortem analysis would be easier and better if a similar recording device was dedicated to analysis of accidents and other abnormal occurrences.

Needs for Improved Instrumentation

There is a need for improved instrumentation of several kinds, which we discuss here.

Need for Disturbance Analysis Systems

On October 5, 1966, the Enrico Fermi Atomic Power Plant, a 200-Mwt sodium cooled liquid metal fast breeder reactor underwent a fuel melting incident. Prior to this incident the Fermi Plant staff had noted anomalous thermocouple readings at the outlets of several fuel subassemblies. While investigating the anomalous thermocouple readings, it was observed that several subassemblies had abnormally high outlet temperatures, and there was leakage of fission products into the reactor buildings. A subsequent investigation found that melting of a portion of two fuel assemblies had taken place. This was caused by a loose zirconium deflector plate that blocked the inlet nozzles of several fuel subassemblies.

As a result of this incident a study was initiated[154] to determine a prompt, reliable, and economic means to detect malfunctions and enable corrective action to be taken to prevent damage to the plant and the environment. This study determined that sufficient information was available from existing instrumentation that, if accurately and rapidly analyzed, could have detected the occurrence of abnormal conditions in sufficient time to reduce and probably eliminate fuel melting. Consequently, an online computer, called a malfunction detection analyzer (MDA), was designed and added to the reactor to detect anomalous conditions involving reactivity, core outlet temperature, and fission product releases.

The MDA utilized an IBM 1800 computer and was put on line within a few years of its conception. It compared measured values with predicted values of subassembly temperature rises on the basis of subassembly power generation, total core power, and primary flow rate. If the difference between predicted and measured temperature rises exceeded prescribed values, the MDA would initiate a warning. Similarly, anomalous reactivity or fission product releases would initiate warnings. The MDA was installed as an operator aid and was not connected to the reactor protection system.

At first, operators considered the MDA and the computer a "black box," and were apprehensive about it. Subsequently, with increased understanding and reliable use, the MDA became a valuable reactor monitoring and data acquisition device that was considered an indispensable aid by the operations personnel.[155]

Findings— Today about a dozen years after the conception of the Enrico Fermi Plant's MDA, no operating reactor in the United States has such an analyzer. This is due in part to the much greater complexity of the neutronics of a large LWR, to regulatory disincentives, and to the reluctance of the utilities to spend money for a system for which they have not felt a real need. The Electric Power Research Institute is spending a great deal of money on a sophisticated neutronic analysis system to improve plant efficiency during normal operations,[156] and General Electric and Combustion Engineering (to name only two) also have developed experimental systems for anticipatory control that have not been sold to any U.S. utility.

In Europe, however, the situation is somewhat different. The Halden Reactor Project (in Norway) has been developing and testing computerized reactor control and disturbance analysis systems on their small reactors for a number of years. The Kraftwork Union (German PWR vendor), working both independently and cooperatively with the Halden Project, has developed and installed computerized xenon transient controls on Biblis A&B reactors (1300 MWe) and disturbance analysis capability in the Grafenrheinfelt reactor.

Recommendations — TMI has shown us that the plant operators need more help in analyzing anomalies, and utilities should be required to install MDAs in each plant to assist the operators in controlling the plant.

Instrument Failures

Very few instrument failures occurred during the accident. This is significant when one considers the duration of the accident, the flooding, radiation in the reactor building, and the degree of core damage.

Conditions of high humidity and radiation have continued at TMI-2 since the accident. There has also been considerable flooding by water that is still far from pure. The possibility of cables being under water in an electrical conducting solution for a matter of months was not considered in the design. The total integrated radiation qualification of many systems may have been exceeded. However, no failures have been ascribed to this condition alone. It is clear, however, that the requirement that systems be operable for 24 hours in an accident environment is far too lenient.

Pressurizer level indicators did fail. These are considered "accident monitoring instrumentation" and, as such, are designed for the postaccident environment. The first such failure occurred at 9:14 p.m., March 29, 1979.[157] This was more than 24 hours after the accident began and, hence, does not technically demonstrate a lack of compliance with the environmental qualifications.

Some incore thermocouples appear to have been damaged in the accident. These were considered not related to safety and would not necessarily be expected to survive environments more severe than normal operation. Moreover, no matter what category these instruments had been placed in, the ferociously severe core environment probably would have damaged them. Temperatures of 3000°F and higher are most challenging for any instruments, given our present technology. The same is true for self-powered neutron detectors. However, consideration should be given to installing thermocouples as a matter of course and to protecting leads from high temperatures to the maximum extent possible.

Lack of Sufficient Range Indication on Temperature

Display instruments for incore thermocouples have an indication range to a maximum of 700°F. Thermocouple temperatures during the accident exceeded 2000°F but were not indicated by the instrumentation available to the operator. However, externally placed instruments (digital voltmeters) with sufficient range recorded these higher temperatures during the accident. Because such temperatures were not anticipated and provisions were not made for use of display, the operators did not place the proper significance on the higher temperature readings recorded. Operators have indicated in interviews that they were reluctant to attribute significance to the readings because the thermocouples were not assessed as important to safety and were not designed to safety standards.

Reactor coolant temperatures also exceeded the indicated range of their display instruments during the accident. The indicated narrow range for hot-leg and cold-leg temperatures is 0–620°F and 0–520°F, respectively. Strip-chart recordings have a range of up to 800°F, which was also exceeded.

Computations for average temperature readings are based on the indicated narrow range of the hot-leg and cold-leg temperatures. Therefore, the average temperatures computed during the accident remained at about 570°F (hot leg) and 10°F (cold leg) lower than normal operating temperatures. These readings of the average temperatures appear to have misled some operators who did not recognize that the average temperature readings of the instruments were in error.[158,159]

Lack of Recording of Reactor Coolant Makeup Flows

Throughout the accident, flow indication of the makeup and high pressure injection was very important to the operators, particularly when these systems were placed on manual control. Makeup flow and high pressure injection were continuously throttled by the operators to control pressurizer level, and the instantaneous flow indication was used for that control. This indication, however, was not recorded for later reference, and it became important for assessment of water inventory in the reactor.

The lack of flow recording for reactor coolant makeup, letdown, and high pressure injection has hampered evaluation of reactor inventory assessment during the accident.

Findings— Very few instrumentation failures occurred, and almost all systems performed far in excess of their requirements. The failures that did occur can be ascribed to too lenient environmental qualification, to exceptionally severe environmental conditions, and to qualifications for too short a time.

The RPS and SFAS systems and, to some extent, accident monitoring systems are environmentally qualified for postaccident environments. Systems required for safe shutdown are not so qualified. No category is established for instrumentation required to maintain stable conditions after shutdown, as existing qualifications call for only 24-hour operation in the accident environment.

Accuracy of instrumentation from preaccident calibration appears to be adequate. However, poor control room practices resulted in difficulties in chart reading.

The reliability of alarms and radiation monitors were perceived to be lower than other data channels. Considerable confidence can be placed on most RCS parameters, within their specified accuracy, and subject to the difficulties in chart timing error. However, PORV block valve opening and closing times cannot be reliably determined.

The utility of the data for operation was compromised both by the lack of some vital data and by a confusing superfluity of low priority data. Little thought appears to have been given to the utility of data for historical reconstruction of the accident. Inappropriate substitutes were used for unavailable data.

Recommendations— Many of the recommendations made in the following sections are covered in Revision No. 2 to Regulatory Guide 1.97[157] and the American Nuclear Society Standard 4.5 (draft).[160] Therefore, if these guides are adopted, the recommendations marked with an asterisk(*) will be superfluous.

Specification of Environmental Qualifications— Design of cables and some sensors for operation after flooding should be considered, and their required time for postaccident operation should be lengthened.*

Categories of Systems—Accident monitoring and safe shutdown systems should be qualified to full accident conditions.* In addition, a category should be established for "systems required to maintain the plant in a stable condition" that are qualified to operate in full accident conditions.* Careful review of instrument and control systems should be carried out to make sure that items such as pressurizer heaters do not get left out or get placed in improper categories.

Accuracy and Reliability of Data—Administrative review of instrument repair records is necessary so that unreliable systems will be upgraded. Stricter control on strip-chart marking should also be instituted.

Utility of Data—Data presented to the operators should be reviewed to make sure that important data are continuously available.* Consideration should be given to layout so that important data can be readily assimilated without distraction by less important displays.*

Recording devices meant to document data for historical reconstruction of accidents or off-normal incidents, such as control room voice recorders, magnetic tape, disk recording of important parameters, and dedicated strip charts, should be installed.

2. CORE DAMAGE AND RECOVERY

a. Data Analysis for the First Sixteen Hours

Introduction

This discussion analyzes the thermal, hydraulic, and neutronic conditions inside the reactor primary system during the period in which damage to the fuel assemblies most probably occurred. The information on the behavior of several of the reactor system parameters has been gathered from several sources. Table II-52 lists the data sources for those parameters found useful in the analysis.

In several cases, the behavior of the reactor system parameters had to be inferred from other data. For example, the opening and closing of the block valve upstream of the pilot-operated relief valve (PORV) had to be inferred from both an analysis of the reactor building pressure strip chart for indications of changes in slope and an analysis of the alarm printer indications of the alarming and clearing of the alarm for the tailpipe temperatures of the PORV. The data given for the system saturation temperature, T_{sat}, are calculated from the system pressure and are given for reference purposes only. Because large temperature differences existed simultaneously among various parts of the system at many times during the accident, the calculated T_{sat} can only be used as a reference point to judge the deviation from condensible vapor behavior at any given time during the accident.

The time conventions used in this discussion are as follows: the time since the start of the accident is given in hours and minutes (e.g., 1 hour 15 minutes), assuming a time zero of 04:00:00 a.m. on March 28, 1979, and clock time is given on the 24-hour clock-time basis and Eastern Standard Time (e.g., 05:33:22 a.m. is 5 hours, 33 minutes, and 22 seconds of a 24-hour day).

The abbreviations used in the status summaries and much of the following text are defined in Table II-53. An isometric drawing of the reactor primary system is presented in Figure II-23, and a schematic showing volumes in the system is presented in Figure II-24. The most important features and elevations are identified. A plan view of the pressurizer is shown in Figure II-25.

The plant parameters that seem to have some correlation to each other and to the total system behavior are plotted in Color Plates III, IV, and V. The time scales of each of the plotted parameters have been matched to the best accuracy possible, but except where otherwise noted, a time coincidence of no better than about 3 minutes should be expected for events or responses that actually were simultaneous.[161]

Other parameters have been plotted and examined for correlation to system behavior, such as pressurizer heater trips and makeup tank levels, but did not correspond to the data presented in Color Plates III and IV. Thus, the data on these parameters are not reported in this section.

General Description of the Accident Sequence

While the available data in the early minutes of the accident are of interest to thermal-hydraulic experts, data of interest to those involved in the estimation and evaluation of the damage to the core does not develop until the last reactor coolant pumps were shut down at 1 hour 40 minutes. There is no evidence to indicate that any damage to the core had occurred earlier.

When the second set of reactor coolant pumps were turned off, the two-phase coolant mixture

TABLE II-52. Sources of data about reactor system parameters

Parameter	Data Sources: Reactimeter Log[1]	Alarm Printer[2]	Utility Typer[3]	Hourly Computer Log[4]	Strip Chart[5]	Inferred from Alarm Printer	Inferred from Strip Chart
Hot- and cold-leg temperatures (OTSG)(T_H, T_C)	X		X	X	X		
Reactor system pressures (RCP)	X		X	X	X		
OTSG pressures and levels	X		X	X			
Pressurizer temperature (T_{pzr})			X				
Pressurizer surge line temperature (T_{surge})			X				
Pressurizer spray valve operation	X						
Pressurizer Pilot Operated Relief Valve (PORV)						X-tail pipe temp alarms	X-building • reactor pressures X-building temperature
Pressurizer Block Valve						X-tail pipe temp alarms	X-building reactor pressure X-building temperature
Makeup pump operation		X					
Reactor Coolant Pump (Operation RC-P1A, 2A, 1B, 2B)	X	X					
Incore Thermocouple temperatures (Incore T/C)		X-plus one set of instrument measurements					
Self-Powered Neutron Detectors (SPND)		X			X		

[1]TMI Reactimeter Patch Log March 28, 1979 (NRC Reel OPS-2-806 283)

[2]TMI Control Room Computer Alarms Data March 28, 1979 (NRC Reel OPS-2-800 2784).

[3]TMI Operator Special Summaries March 28, 1979 (Utility Printer) (NRC Reel OPS-2-802).

[4]TMI Station Log March 28, 1979 (Log/Typer) LSL 0001 (NRC Reel OPS-2-801 2960).

[5]TMI Plant Strip Charts. By name—OTSG and Primary System Temperatures, March 21, 1979 to April 4, 1979. SC-0043 Recorder 10 (NRC Reel OPCP-2-803).

TAB!E II-53. Definitions and abbreviations

RCP—Reactor coolant pressure, reactor primary system pressure.

RC-P—Reactor coolant pumps 1A and 2A, on OTSG A, 1B and 2B on OTSG B.

MU-P—Makeup pumps 1A, 1B, and 1C.

L_{pzr}—Indicated level of water in the pressurizer in inches.

Atmos. Dump Valve—The valve that allows the steam developed in either or both steam generators to be dumped to the atmosphere outside the reactor building.

OTSG A—Once-through steam generator A.

OTSG B—Once-through steam generator B.

T_{surge}—Temperature indicated by a thermocouple strapped on the outer surface of the surge line between the OTSG A hot leg and the pressurizer.

T_{pzr}—Temperature in °F measured in the interior of the pressurizer just above the heaters, by a resistance thermometer called an RTD.

SRM—Counts per second of the Source Range Monitor (SRM), sensing thermal neutrons from the reactor core, primarily from the peripheral bundles. In this accident, SRM is mostly an indicator of water level in the downcomer in the reactor vessel. However, sudden changes in count levels may also be indicative of major changes in geometry of the core.

T_{HA}—Temperature in °F of the hot leg between the reactor vessel and OTSG A, measured by an RTD about 54 inches below the tangent point of the curve at the top of the hot leg.

T_{HB}—Hot-leg temperature for OTSG B.

T_{C1A}, T_{C2A}, T_{CB}—Temperature in °F of cold legs 1A and 2A of OTSG A, and either 1B or 2B (believed to be 1B) for OTSG B, measured a few inches below the inlet to the pertinent reactor coolant pump.

PORV—Pilot-operated relief valve on the pressurizer.

Block Valve—The gate valve positioned in the line between the pressurizer and pilot-operated relief valve (PORV) that was stuck in the open position.

RTD—A platinum resistance thermometer used to measure system temperature.

Engineered Safety (ES) System Actuation—A series of valve and pump actuations automatically performed when certain safety limits in the total reactor system are exceeded. It includes isolation of the reactor containment building, tripping of MU-P1B (unless the trip is bypassed), starting of MU-P1A and 1C, opening of the four "16" valves for maximum makeup flow of about 1000 gpm total from two MU-Ps, start of containment and sprays, start of decay heat pumps.

Steaming to Condenser or Condenser Vacuum—The normal mode of heat removal from the system is by steam production in the OTSG, steam passage through the generating turbines, and condensation in the steam condenser. The flow of steam to the turbines can be bypassed.

PZR Spray Valve—The valve in the pressurizer spray line connecting the outlet side of the RC-P2A to the top of the pressurizer and used for "spraying down" the pressurizer in normal operation to decrease the system pressure.

Pressurizer Vent Valve—A separate venting valve located on the top of the pressurizer which can be used to reduce system pressure.

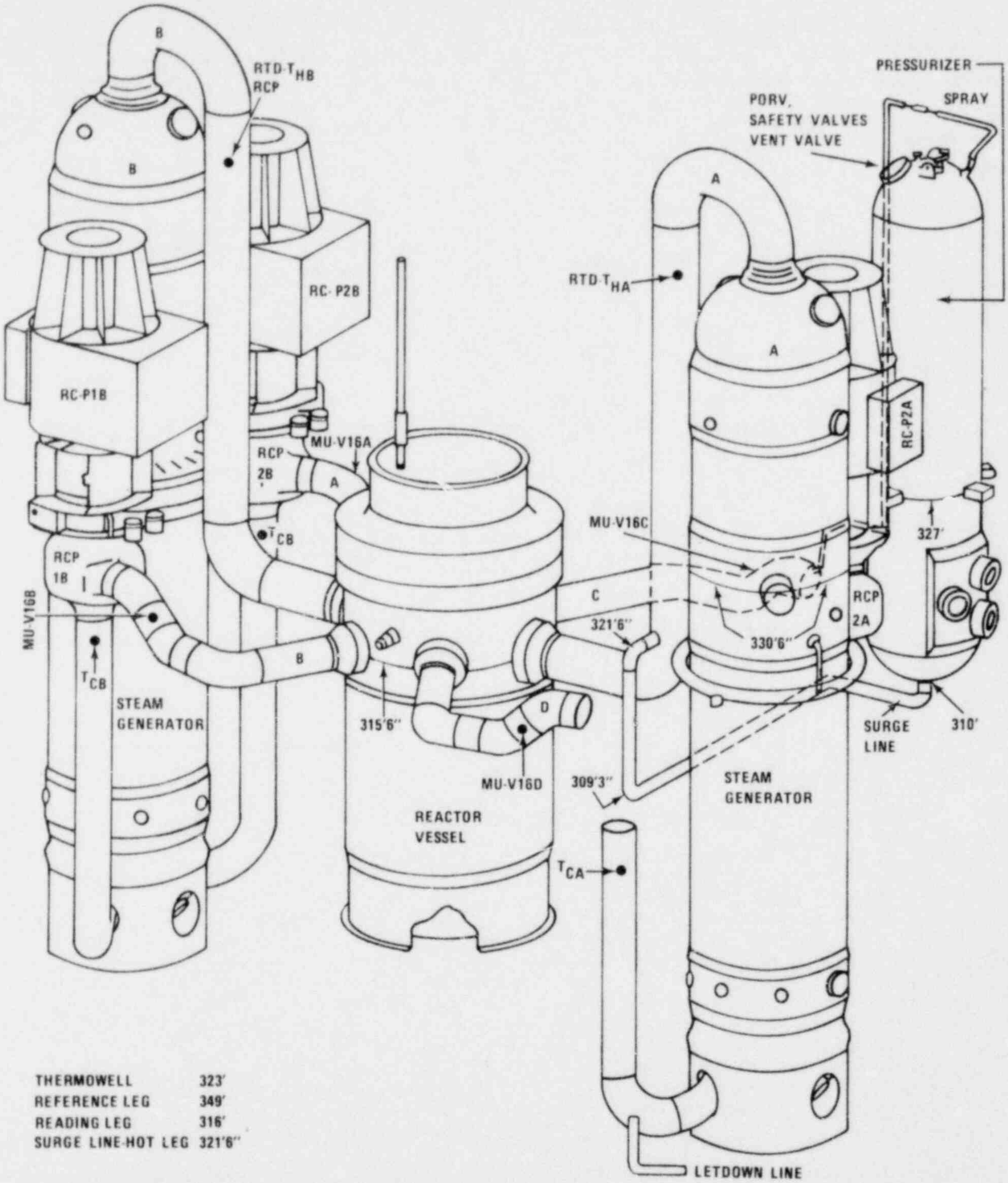


FIGURE II-23. TMI-2 Isometric Schematic Drawing

separated, water going to the lower levels and steam to the upper. We estimate that water filled about one-half of OTSG B, about one-quarter of OTSG A, and to about the top of the core of the reactor vessel. However, the pressurizer still contained a two-phase mixture because the PORV was still open, and the system was still slowly "blowing down."

Almost immediately after the pump shut down, the water level in the core began to decrease, as it boiled off to escape the system or be condensed in an OTSG, and the exposed sections of the fuel rods

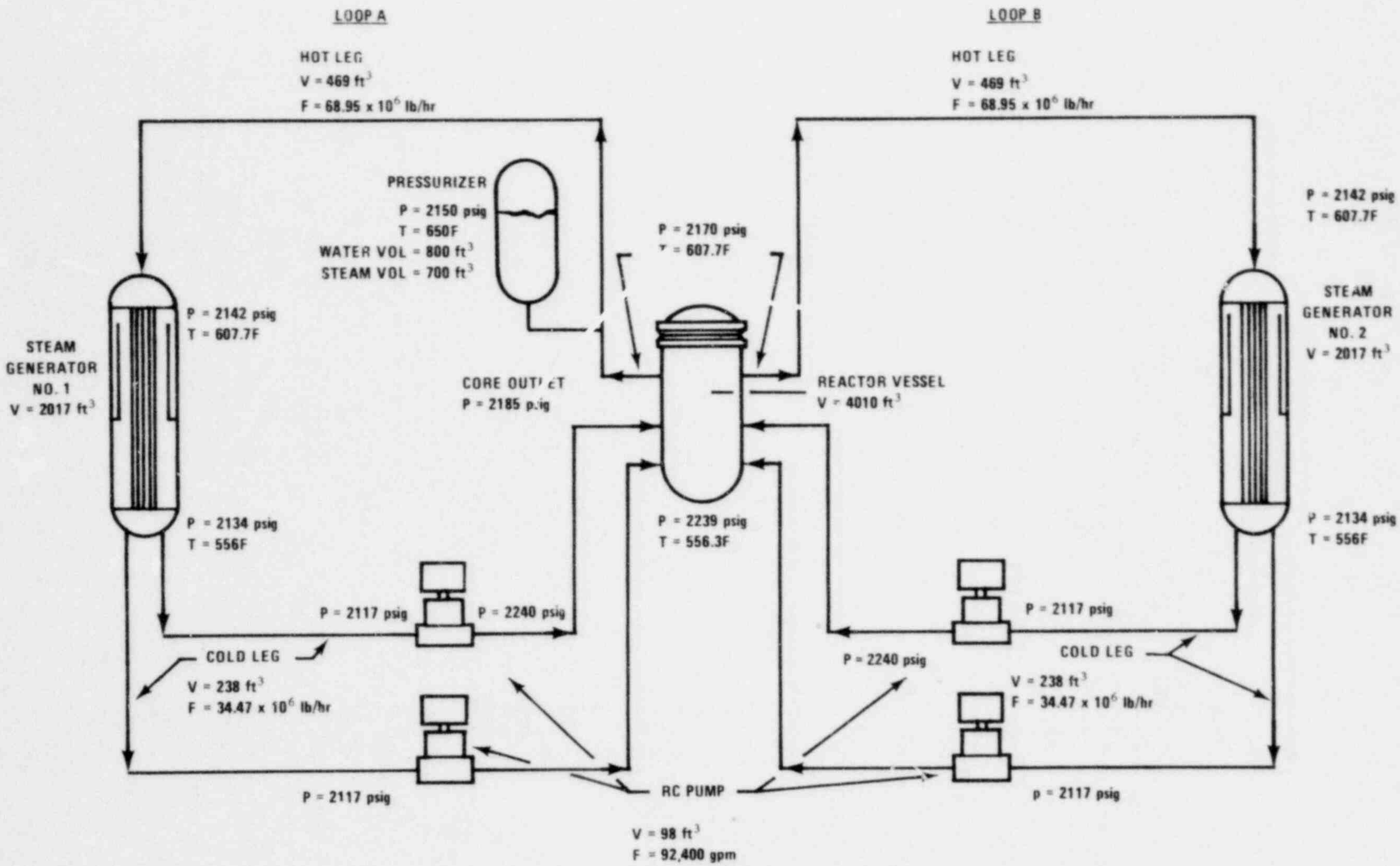


FIGURE II-24. Coolant System Flow Diagram

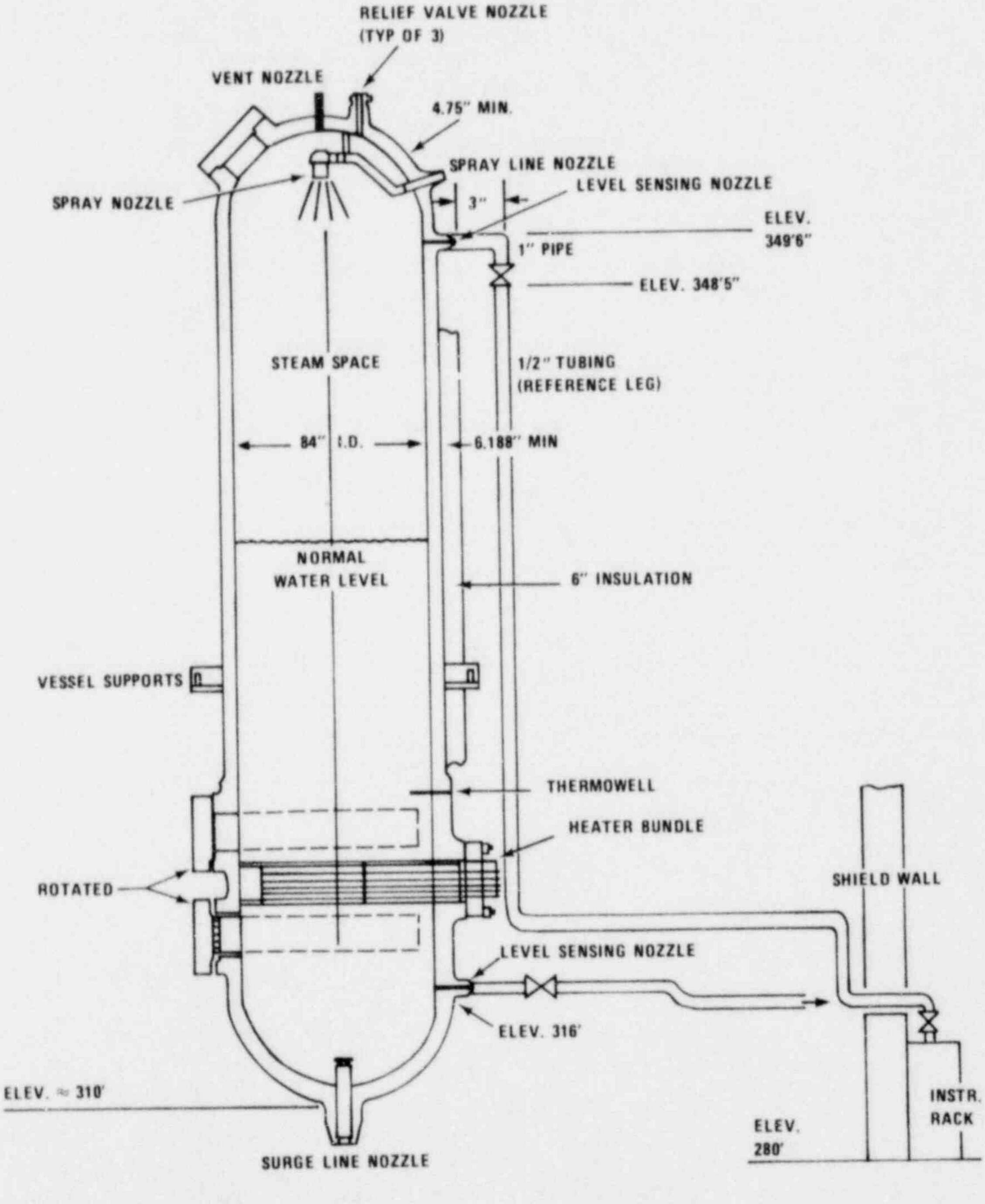
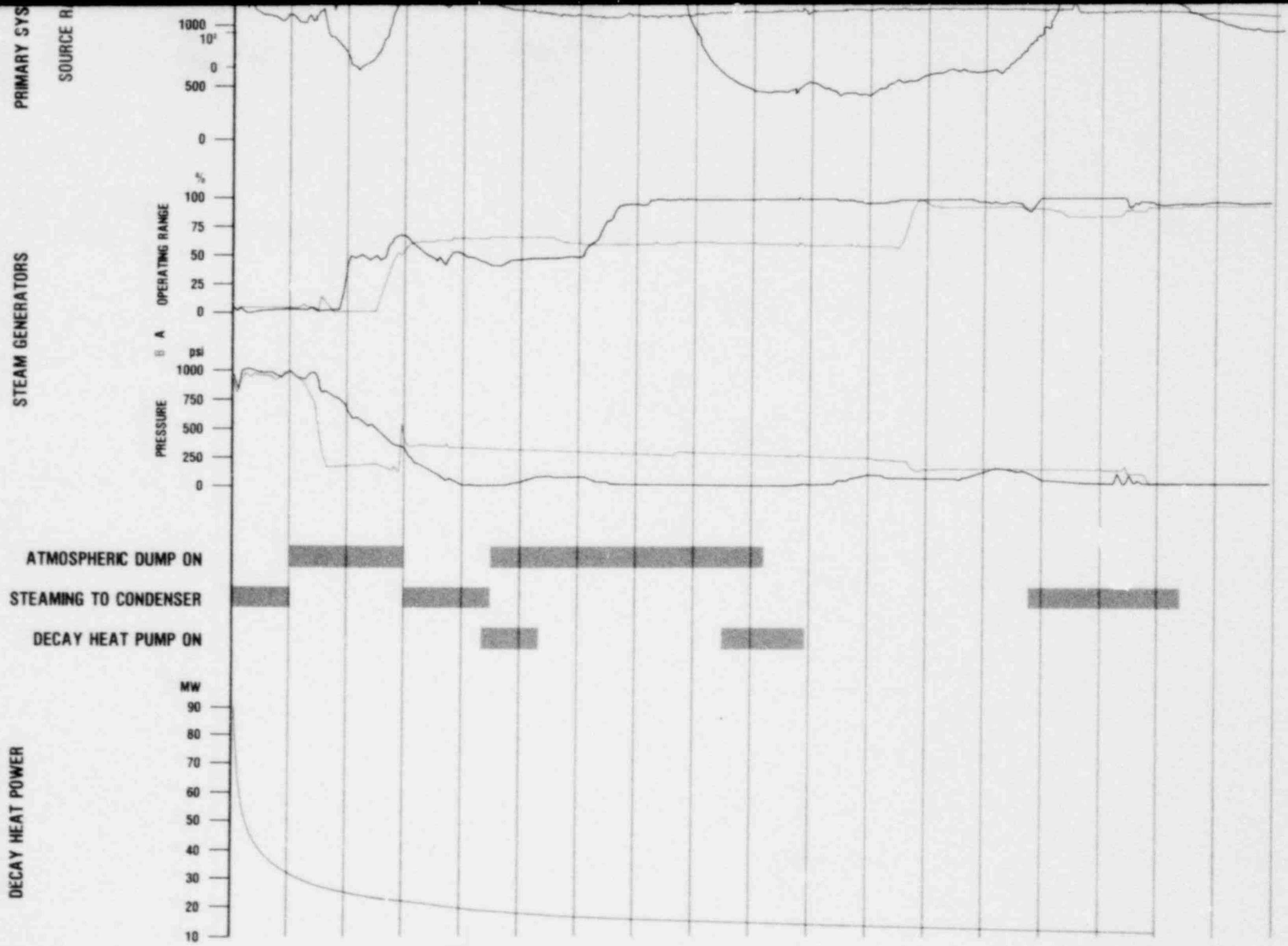
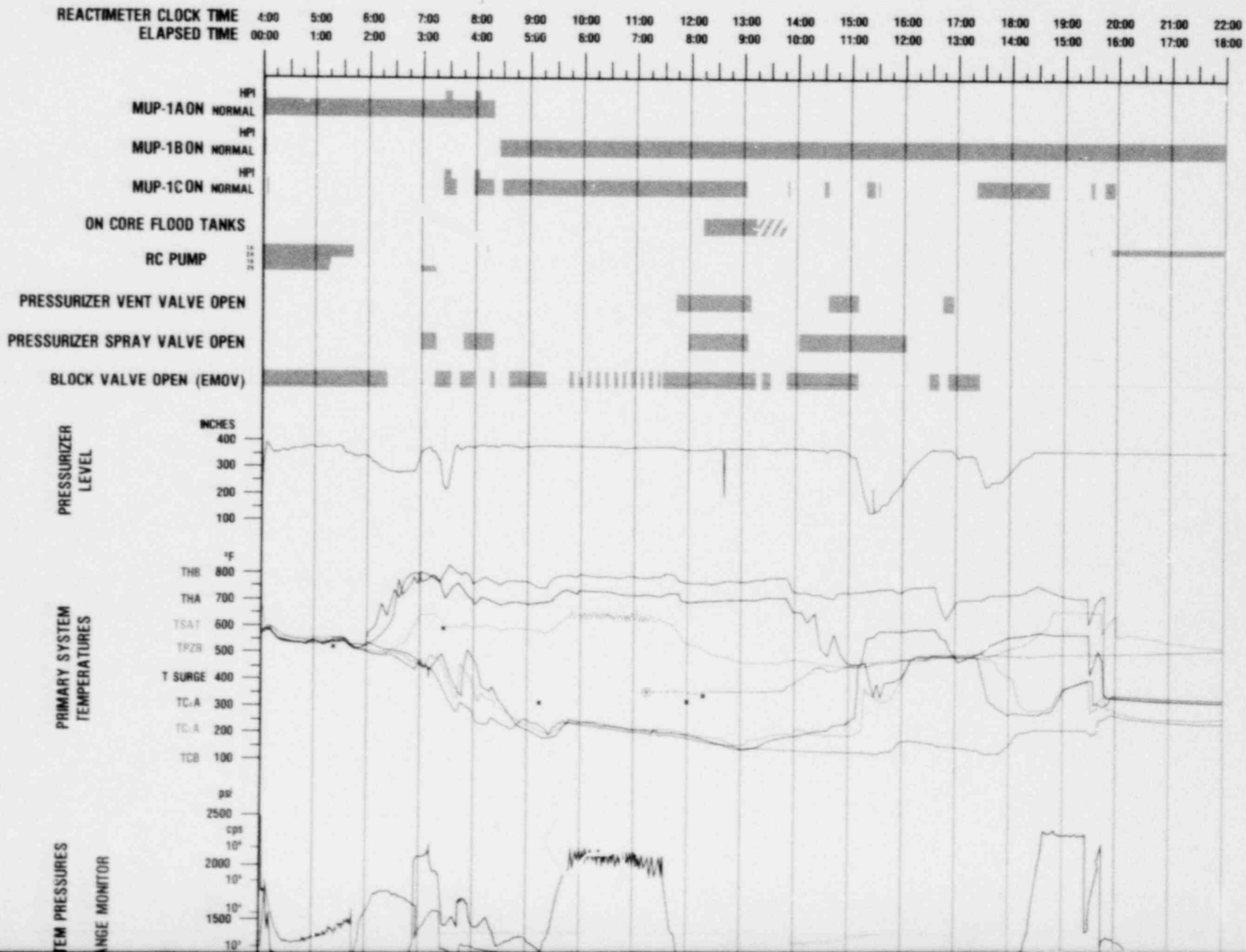


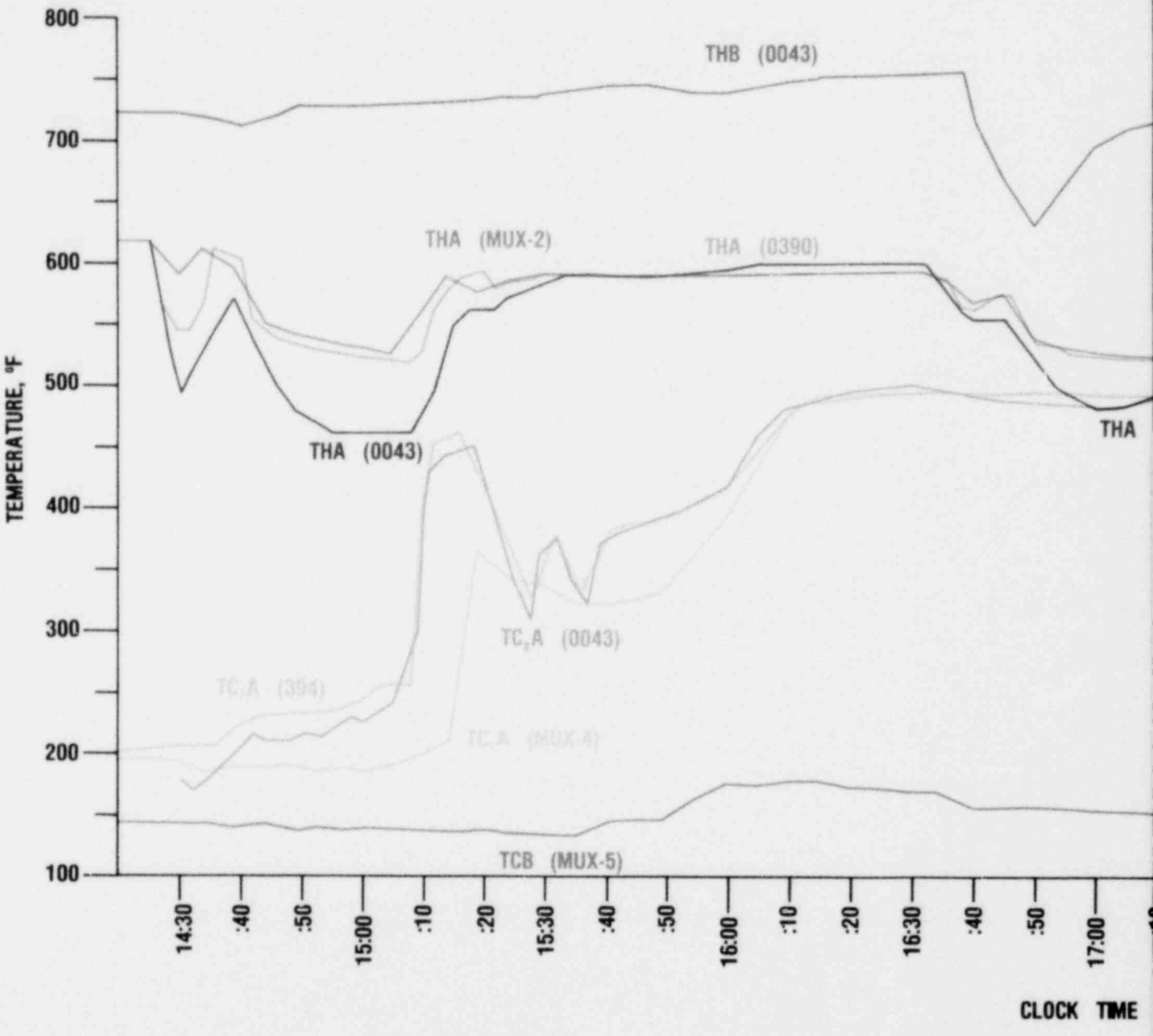
FIGURE II-25. The Pressurizer



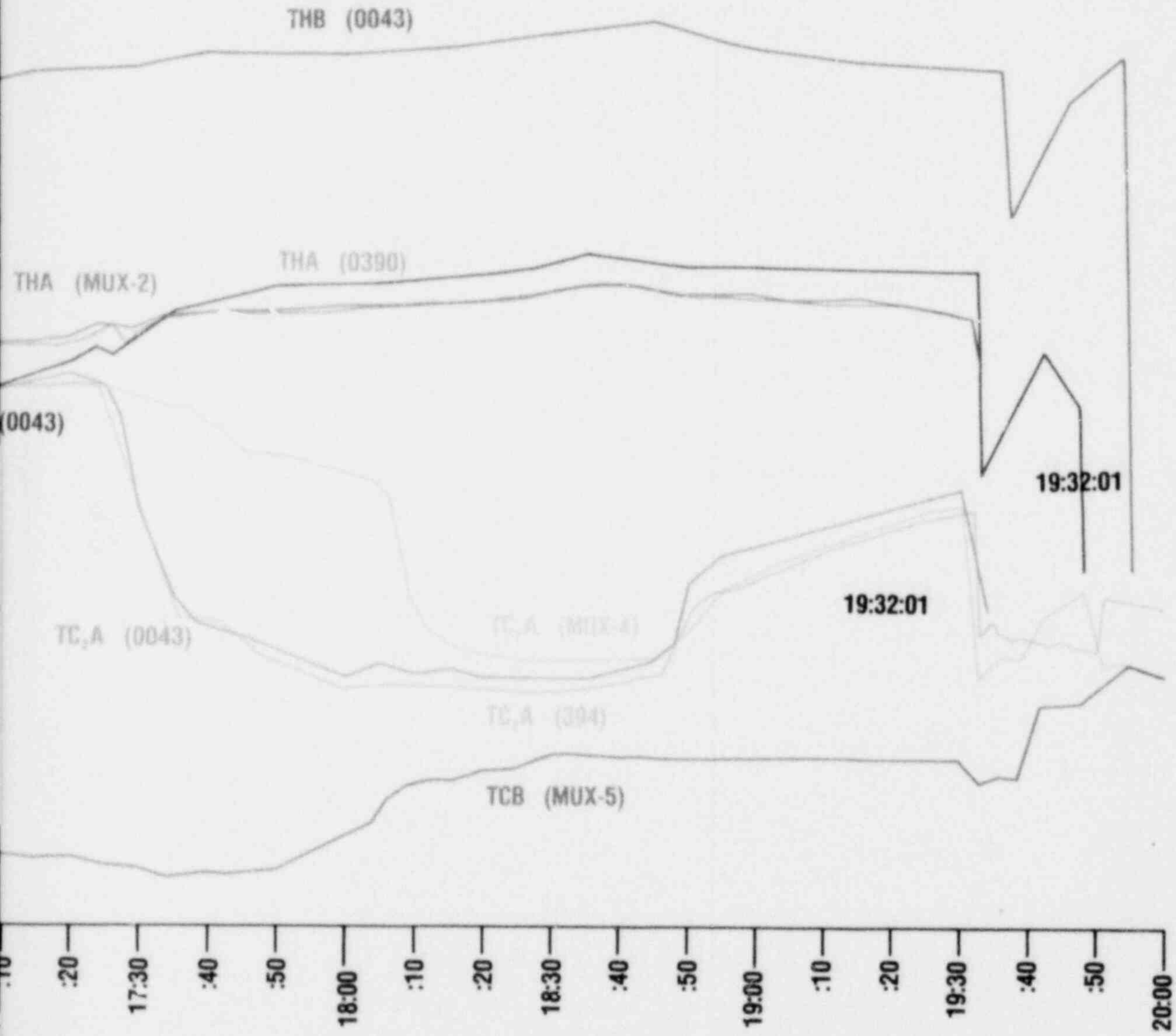
COLOR PLATE III. PLOT OF SYSTEM PARAMETERS FOR THE FIRST 16 HOURS OF THE TMI-2 ACCIDENT.



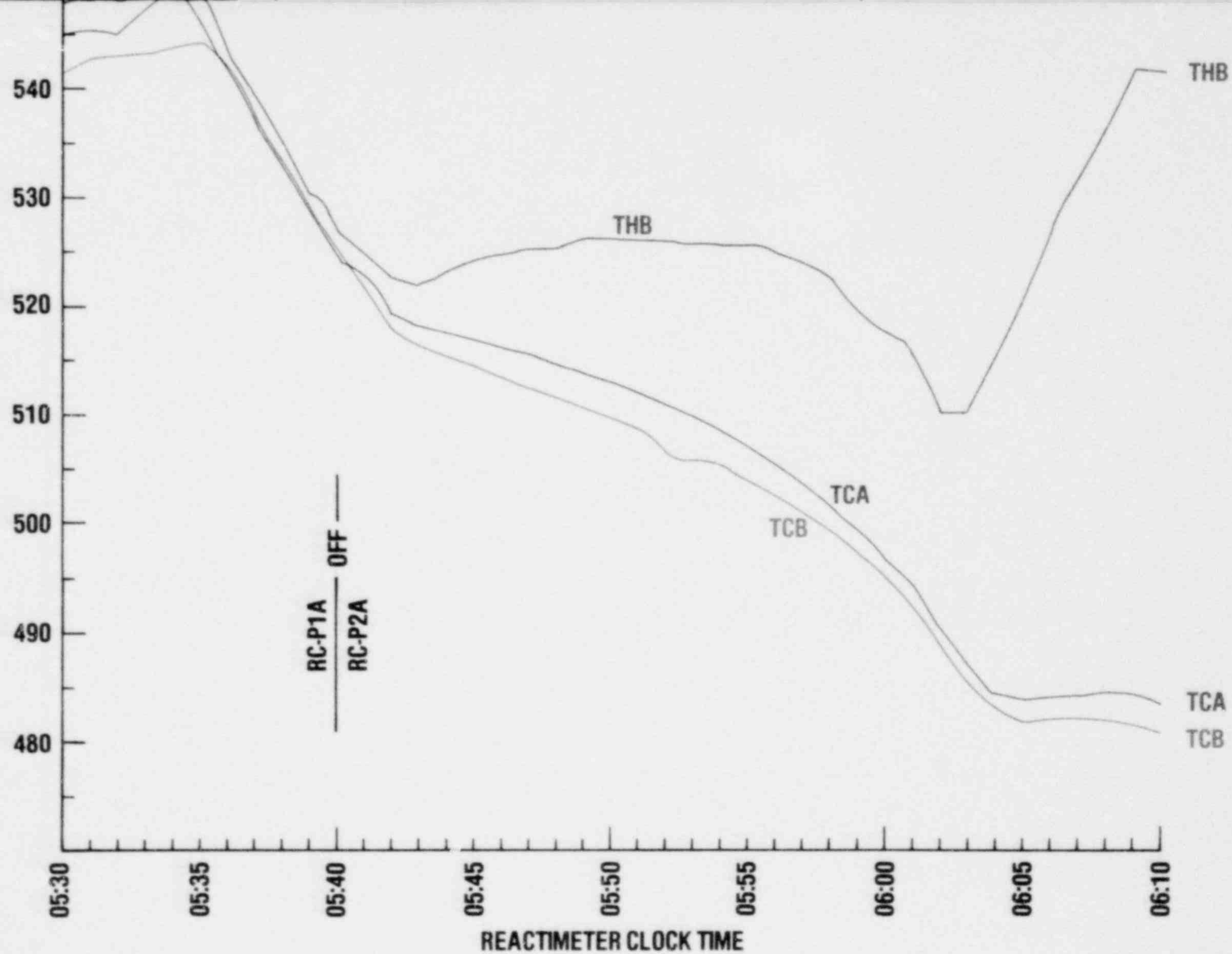
REACTIMETER CLOCK TIME
4:00 5:00 6:00 7:00 8:00 9:00 10:00 11:00 12:00 13:00 14:00 15:00 16:00 17:00 18:00 19:00 20:00 21:00 22:00
ELAPSED TIME
00:00 1:00 2:00 3:00 4:00 5:00 6:00 7:00 8:00 9:00 10:00 11:00 12:00 13:00 14:00 15:00 16:00 17:00 18:00
MUP-1A ON HPI NORMAL
MUP-1B ON HPI NORMAL
MUP-1C ON HPI NORMAL
ON CORE FLOOD TANKS
RC PUMP
PRESSURIZER VENT VALVE OPEN
PRESSURIZER SPRAY VALVE OPEN
BLOCK VALVE OPEN (EMOV)
PRESSURIZER LEVEL
INCHES
400
300
200
100
PRIMARY SYSTEM TEMPERATURES
°F
THB 800
THA 700
TSAT 600
TPZR 500
T SURGE 400
TC A 300
TC A 200
TCB 100
psi
2500
2000
1500
cps



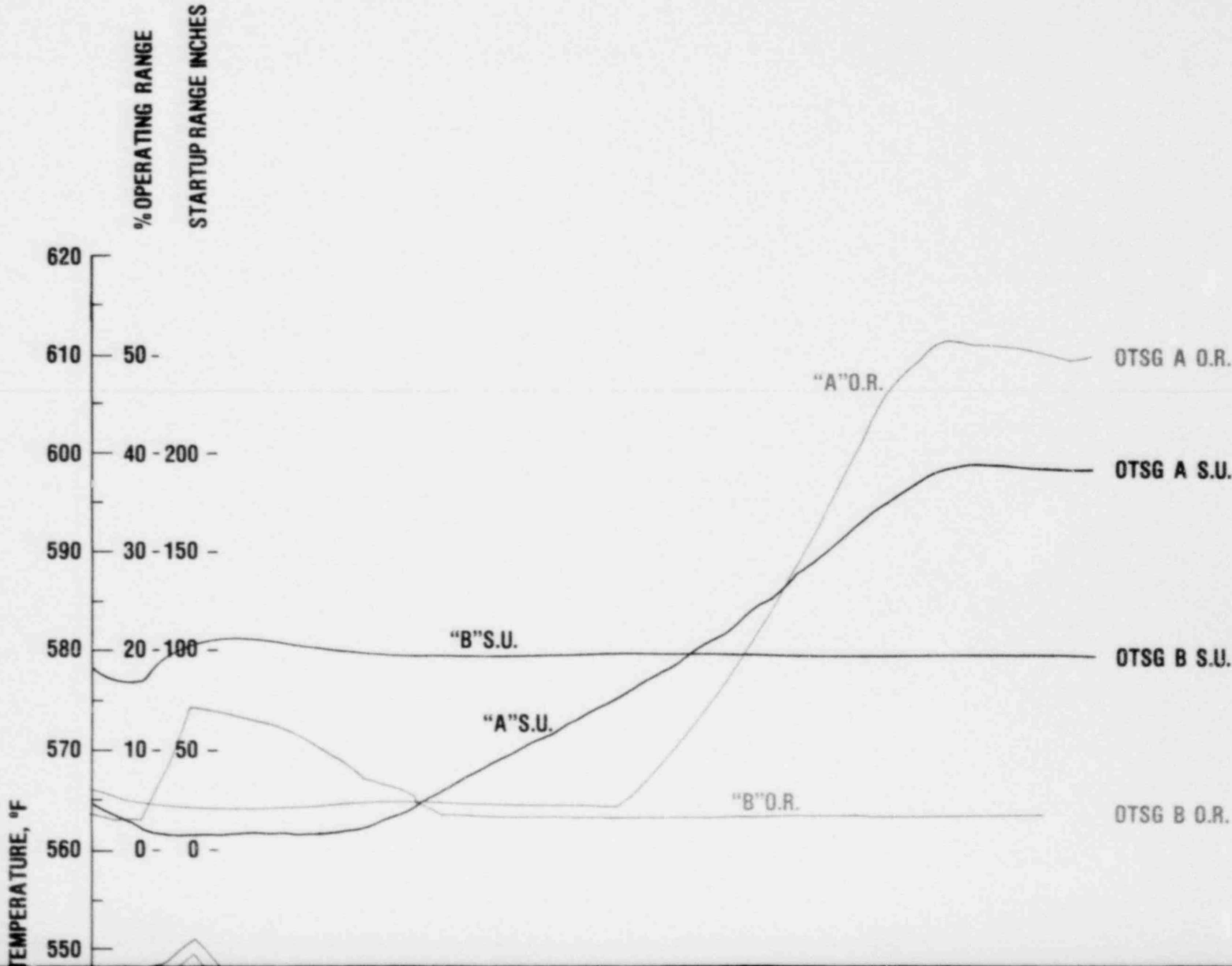
COLOR PLATE IV. HOT AND COLD LEG TEMPERATURES IN THE LATER HOURS OF THE TMI-2 ACCIDENT.



THB (0043)
THA (MUX-2)
THA (0390)
(0043)
19:32:01
19:32:01
TC,A (0043)
TC,A (MUX-4)
TC,A (394)
TCB (MUX-5)
:10
:20
17:30
:40
:50
18:00
:10
:20
18:30
:40
:50
19:00
:10
:20
19:30
:40
:50
20:00



COLOR PLATE V. EXPANDED TIME PLOT OF REACTOR LOOP TEMPERATURES AND STEAM GENERATOR LEVELS.



% OPERATING RANGE
STARTUP RANGE INCHES
TEMPERATURE, °F
620
610
600
590
580
570
560
550
50 -
40 - 200 -
30 - 150 -
20 - 100 -
10 - 50 -
0 - 0 -
"A"O.R.
"B"S.U.
"A"S.U.
"B"O.R.
OTSG A O.R.
OTSG A S.U.
OTSG B S.U.
OTSG B O.R.

began to heat up. When the hottest part of a fuel rod reached 1500±100°F, the rod ballooned and burst, releasing the gases in the gap between the fuel and the cladding to the reactor vessel, and ultimately to the reactor containment via the pressurizer, the open PORV, and the reactor coolant drain tank. At about 2 hours 12 minutes, the pressure in the reactor primary system began to increase even though the PORV was still open. By the time the PORV block valve was closed by a shift supervisor arriving early for his shift duty (2 hours 20 minutes)[162,163] the system pressure had increased from 670 to 750 pounds per square inch (psi).[164] By this time, the two hot-leg temperatures had reached 580° and 650°F. The operation of the RC-P2B at 2 hours 54 minutes produced a great burst of steam and pressure as the coolant from OTSG B caused water in the core to rise and cover parts of the very hot fuel rods. It is believed that the major damage to the fuel rods in terms of oxidation of the cladding in the upper parts of the fuel bundles had occurred by this time and that the rapidly rising water level in the core, coupled with the large increase in steam production, thermally shocked the embrittled fuel rod cladding, shattered it, and produced a bed of fuel rod and fuel pellet debris in the upper part of the core.

The various behaviors of the pressurizer level indication, the hot- and cold-leg temperature time curves, the system pressure versus time, the SRM count rate, the opening and closing of the PORV block valve and the pressurizer spray valve, and the operation of the makeup pumps require close examination in any interpretation of the system behavior. Shortly after the operation of the RC-P2B coolant pump had failed to return the system to normal behavior, the operators attempted to "blow the system down" to a pressure low enough to start removal of core heat by the decay heat removal (DHR) system. After about 2 hours of confused manipulation of valves and pumps, failure of the pressurizer level indication to respond as expected, and failure to get the system pressure low enough for DHR, the operators decided to repressurize the system to "collapse the steam bubbles in the hot legs,"[165] not realizing the presence of large amounts of noncondensible hydrogen. After reaching a system pressure of about 2150 psi, the PORV block valve was cycled open and closed to maintain the system pressure between about 1950 and about 2100 psi. The valve was closed for about 120 to 140 seconds, during which time the system repressurized to the upper limit, and then it was opened for the 70 to 75 seconds required to depressurize to the lower limit. After about 1½ hours of such cycling, the operators became concerned that the valve might fail and decreased the lower limit to about 1850 psi. Eventually, the decision was again made to "blow down" to attempt the DHR system. They never made it. The system pressure finally did get low enough to allow the core flood tanks to inject a small amount of coolant but never low enough to allow the DHR system to take over (which requires a system pressure lower than 400 psi). At about 9 hours 50 minutes, a pressure spike to 28 psig occurred in the containment building, but the operators aware of it considered it a spurious signal and disregarded it.

At about 11 hours into the accident, the hot- and cold-leg temperatures in OTSG A began to approach saturation temperature for the system pressure, indicating that that loop of the system was again approaching the behavior expected for one containing a condensible vapor. OTSG B still seemed to be blocked.

With the final closure of the PORV block valve at 13 hours 22 minutes and the repressurization and the operation of MU-P1C, the system began to refill to the point that a reactor coolant pump could be started at 15 hours 50 minutes. The system was again under control.

The behavior of the system is discussed in more detail in the following pages.

System Status at Successive Time Periods During the Accident

The following discussion of the sequence of events in the reactor primary system related to the damage of the core is broken into 10 time periods for an easier and more comprehensible presentation of important observations, events, and correlations to facilitate a better understanding of the behavior of the system and the interactions occurring therein.

Period I: 0 Hours 0 Minutes to 1 Hour 0 Minutes 4:00 a.m. to 5:00 a.m., March 28, 1979

After the first few minutes of operator action and system response related to a turbine and reactor trip, the reactor primary system came to essentially a steady state condition at about 1100 psi system pressure, about 556°F coolant temperature, a relatively constant leak rate of mixed water and steam or steam only out the open PORV, a slowly increasing buildup of voids (decreasing density of coolant) in the circulating water, and a relatively constant steam pressure in the secondary side of both once-through steam generators. Makeup pump 1A (MU-P1A) was operating with the flow probably throttled to a relatively low rate because the pressurizer level was high at 380 inches. Both OTSGs were filled only to about 4% to 5% on the operating range. (This may be a minimum reading for the instrumentation, rather than a "zero" on the scale used.) System parameters for the first 30 minutes are shown in Figure II-26.

Inferences and Comments—There is no evidence to indicate that the core was damaged at this time, even though there probably had to have been a short period of voiding in the core in the first 5 minutes. The OTSG A was steaming to the condenser.

Period II: 1 Hour 0 Minutes to 1 Hour 40 Minutes
5:00 a.m. to 5:40 a.m.

The following system data are recorded for this period.

RCP—1100 psi ± 25.
RC-P—B's off at 1 hour 12 minutes, A's off at 1 hour 40 minutes.
MU-P1A—on, throttled.
L_{pzr}—380 ± 10 inches.
Atmos. Steam Dump Valve—open.

OTSG A—pressure fell from 1000 to 78 psi, level remained at 5%.
OTSG B—pressure fell from 980 to 160 psi, level rose from 5% to 15%.
SRM—counts rising slowly, oscillating.
T_H—550°F, falling to 510°F.
T_C—550°F, falling to 510°F.
T_{surge}—518°F at 1 hour 18 minutes.
Decay Heat—32 MW at 1 hour.

Inferences and Comments—During this period, the system remained relatively stable except that the vibration of the reactor coolant pumps increased to the point that both B pumps were turned off at 1 hour 12 minutes to prevent damage, and both A pumps were turned off at 1 hour 40 minutes, the end of the period. The source range monitor (SRM) readings became increasingly irregular as the average level increased slowly, indicating the increased amount of voids in the coolant in the downcomer. The condenser vacuum was lost at the beginning of the period, so the atmospheric steam dump valve was opened to permit heat removal from the OTSGs. Hot- and cold-leg temperatures were the same; they decreased about 40°F in the last 8 to 9 minutes of the period.

Period III: 1 Hour 40 Minutes to 2 Hours 20 Minutes
5:40 a.m. to 6:20 a.m.

The following system data are recorded for this period.

RCP—1100 to 670 psi.
RC-P—A's off; B's off.
MU-P1A—on, throttled.
L_{pzr}—370 to 320 inches.
Atmos. Steam Dump Valve—open.
OTSG A—pressure fell from 780 to 530 psi, level rose from 5% to 50% operating range (OR) and held.
OTSG B—pressure varied from 160 to 190 psi, level fell from 15% to 4% OR and held.
SRM—counts fell one decade in 1 to 2 minutes, regained in 6, rose another decade in 15, leveled off.
T_{HA}—rose from 525°F to 680°F, fell to 655°F
T_{HB}—held at 532°F to 536°F for 16 minutes, fell to 513°F, rose to 570°F.
T_{CA}—fell from 510°F to about 490°F.
T_{CB}—fell from 510°F to 480°F.
Decay Heat—25.5 MW at 2 hours.

Inferences and Comments—When the last reactor coolant pumps were shut off at 1 hour 40 minutes, the circulating mixture of water and steam separated. If reverse flow had been induced in OTSG B during the operation of RC-P1A and 2A, the coolant would have drained to the level of the impeller faces of the B pumps, leaving the primary side of OTSG B half full at most. The primary side of OTSG A had to have been nearly empty during pumping in the last few minutes; therefore, only the water in the cold legs would have drained back into the OTSG, possibly leaving it as much as one-fourth full. The drastic decrease in SRM counts indicate that the downcomer was full to about the top of the core immediately after the pumps were turned off. The steady rise in SRM counts over the next 20 or so minutes indicates that the level of the coolant water in the core dropped from about the top of the core to less than half full and leveled off. Various estimates give a level from 7 to 9 feet or more below the top of the 12-foot core. The water boiled off in the core was condensed in the two OTSGs or vented out the pressurizer.

Since the B OTSG may have been filled to the level of the RC-P casing, the condensate in it may have been immediately returned to the core by dribbling through the horizontal section of the B cold legs. However, the level in the primary side of the A OTSG was considerably lower at the start of the

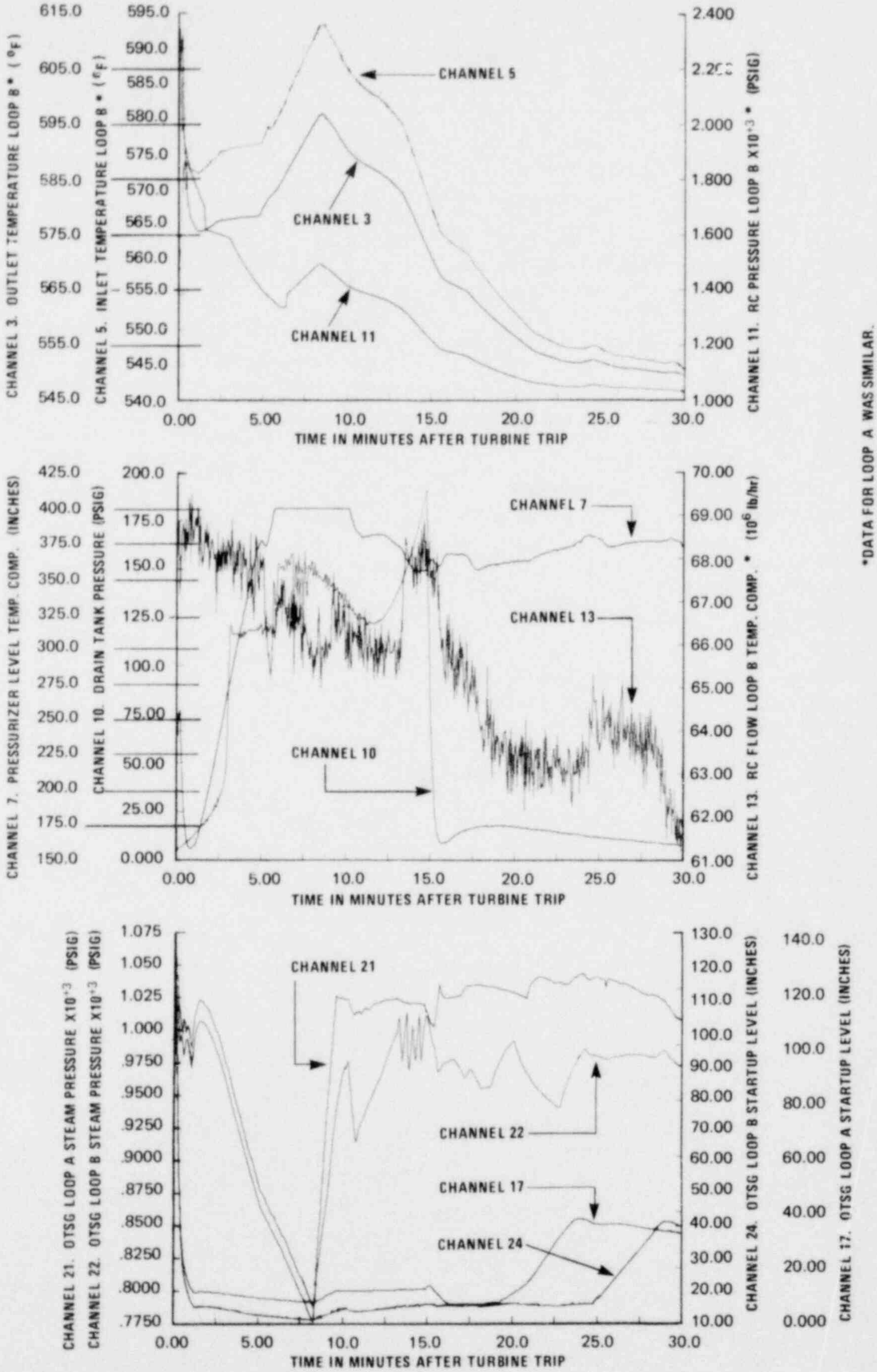


FIGURE II-26. System Parameters for the First Thirty Minutes of the TMI-2 Accident period, so that much more condensate was required to fill it to the point of returning the condensate to the core. The core was then being cooled by refluxing and the loss out the pressurizer and the letdown line. As the coolant in the core was boiled off, the exposed fuel rods began to heat up because they were cooled only by steam at very low flow rates. When the hottest part of the fuel rods reached a temperature of about 1500°F±100°F, the cladding of the fuel rods ballooned and burst and released the gases from the gap between the fuel pellets and the interior surface of the cladding. It is estimated in Section II.C.2.b that the hottest fuel rods in the center bundle (highest power) reached temperatures above 3500°F about 45 minutes after the pumps were turned off, and many others reached such temperatures in the minutes following. The hot and cold legs of the OTSGs were voided, and superheated steam was produced in the top of the core in the first few (about 5) minutes after the top of the core was uncovered. The period ended when the block valve for the PORV was closed, and the loss of system pressure and coolant from the open pressurizer PORV was stopped. However, the loss of coolant from the letdown line continued.

Period IV: 2 Hours 20 Minutes to 2 Hours 54 Minutes 6:20 a.m. to 6:54 a.m.

The following system data are recorded for this period.

RCP—670 to 1300 psi.

RC-P—all off.

MU-P1A—on, throttled.

L_{pzr}—level at 300 inches until 2 hours 51 minutes, then rose to 330 inches.

Atmos. Steam Dump Valve—open.

OTSG A—pressure fell steadily from 530 to 320 psi, level rose from 50% to 68% OR.

OTSG B—pressure held at 190±10 psi, then rose to 300 psi, level rose from 4% to 40% OR.

SRM—counts slowly decreased until 2 hours 54 minutes.

T_{HA}—rose from 655°F to 810°F over the period, then fell to 800°F.

T_{HB}—rose from 570°F to 770°F over the period.

T_{C1A}—fell from 490°F to 400°F and recovered to 430°F.

T_{C2A} rose from 495°F to 500°F, then fell to 450°F.

T_{CB}—fell from 480°F to 440°F.

Block Valve—closed at 2 hours 20 minutes.

Reactivity—detected at 2 hours 25 minutes in primary loop by area monitor.

Inferences and Comments—The leak out the pressurizer PORV stopped when the PORV block valve was closed at 2 hours 20 minutes. If the wide range reactor system pressure recording strip chart can be indexed to an accuracy of ±3 minutes, it appears that the decrease in pressure in the reactor primary system stopped abruptly at 2 hours 12 minutes and began a relatively rapid increase at least 4 minutes before the block valve was closed at 2 hours 20 minutes (8±3 minutes for the strip chart, ±1 minute on block valve closure). The pressure ramp shows two definite inflection points, at 2 hours 25 minutes with 630 psi indicated and at 2 hours 54 minutes with 1300 psi indicated. The first occurred very close in time to abrupt changes in the hot- and cold-leg temperatures for the OTSG, and the second appears to be in time coincidence with the starting of RC-P2B at 2 hours 54 minutes.

The rapidly increasing hot-leg temperatures for both OTSGs can occur only if superheated steam is present in the hot legs and they are voided of water. The pressurizer level indicator showed a rise in the pressurizer of 74 inches in 5 minutes. This change in level is equivalent to 237 cubic feet of water (3.2 ft^3 volume per inch of level in the pressurizer). It is thought that the major oxidation damage to the Zircaloy cladding occurred during this period. This is discussed in detail in Section II.C.2.b.

During normal power operation, the radiation detector HP-R-213 (incore instrument panel area monitor) located above the primary system is sensitive to the short half-life N^{16} isotope formed from the oxygen in the coolant water, which it sees mainly in the hot leg of the primary coolant loop. In this case, the detector would sense radioactivity (gamma) from xenon (Xe) and krypton (Kr) released by burst fuel rods as it was entrained by steam flowing through the hot legs into the OTSGs and still remaining in the primary system. It did so at 2 hours 25 minutes before any activity was released to the containment.

Period V: 2 Hours 54 Minutes to 3 Hours 12 Minutes 6:54 a.m. to 7:12 a.m.

The following system data are recorded for this period.

RCP—1300 to 2100 to 2140 psi.

RC-P—2B on.

MU-P1A—on, throttled.

L_{pzr}—changed from 330 to 380 to 360 inches.

Atmos. Steam Dump Valve—closed at 3 hours.

OTSG A—pressure fell steadily, level fell from 68% to 60%.

OTSG B—pressure changed from 300 to 410 to 380 psi, level rose from 40% to 60%.

Steaming to Condenser.
SRM—count rate dropped one decade in seconds and then rose to recover most of the drop by 3 hours 12 minutes.
T_{HA}—changed from 800°F to 770°F to 780°F.
T_{C1A}—430°F to 480°F.
T_{C2A}—changed from 450°F to 430°F to 440°F.
T_{CB}—changed from 440°F to 470°F to 445°F.
Block Valve—closed.
PZR Spray Valve—open.
Decay Heat—22 MW at 3 hours.

Inferences and Comments—After the RC-P2B was started, the operators reported[161] that there was water flow through the RC-P2B pump for only a very short time (a few minutes at most), as the vibrations and low power in the pump were again observed very shortly after it was started. The reactimeter data show flow in the OTSG A hot leg for less than 9 seconds. If the OTSG B had been half full at the time the 2B pump was started, less than about 1000 ft^3 of water would have been pumped into the core. The very sharp increase in reactor coolant pressure starting at 2 hours 54 minutes was probably due to a very large burst of steam produced when the water from the OTSG B hit the very hot core.

Calculations by R. Cole, Sandia (Appendix II.9) estimate that 1000 ft^3 of gas flowed by the flow meter in the hot leg of OSTG B. Because the flow in the hot leg was induced by the flow of water leaving the OTSG B through the pump, an equal volume of water should have been displaced. At about the same time, there was an abrupt rise in the steam pressure in OTSG B and a small, sharp decrease in level.

Period VI: 3 Hours 12 Minutes to 5 Hours 18 Minutes 7:12 a.m. to 9:18 a.m.

The following system data are recorded for this period.

RCP—fell from 2140 to 2000 psi rapidly, fell more slowly from 2000 to 1500 psi, then fell to 1240 psi with three intermediate periods of increase.
RC-P—all off.
MU-P1A—on until 4 hours 21 minutes, then locked out, 1B and 1C on at 4 hours 27 minutes, 1A and 1C on HPI for 6 minutes at 3 hours 18 minutes and 3 hours 57 minutes, 1C on normal at 3 hours 24 minutes for 12 minutes, 1C on normal for 17 minutes at 4 hours 3 minutes.
L_{pzr}—fell from 360 inches to 230 inches in 13 minutes then rose to 400 inches in 20 minutes and remained above 390 inches for remainder of period.
Atmos. Steam Dump Valve—opened at 4 hours 30 minutes.
OTSG A—fell from 200 to 40 psi at 3 hours 42 minutes, decreased to 20 psi at 4 hours 30 minutes, rose to 80 psi at end of period; level ranged from 60% to 48%.
OTSG B—pressure fell slowly from 380 to 320 psi, level rose from 58% to 65% and held.
Condenser—steaming until vacuum lost at 4 hours 30 minutes.
SRM—count rate dropped one decade abruptly at 3 hours 18 minutes when makeup pumps on HPI, fell steadily about one-third decade to 3 hours 43 minutes, jumped one-third decade abruptly, and slowly decreased to about 2×10^3 cps at end of period.
T_{HA}—780°F at start of period, rose 790°F at 3 hours 18 minutes, fell to 700°F at 3 hours 28 minutes, rose to 760°F at 3 hours 42 minutes, fell to 690°F at 4 hours, then rose and held at 700±10°F for rest of period.
T_{HB}—very similar behavior but with peak temperatures at about 820°F, ending at 745°F.
T_{C1A}—fluctuated from 480°F at start to 320°F at 3 hours 41 minutes to 440°F at 3 hours 45 minutes to 310°F at 4 hours 4 minutes to 350°F at 4 hours 13 minutes to 300°F at 4 hours 30 minutes.
T_{C2A}—rose from 440°F at start to 485°F in 6 minutes, fell to 320°F at 3 hours 43 minutes, rose to 510°F at 3 hours 48 minutes, fell to 450°F at 4 hours, and fell to 190°F at end of period.
T_{CB}—fell from 445°F at start to 220°F at end of period with several oscillations of 20 to 40°F with sharp changes in slope.
Block Valve—open and closed several times in period.
PZR Spray Valve—open from 3 hours 42 minutes to 4 hours 6 minutes.
Decay Heat—20 MW at 4 hours.

Inferences and Comments— At the start of the period, the reactor coolant pressure dropped rapidly to 2000 psi. When the makeup pumps were turned to high pressure injection (HPI) of about 500 gallons per minute from each pump, the influx of water apparently chilled the downcomer region, and the pressure dropped very rapidly to 1500 psi and leveled off as MU-P1C was changed from HPI to normal flow. When HPI by MU-P1A was stopped in another 6 minutes, the pressure in the system rose to about 1560 psi at 3 hours 39 minutes. When the block valve was opened at 3 hours 42 minutes and the pressurizer spray valve was opened at the same time, the system pressure decreased to 1480 psi and then increased to 1710 psi very quickly. When

HPI was again initiated at 3 hours 56 minutes on both MU-P1A and 1C, the pressure again began a decrease to 1510 psi at 4 hours 6 minutes. The pressurizer spray valve was closed at 4 hours 6 minutes and the block valve opened for about 6 minutes between 4 hours 12 minutes and 4 hours 18 minutes. The block valve was opened again at 4 hours 36 minutes and remained open for the rest of the period. The RC pressure decreased rapidly when the MU-P1C was started again at about 4 hours 27 minutes to 1310 psi, and rose to 1390 psi at 4 hours 54 minutes, even though the block valve was opened at 4 hours 36 minutes. The pressure then decreased to about 1250 psi at the end of the period, when the block valve was again closed to repressurize the system.

MU-P1A was shut down and locked out for the remainder of the accident because the operators were having considerable difficulty in keeping it in operation. It tripped off and had to be restarted many times during the first 4 hours of the accident period. When it could not be restarted after the last trip at 4 hours 21 minutes, the operators "locked it out" to prevent its actuation during activation of the engineered safety system, and replaced it with MU-P1B. However, there was a period of about 6 minutes when no makeup coolant was flowing into the system. It should be noted that the major responses of the system seem to occur with the operation of MU-P1C, the block valve, and the pressurizer spray valve. Operation of MU-P1A or 1B seemed to have little or no effect on either system temperatures or the pressure.

The large and sudden increases in the cold-leg temperatures of OTSG A were almost coincident with the opening of the block valve and the pressurizer spray valve. The sharp but relatively small increase in the SRM signal was also coincident with the opening of these two valves.

Period VII: 5 Hours 18 Minutes to 7 Hours 39 Minutes 9:18 a.m. to 11:39 a.m.

The following system data are recorded for this period.

RCP—increased from 1240 psi to 2150 psi, cycled between about 2150 psi and 1850 to 1900 psi with about 2 minutes of pressure increase and about 1 minute of pressure decrease.

RC-P—all off.

MU-P—both 1B and 1C operating, with various degrees of throttling.

L_{pzr}—constant at 400 inches.

Atmos. Steam Dump Valve—open.

OTSG A—pressure dropped slowly from 80 psi to less than 20 psi at 7 hours, remained below 20 psi for rest of period, level remained at 48% until refill started at 5 hours 54 minutes, reaching 100% OR at 7 hours.

OTSG B—pressure dropped slowly from 320 psi to 290 psi at end of period, level fell from 65% to 62%.

T_{surge}—310°F at 5 hours 15 minutes.

T_{pzr}—recorded at 345 to 350°F in last half hour of period.

SRM—count rate dropped slowly from beginning to end of period, with one small "bump" occurring between 6 hours 45 minutes and 7 hours 6 minutes.

T_{HA}—increased from 690°F at start to 735 to 740°F at 6 hours and remained at 735±5°F to end of period.

T_{HB}—paralleled T_{HA} exactly but at 50°F higher temperature.

T_{CA}—temperature record appears that of 1A. Rose from 190°F at start to 220°F at 5 hours 45 minutes and held for rest of period.

T_{CB}—dropped from 220°F at start to 210°F at 5 hours 30 minutes, then gradually fell to 185°F at end.

Block Valve—cycled open and closed to bleed off pressure to prevent opening of safety valves.

PZR Spray Valve—closed.

Decay Heat—17 MW at 6 hours.

Inferences and Comments— The operators stated that during this period they planned to collapse the "steam bubbles" in the hot legs of the OTSGs by pressurizing, so that they could ultimately put into effect the natural circulation mode of cooling the system.[166] Since the system pressure was increasing to the level at which the safety valves would be opened, the block valve was manipulated to keep the pressure as high as possible without "lifting the safeties." The system increased in pressure from about 1900 psi to 2070-2100 psi in 2 to 2½ minutes (114 to 150 seconds) and decreased from about 2100 to about 1980 psi in about 70 to 75 seconds. This procedure was continued for more than 1½ hours. Because the operators feared that the block valve would fail through excessive use, leaving them with no control of system pressure, they decided to depressurize to less than 400 psi so that the system could receive coolant from the core flood tanks. During this period, the OTSG A was filled to 100% of the operating range (OR), but OTSG B was left isolated and at 60% OR. There is a small "bump" in the SRM counts at 6 hours 45 minutes to 7 hours that cannot be keyed to any system parameter and cannot be explained. The block valve was opened at 7 hours 39 minutes and remained open for more than 1½ hours. Pressurizer temperatures were requested from the plant computer by the operators for the first time during the accident. If a bubble existed in the pressurizer, the temperature of about 345°F would be equivalent to a steam pressure of about 130 psia, and the remainder of the pressure would have been due to a noncondensible gas, presumably hydrogen.

Period VIII: 7 Hours 39 Minutes to 10 Hours 21 Minutes 11:39 a.m. to 2:21 p.m.

The following system data are recorded for this period.

RCP—dropped from 2050 psi at the start to 1580 in 4 minutes, to 1460 at 7 hours 51 minutes to 1120 psi at 7 hours 57 minutes, then at an exponential decay to about 500 psi at 9 hours, held 500 to 490 psi to 9 hours 48 minutes (utility typer gives 440 to 450 psi), rose to 550 psi at 10 hours 5 minutes and fell to 520 psi at end of period.

RC-P—all off.

MU-P1C—on until 9 hours 6 minutes, 1B on for entire period; HPI on both at 9 hours 50 minutes.

L_{pzr}—395 to 400 inches.

Atmos. Steam Dump Valve—closed at 9 hours 15 minutes, no heat removal from system except during letdown flow and opening of pressurizer valves.

OTSG A—pressure near atmospheric to 10 hours 18 minutes, rose to 40 psi at 10 hours 21 minutes, level constant at 95% OR.

OTSG B—pressure decreased slowly from 280 to 250 psi except for small increase to 310 psi at 7 hours 54 minutes; level constant at 60% to 65% except for brief rise to 66% at 7 hours 54 minutes.

T_{surge}—requested twice by operators, 310°F at 8 hours and 330°F at 8 hours 18 minutes. Surge line temperature is not reported again.

T_{pzr}—pressurizer temperatures were requested several times by the operators, circa 350°F, and then were reported as "trend data" in Operators Group C Summary afterward. Temperature held at 350°F with slight increase with time until 9 hours 30 minutes when an increasing rate of temperature rise began. At 10 hours 21 minutes, the pressurizer temperature was within a few degrees of, or equal to, saturation temperature for the system pressure. It did not rise higher than saturation temperature for the system for the remainder of the accident.

SRM—count rate increased slowly from start of period until 9 hours 48 minutes, showed a small sharp increase and decrease, then returned to the same curve as before, and remained constant for rest of the period.

T_{HA}—dropped sharply from 730°F at start of period to 700°F in 6 minutes at 7 hours 51 minutes, then very slowly increased to 715°F at 9 hours 51 minutes, dropped sharply to 660°F at 10 hours, and dropped slowly to 650°F at end of period with one excursion to 630°F and return in 9 minutes.

T_{HB}—paralleled T_{HA} behavior except 50 to 80°F higher, ending period at 725 to 730°F.

T_{CA}—fell slowly from 220°F at 7 hours 39 minutes to 160°F at 9 hours, rose to 190°F at 10 hours, and held for remainder of period.

T_{CB}—fell gradually from 185°F at 7 hours 39 minutes to 150°F at 9 hours and held.

Block Valve—closed at 9 hours 15 minutes for 6 minutes, closed at 9 hours 32 minutes for 17 minutes, and opened from 9 hours 49 minutes through end of period.

PZR Spray Valve—opened at 8 hours, closed at 9 hours, and opened at 10 hours.

Pressurizer Vent Valve—opened at 7 hours 54 minutes and closed at 9 hours 9 minutes.

Engineered Safety System Actuation—at 9 hours 50 minutes on high building pressure, decay heat pumps started, reactor building isolated, reactor building sprays started, both makeup pumps on HPI for 1 minute. Reactor building spray pumps stopped at 9 hours 56 minutes.

Reactor Building Pressure—spiked to 28 psi at 9 hours minutes, observable on strip-chart recording reactor building pressure and as an inverse pressure on the OTSG steam pressures (since the pressure sensors use building pressure as the reference pressure).

Decay Heat—14 MW at 9 hours.

Inferences and Comments—During this period, the operators were attempting to "blow the system down" to get to a pressure low enough to allow the system to be opened to the core flood tanks. This would allow injection from the core flood tanks when the system pressure dropped below 600 psi, but significant flow would not occur until the system pressure dropped to less than 200 psi or so. The pressure leveled off at about 490 to 500 psi without dropping below that for about 45 minutes. Also, the system pressure remained at 490 to 500 psi for almost 30 minutes even with the PORV block valve closed for the time period around 9 hours 30 minutes. At about the time of the engineered safety system actuation, when the makeup pumps went onto HPI, the system pressure started rising slowly to about 550 psi at about 10 hours 5 minutes and then slowly dropped down to about 520 psi at the end of the period.

The reactor building pressure pulse recorded at 9 hours 50 minutes on the reactor building strip chart was thought by the operators to be a spurious signal or "electrical noise," both then and later. However, the inverse of the pressure pulse can be seen by plotting the steam pressures of the OTSGs for the time period 9 hours 45 minutes to 9 hours 55 minutes, as shown in Figure II-27, with the data taken from the reactimeter tabulation at 3-second intervals. The pressure sensors of the OTSGs use the reactor building pressure as the reference pressure. The data show that the pressure rose to a peak over a 9-second time interval, decayed to nearly the original pressure in about 100 seconds, and then dropped suddenly to below the original pressure. This was the "hydrogen burn" to be discussed later.

While the reactor core was floating on the core flood tanks from 8 hours 30 minutes to 9 hours 12 minutes because system pressure did not drop below 400 psi, the response of the core flood tank pressure showed that only a small amount of water could have entered the primary system. For a part of the time, the pressure in core flood tanks was rising as indicated by high pressure alarms (the check valves may have leaked).

One of the more important observations of the period may be that the temperature of the pressurizer rose to the saturation temperature for the system (based on the system pressure) for the first time since about the time the primary coolant pumps were turned off at 1 hour 40 minutes.

The "blip" in the SRM count rate strip chart should be noted, but no cause can be assigned to it by our analysis, and it is not quite in time coincidence with the reactor building pressure spike at 9 hours 50 minutes, although it may be within the timing coincidence error of the several strip charts and data acquisition systems.

Period IX: 10 Hours 21 Minutes to 13 Hours 15 Minutes 2:21 p.m. to 5:15 p.m.

The following system data are recorded for this period.

RCP—fell from about 520 psi at the start of the period to 460 to 470 psi with a drop to 409 psi for 1 to 2 minutes at 10 hours 36 minutes, and rose back to 420 to 425 psi. Slow rise starting at 11 hours 10 minutes, leveled off at 650 to 660 psi at 12 hours 39 minutes for rest of period.

RC-pumps—off.

MU-P—1B on for entire period, throttled 1C on for 6 minutes at 10 hours 30 minutes, on for 10 minutes at 11 hours 18 minutes and for 3 minutes at 11 hours 33 minutes; no HPI in the period.

L_{pzr}—380 to 400 inches from 10 hours 21 min to 11 hours 3 minutes, dropped very rapidly to 175 inches at 11 hours 18 minutes, held 175 inches to 11 hr 33 minutes, rose steadily to 400 inches at 12 hours 30 minutes, dropped to 390 to 380 psi for rest of period.

Atmos. Steam Dump Valve—closed.

Condenser Vacuum—pumps started at about 13 hours.

OTSG A—pressure rose from 40 psi at start of period to 80 psi at 10 hours 45 minutes with abrupt change at 10 hours 30 minutes; dropped slowly to about 50 psi at 11 hours 45 minutes, then rose at increasing rate to 160 psi at end of period; level constant at 97% to 98% OR.

OTSG B—pressure dropped slowly from 250 psi at start to 240 at 11 hours 30 minutes, then rapidly to 150 psi at 11 hours 54 minutes, and held at 150 for rest of period; level dropped from 60% at start to 57% at 11 hours 33 minutes and rose rapidly to 96% OR at 12 hours, holding 96% for rest of period.

T_{pzr}—rose slowly or level, within a few degrees of saturation temperature for pressure of the system throughout the period.

SRM—count rate increased only very slightly during the entire period.

T_{HA}—dropped very rapidly from 650°F at 10 hours 21 minutes to 500°F at 10 hours 32 minutes, rose very rapidly to 570°F at 10 hours 40 minutes, fell to 460°F at 11 hours 6 minutes, started rapid rise at 11 hours 15 minutes to 560°F at 11 hours 23 minutes, rose slowly to 590°F and held to 12 hours 33 minutes, dropped at increasing rate to circa T_{sat}=500°F at 13 hours 6 minutes and held at T_{sat} for rest of the period.

T_{HB}—rose slowly from 725°F at start to 755°F at 12 hours 33 minutes, dropped very rapidly to 630°F at 12 hours 42 minutes, rose to 710°F at 13 hours 3 minutes and to 715°F at 13 hours 15 minutes.

T_{CA}—two curves observable, 1A and 2A cold legs, behavior is different. T_{C2A} preceded T_{C1A}, and reached higher temperatures. T_{C2A} reached 440°F at 11 hours 21 minutes, T_{C1A} reached a maximum of 400°F at same time. Both were about 360±10°F at 11 hours 36 minutes, and both reached T_{sat} circa 480°F at 12 hours 15 minutes and held at T_{sat} for remainder of the period.

T_{CB}—fell from 150°F at start of period to 125°F at 11 hours, held 125°F to 11 hours 45 minutes, rose rapidly to peak at 170°F at 12 hours, fell slowly to 145°F at end of period.

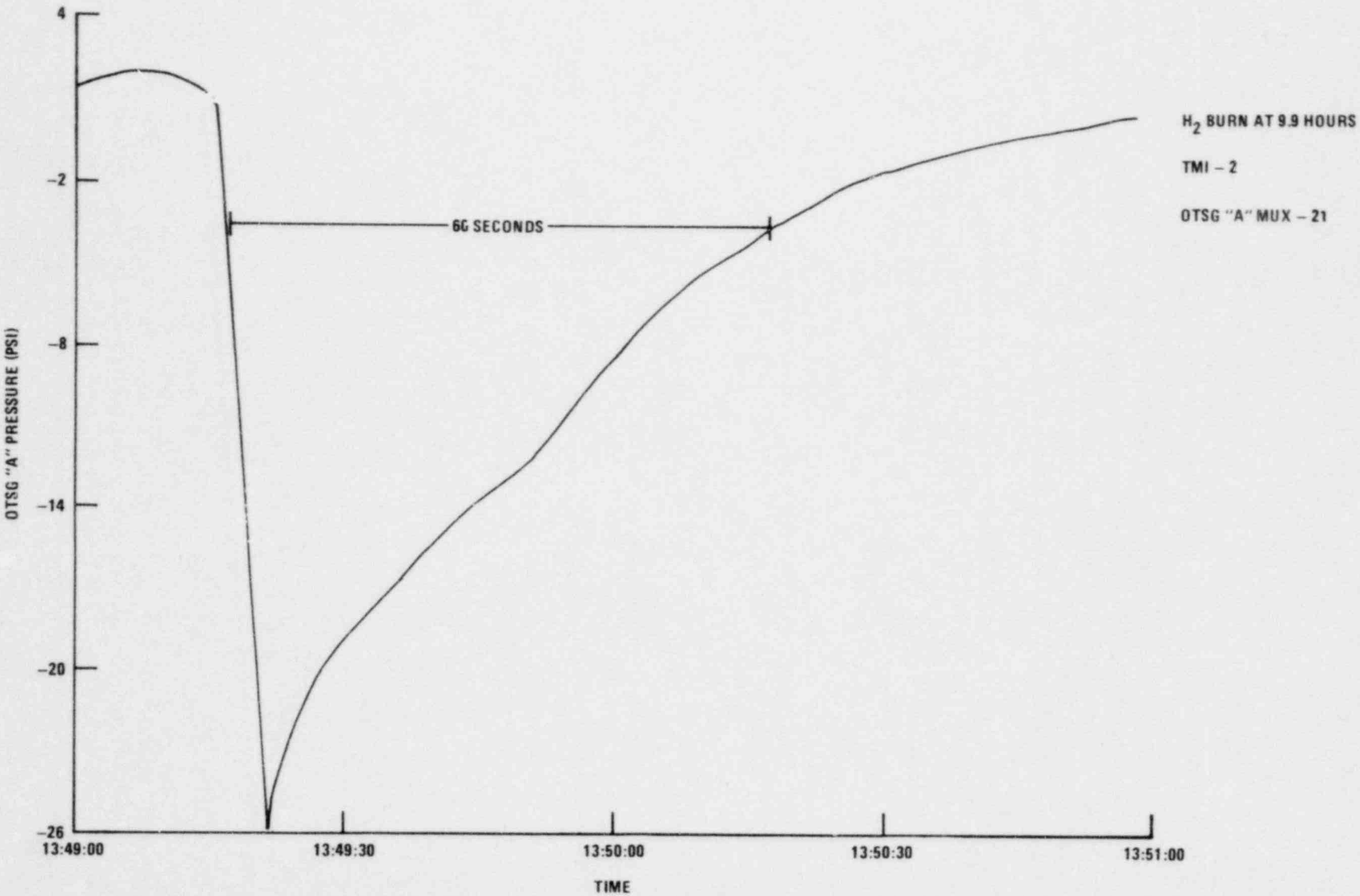


FIGURE II-27. Hydrogen Burn at 9.9 Hours

Block Valve—closed at 11 hours 9 minutes, opened at 12 hours 30 minutes, closed at 12 hours 40 minutes, opened at 12 hours 45 minutes, and open for rest of period.
PZR Vent Valve—opened at 12 hours 45 minutes, closed at 12 hours 57 minutes.
PZR Spray Valve—closed at 11 hours 57 minutes.
Decay Heat—13 MW at 12 hours.

Inferences and Comments— During this period of time, the reactor primary system displayed some of the symptoms of thermal-hydraulic behavior expected of a system having condensible vapor in it. The hot and cold legs of OTSG A showed a behavior indicating that there was again steam flow and condensation in the A steam generator, and the response of the steam generator pressure was in accordance. However, the pressurizer level dropped 230 inches between 10 hours 54 minutes and 11 hours 18 minutes, equivalent to a volume displacement of 736 ft^3. The system pressure rose less than 100 psi, and it was delayed relative to the drop in the pressurizer level. An operator responded to the sudden drop in the pressurizer level indication by greatly increasing the makeup flow rate.[165]

Although the OTSG A hot-leg temperature reached the saturation temperature for the system (based on system pressure) for a short time, it rose to about 100°F superheat again for much of the remainder of the period and again fell to saturation temperature at the end of the period. This may be related to the startup of MU-P1C and the closing of the PORV block valve. The cold-leg temperatures for OTSG A reached the system saturation temperature in the middle of the period and held it for the rest of the period. The OTSG A hot- and cold-leg temperatures, the pressurizer temperature, and the system saturation temperature were the same for the first time since the reactor coolant pumps were turned off.

Period X: 13 Hours 15 Minutes to 16 Hours 5:15 p.m. to 8:00 p.m.

The following system data are recorded for this period.

RCP—the pressure was constant at 660 psi until 13 hours 25 minutes, rose to 2350 psi at 14 hours 48 minutes, fell to 2320 psi at 15 hours 35 minutes, dropped almost instantly to 1500 psi, rose rapidly back to 2120 psi, and fell to 1350 psi at 15 hours 50 minutes.

RC-P—pump 1A "burped" at 15 hours 33 minutes to check operation, started again at 15 hours 50 minutes to run for many days.
MU-P1B—on for entire period, 1C started at 13 hours 21 minutes, throttled at 14 hours 41 minutes, stopped at 14 hours 43 minutes, run for 7 minutes at 15 hours 32 minutes and 11 minutes at 15 hours 45 minutes.
L_{pzr}—dropped rapidly from 390 inches at 13 hours 18 minutes, to 275 inches at 13 hours 30 minutes, rose slowly to 290 inches at 13 hours 54 minutes and rapidly to 400 inches at 14 hours 21 minutes.
Atmos. Steam Dump Valve—closed.
Steaming to Condenser—started at 14 hours for OTSG A.
OTSG A—pressure dropped slowly from 160 psi at start of period to nearly zero at 15 hours, rose from circa 10 psi at 15 hours 30 minutes to 70 psi at 15 hours 42 minutes, and fell to 20 psi at 16 hours; level constant at 95% to 96% except for "dip" to 88% at 13 hours 51 minutes.
OTSG B—pressure constant at 150 psi to 15 hours 30 minutes, dropped to 40 to 50 psi at 16 hours.
T_{pzr}—increased slowly or held steady for entire period—no decrease; started at saturation temperature for the system pressure but did not increase with it as system pressure rose to 2350 psi; reached 520°F at 16 hours.
SRM—count rate was steady or showed only very slight increase over the entire period except for "bump" at 14 hours 30 minutes.
T_{HA}—rose from circa 500°F at start of period to 590°F at 14 hours 45 minutes and fell slowly to 575°F at 15 hours 33 minutes, dropped sharply to 420°F when RC-P1A "burped," rose again to 525°F at 15 hours 50 minutes and dropped to 365°F when RC-P1A started.
T_{HB}—responded as T_{HA} but 150 to 200°F higher.
T_{CA}—T_{C2A} started rapid drop from 490°F at 13 hours 30 minutes to 315°F at 13 hours 45 minutes to 280°F at 14 hours 9 minutes, held to circa 14 hours 45 minutes, and started to rise to 415°F at 15 hours 33 minutes, dropped to 330°F, and ended period at 365°F.
T_{C1A}—behaved much the same way after falling slowly from 490°F at 13 hours 30 minutes to 425°F at 14 hours.
T_{CB}—held 145°F from start of period to 14 hours, rose rapidly to 210°F at 14 hours 15 minutes and slowly to 230°F at 14 hours 39 minutes, fell to 210°F at 15 hours 33 minutes, and rose to 365°F at 15 hours 50 minutes.
Block Valve—closed at 13 hours 24 minutes and remained closed for rest of period.

PZR Vent and Spray Valves—closed.
Decay Heat—12 MW 15 hours.

Inferences and Comments— At the start of this period, the operators decided to repressurize and to increase makeup flow to collapse the "steam bubbles" thought to exist in the hot legs of the steam generators.[167,168] The pressurizer temperature continued its slow rise but did not follow the saturation temperature based on the system pressure. This indicated that the system was not being pressurized by a steam bubble in the pressurizer but by makeup flow and other factors. The system pressure showed a very rapid increase at 14 hours 35 minutes rising from 1400 psi to 1900 psi in less than 2 minutes. The rate of increase then slowed, indicating a massive input of heat to the system vapor phase. The reactor coolant pump 1A was successfully "jogged" at 15 hours 33 minutes and flow, motor amperage, and pump vibration were found to be acceptable. The motor had to cool for 15 minutes before it could be started again.

RC-P1A was started again at 15 hours 50 minutes and ran continuously for more than a week. The hot- and cold-leg temperatures almost immediately merged to within about 5°F of the same value, or 365°F, although the "quenching" of the hot leg of OTSG B appeared to be delayed by 1 to 2 minutes. The system pressure dropped very rapidly to 1350 psi, rose to 1400 psi in about 8 minutes, and then fell smoothly and slowly to 1000 psi at 18 hours (10:00 p.m.). MU-P1B continued to run. The system was "stable," the core was being cooled by flowing water, and OTSG A was steaming to the condenser.

Additional Data

In addition to the facts given above on the various parameters of the reactor primary system, there are certain other sets of data pertinent to any interpretation of the sequence of events and their effects during the accident. Among these are the changing levels of the borated water storage tank (BWST), the indication of the incore thermocouples located just above the top of the fuel rods in the instrumentation tubes of 52 of the fuel assemblies (there were 177 assemblies in the core), and the indications of the self-powered neutron detectors (SPND) located in the same instrumentation tubes as the incore thermocouples.

Borated Water Storage Tank Discharges

The normal reactor trip procedure requires that the supply to the makeup pumps be from the borated water storage tank (BWST), and injection of borated water via the high pressure injection (HPI) valves (MU-V-16A, 16B, 16C, and 16D). The BWST supplies the water necessary for the reactor coolant system (RCS) beyond that available in the normal makeup tank (small capacity—4500 gallons) and the reactor coolant bleed holdup tanks (three tanks at 80 000 gallons each).[169] The changes in level in the BWST can be used to calculate the amount of water injected into the RCS by the makeup pumps, providing that all of the water removed from the BWST is injected and there is no other path for loss of water from the BWST. The BWST levels are not normally recorded except in operator logs and appear on the alarm printer when specifically requested by an operator. The data available were compiled previously[170] and are tabulated in Table II-54.

Only 15 000 gallons of water were removed from the BWST during the first 3½ hours of the accident; 132 000 gallons were removed in the next 3½ hours and 50 000 more in the following 2-1/3 hours. It is important to note that more than twice the volume of the RCS (90 000 gallons) was removed from the BWST in the first 9 hours of the accident and supposedly was injected into the RCS by the makeup pumps. It is also important to note that 37 000 gallons were removed from the BWST in the 1½ hours immediately following the last closure of the PORV block valve to repressurize the RCS. There are at least three paths for water to be removed from the BWST without being injected into the reactor coolant system. These include: (1) a pipe in the A line in the containment (feeding the A cold leg of OTSG B) cracked between two check valves, which leaks significantly only at higher pressures, (2) the DH-V6A and 6B valves, which were opened unwittingly, allowing the BWST water to drain into the sump, and (3) a relief valve on the makeup tank, which opened as it did several days later, to provide

TABLE II-54. Water usage from the borated water storage tank[170]

Accident Time	Level (ft)	Total (gal)	Used in Period (gal)	Avg. (gpm)
3 h 30 min	53.04	15 000	15 000	70
6 h 55 min	37	147 000	132 000	643
9 h 15 min	31	198 000	40 000	357
13 h 20 min	26.5	234 000	37 000	150
14 h 45 min	22	271 000	37 000	560

a path from the BWST through the makeup pumps to the makeup tank and vented to the reactor coolant bleed holdup tanks.

Incore Thermocouples

A type K (Chromel-Alumel) sheathed thermocouple with a grounded bead was located in the top of each of the 52 instrumentation tubes positioned in a specific spiral pattern in the core.[171] Each instrumentation tube was located in the center of a fuel bundle and was permanently fastened into the bottom support plate for the core. Each also contained seven self-powered neutron detectors (SPND) spaced at about 1¾ -foot intervals vertically and located between neighboring grid spacers. The instrumentation was being used in an experimental study of power tilt and power shaping in the core and is not normally present. The incore thermocouples measured water temperatures exiting the bundles, and the SPNDs measured the neutron flux and flux profile in the bundles. The physical elevation of the incore thermocouples was in a flow mixing cup contained in the lower part of the upper end fitting of the bundles and was 12 inches above the top of the fuel in the fuel rods of the bundles. The data from both the thermocouples and the SPNDs could be requested from the plant computer via either the alarm printer or the utility typer at operator option. Both were connected to print out on the alarm printer when the set reading range limits, 700°F and 2×10^{-6} amps, had been exceeded. Data from selected SPNDs were also available on two multiple-point recorders located in the control room.

The incore thermocouples began going off scale (indicating temperatures above 700°F) during the later part of the time the alarm printer was unavailable between 5:15:16 and 6:48:08 a.m. At the time of the earliest record of alarming of the incore thermocouples, between 6:55 and 7:13 a.m. (2 hours 55 minutes and 3 hours 13 minutes accident time), 39 of the 52 incore thermocouples were recorded off scale, i.e., above 700°F. The records thereafter are incomplete because either some thermocouples were missed in an ordered sequence of recording, or only a partial listing was requested, or they simply were not requested by the operators from either the alarm printer or the utility typer for a considerable period of time. The data that are available have been reported elsewhere.[170,172,173] A set of measurements of temperature was made at the computer terminals in the cable spreading room by using a calibrated thermocouple reader instrument and manually recorded.[174] Temperatures as high as 2650°F were measured,[175] as shown in Figure II-28. The trend of the data on incore thermocouples indicating temperatures greater than 700°F either at the time of recording or both before and after the period show that 49 of the 52 thermocouples read above 700°F in the period between 3 hours 13 minutes and 3 hours 21 minutes, 33 between 3 hours 21 minutes and 3 hours 36 minutes, 44 between 3 hours 44 minutes and 3 hours 47 minutes, and 26 between 4 hours 34 minutes and 4 hours 47 minutes. The number above 700°F decreased thereafter in reasonable order, but 11 were still up scale at 00:43 a.m. the next day (March 29, 1979), 3 were still up scale at noon on March 29, 1979, and 1 was still up scale (greater than 700°F); 20 were above 300°F at 10:22 a.m. on April 1, 1979, more than 4 days after the start of the accident. No evidence available at this time can determine whether the temperatures indicated were measured at the thermocouple bead in the mixing cup of the upper end fitting or were those at newly formed junctions located in the "liquefied fuel" region of the core. Attempts to measure the resistances of the legs of the thermocouples could not resolve the question, nor could other types of measurement made to determine the continuity of the thermocouple wires.

Self-Powered Neutron Detectors

The self-powered neutron detectors (SPND) located in the 52 instrumentation tubes are experimental devices used in TMI-2 to measure flux, power tilt, and power shaping in the core. There are seven in each instrumentation tube, located about 1¾ feet apart vertically; each consists of a shielded emitter and collector head about 3 inches long that senses neutrons by a flow of electrons required to replace those emitted from the rhodium surface of the emitter after impingement by a neutron. The emitted electrons travel through an oxide insulator to a grounded sheath. This system can become a thermoionic converter when the temperature is raised to some elevated temperature, currently estimated to be above 1000°F, with the electrons being emitted by thermal excitation.[173] In addition, the current flow changes from negative to positive as the converter temperature is increased. This means that an alarm change from "BAD" to "NORM" can be due to either cooling or continued heating, and there is no way to distinguish between them from the alarm printer notation. The behavior of these systems in both thermal and radiation fields is the subject of an experimental study being con-

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A															
B							*31* 355	*30* 375							
C						*32* 545			*29* 1035	*28* 375			*52*		
D					*33* 1275					*27* 575				*51* 295	
E				*34* 1075			*7* 2055		*5* 2655		*26* 405				
F			*35* 165				*6* 2441	*4* 2453				*24* 405	*23* 625		
G		*36* 455			*9* 2352	*8* 1849			*3* 1930		*25* 1951		*22* 305		
H	*37* 335				*10* 2527			*1* 1370	*2* 2251				*21* 1927		
K					*11* 1886						*19* 705	*20* 1775			
L		*38* 445	*39* 1575			*12* 457					*18* 375		*50* 1855		
M			*40* 395				*13* 2253		*16* 2402	*17* 425				*49* 435	
N				*41* 485				*14* 673	*15* 2242						
O					*42* 425	*43* 535				*47* 1175		*48* 385			
P						*44* 375									
R							*45* 425			*46* 550					
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15

FIGURE II-28. Temperatures Measured by Incore Thermocouples on March 28, 1979, 8:00 a.m.-9:00 a.m., Using Fluke Meter at Computer Terminal Board ducted by EPRI-NSAC.[176] Both the results of the study and of the subsequent analysis of the SPND alarm and strip-chart data (signals from about 40 of the SPNDs were on two multipoint recorders, as well as on the alarm printer) will be reported by EPRI-NSAC later.[176] For the present, it seems adequate to consider only the first time an SPND is reported by the alarm printer as BAD, meaning that the SPND has seen a rise in temperature high enough to cause a flow of negative current of about 2000 nano-amps. It is estimated that this temperature must be well above 1000°F and approaching 2000°F.[172]

During the time between the start of the boildown in the core after 1 hour 40 minutes and the time the alarm information again became available at 2 hours 48 minutes, many SPNDs at levels 4, 5, 6, and 7 went off scale as they heated up, as many of them are shown by the alarm printer as being NORM and many others are shown as being BAD. Because the notation of NORM would not be shown on the alarm printer if the SPND had not been shown as BAD earlier, it can be concluded that these SPNDs had already been heated up significantly when first noted on the alarm printer. The strip-chart data[177] indicate that the level 6 SPNDs near the center of the core first started up scale at 2 hours 30 minutes, and those near the periphery started up scale at 2 hours 34 minutes. In the first 7 minutes after 2 hours 48 minutes, 40 level 4 alarms (both BAD and NORM without reference to specific strings) and 46 level 3 alarms were received. Also, one level 2 alarm (BAD) was received.

Alarms of NORM noted for SPNDs at levels 3–7 immediately after the RC-P2B was started at 2 hours 54 minutes may indicate that part of the core was cooled, but not quenched, and reheating began almost immediately.

In the following hour, many of the SPNDs oscillated between BAD and NORM alarm position, but it is impossible at this time to determine whether they were heating or cooling. However, between 3 hours 44 minutes and 47 minutes, the alarm BAD appeared for the first time for SPNDs at level 1 or 2 in 18 instrumentation tubes (strings). Alarms for three strings had appeared about 20 minutes earlier. Five strings were known to have been inoperable at these levels before the accident. Level 1 SPNDs are about 10 inches from the bottom of the fuel in the fuel rods, and level 2 SPNDs are located about 30 inches from the bottom of the fuel. Although this indication means that the SPNDs at these levels reached temperatures greater than about 1000°F, it does not mean that the water level in the reactor core had reached this level or below. This will be discussed in the interpretations in the following section.

In the later hours, many of the SPNDs at the levels of 3–7 flickered between BAD and NORM, the average alarm number first increasing gradually and then decreasing, particularly after the PORV block valve was last closed at 13 hours 24 minutes and the makeup flow was increased to repressurize the system. Over the next 4 days, SPNDs continued to return on scale, the last one having "quenched out" on April 1, 1979, as shown by one of the multipoint recorders.[177] The physical meaning is debatable, but the pattern is that the number of SPNDs indicating upscale decreased continuously after about 12 to 13 hours accident time. The coincidence with the decrease in the apparent size of the "hydrogen bubble" discussed in Section II.C.2.e is to be noted.

b. Interpretations of Accident Sequence

Introduction

The reconstruction of the sequence of events, interactions, and system behavior of the reactor primary system during the first 16 hours of the TMI-2 accident has been found to be a difficult task requiring many calculations, estimations, and deductions based on too little quantitative and recorded data. Despite the wealth of instrumentation and data available on normal operation, the amount of data important in the accident reconstruction recorded in either the data acquisition systems or on strip charts is appallingly small.

Much valuable information was lost when the alarm printer was inoperable at important times during the first 3 hours of the accident. A very large amount of useful data available from the plant computer through the utility typer (operator special summaries) was either never requested by the operators or requested only 6 to 7 hours after the accident began. No requests asked for previously acquired data. Some of the information important to the postaccident analysis was available to the operators during the accident on their panel as dial indications, but it was neither noted in the operators' log nor recorded permanently in any form. Other data that could be quite useful in reconstruction, in the absence of the losses discussed above, were never taken because no instrument existed, or the data were taken in such a form that reconstruction analysis is not possible.

It has been found to be impossible to establish with an acceptable accuracy even an approximate water inventory in the primary system as a function of time, to determine a heat balance across the OTSGs, or to know with certainty the position (open, closed, or throttled) as a function of time of several valves important in controlling the parameters of the reactor primary system. This, then, forces the reconstruction to be based on inferences, interpretations, arguments, and rationalizations with the use of too little quantitative, recorded data, thus precluding decisive and unequivocal selection of any one interpretation from the several that can be presented.

There appear to be as many interpretations of the events of the first 16 hours of the TMI-2 accident as there are groups examining the problem and attempting a reconstruction. The interpretation given below is based on three separate analyses of core damage: that of the Special Inquiry Group; a base case calculation by Battelle Columbus Laboratories using the MARCH code[178] to evaluate the "Alternative Scenarios" or "what ifs;" and a study conducted by Sandia Laboratories,[179] at the request of Task Group 2 to ensure that some of the less probable scenarios were not missed.

The agreement between these interpretations and those proposed by others[180,181,182] is, in general, much stronger and broader in the important aspects of the accident sequence than is the disagreement. For example, all estimate (1) that between about 50% and 70% of the core has been damaged, with 35% or more of the Zircaloy metal converted to oxide, (2) that temperatures in the neighborhood of 4000°F or higher were reached in the upper part of the core, that significant amounts of "liquefied fuel" were formed and no direct melting of UO_2 occurred (5200°F required), and (3) that about 750 ± 100 pounds of hydrogen were formed by the oxidation of the Zircaloy cladding. The areas of disagreement center primarily on whether the reactor core was covered by coolant after 2 hours 54 minutes and at subsequent times and on the number of periods the core was uncovered. Although these are important in the collation of the system data, they may not be too important in estimating and understanding the extent of damage to the core and the times at which it occurred, as well as the significance of what actually happened to meet our broader needs to understand reactor safety.

Critical Observations

There are several critical observations in the recorded data which must be considered in the reconstruction and interpretation of the accident scenario. Their causes and effects must either be described or the data shown to be in error or due to a false indication by an instrument. Those deemed most important at this time are listed in Table II-55 and can be examined in the curves of Color Plate III.

General Description of the Accident Conditions

Because of the failure of the pressurizer pilot-operated relief valve (PORV) to close again following the initial surge of pressure, reactor coolant was continuously leaked through the valve greatly in excess of the makeup rate for approximately 140 minutes until the block valve (another valve in the same line) was closed. Although the data indicate a high water level was maintained in the pressurizer, the quantity of liquid in the reactor primary system decreased throughout this period. While the reactor coolant pumps were in operation, a mixture of steam and liquid water was pumped through the core, and that flow effectively cooled it. However, when the last set of reactor coolant pumps was shut off at 1 hour 41 minutes, the liquid and steam phases separated, with the liquid phase apparently falling to the level of the top of the core. For the next half hour, some of the steam generated by decay heat in the core was released to the pressurizer and out the open valve, and the remaining steam condensed in the A steam generator. The water level on the primary side of the steam generator was not high enough, however, to permit the condensed water to flow back into the reactor vessel to resupply the core. For this reason, the water level in the core continued to drop to approximately 4 to 6 feet from the bottom of the core.

At 2 hours 18 minutes into the accident the block valve in the relief line was closed and that loss of water from the system was stopped, but the letdown flow continued. The water level in the core apparently began to rise slowly over the next half hour, at which time one of the reactor coolant pumps in the B loop was turned on for 19 minutes. During the first few minutes of pump operation, sufficient water was pumped to fill the annular downcomer region in the vessel and to force some additional water into the core. Although a few feet of core remained uncovered following operation of the reactor coolant pump, the greatest extent of core heatup probably preceded this event and the core was significantly quenched at this time.

Additional damage apparently occurred to the core at 3 hours 45 minutes, as indicated by several sets of system data. We believe that at this time there was slumping and densification of the debris bed produced earlier, with the formation of a steam bubble below a crust in the bed. The displacement

TABLE II-55. Critical observations

1. Source Range Monitor (SRM):
   a. The sharp changes in the count rates of the Source Range Monitor (SRM) at 1 h 40 min, 2 h 54 min, 3 h 18 min, and 3 h 42 min,
   b. the rises in count rate starting at 1 h 45 min and 2 h 54 min, and
   c. the increasing deviation above the "normal decay curve" after 4 h.
2. Pressurizer Level Indications:
   a. The rapid increase in pressurizer level indication at 2 h 30 min,
   b. the decrease beginning at 3 h 6 min followed by the rise starting at 3 h 27 min,
   c. the accelerating rate of decrease in level starting at about 11 h,
   d. the slow rise after 11 h 30 min,
   e. the decrease and subsequent increase between about 13 h 15 min and 14 h 20 min, and,
   f. the "full" reading observed for most of the time after 3 h 45 min.
3. Hot-Leg Temperatures:
   a. The indicated temperatures for the OTSG hot legs,
   b. the changes observed,
   c. the nearly parallel behavior of the two hot legs from 3 h 56 min to 10 h 6 min and the independent behavior thereafter, and
   d. the sudden change in the behavior of the A hot-leg temperature after 10 h 21 min and in the B hot leg at 12 h 3 min.
4. Cold-Leg Temperatures:
   a. The changes in cold-leg temperature behavior for the A legs at 3 h 45 min, 11 h 6 min, and 13 h 30 min,
   b. the separation in both time and magnitude of change for the two A cold legs (1A and 2A), and
   c. the lack of such changes in the B cold-leg temperatures.
5. Reactor System Pressures:
   a. The rapid changes in reactor system pressure starting at 2 h 51 min, 3 h 10 min, 3 h 18 min, 3 h 45 min, 4 h, and 14 h 36 min and
   b. the increases observed in reactor system pressure at 2 h 12 min, 3 h 45, min and 4 h 30 min at times when PORV block valve was open.
6. The behavior of the pressurizer temperature, particularly its apparent independence to changes in system pressure, valve opening and closing, and operation (or flows from) the makeup pumps.
7. The coincidences in time among the several observations.
8. The decrease in levels in the borated water storage tank (BWST).
9. The behavior of the incore thermocouples over 4 days.
10. The behavior of the self-powered neutron detectors (SPND)

of water below the debris bed by the steam allowed more Zircaloy cladding to heat up and oxidize, embrittling cladding to a greater depth and producing more hydrogen.

The high-pressure injection system was actuated for a few minutes at 3 hours 20 minutes into the accident, apparently recovering the core. High pressure injection was again actuated at 3 hours 56 minutes for a short time period. After this time, the core was probably never uncovered again, although some severely damaged regions of the core remained very hot and steam blanketed for approximately 4 days. The steam released from the hot regions was condensed in water in the upper plenum before reaching the hot legs.

At 4 hours 27 minutes, significant makeup flow to the primary system was established from makeup pumps B and C and maintained until 9 hours. The flow through the core during this time period was high enough that all of the decay heat in the core could be removed without boiling the water. After leaving the core, the heated water flowed through the pressurizer and out the relief valve to the reactor coolant drain tank and then to the containment building. In this time period, the upper portions of the two hot legs and steam generators were blocked to steam flow by hydrogen that had been produced earlier from reaction of steam with zirconium. Because the hot legs and steam generators were well insulated, the temperatures measured at the tops of the hot legs remained nearly constant for a number of hours at approximately 750°F, the temperature to which they had been heated during the period of core uncovering.

The attempts made to collapse the steam bubbles in the hot legs of the OTSGs failed, although the system was repressurized and there was more than 1½ hours of feed and bleed operation by cycling the PORV block valve open and closed because of operator failure to recognize that the pressure was not due only to steam but to noncondensible gases as well. The block valve was then opened for a long period of time to "blow the system down" to get to a pressure low enough to bring on the core flood tanks or the decay heat removal (DHR) system. The system pressure did not drop enough in more than 3 hours, but the depressurization did seem to bleed most of the rest of the hydrogen out of the system, at least to the point that the OTSGs were no longer completely blocked by hydrogen.

The hydrogen bled from the system out the PORV during the various depressurizations, accumulated in the containment to reach a concentration high enough to cause a "hydrogen burn" at 9 hours 54 minutes.

Around 13 hours into the accident, the decision was made to increase makeup flow significantly, close the PORV block valve, repressurize to collapse steam or gas bubbles in the hot legs, and attempt to start a reactor coolant pump. The condenser vacuum had been restored to the secondary system, and the permissives in the reactor coolant pump controls had been bypassed to allow them to be started.

At 15 hours 35 minutes, reactor coolant pump RC-PIA was jogged to check starting and operation. At 15 hours 50 minutes, it was turned on and the transient was terminated; that pump worked continuously thereafter for more than a week.

Interpretation of the Data

This accident interpretation is keyed as much as possible to the 10 time periods described in Section II.C.2.a of this report, "Data Analysis for the First Sixteen Hours," each part beginning and ending at the times of certain occurrences thought to be important in the progress of the accident.

At about 4:00 a.m. on March 28, 1979, the TMI-2 plant suffered a turbine trip that was apparently initiated by the tripping of a condensate pump a second or two earlier. About 8 seconds later, the reactor tripped because the system pressure had reached the 2355 psig setpoint. The PORV on the system pressurizer had opened at 2255 psig 5 seconds earlier, but this apparently did not provide enough relief to prevent further pressure rise. When the system pressure decreased after the trip to 2205 psig, the PORV failed to close as it should, and the reactor underwent a small loss-of-coolant accident that was not recognized as such by the reactor operators until more than 2 hours later.

Periods I and II: 0 Hours to 1 Hour 40 Minutes

There is much evidence to indicate that the reactor core was not damaged before the last of the reactor coolant pumps was turned off at 1 hour 40 minutes of accident time. The reactor coolant system had lost a major part of its water inventory out the open PORV, the steam generators had lost almost all of their heat removal capability by being "boiled dry," and the makeup flow was probably automatically throttled by the high pressurizer level indication. (There are no data to show that the pressurizer level control had been taken out of "automatic" control or that the "16" valves (high pressure injection valves) had been left open by the operators). The water inventory and distribution in the primary system near the end of Period II are shown in Figure II-29. In OTSG B, mixed water and steam was "dribbling" over the top of the hot leg and separating into a steam phase in the upper part of the primary side of the OTSG and a water phase in the lower half, which returned to the downcomer of the primary vessel by overflowing through the lower part of the inlet casing of the reactor coolant pumps and into the cold legs. That this normal direction of flow existed is proved by the fact that at 1 hour 30 minutes to 1 hour 35 minutes the temperature of the OTSG B hot leg was 5°F higher than the cold leg, as shown in Color Plate V. In OTSG A, the behavior is similar except that the water in the lower part was being pumped out by the reactor coolant pump to keep that level quite low. Because the OTSG A coolant pumps were running, a mixed phase of water and steam was being fed to the pressurizer by the A hot leg and was being vented out the open PORV into the reactor coolant drain tank (RCDT).

Period III: 1 Hour 40 Minutes to 2 Hours 20 Minutes

When the reactor coolant pumps RC-PIA and RC-P2A were shut down at 1 hour 40 minutes, the

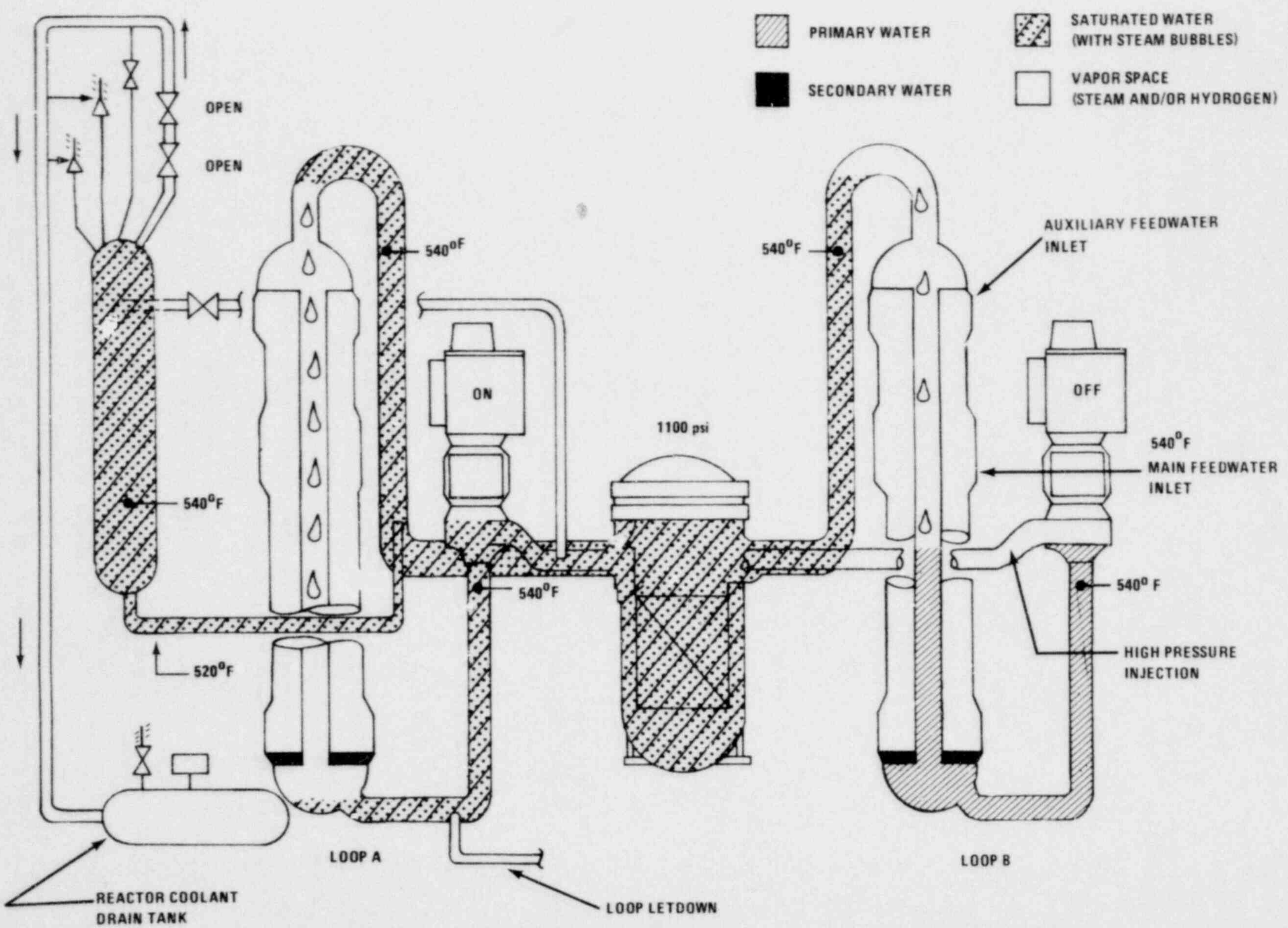


FIGURE II-29. Reactor Primary System at 90 Minutes two-phase mixture of water and steam that had been circulating separated into a steam phase in the upper parts of the reactor primary system and a water phase in the lower parts.

When the A pumps were last operating, any flow in OTSG B had to have been induced rather than forced, while that in OTSG A was driven by the pumps. When the pumps stopped, the fluid in the hot legs would have drained back into the top of the core. OTSG B would have been left about half full on the primary side, with water in the cold legs to the "dribble level" through the pump casings into the horizontal section of the cold legs and into the downcomer.

The pumped flow in the A loop filled the hot leg, fluid passed over the top of the candy cane, and then dropped and separated into a steam phase at the top and a water phase at the bottom of the OTSG. Because the coolant pumps were still operating and the letdown line was also removing water, the water level in the bottom part of the primary side of OTSG A remained quite low. When the pumps stopped, the fluid in the hot leg drained back into the top of the core, and the level in the OTSG A settled out at less than half full, and probably not more than about one-fourth full, as the major amount of water present was that in the cold legs.

The settling out of the water levels and the separation of the fluids into a steam phase and a water phase filled the downcomer with water. This then caused the abrupt and large decrease in SRM count rate observed at 1 hour 40 minutes. Because the SRM count rate started increasing immediately after the abrupt drop and the hot-leg temperature started to increase soon afterward, the water level in the core could not have been much above the top when it settled out.

The hot- and cold-leg temperatures of both OTSGs are plotted on an enlarged scale in Color Plate V for the time period around that of pump shutdown. The temperature for hot-leg B showed a definite increase in temperature by 1 hour 44 minutes, and a definite deviation from the continued de[illegible]ase of hot-leg A and both cold-leg temperatures at 1 hour 42 minutes. The A hot-leg temperature also showed a small but definite increase in temperature at 1 hour 43 minutes before it again decreased at the same rate as that of the cold legs in approximate concert with the refilling of OTSG A as shown by the A startup level (SU). We believe that this indicates that superheated steam could have been present in both hot legs at the location of the RTD near the top of the candy cane no later than 1 hour 42 or 43 minutes.

Because the mass of metal in the upper internals of the reactor above the core and the 50-foot length of the hot leg would absorb considerable heat, the top of the core would have had to have been uncovered immediately upon shutdown of the pumps. However, no conclusion can be drawn as to what level below the top of the core the water settled. There can be no question that superheated steam was in the A hot leg by 1 hour 52 minutes because the temperature of the A hot leg started a rise that did not stop (other than for two small reversals) until the temperature was greater than 800°F. The temperature of the B hot leg fell after its initial rise and did not begin a final rise to more than 800°F until 2 hours 3 minutes, even though OTSG B was not being refilled on the secondary side at this time. We conclude that the time the top of the core was first uncovered must have been between 1 hour 42 minutes and 1 hour 52 minutes. The increase in SRM count rate starting at 1 hour 42 minutes reinforces this conclusion.

