

NUREG-0600

INVESTIGATION INTO THE MARCH 28, 1979
THREE MILE ISLAND ACCIDENT
BY
OFFICE OF INSPECTION AND ENFORCEMENT

Investigative Report No. 50-320/79-10



Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission

Available from
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Available from
National Technical Information Service
Springfield, Virginia 22161

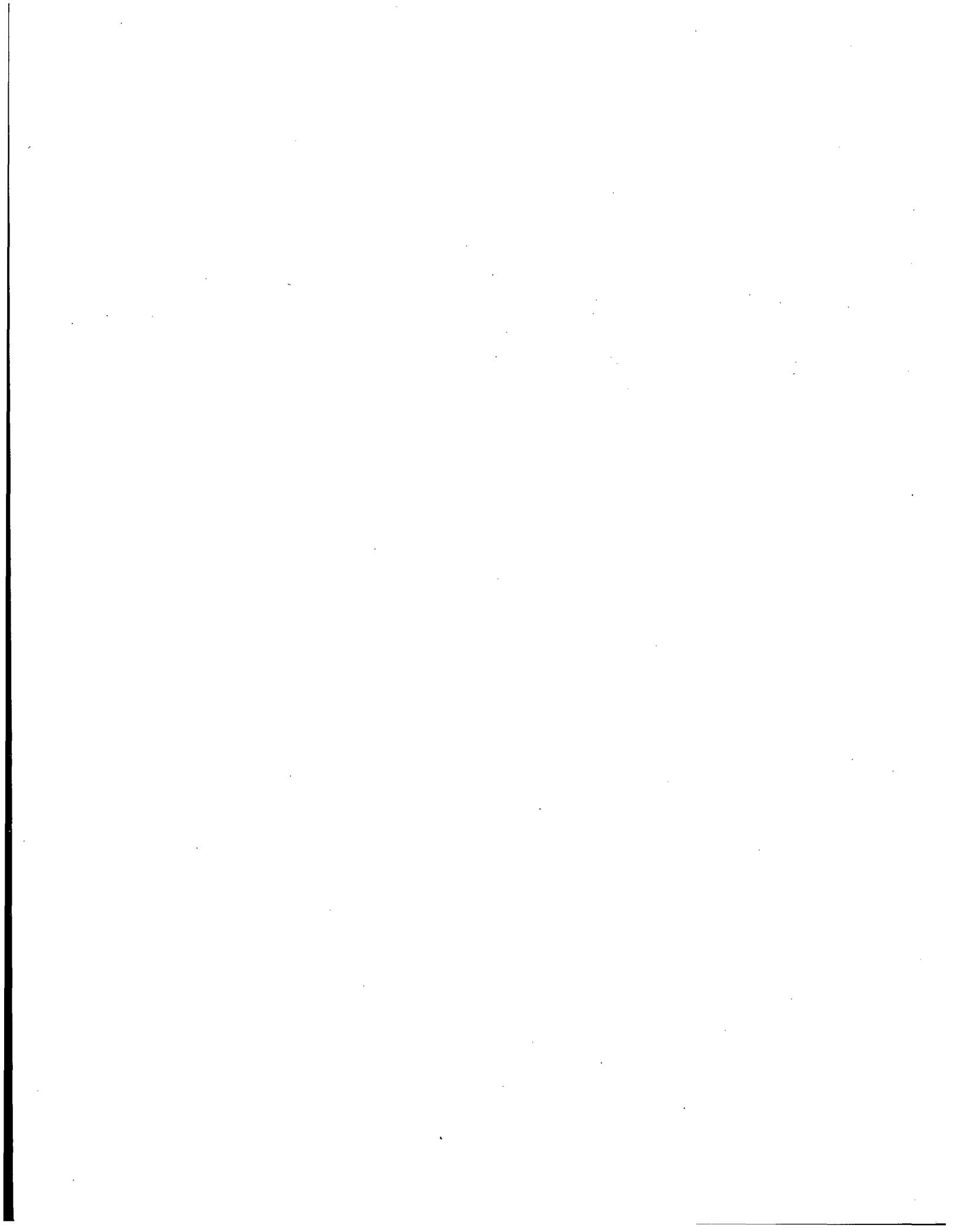
NUREG-0600

**INVESTIGATION INTO THE MARCH 28, 1979
THREE MILE ISLAND ACCIDENT
BY
OFFICE OF INSPECTION AND ENFORCEMENT**

Investigative Report No. 50-320/79-10

**Manuscript Completed: July 1979
Date Published: August 1979**

**Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555**



FOREWORD

On March 28, 1979, the Three Mile Island Unit 2 Nuclear Power Plant experienced the most severe accident in U.S. commercial nuclear power plant operating history. This report sets forth the facts concerning the events of the accident determined as a result of an investigation by the NRC Office of Inspection and Enforcement. The IE investigation, which is based on the information available at this time, is limited to two aspects of the accident:

1. Those related operational actions by the licensee during the period from before the initiating event until approximately 8:00 p.m., March 28, when primary coolant flow was re-established by starting a reactor coolant pump, and
2. Those steps taken by the licensee to control the release of radioactive material to the off-site environs, and to implement his emergency plan during the period from the initiation of the event to midnight, March 30.

These investigation periods were selected because they include the licensee actions which most significantly affected the accident sequence and its results.

The results of the IE investigation supports the reported population dose from the accident, developed by an ad hoc dose assessment group, which included representatives of various cognizant Federal agencies. In its report dated May 10, 1979, this group concluded that, "Based on the current assessment . . . the off-site collective dose associated with the radioactive material released during the period of March 28 to April 7, 1979, represents minimal risks (that is, a very small number) of additional health effects to the off-site population." At the same time, the IE investigation identifies several inadequacies in the inplant radiation protection activities of the licensee and criticizes the measurements of off-site radiation levels made by the licensee. In spite of these identified flaws, no glaring inconsistencies have been found which would significantly alter the conclusions reached by the ad hoc group.

The IE investigation also substantiates earlier conclusions concerning the underlying causes of the accident and those factors that contributed to its severity. Inadequacies in six major areas have been confirmed:

1. Equipment performance (failures and maloperation).
2. Transient and accident analyses.
3. Operator training and performance.

4. Equipment and system design.
5. Information flow, particularly during the early hours of the accident.
6. Implementation of emergency planning.

Perhaps the most disturbing result of the IE investigation is confirmation of earlier conclusions that the Three Mile Island Unit 2 accident could have been prevented, in spite of the inadequacies listed above. The design of the plant, the equipment that was installed, the various accident and transient analyses, and the emergency procedures were adequate to have prevented the serious consequences of the accident, if they had been permitted to function or be carried out as planned. For example, had the operators allowed the emergency core cooling system to perform its intended function, damage to the core would most likely have been prevented. There are other examples set forth in the report where, had a particular operator action been taken, the consequences of the accident could have been significantly mitigated. On the other hand, had certain equipment been designed differently, it too, could have prevented or reduced the consequences of the accident. The results of the investigation make it difficult to fault only the actions of the operating staff. There is considerable evidence of a "mind set," not only by TMI operators but by operators at other plants as well, that overfilling the reactor coolant system (making the system solid) was to be avoided at almost any cost. Undue attention by the TMI operators to avoiding a solid system led them to ignore other procedural instructions and indications that the core was not being properly cooled. Without this "mind set" they might well have acted to preclude or better mitigate the accident. Subsequent actions have been required by NRC to retrain all licensed operators in an effort to preclude recurrence. Upgraded procedural instructions have also been required.

It is clear that substantial effort is needed, by both the NRC and the industry, to assure that these lessons learned concerning the TMI accident are implemented at other facilities. Within the NRC, early action has been taken to inform other nuclear power plant licensees of the circumstances surrounding the Three Mile Island accident and to require immediate implementation of compensatory measures to prevent occurrence of similar accidents elsewhere. In addition, a special Lessons Learned Task Force was established in the NRC Office of Nuclear Reactor Regulation. This group has studied the Three Mile Island accident and has issued a report (NUREG-0578) containing short-term recommendations that will significantly improve continued safe operation of licensed nuclear power plants. The IE investigation adds further emphasis to the need for such plant and procedural modifications.

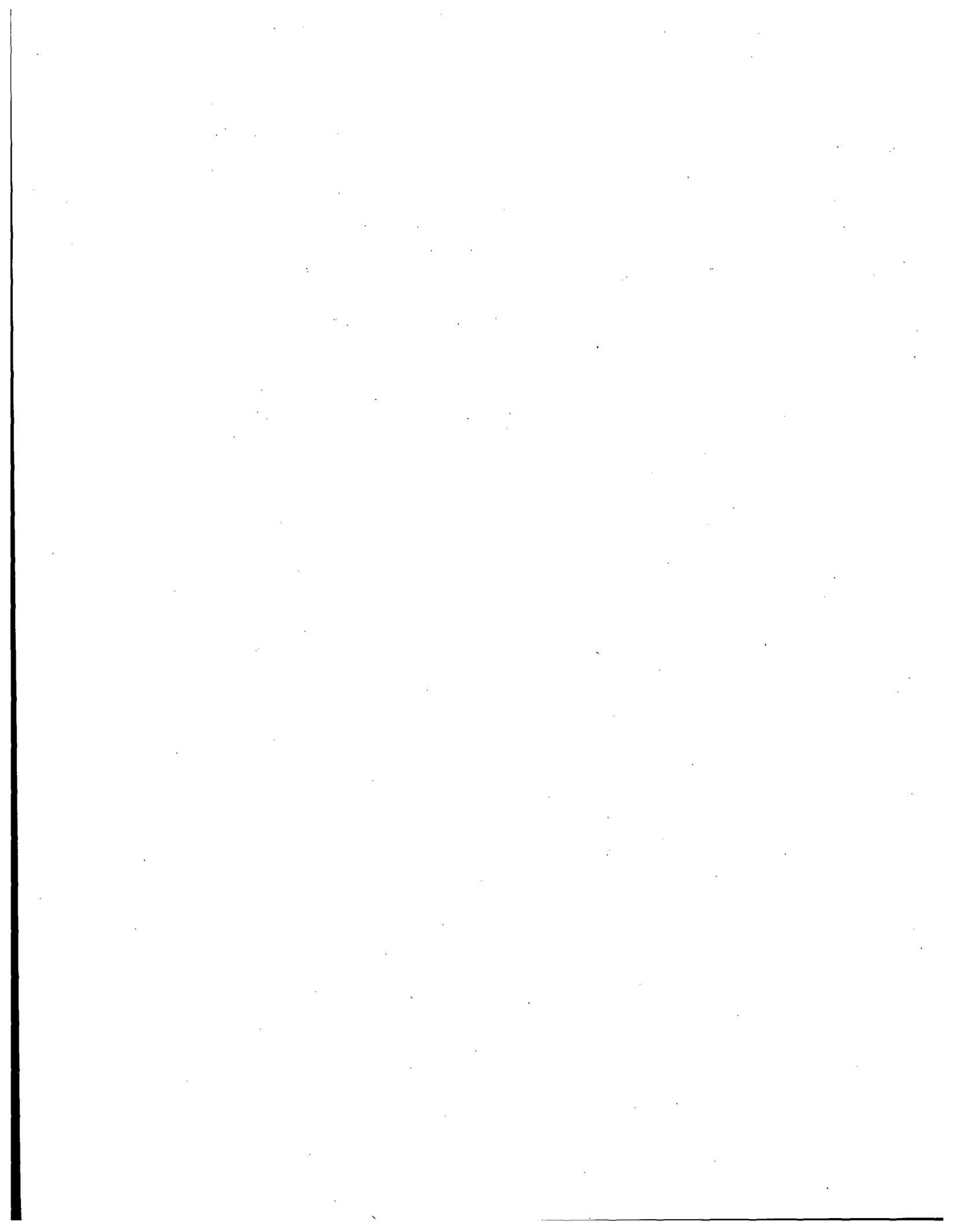
Because they have the benefit of hindsight, most retrospective investigations like this tend to emphasize areas where people and equipment did not perform as desired. The IE investigation team made a concerted effort to evaluate the reasoning processes of the people who were operating the plant during the course of the accident. The report contains the team's conclusion as to whether or not the operating staff's actions were appropriate in light of the training and factual information available to them at the time they had to make decisions as to what course of action to follow.

Further study is clearly needed with respect to the contributions of various other organizations that influence the operation of nuclear power plants, including designers, reviewers, builders, vendors and regulatory agencies. These various studies are now underway; most notably the Presidentially appointed Kemeny Commission, as well as a wide-ranging internal NRC study under Mr. Mitchell Rogovin. A full assessment of all the underlying causes of the Three Mile Island accident must await completion of these studies.

The findings of this IE investigation will be the subject of appropriate enforcement action in accordance with the Commission's regulations (Part 2, Title 10, CFR).



Victor Stello, Jr.
Director
Office of Inspection
and Enforcement



U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

Region I

Report No. 50-320/79-10

Docket No. 50-320

License No. DPR-73

Category C

Licensee: Metropolitan Edison Company
P.O. Box 542
Reading, Pennsylvania 19640

Facility Name: Three Mile Island Nuclear Station Unit 2

Investigation at: Middletown, Pennsylvania

Investigation conducted: March 28-July 31, 1979

Investigators

Operational Aspects Team

J.S. Creswell 7/24/79
J.S. Creswell Date

A.N. Fasano 7/24/79
A.N. Fasano Date

D.R. Hunter 7/24/79
D.R. Hunter Date

D.C. Kirkpatrick 7/24/79
D.C. Kirkpatrick Date

R.J. Marsh 7/24/79
R.J. Marsh Date

T.T. Martin 7/24/79
T.T. Martin Date

Team Leaders

R.D. Martin 7/24/79
R.D. Martin Date

Radiological Aspects Team

D.M. Collins 7/24/79
D.M. Collins Date

D.E. Donaldson 7/24/79
D.E. Donaldson Date

T.H. Essig 7/24/79
T.H. Essig Date

L.L. Jackson 7/24/79
L.L. Jackson Date

O.C. Shackleton 7/24/79
O.C. Shackleton Date

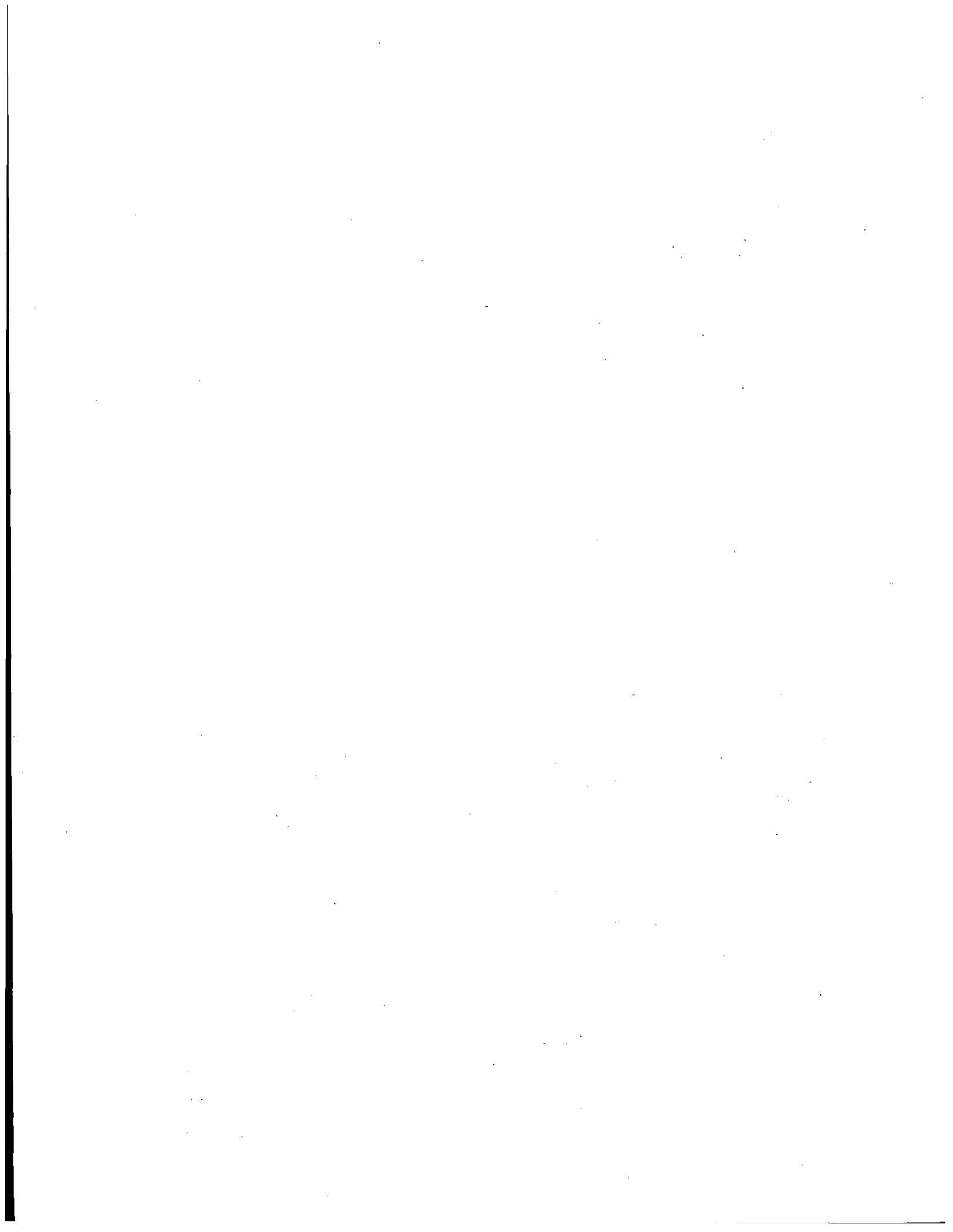
G.P. Yuhas 7/24/79
G.P. Yuhas Date

A.F. Gibson 7/24/79
A.F. Gibson Date

Approved by:

J.M. Allan
J.M. Allan, Deputy Director,
Region I, Investigation Director

7/26/79
Date



PREFACE

This Report was prepared by the investigation team assembled by the Office of Inspection and Enforcement of the U.S. Nuclear Regulatory Commission to investigate certain aspects of the accident at the Three Mile Island Unit 2 facility on March 28, 1979.

The investigation by the Office of Inspection and Enforcement had two basic goals:

- o To establish, in a comprehensive manner, the facts concerning the events of Three Mile Island accident during the period investigated.
- o To evaluate the performance of the Licensee in association with the Three Mile Island accident as a basis for corrective action or enforcement action as appropriate.

This report contains a factual recounting, to the extent it has been possible to establish at this time, of the significant operational and radiological events that transpired during the early hours and days of this accident. The report further provides an analysis of the actions of the licensee staff in light of those facts when compared to the NRC requirements to which they are subject.

This report is critical, in several instances, of the actions taken by the licensee staff during that accident and in the first few days thereafter.

The perspective we ask the reader to retain is to be aware that reports yet to be issued by other investigatory bodies may address evaluation of other organizations associated with the Three Mile Island facility. These

organizations include the designers, reviewers, builders, vendors, and regulatory agencies that are or were involved with this facility.

This report is not a definitive study of every facet of the Three Mile Island accident. Nor is it an engineering evaluation of the accident. Those evaluations are being done by other Offices within this agency as well as by other organizations. The reports from other ongoing investigations and studies will be needed to fully understand the causes of this accident and the appropriate actions to be taken as a result of that understanding.

ACKNOWLEDGEMENT

The investigation team wishes to express its appreciation to the many individuals of the Headquarters and Regional staffs who provided assistance to the team throughout this investigation. Moreover, the assistance extended by other Offices within NRC that provided personnel, review and other forms of assistance is also appreciated. Most noteworthy, however, is the support that was provided by the Word Processing Unit in Region I who, with help from other regional and headquarters personnel, were faced with the formidable task of handling not only this report, but the transcription of over 200 interviews.

J. M. Allan

R. D. Martin	A. F. Gibson
J. S. Creswell	D. M. Collins
A. N. Fasano	D. E. Donaldson
D. R. Hunter	T. H. Essig
D. C. Kirkpatrick	L. L. Jackson
R. J. Marsh	O. C. Shackleton
T. T. Martin	G. P. Yuhas

TABLE OF CONTENTS

Preface	i
Acknowledgements	iii
Table of Contents	iv
Introduction	v
Summary	1
Details I Table of Contents	I-i
Details I - Operational Aspects	I-1
Details II Table of Contents	II-i
Details II - Radiological Aspects	II-1
Enclosure 1 - Memoranda Establishing Investigation	E1-1
Enclosure 2 - Interview List	E1-2

Introduction

a. Location of Plant

The Three Mile Island Nuclear Station is located on Three Mile Island in the Susquehanna River, approximately 10 miles southwest of Harrisburg, Pennsylvania. It is in Londonderry Township of Dauphin County. - Goldsboro is located 1.2 miles west of Three Mile Island and Middletown is located 3 miles north of the facility.

Three Mile Island is one of the largest of a group of islands in the Susquehanna River and is situated about 900 feet from the east bank. It is elongated parallel to the flow of the river, with its longer axis oriented approximately due north and south.

An access bridge connects State Highway Route 441 with the north end of the island. A wood access bridge connects the south end of the island with Route 441. Route 441 is a two-lane black-topped road which runs north and south, parallel to Three Mile Island on the east bank of the Susquehanna River. Route 441 is approximately 2,000 feet from the reactor buildings at the closest point.

On the east bank of the river there is a one-track railroad line adjacent and parallel to Route 441. On the west bank of the Susquehanna River, a distance of approximately 1 1/4 miles, there is a multi-track railroad line and a two-lane, black-topped road.

b. Facility Description

Two pressurized water reactors (designated as Unit 1 and Unit 2) are located on Three Mile Island and, together, constitute the Three Mile Island Nuclear Station. The licensee of the two units is the Metropolitan Edison Company of Reading, Pennsylvania.

Unit 1 was licensed to begin operation on April 19, 1974, at a rated power not to exceed 2,535 megawatts thermal. It achieved criticality on June 5, 1974,

and went into commercial operation on September 2, 1974. Unit 2 was licensed to begin operation on February 8, 1978, at a rated power of 2,772 megawatts thermal. It achieved criticality on March 28, 1978, and went into commercial operation on December 30, 1978. The nuclear steam system supplier for Unit 2 was Babcock and Wilcox and the architect-engineer was Burns and Roe.

c. Purpose/Objectives of Investigation

On the morning of March 28, 1979, Unit 2 of the Three Mile Island Nuclear Station experienced an operational transient which evolved into an emergency situation requiring activation of the Three Mile Island Site Emergency Plan.

The scope of this investigation was described by the Acting Director, Office of Inspection and Enforcement, in memoranda to the Commission dated April 20 and June 8, 1979. These memoranda are included in Enclosure 1 to this report. The stated objectives of this investigation were:

1. To gather facts concerning the incident, its cause, effect(s), and the licensee's response; and
2. To evaluate these facts as a basis for corrective or enforcement action, as appropriate.

d. Scope of Investigation

The operational portion of this investigation examined selected preaccident conditions for a period prior to March 28, 1979, and operational events during the period 0400 hrs to 2000 hrs on March 28, 1979. Areas investigated included: operations staff qualifications and training, operating events, operator actions and management actions.

The radiological portion of this investigation examined selected preaccident conditions and emergency response activities during the period 0400 hrs on March 28, 1979, to 2400 hrs on March 30, 1979. Areas investigated included: detection and classification of the emergency; emergency organization activation; notifications; environmental assessment and protective actions; effluent

monitoring and corrective actions; and in-plant radiological assessment and protective actions.

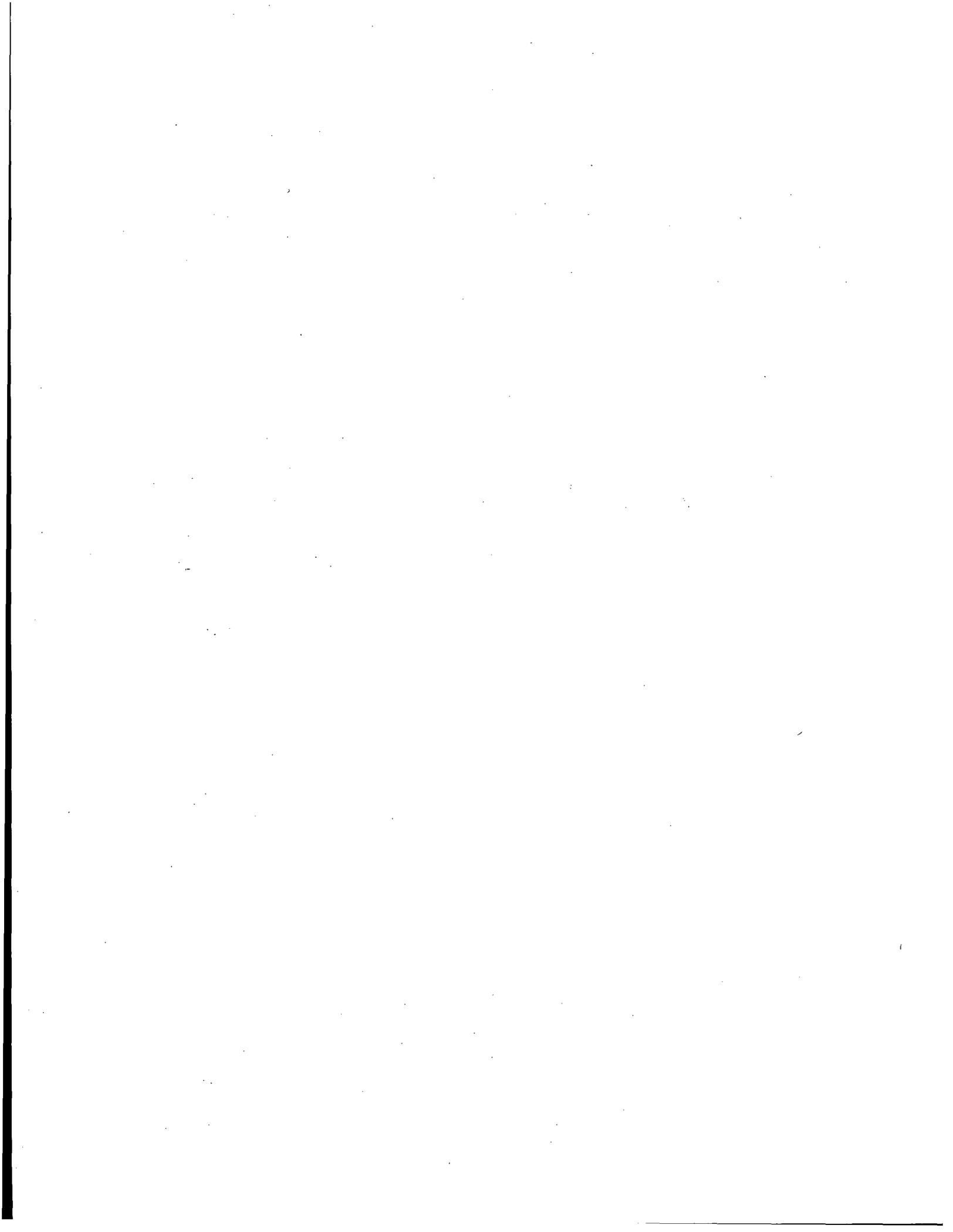
This investigation did not include an engineering evaluation of the accident. Such evaluations are being prepared by others.

e. Sources of Information

Investigators examined records, conducted interviews of and discussed matters with involved personnel, and made observations of facilities and equipment. In the radiological area, scarcity of records caused greater reliance on other sources of information, especially interviews. In making conclusions, the investigators have attempted to reconcile conflicts by using the best available information. A listing of those interviews is included in Enclosure 2 to this report.

f. Noncompliance

Several matters which are under consideration as potential items of noncompliance were identified during this investigation and are listed in Appendices I-B and II-F of this report. These matters are under review and will be handled through the enforcement channels of the Office of Inspection and Enforcement.



TMI INVESTIGATION

SUMMARY OF OPERATIONAL ASPECTS

PREACCIDENT CONDITIONS

On March 28, 1979, during the first 5 hours of the 11-7 shift (2300 hours, March 27, 1979 to 0700 hours, March 28, 1979), the Three Mile Island Unit 2 facility was operating at approximately 97% power with the Integrated Control System in full automatic. Normal makeup, reactor coolant pump seal injection, and letdown were in operation. The reactor coolant system (RCS) boron concentration was 1026 ppm, with the pressurizer spray throttled open and pressurizer heaters energized to equalize RCS and pressurizer boron concentration.

All system and core physics surveillance testing required by Technical Specifications was current, and the facility was in one identified Limiting Condition for Operation ACTION statement of those specifications. The borated water storage tank to spent fuel pool isolation valve (DH-V157) was open to permit BWST recirculation. This ACTION statement time limit would have expired at 1500 hours on March 29, 1979. The RCS leakage as calculated by the licensee was within Technical Specification limits, with the identified leakage being that accumulated in the reactor coolant drain tank (RCDDT). This leakage was identified as being from the electromatic relief valve (EMOV) and/or one or both pressurizer code safety valves. A review by the investigators of the RCS leakage procedure showed the procedure to be in error, and the facility was actually operating with an unidentified leakage in excess of Technical Specification limits.

The assumption by the licensee staff of EMOV and safety valve leakage appears reasonable based on EMOV and safety valve discharge pipe temperatures. The leakage was sufficient to cause the temperature of the discharge pipes ($\sim 200^{\circ}\text{F}$) to be in excess of specified limits (130°F) in plant procedures, and the facility had been operating contrary to these limits for an extended period of time.

The operating staff on duty in the control room during the 11-7 shift of March 28, 1979, was in accordance with Technical Specifications. Each licensed staff member was satisfactorily current with regard to the requirements of the licensee's requalification program for licensed operators. Shift conditions as regards personnel behavior and activities were normal, and the only unusual log book entries indicated an increase in the amount of water being added to the makeup tank when compared to that of previous shifts. No evidence was found showing any maintenance on safety related components was in progress at that time.

The shift foreman and two auxiliary operators were engaged in transferring resin from condensate polisher tank No. 7 to the resin regeneration tank. This activity was a carryover from the previous shift, and a total of about 11 hours had already been expended attempting to complete this transfer. Difficulty was being encountered in this transfer and was attributed by licensee staff to a resin blockage in the transfer line, and shift personnel were involved in an attempt to clear it.

TURBINE TRIP AND PLANT RESPONSE

At this time, and probably as a result of actions taken to clear the resin blockage in the transfer line, the plant experienced a total loss of feedwater initiated by a loss of condensate flow with an almost simultan-

eous trip of the main turbine at 04:00:37. All emergency feedwater pumps started as designed, the reactor continued to operate at full power in accordance with its protection system design, and RCS temperature and pressure increased for approximately 8 seconds. The EMOV opened as designed at its setpoint of 2255 psig. The reactor automatically tripped when the high RCS pressure trip setpoint was reached.

With the trip of the reactor, the RCS experienced an expected coolant contraction, loss of inventory, cooldown, and the attendant reduction in RCS pressure. The EMOV failed to close when its closure setpoint was reached about 13 seconds later. This failure was not recognized by the operating staff for more than 2 hours. At approximately one minute after the start of the accident, the pressurizer level stopped decreasing and began to rise. This rise continued until approximately 6 minutes after the accident, when the level went off scale indicating that the pressurizer was completely filled with water (a "solid" pressurizer). Operator efforts to control the level of the pressurizer, included throttling high-pressure injection which initiated automatically at 2 minutes, and increasing letdown flow to the maximum extent possible. These efforts were largely unsuccessful. RCS pressure began to increase moderately as the pressurizer went solid. At this time, the RCS temperature was also increasing. This increase would also contribute to the pressure rise since saturation conditions now existed in the loops.

This indication of high pressurizer level was caused by voids, either discrete or distributed, that formed in the reactor coolant system coupled with the open EMOV. The open EMOV vented the steam space of the pressurizer causing a rapid insurge into the pressurizer. At 8 minutes into the accident, an operator, upon seeing continuing low once-through-steam-generator (OTSG) levels and decreasing OTSG pressures, searched his panels for the cause.

The block valves on the emergency feedwater headers were found closed. Upon opening the block valves, which admitted emergency feedwater to the OTSG, a rapid cooldown of the RCS and corresponding RCS pressure decrease occurred.

At approximately 14 minutes, the Reactor Coolant Drain Tank (RCDT) rupture disc burst, discharging water and steam into the reactor building and causing a further increase in building pressure (which had begun with lifting of the RCDT relief valve) from the continued discharge of reactor coolant into the open RCDT.

Reactor coolant inventory loss continued with the RCS under saturation conditions. This continuing loss was caused by the discharge of coolant to the reactor building through the open EMOV, coupled with sustained low rates of coolant injection. Reactor coolant pump (RCP) apparent output flow rate decreased while they continued to be operated outside pressure operating limits. The staff secured the RCPs in the B loop at 74 minutes, and the A loop RCPs at 101 minutes, with the staff expecting that natural circulation would occur. However, the plant parameters were outside defined pressure/temperature limits for natural convection. After the trip of the B loop RCPs, the operating staff believed that the B OTSG had developed a secondary-to-containment leak and this generator was isolated.

By this time, after receiving initial early notification of the trip, plant management had become aware of the worsening situation and called for key individuals to come to the site. RCS pressure continued to decrease and temperature increased as a result of the failure of natural circulation to develop because both loops were vapor bound; the lack of any other adequate heat sink being available to accommodate the core decay heat; the continuing unrecognized reactor coolant loss through the EMOV; and the

throttled high-pressure-injection flow. This throttled high pressure injection flow was justified by the operators because of the apparently satisfactory but actually misunderstood pressurizer level. The RCS pressure decreased to a low point of 660 psig at 2 hours and 19 minutes when the leaking EMOV was diagnosed and closed, and RCS pressure began to increase. The pressure increase appears to be associated with the heating of the remaining RCS contents by core decay heat, a change in RCS inventory made by modest increases in high pressure injection, and a zirconium-water reaction. The operating staff believed up to this time, that no substantive inventory loss from the RCS had occurred. This belief was based on the misunderstood pressurizer level, without regard for the low RCS pressure.

At approximately 2-1/2 hours into the accident, substantial fractions of the reactor core were uncovered and had experienced sustained high temperatures. This condition would be expected to result in fuel damage, substantial releases of core fission products, and hydrogen generation. The magnitude of these conditions were not recognized by the plant staff.

With the arrival of senior management, the declaration of a General Emergency, an emergency command team was established with the Station Manager as Emergency Director. Additional unsuccessful attempts to establish sustained forced cooling with one or more RCPs were made. The plant staff was faced with the following conditions:

- o an inability to achieve forced or natural circulation in the RCS
- o high incore and loop temperatures, which were considered to be too high to be realistic
- o an apparent inability to collapse the voids in the loops despite the increased system pressure and the high pressure injection flow which by now was increased.

Unsuccessful efforts to collapse the voids in the loops were continued for approximately 2 hours at about 2000 psig. Because of a growing staff concern over the ability of the EMOV block valve to continue to remain functional under a high use rate, the decision was made to reduce system pressure and float the core flood tanks on the RCS as an assurance of adequate core coverage and as a preliminary step in initiating the use of the decay heat removal system. This depressurization was accomplished in approximately one hour, using the EMOV flow path to the reactor building, and the RCS was held in this low-pressure condition for the next 5 hours.

The failure of the core flood tanks to inject a substantial fraction of their volume was interpreted as an indication that the core was covered. The piping from these tanks contains large loop seals that prevent them from being used effectively for the purpose of ensuring satisfactory core coverage. The design function of these tanks is to supply water to the vessel in the event of a large break LOCA, which did not occur during this accident.

The extended period of low pressure appears to have assisted in the release of hydrogen gas from the RCS. This hydrogen resulted from a significant metal-water reaction with the zirconium fuel cladding. Some of this gas burned in the reactor building at about 10 hours after the accident producing a rapid pressure spike (28 psig) in the containment. This pressure spike received relatively little attention from the majority of the plant staff, with many of them being unaware that it had taken place. However, it is also possible that the release of this noncondensable gas from the RCS contributed to the later apparent success of the staff in collapsing the voids in at least one of the reactor loops (A loop, to which the pressurizer is connected). This increasing success in establishing what appeared to be some degree of natural circulation, despite continuing high temperatures in

portions of the system, led the plant staff to conclude that they had achieved a reasonably stable set of conditions.

The Station Manager left the Emergency Control Center (Unit 2 control room) at approximately 1400 hrs (10 hours after the start of the accident). He was absent for approximately 2-1/2 hours to attend a meeting with the Lt. Governor of Pennsylvania. During his absence, another staff member acted as Emergency Director under additional general guidelines dictated by the Station Manager.

After the return of the Station Manager, the plant staff was directed by corporate management to take the RCS to high pressure to collapse the remaining voids. During this final repressurization, the decision was reached to attempt another start of a reactor coolant pump to establish forced circulation. This was successfully achieved at 1950 hours, on March 28, 15 hours 50 minutes after the start of the accident.

SHIFT CREW ACTIONS

When the original turbine trip occurred, the shift crew on duty took the appropriate initial response actions indicated for a combined turbine trip/reactor trip initiated as a result of a loss of main feedwater. These actions included control manipulations, verifications of automatic actions, and notifications of appropriate personnel.

The misunderstood pressurizer level, and the conditioning instilled in the operators by their training and experience to avoid a solid pressurizer condition at all times caused the shift crew, and those who responded early in the transient to provide assistance, to take a series of actions that were contrary to procedural requirements and/or to prudent operating practices.

These actions led directly to a sufficient loss of reactor coolant inventory to cause core damage. For a period of as much as 2 hours, a reversal of these actions could have prevented the extensive core damage that occurred, although some degree of damage may still have been experienced.

Among the actions taken that contributed to the accident were:

- o The throttling of high pressure injection to a minimum, averaging only 70 gpm net input to the RCS for the first 3 1/2 hours of the accident;
- o The continued operation of RCPs at RCS pressures below the procedural requirement which requires they be tripped which maintained a water supply at the pressurizer surge line and resulted in a sustained higher mass flow rate through the EMOV;
- o The failure to isolate the EMOV after the RCS pressure continued to fall, the RCDT rupture disc had blown, and the reactor building sump pump operation indicated a large discharge of water from the building;
- o The failure to establish the conditions for natural circulation when the combined RCS pressure and temperature conditions were outside the procedural requirements.

Other actions were taken by the shift crew members during the early hours of the accident that did not directly contribute to the accident, but would have severely impaired the response of safety-related equipment had other plant conditions developed. Among these were:

- o Disabling the automatic start features of the emergency diesel generators making them unavailable for rapid starting in the event of a power failure during the course of the accident. (This condition was noted after plant management arrived, but was only partially corrected by restoring control room start capability.)

- o Isolating the core flood tanks early in the event so they were not available to discharge their contents into the vessel. The RCS pressure dropped to within 60 psi of the core flood tank pressure just before the EMOV was isolated, and the core flood tanks apparently had been isolated prior to this time based on the continuing belief of the plant staff that no loss in inventory had occurred.

EMERGENCY STAFF ACTIONS

With the arrival of plant management and the establishment of an Emergency Organization, one of the initial actions taken was to increase high-pressure-injection flow rates to allow ECCS to function as it would if operators were not present. This apparently resulted in eventual reflooding of the core. The actions taken over the next 13 hours that eventually led to the successful operation of one of the reactor coolant pumps (RCP-1A) have been summarized earlier.

The Emergency Director (the Station Manager) formed a management team for overall conduct of the emergency by assigning specific individuals responsibility for different functional areas. A system of periodic meetings with that team for status review and decision-making was established. Decisions were ultimately made by the Emergency Director following consultation with that team, with input from offsite management. Team members then conveyed the decision to the plant staff for implementation.

Within hours of its formation, the management team found themselves between two desired stable states of forced circulation, being unable to use the RCPs at high pressure or the decay heat removal system at low pressure. Natural circulation was similarly unattainable because of vapor binding in the loops.

Their efforts throughout the course of the accident were to move toward one or the other of these desired conditions before the borated water storage tanks (BWST) inventory was exhausted and they would be forced to use the water on the reactor building floor.

Plant parameter information was utilized by the team in planning courses of action to move toward either of these desired conditions with several notable exceptions:

- o The persistent disbelief of high temperature data from incore thermocouples and system RTDs. This was based on the rationale that the former were not safety-grade equipment, while the latter were outside the calibrated range of the detectors;
- o The failure to recognize the fact that a full pressurizer did not provide assurance of core coverage;
- o The failure to recognize the significance and pursue evidence of the pressure spike that occurred in the reactor building;
- o The failure to recognize the fact that small decreases in core flood tank level did not provide assurance of core coverage;

For each of the above areas, the investigation did not attempt to conclude whether the course of subsequent events would have or could have been altered.

In the case of the high temperatures, acceptance of the temperature data as valid might have prompted a higher high-pressure-injection flow rate and a reluctance to subsequently depressurize the plant to use the core flood tanks. However, had that occurred, it cannot be ascertained whether RCP operation could ever have been established in light of the then unrecognized inventory of noncondensibles (hydrogen) that was in the loops and reactor vessel as a result of the zirconium-water reaction.

Similarly, the general recognition of the pressure spike in the reactor building might have led the Station Manager to conclude that conditions were not sufficiently stable to justify leaving the site. His remaining on the site might have altered the subsequent actions taken, or the timing of those actions.

OFFSITE TECHNICAL SUPPORT

The provision of substantive technical support to the management team directing emergency actions on operational matters suffered primarily as a result of communication difficulties. This was evidenced in three ways:

- o Information (both data and plans) transmitted to offsite support, which had been hurriedly mobilized, suffered from time delays. Thus, the offsite groups were dealing with historical and limited data.

- o The individuals who had to provide data to offsite groups had concurrent duties pertaining to the management of the emergency. The emergency duties always took precedence as would be appropriate.
- o The physical communications facilities were inadequate to handle the volume of information requests and transmittals that this kind of accident required.

The investigation has concluded that these communication problems are related to the misconception that the envelope of the analyzed major accidents for this facility are the limiting events. The duration of these analyzed events are projected to occur in a relatively short time frame. The provision of the mechanisms needed to mobilize and communicate with substantial offsite technical support on a real-time basis as an accident progresses had, therefore, not been warranted as a part of emergency planning.

SUMMARY OF RADIOLOGICAL ASPECTS

Health physics operations at TMI Unit 2 were routine prior to 0400 hrs March 28, 1979. The normal complement of radiation protection staff was on site. The emergency plan and implementing procedures had been rehearsed and evaluated during seven drills conducted in the past year. Most plant personnel had received training in their emergency plan duties. However, some workers who would comprise Emergency Repair Party Teams and Radiological Monitoring Teams had not received adequate training in use of emergency survey instrumentation and in radiation protection procedures. Routine retraining of radiation/chemistry technicians was not up to date. While radiation protection training of the plant staff had been sufficient to maintain personnel radiation exposures within limits during normal operations (when radiation levels were low), it had not prepared workers to cope with the high radiation levels that would soon exist inside the Unit 2 auxiliary and fuel handling buildings.

Less than half of the portable radiation survey instruments were operational. Several installed area radiation monitors and airborne radioactivity monitors, which were not essential for normal operations, but would have been useful during the emergency, were out of service for repair.

Fifty self-contained breathing devices and 175 half and full-face respirators were on site. Large quantities of protective clothing were available. All essential communications systems were operational. Three emergency environmental monitoring kits containing survey and counting instrumentation and personnel monitoring devices were in place. One of the three kits was later found to have an inoperable instrument for field measurement of radioactive iodine. Environmental air samplers were operating at eight offsite locations, and environmental TLDs were in position at 20 locations. Tanks in the liquid radwaste system were filled to about 60% of capacity. Valves were aligned to pump the reactor building sump to the auxiliary building sump tank. Ventilation exhaust from fuel handling and auxiliary buildings was through high-efficiency filters and charcoal adsorbers.

At 0400 hrs, the Unit 2 turbine and reactor tripped. At 0411 hrs, there was a reactor building sump high water level alarm. By 0415 hrs, the reactor coolant pressure had dropped from 2435 psig at the time of the reactor trip to approximately 1275 psig. This pressure was below the setpoint for emergency core cooling system initiation (1600 psig). At 0415 hrs, there was a pressure rise of 1.4 psig inside the reactor building. A site emergency should have been declared, based on these indications and criteria in the Site Emergency Plan. However, because the drop in reactor pressure was believed to be under control, and the reactor building pressure increase was considered to be slight, and because there was no evidence of a release of radioactivity from the station, an emergency was not declared. Subsequently, there were several radiation monitor alarms indicative of an emergency situation, but no emergency was declared.

At 0622 hrs the first radiation monitor response to cladding failure occurred. Radiation levels continued to increase and a site emergency was declared at 0655 hrs based on these alarms.

The emergency organization was promptly activated following the declaration of a Site Emergency. The Station Manager arrived in the Unit 2 control room at 0705 hrs and relieved the Shift Supervisor as Emergency Director. Initially, the emergency organization approximated the planned organization described in the TMI Emergency Plan. An exception was that Repair Parties were assembled and controlled by both the Emergency Control Center (ECC) in the Unit 2 control room and the Emergency Control Station (ECS) in the Unit 1 health physics/ chemistry lab area. According to Emergency Plan Implementing Procedures, the Repair Party was to assemble only at the ECS, under the direction of the Supervisor of Maintenance, and coordinated through the Supervisor of Radiation Protection.

Offsite consequences were assessed by performing dose rate calculations. Because of errors in these calculations, the dose rates initially predicted (10 and 40 rem/hr at Goldsboro) were higher than actual dose rates. Radiation measurements by survey teams revealed actual doses were

low (less than 0.001 rem/hr at Goldsboro). Offsite agencies and support groups were notified of the Site Emergency by telephone. At 0724 hrs, a General Emergency was declared based on radiation levels inside the reactor building. Again, offsite agencies and groups were phoned.

Following the turbine trip, about 8000 gallons of reactor coolant were pumped from the reactor building sump to the auxiliary building sump tank. This transfer was terminated at 0438 hrs and was not resumed. The auxiliary building sump tank overflowed to the auxiliary building sump, causing water containing a relatively low concentration of radioactivity to back up through floor drains onto the fuel handling building and auxiliary building floors. Following fuel damage, the concentration of radioactivity in the reactor coolant increased by several orders of magnitude. A flow of this highly contaminated reactor coolant was maintained through the makeup and purification system for several days following the accident. This flow was the principal pathway by which radioactivity was transferred from the damaged reactor core to the auxiliary and fuel handling buildings, and ultimately to the environment.

Gases evolving from reactor coolant in the makeup and purification system were collected in the waste gas system. Small leaks in these systems were of little radiological significance during normal operation. However, following fuel damage, radioactive gas leaks caused very high concentrations of airborne radioactivity inside the auxiliary and fuel handling buildings and resulted in much higher than normal environmental releases via ventilation exhausts from these buildings. Radiation levels in the vicinity of some makeup and purification system components exceeded the limits of the licensee's measurement capability (i.e., greater than 1000 R/hr). High radiation levels inside the Unit 2 auxiliary building caused full scale readings on several station effluent monitors. A full scale reading for the plant vent gas monitor is equal to $2.8 \text{ E-}2 \text{ } \mu\text{Ci/cc}$ of xenon-133. The particulate and iodine monitors were off-scale due to interference from the large amounts of radioactive noble gases.

As hazards from direct radiation and airborne radioactive material developed rapidly in the Unit 2 auxiliary and fuel handling buildings, the licensee attempted to control the in-plant radiation protection program in accordance with Emergency Plan Implementing Procedures.

The ECS was established in the Unit 1 chemistry and health physics area according to Emergency Plan Implementing Procedures. The activities of Radiological Monitoring and Repair Party Teams were to be directed by the Supervisor, Radiation Protection (ECS Director) from this location.

A Unit 2 reactor coolant sample was collected in the nuclear sample room at about 0845 hrs without the knowledge of the ECS Director. The nuclear sample room and primary chemistry laboratory are located in Unit 1 near the ECS. Collection of this sample resulted in an immediate increase in radiation and airborne radioactivity levels at the ECS, causing the ECS to be evacuated to the Unit 2 control room. The high radiation levels disabled the Unit 1 counting room, which contained the only instrument on site capable of performing gamma isotopic analyses.

The individuals who collected and analyzed this sample did not take appropriate precautions. Sample containers were handled directly without use of remote tools or shielding to reduce hand exposure, extremity dosimetry was not worn on hands, and no air sample was collected. If the sample lines had been properly recirculated or flushed prior to sampling, the individuals would likely have received significantly greater radiation exposure.

Shortly after the ECS was established in the Unit 2 control room, airborne radioactivity began to increase, as measured by the control room incoming air monitor. At about 1017 hrs, personnel were requested to put on respiratory protective devices (particulate filter masks), based on an alarm of the control room air monitor and an air sample that indicated high gross beta radioactivity. Control room personnel remained in respiratory protective devices for about six hours. Isotopic analysis of an air sample

would have likely shown that respirators were unnecessary; but, the isotopic analysis capability had been lost.

The ECS was relocated to the Unit 1 control room at 1012 hrs, maintaining the responsibility for coordination of the onsite and offsite environmental survey teams but relinquishing control of the inplant radiation protection program to the Supervisor, Radiation Protection and Chemistry who remained in the Unit 2 control room.

At 1110 hrs, all nonessential personnel were evacuated from the site. Evacuees were surveyed for contamination at the assembly areas, exit gates, and at an area established offsite at the 500 kV substation. Several individuals were found to be contaminated.

During the evacuation, the auxiliary building access control point was relocated from outside the auxiliary building entrance to the Unit 2 control room because of increasing airborne radioactivity in the auxiliary building. This left no positive control over entries into the auxiliary building.

Although the Supervisor, Radiation Protection and Chemistry briefed some individuals and, at times, directed radiation/chemistry technicians to accompany Repair Party Teams into the auxiliary building, several entries were made without his knowledge. These entries were made into areas of high airborne radioactivity and whole-body exposure rates in excess of 100 R/hr. In at least one instance, survey instruments were not used. Two individuals who entered the auxiliary building received a whole-body dose of radiation in excess of a regulatory limit; others became contaminated and received unnecessary doses. At times, high-range pocket dosimeters could not be located and were not worn. Items of protective clothing such as hoods, when not readily available, were not worn, resulting in several instances of head contamination. Extremity monitoring devices were not worn. Air sampling was not performed in the auxiliary building, in the and where workers were exposed during the period from about 0900 hrs on March 28 through midnight on March 30. Appropriate respiratory protective devices were not always worn.

In at least two instances, individuals failed to leave high radiation areas in the auxiliary building when their radiation survey instruments failed or deflected full scale. In one of these instances, this resulted in a whole-body exposure in excess of regulatory limits.

An example which indicates that the radiation protection and chemistry staff was not adequately trained to cope with the hazards which existed occurred during the sampling of reactor coolant on March 29. The sampling was at the direction of the Supervisor, Radiation Protection and Chemistry, and was performed by a chemistry foreman with assistance from a radiation protection foreman.

Although the need for a reactor coolant sample was known for several hours, less than one hour was devoted to planning and preparation for taking it. The two foremen entered the nuclear sample room to survey the area and make the valve lineup to recirculate the sample at about 1600 hrs March 29. They wore protective clothing, full-face respirators with iodine adsorbing cartridges, and high-range pocket dosimeters. No air samples were taken to evaluate airborne radioactivity, and no one was assigned to time their exposure. Neither remote valve operating nor sample handling tools were used.

The exposure rate to operate a sample valve was remembered to be 90 R/hr. About 300 ml of reactor coolant was collected in a hand-held polyethylene bottle. A 100 ml aliquot of this sample in a graduated cylinder produced a radiation exposure rate of 400 R/hr at a distance of 1 foot.

A second sample was collected in a beaker. A portion was removed and placed in a small vial. The remainder was titrated with hydrochloric acid in preparation for a boron analysis. Another chemistry foreman, wearing a particulate filter respirator and no extremity monitoring, performed the boron analysis.

After the operation, the three individuals were found to be contaminated. Decontamination was incomplete, and residual contamination remained

on small areas of one individual's skin for over 30 days. The licensee reported one chemistry foreman received a whole-body exposure, as measured by his TLD, in excess of NRC limits. The NRC evaluated the handling of this sample and concluded that, in addition to the reported whole-body dose, doses to the hands, forearms, and a small area on the skin of the head of the chemistry foreman and to the hands and forearms of the radiation protection foreman exceeded NRC limits.

Prior to and during the emergency, the licensee performed his own onsite personnel dosimetry program. No one individual was assigned programmatic responsibility for this program. During the incident, some radiation/chemistry technicians processed their own TLD badges. Beginning March 29, one radiation/chemistry technician, who had not operated the system in over a year, worked without procedures for over 40 continuous hours.

The Emergency Plan Implementing Procedures did not address sustained in plant radiation hazards. The licensee's radiation protection and chemistry staff was not adequately trained to deal with this degree of hazard, and supplies of equipment and instruments were not sufficient to minimize dose to the workers.

During March 28-30, the licensee's land-based onsite and offsite monitoring teams made about 500 direct radiation measurements. These measurements were made primarily to confirm the predicted location of the noble gas effluent plume and to determine the dose rate produced by the plume. The rate of release of radioactivity (source term) from the station was periodically calculated based on dose rate measurements in the plume and meteorological conditions existing at the time of measurement. The calculated source terms were used to predict dose rates in other areas when meteorological conditions changed. Monitoring team survey results were also used to assess the need for protective actions and to supplement thermoluminescent dosimeter (TLD) results in assessment of accumulated dose. These dosimeters were in place at 15 locations within 3 miles and at 5 locations ranging from 9 to 15 miles from the site prior to the accident.

These TLDs were used to perform an after the fact assessment of direct radiation doses to the public.

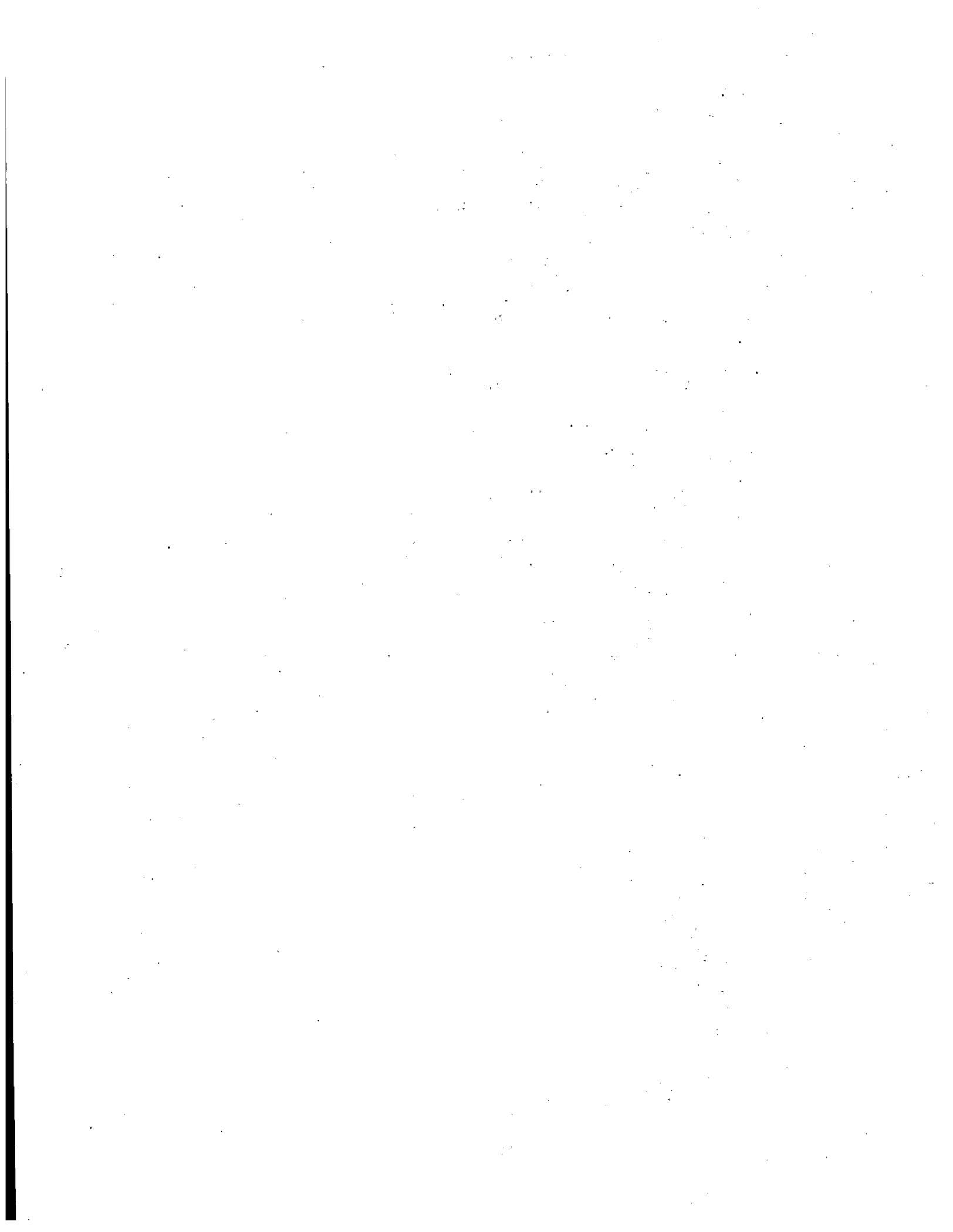
In general, the licensee's onsite and offsite survey teams performed surveys in appropriate areas at appropriate times. However, during a five and one-half hour period from 1700 hrs to 2238 hrs on March 28 and a two-hour period from 0340 to 0540 on March 29, no offsite surveys were performed in the plume. Both of these periods of time were within the interval when the majority of the noble gases were released and when a plume was well defined because of sufficient wind speed and almost constant direction.

Radiation levels on March 28, with the exception of 50 mR/hr measured at 1548 hrs on Pennsylvania Rt 441, about 1500 feet south of the North Gate, were not above background until 2238 hrs when a radiation level of 13 mR/hr was measured near Kunkel School (5.6 mi NNW). Several other radiation levels above background were noted in this general area prior to midnight. However, the 13 mR/hr value was the highest one measured, until 30 mR/hr was measured in Goldsboro at 0600 hrs on March 29. Radiation levels during the remainder of March 29 were generally less than 1 mR/hr, with the maximum noted as 3 mR/hr in Royalton at 2355 hrs. Offsite radiation levels measured on March 30 were also generally below 1 mR/hr, with the maximum noted as 15 mR/hr at location S-11 (one mile south of the plant) at 0906 hrs. The highest radiation level measured onsite (outside of the plant) during March 28-30 was 365 mR/hr (β , γ) at 2325 hrs on March 28 at a location 1,000 feet northwest of the Unit 2 station vent.

Although not a pre-planned consideration in the licensee's Emergency Plan, helicopter-based survey teams were used to track the noble gas plume. Up to three helicopters chartered by the licensee were used during March 28-30, with the majority of surveys taking place on March 30. Over 300 radiation measurements were made by the helicopter teams. The highest measurements reported were 3000 mR/hr (β , γ) at 15 feet above the plant vent at 1410 hrs on March 29 and 1200 mR/hr (β , γ) at 130 feet above the Unit 2 reactor building at 0801 hrs on March 30.

A total of 57 samples were collected on March 28-30 for the purpose of assessing radioiodine concentrations in the environment. Air samples collected on March 28, which were counted in the field with a single channel analyzer having a sodium iodine detector, indicated that "radioiodine" was present at offsite locations with concentrations ranging up to $2.3 \text{ E-7 } \mu\text{Ci/cc}$. The "radioiodine" was subsequently shown to be xenon-133 and xenon-135 at 1400 hrs on March 28 (the time at which the first gamma spectrometry of one of these samples was completed by the Pennsylvania Bureau of Radiological Health). Forty of the fifty-seven samples collected were analyzed by gamma spectrometry, and no radioiodine was detected.

Results of samples from certain portions of the licensee's routine radiological environmental monitoring program collected on March 29 (TLDs, radioiodine in water, and radioiodine in air) were available around mid-day on March 30. The sample results confirmed that the offsite radiological impact was no worse than earlier estimates made using data gathered by the monitoring teams. These data supported the conclusion that radioactive noble gases released to the atmosphere were the principal cause of exposure for individuals in the plant environs.



DETAILS I
OPERATIONAL ASPECTS
TABLE OF CONTENTS

	<u>Page</u>
TABLES LIST.....	I-vii
FIGURE LIST.....	I-viii
1. BACKGROUND.....	I-1-1
1.1 General.....	I-1-1
1.2 Plant Status Prior to Turbine Trip.....	I-1-1
1.2.1 General.....	I-1-1
1.2.2 RCS Boron Concentration.....	I-1-2
1.2.3 RCS Leakage.....	I-1-2
1.2.4 EMOV Leakage.....	I-1-5
1.2.5 Condensate System.....	I-1-7
1.2.6 General Plant Parameters.....	I-1-7
1.3 Status of Surveillance and Inspection.....	I-1-9
1.3.1 General Status.....	I-1-9
1.3.2 Status of Surveillance on Selected Engineered Safety Features (ES) System Components.....	I-1-10
1.3.2.1 Review of Procedures.....	I-1-10
1.3.2.2 Emergency Feedwater Valve Test.....	I-1-11
1.3.2.3 Review of EFW Procedure Change.....	I-1-14
1.3.3 Status of Core Physics Surveillance.....	I-1-16
1.3.4 Status of Surveillance Results and Records Review.....	I-1-19

	<u>PAGE</u>
1.3.4.1 Performance and Review.....	I-1-19
1.3.4.2 Record Retention.....	I-1-21
1.3.5 Inspection of Surveillance Activities.....	I-1-21
1.4 Operations Staff.....	I-1-22
1.4.1 Operations Personnel Training and Qualifications.....	I-1-22
1.4.1.1 General Plant Training Program.....	I-1-22
1.4.1.2 Auxiliary Operator Training.....	I-1-23
1.4.1.3 CRO Training.....	I-1-24
1.4.1.4 Training for Individuals with Previous Experience.....	I-1-25
1.4.1.5 Requalification Program.....	I-1-25
1.4.1.6 Training On Significant Events.....	I-1-27
1.4.1.7 Annual Examinations.....	I-1-28
1.4.1.8 Individuals Involved With the Early Part of the Accident.....	I-1-29
1.4.2 Shift Composition Compared to Technical Specifications.....	I-1-32
1.4.3 Shift Staffing During Prior Trips.....	I-1-33
1.4.4 Physical and Emotional Condition of Operational Staff.....	I-1-33
1.4.5 Possibility of "Anniversary" Celebration on Site.....	I-1-34
1.4.6 Possibility of Plant Sabotage.....	I-1-35
1.5 Significant Operating Events.....	I-1-37
1.5.1 Summary of Selected Licensee Reports.....	I-1-38
1.5.2 Review of Selected Prior Trips.....	I-1-44
1.5.2.1 Previous Event Involving the Electromatic Relief Valve (EMOV).....	I-1-44
1.5.2.2 Previous Loss of Feedwater Transient.....	I-1-44
1.6 Corrective Maintenance Practices.....	I-1-47

1.6.1	General.....	I-1-47
1.6.2	Selected Component Maintenance History	
	Immediately Prior to March 28, 1979.....	I-1-49
	1.6.2.1 Main Feedwater System.....	I-1-49
	1.6.2.2 Electromatic Relief Valve.....	I-1-49
	1.6.2.2.1 Valve Control Revisions....	I-1-49
	1.6.2.2.2 EMOV Locations and	
	Handling.....	I-1-51
	1.6.2.3 EMOV Block Valve.....	I-1-52
	1.6.2.4 Emergency Feedwater Pumps.....	I-1-53
	1.6.2.5 Pressurizer Heaters.....	I-1-53
	1.6.2.6 Waste Tank Rupture Discs.....	I-1-54
	1.6.2.7 Emergency Feedwater Valves.....	I-1-54
	1.6.2.8 Pressurizer Safety Valves.....	I-1-56
	1.6.3 References.....	I-1-56
2.	OPERATOR ACTIONS DURING ACCIDENT.....	I-2-1
2.1	General.....	I-2-1
2.2	Operator Actions Following Loss of Main Feedwater.....	I-2-1
2.3	Operator Actions Following Turbine Trip.....	I-2-2
	2.3.1 Immediate Operator Actions.....	I-2-2
	2.3.2 Followup Operator Actions.....	I-2-5
	2.3.3 Additional Operator Actions.....	I-2-8
2.4	Operator Action Following Reactor Trip.....	I-2-10
2.5	Operator Actions Following ES Actuation.....	I-2-14
2.6	Operator Actions Concerning the Operation of the	
	Makeup System (Following Initial ES Actuation).....	I-2-21
2.7	Operator Actions Concerning Condensate System.....	I-2-27
2.8	Operator Actions Concerning Reactor	
	Building Sump Pumps.....	I-2-29
2.9	Operator Actions Concerning the B OTSG Tube Leak.....	I-2-30
2.10	Operator Actions Concerning Natural Circulation -	
	Decay Heat Removal OTSGS.....	I-2-32
2.11	Operator Actions Concerning the Auxiliary	
	Building Sump Pumps.....	I-2-34

2.12	Operator Actions Concerning Logkeeping.....	I-2-36
2.13	Operator Actions Concerning EMOV.....	I-2-41
2.14	General Location of Operation Personnel.....	I-2-42
2.15	Effect of Training on Operator Actions.....	I-2-47
	2.15.1 Pressurizer Level Versus Pressure Control.....	I-2-47
	2.15.2 Recognition of an Open EMOV.....	I-2-48
	2.15.3 Bypassing of the Engineered Safety Features Actuation System.....	I-2-49
	2.15.4 Shutdown of Reactor Coolant Pumps.....	I-2-49
2.16	Nonlicensed Operator Actions During Accident.....	I-2-50
	2.16.1 General.....	I-2-50
	2.16.2 Activities Prior to Trip.....	I-2-50
	2.16.3 Activities During Accident.....	I-2-51
3.	MANAGEMENT ACTIONS DURING ACCIDENT.....	I-3-1
	3.1 General.....	I-3-1
	3.2 Onsite Operational Activities of Senior Management.....	I-3-1
	3.3 Provision of Technical Support.....	I-3-16
	3.3.1 Onsite Technical Support.....	I-3-16
	3.3.2 Babcock and Wilcox.....	I-3-19
	3.3.3 Corporate Technical Staff.....	I-3-35
	3.3.4 Burns and Roe (Architect Engineer).....	I-3-38
	3.4 Licensee/NRC Interface.....	I-3-39
	3.4.1 Notification of NRC.....	I-3-39
	3.4.2 Communications Between NRC and the Licensee....	I-3-39
	3.4.3 Impact of NRC on Licensee Actions.....	I-3-40
4.	SIGNIFICANT EVENTS DURING ACCIDENT.....	I-4-1
	4.1 Turbine Trip.....	I-4-1
	4.1.1 Apparent Cause.....	I-4-1
	4.1.2 Plant Air Systems.....	I-4-1
	4.1.3 Post Accident Testing.....	I-4-2
	4.1.4 Pump Interlocks and Wiring Errors.....	I-4-4

4.1.5	Evaluation.....	I-4-4
4.2	Closed Emergency Feedwater Block Valves.....	I-4-5
4.2.1	Sequence.....	I-4-5
4.2.2	Cause.....	I4-6
4.2.3	Evaluation.....	I-4-10
4.3	HPI Flow Adjustment.....	I-4-11
4.3.1	Operator Actions Regarding Adjustment of HPI....	I-4-11
4.3.2	Analysis.....	I-4-14
4.3.2.1	HPI Pump History.....	I-4-14
4.3.2.2	Effective Flow Rates to the RCS.....	I-4-18
4.4	Shutdown of the Reactor Coolant Pumps.....	I-4-24
4.5	Attempted and Successful Restarts of the Reactor Coolant Pumps.....	I-4-27
4.6	Closing of the Core Flood Tank Isolation Valves.....	I-4-28
4.7	Use of Core Flood Tanks.....	I-4-30
4.8	Reactor Vessel Level Indication.....	I-4-31
4.9	Sump Pump Operation.....	I-4-32
4.9.1	Sequence.....	I-4-32
4.9.2	Discharge Point.....	I-4-32
4.10	Electromatic Relief Valve Failure.....	I-4-33
4.10.1	History.....	I-4-33
4.10.2	Evaluation.....	I-4-34
4.10.3	Temperature History.....	I-4-34
4.10.3.1	Relief Valve Tailpipe Layout.....	I-4-34
4.10.3.2	Thermocouple Locations.....	I-4-35
4.11	Emergency Director Leaves Site.....	I-4-43
4.11.1	Sequence.....	I-4-43
4.11.2	Precautions Taken.....	I-4-43
4.12	RCS Average Temperature Indication.....	I-4-44
4.13	Plant Computer Records.....	I-4-45
4.13.1	Alarm Status Printout, 05:13:59 to 06:48:08, 3/28/79.....	I-4-46
4.13.2	Alarm Status Printout, 18:48:59 to 19:10:29, 3/28/79.....	I-4-46

4.13.3	Utilities Typer Output, 00:00:00 to 03:24:24, and 20:08:22 to 20:12:*, 3/28/79.....	I-4-46
4.13.4	Analog Trend Recorder Number 2, 3/28/79.....	I-4-47
4.14	Pressure Spike in Containment.....	I-4-47
4.14.1	Sequence.....	I-4-47
4.14.2	Response of Personnel to Pressure Spike.....	I-4-48
4.14.3	Notification of NRC.....	I-4-49
4.15	High Temperatures and Core Response.....	I-4-51
4.15.1	General.....	I-4-51
4.15.2	SRM and IRM Responses.....	I-4-54
4.15.3	Incore Thermocouple Response.....	I-4-54
4.15.4	Incore Flux Detector Response.....	I-4-55
4.15.5	OTSG and Primary Temperature Recorder Display.....	I-4-64
4.15.5.1	Hot-Leg Temperature Behavior.....	I-4-64
4.15.5.2	RCP Suction Temperature Behavior....	I-1-65
4.16	Early Void Formation.....	I-4-72
4.17	Diesel Generator Inoperability.....	I-4-74
4.18	Reactor Coolant Makeup Pump Starting Problems.....	I-4-76
4.19	Operational Usefulness of Computer Alarm Status Printout.....	I-4-78
4.19.1	Timeliness.....	I-4-79
4.19.2	Inaccuracies.....	I-4-79
4.19.3	Misinformation.....	I-4-80
References.....		I-5-1
Appendix IA	Operational Sequence of Events.....	IA-1
Appendix IB	Operational Aspects Potential Items of Noncompliance.....	IB-1

DETAILS I
TABLES LIST

<u>TABLE</u>	<u>DESCRIPTION</u>	<u>SECTION</u>
I.1-1	Recalculation of RCS Leakage Rate	1.2.2
I.2-1	Location of Key Personnel Interviewed	2.14
I.3-1	Notification and Arrival Time of Key Personnel	3.2
I.4-1	HPI Pump History	4.3.2.1
I.4-2	Makeup Flow Through the MU-; V17 Valves and RCP Pump Seals	4.3.2.2
I.4-3	Average Makeup Flows Based on BWST Levels	4.3.2.2
I.4-4	EMOV and Safety Valve Tail Pipe Temperatures	4.10.3.2
I.4-5	Steam Generator and Primary System Temperatures	4.15.5.2
I.4-6	Pressurizer Level Response During First 4 Minutes	4.16
I.4-7	Computer Output Errors	4.19.3

DETAILS I
FIGURE LIST

<u>FIGURE</u>	<u>DESCRIPTION</u>	<u>SECTION</u>
I.4-1	Makeup & Decay Pump Status	4.3.2.1
I.4-2	Water Injected From BWST	4.3.2.2
I.4-3	Emergency Feedwater Pumps - Status	4.3.2.2
I.4-4	Heat Up/Cool Down Curve	4.15.1
I.4-5	Selected Incore T/C Histories from Computer Printouts	4.15.3
I.4-6	" " " " "	4.15.3
I.4-7	" " " " "	4.15.3
I.4-8	" " " " "	4.15.3
I.4-9	Incore Thermocouple Data from Computer at 0834-0836 hrs on March 28, 1979	4.15.3
I.4-10	Incore Thermocouple Data from Computer at 0846-0847 hrs on March 28, 1979	4.15.3
I.4-11	Readings Obtained from Incore Thermocouples at Approximately 4-5 hrs After Start of Accident	4.15.3
I.4-12A	Selected Primary System Temperatures Following Accident	4.15.5.2
I.4-12B	" " " " "	4.15.5.2
I.4-12C	" " " " "	4.15.5.2
I.4-12D	" " " " "	4.15.5.2

1. BACKGROUND

1.1 GENERAL

During the course of this investigation, it appeared that several practices or conditions could have contributed to the initiation or the consequences of the accident. This section of the report addresses only these topics and is not intended to be a complete background description of the Three Mile Island facility.

1.2 Plant Status Prior to Turbine Trip

1.2.1 General

The Three Mile Island Unit 2 (TMI-2) was operating at approximately 97% power (916 MWe) with the the Integrated Control System (ICS) in full automatic in accordance with operating procedure (2105-1.4, "Integrated Control System"; Ref. 58). The normal reactor coolant system (RCS) makeup and reactor coolant pump seal water injection was established with makeup pump (MUP) 1B in service and 70 gpm letdown flow in accordance with normal procedure; (2104-1.2, Makeup and Purification System; Ref. 41).

1.2.2 RCS Boron Concentration

The RCS boron concentration was 1026 ppm (0330 hrs March 28, 1979) with a gross radioactivity concentration of 0.397 uCi/ml in accordance with normal operating procedure. (2304-W1, Borated Water Source, 2304-3D1, RCS-Chemistry and 2304-3D2, RCS Specific Activity; Ref. 59, 60, 61). The pressurizer spray valve (RC-V1) control was in manual and the spray throttled open with the pressurizer heaters energized to equalize the pressurizer and RCS boron concentration in accordance with normal procedure. (2103-1.3, Pressurizer Operation; Ref. 62). This was done to counteract the boron increase in the pressurizer caused by leakage from the pressurizer.

1.2.3 RCS Leakage

One or more of the pressurizer relief valves (EMOV and code safety valves) were leaking into the reactor coolant drain tank (RCDT) at approximately 6 gpm, as determined by completed surveillance procedure (2301-3D1, RCS Inventory, dated March 28, 1979; Ref. 63). This continuous leakage caused the boron concentration to continuously increase in the pressurizer. The relief valve exhaust continuously indicated approximately 180-200°F due to the leakage. One RCDT pump was being operated in manual continuously to cool the RCDT water inventory and transfer the leakage water to the reactor coolant bleed tank (RCBT). Interviews conducted and records review revealed that this condition had existed since the Fall of 1978. The effect of this continuing leakage over a prolonged period on the operation of the EMOV during this accident cannot be evaluated at this time. (Int. 5, 17, 38, 145, 147, 189; 33, 15, 56, 151, 2, 14, 37, 49, 61, 153, 118)

Approximately 2,600 gallons of water were transferred each shift (8 hours) from the RCDT to the makeup tank (MUT) via the RCBT prior to the shift on which the accident occurred as determined by review of the control room log book (2300 hrs March 1979, through 0400 hrs March 28, 1979). During the first 4½ hours of the shift on which the accident occurred, 1,800 gallons were transferred. This review revealed that the transfer of water to the MUT had increased to approximately 3,600 gallons per shift on March 28, 1979, suggesting a substantial increase in leak rate. The RCS leakage behavior for a period (March 22-28, 1979) was reviewed to establish if any other apparent trends could be established prior to the accident. The following data were taken from the completed copies of the RCS Inventory Procedure, 2301-3D1.

<u>Date</u>	<u>Gross Leak Rate (gpm)</u>	<u>Net Unidentified Leak Rate (gpm)</u>
3/22/79	6.73	0.581
3/24/79	6.55	0.160
3/25/79	6.60	0.432
3/28/79*	6.94	0.010

*This measurement performed from 0134 hrs to 0234 hrs on March 28, 1979.

The licensee calculates leak rate in terms of equivalent gpm of water at RCS Operating conditions (i.e., 580°F and 2150 psig) which should be conservative when evaluating results against the requirements of the Technical Specifications. However, the Technical Specifications are silent on the specific basis for comparison. A review of the procedure and Temporary Change Notice (TCN) 2-79-070 to that procedure revealed that the basic procedure is in error resulting in miscalculation of the RCS leak rate. These errors are discussed below.

Review of TCN 2-79-070 showed that changes to the RCDT inventory are appropriately adjusted to RCS conditions for the calculation. However, line 16 of Data Sheet 1 (used when computer is available) and line 29 of Data Sheet 2 (used when hand calculations are performed) incorrectly adds the MUT water additions without correcting those values to gallons equivalent to RCS conditions.

The above results were recomputed. To determine if the omission of this correction leads to substantially different RCS leakage rate calculations. The results are shown on Table I.1-1 on the following page.

The recalculated results, whether using the unidentified leak rate results for hot or cold water, shows the licensee was operating the facility during the March 22-28, 1979, period with the RCS unidentified leakage rate in excess of 1.0 gpm.

TABLE I.1-1

RECALCULATION OF RCS LEAKAGE RATE

<u>Date</u>	<u>Reported Gross RCS Leak Rate</u>	<u>Recalculated Gross RCS Leak Rate</u>	<u>Unidentified RCS Leakage*</u>	<u>Unidentified RCS Leakage**</u>
3/22/79	6.73	8.05	1.90	1.36
3/24/79	6.55	7.91	1.52	1.09
3/25/79	6.60	8.60	2.44	1.75
3/28/79	6.94	8.94	2.01	1.44

I-1-4

* Calculated as gpm at RCS conditions

** Calculated as gpm at 120 degrees F conditions

Evaluation

Technical Specification 3.4.6.2 and surveillance procedure 2301-3D1 require that RCS unidentified leakage be limited to 1 gpm. Operation of the unit during the period March 22-28, 1979, with an unidentified leakage rate in excess of 1 gpm is under consideration as a potential item of noncompliance.

1.2.4 EMOV Leakage

Emergency Procedure 2202-1.5, Pressurizer System Failure, Section A.2.B.1, requires that electromatic relief isolation valve RC-V2, be closed when the following symptoms exist:

Relief valve discharge line temperature exceeding the normal 130°F.

NOTE: The relief valve (RC-R2) discharge temperature was approximately 180°F at the time of the accident. A review of licensee records revealed that no temporary procedure change had been issued to alter this established procedural requirement.

RCDT pressure above normal on the control room radwaste disposal control panel, and the temperature above normal on the local radwaste disposal control panel.

The operators were operating the RCDT transfer pump with flow through the RCDT cooler continuously to maintain the RCDT temperature at ambient conditions with the apparent valve leakage into the tank. The continuous operation of the drain pump in order to maintain the tank temperature at ambient conditions indicated an abnormal condition existed.

RCS makeup flow above normal for the variable letdown flow and RC pump seal in-leakage conditions.

This was indicated by the frequent transfer of reactor coolant between the RCDT and the MUT via the reactor coolant RCBT. Any unusual changes in the MUT level as occurred during the early hours of the shift is the initial indication of a change in the RCS leakage.

Additionally, Emergency Procedure 2202-1.5, Section C.3.3, requires the code relief discharge line temperature to be placed on the analog trend recorder when the discharge line temperatures exceed the computer normal (130 °F). The valve discharge line temperatures were approximately 180 °F prior to the incident, and the code valves were not being trend-recorded as required. The points on the trend recorders as indicated by the computer (group 6) Analog Assignment Summary included and review of the analog trend charts at 0400 hrs on March 28, 1979 included:

EVALUATION

The EMOV block valve (RC-V2) was not closed and the Code Relief Valve Discharge Temperatures were not placed on the Analog Trend Recorders with leakage from the EMOV (RC-R2) and/or both the code relief valves (RC-RV1A and RC-RV1B) on the morning of March 28, 1979.

Technical Specification 6.8.1.a requires written procedures shall be implemented, covering the activities referenced as applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972, "Malfunction of Pressure Control System." The failure to close the EMOV (RC-R2) block valve (RC-V2) and place the Code Relief Valve Discharge Temperatures on the Analog Trend Recorders is under consideration as a possible item of noncompliance.

ATR 1	Condensate Hotwell Level
ATR 2	Condensate Storage Tank IA Level
ATR 3	T-G Bearing No. 2 Vibration
ATR 4	SGFP Suction Header Pressure

1.2.5 Condensate System

The condensate system reject line isolation valve (CO-V60) from the hotwell to the condensate tanks (CO-T-1A/1B) was normally partially closed by the operators to limit the transient on the condensate booster pump and feedwater pump suction pressures if the reject valve cycles or fails open for any reason. The condensate system includes a full-flow polisher (demineralizer) system to provide continuous demineralization of the condensate water supplied to the feedwater system and the OTSGs. A full-flow motor-operated bypass valve (CO-V12) is provided around the polishers which can be operated from the control room. This valve does not open automatically upon polisher system malfunctions (high differential pressure to the condensate booster pump suction). Prior to the accident, operators were working to transfer resin from polisher tank No. 7 to the resin regeneration tank. This had been in progress for approximately 11 hours perior to the trip.

Evaluation

Personnel interviews revealed that the operators were extremely sensitive to the fact that the normal operating conditions of the condensate/feedwater systems were very near the design limits at full power. (Int. 5, 17, 38, 56, 145, 147, 189, 2, 14, 37, 4, 9, 69, 153)

1.2.6 General Plant Parameters

Routine operating and surveillance procedures had been completed prior to the accident establishing specific operating conditions within the plant in accordance with the Plan-of-the-Day schedule dated March 27, 1979, including:

Surveillance Procedure 2301-S1, Shift/Daily Checks (March 28, 1979)
BWST Level 55 ft
BWST Temperature 68°F
Reactor Building Pressure 0.1 psig
Reactor Building Temperature 114°F
Control Room Temperature 70°F
Reactor Building Sump Pump Starts
(The sump routinely pumped approximately 800 gallons per day.)

