Staff Reports To
The President's Commission On

THE ACCIDENT AT THREE MILE ISLAND

Reports Of The Technical Assessment Task Force, Vol. I

THE PRESIDENT'S COMMISSION ON
THE ACCIDENT AT
THREE MILE ISLAND

REPORTS OF THE TECHNICAL ASSESSMENT TASK FORCE

ON

TECHNICAL STAFF ANALYSIS REPORTS SUMMARY

SUMMARY SEQUENCE OF EVENTS

VOLUME I

October 1979
Washington, D.C.

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TECHNICAL ASSESSMENT TASK FORCE

ON

TECHNICAL STAFF ANALYSIS REPORTS
SUMMARY

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October **1979**
Washington, D.C.

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INTRODUCTION

The subjects selected for technical analysis are enumerated by the titles of the individual reports. The subjects studied and the scope of the analysis were directed to those aspects of some specific relevance to the TMI-2 accident. The determination of what actually happened (The Summary Sequence of Events) is therefore a most important contribution to all of the studies. In general, the analyses are directed to describing what happened, explaining why it happened, an assessment of the conditions that made the occurrences possible, an assessment of the results of the accident, and to a limited degree, an examination of what might have happened had the accident worsened.

During the course of the staff study, a number of analyses were executed by the Nuclear Regulatory Commission (NRC), General Public Utilities (GPU)/Metropolitan Edison (Met Ed), Babcock & Wilcox (B&W), Electric Power Research Institute (EPRI) and others. All of those made available to the Commission were examined for inputs. As an example, in the generation of the report on the Summary Sequence of Events (SOE), a catalog was generated of events as described by the major studies to insure consideration of discrepancies or differences.

On the latter point, fortunately there exists a fairly extensive and detailed record of events. The best source for information is a "reactimeter" recorder which recorded 24 channels of data at 3 second intervals. This recorder is used principally for the startup phases of plant development and is not a permanent installation nor does it exist in all plants. Other sources of data such as the "process computer" which records plant instrumentation broadly, a line printer, an alarm printer, and strip chart recorders unfortunately were either subject to lapses in data, uncertainties in time, or limitations of range and recording speed. In addition no audio recording or other detailed log exists detailing control room activities during critical phases of the event. This situation causes considerable reliance on interviews and recollections of individuals to fill in voids in hard records.

The following summarizes each of the technical team's reports. Many of the reports are being published in their entirety. Some of the reports were felt to be adequately represented in these summaries and therefore not published. These are all in the Commission's files and will be available in the National Archives.

TECHNICAL REPORTS

- Summary Sequence of Events
- Core Damage
- Thermal Hydraulics
- Chemistry
- TMI-2 Decay Power and Fission Products' ("TMI-2 Decay Power: LASL Fission Product and Actinide Decay Power Calculations for the President's Commission on the Accident at Three Mile Island," by T. England and R. Wilson)
- Containment: Transport of Radioactivity from TMI-2 to the Environs
- Radiation Releases and Venting of Tanks on Friday Morning, March 30, 1979.*
- Alternative Event Sequences
- TMI-2 Site Management.*
- Selection, Training, Qualification, and Licensing of Three Mile Island Reactor Operating Personnel
- Control Room Design and Performance
- Technical Assessment of Operating, Abnormal, and Emergency Procedures
- Simulators - Training and Engineering Design- ("A Study of Simulation and Safety Margins in Light Water Reactors," by S. Levy, Inc.)
- Equipment Conservatism* ("Analysis Report to the President's Commission on the Accident at Three Mile Island on Equipment Conservation," by System Development Corporation.)
- Safety Design Margin* ("A Study of Simulation and Safety Margins in Light Water Reactors," by S. Levy, Inc.)
- Pilot-Operated Relief Valve Design and Performance
- Condensate Polishing System
- Quality Assurance

*Report in the National Archives.

- **Pre- and Post-Accident Security Status at Three Mile Island*** **(Letter to Dr. William R. Stratton from Donald G. Rose, Los Alamos Scientific Laboratories, Sept. 1, 1979: Letter has attached; "Pre- and Post-Accident Security Status Three Mile Island," by Donald G. Rose, LASL. "Three Mile Island Sabotage Analysis," by Eddie R. Claiborne, Richard L. Cubitt, Roy A. Haarman, and John L. Rand.)**

- **Closed Emergency Feedwater Valves**

- **Past Accidents in Nuclear Reactor Facilities* ("Description of Selected Accidents Which Have Occurred in Nuclear Reactor Facilities," by H. W. Bertine, ORNL/NSIC 176.)**

- **Recovery: TMI-2 Cleanup and Decontamination**

- **Cost of Accident* ("Economic Impact of the Accident at Three Mile Island, by Stanford Research Institute, Final Report," September 1979.)**

- **WASH 1400**

- **Iodine Filter Performance**

***Report in the National Archives.**

SUMMARY SEQUENCE OF EVENTS

THE ONSET

At about 4 a.m. on March 28, a loss of feedwater to the steam generators resulted in a turbine trip (shutdown). The interruption of feedwater to, and of steam out of the steam generators substantially reduced the removal of heat from the reactor coolant system. Response to this was a reactor coolant system pressure increase due to the insufficient rate of heat removal; the opening of the pilot operated relief valve (PORV) to relieve pressure, the automatic reactor shutdown (because of the high pressure signal) dropping the heat generation in the reactor to the decay heat level; and a resultant pressure drop to normal values within a few seconds. To this point, normal reactor protection mechanisms functioned as intended by design.

Shortly thereafter two additional problems were experienced. At approximately 40 seconds into the event, the water levels in the steam generators dropped to the point that which automatically called for water to be supplied from an emergency feedwater system in standby just for such occurrences. Valves, erroneously in a closed condition, between the emergency feedwater pumps and the steam generators, prevented water resupply to the steam generators. (These valves were opened 8 minutes into the event.) The opening of the PORV is a normal response to a loss of feedwater whether or not emergency feedwater was available. Upon reduction of reactor coolant system pressure, the PORV should have closed. Instead it remained open, undetected for 2 hours and 20 minutes allowing a continued loss of coolant from the reactor coolant system, until an upstream block valve was closed at 142 minutes into the accident.

Indications in the control room of the open PORV were ambiguous in the minds of the operators. The valve position light on the control panel indicated that the valve was closed but it only indicates electrical power applied to an actuation solenoid in the valve and not valve position. High temperature readings downstream of the valve were considered ambiguous because of the known opening of the valve a few seconds into the event and because of the existence of excessive temperatures that were existent prior to the event due to leakage through one or more valves in that portion of the system. The pressure of the reactor coolant drain tank into which the escaping coolant was flowing could have been used as an indication of an open PORV but this pressure indicator is located on a back panel in the control room not immediately available to the operators.

The accident could have been terminated with little or no damage to the core by closing the PORV blocking valve as late as 100 minutes into the accident and by maintaining pressure in the system above saturation pressure with the high pressure injection pumps.

The high pressure injection pumps were turned on automatically at about 2 minutes into the event in response to a low pressure (1,640 pounds per square inch gauge (psig)) in the reactor coolant system. At approximately 4-1/2 minutes into the accident, the operator turned off one of the two high pressure injection pumps and the flow from the remaining pump was cut back in response to a high coolant level in the pressurizer. At this reduced injection rate, coolant was flowing out of the system through the PORV and through the coolant let down system faster than it was being resupplied. This situation persisted until full high pressure injection was reinitiated at about 3 hours and 40 minutes. During this period steam voids accumulated in portions of the coolant system other than the pressurizer which negated the use of pressurizer water level as an indication of total coolant in the system. The operators relied upon pressurizer level for assurance of coolant coverage of the reactor core.

For the first 73 minutes all four reactor coolant pumps operated and circulated coolant through the reactor. The open PORV continued to discharge coolant and the coolant system pressure continued to drop increasing the amount of steam in the coolant. At about this time, the fraction of steam (gaseous voids) in the coolant reached the point where it caused high vibration of the reactor coolant pumps. To avoid damage to the pumps the operators turned off the B loop pumps. The A loop pumps continued to circulate coolant through the reactor until about 100 minutes when these pumps were turned off, again because of high vibration and fear of damage to the pumps.

The vibration was due to the mixture of steam and water in the system which caused cavitation in the pumps, but circulation of the mixture had continued to cool the core. When the circulation was stopped the steam separated from the water, i.e., it rose to the high points in the system and the coolant that was left in the lower portions of the system was insufficient to cover the core. Approximately 10 minutes later (at 111 minutes) the reactor coolant outlet temperature began to rise rapidly and in another 38 minutes (149 minutes) the measurements of temperature went off-scale at 620°F. These temperatures indicated a superheated steam environment in the system. It was during this period that portions of the fuel cladding reached temperatures high enough (about 2,000°F) to allow the zircaloy cladding to react with steam to produce hydrogen gas.

<u>ESTABLISHING CONTROL</u>

After the PORV was discovered open and the block valve closed at 2 hours and 22 minutes (142 minutes) attempts were made to reestablish a stable cooling situation.

For about 5 hours attempts were made to establish some circulation so that heat could be removed through the steam generators. Attempts to establish forced circulation or natural circulation were unsuccessful due to the noncondensable gas, hydrogen, in the cooling system. Pockets of gas blocked the flow. Reactor coolant system pressure rises due to temperature increases and attempts at high pressure injection called for opening of the PORV block valve numerous times. Over the next 4 hours attempts were made to reduce pressure sufficiently to effect core flooding and heat removal through the low pressure decay heat removal system. Pressure was lowered again by opening the PORV block valve and during these operations a large fraction of the hydrogen was vented to containment. The reactor coolant system pressure, however, remained too high to initiate cooling using the decay heat removal system.

Another 2 hours passed when at approximately 13-1/2 hours a sustained high pressure injection was made repressurizing the system and a reactor coolant pump was successfully started. This reestablished forced circulation of coolant and made possible subsequent heat removal from steam generator A.

REMOVAL OF HYDROGEN

At about 9 hours 50 minutes into the event, the concentration of hydrogen vented to containment became high enough in some portion of the building to support combustion. It ignited resulting in a measured 28 pound per square inch pressure pulse. This pressure pulse is well within the capability of the reactor building.

The hydrogen gas bubble (or bubbles) which formed in the top of the reactor vessel (and perhaps at other high points in the system) and which had blocked the flow of coolant, was gradually removed over the next week. This was accomplished by forcing some of the hydrogen into solution at high pressure and temperature and then releasing it from solution as coolant was returned to the make-up tank via the letdown system, and by spraying coolant into the pressurizer and then venting the pressurizer.

With cooling reestablished and the hydrogen removed from the coolant system, one had to wait only for the decay heat to reduce to the point where natural circulation could be established and the reactor coolant pumps could be turned off. This took place on April 27.

The plant is in a "cold shutdown" condition which means that the temperature of the coolant is below 200°F. In this condition circulation requirements are minimal but must be continued. A leak at this temperature would spill only liquid coolant, i.e., steam would not form. At some time a low pressure decay heat removal system may be employed.

RADIOACTIVE RELEASES

Coolant escaping from the open PORV was piped to the reactor coolant drain tank from which it subsequently escaped through a ruptured pressure disk. This coolant drained into the reactor building sump. It was low in radioactivity prior to 6 a.m. on March 28. Some of this coolant was pumped from the reactor building sump to the auxiliary building where it flowed out of a tank with a previously ruptured pressure disk and onto the floor. (This flow was terminated at 4:39 a.m. when both sump pumps were turned off.) Radiation survey measurements made just prior to 6:30 a.m. showed normal levels. Shortly after 6:30 a.m., the radioactivity levels in the auxiliary building began increasing, climbing toward 1 rem (R) per hour. At approximately the same time a radioactivity monitor on a reactor coolant sample line also indicated a rapid rise in radioactivity. A "site emergency" was declared at 6:55 a.m. and this was communicated to civil authorities.

Just after 7:00 a.m., radiation monitors **in** the reactor building, the auxiliary building, and the fuel handling building all started increasing rapidly causing a "general emergency" to be declared at 7:24 a.m. Radiation levels off the site did not rise above 1 millirem (mrem) per hour until after 9 a.m. Radiation levels experienced off-site are documented in the Health Physics and Dosimetry report.

Reactor building containment was initiated at 3 hours and 55 minutes into the accident (7:56 a.m.) on a signal of 3.2 pounds per square inch building pressure. The possible escape routes for radiation are complex and they are discussed in the report on containment.

At 7:10 a.m. on March 30 the operator on duty chose to vent the make-up coolant tank to relieve pressure and to preserve the inventory of coolant for later use. The argument was that if pressure continued to rise the coolant would have been forced out of the tank and onto the floor. This action had been preceded by other ventings for the same purpose. The tank was vented to a vent gas header of the waste gas system which was known to leak to the atmosphere. This resulted in a measurement from a helicopter of a 1,200 mrem/hr pulse at the stack. This vent was left open for days. As a result of this high (1,200 mrem/hr) measurement the NRC staff advised the Pennsylvania Emergency Management Agency (PEMA) that evacuation was in order. The knowledge of the radioactivity to be released, the coordination of the intent to perform the venting, and the information (and its sources) used by NRC in considering the impact of the measurements are all questionable.

The plant is in.a stable condition and no further uncontrolled releases are expected. To this date, no one has declared the end of the emergency at TMI-2.

EVALUATION OF PERFORMANCE DURING CRISIS

In conjunction with critical actions taken during the accident, the Summary Sequence of Events contains an evaluation of the actions relative to the information available at the time and the actions that should or could have been taken to terminate or otherwise reduce the effect of the accident.

Clearly the misunderstanding of pressurizer level and the notion instilled as a result of training that running the system "solid" (i.e., totally full with liquid) was undesirable caused the operators and those responding early in the accident to provide assistance, to take actions that instead of terminating the event led directly to a sufficient loss of coolant to cause core damage. Among such actions that were contributors were:

- The failure to recognize the failed open PORV and to take action to isolate it after reactor coolant system pressure continued to fall, the reactor coolant drain tank pressure disk had blown, and the reactor building sump pump operated indicating large quantities of water in the containment building sump.

- The throttling of high pressure injection for the first 3-1/2 hours of the accident.

These actions were a result of:

- Persistant disbelief of high temperatures measured downstream of PORV as an indication of PORV failure to close and a failure to look for corroborating information.

- Inattention to high temperature data from in-core thermocouples.

- Failure to recognize that a high pressurizer level did not ensure coverage of the core.

These are a few of the examples sited in the Summary of Events. The reasons that provided the environment conducive to failure to take correct action and for misinterpretation of information are many and they are found in other portions of this report such as Personnel Training & Qualifications, Management, Simulation Adequacy, Quality Assurance, as well as in the studies of individual components of the system.

CORE DAMAGE

The true extent of the core damage in TMI-2 will not be known until the reactor pressure vessel is opened and the core can be inspected. Any current picture of the core must be a result of analysis of the history of core uncovery and temperature excursions, thermal hydraulic analysis to estimate the history of core uncovery and temperatures realized, and analysis of the fission product releases. From all this material the following judgments are drawn:

- 90 percent or more of the claddings of the fuel rods have probably burst.
- Of all the zirconium cladding, 44-63 percent has been oxidized. The upper 60 percent to 70 percent has lost its structural integrity.
- Fuel temperature exceeded 3,500°F throughout the upper 40 percent to 50 percent of the core. Fuel temperature may have exceeded 4,000°F in 30 percent to 40 percent of the core volume.
- Some of the uranium dioxide fuel may have become liquid at temperatures well below its melting temperature of about 5,200°F due to the formation of a molten partially-oxidized zirconium at about 3,450°F. The uranium dioxide fuel can dissolve into this where it comes in intimate contact with it. It is estimated that the total amount of fuel that melted was small.
- Continuing leaching of radioactive products into the cooling water indicates that some of the fuel may be in finely divided form.
- A section of the core probably fell downward at 226 minutes into the event as a consequence of earlier damage. This is indicated by a rapid change in the readings of both incore and excore neutron detectors at that time.
- Portions of the control rods probably melted, but the constituents of those rods, not being soluble in water, are likely still in the core. Silver from the control rods has been detected in the precipitates from the water in the containment sump.
- The core is not close to becoming critical, even if the control rods (poisons) are somehow removed from the core, as long as the cooling water contains a boron concentration of at least 3,180 parts per million. At present Metropolitan Edison is maintaining a 3,500 parts per million boron concentration in the TMI-2 coolant.

THERMAL HYDRAULICS

This report is concerned with both the flow of water and steam throughout the reactor loops and with the ability (or inability) of these fluids to remove heat from the nuclear reactor. Emphasis is placed on those situations causing trouble or leading to over-heating of the reactor. The term "water" as used herein always refers to water without voids.

The principles for keeping the reactor cool after the reactor is shutdown are simple: (1) keep the reactor full of water; (2) circulate that water throughout the reactor loops; the circulation can be produced by natural convection or by pumping; and (3) provide a heat sink, that is, a place to dump the heat. The heat sink can be supplied by either the steam generators or by injecting water via high pressure injection (HPI) that is boiled and then discharged through the relief valves (either the PORV or the safety relief valves).

The purpose of the study is to assess the ability of the system to deal with and dispose of the heat generated. To do this requires an assessment of the water inventory, its status and its levels in various portions of the system with time.

To do this, a theoretical study was conducted by Los Alamos Scientific Laboratories (LASL) using their TRAC computer code. This computer study used the best information available on the imposed operating conditions (such as HPI flow, reactor coolant pumps on or off, etc.) and estimated the reactor's thermal history. The peak fuel temperature computed was 3,900°F, and the overall results generally corroborate earlier calculations by Picklesimer. This analysis was supplemented by consultants at MIT who examined the general conditions of flow and heat transfer during the period when the reactor was only partially filled with water; they also considered the potential impact on the accident if certain events had not taken place.

An important result of this effort is the confirmation that the TRAC code calculations do in fact reproduce the TMI-2 events, at least up to 3 hours when severe damage to the core occurred.

The report contains the following additional findings:

1. Thermal-hydraulic analysis of the TMI-2 reactor loop by means of the TRAC computer code accurately reproduces the observed operating conditions for about the first 3 hours. Toward the end of this period, a peak fuel temperature was calculated at about 3,900°F. (The Alternative Events study shows that under possible conditions, temperatures may have reached 5,162°F.)

2. At 101 minutes after start of the accident, the inability of the reactor coolant pumps to pump a water-steam mixture having a very high proportion of steam made it necessary to turn off the pumps. Stopping the pumps interrupted the reactor cooling provided by this two-phase mixture, and the reactor fuel elements rose in temperature to 3,500-4,000°F.

3. When the reactor coolant pumps were stopped, water was trapped in the lower portion of each steam generator. The geometry of the reactor loop prevented this water from draining into the reactor vessel and cooling the reactor.

4. Failure to maintain always a pressure (and thereby temperature) in the secondary side of the steam generator lower than on the primary was one of several factors preventing natural circulation from cooling the reactor during the period 100 to 150 minutes from the start of the accident.

5. The low elevation of the steam generators, and the piping arrangement between the steam generators and the reactor, trapped water in the steam generator rather than permitting it to flow back to the reactor. This was another factor preventing natural circulation during the period 100 to 150 minutes from the start of the accident.

6. During the period 150 to 210 minutes from the start of the accident, a large amount of hydrogen in the reactor loop prevented natural circulation from cooling the reactor. Remotely operated vents at the tops of the candy canes would have permitted venting this gas to the containment building.

CHEMISTRY

The TMI-2 accident investigation required looking into the following chemical problems:

- The reaction of the nuclear fuel's zirconium cladding with both the cooling water of the reactor and with the fuel itself, uranium dioxide.
- An analysis of the measurements of fission products released to determine what that can reveal about the damage to the core.
- The hydrogen bubble in the reactor vessel and the likelihood that it might explode.
- The possibilities of hydrogen explosions in the containment building and the potential effect thereof.

ZIRCONIUM-WATER REACTIONS

The Zircaloy-4 cladding used in most reactors today is almost pure zirconium. Zirconium is used because of its desirable structural qualities and its particularly desirable quality of not capturing too many neutrons, thereby saving them for production of fission of the uranium. Thus zirconium is an efficient material to use for cladding. It has a melting temperature of 3,320°F which is about 525° above that of iron.

At high temperatures, zirconium reacts with water to produce zirconium dioxide, hydrogen, and heat. Oxidation of zirconium also makes the cladding brittle. This has long been recognized and the design of water cooled reactors have limited the maximum temperatures to which the cladding should be subjected.

The operating conditions of zirconium were specified to remain within the following limits even during the "design-basis" accident: (1) peak cladding temperatures not to exceed 2,200°F; (2) oxidation nowhere to exceed 17 percent of cladding thickness; and (3) hydrogen generation not to exceed 1 percent of that which would be produced if all the zirconium were to react with water. During the TMI-2 accident all of these limits were exceeded.

The study further finds:

- At high temperature, partially oxidized zirconium can be liquid at about 3,450°F.
- Uranium dioxide fuel can dissolve in the liquid partially-oxidized zirconium.

The significance of this information is that some liquid reactor fuel could result well below the 5,200°F temperature required to melt uranium dioxide alone. This would occur only where the fuel was in intimate contact with the molten, partially-oxidized zirconium. The degree to which this took place in the TMI-2 reactor could not be determined.

FISSION PRODUCTS

Measurements of the fission products released provide some information on the extent of fuel damage. These fission products are in the form of gases that escaped to the atmosphere or substances dissolved in or transported by the reactor's cooling water. The fuel damage is assessed by comparing the measured fission products with the total amount of that species produced by the reactor over its operating history.

A study was performed by Los Alamos Scientific Laboratory of the operating history of the TMI-2 reactor, and from it the quantities of the various fission products and actinides generated were computed. It also determined the amounts of those radionuclides that remain at any given time after the accident as well as the total quantity of decay heat that results from their radioactive decay.

Samples of the reactor coolant at TMI-2 were taken from the letdown line first on March 29 and later on April 10. The first was analyzed by Bettis Laboratories and the second by Savannah River, Oak Ridge National Laboratories (ORNL), Bettis, and Babcock & Wilcox.

On March 31, a gas sample was withdrawn from the air in the containment building and its radioactivity measured by Bettis Laboratories.

Using the March 29 water sample and the March 31 air sample, Bettis estimated:

- Most of the volatile fission products were released to the reactor coolant, and 2 to 12 percent of the fuel reached temperatures of 3,000° to 4,000°F. Based on this and amounts of strontium, barium, and uranium present, little, if any, fuel melted.

- About 90 percent of the 36,816 fuel rods burst their cladding, and about 30 percent of the reactor fuel exceeded 3,500°F.

Cohen, consultant to the Commission staff, concludes that some of the fuel is probably in a finely divided state from which fission products are slowly being leached.

ORNL concludes that the sizable gaseous fission product release could be produced from 40 percent of the fuel at a temperature of 4,350°F (2,400°C) and the remainder at lower temperatures.

Overall, it appears that 50 percent of the core saw temperatures of 3,500° to 4,000°F or higher and 90 percent or more of the fuel rods ruptured.

HYDROGEN BUBBLE

The hydrogen produced was inventoried. Although a set of simultaneous measurements is desirable, they do not exist.

At 9 hours and 50 minutes into the accident, a 28 psig pressure spike was recorded in the containment building. A calculation of the amount of hydrogen burned in producing such a spike showed it would take 294 pound-moles of hydrogen or 5.9 percent by volume.

On March 31, two containment gas samples measured 1.7 and 1.9 percent hydrogen and 15.7 and 16.5 percent oxygen. (Later measurements indicated both higher and lower oxygen concentrations that are unexplained but perhaps within experimental error.) Based on the measured depletion of oxygen (from a standard atmosphere) 436 + 33 pound-moles of hydrogen was consumed in the pulse. The hydrogen burned is thus taken to range from 294 to 469 pound-moles.

To this must be added the 1.8 + 0.1 percent hydrogen in the containment building's atmosphere on March 31, or 79 + 4 pound-moles. Also on March 31, the hydrogen bubble was described as containing a volume of 823 ± 200 cu. ft. at a reference pressure of 875 pounds per square inch absolute (psia). The quantity of hydrogen in the bubble is calculated as 91 ± 22 pound-moles. In addition it is calculated that 36 pound-moles were dissolved in the reactor coolant.

The sum total of all these quantities ranges from 500 + 22 to 642 + 40 or 478 to 682 pound-moles of hydrogen.

If the estimated 49,711 pounds of zirconium in the reactor all combined with water, 1,090 pound-moles of hydrogen would be produced. Thus 44 to 63 percent of the amount of hydrogen possible was produced. The portion of zirconium severely embrittled by oxidation exceeds these proportions because even 18 percent oxidation causes severe embrittlement.

Based on the gas analysis of the containment atmosphere, the hydrogen ultimately released to the containment atmosphere was 642 pound-moles, about 60 times the amount specified by the NRC as a limiting value for design-basis accidents. At its rated capacity of 0.7 pound of hydrogen per hour, the recombiner would have required 11 weeks to consume this much hydrogen, 11 weeks, that is, after it was connected up for use. The planned approach for dealing with the hydrogen was thus of no value during the critical period of the accident.

GETTING RID OF THE HYDROGEN BUBBLE

The hydrogen bubble was removed in part by taking advantage of the differential solubility of hydrogen in water, but the hydrogen disappeared from the bubble more rapidly than this mechanism alone can account for.

Cohen postulates that some hydrogen leaked past 0-ring seals between the head of the reactor vessel and the vessel. This is not a proven hypothesis.

HYDROGEN EXPLOSION IN THE REACTOR

During the period March 29 thru April 1 the NRC became concerned over the possibility of the hydrogen in the reactor vessel exploding and the damage that would result. For this to take place, oxygen would have to accumulate in sufficient quantity and then the mixture ignited. The mechanism postulated for oxygen formation was the radiolysis of water.

Radiolytic decomposition of water always occurs in water reactors, both while they are operating and after they are shut down. Knowledge of this phenomenon and how to deal with it was evolved long ago and is discussed in detail in textbooks. The usual method (as at TMI-2) is to add hydrogen gas to the coolant to react with any oxygen produced and thus prevent its accumulation. Only 0.1 cubic centimeters of hydrogen per kilogram of water will suppress the formation of oxygen; the hydrogen concentration in the reactor coolant was about 200 times this level at TMI-2. No such explosion was possible.

The Argonne National Laboratory review of the handling of the hydrogen bubble in the reactor vessel at TMI-2 reaches the following conclusion:

> It is clear that the erroneous conclusions about dangerous concentrations of 0 in the H_2 bubble originated from a number of calculations neglecting the important back reaction.... Since the radiolysis of water has been studied for decades by radiation chemists, it is hard to understand why none of this country's outstanding radiation chemists were contacted, or, as in the case of KAPL and Bettis, were asked so late in the incident... .Expertise in radiation chemistry is available at each of the National Laboratories....
>
> Certainly, there was nothing in the TMI-bubble incident for which the fundamental science was not well known .... For example, the all important H_2-0_2 back reaction, which was left out of the NRC estimates on oxygen formation, is the basis for adding H_2 to the primary cooling system under normal operating conditions. [G. Closs, S. Gordon, W. Mulac, K. Schmidt, and J. Sullivan, "Report by the Ad Hoc Committee of the Radiation Chemistry Group of the Argonne National Laboratory to the President's Commission on the Accident at Three Mile Island," undated.]

The basis for the NRC's concern for an H -0_2 explosion in the reactor vessel apparently stemmed from their gabitual assumption of worst cases rather than realistic estimates. According to NRC's chronology on the hydrogen bubble, what began as a simple check on the correctness of their presumption of no oxygen in the bubble grew into a major threat through continuing specification to supporting groups to "assume radiolysis" or to "assume stoichiometric proportions" when these were impossible. NRC staff calculations apparently had major impact on NRC's concerns through their predictions of 6 percent oxygen in the bubble on March 31 and 13 percent on April 16. Dissenting views both within and without NRC had little impact, apparently because of NRC's ingrained practice of presuming the worst. Although this approach was conservative in dealing with the physical problem within the reactor vessel, it created problems in the broader community that were apparently not adequately weighed in the balance when judgments were drawn and decisions made. If "best estimates" rather than "worst cases" had guided the judgments and decisions, the hydrogen bubble might have been handled rather differently.

On April 2, the prevailing view shifted, and the threat of an explosion within the reactor vessel disappeared.

Finding

No such explosion inside the reactor vessel was possible at any time at TMI-2. It is clear from the study that adequate information was available beforehand to set aside the fear of an explosion in the reactor vessel and that the concern generated by the public disclosure of such a possibility could have been avoided.

MAXIMUM HYDROGEN EXPLOSION IN THE CONTAINMENT BUILDING

The study assumed the extreme case that all the hydrogen that could have been generated from the water reaction with the 49,711 pounds of zirconium was released to the containment building, uniformly mixed with the atmosphere there and then ignited. The pressures were calculated for two cases as follows:

1. For thermodynamic equilibrium after a constant-volume adiabatic combustion:

   Final Pressure = 79 psig
   Final Temperature = 3,668°F

2. For a one-dimensional detonation:

   Pressure Behind Detonation Wave = 166 psig
   Peak Temperature = 4,042°F

The containment building was designed for an internal steady pressure of 60 psig and has been tested at 69 psig. Inasmuch as the design has a safety factor of 1.5, the building can actually withstand 90 psig. Thus, the 79 psig of (1) above should not be a problem.

Los Alamos Scientific Laboratory evaluated the detonation's impact on the building's structure. A key aspect of that shock loading is that it is imposed for only a brief period in comparison with the natural periods of oscillation of the building. For this reason, the detonation adds only moderately to the load imposed by the steady pressure of 79 psig. A preliminary analysis indicates that the combined loads are within but close to the building's strength. Additional study of this issue is needed.

The possibility of all of this hydrogen accumulating in the building before any ignition took place is extremely remote. As TMI-2 itself demonstrated, the building contains ignition sources, such as limit switches, position indicators, reactors, etc.

TMI-2 DECAY POWER AND FISSION PRODUCTS

This report summarizes calculations on the rate of heat generation in the TMI-2 reactor core and the amount of radioactive material within the core as a function of elapsed time following the accident. The calculations are based on the actual power history of TMI-2 prior to March 28, 1979. The calculations use well known and experimentally verified nuclear data and formula.

These calculations are important for two reasons. First, it is the decay heat, in the absence of adequate cooling, that caused the damage to the reactor core. Hence an accurate calculation of the decay heat is an essential input into a determination of the nature and extent of core damage. It is also an important input into the "what if" series of questions addressed by the Commission to assess how close, or how far away, was the TMI-2 plant and core from a more serious damage situation.

Second, the calculation of inventory of radioactive products as a function of elapsed time following the accident is an essential input into Commission estimates of the amount of radioactivity released during and after the accident. The attached Figure A shows the radioactive decay of some of the more important species, in fact, some of those found in the containment and two (Xe-133 and 1-131) that were released to the environment. The magnitudes plotted are the core inventory; the amount of xenon released to the environment was between 2 and 10 million curies out of a total of 150 million curies. The amount of iodine released to the environment was only about 15 curies out of a total of 64 million curies of 1-131. If the very short-lived species are included, the iodine inventory at the time of the accident is several hundreds of millions of curies. Many fission products decay very rapidly; in fact, the shutdown power decreased by a factor of nearly five in one hour and ten in 7 hours.

A question that has been asked relates to the hypothesis that the TMI-2 accident could have occurred at the end-of-cycle equilibrium core rather than with a relatively new core. The pertinent data are provided in the document. It does not appear that an end-of-cycle accident of the TMI-2 type would have resulted in an accident of significantly higher severity. This subject is covered in the report on "Alternative Event Sequences."

CONTAINMENT
TRANSPORT OF RADIOACTIVITY FROM THE TMI-2 CORE TO THE ENVIRONS

The major radioactive releases from the TMI-2 accident to the environment were airborne noble gas fission products, xenon and krypton, as well as a small fraction of the radioactive iodine isotopes. These isotopes, in addition to other fission products, were dissolved in the reactor primary coolant water. It is believed that the major pathway of radioactivity release from the primary system was through the reactor coolant let-down/make-up system.

Radiation products began appearing in containment at about 2 hours and 4 minutes. The containment isolation signal (3.2 psig reactor building pressure) was not realized until 3 hours and 55 minutes. The let-down system was being used periodically both before and after containment isolation to let coolant out of the system to control pressurizer coolant level. Upon containment isolation, the let-down system is isolated but the isolation was bypassed manually to permit continued removal of coolant from the system. This being the case, earlier containment isolation, i.e., upon radiation alarms at 2 hours and 4 minutes, would not have prevented the release of radioactive gases to the atmosphere.