The sharp drop in SRM count rate occurring at 1 hour 40 minutes is interpreted as indicating that a large increase in fluid density or level occurred at the time the reactor coolant pumps were shut down; i.e., the downcomer was filled with a higher density fluid than that which had been circulating. The following rise in count rate, first rapidly, then more slowly, and then leveling off, is believed to indicate that the boiloff of water in the core occurred over a period of 20 to 30 minutes and then leveled off at some position between the bottom and the midplane of the core. Calculations by Sandia Laboratories (Appendix II.10) in the TMI-2 SRM study indicate that the SRM count rate is quite sensitive to water level in the downcomer within ± 1 foot of the top of the core and relatively insensitive to changes in levels below that. Count rates for levels 2 and 6 feet below the top of the core differ only slightly, especially if the increase in boron content is included as the water boils off and the boron concentration increases. The slight rise in count rate after 2 hours is thought to have been due to the decreasing density of the water left in the vessel as the system pressure continued to decrease, and the slow drop in count rate after 2 hours 18 minutes is thought to be due to increasing fluid density in the core as the system pressure increased. Undoubtedly, there was some adjustment in level in the core due to makeup and letdown flows, but there are no data on which to base a judgment. This interpretation would indicate that the water level in the core reached a steady state level at about 2 hours, which balanced the rate of boiloff in the core with the refluxing of condensate from one or both OTSGs, the loss due to letdown flow and open PORV, and the increase due to makeup flow.

Once the reactor coolant stopped circulating and the water level in the core began decreasing, the portions of the fuel rods no longer covered by water began to heat up. The rate of temperature rise, the degree of oxidation, the formation of "liquefied fuel," and the oxidation damage done to the core are described after Period IV in the section entitled "Core Damage Before Three Hours."

The system pressure began to rise at 2 hours 10 minutes, about 8 minutes before the PORV block valve was closed by the operators after they had finally realized that the PORV had failed to close earlier. Such a pressure rise could have occurred under the circumstances either because of the core's heating up and producing a pressure increase at a rate greater than could be relieved by the open PORV or because of reduced heat removal by the A OTSG when the emergency feedwater spray at the top of the OTSG was stopped at 2 hours 12 minutes. The primary system depressurization rate was being controlled by the OTSG steam pressure. The increase in system pressure was relatively slow at first but then increased more rapidly.

Period IV: 2 Hours 18 Minutes to 2 Hours 54 Minutes

When the block valve to the open PORV was closed at 2 hours 18 minutes, the leak from the system was stopped and the system pressure continued to rise. In an attempt to return the system to its normal cooling mode, the operators attempted to start the reactor coolant pumps sequentially. Only RC-P2B could be started. The operators reported that the pump operated normally only for a very short time and started vibrating. It was finally shut down again at 3 hours 12 minutes.

It is during the period from the first uncovering of the core at 1 hour 42–52 minutes to the thermal shock produced by inflooding water at 2 hours 54 minutes that we believe that the major embrittling damage to the core occurred and much of the hydrogen was produced. The progress of the heatup is discussed in the following section "Core Damage Before Three Hours."

We believe that the condition of the core at 2 hours 54 minutes based on an estimated boildown to 4 feet from the bottom of the core to be as follows: all fuel rods had burst; the Zircaloy cladding in the fuel rods was embrittled to a depth of at least 6 feet from the top of the fuel stack, between 26% and 31% of the Zircaloy in the core had been converted to zirconium dioxide; "liquefied fuel" (UO_2 dissolved in either molten Zircaloy metal or the eutectic liquid formed between Zircaloy metal and its oxide)[183] had been formed to at least 36 inches from the top of the fuel in the fuel rods in the center of the core and to at least 40 inches in the periphery; the Inconel grid spacers had been melted to at least 4 feet from the top; a rubble bed had been formed by fragmented fuel rods on the spacer grids located at about 5¼ feet from the top of the fuel stack; a significant fraction of the fuel rods probably still maintained their original structural geometry above the 4-foot level from the top of the fuel stack, although part or all of the Zircaloy cladding had melted and flowed away, and the UO_2 fuel pellets, for the most part, remained in the original rod geometry; and the control rod guide tubes and instrumentation tubes remained in place and intact, although oxidized to a greater or lesser extent. The notation of the BAD indication on the alarm printer for the SPNDs at the 4 to 7 levels (from midplane to the top of the core) at this time are consistent with this interpretation, as are the alarm printer indications that many of the incore thermocouple temperatures were off scale (above 700°F).

Instrumentation tubes and control rod guide tubes survived longer than the neighboring fuel rods because they were not significant heat sources and because they served as "percolator tubes" during depressurization, in which steam bubbles, formed in the annuli, caused liquid water to percolate above the average level in the core to reach higher temperature regions before evaporating. The net effect was to produce a much higher mass flow of steam, as well as velocity of steam flow, through the annuli between the guide tubes and the control rods (and in the double annuli of the instrumentation tubes) than occurred in the subchannels between neighboring fuel rods. Thus, the guide tubes, control rods, and instrumentation tubes stayed much cooler than otherwise expected during depressurization and, consequently, lagged significantly in temperature rise compared with their surroundings. Their heatup started later, and the heat absorbed by them was transferred by radiation from neighboring fuel rods and by conduction-convection interaction with the steam in the fuel subchannels.

At the end of Period IV, not less than 154 pound moles of the hydrogen gas (308 pounds) was produced during oxidation of the Zircaloy cladding at temperatures less than about 3600°F. A significant amount of hydrogen (probably 100 to 200 pound moles) was produced later by continued oxidation of the zirconium contained in the Zr-ZrO_2 eutectic and "liquefied fuel" formed, but no accurate estimate can be made for two reasons: (1) the actual extent and condition of the "liquefied" material is not known, and (2) there are no data on the oxidation kinetics of such material.

Core Damage Before 3 Hours

On Friday, March 30, 1979, shortly after the "hydrogen burn" was accepted as a real occurrence in the reactor containment building just before 2:00 p.m. on Wednesday, March 28, calculations indicated that the amount of hydrogen present in the containment at the time of the burn and left in the primary system as either a hydrogen gas bubble or as dissolved hydrogen in the reactor coolant was equivalent to 35% to 40% of the Zircaloy present in the core having been converted to zirconium dioxide. This was the first measure of damage to the core, and it applied to the amount of damage to the core at the time of the burn.

Later, a simple set of calculations of the heatup of the fuel rods was made[184] to produce bounding estimates of core damage using simplified assumptions, constant specific heats, constant rate of boiloff, a constant heat loss fraction, and manual and graphical solutions. This estimate gave a total of 25% to 30% of the Zircaloy cladding (fueled length only) converted to zirconium oxide at 3 hours, and estimated the depth of damage to reach as much as 6 feet from the top in the central region of the core. In the worst case estimate, a large part of the cladding in the top half of the 12-foot core reacted with the zirconium oxide to form a liquid eutectic phase at 3455°F. This flowed into the gap between the fuel and the cladding to react with the UO_2 fuel, partially dissolving it, and formed a liquid phase of Zr-U-O termed liquefied fuel.[183] At most, about 10% of the fuel present in the upper half of the core was thought to have formed liquefied fuel. In the least damage case (decay heat only, no heat of oxidation of the Zircaloy added for heatup), it was estimated that the depth of embrittlement of the Zircaloy cladding was essentially unchanged from the worst case, but the extent of formation of liquefied fuel was confined to only the top of the highest power central fuel assembly. No attempt was made to continue the calculations beyond 3 hours of accident time because of a lack of sufficiently accurate information beyond that time. As the damage estimate of 25% to 30% conversion was made at 3 hours, there is no significant disagreement with the previous estimate of 35% to 40% at 9.9 hours.

An engineering code called TMIBOIL[185] was recently written to calculate more precisely the time-temperature relationship for the fuel rods by using relatively precise analytical expressions, few simplifying assumptions, and parametric treatment of several of the system variables. The code was used to estimate the sensitivity of the answers to variations of such parametric variables as depth of boildown, time of boildown to a given depth, convective heat transfer coefficients in steam at low flow rates, and radial peaking factors in the TMI-2 core (related to power in the bundle from center to periphery). The details of the calculations are presented in Appendix II. 8. The principal variation on the amount and extent of damage results from parametric variation in the depth of boildown, all of the other parameters affecting primarily the time at which a given temperature was reached but not the magnitude of the temperature. Time-temperature elevation plots for 1-foot increments on the fuel rod are shown in Figures II-30 and II-31 for the center bundle for a boildown in 20 minutes and a boildown in 33 minutes to 8 feet from the top (4 feet from the bottom) of the core. A summary of the results of the calculations is presented in Appendix Tables II-8 and II-9.

Boildown to 5 feet from the bottom would produce much too little damage, according to our analysis, and boildown to 3 feet from the bottom much too much. We then conclude that the boildown was probably to $4 \pm 1/2$ feet from the bottom of the core. The damage estimates at this level are believed to be consistent with the estimated amount of hydrogen formed, the amount of fission products released, and the data from the incore thermocouples and from the SPNDs.

Data indicate that the first detection of significant levels of radioactivity in the primary loop occurred at 6:25 a.m. on March 28, 1979,[186] which would be consistent with a time of boildown of 33 minutes to 8 feet, and a time of core uncovering of about 1 hour 52 minutes into the accident. A review of the calculations indicate that the major conclusions reached for a time of boildown of 20 minutes can be applied to that for 33 minutes if appropriate corrections are made for the slower rate of uncovering. Thus, the rods would have burst about 30 to 40 minutes after the top of the core was uncovered. The type and extent of damage to the core would be essentially unchanged, since in both scenarios the peak temperatures and greatest depth of damage had been produced before the reactor coolant pump was turned on at 2 hours 54 minutes into the accident.

The principal results of the calculations show that the ranges of time and location of rod burst (1500 ± 100°F) are from about 13 to 25 inches and 20 to about 40 minutes from center to periphery for all ranges of boildown and time to boildown depth.

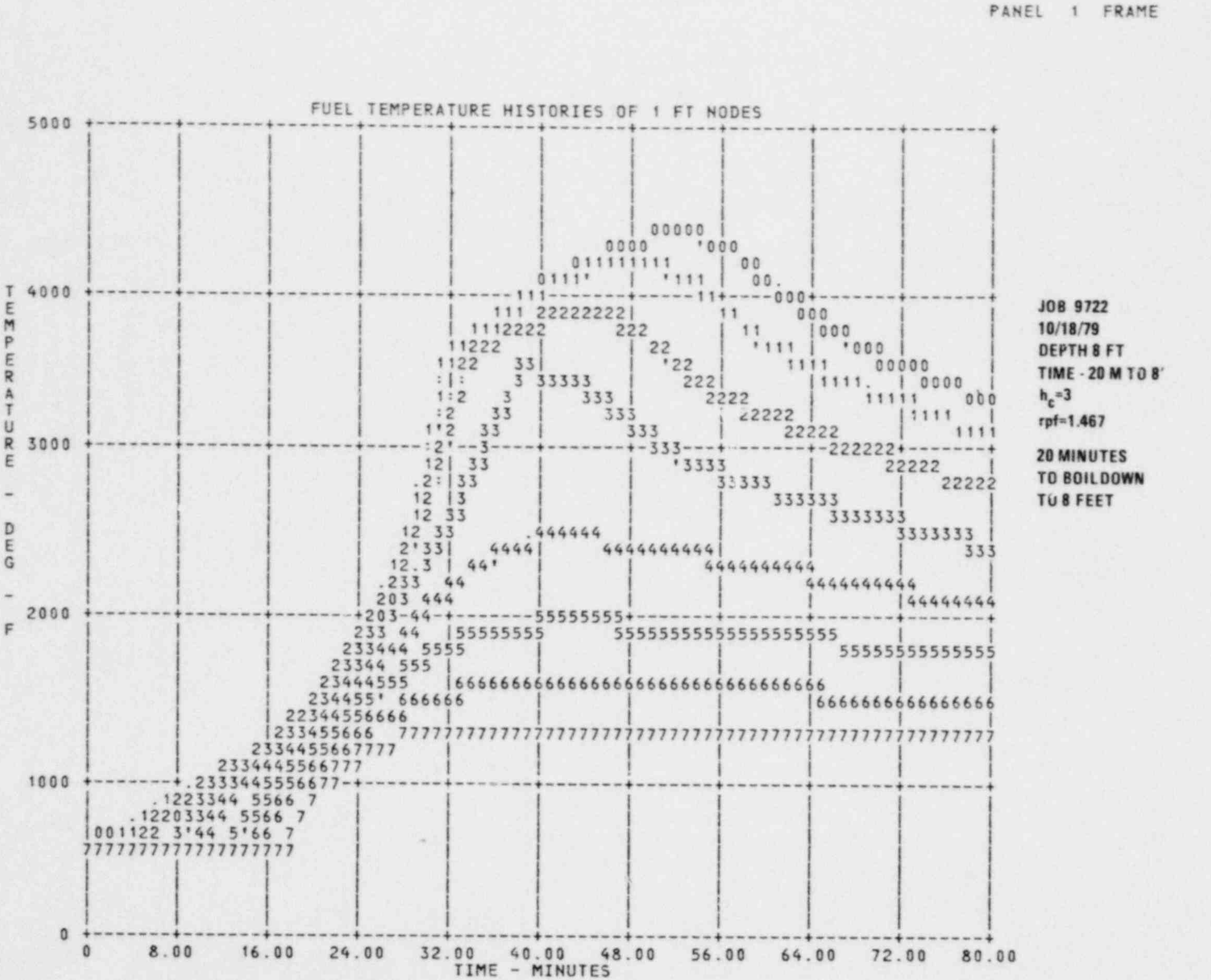
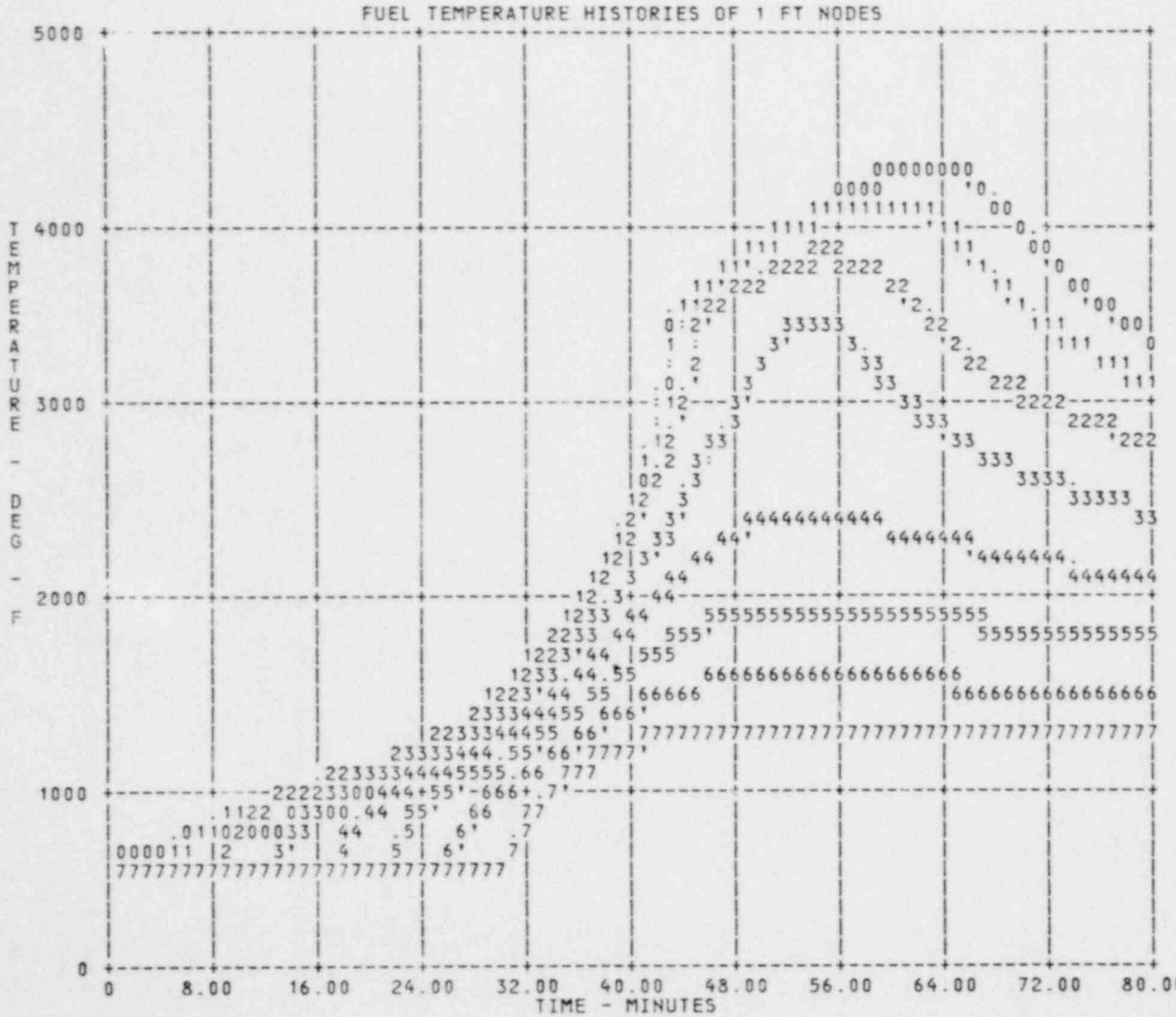


FIGURE II-30. Fuel Temperature Histories



JOB 170 10/24/79
WITHOUT COLD ROD
DEPTH - 8 FT
TIME - 33 MIN to
h_c=3
rpf=1.467

33 MINUTES
TO BOILDOWN
TO 8 FEET

FIGURE II-31. Fuel Temperature Histories

However, the maximum depths for formation of liquefied fuel and the peak temperatures reached vary quite considerably, with boildown to 7 feet producing only a small amount of liquefied fuel in the peripheral bundle while boildown to 9 feet not only producing liquified fuel down to the midplane of the core but also calculated temperatures in excess of the melting point of UO_2 for several feet of length of fuel rod. In addition, the calculated temperatures were still increasing when the calculations were stopped at the time for the reactor coolant pump RC-P2B to be turned on.

The estimate of damage present in the core at 3 hours depends on the time assumed for the first uncovering of the core. The best evidence available for determining this time is shown in Color Plate V where the temperatures of the hot and cold legs of the two OTSGs and the levels of coolant on the secondary side are plotted as functions of time.

There are two possible interpretations of these data. When the prior level in OTSG B is considered (shown in Color Plate V), it can be argued that the first break in the curves for the hot-leg temperatures of both steam generators at 5:42 a.m. (1 hour 41 minutes of accident time) indicates that superheated steam was detected in both A and B steam generators at the top of the hot-legs. The continued rise and subsequent decrease in temperature for OTSG B could indicate flow of superheated steam into a condenser that was saturating in heat. The reversion of OTSG A hot leg temperature to a decreasing temperature–time relationship, paralleling the previous curves and the succeeding curves for the cold legs, could indicate that OTSG A could absorb no significant amount of heat (it was already known to have been boiled dry) until its refilling had begun. Thus, it can be argued that the core was first uncovered at 102 minutes. It can be stated with certainty that the core had been uncovered no later than 5:52 a.m. (1 hour 52 minutes or 112 minutes of accident time) because at that time the OTSG A hot leg temperature began to rise without stopping (other than for two short inversions) until a temperature of about 820°F was reached at 6:52 a.m. (2 hours 52 minutes or 172 minutes accident time). These two times, 102 and 112 minutes of accident time, allow placement of the TMIBOIL zero time and the time at which the RC-P2B pump was started on the time temperature-elevation plots, so bounds for the amount of damage to the core at 3 hours can be estimated. It must be assumed that at least a small amount of water was pumped by RC-P2B into the core to reverse the heatup of the fuel rods, even if for only a few minutes.

If it is then assumed that the TMIBOIL calculations for boiloff to 8 feet in 33 minutes apply (the best estimate based on such information as the amount of hydrogen and radioactivity released, the SRM data, and the first detection of radioactivity in the primary loop), the PORV block valve was closed at 6:20 a.m. (2 hours 20 minutes accident time), and the RC-P2B was started at 6:54 a.m. (2 hours 54 minutes accident time), then the amount of core damage at 7:00 a.m. (3 hours accident time) can be bounded.

With these assumptions, it can be estimated that (1) the great majority of the fuel rods burst at about the time the block valve was closed at 140 minutes and all of the rods were burst within the next 10 minutes, (2) first liquefied fuel formation occurred about 10 minutes after the block valve was closed, (3) the maximum depth of formation of liquefied fuel in the hot assembly occurred about 20 minutes after the block valve was closed and about 10 minutes later in the lowest power assembly, and (4) the maximum temperature reached in the fuel rods was circa 4400°F for a middle power assembly at about 30 minutes after the block valve was closed and at about the time the RC-P2B was started. Additionally, peak temperatures of about 4300°F or more were reached in portions of more than two-thirds of the core by the time the RC-P2B was started. The maximum penetration of the formation of liquefied fuel was to about 40 inches in the lowest powered assemblies on the periphery of the core and to about 35 inches in the center of the core. (The steam production rates decreased greatly as the periphery of the core was approached, and thus the cooling capability of the steam flow. This may be an artifact of the code because crossflow of steam was not allowed.)

The Zircaloy cladding was embrittled by oxidation down to at least 4 feet from the top of the fuel in the fuel rods. Considerable amounts of liquefied fuel had formed and flowed down between remaining oxidized cladding shells to freeze on reaching a lower temperature at a lower level. When the reactor coolant pump was turned on at 2 hours 54 minutes, the embrittled cladding would have been thermally shocked by the influx of coolant (whether steam or water) and would have shattered to produce a rubble or debris bed of cladding fragments, Zircaloy oxide shells, fuel pellets, and liquefied fuel supported by fuel rod stubs, unmelted grid spacers, and intact guide and instrumentation tubes. A significant part of the debris bed would be melded or glued together with liquefied fuel that had frozen after flowing from a higher position and temperature.

Additionally, it is estimated that the amount of Zircaloy converted to oxide as a result of the events to 7:00 a.m. (3 hours accident time) is between 32% and 39% of the Zircaloy in the fueled part of the core, and between 26% and 31% of the total Zircaloy in the core, including plenum regions and end plugs. This estimate includes complete oxidation of the Zircaloy contained in the liquefied fuel. These amounts are equivalent to 300 and 360 pound moles of hydrogen, respectively. Because of the evidence that more hydrogen may have been produced at a later time, this is not to be taken as an estimate of the amount of hydrogen present in the containment and the primary system at 1:34 p.m. (9.9 hours accident time), the time of the hydrogen burn in the containment.

Period V: 2 Hours 54 Minutes to 3 Hours 12 Minutes

The operation of RC-P2B at 2 hours 54 minutes produced a sudden influx of water from OTSG B into the overheated core, causing a great burst of steam to be produced to increase the pressure very rapidly (at approximately 30 psi per second), as shown in Color Plate III. Such a rate of pressure increase could have been produced only by a very large input of energy to the vapor phase of the system and must have been the result of water from the OTSG B raising the level in the core high enough for the water to encounter the overheated parts of the fuel rods. Since the fuel rod cladding was seriously embrittled by the oxidation it received in the preceding time, and much of it had been converted into zirconium dioxide, much of it would have shattered by thermal shock when the water from OTSG B was forced into the core by RC-P2B. There is no evidence to indicate that the core remained covered following the influx of water.

To the contrary, the continued presence of superheated steam in the hot legs supports the argument that the core was not covered. In addition, the data in the reactimeter at this time show flow for only two successive readings taken 3 seconds apart, indicating that flow occurred for at least 3 seconds and less than 9 seconds. As the flowmeter is located in the top portion of the hot-leg candy cane, the flow measured was only gaseous and was probably caused by the displacement of water in the lower part of the OTSG by the pump operation. No actual measure of water flow can be given, but we believe it to be about 1000 to 1100 ft^3. The abrupt changes in cold-leg temperatures may indicate that at least part of the water sucked from the OTSG B by the operation of the 2B pump may have entered the OTSG A through either or both of the A cold legs.

In the 18 minutes of this period, 39 of the 52 incore thermocouples were reported off scale by the alarm printer; i.e., having temperatures greater than 700°F. Five more recorded temperatures between 650 and 700°F, and 8 registered between 500 and 650°F.

The alarm printer showed most of the SPNDs at levels 4 to 7 as BAD or NORM, and alarms for many at level 3 were alarmed, indicating that they were at temperatures well above 1000°F.[180] The combination of the data for the SPNDs and the incore thermocouples constitute measurements showing that the core at the midplane (or a little below) was at temperatures well above 1000°F for the entire period and could not have been filled with water, although the RC-P2B was running, in concurrence with the conclusion in the discussion above. (The debris bed would not have been filled with water even though there was water above the core, since the bed was too hot.)

The PORV block valve was closed, and the level indication in the pressurizer began to rise from 290 inches a few minutes before the start of this period to reach a peak value of 380 inches, a total rise of about 90 inches in about 12 minutes. Because each inch of level in the pressurizer is equivalent to 3.2 ft^3 of volume, this rise amounts to about 290 ft^3 of water. As the hot leg of OTSG A contained only steam plus hydrogen at this time, either the level indication is wrong, some other source must be found for the water required, or some other mechanism must be found for the change indicated. In addition, the pressurizer spray valve was opened at 2 hours 54 minutes 33 seconds, providing an open line between the top of the pressurizer and the 2A cold leg of OTSG A. With the pressure in the system rising, the height of the water leg in the pressurizer spray line must be less than that in the pressurizer plus its surge line (the lowest elevation of the surge line between the OTSG A hot leg and the pressurizer is lower by about 7 feet than that for the spray line at the outlet side of the RC-P2A), and hydrogen plus steam can be entering both the spray line and the surge line into the pressurizer to increase the pressure in the top of the pressurizer.

The pressure and material balances are quite difficult to estimate. It is difficult to see how water can remain suspended in the pressurizer when gas is bubbling through the surge line and the spray line into the pressurizer to increase the pressure in the top of the pressurizer. Also, the temperature of the A hot leg dropped 50°F in the first 6 to 8 minutes of the period, at the same time the pressurizer level indication rose about 80 inches. The drop in temperature would be expected if the pressurizer were voiding into the A hot leg, but this would require a decrease in pressurizer level, not the increase indicated. This apparent contradiction in behavior can be explained if the pressurizer reference leg is somehow being voided at the same time as the pressurizer. A decrease in reference leg level produces the same signal indication as an increase in pressurizer level, because the instrumentation measures only a differential pressure between the reference and reading legs of the level indicator. An alternate explanation is that because the water temperature in the pressurizer was greatly supercooled relative to the steam temperature in the hot leg, a large amount of steam (about 2200 pounds required) condensed to increase the amount of water in the pressurizer and surge line. This does not explain how the water in the pressurizer can remain suspended in the pressurizer when there is an open gas line connection from the top of the pressurizer to the voided 2A cold leg to equalize the pressures throughout the system. The level began to fall at 3 hours 6 minutes when the block valve was opened. Opening the spray valve did not affect the rate of rise of the level in the pressurizer. We are left with different explanations, none wholly satisfactory for this period of the accident.

The opening of the PORV block valve a few seconds before the end of this period allowed the system pressure to drop very rapidly (in seconds to 200 psi) and then more slowly to 1900 psi at the end of the period.

Period VI: 3 Hours 12 Minutes to 5 Hours 18 Minutes

At 3 hours 20 minutes and at 3 hours 56 minutes, the high pressure injection system was turned on for a few minutes and then reduced in flow rate. The rapid pressure drop from 2000 to 1500 psi, which occurred at 3 hours 20 minutes, with the block valve closed is apparently the result of steam in the system flowing through the core barrel check valves and condensing on the water in the downcomer. As shown by the source range monitor, the downcomer was rapidly refilled following the initial actuation of the high pressure injection system and probably remained filled for the rest of the accident. Refilling of the pressurizer after 3 hours 30 minutes is probably an indication that the water level had increased to the surge line at this time. Some severely damaged regions of the core remained very hot for several days following the initial transient as shown by thermocouple reading above the core. Although superheated jets of steam from the damaged regions would be expected to penetrate into the upper plenum, condensation and mixing would occur before the fluid reached the hot legs.

Between 3 hours 42 minutes and 3 hours 46 minutes, something happened to the core, drastically changing the configuration and the state. The SRM signal showed a very sharp rise by a factor of about 2 in count rate; the system pressure rose more than 100 psi in a few seconds (a total rise of about 210 psi in about 7 to 8 minutes); both cold legs of the OTSG A rose very rapidly (130°F in 1 minute in 1A, and 200°F in about 2 minutes in cold leg 2A) as did the cold leg of OTSG B; both hot legs showed definite changes in temperature; and the rate of pressure drop in the secondary side of OTSG A decreased significantly. The pressure increase occurred even though the PORV block valve was open. The increase in the 1A cold-leg temperature started approximately 30 seconds before the pressurizer spray valve opening was recorded by the reactimeter (both are recorded on the reactimeter making the time difference precise to 3 seconds), so the event does not seem to have been precipitated by the opening of the spray valve. It is believed that a condensation or slumping of core geometry occurred just before the pressurizer spray valve was opened, causing the formation of more liquefied fuel in the reheating debris bed, which then dropped into the water in the lower part of the core. We believe that this produced a burst of steam that not only began the pressurization of the system but that may have either flowed through the core barrel vent valves into the A cold legs to condense in the cold water in the partially filled cold legs at a level below the RTDs where the cold-leg temperatures are measured or water in the downcomer was forced into the cold legs and the OTSGs by the expanding steam bubble below the debris bed in the core. The opening of the pressurizer spray valve provided a path for steam and hydrogen flow through the 2A cold leg to the top of the pressurizer, increasing the flow rate of fluid into that cold leg and increasing the temperature rise in it. In addition, the spray valve opening, combined with the opening of the PORV block valve, removed the pressure differential suspending the water in the pressurizer between the OTSG hot-leg pressure and the external pressure.

In the first 7 minutes of this period, 49 of the 52 incore thermocouples were recorded on the alarm printer as being above 700°F, and the remaining 3 as being between 650 and 700°F. In the following

15 minutes (3 hours 21 minutes to 3 hours 36 minutes), 33 were and 2 probably were above 700°F, 9 were between 650 and 700°F, and 1 was between 600 and 650°F. Also, in this period, the temperatures of 51 of the 52 thermocouples were manually measured over a period of 1 to 1½ hours with a direct reading thermocouple instrument by instrument technicians working at the computer terminal board in the cable spreading room. They measured temperatures as high as 2655°F (assuming 75°F cold junction correction); 18 thermocouples showed temperatures greater than 1500°F. As the sequence of measurements was made starting at about 4 hours and ending after about 5 hours into the accident apparently in progression from the center bundle and out the spiral in succession of string number, the temperatures recorded are not representative of the core at any particular time within the period and were influenced by the progression of changes during the time period.

The alarming of 18 SPNDs at levels 1 and 2 between 3 hours 44 minutes and 3 hours 47 minutes can only mean that temperatures greater than 1000°F were reached in the fuel rods at many places at elevations of 10 to 30 inches above the bottom of the fuel.

From the above evidence, we believe that the debris bed and shattered core produced in the prior period were further consolidated by additional formation of liquefied fuel to form a crust in the bed, which spread over much of the core. This crust effectively sealed the debris bed off from cooling by steam percolation, and a steam bubble was formed below the debris bed. This allowed additional oxidation of the fuel rod stubs by dryout, producing damage to a greater depth in the core. The debris bed and crust were penetrated by the increasing steam pressure from below, and much of the bed and crust suddenly slumped to lower levels in the core, part of it dropping or dripping into the water in the lower part of the core. The sudden local pressure generation may have forced a radial displacement of part of the core material into the region between the core and the downcomer. Also, in some assemblies, liquefied fuel flowed down the small channels surrounding the instrumentation tube to reach levels as low as 10 inches from the bottom of the fuel in the rods, or lower, producing the activation of levels 1 and 2 SPNDs observed and forming a hot casing around it below the average water level in the core. This then would have formed a steam jet through the annuli of the instrumentation tube to keep the upper part of the tubes cooled enough to survive until the overall system cooled below about 2600°F. This consolidation and slumping of the core can explain the abrupt change in the SRM data by the dropping of the "source," the increase in incore thermocouple readings, the activation of the level 1 and 2 SPNDs, and the very rapid increase in pressure observed.

As one result of the additional oxidation, core slumping, and formation of liquefied fuel, more hydrogen was formed by reaction between Zircaloy and steam. Additional sources of hydrogen would be oxidation of the stainless steel upper end fittings on each of the fuel assemblies and oxidation of some of the UO_2 to a higher oxidation state. However, bounding calculations (see Section II.C.2.d) indicate that the maximum contributions by these sources could not be more than a few tens of pounds of hydrogen, which is not an important contribution in comparison to the probable 100 to 200 pound error range in estimates for the production of hydrogen from the steam-Zircaloy reaction.

We believe the condition of the core at this time to be roughly as follows: the debris bed plus crust has been lowered in the core so that its lower boundary may be as low as 4½ to 5 feet from the bottom of the fuel in the fuel rods, and its upper boundary may be as low as 3 feet from the top of the fuel stack in the original fuel rods; its density has been increased toward 90% of full density; it rests on fuel rod stubs that may be no more than 5 to 6 feet long; and many assembly sections contain drips of frozen liquefied fuel reaching as far down as 10 inches from the bottom of the fuel. This would indicate that at least 50% and perhaps somewhat more of the Zircaloy in the core has reacted or been embrittled.

The pressure rise that began at about 3 hours 44-45 minutes was stopped and reversed when high pressure injection (HPI) was started at about 3 hours 56 minutes. When the HPI was stopped, the pressure again began to rise, as did the hot-leg temperatures of both OTSGs. This pressure rise was, in turn, stopped by the starting of MU-PIB and MU-PIC at about 4 hours 22 minutes. Because the system pressure again began to rise, within a few minutes of pump startup, it seems likely that the flow from the pumps was throttled.

After 4 hours 30 minutes the makeup flow to the vessel was enough to allow the removal of the entire decay heat from the core without boiling. The injected water flowed into the cold legs through the core, through the hot leg in the A loop to the pressurizer surge line, and out the open PORV. The temperatures measured in the surge line and in the pressurizer in this time period show that this water was subcooled. We believe that the injection rate from the borated water storage tank of 640 gallons per minute reported in NUREG-0600[186] is probably higher than was typical for this time period because the average includes two periods of high injection rate. However, at 5 hours 45 minutes, for a decay heat level of 6.2×10^7 British thermal units per hour, a flow rate of 640 gallons per minute and an inlet temperature of 110°F, the core outlet temperature would be 300°F. The temperature measured at the pressurizer surge line at this time is 310°F.

The secondary side steam pressure of OTSG A increased concurrent with the increase in reactor coolant system pressure after 4 hours 30 minutes, indicating that additional heat had been removed from the primary system through OTSG A. The heat removal capability of OTSG A continued, and the system pressure began to drop at 5 hours. As the pressure dropped about 180 psi in about 18 minutes, it seems likely that the generation of heat by oxidation of zirconium either still in fuel rod geometry or in the liquefied fuel and debris bed geometry had decreased to a negligible rate, and with that the production of hydrogen had stopped.

Although it is quite difficult to estimate with assurance the additional damage to the core produced by this event, it seems certain that some significant amount of damage did occur. If it is assumed that another foot of fuel rod was oxidized as a result of the event, then an estimated 50 pound moles (100 pounds) of hydrogen would have been formed in the next few hours. This would then yield a total production of hydrogen of about 410 pound moles (820 pounds). The hydrogen production estimated would then range from about 354 to about 410 pound moles (700 and 820 pounds), which indicates that 30% to 35% of the total Zircaloy in the core has been converted to zirconium oxide.

The period ends with the closing of the PORV block valve to bring about repressurization to "collapse the steam bubbles" in the primary system, which is believed by the operators to be necessary to allow natural circulation to cool the primary system.

Period VII: 5 Hours 18 Minutes to 7 Hours 39 Minutes

When the PORV block valve was closed at 5 hours 18 minutes, the pressure began to rise immediately at a rate of about 13 to 14 pounds per square inch per minute. It increased to about 2100 psi, at which time the operators began to cycle the PORV block valve open and closed to maintain the pressure between about 2150 and 1975 psi. The pressurization times were about 120 to 130 seconds long and the depressurization times about 70 to 75 seconds. The temperature of the pressurizer was reported in this time period for the first time in the accident sequence on the utility typer as 345°F, and the pressurizer surge line temperature was reported just before the start of this period as 310°F. The pressurizer temperature may have been 20 to 30°F higher at that time. At this temperature, the vapor pressure of steam in the vapor space of the pressurizer would have been no higher than 125 to 130 psia.

As in the previous time period, the core decay heat was removed by the makeup flow passing through the core and out the pressurizer. With the water in the system subcooled, the primary system pressure in this period was determined by the compression of the noncondensible gases trapped in the upper regions of the hot legs and steam generators. Assuming a net makeup flow rate of 565 gallons per minute (based on NUREG-0600)[186] and a perfect gas, a gas volume of 2540 ft^3 can be inferred from the system pressurization rate during the periods of pressure increase. A possible breakdown of this gas volume could have been: the reactor coolant pump volume (400 ft^3), half the volume of the cold legs (476 ft^3), half the volume of the hot legs (469 ft^3), half the volume of the upper head (254 ft^3) and 500 ft^3 in each steam generator. Although it is difficult to reconstruct accurately the distribution of gas among the different volumes in the primary system, the gas volume inferred from the pressurization rate is reasonable for this time period.

Earlier in the accident when the core was uncovered, some of the hydrogen generated from zirconium–water reaction flowed into the hot legs and upper portions of the steam generators. The presence of the hydrogen in the legs effectively blocked the flow of steam from the core to the steam generators. Because the primary system is well insulated (the characteristic thermal decay period for the walls is approximately 150 hours), the hot legs that had been heated to 750 to 800°F during core uncovery remained hot for a number of hours. Even the flow of subcooled water through the A loop hot leg into the pressurizer surge line was ineffective in cooling the upper portion of the hot leg. The thermal conductance along the pipe is too small to have reduced the wall temperature significantly. Furthermore, the hydraulic regime of hot fluid above cold fluid is thermally stable and would not have induced convective cooling.

During this and the previous period, the decrease in level in the borated water storage tank (BWST) indicated that at least 132 000 gallons of borated water had been pumped into the reactor primary system if all of the water removed from the BWST went into the primary system. As the primary system has a water volume of only 90 000 gallons, this amounts to about 1.47 times the total volume of the primary system, without allowing for the water volume present (about 45 000 gallons) at the start of the period.

Period VIII: 7 Hours 39 Minutes to 10 Hours 21 Minutes

With the decision of the operators to "blow the system down" to allow the core flood tanks to flood the core (since it proved impossible to "collapse the steam bubbles" in the hot legs), the PORV block valve was opened, and when the depressurization slowed, the pressurizer vent valve was opened. For a reason we have not determined, the pressurizer spray valve was also opened by the operators, although there was no water flow available to produce a spray in the top of the pressurizer (normal operating procedure would call for pressurizer spray actuation to decrease system pressure).

At 9 hours 4 minutes the makeup flow rate was decreased, and by 10 hours 20 minutes the water temperature in the pressurizer reached saturation. Based upon a decay heat level of 5×10^7 British thermal units per hour, the net makeup flow (including the discharge of flooding tanks) must have been less than 270 gallons per minute to result in saturated conditions at the core outlet. This is consistent with the operation of one makeup pump in this time period. The filtering of hydrogen from the primary system by the water in the pressurizer and surge line continued during this period, the hydrogen content of the system finally reaching a level sufficiently low for the OTSG A to begin to operate as though the vapor space was no longer blocked by a noncondensible gas.

Of particular note is the approximately 45-minute period between about 9 hours and 9 hours 45 minutes in which the system pressure did not change significantly whether the PORV block valve was opened or closed. This behavior did not stop until both MU-PIA and PIC were actuated in HPI at 9.9 hours by the engineered safeguards actuation as a result of the increase in containment pressure after the hydrogen burn discussed below.

The bleeding of hydrogen from the reactor primary system into the containment atmosphere through the PORV block valve in the previous period and the depressurization in this period resulted in a concentration of hydrogen in the containment atmosphere high enough to permit a hydrogen burn to occur. This hydrogen burn, discussed in Section II.C.2.a, caused actuation of the containment building sprays as well as isolation of the containment. The sprays produced a fog that cooled the hot legs between 50 and 60°F, and the cold legs between 25 and 30°F in a period of 6 to 7 minutes.

Neither of the "blips" in the SRM count rate during this and the previous period can be explained.

Period IX: 10 Hours 21 Minutes to 13 Hours 15 Minutes

At the start of this period, three important observations indicate that most of the hydrogen generated earlier by oxidation of the Zircaloy cladding had been removed from the system: the A hot-leg temperature dropped 150°F in about 9 minutes, the pressurizer temperature reached the system saturation temperature calculated from the system pressure, and the OTSG A showed a sharp, although small, rise in steam pressure. All of these indicate that steam was once again flowing through the A loop in significant quantities to be condensed in the OTSG A.

From 11 hours to 11 hours 20 minutes approximately 640 ft^3 of water appears to have drained from the pressurizer. If the pressurizer level reading is correct, a consistent hydraulic picture must be able to explain where this large quantity of water went. A plausible explanation is that the A loop cold legs and pumps also contained hydrogen following core uncovering and the water filled this volume. This may also explain the reason the RC-P1A pump could not be operated at 4 hours 10 minutes.

The reactor primary system pressure increased relatively slowly after the PORV block valve was closed, in contrast to previous behavior, despite operation of MU-PIC for a total of about 15 minutes after closure. In this period, there was no heat removal capability from the entire system when the PORV block valve was closed, except for the flow out the letdown line. However, heat was being removed from the core into the OTSG A, as shown by the increasing steam pressure in the secondary side of A and the rise in the A cold-leg temperature to reach the system saturation temperature at about 12 hours 9 minutes, both of which continued to the end of the period.

Period X: 13 Hours 15 Minutes to 16 Hours

At about the start of this final period of our analysis, the decision had been made to repressurize, increase the makeup flow, and attempt once more to get a reactor coolant pump operating. The condenser vacuum pumps had been started successfully, so vacuum was being established in the secondary steam system to allow the A OTSG to start steaming to the condenser at about 14 hours. Heat removal capability of the OTSG had once again been established.

The PORV block valve was closed at 13 hours 21 minutes, but the system pressure did not begin to increase until about 12 minutes later. At about 13 hours 36 minutes the system pressure began to increase, rising to about 2325 psi before makeup flow was throttled back and the pressure allowed to decrease slightly. At the same time, the hot-and cold-leg temperatures in the OTSG A deviated from the saturation temperature, and the two A cold-leg temperatures began two distinctly different behaviors, the temperature for the 1A cold leg lagging behind that for the 2A cold leg by as much as 30 minutes, or differing as much as 150°F in temperature at a given time. There is no obvious reason for this difference in behavior. Both are on the same OTSG; therefore, both should have been filled to the same level, and the only known difference between the two cold legs is that the letdown line is on the 1A leg.

With the "jogging" of the reactor coolant pump RC-P2A for a few seconds at 15 hours 36 minutes, the hot-and cold-leg temperatures showed immediate changes. The rises in temperature observed in the hot legs immediately after the pump was stopped (100 to 130°F) were caused by the thermal bounce of the walls of the hot-leg pipes reheating in stagnant steam because they had not been significantly chilled by the very short time flows induced by the jog operation of the coolant pump.

When the reactor coolant pump RC-P2A was again started at 15 hours 50 minutes, all hot-and cold–leg temperatures immediately equilibrated (within less than 3 minutes) at about 360°F, and the system pressure dropped to less than 1400 psi. The transient had been terminated and the reactor was finally put under control.

At the time of the very rapid pressure rise, the SRM count rate, increasing very slowly over many hours, had again reached the same value it had had at the time of the core reconfiguration at 3 hours 45 minutes. This was probably due to a concentration of fuel from the top of the core into the debris bed at about 6 to 8 feet from the bottom of the core as a result of the change in core geometry at 3 hours 45 minutes (which would decrease the SRM "view angle" of the fuel in the core), a slow heatup of the water in the downcomer, and little effect of changes in downcomer water level above midcore height.

If the makeup flow rates reported by the operators of about 425 gallons per minute were correct, then more than 34 000 gallons of water were pumped into the reactor coolant system after the PORV block valve was closed for the final time (37 000 gallons were removed from the BWST in this time period). Since the reactor coolant volume is 90 000 gallons, including about 11 500 gallons in the pressurizer, this amount is more than one-third of the total water volume in the system.

The distribution of the water and gas inventory of the primary system at 16 hours is shown in Figure II-32. At that time there were four gas bubbles in the system, one in the top of each OTSG, one in the top of the pressure vessel, and one in the pressurizer. Coolant water was being pumped through the hot leg of OTSG A, flowed over the top of the candy cane, and dropped through the gas bubble to fill the lower part of the OTSG and be recirculated by the pump. The reactor pressure vessel was filled above the hot- and cold-leg nozzles, and hydrogen was trapped in the pressure vessel head.

Summary and Conclusions of the Interpretation of Accident Sequence

The major features of this interpretation are:

1. All of the water removed from the BWST (271 000 gallons) during the 16 hours passed into and through the reactor primary system, through the pressurizer, and out the PORV, when it was open or was used to pressurize the system when the PORV block valve was closed.
2. Makeup flow rates are generally based on the average values reported for the several periods in NUREG-0600 (which, in turn, are based primarily on changes in BWST levels) but are modified as necessary to make the material balance fit.
3. The maintenance of high temperatures in the hot legs of the OTSGs for all times after about 3½ hours was due to the blockage of steam flow by the presence of hydrogen and the very low heat losses through the insulation present on the outer surfaces.
4. The top of the core was uncovered within the first few minutes after the reactor coolant pumps were stopped.
5. The top of the core remained uncovered until about 3 hours 20 minutes and was never uncovered again, although some parts of the damaged core remained steam blanketed and very hot for up to 4 days.
6. The major damage to the core had occurred by the time the reactor coolant pump was started

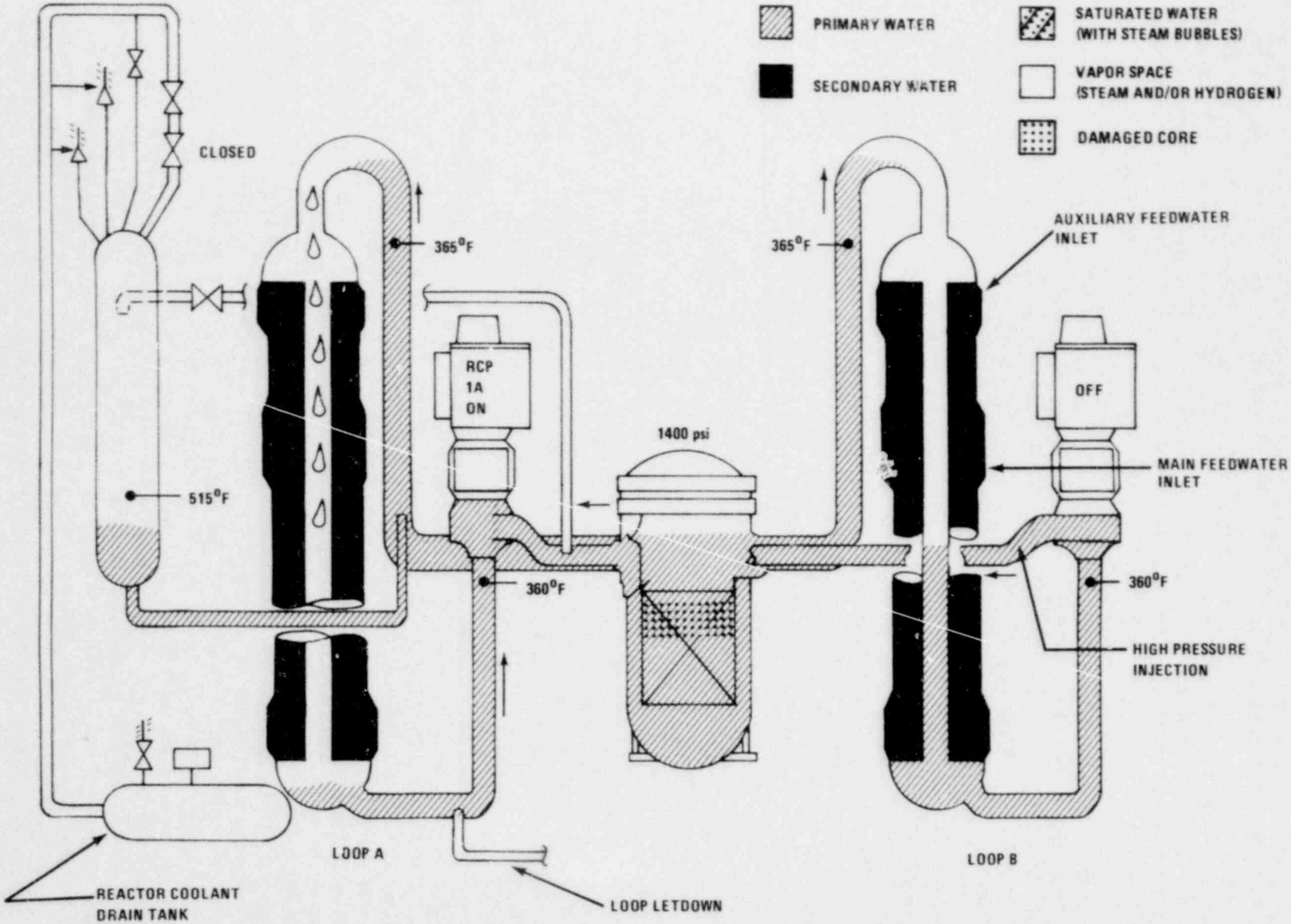


FIGURE II-32. Reactor Primary System at 16 Hours at 2 hours 54 minutes, although additional slumping occurred at 3 hours 45 minutes.

7. All of the fuel rods in the core burst, during an approximately 30-minute (center bundle) to 40-minute (lowest power peripheral bundles) period after the top of the core was uncovered at depths ranging from 1½ feet (center bundle) to 2 feet (peripheral bundle) from the top of the fuel rods.
8. Temperatures at which liquefied fuel (UO_2 dissolved in the zirconium metal–zirconium dioxide liquid eutectic at about 3500 to 3600°F) could be formed were calculated to have first been reached at 6 inches from the top of the fuel in the fuel rods in the central fuel bundle about 33 minutes after the top of the core was uncovered and were reached as low as 36 inches from the top of the fuel. Such temperatures were calculated to have been reached in the peripheral bundles at a depth of about 14 inches from the top of the fuel in about 46 minutes after the core was uncovered and at a depth of about 41 inches in 57 minutes.
9. The peak temperatures calculated for the fuel rods ranged from 4370°F in about 52 minutes for the highest powered bundle to a maximum of 4412°F for a medium powered bundle at 58 minutes to about 4358°F for a lower powered peripheral bundle at about 78 minutes.
10. The amount of hydrogen formed by oxidation of solid Zircaloy cladding during the temperature excursion was calculated to be about 308 pounds, and that formed from all of the damaged Zircaloy, including that contained in the liquefied fuel present at 3 hours, was calculated to be about 720 pounds. This is the minimum amount of hydrogen estimated to have been formed. The maximum could be as high as 820 pounds.
11. The major releases of hydrogen to the containment occurred before 4 hours accident time and during the long depressurization around 8 hours. No significant amount of hydrogen was produced after about 4 hours.
12. The minimum water level occurring in the core up to 3 hours is estimated to have been $4 \pm ½$ ft from the bottom of the fuel in the fuel rods on the basis of the amount of hydrogen produced, the amount of radioactivity released, the time at which significant levels of radioactivity were detected, and the structural damage estimated in the core.
13. The total amount of Zircaloy oxidized is calculated to be not less than 16 400 pounds and may have been as high as 18 700 pounds; i.e., between about 31% and 35% of the total Zircaloy in the core.
14. The damage in the core extends from the top downward at least 7 feet, and probably 8 feet, over most of the core and consists of oxygen embrittled Zircaloy cladding topped by a bed of debris that probably consists of fuel pellet fragments, partially dissolved fuel pellets, shells of Zircaloy oxide, and segments of embrittled Zircaloy cladding with outer skins of Zircaloy oxide, all glued together with liquefied fuel into a relatively tight and compact mass extending entirely across the core from wall to wall and penetrated by only a few vertical passageways, at most. In addition, fingers of liquefied fuel extend downward from the debris bed in several continuous subchannels between fuel rods, encompassing the neighboring fuel rods, to a depth of about 1 foot above the bottom of the fuel stack in the fuel rods. Not less than 32% of the fuel assemblies have such fingers of liquefied fuel.

c. Core Damage Estimates from Fission Product Release

At shutdown the reactor core contained fission products, activation products, and actinides. Some of these, notably krypton and xenon, are gaseous and can diffuse through the fuel pellet to collect in the gap between the fuel and the cladding. To a lesser extent, the halogens (iodine and bromine) can also diffuse into the fuel-clad gap. Any perforation of the cladding can release these fission products into the reactor coolant.

If the fuel temperatures are higher than operating temperatures, but well below melting, other radioactive materials are volatilized and can diffuse out. Also, diffusion of the noble gases and halogens increases so that a larger fraction of these can be released. The release of cesium is quite variable and could be caused by compound formation. Because of this variability and what is now known about cesium, it is not possible to determine precisely the temperature at which a reasonably large fraction of the cesium would be released; however, it is believed temperatures would not be lower than 1300°C (2370°F).[187,188]

At higher temperatures that cause the liquefaction or melting of fuel, some fraction of other fission products such as tellurium can be released. Data reported show that the escape of tellurium depends on many factors other than temperature.[189] Under oxidizing conditions some ruthenium may be released before melting. In general, rather large fractions of both tellurium and ruthenium are released in melting; but under some conditions, these materials can also be released before melt. The presence of ruthenium and tellurium does not prove that melt has occurred, but the absence of them is a good indicator that melt has not occurred. More recent experimental work,[187,190] while tending to confirm previous data, has not resolved all the questions regarding conditions—especially temperature conditions—under which fission products would be released.

Many of the fission products and most of the actinides occur as refractory oxides and are released only in relatively small amounts even at elevated temperatures. However, if damaged fuel pellets are rewetted, some of the more refractory radioactive material can be leached out. This process is slow and only small fractions of these materials find their way into the coolant by leaching. The longer damaged fuel is in contact with water, the more materials are released.

Categories of Fission Product Releases and Their Relation to TMI-2

Fission products and actinides can be divided into typical release groups, based on the ease with which they are volatilized. One such grouping (from Ref. 191) is in order of decreasing volatility.

- I Noble gases (Kr, Xe)
- II Halogens (I, Br)
- III Alkali metals (Cs, Rb)
- IV Tellurium (Te)
- V Alkaline earths (Sr, Ba)
- VI Noble metals (Ru, Rh, Pd, Mo, Tc)
- VII Rare earths and actinides
- VIII Refractory oxides of Zr and Nb

The fraction of gaseous and volatile fission products released depends on the temperature and the size of the fuel fragments. If the temperature is high or if the fuel is highly fragmented, nearly complete release of the volatile materials can be assumed.

Under the conditions that have been calculated for the accident at TMI-2,[188] nearly complete release of groups I and II can be assumed from all fuel that was severely damaged, plus some additional fraction from fuel rods whose cladding was perforated without damage to the fuel. This additional amount from perforated but otherwise undamaged rods is probably partly balanced by the amount not released from severely damaged fuel.

A major fraction of group III and a much smaller fraction of group IV could have been released from the most severely damaged fuel. Small fractions, approximately 10% or less, could have been released from perforated but otherwise undamaged rods, but this cannot be well estimated.

Leaching from Irradiated Fuel

Very small fractions of the remaining groups may have been released from the very hottest fuel. The principal mechanism for release of these refractory materials is probably leaching. Leaching from irradiated UO_2 has not been thoroughly studied. However, the work of Katayama[192,193] and of Forsyth and Eklund[194] has shown that the leaching rates are slow, comparable to those from glass. Quantitative data, especially for the temperatures and conditions existing in TMI-2, are too sparse for a reliable calculation of the rate of leaching, especially when one considers that the condition of the damaged fuel is completely unknown.

An additional complication is presented because the effective surface area of irradiated fuel presented to the water is almost impossible to estimate because of cracking and porosity. The most that can be done with the available data is to form an "educated guess" as to whether the fuel appears to be mainly in the form of very large pieces or in the form of very fine fragments. Without additional data it is not possible to estimate the actual size distribution of the fragments. However, a small fraction of the most refractory material can be expected to have found its way into the reactor coolant. An approximate leaching calculation is presented in Appendix II.7. On the basis of this approximate calculation, it is possible to state, with very low confidence, that a large fraction of the fuel can presently be fragmented and that the size of the fragments is more likely to be a few millimeters than dustlike. A similar calculation has been carried out by Powers.[191] His conclusions, although not identical with these, indicate that the observed activity may have been caused either by leaching from large-sized fragments or by distribution of particle sizes no more than a few percent smaller than 2 millimeters in diameter and none smaller than 0.6 millimeter in diameter.

Expected Dispersion of the Fission Products from the Reactor

Principal fuel damage probably started before 3 hours after turbine trip. There was probably only minor damage before 2 hours. The calculated total inventory[195] of fission products, activation products,

TABLE II-56. Activity in release groups*

Group	Activity
I	2.97×10^8 Ci
II	4.47×10^8 Ci
III	4.6×10^7 Ci
IV	1.61×10^8 Ci
V	3.85×10^8 Ci
VI	6.34×10^8 Ci
VII	2.69×10^9 Ci
VIII	4.80×10^8 Ci
Total	5.11×10^9 Ci**

*A few elements of low total activity, notably Fe, Cu, As, and Sb, have been arbitrarily located on the basis of melting point.

**Total does not quite agree with calculated total activity because of rounding.

and actinides is given in Table II-56 for 3 hours after shutdown.

A detailed discussion of the fission product-release pathways begins in Section II.B of this report where a short summary is included. Radioactive material released to the reactor coolant may have been partially flushed to the containment through the open PORV (RC-R2). Some of the material may have been flushed to the containment prior to the containment isolation and then pumped to the auxiliary building. However, the coolant may have contained only a minute fraction of the total activity at this time; it is highly improbable that a significant fraction of the coolant was released before the reactor building sump pumps were shutdown. There is an unsubstantiated possibility[196] that more water leaked to the auxiliary building after pump shutdown. This leakage would have terminated when the reactor building was isolated after 3 hours 56 minutes.

Most of the material flushed out of the RCS probably remained in the reactor building. Some additional material may have volatilized from the makeup tank. Aside from these losses, which are not expected to be very large, estimates of the total activity released from the fuel can be made by analyzing the reactor building air and water samples, the reactor coolant, and the auxiliary building tanks.

Iodine is quite volatile, and it may be supposed that a significant fraction is found in the air. However, the very high solubility of iodine in water and the strong tendency of atmospheric iodine to plate out on surface quickly reduces the amount of iodine in the air. Cesium, less volatile, is not expected to be present in the air in a significant quantity. On the other hand, the solubility of xenon and krypton is very low; these gases will be found almost entirely in the air.

To summarize, nearly complete release of noble gases, iodine, and cesium from damaged fuel is expected, even if the temperature is below the melting point. Significant releases of tellurium, ruthenium, and more refractory materials will occur only if the temperature approaches the melting point. Most of the noble gases will be found in air, and most of the other fission products will be found in water.

Distribution of Fission Products at the TMI Site

Analyses of samples of containment air, reactor coolant water, and auxillary building tank water are summarized in Ref. 197. Reactor coolant analyses show between 7% and 15% of the calculated inventory of iodine and cesium isotopes to be in the coolant. If these measurements are corrected for dilution by water from the borated water storage tank, the fractions will be a factor of 3 higher. Results for refractory materials show great variation. A sample taken on April 10 was analyzed by four laboratories. There was a large variation from laboratory to laboratory, indicating low confidence in the results. Analyses of krypton and xenon isotopes in the containment atmosphere also showed considerable variation. However, based on the most abundant isotopes (^{85}Kr and ^{133}Xe), there seemed to be 29% to 62% of the core inventory of noble gases in the containment air. Only 2% to 3% of the iodine and cesium was found in the auxiliary building tanks.

On August 28, 1979, a hole was drilled into the reactor building and samples of sump water were removed. Analyses of these samples showed 22% to 48% of the core inventory of iodine and cesium to be in the reactor building sump water.[198] In addition to iodine and cesium, very small amounts of Ru, Zr, Nb, Sb, La, and Ag were found. As expected, little ^{90}Sr was found. At most, the amounts corresponded to a few millionths of the core inventory. About 0.02% of the core inventory of ^{129m}Te was found.

All of these sample analyses were corrected for decay of the radionuclides to the time of analysis. This correction process is certainly more accurate than the analyses themselves; i.e., the accuracy of the estimates does not depend on the accuracy of the decay calculation. Table II-57 is a recapitulation of the release of volatiles.

Findings

From these results, one can cautiously conclude that between 40% and 60% of the core inventory of release groups I-III was released to the coolant; that only a small fraction of group IV was released; and that only minute amounts of the remaining groups were released. The amount of refractory isotopes released is consistent with leaching (see Appendix II.7).

These data tend to confirm other analyses of core damage. The data on radioactivity released are too sparse and variable for a precise conclusion to be made on the amount of core damage; however, the following conclusions appear to be supported.

1. About 50% of the reactor core was damaged sufficiently to release the most volatile fission products.
2. The low fractions of tellurium, ruthenium, and strontium indicate that no significant quantity of fuel reached the melting point of UO_2 (5200°F).
3. The amount of refractory isotopes in the reactor coolant is consistent with leaching.

d. Hydrogen Production, Removal, and Hazard

Introduction

One of the surprises of TMI-2 was the formation of large amounts of hydrogen from the reaction of the cladding around the fuel with the steam generated by the boiling water. In this section several aspects of the hydrogen "problem" are discussed. The following subjects are treated in this section:

1. hydrogen production,
2. hydrogen accounting,
3. calculation of bubble size,
4. removal of the hydrogen bubble, and
5. the hazard from the hydrogen bubble.

Hydrogen Production

Two possible sources of hydrogen are considered: metal-water reactions and radiolysis. Other conceivable sources include oxidation of UO_2, which has not been investigated. The production of hydrogen from metal-water reactions is known to have been large; therefore any hydrogen from other mechanisms is expected to be small in comparison. Radiolysis is not expected to produce large amounts of hydrogen. It is investigated because the possibility of oxygen production was considered at the time of the accident. If oxygen had been released, the hydrogen that was trapped in the reactor coolant system could have become flammable.

Metal-Water Reaction

Many metals are oxidized by water. The reaction is very slow at low temperatures for most metals. Both steel and zirconium are oxidized at an increasing rate as the temperature rises. The oxidation of zirconium, the major constituent of the cladding, oc-

TABLE II-57. Total volatile isotopes released from core

Released To	Isotope (fraction of core inventory) ^{133}Xe	^{131}I	^{137}Cs	^{134}Cs
Environment	0.01[1]	—[2]	—	—
RB Atmosphere	0.46[3]	—	—	—
RB Water	—	0.22[4]	0.48[4]	0.34[4]
RC Water	—	0.14[4]	0.12[4]	0.08[4]
Aux. Bldg. Tanks	—	0.03	0.03	0.02
Totals	0.46	0.39	0.63	0.44

[1]See Ref. 199
[2]Dashes indicate low values (generally less than 1%)
[3]Best estimate from data in Ref. 197.
[4]Average of observations.

curs rapidly as the temperature approaches the melting point. The reaction is

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$$

Each mole of steam produces precisely 1 mole of hydrogen, so that no change occurs in the volume of gas.

If the zirconium is in the form of a fine powder, the reaction takes place very rapidly because the water vapor has instant access to metallic zirconium with, at most, a very thin shield of zirconium oxide. However, the cladding is solid metal and the water has access only to the exterior. Any water vapor on the inside of the cladding is rapidly exhausted and can only oxidize a minute quantity of metal. The initial oxidation of the exterior is very rapid. However, the formation of an oxide layer shields the unreacted metal from access to the steam. This formation causes the reaction to proceed slower as the oxide layer becomes thicker. The shielding is not perfect, however, and some oxidation still occurs even with a relatively thick oxide layer. Experiments have shown that, when the temperature of the zirconium is constant, the thickness of oxide can be approximately represented by the equation

$$h^2 = Kt$$

where h is the thickness of oxide produced up to the time t. The quantity K can be reasonably well represented by

$$K = A\exp(-E/RT)$$

where A and E are experimentally derived constants. R is the gas constant, and T is the absolute temperature. Most investigators now use the Cathcart-Pawel rate constants in which $A = 0.00349\ in^2/s$ and $E/R = 32\,512°R$.

In a reactor accident the temperature of the cladding is not constant. Each kilogram of zirconium oxidized releases about 6½ MW. The release of this energy raises the temperature of the cladding. The table below shows how the oxidation rate increases with increasing temperature.

Temperature °F	K in^2/s
1000	1.9×10^{-13}
1500	1.0×10^{-10}
2000	4.4×10^{-9}
2500	5.3×10^{-8}
3000	3.1×10^{-7}
3500	1.2×10^{-6}
4000	3.2×10^{-6}

For a given oxidized thickness, the speed of the reaction is proportional to the quantity K, so that when the temperature increases from 2000°F to 4000°F, the reaction proceeds nearly 1000 times faster.

Because of the very large energy release in oxidation, the cladding heats up faster than the steam can eliminate heat. (At high temperatures the reaction power can exceed the decay power.) Because of the speed with which oxidation proceeds, once it has started, minor errors or uncertainties in the rate equations are not very important. What is critical is the water level in the reactor vessel. If the water level is low enough for a long enough time to oxidize about 10% of the thickness, the remainder can be oxidized in a short time. Oxidation acceleration is also possible due to fracturing or transformation of the formed oxide layer at high temperatures. However, these mechanisms are not expected to be operative until the runaway oxidation has begun, and they would not change the results appreciably.

These remarks must be tempered by consideration of clad melting. As soon as the melting temperature of the clad (or of the mixture of metal and oxide) is reached the molten material can run down the rod-like candle wax and refreeze in a lower, cooler zone. This procedure takes the unreacted metal away from the runaway oxidation reaction, so that oxidation can be at least partially limited by melting. The molten metal–oxide mixture readily dissolves UO_2, and the rate of oxidation of the resulting mixture is not well known. Therefore, once liquefaction has occurred, there is great uncertainty about the extent of oxidation that follows. This uncertainty, coupled with the lack of precise knowledge of water level, means that rather wide bounds must be placed on our ability to calculate the amount of hydrogen produced.

Hydrogen also can be produced by the reaction of water with steel. However, the amount produced appears to have been small in the TMI-2 accident. Calculations of the steel–water reaction have been performed. The uncertainties are even greater than those involved in the zirconium–water reaction. Because of the low production of hydrogen by this reaction, the overall uncertainty is not greatly affected.

A check on the calculation of hydrogen production occurs. The summed partial pressures of steam and hydrogen must equal the system pressure. The partial pressure of the steam is only approximately known; therefore an exact check is not possible. However, the partial pressure of hydrogen at any time must certainly be less than system pressure; this fact can help to reduce the uncertainty in the calculation.

Radiolysis

Radiation absorbed in water causes it to disintegrate into its constituents—hydrogen and oxygen. Many complex reactions are involved, but the net result is

$$H_2O + \text{radiation} \rightarrow H + OH$$

and

$$H + H \rightarrow H_2$$
$$OH + OH \rightarrow H_2O_2$$

The hydrogen peroxide finally decomposes into oxygen and water. Reverse reactions, or recombination, are

$$H_2 + OH \rightarrow H_2O + H$$
$$H_2O_2 + H \rightarrow H_2O + OH$$

and

$$H + OH \rightarrow H_2O$$

If the radiation is in the form of heavy alpha particles, there are high local concentrations of the radicals H and OH, and the production of H_2 and H_2O_2 is favored. On the other hand, if the radiation is sparsely absorbed, as with gamma rays or slow neutrons, the radicals are dispersed so widely that production of hydrogen and oxygen is not favored.

In addition to the ionization density, the water chemistry influences whether decomposition or recombination governs. The most important chemical regulators are dissolved hydrogen and oxygen.[200,201] If only hydrogen is in the water (above a low threshold concentration), recombination is much more rapid than decomposition, and no net hydrogen or oxygen is produced. If both are present with hydrogen predominating, the production of H_2 and H_2O_2 rises to a peak and then quickly declines essentially to zero. If hydrogen and oxygen are both present in about equal concentrations, both will continue to be produced as long as the radiation is absorbed.

Pressurized water reactors are operated with dissolved hydrogen to promote recombination. Even if this were not so, the metal-water reactions produce hydrogen, thus increasing the hydrogen concentration in water. Furthermore, before clad rupture, the radiation was mostly gamma rays, which do not favor decomposition; after clad rupture some fission products were released, but very extensive prior hydrogen production would have inhibited decomposition.

These conditions are not necessarily true if boiling occurs. The rising steam bubbles scavenge the molecular products, and recombination is suppressed. Under boiling conditions an almost stoichiometric mixture of molecular hydrogen and oxygen could form in the vapor space. The production is always slowed because some recombination exists before the molecular products are removed. This recombination is particularly important when the boiling rate is low, which was typical of conditions in the TMI-2 accident. An excess of hydrogen will reduce the effective yields of hydrogen and oxygen even when boiling is taking place, although the reduction is not as impressive as in the nonboiling regime.

Honekamp, et al. have calculated that the contribution of radiolysis during boiling could have raised the oxygen concentration in the bubble only to 0.7%.[202] Cohen calculated a maximum oxygen concentration of about 1% from all sources.[203]

Iodine and other halogens also promote decomposition, but by another process. Halide ions act as radical scavengers, and thus inhibit recombination. Experiments have been conducted with dilute halide solutions, and marked scavenging of radicals has been observed.[201] However, it would be difficult to quantify the extent to which the trace concentrations of iodine in the TMI-2 accident might have scavenged radicals.

Schwartz has calculated the effect of reactive impurities.[204] He shows that the amount of impurities present is more than 2 factors of 10 too low to prevent recombination.

During much of the TMI-2 accident, a large volume of mixed vapor and gas existed in the reactor coolant system (RCS). Water vapor can also be decomposed by radiation. However, the molecular yield is extremely low, and the only effect is usually the production of H and OH radicals. These radicals recombine to water in the presence of radiation. Impurities might increase the decomposition, but no major hydrogen or oxygen production from radiolysis of water vapor would be expected in the TMI-2 accident.

The net result of all these factors is that probably little hydrogen or oxygen was produced by radiolysis within the reactor coolant system. Some oxygen might have been produced during periods of boiling. The amount so produced cannot be precisely calculated.

Some decomposition might be possible in the water that flowed out of the PORV into the reactor sump. This water was exposed to high linear energy transfer (LET) radiation from entrained fission products and actinides and was exposed to the containment atmosphere. The containment always had more oxygen than hydrogen. Oxygen is also more soluble than hydrogen. Both factors combined to make the sump water oxygen rich, which would have enhanced radiolytic decomposition. However, the concentration of radionuclides was low, and dissolved nitrogen and NaOH inhibit decomposition; therefore radiolytic hydrogen was probably not a major addition to the very large amount released from metal-water reactions.

The radiolytic reactions are far from simple. Yields are complicated functions of the LET characteristics of the radiation, and recombination is a complicated function of water chemistry and state. Estimates of hydrogen and oxygen formation in the TMI-2 accident could be inaccurate and inconsistent unless based on experiments conducted under very similar conditions. Estimates of the maximum and minimum reasonable yields can be made, but it should be understood that these are only estimates.

Hydrogen Accounting

A number of estimates of the amount of hydrogen produced by the metal-water reaction have been made. For example, Picklesimer made an early estimate of 220 to 260 kg of hydrogen in the first 3 hours.[205] Cole estimated 350 kg in the same time frame.[206] A later estimate by Cole was based on more realistic calculations and indicated that 450 kg at 6.5 hours probably was produced.[207] This calculation includes less than 10 kg from oxidation of stainless steel. The President's Commission technical staff estimated that from 434 to 620 kg probably was produced[205]. A calculation made for this study (Section II.C.2.b) produced 330 to 410 kg in the first few hours; this is consistent with a total production of 450.

The calculation of Cole also includes the partitioning of hydrogen between the RCS and containment.[207] This partitioning is important in accounting for the removal of hydrogen. Because Cole's estimate is within the bounds of Picklesimer's,[205] it will be used as a starting point for the analysis.

Cole estimated that at 6.5 hours, 250 kg of hydrogen was in the RCS and 200 kg was in the containment. In later depressurization, between 7.5 and 14 hours, about an additional 100 kg is believed to have been added to the containment. At the time of hydrogen burn, 150 kg might have been in the RCS and 300 kg in the containment. The calculated amount burned, based on the peak overpressure, was 267 kg.[205] Cole estimated that 330 to 360 kg existed at the time of burn.[207] Measurement of the hydrogen concentration on March 31 indicated about 80 kg at that time; therefore the amount consumed according to Cole's estimate would be 250 to 280 kg.[207] The lower estimate of the President's Commission technical staff would have given about 350 kg in containment and, hence, about 100 kg in the RCS. The maximum production according to the President's Commission technical staff, which is considered less likely, would give an RCS content of 270 kg.

These estimates assume that little hydrogen was produced during later depressurization. This premise is believed likely. Even if some of the core was uncovered again, the rods exposed already would have been at least partially oxidized, and further oxidation would have been slow.

The estimated "most likely" amount remaining in the RCS, 100 kg, includes the amount in solution (about 26 kg at 1000 psi and 280°F) as well as that in a bubble (about 74 kg). At a pressure of 1000 psia and 280°F (typical of conditions during the several days following the accident) this measure would be 645 ft^3. If about 1.6 pound moles of fission gases and 3.2 pound moles of helium are added to this, the total of all noncondensible gases in the bubble is 684 ft^3 at 1000 psia and 280°F (29 000 ft^3 at 273 kg and 1 atm pressure [STP]).

The largest amount considered for the RCS, 270 kg, would give 244 kg in the bubble, for a total volume of 2166 ft^3 at 1000 psia and 280°F (92 000 ft^3 at STP).

Bubble size calculations extrapolated back to 16 hours give a volume of 1470 ft^3 at 1000 psia. If the "most likely" hydrogen estimate is correct, this volume would be about 44% hydrogen; the remainder could be any other gas, mostly steam. For example, a 786-ft^3 bubble of steam in a "hot spot" within the damaged core would be possible. The maximum estimate of 270 kg is impossible if the bubble size calculations in the next section are correct. This estimate lends credence to the belief that the smaller quantity is more reasonable.

Based on the "most likely" quantities, the hydrogen accounting is then as follows:

Produced	450 kg
Released to containment	350 kg
Burned	270 kg
Remaining in containment	80 kg
Remaining in RCS at 16 hours	100 kg
In solution at 16 hours	26 kg
In bubble at 16 hours	74 kg

Calculation of Bubble Size

The bubble size was calculated during the course of cooldown and bubble removal by Metropolitan Edison and Babcock & Wilcox. The same physical principle—the compliance of a liquid containing a gas bubble—was used by each organization. After the accident, Sandia carried out an independent investigation[207] at the request of the TMI Special Inquiry Group. The results of the latter study, as given in Figures II-33 and II-34, show that the bubble was about 1470 ft^3 at 2:00 p.m. on March 28 and was completely gone by 6:00 p.m. on April 1.

Although each organization has used the same basic principle, the equations appear different because different simplifying assumptions have been used.

The Met Ed formula[208] is the simplest. It neglects the compressibility and thermal expansion of water and the solubility of hydrogen. These simplifications lead to a consistent 300 ft^3 overprediction of the bubble size at 875 psi.

The B&W formula[209] includes these effects but neglects changes in vapor mass in the pressurizer and the effect of the hemispherical lower head of the pressurizer and does not consider the partial pressure of water vapor. The net result is generally about 5% underprediction of bubble size.

The Sandia formula[207] includes all of these terms but neglects the effect of leakage during bubble size experiments (the compliance of the steel vessel and

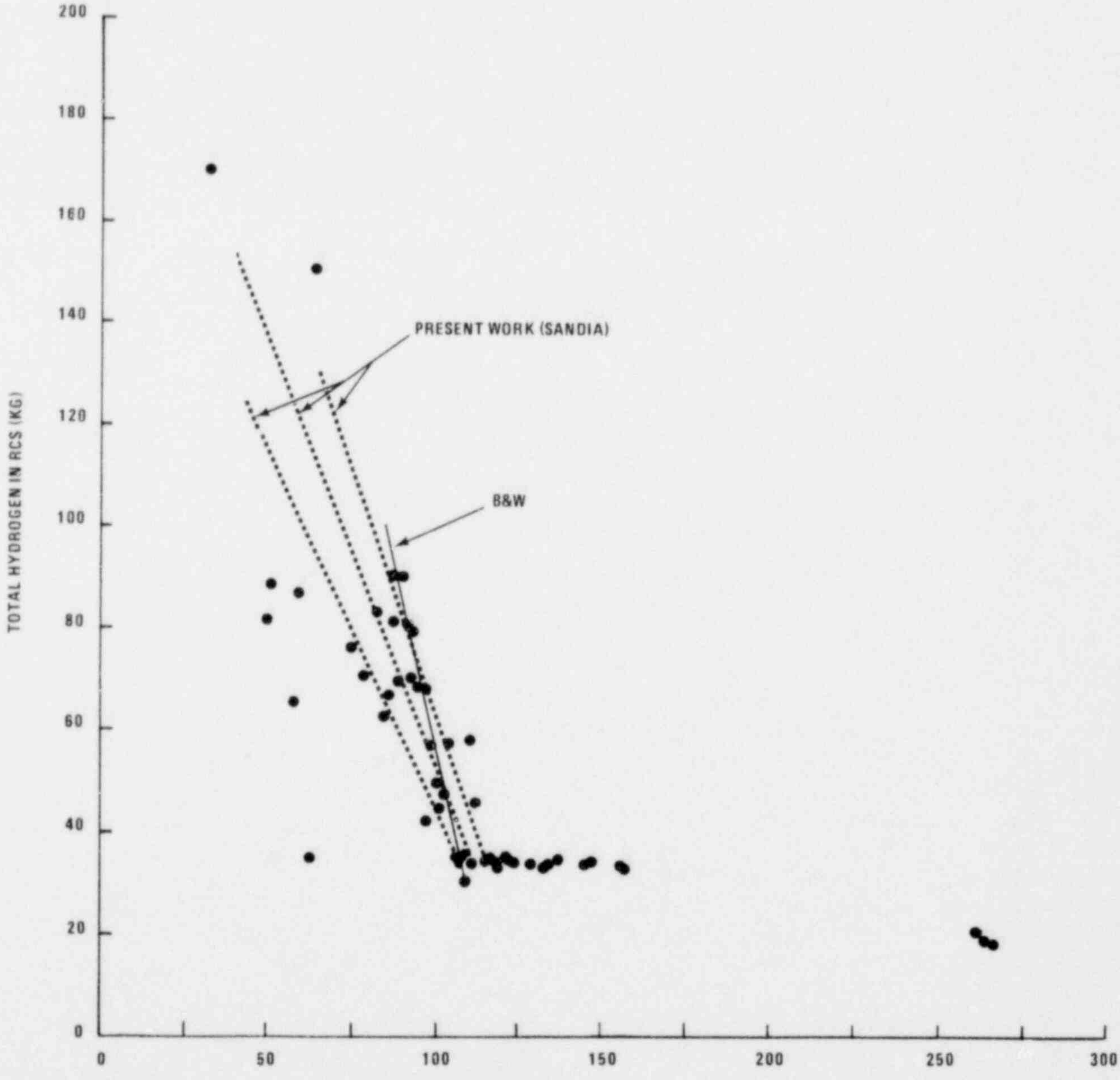


FIGURE II-33. Total Hydrogen in RCS

change of density because of temperature change during experiments). The effect of the last two terms has been evaluated and is known to be small. The leakage effect has not yet been evaluated but is also expected to be small.

Each bubble experiment was performed by subjecting the RCS to a known change in pressure and deducing the associated change in volume. From this change the compliance of the liquid gas system was calculated, and hence the size of the bubble. The size of the bubble decreases with increased pressure for two reasons: because of compression of the gas, and because more of the gas goes into solution at the higher pressure. The latter effect was neglected by Met Ed.

Even if an accurate formula is used that includes all the physical effects, the inherent inaccuracy of the measuring system would make an accurate prediction difficult. One needs to measure small changes in volume that correspond to small changes in pressure in a very large system by using instruments that are not of laboratory quality.

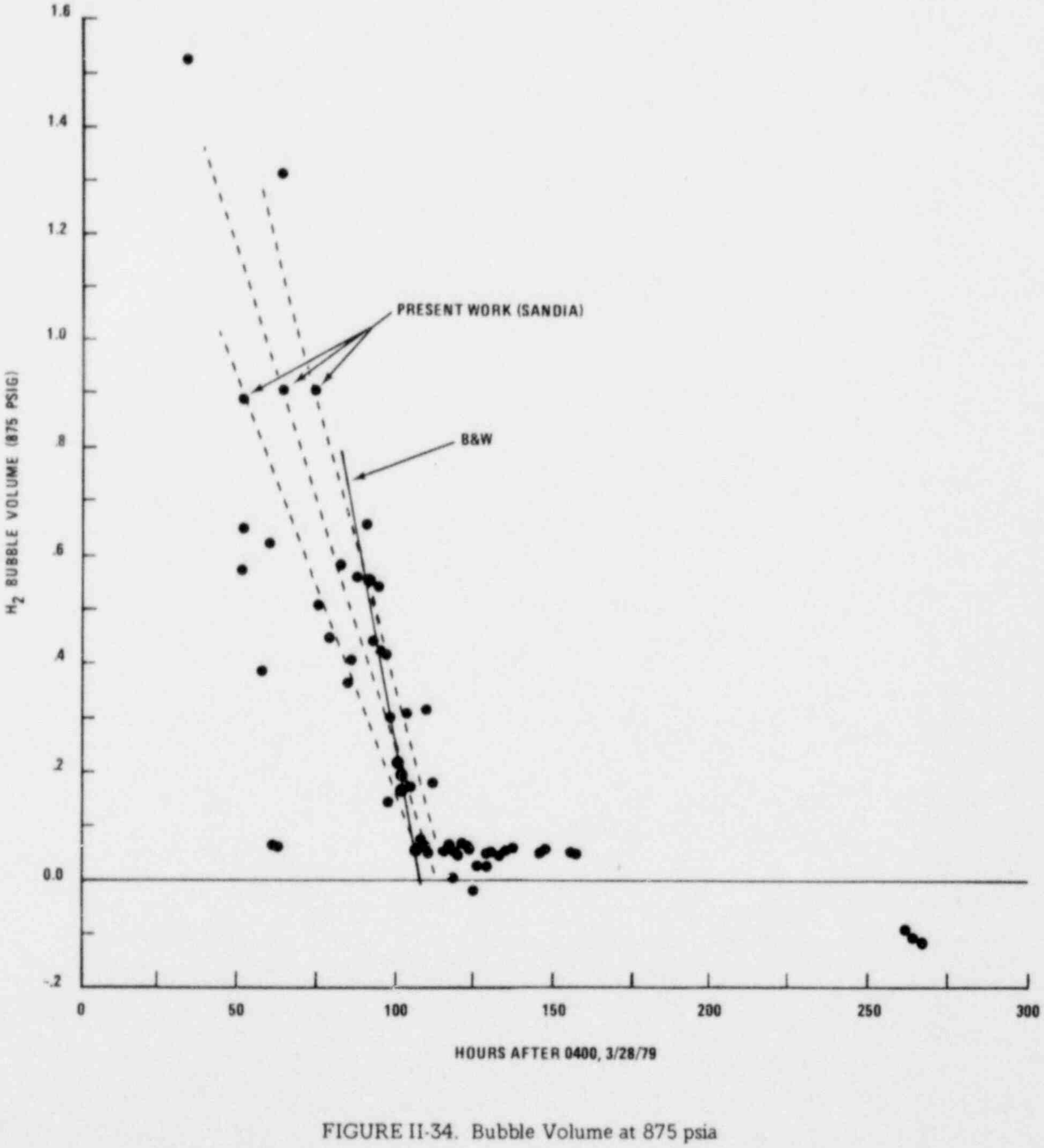


FIGURE II-34. Bubble Volume at 875 psia

Error Analysis

An error analysis of the Sandia formula has been carried out. The errors in bubble size are dependent on the conditions of the experiment and on the size of the bubble. Conditions for most of the bubble size experiments were approximately as follows:

RCS pressure	1000 psi
RCS temperature	280°F
Pressurizer level	250 in
Makeup tank level	45 in
Makeup tank temperature	81°F
RCS pressure	12.2 ft^3/psi error
RCS temperature	1.58 ft^3/°F error
Pressurizer level	97.3 ft^3/in error
Makeup tank level	181.4 ft^3/in error
Solubility	4.43 ft^3/percent error

Errors in each of the measured quantities could be as great as 2% of full range.[210] However, data are normally more accurate than this, and 2% of each reading is considered more likely. An error of 10% in solubility is considered reasonable. Then the possible total errors are:

Error due to RCS pressure error	= 244 ft^3
Error due to RCS temperature error	= 9 ft^3
Error due to pressurizer level	= 486 ft^3
Error due to makeup tank level	= 163 ft^3
Error due to solubility	= 44 ft^3

All errors would probably not occur simultaneously and would not normally all have the same sign. Note, however, that the largest error—that due to pressurizer level error—is nearly as large as the bubble, and several of the errors are large fractions of the bubble size. This clearly explains the great variability in bubble size estimates.

Removal of the Hydrogen Bubble

Except for changes in dissolved hydrogen due to changes in RCS pressure and temperatures, degassing at a constant rate of letdown would give a constant rate of bubble shrinkage. Figure II-33 shows the results of bubble calculation with the Sandia formula, along with a least squares fit for removal rate. Also shown in Figure II-33 is a removal rate calculated by B&W and a one standard deviation error band about the Sandia fit. Figure II-34 shows the same data, except that the ordinate is total hydrogen in the RCS—in the bubble and dissolved in the coolant. The time of removal can be taken to be the intercept with the horizontal axis in Figure II-33. The data show that the bubble had disappeared between 3:00 and 9:00 p.m. on April 1.

The removal of hydrogen was accomplished both by letdown and by pressurizer venting. It is not possible to estimate accurately the amount removed by each. However, from the fact that the hydrogen in the containment atmosphere increased by only a modest amount during venting, it can be assumed that venting was not the principal removal mechanism.

The removal rate by letdown is

$$\frac{dn}{dt}H = \frac{1}{M_H}\frac{dm}{dt}H = \frac{1}{M_w}\frac{dm}{dt}w\,(N_{AR} - N_{AM}),$$

where dn/dt is the molar letdown rate, moles per minute; M is molecular weight; and dm/dt is the mass letdown rate.

N_A is the mole fraction of hydrogen in solution, and the subscripts H and w refer to hydrogen and water and R and M refer to RCS and makeup tank conditions. The mole fraction in solution is, by Henry's law,

$$N_A = P_A/K$$

where P_A is the partial pressure of hydrogen in the gas and K is the Henry's law constant. For RCS conditions $K = 9.3 \times 10^5$, and for makeup tank conditions $K = 11 \times 10^5$.[208] These values are for 300°F and 75°F, the nearest tabulated points to 280°F and 80°F. The partial pressure of water vapor is taken to be equal to the saturation pressure at the indicated temperature. This condition is not strictly accurate but is within a few percent. The partial pressures of hydrogen are 933 psia and 39.6 psia at total pressures of 1000 psia and 40 psia for the RCS and makeup tank. With these values, letdown removes 9.64×10^{-4} moles of hydrogen per mole water. Note that Dalton's law must hold for a bubble of mixed gases. If a separate bubble contains pure steam, Dalton's law cannot be applied to the total.

The letdown rate as given in postaccident notes was about 30 gallons per minute, except for times when the letdown cooler was plugged. An average rate might have been about 25 gallons per minute. This rate is a mole rate of 10.64 pound moles of water per minute or 0.0103 pound moles of hydrogen per minute, referred to RCS conditons, for the 94 hours of bubble removal that would have removed 52.6 kg.

Leakage is estimated to be 5 to 6 gallons per minute. It is assumed that all leakage is due to reactor building conditions, where the partial pressure of hydrogen is so low that it would be considered negligible in comparison with RCS conditions. The molar removal rate is then 0.001 moles hydrogen per mole water, and 5 gallons per minute (again referred to RCS conditions) will remove 10.5 kg in 94 hours. Much of the leakage actually eventually goes to the letdown system. The difference in hydrogen scavenging rates is negligible.

The amount remaining (74–52.6–10.5), or 10.9 kg, could have been removed by venting of the pressurizer. This venting would cause only a 0.2% increase in containment hydrogen content, which explains why a marked increase in hydrogen content due to venting was not observed. Leakage should have caused an additional 0.2% increase in containment hydrogen content.

The amounts removed by using the "most likely" original amount are:

Letdown	52.6 kg	(71%)
Leakage	10.5 kg	(14%)
Venting	10.9 kg	(15%)
Totals	74 kg	(100%)

No exotic or improbable mechanisms need to be invoked to explain the postulated disappearance.

The Hazard of the Hydrogen Bubble

The initial concern expressed on March 29 was that the bubble was growing because of radiolysis of the water in the reactor to produce hydrogen. Later interest focused upon the likelihood of oxygen formation and the hazard of an explosion within the reactor.

Oxygen Content

Assurance had been given as early as March 29 by a B&W scientist that no oxygen problem existed. This information was given to T. Novak but apparently did not reach the NRC officials to inform the public until much later.

On March 30 and 31, Roger Mattson requested both the Office of Research and the Division of Systems Safety of NRR to determine the possibility and consequences of a hydrogen explosion in the reactor. The responses are summarized in Refs. 211 and 212. The early information given to Mattson was based on experiences from a boiling water reactor and from the advanced test reactor (ATR); hence, the information was not applicable to a pressurized water reactor and certainly did not apply to the situation at TMI-2 in which the coolant had a large amount of hydrogen in solution. Some scientists who were questioned were unable to give definitive answers promptly.

Notes taken at the time at the NRC emergency center, including those by Mattson, do not indicate that anyone disagreed with the possibility of a hydrogen–oxygen explosion. Among those queried on the effects and probability of explosion was B&W. The only note found to indicate mild disagreement is the record of a conversation with B&W to the effect that B&W "feels that H_2 recombination is taking place under gamma flux." Notes indicating that other experts basically agreed with the estimates of oxygen production exist. On April 1, the word from B&W was that B&W officially "thinks not flammable."

The opinion was almost universal that the bubble would be explosive, either very soon or in a matter of some days.

Late in the day of March 31, and especially on April 1, other data began to be received that contradicted the belief that the bubble contained oxygen. In the meantime, however, other scientists had been asked about the possibility of an explosion, and still others were delivering opinions on the damaging effects of explosions. It was difficult to sort out the facts in the confusing melange of differing opinions.

In view of the disagreement by the experts, the following summary was prepared on April 1:

Flammability limit	5% O_2 in pure H_2
O_2 production rate	1% per day
Current O_2 concentration	5%
Detonation limit	12% O_2 in pure H_2

Emergency center notes for April 1 show that information was increasingly being received stating that no oxygen was being produced. On April 2 virtually all incoming information stated that no oxygen existed.

A wide cross section of experts was involved: NRC staff, National laboratories, NRC contractors, Department of Energy laboratories, the academic community, and reactor manufacturers.[213] At some time on April 1, the weight of opinion was that oxygen was probably not present. Even then, however, explosion and structural experts, who had not yet been advised of the latest findings, continued to give opinions on the hazard of explosions.

Explosive Hazard in Reactor Vessel

A number of computations were made of the effect that a hydrogen detonation would have on the reactor vessel assuming that an explosive mixture existed (which was highly improbable). These calculations, of which those of Ref. 214 are typical, generally showed that major damage to the reactor vessel was unlikely, although some showed that the strength of the upper head might be marginal. Generally, specialists in explosive damage would be unable to predict the effects on the basis of such calculations without experiments. Less sophisticated analyses—many of which had assumed a stoichiometric mixture—gave rise to excessive fears for the safety of the reactor vessel.

Of equal interest is whether fragments of the reactor vessel could have been propelled with sufficient velocity to breach the containment. Specialists now generally are agreed that this is so improbable that it can be virtually ruled out, especially because any explosive fracture would be highly unlikely.

Because no possibility existed of an explosive mixture being formed, the whole question is academic, and it can be concluded that no explosive hazard was present.

Considering the lack of unanimity on March 31, the decision to consider whether the bubble was potentially explosive was correct. In the face of contradictory opinions, it is proper to give consideration to the worst case.

Explosive Hazard in Containment

A more realistic hazard was the possibility of sudden depressurization, with release of the hydrogen from the RCS to the containment. This depressurization was unlikely but possible. If the entire inventory of hydrogen had been added to the containment, an explosive mixture might have been formed.

Analysis of the containment atmosphere on March 31 showed 1.7% H_2, 15.7% O_2, and 82.6% N_2 for one sample, and 1.7% H_2, 16.5% O_2, and 81.8% N_2 for another. At a temperature of 80°F and pressure of 14.3 psia, the latter would be 86.1 pound moles H_2, 835.9 pound moles O_2, and 4144 pound moles N_2. The addition of all the hydrogen in the RCS—100 kg or 110 pound moles—would raise the hydrogen concentration to 3.8%. This elevation is still below the flammable limit. However, if the entire bubble was hydrogen, an addition of 185 pound moles would occur. This addition would give a hydrogen concentration of 5.2%, which could be flammable. However, the burning of about 290 pound moles on March 28 did not damage the containment. Therefore, the burning of 270 pound moles or less on March 31 likewise would not damage the containment.

Findings

The most likely estimate for hydrogen production is 450 kg, equivalent to oxidation of approximately 50% of the cladding. It is possible that the amount produced could have been as great as 520 kg. A total gas volume of 1470 ft^3 was probably present in the RCS at 8:00 p.m. on March 28. The fraction of hydrogen in this bubble or bubbles could have been 40% to 100%. The hydrogen was removed from the bubble by letdown, leakage, and venting; no unusual mechanisms need to be hypothesized to account for bubble removal.

The variability in estimates of bubble size came from the different methods of computation that were used by different organizations and from the inherent inaccuracy in the method of measurement. The bubble disappeared about 6:00 p.m. on April 1.

Little or no oxygen was present in the bubble and a very low probability of explosion existed. The incorrect perception of an explosion hazard stemmed from contradiction among supposed experts. This perception was known or should have been known to be false by the afternoon of April 1.

A flammable mixture in containment due to release of all the hydrogen would have been possible but very unlikely. Even if it had occurred, the containment would not have been damaged.

e. How Close to a Meltdown?

The extent of damage to the reactor core at 3 and 4 hours after the start of the accident was estimated and discussed in Section II.C.2.b. The estimated damage at 3 hours consisted of embrittled Zircaloy fuel cladding down to about 8 feet from the bottom of the core, with a "debris bed" above consisting of fuel pellet fragments, Zircaloy oxide shells, fractured Zircaloy cladding with an oxide layer on the outer surface, and frozen masses of liquefied fuel (UO_2 dissolved in the Zircaloy metal–Zircaloy dioxide eutectic liquid).

The damage produced later, shortly before 4 hours, lowered the depth of embrittlement and the debris bed, and may have produced additional amounts of liquefied fuel in the debris bed, which then ran down the subchannels between neighboring rods to reach depths of about 1 foot from the bottom of the fuel in the fuel rods. Despite this amount of damage, a core meltdown, as normally considered, did not occur. However, it almost occurred twice.

The first time was in the first heatup between 2 and 3 hours (6:00 and 7:00 a.m.), and was probably stopped by the closure of the PORV block valve and the operation of reactor coolant pump 2B. The second was in the second period of damage at 3 hours 45 minutes and was probably stopped as a result of the core rearrangement and the initiation of maximum HPI flow at 3 hours 56 minutes.

In the following discussion, it is assumed that the PORV block valve was not closed at 2 hours 20 minutes; i.e., the first "close call" is allowed to proceed. The amount of information that provides certain evidence about the condition of the core at 3 hours 30 minutes is so small that a discussion of the second "close call" is considered fruitless.

When the PORV block valve was closed at 2 hours 20 minutes, not only was the loss of coolant from the reactor system stopped but the increasing pressurization raised the boiling point of the coolant. This rise produced an initial decrease in steam flow as the coolant heated up and was followed by a significant increase in steam flow in the damaged core, as a given amount of decay heat from the submerged part of the fuel rods could evaporate a greater amount of steam, by the inverse ratio of the heats of evaporation at the two pressures. The difference is not small, the heat of evaporation being between 20% and 21% less for a system pressure of 2200 psi (compared with about 700 psi). Such an increase in steam flow rate in the later stages of the heatup could have had a significant effect on limiting peak temperatures reached.

Even though the system contained a large amount of hydrogen (a noncondensible gas), the local boiling temperature for the coolant was determined by the system pressure, not the partial pressure of steam in the vapor space. The surge of water accompanying the start of reactor coolant pump 2B at 6:54 a.m. both thermally shocked the embrittled core and reduced the fuel temperatures.

If the PORV block valve had not been closed at 2 hours 20 minutes, the continued loss of water from the PORV would have lowered the level of water in the core. The flow of steam would have decreased as the length of fuel rod submerged to generate it was decreased. The decreased flow of steam would have allowed faster heatup of the exposed parts of the fuel rods and higher temperatures. TMIBOIL computer code calculations reported in Section II.C.2.b indicate that temperatures greater than the melting point of UO_2 (about 5200°F) would be reached about 3 to 3½ feet down the fuel rod, if the water is boiled down to 3 feet from the bottom of the fuel stack in the fuel rod, and liquefied fuel could be formed to the midplane of the core.

Thus, up to approximately one-half of the fuel in the core could have been in liquid form about 50 minutes after the reactor coolant pumps were shut down if the PORV block valve had not been closed when it was. The liquid fuel most likely would have flowed or slumped onto the stubs of fuel rods remaining, adding significant amounts of heat to them and causing more fuel to become liquid. However, some of the liquid fuel probably would have dropped into the water pool below, increasing the generation of steam. Whether the additional steam generated could have produced enough cooling to reverse the meltdown would depend, at least in part, on whether a steam eruption could be produced to disrupt the melting core, thereby improving heat transfer.

With continued water loss, it seems more likely that there would have been, sooner or later, some steam generation rate and refluxing beyond which the steam flow was too low to provide cooling, and a core meltdown would have occurred. This condition could have been reached at almost any time between about 50 minutes and about 70 minutes after the reactor coolant pumps were shut down at 1 hour 40 minutes (5:40 a.m.), depending on the actual rate of coolant loss from the core.

It can thus be concluded that the reactor was probably within about 30 to 40 minutes of having a substantial fraction of the fuel liquefied or molten (which could then have resulted in an irreversible core heatup and meltdown) at the time of the PORV block valve closure at 2 hours 20 minutes (6:20 a.m. on March 28, 1979).

No reasonable estimate can be made at this time as to how close the core came to a meltdown in the second period because too little is known about the condition of the core after 3 hours 30 minutes. However, if the makeup flow had not been increased at about 4 hours, the core could have again heated up as the water level dropped and the flow of steam in the core decreased, given the fact that the PORV block valve was opened at the intervals it was.

Phenomena and Consequences Had a Meltdown Accident Occurred at TMI-2

One conclusion of the Special Inquiry Group is that the accident at TMI-2 may have been approaching a core meltdown accident. In a more technically accurate sense, the TMI-2 accident progression was such that a substantial fraction of the fuel was near the temperature required for formation of fuel-clad eutectic material, so that a loss of coolable fuel geometry was very possible. Because of this proximity to such an accident, a discussion is presented here of the accident progression assuming there had been a meltdown. The physical progression of the fuel, related events within the reactor building such as pressure increases and hydrogen combustion, and the timing associated with various events are described.

Briefly, the following discussion indicates that had a core meltdown occurred in TMI-2, the consequences would likely not have been catastrophic. The reactor building probably would have survived the accident, and the large majority of the radioactive material released from the fuel in the accident would probably have been retained within the reactor building and not released to the surrounding environment.

Present knowledge about the physical phenomena discussed in this section is subject to considerable uncertainty. Although important meltdown accident research is under way in the United States and Europe, much study is still needed in the areas of, for example, fission product release from fuel, large scale fuel melting and liquefaction, fuel–water interactions, and fuel–concrete interactions.

We believe that had a meltdown accident occurred at TMI-2, the likely path followed would not have led to disaster; however, considerable additional research into meltdown accident phenomena is needed to reduce the uncertainties associated with these phenomena and to provide a better basis upon which to consider such accidents.

This discussion of a meltdown accident progression begins at roughly 2 hours into an accident like that described in Section II.D as "Alternative Accident Sequence 6." This accident follows the course of the TMI-2 accident up until the time of PORV block valve closure at 2.3 hours. At this point, it is assumed that the block valve is not closed, so that the loss of coolant continues. Calculations discussed in Section II.D.2.g indicate that this alternative sequence could have led to melting (or liquefying) of a substantial fraction of the fuel.

The conclusions discussed on the extent of core damage in TMI-2 suggest that recovery can occur from a partially molten condition in the core. However, after a certain fraction of the fuel becomes liquefied or molten, measures to prevent additional melting would not succeed. That is, intervention by the operating crew to increase flow of water into the core (if possible) could be expected to stop the accident progression up to a certain point; beyond that (indeterminable) point, full scale melting of the core would likely occur despite attempts to provide cooling. Therefore, beyond this particular point, a "meltdown accident" would occur. Present knowledge of the phenomena of large scale fuel melting or liquefaction is not sufficient to estimate this point of "no return." For the purposes of the following discussion of a meltdown accident progression, the core condition resulting from the accident sequence postulated in Section II.D is assumed to have degraded beyond this point.

Figure II-35 displays the progression of a meltdown accident in terms of specific, important phenomena, showing parameters that can potentially affect the consequences of the accident. The various parameters and their importance are discussed as follows.

Fuel Melting and Slumping

As the amount of fuel reaching eutectic formation or melting temperature increases, the core would experience changes in geometry. In regions where liquefied or molten fuel has formed, such fuel might begin to run down the fuel pins, refreezing upon traveling into cooler areas.[215] Depending on the amount of water and the temperature gradients in the core, refreezing can occur near or relatively far from the place of liquefaction or melting. With available data on fuel melting phenomena, it is not possible to determine definitively the method by which the fuel would slump; i.e., whether the molten fuel would "drip" into lower regions or would initially slump into a large mass (or masses) and collapse en masse.

Because of the extreme temperature gradients experienced during such fuel melting (which would allow the liquid or molten fuel to refreeze near the point of liquefaction), the collection of a large quantity of molten material, with the subsequent collapse into the lower regions, would seem to be the more likely alternative. This collection and collapse of a large mass of molten fuel corresponds to the upper choice in the core slumping mode column in Figure II-35; therefore the most likely accident path would follow this upper route.

Table II-58 indicates that for the particular accident discussed, the time required to result in core collapse into the lower head would be about 1 hour after the beginning of core uncovering, so that about 3 hours would have elapsed from the time of the initiating event to collapse of the core.

In this time period, the reactor building atmosphere would contain increasing levels of steam, hydrogen, and fission products. Steam production

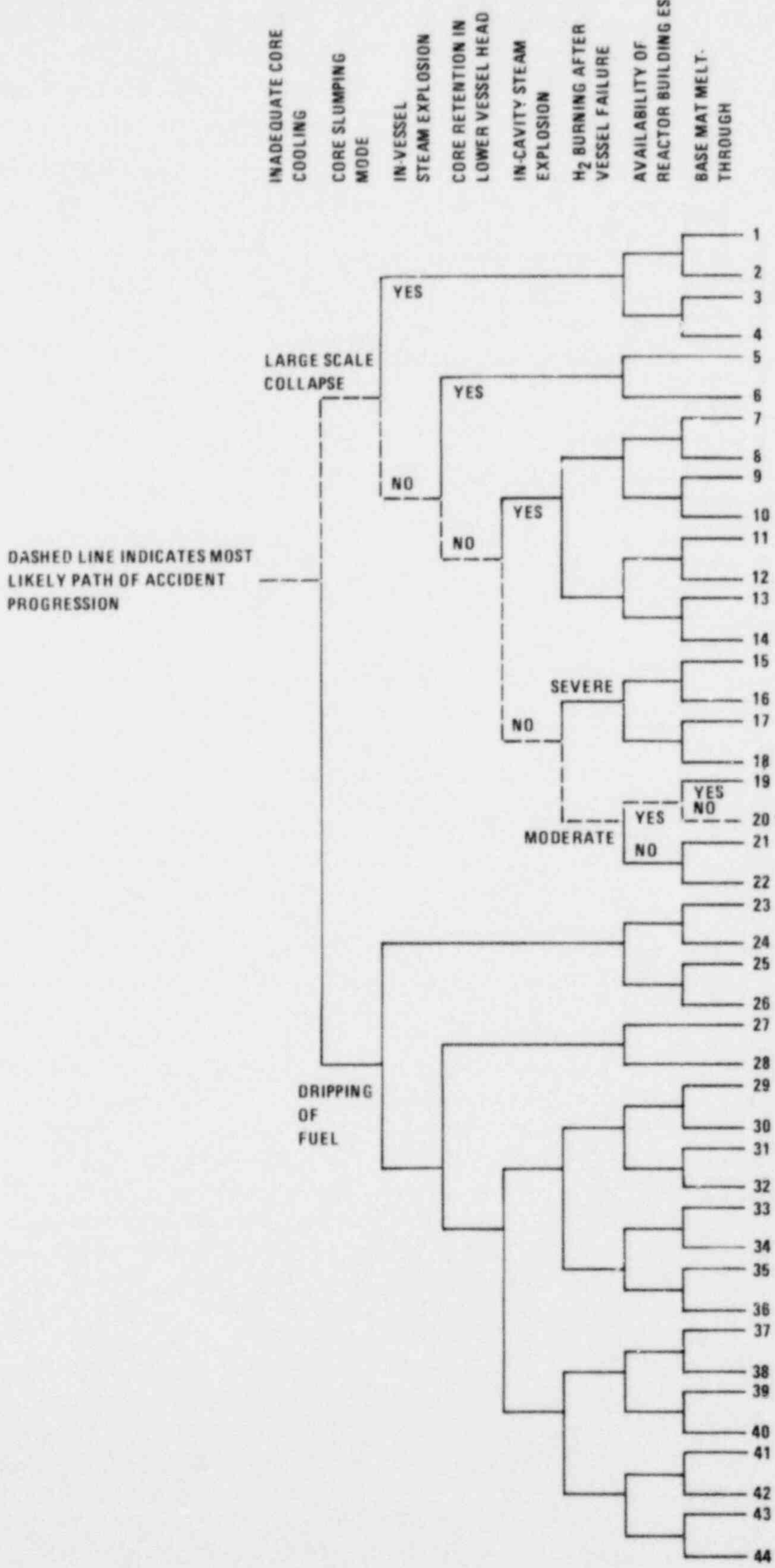


FIGURE II-35. Events in the Progression of a Meltdown Accident

TABLE II-58. Timing of a meltdown accident

Event	Time, minutes	
	Case A*	Case B**
Start core uncovery	101	101
Start core melt	133	133
Core collapse into head	165	165
Head failure	190	167
Start concrete attack	190	220

*Case A: no metal-water reaction in the bottom head, and no debris particulation in reactor cavity.

**Case B: 100% metal-water reaction in the bottom head, and debris particulation in reactor cavity are assumed.

would result from the evaporation of water remaining in the lower regions of the core and, to some extent, the water in the lower vessel head. In the situation when reactor building engineered safety features (ESFs) are operating, Figure II-36 indicates a negligible pressure increase during this time period. In the case of failure of the reactor building ESFs, reactor building pressure would increase slowly, rising to a pressure of about 30 psi (absolute) at the end of this period.

Hydrogen production in significant quantities begins as the core uncovers, heats up, and melts. As Figure II-36 indicates, some of this hydrogen would escape through the "break" in the reactor coolant system (RCS), which in this case is the stuck-open PORV. Upon failure of the lower vessel head, hydrogen not as yet released to the reactor building atmosphere would escape. Furthermore, in this time period the first substantial amount of fission product release from the fuel would occur. Chemical species of relatively high volatility represent the majority of the radioactive materials released; these include primarily the noble gases (e.g., xenon and krypton), halogens (e.g., iodine), and some alkaline metals (e.g., cesium).[217] While some fraction of the latter elements may deposit on (relatively) cold surfaces within the RCS,[218] or be retained in water (possibly) present in it, significant amounts of all these elements might nonetheless escape into the reactor building atmosphere. These releases would be essentially completed by the time of vessel failure. Operation of the reactor building ESFs (especially the spray system) could be expected to reduce amounts of radioactive material in the building atmosphere. Natural deposition processes would, with time, also reduce the amount of material in the atmosphere.

Collapse of the Core into the Lower Vessel Head—Invessel Steam Explosion

As a substantial fraction of the fuel-Zircaloy-structural steel mixture (called in the jargon of this arcane field "corium") becomes molten or liquefied, the dripping or collapse of this material into the lower head of the reactor vessel can occur. Because some water probably would remain in the lower head during the heatup of the core, interaction between the corium and the water is possible. This interaction could be as relatively innocuous as additional steam generation if the fuel mixture were to drip slowly into the water or could result in a highly energetic steam explosion if a large mass of molten corium were to interact rapidly and coherently with the residual water.

For the gradual (dripping) core slumping mode, the interaction between the corium and the water would probably not be severe. Individual small amounts falling into the water would cause some steam generation, and the mixture would be quenched (at least partially) in the process. With continued corium dripping, evaporation of all of the water in the lower head might eventually occur. After this time the remelted fuel would come directly into contact with the steel of the reactor vessel, with the result likely to be failure of the vessel.

For the case of large scale core collapse, the potential damage to the vessel would be more serious. The steam generation rate would be significantly greater, with some possibility of a highly energetic steam explosion. Research on the phenomena of such interactions has been under way to assess their likelihood and consequences.[219,220] These studies suggest that the most likely outcome of such a large scale collapse would be rapid steam generation, lacking the coherence needed for a more serious steam explosion.

In situations such as during the TMI-2 accident, when the postulated meltdown would have occurred with elevated RCS pressures (i.e., when the break size is not sufficient to allow RCS depressurization to the reactor building pressure), experimental and analytical evidence indicates that steam explosions are improbable.[220,221,222] Thus steam explosions of the magnitude of those postulated in the reactor safety study,[223] which were suggested as having the potential for causing reactor building failure, are considered highly unlikely.

An additional effect that is possible in the event of an invessel steam explosion is the release of ad-

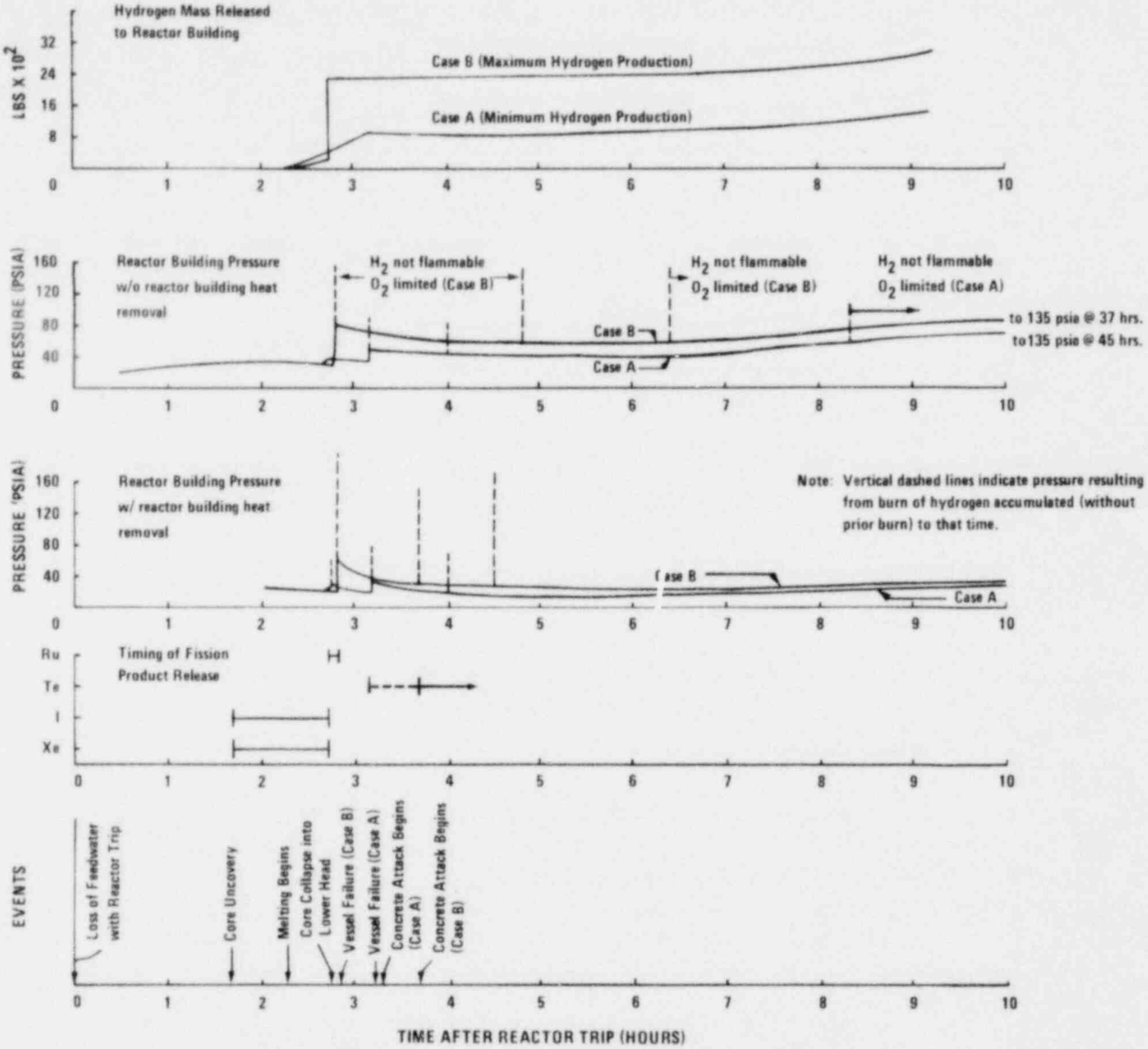


FIGURE II-36. Reactor Building Response

ditional fission products. Such an explosion would subject the molten fuel to a highly oxidizing environment, so that release of relatively low volatility elements (most notably ruthenium) could be expected.[224]

In Figure II-35 the column labeled "vessel steam explosion" indicates two possibilities: experiencing or not experiencing a steam explosion as the molten corium falls into the water in the lower head of the reactor vessel. Because of the significant difference in consequences, this distinction is based on having or not having a steam explosion of sufficient magnitude to cause reactor building failure. The most likely path of a postulated meltdown accident resulting from the TMI-2 accident is thus the lower path (no steam explosion), shown as the dashed line in Figure II-35.

Retention of the Molten Corium in the Lower Vessel Head

As the corium falls and collects in the lower vessel head, it could take two forms: a rubble bed with fragmented pieces of varying size or a molten pool of fuel, Zircaloy, and steel. The form actually taken depends on the amount of water in the lower head and the way in which the molten material falls into the head (i.e., dripping of small amounts or collapsing of large amounts).

In the situation where corium is presumed to drip in relatively small amounts into the lower head, some quenching and fragmentation could be expected. While water remains in the lower head, a rubble bed could be formed. The ability of such a rubble bed to cool and its effect on the underlying vessel steel are matters of considerable uncertainty. Experimental work dealing with the ability of rubble beds to cool in liquid sodium provides some insight into this question,[225] but a definitive answer is not available. Further, eventual evaporation of the water could be expected to result in remelting of the corium. The subsequent effects on the vessel steel make it relatively unlikely that vessel integrity would be maintained.

In the circumstance where a large amount of molten corium is presumed to collapse into the lower vessel head in a short time, the potential for quenching, fragmenting, and substantial cooling of the material is less. Some temporary quenching and fragmenting of the fuel–Zircaloy–steel mixture would likely occur; however, with the evaporation of the residual water and the collection of a large amount of corium in the lower head, remelting of the mixture could be expected. Under these circumstances, the lower vessel head would likely fail.

In an accident such as that at TMI-2, where high RCS pressures were maintained, the mechanical loading applied to the lower head due to this pressure would compound the thermal loadings imposed by the molten material, making structural failure of the lower vessel head essentially certain. For this reason, the "most likely" path shown in Figure II-35 indicates that retention of the fuel–Zircaloy–steel mixture in the lower vessel head would not occur.

The time elapsing before failure of the reactor vessel depends on the extent of metal–water reaction occurring as the molten core falls into the water in the lower head. If all of the remaining Zircaloy is assumed to react at this time, the additional energy releases increases the loading on the head, so that failure could occur within a few minutes. If no additional metal–water reaction occurs at this time, this additional energy is not released, so head failure could take somewhat longer, i.e., about 25 minutes (see Table II-58).

Collapse of the Corium Mixture into the Reactor Cavity

With the failure of the reactor vessel lower head, the corium mixture (containing by now additional molten steel) would fall into the reactor cavity. At this time, corium interaction with water would be possible, either resulting from the corium falling into water collected in the cavity or from the discharge of core flood tank water onto the top of the corium. The latter would be possible in some types of meltdown accidents (like that presumed here) when RCS pressures do not decrease below the discharge setpoint of the tanks until failure of the lower head occurs.

The consequences of a corium–water interaction can vary depending on the method of interaction between the two materials and the extent of fragmentation that may occur; therefore a spectrum of results can occur.

At one end of the spectrum is the case of core flood tank discharge on top of the corium mixture in the cavity. In this instance, fragmentation could be expected to be relatively minor. The water on the top of the corium could be evaporated by film boiling, while penetration of the base mat is under way beneath the corium.

In the middle of the spectrum, the corium–water interaction would result in the quenching and fragmentation of the corium into pieces of intermediate size. This condition then could result in rapid steam generation and a significant increase in reactor building pressure. Some time would be required for remelting of the mixture, so that penetration of the base mat could be delayed somewhat.

At the extreme end of the spectrum of possible events, the quenching of the mixture could cause fragmentation of such magnitude that a steam explosion could result. Under these circumstances, quenching and fragmentation of the corium into very small particles and very rapid generation of steam would occur, resulting in a large pressure pulse in the reactor building. Possibly the quenched and fragmented mixture would not reheat sufficiently to achieve melting temperatures, but it is also possible that it would remelt and begin penetration of the concrete base mat.

The most likely path of the accident progression at this juncture would be the intermediate case discussed. That is, quenching and fragmentation of the corium mixture into intermediate size particles would be expected, with the resultant pressure increase of intermediate magnitude in the reactor building. Because the more severe case of a steam explosion requires a more substantial (and thus less likely) fragmentation of the corium, the most likely accident progression shown in Figure II-35 follows the choice path of no incavity steam explosion.

In Figure II-36, reactor building pressure resulting from minimal fragmentation and fragmentation into roughly 2-inch diameter fragments are shown; pressures just after the collapse into the cavity are indicated as about 31 and 65 psia, respectively. Neither of the pressures by itself would be expected to result in reactor building failure.

Hydrogen Burning at the Time of Vessel Failure

The consideration of hydrogen burning is included here because it is at the time of vessel failure that reactor building integrity would be potentially most threatened. A combination of effects occurring in this time period would have some potential to seriously challenge the reactor building. To do this, most of the Zircaloy in the core would have had to be chemically reduced, and large amounts of hydrogen produced. The hydrogen released to the reactor building before vessel failure would have to remain unburned until this time. Significant fragmentation of the corium mixture as it interacts with water in the reactor cavity would also be required, so that substantial steam generation would occur.

Figure II-36 shows the calculated reactor building response to two particular combinations of hydrogen burning and steam generation. Case A in the figure represents a lower bound to the combined effects; that is, no metal–water reaction is presumed to occur as the core collapses into the lower vessel head, and no significant fragmentation of the corium mixture is assumed as it falls into the reactor cavity. Case B represents an upper bound for combined effects of these phenomena. Total reaction of the Zircaloy is assumed, as is significant fragmentation of the corium as it falls into the reactor cavity.

In Case A, steam pressures would increase to about 30 psia at time of vessel failure. If concurrent burning of all the hydrogen released to that time also occurred, as indicated by the dashed vertical lines, building pressures would increase to about 75 psia. Because the building failure pressure is expected to be about 135 psia,[216] building failure at the time of vessel failure would be unlikely for Case A.

In Case B, the calculated reactor building pressure increase would be more severe. Building pressure would increase to about 60 psia because of rapid steam generation. Burning of the large amount of hydrogen released up to that time could cause an additional increase of up to 100 psi. Thus, if these events were to occur concurrently, the building could fail.

Because the severity of the pressure increase calculated for Case B is primarily due to hydrogen burning, the likelihood of experiencing such a burn must be addressed. As noted, Case B is based on the reaction of all the Zircaloy in the core. The calculations indicate that about 40% of the Zircaloy would have reacted prior to collapse of the core into the lower head, the remaining 60% reacting during the corium–water interaction in the lower vessel head.[216] To react this amount of Zircaloy in the lower head, fragmentation of the corium into very small particles (about 10 mils or 0.010 inches in diameter) is required.[216] Although the size of fragments that actually would result from such an interaction is uncertain, it is unlikely that fragmentation into such very small particles would occur. Fragmentation into larger particles, which is more likely, would reduce the resulting pressure increase because of hydrogen burning; as such, building failure would be unlikely. For this reason, the "most likely" path shown in Figure II-35 indicates that hydrogen burning of sufficient magnitude to cause overpressurization failure of the reactor building would not be expected.

Availability of Reactor Building Engineered Safety Features

Throughout the course of this postulated meltdown accident, steam, fission products, hydrogen, noncondensible gases, and other materials would be released to the reactor building atmosphere. The capability of the building to withstand these insults depends on the functioning of the reactor building engineered safety features (ESFs): the reactor building spray system and the reactor building air cooling system. The former injects chemically treated water into the building atmosphere, provides some cooling capability at early times, and removes radioactive material from the atmosphere. The latter uses fans to force the building atmosphere across coils containing chilled water, and thus provides both short and long term cooling.[226]

Figure II-36 indicates that the operation or failure of the reactor building ESFs does not significantly affect the likelihood of building failure in the period up to and including the time of vessel failure. However, in the longer term, failure of the building air cooling system could lead to failure of the building by overpressurization. For the particular accident discussed, failure of the building coolers is predicted to result in building failure about 1½ to 2 days after the beginning of the accident (presuming no restoration of building cooling is possible). Natural deposition processes could be expected to reduce the amount of radioactive material in the building atmosphere over such long time periods, so that a long term overpressurization would be expected to result in greatly reduced consequences compared with building failure early in the accident. The differing outcomes from having or not having the reactor building ESFs are therefore shown in Figure II-35 to demonstrate the effect that the features can have on the integrity of the reactor building and the resulting consequences of the accident. During the first day of the TMI-2 accident, these safety features were known to have automatically actuated and operated successfully;[227] as such the "most likely" accident path in Figure II-35 indicates that the features would be available.

Reactor Building Base Mat Penetration

As discussed, failure of the reactor vessel would allow the molten corium mixture to fall into the reactor cavity and any water present in it. Temporary quenching and cooling of the corium mixture by the water might be expected; however, eventual remelting of the mixture seems likely upon evaporation of the water (and if no continuous supply of water is available to provide cooling). Thus, at this point, interaction between the molten corium and the concrete of the reactor building base mat would be expected to begin.

As the concrete beneath the corium would begin to rise in temperature, decomposition of its material also would begin. This decomposition would result in the generation of noncondensible gases like carbon dioxide, water vapor, and other materials that flow around and through the corium mixture. Within the corium itself, the water vapor and carbon dioxide would be chemically reduced, oxidizing materials such as steel and the fission product tellurium (enhancing its potential for escaping into the reactor building atmosphere) and releasing hydrogen and carbon monoxide gas into the reactor building atmosphere. Other former constituents of the concrete such as calcium and silicon would also enter the corium mixture, diluting it and altering its chemical composition. The released hydrogen would mix with other noncondensible gases released from the concrete and cause pressure increases and possible additional hydrogen combustion in the reactor building atmosphere. This situation is shown in Figure II-36 by the long term, gradual increase in the total amount of hydrogen released to the reactor building and the building pressure. Experimental evidence[228] suggests that hydrogen burns as it is produced during base mat penetration, so that accumulation of unburned hydrogen after the penetration begins would, under certain circumstances, be unlikely.

As the corium–concrete interaction continues, the decomposition of additional concrete would result in further noncondensible gas generation and dilution of the corium mixture. The combined effect of the reduction in the corium power density (caused by the dilution effect), the expenditure of energy to decompose the concrete and other energy losses (such as radiative and convective cooling) would result in a gradual cooling of the corium mixture. Thus, as the interaction continues, the rate of base mat penetration and the mixture temperature would decrease. This effect is apparent in Figure II-37, which shows that as time progresses the rate of penetration would gradually decrease.