Core Flood Tank Pressure	595-600 psig
Core Flood Tank Level	13.1
Condensate Storage Tank Level	21.5 ft

The major plant parameters immediately prior to the unit trip included:

Reactor Coolant System

Pressure	2155 psig
Temperature, Hot Leg	606°F
Temperature, Cold Leg	557°F
Temperature, Average	582°F
Flow (4 RCP)	107.5-108% 137 x 10 ⁶ lb/hr
Pressurizer Level	220 in.

Reactor

Power	97%
Core Power	2701 Mwt
Control Rod Groups % Withdrawn	
1-5	100%
6	95%
7	95%
8	27%

Power Imbalance (approximate)

Incore	-2.64%
Excure	-3.64%

Secondary Systems

Main Feedwater System (2 pumps)	OTSG	
	A	B
	Flow	5.798 5.727 x 10 ⁶ lb/hr
	Temperature	462.7 462.7°F
Pressure	1045	993 psig

Condensate System

2 of 3 condensate pumps and
2 of 3 condensate booster pumps)
2 of 3 heater drain pumps

Full-flow demineralizer (7 vessels)	8.303 x 10 ⁶ lb/hr
--	-------------------------------

Condenser vacuum	28 in Hg.
------------------	-----------

Main Steam System	OTSG	
	A	B
Temperature	595	594°F
Pressure	910	889.6 psig
OTSG Level	257	264 inches
Circulating Water System		
Inlet temperature	66°F	
Outlet temperature	95°F	

(Ref. 139, 140, 141)

1.3 STATUS OF SURVEILLANCE AND INSPECTION

1.3.1 General Status

During this investigation, a review was conducted to ascertain if the licensee was in any ACTION statement in accordance with the requirements of their Technical Specifications. It was found that the licensee had identified, prior to the accident, that one ACTION statement was in effect.

Technical Specification 4.5.2.a requires that DH-V157, the borated water storage tank to spent fuel pool isolation valve, must be closed during Modes 1, 2, and 3. DH-V157 had been opened on March 26, 1979, at 1500 hrs to permit recirculation of the BWST contents. The ACTION statement (a) for Technical Specification 3.5.2 requires that the system be made operable in 72 hours. This 72-hour period would have expired at 1500 hrs on March 29, 1979. Interviews revealed that this valve was closed at about 0800 hrs on March 28, 1979. (Int: 57, 111, 184)

The investigator selected 82 Technical Specification requirements as a sample, and reviewed the surveillance testing for the period of January 1, 1979 to March 28, 1979, that was performed to satisfy those requirements. All the surveillance had been performed within the schedule required and the results were satisfactory.

In addition, the following documents were reviewed for evidence of information that would indicate surveillance that was not satisfactorily completed or conditions that would indicate inoperable equipment.

- o Shift Foreman's Log Book 12/1/78 - 3/29/79
- o Surveillance Task Index
- o Surveillance Result Index for 1/1/79
3/29/79
- o TMI Unit 2 Histogram 1/1/79 - 3/29/79
- o TMI Unit 2 Core I, Cycle I Burnup History
- o TS Action Statement Summary, Shift Foreman's Logbook

During this review, there were no findings which would indicate that any other ACTION statements, other than the one identified above, (DH-V157), were applicable.

1.3.2 Status of Surveillance on Selected Engineered Safety Features (ESF) System Components

1.3.2.1 Review of Procedures

Selected surveillance procedures performed during the months of January through March of 1979, were reviewed including:

2302-M1A/B, Makeup Pump and Valve Functional Test, Revision 5, September 22, 1978 performed on January 29, February 14, February 22 and March 23, 1979.

2303-M2A/B, Decay Heat Removal Pump Functional Test and Valve Operability Test, Revision 7, June 20, 1978, performed on January 4, January 29, and March 3, 1979.

2302-M24A/B, Reactor Building Spray Pump Functional Test and Valve Operability Test, Revision 3, March 13, 1978, performed on January 5, 1979, February 7, 1979, and March 5, 1979.

1.3.2.2 Emergency Feedwater Valve Test

The review revealed no discrepancies or technical inadequacies with the implementation of the above procedures or their procedural content. One questionable practice was identified. During the review of the procedure for testing the reactor building spray pumps and valves (2303-M24A/B steps 6.3 and 6.4), it was noted that both the manually operated header isolation valves (DH-V133A and V133B) are closed simultaneously during the performance of the test procedure (Ref. 65). In order to perform the test and stroke the individual motor-operated isolation valves (DH-V8A and V8B) for the sodium hydroxide tank (DH-T-1), it is not necessary to close the manual isolation valve in the redundant flowpath.

Technical Specification 3.6.2.2 requires the spray additive system to be operable in Modes 1, 2, 3, and 4. The spray additive system is allowed to be inoperable for 72 hours. The total removal of the spray additive system from the operable condition during the performance of the surveillance procedure by closing the manual isolation valves (DH-V133A and 133B) does not appear to be contrary to the specification requirements. However, the provision of two automatic valves (DH-V8A and V8B) in parallel in order to meet the single failure design criteria also allows the testing of one spray additive flow path while the redundant path is available for automatic initiation if required. A more conservative surveillance procedure, isolating and testing the flow paths independently, would provide an additional margin of safety.

Selected surveillance performed on the emergency feedwater system during the months of January through March of 1979, were reviewed, including:

2303-M 14A/B, Revision 8, November 21, 1978, Emergency Feedwater System Valve Lineup Verification and Operability Test and Turbine Driven Emergency Feed Pump Operability Test, performed on January 3, February 26, and March 26, 1979.

2303-M27A/B, Revision 4, August 30, 1978, Motor Driven Emergency Feedwater Functional Test and Valve Operability Test, performed on January 3, February 26, and March 26, 1979.

Evaluation

Both of the above surveillance procedures contain specific steps that result in the routine isolation of the emergency feedwater supplies to the OTSGs by closing the header isolation valves (EF-V12A and V12B) simultaneously. This occurs as follows:

Procedure 2303-M14A/B requires:

- o Step 6.1.10, Close EF-V12A and 12B.
- o Step 6.1.52, Reopen EF-V12A and 12B.

Identical manipulations are performed in Steps 6.2.9, 6.2.40, 6.3.8, and 6.3.27.

Similarly, procedure 2303-M27A/B requires:

- o Step 6.1.2, Perform Appendix A(B) valve lineup.
(Appendix A(B) includes the closing of EF-V12A and 12B.)

- o Step 6.1.22, Insure EF-V8A(B) is open, EF-V12A(B) is open, EF-V1A(B) is closed, and close EF-V-39 (EF-V40).

Identical manipulations are performed in Steps 6.2.2 and 6.2.17.

NOTE: The above steps which use the designation "V12A(B)" suggest that only one valve was closed during a previous step or steps, which was not actually the case.

During the routine performance of the Surveillance Procedure 2303-M27A/B, the OTSG emergency feedwater flow paths were isolated by closing both header isolation valves (EF-V12A and 12B) during the test performance on January 3, February 26, and March 26, 1979.

Similarly, during the routine performance of Steps 6.1.10, 6.2.9, and 6.3.8 of Surveillance Procedure 2303-M14A/B/C, both OTSG flow paths were also isolated by closing both header isolation valves (EF-V12A and 12B) during the test performance on January 3, February 26, and March 26, 1979.

Evaluation

Technical Specification 3.7.1.2 requires that three independent steam generator emergency feedwater pumps and associated flowpaths be operable in Modes 1, 2, and 3 (OTSG pressure greater than 800 psig). The specific action statement does allow one emergency feedwater system to be inoperable, provided the inoperable system is restored to OPERABLE status within 72 hours or the reactor is to be placed in HOT SHUTDOWN within the next 12 hours.

The routine complete isolation of the emergency steam generator feedwater flow paths on January 3, February 26, and March 26, 1979 is contrary to the requirements of Technical Specification 3.7.1.2 and is being considered as a potential item of noncompliance.

1.3.2.3 Review of EFW Procedure Change

Since a previous procedure change resulted in the closure of both emergency feedwater valves during performance of the above surveillance procedures, an investigator reviewed the evaluation of that change. The review of the specific Plant Operating and Review Committee (PORC) minutes and the procedure change requests (PCR) associated with the procedure revision revealed an apparent inadequate safety evaluation, review, and approval in each instance, including:

Procedure Change Request PCR 2-78-707; Item II B of PORC Meeting No. 281, was entered on August 30, 1978, as Revision 4 of 2303-M27A/B. This procedure change was recommended to provide new pump reference values because the valve lineup had been changed. The emergency feedwater header isolation valves (EF-V12A and V12B) were closed simultaneously to prevent leakage past both the emergency feedwater level control valves (EF-V11A and 11B). (It appears that this leakage is an inherent design feature of these valves to prevent thermal transient to the emergency feedwater nozzles and should have been recognized during this review.)

The safety evaluation performed for this procedure change indicated that the change would enhance safety in that it would ensure that cold water would not be fed into the OTSGs and thermally cycle the emergency feedwater nozzles. The procedure change specifically included the simultaneous closing of both the header isolation valves (EF-V12A and V12B) during the performance of the surveillance activity. The safety evaluation failed to address that aspect of the change and the impact as an unreviewed safety question.

Procedure Change Request, PCR-2-78-895 (Included Temporary Change Notice 2-78-666, October 10, 1978), Item II B, PORC Meeting No. 293, was entered on November 21, 1978, as Revision 8 of 2303-M14A/B/C. (Ref. 67) This procedure

change was recommended to specifically close the header isolation valves (EF-V12A and V12B) during the performance of the procedures, but the simultaneous closing of both the header isolation valves (EF-V12A and V12B) was not specifically addressed in the detailed safety evaluation.

Evaluation

The Plant Operations and Review Committee (PORC) is required to review all changes to procedures which could affect nuclear safety. The Plant Operations and Review Committee (PORC) review and approve of the above procedure changes (2303-M27A/B, Revision 8, dated August 30, 1978, and 2303-M14A/B.C, Revision 8, dated November 21, 1978) placed the emergency feedwater system outside the license requirements, and this constituted an unreviewed safety question. (Ref. 66, 69)

The initial review and request for a procedure change by engineering; the review and approval of the change by the PORC; and the routing performance of the procedure by members of the operating shifts (including licensed operators and supervisors) without realizing that the performance of the procedure placed the plant outside the license requirements is indicative of a serious breakdown in the licensee knowledge level of the facility. This breakdown occurred at numerous levels of expertise and training, including engineering, management, and operations.

The safety evaluation performed associated with the procedure change requests (PCR-2-78-707 and 895, including temporary change notice, TCN-2-78-666) did not address the Technical Specifications requirements concerning feedwater system operability. The safety evaluations were directed towards obtaining test data and limiting thermal transients on the emergency feedwater nozzle. (Ref. 67, 142)

Technical Specification 6.5.1.7.b requires that the PORC shall render determinations in writing with regard to whether or not changes to procedures required by Technical Specification 6.8 constitute an unreviewed safety question. The failure to include the EFW valve lineup changes (simultaneous closing of both feedwater header isolation valves) within the safety evaluation as required and render a written determination, is under consideration as a potential item of noncompliance.

1.3.3 Status of Core Physics Surveillance

The status of the reactor core as revealed by the last surveillance test of that core was reviewed to determine if any core conditions existed that could have contributed to the response of the core to the accident. No core conditions that were outside the acceptable limits defined by the Technical Specifications for the facility were found.

Core performance surveillance was performed at 0810 hrs on March 19, 1979. Conditions during the surveillance were:

Power - 97.5%
Boron Con. - 1037 ppm
RC Total Flow - 138.7×10^6 lb/hr
Core Burnup - 2662 MWD/MT
Pressure - 2268 psia
ALHR - 5.9696 kw/ft
Core up - -21.315 psia
 T_{avg} - 584.3°F

Inoperable Self-Powered Nuclear Detectors (SPND) were:

String 4	level 5 (F-8)
String 5	level 2 (F-7)
String 6	level 2, 4-7 (E-9)
String 11	level 5 (K-5)

String 14 level 2, 3 (N-8)
String 27 level 2-7 (D-10)
String 38 All levels (L-2)
String 46 level 4, (R-10)
String 51 level 6, (D-14)

Inoperable Incore Thermocouples were:

String 26 (E-11)

Hot Fuel Maximum Peaking Factors were:

H - 8	1.485	Note: FSAR does not reference these values in Table 4.4-1 of the FSAR.
B - 8	1.361	
C - 8	1.234	
B - 6	1.198	

Channel Minimum DNB Ratio:

B - 8	3.004	Note: FSAR references 1.75 for maximum design conditions at 100% power. (Table 4.4-1, FSAR).
B - 6	3.356	
C - 8	3.619	
H - 8	2.830	

Hot Channel Maximum Linear Heat Rate (kw/ft)

B - 8	11.653	Note: FSAR references 19.03 kw/ft at design power (Table 4.4-1, FSAR).
C - 8	10.778	
B - 6	10.276	
H - 8	13.144	

Hot Channel Surface Heat Flux (Btu/ft² - hr)

B - 8	353,220	Note: FSAR references 576,885 Btu/ft ² - hr at design power condition (Table 4.4-1).
C - 8	326,690	
B - 6	311,480	
H - 8	398,390	

Hot Channel Maximum Surface Temp (°F)

B - 8	657.66	Note: FSAR references a maximum surface temperature of 654°F. (Table 4.4-1). There is no Technical Specification require- ment for maximum surface temperatures.
B - 6	657.53	
C - 8	653.78	
H - 8	657.72	

Hot Channel Maximum Fuel Temp (°F)

B - 8	3664.4	Note: FSAR references 4,170°F at design conditions. (Table 4.4-1, FSAR).
C - 8	3551.3	
B - 6	3483.7	
H - 8	3850.8	

(These data were extracted from Performance Data Output Segments 1-6; Ref. 68).

Core surveillance was also performed per Procedure 2311-FT (2), Revision 0, at 1500 hrs on March 22, 1979. Conditions during the surveillance were:

Power - 97.52%
 Imbalance - F0.5% all 4 channels
 Quadrant Tilt - F0.2% all 4 channels
 Exposure - 90.16 EFPD
 Maximum Fuel Ass'y to Ave. Power - 1.441
 Minimum DNBR - 2.91
 Max. LHR - 13.02 kw/ft

Max Heat Flux - 394,500 Btu/ft² - hr

Max Clad Temp - 655.2°F

FQ - 1.77

Evaluation

It is not clear that the licensee is monitoring $F_{\Delta H}$ as defined in the Technical Specifications. This item has been discussed with the licensee and Region I personnel and will be the subject of further review by the Regional Inspection staff.

1.3.4 Status of Surveillance Results and Records Review

Through personnel interviews and a review of available surveillance records it was revealed that inadequacies existed in the area of performance, review, and retention of surveillance tests.

The following surveillance procedures were reviewed: (2303-M14A/B/C, 2303-M27A/B, 2303-M1A/B, 2303-M2A/B, and 2303-24A/B; Ref. 69, 66, 70, 71, 65)

1.3.4.1 Performance and Review

The review and personnel interviews revealed that completed surveillance procedures are not routinely reviewed by the shift supervisor/shift foreman except for the completed data sheets. The remaining sheets are routinely discarded. A search of selected surveillance records in storage (2303-M14A/B/C, 2303-M27A/B, 2303-M1A/B, 2303-M2A/B, and 2303-M24A/B) revealed that no completed surveillance procedures had been maintained with exception of the data sheets, the computer schedule sheet, and the procedure cover sheet. Discussions with the licensee representatives indicated that the details of the surveillance procedures (Sections 1 through 6) were not retained because of a record storage space problem. Moreover, it was noted during the interviews that review of the completed procedures by the shift supervisor/shift foreman was not routine.

because the length of some of the procedures and the routine scheduled and unscheduled workload. Discussions revealed that the licensee schedules surveillance on the day shift (7-3) when extra personnel should be available to review the completed procedures as required. (Int. 107, 136, 116, 131; Ref. 66, 69, 70, 71, 65)

The surveillance procedures reviewed contained procedural steps (equipment manipulations) which, if not specifically completed and verified to be satisfactorily completed, could leave the specific engineered safety feature system in an inoperable condition. Because of a lack of documentation, the possibility exists that this occurred during the surveillance performed on March 26, 1979, on the emergency feedwater system and the header isolation valves (EF-V12A and V12B) left closed. (This matter of the closed valves is discussed in greater detail in section 4.2 of this report.) An operator in an interview stated that the header isolation valves (EF-V12A and V12B) were reopened on March 26, 1979, at the completion of the surveillance test (2303-M27A/B). (Ref. 66)

Evaluation

Administrative procedure 1001, Revision 13, March 30, 1978, step 3.8.5 (Procedure Usage), requires the implementation of surveillance procedures including performing the procedures rigorously, providing documentation, and providing shift supervisor/shift foreman review of results and approval of surveillance data sheets. (Ref. 94)

The failure to provide a review of the completed surveillance results by the shift supervisor/shift foreman as required to insure the procedures are implemented rigorously, documentation provided, and results reviewed is being considered as a possible item of noncompliance pursuant to Technical Specification 6.8.1.a.

1.3.4.2 Record Retention

Administrative procedure 1007, Revision 4, September 26, 1978, (Records) requires record storage including handling, storage, and indexing. (Ref. 95)

Technical Specification 6.10.1.d states that records of surveillance activities, inspections, and calibrations required by these Technical Specifications shall be retained for at least five years.

Evaluation

The failure to maintain the complete records of the surveillance activities as required by Administrative Procedure 1007 is contrary to Technical Specification 6.10.1.d and is under consideration as a potential item of noncompliance.

1.3.5 Inspection of Surveillance Activities

The Metropolitan-Edison Three Mile Island Final Safety Analysis Report, Chapter 17.2.15, Inspection, and the Operational Quality Assurance Program, Section X, Inspection (Surveillance), requires, in part, that during normal unit operation or functional testing, the surveillance program include random observation of operations and functional testing. Inspection, examinations, measurements, or tests of materials, products, or activities are required to be performed for each work operation where necessary to assure quality, whether the work is done by the station staff or an outside contractor. In the event a surveillance of processed material or products is impossible or impractical, indirect control by monitoring processing methods, equipment, and personnel is required. Both inspection and process monitoring must be provided when control is inadequate without both.

The Metropolitan Edison Operational Quality Assurance Surveillance Program, GP 4014, Rev 0, December 6, 1977, requires independent observation of activities affecting quality to verify conformance with established requirements utilizing both inspection and auditing techniques for compliance with written procedures and the Technical Specifications.

Evaluation

The review of the selected surveillance activities and the administrative controls and the personnel interviews revealed that the operations surveillance and testing activities were not randomly or routinely inspected by independent methods as required by Criterion X of Appendix B to 10 CFR Part 50; amplified in the FSAR, Chapter 17.2.15, Section X, Inspections (Surveillance); and GP4014, Operational Quality Assurance Surveillance Program. (Int. 131)

As of March 28, 1979, the failure of management to adequately implement the inspection requirement pursuant to Criterion X of Appendix B to 10 CFR 50 is being considered as a potential item of noncompliance.

1.4 OPERATIONS STAFF

1.4.1 Operations Personnel Training and Qualifications

1.4.1.1 General Plant Training Program

The normal training progression starts with Auxiliary Operator C and continues through Auxiliary Operator B and A. The most senior qualified A operator can apply for the position of Control Room Operator (CRO) trainee when one becomes available. CROs can compete for available Senior Reactor Operator (SRO) positions.

1.4.1.2 Auxiliary Operator Training

The requirements for becoming an Auxiliary Operator-C (AO-C) are a high school diploma and a course in high school algebra. Initial training as an AO-C consists of 8 or 9 weeks of classroom training, dealing mainly with secondary plant systems. Other areas covered during this period include:

- o Fundamentals of Mechanics and heat transfer
- o The basic steam cycle
- o Switching and tagging procedures
- o Introduction to the safety features actuation system (SFAS) and the reactor protection systems (RPS)
- o First aid
- o Fire fighting
- o Intermediate health physics

This period is followed by a written examination. The trainee is then assigned to a crew under a foreman, and works under the supervision of an A level AO for the remainder of a year. The trainee must then pass another written and an oral examination to become an AO-B. Failure to pass the examination results in removal from the training program.

Trainees at the AO-B level must have completed courses in high school trigonometry and physics. This training period also covers a year and is initiated by six weeks of full time classroom instruction. The subjects covered deal mainly with the primary system and include:

- o Primary coolant makeup system
- o Decay heat removal system
- o Reactor Building cooling
- o Core flood tanks
- o Reactor Building spray system
- o In depth study of the SFAS and RPS
- o Nuclear instrumentation

- o Control rod drive mechanism
- o Integrated control system
- o Advanced health physics

A comprehensive written examination is given at the end of this period. The remainder of the year is spent in on-the-job training, mainly on primary systems. This year is also followed by comprehensive oral and written examinations which the trainee must pass in order to remain in the program and to be promoted to A0-A. An A0-A is eligible to enter training for the position of Control Room Operator.

1.4.1.3 CRO Training

The CRO Trainee enters the Category IV (CIV) Training Program which lasts nine months. This program is divided into six cycles of 5 to 6 weeks per cycle, with a written and an oral test at the end of each cycle. The program is set up for individual self study. Each section lists the procedures that must be learned and specifies how proficiency in the procedure is to be demonstrated. This is done by actual performance, by simulation, or by discussion.

At the end of Cycle 6, a cumulative examination over the entire course is given. Successful completion of this examination is followed by a two-week course on the reactor simulator. The trainee then takes a mock NRC written exam, and an NRC-type oral exam given by a senior reactor operator. The trainee is then scheduled for the actual NRC licensing examination.

Experienced CROs can compete for SRO positions. SRO Training is individual self study tailored to the individual's experience, but includes:

- o nuclear theory
- o radioactive material handling
- o operating characteristics
- o fuel handling

- o core parameters
- o administrative procedures
- o technical specifications
- o transient analysis

The programs outlined above were compared to the training requirements in ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel," which the licensee is required to meet. No discrepancies were identified.

1.4.1.4 Training for Individuals with Previous Experience

Individuals with previous nuclear power experience are given abbreviated training that takes into account their previous training. The training records onsite indicate that many TMI operators were graduates of the Navy Nuclear Power School. For these individuals, the normal two years of auxiliary operator training and work experience were replaced by a special 26 week course. This course included 40 hours per week of classroom instruction and plant tours. It concentrated on those areas of the auxiliary operator training not covered by the Navy training, primarily the TMI plant systems and operation. This was followed by the CRO training program.

1.4.1.5 Requalification Program

The licensee's requalification program required of all licensed operators is divided into four areas:

- a. Operational Review Lectures
- b. Fundamental and Systems Review
- c. On the Job Training
- d. Annual Evaluation Examinations

The lectures consist of at least 60 hours per cycle and cover the following areas:

- a. Reportable Occurrences
- b. Modifications
- c. Operating Problems
- d. Procedure Changes
- e. Abnormal and Emergency/Procedures Review
- f. Technical Specifications
- g. Operational Evaluations
- h. Federal Regulations
- i. System Review Material
- j. Other topics as necessary

Periodic quizzes that must be passed with an 80% grade are given.

The fundamentals and systems review program is based on the annual written examinations and includes topics in the following areas:

- a. Operation
- b. Facility Design
- c. Operating Characteristics
- d. Instrumentation and Control
- e. Safety and Emergency Systems
- f. Normal and Emergency Procedures
- g. Radiation Control and Safety

The Senior operator program includes:

- a. Reactor Theory
- b. Radioactive Material Handling
- c. Fuel Handling
- d. Core Parameters
- e. Administrative Procedures

Removal from duties and accelerated requalification training is mandatory for failure to score 70% overall on the annual examination.

Quizzes requiring an 80% passing grade are also given in for this program.

The On-the-Job-Training for requalification includes:

1. 10 reactivity manipulations.
2. Participation in surveillance testing, system checkout, and equipment operation relevant to the license.
3. Review of procedure changes, modifications, Technical Specification changes, reportable occurrences, and incidents not covered in the lecture series.

1.4.1.6 Training on Significant Events

The training department reviews the NRC listing of the Licensee Event Reports to identify items for inclusion in the operator training program. However in at least one case this review did not identify an item of significance to the accident. On September 24, 1977, Davis Besse experienced a loss of feedwater transient which resulted in reactor transient behavior which was similar in many respects to the behavior of the transient which initiated the events of the TMI accident. A review of the training records and discussions with the training staff indicate that this event was not brought to the attention of the operators. This particular item was included in the NRC publication "Current Events Power Reactors" for September 1, 1977, through October 31, 1977, published in December 1977. The NRC distribution list indicates that 10 copies of this report were sent to various Metropolitan Edison and GPU addressees. These included the TMI site and the Unit 1 superintendent. However, the training staff member responsible for this area stated that this report had not been made available to the training department.

Another significant event occurred at the Rancho Seco Facility on March 20, 1978. A transient was initiated by a loss of electrical power to part of the non-nuclear instrumentation. As a result of this event B&W sent a memo to the Davis Besse Station Superintendent which included recommendations regarding operation training and emergency procedures. These recommendations discuss the use of alternate instrumentation to replace inoperable instrumentation and the use of RCS pressure as well as pressurizer level to assure that the reactor coolant system is full. (SOM #403 dated August 9, 1978; Ref. 143)

Review of TMI training records showed that the Rancho Seco event had been covered during operator training, but, no record of any specific B&W recommendation, such as those sent to Davis Besse, existed. Training personnel said that they had not been advised of these recommendations. The B&W Site Operation Memoranda (SOMs) to TMI were reviewed for the period covering May 3, 1978, through March 9, 1979. These memoranda did not address the discussion of the Rancho Seco event. The Unit 1 superintendent stated that TMI did not receive any recommendations from B&W based on this event, and this was confirmed by the B&W site representative. The stated basis for this decision was that TMI had a cooldown transient similar in nature to the Rancho Seco event, but caused by different factors. The B&W analysis of the TMI event was covered in a site operations memo (SOM) to TMI. (Ref. 144) (SOM II-140, dated May 2, 1978) As a result, B&W did not consider it necessary to send TMI the Rancho Seco related recommendations. A review of the SOMs to TMI for the period in question indicate that the recommendations specifically relating to operator training were not included in this correspondence.

1.4.1.7 Annual Examinations

An annual written evaluation examination and oral evaluation are administered to all licensed personnel. These examinations simulate the licensing examinations given by NRC. Failure to score higher than 80% in all sections

of the written exam requires participation in the systems review program. Failure to score 70% may require removal from duties to enter an accelerated training program. An unsatisfactory oral evaluation may require similar action.

1.4.1.8 Individuals Involved with the Early Part of the Accident

All four of the NRC licensed individuals who were on shift at the start of the March 28 event were graduates of the Navy nuclear power school. All were initially hired as auxiliary operators and attended the special 26-week course. In addition, each had attended from 5 to 9 weeks of B&W simulator training. The specific experience and training of these individuals as indicated by licensee records and interviews with the individuals are as follows:

o Shift Supervisor E:

He was hired as an Auxiliary Operator on April 14, 1972. He became a Shift Foreman (requires a NRC senior reactor operator's license) on October 1, 1972, and shift supervisor on May 17, 1976. He completed the abbreviated auxiliary operator training and the initial control room operator training on October 19, 1973. By March 1979 he had 7 years of experience at TMI and a total of 13 years of nuclear experience.

o Shift Foreman C:

He was hired as an auxiliary operator on March 5, 1973. He became a control room operator (which requires an NRC senior reactor operator's license) on August 11, 1975, and a Shift Foreman on February 13, 1978. He completed the abbreviated auxiliary operator training program on October 19, 1973, and the initial control room operator (CRO) training on August 11, 1975. By March 1979 he had 6 years of nuclear experience at TMI and a total of 15 years of nuclear experience.

o Control Room Operator C:

He was hired as an auxiliary operator on December 5, 1973. He became a Unit 1 Control Room Operator on October 20, 1975 and worked in this capacity for three months. He regained his control room operator status on September 13, 1976. He completed the abbreviated auxiliary operator training program on August 26, 1974, and the Unit 2 control room operator training program on October 20, 1977. By March 1979 he had 5 years of nuclear experience at TMI and a total of 12 years of nuclear experience.

o Control Room Operator D:

He was hired as an auxiliary operator on November 28, 1973, and became a Control Room Operator on October 6, 1975. He completed the abbreviated auxiliary operator training program on August 26, 1974, and the initial Unit 2 control room operator training program on August 30, 1977. By March 1979 he had had 5 years of nuclear experience at TMI and a total of 10 years of nuclear experience.

Other individuals involved with the early part of the accident include:

o Shift Supervisor A:

He had no nuclear training prior to his employment at TMI on March 3, 1969. He became a Control Room Operator on August 13, 1970, and a Shift Foreman on October 13, 1975. He attained his present position as Shift Supervisor on October 1, 1977. His initial nuclear training was a 48 week Reactor Operator Training course which covered the period from October 20, 1969, to August 18, 1970. He completed his Unit 2 Senior Reactor Operator license training on August 1, 1971. As of March 1979, he had had 10 years of nuclear experience, all at TMI.

o Shift Foreman B:

He had a B.S. degree in Chemistry and no previous nuclear experience when he was hired as an Auxiliary Operator on April 4, 1973. He became a

Control Room operator on August 11, 1975, and a Shift Foreman on August 1, 1978. He completed his initial auxiliary operator training on October 19, 1973, and a Unit 2 Control Room operator training program on August 11, 1975. He completed his senior operator training program for Unit 2 on August 14, 1978. As of March 1979, he had 6 years of nuclear experience, all at TMI.

Management level individuals were also involved in the Unit 2 operations during the early hours of the accident. Their experience and training are as follows:

o Unit 1 Supervisor of Station Operations Nuclear:

He was a graduate of the Navy Nuclear Power School and had 8 years of nuclear experience in the Navy. He was employed at TMI on December 9, 1968. He became a Shift Supervisor on July 15, 1972, and attained his present position as Supervisor of Operations on April 1, 1978. His training records show a total of 659 hours of Senior Reactor Operator license review and training, completed on August 30, 1977, and 2 1/2 weeks of simulator training. As of March 1979, he had 11 years of nuclear experience at TMI and a total of 19 years of nuclear experience.

o Superintendent of Technical Support:

He had BS degree in Mechanical Engineering and no previous nuclear experience when he was employed by Metropolitan Edison on September 1, 1978. He was assigned to TMI in 1970. He was placed in the position of Engineer Nuclear on November 1, 1973, and Engineer Nuclear III on July 1, 1975. He became a Supervisor of Station Operation Nuclear on September 1, 1975, and was assigned his present position as Unit Superintendent of Technical Support on December 1, 1977. His training records show two short college courses totaling 160 hours of reactor classroom instruction and 2 weeks of reactor simulator training. The records did not show completion of any of the more extensive TMI operator training programs. He has a

Senior Reactor Operator license for Unit 1, but had not yet acquired an NRC license on Unit 2. As of March 1979, he had a total of 9 years of nuclear experience, all at TMI.

1.4.2 Shift Composition Compared to Technical Specifications

Technical Specification 6.2 specifies the Organization, and 6.2.2 establishes the Unit Staff, including the shift manning requirements. The shifts consist of a Unit 1-2 Shift Supervisor, a Unit 2 Shift Foreman, licensed control room operators, and "A" auxiliary operators.

Order of Modification of License, issued October 13, 1978, required the commitment to specific operator actions to be completed within 10 minutes from the onset of the small break with a single failure. The licensee designates a small-break LOCA-CRO (switching and tagging Control Room Operator) and a small-break LOCA-AO (primary auxiliary operator). (Int. 5, 17, 38, 56, 145, 147, 189, 3, 15, 4, 9, 61, 153, 151)

Records reviewed and personnel interviews revealed that the shift complement for Unit 2 on the morning of March 28, 1979, consisted of:

- o Unit 2 Shift Supervisor - SRPO
 - o Unit 2 Shift Foreman - SRO
 - o Two Unit 2 Control Room
 - o Four Unit 2 Auxiliary Operators A
 - o Two Unit 2 Auxiliary Operators B
 - o Two Unit 2 Auxiliary Operators C.
- (Int. 5, 17, 38, 56, 145, 147, 189, 2, 14, 37, 189, 3, 15, 151, 4, 9, 61, 153, etc.)

Two Control Room Operators were present in the control room at 0400 hrs on March 28, 1979. The shift supervisor was in the SS office adjacent to the control room. (Int. 5, 17, 38, 56, 145, 147, 189, 3, 15, 4, 9, 61, 153, 151)

Four Rad-Chem technicians were assigned to the 11-7 shift and were present at the facility at 0400 hrs on March 28, 1979.

Evaluation

The total shift personnel complement present on the 11-7 shift on March 28, 1979, and present at 0400 hrs at the initiation of the accident, met the requirements of the Technical Specification.

1.4.3 Shift Staffing During Prior Trips

The available records of past reactor trip events and plant logs and staffing records for Unit 2 were reviewed in an attempt to identify any trend in operating staff assignments that may have significance in understanding the causes of the March 28, 1979, accident. No trends of specific significance were identified.

Of interest though, and outside the scope of this investigation, it was noted that Shift Supervisor C and Shift Supervisor D were the duty Shift Supervisors during eight of the ten reviewed trip incidents. Further, it was noted that in seven of the ten reviewed incidents, the trip was experienced within a nominal 90 minute period before or after shift change.

1.4.4 Physical and Emotional Condition of Operational Staff

During the course of interview sessions, limited observations were made by the investigators to note any readily apparent health or emotional stability (obvious aberrant behavior) problems of the licensee's operating staff. These interviews were conducted during the period early April to early July 1979, when the members of the operational staff were under a high degree of emotional stress directly as a result of the incident and, secondarily, as a result of the extensive investigative activity underway and the high level of local anti-nuclear sentiment that appeared to be impacting their personal activities.

It must be noted that the investigators are not trained medical observers, but rather are lay personnel specifically alert to obvious signs of emotional stress. During these interviews, while staff personnel recalled the events and their actions of March 28 and were subjected to the direct questioning of the investigative staff, no abnormal behavior was noted and all operating staff participated with logical and poised responses.

A review of the licensee's screening policies as detailed in their Policy and Procedural Manual showed that the licensee has a policy of administering a psychological screening test, to all new personnel hired subsequent to August 24, 1978, in an effort to determine their suitability to work at the facility. The results of these tests are evaluated by trained medical staff.

A further review of the licensee's records showed four (one operating and three support) personnel warranted further evaluation. These individuals were identified as a result of a review of all personnel files performed in preparation for the revised NRC regulation on physical security requirements that became effective in August 1978. Three individuals were evaluated and cleared. The single concerned operating staff member subsequently resigned for independent medical reasons.

No reasons were identified to suspect that the health or psychological condition of operational staff members played a contributing role in the initiation of or response of the licensee staff to this accident.

1.4.5 Possibility of "Anniversary" Celebration On Site

Since the date of the incident, March 28, 1979, represented the first year anniversary of the date on which Unit 2 attained criticality, the possibility that unofficial "celebrations" may have been underway at the time of the incident was evaluated.

During the course of this portion of the investigation, no information was obtained that would support a contention that a celebration of any type was in progress. Thus any such contention has no bearing on the initiation or the conduct of plant staff during this accident. In only one instance during the course of the plant staff interviews was there a mention of an anticipated onsite celebration. Further discussion with that individual, (a support staff member, not one of the operating shift complement) both on and off tape, revealed that no alcoholic beverages were anticipated, and by use of the term "celebration," the interviewee had intended to indicate a "coffee and doughnuts" type of get-together rather than anything more extensive.

There was no knowledge of any planned "celebration" stated by any operational personnel, and the investigation has concluded that no such celebration was in progress immediately prior to the accident.

1.4.6 Possibility of Plant Sabotage

As a result of the high degree of public concern associated with the possibility of sabotage or adverse human actions having caused or contributed in some manner to the severity of the March 28 incident at TMI, all members of the investigative team were alert to this possibility.

Within their limited authority, as non-criminal investigators, the assigned investigators pursued the question of possible sabotage during the course of the interview program through liaison with the FBI and other governmental agencies and evaluation of other referred allegations and matters of concern. "Limited authority" means that the NRC investigative staff acted in a noncriminal investigative manner commensurate with their lack of power to administer oaths (i.e., take sworn statements), to subpoena necessary documents such as phone toll records and official business records, or to obtain access to the criminal intelligence records of other governmental and/or state agencies. The goal of this investigation was to attempt to identify any intelligence that would

indicate whether sabotage did play a role in the event. At that point, commensurate with agency policy, the obtained intelligence would be referred to the proper Federal criminal investigative agency.

No information was developed during this investigation indicating that sabotage was a contributing factor to the initiation of the accident or to the subsequent response of plant personnel or equipment to the accident.

This investigation has also addressed several allegations provided to the NRC by confidential sources that alleged sabotage or predictions of catastrophic events by various individuals or groups. All allegations were determined to be unfounded. (Contact of the investigators with the local FBI agents in charge indicated a similar disposition by them of matters under their purview.) "Predictions" in certain newspaper articles were found upon examination to include "March 28" only as a reference to the 1978 date that Unit 2 attained criticality as opposed to "predicting" a catastrophic occurrence. This investigation found no basis for further evaluation of these articles or interviews with their authors.

The interview program addressed the question of potential sabotage on selected occasions using questions directed to all working levels regarding knowledge of possible sabotage.

Responses ranged from direct negative responses to disbelief that sabotage could even be considered. Noticeably lacking during the course of the interviews was any disloyal animosity on behalf of the plant operating staff toward licensee management. The operating staff appeared generally satisfied in their relationships with licensee management, with only minor dissatisfaction expressed over the adequacy of training. A good degree of rapport appeared to exist between all levels of operating staff supervision and workers.

Vocal antinuclear sentiment appears to have been relatively absent prior to the incident. Relationships between the local governments and licensee management appear to have been of a tolerant nature.

Those conditions commonly associated or viewed as causative factors precipitating industrial sabotage were not identified by the investigation as being present at the time of the incident.

This investigation evaluated the effort that would be required to attempt to identify the party or parties who closed the emergency feedwater system block valves (EF-V12 A/B) for whatever undefined reason that may have motivated them. The investigation revealed that the two valves in question were capable of being operated from three specific locations: the control room, the 480V Substation panels (2-21EA & 2-11EA) at the 305 elevation of the auxiliary building, and the physical location of each valve.

Checks of the licensee's security access badging records showed approximately 470 licensee personnel and 260 contractor/vendor personnel would have had unescorted access to one or more of these locations on any of the two work days preceding the March 28 incident. Records exist whereby the identity of the contractor/vendor personnel entering the protected area could be retrieved. However, the 470 licensee personnel are only logged in at the site perimeter and need only display their photo ID badge (issued only after psychological screening and pre-employment checks are complete) to secure access to the TMI protected areas (Units 1 and 2, auxiliary building, turbine building and environs).

Further investigative effort of the magnitude that would be required to specifically identify which of the more than 600 personnel did access the protected areas during the period March 26, 1979, through March 28, 1979, was deemed unwarranted at this time in view of the absence of any intelligence that adverse human activity was involved in the accident.

1.5 SIGNIFICANT OPERATING EVENTS

The investigation team reviewed records of operating events related to reactor trips and events significant enough to be reportable to NRC under the licensee's Technical Specifications 6.9.1.7, 6.9.1.8, and 6.9.2.a. In addition,

the team reviewed reports to NRC by the licensee required by 10 CFR 50.55(e) that might have significance when examined in light of this accident. The review was conducted to identify potential contributions, precursors, and similarities to the accident that occurred on March 28, 1979. The results of this review and the analysis of leads that were followed are presented below.

1.5.1 Summary of Selected Licensee Reports

The following Construction Deficiency Reports (CDRs) and Licensee Event Reports (LERs) were deemed to address elements related to the accident:

<u>CDR/LER</u>	<u>Occurrence Date</u>	<u>Related Elements</u>
77-9c	7/1/77	<p>RCP seal injection determined not to meet single-failure criterion, with resulting potential of RCP seal failure on loss of offsite power.</p> <p>Corrective action required modification to provide redundant seal water sources prior to initial fuel load.</p> <p>NOTE: The investigation team reviewed this report for background information and did not pursue it further.</p>
77-11c	9/21/77	<p>GE type SBM switches used in Unit 2 safety-related equipment determined to be subject to potential cracks and breaks in the cam followers, which position contacts within the switches.</p> <p>Corrective action required replacement of all questionable switches prior to core loading.</p>

NOTE: The investigation team reviewed this report after the potential problem with starting reactor coolant makeup pumps was identified. See Section 4.18, Details I.

78-020-03L 3/29/78

Containment isolation valves found inoperable due to engagement of manual handwheel pins.

Corrective action required revisions to procedures to ensure handwheel pins are disengaged after manual operation.

NOTE: The investigation team reviewed this report for background information on previous examples of valves mispositioned and/or left in an inoperable condition. This lead was not pursued further, since no transient was involved.

78-021-03L 3/29/78

Reactor trip and ES actuation upon loss of a vital bus. The RCS depressurized via the EMOV, which failed open on loss of power. The cause of the depressurization was not obvious to the operators because the EMOV did not have actual position indication. Compensated pressurizer level indication was also lost, since it is powered from the same source.

Equipment modifications subsequent to this event included providing indication that the EMOV actuating solenoid had energized (but not that the EMOV was actually open or shut) and a change to the EMOV failure mode so that it would close on loss of power.

NOTE: The investigation team reviewed records related to this event for background information on earlier EMOV and pressurizer level indication problems. See Sections 1.5.2.1, 1.6.2 and 4.10, Details I.

78-027-01T 4/13/78

B&W informs licensee a small-break LOCA on the discharge side of an RCP is more critical than that previously analyzed. Solution to problem requires operator action to cross-connect and throttle HPI flow during a small-break LOCA.

NOTE: The investigation team reviewed this report for background information and to gain perspective on the conditioning of operators to ensure proper response to the small-break LOCA.

78-033-01T 4/23/78

Reactor trips at power, resulting in ES actuation and RCS depressurization from excessive OTSG cooldown. Operator failed to decrease speed of steam-driven feedwater pump, which was in manual, until excessive OTSG feed had occurred. Preliminary calculations indicated a steam bubble may have been introduced into one or both RCS hot legs.

Corrective action required investigation of the initiating event, additional tuning of controls, a change in Technical Specifications and revision to operating procedures.

NOTE: The investigation team reviewed this report for background information; but did not pursue it further, since the transient did not involve a loss of feedwater or failed open EMOV.

78-055-03L 9/13/78

Control room emergency air handling system dampers failed to respond to a close signal.

Corrective action required resolution of construction problems and re-inspection of air-operated dampers for operability.

NOTE: The investigation team reviewed this report for background information on potential causes of the buildup of radioactivity in the Unit 1 and Unit 2 control rooms which occurred during the accident. See Details II.

78-062-01T 10/19/78

RCS unidentified leakage exceeds Technical Specification limits, without shutdown, due to misinterpretation of requirements.

Corrective action required reduction in unidentified leakage and instruction of personnel.

NOTE: The investigation team reviewed this report for background information. See Section 1.2.3, Details I, for additional problems in this area.

78-065-99X 11/7/78

Reactor trip at power and ES actuation on loss of one steam-driven feedwater pump. This pump tripped following loss of one condensate booster pump. That pump tripped on a low suction pressure trip; resulting from the loss of two heater drain pumps, which supply nearly 30% of the total feedwater flow. Pressurizer level indication was lost during the transient, but licensee calculations indicated the pressurizer was not emptied.

Licensee reported, "No corrective action is required."

NOTE: The investigation team reviewed the records of this event for background information.

78-069-99X 12/2/78

Reactor trip at power and ES actuation due to overfeeding of OTSGs during transfer from the startup to main feedwater valves. The main feedwater regulating valve was found full open and disabled in that position.

Corrective action required procedure revision to prevent reoccurrence.

NOTE: The investigation team reviewed this report for background information on previous ES actuations and examples of mispositioned valves, but did not pursue it further, since the transient did not involve a loss of feedwater or failed open EMOV.

78-073-03L 12/15/78

Sample lines to HP-R-227 suspected to be source of water accumulation, which caused air pump to seize.

Corrective action required pump replacement, sample line heat tracing, periodic inspection for condensation and investigation of appropriate long term fixes.

NOTE: The investigation team reviewed this report for background information on potential sources of water in these lines, found during the accident. See Sequence of Events.

79-001-E 3/21/79

Valve mistakenly left closed following maintenance and clearing of tags, resulting in sulfuric acid discharge to the river during condensate polisher regeneration.

Corrective action required re-instruction of personnel.

NOTE: The investigation team reviewed this report for background information on additional examples of misaligned valves. The lead was not pursued further, since it did not involve an RCS transient.

79-002-03 12/26/78

Adequate documentation was not retained following NI calibration surveillance testing to demonstrate Technical Specification compliance.

Corrective action required re-instruction of personnel.

NOTE: The investigation team reviewed this report for background information on surveillance test record keeping. See Section 1.3.4, Details I for additional problems in this area.

As mentioned at the start of this section, the above LERs and CDRs were selected to identify potential contributors to this accident. No attempt was made by the investigation team to perform an in-depth analysis of the LER history of the licensee. That analysis is being addressed elsewhere.

NOTE: NRC has negotiated a modification to the Teknekron, Inc. contract titled "Licensee Performance Evaluation," that extends their study to include all operating B&W plants, including TMI-2; thereby providing independent in-depth analyses of the performance of these licensees and the differences between them. The methodology developed by Teknekron has been tested on 12

plants and focuses on LER content to identify trends in parameters that reflect performance of facility, operating personnel and management. The results of this study should be available within six months and the investigation team did not attempt to duplicate that effort.

1.5.2 Review of Selected Prior Trips

1.5.2.1 Previous Event Involving the Electromatic Relief Valve (EMOV)

On March 29, 1978, a reactor trip occurred while physics testing was in progress. The power level was much below 1% prior to the event and there was essentially no exposure on the core and hence no decay heat. While reactor building isolation and cooling surveillance were being performed, a fuse blew in a power supply that supplied power to a pressure transmitter feeding a bistable associated with the EMOV. The EMOV opened, and reactor coolant system pressure fell from 2188 psi to approximately 1173 psi at which time power was reestablished to the pressure transmitter and the EMOV reclosed. Saturation conditions in the RCS were not reached (T_{hot} and T_{cold} were isothermal and the maximum temperature reached was 535°F). The valve remained open for approximately 4 minutes. The reactor tripped on the power/flow trip that was brought about by the loss of power when the fuse blew. (The reactor protection system erroneously sensed that a reactor coolant pump had tripped.) During the decrease in pressure, high-pressure injection was initiated at approximately 1600 psi.

Evaluation

A review of computer printouts indicates that the HPI was bypassed. Pressure recorder traces and records indicate that the HPI system was manually shut off. (Ref. 145) This action was apparently based on the concern of injecting sodium hydroxide into the RCS. Pressure recorder traces indicate that RCS pressure was below 1700 psi for approximately 1 hour. It was noted

during the review of computer printouts that high-temperature alarms came in on both safety valves 1.5 minutes after a high outlet temperature was alarmed on the EMOV. In addition, it was noted that the Reactor Building Sump Pump was actuated during this event probably as a result of RCDT relief valve operation. As a result of this event the licensee recognized (Licensee Event Report 78-22/99X) that without position indication for the EMOV the cause of the depressurization was not obvious to the operators. (Ref. 42) Corrective actions taken in this regard are addressed in Section 1.6.2.2 of Details 1. NRC review of this event is documented in IE Inspection Report 50-320/78-15, Paragraph 4.

1.5.2.2 Previous Loss of Feedwater Transient

On November 3, 1978 the reactor tripped due to a loss of feedwater event. (Ref. 147) The loss of feedwater occurred when an instrument technician opened the control power breaker to the condensate polisher control panel by mistake. This action resulted in the closing of the condensate polisher outlet valves. When the flow of water was interrupted the condensate booster pumps and the main feed pumps tripped. The power level prior to the reactor trip was 90%. Upon loss of feedwater to the steam generators the pressure increased in the primary system eventually reaching the high pressure trip setpoint. The EMOV appears to have functioned normally. RCDT pressure increased from approximately 0.8 psi to 2.0 psi in 15 seconds. Maximum RCDT pressure was approximately 11 psi. Reactor pressure increased to approximately 2360 psi then reached a minimum of approximately 1940 psi. Pressure then returned to approximately 2100 psi. There was no initiation of high pressure injection since pressure did not decrease below the initiation setpoint. Pressurizer level, initially at approximately 220 inches reached a minimum of approximately 100 inches and then stabilized at approximately 105 inches. No fuel damage was indicated from the review of the event.

Steam generator A startup feedwater level was approximately 140 inches prior to the event. Within 30 seconds the level was below 18 inches and in 48 seconds was indicating a "dry" condition (10 inches). The dry condition was

maintained for approximately 3 minutes and level did not increase above 18 inches for approximately 4 1/2 minutes. Steam generator B startup feedwater level was approximately 147 inches prior to the reactor trip. A level of below 18 inches was experienced in approximately 50 seconds and stayed below this value for approximately 4 minutes.

Evaluation

Technical Specification 3.4.5 requires that each steam generator shall be operable with a water level between 18 and 440 inches in Modes 1, 2, 3 and 4. Since the reactor was in Mode 3 during this occurrence and the steam generator levels were not maintained above 18 inches the above requirements of Technical Specification 3.4.5 were not maintained.

At the request of an investigator the licensee was requested to review records to determine the level of review that the event had received. This review indicated that with the exception of a Reactor Trip Report (single page) and some hand drawn curves performed by a corporate office engineer there was no detailed review of the event. The review by the licensee also noted there were no Site Problem Reports or Site Operations Memoranda written by B&W as a result of the event.

Technical Specification 6.9.1.9.b requires the licensee report in writing within 30 days the occurrence of event of the type leading to operations in a degraded mode permitted by a limiting condition for operation. Technical Specification 3.4.5 requires each steam generator shall be operable with a water level between 18 and 440 inches. Action statement 3.4.5.b requires that with one or more steam generators inoperable due to the water level being outside of limits, be in at least Hot Standby within 6 hours and in Cold Shutdown within the next 30 hours.

Contrary to the above, the licensee failed to report to NRC the occurrence of steam generator levels below 18 inches for about 4 minutes, a degraded mode allowed by the Technical Specifications, at about 11:48 p.m. on November 3, 1978. This item is under consideration as a potential item of noncompliance.

1.6 Corrective Maintenance Practices

1.6.1 General

Because the accident of March 28, 1979, involved the failure or malfunction of one or more components, the investigation team performed a review of plant maintenance practices. The review concentrated on the area of corrective maintenance since this area addresses the repair of failed or malfunctioning components as well as the interface between maintenance and operational activities.

Corrective maintenance is administered through the use of work requests, also referred to as "job tickets." The work request form allows a plant staff member to request change or modification work. This system allows for classifying the work with respect to level of control (QC or non-QC), change or modification; radiation or nonradiation, and/or corrective maintenance. The work request provides for supervisory review for the issuance of work to the responsible group as well as a review for use of appropriate procedures to meet the defined effort. All work requests are numbered, and recorded in a maintenance logbook which contains, among other things, a short title, type of action required, and the group to perform the effort. The term "cancel" noted in the log indicates either that a duplicate request is on file or that, after review of the request, the problem no longer exists.

Maintenance Station Administrative Procedure No. 1026 and Station Corrective Maintenance Procedure No. 1407-1 define the work request system and responsibilities for the conduct of corrective maintenance. Responsibilities are defined for the Unit Superintendent, Supervisor of Maintenance, Supervisor of Operations, Maintenance Foreman, and Supervisor of Quality Control. Each Unit Supervisor of Maintenance receives the work request, which is logged into the specific unit log. The work request is classified and assigned to the proper maintenance group. The work is performed by Metropolitan-Edison personnel or by a contractor. The maintenance contract work is performed by Catalytic, Inc. The contract

maintenance work is performed under terms, conditions, and provisions of General Maintenance Contract, Metropolitan-Edison, P.O. No. 38735. The same procedures that cover Metropolitan-Edison maintenance are applicable to the contractor. (Int. 193) Catalytic, Incorporated, maintenance personnel can perform maintenance on both safety-related and nonsafety-related systems and equipment.

Corrective maintenance is coordinated with and controlled by the Operations personnel from the initiation of the maintenance efforts to its completion. The work is to start after Technical Specification requirements are met. Operations is to control, by procedure, (Administrative Procedure 1002) that tagging is accomplished, and returning of the repaired system to service is accomplished after a retest. (Ref. 39) The signed, completed work request is returned through QC, if the system is QC classified, to the Maintenance Supervisor for final closeout signatures. The originator or the originators supervisor is not involved unless the need or results of the work requires his expertise.

The Operations Shift Foreman approves the commencement of work. The Maintenance Foreman is responsible for having Operations perform the tagging and reporting status, as work progresses, to Operations.

Thus, Maintenance initiates tag control, and Operations maintains control for the use of the tags. The maintenance and documentation of tags and tag status is kept by Operations to ensure operational knowledge of the plant component/system conditions at all times.

The maintenance function is carried out by each discipline of maintenance under a supervisor who has foremen who ultimately assign the crafts to perform the work. The day shift consists of supervisor, foreman, and craftsmen of the same discipline; the remaining two shifts are composed of a foreman in one discipline with craftsmen from the three major disciplines to perform the work. The personnel are assigned specifically to Unit 1 or to Unit 2, but perform work as assigned to either unit. During active periods such as an

outage it is not unusual for most craftsmen to be assigned to the unit in an outage status.

Operations maintains an ongoing list, called a "no name outage," for high priority work to be performed in the event of an unplanned outage of an operating plant. At the time of a plant trip, maintenance reviews the listed work and plans for work to be performed.

1.6.2 Selected Component Maintenance History Immediately Prior to March 28, 1979

1.6.2.1 Main Feedwater System

The inspector reviewed selected files located in the Unit 2 Mechanical Maintenance Office on May 8, 1979. Information in the files on the feedwater system included work performed on feedwater pump FW-P-1B, work request No. WRC0180, December 13, 1978, related to a stuck shaft problem, and WR1357 which involved the feedwater pump lube oil system. The file contents did not indicate that the feedwater system was not in acceptable operational status prior to the March 28, 1979, event. A work request WR 0334, which addresses FW-P-1B turning gear, remains open.

1.6.2.2 Electromatic Relief Valve.

1.6.2.2.1 Valve Control Revisions

Information pertaining to the Unit 2 installed Electromatic pressurizer relief valve was reviewed. The relief valve is an electrically-actuated pilot operated pressure relief device. This valve is manufactured by Dresser Industries, Inc. It is installed in conjunction with a pressure-sensitive element having set pressure values for electrical signal actuation of a solenoid that actuates a pilot valve allowing for automatic pressure relief. The valve had leaked in

place during the hot functional test of the facility. Corrective maintenance was performed and testing was accomplished.

A QC surveillance report (77-266, October 27, 1977) documents the testing of the electromatic pressurizer relief valve. (Ref. 43) The bench test included electrical checks, pressure application to 2300 psig, and electrical actuation of release and, after release, leak checks.

The original wiring of the installed Unit 2 electromatic pressurizer relief valve was modified (ECM No. S-5934, EMOV), to allow visual control room indication of power across the pilot operator solenoid. This does not provide the operator with valve position, only demand position. The loss of power logic was also changed to have the valve remain in the nonactuated state upon a loss of electrical power. The original installation did not have these features and had resulted in the actuation of the valve during a loss of power occurrence on March 29, 1978. The followup action to the March 29, 1978, event included the performance of corrective action as stated in LER 78-21/3L. Based on discussions with the licensee, the change included the indication of power through the pilot valve solenoid, which appears to function properly.

The investigator reviewed B&R Field Questionnaire, FQ, No. 2529, April 6, 1978, which addresses ECM S-5934 which specified the circuit revisions to be made. The FQ, page 2, states that "Unit wired up per the prints and tested out." Documentation was not available about how and when the retest was done and the results. The Shift Foreman's Log Book, May 6, 1978, contains an entry 1615, page 488, which indicates the "I&C checked out RC-R2 indicating light on Panel 4."

Evaluation

Station Administrative Procedure 1021, Plant Modifications, Revision 5, June 28, 1978, paragraph 4.1.a, requires that "the Supervisor of Maintenance receives a work request which represents a change/modification to a system or component." Documentation of a Work Request was not available. The original

FQ does not designate the change as either minor or major. The lack of designation of the type of change and the absence of a work request is contrary to the requirements of Station Administrative Procedure 1021, Plant Modification. This procedure, if followed, designates if the change/modification must be included in operator training. Failure to follow this procedure is being considered as a possible item of noncompliance with Technical Specification 6.8.1.a.

The technical specification 6.8.1.a requires written procedures shall be implemented covering activities referenced by applicable procedures recommended in Appendix A of Regulatory Guide 1.33, November 1972. Appendix A of Regulatory Guide 1.33, addresses typical safety related activities to be covered by written procedures. Paragraph 9a of RG 1.33 states that "maintenance that can affect the performance of a safety-related equipment should be properly planned and performed in accordance with written procedure...appropriate to circumstances".

Station Administrative Procedure 1021, Plant Modification, Revision 5, June 28, 1979, paragraph 4.1a) requires that "the Supervisor of Maintenance receives a work request which represents a change/modification to a system or component".

Contrary to the above there was no work request for the work performed in completing ECM S-5934 which specified changes to the pressurizer relief valve RC-R2 including the addition of an indicating light and a revised logic on loss of power actuation of the valve.

1.6.2.2.2 EMOV Locations and Handling

During discussions with the licensee and contractors about the pressurizer electromatic relief valve, it appeared that both the Unit 1 and Unit 2 valves had been used on Unit 1 pressurizer during the course of earlier Unit 1 operation. The valves are interchangeable. Both are Dresser Type 2-1/2 31533VX-30.

The investigator reviewed Drawing No. 2841-60501, March 20, 1979, which shows Serial No. BL-08905 for B&W Contract No. 620-005, (Unit 1). The valve was visually inspected in place and the serial number was checked by the investigator to be BL-8905. The serial number for the installed Unit 2 valve is believed to be BN-4233. The Unit 2 B&W Contract Number is 620-006. United Engineers and Constructors, Inc., Order No. 022660LS, does not list the serial number of the valve in place on Unit 2. The investigator inspected the third electromatic valve located in storage. The Serial Number of this valve is BS0-3989. It is concluded that electromatic valve Serial Number BN-4233 is on the Unit 2 pressurizer, the Unit for which it was originally ordered.

Documentation reviewed indicates that valve BN-4233 was sent to the manufacturer for inspection, refurbishing, gasket replacement, reassembly, and testing as documented in Jersey Central Power and Light Purchase Order No. C-0224, June 8, 1976.

Evaluation

The valve history indicates that both electromatic valves have been refurbished and each has been installed on Unit 1 pressurizer. The current status is that the original valves as purchased under the two B&W contracts are installed on their respective pressurizers.

1.6.2.3 EMOV Block Valve.

The pressurizer electromatic relief valve upstream block valve, RC-V2, had work performed on it on September 14, 1977. (Machine History File, WR 1166; Ref. 45). The motor operator was removed to allow repacking the valves. The valve was cycled. Discussions with licensee personnel indicate that there has been a concern that the block valve could stick shut or open if used too often.

1.6.2.4 Emergency Feedwater Pumps.

The machine history file on the electrical emergency feedwater pumps, EF-P-2A and 2B, did not indicate a problem relative to the event of March 28, 1979. The most recent history indicated that motor windings and connecting electrical cables were checked after an event involving steam leakage in the pump area. The steam leakage was due to the atmospheric dump valves (MS-V3A and B) bellows (MS-U7A and B) failing during turbine trip testing on January 15, 1979, and subsequent reactor trip and loss of condenser vacuum. (TMI Reactor Trip Report January 15, 1979, Ref. 6). The steam was reported to have filled the M-20 area where the pressurizer breaker, silicon controlled rectifier (SCR) panels are located as well as the steam-driven emergency feedwater pump, EF-P-1. The adjacent area where EF-P-2A and 2B are located also contained a steam environment.

Checks and repairs were made to equipment in these areas. The checks on the pumps were performed by Catalytic, Inc. The work was completed on January 23, 1979, WR C0711. (Ref. 7) Checks were made on electrical reactor building penetrations in the M-20 area for heat and moisture damage. (WR C0718; Ref. 8). This work request and the Maintenance Procedure Format and Approval form TMI-84-2-78 were reviewed. The documentation indicates that visual checks were completed January 16-17, 1979. Review of the documentation indicates that no problems were found during these visual checks.

1.6.2.5 Pressurizer Heaters.

The Licensee performed maintenance (WR 709) involving the repair of pressurizer heater panels. This effort resulted in the repair of the pressurizer heater transformers and repairs to cabinet 324. The repair description was not specific. Discussions with licensee personnel indicate that there have been problems with the pressurizer heater breakers. It is believed that the problem is associated with the high temperature in the M-20 area due to the proximity of the main steam lines in this area. It was further noted that

control room indication is not available for individual breakers which are subsets of main breakers indicated in the control room. The primary control room indication of individual heater breaker problems is indirect in that it is related to pressurizer pressure recovery rate.

1.6.2.6 Waste Tank Rupture Discs.

The investigator reviewed the Unit 2 maintenance request log for the past year and discussed with the mechanical maintenance department personnel what they knew of a work request to correct rupture discs for auxiliary building sump tank (ABST) WDL-T-5 and miscellaneous waste tank WDL-T-2. Application for Apparatus to be taken out of service, control No. 2385, May 10, 1978, was found in the control room active tag log. This document notes the replacement of a rupture disc as work to be done. The document does not reference a work request number in the blank provided. A work request No. C1117, "WDL-U224, replace ruptured diaphragm," February 8, 1979, is currently an open work item. It was stated by Licensee personnel that there were problems with the rupture discs for these tanks that may possibly be traced to the preoperational phase of TMI-2. (The discs are manufactured by Zook Enterprises and are rated at 19.6 psig \pm 2% at 104°F.) An order for spare discs (P.O. No. JCP 0C211, February 11, 1976) is in existence (Ref. 10). The discs are interchangeable for the tanks. The information indicates that tank WDL-T-5 had a ruptured disc on March 28, 1979. (The plan drawings, B&R Drawing No. 2485, Rev. 9, Radwaste Disposal - Miscellaneous Liquid Auxiliary Building Sump Tank Plan and Section, and B&R Drawing No. 2482; Rev. 13)

1.6.2.7 Emergency Feedwater Valves.

The investigator reviewed work request No. 756, January 16, 1979, requiring maintenance checks of emergency feedwater valves 12A and B, 32A and B, and 33A and B. (The 32 valves bypass the 11 valves and the 33 valves bypass 11 and 12 valves; Burns & Roe Drawing 2005; Ref. 50) The work request was returned to the Operations Shift Foreman on January 24, 1979. The information contained in

the work request indicated that the valves remained to be cycled to complete the assigned work. The information available did not include Meggering of valve operator cables to validate acceptance criteria contained in the work request procedure. (Station Preventative Maintenance Procedure E-2, Dielectric Check of Insulation, Motors and Cables, Ref. 51). Because of the apparent incomplete data and the lack of documented cycling of the valves, the investigator requested tagging documentation. Licensee personnel did not locate the tagging document. Catalytic, Inc., did locate copies of Application for "Apparatus to be Taken Out of Service" No. 4030, January 18, 1979, and No. 4044, January 20, 1979. Both applications were incomplete and would require Operations personnel for proper completion. The investigator did not find suitable documentation as to whether valve cycle, Megger checks, tagging clearance, and acceptance tests to return the valves to service were performed for the listed valves. The Unit 2 maintenance log carries this work request in an open status as of June 20, 1979.

Evaluation

Station Preventative Maintenance Procedure E-2, paragraph 4, requires the use of TMI Administrative Procedure No. 1002, Tagging. Procedure E-2 presents Megger acceptance criteria. The work request documentation failed to validate acceptable values. In that the 12A and 12B valves are known to have been operated and documentation is lacking for retest, satisfaction of acceptance criteria, and tagging clearance requirements, this item is under consideration as a potential item of noncompliance.

Procedure 1002, Revision 14, paragraph 4 "Restoration to Service", states:
"Restoration to Service:

"When work has been completed, grounds removed, and all parties that have received clearance have reported clear, and the Shift Foreman has deemed that the equipment is ready for service, the Switching and Tagging CRO shall so note on the Clearance Control Document. This notation will include the time, date and a statement as to the status and position of

the isolated equipment.... The Shift Foreman shall then sign the original application, indicating that the equipment is approved for operation."

1.6.2.8 Pressurizer Safety Valves.

The investigator reviewed licensee records pertaining to pressurizer safety valves RC-R1A and 1B. WR 340, March 25, 1977, contains results for lift tests of these valves. These results indicate that the settings are within the acceptance criteria $\pm 1\%$ of the 2450 psig setpoint (Int. 139).

1.6.3 References

The references used in this report section are:

109, 99, 55, 100, 110, 111, 112, 113, 114, 115, 116, 117, 118, 119, 120, 121, 122, 123, 124, 125, 126, 127, 128, 129, 130.

2. OPERATOR ACTIONS DURING ACCIDENT

2.1 General

In this section, the use of the term "operator" is being applied liberally to address actions by those licensee staff members, including management level personnel, who are assigned routinely to shift operation responsibility. Subsections 2.2 - 2.13 will identify the major procedural requirements applicable during different phases of the accident, and the manner in which these requirements were addressed by the operators.

2.2 Operator Actions Following Loss of Main Feedwater

Through records review, analysis, and personnel interviews, it was determined that the immediate and following actions prescribed in the procedure (2202-2.2, Loss of S/G Feed, Revision 3, October 13, 1978; Ref. 26) for the loss of Main Feedwater flow to both OTSGs were appropriately addressed as discussed below:

- o When loss of FW is due to loss of both feed pumps, trip the reactor.

The reactor tripped automatically due to high RCS pressure and the full rod insertion was verified by the operators immediately after the trip. The manual reactor trip was delayed approximately 20 minutes. The interviews revealed that this action was delayed because of the rapid sequence of events following the turbine/reactor trip. The reactor trip is treated in Section 2.4 of this report (Int. 3, 15, 9).

- o Verify that turbine trip and stop valves closed.

Turbine trip procedure (2202-2.2; Ref. 28) covers this activity. This is treated in detail in Section 2.3 of this report.

- o Verify that EF-P-1, EF-P-2A, and EF-P-2B start as evidenced by pump discharge pressures.

The operator noted the pumps as operating and "normal" (Int. 4, 9, 61).

- o Verify that emergency feedwater valves (EF-V11A and V11B) are in automatic and controlling OTSG level at 30 inches on the SU range indication.

The operator actions relative to this requirement are described in detail in Section 2.3 of this report.

Evaluation

The investigation concluded that the major procedural requirements applicable to this event were appropriately addressed in the actions taken by the operators immediately following a loss of all main feedwater to the OTSGs. This conclusion does not include those items which are covered in Section 2.3 and 2.4 of this report.

2.3 Operator Actions Following Turbine Trip

The TMI-2 unit experienced an analyzed turbine trip as a result of the loss of main feedwater and experienced a loss of all feedwater for approximately 8 minutes because the emergency feedwater header isolation valves were apparently in the closed position at the onset of the turbine trip transient.

Through records review, analysis, and personnel interviews, it was determined that the immediate and followup actions for a turbine trip on (Emergency Procedure 2203-2.2, Turbine Trip, Revision 7, October 25, 1978; Ref. 28) were addressed by the operators as appropriate including:

2.3.1 Immediate Operator Actions

- o Verify that the turbine stop valves are closed and generator breakers and field breakers are open. Verify that the start of the seal oil backup pump,

the turbine gear oil pump and the bearing lift pumps and the closure of the extraction steam valves.

The operator noted that the turbine trip was normal with the exception of one turbine stop valve which did not indicate closed. The associated control valve was verified closed and, additionally, the turbine was tripped manually. (Int. 4, 9, 61)

- o Monitor pressurizer level, RC pressure, and RC temperature.

After approximately 2 minutes, the high pressure injection (HPI) initiated. As the pressure decreased below 1600 psig, the pressurizer level continued to increase to full scale. The operators were not surprised by the HPI actuation but they did not understand the high pressurizer level and the continuing low RCS pressure. The initiation of ES is covered in more detail in Section 2.5 of this report. (Int. 5, 6, 17, 38, 3, 15, 4, 9, 61)

- o Notify HP/Chemistry to sample RC letdown for dose equivalent iodine between 2 and 6 hours after a power change of greater than 15% power in a one hour power per Surveillance Procedure 2304-302.

Upon the announcement of the turbine reactor trip, the chemists on duty insure that letdown sample is placed in recycle in preparation for obtaining a RCS/letdown system sample. (Int. 5, 17, 38, 2, 14, 37)

- o In the event that the vacuum is lost in the main condenser, verify atmospheric reliefs are controlling header pressure at 1010 psig and reducing reactor power to zero (hot standby). Add required makeup to tank as Tave is reduced to 532°F (885 psig OTSG pressure).

Initially following the turbine trip the main steam code safeties lifted, OTSG pressures reached a maximum 1070 psig, and the operator noted the main steam turbine bypass valves functioning to maintain steam header pressure.

The operator interviews indicated that the main turbine bypass valves (MS-V25A, 25B, 26A, and 26B) were placed in manual and cracked open (apparently at 1.2 minutes) when the operators perceived the valves were not responding adequately to the increasing OTSG pressures. (Int. 4, 9, 61)

- o If turbine trip is due to loss of both feed pumps, verify that emergency feed pumps have started and are delivering water to the OTSG.

The operator noted that all three emergency feed pumps had started and appeared normal with the OTSG levels decreasing and the level control valves EF-V11A and 11B traveling (dual indicating lights). The controllers for the emergency feedwater control valves (EF-V11A and V11B) are located above the lights for the EF-V12A and V12B valves on the same panel. The operator did not note the fact that the EF-V12A and V12B valves were closed at that time. (Int. 4, 9, 61)

The operator watched the steam generator levels for a sign of feedwater and water level increase. After observing no level change the operator quickly reviewed the valve lineup, noting the emergency feedwater control valves (EF-V11A and V11B) opening. The operator placed the controls in manual and raised the demand signals to full open in the attempt to increase the rate of emergency feedwater delivery to the OTSGs.

Evaluation

At approximately 5 minutes into the turbine trip transient, the operator noted that the emergency feedwater level control valves were continuing to travel to the open position. The valves were not supplied adequate air to operate. The valves fail "as is" and only respond if air pressure is available. It can be reasonably postulated that the failure of the condensate polisher outlet valves to the closed position (loss of instrument air pressure) and the failure of the emergency feedwater valves to respond normally (loss of instrument air pressure) were related. Operator interviews revealed that the valves did

not appear to operate properly. The valves were expected to open fully shortly after the OTSG levels decreased below 30 inches and the valves were continuing to travel at about 0405 hours.

2.3.2 Followup Operator Actions

The operator noted the continuing low levels (approximately 10 inches on the startup range, which was assumed to be dry).

At this time in the event (approximately 5 minutes) after the OTSG levels had remained below 30 inches on the startup range from 30-40 seconds after the start of the accident, the feedwater control valves had sufficient time to stroke fully open from their normally closed position.

The Unit 1 Shift Supervisor arrived in the area at about 0408 hours, noted the low steam generator levels, and stated the concern of no apparent delivery of feedwater. Simultaneously, the operator had noted the emergency feedwater header isolation valves (EF-V12A and 12B) closed. The operator announced the condition, and simultaneously opened both feedwater header isolation valves (EF-V12A and V12B). The operator stated that he had looked at the panel valve lineup twice before discovering the valves to be closed. The first scan of the panel was made standing close to the panel and leaning over the console, possibly obstructing his view of the lower valve (EF-V12B). The upper valve (EF-V12A) indicator light (green for closed) was covered by a caution tag hanging from the main feedwater pump 1B miniflow valve (V16B) control station.

NOTE: The caution tag indicated that the control for the main feedpump 1B miniflow valve (V16B) should remain in MANUAL due to a flow instrument problem. The tag was issued on March 20, 1979, for the transmitter valve (CO-V53D) repair (Work Order 1735). (Int. 4, 9, 61, 5, 17, 38, 3, 15).

A review of Administrative Procedure 1037, Revision 1, August 23, 1978, Caution and Do-Not-Operate (DNO) Tags, revealed that no formal guidance is given concerning the method of hanging tags to prevent inadvertent covering of

other information devices (lights, identification tags, etc.) or impeding the operation of other equipment or controls.

Immediately following the opening of the emergency feedwater header isolation valves (EF-V12A and V12B), the operator noted flow noises from a Loose Parts Monitor channel (apparently from the main steam area and/or A OTSG), an increase in the OTSG pressures above the low of approximately 780 psig (8 minutes), and a decrease in the RCS temperature during the subsequent 12 minutes from 597°F to 555°F. (Int. 4, 9, 61)

Evaluation

Technical Specification 3.7.1.2 requires that three independent steam generator emergency feedwater EFW pumps and associated flowpaths be operable in Modes 1, 2, and 3 (OTSG pressure greater than 800 psig). The specific action does allow one emergency feedwater system to be inoperable, provided the inoperable system is restored to OPERABLE status within 72 hours or the reactor is to be placed in hot shutdown within the next 12 hours. Prior to 0400 hours on March 28, 1979, the licensee did not provide a formal routine shift check of engineered safeguards equipment, including the status of the emergency feedwater pumps and valves. Additionally no alarms were provided to annunciate the emergency feedwater system in the inoperable condition.

The plant was operated with the emergency feedwater system in the operable condition as a result of the emergency feedwater isolation valves (EF-V12A and 12B) being closed for a period of time prior to 0408 hrs on March 28, 1979. The continued plant operation with the emergency feedwater header isolation valves (EF-V12A and 12B) in the closed position is an apparent breakdown in the licensee controls over the operability of safety related equipment, and is being considered as a potential item of noncompliance pursuant to Technical Specification 3.7.1.2.

A review of plant records and charts revealed that the recovery of the A OTSG level to 30 inches at 0420 hrs and the B OTSG level to 30 inches at 0425 hrs was delayed because the emergency feedwater flow was apparently limited or

ineffective in reestablishing the OTSG levels even with full emergency feedwater flow available. (All three emergency feedwater pumps operating.)

The steam generator pressures did not recover to the turbine bypass valve setpoint of approximately 1010 psig (or the safety valve setpoint of 1070) or provide significant cooling rates as Tav_g decreased from 597°F to 547°F. The difference between the actual steam generator pressures and the RCS average temperature which determined OTSG saturation pressure, indicate that the OTSGs were "dry."

During this period, between 0408 hrs and 0416 hrs, all three emergency feedwater pumps were operating with discharge pressures greater than OTSG pressure. Their discharge pressures show distinct step changes simultaneously. This uniform variation in EFW pump discharge pressures indicated manual throttling or simultaneous stepwise movement of two automatic valves since OTSG levels were less than 30 inches.

One operator interview revealed that the emergency feedwater flows were maintained at the maximum amount (not throttled) until levels were responding and recovering. However, another interview indicated that the emergency feedwater (EFW) was throttled to limit the thermal shock of the OTSGs and minimize the effect on the RCS pressure and temperature. (Int. 4, 9, 61)

The investigation concluded that the extended times (13 minutes for the A OTSG and 17 minutes for the B OTSG) for the restoration of minimum OTSG levels minimum to more than 18 inches (30 inches normal low limit setpoint) was the result of throttling the EFW flow. The conclusion was based on operator interviews and an analysis of EFW, OTSG, and RCS parameters.

At 0532 hrs during the accident, the A OTSG level decreased to 10 inches (minimum level indication) between 0532 hrs and 0543 hrs, at which time the A OTSG level was increased to about 50% on the operating range to support natural circulation (Ref. Section 2.10). The failure to maintain OTSG A level at 30 inches was due to the feedwater level control valve (EF-12A) in manual and

shrinkage due to a decrease in the RCS temperature 565°F to 535°F. (Int. 4, 9, 61).

Emergency Procedure 2203-2.2, Turbine Trip, Step 3.2 and 3.4 requires the operator to adjust feedwater flow to control OTSG levels at 30 inches (on the startup range).

Contrary to this requirement, the A and B OTSG levels decreased below 18 inches (reaching as minimum of 10 and 12 inches) following the trip until approximately 0420 hrs and 0425 hrs respectively, and the A OTSG level was below 18 inches (reaching a minimum, of 10 inches) between 0532 hrs and 0543 hrs. This failure to maintain required steam generator level, with adequate emergency feedwater available, is being considered as a potential item of noncompliance pursuant to Technical Specification 6.8.1.a.

2.3.3 Additional Operator Actions

After the OTSG levels were established at 30 inches on the startup range, the operator continued to maintain the level in manual control using the emergency feedwater (EFW) system as required with suction for the EFW pumps from the condensate storage tank between 0400 hrs and 0459 hrs. The suction for the pump was then transferred to the condensate pump operating. The condensate pump (CO-P1B) continued to operate during the initial minutes of the event. (Computer Memory Trip Review and Int. 4, 9, 61)

The turbine-driven emergency feedwater pump was stopped at approximately 0426 hrs after the operator noted the "B" OTSG level increasing slowly above the low-level limits.

The shutdown of the turbine driven EFW pump placed the system into ACTION "A" of Technical Specification 3.7.1.2. Operating Procedure 2102-1.1, Revision 14, August 25, 1978 (Unit Heatup), Step 4.53, required the EFW system to be placed in operation at 800 psig in the OTSG. Procedure 2102-3.2, Revision 10, August 29, 1979 (Unit Cooldown), Step 4.14a, removes the EFW system from service to 600 psig in the OTSG.

The motor driven emergency feedwater pump (EF-P2B) was similarly stopped at 0436 hrs.

The emergency feedwater pump (EF-P2A) was tripped at 0711 hrs after the OTSG levels were near the 55-65% level and the OTSG pressures were less than 600 psig. During the operation of the RCP (RC-P2B) for 19 minutes at 0654 hrs, it was noted that the "B" OTSG pressure increased to about 750 psig but not above the Technical Specification of 800 psig, with all of the EFW pumps secured.

The emergency feedwater pump (EF-P2A) was started at 0735 hrs and operated until 0842 hrs to feed the "A" OTSG as needed to maintain 50% level on the operating range.

The "A" OTSG level was raised at 1000 hrs to 94-97% on the operating range using the water from the hotwell via the condensate pump discharge/booster pump suction pressure to enhance decay heat removal. The "A" OTSG pressure during that period decreased from 140 psig to 20 psig and then to 0 psig by 1200 hrs. The emergency feedwater pump (EF-P2A) was started and operated for 8 minutes at 1109 hrs to assist in feeding the "A" OTSG which was near the discharge pressure of the condensate header.

The "B" OTSG level was increased from 60% to 95% on the operating range at 1600 hrs using the emergency feedwater pump (EF-P2A) at 1534 hrs for 18 minutes in preparation for the continued plant shutdown.

Evaluation

The stopping of the steam driven emergency feedwater pump (EF-P1) at 0426 hrs and the motor-driven emergency feedwater pump (EF-P2B) (secondary emergency feedwater pump stopped) by placing the control switches for the steam supply valve (MS-V207) and the pump switch (EF-P2B) to the PULL-TO-LOCK position (nonautomatic) at 1040 psig in the OTSGs (greater than 600 psig) is a violation of the operating procedure for unit cooldown, Step 4.14a. The A OTSG pressure remained above 800 psig until approximately 0545 hrs. The emergency feedwater

pumps start automatically upon loss of two main feedwater pumps if the control switches are in the NORMAL-AFTER-STOP position.

Technical Specification 3.7.1.2 requires that the Emergency Feedwater System be "OPERABLE" in Modes 1, 2, and 3 above 800 psig OTSG pressure with two flow paths, two motor-driven emergency feedwater pumps, and one steam-driven emergency feedwater pump. The failure to maintain the emergency feedwater system in the fully operable status subsequent to 0426 hrs (turbine-driven emergency feedwater pump-EF-P1) stopped, and 0436 hrs (motor-driven emergency feedwater pump-EF-P2B) stopped, by placing the control switches to the non-automatic (PULL-TO-LOCK) position, at 1040 psig in the OTSG (greater than 800 psig) is being considered as a potential item of noncompliance.

2.4 Operator Action Following Reactor Trip

The TMI-2 unit experienced a routine reactor trip as a result of a loss of the turbine when the main feed pumps tripped on loss of suction pressure.

The reactor continued to operate in accordance with control system design at approximately full power for 8-9 seconds when the reactor was tripped as the RCS pressure increased above 2355 psig, the high-pressure reactor trip setpoint. (2202-1.1, Revision 5, October 6, 1978, Reactor Trip, step 1.1.e; Ref. 31)

Through records review, analysis, and personnel interviews, it was determined that the immediate and followup actions for a reactor trip (Emergency Procedure 2202-1.1, Revision 6, October 25, 1978, Reactor Trip; Ref. 30); were approximately addressed by the operators, including:

- o Manually trip the reactor.

The manual reactor trip was delayed approximately 20 minutes due to the sequence of events. (Int. 3, 15)

- o Verify that all "in-limit" lights are actuated (except group 8).

All "in-limit" lights were actuated. (Int. 3, 15, 4, 9, 61)

- o Close letdown isolation valve MU-V376.

The letdown isolation valve (MU-V376) was closed by the operator.
(Int. 4, 9, 61)

- o Start second makeup pump and open MU-V-16B as necessary to maintain 100 inches in the pressurizer.