During normal let-down operation, coolant is removed from the primary coolant system, cooled, and then piped out of containment to the auxiliary building where it goes through a pressure reducing orifice on its way to storage in the coolant make-up tank and in reactor coolant bleed hold-up tanks. Gases released from the stored coolant are compressed and stored in waste gas decay tanks.

The study shows that: (1) initial pressure transients probably caused leaks to the auxiliary building to develop in the header that normally carried gases to the waste-gas decay tanks; and (2) pressures caused by escaping gases could have lifted safety relief valves on the reactor coolant bleed tanks that discharge directly to the atmosphere of the auxiliary building.

The study observes the following:

- Earlier isolation would not have prevented release of gases to the atmosphere.
- The safety relief valves on the reactor coolant bleed tanks should be vented to the containment building rather than directly to atmosphere. (The question of putting the entire let-down/make-up system in containment should be studied.)
- Development of leakage in the auxiliary building vent header due to initial transients might have been avoided if isolated from the reactor building vent header. This would be accomplished with earlier containment signal.

RADIATION RELEASES AND VENTING OF TANKS
FRIDAY MORNING, MARCH 30, 1979

The events on the morning of Friday, March 30, 1979, may have had the greatest impact on the public of any aspect of the accident at TMI.

This study was directed at the sources of the major releases and other elements that are believed to have had a significant bearing on the recommendations to evacuate.

Major releases during the event at TMI-2 consisted primarily of gaseous radio-nuclides, xenon and krypton, and a small fraction of radioactive iodine. These major releases were caused by the continuation of let-down flow from the reactor primary coolant system after a leak developed in the vent gas header system. There had been a number of ventings of the make-up tank to the vent gas header since early on March 29.

The principal findings of this study are:

1. On Friday morning, March 30, 1979, James Floyd, Supervisor of Operations, TMI-2, had operational and technical reasons for venting the make-up tank to the vent gas header at 7:10 a.m., and at other times, based on the decisions to continue let-down which made such venting necessary.

2. Major releases occurred during the venting of the make-up tank in the let-down system to the vent gas header of the waste gas system because of a known leak, which is believed to have developed early in the event, in the vent gas header.

3. The NRC and the licensee had knowledge of this leak as early as the morning of March 29, 1979, and James Floyd admitted in his testimony before the Commission, that he was knowledgeable of this leak prior to 7:10 a.m. on March 30, 1979.

4. The venting of the make-up tank to the vent gas header, at 7:10 a.m., March 30, had been preceded by similar ventings except for one change in procedure. Previous ventings of this tank had been done in a series of short ventings and the venting in question was done in one step.

5. Following the venting in question, radiation readings above the plant, taken from the licensee's helicopter at 7:56 a.m. to 8:01 a.m., indicated a maximum radiation field of 1,200 mrem/hr (beta, gamma).

6. At about the time that NRC received information on the 1,200 mrem/hour measurement above the plant, coincidently another group at Incident Response Center (IRC) produced an estimated radiation dose rate of 1,200 mrem/hour at ground level at the north gate of the site. This estimate was based on an assumed release rate. The coincidence of the two identical numbers,

coupled with an apparent unawareness that the reported measurement was made from a helicopter in the plume above the plant contributed to an erroneous conclusion regarding the severity of the situation.

ALTERNATIVE EVENT SEQUENCES

WHAT MIGHT HAVE HAPPENDED

Very nearly all discussions of the TMI-2 accident touch upon the subject of various possible sequences of events or scenarios that might develop, starting with the actual situation and leading one way or another, from the actual situation to a variety of results -- some more, some less severe than the actual accident. These alternative scenarios can be thought of as being in one of two general classes: those that impose perturbations on the sequence of events that occurred during the development of the accident, and those that postulate somewhat different initial conditions at the time of the accident. These questions can range far and wide and can quickly lead to sequences of events that contain branches too numerous to investigate.

Recognizing both the value of examining these situations and the necessity to bound the number of cases considered, a study was made in which the actual sequence of events at TMI was followed, but at significant times in the accident one more equipment malfunction is assumed or one additional operator action or nonaction is postulated. Also, five variations in plant conditions at the time of the accident were considered. Finally, the bounding case of a fuel melting under a total absence of heat removal is presented.

Based on the approach outlined above, the development of the accident is examined to determine if it was ever close to a much more dangerous condition, and, if so, what would have been the potential consequences for the general public, the plant personnel, and the plant. In common parlance, this paper wishes to determine how close TMI was to a more severe accident and how severe would it have been. Those operator actions or equipment "nonfailures" that would have improved the situation are mentioned as appropriate.

The discussion is restricted to the design of the physical plant and environment at Three Mile Island. Generalizations to other designs and other postulated accident conditions should be made with extreme caution.

MAJOR FINDINGS

A. The temperature of the hottest region of the fuel during the accident may have been as high as the melting temperature of UO_2 (3,123°K = 5,162°F). Some small amount of fuel in the hottest zone may gave melted.

B. No <u>single additional</u> operator action or equipment failure that is tied to the actual sequence of events at TMI would have led unequivocally to large scale fuel melting throughout the core or significantly larger release of fission products to the environment.

C. If the high pressure injection system had not been turned on and if no heat sink were allowed, large scale fuel melting could occur throughout the core. This hypothetical situation was examined and bounded by postulating a fuel melting accident under a total absence of heat removal from the reactor vessel. This study found that containment would not be violated, i.e., opened to the environment by a steam explosion, over-pressure, or by penetration of the basemat (foundation) by the action of molten fuel. Because the containment integrity was not violated, the release of fission products would not be changed by a large factor over what actually occurred at TMI-2.

D. Essentially all of the radioactive iodine released from the fuel in the TMI-2 accident was retained in the water in the primary system, the containment building, and the auxiliary building. This is attributed to the chemical reducing conditions existing in the water near the fuel at the time of release of the iodine, to the high pH of the water, to the high chemical activity of iodine, and possibly to the presence of silver in the reactor vessel.

E. No radioactive cesium, strontium, barium, or lanthanum has been detected in the environment even though significant quantities of these materials were transported to the auxiliary building.

<u>FINDINGS RELATIVE TO SPECIFIC EXTENSIONS OF THE TMI-2 EVENTS</u>

1. Case 1: <u>If the auxiliary feedwater had been available as designed,</u> the accident would not have been changed except in minor detail.

2. Case 2: <u>If the PORV had closed as designed,</u> there would have been no accident. The 8-minute delay in auxiliary feedwater would have been a minor perturbation.

3. Case 3: <u>If the high pressure injection system had not been throttled,</u> a stable condition would have been achieved with no damage to the core. Ultimate recovery would require that the operators recognize the open status of the PORV.

4. Case 4: <u>If the containment had been isolated within a few minutes,</u> and if the operators bypassed isolation by opening the let-down line (as was done at about 4 hours) the accident would have been unchanged.

5. Case 5: <u>If the iodine filters had been in good</u> condition, the release of radioactive iodine to the environment would have been reduced from about 15 curies to less than one curie. Health effects of either of these amounts of radioactive iodine in the environment are insignificant.

6. Case 6: <u>If auxiliary feedwater had remained unavailable,</u> the reactor might have reached a high temperative sooner, i.e., the time scale might have been shorter with the quantity of fuel reaching melting temperatures before the HPI system was restarted being somewhat greater than may have occurred in the actual event.

7. Case 7: <u>If the PORV had remained open</u> (after 2 hours, 22 minutes), the water remaining in the core would have boiled more vigorously, giving more cooling by flow of steam. It is uncertain, however, whether the core would have contained sufficient water to continue boiling until the HPI is turned on at 3 hours, 20 minutes. If sufficient water is available to sustain boiling until HPI is turned on, some fuel could read melting temperatures.

8. Case 8: <u>If the PORV had remained closed</u> (after 3 hours, 12 minutes), the quantity of fuel reaching melting temperatures near the center of the core would have been greater than may have occurred in the actual event. Some fuel melting might have occurred.

9. Case 9: <u>If the high pressure injection system remained throttled</u> (at 3 hours, 20 minutes), the quantity of fuel reaching melting temperatures near the center of the core would have been greater than may have occurred in the actual event. Some fuel might have become molten.

10. Case 10: <u>If the containment sump pump had continued operating</u> until the time of containment isolation, the release of radioactive iodine from the environment would have increased from 15 curies to about 100 curies. The health effect of either of these amounts of radioactive iodine in the environment is insignificant.

11. Case 11: <u>If the containment had not been isolated,</u> there would have been little change in the release of xenon and iodine because the operators had bypassed isolation by opening the let-down line. This action to open the let-down line was taken to preserve a supply of pure water to provide lubrication and cooling to the primary coolant pump seals.

12. Case 12: <u>If the iodine filters had been in much poorer condition</u> (or not in place), the radio-iodine released to the environment would have increased from 15 curies to about 125 curies. The increase could have been larger, except that most of the radio-iodine was retained in water and little actually reached the filter. The health effect of either of these amounts of radio-iodine is insignificant.

13. Case 13: <u>If all the zirconium reacted with water</u> and if all the hydrogen gas generated were burned in the containment building, the building would remain intact. If all the hydrogen detonated, the loads imposed are calculated to be somewhat less than the strength of the building.

14. Case 14: <u>If an adequate hydrogen recombiner had been available, and used,</u> the pressure pulse or detonation at about 10 hours would not have occurred. Because this event apparently did not affect the subsequent sequence of events, the presence of an adequate hydrogen recombiner would not have altered the consequences of the accident.

15. Case 15: <u>If the local meteorology had been different</u> (turbulent instead of nearly stagnant), the individual and population doses would have been reduced, depending on the assumed meteorology. (The meteorology at the time of the accident was unfavorable.)

16. Case 16: <u>If control rods and burnable poisons are removed and the core geometry changed to a most reactive configuration,</u> the TMI-2 reactor is subcritical and will remain subcritical.

17. Case 17: <u>If the reactor fuel had been at end-of-cycle</u> instead of nearly new, the course of the accident would have been changed almost not at all.

<u>FURTHER FINDINGS OF MORE GENERAL APPLICABILITY</u>

18. <u>The presence of silver, probably from the control rods, has been detected</u> in the sump of the TMI-2 containment building. Vaporized silver in a more severe accident could serve as a trap for iodine released from the fuel, and would not cause any adverse conditions in the reactor vessel or containment building.

19. <u>Most of the radio-iodine released from the fuel in the TMI-2 accident was retained in the water,</u> in the primary system, the containment building, and the auxiliary building. This is attributed to the chemical reducing conditions existing in the water near the fuel at the time of release of the iodine. The radioactivity of the iodine has decayed by a factor of nearly 100 million after 7 months.

20. <u>Failure of containment would be unlikely</u> even **in** the event of a steam explosion developing out of a postulated fuel melting accident.

21. Failure of containment to the atmosphere by penetration of the concrete basemat is unlikely even in the event of a postulated large scale fuel melting accident. Significant uncertainties exist in the calculation. Bedrock underneath the TMI plant is judged to be at least equivalent to concrete insofar as penetration by molten fuel is concerned.

22. The fission product decay heat load for a high burnup core is not significantly different at early times after shutdown from that of the TMI-2 core.

SUMMARY

Seventeen variations to the actual sequence of events have been considered in this study, 12 relate to equipment or operator actions and five to matters relating to conditions not tied to the sequence of events. The cases may be classified as:

a. Resulting in no accident or no damage to the core (cases 2, 3).

b. Resulting in insignificant changes in the accident (cases 1, 4, 5, 10, 11, 12, 13, 14, 15, 16, 17).

c. Resulting in potentially more serious consequences (cases 6, 7, 8, 9).

The cases resulting in no accident or minor changes need little discussion; some of these terminate the accident, others create perturbations that damp out in time or reduce the consequences of the accident. Still others involve increased radioactive iodine release, but by amounts not significant to public health and safety.

Four possibly serious cases (6, 7, 8, and 9) require a more detailed study for definitive description than could be made in the time available. At best the accident would have been changed only in detail; at worse, fuel melting in the hottest zone could have occurred. This last possibility is sufficiently uncertain and close enough to that of gross fuel melting that the consequences of a fuel melting accident were investigated. Such an extended accident was caused and bounded by assuming an adiabatic condition (no heat sink or water injection) at 3 hours, 20 minutes. The report documents a best-estimate analysis with detailed identification of possible errors, uncertainties, and alternate paths. Where realistic or best-estimate descriptions were not possible a conservative path was chosen.

This portion of the study of an extended accident examined the physical and chemical effects associated with the melting of fuel and came to the following conclusions: subsequent steam explosions would not be expected to threaten the containment. Collapse of the molten portions of fuel into an uncoolable geometry could have led to penetration of the pressure vessel but the subsequent pressure would be less than

that provided for in the design basis accidents. However, the penetration of the containment concrete basemat by molten fuel is uncertain. If this should occur, the core material would be in a solidified form and the containment rests on solid rock thereby retarding fission product transport. It is unlikely that containment penetration to the atmosphere would have resulted, unless emergency systems designed to accommodate high temperatures and pressures in the containment were unavailable.

TMI-2 SITE MANAGEMENT

In the process of reviewing documents, interviewing operators and management personnel, taking depositions, and inspecting the facility, it became evident that a number of significant deficiencies existed in what might be called "site management." This was not evident when the accident investigation commenced but rather became apparent as the investigation progressed. Although no single discrepancy is of crucial importance, the aggregate of these errors and omission is a matter of concern and merits examination. These shortcomings suggest day-to-day management that may not have adequately supported safe reactor operation and personnel protection from radioactivity.

Most of the deficiencies discussed in this paper arose from interviews and depositions of TMI operators and managers. However, a significant number of comments, namely those concerning preservation and cleanliness, material condition and radiological controls, are based on the personal observation of one or two of the staff members most experienced with the operation of nuclear facilities.

The following summarizes their findings:

- The staff of the Reading headquarters of Metropolitan Edison did not have a significant influence over technical operations at Three Mile Island.
- The unit superintendent did not effectively carry out many of the responsibilities assigned by authoritative documents.
- Neither the station superintendent or the unit superintendent considered himself responsible for the training of operators.
- Shift foremen were tied down by administrative requirements and did not effectively supervise plant operations.
- Surveillance procedures were not adequately supervised or audited to ensure that they were done correctly.
- Procedures for operating shifts did not ensure the continued presense of a "small-break loss-of-coolant accident (LOCA) operator," as required by procedures.
- The relationship between auxiliary operators and control room operators was ill-defined, in practice.
- Shift relief procedures were significantly deficient and required upgrading.
- Valve line-up procedures and operating log maintenance procedures required upgrading.

- Many informal or unsafe work practices were observed.

- The material condition of many valves, pumps, and motors was poor due to inadequate maintenance standards.

- Radiological controls practices required upgrading. Many deficiencies were noted in the TMI-1 reactor building and and TMI-1 auxiliary building.

SELECTION, TRAINING, QUALIFICATION, AND LICENSING OF THREE MILE ISLAND REACTOR OPERATING PERSONNEL

An extensive investigation was undertaken of this area because of the failures of the operators to interpret correctly the early circumstances and to take action that could have terminated the event without severe damage. The investigation covered relevant NRC, GPU/Met Ed, and B&W documentation of training and qualification requirements, procedures, records of training, NRC operator licensing branch records, interviews and depositions of key people, and a visit to the B&W Lynchburg training simulator facility. Three general areas were examined: requirements, implementation, and evaluation of results.

FINDINGS

- There is no regulation concerning minimum eligibility requirements for reactor operators or senior reactor operators (e.g., an operator need not be a high school graduate).
- The NRC has prescribed only limited training requirements for the qualification of operators.
- The NRC does not prescribe any requirements concerning education, experience, reliability, skill, stress fitness, psychological fitness, or criminal records of managers, supervisors, operators, technicians, or repairmen.
- No management personnel other than the operations manager require licenses.
- The minimum required shift composition for TMI-2 while the reactor is at power is one senior operator, two operators, and two nonlicensed operators. Only one operator need be in the control room.
- The NRC licensing process institutionalizes a shallow level of operator knowledge.
- The NRC conducts a paper review of licensee training programs and a one-time-only review of simulator training programs when they are first set up and subsequently observes startup certification tests about every 6 months. There is no written report resulting from these observations.
- The NRC does not conduct in-depth review of licensee or simulator training programs.
- Babcock & Wilcox performs a crucial role in training operators for utilities that do not have their own simulators.
- B&W instructors are not required to qualify as operators.

- The B&W training service section has functioned almost independently of both B&W management and engineering as far as course content and conduct are concerned.

- The B&W simulator was unable to reproduce the TMI-2 accident sequence prior to March 28, 1979.

- Training at B&W did not instruct operators on how to deal with a small-break LOCA in the steam space of a pressurizer. (This was the character of the TMI-2 accident.)

- Training that operators received at Three Mile Island did not prepare them to cope with the accident.

- TMI training department is understaffed in both quality and quantity. (The supervisor of training had been unsuccessful in completing requirements for an operator's license.)

- Auxiliary operator training is sporadic, ill-defined, and does not cover material needed.

- The TMI operator requalification program is of low quality. It does not include topics required by 10 (CFR) 55 and is not related principally to ensuring safe reactor operation. Absenteeism is high.

- The TMI-2 training program did not teach operators about:

  a. Pressurizer level versus reactor coolant system pressure

  b. Recognition of saturation conditions in the reactor

  c. Recognition of the need for and the ways in which to remove decay heat

  d. Recognition of the significance of high radiation levels

  e. Recognition of a loss-of-coolant accident

There can be little doubt that inadequacies in operators and staff training and qualification contributed to the TMI-2 accident. A lack of attention to postulated accident scenarios in such studies as WASH 1400, and to prior experiences such as that of Davis-Besse, permitted training and training aids such as the B&W simulator, to be deficient in areas necessary to the understanding of the TMI-2 events.

CONTROL ROOM DESIGN AND PERFORMANCE

There is evidence the operators of TMI-2 were confused by the indications available to them on March 28, 1979. During the course of the accident that took place that day a number of malfunctions of control equipment occurred. Because of this, the control room design was reviewed to evaluate its adequacy in providing the necessary information to operators and the controls needed to shut down the plant and place it in a safe condition. Performance of the control room during the transient was assessed as was work being performed in the industry to improve control room design.

The TMI-2 control room was reviewed in visits to the plant in June 1979, in interviews of design personnel, discussions with cognizant NRC review team members, and control room design guidelines. This review determined the following key points:

- There are no definitive NRC regulatory requirements for control room design. There has not been standardization; control rooms have generally evolved as certain designs were tailored by the wishes of the utility client and influenced by precedent, designer preference, and nuclear steam supply supplier recommendations.

- The control room at TMI-2 was designed to be operated by a single person during normal operating conditions.

- Review of the March 28 accident sequence indicates that the control room did not lead directly to the onset of the transient or the follow-on events. However, operator confusion, which was evident during the accident, may have resulted in part from the control room layout and design or from equipment malfunctions which occurred.

- Emergency systems controls are not arranged in an orderly manner with all controls and process indications located in one section.

- There are more than 1,500 alarms in the plant with most of them being annunciated in the control room. Alarms are not arranged in the control room in a logical fashion.

- Indicator light colors are not such as to assure that operators are quickly alerted to out-of-position valves or breakers.

- During the accident operators were initially confused by the many alarms that were received. They were misled by incorrect pilot-operated relief valve position indication and ambiguous relief valve discharge line temperature indication. Operators did not notice shut indication for emergency feedwater block valves, perhaps because of the logic with which multi-colored lights are used on the panels. There was no emergency feed-water flow indication available to alert the operators that block valves were shut. Control of the condensate polisher bypass valve from the control rooms failed.

- Instrumentation and aids that might have helped the operators include improved computer diagnostic capability, instruments for detection of inadequate core cooling, improved data displays, a supervisor control panel, and a multi-channel recorder.

TECHNICAL ASSESSMENT OF OPERATING, ABNORMAL, AND EMERGENCY PROCEDURES

SUMMARY

As a part of the effort to identify and evaluate the possible causes for the Three Mile Island accident an analysis of operating, abnormal, and emergency procedures was conducted by the staff. Those significant procedures which were in use at the onset of the accident and the procedures which became applicable as the accident progressed were evaluated for technical accuracy and adequacy with respect to the transient of March 28 and its aftermath.

Summarizing the more significant findings:

a. Seven of the 15 procedures reviewed were adequate for their intended purpose and were not causative factors either in the onset or the severity of the accident as far as their technical accuracy and adequacy are concerned.

b. Four procedures were judged to contain significant deficiencies that could cause confusion or lack of action but which would not preclude their use by competent operators.

c. Four procedures, pressurizer operation, loss of reactor coolant/reactor coolant system pressure, pressurizer system failures, and post-accident hydrogen control -- were assessed to be so deficient as to be inadequate.

d. The provisions of some procedures may have influenced events on March 28. For instance, some procedures emphasized avoiding equipment or component damage over keeping the core covered and cooled. Operators were required by technical specifications not to permit the pressurizer to go solid. Procedures seem to be written to minimize "outage" and maximize "plant availability."

FINDINGS

Analysis of the technical aspects of the operating and emergency procedures that were used or which were applicable on March 28 at TMI-2 suggests the following findings:

1. The following procedures, although they may be deficient in minor respects, are adequate for intended purposes:

   - Operating Procedure 2102-2.1, Power Operations
   - Operating Procedure 2104-1.3, Decay Heat Removal System
   - Operating Procedure 2102-3.3, Decay Heat Removal via OTSG

- Operating Procedure 2104-1.1, Core Flooding System
- Operating Procedure 2104-1.4, Reactor Building Spray
- Operating Procedures 2104-6.3, Emergency Feedwater
- Operating Procedure 2105-1.3, Safety Features Actuation System

2. The following procedures contain significant deficiencies that could cause confusion or lack of action but would not preclude their use by competent operators:

- Operating Procedure 21031.4, Reactor Coolant Pump Operation

  -- Precludes pump operation with excessive vibration.

  -- Whether pump should be tripped under low pressure, LOCA conditions was not clear.

- Abnormal Procedure 2203-2.2, Turbine Trip

  -- Does not require operator to verify that the PORV is shut although it is expected to open.

  -- The operator is directed to use let-down, as necessary, to preclude pressurizer level from exceeding 240 inches following a turbine trip.

- Emergency Procedure 2202-2.2, Loss of Steam Generator Feed.

  -- Requires immediate manual reactor trip on loss of both feedwater pumps.

  -- Does not require verification of proper PORV operation.

- Emergency Procedure 2202-1.1, Reactor Trip

  -- The procedure makes no provision for determining the cause of the reactor trip and correcting it.

3. The following procedures were so deficient as to be inadequate:

- Operating Procedure 2103-1.3, Pressurizer Operation

  States the pressurizer may not be taken solid for any reason except hydrostatic tests.

- Abnormal Procedure 2203-2.6, Post-Accident Hydrogen Control

  The procedure does not recognize rapid generation of hydrogen as occurred at TMI.

-- The procedure does not recognize any difficulties that might be encountered in placing the hydrogen recombiner in operation.

o Emergency Procedure 2202-1.5, Pressurizer System Failures

-- The procedure's basic structure is very confusing, some sections should be in the loss of coolant procedure; symptoms are significantly incomplete, misleading, or erroneous.

-- No guidance is given for actual pressurizer level control problems.

-- Terminology is sloppy.

o Emergency Procedure 2202-1.3, Loss of Reactor Coolant/Reactor Coolant System Pressure

-- Procedure lacks objectives.

-- Symptoms are incomplete, misleading, or erroneous.

-- The procedure is difficult to use. Cases are not defined.

-- The operator is required to throttle HPI to prevent pump run-out regardless of the severity of the accident.

-- The procedure does not promptly ensure that containment is isolated.

-- A section on small-break LOCA response is illogical and cannot be followed.

-- No cautionary guidance is included regarding core covering and cooling.

4. Operators were prohibited by technical specifications from permitting the pressurizer to go solid.

5. Some procedures emphasize avoiding equipment damage over keeping the core covered with water or maintaining core cooling.

6. The procedure for decay heat removal via the once-through steam generators (OTSG) is simple, straight-forward, and if followed can be used to cool the core either with or without running reactor coolant pumps.

7. Procedures recognize that the PORV will open following a turbine trip.

SIMULATORS --
TRAINING & ENGINEERING DESIGN

A light-water nuclear power plant with its many components, subsystems, and systems requires the use of a very large number of analytical models, computer programs, and analytical tools for design, licensing, and training. The models available to describe conditions in the reactor can be broken down into several categories. They are:

1. Steady-state analyses. Such models deal with reactivity, fuel enrichment, heat transfer, power, and flow distribution in the reactor on a steady-state basis. They also provide many input parameters to transient computations. Sometimes, they are employed to describe very slow transients that can be evaluated on a quasi-steady-state basis.

2. Transient analysis. These models deal with most normal and abnormal plant disturbances. They employ a relatively simple representation of the reactor primary system, but include accurate control and safety functions in their modeling. They tend to deal with small departures from normal conditions but not accidents.

3. Accident analyses. These analyses deal with unexpected events such as a leak or break in the primary system, etc. They are transient calculations but they analyze conditions more degraded than those in the transient analyses described above.

4. Damage analyses. Several of the accidents may lead to damage to the reactor core and the calculation of such damage often requires a separate analysis. The accident may alter the reactor configuration and conditions may be quite different from those under normal transient conditions, or the initial stages of the accident.

5. Training simulator models. Such simulators often employ different and more simplified models than those in design or safety analyses, and they are best dealt with as a separate group.

SEGMENTATION OF ANALYSES

The kind of information required and accuracy and details of the calculations can be expected to vary with each kind of analysis. For example, considerable accuracy and details in the reactor core are utilized in steady-state calculations while any accident analyses employ a much more lumped representation of the core. This has led to the development of computerized models (or codes) that are applicable only to certain types of events and often to rather limited scenarios. In some cases the results from one code are required as input to another code. Such segmentation is a serious drawback to being able to calculate the entire course of TMI type accidents. No single code exists that combines a good control system and a good small-break model. While such calculations can be performed by combining several available codes, the analyses are not flexible enough to evaluate readily changes in the possible branches of the fault trees. This is all the more true when operator actions are included.

Superimposed upon this segmentation of analyses for different transient and accident types is the fact that many calculations are performed for licensing purposes rather than on a best-estimate basis. In other words, descriptive sequences of events generated by codes may not be indicative of what the operators will see.

CAPABILITY FOR ANALYSES

The capability for analyses varies from one organization to another. At present, the best capability resides with the reactor suppliers who can perform the entire range of calculations. Next, in terms of capability comes the NRC. While the NRC could call upon national laboratories to attain the same level of proficiency as the reactor suppliers, they have chosen often to assess and audit the results from manufacturers' analyses rather than reproduce them. The widest spread in range of analytical capability exists among the plant owners or operators. Some utilities such as Tennessee Valley Authority (TVA), Duke Power Company, and others have developed good analytical capability while other utilities have almost none. Electric Power Research Institute (EPRI), through its computer codes, is trying to make it possible for all plant owners to have adequate independent analytical tools. However, analytical independence by all utilities is not true today, and several plant operators have to rely very heavily, if not exclusively, upon manufacturers for most of their analytical evaluations. Under such circumstances, the plant-operating engineering support group cannot help but be less responsive and lacking in complete understanding, especially for unexpected type events.

GENERAL FINDINGS

1. There is a strong need for analytical simulation of fault-tree events that involve control systems, operator actions, and equipment failure such as occurred at TMI. Such calculations need to incorporate man-machine interactions and need to be performed on at least a real-time and on a best-estimate basis.

2. The NRC has an inadequate capability to analyze independently transients and accidents.

3. Utilities have an inadequate capability to perform transient and accident analyses.

STEADY-STATE ANALYSES

The steady-state reactor analyses are concerned with calculating the three-dimensional power distribution, reactivity, exposure, and thermal hydraulic characteristics in the core at startup and as fuel burn up progresses. The reactivity computations involve several nuclear group cross-sections and many parallel flow paths. They are multinode calculations and often take several hours on the fastest digital computers available.

The steady-state calculations are of utmost importance to the performance and economics of power plants. They yield the fuel enrichment and operating reactivity strategy, both of which control fuel-cycle costs. They also determine the allowable operating power level.

Findings

1. Commercial incentives motivate the continued development and verification of steady-state models by industry.

2. Several of the outputs from steady-state codes are employed in other performance and safety evaluations. Often, such parameters are taken at their bounding values which make ensuing calculations not representative of what the operators might see. It would be desirable to identify all such outputs, their best estimated values and their range of uncertainty.

TRANSIENT ANALYSES

These analyses are used in the following ways:

1. Investigate total plant dynamics and, in particular, optimize control systems for normal and off-normal operations.

2. Investigate anticipated plant transients and ensure that appropriate safety margins are satisfied.

Findings

1. There are many limitations to the existing transient models. For example, some codes used apply to power levels between 15 and 100 percent and are not suitable for decay power level or low-power natural-circulation studies. No two-phase condition is allowed in the primary system, i.e., it cannot simulate a system piping break or two-phase natural circulation without a break. The pressurizer cannot go solid or entirely empty and the modeling of the emergency core cooling systems (ECCS) is not included. In other words, it is limited to those transients where the primary system remains relatively close to normal.

2. Because of the lack of emphasis on power plant operation under diverse operating conditions, existing transient analyses have not been able to cover the operating ranges encountered at TMI-2. Experimental data on component performance for use in such analyses have not been obtained for sufficiently broad ranges of operating conditions.

3. Several comparisons of the models have been made to startup test data and reactor transients. While the results correspond generally with the events, some discrepancies exist and deserve further investigation.

4. At the time of the TMI-2 accident, elaborate computer codes (TRAC and RELAP) were available for analysis of the severe reactor accidents being investigated by the NRC under their LOFT program. These

computer analyses were tailored to the brief durations of the LOFT tests (just minutes long) and had not been used to explore the longer transients represented by TMI-2 (15 hours long). In the 6 months since the accident, both RELAP and TRAC have been each employed to analyze just a portion of the accident but with a considerable expenditure of computer time; for example, about 30 hours on a CDC 7600 were consumed during TRAC's analysis of the first 3 hours of the accident. Because so much computer time is required by these computer codes, the analyses have been focused on just the TMI-2 accident, and no exploratory analyses have been made of the general class of multi-fault accidents of which TMI-2 is but one example. New computer codes, each specific to the transient conditions being explored, could speed up the process and thereby permit this broader investigation of a range of accidents.

5. In addition to the failure to recognize the need for investigating such multi-fault accidents, a key reason that the capability to analyze such accidents has not evolved is the lack of recorded data from the power plants themselves on which to base the analyses. Because nuclear power plants are so costly (about $1 billion each), they have not been subjected to deliberately imposed transients. Although accidental transients can provide some of the needed information, the power plants are generally not equipped with the data-recording system that would make this possible.

6. Fortuitously, such a data-recording system (B&W's Reactimeter) was installed at TMI-2 at the time of the accident. The data it recorded has been of great value in the accident's postmortem. These data are also potentially useful as input to training simulators. Although use of such recorded data would extend the range of training programs and give an air of realism to an otherwise synthetic process, use of such recorded data in training is not an industry practice. Had, for example, data recorded at Davis-Besse been used in a simulator for training operators at TMI-2, those operators might have learned how to cope properly with the similar accident that did occur at TMI-2.

7. Many of the transient studies are terminated early, and in so doing, do not examine other abnormal conditions that might develop in the course of bringing the plant to cold shutdown.

ACCIDENT ANALYSES

There are many accident analyses performed in the course of safety evaluations and preparations of safety analysis reports for submittal to the NRC. The number of models is also rather large.

The objectives of the extensive safety studies performed in the licensing process are to define the worst cases and to show that they satisfy the requirements of Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix K.

The Appendix K, 10 CFR 50, specifies many of the details of the required analyses. As an example, for a LOCA, it not only specifies initial conditions, rates of power generation, and certain model features, but it also identifies the peak fuel clad temperature not to be exceeded and the malfunction characteristics to be employed. Generally, the LOCA analyses are performed for a specific set of break sizes with the plant at 102 percent of power and with the assumptions of reactor trip, no off-site power, and one single failure such as one complete train of the emergency water cooling system not being available.