The timing associated with base mat penetration is indicated in Table II-58. Initial penetration could begin almost immediately after collapse of the corium into the reactor cavity if no significant quenching and fragmenting occurs. If quenching occurs, delays in initial penetration on the order of 1 hour are possible. Initial penetration would be relatively slow in either case, as Figure II-37 shows. This initial slowness results from the relatively low temperatures of the corium; within 1 to 2 hours, internal heating of the mixture could bring it to temperatures where decomposition of the concrete is more rapid. The possibility of the corium mixture penetrating through the entire depth of the base mat is clearly dependent on the rate of cooling of the mixture. For the particular case of the TMI-2 base mat, solidification of the corium mixture would be likely (but not ensured) before complete penetration occurs. For

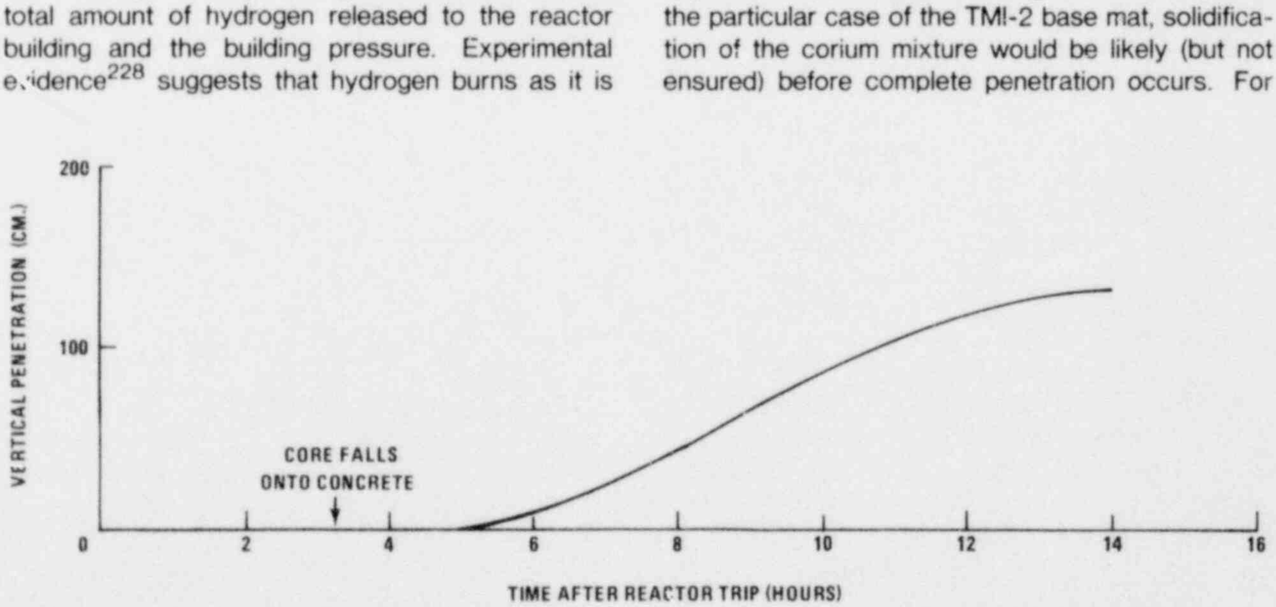


FIGURE II-37. Penetration of the Reactor Building Base Mat (Timing Related to Case A in Figure II-36) this reason, the most likely path shown in Figure II-35 indicates that complete penetration of the reactor building base mat would not have occurred.

In the particular case of TMI-2, the bottom of the base mat is directly in contact with bedrock. If the corium mixture were to penetrate the base mat completely, it would then begin penetration of the bedrock, where solidification would be a virtual certainty. This bedrock could also act as an effective block against transport of radioactive material, mitigating possible releases of this material into the surrounding environment.

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[99]NRC, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director, NRR Staff," NUREG-0138, November 1976, at 4-1 through 4-11.

[100]Hearing before the Committee on Government Operations, U.S. Senate, 94th Cong., 2nd Sess. (Dec. 13, 1976) at 260–262.

[101]Met Ed, "TMI-2 Emergency Procedure 2202-1.3—Loss of Reactor Coolant/Reactor Coolant System Pressure, Sec. 3.4, Rev. 11, October 6, 1979.

[102]Faust, Frederick, Scheimann, and Zewe dep. at 99.

[103]Memorandum from R. C. DeYoung, NRC, to L. V. Gossick, NRC, "Potential 10 CFR Part 21 Violations by B&W," Enclosure 1, September 10, 1979.

[104]Met Ed, "TMI-2 Emergency Procedure 2202-1.3—Loss of Reactor Coolant/Reactor Coolant System Pressure, Sec. B, Rev. 11, October 6, 1978.

[105]Memorandum from H. Denton, NRR, to Hendrie, et al., NRC, "Response to Questions Raised by Congressman Udall," October 4, 1979.

[106]Letter from J. H. Taylor, B&W to R. J. Mattson, NRC, Subject: Operating Procedure Guidelines for Small Breaks, dated May 5, 1979.

[107]Memorandum from H. Denton, NRR, to NRC Commissioners, "Interim Report on Sensitivity Studies of the B&W Reactor Design," Enclosure C, October 25, 1979.

[108]Memorandum from R. C. DeYoung, NRC, to L. V. Gossick, NRC, "Potential 10 CFR Part 21 Violations by B&W," Enclosure 1, September 10, 1979.

[109]Met Ed, "TMI-2 Emergency Procedure 2202-1.3—Loss of Reactor Coolant/Reactor Coolant System Pressure, Sec. B, Rev. 11, October 6, 1978.

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[111]Memorandum from R. C. DeYoung, NRC, to L. V. Gossick, NRC, "Potential 10 CFR Part 21 Violations by B&W," Enclosure 6, September 10, 1979.

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[113]Memorandum from R. C. DeYoung, NRC, to L. V. Gossick, NRC, "Potential 10 CFR Part 21 Violations by B&W," Enclosure 9, September 10, 1979.

[114]NRC, "Investigation into the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement," NUREG-0600, Sec. 1.3.2.3, August 1979.

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[117]Met Ed Specification 2555-1.5, "Condensate Polishing Plant," Amendment No. 1, at II-12.

[118]Met Ed Calibration Data Sheets, TMI-2, Instrument Nos: WT-PS-7410, 7412, 7414, 7416, 7418, 7420, 7422, and 7424.

[119]Met Ed photographs of MTX 24.2 drawings depicting electrical system tested.

[120]Met Ed MTX 24.6, Condensate Polishing System Functional Test.

[121]Photographs of Condensate Polisher Cabinets 1 through 8 with disconnected wiring.

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[123]TMI-2 Work Requests Nos: 0027, 0028, 0036, 0037, 0039, 0172, 0173, 02[illegible]9, 0282, 0335, 0204, 0436, 0468, 0478, 0488, 490, 495, 496, 500, 602, 730, 890, 956, 957, 1053, 1076, 2151, and 1296.

[124]Met Ed MTX 24.6, Condensate Polishing System Functional Test.

[125]Memorandum from M. J. Ross, Met Ed, to G. P. Miller and J. L. Seelinger, Met Ed, "Water in the Instrument Air Lines at the Condensate Polisher Control Panel and Regeneration Skid Resulting in a Loss of Feedwater Condition in Unit No. 2 on October 19, 1977," November 14, 1977.

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[128]Memorandum from W. H. Zewe, Met Ed, to J. Seelinger, "Water in Service and Instrument Air," May 15, 1978.

[129]Memorandum from J. Seelinger, Met Ed, to J. Brummer, GPU, "Water in IA," May 16, 1978.

[130]TMI-2 Reactor Trip Report, dated November 3, 1978.

[131]TMI-2 Operating Procedure 2106-2.2, "Condensate Polishing System," Rev. 9, para. 4.1.4, December 21, 1978.

[132]TMI-2, The Condensate Polisher Operator's Log Book for the Period March 27-28, 1979, at 383.

[133]NRC, "Investigation into the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement," NUREG-0600, Appendix I-A, item 4, August 1979.

[134]*Id.* at I-2-4 and 5.

[135]TMI-2, Differential Pressure Recorder 89, SC-0401, March 28, 1979, File No. 62-0070-404-89-00, Reel OPCP-2-818.

[136]NRC, "Investigation into the March 28, 1979 Three

Mile Island Accident by Office of Inspection and Enforcement," NUREG-0600, Appendix I-A, item 14, August 1979.

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[138]Faust, Frederick, Scheimann, and Zewe dep. at 170–171.

[139]Letter from G. F. Trowbridge; Shaw, Pittman, Potts, and Trowbridge; to E. Case, NRC, Subject: TMI Staff Interviews, dated May 7, 1979, Scheimann interview at 5.

[140]*Id.*, Zewe interview at 3.

[141]Met Ed, "Final Safety Analysis Report, Three Mile Island Nuclear Station-Unit 2," Sec. 10.4.7.

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[143]NRC, "Investigation into the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement," NUREG-0600, August 1979, at 4.1.1.

[144]Letter from G. F. Trowbridge; Shaw, Pittman, Potts, and Trowbridge; to E. Case, NRC, Subject: TMI Staff Interviews, dated May 7, 1979, Zewe interview at 3.

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[146]*Id.* at IA-6.

[147]Letter from G. F. Trowbridge; Shaw, Pittman, Potts, and Trowbridge; to E. Case, NRC, Subject: TMI Staff Interviews, dated May 7, 1979, Faust interview at 8.

[148]*Id.*, Zewe interview at 2.

[149]Electric Power Research Institute, "Nuclear Safety Analysis Center, Analysis of Three Mile Island-Unit 2 Accident," NSAC-1, Appendix STEAM DUMP, July 1979.

[150]Met Ed, "Final Safety Analysis Report, Three Mile Island Nuclear Station Unit 2," at Sec. 7.0, Docket 50-320.

[151]IEEE Standard No. 279-1968.

[152]NRC, "Three Mile Island Unit 2 Technical Specifications," NUREG-0432, 1978.

[153]Met Ed, "Three Mile Island Nuclear Station Unit 2 Surveillance Procedure 2302-R28," July 1978.

[154]Power Reactor Development Company, Enrico Fermi Atomic Power Plant, "Report on the Fuel Melting Incident in the Enrico Fermi Atomic Power Plant on October 5, 1966," December 15, 1968.

[155]Atomic Power Development Associates, Inc., "Design and Capabilities of the Malfunction Detection Analyzer Installed on the Enrico Fermi Fast Breeder Reactor," July 1969.

[156]Electric Power Research Institute, "On-Line Power Plant Alarm and Disturbance Analysis System," EPRI-NP-613, Project 891, Interim Report, February 1978.

[157]NRC, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," Reg. Guide 1.97, Rev. 2. (Proposed).

[158]Zewe, Scheimann, Frederick, and Faust interview (IE) (April 3, 1979) Tape 128 at 17.

[159]Faust interview (IE) (March 30, 1979) Tape 145 at 13.

[160]American National Standard, "Functional Requirements for Post Accident Monitoring Capability for the Control Room Operator of a Nuclear Power Generating Station," ANS-4.5 (Draft), September 1979.

[161]Note on Timing Problems—Time marks on most of the strip charts can be placed with no more than about ±3 minutes accuracy, even though sudden changes in the recorded parameter were fitted to similar changes in other parameters to ±1 minute or better. Better matching is not possible, since the accuracy of the chart drives is not known, while "fits" between neighboring "accurate event" time points may be several feet apart on charts having nominal speeds of 4 to 8 inches per hour. In addition, the same signal from one sensing instrument recorded on two separate data acquisition systems was in one case displaced approximately 63 seconds at the start of the accident (03:59:33 a.m. and 04:00:36 a.m.) and approximately 2 minutes about 10 hours later (02:36:20 p.m. and 02:38:14 p.m.—both values extrapolated). This means that not only were the internal clocks of the two data acquisition systems indicating different times but they also had different rates. These data are contained in channel 390 of the utility typer and channel MUX-2 of the reactimeter. They are plotted in Color Plate IV for times after 02:30 p.m. The signal calibrations and setting accuracies of the strip chart recording instruments on March 28, 1979, are not known exactly, but the results of the most recent calibration have been summarized in Section II.C.1.d. The errors may be as large as 5%, as the wide-range chart for the reactor coolant system pressure records a pressure of 490 to 495 psig at 13 hours 28 minutes and channel 398 of the utility typer reports a pressure of 445 psig at that time. Also, the wide-range reactor coolant pressure chart indicates a pressure of 2200 psig at 10 hours 13 minutes while the reactimeter reports a pressure of 2145 psig at that time.

[162]TMI Control Room Logs March 28, 1979 (NRC Reel OPS-2-801.2068).

[163]Mehler dep., October 11, 1979, at 5–9.

[164]TMI Plant Strip Charts: By name—OTSG and Primary System Temperatures, March 21, 1979 to April 4, 1979, SC-0043 Recorder 10 (NRC Reel OPCP-2-803).

[165]Letter from G. F. Trowbridge; Shaw, Pittman, Potts, and Trowbridge; to E. Case, NRC, Subject: TMI Staff Interviews, dated May 7, 1979, Zewe interview at 6, 8.

[166]Faust interview (IE) April 21, 1979, Tape 36 at 4–7.

[167]Chwastyk dep., Oct. 11, 1979, at 38–41.

[168]Letter from G. F. Trowbridge; Shaw, Pittman, Potts, and Trowbridge; to E. Case, NRC, Subject: TMI Staff Interviews, dated May 7, 1979, Zewe interview at 6.

[169]Met Ed, TMI-2 Final Safety Analysis Report (FSAR), Table 9.3.

[170]NRC, "Investigation into the March 28, 1979, Three Mile Island Accident" by the Office of Inspection and Enforcement, NUREG-0600 August 1979, at I-4-20.

[171]These 52 instrumentation tubes are reasonably representative of the 177 fuel assemblies in the core, and the fraction of them to which events are happening should be multiplied by 177/52 to estimate the behavior in the entire core.

[172]NRC, "NRC Special Inquiry, TMI-2: Accident Delineation," NUREG CR-1239 (Sandia Laboratories, SAN 80-0094), January 1980.

[173]Electrical Power Research Institute, "Analysis of Three Mile Island-Unit 2 Accident," NSAC-1, July 1979, Appendix CI.

[174]I. Porter interview, May 21, 1979 (tape 237) at 17–21.

[175] *Id.* at 18–21.

[176] Electrical Power Research Institute, Nuclear Safety Analysis Center, (EPRI), Palo Alto, Calif., research in progress.

[177] TMI Plant Strip Charts, Self-Powered Neutron Detector (SPND) Data, March 28, 1979, SC-0037, Recorder 23.

[178] R. O. Wooten, R. S. Denning, and P. Cybulskis, Battelle Columbus Laboratories, "Analysis of the Three Mile Island Accident and Alternative Sequences," NRC Report NUREG/CR-1219, December 19, 1979.

[179] See Appendix II-10.

[180] Electrical Power Research Institute, "Analysis of Three Mile Island Unit No. 2 Accident," NSAC-1, July 1979.

[181] Pres. Comm. "Technical Staff Analysis Report on Core Damage," October 1979.

[182] R. O. Meyer, calculation of Zircaloy oxidation from hydrogen burned.

[183] S. Hagan, et al., Projekt 4241, "Experimentalle Undersuchung dew Abschmenlsphize van UO_2 Zircaloy Brennelemeuten bei versagender Not Kuhlung," Projekt Nukleare Sicherheit Halbjahresbericht, 1977/2 Kfk 2600, pp. 416–428.

[184] Memorandum from M. L. Picklesimer, NRC, to File, "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," June 22, 1979.

[185] Memorandum from G. P. Marino and J. M. Marks, NRC, to File, "Calculation of Fuel Rod Temperatures Reached in the Three Mile Island-2 Incident," October 25, 1979.

[186] NRC, "Investigation into the March 28, 1979 Three Mile Island-2 Accident by the Office of Inspection and Enforcement," NUREG-0600, August 1979.

[187] A. P. Malinauskas, R. A. Lorenz, J. A. Collins, "Fission Product Release from Defected LWR Fuel Rods," 7th Water Reactor Safety Information Meeting, November 5–9, 1979.

[188] 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, on March 28, 1979," Memorandum for file, NRC June 1979.

[189] D. M. Haaland, "Release of Radioactivity from the Core," Chapter 7 of *Core Meltdown Experimental Review*, SAND 74-0382, Sandia Laboratories, Albuquerque, N.M., August 1975.

[190] I. Albrecht, V. Matschoss, H. Wild, Experimental Investigations of LWR-Core Material Release at Temperature Ranging from 1500 to 2800°C, 7th Water Reactor Safety Information Meeting, November 5–9, 1979.

[191] NRC, "Reactor Safety Study," Appendix VII, WASH-1400 (NUREG-75/014), October 1975.

[192] Y. B. Katayama and J. E. Mendel, "Leaching of Irradiated LWR Fuel Pellets in Deionized Water, Sea Brine, and Typical Ground Water," *ANS Trans*, 27:447, Nov.-Dec. 1977.

[193] Y. B. Katayama, *Leaching of Irradiated LWR Fuel Pellets in Deionized and Typical Ground Water*, BNWL-2057, Battelle Northwest Laboratories, Richland, Washington, July 1976.

[194] E. B. Eklund and R. Forsyth, *Lackning av bestraalad UO2—braens* (Leaching of irradiated UO_2 fuel), TPM-B, AB Atomenergie, Studsvik Sweden, February 1978 (in Swedish).

[195] Letter from D. E. Bennett, Sandia Laboratories, to J. Murphy, NRC (PAS), Subject: Sandia-ORIGEN code calculations, dated April 4, 1979.

[196] NRC, "Investigation into the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement," NUREG-0600, August 1979.

[197] Letter from Harold R. Denton (NRR) to Vincent L. Johnson, September 28, 1979.

[198] Letter from J. T. Collins (TMI-2 support) to J. Y. Lee (TMI-2 support), "Preliminary Analysis Results of TMI-2 Containment Building Water Samples," August 31, 1979.

[199] J. A. Auxier et al., "Report of the Task Group on Health Physics and Dosimetry to President's Commission on the Accident of Three Mile Island," October 1979, at 139–148.

[200] C. J. Hochanadel, "Effects of Cobalt Gamma Radiation on Water and Aqueous Solutions," *J. Phys. Chem.* 56:587–594, 1952.

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[202] J. R. Honekamp, S. Gordon, K. H. Schmidt, and D. J. Malloy, "An Analysis of the Hydrogen Bubble Concerns in the Three-Mile Island Unit-2 Reactor Vessel" undated, contribution to President's Commission Technical Staff Analysis Report on Chemistry.

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[205] M. L. Picklesimer, "Bounding Estimates of Damage to Zircaloy Fuel Cladding in the TMI-2 Core at Three Hours After the Start of the Accident, March 28, 1979," Memorandum for File, NRC, June 1979.

[206] R. K. Cole, Generation of Hydrogen During the First Three Hours of the Three Mile Island Accident, NUREG/CR-0913, (Sandia Laboratories, SAND 79-1357) July 1979.

[207] Letter (with attachments) from R. K. Cole, Sandia Laboratories, to M. Picklesimer, NRC-TMI-SIG, dated October 15, 1979.

[208] Met Ed, "Bubble Size Calculations," various dates.

[209] Letter from James H. Taylor, Manager of Licensing, B&W, to John Bickel, ACRS, Subject: "TMI-2 Incident, Gas Bubble Volume Estimate," dated July 20, 1979.

[210] Met Ed, "Final Safety Analysis Report, Three Mile Island Nuclear Station-Unit 2," Chap. 7.

[211] Memorandum for Files, T. E. Murley, "Record of Actions, Three Mile Island Accident, March 28–April 6," NRC (RES), June 1979.

[212] Appendix III.4, "Chronology of TMI-2 Hydrogen Bubble Concerns."

[213] Notes made of the time and later recollections indicate that the organizations contacted included several groups within NRC, GE, Westinghouse, Bettis, KAPL, SAI, B&W, ORNL, EG&G, INEL, Battelle, Sandia, AVCO, IBM, Mobil Oil, Lawrence Berkeley Laboratory, BNL, Picatinny Arsenal, Columbia University, and Exxon Research.

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[214]P. Cybulskis, R. G. Jung and B. D. Trott, "Hydrogen Explosion Analysis," attachment to letter from Richard S. Denning, Battelle Columbus Laboratories, to Mark Cunningham, NRC (PAS), dated May 22, 1979.

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[216]NRC, "Analysis of the Three Mile Island Accident and Alternative Sequences" (by Battelle Columbus Laboratories), NUREG/CR-1219 (in press), Sec. 6.0.

[217]NRC, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), October 1975, Appendix VII, Sec. 1.2.

[218]NRC, "TRAP-MELT Users Manual" (by Battelle Columbus Laboratories), NUREG/CR-0632, February 1979.

[219]NRC, "Molten Core/Water Contact Analysis for Fuel Melt Accidents" (by Sandia Laboratories), NUREG/CR-0391 (SAND 77-1842), February 1979.

[220]Argonne National Laboratories, "An Experimental Study of Combined Physical-Chemical Explosions in an Aluminum-Water System," ANL/RAS/LWR 79-2, April 1979.

[221]H. Kottowski, et al., "Importance of the Coolant Impact on the Violence of the Vapor Explosion," 4th CSNI Specialist Meeting on Fuel-Coolant Interactions in Nuclear Reactor Safety, April 1979.

[222]NRC, "Preliminary Analysis of the Containment Failure Probability by Steam Explosions Following a Hypothetical Core Meltdown in a LWR" (by Sandia Laboratories), NUREG/CR-1104, (in press).

[223]NRC, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," Wash-1400 (NUREG-75/014), October 1975, Appendix VII, Sec. 1.2.

[224]*Id.*, Appendix VII, Sec. 1.4.

[225]NRC, "Post-Accident Heat Removal: Debris Bed Experiments D-2 and D-3" (by Sandia Laboratories), NUREG/CR-0421, November 1978.

[226]Met Ed, "Final Safety Analysis Report (FSAR), Three Mile Island Nuclear Station, Unit 2," Sec. 6.2.2.

[227]NCR, "Investigation into the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement," NUREG-0600, August 1979, at I-4-47.

[228]NRC, "Sustained Molten Steel/Concrete Interactions Tests" (by Sandia Laboratories), NUREG/CR-0166, June 1978.

D ALTERNATIVE ACCIDENT SEQUENCES

In this section, a number of accident sequences are discussed that are somewhat different from the actual TMI-2 accident progression. These alternative sequences have been established and evaluated to address particular questions that arise from the Special Inquiry Group's (and other groups') investigations of the accident. None of these alternative sequences actually happened, but some of them could have, and others have much less probability of occurring. Each alternative has been evaluated here to provide insights into reactor performance and safety.

The section "Amelioration of Fuel Damage" discusses methods by which damage to the fuel could have been ameliorated and some reasons for the lack of success of the anticipated procedures. The "Analysis of Alternative Accident Sequences" section describes analyses performed that address specific questions concerning the effect of certain operator actions (or inactions) or equipment failures. This section addresses a number of "what if" questions such as, what if the operators had not reduced the high pressure injection system flow? Or, what if the pilot-operated relief valve (PORV) block valve had not been closed when it was? Only a few of the many possible "what ifs" have been evaluated. We have selected alternatives to analyze that we believe provide the greatest insights.

1. AMELIORATION OF FUEL DAMAGE

The integrity of the TMI-2 core was threatened primarily during the first 16 hours of the accident. The majority of the damage was done between 2 and 4 hours into the accident; the remainder of the 16 hours was then spent in attempts to recover from this damage. As discussed in Section II.C.2, the actual time of the final refilling of the core region is a point of controversy. Because of this controversy, the effectiveness of the various core cooling methods utilized between 4 and 16 hours also remains unresolved. However, apparently there are some methods of cooling that were not used that would have been more likely to succeed. These are discussed in this section.

Early PORV Block Valve Closure

The PORV is located at the top of the pressurizer and is accompanied by an upstream block valve. The discharge line temperature downstream of the PORV (which provides information on the valve status, stuck-open or closed) was obtained from the plant computer a number of times between the beginning of the accident and when the block valve was closed at 2.3 hours; the first time was at 25 minutes. Had the block valve been closed at that time, the loss of coolant would have stopped before significant amounts of water were lost. During similar incidents involving stuck-open PORVs (Davis Besse on September 24, 1977, Oconee 3 on June 13, 1975),[1] block valve closure occurred during roughly comparable time periods. Because no fuel damage resulted from these similar events, it can be stated reasonably that no fuel damage would have resulted at TMI-2 had the block valve been closed at 25 minutes.

Computer analysis has been performed for the Special Inquiry Group (SIG) to evaluate further the effect of block valve closure at 25 minutes. This analysis, discussed in greater detail in "Alternative Accident Sequence 5, PORV Block Valve Closure at 25 Minutes," Section II.D.2.f, also indicates that valve closure at 25 minutes would have stopped the accident before serious damage to the fuel began.[2,3]

By the time the operators made subsequent requests for discharge line temperatures, additional coolant had been lost from the reactor coolant system. After some time period, this loss of coolant would have become sufficiently great that closure of the block valve would not, in itself, have reversed the deteriorating situation. In these cases, the use of the high-pressure injection system in one of the modes discussed below would also have been necessary.

Use of the High-Pressure Injection System

Effective use of the high-pressure injection (HPI) system would have provided, and eventually did help provide, the means to cool down the reactor coolant system (RCS) and the core. Several modes of HPI system use, by itself or in conjunction with other systems, were possible. These include both of the following modes.

Continuous HPI System Flow at High Flow Rates

The HPI system was automatically actuated a number of times during the first 16 hours of the accident, as discussed in Section II.C.2. Because of the continued reliance on pressurizer level instrumentation by the operating crew, the flow rate from the system was substantially reduced following each actuation.

Operation of the HPI system at its full flow rates would have repressurized the RCS and refilled it with coolant. With the RCS again filled with water (along with some pockets of noncondensible gas after 2 to 3 hours into the accident), a flow path from the HPI system into the RCS and out the PORV and safety valves would have been established. In this mode of HPI system use, heat removal from the RCS should have been achieved by the heatup of water as it passed through the RCS. After depletion of the normal source of water from the HPI system (the borated water storage tank), a flow path from the reactor building emergency sump by using the water lost out the PORV and safety valves could have been established. This method of cooling was not attempted on the first day of the accident apparently because of a failure to recognize that a loss-of-coolant accident was occurring.

Use of the HPI System in Conjunction with Reactor Coolant Pump Operation

Between 100 minutes and 16 hours, all reactor coolant pumps (RCPs) were off (with two brief exceptions), so that forced flow cooling of the core was not occurring. Restart of one of these pumps would have reestablished forced flow cooling to the core, with heat removal being achieved through the once-through steam generators (OTSGs). However, attempts to restart an RCP in this time period met with limited success apparently because of low water inventories and pressures in the RCS. Use of the HPI system in support of the restart of an RCP may have provided the needed additional coolant and pressure. Thus the combination of HPI system use and restart of an RCP should have been, and eventually was, successful in cooling down the reactor core.

Between the 4- and 16-hour period, RCS pressure was increased to over 2000 psi twice, once at about 5 to 6 hours and maintained for over 2 hours and again at about 14 to 15 hours. During the latter repressurization, a reactor coolant pump was started, providing the long term, stable method of cooling. There is no evidence that attempts to start a reactor coolant pump were made during the earlier repressurization.

The emergency procedure to be followed during a loss-of-coolant accident due to a small break in the RCS is to allow the automatically actuated high pressure injection system to operate. For a break such as a stuck-open PORV, the HPI would restore RCS pressure and coolant inventory and maintain core cooling. When RCS pressure and pressurizer level are restored to specific levels (as defined in the emergency procedures), HPI system flow is supposed to be decreased by valve manipulation. This anticipated procedure apparently was not followed for a number of reasons. We believe that these reasons include:

- failure of operators to recognize that the PORV was stuck open and thus that a loss-of-coolant accident was occurring;
- pressurizer level indication was misleading; and
- lack of understanding by operators on how to recover from such an accident, once recognized.

2. ANALYSIS OF ALTERNATIVE ACCIDENT SEQUENCES

a. Introduction and Summary

The analysis of a set of alternative accident sequences has been undertaken as part of the Special Inquiry Group's work. The purposes of this analysis are:

- to assess specifically the importance of various equipment failures and/or human actions (or inactions);
- to provide additional information on the physical phenomena occurring during the accident; and
- to aid in the assessment of how close this accident was to becoming a "core meltdown" accident.

To assist in the evaluation of certain alternative accident sequences, computer analyses were performed at the following locations:

- MARCH code calculations at Battelle Columbus Laboratories;
- RELAP code calculations at the Idaho National Engineering Laboratory; and
- TRAC code calculations at the Los Alamos Scientific Laboratory.

The MARCH code is a relatively simplistic code that models the progression of core meltdown accidents, including reactor coolant system thermal-hydraulics, fuel heating, melting, and collapse, containment base mat penetration, and containment pressure and temperature response. The RELAP and TRAC codes are more sophisticated thermal-hydraulic codes used to analyze design basis accidents, employing multidimensional modeling of coolant flow (as liquid water, steam, and mixtures) and calculating detailed fuel temperature profiles. The results of these calculations are discussed in the following sections. Detailed results may be found in Refs. 2, 3, and 4.

The method by which most of the alternative sequences were determined was through the combination of critical parameters in an "event tree" logic. Such a tree displays the progression of the early portion of the accident (the first few hours) in terms of system operations and human actions. Variations in these parameters are displayed as "branches" in the event tree; thus, any variation becomes a different branch, or alternative accident sequence, in the overall event tree.

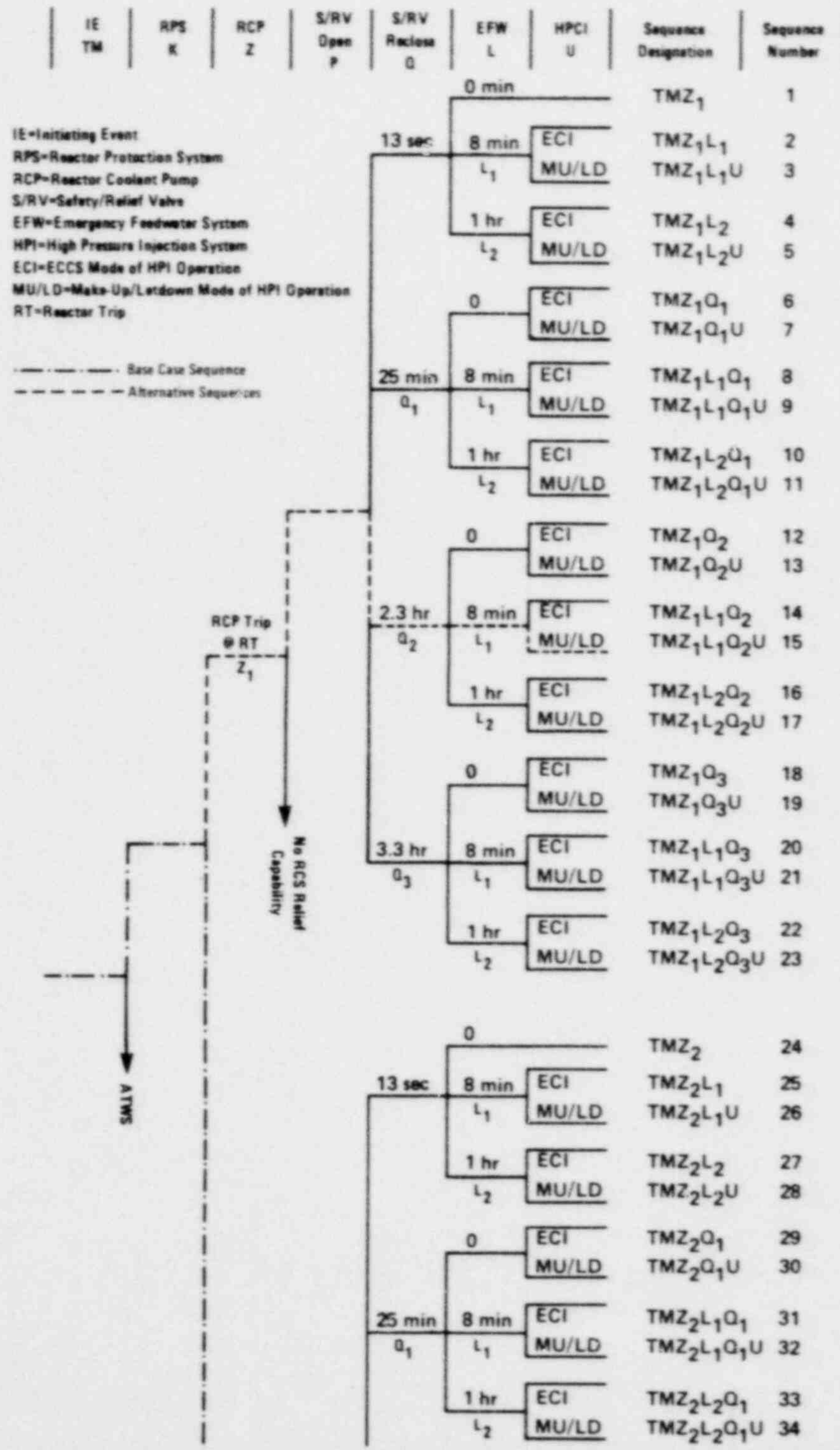
The progression of the early portion of the TMI-2 accident is shown in Figure II-38. Four parameters, all of which were related to human actions, were identified as critical to this progression. The four parameters chosen were:

- timing and method of tripping the reactor coolant pumps;
- timing of closure of the PORV or its block valve;
- timing of initial delivery of emergency feedwater system flow to the steam generators; and
- flow rate delivered from the high pressure injection system.

For each of these parameters, a number of alternative values were chosen. With respect to the timing and method of tripping the two pairs of reactor coolant pumps, three variations were chosen: (1) total pump trip concurrent with reactor trip, (2) one of the two pumps in each loop tripped at 73 minutes, and (3) B-loop pump trip at 73 minutes followed by A-loop pump trip at 100 minutes. Case 1 relates to the tripping of all four of the reactor coolant pumps very early in the transient at the time of reactor trip. Case 2 relates to the possibility of prolonged pump operation from selective tripping of one pump in each loop. Case 3 is the base case; that is, the actual timing of pump tripping during the accident.

Four variations in the time of closure of the PORV or the PORV block valve were defined. Times of closure were (1) 13 seconds, (2) 25 minutes, (3) 2.3 hours, and (4) 3.3 hours. Case 1 relates to the normal timing of PORV closure following an interruption in flow from the main feedwater system and subsequent reactor trip. Case 2 relates to the timing of the first operator request for printout of the PORV discharge line temperature from the utility printer. Case 3 is the base case; that is, the actual time of PORV block valve closure. Case 4 adds an additional 1-hour delay in closure of the PORV block valve beyond that actually experienced.

Three variations in the timing of initial emergency feedwater (EFW) flow into the steam generators were analyzed. Times of delivery were (1) 40 seconds, (2) 8 minutes, and (3) 1 hour. Case 1 relates to the normal time of EFW flow initiation into the once-through steam generators (OTSGs), had the discharge line block valves not been closed. Case 2 is the base case time of EFW delivery to the OTSGs. Case 3 relates to a delay in opening of the block valves to 1 hour rather than 8 minutes.


IE TM
RPS K
RCP Z
S/RV Open P
S/RV Reclose Q
EFW L
HPCI U
Sequence Designation
Sequence Number
IE=Initiating Event
RPS=Reactor Protection System
RCP=Reactor Coolant Pump
S/RV=Safety/Relief Valve
EFW=Emergency Feedwater System
HPI=High Pressure Injection System
ECI=ECCS Mode of HPI Operation
MU/LD=Make-Up/Letdown Mode of HPI Operation
RT=Reactor Trip
Base Case Sequence
Alternative Sequences
RCP Trip @ RT Z_1
No RCS Relief Capability
ATWS
13 sec
25 min Q_1
2.3 hr Q_2
3.3 hr Q_3
0 min
0
8 min L_1
1 hr L_2
ECI
MU/LD
TMZ_1 1
TMZ_1L_1 2
TMZ_1L_1U 3
TMZ_1L_2 4
TMZ_1L_2U 5
TMZ_1Q_1 6
TMZ_1Q_1U 7
$TMZ_1L_1Q_1$ 8
$TMZ_1L_1Q_1U$ 9
$TMZ_1L_2Q_1$ 10
$TMZ_1L_2Q_1U$ 11
TMZ_1Q_2 12
TMZ_1Q_2U 13
$TMZ_1L_1Q_2$ 14
$TMZ_1L_1Q_2U$ 15
$TMZ_1L_2Q_2$ 16
$TMZ_1L_2Q_2U$ 17
TMZ_1Q_3 18
TMZ_1Q_3U 19
$TMZ_1L_1Q_3$ 20
$TMZ_1L_1Q_3U$ 21
$TMZ_1L_2Q_3$ 22
$TMZ_1L_2Q_3U$ 23
TMZ_2 24
TMZ_2L_1 25
TMZ_2L_1U 26
TMZ_2L_2 27
TMZ_2L_2U 28
TMZ_2Q_1 29
TMZ_2Q_1U 30
$TMZ_2L_1Q_1$ 31
$TMZ_2L_1Q_1U$ 32
$TMZ_2L_2Q_1$ 33
$TMZ_2L_2Q_1U$ 34

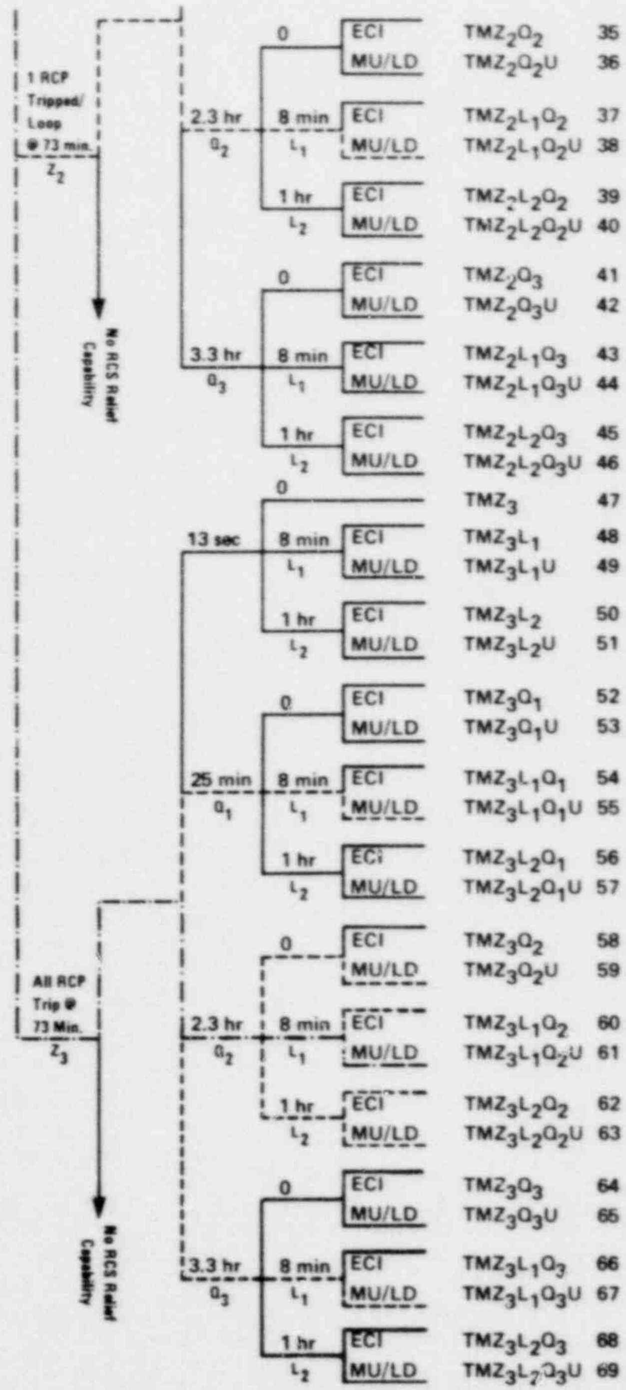


FIGURE II-38. Event Tree for Parameters Critical to Early Accident Progression

Two variations in the flow rates from the high-pressure injection (HPI) pumps were examined. These cases were (1) full HPI flow rate after actuation, and (2) degraded HPI flow rates. Case 1 relates to the functioning of the HPI system without interference to throttle flow rates. Case 2 involves the base case flow rates as actually experienced because of operator actions to reduce this flow.

Figure II-38 displays possible variations in the progression of events following the TMI-2 accident initiating event (i.e., the interruption of main feedwater flow) resulting from the parametric variations discussed above. From all of the possible accident sequences shown in Figure II-38, a set of nine sequences was chosen to examine the effects of variations in the four specific parameters. These nine cases are described in the following discussion.

In addition to the alternative accident sequences based on the four parameters critical to the early progression of the accident, a number of alternative sequences related to other specific concerns have been addressed.

Base Case Accident Sequence

Accident sequence 61 illustrated in Figure II-38 is the sequence of events that actually occurred during the early portion of the TMI-2 accident. This sequence has been studied with the use of various computer codes to provide a basis to which the alternative accident sequences could be compared and to assist in the overall understanding of the accident. The MARCH, RELAP, and TRAC codes have all been used to recreate the base case accident sequence, with RELAP and TRAC being used to analyze the time period of 0 to 2½ hours and MARCH the time period of 0 to 16 hours. Detailed discussions of these analyses may be found in Refs. 2, 3, and 4.

Alternative Accident Sequence 1

Accident sequence 60 in Figure II-38 has been analyzed as alternative accident sequence 1. All parameters remain the same except that the high-pressure injection system is allowed to operate at full flow rates rather than in the degraded mode resulting from operator actions to throttle back flow. This analysis shows the effect of the operator decision to throttle back the HPI flow. Detailed results of the analysis of this sequence may be found in Section II.D.2.b.

Alternative Accident Sequence 2

Accident sequence 62 in Figure II-38 has been analyzed as alternative accident sequence 2. Two parameters are varied, so that the high-pressure injection system operates at full flow rates (rather than throttled back) and the delivery of emergency feedwater is delayed from the actual 8 minutes until about 1 hour into the accident. This sequence addresses the capability of the HPI system to provide adequate core cooling without heat removal through the steam generators. Detailed results of the analysis of this sequence may be found in Section II.D.2.c.

Alternative Accident Sequence 3

Accident sequence 59 in Figure II-38 has been analyzed as alternative accident sequence 3. Only one parameter is varied, this being the time of delivery of emergency feedwater flow. In this sequence, EFW is assumed to be delivered beginning at about 40 seconds, which would have happened if the EFW discharge line block valves had not been closed. This analysis shows the effect of these block valves being closed until 8 minutes into the accident. Detailed results of the analysis of this sequence may be found in Section II.D.2.d.

Alternative Accident Sequence 4

Accident sequence 63 in Figure II-38 has been analyzed as alternative accident sequence 4. In this sequence, only the time of opening the EFW system discharge line block valves is varied. Delivery of EFW to the OTSGs is assumed to begin at 1 hour into the accident, rather than at 8 minutes. This alternative sequence shows the effect of a more prolonged unavailability of emergency feedwater resulting from the prolonged closure of the discharge line block valves. Detailed results of the analysis for this sequence may be found in Section II.D.2.e.

Alternative Accident Sequence 5

Accident sequence 55 in Figure II-38 has been analyzed as alternative accident sequence 5. In this sequence, only the time of PORV block valve closure is varied. Time of closure is assumed to be 25 minutes, which is the time when an operator first queried the plant computer as to the temperature of the PORV discharge line. Because the indicated temperature was believed by the operating crew to have been a result of the initial opening of the PORV and not a continuous steam discharge, the crew did not close the PORV block valve. This sequence shows the impact of this ..ure to understand the meaning of the temperatures and act accordingly. Detailed results of the analysis of this sequence may be found in Section II.D.2.f.

Alternative Accident Sequence 6

Accident sequence 67 in Figure II-38 has been analyzed as alternative accident sequence 6. In this sequence, the time of PORV block valve closure is delayed 1 hour beyond the 2.3 hour time in the actual accident, so that closure would occur at 3.3 hours. This sequence assesses the effect of a continued loss of coolant out the PORV and the associated effect on core water level and fuel temperature, resulting from the operator failing to close the PORV block valve for an additional hour. Detailed results of the analysis of this sequence may be found in Section II.D.2.g.

Alternative Accident Sequence 7

Accident sequence 38 in Figure II-38 has been analyzed as alternative accident sequence 7. In this sequence, only the method of tripping the reactor coolant pumps is varied. At 73 minutes, one of the two RCPs is assumed to be tripped in each loop. In the actual accident, both B-loop pumps were tripped at 73 minutes and both A-loop pumps at 100 minutes. This alternative method of RCP trip is preferable for two reasons. First, the operation of one pump per loop should increase the net positive suction head (NPSH) available from the outlets of the steam generators. Second, this type of operation would prevent the forced pumping of coolant into an idle loop where it would be, in effect, lost in terms of cooling capability. The analysis of this sequence will indicate the impact of this variation in RCP operation. Detailed results of the analysis of this sequence may be found in Section II.D.2.h.

Alternative Accident Sequence 8

Accident sequence 15 in Figure II-38 has been analyzed as alternative accident sequence 8. In this sequence, all reactor coolant pumps are tripped at the start of the accident, concurrent with reactor trip. This sequence in effect eliminates the core cooling resulting from the RCP forced flow during the first 100 minutes of the accident and thus will help to assess the contribution of these pumps to the course of the accident. Detailed results of the analysis of this sequence may be found in Section II.D.2.i.

Alternative sequences related to other specific concerns are described briefly in the following paragraphs.

Alternative Accident Sequence 9

Alternative accident sequence 9 deals with events occurring slightly later in the accident than are shown in Figure II-38. In this alternative sequence, the PORV block valve is assumed to be closed at 2 hours 18 minutes (as it actually was) but not reopened at later times. Further, reactor coolant pump restart at 2 hours 54 minutes and high pressure injection actuation at 3 hours 20 minutes are also assumed not to occur. This alternative sequence examines the effect of the possibility that after block valve closure the operating crew, failing to recognize and cope with the accident, returns to normal operating procedures. Detailed results of this sequence may be found in Section II.D.2.j.

Alternative Accident Sequence 10

This alternative sequence is similar to alternative accident sequence 9 discussed above the one difference being the allowance for reopening of the PORV block valve, as was done in the actual accident. In this case, operator intervention to control RCS pressure is assumed, but other operator actions to cope with an accident (rather than a somewhat unusual shutdown) are assumed not to occur. Detailed results of this sequence may be found in Section II.D.2.k.

Alternative Accident Sequence 11

At 16 hours into the accident, one reactor coolant pump was restarted and forced-flow through the core reinstated. Alternative accident sequence 11 involves failure of an RCP to be restarted at this time. This sequence addresses the state of core cooling at 16 hours and the importance of the pump restart at that specific time. Detailed discussion of this sequence may be found in Section II.D.2.l.

Alternative Accident Sequence 12

Alternative accident sequence 12 assumes the loss of offsite alternating current (ac) power in the time period of ½ to 5 ½ hours. Although sequence 12 is less closely related to the actual accident progression than alternative sequences 1 through 11, investigation was considered useful because of the potentially serious consequences of such a power loss.

During the time period of ½ to 5½ hours, the emergency onsite ac power system (the diesel generators) had been manually disabled by the operating crew, so that the onsite system would not have started automatically, as designed, on loss of offsite power. Restoration of the onsite system would have required manual action in the diesel-generator building. Thus, had a loss of offsite power occurred in this time period, a total loss of ac power would have been experienced until either the onsite ac power supply was started in the diesel-generator building or the offsite power system restored. The implications of this alternative accident sequence are discussed in Section II.D.2.m.

Alternative Accident Sequence 13

Alternative accident sequence 13 is also related to a loss of offsite power. In this sequence, the time period of interest is March 30 through April 1 (the third through fifth days). In this period, core cooling was being maintained by the operation of one reactor coolant pump. Concern existed that a loss of offsite power would cause the loss of the RCP and thus might compromise the capability to maintain adequate cooling. Detailed discussion of this sequence may be found in Section II.D.2.n.

Alternative Accident Sequence 14

Alternative accident sequence 14 examines the potential for recriticality (the reinitiation of the nuclear chain reaction) in the TMI-2 core during and after the time of the accident. Of concern here are the possible effects of the changes in core geometry resulting from fracturing of the fuel in some regions and possible distortion or destruction of some control rods. This sequence is addressed in Section II.D.2.o.

Alternative Accident Sequence 15

Alternative accident sequence 15 evaluates the effect of reactor building design. Specifically, consideration has been given to the possibility of an accident such as that at TMI-2 occurring in a pressurized water reactor with a different reactor building design. Of particular interest was the ice condenser type used at some pressurized water reactors designed by Westinghouse. Discussion of this evaluation may be found in Section II.D.2.p.

Analysis Results

The results of the alternative accident sequences are summarized in Table II-59. These results can be generalized to indicate the importance of certain operator actions during the first few hours of the accident. The following insights are noteworthy.

- Operator actions that substantially reduced flow from the high-pressure injection system were clearly the primary cause of the fuel damage sustained.
- Failure by the operators to recognize the significance of the PORV discharge line temperature readings was an additional highly significant contributor to the severity of fuel damage.
- If the PORV block valve had not been closed at the time it was, substantial fuel melting would likely have occurred within an hour.
- The trip of a single reactor coolant pump in each loop at 73 minutes might have prevented high fuel temperatures and minimized fuel damage.
- The delay in delivery of emergency feedwater to the steam generators until 8 minutes had no appreciable effect on the accident progression.

b. Alternative Accident Sequence 1: High-Pressure Injection System Allowed to Operate at Full Flow Rates

At approximately 2 minutes into the accident, the high-pressure injection (HPI) system was automatically actuated on a low reactor coolant system (RCS) pressure signal, resulting in the flow of approximately 1000 gallons of water per minute into the RCS. Within 2 to 3 minutes, the operators had substantially reduced the flow from the HPI system to the degree that the amount of water lost out the stuck-open PORV was greater than that supplied by the HPI system. Throughout the first 16 hours of the accident, the HPI system was automatically actuated a number of times; each time the high flow rates from the system were subsequently reduced by the operators.

In this alternative accident sequence, the high-pressure injection system is assumed to have operated at full capacity from the initial actuation. Other events such as the delay of 8 minutes in delivery of the emergency feedwater are assumed to remain the same.

The results of the RELAP[3] and MARCH[2] calculations both indicate that the use of the HPI system in TMI-2 at full capacity would have prevented the overheating of the fuel and the resulting release of radioactive material. These analyses show that the

TABLE II-59. Description of alternative accident sequences and results

Accident Sequence	Parameter Analyzed	Computer Code Used			Results
		RELAP (Ref. 3)	TRAC (Ref. 4)	MARCH (Ref. 2)	
Base Case —Reactor coolant pumps tripped at 73-100 min —Emergency feedwater delivered at 8 min —PORV block valve closed at 2.3 h —High-pressure injection system in "degraded" mode (throttled back from full flow)		X	X	X	
Alternative Sequence 1 (Section II.D.2.b) —High-pressure injection system allowed to operate at full flow rates	Effect of operator decision to substantially throttle back HPI flow	X		X	Core continuously cooled—no fuel damage
Alternative Sequence 2 (Section II.D.2.c) —High-pressure injection system allowed to operate at full flow rates, and —Emergency feedwater delivery to OTSGs at 1 h	Capability of HPI system to cool core without heat removal from OTSGs		X	X	Core continuously cooled—no fuel damage
Alternative Sequence 3 (Section II.D.2.d) —Emergency feedwater delivery to OTSGs at about 40 s	Effect of closure of EFW block valves until 8 minutes		X	X	Little significant change from base case
Alternative Sequence 4 (Section II.D.2.e) —Emergency feedwater delivery to OTSGs at about 1 h	Effect of a more prolonged closure of the EFW block valves		X	X	Definitive conclusions not possible
Alternative Sequence 5 (Section II.D.2.f) —Closure of the PORV block valve at 25 min	Effect of operator error is not closing the block valve after the first check of PORV discharge line temperature	X		X	Core continuously cooled—no fuel damage
Alternative Sequence 6 (Section II.D.2.g) —Closure of the PORV block valve at 3.3 h	Effect of a more prolonged operator error before closure of the block valve			X	Substantial fuel melting could result
Alternative Sequence 7 (Section II.D.2.h) —One reactor coolant pump per loop shutdown at 73 min	Effect of method of shutting down RCPs, i.e., both B loop pumps first, then A loop pumps 28 min later	X			Core continuously cooled —no fuel damage
Alternative Sequence 8 (Section II.D.2.i) —All reactor coolant pumps shut down at time of reactor trip	Effect of cooling provided by forced flow from the RCPs	X	X	X	Definitive conclusions not possible

TABLE II-59. Description of alternative accident sequences and results

Accident Sequence	Parameter Analyzed	Computer Code Used			Results
		RELAP (Ref. 3)	TRAC (Ref. 4)	MARCH (Ref. 2)	
Alternative Sequence 9 (Section II.D.2.j) —PORV block valve not reopened after 2.3 h, no reactor coolant pump restart at 2.9 h, no high pressure injection actuation at 3.3 h	Effect of operator actions to cope with LOCA after 2.3 h			X	Substantial fuel melting could result
Alternative Sequence 10 (Section II.D.2.k) —No reactor coolant pump restart after 2.3 h, no high pressure injection actuation at 3.3 h	Effect of operator actions to cope with LOCA after 2.3 h			X	Substantial fuel melting could result
Alternative Sequence 11 (Section II.2.D.l) —No reactor coolant pump restart at 16 h	Effect of timing of the pump restart			X	Definitive conclusions not possible
Alternative Sequence 12 (Section II.D.2.m) —Loss of offsite ac power at ½ to 5 h	Effect of crew decision to negate emergency ac power actuation system			X	Operator action to restore diesels required within about 15 min to prevent substantial fuel melting
Alternative Sequence 13 (Section II.D.2.n) —Loss of offsite ac power during March 30 to April 1 (third to fifth days)	Effect of loss of forced flow from the one operating reactor coolant pump				Options available to prevent further core damage
Alternative Sequence 14 (Section II.D.2.o) —Recriticality	Recriticality resulting from fuel and control rod damage				Recriticality potential minimal
Alternative Sequence 15 (Section II.D.2.p) —Effect of containment design	Design characteristics of various containment types			X	Some containment designs might have been severely damaged

reactor coolant system would have remained essentially full and cool throughout the incident.

On the basis of the analysis performed for this alternative accident sequence, we find that the operating crew's decision to reduce the flow from the HPI system was a major contributor to the severity of this accident.

c. Alternative Accident Sequence 2: HPI System Operated at Full Flow Rates and Emergency Feedwater Delivered at 1 Hour

In this alternative sequence, the effect of HPI flow analyzed in alternative accident sequence 1 is compounded with the effect of a human error in prolonging the failure to open the emergency feedwater (EFW) system discharge line block valves. In the actual accident, these block valves were opened at approximately 8 minutes. In this alternative sequence, opening of these valves is delayed until 1 hour. This sequence in effect analyzes the capability of the HPI system to cool the core in the absence of heat removal through the steam generators.

The TRAC[4] and MARCH[2] analyses of this alternative accident sequence are in general agreement; both indicate that fuel temperatures remain significantly lower than those achieved during the actual accident. Figure II-39 shows this difference in temperature based on the TRAC calculations.

The analysis of this alternative accident sequence indicates that, for the HPI system design in TMI-2, adequate core cooling would have been achieved by the use of the system at full capacity, even in the absence of heat removal through the steam generators (i.e., without the use of the EFW system).

d. Alternative Accident Sequence 3: EFW Delivered at 40 Seconds

In alternative accident sequence 3, it has been assumed that the emergency feedwater system discharge line block valves were not closed, so that EFW could have been delivered as designed at about 40 seconds into the accident. The comparison of the results of this sequence to those of the base case shows the effect of the 8-minute delay in the initial delivery of EFW to the steam generators.

Analysis of this alternative accident sequence has been performed using the TRAC[4] and MARCH[2] codes. The results of these analyses indicate that, although some differences in the early progression of the accident result from this variation in delivery time, the progression beyond about 80 minutes is essentially the same (see Figure II-40). Since fuel damage did not occur until later than 80 minutes, the delay of 8 minutes in initial delivery of emergency feedwater does not appear to have significantly affected the overall course of this accident. However, since the lack of heat removal through the steam generators apparently had some influence on the initial pressurizer increase off scale and its remaining off scale, the lack of EFW for 8 minutes did, to some degree, influence the decisions of the operating staff. In this sense, closure of the EFW block valves did contribute to the accident progression.

e. Alternative Accident Sequence 4: EFW Delivered at 1 Hour, HPI System in Degraded Mode

In alternative accident sequence 4, it has been assumed that the closure of the EFW discharge line block valves was not corrected until about 1 hour into the accident, rather than 8 minutes. This sequence indicates the effect of a more prolonged failure by the operator to discover the block valve closure.

Analysis of this alternative accident sequence has been performed using the TRAC[4] and MARCH[2] codes. In this instance, the two code calculations differ in their results. The TRAC results indicate that this assumed delay in emergency feedwater somewhat changes the accident progression during the first hour but that after 1 hour the accident assumes characteristics essentially like those of the actual accident. The MARCH results indicate a substantially greater repressurization of the reactor coolant system in this alternative case, to the degree that the RCS safety valves open and remain open for some period of time. This results in a larger loss of water from the RCS and a shorter time before initial uncovery of the core. MARCH then predicts that liquefaction of fuel begins at about 70 minutes into the accident, with a large fraction of the core molten by about 100 minutes. The differing results of the two code calculations appear to result from differences in assumptions regarding the quality of the fluid (steam, steam–water mixture, or liquid water) leaving through the stuck-open PORV and the extent of heating of the coolant by the reactor coolant pumps during the period of flow degradation. Because of the significant uncertainties in the data obtained on the actual accident progression and be-

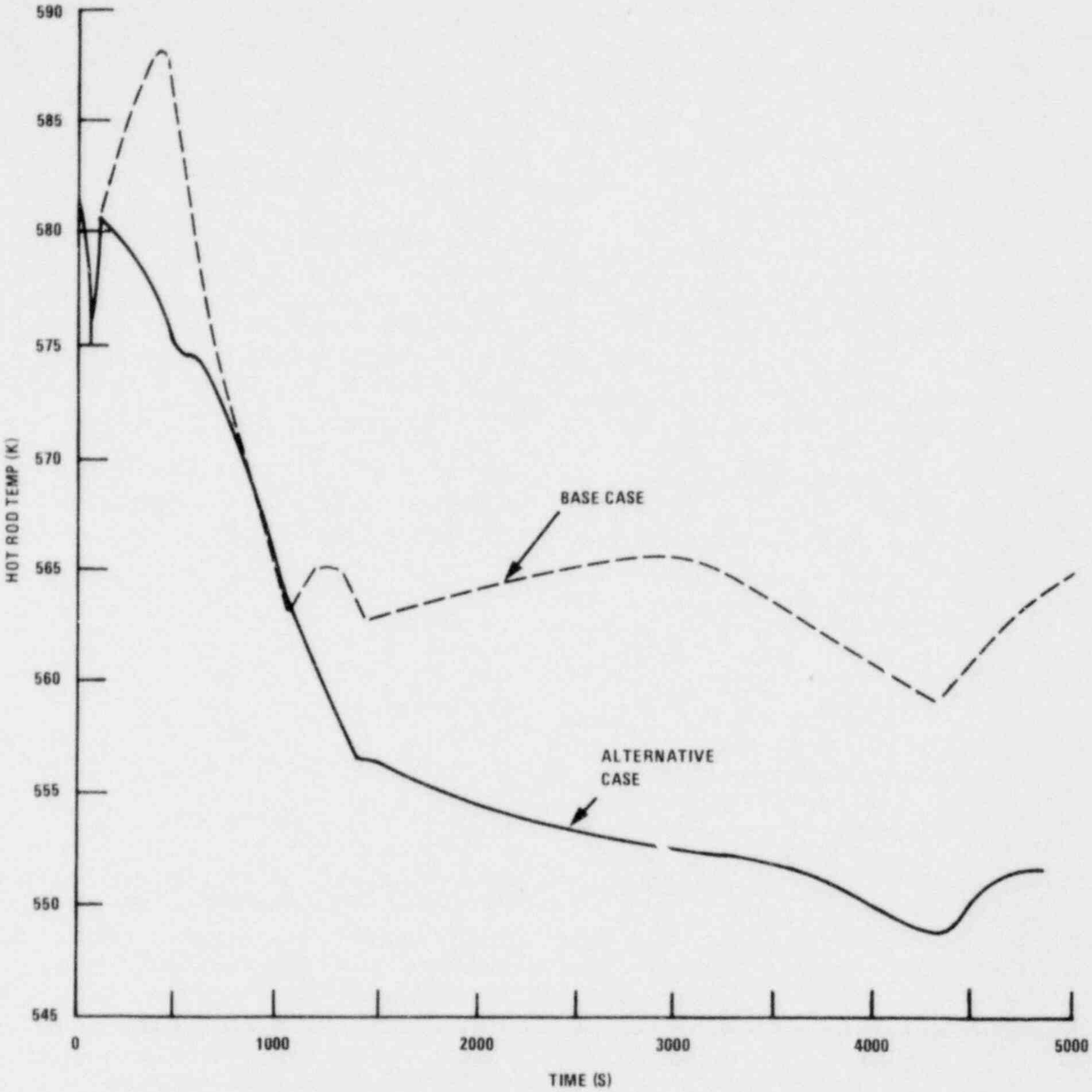


FIGURE II-39. Comparison of Base Case to Alternative Sequence 2 (EFW Delay Until 1 Hour; Full HPI Flow) (Ref. 4)

cause of the limited time available for the Special Inquiry Group's analysis, definitive resolution of these differences has not been possible.

One should note, however, that it would seem likely that the repressurization of the RCS and opening of the safety valves predicted by MARCH would be sufficiently unusual to expect operator intervention. Actions to reduce RCS pressure below the safety valve setpoint would reduce the mass loss from these valves, so that the significantly shorter time to core uncovering predicted by MARCH would be somewhat tempered.

It therefore appears that a delay of 1 hour in emergency feedwater delivery to the steam generators could have produced somewhat worse consequences than those actually experienced in the TMI-2 accident. However, the magnitude of the increase in consequences cannot be determined at this time.

f. Alternative Accident Sequence 5: PORV Block Valve Closure at 25 Minutes

In this alternative sequence, it has been assumed that closure of the PORV block valve occurred at approximately 25 minutes. At this time in the ac-

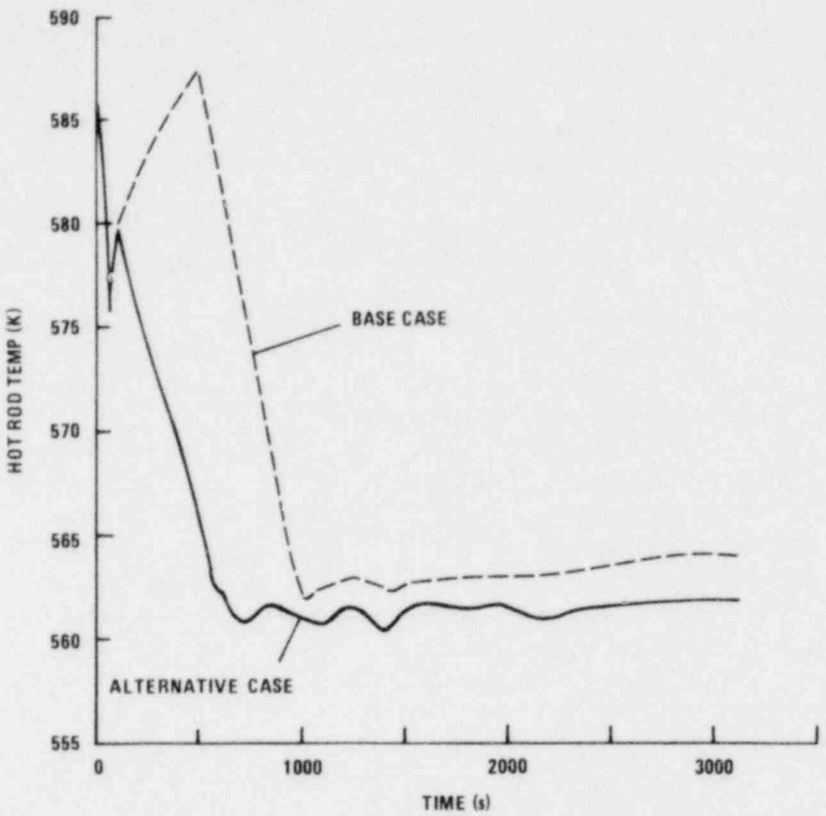


FIGURE II-40. Comparison of Base Case to Alternative Sequence 3 (EFW on at 40 Seconds) (Ref. 4)

cident, the staff in the control room first requested from the computer the PORV discharge line temperature. This sequence has been compared to the base case in order to assess the effect had there been closure of the PORV block valve at this early time.

The analysis of this sequence was performed using the RELAP[3] and MARCH[2] codes. The results of these analyses indicate that the temperature in the core does not become sufficiently high to produce damage to the fuel. With the flow rates from the high-pressure injection system as they are believed to be in the accident, recovery to normal conditions in the reactor coolant system would have taken roughly 90 minutes. Thus, had the PORV block valve been closed at 25 minutes, we find that the event would have produced no significant consequences to the plant.

g. Alternative Accident Sequence 6: PORV Block Valve Closure at 3.3 Hours

In this alternative accident sequence, the time of closure of the PORV block valve is assumed to be delayed by 1 additional hour; therefore closure occurs at about 3.3 hours into the accident. The subsequent course of the accident in the time period between 2.3 and 3.3 hours has been evaluated using the MARCH[2] code so that the importance of the timing of the operator action to close the block valve can be better understood.

The MARCH code analysis indicates that the accident progression after 2.3 hours is particularly affected by the makeup flow; it is also dependent on emergency feedwater flow and the availability of the core flood tanks (CFT). In the actual accident, emergency feedwater flow to the one operable steam generator (steam generator A) was stopped (or significantly reduced) just before the time of PORV block valve closure at 2.3 hours.[5] After this, steam generator heat transfer was decreased, resulting in higher RCS pressures. Also, the availability of the core flood tanks is uncertain because of operator actions prior to 2.3 hours. It appears that the CFT isolation valves were closed early in the accident; therefore the possibility exists that the tanks would not have operated if RCS pressure decreased below their setpoints.

The best estimate MARCH calculation of the alternative sequence indicates that, because of the lack of emergency feedwater after about 2 hours, RCS pressures do not decrease after this time (with the PORV block valve remaining open) but begin a slow increase. Because the pressure level required for core flood tank discharge is not reached, water from these tanks is considered not to be available.

Based on the MARCH results shown in Figure II-41, it appears likely that the failure to close the PORV block valve until 3.3 hours would have resulted in a substantial fraction of the fuel achieving temperatures where fuel-clad eutectic formation, i.e., fuel liquefaction, would occur. Thus, it appears that the TMI-2 accident could have been within an hour of becoming what is called in general terms a core meltdown accident. It should be noted that the likely consequences of such a meltdown would not necessarily be catastrophic because of the likely ability of the reactor building to maintain its integrity and retain a great majority of the radioactive material released during the accident. Discussion of the physical events expected to occur in such a meltdown accident may be found in greater detail in Section II.C.

h. Alternative Accident Sequence 7: One Reactor Coolant Pump Per Loop Tripped at 73 Minutes

In this alternative sequence, the method of tripping the reactor coolant pumps has been varied from that in the actual sequence. During the accident, both pumps in the B loop were tripped at about 73 minutes, with both A-loop pumps tripped 27 minutes later. In tripping both B-loop pumps first, the water subsequently available to the A-loop pumps may have been reduced. In this alternative sequence, one pump per loop is assumed to be tripped at 73 minutes, potentially increasing the water subsequently available to the two running pumps. This may result in prolonged cooling of the core and delayed core uncovering.

Analysis of this sequence has been performed using the RELAP[3] computer code. The results indicate that the fluid density at the suction of the reactor coolant pumps remains higher than in the actual accident. As may be seen in Figure II-42, the fluid density at the A-loop pump suction is calculated to be about 5 pounds per cubic foot at the time of trip of these pumps in the actual accident (about 100 minutes). In contrast, this density is not achieved in the alternative sequence until roughly 135 minutes.

Also obtained in the RELAP calculations is the core inlet mass flow rate, shown in Figure II-43. This figure indicates that in the alternative sequence case the inlet flow rates decrease at a slower rate than in the actual accident and remain almost constant after the trip of the first two reactor coolant pumps.

The calculated pump suction fluid densities and core inlet flow rates discussed above suggest that relatively good flow could have been sustained until

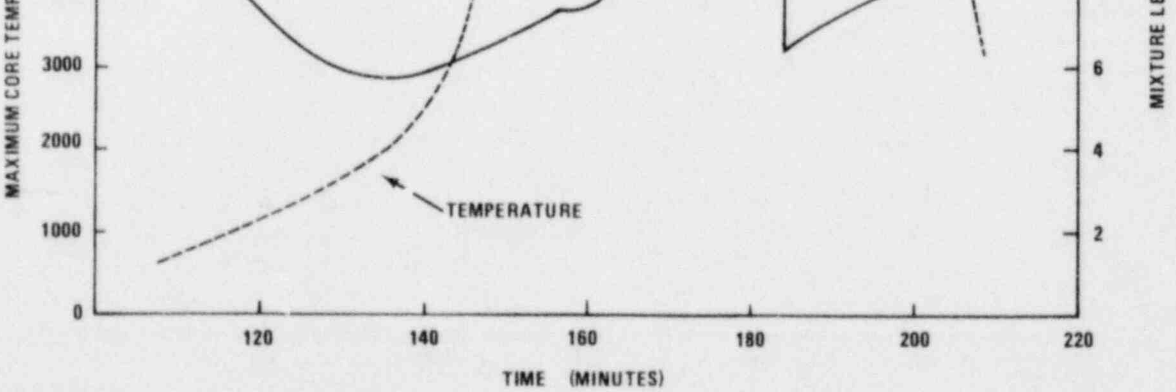


FIGURE II-41. Effect of Delay in PORV Block Valve Closure Until 3.3 Hours (Ref. 2) the time of block valve closure (at about 138 minutes) if the alternative method of pump trip had been used. Because reactor coolant pump flow of this magnitude would have likely prevented high fuel temperatures, fuel damage might not have occurred had one pump been tripped in each loop rather than both pumps in one loop.

We note that, as part of its analysis of the issue of tripping the reactor coolant pumps during small break loss-of-coolant accidents (LOCAs), Combustion Engineering has been examining the effects of running one pump in each loop. Initial indications are that this method may provide an acceptable alternative method to requiring the trip of all pumps.[6] Further, such an alternative method may help to resolve problems associated with the identification of small-break LOCAs vis-à-vis non-LOCA transients. We believe that the analysis of this alternative accident sequence lends credence to the Combustion Engineering analysis and that additional consideration of this method of pump trip in all types of PWRs has distinct merit.

i. Alternative Accident Sequence 8: All Reactor Coolant Pumps Tripped Concurrently with Reactor Trip

In this alternative sequence, the reactor coolant pumps are assumed to have been tripped at the time of reactor trip; i.e., about 8 seconds into the accident. This has two effects. First, the forced flow of water provided by the pumps during the actual accident was a positive factor in keeping core temperatures relatively low. However, this same flow was forcing liquid water into the pressurizer and out the PORV, increasing the mass loss out of the reactor coolant system. This analysis indicates the relative significance of these competing effects.

Analysis of this alternative sequence has been undertaken using the MARCH,[2] RELAP,[3] and TRAC codes.[4] The conclusions reached by the three calculations differ somewhat, with RELAP and MARCH suggesting a somewhat less severe accident and TRAC suggesting a worse accident than the actual TMI-2 accident. The source of these differences appears to be the modeling of the mass flow out the stuck-open PORV. Depending on the calculated quality (steam, liquid water, or a mixture) of the exiting fluid, the mass loss from the reactor coolant system can vary significantly. This uncertainty in modeling, coupled with the significant uncertainties in the RCS mass balance during the accident, results in the differing results obtained by the three analyses. Thus, TRAC calculations indicate that core uncovering could have occurred significantly earlier, and the RELAP and MARCH calculations indicate some additional delay in the beginning of core uncovering.

The significant dependence of the code results on the break-flow model used suggests that more general conclusions regarding the desirability of reactor coolant pump trip concurrent with reactor trip should be approached with great care. Our concerns regarding the long term resolution of this

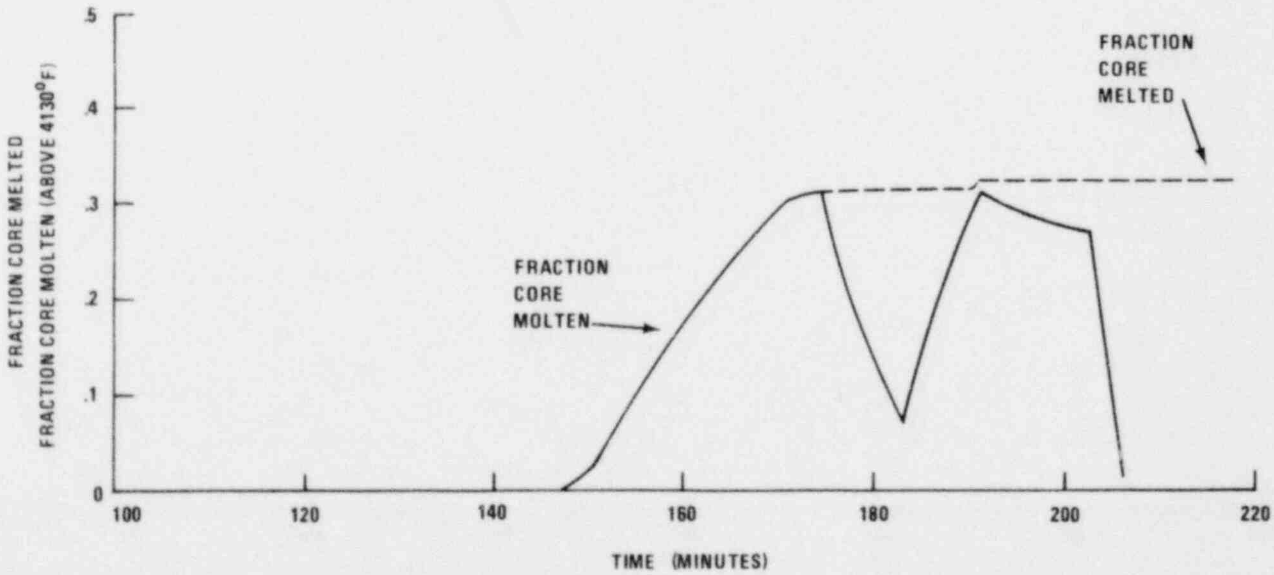


FIGURE II-41--Continued

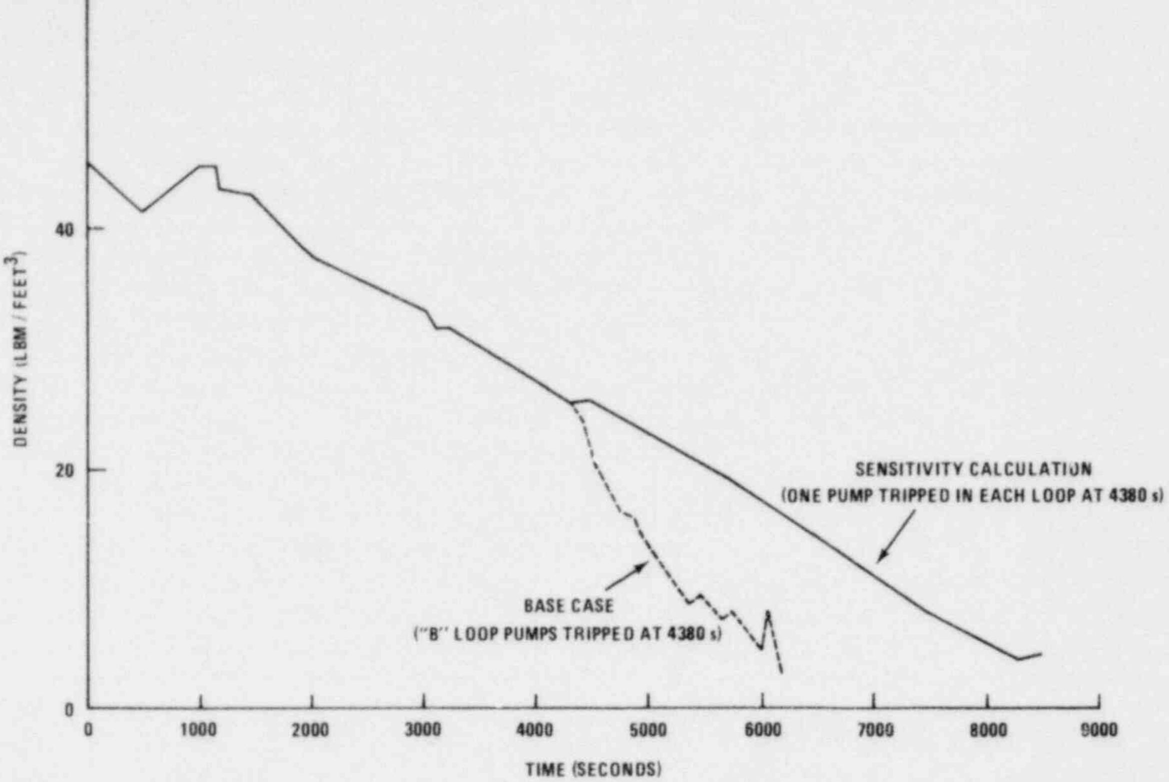


FIGURE II-42. Effect of Pump Trips on Fluid Density in the A-1 Loop Pump (Ref. 3)

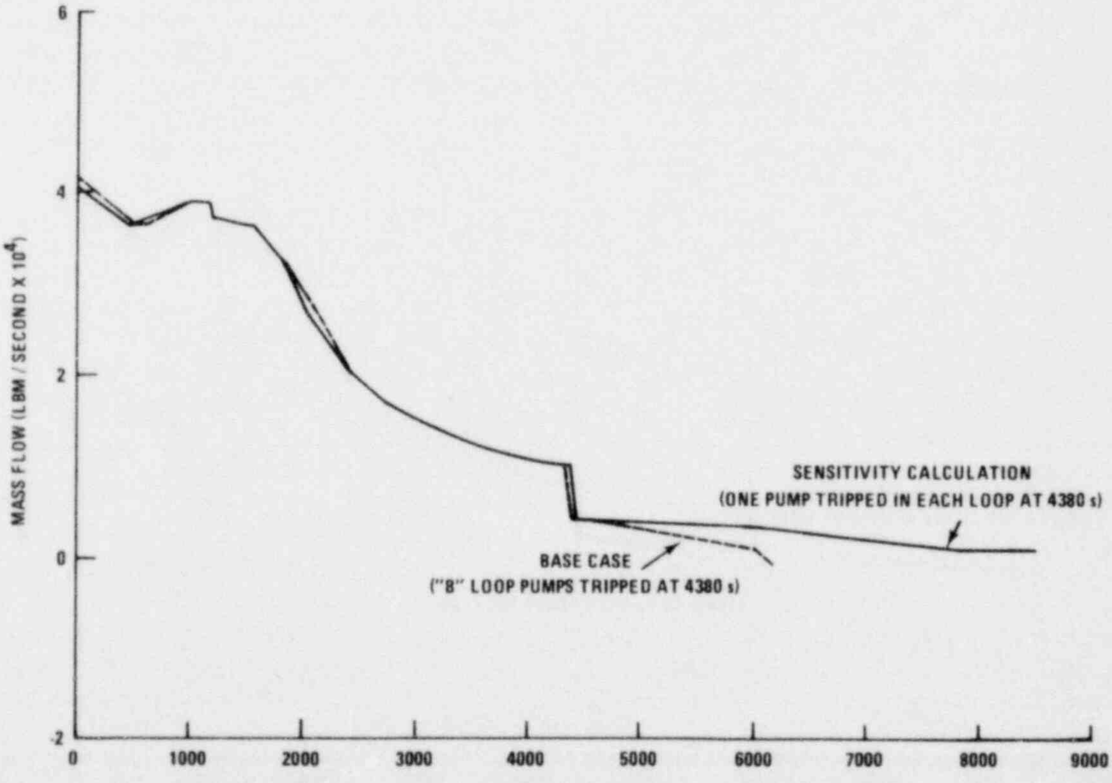


FIGURE II-43. Effect of Pump Trips on Core Inlet Mass Flow Rate (Ref. 3) issue are further discussed in Section II.C.1.b, in the subsection entitled "Reactor Coolant Pump Control."

j. Alternative Accident Sequence 9: PORV Block Valve Remains Closed After 2.3 Hours, No Reactor Coolant Pump Restart at 2.9 Hours, No High-Pressure Injection Actuation at 3.3 Hours

In this sequence, it has been assumed that after the PORV block valve was closed at 2.3 hours, no reopening of the block valve occurred and no attempts were made to start a reactor coolant pump at 2.9 hours or actuate high pressure injection at 3.3 hours. Rather, it has been assumed that the operating crew acted as if a somewhat unusual cooldown following a reactor trip were occurring, rather than an accident. Calculations have been performed with the MARCH code to assess the consequences of this alternative sequence.[2]

With closure of the PORV block valve at 2.3 hours (138 minutes), normal makeup flow (at about 90 gallons per minute) begins to refill the reactor vessel. By about 185 minutes much of the core is re-covered with water. However, because of the closed block valve and the combination of hydrogen blockage of the steam generators and relatively ineffective use of the cooling capability of the steam generator secondary coolant, little heat transfer from the RCS is being accomplished. For this reason, the RCS pressure increases to the safety valve setpoint and mass loss from the RCS begins to occur. Water level in the core subsequently begins to drop again.

Assuming no further corrective actions, a substantial fraction of the core (about 45%) has melted by about 5 hours into the accident. Thus, given the conditions assumed here (i.e., no operator intervention after PORV block valve closure), the eventual complete meltdown of the core is likely.

One should note that the pressurization of the RCS to the safety valve setpoint predicted for this alternative sequence would be a clear signal to the operating crew that a normal cooldown was not occurring. Subsequent intervention to increase heat removal through the steam generators or to reopen the PORV block valve might then be expected, potentially mitigating the severity of core damage. The likelihood of experiencing the eventual complete melting of the core is thus predicated to some degree on the (unpredictable) extent of crew intervention.

k. Alternative Accident Sequence 10: No Reactor Coolant Pump Restart at 2.9 Hours, No HPI Actuation at 3.3 Hours

This alternative sequence is similar to alternative sequence 9 except that we assume operator opening and closing of the PORV block valve, as was done in the first 4 to 5 hours of the actual accident. Analysis using the MARCH code was performed to evaluate the outcome of this sequence.[2]

The general progression of this alternative sequence is similar to that of alternative sequence 9. However, because of the opening of the PORV block valve, mass loss from the RCS occurs somewhat more rapidly, resulting in a shorter time to the beginning of core uncovering. By about 5 hours into this accident progression, about 55% to 60% of the core has melted. Therefore, with this assumed course of events, the complete meltdown of the core is again likely.

l. Alternative Accident Sequence 11: No Reactor Coolant Pump Restart at 16 Hours

In this sequence, it has been assumed that it was not possible to restart a reactor coolant pump at 16 hours. In the actual accident one reactor coolant pump was started at that time and forced cooling of the core reestablished. This analysis assesses the state of core cooling at 16 hours; i.e., whether actions had begun to repressurize the RCS by increasing high-pressure injection flow to cool the core effectively or whether core conditions were continuing to deteriorate. Consideration of this alternative sequence has been undertaken as part of the MARCH re-creation of the first 16 hours of the accident,[2] the additional insights being obtained from other evaluations of this time period by the Special Inquiry Group staff.

Neither the MARCH analysis nor the work by members of the Special Inquiry Group provides conclusive answers to the question of concern (see Section II.C.2). The trends in hot-leg temperatures appear to indicate that some cooldown of the RCS was occurring as a result of the RCS repressurization beginning at about 14 hours and before the restart of the reactor coolant pump at 16 hours. However, this apparent decrease in the hot-leg temperatures is not necessarily an indication of decreasing fuel temperatures. Information from incore thermocouples and self-powered neutron detectors indicate that a substantial region of the core remained very hot in this time period, with quenching of some regions occurring as the reactor coolant pump was restarted. However, no clear trend in quenching of regions is apparent before start of the pump. For this reason, one cannot conclude definitively that the core was (or was not) cooling down in this time period. As such, whether or not reactor coolant pump restart at 16 hours was a critical event cannot be determined conclusively.

m. Alternative Accident Sequence 12: Loss of Offsite Power at ½ to 5½ Hours

In this alternative sequence, a less directly related, less likely event has been postulated. Be ween about 4:30 and 9:30 a.m. of the first day (March 28), the emergency onsite ac power system diesel generators) was disabled by the operating c ew in such a way that, had offsite power been lost all ac power would have been temporarily lost.[7] Such a loss of offsite power was unlikely during this time period; however, the resulting loss of all ac power would have seriously affected an already severe situation.

MARCH analysis has been performed to assess the time required of a significant fraction of the fuel to reach eutectic-formation temperatures.[2] This analysis indicates that, in the event of a total loss of ac power beginning at about 2 hours, some fuel would reach such temperatures in about 24 minutes. The majority of the fuel is predicted to reach these temperatures within about 54 minutes after the loss of ac power.

The onset of such high fuel temperatures could be prevented by the restoration of an ac power source. When questioned about the time required to restore the diesel generators to operation, operators from TMI-2 estimated this to require about 5 minutes.[8] Therefore, we find it likely that a loss of all ac power during the early portion of the accident could have been compensated for by prompt operator action before fuel eutectic formation occurred.

n. Alternative Accident Sequence 13: Loss of Offsite Power During March 30 to April 1

In this alternative sequence, it has been assumed that a loss of offsite power occurred during the time period of March 30 through April 1 (the third through fifth days). In this period, core cooling was being maintained by the operation of one reactor coolant pump. A loss of offsite power during this time would have shut down this pump and other equipment such as the pressurizer heaters and the PORV block valve.

The loss of offsite power postulated here would have had varying degrees of impact on the systems potentially available for core cooling. Table II-60 shows the possible system options and the associated impacts of a loss of offsite power. We believe that the most reasonable option would be the use of the high-pressure injection system. Natural circulation cooling may have been a viable option; however, loss of RCS pressure control and the presence of some hydrogen in the RCS may have inhibited this option. Further, the lack of forced flow in parts of the damaged core region may have resulted in localized higher temperatures following the loss of offsite power. The use of the low-pressure injection system would not have been possible because of the inability to depressurize the RCS by using only the PORV and its block valve.

Restoration of offsite power would of course have increased the number of options available to the operating crew. Restart of a reactor coolant pump, as well as the use of low-pressure injection system, would have been possible.

Analysis using the MARCH code indicates that had a total loss of core cooling occurred on March 31 (the fourth day), at least 20 hours would have had to elapse before fuel temperatures would have reached those needed for eutectic formation.[2] With this amount of time available for restoration of offsite power or the actuation of the HPI system, it appears likely that core cooling could have been restored without further core damage. For this reason, we find that the loss of offsite power on March 30 to Apr l 1 would not have been a serious problem.

o. Altern ative Accident Sequence 14: Recritica ity

This al ernative accident sequence assesses the potential for recriticality (the reinitiation of the nuclear chain reaction) after the accident. Because the high fuel temperatures experienced early in the accident distorted the core geometry and damaged control rods, we have evaluated possible core reactivity changes.

A number of analyses of criticality potential were performed after the accident by the NRC staff[9] and by Babcock & Wilcox.[10] These analyses considered degrees of fuel damage ranging from essentially no geometric distortion to a substantially collapsed core. In general, no credit was given in

TABLE II-60. Possible systems options to mitigate a postulated loss of offsite power on March 30–April 1

System	Effect of Loss of Offsite Power
(1) High-pressure injection system	None
(2) Natural circulation	Loss of RCS pressure control may prevent natural circulation
(3) Low-pressure injection system with depressurization caused by PORV block valve opening	No power to PORV block valve

these analyses for control rod or burnable neutron poison material; dissolved boron was the only presumed poison in the core. The results of these calculations indicate that subcriticality could be maintained with boron concentrations of 1500 parts per million (ppm) for an essentially undisturbed core and range up to about 3500 ppm for a fully damaged core in its most susceptible configuration.

For the core configuration suggested in Section II.C.2 as now thought to exist in the TMI-2 vessel, the criticality calculations indicate that boron concentrations in the range of 1500 to 2200 ppm is required to maintain subcriticality.[10] Since no credit is given for control rod and burnable poison material, it is likely that these estimates are conservative; i.e., a more realistic requirement for boron concentration would be somewhat less.

Reactor coolant samples taken on April 7 indicated that the coolant was being maintained at approximately 2200 ppm,[10] suggesting that the potential for recriticality was not a serious concern. Subsequent to the analysis, the boron concentration was increased to over 3000 ppm to provide an even greater margin of subcriticality.

The possibility of an inadvertent dilution of the RCS could have caused the possible return of the core to criticality and caused additional problems in the recovery process. However, since such a dilution would have to go undetected for some time to result in recriticality, it seems reasonable that operator detection and correction would be likely prior to the return to criticality.

p. Alternative Accident Sequence 15: Effect of Containment Design

Consideration has been given in this section to the effect of various containment designs on the course of the accident. Specifically, it has been postulated that the containment design was different from what actually exists at TMI-2; this then indicates the relative vulnerability of different containment designs to this type of accident.

The principal threat to the TMI-2 containment occurred at about 1:50 p.m. on the first day (March 28), when a hydrogen deflagration resulted in a 28 psig pressure spike.[11] Subsequent analysis of this event indicates that uncertainty exists with respect to the amount of hydrogen burned in the deflagration and the volume in the reactor building within which the burn occurred. Depending on the type of data used (e.g., shape of the pressure pulse, oxygen depletion in the reactor building), estimates of the amount of hydrogen burned range from about 550 to 1000 pounds[2,12] (see Section II.C.2 for additional discussion). Seemingly conflicting data also exist regarding the region within which the deflagration occurred. Some data suggest that the burn occurred in a relatively small section in the building (a local burn), while other data suggest that it occurred throughout the building (a global burn). The assessment of hydrogen burning presented here reflects these uncertainties, so that definitive conclusions on the capability of certain building designs are not possible.

Table II-61 shows typical design characteristics for the variety of containment buildings used in large commercial reactors in this country. This indicates that the containment buildings grouped under the category of "large free volume" have volumes and design pressures comparable to that of TMI-2. Such designs would not be seriously threatened by a hydrogen deflagration such as that experienced during the TMI-2 accident—just as the TMI-2 containment was not threatened.

The data in Table II-61 suggest that the various pressure suppression types of containment building are more susceptible to damage from a hydrogen deflagration of the magnitude experienced at TMI-2. For this reason, each type of pressure suppression containment will be discussed below.

TABLE II-61. Typical containment design parameters

Containment Type	Example Plant	Free Volume (ft^3)	Design Pressure (psig)
Large Dry Containment			
Prestressed Concrete	TMI-2[13]	2×10^6	60
Free Standing Steel	St. Lucie[14]	2.5×10^6	44
Subatmospheric,			
Reinforced Concrete	Surry	1.8×10^6 [15]	45[16]
Spherical Steel Shell	Perkins[17]	3.3×10^6	47
Pressure Suppression			
Ice Condenser	Sequoyah	1.2×10^6 [18]	12[19]
BWR Mark I	Peach Bottom[20]	2.8×10^5	56
BWR Mark II	Zimmer[21]	3.9×10^5	55
BWR Mark III	Grand Gulf[21]	1.7×10^6	15

Analysis of the capability of one ice condenser containment design to withstand pressure loadings due to hydrogen burning has been performed at Battelle Columbus Laboratories (BCL), using the MARCH code.[2] This analysis indicates that, if 550 to 1000 pounds of hydrogen were burned globally in an ice condenser containment, failure of the building would be likely. If the TMI-2 burn was local, a similar event in an ice condenser would be much less likely to fail the building. Further, if a comparable concentration of hydrogen were burned in an ice condenser, building failure would be much less likely. Resolution of the question of local versus global burning may be obtained when the TMI-2 reactor building is reentered (expected in the spring of 1980) and examinations conducted.

The BWR Mark I containment is of relatively high design pressure but of very small free volume, suggesting that this design could also be vulnerable to hydrogen burning. However, the majority of plants with Mark I containments have been required to inert the containment atmosphere by replacing the air with nitrogen, so that the potential for hydrogen burning is not of concern. Analysis of the possible vulnerability of a noninerted Mark I containment to hydrogen burning was performed by Battelle Columbus Laboratories for the reactor safety study.[22] This analysis indicates that, because of the combination of high design pressure and strength of the steel Mark I containment and the limited oxygen within the building available for combustion, it is possible that this containment could withstand the burning of large amounts of hydrogen.

The BWR Mark II containment design is characterized by a somewhat larger free volume than the Mark I and a design pressure comparable to the Mark I design. The Mark II is constructed of prestressed concrete rather than the steel of the Mark I. Because of the lack of an inerted atmosphere, the Mark II would be somewhat more vulnerable to hydrogen burning. Because no specific analysis is available on this containment design, we cannot conclude whether or not a hydrogen deflagration of the magnitude of that in TMI-2 would have caused containment failure.

The BWR Mark III containment is the largest of the BWR containments, being roughly comparable in free volume, design pressure, and construction to the analyzed ice condenser design. This comparability in design characteristics suggests that the Mark III containment would respond in a manner similar to that predicted for the ice condenser; that is, a global deflagration of 550 pounds of hydrogen, which may have occurred in TMI-2, could cause the failure of a Mark III containment. As was discussed for the case of the ice condenser design, resolution of this issue awaits the examination of the TMI-2 reactor building.

Since the time of the TMI-2 accident, the NRC Lessons Learned Task Force has included as one of its short term recommendations the need for the inerting of all BWR Mark I and Mark II containments.[23] Consideration of a similar requirement for the ice condenser and Mark III containments was included in the final report of that task force as part of its Recommendation 10.[24] This recommendation calls for the use of the rulemaking process to consider the inclusion in the licensing process of "certain design features for mitigating accidents that are not provided by the set of design basis events."[25] Our analysis and conclusions here support these recommendations of the Lessons Learned Task Force.

REFERENCES AND NOTES

[1]NRC, "Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock and Wilcox Company," NUREG-0560, May 1979, at 3–14.

[2]Battelle Columbus Laboratories, NRC, "Analysis of the Three Mile Island Accident and Alternative Sequences," NUREG/CR-1219.

[3]EG&G Idaho, Inc., "TMI Sensitivity Calculations," December 1979.

[4]Los Alamos Scientific Laboratory, "Preliminary Calculations Related to the Accident at Three Mile Island," LA-UR-79-2425, December 1979.

[5]NRC, "Investigation of the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement," NUREG-0600, at IA-40, August 1979.

[6]NRC, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of Coolant Accidents in Pressurized Water Reactors," NUREG-0623, November 1979, at 28.

[7]NRC, "Investigation of the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement," NUREG-0600, August 1979, at I-4-74.

[8]Faust, Frederick, Scheimann, and Zewe dep. at 248.

[9]Memorandum from C. R. Marotta, NRC/NMSS, to K. Kniel, NRC/NRR, Subject: Recriticality Potential for TMI-2 Core, dated May 14, 1979.

[10]NRC, "Evaluation of Long Term Post Accident Core Cooling of Three Mile Island, Unit 2," NUREG-0557, Sec. 6.8., May 1979.

[11]NRC, "Investigation of the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement," NUREG-0600, August 1979, at I-4-47.

[12]President's Commission on the Accident at Three Mile Island, "Technical Staff Analysis Report on Chemistry," October 1979, at 18.

[13]Met Ed, "Final Safety Analysis Report (FSAR), Three Mile Island Nuclear Station-Unit 2," Vol. 1, Sec. 1.3-11.

[14]Florida Power and Light Company, "Final Safety Analysis Report (FSAR), St. Lucie Plant," at 1.3-7.

[15]Virginia Electric Power Company, "Final Safety Analysis Report (FSAR), Surry Power Station, Units 1 and 2," at 14.5.2.1-3.

[16]*Id.* at 5.4-1.

[17]Duke Power Company, "Preliminary Safety Analysis Report (PSAR), Perkins Nuclear Station, Units 1, 2, and 3," Table 1.3-1.

[18]Tennessee Valley Authority, "Final Safety Analysis Report (FSAR), Sequoyah Nuclear Plant," at 6.2-111.

[19]*Id.* at 6.2-3.

[20]Philadelphia Electric Company, "Final Safety Analysis Report (FSAR), Peach Bottom Atomic Power Station," Table 1.7.4.

[21]Mississippi Power & Light Company, "Final Safety Analysis Report (FSAR), Grand Gulf Nuclear Station," Table 1.3-4.

[22]NRC, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), Appendix VIII, October 1975, at VIII-155.

[23]NRC, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," NUREG-0578, July 1979, at A-20.

[24]NRC, "TMI-2 Lessons Learned Task Force Final Report," NUREG-0585, October 1979, at 3-6.

[25]*Id.* at 3-1.

E HUMAN FACTORS

1. INTRODUCTION

From the earliest accounts of the events at Three Mile Island Station (TMI) on March 28, 1979, it was apparent that actions or inactions by the control room operators were an integral part of the accident sequence. It was equally apparent during early investigative efforts by the Special Inquiry Group (SIG) and others[1,2] that many underlying factors were present that actually or potentially precluded the operators from preventing or ameliorating the accident. Accordingly, the objective of this analysis has been to establish the nature and degree of operator "error" and gain an indepth understanding of all pertinent factors.

Because of the lack of personnel within the NRC, with the proper background and experience to conduct a human factors investigation, the SIG acquired outside assistance from the Essex Corporation.

The report here draws extensively upon the Essex Corporation work but includes inputs derived from other SIG activities. The report is organized into the following sections:

1. Introduction
2. Human Factors and the TMI-2 Accident
3. Control Room Design
4. Emergency Procedures
5. Operator Selection and Training
6. Human Factors Precursors
7. Recommendations

2. HUMAN FACTORS AND THE TMI-2 ACCIDENT

"Human factors" is an interdisciplinary approach to optimizing human performance in man–machine systems. It includes application of principles relating to psychology, physiology, instrumentation, control and workspace design, personnel selection, and personnel training. When discussing the causes of the TMI-2 accident, several factors within these principles can be singled out as directly contributing to the accident. Others can be identified as possible contributors to the general confusion of the operators, confusion that impaired their ability to analyze the problem and take corrective actions.

Although the critical condition of the plant continued for some 16 hours,[3] investigation into the human factors aspects focused on the first 150 minutes of the accident. This limitation was chosen because of time and resource constraints on the SIG, as well as the fact that the major operator decisions affecting the accident occurred during that period.

Significant Operator "Errors"

Two situations clearly had a significant impact on the accident. First, the operators failed to recognize that the pilot-operated relief valve (PORV) on the reactor pressurizer had not automatically closed, as it is designed to do, in the course of recovery from a reactor trip. Consequently, the operators did not close the PORV block valve for more than 2 hours after the accident began, and the resulting water loss caused significant damage to the reactor.[4]

The second significant and more fundamental action was operator throttling (curtailment) of the high-pressure injection (HPI) of water into the reactor coolant system. Had the HPI been allowed to operate automatically as intended, the reactor core would have remained covered, and serious core damage would have been prevented.

Failure to Isolate the PORV

As part of their training, the operators memorize the immediate actions and symptoms in the plant's emergency procedures and use them as a basis for diagnosing and responding to emergencies. Failure to close the PORV block valve can be attributed to failure to recognize and respond to the symptoms described in the plant's emergency procedure for pressurizer system failure.[5] According to this procedure, the operator must recognize the following conditions:

1. that the PORV valve has failed to close;
2. the elevated reactor coolant drain tank pressure and temperature; and
3. the elevated PORV pipe discharge temperature above the 200°F alarm setpoint.

For each condition, there is a logical human factors explanation of why the operators failed to take corrective action.

The initial failure to notice the open PORV can be traced to a misleading instrument that indicates the valve's position—a single red PORV status indicator light. This light is on when an electrical signal is sent to open the PORV, and it is off when the signal is terminated. The light does not, as may be inferred from its label as shown in Figure II-44, indicate the actual position of the PORV.[6] Consequently, about 13 seconds into the accident, when the PORV indicator light went out, the operator believed the valve had actually closed. In fact, it had stuck open.[7] Originally, the TMI-2 control room design contained no indicator light. Following a March 29, 1978 trip where the PORV had failed open,[8] the light and labeling were installed.

A valve indicator system that directly sensed the open and closed positions of the valve, i.e., microswitch on relief valve stem, probably would not have incorrectly indicated valve closure. With such an indication system, the operators would have noticed the open valve indication (or lack of closed indication), closed the block valve much earlier, and terminated the accident well before any core damage occurred.

The failure of the operators to recognize and respond to the second symptom, elevated reactor coolant drain tank temperature and pressure, also was compounded by human engineering and design factors: inadequate and poorly placed instrumentation and the preaccident history of a leaking PORV or code safety valve.

Water discharged from the pressurizer through the PORV eventually collects in the reactor coolant drain tank (RCDT). Thus, if the PORV fails open, the temperature, pressure, and water level in the RCDT are expected to increase. However, at TMI-2, one of the code safety valves (or possibly the PORV) that also drains into the RCDT had been leaking since the fall of 1978 and had been scheduled for repair during the next reactor shutdown.[9] For this reason, elevated temperature, pressure, and level in the RCDT were not unusual observations. About once every shift, operators had been forced to pump the accumulated water from the RCDT.[9]

Moreover, the instrumentation for RCDT conditions and the corresponding alarms are behind the control panel and cannot be read unless the operator leaves his normal operating area in front of the control panel and walks about 50 feet (see Figure II-47). To further compound the problem, the RCDT instrumentation on the back panel only gives instantaneous information. It does not record the parameters that indicate the previous conditions and thereby indicate trends in the RCDT temperature, level, and pressure. Consequently, when the operator went to check the RCDT status, he had difficulty telling whether the RCDT conditions were a result of an expected single opening and closing of the PORV at the beginning of the accident or whether they were a result of an unexpected longer, continuous leak from a stuck-open PORV.

In fact, in the period from 10 to 15 minutes into the accident, one operator did check the RCDT and noted that it was full.[10] After the RCDT rupture disk had failed (at about 15 minutes), the shift supervisor from Unit 1 checked the panel and noted that the tank was empty.[11] This occurrence was immediately followed by an increase in reactor building pressure and the sounding of an associated alarm. The shift supervisor consulted with the control room operators and correctly concluded that the RCDT rupture disk had failed. However, they incorrectly concluded that the RCDT had been nearly full of water from the previously leaking PORV or code safety valve and that the subsequent momentary opening of the PORV (at the time of reactor trip) had added enough

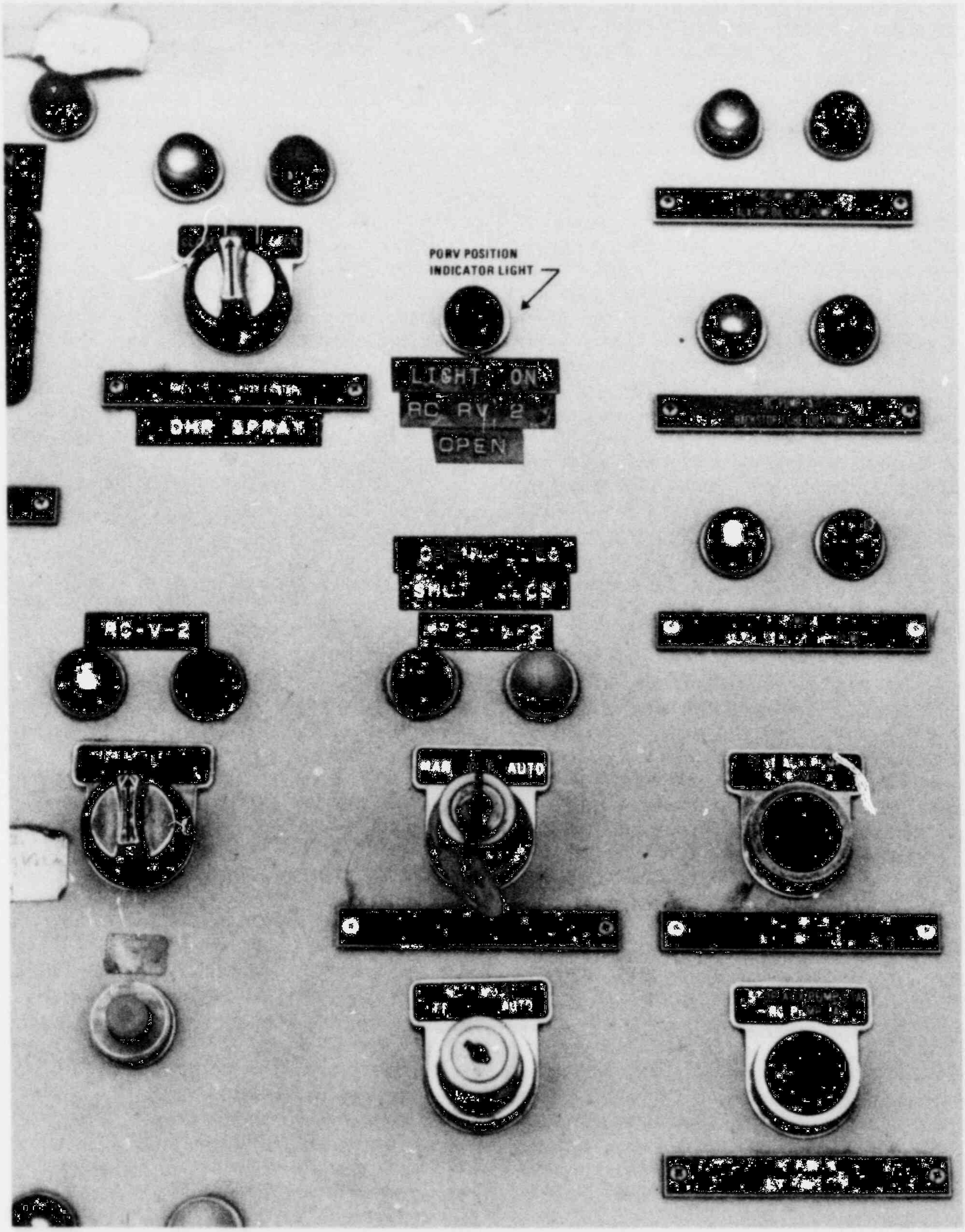


FIGURE II-44. PORV Indicator Light and Controls water to overfill the tank, causing its emergency rupture disk to break,[12,13] and result in the tank indicating empty, i.e., the reading was off scale low (below 60 inches).

If the RCDT monitoring instrumentation had been recorded, the operators may have noticed the time trend of RCDT parameters and correctly realized that the PORV was stuck open when they investigated RCDT conditions. On the other hand, had the instrumentation been located within view of the operators, it is more likely they would have noticed the increasing water level.

The third symptom was the elevated temperature of the discharge pipe from the PORV. Contrary to procedure, the TMI-2 plant had been operating with a leaking valve that had been causing high PORV discharge temperature (180°F) for several months.

Consequently, the TMI-2 operators were misled into believing that the rise in temperature in the discharge line following the reactor trip was caused by a combination of high temperatures before the accident and a momentary opening of the PORV. Furthermore, the situation leading the operators to this conclusion was compounded by their incorrect expectation that the highest possible temperature in the discharge line as a result of an open PORV was well over 300° F.[14] In fact, because of the throttling action of the PORV relief valve, the maximum achievable temperature was closer to 300°F. The operators were unaware of this fact; it is not explained in their emergency operating procedures, and it apparently was not learned in their training.

After the accident began, the operators monitored the discharge line temperature and twice misread a 285°F temperature as being only 235°F.[15] After almost 2½ hours, the oncoming shift supervisor noticed that the PORV discharge temperature was 229°F, about 25°F hotter than the code safety discharge temperature. He correctly interpreted the reading and the PORV block valve was closed,[17] thereby isolating the malfunctioning PORV.

Throttling of High-Pressure Injection

Manual throttling or curtailment of the flow of emergency core cooling water into the reactor coolant system was a second significant operator action that caused the core damage. This action is significant because it involved not only an inability to diagnose the specific leak point but an inability to understand that a leak was occurring at all.

At approximately 2 minutes into the accident, operators took manual control of the automatic high-pressure injection (HPI) system (which had started automatically when RCS pressure dropped below 1640 psig) and reduced the water flow to the reactor. For most of the first 1½ hours, the net flow rate was reduced from about 1000 gallons per minute to only about 25 gallons per minute.[17] Technical analysis indicates that if such severe throttling had not occurred, core damage probably would have been avoided.[18]

The factors that led the operators to take this action include improper training, lack of instrumentation, inadequate procedures, poor operating practices, and fundamental misunderstanding of reactor thermal hydraulics by the operators.

The operators' basic mistake was failing to recognize that the reactor was experiencing a small loss-of-coolant accident (LOCA) that could lead to core uncovery and overheating. This mistake was further compounded by their inability to realize that the existing low-pressure condition in the reactor would lead to boiling reactor coolant that had the potential for uncovering the core. Consequently, they turned off the automatic safety device (HPI) even though the low-pressure condition that had activated it was persisting.

The TMI-2 plant did not have instrumentation for directly measuring water inventory or water level. Thus, for operators to realize a LOCA was occurring, they had to recognize and properly diagnose LOCA symptoms that include decreasing pressurizer level, decreasing reactor coolant system pressure, increasing reactor building pressure, increasing reactor building temperature, and water accumulating in the reactor building sump.

The TMI-2 operators were faced with all but the first of these symptoms. The question is, why then did they fail to diagnose the LOCA properly? One answer lies in the fact that TMI-2 operator training and written emergency procedures relied on a misconception that water level in the pressurizer would serve as a true indication of total volume of water in the reactor coolant system.

Subsequent analysis reveals that for the TMI-2 type LOCA, the belief that high pressurizer level signifies that the reactor vessel is full of water is not correct. Although this fact was known in some segments of the industry, the information had not been incorporated in the TMI-2 operator training or emergency procedures. Thus, the operators mistakenly throttled HPI in an attempt to maintain pressurizer level within the normal range.[19] For example, the emergency procedure for a LOCA contains two alternative sections, each of which warns the operators to look for a combination of low reactor pressure and low pressurizer level.[20] At TMI-2, reactor pressure did fall, but pressurizer level increased. The operators did not observe the symptoms applicable to this written procedure and naturally did not follow the prescribed corrective actions.

Lacking unambiguous written emergency procedures, operators instead followed other dictates of their training and procedures, which were in our opinion an ill-considered fix to a basic engineering problem of pressurizer level sensitivity.[20]

The TMI-2 pressurizer is not large enough to maintain proper level and reactor coolant pressure following turbine trips and other transients that frequently occur. Consequently, to avoid excessive drops in the pressurizer level and pressure, the operators immediately start an additional high pressure injection pump and increase high pressure injection flow until proper pressurizer level and pressure are restored. Thus during the accident when the pressurizer level came back up, the operators were conditioned to reduce the high pressure injection flow and apparently ignored the fact that high pressure injection flow had been initiated automatically because of low reactor coolant system pressure. The operators waited until the pressurizer was nearly full before they throttled the high pressure injection flow to prevent filling the pressurizer solid with water, which their training and experience had taught them to avoid. These events all happened in the first few minutes when there was little time to think and the operators were simply following their normal operating procedures.

As the accident progressed, the reactor coolant system pressure continued to drop. The operators knew that the RCS pressure was abnormally low and that other LOCA symptoms were present, but they did not make the correct diagnosis or take corrective action. At 38 minutes after the reactor trip, the containment sump filled with water and the operators stopped the sump pumps, attributing this LOCA symptom also to the initial opening of the PORV.

In addition to these actions, the operators delayed following written procedures requiring them to declare a site emergency when high containment pressure and temperature symptoms were present. The operators also failed to either understand or react to the basic design concept of a pressurized water reactor—that it is imperative to keep the pressure high to prevent the hot reactor coolant water from boiling and potentially uncovering the core.

These actions could be attributed to "operator error," as was done in NUREG-0600.[9] However, our view is that the overall system of operator training, procedures, control room design, and maintenance is the major problem—a view that has become more evident as this study has progressed.

Other Factors Contributing to the Accident

Other human factors had strong potential for contributing to the general confusion of the operators and impaired their ability to respond correctly to the problems.

The Essex Corporation's study[21] describes several such factors. The confusion of the first hour was compounded by the discovery that the emergency feedwater block valves were closed. Although technical analysis suggests that closure of these valves did not directly influence the severity of the accident,[22] discovery of their closure after 8 minutes and the resultant diversion of operator attention to feedwater problems may have diverted the operators from analysis of and reaction to more fundamental factors contributing to the accident.[23]

The failure to discover closure of emergency feedwater va s earlier can be directly attributed to several human engineering control room factors:

1. Adequate quality control of valve lineup and proper procedures could have led the operators to discover the closed valves sooner.
2. The control room did not contain any direct indication of flow status. Thus, it was necessary to either notice the valve position lights or check steam generator level to determine whether there was adequate feedwater flow.
3. The indicator lights that tell the operator whether or not the emergency feedwater block valves are closed may have been hidden by one of the out-of-service tags that cluttered the control panel (Figure II-45).
4. The feedwater control panel is not laid out in a logical fashion. For example, control locations do not mimic actual valve and pump positions in the plant. In fact, the control and display placement on the emergency feedwater panel is inconsistent (Figure II-46). The absence of any logical panel layout forced operators to rely on memory or random search to locate a particular control. This panel layout problem also existed elsewhere in the control room and increased operator workload and the probability for mistakes, particularly during emergency situations.

Another condition that contributed to the confusion in the control room was the alarm system that hampered the operators during the early stages of the accident. The control room contains more than 750 alarms. These alarms are not prioritized, and many are difficult to read from normal operator positions. During the first few minutes, more than 100 alarms went off.[24]

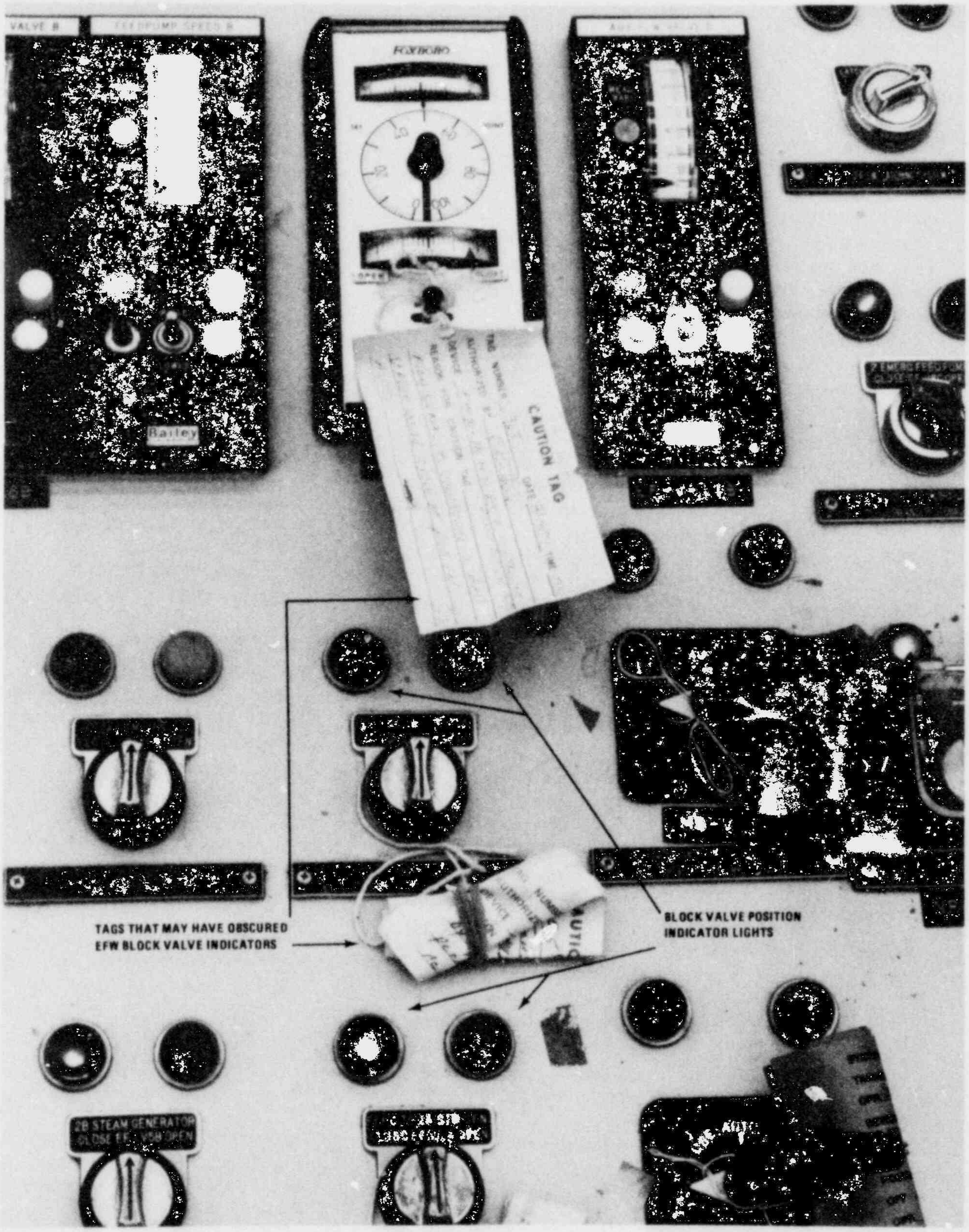


FIGURE II-45. EFW Block Valve Controls and Indicator Lights Showing Caution Tags

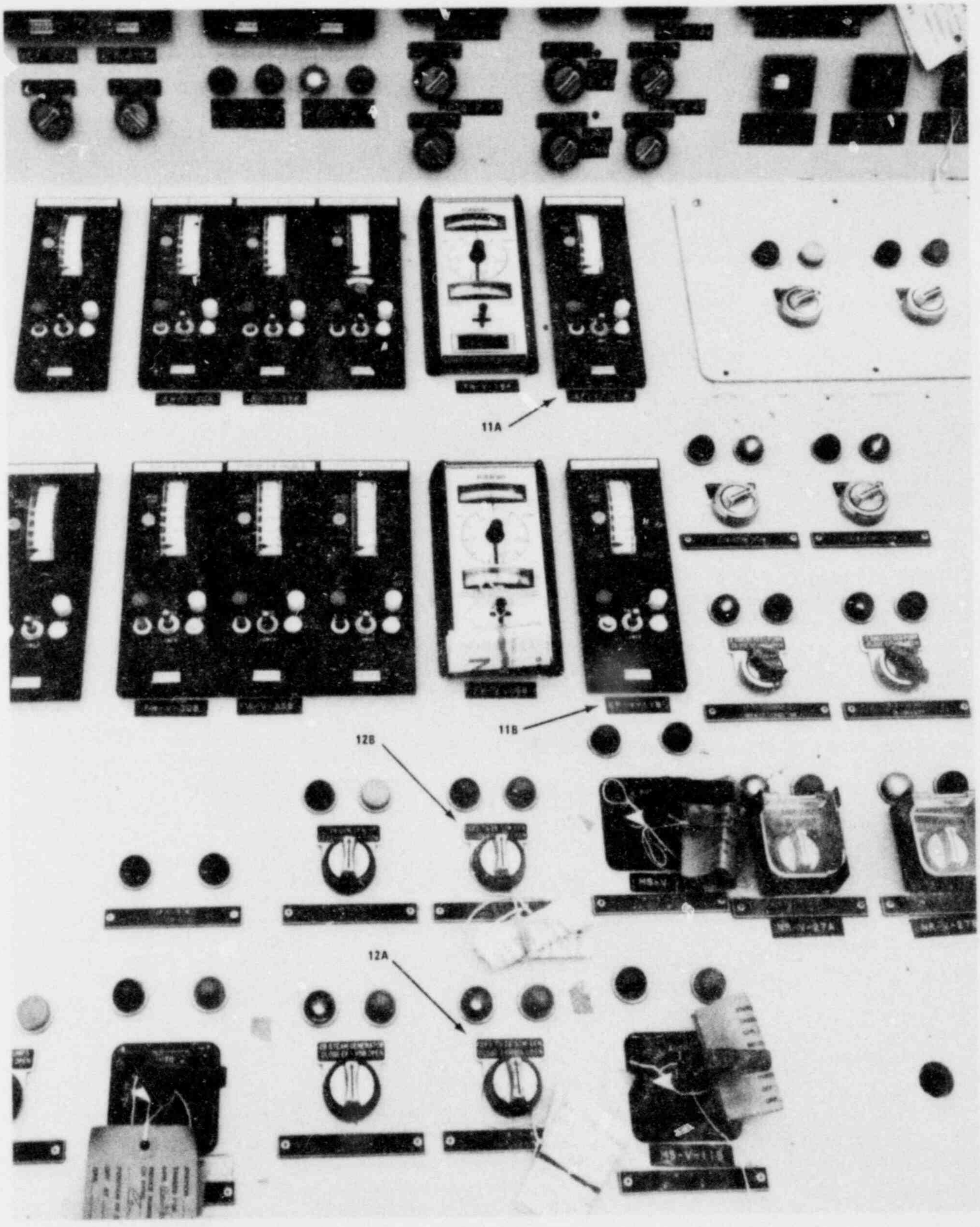


FIGURE II-46. EFW Control Station Showing Block Valve (11A and B) and Control Valve (12A and B) Layout

This problem with the alarm systems prompted one operator, a year before the accident, to write:

> The alarm system in the control room is so poorly designed that it contributed little in the analysis of a casualty. The other operators and myself have several suggestions on how to improve our alarm system—perhaps we can discuss them sometime—preferably before the system as it is causes severe problems.[25]

When the accident occurred, the control room alarm system had not been significantly changed.

The Essex Corporation found other examples of poor control room design that contributed to confusion. These situations include poor lighting, numerous examples of illogical panel layout, confusing use of indicator color coding, and situations where operator ability to read meters and observe indicator lights were impaired.[26]

Furthermore, the Essex report found that several operator errors were caused or influenced by expectancy or set.[27] Set is a psychological construct defined as a temporary, but often recurrent, condition of individuals that orients them toward certain information and events rather than others and increases the likelihood of certain responses over others. The influence of set in the TMI accident is evident in the tendency to evaluate indications of present plant status in terms of events or conditions occurring in the recent past. For example, the high exhaust pipe temperature of the PORV was not considered excessive because the safety valve had been leaking for some time.

Operators also seemed conditioned to expect problems in the secondary system and not in the primary system because of their experience with both systems. In addition, testimony of plant personnel indicates that high-pressure injection initiation was not unexpected because it had occurred before. Rapid cooldown events and normal reactor trips had conditioned the operators to take immediate actions (manually start an HPI pump and isolate letdown) and to key on pressurizer level as the main reactor coolant system parameter to be controlled. For the previous rapid cooldown events, the HPI system was stopped without harmful effects after pressurizer level recovered.[28] Thus, it was natural during the accident that the operators would have throttled HPI to avoid increasing pressurizer level to the point that they would have solid conditions indicated. Such expectancies, combined with the slow response of the system, obscured the real problems.

Development of these erroneous expectancies, however, does not reflect on the operators themselves but on their training. In the absence of adequate training, operators must use whatever information is at their disposal, including their knowledge of what has happened in the plant in the recent past and during their involvement with the system. The function of training is to provide the ability to integrate displayed information to arrive at an understanding of present events and required actions independently of what has happened in the recent past. The training provided to the TMI operators was obviously deficient in this regard.

The importance of operator set in the TMI-2 incident is also evident from the fact that several conclusions, including the determination that the PORV was open, were reached by personnel who were new to the problem, did not have the recent experience with the plant, and were able to assess available information on its own merits without reference to prior influences.

The Essex Corporation found that the influence of psychological stress as a determinant in the TMI accident was difficult to evaluate on the basis of available data.[29] The operators were under increasing stress over the course of the accident; however, inappropriate actions or inactions can be attributed only indirectly to stress.

Summary and Conclusions

A description of the problems facing the operators was expressed a year before the accident. A TMI operator, addressing problems experienced during an April 23, 1978, reactor trip, stated in a letter to his supervisor:

> I feel that the mechanical failures, poor system designs and the improperly prepared control systems were very much more the major cause of this incident than was operator action. Although training is always essential and welcome—nothing we study or learn to practice could have prepared us for this unfortunate chain of events.... You might well remember this is only the tip of the iceberg and the best operator in the world can't compensate for multiple casualties which are complicated by mechanical and control failures.[25]

Our analysis has documented that many of the operator actions can be attributed not only to the poor quality of instrument displays and inadequate control room design but also to improper operator training and inadequate emergency procedures. We believe that the system that permitted these deficiencies must share a large part of the responsibility for the operator actions at TMI-2.