The operator started the second makeup pump (MU-P1A) and also opened the suction valve (DH-V5A) from the Borated Water Storage Tank (BWST) for the two operating makeup pumps (MU-P1A and P1B) in anticipation of the expected low pressurizer level and low RCS pressure transient that follows a reactor trip. The integrated control system normally returns the RCS average temperature from the operating (greater than 15 percent) value (582°F) to the initial hot shutdown value (549°F), which produces a decrease in the pressurizer level from 240 inches to 85 inches (about 5 inches per °F) and an RCS pressure at or near 1640 psig (ESF initiation setpoint). During the rapid sequence of events, the operators did not realize that the pressurizer level decreased to only 158 inches and RCS pressure was stabilized for about 60 seconds at approximately 1500 psig. (Int. 5, 6, 7, 38, 3, 15)

- o Verify that pressurizer heaters are off at 80 inches in the pressurizer.

[The pressurizer minimum level was 158 inches.]

- o Announce reactor trip on page system.

The Shift Supervisor announced turbine trip-reactor trip. (Int. TMI 6, 7, 60, 61, 96)

- o Monitor makeup tank level and maintain level higher than 55 inches by using waste transfer pumps (WDL-P5A/B) and feeding from an RCBT with boron concentration equal to or greater than RCS concentration.

The operator opened the suction valve (DH-V5A) to the BWST to provide makeup pump suction (MU-P1A and P1B) in anticipation of the expected pressurizer level transient.

- o Verify that the pressurizer heaters and spray have returned RCS pressure to normal operating pressure of 2155 psig.

The pressurizer heaters and spray were returned to AUTO by the operator after the reactor trip to provide normal pressure control and response to the transient. The spray valve and the heaters were being maintained in manual (2103-1.3, Pressurizer Operation, Revision 1, November 1, 1977, step 4.2.4) to continuously recirculate the pressurizer. Boron concentration was consistently higher than that in the RCS because of the leak through the pressurizer relief valves (exhaust tailpipe temperatures indicating consistently above 180°F) and concentrating the boron in the pressurizer (Ref. 53). The operators indicated that this boron concentration in the pressurizer was often as much as 200-300 ppm higher than that in the RCS. The RCS pressure did not recover to normal (2155 psig) during the event under normal pressure control. (Int. 5, 17, 38, 3, 15, 2, 14, 37)

- o Reduce pressurizer level setpoint to 100 inches (25%).

The pressurizer level increased to full scale or near full scale within 6 minutes. The operators did not understand the conditions. (Int. 5, 6, 17, 38, 3, 15, 2, 14, 37)

- o Verify normal electrical lineup, i.e., no substation or inplant distribution breakers are open (except generator breakers).

The operators verified normal electrical system conditions.

(Int. 4, 9, 61)

- o Check that all RMS channels are normal and that no unplanned or uncontrolled radioactive release is in progress.

The only radiation alarm that annunciated shortly after the reactor trip [at approximately 0420 (ALERT) and 0500 (HIGH)] was the intermediate closed cooling system monitor for the letdown heat exchangers. This alarm was not considered unusual because of its location near the containment sump, which was filled from the RCDT rupture disc discharge. Previous experience had also noted that this monitor alarmed following a unit trip due to increased reactor coolant activity levels (crud or iodine spikes). The operator did note that the high alarm was annunciated rather than the alert alarm as occurred in the past, but the alarm was still considered normal. (Int. 5, 6, 17, 2, 14, 37, 3, 15, 4, 9, 61)

- o Compute shutdown margin calculations per 2103-1.9 (Reactivity Balance Calculations). If shutdown margin is less than 1%, boron should be added to the RCS.

Based on the rod worths and positions at the time of the reactor trip (see Section 1.33, Details I), the shutdown margin was satisfactory with all rods inserted and 1026 ppm RCS boron concentration. (Int. 5, 6, 17, 97)

- o If reactor startup is not intended within four hours, raise OTSG level to 97-99% on the operating range using the feedwater valve bypass and the tubesheet drains for level control.

A return to power was initially planned following the trip, but the system conditions quickly deteriorated. The OTSG levels

were increased to the 97-99% level later in the day. (A OTSG at 1000 hrs and the B OTSG at 1600 hrs). (Ref. Section 2.4 of this report)

- o Notify HP/Chemistry to sample RC letdown for dose equivalent iodine between 2 and 6 hours after power change greater than 15% within one hour period per Surveillance Procedure 2304-3D2.

Following the announcement of the reactor trip, the onshift chemistry technicians placed the RCS letdown sample line on recirculation and obtained a sample. (Int. 5, 6, 17, 38, 3, 15)

Evaluation

The investigation concluded that the major procedural requirements applicable to this event were appropriately addressed in the actions taken by the operators immediately following the reactor trip on high pressure.

2.5 Operator Actions Following ES Actuation

The high-pressure safety injection initiated at a reactor coolant system pressure of approximately 1600 psig (decreasing) following the reactor trip at approximately two minutes after the start of the accident.

The automatic initiation of safety injection due to low RCS pressure following a turbine-reactor trip was not foreign to the operators, based on previous plant experience. The turbine trip transient the TMI-2 unit was characterized by the operators as "severe", resulting in a substantial decrease in pressurizer level (expected to decrease below 80 inches and trip the pressurizer heaters) and RCS pressure (expected to decrease below the ESF actuation trip point of approximately 1640 psig). (Int. 5, 6, 17, 38, 2, 14, 37, 3, 15, 4, 9, 61)

Through records review, analysis, and personnel interviews, it was determined that, with two major exceptions, the majority of the immediate and

followup actions for the procedure Loss of Reactor Coolant/Reactor Coolant System Pressure (2202-1.3, Revision 8, May 12, 1978, Sections A and B), were appropriately addressed. These procedure sections addressed actions considering "automatic" initiation of the engineered safety feature systems, which occurred at 0406 hrs (Ref. 34).

The symptoms of a significant leak or rupture resulting in an automatic initiation of the ES systems include a rapid continuing decrease of reactor coolant pressure, with the following indications:

- a. Lo alarm 2055 psig.
- b. Lo-lo alarm 1700 psig.
- c. Safety injection actuation at 1640 psig.
- d. Lo alarm 200".
- e. Lo-lo alarm 80" (interlock heater shutoff).

Within two minutes (initiation of safety injection at approximately 1640 psig), the RCS pressure had peaked at approximately 2435 psig and decreased rapidly to 1900 psig at 15 to 20 seconds. RCS pressure then decreased more slowly from 1900 psig to 1600 psig at 20 to 120 seconds. During this time interval RCS pressure stabilized at approximately 1750 psig for about one minute. This pressure stabilization indicated the formation of voids within the reactor core, upper vessel head, or upper OTSG and hot leg ("candy cane") and continued as the OTSGs secondary side boiled dry in the 1-2 minute time frame. RCS pressure continued to decrease after ESE initiation to a low of approximately 1400 psig at 2 to 6 minutes. Pressure then increased from 1400 psig to 1600 psig at 6 to 8 minutes as the operators throttled the HPI flow to minimum and removed one (MUP-1C) HPI pump from operation. At this point, the bulk saturation temperature of the coolant determined the RCS pressure. RCS pressure subsequently decreased from 1600 psig to 1100 psig from 8 to 18 minutes as the emergency feedwater was initiated to the OTSG and the RCS average temperature decreased from 597°F to 555°F. (Reference: See Appendix I-A for the time period described; 0-18 minutes).

The operators had noted the pressurizer level increase to 258 inches at 12 seconds followed by a decrease to 158 inches from 12 to 40 seconds, after which the level commenced a continuous rise to full scale indication from 1 to 5 minutes. The HPI flow between 2 and 5 minutes was throttled to minimum and one HPI pump stopped (MUP-IC), based on the increasing pressurizer level and RCS pressure decreasing very slowly toward 1400 psig (6 minutes). With the pressurizer level off scale high and the RCS pressure low (1400-1600 psig from 6 minutes through 18 minutes), the operators concluded that the problem was the collapse of the pressurizer bubble during the severe transient. Substantial time and effort was expended attempting to regain the bubble and pressurizer level using the pressurizer heaters and letdown system.

The operators had previously experienced pressurizer heater breaker failures, ascribed by the plant staff to be due to hot and humid local (control building M-20 area) conditions therefore, the lack of initial response of the RCS pressure to the heaters was not unexpected by the operators.

Based on previous experience and training the operators expected the pressurizer level to decrease below 80 inches following the transient. They did not understand why the level decrease ceased at 158 inches and began to rise steadily. (Int. 5, 17, 38, 2, 14, 37, 3, 15, 4, 9, 61)

The continuing high pressurizer level caused the operators to conclude that no RCS inventory problem existed. This decision was based on a number of other events that had occurred but were considered normal or not understood. These items included:

- o High radiation alarm in Reactor Building.

No high radiation alarm was received (the HPR-227 monitor was subsequently reported flooded, under positive pressure, and manually isolated at 0600 hrs). (Int. 5, 17,38)

o Reactor Building (RB) Ambient Temperature Alarm.

An increase in RB ambient temperature was noted (5 minutes), and high temperature fire alarms were received and evaluated by the Shift Supervisor who was in the control room at that time. That Shift Supervisor was aware at that time that the rupture disc had blown on the RCDT following the initial turbine trip transient and normal opening of the EMOV (RC-R2). (Int. 5, 17, 38)

o High Reactor Building Sump Level.

The high RB sump level was attributed to the blown rupture disc on the RCDT after the opening of the EMOV (RC-R2). (Int. 5, 17, 38, 3, 15, 94)

o High Reactor Building Pressure (RCS or main steam line rupture).

With the higher-than-normal RB pressure and temperature continuing to increase with no apparent RCS inventory problem (pressurizer level off scale high), the shift personnel became suspicious of a steam line or OTSG leak. Subsequent to the stopping of the RCPs in the B loop at 74 minutes, the B OTSG was isolated at 0527 when the B OTSG pressure was noted to be 300 psig lower than the A OTSG pressure. The operators noted a small decrease in the RB pressure increase at the time the B OTSG was isolated and felt that the steam (feed) leak to the reactor building had been isolated. (Int. 5, 17, 38, 2, 14, 37, 3, 15, 4, 9, 61)

o Rapidly Decreasing Makeup Tank Level.

The operators were maintaining the MUT level by cycling the makeup pump suction valve (DH-V5A) from the BWST. With the normal suction valve (MU-V12) open from the makeup tank, the makeup pump 1A or 1B suction was momentarily shifted to the MUT as needed to control the tank level. This mode of operation was familiar to the operators. (Int. 5, 17, 38, 3, 15)

- o The operators are required to verify that the high-pressure injection system is operating properly as evidenced by injection flow in all four legs (MU-V16A/B/C/D). Flow was indicated on MU23 FE1, 2, 3, 4.

The high-pressure injection system was verified to be operating properly with 250 gpm per HPI leg as evidenced by flow through all four injection paths. (Int. 3, 15)

The operators are also required to verify that the safety injection equipment is in its ESF position as shown in a table attached to the procedure.

The interviews indicated that the equipment was verified in the ESF position, but no documentation was available for this action. (Int. 3, 15, 2, 14, 37)

Procedure Section B, step 3.4, calls for bypassing the safety injection by depressing the group reset pushbuttons (step 3.5) and throttle MU-V16A/B/C/D as required to prevent pump runout.

NOTE: The emergency procedure is deficient as this step (3.5) alludes to HPI flows at less than 250 gpm.

The operators throttled the HPI valves in an attempt to control the pressurizer level without regard to the RCS pressure being below 1640 psig. They felt that the pressurizer level had been "caught" when the minimum was 158 inches from 40 to 60 seconds and started to increase. They further concluded that flow from the HPI system was not required based on the pressurizer level continuously increasing to full scale in 5 minutes. The operator throttled the HPI flow to minimum by approximately 4.5 minutes and immediately established maximum letdown flow (140-160 gpm) in an attempt to regain and control pressurizer level using the normal orifice path, the orifice bypass valve (MU-V5), and two letdown coolers.

Evaluation

The emergency procedures require the operators to trip reactor coolant pumps before pressure decreases below the net positive suction head (NPSH) requirements and before reaching 1200 psig. The reactor coolant system pressure decreased to 1200 psig within approximately 15 minutes after the reactor trip and remained between 1000 psig and 1100 psig until the last two RCPs (RCP-1A and 2A) were tripped (101 minutes) by the operators. The operators did not perceive that a RCS inventory problem existed because of the full pressurizer and felt that the bubble in the pressurizer had been collapsed (lost). The recovery of the RCS to normal was considered a matter of energizing the pressurizer heaters and providing letdown to reestablish pressurizer level and pressure. The operators did not feel they were involved in a LOCA condition. (Int. 5, 17, 38, 2, 14, 37, 3, 15, 4, 9, 61)

The failure to follow procedures (2202-1.3, Loss of RC/RCS Pressure Revision 8, May 12, 1978, Section B, Step 2.2.4) and trip the reactor coolant pumps at 1200 psig as required is under consideration as a potential item of noncompliance pursuant to Technical Specification 6.8.1.a (Ref. 35).

The rate of increase of pressurizer level slowed at approximately 5 minutes, and the letdown flow was returned to near normal (70 gpm) shortly afterwards; however, the pressurizer level continued to increase and went off scale (400 inches). With a high pressurizer level, the operators did not consider that a reactor coolant inventory problem existed, even though the system pressure was decreasing steadily. The opinion of the operators was that the bubble in the pressurizer had been collapsed during the severe plant transient. (Int. 2, 14, 37, 3, 15)

The operator removed one of the makeup pumps MU-PK at 4.5 minutes after closing the two associated HPI valves (MU-V16C and V16D). (Int. 2, 14, 37, 3, 15)

Section A, step 3.5 of the procedure contains the following statement:

"CAUTION: Continued operation of HPI depends upon the capability to maintain pressurizer level and RCS pressure above the 1640 PSIG Safety Injection Actuation setpoint.

"If pressurizer level can be maintained above the low-level-alarm point and the RCS pressure above the Safety Injection Actuation point; then initiate a plant shutdown and cooldown per normal procedures.

NOTE: The plant did not reach a stable hot standby RCS condition.

"If pressurizer level can not be maintained above the low level alarm point and the RCS pressure above the Safety Injection point then the plant has suffered a major rupture of a significant size leak such that Engineered Safety Features Systems are Automatically Initiated."

The operating procedures require operability of the safety features actuation system (2105-1.3, Safety Features Actuation System, Revision 2, Revision 13, March 19, 1979) during plant operating conditions in accordance with Technical Specifications 3.5.2. This requirement is amplified by the procedure for unit heatup (2102-1.1, Unit Heatup, Revision 19, March 7, 1979) which addresses ES system operability. (Ref. 79, 28)

The procedure for unit cooldown (2101-3.2, Unit Cooldown, Revision 13, March 10, 1979, step 4.13) provides for blocking the SFAS channels during normal plant cooldown activities between 1920 psig and 1800 psig; this allows for normal plant cooldown and depressurization under controlled conditions without an unwarranted ES initiation. (Ref. 27)

No procedural steps exist which permit the reduction in the HPI flow with RCS pressure below 1640 psig until the RCS is placed on the Decay Heat Removal System (DH).

The shift personnel throttled the high-pressure-injection flow to the reactor coolant system and the HPI system was in the degraded (throttled)

condition until approximately 100 minutes, even though the reactor coolant system pressure never recovered above 1640 psig. Moreover, the RCS pressure did not recover above approximately 1200 psig after 15 minutes into the accident until the plant was repressurized at 0750 hrs.

The failure to maintain the operation of the high pressure injection system delivering the design injection flow to the reactor core of 250 gpm per HPI flow leg (4), during the accident, in accordance with the requirements of emergency procedure 2202-1.3, Rev. 8, Section B, step 2.2.3 and 3.5 resulted in apparent serious core damage and onsite and offsite exposures to radioactive materials. This failure to maintain the HPI system in accordance with procedure 2202-1.3 is under consideration as a potential item of noncompliance pursuant to Technical Specification 6.8.1.a.

2.6 Operator Actions Concerning the Operation of the Makeup System (Following Initial ES Actuation)

The operators continued to operate the HPI/makeup system in the throttled condition between 0405 hours and 0530 hours. At about 0525 hours, emergency borating operations were performed when the RCS boron sample indicated 700 ppm and the source range monitors had shown an oscillatory behavior with an increased count level. The chemistry analysis indicated a decrease in boron concentration in the RCS, the neutron flux levels increased, and the pressurizer and/or makeup tank levels increased. The interviews and records review indicated that the operators appropriately addressed the actions required by the abnormal procedure (2203-1.1, Revision 5, September 12, 1978, Loss of Boron-Moderator Dilution, as follows:

- o Attempt to stop the inflow of water into the RCS.

The operators attempted to verify all sources of water into the RCS that could be causing a high pressurizer level (full-scale after 6 minutes) and an increase in the neutron flux level. (Int. 5, 17, 38, 2, 14, 37, 3, 15, 10)

- o Determine the amount of shutdown margin available in accordance with 2103-1.9, Reactivity Balance Calculations.

The calculated shutdown margin for the boron sample taken at 0515 hours (700 ppm) was approximately 6% reactivity. (Int. 5, 17, 38, 2, 14, 37, 97)

- o Initiate additional sampling.

An additional sample for boron was obtained at 0600 hours (404 ppm) and the shutdown margin was calculated to be 2.4% reactivity at 0630 hours. (Int. 5, 6, 17, 38, 2, 14, 37)

Emergency borating operations were conducted at approximately 0525 hrs through the normal flowpath to the MU system from the boric acid mix tank (BAMT) and the boric acid pumps (CA-P-4A or 4B) by opening MU-V10. Additionally, at 0600 hrs after the neutron flux level had continued to increase, the operators borated again and personnel were dispatched to the auxiliary building to open the alternate borating path (MU-V127) and verify the valve lineup to the makeup system from the boric acid mix tank (BAMT). (Int. 5, 6, 17, 38, 2, 14, 37, 3, 15)

The suction for the MU system, with one or two pumps operating from shortly after the reactor trip occurred, had been the borated water storage tank (BWST). Therefore, the water being supplied to the reactor coolant system was approximately 2300 ppm boron. The operators did not understand specifically why the neutron flux level was increasing or why the indicated RCS boron concentration had decreased from 1028 ppm to 700 ppm and then to 404 ppm by about 0600 hrs. (Int. 5, 6, 17, 38, 2, 14, 37, 27, 3, 15)

- o The operator interviews indicated the HPI flow was increased approximately 0530 hrs because neutron flux level was oscillating and increasing, and the reactor coolant system flow through the A loops had decreased to approximately 35 million pounds per hour. Subsequent to initiating the increased high-pressure-

injection flow, the operators noted that the A loop RCS flow continued to decrease rapidly from about 35 million pounds per hour to 25 million pounds per hour. (Int. 5, 6, 17, 38, 2, 14, 37, 3, 15, 27)

The operators continued to operate the MU system in the high-pressure-injection mode from 0530 hours through 2000 hours with at least one makeup pump operating. The apparent exceptions to this include:

- o High pressure safety injection was initiated at 0720 hrs and MU-P1A started and operated until 0737 hrs, when the operator, having reset the ES signal at 0727 hrs, stopped the 1C MUP. The MU-P1A continued to operate until 0756 hrs, when the building isolation occurred and the MU-PIC started. The MU-P1A and MU-PIC pumps operated in the injection mode until about 0817 hrs, when the operator stopped both 1A and 1C pumps. The operator immediately attempted unsuccessfully to start the MU-P1A to reestablish RCP seal water injection. The MU-P1A control switch was placed in the PULL-TO-LOCK position and apparently remained in that position until after 2000 hrs due to the failure of the pump to start manually and the belief the pump had a mechanical problem. The investigation of this area indicates that the pump breaker could not have closed, which led the operator to misunderstand indications. (Int. 2, 14, 37, 3, 15, 4, 9, 61)

- o Another ES initiation (Channel A) occurred at 0819 hrs on high building pressure. The makeup pumps (MU-P1A and P1B) failed to start and the 1C makeup pump was not operating or started. (Int. 5, 6, 17, 38, 2, 14, 37, 4, 9, 61)

- o The makeup pumps were off for approximately 4.5 minutes when, at 0822 hrs, the operator started the 1B pump, and this pump remained in operation until after 2000 hrs. (Int. 2, 55, 95; 4, 35, 36, 37 and 145)

- o The operator started the 1C makeup pump at 0827 hrs and the 1B and 1C makeup pumps were operated in the injection mode, delivering borated water heavily to the loop A injection lines (MU-V16A and MU-V16C). The 1C pump was operated continuously in support of the 1B pump until 1304 hrs, when the RCS pressure had been decreased to about 450 psig. This was during a period of intentional

pressure reduction of the RCS in an attempt to reach the discharge pressure of the core flood tanks. It was intended by the plant staff to continue down in pressure to place the decay heat removal system in service at approximately 350 psig. (Int. 26, 5, 6, 17, 38)

o After the restart of the 1C MUP at 1350 hours on high building pressure ESF actuation, the operators reviewed the plant status and stopped MU-PIC at 1351 hrs after resetting the ES signal, leaving the 1B pump in operation and the 1C pump in standby. (Int. 119, 26)

o At 1432 hrs, the operators restarted the MU-PIC pump and injected additional HPI water into the B loop through valves (MU-V16C and MU-V16D) as pressurizer level decreased from offscale high (more than 400 inches) to 375 inches. At 1436 hrs, the operator stopped the 1C makeup pump after pressurizer level increased to more than 400 inches. (Int. 119, 26)

o The operators operated the 1C makeup pump for 9 minutes at 1519 hrs, for 3 minutes at 1533 hrs, and for 83 minutes at 1723 hrs in order to prevent pressurizer level from decreasing below the 80-inch heater cutoff point. (Int. 119)

o At 1933 hrs, when the RCP was bumped (10 seconds), the RCS pressure dropped to 1450 psig and the actuation of ES attempted to start the 1C MUP. The 1C makeup pump tripped; six seconds later the operator restarted the 1C pump and the pump operated for six minutes to recover RCS pressure to normal (2250 psig). The 1C pump was operated again as the pressure decreased to 1300 psig. The operation of the additional pump (MU-PIC) in support of the 1B pump was to prevent the RCS pressure from decreasing drastically. The RCS pressure finally stabilized at approximately 1000 psig. (Int. 119, 232, 233)

Evaluation

When the ES initiation (train A) occurred at 0819 hrs on high building pressure; the makeup pumps 1A and 1B failed to start and the 1C makeup pump was not operating or manually started. The failure of the 1B makeup pump to

start indicated that the pump control switch was in the PULL-TO-LOCK position, the pump was not selected for standby, or the motor breakers failed to close.

The automatic initiation of the ES Train A occurred normally. The failure of the ESF train B to initiate was due to the manual defeat of two of the three channels in train B of the Reactor Building Cooling and Isolation Actuation System (ESFAS) at 0800 hrs by the operators. Subsequent to the ESF initiation on Train A at 19 hrs, two of the three channels in Train A of the Reactor Building Cooling and Isolation Actuation System (ESFAS) were also defeated at 0820 hrs by the operators.

NOTE: The first of the two channels in the Train A was defeated at 0800 hrs concurrently with the defeat of the channels in Train B by the operators.

The Emergency Procedure (2202-1.3, Revision 8, May 12, 1978; Loss of Reactor Coolant/Reactor Coolant System Pressure, Step 3.4 of Section B) requires the "DEFEAT" of any two channels of Reactor Building Isolation and Cooling, then bypass of all three Safety Injection Channels.

NOTE: Both Train A and B remained defeated until 0724 hrs when the channels were automatically reset.

Technical Specification 3.3.2, Engineered Safety Feature Actuation System Instrumentation Procedures, section 3.3.2.1 requires that the ESFAS instrumentation channels shall be OPERABLE as shown in Table 3.3-3 of those specifications. This includes the safety injection and Reactor Building Cooling and Isolation from Reactor Building Pressure High with a minimum of two channels OPERABLE in MODES 1, 2, and 3.

The manual defeat of two of the three channels in Train B at 0800 hrs and two of the three channels in Train A at 0820 hrs in MODE 3 is under consideration as a possible item of noncompliance pursuant to Technical Specification 3.3.2.1.

As noted above in between 0654 hrs and 2000 hrs, the HPI system flow was routinely modified during the accident on March 28, 1979, by throttling the

HPI flow and starting and stopping makeup pumps. Except during the time periods listed below when the RCS pressure was above 1640 psig, the continuous operation of the full, available HPI system was required to be OPERABLE to provide core protection.

- (1) 0654 hrs through 0725 hrs
- (2) 0740 hrs through 0755 hrs
- (3) 0930 hrs through 1140 hrs
- (4) 1840 hrs through 1933 hrs
- (5) 1933 hrs through 1937 hrs
- (6) 1950 hrs through 2000 hrs

The failure to maintain the operation of the HPI system delivering the design injection flow to the reactor core of 250 gpm per HPI flow leg, including the period between 0819 hrs and 0822 hrs when no HPI pumps were operating in accordance with the requirements of the emergency procedure 2202-1.3, Loss of RC/RCS Pressure, step 2.2.3, is under consideration as an apparent item of noncompliance pursuant to Technical Specification 6.8.1.a.

At approximately 0800 hrs following the initiation of the ESFAS from reactor building high pressure at 0819 hrs the operator blocked the reactor building engineered feature actuation system channels in order to reestablish manual control of certain components and equipment. Technical Specifications 3.3.2 requires that the Engineered Safety Feature Actuation System (ESFAS) channels, including Reactor Building High Pressure, shall be operable in Mode 1, 2, and 3. Operating procedure 2105-1.3, Revision 0, April 9, 1977, Safety Features Actuation System, Step 4.3, requires that the SFAS channels will remain in the PROTECTION FUNCTION FULLY ENABLED condition except during maintenance or testing which will be required to demonstrate the systems ability to actuate when required.

The failure to maintain the SFAS channels operable as required is under consideration as a possible item of noncompliance pursuant to Technical Specification 3.3.2.

2.7 Operator Actions Concerning Condensate System

The sequence of events that resulted in a turbine trip are discussed in Section. 4.1.

The steaming to the main condenser via the turbine bypass valves (MS-V25A, V25B, V26A and V26B) to maintain steam header pressure at 1010 psig apparently caused the hotwell level control valve controller to fail (low level) and the hotwell was flooded from the condensate storage tank. The hotwell level increased from 21.82 inches to the completely flooded condition (greater than 50" indicated level) within 1 minute and the hotwell level was not recovered until 0653 hrs.

Interviews, record reviews, and analysis indicate that attempts were made to reestablish the condensate system at 0405 hrs as indicated by the start of the 1A condensate pump through 0653 hrs, when the hotwell level reject valve was finally recovered and the hot well pumped down to normal. This was done to provide normal feedwater system flow to the generators and to prevent losing condenser vacuum. The loss of condenser vacuum (approximately 18 inches) terminates the release of decay heat steam to the condenser and automatically shifts the decay heat steam to the atmosphere. The operators were involved in the following activities:

- o Attempts were made to start a condensate booster pump.

The operators responded quickly to the turbine building basement area to attempt to recover the condensate feedwater system. (Int. 4, 9, 5, 6, 17, 38)

- o The operators discovered a leak in the 2A condensate booster pump suction line and isolated the pump locally to prevent spilling condensate water to the turbine building floor and sump. (Int. 109, 102, 10, 5, 6, 17, 38)

The OTSG B turbine bypass valve isolation valve (MS-V15B) was closed by the operator 8 minutes later (Section 2.9 of Details I). (Int. 5, 6, 17, 38)

o The condenser vacuum started to decrease at about 0720 hrs, and the operator secured the condenser vacuum pumps at 0930 hrs. The release of decay heat steam was automatically transferred to the atmospheric relief valve (MS-V3A). The vacuum decreased rapidly, and at approximately 1050 hrs, the auxiliary boilers were also lost. The supply of auxiliary steam to the turbine seals in order to maintain vacuum was a continuing problem until approximately 1700 hrs when the auxiliary boilers were recovered. (Int. 57, 111, 5, 6, 17, 38)

o The release of decay heat through the atmospheric relief valve (MS-3A) was stopped at approximately 1315 hrs after the State had requested that steam release be terminated. This limited the removal of decay heat from the reactor to the HPI flow and RCS blowdown into the reactor building. (Int. 26, 29, 113, 5, 6, 17, 38)

o The condenser vacuum was reestablished at approximately 1700 when auxiliary steam was available for the main turbine seals. The release of decay heat was transferred back to the turbine bypass valves (MS-V25A and 26A) to the main condenser at 1750 hrs. (Int. 119, 26)

o The condensate booster pump (2B) was started at 1917 hrs in order to feed OTSG A, which was at approximately 100 psig at that time. (Int. 119, 20)

Evaluation

The problems encountered with the condensate system and condenser vacuum significantly detracted the operator's attention from the accident.

The shift supervisor, Unit 2, spent approximately 45 minutes from 0415 to 0500 hrs in the area of the condensate polishers and hotwell due to the persistent high hotwell level condition and lack of reject capability to the storage tank. The reject capability was recovered at about 0655 hrs. The Unit 1 shift supervisor was in the Unit 2 control room during the absence of the Unit 2 shift supervisor.

2.8 Operator Actions Concerning Reactor Building Sump Pumps

The reactor building sump pump (WDL-P-2A) came on automatically at 0407 and commenced pumping the RB sump to the auxiliary building. The source of the water appeared to be the normal water accumulation in the sump supplemented by the discharge from the relief valve on the reactor coolant drain tank (WDL-T-3) lifting at 120 psig.

NOTE: The level in the miscellaneous waste holdup tank (WDL-T-2) did not change during the accident. (Int. 109, 40, 10)

The second RB sump pump (WDL-P-2B) started at 0410 hours and pumped in parallel with the 2A sump pump to the auxiliary building. An auxiliary operator noted an excessive in water level in the auxiliary building sump with the RB sump pumps operating, and the operator notified the control room. The RB sump pumps were stopped at that time by the auxiliary operator at the direction of the control room operator to cease water transfer into the auxiliary building. At that time (0438 hrs), the operators did not note any unusual conditions in the auxiliary building. (Int. 5, 6, 17, 38, 3, 15, 10)

Reactor building isolation occurred at 0756 hrs (3.58 psig). This isolated the RB sump pump discharge line to the auxiliary building by closing the RB isolation valves (WDL-FHS-3189 and WDL-FHS-1332). The operators verified reactor building isolation using the Emergency Procedure 2202-1.3, (Loss of Reactor Coolant/Reactor Coolant System Pressure, Revision 3, May 12, 1978, Appendix B, of Section B.) (Int. 2, 14, 40, 3, 11, 15)

The 480V motor control centers 2-32A and 2-42A feeder breakers tripped at approximately 1351 hrs, terminating the power to the RB sump pumps (WDL-P2A and P2B) and other equipment supplied by the motor control centers. The power was not subsequently returned to the RB pumps (WDL-P2A and P2B) because of the possibility of electrical grounds on the pump motors after the operation of the reactor building spray system.

Evaluation

The operator actions concerning the reactor building sump pumps was very general in nature. The stopping of the RB sump pumps was viewed as needed to cease water transfer to the auxiliary building (AB) to limit the liquid waste handling conditions.

2.9 Operator Actions Concerning the B OTSB Tube Leak

The sequence of events following the turbine/reactor trip led the operators to conclude that an OTSG leak from the shell side to the containment atmosphere. This conclusion developed based on a number of events, including:

- o The persistent increased reactor building (RB) pressure and temperature including high-temperature fire alarms, which were noted and acknowledged by operations personnel. (Int. 95, 315, 109)

- o There were no radiation alarms in the RB with the exception of the intermediate closed cooling alarms, which were considered normal and routine. This absence of alarms led the operators to conclude there was not an RCS leak. These detectors were located near the RB sump and it was known by the staff in the control room that the rupture disc had blown on the RCDDT at 0415 hrs (Int. 3, 15, 30, 95). The operators also noted that these alarms had occurred previously. The RB radiation monitor (HP-R-227) had been found flooded, under positive pressure, and manually isolated by the operators at 0550 hrs. Records review recalled that the monitor had reached 50,000 cpm at 0518 hrs. The monitor had been flooded previously and reported to the NRC (LER 78-073-036, Section 1.6 of this report).

- o The operators noted that the B OTSG pressure had dropped 300 psig below the A OTSG and continued to drop markedly after the 1B and 2B RCPs were stopped at 0514 hrs. (Int. 5, 6, 17, 38, 4, 9, 61)

- o After isolating the B OTSG, the RB pressure was noted to have leveled out, reaffirming the conclusion of the operating staff that the B OTSG shell side was leaking into the containment building. (Int. 5, 6, 17, 38, 4, 9, 61)

o Subsequently, at about 0620 hrs, the EMOV (RC-R2) was closed and the RB pressure decreased from approximately 2.5 psig to 1.0 psig. The reactor building temperature decreased from 160°F to 120°F. The operators then concluded that the B OTSG had not been leaking earlier as suspected.

o At about 0656 hrs, after pumping down the hotwell, two additional circulating water pumps were started. The B OTSG was momentarily steamed to the main steam header when the main steam stop valves (MS-V4B and MS-V7B) were cycled open and closed within 7 seconds. The B OTSG was reisolated when the condenser offgas monitor (VA-R-748) alarmed indicating a leak in the B OTSG. The operators were convinced that the B OTSG had a primary-to-secondary leak and completely isolated the steam generator, including closing the supply to the turbine bypass valves by closing the header isolation valve (MS-V15B) at 0704 hrs. (Int. 81, 57, 111, 184)

o Throughout the remainder of the day, the B OTSG remained isolated and the A OTSG was utilized as necessary for decay heat removal.

Evaluation

The operations staff initially concluded that the B OTSG had a shell side leak based on the increasing reactor building temperatures coincident with a normal RCS pressurizer level. This conclusion by the staff was also supported by the lack of any reactor building radiation alarms early in the accident.

The operating staff did not relate the continuing heat load in the reactor building to the loss of coolant accident even though the assumption of a B OTSG shell side leak to reactor building could not be supported. The RCS temperatures were stable and the B OTSG level was increasing with minimal feedwater required. Furthermore, the feedwater to the B OTSG was completely isolated at 0428 hrs by the operators because of difficulties in maintaining OTSG level. (Ref. Appendix IA)

The substantial evidence of a LOCA was indicated by the continuing RB high temperature with the full reactor building cooling system in operation

(10 million to 100 million BTU per hr heat removal capability) and the continuing low RCS system pressure.

The operators completed isolation of the B OTSG at about 0527 hrs based on 300 psig difference between the A OTSG and the B OTSG. The operators did not recognize that the pressure differential was caused by inadequate backflow through the B loop after the RCPs were stopped in the B loop at 0514 hrs.

The brief unisolating of the B OTSG at about 0656 hrs indicated to the operators that the B OTSG had suffered a tube leak; since the condenser off gas monitor (UA-R-748) alarmed.

2.10 Operator Actions Concerning Initial Natural Circulation - Decay Heat Removal Via OTSGs

At approximately 0541 hrs, the loop A reactor pumps (RC-P1A and P2A) were stopped because of the decrease to 950 psig in RCS pressure, reduction of the indicated RCS loop A flow from 35 to 25 million pounds per hours, and pump vibration alarms including indication at maximum displacement. The situation was discussed between the shift supervisor, superintendent-technical support, and the operating crew. The discussion included the review of the Heatup/Cooldown Curve (RCS Pressure-Temperature Limits) within the operating procedure for RCPs (2103-1.4, Reactor Coolant Pump Operation, Revision 6, August 16, 1978). This heatup/cooldown curve was also provided in the procedure for decay heat removal (2102-3.3, Decay Heat Removal Via OTSG, Revision 5, March 17, 1978). During this period of time, immediately prior to stopping the loop A RCPs, it was decided to go to natural circulation. Procedure 2102-3.3, contains specific references, limitations and precautions, prerequisites, and procedural steps, including:

- o Reactor Coolant System Cooldown Limitations. The reactor coolant temperature, pressure, and cooldown rates must be maintained within the limits specified in Figure 3.4.2 of Technical Specification 3.4.9.1.

The heatup/cool-down curve attached to the operating procedure include the maximum and the minimum RCS pressure-temperature limits for the RCS, RCPs, fuel clad compression, and control rod drives during the decay heat removal operation.

o The emergency feedwater pumps (EF-P1, EF-P-2A, and EF-P-2B) are running (as indicated by Panel 3 indicating lights), supplying emergency feedwater to the OTSGs from the condensate storage tanks through the emergency feedwater valves (EF-V11A and V11B). OTSG level is increasing to or at 21 feet (50% as indicated on operating range level instrumentation SG-1A/1B-LT2 or LT3).

The motor-driven emergency feedwater pump (EF-P2A) was operating, taking suction from the condensate booster pump suction header and the main hotwell via the operating condensate pump. (Int. 4, 9, 61)

o All four RC pumps are tripped, the reactor is tripped, and the turbine is tripped. Steam pressure is being maintained at turbine header setpoint (i.e., 855 psig during normal operation plus 125 psig) dumping steam to the main condenser through the turbine bypass valves (MS-V25A and V25B and MS-V26A and V26B) or dumping steam to the atmosphere through the atmospheric dump valves (MS-V3A and V3B) if a low condition of less than 24 inches Hg exists in the main condenser or less than three circulating water pumps are running. Steam safety valves and pressurizer electromatic and code safety valves are closed.

Shift personnel had established that the reactor and turbine were tripped and had secured the four RCPs. The turbine bypass valves were in manual control with five circulating water pumps operating. The operators had previously concluded (incorrectly) that the safety valves and the pressurizer electromatic relief valve (EMOV) were closed. The EMOV (RC-R2) was still open at this time.

The operators stated that they did not feel that natural circulation was established because of the differential temperature across the A OTSG (T_h -530°F, T_c -520°F at 0652 hrs and diverging), the low A OTSG secondary pressure (800 psig and decreasing), and the minimum feeding and steaming rates for

the A OTSG. At approximately 0610 hrs the A loop T_h had increased to 620°F, the recorder off scale high; T_ψ had decreased to about 480°F differential temperature of 140°F); and the operators expected a differential temperature of 25°F. Some heat was being removed through the steam condensing mode within the steam generators, but the continuous decrease in the A OTSG pressure indicate that the heat removal from the RCS was limited. (Int. 5, 17, 38, 4, 9, 61)

Evaluation

Technical Specification 6.8.1.a requires that procedures be implemented covering the applicable safety-related activities as recommended by Regulatory Guide 1.33, 1972. The failure to establish the plant conditions as required by the operating procedure (2102-3.3, Decay Heat Removal, Revision 5, March 17, 1978, step 2.1.3), of approximately 1100 psig or 558°F (saturation pressure for 558°F - 528°F plus 30°F is under consideration as a possible item of noncompliance.

Interviews revealed that the shift personnel associated with the accident had not received specific training in the natural circulation aspects on the facility on site or at the simulator. This lack of specific training supports the questions concerning the establishment of natural circulation when the last two RCPs were secured at 0541 hrs.

Furthermore, the lack of specific training contributed to the attempt by the plant operators to place the plant in the natural circulation mode of decay heat removal, when parameters were outside the procedural requirements.

2.11 Operator Actions Concerning the Auxiliary Building Sump Pump

The auxiliary building sump pumps (WDL-P-3A and 3B) are normally aligned to automatically pump the auxiliary building sump through a set of two parallel filters to the auxiliary building sump tank (WDL-T-5) as described in the plant operating procedure (2104-4.1, Miscellaneous Liquid Rad Waste Disposal, Revision 2, March 14, 1979) (Ref. 37) and the plant system description number A4 (Rad Waste - Miscellaneous Liquid, 10/75).

Personnel interviews revealed the following information concerning the auxiliary building sump pumps, including:

o At approximately 0420-0440 hrs, when operators were in the area of the rad waste panel in the auxiliary building, the auxiliary building sump tank (WDL-T-5) level was extremely high with the reactor building sump pumps (WDL-P2A and P2B) operating. The control room was notified and the RB sump pumps were secured to stop the discharge of liquid from the RB sump. At that time, the operators had noted no abnormal conditions in the area. The miscellaneous waste drain tank (WDL-T-2) level was noted to be normal. (Int. 10, 109)

o At approximately 0656 hrs, the operator evacuated the area because of high radiation levels in the auxiliary building areas. Prior to leaving the rad waste panel, the auxiliary building sump pumps (WDL-P-3A-3B) were secured by the operator because water was backing up through the auxiliary building floor drains (small amounts of water were reported in the low points near the drains). (Int. 10, 109, 3, 15)

o The initial water in the auxiliary building sump system was concluded not to be highly radioactive since the high radiation levels were observed to be from specific areas such as the makeup tank (MUT) room and the pipe alley. This is treated in greater detail in Details II of this report.

NOTE: It was concluded by the staff that the relief valves may have lifted because of a high differential pressure across the makeup filters (MU-F-2A and 2B) as a result of deposits of materials from the RCS of undetermined origin within the filters. Pulsating letdown flows noted by the operators would indicate the lifting of a relief valve, either to the reactor coolant bleed tank/(RCBT) or to the auxiliary building sump system. (Int. 26, 4, 9, 61)

o At approximately 1300 hrs, a tour of the auxiliary building was performed. It was determined that there was water near the floor drains. Because of high

radiation levels in the area (10 rem per hour) the stay was brief. The reactor building and auxiliary building sump pumps were verified in the OFF-position. (Int. 13, 140, 173)

o Following the trip of 480-volt motor control centers 2-32A and 2-32B (immediately after the initiation of the reactor building spray system on high pressure - 30 psig) at approximately 1350 hrs, the operating engineer was dispatched to the auxiliary building to return power to certain of the equipment on the buses. The reactor building sump pumps (WDL-P-2A and 2B) were not energized. The buses subsequently remained energized with the auxiliary building sump pumps (WDL-P-3A and 3B) in the OFF position. (Int. 102)

Evaluation

The operator actions concerning the auxiliary building sump pumps was routine. The stopping of the reactor building sump pumps to prevent continued transfer of water from the reactor building to the auxiliary building until it could be processed was not unusual.

The water in the auxiliary building sump and floor drains became a problem when the highly radioactive reactor coolant was discharged into the auxiliary building sump system sometime after the reactor core was seriously damaged. The discussion of leakage of contaminated RCS water to the other areas of the facility is treated in Details II of this report.

2.12 Operator Actions Concerning Logkeeping

Administrative Procedure 1012 (Shift Relief and Log Entries, revision 8, October 4, 1977, (Ref. 77) establishes the requirements for shift relief and recording station operating activities in logs or other controlled documents on a shift basis (section 1.1) and describes the various shift records and logs involved and instructions required to maintain these records to conform with the Technical Specifications and to ensure adherence to the requirement of the FSAR (Section 1.2).

The general requirements are described in section 3.1.1 and specify shift records as hourly log, control room log, checkoff lists, recorder charts, and computer printouts that describe or record operating information and events. These records comprise the information that is necessary for evaluating operations or for analysis of previous operations.

The hourly log requirements (action 3.2) specify that the log will reflect plant parameters on an hourly basis. It will normally be prepared by the plant computer but can be manually prepared by the control room operator in the event that the computer is not functioning. If manual preparation is necessary, it will be performed by the control room operators and auxiliary operators. The inspectors reviewed the hourly computer log for March 28, 1979.

The control room log requirements (section 3.3) specify that the log will include information concerning reactivity, alarms pertaining to reactor core conditions with detailed explanation, any abnormal condition of operation, releases of radioactive waste, gaseous or liquid. The administrative procedure (1012) requires that the control room log contain specific information (steps 3.3.1-3.3.18 inclusive) and be considered an official document required by the FSAR. The required information includes: (Ref. 38)

- o All alarms that involve reactor core conditions must be recorded by the operator along with an explanation or reason for the alarm, e.g., Tave, reactor coolant system pressure, flow, or power.
- o Plant shutdown - Record the major steps in shutdown and the associated
- o Each system startup, significant status changes, and shutdowns must be recorded. Also record major unit status changes such as opening of the primary system and flooding of the fuel transfer canal and the time of the event.

- o Equipment/malfunction - List the equipment and problem and any restriction placed on the plant.
- o Abnormal operation - Record any condition that causes principal primary or secondary parameters to vary from normal.
- o Reactor trip & turbine trip - Record the conditions prior to the trip, cause of trip (if determined), corrective action taken, and time of the events.
- o Start and stop of any radioactive gaseous or liquid releases must be recorded in the Control Room Log along with release permit number.
- o Any abnormal valve line ups and equipment out of service or returned to service must be recorded.
- o Changes of position of any "defeat", or "bypass" switches must be recorded.
- o Accomplishment of testing - Record title and number of the test performed and the start and completion times or time of suspension of the test. The performance of all periodic tests and inspections required by the Technical Specifications just be recorded.

The above sections are not meant to be inclusive but merely indicate the type of entries that should be made. When doubt exists, an entry should be made in the log.

The shift foreman log requirements (Administrative Procedure 1012, Section 3.5; Ref. 39) require that the shift foreman log contain:

- o A summary of the station operation and major events that occur on each shift. Significant abnormalities that occur will be explained in greater detail than would be expected in the control room log.

o When equipment covered by Technical Specifications is taken out of service, the reason, time, Technical Specification requirements, and sample results (if applicable) will be noted on the left-hand page of the shift foreman's log. Additionally, all requirements for running, sampling, testing will be noted, delineating times, when the above must be accomplished.

On March 28, 1979, the Control Room Log contained two entries between 0400 hrs and 1314 hrs, including:

- o 0400 hrs - Turbine trip, reactor trip, ES
- o 0527 hrs - Isolated OTSG B

The next entry was at 1315 hrs and was noted as a late entry.

The shift foreman log contained two entries and a plant status stamp between 0400 hrs and 1315 hrs, including:

- o 0400 hrs - Turbine trip, reactor trip, HP injection ES.
- o 0527 hrs - Isolated S/G "B".

Evaluation

Technical Specification 6.8.1 requires that written procedures be established, implemented, and maintained covering certain safety-related activities.

Administrative procedure 1012, Shift Relief and Log Entries, specifies the requirements for documentation, evaluation, and analysis of significant plant operations, activities, and abnormalities.

The operators failed to provide adequate shift log entries of the significant events between 0400 hrs and 1315 hrs including:

- o HPI flow rates and paths
- o Manual initiation of HPI at 0530 hrs
- o Shutting and opening the core flood tank isolation valves.
- o Emergency feedwater valves EF-V12A and 12B discovered closed at 0408 hrs.
- o BWST level decrease between 0400 hrs and 1315 hrs.
- o Method of decay heat removal (atmospheric, condenser, HPI)
- o Emergency borating activities
- o Reactor building isolation at 0756 hrs
- o Disabling the auto start capability of the emergency diesel generators.
- o Isolation of the EMOV (RC-R2) at 0618 hrs by shutting the block valve (RC-V2).
- o The extent to which HPI was throttled during various periods throughout the event.

The above are examples of items having significant safety implications which were not available to the licensee or the investigators after the accident.

The failure to provide the required logs of events and activities as required by Administrative Procedure 1012, Shift Relief and Log Entries, is under consideration as a possible noncompliance.

NOTE: The examples chosen are those that are not recorded on the alarm printer or shown on other instrumentation. The operators are authorized by procedure to clear the computer memory in order to have the alarm printer catch up with current events. Had that been done, much of the

information utilized this and other investigations; including that of the licensee, would have been lost.

2.13 Operator Actions Concerning EMOV

The operators stated that they realized that the RCDT rupture disc had blown, but they were not alarmed by this because of the anticipated lifting of the EMOV following the apparent severe transient (routine following a Turbine-Reactor Trip). (Int: 17, 38, 15, 95)

The subsequent high temperatures on the EMOV and code relief valves were similarly not considered unusual. The operators stated that a key issue was that so many things were happening within the first hours of the transient that time passed very quickly. The awareness of elapsed time after the unit trip became secondary to the operations group. (Int: 17, 38, 95)

An operator stated that the EMOV discharge temperature indicating 283°F, 60°F above the code relief valves within minutes after the trip and the time was actually 0521 hrs did not appear abnormal following the transient. It would appear the operator did not realize that it was 81 minutes into the accident due to the continuing sequence of events. (Int. 95, 196)

Also, the operators indicated that the EMOV and code relief valve discharge temperatures were not significantly above the normal operating temperatures of approximately 200°F, which is just above the alarm setpoint. (Int: 17, 38, 15, 95)

The EMOV block valve was closed at about 2 hours and 19 minutes after the accident (0619 hrs). The closing of the block valve at that time occurred with a smaller differential temperature (26°F) between the relief valve exhaust lines than had been previously observed (0521 hrs). The interviews also revealed that the check of the relief valve temperatures at 0618 hrs were performed by another shift supervisor. (Int: 105)

Evaluation

The operation of the plant with the EMOV and code relief valve exhaust temperatures (about 200°F) significantly above the ambient conditions (less than 130°F by procedures) complicated the diagnosis of the EMOV-LOCA event and subsequent isolation of the EMOV by closing the block valve (RC-V2). The operation of the plant with the high (200°F vs. 130°F) exhaust temperatures on the relief valves is addressed in another section of this report (Section 1.2 of Details I).

Failure to shut the EMOV block valve until about 0619 hrs, (2.3 hours after the EMOV initially opened) is a second example of failure to implement Emergency Procedure 2202-1.5, Pressurizer System Failure, as discussed in Section 1.2 of Details I, and is under consideration as a possible item of noncompliance pursuant to Technical Specification 6.8.1.a.

2.14 General Location of Operation Personnel

As noted in Section 1.4.2, the shift complement on the morning of March 28, 1979, met the Technical Specification requirements. The location of the operations personnel during the event shown in Table I-2-1 was determined based on selected activities performed and interviews with the personnel involved. The tabulation does not include all operations personnel on site during the accident. The listing provides a general guide to the location of certain individuals.

The licensed operators performed their functions from the control room area, with a few exceptions, including:

Shift Supervisor E worked with the condensate polisher bypass valve (CO-V12) between 0420 hrs and 0500 hrs.

Shift Foreman A unlocked and closed the electrical breakers for the decay heat removal suction valves (DH-V102A and DH-V102B) at MCC 2-11EA and 2-21EA at about 1000 hrs.

Shift Supervisor C made a limited tour of the auxiliary building rad-waste panel and accessible floor drain areas at about 1300 hrs.

The auxiliary operator actions during the first 16 hours of the accident were directed by the control room staff. The auxiliary operators who were in their assigned areas performed checking activities in their respective areas following the announcement of the turbine and reactor trip. The auxiliary operators outside their assigned areas, in general, went to the Unit 2 control room for standby. The more experienced auxiliary operators made checks on equipment as they proceeded to the control room.

TABLE I.2-1

LOCATION OF KEY PERSONNEL INTERVIEWED

Individual	PRI	Control Room		Auxiliary Bldg.		Turbine Bldg.
		SEC	Back Panels	Waste	Elect	Makeup
Shift Supervisor E		0400-0420				0420-0500
		0500-1600				
Shift Supervisor A		0408-0630				
Shift Supervisor D		0550-1600				0400-0402
Shift Supervisor B		1200-2000(+)				
Shift Supervisor C				1300-1330 (TOUR)		
Shift Foreman C		0402-1500				
Shift Foreman A		0545-1000				1000-1030
Shift Foreman B		0700-2000(+)				
Control Room Operator D	0400-0600		0600-1500			
Control Room Operator C	0600-1500		0400-0600			
Control Room Operator I			0615-1930			
Control Room Operator A	1800-1900		0630-1800			

TABLE I.2-1 (Continued)
LOCATION OF KEY PERSONNEL INTERVIEWED

Individual	Control Room			Auxiliary Bldg.		Turbine Bldg.
	PRI	SEC	Back Panels	Waste	Elect	Makeup
Control Room Operator F		0600-1600				
Control Room Operator H	1600-1800					
Control Room Operator G	1800-2000(+)					
Control Room Operator K	1300-1700		0700-1300			
Auxiliary Operator G		0410-0420				0400-0410
						0420-0500
					0500-0600	0600-0700
		0700-1500				
Auxiliary Operator B		0410-0420				0400-0410
						0420-0430
				0430-0500		0500-0530
						0620-0630
		0700-1500		0630-0650		
Auxiliary Operator K						0400-1030

I-2-45

TABLE I.2.1 (Continued)
LOCATION OF KEY PERSONNEL INTERVIEWED

Individual	Control Room			Auxiliary Building			Turbine Building
	PRI	SEC	Back Panels	Waste	Elect	Makeup	
Auxiliary Operator E						0500-0600 0600-0620 0620-0640	0400-0500
Auxiliary Operator C		1400-2000+					
Auxiliary Operator H		0500-0600 0700-1300					0600-0700
Auxiliary Operator D							0400-0800
Engineer K		0550-0600				0600-0700
		0700-0900				
		1030-1630		1630-1645		

I-2-46

2.15 Effect of Training on Operator Actions

Members of the plant training staff were interviewed regarding the effect of the operator's training on their actions during the incident. (Int. 80)

Among the areas discussed were:

- . pressurizer level versus pressure control
- . recognition of an open EMOV
- . bypassing of emergency safety features actuation system
- . shutdown of reactor coolant pumps.

2.15.1 Pressurizer Level Versus Pressure Control

The plant procedures (2103-1.3) and the operator training program specify that the pressurizer shall not be allowed to go solid at any time except for hydrostatic testing of the RCS (Ref. 53). The training emphasizes this prohibition, stressing the possibility of exceeding the high-pressure safety limit of 2750 psig because the pump discharge head is 2900 psig. For high pressurizer level, the procedures (2103-1.3) require securing makeup and increasing letdown (Ref. 53). For low pressurizer pressure, the procedures require the opposite: isolate letdown, increase makeup, and, in addition, turn on the heaters.

The training staff was asked what the operators would be expected to do, based on their training and experience, if a high pressurizer level indication called for one set of actions and a low RCS pressure called for another. Members of the training staff stated that the operators would definitely have reacted to the high level to avoid going solid. This is based on both the TMI training and the B&W operating procedures at the simulator. The necessity of maintaining pressure is stressed in connection with the avoidance of departure from nucleate boiling (Int. 80).

The staff was asked if the operators were taught the significance of saturation pressure. The training staff stated that the operators receive

this instruction as part of the basic thermodynamics training. The staff was asked if the saturated condition would cause the operators to suspect steam voids in the primary system. They indicated that under the conditions existing at the beginning of the event, the training staff would not expect the operators to check for the saturation condition immediately. The operators would not expect voiding with the pressurizer full. With the injection of cold auxiliary feedwater supply and high pressure injection the operators would expect a pressure reduction. (Int. 80)

The training staff was asked if the possibility of a level rise in the pressurizer caused by steam flashing in another part of the primary system had ever been recognized and brought to the operators' attention. The answer was "never." The only training in this area is the discussion of the possibility of flashing in the hot legs if the pressurizer level is not maintained. (Int. 80)

2.15.2 Recognition of an Open EMOV

The training staff was asked if the operators were trained to verify the closure of the electromatic relief valve following events that can be expected to result in its opening. They stated that they were trained to check if it was open, but considering the other events that were occurring during the accident, they would not have expected the operators to check this right away. The means available to check this were the console demand signal, which indicated closed, and the discharge line temperatures. The training staff stated that high discharge line temperatures were not very meaningful because the EMOV had been leaking prior to the incident, which resulted in temperatures that were not much lower than those existing with the valve open. Moreover, these temperatures and their status are printed out by the alarm printer, and these alarms would not get printed out for 20 to 30 minutes. This time delay, under the conditions of the accident is caused by the large number of alarms to be printed, and the limits on the typing speed of the Alarm Typer.

The training staff was asked if the operator training included actions to be taken if there was a pressure rise in the reactor coolant drain tank (RCDT).

They stated that the training on the "Response to High RCDT Alarm" procedure (No. 2204-301B, Ref. 26) covered this. However, the alarm and indicators for this system are located behind the back panels. Also, determining the source of the leakage requires a process of elimination because the RCDT receives other leakage such as the RCP seals and valve packing leakoffs.

2.15.3 Bypassing of the Engineered Safety Features Actuation System

It was noted that the engineered safety features actuation system was bypassed by the operators promptly after actuation, even though the coolant injection might not be throttled back until later. The training staff was asked if the operators were trained to do this. The staff stated that the operators were trained to reset as soon as possible. This is done to prevent injection of sodium hydroxide into the reactor. In addition, the operators were trained to be prepared to maintain a 220-inch level in the pressurizer by throttling the HPI valves. The operators also had to be prepared to throttle the flow to the makeup pumps to prevent exceeding the 550-gpm flow limitation, as the flow would increase if the RCS pressure decreased.

2.15.4 Shutdown of the Reactor Coolant Pumps

The training staff was questioned as to the training given the operators regarding shutdown of the RCPs. They stated that the procedures (2203-1.4, Revision 3) and training, required pump shutdown for high vibration, low amperage or low reactor coolant flow, all conditions that existed during the event. (Ref. 78) In the training staff's view, the conditions that existed during the March 28 event did require shutdown of the pump. They stated that the operator is trained that failure to trip the pump under these conditions could lead to pump seal failure or loss of the impeller.

Evaluation

The operator training program had a substantial impact on decisions made during the early phase of the accident. The prohibitions against allowing the

pressurizer to go solid caused the operators to neglect the low RCS pressure in their attempt to control pressurizer level. In addition the operators were conditioned to promptly bypass ES without first determining the condition of the RCS.

However, it must be recognized that operator experience also played an important role in these decisions. The normal course of most ES initiations, those which did not involve a loss of coolant, required bypassing of ES and securing of HPI to prevent an extended plant outage and/or possible damage to the pressurizer safety valves. Past failure to bypass ES promptly had, in fact, resulted in the injection of sodium hydroxide into the RCS at TMI2.

The operator training did not address the phenomenon which led to the pressurizer level transient. The investigation has not established whether operators at other B&W plants had received such training.

2.16 Nonlicensed Operator Actions During Accident

2.16.1 General

The nonlicensed operator (auxiliary operator) actions during the first 16 hours of the March 28, 1979, event were directed from the control room. The auxiliary operators who were in their assigned area performed checking activities in their respective areas following the announcement of the turbine and reactor trip. The auxiliary operators outside their assigned areas, in general, went to the Unit 2 control room for standby. The more experienced auxiliary operators made checks on equipment as they proceeded to the control room.

2.16.2 Activities Prior to Trip

The major activity just prior to the turbine trip was being performed in the turbine building at the condensate polisher unit. There was a problem in transferring resin out of condensate polisher vessel No. 7. The transfer line was plugged with resin for about 11 hours (Int: 123). There were at least three individuals working on the problem just before 0400 hrs. This included

two auxiliary operators and the operation shift foreman. The unclogging process was making use of station air and demineralized water. Other activities being performed just prior to the event included adding hydrogen to the main generator hydrogen system, performing readings on industrial waste treatment systems, and logging and maintaining the radwaste panel in the auxiliary building. The consensus of opinions from the interviews involving the auxiliary operators indicate that the plant conditions were, in general, fair to good (Int. 10, 36, 38, 104, 109, 115, 123, 125 and 128). There was a large inventory of water in the auxiliary building. Water was estimated to be within 8 inches of overflow in the auxiliary building sump. This situation has existed in the past because of the way the units are restricted in the release of waste water. The units share release of water in that only one plant can release at a time. The schedule for release of Unit 2 water and the status of the Unit 1 water inventory was not known by the auxiliary operator on March 28, 1979 (Int. 10).

2.16.3 Activities During Accident

The following paragraphs are summaries of auxiliary operator interview transcripts. They give a general picture of typical activities on March 28, 1979:

Auxiliary Operator D (Int. 61, 128) was on the 2300 hrs to 0700 hrs shift on March 27-28, 1979 (Int. 61). The assigned area was the secondary side of the plant. Operator D was on the way to and very near the control room at 0400 hrs and apparently just outside the control room at the time of the trip announcement. This operator was assigned to go to the turbine building to turn the main feedwater pump B shaft. This effort continued until about 0800 hrs. Independent checks were made on the main condenser vacuum pump system and the operator aided in checking the hydrogen oil seal system.

Auxiliary Operator G was involved in the condensate polisher problem and actively engaged in unblocking the resin transfer line (Int. 36). He checked the condensate polisher panel and the status of the outlet valves at the condensate polisher tanks immediately after the trip was announced. The condensate polisher valves were reported to be closed. After completing a

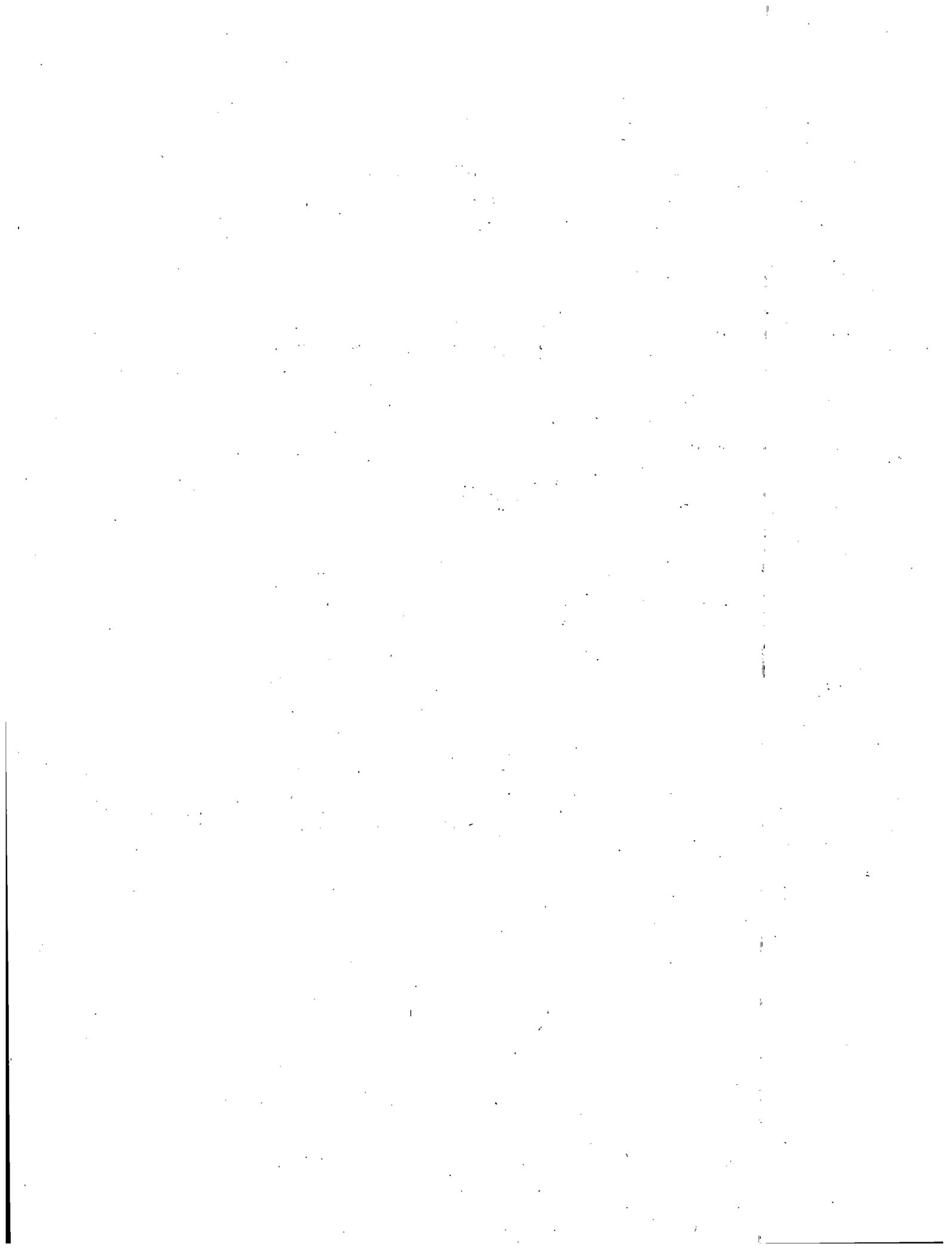
lineup of the condensate polisher for restart, he proceeded to the control room at about 0415 hrs to 0430 hrs and was sent back to recheck the condensate polisher valve lineup. He noticed leakage from condensate booster pump 2A suction line and proceeded to isolate the leak, with help from others (Int. 10), by closing the booster pump suction valve. He associated this leak with a major movement of this pipe that he had observed just after 0400 hrs prior to going to the control room the first time. He then, with help, manually opened the condensate bypass valve (C0-V12) (Int. 10). He also checked the pressurizer level indicator inside the auxiliary building. At about 0530 hrs he was told to open MU-V127 to allow for emergency boration.

Auxiliary Operator B was assigned to the radiation waste panel in the auxiliary building on the 2300 hrs to 0700 hrs shift (Int. 10) The status of his area has been described above. The miscellaneous waste storage tank level was stated to be at 7.4 feet just before 0400 hrs and lined to the RB sump pumps. At 0400 hrs he was just outside the Unit 2 control room, by the I&C area. He returned to the control room at about 0410 hrs and was sent by the control room operator to check breakers for condensate booster pump 2B. He found the breakers to be in good status. He was involved in throttling river water to the intermediate closed coolers. He went to the auxiliary building, his assigned area, to make a check of his area. He found indication that both reactor building sump pumps were on, the local reading of the reactor building sump level was pegged high (over 6 feet), and the background activity in his assigned area of the auxiliary building had increased. He estimates this time to be about 0430 hrs to 0500 hrs. He communicated his finding back to a control room operator. He turned off the reactor building sump pumps at about 0438 hrs at the request of the CRO. The pumps were said to be lined up to the miscellaneous waste holdup tank, but he did not notice a level change from the 7.4 feet seen earlier. He did not see any overflow of the auxiliary sump at that time. He was told by a CRO to check for makeup valve alignment for demineralized water to the makeup system. He noticed while assisting closure of C0-V12 that efforts were underway to lower the hotwell level. He also noticed that the air supply line to the pneumatic actuator to the normal condensate reject line was failed and blowing air. He noted that the shift supervisor was also in this area at this time

(TMI 220, 221). At about 0600 hrs, he was back in the auxiliary building and found water backing up out of the floor drain. He was at the radwaste panel at the time the site emergency was announced. He then went back to Unit 2 control room to report the status of his area to the shift supervisor and CRO.

Auxiliary Operator H was on the 2300 hrs to 0700 hrs shift assigned to the industrial waste area (Int. 38 and 125). He reported to the Unit 2 control room at approximately 0445 hrs after being paged by another auxiliary operator at about 0420 hrs. He was directed by the shift supervisor to check the main turbine and to ensure that it went on turning gear. He was also told to isolate the second stage reheat of the moisture separator and reheater units at about 0510. He was told, by the operating engineer, to isolate the outlet valves on the condensate polisher system, which he did. He had found all outlet valves closed and proceeded to drop air pressure to the valves, which would ensure closure.

Auxiliary Operator E reported to the control room after hearing the noise from the main steam safety relief valves (Int. 109). He was assigned specific tasks by the control room operations personnel. He stated that he suggested the stopping of the reactor building sump pumps based on the auxiliary building high water inventory. He recalled that one condensate pump remained on line. He assisted in opening the main condensate reject valve at about 0500 hrs to 0530 hrs. He closed the breakers for the core flood tank isolation valves at about 0600 hrs. He searched for the cause of the fire alarm. He checked the valve lineup for boric acid, level of the miscellaneous waste holdup tank, and valve lineup for possible demineralized water entering the makeup system. He recalled seeing water coming out of the auxiliary building drains (before 0730 hrs).



3. MANAGEMENT ACTIONS DURING ACCIDENT

3.1 General

This section of the report deals with the actions and management decisions undertaken by those members of licensee management who were called to the site to provide emergency direction to cope with the operational aspects of the accident. Moreover, this section will address the additional support that was provided through the licensee organization and by other parties to support the onsite operational activities. Those actions taken to cope with the radiological aspects of the accident are addressed in Details II of this report.

The use of the word management in this section refers to those individuals who would be summoned to the site for emergency and technical direction as well as those contacted to organize appropriate support activities. The investigation team recognizes that members of the routine shift organization do represent levels of licensee management.

3.2 Onsite Operational Activities of Senior Management

Management notification was initiated by the Unit 2 Shift Supervisor shortly after the turbine and reactor trips. At the request of the Unit 2 Shift Supervisor, the Unit 1 Shift Foreman called the Station Manager (approximately 0401 hrs) and the Unit 1 Operations Supervisor (approximately 0435 hrs). Nuclear engineers, coming to Unit 2 from the Unit 1 refueling startup crew, called the Unit 2 Superintendent and the Unit 2 Superintendent - Technical Support (approximately 0410 hrs). These senior managers were informed of the Unit 2 turbine and reactor trips, as a matter of normal site policy and not as a result of any suspicion that this trip was unusual. The Unit 2 Operations Supervisor was not called, since he was in Lynchburgh, Virginia attending B&W simulator training (Int. 1, 17, 26, 27, 83, 148).

The Unit 2 Superintendent - Technical Support, the assigned on-call "duty section head" at that time, was the first to arrive on site (approximately 0450

hrs). He was informed of the problems being experienced with pressurizer level being high, reactor coolant system (RCS) pressure being low, reactor coolant drain tank (RCDT) rupture disc being blown, and that an automatic ES actuation/ high pressure injection (HPI) had occurred. He was not informed of the earlier failure of the emergency feedwater system to perform as designed, due to the improper position of the EF-V12A and 12B valves; this fact was not learned by onsite management until engineers began reviewing the sequence of events, a day or so later.

NOTE: The Unit 2 Superintendent - Technical Support is the chairman of the Unit 2 Plant Operating Review Committee and holds a senior reactor operator license on Unit 1. He had just started an informal training program for Unit 2 and he doubted his capability to recognize the meaning of alarms or to interpret everything he was being told. Technical Specifications do not require this individual to be licensed.

Lacking familiarity with the plant, but recognizing an unusual situation, the Superintendent - Technical Support directed additional technical and operations personnel be called in. Those called included the Unit 2 Operations Engineer (approximately 0501 hrs), the Station Chemistry/Health Physics Supervisor (approximately 0510 hrs), the Maintenance Superintendent (approximately 0515 hrs) and at least ten others. (Int. 27, 129; Ref. 21)

As RCS pressure continued to fall, the Unit 2 Shift Supervisor secured the reactor coolant pumps (RCPs) in the B loop. The Unit 2 Superintendent-Technical Support reviewed net positive suction head requirements for the RCPs and concurred with the decision made and action taken by the Shift Supervisor. During this period, the Superintendent - Technical Support was attempting to become better informed relative to the status of the plant, making suggestions and offering technical advice, and requesting additional plant personnel report to the site. The direction of plant operations continued to rest with the Unit 2 Shift Supervisor. (Int. 27, 135)

Having received no additional information since the post-trip notification, the Station Manager called Unit 2 (approximately 0515 hrs, lasting to approximately 0535 hrs) to ascertain the status of the plant. The Station Manager was informed by his management representative of the current plant conditions and the fact that an automatic actuation of HPI had occurred. The Station Manager was disturbed by the coincidence of low RCS pressure and high pressurizer level. The automatic actuation of HPI was not unexpected, based on experience with previous turbine/reactor trips. The Station Manager decided to initiate a conference call with selected technical people; but first (approximately 0545 hours) called the Unit 1 Superintendent, informing him of the peculiar trip conditions and requesting that Unit 1 cooperate in giving Unit 2 heating steam, and called the Maintenance Superintendent, requesting he report to the plant.

NOTE: During the period of the Station Manager's call to the Superintendent - Technical Support, the Shift Supervisor had isolated the B once thru steam generator (OTSG). The Shift Supervisor had taken this action on finding reactor building pressure increasing and B OTSG pressure below that in A OTSG; which indicated to him a secondary-to-containment leak. (Int. 1, 27, 71, 77, 120)

Following his conversation with the Station Manager, the Superintendent - Technical Support found the RCS pressure dropping again. With indicated flow in the A Loop at about 30% and dropping, the Shift Supervisor tripped the A loop RCPs. Everyone just assumed natural circulation would occur; despite the fact that RCS parameters were outside necessary conditions for subcooled natural circulation. (Int. 27, 38, 135)

With all RCPs tripped, the core was being cooled by HPI water steaming through the still partially or fully open pressurizer electromatic relief valve (EMOV). The Unit 2 Superintendent (senior reactor operator licensed on Unit 2) arrived onsite (approximately 0545 hrs) and proceeded to the Unit 2 control room, where he was briefed by the Shift Supervisor and the Superintendent - Technical Support.

NOTE: The Unit 2 Superintendent lives approximately 50 miles from the site and his initial notification provided no evidence of an abnormal trip, which might have prompted him to arrive earlier.

At the request of the Unit 2 Shift Supervisor, the Unit 1 Operations Supervisor (senior reactor operator cross-licensed on Units 1 and 2), who was in Unit 1 since about 0530 hrs, to assist in its startup, reported to the Unit 2 control room (approximately 0600 hrs) to assist as needed. The abnormal pressurizer level versus RCS pressure, the tripped RCPs, and the results of an RCS post-trip boron sample showing a drop of 300 ppm from pre-trip conditions, were discussed. Shortly thereafter, on seeing indication on the source and intermediate range nuclear instrumentation of what appeared to them to be a possible reactor restart, the Shift Supervisor was directed by the Operations Supervisor and the Superintendent - Technical Support to high pressure inject and emergency borate. (Int. 5, 9, 26, 27, 83)

The Station Chemistry/Health Physics Supervisor had arrived onsite (approximately 0545 hrs), following the earlier call for technical and operations personnel, directed by the Superintendent - Technical Support. His initial assignments, from the Superintendent - Technical Support, included making preparations for a reactor building (RB) entry and confirmation of RCS boron sample results. (Int. 20, 102, 129, 41; Ref. 21)

With the Unit 2 Superintendent and the Unit 1 Operations Supervisor following operations, the Superintendent - Technical Support retired to the Unit 2 Shift Supervisor's office to participate in the conference call established by the Station Manager. The conference call (approximately 0600 hrs, lasting to approximately 0635 hrs) additionally included the B&W Site Operation Manager and the Vice President - Generation, Metropolitan Edison (the first corporate individual to be notified). The status of the plant was discussed and it was decided that forced circulation must be reestablished. The fact that all available indicators of pressurizer level were checked and found in close agreement was discussed and it was decided they were to be believed. The condition of the EMOV was questioned and it was reported to be shut, reportedly based on a demand position light indication. The B&W Site Manager and the Station Manager were encouraged to report to the site.

NOTE: During the period of this conference call, the EMOV block valve was closed; finally isolating the loss-of-coolant accident path that had existed since 0400 hrs. Whether this action was keyed by this discussion has not been established. (Int. 1, 27, 53, 71)

With the EMOV block valve shut, RCS pressure began to increase and Reactor Building pressure began to drop. The results of a second RCS boron sample indicated a further decrease in boron concentration; which the Station Chemistry/Health Physics Supervisor had confirmed by splitting the sample and observing the analysis performed by two technicians. Following the report of these results, the Superintendent - Technical Support then directed the Unit 2 Operations Engineer (senior reactor operator licensed on Unit 2), who had arrived onsite (approximately 0545 hrs) following his earlier call out, to investigate possible sources of demineralized water that might be diluting the RCS boron concentration. The results of this investigation reportedly were never made; presumably the investigation was aborted by subsequent events related to the declaration of the Site Emergency. (Int. 20, 27, 102, 129)

With the RCS sample lines on recirculation to ensure a representative sample, radiation levels in the vicinity of the sample lines began to increase. Responding to an alarm, the Station Chemistry/Health Physics Supervisor discovered radiation levels of 600 millirem per hour (mr/hr) and reported these results to the Superintendent - Technical Support. He, in turn, then reported the information to the Unit 2 Superintendent, along with his belief that they were experiencing fuel failures. The Unit 2 Superintendent directed the Maintenance Superintendent, who had arrived on site (approximately 0615 hrs) at the earlier direction of the Station Manager, to call the Station Manager and inform him of this event. The call was made as directed (approximately 0650 hrs). (Int. 20, 27, 71, 120, 135)

RCS pressure had now increased to the point that RCPs could be operated. Each manager recognized the need to re-establish core cooling. Initial attempts to start RCPs under the direction of the Unit 2 Superintendent, the Superintendent - Technical Support and the Unit 1 Operations Supervisor were unsuccessful. The

operators finally succeeded in starting RCP-2B (0654 hrs) and the source and intermediate range nuclear instrumentation showed significant drops in flux levels, convincing the Superintendent-Technical Support that the apparent reactor restart had been terminated. Within minutes radiation monitors throughout the plant entered alarm status and a Site Emergency was declared (approximately 0656 hrs) by the Unit 2 Shift Supervisor at the direction of the Superintendent - Technical Support. (Int. 26, 27, 83, 129, 138)

Immediately following the Site Emergency declaration, the Shift Supervisor returned his attention to the primary plant and reinitiated HPI at the direction of the Unit 1 Operations Supervisor. B OTSG was again isolated due to radiation alarms on the condenser vacuum pump exhaust; it had been unisolated following discovery that the leaking EMOV was apparently the source of the Reactor Building pressure increase. The Unit 1 Operations Supervisor stationed operators at each Unit 2 control room panel, coordinating their activities and backing them up with normal shift and relief shift foremen and supervisors. The Superintendent - Technical Support directed various engineers (then present due to his earlier call-out of technical and operations personnel) to make the required Emergency Plan notifications. With the Station Manager in transit to the site, the Unit 2 Superintendent became the Emergency Director (per TMI Emergency Plan, Procedures 1670.2, Revision 8). (Int. 17, 26, 27, 71, 83, 38; Ref.)

The Unit 1 Superintendent (senior reactor operator cross-licensed on Unit 1 and Unit 2) had arrived on site (approximately 0645 hrs), heard the announcement of the Site Emergency, proceeded to the Unit 1 control room (per TMI Emergency Plan, Procedure 1670.2 Rev. 8) and declared himself the Emergency Director of that unit after being briefed on the situation. The Station Manager (not currently licensed or required to be licensed on either unit) arrived onsite (approximately 0705 hrs) and proceeded to the Unit 2 control room, where he was briefed by the Shift Supervisor and his managers. The Station Manager then declared he was the Emergency Director and established an emergency command team. The Unit 1 Operations Supervisor was put in charge of operations to direct the Shift Supervisor; the Station Chemistry/Health Physics Supervisor was put in charge of onsite and offsite radiation and environmental concerns;

the Unit 1 Superintendent was put in overall charge of Unit 1 and the Emergency Control Station, then being established in Unit 1; the Unit 2 Superintendent was put in charge of reviewing and verifying personnel complied with procedures and plans; the Maintenance Superintendent was put in charge of emergency maintenance; the Superintendent - Technical Support was put in charge of notifications, communications and technical support; and the B&W Site Manager, who had just arrived onsite (approximately 0715 hrs) was requested to provide technical assistance and communications with B&W. The Station Manager further declared these personnel were to be the funnels through which information relative to areas under their charge would be directed to or from him. During this period, the Unit 1 Operations Supervisor and the Shift Supervisor secured the running RCP, since it showed no flow and a running current of about 100 amps, and established the dominant mode of core cooling to be utilized for the next 13 hours, with HPI feeding cool water to the RCS and the EMOV and/or its block valve open to remove hot steam. Attempts to notify the Vice President - Generation, Metropolitan Edison of plant status at this time were unsuccessful. (Int. 20, 26, 27, 53, 71, 77, 83, 120, 38)

By about 0724 hrs, conditions had deteriorated to the point where the Station Manager declared a General Emergency.

NOTE: Management actions relative to radiation protection, environmental protection and implementation of the Emergency Plan are discussed in Details II of this report and will not be repeated here.