Findings

1. A large number of worst-case accidents are examined with the assumption that lesser accidents are covered if the large ones can be handled.

2. The available models take several hours on the fastest computers to carry out simulations of accidents.

3. Event tree/fault tree analyses of accidents have not been employed fully. They have not adequately considered operator information and operator actions. Such analyses can surface sequences of events that are more probable or more severe than those presently prescribed in the licensing process.

4. Modeling of accidents for licensing purposes on a "best estimate basis" and adding a safety margin may be superior to the present mode of adding conversatism to the scenario, i.e., taking a worst case for a "design basis accident". This approach can lead to a better understanding of phenomena and make more information available to operators.

DAMAGE ANALYSIS

Following the accident at TMI-2, there was a need to estimate the degree of core damage and, in particular, the reactor core configuration. This knowledge was necessary to evaluate alternate modes of transition to cold shutdown at TMI-2. Models have been developed to deal with such post-accident damage. These models vary with the type and degree of resultant damage. For example, during a LOCA, the fuel cladding will balloon and fail and lead to flow blockage in the fuel assembly. If the fuel clad temperature continues to rise, metal (zirconium)-water reaction takes place and brittle clad failure occurs. In the case of a very strong reactivity accident, the fuel clad will rupture and some fuel fragments might be dispersed in the coolant. Many out-of-reactor and in-reactor experiments have been performed to help predict the resulting damage and to verify the many available models.

In performing such predictions, one of the key results is to define the prevailing geometry because it will determine the flow at each location and the fuel capability to transfer heat to the coolant. As expected, uncertainty in geometry increased rapidly with degree of core damage.

Findings

1. Damage models have been developed and they are validated against experimental data. These models tend to deal with the early stages of damage and to overestimate the consequences in order to satisfy licensing requirements.

2. There are uncertainties in the models and continued experimentation and modeling efforts need to be carried out. Several in-reactor experiments have been performed to simulate LOCA accidents. These experiments are being sponsored by NRC. In the past, overemphasis may have been placed upon modeling and testing the rapid-damage scenarios rather than slowly developing damage as occurred at TMI-2.

TRAINING SIMULATOR MODELS

The use of nuclear reactor training simulators began in 1968. The purpose of simulators is to provide a realistic facility for training reactor operators. The major advantage of a simulator over a real control room is that it can provide the operator with exposure to unusual events which might otherwise take an entire career to experience or endanger a facility.

The models that are used in these simulators to represent the water flow, steam flow, core power, valve position, control rod position, etc., are much simpler than the models described earlier in this report. There are two reasons for this: first, there is less need for detailed information in a training simulator, and second, it must be simple in order to perform the calculations in real time.

A set of prescribed transients is run on the simulator model once it is assembled, and adjustments are made to make the control room indications to be the same as that expected on the real reactor, within the tolerance limits allowed.

Findings

1. The current-generation training simulator models are very capable of modeling operational manuevers such as startup, shutdown, turbine trip, and load demand changes. To varying degress they are also capable of simulating multiple component failures and instrumentation and control malfunctions.

2. The March 28, 1979, capability of simulation of the TMI-2 event with the training simulator was virtually non-existent. For example, the Lynchburg training simulator could not take into account steam void formation or simulate the breakdown of natural circulation when the plant is employing that mode of cooling. Even the most recent generation of simulators, which do a much better job of simulating the TMI accident, have a problem of coarse noding in the primary loop, which makes the natural circulation calculations marginal.

3. The computer/simulator industry appears to have the capability of designing simulators that are much more sophisticated through the use of faster, larger computers and improved programming techniques.

4. In aerospace applications, the models for aircraft simulation are relatively simple, with the flight motion and visual simulator requiring large computing capacity. The nuclear simulation models are generally more complex than aerospace simulation models, but have the advantage of not needing complex visual and cockpit motion simulation. Aerospace simulation appears to be more advanced than nuclear simulation because of the need for speed and capacity for cockpit simulation. However, the overall level of technology appears to be equivalent between the aerospace and nuclear industries.

5. Simulators are often calibrated against analytical results that are presented in licensing documents. These licensing calculations are usually very conservative, rather than being "best estimate," and therefore often introduce a bias into what is presented to the trainee as a normal event.

6. NRC reviews the degree to which a simulator duplicates the type of plant only once. There is no periodic reassessment of the ability to cope with new experiences and accidents.

7. Flow of information between designers, operators, and simulator designers could be increased to the benefit of all.

<u>SIMULATOR LOCATION</u>

Although some utilities own a training simulator, Metropolitan Edison relied on Babcock & Wilcox's simulator in Lynchburg, Va., a common practice in the nuclear-power field. The result was that the operators at TMI-2 had fewer training opportunities than if such a training simulator had been located on site at TMI.

Considering the aid and counsel that the current state of the art in computer simulation could have provided the operators, they were also handicapped in their ability to judge just what actions they should have taken. An appropriate computer (or simulator) on-site could have (1) displayed to the operators the severity of imposed reactor operating conditions and recommended corrective actions, and (2) could have traced at high speed the future course resulting from contemplated actions. Two examples illustrate the potential impact on TMI-2. When boiling took place in the reactor and the pressurizer went off-scale high, a computer could have displayed this condition and cautioned the operators not to turn off high-pressure injection from ECCS, as they did. When at 7 hours and 30 minutes the operators opened the PORV block valve in an attempt to lower reactor-loop pressure to 400 psig and then to shift the decay heat removal, a computer at the site could have raced ahead to trace the future course of this plan and predicted for the operators that their attempt would fail.

Findings

1. In contrast with the practice of some utilities, the simulator for training operators for TMI-2 is in Lynchburg, Va. This remote location diminished the opportunities for operator practice and training compared with what could have been achieved if the training simulator had been at the TMI site.

2. In forming judgments on how to handle the accident, the operators were handicapped by a lack of display of relevant data, in comparison with the information that an on-site computer could have provided.

In general, all simulation could benefit from the following:

1. Greater use of fault-tree/event-tree analyses to point up most probable events and sequences.

2. A rigorous requirement to ensure that experienced events, transients, and accidents can be duplicated by all of the relevant models and for new sequences of events to be exercised on the simulators to test adequacy of understanding of designers and operators as well as the adequacy of the relevant procedures used by operators.

3. To provide for 2 above, an adequate engineering record of transients experienced by all plants should be assured.

4. Continuous updating of simulators based on new experience involving the utility's own engineers has been found to assure the highest level of understanding of the plant and its responses to accidents at the plant when it is needed.

EQUIPMENT CONSERVATISM

The objective of this study was to evaluate equipment conservatism or margins built into the equipment utilized in TMI-2. The study looked into design considerations and evaluated the controls exercised to ensure that margins were preserved.

This study was performed by a contractor with considerable experience in design of nuclear systems as well as other systems requiring high reliability. Three components that had experienced some problems during the TMI-2 accident were selected for this study: the high pressure injection pumps (HPIP), the pressurizer heaters (PH), and the condensate polisher (CP). The first two are designated "safety related" and the third is not.

SIGNIFICANT FINDINGS

- Statements of design conservation (margins) are not stated in quantitative figures for functional equipment or system performance. This statement applies to the various regulatory guides, codes, specifications, etc. utilized by the Nuclear Regulatory Commission (NRC). The NRC does require compliance with the ASME Boiler and Pressure Vessel Code for pressure vessels, pumps, piping, and containment structures.

- Design Conservatism Margin* - Components evaluated.

  For the components/subsystems studied, the design margin in terms of performance and structural adequacy is summarized below.

Pressurizer Heater System

Based on the design criteria specified, the heater system capacity provided represents a design margin for the worst demand condition of 0.5 or 50 percent greater than what is required of the system.

The pressurizer heater system has considerable redundancy and the heater elements themselves are of a rugged construction. The system has sufficient over-design in terms of heat capacity and redundancy such that a number of elements could fail without adversely affecting system performance. In summary, it appears that the system is over designed, and includes good redundancy provisions.

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* Note: Design conservatism for purpose of this report is defined as the amount of margin or excess capacity over demand(s) that a system or component is capable of and has included within the design.

Make-up Pump (MUP)/HPIP Margins

In the normal make-up mode, the MUP/HPIP units can each provide a factor of 1.62 times the flow required to maintain normal system make-up requirements. The data show that the two pumps' combined capacity exceeds the system demand by a factor varying from 1.57 to 2.57 depending on system pressure conditions.

Condensate Polishing System (CPS)

The hydraulic performance of the CPS to handle the maximum required condensate flow is assessed to be satisfactory. The polishing system is sized in accordance with an empirical industry standard of 50 CFM of flow per square foot of polisher bed area (in the direction of flow). The design flow capacity of the CPS of TMI-2 per the above standard is 17,572 GPM.

The hydraulic performance of the CPS, to assure at all times and for all operational flow conditions that enough parallel polisher legs will be available to adequately feed the condensate booster pumps, is assessed to be inadequate. This assessment is based on the fact that the CPS is a full flow system, but does not have an active (automatic) by-pass system. Secondly, the availability of sufficient numbers of parallel polisher legs cannot be ensured because it is subject to manual operator procedure and adjustment.

FURTHER SIGNIFICANT FINDINGS

- The condensate polisher instrumentation and control system does not have adequate fail-safe provisions incorporated into the design. No analysis was performed or requirements formulated to assure that an adequate number of polisher legs would be in operation to feed adequately the condensate booster pumps under all operations flow conditions.

- End-to-end functional schematics or flow diagrams do not exist that cover all aspects of a given system.

- Independent design reviews were conducted by the A&E contractors and outside consultants. Comments were constructive and indicative of a review of fair depth.

- Design practices/controls imposed on the TMI-2 design by either the NRC or GPUSC were less rigorous than those imposed by other high reliability and safety-oriented programs.

In summary, the two safety-related systems, the high pressure injection pump and the pressurizer heaters are over designed. Their design is governed by codes and standards rather than by a quantitative statement of margin requirements. The condensate polisher meets the flow requirements, but has a control system that permits total flow disruption upon failure. This may be an oversight or symptomatic of a lower degree of attention period to non-safety related programs. A sample of three is hardly enough from which to draw conclusions of this type.

SAFETY DESIGN MARGINS

The Levy Report (Chapter 4) was prepared in response to a request by the staff to "assess design margins in the TMI nuclear plant".

The Levy Report addresses three aspects of safety margins. First, it addresses the approach taken during the present nuclear plant licensing process towards defining safety margins aimed at improving this approach. Second, it addresses the actual design margins present in existing nuclear power plants in these specific areas: peak fuel duty (kilowatts/foot), critical heat flux margins, and peak clad temperature (during LOCA) margins. The report concludes that "the course of events at TMI-2 would not have been changed considerably or the consequences seriously reduced if the design margins at TMI-2 had been greater." Third, the report compares the equipment margins among modern nuclear plants (PWR's) produced by B&W, Westinghouse, and Combustion Engineering in several important areas, such as secondary side boil-off time at full power, high pressure injection capability, etc., and makes some general observations about the relative safety margin in these three plants.

Some of the more signficiant conclusions, and staff comments, in each of these three areas, are as follows:

1. <u>Approach Taken During Licensing In Determining Adequacy of Safety Margin.</u> The report criticizes the present approach used by NRC and industry to estimate safety margins as being too narrow, and suggests that a broader and more systematic approach would be beneficial. For example, the report suggests that there may be accident scenarios other than those presently focused on the licensing process, and other than the TMI scenario, that are of higher probability and may produce extensive damage. The report suggests a systematic approach towards evaluating an expanded set of accident scenarios and that careful consideration of operator errors be included in this systematic approach. In the past, several organizations have recommended that quantitative safety goals be defined in the licensing process so that rigorous safety evaluation methods would have some meaning (e.g., Atomic Industrial Forum Report on Reactor Licensing, 1978, and letter from the ACRS Chairman to Joseph Hendrie dated April 1979) but this has still not been implemented by the NRC.

2 <u>Design Margins at TMI.</u> The Levy Report states that even if design margins at TMI-2 had been greater, the course of events would not have been much different. While this may be true for the particular parameters selected by Levy (peak fuel duty, critical heat lux, and peak clad temperature), it is not true for the other types of design margins that should normally be considered in designing a power plant. For example, this staff assessment points out elsewhere that the condensate system at TMI-2 has inadequate design margin, and this contributed to the initiating events on March 28. If this is true for the condensate system at TMI-2, then it may well be true for many other auxiliary systems at TMI and elsewhere, systems that are not generally considered to be safety-related, and hence do not receive the same focus of attention during design and licensing as the primary heat transport system.

3. Equipment Margins. The Levy Report notes substantial differences in equipment margins between B&W, Westinghouse, and Combustion Engineering designs. In some cases, such as thermal inertia in the steam generator, the B&W units have less margin. In other cases, such as high pressure injection system capability, or natural circulation thermal driving head, the B&W units have more margin. The integrated control system of the B&W plants not only makes the plants easier to operate but also by making the plants more responsive to load change, diminishes the frequency of reactor trips.

Levy further notes that raising the steam generators, as in the Davis-Besse power plant, increases the operating margins during abnormal or accident conditions. The capability for removing heat from the reactor by means of natural circulation is substantially augmented, thereby improving on an already existing advantage of once-through steam generators over the U-tube type. In addition, during a severe accident, as at TMI-2, nearly the entire inventory of reactor coolant in the steam generators could drain into the reactor rather than being trapped there as at TMI-2. Overall, the report concludes that margins appear to be about equivalent.

The staff has no basis for disagreeing with this conclusion. It does not consider it to be possible, nor would it be useful, to attempt a critical evaluation of basic plant design features of the different types of plants now in use. The general view is that the safety of these plants is more dependent on how the design features are implemented in actual practice (i.e., the details of control systems, quality assurance, operating procedures, etc.) than on the basic features themselves. The staff did, however, find some system component inadequacies and questionable designs. For example, the staff reports on the polisher and the PORV describe some of these design inadequacies.

GENERAL COMMENT

The staff found the Levy Report to be quite useful in providing insights and innovative viewpoints on the subject of reactor systems, regulation, and reactor safety. Many individual observations, conclusions and findings are included that are not mentioned in this summary, and which the staff believes may be useful for the Commission and others considering the matter of nuclear power plant safety.

ADDITIONAL COMMENTS

The report also contains two papers on NRC's approach to systems safety consideration and their approach to changes in requirements deemed necessary in the interests of safety. These relate to actions taken by NRC to improve safety margins as a result of TMI-2 as follows:

1. PORV Margins. For operational convenience, B&W reactors use PORV and ECCS to avoid using scram in normal transients. Other PWR designs do not, because their PORV pressure settings are above the reactor scram setting and reactor scram is tied to loss of secondary cooling. Essentially, this exposes the B&W reactors to more demands on PORV, hence more chance of small LOCA, hence more demand on HPSI, hence

more risk of fuel melting (reference event tree report). There is more chance of a small LOCA in the B&W design without anticipatory scram. Since TMI-2, NRC has required B&W reactors to include anticipatory scram.

2. <u>ECCS Margins.</u> High-pressure ECCS coupled with proper PORV operation is sufficient to cool decay heat from B&W reactors during transient loss of secondary-side cooling. In Westinghouse reactors, secondary-side cooling must be restored within about an hour in order to prevent fuel melting (reference WASH 1400, TML sequences). There is a better capability in B&W designs than in Westinghouse to handle transient loss of secondary-side cooling. This capability is decreased since NRC required the PORV setpoint to be raised.

3. <u>General Observation.</u> Design changes to improve the safety margin in a reactor have been made without detailed analysis of the effect of the changes on other parts of the system, and can actually lead to degraded safety. To lessen the demand on PORV and, hence, the likelihood of PORV failure in B&W reactors, NRC currently requires that the pressure setpoint for opening PORV be set above the reactor scrar-pressure setpoint. However, this raises the pressure against which ECCS must work in feedwater transients, and reduces the capability of ECCS coupled with the PORV to cool the reactor for an extended period of time.

Changes such as these can have other effects on the whole plant, making desirable the use of WASH 1400 risk assessment methodologies for evaluating safety margins between existing and alternate systems.

PILOT-OPERATED RELIEF VALVE DESIGN AND PERFORMANCE

The failure of the pilot-operated relief valve (PORV) to close when the pressurizer pressure returned to safe operating levels was a major contributor to the TMI-2 accident. The failure of the operators to recognize this fact for 2 hours and 20 minutes is discussed in the sections of the report dealing with operator performance, training, and procedures. Because of the criticality of the PORV to this event, a detailed study was made of its purpose, history of occurrences, reliability, performance of other available values, and its recognition as a safety-related item.

PURPOSE

The purpose of the PORV is to relieve reactor coolant system (RCS) pressure increases due to transients without operating the code safety valves. When the code valves are operated they often leak upon reseating and the plant must be returned to a cold shutdown for refurbishment of replacement of these valves. Safety precludes the use of block valves in series with code valves. Since at TMI-2 the reactor is not automatically tripped on the onset of transients that induce RCS pressure excursions, the PORV is used operationally to avoid a reactor shutdown and the resultant time that it would take to shutdown and restart (approximately 8 hours). All PORVs leak after a few operations. Small leaks of the PORV (up to 1 gpm) are allowed in the technical specifications. A procedure exists calling for the closing of PORV block valves if the discharge line temperature exceeds 130°F. Practice, however, tolerated higher discharge line temperatures.

HISTORY

PORVs have been operated hundreds of times in operating plants. Although there has been some improvement in reliability over the last few years, failures still occur. Nine failures have been identified in B&W plants, one in a Combustion Engineering Plant and one in a Westinghouse plant.* Boiling water reactors (BWRs) are much more responsive to transients than pressurized water reactors (PWRs) because of the absence of stream generators so they make much more use of PORVs (they call them pilot-actuated relief valves (PARVs) and 21 failures of these between 1970 and 1978 are identified.

Since these valves, when operated, are breaching the primary RCS boundary, a clear line of defense or safety, the acceptance of the safety-of-the-art in PORV reliability and its use for operational convenience needs to be carefully reexamined. The NRC has already moved in this direction by ordering an increase of the pressure level at which the PORV is opened and by ordering that the reactor be tripped on loss of feedwater. These actions both minimize the number of PORV operations called for and the requirement for it to be used at all.

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* On August 24, 1974, the PORV failed to close at the Beznau plant in Switzerland. This was not reported to NRC until after TMI-2.

OBSERVATIONS

- PORVs are subject to leakage and failures.
- Existing procedures did not realistically take into account operations under leaking conditions.
- Training and simulation did not reflect probability of PORV failure.
- Acceptance of operational convenience use of PORV should have been critically weighed in light of its breaching of a safety boundary and the possiblity of operator error.
- The PORVs may not be required when taking into account the new requirement for anticipatory trip of the reactor.
- The PORV at TMI-2 was not recognized as a "safety-related" component.

CONDENSATE POLISHING SYSTEM

The function of the condensate polishing system is to maintain water quality by removing impurities from the condensate; the objective being the prevention of problems in the power conversion system caused by scale formation, corrosion carryover, and caustic embrittlement. In addition, the system design provides for removing impurities in the condensate caused by in-leakage in the steam generator of reactor coolant liquid, and intermittent in-leakage in the condenser of cooling water from the circulating water system.

The condensate polishing system is composed of eight parallel condensate polisher units. The design of the polisher units and regeneration equipment is based on a 28-day resin in-service life, this is, each of the eight polisher units, or tanks, nominally needs to have resin removed and replaced once each 28-day period. The system is so designed that seven of the polisher units can handle the full condensate flow, while the remaining one is being replenished.

The equipment in the condensate polishing system is in accordance with ASME (American Society of Mechanical Engineers) boiler and pressure vessel code, Sections VIII and IX. However it is not classified as "safety-related" and thus does not receive the same care and attention, as from quality control during its operational life, nor did it receive such when it was designed, fabricated, transported, stored, installed, and checked out.

As reported by several sources, and as noted in various interviews, depositions, and hearings, the TMI-2 plant operating staff had been working for some time when the accident was initiated to clear resin from polisher tank Number 7. This work was reportedly being accomplished in accordance with operating procedures and had been in progress for about 11 hours prior to the accident. The work involved the use of compressed air and water, as per the procedure to force the spent resin from the tank.

At the time the turbine trip was announced, an operator reported that the condensate polisher panel indicators showed condensate polisher isolation, which indicates no flow through the polisher. This condition could be caused by closed polisher effluent valves. This state of no flow at this time was confirmed through a review of records. This no flow condition would then result in the condensate pump trips that did occur initiating the loss of feedwater.

THE INVESTIGATION

Investigation into the causes for polisher failure included examination of design, history, procedures, and post-accident tests resulting in the following findings.

DESIGN

Without using the manual bypass valve provided in the TMI-2 condensate polisher, the system had essentially no capacity beyond that required of normal 100 percent operations. In addition, the polisher system did not have the automatic fast acting bypass of TMI-1.

HISTORY

On at least two occasions (June 15, 1978, and Nov. 4, 1977) prior to the accident, operators documented serious concern over loss of flow through the polisher system due to sudden closure of polisher effluent valves resulting in loss of condensate flow as it did on March 28, 1979. Design deficiencies were not effectively corrected.

PROCEDURES

A recently revised procedure was being used for work on the polisher. No quality assurance audit was being performed since this system is not classified as "safety-related" equipment.

POST-ACCIDENT TESTS AND INSPECTIONS

Water was indeed found in the service air system although attempts to repeat the failure with water in the system have not yet reproduced the closure of the effluent valves. Other differences were found to exist between equipment and drawings.

The effluent valve solenoids were not wired per the drawings. The effect of this is not yet known.

SUMMARY

It is still not proven at this time that the work on the polisher caused the initiating event of the accident although it is most probable. The study of component conservatism contains an analysis of a possible mode for failure of the polisher in the manner experienced at TMI-2.

The reliability of this system is questionable. There was a lack of management attention to recurring problems with the system prior to TMI-2 and a corresponding lack of attention given to operator's expressed concerns over the consequences of a malfunction of the system. There is an absence of quality assurance overview of the equipment or procedures involving it because of its nondesignation as "safety related".

Perhaps the most questionable design decision was that of not providing for a routine and automatic bypass capability on the TMI-2 polisher system. Various arrangements of this bypass capability that provide operational margin are in use in other nuclear plants.

QUALITY ASSURANCE

A review of the independent assessment program at TMI-2 and the requirements thereof as defined by NRC regulations was conducted. A team of the Commission staff and consultants studied the regulations, organizations, procedures, and practices involved in both the NRC and the utility's activities which are intended to assure the safe operation of nuclear plants.

Requirements for quality assurance and reliability activities are contained in: 10 CFR 50, primarily Appendix B, Quality Assurance, for the utility; in the standard review plan for the design and review process conducted by the Office of Nuclear Reactor Regulations; and in the Inspection and Enforcement Manual for the audit program conducted by the NRC Office of Inspection and Enforcement.

The review team found that the regulations and overall review process apply only to those portions of the plant defined as "safety-related"* and that it does not call for the rigorous safety analysis and reliability engineering techniques currently being applied in other safety critical programs and industries. The review shows that management structures that have evolved as a result of the narrow definition of NRC responsibilities do not provide for an independent assessment and check of many critical systems, functions, and operations.

The narrow approach by NRC is reflected in the response of the utility in the scope of responsibility, staffing, and management attention to this very important area. It is believed that this situation made possible some of the conditions contributing to the TMI-2 accident.

An adequate quality assurance and reliability program provides management with insight into the performance of existing organizations, procedures, and practices as well as of the performance of plant equipment. This was not available either to Met Ed or NRC.

FINDINGS

- The NRC organization, procedures, and practices do not provide the necessary management, engineering, and quality assurance review of utility performance to assure early identification and qorrection of deficiencies in utility systems, procedures, and practices.

- There is a lack of <u>independent</u> on-site quality assurance or safety assessment of non "safety-related" equipment and systems.

- There is no <u>independent</u> quality assurance or safety assessment of plant operations and procedures.

'See Appendix B on Definition and Application of "Safety-Related".

- There is a lack of detailed safety and failure modes analyses of all plant systems.

- Systems engineering, such as systems interaction, and the interaction between the many facility systems themselves and with operators has generally not been considered in the NRC overview process.

- There is no comprehensive non-conformance, problem-reporting, failure analysis, corrective action review. The current licensee event report (LER) system does not assure adequate total systems consideration to the event nor dissemination of and attention to the lessons learned by all elements of the industry.

- Full use is not being made of management, engineering, safety, reliability, and quality assurance practices that are used in other industries where safety and reliability are critical.

PRE- AND POST-ACCIDENT SECURITY STATUS AT THREE MILE ISLAND

A review was made by LASL of the physical security measures in place at Three Mile Island before and after the accident on March 28, 1979. The study concluded that before the accident, the plant security complied with 10 CFR 73 and was protected from external attack, but that there was not adequate protection against sabotage by an insider. The same situation exists today except that protection from external attack has been enhanced. Some details of the study are summarized below.

In regard to sabotage by an insider, it is the control of vital areas that is of concern at TMI. A vital area is defined in 10 CFR 73.2 as "any area which contains vital equipment within a structure, the walls, roof, and floor of which constitute physical barriers." Vital equipment is in turn defined as:

> any equipment, system, device, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation. Equipment or systems that would be required to function to protect public health and safety following such a failure, destruction, or release or which could directly or indirectly endanger the public health and safety by exposure to radiation are considered vital. Equipment or systems that would be required to function to protect public health and safety following such a failure, destruction, or release are also considered vital.

The NRC defines Type I vital areas as those wherein successful sabotage can be accomplished by compromising or destroying the vital systems or components located within this area. All other vital areas are Type II.

Only the control rooms and containment buildings are considered to be Type I at TMI although the turbine buildings, the diesel generating buildings, and the auxiliary and fuel handling buildings also meet the definition. The "two-man" rule requiring that no one be allowed in a Type I vital area without another person accompanying him had not yet been implemented.

The plant was inspected by Region I of the Inspection and Enforcement Office of the NRC in March 1979 and was found to be in compliance with the existing rules except that some vital area doors that should have been locked and guarded were found to be open and unguarded. There was in fact very little protection against the sabotage actions of the insider. There was little or no control of the whereabouts of people inside the vital area; so it cannot be said that sabotage to the auxiliary feedwater system was impossible.

During the period March 28 through April 6, 1979, all persons not required for safety operations were barred from the Island. During this time compensatory measures for loss of intrusion detection were not **in** place. On April 6, 1979, conditions reverted to what they had been before the accident.

The accident seriously delayed the installation of security hardware at TMI but certain measures have been taken to enhance security beyond the status prior to the accident. The same situation exists now, however, as existed before with respect to vital areas. Although the plant is considered in compliance with 10 CFR 73.55, there is still little protection against the actions of an insider. Approximately 1,500 persons have unescorted access to the Island, 900-1,000 have unescorted access to TMI-2 and 500 to TMI-1. Most of these people are contractor personnel.

The general attitude seems to be that under the current circumstances, with large numbers of contractors having to have access to TMI-2, not much can be done to improve security. Repairs and cleanup should be executed as rapidly as possible to return to a manageable security situation.

THREE MILE ISLAND SABOTAGE ANALYSIS

TMI-1 and TMI-2 were evaluated to determine where sabotage could be accomplished. Sabotage for this study is considered to be any act resulting in the unplanned release of radioactivity or the compromise of plant radiological safety. Information from site visits, FSAR studies, and other documentation was used to prepare detailed sabotage event trees for both TMI units. Output of the analysis is an ordered list of combinations of locations where successful sabotage can be accomplished. The analysis assumed that an adversary is a knowledgeable insider possessing explosives.

The study concluded that successful sabotage can be performed at either TMI-1 or TMI-2. Many of the successful sabotage acts described would probably not result in significant radiation release to the public.

Because of the unusual conditions at TMI-2, the event tree for that unit is more complex than for TMI-1; furthermore, the results of sabotage would be more severe, due to the larger quantities of stored radioactive water, gas, and solid material. The following events were identified as possible sabotage events and have been evaluated in depth:

- release of radioactive gas from the reactor building;
- release of radioactive water from the reactor building;
- release of primary coolant water to areas exterior to the reaction building;
- return to reactor to criticality;
- loss of all ac and dc electrical power;
- disablement of all decay heat removal systems;
- release of primary coolant through pressurizer relief line and disablement of make-up systems; and
- radioactive release from liquid, gas, and solid radwaste systems.

CLOSED EMERGENCY FEEDWATER VALVES

These valves were in a closed position at the initiation of the accident at TMI-2 preventing the supply of emergency feedwater to the steam generators when normal feedwater flow was interrupted. It took operators approximately eight minutes to recognize and correct the situation. The control panel lights correctly showed the valves closed at the time of the accident.

A surveillance (routine test) of the emergency feedwater pumps was performed on March 26, 1979, in which the EFVs were to be temporarily closed. Statements at the Commission hearings of those involved were that the EFVs were returned to open positions at the conclusion of that surveillance. An investigation was undertaken to try to determine why the valves were in closed condition at 4 a.m. on March 28, 1979.

THE INVESTIGATION

The investigation looked into a number of possible causes for the closed valves. Each was examined in some considerable detail. In addition, some employee records were examined. The findings are as follows:

- there is no record kept of valve line-up;
- operators are not required to review systematically periodically control room status. There is no checklist for such a review;
- incorrect switch and valve positions have been experienced possibly more often than formal documents indicate;
- many people have access to plant positions from which valves can be operated;
- auditing and inspection of procedures are inadequate;
- procedure changes receive inadequate review; and
- deliberate valve mispositioning cannot be confirmed or completely dismissed.

In regard to the last point, the Commission chairman requested that the FBI reexamine this possibility. The FBI response indicates that they have not found sufficient grounds for further investigation.

SUMMARY

The findings from this analysis are as follows:

- There has been no positive identification of an explanation for the valves being in the closed position.

- Of all the explanations analyzed the most likely explanations, each with comments to the contrary, are:

  a. The valves were not reopened at the conclusion of the most recent surveillance procedure, requiring them to be closed, conducted prior to the accident.

  b. The valves may have been mistakenly closed by control room operators during the very first part of the accident.

  c. The valves may have been mistakenly closed from other control points within the plant.

PAST ACCIDENTS IN NUCLEAR REACTOR FACILITIES

Part I of this report describes selected accidents that have occurred in nuclear reactor facilities worldwide. Included are accidents involving central station power plants, plutonium production reactors, demonstration plants, and experimental and research reactors. The condition for inclusion in this compilation is that the accident fulfill one of the following criteria:

- Caused death or significant injury;
- Released significant radioactivity off-site;
- Results in core damage;
- Causes severe damage to major equipment;
- Was a precursor to a potentially serious accident;
- Resulted in inadvertent criticality; and
- Resulted in significant recovery cost;

Of the 40 accidents considered, 22 resulted from equipment failure, 10 from human failure, and 7 involved both equipment failure and human failure.

By type, there were 27 nuclear accidents and 13 nonnuclear. The latter are defined as cases where criticality of the core was not a factor; either the reactor was unfueled, shutdown, or systems not associated directly with reactor operation were involved. In this type of accident, two people were killed and eight injured. Radioactive release accompanied three.of these but in no case was it significant off-site. All of the nonnuclear accidents involved central station power plants.

Nuclear type accidents to central station power plants resulted in no personnel injuries or deaths. Three Mile Island received by far the most attention because of the nature and duration of the accident and the number of people involved. It was the first central station power plant accident to release more than trivial amounts of radiation. Inadvertent criticality at two power plants did not release any activity or cause any core damage.

The most serious accident radiologically happened to the Windscale production reactor in England when part of the uranium-graphite core was destroyed by a smoldering fire. Milk consumption in a 200 square mile area was restricted because of iodine contamination through animal feed.

The only serious criticality accidents have occurred with experimental and research reactors. In one of these, there were three fatalities and the reactor was destroyed. Serious core damage was incurred at four other reactors of this type when they became supercritical. No serious off-site contamination resulted, however, for any of these accidents.

In the second part of this report, yearly releases of noble gases and halogens are tabulated for power plants operating in the United States. Some of the higher routine yearly releases from operating nuclear power stations have been comparable to the single event release of the Three Mile Island accident.

NUCLEAR POWER PLANT AIRBORNE RADIOACTIVITY RELEASES

The release of fission products from the TMI-2 accident consisted of 2.5 million curies of noble gases, primarily xenon and about 15 curies of iodine-131. The question may be asked, how does this short-term, single-event release compare with the historical record of allowed annual release of fission products from operating reactors? The following two tables present information taken from NRC reports 1,2/, concerning routine releases of noble gases, halogens, and particulates from operating nuclear reactors in the United States. Annual releases that are comparable to releases resulting from the TMI-2 accident are underlined.