The Essex Corporation Study reached a similar conclusion.

> The overall conclusions are: (1) operators did commit a number of errors which certainly had a contributory if not causal influence in the events of the

accident; and (2) these errors resulted from grossly inadequate control room design, procedures, and training rather than from inherent deficiencies on the part of the operators.[30]

3. CONTROL ROOM DESIGN

a. Requirements and Criteria

The AEC (NRC) review and approval of the application for a construction permit, submitted by Met Ed for TMI-2 in April 1968, was completed, and the TMI-2 construction permit (CP) was issued on November 4, 1969.[31] At the time of the CP review, the criteria most relevant to control room design were found in the proposed Appendix A to 10 C.F.R. Part 50, "General Design Criteria for Nuclear Power Plants Construction Permits."[32] Typical examples of these criteria indicate that Federal regulations for control rooms were vague, lacked specificity, and contained little, if any, indication of concern for human engineering issues associated with the interface between operators and the control room. For example, criterion 12[32] requires that "instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges." Another example is criterion 11, which states in part:

> The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled.[32]

Although these criteria were only proposed by the AEC at the time, they were published with the notation that they "would not add any new requirements, but are intended to describe more clearly present Commission requirements. . . ."[32] Thus they were, in effect, AEC requirements. In addition to these Federal regulations, the industry also developed standards that could have affected the human engineering of the TMI-2 control room. One example cited in the TMI-2 PSAR[33] was IEEE standard 279,[34] which required that:

> If the protective action of some part of the system has been bypassed or deliberately rendered inoperative for any purpose this fact shall be continuously indicated in the control room.[33]

The thrust of this standard was to provide an effective means of warning operators of an inoperative system. However, this standard applied only to the reactor protection system (a system for rapidly shutting down the reactor in the event safety limits are exceeded) and not to other safety systems such as the emergency core cooling system (ECCS).

Another industry standard that exhibited a concern for human engineering was IEEE standard 603 "Criteria for Safety Systems for Nuclear Power Generating Stations"[35] This standard applied to other safety systems besides the ECCS and suggested that the display instrumentation provided for the manually initiated protective actions required for a safety system should be considered part of the safety system; furthermore, that design should minimize the possibility of anomalous indications that could be confusing to the operator. However, unlike IEEE-279, this standard was not required for the control room design and was not cited in the TMI-2 PSAR.

In addition to these standards, Section 7.4 of the TMI-2 PSAR outlines the general philosophy to be used in designing the TMI-2 control room. Similar to the standards just described, this general design philosophy contains only a vague and general reference to the man-machine interface problem.

Section 7.4 states that all controls and instruments be located in one room. This room was to be designed so that one operator would suffice during normal operations. During "other than normal steady state operating conditions," other operators were to be available to assist the control operator. This section also contains general prescriptions for the shape of the control room; the relative placement of various systems; a brief description of an audible alarm system; requirements to allow occupancy during abnormal conditions such as fire protection, radiation shielding, and ventilation; provisions related to evacuation of the control room; and provisions for auxiliary control stations.

The final portion of Section 7.4 provides a typical example of the general nature of the specifications provided in the PSAR and the limited extent to which they addressed the human engineering problems. Section 7.4.7 "Safety Features" states in part:

> The primary objectives in the control room layout are to provide the necessary controls to start, operate, and shut down the nuclear unit with sufficient information display and alarm monitoring to insure safe and reliable operation under normal and accident conditions. Special emphasis will be given to maintaining control integrity during accident conditions. The layout of the engineered safety features section of the control board will be designed to minimize the time required for the operator to evaluate the system performance under accident conditions. Any deviations from predetermined conditions will be alarmed so that the operator may take corrective action using the controls provided on the control panel.[36]

From 1970 to 1978, the number of requirements and guidance related to control room design increased significantly within both the AEC-NRC and the nuclear industry. As shown in the Essex report,[21] a large number of these criteria were related to human engineering. Although these requirements and guidelines provided more substance than previously existed, the majority of these criteria still suffer from the same deficiency identified previously. That is, they were too vague and too general to require the direct application of human engineering technology that had been extensively developed and used as requirements in other fields.[37]

During this time period, the NRC issued various documents containing recommended practices or guidance in safety matters. These included reactor technology memoranda followed by safety guides and then regulatory guides. In 1975, the NRC consolidated its criteria in a standard review plan[38] aimed at providing guidance to its technical staff who review and approve applications for nuclear powerplant licenses.

The more substantive of these criteria pertaining to human factors considerations include the following:

- Requirement of IEEE-279 that bypasses be indicated was expanded in Regulatory Guide 1.47 "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems"[39] to include safety systems.
- Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident,"[40] included a provision for analysis of what instruments are required. This provision could have conceivably given rise to a requirement for a task analysis. That is, a description of what functions need to be done, initially on a time-line basis, and what aids (including instrumentation) are needed to optimize the man-machine relationship. However, the regulatory guide was not interpreted by the NRC or the industry to cover the use of a task analysis.[41]
- Regulatory Guide 1.114, "Guidance on Being Operator at the Controls of a Nuclear Power Plant,"[42] also provides insight into NRC regulatory attempts to address the man-machine interface. The basic thrust of this regulatory guide is to place the responsibility for safe operation of the plant on the control room operator. It assumes that the control room is properly configured and the operator will be provided all the aids needed to perform his job.

For example, the guide states:

> The operator at the controls of a nuclear power plant should have an unobstructed view of and access to the operational control panels, including instrumentation displays and alarms, in order to be able to initiate prompt corrective action when necessary, on receipt of any indication (instrument movement or alarm) of a changing condition.[42]

and that:

> The operator at the controls should not normally leave the area where continuous attention (including visual surveillance of safety-related annunciators and instrumentation) can be given to reactor operating conditions and where he has access to the reactor controls. For example, the operator should not routinely enter areas behind control panels where plant performance cannot be monitored.[42]

Analysis of the control room at TMI-2 and operator actions performed during the early stages of the accident clearly suggest, for example, that the TMI-2 unit was not designed so that operators would have an unobstructed view of instrumentation displays and alarms. Furthermore, operators had to enter the area behind reactor controls to observe the reactor drain tank instrumentation critical to an assessment of the accident.

The Essex Corporation conducted a detailed review of these regulations, regulatory guides, and the standard review plan, and found no examples of criteria written with a clear intent to include human engineering considerations in the licensing and regulatory system.

The expansion in guidance related to human factors from pre-1970 to pre-1978 that was experienced by the AEC-NRC also occurred in the codes and standards of the nuclear industry. The Essex Corporation found that a significant number of industry standards relating to human factors were developed during this time. As in the other cases discussed, however, few of these standards were thought to be important by those at whom they were aimed. The standards were too vague to require effectively the application of human engineering in the design process. They were narrowly drawn guidelines addressing a specific component or group of components and did not adequately address the man-machine system interface problems.

The most significant industry guidelines in existence during the operating license review of TMI-2 are found in IEEE standard 566, "Recommended Practice for the Design of Displays and Control Facilities for Central Control Rooms of Nuclear Power Generating Stations."[43] This standard contains guidance directly related to human engineering, but the Essex Corporation's review of it found serious deficiencies. The Essex Corporation noted that the standard was incomplete and that it did not include guidance on the use of some very important human factors tools[44] such as:

- analysis of the tasks operators must perform;
- the use of existing human engineering standards;[37]
- control and display layout conventions; and
- alarm placement rules.

The Essex Corporation concluded that the generalizations, ambiguities, and oversights of IEEE 566 result in little more than an admonishment that the designer consider the operator, with little guidance on just how to prevent operator error.[45]

Nearly all of the industry standards were published after the application for the operating license for TMI-2 had been submitted to the NRC in 1974. Thus, none of the more recent standards were applied to the TMI-2 design except as deemed necessary by the NRC or the utility to address significant safety issues.

Conformance of TMI-2 to Human Factors Criteria and Standards

As noted previously, the TMI-2 design was found by the AEC to meet the applicable criteria prior to issuance of the construction permit in 1968. Furthermore, the design development by the utility and its contractors and the review of this design by the AEC were conducted with essentially no human engineering considerations. Thus, NRC found that TMI-2 satisfied the existing criteria even though a review of the current design today by human engineering specialists against these limited criteria would find serious deficiencies.

When a nuclear powerplant application is received by the NRC for an operating license, the practice has been to require conformance of the design to the criteria specified at the time the construction permit is issued and to address the necessity for meeting subsequent criteria on a case-by-case basis. The necessity to conform to post-CP criteria is determined by the NRC and the industry on the basis of a perceived level of safety improvement that can be achieved by such conformance. Given the absence of any human engineering expertise on the NRC staff, it is not surprising that the NRC had no perception that human factors criteria could improve safety.

In summary, we found a lack of substantive human factors criteria and guidance both within the NRC (AEC) and the nuclear industry, and more important, a lack of appreciation for the importance of human factors to the safe operation of nuclear powerplants. Furthermore, the personnel resources to employ human factors techniques that would be required to implement even the existing criteria did not exist within the NRC and were limited within the nuclear industry.

b. The TMI-2 Control Room

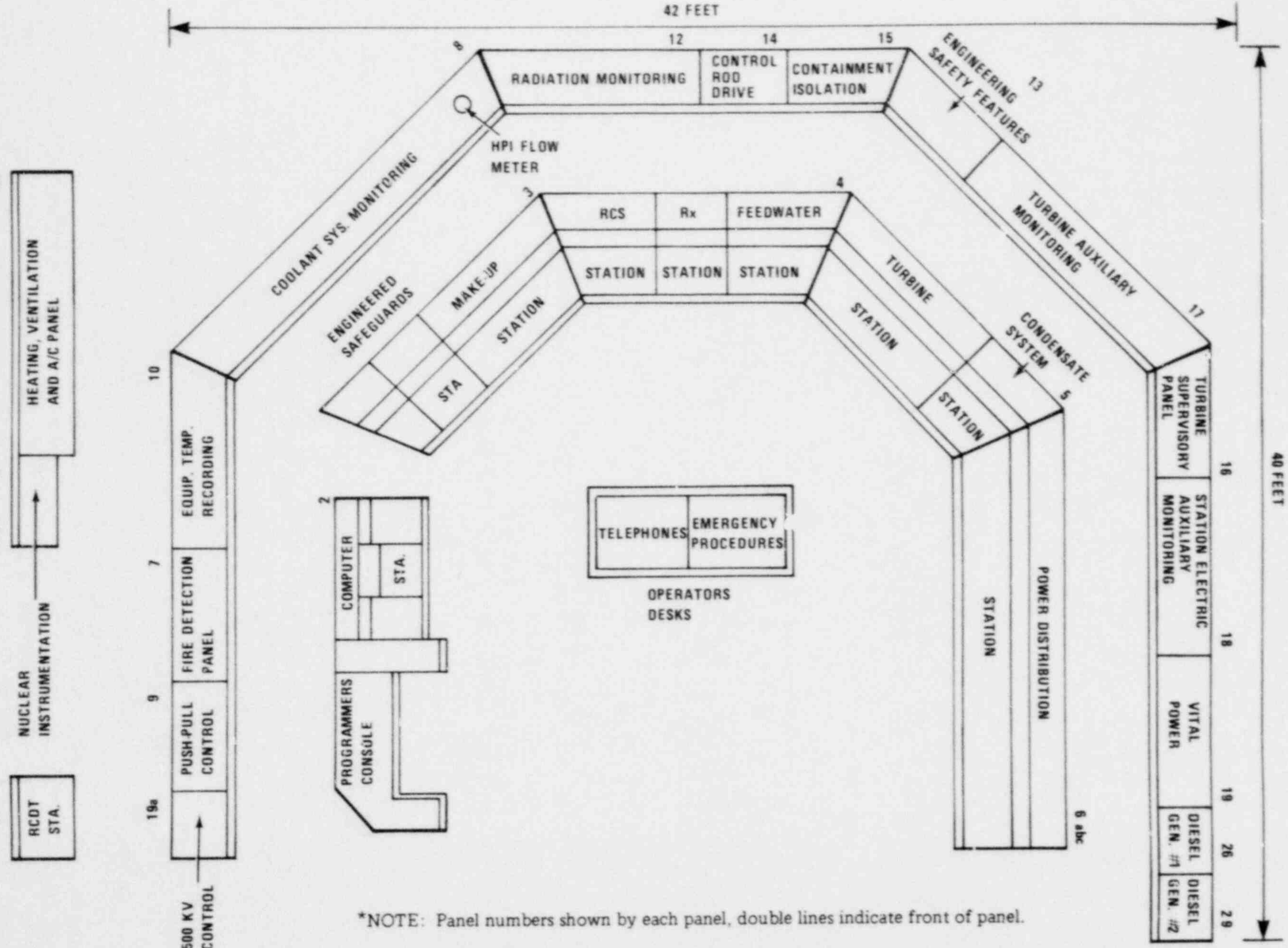
General Layout

At the TMI-2 nuclear powerplant, the control stations, switches, and indicators necessary to start up, operate, and shut down the nuclear unit are located in one control room. Controls for certain auxiliary systems are located at remote control stations.

As can be seen from Figure II-47 and the photograph in Figure II-48, the TMI-2 control room is spacious and contains a large number of instruments, controls, and alarms. The control room consoles are arranged in a U-shaped pattern with vertical panels following the same pattern behind the consoles, separated by a passage aisle. The operator's desk is located in front of the U-shaped console and panel arrangement. These figures show the floor plan and layout of the control room and give an idea of its size.

According to the TMI-2 Final Safety Analysis Report (FSAR),[46] the control room was to be designed so that one man could supervise operation of the unit during normal steady-state conditions. During abnormal operating conditions, additional operators are expected to be available for assistance. The control room is arranged to include the operating consoles, which house frequently used controls and indicators, as well as start-up and emergency controls and indicators. The FSAR also notes that the controls and indicators were to be located in a logical arrangement, accessible, and readily visible to the operator. Recorders and radiation monitoring equipment, infrequently used control switches, remaining indicators, temperature recorders, annunciators, and reactor building isolation valves position indicators are mounted on the vertical panels behind the consoles. Table II-62 lists the descriptions of the panels that were most important during the March 28, 1979, accident.

Visible and audible alarm units are incorporated into the control room to warn the operator of unsafe or abnormal conditions. The control room was supposedly designed so that information readouts contain all the indications required by the operator for monitoring conditions in the reactor, reactor coolant system, containment, and safety-related process systems throughout all operating conditions of the plant.[46]



*NOTE: Panel numbers shown by each panel, double lines indicate front of panel.

FIGURE II-47. TMI-2 Control Room Layout*

FIGURE II-48. TMI Control Room (Postaccident)

Plant Computer

The plant computer system is used for monitoring alarms, plant performance, logging data, and performing simple calculations and is located near the center of the control room on console one. This system uses a Bailey 855 computer which is linked to a smaller NOVA computer. The NOVA computer was added to the original design to provide more capacity for monitoring the balance-of-plant conditions.

The computer has two output modes—an alarm printer and a utility printer. Both printers are automatic typewriters, and if either fails, its output is automatically transferred to the other. A small cathode ray tube display duplicates the output of the printers or can be used for independent display.

For all monitored parameters that have an alarm function, the alarm printer automatically prints an alarm message when the parameter has gone into an alarm condition. The computer also samples each parameter, such as temperature, pressure, and level, and compares the reading to a preset alarm value. If the reading is outside acceptable limits, a notation to that effect is typed out on the alarm printer. When the parameter again comes within acceptable limits, another notation is typed. The alarm printer also makes a record of starting, stopping, or tripping of major equipment.

The computer alarm printout is capable of typing only one line about every 4 seconds. Consequently, in situations where alarms are initiated rapidly, the printer is unable to keep up and the alarm printout is delayed. An operator can bring the printout up to real time, but only at the cost of clearing all alarms awaiting printout from the computer memory. At one point during the accident, the alarm printer was more than 2 hours behind.

The utility printer provides output on request. The value or condition of any monitored parameter can be requested. Special subroutines allow the operator to request output values in specific preprogrammed groups called Operator Special Summaries or to trend output values in preprogrammed groups called Operator Group Trends.

The computer is also programmed to record automatically all changes in state of a predesignated group of parameters called Sequence of Events inputs. These event inputs are stored in the computer and can be printed on request. The sequence is

TABLE II-62. TMI-2 control ro' m key panel descriptions*

Panel	Description
2	Computer console.
3	Reactor coolant makeup and purification system and the control room equipment related to the safety features actuation system.
4,5,6	Controllers, recorders, and indicators necessary for control and supervision of the reactor power output, feedwater, condensate, steam generators, and turbine generator.
7	Indicates a fire in the unit and the automatic steps being taken to control it.
8	Annunciators and indicators for status of the various nuclear and conventional cooling systems of the unit.
8a	Reactor coolant drain tank controls, indicators, and alarms.
10	Records temperatures of major equipment, reactor vent valves, control rod drives, and self-powered neutron detectors; each temperature monitored is alarmed if the temperature exceeds a preset limit.
12	Station radiation monitoring equipment and recorders, including equipment required to annunciate and indicate the status of equipment and interlocks intended to prevent any release to the environment that exceeds preset limits.
13	Status of the engineered safety features panel.
14	Individual control rod positions, fault lights, and inserted and withdrawn limit lights.
15	Graphic panel that shows the position of all reactor building isolation valves.

*Panel numbers refer to those shown in Figure II-47.

started by any one of the Sequence of Events inputs changing state and continues until called up by the operator.

The plant computer provides the operator with an efficient means of keeping logs and showing trends on a large number of plant parameters under normal operating conditions. The computer was not designed to accommodate the operator's data needs during an accident situation. Using the computer in an accident situation requires that the operator leave his control panels to request computer output; it takes the computer several seconds to supply the requested output, and as noted, the automatic alarm printout can be several minutes or even hours behind real time. All of these factors tend to limit the computer's usefulness in an accident situation. If properly designed and programmed, the computer could provide information useful for diagnosing and responding to an emergency situation. However, the TMI-2 computer was not programmed to establish a hierarchy of critical parameters to be monitored in the event of an emergency. Thus, during the March 28, 1979, accident the large number of unimportant alarms and the resulting backlog made the computer nearly useless as a diagnostic tool.

TMI-2 Control Room Design Evaluation

The likelihood of operator errors can be reduced by the systematic integration of human factors engineering into the planning and design of a plant. To determine the extent to which TMI-2 was designed to prevent or minimize operator errors, the Essex Corporation evaluated the TMI-2 control room and compared it with human factors engineering criteria and guidelines generally applied in other industries. The following discussion of human engineering aspects of the TMI-2 control room design has been divided into categories that reflect different aspects of the design. They summarize the findings of the Essex report.[36]

Workstation Design

A fundamental tenet of human factors engineering is that workstation design should facilitate operator performance and reduce the probability of operator error. To accomplish this, controls and displays should be logically organized according to function or sequence or in relation to the system they control (i.e., mimic). Furthermore, controls should be placed to minimize the operator's need for reaching and to shorten the visual span between the operator and the instruments the operator must read, thus reducing time to locate and manipulate specific controls or displays.[47]

The Essex Corporation found that little, if any, attention was paid to this aspect of workstation layout. Apparently no analysis was made of the tasks that must be performed at the various TMI-2 workstations or the capabilities and limitations of the operators performing such tasks. The following deficiencies are indicative of their findings:[48]

- In many cases, workstation design appears to maximize visual scan, reach, and walking requirements.
    - —RC pump seal pressure is on panel 10, and seal temperature is on panel 8, but the pump controls are on panel 4.
    - —Makeup control is on panel 3, but makeup flow indication is displayed on panel 8. See Figure II-49.
- Controls and displays are not logically or consistently sequenced.
    - —Pressurizer heater controls are sequenced from right to left rather than from left to right. See Figure II-50.
    - —Pressurizer narrow range indicators are B, A instead of A, B
- Indicator lights are inconsistently placed above, beside, or below their associated controls. See Figure II-51.

Reaching over benchboards to actuate switches or to manipulate recorders not only obscures the displays under the reaching operator but also increases the risk that the operator will unintentionally actuate a switch. Frequently, it prevents the operator from monitoring important displays during switch operation.[48]

The Essex Corporation examined the benchboards and the attached vertical panels in TMI-2 for reaching requirements. The levels of excessive reach requirements were defined by using the stature of the fifth percentile male (street clothes) as a basis.

They found that 18 chart recorders, 10 control stations (10 switches) and 31 switches (most with frequent use) required a reach of 10 to 14 inches greater than that of the fifth percentile male standing erect, requiring him to bend over the panel to actuate the control or switch.[48]

Control and Display Design

Poor selection of controls and displays can impede the performance of tasks assigned to a particular workstation. The Essex Corporation evaluation of the TMI-2 control room identified several such deficiencies in the control and display design at TMI-2.[49] Examples include the following:

- Controls were selected without regard for the relationship between size and performance. As a consequence, many controls (e.g., "j-handle" switches) are unnecessarily large and require extensive panel space.
- Displays were selected without concern for the information processing requirements of the operator. As a result, rarely used or noncritical displays (e.g., electrical displays on panel 6) are unnecessarily large and prominent in the workspace, whereas critical displays (e.g., pressurizer level) are smaller and less easily seen.
- Bulbs are difficult to change in pushbutton-legend light control indicators—in some cases resulting in shorting out of switches. (Note: Control room operators stated that the process is so unmanageable that they generally wait until the plant is shut down before attempting to replace burned out bulbs.)
- Auditory displays associated with annunciators are not prioritized to assist the operator in discriminating critical alarms.
- Controls having common operating modes (i.e., automatic and manual) are not designed so that mode selection is consistent between controls. In some cases, controls having similar functions are turned clockwise to place the system in manual, and in others, counterclockwise. See Figure II-50.

Displays

A critical design requirement for the nuclear powerplant control room is the effective display of information to the operator. This requirement is most pronounced during emergency conditions

FIGURE II-49. Visual Scan Necessary for Operator (on Left) Controlling Makeup To Monitor Makeup Flow (Operator on Right)

where prompt, accurate diagnosis of a problem may be critical. To perform tasks effectively, the operator must have immediate access to information regarding all system parameters reflecting plant status; the information must be easily seen and read, well organized, and unambiguous.

The Essex Corporation found that "the design of the TMI-2 control room evidences a patent disregard for the information processing requirements of the operator."[50] The following serves to underscore the magnitude of this problem:[51]

- In some cases, the status of critical parameters must be inferred from changes in associated parameters.
  - —There is no displayed indication of emergency feedwater flow.
  - —There is no displayed indication of flow through the pressurizer relief valve discharge line.
  - —There is no displayed indication that the reactor coolant system has reached saturation conditions.
- Displays are incorrectly located, both with respect to their associated controls as well as the operator's optimal field of view.
  - —RC pump vibration-eccentricity indicators and alarms are on the back of panel 10, approximately 20 feet from the RC pump controls on panel 4.
  - —ESF indicator board on panel 13 consists of 16 rows of indicator lights. Due to placement and organization of this panel, a 6-foot-tall operator can see only eight rows of lights from his normal operating position. See Figure II-52.
  - —RCDT instrumentation is located on panel 8A, which is completely outside the main operating area. See Figure II-47.

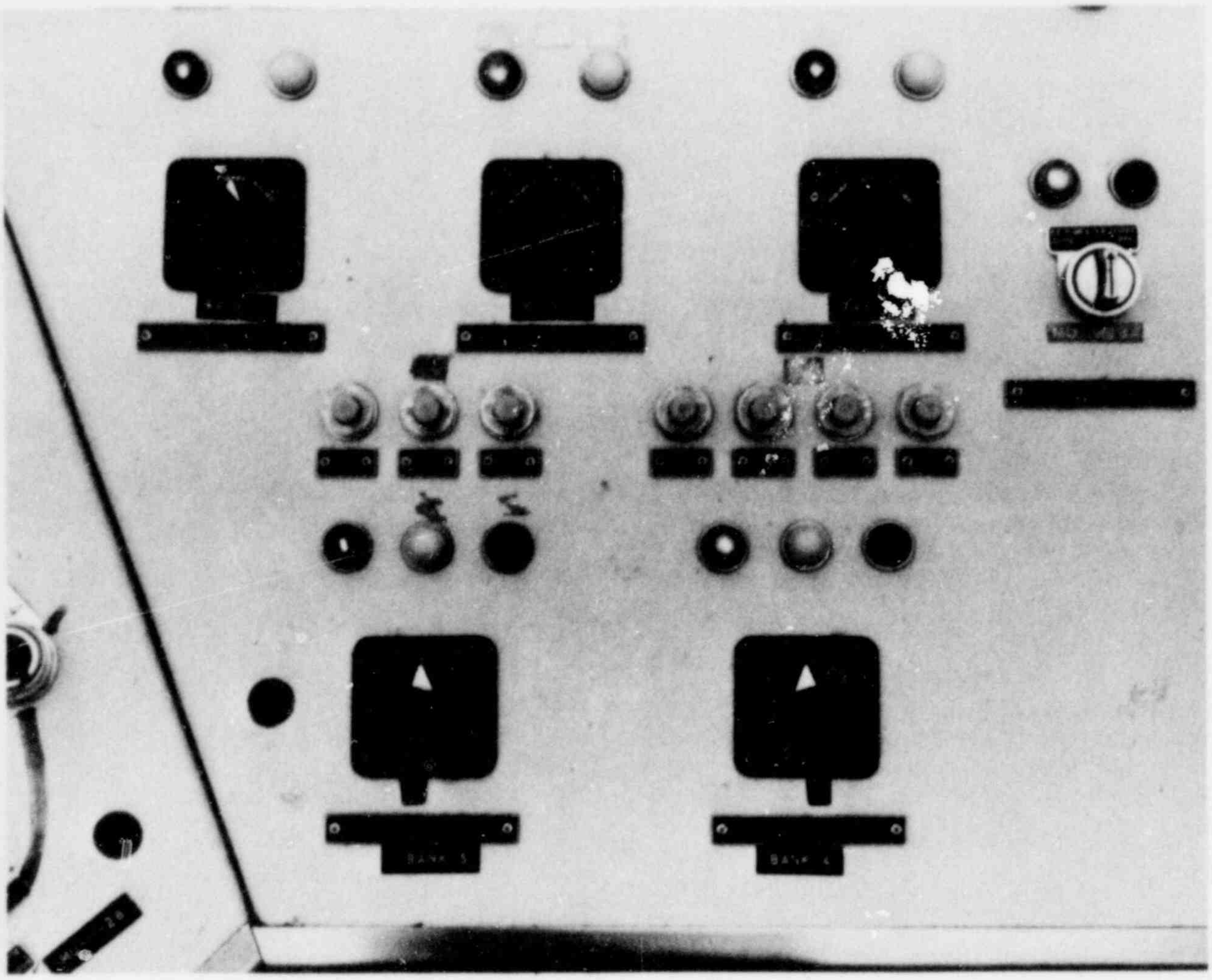


FIGURE II-50. Pressurizer Heater Control*

*Note the right to left sequence and inconsistency of control movement to auto and manual.

- Information is inadequate and/or ambiguous, making precise determination of plant status difficult or impossible.
  - —Strip charts are overloaded, in some cases displaying up to 72 separate channels on the same chart.
  - —Critical controls have no obvious indication of being in manual (e.g., when the pressurizer spray valve is set to manual, the handle is "up" (out), but the point is at "AUTO").
- The annunciator system, which includes over 750 annunciator lights (some of which are outside the main operating area, e.g., RCDT panel), is poorly organized both in terms of grouping and relationship of alarms to associated subsystems. In addition, critical alarms have not been color coded or otherwise prioritized to permit immediate identification. In many cases, legends are excessively wordy or contain inconsistent abbreviations, increasing the time required to ascertain their meaning. See Figure II-53 for an example of one alarm panel out of some 20 of similar size.
- Extinguished lights are used as positive indication of system status (e.g., PORV seated).
- Displays on several panels were evaluated against standard human engineering display criteria. Some 89 deficiencies were found in evaluating three systems on panel 4.

Parallax

In the TMI-2 control room, moving-pointer and arc-scale vertical indicators are used extensively. Unless these indicators are viewed on a line passing through the pointer and perpendicular to the scale plate, parallax problems will occur. This parallax problem will produce a difference between the actual and the perceived indicator reading. With vertical

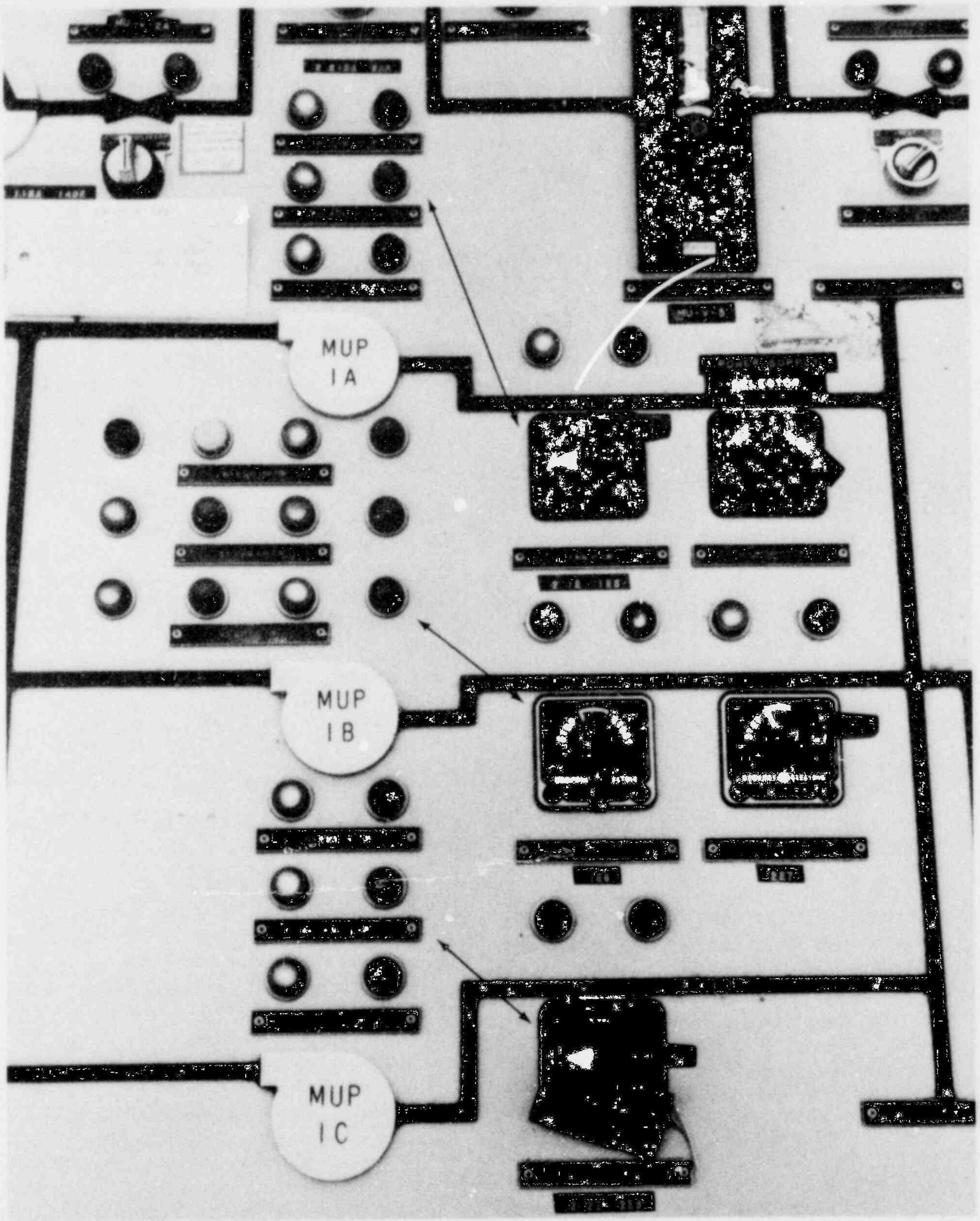
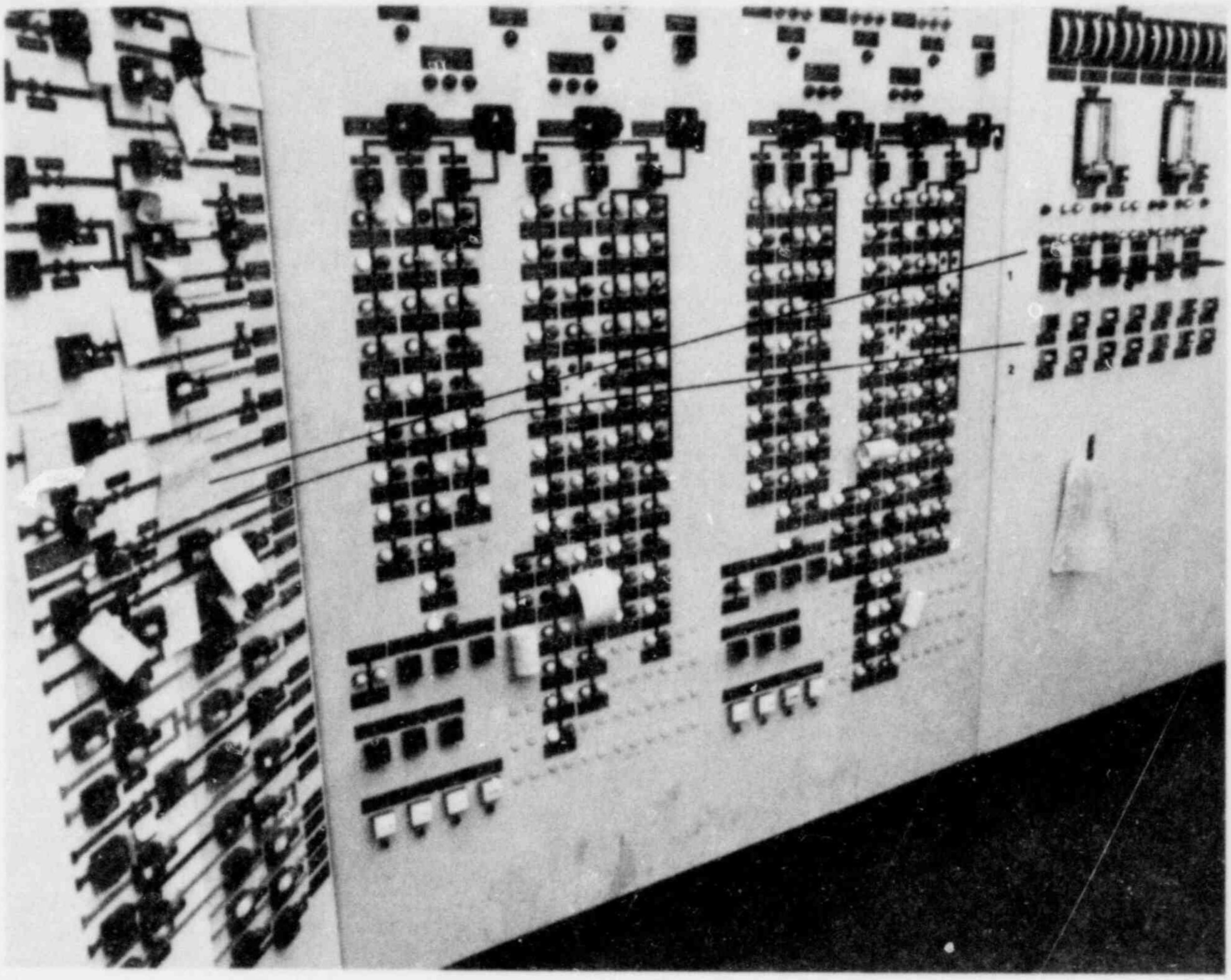


FIGURE II-51. Relationship of Makeup Pump Controls and Indicator Lights



1. INDICATION BELOW THIS LINE CANNOT BE SEEN BY A 6 FT. OPERATOR STANDING AT THE ESF OPERATING STATION

2. INDICATION BELOW THIS LINE CANNOT BE SEEN BY A 6 FT. OPERATOR FROM THE CLOSEST POSITION IN FRONT OF THE CONTROL CONSOLES

FIGURE II-52. ESF Station Indicator Board

indicators, parallax error will occur when the indicator is placed too low on the panel.

Aside from placing the vertical indicator on the panel so it can be read easily, parallax can be minimized by using a mirrored backing so that the operator will know that his reading is accurate when the pointer is lined up with its scaled image.

The parallax survey conducted by the Essex Corporation identified 115 vertical meters in the primary area above the eye level of the fifth percentile male, none of which had mirrored scales.[52]

Obscured Displays

In its evaluation of the control room at TMI-2, the Essex Corporation found that the vertical panels behind the benchboard contain about 1900 displays, including indicator lights. Depending on their mounting height, displays on the vertical panels can be obscured by the vertical portion of the front benchboard from viewing by an operator standing at the benchboard.[51]

The Essex Corporation found a large number of displays below the line of sight of a fifth percentile male standing at the benchboard and looking directly at the vertical panel. Specifically, the following were obscured.[51]

470 indicator lights
24 legend switches
3 control display units
3 vertical indicators

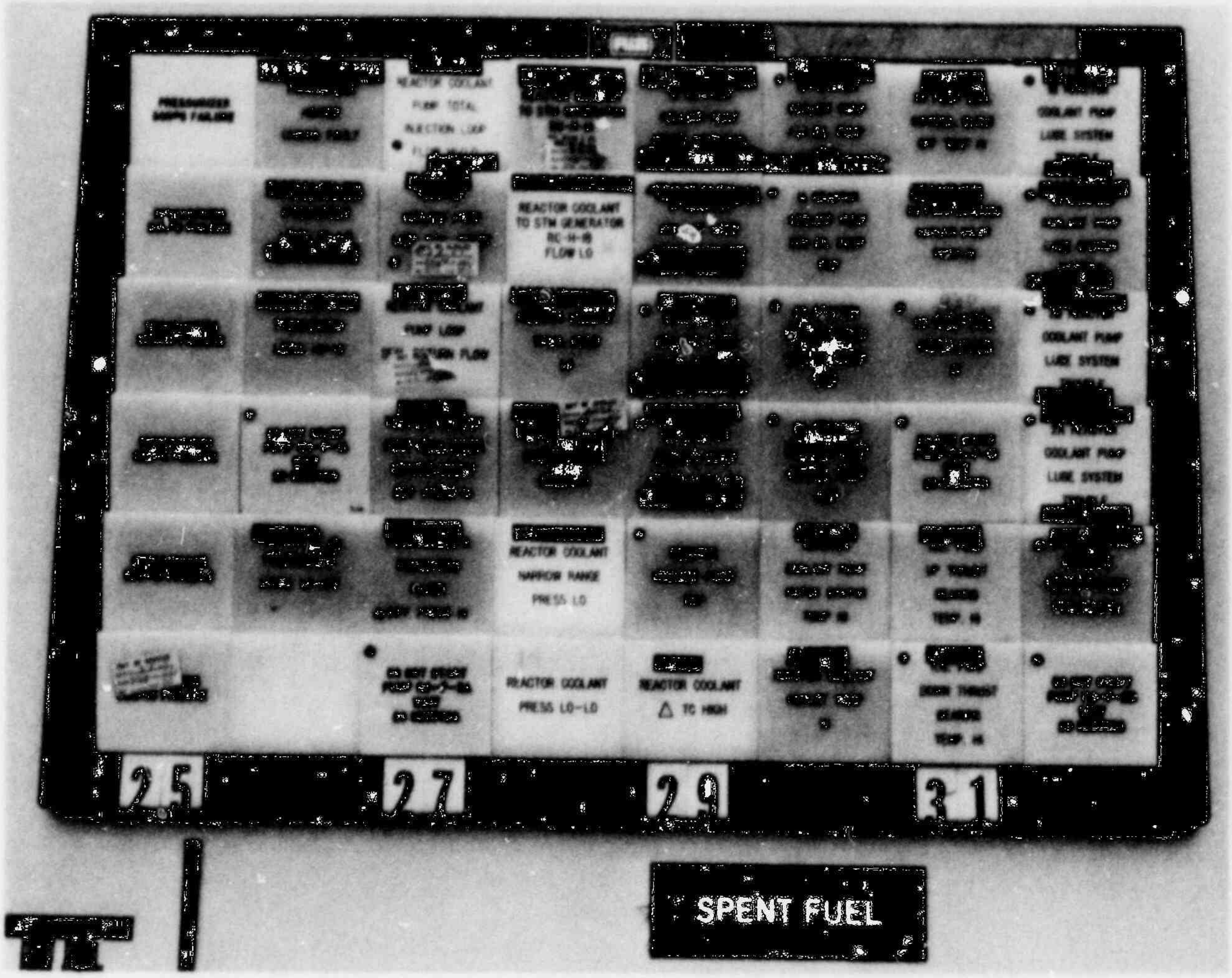


FIGURE II-53. Typical TMI-2 Alarm Panel

1 strip chart
1 dial
1 counter

Viewing Distance

Although the Essex Corporation did not have the opportunity to conduct a thorough analysis of display viewing distance, indications are that the TMI-2 control panel presents many opportunities for misreading displays. For example, at least 250 meters are located on vertical panels that must be viewed from a minimum reading distance of about 10 ½ feet from the primary benchboard.[53]

Labeling

Labeling, although actually a subset of information display, has unique characteristics and requirements and significantly affects operator performance. To ensure efficient, accurate operator performance, labeling must be consistent in location with respect to associated controls and displays; characters must be of adequate size to be read easily from the operator's normal operating position (for normal 28-inch viewing distance, ⅛-inch characters should be used); coding and abbreviations must be consistent throughout the system; and labels should be graduated in size.[54]

Labeling used in the TMI-2 control room was judged by the Essex Corporation to be inadequate in a number of areas, including the following:[54]

- Labeling on back panels is difficult or impossible to read from main operating positions.
- Labels are inconsistently placed in relation to their associated controls and displays; 34% of the labels were located above associated components and 55% were located below.

- Labels do not always correspond to their associated indicator lights (e.g., diesel fire pump labeling contradicts its indicator lights).
- Labels and markings on several panels were evaluated against standard human engineering criteria. Some 62 deficiencies were found in evaluating three systems on panel 4.

Labeling is often treated as an adjunct to control panel design, rather than as an important communications link necessary for the efficient and reliable operation of the plant.

Color Coding

The Essex Corporation noted that human engineering criteria, developed by the military and aerospace industry, is at odds with the color coding practices evidenced at TMI-2. The design of the TMI-2 control room sharply reduced the value of color coding to the operator. The number of meanings associated with each color as well as the number of colored lights "combine to produce considerable ambiguity in the man–machine communication link."[55]

The color coding deficiencies noted by the Essex Corporation include the following:[56]

- Many different meanings were given to each color: for red, 14 titles; for green, 11; and for amber, 11.
- Annunciators, when alarming, intend to draw attention to the window of interest. TMI-2 uses flashing white on a white background. Contrast is particularly bad if several lights around the alarming window are on.
- The "Christmas tree" effect in the control room is overwhelming to the observer and must be distracting and, at times, confusing to the operator. The number of lights make it virtually impossible to determine with confidence, the status of any switch or system from across the control room, particularly if the component is benchboard mounted.

The Essex Corporation conclusions concerning the control room design are summarized as follows:[57]

- The TMI-2 control room was designed and built without an appreciation of the needs and limitations of the operator, particularly during emergency situations.
- In the absence of a detailed analysis of information required by the operators, some critical parameters were not displayed, some were not immediately available to the operator because of location, and the operators were burdened with unnecessary information.
- The control room panel design at TMI-2 violates a number of human engineering principles resulting in excessive operator motion, workload, error probability, and response time.

c. Control Rooms at Other Plants

Evaluation of Specific Plants

To assess the adequacy of the application of human factors principles to control room (CR) design in the nuclear industry and to compare these CRs with the TMI-2 CR, the Essex Corporation studied two additional plants.[58] The plants chosen for the investigation were Calvert Cliffs 1 and Oconee 3. Both of the plants are pressurized water reactors of approximately the same power output and the same vintage as TMI-2. However, these plants had different architect-engineers, and utilities, and the management philosophy utilized in the CR design was different from that employed at TMI-2. At TMI-2, the CR layout was the responsibility of a senior engineer on the staff of the architect-engineer and all decisions were made by him. On the other hand, Calvert Cliffs 1 and Oconee 3 were designed by a management and operator team. No changes were made to the CR or indicator arrangement without approval by the management and operator team after all had an opportunity to criticize the change. Furthermore, these two CRs were developed with the aid of mockups.[59]

The comparison between TMI-2 and the other two plants included a human factors assessment of features, such as reach and visibility, and the placement and readability of meters and indicators in the control rooms.

The ability of the control room operators to reach controls easily and see displays from operational distance is basic to reliable and timely performance. The reach survey of the control room indicated that Calvert Cliffs was better than the other two. It had fewer switches and controls beyond the reach of the fifth percentile male standing at the control boards. Oconee was the worst offender, having some 22 recorders and 113 switches and controls beyond 10-inch reach of the fifth percentile male. In the TMI-2 control room, 18 recorders and 41 switches were beyond the 10-inch measurement.[60]

The parallax survey of the three plants focused on vertical meters in the primary area above the eye level of the fifth percentile male. Oconee was better than the other two, having only one indicator above the limit, while Calvert Cliffs had 75 indicators above the level; however, to minimize the parallax problem, all had mirrored scales and 25 of these had limit switches. TMI-2 had 115 vertical indicators above the eye level, none of which had mirrored scales or limit switches.[61]

Depending on their mounting height, displays on the vertical panels can be obscured by the vertical position portion of the benchboard from viewing by an operator standing at the bench. To determine the degree to which displays are obscured, those displays were counted that were below the sight of a fifth percentile male standing at the benchboard looking directly at the vertical panels were counted. Calvert Cliffs and Oconee were better than TMI-2 in this regard. Calvert Cliffs had no obscured displays, and Oconee had only two obscured indicator lights. In the TMI-2 control room, there were 470 indicator lights obscured, as well as a number of other switches and indicators.[62]

It seems clear that the TMI-2 design gives insufficient attention to the requirements for reach and visibility. Under normal conditions, operators are likely to compensate for design inadequacies such as these. However, under pressure, the operators may take risks with reaching and display reading because of time constraints that could compound the problem.

The three plants were also compared for the adequacy of the aids, such as labels, color coding, and procedures, provided for the CR operator and for the means to display the procedures provided to assist the operator in running the plant.

The Essex Corporation survey of CR labeling found significant and comparable deficiencies at all three plants.[63] For example, labels were left off some components, not attached in any consistent order, and so poorly planned that 34% to 65% of the panel components needed backfits.

In evaluating the color code practice, it was found that all three plants attached several meanings to each color used.[64] In fact, the operator in many cases would have to know the specific component being observed to know how to interpret the color, and in many instances the colors have contradictory meanings.

A summary of the results of the Essex color survey are shown in Table II-63.[64] As can be seen, the TMI-2 control room attached more meaning to each color than do either of the other two plants.

In summary, the Essex Corporation's limited review of the features that aid the operator in reliability and timely performance showed Calvert Cliffs 1 and Oconee 3 to be superior in human engineering to TMI-2. Despite their good features, however, Oconee 3 and Calvert Cliffs 1 had some shortcomings and a detailed analysis would no doubt uncover more.

TABLE II-63. Number of different meanings given to each color

	Red	Green	Amber
Calvert Cliffs	6	4	5
Oconee-3	4	3	4
TMI-2	14	11	11

Evaluation of Additional Plants

In light of the advancement in human factors in the aerospace industry at the time that the three plants were being designed, it appears that none took advantage of the technology available. The limitation of the Essex Corporation study to the two additional nuclear powerplants does not permit a conclusive decision as to the state of nuclear powerplant control rooms in general. Therefore, we reviewed the EPRI[65] study of five additional powerplants and the Sandia Laboratories analysis of the Zion nuclear powerplant.[66] In November 1976, the Electric Power Research Institute (EPRI) published a report, EPRI NP-309,[65] of a 16-month study of five nuclear powerplants. EPRI had contracted with the Lockheed Missiles and Space Company, Inc., of Sunnyvale, Calif., to conduct the study and write the report. The intent of the study was to uncover general problem areas where human factors guidelines could profitably be applied to the next generation of nuclear powerplants. A secondary objective was to identify problems within existing powerplants where minor modifications at low cost would upgrade the quality of the man–machine interface. A review of this study allows a better evaluation of the TMI-2 control room design in comparison with the state of the art in the nuclear industry and permits a better evaluation of the nuclear powerplant CR design in general.

The EPRI study made the following findings:

- Insufficient attention was paid to the abilities and limitations of the operator in developing the control room configuration. Serious difficulty in the plants' normal and emergency operations resulted from the poor positioning of controls and instruments on back or remote panels requiring the operators to leave their primary operating stations to use these controls or monitor these instruments. In addition, the study found that four of the five plants were inadequate because of glare and reflections on instruments.[67]

- In general, the control board designs were too large, requiring too great a visual and control span for the operators, and were not optimized for minimum manning. Control boards had arrays of identical components that were not discriminated into clearly identified panels and subpanels containing related elements. Closely related controls and displays were often widely separated. Although some mimicking was provided by the designer, there usually was not enough to satisfy the operators so that some attempt was made to modify panels with tape to superimpose mimic logic.[68]
- Although no data on the physical dimensions of typical control room operators were available, the placement of instruments was too high or too low for convenience. This problem was predominant on the back panels and peripheral consoles. Footstools and ladders were often required to permit the operators to reach these controls and displays.[69]

  Placement of controls made them susceptible to accidental actuation. Adjacent controls having identical appearance, shape, and texture but different functions can cause inadvertent actuation. Placement of some controls makes them susceptible to accidental contact by operators and visitors to the control room.[70]
- Meters currently utilized in nuclear powerplants have tremendous potential for human factors improvements. The most common problems observed in the five plants examined were improper scale markings in association with scale numerals; selection of scale numeral progressions that were difficult to interpret; parallax problems resulting from placing the meters above or below eye level; meters that fail with the pointer reading in the normal operating band of the scale; and glare and reflection from overhead illumination.

  The most serious problem observed in all of the plants was the lack of meter coding to enable the operator to readily differentiate between normal, marginal, and out-of-limits segments of the meter scale.[71]
- Each of the five control rooms had an annunciator warning system consisting of a horizontal band of hundreds of indicators spanning the uppermost segment of the control board. This system was too complex and had become a catchall for a wide variety of qualitative indicators, which compounded the difficulty to diagnose malfunctions. When emergencies occurred, the large number of indicators illuminated, in concert with blaring horns, startling the operators and overloading their sensory mechanisms rather than shedding light on the problems at hand.[72]
- Indicator reliability is a problem in nuclear powerplant control displays. There were a surprising number of burned-out single-lamp indicators at any given time. The replacement of these lamps was difficult and presented problems for the operator. There are examples in the plants of negative indicators (the absence of indication to convey information to the operator).[73]
- The control room designs underutilize coding techniques that could help the operator discern plant status and prevent misidentification of control elements. Color codes have not been applied systematically, and code meanings vary from panel to panel. Present coding of indicators tells the operator whether a valve is closed or open but does not convey any information as to whether the valve should or should not be closed.[74]
- Labels were not placed consistently above or below the panel elements being identified, which could result in misidentification of the panel element. Some labels were obscured by adjacent control levers. The best indication of labeling inadequacies is the extensive handmade labeling that operators add to the consoles to clarify identification of given controls or their operation.[75]

The NRC contracted with the Sandia Laboratories to conduct a study of the Zion nuclear powerplant.[66] The scope of the study was limited to the human factors problems associated with engineered safety panels in the control room and associated procedures for coping with a LOCA. The Sandia report was published as NUREG-76-6503 in October 1975.

Sandia Laboratories reported that in the Zion situation, as in other nuclear powerplants we have visited, little attention was paid by the designers to the human engineering practices that have maximized reliable human performance in other complex systems.[76] The report lists the following design features that deviate from sound engineering practices and are regarded as likely to cause errors:[77]

- poor layout of controls and displays;
- poor and inconsistent color coding;
- too many annunciators;

- too many exceptions to the go-no-go coding scheme for rapid assessment of monitor panel status;
- labeling that provides little or no location aid;
- misleading labeling due to violation of populational stereotypes; and
- insufficient labeling of valves.

It can be seen that the design problems existing at the Zion plant are similar to those discussed in the Essex Corporation report on TMI-2.

Summary

A broader base of investigation might be needed to compare TMI-2 with the state of the art in the nuclear industry in the late 1960s. From the limited study by the Essex Corporation of three plants, the EPRI study of five plants, and the Sandia study of Zion, it can be concluded that the TMI-2 control room is representative of contemporary nuclear powerplants and that there are serious human factor problems throughout the nuclear industry.

4. EMERGENCY PROCEDURES

Introduction

The actions of TMI-2 operators during the accident suggest that emergency procedures were of little use for diagnosing the problem being faced or for deciding on the appropriate corrective actions. This is not surprising since, as the analysis in Section II.E.2 suggests, the written emergency procedures for TMI-2 had serious deficiencies. We did not perform a detailed analysis of all the TMI-2 emergency procedures. We did, through our contract with the Essex Corporation, perform an evaluation of one procedure, 2202-1.3 "Loss of Reactor Coolant/Reactor Coolant System Pressure."[78] Essex also performed an assessment of the impact that procedures had on the accident and of Met Ed's process for developing and updating procedures. The discussion that follows draws substantially from the Essex review.

We feel that one of the two emergency procedures that were most relevant to the situation at TMI-2 was 2203-1.3 "Loss of Reactor Coolant/Reactor Coolant System Pressure." The Essex Corporation evaluation of this emergency procedure from a human factors engineering standpoint revealed a number of deficiencies including:[78]

- The procedure was not complete in several regards:
  - —It failed to define a leak or rupture that is within the capability of system operation.
  - —It lists symptoms but does not address diagnostic procedures and tests.
  - —It indicates that the control room operator (CRO) should monitor liquid levels, reactor building parameters, and safety feature flow rates, but does not indicate acceptable and nonacceptable values.
- The procedure has several content coverage problems, notably:
  - —Step 2.2.2 under A ("close MU-V376 letdown isolation valve and start the backup MU pump if required") does not discuss how to determine if required.
  - —Section 3.2.5 (A) states that continued operation depends on the capability to maintain pressurizer level and reactor coolant system (RCS) pressure above the 1640 psig safety injection actuation setpoint. The procedure completely ignores the situation where level is maintained well above its low level alarm point while pressure is below 1640 psig, the situation that was present from 2 minutes after the accident initiation through the 150-minute point.
- Problems with procedure clarity and conciseness:
  - —Too many subjective statements are used in symptoms, such as "becoming stable after short period of time."
  - —It is not clear if all symptoms must be present, or only some subset, or only one of the symptoms to diagnose the problem.
  - —Section 2.2.2.1 of Section B states that the CRO dedicated to recognizing a LOCA must accomplish four steps within 2 minutes. Step four states that MU pump discharge cross connect valves must be opened within 5 minutes of the LOCA. It is not clear how a step taking 5 minutes must be accomplished within 2 minutes.
- Problems with procedure consistency include:
  - —Nomenclature used in the procedure is consistently different from panel nomenclature, control and display labels, and annunciator designators.
  - —The procedure itself is not internally consistent at times in identifying valves to be

monitored and at other times in omitting such valves.

- Problems with correctness of procedure:
  - —Section B symptoms are not correct. Symptoms for leak or rupture include "rapid continuing decrease of pressurizer level."
- Problems with compliance with ANSI N18.7:
  - —The procedure includes the reactions designated for emergency procedures but totally ignores the sections required for procedures in general, such as the statement of applicability: prerequisites, precautions, limitations and actions, and acceptance criteria.

The Essex Corporation also found that the emergency procedures fail to identify in clear and concise terms what decisions are required of the operator, what information is needed by the operator to make the decision, what actions need to be taken to implement the decisions, and how the operator verifies the correctness of his decision and actions.[79]

The Essex Corporation evaluation of the use of procedures addressed the following:

- accessibility of procedures,
- management of the update of procedures, and
- use of procedures as job performance aids.

Procedures should be written to allow easy identification of which procedure should be followed. The emergency procedures at TMI-2 were not easily accessible. There was no organized listing or catalog of symptoms that would help the operator determine which procedure to apply. The operator is forced to rely on memory. While this approach may be acceptable during normal operations for single-fault situations, the Essex Corporation maintains, and we agree, that it fails miserably in multiple-failure conditions, as was the case at TMI-2 on March 28, 1979.[80]

There was no formal method for getting operator input to update the procedures at TMI-2. Since the purpose of procedures is to aid the operators in controlling the powerplant during normal and emergency operations, the Essex Corporation felt that a mechanism is needed to identify the need for procedure change, to include operator input in the change process, to complete the required change, and to obtain operator evaluation of the changed procedure.[81]

In an emergency situation the operator has only three aids available to enable him to cope with the emergency—emergency procedures, training in similar situations, and knowledge of the plant operation and status. The operator must detect and isolate the problem by diagnosis. The Essex Corporation pointed out that the operator cannot depend entirely on his knowledge of the plant or his training to make the diagnosis or to determine what action is necessary to isolate the problem. He must rely on the emergency procedures.[81] For this reason he needs accurate and readily accessible procedures to supplement his knowledge and training. They should provide him with criteria and steps to be taken in formulating hypotheses concerning what is happening in the plant and in testing the hypotheses using displayed data and test sequences.

The underlying questions are: Were there procedures available to cope with the situation at TMI on the morning of March 28, 1979, and did procedures or lack of procedures have an impact on the accident. We believe that the procedures were grossly deficient in assisting the operator in diagnosing problems with the feedwater system, the emergency feedwater system, and OTSG level responses when emergency feedwater pumps were activated. The procedures were of no help in diagnosing the PORV failure, nor did they provide guidance in analyzing the situation of pressurizer level increasing while RC pressure decreased. Furthermore, the procedures gave no guidance regarding overriding the automatically initiated HPI, when to trip the RC pumps while temperature and level are high and pressure is low, and when and how to establish natural circulation.[82]

5. OPERATOR SELECTION AND TRAINING

Regulations and Requirements

The statutory requirements for licensing operators of nuclear powerplants are contained in Section 107, "Operators' Licenses" of the Atomic Energy Act of 1954, which states:

> The Commission shall
> a. prescribe uniform conditions for licensing individuals as operators of any of the various classes of . . . facilities licensed in this Act;
> b. determine the qualifications of such individuals;
> c. issue licenses to such individuals in such form as the Commission may prescribe; and
> d. suspend such licenses for violations of any provision of this Act or any rule or regulation issued thereunder whenever the Commission deems such action desirable.[83]

That Act also defines the term "operator" as "any individual who manipulates the controls of a utilization or production facility," which includes nuclear power reactors.[84] Although "controls" is not defined in the Atomic Energy Act, the term is defined in the Commission regulations as meaning "apparatus and mechanisms the manipulation of which directly affect the reactivity or power level of the reactor."[85] The Commission has implemented this statutory requirement in Part 55 "Operators' Licenses" of its regulations. Part 55 establishes the procedures and criteria for the NRC's issuance of two types of licenses, one for "operator" and one for "senior operators."[86]

The NRC has also issued regulatory guides that provide details concerning the methods that are acceptable to the NRC staff for satisfying the specific requirements in Part 55.[87] These regulatory guides in turn refer to other NRC documents (NUREGs) that provide further guidance on the information needed by the NRC staff for its review and evaluation of applications for licenses.[88] These guides also refer to relevant national standards developed under the aegis of the American National Standards Institute (ANSI), which serves as a clearinghouse to coordinate the work of standards development in the private sector.[89]

Part 55 requires an applicant to pass a written examination and operating test "to determine that he has learned to operate, and in the case of a senior operator, to operate and to direct the licensed activities of licensed operators in a competent and safe manner."[90] The guidelines that apply to experience and education are contained in ANSI-N18.1, which specifies that licensed operators have a high school diploma or the equivalent. Two years of powerplant experience are specified for reactor operators and 4 years for senior reactor operators. Under this industry standard, operators must possess a high degree of manual dexterity and mature judgment. There are no requirements, however, that operators possess any other aptitudes, such as problem solving or spatial orientation capabilities.

Additional requirements for operator license application are set forth in broad terms in 10 C.F.R. 50.11. Generally, the physical condition and general health of the applicant must not be such that could "cause operational errors endangering public health and safety" or that could "cause impaired judgment or motor coordination."[91] These requirements are elaborated on in Regulatory Guide 1.134 and ANSI N546-1976.[92]

The 10 C.F.R. 55.33 requires that each licensed operator demonstrate his continued competence every 2 years to have his license renewed. The NRC accepts certification that an operator has satisfactorily completed an approved requalification training program as evidence of such competence in lieu of reexamination. Appendix A of 10 C.F.R. 55 presents requirements for operator requalification training.[93]

NRC Examinations

After accepting an application, the NRC staff (or an NRC consultant) prepares, administers, and grades the operator license written examinations and oral operating examinations to test the applicant's understanding of the design of the reactor for which the applicant seeks an operator license and the applicant's familiarity with its controls and operating procedures.[94]

The regulations indicate the general topical content of the written examination for operators and the supplemental topics for the senior operator examination that covers in greater depth areas such as reactor theory and operating characteristics.[95] Similarly, the regulations give topical guidance for the operating tests for both an operator and a senior operator.[96]

The scope of the 8-hour written examination is outlined in 10 C.F.R. 55.21 and NUREG-0094, Chapter IV and covers the following seven topics:

1. principles of reactor operation,
2. features of facility design,
3. general operating characteristics,
4. instrumentation and control,
5. safety and emergency systems,
6. standard and emergency operating procedures,
7. radiation control and safety.

An individual passes the written examination if he receives an overall grade of 70%. A grade of less than 70% in a given category is not grounds for failure if it is compensated by a grade higher than 70% in another category.

Candidates for the senior reactor operator's license must, in addition to passing this 8-hour written examination, pass a 5-hour written examination, the scope of which is outlined in 10 C.F.R. 55.22 and NUREG-0094, Chapter IV. This examination covers the following five topics:

1. reactor theory,
2. radioactive material handling, disposal, and hazards,
3. specific operating characteristics,
4. fuel handling and core parameters,
5. administrative procedures, conditions, and limitations.

In addition to these written examinations, the NRC staff or its consultants administer a 4- to 6-hour oral examination as required by 10 C.F.R. 55.[97] The scope of the examination is described in NUREG-0094, Chapter VIII.

The scope of the operating examination for the applicant for a senior reactor operator's license will be generally the same as that for operators. However, the senior operator candidates are required to demonstrate a higher degree of competence, knowledge, and understanding than that required of reactor operators.[98] The oral examination is conducted at the applicant's plant, primarily in the control room where the applicant is asked to point out and explain the function and use of plant instrumentation and controls. The test includes hypothetical accident scenarios and mock manipulation of controls.[99] If the applicant has not been to a simulator, he is required to demonstrate his capability by actually manipulating the controls of the reactor during a startup.[100] However, most applicants have had simulator training and the examiners do not usually witness the applicants' manipulating the controls of the plant.

The applicant is examined on proper use of normal, abnormal, and emergency procedures and his knowledge of plant technical specifications, administrative procedures, and emergency plans. During a tour of the plant, the applicant's knowledge of radiological practices and monitoring equipment are also evaluated.[99]

Throughout the oral examination, the applicant's knowledge and understanding are subjectively evaluated by the NRC examiner and are noted on an examination summary report, NRC form 157.[101] No objective measures are used. If in the judgment of the examiner, an applicant performs unsatisfactorily on any facet of the oral examination, the examiner documents the performance in the comments section of the summary report. This report is then reviewed by NRC's Operating Licensing Branch. Three "unsatisfactory" grades (U's) or six "marginal" grades (M's) out of approximately 70 constitutes a failure.[102]

The licensing examinations are made up, administered and graded by NRC employees (NRR's Operator Licensing Branch) or their consultants who are usually employees of national laboratories and university professors who work with research reactors. The staff and the consultants are not required to have a current or expired operator's license. Furthermore, there is no training program to qualify NRC examiners or any requirement that they maintain an expertise in the areas covered in the examinations.

Unless specifically requested, neither the applicant nor the utility receive copies of the examination summary report. Consequently, the weak points cited during an examination are not necessarily told either to the operator or his management. R. Zechman of TMI's training staff has stated that he knew of only three cases where TMI operator examination papers were obtained by his department.[103] These three were obtained through FOIA requests.

Operator Training Program

As noted earlier, in addition to passing the NRC's written and oral examinations, an operator license applicant must demonstrate he "has learned to operate the controls in a competent and safe manner."[104]

This requirement is normally fulfilled by certification from the utility that the applicant has completed a utility-administered training program. The training program for TMI-2 is carried out by the Met Ed training department, which relies partially on training services offered by Babcock & Wilcox. The program Met Ed developed was divided into two phases: "cold training" and "hot training" corresponding to periods before and after reactor criticality. For both of these programs the normal progression of personnel involves gaining experience as an unlicensed auxiliary operator for about 1½ to 2 years before applying and being accepted into the licensed operator training program.

In addition to providing the necessary training for new operator license applicants, the utility's training program must also meet regulatory requirements for the operator requalification program.[105] This portion of the program, as its name suggests, is focused on training operators who must renew their licenses every 2 years. The TMI-2 cold training program was described in the TMI-2 FSAR,[106] which was reviewed and approved by the NRC staff. The program was geared for Unit 1 auxiliary operators who had applied to become Unit 2 operators, and involved candidate participation in:

1. approximately 200 hours of formal classroom training in areas such as abnormal occurrences, plant modifications, major operational evolutions, emergency procedures, radiation control, and safety;
2. one week of training at Penn State University's research reactor which included core physics and detector experiments and assured that each candidate participated in 10 reactor startups;
3. one month of practical onshift observation experience at the TMI-1 control room;
4. an 8-week simulator training course at B&W (100 hours simulator operation, 170 hours classroom instruction, 40 hours simulated NRC examination and review);

5. classroom training course on TMI-2 NSSS, secondary system and balance of plant systems (160 hours);
6. nuclear theory review course (60 hours);
7. advanced systems, procedure, and nuclear theory training (about 8 weeks).

The cold training program also had the provisions for operators who were licensed on TMI-1 to become licensed senior reactor operators for TMI-2. Their program included a utility administered unit differences course, which stressed the difference between the Unit 1 and Unit 2 NSSS, secondary and balance-of-plant systems.[107]

Subsequent to reactor criticality on March 28, 1978, Met Ed developed its hot training program for providing replacement reactor operators and senior reactor operators. This program is geared to provide the same technical training as the "cold" program but since the reactor had become operational, it relied more heavily on the utilities' reactor and less on B&W with its reactor simulator.

Replacement candidates for control room operators are chosen from the ranks of the experienced auxiliary operators. The operators are required to complete a 9-month formal training program covering the same general material and fundamental program that was outlined for training the initial TMI-2 staff. The program includes normal and emergency operating procedures, system operation, simulator training, and classroom training on topics such as reactor theory and health physics. This program is administered primarily by the TMI training staff, with a short simulator training course at B&W's Lynchburg facility.[108] As part of the operator's license application, Met Ed provides certification of the applicant's successful completion of the replacement training program. This certification, however, does not contain information on how well the candidate performed in the training program.

Replacement candidates for senior reactor operators licenses are filled by more experienced personnel. In 1977, senior reactor operator candidates who were licensed on Unit 1 were required to complete a training course on Unit 2 systems, technical specifications, and procedures, with emphasis on the differences between Unit 1 and Unit 2, as well as to attend a short simulator training course at B&W's Lynchburg facility.

In July 1977, the TMI training department requested that the NRC's Operator Licensing Branch (OLB) grant senior reactor operators' licenses valid for Unit 2 to senior reactor operators who were licensed on Unit 1 and who had successfully completed a "differences" course. Agreement was reached between NRC-OLB and TMI that the oral walk-through examination would be waived. However, NRC-OLB required the TMI training department to make up and administer a "differences examination" to demonstrate operator proficiency on Unit 2.[109] The NRC performed an audit review of the examination of 12 candidates who took the cross-licensing examination; 11 passed and were granted senior reactor operator licenses for Unit 2.[110]

The NRC did not approve TMI-2's hot training or replacement operator program. In fact, firm details of this program were never submitted to NRC. Instead, Met Ed submits a synopsis of the replacement operator's training background and experience along with the operator's license application. NRC approval of the program is essentially performed on a case-by-case basis without any prior NRC review.

TMI Requalification Program

The TMI requalification training program extends over the 2-year duration of each operator's license. It is administered by the utility's training department with the use of external training resources. As described in the TMI-2 FSAR, the program includes:[111]

1. Operational review lectures (60 hours per year, including films and videotapes) and self-study of the following items: reportable occurrences; unit modifications; operating history and problems; procedure changes; abnormal and emergency procedure review; technical specifications; major operational evolutions (such as refueling); applicable NRC regulations, 10 C.F.R.; and fundamentals and system review. (These lectures may be given on shift by shift foremen and shift supervisors instead of by the training department.) Additional preplanned lectures are provided in areas in which an operator's annual written examinations indicate that strengthening of the operator's knowledge is necessary in any area that NRC's written examination covers.
2. On-the-job training where each licensed operator manipulates the reactor controls or the B&W simulator controls to effect reactivity changes on 10 occasions during the 2-year requalification program. In addition, to ensure diversity of operator performance, operators may be assigned to surveillance testing, makeup-purification system operation, decay heat removal system, feedwater system operation, and reactor cooling system.
3. The licensed operators undergo annual written examinations to identify areas that are covered in NRC's written examinations for which retraining

and upgrading are required. These examinations are made up, administered, and graded by the TMI training department; NRC's OLB performs only an audit review. An overall written examination grade of 70% is passing. Grades under 70% require that the operator be relieved of his licensed duties until he successfully completes an accelerated retraining program. Any operator scoring less than 80% in any category is required to attend fundamentals and systems review lectures on that subject.

In addition, the licensed operators undergo annual oral examination given and graded by the TMI training staff. The examination covers operator actions during abnormal and emergency conditions, response to transients, instrumentation signal interpretation, procedure modifications, technical specifications, and emergency plans.

An unsatisfactory evaluation on the annual oral examination requires discussions of deficiencies between the operator and supervisory personnel and administration of a second examination. Unsatisfactory performance on the second examination results in the operator being relieved of responsibilities until he successfully completes an accelerated requalification program.

B&W Training Program

As previously mentioned, the TMI training program relies in part on training services offered by B&W. The following list outlines the basic elements of the program offered by B&W:[112]

1. basic nuclear theory, lectures and operation at vendors' training reactor (not simulator) (3 months) at Lynchburg or with B&W personnel at utility site;
2. nuclear powerplant observation (2 months) at a B&W-type plant;
3. reactor simulator training, including mock NRC written and oral examinations (2 months), at Lynchburg;
4. design details of the specific plants (1 month) at Lynchburg or with B&W personnel at utility site;
5. on-the-job experience at plant during testing as well as writing operating and test procedures (10 months) as a B&W resident at utility site.

The TMI-2 PSAR noted that the Met Ed training department would consider using B&W's assistance in the basic nuclear theory portion of the cold license training program.[113] However, Met Ed staff and the services of NUS Corporation were used instead.

The TMI staff used B&W's 8-week cold license simulator certification training course, which was conducted at B&W's Lynchburg simulator training facility. Certification by B&W of the simulator training and mock examination results were forwarded by Met Ed to NRC with the applications for operator licensing examination. The TMI operators also took 1- or 2-week simulator courses as a refresher in preparation for cross-licensing senior reactor operator examinations and operator requalification. Met Ed also purchased an 8-hour B&W videotape series illustrating the performance of the integrated control system during transients.[103]

Met Ed also hired the services of the General Physics Corporation to give operator license candidates mock NRC oral examinations several weeks prior to the actual NRC examinations. NRC was not apprised of the results of those examinations.[114]

Evaluation of Training

One important objective of a good training program is to provide operators with the skills and the knowledge to deal with emergency situations like those that occurred at TMI-2 on March 28, 1979. In order to understand how well TMI's training program met this objective, the Essex Corporation analyzed six TMI-2 emergency procedures relevant to the accident. This analysis identified tasks or actions that operators must perform to respond correctly to emergency conditions that should form the basis of a well designed emergency training program. Fifty-three such emergency tasks or actions were identified. Twenty-three of these require the operator to have and use diagnostic skills. The other tasks require memorizing procedural skills for following sequences of activity and control skills involving motor and perceptual capabilities. The significant need for diagnostic skills during emergencies is further underscored by the discussion in Section II.E.2 of this report on the significant deficiencies in the ability of the TMI operators to diagnose the difficulties experienced in the plant.

Essential features of diagnostic skill training are the ability to reproduce symptoms of a fault condition in training and to challenge the operator to detect and isolate the problem based on his understanding of what is happening in the plant. Such training for control room operators can utilize lectures, study of nuclear engineering, and practice in responding to emergencies. Practice can be accomplished either through simulated emergencies at a training facility (simulator) or through experience from repeated emergency situations in an operating plant. Obviously, the latter is of little use in training applicants for CRO licenses for nuclear powerplants because of the practical limitation on what can be done at an operating plant.

The educational backgrounds of most of the TMI-2 operators, and the operators on shift the morning of the accident were no exception, are limited to completion of high school. This fact suggests that heavy emphasis on lectures and classroom study may not be the most effective means of teaching diagnostic skills. Similarly, the Essex Corporation concluded that diagnostic training of nuclear powerplant operators can best be accomplished by the use of a simulator that accurately reproduces the system response and format of the information available in the operator's own plant control room.[115] The simulator need not, however, accurately reflect the control room's physical characteristics. Thus, the B&W Lynchburg simulator shown in Figure II-54 could serve as an effective training tool for diagnostic skill acquisition even though it does not physically represent the TMI-2 panel configuration.

The Essex Corporation found that only 6% of the TMI operator training program was devoted to simulator training.[116] They also found, however, that even this time was not used effectively to provide diagnostic skills. Most of the time was used to give simple demonstration of plant response. Little or no simulator time was used to practice control techniques, procedure sequences, or fault identification and isolation, any of which would improve the operator's diagnostic skills.

Simulators can also be used to develop procedural and control skills and for evaluating and improving procedures. Because all of these uses require a high degree of similarity between the simulator and the plant's real control room, the B&W simulator was of little value in this training area.

As previously noted, another method of acquiring skills is through on-the-job training in the reactor facility itself. On-the-job training can be an excellent followup training approach because of the high fidelity of the control room configuration, system response, and procedures to those that the operator will face. Essex found, however, that the operators' formal on-the-job training accounted for about 10% of the total training time.[117] Furthermore, since on-the-job training is conducted during normal operation with few transients, it was of little direct benefit in dealing with accident situations. The TMI-2 FSAR requires that an operator have experienced certain specified events during the 2-year term of his license for requalification. Specifically, he must participate in a minimum of 10 reactivity manipulations that are judged to demonstrate skill and familiarity with reactivity control systems.[117] The Essex Corporation did not directly measure the actual operator skills for manipulating reactivity controls. The overall evaluation of the deficiencies in simulator and on-the-job training, however, led to the conclusion that the TMI training program did not ensure adequate development of their skills.[118]

The crew on duty at the time of the accident had spent an average of about 85% of their training time in classroom instruction.[119] This training time was spent either attending training sessions during a training shift or studying for several hours on shift. To determine the scope, accuracy, and clarity of these lessons, Essex evaluated the lesson plan and lecture outline for a number of reactor systems. This evaluation compared the plans and lectures with the 53 emergency tasks or objectives identified by the review of emergency procedures. Twelve of those objectives were related to the feedwater system, yet none was addressed in the lecture outline. In addition, Essex found that the general format and organization of the lesson outline was seriously lacking and concluded that, even though the author of the outline may have been familiar with the feedwater system, the author did not display any expertise in presenting training material to enhance operator interest and retention. A similar evaluation of the lecture outline for the reactor coolant system yielded identical results. Not one of the 20 training objectives identified to deal with the RCS was addressed in the outline.[120]

An important aspect of any training program is the effectiveness of the training evaluation methods. With a good set of measures to establish operator performance capability, the training program can be evaluated, as well as the effectiveness of the operators. Although Met Ed has developed a number of tests and quizzes to measure the effectiveness of the operator, the Essex Corporation found the tests deficient.[121]

The emergency training objectives identified from emergency procedures were not reflected in the examination. The operators did not get feedback on their own strengths and weaknesses. Also, the examinations did not measure the operators' ability to diagnose a transient, control the plant, follow procedures, or anticipate responses. The only required measure of operator capability is the NRC licensing examination. Essex's review of these examinations concluded that on average, each examination covered only 1 or 2 of the 53 emergency training objectives they had identified from the emergency procedures. The Essex Corporation study conclud-

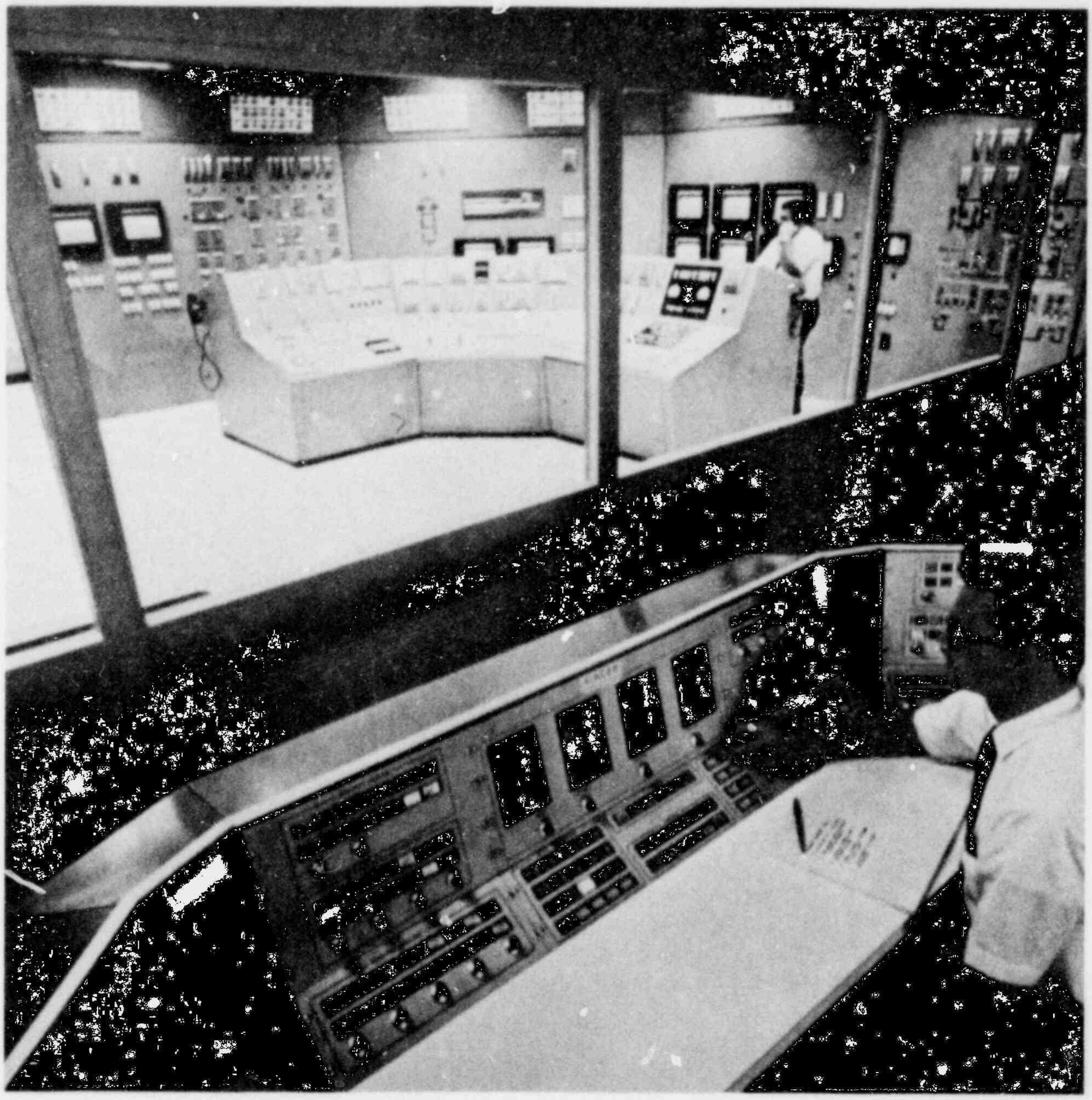


FIGURE II-54. B&W Simulator—Showing Instructors Console

ed that the NRC examinations provide a poor assessment of the operator's ability to use emergency procedures.[121]

On the basis of its evaluation of the TMI training program, Essex concluded that the program was deficient in the following respects:[122]

1. It was not directed at the skills and knowledge required of the operators to satisfy job requirements.
2. Too little simulation was provided and where it was used it was misused.
3. It failed to provide the operators with the skills they needed in the accident; e.g., skills in developing a hypothesis and acquisition of feedback data to verify the hypothesis.
4. It failed to provide for measurement of operator capability.
5. It provided no instruction for the instructors on how to reinforce lesson objectives or how to assist trainees in understanding the system.

6. It took an archaic approach to learning, in that no applications of instructional technology were included in the program.
7. It was not closely associated with procedures used by the operators, and no guidance was provided in what to do if procedures do not apply.
8. It did not provide for formal updating and upgrading of training methods, materials, and content.
9. It failed to establish in the crew the readiness necessary for effective and efficient performance.

These conclusions were aptly summarized by Essex as follows:

> Operators were exposed to training material but they certainly were not trained. They were exposed to simulators for the purpose of developing plant operation skills, but they were not skilled in the important skill areas of diagnosing hypothesis formation and control technique. They were deluged with detail yet they did not understand what was happening. The accident at TMI-2 on the 28th of March 1979, reflects a training disaster.[123]
>
> The overall problem with the TMI training is the same problem with information display in the TMI-2 control room application of an approach which inundates the operator with information and requires him to expend the effort to determine what is meaningful.[124]

Manning Levels and Operator Qualifications

The NRC regulations require that a licensed control room operator (CRO) or senior operator (SRO) be present at the controls at all times during plant operation.[125] This requirement is implemented through the TMI-2 technical specifications,[126] which require that one SRO and two CROs be on shift during reactor operations. Under most circumstances, no more than one licensed operator needs to be in the control room. The NRC requirements allow, however, for the utility to have only one licensed operator on site (in the control room) for short periods.[126]

Although the size and combined experience of the shift of two licensed operators and two senior operators on duty when the accident occurred was considerably larger than the minimum required by the NRC, the difficulty experienced by this shift in responding to the accident raises serious questions about the adequacy of the minimum NRC manning requirements.

We found that the licensed operators on duty at TMI-2 at the time of the accident met or exceeded all NRC requirements with respect to background, training, and qualifications. All four of these operators performed reasonably well on the NRC licensing examination. Their grades ranged from a low of 79% to a high of 91% with an average of 83.5%.[127] This compares favorably with a passing grade of 70%[128] and an industrywide average of about 80%.[129]

Before the accident all of the operators had completed training courses that met NRC requalification requirements, and all were slated to take a 1-week simulator course at B&W's Lynchburg facility on April 9, 1979. Each operator had received simulator training totaling 5 to 9 weeks. Three of them had a week's training at Penn State University's research reactor.[130]

All four operators had high school diplomas and two had completed about a year of college coursework. This educational background is similar to industrywide experience where 50% of the CROs and 80% of the SROs have formal education beyond high school but most have not graduated from college.[131]

Comparison with the Nuclear Navy

The most striking difference between the backgrounds and experience of the four operators who were on duty at TMI-2 and those of most reactor operators is their U.S. Nuclear Navy experience. Each of these licensed operators had more than 5 years of experience in the U.S. Nuclear Navy for a combined total of 26 years. This is considerably higher than the industry average of 2.5 and 4.4 years of noncommercial (mostly Navy) nuclear reactor experience of CROs and SROs, respectively.[131] Each had graduated from the Navy's 6-month basic enlisted nuclear power school and the 6-month program at a Nuclear Power Prototype School. Their specialty positions in the Navy included an electronic technician, an electrician, an interior communications technician, and a machinist.[132] Although we have not inspected confidential Navy records on these operators, we have found no evidence that would indicate that they did not perform satisfactorily in the Navy.

The Navy nuclear power program is generally considered to be highly successful with a reputation for having well-trained and disciplined operators.[133] Since operator actions at Three Mile Island led to the severe core damage, some have suggested that training and discipline similar to that used in the Navy should be adopted in the commercial nuclear plant program.

The significant differences in Navy nuclear propulsion plants and civilian nuclear powerplants, however, suggest that personnel who may be highly qualified to operate the Navy plants may not be the most qualified to operate large complex civilian nuclear powerplants. The Navy nuclear plants are designed to accommodate expected transients without the need for immediate operator actions or automatic system responses. The plants rely on inherent system stability rather than automatic or manual activation of complex control systems to control the plant during most transients.[134] In addition, there has been a significant effort to simplify system design to give confidence in the ability of operators to operate the plant properly.[135]

The implication here is subtle but significant. The designers of commercial plants have assumed that the operators are only a backup to the automatic control. They expect operator action only if the automatic systems fail to perform properly and then only to the extent needed to correct the immediate problem. In addition, the commercial system is harder to understand and operators often have to react to unexpected operation of automatic systems. This includes intervening in the operation of automatic controls that are otherwise operating as designed. Rapid manual operator action is often necessary. Examples are the need to reset and throttle HPI to prevent pump runout, and the rapid actions required to prevent excessively low pressurizer level following a trip. The complexity of civilian nuclear plants is associated with some safety advantages, however. Much of this complexity stems from the presence of more safety systems. In addition, the typical commercial reactor control room, with its multitude of alarms and indicators provides significant automated warnings of insipient problems and status changes. Navy reactors rely more heavily upon operator surveillance for such warning.

Navy nuclear facilities typically require about 10 trained operators at all times with 4 on duty in the control room. The operators are enlisted personnel who have been trained to operate specific portions of the plant. The plant shift also contains more experienced roving watch personnel who have a better understanding of the entire plant operation and help to supervise individual operators. All of these personnel are directly supervised by the engineering officer of the watch (EOOW), a commissioned officer who is a graduate engineer and who has received special training at the undergraduate and graduate level in the fields necessary to support nuclear operations. The EOOW also has had specialized training to ensure his capability to supervise operators and plant operations under normal and emergency situations.[136,137] The EOOW is stationed in the control room directly overseeing the plant operators of the main control panels. All changes in plant status, responses to abnormal conditions, requests to do maintenance, etc., are approved and discussed with the EOOW.

A number of elements of this structure are relevant. The EOOW is not an operator, he does not operate equipment (although he was required to do so at every watch station to become qualified).[138] He is a supervisor and an integrator of the entire plant operation. In contrast, the enlisted operators' responsibilities are for only a specific section of the plant.[138,139] Within each section they have responsibility for operating, monitoring, following procedures, and taking actions as directed by the EOOW.

In contrast to the Navy, the NRC requires only three licensed operators on site during operations of commercial reactors. Only one of these must be in the control room. There are additional unlicensed auxiliary operators on shift who perform many of the plant operations outside the control room.[126] The senior reactor operator is given the responsibility of supervising plant operations and the licensed operators in the control room. His training and qualifications, while more stringent than those for the reactor operators, are not comparable to those required for the engineering officer of the watch in the U.S. Navy. Senior reactor operators are promoted from the ranks of reactor operators primarily on the basis of having demonstrated competent performance and passing a more rigorous examination. In many ways, the SROs have been given the responsibility of the EOOW without the benefit of an engineering degree or the specialized supervisory training received by EOOWs. In addition, they are supervising generalists in plant operations rather than specialists with assignments to specific parts of the plant.

The training and selection of personnel are also significantly different in the Navy and in the civilian programs. Navy programs have a highly organized system of selection, training, qualification, continuing training, and examination of its personnel. This system includes detailed selection criteria and screening, and formal training programs using trained and skilled instructors with formally approved training course materials. All facets of the program are frequently reviewed, audited, and inspected by trained and experienced personnel.[140] Before assignment of personnel to operating ships, they must complete rigorous academic and practical training programs. Each person must complete a 6-month classroom course (equivalent to 50 semester hours of classroom instruction)[141] and a 6-month training course at an operating land-based prototype reactor[142] to demonstrate their fitness for duty onboard ship and to prepare them for such duty.[137] These prototypes may be compared to nuclear powerplant simulators, although, unlike commercial simulators, they are functioning nuclear reactors that closely resemble powerplants onboard ships.

Once an operator arrives onboard ship, the operator must completely requalify on the engineering plant of the particular ship. This includes written, oral, and practical examinations administered by the ship's personnel at each level of the qualification program.[143]

In addition, the crew is examined annually by an independent examining team. This examination (operational reactors safeguards examination) emphasizes both individual knowledge as well as the ability of entire watch sections to function during actual, self-initiated casualty situations.[144,145] A large part of the examination covers diagnosis of problems, since the operators are not warned of casualty drills before they are initiated.[145] Crews that have significant weak areas are required to take immediate corrective action and to report such action to headquarters. Failure of the examination will result in extensive retraining, requalification, and reexamination, until it is determined that the crew meets acceptable levels of performance.[145]

In every area described above concerning the Navy nuclear program, the civilian nuclear program appears to fall short. There are no standardized performance criteria and guidelines, and there is no systematic meaningful review of training programs. Simulators, which are the only available means of training operators in actual plant emergency operation, are not required to be used in an effective manner (they are used more as a demonstration device than as a tool to develop proficiency in diagnosing and coping with accidents).

In comparing related but dissimilar situations, care must be exercised in drawing unqualified conclusions. Because of the design differences between Naval and commercial reactors, we believe it would be inappropriate to incorporate all aspects of the Navy system into the civilian program.

6. HUMAN FACTORS PRECURSORS

Introduction

Before March 28, 1979, accident precursors, in the form of reports of reactor instances, Congressional testimony, and correspondence, contained warnings that an accident of the type that occurred at TMI-2 could happen. Section I.C of this report addresses precursors relating to the design and function of the TMI-2 reactor. This section addresses those precursors relating specifically to the human factors application in control room design, operator training, emergency procedures, and the issue of the man-machine interface. This discussion and analysis documents the fact that, before the accident, the NRC and the industry had been alerted to the human factors problems, many of which existed at TMI-2.

"Evaluation of Incidents of Primary Coolant Release from Operating Boiling Water Reactors," WASH-1260

In May 1972, the Atomic Energy Commission appointed a seven-member study group[146] under the auspices of the Office of Operations Evaluation to conduct an evaluation of incidents involving the unintentional discharge of significant release of reactor coolant from the primary coolant system of operating nuclear powerplants. Of 50 reported inadvertent releases or leakages, the study group identified and studied eight that involved significant releases. On October 30, 1972, the AEC published the study group report WASH-1260.

The study group made many findings and recommendations, several of which dealt with control room design, manning of the control room, operator training, operating procedures, and feedback of operational experience.

The study group found that insufficient consideration had been given to displaying information on control panels and to the location of controls in relation to each other, particularly when only one operator is required in the control room during operation.[147] The group recommended that the industry develop control panel and control room design standards or guides that address the human engineering aspects of reactor operation during abnormal operating occurrences.[148]

The report also discussed the need for further consideration, during the control room design phase, for the instrumentation and controls and their layout, taking into consideration the number of operators, the information required by them to rapidly diagnose and take proper corrective action in response to unusual occurrences, and other human engineering aspects of plant control system design.[149] The study group made specific recommendations addressing the instrumentation needed to provide the operator with the information essential to reaching proper operating decisions during transients and postulated accidents.[150]

The NRC regulations require only one licensed operator to be on duty in the control room during operation. In view of the fact that more than one licensed operator was on duty in each of the eight instances, the study group found that the number of personnel in the control room was not a factor. The study group recommended, however, that a guide be developed to assist in evaluating the number of reactor operators needed to cope with anticipated transients. They listed the criteria to be taken into account in determining the size of the control room staff. They further recommended that utilities of currently operating plants and applicants for new plants should be required to evaluate their control room manning needs based on the these criteria.

It was found that the training and experience of the reactor operators in the eight incidents studied appeared to be adequate and met the AEC guides and standards.[151] They also found, however, that the transients studied tended to be aggravated and prolonged by operator actions. The study group felt that one of the causes for this could have been insufficient training.[152]

It was recommended that the licensees and applicants should, to the extent practicable, use simulations to train and evaluate operator performance and verify the adequacy of operating procedures. Simulators should also be utilized to evaluate operator performance and adequacy of training during operator licensing.[153]

Additionally, the report contained a recommendation that licensees and applicants for licenses be required to submit plans and schedules for training of technicians and repairmen engaged in the testing and maintenance of safety related systems and components.[154]

During the incidents studied, a number of deviations from operating procedures and technical specifications were experienced.[155] The report indicated that operating procedures were either incomplete or deficient for coping with anticipated transients, and although some improvements had been made, further improvements were needed.[156]

The report indicated that there was insufficient information available to determine whether incident reports were disseminated to facilities in a timely manner or whether corrective action had been taken or planned to minimize the probability of recurrence in the plant where the transient occurred.[157]

The study group made a number of recommendations regarding reporting and dissemination of operating experience. It recommended that a system be developed and implemented to fully inform licensees of incidents and unusual occurrences. It further recommended that an incident reporting guide be developed by the AEC, and enumerated specific information to be reported.[158] Finally, it recommended that regulatory policies and procedures be revised to identify more clearly the responsibility for review, decision making, investigation and documentation with respect to incidents and unusual occurrences.[159]

On November 28, 1972, the director of regulation, in a memorandum to three directors, indicated that the recommendations of WASH-1260 are to be implemented by the appropriate regulatory directorates.[160]

Some actions were taken to implement the recommendations of WASH-1260, including the following:

1. The NRC contracted with Sandia Laboratories to conduct a study of human factors problems of the Zion nuclear powerplant.[161] This will be discussed in a later portion of this section.
2. The AEC interacted with industry to develop industry standards for control room displays.[43] However, to date these standards have not been endorsed by the NRC.
3. Incident and abnormal occurrence reporting requirements underwent evolutionary changes regarding reporting times and information requirements; however, the details and mechanism for utility review of events at other facilities do not appear to have been addressed by the NRC regulations. Furthermore, circumstances surrounding the handling of the 1977 incident of the Davis Besse plant indicate the existing process fell short of the recommendation.[162]
4. Regarding information available to the operator at a nuclear powerplant during and subsequent to a transient or accident, the NRC has written Regulatory Guide 1.97 "Instrumentation to Follow the Course of an Accident." However, as of March 28, 1979, this standard had not been fully implemented in either old plants or those undergoing licensing review.
5. Reactor simulators have found widespread use. However, the recommendations of WASH-1260 in the area of simulators have not been implemented; i.e., the NRC has virtually no requirements regarding simulators. They are not used to evaluate reactor operators' performance; they are not generally used to verify operating procedures for coping with anticipated transients;[163] the NRC examiners seldom observe and evaluate operators on the simulator for their licensing examination and receive only scant information regarding specific operators' performance. Furthermore, the licensees do not use the simulator as a basis for modifying operating procedures or for

evaluating the need for operator training or retraining.

Human Performance March 13, 1975, Memorandum from Hanauer to Commissioner Gilinsky

On March 13, 1975, Dr. Stephen H. Hanauer, Technical Advisor to the Executive Director for Operations of the NRC, initiated a memorandum to Commissioner Gilinsky to which he attached his views on important technical reactor safety issues facing the Commission and reactor safety policy issues.

In his list of technical reactor safety issues, Hanauer addressed the subject of human performance, stating:

> Present designs do not make adequate provision for the limitations of people. Means must be found to improve the performance of the people on whom we depend and to improve the design of equipment so that it is less dependent on human performances.... The relative roles of human operation and automation (both with and without on-line computers) should be clarified. Criteria are needed regarding allowable computerized safety-related functions and computer hardware and software requirements for safety-related applications.[164]

At the time of the TMI-2 accident, no substantive action had been taken by the NRC as a result of this memorandum addressing the human performance issue. No criteria have been developed by the NRC regarding the roles of human operation and automation or computer aids for the operator.

Hearings Before the Joint Committee on Atomic Energy, Congress of the United States, February 18, 23, and 24, and March 2 and 4, 1976

Three former General Electric employees, Dale G. Bridenbaugh, Richard B. Hubbard, and Gregory C. Minor (BH&M), testified before the Joint Committee on Atomic Energy. They cited numerous examples of human factors deficiencies in the nuclear power industry. They pointed out examples of incidents resulting from human error that could have resulted in major accidents. To minimize these errors, they made specific recommendations in the area of control room design, the availability of up-to-date simulators and their utilization for more frequent training of control room operators, the adequacy of operational and maintenance procedures, and the training of operators to use these procedures. The NRC, on March 2, 1976, testified before the Joint Committee in response to the testimony of BH&M.

The NRC concluded that nuclear reactors are designed to keep the likelihood of operator errors relatively low and took issue with the statement that the human error that has occurred "has seriously jeopardized plant and public safety" because the "engineered safety features, redundant systems and containment design features have always, singly and in combination, been available to protect plant and public safety."[165]

BH&M testified that improvements in control room design were one method of reducing the likelihood of human error. They noted the complexity of nuclear powerplant control rooms, the differences in control room layout throughout the industry and the utilization of mirror images in common control rooms for two nuclear units. They also maintained that "standardization of control rooms is a vital element of safety...."

The NRC response supported standardization in general but claimed that standardization of control rooms and controls and displays had not been demonstrated to have a significant impact on operator performance.[166] The NRC testimony also pointed to studies sponsored by the NRC and industry to evaluate control room design and indicated that the IEEE was developing a standard guide for design and control facilities for control rooms.[167]

In discussing control room design, the NRC stated that due to the automatic initiation of the engineered safety features, the consequences of an accident are mitigated and the only functions of the operator are to ensure that these systems function properly and to initiate any action that failed to occur. It therefore concluded that "the control room design arrangement or operator-process interface is not as critical (or vital) to safety as may be inferred from the February 18, 1976 testimony."[167]

The NRC did, however, recognize the importance of human engineering principles, control room design standardization, and optional arrangement of design to minimize the probability of human error.[168]

BH&M testified that providing up-to-date simulators and more frequent training of operators is another method of reducing the likelihood of human error. Specifically, they indicated that the present simulators were outdated and did not represent the control philosophy that has evolved over the last 10 years. Additionally, they questioned the ability of the operator to remember the accident procedures through time without very frequent update, indicating that retraining periods are too infrequent to keep the operator aware of his special procedures under accident conditions.[169]

In response, the NRC disagreed with the contention that the simulators are outdated for training programs, pointing out that the design philosophy for data display and plant control for operating plants and those in the operating licensing stage of review are very similar to the design philosophy of existing nuclear powerplant simulators.[170]

The NRC pointed out that there was no requirement for simulator training, and if simulators are used, the operator is also trained at the plant for which he seeks his license. The NRC testified that it ensures that transition from simulator to plant has been made by the trainee through examination at the facility for which the individual seeks a license.[171]

The NRC agreed that it is unrealistic to expect the operator to remember details of accident procedures over a long period of time. In 1973, the NRC promulgated an amendment to 10 C.F.R. 55 by adding Appendix A, "Requalification Programs for Licensed Operators of Production and Utilization Facilities." This program requires periodic review of all abnormal and emergency procedures. The NRC has not conducted any tests, nor are they aware of any tests by others to determine how long an operator is able to retain procedural details.[172]

BH&M further testified, "Most human errors in reactor plants result from one of two causes: inadequate procedures or insufficient knowledge of existing procedures."[173] They recommended that the NRC review operational and maintenance procedures to ensure adequacy of both scope and content and that it step up its surveillance of training processes to ensure that the procedures are fully understood and implemented.[174]

The NRC responded that guidance in the preparation of procedures is provided to the applicant in Regulatory Guide 1.33 which incorporates industry standards. It pointed out that the utility plans are reviewed to assure compliance with this guide and that NRC inspectors conduct an audit of the detailed procedures to assure their completeness prior to the issuance of an operating license.[175] Review and approval of procedures and amendments thereto is conducted by utility management according to the NRC testimony.[176]

The NRC testified that training programs are reviewed to ensure that all personnel receive satisfactory training on all procedures appropriate to their respective job classification and responsibility. Additionally, the requalification program includes lectures on procedures, annual written examinations which include a section on procedures, requirements for licensed individuals to review procedure changes, and an evaluation by supervisors of licensed individuals to ensure proficiency in plant procedures.[176]

In reviewing the foregoing testimony, we believe that it provides a useful insight into the NRC's attitude towards human factors and nuclear reactor safety. In essence, the NRC staff's response is that operators are well trained, there have been no serious accidents, and that automated systems can be depended upon to assure plant and public safety. Other than the fact that there were ongoing studies in the area of human factors application to control room design, the NRC did not develop programs responsive to the BH&M recommendations because the agency maintained human error was not a danger to safe operation of nuclear powerplants.

Although the NRC stated that it would implement the recommendations resulting from the aforementioned studies, virtually none of these recommendations for improvement in control room design, operator training and procedure improvement has been implemented by regulations as of March 28, 1979.

"Preliminary Human Factors Analysis of Zion Nuclear Power Plant," NUREG 76-6503, October 1975

The NRC contracted with the Sandia Laboratories to conduct a study of the Zion nuclear plant. The scope of this study was limited to the human factors problems associated with engineered safety panels in the control room and associated procedures for coping with a LOCA. The Sandia report was published in October 1975.[66]

The report contained a number of significant conclusions and recommendations for human factors improvements in the Zion plant that are equally applicable to other nuclear powerplants. It was found that the control panels and other man-machine interfaces deviated from accepted human engineering standards and increased the probability of human error. Improvement in human performance could be achieved by relatively minor and inexpensive changes to the control room, practicing for emergencies, and changes in format and content of written procedures. The report concluded that industrywide standards covering all aspects of human reliability could serve to materially improve the impact of human performance on system availability and safety.[177]

The study found that the major human engineering problems fell into seven major areas:

- poor layout of controls and displays,
- poor and inconsistent color philosophy,
- too many annunciators,
- too many exceptions to the go—no-go coding scheme employed for rapid assessment of monitor panel status,
- labeling that provides little or no location aid to controls and displays,

- misleading labeling due to violation of population sterotypes, and
- insufficient labeling on valves.[178]

The report also pointed out that the human factors problems uncovered in the study were not peculiar to the Zion nuclear powerplant. Previous visits to other plants by the same investigators revealed similar human factors problems in each plant.[179]

The report contained the following four recommendations for consideration by the NRC:

1. Investigate the need for additional human factors data, and develop, on an exploratory basis, a method for acquiring the necessary information. Part of the study should be the determination of what level of information is needed. Whatever level of human error data collection system is deemed necessary, the suggested study should include the procedures and data forms for collecting human performance information.
2. Develop the procedures and format for incorporating human performance information (as determined in item 1 above) into the NPRDS.
3. Perform a complete human factors analysis at the Zion Plant (that is, expand the present preliminary analysis) to:
   a. Identify all major error-likely situations related to the safeguards systems.
   b. Estimate the relative likelihood of human errors and associated recovery factors for those errors identified as important by the reliability models.
   c. Provide recommendations (based on the above) for improving human reliability at the Zion (and similar) plant(s) and for design of future plants.
   d. Develop a procedure for a human factors analysis of nuclear powerplants which could be used during all phases of design and development to improve human reliability consistent with other systems engineering requirements.
4. Upon satisfactory completion of item 3 above, develop industrywide standards for human engineering of equipment, written procedures, operating methods, and onsite training and practice provisions in nuclear powerplants to insure the highest levels of human reliability consistent with other system requirements.[180]

We found that the human factors problems identified in this study are similar to those identified in other studies that predate the TMI-2 accident and those found in subsequent studies by Essex Corporation. On August 24, 1976, the Chairman of the NRC, Marcus A. Rowden, wrote to the Honorable Virginia H. Knauer, Special Assistant to the President for Consumer Affairs. In his letter Chairman Rowden stated in part, "We believe that human error analyses must not be neglected and indeed a special research review group on human error assessments has been established to coordinate and expedite our efforts. Programs are underway to systematize human error analysis and human error data evaluations through contracts, including that with Dr. Swain at Sandia Laboratory. If the results of these programs or actual experience with operating reactors indicate situations in which equipment design or operator interfaces should be improved, we will, in accordance with our statutory responsibilities and our implementing review procedures, require changes to the design or operation of the plants as required."

To date, virtually none of the report's recommendations have been implemented. It should be noted that even though the 1975 Sandia report on the Zion plant found that minor inexpensive improvements would enhance plant safety and operations, to our knowledge not one has been implemented, and as of March 28, 1979, none had been planned for implementation.

"Plan for Research to Improve the Safety of Light Water Nuclear Power Plants," NUREG-0438

On April 12, 1978, the NRC made its first annual report to Congress on its recommendations for research on improving the safety of light water nuclear powerplants. Among the recommendations was one dealing with improved inplant accident response.

The research recommendation covered operator response during an accident situation, information available to the operator on plant status, operator training and procedures, and human response under stress conditions. It was proposed that the research include not only operators in the control room, but also personnel involved in the testing and maintenance of the plant. It was pointed out that analyses have shown components may be left in an unavailable state by test and maintenance personnel through carelessness, improper training, use of improper procedures or failure to follow procedures.[181]

The proposed research would encompass computerized processing of data, control room layout and data presentation, and attention to human factors in the design of annunciators, warning lights, and display panels.

This research project was assigned a high priority by the NRC report because of its high potential for risk reduction and its low cost. The report proposed a project to review studies completed and in process on the following topics to establish the need for further research:[182]

- human errors in testing and maintenance;
- monitoring and diagnostic systems to assist the operator under accident conditions;

- operating and emergency procedures for responding to accident situations;
- improved use of simulators in studying operator response to accident situations and for related training;
- man–machine interface, information presentation, pattern recognition, control room design, and automatic controls for safety systems; and
- human initiation of accidents.

This research project was scheduled to begin in early FY 1980. The TMI-2 accident reinforced the need for high priority and resulted in accelerating the project initiation to the end of FY 1979.

We note that the purpose of this research project was to identify new areas for research in human factors while ignoring the large body of information being utilized by other industries that could be readily adaptable to the nuclear powerplant industry.

"1978 Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program," NUREG-0496

In December 1978, the Advisory Committee on Reactor Safeguards sent to the Congress its evaluation of the NRC safety research program.[183] This evaluation recommended that research be conducted on a high priority basis in the area of the man-machine interface. Such research would include an examination of the potential for and consequences of human errors. Furthermore, the ACRS recommended exploration of computer-controlled automation in the control room and that control room equipment emphasize diagnostic information that would simplify decisionmaking. The ACRS indicated that, along with development of advanced computers and graphic displays for the control room by industry, independent NRC research is necessary; i.e., research to support the "licensing review" of the advanced control room designs and to develop criteria, guides, and standards. The ACRS also recommended that the NRC conduct a more systematic review and evaluation of operational experiences at U.S. and foreign nuclear powerplants.

Analysis of the TMI-2 accident, in our opinion, has highlighted the importance of the application of human factors principles to control room design, operator training, and procedures. Although additional research in this area may be justified, the time has come to write standards and modify existing and new powerplant control room design, procedures, and training programs.

Other Precursors

In addition to the precursors discussed previously, others should be mentioned. The Electric Power Research Institute (EPRI) has sponsored a number of research projects to evaluate the application of human factors in control room design. One such report, EPRI NP-309,[65] describes a study conducted by the Lockheed Missiles and Space Company, Inc. of Sunnyvale, Calif. Lockheed evaluated five recently operational nuclear powerplants using human engineering expertise and standards developed in other industries.[184]

The report discusses various deficiencies found in the five plants. The findings are typical of those in the precursors discussed earlier. These include lack of attention to control room design, poor designs of individual control panels, inappropriate placement of instruments and controls, unreliable indicators and use of negative indications, complexity of the annunciator-warning systems, underuse of proven coding techniques, and inconsistencies in labeling.

The EPRI report concluded that:

> As a first priority, a detailed set of applicable human factors standards must be developed and industry-wide acceptance should be promoted.... In addition to a comprehensive set of standards, a need is perceived for human factors engineering design guides specific to the needs of the nuclear power industry.[185]

Another study "Human Engineering of Nuclear Power Plant Control Rooms and its Effects on Operator Performance," prepared for the NRC by the Aerospace Corporation of El Segundo, Calif., was published during February 1977 as Aerospace Report No. ATR-77(2815)-1. The Aerospace Corporation evaluated the effects of human engineering on operator performance in the control room. It specifically examined what Aerospace considered to be the three general groups of factors that influence operator performance in fulfilling their responsibilities in the control room:[186]

- control room and control system design,
- operator characteristics, and
- job performance guides.

In conducting its study, the Aerospace Corporation's study group visited 10 facilities containing 18 control rooms and 3 control room simulators.[187]

As a result of its study, the Aerospace Corporation made three recommendations to NRC:

1. Development of a regulatory guide to provide directions to the utilities in human engineering of control rooms; the guide should be designed to

encourage an increased rate of incorporation of advanced control and display concepts.[188]

2. A thorough analysis of LER data on personnel errors to establish meaningful cross-correlation of results of plant status in relation to licensing at the time of the accident, operational power levels, equipment and control elements involved, event significance, radioactivity release, etc.[189]
3. A detailed study of the programmed malfunctions provided in the software routines of current simulators to determine whether they "have the capability ... to provide student operators with the level of training needed to minimize operator errors under conditions of severe stress." It was further recommended that the study evaluate the "effectiveness of operator training in severe accidents on a simulator which does not realistically model the control board layout of the plant for which the operator is to be licensed (or relicensed)."[190]

We found that virtually no action had been taken by the NRC to implement these recommendations.

7. RECOMMENDATIONS

Our investigation found that operator actions and inactions had significant impact on the course of the TMI-2 accident. Actions that adversely affected the course of the accident should not be simply viewed as operator error. Facets of control room design, emergency procedures, operator training, and previous operator experience and practices had a significant impact on the operator's response to the accident. When viewed from a human factors perspective, this impact may have effectively precluded the operator from preventing or ameliorating the accident.

Thus, we conclude that the integration of human factors principles and disciplines into all facets of the design, construction, operation, maintenance, testing, and regulation of nuclear powerplants will significantly improve nuclear safety. Within this context, the following recommendations should be implemented:

1. The NRC should develop an interdisciplinary human factors capability. The organizational unit should be placed at a sufficiently high level within the NRC to ensure its impact throughout the NRC.
2. The NRC should require the development and implementation of formal human factors programs by utilities, vendors, and architect-engineer organizations. These programs should ensure the application of human factors principles to all aspects of plant design, construction, and operation including plant maintenance, health physics protection, and radioactive waste handling.
3. The NRC should promulgate detailed regulations requiring the application of human factors principles to design of new nuclear powerplant control rooms.
4. The NRC should initiate a program of control room enhancement. This program should have near term and long term goals. In the near term, the NRC should conduct an onsite human factors evaluation of control rooms in operating plants and plants for which operating licenses are imminent. This evaluation should be staffed by experienced human factors personnel. Where human engineering deficiencies in accident-related information display are found, expeditious corrections should be required. On a long term basis, the NRC should conduct an indepth evaluation of nuclear powerplant control rooms to determine the adequacy of the man-machine interface. On the basis of this evaluation, the NRC should require modifications in those control rooms that NRC determines necessary to ensure adequate safety.
5. Additional diagnostic operational aids, such as logic trees or disturbance analyzers, should be required in all control rooms. To expedite this recommendation, it may not be necessary initially to apply safety criteria such as redundancy to hardware additions.
6. The NRC should certify and approve operator training facilities, training instructors, and training curricula. The NRC should evaluate the overall training programs periodically.
7. The NRC should require increased emphasis on diagnostic, hypothesis testing, and accident response training of control room operators. Such training should include simulator operations that reflect the operating characteristics of the control room for which the operators are licensed or for which licensing is sought.
8. Analysis and research should be performed to determine operator responsibilities and actions during normal and abnormal conditions. The results of this analysis should be used as a basis for determining operator selection and training criteria, manning levels, and procedural format and content.
9. Until recommendation 8 can be implemented, the NRC should require that all hot operations

shifts be manned by a minimum of one SRO, two CROs, and one additional individual with demonstrated and tested capabilities in abnormal system diagnosis. Two of these individuals should be required in the plant control room at all times. Less than this minimum should not be allowed at any time during hot operation.

10. The NRC should require powerplant operations supervisors and management personnel to be trained in investigation techniques and reporting methods for events involving human behavior.

11. The NRC should conduct an immediate review of the emergency procedures of all operating plants to identify and correct problems associated with symptoms identification, technical accuracy, and systems compatibility.

12. The NRC should develop improved methods for measuring operator performance and the effectiveness of training programs in meeting training objectives. These methods should use written examinations, oral examination in the operators' plant, and assessments of performance on simulators reflecting normal and abnormal plant conditions.

13. The NRC should consider the licensing of auxiliary operators and testing and maintenance personnel for specific plants.

REFERENCES AND NOTES

[1]NRC, Investigation into the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement, NUREG-0600, August 1979.

[2]Report of the President's Commission on the Accident at Three Mile Island, "The Need for Change; The Legacy of TMI," October 1979.

[3]Sec. II.C, Plant Behavior and Core Damage, Subsec. II.C.2.

[4]Sec. II.D, Alternative Accident Sequences, Subsec. II.D.1.

[5]Three Mile Island Nuclear Station Unit 2 Emergency Procedures 2202-1.5, Pressurizer System Failure, Rev. 3, September 29, 1978.

[6]This design is a specific violation of accepted human factors principles as contained in Mil-Std-1472B, para. 5.2.2.1.5.

[7]Bryan interview (IE), Tape 278, at 12.

[8]Sec. I.C Precursor Events, Subsec. I.C.15.

[9]NRC, "Investigation into the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement," NUREG-0600, August 1979, at I-1-3.

[10]Bryan interview (IE), Tape 278, at 6.

[11]Bryan interview (IE), Tape 198, at 25–28.

[12]Faust, Frederick, Scheimann, and Zewe dep. at 154–155.

[13]Oversight Hearings Before a Task Force of the Subcommittee on Energy and the Environment of the Committee on Interior and Insular Affairs, House of Representatives, 96th Congress, 1st Sess. 96-8, Part I (May 9, 10, 11, and 15, 1979) at 170.

[14]Chwastyk dep. at 71–72.

[15]Sec. II.A, Appendix: Tabular Chronology, 24 minutes 58 seconds.

[16]Sec. II.A, Appendix: Tabular Chronology, 2 hours 18 minutes.

[17]NRC, "Investigation into the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement," NUREG-0600, Sec. 4.3, August 1979.

[18]Sec. II.D.

[19]Pres. Com. Hearing (May 30, 1979) at 194.

[20]Three Mile Island Nuclear Station Unit 2, Emergency Procedure 2202-1.3, Loss of Reactor Coolant/Reactor Coolant System Pressure.

[21]The Essex Corp., "Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2," NUREG/CR-1270, Vol. 1: Final Report, December 1979.

[22]Sec. II.D.

[23]See Faust, Frederick, Scheimann, and Zewe dep. at 124.

[24]Hearings Before Committee on Interior and Insular Affairs, Task Force on Three Mile Island (May 11, 1979, 10:00 a.m.) at 43.

[25]Letter from E. Frederick to J. Seelinger, TMI-2 Superintendent for Technical Support, April 1978.

[26]The Essex Corp., "Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2," NUREG/CR-1270, Vol. 1: Final Report, December 1979, at 25.

[27]*Id.* at 23.

[28]Sec. II.C, Plant Behavior and Core Damage, Subsec. II.C.1.

[29]The Essex Corp., "Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2," NUREG/CR-1270, Vol. 1: Final Report, December 1979, at 24.

[30]*Id.* at 27.

[31]NRC, "Program Summary Report," NUREG-0380, Vol. 3, No. 5, May 1979, at 3–2.

[32]Proposed Amendment to 10 C.F.R. Part 50, "Licensing of Production and Utilization Facilities," to add Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits," (32 Fed. Reg. 10213) July 11, 1967. pp. 10213–10216.

[33]Met Ed, "Preliminary Safety Analysis Report (PSAR), Three Mile Island Nuclear Station-Unit 2."

[34]Institute of Electrical and Electronic Engineers, Inc., "Proposed IEEE Criteria for Nuclear Power Plant Protective Systems," IEEE 279, August 1968.

[35]Institute of Electrical and Electronic Engineers, Inc., "Trial Use Standard Criteria for Safety Systems for Nuclear Power Generating Stations," IEEE Standard 603-1977.

[36]Met Ed, "Preliminary Safety Analysis Report (PSAR), Three Mile Island Nuclear Station-Unit 2," Vol. 3, Sec. 7.4.

[37]For example: U.S. Military Standard 1472B, Human Engineering Requirements for Military Systems, Equipment and Facilities, December 31, 1974. This standard includes detailed design guidelines, principles, and requirements.

[38]NRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," NUREG-75-087, September 1975.

[39]Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," May 1973.

[40]Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," Rev. 1, August 1977.

[41]Military Specification "Human Engineering Requirements for Military Systems, Equipment and Facilities," MIL H-46855, para. 3.2.1.3, May 2, 1972.

[42]Regulatory Guide 1.114, "Guidance on Being Operator at the Controls of a Nuclear Power Plant," Rev. 1, Nov. 1976.

[43]Institute of Electrical and Electronics Engineers, Inc., "IEEE Recommended Practice for the Design of Display and Control Facilities for Central Control Rooms of Nuclear Power Generating Stations," IEEE Standard 566-1977, July 1977.

[44]The Essex Corp., "Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2," NUREG/CR-1270, Vol. 1: Final Report, December 1979, at 116–118.

[45]*Id.* at 118.

[46]Met Ed, "Final Safety Analysis Report (FSAR), Three Mile Island Nuclear Station-Unit 2," Vol. 6, Secs. 7.4, 7.5.1.2, 7.5.2.

[47]The Essex Corp., "Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2," NUREG/CR-1270, Vol. 1: Final Report, December 1979, at 35.

[48]*Id.* at 35–37.

[49]*Id.* at 37–39.

[50]*Id.* at 40.

[51]*Id.* at 40, 42.

[52]*Id.* at 38, 40.

[53]*Id.* at 43.

[54]The Essex Corp., "Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2," NUREG/CR-1270, Vol. 1: Final Report, December 1979, at 43–46.

[55]*Id.* at 48.

[56]*Id.* at 47.

[57]*Id.* at v–vi.

[58]*Id.* at 31.

[59]*Id.* at 62.

[60]*Id.* at 37.

[61]*Id.* at 40.

[62]*Id.* at 42.

[63]*Id.* at 43–46.

[64]*Id.* at 47.

[65]Lockheed Missiles & Space Co. Inc., "Human Factors Review of Nuclear Power Plant Control Room Design," EPRI NP-309 (Research Project 501), November 1976.

[66]A. D. Swain, Sandia Laboratories, "Preliminary Human Factors Analysis of Zion Nuclear Power Plant," 76-6503, October 1975.

[67]Lockheed Missiles & Space Co. Inc., "Human Factors Review of Nuclear Power Plant Control Room Design," EPRI NP-309 (Research Project 501), November 1976.

[68]*Id.* at 1-6 to 1-11.

[69]*Id.* at 1-12.

[70]*Id.* at 1-12 and 1-13.

[71]*Id.* at 1-13 to 1-16.

[72]*Id.* at 1-18.

[73]*Id.* at 1-13.

[74]*Id.* at 1-20.

[75]*Id.* at 1-21.

[76]A. D. Swain, Sandia Laboratories, "Preliminary Human Factors Analysis of Zion Nuclear Power Plant," 76-6503, October 1975, at 13.

[77]*Id.* at 6–7.

[78]The Essex Corp., "Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2," NUREG/CR-1270, Vol. 1: Final Report, December 1979, at 72.

[79]*Id.* at 74.

[80]*Id.* at 77–78.

[81]*Id.* at 78.

[82]*Id.* at 79.

[83]42 U.S.C. Sec. 2137.

[84]Sec. 11(r) of the Atomic Energy Act of 1954, 42 U.S.C. 2014 o.

[85]10 C.F.R. 55.4(f).

[86]10 C.F.R. Part 55, 28 FR 3197, April 3, 1963.

[87]Examples are: Regulatory Guide 1.8, "Personnel Selection and Training," and Regulatory Guide 1.134, "Medical Evaluation of Nuclear Power Plant Personnel Requiring Operating Licenses."

[88]See NUREG-0094, "NRC Operator Licensing Guide—A Guide for the Licensing of Facility Operators Including Senior Operator," July 1976. This document provides guidance on details such as the logistics of scheduling and administering examinations, topical content of examinations, and copies of pertinent NRC forms.

[89]For example, Regulatory Guide 1.8 refers to ANSI 18.1-1971, "Selection and Training of Nuclear Power Plant Personnel," as providing an acceptable basis for the selection and training of nuclear powerplant personnel except for the position of supervisor-radiation protection. Also, Regulatory Guide 1.134 refers to ANSI N546-1976, "Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plant," for use by an examining physician in determining whether an applicant is medically qualified.

[90]10 C.F.R. Sec. 55.11(b).

[91]10 C.F.R. Sec. 55.11(a).

[92]Regulatory Guide 1.134 "Medical Evaluation of Nuclear Power Plant Personnel Requiring Operating Licenses," Rev. 1, March 1979.

[93]Appendix A to Part 55, "Requalification Programs for Licensed Operators of Production and Utilization Facilities," was effective in 1973 at the same time requalification program requirement for utilities was established in 10 C.F.R. Sec. 50.54.

[94]10 C.F.R. Sec. 55.20.

[95]10 C.F.R. Sec. 55.21 and 55.22.

[96]10 C.F.R. Sec. 55.23. The Commission may administer a simulated operating test prior to the initial criticality of the reactor upon a showing by the utility involved of immediate need for the applicant's (for an opera license) service and of the applicant's qualifications, 10 C.F.R. Sec. 55.25.

[97]10 C.F.R. 55.11(b).

[98]NRC, "NRC Operator Licensing Guide—A Guide for the Licensing of Facility Operators, Including Senior Operators," NUREG-0094 (Rev. 1 of WASH-1094), Sec. IX, July 1976.

[99]*Id.* at Sec. VIII.

[100]*Id.* at Appendix F.

[101]*Id.* at Appendix G.

[102]NRC Public Meeting, "Briefing on Procedures for Qualifying Reactor Operators," April 20, 1979, at 65.

[103]Summary of Telephone Conversation Between H. Ornstein, NRC/SIG, and R. Zechman, TMI Training Staff, December 13, 1979.

[104]10 C.F.R. 55.10(a).

[105]10 C.F.R. Part 55, Appendix A.

[106]Met Ed, "Final Safety Analysis Report (FSAR), Three Mile Island Nuclear Station-Unit 2," Vol. 9, Sec. 13.2.

[107]*Id.* at Sec. 13-2.1.1.1.d.

[108]A 1- or 2-week simulator course encompassing 40 hours per week, half of which is devoted to classroom work and the other half to actual simulator work.

[109]Letter from Miller, Met Ed, to Collins, NRC, dated July 5, 1977.

[110]Letter from Miller, Met Ed, to Collins, NRC, dated September 7, 1977.

[111]Met Ed, "Final Safety Analysis Report (FSAR), Three Mile Island Nuclear Station-Unit 2," Vol. 9, Secs. 13.2.2-13.2.2.4.

[112]Enclosure (B&W Cold Licensing Training Program) to Memorandum from Elliot, B&W, to Collins, NRC, March 14, 1977.

[113]Met Ed, "Preliminary Safety Analysis Report (PSAR), Three Mile Island Nuclear Station-Unit 2," Vol. 3, Sec. 12.2.1.

[114]This conclusion was reached upon examination of TMI operator license applications and supporting documents; confirmed in a telephone conversation, NRC-SIG (Ornstein) to NRC-OLB, (P. Collins), December 18, 1979.

[115]The Essex Corp., "Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2, NUREG/CR-1270, Vol. 1: Final Report," at 83, December 1979.

[116]*Id.* at 90.

[117]*Id.* at 92.

[118]*Id.* at 91.

[119]*Id.* at 93.

[120]*Id.* at 94.

[121]*Id.* at 95.

[122]*Id.* at 97-98.

[123]*Id.* at 99.

[124]*Id.* at 96.

[125]10 C.F.R. 50.54(k).

[126]Three Mile Island Nuclear Station Unit 2, Technical Specifications, Appendix A to License No. DPR-73, NUREG-0432, Sec. 6.2.2. February 8, 1978.

[127]Operator Examination Scores were obtained from NRR-OLB's files.

[128]NRC Public Meeting, "Briefing on Procedures for Qualifying Reactor Operators," April 20, 1979, at 52.

[129]*Id.* at 67, 78.

[130]Records of operators training courses were obtained from NRR-OLB's files (license applications).

[131]NRR Information Report (Denton) to Commissioners "A Statistical Profile of Licensed Operators and Senior Operators and a Statistical Profile of Commercial Airline Pilots and Merchant Marine Engineering Personnel," SECY 330A, dated May 9, 1979.

[132]Records of operators experience were obtained from NRR-OLB's files (license applications).