NOTE: From an operational standpoint, TMI Emergency Plan, Procedure 1004, Revision 2 states ... "The Operational personnel are responsible for the safe operation and recovery of all systems during an emergency situation." The Emergency Plan also requires a significant effort in initial notification of and maintaining communications with, outside agencies and offsite personnel; both of which distracted technical people from the task of grasping the significance of events, understanding the situation, and containing the accident. (Int. 27, 71)

The Station Manager called and directed the Unit 1 Superintendent (approximately 0745 hrs) to join him in Unit 2. The Station Manager then requested his emergency command team caucus with him in the Shift Supervisor's office. The meeting was held away from control room activities to allow more considered thought relative to plant status, long term goals, potential actions and their consequences. It was decided to attempt another RCP start. RCP-1A was finally started, but loop flow and motor current failed to respond normally and everyone was forced to acknowledge the loops were vapor bound. (Int. 26, 27, 53, 71, 77)

Unable to resolve what appeared to them to be discrepancies in RCS parameters, the emergency command team opted to direct emergency core cooling systems be allowed to function as they would had the operators not been present. The Station Manager directed the Unit 1 Operations Supervisor to keep HPI on and not to secure it without his permission. Instrument technicians were directed to obtain local readout of core thermocouples (T/Cs) and RCS loop hot leg resistance temperature detectors (RTDs). The emergency command team would meet frequently during the following hours, usually once or twice an hour. Plant status would be reviewed, opinions and fears expressed, courses of action considered, decisions made and orders issued. Subsequent meetings would discuss results of actions taken and/or communication held with offsite individuals or groups. These meetings allowed a sharing of knowledge, but reportedly none of the participants doubted that the Station Manager was in charge and that, ultimately, it would be his decision they would carry out. It should be noted that the character of these meetings reportedly did not change, even after the arrival of NRC and GPU personnel. (Int. 26, 27, 53, 71, 129)

By 0900 hrs, the plant had experienced one manual ES actuation and two automatic ES actuations/reactor building (RB) isolations, since the declaration of the Site Emergency. Temperatures in the RCS hot legs and the core indicated superheated vapors were present. The core cooling that did exist was provided by HPI water that flashed to steam in the core and was being vented to the reactor building via the EMOV. The emergency command team thought the core

was covered, but were not sure. They acknowledged that temperatures were high, but didn't believe temperatures were as high as some readings had indicated (2620°F on one core thermocouple). They acknowledged the RCS loops were vapor bound, but couldn't figure a way to refill them. They knew HPI was being injected, but feared it might be bypassing the core. They recognized that without changing their tactics, clean Borated Water Storage Tank (BWST) water would ultimately be exhausted and radioactive dirty water from spillage on the reactor building floor would then have to be utilized, increasing auxiliary building radiation levels significantly. Finally, they were nowhere nearer the two possible stable RCS conditions of forced circulation, using either RCPs at high pressure or decay heat removal (DHR) pumps at low pressure. The Vice President - Generation called (approximately 0910 hrs) and learned the current status of the plant from the Superintendent - Technical Support and possibly others. Shortly thereafter, the Station Manager directed the plant be repressurized using HPI, maintaining pressure between 2000 and 2100 psig with the EMOV block valve. This pressure range was picked to prevent actuation of the code safety valves and the attendant danger of uncontrolled depressurization should they stick open. (Int. 26, 27, 53, 71, 70, 83, 113, 129; Ref: 21, 23, 22)

RCS pressure was increased in an attempt to collapse the voids in the RCS loops. The decision to repressurize lacked unanimity, since it was pointed out that if temperatures were anywhere near correct, the code safeties would not allow sufficient pressure to condense the steam of which the voids were assumed to be composed. Further, the history of poor reliability of the EMOV block valve raised concerns for the consequences of its failure in an indeterminate position. The Vice President-Generation again called (approximately 1010 hrs) to obtain an update on plant status from the Station Manager. (Initial notification, communications and activities of other Metropolitan Edison offsite personnel and those of General Public Utilities are covered elsewhere in this report and will not be repeated here.) (Int. 26, 27, 31, 53, 59, 71, 83, 129, 38, Ref: 21)

The B OTSG remained isolated at the direction of the Station Manager after the Site Emergency was declared. Samples from A OTSG confirmed it was not contaminated. It was decided to attempt to gain natural circulation cooling by filling the A OTSG to a 90% operating level and steaming it through the atmospheric steam dumps. As a precaution, an operator was stationed on the roof with a radiation detector as close as practicable to the exhaust, should this become a radioactivity release path. (Int. 16, 27, 31, 77, 83, 41)

The concern for the failure of the EMOV block valve persisted. By about 1120 hrs, the emergency command team decided to increase the pressure control range from 2000 to 2100 psig to a range of 1900 to 2100 psig, thereby cutting the frequency for cycling this valve in half. (Int. 27, 83; Ref. 23)

The Vice President - Generation (not currently licensed or required to be licensed on either unit) arrived at the Observation Center (approximately 1140 hrs) and called the Station Manager to learn the current plant status. The Vice President - Generation indicated during interviews that he did not proceed to the Unit 2 Control Room and insert himself in the command chain, since competent people were in charge, performing tasks for which they were trained. About this time it was decided to attempt to depressurize the RCS to obtain core flood tank injection and possibly enable DHR. This decision was reportedly the result of concerns related to the lack of positive indication of core coverage or natural circulation; fears that the EMOV block valve might fail, that boron might be plating out through crystallization, and that HPI might be bypassing the core; and recognition that the plant appeared no closer to forced circulation by RCPs or DHR pumps, even though a significant drop in BWST inventory had been experienced. A restart of RCPs was not considered here, since RCS loop and pressurizer parameters had not changed significantly. Consideration of an RCP restart would be delayed until the emergency command team was convinced the loops were sufficiently filled and subcooled to prevent RCP cavitation and seal destruction. (Int. 26, 27, 53, 71, 77, 83, 91, 113, 38, 41)

While maintaining HPI, the RCS was depressurized using the EMOV, EMOV block valve and the pressurizer vent valves. RCS pressure bottomed out at 440 psig, but not before some small (approximately 1 foot based on interviews) drop in core flood tank level was experienced, convincing the emergency command team that the core was indeed covered. HPI was reduced to minimum, but the pressure would not drop further. (Int. 26, 27, 53, 71, 77, 113)

As the afternoon progressed, the Vice President-Generation worked to improve communications with offsite and onsite groups. The Superintendent - Technical Support, recognizing the need for information following recovery from this event, directed an engineer to begin a chronological log of events. The Station Manager came under increasing pressure to secure steaming A OTSG through the atmospheric dump. Without condenser vacuum, the Station Manager hesitated to secure this system, since, in so doing, he thought progress in removing core heat through natural circulation would be lost. Ultimately, the Vice President - Generation directed it be secured and it was. Efforts to reestablish condenser vacuum were then redoubled. (Int. 16, 26, 31, 71, 91, 113; Ref. 21)

With the secondary heat sink isolated, it was decided to increase HPI flow while at low pressure and attempt to collapse the voids in the loops. Core cooling was maintained by opening the EMOV and/or its block valve to stimulate flow by the fuel assemblies. The Station Manager was directed by the Vice President - Generation to prepare for a briefing trip to the Lieutenant Governor's office. The Station Manager directed the Superintendent - Technical Support to gather the necessary information and prepare to accompany him. (Int. 27, 71, 91)

In a further attempt to improve circulation and collapse loop voids, the emergency command team decided to maintain HPI flow, but alternate its injection point to the RCS. The EMOV block valve was cycled to control pressure and during one opening (approximately 1350 hrs) a double "thump" was heard. Reactor building spray initiated, and ES actuation and reactor building isolation occurred. The Station Manager, then in the control room verifying plant

status, reportedly exclaimed: "What was that?" The connection between the noise, the start of reactor building spray pumps and a hydrogen burn would not be made until the following day when engineers reviewed plant data. The noise was attributed to shifting ventilation dampers, which were being worked on at the time. The start of the reactor building spray pumps and the spike in the reactor building pressure were attributed to an electrical noise spike by one supervisor. Another supervisor observed the pressure spike and recommended to the Station Manager that the EMOV not be cycled again, because he noted the rapid rise in building pressure corresponded to the timing of the opening of that valve. (Int. 16, 26, 71, 83, 102, 111, 113, 119)

The Vice President - Generation, Station Manager and Unit 2 Superintendent - Technical Support left the site (approx. 1400 hours) and proceeded to the Lieutenant Governor's office. Prior to leaving, the Station Manager satisfied himself that conditions were stable; ordered the Unit 1 Operations Supervisor to attempt to maintain the status quo and to not reinitiate atmospheric steam dump; appointed the Unit 2 Superintendent the acting Emergency Director; and called the Unit 1 Superintendent to tell him where he was going and how to reach him. The Station Manager carried a beeper and had the Superintendent - Technical Support reestablish phone communication, once he had reached the Lieutenant Governor's Office. During the absence of the Station Manager, the Unit 2 Shift Supervisor began to have some success in changing indicated A loop hot and cold leg temperatures and indicated pressurizer level, by directing HPI preferentially to the A loop. It was decided to attempt the same maneuver on the B loop, the execution of which was without success and, all previous progress observed on the A loop was lost. Increasing core flood tank overpressure was also discussed, but whether this action was ever taken has not been determined. No evidence or statements were obtained which indicate such action was taken. (Int. 16, 26, 27, 31, 53, 71, 77, 83, 91, 111, 113)

At the request of the Unit 1 Operations Supervisor, the Unit 1 Superintendent returned to the Unit 2 control room from Unit 1, where he had been since about 1000 hrs. The Unit 1 Superintendent assisted the Unit 2 Superintendent in maintaining communications and joined the emergency command

status, reportedly exclaimed: "What was that?" The connection between the noise, the start of reactor building spray pumps and a hydrogen burn would not be made until the following day when engineers reviewed plant data. The noise was attributed to shifting ventilation dampers, which were being worked on at the time. The start of the reactor building spray pumps and the spike in the reactor building pressure were attributed to an electrical noise spike by one supervisor. Another supervisor observed the pressure spike and recommended to the Station Manager that the EMOV not be cycled again, because he noted the rapid rise in building pressure corresponded to the timing of the opening of that valve. (Int. 16, 26, 71, 83, 102, 111, 113, 119)

The Vice President - Generation, Station Manager and Unit 2 Superintendent - Technical Support left the site (approx. 1400 hours) and proceeded to the Lieutenant Governor's office. Prior to leaving, the Station Manager satisfied himself that conditions were stable; ordered the Unit 1 Operations Supervisor to attempt to maintain the status quo and to not reinitiate atmospheric steam dump; appointed the Unit 2 Superintendent the acting Emergency Director; and called the Unit 1 Superintendent to tell him where he was going and how to reach him. The Station Manager carried a beeper and had the Superintendent - Technical Support reestablish phone communication, once he had reached the Lieutenant Governor's Office. During the absence of the Station Manager, the Unit 2 Shift Supervisor began to have some success in changing indicated A loop hot and cold leg temperatures and indicated pressurizer level, by directing HPI preferentially to the A loop. It was decided to attempt the same maneuver on the B loop, the execution of which was without success and, all previous progress observed on the A loop was lost. Increasing core flood tank overpressure was also discussed, but whether this action was ever taken has not been determined. No evidence or statements were obtained which indicate such action was taken. (Int. 16, 26, 27, 31, 53, 71, 77, 83, 91, 111, 113)

At the request of the Unit 1 Operations Supervisor, the Unit 1 Superintendent returned to the Unit 2 control room from Unit 1, where he had been since about 1000 hrs. The Unit 1 Superintendent assisted the Unit 2 Superintendent in maintaining communications and joined the emergency command

team that would periodically caucus during the period of absence of the Station Manager. By 1700 hrs, problems with the station auxiliary boilers were temporarily solved and a vacuum was drawn in the condenser. The A OTSG was then steamed thru its turbine bypass valves to the condenser in an attempt to establish natural circulation. The A loop temperatures and pressurizer level again appeared to respond to the preferential injection of HPI to the A loop. The emergency command team believed they were seeing the first signs of natural circulation. (Int. 16, 26, 27, 31, 77, 113)

The Vice President - Generation, Station Manager and Unit 2 Superintendent - Technical Support returned (approximately 1630 hrs) from their trip to the Lieutenant Governor's office and were briefed on current plant status and trends. The Vice President - Generation, Metropolitan Edison communicated this information to the Vice President-Generation, General Public Utilities, and between them, it was decided the plant must be repressurized. The emergency command team recommended continuation of their current policy of attempting to establishing natural circulation, since they believed they were now making progress and earlier attempts to establish stable conditions at high pressure were believed unsuccessful. The Vice President - Generation, Metropolitan Edison directed the Station Manager to repressurize. The decision to repressurize was based on a recognition that RCS water and metal temperatures were too high to allow pressure to be brought below the interlock for initiating DHR. Without an effective means to cool down the RCS, this goal of using DHR was unreachable, leaving high pressure RCP forced circulation as the only viable option. As pressure was increased using HPI with the EMOV block valve shut, it appeared the loops were filling and the emergency command team then recommended that an RCP be started once pressure was stabilized. Everyone, including B&W, appeared to concur on this course of action. (Int. 16, 26, 27, 31, 44, 53, 71, 83, 91, 113, 38)

RCS pressure was increased to about 2300 psig and maintained there by throttling HPI flow at greater than or equal to 400 gpm. B&W calculation results, now available to the emergency command team, indicated this was sufficient flow for the existing core decay heat. The history of RCP operations was reviewed

with B&W and it was determined that RCP 1A should be bumped. An RCP in A loop was chosen since it would provide the strongest pressurizer spray and previous indications convinced them A loop had liquid in it. The decision to initially bump the pump for 10 seconds was precautionary and followed normal practices utilized in RCS filling and venting procedures. Before the pump could be bumped, a number of interlocks had to be bypassed, based largely on the unavailability of the a.c. lube and lift oil pumps, which had previously lost power because of electrical faults on their motor control centers. At 1933 hrs, RCP 1A was run for 10 seconds. (Int. 16, 26, 27, 31, 53, 71, 83, 91)

As the RCP was bumped, RCS pressure and temperature dropped, OTSG pressures increased, loop flow and RCP current acted normally. Discussions were held as to the appropriate delay for restarting the RCP, a normal consideration relative to motor overheating, and it was decided to delay 15 minutes and then run RCP 1A, again. RCS pressure was increased to about 2200 psig and RCP 1A was started and left running at 1950 hrs. RCS pressure again dropped, but stabilized at about 1300 psig; hot and cold leg temperatures converged and stabilized at about 340°F; A and B OTSG pressures stabilize at 94 and 50 psig, respectively; the pressurizer level remained full scale. Stable plant conditions, as the emergency command team had defined them earlier, had finally been reached. (Int. 16, 26, 27, 31, 53, 71, 83)

Subsequently, pressurizer level would be regained and a plant cooldown on A OTSG would occur. Activities beyond 2000 hrs on March 28, 1979 are beyond the scope of this report and, therefore, are not covered here.

A summary of notification and arrival times of selected individuals during this accident is included on Table I.3-1.

TABLE I.3-1

NOTIFICATION AND ARRIVAL TIMES (APPROXIMATE) OF KEY PERSONNEL

<u>INDIVIDUAL</u>	<u>NOTIFICATION(S)</u>	<u>APPROX. TRAVEL DISTANCE (MILES)</u>	<u>ARRIVAL TIME</u>
Station Manager	0401 (initial notification; see Section 3.1 for subsequent contacts & actions)	10	0705
Unit 2 Superintendent	0410	50	0545
Unit 1 Superintendent	0545	15	0645
Maintenance Superintendent	0515	10	0615
Superintendent - Tech. Support (Unit 2)	0410	10	0450
Unit 1 Operations Supervisor	0435	20	0600
Chemistry/HP Supervisor	0510	15	0545
Site Manager (NSSS Vendor)	0600-0635 (Conference Call)	15	0715
Instrument/Control Engineer	0600	10	0630
On-Call Operating Engineer	0501	15	0545

3.3 Provision of Technical Support

3.3.1 Onsite Technical Support

Basically four engineering disciplines (mechanical, electrical, nuclear and instrumentation) were represented during the event. In addition to the plant staff, there were two individuals employed by B&W and assigned to the plant staff who were present in the control room. The major discipline actively participating in operational related matters was instrumentation. The instrumentation engineer (Int. 121, 192) was involved in the following areas:

- o Verifying RCS temperatures and pressures by comparing meter indications in the reactor protection system (RPS) cabinets for two or more instrument channels. He informed the Unit 2 Superintendent-Technical Support that he could see no reason not to believe the instruments.
- o Deploying personnel and equipment to measure hot leg RTD temperatures and evaluating the results. He could see no reason not to believe the instruments and informed the Unit 2 Technical Superintendent of this fact.
- o Deploying personnel and equipment to measure incore thermocouple temperature and evaluating the results. He informed the Station Manager of temperatures greater than 2000°F in the core. It should be noted that during the measurement of the incore thermocouple temperatures, statements were made by the individuals taking the measurements that they felt the core was uncovered. They stated that they so informed the instrumentation engineer. (Int. 181, 183)
- o Evaluating conditions associated with RCP interlocks to allow a RCP start. He initially tried unsuccessfully to manually energize the K-3 relay on an RCP. He was later involved in jumpering interlocks to allow a RCP start.

The Lead Mechanical and Electrical Engineers (Int. 70) were involved in notification and communication as required by the Emergency Plan. There is some evidence that the Lead Electrical Engineer was involved in the RCP interlock jumper. (Int. 193) In addition, the Lead Electrical Engineer recommended that the diesel generators' control switches be placed in the maintenance position and the fuel rack be reset (Int. 195)

A nuclear engineer (Int: 97) was involved in the performance of a shutdown margin calculation that was performed shortly after the trip. This calculation showed the reactor shutdown margin to be 8-10%. Later during the event, when the SRM and IRM showed an anomalous response, he was asked to verify the calculation. He requested boron samples and performed a calculation based on a boron concentration of 404.5 PPM. This calculation showed the reactor was shut down by $-2.445\% \Delta k$. This fact was reported to the Unit Operations Supervisor. (Int. 26)

The Lead Nuclear Engineer (Int. 48) was involved in offsite release calculations as required by the Emergency Plan. After performing the above duties, he evaluated the following conditions and drew the indicated conclusions:

- o The SRM and IRM showed anomalous indication. There was increased radiation levels at the detectors causing the SRM and IRM response and not a return to criticality.
- o The incore neutron detector responses, as indicated by the backup recorders, were giving erroneous indications.
- o The T_{hot} and T_{cold} temperatures. That T_{cold} was as expected. T_{hot} was high.

There is no evidence that a staff nuclear engineer evaluated the incore thermocouple readings during March 28, 1979.

The B&W Site Operations Manager was consulted by telephone (Sequence of Events; Ref. 53) prior to coming on site. During this consultation he questioned whether the block valve on the EMOV relief line had been isolated. He also participated in the decision to reestablish RCS flow (by the start of a reactor coolant pump). At approximately 0745 hrs, the Site Operations Manager advised the B&W corporate offices of the event (See Section 3.3.2). (Int. 53) (The B&W Site Operations Manager reported to the B&W Manager of Plant Startup Services and as a result of a contractual agreement with Metropolitan Edison Company reported to the Station Manager. The B&W Site Operations Manager received direction from the Metropolitan Edison Company in the form of written communication.)

During the day, the Site Operations Manager was in contact with a B&W employee offsite who relayed information to and from the corporate offices. (See also Section 3.3.2.) During one of these calls a request was made for radiochemical expertise at the request of the licensee. The personnel requested did arrive from B&W in Lynchburg in the area in the afternoon of March 28, 1979, but they did not enter the plant that day. They were involved in organizing mobile laboratory assistance offsite.

The Site Operations Manager participated in the following decisions:

- o To increase pressure to collapse the steam void (he did not believe the T_h indications and he evidenced some concern about increasing the pressure to the point that the safety relief valves would lift.)
- o To decrease RCS pressure and "float" the core flood tanks on the core.

The Site Operations Manager participated in the evaluation of HPI flow paths that would have resulted in the low pressurizer temperatures.

The B&W Physics Test Coordinator (Int. 16) was also present in the control room after approximately 0900 hrs. He evaluated the following areas with the conclusions as shown:

- o The SRM and IRM indication anomaly. The SRM and IRM anomaly was caused by a change in neutron leakage sensed by the detectors.
- o The high T_h indication. The high T_h temperature indicated super-heated steam in the hot legs and he advised that the steam could not be condensed by increasing system pressure.
- o Steam generator performance during natural circulation. The steam generators were not promoting natural circulation. He recommended that feedwater levels be increased to increase the steam generator heat removal capability.
- o The incore thermocouple indications. He recommended that the incore thermocouples be monitored to indicate core conditions.
- o Indications from the backup recorders for the incore neutron detectors. No information was derived from the backup recorders.

He stated that these recommendations and conclusions were discussed with the B&W Site Operations Manager.

3.3.2 Babcock and Wilcox

At 0745 hrs, the B&W Site Operations Manager contacted the B&W Manager of Plant Startup Services located in the Lynchburg, Virginia, Corporate Offices. The following information was transmitted (Int. 185) as indicated by notes* taken in the corporate offices:

- o There was a loss of feedwater caused by condensate polisher isolation valve malfunction.
- o There was a turbine trip.

*These notes contain preliminary information and as a result may not be entirely accurate.

- o There was a reactor trip on high pressure.
- o There was initiation of high pressure injection.
- o The pressurizer went solid.
- o RCS pressure went to approximately 2500 psi.
- o The reactor coolant drain tank rupture disc had blown.
- o There was an indication of fuel failure.
- o There was a reading of 800 R/hr at the dome of the reactor bldg.
- o There was a loss of RCS flow indication.
- o The reactor coolant pumps had been tripped.
- o There was an indication of a primary-to-secondary leak.
- o Present RCS conditions were T_{cold} : 300°F, pressure: 1500 psig.
- o The site was in a state of emergency.

After this information was received, a task force was formed at B&W in Lynchburg. At 0900 hrs a task force information transmittal meeting was held in a classroom adjacent to the simulator at the B&W Training Center. The Manager of Plant Startup Services presented the information as received from the Site Operations Manager.

Notes taken during the day indicate the task force placed a high priority on ensuring that the core remained covered and that it was being cooled

adequately. During the meeting three engineers were selected to go to the site. The personnel were selected on the basis of their knowledge of systems, analysis capability, experience with similar events at Oconee and Davis-Besse, and, in some instances, having qualifications for entry onsite. These personnel arrived in the site area at about 1330 hrs, but they did not gain entrance until after 2000 hrs, the period covered by this portion of the report. A list of data to be requested during the next contact with the site was developed. An unsuccessful attempt was made to telecopy this request to the site. (Int. 84, 85, 86)

At approximately 1030 hrs, information was obtained from the Unit 2 Operations Supervisor, who was in Lynchburg for simulator training. The Operations Supervisor had been in contact with the site and via the Unit 1 control room had obtained the following additional information which was given to the task force:

- o There was a primary-to-secondary leak in the B steam generator.
- o There was 60,000 R/hr in the dome of the reactor building. (This was noted as doubtful).
- o There were 10^4 counts on radiation monitor 748, (200 is background).
- o There was 100 mR/hr at the personnel hatch outside the reactor building.
- o "Iodine was high off-scale."
- o The plant was in natural circulation cooldown.
- o "Incore temperature was 450°F" (computer output).
- o T_{cold} was 250°F.

- o On and offsite area radiation was negligible.
- o There was water in the control air lines of the condensate polisher isolation valves, which caused the valves to shut and cause a loss of feedwater.
- o There was an ESFAS actuation.
- o The makeup pumps were stopped and letdown established (during RCS high-pressure condition).
- o The EMOV block valve (RC-V2) was shut after the quench tank rupture disc blew.
- o Auxiliary feedwater initiated, but there was no flow to the steam generators until approximately 12 minutes after the trip. (Later information indicates 8 min.)
- o The steam generators did not go dry.
- o RCS pressure went as low as 1200 psi; saturation conditions were possibly reached.
- o The RCS flow decreased by approximately 1/3; RCPs were tripped.

The Unit 2 Operations Supervisor (Int. 34) performed a rough calibration based on radiation levels he had been informed of and estimated that 1/8 of the cladding had failed. Attempts were made by him to duplicate the event on the simulator based on the information he had obtained but he was unsuccessful.

At 1145 hrs, a message was relayed to the task force from the B&W Site Operations Manager via a B&W employee offsite. The following information was conveyed:

- o The RCPs were shut down.
- o The plant personnel were trying to go solid; there were indications of steam bubble in B loop.
- o Low-level radioactivity was reported in the atmosphere.
- o The Metropolitan Edison Company had made a public announcement to the news media.
- o Radiation teams were performing surveys on and off site.
- o An NRC team had been sent to the site to investigate.
- o The primary-to-secondary leak in the B steam generator had been confirmed by sample analysis.
- o The B steam generator was isolated.
- o Natural circulation was being used to cool down the plant using the A steam generator.
- o Suction for HPI was being taken from the BWST.
- o The pressurizer heaters were shorted out.
- o The electromatic block valve was being used to control pressure.
- o Present RCS conditions were T_{cold} :300°F; pressure:2100 psig.
- o There were plans to cool down and depressurize.
- o There was speculation of fuel leakage but no further information on radiation levels.

- o There was some level increase in the reactor building sump.
- o Component cooling water and seal injection had been maintained.
- o There was high moisture level in the reactor building.
- o Radiochemical expertise was requested.

At approximately 1200 hrs, the B&W task force was involved in determining a recommended course of action for cases with and without RCPs. An assignment was also made to a task force member to determine the prerequisites for starting a reactor coolant pump.

At approximately 1330 hrs, the following information was relayed from the site by the B&W site Operations Manager to the B&W task force via the B&W employee offsite:

- o RCS pressure was 495 psig on core flood tank float.
- o The T_{hot} was 700°F as measured by a digital voltmeter.
- o The pressurizer was full.
- o Pressure was being controlled by the electromatic relief block valve.
- o The HPI suction was on the BWST in process of switching to the RC bleed holdup tanks.
- o B steam generator was isolated; the level was approximately 60% of the operating level.
- o There was minimum cooldown from the A steam generator.
- o The atmospheric dump valves were being used as the heat sink.

- o The gland sealing steam was lost as the turbine-condenser vacuum was broken.
- o Emergency feedwater to the A steam generator was being supplied through the main feedwater nozzles.
- o There was less than 1 mR/hr at the site security fence.
- o Radioactivity was reported in the feedwater system.
- o There was airborne activity in the auxiliary building and control room (reactor building sump was being pumped to the auxiliary building sump before it was discovered and isolated).
- o Attempts had been made to start RCPs (one in each loop) but drew only about 100 amps (no-load current).
- o It was not clear why the RCPs were originally stopped.
- o That the three people originally sent to the site from B&W had arrived in the area.
- o The B&W Physics Test Coordinator was on site.
- o It was planned to go to decay heat removal system operation as soon as possible.
- o The BWST level was last known at about 37 feet.
- o Arrangements had been made for another call from the Site Operations Manager.

The following recommendations of the B&W task force were then made to be relayed to the B&W Site Operations Manager via the B&W employee offsite:

- o Obtain cooldown data.
- o Ensure accurate RCS temperature before going to decay heat removal system operation.
- o Confirm core outlet temperature by pressurizer temperature since this is now the flow path.

The information received by the B&W task force regarding the planned operation to go on the decay heat removal system, and the recommendation to ensure accurate RCS temperature before going on to DHRS operation tends to indicate that there was an intent on the part of the licensee to go into DHRS operation. In addition, the recommendation regarding inferring the core outlet temperature by reading the pressurizer temperature indicates that the task force had some knowledge about the flow of coolant in the reactor coolant system at that time.

At approximately 1400 hrs, the Manager of Plant Systems Design (Int. 87) recommended establishing at least 400 gpm of HPI flow. This value had been provided by the ECCS Systems Manager based on the decay heat at that time and included 50% conservatism. This recommendation was relayed to the site in two ways. The B&W Manager of Project Management (Int. 88) contacted a vice president for GPU (Int. 90). The message was also communicated to the Unit 1 control room by the Unit 2 Operations Supervisor for relaying to the Unit 2 Control Room.

At this time, the Manager of Project Management contacted the Manager of General Engineering at Metropolitan Edison and requested that a direct communication link be established between the Manager of Nuclear Services and the site.

During the time period 1400 hrs to 1600 hrs, additional information was provided by the Unit 2 Operations Superintendent based on contacts with the site:

- o The EMOV had stuck open (resulting in the rupture of the reactor coolant drain tank rupture disc).
- o There had been a delay in emergency feedwater because of incorrect indication (pumps running - valves indicated open but were not).
- o The RCPs were stopped because of an indication of cavitation (indicated drop in RCS flow); condenser offgas alarm first indicated near the steam generator.
- o The 60,000 R/hr reactor building dome reading was bad because of moisture (pegged high).
- o The best estimate of reactor building radiation level was approximately 100 R/hr based on 100 mR/hr at the personnel hatch.
- o The reactor building pressure went initially to approximately 2.5 psig and then to approximately 4 psig about 6 hours into the transient.

Sometime prior to 1600 hrs, the individual assigned the task of making recommendations regarding the restart of the reactor coolant pump reported to the Task Force that:

- o There was a concern about moisture being a problem in starting the RCPs (motors).
- o The electrical circuit couldn't be meggered (checked for faulted condition). It was suggested that the component cooling water (Int. closed cooling water) leak alarm (moisture detector) could be checked.
- o Up to 30 mils vibration could be tolerated for a short period of time. In addition, a reactor coolant pump associated with the spray line should be chosen for operation if possible.

- o There is a possibility that the reactor coolant pumps were not connected (sheared drive pin) because no-load amperage should be approximately 100 amps, which is the indication received on earlier attempts to restart, if connected, RCP starting current should be greater than 100 amps.

At approximately 1600 hrs, it was reported from the B&W Site Operations Manager to the task force via the B&W employee offsite that:

- o It was difficult to establish direct communications with the site.
- o The "bubble" had been collapsed in the "A" leg -- good indication from temperatures and pressures.
- o There was normal letdown and pressurizer electromatic isolation valve pressure control.
- o There was an attempt to collapse the bubble in the B loop.
- o The pressurizer level was 190 inches; pressure: 560 psig; temperature: 460°F.
- o Plans were to get on DHR via the suction from the BWST.
- o The core flood tanks were still "floating." Pressure: 400-500 psig; temperature: 540°F.
- o RCP-2A had indications of leakage but unsure.
- o The cooldown data was available on the reactimeter tapes.
- o The level in the A steam generator was at 80% of the operating level.

- o The level in the B steam generator was at 50% of the operating level.

The following recommendations were made to the Site Operations Manager via the B&W employee offsite by the task force during this telephone call:

- o Establish 400 to 500 gpm of HPI flow.
- o The minimum recommendations for RCP start:
 - 30 mils vibration limit.
 - HPI injection providing greater than saturated conditions for the loop.
 - "Good" amperage reading.
 - Component cooling water (int. cooling water) flow.
 - Normal prerequisites for start.
 - No steam flashing at the seals.

During the call, a request from the licensee was relayed to the task force by the B&W employee offsite concerning what were the considerations for running an RCP (at least one) in the B loop (decay heat drop line side).

At 1640 hrs, information originating in the Unit 2 control room was received that the T_{hot} temperature was 550°F and the pressure was 450 psig (superheated conditions).

In the period from about 1630 hrs to about 1700 hrs, the B&W Manager of Project Management again emphasized to the GPU Vice President that HPI flow of 400-500 gpm should be initiated. The GPU Vice President advised that HPI flow had been initiated.

At 1745 hrs, the following information was relayed from the Site Operations Manager via the B&W employee offsite:

- o A vacuum was being established in the condenser.
- o High pressure injection flow was 100 gpm.
- o Reactor pressure increased from 560 to 670 psig.
- o The core flood tanks were no longer "floating" on the RCS.
- o There was a bubble in the pressurizer.
- o A "bubble" was suspected in the A loop.
- o The T_{cold} was 520°F and T_{hot} was 530°F.
- o The A steam generator was at 180 psi, and the level was at 80% of the operating range.
- o The HPI was taking suction from the BWST.

During this call, it was recommended that the HPI be immediately increased, letdown be terminated, and subcooled conditions be attained in the RCS.

At 1810 hrs, a telephone call was received from a vice president of GPU indicating that, since 1620 hrs, HPI had been maintained at 400 ppm.

At 1835 hrs, a telephone call was received directly from the Site Operations Manager in the Unit 2 Control Room. He advised that:

- o Cooling was from the A steam generator via the dump valve to the condenser.

- o The A loop conditions were T_{cold} : 300°F; T_{hot} : 540 to 550°F.
- o The RCS was going "solid."
- o Pressurizer level was stable at a temperature of approximately 500°F.
- o The pressurizer heater capacity was limited because of shorts.
- o There were "bubble" indications in the B loop with T_{cold} at 200 and T_{hot} was off scale high (>650°F).
- o RCS pressure was 1800 psig and increasing
- o HPI flow was 400 gpm.
- o A steam generator pressure was 50 psig (consistent with a T_{cold} of 300°F).
- o A steam generator T_{cold} was decreasing with time
- o The A steam generator T_{hot} was consistent with pressurizer pressure.
- o There were two incore thermocouple readings higher than 500°F. The computer was printing out "?????" for the others.
- o Seal injection had been maintained since 0700 hrs, and before the transient.
- o The RCPs had good bearing temperatures and seal pressures.
- o There was no indication that the ESFAS had isolated the seal injection to the RCPs.

- o There was 100 R/hr in the auxiliary building (reactor building sump automatically pumped to the auxiliary building sump; secured later).
- o There was 100 mR/hr in the plant.
- o There was 70 mR/hr outside the gate.
- o There was more than 1 mR/hr on State routes near the site.
- o Moisture and high radiation levels were limiting the use of some equipment.
- o There was an indication of a primary-to-secondary leak from B steam generator by sampling, but the magnitude of the leak was not known.
- o The B steam generator was isolated before 0700 hrs.
- o The electrical buses supplying the oil lift pumps needed for RCP start were lost.

At 1900 hrs, the following information was received during the same telephone call:

- o The T_{hot} was 560°F.
- o The oil lift pumps were now running.

At about this time, the B&W staff member assigned the task of making recommendations regarding the start of an RCP provided the following information to the task force:

- o Seal injection was to be maintained at 12 gpm.

- o The maximum seal return flow was to be less than 1.9 gpm.
- o There should be a supply of nuclear service water.
- o There was no alarm on the component coolant water (CCW) leak alarm.
- o The starting current should be greater than 600 amps.
- o The vibration limit was 30 mils (peak to peak).
- o There should be clear motor permissive start interlocks.
- o The seal injection temperature should be less than 150°F.
- o The seal return temperature should be less than 185°F.

During this same call it was reported that:

- o The 2A and 2B RCPs were the pumps used earlier in attempted restart.
- o The A steam generator pressure was 50 psig, and the level was 80% on the operating level and decreasing.
- o The B steam generator was at 70% level.

The recommendation was made to the Site Operations Manager by the task force that the RCP-1A be given a five-second start ("bump") and then that it be stopped to let the RCS parameters stabilize.

At 1930 hrs, it was reported to the task force by the Site Operations Manager that the RCP-1A was given a 10 second "bump" start/stop by the operators. In addition, the steam generators were at 200 psi and the A steam generator steaming rate increased. It was reported that the RTD indications were consistent with conditions.

At 1940 hrs, the following information was received during the same telephone call:

- o The A loop T_{hot} was less than 520°F and the T_{cold} was 320°F. The A steam generator pressure was 60 psig, and temperature was 325°F.
- o The B loop T_{hot} was 620°F and T_{cold} was 225°F. The B steam generator pressure was 160 psig and temperature was 390°F.
- o The RCS pressure was 1850 psig and it had dropped to 1400 psig
- o RCS flow went to 40%.
- o The RCP starting current was 1200 amps (normal).

At this time the task force evaluated conditions for a second "bump" of the RCP or for a start with continued running of the RCP. The recommendation was made by the task force to the Site Operations Manager to start and run the 1A RCP. However, the Station Superintendent had already ordered that this be done.

At 1950 hrs, it was reported by the Site Operations Manager during the same telephone call that the 1A RCP had started and was running.

At 1955 hrs, it was reported by the Site Operations Manager during the same telephone call that the plant conditions were as follows:

- o RCP-1A vibration was 18 mils.
- o Both loops had T_{hot}
- o The steam generators were at 100 psi.

- o The plant was stable.
- o The RCS pressure was 1800 psig.
- o The RCP current was 620 to 580 amps.
- o The RCS flow was 40%.
- o The pressurizer temperature was 500°F.
- o The incore thermocouples indicated approximately 400°F.

The Site Operations Manager stated that it was planned to stay on one RCP and cool down to get DHR in operation, which was expected in a few hours.

In addition, the Site Operations Manager was asked by the Manager of Plant Startup Services to call when DHR was in operation or if there were any other problems or assistance was needed. The Site Operations Manager stated no further assistance was needed at this time and the Task Force was adjourned.

3.3.3 Corporate Technical Staff

The licensee Emergency Plan requires that notification be made to Division Headquarters, in particular, the Metropolitan-Edison Vice President - Generation, in the event of a Site or General Emergency. Contact with GPU Service Corporation is implied as required but is not stated explicitly. Notification is based on the requirements stated in Emergency Procedure 1670.2, Revision 9, Figure 4, and in Administrative Procedure 1014, Recall or Standby Personnel to Plant, Revision 3. The Emergency Plan also requires, as needed, Metropolitan Edison, Division of Engineering support, and Metropolitan Edison, GPU Service Corporation engineering/technical services. The Emergency Plan is silent on specific requirements to be provided by the Metropolitan Edison, Division of Engineering, and GPUSC engineering/technical services.

The first call to Metropolitan Edison corporate management was between approximately 0600 hrs and 0630* hrs. The Metropolitan Edison Vice President - Generation was called again at 0730 hrs to be informed of the declaration of a General Emergency but this call was not completed. The Metropolitan Edison Vice President - Generation was told to leave Philadelphia by helicopter and report to Three Mile Island. He left by helicopter from Philadelphia at approximately 1100 hrs, arrived at the TMI Observation Center at about 1140 hrs and established phone contact with the Station Manager who was located in the Unit 2 control room, the Emergency Control Center. The Metropolitan Edison Vice President - Generation had become aware sometime between 0830 hrs and 1100 hrs that a general emergency had been declared. (Int. 91)

Emergency assistance was not requested from the Metropolitan Edison headquarters staff for direct assistance to operations for the first 16 hours. The Metropolitan Edison Vice President - Generation, provided consultation, and through joint agreement with the Site Emergency Director, decisions were made on operational evaluations until 2000 hrs. (Int. 91)

The Metropolitan Edison Manager of Generation, (Operation) was called at about 0700 hrs. The call was taken by Metropolitan Edison Manager of Generation (Engineering) because the Manager of Generation (Operation) was out of town. The Manager of Generation (Engineering) was informed that a Site Emergency had been declared for TMI-2. A second call at approximately 0730 hrs informed the Manager of Generation (Engineering) that a General Emergency had been declared (Int. 159)

The involvement of the Metropolitan Edison engineering corporate staff was of a standby nature. There was no technical feedback to operational activities nor were decisions being made by the Metropolitan Edison corporate staff for transmittal to the TMI-2 control room. The Manager of Generation (Engineering) was advised that Unit 2 had a loss of feedwater and a turbine trip followed by a reactor trip and that the levels of a General Emergency had

* Times may not be in agreement with the time as stated in the referenced interviews but are based on data acquired from various sources and represents the most probable hour.

been reached. The President of Metropolitan Edison was apprised of the situation at approximately 0755 hrs. Contact with the Station Superintendent at approximately 0940 hrs resulted in additional information being given to the corporate staff. (Int. 159)

The Metropolitan Edison Manager of Generation (Engineering) and Staff was contacted by B&W and was advised that three B&W personnel were being sent to TMI to provide assistance. He was also called by the Metropolitan Edison Vice President - Generation at about 1200 hrs and given a status of Unit 2 that was similar to the information given to him by the Station Manager. Specific data with regard to temperature and pressures were not transmitted, but general evaluations of the plant were transmitted to the corporate staff. The support provided by the Metropolitan Edison corporate staff from late afternoon and into the evening was that of assisting communications service personnel and answering calls from the public and the press. (Int. 159)

The General Public Utility Service Corporation (GPUSC) was contacted by the Metropolitan Edison Manager, Operational Quality Assurance at approximately 0759 hrs. Contact was made by phone to the Vice President - Generation, GPU Service Corporation. He was informed that a Site Emergency existed at TMI and that TMI-2 had undergone a turbine and reactor trip, the steam generator appeared to have a primary to secondary leak, and there were increased radiation levels in the reactor building. (Int. 90)

The Emergency Plan does not identify specific responsibilities on the part of GPU Service Corporation, but it is implied that engineering/technical services would be available as necessary. There are no dedicated communication links between the GPU Service Corporation office and the TMI site.

There was contact with the control room on behalf of B&W to communicate with the B&W Site Manager. Contact was made with the B&W Site Manager and the Station Manager. This phone contact was made at about 1000-1030 hr. At this time, the Vice President - Generation, GPU Service Corporation, did not know that a general emergency had been declared. (Int. 90). Later (about 1145 hrs), he became aware that there were indications of offsite releases. The GPU

Service Corporation staff had general information about the status of the plant, but they did not have specific hard data. The information they had included the following: turbine trip/reactor trip, instrument air problem, condensate pump trip, RCP trip, OTSG possible leak. (Int. 165) This information was known and discussed at a meeting held around 0900-1000 hrs at the GPU Service Corporation headquarters. Notes from a GPU Service Corporation staff member indicates that at a meeting held at 0915 hrs they also were told that the condensate polisher isolated, there was low reactor coolant pressure, there was high pressure injection, the reactor coolant was solid, reactor coolant at 350°F and containment dome radiation alarmed. The GPU Service Corporation management decided to dispatch five of their technical staff to the site to provide assistance related to the information they had available.

The GPU Service Corporation engineering group arrived, by automobile, at the observation center over a period from 1400 to 1730 hrs. They were given a briefing by the Superintendent - Technical Support at the Observation Center at approximately 1805 hrs. (Int. 176) The dispatched group did not provide assistance or make decisions with respect to plant operation during the first 16 hours of the event. One individual of the group did enter the TMI-2 Control Room at about 1900 hrs to obtain data so that the GPU Service Corporation group could have specific information to allow an evaluation of the event. He was in the control room for about four hours.

The GPU Service Corporation management personnel that remained at their Headquarters were involved in the recommendation to increase reactor coolant makeup flow, take the plant to 2000-2300 psig and to restart a reactor coolant pump. This was around 1630 hrs. (Int. 90) The decision to recommend this action was transmitted to the Metropolitan Edison Vice President - Generation.

3.3.4 Burns and Roe (Architect Engineer)

The investigator made contact by telephone with a Burns and Roe (B&R) site supervisor with respect to his or Burns and Roe's involvement with the event of March 28, 1979, at TMI-2. He stated that he made contact through a

representative of Met Ed Quality Control to inform the Unit 2 Superintendent of B&R willingness to provide assistance if it were deemed necessary. Initial contact was at about 0730 hrs at the TMI observation center. B&R was not requested nor did they provide operational assistance during the first 16 hours of the March 28, 1979, accident at TMI-2.

3.4 Licensee/NRC Interface

3.4.1 Notification of NRC

A Site Emergency was declared by the Unit 2 shift supervisor at about 0656 hrs. At 0704 hrs, the first call from the site was received by the NRC Region I answering service. (Ref. 3, 54) The answering service first tried to reach the Region I Duty Officer at his home, but he had left for the Region I Office. The answering service then tried to signal the Duty Officer's beeper, starting at 0720 hrs, but the signal was not received until 0738 hrs, at which time the Duty Officer had almost reached the Region I Office and he decided to proceed to the office to answer the call.

Meanwhile, a General Emergency had been declared at 0724 hrs by the Station Manager. (Ref. 2) The site called the Region I Office regarding the General Emergency at 0740 hrs. (Ref. 3) At 0745 hrs, the Region I telephone operator took over for the answering service and notified the Reactor Operations and Nuclear Support Branch Chief of the General Emergency. The Region I Director was informed of the General Emergency about 5 minutes later and ordered activation of the Regional Emergency Center at about 0800 hrs.

3.4.2 Communications Between NRC and the Licensee

Communications were established between the Region I Incident Response Center (IRC) and the licensee and between the IRC and the NRC Headquarters Office of Inspection and Enforcement by 0810 hrs. (Ref. 4)

By 0839 hrs, a permanent line had been established between the Operations Center (OC) in NRC headquarters and the Region I IRC, and it remained open for

the rest of the day (IRACT tapes). The IRC relayed information between OC and the site. This procedure as well as the time spent in clarifying information resulted in considerable delay (one example was about 15 minutes) in transmitting information between the OC and the site. At 1015 hrs, Unit 2 control room was evacuated and communications were interrupted until contact could be made with the Unit 1 control room. Communications gradually improved as the day progressed.

At about 1030 hrs, contact was established with Region I personnel who would obtain information and provide it directly to NRC (IRACT tapes). Later (prior to 0130 hrs), a conference call was established between the HQ OC, the Region I IRC, and the site. At times (for example about 1100 hrs), communications were hampered by a high noise level and the necessity of using respirators. About 1500 hrs, a phone contact was established with the Unit 1 Shift Supervisor's office, which remained open for the rest of the incident. (TMI phone bill; Ref. 3)

3.4.3 Impact of NRC on Licensee Actions

Prior to the arrival on site of NRC Region I personnel, the NRC's principal contacts with the licensee were for the purpose of obtaining information. The first team of Region I inspectors arrived on site shortly after 1000 hrs. Two of them went to the Unit 2 control room (Int. 31, 62), the first arriving in the control about 1100/1130 hrs. Their main function was to gather information and relay it to NRC through the direct phone contact that had been established. They reported that there were around 20 to 25 people in the control room and that, for the most part, actions taken were calm and deliberate. They dealt mainly with a relatively small group of people in the middle of the control room. These inspectors spent considerable time getting temperature readings as these were of considerable interest to NRC. Some were taken from instruments, but most were obtained directly from the computer, which was manipulated for this purpose by NRC personnel. They reported that they were asked by licensee personnel for their ideas and they made comments in response, but no significant recommendations were made. One inspector recalls the recommendation being made that the licensee think hard before putting the decay heat removal system

into operation. At one point an inspector pointed out to the Station Manager that the steam being released to the atmosphere might be contaminated.

As the day progressed NRC headquarters expressed concern over various areas. These concerns were made through the inspectors or directly to licensee representatives. The following examples were derived from the tape recordings made of phone conversations in the NRC:HQ Operators Center (OC):

- o At 1215 hrs NRC Headquarters requested that licensee be asked if they considered blowing down the primary system, and if they considered bumping the RCP's. It appeared that this is the first instance of NRC Headquarters asking questions of a planning nature rather than a status nature.
- o At 1317 hrs NRC expressed concern that leaving core flood tank isolation valves open may lead to injecting N₂ into vessel. This concern was reiterated at approximately 9 hrs. 30 min. In both cases, it was responded to by indicating the licensee's conclusion that N₂ injection was not possible, based on system design and plant conditions.
- o At 1400 hrs NRC Headquarters expressed their concern that continued HPI injection would prevent discharge of the core flood tanks, with the result that the system will remain above the DHR interlock for a long period. OC asks how licensee is going to get the pressure down so that DHR can be put into use.
- o At 1559 hrs NRC Headquarters asked the Unit 1 Shift Supervisor (over the telephone) to communicate the NRC concern that the pressurizer level indication did not preclude a bubble in the core, and that the temperature readings indicating superheat may have been real and imply that the core was uncovered. They would then need to find a way to put more water into core and to get the core level back up. The question was asked if the licensee had talked to B&W,

because B&W had been trying to get in touch with the licensee and had the same concern.

- o At 1615 hrs NRC Headquarters asked if the licensee had considered blowing the system down, and the supervisor responded that it had been discussed and rejected. NRC:HQ requested that it be considered again, stressing this was a request to consider it, not an order to do so. Approximately 5 minutes later the supervisor returned stating the licensee thought it was a good idea and decided to do it. NRC:HQ then relayed their concern about valving out the CFT's before the blowdown to prevent nitrogen from getting into the vessel.

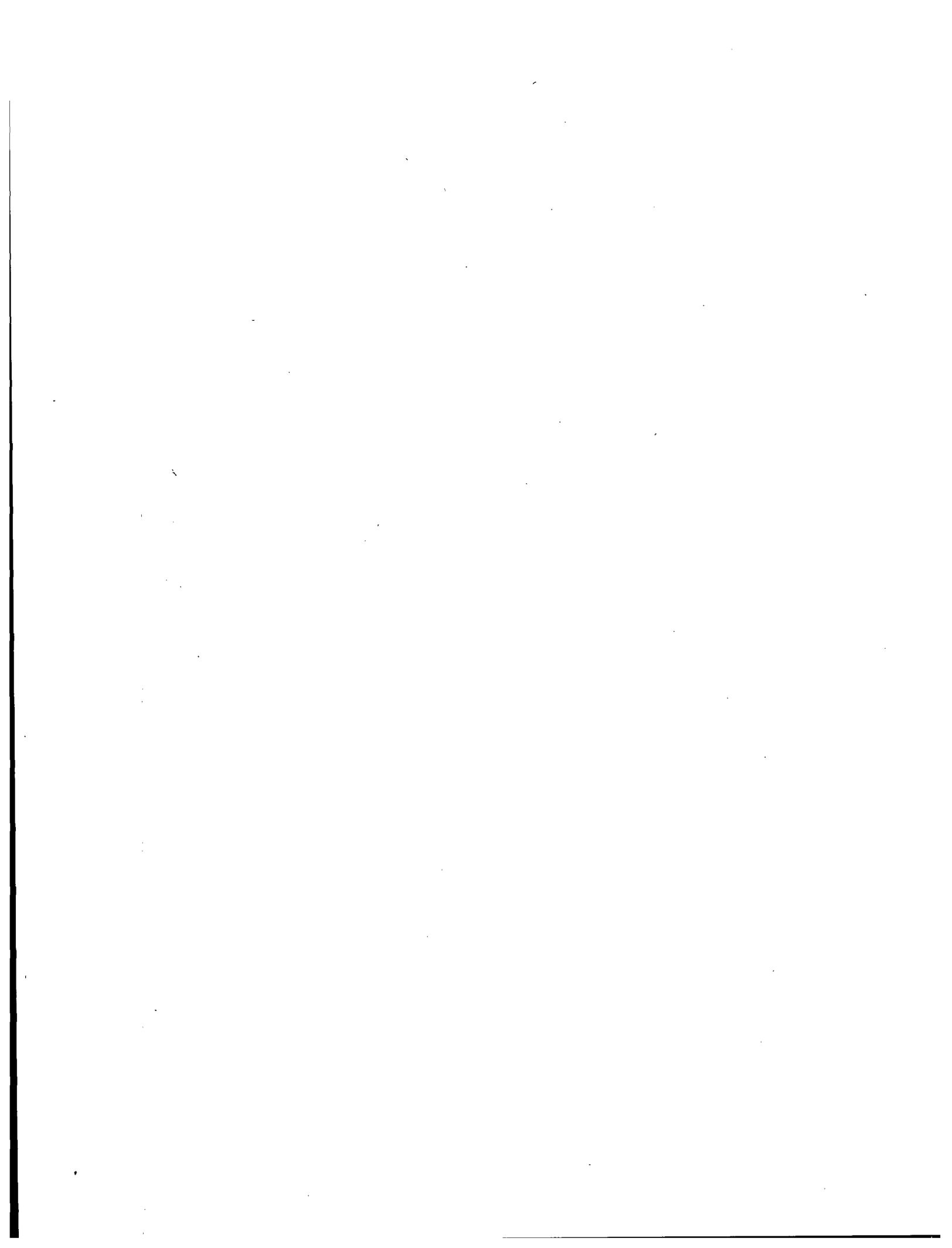
- o At 1845 hrs NRC Headquarters asked the inspector onsite if the licensee had considered running an RCP. Inspector stated that those preparations were underway but that they were having trouble with the oil lift pumps. NRC:HQ asked if the licensee had considered that they might have a gas bubble rather than a steam bubble in the "B" loop. They were concerned that it could be nitrogen, xenon, or hydrogen. Inspector agreed to pass the concern along. This concern apparently developed from evidence that only the A leg bubble quenched while the B loop remained superheated. Shortly thereafter, the inspector reported back that there was nothing licensee could do about it. There was no way to vent that leg, and the only way to seep it out, whether it be steam or gas, would be to use the pumps.

Evaluation

NRC's initial impact on the licensee was to monopolize the attention of certain key licensee technical personnel in providing notification and communications. Licensee personnel involved in maintaining these communications were faced with difficult problems of nomenclature and spent much of their time in training NRC personnel on plant systems. As time progressed, the nature of the communications shifted from training to information request and transmittal, and finally to information exchange.

NRC operation inspectors arrived in the Unit 2 Control Room too late to have an effect on the events that led to the accident. Subsequently, their time was occupied with gathering information and communicating with the NRC. The inspectors occasionally made suggestions but it appears their presence had little impact on the decisions made or actions taken by the licensee.

NRC management began to make recommendations to the licensee after 1200 hrs. These recommendations were directly at three areas; blowdown of the RCS, operation of the core flood tanks, and HPI usage. At the time of the blowdown recommendations, the licensee had already initiated blowdown in an attempt to cause Core Flood Tank injection and ultimately to go on the decay heat removal system. There is no evidence that the suggestions pertaining to the core flood tanks were followed. Licensee management did finally decide to increase HPI and repressurize the RCS. It is doubtful the NRC recommendation affected their decision.



4. SIGNIFICANT EVENTS DURING ACCIDENT

4.1 Turbine Trip

4.1.1 Apparent Cause

The cause of the initial trip of the steam generator feed pumps (FW-P-1A/1B) that initiated the turbine trip followed by a reactor trip has been attributed by some members of the plant staff to water in the instrument air system. It has been stated that water in the instrument air system at the condensate polishing system will cause the condensate polisher air-operated effluent valves M12, M22, etc., to go closed (Int. 102, 123). While a turbine trip is an anticipated operational event that occurs relatively frequently during the operation of a plant, the investigation studied the possible causes of the turbine trip to determine if a precise initiating cause could be determined, thereby eliminating the need for conjecture.

4.1.2 Plant Air Systems

A partial review of the history of the instrument air service air system was conducted during this investigation.

A licensee concern as to the capacity of the air system was recognized early in the construction/preoperational phase of TMI-2. (Problem Report No.

913; Ref. 18) The solution of the capacity problem was cross-connecting the station service air system to the instrument air system as a normal mode of operating the two systems. Discussions with licensee personnel indicate that there is a change pending that would isolate part of the station service air system. This change and its status were not pursued for details. The mode of air supply operation on March 28, 1979, was the cross connected system.

The investigator reviewed Instrument and Control Log Book I, (designated as "Mercury" Daily Log), which indicates that there was water in the air system at the condensate polishers on October 19, 1977. While the log called for attempts to determine the cause, discussions with licensee personnel knowledgeable of this occurrence indicate that the cause was not determined. The air lines were cleared of water, and the system functioned correctly.

The investigator noted that reference is made in the I&C Shop Log, May 12, 1978, that condensate polisher air lines were again filled with water, at a time when operations staff members were working on the condensate polishing system.

The licensee installed local air dryers at various points in the instrument air system to prevent the accumulation of moisture. In particular, an air/water separator was installed in the condensate polisher instrument air line in series with two pressure regulators. This arrangement processed all air to the condensate valve controls and instruments located on the condensate polisher local control panel. This water/air separator utilizes a buoyant valve plug that rises and allows water to exit the line while keeping the air above the water. It also contains a fine mesh screen. Air lines at the control panel are 1/4 and 3/8 inch diameter copper tubing. Observation of this separator in its current status shows an accumulation of iron-rust-colored particles at the screen location of the air/water separator.

4.1.3 Post Accident Testing

The licensee has performed tests on the condensate polisher instrument air system subsequent to March 28, 1979. Discussions with the licensee staff

working on these tests indicate that, on isolating instrument air from the condensate system, the condensate outlet valves for each polisher tank go closed.

Their tests indicated that introduction of water into the air system did not affect the polisher outlet valves, in that the air-water separator functioned properly. They proceeded to change the test procedure and they removed the air/water separator. In that test, on introduction of water, the outlet valves fluttered and then stabilized open. In all testing cases involving water, the polisher inlet valves remained open. The testing with water consisted of allowing about 15-gallons of water to enter, with the air, at the condensate polisher control cabinet.

The licensee, during testing, inspected the service station air supply connected to the condensate system. He found that the check valve (Tag No. R0-I 51) in the 1 inch station service air line, approximately 115 feet from condensate resin tank No. 7, was "frozen" open. The shutoff valve in that line is open when resin transfer is in progress. The open shutoff valve and open check valve allowed a path for water to enter the station service air-instrument air system on March 28, 1979 (Procedure 2106-2.2, Revision 9; Ref.).

The licensee report on their testing and investigation effort has not been completed.

The actual circumstances described at the time of the trip (Int. 102, 143) indicate that significant amounts of water were found in the station service and instrument air receivers. The No. 7 condensate polisher was being "flushed" with water having an exit pressure of 160 psig into the condensate polisher No. 7 tank. The 160 psig water and the service air, at 80 to 100 psig, were being used simultaneously. Discussion with the auxiliary operator who was working on the condensate polisher tank No. 7 on March 28, 1979, revealed that the air was being used to "fluff" the resin while introducing high-pressure demineralized water to transfer the resin.

Discussion and taped interview (Int. 123) indicate that at the announcement of the turbine trip/reactor trip, the condensate polisher panel indicators showed condensate polisher isolation, which indicates no condensate flow. Condensate flow charts have been reviewed by the investigator and confirm an abrupt termination of flow (Flow Charts for Condensate Polisher Tanks 1, 2, 3, 4, 5, and 8; as well as Polisher Outflow Chart; Ref. 19).

4.1.3 Pump Interlocks and Wiring Errors

Discussion with a licensee engineer indicates that he also found that the solenoid switch wiring for the polisher valve controls was not in accordance with drawings in at least two polisher units. This could affect the status of the valves on power failure. The details on this aspect are not available at this time, but should be available in his pending report. He also stated (Int. 166) that there was a wiring error related to the condensate/condensate booster pump auto/manual switch such that on a trip of condensate booster pump CO-P-2A, its paired condensate pump CO-P-1A would trip. This wiring error was isolated to the A pump pair so that the condensate B pump would remain on line when its paired booster pump tripped.

Once the condensate polishing system fails to allow condensate water to flow, the condensate booster pumps will trip on low suction pressure and condensate pump 1A will trip because of the wiring error. Once the booster pump trips, the main feedwater pumps will trip on low feedwater pump suction pressure. The loss of both feedwater pumps trips the turbine.

4.1.4 Evaluation

The tests performed to date do not demonstrate conclusively that condensate polisher valve closure will not occur with water in the condensate polisher valve instrument air lines as reported by the interviewed operators.

To validate the initiating cause of the turbine/reactor trip as the presence of water in the air system producing condensate polisher flow isolation,

a confirmatory test would be needed. The test performed by the licensee casts some doubt on the apparent initiating cause, but the investigators consider the testing performed thus far to be insufficient to be conclusive.

This investigation was not able to identify the specific initiating cause which led to the turbine trip on March 28, 1979. The initiating cause is of interest for reasons stated in Section 4.1.1. However, failing to identify a specific cause is not a concern in that this plant was designed to sustain a turbine trip irritated by a loss of main feedwater as a routine operational event. The availability of the condensate system is not critical to the sequence of events in this accident.

4.2 Closed Emergency Feedwater Block Valves

4.2.1 Sequence

At approximately 8 minutes after the start of the accident, the operator found the OTSG level at 10 inches on the startup range. The operator considered this level to indicate that the OTSG was "dry." (This level indication is appropriate for a steam filled OTSG with no liquid phase present.) The operator verified that the emergency feedwater (EFW) pumps were running and examined the valve lineup, and found that the EF-V12A and 12B valves were shut. The position-indicating lights on 12A were obscured by a caution tag hanging from another valve controller. The position indicating lights for 12B may have been obscured by operator as he leaned over the panel. (This was due to the light positions on the panel and the operator's location, not his physical stature.) The operator drove the valves open, resulting in the dry OTSG being fed with relatively cool water. (Int. 4, 9)

It should be noted that on March 23, 1979, an NRR Operator Licensing Branch (OLB) examiner conducted operator examinations which included the EFW System. The valves were open on that day based on control room indications. Moreover, a routine scheduled surveillance test was performed on the A and B

electric-motor-driven emergency feedwater pumps on March 26, 1979 at approximately 1000 hrs, by the training shift control room operator and auxiliary operator. Implementation of this surveillance test procedure results in closure of both the 12A and 12B valves regardless of which pump is being tested. The procedure calls for reopening of the valves along with ensuring the proper status of at least three other valves. The procedure for the electric-motor-driven emergency feedwater pumps is not sufficiently specific to provide documentation of valve opening. Instead, the procedural requirement is in sentence form with one signoff signifying proper positioning of the valves. The documentation which is retained contains the completed data sheets, but does not include the check lists which document the appropriate steps to return the valves to their proper position.

While the surveillance procedure is deficient as to the manner in which the test is done, and the documentation which is retained by the licensee, the control room operator who was on duty in the control room when the March 26, 1979 surveillance was performed stated during interviews that he left the 12A and 12B valves open at the end of the test (Int. 184) For additional details see Section 1.3.2 of this report.

4.2.2 Cause

The investigation attempted to ascertain the circumstances under which these valves were closed. The following possible reasons were considered:

- o The valves were left closed after the last surveillance test of the emergency feedwater system.
- o The valves were closed by the overt act of an individual.
- o The valves were left closed after maintenance work on the system.
- o The valves malfunctioned as a result of an improper design change or plant modification.

- o The valves malfunctioned because they were exposed to elevated temperatures prior to or during the accident.
- o The valves were closed as an operator action prior to or during the transient.

Review of all these possible causes revealed no reason to believe that any of them was the specific cause of the closed valves. The findings are summarized below.

The operators and supervisors responsible for conducting the surveillance test on March 26, 1979 were interviewed. (Int. 107, 136) The operator who actually manipulated the valves involved stated that he specifically recalled opening that valve. The investigation found no basis for rejecting his assertion. If his assertion was incorrect and the valves were left closed after the test, the investigation found no information to explain how the closed valves would have gone unnoticed during the 42 hours between the test and the accident. However, routine panel inspections are not required of the staff by this licensee. The matter of inspections of safety related matters is addressed in Section I-1.3.5 of this report.

The possibility that the valves were closed by an overt act of an individual was addressed in Section 1.4.6. No information was developed during this investigation indicating that sabotage was a contributing factor to the initiation of the accident or to the subsequent response of plant personnel or equipment to the accident.

The possibility of maintenance work being done on the valves was addressed in Section 1.6.2. No evidence was found of such maintenance after both record reviews and interviews.

The possibility that the valves were closed as the result of an improper response of the valve control circuits to the turbine trip was addressed. A change was made in the logic circuitry related to the operation of the emergency feedwater valves (FCR 2329.1; Ref. 9). The change included defeating the

automatic closure of the emergency valves EF-V12A/B with a low OSTG pressure signal. This feature had been part of the protective circuitry involved with the plant response to a steam break accident. If the modification required by FCR 2329.1 had not been properly performed there would be a possibility of the valves closing. Since the accident, the licensee has written and performed a test to determine if the closure demand feature had been removed from the EP-12VA/B valves (SOP No. R-2-79-31, June 29, 1979; Ref. 12). The results indicate that the valves did not close when the feedwater latching logic was introduced, indicating that the changes affecting EF-V12 A/B appear to be correct. Moreover, the pressure in the OTSGs during the first 8 minutes did not reach the initiation point for this control system, even if the change had not been properly completed.

The possibility that the valves were closed, as a result of temperature problems as might occur from system backflow was addressed. Information was obtained that suggests at least one of the valves might have undergone a thermal transient. This was based on observed discoloration of the valve and piping. The visual inspection by an investigator confirmed that a plastic instruction tag on valve EF-V11B, the EFW control valve, was "melted". The investigation included a review of possible reverse flow paths to the B OTSG, a check of maintenance requests, and interviews with mechanical and electrical maintenance personnel and operations personnel. Burns and Roe Drawing No. 2005, Flow Diagram Feedwater and Condensate, shows the possible flow paths from B OTSG. A backflow from inside containment would have to travel through reactor building penetration R-616B, check valve EF-V13B to EF-V12B which would be the most direct path to the EF-V11B valve. An alternative path could involve the same penetration, EF-V13B, EF-V12B, EF-V32B, and end at the backside of EF-V11B on to EF-V12B. A third path could include the penetration R-616B, EF-V13B, EF-V32B, and back up to EF-V11B and/or through EF-V32B. The discoloration of the pipe appears to indicate heating along the pipe from penetration R-616B to the check valve EF-V13B through EF-V12B to EF-V11B, the most direct route.

The possibility that oil staining might indicate an overheating of these valves was addressed. The EF-V12B valve appears to have oil leakage from the

limitorque operator motor which stained the valve body and piping. No evidence of a Work Request for the EF-V12 valve just prior to March 28, 1979 was found. Operations auxiliary operators who performed the EFW surveillance test that required them to be in the vicinity of the EF-V12B valve were interviewed regarding the valve oil leak. Five stated they did not recall seeing an oil stain, while the sixth did not recall looking at that valve. They did performed the surveillance over a period from January 3, 1979, to March 3, 1979. Additional information presented to the investigator indicated that the valve EF-V12B did not have such an oil stain on March 26, 1979. The investigator did note that the instruction tag on EF-V11B was deformed and showed signs of being burnt (brown) on the rear side where it is in contact with the valve housing.

The condition of the EF-V12A valve and piping was inspected and found sound with no similar condition. On March 28, 1979 both EF-V12A and EF-V12B were in a closed status.

There was no evidence to cause the investigator to conclude that either 12 valves would be closed because of the condition of EF-V12B or the condition of the B emergency feedwater piping. All information indicates that both valves opened, when actuated by the control room operator, on March 28, 1979, at about 0408 hrs. This review did not conclude how the emergency feedwater pipe became discolored, how the oil leaked, nor how the tag deformed. The purpose of this study was to determine if the condition could have been a reason for the EF-V12B valve to be in a closed position at 0400 hrs on March 28, 1979. The findings do not indicate a relationship. The possibility of a correlation to the status of the B OSTG emergency feedwater piping after its isolation during the accident was not pursued.

The possibility that the valves were closed as an operator action during the transient was addressed. The operating staff on duty during the period when the valves were found closed were interviewed to determine whether these valves could have been closed as an operator action to prevent an excessive cooldown rate of the RCS and an attendant pressurizer level drop. The investigators pursued the possibility that the action was initially taken and then

forgotten by the operator for 8 minutes (Int. 189). No information was obtained during this interview that would indicate that this operator action took place during the accident.

The investigation has not been able to establish the specific cause of the EF-V12A and EF-V12B valves being closed during the initial part of the transient. The investigators concluded that the design or mechanical malfunction possibilities reviewed are not probable causes of the valves being closed.

4.2.3 Evaluation

The delay in automatic initiation of emergency feedwater for 8 minutes contributed to an early recovery towards normal values of certain RCS parameters upon which the operators concentrate. This recovery of key turbine trip/reactor trip RCS parameters misled the operators into believing that their actions had been successful in limiting the severity of the transient. This erroneous belief led them to initiate the routine subsequent operator actions that were normal for the assumed transient. These actions occupied the operators' attention and detracted from their opportunity to establish a correct analysis of the plant conditions.

This investigation did not conclude what the ultimate course of events of the accident would have been, had emergency feedwater been introduced to the OTSGs as designed.

Based on interviews conducted with the operating staff, it is concluded that once the accident was underway and after the valves were open, the knowledge that the valves were previously shut had no substantive impact on the actions of the operators during the remainder of the accident.

4.3 HPI Flow Adjustment

4.3.1 Operator Actions Regarding Adjustment of HPI

The transient which followed a routine Turbine/Reactor Trip at the TMI-2 facility, was described as "severe" by the operations personnel. Normal plant response to the loss of feedwater transient should result in the following:

- o T_{av} decrease of 33°F from 582°F to 549°F based on a turbine bypass valve (MS-V25A, V26A, V25B, V26B) setpoint of 1010 psig.
- o Rapid pressurizer level decrease from 220 inches to 55 inches (25 inches below the pressurizer heater cutoff point of 80 inches), based on a pressurizer level change of 5 inches/°F change in T_{av} (33°F T_{av} change for a 165-inch change in level).
- o Reactor pressure decrease from 2200 psig to near the ES setpoint pressure of 1640 psig, during the RCS cooldown. (ES initiation had occurred at TMI-2 during this type of transient.)

The routine pressure transient resulted in a philosophy of operation that was formulated to prevent the initiation of ES after each reactor trip. This philosophy specifically developed following the admission of sodium hydroxide into the RCS twice following a unit trip with an ES initiation. A further consideration was the time and effort that was subsequently required to clean-up and deborate the RCS after addition of substantial amounts of sodium hydroxide and borated water (2300 ppm) into the RCS following ES initiations. (LER 78-021-03L, April 29, 1979; 78-033-01T, April 23, 1978; Ref. 134)

- o The Reactor Trip Procedure (2202-1.1) contained steps that required starting the second or third makeup pump, based on pressurizer level decrease to 100 inches or 20 inches, respectively, supplying suction for the makeup pumps from the BWST, and injection of borated water to the RCS via the HPI valves (MU-V-16A, 16B, 16C, and 16D). (Ref. 76) This operating philosophy

required that the makeup pumps and HPI flow be stopped or throttled, as soon as possible, on indication of the recovery of pressurizer level or RCS pressure toward normal conditions of 100 inches and 2155 psig, respectively, in order to limit the addition of borated water (2300 ppm) to the RCS.

o The requirements for returning the makeup system (HPI system if ES has initiated) to normal required the immediate throttling of the HPI valves and stopping of one or two makeup pumps. If an ES had been initiated, the operator quickly blocked and bypassed the ES signal to allow manual control of the HPI valves (MU-V16 A, 16B, 16C, or 16D as appropriate) and the control of the makeup pumps.

Evaluation

It must be noted that the reactor trip procedure does not recognize the initiation of ES following an analyzed plant transient. Furthermore, the automatic initiation of the ES was recognized by procedure only in the loss of reactor coolant/reactor coolant system pressure and steam line break accidents. The procedure stated that the automatic initiation of ES was indicative of a large break, one beyond the capability of the makeup system. For further details see Section 2.6 of this report.

Based on the operating philosophy developed to cope with the associated severe RCS parameter transients, the operators have bypassed the ES signal (within seconds), when the RCS pressure was continuing to decrease and had not recovered above 1640 psig as indicated in the loss of RC/RCS pressure procedure. Additionally, the operators have limited (throttled) the injection of borated water into the RCS, based on the recovery of the pressurizer level.

Additionally, the operators were very sensitive to taking the pressurizer "water solid." This sensitivity was developed through the formal training program.

Operating Procedure 2104-1.2, Makeup and Purification, Step 2.2.1, states that a MUP should not be run with the RCS (Pressurizer) in a water solid

condition. Operating Procedure 2103-1.3, Pressure Operation, Step 2.1.8, states that the pressurizer will not be taken to solid conditions except as required for system hydrostatic tests.

Technical Specification 3.4.4 requires that the pressurizer level be maintained between 85 and 380 inches in Modes 1, 2, and 3.

Interviews with the operating and training personnel (see Section 2.15) indicated that the operators would respond specifically to the pressurizer level in almost any circumstances, because they were trained to believe that water in the pressurizer indicated water over the reactor core. (Int. 5, 17, 38, 2, 14, 37, 3, 15)

On the morning of March 28, 1979, the operators responded as previously during plant trips; after starting the second makeup pump and receiving an ES actuation, they noted the pressurizer level increasing from 155 inches and quickly bypassed the ES signal in order to throttle the HPI valves (MU-V16A, 16B, 16C, and 16D) and stop the additional makeup pump. The HPI flow was essentially stopped within 4.5 minutes following the reactor trip in response to the increasing pressurizer level to the near-full-scale condition, but not to the continuing low RCS pressure (below the ES setpoint of 1640 psig).

As noted in this report (Section 2.5), the operators had throttled the makeup pump flow/HPI flow to a minimum within 4.5 minutes after the reactor trip and were down to the one pump (MU-P1A). Additionally the letdown system was also utilized in an attempt to recover and maintain normal pressurizer level between 0403 hrs and 0407 hrs.

The interviews revealed that the makeup pump flow and letdown flow were operated in a normal manner between 0410 hrs and 0533 hrs when the HPI flow was increased manually to approximately 300 gpm. Moreover, records review revealed that the borated water delivered from the BWST to the RCS between 0400 hrs and 0730 hrs was 15,000 gallons or approximately 70 gpm average with

the major portion of the injection occurring from 0402 hrs to 0405 hrs, and 0533 hrs to 0730 hrs. During the majority of this 3 1/2 hour period, the average net addition to the RCS was 25 gpm.

4.3.2 Analysis

4.3.2.1 HPI Pump History

The status of all three makeup pumps was established by the investigation for the entire time of the incident using information from the alarm printer, operator logs and operator interviews. Graphs of the status of the three pumps are shown on Figure I.4-1. These graphs show three pump states: off, on, and the high pressure coolant injection (HPI) condition. During HPI, two pumps are on and the throttling valves for the pumps automatically go to full open, at which position each pump puts at least 500 gpm into the RCS. When a pump is on, but HPI is not actuated, the flow can be throttled back by the operators. According to operator statements (Int. 15) the minimum flow that can be adequately controlled is about 100 gpm per pump. Table I.4-1 lists the specific on and off times for each makeup pump.

At least one makeup pump was running during the entire time of the incident, with the exception of a 5-minute period starting at 0817 hrs. Until 0800 hrs, MUP-1A was the pump selected. However, after two unsuccessful attempts to start this pump at 0818 hrs, it was taken out of service. MUP-1B was then selected and started at 0822 hrs, after which it continued to run during the remainder of the incident. MUP-1C was used from time to time throughout the accident to supplement the selected A or B pump.

MAKE-UP AND DECAY HEAT PUMP STATUS

I-4-15

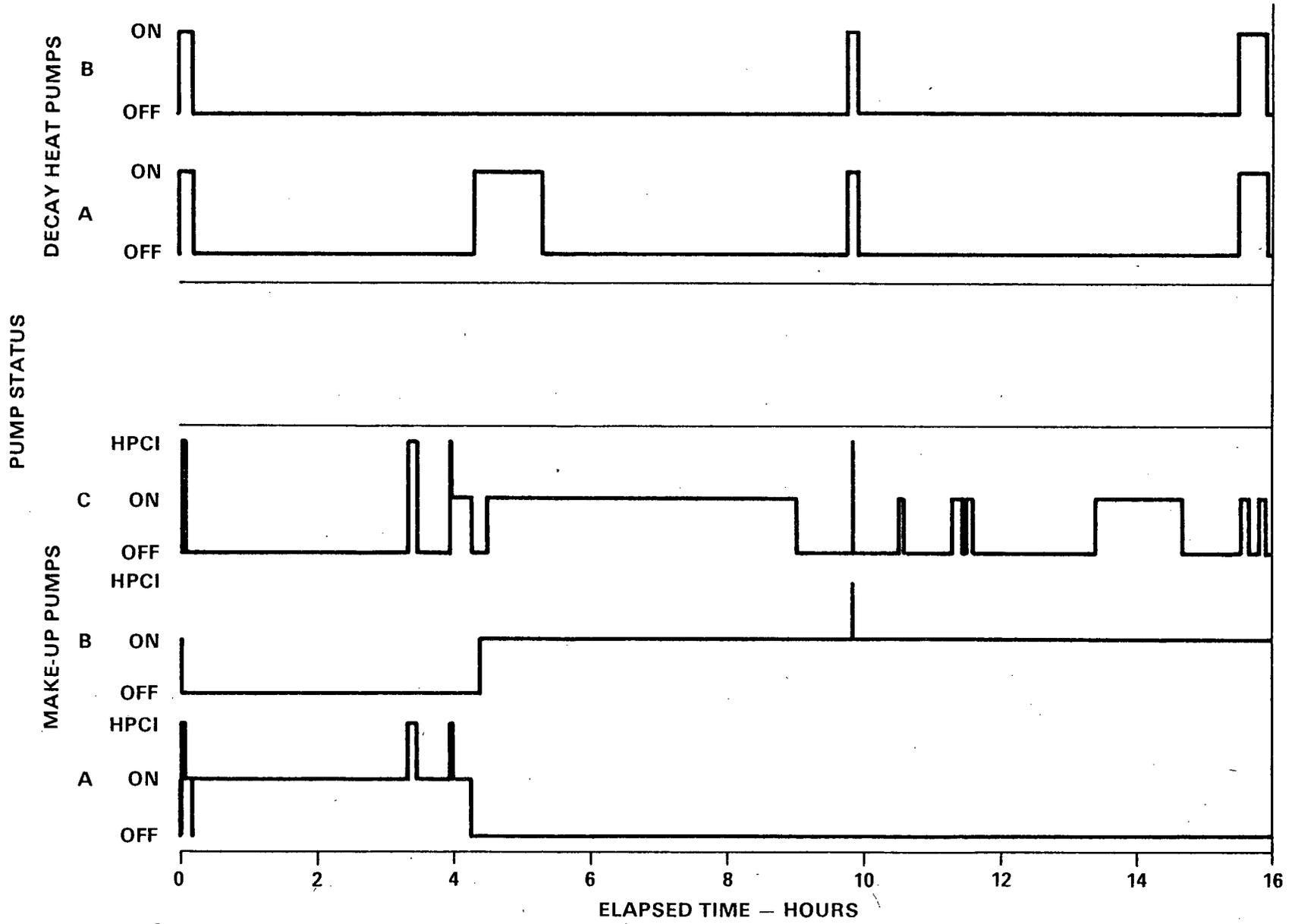


FIG I.4-1

TABLE I.4-1
HPI PUMP HISTORY

<u>Elapsed Time</u>	<u>MUP A</u>	<u>Pump Status MUP B</u>	<u>MUP C</u>	<u>Source of Information</u>	<u>Comment</u>
0	Off	On	Off	Interviews (<u>GPU 1, 20;</u> <u>Int. 15</u>)	MUP B acting as normal makeup pump
41 seconds	On	-	-	Alarm Printer Interviews (<u>GPU 1, 20;</u> <u>Int. 15</u>)	MUP-1A started by operator; MU-V16B opened to increase flow.
2 minutes 2 seconds	-	Off	On	Alarm Printer Interviews (<u>GPU, 20</u>)	ES initiation; control valves full open.
4 minutes, 38 seconds	-	-	Off	Alarm Printer Interviews (<u>Int. 15</u>)	MUP A throttled.
10 minutes, 24 seconds	Off	-	-	Alarm Printer	MUP 1A stopped, restarted and stopped within 4 seconds.
11 minutes, 43 seconds	On	-	-	Alarm Printer	
3:20	-	-	On	Alarm Printer	ES initiated by operator; valves full open.
3:37	-	-	Off	Alarm Printer	ES reset.
3:56	-	-	On		ES on high reactor building pressure; valves full open.
4:00	-	-	-	Alarm Printer	ES defeated.

I-4-16

TABLE I.4-1 (Continued)

HPI PUMP HISTORY

<u>Elapsed Time</u>	<u>MUP A</u>	<u>Pump Status MUP B</u>	<u>MUP C</u>	<u>Source of Information</u>	<u>Comment</u>
4:17	Off	-	Off	Alarm Printer	No pumps running.
4:18	-	-	-	Alarm Printer (Int. 15)	After two unsuccessful start attempts, MUP-1A pulled to lock; remains off for rest of incident.
				Interviews	
4:22	-	On	-	Alarm Printer	MUP-1B remains on for rest of incident.
				Interviews	
4:27	-	-	On	Alarm Printer	
9:04	-	-	Off	Alarm Printer	
9:50	-	-	On	Alarm Printer	ES on high reactor building pressure; valves full open.
9:51	-	-	Off	Alarm Printer	MUP-1C was on 1 minute.
10:32	-	-	On	Alarm Printer	
10:36	-	-	Off	Alarm Printer	MUP-1C was on 4 minutes.
11:19	-	-	On	Alarm Printer	MUP-1C was on 9 minutes.
11:28	-	-	Off	Alarm Printer	
11:33	-	-	On	Alarm Printer	

I-4-17

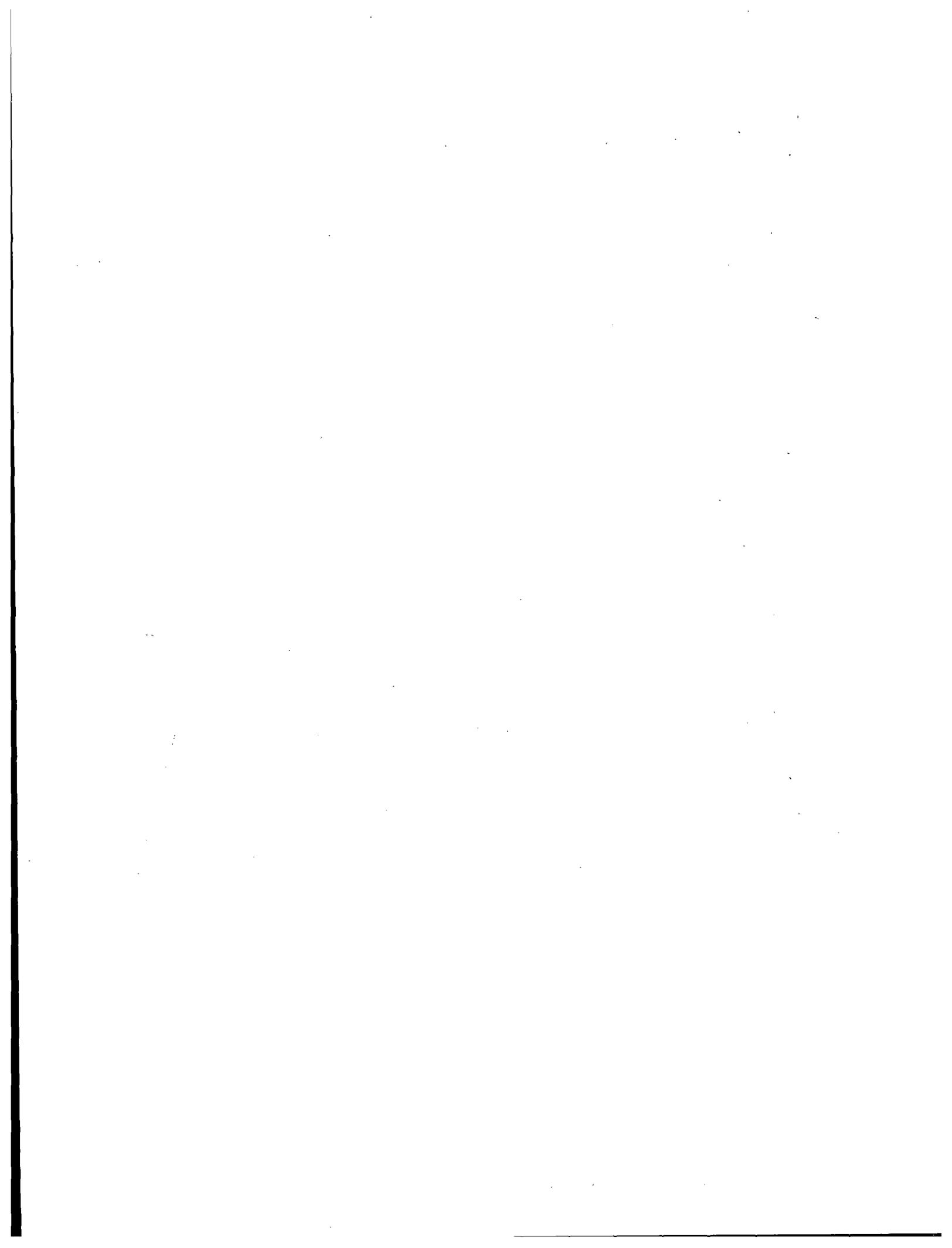


TABLE I.4-1 (Continued)

HPI PUMP HISTORY

<u>Elapsed Time</u>	<u>MUP A</u>	<u>Pump Status MUP B</u>	<u>MUP C</u>	<u>Source of Information</u>	<u>Comment</u>
11:36	-	-	Off	Alarm Printer	MUP-1C was on 3 minutes.
13:23	-	-	On	Alarm Printer	
14:41	-	-	-	Operating Logs	MU-V16C throttled to 105 or 120 gpm.
14:43	-	-	Off	Alarm Printer	
15:33	-	-	On	Operating Logs	MUP-1C started by operation; ES initiation at this time was immediately bypassed.
15:39	-	-	Off	Alarm Printer	MUP C was on 6 minutes.
15:49	-	-	On	Alarm Printer	
15:56	-	-	Off	Alarm Printer	MUP-1C was on 7 minutes; remained off for rest of incident; MUP-1B still running.