In 1975 and 1976, amendments to 10 CFR Part 50 (Appendix I) severely limited the allowed releases from routine operations. The concept of "as low as practicable" releases required power stations to install equipment limiting releases to low values.

The release of radioactive noble gases from TMI-2 led to a low average radiation dose to individuals in the neighborhood and to a collective dose to the total population within a 50-mile radius of about 3,300 person-rems. 3,5/ For comparison purposes, the population doses from operating nuclear power plants in 1975 have been estimated. 4/ These ranged from a high of 750 person-rems to a low of 0.008 person-rems.

1/ NUREG-0077, "Radioactive Materials Released from Nuclear Power Plants," 1974, NRC, June 1976.

2/ NUREG-0521, "Radioactive Materials Released from Nuclear Power Plants, Annual Report 1977," NRC, January 1979.

3/ Report of the TMI Ad Hoc Population Dose Assessment Group, "Population Dose and Health Impact -- the Accident on the Three Mile Island Nuclear Station", May 10, 1979.

4/ "Population Dose Commitment Due to Radioactive Releases from Nuclear Power Plant Sites in 1975," PNL-2439, October 1977, by Baker, Soldat, and Watson.

5/ "President's Commission on the Accident at Three Mile Island - Report of the Task Group on Health Physics and Dosimetry," J. A. Auxier, et al., September 28, 1979, gives an estimated collective dose of 2,800 person-rems.

Table 1: Nuclear Power Plant Airborne Releases
(Curies of Noble Gases (Kr, Xe, etc.)

Plant	1970	1971	1972	1973	1974	1975	1976	1977
Big Rock Point 1	280,000	284,000	258,00	230,000	188,000	50,600	15,200	13,400
Browns Ferry 1, 2, 3					64,000	92,400	<80,500	<166,000
Cooper Station					2,000	19,800	38,000	1,270
Dresden 1	900 000	753,000	877,000	840 000	98,000	520,000	452,000	520,000
Dresden 2, 3		580,000	429,000	880 000	627,000	369,000	323,000	313,000
Humboldt Bay 3	540,000	514,000	430,000	350,000	572,000	297,000	93,000	?
Lacrosse	1,000	1,000	31,000	91,000	49,000	57,100	124,000	42,500
Millstone Point 1		276,000	726,000	79,000	912,000	2,970,000	507,000	620,000
Monticello		76,000	751,000	870 000	1,480,000	155,000	11,400	6,870
Nine Mile Point 1	10,000	253,000	571,000	872 000	558,000	1,300,000	176,000	3,530
Oyster Creek	110,000	516,000	866,000	810 000	279,000	206,000	167,000	177,000
Peach Bottom 2, 3				<1,000	<1,000	13,000	209,000	71,100
Pilgrim 1			18,000	230,000	546,000	46,000	183,000	413,000
Quad Cities 1, 2	--		132,000	900,000	950 000	110,000	33,600	25,600
Vermont Yankee			55,000	180,000	64,000	4,080	3,030	3,350
Arkansas 1					196	1,030	5,690	13,900

Table 1 (cont'd)

Connecticut Yankee	1	3	1	32	7	480	452	3,120
Fort Calhoun				67	303	429	1,940	3,810
H. B. Robinson		< 1	< 1	3,100	2,310	1,170	640	476
Indian Point 1	2	< 1	1	122	611		--	
Indian Point 2		--		15	5,580	8,200	11,600	16,000
Kewaunee				--	3,350	2,450	1,400	2,430
Maine Yankee			< 1	161	6,360	4,090	1,300	286
Oconee 1, 2, 3				9,300	19,400	15,100	43,900	35,600
Palisades			1	454	<1	2,610	30	60
Point Beach 1, 2		1	3	5,750	9,740	44,500	1,910	1,130
Prairie Island 1, 2				9	358	2,170	1,740	673
R. E. Ginna	10	32	12	576	757	10,400	5,520	3,200
San Onofre 1	<1	8	19	11,000	1,780	1,110	416	154
Surry 1, 2			<1	866	55,000	8,040	19,100	10,000
Three Mile Island 1					916	3,630	2,760	16,600
Turkey Point 3, 4				530	4,660	13,400	15,600	23,300
Yankee Rowe	< 1	< 1	< 1	35	40	22	26	12
Zion 1, 2				4	2,290	48,800	114,000	32,200

Table 2: Nuclear Power Plant Airborne Releases
(Curies of **Halogens and Particulates** (half life > 8 days)

Plant	1970	1971	1972	1973	1974	1975	1976	1977
Big Rock Point 1	0.13	0.61	0.15	4.60	0.16	0.12	0.05	0.01
Browns Ferry 1, 2, 3	--				0.12	0.27	< 0.07	0.10
Cooper Station					0.24	0.05	< 0.04	<0.02
Dresden 1	3.3	0.67	2.75	0.04	0.68	0.96	0.84	4.93
Dresden 2, 3	1.6	8.68	5.89	6.70	6.50	4.31	5.49	6.86
Humboldt Bay	0.35	0.3	0.48	0.29	0.84	1.06	0.08	0.004
Lacrosse	< 0.06	< 0.01	0.71	0.20	0.04	0.10	< 0.07	0.17
Millstone Point 1		4.0	1.32	0.20	3.26	9.98	2.33	4.86
Monticello	--	0.05	0.59	1.20	5.69	3.71	0.17	0.08
Nine Mile Point 1	< 0.01	0.06	0.97	1.98	0.89	2.78	2.20	0.20
Oyster Creek	0.32	2.14	6.48	7.02	3.51	5.64	6.39	9.05
Peach Bottom 2, 3				< 0.01	0.01	0.04	0.98	0.27
Pilgrim 1			0.03	0.47	1.45	2.58	0.67	0.69
Quad Cities 1, 2			0.75	5.5	8.88	1.31	1.33	1.69
Vermont Yankee			0.17	0.07	0.36	0.01	<0.01	0.01
Arkansas 1			--		0.05	0.74	0.06	0.01

Table 2 (cont'd.)

Connecticut Yankee	< 0.01	0.03	0.02	0.05	<0.01	<0.01	<0.01	0.002
Fort Calhoun		--		< 0.01	<0.01	<0.01	<0.02	0.01
H. B. Robinson		0	0.03	0.30	0.05	0.02	0.10	0.004
Indian Point 1	0.08	0.21	0.93	0.01	0.11		--	
Indian Point 2				.01	0.43	1.62	0.24	0.06
Kewaunee	--				0.02	0.66	<0.01	0.02
Maine Yankee			< 0.01	0.94	0.05	<0.01	<0.01	0.005
Oconee 1, 2, 3			--	0.01	0.03	0.01	0.27	0.54
Palisades			< 0.01	0.31	0.01	0.38	0.04	0.01
Point Beach 1, 2	--	< 0.01	0.03	0.55	0.16	0.07	0.02	0.005
Prairie Island 1, 2				< 0.01	< 0.01	0.02	0.01	0.008
R. E. Ginna	0.05	0.17	0.04	< 0.01	< 0.01	0.02	0.03	0.03
San Onofre 1	< 0.01	< 0.01	< 0.01	1.61	< 0.01	0.04	< 0.01	0.0002
Surry 1, 2			< 0.01	0.04	0.14	0.05	0.35	0.12
Three Mile Island					< 0.01	< 0.01	0.01	0.03
Turkey Point 3, 4				0.06	3.63	0.43	0.42	1.04
Yankee Rowe	< 0.01	< 0.01	< 0.01	0.19	0.53	0.01	< 0.01	0.0001
Zion 1, 2				< 0.01	0.01	0.14	0.09	0.05

RECOVERY: TMI-2 CLEANUP AND DECONTAMINATION

As a result of the accident on March 28, large quantities of radioactive fission products were released from the damaged reactor fuel rods and distributed throughout portions of the TMI-2 facility. The major fraction of these fission products were short-lived and have largely decayed away. For example, of the approximately 35 million curies of Iodine-131 estimated to have been released from the fuel during the accident, less than one curie remains as of mid-September.

At present, the radioactive material remaining in the facility includes: the damaged core itself; fuel debris that has possibly been transported to locations in the primary coolant system; fission products dissolved and suspended in the primary coolant and in water contained in the reactory containment building and the TMI-2 auxiliary building; gaseous radioactivity in the containment building atmosphere; and radioactively contaminated materials in various forms that have precipitated and settled onto numerous surfaces (equipment and building interiors) in the TMI-2 containment, auxiliary, fuel handling, and diesel generator buildings.

The bulk of the remaining radioactive material that is distributed outside of the fuel is contained in.several volumes of water. This water contains in total approximately 850,000 curies of longlived fission products (mostly cesium-137 and strontium-89 and -90) and consists of approximately: 90,000 gallons in the primary coolant system, 600,000 gallons in the reactor containment building, and about 380,000 gallons in several large tanks located in the TMI-2 auxiliary and fuel handling buildings. The atmosphere in the containment building contains about 51,000 curies of krypton 85 (half life 10.7 years, a noble gas).

Floors, sumps, and equipment surfaces in the above mentioned facilities were extensively contaminated largely due to flooding and subsequent water leakage from tanks. No estimate is available regarding the total amount of radioactive material that is involved in this contamination nor of the total number of curies remaining, but it is generally comprised of the same isotopes as contained in the inventory of contaminated water in the TMI-2 facility.

Work has been under way by Met Ed/GPU, supported by several contractors, since April on the decontamination of floors and accessible equipment in the TIM-2 diesel generator, fuel handling, and auxiliary buildings. Decontamination of floor areas in the buildings is about 80 to 85 percent complete as of October 1. Additional work is limited because of high radiation dose rates in the vicinity of tanks containing contaminated water.

Major remaining cleanup tasks include processing the contaminated water in the primary coolant system, the containment building, and in the auxiliary building and fuel handling building tanks; completion of the decontamination of building interiors and equipment; processing and disposition of the contaminated containment building atmosphere; removal of the damaged reactor core; and disposition of the core and the large volumes of solid radioactive wastes generated by decontamination and cleanup operations.

A system to process (decontaminate) the approximately 380,000 gallons of water contained in the TMI-2 auxiliary and fuel handling building tanks has been designed and installed on the site. This system has been designed to process large volumes of water containing concentrations between one and 100 microcuries per milliliter of cesium-137. The system is known as EPICOR II (named after the company who developed the basic process). Processing of water by the EPICOR II system has been approved by the NRC.

Design work has started for a system to decontaminate the water in the containment building and the primary coolant system. This water contains concentrations of cesium and strontium (the principal isotopes remaining) at higher levels than can be effectively handled by the EPICOR II system, hence the need for a separate treatment system.

Met Ed/GPU, with the assistance of several contractors, has examined alternatives for disposition of the contaminated air in the containment building. A proposal for treatment by filtration to remove radioactive particulates with subsequent venting to the outside atmosphere under controlled conditions to dispose of the krypton has been prepared.

At present, the containment building remains sealed to contain the contaminated air and water. Human entry has not been made because of high radiation levels therein and the need to maintain the building integrity. A preliminary assessment of the containment building entry and decontamination was completed for GPU by Bechtel Corp in July. Development of detailed plans and procedures is continuing by Met Ed/GPU with assistance by Bechtel and other contractors. Detailed assessment of the containment building decontamination effort must await actual entry, radiation mapping, and direct examination of conditions inside the building. Entry is not contemplated until the containment building atmosphere has been decontaminated or purged and the contaminated water removed. Entry is not expected until January 1980, at the earliest.

A joint agreement has recently been reached between GPU, DOE, NRC, and EPRI regarding research and development needs related to TMI cleanup and recovery among other things. One of the primary tasks under this agreement is a preliminary assessment of the handling and disposition of the damaged reactor core and associated components. This assessment will also identify the facilities necessary to handle the core and accomplish the ultimate disposition of the core fuel material. The task group will also prepare recommendations concerning the necessity to use government-owned facilities for receipt and disposition of the fuel.

The final major aspect of the TMI cleanup is the handling and disposition of radioactive waste generated as a result of decontamination and recovery activities. This waste consists of several major categories of material. These include: filter beds and spent ion-exchange resins produced from processing contaminated water; filters from air cleaning systems; clothes, rags, shoe covers, tools, and small equipment used in decontamination work; damaged equipment removed from contaminated areas; temporary shielding and construction materials used in cleanup and recovery operations; and sludges and residues from decontamination solutions. Preliminary design work has begun for a large-scale liquid evaporator facility (30 gallons per hour) to concentrate the radioactivity from the large volumes of liquid decontamination solutions that are expected to be generated in the containment building cleanup. It has been estimated that about 500,000 cubic feet of radioactive waste material will be produced in the cleanup of all TMI-2 facilities.

To date, 12 truck loads of solid radioactive waste have been shipped from TMI since the accident. Initially, it was intended to ship wastes from the cleanup in accord with the previous practice of shipping routine radioactive wastes from normal plant operations to the nearest commercial disposal site at Barnwell, South Carolina. However, the governor of South Carolina intervened and the Barnwell site was, in effect, prohibited from receiving any post-accident wastes from TMI. Subsequently, arrangements were made to ship waste from TMI cleanup to the commerical burial site at Richland, Washington. The initial arrangement, with which Washington public officials have concurred, is for the receipt of approximately 200 shipments (truckloads) over the next 2 years. This agreement is intended to accommodate the bulk of solid wastes expected from the use of the EPICOR II system in processing the water currently stored in the TMI-2 auxiliary and fuel handling buildings and from the decontamination of the auxiliary, diesel generator, and fuel handling buildings. This does not include wastes from the processing of water in the containment building and primary coolant system, nor wastes from the containment building decontamination and refurbishment. Radioactive wastes from these operations are expected to amount to as many as 2,000 shipments. Specific agreements and plans for the disposition of this material have not yet been made.

A special problem may be presented in the disposal of ion-exchange resins used to process the higher activity contaminated water. The radioactivity removed from the water will be concentrated onto a relatively small volume of resin. These resin beds are expected to contain concentrations of cesium and strontium ranging from several curies per cubic foot up to several thousand curies per cubic foot. Present federal policy and regulations on the subject are not well defined, but it appears that this material would be classified as high-level waste and hence precluded from disposal at existing commercial radioactive waste disposal facilities. Only very preliminary estimates of the amount of this material expected from the cleanup are available. These estimates indicate that from 2,000 to 6,000 cubic feet of high activity resin will be produced. The resins will be dewatered and solidified on the TMI site before shipment.

Cleanup operations entail work in high radiation areas and the handling of highly radioactive materials and will present risks to workers from exposure to radiation, in addition to the accident risks associated with a large scale industrial operation. Precautions will be taken for worker protection such as protective clothing, extra shielding, remote handling, and respiratory protection (such as filter masks and self-contained breathing apparatus), where appropriate.

Some information is available regarding the radiation exposure experience associated with cleanup and recovery efforts conducted thus far. For the third calendar quarter of 1979, the collective exposure for decontamination workers was 26 person-rems. A total of 182 workers were involved. By way of comparison, the 3-month total (June - August) for all on-site personnel at TMI was 285 person-rems. The average on-site population during this period was about 3,000. Thus far whole-body counting and bioassay results on decontamination workers have not shown detectable uptake of radionuclides. However, in August, five workers involved in a maintenance procedure received overexposures (in excess of NRC limits) to the skin and extremities.

In the aftermath of the accident, extensive environmental radiation surveillance programs were established by several federal agencies in response to the accident. The various agency efforts have since been consolidated in a comprehensive long-term surveillance program. It is designed to provide monitoring of air, water, and direct radiation, as well as selected food pathways. The plan contains provisions to follow cleanup and recovery operations and contains emergency notification and response procedures. An agreement has recently been signed by the Federal EPA, NRC, DOE, HEW, and the state of Pennsylvania. In addition, a protocol has been established between Met Ed, NRC, and the state of Pennsylvania regarding notification and monitoring of all radioactive waste shipments leaving the site.

The volume of contaminated water that must be contained continues to grow. This is from two sources. First, in-leakage of noncontaminated water from miscellaneous sources continues to flow into the numerous sumps and drains in the TMI-2 auxiliary building. This in-leakage which has been reduced to the extent possible, amounts to about 800 to 1,000 gallons per day. The in-leakage mixes with a smaller amount of contaminated water which has leaked from the contaminated water holding systems into the sumps. It then must be collected and routed to the radioactive water storage tanks. Met Ed has projected that at current in-leakage rates, they will be down to a 10,000 gallon reserve unused storage capacity in TMI-2 by mid-October. The recent NRC decision to permit processing of the intermediate level water by EPICOR II should provide an additional margin of storage capacity for the remaining untreated water.

The other source of increase in the contaminated water is the approximately 1,000 gallons per day that leaks from the primary coolant let-down system directly into the containment building from pump seals and glands in the primary system. This leak rate represents the minimum loss rate from the let-down system under present plant conditions. This loss must be replaced in the primary system through the make-up system to maintain coolant inventory and boron concentration in the coolant. This does not appear to present any immediate problems, however.

A number of preliminary conclusions regarding the cleanup and recovery can be drawn. It is clear that the clean-up and recovery of the TMI-2 facility from the accident of March 28, 1979, represents a task in both magnitude and complexity that has not been previously encountered by the U.S. civilian nuclear power industry. This is easily borne out on the basis of preliminary cost estimates for the cleanup which range from about $100 to $200 million. It is also apparent that extensive experience in the decontamination and recovery of a large number of nuclear facilities has been gained over the past 30 years by both governmental and civilian organizations. Successful completion of cleanup and recovery operations that include tasks faced by the TMI-2 cleanup have been performed at various facilities, including the handling of damaged irradiated reactor cores.

It can be concluded on the basis of present information, and with appropriate caveats, that the clean-up and recovery can be successfully completed using presently available technology. That is to say, the scientific and practical experience base in the United States is adequate to do the job. Engineering and chemical process development work is required, however, and is underway for various tasks. It is possible that facilities and expertise of DOE and its contractors will be necessary for the removal, handling, and disposition of the damaged reactor core. This depends in part upon decisions yet to be made regarding the interim and ultimate disposition of the fuel material after it is removed from TMI.

Additional engineering development work may be required in order to satisfy environmental release constraints that could be applied to the TMI-2 cleanup. For example, if Met Ed is precluded from disposing of the 51,000 curies of krypton-85 presently in the containment building air by venting to the outside atmosphere, alternatives such as cryogenic trapping, absorption on charcoal, or concentration and storage under pressure will have to be considered. None of these potential alternatives have been successfully demonstrated on the scales necessary for TMI-2.

Attention should also be given to institutional and political issues, as well as health and safety, engineering, and financial aspects when assessing the likelihood of successful cleanup of TMI. One aspect of this is the capability of the utility organization in terms of both its financial capability and its ability to manage the complex cleanup and recovery task. The second aspect is the uncertain regulatory and political climate in which the cleanup and recovery from the accident is conducted.

The Met Ed/GPU organization recognized that it did not possess the in-house experience and capability to manage and perform the cleanup and recovery from the accident. They have hired a number of commercial firms and consultants (including consultants from DOE) to assist in planning and implementing the cleanup and recovery activities on the site.

One area of concern that has been noted is the lack of experience in the Met Ed/GPU organization in operating in radiation environments such as presented by the post-accident situation at TMI-2. The GPU organization appears to have recognized this problem as it has taken steps to strengthen its health physics organization through the acquisition of additional professional staff members.

Continued presence of materials in the TMI facility dispersed in the large volumes of air and water present increased risk (however small) of uncontrolled release to the environs. The orderly, systematic cleanup and decontamination of the facility with concentration and confinement of the radioactive materials would result in an overall reduction in exposure risk to both workers and members of the public living in the vicinity of TMI.

FINDINGS

1. The TMI-2 facility cleanup and decontamination represents a task that is greater in magnitude and complexity than previously encountered in the U.S. commercial nuclear power industry.

2. Cleanup cost is expected to be between $100 and $200 million. This does not include costs for refurbishment and return of TMI-2 to service.

3. Overall planning and task definition, and the development of a preliminary schedule have been completed. The entire cleanup is expected to take at least 2 years.

4. The cleanup, concentration, and confinement of radioactive materials presently dispersed in large volumes of air and water contained in the facility will result in the reduction of radiation exposure risks to both workers and members of the public.

5. The cleanup will produce large volumes of radioactive waste materials (over 500,000 cubic feet) which must be disposed of. Final disposition of the radioactive waste and the damaged reactor core is yet to be determined.

COST OF THE ACCIDENT

The accident at Three Mile Island on March 28, 1979, generated considerable economic disturbance. Some of the impacts were short term, occurring during the first days of the accident, while others have yet to occur. Many of the impacts were experienced by the local community; others will be felt at the regional and national levels. To add to the understanding of the effects of the accident, the Stanford Research Institute was asked to make a study of the costs of the accident.

The assessment was carried out during a 7 week period 4 months after the accident. The purpose of the effort was not to develop an exhaustive data base from which the costs of the accident could be determined precisely, but rather to estimate the approximate magnitude of the accident, using the best data available at this time. The estimates are based on cost data provided by affected parties. Wherever possible, an attempt was made to verify the estimates through consistency checks and checks with other sources. The information on which this report is based is necessarily sketchy, as not all of the impacts have occurred, nor have all of the affected parties fully accounted for their costs. We therefore expect that future assessments of the economic impacts of the accident will change as later technical and accounting information become available.

The economic impact of the accident was organized into two broad categories: those incurred as a direct result of the accident; and indirect income losses and other potential impacts on the growth of regional and national economies.

The following lists the categories of expenditures directly attributable to the accident.

Emergency Management	Replacement Power	Replacement Capacity	Health Effects
Evacuation Management	General Public Utilities	Decontamination	Physical Effects
Radiation Monitoring	Other Babcock & Wilcox Plants	Waste Disposal	Mental Effects
Plant Management		Refurbishment or Decommissioning and Replacement	Health Monitoring
Investigative Studies			

Other impacts considered are both directly and indirectly realized at the local, regional and national levels. The following were considered here:

<u>Loss of Local Income</u>

<u>Business Disruption</u>

- Manufacturing
- Tourism
- Agriculture
- Real Estate

<u>Changes in Regional Growth</u>

- Gross Regional Product
- Employment

In the cases in Table 3, the costs of decontamination of TMI-2 are included with the Plant Refurbishment costs or the Plant Replacement costs.

DIRECT AND INDIRECT EFFECTS ON THE LOCAL, REGIONAL, AND NATIONAL ECONOMY

The accident at Three Mile Island resulted in a number of economic impacts for which direct expenditures were not made. These impacts were felt at the local, regional, and national levels. The following summarizes those effects.

Direct Impacts on the Local Economy

Manufacturing Sector

Based on a survey by the State of Pennsylvania and preliminary survey data gathered, it is estimated that lost wages in the local manufacturing sector range from $5.7 million to $8.2 million, with a most likely estimate of $6.3 million. The Pennsylvania Department of Commerce survey covered 383 manufacturing establishments, representing 80,720 employees, in the affected area. The survey covered all firms with 100 or more employees, all food processors, and a representative sampling of the remaining firms. The data gathered appear to correlate well with the state findings. The surveys found that most of the losses occurred within a few days of the accident and quickly subsided thereafter. The available data indicate no evidence of permanent layoffs resu.lting from the accident.

A few food processors incurred extraordinary expenses as a result of the accident. Some firms purchased equipment to detect radiation levels and converted their dairy production to powdered milk. These expenses are estimated to be $250,000.

TABLE 3: Summary of Direct Expenditures 1/
Assuming Refurbishment
(Millions of Dollars)

	Low 3/	Medium 4/	High 5/
Emergency management	$ 120	$ 160	$ 225
Replacement power 2/	678	966	1128
Plant refurbishment	249	306	503
Health effect 7/			
Total	$ 1047	$ 1433	$ 1858

Summary of Results Assuming Capacity 1/
Replacement with Coal 6/
(Millions of dollars)

	Low	Medium	High
Emergency management	$ 120	$ 160	$ 225
Replacement power 2/	1386	1506	1746
Plant replacement	486	503	614
Health effects 7/			
Total	$ 1974	$ 2170	$ 2585

Summary of Results Assuming Capacity 1/
Replacement with New Nuclear Facility 6/
(Millions of Dollars)

	Low	Medium	High
Emergency management	$ 120	$ 160	$ 225
Replacement power 2/	1626	1746	2106
Plant replacement	538	593	719
Health effects 7/			
Total	2284	2500	3050

1/ Cost is in 1979 dollars.

2/ Assumes replacement costs for TMI-1 are $10 million/month; $14 million/month for TMI-2.

3/ Low estimate assumes TMI-1 resumes service in January 1980; TMI-2 in January 1983.

4/ Medium estimate assumes TMI-1 resumes service in January 1981; TMI-2 in January 1984

5/ High estimate assumes TMI-1 resumes service in April 1981; TMI-2 in January 1985.

6/ For replacement cases, Low, Medium, and High are a range of estimates.

7/ The costs associated with health effects have been deleted from this table. The costs projected by the study had a minimal effect on the total costs projected. The Commission believes that the analysis of health effects costs was insufficient to reach the conclusion set out in the study.

Nonmanufacturing Sector

a

Surveys of the nonmanufacturing sector have been begun by the State of Pennsylvania, but none is yet completed.

One industry of particular concern is tourism. Lacking estimates from state and local officials on the accident's impact on tourism, SRI contacted a major resort representing approximately 10 percent of the total south-central Pennsylvania tourist trade. Declines in tourism during the period of the accident were noted by the resort, but it should also be pointed out that there was a gasoline shortage and polio scare during the same period. Based on these discussions and extrapolating to seasonally adjusted tourism figures for the five-county area, it is estimated that total lost tourism revenues of $6 million to $8 million, with a most likely estimate of $6.5 million. Lost wages in the tourism sector are estimated to range from $2.8 million to $3.8 million.

Losses to the agricultural sector from the accident appear to be minimal. For nondairy agricultural firms, the accident occurred during the off season, with both employment and production relatively low. Initial results of a state Department of Agriculture study indicate that the losses in agriculture were "significantly less" than $1 million.

Immediately following the accident, considerable concern was expressed by members of the community that land values in the vicinity of the plant would decline. However, an inspection of county records and interviews with realtors, community development officials, state and local government officials, and private home owners indicate that no such decline in property values has occurred.

Total direct impacts on the local economy are summarized in Table 4.

Impact of Capacity Replacement on the Local Economy

The replacement of capacity will have a significant impact on the economy of the Harrisburg area. Refurbishment, or the construction of a new plant, will involve hundreds of workers on-site at the plant and expenditure of tens of millions of dollars for equipment and materials in the vicinity of the plant. These effects are summarized below for the various conditions of refurbishment, decommission, and replacement of plant with another nuclear or a coal facility.

Impact of Plant Disposition on Local Economy

- Impact of refurbishment effort on local economy

  Increased employment = 1,900 persons per year

  Increased earnings = $13 million per year

TABLE 4: Direct Impact on the Local Economy
(Millions of Dollars)

	Low	Medium	High
Manufacturing			
Lost wages	$ 5.7	$ 5.3	$ 8.2
Extraordinary purchases	0.2	0.3	0.5
Nonmanufacturing			
Tourism			
Lost revenues	6.0	6.5	8.0
Lost wages	2.8	3.1	3.8
Agriculture			
Lost wages	0.0	0.1	1.0
Real Estate	(No appreciable effect)		
Other nonmanufacturing			
Lost wages	8.6	9.6	14.6
Total lost wages and extraordinary purchases	$17.3	$19.4	$28.1

Note: Lost revenues in the tourism sector are not included in the total of lost wages and extraordinary purchases.

- Impact of decommission on local economy

  Increased employment = 2,000 persons per year

  Increased earnings = $20 million per year

- Impact of replacement construction on local economy

  Nuclear:

  Increased employment = 1,800 persons per year

  Increased earnings = $17 million per year

  Coal:

  Increased employment = 1,800 persons per year

  Increased earnings = $18 million per year

<u>Indirect Impact of Higher Electricity Prices on the GPU Service Area</u>

The long run, indirect effect of the replacement power costs on the GPU service area is uncertain at this point, due to regulatory and financial uncertainties.

<u>Summary</u>

It appears clear that the major costs of the TMI-2 accident are associated with the emergency management replacement power and with plant refurbishment or replacement. The minimum cost estimate of nearly $1 billion supports the argument that considerable additional resources can be cost effective if spent to guard against future accidents.

IODINE FILTER PERFORMANCE

During the accident at Three Mile Island a quantity of Iodine 131 was detected in the gaseous effluent. This quantity was more than what would be expected to pass through the filtering system if it performed as designed. Replacement charcoal in the auxiliary building and the fuel handling building ventilation systems reduced iodine effluent levels significantly suggesting that charcoal in the filter trains at the onset of the accident did not perform as it should.

Investigation determined that airflow normally bypasses the filters for control room, auxiliary building and fuel handling building exhaust and, if the level of radioactivity in the airstream reaches a predetermined level, airflow is directed to pass through the filter. Charcoal in use in the filters was purchased in 1975. It met the regulatory requirements in existence at that time but did not conform to the requirements in effect at the time the TMI-2 license was issued. The NRC approved use of the charcoal that was installed and waived the surveillance requirements in the operating license technical specification for the fuel handling building and control room air cleanup systems. Such surveillance was intended to verify correct filter performance. There was no such surveillance required for the auxiliary building system filter performance.

Although the air filtering systems were designed to be used only when needed to remove airborne radioactivity because of limited filtering lifetime for charcoal, the filters had been in use about one year. This fact coupled with the initial underspecification charcoal and the lack of surveillance to verify system performance could explain apparently inadequate filter performance during the accident.

Samples of charcoal filters removed from the auxiliary building and fuel handling building filters trains during the accident were tested for removal efficiency. These tests showed a degradation in removal efficiency for methyl iodide (which is a standard test medium). Charcoal samples from each filter train indicated significant reduction in removal efficiencies, with the highest, 75.6 percent, and lowest, 49.1 percent, from fuel handling building A and B trains, respectively. Removal efficiencies obtained for the auxiliary building A and B trains were 69.5 percent and 56.0 percent, respectively. New charcoal meeting current specifications should have a filtering efficiency for methyl iodide of 99 percent.

WASH 1400

WASH 1400 (The Rasmussen Report) was published in 1975. It was intended to estimate the probabilities of occurrences of accidents involving radioactivity release and to assess the risk of such accidents relative to other risks. The study involves (1) a list of potential accidents in nuclear reactors, (2) estimation of the likelihood of accidents resulting in radioactivity release, (3) estimation of health effects associated with each accident, and (4) comparison of nuclear accident risk with other accident risks. The study determined that the nuclear accident risk was small - almost negligible compared with more common risks.

The WASH 1400 risk assessment was subsequently reviewed by a Risk Assessment Review Group in 1977 (the Lewis Report) that concluded that "they were unable to determine whether the absolute probabilities of accident sequences in WASH 1400 are high or low, but believes that the error bounds on those estimates are, in general, greatly understated." They went on however to say:

> WASH-1400 was largely successful in at least three ways: in making the study of reactor safety more rational, in establishing the topology of many accident sequences, and in delineating procedures through which quantitative estimates of the risk can be derived for those sequences for which a data base exists.
>
> Despite its shortcomings, WASH-1400 provides at this time the most complete single picture of accident probabilities associated with nuclear reactors. The fault tree/event tree approach coupled with an adequate data base is the best available tool with which to quantify these probabilities.
>
> WASH-1400 made clear the importance to reactor safety discussions of accident consequences other than early fatalities.

The NRC accepted the findings of the Risk Assessment Review Group and issued a statement which said in part:

> The Commission accepts the Review Group Report's conclusion that absolute values of the risks presented by WASH-1400 should not be used uncritically either in the regulatory process or for public policy purposes and has taken and will continue to take steps to assure that any such use in the past will be corrected as appropriate. In particular, in light of the Review Group conclusions on accident probabilities the Commission does not regard as reliable the Reactor Safety Study's numerical estimate of the overall risk of reactor accident.
>
> With respect to the component parts of the study, the Commission expects the staff to make use of them as appropriate, that is, where the data base is adequate and analytical techniques permit. Taking due account of the reservations

expressed in the Review Group Report and in its presentation to the Commission, the Commission supports the extended use of probabilistic risk assessment in regulator decisionmaking.