[133]Naval nuclear powered ships have accumulated over 1800 reactor-years of operation since the USS *Nautilus* first put to sea in 1955, as compared with more than 460 reactor-years accumulated in land-based nuclear powerplants (NRC, "Program Summary Report" Vol. 3, No. 7, July 20, 1979). Admiral Rickover indicated that there has never been an accident or any significant release of radioactivity that has had a significant effect on the environment during that time. (Ref. 134 at 2).

[134]Statement of Admiral H. G. Rickover, USN Director, Naval Nuclear Propulsion Program Before the Subcommittee on Research and Production of the Committee on Science and Technology, U.S. House of Representatives, May 24, 1979, at 14.

[135]*Id.* at 15.

[136]*Id.* at 28-30, 43, 56.

[137]NRR Information Report to the Commission, "Comparison of the Navy/Industry Training and Requalifications Program," SECY-79-330D, July 5, 1979, at 3.

[138]*Id.* at 4.

[139]Statement of Admiral H. G. Rickover, USN Director, Naval Nuclear Propulsion Program, before the Subcommittee on Research and Production of the Committee on Science and Technology, U.S. House of Representatives, May 24, 1979, at 87.

[140]*Id.* at 81-84.

[141]*Id.* at 42.

[142]*Id.* at 53.

[143]*Id.* at 85-88.

[144]*Id.* at at 93-95.

[145]*Id.* at 105-107.

[146]U.S. Atomic Energy Commission, "Evaluation of Incidents of Primary Coolant Release From Boiling Water Reactors," WASH-1260, October 1972, at 1.1.

[147]*Id.* at 43.

[148]*Id.* at 44.

[149]*Id.* at 29.

[150]*Id.* at 43.

[151]*Id.* at 28.

[152]*Id.* at 28.

[153]*Id.* at 43.

[154]*Id.* at 42.

[155]*Id.* at 28.

[156]*Id.* at 43.

[157]*Id.* at 44.

[158]*Id.* at 45.

[159]*Id.* at 46.

[160]Memorandum from L. M. Muntzing, NRC, to F. E. Kruesi, J. F. O'Leary, and L. Rogers, "Implementation of Recommendations of the Regulatory Study Group," November 28, 1972.

[161]A. D. Swain, Sandia Laboratories, "Preliminary Human Factors Analysis of Zion Nuclear Power Plant," 76-6503, October 1976.

[162]Sec. I.C, Precursor Events.

[163]Lind dep. at 59-66 (Pres. Com.).

[164]Attachment "Important Technical Reactor Safety Issues Facing the Commission Now or in the Future," to Memorandum from S. H. Hanauer, Technical Advisor EDO, NRC to Commissioner Gilinsky, NRC, Subject: Technical Issues, dated March 13, 1975, at 2

[165]Investigation of Charges Relating to Nuclear Reactor Safety Hearings Before the Joint Committee on Atomic Energy, 94th Cong., 2nd Sess. (February 18, 23, and 24, and March 2 and 4, 1976), Vol. 1: Hearings and Appendixes 1-11, at 913.

[166]*Id.* at 929.

[167]*Id.* at 930.

[168]*Id.* at 930-931.

[169]*Id.* at 555.

[170]*Id.* at 933-934.

[171]*Id.* at 935.

[172]*Id.* at 936.

[173]*Id.* at 555.

[174] *Id.* at 556.

[175] *Id.* at 937.

[176] *Id.* at 938.

[177] A. D. Swain, Sandia Laboratories, "Preliminary Human Factors Analysis of Zion Nuclear Power Plant," 76-6503, October 1975, at 3.

[178] *Id.* at 6–7.

[179] *Id.* at 1.

[180] *Id.* at 10.

[181] NRC, "Plan for Research to Improve the Safety of Light-Water Nuclear Power Plants—Report to the Congress of the United States of America," NUREG-0438, April 1978, at 23.

[182] *Id.* at 42.

[183] Advisory Committee on Reactor Safeguards, NRC, "1978 Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program. A Report to the Congress of the United States of America," NUREG-0496, December 1978.

[184] Lockheed Missiles and Space Co., Inc., "Human Factors Review of Nuclear Power Plant Control Room Design," EPRI NP-309 (Research Project 501), November 1976.

[185] *Id.* at 1-28.

[186] The Aerospace Corporation, "Human Engineering of Nuclear Power Plant Control Rooms and its Effects on Operator Performance." ATR-77 (2815)-1, February 1977, at 1-1.

[187] *Id.* at 1-7.

[188] *Id.* at 7-13.

[189] *Id.* at 7-14.

[190] *Id.* at 7-15.

F ENVIRONMENTAL AND SOCIOECONOMIC IMPACTS

1. INTRODUCTION

This section addresses the social and economic effects and the effects on aquatic biota and fisheries of the Three Mile Island accident. The sequence of events during the accident is described in other parts of the report. The purpose of this section is to explore response to the accident and the resulting economic, social, and associated effects—both immediate and continuing—on the public, local government, and institutions. Although a number of studies dealing with these various aspects of the accident are continuing, the data and information compiled to date are sufficient to draw general conclusions with considerable confidence.

The sources of information provided herein are several. In the social and economic areas, most of the information was developed or obtained within the framework of an NRC-sponsored study of the local and regional, social and economic effects of the TMI accident. This study, conducted by Mountain West Research, Inc. of Tempe, Ariz., and coordinated with the Commonwealth of Pennsylvania through the Office of Policy and Planning, was commissioned immediately after the accident and is scheduled to be completed in the summer of 1980. The study has several components that are used in this section. Most of the information on evacuation behavior, individual costs, and concerns and attitudes are from a telephone survey conducted by Mountain West in the vicinity of TMI in late July and early August 1979. A randomized quota sample of 1500 households within 55 miles of TMI was used. The quota's size was greater for households closest to TMI. Additional information on the accident's effects on individuals was developed in a number of personal interviews with area residents conducted by investigators from Mountain West Research, Inc. The report of the Task Group on Behavioral Effects to the President's Commission on the Accident at Three Mile Island was a major information source on mental health and psychological stress. Other studies concerned with mental health and psychological stress currently in progress were also reviewed, and discussions have been held with officials of the Pennsylvania Department of Health and staff members of the Hershey Medical Center.

Information on the economic effects on various sectors of the local economy has been collected by several agencies of the Commonwealth of Pennsylvania as part of the State's analysis of the accident. The overall effort is being coordinated by the Office of Policy and Planning. Additional information on the economic effects was compiled by Mountain West Research, Inc. through interviews with members of the business community.

Information on the accident's governmental and institutional effects was obtained from several sources. The primary source of information was local government officials, including those in civil defense, through interviews by Mountain West Research, Inc. Supplementary information was obtained from documentation developed on emergency preparedness by the President's Commission, the Federal Emergency Management Agency (FEMA), and the NRC Special Inquiry Group. Care has been taken to differentiate clearly between actual observed effects and effects believed by the public to have occurred but that have not actually been documented.

2. BACKGROUND

The Three Mile Island Nuclear Power Station occupies a site consisting of Three Mile Island and adjacent islands in the Susquehanna River, approximately 10 miles southeast of Harrisburg, Pa. Three Mile Island is located in Londonderry Township, the southernmost township of Dauphin County (Figure II-55). York County is across the river to the west of TMI, and Lancaster County is immediately to the south on the east side of the river. The nearest population concentrations are in the boroughs of Goldsboro (population 576) 1.25 miles to the east in York County; Royalton (1975 population 1131) 2 miles to the north; and Middletown (1975 population 9877) 3 miles to the north. The Harrisburg International Airport is about 3 miles upriver from TMI.

A high percentage of the land in the region either is in agricultural use or is woodland. Population is mostly concentrated in the cities and boroughs. Housing developments in townships, which tend to be more rural, have absorbed much of the population growth in the area as well as some of the outmigration from Harrisburg. The two largest concentrations of population within 15 miles of the TMI center are in the cities of Harrisburg in Dauphin County (1975 population 58 274) and York in York County (population more than 50 000). The heaviest concentration of population is to the north and northwest of TMI, centering on Harrisburg. Population is estimated to be 38 000 within 5 miles of TMI, 165 000 within 10 miles, and 636 000 within 20 miles.

Local government is basically decentralized. General administrative responsibilities reside in counties, cities, boroughs, and townships. Independent school districts typically are not contiguous with municipalities. Dauphin County has 23 townships, 1 city, 16 incorporated boroughs, and 10 independent school districts. Within 20 miles of TMI, there are 6 counties and more than 90 municipalities.

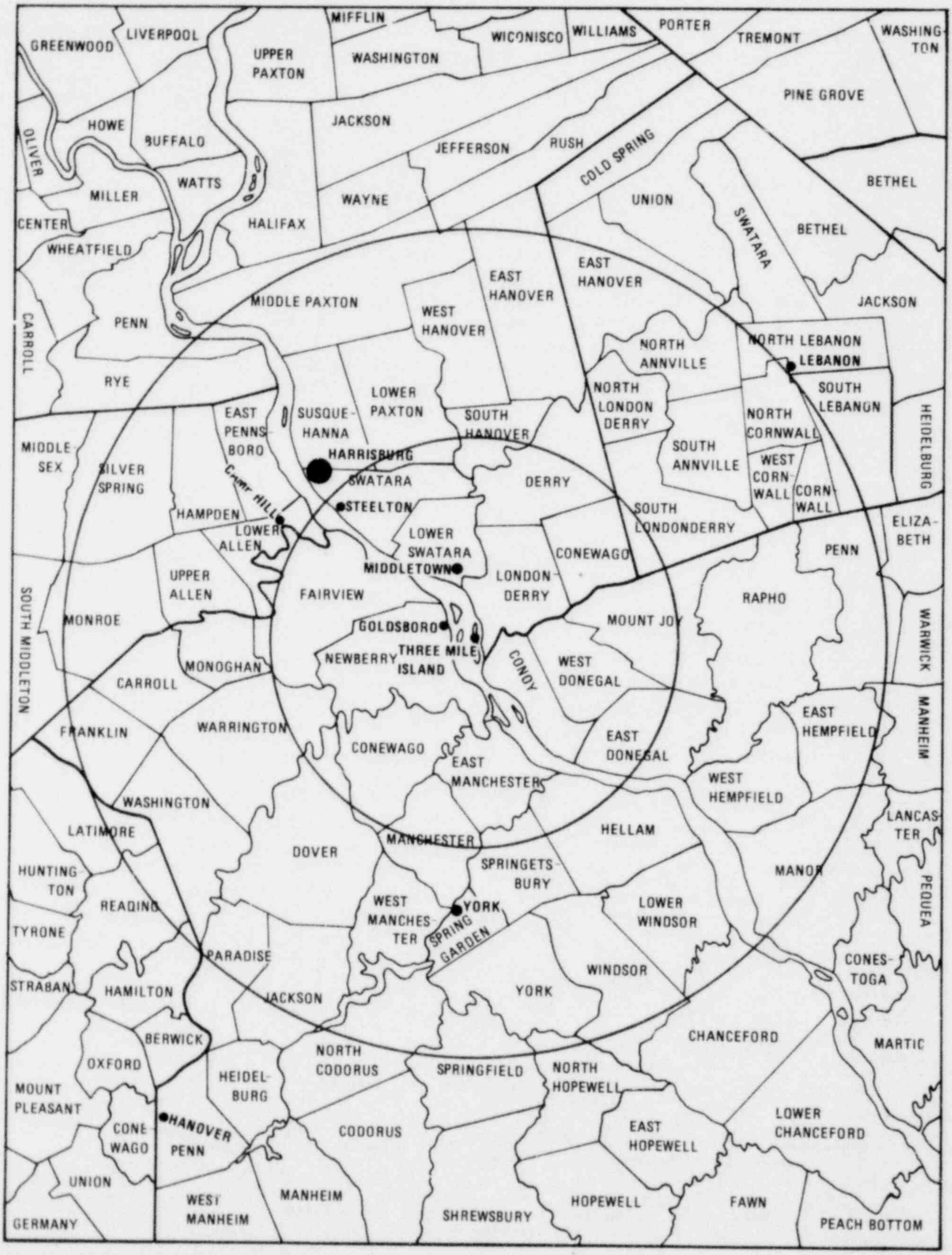
The economic base of the region is diversified. Agriculture, manufacturing, recreation–tourism, and State and Federal Governments all contribute to the region's strong economic performance. The manufacturing sector continues to be the dominant part of the economic base of the area. Unemployment rates have been lower than both State and National rates.

3. INFORMATION FLOW DURING THE EMERGENCY

The character of the information available to the public on the accident was a major determinant of evacuation behavior and the public's perception of danger from TMI-2. To a considerable extent, information available through the media was confusing and frequently conflicting. Met Ed, the basic source of information on the status of the plant, quickly lost credibility. The NRC was a source of contradictory and, upon several occasions, alarming information. Whereas the local media tended to be restrained and nonspeculative in its coverage of the accident, the national media and the media outside the area tended to be more speculative. Such speculative accounts of the accident were fed back to local residents by more distant friends and relatives. A number of aspects of the public information environment, especially those relating to the utility–Federal–State interface with the media, were described in Section III.D, "Information Provided to the News Media."

Several local media events not elsewhere covered were significant in forming the public's perception of possible danger from the accident. The first public broadcast of the accident was aired by radio station WKBO at 8:25 a.m., Wednesday, March 28.[1] Following up on evidence that something was wrong at TMI, the station broadcast Met Ed's assurances that there was no danger to the general public.

The second media event not previously covered was a Thursday afternoon Harrisburg radio station broadcast of an interview with Dr. Ernest Sternglass, of the University of Pittsburgh, who recommended evacuation of pregnant women and preschool children.[2] This interview was used in the



NOTE: The inner circle encloses the 10-mile radius, and the outer circle encloses the 20-mile radius.

FIGURE II-55. Map of Area

station's hourly newscast. A comment on Sternglass's statement by the station's disc jockey gave listeners the impression it was an official order for women and children to leave the area.[3] By this time, rumors of evacuation were growing.

The number of calls received by local authorities from area citizens, radio stations, and civil defense officials indicated strong outside reinforcement of local anxieties by rumors from more distant family and friends who were themselves responding to a considerable amount of speculative news. In response to the Sternglass interview, a representative of the Pennsylvania Department of Health went on the air to assure the public that evacuation was not necessary. Aware of increasing public alarm, Governor Thornburgh held a press conference at 5:15 p.m. Thursday to assure the public there was no cause for alarm, no danger to public health, and no reason to disrupt daily routines.

On Friday, March 30, in the early morning, confusion over the release of radioactive gases and the NRC's advice to evacuate resulted in the Pennsylvania Emergency Management Agency (PEMA) informing the civil defense director of Dauphin County that an official order to evacuate a 5-mile area around the plant was probably imminent.[4] The Dauphin County director put fire departments within a 10-mile zone on standby and advised all school districts to keep students inside and buses ready to move. He then went on a local radio station to inform the public of the possibility of evacuation and to give basic information on where to go and what to take. It was emphasized that the broadcast was a warning, not an order to evacuate. Telephone service in the Harrisburg area immediately deteriorated because of the large number of incoming calls in response to the announcement. More than six times the usual number of calls were placed following the radio message.[5] At 10:46 a.m., PEMA informed the local media that a general evacuation was not imminent.

A survey of the TMI-area population by the Michigan State University Department of Geography shortly after the accident provides information on the source and time of initial information received by the public.[6] The survey area covered a 25-mile radius from TMI, the most intense sampling being near the plant. Of the total sample, 35% of all respondents first heard of the accident on Wednesday morning, 62% had heard by Wednesday night, and all had heard by Friday (Figure II-56). In spite of the wide news coverage, 17% did not hear of the accident until Friday. The survey also showed that those residents close to TMI, with the greatest supposed risk, did not hear of the accident as early as

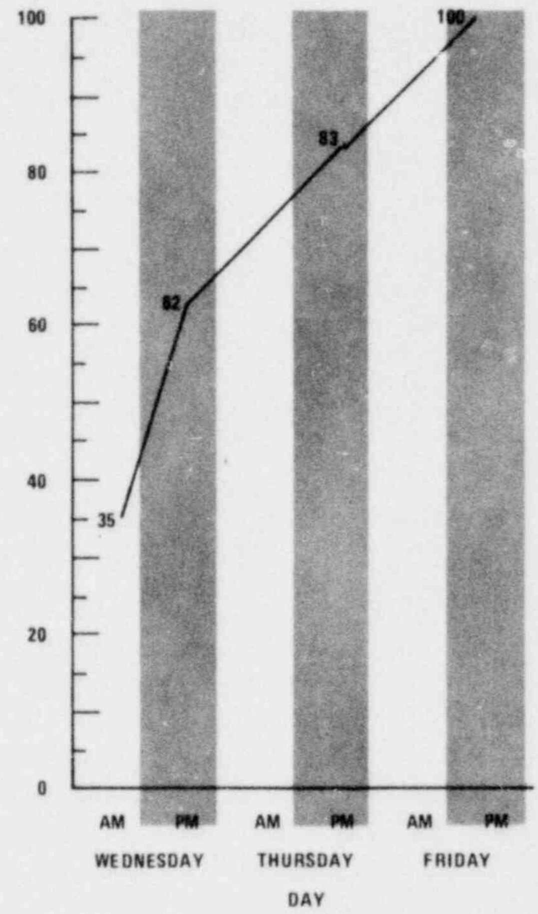
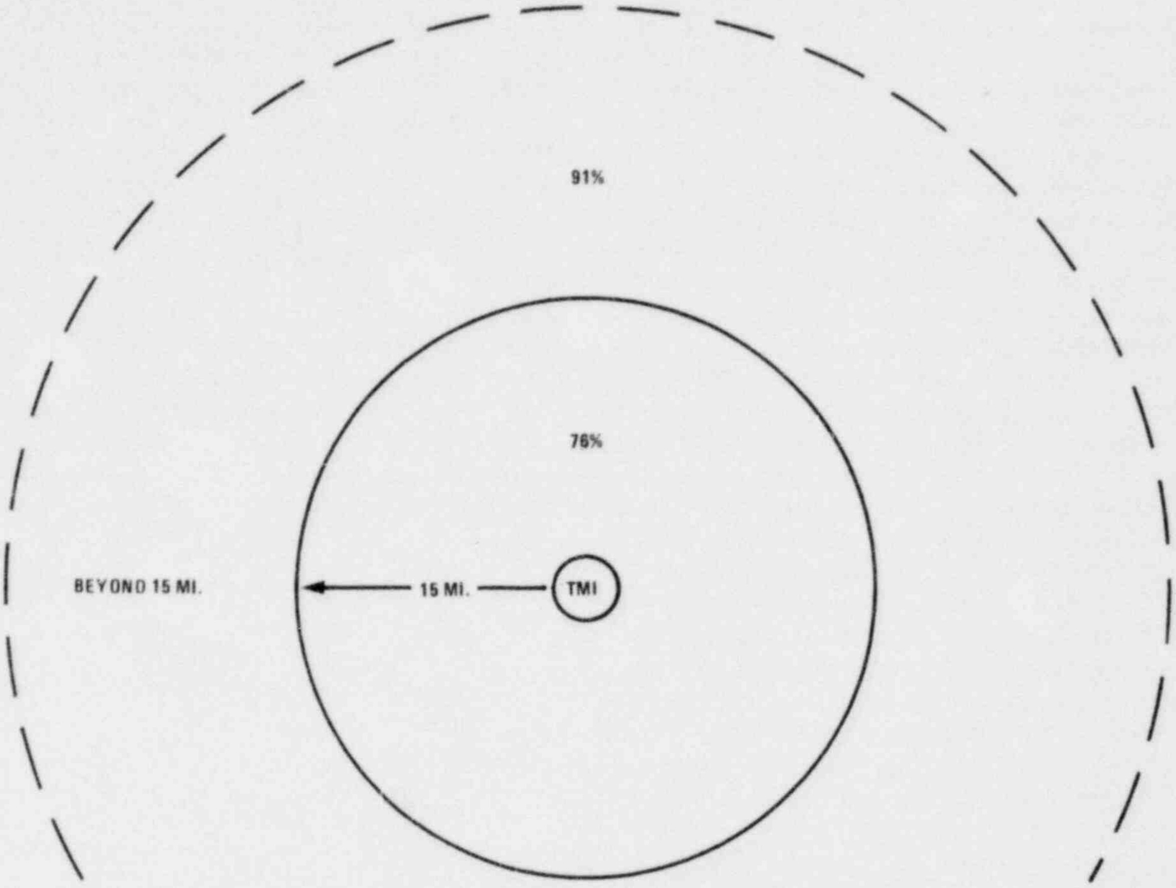


FIGURE II-56. Cumulative Percent of Local Population Who Received Information on Accident, by Day

more distant residents. Twenty-four percent of the respondents within 15 miles of TMI did not know of the accident until Friday, compared with 9% of the respondents beyond 15 miles (Figure II-57). A slightly smaller percentage of respondents within 6 miles of TMI first received the news on Wednesday and on Thursday, compared with respondents residing over 15 miles away. Initial sources of information were the following: radio (56%), friends and family (26%), television (14%), and newspaper (3%) (Figure II-58).

4. PUBLIC RESPONSES

Data in this section, unless otherwise cited, are from the NRC-TMI telephone survey conducted under contract by Mountain West Research, Inc. Many of these data have previously been reported in greater detail.[7]



FICURE II-57. Percent of Respondents Aware of TMI Accident Before Friday, by Distance From Plant Site

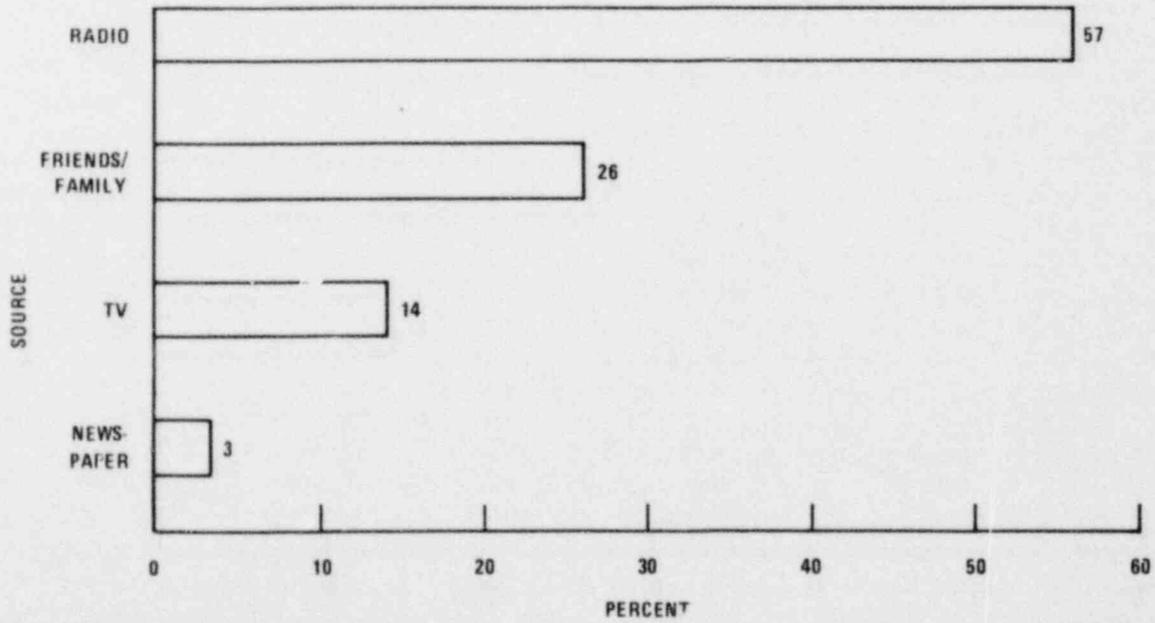


FIGURE II-58. Initial Sources of Information, by Percent of Population Sampled

a. Evacuation

Local interest in news of the accident was undoubtedly high on Wednesday and Thursday, but news from TMI was generally reassuring, and as a rule the public went about business as usual. Concern on the part of some residents, however, was high. Of the households covered in the NRC telephone survey, 7% reported at least one member evacuating on Wednesday, and another 7% reported evacuating on Thursday.

During the emergency period, there was no evacuation order issued. An evacuation advisory by the Governor on Friday concerned only pregnant women and preschool children within 5 miles of TMI: approximately 4200 individuals in nearly 2800 households. This target group, however, accounts for only about 3% of evacuees within 15 miles of TMI. An estimated 20% of this group did not evacuate. Within 15 miles of TMI, however, about 144 000 individuals (almost 39% of the population) in about 50 000 households evacuated. The advisory covered 1% of the population and involved less than 2% of the households within 15 miles.

The percentage of the population that evacuated decreased with increased distance from TMI (Figure II-59). The percentage of individuals evacuated by distance was 60% within 5 miles, 44% within 5 to 10 miles, and 32% within 10 to 15 miles. (The 60% estimate for the 5-mile radius is consistent with the 5-mile census conducted by the Pennsylvania Department of Health.) The percentage of individuals evacuated and households affected by evacuation decreased significantly beyond 15 miles. The percentage of households having at least one evacuee was 64% within 5 miles, 48% within 5 to 10 miles, and 32% within 10 to 15 miles. A much higher proportion of pregnant women and of children under 6 years of age evacuated at each distance (Figure II-60) compared with the proportion evacuated of the general population. The proportion evacuated of the general population decreased significantly over the 15-mile radius. The percentage of pregnant women and children under 6 years of age who evacuated was 83% within 5 miles, 70% within 5 to 10 miles, and 55% within 10 to 15 miles.

Although there was a declining percentage of individuals and households affected by the accident as distance from TMI increased, the absolute number of individuals and households that evacuated within 15 miles increased with distance. The total estimated number of individuals who evacuated is 21 000 within 5 miles, 56 000 within 5 to 10 miles, and 67 000 within 10 to 15 miles (Figure II-61).

A substantial number of additional persons were directly affected because they remained at home during the emergency after other household members had evacuated. It is estimated that an additional 18 000 persons within 15 miles of the station were affected in this way. The percentage of the total population affected by having households separated during a stressful time was 9% in the 0–5-mile ring, 5% in the 5–10-mile ring, and 4% in the 10–15-mile ring (Figure II-62).

Evacuations began at a slow pace and accelerated to a peak on Friday, March 30, when more than half of the total evacuees moved away from their households. There was a decline after Friday, but evacuations continued until April 10. The major outflow of evacuees, by percentage of the total, was as follows.

Date	*Evacuees Departing*
March 28–29	14
March 30	51
March 31	18
April 1	11
April 2–10	6

Return of the evacuees to their households began before the evacuations were completed; 15% had returned by Sunday night, April 1, 1979. The returns, by percentage of the total evacuees, accumulated as follows.

Date	*Evacuees Returned*
April 1	15
April 4	54
April 8	80
April 30	99

The median distance traveled by evacuees from the 15-mile area was 100 miles; 23% evacuated no farther than 45 miles, and 52% evacuated 90 miles or more. Persons living closer to TMI tended to travel shorter distances than those living farther from the plant. Thirty-four percent of evacuees from within 5 miles of TMI evacuated 45 miles or less; the corresponding rate for those from the 5–10-mile ring was 24%, and for the 10–15-mile ring 19%.

Evacuees stayed in all parts of the country, but the largest number (72%) remained in Pennsylvania. Pennsylvania was followed by other States nearby: New Jersey (6.6%), Maryland (5.8%), and Virginia (3.8%). Other more distant destinations included California, Oklahoma, and Florida. In all, 21 States received evacuees (Figure II-63).

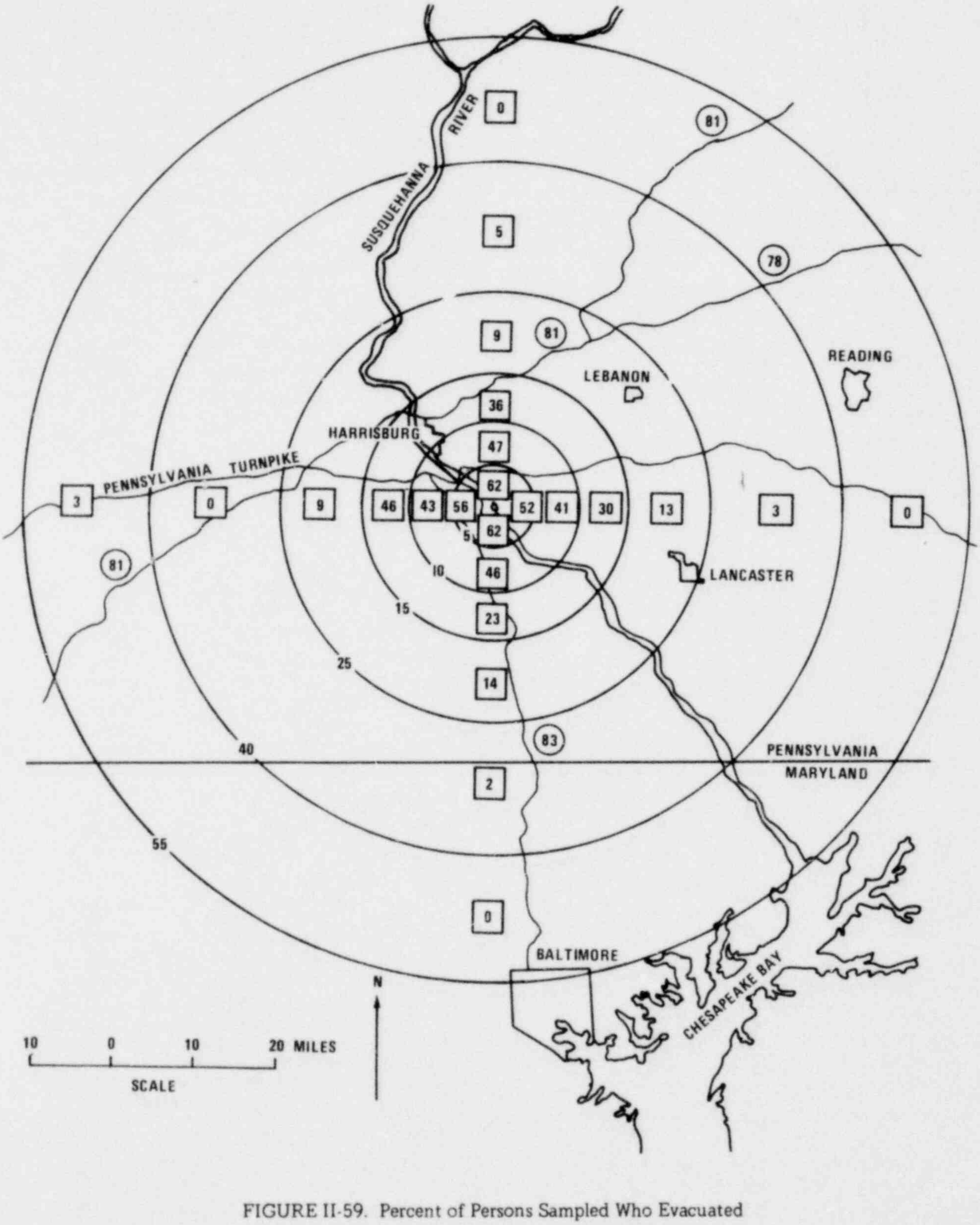


FIGURE II-59. Percent of Persons Sampled Who Evacuated

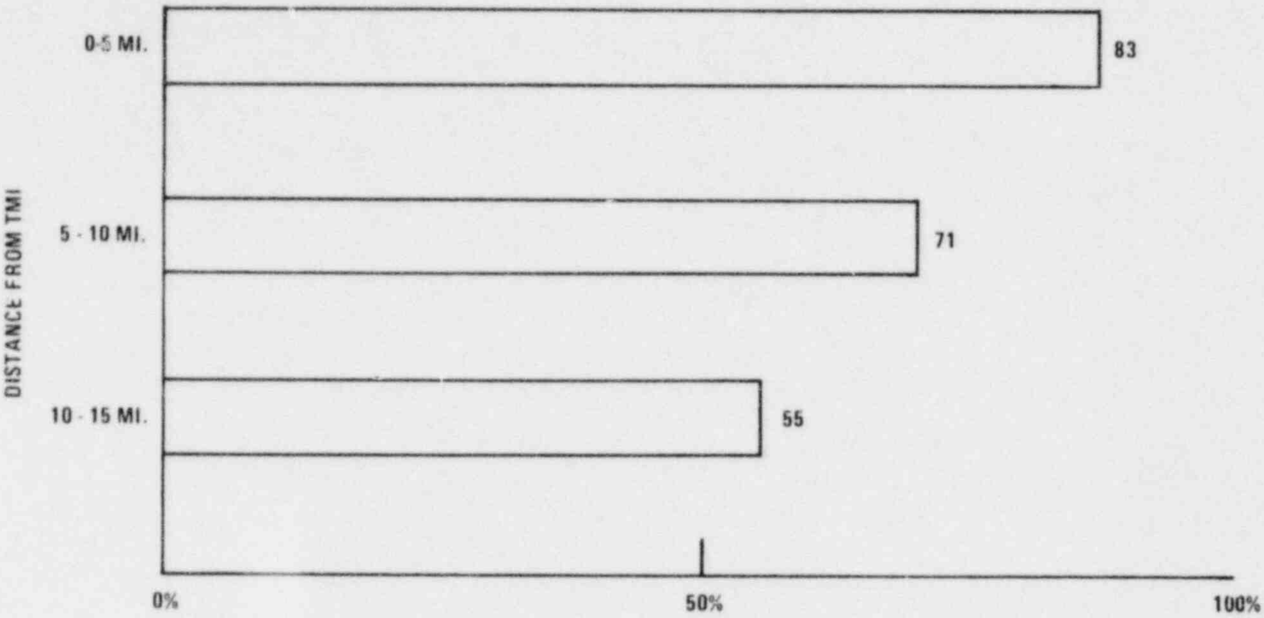


FIGURE II-60. Percent of Pregnant Women and Preschool Children Sampled Who Evacuated, by Distance From TMI

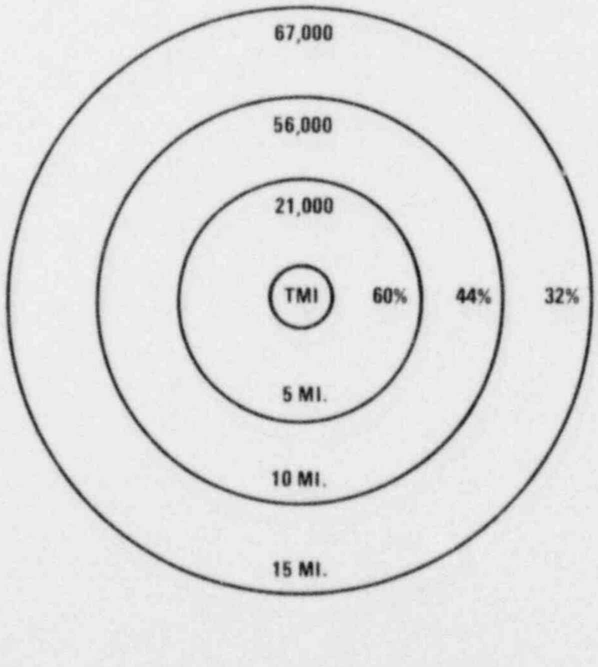


FIGURE II-61. Estimated Number and Percent of Population Evacuated, by Distance From TMI

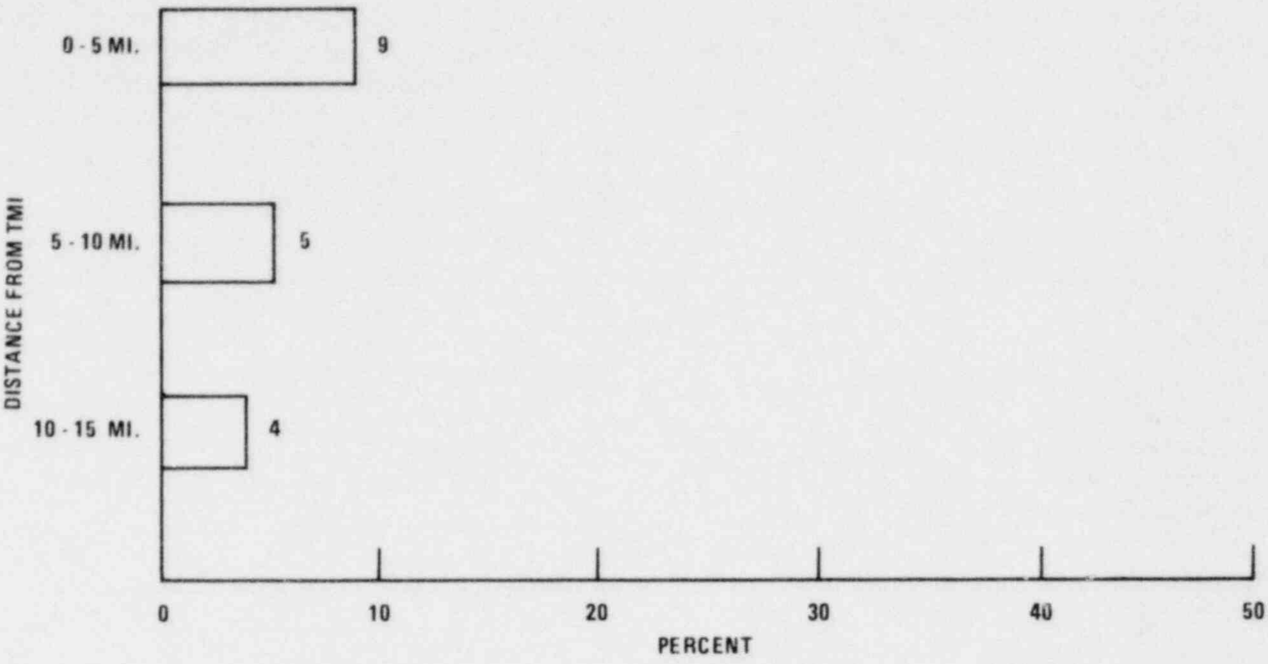


FIGURE II-62. Percent of Individuals Sampled Who Experienced Separation of Households, by Distance From TMI

FIGURE II-63. Distribution of Evacuees, by Percent of Total

The majority of persons (78%) evacuated to the home of a friend or relative. Hotels or motels were the destination of only 15% of the evacuees.

The following reasons for evacuation were offered by respondents to the survey.

Reason	*Percentage of Respondents*
Situation seemed dangerous	91
Information on the situation confusing	83
Wished to avoid the confusion or danger of a forced evacuation	76
To protect children	61
Pressure from outside the family	28
Previously planned trip	5

Confusing information influenced a higher percentage (89%) of respondents within the 5-10-mile ring in their decision to evacuate than it did evacuees from within 5 miles (74%) or from 10-15 miles (81%). In both the 5-10-mile and 10-15-mile rings, 78% of respondents said they wanted to avoid the confusion or dangers of a forced evacuation, whereas 65% within 5 miles gave this reason. Other reasons indicated for evacuation were more evenly distributed in responses from the three distance categories.

The reasons for deciding not to evacuate were compared for two categories of respondents: members of households in which some persons evacuated, and those in which no one evacuated. Clear differences in the reasons for not evacuating were apparent in the two groups. Although households in which some evacuated and some did not were very sensitive to the danger of the situation (only 14% "saw no danger"), the primary reasons they remained behind were that they were unable to leave their jobs (64%) or would have left only if they had received an evacuation order (52%). Many (45%) felt that whatever happened was in God's hands; 34% were concerned about looters (Figure II-64).

The households where none evacuated exhibited a quite different pattern. The overriding reason given for staying was that they were waiting for an evacuation order (71%), followed by the feeling that whatever happened was in God's hands (65%). The third reason for staying was that they saw no danger (36%), which was mentioned two and one-half times as frequently by households in which no one evacuated, compared with households where some members evacuated and others did not. Together, these three reasons suggest that households where everyone stayed placed greater confidence in authority than households which evacuated. Although the ability to leave their jobs was a consideration for this group, it was not the overriding concern that it was for nonevacuees in households where some persons evacuated (Figure II-64).

When asked if a particular piece of information influenced their decisions to evacuate, respondents gave a variety of answers. Evacuees gave the following reasons most frequently.

Information Influencing Decision	*Percentage of Respondents (two coded for respondents citing two or more)*
Hydrogen bubble	30
Conflicting reports	19
Governor's advice to evacuate	14
Threat of forced evacuation	14
News bulletins	9
Urging by family member	6
No particular information	25

Only 14% considered the Governor's advice to evacuate as critical in their decision to do so. News of the hydrogen bubble, however, was thought to be critical in the evacuation decision of 30% of evacuees. This percentage accounts essentially for all of the evacuees after Friday, March 30.

Specific questions about communication of the Governor's advice to evacuate were asked in households with pregnant women or with children younger than 6 (98% of the respondents in such households were aware of the Governor's advice). Most respondents heard the statement virtually as soon as it was given: about two-thirds of the sample heard it on TV or radio; about 11% heard from friends; and the rest heard in some other way. Two-thirds said they were told neither to listen to a specific radio or TV station for additional information nor that they would be transported to an evacuation center. However, two-thirds of the respondents were aware where they could expect to be evacuated. Only one-fourth said they were told who would be responsible for conducting the evacuation.

All respondents were asked about expected notification procedures in case of a general evacuation. Radio (62%) and TV (56%) were seen as the primary means of notification. Respondents were asked additional questions about persons they expected would be responsible for emergency services. A majority of respondents (64%) felt that an

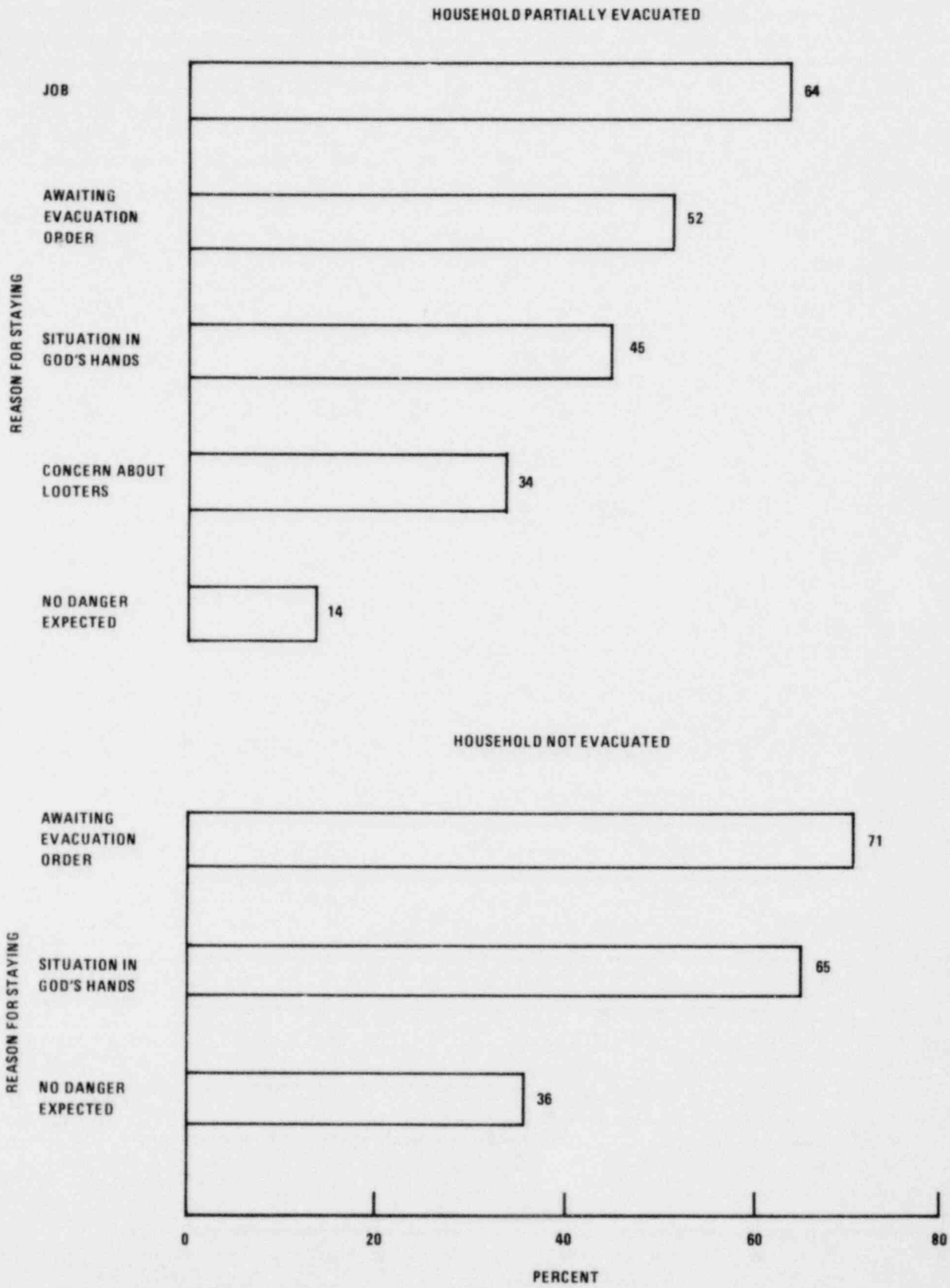


FIGURE II-64. Respondents' Reasons for Not Evacuating, by Percent emergency group would be responsible for their food and shelter during an emergency but that they themselves would be responsible for their transportation (66%).

b. Credibility of Information

The Governor of Pennsylvania and the NRC were cited as the most useful sources of information during the emergency period (Figure II-65). Fifty-seven percent of the informants rated information from each of these sources as useful or extremely useful. Only 11% of the respondents found information from Met Ed to be useful or extremely useful. Sixty percent of the respondents found Met Ed information totally useless. Respondents within (compared with those beyond) 15 miles of TMI were more likely to say that the information given by the Governor and the NRC was extremely useful.

During the emergency period, respondents found local TV and radio to be the most useful media forms (Figure II-66). Sixty-seven percent of the respondents found each of these forms useful or extremely useful. Less than 10% found them totally useless. National network TV was slightly less useful, with 55% of respondents answering useful or extremely useful. The print media ranked behind all radio and TV. Comments offered by respondents suggested that poor scores for information received from friends and relatives resulted because this information was perceived as rumor rather than fact.

When asked of their overall satisfaction with the way they were given information during the emergency, half the respondents were either very satisfied (12%) or mostly satisfied (37%), and the other half were either very dissatisfied (22%) or mostly dissatisfied (29%). Generally, those farther from TMI were more likely to be satisfied with the information they received than were those closer to TMI. Those who were most likely to be dissatisfied were pregnant women (71%) and students (75%). Also, evacuees were more likely to be dissatisfied (64%) than were those who did not evacuate (47%).

c. Levels of Public Concern During Emergency Period

Considerable attention has been focused on the nature and extent of psychological distress resulting from the accident at TMI. The NRC–TMI survey provides a perspective on the levels of concern within the affected population (Figure II-67). At the

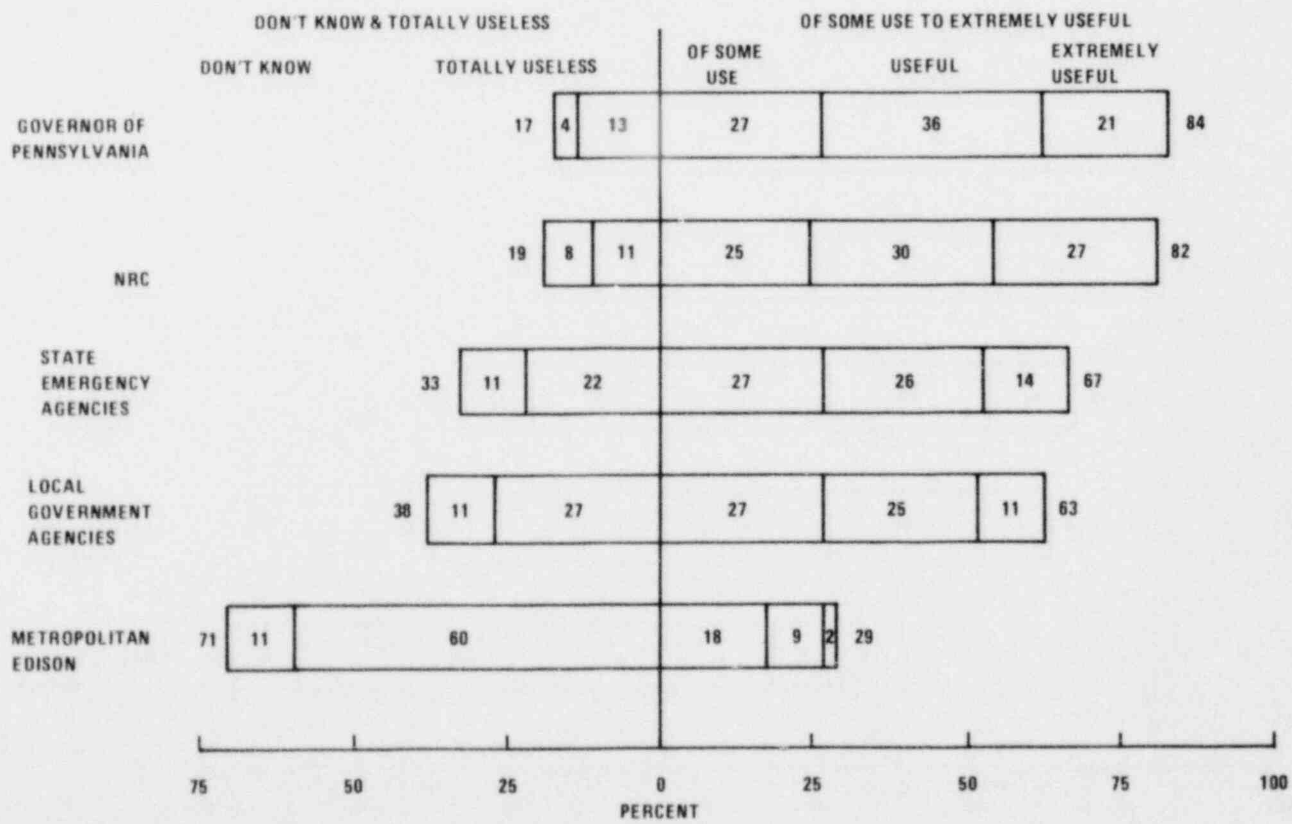


FIGURE II-65. Respondents' Evaluation of Information Sources, by Percent of Respondents

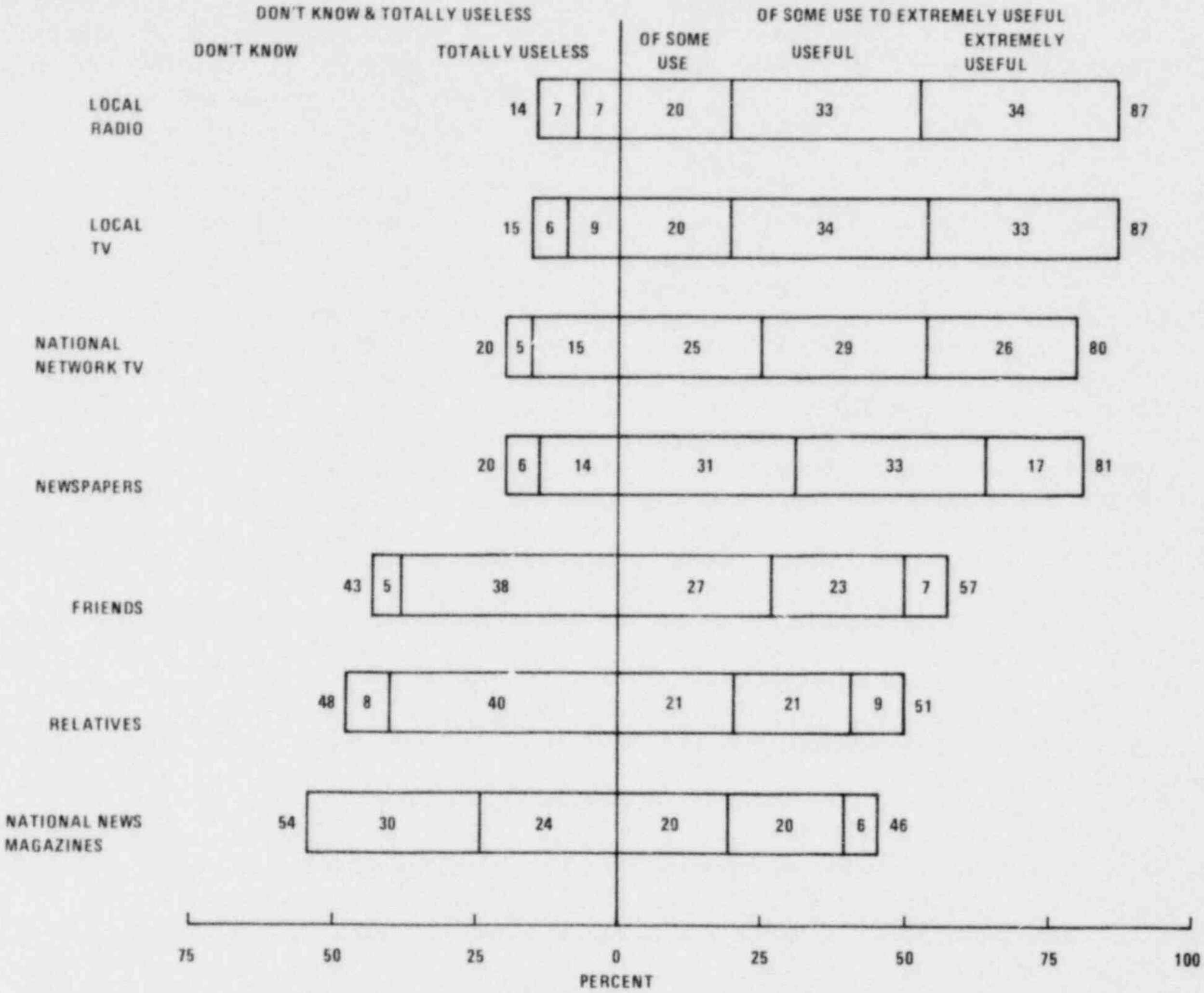


FIGURE II-66. Respondents' Evaluation of News Media, by Percent of Respondents

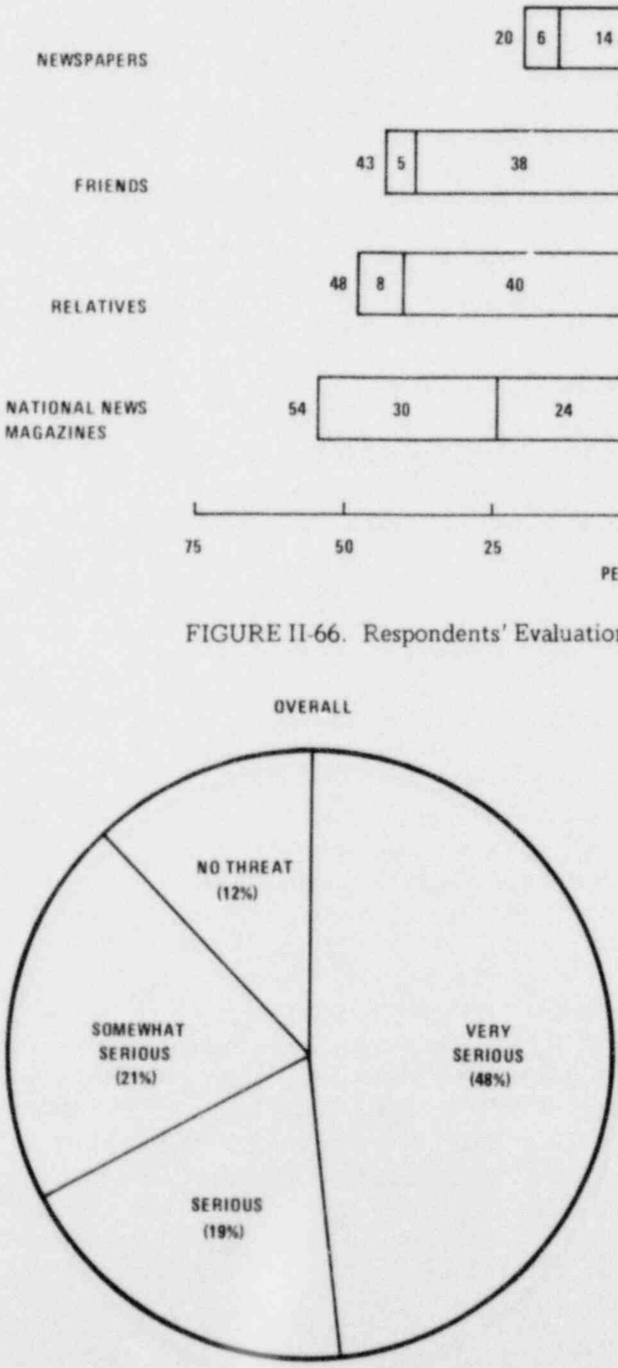


FIGURE II-67. Public Assessment of TMI Accident as a Threat to Family Safety time of the accident, 48% of respondents believed that the situation at TMI was a "very serious" threat to family safety; 19% believed the threat was "serious." The perception of threat was clearly related to the distance from the Three Mile Island station. Within the 5-, 10-, and 15-mile rings, there was little difference in the percentage of respondents seeing the accident as a "very serious" threat (slightly less than 50% overall). In the 15 to 25 mile ring, however, those seeing a "very serious" threat fell to 28%. Beyond 25 miles, roughly 20% of respondents perceived a "very serious" threat. Pregnant women were much more likely than average (64%) to view the accident as a "very serious" threat. Evacuees (63%) were nearly twice as likely as nonevacuees (38%) to perceive a "very serious" threat at TMI.

Despite the high degree of perceived threat of TMI to family safety during the accident period, most individuals did not tend to be very upset. Twenty-two percent were extremely upset, and 29% were not at all upset. Essentially, there was an even split among those quite upset, somewhat upset, and a little upset. Seventy-two percent of pregnant women were extremely or quite upset, however. Distance was a consideration in the degree of distress experienced. Households within 15 miles of TMI were twice as likely as those beyond 15 miles to have at least one member who was quite or extremely upset. Those households in which no one evacuated were more than twice as likely as evacuating households to have no member upset.

Noteworthy studies dealing with the psychological effects of the accident on the population surrounding TMI have been or are being supported by the Pennsylvania Department of Health, the Hershey Medical Center, and the President's Commission on the Accident at Three Mile Island. Some of these studies are multiyear; others have not yet been reported; and those that are now available have, in most cases, been completed by the President's Commission's Task Group on Behavioral Effects and reported in the group's staff analysis report.[8]

The Task Group on Behavioral Effects expanded upon the sample survey studies undertaken by several researchers from colleges and universities near the TMI site. These surveys employed measures of psychological effects with small samples of the general population or high risk groups such as mothers of preschool children. Because studies of the behavioral effects on workers had not been initiated, the task group undertook such studies. It was found that the accident increased stress and had a strong demoralizing effect on the population in the vicinity of TMI, especially on teenagers and mothers of preschool children. These ill effects diminished rapidly in the months following the accident for all groups other than TMI workers, but higher than normal distrust of authorities involved with TMI continued. Workers involved at TMI, however, showed high trust in the utility.

Asked whether anyone in the household had considered moving because of the accident, 19% of the respondents said "yes." Within 5 miles, 30% answered "yes." Affirmative answers were given more frequently in the north and the west. Those who had considered moving were likely to be younger and more highly educated than those who had not. Evacuees were more than three times as likely to say they had considered moving as nonevacuees (33% versus 9%). Among those who had considered moving, 22% had definitely decided to move (4% of total households). This percentage, extended to the total population, implies that a total of 5100 households within 15 miles of the plant had decided to move.

Preliminary tabulations of the population census conducted by the State Department of Health within 5 miles of TMI identified 147 households as having moved between April 1, 1979 and the end of July of the same year. This figure is about 1% of the estimated total number of households in the 5-mile area. Only 29% of moved households that had been contacted by late August 1979 indicated that their move was motivated by the accident; therefore, less than three-tenths of 1% of the households within 5 miles are estimated to have moved by the end of July because of the accident. It is likely that at least seven-tenths of the movement was normal turnover.

d. Continuing Effects

Continuing disruption from the TMI accident of individuals within the region appears to be slight. No damage to public or private facilities and no loss of life or injury was incurred. At the time of the NRC-TMI survey in July and August, however, there was continuing concern about the safety of TMI and the effects the accident would have on the local economy. The NRC-TMI survey identified a small percentage of respondents who believed that their households were continuing to experience effects of the accident (12% of the households that evacuated and 4% of those that did not). The most frequently mentioned effects were higher electric bills, reduced real estate values, and declines in business. A small group of respondents (3%) had considered changing jobs as a result of the accident, and about half of these were taking definite steps to do so. Evacuees considered changing jobs more frequently (6.4%) than nonevacuees (1.5%), but were no more likely to have taken definite steps toward that end. Ninety percent of the respondents said that their normal daily activities were unchanged by the accident. Those living 0–5 miles to the west of the plant were more likely to say that there was substantial change in their day-to-day activities. Changes most frequently mentioned were that TMI was always in the back of their minds (6%) and that they avoided the area (2%). Evacuees were more likely than nonevacuees to report at least a minimal disruption.

A majority of the respondents said that the economy of the area will be hurt by the accident (60%) rather than helped (6%) or not affected (34%) (Figure II-68). Those residents within 15 miles of TMI were more likely to respond that the area will be hurt by the accident than those farther away. Evacuees were more likely to think that the area's economy will be hurt and less likely than nonevacuees to think that there will be no effect.

Continuing concerns with the economic effects of TMI on the area were related, at least in part, to the continuing concern with the safety of the Three Mile Island station (Figure II-69). Twenty-two percent of respondents said TMI continued to be a very serious threat to their families, and 19% thought it continued to be a serious threat. On the other hand, 28% said it was not a threat. Concern about the safety of TMI is closely related to the perception of

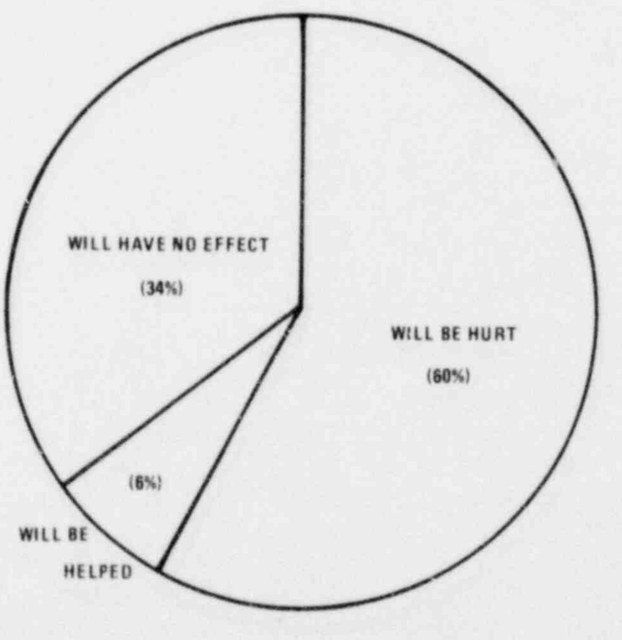


FIGURE II-68. Public Assessment of the Economic Effects of the TMI Accident.

radioactive emissions. Figure II-70 compares the postaccident level of concern about radioactive emissions with both preaccident concerns and those during the emergency period. Four months after the accident, the level of concern about emissions was slightly less than during the accident, but much higher than before the accident. Forty-one percent of respondents were still very concerned about emissions. Those persons either very concerned or somewhat concerned decreased only from 86% during the accident to 75% 4 months later. Evacuees were more likely to be concerned than nonevacuees about emissions before, during, and after the accident. Both during and after the accident, respondents within 15 miles of TMI expressed greater concern with emissions than those farther than 15 miles away.

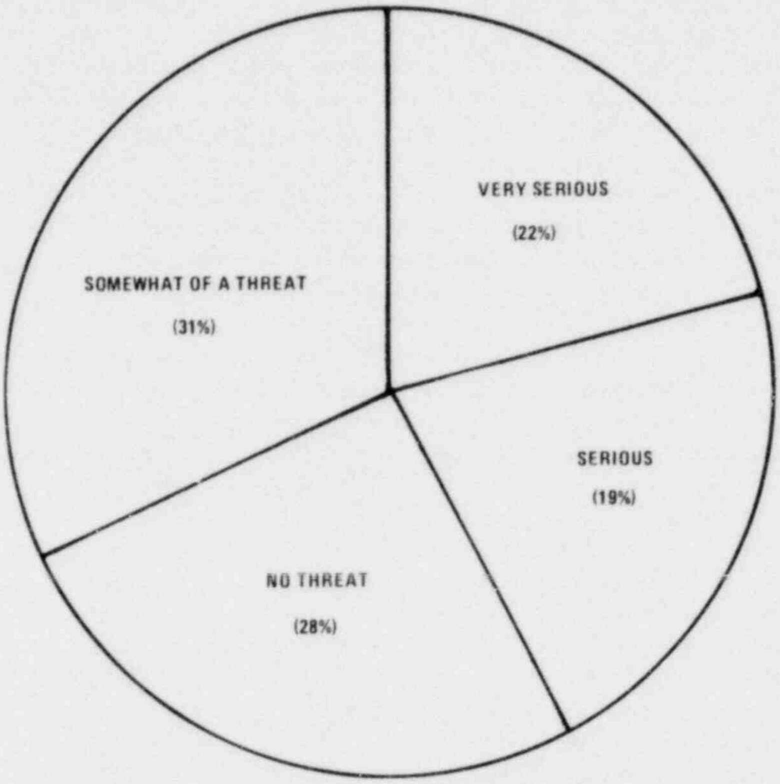
5. ECONOMIC EFFECTS

a. Background

The immediate and continuing effects of the TMI accident on the local economy have been well documented by various departments within the State Government of Pennsylvania. These studies have been coordinated by the Office of Policy and Planning and the data have been combined with additional field investigations by Mountain West Research, Inc.[9] Although there was short term and localized economic disruption, the overall economic effects are apparently of little consequence.

b. Emergency Period

Disruption to local commerce and industry was generally moderate and short lived. Few businesses shut down completely, but those closer to TMI generally suffered from loss of customers or loss of workers due to evacuation. Emergency period economic effects on residents within 15 miles of TMI have been completed from data gathered in the NRC telephone survey. These economic effects consist of income losses (or gains) plus extraordinary expenses uncompensated by insurance. The survey estimated that within 15 miles, 34 000 evacuees lost 141 000 person-days of work. Of the evacuees who lost work, 19 000 also lost pay. The median pay loss for this group was $110, although the mean loss was $271. Eleven percent of the respondents reported losing more than $500. Additionally, 8000 nonevacuees are estimated to have lost income because of loss of work. Only 7% of



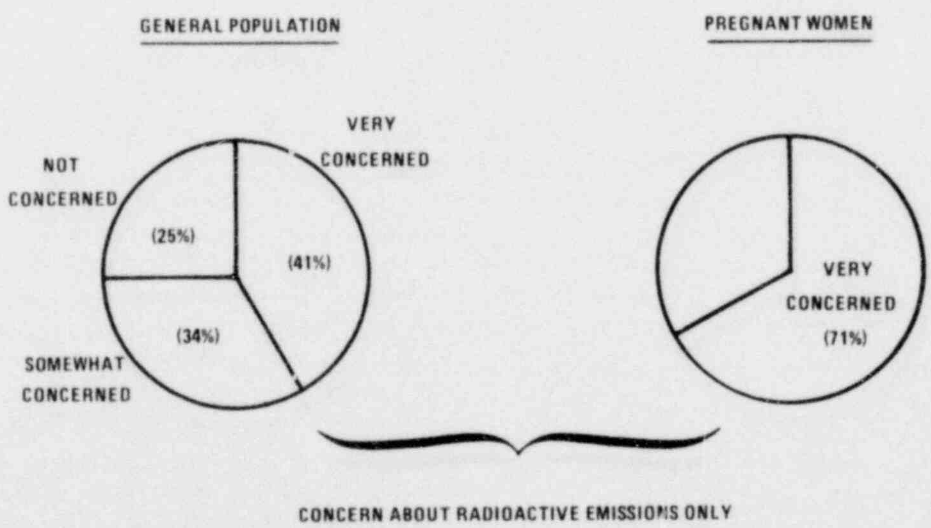


FIGURE II-69. Public Assessment of TMI as a Continuing Threat to the Area.

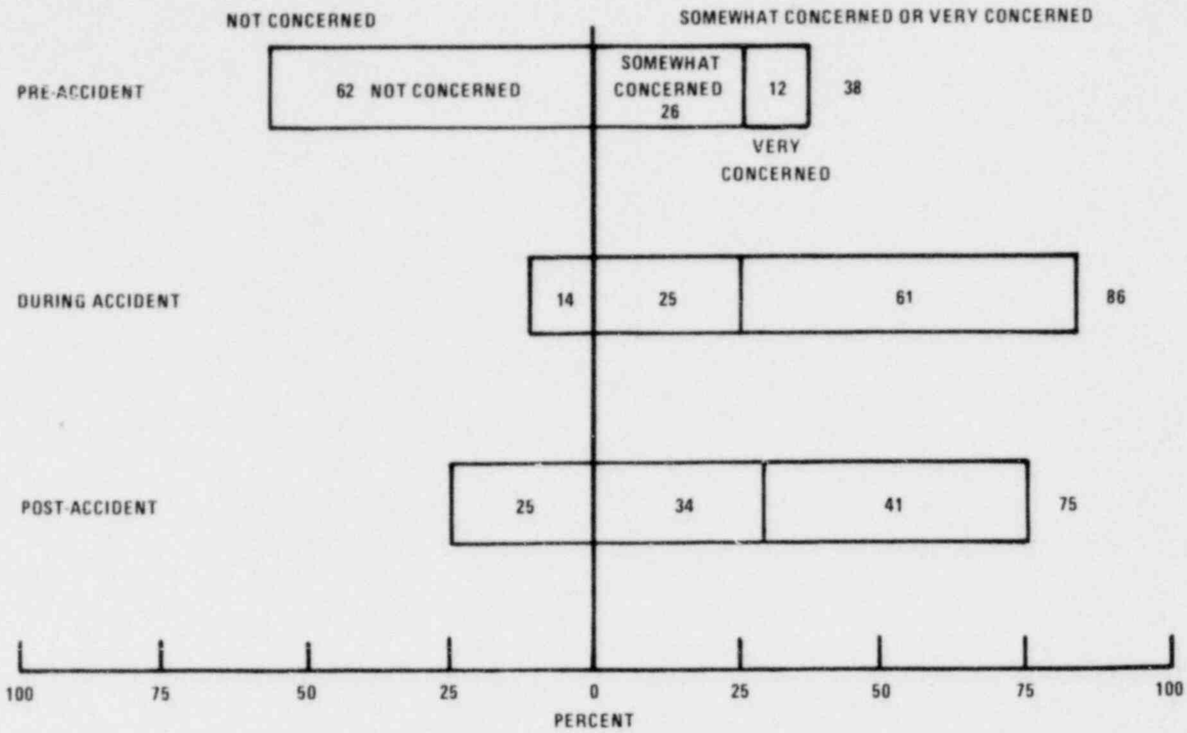


FIGURE II-70. Respondent's Concern about TMI Radioactive Emissions.

nonevacuating households reported extraordinary expenses during the emergency period, and about 8% reported a loss of family income. For those suffering losses, median extra expense was $51 and median income loss was $142. Nearly all evacuating households, however, experienced extra (out-of-pocket) expenses associated with the evacuation. Median household extra expense for evacuees was $100, but the mean, at $198, was nearly twice as high. Total costs per evacuating household increased with distance from TMI: $247 for 0 to 5 miles, $259 for 5 to 10 miles, and $342 for 10 to 15 miles (Figure II-71). This is probably related to the finding that evacuees farther from TMI traveled farther than persons living closer to the site.

Table II-64 summarizes the economic costs of the accident at TMI for households within 15 miles. The table shows that income loss contributed to about half of the short term economic costs suffered by households. The other half was due to evacuation costs and other accident-related expenses. Data from the NRC–TMI telephone survey indicate that households within 15 miles had received a total of $1 215 000 in insurance compensation at the time of the survey. Data collected by the Pennsylvania Department of Insurance are consistent with these findings. As of August 10, 1979, the Department of Insurance reported a total of $1 298 325 in private claims paid within 20 miles of TMI. When the approximately $1.2 million of insurance payments is subtracted from income loss and accident-related expenses, short term economic costs borne by households within 15 miles of TMI are about $18 million.

Assuming that the mean household income of $17 000 found in the survey holds for each of the three rings, expenses as a percentage of evacuees' annual household income were 1.4% for 0 to 5 miles, 1.5% for 5 to 10 miles, and 2.0% for 10 to 15 miles. Averaged over all households within 15 miles, expenses amounted to a little less than 1% of annual household income (Figure II-72).

The effects of the accident on local business and economy are based to a considerable extent on the evacuation of workers and customers and the threat of enforced evacuation and to a limited extent on concern for radiological protection of product. Although detailed data on daily developments are not available, it appears that for those businesses within approximately 5 miles of TMI, activity was down only slightly on Thursday, March 29, and that disruption began with the increasing concern over

TABLE II-64. Economic costs of the accident at TMI for households in the 15 mile ring

Cost	0–5 Mile Ring	5–10 Mile Ring	10–15 Mile Ring	Total for 15 Mile Ring
Costs for evacuees				
Pay loss (or gain)	$ 726 000	$1 861 000	$1 270 000	$ 3 857 000
Evacuation costs	1 719 000	2 990 000	4 119 000	8 828 000
Other expenses	108 000	75 000	763 000	946 000
Other income loss (or gain)	34 000	600 000	2 162 000	2 796 000
Insurance payments to evacuees	643 000	424 000	148 000	1 215 000
Total costs net of insurance	$1 944 000	$5 102 000	$8 166 000	$15 212 000
Costs for non evacuees				
Income loss (or gain)	140 000	1 043 000	1 412 000	2 595 000
Other expenses	29 000	122 000	255 000	406 000
Total costs for non evacuees	169 000	1 165 000	1 667 000	3 001 000
Total costs net of insurance compensation (evacuees and non evacuees)	$2 113 000	$6 267 000	$9 833 000	$18 213 000

Source: C. B. Flynn, "Three Mile Island Telephone Survey: Preliminary Report on Procedures and Findings," U.S. Nuclear Regulatory Commission, 1979.

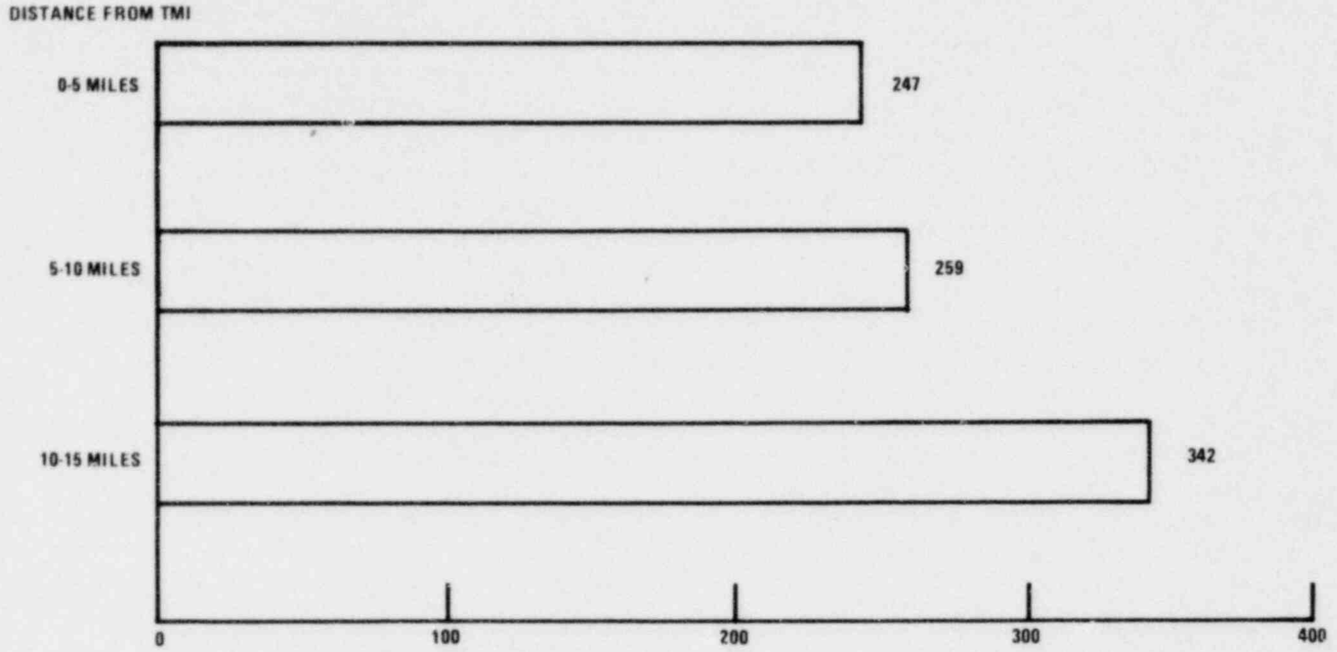


FIGURE II-71. Costs Per Evacuating Household, by Distance from TMI

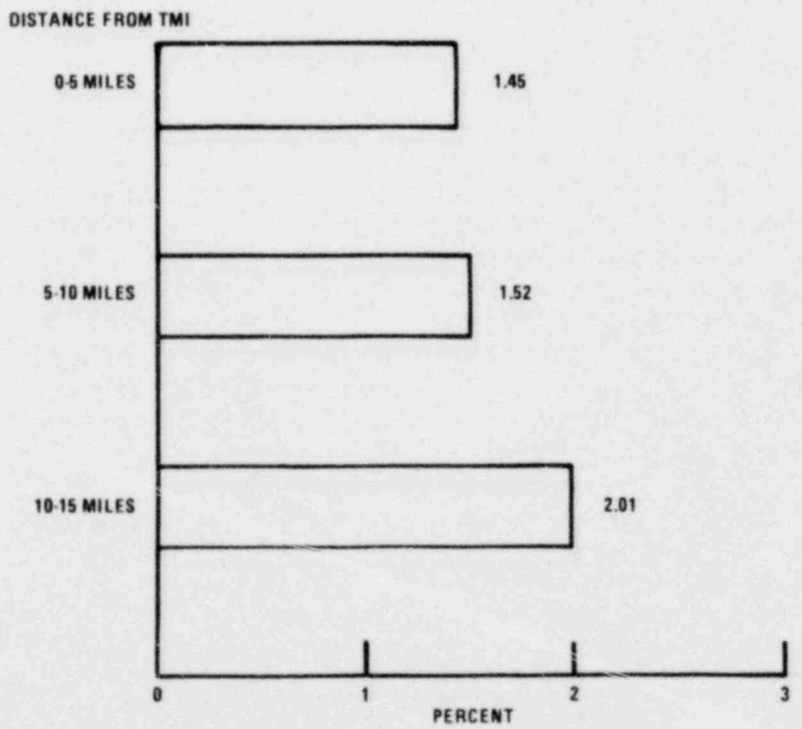


FIGURE II-72. Percent of Annual Income Lost Per Evacuating Household, by Distance from TMI.

the prospect of evacuation at midday. Preoccupation of the local population with developments concerning the accident diverted workers' and customers' attention from their normal routines. Increasing numbers of employees left their places of work on Friday afternoon. Most apparently did so with the concurrence of their employers. The only large employer known to have shut down operations was Freuhauf Corporation, located 3 miles north of TMI. (The plant was closed Monday, April 3, through Wednesday, April 5.) When the plant reopened on Thursday, the work force was near normal. Other large firms in the area remained open and attempted to maintain production in spite of substantial absenteeism. Most firms did not discourage absenteeism but had a policy of no work, no pay. Some firms paid those who were covered by the Governor's advisory, but not others. A few firms continued to pay all those employees who evacuated.

Many firms had to contend with evacuation preparations and materials protection. Some firms had production methods such as food processing, which could neither be easily shut down nor left unattended. A forced evacuation would have been costly to these firms in damaged equipment and loss of goods in process. Business interruption claims filed with nuclear insurers show that wages paid to absent workers were uncommon. More than three-quarters of the claims have been for loss of sales. A few claims were for interruption or loss of production and for expenses in preparing for evacuation or in product testing.

Large demands for cash to be used in evacuation were anticipated by banks. As an example, the largest bank in Middletown, the Commonwealth National Bank, requested employees to work their regular hours and overtime to service their customers and help reduce a stressful situation. The bank held the deposits of a large proportion of the town's residents.

The role played by Hershey Park is another dramatic example of involvement of the business sector in evacuation response. The Hershey Park Arena is a subsidiary of Herco, the corporation that owns the Hershey Park complex and Hershey Chocolate Company. Shortly after 9:00 a.m. on Friday, March 30, the Derry Township police requested that the sports arena be designated an evacuation center. The manager was informed that as many as 14 000 persons might arrive. Preparations to receive evacuees were completed by 11:00 a.m. the same day. Cots and blankets were brought from nearby Indiantown Gap Military Reservation in the afternoon. A communications center and press room were set up. While the Red Cross administered the management and direct care of evacuees, the Hershey Park management attended to facilities and logistics. The fast and effective establishment of this evacuation center was due to the facility's design and the management's experience in servicing large crowds. Although initially as many as 14 000 evacuees were anticipated, the maximum number of people at Hershey Park at one time was about 180. A total of 800 people may have stayed there at some time during the emergency period. After it became known that nuclear insurers were making cash payments to those covered by the Governor's advisory, there was a substantial decrease in the number of evacuees at the arena.

When the possible evacuation area was extended to a 20-mile radius, arena management began developing a plan for evacuation of the center, completing the plan by Sunday morning. It was estimated that everyone could be moved within 15 to 30 minutes. Given 1 hour, it would have been possible to move the materials from the shelter, including food and equipment, in the tractor trailers standing by.

Income and employment losses within the region have been estimated from the Pennsylvania Department of Commerce studies as well as from the NRC telephone survey. Estimated lost employment for firms within 20 miles of TMI, from the State study, is 1.25 million person-hours for both evacuees and nonevacuees in the 1-week period following the accident. This estimate is reasonably consistent with the 1.13 million lost person-hours of evacuees living within 15 miles of TMI, as calculated from the NRC-TMI telephone survey. Therefore, approximately 8.5% of employment was lost during the week within 20 miles. This is about one-tenth of 1% of annual employment for the area. Employment loss does not necessarily lead either to income or to production loss. Some employees continued to be paid despite absence from work. The same industries' production can be sustained on a short run basis despite a reduction in work force. Also, compensatory increases in output through higher production rates possibly occurred in some firms after the evacuation period.

Both the State business firm and the NRC household telephone survey estimates of personal income losses are also consistent. The State studies indicate about $7.0 million in wages lost. The NRC study indicates that, within the 15-mile area, evacuees lost $3.9 million in wages and $2.8 million in nonwage income. Additionally, nonevacuees lost $2.6 million in wage and nonwage income. Total income loss estimated from the NRC-TMI survey is, therefore, $9.3 million. This represents about three-tenths of 1% of annual personal income in the area.

In the State surveys of manufacturing and nonmanufacturing firms, each was asked the value of production (or business) lost during the first week after the TMI accident. Manufacturing lost an estimated total of $7.7 million in gross output. This figure overestimates the real loss, however, because it includes the value of purchased inputs that were still available for use. A gross state product for Pennsylvania in 1977 of $11 per person-hour in manufacturing, compared with an estimated $41 loss per person-hour for the manufacturing survey, suggests overstatement of actual losses by a factor of 3 to 4. In addition, some percentage of lost production can be made up with little or no additional expenditure of resources. The extent to which there has been compensating output is unknown. Manufacturing firms lost business valued at $106.1 million during the first week after the accident. Much of this was due to lost sales to those who evacuated and those who postponed purchases in the atmosphere of uncertainty. Again, this is an estimate of gross business volume, and greatly overestimates the real economic loss. Most inventory was carried over for later sale, and some purchases were only delayed rather than completely foregone.

Following the accident there was concern that certain sectors of the local economy were particularly vulnerable to the effects of the accident. Farmers, processors, consumers, and industrial users of the area's agricultural products raised concerns about potential radiological contamination. A testing and monitoring program (principally of milk) initiated on Thursday, March 29, by the Pennsylvania Department of Agriculture uniformly failed to show levels of radiation that would be of any concern. Potential concentrations of iodine-131 in milk received the most attention. The highest reading found in any sample was 29 picocuries per liter, which is very low compared with the State's standard of 8300 pCi/L and the 12 000-pCi/L level at which the Food and Drug Administration becomes concerned about protecting the public's health. Local industrial concerns were careful to segregate, test, and monitor the use of locally produced milk, but there were several instances of canceled orders by out-of-State dairies for Pennsylvania milk. One large dairy serving Harrisburg reported an 18% decline in sales the first week and a 15% decline the second week after the accident. This dairy advertised that they did not use milk from farmers within 10 miles of TMI and had disposed of milk produced within this area.

Other fresh agricultural products were similarly affected. Noticeable effects on sales of agricultural products were largely limited to the week immediately following the accident and appeared to have been gone by the end of the week. A survey of full-time farmers within 25 miles of TMI conducted by the Pennsylvania Department of Agriculture showed the emergency period economic impacts on farmers not to be serious. Within 10 miles of TMI, 9% reported some loss; over the 25-mile area, only 4% reported any loss.

The accident did have an immediate impact on the tourist industry during April. Ten major lodging and convention centers surveyed reported losses of nearly $2 million in gross sales. These losses included the cancellation of a major trade show scheduled for the Pennsylvania Farm Show Building in Harrisburg, as well as cancellation of other conferences and individual reservations. Extrapolated to the total tourist industry, the loss may have been $4 to $5 million. Although there was a major interruption in the convention business for lodging and restaurant facilities, this was partially offset by the influx of transient workers connected with TMI.

Disruption to the local economy during the emergency period was generally moderate and short lived. A large part of the disruption that did occur is directly attributable to the loss of workers and customers who were evacuated. In monetary terms, the net loss of personal income was about three-tenths of 1% of the annual level. In terms of gross area product, this would be about 0.5% of the annual level.

c. Postemergency Period

There is little or no evidence of continuing direct negative effects of the accident on the economic base of the area surrounding TMI. A small proportion of manufacturing firms (9.8%) and of nonmanufacturing firms (4.1%) reported in the Department of Commerce study a perceived short term effect in their product. It is likely that these percentages would be now greatly diminished, as they were quite low when collected shortly after the accident. The Pennsylvania Department of Agriculture concluded in its study of impacts, reported in August, that it did not appear that there had been a permanent decrease in sales or a resistance to the buying of agricultural commodities produced or processed in the TMI vicinity. A travel industry-sponsored survey of potential travelers to Pennsylvania conducted April 26 through April 30 indicated only 2% of the respondents would avoid traveling to

Pennsylvania because of concern over the TMI accident. If there has been any continuing effect on tourism at all, it would be nearly impossible to separate from other more important adverse factors last summer, especially a polio outbreak in Lancaster County, gasoline shortages, and bad weekend weather.

Case activity of the Small Business Administration (SBA) and the Bureau of Employment Security (BES) supports the contention that there has been little or no measurable impact on the area's economy. SBA reported that only 15 firms, mostly retailers or service-related businesses, had applied for a total of $423 000 in loans. These loans were for short term impacts suffered immediately after the accident. As a comparison, 35 000 loans were made as a result of Hurricane Agnes in 1972 and 1500 loans were made as a result of Hurricane Eloise in 1975. BES case experience also supports the conclusion of no continuing economic dislocation. A total of 704 initial claims (95% made during the first week of April) were filed in the Harrisburg, Lancaster, Lebanon, and York offices for TMI-related reasons. Very few of the claims continued beyond the end of April.

Surveys of area residents identified a concern with adverse effects of TMI on local real estate values. Currently, evidence indicates that, if TMI has had an effect on real estate, it has not been substantial. The local real estate industry maintains that the market has not been affected, citing the continuing rise in sales prices. Monthly data on listings, sales, and settlements over the period 1977–1979 show a noticeable dip in April and a return to normal since then. The Pennsylvania Department of Community Affairs has completed data comparing certain characteristics of property sales within 5 miles of TMI relative to the same characteristics for the entire Central Penn Multilist area. Although the comparative analysis appears to show a slight effect of TMI within 5 miles, the movement, with one exception, is within the experience of the past 2½ years. The one exception is a considerable relative increase in the average number of days property was on the market. Because the data only extend through the second quarter of 1979 (April through June), nothing can be concluded about any basic changes in the local market.

The area's image as a place to work and live among possible recruits to the area does not appear to have suffered. A survey of the personnel directors of 11 large firms was made to determine if there was any resistance from out-of-region potential recruits or any unusual turnover among existing employees. Only in one instance was there thought to be any turnover (4 or 5 employees out of 1200) because of TMI. The personnel directors could not cite a single instance in which resistance to the area affected a potential job recruit. Of course, there is considerable opportunity to find a residence at some distance from TMI; therefore, this indicator would not support any conclusion about potential redistribution within the area.

There is little evidence of continuing economic effects of the accident. All sectors of the local economy appear to have quickly recovered from whatever disruption was experienced. If there has been any influence on the real estate market close to TMI, it has not yet become apparent.

6. EFFECTS ON LOCAL GOVERNMENT AND INSTITUTIONS

a. Background

The primary consequence of the accident on local government and institutions was the burden placed upon them to develop coordinated evacuation plans under great time pressure and in a climate of sparse and confusing information concerning the accident and its potential danger to the public. An additional burden was placed upon local authorities who were sought by local residents as a source of information and advice.

b. Emergency Period

Government operations at all levels were severely affected during the emergency period. Much of the attention of State, county, and municipal governments in south central Pennsylvania was directed to emergency management. At the same time, essential services had to be maintained. Nevertheless, State employees working in the Harrisburg area were exposed to the same evacuation pressures as the rest of the population. The State granted 21 938 hours of administrative leave at a cost of $161 257 from March 30 through April 9, 1979.

The complexity of local jurisdictional responsibilities was an important consideration in local emergency management. Dauphin County includes 20 municipalities (cities, boroughs, and townships) within 20 miles of TMI; York County includes 45 municipalities. Fewer municipalities were involved in Lancaster, Cumberland, Lebanon, and Perry Counties. Several authorities played a role in each municipality, although one individual usually serves as the civil defense liaison. The individual, who is proposed by the municipality and approved by the Governor, coordinates with the county director of civil defense. Borough mayors and township managers have responsibility for preserving order and protecting the public, including control of the police department. In the local government, mayors and managers are generally the focal point for the public. About half of the municipalities in the area have a police force; the others depend on the State Police for protection. Most of the area is serviced by volunteer fire departments and rescue squads whose territories do not necessarily correspond to municipal boundaries and who are independent from municipal control. Coordinated dispatch of emergency personnel (police, fire, and rescue) is achieved through county or subcounty communication centers. School buses, which are under the control of district school superintendents, played an important role in emergency planning. Consolidated school districts generally correspond to municipality lines but are not directly accountable to municipal authorities.

During the emergency period, local government officials experienced numerous problems and anxieties in fulfilling their responsibilities. Notification and communication concerning the emergency took place through prespecified civil defense channels. Some municipalities did not have an approved civil defense coordinator. In some communities having a coordinator, communications among officials within the municipality were not always ideal. At least one municipality (Royalton), which did not have a coordinator, apparently was never formally notified of the accident. Notification on Wednesday, March 28, concentrated on municipalities within 5 miles. No apparent effort was taken to advise municipal officials at greater distances. In the absence of a formal declaration of emergency by the Governor, the regular municipal authority charged with public safety (rather than the civil defense coordinator) remained legally in charge. Division of authority and responsibility was ambiguous. Although the civil defense coordinators had limited authority to take action and to make decisions, all the emergency preparedness measures from the county level were coordinated through them. Clarification of responsibilities and good working relationships—both within individual municipalities and among municipalities—appeared to be greatly influenced by personalities and the extent to which individuals had worked together in the past and were comfortable with each other. In some municipalities, all parties with any responsibility for public safety worked together in one location and made decisions jointly. Weak communications channels and problems with the timeliness and quality of information were no less a problem for local officials than for responsible parties at higher levels of government. Communications appear to have been particularly difficult for officials on the west side of the river. News briefings and briefings for public officials were held in Middletown, a considerable commute for west shore officials anxious about fulfilling their responsibilities in the emergency.

Much of the emergency planning burden placed on local governments was due to the expanding zone of possible evacuation. Early in the accident (before Friday, March 30) officials were operating within the framework of the 5-mile evacuation plan previously developed for TMI. The degree to which details were initially worked out appears to have varied considerably among communities. The population of the west shore within 5 miles of TMI was less than on the east shore where responsibilities were spread out and thus required a degree of coordination among authorities. Initial evacuation details included evacuation routes, staging points for mass transportation, procurement of school buses to transport those without other means, and evacuation centers. Officials within 5 miles reviewed their evacuation plans, and some made attempts to brief citizens on what actions they might be expected to take.

After being forced on Friday to revise 5-mile evacuation plans to 10 miles, county and local officials were told that night to extend their planning range to 20 miles. Within 3 days, evacuation planning changed from the direct concern of 3 counties and 11 municipalities with existing plans to the direct concern of 6 counties and about 90 municipalities, including the cities of Harrisburg, York, and Lebanon. An additional 26 counties were involved as host counties for the evacuees. Information was not collected to determine what demands, if any, were put upon municipal officials beyond 10 miles. Once revised evacuation plans were prepared (some communities had their individual plans and instructions to citizens completed on Sunday, April 2), firefighters in several communities distributed mimeographed instruction sheets or went door to door giving oral instructions. In other communities, instructions were given over loudspeakers.

The Pennsylvania Department of Community Affairs conducted a survey of TMI-related expenses incurred by county and local governments within 20 miles of TMI. Based on the response of 68 units, out-of-pocket costs were generally found to be modest. In the six municipalities nearest TMI, expenditures were less than $10 000 each. These dollar expenditures considerably underestimate the local resource commitment, however, because of local government dependence upon volunteers or labor for which compensation is not tied to hours worked. For example, the Londonderry Township Emergency Operations Center was staffed by 18 volunteers who worked a total of 510 hours without pay.

The public school system in the vicinity of TMI faced an especially trying situation. As early as Thursday, there was concern over the prospect of an evacuation and the procedures to be followed by the individual schools. It was assumed that normal emergency procedures would be followed, such as those for a snowstorm. At the time of the Governor's Friday morning press conference, in which he advised people to stay indoors, school districts within 5 miles of TMI were notified to have their schools shut down ventilating systems, shut windows, and allow only indoor recess. Procedures followed in the Middletown School District are probably representative of other districts. Bus drivers, crossing guards, and cafeteria staff were notified to stand by. Children were accounted for by checking absentee lists. Parents and friends began arriving to pick up children even before the Governor's 12:30 p.m. advisory for pregnant women and preschool children to evacuate. Varying degrees of hysteria were experienced in elementary schools. In smaller schools, principals were able to patrol the halls and reassure parents, teachers, and children. In larger schools, the anxiety level was apparently higher because of greater difficulty in handling the large number of parents, teachers, and students involved.

The Middletown School District followed normal emergency procedures; parents were notified of the schools' closings by local radio stations. Official dismissal began about 12:30 p.m., with buses following their normal routes and making three or four trips each. All children were gone by 1:30 p.m. School officials assumed that children would be cared for when arriving home, which was not always the case. Although there apparently was no consideration of leaving the children at school, most schools do have fallout shelters.

On the west shore, children from Newberry and Fishing Creek Elementary Schools were evacuated to another school more than 10 miles from TMI. Although this action ensured the safety of the children, it did create some panic among a few parents who had difficulty in locating their children.

Adequate and appropriate emergency planning must take into consideration the vulnerability of children, parents, and teachers to stress from uncertain danger. Given the characteristics of any specific radiological emergency, one or more protective actions available to the school may be effective. Also, if school buses are fully used to evacuate children, they will not be available for general evacuation service.

There are no inpatient medical facilities within 5 miles of TMI. Once evacuation planning extended to 20 miles, a number of hospitals were affected and had to prepare for possible evacuation. The Hershey Medical Center, the only hospital prepared to treat for radiological exposure, could be sealed and pressurized and was to continue in operation. In addition to identifying host general care hospitals, medical planners also had to identify special care needs and to line up capable host facilities and the requisite special transportation. On Friday and Saturday, hospitals generally began to reduce their number of patients by receiving only emergency cases and releasing those recuperating patients who could be sent home. Although there was some concern with hospital staff members evacuating with their families on Friday, the experience of at least one hospital (Holy Spirit Hospital) was that the staff absentee rate never exceeded 20% and that many evacuating staff members returned after ensuring their families' safety. Hospitals began to resume normal operations about Wednesday of the following weeks, and by Friday, April 6, most were back to normal.

Nursing homes and homes for the mentally retarded were also included in evacuation planning. Two nursing homes in Lower Swatara Township were actually evacuated because the administrators wanted to avoid the confusion of a forced evacuation and because they were short of staff. Several supervisors of homes for the mentally retarded evacuated their charges in anticipation of a forced evacuation.

c. Postemergency period

Continuing effects of the accident on local government, health services, and other institutions appear to be limited. Perhaps the two most noteworthy effects are the stronger emphasis on emergency plans and emergency management capabilities, and the interjection of TMI and nuclear power as sensitive local political issues. Many municipalities have put considerable effort into reviewing and revising evacuation plans. In such cases, need for additional volunteer personnel, training, and equipment has been identified. Even though much of the upgrading of plans and preparedness can be achieved with volunteer labor, additional communications and other types of equipment will require additional funding.

Since the accident, several new antinuclear groups have formed, and the anti-TMI movement has become a political force to be reckoned with. Membership of these groups appears to be substantial and broad based within the communities. Local elected officials are on notice that they will be opposed politically if they support the restart of TMI. Pressure has been exerted at the municipal level to pass resolutions in opposition to the reopening of TMI, and at least one municipality, Lower Swatara Township, has passed a resolution against its reopening. Opposition to reopen TMI is not universal, however; a number of local officials see the benefits of lower cost electricity from TMI as offsetting whatever risks are present. Local governing bodies have held meetings to allow citizens to air their views on TMI. Such strong pressure was put on the Middletown Borough Council that it passed a resolution in August 1979 opposing the restart of TMI-1. The Council's intent had been to defer passing a resolution until research findings from the State of Pennsylvania and the President's Commission were available.

7. EFFECTS ON AQUATIC BIOTA AND FISHERIES

a. Background

This account of the chemical and thermal effects of the TMI-2 accident on aquatic biota and fisheries in the surrounding area is based upon an NRC staff assessment.[10] Very high core coolant temperatures and releases of liquid industrial wastes into the Susquehanna River occurred during and following the accident. The study covers March 28, 1979 through July 1979, during which time Unit 1 was in a cold shutdown mode. Data used in the study were obtained from Met Ed, the Commonwealth of Pennsylvania, the U.S. Geological Survey, knowledgeable persons within State and Federal agencies, and from various other published studies of the aquatic biology of the Susquehanna River. Met Ed data were developed in an ongoing operational monitoring program required by the NRC and the EPA. In addition to these sources, the staff had available to it the inhouse environmental analysis for TMI-2 operating impacts completed in 1976.

b. Thermal and Chemical Discharges

The Commonwealth of Pennsylvania's water quality certification for TMI, under Section 401 of Public Law 92-500, contains five criteria about thermal discharges, limiting both absolute discharge temperatures and temperature differentials between discharge and ambient river temperature. During and following the accident, none of the thermal criteria was violated. In fact, the temperature differentials generally were smaller than during most of the month preceding the accident. Thermal discharges during and following the accident were also within the required limits of the NRC environmental technical specifications for TMI.

A number of chemical parameters have been well monitored at several points in the river around TMI. Monitoring of various chemical parameters is required under both the Commonwealth of Pennsylvania's national pollution discharge elimination system (NPDES) permit and the NRC environmental technical specifications (ETS) program. Data were also collected by the Pennsylvania Department of Environmental Resources. The volumes of industrial wastewater released during March and April were neither unusual nor significantly different from those released during normal operation. Releases thereafter were lower than normal. Total releases between March 28 and May 19, 1979 were 7 431 490 gallons. Apparently, toxic concentrations of nonradiological effluents were not released into the river, and violations of water quality limitations did not occur.

c. Consequences

Biological data collected through July confirmed the absence of any detectable effects of benthic invertebrates and fish. No unusual conditions of fish diseases or mortalities were noted in the river following the accident. Significant impacts from thermal and chemical discharges were not expected because they neither exceeded normal operations nor violated effluent limitations.

Recreational fishing in the river near TMI following the accident departed from historical trends. Fishing effort shifted away from TMI to other areas of the York Haven Pond. Not only did anglers fish less, but they also returned more of their catches than in previous years. Such alterations probably were related to the fishermen's knowledge of the occurrence of the accident and to their awareness of the liquid releases of industrial wastes to the river. By July the patterns of recreational fishing other than catch retained had returned to near normal.

8. LONGER TERM SOCIAL AND ECONOMIC EFFECTS

Immediate emergency period effects of the accident generally were accommodated by the populace. The accident does have the potential, however, to continue to affect their lives in various ways. Current replacement power costs with both TMI-1 and TMI-2 out of service are about $24 million monthly. If TMI-1 is restarted, the monthly replacement power costs for TMI-2 will be about $10 million. Customers in the Met Ed and the Penelec service areas are now paying more for electricity than they were before the accident. The actual effect on the cost of electricity is dependent on if and when each unit restarts. Price increases could be substantial, and given the energy intensity of industry in the area, long term net economic effects of these increases could be important. Local business interests are concerned about the implications for a spatial redistribution of growth in favor of utility service areas other than Met Ed and Penelec. A study of manufacturing and nonmanufacturing firms by the Pennsylvania Department of Commerce showed a strong influence of electricity price increases on intent of business to remain and to expand in the area. A hypothesized 10% increase in the price of electricity resulted in 22% of those manufacturing firms considering expansion indicating that they would not expand, and resulted in 30% reporting that their plans to remain in the area would be affected. Thirteen percent of nonmanufacturing firms reported they would not expand, and 33% reported that their plans to remain in the area would be affected by a 10% increase in the price of electricity. Sixty-two percent of nonmanufacturing firms reported that their plans to remain would be affected by a 25% increase in the price of electricity. Although it would be conjectural at this point to assume that expressions of intent would actually be carried out by all firms, it is clear that future movements in the price of electricity will weigh heavily in businesses' decisions to expand or to relocate.

Evidence to date indicates that, if the accident has affected the local real estate market, the effect has been minor. Further studies with more transaction information may, however, be better able to discern any effect. The aftermath of the accident—in terms of the public's confidence in the decisional process surrounding the cleanup of TMI-2 and safety conditions established for the possible restart and operation of both TMI-1 and TMI-2—will likely be the decisive factor in the future strength of real estate close to the plant. Local growth policy concerning development in the immediate vicinity of TMI will also have an important effect on residents. A question has been raised in at least one municipality (Newberry Township) as to whether it should encourage growth within 5 miles of the plant.

Despite the findings of the Behavioral Effects Task Group that the level of distress within the population significantly diminished since the accident, there is continuing strong concern and anxiety about the safety of future operation of TMI-1 and TMI-2. This concern has partially manifested itself in a dramatic increase in the number of local antinuclear groups and in their membership. Prior to the accident, opposition to TMI included the Three Mile Island Alert (a Harrisburg-based group) and the Environmental Coalition on Nuclear Power (a statewide organization), both of which have substantially increased their membership and operating funds since the accident. Three additional groups have formed in the immediate area: Persons Against Nuclear Energy (PANE), Concerned Citizens of Londonderry, and the Newberry Township Steering Committee. Two additional groups to the south, Anti-Nuclear Group Representing York and the Susquehanna Valley Alliance, are also concerned about the Peach Bottom Nuclear Powerplant. These groups are all committed to the permanent closure of TMI as a nuclear station. Their size, broad-based representation of the community, and commitment will ensure a high degree of publicity and controversy about the safety of TMI.

9. FINDINGS AND RECOMMENDATIONS

The direct social and economic effects of the TMI-2 accident were dramatic in terms of short term disruption but were mostly transitory. Lasting effects of the accident will be determined within the context of the issues of reopening TMI-2 and restarting TMI-1. Environmental effects of thermal and chemical effluents on aquatic biota and water quality were not detectable. During and following the accident, thermal and chemical effluents remained within NPDES limitations.

The accident's most significant effect on the people was the evacuation experience. In a climate of confusing and conflicting information, pressures to evacuate mounted within the population from the first day, Wednesday, March 28, 1979. By the time of the Governor's advisory at 12:30 p.m. on Friday, well over 14% of those who would evacuate had already done so. Within 15 miles, an estimated 144 000 people evacuated (39% of the population). The rate of evacuation decreased significantly beyond 15 miles, but instances of evacuation were found beyond 50 miles. Evacuation rates were higher closer to TMI; within 5 miles, 60% (21 000 persons) of the population evacuated. The absolute number of evacuees, however, was greater (67 000 persons) in the outer (10 to 15 mile) ring. Only 2% of the total households within 15 miles included individuals within the Governor's advisory (pregnant women or preschool children within 5 miles of TMI). A high percentage evacuated in anticipation of both worsening conditions at TMI-2 and a forced evacuation.

On the average, evacuees traveled a considerable distance, averaging 100 miles; were gone from home an average of 5 days; and spent an average of $300 extra. Many lost work and pay. The total cost to evacuees (after insurance payments) and to nonevacuees was more than $18 million. Insurance payments to evacuees were more than $1.2 million.

Another major result of the accident has been an increased level of psychological distress within the population. The level of concern and mental illness symptoms were high during the emergency period. Symptoms of mental illness within the population as a whole decreased after the accident, although many individuals, especially mothers of young children, still experience anxiety. The level of concern about the safety of TMI, although lower than during the accident, remains high. As a group, evacuees appear to have greater continuing concerns than nonevacuees.

Direct and immediate economic effects of the accident included interrupted local production and reduced local income and employment. Most losses occurred in the first week of April. Particularly vulnerable to the accident were the agricultural and tourism sectors of the economy. Each was significantly affected during the emergency period, but there are no noticeable continuing effects. No evidence of continuing disruption to economic activity exists. There is, however, concern within the business community that higher electricity prices and image problems due to TMI might have a negative effect on the continued growth and development of the area.

The institutional effects of the accident, which continue to be high, have primarily been in the form of increased concern with emergency planning and the local politicization of TMI. The poor quality of information available for use by officials in fulfilling their responsibility for the health and welfare of their constituency placed considerable stress on local political and emergency management officials during the period. Whereas these officials were in a position to be a reassuring influence on the populace, they did not have the necessary information. Since the accident, there has been continuing concern with emergency planning at the county and local levels. Perhaps the most significant longrun effect of the accident is the politicization at the local level about the future of TMI.

From the above findings, we make the following recommendations aimed at reducing the socioeconomic impacts resulting from any future accident at a nuclear powerplant.

1. Improved Public Education—The public in the vicinity of nuclear power reactors should be well informed about several aspects of reactor operation and malfunctions including the following:
   - general information about a reactor and how it works;
   - the kinds of accidents that might have offsite radiological consequences and their likelihood of occurrence;
   - potential health consequences of various types and levels of radiological release;
   - basic elements of the emergency preparedness plan for the reactor in question;
   - offsite protective action that may be required for a range of radiological release scenarios;
   - expected benefits and costs of protective action measures including reduction in radiological exposure and associated health implications and societal costs in terms of social and economic disruption.

   We believe a public well informed on those topics will be less prone to psychological distress and spontaneous evacuation.
2. Improved Information Flow—Timely, relevant, and understandable information about the status of an accident and likely offsite consequences must be available to State, county, and local decisionmakers responsible for recommending or implementing offsite protective action. This information should be adequate for State, county, and local emergency officials not only for making decisions on the need for specific protective actions but also for responding in a knowledgeable manner to questions from the general public. Also, the information flow to the public should be sufficient for them basically to reasonably understand the situation at the plant and the purpose and need for any protective actions that may be ordered or advised.

REFERENCES AND NOTES

[1]The President's Commission on the Accident at Three Mile Island, "Report of the Public's Right to Information Task Force" at 118.

[2]The President's Commission on the Accident at Three Mile Island, "Report of the Office of Chief Counsel on Emergency Response" at 33.

[3]The President's Commission on the Accident at Three Mile Island, "Report of the Public's Right to Information Task Force" at 296.

[4]*Id.* at 185.

[5]*Id.* at 186.

[6]Stanley D. Brunn, James H. Johnson, Jr., and Donald J. Zeigler, "Final Report on a Social Survey of Three Mile Island Area Residents," Department of Geography, Michigan State University, August 1979.

[7]C. B. Flynn, "Three Mile Island Telephone Survey: Preliminary Report on Procedures and Findings," NUREG/CR-1093, prepared for NRC by Mountain West Research, Inc., October 1979.

[8]The President's Commission on the Accident at Three Mile Island, "Technical Staff Analysis Report on Behavioral Effects."

[9]C. B. Flynn and J. A. Chalmers, "The Social and Economic Effects of the Accident at Three Mile Island: Findings to Date," NUREG/CR-1215, prepared for NRC by Mountain West Research, Inc., January 1980.

[10]C. R. Hickey, Jr. and R. B. Samworth, "Non-Radiological Consequences to the Aquatic Biota and Fisheries of the Susquehanna River from the 1979 Accident at Three Mile Island Nuclear Station," NUREG-0596, November 1979.

APPENDIX II.1
INTRODUCTION TO SEQUENCE OF EVENTS

The following sequence of events has been compiled from a number of sources; the NRC report NUREG-0600,[1] the EPRI report NSAC-1,[2] the Met Ed sequence of events,[3] various logs, plant computer output, the reactimeter, plant strip charts and operator interviews (especially those conducted by GPU on March 30, 1979). Times and events from other sequences have been checked insofar as possible from hard data sources such as the alarm and utility printers, reactimeter, and strip charts. An attempt has been made to reconcile discrepancies found in other published sequences. Where major discrepancies have been found they have been signaled in this sequence of events by the symbol (#).

Some events are not confirmed by hard data. In such cases, the times as given in the logs or by operator recollection have been used. It should be understood that these times may be subject to wide error. When events occur in rapid succession, the order of occurrence has been taken to be that given by the alarm printer. Because of the alarm sampling procedure, the order in which events are printed is not necessarily the order of occurrence.

Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
3/28/79	---	Plant status prior to accident: TMI-1 is shut down for refueling. TMI-2 is operating at between 97-98% of full power. The Integrated Control System (ICS) was in automatic. Pressurizer heater and spray controls were in manual. Feedwater pumps FW-P1A and FW-P1B, condensate pumps CO-P1A and CO-P1B and condensate booster pumps CO-P2A and CO-P2B were in operation. Makeup pump MU-P1B was in service. Operators were attempting to transfer spent resins from a condensate polisher to the resin regeneration tank. In this operation air at 100 psig and demineralized water at approximately 160 psig are used. Plant parameters as printed by the hourly log typer at 0300: RCS Pressures: Loop A = 2165 psig Loop B = 2148 psig Flow = 137 million lb/h Temperatures: Loop A TH = 606°F TC = 556-558°F Loop B TH = 606°F TC = 557°F Pressurizer level = 229 inches Makeup Tank at 77 inches Makeup Flow = 70 gpm Steam Generators Pressure: A = 908 psig B = 905 psig Temperature: A = 595°F B = 594°F Levels: A = 257 inches B = 264 inches Percent of full power = 97.928				1,2,3,10

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
1	3/28/79	-1 s (0400:36)	Condensate pump CO-P1A tripped.	Annunciator (panel 17) Status lights (panel 5) Alarm printer (operating without delay at this time)	Check valve in air line to condensate polisher was found to be frozen in the open position. This could have admitted water to the control air system. Condensate booster pumps CO-P2A and CO-P2B found tripped after turbine trip.	It has been postulated that the cause of the trip was closure of the polisher outlet-inlet valves because of water in the control air system. The polisher outlet and inlet valves were found to be closed after the turbine trip, but tests of similar valves have not substantiated this hypothesis.	1,2,3,4
2		0 s (0400:37)	Feedwater pumps FW-P1A and FW-P1B tripped.	Annunciator (panels 15 and 17) Pump discharge meter (panel 5) Alarm printer (delay--4 s)		Could have tripped on low suction pressure or trip of condensate booster pumps.	1,2,3,4
3		0 s	Turbine trip.	Annunciator (panels 5 and 17) Meters (panel 5) Status lights (panel 5) Alarm printer (delayed)		Normal following trip of feedwater pumps.	1,2,3,4,5,6
4		0 s	Emergency feedwater pumps EF-P2A, EF-P2B and EF-P1 came on.	Status lights (panel 4) Alarm printer (delayed)	Block valves EF-V12A and EF-V12B were closed.	Startup of emergency feedwater is automatic on loss of main feedwater pumps.	1,2,3,4
5		+1 s	Turbine throttle and governor valves closed.	Meters (panel 5) Alarm printer (delayed)		One throttle valve did not show closed.	1,4,11

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
6	3/28/79	3 s	RCS pressure reaches the setpoint of the pilot-operated relief valve (PORV) RC-R2. PORV opens. (Setpoint = 2255 psig)	Status light (panel 4)		Pressure in reactor coolant drain tank (RCDT) begins to increase.	1,2,3,6
7		8 s	Reactor trips on high pressure. (Setpoint = 2355 psig)	Annunciator (panel 8) Status light and meter (panel 14) Neutron flux meter (panel 4)	Reactimeter indicates peak pressure of 2346 psig. Wide range strip chart shows a peak of 2435 psig.	Reactimeter sampling rate may be too coarse to catch peak. The code safety valves may have lifted momentarily, if the higher indicated pressure is correct.	1,2,3,4,5,6
8		8 s	Pressurizer heater banks 1-5 tripped.	Status light (panel 4)		Pressurizer was evidentally switched from manual to automatic control.	1,2,3,4
9		9 s	Main steam pressure peaks at 1070 psig.	Meter (panel 4) Strip chart (panel 17)			1,2,3,6
10		9 s	Confirmed all rods inserted.	Status lights (panel 4) Alarm printer (delayed)			1,3,4
11		13 s	Let down secured. Operator attempts to start makeup pump MU-P1A.	Annunciator (panel 8) Status light (panel 3) Alarm printer (delayed) Letdown flow meter (panel 3)	Pump failed to start.	The switch for the makeup pump must be held in the start position for 2.5 s. Observation of status light would have shown that pump did not start. The purpose of these actions is to minimize pressurizer transient.	1,2,3,4
12		13 s	RCS pressure reaches setpoint for PORV closure (setpoint = 2205 psig).	Status light (panel 4)	Valve did not close.	Light "off" indicates solenoid deenergized. There is no actual position indicator.	1,2,3,6

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
13	3/28/79	13 s	Condensate hotwell low level alarm (21.72 inches).	Meter (panel 5) Alarm printer (delayed)			2,3,4
14		14 s	Pressurizer heater groups 1-5 returned.	Status light (panel 4) Alarm printer (delayed)		Automatically energized on decreasing pressure. Setpoints = 2105 psig for 1-3 and 2120 psig for 4-5, with pressure decreasing.	1,2,3,4
15		14 s	Emergency feedwater pumps reach full discharge pressure.	Meters (panel 4) Alarm printer (delayed)		Emergency feedwater valves EF-V11A and EF-V11B will not open until OTSGs reach 30 inches.	1,2,3,4
16		15 s	Pressurizer spray valve closed.	Status light (panel 4)			2,6
17		15 s	Pressurizer peaks at 255 inches.	Meter (panel 5) Strip chart (panel 4)		RCS parameters are normal.	1,3,6
18		28 s	OTSG A reaches 30 inches.	Meter (panel 4) Annunciator (panel 17)		Emergency feedwater valves EF-V11A and EF-V11B should begin to open. These valves apparently opened more slowly than usual; however, no flow was possible because the block valves were closed.	3,6
19		28 s	Condensate hotwell level returned to normal.	Meter (panel 5) Alarm printer (delayed)			3,4
20		30 s	High temperature alarms on outlet temperatures for PORV (239.2°F) and one code safety valve.	Strip chart (panel 10) Alarm printer (delayed)		Alarms were not considered abnormal, because the PORV had previously opened.	1,2,3,4
21		30 s	RCS pressure reaches . . pressure trip setpoint (1940 psig).		Reactimeter data.		1,2,4,6

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
22	3/28/79	40 s	Both OTSGs alarm low.	Annunciator (panel 17) Meter (panel 4) Alarm printer (delayed)			1,4
23		41 s	Start makeup pump MU-P1A. Open valve MU-V16B to increase makeup flow.	Annunciator (panel 8) Status light (panel 3) Alarm printer (delayed)		Pump was started by a second operator, who saw that the first at-attempt was unsuccessful. Pumps A and B are now both operating.	1,2,3,4
24		41 s	Open valve DH-5A.	Status lights (panel 8)		Allows makeup to be drawn from BWST.	1
25		48 s	Pressurizer level reaches minimum 158.5 inches and starts to increase.	Meter (panel 5) Strip chart (panel 4)		Minimum level is not as low as usual for this transient.	1,2,3,6
26		1 min	Code safety valve (RC-R1A) outlet temperature alarms high (204.5°F).	Strip chart (panel 10) Alarm printer (delayed ~ 1 min)		This does not necessarily indicate that the code safety valves lifted; opening of PORV would also cause increase in code safety valve outlet temperature.	1,2,3,4
27		1 min, 13 s	Condensate high level alarm.	Meter (panel 5) Alarm printer (delayed ~ 1 min)	Hotwell level reject valve was later found to be inoperative. Instrument air line to level controller was broken.	Condensate hotwell level control and other secondary side problems were constantly occurring, distracting operators' attention from the accident.	2,3,4
28		1 min, 18 s	OTSGs reach minimum level on start-up range instrumentation (A: 11 inches; B: 15 inches).	Steam pressure: Meter (panel 4) OTSG level: Meter (panel 4)		Indicates dryout. No feedwater was being admitted. Dry-out indicated by low steam pressure, low level, increasing RCS temperature. Operator verified EF-V11A and B opening.	1,2,3,6 Ref. 1 and Ref. 3 times are in error.

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
29	3/28/79	1 min, 26 s	Reactor coolant drain tank (RCDT) temperature reaches 85.5°F.	Meter (panel 8A) Alarm printer (delayed ~1 min)		RCDT temperature was gradually increasing as RCS coolant was released from the EMOV.	1,2,3,4 (Ref. 1 time is in error)
30		1 min, 30 s	RCS pressure reaches 1727 psig.	Meter (panel 4) Strip chart (panel 4) Alarm printer (delayed ~1 min)		RCS pressure was decreasing, pressurizer level was increasing, RCS temperature was increasing. Pressure normally trends in the same direction as level and temperature following feedwater transient and reactor trip.	1,4,6 (Ref. 1 time is in error)
31		2 min, 2 s	ESF actuation. Makeup pump MU-P1B trips. Makeup pump MU-P1C starts at 2 min 4 s. DH removal pumps 1A and 1B start.	ESF: Annunciator (panel 13) Status lights (panels 3 and 13) MU pumps: Annunciator (panel 8) Status lights (panel 3) DH pumps: Status lights (panels 3 and 13) Meters (panels 3 and 8) Alarm printer (delayed 2 min)		Actuation on low RCS pressure (setpoint = 1600 psig.) Makeup pumps A and 1C operating with valves MU-V16 wide open.	1,2,3,4,5
32		3 min, 13 s	RCDT relief valve opens (120 psig).	Pressure: Meter (panel 8A)	Reactimeter--not available to operators.		2,3
33		3 min, 13 s	ESF emergency injection bypassed by operator.	Operator action Alarm printer (delayed 3 min)		Bypass leaves all equipment operating, but generator now has control.	1,2,3,4
34		3 min, 26 s	RCDT high temperature alarm (127.2°F).	Meter (panel 8A) Alarm printer (delayed 3 min)	Reactimeter shows oscillations, possibly caused by RCDT safety valve lifting momentarily.	Further indication of open PORV.	3,4

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
35	3/28/79	4 min 30 s (approx.)	Operator throttles makeup valves (MU-V16) to reduce injection flow.	Flow meter (panel 8)		Purpose of throttling is (a) to reduce rate of rise of pressurizer level (b) to prevent pump damage as RCS pressure drops.	1,2,3
36		4 min, 38 s	Operator stops makeup pump MU-P1C.	Operator action Annunciator (panel 8) Status light (panel 3) Pressurizer level: Meter (panel 5) Strip chart (panel 4)		Makeup valves MU-V16C and MU-V16D were fully closed. Operator throttles valves MU-V16A and M-V16B in an attempt to control rising pressurizer level.	1,2,3,4
37		4 min, 52 s	Operator starts intermediate closed cooling pump IC-P-1A.	Annunciator (panel 8) Status lights (panels 8 and 13) Meters (panel 8) Alarm printer (delayed 5 min)		Operator is preparing to put a second letdown cooler in operation, so that letdown flow can be increased. He is attempting to recover control over the still-increasing pressurizer level.	1,2,3,4
38		4 min, 58 s	Letdown flow alarms high (Greater than 160 gpm).	Meter (panel 3)		Alarm printer is now so severely delayed that it is of little value to operators; alarm printer will not be listed as information available to operator from now on.	1,2,3,4 (Ref. 1 time is in error)
39		5 min, 0 s	Pressurizer level hits peak of 377 inches, momentarily decreases to 373 inches at 5 min 18 s, and then begins to increase again.	Meter (panel 5) Strip chart (panel 4)			1,2,3,6
40		5 min, 15 s	Start condensate pump CO-P1A.	Annunciator (panel 17) Status light (panel 5) Meter (panel 5)			2,3,4

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
41	3/28/79	5 min, 17 s	Attempt to start condensate booster pump CO-P2B (trips at 5 min 20 s).	Annunciator (panel 17) Status light (panel 5) Meter (panel 5)		Cause of trip is apparently low suction pressure. Auxiliary operator has reportedly realigned polisher correctly for restart.	1,2,3,4
42		5 min, 50 s	RCS pressure reaches minimum (~1350 psig), then begins to increase. Temperature reaches saturation.	Meters and strip charts (panel 4)		Reaching saturation temperature means that steam voids can form in system; steam is being formed rapidly enough to reverse pressure decline.	1,3,7
43		5 min, 51 s	Pressurizer level goes offscale high (greater than 400 inches).	Meter (panel 5) Strip chart (panel 4)			1,2,3,6
44		6 min	RCDT pressure begins erratic, rapid rise.	Meter (panel 8A)	Reactimeter data (not available to operators).	Indicative of two-phase flow through PORV.	1,2,3,6
45		6 min, 24 s	Condensate booster pump CO-P2B trips after another attempted start.	Annunciator (panel 17) Meter (panel 5) Status light (panel 5)			1,2,3,4
46		6 min, 54 s	Letdown cooler high temperature alarm (139°F).	Strip chart (panel 10)		This would have isolated letdown flow.	2,3,4
47		6 min, 58 s	Letdown flow decayed to 71 gpm.	Meter (panel 3)		Because of closure of MU-V 346.	1,2,3,4
48		7 min, 29 s	Reactor building sump pump WDL-P1A starts.			Pump outlet was believed aligned to the miscellaneous waste holdup tank; however, the latter tank's level does not show the appropriate changes. Pump was apparently aligned to the auxiliary building sump tank, which had a blown rupture disk.	1,2,3,4

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
49	3/28/79	8 min	Operator finds emergency feedwater block valves EF-V12A and EF-V12B shut and opens them.	Status lights (panel 4)		Clues to blocked feedwater flow: low OTSG level, low steam pressure, high emergency feedwater discharge pressure. Clues to initiation of flow: drop in discharge pressure, noise from loose parts monitor, increase in steam pressure.	1,2,3,6
50		8+min	RCS temperatures begin to decrease.	Strip charts (panel 4 and panel 10) Meter (panel 4)		Resumption of heat transfer through steam generators.	3,6
51		8 min, 58 s	Condensate pump CO-P1A trips again.	Annunciator (panel 17) Meter (panel 5) Status light (panel 5)		Another recurrence of secondary side problems, apparently unrelated to accident, but very troublesome to operators.	1,2,3,4
52		9 min, 7 s	Intermediate range NIs drop below scale, source range NIs energized.				1,4
53		9 min, 13 s	Condensate booster pump low suction pressure alarm (14.7 psig).			See remarks above.	2,3,4
54		9 min, 23 s	Letdown flow reestablished.				4
55		9 min, 30 s	Turbine bypass valves placed in manual.	Operator action		ICS was not responding adequately to increased steam pressure.	1
56		10 min	RCP high vibration alarm.	Annunciators (panel 8 and panel 10) Meter (panel 10)		Indication of voids in system. Apparently not recognized.	3
57		10 min, 15 s	Pressurizer level comes back on scale and drops rapidly.	Meter (panel 5) Strip chart (panel 4)			2,3,6
58		10 min, 19 s	Reactor building sump pump WDL-P2B starts.				1,2,3,4

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
59 60	3/28/79	10 min, 24 s to 11 min 43 s	Makeup pump MU-P1A was stopped, started, stopped, and restarted.	Annunciator (panel 8) Status light (panel 3) Meter (panel 3)		After final restart, pump 1A runs throttled for 3 h 23 min.	1,2,3,4
61		10 min, 48 s	Reactor building sump high level alarm (4.65 feet).		Alarm printer.	PORV discharge goes into RCDT, then out RCDT relief valve.	1,2,3,4
62		11 min, 24 s	Intermediate cooling water temperature from RCDT cooling is off scale (~225°F).		Alarm printer.		1,4
63		13 min	The operators are attempting to establish a 30-inch level in the OTSGs.	Meter (panel 4) Strip chart (panel 4) Strip chart (panel 5)		Operators may have throttled valves EF-V11A and B.	1
64		13 min, 13 s	Operators stopped decay heat removal pumps DH-P1A and B.	Operator action Status lights (panel 13 and panel 3) Meters (panel 8 and panel 13)			1,2,3,4
65		13 min, 27 s	Condensate booster pump suction header low pressure alarm clears.		Alarm printer.		2,3,4
66		14 min, 50 s	Reactor coolant pump alarms begin to be received on pumps 2A and 1B.	Annunci .ors (panel 8 and panel 10) Meter (panel 10)	Alarm printer.	Many rapidly alternating "norm/high" or "norm/low" alarms on pump speed, seal leak tank level, backstop oil flow, etc. These were probably caused by high vibration levels of the RCPs and all associated equipment, but might not be perceived as such. The great number of RCP related alarms was a major factor in the alarm printer getting so far behind time.	1,2,4

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
67	3/28/79	15 min, 27 s	RCDT rupture diaphragm bursts. RCDT pressure drops suddenly; reactor building pressure increases 1 psi.	Annunciator (panel 8A) Meter (panel 8A)	Reactimeter (not available to operators).	The drain tank is now completely open to the reactor building atmosphere, and will overflow.	1,2,3,6
68		15 min, 43 s	Condensate booster pump low discharge pressure alarm (307 psig).		Alarm printer.		2,3,4
69		16 min, 4 s	Restarted condensate pump CO-P1A.	Operator action Annunciator (panel 17) Meter (panel 5) Status light (panel 5)			2,3,4
70		16 min, 30 s	RCS becomes subcooled.	RCS Pressure: Meter and strip chart (panel 4) RCS Temperature: Meter and strip chart (panel 4) Strip chart (panel 10)	Calculations based on reactimeter temperature data and strip chart pressure data.	RCS has been near saturation or saturated for ~10 min. For 30 min to 1 h hereafter, the RCS remains either slightly subcooled or saturated.	1,6
71		16 min, 12 s	Condensate booster pump suction low pressure alarm.		Alarm printer.		2,3,4
72		19 min, 23 s	Reactor building air exhaust duct shows increased radiation level.	Meter (panel 12) Strip chart (panel 12)		Probably due to dislodged "crud." Possibly slight cracking of fuel clodding.	1,2,3
73		22 min, 17 s	Source range NI was higher than expected: the operator manually tripped the reactor.	Meter (panel 4)	Postaccident analysis indicates increase was due to voids in coolant in downcomer.	Operator was not aware of reason for increase.	1,2,4
74		22 min, 44 s	OTSG A low level alarm cleared.	Annunciator (panel 17)		OTSG B clears 4 min later.	1,4

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
75	3/28/79	24 min, 58 s	Operator requested PORV and code safety valve outlet temperatures. PORV outlet was 285.4°F, code safety outlets were 263.9°F and 275.1°F.	Utility printer	High temperatures, coupled with blown RCDT rupture disk and increasing reactor building pressure, give sufficient indication of PORV being open. Note, however, that RCDT parameters are displayed behind the control panels. RB pressure can be read on panel 3.	Operator believed high temperatures were due to (a) slow cooldown from the opening at 3 s, and (b) known leakage.	3,8
76		25 min	Letdown cooler high radiation alarms.		Analyses indicate that little significant fuel damage could have occurred at this time, although there is some possibility of clod rupture on high-powered fuel rods.	This is probably due to a "crud burst," which would not be unexpected in a transient event.	3
77		25 min, 44 s	Emergency feedwater low discharge pressure alarm received.			Operator has apparently shut steam driven emergency feedwater pump to slow the rate of rise of water in the steam generators.	1,2,4
78		26 min, 26 s to 27 min, 51 s	The operators request the computer print RCS temperatures, PORV outlet temperature, and pressurizer level.	Utility printer			1,2,8
79		26 min	Stopped steam driven emergency feedwater pump.				
80		28 min	Operator closes valves supplying emergency feedwater to OTSG B.	Operator action		Intend to slow rate of rise in level.	1
81		30 min, 21 s	Diesels manually shut down.			Auxiliary operator has been sent to diesels to shut them down locally. Diesels cannot now be started from control room.	1,4

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
82	3/28/79	31 min	Operator requests "Sequence of Events Review" from computer.	Utility printer			1,8
83		32 min, 36 s	Incore thermocouple R-10 alarms offscale high (T greater than 700°F).		Alarm printer not received by operator for nearly 30 min.		1,2,3,4
84		36 min, 8 s	Emergency feedwater pump EF-P2B stopped by operator.	Meters and status lights (panel 4)		Further actions to stop rate of rise of OTSG levels.	1,2,3,4
85		38 min, 10 s	Reactor building sump pumps turned off.		Approximately 8300 gallons have been pumped to auxiliary building.		1,2,3,4
86		40 min	Operator checked RCDT pressure and temperature.	Meters (panel 8A)			3
87		40 min	Increased source range count rate.	Meter and strip chart (panel 4)	RCS conditions again at or approaching saturation.	Possible voiding in core region.	2
88		44 min	Operator requests printout of pressurizer level indicator differential pressures.	Utility printer		Operator attempts to determine if level indication is correct. Conclusion: all instruments agree.	1,8
89		45 min	Letdown cooler count rate increased slowly over an order of magnitude. Peak ~ 2 x 10^4 cpm. (IC-R-1092)	Meter and strip chart (panel 12)		Increase and recovery are more indicative of crud burst than of fuel failure.	2,3,8
90		50 min	OTSG A level trending downward; OTSG B level trending upward.	Meters and strip charts (panel 4) Strip chart (O.R.) Channel 5			1,6
91		52 min	Operator requests computer print condenser pressure and emergency feedwater pump #1 discharge pressure.	Utility typer		EF-P1 has previously been shut down.	1,8
92		59 min	Condensate High Temperature Alarm.				