I-4-17a

4.3.2.2 Effective Flow Rates to the RCS

Instantaneous makeup flows are recorded hourly by the Computer Log Typer. These are listed in Table I.4-2 for the period of the incident. They varied from 8 to 128 gpm, but averaged about 80 gpm during the incident. These are the flows through the 17 valves used for normal makeup. They do not include the RC pump seal water injection, which totals about 32 gpm, or the HPI flow through the 16 valves, which is known accurately only immediately after ES actuation, since these valves can be throttled once ES is reset.

The net makeup from the borated water storage tank (BWST), generally injected through the 16 valves, could be derived for various time periods from the known changes in the BWST levels. Table I.4-3 lists the average flow rates for various time periods and the data and data sources, used to derive them. These rates and the total quantities of BWST water expended during each period are plotted on Figure I.4-2. These rates, the makeup pump status discussed and pressurizer or core flood tank volume changes, were used to develop net addition flow rates to the primary system. These can be divided into five main time periods for convenience of discussion:

TABLE I.4-2 MAKEUP FLOW THROUGH THE MU-V17
VALVES AND RCP PUMP SEALS (GPM)

<u>Elapsed Time (Hours)</u>	<u>MU-V17 Valve Flows</u>	<u>Pump Seal Flow</u>	<u>Total</u>
0	16	32	48
1	118	32	150
2	97	32	129
3	125	32	157
4	8	32	40
5	58	32	90
6	52	32	84
7	49	32	81
8	49	32	81
9	97	32	129
10	113	32	145
11	95	32	127
12	116	32	148
13	128	32	160
14	43	32	75
15	72	32	104
16	95	32	127

TABLE I.4-3

AVERAGE MAKEUP FLOWS
BASED ON BWST LEVELS

Elapsed Time Hours:Minutes	Level (ft)	Total Used (gal)	Used in Period (gal)	Avg Flow in Period (gpm)	Source of Information
3:30	53.04	15,000	15,000	70	Alarm Printer
6:55	37	147,000	132,000	640	RI Incident Response Center
9:15	31	197,000	50,000	360	RI Incident Response Center
13:20	26.5	234,000	37,000	150	RI Incident Response Center
14:47	22	271,000	37,000	430	Operator Logs

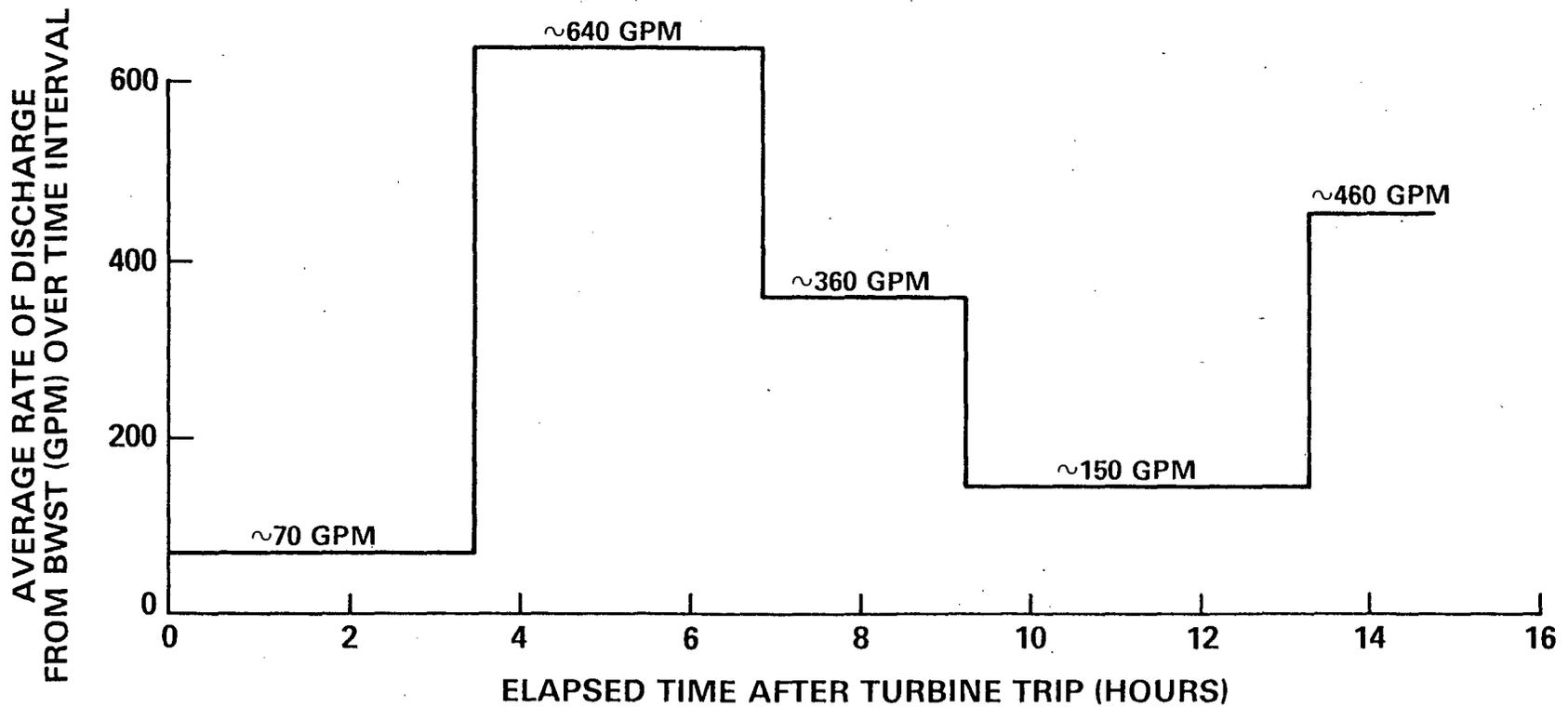
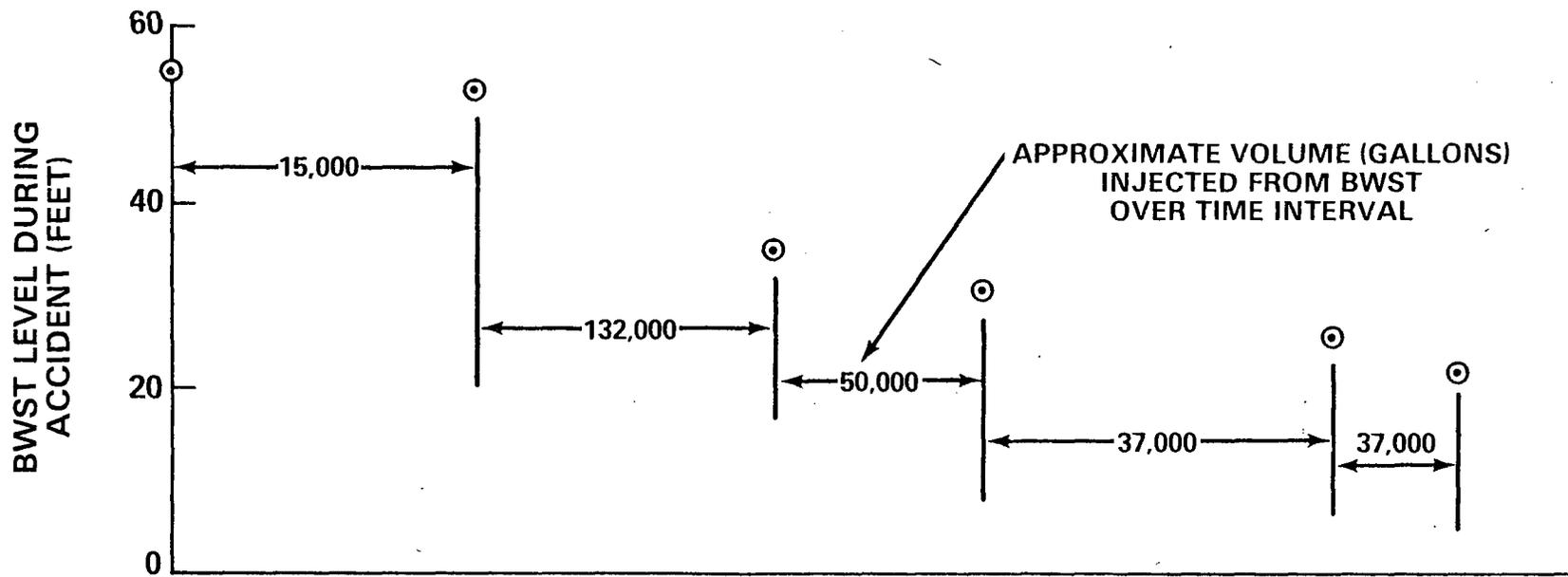


FIG 1.4-2

- . The first 3 1/2 hours of generally very low flow,
- . The following 3 1/2 hours of high flow rates,
- . A 2-hour period of moderate flow rate,
- . An interim 4-hour period of low flow rates,
- . The final repressurization.

These periods are detailed below.

Only 15,000 gallons of water were injected into the reactor from the BWST during the first 3 1/2 hours, approximately 2/3 of it during two HPI actuations. As indicated in Section 4.3.1, the overall average flow for this period was 70 gpm. However, during most of this time the flow was much less. During the first minute of the accident, the operators started MUP-1A and opened MU-V-16B. This was followed by ES initiation at 2 minutes, which was throttled back at 4 minutes (Int. 61, 15). Thus, the net flow during the first 4 minutes was about 3000 gallons. At 0720 hrs, ES was initiated and maintained for 7 minutes, resulting in a flow rate of 1,000 gpm for a total injection of 7,000 gallons. The remaining 5,000 gallons injected during the period from 0404 hrs until 0720 hrs, when ES was not actuated, results in an average net flow rate of about 25 gpm for that period.

The BWST level at 1055 hrs shows a total expenditure between 0730 hrs and 1055 hrs of 132,000 gallons. This volume and time period can be adjusted for the known or maximum flows identified below.

o From 0727 hrs to 0756 hrs, one pump (MUP-1A) was running, which limited the flow to 500 gpm. In addition, between 0730 hrs and 0740 hrs, there was a 3600-gallons increase in indicated pressurizer water volume. This occurred during a period of stable pressure and decreasing cold leg temperatures with the EMOV closed. (Plant Strip Charts and Multipoint Recorder; Ref. 13. Also see discussion of EMOV status in Section 4.8) Thus, the flow during this period is estimated to be between 360 and 500 gpm.

o ES was initiated at 0800 hrs, resulting in a 1000-gpm flow until 0804 hrs, when ESF was defeated. Both pumps continued to run until 0817 hrs, but it is not known whether or not the flows were throttled. (Alarm Printer; Ref. 14)

o There was no flow for 5 minutes from 0817 hrs until 0822 hrs, as all of the makeup pumps were off. (Alarm Printer; Ref. 14)

NOTE: MUP-1B was turned on at 0822 hrs and MUP-1C was turned on 5 minutes later. Both pumps continued to run until 1304 hrs when MUP-1C was shut off.

These adjustments result in a minimum net average flow rate during two-pump operation of 680 gpm for the period between 0730 hrs and 1055 hrs.

(4) Between 1055 hrs and 1315 hrs, 55,000 gallons of BWST water were expended for an average flow rate of about 360 gpm during this period. At the end of this period, the RCS was depressurized in order to inject the core flood tanks, and a partial injection was achieved. The available data on core flood tank levels is not sufficiently consistent to be used to establish the amount of water injected by this means. However, a core flood tank pressure measured some time after the incident (Int. 114) was reported to be about 400 psi. The normal CFT gas volume is 370 cubic foot. (Core Flood System Description, Index No. 2813; and Technical Specifications; Ref. 15) The volume injected by both tanks, assuming isothermal gas expansion from 600 psi, is about 2,670 gallons.

(5) During the 4-hour period between 1315 hrs and 1720 hrs, 37,000 gallons of BWST water were expended, for an average net flow rate of 150 gpm from the BWST. MUP-1C was used intermittently for a total of 17 minutes, including a 1-minute ES actuation at 1350 hrs.

(6) Following the decision to repressurize the RCS, the flow rates were again increased. MUP-1C was turned on at 1723 hrs (MUP-1B was already on) and

run continuously until 1841 hrs. The BWST level drops during this period showed a total of 37,000 gallons injected. Based on the period of a two-pump operation, this results in an average injection rate of about 470 gpm. No BWST levels were recorded after 1841 hrs.

4.4 Shutdown of the Reactor Coolant Pumps

During normal operation at full power (97% on 0400 hrs on March 28, 1979), all four reactor coolant pumps (RC-P1A, P-2A, P-1B, and P-2B) were operating and following the routine loss of feedwater turbine/reactor trip at 0400 hrs. The pumps continued to operate until manually tripped at 0514 hrs (P1B and P2B) and 0541 hrs (P1A and P2A) by the operators.

Subsequent to the unit turbine/reactor trip, resulting in the normal RCS pressure increase (2250 to 2435 psig), and the apparent opening of the EMOV (RC-R2) and the subsequent failure to reclose resulted in the continuing loss of mass from the RCS.

Records review and personnel interviews revealed that conditions of decreasing coolant flow, increasing reactor coolant pump vibration, and pump motor vibration increases began almost immediately following the reactor trip at 0400 hrs.

At approximately 0406 hrs, the RCS conditions were at saturation temperature and pressure conditions (585°F and 1050 psig) and within 15 minutes the RCS pressure also dropped below 1200 psig again and remained below 1200 psig until about 0605 hrs. The existence of saturation conditions and the RCS pressure below 1200 psig requires, by procedures, the tripping of the reactor coolant pumps.

The RCS flow began to decrease steadily following the reactor trip at 0400 hrs and had decreased to approximately 65 million pounds per hour by 0420 hrs as the RCS pressure decreased to less than 1100 psig.

The normal operating procedure (2103-1.4, Revision A, April 18, 1978, Reactor Coolant Pump Operation, Step 2.1.1.1) specifically addresses RCS system pressure requirements that must be maintained for required RCP operation. (Ref. 91) The heatup/cool-down curve addresses the RCS pressure-temperature limits for reactor coolant pump operations.

NOTE: The RCS pressure-temperature limits for fuel pin compression considerations are more conservative than the limits for RCP operation.

The minimum pressure-temperature limits for continued RCP operation were not met.

The Emergency Operating Procedure (2202-1.3, Revision E, May 12, 1978, Loss of RC/RCS Pressure, Step A.3.2.7), requires the operator to trip the RCPs before RCS pressure decreases below pump NPSH conditions (as indicated by the heatup/cool-down curve). (Ref. 87) Step B.2.2.4 of that procedure requires that the operator trip the reactor coolant pumps before reaching 1200 psig decreasing.

Unit 2 Abnormal Procedure 2203-1.4, RCP and Motor Emergencies, Revision 3, addresses pump and motor vibration. This procedure categorizes the Allis-Chalmers motor installed vibraswitch as an alarm and as a symptom related to a dropped impeller. The alarm was recorded at 0654 hrs on the Computer Utility Typer. This procedure also addresses shaft vibration sensed by the vibration sensors (IRD). The procedure allows for the operation of one pump per loop for the first four hours at a vibration level greater than 30 mils and greater than 26 mils for one or two pumps per loop with no required automatic action. The procedure also requires the resetting of the IRD vibration alarm and, if vibration alarms immediately reoccur, reactor power must be reduced and the affected RCP tripped per normal procedure. This procedure addresses pump vibration as a symptom of seal failure. A vibration level is not specified and no automatic action occurs. The indication for action is based on seal cavity pressure exceeding 2500 psig. The action criteria related to vibration includes a bleed off flow and seal leak off flow exceeding 3 gpm which was not evident from records for March 28, 1979.

A second procedure, Operating Procedure 2101-1.1, Nuclear Plant Limits and Precautions, Revision 4, addresses shaft vibration. The procedure requires that B&W engineering be notified when RCP vibration at the pump coupling (shaft vibration) reaches 15 mils peak to peak amplitude. The procedure allows a shaft vibration of 20 mils for a period not to exceed four hours.

Operating Procedure 2103-1.4 requires the contacting of the pump manufacturer if the RCP vibration at the pump coupling reaches 15 mils peak amplitude. This procedure allows shaft vibration of 20 mils for a period of 4 hours on initial start. The procedure requires a RCP trip if the motor stand vibration exceeds 3 mils. The motor stand vibration is noted on control room panel 10 as Motor Brg. "x" and "y". This procedure requires the tripping of RCP if shaft vibration exceeds 20 mils for the first four hours and must be tripped if shaft vibration exceeds 30 mils under any condition.

The pump vibration IRD values are read and recorded by the control room operator as a routine parameter. The vibration indicated values are not recorded through any automatic printout.

From the results of reviewing Abnormal Procedure 2203-1.4, Operating Procedure 2101-1.1 and Operating Procedure 2103-1.4 it is evident that the procedures have disagreements as to vibration limits and actions to be taken. This is mentioned as an item to be resolved and is left for appropriate licensee action.

Evaluation

Based on the records review of the RCS/temperature relationships and discussions with licensee personnel, the low pressure RCP trip criteria was exceeded early in the event. Also, the discussion indicated that the pump vibration indicators had increased to their full scale indicating condition. Section 2.5 of Details I describes an item being considered as a possible item of noncompliance related to the failure to follow procedure and trip the RCPs.

4.5 Attempted and Successful Restarts of the Reactor Coolant Pumps

Following the stopping of the RCPs at 0514 hrs and 0541 hrs, there were doubts that adequate natural circulation was occurring. It was decided that forced cooling must be provided to the reactor. Reactor coolant pump operation occurred twice in the morning (0654 hrs and 0808 hrs) and twice in the evening (1933 hrs and 1950 hrs).

The first restart of a reactor coolant pump was attempted following the closing of the RCS EMOV (RC-R2) at 0612 hrs. The RCS pressure was increasing from a minimum of 660 psig as a result of HPI flow (increased at approximately 0533 hrs) into the RCS. The attempt to restart the RCP commenced at about 0630 hrs and included jumpering the starting interlocks for the pumps, to allow pump start. The specific conditions which prevented the RCP start capability was not determined.

After attempting to start sequentially RCP-1A, RCP-2A, and RCP-1B, RCP-2B finally started at 0654 hrs and was operated until 0713 hrs in an attempt to supply cooling to the reactor core.

The reactor coolant pump (RCP-2B) starting current returned to about 100 amps after approximately 6 seconds, and the reactor coolant flow in the B loop increased (spiked) to approximately 10 million pounds per hour and decreased quickly to minimum. It was concluded that the pump was not operating properly, was vapor bound and RCP-2B pump was stopped after 19 minutes at 0713 hrs.

The second attempt to run a reactor coolant pump occurred at 0808 hrs, and the pump was left on for only 37 seconds, after the pump conditions indicated unsatisfactory pump performance again, including no flow and no pump motor current above about 100 amps. It was concluded that the pump was vapor bound (Int. 121, 192).

The third attempt to run a reactor coolant pump occurred at 1933 hrs. The system pressure had been increased via the HPI pumps to about 2300 psig at

1930 hrs. The decision to repressurize was apparently made at approximately 1730 hrs. The decision was made to bump the RCP for 10 seconds to sweep any steam/gas from the pump and ensure that the pump would operate normally. The RCP-1A pump was started for 10 seconds and indicated normal starting current and the loop flow increased to about 30 million pounds per hour.

RCP-1A was restarted at 1950 hrs, following the normal waiting period after starts, and the RCS was returned to stable conditions at approximately 1000 psig and 280°F with the one pump operating.

4.6 Closing of the Core Flood Tank Isolation Valves

During the first two hours of the accident, the RCS pressure decreased from the initial value of 2200 psig (normal operating condition) to approximately 660 psig prior to closing the EMOV (RC-R2) at 0618 hrs.

The interviews of operations personnel revealed that the core flood tank valves (CF-V1A and V1B) were closed by operator actions at approximately 0600 hrs to prevent the addition of additional borated water to the RCS when the RCS was believed to be in the solid water condition, as indicated by the nearly full pressurizer (level indicated at or near 400 inches). (Int. 95, 196, 109)

The specific requirements and steps required to close the valves included unlocking and closing the valve motor operator electrical breakers at the motor control centers (2-11EB: CF-V1A and 2-21EB: CF-V1B) and closing the valves remotely (electrically from the control room). (Int. 95, 196, 109).

The interviews did not determine the specific time the core flood tank valves (CF-V1A and V1B) were reopened, but it was apparent that the valves were opened prior to depressurization of the RCS at approximately 1300 hrs resulting in the discharge of some amount of core flood tank (CF-T1A and T1B) water into the reactor vessel. The discharge of water into the RCS from the CFTs was confirmed through interviews and computer alarms. (Int. 57, 111, 184, 119)

Operating Procedure 3102-1.1, Unit Heatup, Steps 4.38, Mode 4 to Mode 3 checklist (Appendix C) and Step 4.41, places the core flood system into operation. This is done by Procedure 2104-1.1, Core Flood, at RCS pressure of 750 psig and puts the core flood tanks into an OPERABLE status. Operating Procedure 2102-3.2, Unit Cooldown, Steps 4.16 and 4.17, remove the core flood tanks from the OPERABLE status condition between 750 psig and 700 psig, decreasing.

Evaluation

The Operating Procedures referenced above, 3102-1.1 and 2102-3.2 address the isolation (and unisolation) of the core flood tanks during normal plant evolutions of heat-up and cooldown. These procedural requirements indicate that the core flood tank isolation valves would be maintained in the open positions in Modes 1, 2, and 3.

Technical Specification 3.5.1, action statement b, indicates that with any core flood tank inoperable due to the isolation valve being closed, the plant shall be in HOT SHUTDOWN ($280^{\circ}\text{F} > T_{\text{avg}} > 200^{\circ}\text{F}$) within the next 12 hours. The normal cooldown procedures would appear to contain this required condition even though the procedure would allow the plant to be maintained at 800 psig or less and higher than 280°F for an indefinite period of time.

Emergency Procedure 2202-1.3, Loss of Reactor Coolant/Reactor Coolant Pressure, Section B, Step 3.1, requires that the verification that all ESF equipment is in its ESF position, by observing that all equipment status lights indicate as shown in a table in the procedure; including verification that the core flood valves (CF-VIA and VIB) are open following the automatic initiation of ESAS.

Technical Specification 6.8.1.a requires that written procedures shall be implemented covering the activities referenced as the applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972, which cover procedures for combating emergencies such as "LOSS OF COOLANT".

The closing of the core flood tank valves (CF-VIA and VIB), during the accident on March 28, 1979, contrary to the Emergency Procedure 2202-1.3, Section B, Step 3.1, Table B-1, is under consideration as a potential item of noncompliance pursuant to Technical Specification 6.8.1.a.

4.7 Use of Core Flood Tanks

At approximately 7 hours and 30 minutes after the start of the accident, the decision was made to depressurize the RCS so that the core flood tanks would be available to discharge their contents if the vessel required large quantities of water. Once having done so, further steps would then be taken to enable the staff to put the decay heat removal system into service. (Further detail is available in Section 3.2)

The RCS depressurization took approximately one hour, the RCS pressure reaching 600 psig at 1241 hrs.

Starting at about 1212 hrs and lasting for the next several hours, multiple level alarms were received on the A core flood tank. No such alarms were received on the B core flood tank. The investigation has concluded that these alarms are probably erroneous since the nitrogen pressure on both tanks was reported to have decreased to approximately 400 psig. This would imply a level decrease well below the alarms that were indicated.

This condition of the core flood tanks "floating" on the RCS remained in effect until approximately 13-1/2 hours when the Station Manager was directed to take the plant solid. The plant staff has stated that, at that time, they believed the behavior of the core flood tanks demonstrated that the core was covered with water. The presence of the loop seals in the piping from the core flood tank to the reactor vessel precludes the tank level change behavior from assuring that the core was covered at that time. This is discussed further in Section 4.8 of this report.

Evaluation

It is not possible for the investigation team to assess the direct impact of the use of the core flood tanks, as they were used during this accident, on the outcome of the accident. To do so would require treatment of the following areas:

- o Analysis of the vessel water levels during the low-pressure period from approximately 8 hours to 14 hours after the start of the accident and subsequent analysis of the core thermal behavior under those conditions.

- o Analysis of the extent of venting of noncondensable gases from the RCS during the low-pressure period and the subsequent impact that venting had on the ability of the staff to apparently refill the A RCS loop and successfully operate reactor coolant pump PIA.

The analysis of these matters is beyond the scope of this investigation. Consequently, no conclusions have been drawn in this area.

4.8 Reactor Vessel Level Indication

Plant operators attempted to use pressurizer level indication, early in the accident sequence, and core flood tank level indication, approximately half way through the accident sequence, to infer reactor vessel water level conditions. The operators knew the pressurizer and both core flood tanks were positioned well above the core. They assumed the core had to be covered with water; if a water level existed in the pressurizer, regardless of system pressure, or in both core flood tanks, at a system pressure below the nominal 600 psig cover gas pressure. The operators failed to recognize that the existence of saturated, even superheated conditions within the RCS, coupled with the actual piping configuration from these vessels to the RCS, allowed loop seals to form. The loop seals for the pressurizer, core flood tank 1A

and core flood tank 1B are about 11 feet 5 inches, 14 feet 4 inches, and 17 feet 10 inches; respectively. Once loop seals formed, RCS pressure needed only to exceed the pressurizer vapor space pressure by about 13 psi and the core flood tank cover gas pressure by about 8 psi to force both water levels to full scale, even with the RCS completely voided.

Admittedly, the RCS should not normally experience saturated or superheated conditions and, without those conditions, the piping configuration would not have mattered.

4.9 Sump Pump Operation

4.9.1 Sequence

There are two reactor building sump pumps, WLD-P2A and WLD-P2B, each having a design pumping capacity of 140 gpm. The A pump started at 0408 hrs. and the B pump started at 0411 hrs as indicated on the Alarm Printer. Both pumps continued to operate until they were shut-off by the operators at 0438 hrs. The amount of water transferred by these pumps totaled about 8000 gpm based on the 140-gpm per pump capacity. (Int. 5, 15)

This sump water was a mixture of reactor coolant drain tank (RCDT) quench water and primary coolant vented through the RCDT relief valve until 15 minutes into the event. This valve has a water relieving capacity of 2270 gpm. (Ref. 137; Section 5.5.11.3) The RCDT pressure reached the relief level about 4 minutes before the A pump started. This pump starts at a sump level of 32 inches at which point the sump contains less than 1,000 gallons of water. The second pump comes on at a level of about 44 inches. The flow to the sump was through the rupture disc after it opened at 15 minutes into the event.

4.9.2 Discharge Point

The destination of this water had not been definitely established. The sump tank discharge is normally aligned to the miscellaneous waste holdup tank WDL-T-2. However, the tank level records do not show any level change in this

tank. An alternative discharge point is the auxiliary building sump tank, which has a 3,200-gallon capacity and was already 3/4 full at this time. This subject is addressed in greater detail in Details II of this report.

Evaluation

It is of interest to note that, during the ten-day period prior to the accident, sump pump operation would occur between 2 and 6 times per day, each operation discharging approximately 200 gallons of water. The average for this period was 4.1 transfer operations per day, or approximately 800 gallons per day. The starting of a sump pump would not key the operators to a problem - only its continued operation and the start of the second pump.

During the review of the sump pump operation history for the 10-day period, no trend was noted that would suggest a steadily increasing rate of water accumulation in those sumps.

4.10 Electromatic Relief Valve Failure

4.10.1 History

The EMOV is described in Section 1.6.2.2, which provides a selected maintenance history on that valve immediately prior to March 28, 1979. The section also presents the history of the valve believed to be installed on the TMI-2 pressurizer.

The calibration data (TMI-2 Instrument Calibration Data Sheet, Instrument No. RC3-858) performed on September 24, 1978, shows the valve setpoint values to be 2255 psig high and 2205 psig low (Ref. 16). During the March 28, 1979 event, the high reactor pressure exceeded the high setpoint of 2255 psig in less than 6 seconds after the turbine trip. The EMOV then did not reset when the RCS pressure was below the low pressure setpoint of 2205 psig, approximately 13 seconds later. The EMOV remained open; this was unknown to the operators at that time.

Alarm Printer data indicated that the EMOV exhaust pipe temperature reached 239°F which would be expected within approximately 30 seconds following an EMOV actuation. The EMOV exhaust pipe temperature was printed out as 283°F at 81 minutes (Utility Typer), and remained above the code safety valve exhaust temperature. The exhaust temperature of the EMOV was printed out (Utility Typer) again about 2 hours 18 minutes into the event. The EMOV was isolated at this time by operator actuation of the EMOV block valve (RC-V2).

Evaluation

The history of this valve included indications of leakage. Just prior to the March 28, 1979 event it was suspect, along with the code safety valves, of leaking to the reactor coolant drain tank. The effect of the extended period of leakage on the performance of this valve during the accident cannot be determined at this time.

The evidence clearly supports that the EMOV failed to close at the low-pressure closure setpoint, but a specific cause of that failure has not been determined at this time. Final determination will require access to the valve itself.

4.10.3 Temperature History

4.10.3.1 Relief Valve Tailpipe Layout

The effluent from the EMOV RC-R2 and the two code safety valves (RC-RV1A and RC-RV1B) is piped to the reactor coolant drain tank via a common 14-inch header. (Burns and Roe, P & ID Dwg. No. 2024; Ref. 72) A 4 inch exhaust line having a total run of about 17 feet leads from the EMOV to the 14-inch header. (M.W. Kellogg Isometric Drawings No. 223-1 and 2; Ref. 73) Six-inch exhaust lines having runs of about 20 feet lead from each of the code safety valves to the 14-inch header, entering 3-1/2 and 5 feet below the EMOV exhaust line entry point. Thus the exhaust points for the EMOV and two code safety valves are separated by approximately 40-42 feet.

4.10.3.2 Thermocouple Locations

One chromel constantan thermocouple (T/C) is strapped to each relief valve exhaust line. The T/C on the EMOV line is located about 3 feet downstream of the valve. (Plant Training Photograph No. 443; Ref. 74). The T/Cs on the code safety exhaust lines are located about one pipe diameter from these valves. (Plant Training Photograph No. 81; Ref. 75) Thus, about 40 feet of piping separate any one T/C from the other two.

NOTE: This location is based on discussions with plant staff and disagrees with plant drawings which show the T/C located several feet further downstream of the valves.

These temperatures are normally recorded on a multipoint temperature recorder in the control room. However, this data is unavailable for the period of the incident. The chart drive stopped before the accident and entries were printing over each other during the accident. No information has been obtained to explain the cause of this recorder behavior. Operation of the chart drive resumed after the accident. No chart paper is missing. These three T/Cs are also connected to alarms that are printed out by the Computer Alarm Printer. "High temperature" is printed when one of these points exceeds about 200^oF. The alarm clearing is printed when the temperature decreases below about 193^oF. These status changes are listed in Table I.4-4 for the period of the event. In addition these temperatures can be called up by the operators through the computer and printed by the Utility Typer. This was done several times during the accident, and these temperatures are also included in the Table.

During the accident, the opening of the EMOV or the associated block valve generally caused all three of these T/Cs to alarm. They are connected to the previously described common exhaust header. If the EMOV/block valve pathway remained shut for a sufficient time, these alarms would clear. This generally required 10 to 20 minutes for the code safety exhaust line temperatures

and up to 37 minutes for the EMOV exhaust line. The increased time required to cool the EMOV exhaust line may have been caused by continuing lower level leakage from this valve. In general it is not known whether the relief line closure was by the EMOV or by the block valve, since partial use of the EMOV was apparently recovered during the accident. For this reason the term "EMOV flowpath" will be used in discussing the open/closed status of this pathway.

The safety valve temperature alarm status was used to establish the EMOV flowpath opening times. Closure times were established by changes in the RCS or reactor building pressure trends. The open/closed status of the EMOV flow path and the method used to determine the time of the changes are included in Table I.4-4.

The EMOV opened within a few seconds of when the RCS pressure reached the 2255 psig EMOV opening setpoint. It remained open until 0619 hrs when the operators discovered that the EMOV was stuck open and closed the block valve.

The EMOV flow path was opened twice for brief periods at 0713 hrs and 0719 hrs in order to reduce pressure following the rapid rise that occurred at the time of the restart of reactor coolant pump 2B at 2 hrs 54 minutes.

The EMOV flowpath was opened again at 0741 hrs and apparently left open until 0918 hrs when the decision was made to repressurize the reactor.

From 0944 hrs until ~~1135~~ hrs, the block valve was apparently cycled open and closed intermittently to maintain the pressure between 2200 psi and 1940 psi. (RCS and Reactor Building Pressure Charts; Ref. 135) During this period the block valve was opened and closed on a 3-to-6 minute cycle, with about a 40% open time. The cycle time was too short for the temperature alarms to clear during this time.

Following the decision to depressurize the RCS, the EMOV flowpath was opened at 1135 hrs. During the depressurization the code safety A and B valve temperature alarms cleared at 1217 hrs and 1223 hrs respectively, well before the RCS pressure had reached its minimum value. There were no changes in the

pressure trends at this time to indicate EMOV pathway closure. At 1310 hrs, the beginning of a drop in reactor building pressure indicated that the EMOV pathway did close at this time and the EMOV temperature alarms cleared 7 minutes later to confirm this. However, the RCS pressure was still above the decay heat removal pressure of 400 psi. A possible explanation is that the EMOV had gone shut spontaneously and the previous clearing of the code safety temperature alarms indicated a gradual partial EMOV closure at that time. An EMOV high-temperature alarm and a small rise in reactor building pressure at 1320 hrs, shows that the EMOV pathway opened briefly at that time and closed again at 1323 hrs. The EMOV temperature alarm cleared 9 minutes later.

The operators opened the EMOV manually at 1350 hrs (the time of the building pressure spike). The operators reported that the block valve was already open at this time and that they opened the EMOV in order to increase the RCS venting rate (Int. 14, 15, 17). High alarms on all three relief valve temperatures, confirmed that the EMOV valve opening at this time. The reactor building spray, which actuated at this time, apparently affected the safety valve temperatures, since they cleared at abnormally low temperatures and two of them alarmed high again in the following several minutes. (The EMOV alarm cycled twice.) The EMOV pathway was closed some time between 1401 hrs and 1424 hrs when the EMOV temperature alarm cleared.

The EMOV pathway was opened for the final time between 1635 hrs and 1648 hrs. Approximately a half hour later the final repressurization of the RCS was started. Stable reactor building pressures and the absence of temperature alarms indicate that the EMOV pathway remained closed for the remainder of the incident.

TABLE I.4-4

EMOV AND SAFETY VALVE TAIL PIPE TEMPERATURES

ELAPSED TIME	BLOCK VALVE OR EMOV STATUS	TAIL PIPE STATUS AND TEMPERATURES			INFORMATION	
		EMOV V2	CODE SAFETY V1A	CODE SAFETY V11B	SOURCE	COMMENT
6 seconds	OPEN					Setpoint for opening reached.
30 seconds	--	High 239		High 203	Alarm Printer	
0:25	--	- 285	- 264	- 275	Utility Typer	
1:21	--	- 283	- 211	- 218	Utility Typer	
2:18	SHUT	- 229	- 190	- 194	Utility Typer	Rx building pressure drops.
2:20	--	- 228	--	--	Utility Typer	
2:21	--	- 227	--	--	Utility Typer	
2:24	--	- 220	--	--	Utility Typer	(From 1:13 to 2:47 alarm
3:13	OPEN	High 247	High 203	High 206	Alarm Printer	printer missing.)
3:15	SHUT	--	--	--	Rx Bldg Press	Termination of pressure rise.
3:19	OPEN	--	--	--	Rx Bldg Press	Beginning of pressure rise.

TABLE I.4-4

EMOV AND SAFETY VALVE TAIL PIPE TEMPERATURES

ELAPSED TIME	BLOCK VALVE OR EMOV STATUS	TAIL PIPE STATUS AND TEMPERATURES			INFORMATION	
		EMOV V2	CODE SAFETY VIA	CODE SAFETY V11B	SOURCE	COMMENT
3:20	SHUT	--	--	--	Rx Bldg Press	Termination of pressure rise.
3:31		--	CLR 193	--	Alarm Printer	
3:35	--	--	--	CLR 193	Alarm Printer	
3:41	OPEN	--	High 202	High 205	Alarm Printer	May not have been time for EMOV T/C to cool.
5:15	SHUT	--	--	--	RCS Press. and Rx Bldg Press	Beginning of RCS pressure rise and Rx Bldg Pressure drop.
5:35	--	CLR 192	CLR 193	--	Alarm Printer	
5:40	--	--	--	CLR 193	Alarm Printer	
5:44	OPEN	High 215	--	High 205	Alarm Printer	(Cycling block valve with about 40% open time until 7:35.)
8:00	--	--	--	206.1	Utility Typer	

I-4-39

TABLE I.4-4

EMOV AND SAFETY VALVE TAIL PIPE TEMPERATURES

ELAPSED TIME	BLOCK VALVE OR EMOV STATUS	TAIL PIPE STATUS AND TEMPERATURES			INFORMATION	
		EMOV V2	CODE SAFETY VIA	CODE SAFETY V11B	SOURCE	COMMENT
8:17	--	--	CLR 193	--	Alarm Printer	May have cleared due to partial EMOV closure.
8:23	--	--	--	CLR 193	Alarm Printer	
9:10	SHUT	--	--	--	Rx Bldg Pressure	Termination of rise in Rx Bldg pressure.
9:17	--	CLR 193	--	--	Alarm Printer	
9:20	OPEN	High 220	--	--	Alarm Printer	
9:23	SHUT	--	--	--	Rx Bldg Pressure	Termination of rise in Rx Bldg pressure.
9:32	--	CLR 193	--	--	Alarm Printer	
9:49-9:50	OPEN	High 226	High 204	High 205	Operator, Alarm Printer, and Utility Printer	(H ₂ burn and ES actuation.)
9:52	--	CLR 181	CLR 179	--	Alarm Printer	(EMOV temperature alarms again in 20 sec.) cleared due to Rx Bldg Spray
9:55	--	CLR 177	--	CLR 190	Alarm Printer	Clears again in two minutes

I-4-40

TABLE I.4-4

EMOV AND SAFETY VALVE TAIL PIPE TEMPERATURES

ELAPSED TIME	BLOCK VALVE OR EMOV STATUS	TAIL PIPE STATUS AND TEMPERATURES			INFORMATION	
		EMOV V2	CODE SAFETY V1A	CODE SAFETY V11B	SOURCE	COMMENT
10:01	--	High 207	--	High 200	Alarm Printer	Finally returns to high status, caused by termination of building spray.
10:+	SHUT	--	--	--		Clearance of EMOV alarm at 10:24.
10:24-10:25	--	CLR 192	--	CLR 193	Alarm Printer	
11:27	--	NORM 190.7	NORM 170.6	NORM 174.7	Utility Printer	
12:35	OPEN	High 233	High 203	High 201	Alarm Printer	
12:48	SHUT	--	--	--	Rx Bldg. Pressure	End of Pressure rise.
12:59	--	--	CLR 192	--	Alarm Printer	
13:05	--	--	--	CLR 193	Alarm Printer	
13:25	--	CLR 193	--	--	Alarm Printer	Stable Rx Bldg Pressure after this time indicate EMOV remained closed.

I-4-41

EMERGENCY FEEDWATER PUMPS — STATUS

I-4-42

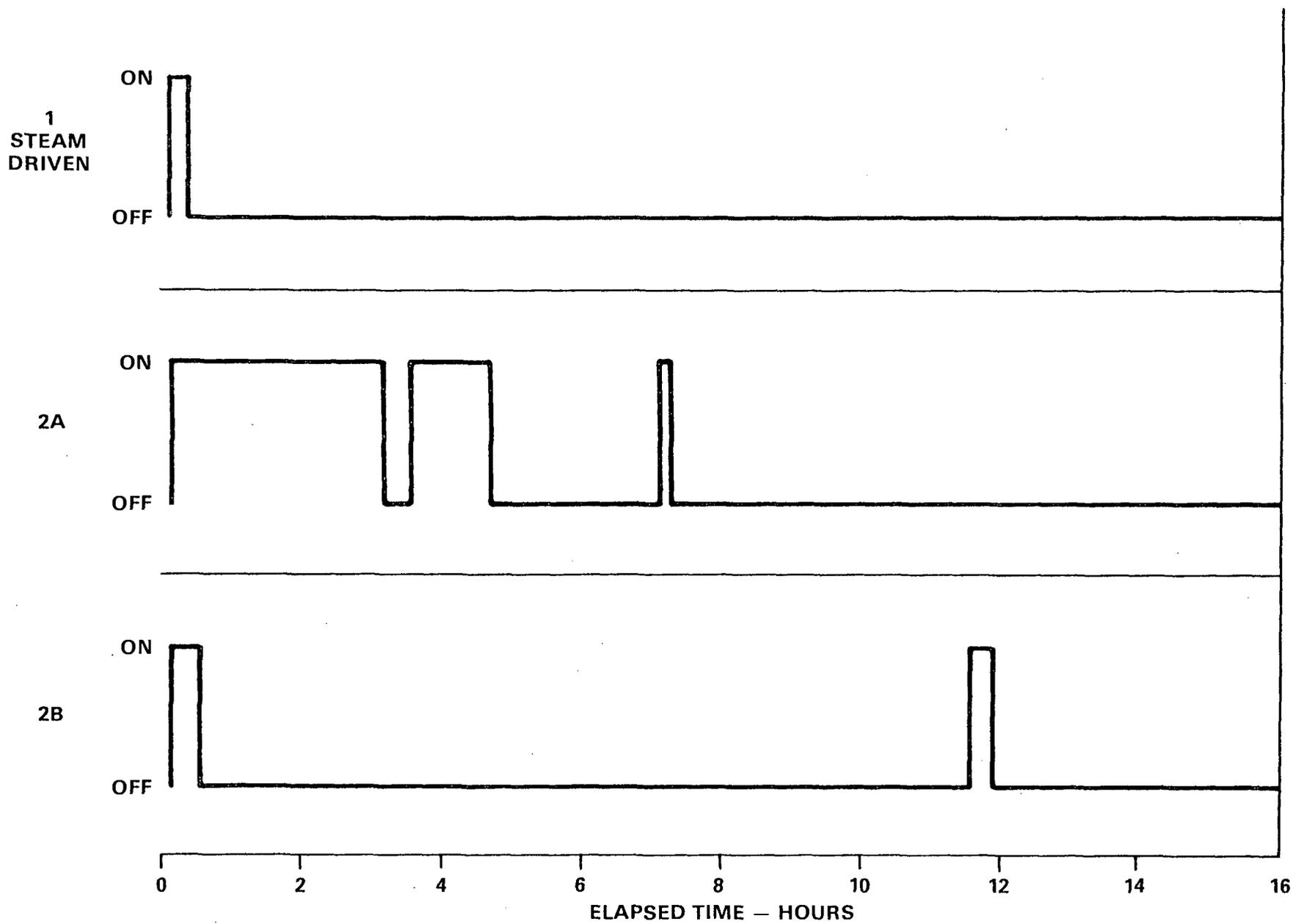


FIG 1.4-3

4.11 Emergency Director Leaves Site

4.11.1 Sequence

According to interviews, the Lieutenant Governor of Pennsylvania called the President of the licensee's organization requesting a meeting on the status of the plant and related matters. It appears that an initial meeting time of approximately noon (8 hours after the start of the accident was established) but was changed to about 1430 hrs (10-1/2 hours after the start of the accident) when the Plant Manager objected to the earlier meeting time. (Int. 1, 21, 34)

It appears that it was the decision of the Vice President - Generation to incorporate onsite plant management in that meeting. As a consequence of that decision and at some time prior to the meeting, the Unit 2 Superintendent - Technical Support was directed to start collecting technical materials for that meeting.

Based on the interviews, it appears that the Station Manager and the Unit 2 Superintendent - Technical Support left the control room shortly before 1400 hrs. They subsequently met the Vice President - Generation at the North Gate (he was previously at the Observation Center). They then proceeded to the Lt. Governor's office where the meeting began at approximately 1430 hrs and lasted 30-45 minutes.

It further appears that the Station Manager and the Unit 2-Superintendent - Technical Support returned to the control room at or about 1630 hrs.

4.11.2 Precautions Taken

Prior to leaving for the meeting, the Station Manager took the following actions:

reviewed plant status to determine if conditions were stable. (The Station Manager stated that he would not have left for the meeting if that were not the case.)

- . ordered the Unit 1 Operations Supervisor to maintain the status quo as regards reactor conditions.
- . appointed the Unit 2 Superintendent Acting Emergency Director.
- . informed the Unit 1 Superintendent of where he was going.
- . was provided with a beeper to enable contact while enroute to and from the meeting.
- . arranged for the Unit 2 Superintendent - Technical Support to make contact with the site from the Lt. Governor's office. (This was done, and contact was maintained for approximately 15 minutes while the meeting was in progress.)

Evaluation

It was not possible for the investigation team to establish the impact that the absence of the individuals involved had on the course of the accident. Prudent precautionary steps were taken prior to his departure. The steps taken were comparable or in excess of those that would be expected in the event an Emergency Director were incapacitated in some fashion.

The decision that repressurization and an RCP start should be pursued occurred after the meeting participants returned to the site and were briefed on the status of the plant. Whether this decision would have been reached earlier, or the attempts to refill the loops that proceeded during the meeting would have been abbreviated would be conjecture and thus, outside the scope of this investigation.

4.12 RCS Average Temperature Indication

Operator interviews indicate they were confused and possibly misled by the behavior of the RCS Average Temperature (Tav) indications. A review of

Plant Strip Charts recorded on March 28, 1979, indicates that Tav "sat for the longest time" at 570°F; about 11 hours out of the 16 hours reviewed by the investigation team.

Tav indication is developed from signals generated by narrow-range RCS hot-leg and cold-leg temperature instrumentation. These source instruments produce minimum to maximum strength signals corresponding to their allowable temperature range of 520°F to 620°F. If temperature at the point of measurement drops below 520°F, the signal delivered to the Tav computation circuit remains at minimum levels, implying that actual temperatures stop dropping at 520°F. If temperature at the point of measurement rises above 620°F, the input to the Tav computation circuit is effectively clamped at 620°F. Normally, RCS hot-leg and cold-leg temperatures will track essentially together, since major changes in these parameters are usually due to plant heatup or cooldown. Tav indication during these normal plant evolutions would not appear anomalous, although the indication would be in error as soon as either input left the allowable range for its instrument. (Int. 56, 61.)

Evaluation

During the accident, RCPs were tripped and hot-leg and cold-leg temperatures quickly diverged. With the hot leg above 620°F and the cold leg below 520°F, the Tav computation circuit functioned as designed and indicated Tav at 570°F. That the operators found comfort in this near normal indication indicates they were unaware of the limitations of this instrument.

4.13 Plant Computer Records

The investigation team relied heavily on computer generated records related to this accident to substantiate reported events and their timing, to determine information of which the operators should have been aware (based on control room annunciators), and to identify plant problems not yet reported to NRC. The records were typed on the Alarm Typer, the Utility Typer and the Log Typer and were inked on the four Analog Trend Recorders. Copies of these

records were made available to the investigation team following extensive searches of plant records by NRC and licensee personnel. They are essentially complete, with the exceptions discussed below.

4.13.1 Alarm Status Printout, 05:13:59 to 06:48:08, 3/28/79

Analysis of Alarm Typer and Utility Typer output prior to and during this period indicates that alarm status printout records for this period, if they ever existed, would cover only the period 05:13:59 to about 05:15:16. The indication is consistent with an assumption that the operator actuated the alarm suppress function at about 0648 hrs, thereby dumping the alarm status printout memory and restarting the output at the current time. This action is permissible and would be in the best interest of the operators, who needed current information, not the history of nearly one and three quarter hours before. Interviews with plant personnel have not determined the actual cause of data loss. No evidence has been identified that the data was purposely destroyed or withheld to hamper the investigation.

4.13.2 Alarm Status Printout, 18:48:59 to 19:10:29, 3/28/79

Analysis of Alarm Typer and Utility Typer output prior to, during, and after this period indicates that alarm status printout was probably generated within this period, but may have been so unintelligible as to be useless. The Alarm Typer, on which the alarm status printout was being typed, had a history of paper jamming, causing overstrikes, tearing, and general havoc with the resulting record. The record from 19:10:29 to 19:14:24 shows clear evidence of this condition. Whether the missing records were lost, thrown away as useless, or otherwise disposed of has not been determined. Records of alarms during this period were not critical to the understanding of the accident sequence and did not significantly impact on this investigation.

4.13.3 Utility Typer Output, 00:00:00 to 03:24:24, and 20:08:22 to 20:12:*, 3/28/79

The records from this typer during the periods of interest, if they ever existed, were not found. They are not considered critical to this investigation.

*NOTE: Paper is torn and time in seconds is missing.

4.13.4 Analog Trend Recorder Number 2, 3/28/79

The strip chart from this recorder has reportedly not been found.

4.14 Pressure Spike in Containment

4.14.1 Sequence

The 28-psig pressure spike in the TMI-2 containment occurred at 13:50:21 (approximately 9 hours 50 minutes after the start of the accident) on March 28, 1979 based on data. (Alarm Printer, Ref. 14) An operator had just been directed to open the EMOV valve. Concurrent with his manipulation of the controls, some operators and the Station Manager heard a "double thump," and the alarms and automatic equipment functions associated with high pressure in the containment actuated.

The Alarm Printer shows the following sequence of events:

13:50:21 Channels A&B HPI and building isolation setpoints reached (~4 psig).

13:50:27 Channel A&B high-pressure setpoint (~28 psig) actuated.

13:50:32 High-pressure setpoint (~28 psig) clears owing to rapid decrease in pressure after pressure spike.

13:50:46 A channel 4-psig setpoint clears.

13:50:48 B Channel 4-psig setpoint clears.

13:56:07 Spray pumps A&B tripped.

(It should be noted that these entries from the Alarm Printer are included here to establish precise times of events. The operators would not have been aware of these entries at the time because the computer was running behind real time due to the number of alarms being printed.)

The alarm that initiated building spray cleared in 5 seconds. The alarm that initiated the other protective functions (lower setpoint) was in the alarm status for a total of approximately 25-27 seconds. Because of the need to restore other equipment to operation, the building spray pumps were not shutdown until approximately 5-1/2 minutes after they started.

This event occurred shortly after the licensee had reduced reactor pressure sufficiently to attempt to "float" the core flood tanks on the reactor coolant system. The effectiveness of floating the core flood tanks was receiving most of the licensee staff attention. Moreover, all personnel in the control room were wearing respirators at this time because of airborne radioactivity problems which caused substantial communications problems.

4.4.2 Response of Personnel to Pressure Spike

Based on interviews conducted during this investigation, it appears that the response of personnel present in the control room to this pressure spike was varied, including a total lack of awareness that it occurred. No statements that have been obtained indicate that anyone present postulated that the pressure spike was due to the rapid burning of hydrogen.

One supervisor, in the area of the controls but not immediately at them, ascribed the actuation of the building spray pumps to an electrical problem. Apparently, the loss of the electrical buses 2-32A and 2-42A, which occurred at 1351 hrs (approximately 30 seconds after the spike) contributed to this conclusion. Others attributed the spike to various sources including electrical interference on the recorder or the switches that actuate building spray.

Another shift supervisor, who had entered the control room earlier and assumed responsibility for the controls, saw the pressure spike occur in coincidence with the opening of the EMOV. (Int. 119)

Based on this latter supervisor's interview, it appears that he directed the operators to leave the spray pumps on until he was assured that building pressure had stabilized and no further pressure spikes were occurring. When he felt satisfied that conditions had reestablished, he directed the spray pumps be shutdown. He stated that he recommended to the Station Manager that the EMOV no longer be cycled because the rapidly rising building pressure corresponded to opening the EMOV. This apparently was brought to the Station Manager's attention immediately prior to his leaving for the meeting at the Lt. Governor's office. It appears that the incident was not recalled until late the following day when the building pressure trace was reviewed.

In general, the investigation has concluded that the staff on duty in the control room did not attach any special significance to the pressure spike. Those who actually saw the spike related it to opening of the EMOV, and did not attach significance to another cause. It should also be noted that the recorder displaying the spike displays the trace for approximately 2 hours, after which the trace must be removed from the recorder or the recorder pulled out to the extended position in order to be viewed.

NRC personnel were also in the control room during this period. During interviews with those individuals, they stated that they were not aware of the pressure spike. (Int. 31)

4.14.3 Notification of NRC

It appears based on interviews and log books that the NRC was notified of the pressure spike early in the morning of March 30, 1979. It was referred to in Preliminary Notification PNO-79-67D prepared late that same day. An interview with an NRC inspector indicated he recalled discussing the matter with the Station Manager on that date, after the licensee investigation team had reviewed the Plant Strip Charts the evening before (March 29, 1979).

An interview (Int. 139) of a control room operator indicated that copies of the RB recorder having the spike were made and in the control room at about 2230*. The person stated that he arrived at the Observation Center at approximately 1400 hrs and went to Unit 2 Control Room, wearing a breathing mask, at about 1445 hrs. He stated that he was told about a reactor building pressure spike and after 2000 hrs that night, he stated that there was discussion as to what the spike was. He stated that a hydrogen explosion was discussed. He stated that "We pulled it out and I don't know who wanted a copy but we made a couple copies of the chart."

He stated that he discussed the spike with a trainee and the person who asked for the information. He did not remember if this was an NRC inspector or a B&W representative.

The investigation looked into this allegation and has concluded that the individual has confused the events of the evenings of March 28 and 29, 1979. This conclusion is based on the following information.

The investigator reviewed the original combined wide-range 0-100 psig (BS-PT-4388-2) and narrow-range -5-0 +10 psig (BS-PT-4388-1) chart. The chart shows that, on March 28, 1979, at approximately 1350 hrs, two peaks occurred. The narrow range goes off scale and the wide range peaks at about 28 psig. A portion of the charts had been cut out from the original Strip Chart. The cutout section of the chart was matched to the two adjacent sections and carefully reviewed for pen disruptions that could be expected if the chart was removed on March 28, 1979, in order to make copies and then returned to the recorder. The chart has a written notation that indicates "chart removed, March 29, 1979 at noon."

*NOTE: CRO stated that copies were made and in the control room after the establishment of a bubble in the pressurizer. This would establish a time of 2218 hrs or later based on the CFR log, March 28, 1979.

*CRO stated copies were made in the control room after the establishment of a bubble in the pressurizer. This establishes a time of 2218 hours or later, based on the CRO Log March 28, 1979.

The cut section has an early cut line at 0200 hrs March 28, 1979, and a late cut line at 2215 hrs. The chart runs out about 1200 hrs (noon) on March 29, 1979.

The possibility that copies of the reactor building pressure trace were made on March 28, 1979, appears possible but unlikely. The reviewed originals had no obvious pen disruption to confirm this. The CRO later expressed reservations as to whom he talked to and when. NRC personnel do not recall a xerox copy in the control room on March 28, 1979.

Further discussions were held with licensee and GPUSC technical staff who were identified as having knowledge of the copying of the pressure trace. A GPU staff member stated in a discussion that he was not in the control room on March 28, 1979. He did get copies of the reactor building pressure spike around the 29th or 30th of March. He also stated he made copies for NRC staff.

Based on these findings, it appears the individual who expressed the view that the pressure spike was distributed at the site on March 28, 1979 was mistaken, and that the actual analysis and review took place on March 29, 1979 as previously mentioned.

4.15 High Temperatures and Core Response

4.15.1 General

There were four indications available to plant personnel to indicate that the core was not being adequately cooled:

- o SRM and IRM anomalous indications.
- o Offscale (greater than 700°F) incore thermocouple readings
- o Hot-leg RTD readings that indicated superheated conditions of the coolant

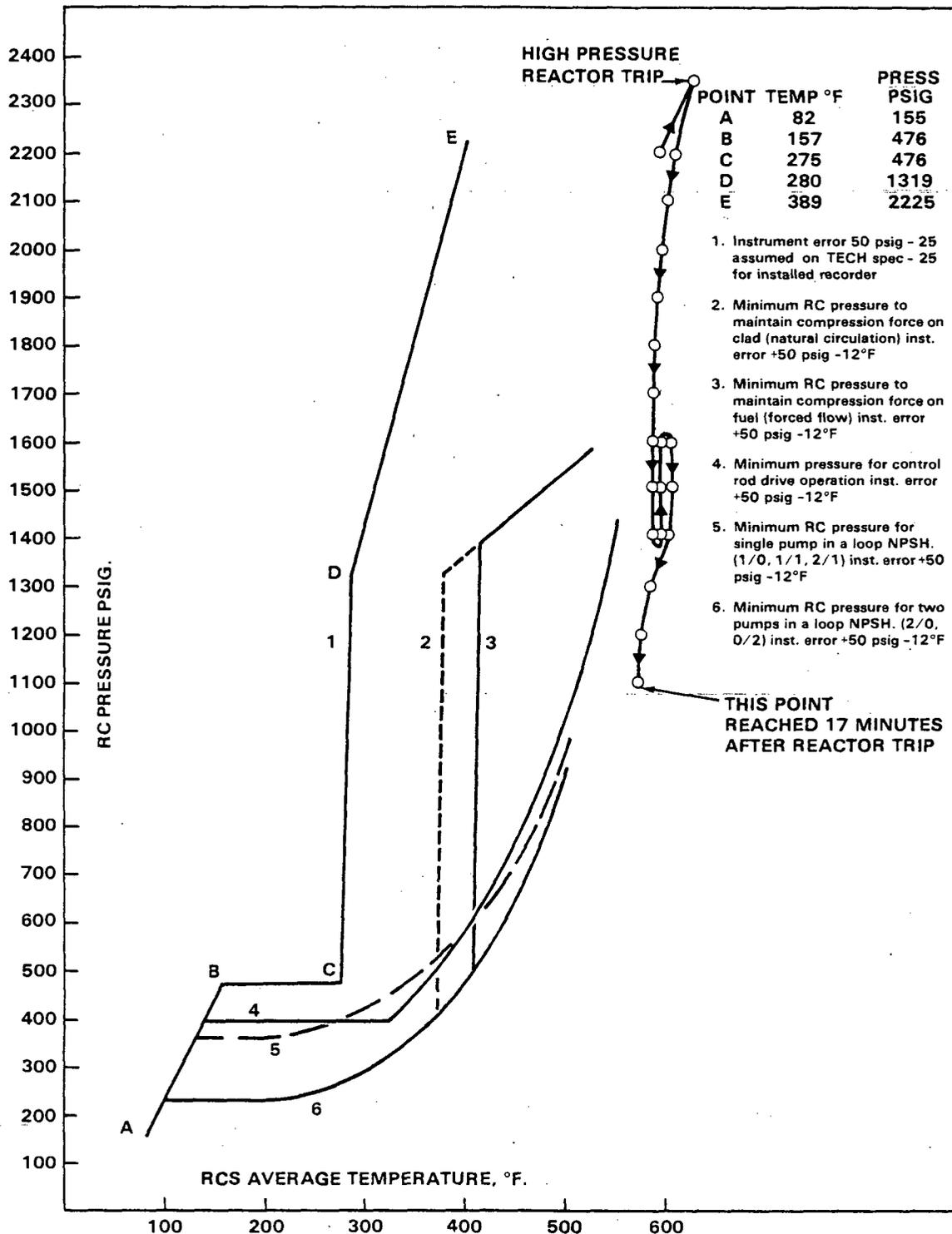
- o High radiation readings in the containment and elsewhere indicating a large-scale release of radioactivity to the RCS.

The onset of significant release of activity from the fuel because of inadequate cooling cannot be fixed in time. Review of radiation monitor charts for the hot machine shop, incore detector cabinets, and the reactor building dome monitor indicates fuel failures occurring prior to 0630 hrs. Basically two failure mechanisms can be postulated for release of activity from the fuel.

The first failure mechanism is associated with the pressure differential across the cladding and is related to the pin compression factors. Pin compression factors appear to have been violated within the first 20 minutes of the accident. Technical Specifications do not address pin compression factors. The temperature-pressure relationship as experienced in the transient are plotted in Figure I.4-4. Review of activity levels in the RCS as monitored by chemical analysis and radiation monitors does not indicate untoward releases of activity prior to core becoming uncovered. Lower pressures and higher temperatures were experienced during the period when core became uncovered. There is a likelihood that there was some mechanical failure of the cladding during this period.

There is almost virtual certainty that a zirconium-water reaction took place based on the release of hydrogen from the RCS. The temperature design limit for loss of coolant accidents is 2200^oF peak clad temperature (Ref. 138). Temperatures of this order were indicated by the incore thermocouples at approximately 0900 hrs after the core was reflooded. It is thought that this temperature was reached earlier when the core was uncovered, although there is no direct evidence of this. Significant structural damage to the core is thought to have occurred when an RCP was started at 0654 hrs. When the pump was started, a rapid quenching of the core is indicated by the SRM and IRM response addressed below. In addition, the core reflood was attended by a rapid pressure change.

HEAT UP/COOL DOWN CURVE



COMPARISON OF RCS TEMP/PRESS. BEHAVIOR TO LIMITS (1ST 17 MIN.)

FIG 1.4-4

4.15.2 SRM and IRM Responses

Within 45 minutes of the reactor trip, SRM Channel 1 was showing an anomalous response. From this time until the last two RCPs were tripped at 0541 hrs, increased radiation transport due to reduced density in the radiation path from the core to the detectors is evident. An increase in the indication of IRM Channel 3 noted during the latter part of this period is due to the same effect. When the last two pumps were tripped, it is felt that many of the steam voids collapsed in the radiation path; this is noted by the increased attenuation (decreasing signal) of the SRM and IRM. The signal levels then increase, and it is believed that, prior to 0600 hrs, the core became uncovered, although the exact time cannot be determined. The exact level is also unknown. At 0654 hrs, an RCP was started and, because of a sharp increase in attenuation observed on the SRM and IRM, it is believed that a reflood of the core took place. After this time it is believed that the core became uncovered again. This condition was terminated at about 0715 hrs when substantial HPI flow was established and the effect of increased attenuation is noted. At approximately 0745 hrs, an increased signal is noted on the SRM trace. The cause for this increase is unknown but it is postulated that the change is consistent with a core geometry change.

4.15.3 Incore Thermocouple Response

Chromel-Alumel thermocouples are located in 52 selected fuel assemblies and are positioned approximately 6 inches above the active fuel. At

04:33:13 the following computer alarm was received:

BAD 0538 IM INCORE T/C 10-R TEMP-???

The cause for this alarm is unknown. This detector is located in a peripheral fuel assembly and thus is at a relatively low power level. At this time, core flow is relatively high and it is doubtful, assuming normal core geometry, that the alarm is indicative of an uprange (>700°F) temperature.

Incore thermocouple (T/C) data from the Alarm Typer does not exist for the time period prior to 06:48:08 except as previously mentioned. A list of incore T/C values as output by the process computer are listed in Figures I.4-5 to I.4- 8. These values cover the period from 0655 hrs to 0927 hrs. This figure indicates values obtained on normal scan of the computer.

From 8:34:07 to 8:36:40, sixteen incore T/C temperatures were requested from the computer, probably by the Instrumentation Engineer. It has not been possible to ascertain exactly who requested the data. In addition, from 08:46:44 to 08:47:58, eleven incore T/C temperatures were requested, again probably by the Instrument Engineer. These are shown in Figures I.4-9 and 10. A comparison of the data requested shows that, for locations that are duplicated in the requests, there is an indicated temperature drop between the time the data was first obtained and the second time the data was requested. Between 0800 hrs and 0900 hrs, readings were taken at the computer input in the cable room. This data was obtained using a thermocouple reader and digital voltmeter measurements with suitable conversion to temperature. This data is indicated in Figure I.4-11. A review of this data compared with the data output between 08:46:44 and 08:47:58 reveals an indicated temperature drop for common locations to the two sets of data. This fact would indicate the measurements at the computer input were taken after 8:47:58. Review of this data indicates significant flow blockage in the center of the core. It should be noted when examining Figure I.4-11 that the values are approximate because corrections were not made for the reference junction ($\sim 75^{\circ}\text{F}$ correction).

4.15.4 Incore Flux Detector Response

The reactor core contains 364 self powered nuclear detectors (SPND) with 7 detectors equally spaced in 52 strings.

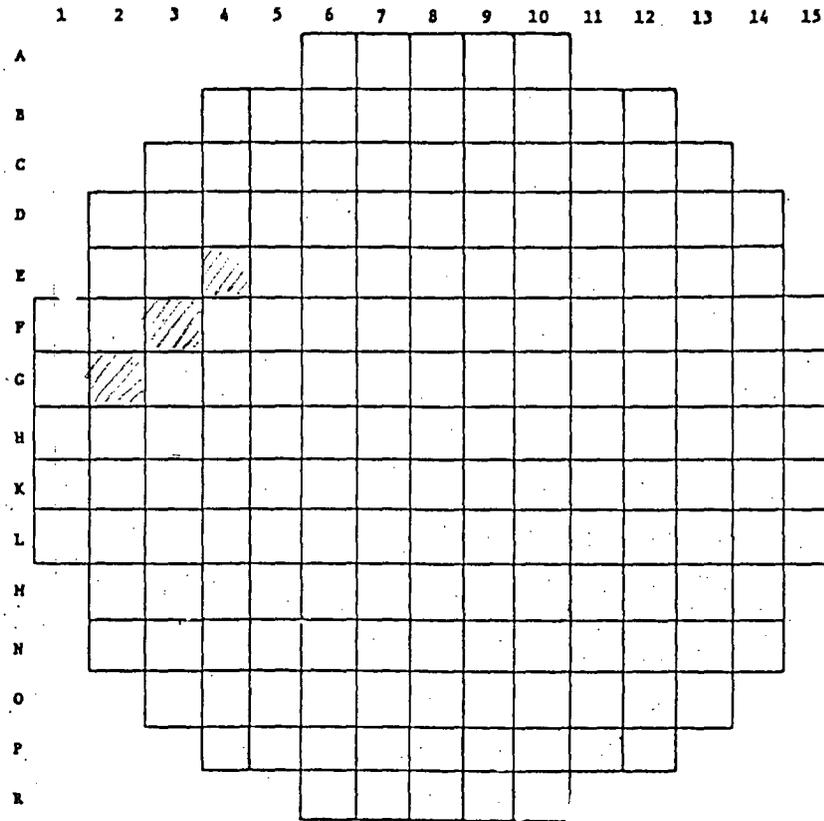
A review of backup recorder traces of incore flux detectors (SPNDs) outputs indicate a normal response to the reactor trip at approximately 0400 hrs. At approximately 0640 hrs, some of the detectors were producing a measurable output. By 0700 hrs, the SPND's at locations F-3, G-5, and K-11,

level 4 are off-scale. Shortly after the SPND at location H-8 level 2 was off-scale. At 0810 hrs, the SPND at H-8 level 6 is off-scale. By 0830 hrs, the SPND's at levels 2 and 4 of H-9 are off-scale. The output signal of the SPND's is believed to be caused by thermionic emission by the rhodium wire in the detector. Measurements taken based on the current output of the detectors prior to the trip indicate an off-scale current of approximate 2000 nanoamperes.

Other detectors exhibiting on-scale outputs are:

<u>Location</u>	<u>Level</u>
H-8	4
G-11	2
G-11	4
C-11	2
K-11	6
M-9	6
F-3	6
L-13	6
K-5	6
F-13	4
M-7	6
K-5	6
E-7	6
F-13	6
L-13	6

It should be noted that the SPND at location H-8 level 6 remains upscale until approximately 1315. (The SPND at location L-13 level 6 remained upscale until the afternoon of March 29.) It is believed that the core was uncovered to a depth of at least 9 feet below the active fuel height based on the response of the level 2 detectors previously referenced.



NOTES:

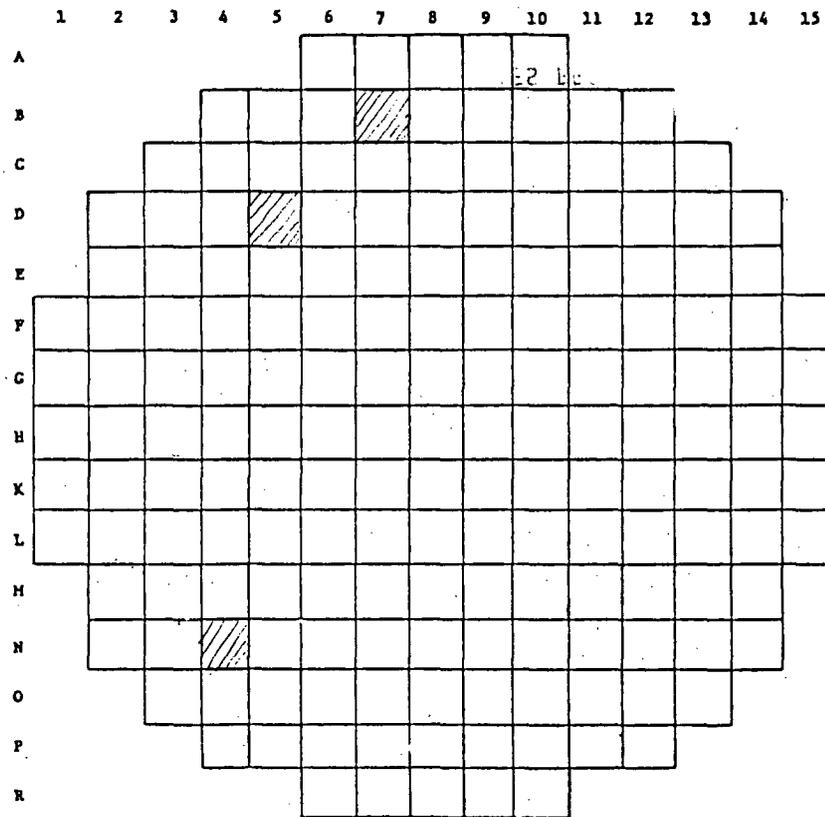
- (1) "?" indicates value outside of computer software range (700°F).
- (2) Time shown is actual time. Turbine trip occurred at 04:00:37.

I-4-57

LOCATION:

2-G		3-F		4-E	
Time	Temperature (°F)	Time	Temperature (°F)	Time	Temperature (°F)
06:55:40	623	07:03:40	681.9	07:43:40	680.2
07:16:40	?	07:09:40	?	07:44:40	?
07:21:40	630.3	07:10:10	694.1	08:05:40	610.3
08:36:40	518.6	07:10:40	?	08:31:10	?
08:48:39	?	07:11:10	643.9	08:36:22	?
		08:36:38	309.7	08:42:00	699.3
				08:45:09	?

FIGURE I.4.-5 SELECTED INCORE T/C HISTORIES FROM COMPUTER PRINTOUTS



NOTES:

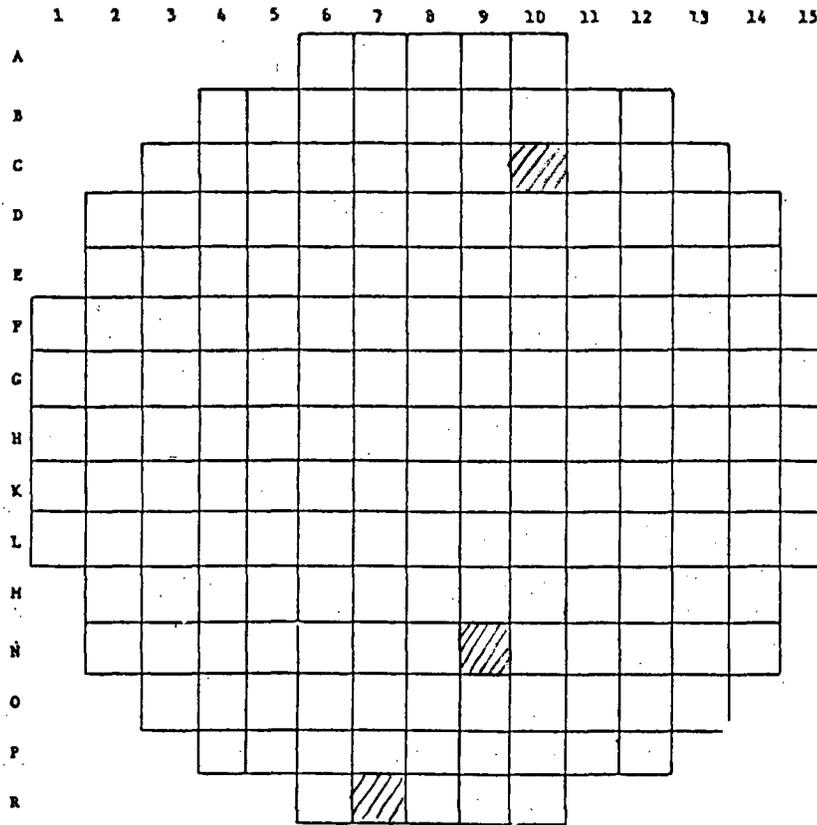
- (1) "?" indicates value outside of computer software range (700°F).
- (2) Time shown is actual time. Turbine trip occurred at 04:00:37.

I-4-58

LOCATION:

4-N		5-D		7-B	
Time	Temperature (°F)	Time	Temperature (°F)	Time	Temperature (°F)
07:23:41	693.4	07:03:40	699.5	06:55:10	623
07:27:41	?	07:05:40	664.2	07:15:40	?
07:29:11	658.2	07:12:40	666.1	07:21:40	627.7
07:30:41	?	07:13:10	?	08:36:00	436.3
07:33:41	695.3	07:23:40	645.0	08:47:43	377.6
08:03:11	?	07:44:40	?		
08:06:41	23.7	08:36:15	?		
		08:47:58	?		

FIGURE I.4-6 SELECTED INCORE T/C HISTORIES FROM COMPUTER PRINTOUTS



NOTES:

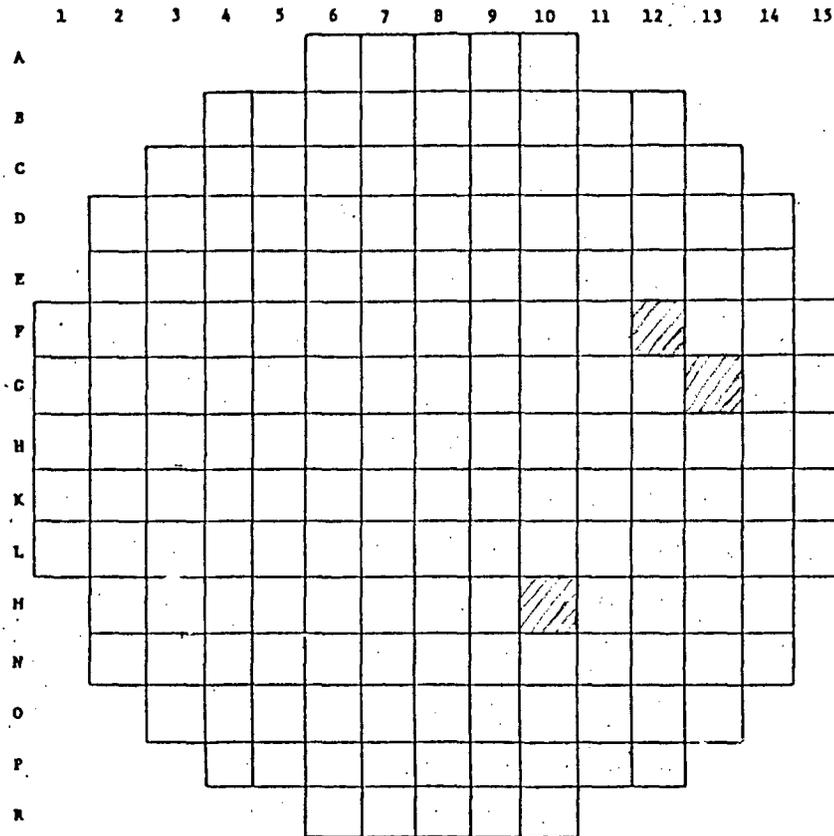
- (1) "?" indicates value outside of computer software range (700°F).
- (2) Time shown is actual time. Turbine trip occurred at 04:00:37.

I-4-59

LOCATION:

7-R		9-N		10-C	
Time	Temperature (°F)	Time	Temperature (°F)	Time	Temperature (°F)
06:57:11	679	09:11:39	611.7	06:55:40	599
07:20:41	?	09:12:09	?	07:19:10	?
07:21:11	630.5	09:16:09	639.0	07:21:10	637.9
07:45:41	?	09:25:39	?	08:35:38	465.1
07:51:41	641.0	09:26:39	675.4	08:47:19	398.9

FIGURE I.4-7 SELECTED INCORE T/C HISTORIES FROM COMPUTER PRINTOUTS



NOTES:

- (1) "?" indicates value outside of computer software range (700°F).
- (2) Time shown is actual time. Turbine trip occurred at 04:00:37.