It is important to note that the Risk Assessment Review Group, while critizing the risk assessment of WASH 1400 per se, commended the description of accident sequences and the "fault tree/event tree" approach as an analytic tool for quantifying probabilities of accidents.

The failure of a pressurizer relief valve to close is discussed in WASH 1400 and its likelihood was predicted on the basis of actual experience with relief valves. WASH 1400 goes on to state that normal response to this failure is actuation of emergency core cooling to avoid excessive loss of water from the reactor. It states that failure to remove heat from the core could lead to core meltdown or damage and that operator action is required to prevent meltdown. (In TMI-2 the operators turned off the emergency core cooling system.)

Although the absolute risk assessment of WASH 1400 was questioned, the message that the reactor accident risk is dominated by the small-break loss-of-coolant accident and by transients initiated accidents is quite clear and was not contested. Thus, emphasis should have been given in reactor research, design considerations, operator training, and safety procedures to the amelioration of these events. This does not seem to be the case. NRC and the industry are still focusing on the "design basis (large) accidents" that admittedly have great consequence but low probability of occurrence.

LESSONS THAT SHOULD HAVE BEEN LEARNED FROM WASH 1400

WASH 1400 contains three important messages. These involve expected frequency of accidents, methods for improving reactor safety, and the most likely types of accidents. Perhaps it is a fault of the report that these messages were not emphasized, because the conclusions most often associated with WASH 1400 -- reactors are safe -- receives the primary emphasis in the report. Perhaps it is the fault of the NRC that more effort was dedicated to criticize, WASH 1400 then was applied to understand its messages. In fact, WASH 1400 predicted that accidents could happen although most would present little or no public hazard. One message of WASH 1400 is that while nuclear accident risk is small compared to other societal risks, accidents similar to Three Mile Island should have been expected. These accidents were not emphasized in WASH 1400, because they do not contribute as significantly to risk as the more severe core melt accidents (See Rasmussen deposition, Sept. 15, 1979, pp. 35-36).

The WASH 1400 study, in using the "event tree" and "fault tree" methodologies, borrowed from the aerospace industry, actually revealed a "weak link" in the safety of the Surry reactor. This led directly to a change in inspection procedures at Surry and reduced the probability of one major risk contributing accident (see Rasmussen deposition, Sept. 15, 1979 pp. 26-29) by a factor of 20 (p. 63, Main Report). Thus, another message of WASH 1400 is that application of these methods to analysis of a specific reactor should be used to reveal "weak links" in safety.

Recently, NRC officials have endorsed a plan to apply WASH 1400 techniques to the analysis of other existing reactors for this purpose (see Levine deposition, Sept. 15, 1979, pp. 25-26). Since the accident at Three Mile Island, NRC has applied reliability analysis to the study of auxiliary feedwater availability in all U.S. commercial reactors.

Reactor safety research, both before and after WASH 1400 was published, has concentrated on the double ended pipe break, or large loss-of-coolant accident. Safety systems were designed specifically to accommodate this accident. Yet, the WASH 1400 results published in 1975 indicated that reactor accident risk is dominated by small-break loss-of-coolant accidents and transient initiated accidents, like Three Mile Island. <u>A third message of WASH 1400 is the relative efforts in reactor safety research for large loss-of-coolant accidents, and transient initiated accidents should be consistent with priorities suggested by their relative risk contributions.</u> Generally, NRC has based priorities on "good engineering judgment" (see Rasmussen deposition, Sept. 15, 1979, pp. 56-57), although the Lewis Report and the NRC commissioners have recently endorsed the use of WASH 1400 techniques to carry out more effectively the licensing. In fact, the NRC staff has successfully applied the techniques to prioritize safety issues, overpressurizing of vessels, and optimization of inspection time intervals (see Rasmussen deposition, Sept. 15. 1979, pp. 58-59).

It should be noted with regard to small-break loss-of-coolant accidents (LOCAs) that it was thought by NRC that safety systems designed to accommodate large LOCAs would necessarily be adequate to deal with small LOCAs (see Budnitz deposition, Aug. 27, 1979, pp. 28-30). It should have been clear from WASH 1400 treatment of PORV transient-initiated LOCAs that such was not the case. Instead, WASH 1400 was taken by NRC as an affirmation of their good regulatory work (see Budnitz deposition, Aug. 27, 1979 pp. 33).

Further, procedural considerations inhibit the application of WASH 1400 techniques. It is very difficult to apply properly the techniques, and few people are trained or experienced in such work (see Levine deposition, Sept. 15, 1979 pp. 20-21). Also, the criticisms of WASH 1400 techniques by NRC Commissioners left the NRC staff unmotivated to develop ways to apply the techniques. Since the Lewis Report and the Three Mile Island accident, this trend appears to be reversing.

<u>TMI-2 AND WASH 1400 RISK ASSESSMENT</u>

If WASH 1400 predictions of the best estimate probabilities are valid, there was a 13 percent chance of having an accident at the time of TMI-2. Further, there was an 80 percent chance that the accident would occur in a PWR rather than a BWR. The WASH 1400 upper bound probabilities yield a predicted 80 percent chance of having an accident after 400 reactor-years of operations of nuclear power systems in the United States. The TMI-2 occurrence is therefore within the bounds of the WASH 1400 predictions.

Fault tree analysis techniques of the WASH 1400 type are extremely valuable to determine where effort can best be put to insure reduction of failure rates of critical elements of existing plants and proposed new designs as rapidly as experience and technology permit.

APPENDIX A

MELTDOWN: A PERSPECTIVE IN HISTORY

Power reactors in the early 1960s did not have emergency core cooling systems, The emphasis, vis-a-vis engineered safeguards, was on depressurizing the containment and containment cooling systems. Dose calculations were made for the "maximum credible accident," a rupture of the largest recirculation cooling pipe, from the standpoint of fission product inventory in the containment after the accident. It was assumed that the cladding failed and fuel melted to provide the source term of fission products, but any other consequences of fuel melting were not considered.

By 1965, the loss-of-coolant accident was highly emphasized in analytical safety studies. The concentration was on modeling, and on evermore sophisticated computer techniques. (LOFT was just getting under construction.) The major emphasis was on coolant blowdown phenomena and heat transfer and fluid flow. Meltdown models were developed in an attempt to describe premelting, sumping of fuel, and melting of the core support plate. The consideration of melting through the vessel was not being handled analytically.

At this time, meltdown was being described in a very simple fashion. From BMI-1779:

> In the course of a loss-of-coolant accident in a power reactor it is very likely that in the absence of preventative measures fuel-pin temperatures will rise above the melting points of the constituent materials and geometry change and release of fission products will result.

By early 1966, the AEC regulatory staff (and the ACRS) began concentrating on requirements for emergency core cooling systems. The question focused on the possibility of "core melt" as opposed to "fuel melt." The Advisory Task Force on Power Reactor Emergency Cooling, under Dr. Bill Ergen focused on how ECCS would be improved to **prevent substantial meltdown,** and looked into what might happen in a LOCA involving large molten masses of fuel.

The emphasis of core melt studies shifted to assuring that ECCS would keep fuel temperatures well below clad melt temperatures. **A** limit of 2,600°F shrunk to 2,300°F, and finally in the early 1970s, after the rulemaking hearing, to 2,200°F.

It was recognized in small reactors such as the N.S. SAVANNAH, without ECCS, that a major loss-of-coolant accident would result in core melt, and thus time-to-melt criteria were developed such that the ship could be moved away from populated areas before fuel melt occurred.

"Meltdown" has become synonymous with "core melt." It is not meant to be, or should not be considered, the same as "fuel melt." In Fermi 1 and the MTR there were incidents involving a small amount of fuel melting.

These were not considered core melt accidents, although the distinction is only a matter of degree. No one has clearly defined the percentage of a core that must experience fuel melt (not clad melt) to have a definable meltdown situation. In general, however, the context is usually stated in terms of a "substantial fraction of the core."

APPENDIX B

DEFINITION AND APPLICATION OF "SAFETY-RELATED"

"Safety-related," "safety-grade," and similar descriptive terms are applied to equipment by NRC and the TMI-2 utility company and their contractors. This application has a significant affect on the way the equipment is treated and on the way related procedures and training are effected. The significance of these terms has been investigated.

A description of the NRC meaning of safety or nonsafety is described by the following quotation:

> In the licensing process, the specification of the design basis event has resulted in the classification of systems into two types -- safety related and nonsafety-related. The reliability and quality of safety systems are controlled through NRC requirements for their design, construction and operation. The NRC requirements for nonsafety systems are generally limited to assuring that they do not adversely affect the operation of safety systems.

The investigation made use of pertinent document review, interviews, and depositions.

The results of this investigation are shown in the findings and conclusions shown below:

<u>FINDINGS</u>

- Significant misunderstanding exists among NRC and TMI-2 personnel regarding the meaning and application of terms such as safety-related", "safety-grade", and similar terms.
- Misunderstanding exists among NRC and TMI management and project personnel as to what specific hardware is considered safety-related and what specific document defines that hardware.
- The lack of clear designation of "safety-related" equipment and specifically what that means contributed to inadequate hardware and procedure review and failure analysis and corrective action that are necessary to assure safe operation of the plant.

REPORT OF THE
TECHNICAL ASSESSMENT TASK FORCE

ON

SUMMARY SEQUENCE OF EVENTS

BY

Jasper L. Tew

October 1979
Washington, D.C.

TABLE OF CONTENTS

INTRODUCTION

This report summarizes the technical sequence of events of the TMI-2 accident of March 28, 1979. It also provides some commentary and/ or explanation of critical events. Except where otherwise noted, the commentary and explanations are based upon analysis by the technical staff. The summary includes a statement of the conditions of the plant at the start of the accident and carries through approximately the first 16 hours, at which time forced cooling of the reactor was reestablished placing the plant in a relatively stable condition. The events herein are extracted from the much longer "Catalog of Events" available in the Commission files, at the National Archives.

Three appendices are attached which provide further discussion of:

A. Decay Heat Removal Methods -- TMI-2

B. Significant Equipment Problems

C. Incorrect Operational Actions

The "Summary Sequence of Events" has been separated into the time periods where significant events occurred.

A. PLANT CONDITIONS PRIOR TO THE INITIATING EVENT

Three Mile Island Unit Two (TMI-2) was at 97 percent (of 2,772 megawatts) power with the integrated control system in full automatic. Rod groups one through five were fully withdrawn, rod groups six and seven were 95 percent withdrawn, and rod group eight was 27 percent withdrawn. Reactor coolant system (RCS) total flow was approximately 138 million pounds per hour and the reactor coolant system pressure was 2,155 psig. Reactor coolant make-up pump B (MU-P-1B) was in service supplying make-up and reactor coolant pump seal injection flow. Normal reactor coolant system let-down flow was approximately 70 gpm. The reactor coolant system boron concentration was approximately 1,030 parts per million. The pressurizer spray valve (RC-V1) and the pressurizer heater bank number 4 was in manual control while spraying the pressurizer to equalize boron concentrations between the pressurizer and the remainder of the reactor coolant system. The pressurizer relief valve and safety valves discharge header temperatures were high enough to give periodic alarms (at about 200°F) and continuously indicated temperatures in the range of 180°F to 200°F. These temperatures are significantly above the maximum allowable operating temperature of 130°F.

The steam generator and associated secondary plant conditions prior to the accident were as follows:

	Steam Generator A	Steam Generator B
Feedwater Flow	5.7459 MPPH*	5.7003 MPPH*
Operating Range Level	56.0%	57.4%
Startup Range Level	158.8 Inches	163.4 inches
Steam Pressure	910 psig	889 psig
Feedwater Temperature	462°F	462°F
Steam Temperature	595°F	594°F

*million pounds per hour

Main feedwater pumps (FW-P-lA and FW-P-1B), condensate booster pumps (CO-P-2A and CO-P-2B) and condensate pumps (CO-P-lA and CO-P-1B) were in service.

B. THE INITIATING EVENT

For approximately 11 hours prior to 4:00 a.m. on March 28, 1979, TMI-2 operating personnel had been attempting to transfer spent resin from an isolated condensate polisher unit to the resin regeneration system. The resin was apparently clogged in the outlet of the polisher and the operators were injecting a water-and-air mixture into the polisher to break up the clogged resin. The air system is isolated only by a check valve (Tag. No. R-I-50) while performing the operation, and water can leak into the service and instrument air system through the check valve. Water entering the instrument air system can restrict air flow to the control valves for the polisher outlet valves causing them to shut. This is the most probable cause of the outlet valve closure and loss of feedwater.

A few seconds before 4:00:37 a.m. on March 28, 1979, the condensate polisher outlet valves shut causing the condensate booster pumps to trip due to low suction pressure which caused the main feedwater pumps to trip from low suction pressure at 4:00:37. (TMI-2 Emergency Procedure 2202.2-2 "Loss of Main Feedwater to Both OTSG's, Section 2.B," states in <u>Manual Action</u> 1: "If loss of FW is due to loss of both feed pumps: a. Trip the Reactor." The operator did not manually trip the reactor, although it tripped on high pressure 8 seconds later.)

The polisher outlet valves had been accidentally shut from water entering the air system on at least one other occasion. (See Technical Assessment Task Force report on "Condensate Polishing System.")

C. INITIAL PLANT RESPONSE TO THE ACCIDENT

04:00:37- (00:00:00)	Both main feed pumps trip, automatically tripping the main turbine. Three emergency feed pumps start automatically. With the reactor still operating, the primary coolant began to heat up because the turbine was no longer extracting heat from the system.
04:00:40- (00:00:03)	Reactor coolant system pressure increases to 2,255 psig, (Figure 1) opening the PORV as designed. The pressure in the reactor coolant drain tank began to increase (Figure 2).
04:00:45- (00:00:08)	Pressure reached 2,355 psig and the reactor tripped on a high pressure signal as designed (Figure 1). After the reactor tripped the plant began to cool down due to heat rejection through the steam generator relief valves, which had lifted, and the turbine bypass valve, decreasing the plant pressure (Figures 3, 4, and 7).
04:00:49- (00:00:12)	Reactor coolant system pressure was reduced to 2,205 psig, where the PORV should have closed (the PORV did not close) (Figure 2). The expected insurge of reactor coolant into the pressurizer peaked at about 260 inches and began to decrease. The operators, as specified in the operating procedures, stopped let-down flow and started another make-up pump (1A) to compensate for the expected pressurizer out-surge as the plant continued to cool down.
04:01:07- (00:00:30)	PORV and one pressurizer safety valve high outlet temperature alarm were received (temperatures were 239.5 °F and 203.5°F respectively). The operators were aware that the PORV had lifted but thought the valve had closed because the valve position indicator light was extinguished. This indicator only indicated that power was applied to the pilot solenoid for the PORV and did not indicate valve stem position. The high temperature was assumed to be a result of the temporary opening of the PORV and an existing leak in either the PORV or a code safety valve. Although the TMI-2 procedures indicate the PORV will open on a severe transient (Abnormal Procedure 2203-2.2 Turbine Trip, Section 2.0 automatic action A.3 states "Pressurizer Pilot Operated Relief Valve Open") none of the operating procedures required the operators to make positive checks to ensure the PORV had closed after an increasing pressure transient that approached or exceeded the PORV set point.
04:01:07- (00:00:30) through 04:01:10- (00:00:33)	Both steam generators reached the water level control set point of 30 inches, (Figures 3 and 4), where the emergency feedwater control valves TEF-V-11 A and 11 B opened. No water was added to the steam generators because the downstream block valves EF-V-12A and 12B were closed. The operators were not aware that the block valves were closed. The question of when the

EF-V-12A and 12B valves were closed is addressed in a separate paper included in the staff technical report.

04:01:25- (00:00:48)	With two make-up pumps (1A and 1B) running the rate of pressurizer level decrease was reduced and after reaching a minimum of about 160 inches it began to increase (Figure 1).
04:01:37- (00:01:00)	A second pressurizer safety valve high outlet temperature alarm was received. The indicated temperature was 294.5°F. The safety valve outlet temperature increase was probably due to the hot reactor coolant being discharged through the PORV, which increased the temperature of the safety valve outlet piping.
04:02:22- (00:01:45) approx.	Both steam generators boiled dry and effective heat transfer from the reactor coolant system to the secondary system stopped (Figures 3, 4, and 5).
04:02:38- (00:02:01)	The open PORV continued to reduce reactor coolant pressure to 1,640 psig where engineered safeguards features (ESF) for high pressure injection (HPI) activated. ESF actuation automatically stopped make-up pump 1B, started makeup pump 1C (make-up pump 1A was started previously at 4:00:49)and fully opened the make-up valves providing a total injection flow rate of about 1,000 gpm to the reactor coolant system.
04:03:50- (00:03:13)	The HPI portion of the engineered safety features was bypassed. (TMI-2 Emergency Procedure 2202-1.3 "Loss of Reactor Coolant/Pressure" <u>requires</u> the operator to "bypass" the engineered safety features and throttle the valves to <u>prevent pump runout.)</u> Note: "bypass" only returns HPI to manual control -- this action does not change any valve settings (see action at 4:05:15).
04:03:50- (00:03:13)	Reactor coolant drain tank relief lifts at about 122 psig (Figure 3).
04:04:03- (00:03:26)	Reactor coolant drain tank high temperature alarm occurred.
04:04:05- (00:03:28)	Pressurizer high coolant alarm occurred (260 inches) (Figure 1). Note: Over 100 alarms occurred during the first few minutes of the accident.

> Note: TMI-2 operating procedure 2103-1.3 "Pressurizer Operation" requires the operator to maintain level between 45 and 385 inches, see page 7, para. 2.2.7. In addition, page 5 para. 2.1.8., states, "the pressurizer/RC System must not be filled with coolant to solid conditions (400 inches) at any time except as required for system hydrostatic tests."

04:05:15- (00:04:38) Operator stopped make-up pump 1C and throttled the high pressure injection valves. (Operators had previously bypassed HPI at 4:03:50.)

> Note: TMI-2 Alarm Procedure 2201-13 Alarm 13.A2 (engineered safeguards features actuation) states that the cause for the ESFA alarm actuation (other than test or channel failure) is "LOCA" and requires followup action with TMI-2 Emergency Procedure 2202-1.3 "Loss of Reactor Coolant/Reactor System Pressure."
>
> Prior to automatic initiation of HPI at about 4:02, the plant response to the turbine trip appeared to be normal and the operators probably had no reason to suspect a reactor coolant system leak.
>
> After the automatic initiation of HPI, the rapidly decreasing reactor coolant pressure with constant reactor coolant system temperature was an unambiguous symptom that coolant was leaking from the system.
>
> Subsequent to overriding and reducing the HPI flow the operators increased let-down flow to its maximum value (about 160 gpm) in response to high pressurizer level, which further exacerbated the loss of coolant.
>
> During the period HPI was activated, from about 2 minutes to about 4 minutes, the reactimeter traces indicate there was no net heat up of the reactor coolant system (Figure 5), indicating that the plant had achieved a heat rejection rate equal to the decay heat and reactor coolant pump heat input.

04:06:07- (00:05:30) The indicated reactor coolant system hot leg temperatur and pressure reached saturation conditions of 582°F and 1,340 psig (Figures 1 and 5). As steam bubbles formed in the reactor coolant system, they took control of the plant pressure increasing the pressurizer level as the bubble expanded.

04:06:28- (00:05:51) Pressurizer level indication was off-scale high (greater than 400 inches)(Figure 1).

04:08:06- (00:07:29) Reactor building sump pump 2A started automatically at a water level of 38 inches.

04:08:37- (00:08:00) The operators discovered that the emergency feedwater block valves EF-V-12A and 12B were shut, and began opening the valves. Addition of the cold feedwater to the steam generators sub-cooled the reactor coolant system over the next 15 minutes and the system pressure followed saturation temperature (Figures 1 and 5). Since the pressurizer could not regain control of plant pressure, due to flow out of the open PORV, it appears that the 8-minute delay in providing feedwater to the steam generators did not materially affect the outcome of the accident.

04:10:56- (00:10:19) Reactor building sump pump 2B started automatically at a water level of 53 inches.

04:11:25- (00:10:48) Reactor building sump high-level alarm occurred. This alarm is one of the symptoms of a loss of coolant shown in TMI-2 Procedure 2202-1.3 "Loss of Reactor Coolant/Reactor System Pressure."

04:15:25- (00:14:48) The reactor coolant drain tank (RCDT) rupture disc failed as designed when pressure increased to about 191 psi (Figure 2). The reactor building ambient temperature began to increase rapidly as a result of released steam (Figure 6).

04:15:27- (00:14:50) Reactor coolant pump alarms occurred. Reactor coolant system pressure was about 1,275 psig and the temperature was about 570°F at this time (Figures 1 and 5). These conditions were very close to the lower limit for operating the reactor coolant pumps. The pumps were apparently vibrating due to the voids being formed in the reactor coolant.

04:20:37- (00:20:00) The out-of-core neutron instrument flux levels on the source range began to increase, (Figure 7). The reactor coolant system contained significant steam voids at this time, and the source-range nuclear instruments located outside the reactor vessel and measuring the radiation levels, as attenuated by any water in the reactor, were responding to this decrease in density. Electric Power Research Institute (EPRI) report NSAC-1, dated July 1979, Appendix CI pages 8-15, provides an analysis of the out-of-core neutron detectors and their response. The operators depressed the reactor manual trip pushbutton at 4:22:54 as a precautionary action.

04:23:21- (00:22:44) The steam generator A water level reached 30 inches and could have been used for heat transfer. However, the turbine bypass valve control was set to automatically control the steam generators pressure at a value about equal to saturation pressure of the primary coolant. Therefore, the valve was not effectively used to remove heat from the reactor coolant system.

04:25:35- (00:24:58) The operators requested PORV outlet temperature. The PORV outlet temperature was 285.4°F.

04:27:03- (00:26:26) Plant status information requested by the operator was printed out by the utility typewriter:

Reactor coolant loop A hot leg temperature	51.9°F
Reactor coolant loop B hot leg temperature	550.9°F
Reactor coolant loop A cold leg temperature	541.1°F
Reactor coolant loop A cold leg temperature	547.0°F
Reactor coolant loop B cold leg temperature	547.0°F
Reactor coolant loop B cold leg temperature	546.8°F
Reactor coolant loop A pressure	1,040 psig
Reactor coolant loop B pressure	1,043 psig

04:30:00- (00:29:23) Reactor building temperature and pressure were increasing rapidly. The operators responded by starting the reactor building emergency cooling booster pumps and switching all five reactor building cooling fans to high speed. The rate of pressure increase in the reactor building slowed down as a result of these actions (Figure 6).

04:33:13- (00:32:36) In-core thermocouple 10 rems read greater than 700°F, which is the highest reading the computer software was programmed to record. The significance of this reading is still not understood since the core was covered and being cooled at this time.

04:38:47- (00:38:10) to 04:38:48 (00:38:11) Both reactor building sump pumps were stopped. Based on the run time and the pumping capacity of these pumps, they could have transferred as much as 8,100 gallons of water out of the reactor building. The pumps were apparently aligned to discharge to the auxiliary building sump tank (which had a failed rupture disc) instead of the miscellaneous waste hold-up tank (the level of this tank did not change during the March 28, 1979, operations). The sump pump, by procedural guidance, could be aligned to either tank.

04:40:37- (00:40:00) through 05:00:37 (01:00:00) The source range out-of-core nuclear instruments continued to show an increasing count rate (Figure 7) due to the continuing decrease in reactor coolant density.

Increases in the reactor building background radiation level were shown on the reactor intermediate closed cooling system let-down monitor (1C-R-1092).

Reactor coolant flow had decreased from a normal rate of about 69 million pounds per hour to less than 50 million pounds per hour (Figure 8).

05:14:00- (01:13:23) (approx) Reactor coolant pumps 1B and 2B were stopped because of the vibration readings and because the plant conditions (temperature and pressure) were outside the specified range for pump operation.

05:14:00- (01:15:23) to 05:41:00 (01:40:23)	The out-of-core nuclear instruments, both source and intermediate range, increased their readings as the to reactor coolant density continued to decrease (Figure 7). The PORV discharge line temperature remained at about 283°F. Reactor coolant flow continued to decrease (Figure 8). There were some momentary indications of steam flow from steam generator A. The feed flow rate to steam generator B was increased. Steam generator A boiled dry and a few minutes later feedwater flow to steam generator A was increased (Figures 3 and 4) and was apparently used effectively for a few minutes to remove heat from reactor coolant loop A (Figure 5).
05:15:00- (01:44:23) through 05:18:00 (01:17:25)	Intermediate closed cooling system radiation monitor (1C-R-1092) began increasing from 3,500 counts per minute. The monitor reached its alarm point of 5,000 counts per minute at 5:18 a.m.
05:18:00- (01:17:23)	The reactor building air particulate monitor HP-R-227(P) reached its alarm point of 50,000 counts per minute. Due to the fact that the reactor coolant system had been below the required pressure conditions for fuel rod compression for some time and the core temperature was increasing above normal (at least one core exit thermocouple was reading off-scale high at 4:33:03) it is inferred that these radiation monitor readings indicate that fuel cladding was being ruptured mechanically by internal pressure. The ruptures at this time were probably small but did allow some of the fission gases accumulated in the fuel-to-cladding gap to escape into the reactor coolant system.
05:41:22- (01:40:45) (approx)	Reactor coolant pumps IA and 2A were stopped (Figure 8). Forced cooling of the core was terminated. The source and intermediate range nuclear instruments decreased significantly as the cooler water being held up in coolant loop A hot leg fell back into the core, temporarily increasing the coolant density in the core. Reactor coolant loop A hot and cold leg temperature both decreased for about 12 minutes. Then the hot leg temperatures indicated in the control room began to rise rapidly going off-scale high (greater than 620°F) within 38 minutes. The loop A cold leg temperature continued to decrease slowly over the next hour (Figure 5). Reactor coolant loop B hot leg temperature continued to decrease until about 6:05 a.m., at which time it began to increase rapidly and the indicated temperature in the control room went off-scale high (greater than 620°F) within about 25

minutes. The loop B cold leg temperature continued to decrease (Figure 5).

Figure 9 data, which were not available to the operators in the control room, indicates that superheated conditions existed in both the A and B loop hot legs from about 6:15 a.m. to about 2:30 p.m.

About 2 minutes after the reactor coolant pumps stopped, the out-of-core nuclear instruments, both source and intermediate range, began to increase rapidly, indicating boil off of the reactor vessel inventory.

Reactor system pressure at the time the A loop reactor coolant pumps were stopped was about 1,000 psi and it continued to decrease rapidly (Figure 1).

06:02:00- (02:01:23) (approx)

Analysis of a reactor coolant sample showed the gross beta-gamma activity to be 4 micro ciroes per milliliter, which is about 10 times the normal expected reading. This sample is a further indication that some mechanical damage to the fuel cladding had been sustained and fission products had been released into the reactor coolant from the fuel to cladding gap space.

06:15:00- (02:14:23) (approx)

The self-powered neutron detectors (SPNDs) installed in the core began responding to high temperatures (see EPRI report NSAC-l July 1979, Appendix CI pages 19-22, for a description and temperature response of these devices) indicating that the water level in the reactor vessel was below the top of the active core.

The response of the SPNDs is consistent with the sharp rise in the reactor coolant loop hot leg temperatures which started at about 6:15 a.m. (Figures 5 and 9).

06:22:37- (02:22:00)

The PORV block valve was shut. Reactor system pressure began to increase (Figure 1) and the reactor building pressure began to decrease (Figure 6) indicating that the PORV was the source of coolant leakage from the system.

06:24:00- (02:23:23)

The reactor building air particulate sample monitor (H0-R-227 (P)) reached its alarm point of 50,000 counts per minute for the second time.

06:26:00- (02:25:23) (approx)

The area monitor in the reactor building on the 347 ft. level reached its alarm point of 50 mrem/hr.

06:30:00-
(02:24:23)
to
07:00:00-
(03:59:23)

General radiation levels in the auxiliary building increased and ranged from about 10 mrem/hr to more than 5 rem/hr at the purification valve room door.

06:43:00-
(02:42:23)

Analysis of a reactor coolant sample taken at this time showed gross betagamma activity of 140 microcuries per milliliter. The area monitor for TMI-1 sample room (RM-G3), which contained in TMI-2 sample lines, reached its alarm set point at 2.5 mrem/hr.

06:48:00-
02:47:23)

TMI-1 hot machine shop area monitor RM-G4 reached the reactor alarm set point of 2.5 mrem/hr. A survey of coolant sample line running through this area read 1.5 rem/hr.

Note: For further details of various radiation alarms and information see the NRC Radiological Sequence of events starting on page II-A-I of NUREG 0600.

Summary: The reactor vessel inventory continued to boil off after the reactor coolant pumps were stopped at 5:41 a.m., and by 6:15 a.m. there was evidence of super heated steam in the coolant loop hot legs. Subsequent to closing the PORV block valve at about 6:22 a.m. no significant heat was removed from the core until the block valve was again opened at 7:12 a.m.

By 7:00 a.m. the temperature of the hot leg was at least 750°F in the B loop. The A loop temperature was about 775°F.

No significant change in make-up flow to the loops is evident until almsot 7:20 a.m., one hour after the PORV was closed.

Response of the SPNDs at 6:48 a.m. indicates that the reactor vessel water level may have been 8 to 9 feet below the top of the active core. Radiation monitoring instruments and reactor coolant sample analyses had previously indicated that some mechanical damage (ruptured cladding from internal pressure) to the core was occurring about 6:02 a.m.

By 6:50 a.m., the high radiation levels indicated by the radiation monitoring instruments indicate that severe core damage was taking place.

It is unclear why the operators or engineers and supervisors who were present did not immediately start high pressure injection when the PORV block valve was closed and plant pressure began to increase indicating that the PORV was the source of the leak.

06:48:23-
(02:47:50)
to
06:55:37-
(02:55:00)

The operators managed to get the condensate system to function automatically by 6:50 a.m. (see section B of this report). The difficulty with this system was found to be a broken air line which supplies operating air to the valve air operator. The valve was then opened manually and the system began to control the condenser hot well level automatically. The operating air line was apparently broken during the transient since the hot well level control was functioning normally prior to the turbine trip.

After jumpering interlocks in the control circuits reactor coolant pump 2B was started at 6:54:46 and allowed to run for 19 minutes.

The reactor out-of-core nuclear instruments showed a sharp decrease in level as the colder water trapped in the reactor coolant loop cold legs and the B steam generator was transferred into the reactor vessel.

Radiation level increases and alarms in several areas of the plant including reactor building atmospheric sample monitors and the hot machine shop area radiation monitor led the shift supervisor and the TMI-2 Superintendent, technical support to decide to declare a site emergency at approximately 6:56 a.m.

D. DECLARATION OF EMERGENCY AND STABILIZATION OF THE PLANT

06:56:00- (02:55:23) (approx)	A site emergency was declared.
06:56:00- (02:55:23) to 07:00:00-	Radiation levels continued to increase in the reactor building, the auxiliary building and in the fuel handling building.
07:05:00- (03:04:23)	The TMI station superintendent arrived in the TMI-2 control room and assumed the role of emergency director.
07:05:00- (03:04:25) to 07:24:00 (03:23:23)	Radiation levels throughout the plant continued to increase. The PORV block valve was opened at 7:13 a.m. and closed at 7:17 a.m. High pressure injection was manually initiated at 7:20 a.m.
07:24:00- (03:23:23)	A general emergency was declared by the TMI station superintendent. The radiation monitor in the dome of the reactor building had reached a reading of 8R/hr, which is a specified condition in the TMI-2 emergency plan that requires declaration of a general emergency.
07:24:00- (03:23:23) to 07:30:00- (15:49:23)	The operation of plant systems and components during this period are summarized as follows:

A. From 7:12 to 11:08 a.m., a combination of high pressure injection flow into the loop and flow out of the PORV was the principal means of cooling the core. Based on the out-of-core nuclear instrument readings, the reactor vessel inventory appears to have been recovered to a level above the active core by about 11:00 (Figure 7).

B. Starting at about 11:40 a.m., a prolonged depressurization of the reactor coolant system began, with a relatively low high pressure injection flow, which may have resulted in some core uncovery as indicated by the out-of-core nuclear instruments (Figure 7).

C. Even though substantial quantities of steam were discharged into the reactor building through the open PORV until it was isolated at 6:22 a.m., the operators precluded a reactor building pressure rise to 4 psig (the actuation set point for engineered safeguards features actuation for reactor building isolation) by manipulation of the reactor building ventilation system. However, a prolonged discharge out of the PORV which started at 7:40 a.m.caused the first reactor building isolation to occur at 7:56 a.m.