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
93	3/28/79	59 min	Polisher inlet and outlet valves were manually opened.	Report from personnel at polisher.		Still could not establish hotwell level control--broken air line to reject valve.	1
94		1 h	Plant status: All RCPs running, makeup pump MU-P1A operating. Feeding OTSG A directly. Feeding OTSG B via cross-connect. Hourly log typer has the following data for 0500: Reactor Coolant Flow = 103×10^6 lb/h Loop A: TH = 550°F TC = 546-547°F Pressure = 1061 psig Loop B: TH = 550°F TC = 547°F Pressure = 1041 psig Steam Pressure A = 1003 psig B = 1011 psig Steam Temperature A = 579°F B = 580°F Makeup Flow = 102 gpm The PORV is open (unknown to the operators). The RCS is near saturation, probably having extensive voids in the core. Coolant pump operation has become severely degraded, with reduced flow and high vibration. Difficulties with the condensate system have plagued the operators the past hour; the condition of the RCS appeared outwardly stable, i.e., pressure and temperature were not changing rapidly.		Calculated decay power = 32.8 MW	Note that steam temperature is actually higher than hot-leg temperature; this shows that cooling at this time is being provided by makeup water, which is being blown out through the PORV.	10,12

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
95	3/28/79	1 h, 1 min	The operator stopped circulation water pumps CW-P1B, C, D and E (opens atmospheric dump).	Operator action Meters and Status lights (panel 17)		This is done to permit use of the power operated emergency main steam dump valves (MS-V3A and MS-V3B) to control main steam pressure. The continuing deterioration of the condensate system has made an atmospheric dump necessary.	
96		1 h, [illegible] min	Alarms from 1 h 2 min to 1 h 13 min are on the utility printer.	Utility printer	Alarms are delayed about 1-1/2 h.	The action causing this change is taken at 2 h 39 min after initiation.	4,8
97		1 h, 11 min	Reactor building air cooling coil B emergency discharge alarm.	Flow meter (panel 25)		The fact that this alarm clears in 30 s indicates that it may be spurious.	1,3,4
98		1 h, 12 min	Operator requests alarm status of reactor coolant pumps and motors.	Utility printer	RCP flow is down 35% from normal.	All pumps show oil lift pump discharge pressure alarms; 1A, 2A, and 1B show full speed alarms; all pumps show backstop oil flow alarms; 2A shows seal leak tank level alarms. These alarms may not be intrinsically valid, but were probably caused by severe vibration conditions.	1,8
99		1 h, 13 min	Operator stops reactor coolant pump RC-P2B because of increasing vibration and decreasing flow and amperage. Pump RC-P1B is stopped a few seconds later.	Flow: meter and strip chart (panel 4) Vibration: Annunciators (panel 8 & 10) Amperage: Meter (panel 4) Status light (panel 4)	Coolant has now been clearly saturated again since about 1 h after initiation.	Loop B pumps were tripped in order to maintain pressure on pressurizer spray line from loop A. Steam pressure on B side began to drop, indicating stagnation.	1,2,3,6

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
100	3/28/79	1 h, 13 min	Alarms are lost for 1 h, 24 min. (#)			Ref. 1 has the time for this event at 1 h 2 min, with a duration of 2 h 49 min. These times are obviously wrong.	2,3,4,8
101		1 h, 15 min	Boron concentration measured at 700 ppm. Source range high (10^4 cps) and increasing.	B conc: sample results Source range: Meter and strip chart (panel 4)	Low boron concentration may have been due to dilution of liquid in sample line by condensed steam. High count rate may have been caused by voids in the downcomer.	Operators fear restart, do not realize that voids are forming in coolant. Checked reactor trip procedures.	1,2,7
102		1 h, 20 min	An increase in the letdown line radiation monitor was observed.	Strip chart (panel 12)		Increased steadily for the next 45 min, then stays offscale high.	1,9
103		1 h, 20 min	Operator gets printout of: Pressurizer surge line temperature (514°F) PORV outlet temperature (283°F) Code safety outlet temperature (211, 219°F) Pressurizer spray line temperature (497°F) Condensate pump outlet pressure (164 psi).	Utility printer		The operators now have adequate information to deduce that the PORV is open--(a) No reduction in outlet temperature, and (b) PORV outlet 70°F hotter than code safety outlets.	1,2,3,7
104		1 h, 27 min	OTSG B was isolated.	Operator action	Assumption of leak in steam generator cannot be supported by later information.	Operators assume that low steam pressure and high reactor building pressure are caused by steam leak.	1,3,5
105		1 h, 30 min	Intermediate range and source range neutron instrumentation both increase.	Strip chart (panel 4)		Ref. 2 postulates that increased voiding makes the downcomer annulus more transparent.	1,2,3,7
106		1 h, 30 min	Boron concentration down to 400-500 ppm in RCS; activity 4 μCi/ml (factor of 10 increase).	RCS sample analysis.	Analysis indicates gross fuel failure improbable at this time.	Increased activity could be the result of a crud burst; count level recovers.	1

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
107	3/28/79	1 h, 30 min	Secondary side steam flow from OTSG A increased.	Pressure: Meter (panel 4), strip chart (panel 17) Level: Meters (panel 4), strip charts (panels 4 & 5)	Inferred from pressure and level changes.	Indicates increased heat transfer in OTSG A. Temperature of reactor coolant in A loop has been trending upward slightly; apparently temperature difference across OTSG is now sufficient for OTSG to remove a significant amount of heat from RCS. Loop A cold-leg temperatures decrease next, because of increased heat loss.	2,6
108		1 h, 34 min	OTSG A boils dry again.	Meters (panel 4) Strip charts (panels 4 & 5)			1,2,3,6
109		1 h, 37 min	Intermediate range neutron instrumentation drops off scale; source range decreases suddenly by a factor of 30.	Meters and strip charts (panel 4)	Analysis (Ref. 2) indicates that separation of liquid and vapor probably occurred.	Flow in A loop may have now deteriorated to the point that vapor is no longer being circulated. Loop A flow is 30000 lb/h. Normal flow is 60000 lb/h. Flow is now dropping rapidly.	
110		1 h, 37 min	Operator increases flow to OTSG A in an effort to reestablish level.	Operator action Meters (panel 4) (EF-V11A)		Will raise level to ~50% on operating range, in an effort to establish natural circulation.	2,3

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
111	3/28/79	1 h, 41 min	Operator stopped both Loop A coolant pumps because of high vibration, decreasing and erratic flow.	Vibration: Annunciator (panel 8) Annunciator and Meter (panel 10) Flow: Meter (panel 4) Pump operation: Status lights and meters (panel 4)		The pump has been operating without adequate suction head. Further operation could cause severe damage.	1,2,3,6,7
112		1 h, 42 min	Samples taken from condenser vacuum pump.			Primary to secondary OTSG leak suspected.	1
113		1 h, 42 min	Source range count rate increased two decades. Intermediate range comes on scale and increases one decade. Operators commence emergency boration. (#)	Meters and strip charts (panel 4)	No data available to substantiate either of conflicting hypotheses. However, the fact that recorded source range and intermediate range (NI-3 and NI-7) follow each other closely lends credence to Ref. 2.	Increase has been postulated (Ref. 2) to be due to lower downcomer water level as the core dries out. This conclusion is disputed by Ref. 1--believes instrument error.	1,2,3,7
114		1 h, 43 min	Hot- and cold-leg temperatures begin to diverge. The cold-leg temperature drops and hot-leg temperature rises.	Meter and strip chart (panel 4) Strip chart (panel 10)		After temperatures exceed the narrow range indications, average temperature is the average of the narrow range limits, rather than the average of hot- and cold-leg temperatures.	1,2,3,6,7
115		1 h, 52 min	Trying to achieve 50% level on OTSG A.	Operator action Meters (panel 4) Strip charts (panels 4 & 5)			1
116		1 h, 54 min	Operator requests computer print Sequence of Events Review. (#)	Utility printer		Review disproves contention of Ref. 3 that makeup pump 1C has been started.	1,8

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
117	3/28/79	2 h	Plant Status: Makeup pump MU-P1A is operating, no reactor coolant pumps are operating, OTSG A is being dumped to the atmosphere because of a general breakdown of the condensate system. The hourly log typer shows the following data for 0600: Reactor coolant temperatures: Loop A: TH = 558°F TC (off scale) Loop B: TH = 528°F TC (off scale) Pressures: Loop A = 735 psig Loop B = 715 psig Makeup flow = 99 gpm Steam Pressures: A = 685 psig B = 190 psig Steam Temperatures: A = 536°F B = 532°F OTSG Levels: A = 154 inches B = 79 inches (OTSG B is isolated) PORV is still open. Boron sample = 400.		Calculated decay power = 25.7 MW		10,12
118		2 h, 5 min	OTSG A reaches 50% level. Throttled back EF-V11A.	Meters (panel 4) Strip charts (panels 4 & 5)			1,2,6,7
119		2 h, 11 min	Loop A hot-leg temperature offscale high.	Annunciator (panel 8) Strip charts (panels 4 & 10) Meter (panel 4)		TAVE will not be correctly shown.	3,6
120		2 h, 15 min	Reactor building air sample particulate radiation monitor goes offscale high. (HP-R-227)	Annunciator, meter, and strip chart (panel 12)		Possibility of gross fuel damage at this time.	2,3,7
121		2 h, 18 min	Operator requests computer printout of PORV and safety valve outlet temperatures. PORV-228.7°F, safety valves 189.5°F, 194.2°F.	Utility typer			1,2,3,8

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
122	3/28/79	2 h, 18 min	Operator closes PORV block valve RC-V2.	Status light (panel 4)			1,2,3
123		2 h, 18 min	Reactor building temperature and pressure immediately decrease.	Strip chart (panel 3)			1,2,3,7
124		2 h, 18 min	RCS pressure begins to increase.	Meter and strip chart (panel 4)		Because of block valve closure. No physical evidence of an increase in makeup flow.	1,2,3,7
125		2 h, 24 min	Reactor building air sample gas channel monitor increased and went off scale. (HP-P-227)	Annunciator, meter, and strip chart (panel 12)			3,7
126		2 h, 28 min	Loop B hot-leg temperature goes offscale high.	Annunciator (panel 8) Strip charts (panels 4 & 10) Meter (panel 4)		There is now clear evidence of superheating in the hot legs. This could have shown the operators that the core was beginning to become uncovered.	3,6
127		2 h, 30 min	Self-powered neutron detectors begin to go off scale.	Strip charts (panel 14)		Indicative of high core temperatures.	7
128		2 h, 34 min	Additional makeup pump started.			Unable to determine which pump. This event not substantiated.	2
129		2 h, 35 min	Operator begins feeding OTSG B to 50% level.	Meters (panel 4) Strip charts (panels 4 & 5)			1,2,3,6,7
130		2 h, 38 min	Letdown cooler A radiation monitor went offscale high. Numerous area radiation alarms received.	Annunciators, meters, and strip charts (panel 12)	Calculations suggest possibility of fuel damage at this time.	Letdown sampling secured due to high radiation.	1,3,7

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
131	3/28/79	2 h, 40 min	Emergency boration started.			Increasing levels of neutron instrumentation lead operators to fear restart.	1,3
132		2 h, 43 min	Reactor coolant sample taken. (140 μCi/ml)				1
133		2 h, 44 min	Incore instrumentation panel monitor goes offscale high.	Meter and strip chart (panel 12)			2,3
134		2 h, 44 min	Makeup pump stopped.			See event number 128 above. Not substantiated.	2
135		2 h, 45 min	See Remarks. (#)			Ref. 3 has makeup pump 1C tripped at this time. Ref. 3 had start at 1 h 54 min This contention is disputed by Ref. 1 (which erroneously states that Ref. 3 has no start time).	
136		2 h, 45 min	Numerous radiation alarms begin. (HP-R-225, HP-R-226, HP-R-222)	Panel 12		Radiation alarms are now indicative of extensive fuel damage.	3
137		2 h, 46 min	Unsuccessful attempt to start reactor coolant pump RC-P1A.	Status lights and meters (panel 4)			2,3
138		2 h, 47 min	Alarms back on alarm printer. Alarm printer brought up to date.	Operator action		Alarms for the period from 1 h 13 min to 2 h 47 min were irretrievably lost.	1,2,3,8

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
139	3/28/79	2 h, 48 min	90% of core T-Cs offscale high. Self-powered neutron detectors indicate readings.	Alarm printer		Readings on SPNDs caused by high temperatures. Flood of readings swamps alarm printer, causing it to lose time again. Core T-C reading not available to operators.	2,4
140		2 h, 48 min	Station vent monitor (HP-P-219) alarmed at 0.3 $\mu Ci/sec$--limit for I-131.				1
141		2 h, 52 min	Unsuccessful attempt to start RC-P2A.	Status light and meters (panel 4)		Does not appear on alarm printer.	1,2,3
142		2 h, 53 min	Control of hotwell level regained.	Alarm printer (delayed several minutes) Meter (panel 5)		Broken air line to reject valve was repaired.	1,2,3,4
143		2 h, 53 min	Unsuccessful attempt to start RC-P1B.	Status lights and meters (panel 4)			1,2,3
144		2 h, 54 min	Start RC-P2B.	Status lights and meters (panel 4)		Had to jump-start interlocks to start pump. Flow was shown momentarily and then dropped to near zero. The pump ran with high vibration.	1,2,3,4,6
145		2 h, 54 min	Pressurizer heater groups 1-5 tripped.	Status lights (panel 4)			1,2,3,4
146		2 h, 55 min	Pressurizer spray valve opens.			Operation of pressurizer spray is impossible without reactor coolant pump operation.	6

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
147	3/28/79	2 h, 55 min	7 incore T-Cs on scale.	Alarm printer (delayed 1/2 hour)	Slug of water from RC-P2B apparently gave some cooling.	These readings were back on scale indicating that they had just returned from an offscale condition.	1,3,4
148		2 h, 55 min	RCS pressure suddenly increased to 2140 psig.	Annunciator (panel 8) Meter and strip chart (panel 4)		Slug of water from cold leg gave rise to rapid boiling.	1,3,4
149		2 h, 55 min	HPI reset by increased pressure.	Annunciator (panel 13) Status lights (panels 3 & 13)		Setpoint 1845 psig.	2,3,4
150		2 h, 55 min	Source range and intermediate range neutron instrumentation dropped sharply.	Meters and strip charts (panel 12)		Slug of water filled downcomer, giving better shielding.	1,3,4
151		2 h, 56 min	Start circulating water pump CW-P1B, and CW-P1E. (Close atmospheric dump, resume steaming to condenser).	Meters and status lights (panel 17)		This allows control of main steam pressure by turbine bypass valves, and use of condenser.	1,2,3,4
152		2 h, 56 min	Site Emergency declared.			Reason: radiation alarms.	1,2,3
153		2 h, 59 min	Reactor building purge unit area monitor and fuel handling building area monitors increased. (HP-R-3236 and HP-R-3240) Fuel handling building air supply fans turned off.	Annunciators, meters, strip charts (panel 12)			

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
154	3/28/79	3 h	Plant Status: Makeup pump MV-P1A is operating, t. PORV block valve is closed, reactor coolant pump RC-P2B is operating but with very little flow. Both steam generators were being fed, with dump to condenser from steam generator A only. The hourly log typer gives the following information for 3 h 1 min: RCS Pressures: Loop A = 2055 psig Loop B = 2051 psig RCS Temperatures: Loop A: TH offscale high TC offscale low Loop B: TH offscale high TC offscale low Pressurizer level = 375 inches. MU Flow = 125 gpm Steam Pressures: A = 308 psig B = 416 psig Steam Temperatures: A = 495^{o}F B = 520^{o}F		Calculated decay power = 22.3 MW		10,12
155		3 h +	Pressurizer level offscale high.	Meter (panel 5) Strip chart (panel 4)			1,6,7
156		3 h, 2 min	RCS Loop B hot-leg temperature reaches 800^{o}F.	Strip chart (panel 4)		This is the limit on multipoint recorder.	1,7
157		3 h, 3 min	Hotwell low level alarm.				2,3,4
158		3 h, 4 min	Shut turbine bypass valves from steam generator B. Shut emergency feedwater valves to steam generator B.	Status lights (panels 4 & 5)		Condenser vacuum pump exhaust radiation monitor had increased. A leak from primary side was suspected. This completely isolates steam generator B.	1,2,3

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
159	3/28/79	3 h, 5 min	Source range and intermediate range detectors increasing.	Meters and strip charts (panel 4)		Indicates dropping water level.	1,3,7
160		3 h, 7 min	Condensate storage tank low level alarm.		Alarm printer delayed 50 min.		2,3,4
161		3 h. 10 min	Emergency feedwater pump (EF-P2A) was stopped.	Status lights and meters (panel 4) SG level: panels (panels 4 and 5)		Steam generator level above 50% on startup range.	1,2,3,4
162		3 h, 11 min	Condenser hotwell low level alarm cleared.	Meter (panel 5)			2,3,4
163		3 h, 12 min	Opened PORV block valve.	Status light (panel 4) Operator action	Inferred from pressure and temperatures. Time cannot be specified accurately.	Attempt to control RCS pressure. Outlet high temperature alarm, pressure spike in RCDT, drop in RCS pressure, increase in reactor building pressure.	1,2,3,4,7
164		3 h, 13 min	Pressurizer spray valve closes.				6
165		3 h, 13 min	Stopped reactor coolant pump RC-P2B.	Status lights, meters, and strip charts (panel 4)		Zero flow, low current, high vibration.	1,2,3,4
166		3 h, 14 min	Intermediate closed cooling pump area radiation monitor increased. (HP-R-207)	Annunciator, meter, strip chart (panel 12)			2,3,7
167		3 h, 20 min	ESF manually initiated. Makeup pump MU-P1C starts. Loop A hot-leg temperature drops.	Annunciator (panel 13) Status lights (panels 3 & 13)	Rapid quenching probably caused major fuel damage.	Reason for actuation was low RCS pressure. HPI gives water in core.	1,2,3,4
168		3 h, 20 min	Source range and intermediate range detectors drop suddenly.	Meters and strip charts (panel 4)		Indicates reflooding.	1,2,3,7

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
169	3/28/79	3 h, 21 min	Many radiation alarms received. (MU-R-720HI, MU-R-720LO, IC-R-1091, IC-R-1092, IC-R-1093, WDL-R-1311, DC-R-3399, DC-R-3400, NS-R-3401, SF-R-3402, HP-R-225, HP-R-226, HP-R-222) The control building (except the control room) was evacuated.	Annunciators, meters, strip charts (panel 12)		Indication of major core damage.	1,2,3,7
170		3 h, 24 min	General Emergency declared.			Based on high reading on HP-R-212.	1,2,3
171		3 h, 26 min	Pressurizer high level alarm clears.	Meter (panel 5) Strip chart (panel 4)			1,2,4
172		3 h, 27 min	ESF actuation reset.	Operator action			1,4
173		3 h, 30 min	BWST low level alarm at 53 feet.				1,4
174		3 h, 30 min	Shut PORV block valve.	Status light (panel 4)		Time of closing is very uncertain. May have been closed at 3 h 17 min. Definitely closed before 3 h 34 min.	1,2,3,4
175		3 h, 33 min	Pressurizer high level alarm level increasing rapidly.	Meter (panel 5) Strip chart (panel 4)			1,2,4
176		3 h, 35 min	Auxiliary building basement flooded. High radiation readings in many areas of auxiliary building.	Meters and strip charts (panel 12) Annunciators (panel 12)			1,7
177		3 h, 35 min	Start emergency feedwater pump EF-P2A.	Status lights and meters (panel 4)		OTSG A level had been falling.	1,2,3,4
178		3 h, 37 min	Makeup pump MU-P1C stopped.	Status lights and meters (panel 3) Annunciator (panel 8)		Apparently stopped to slow rate of rise in pressurizer level.	1,2,3,4

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
179	3/28/79	3 h, 41 min	Open PORV block valve. Outlet temperature alarms.	Status lights (panel 4)		Time is in doubt. Could have been opened earlier.	1,2,3,4
180		3 h, 45 min	Pressurizer spray valve opens.				6
181		3 h, 46 min	Sudden jump in source range detectors.	Meters and strip charts (panel 4)		May have been due to sudden steam flashing or change in core geometry.	1,2,7
182		3 h, 54 min	Reduced feed to OTSG A.	Operator action			1,6
183		3 h, 56 min	ES actuates on high reactor building pressure. Reactor building isolated.	Annunciator (panel 13)		This is the first time the reactor building has been isolated.	1,2,3,4
184		3 h, 56 min	Makeup pump MU-P1C starts.	Annunciator (panel 8) Strip charts and meters (panel 3)			1,2,3,4
185		3 h, 56 min	Close PORV block valve.	Status light (panel 4)		Inferred from reactor building pressure.	7
186		3 h, 56 min	Intermediate closed cooling pumps 1A and 1B tripped.	Annunciators, status lights, meters (panel 8)		By building isolation.	1,2,3,4
187		4 h	Plant status: makeup pumps MU-P1A and B operating. No reactor coolant pumps operating. The hourly log typer gives the following information: RC Pressures: Loop A = 1460 psig Loop B = 1453 psig RCS Temperatures: Off scale Pressurizer level 381 inches Steam Pressures A = 30 psig B = 358 psig (isolated) Steam Temperatures A = 468°F B = 499°F		Calculated decay heat = 20.3 MW	ES actuation and RB isolation have just occurred.	10,12
188		4 h	ESF and reactor building isolation defeated.	Annunciator (panel 13)			1,2,4

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
189	3/28/79	4 h	Started intermediate closed cooling pumps 1A and 1B.	Annunciators, status lights, meters (panel 8)		Necessary for letdown cooling.	1,2,3,4
190		4 h to 5 h, 30 min	Incore thermocouples being manually read.			This permits reading beyond range of 700°F. Range from readings was 80-2580°F.	1,2,3
191		4 h, 8 min	Start reactor coolant pump RC-P1A.	Status lights, meters, strip charts (panel 4)		Purpose of start was to observe current and flow. Started satisfactorily, but running current was low, and flow was zero.	1,2,3,4
192		4 h, 9 min	Stop reactor coolant pump RC-P1A.				1,2,3,4
193		4 h, 15 min	Open PORV.	Status light (panel 4)		Inferred from reactor building pressure.	7
194		4 h, 17 min	Stop makeup pumps MU-P1A and 1C.	Status lights and meters (panel 3) Annunciator (panel 8)		No makeup pumps now running.	1,2,3,4
195		4 h, 18 min	Attempt to restart MU-P1A.			Switch apparently then put in "pull-to-lock" position. MU-P1A will not now start on ESF actuation.	1,2,4,8
196		4 h, 19 min	ESF actuates on high building pressure. Decay heat pump DH-P1A starts. Intermediate cooling pump 1A trips. MU-P1A and C do not start.	Annunciator (panel 13)		One channel actuated, one channel defeated. 2/3 logic satisfied. Immediately bypassed.	1,2,3,4
197		4 h, 19 min	Cleared ESF actuation.	Annunciator (panel 13)			1,2,3,4
198		4 h, 19 min	Restart intermediate cooling pump 1A.	Annunciators, status lights, meters (panel 8)			1,2,3,4

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
199	3/28/79	4 h, 20 min	Close PORV.	Status light (panel 4)		Inferred from reactor building pressure.	7
200		4 h, 22 min	Pressurizer spray valve closes.				6
201		4 h, 22 min	Operator starts makeup pump MU-P1B.	Status lights and meters (panel 3) Annunciators (panel 8)			1,2,3,4
202		4 h, 24 min	Pressurizer heater groups 1-5 return to service.	Annunciator (panel 8) Status lights (panel 4)		All heaters now in service.	1,2,3,4
203		4 h, 26 min	Letdown cooler high temperature alarm.			Probably a late alarm when ESF was cleared.	1,2,4
204		4 h, 27 min	Start makeup pump MU-P1C.	Status lights and meters (panel 3) Annunciator (panel 8)			1,2,3,4
205		4 h, 31 min	Pressurizer heater group 10 trips.	Status lights (panel 4) Annunciator (panel 8)			1,2,3,4
206		4 h, 31 min	Stop condenser vacuum pumps. Broke condenser vacuum.	Status light (panel 17) Annunciator and strip chart (panel 17)		Condenser vacuum had been seriously degraded previously. Auxiliary boiler out of service.	1,2,3,4
207		4 h, 31 min	Opened main steam dump valve (MS-V3A).	Meter (panel 5)			2,3
208		4 h, 35 min	Incore thermocouple readings printed out.	Utility printer		Range from 310°F to off scale.	1,8
209		4 h, 36 min	Letdown high temperature alarm clears.				1,4

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
210	3/28/79	4 h, 36 min	Open PORV block valve.	Status light (panel 4)		Inferred from reactor building pressure.	7
211		4 h, 42 min	Emergency feedwater pump EF-P2A stopped.	Status lights and meters (panel 4)	Steam generator could only be operating in reflux mode. Heat removal capability is low.	Steam generator level had risen and would now remain up.	1,2,4
212		4 h, 44 min	Letdown cooler A radiation monitor went offscale low. (IC-R-1092)	Strip chart and meter (panel 12)		Apparently failed.	2,3,7
213		4 h, 46 min	Pressurizer heater groups 4-5 trip.	Annunciator (panel 8) Status lights (panel 4)		Did not come on again for rest of 3/28.	1,3,4
214		4 h, 47 min	Incore temperature readings again printed out.	Utility printer		Range from 378°F to off scale.	1,8
215		4 h, 59 min	Intermediate cooling pump area radiation monitors and reactor building emergency cooling monitors increase. (HP-R-207 and HP-R-204)	Strip chart (panel 12)			3,7
216		5 h	Plant status: No reactor coolant pumps running, makeup pump MU-P1B and 1C running, steaming through atmospheric dump valve, only reflux circulation, many radiation monitors off scale (containment dome monitor up to 6000 R/h), RCS pressure 1266-1296 psig, steam pressure (A) = 43 psig, temperature 454°F, hot leg superheated, PORV block valve open.		Calculations indicate that a large quantity of hydrogen was now in the RCS. Calculated decay power = 18.9 MW.		
217		5 h, 15 min	RCS pressures (1203, 1164, 1126 psig) and pressurizer surge line temperature (303°F) printed out.	Utility printer			1,8
218		5 h, 15 min	Decision made to repressurize system.	Operator action	Bubble contained hydrogen. There was no possibility of collapsing the bubble.	Believed hot legs contained steam bubble. Hoped repressurizing would collapse bubble.	1

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
219	3/28/79	5 h, 17 min	The alarm printer returned to service.	Alarm printer		Alarms had previously been on utility printer. Alarms are 1 h 26 min behind time.	3,4
220		5 h, 18 min	Closed PORV block valve.	Status light (panel 4)			1,2,3
221		5 h, 19 min	Decay heat pump DH-P1A stopped.	Status light (panel 3)		Switch placed in "pull-to-lock".	1,4
222		5 h, 24 min	ES actuates on building pressure. Pumps MU-P1A and DH-P1A do not come on. ES immediately defeated. Intermediate cooling pump 1A trips and is immediately restarted.	Annunciator (panel 13) RB Pressure: strip chart (panel 3)			1,2,4
223		5 h, 29 min	Diesels are placed in "MAINT EXERCISE" position.	Operator action		Diesels can now be started from the control room, but will not automatically start.	1
224		5 h, 31 min	Pressurizer heater group 3 trips.	Annunciator (panel 8) Status lights (panel 4)		Remains out of service.	1,3,4
225		5 h, 35 min	PORV outlet temperature alarms clear.			Evidence of closure at 5 h 18 min.	1,4
226		5 h, 35 min	Condensate storage tank low level alarm.				4
227		5 h, 43 min	System is repressurized. Pressure maintained by cycling PORV block valve.	Meter and strip chart (panel 4)		Intention is to hold pressure at about 2050 psig. Intermittent outlet temperature alarms and pressure fluctuations show block valve cycling.	1,2,3,4,7

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
228	3/28/79	5 h, 49 min	Control room intake radiation monitors (gas, particulate, iodine) all increased.	Strip charts (panel 12)		Nonessential personnel cleared from control room. Emergency control station moved to TMI-1.	2,3,7
229		5 h, 59 min	Auxiliary building exhaust fans stopped because of high radiation.	Strip charts (panel 12)			2,3
230		6 h	Plant status: RCS pressure being maintained between 2050-2200 psig by cycling PORV block valve. No reactor coolant pumps running, makeup pumps MU-P1B and 1C running, hot legs superheated. Atmospheric dump from OTSG A.		Calculated decay power = 17.8 MW		
231		6 h, 14 min	Raise OTSG level to 97%, using condensate pumps for feeding.	Meters and strip charts (panels 4 and 5)			1,2
232		6 h, 14 min	Pressurizer heater groups 1 and 2 trip, but are immediately returned.	Annunciator (panel 8) Status lights (panel 4)			2,3,4
233		6 h, 14 min	Auxiliary building fans restarted.	Status lights (panel 25)			3,4
234		6 h, 17 min	Control room personnel don respirators.				1,3
235		6 h, 18 min	Operator gets "Sequence of Events Review" from computer.	Utility typer			1,8
236		6 h, 23 min	Temperature on reactor building air cooling coils B emergency discharge goes off scale, then returns.			Indicative of severe temperature transient in reactor building.	1,4
237		6 h, 39 min	Start fuel handling building air exhaust fans.	Status light (panel 25)			3
238		6 h 54 min to 6 h 56 min	Steam generator A downcomer and shell temperatures printed by computer.	Utility printer			1,8

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
239	3/28/79	7 h	Plant status: No reactor coolant pumps operating, makeup pumps 1B and 1C operating. Maintaining RCS pressure between 2000 psig and 2100 psig by cycling EMOV block valve.		Calculated decay heat = 16.9 MW		10,12
240		7 h	Confirmed OTSG A not contaminated.	Measurement of steam plume.			1
241		7 h, 9 min	Started emergency feedwater pump EF-P2A to raise steam generator level higher.	SG level: Meters and strip charts (panels 4 and 5)			1,2,4
242		7 h, 30 min	Steam generator A filled.	Meters and strip charts (panels 4 and 5)			1,2,3
243		7 h, 30 min	Note: Natural circulation cannot be achieved by repressurizing. Reactor coolant pumps have proven to be inoperable. At this time it is planned to depressurize via the PORV, with the hope of getting the pressure low enough to inject core flood tank water.				
244		7 h, 30 min	Open PORV block valve and pressurize spray valve.	Status lights (panel 4)		On orders of station manager.	1,2,3,6
245		7 h, 42 min	Defeated ESF actuation.	Operator action Status light (panel 3)		ES would have been actuated 1 min later if it had not been defeated.	1,2,3,4
246		7 h, 44 min	Pressurizer heater groups 1 and 2 trip but immediately return.	Status lights (panel 4)			1,3,4
247		7 h, 44 min	Auxiliary building air exhaust fans stopped.	Status lights (panel 25)			3
248		7 h, 50 min	Pressurizer heater groups 1 and 2 trip.	Status lights (panel 4)		May be depressurizing via pressurizer vent now.	1,3,4

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
249	3/28/79	7 h, 53 min	Operator gets "Sequence of Events Review."	Utility printer			1,8
250		7 h 54 min to 8 h 19 min	Print out RCS and pressurizer pressures and temperatures.	Utility printer			1,8
251		7 h, 58 min	Pressurizer spray valve opens.				6
252		8 h	Plant status: No reactor coolant pumps operating. Makeup pumps MU-P1B and 1C operating. PORV block valve open. Pressurizer vent valve probably open. Depressurizing RCS hourly log typer gives the following data: RCS Pressure: Loop A = 1035 psig Loop B = 1038 psig RCS temperatures off scale. Pressurizer level = 395 inches Steam Generators: Pressures: A = 7 psig B = 320 psig Temperatures: A = 422°F B = 458°F Levels: A = 374 inches B = 228 inches Steaming through atmospheric dump valve from OTSG A.		Calculated decay heat = 16.2 MW		10,12
253		8 h, 1 min	Letdown cooler high temperature alarm.				1,4
254		8 h, 12 min	Core flood tank high level alarm. (13.32 feet).	Annunciator and meter (panel 8)		Possibility of check valve leakage.	1,3,4
255		8 h, 31 min	Start decay heat cooling pumps DH-P1A and 1B.	Status lights (panels 3 & 13)		Hoped to be able to go on decay heat removal system.	1,2,3,4
256		8 h, 40 min	Operator requests incore thermocouple readings.	Utility printer		Most are off scale.	1,8
257		8 h, 40 min	RCS pressure is down to 600 psig.	Meter and strip chart (panel 4)		Indicates RCS now floating on core flood tanks. Level of CR tanks decreased very little.	1,2,3,7

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
258	3/28/79	8 h, 43 min	BWST level down to 32 feet.	Meter (panel 8)			1
259		8 h, 55 min	Core flood tank high level alarm clears (13.13 feet).	Annunciator and meter (panel 8)		Very little water injected.	1,3,4
260		8 h, 58 min	Printout of RCS pressure and pressurizer temperature. Pressure: 483-526 psig, temperature 350°F.	Utility printer			1,8
261		9 h	Plant status: RCS has been depressurized. Makeup pumps MU-P1A and 1C operating. EMOV block valve open, vent valve may be open. Steaming through atmospheric dump valve. RCS Pressure: Loop A = 473 psig Loop B = 480 psig RCS temperatures off scale. Pressurizer level = 399 inches. Steam pressures: A = 13 psig B = 296 psig Steam temperatures: A = 413°F B = 449°F		Calculated decay heat = 15.6 MW		
262		9 h, 4 min	Stopped makeup pump MU-P1C.	Annunciator (panel 8) Status lights and meter (panel 3)			1,2,3,4
263		9 h, 7 min	Pressurizer spray valve closes.				6
264		9 h, 8 min	Stopped taking makeup from BWST.			Concerned that BWST would run out.	1
265		9 h, 15 min	Closed atmospheric dump valve. (#)	Meter (panel 5)		There is now no heat sink for the steam generators. Refs. 2 and 3 have this event at 8 h 30 min.	1,2,3

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
266	3/28/79	9 h, 15 min	Shut PORV block valve.	Status light (panel 4)		Cannot get RCS pressure low enough to go on decay heat removal system. The line at closure is not well fixed.	1,2,3,5
267		9 h, 17 min	PORV outlet high temperature alarm clears.			Indicates that PORV block valve was definitely closed by this time.	1,3,4
268		9 h, 20 min	Letdown cooler high temperature alarm clears.				1,4
269		9 h, 21 min	PORV outlet high temperature alarm. (EMOV block valve open-time not accurately known)			Block valve must have been reopened prior to this time.	1,4
270		9 h, 32 min	PORV outlet high temperature alarm clears (block valve closed-time not accurately known).			Valve must have been closed again.	1,4
271		9 h, 40 min	Start intermediate closed cooling pump IC-P1B.	Annunciators, status light, meters (panel 8)		This clears letdown alarm.	1,4
272		9 h, 49 min	PORV outlet high temperature alarm (block valve reopened).			Valve was opened again.	1,4
273		9 h, 50 min	Pressure and temperature in containment show sudden spike.	(RB Pressure: Strip chart (panel 3) RB Temperature: Strip chart (panel 25) Audible "thump"	Hydrogen combustion in containment.	Pressure spike was believed to be "electrical noise." Max. pressure 28 psig. Not ascribed to detonation at the time.	1,2,3,6,7

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
274	3/28/79	9 h, 50 min	ESF actuation on high/high building pressure (setpoint = 28 psig). Decay heat pumps DH-P1A and 1B start. Int. cooling pumps IC-P1A and 1B trip. Reactor building isolates. Reactor building sprays start. Makeup pump MU-P1C starts. Reactor building isolated.	ES: Annunciator and status lights (panel 13) Status light (panel 3) MU-P1C: Annunciator (panel 8), meters and status lights (panel 3) Spray: Status lights (panels 13 and 15) Meters panel 3)		Later "Sequence of Events" review showed only that 4 psi had been received on 4 channels. Log entry has "4 psi" apparently based on this print-out. Reactor coolant pump air temperatures alarmed high (cleared in 1 min).	1,2,3,4,5
275		9 h, 51 min	Stopped makeup pump MU-P1C.	Annunciator (panel 8) Status lights and meter (panel 3)			1,2,3,4
276		9 h, 51 min	480 v motor control centers 2-32A and 42A trip.				1,4
277		9 h, 55 min	Pressurizer heater group 8 trips.	Status light (panel 4)			1,2,3,4
278		9 h, 56 min	Reactor building spray pumps stopped.	Meters (panel 3) Status lights (panels 13 & 15) Annunciator (panel 8)	Sprays operated for 5 min 40 s.	Observed that pressure and temperature had been brought down.	1,2,3,4
279		9 h, 57 min	Stopped decay heat pumps, DH-P1A and 1B.	Status lights and meters (panel 3) Status lights (panel 13)		Cannot get pressure down far enough for decay heat system. Core Flood tanks remain floating, with intermittent changes of level.	1,2,3,4
280		10 h	Plant status: RCS pressure 512-522 psig, temperatures off scale, no RCPs running, makeup pump MU-P1B running, pressurizer shows 400 inches, no secondary heat sink.		Calculated decay power = 15.1 MW		

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
281	3/28/79	10 h	Opened PORV block valve.	Status light (panel 4)		Outlet temperature alarms high 1 min later.	2,3,4
282		10 h, 3 min	Operator gets "Sequence of Events" covering last ESF actuation.	Utility printer			1,8
283		10 h, 5 min	Pressurizer spray valve opens.				6
284		10 h, 6 min	Pressurizer heater groups 1 and 2 returned to service, but trip again less than 2 min later.	Status light (panel 4)			1,3,4
285		10 h, 28 min	RCS loop A outlet temperature comes back on scale. Goes to minimum of 548°F and stays on scale for 10 min.	Strip chart and meter (panel 4)		Ref. 1 postulates pressurizer dumped to loop. Operators may have believed they now had control of pressurizer level.	1,2,3,7
286		10 h, 32 min	Makeup pump MU-P1C started. RCS pressure had dropped to 440 psig.	Annunciator (panel 8) Meter and status light (panel 3) RC Press: Meter and strip chart (panel 4)			1,2,3,4
287		10 h, 33 min	Pressurizer heater groups 1 and 2 return to service.	Status light (panel 4)			1,3,4
288		10 h, 35 min	RCS pressure drops to 409 psig, then starts to rise again.	Meter and strip chart (panel 4)			2,7
289		10 h, 36 min	Makeup pump MU-P1C stopped.	Annunciator (panel 8) Meter and status light (panel 3)			1,3,4
290		10 h, 38 min	RCS loop A outlet temperature goes off scale again, then comes back on and continues to drop.	Strip chart and meter (panel 4)			3,6,7
291		10 h, 39 min	Pressurizer heater groups 1 and 2 tripped again.	Status light (panel 4)			3,4

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
292	3/28/79	10 h, 40 min	Auxiliary building sumps now full.	Observation			1
293		10 h, 44 min	Auxiliary building fans came on.	Strip chart (panel 25)		Ran for 30 min.	3,7
294		11 h	Plant status: Makeup pump MU-P1B operating. EMOV block valve open. RCS Pressure: Loop A = 415 psig Loop B = 421 psig RCS Temperatures: TH-A = 525°F All others off scale. Pressurizer level 378 inches. Steam pressures: A = 63 psig B = 266 psig Steam temperatures: A = 404 psig B = 431 psig OTSG levels: A = 371 inches B = 224 inches		Calculated decay heat = 14.6 MW		10,12
295		11 h, 6 min	Pressurizer decreased to 180 inches in next 18 min. RCS loop A temperature increases.	Annunciator (panel 8) Strip chart (panel 4) Meter (panel 5)			1,3,6,7
296		11 h, 10 min	Respirators removed in control room.	Operator action			1,3,5
297		11 h, 10 min	Shut PORV block valve.	Status light (panel 4)			1,2,3
298		11 h, 18 min	Start makeup pump MU-P1C pressurizer low level alarm.	Annunciator (panel 8) Meter and status light (panel 3) PZR level: Annunciator (panel 8) Strip chart (panel 4)			1,2,3,4
299		11 h, 27 min	Computer printout of PORV and pressurizer safety valve outlet temperatures.	Utility printer		EMOV outlet 191°F safety valves 171°F and 175°F.	1,8

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
300	3/28/79	11 h, 28 min	Stopped makeup pump MU-P1C.	Annunciator (panel 8) Meter and status light (panel 3) Pressurizer (panel 4)		Pressurizer level increasing.	1,2,3,4
301		11 h, 29 min	Pressurizer heater groups 1 and 2 tripped again.	Status light (panel 4)			1,3,4
302		11 h, 33 min	Start makeup pump MU-P1C.	Annunciator (panel 8) Meter and status light (panel 3)			1,2,3,4
303		11 h, 34 min	Start emergency feedwater pump EF-P2B.	SG level: Strip charts and meters (panels 4 and 5) EF-P2B: Status lights and meters (panel 4)		To raise level in OTSG B to 97% to 99% range.	1,2,3,4
304		11 h, 36 min	Stop makeup pump MU-P1C.	Annunciator (panel 8) Meter and status light (panel 3)		Pressurizer level continues to climb.	1,2,3,4
305		11 h, 44 min	Pressurizer low level alarm clears at 206 inches.	Annunciator (panel 8)			1,4
306		11 h, 52 min	Stopped emergency feedwater pump EF-P2B.	SG level: Strip charts and meter (panels 4 and 5)		Steam generator B at 97%.	1,2,4
307		11 h, 54 min	Pressurizer high level alarm at 260 inches.	Annunciator (panel 8)			1,4

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
308	3/28/79	12 h	Plant status: No reactor coolant pumps running. Makeup pump MU-P1B running. RCS pressure 560 psig and rising. Pressurizer level 294 inches and rising. RCS Temperatures: Loop A TH = $590^{\circ}F$ TC = $340^{\circ}F$ and rising Loop B TH = $620^{\circ}F$ TC = $180^{\circ}F$ OTSG B isolated and full. OTSG A without heat sink, pressure 44 psig and falling, and nearly full.		Calculated decay heat = 14 MW	There is no indication of natural circulation. Very little of the decay heat is being removed, except by makeup water and by occasional opening of PORV block valve. Gradual heatup of RCS is causing temperature and pressure to rise. Attempting to control pressure by juggling makeup and PORV block valves.	
309		12 h, 6 min	Pressurizer spray valve closes.				6
310		12 h, 11 min	Computer printout of selected incore thermocouples.	Utility printer		Almost all off scale.	1,8
311		12 h, 14 min	Auxiliary building exhaust fans restarted.	Strip chart (panel 25)			3
312		12 h, 22 min	Pressurizer level goes off scale.	Strip chart (panel 4) Meter (panel 5)			1,6
313		12 h, 30 min	Open PORV block valve.	Status light (panel 4)		Attempting to depressurize further. PORV outlet alarms 5 min later.	1,2,3,4
314		12 h, 40 min	Close PORV block valve.	Status light (panel 4)		Closing time in doubt; Ref. 2 has 12 h 46 min.	1,2
315		12 h, 48 min	Pressurizer level back on scale.	Strip chart (panel 4) Meter (panel 5)			3,6
316		12 h, 52 min	Open PORV block valve. (?)	Status light (panel 4)	RCDT temperatures suggest the block valve remains closed the rest of 3/28.	This is extremely doubtful. Safety valve alarms clear a few minutes later, which is inconsistent with opening.	2

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
317	3/28/79	13 h	Plant status: RCS at low pressure without secondary heat sink. Make-up pump MU-P1B operating. RCS Pressure: A = 613 psig B = 613 psig RCS Temperatures: THA = 522°F All others off scale. Pressurizer level = 379 inches Steam pressures: A = 95 psig B = 172 psig Status of PORV block valve not positively known; believed to be closed.		Calculated decay heat = 13.8 MW		10,12
318		13 h, 2 min	Start condenser vacuum pumps VA-P1A and 1C.	Status light (panel 17)	Condenser vacuum will be restored in a few minutes.	The auxiliary boiler has finally been returned to service and is now supplying turbine gland seal steam (this is a necessary prerequisite to using the condenser). Pump 1A trips, but is restarted 10 min later.	1,2,3,4
319		13 h, 20 min	Reactor building pressure starts to go negative. Pressurizer level starts to drop. RCS pressure 637 psig and falling. Pumping 425 gpm with two HPI pumps. It is now the the intention to repressurize, collapse bubbles (hopefully) and begin steaming from OTSG A.		High points were actually hydrogen filled. Collapse of loop bubbles was still impossible.	It is operators' belief that main condenser will soon be available.	1,5,7
320		13 h, 23 min	Start makeup pump MU-P1C.	Annunciator (panel 8) Meter and status lights (panel 3)			1,2,3,4
321		13 h, 25 min	PORV outlet temperature alarm clears (block valve closed-time not known).			Indicates valve is definitely closed.	1,4
322		13 h, 26 min	Pressurizer heater groups 1 and 2 trip.	Status light (panel 4)			1,2,3,4

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
323	3/28/79	13 h, 38 min	RCS pressure bottoms out at 611 psig and begins increasing.	Utility printer Meter and strip chart (panel 4)			1,5,8
324		13 h, 45 min	OTSG A now steaming.	Meter (panel 4) (Steam pressure)		Some difficulty earlier encountered with outlet valve--now cleared.	1,5,7
325		13 h, 52 min	OTSG A high level alarm clears at 81.3%.	Annunciator (panel 17) Meter (panel 4)		Indicates some heat transfer--steaming down.	1,4
326		13 h, 59 min	OTSG A high level alarm again.	Annunciator (panel 17) Meter (panel 4)		Indicates now feeding OTSG.	1,6
327		14 h	Plant status: RCS at low pressure; pressure increasing. EMOV block valve closed. Makeup pump MU-P1B operating. RCS Pressure: Loop A = 852 psig Loop B = 868 psig RCS Temperatures: THA = 549°F All others off scale. Pressurizer level 312 inches.		Calculated decay heat = 13.5 MW		10,12
328		14 h, 20 min	RCS pressure 1200 psig, BWST 23 feet.	Meter and strip chart (panel 4)			1,5
329		14 h, 25 min	Pressurizer heater groups 1 and 2 returned to service.	Status light (panel 4)			1,3,4
330		14 h, 39 min	Closing valve MU-V16B. MU flow now 120 gpm. RCS pressure 2080 psig.	MU-V16B: Status light (panel 3) Flow: Meter (panel 8) RCS P.: Meter and strip chart (panel 4)		RCS is now fully repressurized. Valve is throttled to reduce flow.	1,5,6
331		14 h, 41 min	Cutting back on valve MU-V16C. MU flow 105 gpm.	MU-V16C: Status light (panel 3) Flow: Meter (panel 8)			1,5

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
332	3/28/79	14 h, 43 min	Stop makeup pump MU-P1C. MU-V16C closed. RCS pressure 2275 psig.	MU-P1C: Annunciator (panel 8) Meter and status light (panel 3) RCS P.: Meter and strip chart (panel 4)			1,4,5
333		14 h, 47 min	Holding at 2300 psig. Operators have now decided to "bump" a reactor coolant pump.	Meter and strip chart (panel 4)		BWST level 22 feet. 80 gpm letdown flow . 32 gpm seal injection, 20 gpm makeup flow.	1,5
334		14 h, 48 min	Alarm printer fails. Not available until 15 h 10 min.				1,4
335		14 h, 59 min	Many radiation monitors come back on scale. (HP-R-3236, HP-R-232, HP-R-218, HP-R-3240, HP-R-215, HP-R-234)	Strip charts (panel 12)			2,3
336		15 h	Plant status: PORV block valve closed. MU-P1B operating. RCS pressures: A = 2285 psig B = 2304 psig Attempting to collapse bubbles.		Calculated decay heat = 13.2 MW		10,12
337		15 h, 10 min	Alarm printer back in service, but almost illegible.				1,4
338		15 h, 11 min	Computer prints out reactor coolant pump and makeup pump status on request.	Utility printer		Only MU-P1A now operating.	1,8
339		15 h, 15 min	Start DC reactor coolant pump oil lift pumps.	Auxiliary operator action		AC pumps not operable, due to power loss at motor control centers. Had to send personnel to auxiliary building to start. RCPs will not start without oil lift pump running.	1
340		15 h, 16 min	Full condenser vacuum reestablished.	Strip chart (panel 17)			1,5

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
341	3/28/79	15 h, 22 min	Start condensate booster pump CO-P2B.	Annunciator (panel 17) Meter and status light (panel 5)		To complete filling of OTSG B.	1,4
342		15 h, 32 min	Start makeup pump MU-P1C.	Annunciator (panel 18) Meter and status lights (panel 3)			1,2
343		15 h, 32 min	Stop condensate booster pump CO-P2B.				1,4
344		15 h, 33 min	Start reactor coolant pump RC-P1A. Ran for 10 s, then stopped.	Status light (panel 4) Meters: amperage, flow (panel 4)		Starting amperage normal, flow OK. RCS pressure and temperature immediately drop, then start to rise again. ESF actuates, but was bypassed.	1,2,3,4,5
345		15 h, 39 min	Stop makeup pump MU-P1C.	Annunciator (panel 8) Meter and status light (panel 3)			1,3,4
346		15 h, 49 min	Start makeup pump MU-P1C.				1,3,4
347		15 h, 50 min	Start reactor coolant pump RC-P1A.	Status light (panel 4) Meters and strip chart (panel 4)	Adequate core cooling now has been established.	Satisfactory operation.	1,2,3,4
348		15 h, 55 min	Stop decay heat pumps DH-P1A and 1B.	Status lights and meters (panel 3)			1,4
349		15 h, 56 min	Stop makeup pump MU-P1C.	Annunciator (panel 8) Status light and meter (panel 3)			1,3,4

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
350	3/28/79	16 h	Plant status: Reactor coolant pump RC-P1A is operating, makeup pump MU-P1B is running, the plant is now being well cooled with a heat sink to the condenser. Reactor coolant flow 28 million lb/h. Pressurizer level = 400 inches. RCS pressure 1310-1330 psig. Loop A: TH = 520°F TC = 286°F Loop B: TH = 520°F TC = 282°F Steam pressure (OTSG A) = 76 psig (OTSG B) = 99 psig Steam generator levels: A = 414 inches B = 393 inches.		A bubble of noncon-sensible gas had col-lected in the upper head of the reactor pressure vessel. Calculated decay power = 12.9 MW.	Steam generator B is isolated.	6,7
351		17 h, 25 min	Valve DH-V187 from the decay heat pumps to the RCS was opened.			Indicates intention to depressurize.	5
352		17 h, 29 min	Commenced transfer of material from auxiliary building neutralizer tank WDL-T8B to TMI-1. This tank had been filled before the accident, and was now being emptied to accept water from the auxiliary building sump.	Operator action			1,3
353		18 h, 18 min	Bubble reestablished in pressurizer.	Meter (panel 5) Strip chart (panel 4)		Went back off scale at 18 h 30 min.	1,5,6
354		18 h, 34 min	Letdown flow is lost.	Flow meter (panel 3)	Probably due to plugging with boric acid.		3

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
355	3/29/79	(All times after 0000 on Mar. 29 are given as time of day 0000.)	RCS pressure = 1026 psig RCS temperature = 240oF Pressurizer level = 362 inches Steam pressure (A) = 25 psig				
356		0020	Stopped transfer fom WDL-T8B to TMI-1.				1,5
357		0051	High pressure drop observed across letdown prefilters.	Alarm printer			5,4
358		0055	Secured auxiliary building and fuel handling building ventilation.				1,5
359		0210	Restarted auxiliary building and fuel handling building ventilation.				1,5
360		0211	The control room gas and particulate radiation monitors showed high levels. Control room personnel donned masks.	Strip charts (panel 12)			5
361		0300	Pressurizer level [illegible] RCS pressure dropping slowly. Loop A: TC = 2[illegible] Flow = 28 million lb/h Pressurizer temperature = 549oF RCS pressure = 1028 psig Pressurizer level = 400 inches OTSG A at 95%.				5,6
362		0315	Control room radiation monitors dropped and respirators were removed.	Strip charts (panel 12)			5
363		0400	Plant status: RCP-1A running. Loop A: TC = 234oF Loop B: TC = 233oF Flow = 28 million lb/h RCS pressure = 998 psig Pressurizer temperature = 547oF Pressurizer level = 394 inches.				5,6

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
364	3/29/79	0435	Vented makeup tank MU-T1 to vent header by opening MU-V13.				1,5
365		0443	Seal water high temperature alarm on RCP 2A. Requested seal water temperatures on all RCPs.	Alarm printer Utility printer		High temperatures on RC-P1B, 2A, 2B (all nonoperating).	4,5,8
366		0504	RCP seal water temperature alarms cleared.	Alarm printer			4
367		0510	RCS pressure = 969 psig TC B = 284°F Pressurizer temperature = 543°F Pressurizer level = 352.5 inches.				5
368		0615	RCS pressure = 945 psig TC B = 284°F Pressurizer temperature = 540°F Pressurizer level = 341 inches BWST = 20.5 feet.				5
369		0630	Sprayed down pressurizer. Level rose from 345 inches to 367 inches, pressure dropped 50 psi.	Level: Meter (panel 5) Strip chart (panel 4) Pressure: Meter (panel 4) Strip chart (panel 4)			5
370		0631	Letdown flow (~25 gpm) reestablished after raising intermediate cooling temperature.	Meter (panel 3)			5
371		0710	RCS pressure = 899 psig TC B = 283°F Pressurizer level = 352 inches.				5
372		0715	Pumped auxiliary building sump tank to auxiliary building neutralizer tank.				1,5

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
373	3/29/79	0716	Letdown flow shifted to RCBHT B.			When makeup tank was vented, auxiliary bldg. radiation increased.	1,5
374		0845	Commenced transfer from WDL-T8A to TMI-1.		Calculated decay heat at 1000 = 10.3 MW.	Preaccident water being transferred to make room in tank.	1,5
375		1215	Plastic sheet put down on auxiliary bldg. floor to reduce rate of release.				1
376		1240	Shut off turbine building, control bldg., control and service bldg. sump pumps.			High level in industrial waste treatment system. Overflowing and draining to settling pond. Leakage to river.	1
377		1315	Started industrial waste treatment system.			Discharges to river.	1
378		1410	Shut down industrial waste treatment system.		Xenon measurement was false.	Because of apparent Xenon release.	1
379		1458	Shifted letdown from RCBHT B to C.				5
380		1600	Pumped auxiliary building sump tank to WDL-T8A.			Will later pump sump to sump tank.	1,5,9
381		1610	Restarted industrial waste processing system.				1
382		1815	Stopped industrial waste treatment system.				1
383		1900	Washed down auxiliary building floor under the plastic.			After pumping sump to sump tank.	5
384		1920	Letdown flow was 20 gpm.	Meter (panel 3)			9
385		1945	Lined up M.U.T. degassing system through TMI-1 sample system to TMI-2 vent header.				9

IMAGE EVALUATION
TEST TARGET (MT-3)

6″

MICROCOPY RESOLUTION TEST CHART

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
386	3/29/79	2020	Started degassing M.U.T. via sample system.			Secured 10 min later.	1,5
387		2035	Opened MU-V13 for 5 s.			Vents makeup tank to vent gas system.	9
388		2036	Isolated nitrogen to waste gas header.			To keep pressure down.	9
389		2040	Significant increase in fuel handling building exhaust gas monitor.	Strip chart (panel 12)		From 300 mr/h to 1 r/h.	9
390		2045	MU-T1 vented to waste gas header.			Cautiously (to keep in waste gas header down). Reduce M.U.T. pressure to 55 psi.	1,5
391		2105	Secured venting MU-T1.				9
392		2114	LT2 pressurizer level indicator failed.	Alarm printer		Returned to service at 2230.	4,5,9
393		2200	Decided no leak in OTSG B.			Pressure steady at 25 psig. Level steady at 380 inches.	5
394		2330	Vented MU-T1 to waste gas vent header.			Cycling MU-V13 at 2 s periods.	5,9
395		2400	Now believed steam bubble in reactor vessel. TC A = 325°F RCS pressure = 1105 psig Pressurizer level = 325 inches.				5
396	3/30/79	0058	MU-T1 level decreasing, pressure increasing.				5,9
397		0130	Shut turbine bypass value for 5 min.			Temperature increased 8°F.	5,9
398		0150	Vented MU-T1 to waste gas decay tank WDG-T1B.			Secured at 0215.	1,5,9

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
399	3/30/79	0155	Secured transfer from aux. bldg. sump tank to neutralizer tank WDC-T8A.				1
400		0215	Shut off all sump pumps from turbine building and control building area.				1,5,9
401		0315	Pumped control building area sump to turbine building sump.			Using temporary pump.	1,5,9
402		0330	Vented MU-T1 to waste gas decay header.			Secured at 0350 Tank pressure: A = 50 psig, B = 80 psig.	1,5,9
403		0346	Cycling MU-V376.			To try to reestablish letdown.	5,9
404		0430	Started industrial waste discharge filter system, discharging to river from mechanical draft cooling tower blowdown line.			Sump level = 76%.	1,5,9
405		0435	Liquid pressure relief valve MU-R1 on MU-T1 opened, venting MU-T1 to reactor coolant bleed holdup tanks. MU-T1 level dropped to zero. Shut MU-V12. Seal flow dropped. Pressure in RCBHTs went off scale. Realigned make up to BWST.			Increase in gas discharge, coincident with venting.	1,5,6,9
406		0530	Flow to RCP seals adjusted to 7.2 gpm each, using needle values.				5,9
407		0710	Venting MU-T1 to vent header via MU-V13.				1,5,9
408		0750	Started waste transfer pump WDL-P5A pumping from RCBHT to MU-T1.		Calculated bubble at 0730--893 ft^3 (GPU).	Unsuccessful because of high pressure in MU-T1 (80-84 psi) stopped at 0753.	1,5,9
409		0753	Added 371 gal demineralized water to MU-T1 and boric acid from CA-T1.			So as not to draw from BWST. Finished at 0800.	5,9

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
410	3/30/79	0805	Secure seal water injection to non-operating pumps RC-P2A, 1B, 2B.			Log says 1A; o'vious error.	5,9
411		0815	Open MU-V12, shut DH-V5A, switch MU-P1A to MU-T1. Commenced adding water and boric acid to MU-T1.			Finished at 0820. Added 300 gal shift suction from BWST to MU-T1.	5,9
412		0855	Sent personnel to start hydrogen recombiner.				5
413		0900	Venting MU-T1.				5,9
414		0908	Shut DH-V5B.				5,9
415		0940	Shut off OTSG A.			To heat RCS to 280°F for 7 min.	5,9
416		1045	Closed MU-V17. Commenced bleeding letdown to RCBHT A. Began reducing pressurizer level to 100 inches.				
417		1120	RCS status: TC A = 280°F Pressure = 1043 psig. Pressurizer level = 390 inches. Pressurizer temperature = 560°F.				9
418		1220	Started transfer of miscellaneous waste tank to TMI-1.		Calculated bubble at 1240 (GPU) = 308 ft^3.		1,5
419		1405	Attempted to open WDG-V30B to vent WDG-T1B into reactor building.			Unsuccessful. Finally opened at 1442.	1,5,9
420		1410	Switched letdown from MU-T1 to RCBT A.			Switched back to MU-T1 at 1420.	9
421		1442	Venting waste gas decay tank B WDG-T1B to reactor building.			Stopped at 1450.	1,9
422		1502	Added 462 gal from RCBHT A to MU-T1.				5
423		1530	Fuel handling exhaust unit ARM and aux. bldg. access corridor ARM climb from 240 mr/h and 70 mr/h at 1145 to 700 and 160 mr/h.	Strip charts (panel 12)		Decline slowly to 100 and 35 mr/h on 4/1/79.	9

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
424	3/30/79	1600	RCS status: TCA = $280^{\circ}F$, pressure 1049 psig, pressurizer level 215 inches, pressurizer temperature $557^{\circ}F$, BWST level 15.5 feet.				9
425		1634	Turned off all pressurizer heaters to calculate rate of RCS pressure drop.	Status lights (panel 4) RCS pressure: Meter and strip chart (panel 4)			5,9
426		1650	Letdown temperature high. Opened valve MU-V376 to cool down.	Strip chart (panel 10)		Cleared at 1655.	9
427		1704	Started RCP-2A oil pump. K3 relay failed and pump tripped.			Ground fault.	5,9
428		1719	Added 200 gal from RCBT A to MU-T1.				5
429		1730	Lining up to pump from TMI-1 spent fuel tank to TMI-2 surge tank and then to TMI-2 BWST.				9
430		1810	Found and replaced blown fuse on RC-P2A control circuit.				
431		1850	Starting to refill BWST at 4000 gal/h.		Calc. bubble at 1907 (GPU) = 1806 ft^3.	CR log says being filled from Halliburton truck. BWST level 15.5 feet.	5,9
432		1920	Switched letdown to RCBHT C.			RCBT A filled.	5
433		1945	Isolated letdown from RCBHTs.				5,9
434		2036	Added 300 gal from RCBHT A to MU-T1.				9
435		2053	Shut off feedwater to OTSG A.			Steaming down.	9
436		2132	Venting pressurizer to RCDT.				
437		2200	Oil pumps on RCP-2A tested satisfactorily.			Until 2217.	5,9
438		2229	Transferring misc. waste tank to TMI-1.				5

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
439	3/30/79	2240	Restored feed to OTSG A. TCA 285°F, pressure 1029 psig, pressurizer level 215 inches, BWST 16.5 feet.				9
440		2310	Starting to vent pressurizer again.			Completed 0140, 3/31.	5,9
441		2330	"Gas bubble" noted for first time in C.R. log.			Volume given as 400 ft^3.	5
442		2347	Added 300 gal from RCBHT A to MU-T1. Pressure in MU-T1 at 43.5 psig.				5
443	3/31/79	0145	Venting RCS.				5,9
444		0205	Reactor building equipment hatch contact reading 60 r/h. WDG-T1A and 1B contact readings 40 r/h.				5,9
445		0315	Secured venting. Waiting for hydrogen recombiner to be placed in operation.				5,9
446		0325	Shift supervisor, shift foreman, and CROs reviewed Emergency Procedure for loss of RC-P1A.				5,9
447		0400	RCS status: TCA 282°F, pressure 1060 psig, pressurizer level 215 inches, pressurizer temperature 550°F, BWST level 18 feet.		Calculated decay power = 7.43 MW.		9
448		0423	Auxiliary operators instructed not to enter auxiliary building without a "teletector".				9
449		0546	Pressure in MU-T1 is 32 psig.				5
450		0518-0638	Taking hydrogen samples from reactor building.				5,9
451		0548	Turbine bypass valves from OTSG A closed from 47% open to 44% open to heat up RCS.				9

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
452	3/31/79	0735	Reduced pressure in RCS to 1025 psig using pressurizer spray. Level after spraying was 233 inches.				5,9
453		0753	Commenced venting pressurizer while heating and spraying simultaneously.			Secured at 0803.	9
454		0828	Venting pressurizer (same as 0753).			Secured at 0846.	5,9
455		0907	Venting pressurizer.			Secured at 0917.	5,9
456		0930	Sump and tank levels: MWHT, 7 feet; Aux. Bldg. sump 3.2 feet; auxiliary bldg. sump tank 3.4 feet; waste gas vent heater ~20 psig. Auxiliary sump tank lined up to MWHT.				5,9
457		0935	Venting pressurizer.			Secured at 0957.	5,9
458		0950	Drained spent fuel surge tank to TMI-2 BWST.		Calc. bubble (GPU) at 1032 = 860 ft^3.		5,9
459		1312	Venting pressurizer.			Secured at 1350.	
460		1344	Transferring water from TMI-1 spent fuel fuel pool to TMI-2 spent fuel surge tank with two sump pumps. Pumping (intermittently) to TMI-2 with SF-P1A.				5,9
461		1425	Venting pressurizer.			Secured at 1500.	5,9
462		1511	Halted transfer from TMI-1 spent fuel to TMI-2 BWST, until spent fuel refilled. BWST level 26.5 feet.				5,9
463		1537	Cracked pressurizer vent valve.			Closed at 1619.	5,9
464		1542	Secured turbine bldg. ventilation.				5,9
465		1656	Venting pressurizer.			Secured at 1737.	5,9
466		1741	Pressure in MU-T1 vent to zero. Closed MU-V13.				5,9
467		1815	Cracked pressurizer vent valve.			Closed at 1850.	5,9

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
468	3/31/79	1858	Opened MU-V13. MU-T1 pressure equalized with waste gas vent header. Discharge level increased.				9
469		1950	Cracked pressurizer vent valve.			Closed at 2034.	5,9
470		2110	Pressurizer venting: 2110-2139 2221-2352				5,9
471		2124	Transferring water from SF surge tank to BWST.		Calc. bubble (GPU) at 2245 = 894 ft^3. B+W = 487 ft^3.		5,9
472	4/1/79	0016	Pressurizer vented at same frequency throughout the day.				5,9
473		0029	Opened bypass valve on OTSG A slightly to compensate for higher RCS temperature.				9
474		0750	Stopped transfer of water to BWST. BWST level 41.5 feet.		Calc. bubble (GPU) at 0731 = 564 ft^3.		5
475		0930	Transferring MWHT WDL-T2 to TMI-1. Reactor building hydrogen concentration 2% throughout day.				5
476		1500	Reduced RCS pressure to 1000 psig.				9
477		2020	WDG-T1A and 1B is 86 psig.				9
478	4/2/79	1000	Lost auxiliary boiler for 2 min.				9
479		1347	Hydrogen recombiner in service.		Calc. bubble (GPU) at 1315 = 174 ft^3.		9
480	4/3/79	0906	Reduced steaming on OTSG A.		Calculated decay power = 5.4 MW.		9
481		0950	Slowly raised OTSG level to 97%.				9
482		1830	DC ground faults, RCP alarms.				9

Event Number	Date	Time after initiation	Event	Information available to operators	Postaccident calculations and data	Remarks	References
483	4/3/79	2400	Plant status: TC A = 281°F TH A = 281°F TC B = 281°F TH B = 278°F Pressure 1050 psig BWST level = 54 feet Reactor building pressure = 1.3 psig Reactor building temperature = 87.5°F.		Bubble gone.		

[1]NRC, Investigation into the March 28, 1979 Three Mile Island Accident by Office of Inspection and Enforcement, Investigative Report No. 50-320/79-10, NUREG-0600, Washington, DC, August 1979.

[2]Nuclear Safety Analysis Center, Analysis of the Three Mile Island--Unit 2 Accident, NSAC-1, Electric Power Research Institute, Palo Alto, CA, July 1979.

[3]Metropolitan Edison Co., Annotated Sequence of Events, Rev. 1, July 1979.

[4]Alarm Printer.

[5]Control Room Logs.

[6]Reactimeter data.

[7]Plant strip charts.

[8]Utility printer.

[9]Emergency Control Station (ECS) logs.

[10]Log typer.

[11]Operator Interviews.

[12]Letter from D. E. Bennett, II, Sandia Laboratories, to J. Murphy, NRC (PAS), dated April 24, 1979, with attachments.

APPENDIX II.2
CARBON PERFORMANCE WITH TIME

A. INTRODUCTION

Activated carbon is used in filtration systems to reduce the amount of radioiodine released to the environment. This carbon degrades in performance with time of service as contaminants build up on the carbon. Normal atmospheric contaminants have been shown to severely degrade the ability of the carbon to retain radioiodine.[1] The performance of carbon in four separate filter systems operated at TMI-2 after the initial stages of the accident (i.e., after mid-April) has been followed.[2] This appendix contains the available data for the auxiliary building, fuel handling building, supplementary auxiliary building, and condenser vacuum pump air exhaust filtration systems.

B. AUXILIARY AND FUEL HANDLING BUILDING VENTILATION FILTER SYSTEMS

The carbon in all four trains of these filter systems was replaced with new (unused) carbon in April-May 1979. All trays except for 79 trays in the auxiliary building B train were refilled with carbon coimpregnated with potassium iodide (KI or KI_3) and triethylenediamine (TEDA). These 79 trays were refilled with carbon impregnated only with KI_3.

Met Ed has been obtaining samples of the carbon after every 720 hours of filter system operation since the carbon was replaced. Difficulties in following the performance of the filter systems are attributed to the two types of impregnated carbons being obtained from two sources, and the possibility of the carbon samples not being representative[3] (see Section II.B.2.g). The available data on filter system efficiencies over time are given in App. Table II-1 for the auxiliary and fuel handling building filter trains. The test procedures used were in accordance with Regulatory Guide 1.52 (Revision 1), including a 16-hour preequilibration at the stated relative humidity. Each sample has been analyzed after successive 720-hour periods of exposure, with the exception of fuel handling building train B. For this train, the 3-month and 4-month samples each served only 2 months. The data lack consistency but generally indicate decreasing efficiencies with time. As a result, the filter banks were changed out and replaced with NUCON KITEG carbon (KI + TEDA impregnant) during late October-early November 1979. Additional data on moisture content, pH, and iodine activity for the replacement carbon is

APP. TABLE II-1. Filter system operation with time of service

	One Month (May)		Two Months (May-June)		Three Months (May-July)		Four Months (May-August)		Five Mos. (May-September)	
	95% R.H.[1]	30% R.H.	95% R.H.	30% R.H.	95% R.H.	30% R.H.	95% R.H.	30% R.H.	95% R.H.	30% R.H.
Aux. Bldg. Train A	97.9	99.5	88.8	99.9	94.1	99.9	92.2	99.7	93.6	99.9
Aux. Bldg. Train B	88.7	99.9+[2]	95.9	99.9+	96.5	99.9+	94.5	99.9+	90.0	99.9
Fuel Handling Bldg. Train A	98.7	99.9	93.3	99.9+	80.4	99.3	82.4	99.7	84.9	99.9
Fuel Handling Bldg. Train B	98.7	99.9+	91.2	99.9	86.0[3]	99.9+[3]	82.7[4]	99.4[4]	83.5	99.9

[1] R.H. means relative humidity.
[2] 99.9+ indicates the laboratory result reported was 99.9%, the upper limit of accuracy and detection.
[3] Experienced only 2 months service: June-July.
[4] Experienced only 2 months service: July-August.

presented in App. Table II-2. The general decrease in pH with increasing time of exposure indicates the degrading of the carbon by normal atmospheric contaminants,[4,5] and supports the decision to change the carbon in late October 1979.

C. SUPPLEMENTARY AUXILIARY BUILDING AIR FILTRATION UNITS

The four 30 000-cfm filter units installed on the roof of the auxiliary building provide a second filter system in series with the auxiliary and fuel handling building filter systems prior to release of ventilation exhaust to the environment. Before installation, the KI_3-impregnated carbon was certified to remove at least 96.3% methyl iodide and 99.9% elemental iodine when tested at 95% relative humidity and 212°F. Samples have been obtained monthly since May, but the samples are not representative (see Section II.B.2.g). The available test results for the used carbon in the supplementary auxiliary building filter systems are presented in App. Table II-3. Testing was performed for methyl iodide removal efficiency primarily at 95% relative humidity and 25°C and with 16 hours' preequilibration at the stated relative humidity. Filter trains 3 and 4 show the expected trend of decreasing carbon performance with time of exposure until the September samples. Carbon performance of trains 1 and 2, however, is not explainable because of the sampling problems. Because the samples did not contain sufficient carbon, only limited investigation of the physical properties of the carbon was possible. App. Table II-4 presents the available data. The pH values of approximately 7 indicate degraded carbon due to normal atmospheric contaminants (pH of unexposed carbon is approximately 9.5).[1,4] Although the carbon has not yet been replaced, this action is under discussion (November 1979).

D. CONDENSER VACUUM PUMP FILTER SYSTEM

The carbon in the first bed of the condenser vacuum pump filter system has been sampled monthly since May and has been tested in the laboratory for methyl iodide removal efficiency at 25°C and 95% relative humidity. The results are presented in App. Table II-5. Before installation, the KI_3-impregnated carbon was certified to remove 98.7% methyl iodide when tested in the laboratory at 130°C and 95% relative humidity. The carbon is not considered (November 1979) to be sufficiently degraded to require replacement.

APP. TABLE II-2. Physical properties of the replacement carbon with time of service

	One Month (May)			Two Months (May-June)			(Three Months (May-July)			Four Months (May-Aug.)		
	Moisture %	pH	Activity Ci/gm	Moisture %	pH	Activity Ci/gm	Moisture %	pH	Activity Ci/gm	Moisture %	pH	Activity Ci/gm
Aux. Bldg Train A	7.6	9.3	2.6(-6)	8.2	9.4	9.2(-11)	22.8	8.6	3.6(-10)	13.2	.9	1.6(-10)
Aux. Bldg. Train B	13.2	9.2	1.3(-5)	8.7	10.0	1.2(-11)	21.8	9.1	8.2(-11)	12.4	9.2	3.7(-11)
Fuel handling Bldg Train A	30.6	9.2	1.3(-7)	22.0	9.5	3.0(-11)	20.0	9.0	1.9(-10)	17.2	9.2	6.0(-11)
Fuel Handling Bldg. Train B	15.1	9.3	3.3(-10)	25.0	10.1	3.7(-11)	23.2*	6.8*	2.6(-10)*	18.0**	7.1*	4.1(-10)

*Experienced only 2 months service. June-July

**Experienced only 2 months service. July-August

APP. TABLE II-3. Supplementary auxiliary building filter system performance

	May		June		July		August**		September	
Days of Operation	CH_3I* Efficiency	Days of Oper.	CH_3I Efficiency	Days of Oper.	CH_3I Efficiency	Days of Oper.	CH_3I Efficiency	Days of Oper.	CH_3I Efficiency	
Train 1	29	No samples taken	29	70.9	32	88.0	***	93.8 (99.8)	***	94.1 (99.4)
Train 2	30		17	81.5	8	96.2	***	95.0 (99.9)	***	87.2 (99.9)
Train 3	24		29	96.3	32	95.7	***	90.7 (99.9)	***	96.8 (99.9)
Train 4	12		15	99.3	32	97.2	***	84.3 (99.9)	***	98.1 (99.9)

*Performed at 95% relative humidity and 25°C

**Efficiencies in parentheses were obtained at 30% relative humidity at 25°C

***Not available

APP. TABLE II-4. Physical properties of the supplementary auxiliary building filter carbon

	June	July	August		
	Activity Ci/gm	Activity Ci/gm	Moisture %	pH	Activity Ci/gm
Train 1	4.4(-12)	1.2(-10)	7.6	7.9	6.6(-11)
Train 2	5.0(-12)	1.5(-10)	6.8	7.5	3.3(-11)
Train 3	1.3(-12)	2.7(-10)	7.6	7.1	1.0(-10)
Train 4	2.1(-12)	1.9(-10)	6.4	6.8	6.1(-11)

APP. TABLE II-5. Condenser vacuum pump filter system

	May	June	July	August	September
CH_3 Efficiency at 25°C, 95% Relative Humidity	99.6	89.0	85.62	89.7	90.2
Activity on Carbon Ci/gm	4.9(-9)	1.3(-11)	2.3(-11)	2.6(-11)	Not Available

REFERENCES AND NOTES

[1]R. R. Bellamy and V. R. Deitz, "Confirmatory Research Program—Effects of Atmospheric Contaminants on Commercial Charcoals," in *Proceedings of the 15th DOE Nuclear Air Cleaning Conference*, DOE Conf-780819, August 7–10, 1978, Boston, Mass.

[2]Nuclear Consulting Services, Inc.., "Summarized Post Accident TMI Unit 2 HVAC Adsorber Systems Sample Data," NUCON 6MT611/13, October 1979.

[3]A representative sample of carbon is considered as a sample from the carbon bank that has experienced the same service conditions due to the air flow as the rest of the carbon bank.

[4]NRC, "Effects of Weathering on Impregnated Charcoal Performance," NRC Report NUREG/CR-0025, V. R. Deitz, Naval Research Lab, Washington, D.C., March 1978.

[5]NRC, "Effects of Weathering on Impregnated Charcoal performance," NRC Report NUREG/CR-0771RQ (also published as NRL Memorandum Report 4006), V. R. Deitz, Naval Research Lab, Washington, D.C., May 10, 1979.

APPENDIX II.3

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM FOR THREE MILE ISLAND STATION

The objectives of the TMI Station Radiological Environmental Monitoring Program (REMP) are to:

1. Comply with the radiological environmental requirements of the environmental technical specifications (ETS) for TMI-1 and TMI-2;
2. Determine whether any statistically significant increase occurs in the concentration of radionuclides in critical pathways;
3. Detect any buildup of long-lived radionuclides in the environment;
4. Detect any change in ambient gamma radiation levels; and
5. Verify that radioactive releases are within allowable limits and to evaluate any effects of the Three Mile Island Station on the health and safety of the public and the environment.[1]

To meet these stated objectives, an operational REMP was developed by a consultant for Met Ed.[1] Samples for the operational REMP were taken from the aquatic, atmospheric, and terrestrial environments. Samples of various media were selected to obtain data for evaluation of the radiation dose to man and important organisms. Sample types were based on (1) established critical pathways for the transfer of radionuclides through the environment to man, and (2) experience gained during the preoperational and initial operational phases. Sampling locations were determined from site meteorology, Susquehanna River hydrology, local demography, and land use.[1]

Sampling locations were divided into two classes—indicator and control. Indicator stations are those that are expected to show the effect of station operation; control samples are collected at locations that are believed to be unaffected by station operations. Indicator station data are also evaluated relative to background characteristics established prior to station operation. Audit samples beyond those required by the ETS may be collected and analyzed. The REMP sampling locations and requirements are shown in App. Table II-6.

APP. TABLE II-6. Radiological environmental monitoring program sampling locations [1]

Sample Medium	Location Code	Map No.[2]	Description[2]
AI, AP, ID	1S2[3]	2	0.4 mile N of site, North Weather Station
ID	2S2	3	0.7 mile NNE of site on light pole in middle of North Bridge
ID	4S2[3]	5	0.3 mile ENE of site on top of dike, East Fence
ID	5S2[3]	6	0.2 mile E of site on top of dike, East Fence
ID	9S2	8	0.4 mile S of site at South Beach of Three Mile Island
ID	11S1[3]	9	0.1 mile SW of site, west of Mechanical Draft Towers on dike
E	14S1	10	0.4 mile WNW of site at Shelly's Island picnic area
ID	16S1[3]	11	0.2 mile NNW of site at gate in fence on west side of Three Mile Island
AQP, AQS	1A2	12	0.7 mile N of site at north tip of Three Mile Island
ID	4A1	13	0.5 mile ENE of site on Laurel Rd., Met. Ed. pole #668-0L
AI, AP ID, RW	5A1[4]	14	0.4 mile E of site on north side of Observation Center Building
AQP, SW	9A2	15	0.5 mile S of site below Discharge Pipe
ID	16A1	17	0.4 mile NNW of site on Kohr Island
M	4B1	18	1.1 miles ENE of site, west of Gringrich Road
FPL, M	5B1	19	1.0 mile E of site on Peck Road
FBL, M	7B3	20	1.6 miles SE of site on east side of Conewago Creek
AQF, AQP, AWS, SW	9B1	21	1.5 miles S of site, above York Haven Dam
ID	10B1	23	1.1 miles SSW of site on south beach of Shelly's Island
AP, ID	12B1[4]	24	1.6 miles WSW of site adjacent to Fishing Creek
AQF	16B1	25	1.1 miles NNW of site below Fall Island
AI, AP, ID	1C1[4]	26	2.6 miles N of site at Middletown Substation
SW	1C3	27	2.3 miles N of site at Swatara Creek
AI, AP, ID, RW	8C1[4]	28	2.3 miles SSE of site
FPL, N	14C1	29	2.7 miles WNW of site near intersection of Routes 262 and 392
SW	8E1	30	4.1 miles SSE of site at Brunner Island
AI, AP, ID, RW	7F1	34	9 miles SE of site at Drager Farm off Engle's Tollgate Road
SW	15F1	35	8.7 miles NW of site at Steelton Municipal Water Works
FPL, M	2G1	36	2 miles NNE of Hershey on Rt. 39 Hummelstown
ID	4G1	37	10 miles ENE of site at Lawn - Met. Ed. Pole #J1813
ID, SW	7G1	38	15 miles SE of site at Columbia Water Treatment Plant
AP, ID	9G1	39	13 miles S of site in Met. Ed. York Load Dispatch Station
AI, AP, ID, RW	15G1	40	15 miles NW of site at West Fairview Substation
SW	8C2	43	2.3 miles SSE of site — York Haven Hydro
AQS	10A1	44	0.8 mile SSW of site
M	1B1	45	1.2 miles N of site — along Rt. 441

[1] All distances are measured from a point that is midway between the Reactor Buildings of Units One and Two.
[2] Refers to number on maps in Figure II-14.
[3] Also had RMC quality control TLD.
[4] Also had Commonwealth of Pennsylvania and DOE TLD's.

Sampling locations are shown in Figure II-14 and Color Plates I and II.

REMP samples are identified by a three-part code. The first two letters are the power station identification code, in this case, TM. The next one to three letters are for the media sampled.

AI = Air iodine	FPL = Green leafy vegetables
AP = Air particulates	ID = Immersion dose (TLD)
AQF = Fish	M = Milk
AQP = Aquatic plants	RW = Precipitation
AQS = Sediment	SW = Surface water
E = Soil	V = Fodder crops
FPF = Fruit	MG = Milk (goats)

The last four symbols are a location code based on direction and distance from the site. Of the last four symbols, the first two represent each of the 16 angular sectors of 22.5° centered about the reactor site (numbered in a clockwise direction from the north axis). The next symbol is a letter which represents the radial distance from the plant:

S = Onsite location	E = 4-5 miles off site
A = 0-1 mile off site	F = 5-10 miles off site
B = 1-2 miles off site	G = 10-20 miles off site
D = 3-4 miles off site	H = Further than 20 miles

The last symbol is the station numerical designation within each sector and zone. The location codes are shown in App.Table II-6.

Fish—Fish samples are collected at two locations each August and October and are separated into classes of bottom feeder versus predator-game species. Gamma spectrometry and ^{89}Sr and ^{90}Sr analyses are performed.[3]

Sediment—Three sediment samples are taken in both July and October and are analyzed for ^{89}Sr and ^{90}Sr and γ-emitting nuclides.[3]

Air Particulates—Air particulate samples are collected weekly at eight locations with low volume air samplers and are analyzed for gross β-activity. Monthly composites of all indicator and control samples are examined for γ-emitting nuclides. On a quarterly frequency, the air particulate samples are composited by individual stations and are analyzed for γ-emitting radionuclides, and then composited for gross **a**-particles and ^{90}Sr by indicator and control locations.[4]

Air Iodine—Gaseous iodine samples are collected on charcoal cartridges at four locations and are analyzed weekly for ^{131}I.[5]

Precipitation—Precipitation is collected at each sampling station. Monthly composite samples are analyzed for gross β-activity, and quarterly composite samples are analyzed for tritium and γ-emitting nuclides. Concentrations of ^{89}Sr and ^{90}Sr are determined in semiannual composites from each sampling station.[5]

Terrestrial Environment—The terrestrial environment is examined by analyzing samples of milk from six locations on a monthly basis and green leafy vegetables on an annual basis. Each sample is analyzed for ^{131}I and for other γ-emitting radionuclides. Quarterly composite samples are analyzed for ^{89}Sr and ^{90}Sr. Green leafy vegetables (cabbage) are collected in July and August from five stations and are analyzed for γ-emitting radionuclides.[5]

Direct Radiation—The ambient radiation levels in the area of Three Mile Island Station are determined with energy compensated calcium sulfate TLDs. Twenty TLD packets of four TLD sections each are placed quarterly at 20 locations. In addition, prior to the accident, RMC TLDs were placed at 10 of the REMP locations for quality control (see App. Table II-6). The Commonwealth of Pennsylvania had TLDs at four of the REMP locations (see App. Table II-6).[6]

REFERENCES AND NOTES

[1]Met Ed Radiological Environmental Monitoring Report, 1978 Annual Report, Three Mile Island Nuclear Station, IWL-5590-443, Teledyne Isotopes, Westwood, at 4.

[2]*Id.* at 27.

[3]*Id.* at 13.

[4]*Id.* at 14.

[5]*Id.* at 16.

[6]*Id.* at 17.

APPENDIX II. 4

OFFSITE RADIOLOGICAL MONITORING ACTIVITIES OF DOE ORGANIZATION

This Appendix describes the contribution of units within the DOE force which increased the offsite radiological monitoring effort after the TMI-2 accident.

Brookhaven National Laboratory (BNL)— A seven-person radiological assistance team from BNL arrived at the Capital City Airport via Coast Guard helicopter on the afternoon of March 28, and immediately began to collect air, soil, and vegetation samples and to make field measurements for external radiation. After coordination with the Pennsylvania Department of Environmental Resources (PDER), BNL dispatched two teams to the Three Mile Island site boundary.

The teams worked in an area in the downwind direction. External exposure rate measurements, gamma scans with an NaI(Tl) portable analyzer, TEDA-impregnated charcoal air samples, silver-loaded silica gel air samples, and water, soil, and vegetation samples were obtained at several locations within a 15-mile radius of the plant. Results of the radiation survey were reported at or near the time of measurement and samples were turned over to PDER at midnight on March 28, 1979.

A member of the BNL team served as DOE representative and remained in Harrisburg to keep PDER officials informed of BNL field measurement results throughout the night. A second BNL team consisting of five people arrived in Harrisburg on the morning of March 29 to supplement the first team's effort. The BNL team continued to perform offsite monitoring until March 30.[1]

Aerial Measurement System (AMS)–Nuclear Emergency Search Team (NEST)—The DOE AMS–NEST unit stationed at Andrews Air Force Base sent a helicopter equipped with sensitive gamma-radiation detectors and an onboard computer for data acquisition to Three Mile Island. This system was too sensitive to operate within the release plume and a second helicopter was sent. This helicopter was equipped with a greater variety of radiation detection equipment, including hand-held gamma detectors with sensitivities ranging from μR/h to mR/h, and an NaI(Tl) detector with a multichannel analyzer for spectral measurements. In addition, an air sampler provided by BNL was also used to specifically monitor airborne iodine.

From the period March 28 through April 15, the AMS-NEST unit made 72 sorties, with 17 of these on March 31 and April 1, that provided both routine and specially requested aerial radiological measurements.[2]

Bettis and Knolls Atomic Power Laboratories (BAPL and KAPL) with Pittsburgh and Schenectady Naval Reactors Offices— The Pittsburgh Naval Reactors Office (PNR)-BAPL team arrived at the DOE command post at the Capital City Airport on the afternoon of March 29, and deployed field radiation survey meters, air samplers, environmental media collection materials, Ge(Li) detectors, Na(Tl) detectors, γ-spectrum analyzers, and self-sufficient power generators to perform specific γ-emitting isotopic analyses in the field.[3]

The low level radiochemistry laboratory was operational that evening. Four environmental monitoring and sample collection teams were dispatched with State representatives and airport security personnel as guides. They remained in the field, 1 mile north to 8 miles south of Three Mile Island Station, in regions most recently covered by the plume, until early on the morning of March 30. Field measurements for external radiation were made and samples of air, soil, water, and vegetation were collected. Samples collected the previous day and that evening were analyzed for evidence of radioiodine deposition and the results were provided verbally to PDER the morning of March 30.[3]

PNR–Bettis personnel and equipment were supplemented and replaced in part by KAPL and Schenectady Naval Reactors Office (SNR) resources in subsequent days.

Argonne National Laboratory (ANL)— An initial response team of four ANL and DOE personnel arrived at the Capital City Airport command center on the evening of March 30. This initial ANL response team was supplemented the morning of March 31 with a mobile van from ANL equipped to support field measurements of direct radiation and radioactivity in air, water, vegetation, and soil. The ANL team was assigned to assist the NRC field command post in performing terrestrial monitoring in areas covered by the plume. Results of direct radiation measurements and air samples as well as some soil, water, and vegetation samples were provided to the NRC field command post as available on a 24-hour basis.[4]

Oak Ridge National Laboratory (ORNL)—The ORNL team (a DOE representative and five ORNL health physics personnel) arrived at the DOE command center on the evening of March 30. The ORNL team brought two multichannel analyzers, eight GM survey instruments, air sampling equipment (pump and charcoal cartridges), and miscellaneous support gear. The ORNL team began operations the afternoon of March 31, and conducted radiological environmental monitoring in the offsite area.[5]

Mound Laboratory—A team of six from the Mound Laboratory reported to the DOE command center on the morning of March 31 and was assigned to the DOE Region I Radiological Assistance Team. The Mound team assisted the FDA and BRH on April 1, 1979 by placing TLDs in the environment 10 to 20 miles from Three Mile Island Station and by recording the TLD locations for FDA and BRH.

DOE Environmental Measurements Laboratory (EML)—In response to the accident, EML deployed several continuous exposure rate monitors and performed field gamma-ray spectrometry in the vicinity of the plant. Field measurements of deposition and radionuclide identification were begun by EML on the morning of April 2 and were performed with a 130 cm^3 Ge(Li) gamma-ray detector. The energy range for this spectrometer was set for 50 keV–4 MeV, and analysis was performed using a 4000-channel analyzer and a programmable calculator. Exposure rate measurements were performed with high pressure ionization chambers (HPIC).[6] Based on estimates of possible maximum deposition, most measurements with the Ge(Li) system were made in various directions within 6 to 7 miles of the plant. Particular emphasis was placed on making representative measurements, so the sites chosen (e.g., lawns, pastures and fields) were relatively large, flat, and open. Because the HPIC monitors are sensitive and saturate at 200 μR/h, most of these were positioned at distances of about 12.4 miles in the prominent wind directions and near population centers. App.Table II-7 contains a list of the sites monitored and App.Figures II-1 and II-2 indicate their locations.

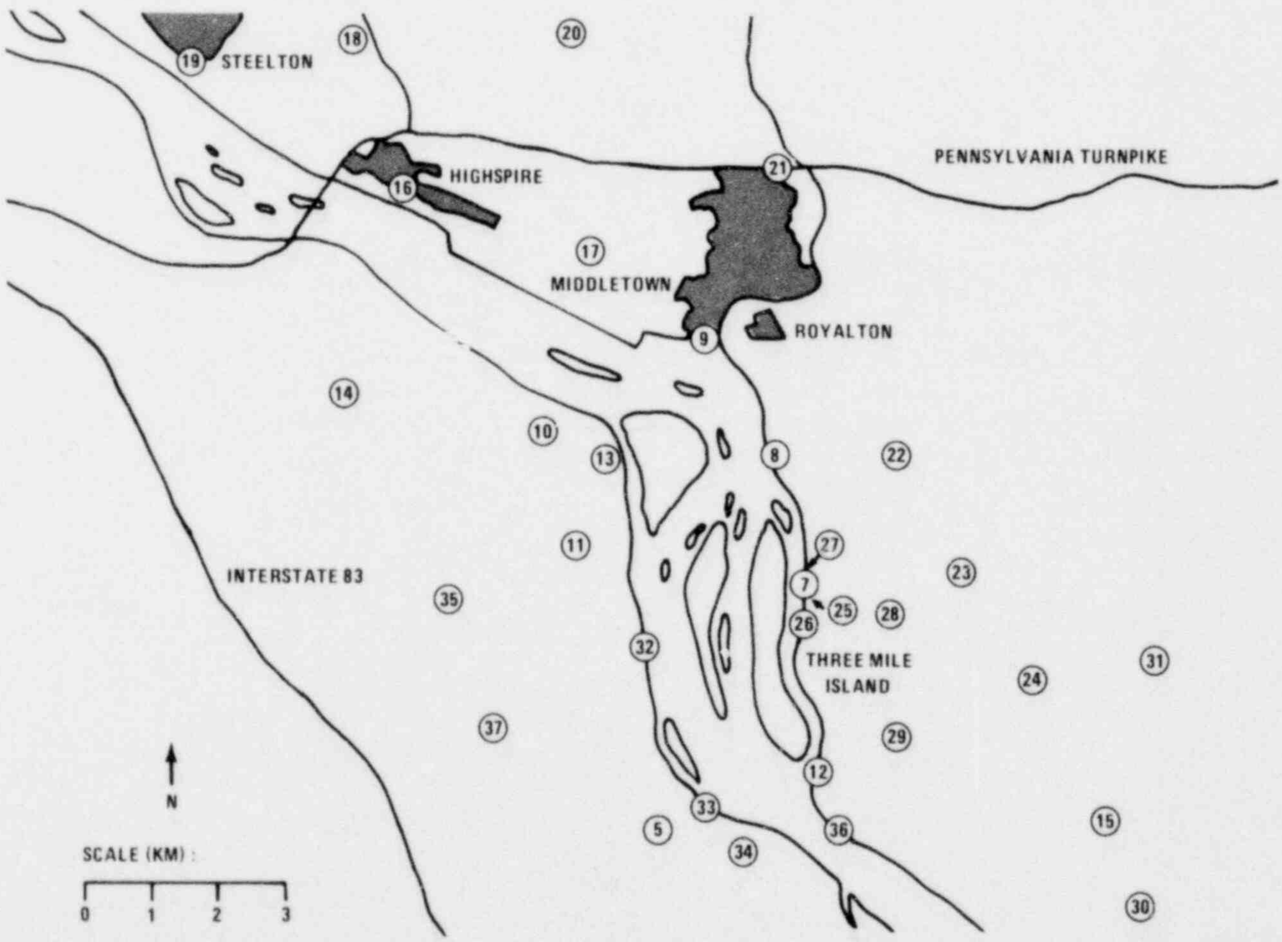
Lawrence Livermore Laboratory (LLL)—LLL, using its atmospheric release advisory capability (ARAC) computerized system, provided meteorological forecasts and predictions of plume trajectories for gui-

APP. TABLE II-7. Field Ge(Li) and HPIC measurement locations and time[7]

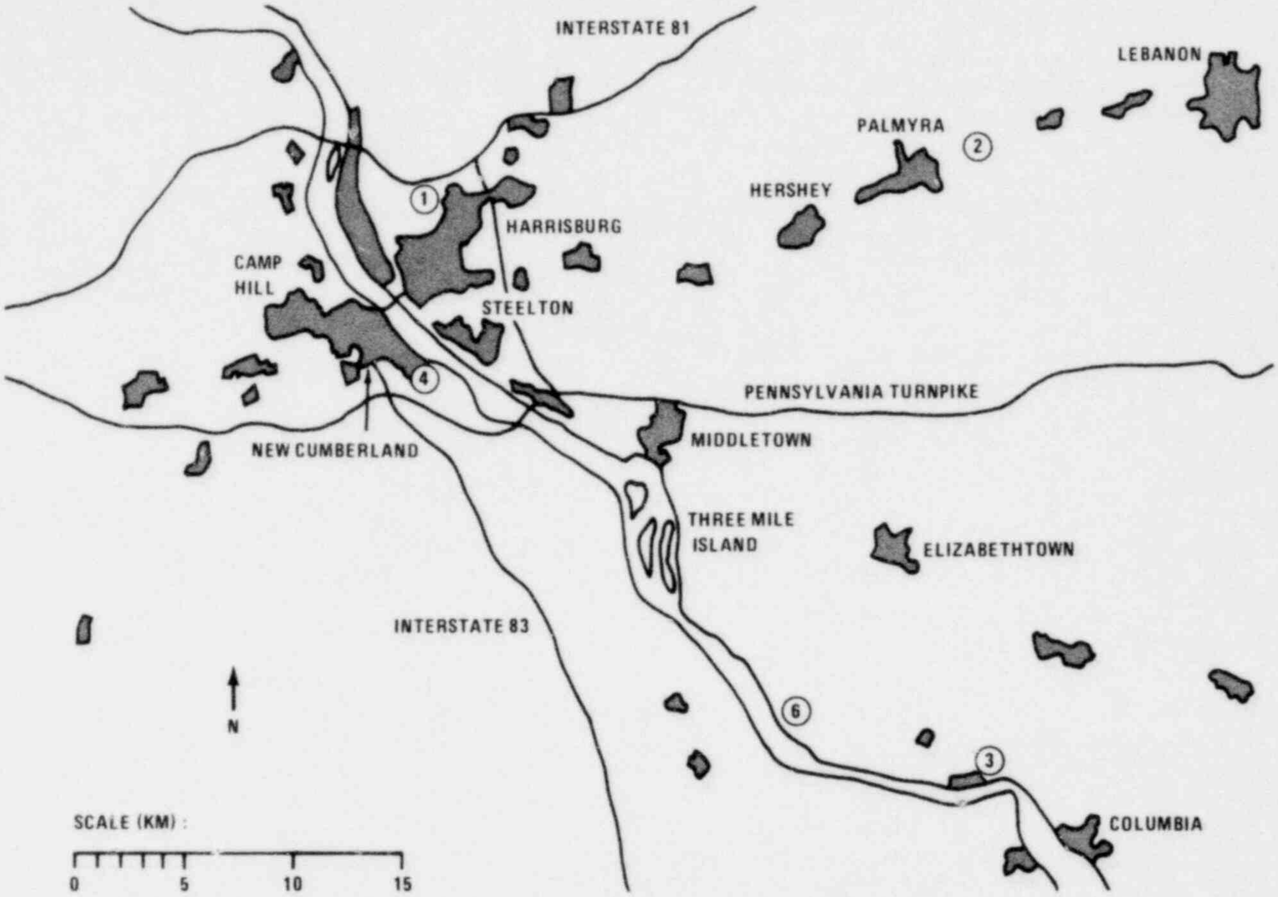
EML Location No.	Distance (miles)	Direction (degrees)	Sector	Date Monitored	Time (EST)
1	12.1	325	NW	4/2	11:05a.m.
2	15.5	40	NE	4/2	1:30p.m.
3	11.3	124	SE	4/2	4:20p.m.
4	8.14	305	NW	4/2	6:10p.m.
5	2.5	203	SSW	4/3	10:35a.m.
6	6.52	143	SE	4/3	12:40p.m.
7	0.4	90	E	4/3	2:50p.m.
8	1.2	5	N	4/3	3:40p.m.
9	2.3	346	NNW	4/3	4:30p.m.
10	2.5	304	NW	4/4	9:45a.m.
11	1.8	281	W	4/4	10:20a.m.
12	1.8	162	SSE	4/4	12:20p.m.
13	1.9	309	NW	4/5	2:00p.m.
14	4.3	294	WNW	4/5	4:00p.m.
15	3.9	125	SE	4/6	2:10p.m.
16	5.0	318	NW	4/7	9:15a.m.
17	3.5	332	NNW	4/7	10:45a.m.
18	6.34	323	NW	4/7	12:25p.m.
19	6.9	311	NW	4/7	2:40p.m.
20	5.4	341	NNW	4/7	3:50p.m.
21	3.7	1	N	4/8	9:25a.m.
22	1.7	46	NE	4/8	11:00a.m.
23	1.8	87	E	4/8	2:05p.m.
24	2.6	108	ESE	4/8	3:30p.m.
25	0.5	109	ESE	4/9	9:30a.m.
26	0.6	133	SE	4/9	10:40a.m.
27	0.4	70	ENE	4/9	11:45a.m.
28	1.2	104	ESE	4/10	11:55a.m.
29	1.9	139	SE	4/10	3:25p.m.
30	4.7	131	SE	4/10	4:35p.m.
31	3.7	100	E	4/10	6:05p.m.

APP. TABLE II-7. Field Ge(Li) and HPIC measurement locations and time—Continued

EML Location No.	Distance (miles)	Direction (degrees)	Sector	Date Monitored	Time (EST)
32	1.3	243	WSW	4/11	11:25a.m.
33	2.2	195	SSW	4/11	12:10p.m.
34	2.5	186	S	4/11	2:35p.m.
35	3.0	267	W	4/11	4:10p.m.
36	2.4	163	SSE	4/11	6:10p.m.
37	2.9	241	WSW	4/13	8:55a.m.



APP. FIGURE II-1. EML Measurement Sites Close to TMI Station



APP. FIGURE II-2. EML Measurement Sites in Harrisburg Metropolitan Area

dance in radiation monitoring and evacuation planning. The purpose of ARAC is to provide estimates of the effects of accidental or routine atmospheric releases of hazardous materials, including radioactive materials. The DOE used ARAC in its accident response functions by directing aircraft and terrestrial radiological monitoring and sampling efforts in the vicinity of Three Mile Island.[10] Other agencies, including the NRC, unfamiliar with ARAC capabilities for predicting plume behavior did not use it during the response to the accident. Appendix II.5 contains a detailed discussion of ARAC.

REFERENCES AND NOTES

[1]"Region I Radiological Assistance Program (RAP) Response to the Incident at Three Mile Island, March 28-April 18, 1979," DOE, Appendix G, at 1ff.

[2]Department of Energy Radiological Response to the Three Mile Island Accident, April 14, 1979, at Fig. 3.

[3]"Region I Radiological Assistance Program (RAP) Response to the Incident at Three Mile Island, March 28-April 18, 1979," DOE, Appendix G at 1, 2.

[4]*Id.* at 1.

[5]*Id.* at 1-2.

[6]Kevin M. Miller, et al., "Radiation Measurements Following the Three Mile Island Reactor Accident," (EML-357) Environmental Measurements Laboratory, New York, at p. 2.

[7]*Id.*, Table I at 7, 8.

[8]*Id.*, Figure 1 at 12.

[9]*Id.*, Figure 7 at 13.

[10]Memorandum from O. D. T. Lynch, Jr., TMI SIG, to Files, Task Group 3, TMI SIG, "Meeting with Lawrence Livermore Laboratory (LLL) Officials on Atmospheric Release Advisory Capability (ARAC) Response to TMI Accident," August 22, 1979.

APPENDIX II. 5

ATMOSPHERIC RELEASE ADVISORY CAPABILITY (ARAC) UTILIZATION DURING THE TMI ACCIDENT

ARAC is a computer service developed by the Lawrence Livermore Laboratory (LLL) under a Department of Energy (DOE) contract. The purpose of ARAC is to provide accurate estimates of the effects of accidental or routine atmospheric releases of hazardous materials, including radioactive materials.

During the accident, ARAC was utilized extensively by DOE and the results were provided to the NRC and other agencies. As a sponsor of the system, DOE personnel were familiar with ARAC and used the ARAC products to direct aircraft and terrestrial radiological monitoring and sampling efforts. Other agencies, including the NRC, were much less knowledgeable of ARAC and its capabilities for predicting plume development and did not use it effectively during the response to the accident.

ARAC System—The ARAC system provides real-time regional assessments of plume development using numerical models and local site data. The models used by ARAC vary in complexity from a single trajectory model to a set of advanced regional transport and diffusion models.[1]

ARAC Product— The ARAC system produces several graphical presentations (plots) that include the following:

1. Simple Gaussian Curves—This plot is a concentration-downwind distance curve that indicates cloud center concentration, ground level concentration on center line, and integrated ground levels.
2. Relative Concentration—This plot is a plan view of the plume (as a distribution of dots) intended to depict an overview of general plume behavior.
3. Instantaneous Concentration (χ / Q)—This plot indicates contour lines of instantaneous concentrations at a particular elevation. The instantaneous concentrations are used, with field measurements in the plume, to predict exposure rates on the ground for use as health physics advisory information.

4. Vertical Plume Distribution—This plot provides a side view of the plume.
5. Dose—This is a health physics advisory plot similar to the instantaneous concentration plot. A source term is required, and exposure-dose rates on the ground or above the ground are provided. Integrated exposure and dose can also be provided.

ARAC Input-Output for TMI—ARAC was activated at 11:20 a.m. (EST) on March 28, 1979, by J. Bufait, DOE, Germantown, Md. The first product was provided to Bufait at the Emergency Operations Center (EOC), DOE Headquarters, Germantown, at approximately 1:00 p.m. This product was an integrated air and surface concentration calculation using a simple Gaussian model.[2]

LLL transmitted an improved ARAC product (using TMI-2 local meteorology and topographic data) to Bernard Weiss at the NRC Incident Response Center, in Bethesda, Md. at 4:07 p.m. There were transmission-reception problems due to facsimile machine incompatibility and the product was finally received at the NRC Incident Response Center (IRC) at approximately 6:00 p.m.[3] (Correct facsimile machine matching was never realized, and the ARAC plots received at the NRC IRC were continually seriously distorted.) The NRC staff using the product had never seen it before and never realized that distortion existed.[4] The product transmitted to NRC at this time consisted of two ^{131}I contour plots: (1) integrated surface air concentration at the surface (about 2 meters above) for an assumed unit release at 7:00 a.m. based on 12:00 noon meteorology; and (2) deposition from an assumed unit release at 7:00 a.m. (of 1 Ci/s of ^{131}I) based on 12:00 noon meteorology.

By 8:00 p.m., USGS-digitized terrain data was added to the ARAC computer bank. By this time the DOE at Capital City Airport had chosen the grid size of 2 kilometers for the ARAC product. The USGS terrain data, which were on a 62.5-meter grid, were averaged over a 2-kilometer grid to match the computer model grid network.[5]

On March 29, ARAC runs were made every 2 hours during the day and provided to the DOE. Calculations were not performed at night; however, LLL did have the capability to provide 24-hour service.

At 2:00 p.m. on March 30, ARAC received the TMI-2 meteorological tower data from Pickard, Lowe, and Garrick, consultant to Met Ed. These data were provided hourly to LLL until 10:00 p.m. on March 31, when the tower data were transmitted automatically. At 8:15 p.m., on March 30, the ARAC products were transmitted, for the first time, to Robert Bores at the NRC Region I office, King of Prussia, Pa. Again, some difficulties were encountered due to facsimile machine interfaces,[6] but plots were not distorted.[7]

On March 31, eight ARAC runs were made between 9:00 a.m. and 6:00 p.m. ARAC products continued to be provided through April 4, after which they were only provided to the DOE on an as-needed basis until April 18, 1979. During the emergency, ARAC products were provided to the following agencies: DOE Command Center, Capital City Airport; NRC IRC, Bethesda, Md.; EG&G, Las Vegas, Nev.; DOE EOC, Germantown, Md.; DOE Nevada, Las Vegas, Nev.; NRC, Region I, King of Prussia, Pa.[8]

ARAC was typically able to produce its product and provide it to Harrisburg within 55 to 60 minutes from input of meteorological data. It was assumed that the meteorology was consistent for a 2-hour interval.

For the TMI-2 accident, ARAC provided (1) the "dot plot" relative concentration plot and (2) the instantaneous concentration plot for a 65-meter height above the surface. The grid size used for the TMI-2 accident was 2 kilometers on the horizontal and 35 meters on the vertical, although a smaller grid size was available, down to 62.5 meters for the horizontal.

Agency Use of ARAC Products During the Accident

DOE—DOE at Capital City Airport, the primary user of ARAC, utilized ARAC plots to vector their aircraft and ground sampling operations. Aircraft operations by EG&G and reports from ground survey teams indicated agreement with ARAC predictions on the location and extent of plume.[9]

NRC Incident Response Center (IRC), Bethesda, Md.— The ARAC information was received at the IRC in Bethesda, Md. by Bernard Weiss. Weiss passed the ARAC plots on to the Meteorology Section of the Hydrology-Meteorology Branch, NRR (HMB(M)).[10]

Hydrology-Meteorology Branch [*HMB(M)*]—Personnel were providing meteorological support to the IRC. The predictions performed by the HMB(M) personnel were based on a simple straight-line Gaussian model that did not account for changes in wind direction at distances away from the site or topography. HMB(M) personnel had confidence in their model and calculations. Although several HMB(M) personnel had some knowledge of ARAC, they did not have enough familiarity with it to make it their primary source of information. Thus, the ARAC products were used only as a check. In addition, the model used by HMB(M) was sufficiently accurate close in (within 5 miles) and they believed that ARAC could not be utilized close in, but was useful at ranges over 10 miles.[11]

The NRC was not a subscriber to the ARAC system (it was a DOE-sponsored effort being used for DOE facilities), and no one at the NRC had sufficient familiarity with ARAC to make effective use of the system.

Because the NRC staff was not familiar with the ARAC product, they did not recognize the usefulness of the ARAC plots that were available. Although it was recognized that 2-kilometer grid spacing was being used, no one knew that the grid size could be varied, down to 62.5 meters, if necessary. ARAC was not effectively used by the NRC IRC staff. One individual indicated his reasons for limited use:

> This is one of the reasons why, I will be quite honest, I hesitate to run with new data when we don't understand exactly how it is coming in, how it is being developed, the quality control that goes into it because you get into a lot of trouble making erroneous decisions. That is why I say the main value of the ARAC data was to confirm the general direction that we would have a problem if we had one.[12]

ARAC was a valuable tool that utilized wind fields developed from a variety of onsite and offsite sources. ARAC provided near real-time predictions of plume behavior, and, if necessary, close-in downwind concentrations, but was relegated to a position of low importance. ARAC was an available tool for use in responding to the accident, but was not effectively used by the NRC.

REFERENCES AND NOTES

[1]M. H. Dickerson, Atmospheric Release Advisory Capability (ARAC), Update 1977, *IEEE Transactions on Nuclear Science* NS-25, 850 (1978). Atmospheric Release Advisory Capability (ARAC) Operator's Guide, Site Facility, UCID-17490 Lawrence Livermore Laboratory, July 13, 1977.

[2]Memorandum from O. D. T. Lynch, Jr., TMI SIG, to Files, Task Group 3, TMI SIG, "Meeting with Lawrence Livermore Laboratory (LLL) Officials on Atmospheric Release Advisory Capability (ARAC) Response to TMI Accident," at 1, August 22, 1979.

[3]LLL TMI Incident Log Book, entry for 3:00 p.m., March 28, 1979.

[4]Weiss dep. at 223.

[5]Memorandum from O.D.T. Lynch, Jr., TMI SIG, to Files, Task Group 3, TMI SIG, "Meeting with Lawrence Livermore Laboratory (LLL) Officials on Atmospheric Release Advisory Capability (ARAC) Response to TMI Accident, at 2, August 22, 1979.

[6]*Id.* at 2.

[7]Memorandum from J. P. Stohr, NRC IE, to J. C. Guibert, Office of Commissioner Kennedy, "Request for DOE Illustrative Material," April 12, 1979.

[8]*Id.* at 4.

[9]*Id.* at 5.

[10]Weiss dep. at 215.

[11]Memorandum of telephone call from P. Murray, NRC, to J. E. Fairobent, ARAC, "ARAC Data Received from B. Weiss," July 13, 1979.

[12]Sniezek dep. at 40.

APPENDIX II.6
RADIOLOGICAL CHRONOLOGY OF EVENTS

ITEM	DATE AND TIME	EVENT DESCRIPTION	REFERENCES
	3/28/79		
1	4:00 a.m.	TMI-2 at 97% full power, auxiliary and fuel handling building ventilation systems exhausting through HEPAs and charcoal adsorbers.	1
2	4:00 a.m.	Reactor trip.	1
3	4:08 a.m.	Reactor containment sump pump WDL-P-1A started to pump coolant resulting from stuck-open PORV to auxiliary building sump tank.	1
4	4:10 a.m.	Reactor containment sump pump WDL-P-2B also started.	1
5	4:38 a.m.	Sump pumps manually turned off after transfer of approximately 8000 gal to auxiliary building. This liquid was not highly radioactive.	1
6	5:42 a.m.	Primary to secondary steam generator leaks indicated by samples from the condenser vacuum pump.	2

ITEM	DATE AND TIME	EVENT DESCRIPTION	REFERENCES
7	6:30 to 7:00 a.m.	Technicians reported rapidly increasing levels of radiation in the auxiliary building, up to 10 R/h.	1
8	6:43 a.m.	Reactor coolant sample taken that alarmed TMI-1 sample room area monitor, sample analyzed at 140 μCi/ml gross activity.	1
9	6:48 a.m.	Particulate channel of station vent monitor HP-R-219 alarmed at its setpoint of 0.3 μCi/sec, the technical specification limit for ^{131}I and particulates.	1
10	6:50 a.m.	Technician walked through liquid in auxiliary building, but was not contaminated.	1
11	6:51 a.m.	Particulate channel of auxiliary building exhaust monitor HP-R-228 alarmed at 0.3 μCi/sec setpoint.	1
12	6:54 a.m.	Condenser vacuum pump discharge monitor VA-R-748 alarmed at 0.024 μCi/sec setpoint.	1
13	6:55 a.m.	Site emergency declared by shift supervisor Zewe.	1
14	6:58 a.m.	R. Dubiel, Supervisor of Radiation Protection and Chemistry noted that containment dome monitor HP-R-214 was in alert and increasing.	1
15	7:01 to 7:06 a.m.	Fuel handling building iodine monitor downstream of filters, fuel handling building particulate monitor upstream of filters, and reactor containment purge particulate monitor alarmed.	1
16	7:12 a.m.	Noble gas channel of station vent monitor HP-R-219 alarmed at $2.8x10^{-4}$ μCi/cc setpoint. Reactor coolant letdown monitor alarmed.	1

ITEM	DATE AND TIME	EVENT DESCRIPTION	REFERENCES
17	7:00 to 8:20 a.m.	Numerous TMI-1 and TMI-2 area and exhaust monitors alarmed for particulates, iodines, and noble gases. Of importance, iodine channel of station vent stack monitor HP-R-219 alarmed at 7:35 a.m. Other monitors that alarmed included: 1. Reactor containment purge area monitor (7:19 a.m.). 2. TMI-1 fuel handling building particulate monitor (7:20 a.m.). 3. Fuel handling building exhaust area monitor (7:23 a.m.). 4. Reactor containment purge gas monitor (7:23 a.m.). 5. Fuel handling building exhaust gas monitor downstream of filters (7:23 a.m.). 6. Fuel handling building exhaust gas monitor upstream of filters (7:25 a.m.). 7. Auxiliary building exhaust gas monitor (7:28 a.m.). 8. Reactor containment purge iodine monitor (7:29 and 7:37 a.m.). 9. Auxiliary building exhaust iodine monitor (8:00 a.m.). 10. TMI-1 fuel handling building exhaust particulate monitor (8:19 a.m.).	1
18	7:24 a.m.	General emergency declared by Station Manager Miller.	1
19	7:55 a.m.	Offsite survey team reported less than 1 mR/h at both the north gate and observation center.	1
20	7.56 a.m.	Reactor building isolated automatically on high reactor building pressure.	1
21	8:07 a.m.	Model room door between auxiliary and fuel handling buildings recorded as closed, separating the buildings with respect to ventilation.	1
22	Approx. 8:00 a.m.	Control Room Operator Hugh A. McGovern manually activated control room recirculation by starting fan AH-E-4B.	3

ITEM	DATE AND TIME	EVENT DESCRIPTION	REFERENCES
23	8:30 a.m.	Onsite radiation readings of 7-14 mR/h. Offsite readings less than 1 mR/h, with a few locations at 1-3 mR/h.	2
24	8:30 a.m.	NRC mobile lab left Millstone Station for the Three Mile Island Station.	1
25	8:43 a.m.	TMI-1 nuclear sampling room monitor increased to 10 R/h (5 feet from TMI-2 coolant sample lines).	1
26	8:50 a.m.	Reactor coolant sample showed high radioactivity levels (over 500 μCi/ml).	1
27	9:22 a.m.	Goldsboro air sample indicated 10^{-8} μCi/cc iodine-131.	1
28	9:48 a.m.	Particulate channel of control room air intake monitor HP-R-220 alarmed.	1
29	10:10 a.m.	Noble gas channel of control room air intake monitor alarmed.	1
30	10:12 a.m.	ECS moved from TMI-2 control room to TMI-1 control room.	1
31	10:17 a.m.	TMI-2 control room personnel donned face masks with particulate filters.	1
32	11:10 a.m.	Island evacuated of all nonessential personnel.	1
33	11:25 a.m.	Onsite radiation readings of 5-10 mR/h recorded; highest 365 mR/h at western boundary. Offsite readings increasing average of 1-5 mR/h; highest 13 mR/h 6 miles WNW. The radioactive releases from the plant are confused with the plume of steam being released by atmospheric steam dumping.	2
34	12:00 noon	Entries into auxiliary building made without high range pocket dosimeters. Areas up to 1000 R/h surveyed. The three individuals each received 800 mrem (10-min entry).	1

ITEM	DATE AND TIME	EVENT DESCRIPTION	REFERENCES
35	2:27 p.m.	Offsite radiation readings in Middletown indicated 1-2 mR/h.	2
36	3:10 p.m.	Masks removed by control room personnel.	1
37	3:28 p.m.	Met Ed personnel report 50 mR/h readings on Pa 441 east of plant.	2
38	4:45 p.m.	Met Ed reported to Commonwealth of Pennsylvania that radiation levels onsite at north gate have increased from 30 mR/h to 50 mR/h. Offsite readings less than 1 mR/h and a maximum of 9.6×10^{-9} $\mu Ci/cc$ ^{131}I.	2
39	5:20 p.m.	Onsite radiation level of 210 mR/h at northwest boundary.	2
40	6:00 p.m.	BNL team reported radiation levels of 1-2 mR/h in the plume 5-10 miles from the site with less than MDA (10^{-10} $\mu Ci/cc$ ^{131}I.)	2
41	7:00 p.m.	NRC inspectors reported 2 mR/h on Pa. Turnpike and 10-15 mR/h at Olmstead Plaza.	1
42	7:30 p.m.	NRC mobile lab in operation.	1
43	7:43 p.m.	Onsite radiation levels decreased to 10-20 mR/h with maximum of 42 mR/h behind TMI-1 warehouse. Offsite readings less than 1 mR/h.	2
44	9:00 p.m.	Auxiliary operator entered auxiliary building alone and without a safety man or high range dosimeter. After passing through 100 R/h (measured) radiation fields, he discovered his pocket dosimeter off scale, and also caused the GM counter to alarm on entering the control room. Instead of decontaminating, he reentered the auxiliary building after he zeroed his pocket dosimeter. Again, the pocket dosimeter went off scale. His TLD indicated a dose of 3.2 rem.	1
45	11:25 p.m.	Onsite radiation readings increased to 365 mR/h beta-gamma and 50 mR/h gamma 1000 feet NW of TMI-2 vent.	1

ITEM	DATE AND TIME	EVENT DESCRIPTION	REFERENCES
46	During Day	D. Fredrickson, NIH, advised Secretary Califano of HEW that as a precautionary measure there should be supplies of potassium iodide in the Harrisburg area as a thyroid blocking agent.	4
47	0:55 a.m.	Auxiliary and fuel handling building ventilation fans were stopped in an attempt to reduce releases.	1
48	2:11 a.m.	Particulate channel of control room air intake monitor HP-R-220 alarmed, and all personnel in TMI-2 control room donned masks. Ventilation fans in auxiliary and fuel handling buildings restarted.	5
49	3:15 a.m.	Levels of particulate radioactive material in TMI-2 control room quickly decreased, allowing the removal of masks.	5
50	During morning	Onsite radiation levels gradually decreased to 5-10 mR/h. Offsite levels were 1-3 mR/h with no detectable ^{131}I.	2
51	4:35 a.m.	Makeup tank MU-T-1 vented to the vent header and the waste gas decay tanks for the first time.	1
52	8:30 to 11:30 a.m.	Survey at letdown filter cubicle indicated $>$1000 R/h through a porthole, 2-5 R/h general area. The technician received 1.4 rem obtaining these readings.	1
53	Approx. 7:00 a.m.	Technician surveyed the auxiliary building, and reported several areas with $>$100 R/h radiation fields.	1
54	9:03 a.m.	TLD reader moved to observation center due to 40 mR/h background on site.	1
55	12:15 p.m.	Plastic sheets placed over water in the auxiliary building to reduce gaseous releases.	1

ITEM	DATE AND TIME	EVENT DESCRIPTION	REFERENCES
56	12:40 p.m.	Sumps turned off from turbine building, control building, and control and service building.	1
57	1:30 p.m.	IWTS discharge sampled. No iodine was detected.	1
58	2:00 p.m.	RMC set up their whole-body counter and mobile lab at observation center.	1
59	2:10 p.m.	A helicopter measured 3 R/h beta-gamma and 400 mR/h gamma, at 15 feet above the TMI-2 stack.	1
60	3:00 p.m.	Met Ed retrieved 17 TLDs from fixed positions located within 15 miles of the plant. These TLDs had been exposed for 3 months, including the first 1-1/2 days after the accident.	1
61	4:15 p.m.	100 ml reactor coolant sample taken by a radiation protection foreman and a chemistry foreman. Although they were informed that exposure rates of 800 to 1000 R/h were probable, they wore no extremity dosimeters and took no air samples. The sample read >1000 R/h on contact, 400 R/h at 1 foot and 10-15 R/h at 3 feet. The chemistry foreman received 4.1 rem whole-body and had nonremovable contamination measuring 25 mR/h contact on his hands. The radiation protection foreman had 150 mR/h contact nonremovable contamination on his forearm.	1
62	5:55 p.m.	NRC executive management team directed Met Ed to stop discharging all water.	1
63	8:20 to 8:45 p.m.	Makeup tank vented intermittently to vent header. Radiation monitors increased during each attempt, indicating leaks into the vent header system.	1
64	10:04 to 10:13 p.m.	The contaminated chemistry foreman, with 25 mR/h hand contamination, was whole-body counted and sent home.	1
65	11:00 p.m.	Two engineers surveyed the auxiliary building for water leaks, and one engineer received 3.14 rem.	1

ITEM	DATE AND TIME	EVENT DESCRIPTION	REFERENCES
66	12:00 midnight	Onsite and offsite readings generally less than 0.5 mR/h; some onsite readings 1-30 mR/h.	2
67	12:00 midnight to 7:00 a.m.	Makeup tank vented intermittently numerous times.	1
68	4:35 a.m.	The liquid pressure relief (MU-R-1) on the makeup tank opened to the reactor coolant bleed holdup tanks. The makeup tank level dropped to zero, and suction for the makeup pumps was supplied by the borated water storage tank. The borated water cycled directly to the makeup tank and RCBHTs.	1
69	7:10 a.m.	Operator Faust opened the makeup tank vent valve MU-V-13 (with supervisory concurrence) in order to stop depleting the BWST. The vent was left open continually except for possible short periods.	1
70	7:22 a.m.	Readings of 150-180 mR/h at 130 feet above TMI-2 stack.	2
71	7:56 a.m.	Reading of 1 R/h (beta-gamma) at 130 feet above stack.	2
72	8:01 a.m.	Reading of 1200 mR/h (beta-gamma) at 130 feet above stack. Helicopter cannot duplicate reading. Onsite readings 10-30 mR/h at west boundary, offsite locations close to plant increased to 5-18 mR/h.	1, 2
73	8:34 a.m.	Unit supervisor of station operations called Civil Defense to discuss evacuation.	1
74	12:30 p.m.	Met Ed received TLD analyses from TLDs pulled on 3/29. Results showed less than 25 mrem per quarter offsite, maximum onsite dose was 1044 mrem per quarter. Iodine air samples on the island all less than 0.03 p Ci/m^3 except one location NNE on island of 0.47 p Ci/m^3 (unrestricted area MPC is 100 p Ci/m^3).	1

ITEM	DATE AND TIME	EVENT DESCRIPTION	REFERENCES
75	2:40 p.m.	The contents of waste gas decay tank B were transferred into the reactor building.	1
76	7:45 p.m.	Air sample at observation center showed $1x10^{-9}$ $\mu Ci/cc$ ^{131}I activity.	2
77	During Day	Secretary Califano directed FDA to make potassium iodide available to Commonwealth of Pennsylvania.	4
78	12:00 midnight	Radiation work permit system back in use by Met Ed.	6
79	6:20 a.m.	Argonne team began terrestrial monitoring under the plume.	2
80	9:00 a.m.	Offsite readings increased to 5-10 mR/h, up to 38 mR/h on Pa 441.	2
81	11:15 a.m.	100 mR/h observed at east site boundary.	2
82	Approx. 12:00 noon	EPA, NRC, FDA-BRH all distributed TLDs. RMC and Met Ed TLDs collected.	2
83	2:37 p.m.	56 mR/h observed at east site boundary.	2
84	During day	FDA requested Mallinckrodt Corporation to manufacture potassium iodide.	4
85	1:30 a.m.	11000 1-ounce bottles of potassium iodide delivered to Harrisburg Airport.	4
86	Approx. 12:00 Noon	NRC TLDs collected, 37 additional stations established up to 12 miles from the plant.	2
87	4/1/79-4/5/79	Six shipments of potassium iodide received at Harrisburg Airport.	4
88	4/1/79-4/18/79	TLDs distributed and collected periodically.	2
89	4/8/79	Science Applications, Inc., obtained auxiliary building samples upstream and downstream of the exhaust filters, which showed an overall decontamination factor of 1.2.	7

ITEM	DATE AND TIME	EVENT DESCRIPTION	REFERENCES
90	Early April	Supplementary air filters arrived by air transport for installation on auxiliary building roof.	8
91	4/20/79	Filters (carbon adsorbers and HEPAs) changed in the A train of the auxiliary building exhaust system.	9
92	4/24/79	Filters (carbon adsorbers and HEPAs) changed in the A train of the fuel handling building exhaust system.	9
93	4/25/79	Filters (carbon adsorbers and HEPAs) changed in the B train of the auxiliary building exhaust system.	9
94	5/1/79	Supplementary auxiliary building filtration system put into operation.	10
95	5/20/79	TMI-2 stack capped to ensure all releases go through the supplementary auxiliary building filtration system.	11
96	5/23/79	Filters (carbon adsorbers and HEPAs) changed in the B Train of the fuel handling building exhaust system.	9

REFERENCES AND NOTES

[1]NRC, "Investigation into the March 28, 1979 Three Mile Island Accident by Office of Inspection and Enforcement," NRC Report NUREG-0600, August 1979.