LOCATION:

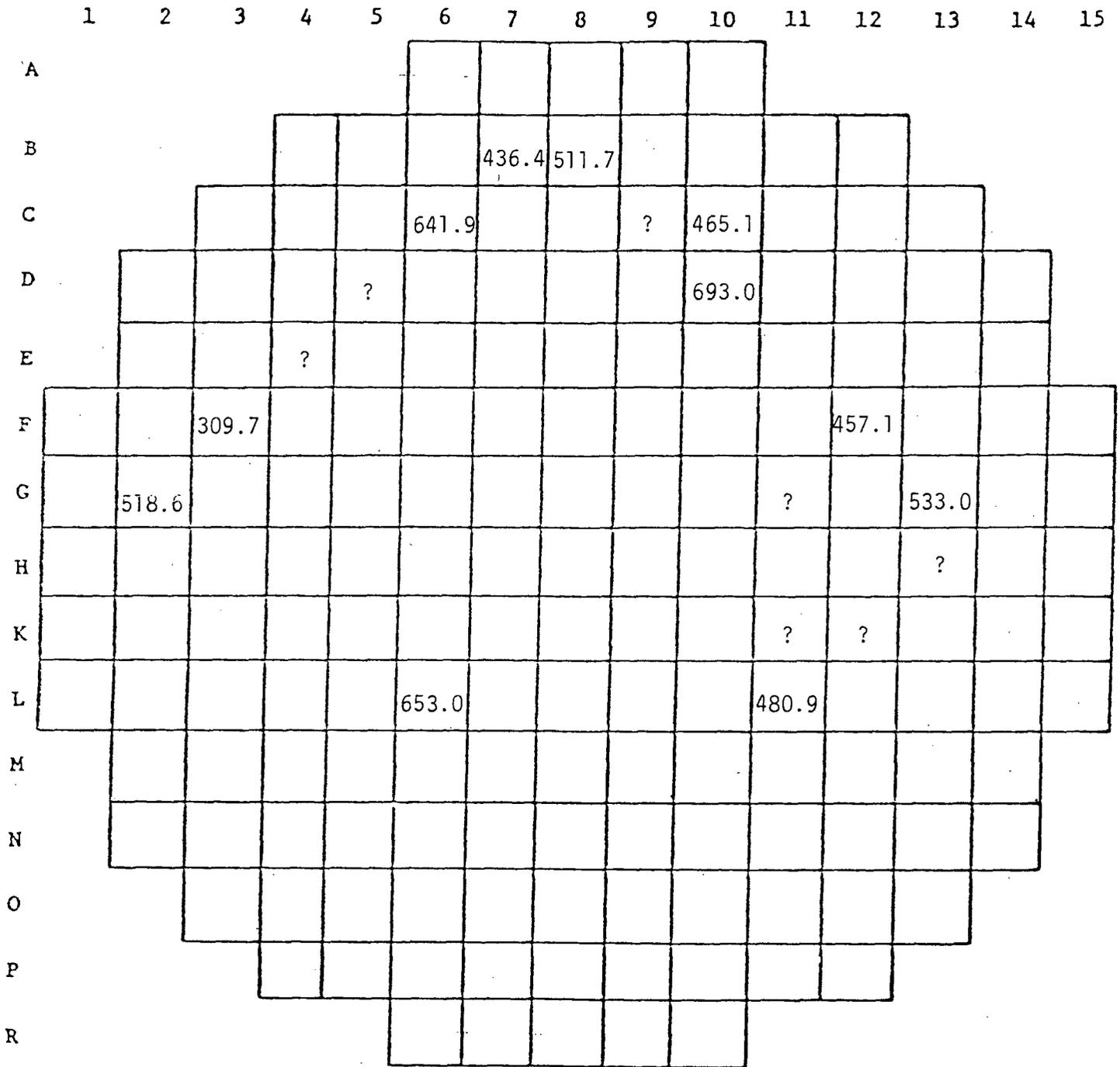
10-M	
Time	Temperature (°F)
07:32:10	689.0
08:15:10	693.2

12-F	
Time	Temperature (°F)
08:06:10	694.9
08:15:10	?
08:15:40	588.5
08:34:44	457.1
08:46:51	393.1

13-G	
Time	Temperature (°F)
06:57:10	670
07:06:10	?
07:06:40	675.6
07:19:40	?
07:21:40	682.1
08:01:10	?
08:34:37	533.0

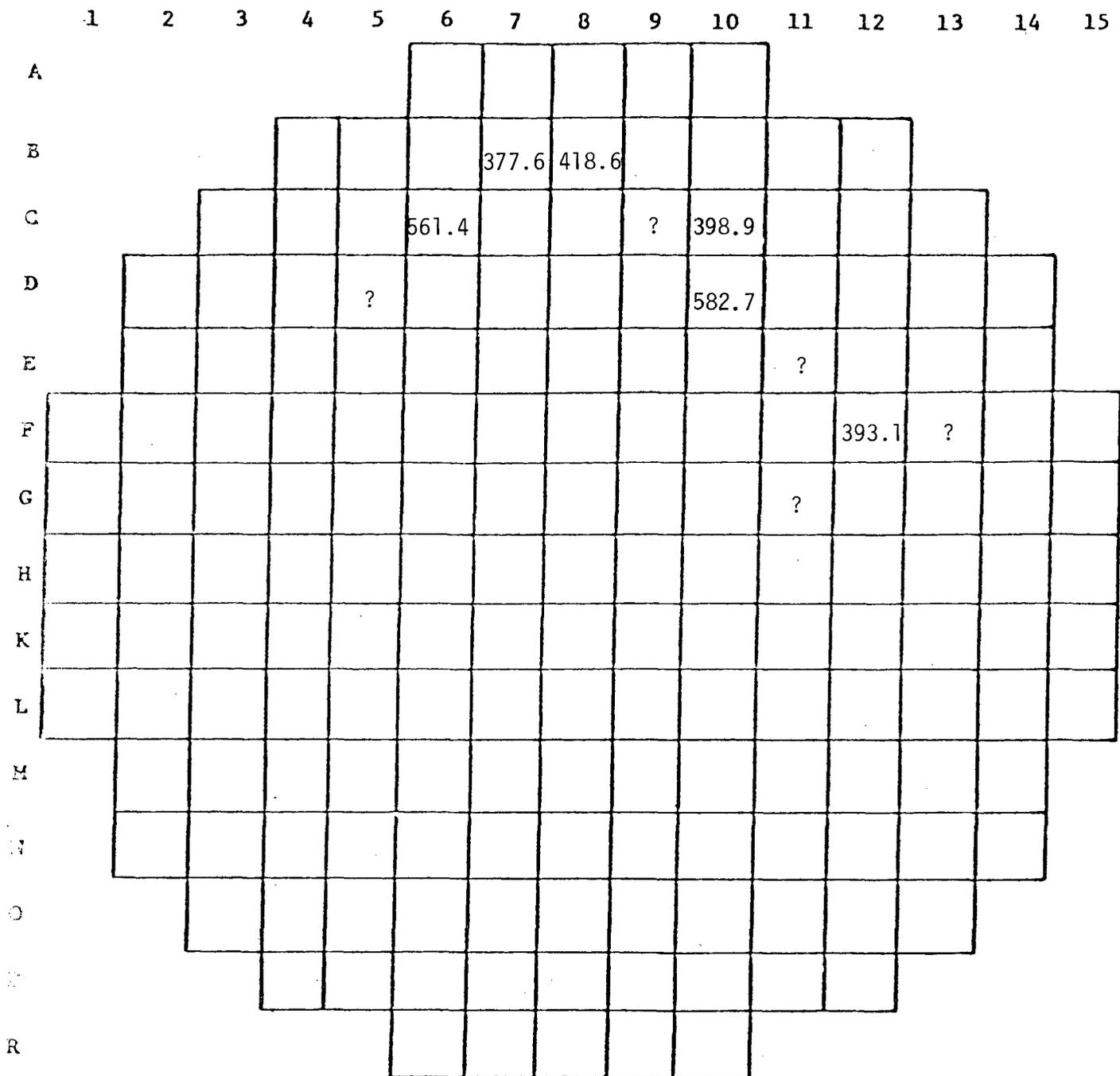
I-4-60

FIGURE I.4-8 SELECTED INCORE T/C HISTORIES FROM COMPUTER PRINTOUTS



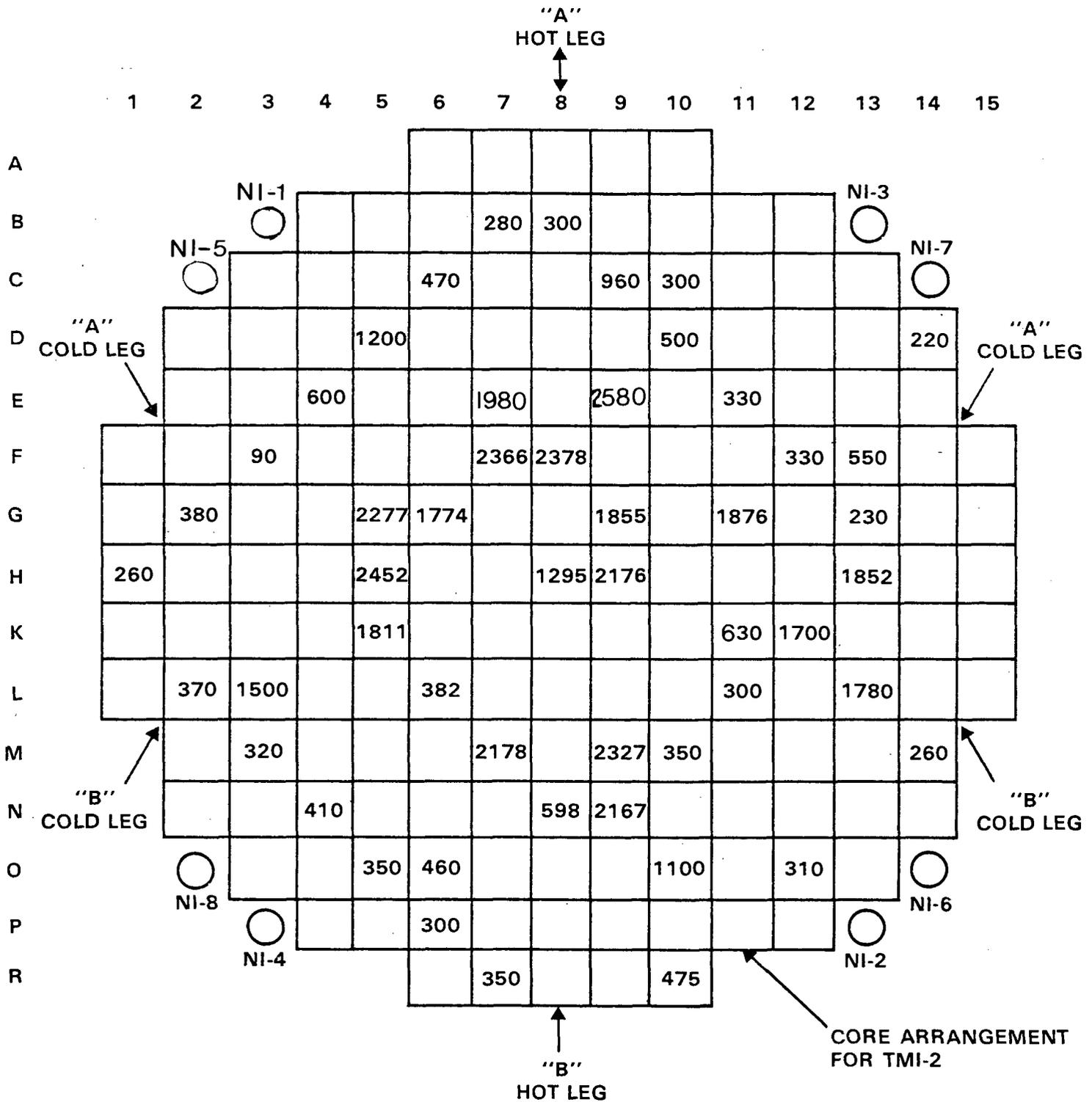
NOTE: TEMPERATURES IN °F
 ("?" INDICATES OFFSCALE)

FIGURE I.4-9: INCORE THERMOCOUPLE DATA
 FROM COMPUTER AT 08:34-08:36
 ON MARCH 28, 1979



NOTE: TEMPERATURES IN °F
 ("?" INDICATES OFFSCALE)

FIGURE I.4-10: INCORE THERMOCOUPLE DATA
 FROM COMPUTER AT 08:46-08:47
 ON MARCH 28, 1979



READINGS OBTAINED FROM INCORE THERMOCOUPLES AT APPROXIMATELY 4-5 HOURS AFTER START OF ACCIDENT

FIG 1.4-11

4.15.5 OTSG and Primary Temperature Recorder Display

Various primary and steam generator temperatures are recorded on a 24-point dual-range recorder on panel 10 in the west side of the control room, behind the console facing the center of the room. The point designations and ranges are listed in Table I.4-5. Of particular interest are the six primary coolant temperatures, which comprise one point for each of the hot-legs and one for each pump suction. All of these are displayed with a 0°F to 800°F range. All of the six remained within range, except for several 3- to 6-minute periods, during the entire incident. A brief description of the behavior of four of the traces is given below. This is based on a review of the original recorder trace and which are shown on Figures I-4-12A, B, C, and D for the period 0400 hrs to 2030 hrs on March 28, 1979. The original chart is multicolored, and traces outside the period given above were used to correlate the point numbers with the colors. All times are given in hours and minutes of elapsed time after the start of the accident.

4.15.5.1 Hot-Leg Temperature Behavior

Both of the hot-leg temperature indications were in excess of saturation temperature a few minutes after the A loop pumps were tripped at 101 minutes. The A loop temperature increased from 540°F to 800°F over the next hour, going off scale briefly at about 0648 hrs.

The temperature rise was interrupted by a temporary reduction of 40°F at the time that the EMOV was closed at 0619 hrs. A similar reduction occurred about 10 minutes later. The B loop temperature rose more slowly, going off scale briefly at about 0700 hrs. During the rise, the temperature peaked sharply at 0634 hrs then dropped 60°F over the next 10 minutes.

Both A and B loop hot-leg temperatures had sudden reductions (of 60°F and 30°F, respectively) when the attempt was made to start RCP-2B at 0654 hrs. Another sharp drop of 90°F occurred in A loop temperature when ES was initiated

by the operator at 0720 hrs. For the next 6 hours, both hot leg temperatures remained relatively stable, with A loop varying between 680°F and 760°F and B loop tracking A loop about 60°F higher.

The ES actuation resulting from the pressure spike in the reactor building at 1350 hrs caused another sharp drop in both hot leg temperatures of about 50°F. This may have been caused by cooling by the building spray, which was actuated by this ES actuation. Following this, the A loop began an irregular downward trend reaching 450°F by 1510 hrs. Large swings of between 470°F and 600°F occurred during the period of this trend. A drop of 150°F starting at 1428 hrs was also indicated by the A loop strip chart recorder; it was attributed by the operators to increased makeup water injection through MU-V16C. However, the start of a sudden increase of 85°F coincides with the start of MUP-1C at 1432 hrs, and this increase is terminated at the time that MUP-1C is secured. A similar rise coincides with the start of MUP-1C at 1519 hrs. This rise also terminates about the time MUP-1C is stopped at 1128 hrs. During this time (1350 hrs to 1630 hrs), B loop hot-leg temperature remained generally constant above 700°F.

Both hot-leg temperatures began a drop of about 120°F when the operator opened the EMOV block valve at 1635 hrs. This was followed by a return to the previous conditions for the next 2 hours, with A loop hot-leg between 550°F and 590°F and B loop between 720°F and 760°F.

The momentary start of RCP-1A at 1933 hrs caused another sudden decrease in both A and B loop hot-leg temperatures of 150°F and 100°F, respectively. When this pump was finally restarted at 1950 hrs and left running, all of the primary temperatures rapidly converged at 350°F.

4.15.5.2 RCP Suction Temperature Behavior

The RCP suction temperatures began a gradual decrease after the RCPs were tripped. They were less responsive to system changes than the hot-leg temperatures, but several trends will be noted. RCP-2B suction temperature began diverging

from the rest of the primary temperatures at about 0540 hrs, either shortly before, or at the time of, the A loop RCP shutdown. RCP-2A suction temperature did not begin diverging until 0600 hrs.

Both RCP-2A and RCP-2B suction temperatures began a rapid decrease in temperature at 0720 hrs, the time that ES was manually initiated by the operator. There was a rise in both pump suction temperatures when MUP-1C was shut off at 0737 hrs. However, RCP-2B suction temperature rose only 30°F while RCP-2A suction temperature increased about 180°F. The pump suction temperature had decreased to around 200°F by 0900 hrs, and, in general, remained around this level until about 1430 hrs.

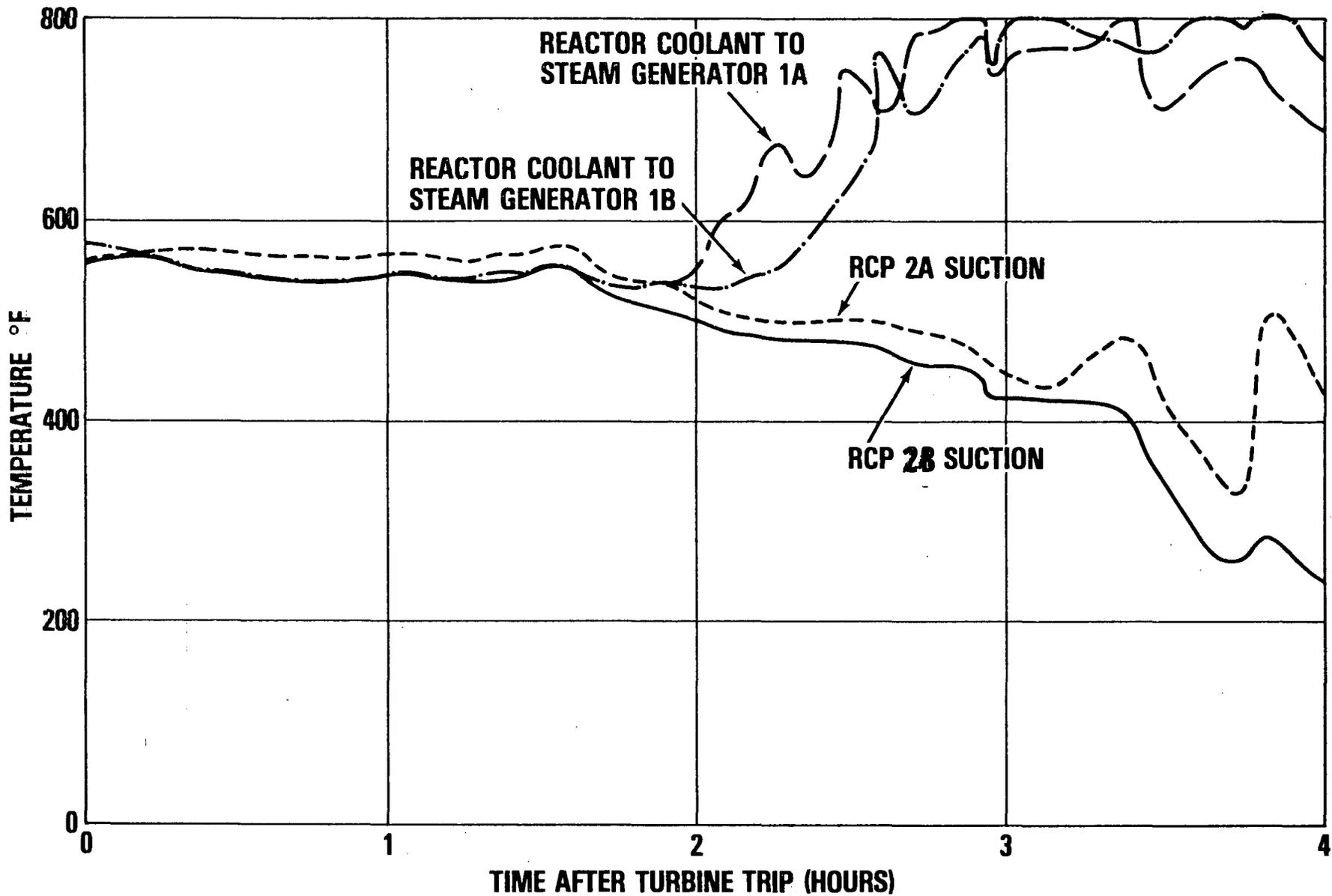
At 1430 hrs, the RCP-2A pump suction temperature began a long trend upward, increasing from 170°F to 490°F by 12 hours 10 minutes. This trend was interrupted at 1510 hrs, by a 200°F rise over a several minute period, that corresponds to a simultaneous pressurizer level drop of nearly 200 inches. A large decrease in the RCP-2A suction temperature occurred after 1723 hrs, when the operators turned on MUP-1C to begin repressurizing the reactor. A final sharp drop in this temperature occurred when the RCP-2A was bumped at 1933 hrs.

TABLE I.4-5
STEAM GENERATOR AND
PRIMARY SYSTEM TEMPERATURES

<u>Point</u>	<u>Service</u>	<u>Range</u>
1	SG 1A Outlet (Secondary)	0 - 800
2	SG 1A Outlet (Secondary)	0 - 800
3	SG 1B Outlet (Secondary)	0 - 800
4	SG 1B Outlet (Secondary)	0 - 800
5	RX Coolant to SG 1B	0 - 800
6	RX Coolant to SG 1A	0 - 800
7	SPARE	
8	RC Pump 1B Suction	0 - 800
9	RC Pump 2B Suction	0 - 800
10	RC Pump 1A Suction	0 - 800
11	RC Pump 2A Suction	0 - 800
12	DH Coolant from RX DH Pump A	0 - 400
13	DH Coolant from RX DH Pump B	0 - 400
14	DH Coolant from DH Cooler A	0 - 400
15	DH Coolant from DH Cooler B	0 - 400
16	FW to SG 1A	0 - 800
17	FW to SG 1B	0 - 800
18	SPARE	
19	Letdown Cooler Outlet A	0 - 400
20	Letdown Cooler Outlet B	0 - 400
21	Letdown Cooler Inlet	0 - 800
22	SPARE	
23	RX Coolant Makeup Tank	0 - 400*
24	Mechanical Draft Cooling Tower	Not Listed*

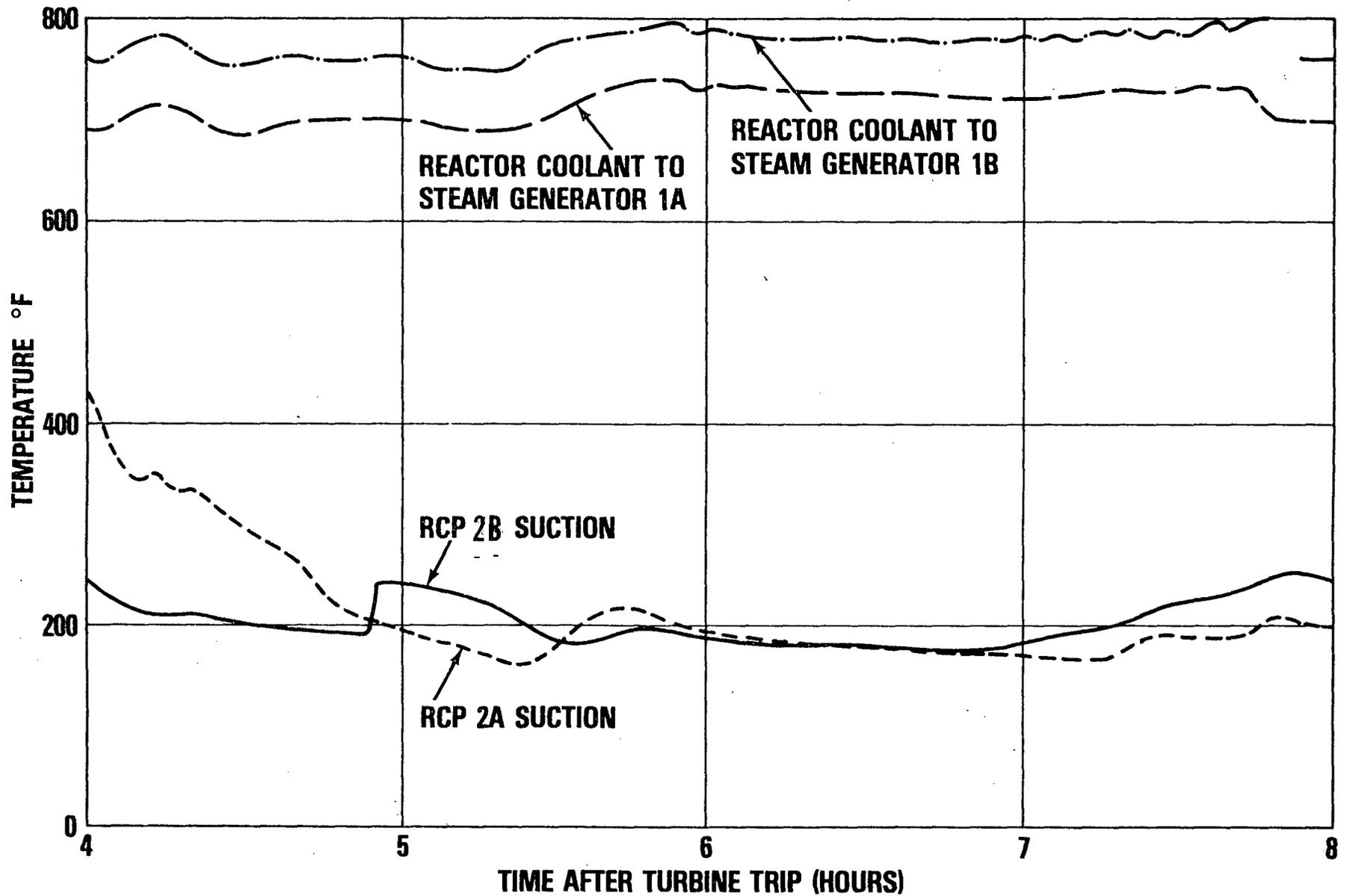
*These are currently printing 0 - Operator says that they are not used.

I-4-68



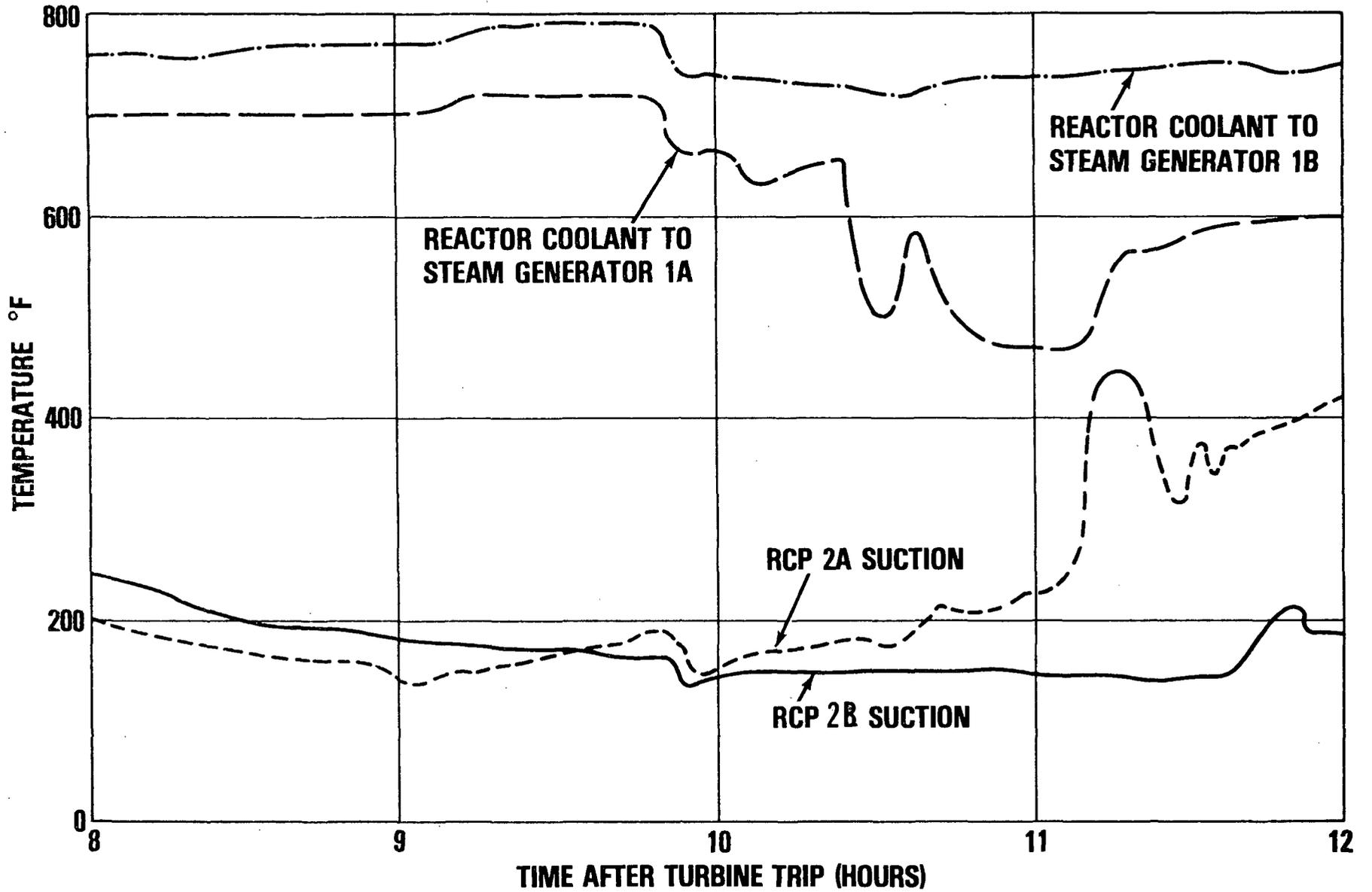
**SELECTED PRIMARY SYSTEM TEMPERATURES FOLLOWING ACCIDENT
FIG 1.4-12A**

I-4-69



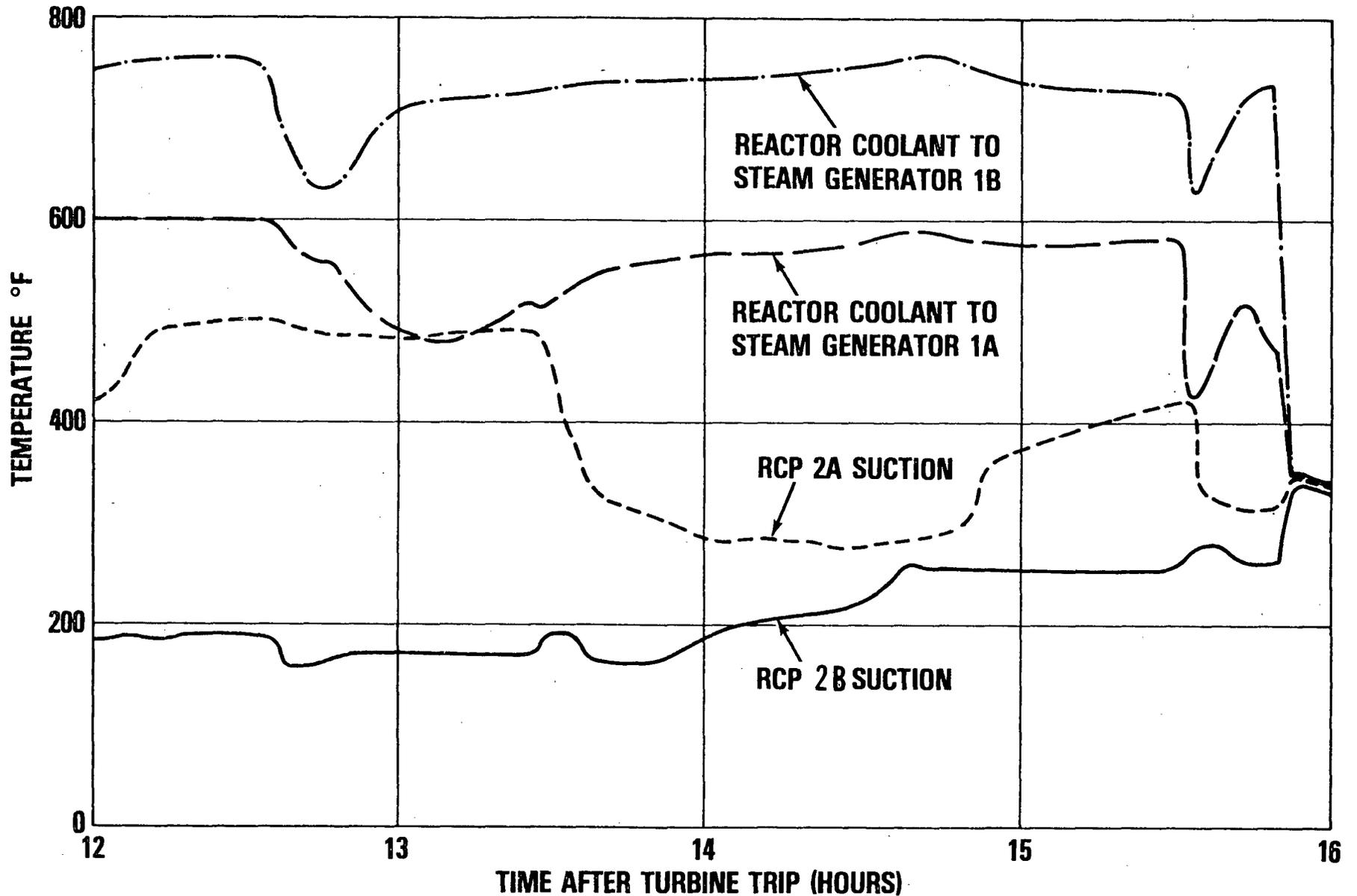
**SELECTED PRIMARY SYSTEM TEMPERATURES FOLLOWING ACCIDENT
1.4-12B**

I-4-70



**SELECTED PRIMARY SYSTEM TEMPERATURES FOLLOWING ACCIDENT
1.4-12C**

I-4-71



**SELECTED PRIMARY SYSTEM TEMPERATURES FOLLOWING ACCIDENT
I.4-12 D**

4.16 Early Void Formation

The Operational Sequence of Events, Appendix I-A to this report, will identify the conclusion of the investigation team that early formation of voids appears to occur in the reactor coolant system during the 1 to 4-minute time frame. No assertion is made as to the nature of these voids because objective evidence is unavailable to distinguish whether they are distributed voids in the circulating reactor coolant or discrete voids forming in the reactor vessel or upper portions of the reactor hot-legs ("candy canes").

This conclusion was reached by doing a mass balance on the reactor coolant system as indicated by RCS Temperature, pressure and recorded pressurizer level at selected times during the 1 to 4 minute time interval. Using the data at 1 minute as the basis for comparison, this mass balance would indicate the mass that would have to be added to the RCS, assuming no voids other than in the pressurizer, to give the indicated levels. This required mass addition was then compared to the upper estimate of the mass flow rates being added to the core as a result of operator and automatic actions as determined from interviews and objective evidence.

The results are shown in Table I.4-6 and suggest a mass discrepancy of approximately 4,000 pounds. Thus the indicated mass is greater than the calculated maximum possible mass by this amount. If one includes the probable mass loss through the EMOV during this time (if the EMOV was fully open which is indeterminate), the calculated mass discrepancy would increase to not more than 9,200 pounds.

It was therefore concluded, because of the magnitude of the discrepancy, that the system was not, in fact, solid and was already experiencing significant void formation of an indeterminate nature.

Some additional support for this conclusion comes from the fact that the observed RCS heatup rate during this period is low when compared to that expected for the decay heat load at this time after shutdown and the makeup

TABLE I.4-6

PRESSURIZER LEVEL RESPONSE DURING 1-4
MINUTES

<u>Time</u>	<u>Makeup Pump Letdown Status</u>	<u>Press' er Level</u>	<u>RCS Liquid Volume (Ft3)</u>	<u>RCS Pressure</u>	<u>RCS Tav</u>	<u>RCS Mass (lb)</u>	<u>Apparent Mass Addition*</u>	<u>Possible Mass Addition*</u>
60 sec	1A & 1B on - max Flow N420 gpm (flow thru 16B & MU Line only or letdown secured)	165"	10,938	1800 psig	576°F	489,000	---	---
122 sec	1A&1C full on (ECCS) - 1000 gpm; letdown secured.	210"	11,095	1720 psig	578°F	494,000	5,000	3,600
195 sec	1A & 1C on Operator; Starting to throttle (less than 1000 gpm); letdown secured	290	11,346	1600 psig	578°F	503,000	9,000	10,100
278 sec	1C off; 1A throttled (less than 500 gpm)	370"	11,656	1450 psig	579°F	518,000	15,000	11,500

*This does not include mass loss out of EMOV during this period, which, if fully open, should be approximately 1500 lb/minute.

flow rates. This temperature discrepancy during the same period could be accounted for by the energy required to achieve a change in state of 4000-9000 pounds of water over a 3-minute time period.

The apparent stabilization of RCS pressure at 1750 psig (although not confirmed by all the pressure monitoring channels) would also suggest flashing or void formation resulting in momentary maintenance of RCS pressure.

4.17 Diesel Generator Inoperability

At 0402 hrs, the first automatic ES actuation was experienced. Diesel generators DF-X-1A and DF-X-1B started, but did not load, since the emergency buses continued to be powered from offsite sources. (Int. 3, 15, 189, 195)

At 0430 hrs, the diesel generator fuel racks were manually tripped, by an operator dispatched for that purpose, after the diesels had operated unloaded for 28 minutes. (Int. 189, 195; Ref. 14)

NOTE: The following is extracted from Unit 2 Operating Procedure 2104-6.2, "Emergency Diesels and Auxiliaries," Revision 9:

"2.1.1.4 Do not run the Diesel Generators unloaded for extended periods. Unload operation causes carbon deposits in the exhaust system which cause exhaust system overheating as the carbon burns off when full load is resumed. Unloaded operation for periods of one half hour can be considered completely safe if they are followed by a period of operation under load."

"4.10.6 The Unit can be shutdown after the ES Actuation has been cleared. Safety Injection - Bypassed. R.B. Isolation & Cooling - Defeated."

NOTE: Following an automatic diesel start on an ESF actuation, diesels can only be shut down locally by an operator tripping the fuel racks. (Int. 189, 195; Ref. 148)

At about 0930 hrs, an engineer noted the energized overspeed trip alarm (indicating the fuel racks were still tripped) and questioned a Shift Foreman relative to the status of the diesel generators. In that condition, the diesel generators would not start automatically on ES actuation, loss of offsite power or manual start initiation. It was agreed to (1) place the Unit 2 Control Room Emergency Standby/Maintenance Exercise switch for each diesel in the Maintenance Exercise position (thereby defeating its automatic start capability) and (2) to reset the diesel fuel racks (thereby regaining manual start capability). An operator was dispatched to reset these fuel racks, which occurred at 0949 hrs. From this point on, the diesel generators could only be started manually. These actions reportedly solved the Shift Foreman's concern for unnecessary exposures of operators to high radiation fields. These exposures might be necessary to shut down diesels that start unnecessarily and run unloaded, should subsequent ES actuations occur with offsite power available. (Int. 195, 200; Ref. 14, 148)

NOTE: The following is extracted from Unit 2 Operating Procedure 2104-6.2, Revision 9.

"2.2.3 The requirements of Technical Specifications 3.8.1.1 or 3.8.1.2 must be met depending upon the mode of operation. In modes 1, 2, 3 and 4, two separate and independent diesel generators each with: A separate day fuel tank containing a minimum volume of 500 gallons of fuel, a fuel storage system containing a minimum 19,000 gallons of fuel in DF-T-2A and 19,000 gallons of fuel in DF-T-2B, and a separate fuel transfer pump must be operable. In Modes 5 and 6, one diesel generator and its associated fuel oil system must (sic) be operable. If the above requirements can not be met, proceed as outlined in the applicable Action Statement."

"4.10 Diesel Generator - Automatic Start upon Engineered Safety Features Actuation.

NOTE: Unit must be in Emergency Standby and reset for auto start to occur."

With the exception of the ES actuation at 0402 hrs, no subsequent ES actuation (0720 hrs, 0756 hrs, 0807 hrs, 0924 hrs, 1350 hrs, 1933 hrs, and 1950 hrs) shows evidence of a diesel generator start. This evidence would be in the form of indications of diesel generator room air compressor starts and stops, diesel generator oil pressure alarms and indications of diesel generator shutdowns in accordance with operating procedure 2104-6.2, Revision 9.

Evaluation

Operating Procedure 2104-6.2, "Emergency Diesels and Auxiliaries," Revision 9, requires that the diesels be operable while in Hot Standby; and that if that requirement can not be met, then a degraded mode of operation is allowed. Licensee operators willfully defeated the automatic start function of the emergency diesel generators, when loss of site power was still possible and power for engineered safety features was still needed. Since the diesels could have been operable, had the operators not taken the actions described, this is under consideration as a potential item of noncompliance.

4.18. Reactor Coolant Makeup Pump Starting Problems

During the course of this accident, the operators reportedly experienced multiple problems with starting reactor coolant makeup pumps (MUPs).

At 13 seconds into the event, the computer alarm status printout indicates an operator unsuccessfully attempted a start of MUP-1A. At 39 seconds into the event, the alarm status printout indicates an operator again tried to start MUP-1A and was successful. This sequence of events is confirmed by operator interviews, which further indicate that more than one unsuccessful attempt to start MUP-1A was experienced. A review of the computer generated sequence of events for the period of interest appears to show a condition of contact bounce related to the control switch disagreement relay. Assuming

that the Burns & Roe Drawings for this pump and its related components represent actual wiring, that computer indications of changes in electrical contact position are good, and that the memory of two separate operators is correct; focuses attention on the control room switch for MUP-1A, whose malfunction alone could defeat a manual or automatic start, fail to indicate additional operator start attempts, and provide the apparent anomalous indications to the sequence of event program.

At 12 minutes into the event, the alarm status printout and the sequence of events review present similar indications of a start failure followed by a start success. Operator interviews confirm an event like this occurred, but their memory of timing does not agree with the computer listing.

At 4 hours 18 minutes into the event, the alarm status printout and the sequence of events review present an additional example of these indications; with the exception that, at these times, the pump does not start. Operator interviews confirm these events and indicate the control room switch was then placed in the pull-to-lock position, thereby defeating MUP-1A capability to start automatically and enabling MUP-1B to take its place. The alarm status printout and sequence of event review indications on the following automatic ES actuation, 20 seconds later, supports that testimony.

MUP-1A was left in pull-to-lock for the rest of the 16 hours reviewed by the investigation team. MUP-1A has reportedly been run since the event, indicating the problem may be intermittent.

At 15 hours 33 minutes into the event, the alarm status printout indicates an operator was attempting to start MUP-1C as an automatic ES actuation occurred. The pump did not start; but the alarm status printout indicates it was manually started by an operator 6 seconds later. The failure of MUP-1C to start during this ES actuation may be indicative of an example of a similar malfunction of the same type switch on another pump; but could also be related to the malfunction of several other components, such as a low lube oil pressure switch or one of several relays, which are not bypassed by the ES actuation signal.

Evaluation

These switches are replacements for switches of the same type and manufacturer, of earlier vintage, that had experienced contact-cam-follower failures. These switches are standard items used throughout the plant, but which are uniquely modified for the MUP application above. (Int. 61, 195; Ref: 11, 136)

The information developed by the investigation team relative to this problem has been passed to the licensee and NRC management to ensure the ultimate resolution of the problem.

During discussion with the licensee, the licensee indicated the problems with starting MUP-1A had been determined. The licensee investigation indicates the ammeter for MUP-1A was found stuck downscale and that the MUP had, in fact, started. The operator had seen the ammeter downscale and had secured the MUP.

Analysis results of computer Alarm Typer records from March 28, 1979, challenge the credibility of the licensee's determination. The Alarm Typer record indicates the MUP-1A power supply breaker did not close at 04:00:05 and 08:18:53 during operator attempts to start the MUP. Clearly, if the breaker does not close, the ammeter will not register current flow. It appears the licensee has not completed an adequate investigation of this matter.

4.19 Operational Usefulness of Computer Alarm Status Printout

The usefulness of the alarm status printout to the operators during this accident transient appears limited. This resulted from the computer typewriter combination inability to keep up with real time and the inaccuracies and misinformation that the output contained.

4.19.1 Timeliness

At the beginning of this accident, the Alarm Typer output was in close agreement with real time. By 0646 hrs, the Utility Typer, then outputting the alarm status printout, was running 1 hour and 33 minutes behind. At 0648 hrs, the Utility Typer output of the alarm status printout was brought in synchronization with real time by equipment failure or operator action. By 1315 hrs, the Alarm Typer, then outputting the alarm status printout, was running 2 hours and 39 minutes behind. Had the alarm status printer memory not been lost at 0648 hrs, the lag in event information would have been in excess of 4 hours, since the lags are cumulative. The information presented in the alarm status printout is useful as an historical record, but lacks timeliness for operators to utilize it for current decisions. During interviews, the operators stated that they would use the alarm suppress function to wipe out historical data, if they needed real-time alarm information for current decisions. (Int. 189; Ref: 14, 136)

4.19.2 Inaccuracies

Analog instruments are scanned by the computers at various frequencies, varying from once per second to once per minute, depending on group assignment and, supposedly, importance or expected rapidity of output change. The results of this scan are stored in assigned memory locations. Computed values are recalculated using data currently stored in memory locations every 15 or 30 seconds, again, depending on group assignment. The resulting digitized analog values and the host of open or closed contacts are then scanned once per second for alarm conditions and changes of contact state. The resulting inaccuracies are ones of association. Alarm conditions and corresponding analog values are recorded at indicated times that are long past the time when the transient parameter was actually at that point. Further, the sequential nature of the alarm status printout implies event sequence, when near concurrent events may not have actually occurred in the order presented due to the low frequency of the alarm scan. These errors are not apparent in slow transients but occurred frequently during this accident.

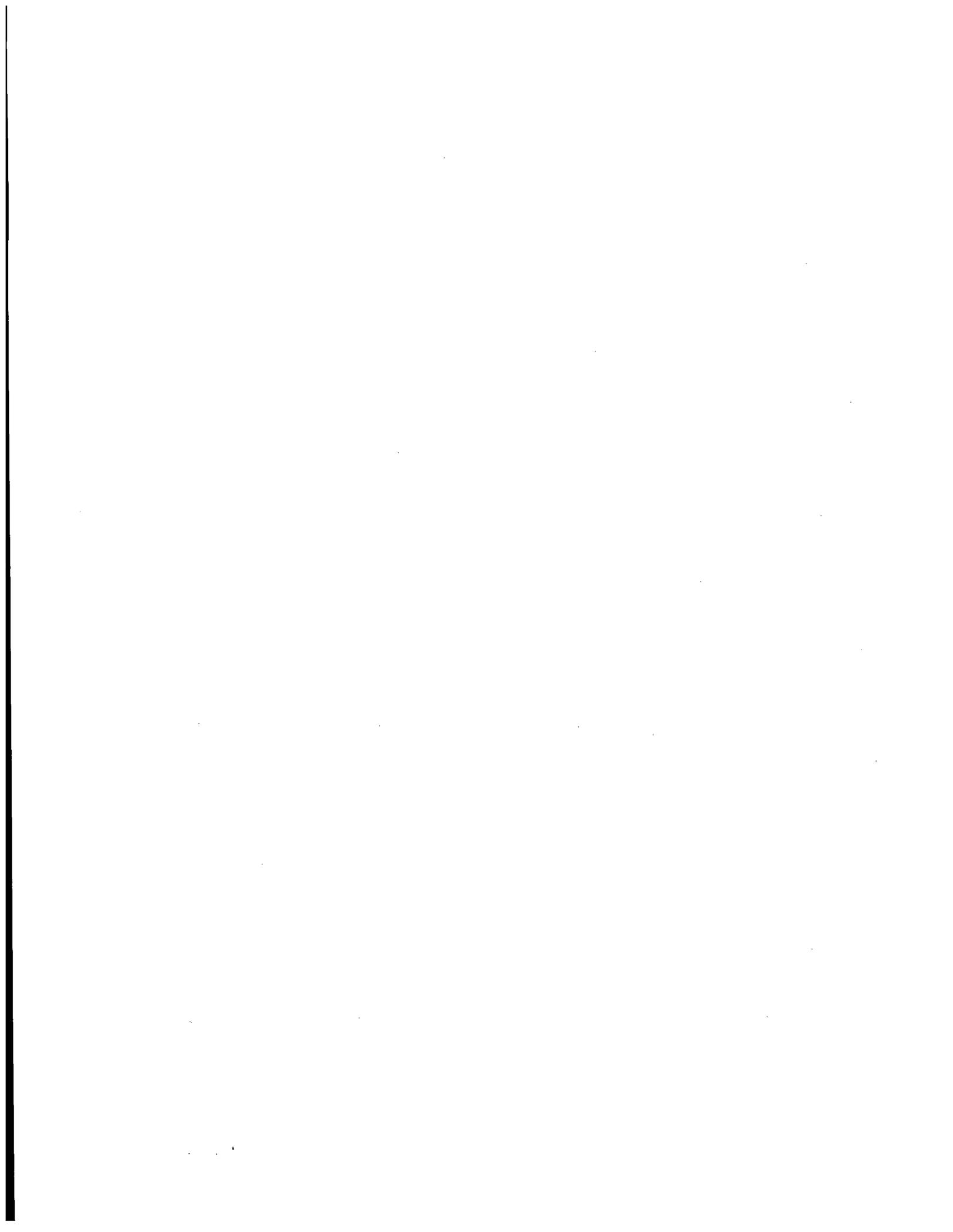
4.19.3 Misinformation

In researching anomalous indications in the alarm status printer output, it became apparent that certain indicated events could not have happened, could not have happened in the indicated sequence, or should have been indicated but were not. Some of the problems indicated in Table I.4-7 were known by licensee personnel; but the majority were not, having only recently been confirmed by the licensee, following inquiries by the investigation team.

TABLE I.4-7
Computer Output Errors

<u>Computer Point Designation</u>	<u>Point Descriptor</u>	<u>Associated Problem</u>
3092 3093 3094	Condensate Booster Pump CO-P-2A Condensate Booster Pump CO-P-2B Condensate Booster Pump CO-P-2C	Pump "TRIP" information is available only in automatic or while starting in manual.
3156	Diesel DF-X-1A Lubeoil Press. Low	"NORM" indications means Diesel DF-X-1B is tripped.*
3159	Diesel Gen DF-X-1B Fault	Associated with Diesel DF-X-1A Lube Oil Pressure Switches. Why it indicates at all during this transient is unclear, since its twin, Diesel DF-X-1B Lube Oil Pressure low alarm, is never received.
3172 3173 3174	RC Makeup Pump 2A Tripped RC Makeup Pump 2B Tripped RC Makeup Pump 2C Tripped	Contact meaning reversed. "NORM" means tripped. Point 3173 is no good.
3213	RC Pump 2A Off	With the pump running, the contact appears to make and break at high frequency. The meaning of this phenomenon has not been determined.

*NOTE: Besides the software and hardware errors indicated here, the actual contact arrangement does not agree with Burns and Roe drawings, either.



DETAILS I

REFERENCE LIST

1. Plant Charts & Computer Demand Logs for 0300 hrs & 0400 hrs on March 28, 1979
2. Statement by Plant Manager. Written testimony prepared by the Plant Manager for and presented to the Presidents Commission. The majority of the testimony is a verbatim copy of notes generated from a meeting of managers shortly after the accident.
3. TMI Phone Bill, March 28, 1979.
4. Steam System Operating Procedure 2106-2.2, Condensate Polishing System, Revision 9
5. (duplicate - See #44)
6. TMI Reactor Trip Report January 15, 1979 (duplicate - See #48)
7. Maintenance Work Request, (WR) CO 711, Megger Checks for Water/Heat Damage on Emergency Feedwater Pumps EF-P2A/B (duplicate - See #52)
8. WR CO 718, Reactor Building penetration checks, January 15, 1979
9. Field Change Request, FCR, 2329.1
10. PO No. JCP 0C211, February 11, 1976, Purchase Order for Spare Liquid Waste Tank Rupture Discs (duplicate - See #49)
11. B&R Drawing No. 2485, Revision 9, Radwaste Disposal-Miscellaneous Liquid Auxiliary Building Sump Tank & Section & B&R Drawing No. 2482, Revision 13
12. Special Operating Procedure, SOP, No. R-2-79-31, June 29, 1979, Feedwater Latching Functional Test
13. Plant Strip Charts and Multipoint Recorder. Also see discussion of EMOV status in Section 4.8
14. Alarm Printer. Same as Alarm Typer. A computer output terminal which lists alarm status.
15. Core Flood System Description, Index No. 2813, and Technical Specifications

16. TMI-2 Instrument Calibration Data Sheet, Instrument No. RC3-858
17. March 29, 1976, letter Dresser Industries to Metropolitan Edison Company
18. Problem Report No. 913, Instrument Air & Service Air, Low Capacity, April 29, 1974
19. Flow Charts for Condensate Polisher Tanks 1, 2, 3, 4, 5, and 8, as well as Polisher Outflow Chart
20. Operating Log
21. Transient Logs - A collection of informal handwritten notes, documenting observations and sequential events or actions, which were generated by licensee personnel during the course of the accident. (Titled LT-0001 in licensee's files of the records of the event.)
22. T/C list A list of incore Thermocouple readings which were taken at about 0830 hours on 3/28/79 by technicians using portable equipment.
23. Plant Strip Charts. Various charts of plant analog parameters, recorded automatically and continuously by installed equipment.
24. Instrument Drawings
25. Plant Abnormal Procedure 2203-1.4, "Reactor Coolant Pump and Motor Emergency"
26. Plant Operating Procedure 2204-301B, "C5 Reactor Coolant Drain Tank Hi/Lo", Response to High RCDT Alarm
27. Plant Operating Procedure 2101-3.2, "Unit Cooldown", Step 4.13, Revision 13, March 10, 1979
28. Plant Operating Procedure 2102-1.1, "Unit Heatup", Revision 19, March 7, 1979
29. Plant Abnormal Procedure 2203-2.2, "Turbine Trip", Revision 7, October 25, 1978
30. Plant Emergency Procedure 2202-1.1, "Reactor Trip", Revision 6, October 25, 1978
31. Plant Emergency Procedure 2202-1.1, "Reactor Trip", Revision 5, October 6, 1978

32. Plant Abnormal Procedure 2203-1.1, "Loss of Boron Moderator Dilution", Revision 5, September 13, 1978
33. Surveillance Procedure 2304-3D2, "Reactor Coolant Specific Activity", Revision 0, November 8, 1977 (duplicate - See 61)
34. Plant Emergency Procedure 2202-1.3, "Loss of Reactor Coolant/Reactor Coolant System Pressure", Sections A & B, Revision 8, October 6, 1978
35. Plant Emergency Procedure 2202-1.3, "Loss of Reactor Coolant/Reactor Coolant Pressure", Revision 8, October 6, 1978
36. Plant Operating Procedure 2103-1.4, "Reactor Coolant Pump Operation", Revision 6, August 16, 1978
37. Plant Operating Procedure 2104-4.1, Miscellaneous Liquid Rad Waste Disposal, Revision 2, March 14, 1979
38. Administrative Procedure 1012, "Shift Relief and Log Entries", Revision 8, November 4, 1977
39. Administrative Procedure 1012, "Shift Relief and Log Entries", Revision 8, Section 3.5
40. Plant Emergency Procedure 2202-1.5, "Pressurizer System Failure", Revision 3, September 29, 1979
41. Operating Procedure 2104-1.2, "Makeup & Purification System, Revision 13, March 19, 1979
42. Licensee Event Report 78-22/99X and 78-21/3L, March 29, 1978 Reactor Trip
43. QC Surveillance Report No. 77-266, October 27, 1977, Mechanical Maintenance of RC-RV2, Inspection, Repair and Testing, WR 1467
44. Fourth EMOV Valve on order, Reference Number 897028-5000-1, P.O. No. 239715, August 31, 1978 (duplicate - See #5)
45. Machine History File, WR 1166, September 14, 1977, Removal of Valve Operator for Repacking
46. Temporary Change Notice 3-78-666, October 10, 1978, entered a Revision 8 to Procedure 2303-M14A/B/C

47. Performance Data Output Segments 1-6, Process Computer Program
48. TMI Reactor Trip Report, January 15, 1979, Failure of the Atmosphere Dump Valve Bellows. (duplicate - See item 6)
49. Spare Discs, P.O. No. JCP 0C211, February 11, 1976 (duplicate - See #10)
50. Burns & Roe Drawing 2005, Feedwater and Condensate, Revision 30
51. Station Preventative Maintenance Procedure E-2, "Dielectric Check of Insulation, Motors & Cables", Revision 1, January 6, 1978 (duplicate - See item 97)
52. Machine History Card, January 23, 1979 (see WR C0711 Item No. 7) (duplicate - See item 7)
53. Plant Operating Procedure 2103-1.3, "Pressurizer Operation", Step 4.2.4, Revision 1, November 1, 1977 (duplicate - see item 62)
54. Region I Notes: A series of notes compiled by the Reactor Operations and Nuclear Support Branch Chief of Region I in the early days of the accident.
55. Administrative Procedure 1002, "Rules for the Protection of Employees Working on Electrical & Mechanical Apparatus", Revision 9, February 23, 1977 (duplicate - see #98)
56. Plant Emergency Procedure 2202-2.2, "Loss of Steam Generator Feed", Turbine Trip/Abnormal Procedure, Revision 3, October 13, 1978
57. Plant Operating Procedure 2102-3.3, "Decay Heat Removal Via OSTG", Revision 6, April 17, 1978
58. Plant Operating Procedure 2105-1.4, "Integrated Control System", Revision 3, September 1, 1978
59. Surveillance Procedure 2304-W1, "Borated Water Source Concentration Verification", Revision 1, March 14, 1979
60. 2304-3D1, "RCS Chemistry", Revision 2, February 5, 1979
61. Surveillance Procedure 2304-3D2, "RCS Specific Activity", Revision 0, November 8, 1977 (duplicate - see #33)

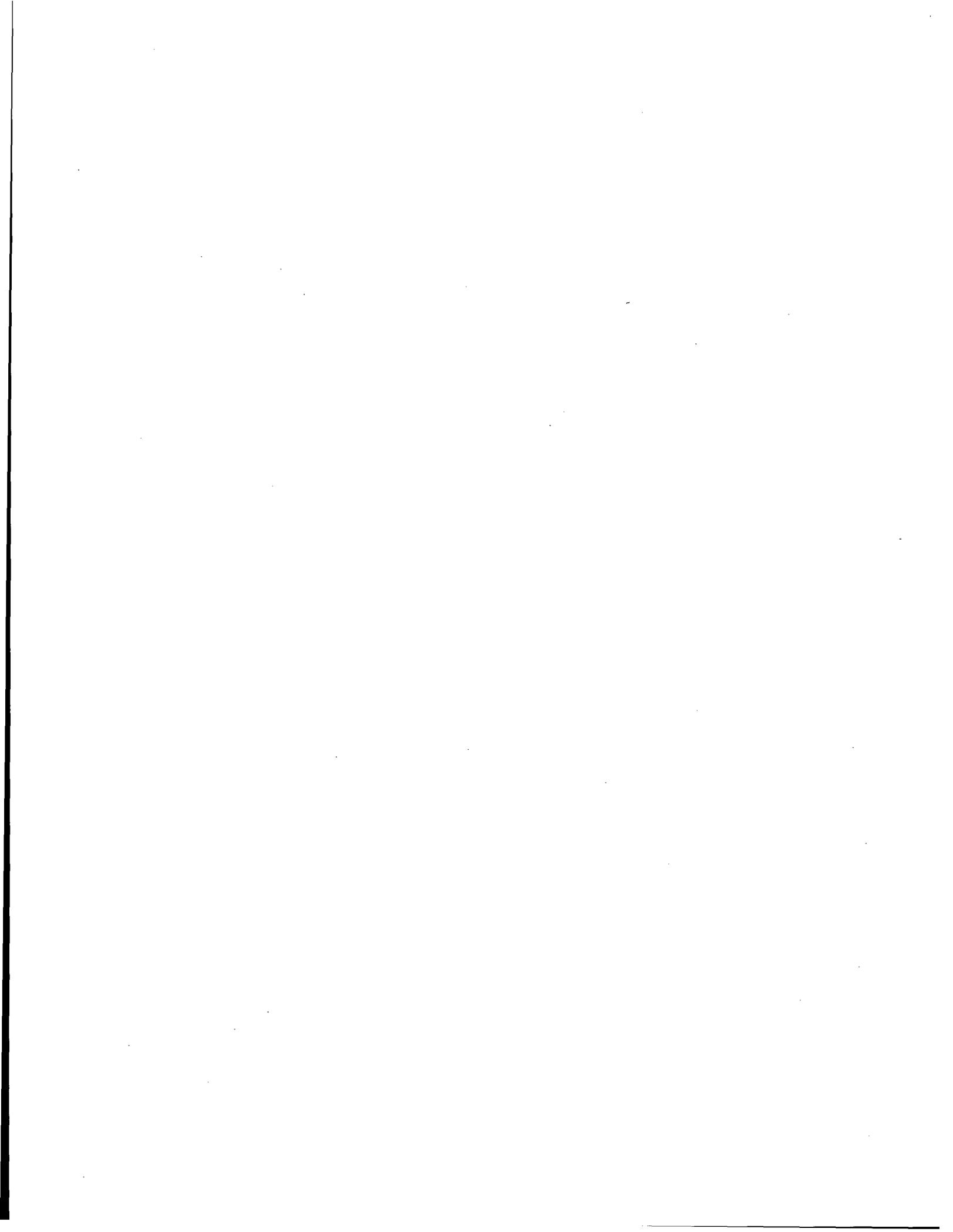
62. Plant Operating Procedure 2103-1.3, "Pressurizer Operation", Revision 3, July 19, 1978
63. Surveillance Procedure 2301-3D1, "RCS Inventory", Revision 3, February 5, 1978
64. Plant Emergency Procedure 2202-1.5, "Pressurizer System Failure", Revision 3, September 29, 1978
65. Surveillance Procedure 2303-M24A/B, "Reactor Building Spray Pump Functional Test and Valve Operability Test", Revision 7, October 6, 1978, Steps 6.3 & 6.4
66. Surveillance Procedure 2303-M27A/B, "Motor Driven Emergency Feedwater Functional Test & Valve Operability Test", Performed January 3, 1979, February 26, 1979, and March 26, 1979, Revision 4, August 30, 1978
67. Temporary Change Notice 2-78-666, October 10, 1978
68. Same as 47
69. Surveillance Procedure 2303-M14A/B, "Emergency Feedwater System Valve Lineup Verification & Operability Test & Turbine Driven Emergency Feed Pump Operability Test", Performed January 3, 1979, February 26, 1979, and March 26, 1979, Revision 8, November 21, 1978
70. Surveillance Procedure 2303-M1A/B, "Makeup Pump & Valve Functional Test", Revision 5, September 22, 1978 (duplicate - See item 93)
71. Surveillance Procedure 2303-M2A/B, "Decay Heat Removal Pump Functional Test", Revision 7, June 20, 1978
72. Burns & Roe Pipe and Instrument Drawing (P & ID) Drawing No. 2024, Reactor Coolant Make-Up & Purification, Revision 25
73. M. Kellog Isometric Drawings No. 223-1 and 2
74. Plant Training Photograph No. 443 (Thermocouple on EMOV line located 3 feet downstream of valve)
75. Plant Training Photograph No. 81
76. Plant Emergency Procedure 2202 1.1, "Reactor Trip", Step 1.1.e, Revision 6, October 25, 1978

77. Shift Relief & Log Entries, Revision 8, October 4, 1977
78. Plant Abnormal Procedure 2203-1.4, "RCP & Motor Emergencies, Revision 3, May 4, 1978
79. Plant Operating Procedure 2105-1.3, "Safety Features Actuation System", Revision 2, October 25, 1978
80. Plant Emergency Procedure 2202-1.3, "Loss of Reactor Coolant/Reactor Coolant System Pressure, Revision 11, October 6, 1978
81. Plant Operating Procedure 2103-1.3, "Pressurizer Operation", Revision 3, July 19, 1978
82. Administrative Procedure 1670.2, Site Emergency Procedure, Figure 4, Revision 9, November 22, 1978 (duplicate of 108)
83. Administrative Procedure 1014, "Recall of Standby Personnel to Plant", Revision 3, June 21, 1977
84. Makeup and Purification, Step 2.2.1
85. Pressure Operation, Step 2.1.8
86. Operating Procedure 2103-1.4, "Reactor Coolant Pump Operation, Step 2.1.1.1, Revision A, April 18, 1978
87. Emergency Procedure 2202-1.3, "Loss of Reactor Coolant/Reactor Coolant System Pressure", Step A.3.2.7, Revision E, May 12, 1978
88. Technical Specifications; Appendix A to the Operating License for Unit 2.
89. Procedure Step 6.1.3, "Addresses the RCP Vibraswitch Alarm"
90. Operating Procedure 2101-1.1, "Nuclear Plant Limits & Precautions", page 27, Revision 4, August 8, 1978
91. Operating Procedure 2103-1.4, "Reactor Coolant Pump Operation", Revision 6, August 16, 1978
92. Operating Procedure 2104-6.2, "Emergency Diesels and Auxiliaries", Revision 9, March 14, 1979
93. Surveillance Procedure 2303-MIA/B, Make-Up Pump & Valve Functional Test, Revision 5, September 22, 1978 (duplicate - See item 70)

94. Administrative Procedure 1001, "Document Control", Revision 18, March 21, 1978
95. Administrative Procedure 1007, "Control of Records", Revision 4, July 14, 1978
96. Engineering Change Memo, ECM No. S-5934, Electromatic Valve, RC-RV2, Signal Light, Result of March 29, 1979 Open Electromatic Valve (EMOV).
97. Station Preventive Maintenance Procedure E-2, "Dielectric Check of Insulation, Motors and Cables" (duplicate - See item 51)
98. SPMP, Para. 4, "Tagging" (duplicate - See item 55)
99. Administrative Procedure 1026, "Corrective Maintenance & Machinery History", Revision 8, November 29, 1978
100. Administrative Procedure 1021, "Plant Modification", Revision 0, July 22, 1977
101. Corrective Maintenance Procedure 1407-1, "Station Corrective Maintenance Procedure", Revision 0, November 29, 1978
102. B&R Drawing No. 2006, "Flow Diagram Makeup Waste Treatment & Condensate Polishing", Revision 24
103. "Operation Quality Control Surveillance Report", No. 77-266, "Operation Quality Assurance" October 26, 1977
104. Plant Abnormal Procedure 2203-2.2 "Turbine Trip", Revision 7, October 29, 1978
105. Administrative Procedure 1037, "Control of Caution and DNO Tags", Revision 1, August 23, 1978
106. Plant Operating Procedure 2102-1.1, "Unit Heatup", Revision 19, March 7, 1979
107. Plant Operating Procedure 2102-3.2, "Unit Cooldown", Revision 13, March 10, 1979
108. Emergency Plan Procedure 1670.2, Revision 9, November 22, 1978 (duplicate of 82)
109. The Maintenance, "Work Request Log, Unit 2", starting with 1979 input.

110. Station Corrective Maintenance Procedure 1407-1, Revision 1, February 2, 1979
111. Unit 2 Machine History Files (selected sections), May 8, 1979.
112. Unit 2 Valve List 2075, two volumes.
113. See 102
114. Shift Maintenance Log, March 27-28, 1979
115. "Instructions for Operating and Repairing Consolidated Electromatic Relief Valve", B&W Instruction Book No. 620-0006:01-00400.
116. See 43
117. Machine History Card, Mechanical, RC-RV2 and RC-V2, October 26, 1974 and September 29, 1975.
118. Machine History Card (Unit 1), RC-RV2 and RC-V2, October 26, 1974 and September 29, 1975.
119. Letter, B&W to GPU, Dresser Valve RC-V2, September 2, 1975.
120. Letter, GPU, to UE&C, Concerning Dresser Valve RC V2.
121. Drawing No. 3079, Sheet 14, Revision 13.
122. Megger Test, January 23, 1979, Control Building East, EF-P-2A.
123. UE&C Equipment Data Sheet, Mechanical, July 28, 1977, Ingersoll Rand, 470 gpm at 2510 feet and 3560 rpm, discharge pressure 1500 psi, 100°F (EF-P-2A).
124. UE&C Equipment Data Sheet, Motors, July 28, 1977, EF-P-2A, July 28, 1977; EF-P-2B, July 27, 1977.
125. UE&C Equipment Data Sheet, Mechanical, July 27, 1977, discharge pressure 1470 psi, EF-P-2B.
126. UE&C Equipment Data Sheet, Motors, July 27, 1977, EF-P-2B.
127. UE&C Equipment Data Sheet, Vibration Record, July 27, 1977, EF-P-2B.
128. UE&C Motor Bumping Check List, July 27, 1977, EF-P-2B.

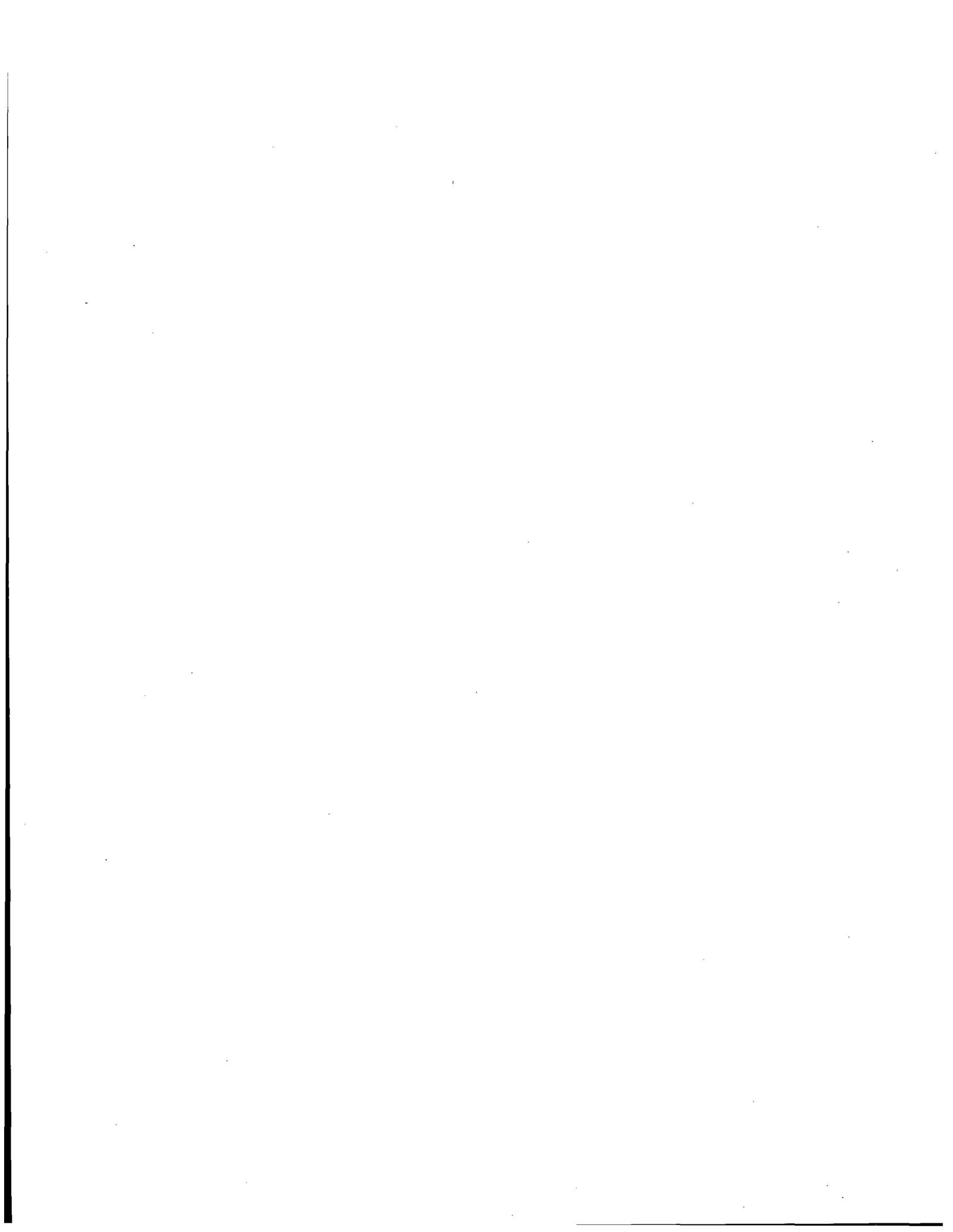
129. Manual 31700, "Consolidated Closed Bonnet Maxiflow Safety Valve", B&W Instruction Book No. 620-0006, 01-003-00 (Dresser Valves).
130. WR No. 340, March 25, 1977, w/Procedure, "Hot N₂ Gas Testing of Pressurizer Code Relief Valves".
131. Sequence of Events; Appendices I-A and II-A to this report
132. 2103-1.9, "Reactivity Balance Calculations"
133. 2302-M1A/B, Makeup Pump & Valve Functional Test, Revision 5, September 22, 1978 (duplicate - See #93 & #70)
134. LER 78-021-03L, April 29, 1979; 78-033-0IT, April 23, 1978
135. RCS and Reactor Building Pressure Charts. Plant Strip Charts; see 23.
136. Utility Typer. Computer output terminal for data specifically requested by operator. Also output terminal for alarm status printout should the Alarm Typer fail.
137. Final Safety Analysis Report
138. 10 CFR 50.46
139. Computer Logs/Daily Record Storage; March 30, 1979
140. System Description - Radwaste Disposal Gas System, October 1975
141. System Description - Nuclear Sampling System, February 1976
142. Procedure Change Request PCR-2-78-707 and 895 (duplicate - See #46)
143. Site Operations Memorandum #403 dated August 9, 1978
144. Site Operations Memorandum
145. Plant Trip Report for March 28, 1979 Trip
146. Problem Report #2847
147. Reactor Trip Report - Unit II #12, November 3, 1978
148. Burns & Roe Drawings



Appendix I-A
Operational Sequence
of
Events

Note:

This sequence of events was prepared based on elapsed time from the initiating event, the trip of the turbine, which occurred at 04:00:37 on March 28, 1979.



APPENDIX I-A
OPERATIONAL SEQUENCE OF EVENTS

<u>ITEM</u>	<u>ELAPSED TIME</u> 3/28/79	<u>EVENT DESCRIPTION</u>	<u>REFERENCES</u>
1	Prior to Turbine Trip	Three Mile Island Unit Two (TMI-2) was operating at 97% power with the Integrated Control System (ICS) in full automatic. Normal reactor coolant system (RCS) makeup, seal injection and letdown were established with makeup pump (MUP) "1B" in service. Pressurizer spray valve and heater controls were in manual to equalize pressurizer and RCS boron concentrations. RCS boron concentration was 1026 ppm and radioactivity concentration was 0.394 uc/ml. Identified RCS leakage, believed to be from the electromatic relief or code safety valves on the pressurizer, was approximately 6 gpm. RCS pressure was 2148 psig.	<ol style="list-style-type: none"> 1. Interview 58 2. Operating Logs 3/28/79 3. Surveillance Procedure 2301-3D1 4. Reactimeter 5. Operating Logs 6. GPU Sequence of 7/16/79
2		On the secondary side, once thru steam generator (OTSG) levels were at about 160 inches with pressures of about 900 psig. Steam generator feedwater pumps 1A & 1B, condensate booster pumps 2A & 2B and condensate pumps 1A & 1B were in service.	
3		Operators had been trying for about eleven hours to transfer an isolated condensate system polisher's spent resins to the regeneration receiving tank. The transfer procedure utilizes station compressed air to fluff the resin and demineralized water to transfer the resin between tanks. A resin block had developed in the transfer line.	<ol style="list-style-type: none"> 1. Interviews 2, 5, 14, 17
4		At this point, plant operators hypothesized that the water pressure, then the discharge pressure of two demineralized water pumps or the	<ol style="list-style-type: none"> 1. Interviews 166, 189

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	condensate pumps, either at approximately 160 psig, exceeded the station air pressure (approx. 100 psig), forcing water into the air system. Further, the water then made its way through a check valve (since found stuck open) and through the open cross-connect between station and instrument air systems to the polisher isolation valve controls, causing these valves shut. (This problem of water entering the air system has occurred at least twice before.)	
5	~ -1 sec.	Polisher outlet and/or inlet valves go shut.	1. Analysis (see notes)
6		Note: Condenser hotwell level dropped below the low alarm within 13 seconds (21.72"); turned and entered the normal range in 28 seconds (26.44"); and passed the high alarm in 73 seconds (37.77"). Even with this increasing level (which should initiate rejection flow), normal coastdown flow of the condensate booster pumps, and one condensate pump still running, the condensate flow was reading "0" at 2 minutes into the event. This strongly supports contention that polisher valves were shut.	1. Alarm printer 2. Hourly computer log 3. Analysis 4. Computer Analog Trend Recorder
7		Note: All polisher inlet and outlet isolation valves were found shut after the turbine trip.	1. Interview 36
8		Note: Polisher bypass valve on TMI-2 does not open automatically on high differential pressure.	1. Interview 8
9		Note: Polisher outlet pressure was reportedly so close to the condensate booster pump suction pressure trip limit that the full capacity of the condenser level control reject valve could not be used. A valve in series with the reject valve was throttled to limit reject flow which came from the polisher outlet to prevent spurious trips of the condensate booster pumps.	1. Interviews 5, 17

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
10	-1 sec.	Condensate booster pumps trip. (See notes below.)	1. Analysis
11		Note: Each main feed pump is served by its own low-suction-pressure trip switch. Since these switches are purposely set differently, to prevent loss of both pumps on minor transients and since the two pumps tripped together, within 152 milliseconds, the suction pressure transient must have been exceedingly fast. This transient could only result from the condensate booster pumps tripping at minus 1 second.	1. Utility typer 2. P&ID 3. Interview GPU-17
12		Note: Loss of suction pressure will cause a condensate booster pump trip.	1. Alarm Procedure 17.E9 2. Burns & Roe drawings
13		Note: Condensate booster pumps were subsequently found tripped.	1. Interview 109 2. See restart attempt 5 min. 15 sec.
14	-1 sec.	Condensate pump 1A trips (one condensate pump remains running). (Actual time 04:00:36, 3/28/79).	1. Alarm printer 2. Interview GPU-2 3. Interview 195
15		Note: The computer trip indication received indicates this pump tripped due to something other than control room switch operation. No interview has indicated relay flags were reset; therefore electrical faults can be ruled out. Local breaker switches require handles that were reportedly not present.	1. Burns & Roe drawings 2. Inspection 3. Discussion with operators

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
16		Note: A trip of an associated condensate booster pump with the control switch in AUTO will cause that condensate pump to trip. (Procedures require operation in AUTO, but plant problems have trained operators to normally operate in MANUAL.)	1. Alarm Procedure 17.E8 2. Interview GPU-11 3. Burns & Roe drawings
17		Note: Wiring error in trip circuit for condensate pump 1A bypassed AUTO switch contacts, making it appear to that trip circuit alone that paired condensate pump 1A and condensate booster pump 1A were in AUTO, no matter that position the switch really was in.	1. Inspection by GPU 2. Burns & Roe drawings 3. Interview 195
18	0 sec.	Main feedwater pumps are tripped resulting in an almost simultaneous trip of the turbine. (Actual time 04:00:37, 3/28/79)	1. Alarm printer 2. Interview GPU-20
19		Note: Low feedwater pump suction pressure or loss of the condensate booster pumps, while in automatic, will cause the feed pump turbines to trip.	1. Alarm Procedure 17.A15 & 17.A16 2. Burns & Roe drawings
20		Note: Both feed pumps tripping will cause the main turbine to trip.	1. Abnormal Procedure 2203-2.2 2. Burns & Roe drawings
21		Note: Tripping both feedwater pump turbines will start all three emergency feedwater pumps.	1. FSAR Section 7.7.1.2.1.3 2. Burns & Roe drawings

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
22		Note: Loss of one or more main feedwater pumps will cause the ICS to run back reactor power, causing the control rods to be driven into the core. None of the interviews conducted indicated that the operators specifically observed the runback in progress prior to the trip of the reactor at 8 sec.	1. FSAR Table 7.7-2
23	0+ sec.	All three emergency feedwater pumps start. Pressurizer level and pressure increasing rapidly. Condenser hotwell level begins rapid transient resulting in filling the hotwell to a fully flooded condition in approximately one minute.	1. Alarm printer 2. Interview 9
24	1 sec.	Turbine stop and intercept valves closed. 500 KV breakers are tripped.	1. Alarm printer
25	3 sec.	Pressure in reactor coolant drain tank (RCDDT) begins to increase.	1. Reactimeter data
26	3-6 sec.	RCS reactor pressure reaches electromatic relief valve (EMOV) opening setpoint (2255 psig).	1. Plant strip charts 2. Reactimeter data
27	8 sec.	Reactor trips from reactor high pressure (setpoint = 2355 psig). Indicated reactor pressure on the wide range RCS pressure strip chart from the control room shows an increase to approximately 2435 psig, which would normally suggest one of the two safety valves may have lifted. However, this maximum value is not confirmed by the reactimeter or by the narrow range RCS pressure recorder trace.	1. Alarm printer 2. Plant strip charts 3. Interviews 4. Reactimeter
28		Note: The reactimeter sampled data every 3 seconds and may have missed the peak. The narrow range pressure instrument did not indicate a peak high enough to trip the Reactor, which is known to occur. The wide range pressure strip chart, when compared to other pressure sensor outputs, appears to indicate about 35 psig high.	