D. A pressure spike of at least 28 psig occurred in the reactor building at about 1:50 p.m. The pressure spike was apparently the result of a hydrogen burn caused by flammable concentrations of hydrogen which were generated by the zirconium/water reactions that took place during the times the core was uncovered and overheated (see Technical Assessment Task Force report on "Chemistry").

E. Continued operation of the let-down system and other systems such as the waste gas decay system after the core was damaged contributed significantly to the escape of radiation to the environment. A leak in the waste gas header system was an important factor in this (see Technical Assessment Task Force report on "Containment: Transport of Radioactivity from the TMI-2 Core to the Environs").

F. Plant repressurization was started at 3:08 p.m. and at 7:50 p.m. forced circulation was reestablished when reactor coolant pump IA was started and run continuously placing the reactor in a stable cooling mode.

APPENDIX A

DECAY HEAT REMOVAL METHODS -- TMI-2

BACKGROUND

Subsequent to a reactor shutdown, decay heat must be extracted from the reactor core to preclude overheating. While the decay heat, due to the energy produced by radioactive decay of fission byproducts, after shutdown is only a fraction of the heat produced by the core at full power, this heat source of 168 megawatts at shutdown drops to 13 megawatts in one day and to 0.14 megawatt in one year (LA8041MS). Consequently, this heat must be removed from the core and rejected out of the system or the core will overheat and be damaged. Three basic rules must be followed in order to protect the reactor core in the event of an upset of normal operating conditions. These rules are (1) stop the nuclear reaction; (2) keep water over the core; and (3) remove the heat generated by the core. Stopping the nuclear reaction occurs by actuation of a separate system and is not included in the discussions below. Keeping the core covered with water and removal of the decay heat are discussed in the succeeding paragraphs.

NORMAL DECAY HEAT REMOVAL METHODS

When the reactor is shut down, after power operation, by a normal planned shutdown or an event that causes a reactor trip, the decay heat generated in the core is transferred to the reactor coolant and transported to the steam generators as the reactor coolant is circulated by the reactor coolant pumps.

The heat in the reactor coolant is given up to the water in the secondary side of the steam generator which becomes heated and produces steam.

The heat from the secondary water is removed from the steam generator in the form of steam and transported to the condenser through a flow path controlled and regulated by the turbine bypass valve.

The steam is condensed to water in the condenser, with the heat in the steam being given up to the condenser cooling water system which rejects its heat to the environment.

The condensed water or condensate in the condenser is returned to the steam generator through the condensate and feedwater system where the cycle starts over.

The rate of cooldown of the reactor coolant or the rate of decay heat removal is dependent on the rate of steam flow out of the steam generators.

The emergency feedwater sytem is a backup to the main feedwater system that can use the flow path described for the normal condensate and feedwater sytem or it can provide feedwater separately from the condensate storage tanks.

In the event that the condenser is not available, steam can be dumped directly to the atmosphere through atmospheric dump valves tied into the main steam lines.

Water level in the steam generators is controlled automatically, under normal decay heat removal conditions, as is the rate of heat rejection (steam flow) from the steam generators. The automatic control system will normally reject only the amount of decay heat being generated, thus keeping the reactor coolant very close to its normal operating temperature. This designed mode of operation facilitiates returning the plant to service after a reactor shutdown.

Should the plant operators desire to cool the plant to a lower temperature, they can adjust the controls to maintain a constant cool down rate to the desired temperature.

The reactor coolant system is intact and pressurized during normal decay heat removal and water is added to the loop by the makeup pumps as required during plant cooldown.

When the reactor coolant temperature and pressure has been reduced to a level consistent with the operational capability of the low pressure injection system (about 200°F and 350 psig) it is used in its companion role of a decay heat removal system.

BACKUP DECAY HEAT REMOVAL METHODS

The major backup system for decay heat removal is the emergency core cooling system, comprising: (1) a water supply (the borated water storage tank which holds 472,000 gallons), (2) the high pressure injection system composed of pumps and appropriate piping which can inject water into the system up to a rate of 1,000 gpm at 1,600 psig, (3) the low pressure injection system composed of high flow rate (3,000 gpm) pumps and appropriate piping, (4) the core flood tanks which can discharge directly to the reactor vessel if a sufficiently low pressure is reached, and (5) the reactor building ventilation system to reject heat dumped into the building to the environment.

The emergency core cooling system is designed to function automatically if plant pressure decreases to a preset value of about 1,600 psig. At this predetermined pressure the high pressure injection pumps start automatically, appropriate valves open, and water is injected into the reactor coolant system. The low pressure injection pumps also start upon receiving the low pressure signal, but do not add water to the system at this time. The core flood tanks do not actuate unless system pressure decreases to a value lower than the nitrogen pressure in the tanks.

Should plant pressure continue to decrease to the appropriate value, the low pressure injection system and the core flood system will be used to inject water into the reactor coolant system to keep the core covered.

Heat is removed from the core, during periods when the various components of the emergency core cooling system are designed to function, by water flowing into the reactor vessel absorbing heat from the core and flowing out a hole in the reactor coolant system as water or steam.

The design of the emergency core cooling system is based on the ability to keep the core covered under serious accidents postulated to be caused by ruptures that might occur in the reactor coolant system pressure boundary. The system design is analyzed to show that it is adequate to pgrform its function for a wide range of break sizes from about 0.04 ft up to and including a 14.1 ft split in the reactor coolant system hot leg.

The emergency core cooling system's design and operation presumes that a hole in the reactor coolant system pressure boundary is available as a place to reject the heat.

An additional assumption concerning the analysis of the emergency core cooling system design is that there is no middle ground between availability and use of the normal decay heat removal systems and the need for emergency core cooling. However, a leak in the coolant system pressure boundary, which can be compensated for by running the normal makeup pump, but which is too small to remove all the decay heat generated by the core, requires that the normal steam generator decay heat removal path be functional; otherwise, the decay heat energy of the core will heat up the reactor coolant, expanding it and increasing the system pressure. This process will continue until the PORV and/or one or both of the code safety valves on the pressurizer opens and makes a "hole" in the system to remove decay heat. The high pressure injection system under these conditions should be operated in a manner that maintains the reactor coolant inventory constant (i.e., as much water must be charged into the system as is rejected out through the hole and/or relief valve). This postulated leak is not analyzed in the TMI-2 Final Safety Analysis Report (FSAR).

Various parts of the emergency core cooling system are designed for multiple use. Parts of the high pressure injection system are used as a normal make-up system, to supply water for purification system let-down flow, reactor coolant pump seal cooling, and to make up losses due to small leaks. The low pressure injection system is used as a normal decay heat removal system.

When the normal decay heat removal path through the steam generators is lost and there is no rupture of the reactor coolant pressure boundary, the high pressure injection system used in this manner requires the operators to create a hole in the reactor coolant system pressure boundary by opening the pilot-operated relief valve (PORV) and charging into the reactor coolant system an amount of water equal to that rejected through the relief valve. This event is not analyzed in the TMI-2 FSAR because total loss of feedwater is not considered to be a credible accident.

DECAY HEAT REMOVAL DURING THE TMI-2 ACCIDENT

During the period of time between trip of the main turbine and the reactor trip at 8 seconds into the accident some of the heat generated by the reactor operating at power was removed by (1) the turbine bypass line discharging to the condenser (2) the main steam safety valves in the secondary loop opening and discharging steam to the atmosphere and (3) the PORV on the pressurizer which opened at about 3 seconds into the transient.

After the reactor tripped 8 seconds into the transient, the steam generators began to cool down the reactor coolant and the main steam relief valves closed. The turbine bypass system continued to remove heat until the steam generators boiled dry at about one minute 45 seconds into the accident. The steam generators boiled dry because the emergency feedwater block valves were left shut improperly. The open PORV continued to reject heat and mass from the reactor coolant system until flow through it was stopped after about 2 hours and 22 minutes by closing the block valve down stream of the relief valve.

From one minute 45 seconds to 2 minutes, the discharge of the PORV, together with modest make-up flow, probably on the order of 300 gpm, was the heat removal method and was clearly not sufficient as the plant continued to heat up (Figure 5).

High pressure injection was initiated automatically at about 2 minutes, delivering on the order of 1,000 gallons per minute flow to the reactor coolant system. This high injection flow rate in conjunction with the continuing flow out of the open relief valve removed an amount of heat equal to the decay heat at that time. Figure 5 shows that the reactor coolant temperature rise stopped after high pressure injection commenced. It then leveled off and was essentially constant when high pressure injection was terminated at about 4 minutes and 38 seconds.

Terminating high pressure injection at 4 minutes and 38 seconds upset the equilibrium of this decay heat removal mode and the reactor coolant system started to heat up (Figure 5). The plant continued to heat up until feedwater was added to the steam generators (Figures 3 and 4), after opening the emergency feedwater block valves, at about 8 minutes. Between 8 minutes and 30 minutes, the reactor coolant system temperature was reduced from about 597°F to 550°F by use of the steam generators (Figure 5) and the open PORV. During this period the make-up flow rate was very low, probably less than 100 gpm. Let-down flow of about 160 gallons per minute was started at about 4 minutes 38 seconds (to reduce pressurizer level), further increasing the rate of coolant loss from the reactor coolant system.

The reactor coolant system pressure continued to decrease due to flow out the open relief valve while the loop was being subcooled and reached 1,100 psig at 18 minutes. When feedwater was rapidly added to the steam generators starting at about 18 minutes pressure was reduced to about 1,050 psig, but returned to 1,100 psig with a few minutes as the added feedwater heated up.

A fairly constant heat balance was maintained from about 18 minutes until the B loop reactor coolant pumps were stopped. During this period the A and B steam generators, together with the boil off through the open relief valve, were removing essentially all the decay heat generated by the core and the heat input of the reactor coolant pumps (Figure 5).

The B loop reactor coolant pumps were stopped at one hour and 14 minutes and both the A & B loop temperatures began to increase (Figure 5), indicating that the heat balance was upset by the sharply reduced forced circulation in the reactor coolant system.

The atmospheric dump valves were opened on the secondary side of the a steam generator at about one hour and 31 minutes as evidenced by sharply decreasing steam generator pressure. The combination of increased feed flow and possible opening of the B loop atmospheric dump valves at one hour and 14 minutes appears to account for the rapidly decreasing B steam generator pressure (Figures 3 and 4). It is concluded that there was flow in both loops until about one hour and 31 minutes when the B steam generator appears to have been isolated and the A steam generator boiled dry. The conclusion is based on the fact that the average temperatures and the differential temperature across the steam generators were essentially equal in both loops. For average loop temperatures and delta temperatures across the steam generators to be equal each generator must have been removing equal amounts of heat.

The temperature in both loops had been reduced to about 530°F just prior to stopping the A loop reactor coolant pumps at one hour and 41 minutes. The reactor coolant system pressure followed the decrease in saturation temperature indicating that this heat removal mode was capable of subcooling the system (Figures 1 and 5).

When the A loop reactor coolant pumps were stopped, followed closely by a rapid feeding of the A steam generator at about one hour and 42 minutes, the A and B loop hot leg temperatures began to diverge.

The B loop hot leg temperatures began to increase and then stabilized for a few minutes. The A loop hot leg temperature decreased until one hour and 52 minutes at which time it began to increase rapidly from 530°F and was greater than 800° by about 5 hours. The B loop temperature began to decrease at about one hour and 55 minutes and reached 620°F at 2 hours and 22 minutes. It then began to increase rapidly and reached about 790°F by about 3 hours and 15 minutes. (Figures 5 and 9).

The extremely rapid heat up of the reactor coolant loop hot legs after the last reactor coolant pumps were stopped indicates that any fluid that had been in the loops collapsed leaving only steam in the hot legs which achieved superheated conditions within a few minutes. In order for superheated steam to be present, a heat source with a temperature greater than saturation temperature had to be available to heat the steam, thus it is concluded that a portion of the active core was exposed (uncovered) between one hour and 55 minutes and 2 hours.

After flow in the loops was stopped at one hour and 40 minutes, boiling off the water inventory in the reactor vessel was apparently an effective heat-removal method until loss of mass began to expose (uncover) the active core. Steam flow, with its low heat transfer coefficient, apparently was inadequate to remove the heat generated in the exposed fuel since the fuel began a rapid heat up during this period. This mode continued until the PORV was shut at 2 hours and 22 minutes.

While it is not possible to show the precise water level in the core, from one hour and 30 minutes to 2 hours and 22 minutes, the level can be inferred by use of data from the out-of-core neutron detectors and the response of the self powered neutron detectors (SPNDs) located at various elevations in the core as indicators of what parts of the core were covered at various times. These instruments indicate that the water level at 2 hours and 22 minutes could have been as low as 8 to 9 feet below the top of the active core. The Electric Power Research Institute report NSAC-1, dated July 1979, Appendix Ci, pages 16-22, provides a description and an interpretation of indications from these instruments.

No effective cooling occurred between 2 hours and 22 minutes (block valve shut) and 3 hours and 19 minutes when high pressure injection was started and maintained for 18 minutes. This was a period that produced large quantities of hydrogen from the Zirconium-water reaction at high temperature and significant damage to the core (see the Technical Assessment Task Force reports on "Chemistry" and "Core Damage").

Sustained high pressure injection flow was started at 4 hours and 26 minutes and maintained until about 9 hours and 4 minutes. The core appears to have been recovered by about 6 hours and 30 minutes as indicated by the out-of-core nuclear instrument readings. Heat removal during this period was by (1) let-down flow, which is believed to have been close to maximum level of 160 gpm, (2) periodic opening of the PORV (open 5 minutes starting at 3 hours and 12 minutes, open 98 minutes starting at 3 hours and 40 minutes, periodic cycling (about 30 cycles between 5 hours and 43 minutes and 7 hours and 38 minutes), (3) and some steam refluxing to the steam generators.

Reactor coolant pump 2B was restarted at 2 hours and 54 minutes and some of the water trapped in the steam generator B and loop B cold legs was returned to the reactor vessel. See Electric Power Research Institute Report NSAC-1, dated July 1979, Appendix Th, pages 60-63, for an explanation of the reaction of the loops to the pump's start.

Temperature measurements from the core exit thermocouples readings obtained between 4 hours and 5 hours and 30 minutes show at least 9 temperatures above 2000°F with the highest being 2,580°F and several above 1,000°F. The Electric Power Research Institute report NSAC-1, dated July 1979, Appendix CI, pages 16-17, provides an interpretation of this data; the staff agrees with EPRI's interpretation. Their Appendix CI, figure CI-10, is a core map of the observed readings.

During this period high pressure injection flow into the reactor vessel and out through the loop A hot leg then via the surge line out the pressurizer when the PORV is open is the cooling flow path.

The continuous depressurization of the reactor coolant system, in an attempt to dump the core flood tanks into the system, which was started at about 7 hours and 38 minutes, may have uncovered the core again as evidenced by the increasing out-of-core neutron instruments. The high pressure injection flow during this period was very low and cooling was accomplished by boil off of the core inventory.

Since the conditions of this depressurization and possible uncovery are similar to the earlier uncovery,,it must be assumed that conditions for zirconium-water reactions also existed during this period. The continuous depressurization allowed the hydrogen generated by the zirconium-water reaction to be expelled into the reactor building. This new hydrogen combined with any hydrogen generated and expelled during the earlier uncovery reached a level sufficient to support combustion. A 28-lb. pressure spike shown on the reactor building pressure instrument at about 9 hours and 50 minutes probably indicates a burn of the hydrogen in the reactor building atmosphere (Figure 7). (See Technical Assessment Task Force report on "Chemistry.")

The depressurization attempt was terminated at 11 hours and 8 minutes with the closure of the PORV block valve. For the next 3 hours and 35 minutes there was very little heat removed from the system. The PORV block valve was opened for two periods of about 10 minutes each. There was no forced circulation flow, and high pressure injection flow was sporadic until about 13 hours and 23 minutes.

It can be inferred from the actions during this period that substantial continued core heat-up occurred at least until the system was finally filled and reactor coolant pump 1A was started and remained running at 15 hours and 50 minutes.

After the IA main coolant pump was started, a slow cooling trend was later established and the plant was placed in the natural (no pumps running) circulation mode of cooling on April 27, 1979.

APPENDIX B

SIGNIFICANT EQUIPMENT PROBLEMS

DISCUSSION

During the course of the accident there were several equipment problems that may have drawn the operators' attention away from those principle actions necessary to protect the reactor core. Some of the more important of these problems are discussed below:

1. Condensate System

The initiating event (condensate polisher outlet valves shutting) at a few seconds before 4:00:37 a.m. blocked the condensate discharge path from the condenser hot well to the section of the condensate booster pumps. (The condensate polisher is further discussed in the staff report on the subject.)

The polisher tank bypass valve does not open automatically in the TMI-2 plant and apparently could not be opened by the operators in the control room, probably due to high differential pressure across the valve. This valve was apparently opened manually by the auxiliary operators about 59 minutes into the event.

Hot well level control could not be maintained automatically due to broken air operating lines to the air operator for the condensate reject valve C0-C057. These air lines apparently broke at the start of the transient since the system was operating normally prior to turbine trip.

An additional problem was an excessive seal leakage problem that developed on condensate pump 2A which started after the turbine trip.

The actions required to reestablish condenser hot well level control lasted from about 4:05 a.m. to about 6:50 a.m. See Electric Power Research Institute report NSAC-1, dated July 1979, Appendix C/FDW, for a description of the system.

The following sequence of events illustrates the extensive efforts devoted to the condensate problem. Note that the TMI-2 shift supervisor left the control room at about 4:20 a.m. (a critical time) and spent about 45 minutes in the turbine building trying to fix the condensate system. (The TMI-1 shift supervisor was in charge in the TMI-2 control room during this time.)

Time	Sequence of events related to the condenstate system
04:00:30	Condensate pump 1A tripped.
04:00:37	Main Feedwater pumps 1A & 1B tripped.

04:00:50	Condenser hot well low-level alarm.
04:01:05	Condenser hot well low-level alarm cleared.
04:01:50	Condenser high-level alarm.
04:05:52	Condensate pump 1A started.
04:05:52	Condensate booster pump trip signal received three three times between 04:05:52 and 04:07:36 indicates operators were trying to reestablish flow in the condensate system.
04:09:35	Condensate pump 1A tripped.
04:09:50	Condensate booster pump suction header pressure low alarm.
04:14:04	Condensate booster low-suction pressure alarm cleared.
04:16:20	Condensate booster pump low-discharge pressure alarm.
04:16:49	Condensate booster pump low-suction pressure alarm.
04:20:00 approx.	TMI-2 shift supervisor leaves control room to go to turbine building.
04:59:00	Condensate booster pump low-suction pressure alarm cleared.
04:59:00	Condensate polisher bypass valve opened manually.
04:59':58	Condensate high temperature alarm.
05:00:00 approx.	TMI-2 shift supervisor returns to control room.
06:53:07	Condensate hot well high-level alarm cleared.
06:55:00 approx.	Condensate reject valve C0-v-57 opened manually.
06:56:00	Condenser hot well low-level alarm.
07:01:33	Condenser hot well level was off-scale low.
07:01:48	Condenser storage tank low level alarm.
07:03:04	The condensate storage tank low-level alarm cleared.
07:11:47	Condenser hot well low-level alarm cleared.

2. Pressurizer Level

The TMI-2 operating procedure 2101-1.1 "Nuclear Plant Limits and Precautions" requires that the pressurizer level not be taken solid, defined as greater than 400 inches indicated level, except for hydrostatic tests. The Technical Specification requires that pressurizer level be maintained between 45 inches and 385 inches for normal operation and not allowed to exceed 385 inches any time the plant is in operation mode 1, 2 or 3. Consequently the operators were extremely concerned with pressurizer level during the course of the accident.

The following sequence of events indicates the changes in pressurizer level observed by the operators and their responses to these indicates.

Time	Sequence of events related to pressurizer level
04:01:25	Pressurizer level reached minimum level of 158 inches.
04:04:05	High pressurizer level alarm. Set point is 260 inches. The operators responded by stopping HPI flow at 4:04:38 even though plant pressure continued to decrease.
04:05:37	Pressurizer level reached 377 inches, decreased momentarily, then continued to rise.
04:06:28	Pressurizer level off-scale high (greater than 400 inches). The operator responded by initiating maximum let-down flow, then reduced it at 4:07:35.
04:10:52	Pressurizer level came back on scale and remained between 350 inches and about 390 inches until about 7:33 a.m.
07:33:33	Pressurizer high-level alarm. The level was 271 inches but it went off-scale high within a few minutes.
08:00:00	Pressurizer level was 380 inches with a reactor coolant system pressure of 1,500 psig.
15:10:00	Pressurizer level begins a rapid decrease.
15:11:00	Pressurizer high-level alarm clears.
15:19:00	Pressurizer level low-level alarm. Operator responded by starting a make-up pump.
15:29:00	Pressurizer level beginning a steady increase.

15:44:00	Pressurizer low-level alarm clears at about 206 inches.
15:54:00	Pressurizer high-level alarm, level is 260 inches.
16:22:00	Pressurizer level reaches 400 inches.

3. Pressurizer Heaters

Throughout the sequence the operators experienced trouble with the pressurizer heaters tripping. This tripping could be attributed to grounding due to the moisture being injected into the reactor building during the course of the accident.

The following sequence indicates the magnitude and persistance of this problem.

Time	Sequence of events related to pressurizer heaters
04:00:45	Pressurizer heater groups 1 through 5 off. These heaters placed in automatic control by the operators at the start of the event and because pressure is high at this time the automatic control has the heaters turned off.
04:01:00	Pressurizer heater groups 1-5 are automatically energized.
06:54:56	Pressurizer heater groups 1-5 tripped.
08:24:00 (approx.)	Pressurizer heater groups 1-5 on. All heaters are operable at this time.
08:31:00	Pressurizer heater group 10 tripped.
08:46:00	Pressurizer heater groups 4 and 5 tripped.
09:30:00	Pressurizer heater group 3 trips and remains off throughout the remainder of the sequence.
10:14:16	Pressurizer heater groups 1 and 2 tripped.
10:14:43	Pressurizer heater groups 1 and 2 on.
11:44:21	Pressurizer heater groups 1 and 2 off, came on again in 2 seconds.
11:50:53	Pressurizer heater groups 1 and 2 tripped.
13:55:47	Pressurizer heater group 8 trips.
14:06:02	Pressurizer heater groups 1 and 2 on.

14:07:56	Pressurizer heater groups 1 and 2 off.
14:32:13	Pressurizer heater groups 1 and 2 on.
14:39:34	Pressurizer heater groups 1 and 2 off.
14:40:28	Pressurizer heater groups 1 and 2 on.
15:29:29	Pressurizer heater groups 1 and 2 off.
15:45:54	Pressurizer heater groups 1 and 2 on.
17:26:46	Pressurizer heater groups 1 and 2 off.
18:26:03	Pressurizer heater groups 1 and 2 on.

APPENDIX C

SUMMARY OF INCORRECT OPERATIONAL ACTIONS

The purpose of this section is to summarize and evaluate those actions taken by the plant which were not correct and either contributed to the onset of the accident or worsened the final outcome. It should be understood that this paper does not attempt to show causes or attempt to explain why these actions were taken. Control room operators, shift foremen, shift supervisors, duty officers, engineers and managers were all involved to various degrees in manipulations or decisionmaking pertinent to the accident. Also highlighted are inactions which affected the outcome. Not mentioned in this analysis are the actions and decisions which were correct.

With respect to errors committed before the accident transient three are important: permitting the emergency feedwater block valves to be shut during operations; not shutting the pilot-operated relief valve (PORV) block valve; and permitting water to enter the condensate polisher control air system.

The emergency feedwater block valves, EF-V-12A and 12B were almost certainly closed at the commencement of the transient and were most probably not reopened following a surveillance test on March 26, 1978 (although this is not proven and there are other possible courses). In any event shift reliefs were not conducted in such a fashion as to ensure the relieving control room operator was appraised of plant status as required by Administrative Procedure 1012, "Shift Relief and Log Entries." Not noting that indicating lights on the front control panel showed the block valves to be out of position suggests a lack of rigor regarding reactor operators' attention to the conditions of the control panel indicators (see staff report on these valves for further details).

For a lengthy period before March 28, temperature detection in the discharge piping downstream of the PORV and the code safety valves indicated that one of these valves was leaking. The temperature of the PORV tailpiece was nearly 200°F; periodically, safety valve discharge header temperature alarms, which occur at 200°F, had been received. The PORV isolation valve, RC-V2, had not been shut as Emergency Procedure 2202-1.5, "Pressurizer System Failure," requires it to be when the relief valve discharge line temperature exceeds 130°F.

Immediately prior to commencement of the transient, operators were attempting to transfer spent resin from an isolated condensate polisher unit to the resin regeneration system. The resin was apparently clogged in the outlet of the polisher and the operators were injecting a water-and-air mixture into the polisher to break up the clogged resin. The air system is isolated only by a check valve during this operation and water can leak into the service and instrument air systems. Water entering the instrument air system can restrict air flow to the control valves for the polisher outlet valves, causing them to sense a requirement to shut the valves. Inadvertent isolation of the condensate polishers had occurred during other similar spent-resin transfers. Procedures for the clearing of resin from the system failed to take this into account. The operators could have been instructed to bypass and remove the polishers from service prior to attempting this evolution to avoid the problem.

Following isolation of the condensate polishing system on March 28 there followed other operational errors which will be discussed below.

Upon the loss of both feedwater pumps the immediate actions of Emergency Procedure 2202-2.2, "Loss of Steam Generator Feed," require the operator to manually trip the reactor regardless of whether an automatic trip took place or not. The operator did not immediately trip the reactor as required by the procedure.

Following a turbine trip, Abnormal Procedure 2202-2.2 states that the PORV will open, indicating that this is an expected phenomenon. The valve should be verified shut following actuation. Operators relied upon an inadquate panel indication that the valve had been commanded to close but failed to recognize symptoms which indicated that the valve was open. These symptoms included a rapidly decreasing reactor coolant system pressure to saturation, reactor coolant drain tank pressure increase, PORV discharge high-temperature alarm, pressurizer safety valve high-temperature alarm, reactor coolant drain tank high temperature alarm, reactor building radiation alarms, reactor building sump high-level alarm and reactor building temperature and pressure increase. There was sufficient information in the control room to know that the PORV was stuck open.

The operators failed to verify that the emergency feedwater pumps were not only running as automatically required by the transient but that they were also delivering feedwater to the steam generator as required by Abnormal Procedure 2203-2.2, "Turbine Trip," and Emergency Procedure 2202-2.2, "Loss of Steam Generator Feed." The lack of emergency feedwater flow resulted in the steam generator boiling dry.

Neither the reactor operators, senior reactor operators, the duty officer, unit superintendent or station manager recognized that a loss-of-coolant accident (LOCA) was in progress. Emergency Procedure 2202-1.3, "Loss of Reactor Coolant/Reactor Coolant System Pressure" lists 10 symptoms of a loss of reactor coolant, of these the following were present and recognized by the operators: decreasing reactor coolant system pressure, high radiation levels, high reactor building temperature, high reactor building sump level, high reactor building pressure. One important symptom, decreasing pressurizer level, was not present. This latter symptom confused the situation and they concluded incorrectly that they had a steam leak. Also confusing them was the fact that one of the 10 symptoms, high reactor building pressure, is listed as common to both a loss of coolant or a steam leak. It is not clear what the operators considered as the cause for rapidly decreasing reactor coolant system pressure. This could only be due to a loss of coolant as is recognized by the title of the emergency procedure.

The control room operators bypassed the high pressure injection portion of the safety features actuation system (SFAS) less than 5 minutes after the commencement of the accident. This was in violation of Operating Procedure 2105-1.3, "Safety Features Actuation System" which requires that SFAS be fully enabled except during maintenance or testing. However, it was consonant with the requirements of Emergency Procedure 2202-1.3, "Loss of Reactor Coolant," Section B, paragraph 3.4, which stipulates that the safety injection channels should be bypassed in order to protect (prevent pump runout) the high pressure injection pumps (HPI). Statements from the operators suggest that they were concerned about the abnormal pressurizer level increase rather than HPI pump limits and therefore they considered the provisions of Emergency Procedure 2202-1.3 were of lesser priority in guiding their actions.

Apparently all those present in the control room were concerned about the high pressurizer level indication. They were confused and did not understand the significance of this phenomenon. However, they realized that the system was not reacting as if it were solid. "We knew that we weren't solid."1/ The apparent principal concern was that a full pressurizer and a non-solid system indicated the presence of a bubble in the reactor coolant system other than in the pressurizer. "We were sitting there trying to figure out how the heck we were going to cool this thing down -- get that thing [pressurizer bubble] back -- without aggravating our problem."2/ The shift foreman (Schiemann), when interviewed immediately after the accident, indicated that when the pressurizer "was up to the top" he was not concerned principally about pressurizer level *per* se but rather where the water was coming from. "We were also watching all our other tanks to see where we were getting the water from . . . We couldn't find any other water source coming in and we continued maximum let-down."3/ The increased let-down flow coupled with the reduced HPI flow hastened core uncovering.

After emergency feedwater restored steam generator levels approximately 23 minutes into the transient, the steam generators were not used effectively to remove decay heat. The turbine bypass valve control was set to automatically control the steam generator pressure at a value about equal to saturation pressure of the primary coolant. Therefore the valve could not open to remove heat from the reactor coolant system. If the plant operators had used the steam generators as an effective heat sink, cooling of the reactor coolant system would have taken place, ameliorating plant saturation conditions, and as a minimum, would have delayed core uncovering.

It was not recognized that heat was not being effectively removed from the reactor early in the accident. When Faust was interviewed on March 30, he offered some insight concerning removal of decay heat. "As far as I'm concerned, once the turbine is down I don't have a source of steam going out there -- so I'm safe there as far as pulling any more heat off or too much heat out of the core. The reactor starts cooling herself, so the idea is just to stabilize out down at saturation for about 547°F temperature (in the secondary system). O.K.?"4/ The plant status information requested from the computer by the operator 27 minutes into the accident indeed showed that the primary reactor coolant system was also at conditions of about 547°F and 1,040 psig indicating that no heat was being removed and that the primary system was also at saturation conditions.

The increased readings on the out-of-core source range and then the intermediate range nuclear instruments, as would be expected when the coolant density decreased significantly or as the core becomes uncovered, were not understood. These readings were misinterpreted to indicate that the reactor was starting up (becoming critical) although control rods were inserted.

Thirty minutes after the commencement of the accident, reactor building high-temperature alarms were received. This was also not recognized as an important symptom of a LOCA. In response the operators merely started the reactor building emergency cooling booster pumps and placed the reactor building cooling fans in high speed. This slowed the rate of building pressure increase and delayed containment actuation.

When plant conditions degraded to outside the range specified for reactor coolant pump operation, and the coolant contained steam-produced voids causing cavitation and vibration of pumps, the operators tripped the pumps. However, they failed to recognize that the cause for these degraded conditions was a loss of coolant. The act of tripping the pumps was in response to the equipment protection provisions of Operating Procedure 2103-1 4, "Reactor Coolant Pump Operations." The operators did not take appropriate action to be reached, that is, a loss of coolant.

After the PORV block valve was shut at about 6:22 a.m., no significant heat was removed from the core until about 7:12 a.m. when the block valve was again opened. It is not clear why the operators did not immediately start high pressure injection when shutting the PORV block resulted in a sharp pressure increase and clearly indicated that there had been a loss of coolant for over 2 hours through the open PORV.

As hot-leg temperatures increased to off-scale (high) values there is no evidence that the operators recognized the significance of this indication. Rather, it appears that they inferred that average coolant temperature had stabilized near the top of the indicated range.