[2]"Report of the Task Group on Health Physics and Dosimetry" to President's Commission on the Accident at Three Mile Island, J. A. Auxier, et al., October 1979.

[3]H. A. McGovern, "Chronology of Actions and Observations of System Beginning 7:00 a.m.," March 28, 1979, NRC Accession Number 7906130096.

[4]G. K. MacLeod, Secretary of Health, Commonwealth of Pennsylvania, "The Decision to Withhold Distribution of Potassium Iodide During the Three Mile Island Accident: Internal Working Document."

[5]TMI-2 Control Room Log, NRC Accession Number 7906140471.

[6]Radiation Work Permit Log, NRC Accession Number 7906150259.

[7]Memorandum from R. R. Bellamy, NRC, to F. Miraglia, "Telephone Conversation Regarding TMI Iodine Species," November 13, 1979.

[8]Collins dep. at 71.

[9]"Technical Staff Analysis Report on Iodine Filter Performance," to President's Commission on the Accident at Three Mile Island, William M. Bland, Technical Assessment Task Force, October 1979, p. 9.

[10]Memorandum, J. T. Collins, NRC to Ben Rusche, GPU, "Charcoal Test Cartridges from Supplementary Auxiliary Building Filtration System," June 3, 1979.

[11]J. T. Collins, W. D. Travers, R. R. Bellamy, "Report on Preliminary Radioactive Airborne Release and Charcoal Efficiency Data: Three Mile Island Unit 2", NRC, Washington, D.C. 20555.

APPENDIX II.7

CALCULATION OF LEACHING FROM REACTOR FUEL

Analysis of a sample of reactor coolant taken on March 29 showed only a minute fraction of a percent of the core inventory of refractory elements (Sr, Ru, Ba) in the reactor coolant. A sample taken on April 10 showed about 1% of the core inventory of Sr, Ba, La and Mo. The fractions were quite variable both from element and from laboratory to laboratory but a figure of 1% represents a reasonable average.

A fit to the data of Katayama[1] gives, for early time,

$$\dot{w} = 3.2 \times 10^{-4} t^{-.9} A$$

(average of ^{90}Sr and ^{137}Cs)

$\dot{w}$ = leaching rate, g/day

t = time in days

A = surface area, cm^2

Then the total amount leached is

$$w = \int_0^T \dot{w}\,dt = .0032 t^{0.1} A$$

where w is total leached in grams.

The total weight of fuel is 9.31×10^7 g.[2]

If it is assumed that 1/3 of the fuel is damaged, the mass of damaged fuel is 3.1×10^7 g.

If the damaged fuel is in the form of spheres of uniform size, the volume of each is

$$v = 4\pi \frac{r^3}{3}$$

and the surface area is

$$a = 4\pi r^2$$

The number of spheres is then

$$n = \frac{V}{v} = \frac{M}{pv}$$

where M is the mass of fractured fuel, and p is fuel density. The total surface area is then

$$A = na = \frac{3M}{pr}$$

The apparent surface area is

$$A_{app} = \frac{(0.01)(3.1x\,10^7)}{(.0032t^{0.1})}$$

For April 10 ($t = 14$)

$$A_{app} = 7.4x\,10^7 \mathrm{cm}^2$$

The equivalent radius sphere is

$$r = \frac{3M}{\rho A} = \frac{(3)(3.1x\,10^7)}{(7.4x\,10^7)(10.9)} = 0.12\,\mathrm{cm}$$

This measurement appears to be rather small for an average size, a diameter of about .090"; however, the precision of the concentration data is so poor that a factor of 2 larger would be completely reasonable also. If the fraction of fuel damaged enough to be leached is larger, the average radius would be larger.

Experimental data[3] indicates that particle sizes under similar (LOCA) conditions may be of the order of 0.2 centimeters.

Although the calculations cannot be made with any precision, it appears that the presence of refractory elements in the reactor coolant can be explained by leaching alone.

A similar calculation has been carried out by Powers.[4] His results are not precisely the same as those found above, though they are certainly within the expected error bounds for such an uncertain computation. Powers has computed the possible fraction of the core in any particle size range. A lower limit of about 0.03 centimeter radius is imposed, because particles smaller than this would be levitated by the flow and would be distributed throughout the system. As will be seen, if the particles are larger than about 0.2 centimeter radius, almost any amount of the core can be involved. If the particles are smaller than about 0.1 centimeter radius, only a small fraction of the cores can be involved. Therefore, it appears improbable that much of the core has been broken up into extremely fine particles.

REFERENCES AND NOTES

[1]Y.B. Katayama and J.E. Mandel "Leaching of Irradiating LWR Fuel Pellets in Deionized Water, Sea Brine, and Typical Ground Water," *ANS Trans.*, 7:447, Nov.-Dec. 1977.

[2]Met Ed, Jersey Central Power and Light Co., Pennsylvania Electric Co., "Final Safety Analysis Report, Three Mile Island Nuclear Station—Unit 2."

[3]T.R. Yackle & P.E. MacDonald, Influence of Internal Pressure and Prior Irradiation on Deformation of Zircaloy Cladding Doc-3. Results Presentation at 7th Water Reactor Research information Meeting, Gaithersburg, Maryland, November 5-9, 1979. Available from the NRC Public Document Room.

[4]D.A. Powers, Sandia Laboratories, "Status of the Reactor Core Based on Fission Product Analysis," November 29, 1979.

APPENDIX II.8
TMIBOIL CODE CALCULATIONS OF CORE DAMAGE AT 3 HOURS

A code called TMIBOIL was written recently to calculate more precisely the time–temperature relationship for the fuel rods in TMI-2, using relatively precise analytical expressions, few simplifying assumptions, and parametric treatment of several of the system variables. The code has been written so that the accident "scenario" can be varied over wide ranges, and the calculations fit parametrically into the scenario. Specifically, the code does not require an exact knowledge of the makeup and letdown flows, but it does require a stated rate of change (as one of the parameters) of level of coolant in the core. In addition, the functions of TMIBOIL include the following:

- calculation of the steam production rate as a function of the length of the fuel rod submerged in coolant, the system pressure, the time in the scenario, and the rate of coolant level change;
- calculation of the specific heat of the fuel rod;
- analytical calculation of the heat of oxidation at each node, time, and temperature;
- calculation of the radiative heat transfer coefficient and addition to the conductive heat transfer coefficient;
- parametrical use of the conduction heat transfer coefficients, the final depth of boil off, the rate of boil off, the assembly power (radial peaking factor times a fixed axial power profile), and the presence of "chilling" rods (such as control and poison rods);
- calculation of the total steam produced in each time increment, and the surplus of steam exiting the fuel subchannels for each time increment;
- report of the axial node in 1-inch increments, the elapsed time in minutes, the fuel (cladding) temperature in °F, the steam temperature in °F, the steam flow in pounds per hour, the thickness of Zircaloy metal left in the wall (not converted to oxide), and the ratio of the oxidation heat to the decay heat at each node;
- calculation of the total number of gram moles of hydrogen produced;
- cutoff of oxidation heat of Zircaloy-steam reaction at 3600°F, assumption that molten material is formed between oxide and metal that leaves the node, and thereafter, at that node, report of the thickness of metal remaining when the node reaches 3600°F (3600°F ensures melting of the

alpha Zircaloy whether or not the eutectic with the Zircaloy oxide is formed); and

- assumption of the time as zero at the time the top of the fuel stack is first uncovered.

The code has been used to calculate the time–temperature relationship for the fuel rods using the following set of parametric values:

- boil down to 7, 8, or 9 feet from the top of the fuel stack;
- a time of boil off of 20 minutes for most scenarios, but 30 or 33 minutes for certain scenarios;
- radial peaking factors in the assemblies of 1.467, 1.2, 1.0, and 0.622 (a spread reasonably representative of the core). Power in the assemblies at each node is obtained by multiplying the radial peaking factor (rpf) by the axial power profile value at each node;
- conduction heat transfer coefficients over a range of representative low steam flow rates (3 and 10); and
- the boil down and refill scenario proposed by EPRI in NSAC-1.

Results

The principal results are presented in summary form in App. Tables II-8 and II-9, and in App. Figures II-3 to II-19. The effects of varying the parameters can be seen in App. Tables II-8 and II-9 on the time and location of bursting of the fuel rods at 1500°F (assuming that bursting occurs at 1500°F), the time and location of the first formation of the Zr-U-O liquid phase (assumed to have formed at 3600°F) and of the maximum depth of formation from the top of the fuel stack, and of the time and location of the maximum temperature reached in the fuel rod. The App. Figures II-3 to II-19 show the time–temperature curves for 1-foot nodes on the fuel rods over a time interval of 80 minutes.

Because the time zero for the TMIBOIL calculation is the time at which the top of the fuel stack was first uncovered, the time scale can be moved along the clock-time axis (or accident time axis) as needed to examine the effects of modifying an accident scenario.

Discussion

In general overview of the TMIBOIL calculational results, and the known "facts" of the TMI-2 accident sequence, it is believed that boil off of 7 feet produces too little damage (considering the amount of hydrogen produced and the amount of core inventory of radioactivity released), and the boil off to 9 feet produces too much. It appears that the boil off to 8 feet ± ½ foot produces damage values not inconsistent with known levels such as hydrogen, radioactivity release, and maximum temperatures.

App. Tables II-8 and II-9 present most of the same data in somewhat different order, so that comparisons of several parameters are made easier.

In App. Table II-8, the effect of changing the power in the assembly on the significant points can be seen by comparing lines 1 through 4. As the power in the assembly increases, the location of the burst (defined as the first position on the rod to reach 1500°F) can be seen to rise toward the top of the fuel rod, and the time to burst decreases from 29 to 20.6 minutes. Also, the location of the first formation of liquid phase (3600°F) rises, and the time to formation decreases. It may seem surprising that the maximum depth of liquid phase formation decreases with increasing power in the assembly, but this is due to the increasing rate of steam production with increased power.

The effects of changing the maximum depth of boil off can be seen by comparison of lines 1–4 with 8–11 and 12–15. At the 7-foot level of boil off, the peak temperature on the fuel rod increases from 3042°F to 3600°F, with decreasing assembly power from rpf = 1.467 to rpf = 0.622, and only the lowest power assemblies on the periphery of the core reach temperatures high enough to form the Zr-U-O "liquefied fuel" phase.

The ranges of time before the burst and the location of the burst for the different levels or rates of boil down vary about 4 inches of range of level across the core, with differences of 7–10 minutes between first and last bursts for each of the boil down levels. Changes in most of the parameters do not have a large effect on time versus temperature, or on burst time and elevation. The largest effects are observed in the influence of level of boil down on the first and maximum levels of liquefaction and on the peak temperature reached. The calculations for the 9-foot level of boil down (from the top of the core) indicate that more than three-fourths of the core had exceeded temperatures of 5200°F (melting point of UO) for a depth of about 2 feet at an elapsed time of 78 minutes from the start of the uncovering of the core.

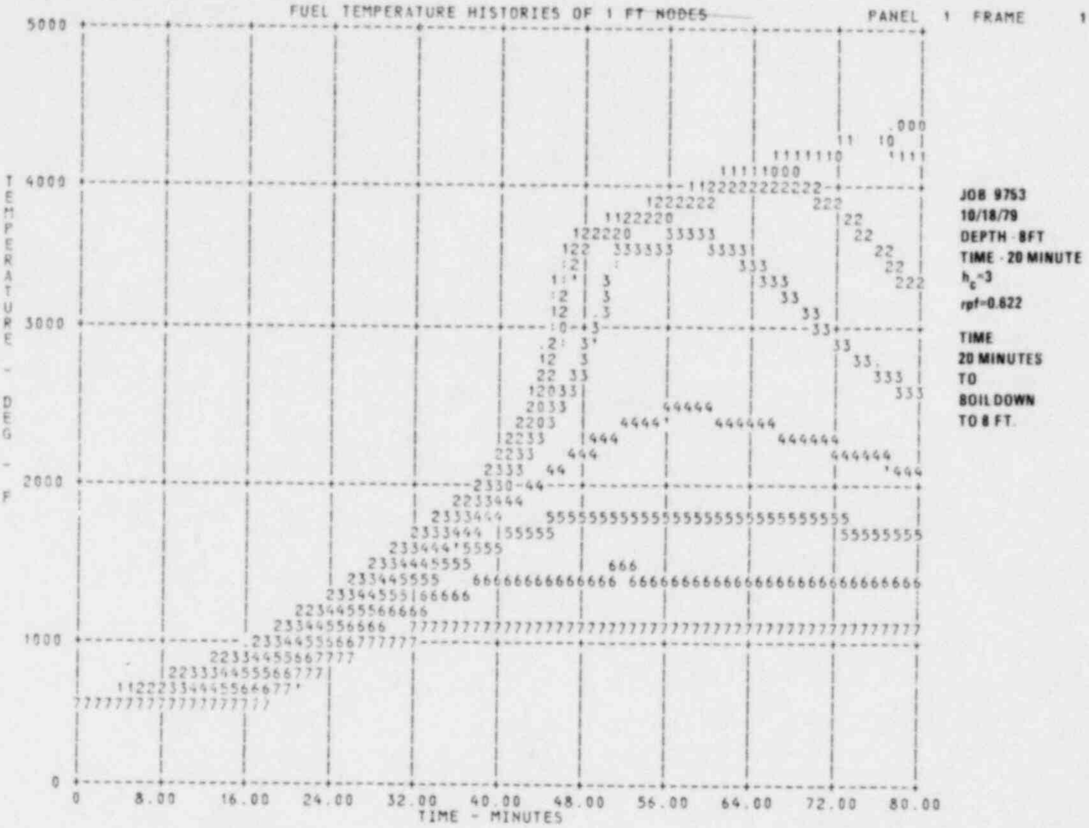
The estimate of damage present in the core at 3 hours depends on the time assumed for the first uncovering of the core. The best evidence available for determining this time is shown in Color Plate 5, where the temperatures of the hot and cold legs of the two OTSGs and the levels of coolant on the

APP. TABLE II-8. TMIBOIL calculations on core damage at 3 hours

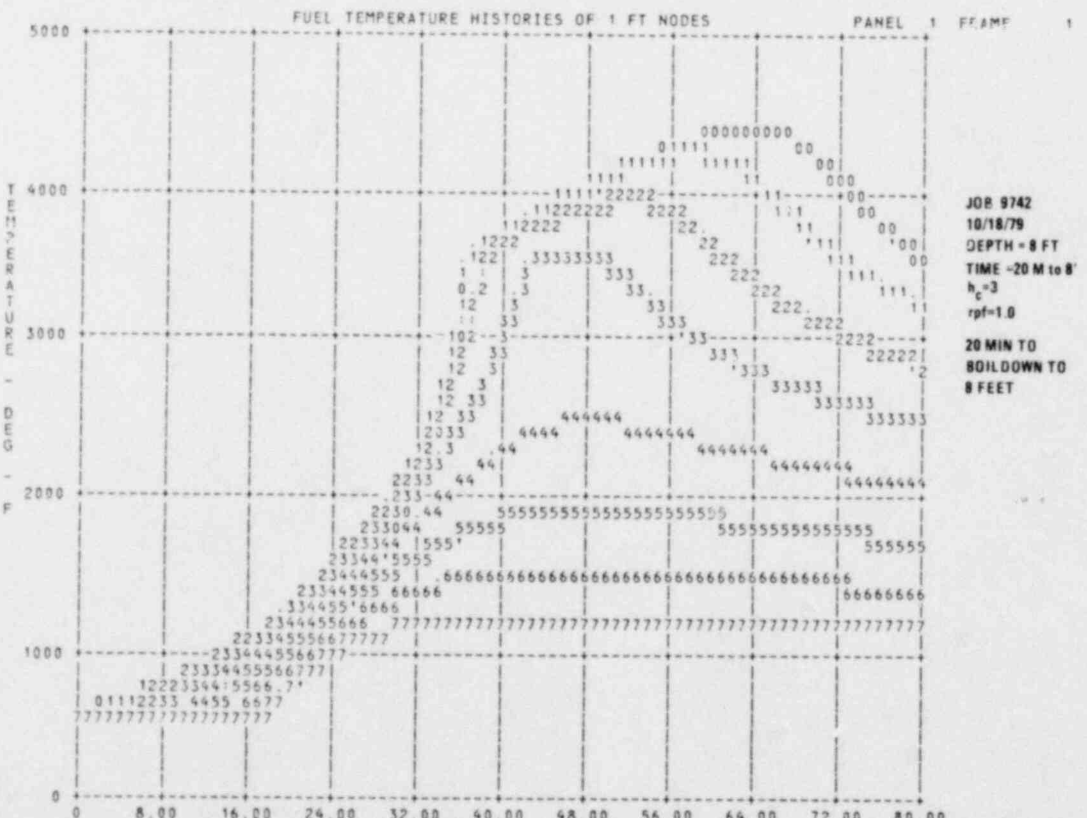
Boiloff				1500°F Burst		Liquefaction First		Liquefaction Maximum		Peak Temperature			
Depth (ft)	Time (min)	Power (Rpf)	h_c	Depth (in)	Time (min)	Depth (in)	Time (min)	Depth (in)	Time (min)	°F	Depth (in)	Time (min)	Comments
8	20	0.622	3	22	29	14	46.2	41	57	4358	1	77.5	
8	20	1.0	3	20	23.2	10	36.5	39	48	4410	1	62.5	
8	20	1.2	3	19	21.5	7	33.8	37	42	4412	1	57.5	
8	20	1.467	3	18	20.6	6	31.3	36	38	4370	1	52.5	
8	20	1.467	10	17	21.5	9	31.6	36	38	4362	1	50	
8	33	1.467	3	13	30	3	44	35	52	4280	1	61	Without cold rod
8	33	1.467	3	16	31.9	4	48	31	56	4195	1	70	With Cold rod
7	20	0.622	3	16	32.2	1	55.0	2	55.4	3600	2	55.4	Steam flow at peak Temp. 1.02 lb/h
7	20	1.0	3	15	26.8	–	–	–	–	3549	1	50.0	Steam flow at peak Temp. 1.60 lb/h
7	20	1.2	3	15	24.1	–	–	–	–	3265	1	45.0	Steam flow at peak Temp. 1.89 lb/h
7	20	1.467	3	16	22	–	–	–	–	3042	1	40.0	Steam flow at peak Temp. 2.30 lb/h
9	20	0.622	3	25	27.0	24	43.8	74	56.1	4796	29	77.5	Temp. still increasing at 77.5 min
9	20	1.0	3	22	21.5	18	33.6	72	45	5590	20	77.5	Temp. still increasing at 77.5 min
9	20	1.2	3	22	20.5	17	30.1	72	42.5	5892	15	77.5	Temp. still increasing at 77.5 min
9	20	1.467	3	21	20	17	28.5	71	39	6194	9	78	Temp. still increasing at 78 min
9	30	1.467	3	16	24.7	16	36.5	70	48	5444	1	70	EPRI NSAC-1
8	33	0.622	3	22	38	4	58	36	68	4200	1	80	

APP. TABLE II-9. TMIBOIL calculations on core damage at 3 hours

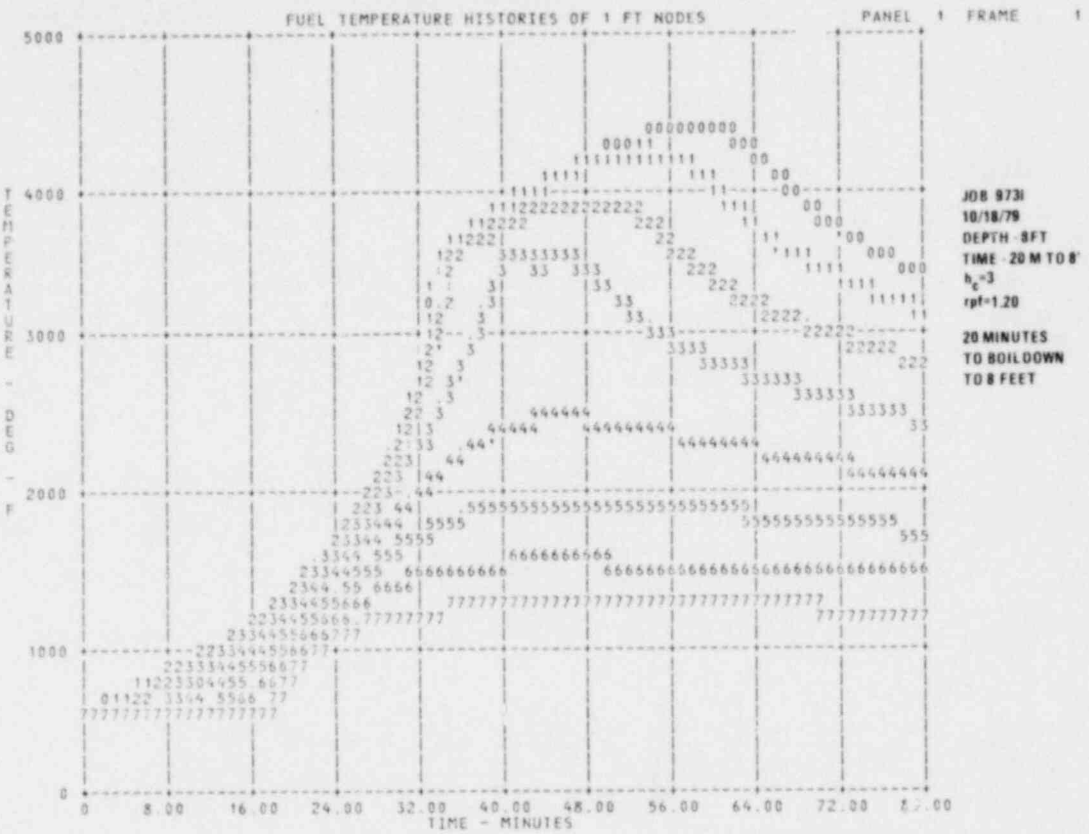
Boiloff				1500°F Burst		Liquefaction First		Liquefaction Maximum		Peak Temperature			
Depth (ft)	Time (min)	Power (Rpf)	h_c	Depth (in)	Time (min)	Depth (in)	Time (min)	Depth (in)	Time (min)	°F	Depth (in)	Time (min)	Comments
7	20	1.467	3	16	22	–	–	–	–	3042	1	40.0	
8	20	1.467	3	18	20.6	6	31.3	36	38	4370	1	52.5	
9	20	1.467	3	21	20	17	28.5	71	39	6194	9	78	
9	30	1.467	3	16	24.7	16	36.5	70	48	5444	1	70	EPRI NSAC-1
7	20	1.2	3	15	24.1	–	–	–	–	3265	1	45.0	
8	20	1.2	3	19	21.5	7	33.8	37	42	4412	1	57.5	
9	20	1.2	3	22	20.5	17	30.1	72	42.5	5892	15	77.5	
7	20	1.0	3	15	26.8	–	–	–	–	3549	1	50.0	
8	20	1.0	3	20	23.2	10	36.5	39	48	4410	1	62.5	
9	20	1.0	3	22	21.5	18	33.6	72	45	5596	20	77.5	
7	20	0.622	3	16	32.3	1	55.0	2	55.4	3600	2	55.4	
8	20	0.622	3	22	29	14	46.2	41	57	4358	1	77.5	
9	20	0.622	3	25	27.0	24	43.8	74	56.1	4796	29	77.5	



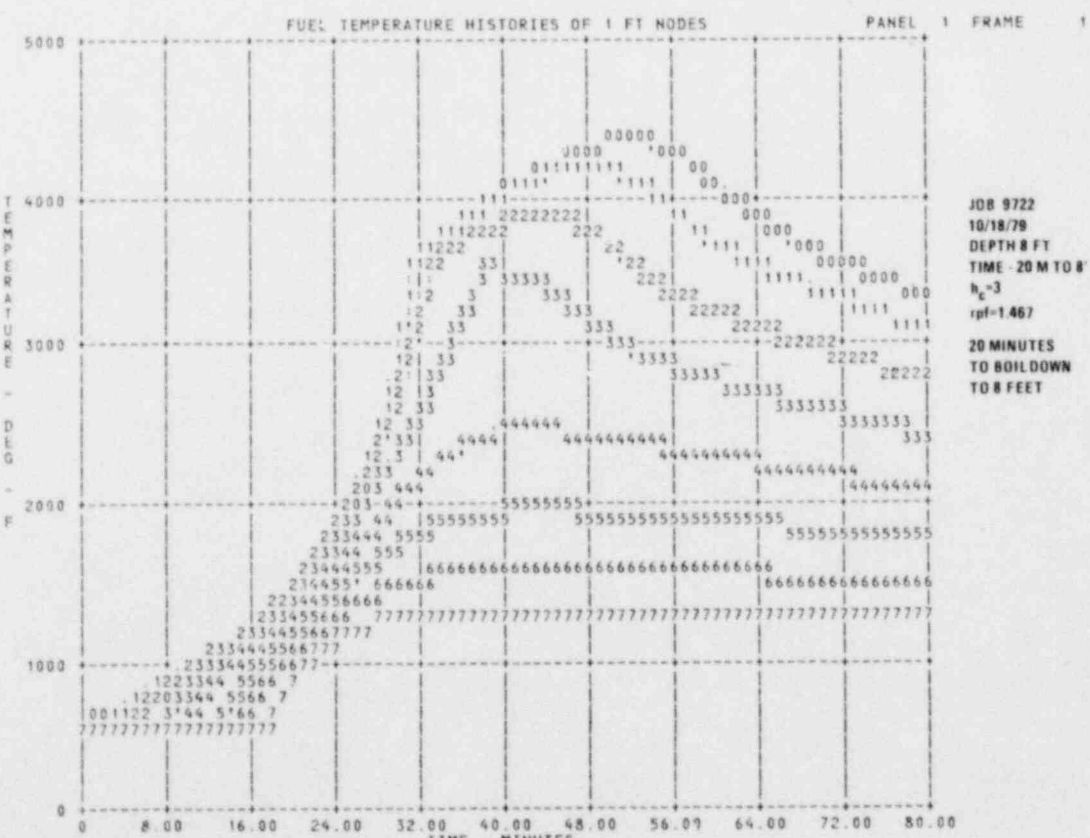
APP. FIGURE II-3. Fuel Temperature Histories



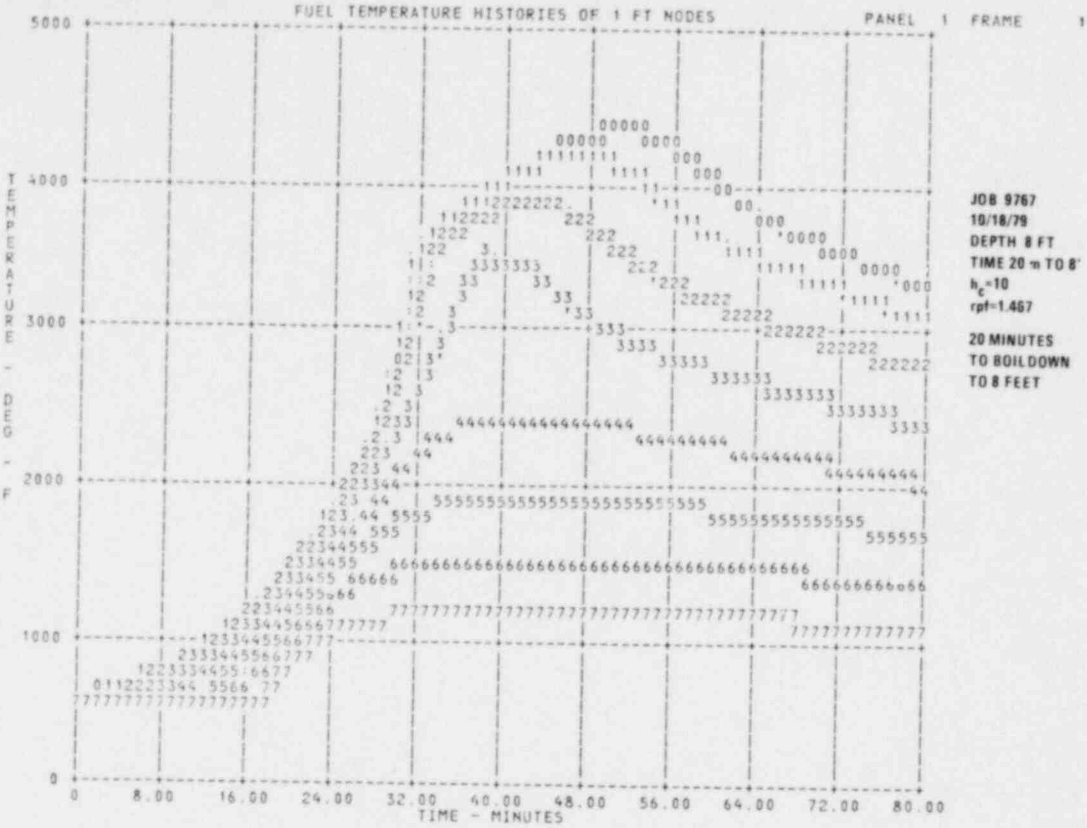
APP. FIGURE II-4. Fuel Temperature Histories



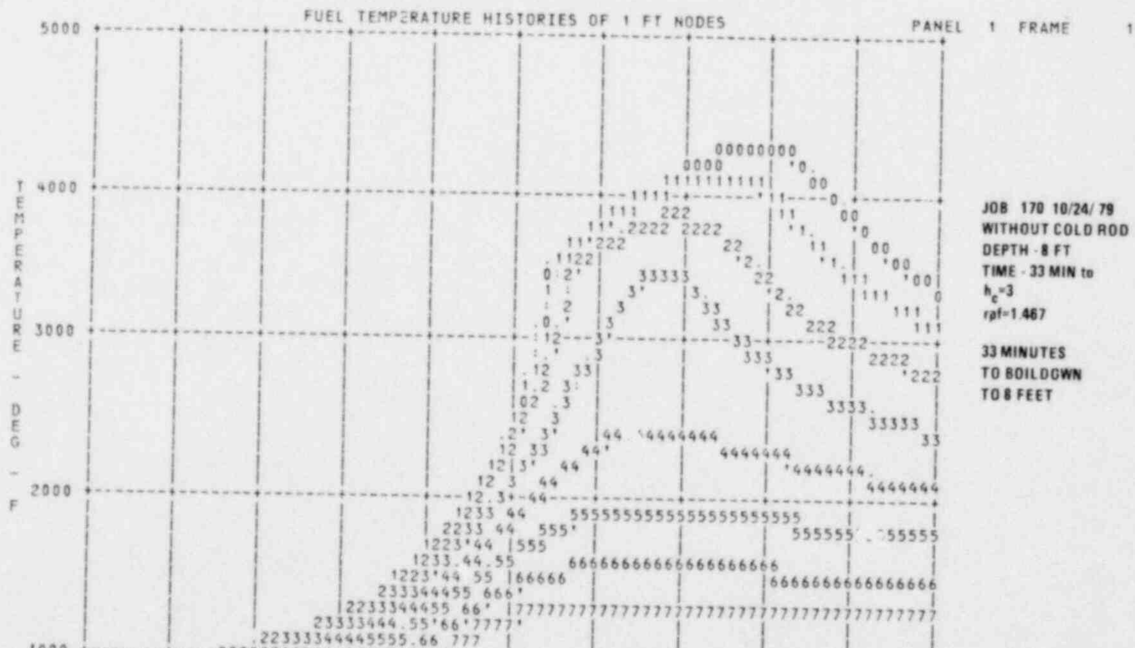
APP. FIGURE II-5. Fuel Temperature Histories



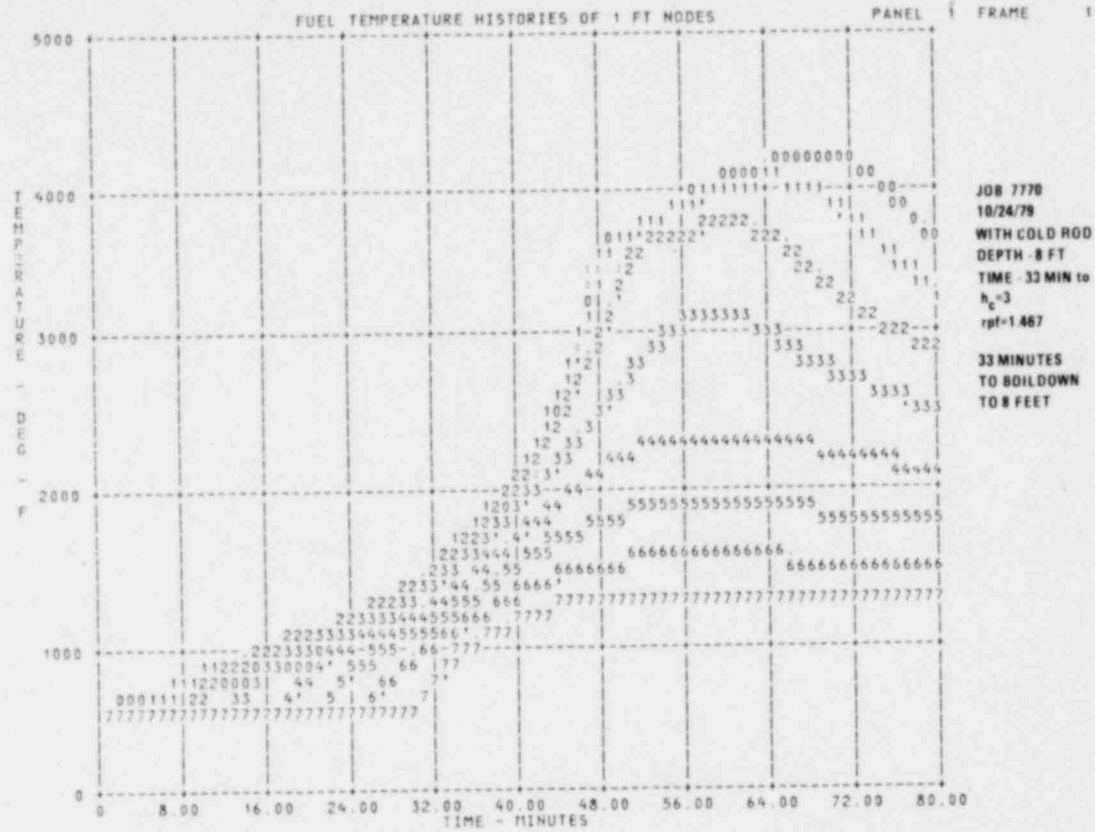
APP. FIGURE II-6. Fuel Temperature Histories



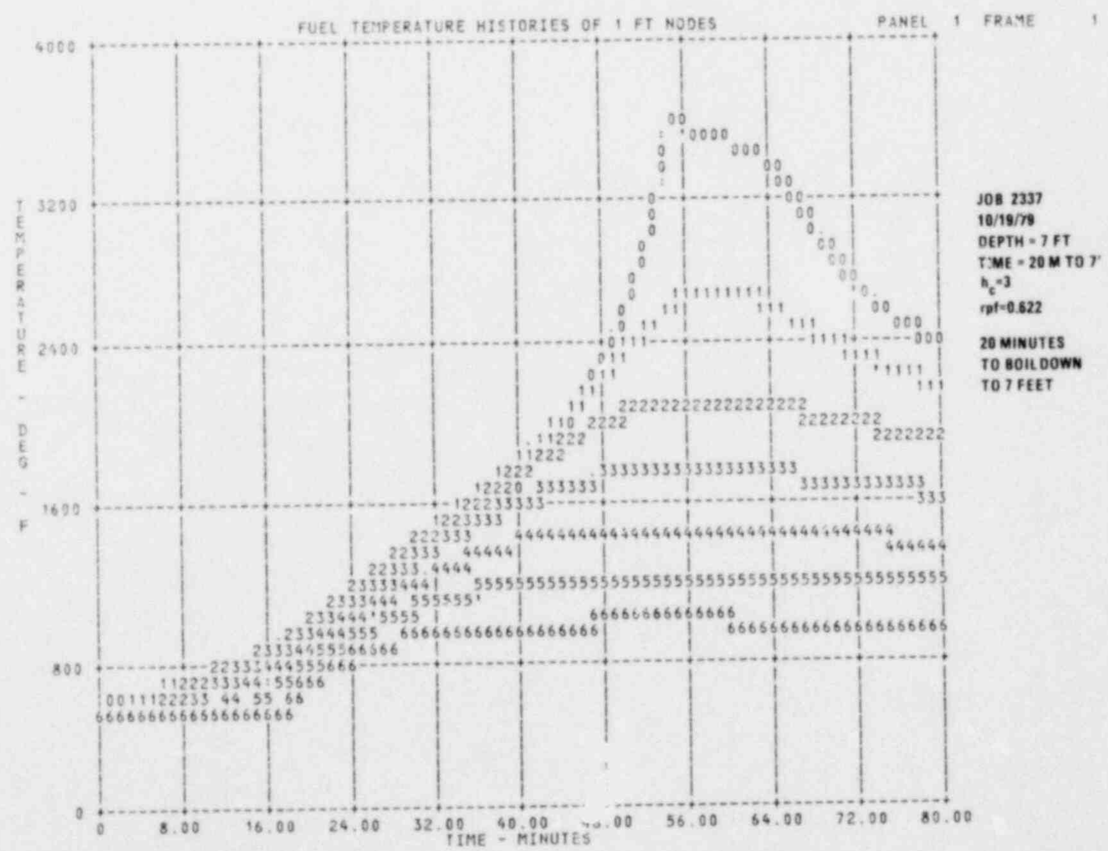
APP. FIGURE II-7. Fuel Temperature Histories



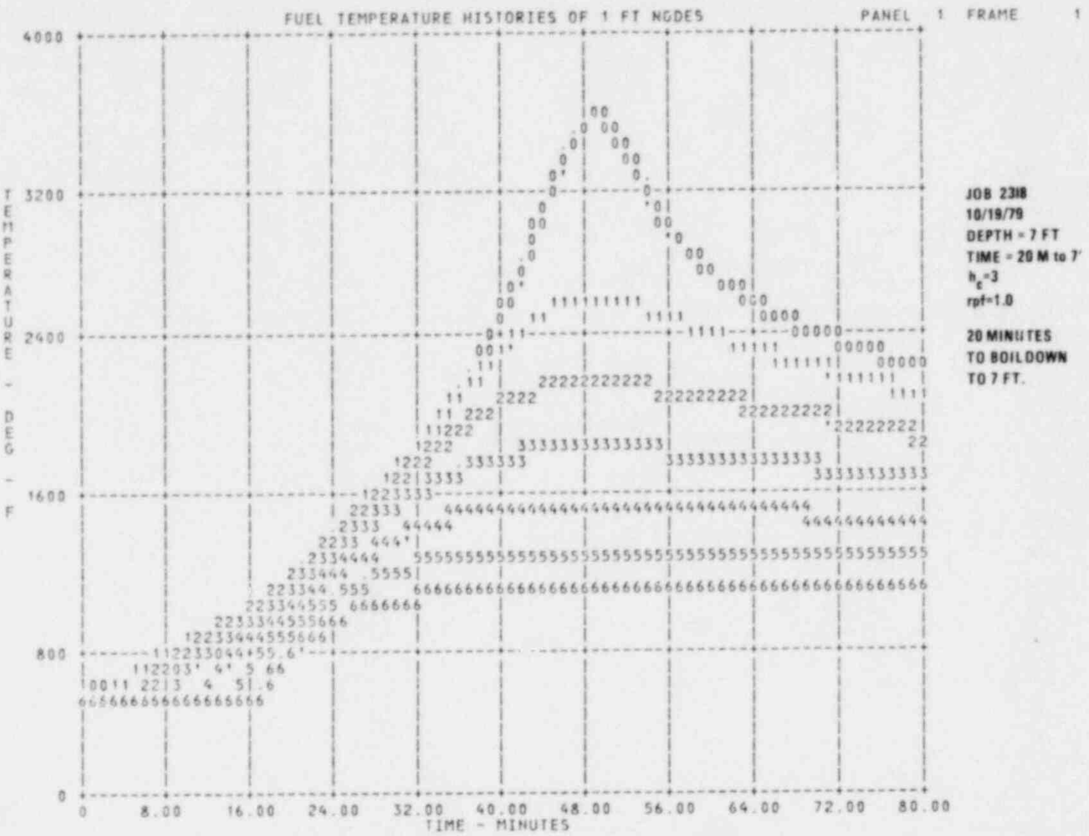
APP. FIGURE II-8. Fuel Temperature Histories



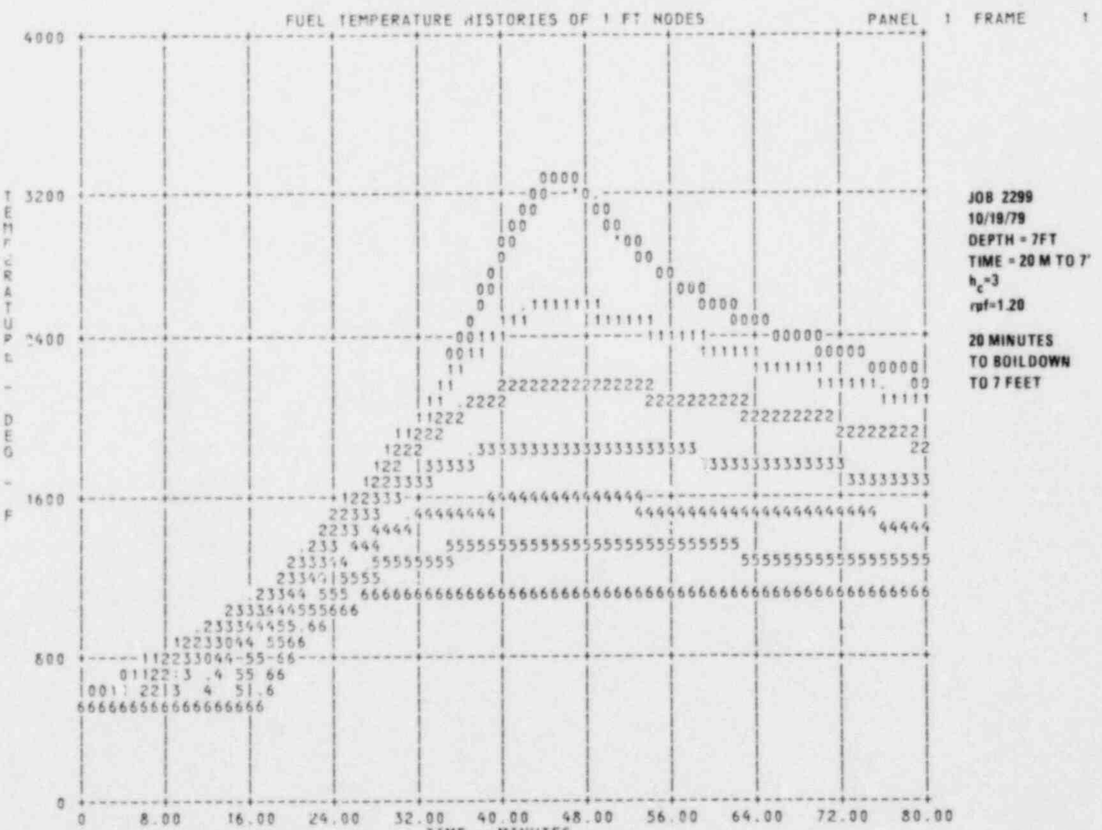
APP. FIGURE II-9. Fuel Temperature Histories



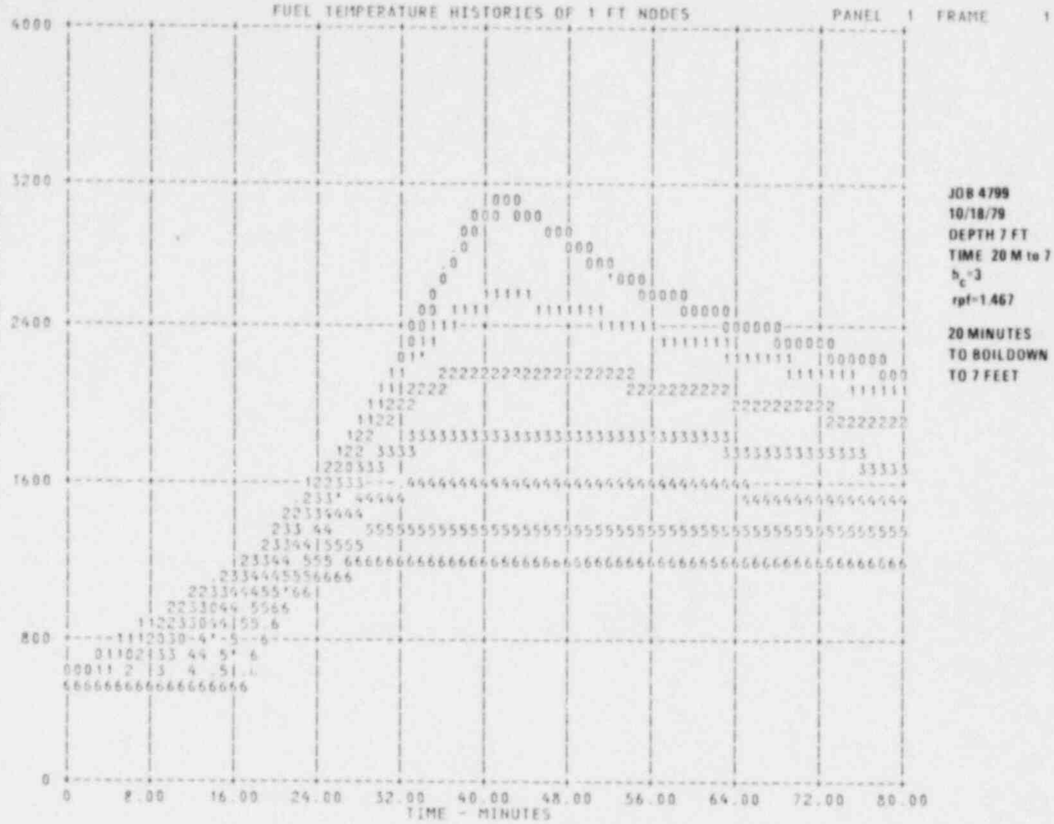
APP. FIGURE II-10. Fuel Temperature Histories



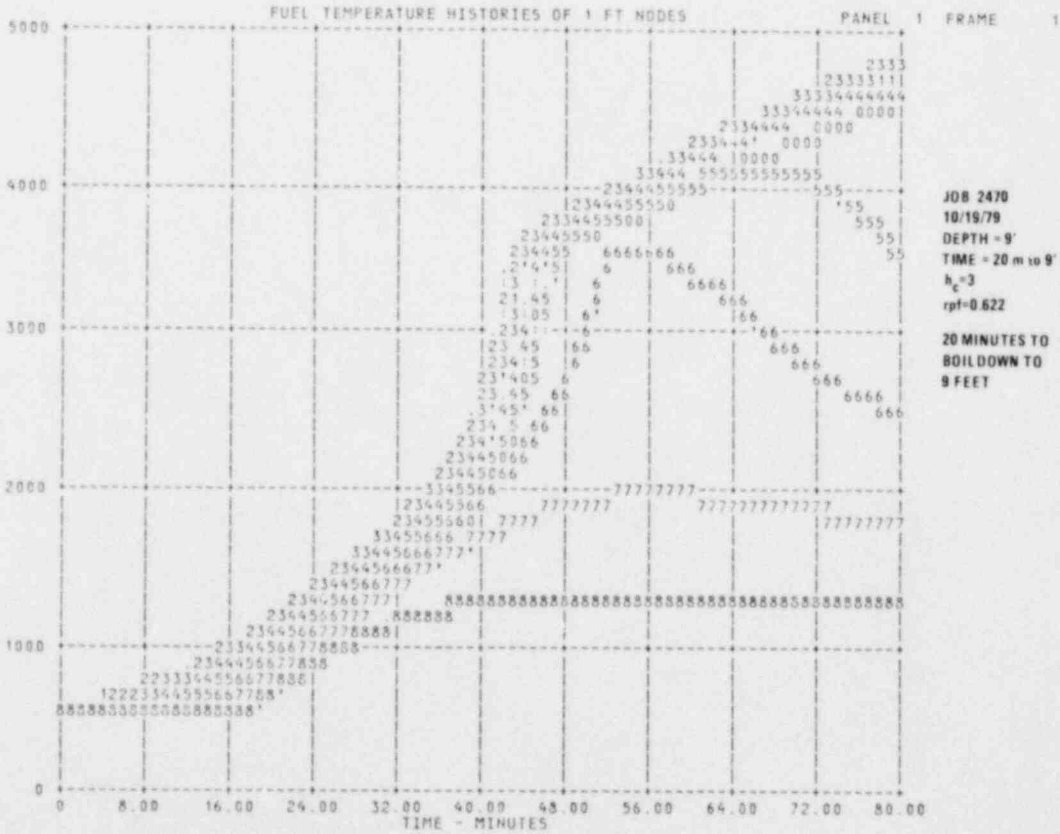
APP. FIGURE II-11. Fuel Temperature Histories



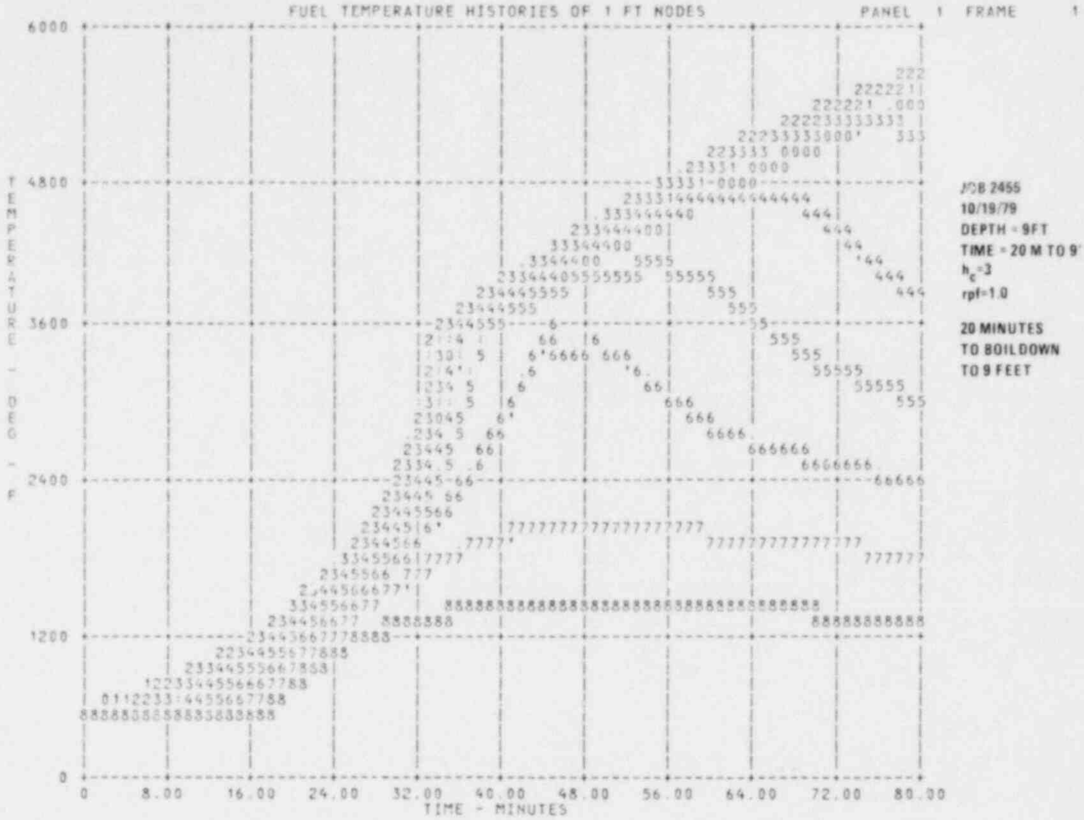
APP. FIGURE II-12. Fuel Temperature Histories



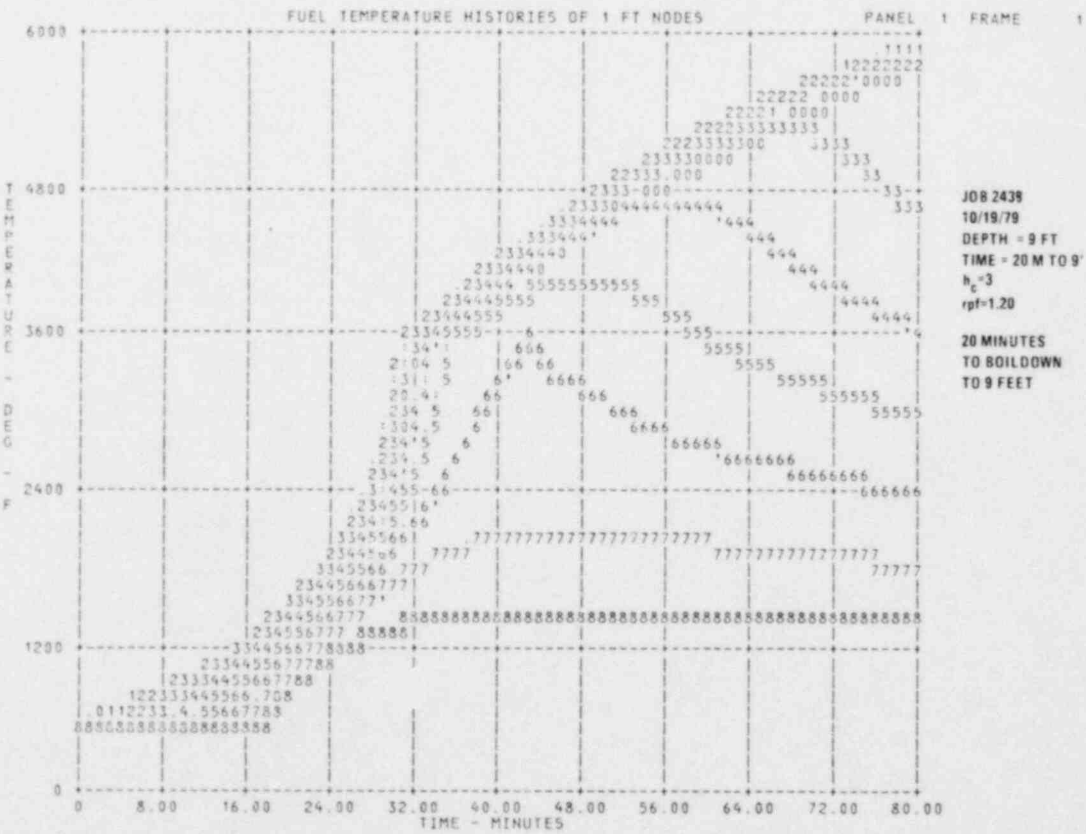
APP. FIGURE II-13. Fuel Temperature Histories



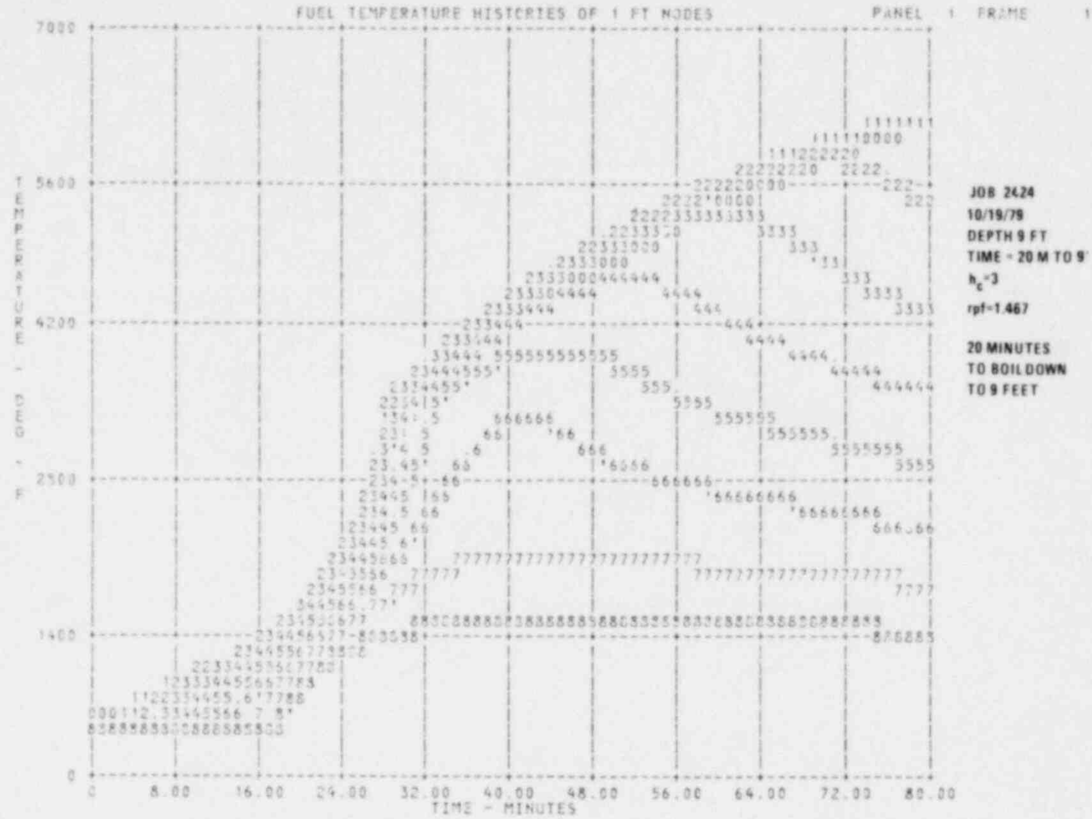
APP. FIGURE II-14. Fuel Temperature Histories



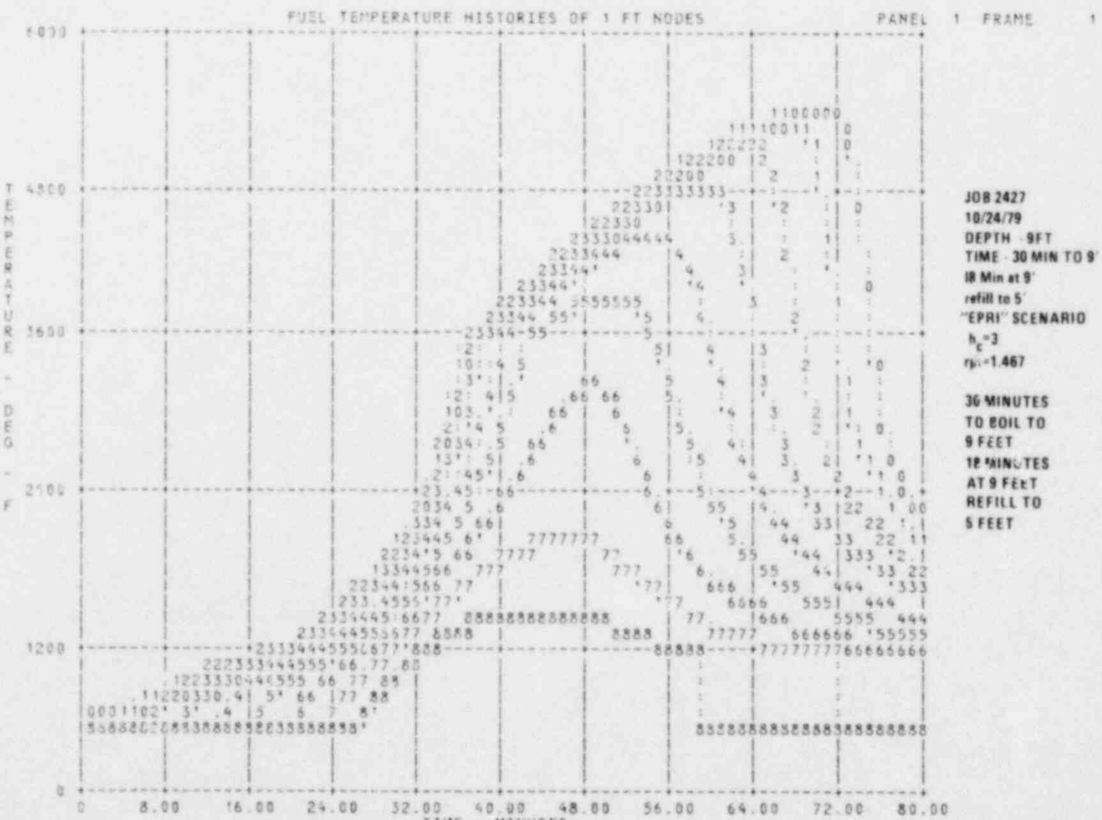
APP. FIGURE II-15. Fuel Temperature Histories



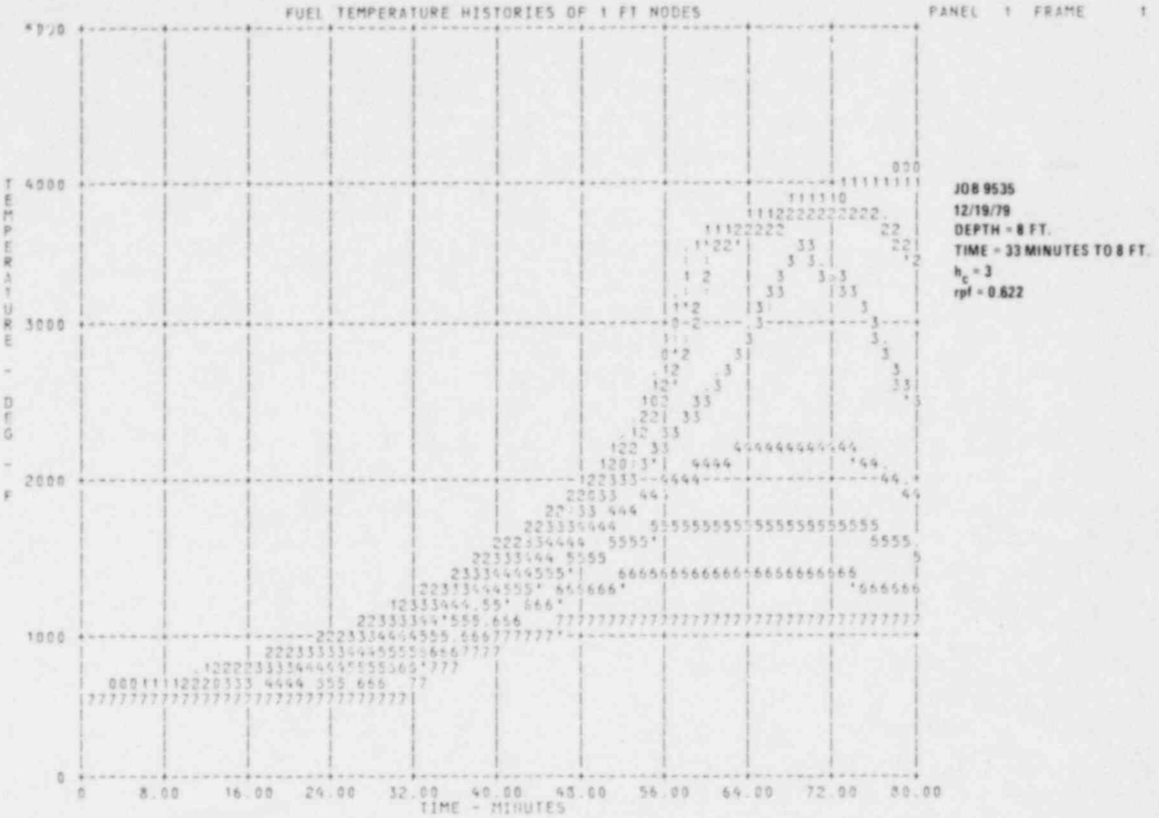
APP. FIGURE II-16. Fuel Temperature Histories



APP. FIGURE II-17. Fuel Temperature Histories



APP. FIGURE II-18. Fuel Temperature Histories



APP. FIGURE II-19. Fuel Temperature Histories

secondary side are plotted as functions of clock time.

There are two possible interpretations of these data. When the prior level in OTSG B is considered (shown in Color Plate 5), it can be argued that the first break in the curves for the hot-leg temperatures of both steam generators at 5:42 a.m. (1 hour 42 minutes of accident time) indicates that superheated steam was detected in both A and B steam generators at the top of the hot legs. The continued rise and subsequent decrease in temperature for OTSG B could indicate the flow of superheated steam into a condenser that was heat saturated. The reversion of OTSG A hot-leg temperature to a decreasing temperature-time relationship paralleling the previous curves and the succeeding curves for the cold legs could indicate that OTSG A could absorb no significant amount of heat (it was already known to have been "boiled dry") until its refilling had begun. Thus, it can be argued that the core was first uncovered at 102 minutes. It can be stated with certainty that the core had been uncovered no later than 5:52 a.m. (1 hour 52 minutes, or 112 minutes of accident time), because at that time the OTSG A hot-leg temperature began a rise that did not stop (other than for two short inversions) until a temperature of about 820°F was reached at 6:52 a.m. (2 hours 52 minutes, or 172 minutes accident time). These two times (102 and 112 minutes of accident time) allow placement of the TMIBOIL zero time and the time at which the RC-P2B pump was started on the time-temperature elevation plots, so that bounds for the amount of damage to the core at 3 hours can be estimated. It must be assumed that at least a small amount of water was pumped by RC-P2B into the core to reverse the heatup of the fuel rods, even if for only a few minutes.

If it is then assumed that the TMIBOIL calculations for boil off to 8 feet in 33 minutes apply (the best estimate from such variables as the amount of hydrogen and radioactivity released, the SRM data, and the first detection of radioactivity in the primary system at 6:25 a.m.), the PORV block valve was closed at 6:20 a.m. (2 hours 20 minutes accident time), and the RC-P2B was started at 6:54 a.m. (2 hours 54 minutes accident time), then the amount of core damage at 7:00 a.m. (3 hours accident time) can be bounded.

With these assumptions, it can be estimated that (1) the great majority of the fuel rods burst at about the time the block valve was closed at 140 minutes and all of the rods burst within the next 10 minutes; (2) the first "liquefied fuel" formation occurred at about 10 minutes after the block valve was closed; (3) the maximum depth of formation of "liquefied fuel" in the hot assembly occurred about 20 minutes after the block valve was closed and about 10 minutes after that time in the lowest power assembly; and (4) the maximum temperature reached in the fuel rods was about 4000°F for a "middle power" assembly at about 72 minutes after the block valve was closed, and at the time the RC-P2B was started. Additionally, peak temperatures of about 4300°F or more were reached in more than two-thirds of the core by the time the RC-P2B was started. The maximum penetration of the formation of "liquefied fuel" was to about 41 inches in the lowest powered assemblies on the periphery of the core and to 36 inches in the center of the core. (The steam production rates decreased greatly as the periphery of the core was approached, and thus the cooling capability of the steam flow also decreased.)

Additionally, it is estimated that the amount of Zircaloy converted to oxide as a result of the events to 7:00 a.m. (3 hours accident time) is between 32 and 39% of the Zircaloy in the fueled part of the core, and between 26 and 31% of the total Zircaloy in the core, including plenum regions and end plugs. This estimate includes complete oxidation of the Zircaloy contained in the "liquefied fuel." These amounts are equivalent to 300-pound moles and 360-pound moles of hydrogen, respectively. Because there is evidence that more hydrogen may have been produced at a later time, this is not to be taken as an estimate of the amount of hydrogen present in the containment and the primary system at 1:54 p.m. (9.9 hours accident time), the time of the "hydrogen burn" in the containment.

The Zircaloy cladding was embrittled by oxidation down to at least 4 feet from the top of the fuel in the fuel rods. A considerable amount of "liquefied fuel" had formed and flowed down between oxidized cladding shells, and would have frozen upon reaching a lower temperature at a lower level. When the reactor coolant pump was turned on at 2 hours 54 minutes, the embrittled cladding would have been thermally shocked by the influx of coolant (whether steam or water) and would have shattered to produce a "rubble" or "debris" bed of cladding fragments, Zircaloy oxide shells, fuel pellets, and "liquefied fuel," supported by fuel rod stubs, unmelted grid spacers, and intact guide and instrumentation tubes. A significant part of the "debris" bed would have been "melted" or "glued" together with "liquefied fuel" that would have frozen after flowing from a higher position and temperature.

REFERENCES AND NOTES

Memorandum from G. P. Marino and J. M. Marks, NRC, to File, "Calculation of Fuel Rod Temperatures Reached in the Three Mile Island-2 Incident," October 25, 1979.

APPENDIX II. 9

HYDROGEN CALCULATIONS

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The Hydrogen Bubble

During the period from 29 March through at least 8 April, measurements were made to determine the size of the (assumed) hydrogen bubble in the TMI-2 reactor coolant system (RCS). The procedure was to define a mass balance for the RCS (exclusive of pressurizer) by recording changes in the levels of the pressurizer (PZR) and makeup tank (MUT). From this, the change in bubble size (and therefore the apparent compressibility) could be calculated and the bubble size inferred.

We have independently derived a "bubble formula," compared it with the Met Ed and B&W formulae, and used it to reduce the raw data in the "Bubble Book." The total hydrogen content of the RCS, including hydrogen in solution, was then fit by a straight line (as a function of time) to estimate the average removal rate.

The Bubble Formula

We assume that the bubble is a mixture of hydrogen gas and water vapor. Because of the low temperature and water vapor pressure (280°F, 50 psia), Dalton's Law should apply. The change in water vapor pressure due to the partial pressure of hydrogen*

$$P_v - P_{sat}(T) \simeq (P - P_{sat}(T))\, \rho_{v(sat)}\, \rho_{\ell(sat)} \qquad (1)$$

is negligibly small—about 2 psi at a total pressure of 1000 psia. The partial density of water vapor in the bubble is essentially the saturation density $\rho_{v(sat)}(T)$.

The total water mass in the RCS is given by

$$M_W^{RCS} = (V^{RCS} - V^B)\, \rho_\ell\, (P^{RCS}, T^{RCS}) + V^B\, \rho_{v(sat)}\, (T^{RCS}) \qquad (2)$$

If Eq 2 is evaluated for two (P,T) states and the results subtracted, one finds

$$(\rho_\ell - \rho_v)_2^{RCS}\, V_2^B - (\rho_\ell - \rho_v)_1^{RCS}\, V_1^B = (V\rho_\ell - M_w)_2^{RCS} - (V\rho_\ell - M_w)_1^{RCS} \qquad (3)$$

With the general notation (f is any quantity)

$$\bar{f} \equiv (f_1 + f_2)/2 \qquad (4)$$

$$\Delta f \equiv f_2 - f_1 \qquad (5)$$

Eq 3 becomes

$$(\bar{\rho}_\ell - \bar{\rho}_v)\Delta V^B + \bar{V}^B\, \Delta(\rho_\ell - \rho_v)^{RCS} = \bar{V}^{RCS}\, \Delta\rho_\ell^{RCS} + \bar{\rho}_\ell^{RCS}\, \Delta V^{RCS} - \Delta M_w^{RCS} \qquad (6)$$

*P. M. Morese, *Thermal Physics,* W. A. Benjamin Inc., 1969, p. 124.

This equation (which contains no approximations) may be simplified by elimination of several small terms.* At 1000 psia and 280°F, $\rho_\ell \simeq 58\ lb_m/ft^3$ and $\rho_{vsat} \simeq 0.12\ lb_m/ft^3$. Also, $V^{RCS} \simeq 10\,300\ ft^3$ and (we will find) V^B is typically several hundred cubic feet while ΔM_w^{RCS} is typically a few thousand pounds for a pressure change of 100 psi. The expansion of a cylindrical vessel is given by

$$\Delta V \simeq 2 \frac{R}{t} \frac{\Delta P}{E} V \qquad (7)$$

where R/t is the ratio of radius to thickness, about 8 for the RCS, and E is Young's modulus, roughly 3×10^7 psi for steel. Thus $\bar{\rho}_\ell \Delta V^{RCS}$ is typically 30 lb_m, a 1% effect. Temperature changes were almost never greater than 1°F. Therefore, from steam tables, we find $\Delta\rho_\ell \gtrsim .05\ lb_m/ft^3$ and $\Delta\rho_v \gtrsim .002\ lb_m/ft^3$. Thus $\bar{V}^B \Delta(\rho_\ell - \rho_v)^{RCS}$ is a few tens of pounds, again a 1% effect, while $V^{RCS} \Delta\rho_\ell^{RCS}$ may be several hundred pounds and is more significant. Finally, neglecting ρ_v compared to ρ_ℓ we find

$$\Delta V^B \simeq (V^{RCS} \Delta\rho_\ell^{RCS} - \Delta M_w^{RCS})/\bar{\rho}_\ell^{RCS} \qquad (8)$$

accurate to a few percent.

The mass increase in the RCS is simply the net mass loss from the makeup tank and pressurizer, reduced by net leakage. Because the pressure and temperature in the makeup tank are essentially constant, the change in its density may be neglected, but this is not the case for the pressurizer. The resulting expression is

$$\Delta M_W^{RCS} \simeq -(\rho_\ell A \Delta L)^{MUT} - \Delta(\rho_\ell V_\ell + \rho_v V_v)^{PZR} - \Delta M_v^{LEAK} \qquad (9)$$

Here

$$V_\ell^{PZR} = \left\{A(L+L_o)\right\}^{PZR} \qquad (10)$$

$$V_v^{PZR} = V^{PZR} - V_\ell^{PZR} \qquad (11)$$

where the A's are horizontal cross-sectional areas, L's are levels, L_o is the effective height of the hemispherical section of the pressurizer below the lower sensing nozzle (2/3 the radius or 28 in.) and ΔM_w^{leak} is the unknown leakage term.

The hydrogen content of the bubble is simply

$$M_H^B = P_H V^B / RT \qquad (12)$$

*The reader may skip to Eq 8, an obvious approximation, and miss only a discussion of the accuracy of that approximation.

moles where R is the gas constant and

$$P_H \equiv P - P_{sat}(T) \qquad (13)$$

is the partial pressure of hydrogen. The solubility of hydrogen is proportional to its partial pressure and (neglecting the compressibility of the water) is given by $S'(T)P_H$ moles per unit volume. The total hydrogen content of the RCS (exclusive of the pressurizer) is given by

$$M_H^{RCS} = M_H^B + S'(T)P_H(V^{RCS} - V^B) \qquad (14)$$

assuming that a bubble is present. If M_H^B is eliminated from Eq 14 by using Eq 12, the resulting equation is

$$V^B = \max\left\{\frac{M_H^{RCS}\ RT}{(1-S'RT)P_H} - \frac{S'RT}{1-S'RT} V^{RCS}, 0\right\} \qquad (15)$$

where the dimensionless quantity

$$S'RT \equiv S'P_H/(P_H/RT) \qquad (16)$$

is simply the ratio of the volumetric concentration of hydrogen in solution to that in the vapor phase, and is approximately 0.03 at 280°F.* Equation 15 explicitly includes the possibility that all hydrogen is in solution with no bubble present.

If measurements are made at two pressures but nearly equal temperatures so that changes in S' and T may be neglected, Eq 15 may be used to show that**

$$M_H^{RCS} = \frac{1-S'RT}{RT} \max\left\{-P_{H2}P_{H1} \frac{\Delta V^B}{\Delta P_H}, P_{H\min} \left(|\Delta V^B| + \frac{S'RT}{1-S'RT} V^{RCS}\right)\right\} \qquad (17)$$

where the Δ notation of Eq 5 has been used and $P_{H\min}$ is the lesser. Once the hydrogen content of the system has been calculated from Eq 17, Eq 15 may be used to calculate the size of the bubble at any pressure. If a bubble is present in both states 1 and 2, the first term in brackets in Eq 17 is the greater, and the bubble volume is given by

$$V^B = -\frac{P_{H2}P_{H1}}{P_H} \frac{\Delta V^B}{\Delta P_H} - \frac{S'RT}{1-S'RT} V^{RCS} \quad \text{(Bubble at } P_{H1}, P_{H2}, P_H) \qquad (18)$$

*Calculated from H. A. Pray *et al.* "Solubility of Hydrogen, Oxygen, Nitrogen, and Helium in Water," *Industrial and Engineering Chemistry 44* (5) 1146-1151, 1952.

**We assume that M_H^{RCS} is constant during the measurement. Changing hydrogen content could be included with minor changes in the following equations.

If a bubble is present only at the lower pressure, the second term in Eq 17 is the greater, and

$$V^B = \frac{P_{H\min}}{P_H} |\Delta V^B| + \frac{P_{H\min} - P_H}{P_H} \tag{19}$$

$$\frac{S'RT}{1-S'RT} V^{RCS} \text{ (Bubble at } P_{H\min}, P_H)$$

Finally, if no bubble is present in either state, Eq 17 yields an upper bound on hydrogen content given by the amount soluble at the lower pressure, and Eq 15 gives zero volume for any pressure about which we have knowledge. In practice, because of experimental errors, it may be difficult to distinguish the latter two cases, and it may be better to interpret M_H^{RCS} from Eq 17 as an upper bound whenever the second term dominates.

For the case where a bubble is present at both states 1 and 2, our bubble formula is given by Eqs 8, 9, and 18. The B&W formula is in close agreement, although matching the solubility terms takes a little work. The main differences are B&W's neglect of changes in vapor mass in the pressurizer, and of liquid below the lower sensing nozzle (L_o in Eq 10), which are offsetting effects of a few percent each. Also, they appear to use total pressure rather than hydrogen partial pressure in their equivalent of Eq 18, which would lead to a 5% underprediction of bubble size.

The Met Ed formula also uses total pressure, and further neglects *all* compressibility and thermal expansion terms for the water. The most important of these neglected terms is thermal expansion in the pressurizer under increasing saturation temperature, leading to a 10% underprediction of bubble size. The next largest term, $\Delta\rho_\ell^{RCS}$ is dominated by temperature changes and therefore not consistent in sign. Much more important is the neglect of the solubility of hydrogen. The effect is shown by the last term of Eq 18 to be a systematic 300 ft^3 overprediction of bubble volume ($S'RT \simeq .03$ and $V^{RCS} \simeq 10\,300\ ft^3$). While the previously mentioned errors tend toward underprediction, they are substantially smaller so that the overprediction through the neglect of solubility is the overriding effect. Finally, the data reduction in the Bubble Book contains a number of arithmetic and/or transcription errors.

Results

The bubble formula derived in the preceding section has been used to reduce the raw data presented in the Bubble Book. The RCS temperature and the makeup tank temperature were taken as 280°F and 80°F respectively, when these data were not recorded. We observed that the pressurizer temperature and system pressure were not consistent in that the pressure was consistently 50 psi lower than $P_{sat}(T^{PZR})$. This discrepancy, in the *wrong* direction to be explained by a partial pressure of hydrogen, could be due to a 5°F error in T^{PZR}. However, we also noted that for the earliest data sets pressure was reported from the wide range recorder as well as from the computer, and is typically 50 psi higher. This problem has not been resolved. Therefore, in those cases where T^{PZR} was reported as well as P, the pressurizer mass term was evaluated twice, first using P to determine saturation densities and then using T, and the average used. In no case was the difference significant.

The results in these calculations* are presented in Figures 1-3. The first shows total mass while the second and third given bubble volumes at the average system pressure of 1000 psia and at the Met Ed-established standard 875 psia. Also shown in these figures are generalized** least-squares fits to the data from 3/31/79 through 4/3/79 with an approximate one-standard-deviation confidence band on the fit, and a fit presented by B&W.***

We feel that the difference between the two fits may be due to B&W's apparent use of only 5 data points. The standard deviation of the points used in our fit from the fit line is ±12 kg, in reasonable agreement with the error bars shown by B&W, corresponding to a mass error of ±8 kg.

We find an average removal rate of 1.7 (±0.3) kg/hr. This corresponds to the complete degassing of 60 (±10) gpm of letdown flow. This is not meant to imply that all the hydrogen was removed through letdown, but merely to note that typical letdown rates are sufficient to remove most of it. For comparison, the B&W line implies a significantly larger removal rate of 3.0 kg/hr, again perhaps due to the small data set used.

Our fit suggests that the bubble was gone at 1000 psia at 1800 on 4/1/79 (±3 hrs). During this time period, the pressure was being cycled between 950 and 1050 psia. The bubble would be eliminated at the higher pressure about an hour earlier.

If the constant removal rate is used to estimate the hydrogen content of the RCS at 16 hours—a very questionable extrapolation of inaccurate data—one finds a total mass of 190 (±40) kg. At 1400 psia and 360°F, there would be roughly 45 kg in solution and an 1100 ft^3 bubble. Two or 3 hours later when the pressure had fallen to 1000 psia, and the temperature was also lower, we would estimate a bubble size of 1300 ft^3.

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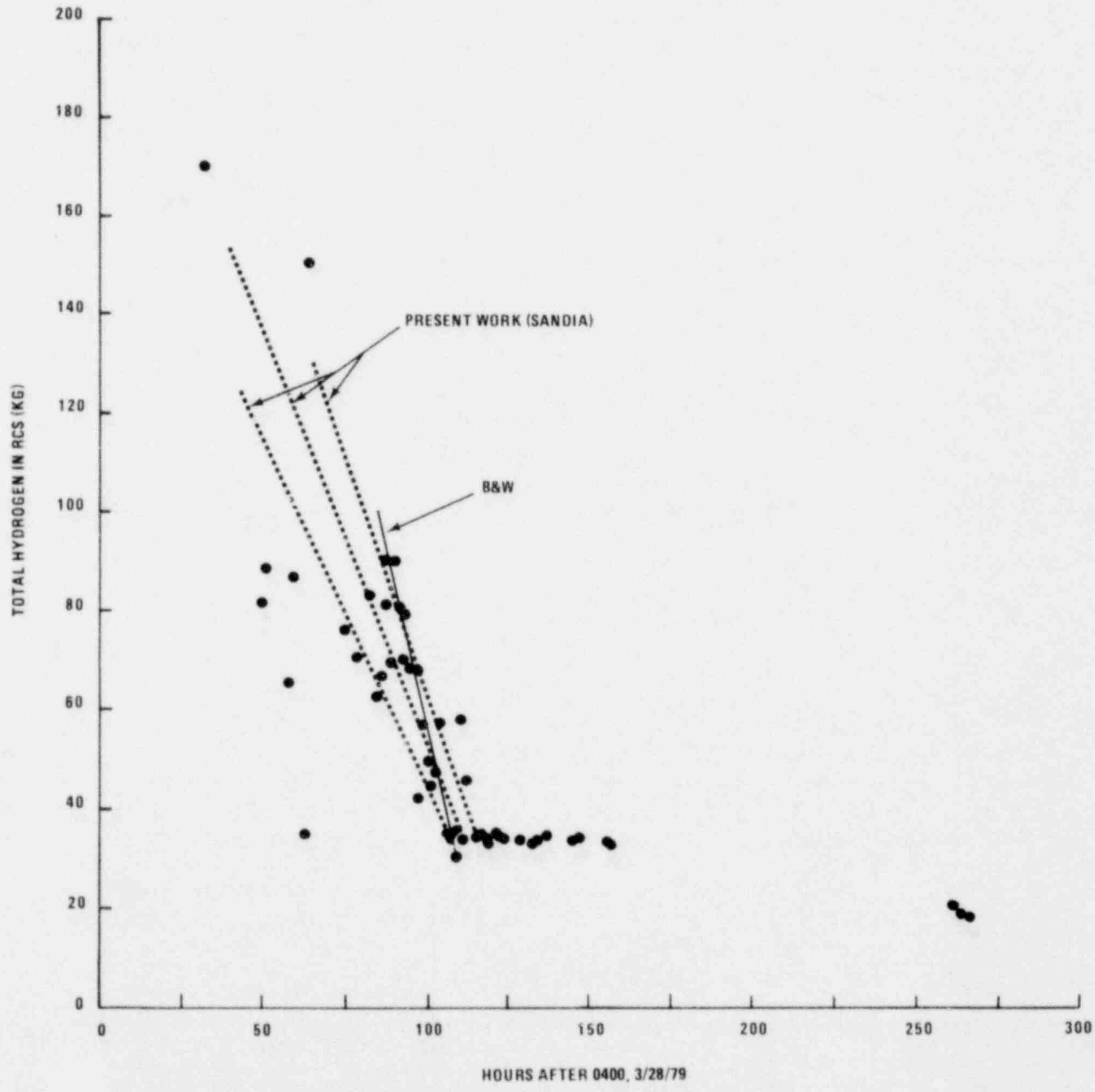
*The mass-leak term was taken as zero. We intend to repeat the calculations using the value mentioned later in the text.

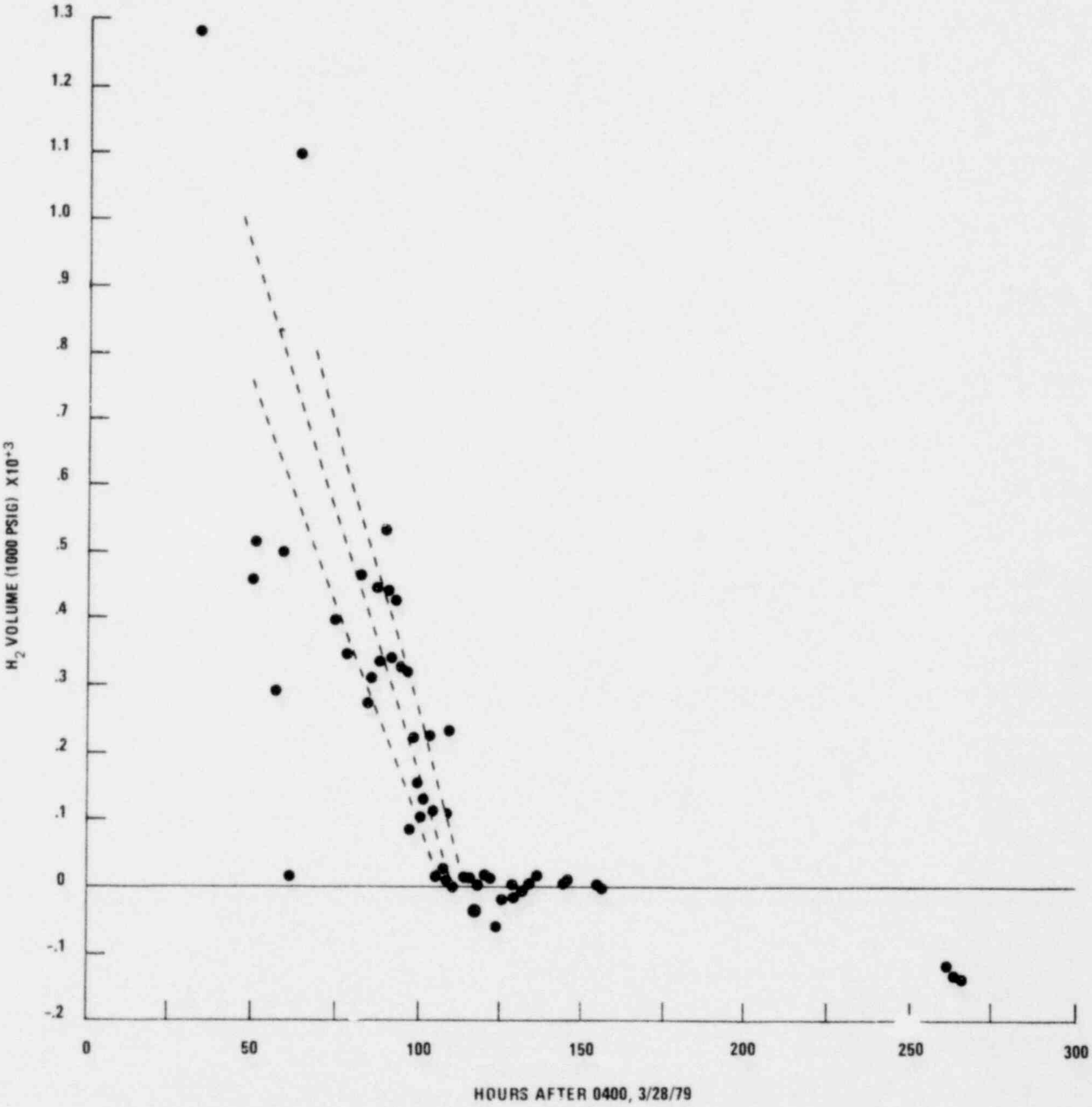
**The generalization involves points where the bubble was "almost gone" and the formula yields an upper bound.

***Memo from James H. Taylor to John Bickel, 20 July 1979. We consider only the data "With Solubility Correction."

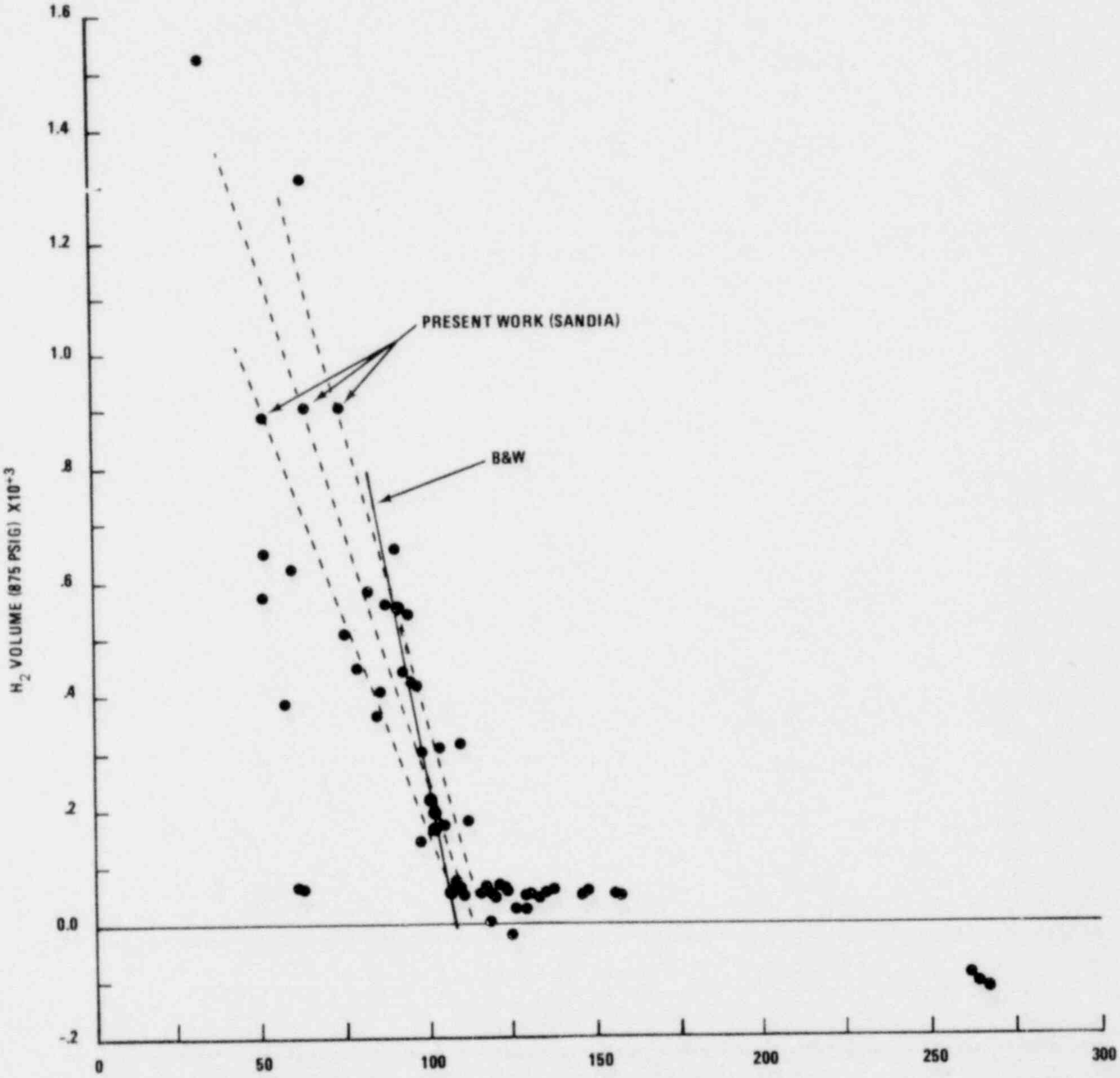
We have not included the effects of any leakage of water or loss of hydrogen during the course of a measurement. The former clearly exists; an analysis in the Bubble Book shows replenishment of the makeup tank at an average rate equivalent to 46 ft^3/hr at RCS temperature and pressure. The excess over the bubble shrinkage rate of 16 ft^3/hr is presumably unaccounted leakage, a mass loss of perhaps 1700 lb_m/hr (3 or 4 gpm). This is confirmed in the data reduction in that, for cases where it is clear that no bubble exists (on 4/2/79 and 4/3/79), the RCS still appears to accept an excess 500-1000 lb_m of water during a typical pressure excursion. When time permits, we intend to repeat the analysis, including this average leakage rate in the mass balance from which the bubble size is inferred. We do not anticipate any large change in results.



APP. FIGURE II-20. Total Hydrogen in RCS (Figure 1)



APP. FIGURE II-21. Bubble Volume at 1000 psig (Figure 2)



APP. FIGURE II-22. Bubble Volume at 875 psia (Figure 3)

APPENDIX II. 10

ANALYSIS AND CALCULATIONS BY AND FOR SANDIA LABORATORIES

1. HYDROGEN EFFERVESCENCE AND THE PRESSURIZER LEVEL DETECTOR

D. A. Powers

During the accident at Three Mile Island the pressurizer level detector indicated several changes in water level that seem to coincide with depressurization of the primary reactor coolant system. It has been suggested that effervescence of hydrogen from the reference leg of the pressurizer level detector may be responsible for these apparent changes in water level. In this analysis, it will be shown that the magnitude of hydrogen effervescence is insufficient to support this suggestion.

Description of Level Detector

The pressurizer level detector is schematically diagrammed in Figure 1. The level detector consists of two ½" pipes and a differential pressure transducer. The reference leg passes up the side of the pressurizer and the measuring leg attaches near the base of the pressurizer. Both lines extend out to the wall of the containment building.

Hydrogen Solubility in Water

At the modest temperatures and pressures encountered in the reactor situation hydrogen solubility in water is well-described by Henry's Law:

$$P_{H_2}\gamma = Hx$$

Where

P_{H_2} = the hydrogen partial pressure in equilibrium with the solution.
γ = fugacity coefficient for hydrogen gas
H = Henry's Law coefficient
X = mole fraction of hydrogen in solution

Data concerning the solubility of hydrogen in water are summarized in Table 1. Conclusions that may be drawn from this data are:

a) The fugacity coefficient of hydrogen gas may be taken as unity

b) The Henry's Law coefficient is a function of the absolute temperature only

The data in Table 1 may be correlated by the expression (1):

$$-0.1233[\log_{10}(H\times10^{-4})]^2-0.1366(10^3/T)^2$$
$$+0.02155[\log_{10}(H\times10^{-4})](10^3/T)$$
$$-0.2368\,[\log_{10}](H\times10^{-4})]$$
$$+0.824g\,(10^3/T)=1$$

Where

$T=$ absolute temperature (K)
$H=$ Henry's Law coefficient (atmospheres/mole fraction)

A solubility map for hydrogen in water at temperatures of 0-700°F and hydrogen partial pressures of 1 to 2200 psia based on the above correlation is shown in Figure 2. The map and the correlation are strictly applicable only to hydrogen solubility in pure water. Data exist showing that hydrogen solubility decreases when strong electrolytes are dissolved in the water (8)[9]. The reduction in solubility for dilute electrolyte solutions is approximately additive based on the molar concentration of the solution. No data on the reduction of hydrogen solubility with addition of sodium borate, boric acid and sodium hydroxide have been found. Discussions below are based, then, on hydrogen solubility in pure water. Water ejection predicted below is conservative in that predictions of ejection will be too high.

Water Ejection

When the partial pressure of hydrogen in equilibrium with a water solution of hydrogen is reduced, hydrogen bubbles may form in the solution. In appropriate geometrics such as the pressurizer level detector, hydrogen bubble formation may cause water to be ejected from the system.

Water will sustain some super-saturation of hydrogen. However, in the high radiation environment of the nuclear reactor and the strong system vibrations of the TMI accident, nucleation would be expected to be easy and super-saturation unimportant.

Water ejection that occurs equally in the two legs of the level detector will not produce a change in level indication. Consequently, it is only water ejection in the incremental 30' length of the reference leg of the detector that is of concern here. Hydrogen solubility calculated here is based on the reported system pressure and the assumed temperature of the level detector plumbing—160°F. The assumed temperature of the plumbing is much lower than the water temperature in the reactor coolant system since:

1) the detector plumbing is insulated from the pressurizer, and
2) most of the level detector extends well away from the primary coolant system into cooler regions of the secondary containment building.

Water ejection during the depressurization may be considered as the result of two processes. When hydrogen bubbles form they displace water. This will be termed "static ejection" of water. During rapid bubble formation, the expansion of the bubble may impart a kinetic velocity to the slug of water above the bubble which is dissipated by gravity and drag forces of the plumbing walls. The combination of static ejection and ejection due to an imparted velocity to the water will be termed "kinetic ejection."

The magnitude of the "static ejection" is simply determined by the difference in hydrogen solubility before and after the depressurization event. The volume of water ejected is equal to the volume of hydrogen at the system pressure and temperature that must be removed from solution to re-attain equilibrium.

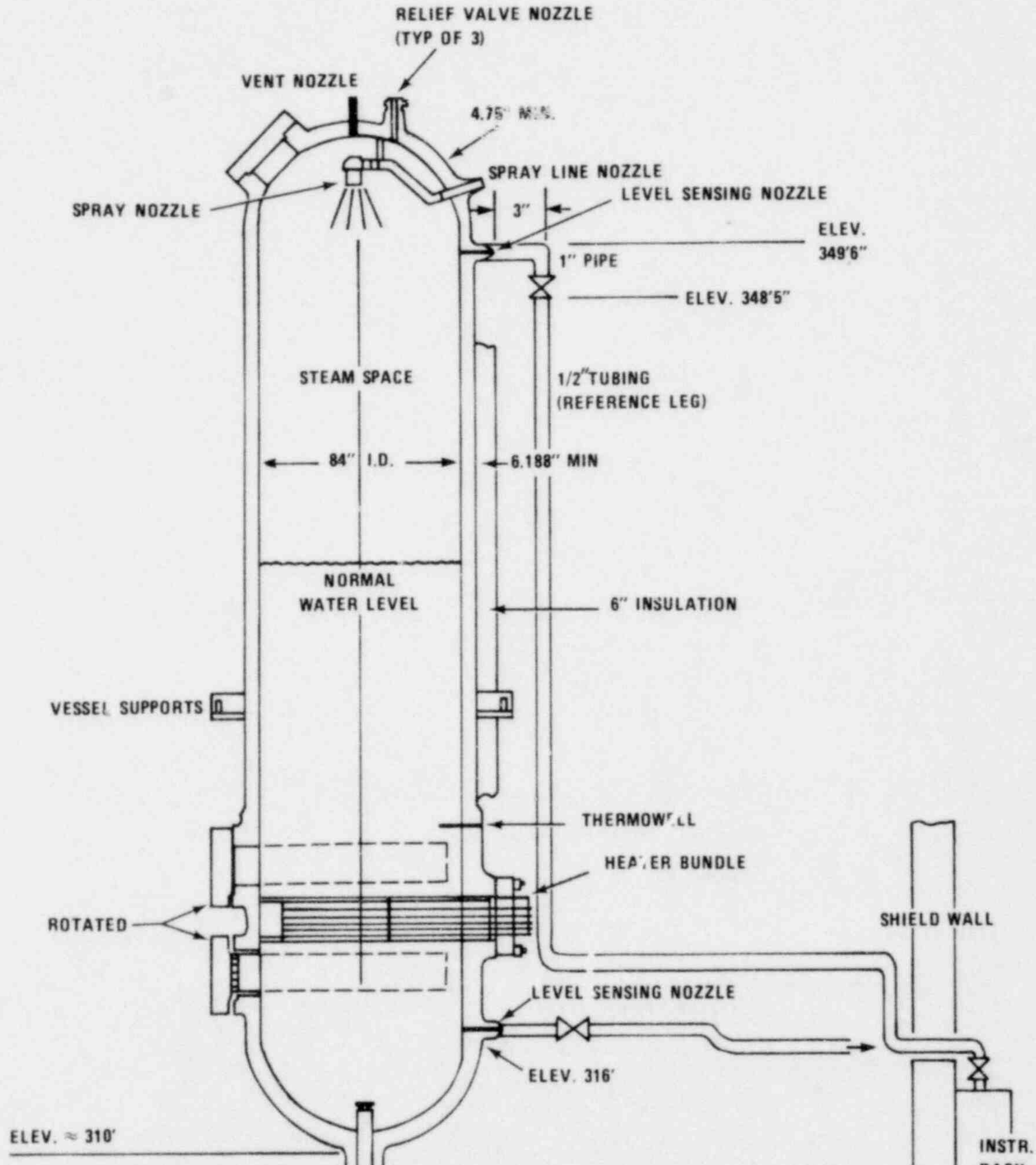
Calculations of static ejection were made assuming the system pressure was due to water vapor and hydrogen. Pressure due to the water head was neglected. Water vapor partial pressure was taken as the saturation pressure at the temperature of the detector. This treatment of water vapor partial pressure should also yield over-prediction of water ejection.

To compute the "kinetic ejection," the maximum work done by an expanding bubble is computed. To insure an upper bound on the work is computed, assumptions that are not consistent are made:

1) bubble expansion is reversible and adiabatic
2) bubble formation is rapid in spite of the assumption of reversibility
3) bubble formation occurs in a single location and the water above the bubble behaves like a solid slug
4) hydrogen gas in the bubble is at the system temperature in spite of the adiabatic assumption.

The work done by adiabatic bubble expansion during a depressurization event from P_i to P_f is:

$$g\frac{LA\rho}{A}\Delta V(P_i,P_f) = W$$



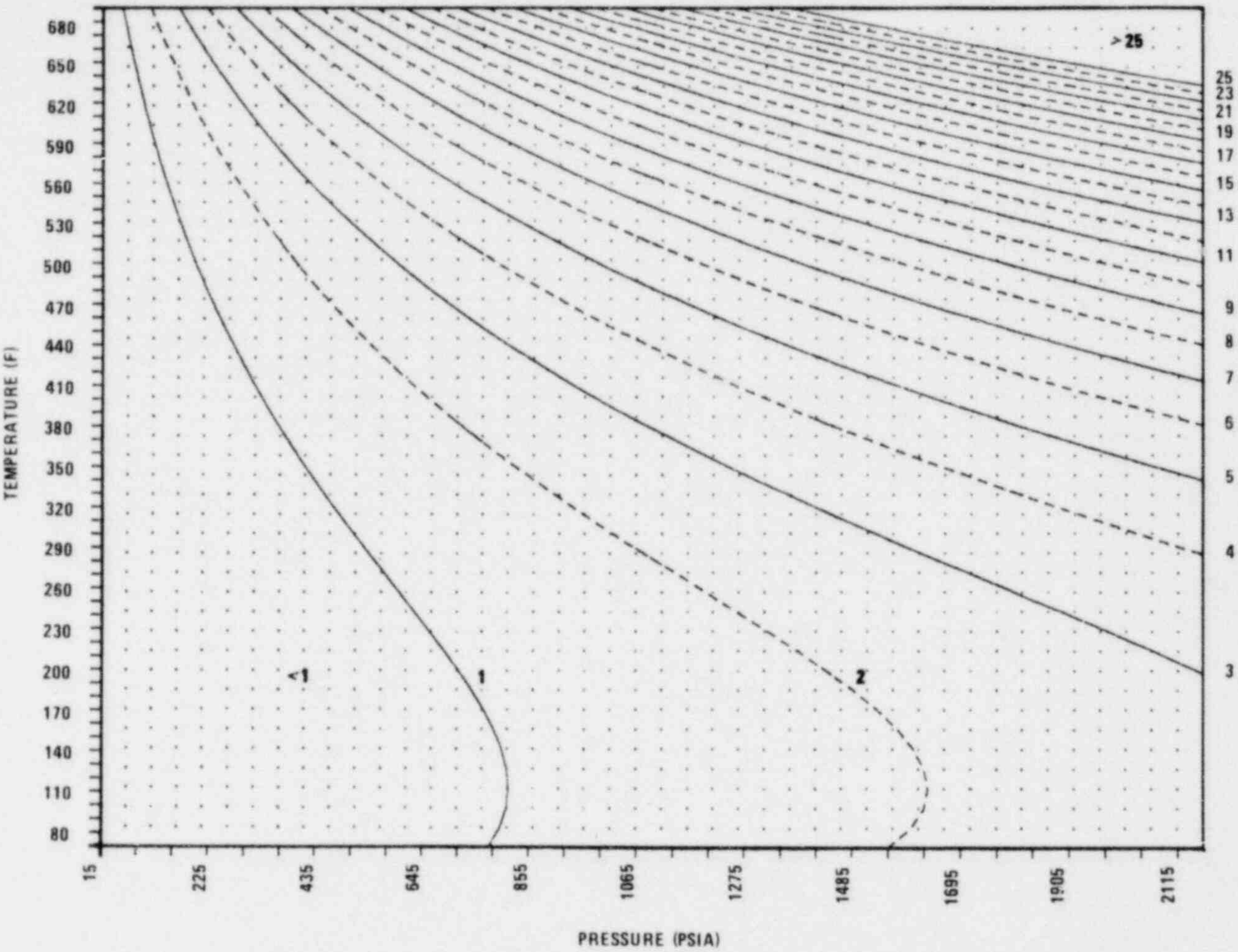
APP. FIGURE II-23. The Pressurizer (Figure 1)

TABLE 1. Hydrogen solubility data

Temperature (c)	Partial Pressure Hydrogen (atm)	H x 10^{-4} (atm/mol frac)	Solubility (ml H_2 STP/ml H_2O)	Reference
0		5.79		2
5		6.08		2
10		6.36		2
15		6.61		2
20		6.83		2
25		7.07		2
30		7.29		2
35		7.42		2
40		7.51		2
45		7.60		2
50		7.65		2
60		7.65		2
70		7.61		2
80		7.55		2
90		7.51		2
100		7.45		2
19.5	1.184	7.42		3
19.5	2.632	7.42		3
19.5	3.947	7.43		3
19.5	5.263	7.47		3
19.5	6.579	7.56		3
19.5	7.895	7.70		3
19.5	9.210	7.87		3
19.5	10.855	8.17		3
23.5	1.447	7.75		3
23.5	2.632	7.76		3
23.5	3.947	7.77		3
23.5	5.263	7.81		3
163	15.31		0.48	4
163	17.35		0.498	4
163	17.35		0.51	4
163	35.71		1.04	4
163	38.71		1.08	4
163	39.73		1.08	4
163	40.14		1.097	4
163	41.16		1.17	4
163	44.56		1.285	4
163	54.15		1.64	4
163	67.35		1.89	4
163	68.37		1.93	4
163	74.15		2.12	4
163	76.53		2.25	4
163	83.27		2.45	4
163	87.76		2.42	4
163	89.80		2.52	4
135	13.94		0.362	4
135	14.63		0.363	4
135	15.31		0.36	4
135	15.99		0.414	4
135	35.03		0.80	4
135	35.71		0.84	4
135	36.39		0.87	4
135	60.20		1.42	4
135	60.88		1.38	4
135	62.24		1.38	4
135	63.60		1.42	4
135	93.54		2.15	4
135	94.22		2.17	4

TABLE 1. Hydrogen solubility data—Continued

Temperature (c)	Partial Pressure Hydrogen (atm)	$H \times 10^4$ (atm/mol frac)	Solubility (ml H_2 STP/ml H_2O)	Reference
135	95.58		2.05	4
135	96.26		2.10	4
100	15.31		0.32-0.33	4
100	15.99		0.35	4
100	16.67		0.306-0.33	4
100	35.71		0.67	4
100	37.07		0.71	4
100	38.43		0.74-0.76	4
100	69.73		1.38	4
100	70.41		1.35	4
100	71.08		1.34	4
100	71.77		1.35	4
100	96.46		1.81	4
100	98.98		1.93	4
24	20.41		0.32	5
24	24.96		0.44	5
52	13.6		0.33	6
52	20.4		0.41	6
52	23.81		0.45	6
149	6.8		0.13	7
149	13.6		0.28	7
149	20.4		0.40	7
149	25.51		0.52	7
149	34.01		0.75	7
174	6.80		0.15	7
174	13.60		0.30	7
174	20.40		0.43	7
174	25.51		0.56	7
174	34.01		0.75	7
199	6.80		0.18	7
199	13.60		0.34	7
199	20.40		0.52	7
199	25.51		0.68	7
224	6.80		0.22	7
224	13.60		0.49	7
224	20.40		0.75	7
224	25.51		0.94	7
224	34.01		1.26	7
260	6.80		0.39	6
260	13.60		0.91	6
260	20.40		1.25	6
315.5	6.80		0.65	6
315.5	13.60		1.32	6
315.5	20.40		2.01	6
343	6.80		1.40	6
343	7.82		1.63	6
343	8.16		1.68	6
343	8.50		1.74	6



APP. FIGURE II-24. Hydrogen Solubility Versus Temperature and Pressure ($ml\ H_2(STP)/g\ H_2O$) (Figure 2)

Where

g = gravitation constant

L = length of water column

ρ = density of water

ΔV = volume of hydrogen effused in the depressurization

w = work done

Assume then that all of this work is used to impart a velocity to the water column above the bubble. Then:

$$gL\,\rho\Delta V = \frac{LA}{2}\rho V_i^2$$

where V_i = initial velocity of the water column. Then,

$$V_i = \sqrt{\frac{2g\,\Delta V}{A}}$$

The water column will move upward until the forces of gravity and friction reduce the velocity to zero. If the water column is full, the distance the water moves is equal to the head of water lost. Assume the friction forces are negligible. Then,

$$d = \int_0^{\tau_0} v(\tau)d\tau = \int_0^{v_i/g} v_i - g\tau d\tau$$

$$= \frac{v_i^2}{2g} = \frac{\Delta V}{A}$$

Thus, the maximum head-loss due to the ejection of water because velocity imparted by bubble formation is equal to that loss due to "static ejection." Thus,

"kinetic ejection" = 2 ("static ejection").

a. Comparison of the Model with Experiments

Babcock and Wilcox have tested the performance of the pressurizer level detector during depressurization (10). The model described above was used to predict head-loss due to hydrogen effervescence in these experiments. System pressure was used as input data for the model. Results are shown in Figures 3-6. "Static ejection" and "kinetic ejection" are shown in these figures as solid and dashed lines, respectively.

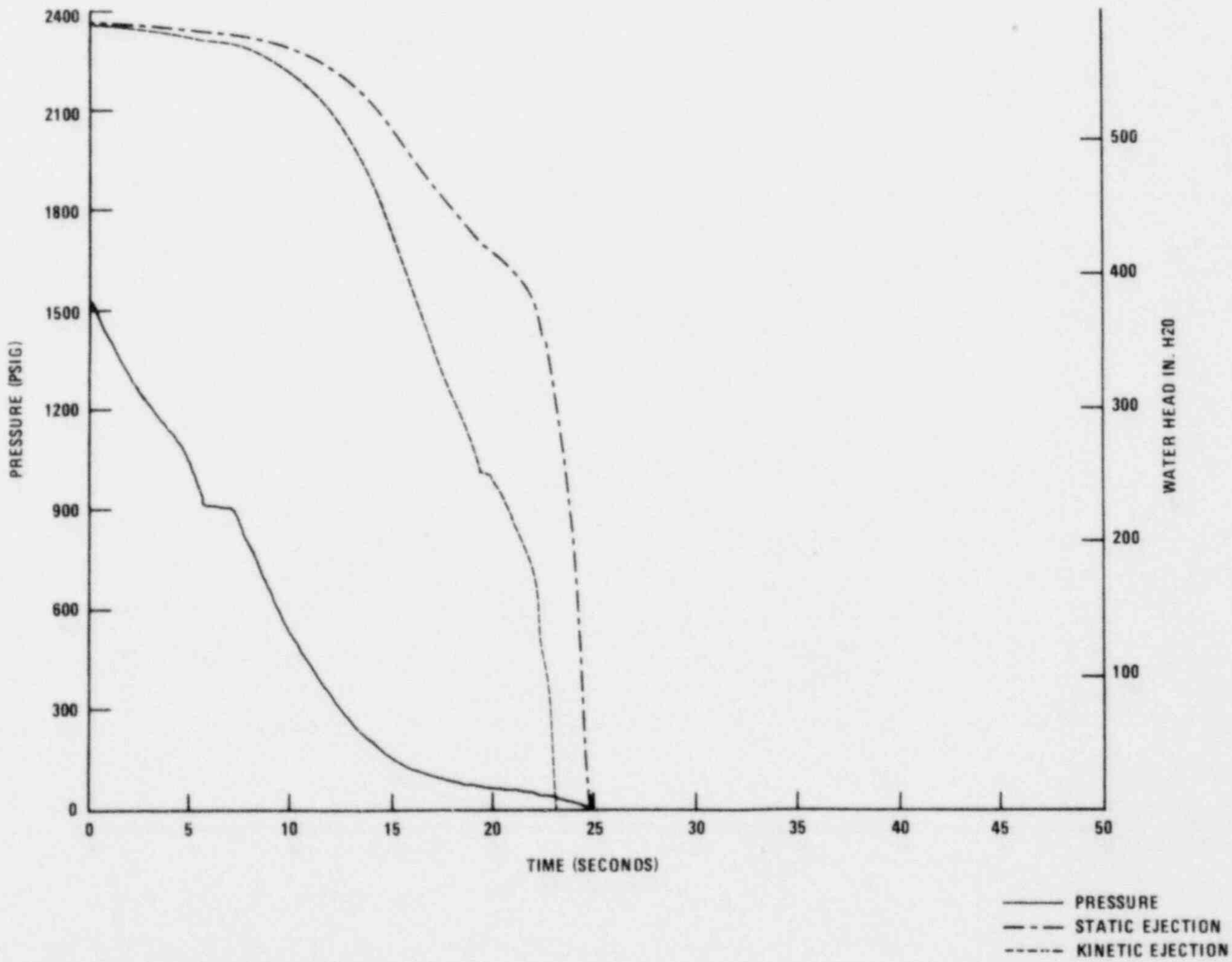
The following conclusions may be drawn from the comparison of model predictions with experimental results:

1) The experimental headloss is less than head-loss predicted by "kinetic ejection."
2) Kinetic effects other than those considered here inhibit head-loss so that "static ejection" is usually an upper bound to head-loss.
3) The rate at which re-equilibrium is attained in the experiments is rapid in comparison to events of the reactor accident, but the rate is not always well described by the model.

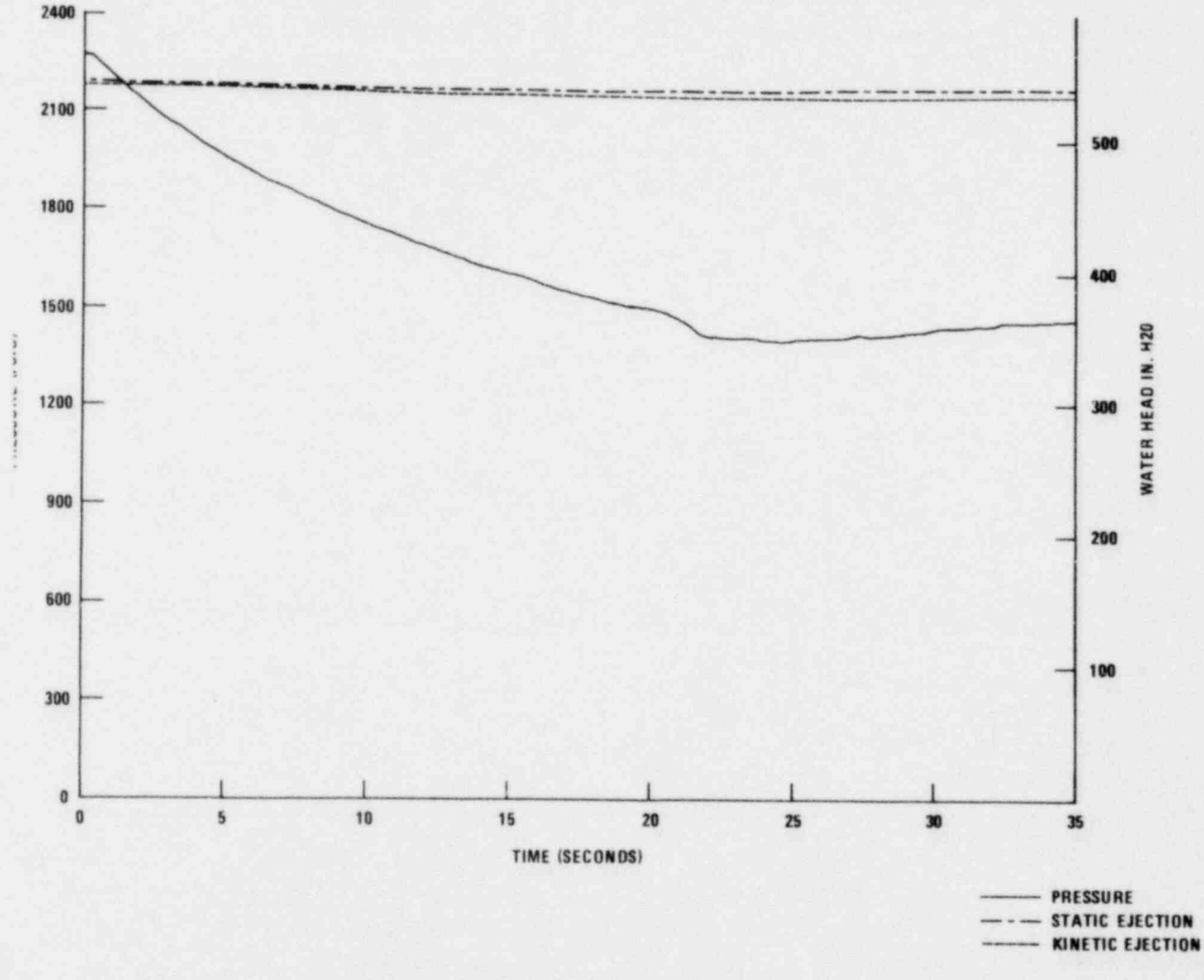
These results give some confidence that head losses predicted by the model will be useful in assessing whether pressurizer level indications were in error during the accident due to hydrogen effervescence.

b. Head Loss During the Accident

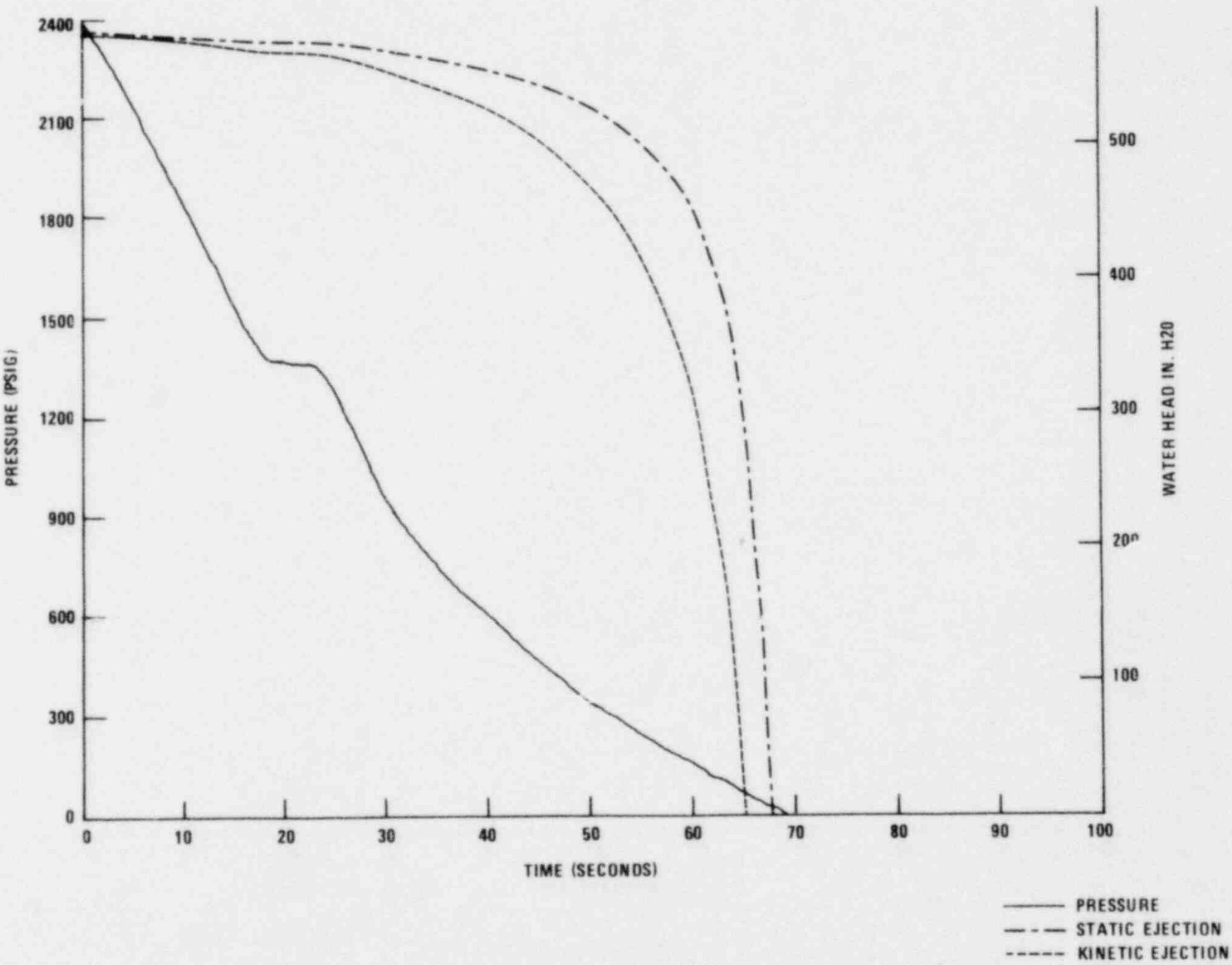
The possible head-loss, based on the coolant system pressure, due to hydrogen effervescence is shown in Figures 7 and 8. The losses were computed assuming the water was saturated with hydrogen at about 2200 psi and an instantaneous depressurization to the observed pressure at any time occurred. The maximum predicted head-loss based on "kinetic ejection" was 57 inches. It is apparent that head-loss due to hydrogen effusion is too small to be responsible for the large level changes reported for the accident.