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
29	8 sec.	Pressurizer heater groups 1-5 trip. This event is believed to be related to the operator resetting the pressurizer spray and heater controls to "Automatic," as an operator action following the turbine trip, and pressure exceeding 2125 psig. A continuous pressurizer spray had been in effect prior to the incident to eliminate a difference in boron concentration between the pressurizer and the rest of the reactor coolant system.	1. Alarm printer 2. Burns & Roe drawing
30	~ 8 sec.	Shift Supervisor announces a Unit 2 turbine trip - reactor trip.	1. Interview 5
31	9 sec.	Secondary steam pressure peaks at about 1070 psig. The main steam relief valves are reported to have lifted.	1. Reactimeter 2. Interview GPU-1 3. Plant strip charts 4. Emergency Procedure 2202-1.1
32		Note: With ICS in AUTO, turbine bypass valves (MS-V25A&B, 26A&B) would attempt to control main steam pressure at 1010 psig \pm 10 psi.	
33	~ 10 sec.	Operator verifies all control rods tripped and bottomed.	1. Interview 9 2. Emergency Procedure 2202-1.1
34	13 sec.	Letdown from reactor secured (MU-V376 'shut). Operator unsuccessfully attempts to start makeup pump 1A.	1. Alarm printer 2. Interviews 9 & 15 3. Emergency Procedure 2202-1.1

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
35		Note: These actions, according to interviews, are immediately taken by the operators, following a reactor trip, for the purpose of minimizing a large pressurizer level transient.	
36		Note: Alarm Printer output for makeup pump 1A, 1B, and 1C status (norm/trip) found to be reversed due to software error, potentially misleading operators who read printout.	1. Burns & Roe drawings 2. Analysis of alarm printer output 3. Inspection by Licensee
37		Note: Cause of start failure is indeterminate at this time but this investigation indicates that switch contact "bounce" is highly probable. Troubleshooting switchgear must await reduction in contamination levels. Operator may have believed pump has started, because of his actions, but panel indications would dispel this belief.	1. Inspection
38	~ 13 sec.	RCS pressure reaches setpoint for EMOV closure (2205 psig). However, later events will show that closure did not occur.	1. Reactimeter 2. Plant strip charts
39	14 sec.	Pressurizer heater groups 1-5 restart in AUTO at 2105 psig, decreasing.	1. Alarm printer
40		Emergency feedwater pumps reach full pressure.	1. Alarm printer
41		Note: Emergency feedwater pump discharge pressure indication is available to operator on panel, but the emergency feedwater flow must be inferred from once through steam generator (OTSG) level changes.	1. Interview 2. Inspection of panel
42		Note: During the transient, the operators reported they did have apparent flow indication in the "A" OTSG from the noise	

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	<p>("hammering") heard from the vibration and loose parts monitor system, which was on to monitor one of the steam generators. (Monitor had been left on by operators to detect OTSG relief valve operation during any future transients. This monitor was audible in the control room but was not recorded on tape.)</p>	
43	~ 15 sec.	<p>Pressurizer level reaches peak of approximately 255 inches. (The RCS pressure, average temperature, and pressurizer level appear to be tracking normally at this time.)</p> <p>PLANT STATUS</p>	<ol style="list-style-type: none"> 1. Reactimeter 2. Plant strip charts
44		<p>The Unit has just experienced a Turbine/Reactor Trip. RCS pressure and pressurizer level were decreasing rapidly after reaching their peaks. Unknown to the plant operators the electromatic relief valve (RC-RV2) was not shut and was passing reactor coolant from the steam space at the top of the reactor coolant system pressurizer. Based on control room indications, the RCS pressure and pressurizer level were trending together and decreasing as was expected after a reactor trip. The OTSG water levels were at about 90 inches and decreasing at about 4 inches/second. The OTSG steam pressures were about 1030 psig and decreasing at 4 psi per second. The turbine bypass valves were open relieving steam. The OTSG water levels had not yet reached the ICS setpoint of 30 inches for the programmed opening of the emergency feedwater valves (EF-V11A and EF-V11B) that would admit feedwater to the OTSGs.</p>	<ol style="list-style-type: none"> 1. Plant strip charts 2. Reactimeter
45	30 sec.	<p>Reactor pressure reaches low-pressure-trip setpoint (1940 psig), pressurizer relief valve exhaust pipe temperature reaches 239^oF, and</p>	<ol style="list-style-type: none"> 1. Alarm printer

<u>ITEM :</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	one code safety relief valve exhaust pipe temperature reaches 203 ⁰ F. (These temperature indications would be expected, whether the EMOV reseats or not.)	
46	38 sec.	Steam generator "A" level at 23.8 inches and decreasing.	1. Alarm printer 2. Interview 9
47		Emergency feedwater valves EF-V11A&B start to open as level decreases below 30 inches and give dual indication on panel.	1. Interview 9
48	40 sec.	Steam generator "B" level at 23.7 inches and decreasing.	1. Alarm printer 2. Interview 9
49	41 sec.	Operator starts makeup pump 1A and opens MU-V16B by operator to increase injection flow to loops.	1. Alarm printer 2. Interviews 61, 15 3. Emergency Procedure 2202-1.1
50		Operator opens DH-V5A to control makeup tank level using BWST head pressure.	
51		Note: During the interviews, the operators noted that they had several problems with keeping makeup pump 1A operating when desired. This sequence will note several trips of that pump. The last trip occurs at 4 hr. 18 min. after the start of the incident, and the pump is not restarted again for the remainder of the incident.	
52	54 sec.	Pressurizer level is at minimum of 158 inches (lowest level reached) and starting to rise.	1. Reactimeter

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
53	3/28/79	Note: Minimum level experienced was significantly higher than experienced in previous trips of the same nature.	1. Interview
54	58 sec.	Pressurizer low level alarm at 185.3" received.	1. Alarm printer
55		Note: Discrepancy between alarm printer and reactimeter data results from frequency of calculation. The computer takes differential pressure readings from each level transmitter (updated every 30 seconds) temperature compensates them and averages the results, performing this calculation once every 15 seconds. The alarm printer routine samples the most recent result of this calculation once per second and compares it to the alarm setpoint. The reactimeter routine samples a continuously calculated pressurizer level signal once every 3 seconds and therefore, can be considered closest to real-time data. In this instance, the alarm printer has just recognized level is below setpoint, even though level is rising at this point. The memory trip review confirmed the level dropped lower than the 185.3" reported at this point on the alarm printer.	1. Discussion with Computer Supervisor 2. Log typer
56	60 sec.	One pressurizer code safety valve tail pipe high temperature alarm received at 204.5 ⁰ F with RCDT pressure at 12 psig and increasing.	1. Alarm printer 2. Interviews 3. Reactimeter
57		Note: This alarm would be expected following an EMOV opening due to backflow of steam from the common tail pipe discharge header. The temperature elements for the EMOV are located at the EMOV discharge, which is approximately 40 feet from the temperature elements for the code safety valves.	1. System Description 2. Burns & Roe drawings 3. Plant training photographs

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
58	1 min.	Steam generators A and B levels at approximately 10 inches on start-up level instrumentation.	1. Reactimeter 2. Plant strip charts
59		Note: This level indication is due to weight of steam alone (dry OTSG).	1. Analysis
60		Differential temperature, hot to cold leg, rapidly approaching zero, indicating OTSG's are going dry.	1. Reactimeter 2. Analysis
61	1-4 min.	Pressurizer level rising very rapidly. The change in RCS mass inferred from pressurizer level occurring during this period cannot be accounted for under the conditions of makeup, temperature changes, and injection flow. The mass discrepancy is further increased when the mass loss through the EMOV is also considered. Moreover, reactor coolant wide range pressure shows, at 1770 psig (saturation temperature approximately 619°F), a reversal in direction and subsequent stabilization of RCS pressure for what appears to be 2-3 minutes, suggesting void formation. Moreover, the memory trip review printed from the computer shows pressure at approximately 1750 psig from 68 seconds to 113 seconds. This stabilization, however, is not confirmed by the reactimeter or narrow range pressure instrumentation.	1. Interviews 1, 14, 15 2. Calculations 3. Plant strip charts 4. Utility typer
62		Operating staff interviews covering this period disclose that the operators had felt that they had "caught" the pressurizer level decrease and that they had actually expected it to drop much further before being "turned around," based upon previous experience. Further, the Shift Supervisor noted that the rate of pressurizer level increase appeared to be wrong for the reactor trip that had occurred.	
63		Note: It appears that beginning at approximately one minute and continuing thereafter, the reactor coolant system experienced	

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	<p>either a general reduction in density, as might occur with distributed voids, or the formation of one or more discrete steam vapor voids.</p> <p>PLANT STATUS</p>	
64		RCS pressure was still dropping and pressurizer level had just turned and begun to rise. This behavior is unexpected, since RCS pressure and pressurizer level should trend together, during a loss-of-feedwater-flow transient.	<ol style="list-style-type: none"> 1. Plant strip charts 2. Reactimeter
65		<p>Note: The deviation from expected behavior was due to the formation of voids within the RCS loops and/or reactor vessel as RCS pressure plummeted. The inability of the pressurizer to effectively damp this pressure transient is attributed to a massive loss of its energy content, first by dilution during the initial power generation/removal mismatch as the reactor trip is delayed 8 seconds after the turbine trip, and relatively cool hot leg fluid is injected into the pressurizer, second by bulk energy removal as the EMOV lifts and fails to reseat, third by boiling and ejection of hot water as level and pressure drop to compensate for the RCS cooldown from the steaming OTSGs, and finally by the short failure to generate additional energy during the period when pressurizer heater groups 1-5 trip on high pressure.</p>	
66		The reactor coolant drain tank (RCDT) pressure and temperature started to increase, showing the effects of continued discharge or reactor coolant thru the EMOV. Plant operators did not associate these indications with leakage past the EMOV, but rather with the initial opening of the valve.	

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
67	1 min.	At request of Unit 2 Shift Supervisor, the Unit 1 Shift Foreman called the Station Manager to inform him of Unit 2 trip, apparently as a matter of policy and not as a result of any suspicion about the nature of the trip. The Superintendent of Technical Support - Unit 2, the on-call Duty Officer, was also called at about this time.	1. Interviews 1, 27
68	1 min.	The hourly plant log typer displayed the following information as being applicable for this time: RCS Flow: 135 MLBH Loop "A" T _h : 577 ⁰ F T _{ca} : 574 ⁰ F T _{ca} : 573 ⁰ F Loop "B" T _h : 576 ⁰ F T _{cb} : 573 ⁰ F T _{cb} : 573 ⁰ F MU Flows: 16 GPM OTSG "A" Level: 8 inches OTSG "B" Level: 5 inches Condensate Flow: 0 KLBH	1. Log typer
69	1 min. 40 sec.	Reactor coolant drain tank (RCDT) indicates 86 ⁰ F and increasing. (RCDT temperature prior to event was about 70 ⁰ F.)	1. Alarm printer 2. Utility typer
70	2 min.	Reactor coolant pressure reaches average of 1726 psig.	1. Alarm printer 2. Plant strip char

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
71		Operator requests computer print RCS loop B inlet temperature (573.6°F).	1. Utility printer
72	2 min. 2 sec.	ECCS initiation (HPI - nominal setpoint = 1600 psig). Pressure confirmed by reactor pressure instrumentation. Decay heat removal pumps start, and makeup pump 1B trips. Makeup pump 1C starts leaving makeup pumps 1A and 1C now running as HPI with discharge valves open to the 250 gpm/loop throttled position.	1. Alarm printer 2. Interview 61 3. Utility typer
73		Note: When HPI is initiated, any discharge valves (MU-V16A,B,C or D) which may have been previously throttled, are designed to return automatically to their open position. Based on interviews the valves on the 1A pump had not been manually throttled closed up to this point in the incident.	
74		Note: The trip of makeup pump 1B is appropriate action for control logic at this point.	1. System Description
75		Steam generators appear "dry" based on zero Hot to Cold leg RCS temperature difference. OTSG pressures have peaked and are dropping rapidly..	1. Reactimeter 2. Analysis of parameter trend
76	2 min. 28 sec.	Pressurizer level at 225.7 (207.7)* inches and increasing.	1. Reactimeter 2. (Alarm printer)*
77		Shift Foreman enters control room and obtains the emergency procedures for turbine trip, reactor trip, and safeguards actuation, to confirm that all appropriate actions occurred.	1. Interviews 14, 15
78	3 min. 13 sec.	ES bypassed by operator actions. Operator verifies all ECCS equipment has started normally.	1. Alarm printer 2. Interviews 14, 15

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
79		Note: Bypass of the ES does not change any ECCS equipment status, but is required to allow operator to control plant equipment.	
80		Note: Operating procedure allows ES Bypass if RCS pressure and Pressurizer level have recovered. Pressure had not recovered at this time.	1. Operation Procedure 2202-1.3
81		Note: Operator takes control of MU-V16 valves and throttles them as pressure drops, to prevent MUP runout. (Runout will cause cavitation and; potentially, pump failure.)	1. Interview 80
82	~ 3 1/2 min.	Pressurizer level in high alarm at 266.1". RCDT temperature 127 ⁰ F, tank pressure rapidly increasing and stabilizes at approximately 120 psig for 3 minutes. The reactimeter plot of RCDT pressure exhibits an oscillatory behavior at 120 psig, similar to what would be expected if the RCDT safety valve had actuated. The setpoint for the RCDT safety valve is listed as 150 psig. The actual safety valve setpoint is not known at this time, but reactor building pressure does start increasing at about this time. The operators indicate during interviews that the RCDT cooling system (a manually initiated system) was in operation prior to the transient due to relief valve leakage.	1. Alarm printer 2. Reactimeter 3. FSAR 4. System Description 5. Interview 15 6. Plant strip charts
83	~ 4 min.	Operator throttles MU-V16 valves in attempt to control pressurizer level increase.	1. Interview 15
84		Note: Operator training emphasizes avoidance of "solid" operations.	1. Interview 80
85	4 min. 38 sec.	After MU-V16C&D are fully closed operator trips makeup pump 1C. Makeup pump 1A still running with MU-B16A&B in throttled condition.	1. Alarm printer 2. Interview 15

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
86	3/28/79	Note: Based on interviews, operators stated that when the flow from a makeup pump operating in the High Pressure Injection mode is throttled, the operators do not reduce flow below approximately 100 gpm in order to protect the pump. This minimum flow is accomplished by shutting the valve in one of the two injection paths and throttling the flow in the remaining path to 100 gpm. Based on BWST level changes during the early hours of the incident, this assertion cannot be supported.	1. Interview 15
87	4 min. 52 sec.	Operator starts second intermediate closed cooling pump in preparation for putting a second letdown cooler in service, allowing increased letdown to halt pressurizer level rise.	1. Alarm printer
88	5 min. 15 sec.	Operator restarts condensate pump 1A. Two condensate pumps should now be running. Operator tries to restart condensate booster pump 2B, but it trips; probably on low suction pressure.	1. Alarm printer
89		Note: Operators attempting to regain normal condensate and feedwater flow to pump down condenser, preventing loss of vacuum or turbine damage on high water level, and to enable use of main feedwater pumps to feed the OTSGs.	
90	5 min. 15 sec.+	Pressurizer level continues to increase. The operator opens MUP1A recirculation valve, throttles the HPI flow to minimum and initiates letdown at a rate in excess of 160 gpm, in attempt to halt the pressurizer level rise. Orifice and bypass valves are utilized together with a second heat exchanger now in operation.	1. Alarm printer
91		Note: At five minutes after the transient, pressurizer level starts dropping at a rate of approximately 60 inches per minute for	1. Reactimeter 2. Interview 14

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	15 seconds. At this same time, an apparent decrease in loop "A" flow and increasing reactor coolant temperatures are observed. The pressurizer level rate of change is not in agreement with the actions, system temperature changes, and makeup pump operation occurring during this period, unless voids exist in the RCS.	
92	5 min. 20 sec.	Operator again tries to start condensate booster pump 2B and it trips again.	1. Alarm printer
93	6 min.	Pressurizer level appears lost (bubble lost). RCDT pressure starts rapid rise above 120 psig to the RCDT safety valve setpoint of approximately 150 psig, most probably the result of the open EMOV now passing liquid versus steam. The combination of full Pressurizer indication and increasing RCS pressure, the result of rising hot leg temperatures while charging with the one MU Pump, appears to convince operators that they have a solid plant.	1. Reactimeter 2. Plant Charts 3. Interview 15
		PLANT STATUS	
94		The RCS hot leg temperature and pressure have reached saturation conditions, as indicated by the reactimeter and the wide range RCS pressure stripchart. The RCS indicated flow rate decreased sharply, suggesting a reduction in reactor coolant density. The OTSGs had boiled dry as indicated by a continuously decreasing steam pressure, while RCS hot leg and cold leg temperatures increased.	1. Reactimeter 2. Plant strip charts
95	6 min. 24 sec.	Operator again tries to start condensate booster pump 2B and it trips again.	1. Alarm printer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
96	6 min. 58 sec.	Operator throttles letdown flow to 71 gpm following high temperature alarm.	1. Alarm printer 2. Interview 15
97	~ 7 min.	Operators noted plant conditions did not match Emergency Procedures.	1. Interview 15
98	7 min. 29 sec.	Reactor building sump pump (WDL-P-2A) turns on, presumably pumping 140 gpm from the reactor building sump to the miscellaneous waste holdup tank (WDL-T-2) via the normally open reactor building isolation valves. However, tank level records and operator interviews show this tank not to have changed level significantly during the incident, and there is reason to believe the sump pump discharge was aligned to the auxiliary building sump tank (an alternate path). (This tank had a blown rupture disc which was scheduled for later repair.) These isolation valves do not close on the SFAS signal which initiates HPI. These valves will close when 4 psig in the reactor building SFAS setpoint is reached.	1. Alarm printer 2. Radiological sequence of events
99		Note: The reactor building sump pump had started (WDL-P-2A) and run for 1 1/2 minutes at 01:23 hours; and at this time the pumping of the sump was not considered to be abnormal.	1. Alarm printer 2. Interviews 5, 15
100	~ 8 min.	Second Shift Supervisor, who was onsite because of start-up in progress on Unit 1, arrives in Unit 2 control room to assist as necessary.	1. Interview 95
101		Note: The availability of a second Shift Supervisor is not required by Technical Specifications, but he was available due to the pending startup of Unit 1.	
102	8 min.	Operator finds OTSG level at 10 inches on the startup range. Operator considers this level means OTSG is "dry" per his training. Operator	1. Interview 61 2. Reactimeter

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	verifies emergency feedwater pumps are running and examines valve line-up. Operator finds and announces that EF-V12A & 12B valves are shut. Position indicating lights on 12A were obscured by a caution tag hanging from another valve controller. Position indicating lights for 12B may have been obscured by operator's body as he leaned over panel. Operator drives valves open, resulting in dry OTSG being fed with relatively cool water. Hot and cold leg temperatures drop. RCS pressure, now under control of loop saturation conditions, follows accordingly.	<ol style="list-style-type: none"> 3. Plant strip charts 4. Surveillance test procedure 5. Interviews 5, 9, 14 6. Memory trip review
103		Note: On 3/23/79, an NRR Operator Licensing Branch (OLB) examiner conducted examinations and "walked thru" the EFW System. The valves were open on that day.	<ol style="list-style-type: none"> 1. Memo from OLB to Investigation Team (5/21/79)
104		Note: A routine, scheduled surveillance test was performed on the A & B electric emergency feedwater pumps on 3/26/79, at approximately 10:00 a.m., by training shift CRO and AO.	
105		Implementation of this surveillance test procedure results in closure of both the 12A and 12B valves, regardless of which pump is being tested. The procedure calls for reopening of the valves, along with ensuring the proper status of at least three other valves. The procedure for the electric driven emergency feedwater pumps is insufficiently specific to provide documentation of valve opening, in that the procedure does not require individual signoffs for each valve; rather, the procedural requirement is in sentence form with one sign-off signifying proper positioning of the valves. Additionally, no documentation is available that the ST steps were completed, with the exception of the completed data sheets.	

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
106		The investigation has not found any evidence of willful closure of the valves over the period of 3/26/79 to 3/28/79.	
107		Note: The Control Room Operator (CRO) who was on duty in the control room when the 3/26/79 surveillance was performed, stated during interviews his specific recollection that he left the 12A & 12B valves open at the end of the test.	1. Interview 131
108	8+ min.	Rapid rise in OTSG pressure observed indicating feed flow to generators.	1. Reactimeter
109		Confirmed by EFW pump discharge pressure decreasing and "hammering" and "crackling" heard from the vibration and loose parts monitor speaker aligned to listen to "A" OTSG.	2. Interview 9
110	8 1/2 min.	Hot leg temperature dropping rapidly at a rate of approximately 110 ⁰ F/hr.	1. Reactimeter
111	9 min.	Condensate pump 1A trips. (Analysis would suggest this could only occur if operator holds start-stop switch of condensate booster pump 2A in start for at least 10 seconds or pump is tripped by operators locally.) Memory trip review records that pressurizer level reaches 404.9 inches.	1. Alarm printer 2. Burns & Roe drawings 3. Memory trip review
112	9+ min.	Source Range NI's energized as Intermediate Range NI's drop below 5X10 ⁻¹⁰ amps.	1. Alarm printer
113	~ 9 1/2 min.	Turbine bypass valves placed in MANUAL OPEN, because OTSG pressure was increasing and valves were not responding.	1. Interview GPU-20
114	~ 10 min.	Plant Nuclear Engineer arrives from Unit 1 and commences routine post-trip review. Another engineer calls the Unit 2 Superintendent to	1. Interviews 97, 1, 83

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	<p>notify him of the trip, apparently as a routine action with no mention of anomalous conditions.</p> <p>PLANT STATUS</p>	
115		The RCS pressure was at saturation conditions with respect to RCS hot leg temperatures. This was the result of the EMOV remaining open, high letdown flow rate, throttled high pressure injection isolation valves and emergency feedwater addition to the OTSGs. Emergency feedwater flow had been established to both OTSG's resulting in increased steam pressures and rapid RCS cooldown and depressurization.	<ol style="list-style-type: none"> 1. Reactimeter 2. Plant strip charts
116	~ 10 min.	On-call TMI Duty Officer (then the Unit 2 Superintendent - Technical Support) called by site personnel. He arrives 40 minutes later.	<ol style="list-style-type: none"> 1. Interviews 27, 83
117	10 min. 19 sec.	Second reactor building sump pump (WDL-P-2B) starts. Now pumping reactor building sump to the auxiliary building at 280 gpm.	<ol style="list-style-type: none"> 1. Alarm printer
118		Note: Second reactor building sump pump starts based on rising level in sump at 4.4'.	
119	10 min. 24 sec.	Operator stops, restarts and stops makeup pump 1A within a 4 second interval.	<ol style="list-style-type: none"> 1. Alarm printer
120	10 min. 48 sec.	Reactor building sump alarms high at 4.65 ft.	<ol style="list-style-type: none"> 1. Alarm printer
121		Note: A control room operator reported that the sump overflowed (6 feet) sometime after this point.	<ol style="list-style-type: none"> 1. Interview 15

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
122	10 to 11 min.	Pressurizer level indication comes back on scale and drops rapidly (20" in 1 min.), as RCS loop temperatures continue to drop from heat removed by the OTSGs. Apparently, the two-phase hot leg water, received from the surge line, remains separated as it flows through the pressurizer, giving the external appearance of a reestablished level indication.	1. Reactimeter
123	11 1/2 min.	RCDT cooler intermediate cooling temperature indicates offscale (>225°F).	1. Alarm printer 2. Reactimeter
124		Note: RCDT at 125 psig and increasing.	
125	11 min. 43 sec.	Makeup pump 1A restarted.	1. Alarm printer
126		Pump 1A remains in throttled condition until second HPI initiation at 3 hrs. 23 min. 16 seconds.	
127	8 to 12 min.	Operators attempting to establish 30 inches in each steam generator per procedure. Level increase in "A" OTSG leads that in "B" OTSG. Interview and records review indicates operators throttled EFW before reaching minimum steam generator levels to limit RCS pressure and temperature decrease.	1. Interview 196 2. Plant strip charts 3. Reactimeter 4. Memory Trip Review
128	13 min.	Decay heat (DH) removal pumps 1A and 1B turned off with RCS pressure about 1400 psig.	1. Alarm printer 2. Plant strip charts
129	13 min.	The memory trip review for this period shows that the discharge pressure of the EFW pumps dropped after the opening of the EF 12A and B header isolation valves. This drop is appropriate and signifies the start of	1. Log typer 2. Interviews 189, 196

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	flow to the OTSG's. However, the discharge pressure increases again reaching a peak (less than the previous shut-off head) suggesting an interference with discharge flow. Operator statements differ as to whether the control valves (EF11A & B) were changed by them, but the low level point in the OTSG's had not yet been reached, inferring that the ICS should not have automatically modified the EF11A & B position.	
130	14 min.	RCDT rupture disc blows at a RCDT indicated pressure of 192 psig, dropping RCDT pressure to approximately 10 psig in 36 seconds. Reactor building pressure appears to jump one full psi. At this time, to about 1.2 psig pressure rise seen in reactor building.	<ol style="list-style-type: none"> 1. Reactimeter 2. Plant strip charts 3. Interview 5
131	15 min.	RCS pressure approximately 1275 psig with cold leg temperature of approximately 567°F. Saturation pressure for 567°F is 1185 psig.	<ol style="list-style-type: none"> 1. Reactimeter 2. Plant strip charts
132		<p>Note: A review of the control room instruments and alarms and discussions with operating personnel relative to the RCDT reveals the following:</p> <ol style="list-style-type: none"> 1. RCDT pressure is not recorded. 2. RCDT parameters do not alarm on the panels in immediate view of the operators. 3. To determine there is an alarm on the RCDT annunciators, operators must clear all audible alarms on the front panels. 	<ol style="list-style-type: none"> 1. Inspection
133		Note: Rupture Disc located on top of tank. Once blown, tank will fill and overflow.	<ol style="list-style-type: none"> 1. Construction Drawing

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
134	15+ min.	Reactor coolant pump related alarms received. (During entire transient there was a slow reduction in indicated loop flows. These alarms were received at various times after this point until the pumps were shut down.	1. Alarm printer 2. Reactimeter 3. Interview 61 4. Interview 5
135	18 min.	Fuel handling exhaust monitors show small ramp increase in iodine reading. Reactor building exhaust shows factor of 10 increase in reading. (Instrument location on the lower part of the vertical back panel would prevent operator standing at front panel from viewing these trends.)	1. Plant strip charts
136		Note: RCS was effectively being degassed into the reactor coolant drain tank and, when rupture disc blew, gas was released into the reactor building.	1. System Descriptions
137		RCDT vent lines should isolate at 10 psig automatically; but after the rupture disc blew, the vent line valve would reopen (6 psig) venting the RCDT to the waste gas system. The waste gas system header is maintained at 2 psig and as long as the reactor building isolation had not occurred, the waste gas compressors pumped gas from the reactor building to the gas surge tank.	
138	22 min.	Increase seen on SRM nuclear instrumentation and reactor manual trip performed. Radiation level on letdown monitor increases and returns to previous level. RCP full speed and vibration alarms now begin to appear and will continue until the RCPs are tripped.	1. Alarm printer 2. Plant strip charts
139		Note: SRM count rate increases are believed as reported by others to be associated with increases in coolant void fraction, reducing	1. TMI Core Assessment, EG&G, 4/6/79

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	shielding of detectors. This evaluation is outside the scope of this investigation and is presented here for information only.	
140		Note: From approximately 20 to 60 minutes, system parameters generally stabilized at a reactor pressure 1015 psig, reactor coolant average temperature of 550 ⁰ F, and a pressurizer level of 380-395 inches.	1. Plant strip charts 2. Reactimeter
141	23 min.	Low level alarm on "A" OTSG first to clear at 26.6".	1. Alarm printer
142	26 min.	Steam driven emergency feedwater pump turned off. Operator trying to establish low level limit in steam generators but the level in "B" OTSG kept creeping up.	1. Alarm printer 2. Interview 61 3. Plant strip charts
143	27 min.	Low level alarm on "B" OTSG clears at 26.6".	1. Alarm printer
144	27 min. +	Operator requests computer print pressurizer relief and code safety outlet temperatures, hot leg and cold leg temperatures, loop pressurizer and pressurizer level.	1. Utility printer
145	28 min.	Operator secures feeding "B" OTSG by closing EFV-11B and then EFV-12B as level continues to rise. (Later operators will shut down one electric pump. Steam generator pressure being controlled manually through the turbine bypass valves.)	1. Plant strip charts 2. Interviews 5, 14
146		Note: Operators have been dispatched to the auxiliary building to confirm pressurizer level indications and/or determine source of water which has filled pressurizer. The shift personnel were concerned with the continuing high pressurizer level.	1. Interview 15

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
147	30 min.	Operator requests memory trip review printout. (Printout shows one condensate pump running at shutoff head.)	1. Log typer
148	30 min.	Diesels manually tripped locally by operators. They had been running unloaded since original ES actuation and are not needed at this time. When the diesels start on an HPI signal, they must be shutdown locally. At this point, the operating staff did not reset the trip. The diesels would not have started by any automatic or manual start signal. This condition lasted until ~ 5 1/2 hrs.	1. Alarm printer
149	31 min.	Operator requests computer print Sequence of Events Review. There will be several occasions during this sequence when the operator is pulling "Sequence" information from the computer. In each instance, the information once typed is eliminated from the "Sequence" portion of the computer memory. Subsequent printouts then provide information from the time of the last printout.	1. Utility typer 2. Discussion with computer engineer
		Alarm Printer shows Incore Thermocouple R-10 go offscale at 32 1/2 minutes.	1. Alarm printer
150	~ 35 min.	Unit 1 Shift Foreman called Unit 1 Operations Supervisor to inform him of unit trip.	1. Interview 26
151	36 min.	EFW pump "2B" turned off; both OTSGs approximately 35".	1. Alarm printer 2. Reactimeter 3. Interviews 5, 14
152	38 min.	Reactor building sump pumps turned off by auxiliary operator. The two sump pumps had operated for 31 and 28 minutes, respectively, pumping	1. Alarm printer 2. Analysis

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	approximately 8260 gallons of water to the auxiliary building. Shutdown of the pumps does not result in isolation of the pump discharge line. The possibility of continued water transport to the auxiliary building, caused by elevated containment pressure and considering tank elevations in the auxiliary building, appears unlikely.	3. Interviews 5, 15
153		Note: At this point, the criteria for isolating associated with sump pump operation EMOV were satisfied.	1. Emergency Procedure 2202-1.5
154		Note: The automatic pumping of the RB sump to the auxiliary building was not considered abnormal since operators knew RCDT rupture disc had blown.	1. Interviews 5, 15
155	44 min.	Operator requests computer print pressurizer level differential pressures in attempt to determine if level indication in error. (Operators stated during interviews that all level indicator outputs supported they actually had high pressurizer level. Operators stated all the sensors couldn't be wrong.)	1. Utility typer
156	50 min.	Steam generator "A" level starts trending down while generator "B" level starts trending upward.	1. Interview 5 2. Reactimeter
157		Superintendent of technical support, the on-call Duty Officer, reports to the site. He had been called at approximately 10 min. and left home for the site approximately 30 minutes later.	1. Interview 5
158	52 min.	Operator requests computer print condenser pressure and steam drive emergency feedwater pump discharge pressure.	1. Utility typer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
159	59 min.	Condensate booster pump suction header low pressure alarm clears and slug of hot condensate passes condensate temperature detector. (The hot slug of water would be created by condensate pump 1B pumping against a shutoff head for the last 59 minutes.) (Operators still attempting to establish normal condensate flow to restore normal hotwell level. It appears that operators have finally been successful in manually opening the polisher bypass valve.)	1. Alarm printer 2. Interviews 5, 14
160		Note: Hotwell had been flooded when instrument air line for level controller had broken, probably during initial water hammer that reportedly moved some pipes 2 to 3 feet. (This degree of movement unlikely based on inspection of area.)	1. Interview 5 2. Inspection
161		Note: By design, the polisher bypass valve on Unit 2 does not open automatically to limit polisher differential pressure. Attempts to open valve using valve controls in control room had been unsuccessful. Valve motor reportedly unable to open valve with differential pressure across it.	1. Interviews 109, 123
162		Note: All polisher inlet and outlet valves had been found shut. Manual handwheel for operating the polisher bypass was missing from the valve, but found on top of ventilation duct. The Shift Supervisor was present and personally involved in manually opening this valve locally.	1. Interview 109
163		Note: This operation should have enabled normal reject flow to the condensate storage tanks and did automatically shift the emergency feedwater pump suction to the discharge of the condensate pumps. Operators have yet to discover broken air line to AUTO-REJECT valve (it fails shut) and will wonder why hotwell level doesn't go down.	1. Burns and Roe drawings 2. Transient logs

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
164		Note: A leak had developed on the suction piping of condensate booster pump 2A.	1. Interviews 109, 123
		PLANT STATUS	
165		The RCS was near a saturation temperature-pressure relationship at 550 ⁰ F and 1050 psig. The indicated loop flow rates had decreased from about 69 million pounds per hour to approximately 47 million pounds per hour and continued to decrease. The reactor building temperature and pressure had increased from 120 ⁰ F to 170 ⁰ F and 0 psig to 2.5 psig, respectively, as a result of releasing contents of the reactor coolant drain tank to the reactor building atmosphere. The EMOV was still open and makeup pump 1A was operating. The operator was having difficulty controlling the level of steam generator B. Emergency feedwater valves (EF-V11B and EF-V12B) were shut and the operator was admitting feedwater to steam generator B by cycling emergency feedwater crossconnect valve (EF-V5B).	
166	~ 1 hr.	On-call operating engineer called to come to site. He arrived at site approximately 45 minutes later.	1. Interview 5
167	61 min.	Circulating water pumps B, C, D, E off. Letdown line radiation monitor shows small increase in level.	1. Alarm printer 2. Interview GPU-1 3. Interviews 5, 9
168		Note: The circulating water pumps were turned off to automatically switch steam dump to the atmospheric relief valves to prevent flooding of the condenser or an uncontrolled shift to those same valves on loss of con-	

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	denser vacuum; the latter concern relating to inability of Unit 1 to supply adequate gland sealing steam pressure from their auxiliary boilers.	
169		Note: The intermediate cooling water radiation monitors (1082 and 1083) alarm were believed to be due to high background radiation levels near the containment sump, which was being overflowed from the reactor coolant drain tank rupture disk failure. It was believed normal RCS coolant activity levels would be sufficient to cause these alarms. Further, the operators were aware of a design change request to reduce the sensitivity of these instruments to reduce spurious alarms. The operator did verify no increase in the intermediate closed cooling surge tank level to ensure no primary system to closed cooling water system leak.	
170	62 min.	Alarm printer unavailable starting at this time. Not returned to service until 3 hr. 51 min. Operator interviews indicate that there was malfunction of the paper feed. Review of other alarm printer outputs for various other periods, shows evidence of paper feed problems.	1. Interview 5 2. Record review
171	71 min.	Temperature transient on cooling water indicates reactor building air cooling coil B now operating. The reactor building cooling was initiated with emergency river water cooling through the reactor building cooler.	1. Utility typer 2. Interview 14
172		Note: Operators were unable to account for increased reactor building temperature.	
173		Note: This data point based on transient in alarm printout of outlet temperatures and may be erroneous conclusion. Further alarms	

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	will be shown elsewhere in this sequence. However, the investigation has not determined an explanation for this recurring alarm. They are shown for the sake of completeness should a later investigation determine their cause. It should also be noted that no similar alarms are received from any of the other cooler units throughout the accident.	
174	72 min.	Operator requests computer print those conditions currently in alarm related to RCPs. (This action believed associated with desire for more information to base decision on which RCPs to be tripped.)	1. Utility typer
175	Prior to 74 min.	Reactor coolant pump flow indication has been steadily decreasing and indicate flow in loop "B" at 74 min. approximately 60% of normal value. Operators report receiving vibration alarms on the reactor coolant pumps (RCP). Chart recorders for source range nuclear instrumentation show increasing average count rate with increasing amplitude of oscillation.	1. Reactimeter 2. Interview GPU-1 3. Plant strip chart
176		Operators reportedly recognized that they were violating 4 RCP pressure-temperature limits and the pin compression curves. The Shift Supervisor and the Technical Superintendent specifically reviewed these curves.	1. Interviews 53, 5 2. Operating Procedure 2202-1.3 3. Transient Logs
177	74 min.	RCP's in B loop chosen to be tripped to be able to maintain maximum pressurizer spray capability which comes from "A" loop. RCP 1B and 2B tripped. (Utility typer indicates that RCP 2A also tripped. This inconsistency believed to be associated with inductive pickup in alarm wiring or bouncing contacts. Operators reported earlier examples of this indication were experienced with RCP 2A known to be running.)	1. Interview GPU-20 2. P&ID 3. Utility typer 4. Record review 5. Interview 6

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
178		Note: RCPs were running without adequate net positive suction head, subjecting these pumps to potential damage from cavitation.	
179	74 min.	<p>The hourly plant log typer displayed the following information as being applicable for this time:</p> <p style="margin-left: 40px;">RCS Flow: 89 MLBH</p> <p style="margin-left: 40px;">Loop "A"</p> <p style="margin-left: 80px;">Th : 543 °F</p> <p style="margin-left: 80px;">Tc : 540 °F</p> <p style="margin-left: 80px;">Tc_a: 539 °F</p> <p style="margin-left: 40px;">Loop "B"</p> <p style="margin-left: 80px;">Th: 543 °F</p> <p style="margin-left: 80px;">Tc : 540 °F</p> <p style="margin-left: 80px;">Tc_b: 540 °F</p> <p style="margin-left: 40px;">MU Flow : 118 GPM</p> <p style="margin-left: 40px;">OTSG "A" Level: 34 inches</p> <p style="margin-left: 40px;">OTSG "B" Level: 70 inches</p> <p style="margin-left: 40px;">Condensate Flow: 513 KLBH</p>	1. Log typer
180	74 min. +	<p>Operators reported reactor coolant flow oscillations decreased immediately following pump trip. Moreover, Source Range (SRM) instrumentation count rate oscillatory behavior significantly decreased. Average SRM count rate was approximately three times the value that would be expected at this time after reactor trip. RCS pressure indicates an upward pressure trend, and "B" steam generator pressure drops from approximately 960 psig to 140 psig over the next 18 minutes.</p>	<ol style="list-style-type: none"> 1. Interviews 2. Plant strip charts 3. Reactimeter

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
181	~ 75 min.	RCS sample yields 700 ppm Boron. Sample required by emergency procedures within 2 hours of trip. Sample results were questioned by technician and he started to get additional sample. (Calculated shutdown margin at this time for the reactor was ~ 6% reactivity.)	1. Interviews 58, 5 2. Emergency Procedure 2202-1.1
182		Note: Apparent reduction in boron concentration believed associated with dilution of liquid phase of sample by condensation of steam in sample stream. However, this was not a conclusion drawn by the operator staff at that time.	
183		Station Manager calls Unit 2 (call could have been as late as 95 min.) to ascertain plant status not having heard further since original notification. When informed of status, Station Manager directs Duty Officer to establish conference call between selected staff. (This conference call occurs at ~ 2 hrs.)	1. Interview 1
184	80 min.	Factor of 7 increase in radiation level as shown on letdown line monitor.	1. Plant strip charts
185	81 min.	Operator requested computer print Pressurizer Surge Line Temperature (514 ⁰ F), EMOV Outlet Temperatures (283 ⁰ F), Code Safety Outlet Temperatures (211 ⁰ F & 219 ⁰ F), Pressurizer Spray Line Temperature (497 ⁰ F), and Condensate Pump Outlet Header Pressure (165 psig).	1. Utility Typer
186		Note: EMOV outlet at least 64 ⁰ F hotter than code safety temperatures. (Why EMOV block valve was not shut at this point is unclear.)	

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
187	3/28/79	Note: Operators knew the EMOV and/or the code safeties leaked prior to the event. They reportedly had to add approximately 2400 gallons of makeup, per shift to the RCS and the tail pipe temperatures were always above ambient. Under these conditions, operators may have been hard-pressed to decide what temperatures constituted abnormal indications given their recognition of an earlier lifting of the EMOV.	1. Interview 80
188	87 min.	The "B" steam generator was isolated at this point. Operators had observed increases in reactor building pressure and noted that pressure in generator "B" was 300 psig lower than "A". They believed steam was leaking into the reactor building from "B" OTSG secondary side. After isolating steam generator "B", they note building pressure leveling off and "B" generator level trending upward, so they secured the third EFW valve to the generator, EFW-5B.	1. Control room logs 2. Plant strip charts 3. Interview 5 4. Transient Logs
189		Note: Reverse flow should have occurred in the "B" loop as a result of continued operation of the pumps in the "A" loop. No substantial changes in steam generator level or pressure occurred over the next hour. It appears that reverse flow of RCS coolant at a temperature of approximately 550°F was severely curtailed through the "B" generator, as was indicated by the rate of pressure drop in that generator. Some flow was occurring as indicated by loop "B" RTDs tracking those of loop "A".	1. Reactimeter 2. Analysis
190	~ 90 min.	RCS sample yields 400-500 ppm boron and 4.0 uc/ml; a further drop in boron concentration and a factor of 10 increase in activity. There is no clear evidence that this increase in activity was related to any fuel failure. The possibility of a crud burst or an iodine	1. Interview 58

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	spike cannot be discounted.	
191	91 min.	Operator requests Computer printout Sequence of Events Review.	1. Utility typer
192	~ 93 min.	Operators report increasing HPI flow; RCS pressure peaks and drops rapidly; SRM & IRM indication spike upward, and injection flow was maintained at 150-200 gpm/loop.	1. Interviews 5, 9
193		Note: The actual peak in reactor building pressure appears to occur here, versus immediately following isolation of "B" OTSG, as the operators believed.	1. Plant strip charts
194	~ 94 min.	"A" OTSG appears to boil dry, again, based on OTSG level, pressure and RCS temperature trends.	1. Reactimeter 2. Plant strip charts
195	96 min.	Operator again requests computer print pressurizer Level Indication (372.9").	1. Utility typer
196	Prior to 101 min.	Approximately 5-10 minutes after the trip of the "B" loop RCP's, SRM count rate instability increases again as well as continuing an upward trend. IRM current also begins to show upward trend. Operators report loop flow instability increasing again, and the indicated loop flow continues to show a decrease. Operators asserted during interviews they were concerned about a "seal failure LOCA" and decided to go on natural circulation. Operator stated that he started "emergency boration" during this period, based on SRM increases and fear of a restart accident.	1. Plant strip charts 2. Interview 9

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
197	3/28/79	Note: Conditions for initiation of natural circulation cooling of the RCS did not exist at this point.	1. Procedure 2102-3.3
198		Note: Operator training and plant procedures require securing RCPs on sustained high vibration.	1. Abnormal Procedure 2203-1.4 2. Interview 80
199		Note: Tech. Supt. Transient Log indicates tripped last two RCPs due to low suction pressure.	1. Transient logs
200		Note: The indicated increase in NIs has been reported elsewhere to be due to increased neutron leakage from the core as a result of increases in void content. This conclusion may be subject to some question. Review of the outputs of 2 IRMs suggests a difference in gamma compensation of the two chambers. The two chambers behave differently during this period which implies that their output may be dependent, to some degree, on changes in the leakage gamma flux from the core. The chamber which appears undercompensated exhibited the behavior described while the chamber which appears overcompensated did not.	
201		Note: Emergency Boration action consistent with procedure requirements.	1. Procedure 2203-1.1
202	101 min.	"A" loop RCPs tripped. SRM count rate spikes upward to peak at least one decade over count rate expected following a normal reactor trip. All radiation monitors exhibiting substantial ramp increase.	1. Utility typer 2. Reactimeter 3. Plant strip charts 4. Interview 9 5. Emergency Procedure 2203-1.3

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
203		Operators stated they did not believe that natural circulation had been established due to the differential temperature across the steam generator and the low steam generator pressure with minimum heat removal, if any.	1. Interviews 9, 15
204		Note: RCS was at 1020 psig and 535 ⁰ F; well outside prerequisite pressure/temperature conditions for establishing natural circulation.	1. Operating Procedure 2102-3.3
205	101+ min.	RCS Pressure continues to drop. SRM counts and IRM current drop significantly.	1. Reactimeter 2. Plant strip charts
206	102+ min.	The SRM count rate level, followed by intermediate range monitor (IRM) output, start upward trend. Over the next 15 minutes, SRM count rate level increases 2 decades without exhibiting previously observed oscillatory behavior. Operator reports he "emergency borated" again, based on fear of restart accident.	1. Interview 5 2. Procedure 2203-1.1
207		Note: Emergency boration action consistent with procedure.	
208	~ 103 min.	RCS hot and cold leg temperatures begin to diverge widely. Loop "A" hot leg temperature begins to rise to 620 ⁰ F (off scale within 14 minutes) and cold leg temperature drops to 150 ⁰ F (apparently from effect of HPI water). Loop "B" divergence is much slower, demonstrating little coupling between loops and reactor. The operators reported average RCS temperature "stabilized" at 570 ⁰ F.	1. Reactimeter 2. Plant strip charts 3. Interview 61 4. Multipoint recorder
209		Note: The average RCS temperature (Tav) recorder trace could not be relied upon by the operators for the period from 2 hours to 16 hours. Tav stabilizes at an indication of 570 ⁰ F as soon as both narrow range T hot and T cold instruments reached their	

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	display limits (620 ⁰ F and 520 ⁰ F, respectively).	
210		Note: RCS Hot leg temperatures already appear superheated.	
211	112+ min.	Per procedure, operator begins feeding "A" steam generator to reach 50% level. Operator states he is still controlling steam generator manually with turbine bypass valves. Operator intends to remove decay heat via the OTSG using natural circulation.	<ol style="list-style-type: none"> 1. Interview 61 2. Plant strip charts 3. Interview 5 4. Operating Procedure 2102-3.3
		PLANT STATUS	
212		The RCS had no forced convection cooling. All RCPs were stopped. The RCS average temperature and pressure were approximately 520 ⁰ F and 750 psig. Makeup pump 1A was operating. The operators were attempting to establish natural circulation flow to cool the reactor core. OTSG B was isolated because of a suspected leak to the reactor building.	<ol style="list-style-type: none"> 1. Reactimeter 2. Plant strip charts
213	114 min.	Operators requests computer printout Sequence of Events Review.	1. Utility typer
214	~ 2 hrs.	Nuclear Engineer called to site to gather information required for standard post-trip report. Some question then whether the reactor had experienced a restart based on SRMs.	1. Interview
215		Third boron sample indicates approximately 400 ppm.	1. Interview GPU-19
216	~ 2+ hrs.	Conference call established between Unit 2 Technical Supt. (Unit 2 Control Room) and Station Manager, Vice President of Generation and B&W site representative (at their homes), lasting approximately 38	<ol style="list-style-type: none"> 1. Interview 53 2. Statement Plant Manager

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	minutes. Knew trip was abnormal since RCPs were off and unable to draw pressurizer bubble. Blown rupture disk and water on floor were not surprising since this had happened before. The condition of the EMOV block valve was questioned and reported to be shut. Group decided need existed to restart an RCP and all should report to TMI. (Conference call had been initiated by Station Supt. following discussion with Unit 2 Tech. Supt. (on-call Duty Officer) around 1 hour 15 minutes into event. The Unit 2 Tech. Supt. had been on site since 50 minutes into the event, following his call to the site shortly following the initial trip.)	3. Telephone bills
217		Note: B&W representative reports that the fact that EFV-12A & B were shut with no feeding to the OTSGs for 8 minutes was not discussed the first day. It was discovered by B&W representatives later, while delogging the reactimeter. However, it appears that B&W in Lynchburg had this information as it was given to them by Operations Supervisor after he had made contact with the site. Information was that emergency feedwater had been delayed approximately 12 minutes.	1. Interview 53
218	2 hrs. 1 min.	The hourly plant log typer displayed the following information as being applicable for this time: RCS Flow: 3 MLBH Loop "A" Th: 570 °F Tc _a : ? °F Tc _a : ? °F Loop "B" Th: 526 °F Tc _b : ? °F	1. Log typer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	<p style="text-align: center;">T_c: ? °F MU Flow: 97 GPM OTSG "A" Level: 173 inches OTSG "B" Level: 79 inches Consensate Flow: 318 KLBH</p>	
219	2 hrs. 5 min.	"A" OTSG level reaches 50%. Operator either stops or reduces significantly the feed rate to that OTSG. Appears operator alternately feeds and steams down "A" OTSG level for next 50 minutes.	<ol style="list-style-type: none"> 1. Plant strip charts 2. Reactimeter
220	2 hrs. 14 min.	Operator requests computer printout Sequence of Events Review.	<ol style="list-style-type: none"> 1. Utility typer
221	2 hrs. 18 min.	Operator requests Computer print EMOV and code safety outlet temperatures (229° ⁰ F, 190° ⁰ , and 194° ⁰ , respectively).	<ol style="list-style-type: none"> 1. Utility typer
222		Electromatic relief valve is now isolated by operator by closing the block valve (RC-V2). Operators had noted tail pipe temperature on relief valve 35° ⁰ F higher than others and believed valve to be leaking. Operators noted a drop in reactor building pressure after closure of the block the block valve.	<ol style="list-style-type: none"> 1. Utility typer 2. Interview 5
223		The pressure change in the reactor building was more marked than when "B" OTSG was isolated. The plant operations group decided that the "B" OTSG not have a leak from the shell into the reactor building. The "B" OTSG still had a level.	<ol style="list-style-type: none"> 1. Interview 5 2. Plant strip charts
224	2 hrs. 18 min.+	With closure of EMOV, RCS pressure begins increase from low point of 660 psig and reaches 1300 psig. During this period from 40 minutes, no change in pressurizer level (about 290") is observed. At some time prior to this point, the isolation valves for the core flood tanks had	<ol style="list-style-type: none"> 1. Reactimeter 2. Plant strip chart

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	been isolated to prevent what was judged to be an unneeded discharge of water from the core flood tanks. Time of isolation and reopening has not been determined.	
225		Prior to this point, standard post-reactor-trip boron samples taken from the reactor coolant system had been analyzed with no problems indicated or detected relative to radiation levels.	1. Interview 7
226		Fifteen to twenty people in control room at this time.	1. Interview 12
227 228	2 hrs. 18 min.- 2 hrs. 30 min.	Discussions were held considering an entry into containment to manually vent the RCS hot legs. The occurrence of radiation alarms halted further consideration of this option. Area radiation monitor (ARM) alarms were received at the sample station and the letdown sample lineup was secured. Loop "A" hot leg temperature peaked at about 750°F.	1. Interviews 58, 105, 97 2. Multipoint recorder
229	2 hrs. 35 min.	Operator begins feeding isolated "B" OTSG to 50% level. Intent is to unisolate OTSG and improve natural circulation by using both OTSGs to remove core decay heat through natural circulation. Operator may have opened the turbine bypass isolation valve (MS-V15B) at this time, also. These actions are consistent with procedure requirements.	1. Plant strip chart 2. Reactimeter 3. Procedure 2102-3.3
230	~ 2 hrs. 40 min.	Operator initiates emergency boration based on increasing NI indication, low boron sample results report and calculated shutdown margin of only 2.4% reactivity.	1. Statement Plant Manager 2. Interview 5
231		Note: These actions consistent with requirements of emergency procedure.	1. Emergency Procedure 2203-1.1

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
232	2 hrs. 45 min.	Makeup pump 1C tripped by operator action according to GPU sequence. However, the GPU sequence does not indicate a start time. The last identified change of status was at 4 min. 38 sec. when the pump was tripped. No other status change is noted up to 1 hr. 13 min. 27 sec. when the Alarm Printer stopped working.	1. GPU sequence of 7/16/79
233		NOTE: This information not supported by Sequence of Events Review printout from the plant computer.	
234	2 hrs. 46 min.	The utility typer stops and restarts. Alarm data from 1 hour 13 min. to 2 hours 48 min. is lost. Alarm printer had been lost earlier due to paper feed problem.	1. Utility Typer
235	~ 2 hrs. 48 min.	RCS loop "A" hot leg at 800 ⁰ F and increasing.	1. Multipoint recorder
236		High radiation readings obtained in hot machine shop. Hot machine shop high radiation alarms were received in Unit 1 control room.	1. Plant strip charts
237	~ 2 hrs. 50 min.	Operators finally solve condensate reject problem and pump condensor hotwell level down. Had found automatic reject valve closed due to broken air line.	1. Alarm printer
238	2 hrs. 52 min.	Operator places RCS loop "B" cold leg tempature in analog recorder trend (#2).	1. Alarm printer
239	~ 2 hrs. 53 min.	Operators finally get condensate reject valve open with technicians, help which allows them to drop hotwell level.	1. Alarm printer
240	2 hrs. 54 min.	Electricians jumper RCP start interlocks ("K" relays) to allow pump start. Operators try to start 1A, 1B & 2A RCPs without success.	1. Alarm printer 2. Interview 61

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
		Operators start RCP 2B and it runs for 19 min. Operators state they received all radiation alarms. Operators started RCP because they were not sure that natural circulation had been established and NI counts were trending up. During this operation of the pump, RCS pressure went up to 2200 psig and SRM counts and IRM current dropped significantly.	3. Plant strip chart 4. Interviews 5, 105
241		Groups 1-5 of the pressurizer heaters are tripped and remain in a tripped state for 1 1/2 hours. Throughout the course of the incident, difficulty is experienced with multiple pressurizer heater trips, presumably due to moisture content in the reactor containment; a problem previously experienced on Unit 1 at this site and other facilities. As of this time, no more than eight groups of heaters are operable.	1. Alarm printer 2. Interview 53
242		NOTE: The Shift Supervisor felt that the pressurizer heaters should have been more effective in recovering RCS pressure and dispatched the the Shift Foreman to the pressurizer heater control cabinet to confirm that their breakers were shut. The Shift Foreman verified breakers were closed; noted the area was extremely hot and humid (believed due to proximity to main steam relief lines); noted local vent fans were tripped (due to fire heat sensor trip); and he defeated the trips and re-started fans.	1. Interview 107
243	2 hrs 54 min.	Operator places RCS loop "A" cold leg temp. in analog recorder trend (#2).	1. Alarm printer
244	~ 2 hrs. 54 min.	Incore thermocouple data collected by computer that will not be printed for at least 1 hour:	1. Alarm printer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>																										
	3/28/79																												
		<table border="1"> <thead> <tr> <th><u>Thermocouple Location</u></th> <th><u>Temperature (°F)</u></th> </tr> </thead> <tbody> <tr><td>TC 7B</td><td>624</td></tr> <tr><td>TC 1-H</td><td>596</td></tr> <tr><td>TC 12-0</td><td>625</td></tr> <tr><td>TC 14-M</td><td>654</td></tr> <tr><td>TC 14-D</td><td>688</td></tr> <tr><td>TC 13-C</td><td>577</td></tr> <tr><td>TC 11-L</td><td>584</td></tr> <tr><td>TC 10-C</td><td>599</td></tr> <tr><td>TC 2-G</td><td>624</td></tr> <tr><td>TC 13G</td><td>671</td></tr> <tr><td>TC 3F</td><td>682</td></tr> <tr><td>TC 5D</td><td>700</td></tr> </tbody> </table>	<u>Thermocouple Location</u>	<u>Temperature (°F)</u>	TC 7B	624	TC 1-H	596	TC 12-0	625	TC 14-M	654	TC 14-D	688	TC 13-C	577	TC 11-L	584	TC 10-C	599	TC 2-G	624	TC 13G	671	TC 3F	682	TC 5D	700	
<u>Thermocouple Location</u>	<u>Temperature (°F)</u>																												
TC 7B	624																												
TC 1-H	596																												
TC 12-0	625																												
TC 14-M	654																												
TC 14-D	688																												
TC 13-C	577																												
TC 11-L	584																												
TC 10-C	599																												
TC 2-G	624																												
TC 13G	671																												
TC 3F	682																												
TC 5D	700																												
245		NOTE: The majority of readings from this system indicate data points have just failed or are off-scale high. The data point values, displayed above, indicate these thermocouples have returned on scale and the computer has just noted their return.																											
246	2 hrs. 56 min.	Started circulating water pump 1B & 1E. (This automatically shifts "A" & "B" OTSG pressure controls to the turbine bypass valves.)	1. Alarm printer																										
247		SRM and IRM indications drop significantly.	1. Plant strip chart																										
248		Operator opens and then closed "B" OTSG MSIVs (MS-V4B&7B), within 7 seconds. (Operator actions in cycling these valves have not been explained. Operators speculate this event may represent inadvertent actions associated with attempts to cope with the rapidly deteriorating	1. Alarm printer 2. Reactimeter 3. Interviews 5, 189, 111																										

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	situation, which included offgas radiation monitor alarms. It should be noted that operation of these MSIV's does not affect steam flow to the condenser if the turbine bypass controls are in "Hand," which they reportedly were at this time.)	4. Multipoint recorder 5. Burns and Roe drawings 6. Transient logs
249		HP Technician reports over page that letdown sample lines from Unit 2 was reading 600 R. Auxiliary building evacuated.	1. Interviews 13, 105
250		Unit 2 Shift Supervisor and Technical Superintendent then declared a site emergency. Declaration based on radiation alarms in more than one area. Notification of offsite authorities begins.	1. Statement Plant Manager 2. Interview 48
251	2 hrs. 56 min.	North Bridge Gates being closed due to Guards responding to the establishment of site emergency.	1. Interview 8
		Secured liquid release from Unit 1 and Unit 2 Operators verified appropriate valve closed. Action was taken by Unit 1 operators at direction of Unit 2.	1. Interview 15
252	~ 2 hrs. 57 min.	Radiation level in vicinity of Unit 2 sample area, where lines were on recirculation for another boron sample, reportedly showed marked increase.	1. Interview 7
253		Fifty to sixty people estimated to be in control room at this time with noise level very high.	1. Interviews 7, 12, 105
254	3 hrs.	Operator places RCS Loop "A" cold leg temp. in analog recorder trend (#3). Both RCS loop hot leg temps. go off scale.	1. Alarm printer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
255	3 hrs. 1 min.	<p>The hourly plant log typer displayed the following information as being applicable for this time:</p> <p style="margin-left: 40px;">RCS Flow: 3 MLBH</p> <p style="margin-left: 40px;">Loop "A"</p> <p style="margin-left: 80px;">Th: 570 °F</p> <p style="margin-left: 80px;">Tc : ? °F</p> <p style="margin-left: 80px;">Tc^a: ? °F</p> <p style="margin-left: 40px;">Loop "B"</p> <p style="margin-left: 80px;">Th: 526 °F</p> <p style="margin-left: 80px;">Tc : ? °F</p> <p style="margin-left: 80px;">Tc^b: ? °F</p> <p style="margin-left: 40px;">MU Flow: 125 GMP</p> <p style="margin-left: 40px;">OTSG "A" Level: 245 inches</p> <p style="margin-left: 40px;">OTSG "B" Level: 220 inches</p> <p style="margin-left: 40px;">Consensate Flow: 0 KLBH</p>	1. Log typer
256	3+ hrs.	Eighteen to twenty-four people reported to be in control room at this time. Not known if excess people ordered from control room or if differences previous estimate related to individual estimates.	1. Interview 48
257		Operators trying to recover pressurizer level indication which was now off scale high.	1. Interview 48
258		Guards at North Gate delay, then allow B&W site operations manager access to site. He was arriving at site at Station Manager's request after earlier conference call which ended at ~ 2 hrs. 58 min. He reached the control room at about 3 hrs. 45 minutes after contacting B&W in Lynchburg. Guards were sending other people arriving at site to Observation Center.	1. Interview 48

<u>ITEM</u>	<u>DATE AND TIME</u> 3/28/79	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
259	3 hrs. 2 min.	RCS loop "B" hot leg temperature reaches 800 ⁰ F.	1. Multipoint recorder
260	3 hrs. 4 min.	Region I Answering Service receives call from Licensee.	1. Telephone bills 2. Statement Plant Manager 3. Region I notes
261	~ 3 hrs. 5 min.	Station Manager arrives at site and proceeds to control room to assume duties as Emergency Director.	1. Interview 1
262		OTSG "B" turbine bypass isolation valve (MS-V15B) is closed by operator, completing the isolation of "B" OTSG . IRM current returns on scale, increasing. Operator secures feeding "B" OTSG.	1. Alarm printer 2. Strip charts 3. Reactimeter
263	3 hrs. 11 min.	EFW pump (EF-P-2A) tripped by the operator. At this point, OTSG levels were in the 55% to 65% range and no further feed appeared necessary.	1. Alarm printer 2. Reactimeter
264	3 hrs. 13 min.	SRM counts and IRM current peaked; both then dropped significantly as pressure peaked. IRM current drops off scale just prior to the RCP being tripped.	1. Alarm printer 2. Reactimeter
265		High temperature noted on tail pipe of EMOV. Rate of pressure increase in reactor building suggests operator has opened EMOV block valve and EMOV, in manual. RCS pressure and pressurizer level both drop (RCS pressure to 1975 psig; pressurizer level - 300"). A pressure spike of 5.5 psig also seen in RCDT.	1. Alarm printer 2. Plant strip charts 3. Reactimeter
266		Note: Pressure increase in reactor building stops about 3 minutes later, suggesting the EMOV discharge path is again isolated. A quick	1. Plant strip charts

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	bump upward in reactor building pressure about 4 minutes later, indicated another short RCS blowdown using this path. The validity of the RCDT pressure spike is very doubtful. From approximately one hour on, the RCDT pressure, as recorded on the reactimeter, approaches zero and starts to record negative values for much of the remainder of the accident. No correlation can be drawn between RCDT pressure and reactor building pressure which it should now be tracking in some general fashion.	2. Reactimeter
267		RCP 2B tripped by operators after vibration alarms received and motor only drawing 100 amps. Operator notes no change in Tave; still at 570° F (a false indication).	1. Alarm printer 2. Utility typer
268		Operator stated that coincident with tripping of RCP further radiation alarms received and they started following radiological emergency procedures.	1. Interviews 2. Plant strip charts
269	3 hrs. 14 min.	Received high temperature alarms on code safety outlets.	1. Alarm printer
270	3 1/4 hrs.	Control building evacuated with exception of control room. Building reported to be clear by operator who has just returned.	1. Interview 7 2. Radiation survey log
271		After RCS pressure holds at 1975 psig for approximately 5 minutes, a rapid pressure decrease begins and RCS pressure reaches 1500 psig. Pressurizer level dropped during this period to 225 inches. A RCDT pressure spike of 11 psig is seen at this time and reactor building pressure increases from 1 to 3 psig.	1. Operating logs 2. Reactimeter 3. Plant strip charts

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
272	3 hrs. 16 min.	Answering Service attempts to reach Region I Duty Officer, but he has already left for Region I Office.	1. Region I notes
273	3 hrs. 18 min.	IRM current (~14) returns on scale, increasing.	1. Alarm printer
		PLANT STATUS	
274		After attempts to establish natural circulation failed, the operator started RCP 2B. However, based on a no flow indication RCP 2B was stopped after 19 minutes. Superheated steam/gas was present in the reactor vessel head and RCS hot leg. Both reactor coolant system hot leg temperatures were off scale high (i.e. greater than 620F). The RCS cold leg temperatures were about 375 ^o F for loop "A" and 330 for loop "B". OTSG "B" was isolated due to a suspected RCS to OTSG leak. OTSG "A" pressure control was by means of the power operated emergency main steam dump valve "A" (MSV-3A). An attempt was in progress to control pressurizer pressure and level with the EMOV.	1. Reactimeter 2. Plant strip charts
275	3 hrs. 20 min.	Answering Service signals Region I Duty Officer beeper but beeper reportedly did not alarm.	1. Region I notes
276		Operator secures borated water recirculation pump. (Believed to have been in operation prior to event to clean up borated water storage tank).	1. Alarm printer 2. Interview 111

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
277	3/28/79	ES manually initiated by operator. Makeup pump 1C starts. Makeup pumps 1A & 1C now running with MU-V-16 valves wide open. Pressure begins to drop.	1. Alarm printer
278		RCS loop "A" hot leg temperature drops, 800 ^o F to 710 ^o F.	1. Multipoint recorder
279		NOTE: Makeup pump 1B has been off since ES initiation at 2 minutes after the start of the incident.	
280		NOTE: RCS loop "A" hot leg temperature varies between 680 ^o F to 760 ^o F over next 6 hours. Loop "B" will follow loop "A", but about 60 ^o F hotter.	1. Multipoint recorder
281	3 hrs. 21 min.	IRM current (~14) drops off scale.	1. Alarm printer
282	~ 3 hrs. 24 min.	General Emergency declared by Station Manager.	1. Statement Plant Manager 2. Region I tapes
283	3 hrs. 26 min.	Pressurizer level and pressure dropping. Pressurizer high level alarm clears indicating 254". Pressurizer pressure has dropped to 1500 psig. Pressurizer surge line temperature has returned to normal at 581 ^o F.	1. Alarm printer 2. Plant strip charts
284	3 hrs. 27 min.	ES reset by operator.	1. Alarm printer
285	3 hrs. 30 min.	Borated water storage tank low level alarms received at 53.03' and 53.06'.	1. Alarm printer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	<u>3/28/79</u>		
286		NOTE: Calculations based on these BWST levels indicate the average net flow rate into the RCS from the BWST has only been about 75 GPM over this 3 1/2 hour period.	
287		Operator places RCS loop "B" cold leg temp. in analog recorder trend (#2).	1. Alarm printer
288	3 hrs. 31 min.	First code safety outlet temperature high alarm returns to normal at 192°F, indicating EMOV had been shut sometime prior to this.	1. Alarm printer
289	3 hrs. 32 min.	Makeup tank radiation level 3 R/hr and auxiliary building basement reported as flooded. Auxiliary building airborne activity at about 10E-8 uC/cc. Spent fuel demineralizer area monitor reading 250-900 mr/hr. SRM count rate found to have increased by about a factor of 3. (Above radiation level readings from hallway of elevation 305', auxiliary building. Basement referred to above is elevation 281', auxiliary building.	1. Radiation survey logs 2. Plant strip charts 3. Interview 7
290	3 hrs. 33 min.	Pressurizer high level alarm received at 267", rapidly increasing.	1. Alarm printer 2. Plant strip charts
291	3 hrs. 35 min.	Second code safety outlet temperature alarm returns to normal at 193°F. Operator starts emergency feedwater pump 2A. Pump discharge initially rises to shutoff head conditions. Operator has started pump to re-establish feeding OTSGs. Main condenser vacuum is seen to begin degrading.	1. Alarm printer 2. Plant strip charts
292	3 hrs. 37 min.	Makeup pump 1C is tripped; presumably by operator, based on high pressurizer level. Pump 1A is still in operation.	1. Alarm printer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
293	3 hrs. 38 min.	Answering Service signals Region I Duty Officer bepper again. Signal received by Duty Officer who is now in heavy traffic and close to office. He decides to proceed to office to answer call.	1. Region I notes
294	~ 3 hrs. 40 min.	Site calls Region I Office to inform them of declaration of General Emergency.	1. Telephone bills
295	3 hrs. 40 min.	Reactor building air sample line isolated after reported as blowing air into auxiliary building.	1. Radiation survey log
296	3 hrs. 41 min.	Code safety outlet high temperature alarms return at 202 ⁰ F and 205 ⁰ F. Rate of reactor building pressure increase indicates EMOV discharge path is again open, with EMOV is manual and block valve open. Increase in RCDT pressure also seen for approximately 20 minutes.	1. Alarm printer 2. Plant strip charts
297	3 hrs. 42+ min.	Not receiving response from Duty Officer. Answering Service calls Region I Deputy Director, in accordance with procedure, and learns the Deputy Director is enrouté to the Regional Office.	1. Region I notes
298	~ 3 hrs. 43 min.	At this time, there is an indication of rapidly increasing SRM count rate concurrent with a rapid pressure increase. This is the only time this occurs during the accident. All other increases in SRM count rate appears to correlate to periods of apparent increased voiding associated with decreasing RCS pressure and/or loss of heat sink. Further, the alarm printer indicates a proliferation of "BAD" incore flux and temperature indications. These indications are consistent with an assumption that a core geometry shift has occurred.	1. Alarm printer 2. Plant strip
299	~ 3 hrs. 43 min.	Operator begins raising "A" OTSG level back to 50% to 55% range.	1. Plant strip

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
			chart
			2. Reactimeter
300	3 hrs. 44 min.	Answering Service receives call informing Regional Office that a General Emergency has been declared.	1. Region I notes
301	~ 3 hrs. 45 min.	B&W Site Operations Manager is now in Unit 2 control room. Station Manager was in control room having arrived about 3 hours, 5 minutes into the event. Operators reported they had run a RCP as directed with 100 amp indication. Operators report that "many" pressurizer heaters are "out of commission."	1. Interview 53
301A		B&W Site Operations Manager establishes first contact with B&W offices, informing them of plant status.	1. B&W Telephone Log
302	3 hrs. 45 min.	Region I telephone operator takes over from Answering Service. Region I operator is informed of General Emergency and she notifies Reactor Operations and Nuclear Support Branch Chief.	1. Region I notes
303	~ 3 hrs. 50 min.	Region I Director informed of General Emergency. Reactor Operations (RO&NS) and Fuel Facility and Materials Safety (FF&MS) Branch Chiefs call site (informed that Site Emergency was declared at 6:45 a.m. and General Emergency at 7:45 a.m.).	1. Region I notes
304	~ 3 hrs. 54 min.	Operator secures or significantly reduces feed rate to "A" OTSG.	1. Plant strip chart 2. Reactimeter
305	3 hrs. 55 min.	Station Manager directed outside agencies be contacted to report	1. Interview 53

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	incident. Established open line to NRC:RI.	2. Statement Plant Manager
306	3 hrs. 56 min.	ES and reactor building isolation initiated by high reactor building pressure (~ 3.58 psig). Makeup pump 1C starts automatically, joining makeup pump 1A in injecting through now wide open MU-V16 valves. Intermediate closed cooling pumps 1A & 1B trip.	1. Alarm printer
307		Note: Reactor building pressure trace suggests EMOV is shut by operator at this point, but EMOV block valve is left open.	1. Plant strip charts
308	4 hrs.	ES and reactor building isolation defeated by operator. Operator re-starts intermediate closed cooling pumps to ensure RCP seal and letdown flow cooling.	1. Alarm printer
309	~ 4 hrs.	By this time, pressurizer level is approximately 380 inches with a reactor pressure of 1500 psig.	1. Reactimeter
310		Detector shielded with 2 inches of lead located in containment dome reading 200 R/hr.	1. Region I Incident Messageform
311		Further degradation of main condenser vacuum is noted.	1. Plant strip charts
312		Station Manager requests B&W Site Operations Manager and other senior supervisors caucus with him in Shift Supervisor's office. It was decided to try another RCP start since pressure was high enough to satisfy NPSH requirements.	1. Interview 53 2. Statement Plant Manager

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
313		NOTE: Nature of subsequent discussions were reported to consider current situation and options available for return to normal conditions; discussions were not retrospective to determine what actions or conditions resulted in the plant reaching its current status.	
314	4 hrs.	NRC Regional Director calls for activation of Regional Emergency Center and NRC:OIE Director called by Regional Director.	1. Region I notes
315	4 hrs. 1 min.	The hourly plant log typer displayed the following information as being applicable for this time: RCS Flow: 3 MLBH Loop "A" Th: ? °F Tc _a : ? °F Tc _a : ? °F Loop "B" Th: ? °F Tc _b : ? °F Tc _b : ? °F MU Flow: 8 GPM OTSG "A" Level: 228 inches OTSG "B" Level: 241 inches Condensate Flow: 711 KLBH	1. Log typer
316	4 hrs. 4 min.	Operator requests computer print RCS loop "A" hot leg temp. (offscale).	1. Utility typer
317		Operator requests computer print control rod 13-E percent withdrawal.	1. Utility typer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
318	4 hrs. 5 min.	Operator requests computer print control rod drive 3-M motor temperature.	1. Utility typer
319	4 hrs. 8 min.	Electricians and technicians had jumpered interlocks to allow RCP re-start. Operating staff had attempted to start RCPs in each loop. Successfully started RCP-1A, but the pump ran with a current less than 100 amps and without flow indication. Operating staff believed RCP was cavitating and tripped RCP-1A 37 seconds later.	1. Alarm printer 2. Statement Plant Manager 3. Interviews 53, 181
320		NOTE: Alarm printer is now running nearly 2 hours behind real time. Information presented here not typed and available to operator for his analysis until that later time. As a result of this delay time, the alarm printer is not used as an operational aid by the operators.	1. Alarm printer
321	4 hrs. 10 min.	Region I Incident Response Center activated. Communications established between Region I, IE:HQ and licensee.	1. IRACT tapes 2. Region I notes
322	4 hrs. 10 min.	Appears reactor building air cooling coil B placed in service by operator in attempt to halt building pressure increase, based on temperature transient indication on cooling water. (This is similar to that reported at ~ 71 minutes.)	1. Alarm printer
323	4 hrs. 17 min.	Operator secures makeup pumps 1A & 1C, leaving no makeup pumps running.	1. Alarm printer
324	4 hrs. 18 min.	Operator twice attempts unsuccessful starts of makeup pump 1A to re-establish normal RCP seal flow. Operator then shifts the control switch to the pull-to-lock position and this pump remains off for the remainder of the incident.	1. Alarm printer 2. Interview 61 3. Utility typer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
325		NOTE: From 4 hrs. 17 min. 22 sec. until 4 hrs. 21 min. 53 sec., no makeup pumps were operating, a period of approximately 4 1/2 minutes.	
326		NOTE: Based on interviews, it appears that the operators were monitoring the rise in building pressure and they placed the makeup pump selector switches in the "pull-to-lock" position. This action apparently was taken to prevent HPI during a time when the pressurizer appeared to be "solid". (RCS pressure about 1500 psig).	1. Interview GPU-1 2. Reactimeter 3. Plant strip charts
327	4 hrs. 19 min.	Third SFAS channel in the "A" train actuates on high reactor building pressure. With one channel in defeat, the two out of three logic of the "A" train is satisfied. Full ES and building isolation is initiated on the "A" train. Decay heat pump 1A starts and will run for 60 minutes. Intermediate cooling pump 1A trips. The fact makeup pump-1A does not attempt to start confirms that its control switch is in the pull-to-lock position.	1. Alarm printer
328		Eighteen seconds later, operator defeats ES and building isolation. Twenty-three seconds later operator restarts intermediate cooling pump 1B to maintain RCP seal and letdown flow cooling.	1. Alarm printer
329	4 hrs. 20 min.	Containment dome radiation monitor reads 600 R/hr.	1. Operating logs 2. Region I Incident Messageform
330	4 hrs. 22 min.	With the unsuccessful attempt to start makeup pump 1A, the operator	1. Alarm printer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
		starts pump 1B. This pump remains in operation for the remainder of the period covered by this sequence.	2. Interview 61
331	~ 4 hrs. 22 min.	Technicians lift leads on RCS loop "A" hot leg RTD to take reading (720 ⁰). Initially, temperature readings were not believed due to their magnitude and the fact that the readings were outside the calibrated range of the instruments. All involved site personnel felt the core was covered at this time.	1. Interviews 53, 121, 181 2. Statement Plant Manager
332	4 hrs. 24 min.	Pressurizer heater groups 1-5 are returned to normal. All pressurizer heaters are operable at this time.	1. Alarm printer
333	4 hrs. 26 min.	Letdown high temperature alarms received. (Appears operator may have increased letdown in excess of one cooling pump's capability).	1. Alarm printer
334		NOTE: Operator interviews indicate they experienced letdown flow oscillations of 40 GPM, peak to peak, indicating relief valve operation.	1. Interviews 9, 21
335	4 hrs. 27 min.	Operator starts makeup pump 1C. This pump remains in operation until 9 hrs. 4 min. after the start of the incident, feeding 250 gpm thru valves 16A & C, only, using 1B & 1C makeup pumps.	1. Alarm printer 2. Interview 61
336	4-5 hrs.	Station Manager had requested verification of incore T/C readings, which were indicating offscale. Technicians take incore T/C readings at terminal strips in control building. Readings indicate temperatures from 80 ⁰ F to 2620 ⁰ F. Indications were not considered reliable by the senior staff present, reportedly because the thermocouples were not safety grade components.	1. Technician Records 2. Interviews 121, 181 3. Statement Plant Manager

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
337	4 hrs. 31 min.	Group 10 of the pressurizer heaters trip. This group does not become operational again during the period covered by this sequence.	1. Alarm printer
338	4 hrs. 31 min.	Condenser vacuum pumps 1A & 1C secured. Plant staff was experiencing problems with Unit 1 auxiliary boiler which supplied turbine seals. Main condenser vacuum decreases, reaching atmospheric pressure within 15 minutes.	1. Alarm printer 2. Plant strip charts
339	4 hrs. 35 min.	Operator requests computer print 16 specific incore T/C reading. Output data ranges from 310 ⁰ F to offscale.	1. Utility typer
340	4 hrs. 36 min.	Letdown high temperature alarms clear.	1. Alarm printer
341	4 hrs 39 min.	NOTE: At this time, a telephone line between the IE:HQ Operations Center (OC) and the Region I Incident Response Center (IRC) was opened and remained open throughout the incident. The IRC had already opened one or more telephone lines to the Three Mile Island site to receive information, and the IRC relayed that information and questions between the OC and the site. Later in the day, between 8 1/2 to 9 hours after the start of the accident, these lines were tied together putting the OC, the IRC and the site on a common conference line.	1. IRACT tapes
342		Because the IRACT tapes were transcribed in half-hour intervals only approximate times of information exchanges can be given within this half hour interval. The initial information and question exchanges were for the purpose of obtaining operational and radiological status information	

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	which are redundant to the material in this sequence, and will not be repeated unnecessarily.	
343	4 hrs. 40 min.	Containment dome radiation monitor reads 1000 R/hr.	1. Region I Incident Messageform
344	4 hrs. 42 min.	EFW pump 2A turned off. Steam generator levels appear stable in desired range. Level changes over next 90 minutes in both "A" and "B" OTSGs appear to relate to heating of OTSGs thru steam condensing natural circulation. Neither "A" or "B" OTSGs would be steaming at this point.	1. Alarm printer 2. Plant strip charts 3. Reactimeter
345	4 hrs. 43 min.	Vacuum dropping in condenser as evidenced by feedwater turbine low vacuum trip.	1. Alarm printer 2. Plant strip charts
346	4 hrs. 46	Groups 4 and 5 of the pressurizer heaters trip and do not become operational for the remainder of the period covered by this sequence. No more than 10 heater groups are now operational.	1. Alarm printer
347	4 hrs. 47 min.	Operator requests computer print 11 specific incore T/C readings which range from 378°F to off scale; the latter indicated by "????".	1. Utility typer
348		NOTE: IRC relays 1000 R/hr dose rate to OC. This entry made to indicate that at this time, certain aspects of key information being relayed were current. In general, however, responses to questions asked about plant conditions were not responded to promptly because of the activities in the control room and the need to relay information, both between OC and IRC and the site,	1. IRACT tapes

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	as well as the relaying required within the control room.	
349	4 hrs. 59 min.	Information relayed to NRC:Region I office is that HPI flow is 250 gpm/leg and RCS pressure 1400 psig.	1. Region I Incident Message form
350	5 hrs	<p>General plant parameters at this point are:</p> <ol style="list-style-type: none"> 1. RCS pressure 1375 psig and generally stable; 2. Reactor building pressure approx. 4 psig and stable; 3. Pressurizer full; 4. Steam generator levels 40-60% and essentially steady; 5. Steam generator pressures - "B" decreasing slowly from 325 psig and "A" at 50 psig and increasing. 6. Hot leg temperatures > 620⁰F; 600⁰F and 660⁰F on multipoint recorder. 	<ol style="list-style-type: none"> 1. Strip charts 2. Reactimeter 3. Multipoint recorder
351	5 hrs.	Containment dome radiation monitor reaches 6000 R/hr.	1. Region I Incident Messageform
352	~ 5 hrs.	Licensee reported to Region I at 0900 (elapsed time = 5 hours) that the auxiliary building was isolated at 0800. Reported that "B" steam generator leak was noted at 0800 by vacuum pump monitor. Unit 2 control room placed on recirculation air.	<ol style="list-style-type: none"> 1. Region I Incident Messageform 2. Radiological.