The station manager was informed of the turbine trip and reactor trip about 2 minutes after these events occurred. He issued no order because he assumed the unit superintendent would.5/

At approximately 5:00 a.m. the station manager was appraised by the duty officer of conditions in the plant. Although he was disturbed, the station manager again issued no order. 6/ Subsequently the station manager set up a conference call including the vice president, Generation, the B&W on-site representative, and the duty officer in the control room. No instructions were given to the duty officer as a result of this lengthy telephone conversation. 7/ Approximately 3 hours after the beginning of the accident, the station manager arrived in the control room. A site emergency had been declared and the station manager assumed the role of emergency director. Again, he issued no direction concerning operation of the plant.8/

The TMI-2 superintendent was called shortly after 4:00 a.m. and was informed only that there had been a turbine trip and a reactor trip. He assumed that the duty officer would be called. When the unit superintendent arrived in the control room at about 5:45 a.m. he tried to ascertain what was taking place. He stated that he did not pay attention to reactor coolant system temperature. He did not issue any order concerning reinitiating high pressure injection. When the station manager arrived, the unit superintendent was placed in charge of ensuring the emergency plan steps were carried out.9/

The duty officer, who was not licensed on TIM-2, arrived in the control room at about 4:50 a.m. and was the first engineer on-site. He was briefed on the situation,10/ considered that he was not sufficiently familiar with the plant and directed that additional technical and operation support be called in.11/ He offered no specific guidance to the operator either before, during or after the conference call with the station manager, et al.

In summary, key management personnel expected the operators to keep them informed of the situation but they did not in turn provide the shift supervisor with instructions which would place the plant in a safe condition.

Operational errors, then, appear to be a significant factor in the accident. It can be argued that overall actions taken worsened the situation during the first 16 hours.

The core was damaged because adequate cooling was not provided to remove decay heat. The operators apparently did not understand that they were not providing adequate cooling to the core. They were confused- by the phenomenon of high pressurizer level and low reactor coolant system pressure which they had not experienced before. Nevertheless, they believed that high pressurizer level indicated a full reactor coolant system. The procedures used by the operator did not recognize the phenomenon of high pressurizer level and low reactor coolant system pressure could occur. 12/ Therefore, the operating procedures did not provide adequate guidance and reactor safety was dependent on the operators' ability to comprehend the significance of key parameters such as reactor coolant system pressure and temperature. The early throttling of high pressure injection for a period of time is understandable; operators had committed the same error at Davis-Besse-1 in 1977. However, in the face of continuing indications that the core was being hazarded, serious errors were committed in continuing letdown and not injecting water into the core.

APPENDIX C NOTES

1/ Frederick interview, 0124, March 30, 1979.

2/ Faust interview, 0400, March 30, 1979.

3/ Scheimann interview, 0230, March 30, 1979.

4/ Faust interview, Ibid.

5/ Miller deposition at 235.

6/ Ibid. at 239.

7/ Ibid. at 247.

8/ Ibid. at 265-266.

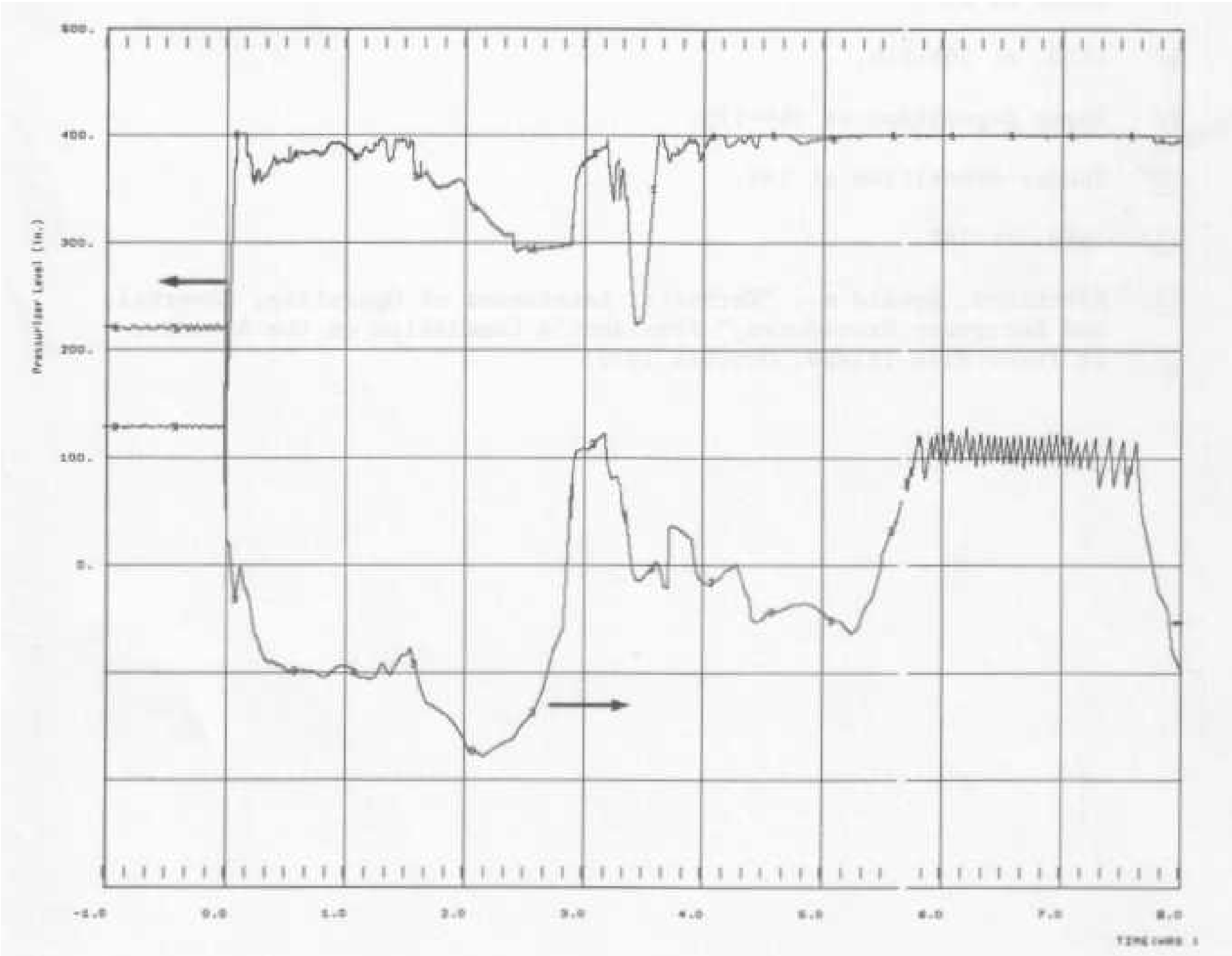
9/ Logan deposition at 169-175.

10/ Kunder deposition at 143.

11/ Ibid. at 149.

12/ Eytchison, Ronald m., "Technical Assessment of Operating, Abnormal, and Emergency Procedures," President's Commission on the Accident at Three Mile Island, October 1979.

FIGURE 1: Composite Primary Coolant Pressure and Pressurizer Level



Source: NSAC, "Analysis of Three Mile Island-Unit 2 Accident," NSAC/Electric Power Research Institute, NSAC-1, July 1979.

FIGURE 1 (Continued)

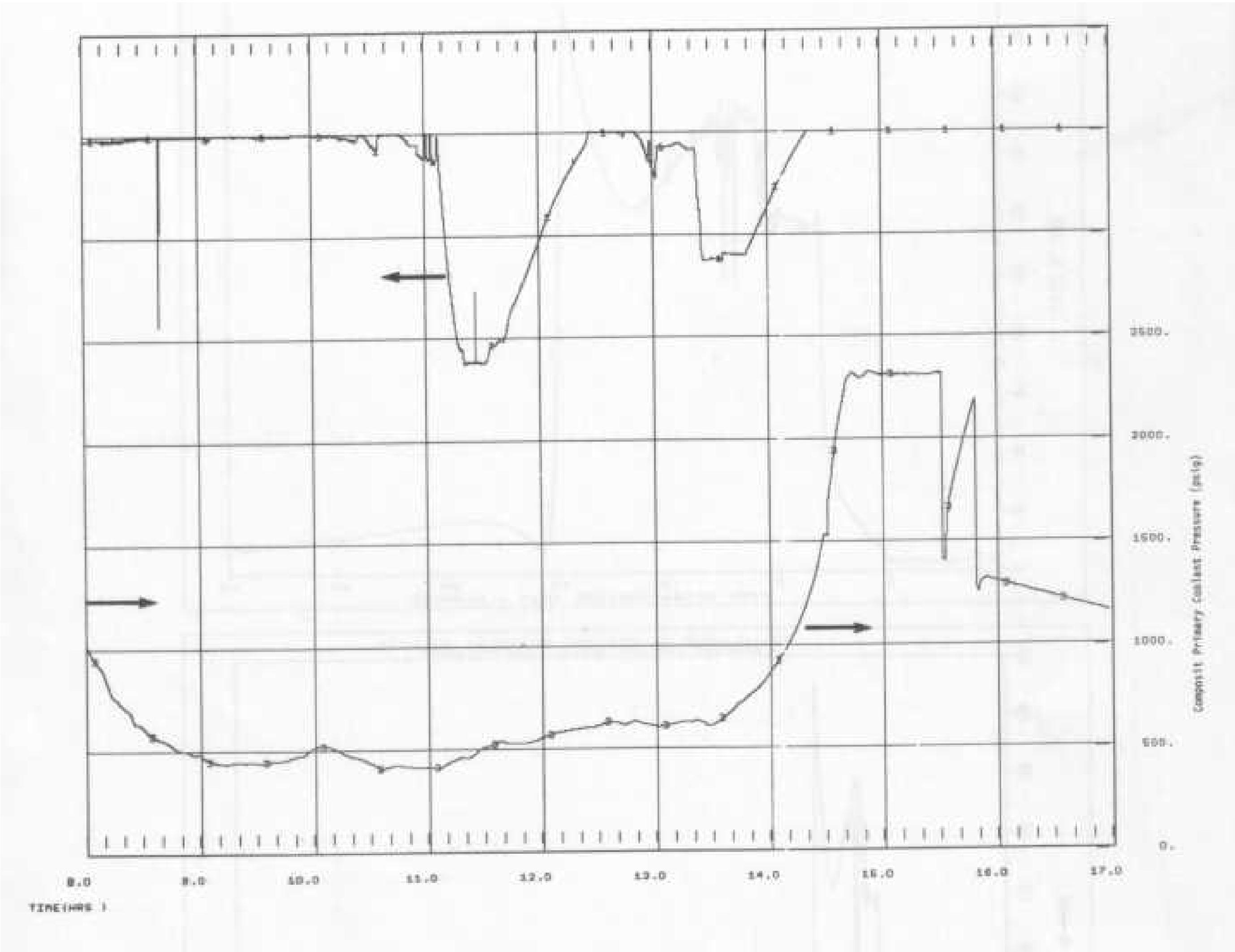
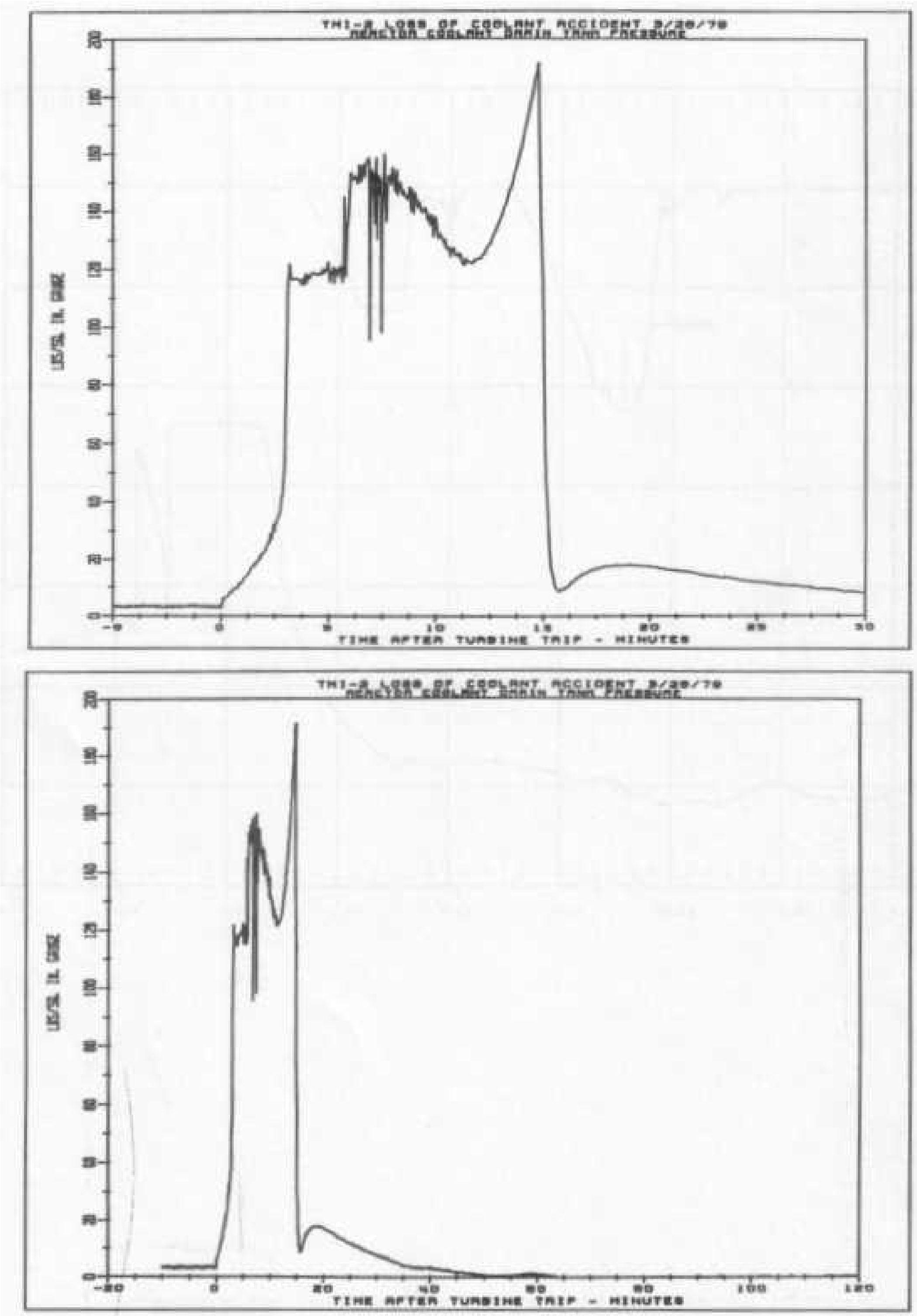
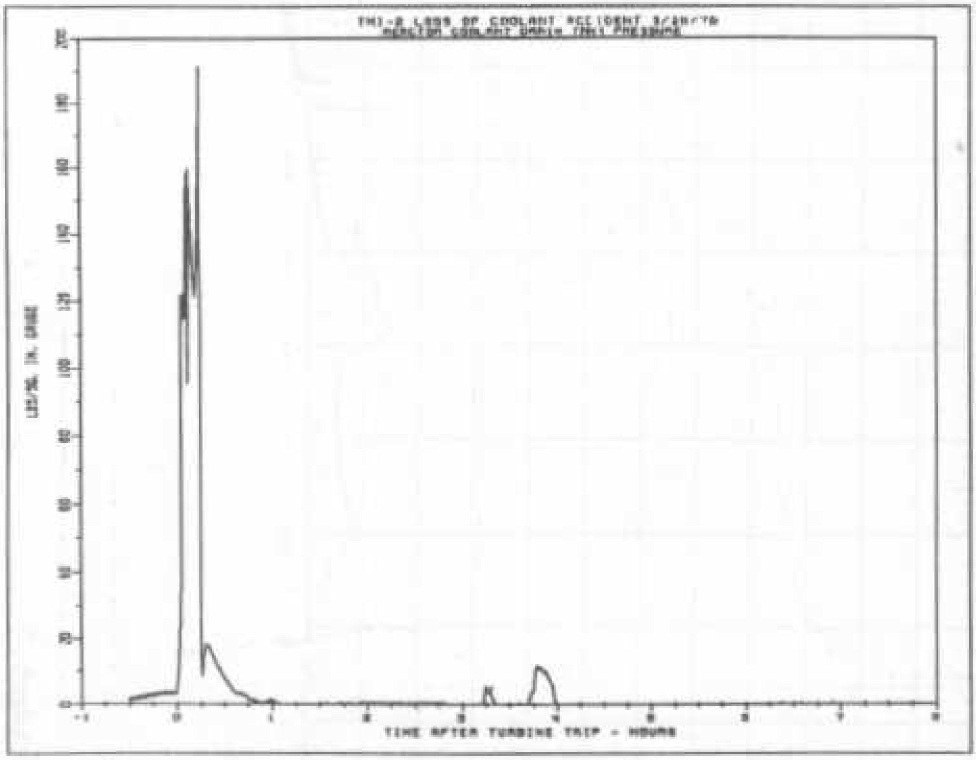


FIGURE 2: TMI-2 Loss-of-Coolant Accident of March 28, 1979, Reactor Coolant Drain Tank Pressure



Source: GPU, "Preliminary Annotated Sequence of Events, March 28, 1979," Revision 0, May 1979.

FIGURE 2 (Continued)



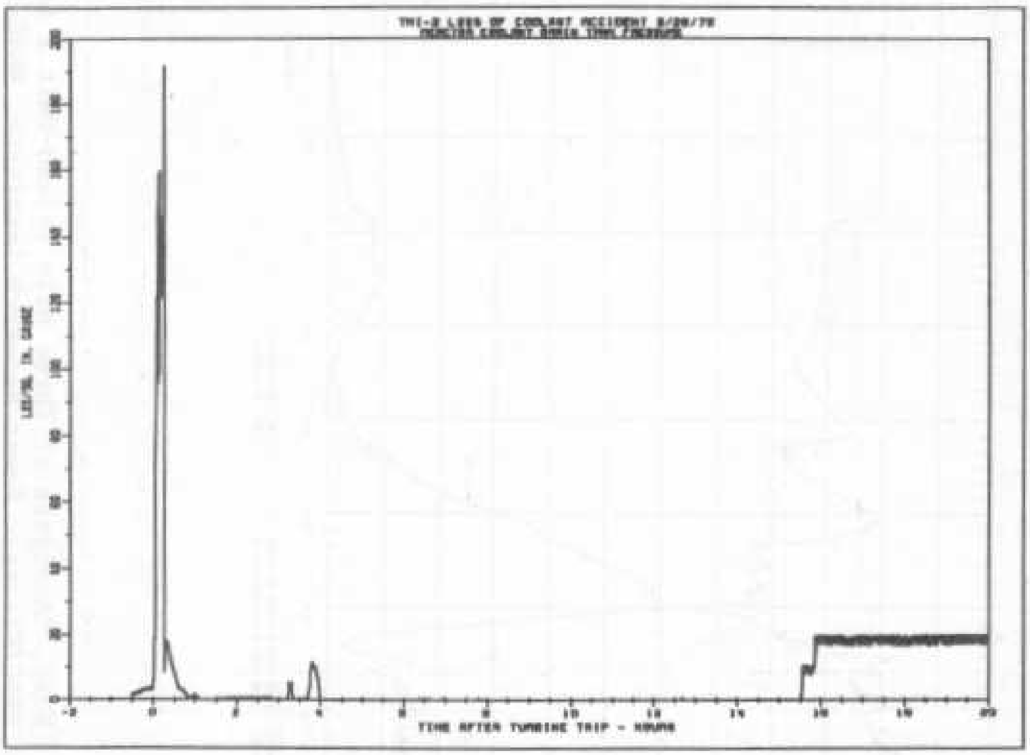
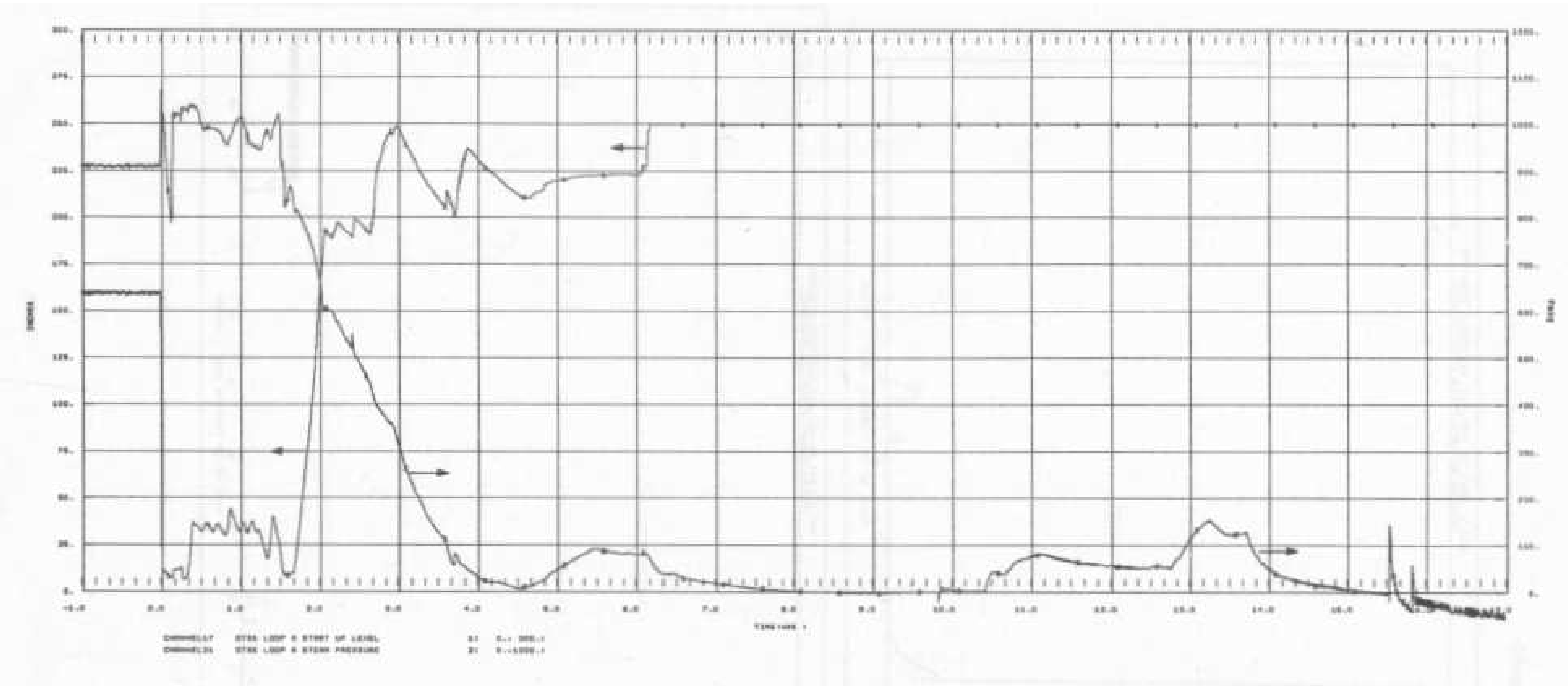


FIGURE 3: Loop A Steam Pressure
Startup Level



Source: NSAC, "Analysis of Three Mile Island-Unit 2 Accident," NSAC/Electric Power Research Institute, NSAC-1, July 1979.

FIGURE 3 (Continued)
operating Level

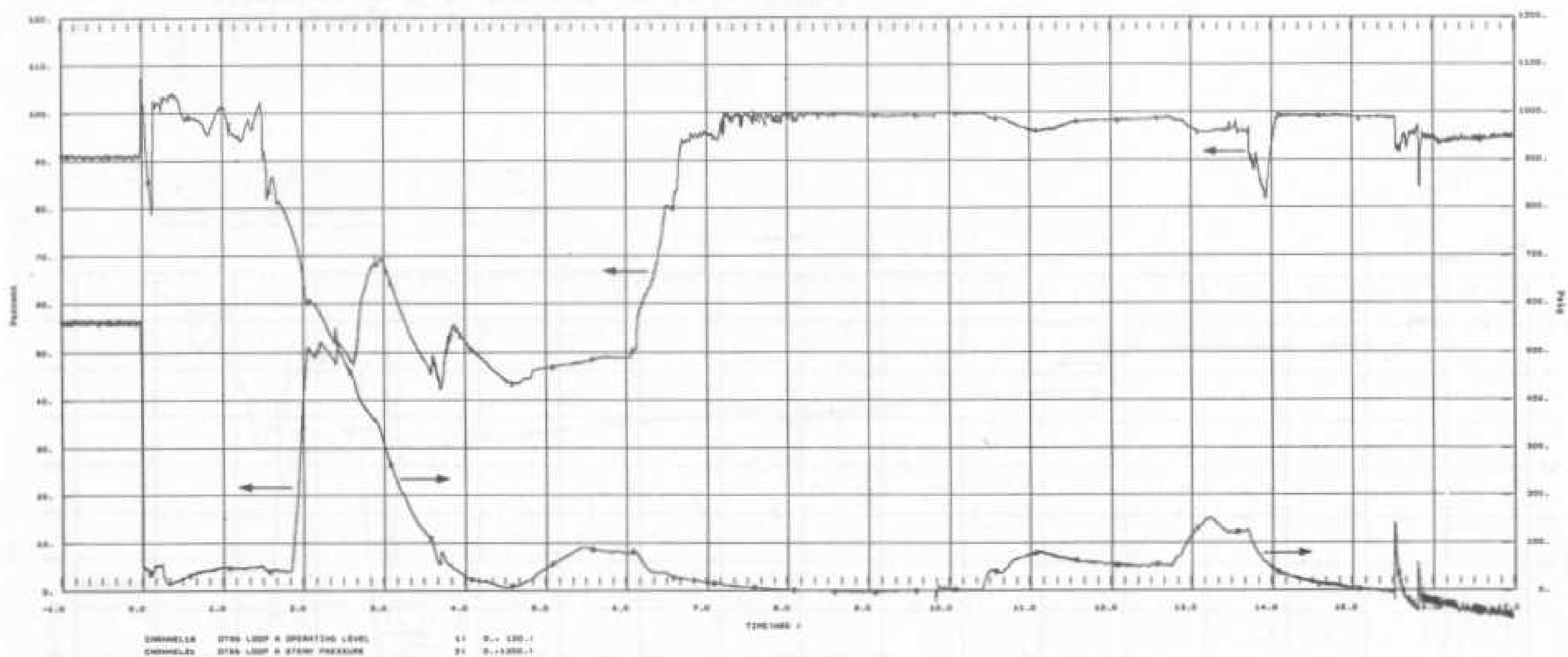
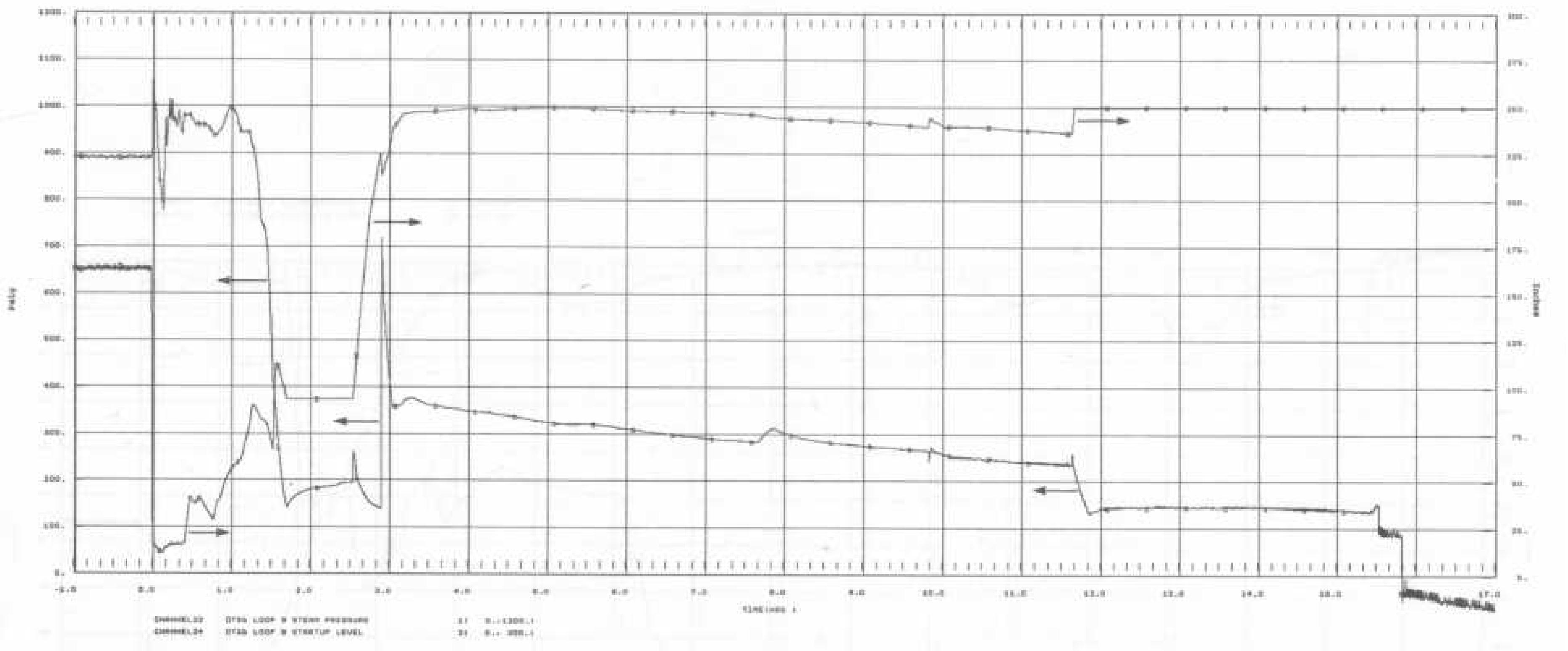


FIGURE 4: Loop B Steam Pressure
Startup Level



Source: NSAC, "Analysis of Three Mile Island-Unit 2 Accident," NSAC/Electric Power Research Institute, NSAC-1, July 1979.