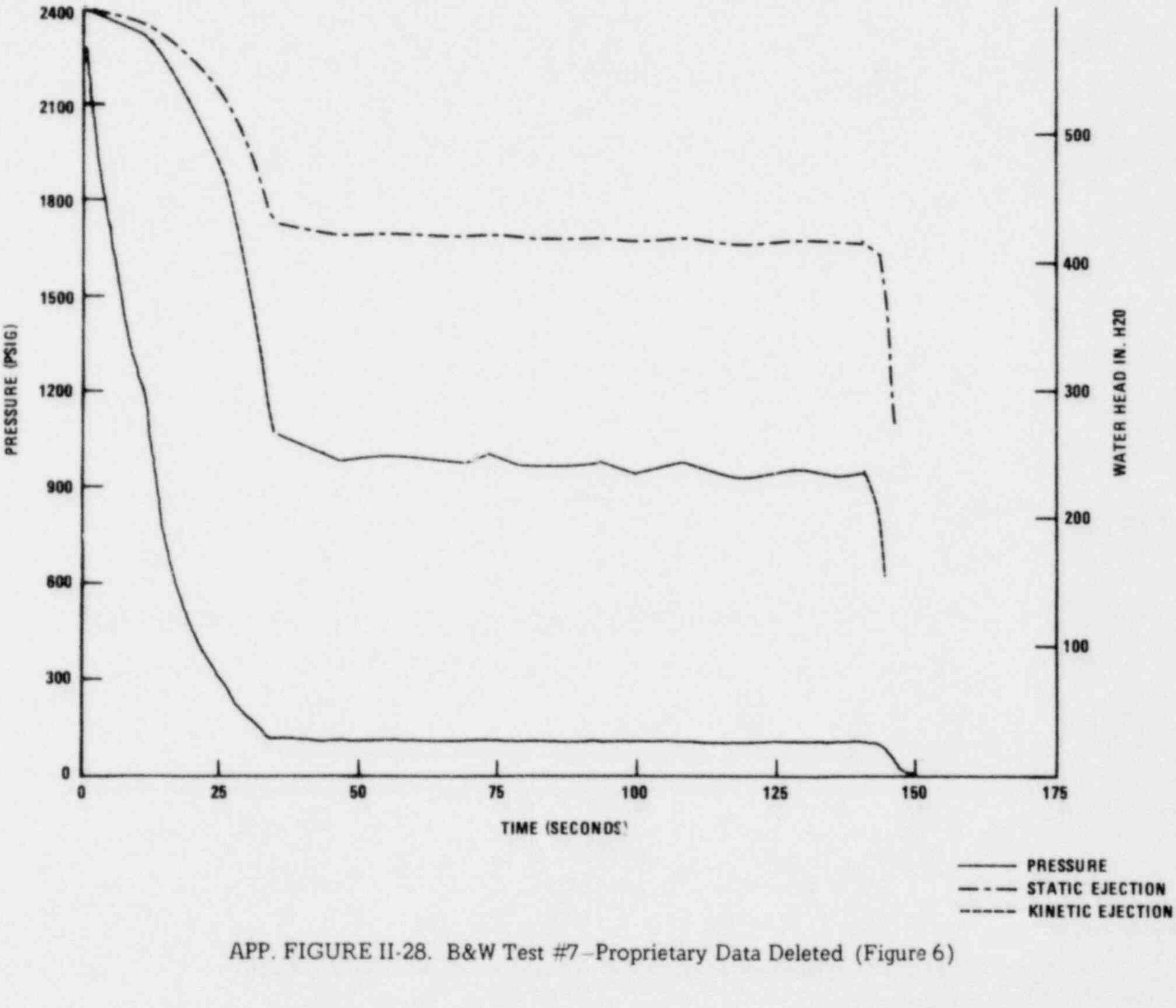
APP. FIGURE II-25. B&W Test #2–Proprietary Data Deleted (Figure 3)



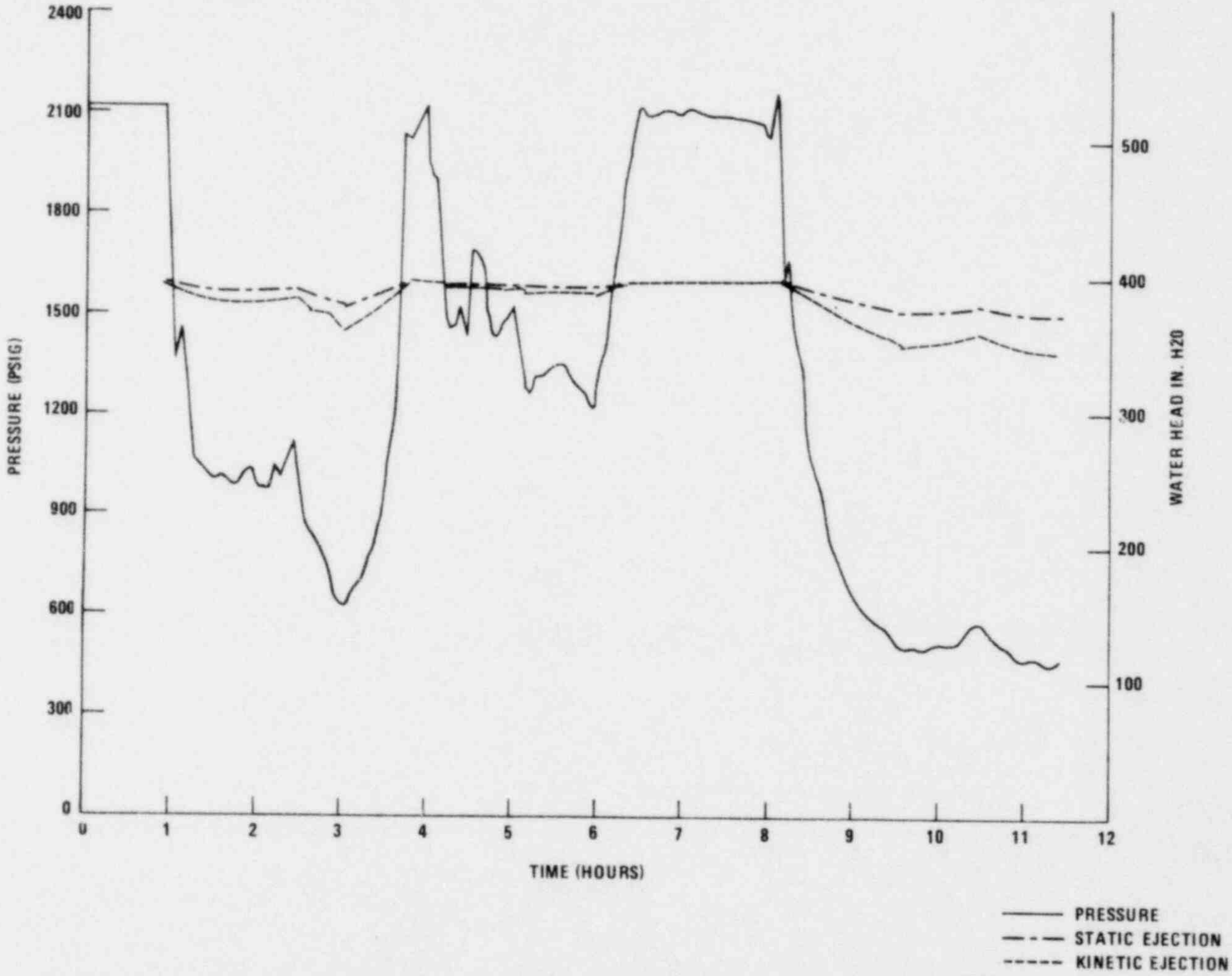
APP. FIGURE II-26. B&W Test #5–Proprietary Data Deleted (Figure 4)



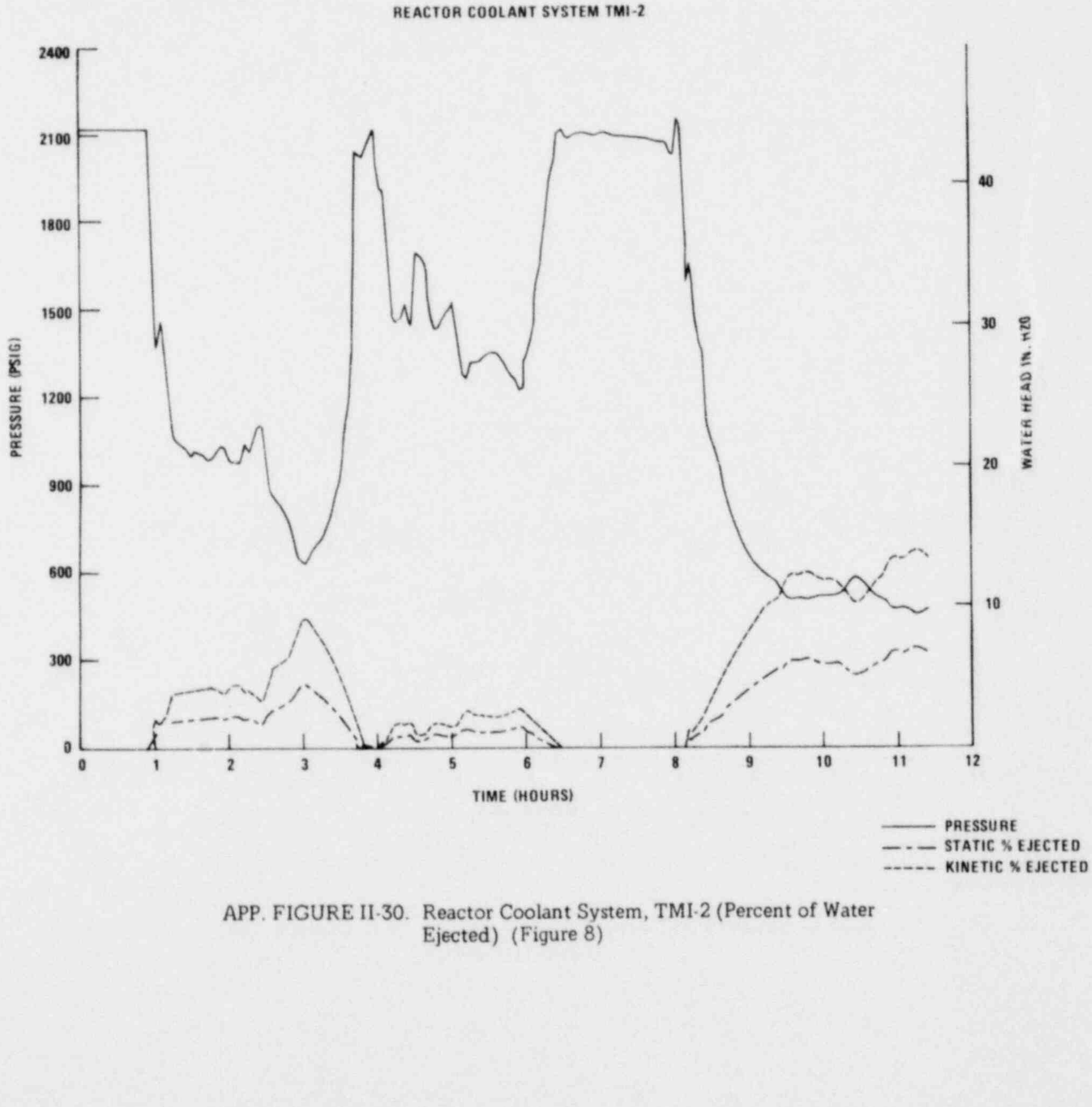
APP. FIGURE II-27. B&W Test #3–Proprietary Data Deleted (Figure 5)



APP. FIGURE II-28. B&W Test #7–Proprietary Data Deleted (Figure 6)



APP. FIGURE II-29. Reactor Coolant System, TMI-2 (Water Head) (Figure 7)



APP. FIGURE II-30. Reactor Coolant System, TMI-2 (Percent of Water Ejected) (Figure 8)

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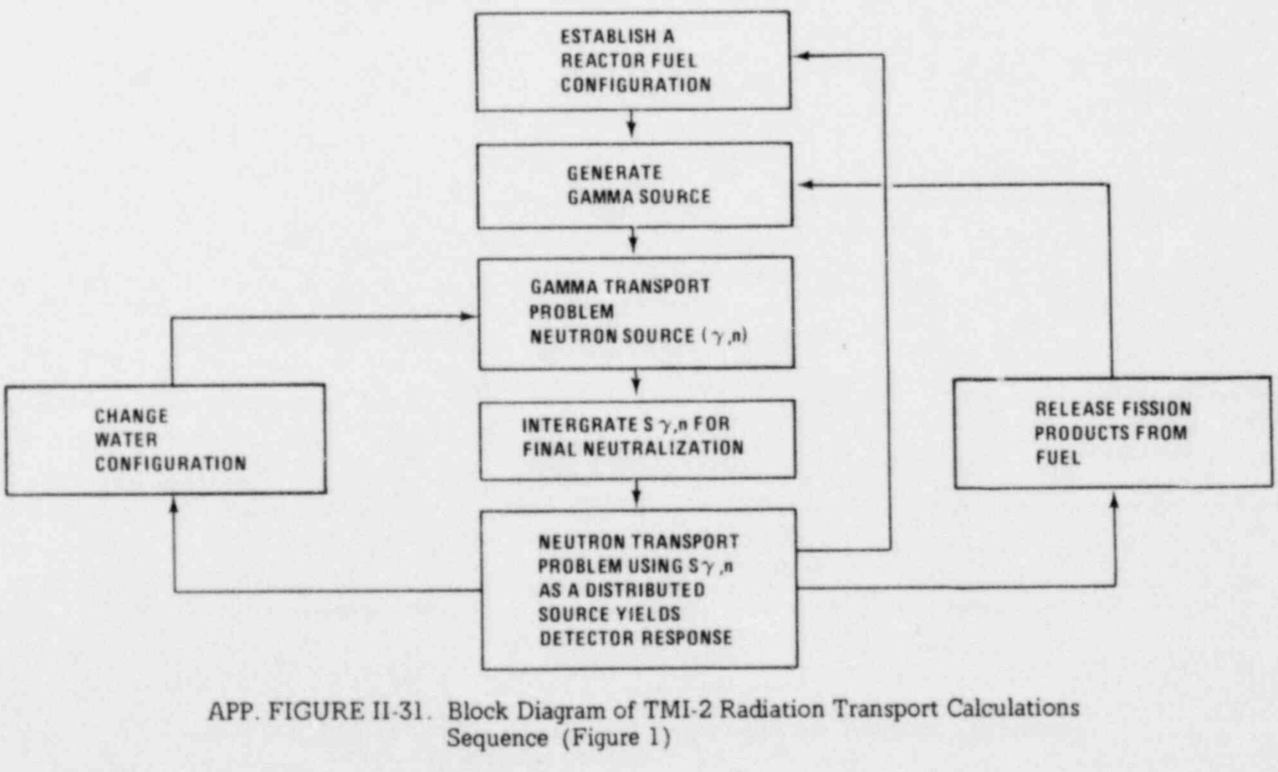
2. EXCERPTS FROM TMI-2 SENSITIVITY STUDY

Purpose

The purpose of this phase of the review group investigation was to examine the sensitivity of the Source Range Monitor (SRM) to a range of possible reactor configurations following the TMI-2 accident. A detailed analysis of the SRM trace could then be undertaken utilizing these sensitivity results to provide additional input to the explanation of the sequence of events in TMI-2. A series of gamma and neutron transport calculations were performed for several possible reactor configurations. This report briefly summarizes these SRM sensitivity calculations. Configurations and parameters examined included the following: homogeneous voiding, core uncovery, fuel relocation, the relative importance of the distributed source and core multiplied neutrons, the hotoneutron effect of changing the boron content in the coolant water, and the effect of removing control poison from the core.

Sequence of Calculations—(Fig. 1)

1. A gamma source from the fission product inventory was calculated at 2 hours after the accident. A core-averaged inventory was obtained from ORIGEN. Power distribution data (R-Z) was used to distribute the gamma source spatially within the core regions for the intact core. For disrupted configurations the source was redistributed accordingly.
2. R-Z Gamma Transport calculations were performed with TWOTRAN II. This yielded the space- and energy-dependent gamma flux for a given configuration.
3. A space and energy dependent γ-n neutron source was calculated from the gamma flux and the photoneutron response function for deuterium. The threshold energy for this reaction is 2.2 MeV.
4. The (γ-n) neutron source was integrated over energy and space to obtain a normalization factor. This factor is a function of the water density, water level, and core configuration. This factor is required to correctly normalize the detector response in the neutron transport problem.
5. An R-Z neutron transport problem with TWOTRAN II was then performed. A normalization of 1.0 was used and results were corrected by the factor calculated in step 4. This calculation provides the detector response for a given core configuration. In general, each change in reactor configuration required a repeat of steps 2 through 5. The ENDF/B cross section sets used in these calculations were not changed as the core configuration changed.



APP. FIGURE II-31. Block Diagram of TMI-2 Radiation Transport Calculations Sequence (Figure 1)

Calculational Model

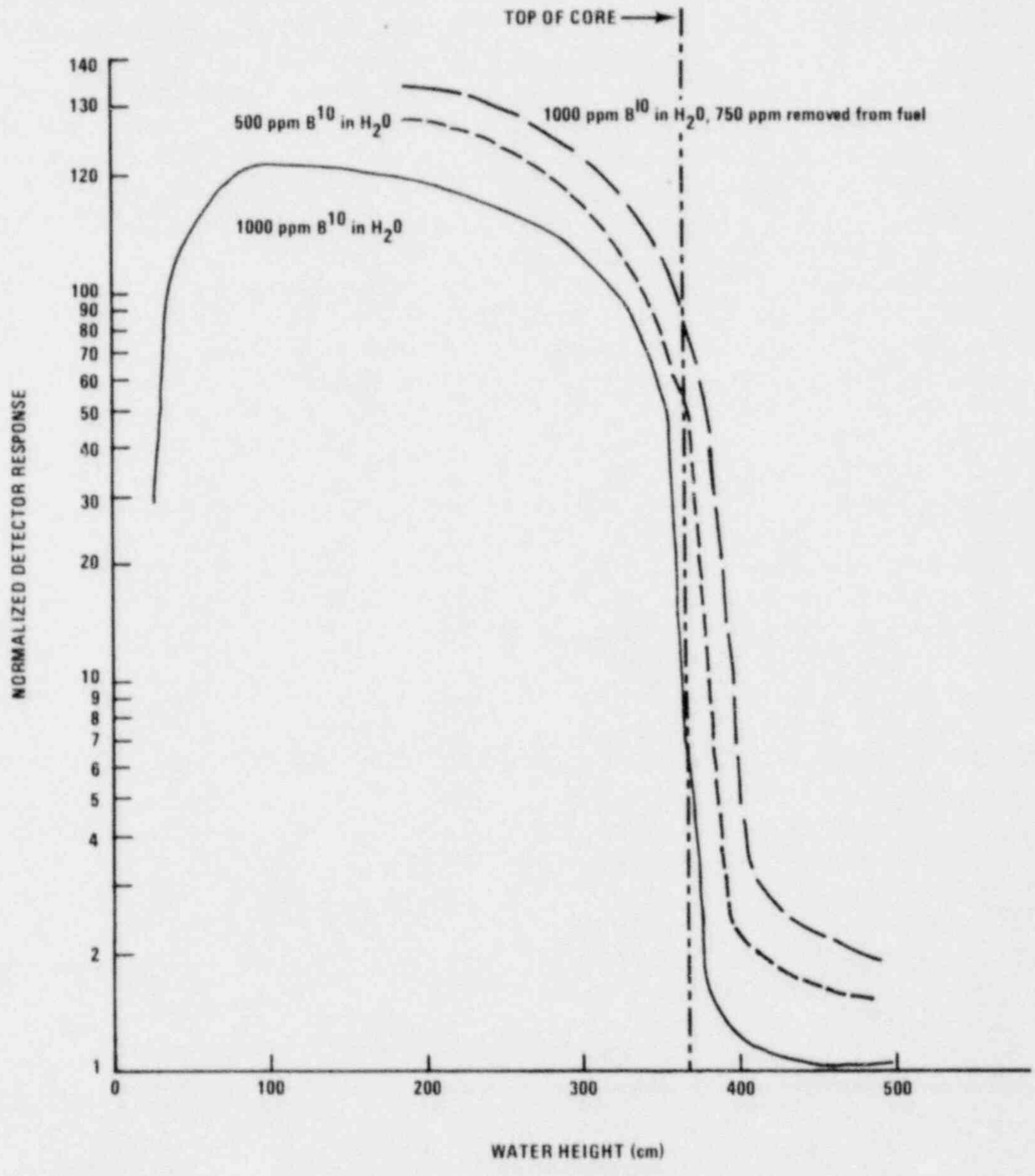
Overall dimensions of core, downcomer, vessel, and detector locations were based on data from the TMI-II SAR. Enrichments for simplified two fuel zone model were derived from a B&W memo from E.J. Baieman to A.W. Snyder dated October 18, 1979.

The two dimensional transport code, TWOTRAN-II was used in R-Z geometry for all gamma and neutron transport problems. Special software was developed to generate gamma and neutron sources in standard interface file format.

The initial K_{eff} was set at 0.91 and the equivalent natural boron content of the water was set at 1000 ppm. A distributed source of photoneutrons was established as the only neutron source for most of the calculations relevant to early accident times ($<$ 4 hrs) and a fixed startup source alone was assumed for a selected number of the remaining cases. The photoneutron source was due to gammas ($E > 2.2$ MeV) interacting with the deuterium in the coolant. The spectrum and total intensity of the gammas were derived from an ORIGEN calculation for the TMI-II core at a time of 2 hours after the reactor trip. This reference gamma source was used for the majority of the sensitivity studies. The spatial distribution of the gamma source was varied for the cases where fuel disruption and fission product releases were postulated but the spectrum and integrated total gamma source were fixed. The total photoneutron source (used in the neutron transport problems) varies, however, due to changes in water density, water level, etc., this change in neutron source level is accounted for thru a normalization factor which is calculated for each configuration. Normal water density as referred to in this memo is assumed to be $\rho = 0.72$ g/cc. Void fractions of 25%, etc. refer to 0.75 ρ (normal), 0.5 ρ (normal), etc.

Group II—Coolant Level Change Series-(Fig. 2) [Including Soluble Boron and Control Position Effects]

These calculations apply primarily to the transients in the SRM data beginning at about 1.8 hours. After the pumps were turned off phase separation occurred and coolant began to boil off. Pump 2B was turned on briefly apparently injecting a slug of coolant into the core/downcomer causing the sharp dip at 2.85 hours. The process of uncovery then continued. During the SRM peak the reading was greater than two orders of magnitude above the normal shutdown trace ($\sim$ 135-140) and about a factor of 115 higher than the reference SRM reading at 2 hours. (Note: The gamma source for the sensitivity study calculations was not adjusted for decay). The solid line is for the reference natural boron concentration of 1000 ppm. The other lines show the sensitivity to reduced poison in the core region alone. There is a large increase in the SRM activity near the top of the active core. An increase of [illegible] is observed by the time the water level is just 30 cm (1 ft) below the top of the core. A higher multiplication factor ($k_{eff} = 0.95$) due to a reduced boron content or poison rods melted during the period of uncovery could increase the readings to 170 or higher at that level. There is poor sensitivity once the core is uncovered. It should be noted that core and downcomer water levels were lowered together in these calculations, essentially assuming a zero void fraction in the core. A less severe initial slope to the curve is expected if a nonzero void fraction in the core is calculated.



APP. FIGURE II-32. Relative SRM Response as Function of H_2O Level in Core and Downcomer (Figure 2)

3. INTEGRATED ANALYSIS OF SOURCE RANGE MONITOR DATA FROM TMI-II

Prepared for Sandia Laboratories
as part of Rogovin Study Review Group

November 28, 1979

E. A. Straker and W. K. Hagan

Introduction

There are numerous factors which affect the source range monitor (SRM) count rate. These factors include water level in the core and downcomer, void fraction in the core, bypass and downcomer, boron concentration in the core and the physical condition of the core. The objective of this investigation was to determine the consistency between the reactor conditions and the observed SRM count rate.

Because of the large range in possibilities for core condition and coolant characteristics, a parametric systems analysis approach was taken. Since the SRM count rate is determined by photoneutrons and source neutrons both of which are multiplied by the fissile material in the core, the response must be calculated using a radiation transport code capable of performing both deep penetration shielding analysis and core multiplication. If transport results are obtained for a large number of conditions, and the results utilized in a system model in conjunction with postulated reactor conditions to calculate an expected SRM count rate. The postulated reactor conditions may be determined by other analyses or by engineering judgment.

To implement the approach a computer code was developed at Science Applications, Inc. (SAi), to quantitatively predict the count rate at the source range monitor as a function of several reactor parameters. This effort has been made with the flexibility that such a tool could also be used "in reverse," i.e., knowledge of the SRM count rate would then allow the calculation of some reactor parameters for the time period of interest.

Analysis Procedure

The approach used in the SRM code is to base the expected count rate at the SRM on the count rate which would be expected if all reactor parameters (e.g., water level, void fraction, etc.) were at their nominal values. This nominal count rate is then multiplied by correction factors for each reactor parameter which is not at its nominal value, as shown in Eq. (1).

$$C(t) = N(t)F_1(v,t)F_2(w,t)F_3(b,t)F_4(c,t) \quad (1)$$

where,

- $C(t)$ is the expected count rate,
- $N(t)$ is the nominal or normal count rate,
- $F_1(v,t)$ is the factor associated with void fraction changes,
- $F_2(w,t)$ is the factor associated with water level changes,
- $F_3(b,t)$ is the factor associated with boron concentration changes,
- $F_4(c,t)$ is the factor associated with changes in the core conditions.

These multiplicative factors include the effect on the count rate of deviation from their nominal value and could be considered correction factors for non-normal conditions. The code can handle any number of parameters; the four used here are representative and were used in the TMI analysis. Thus, to predict the SRM response for an abnormal trip, information on the count rate for a normal trip is required. By using normal trip results the time dependent behavior of the photoneutron source is properly treated for a non distorted core.

The approach utilized was "one-dimensional" in the sense that all of the reactor parameters affect the count rate independently, e.g., the effect of varying the boron concentration is taken to be independent of the water level. This simplification was required for this study since there was not yet enough data to quantify the interdependencies. However, once the information is available it can be easily incorporated into this model. For example, the void fraction and the water level effects may be interdependent and Eq. (1) would be modified to take the form of Eq. (2).

$$C(t) = N(t)G_1(v,w,t)F_3(b,t)F_4(T,t) \quad (2)$$

where

$$G_1(v,w,t) \neq F_1(v,t)F_2(w,t).$$

Also, the interdependence between water level in the downcomer and the core water level and void fraction could be treated as dependent effects.

For this project the independence of variables was determined by the transport data base generated at Sandia. If additional transport results were available then other approaches could be utilized. For example, work funded by EPRI is oriented toward including all possible core condition dependencies in the transport calculation and thus the integration over time dependent variables is considered in the transport results. Results are also presented for comparison in this paper for some EPRI transport results.

Data Base

The data base information was obtained from Sandia Laboratory calculations and documented by Paul Picard in a handout of 20 November 1979. The data base was supplemented by alternative data obtained from Technology for Energy Corporation (TEC). The data utilized in our analysis are given in Tables 1 through 5. Linear interpolation was utilized.

TABLE 1. Normal reactor trip data

Time (min)	Intensity
5	1.0 (+5)
10	1.0 (+4)
12	3.5 (+3)
14	1.8 (+3)
16	1.2 (+3)
18	9.5 (+2)
20	8.5 (+2)
22	7.7 (+2)
24	7.2 (+2)
26	6.9 (+2)
28	6.5 (+2)
30	6.2 (+2)
32	6.0 (+2)
34	5.8 (+2)
36	5.6 (+2)
40	5.2 (+2)
60	4.1 (+2)
80	3.3 (+2)
100	2.7 (+2)
120	2.3 (+2)
140	1.9 (+2)
180	1.5 (+2)
240	1.25 (+2)
360	1.0 (+2)
480	8.7 (+1)

TABLE 2. Correction factor for water level

Water Level (feet above bottom of core)	Factor (Sandia)	Factor (TEC)
16	1.0	1.0
13	1.3	1.1
12	8.0	1.6
11	85.0	3.5
10	120.0	10.0
7	190.0	
5		93.0
4	220.0	95.0

TABLE 3. Correction factor for void fraction

Void Fraction (%)	Factor
0	1.0
10	2.0
20	3.5
30	6.0
40	10.0
50	15.0
60	25.0

TABLE 4. Correction factor for boron concentration

Boron Concentration (ppm)	Factor
500	1.5
1000	1.0
2000	0.5

TABLE 5. Correction factor for core condition

Percent Core Displaced	Factor
0	1.0
5	2.4
10	3.7
15	4.2
20	4.6

The two sets of data in Table 2 on water level should not be directly compared since the TEC data includes a hydrostatically balanced core and downcomer and thus the water level in the downcomer results in a higher water level in the core. Also, the TEC data were modified for levels near the top of the core after discussions with Jim Robinson of TEC. The core exit void fraction varies up to 40% in the TEC transport calculation.

Analyses of TMI-II

The analysis is separated into two parts. The first analysis was based on the trial and error choice of water level and void fraction versus time in order to obtain a reasonably good agreement with the observed count rate. Figure 1 shows the plots of the data base. Figure 2 shows the comparison between the observed (dashed curve) and calculated SRM count rate using the Sandia data base. The time dependent boron concentration, core condition, water level and void fraction are given in Fig. 3.

As noted previously the data derived from the Sandia calculations is valid for the water level being the same in the downcomer and the core. Analysis was also performed using the TEC data base and Figs. 4 and 5 show the results. Due to the maximum variation in the TEC data being only a factor of 95, it was not possible to obtain the full variation between a normal trip and TMI-II data without adding a factor for void fractions. This void fraction could account for the difference in behavior in the core and bypass region. There is no sound basis for the void fraction and therefore it might indicate the range of uncertainty that might be associated with the data.

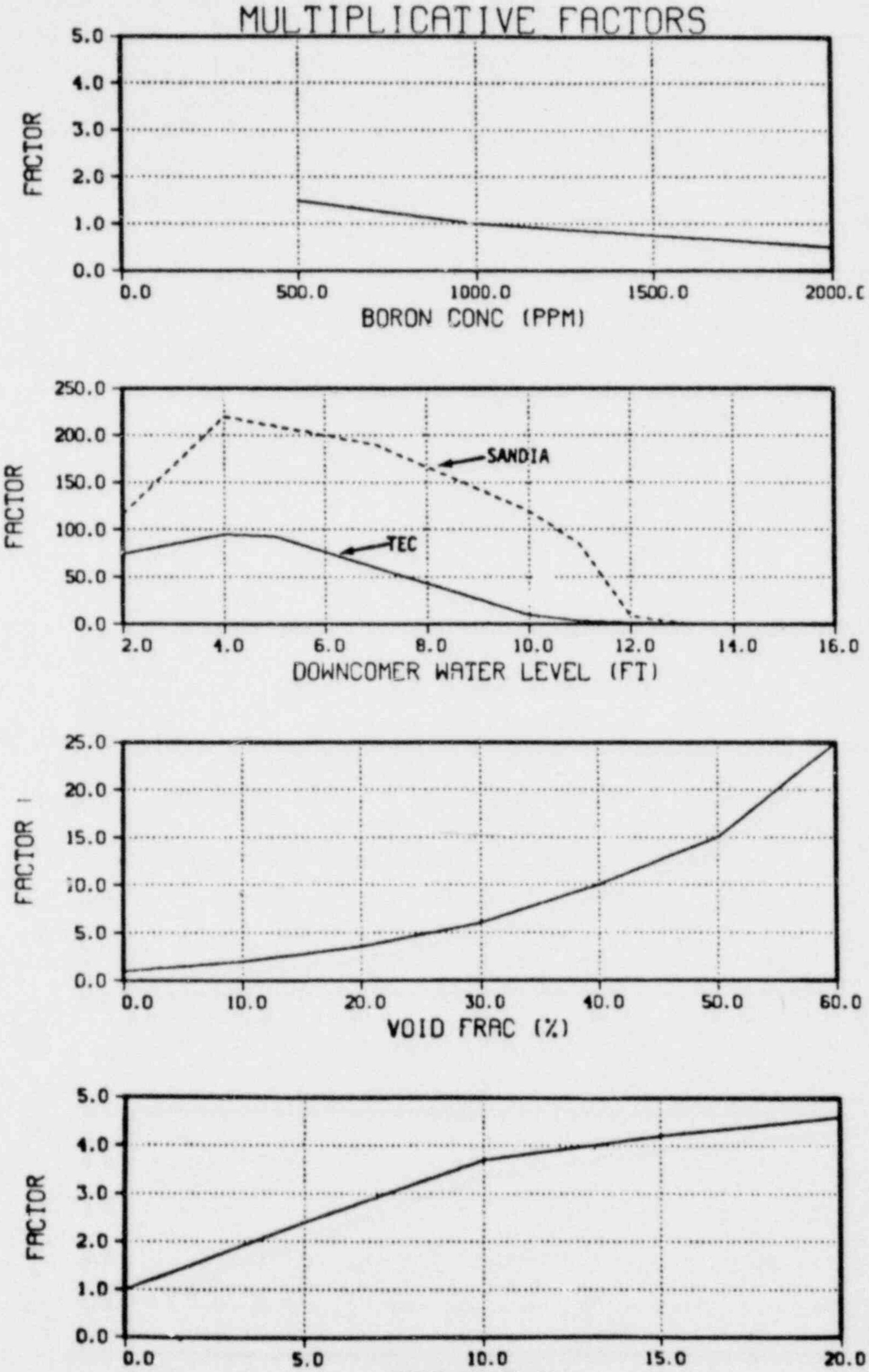
A comparison of the water levels derived from the two data bases is given in Fig. 6. Note that the TEC data base would indicate that about 5 feet more of the core would be uncovered since for low water heights the core water level is approximately the same as the downcomer water level in the TEC transport calculations. It is important to realize the lack of uniqueness of conclusions to be drawn from the analysis. Besides an uncertainty in the normal trip data and the observed TMI-II data, there are uncertainties associated with the transport modeling—especially the relationship between void fraction and water level. Assumptions on the characteristics of the water in the bypass region could affect the transport results by about a factor of 2.

Early analysis of fuel assembly and exit core temperature have been used by others to estimate the amount of core uncovery. The core water level versus time has been postulated as that given in Table 6 by different investigators (assuming linear interpolation between data points). Using these postulated water levels and the TEC data, a comparison of the resulting SRM count rate and the observed count rate is given in Figs. 7 and 8. The parameter changes which yield the count rates in Figs. 7 and 8 are given in Fig. 9. Note that time dependent water levels were input and it was assumed that core water level was the same as downcomer water level when the data base is utilized. Although this is not correct, there is not enough other data available to do otherwise.

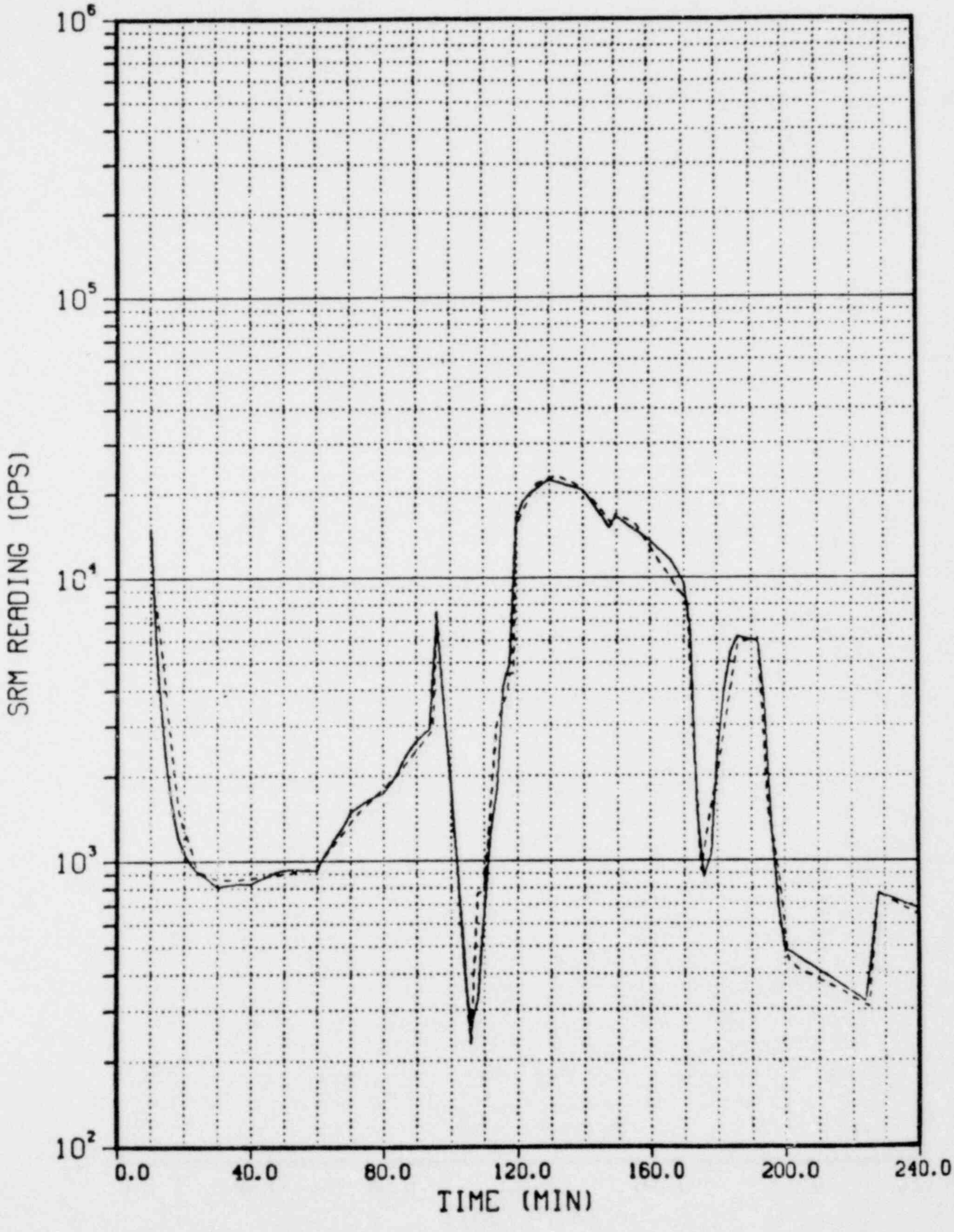
TABLE 6. Core water level versus time

	Water Level (ft)	
Time (min)	Case 1	Case 2
100	16.0	16.0
105	12.0	12.0
120		7.5
130	5.5	
140		5.0
160		6.0
170		6.0
176	5.5	
180	9.0	
190	7.5	9.0
200		9.0
205	7.5	12.0
210		16.0
215	9.5	
270	12.0	
280	16.0	

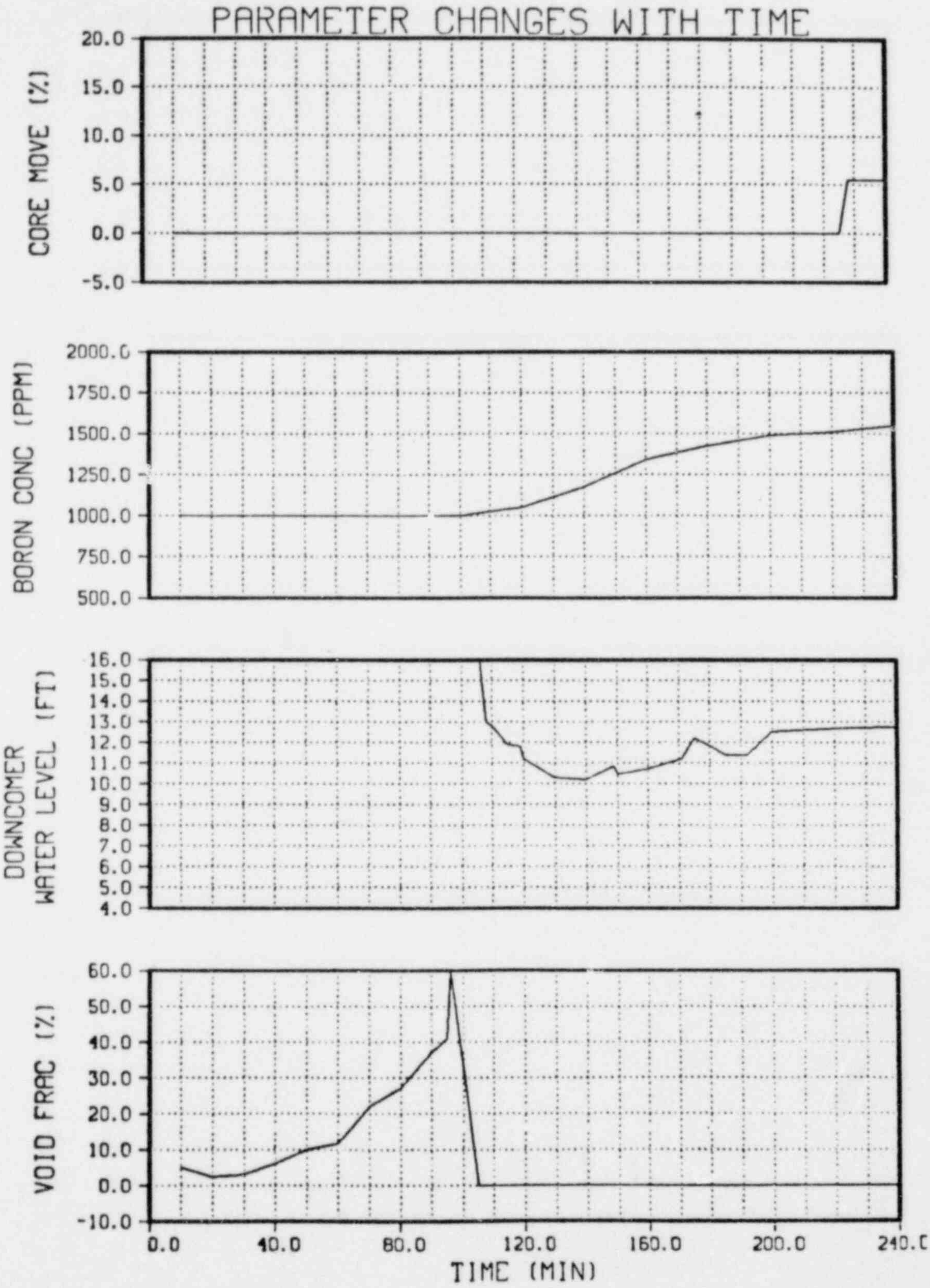
The integration of postulated core conditions and transport results indicate that a number of different core conditions could lead to the observed SRM count rates. The differences in the data bases lead to significantly different water levels. As transport data changes, other analyses can be performed easily using the approach discussed in this paper.



APP. FIGURE II-33. Plots of Input Data Base (Figure 1)

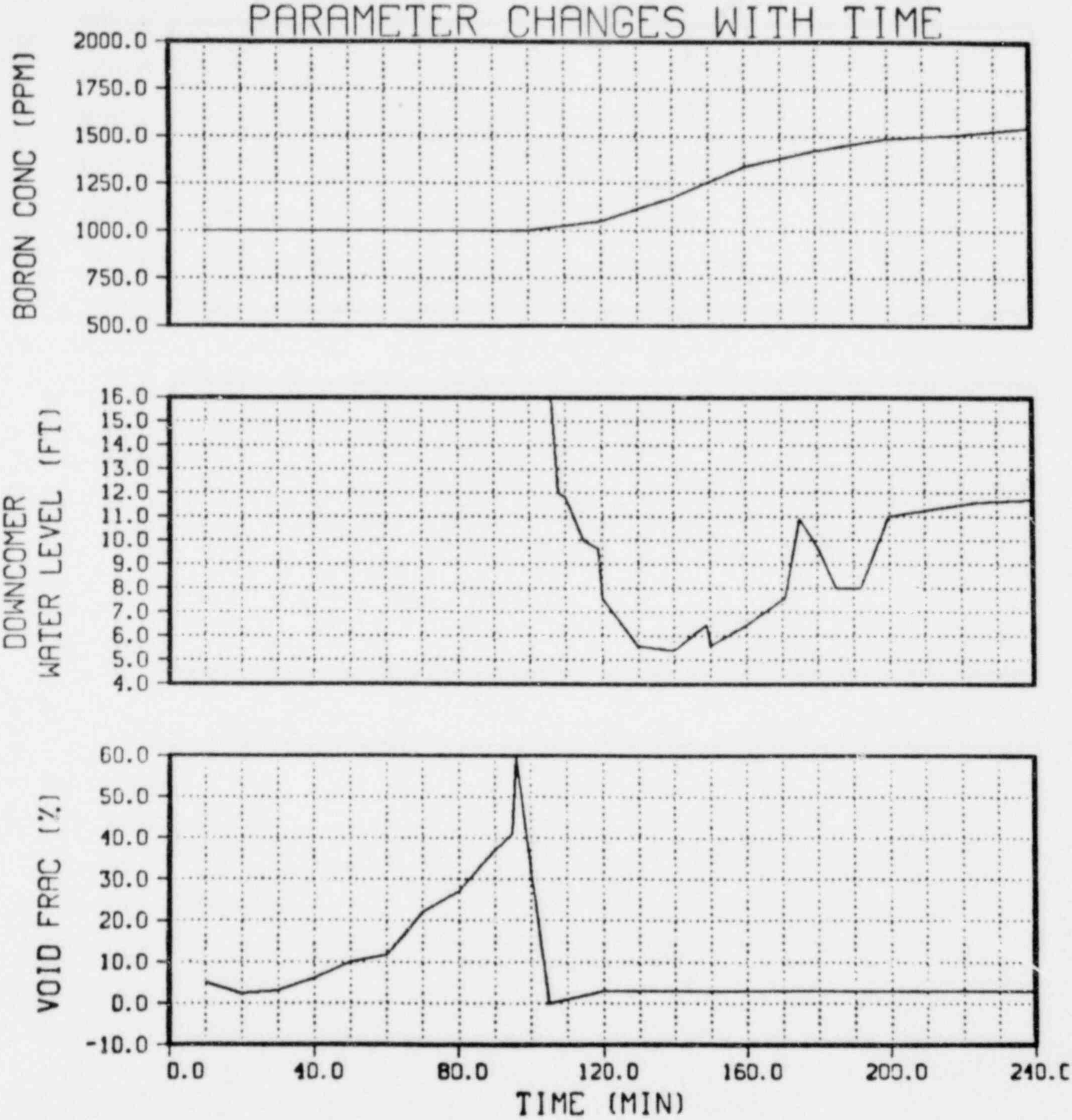


APP. FIGURE II-34. SRM Reading Versus Time Using the Sandia Data Base (Figure 2)

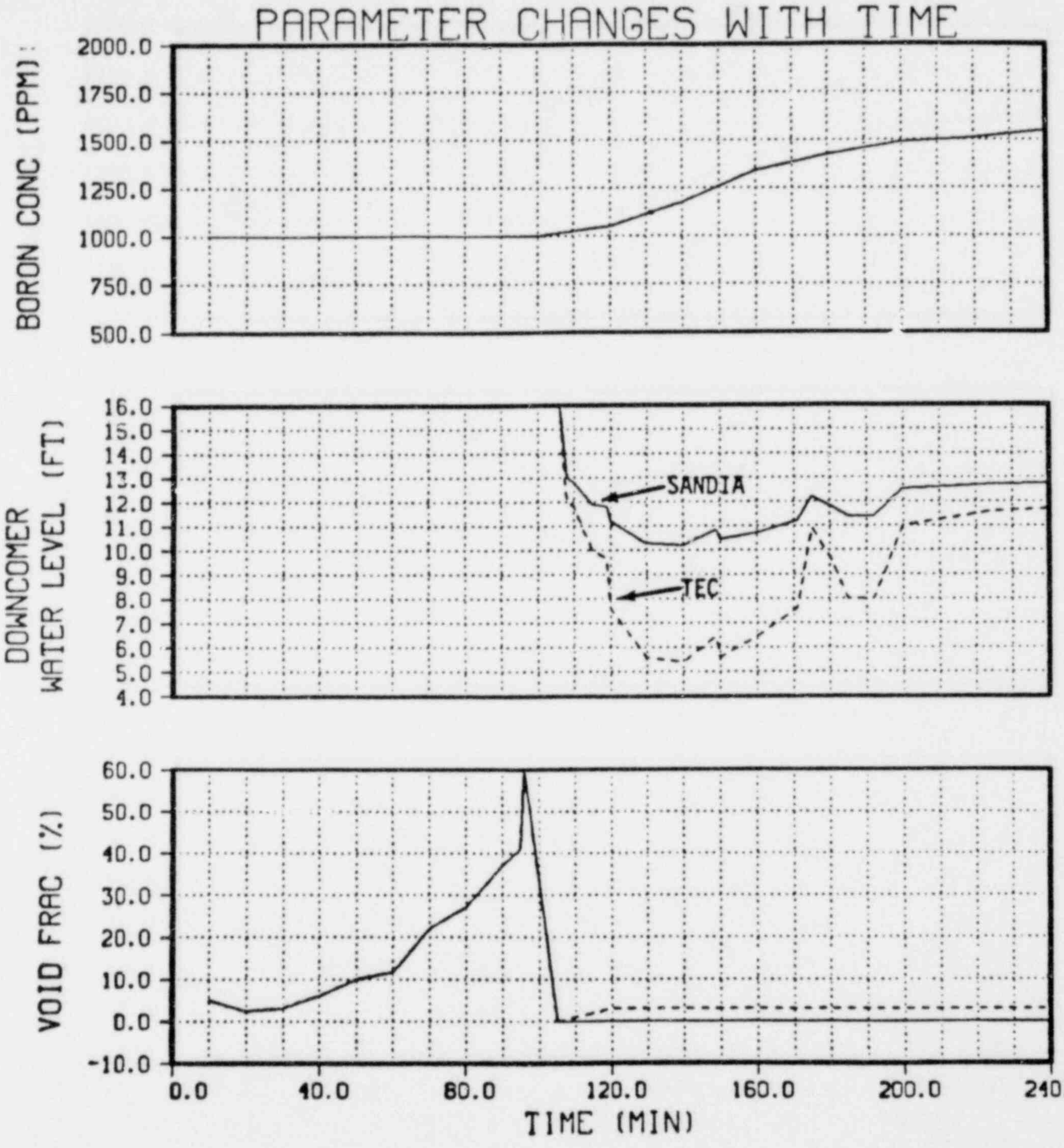


APP. FIGURE II-35. Postulated Core Conditions for Calculating SRM Count Rate Shown in App. Figure II-34 (Sandia Data Base) (Figure 3)

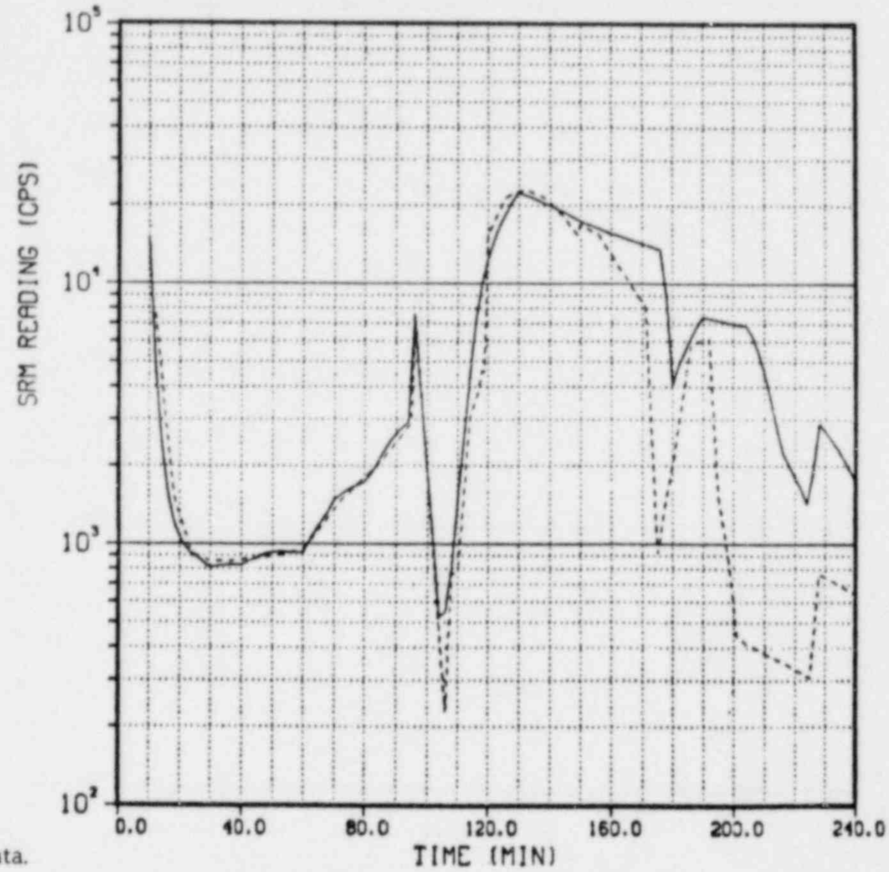
APP. FIGURE II-36. SRM Reading Versus Time Using the TEC Data Base (Figure 4)



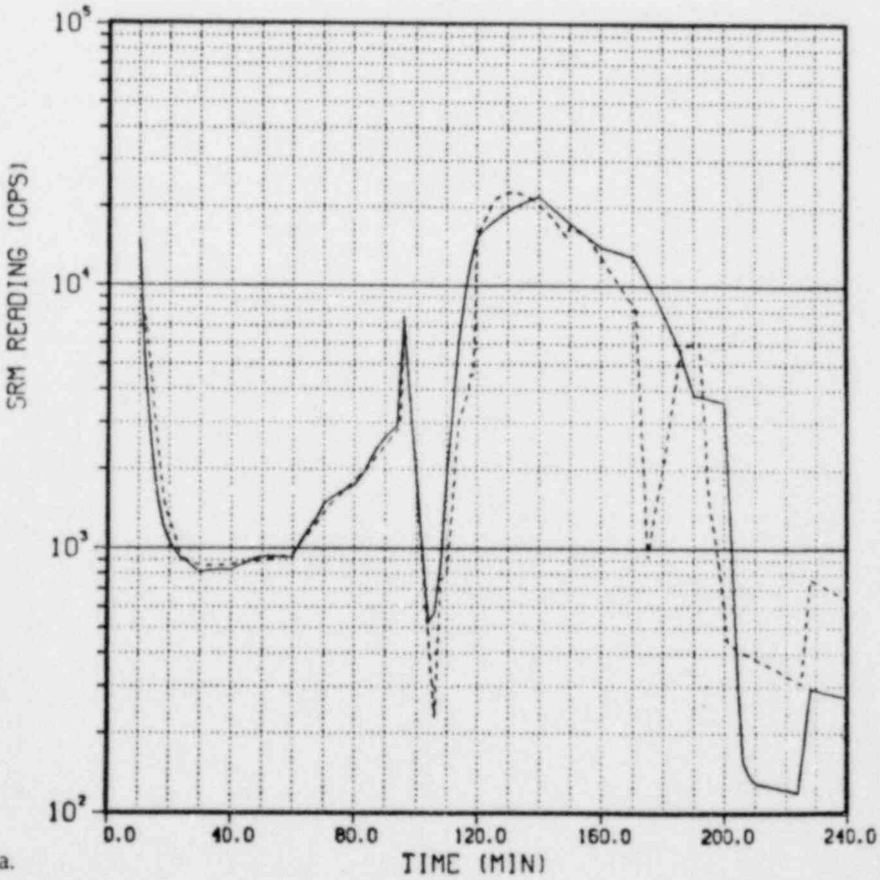
APP. FIGURE II-37. Postulated Core Conditions for Calculating SRM Count Rate Given in App. Figure II-36 (TEC Data Base) (Figure 5)



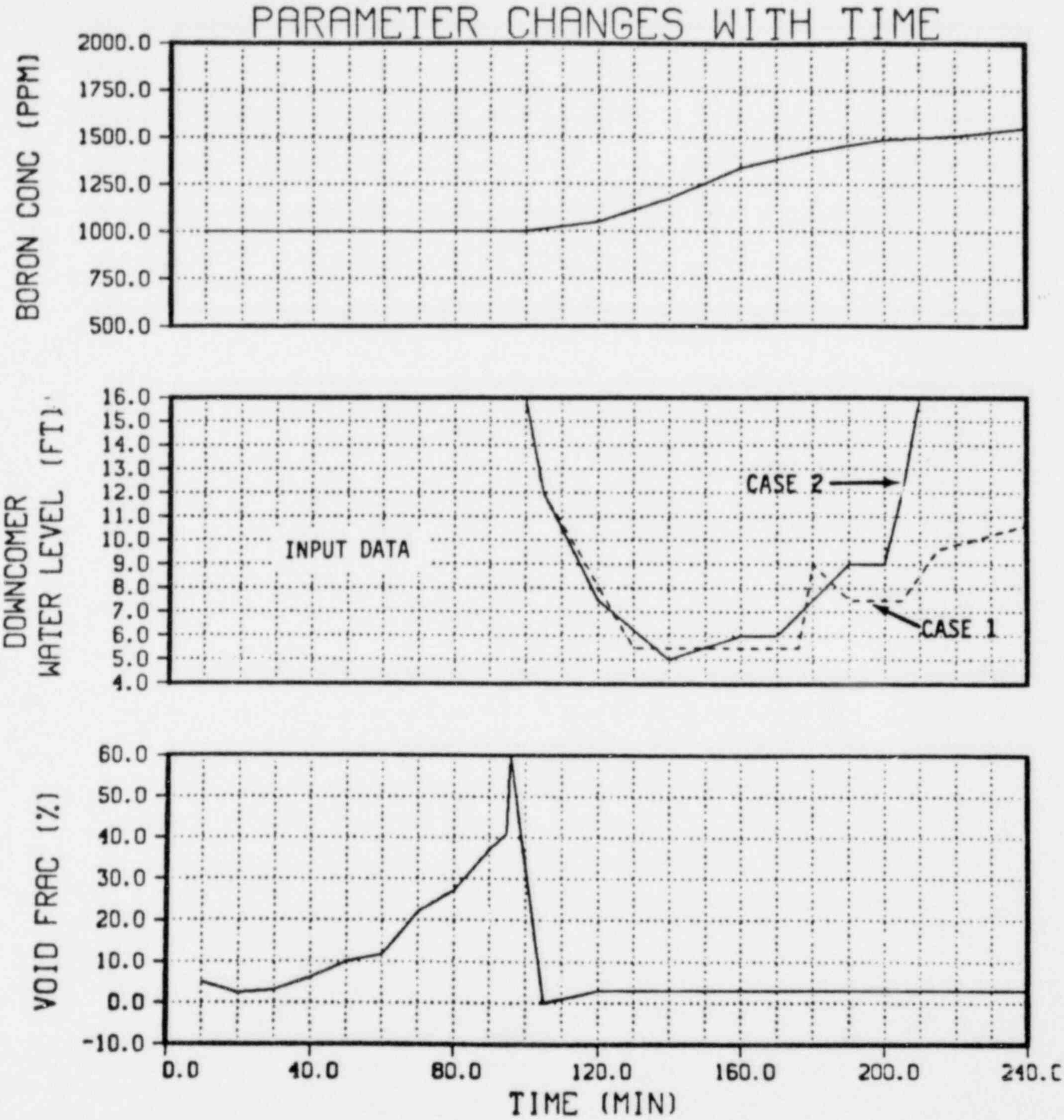
APP. FIGURE II-38. Comparison of Parameter Inputs for SRM Count Rates Shown in App. Figures II-34 and II-36 (Figure 6)



APP. FIGURE II-39.
SRM Reading Versus Time for
Case 1 Water Level Data*
(Figure 7)
*Dashed curve is TMI-2 observed data.



APP. FIGURE II-40.
SRM Reading Versus Time for
Case 2 Water Level Data*
(Figure 8)
*Dashed curve is TMI-2 observed data.



APP. FIGURE II-41. Input Conditions for Calculations Shown in App. Figures II-37 and II-38 (Figure 9)

4. ALTERNATE INTERPRETATION OF THE ACCIDENT SEQUENCE
Sandia Laboratories

A special study group at the Sandia Laboratories, Albuquerque, New Mexico, was asked by the NRC TMI Special Inquiry Group (through the Office of Nuclear Regulatory Research) to conduct a very short term (2 months) examination of the data available on the first 16 hours of the TMI-2 accident on March 28, 1979, to determine, in consultation with the Task Group 2 of the NRC/SIG, if any additional interpretations or aspects of the accident scenario could be developed logically beyond those developed by Task Group 2 and by the MARCH code analysis being conducted at Battelle Columbus Laboratories (BCL). The intent of the request was to try to insure that a minimum of "surprises" would be encountered when the TMI-2 core is examined at some time in the future.

Lacking the time to conduct an intensive investigation independently, the study group was briefed on the interpretations developed at that time, the types and range of "hard" data available in the way of reactimeter data, plant computer and alarm printer, strip and multipoint recorder charts, etc., and furnished with copies for their own examination and analysis.

A summary of their interpretations of the accident sequence is given below, with the addition of a set of figures presenting various system summaries as calculated by M. I. Baskes, Sandia, Livermore, using a new code called "TMI". This interpretation also requires that all of the water removed from the BWST pass through the reactor primary system or be used for repressurization of the system.

Summary of TMI-2 Accident Scenario

Upon stopping both Loop A coolant pumps at 05:41:03±3 a.m. (about 101 minutes after turbine trip), the coolant level above the core subsequently collapsed to a mixture level no less than about 50 inches above the top of the fuel. The Source Range Monitor data suggest that the steam voids, entrained in the liquid by the prior operation of the pumps, separated from the liquid in the downcomer during a period of several minutes. At approximately 05:52 a.m. (112±4 minutes after turbine trip), the core was uncovered for the first time and remained partially uncovered until approximately 06:56 a.m. (176±2 minutes after turbine trip). Reactor coolant pump RCP-2B was restarted at 06:54 a.m. In the 64-minute interval, 05:52 a.m. until 06:56 a.m., maximum uncovery occurred for 38 minutes of the interval (06:10 a.m. until 06:48 a.m.). During this 38-minute period, the collapsed equivalent void-free liquid level in the core was 60 inches, leaving 7 feet (less liquid level swell due to steam due to steam bubble entrainment) uncovered. During this period, the major clad, fuel, and in-core structural damage and ex-core structural damage (if any) occurred.

In a subsequent, approximately 26-minute period, 08:20 a.m. (260 minutes after turbine trip) until 08:46 a.m. (286 minutes after turbine trip), the core appears from computations to have been uncovered to a collapsed equivalent void-free liquid level of 110 inches. Approximately the top three (3) feet (less liquid level swell due to steam bubble entrainment) of the fuel appears to have been uncovered. During this 26-minute period, little, if any, additional significant damage was sustained by the clad fuel and structure. However, uncertainties in the coolant makeup/letdown quantities make this estimate of liquid level uncertain.

In a third period, approximately 02:00 p.m. until 05:30 p.m., it appears from computations that the upper levels of the core might have again been uncovered, but the estimates are uncertain.

Even though additional uncovery of the core might have occurred during the periods, 08:00 a.m. until 09:30 a.m. and 2:00 p.m. until 5:30 p.m., the extent of the uncovery was not likely to have been sufficient to cause significant additional clad, fuel, and structural damage. However, these additional periods during which damaged fuel could have been above the coolant mixture level could account for the continued existence of superheated steam at the tops of the outlet (hot) legs of both Loops A & B, for the period, approximately from 06:00 a.m. until 7:30 p.m.

During the 38-minute period of maximum core uncovery, within the total period of 64 minutes of some core uncovery (05:52 a.m. until 06:56 a.m.), substantial core heatup and clad/fuel/in-core structural damage occurred. Within the maximum units of uncertainty of core heatup estimates (top core temperatures to 3600°F), it is possible that ex-core thermal/structural damage could have occurred to the upper grid assembly and control rod guide tubes, to the top portion of the core basket, to the core support assembly (core barrel) and to the guide lugs, due to loading by the axial thermal expansion of the core support assembly. Constraints by the two diametrically opposite outlet nozzles on the radial thermal expansion of the core support assembly at elevated temperature might have produced a permanent elliptical set to the core support assembly and opened a gap at the vessel/support assembly mating surfaces of the outlet nozzles. Upon core reload, such a gap, if it were caused, would permit coolant to bypass the core.

Using a highly modified version of the computer code, BOIL, calculations indicate that the earliest clad rupture (about 1500°F) occurred approximately 17 minutes (about 06:09 a.m.) after core uncovery. Reactor building radiation monitors, indicating probably gross fuel damage, went off-scale "high" at 06:15 a.m. At 06:18 a.m. the PORV block valve (RC-V2) was closed for the first time. Strip chart recordings of the Self-Powered Neutron Detectors (SPND) subjected to high temperature environments indicated outputs at 06:15 a.m. At 06:48 a.m. (as indicated by the updated Alarm Printer) SPNDs at levels 3 (52 inches above the core bottom) thru 7 (near the top of the core) indicated temperatures deduced to be greater than approximately 1700°F. Likewise, at some indeterminate time prior to 06:48 a.m. (as indicated by the Alarm Printer), a large fraction of the core thermocouples had experienced temperatures in excess of 700°F.

At about 1750°F, an Inconel (grid spacers)/zirconium (clad) eutectic forms and consequently some liquefaction and weakening of the grid spaces in the area of contact would be expected. At increased temperatures, ca 1800°F, the reaction of the steam with the zirconium occurred on the clad exterior surfaces of the control rod guide tubes and the instrument tubes. The estimated quantities of hydrogen produced, ranging from 4-4.5 $\times$ 10^5 grams, imply oxidation of 45-50% of the available core zirconium. If, however, temperatures of the upper grid reached about 2500°F, some fraction of the hydrogen could have been produced by the steam/iron (stainless steel) reaction. The implied oxidation of 45-50% of the available core zirconium is consistent with an estimated 40-50% of core damage derived from measurements of cesium concentrations in the reactor coolant.

The estimated depth (7 ft.) and duration (38 to 64 minutes) of the initial core uncovery was sufficent to expect, within the limits of computational error, upper core temperatures in the range of 3200 to 3600°F. At, or below, these temperatures, $Zr(O) + UO_2$ and $Zr(O) + ZrO_2$ melts occur. At temperatures ca 2600°F, the steam/zirconium reaction power exceeds the decay heat power, thus accelerating the core heatup. As melts form, slumping along the fuel pin surface will occur with resolidification occurring above the coolant mixture level. Due to the fuel pins occupying approximately half (45%) of the core cross sectional area, the resolidified melts could produce a tight crustal zone of fuel pin stub and interstitial eutectics.

At 07:44:00 a.m., approximately 48 minutes after reflood at 06:56 a.m., an anomalous event occurred within the reactor core. The event appears to have occurred spontaneously since no external changes were made to the primary system. Key responses observed were rises in both cold leg temperatures, system pressure rise, SPND responses at levels 1 and 2, and core thermocouple responses. Since prior to this event the coolant liquid level was well above the top of the core, a plausible explanation of the event is localized dryout beneath the impermeable crustal zone conjectured to have occurred during the prior core uncovery. Such a condition would allow superheating of the dry region. Accompanying pressure and/or temperature increases could have caused a breach in the crustal zone followed by a rapid reflood of the previously dried out zone and by a sudden generation of steam. If temperatures in the dried out zone beneath the crustal layer reached ca 2600°F in the presence of zirconium, the power generated by the steam/zirconium reaction could have enhanced the effect of low crustal zone and debris permeability and additional hydrogen could have been generated. One descriptive prognosis of the condition of the TMI-2 core is thus:

- oxidation of up to 50% of the zirconium; lesser quantities of oxidized zirconium would be consistent with the hydrogen mass estimates, if temperatures at the upper grid were sufficient to cause the steam/iron reaction. At such temperatures, ca 2500°F, structural failure and slumping of the stainless steel onto the top of the fuel debris would have occurred.
- the top of the core is extensively disarrayed, consisting of predominantly whole and fractured fuel pellets and stainless steel debris predominantly from the fuel element, end pieces and the upper grid. Beneath the zone of predominantly fuel pellets and fractured pellet debris there would be a crustal zone of eutectic mixtures of zirconium, oxygen and uranium, breached by the event which occurred at 07:44:00 a.m.
- less probable, but possible, extreme temperatures induced structural deformations of the core basket, the core support assembly, and the guide lugs.

Summary

The major features of this Alternate Interpretation are:

(a) all of the water removed from the BWST passed through the reactor primary system or was used for its repressurization,

(b) the top of the core was first uncovered at about 108 minutes, recovered at 174 minutes (2 hr 54 min), uncovered again beteen about 250 and 290 minutes (4 hr 10 min to 4 hr 50 min) and again, between about 740 and 775 minutes (12 hr 20 min to 12 hr 55 min),

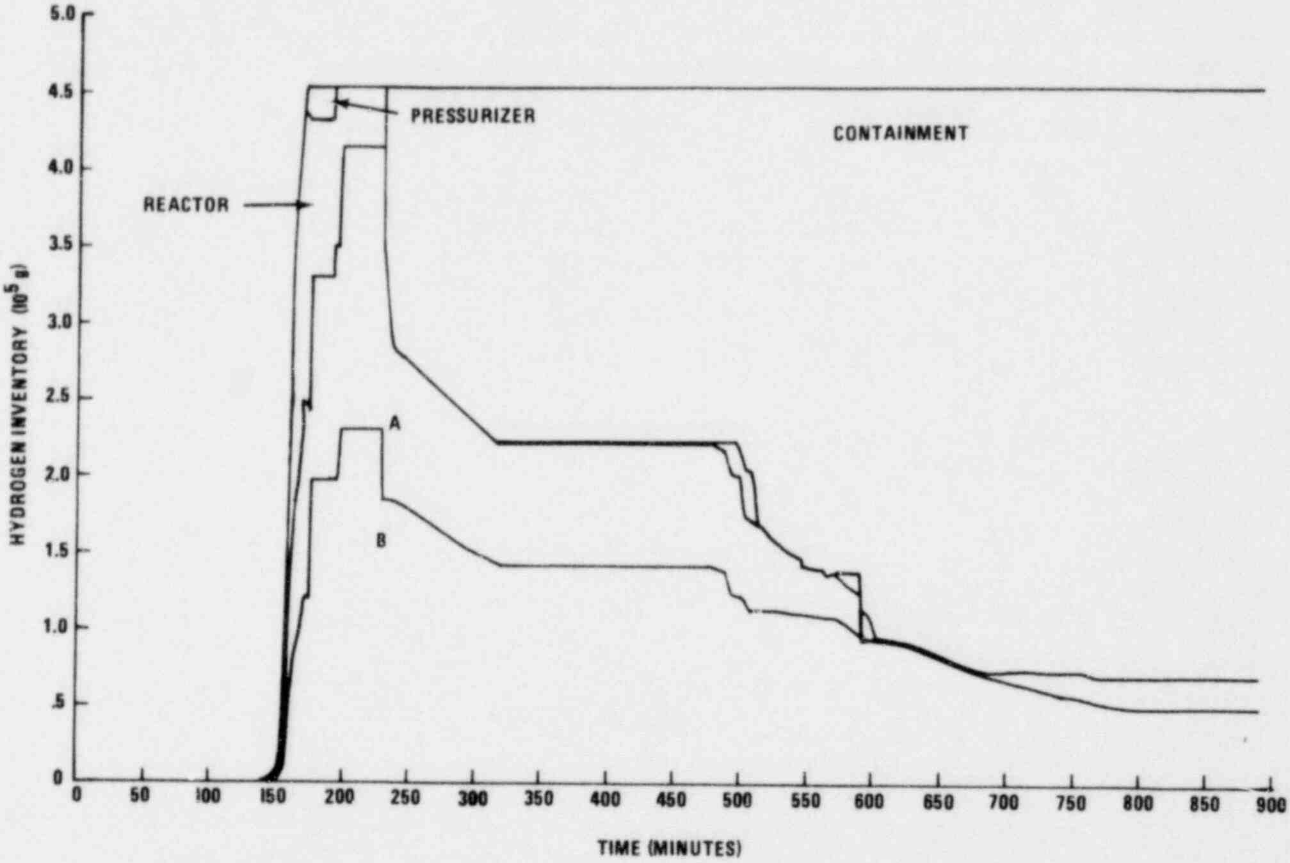
(c) the water level in the core dropped to about 5 feet from the bottom in the first period of uncovering between 108 and 174 minutes (1 hr 48 min and 2 hr 54 min), to about 9 feet from the bottom at about 260 minutes (4 hr 20 min), and to about 10 feet from the bottom at about 750 minutes (12 hr 30 min),

(d) approximately 990 pounds of hydrogen were generated (495 lb-mole), about 1/2 of this was released to the containment between 225 and 315 minutes (3 hr 45 min and 5 hr 15 min), about 1/5 was released to the containment between 470 and 550 minutes (7 hr 50 min to 9 hr 10 min), about 1/7 between about 590 and 600 minutes (9 hr 50 min to 10 hr), and about 1/6 remained in the primary system when the PORV block valve was closed at about 800 minutes (13 hr 24 min),

(e) the major damage to the core occurred by the time the reactor coolant pump was turned on at 174 minutes (2 hr 54 min), but additional damage occurred in the time period around 3 hr 44 minutes, when an impermeable crustal zone formed by melting in the debris bed, sealing off the core and allowing the development of a "dryout" zone below the debris bed. It was ultimately breached by increasing pressure or temperature. The total core damage was estimated to be about 50% of the Zircaloy converted to oxide.

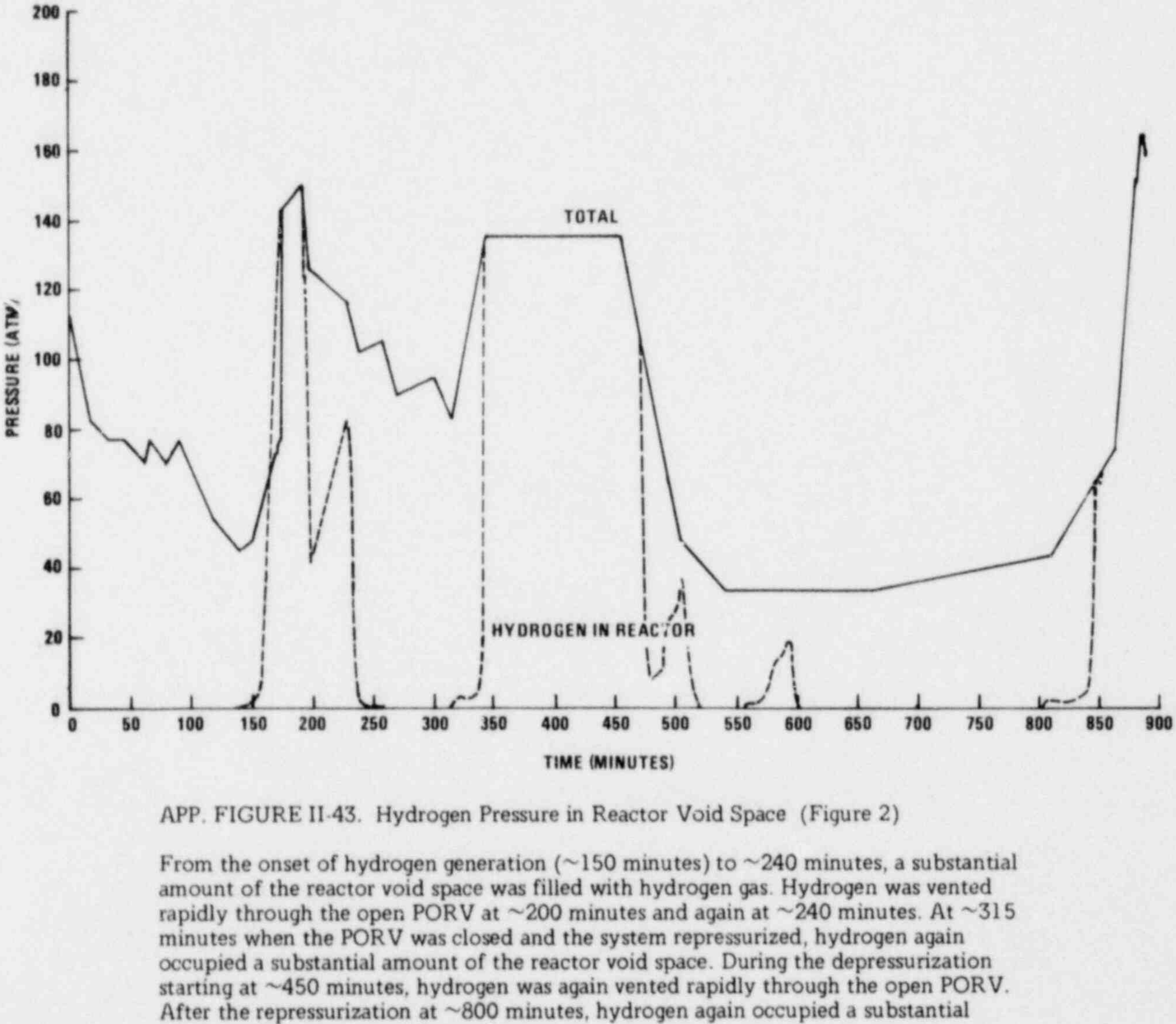
(f) the picture of the core damage is a disarrayed top consisting mostly of whole and fractured fuel pellets covered with stainless steel debris from the upper end fittings and upper grid structure, a lower zone of fuel pellets and fractured cladding, and Zircaloy oxide, containing within it a crustal zone of eutectic mixtures of zirconium, oxygen and UO_2, and a still lower zone of oxidized and embrittled Zircaloy clad fuel rod stubs,

(g) structural deformations may have been induced in the core basket, the core support assembly, and the guide lugs.



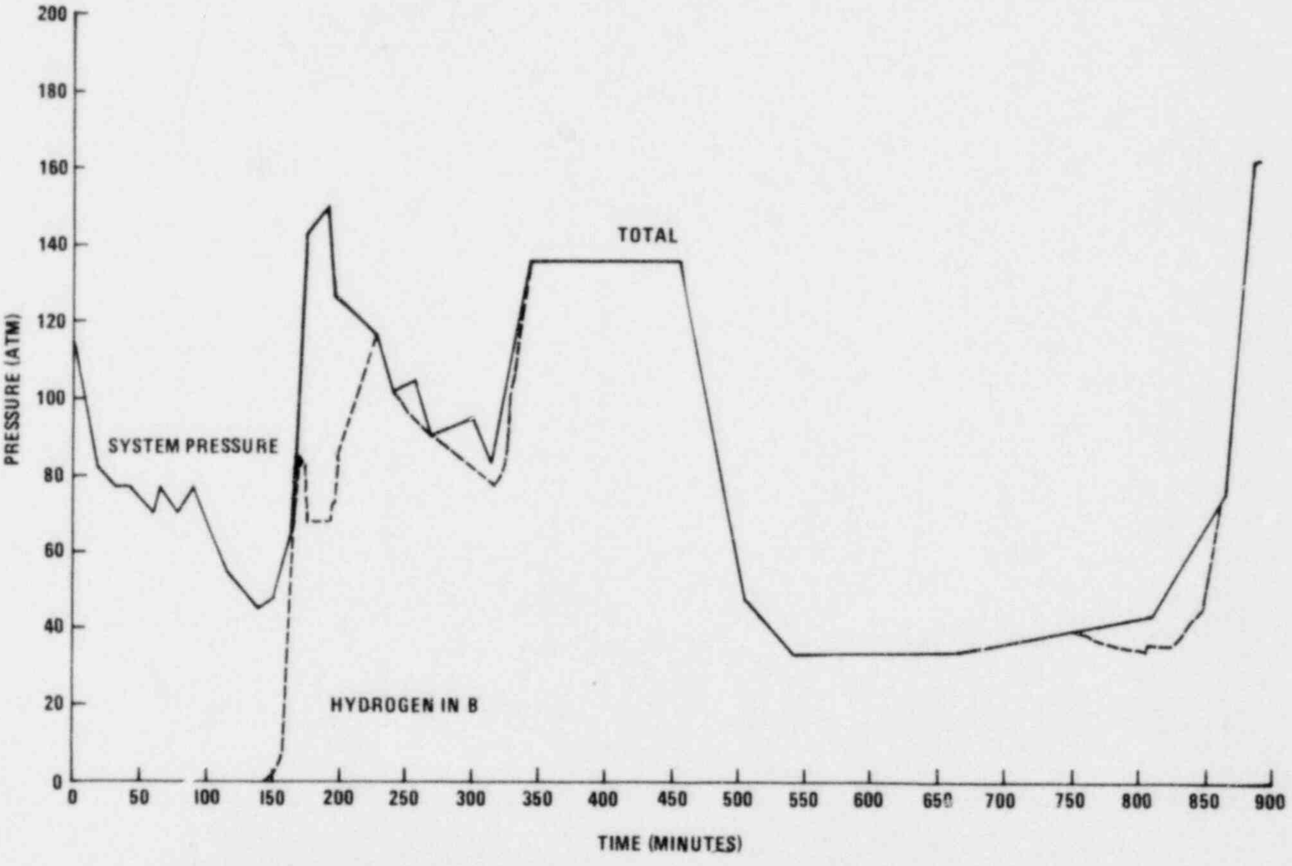
APP. FIGURE II-42. Hydrogen Inventory (Figure 1)

Hydrogen was generated over a short period of time (~150-175 minutes). The model had been "tuned" so that 4.5×10^5 g of hydrogen was produced. When the PORV was opened at ~220 minutes, hydrogen was rapidly vented from the reactor to containment. From ~220 to ~300 minutes the hydrogen venting continued from both A and B steam generators and hot legs. On opening of the PORV at ~450 minutes (system depressurization), the venting continued and resulted in a hydrogen deflagration in containment at ~600 minutes. About 6×10^4 g of hydrogen remained in the system after 900 minutes.



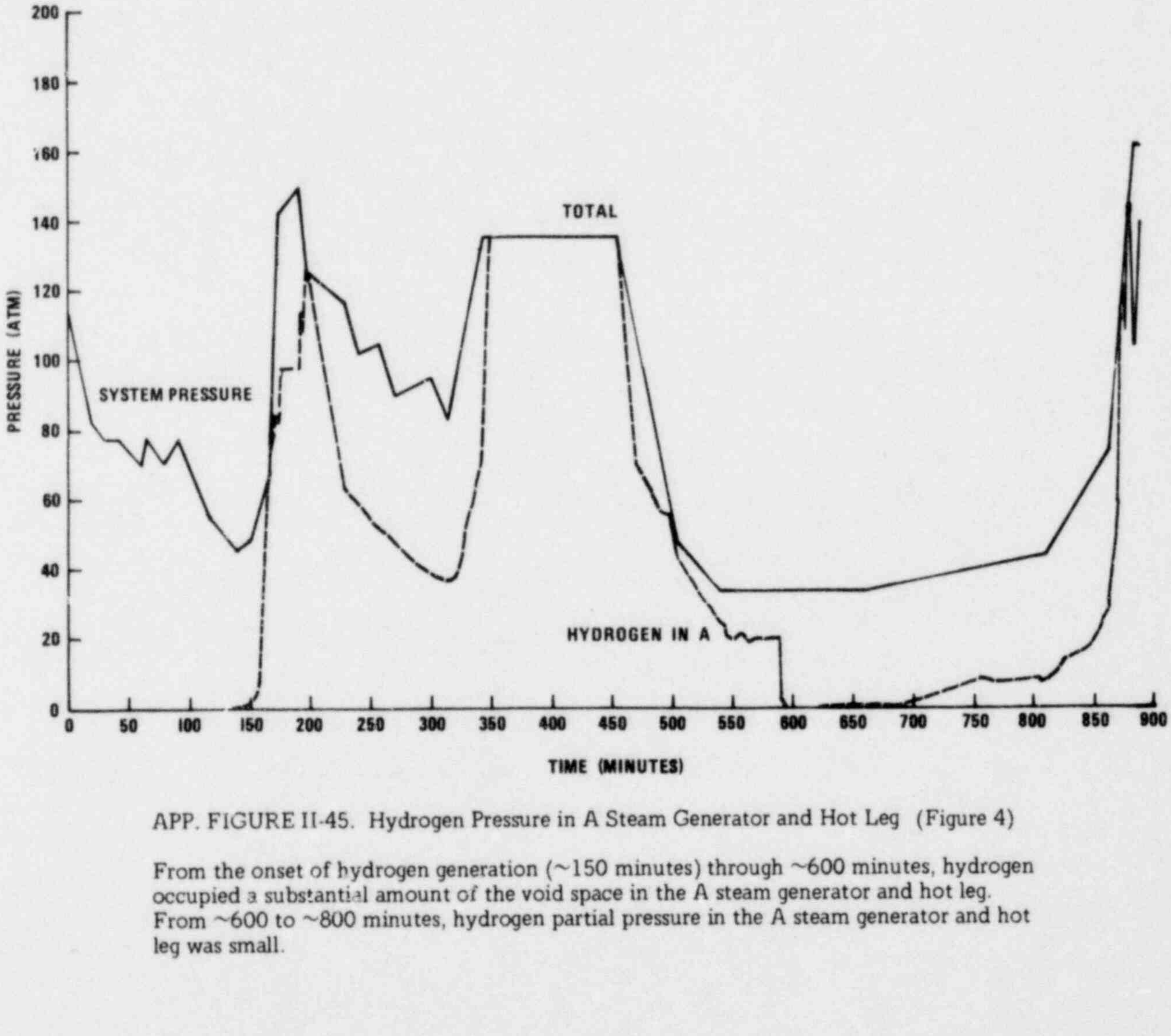
APP. FIGURE II-43. Hydrogen Pressure in Reactor Void Space (Figure 2)

From the onset of hydrogen generation (~150 minutes) to ~240 minutes, a substantial amount of the reactor void space was filled with hydrogen gas. Hydrogen was vented rapidly through the open PORV at ~200 minutes and again at ~240 minutes. At ~315 minutes when the PORV was closed and the system repressurized, hydrogen again occupied a substantial amount of the reactor void space. During the depressurization starting at ~450 minutes, hydrogen was again vented rapidly through the open PORV. After the repressurization at ~800 minutes, hydrogen again occupied a substantial amount of the reactor void space.



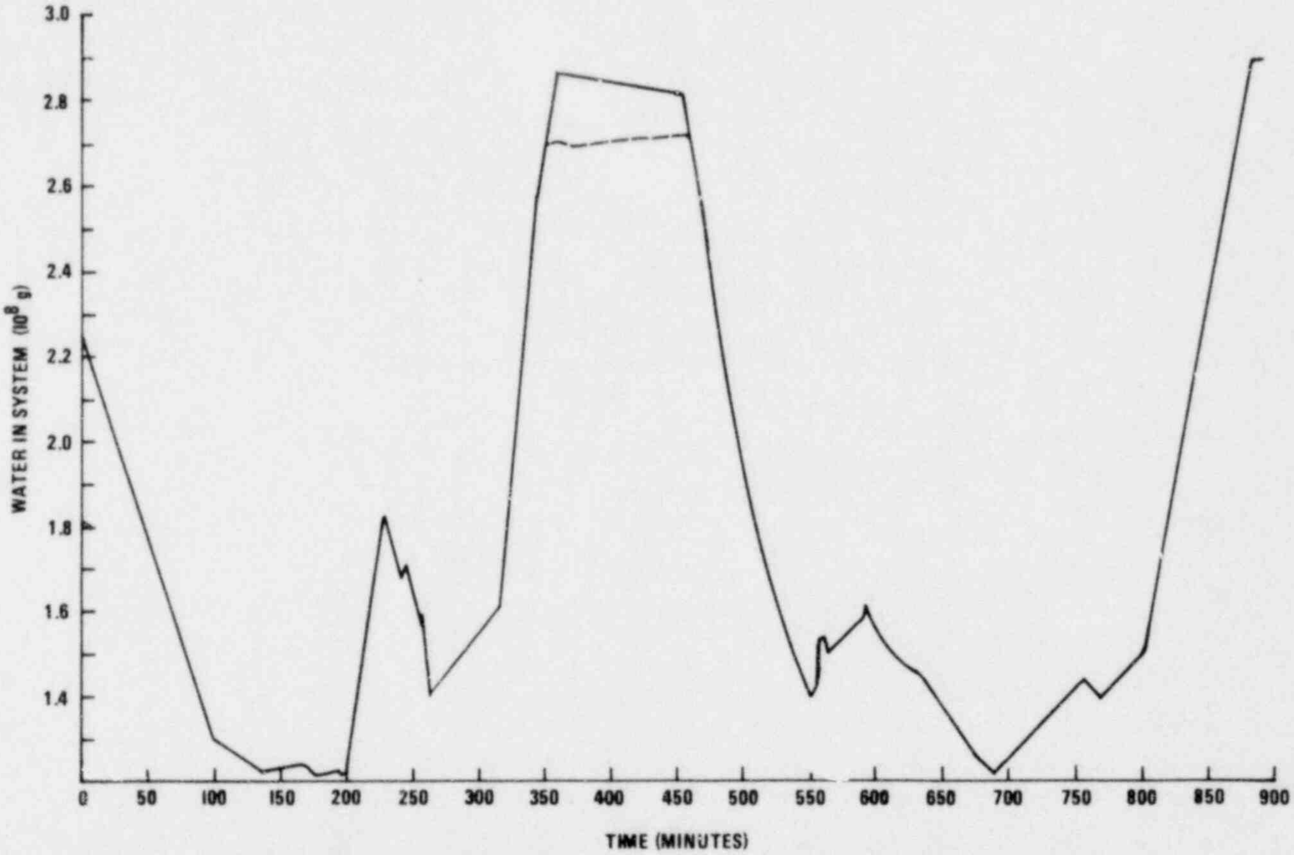
APP. FIGURE II-44. Hydrogen Pressure in B Steam Generator and Hot Leg (Figure 3)

From the onset of hydrogen generation (~150 minutes) through ~900 minutes, hydrogen occupied a substantial amount of the void space in the B steam generator and hot leg.



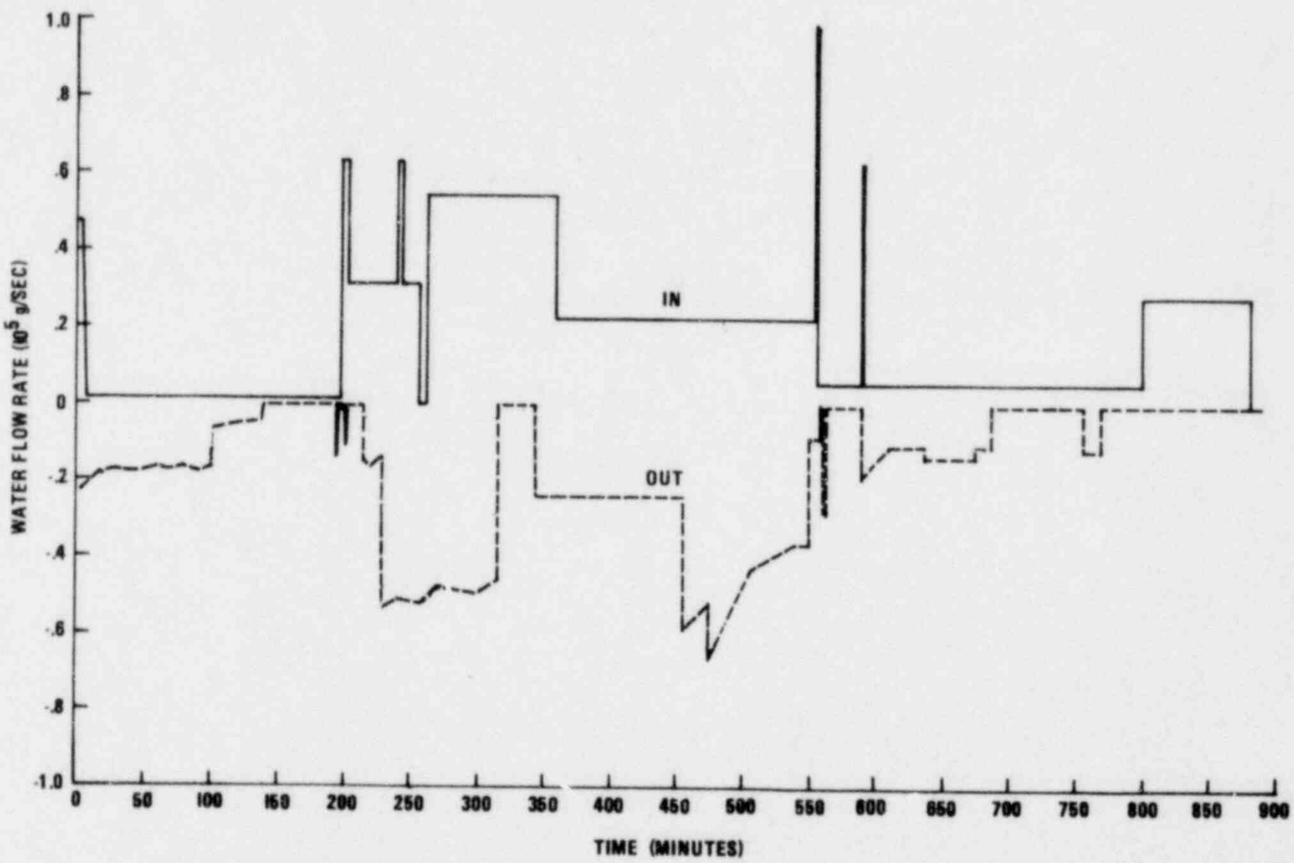
APP. FIGURE II-45. Hydrogen Pressure in A Steam Generator and Hot Leg (Figure 4)

From the onset of hydrogen generation (~150 minutes) through ~600 minutes, hydrogen occupied a substantial amount of the void space in the A steam generator and hot leg. From ~600 to ~800 minutes, hydrogen partial pressure in the A steam generator and hot leg was small.



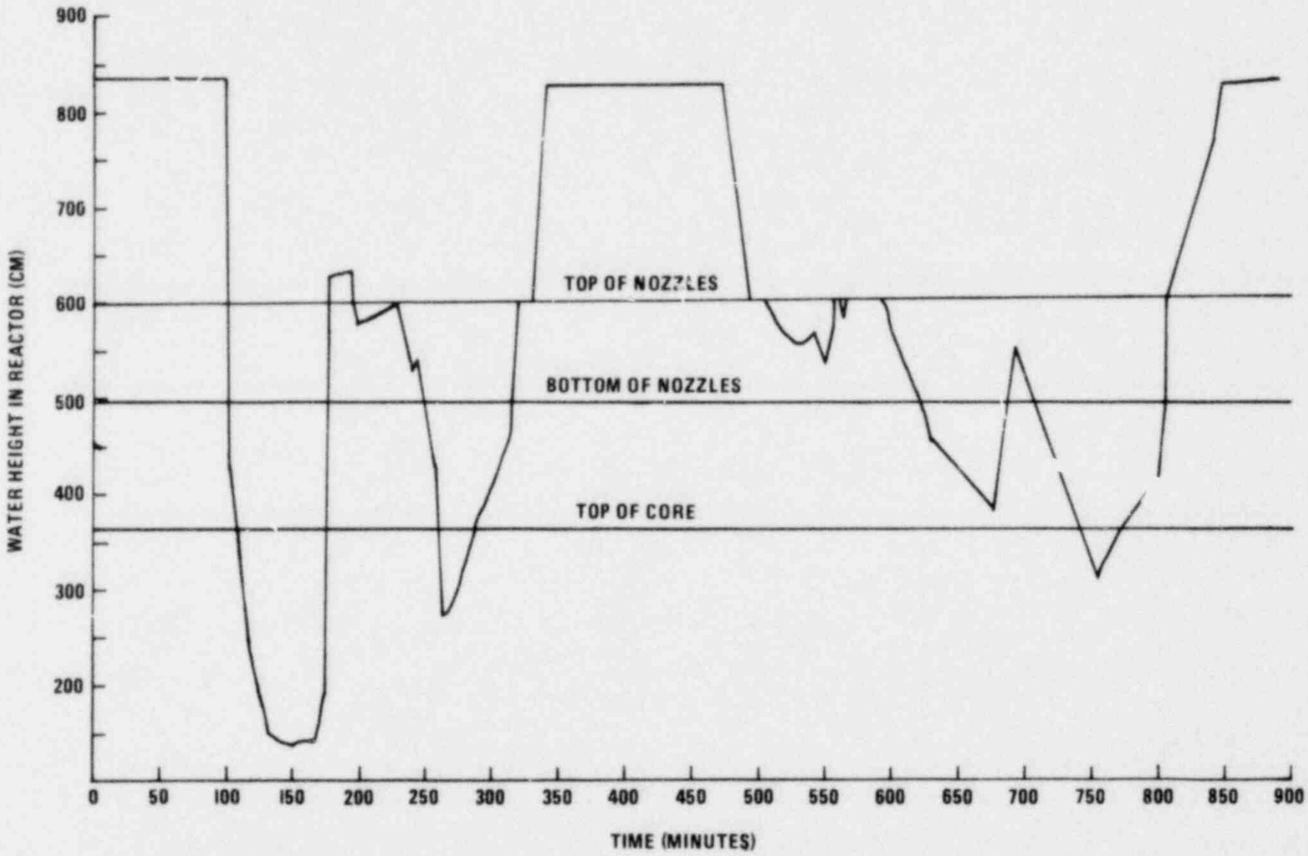
APP. FIGURE II-.6 Water Inventory in System (Figure 5)

The water inventoi 'he system was calculated assuming net makeup/letdown from BWST levels and flow .hrough the PORV and vent valve. A saturated vapor model, a saturated liquid model, or a subcooled liquid model, depending on the recorded pressurizer level, temperature, and pressure was used. The system volume was large enough to contain this amount of water in addition to the calculated hydrogen (at 700° K), except during the short period from 350 to 500 minutes. This excess water could have a number of explanations, for example, a net gain in the makeup tank, a lower average hydrogen temperature, a smaller amount of hydrogen generated, or a larger amount of hydrogen released to containment.



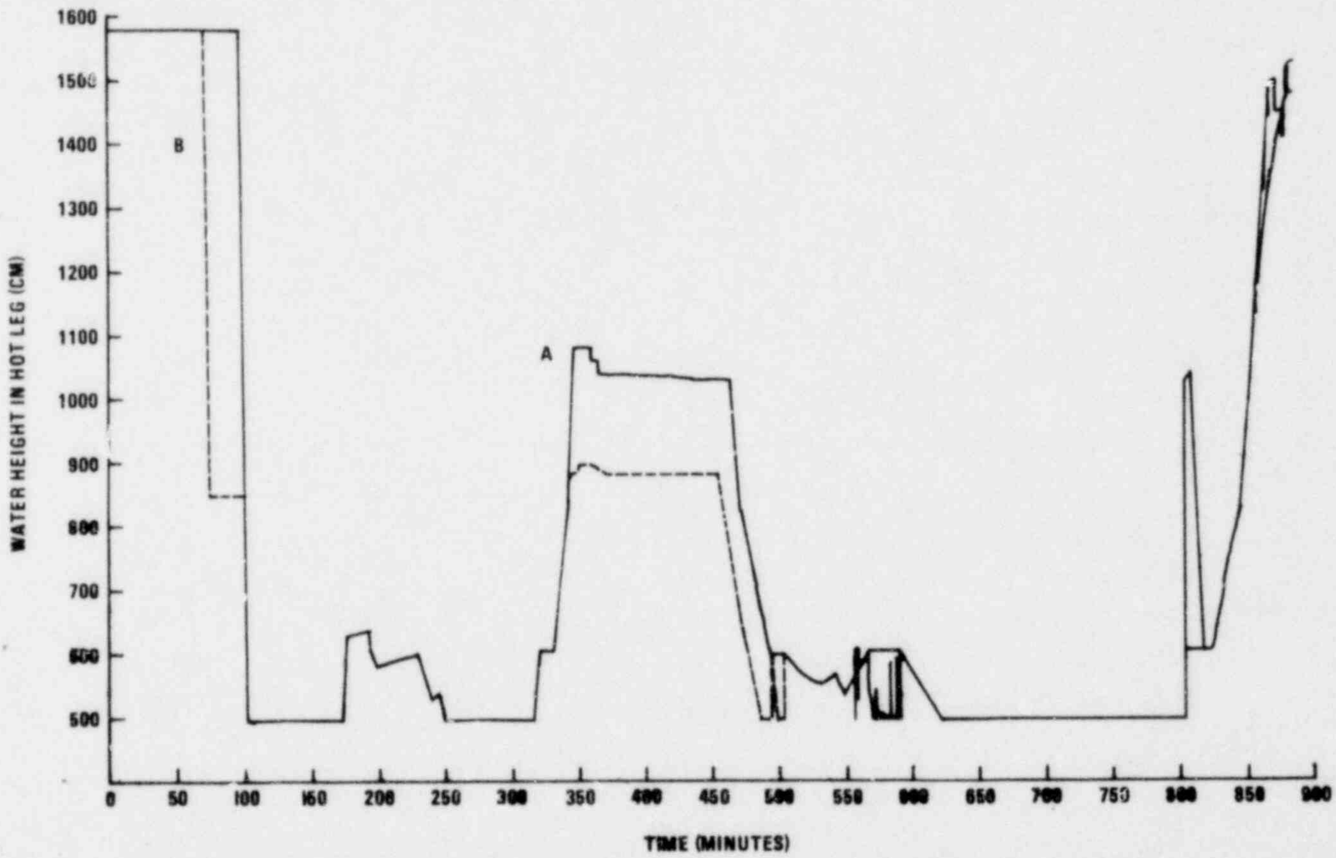
APP. FIGURE II-47. Water Flow Rate, Used To Construct Water Inventory (App. Figure II-46) (Figure 6)

"IN" represents the net makeup/letdown flow from the BWST, and "OUT" represents flow out the pressurizer PORV and vent valve.



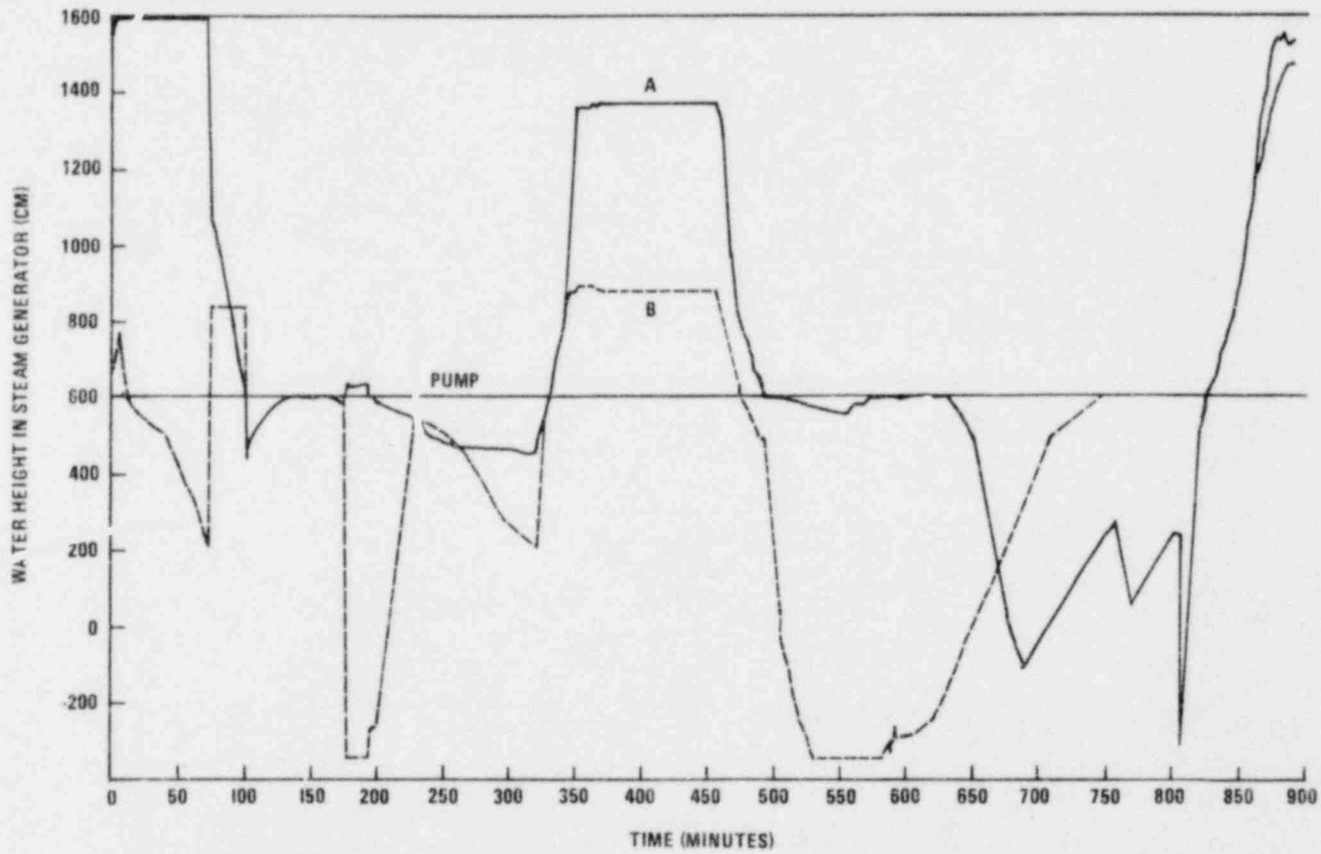
APP. FIGURE II-48. W... er Height in Reactor (Figure 7)

The water level in the reactor fell rapidly at ~100 minutes when the steam generator A RCP was turned off and the voids collapsed. Boiloff then reduced the level to ~150 cm above the bottom of the core. The top of the core was uncovered at ~108 minutes. The starting of the steam generator B RCP and the subsequent HPI recovered the core rapidly. A slight uncovery occurred at ~250 minutes; however the makeup/letdown details could easily have resulted in the core not being uncovered. The reactor was filled at ~350 minutes and remained filled until ~475 minutes when the system was depressurized. The level slowly rose until ~750 minutes when the core was again uncovered. As shown above, makeup/letdown details could easily have resulted in the core not being uncovered. The level rose rapidly as the system was repressurized at ~800 minutes.



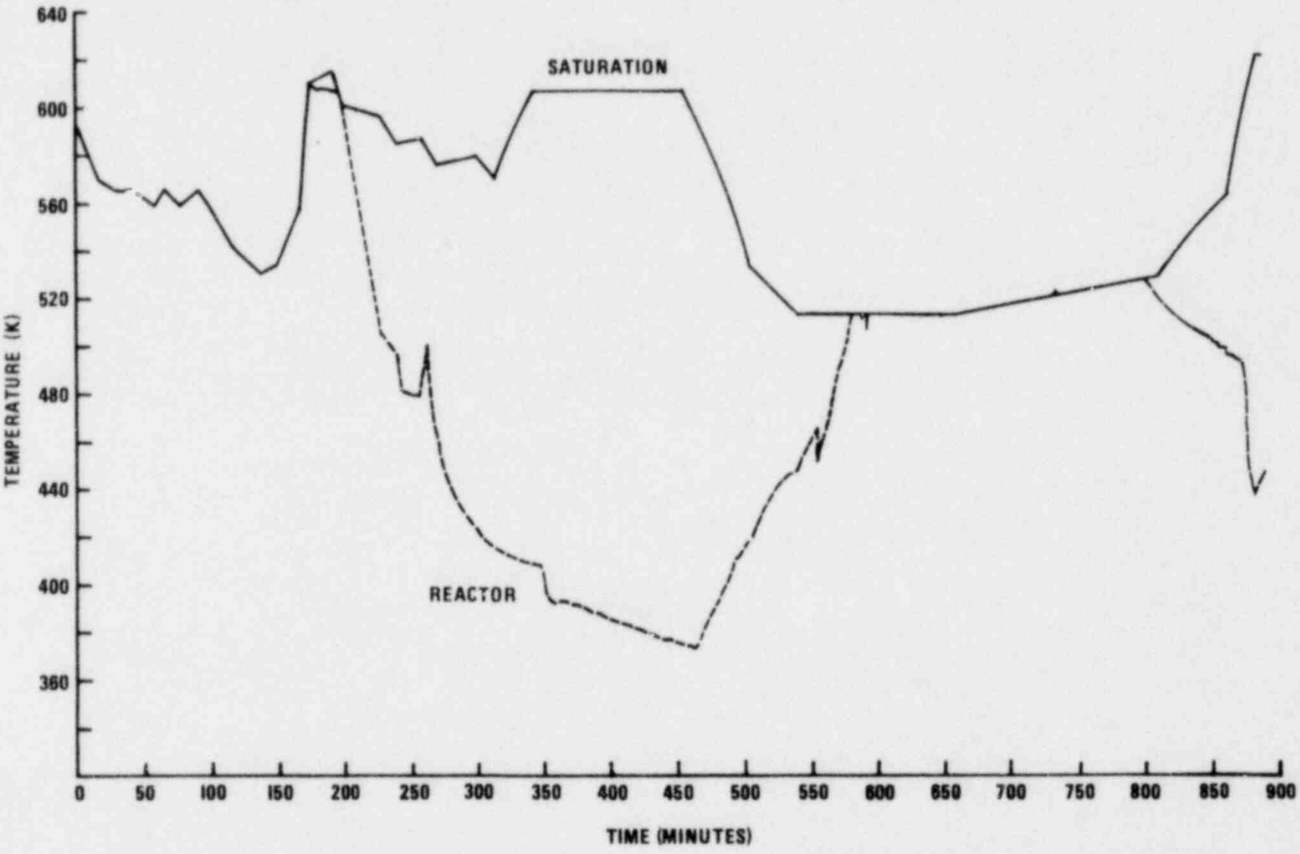
APP. FIGURE II-49. Water Height in Hot Leg (Figure 8)

The hot legs filled with water above nozzle elevation prevented the flow of hydrogen gas for most of the accident. However, from ~200 to ~300 minutes the water level in the hot legs allowed gas flow. In addition, the depressurization at ~500 to ~800 minutes emptied the hot legs and allowed hydrogen redistribution and flow to containment.



APP. FIGURE II-50. Water Height in Steam Generator (Figure 9)

Steam generator water levels generally fluctuated about the pump elevation. An exception occurred at ~175 to ~200 minutes when the steam generator B RCP emptied the B steam generator. It appears that a calculational error emptied the B steam generator rather than the A steam generator during the depressurization at ~500 minutes. When the water level in the reactor was below pump elevation, the A steam generator emptied due to letdown (~600 to ~800 minutes).



APP. FIGURE II-51. Temperature Levels in Reactor (Figure 10)

Until ~200 minutes, cold water flow into the reactor was insufficient to remove decay heat; thus the reactor remained at saturation temperatures with the excess heat being used to vaporize water. From ~200 to ~600 minutes the cold water flow was sufficient to remove decay heat. This removal resulted in subcooling of the reactor. The subcooled reactor water flow through the open PORV during this period resulted in the subcooling of the pressurizer. From ~600 to ~800 minutes the decay heat again became greater than the cooling, and vaporization resumed. The saturated vapor flowed through the hot legs to condense in the steam generators; thus the hot leg temperature was lowered. Because flow to the B steam generator was much less than that to the A steam generator due to hydrogen blockage. A significantly larger hot leg temperature drop occurred in the A hot leg. After ~800 minutes the reactor again became subcooled.

5. STATUS OF THE REACTOR CORE BASED ON FISSION PRODUCT ANALYSIS

D. A. Powers

Coolant water in the reactor coolant system and the reactor sump at TMI-2 contains large quantities of non-volatile and sparingly volatile fission products and fuel materials. This effluent from the reactor fuel can give some insights into the nature and the extent of damage sustained by the reactor core.

The mere presence of the fission products in the coolant water is evidence of fuel cladding failure. However, it is desirable to obtain indications of what regions of the reactor core sustained damage, how extensive was that damage, and what is the state of the reactor fuel following the accident.

The following analyses were attempted to answer these questions:

(a) Isotopic ratios of plutonium and uranium in the sump were compared to the ratios expected for the TMI-2 fuel to determine the regions of the core that were damaged.

(b) The time variations of ^{131}I, ^{134}Cs, ^{137}Cs concentrations in the reactor coolant system were used to determine the rate of coolant leakage.

(c) The inventories of fission products in the sump and the coolant system were used to determine the "prompt" losses of these species and thereby the extent of fuel damage.

(d) The rates of leaching of ^{89}Sr and ^{90}Sr were used to derive the effective particle size of the damaged fuel in the core and to establish a bound on the fine particulate material in the core.

The fuel in the TMI-2 core is enriched in ^{235}U to three different levels: t 2.96, 2.64, and 1.98% ^{235}U. The enrichments are located in the core as shown in Figure 1. The isotopic ratios of Pu and U calculated for fuel assemblies of various enrichments are compared with the ratios found for Pu and U in the sump of the reactor. The observed ratios compare favorably with an average of the expected ratios for fuel initially enriched with 2.64 and 1.98% ^{235}U. If more error is tolerated between observed and calculated ratios, the observed ratios also compare well with expected ratios for a uniformly damaged core.

This comparison of observed and calculated isotopic ratios suggests that the central region of the reactor core was certainly damaged. With less certainty the comparison is consistent with a core damaged across its entire cross-section.

The Pu and U gradients in the sump water are consistent with solids dissolving in the water rather than solids precipitating from the water. This indicates the Pu and U in the sump came either from solids ejected from the reactor coolant system or dissolved species that initially precipitated in the sump and subsequently began to redissolve.

The consistency of the uranium isotopic ratios, with the exception of U^{236} which may be in error, with those expected of the fuel provides no evidence for the intrusion of river water into the reactor sump.

The concentration of ^{131}I, ^{134}Cs, ^{136}Cs, and ^{137}Cs in the reactor coolant system decrease with time even when corrected for radioactive decay. This suggests that little leaching of these materials is occurring. Since they are among the most leachable species in the fuel, it appears that these species were volatilized nearly to completion from the damaged fuel during the accident.

The decreasing concentrations of these species in the reactor coolant system is consistent with leakage of coolant. Constant leak rates estimated from the concentrations of the species are:

Species	*Leak Rate (gal/min)*
^{131}I	1.02
^{134}Cs	1.34
^{136}Cs	1.28
^{137}Cs	1.42
	mean =1.26
std. dev.	std. dev. =0.13

These leak rates apply to the first 43 days after the accident. Concentration data taken from later times suggest that leakage may have slowed considerably.

The inventory of the Cs and I in the sump and coolant system may be used to determine the extent of damage to the reactor core. Results of such calculations are shown below:

Species	*% of Core Inventory in Water*	*Expected Release Fraction*	*% of Core Damaged*
^{131}I	58	1.00–0.95	61–58
^{137}Cs	43	0.89–0.96	48–45
^{134}Cs	36	0.89–0.96	40–38

The expected release fractions listed in the table above were taken from experimental data for melting fuel (G. W. Parker et al. ORNL-3981, July 1967). The computation of "% of core damage" was made under the presumption that the release of the species occurred only from the damaged fuel. Consequently, the result for ^{131}I is an upperbound since iodine would be released from the fuel-clad gap even from relatively intact fuel. The results for cesium are lower bounds since only the sump and coolant system inventories of cesium were considered. The computed extent of core damage based on these fission products are in remarkably good agreement with similar calculations based on the extent of hydrogen formation.

Attempts to calculate the extent of core damage using other isotopes were not fruitful. The range of uncertainty for the "Expected Release Fraction" was too great for other species to provide useful estimates of the percentage of core damage.

The concentrations of ^{89}Sr, ^{90}Sr, and ^{140}Ba in the reactor coolant system increase with time. These species are leaching from the damaged fuel exposed to the coolant water. Leaching data are available for strontium which allow the time dependencies of ^{89}Sr and ^{90}Sr to be used to compute the surface area of the fuel exposed to the water. The surface area to weight ratio so found can be used to derive an equivalent spherical particle radius for the damaged fuel. Results are shown below:

Species	*Surface Area to Weight Ratio†* *(cm^2/g)*	*Equivalent Spherical Particle Size * (cm)*
^{89}Sr	3.0	0.1
^{90}Sr	2.0	0.15

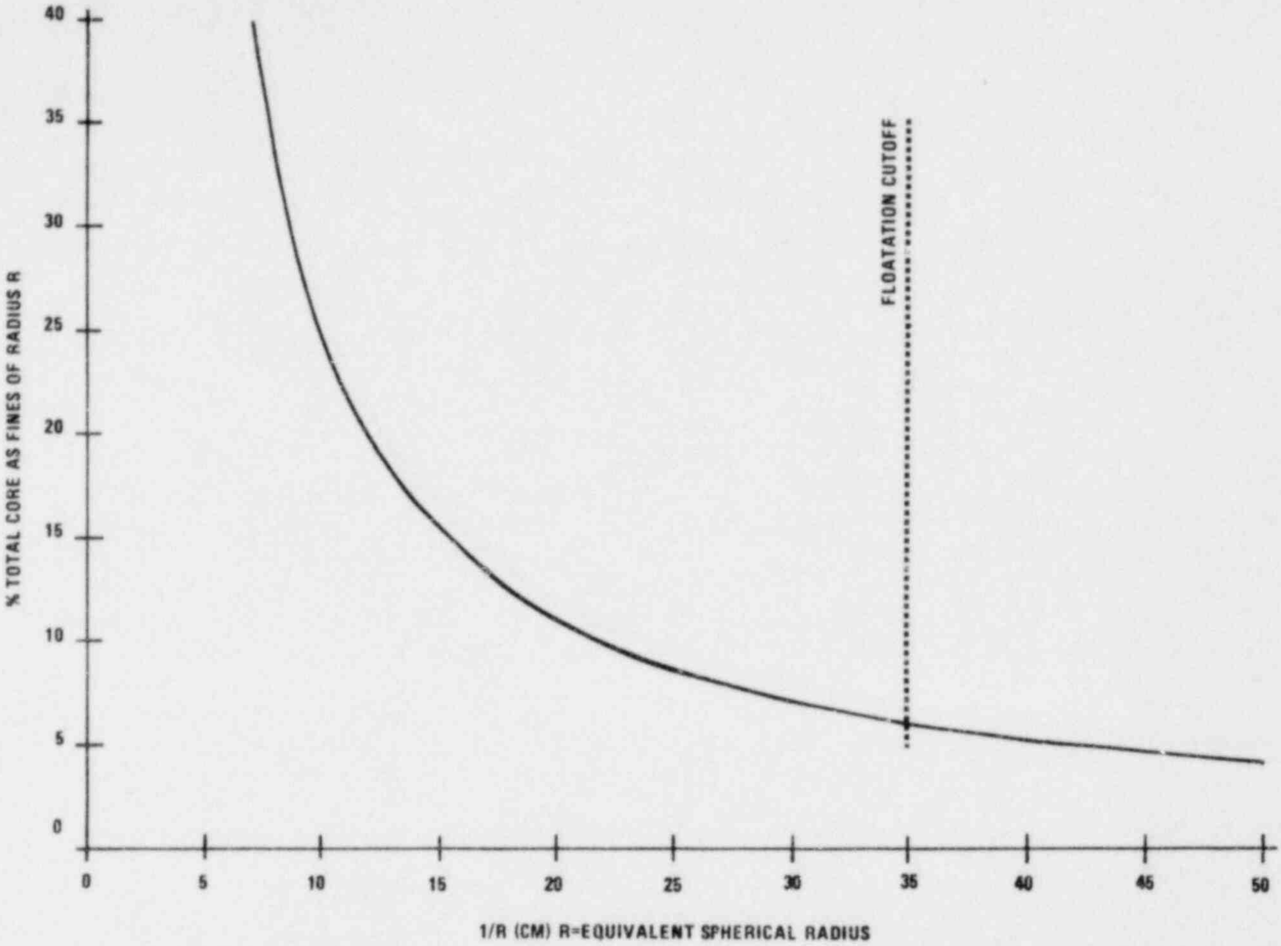
† assumes that 40% of the core is exposed to coolant.

* spherical particle.

If it is assumed that the core debris is made up of intact pellets and fuel fragments of a particular size, an estimate of the volumetric fraction of fine fragments in the core may be made. The volume fraction of the core that could have a size r is plotted versus r in Figure 2. Very fine fragments are not likely to have developed. Fine materials would be levitated by the coolant flow and would have escaped the reactor coolant system to a much greater extent than observed. Consequently, a cut-off in the possible size of the fines of about 0.03 cm is shown in Figure 2.

	% of Isotope relative to total elemental abundance in:					
Species	**1.98% Enriched Fuel**	**2.64% Enriched Fuel**	**2.96% Enriched Fuel**	**Observed in Sump**	**Average of 1.98 and 2.64% Enriched Fuel**	**Core Average**
^{235}U	1.605	2.254	2.572	2.207	1.943	2.156
^{236}U	0.074	0.081	0.083	0.064	0.078	C 080
^{238}U	98.32	97.665	97.345	97.71	97.98	97.763
^{239}Pu	87.916	90.274	91.096	90.6	87.145	89.807
^{240}Pu	9.684	7.97	7.341	7.84	8.790	8.299
^{241}Pu	2.292	1.697	1.497	1.46	1.982	1.818

APP. FIGURE II-52. Comparison of Calculated and Observed Isotopic Ratios (Figure 1)



APP. FIGURE II-53. Percent of Total Core as Fines of Radius (Figure 2)