FIGURE 4 (Continued)
Operating Level

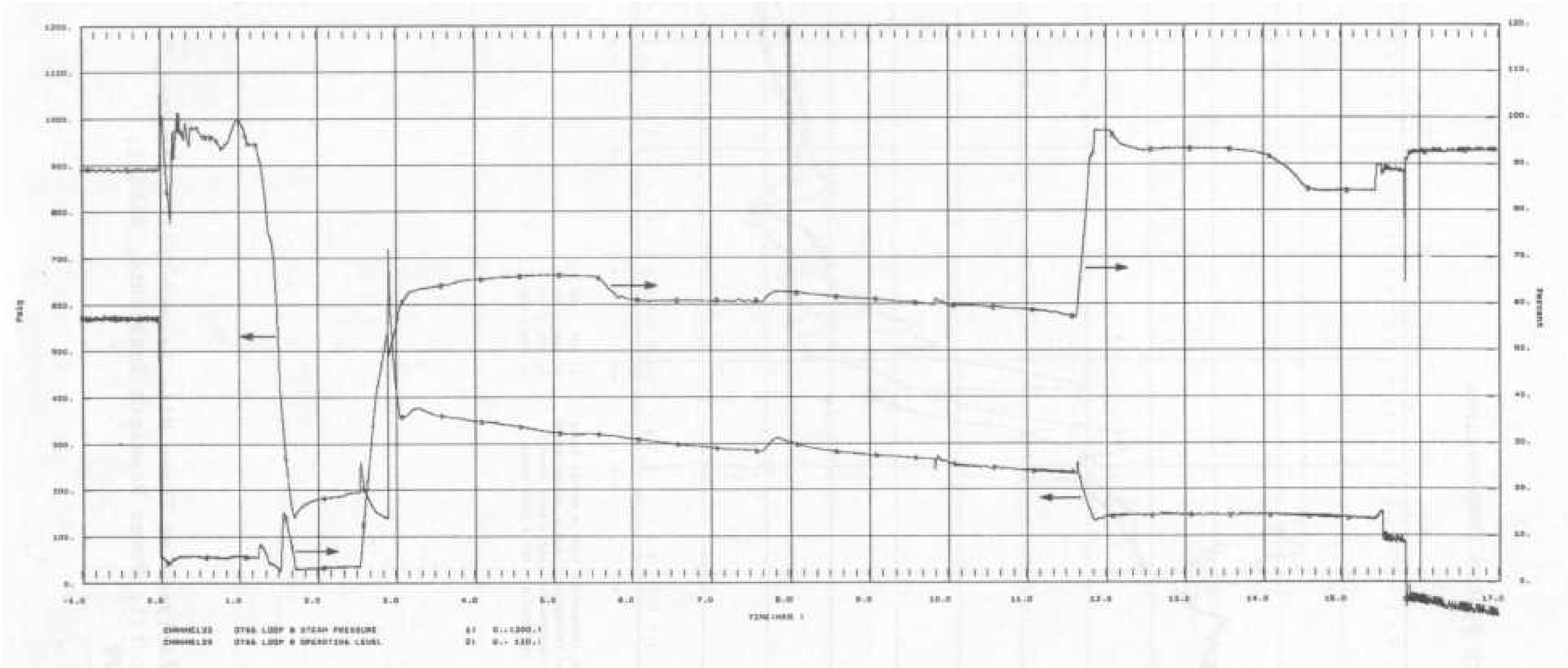
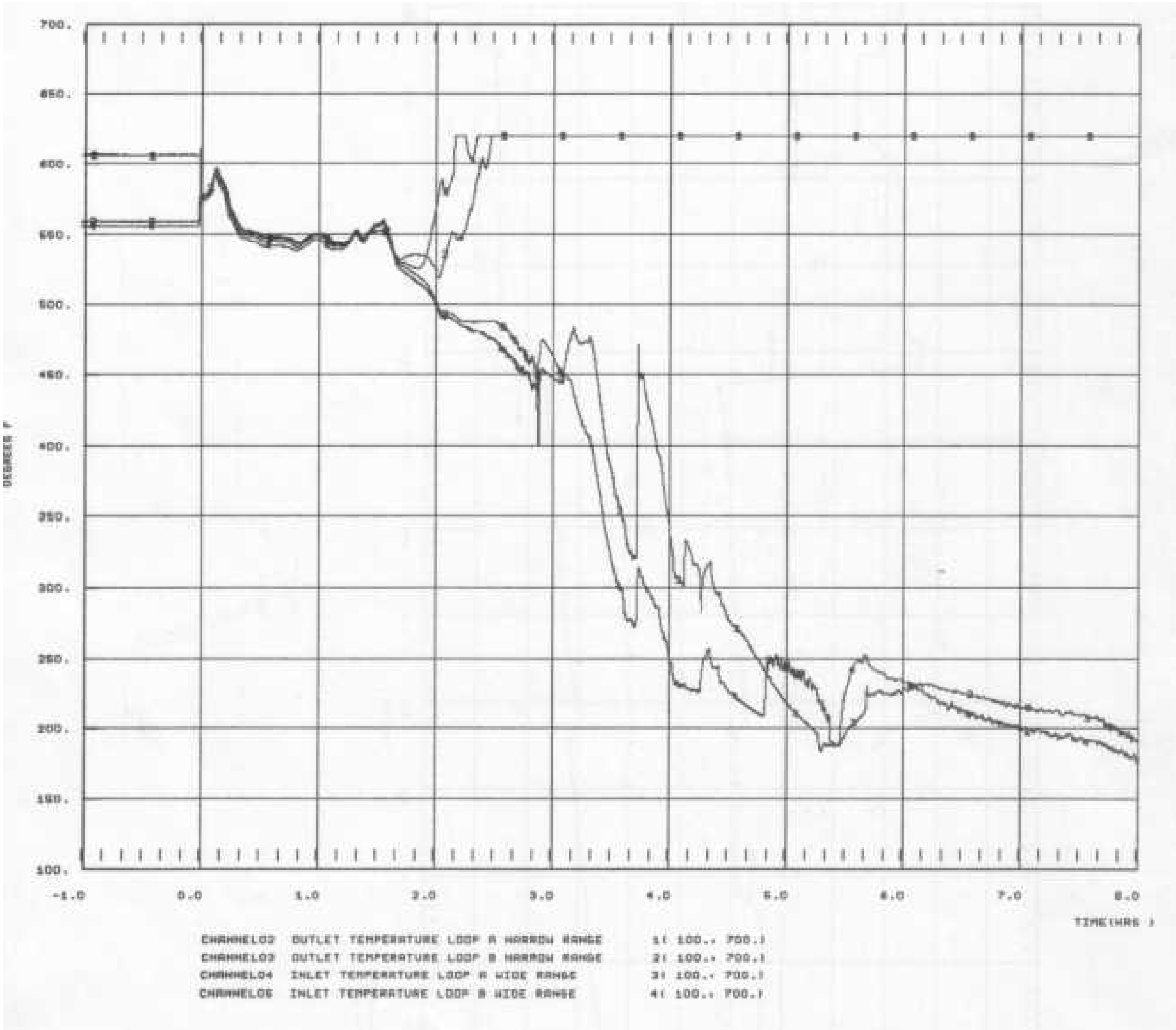


FIGURE 5: Primary System Temperatures



Source: NSAC, "Analysis of Three Mile Island-Unit 2 Accident," NSAC/Electric Power Research Institute, NSAC-1, July 1979.

FIGURE 5 (Continued)

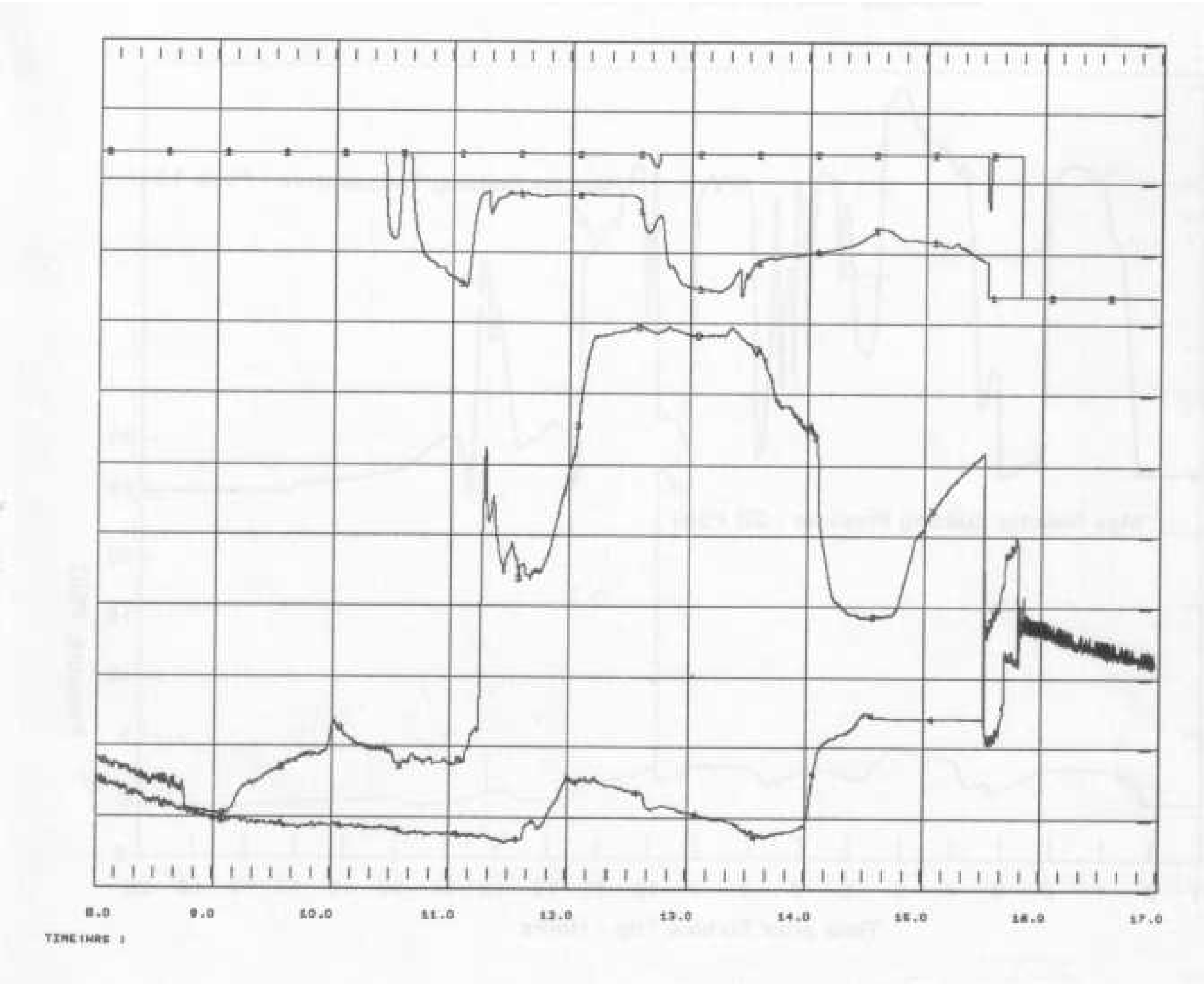
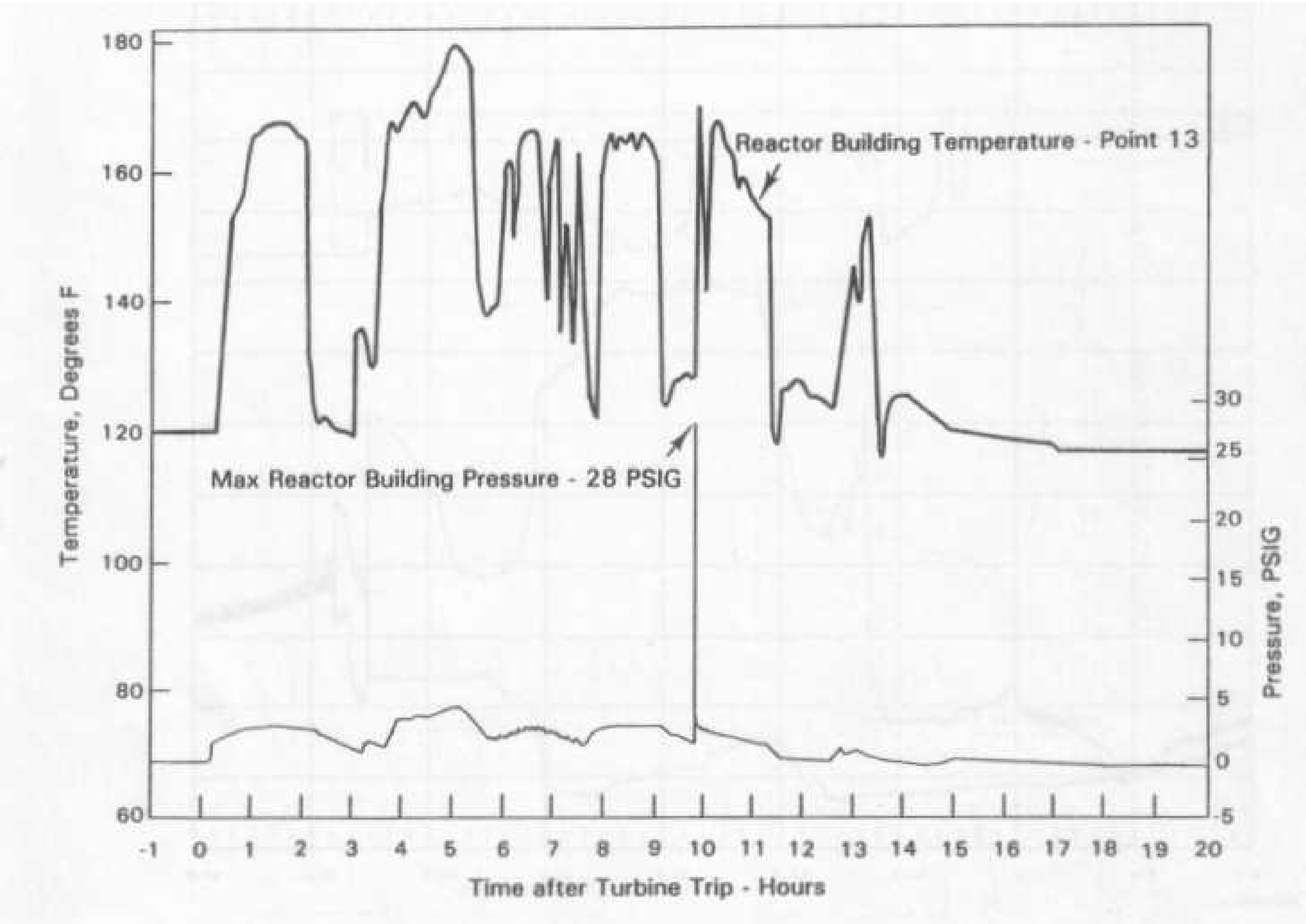
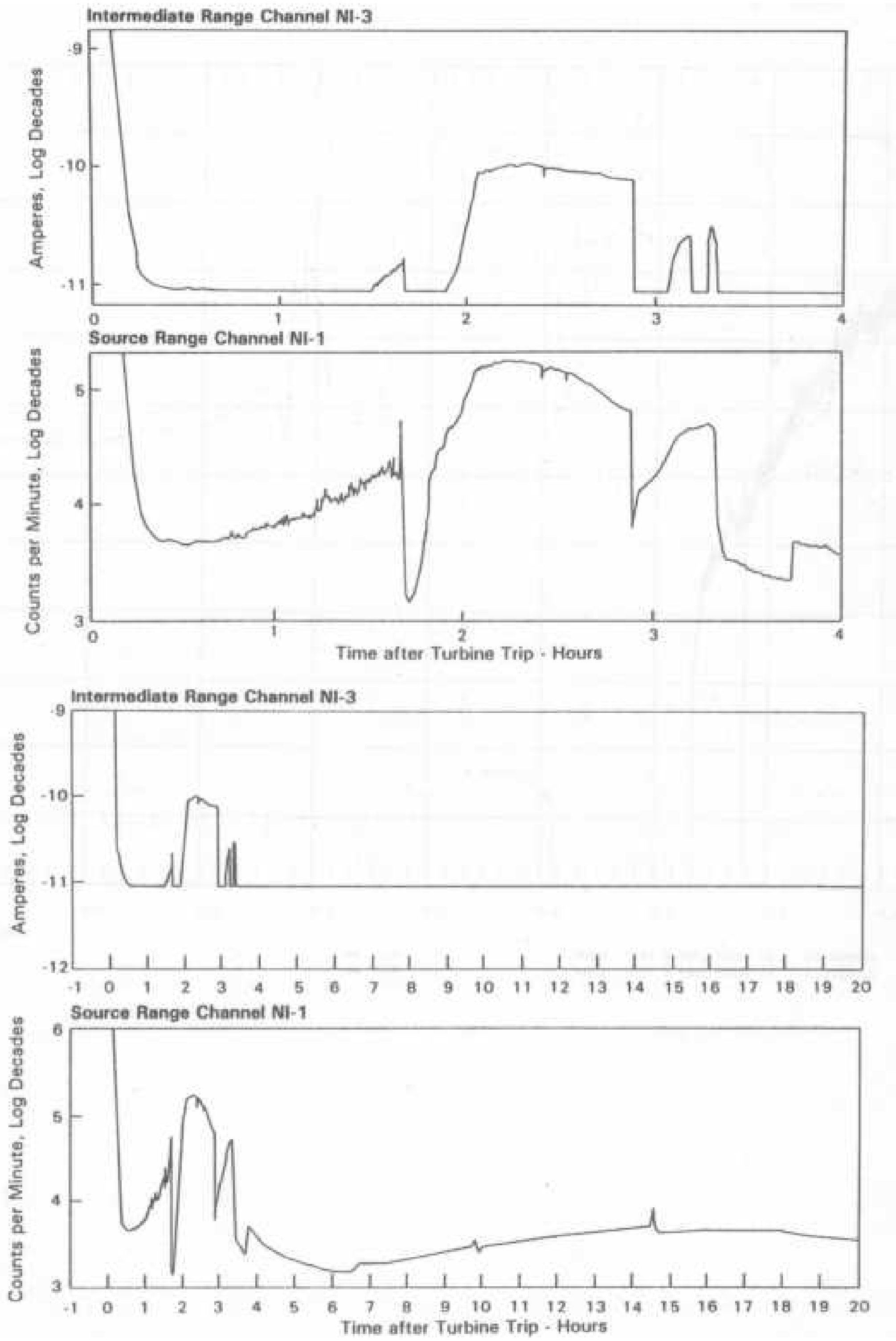


FIGURE 6: TMI-2 Loss-of-Coolant Accident of March 28, 1979, Reactor Building Temperature and Pressure



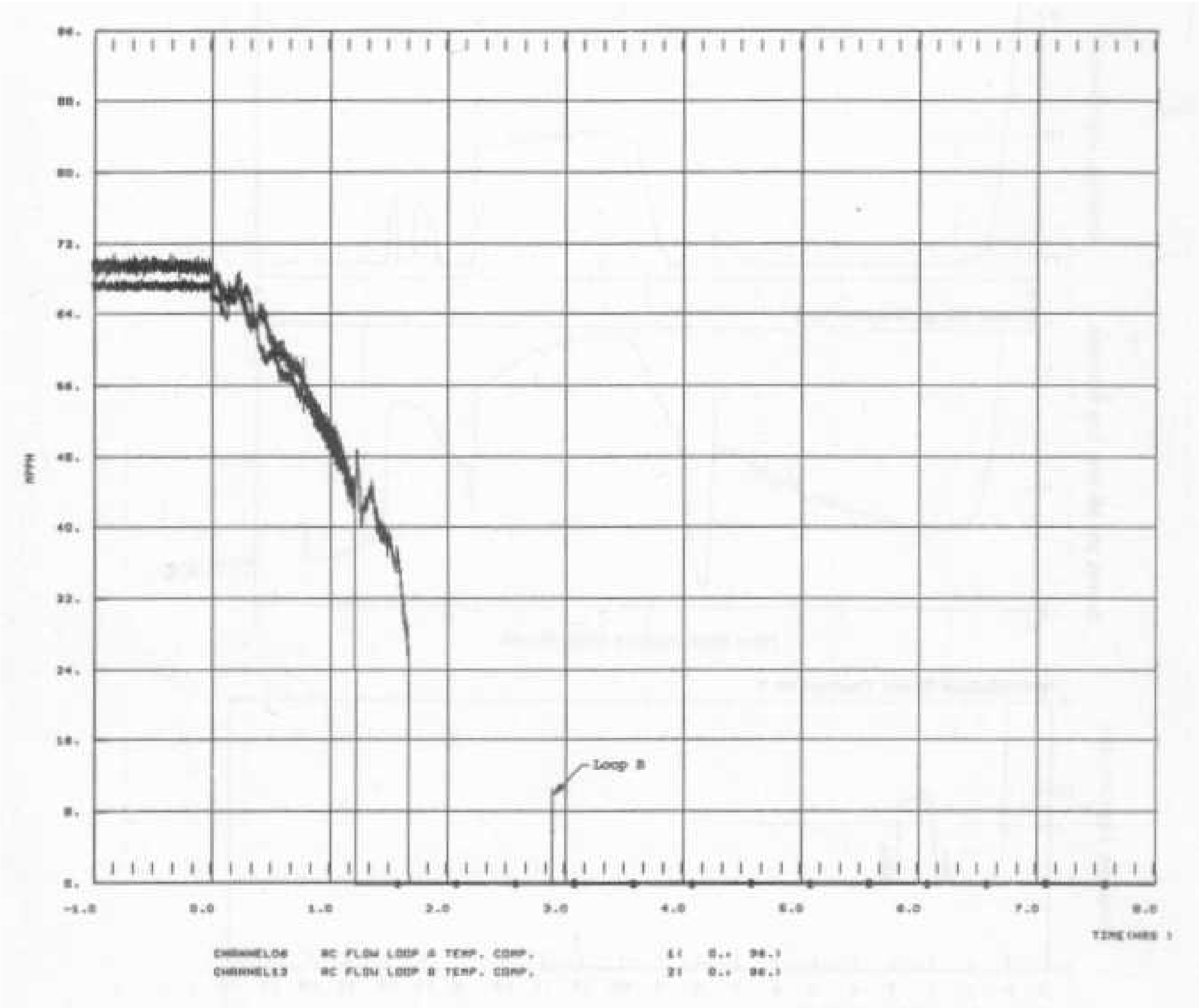
Source: GPU, "Preliminary Annotated Sequence of Events, March 28, 1979," Revision 0, May 1979.

FIGURE 7: TMI Loss-of-Coolant Accident of March 28, 1979, Intermediate and Source Range Nuclear Instrumentation



Source: GPU, "Preliminary Annotated Sequence of Events, March 28, 1979," Revision 0, May 1979.

FIGURE 8: Primary System Flow



Source: NSAC, "Analysis of Three Mile Island-Unit 2 Accident," NSAC/Electric Power Research Institute, NSAC-1, July 1979.

FIGURE 8 (Continued)

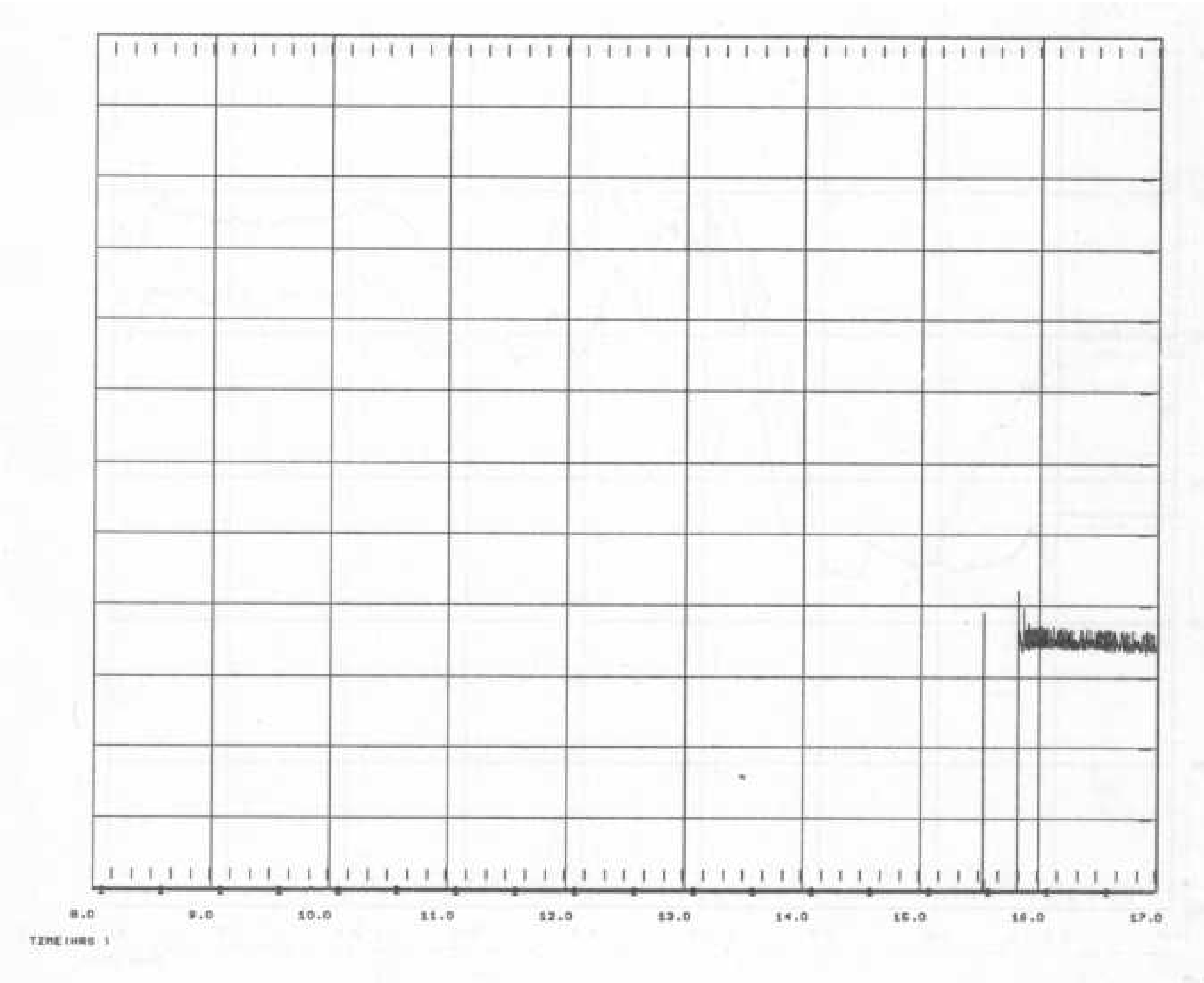
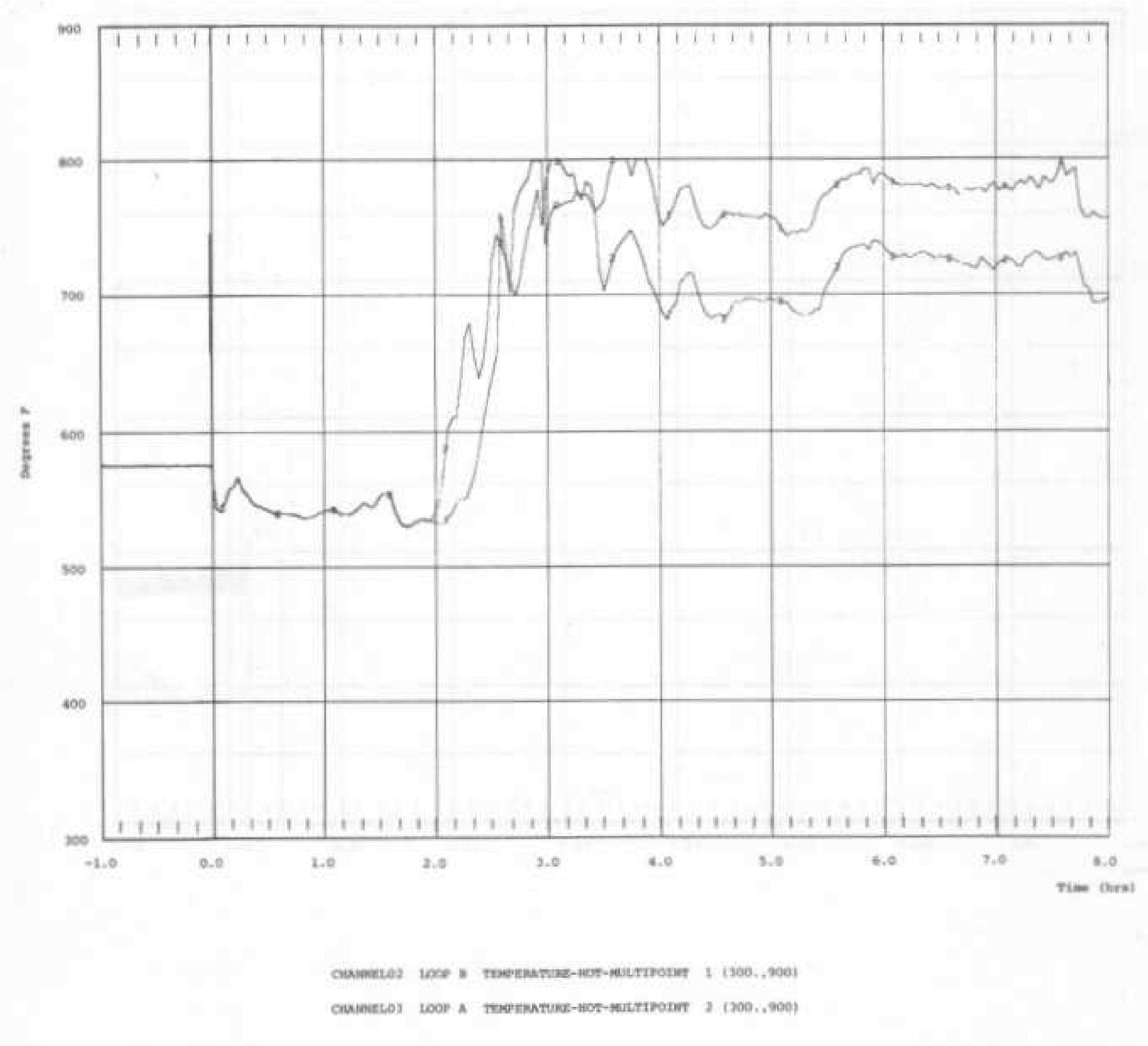
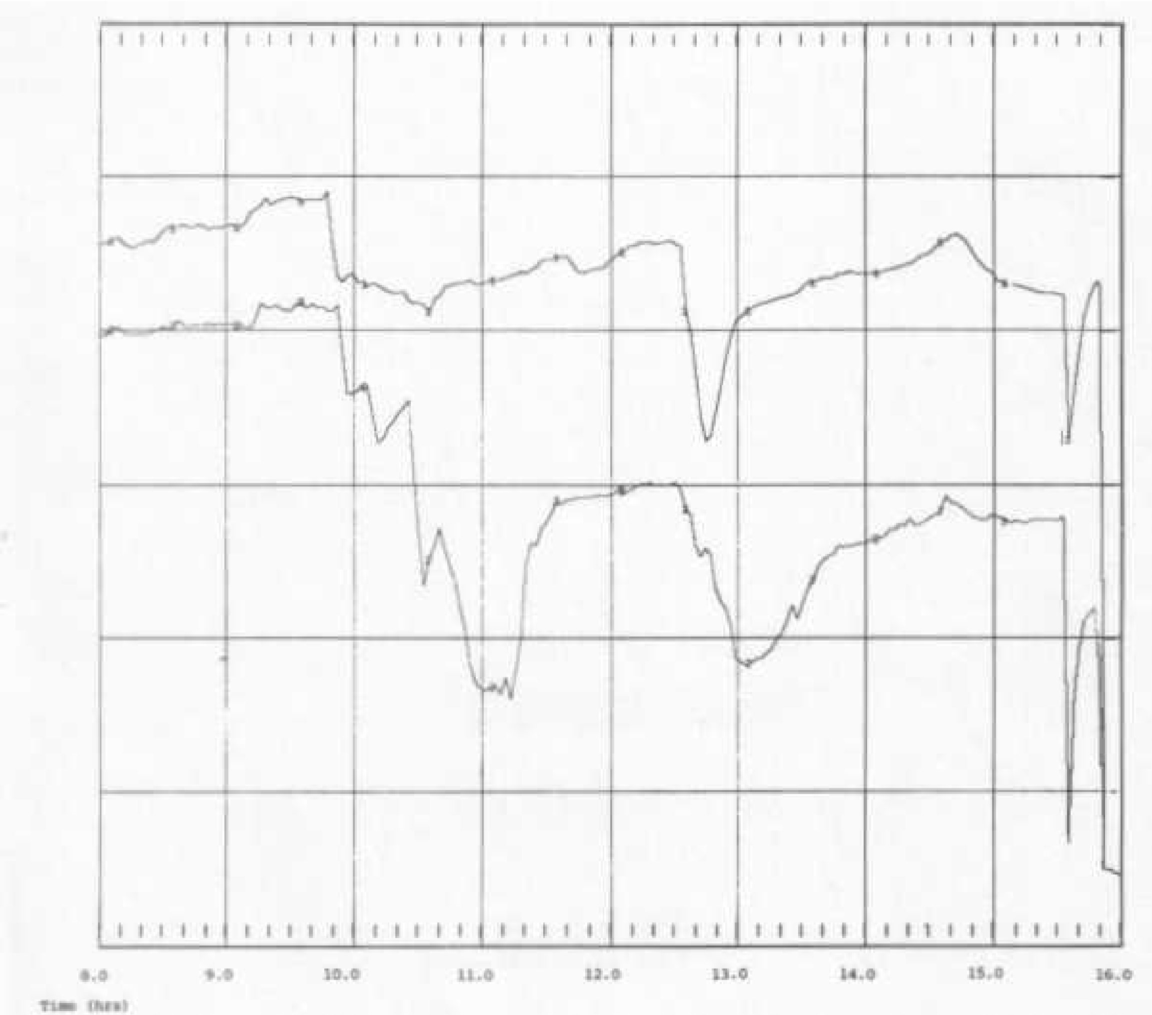


FIGURE 9: Primary System Hot Leg Temperatures (Multi-Point Recorder)



Source: NSAC-1, "Supplement to Analysis of Three Mile Island -- Unit 2 Accident," Electric Power Research Institute (Nuclear Safety Analysis Center), Figure TH2O, October 1979.

FIGURE 9 (Continued)



Staff Reports To

THE PRESIDENT'S COMMISSION ON
THE ACCIDENT AT
THREE MILE ISLAND

The Nuclear Regulatory Commission, Report of the Office of Chief Counsel

The Role of the Managing Utility and Its Suppliers, Report of the Office of Chief Counsel

Emergency Preparedness, Emergency Response, Reports of the Office of Chief Counsel

Reports of the Technical Assessment Task Force, Vol. I
- "Technical Staff Analysis Reports Summary"
- "Summary Sequence of Events"

Reports of the Technical Assessment Task Force, Vol. II
- "Chemistry"
- "Thermal Hydraulics"
- "Core Damage"
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Report of the Public's Right to Information Task Force