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		sequence of events
352A		B&W attempts to fax list of information, required to make recommendations, without success.	1. B&W Telephone Logs
353	5 hrs. 1 min.	The hourly plant log typer displayed the following information as being applicable for this time: RCS Flow: 3 MLBH Loop "A" T _h : ?°F T _{ca} : ?°F T _{ca} : ?°F Loop "B" T _h : ?°F T _{cb} : ?°F T _{cb} : ?°F MUS Flow: 58 GPM OTSG "A" Level: 216 inches OTSG "B" Level: 241 inches Condensate Flow: 586 KLBH	1. Log typer
354	5 hrs. 10 min.	Emergency Control Station (ECS) moved to Unit 2 control room.	1. Radiological sequence of events 2. IRACT tapes
355	5 hrs. 12 min.	Operator places incore T/C 3-L computer point in digital cal.	1. Utility typer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	display.	
356	5 hrs. 15 min.	Based on first information recorded on Region I Incident Response Center (IRC) Tapes, the status of the plant reported to Region I is as follows: <ol style="list-style-type: none"> 1. MSL Safety Valves said to be shut. 2. OTSG "A" Level reported at 44%. 3. OTSG "B" Level reported at 66%. 4. OTSG are reported to be isolated. 5. RB Spray had not actuated yet. 	<ol style="list-style-type: none"> 1. Region I IRC tapes 1-1 2. Plant strip charts
357	5 hrs. 15 min.	Operator requests computer print RCS Pressures (1203, 1164 & 1126 psig) and pressurizer surge line temperature (303°F).	<ol style="list-style-type: none"> 1. Utility typer
358	~ 5 hrs. 15 min.	First instance of a request from OC thru IRC to site for specific information on steam generator levels and meteorological data.	<ol style="list-style-type: none"> 1. IRACT tapes
359	~ 5 hrs. 15 min.	Convinced that steam was in each loop, the licensee's staff decided to raise RCS pressure and collapse unwanted steam bubbles. They verified again that the EMOV was shut. (Recall that position "indication" shows valve position demand rather than actual position.) An increase in high pressure injection flow was directed; however, they were apparently concerned that if pressure went too high, it could cause code safeties to	<ol style="list-style-type: none"> 1. Interview 53 2. Statement Plant Manager 3. Tech. Supt.'s Log

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	be opened. They decided to control pressure at 2000-2100 psig using the EMOV block valve, with the EMOV open.	
360	5 hrs. 17 min.	Appears reactor building air cooler B started again.	1. Alarm printer
361	5 hrs. 18 min.	Building pressure starting to decay and RCS pressure rising. Appears EMOV block valve has just been shut to allow plant pressurization.	1. Plant strip chart 2. Reactimeter
362	5 hrs. 19 min.	Decay heat pump 1A turned off and pulled-to-lock.	1. Alarm printer 2. Analysis of next ES initiation
363	5 hrs. 24 min.	ES and reactor building isolation initiated in train "A", apparently the result of a contact race in a reactor building 4 psig pressure switch or a misaligned or dirty contact in a 63X1 relay, when the pressure switch is changing state as reactor building pressure decreases. Intermediate cooling pump 1A trips. The fact that decay heat pump 1A or makeup pump 1A do not start and makeup pump 1B doesn't trip confirm decay heat pump 1A and makeup pump 1A are still in pull-to-lock. Makeup pumps 1B & 1C are now injecting with the MU-V-16 valves wide open. The operator resets the channel, clearing the ES and reactor building isolation thirteen seconds later. The operator then restarts intermediate cooling pump 1A ten seconds later to ensure RCP seal and letdown flow cooling.	1. Alarm printer
364	~ 5 hrs. 29 mins.	At this point an engineer noted the indication of an overspeed trip condition on the Diesels (tripped at 30 min.) and questioned Diesel status. It was agreed to place the Diesel control switch in the MAINTENANCE EXERCISE position and reset the fuel racks. This rendered	1. Interview 195 2. Alarm printer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	the Diesels capable of manual starts; however, they remained incapable of any automatically initiated starts for the remainder of the accident.	
365	~ 5 hrs. 30 mins.	RCS pressure ~1500 psi	1. NRC inspector notes
366	~ 5 hrs. 30 mins.	During the OC-IRC information exchanges, some period of time (~ 15 min.) is spent clarifying information. This appears to some extent to be due to the quality of communications and to clarification of terminology and plant layout and configuration.	1. IRACT tapes
367	5 hrs. 31 min.	Group 3 of the pressurizer heaters trips and remains inoperable for the remainder of the period covered by this sequence. No more than 9 heater groups are now operational.	1. Alarm printer
368	5-6 hrs.	NOTE: Reactor building pressure peaked at 4.4 psig and trended downward to 1.6 psig during this period. SRM count rate showed a general decrease with no anomalies.	1. Plant strip charts 2. Reactimeter
369		NOTE: Appears vapor lock in "A" RCS Loop is significantly larger than in "B" RCS Loop, based on analysis of OTSG level and pressure trends.	1. Plant strip charts 2. Analysis
370	5 hrs. 35 min.	Licensee reports to Region I that: 1. Belief expressed that OTSG pressure transient probably lifted OTSG Safety Reliefs. 2. No one had been evacuated from site yet. 3. No consideration of offsite evacuation yet.	1. Region I IRC Tapes 1-1 & 2

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	<p>4. Licensee reported to have stated to have no significant leakage. (Apparently referring to radiological released)</p> <p>5. No significant offsite activity reported yet.</p>	
371		The EMOV and one code safety outlet temperature high alarm clear at 192°F and 193°F, respectively. Remaining code safety outlet temperature high alarm clears at 193°F, 4 1/2 minutes later.	1. Alarm printer
372	5 hrs. 39 min.	Temperature transient again indicated on reactor building air cooling coil B, suggesting it has again been placed in service.	1. Alarm printer
373	5 hrs. 40 min.	RCS pressures now read 1791 psig, 1781 psig and 1785 psig; increasing.	1. Alarm printer
374	5 hrs. 42 min.	RCS Pressure at 1950 psi, pressurizer level at 400 inches.	1. Plant strip charts 2. Reactimeter
375	5 hrs. 44 min.	RCS pressure rises to 2050 psi and exhibits oscillating behavior between 2050 psig and 2200 psig. EMOV and safety valve tailpipe temperatures alarm high (205-215°F). Operators are cycling EMOV block valve with the EMOV open to control pressure within limits dictated by plant staff.	1. Plant strip charts 2. Alarm printer 3. Tech. Supt.'s Log
376	~ 5 hr. 45 min.	Note: The IRACT tapes at this point show that a second line was opened to the IRC in addition to the primary line from the OC. This resulted in a different set of questions being asked to be relayed to the site (radiological matters) in addition to those being transmitted from the OC (operational matters). This produced some conflict in establishing priorities for the two	1. IRACT tapes

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	requests, but appears to have been resolved fairly promptly.	
377	5 hrs. 47 min.	Operator secures Intermediate Cooling Water Pump 1B.	1. Alarm printer
378	5 hrs. 51 min.	Temperature transient on reactor building cooling coil B experienced again.	1. Alarm printer
379	6-7 hrs.	NOTE: RCS pressure cycled between approximately 2050 and 2200 psig throughout the period. Reactor building pressure remained in the range of 1.8 to 2.4 psig. Pressurizer level at 400 inches.	1. Plant strip charts
380	6 hrs. 1 min.	The hourly plant log typer displayed the following information as being applicable for this time: RCS Flow: 3 MLBH Loop "A" Th: ? °F Tc: ? °F Tc _a : ? °F Loop "B" Th: ? °F Tc: ? °F Tc _b : ? °F MU Flow: 52 GPM OTSG "A" Level: 218 inches OTSG "B" Level: 236 inches Condensate Flow: 0 KLBH	1. Log typer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
381	6 hrs. 10+ min.	Airborne levels in Unit 2 control room require evacuation of all but essential personnel.	1. Region I IRC Tapes 1-2
382	6 hrs. 12 min.	ESC moved from Unit 2 control room to Unit 1 control room.	1. Radiological sequence of events
383	~ 6 hrs. 14 min.	Operator begins raising "A" OTSG level to 97% to 99%. Believed now using condensate pumps, only.	1. Alarm printer 2. Plant strip charts
384		NOTE: Reactor trip procedure requires raising OTSG levels to this range, if reactor restart is not anticipated within 4 hours.	1. Emergency Procedure 2202-1.1
385		Pressurizer heater groups 1 and 2 trip, but are reenergized in 27 seconds. Licensee reports to Region I: 1. 140 uc/cc gross β - γ activity in RCS. 2. Concerned "B" OTSG primary to secondary leak may lead to burp of activity from safety reliefs. Water level high.	1. Alarm printer 1. Region I IRC tapes 2-1 2. Region I IRC tapes 2-1
386	6 hrs. 15 min.	"Still injecting" and BWST level dropping, based on report from licensee to Region I.	1. Region I Incident Messageform 2. Region I IRC tapes 2-1

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
387	~ 6 hrs. 15 min.	NOTE: OC notified by IRC that evacuation of Unit 2 control room has been accomplished. Key events continue to be reported promptly but responses to information requests still require longer periods. At about this time, OC was told that the operators had "secured" HPI within 5 minutes after it was initiated following the turbine trip. Shortly thereafter, OC is informed by IRC that site communications will be interrupted while they change telephones to Unit 1 Control Room due to evacuation of Unit 2 Control Room.	1. IRACT Tapes
388	6 hrs. 17 min.	Licensee reports to Region I: 1. Unit 2 people having to don masks. 2. Using "A" OTSG for cooling. Staff believes natural circulation was working.	1. Region I Incident Messageform 1. Region IRC Tapes 2-1
389	6 hrs. 18 min.	Operator requests computer printout Sequence of Events Review.	1. Utility typer
390	6 hrs. 23 min.	Temperature transient again experienced on reactor building cooling coil B.	1. Alarm printer
391	6 hrs. 27 min.	Everyone moved to Unit 1 CR except essential personnel.	1. Region I IRC Tapes 2-1 & 2
392		NOTE: Information at this point will not be typed by alarm printer and available for operator analysis for nearly 2 hours 40 minutes.	1. Alarm printer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
393	~ 6 hrs. 30 min.	NOTE: OC informed of evacuation of non-essential personnel from site. OC also informed that first NRC team from Region I has arrived on site. Within next few minutes, questions from OC are being brought into Unit 2 control room from the Unit 1 control room by NRC inspectors. However, this process is delayed because of the air-borne radioactivity in the Unit 2 area. Shortly thereafter, OC decides to contact telephone operator to attempt to establish 3-way conference call between OC, IRC, and the site.	1. IRACT tapes
393A	~ 6 hrs. 30 min.	B&W offsite personnel receive update on plant status as relayed by Metropolitan Edison employee at simulator who has been able to get thru to site.	1. B&W Telephone Logs
394	6 hrs. 39 min.	Operator requests computer print RCS pressures (2010-2063 psig).	1. Utility typer
395	6 hrs. 40 min.	Level in "A" OTSG 82.5% and increasing. Operator still filling OSTG to 97% to 99% level.	1. Alarm printer
396	6 hrs. 50 min.	BWST level 37'. reactor building sump > 6'.	1. Transient log
397	6 hrs. 53 min.	Low vacuum trip, feedwater turbine "A" alarm received were steaming thru atmospheric relief from "A" OTSG.	1. Line printer 2. Statement Plant Manager
398		NOTE: Why this alarm was not received earlier is unexplained. Vacuum has been below setpoint for 2 hours.	
399		NOTE: Not many individuals actually left control room as result of evacuation order.	1. Interview 59

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
400	3/28/79	NOTE: Door to Unit 2 control room without automatic recloser. People failed to close door on their own, possibly compromising recirculating ventilation system.	1. Interview 59
401	6 hrs. 55 min.	Operator requests computer print OTSG parameters. "B" OTSG downcomer temperature ~ 430°F. "A" OTSG downcomer temperature ~ 265°F. "A" OTSG shell temperatures increase with elevation, 258°F at bottom to 441°F at top.	1. Utility typer
402		NRC inspector now on site reports trying to cooldown using "A" OTSG and atmospheric dumps. Possible bubble in both RC pump legs; 37 feet left in BWST. Pressurizer level > 400". Feeding "A" OTSG with emergency feed.	1. Region I Incident Messageform 2. Region I IRC tapes 2-2
403		Level in reactor building indicating > 6', which is maximum possible level indication.	1. Region I IRC Tapes 2-2
404		Noise level in Unit 2 control room reported to be high with 20+ people.	1. Interview 59
405	~ 7 hrs.	Station Manager in charge in Unit 2 control room with major decisions made in Shift Supervisor's office in conference with senior staff members. Once decision was made, operators were then directed to perform required manipulations.	1. Interviews 31, 59
406		Tour of auxiliary building found 10 R/hr at the Rad Waste Panel, water standing on the floor in areas with floor drains, and the auxiliary building sumps full.	1. Interviews 13, 59

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
407	3/28/79	Operators stated that during this period they were reluctant to start RCP for fear of vibration induced seal failure and resultant LOCA. They recognized they had a steam bubble in both loops. Personnel believed core was covered and considered potential of uncovering core as each option was reviewed. Operators were experiencing trouble maintaining adequate letdown flow. Concern for failure of EMOV block valve in open position may have lead to decision to use pressurizer vent valves for pressure reductions versus EMOV block valve.	1. Interviews 31, 59
408		Unit 2 control room in respirators. Communications in Unit 2 control room are hampered by respirators. Communication problems in Unit 2 control room lead some personnel to remove respirators for short periods.	1. Interviews 53, 31 2. Region I Tape 3-1
409	7 hrs.	Inspector at site reports licensee still feeding "A" OTSG and using atmospheric steam dumps.	1. Region I Incident Messageform
410		NOTE: OTSG samples had confirmed "A" OTSG was not contaminated. Had stationed man on roof to monitor for radiation in steam plume should it develop. Readings also confirmed no activity in "A" OTSG. "B" OTSG sample confirms have or had primary to secondary leak. "B" OTSG remains isolated thru remainder of accident.	1. Statement Plant Manager 2. B&W Telephone Logs
411	~ 7 hrs.	Operators controlling pressure using EMOV Block Valve (RC-V2) at ~ 2000 psi. Opening valve 3 min. for every 5-8 min. About 7½ hrs, shifted to pressurizer vent valve for fear of RC-V2 failure.	1. NRC Inspector notes
412	7 hrs. 1 min.	The hourly plant log typer displayed the following information as being applicable for this time:	1. Log typer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	<p>RCS Flow: 3 MLBH</p> <p>Loop "A"</p> <p>$T_h = ? \text{ } ^\circ\text{F}$</p> <p>$T_{ca} = ? \text{ } ^\circ\text{F}$</p> <p>Loop "B"</p> <p>$T_h = ? \text{ } ^\circ\text{F}$</p> <p>$T_{cb} = ? \text{ } ^\circ\text{F}$</p> <p>MU Flows = 49 GPM</p> <p>OTSG "A" Level = 351 inches</p> <p>OTSG "B" Level = 233 inches</p> <p>Condensate Flow = 0 KLBH</p>	
413	7 hrs. 4 min.	Unit 2 ventilation system isolated. NRC expresses concern that above may lead to ground level release.	<ol style="list-style-type: none"> 1. Region I Incident Messageform 2. Region I IRC Tapes 3-1
414		NOTE: State reportedly had urged shutdown of auxiliary building ventilation to limit further releases.	<ol style="list-style-type: none"> 1. Interview 59
415	7 hrs. 9 min.	EFW Pump 2A on. Operator appears to raise "A" OTSG level to 93%.	<ol style="list-style-type: none"> 1. Alarm printer 2. Plant strip charts
416	7 hrs. 11 min.	Operator requests computer print 14 separate incore T/C parameters; all indicate out of range.	<ol style="list-style-type: none"> 1. Utility printer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
417	7 hrs. 14 min.	Operator requests computer print the 3 pressurizer level differential pressures, and the one compensated pressurizer level all of which indicate full (> 400 ⁿ), and pressurizer temperature, which indicates about 342°F.	1. Utility printer
418	7 hrs. 15+ min.	100 ml RCS sample reads ~ 200 mr/hr. Translates to ~ 72 uc/ml.	1. Region I IRC Tapes 3-1
419	~ 7 hrs. 15 min.	NOTE: OC informed at this point that licensee planned to depressurize to utilize decay heat system. Then the person making this statement (unidentified) indicates he wants to confirm it, and goes off to do so. A specific confirmation or denial is not relayed back to the OC. For the next 2 hours, OC personnel appear to be under impression that DHR will be used until they learn, at approximately 9 hrs. 15 min, that licensee is floating CF tanks on the core.	1. IRACT Tapes
420	7 hrs. 16 min.	Operator requests computer print OTSG pressures (1 psig (A) & 302 psig (B)) and feedwater temperatures (287°F(A) & 415°F(B)).	1. Utility typer
421	7 hrs. 17 min.	EFW pump 2A off.	1. Alarm printer
422	~ 7 hrs. 20 min.	Operating staff was concerned EMOV block valve might fail. The EMOV block valve reportedly had a history of sticking. Appears it was decided to remove valve cycling frequency by utilizing wider pressure control band (1900 to 2100 psig).	1. Plant strip charts 2. Interview 1
423	~ 7 hrs. 26 min.	Operator requests computer print "B" OTSG parameters. OTSG shell essentially isothermal with slight rise in temperature with elevation.	1. Utility typer
424	~ 7 hrs. 30 min.	Operators indicated that at this time they believed that pressurizer	1. Interviews 31,

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	heaters were unavailable and concerned that EMOV block valve might fail open. Didn't appear to be getting anywhere so the group decided to drop pressure to get core flood tanks (CFT) to float on core.	53
425		Operators were controlling pressure with the HPI and EMOV. The staff was concerned that the HPI flow might be bypassing the core; since $T_h \sim 620$, $T_c \sim 220$ and the pressurizer water space RTD was relatively constant at 350°F . They wanted to make sure the core was covered; so they decided to depressurize in an attempt to get Core Flood initiation and ultimately to initiate DHR below 400 psig.	1. Interviews 53, 107 2. Statement Plant Manager
426		NOTE: Interviewee asserted that Steam Tables were referred to several times by control room personnel during sequence.	1. Interview 53
427		NOTE: BWST level had dropped 20', and at present rate of use, sources of clean water would be exhausted in matter of hours, requiring the use of dirty containment sump water.	1. Statement Plant Manager
428		NOTE: Calculations, based on level changes in the BWST over the last 90 minutes, indicate current volumes would support at least 10 hours of additional makeup at current rates.	1. Calculations
429	~ 7 hrs. 30 min.	Shift Foreman unlocks and closes breakers for DHR isolation valves (DH-V102A & B).	1. Interview 107
430	7 hrs. 30 min.	Operator open both the EMOV and its block valve to depressurize the RCS. Reactor building pressure shows rise from low point of 0.2 psig to 2.5 psig during this RCS depressurization.	1. Reactimeter 2. Plant strip charts

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
431	3/28/79	NOTE: Station Management asserted that operators were directed to maintain HPI during depressurization.	1. Statement Plant Manager
432	7 hrs. 40 min.	"B" OTSG is bottled, but it appears that it may be necessary to relieve its pressure.	1. Region I IRC Tapes 4-1
433	7 hrs. 40 min.	Region I inspector reports that licensee thinks they have release under control.	1. Region I Incident Messageform
434	7 hrs. 42 min.	ESF A&B bypassed. Reactor pressure at that time is 1735 psig. Believed to be operator action to prevent inadvertent actuation of HPI during depressurization.	1. Alarm printer 2. Plant strip charts
435	7 hrs. 44 min.	Pressurizer heater groups 1 and 2 trip, but are reenergized within 2 seconds.	1. Alarm printer
436	7 hrs. 45 min.	Reactor coolant drain tank leakage transfer pumps 9A & 9B trip.	1. Alarm printer
437		NOTE: These pumps are started by operators to recirculate RCDT water for temperature control and/or to transfer water to the in-service bleed tank. NOTE: Following initial reactor building isolation, transfer of water to the bleed tanks would be impossible unless operators reopened intervening containment isolation valves. No interview has indicated these valves were reopened.	1. System description
439	~ 7 hrs. 45 min.	B&W Site Operations Manager relays site status information to B&W	1. Interviews 53,

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	at Lynchburg via home phone of another B&W site representative.	67 2. Mgr of Nuclear Services phone log
440	~ 7 hrs. 45 min.	NOTE: Operational data is relayed to OC. This is the first operational data relayed to OC in more than 1/2 hour. At about this same time there is a discussion in the background (location unknown) concerning injection at maximum rate despite the risk of blowdown. This view, however, is not relayed to the site.	1. IRACT tapes
441	7 hrs. 50 min.	Pressurizer heater groups 1 and 2 trip and remain in a tripped state for 2 hrs. 16 min. At this time, there are no more than 7 heater groups operable.	1. Alarm printer
442		NOTE: Throughout depressurization, pressurizer level did not change from 400 inch indication. Reactor pressure decrease also shows discontinuity at 1400 psig at about this time; this indication may be result of operator shutting the EMOV block valve and opening the pressurizer vent valve to continue pressure decrease. The drop in rate of increase in reactor building pressure further suggests a change in blowdown technique.	1. Reactimeter 2. Plant strip charts
443	7 hrs. 52 min.	Operator requests computer printout Sequence of Events Review.	1. Utility typer
444	7 hrs. 55+ min.	Using atmospheric dumps now. Unit 1 reported to be in Hot Standby.	1. Region I IRC Tapes 4-2
445	~ 8 hrs.	Operator requests computer print pressurizer parameters. Surge line temperature seen to be increasing, demonstrating flow thru the Pressurizer.	1. Utility typer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
446	~ 8 hrs.	Station Manager directed, purportedly over his objection, by the Vice President-Generation to be prepared to meet with the Lt. Governor in Harrisburg. The meeting was apparently established after a telephone call from the Lt. Governor to the President of the company. The Station Manager directed the Superintendent-Technical Support to collect technical materials prior to the meeting.	1. Interviews 71, 91
447	8+ hrs.	Letdown cooler 1B outlet temperature high alarm received. As the operators reduced pressure, they had even greater problems maintaining adequate letdown flow. Personnel reportedly hesitated to take corrective action because of their concerns relative to whether NRC would tolerate the deliberate overexposure required to manipulate the valves necessary to regain adequate flow.	1. Interview 31 2. Alarm printer
448	8 hrs.	NOTE: OC is notified of arrival of second Region I team at the site.	1. IRACT tapes
449	8 hrs. 1 min.	The hourly plant log typer displayed the following information as being applicable for this time: RCS Flow: 3 MLBH Loop "A" T _h : ? °F T _{ca} : ? °F T _{ca} : ? °F Loop "B" T _h : ? °F T _{cb} : ? °F	1. Log typer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	<p style="text-align: center;">T_{cb} ? °F</p> MU Flows: 49 GPM OTSG "A" Level: 373 inches OTSG "B" Level: 228 inches Condensate Flow: 0 KLBH	
450	8 hrs. 12 min.	High level indicated on core flood tank "A" (13.3 feet of water). RCS pressure at that point was approximately 800 psi.	1. Alarm printer 2. Plant strip charts
451	8 hrs. 15 min.	Letdown cooler 1A outlet temperature high alarm received.	1. Alarm printer
452		NOTE: Staff recognized letdown was providing some cooling.	1. Transient log
453	~ 8 hrs. 15 min.	NOTE: OC requests that licensee be asked if they considered blowing down the primary system, and if they considered bumping the RCP's. It appears that this is the first instance of the OC asking questions of a planning nature rather than a status nature. IRC agreed to relay request to Inspector at site.	1. IRACT Tapes
454	8 hrs. 17 min.	Over next 6 minutes, both code safety outlet temperature high alarms clear at 193 ⁰ F; believed associated with operator shutting EMOV block valve at some time earlier. Appears operators are continuing to drop pressure using the pressurizer vent valves.	1. Alarm printer 2. Plant strip charts
455	8 hrs. 20 min.	Pressurizer level is still off scale high. Inspector believes bubble exists in OTSG "candy canes".	1. Region I IRC Tapes 4-2
456	~ 8 hrs. 20 min.	NOTE: OC informed of hot leg temperature of 620° F. (No comment is	1. IRACT tapes

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	made that this is the limit of display for this detector) OC requests information be provided on fuel assembly outlet thermocouples. There appears to be a discussion in the background of why the licensee throttled back on HPI flow (as understood by OC) but no question is relayed to site.	
457	8 hrs. 25 min.	Operator requests computer print pressurizer temperature (350°F) and RCS pressure (618-640 psig). Available heaters appear incapable of establishing saturation conditions in pressurizer with existing flow thru it.	1. Utility typer
458	8 hrs. 30 min.	Operator requests computer print OTSG parameters. "A" OTSG shell temperatures range from 215°F at bottom to 404°F at top. "B" OTSG shell temperatures range from 401°F at bottom to 492°F at top.	1. Utility typer
459	~ 8 hrs. 30 min.	NOTE: OC is informed that "B&W representative says please do start reactor coolant pumps." We don't know what the status is and we may end up blowing the seal...." OC also informed HPI flow to core is 400 gpm. Minutes later, a direct connection between OC and the site is established.	1. IRACT tapes
460	8 hrs. 31 min.	Operators start decay heat closed cooling water pumps in preparation for initiation of DHR.	1. Alarm printer
461	8 hrs. 40 min.	Operator requests computer print 25 separate incore T/C reading. Printed data ranges from 322°F to 635°F, with the majority reading offscale.	1. Utility typer
462	8 hrs. 41 min.	RCS pressure reaches 600 psig, which is equivalent to the nominal gas pressure maintained in the core flood tank (CFT) nitrogen cover gas.	1. Reactimeter 2. Plant strip

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		charts
			3. NRC Inspector notes
463		NOTE: Throughout the entire period when the core flood tanks were floating on the reactor coolant system, operators report that the tank levels did not drop more than a few feet, thus eliminating any possibility that portions of the nitrogen cover gas used in the tanks entered the reactor coolant system. An operator reported refilling the CFT nitrogen from a pressure of 400 psig to 600 psig at approximately 17 hours after the start of the accident.	1. Interviews 114, 119
464	8 hrs. 43 min.	32' BWST, RCS pressure 576 psig.	1. Region I Incident Messageform 2. Region I IRC Tapes 5-1
465	8 hrs. 45 min.	Staff decides not to open the EMOV as method of cooldown. Concerned "where will we be when core flood is gone?"	1. Transient logs
466	8 hrs. 55 min.	"A" core flood tank high alarm clears at 13.13'.	1. Alarm printer
467	8 hrs. 58 min.	Operator requests computer print RCS pressure (483 psig) and pressurizer temperatures (350°F).	1. Utility typer
468	~ 9 hrs.	Were feeding "A" OTSG with condensate pumps and still steaming out atmospheric reliefs. State had made several calls requesting releases be secured, each time more militant. Inspector pointed out to Station	1. Interview 26 2. IRACT tapes

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	Manager that source of water was probably contaminated. Station Manager may have been unaware that OTSG feed had been shifted from the condensate storage tanks to the main condenser.	
469	9 hrs.	Operator requests computer print OSTG shell temperatures. "A" OSTG is cooling down, all levels. "B" OSTG temperatures are holding.	1. Utility typer
470		General plant parameter conditions:	
		1. RCS Pressure 500 psig.	1. Reactimeter
		2. Reactor Building pressure 2.6 psig.	2. Plant strip charts
		3. Pressurizer level 400 inches.	3. Multipoint recorder
		4. Steam Generator "A" - 100% level indicated on operating range instrument; indicated pressure 0 psig.	
		5. Steam Generator "B" - 60% level and 250 psig (pressure slowly decreasing).	
		6. RCS Hot Leg Temperatures per multipoint recorder are 605 ^o F and 665 ^o F.	
471	~ 9 hrs.	Inspector reports from site that: Unit 1 control room going on masks. Excess personnel moved to Observation Center.	1. Region I Messageform 2. Region I IRC tapes 5-1

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
472	9 hrs. 1 min.	<p>The hourly plant log typer displayed the following information as being applicable for this time:</p> <p>RCS Flow: 3 MLBH</p> <p>Loop "A"</p> <p> T_h: ? °F</p> <p> T_h: ? °F</p> <p> T_{ca}: ? °F</p> <p> T_{ca}: ? °F</p> <p>Loop "B"</p> <p> T_h: ? °F</p> <p> T_h: ? °F</p> <p> T_{cb}: ? °F</p> <p> T_{cb}: ? °F</p> <p>MU Flows: 97 GPM</p> <p>OTSG "A" Level: 380 inches</p> <p>OTSG "B" Level: 226 inches</p> <p>Condensate Flow: 0 KLBH</p>	1. Log typer
473	9 hrs. 4 min.	Makeup pump 1C tripped by operator. Pump 1B still operating.	1. Alarm printer
474	9 hrs. 8 min.	Have stopped makeup from the BWST and are floating RCS in the CFTs.	1. Transient log
475	9 hrs. 9 min.	Operator places "A" loop cold leg temperature in analog recorder trend (#2).	1. Alarm printer
476	~ 9 hrs. 15 min.	VP directs Site Manager to secure atmospheric steam dump. Repairs to the instrument air line on the auto-reject valve had allowed operators to reestablish normal hotwell level. Log indicates "Electromatic relief shut."	1. Interview 61 2. Statement Plant Manager 3. Transient logs

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
477	3/28/79	NOTE: Reactor building pressure trace indicates all pressurizer blowdown paths were isolated about 8 minutes earlier.	1. Plant strip charts
478		Region I informed that: 1. Condenser is isolated. (Thus not available as a heat sink.) 2. Reported Harrisburg TP Exit at 25 mr/hr using uncalibrated instrument. 3. Core flood tanks have partially injected. 4. BWST now at 31 feet. 600 psig in RCS. 5. Hoping core flood injection will drop temperature allowing use of decay heat removal (DHR) system. 6. General plant announcement heard, "SHUTTING DOWN ATMOSPHERIC DUMPS".	1. Region I Incident Messageform 1. Region I IRC tapes 5-2
479		NOTE: With the termination of the use of the atmospheric steam dump, the only cooling mechanism available to the licensee was the injection of relatively low temperature water from the BWST into the RCS. This condition lasted until approximately 14 hrs. after the start of the accident when a heat sink was reestablished by drawing a vacuum in the condenser and steaming the "A" OTSG through the condenser steam dump valves. Thus a heat sink was not available for a period of approximately 4 3/4 hours.	
480	~ 9 hrs. 15+ min.	Personnel felt core was demonstrated to be covered when CFT level dropped	1. Interviews

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	only slightly upon dropping RCS pressure below CFT blanket pressure. They attributed the inability to drop below 450 psig as sign they had reached saturation pressure of loops. They then would be unable to depressurize further without some mechanism for cooling those loops. Personnel apparently did not realize at the time that both CFT flow paths have large loop seals which preclude assurance that core is covered.	31, 53 2. Transient logs 3. NRC Inspector notes 4. Burns & Roe drawings
481	9 hrs. 17 min.	EMOV outlet temperature high alarm clears at 193 ⁰ F. Core flood tanks indicate slow discharge. No indications of changes seen on RCS pressure or pressurizer level instrumentation. Pressurizer still at 400 inches. Reactor building pressure at 2.6 psig and decreasing. Logs indicate " <u>Stopped HPI</u> " (Initiation time not known - not confirmed by line printer).	1. Operating logs 2. Plant strip charts 3. Reactimeter 4. Alarm printer
482	~ 9 hrs. 17 min.	NRC inspector reports atmospheric steam dump closed.	1. Region 1 Incident Messageform
483		NRC expresses concern that leaving core flood tank isolation valves open may lead to injecting N ₂ into vessel. This concern was reiterated at approximately 9 hrs. 30 ² min. In both cases, it was responded to be indicating the licensee's conclusion that N ₂ injection was not possible, based on system design and plant conditions.	1. Region I IRC Tapes 6-1 2. IRACT tapes
484	9 hrs. 20 min.	Letdown cooler 1A outlet temperature high alarm clears.	1. Alarm printer
485	9 hrs. 21 min.	Received EMOV outlet temperature high alarm at 220.4 ⁰ F; this alarm believed to be associated with opening the EMOV block valve at this time.	1. Alarm printer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
486	9 hrs. 24 min.+	Operator places "A" & "B" OTSG pressurizer and RCDT parameters in 2 minute trend.	1. Utility typer 2. Alarm printer
487	9 hrs. 30 min.	Reactor building pressure 1.6 psig. SRM count rate continuing to show slowly increasing level.	1. Plant strip charts 2. Operating logs
488		Region I inspector reports from site to Region I that State of Pennsylvania was concerned about steam dump. (At this time, atmospheric steam dump had been secured for approximately 15 minutes. This message appears to be for the purpose of explaining why the dump was secured).	1. Region I IRC tapes 6-1
489	9 hrs. 32 min.	EMOV outlet temperature high alarm clears at 193 ⁰ F; alarm is believed to be associated with earlier closure of the EMOV block valve by an operator.	1. Alarm Printer
490	9 hrs. 35 min.	Region I inspector, in response to question, states to Region that pressurizer relief valve is closed.	1. Region I Incident Messageform 2. Region I IRC tapes 6-1
491	9 hrs. 40 min.	Operator starts intermediate closed cooling pump 1B, clearing high temperature alarm on letdown cooler 1B outlet.	1. Alarm printer
492	9 hrs. 45 min.	Region I inspector reports to Regional Office that DHR will be initiated when RCS pressure gets to about 350 psig.	1. Region I Incident Messageform

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
493		<p>Inspector reports general status to Region I as follows:</p> <ol style="list-style-type: none"> 1. Rx Press ~ 500 psig. 2. Temperature ~ 250⁰F. 3. Pressurizer being vented to vent header. 4. Normal letdown in operation. 5. Pumping bleed tank to core flood tanks to complete water cycle. 	<ol style="list-style-type: none"> 1. Region I IRC tapes 6-2 2. NRC Inspector notes
494		<p>NOTE: The path described here by the inspector was not confirmed by any interviews conducted. It appears to be a misunderstanding on the part of the inspector.</p>	
495	9 hrs. 49+ min.	EMOV outlet temperature high alarm received at 226 ⁰ F. RCDT and pressurizer temperature transient indicate EMOV block valve is now open.	<ol style="list-style-type: none"> 1. Alarm printer 2. Utility typer
496	9 hrs. 50 min.	ES actuation on reactor building isolation by high building pressure (4 psig). Building pressure experienced spike to 28 psig (indicated) tripping reactor building spray pressure switches (nominal 30 psig set-point) some 6 seconds later. Building spray pumps start and the 30 psig actuation clears within 4 seconds. RCS pressure indication shows rapid 40 psi dip (probably due to increased pressure on RCS pressure detector reference leg) and recovery to 500 psig at that same time. Makeup pump 1C starts (1B already operating and 1A in pull-to-lock). Decay heat pumps 1A and 1B start, intermediate closed cooling pumps 1A and 1B trip. Code	<ol style="list-style-type: none"> 1. Alarm printer 2. Plant strip charts 3. Interview 61 4. Statement Plant Manager 5. Utility typer 6. Interviews 111, 119 7. Operations log

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	safeties and RCP air temperatures alarm high. Operator defeats building isolation and restarts intermediate closed cooling pumps 1A & 1B to maintain RCP seal water and letdown cooling.	
497		Operator reports that the alarm occurred simultaneous with his opening of the EMOV. Shift Supervisor believed he had a "noise" or electrical cross connection which yielded "false" signal. Supervisor requested aid from instrumentation people. Some question exists as to whether this information was passed to management onsite. Another Shift Supervisor, then present in the control room and directing console activities, stated that he saw the spike occur in coincidence with the opening of the EMOV. He did not know what caused it, but he believed it was a real pressure spike. He stated that only after he was convinced that he no longer needed the spray did he secure the RB Spray Pumps. This would appear to explain why the pumps are not secured until 6 minutes later.	1. Interview TMI-226, 232
498		The Supervisor stated that he recommended to the Station Manager that the EMOV no longer be cycled because of the rapid rising building pressure when it was opened.	
499		Oxygen analysis days later indicate combustion has occurred. The Station Manager recalls hearing a "thump", but attributed it to a change in ventilation damper position. Recognition that hydrogen combustion has occurred, will not come until the next day as data is analyzed.	1. Reactor building atmospheric sample results 2. Interview 3. Statement Plant Manager
500		NOTE: Review of IRACT and Region I IRC tapes shows that this event was not brought to the attention of NRC Management.	1. IRACT tapes 2. Region I tapes

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
501	9 hrs. 51 min.	Makeup pump 1C is tripped; probably by operator, since pressurizer level indicates full scale. 480V motor control centers (MCCs) 2-32A & 42A feeder breakers trip along with leakage closed cooling pumps 2A & 2B supply breakers.	1. Alarm printer
502		NOTE: Leakage closed cooling pumps 2A and 2B receive power from the 480V MCCs which lost power.	1. Burns & Roe drawings
503	9 hrs. 52 min.	EMOV and one code safety outlet temperature high alarms rapidly clear together. The EMOV outlet showing a 45 ^o F drop in 3 minutes; probably the result of building spray quenching. The EMOV outlet alarm appears again within 20 seconds clearing again in 2 minutes, showing a temperature swing of nearly 30 ^o F, further supporting conclusion that rapid clearing due to actuation of spray.	1. Alarm printer 2. Utility typer
504	9 hrs. 55 min.	Unit 1 CR still in respirators.	1. Region 1 IRC tapes 7-1
505	~ 9 hrs. 55 min.	Station Manager and Superintendent-Technical Support leave Control Room to meet Vice President-Generation (who left from Observation Center) at the North Gate to proceed to the Lt. Governor's Office. Emergency Director Designee (Unit 2 Superintendent) was directed to maintain status, if possible, during their absence. Station Manager was fitted with a beeper to permit him to be contacted in the event of a change in conditions. Upon arrival at the Lt. Governor's Office, the Superintendent-Technical Services called the plant and remained on the phone approximately 15 minutes after the meeting started. It appears the meeting started at 2:30 p.m. (10.5 hours after the start of the accident) and lasted 30-45 minutes.	1. Interviews 91, 171, 53, 27 2. Chronological log in Lt. Governor's Office. 3. Control Room muster log

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
506	9 hrs. 55 min.	Pressurizer heater group 8 trips and remains tripped for the remainder of the incident. At this time, no more than 6 heater groups are operable. The remaining code safety outlet temperature high alarm now clears.	1. Alarm printer
507	9 hrs. 56 min.	Reactor building spray pumps shut down and placed in pull-to-lock by operator to secure NaOH washdown of building when pressure drop indicates no longer needed. It appears these switches were returned to normal within minutes. Operators close NaOH supply valves and the reactor building spray line valves.	1. Alarm printer 2. Interviews 61, 111 and 119
508	9 hrs. 57 min.	Core flood tank "A" level alarms indicate varying levels over the next 1 1/2 hours, between a low of 12.3 feet to a high level at or above 14 feet. During this period RCS pressure slowly increased to 560 psig. These alarms continued intermittently over a period of approximately 1 1/2 hours. No core flood tank "B" level alarms were received on the alarm printer during the incident.	1. Alarm printer 2. Plant strip charts
509		Operator secures decay heat pumps 1A & 1B. Pressure and temperature were still too high to initiate DHR and pumps were just running on recirculation.	1. Alarm printer
510	~ 10 hrs.	B&W at Lynchburg relays message licensee to establish at least 400 gpm of HPI flow. This done in two ways: 1. B&W Manager of Project Management contacts GPU Vice President. 2. Unit 2 Operations Supervisor (still at Lynchburg) calls Unit 1 control room to relay message to Unit 2.	1. Interview 88

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
511	~ 10 hrs.	NOTE: OC expresses their concern that continued injection will prevent discharge of the core flood tanks, with the result that the system will remain above the DHR interlock for a long period. OC asks how licensee is going to get the pressure down so that DHR can be put into use.	1. IRACT Tapes
512	10 hrs. 1 min.	EMOV and two code safety outlet temperature high alarm received again. RCDT and pressurizer temperature swings indicate EMOV block valve reopened.	1. Alarm printer 2. Utility Typer
513	10 hrs. 2 min.	The hourly plant log typer displayed the following information as being applicable for this time: RCS Flow: 3 MLBH Loop "A" T _h : ? °F T _h : ? °F T _{ca} : ? °F T _{ca} : ? °F Loop "B" T _h : ? °F T _h : ? °F T _{cb} : ? °F T _{cb} : ? °F MU Flows: 113 GPM OTSG "A" Level: 395 inches OTSG "B" Level: 225 inches Condensate Flow: 0 KLBH	1. Log typer
514	10 hrs. 3 min.	Operator requests computer printout Sequence of Events Review.	1. Utility typer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
515	10 hrs. 6 min.	Pressurizer heater groups 1 and 2 return to normal, but trip again within 1 minute 54 seconds.	1. Alarm Printer
516	10 hrs. 15 min.	Status of plant as reported to NRC: RCS Pressure ~ 500 psig. T 230°F Suspect Bubble in Loops Reported to be venting pressurizer	1. Region I IRC tapes 7-2 2. Utility Typer
517	10 hrs. 24 min.	One code safety outlet temperature high alarm clears.	1. Alarm printer
518	~ 10 hrs. 26 min.	General status as reported to NRC: 1. Licensee reported to believe release coming from auxiliary building's floor drains. Reading 1-2 R/hr in area of rad waste control panels. 2. All TMI-2 water tanks are full. 3. Going to dump TMI-2 neutralizing waste tank to TMI-1 miscellaneous waste tank, then dump that water to floor of TMI-1 bleed tank room (apparently this technique of shifting water had been used year before), then pump TMI-2 auxiliary building sumps to various TMI-1 and 2 tanks. 4. Some places in auxiliary building water was 6-8" deep, usually in areas of floor drains.	1. Region I IRC tapes 8-1
519	10 hrs. 28 min.	Rapid drop in RCS "A" loop outlet temperature noted and operators believe bubble has moved. Operators believe bubble collapse resulted from their actions of injecting heavily thru MU-V16C, only.	1. Operating logs 2. Plant strip charts 3. Utility typer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		4. Statement Plant Manager 5. NRC Inspector notes
520		NOTE: Although hot leg temperature came on scale, Tav indication was still erroneous due to cold leg temperature remaining below 520°F.	
521		Pressurizer level showed slight decrease to 380 inches and then began to rise again to 400 inches. This level change coincides with hot leg temperature coming on scale, but is a lesser magnitude than would be expected under normal conditions for the indicated hot leg temperature change. Appears pressurizer heaters have finally established saturation conditions in the pressurizer, blowing pressurizer water to the loop	1. Reactimeter 2. Utility typer
522		"A" OTSG pressure begins to increase following reported collapse of bubble in "A" loop.	1. Interviews 31, 127 2. Reactimeter 3. Utility typer
523	~ 10 hrs. 30 min.	NOTE: During conversations between the OC and IRC, the OC is informed of contaminated water being on the floor of the auxiliary building. This is the first reference to this condition found in the telephone tapes.	1. IRACT tapes
524	10 hrs. 32 min.	Makeup pump 1C started by operator. Pumps 1B and 1C now operating, 1A in pull-to-lock.	1. Alarm printer
525	10 hrs. 33 min.	Pressurizer heater groups 1 and 2 return to normal, but trip again	1. Alarm printer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	within 6 1/2 min. They are returned to normal within another minute. There are now not more than 8 operable pressurizer heater groups.	
526	10 hrs. 36 min.	Makeup pump 1C secured by operator.	1. Alarm printer
527	10 hrs. 38 min.	Trend in RCDT and pressurizer temperatures, and failure to receive the one cleared code safety outlet temperature alarm, suggest operators were venting pressurizer with vent valves versus the EMOV.	1. Utility typer
528	~ 10 hrs. 41+ min.	1. All the auxiliary building sumps are full. 2. A Region I HP indicated his tour of auxiliary building identified no visible leaks, just water on floor.	1. Region I IRC Tape 8-2
529	10 hrs. 45 min.	Operator notes that "A" loop hot leg temperature goes off scale high and returns again. Pressurizer level starts steady decrease to 180 inches from 400 inches (minimum level reached at about 11 hours 24 minutes).	1. Interview 2. Reactimeter 3. Utility typer
530	10 hrs. 45 min.	NOTE: OC repeats its request to site for status of licensee's plans to reduce pressure and use DHR system.	1. IRACT tapes
531	~ 10 hrs. 57 min.	Region I inspector reports current status to be: $T_h = 530^{\circ}\text{F}$ $T_c = 236^{\circ}\text{F}$	1. Region I IRC tape 9-1 2. Utility typer 3. Interview 111
532		RCS press 418 psig; plan to go into Decay Heat Removal via BWST; will	

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	use flow path from BWST to DH Removal Pumps to core.	
533	11+ hrs.	Licensee plan reported by inspector to Region I is to initiate low pressure injection followed by normal hot leg cooldown using DHR.	1. Region I IRC tape 9-1
534		NRC Inspector notes that pressurizer level being established and venting of pressurizer has been stopped.	1. NRC Inspector Notes
535		NRC:HQ requests clarification of 16-18 hour estimate to get on DHR. Told (1) cooling down at 3 ^o F/hr; (2) that there is an interlock on DHR valves at 430 psig; (3) that plant currently at 450 psig; and (4) that to be conservative they had assumed they would wait to shift over to DHR until they arrived at 400 psig. Thus 18 hr. estimate.	1. Region I IRC tape 9-2
536	~ 11 hrs.	Note: Telephone contact with the site is lost for approximately 5 minutes. When contact is established, it will be with the Unit 1 Shift Supervisor. This will be the first time that the OC has had directed contact with a licensee representative.	1. IRACT tapes
537	11 hrs. 1 min.	The hourly plant log typer displayed the following information as being applicable for this time: RCS Flow: 3 MLBH Loop "A" Th: 524 ^o F Tc: ? ^o F Tc ^a : ? ^o F Loop "B"	1. Log typer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	Th: ? °F Tc _b : ? °F Tc _b : ? °F MU Flow: 95 GPM OTSG "A" Level: 370 inches OTSG "B" Level: 224 inches Condensate Flow: 0 KLBH	
538	11 hrs. 3 min.	Operator places RCS loop "B" cold leg temp. into analog recorder trend (#2).	1. Alarm printer
539	11 hrs. 10 min.	Operators authorized to remove respirators in control room. Personnel in control room had to use respirators for approximately 5 hours.	1. Operating logs 2. Region I IRC tapes 9-1 3. Interview 36
540	~ 11 hrs. 10 min.	Loop "A" cold leg temperature starts rapid increase from ~ 256°F to ~ 400°F. Loop "A" hot leg temperature starts rapid increase from ~ 520°F to ~ 590°F. RCS pressure is increasing. Pressurizer level drops nearly 200" in 15 minutes. Change in rate of reactor building pressure drop indicates all blowdown paths from the pressurizer have been isolated.	1. Reactimeter 2. Utility typer 3. Plant strip charts
541	~ 11 hrs. 12 min.	Inspector reports to region I that: 1. Feed coming from condensate storage. 2. Steaming out of atmospheric reliefs. 3. Reported no radioactivity at these reliefs.	1. Region I IRC tape 9-2

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	4. Using atmospheric reliefs since have no condenser vacuum. Lost turbine seals after no longer able to get steam from Unit 1.	
542		Note: Actually items 2 and 4 are not correct in this report.	
543		The licensee is not now cooling down OTSGs. They have stopped because contamination. (Note: In fact, use of the atmospheric dumps had been terminated for almost 2 hours at the time of this report.) RCS pressure begins upward trend from about 415 psig. "A" OTSG pressure begins to decay due to vapor lock compression.	1. Region I IRC tape 9-2 2. Utility typer 3. Reactimeter
544	~ 11 hrs. 15 min.	Commenced efforts to draw vacuum in condenser.	1. Interview 31
545	11 hrs. 15 min.	Status report to NRC: Suction from BWST, DH-V5A and B now open, MUP running, supplying water to RCS via 16 valves.	1. Region I IRC tape 9-2
546	11 hrs. 15 min.	Pressurizer high level alarm clears at 254" with level dropping. Pressurizer heaters have again established saturation conditions and are blowing water to the loop.	1. Alarm printer 2. Utility typer
547	11 hrs. 19 min.	Pressurizer low level alarm received with level dropping rapidly. Makeup pump 1C started by operator and runs for less than 10 minutes. Operator reportedly wanted to prevent pressurizer heater trips on low level.	1. Alarm printer
548	11 hrs. 25 min.	EMOV outlet temperature finally returns to normal range.	1. Alarm printer
549	11 hrs. 27 min.	Operator requests computer print EMOV and code safety outlet temperatures; all indicate below alarm point.	

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
550	~ 11 hrs. 28 min.	Only one 16 valve open. Pressurizer level 175" (Per Operator in Unit 1 control room speaking to Region I Office - potential time lag). Periodically bumping EMOV. Worried continued bumping of relief might cause it to fail; Failure mode uncertain. Not using OTSG cooling at all at this point.	1. Region I IRC tape 10-1
551	11 hrs. 28 min.	Operator stops makeup pump 1C. Level in the pressurizer had started increasing again.	1. Alarm printer 2. Plant strip charts
552	11 hrs. 29 min.	Pressurizer level begins steady increase to 400 inches from 180 inches, reaches 400 inches at 12 hours, 30 minutes (59 minutes later). RCS Loop "A" hot leg temperature rises to 600°F during next 30 minutes. Loop "A" Cold Leg increases 144°F in 2 minutes as steam condensing natural circulation appears to develop.	1. Reactimeter 2. Utility typer 3. Multipoint recorder
553		Pressurizer heater groups 1 and 2 trip, but are returned to normal within 16 min.	1. Alarm printer
554	11 hrs. 32 min.	Operators indicate method of cooldown at this time is one high pressure injection pump and the core flood tanks. (Region I Incident Messageform states pressurizer level 166" and periodically cycling EMOV. Information coming from Unit 1 Control Room with potential significant time delay.)	1. Operating logs 2. Region I Incident Messageform
555	11 hrs. 33 min.	Operator starts makeup pump 1C.	1. Alarm printer
556	11 hrs. 34 min.	North Gate 30 mr/hr.	1. Region I Incident Messageform 2. Region I IRC

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		tape 10-1
557		Operator starts emergency feedwater pump 2B and commences raising "B" OTSG level to 97 to 99% range. Will reach desired level within 20 minutes. OTSG pressure will drop from about 250 psig to about 150 psig and stabilize.	1. Alarm printer 2. Plant strip charts 3. Reactimeter
558	11 hrs. 36 min.	Operator stops makeup pump 1C.	1. Alarm printer
559	~ 11 hrs. 43 min.	Reported readings from Unit 2. RB Pressure 0 psig PZR Level 170" (Increasing)* PSR Temp. 460°F (463°F, Slowly Decreasing)* PZR Press. 450 psig (534 psig, Slowly Decreasing)*	1. Region I IRC tape 10-2 2. (Utility typer)*
560	11 hrs. 44 min.	Pressurizer low level alarm clears at 206" and increasing.	1. Alarm printer
561	11 hrs. 52 min.	Operator secures emergency feedwater pump 2B after raising "B" OTSG to 97%.	1. Alarm printer 2. Plant strip charts
562	11 hrs. 54 min.	Pressurizer high level alarm received at 260" with level increasing.	1. Alarm printer
563	~ 11 hrs. 59 min.	NRC:HQ asks Unit 1 Shift Supervisor (over telephone) to communicate NRC concern that pressurizer level indication does not preclude bubble in core, and that temperature readings indicating superheat may be real and imply core is uncovered. Would then need to find a way to put more water into core and get core level back up. Question is asked if licen-see has talked to B&W, because B&W had been trying to get in touch with	1. Region I IRC tape 11-1 2. IRACT tapes

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	the licensee and had the same concern.	
564	~ 12 hrs.	HPI alignments still heavily biased to "C" injection leg. Operators believed natural circulation did exist at this time. Rate of BWST level loss had been reduced, but still concerned would eventually run out of clean water.	1. Statement GPM 2. IRACT tapes 3. Region I IRC tapes
565	~ 12 hrs.	RCDT temperature trending down. Pressurizer level trending up as RCS pressure increases. "A" loop hot leg temperatures are stable and the pressurizer is heating slowly. "B" OTSG shell and downcomer temperatures are slowly decaying with level now at 95%. "A" OTSG pressure is slowly decaying with level stable at 92%. Appears operator is compressing pressurizer bubble and loop vapor lock with HPI flow.	1. Utility typer 2. Plant strip charts
565A	~ 12 hrs.	B&W requests licensee establish site to B&W communications link. Recommends 500 gpm (atleast 400 gpm) HPI flow be established immediately. Whether this information gets passed to the site at this time has not been determined.	1. B&W Telephone logs
566	~ 12 hrs.	B&W Site Operations Manager relays information to B&W on plant status. B&W requested to identify consideration for running RCP in the "B" loop (DH drop line side). Various recommendations were relayed back to Site Management at this time.	1. Interviews 53, 67 2. B&W Telephone logs
567	12 hrs. 2 min.	The hourly plant log typer displayed the following information as being applicable for this time: RCS Flow: 3 MLBH Loop "A" Th: 590 °F Tc _a : ? °F	1. Log typer

ITEM	DATE AND TIME	EVENT DESCRIPTION	REFERENCE
	3/28/79	<p style="margin-left: 40px;">Tc_a: ? °F</p> <p style="margin-left: 40px;">Loop "B"</p> <p style="margin-left: 80px;">Th: ? °F</p> <p style="margin-left: 80px;">Tc_b: ? °F</p> <p style="margin-left: 80px;">Tc_b: : °F</p> <p style="margin-left: 40px;">MU Flow: 116 GPM</p> <p style="margin-left: 40px;">OTSG "A" Level: 365 inches</p> <p style="margin-left: 40px;">OTSG "B" Level: 336 inches</p> <p style="margin-left: 40px;">Condensate Flow: 0 KLBH</p>	
568	12 hrs. 10 min.	<p>Reported incore temperatures unavailable. Supervisor reports to NRC they are all printing question marks which means either the computer point or the sensor is malfunction. The supervisor did not indicate that the same result occurs when the temperature exceeds the range of the software calibration for those points. This is the first thermocouple data or comments to the NRC. First request had been made at approximately 8 hrs., 20 min. States core flood tanks floating on core, on T_a 590 °F, other "pegged" and that staff is convinced there is no boiling in the core.</p>	<ol style="list-style-type: none"> 1. Region I Incident Messageform 2. Region I IRC tape 10-1 IRACT tapes
569	12 hrs. 10+ min.	<p>Region I inspector reports to Region that:</p> <ol style="list-style-type: none"> 1. Licensee believes no bubble in core since core flood tanks sit on top of core and only a little water "slid in." (Recall piping design utilizes "loop seal.") 2. Licensee NOT planning to use electromatic to blowdown because could fail open with resulting rapid cooldown that could overstress RCS. 	<ol style="list-style-type: none"> 1. Region I IRC tape 11-1 1. Region I IRC tape 11-1

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
570		Note: (Information from Unit 1 control room with attendant time delay)	
571	12 hrs. 11+ min.	Operator requests computer print 15 separate Incore T/C readings. All but two (572°F and 597°F) read off scale.	1. Utility typer
572	~ 12 hrs. 15 min.	NRC:HQ asks if licensee has considered blowing the system down, and supervisor responds that it had been discussed and rejected. NRC:HQ requests it be considered again, stressing this was a request to consider it, not an order to do so. Approximately 5 minutes later the supervisor returned stating the licensee thought it was a good idea and decided to do it. NRC:HQ then relayed their concern about valving out the CFT's before the blowdown to prevent nitrogen from getting into the vessel.	1. IRACT tapes Region I IRC tapes
573	12 hrs. 22 min.	Pressurizer level reaches 400 inches.	1. Reactimeter
573A	~ 12 hrs. 25 min.	Licensee will subsequently inform B&W that HPI of 400 GPM had been established and maintained since this time.	1. B&W Telephone logs
574	12 hrs. 30 min.	License representative reports to Region I: "We're going to decrease pressure. What we're going to do is open that electromatic."	1. Region I IRC tape 11-2
575	~ 12 hrs. 30 min.	NRC inspector reported Unit 1 control room is back in masks.	1. Region I IRC tape 12-1
576	12 hrs. 30 min.	Operator opens EMOV to depressurize. Rate of increase of reactor building pressure indicates both the EMOV and its block valve are open.	1. Region I Incident Messageform 2. Region I IRC tape 11-2

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
577	~ 12 1/2 hrs.	Station Manager and Superintendent-Technical Support return from Lt. Governor's Office.	1. Interviews 27, 53
578		Note: Review of IRACT tapes and Region I IRC tapes shows that NRC was not aware that the Station Manager and the Superintendent-Technical Support had been away from the site for the last 2 1/2 hours.	
579	~ 12 hrs. 30+ min.	Operator dispatched to manually reset the DHR inlet valve (DH-V108A&B) pressure interlock when RCS pressure drops below 385 psig.	1. Interview 81
580	12 hrs. 30+ min.	NRC inspector on line in Unit 2 Control Room. Reports following: <ul style="list-style-type: none"> 1. Originally had bubble in both loops. 2. Have collapsed bubble in "A" loop. 3. Had bubble in pressurizer, but just lost it. 4. Floating flood tanks on core and got some injection. 5. Normal letdown at 120 gpm. 6. Makeup Pumps feeding from BWST. 7. B&W calculations indicate 120 gpm insufficient to remove current core decay heat. 8. Operating electromatic relief and pressurizer vent valves 	1. Region I IRC Tape 12-1 2. NRC Inspector Notes

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	periodically to remove additional heat.	
		9. Plan is to establish vacuum in condenser, steam "A" OTSG and remove core heat through natural circulation.	
581		NOTE: Inspector reported to Region I that licensee believes "A" loop is solid since: Original T_h (A&B loops) $\sim 700^\circ$ with $T_c \sim 225^\circ$; Directed MU to "A" loop to collapse steam bubble and its T_h dropped quite rapidly to $\sim 575^\circ$ F. At the same time PZR level dropped considerably and came back on scale.	
582	12 hrs. 35 min.	Pressurizer level tops out. RCDT temperature begins to rise and reactor building pressure begins to rise. Tailpipe temperature alarms received on EMOV and both pressurizer safety valves; believed associated with operator opening EMOV Block Valve and possibly EMOV, also.	<ol style="list-style-type: none"> 1. Alarm printer 2. Utility typer 3. Plant strip charts 4. Reactimeter
583	12 hrs. 40 min.	Rate of reactor building pressure change indicates EMOV block valve was just shut.	<ol style="list-style-type: none"> 1. Plant strip charts
583A	\sim 12 hrs. 40 min.	B&W informed by site of plant status, recognized now to represent super-heat conditions by B&W.	<ol style="list-style-type: none"> 1. B&W Telephone logs
584	\sim 12 hrs. 45 min.	"A" loop T_h reported $\sim 580^\circ$ F. Utility Typer indicates $\sim 572^\circ$ F.	<ol style="list-style-type: none"> 1. Region I IRC Tape 12-1 2. Utility Typer
584A	\sim 12 hrs. 45 min.	B&W informed of plant status via B&W employee offsite. B&W attempts to relay recommendation that HPI must be immediately increased, letdown stopped and RCS subcooled condition established.	<ol style="list-style-type: none"> 1. B&W Telephone Logs

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	Rate of reactor building pressure increase suggests pressurizer vent valves may have just been opened.	1. Plant strip charts
585	12 hrs. 45 min.	Region I inspector reports conditions are: pressure 450-500, (Utility Typer and Plant Strip Chart indicates RCS pressure is 620 psig.) letdown (120 gpm), and EMOV opening intermittently. Lowest pressure reached during blowdown was 450 psig and it just "hung" there. "Nothing tried so far could drive it lower."	1. Region I Incident Messageform 2. Region I IRC tape 12-1
586	12 hrs. 50 min.	Reported natural circulation cooldown being conducted using "A" OTSG. Working on getting main condenser vacuum. Until can get temperature and pressure reduced further, can't go on decay heat removal.	1. Region I Incident Messageform 2. Region I IRC Tape 12-1
587	12 hrs. 56+ min.	<p>NRC inspector reports the following:</p> <ol style="list-style-type: none"> 1. Licensee concerned that current cooldown process is too slow and believe will be faster steaming OTSG. 2. Licensee working to get rid of bubbles in loops, establish bubble in pressurizer and go on natural circulation. 3. Licensee concerned for further use of electromatic since water dumps to floor, and with sources of clean water being exhausted, would be forced to use dirty sump water for recirculation. 4. Licensee concluded core was covered. Discussed and rejected further blowdown since would ultimately entail recirculation of sump water, 	1. Region I IRC tape 12-2

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	<p>greatly increasing magnitude of cleanup. Discussed and rejected repressurization to collapse all bubbles. Latter decision appears to be contrary to recommendations of B&W, of which the licensee is aware.</p> <p>5. Have regained some pressurizer heaters.</p> <p>6. Plan to draw condenser vacuum using TMI-2 mechanical vacuum pumps and TMI-1 steam for turbine gland seal. Having problems because of high radiation and contamination levels, restricting free movement thru plant.</p>	
588		Note: IRC notifies OC of this information. Asks if licensee can explain hot leg temperature as superheat without core being uncovered, and IRC relates to OC the licensee discussion and conclusion they were confident the core was covered.	1. IRACT tapes
589	12 hrs. 59 min.	One code safety outlet temperature high alarm clears.	1. Alarm printer
590	13 hrs. 1 min.	<p>The Hourly Plant Log Typer displayed the following information as being applicable for this time:</p> <p>RCS Flow: 0 MLBH</p> <p>Loop "A"</p> <p>$T_H = 522 \text{ }^\circ\text{F}$</p> <p>$T_{Ca} = ? \text{ }^\circ\text{F}$</p> <p>$T_{Ca} = ? \text{ }^\circ\text{F}$</p> <p>Loop "B"</p> <p>$T_H = ? \text{ }^\circ\text{F}$</p>	1. Log typer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	$T_{bc} = ? \text{ } ^\circ\text{F}$ $T_{cb} = ? \text{ } ^\circ\text{F}$ MU Flow: 128 GPM OTSG "A" Level: 345 inches OTSG "B" Level: 321 inches Condensate Flow: 0 KLBH	
591	13 hrs. 2 min.	Operators attempt unsuccessful start of condenser vacuum pump 1A. Operator then starts condenser vacuum pump 1C.	1. Alarm printer 2. Plant strip charts
592	13 hrs. 4 min.	Reactor building air cooling fan AH-E-11B trips. RCDT temperature and reactor building pressure start long term trend down, which will continue through the rest of this scenario; suggesting neither the EMOV or pressurizer vent valves are used again.	1. Alarm printer 2. Utility typer 3. Plant strip charts
593	13 hrs. 5 min.	Operators starting to draw vacuum. Staff remains confident no bubble in "A" OTSG loop. RCS pressure 650 psig and pressurizer level 400" decreasing. (Pressurizer heaters appear to have raised pressurizer temperature to the point it is blowing water to the "A" loop Hot Leg.) Continuing normal letdown and makeup. "A" loop $T_H \sim 530^\circ\text{F}$ and dropping; "B" loop $T_H \sim 700^\circ\text{F}$.	1. Region I Incident Messageform 2. Region I IRC tape 12-2 3. Interview 44 4. Utility typer 5. Plant strip charts
594		Second code safety outlet temperature high alarm clears.	1. Alarm printer
595	13 hrs. 8 min.	Main feedwater pump turbine low vacuum trips clear as vacuum increases.	1. Alarm printer
596	13 hrs. 13 min.	Operators succeed in starting condenser vacuum pump 1A.	1. Alarm printer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
597	13 hrs. 20 min. 3/28/79	Reactor building pressure zero and decreasing to vacuum. Operators had noted reactor cooling system going solid again. Previous small change in pressurizer level was noted. Operators now note pressurizer level starting to drop and RCS pressure dropping. (Present value 636 psig). Seal injection flow 60 gpm. HPI flow 425 gpm and BWST level 26½ ft. Inspector at site reports continued operation of electromatic relief (RB pressure trace does not support) steam "A" OTSG to condenser. "A" OTSG pressure begins to drop. RCS Loop "A" cold leg temperature begins to drop as heat is removed from OTSG by steaming, and from the RCS by what appears to be steam condensing natural circulation.	<ol style="list-style-type: none"> 1. Operating logs 2. Region I Incident Messageform 3. Statement Plant Manager 4. Utility typer 5. Multipoint recorder 6. Plant strip charts
598	~ 13 hrs. 23 min.	<p>Status reported by Region I inspector:</p> <ol style="list-style-type: none"> 1. 15" Vacuum in Condenser. 2. Just starting to steam now. 3. RCS ~650 psig. (635 psig)* <ul style="list-style-type: none"> T_{hb} 650-700°F T_{hb} 570-580°F (529°F)* T_{ha} & T_{cb} & T_a 225°F (T_{ca} 494°F)* 4. Electromatic relief indicates closed. <p>Station Manager directed to take plant solid at high pressure.</p>	<ol style="list-style-type: none"> 1. Region I IRC Tape 13-1 2. (Utility typer)* 3. Plant strip charts <ol style="list-style-type: none"> 1. Statement Plant Manager

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
599	13 hrs. 23 min.	Makeup pump 1C started by operator.	1. Alarm printer
600	13 hrs. 25 min.	EMOV outlet temperature high alarm clears.	1. Alarm printer
601	13 hrs. 26 min.	Pressurizer heater groups 1 and 2 trip. At this time there are now no more than 6 operable pressurizer heater groups. Pressurizer temperature begins to drop. Pressurizer level stops dropping and starts to rise.	1. Alarm printer 2. Utility typer
602	13 hrs. 30 min.	Staff repressurizing RCS to collapse voids and prepare to start RC pump. Plant charts confirm pressure increase.	1. Utility typer 2. Plant strip charts 3. Statement Plant Manager
603		NOTE: Regarding decision to repressurize RCS; interviews indicate that control room personnel had desired to remain in depressurized state since they had regained pressurizer level, appeared to have collapsed bubble in "A" loop, and were progressing on doing same to "B" loop.	1. Interviews 31, 44, 53
604		After decision to pressurize, control room personnel stated they recommended restart of RCP to sweep vapor from loop.	
605	13 hrs. 32 min.	RCS pressure 623 psig and increasing. RCS pressure is increased to about 2300 psig over the next 74 minutes, using two makeup pumps, with the EMOV block valve shut.	1. Operating logs 2. Utility typer
606	13 hrs. 40 min.	"A" OTSG outlet valve not opening - not steaming. Operators have started investigating, assuming indicator must be in error. ("A"	1. Region I Incident Messageform

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	OTSG pressure had dropped 30 psi from 144 psig, but now is slowly rising.)	2. Region I IRC tape 13-2 3. Utility typer
607	13 hrs. 45 min.	Started steaming "A" OTSG.	1. Operations logs
608	~ 13 hrs. 45 min.	Reactor building pressure is -0.2 psig.	1. Region I IRC Tape 13-2
609	13 hrs. 50 min.	RCS pressure 765 psig. Pressurizer level 285 inches. As pressure increases above this point, pressurizer level climbs, reaching 400 inches in about 32 minutes.	1. Operating logs 2. Utility typer 3. Plant strip charts
610		"A" OTSG started steaming using turbine bypass valves to condenser. Indication of level change in steam generator and indication of feedwater flow. ("A" OTSG pressure showed marked downward trend at 13 hrs. 42 min.)	1. Region I Incident Messageform 2. Region I IRC Tape 14-1 3. Utility typer
611	13 hrs. 50+ min.	Not now wearing respirators in TMI-2 control room.	1. Region I IRC tape 14-1
612	13 hrs. 52 min.	"A" OTSG high level alarm clears at 81.3%; evidence of steaming "A" OTSG.	1. Alarm printer
613	13 hrs. 59 min.	"A" OTSG high level alarm received at 83.5%; evidence of feeding "A" OTSG.	1. Alarm printer
614	14 hrs.	RCS Pressure 860 psig. BWST level 24 feet.	1. Operating logs

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
615		Region I inspector reports BWST 23'.	2. Utility typer 1. Region I Incident Messageform
616	14+ hrs.	Region I Inspector reports his understanding that: 1. Either licensee staff or senior management have concern whether core covered or not. (NRC notes show VP-Generation has directed repressurization of RCS.) 2. The plan has changed now. Licensee plans to continue cooldown by steaming. "A" OTSG; T _h now 548 ^o F, T _c now 446 ^o F. They increased makeup to 480 gpm with letdown at 40 gpm, letting pressure increase and plan to take plant solid at ~2000 psig to collapse all bubbles. No longer concerned with remaining volume of clean water since going solid. (This information was passed from IRC to OC.) Note: This information appears to lag actual decision about 1/2 hour.	1. Region I IRC tape 14-1 2. Statement Plant Manager 3. NRC inspector notes
617	14 hrs 2 min.	The hourly plant log typer displayed the following information as being applicable for this time: RCS Flow: 0 MLBH Loop "A" T _h : 549 ^o F T _{ca} : ? ^o F T _{ca} : ? ^o F	1. Log typer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	Loop "B" T _h : ? ⁰ F T _{cb} : ? ⁰ F T _{cb} : ? ⁰ F MU Flow: 43 GPM OTSG "A" Level: 338 inches OTSG "B" Level: 321 inches Condensate Flow: 0 KLBH	
618	14 hrs. 7 min.	RCS Pressure 945 psig. BWST 24 ft.	1. Operating logs 2. Utility typer
619	14 hrs. 10 min.	RCS Pressure 1012 psig. Pressurizer level 370 inches.	1. Operating logs 2. Utility typer
620	~ 14 hrs. 10 min.	Note: Inspector onsite reports to OC that "B" OTSG hot leg still appears to have bubble with temperatures in the range of 650 ⁰ F to 700 ⁰ F. States that B&W people are taking thermocouple readings on the back panel and they have obtained 700 ⁰ F to 750 ⁰ F readings.	1. IRACT tapes
621	~14 hrs. 17+ min.	Region I inspector reports: 1. Continuing cooldown of "A" OTSG. The ΔT for "A" loop is increasing. T _{ha} 555 ⁰ (554 ⁰ F)** T _{hb} 650-700(*) ⁰ F T _{ca} 300 ⁰ F (271 ⁰ F)** T _{cb} 200 ⁰ F	1. Region I IRC tape 14-2 2. NRC Region I Incident Messageform 3. (Utility typer)**
		(*) = Old data from last time B&W employee manually reads back panel RTD parameters.	

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	<p>2. Makeup at 450 gpm. Pressure 1100 psig increasing, pressurizer full but pressure increase indicates bubble still exists somewhere in system.</p> <p>3. Using condensate pumps, not emergency feed, for OTSG level control.</p> <p>Note: Appears time lag in information flow to NRC is much shorter for data than it is for changes in management plans.</p>	
622	14 hrs. 20 min.	RCS Pressure 1200 psig. Pressurizer level 400 inches and BWST level 23 ft.	<p>1. Operating logs</p> <p>2. Utility typer</p>
623	14 hrs. 26 min.	Pressurizer heater groups 1 and 2 return to normal. There are now no more than 8 operable pressurizer heaters. Pressurizer temperature begins increasing.	<p>1. Alarm printer</p> <p>2. Utility typer</p>
624	~14 hrs. 29 min.	<p>Region I inspector reports:</p> <p>1. OTSG level ~90%.</p> <p>2. 27" Vacuum in condenser.</p> <p>3. Main steam bypass valve indicates 30% open.</p>	<p>1. R:I IRC tape 15-1</p> <p>2. Plant strip chart</p>
625	14 hrs. 30 min.	RCS pressure 1520 psig (at approximately this time, the SRM count rate showed a rapid rise and then decayed to a point below its starting value).	<p>1. Operating logs</p> <p>2. Plant strip charts</p> <p>3. Utility typer</p>
626	14 hrs. 33 min.	RCS pressure 1700 psig (Line printer shows 1600 psi interlock cleared at	1. Operating logs

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	32 minutes).	2. Alarm printer 3. Utility typer
627	~14 hrs. 34 min.	Region I inspector reports: 1. Still steaming "A" OTSG, but indications are that they are not getting much natural circulation. ("A" OTSG pressure essentially zero now).* 2. T_{ha} ~570 ^o F (566 ^o F), T_{ca} ~220 ^o F (273 ^o F), one Incore T/C reads 570 ^o , indicating T_{ha} may be accurate; other T/Cs can not be read. (This apparently refers to the "??" output for the T/C's indicating they are outside the range of the computer software.)	1. R:I IRC tape 15-1 *2. Utility typer
628		B&W Site Operations Manager contacts B&W at Lynchburg who indicate that had also come to conclusion that licensee should repressurize and start a RCP.	1. Interview 53
629	14 hrs. 35 min.	RCS Pressure 1800 psig (increasing), T_{ha} =570 (567 ^o F) and T_{ca} =220 (273 ^o F).* Indication some bubbles may be collapsing since have seen some temperature changes (T_{ca} showing slow upward trend).*	1. Operating logs 2. (Utility typer)* 3. R:I Incident Messageform 4. R:I IRC Tape 15-1
629A	~ 14 hrs. 35 min.	Unit 2 Control Room to B&W open line established. (Will be maintained through rest of sequence.)	1. B&W Telephone Logs
630	14 hrs. 39 min.	Operator closes MU-V16B, reducing HPI flow to 120 gpm at 2080 psig.	1. Operations log 2. Utility typer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
631	14 hrs. 41 min.	Operator throttles MU-V16C, reducing HPI flow to 105 gpm at 2220 psig.1.	Operations log 2. Plant strip charts
632	14 hrs. 43 min.	RCS pressure 2275 psig. Makeup pump 1C tripped and valve 16C closed by operator. Makeup pump 1B still running.	1. Operating logs 2. Alarm printer 3. Plant strip chart
633	~ 14 hrs. 45 min.	Note: OC asks inspector onsite if licensee has considered running an RCP. Inspector states that those preparations are underway but they are having trouble with the oil lift pumps. OC asks if they considered that they might have a gas bubble rather than a steam bubble in the "B" loop. Concerned it could be nitrogen, xenon, or hydrogen. Inspector agrees to pass concern along. This concern apparently developed from evidence that only the A leg bubble quenched while the B loop remained superheated. Shortly thereafter, the inspector reported back that there was nothing licensee could do about it. There was no way to vent that leg, and the only way to sweep it out, whether it be steam or gas, would be to use the pumps.	1. IRACT tapes
634	14 hrs. 45 min.	Operators beginning preparation to start a reactor coolant pump.	1. Operating logs
635		Control room personnel recommended only bumping RCP, as would normally be done following routine loop refill.	1. Interview 53
636		Note: Earlier loss of vital switchgear has disabled RCP lift and back-stop oil pumps. Attempts are underway to regain these components to allow RCP starts, should they become desirable.	1. Statement Plant Manager

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
637	3/28/79	Note: Oil pump starting interlocks (K3 Relays) would still be jumpered at this point.	
638	14 hrs. 45+ min.	<p>Region I inspector reports:</p> <ol style="list-style-type: none"> 1. Press ~2300 psig. (2291 psig)* 2. Appears still have bubble in "B" loop. 3. Makeup and Letdown at 150 gpm. 4. T_{ha} 560 (560°F)*, T_{ca} 320° (287°F)*. 5. Steaming ~100,000 #/hr based on constant OTSG level and rate of feeding. 6. Level in "A" OTSG at ~93%. 	<ol style="list-style-type: none"> 1. Region I IRC tape 15-2 2. See later confirmation at 14:52 3. NRC Region I Incident Messageform 4. (Utility typer)* 5. Plant strip charts
639	14 hrs. 47 min.	Holding RCS Pressure at 2300 psig. Increasing steaming rate from "A" steam generator to condenser by opening bypass valves to provide more cooldown; one makeup pump in operation; 80 gpm letdown; 32 gpm seal injection; 20 gpm makeup; BWST 22 ft.	<ol style="list-style-type: none"> 1. Operating logs
640		Note: "A" OTSG Pressure indicates zero; will begin slow trend upward in 3 minutes.	<ol style="list-style-type: none"> 1. Utility typer
641	14 hrs. 48 min.	Alarm printer output for the period 14 hrs. 48 min. 23 sec. to 15 hrs.	<ol style="list-style-type: none"> 1. Alarm printer

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	10 min. 29 sec was not available for NRC review. Printer output available after this period demonstrates gross paper feed problems and uses black ink only compared to the normal red for alarms and black for "return to normal" seen in previous records. Utility typer output, which may have information gaps, does not indicate the alarm printer became unavailable during this time.	2. Utility typer
642	14 hrs. 52 min.	Region I inspector reports RCS 2300 psig, bubble still in "B" loop, "A" loop $T_h=560$, $T_c=320$, "A" steam generator 50 psi.	1. NRC Region I Incident Messageform 2. Utility typer
643	15 hrs.	Region I inspector reports: 1. T_h dropping slowly, T_c rising slowly. Believe getting some natural circulation flow, but still very little. 2. Licensee has considered starting RCP, but unable to do so since has lost power to RCP oil pumps. Having problems restoring power to oil pump bus because of high radiation and contamination levels.	1. R:I IRC Tape 15-2 2. Utility typer
644		Inspector reports that B&W in Lynchburg has been contacted and is doing calculations for TMI. (Transcripts are not clear as to nature of calculations being performed.)	1. Region I IRC tape 16-1 2. IRACT tapes
645		Inspector reports $T_{ha}=560^{\circ}\text{F}$, $T_{ca}=360^{\circ}\text{F}$, $T_{cb}=220^{\circ}\text{F}$, T_{hb} off scale high, 2300 psig and proceeding in attempt to start a RCP.	1. NRC Region I Incident Messageform 2. Region I IRC tape

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		16-1
			3. Utility typer
646	15 hrs. 2 min.	<p>The hourly plant log typer displayed the following information as being applicable for this time:</p> <p>RCS Flow: 0 MLBH</p> <p>Loop "A"</p> <p>T_h: 556^oF</p> <p>T_{ca}: ?^oF</p> <p>T_{cb}: ?^oF</p> <p>Loop "B"</p> <p>T_h: ?^oF</p> <p>T_{cb}: ?^oF</p> <p>MU Flow: 72 GPM</p> <p>OTSG "A" Level: 547 inches</p> <p>OTSG "B" Level: 320 inches</p> <p>Condensate Flow: 0 KLBH</p>	1. Log typer
647	15 hrs. 4 min.	NRC:HQ was of opinion might have to bump RCP to clear "candy-canes" of voids to allow full natural circulation, since trend of parameters strongly suggest little or no natural circulation is occurring.	1. Region I IRC tape 16-1
648	15 hrs. 10 min.	<p>Inspector reports T_{ha} 557^oF, T_{ca} 380^oF, T_{cb} 240^oF, 2300 psig.</p> <p>Steam Generator "B" 180 psig</p> <p>Steam Generator "A" 20 psig (10 psig)*.</p>	<p>1. NRC Region I Incident Messageform</p> <p>2. Region I IRC tape 16-1</p>

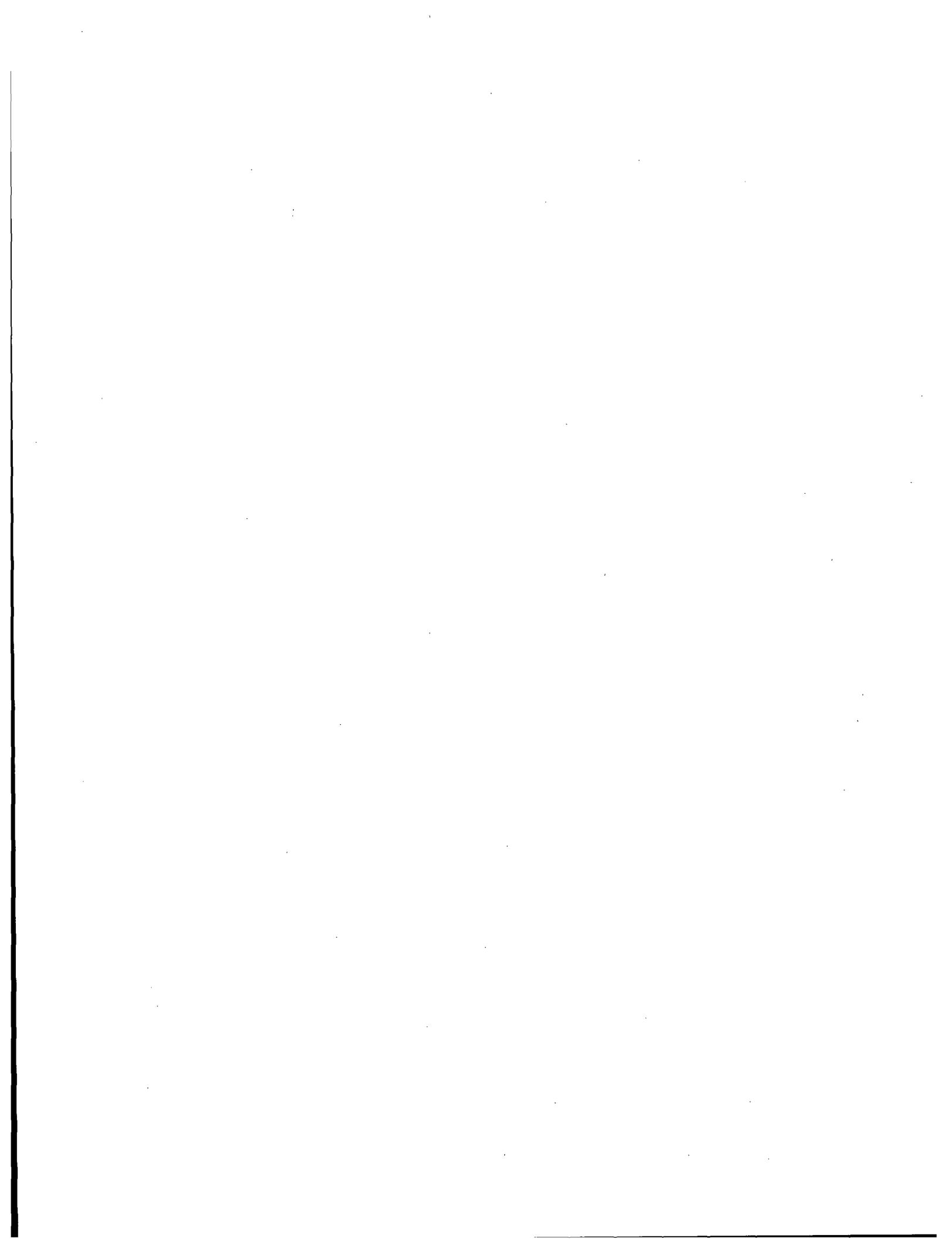
<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
			3. (Utility typer)*
649	15 hrs. 11 min.	Operator requests computer summarize reactor coolant pump and makeup pump status.	1. Utility typer
650	~ 15 hrs. 15 min.	1. Heaters now on in an attempt to draw a bubble, but currently pressurizer level is off scale high. 2. Since no power to A.C. RCP oil pumps, trying to start D.C. RCP oil pumps. Interview indicates licensee had to send operators to auxiliary building to assist in starting RCP oil pumps.	1. Region I IRC tape 16-1 1. Interviews
651	15 hrs. 16 min.	Re-established normal vacuum in condensor.	1. Operations log
652	15 hrs. 20 min.	Region I inspector reports BWST 22 ft. Still steaming; "A" loop ΔT decreased, RCS Pressure 2300. Increasing temperature with heaters. Trying to steam down "B" OTSG level and have secured feeding it. T_h 560°F; T_{ca} 395°F; T_{cb} 220°F; Makeup and Letdown ~60 gpm Reactor building pressure -0.2 psig Pressurizer temperature (505°) increasing with heaters on	1. NRC Region I Incident Messageform 2. Region I IRC tape 16-2 3. Utility typer 4. Operations log
653		Note: Reported two NRC inspectors scheduled to go to Governor's Office for briefing at ~7:30 p.m. (Would be ~ 15 hrs. 30 min. after start of accident.)	1. Region I IRC tape 16-2
654		Note: Reported that at 1600 hours (12 hrs. after start of accident) the	1. Region I IRC tape

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	Gross β was less than minimum detectable at condensate pump discharge (MDA= 10×10^5 uc/ml).	16-2
655		Note: Reported that have been unable to get a primary sample since ~0900 hours (~5 hrs. after start of accident) this morning.	1. Region I IRC tape 16-2
656	15 hrs. 22 min.	Started condensate booster pump 2B to fill "B" OTSG.	1. Operating log 2. Alarm printer
657		Note: Reported to NRC early morning sample of OTSGs were checked with frisker. "B" sample pegged meter, "A" OTSG sample did nothing to meter. Based on results, had isolated "B" OTSG.	1. Region I IRC tape 16-2 2. Section 3.2.3 of Details II
658	15 hrs. 32 min.	Secures condensate booster pump 2B after raising "B" OTSG level.	1. Alarm printer 2. Operations log
659	15 hrs. 33 min.	Operator starts RCP 1A, runs it for 10 seconds and then trips it. RCS pressure begins to drop and operator attempts unsuccessful start of makeup pump 1C. Pressure drop actuates ES and starts decay heat pumps 1A & 1B. Operator bypasses ES and starts makeup pump 1C within 6 seconds. Pressure turns at a minimum of ~1409 psig and climbs to ~2200 psig over next 17 minutes.	1. Operating log 2. Plant strip charts 3. NRC Region I Incident Messageform 4. Region I IRC tape 16-2 5. Statement Plant Manager 6. IRACT tapes 7. Interview 119
660	15 hrs. 35 min.	"A" loop - T_h now below 520°F, the lowest meter can read. T_c did not go	1. Region I IRC tape

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	up much. Staff believes they got some flow. Must wait 15 minutes due to RCP motor start limitations until can bump RCP again.	16-2 2. Utility typer
661		Station Manager consulted with B&W Site Operations Manager on appropriate delay before pump restart.	1. Interview 53
662	15 hrs. 37 min.	Operator places RCS loop "B" cold leg temperature in Analog Recorder Trend (#1).	1. Alarm printer
663	15 hrs. 38 min.	RCS pressure 1783 psig and climbing.	1. Alarm printer 2. Utility typer
664	15 hrs. 38+ min.	Inspector reports that he believes current plans are to run that RCP and conduct normal cooldown. When bumped RCP, OTSG pressure jumped from 20 psig to 200 psig, indicating they did get heat transfer. (Still steaming OTSG to condenser.)	1. Region I IRC tape 17-1 2. Utility typer
665	15 hrs. 39 min.	Makeup pump 1C turned off. RCS pressure (Loop A) 1948 psig (1846 psig)*.	1. Operating logs 2. (Utility typer)* 3. Alarm printer 4. Interview 125
666	15 hrs. 40 min.	Operator places RCS wide range pressure in Analog Recorder Trend (#2). Operator places pressurizer temperature in Analog Recorder Trend (#3).	1. Alarm printer
667	15 hrs. 45+ min.	Setting up to start and run an RCP in "A" loop.	1. Region I IRC tape 17-1 2. Statement Plant

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		Manager
668		Station Manager waited appropriate delay time on RCP restart and then directed operators to start RCP 1A.	
669	15 hrs. 49 min.	Makeup pump IC started by operator.	1. Operating logs 2. Alarm printer
670	15 hrs. 50 min.	<p>Started RCP 1A and let it run. Flow and amps look good. Bypassed ES prior to its actuation on train "A", but train "B" actuates and then clears due to contact race. Reactor coolant pressure drops to 1123 psig. "B" OTSG indicated level drops below 79% and recovers to above 85 % within 8 second span. "A" OTSG indicated level drops below 81% and recovers to above 82% within a 9 second span, 40 seconds after "B" OTSGs transient.</p> <p>1. Loop Temperatures: "A" Loop Cold Leg: 340⁰F "B" Loop Cold Leg: 338⁰F</p> <p>2. Steam Generator Pressures: "A" Generator increases from 0 to approximately 40 psig (94 psig)*. "B" Generator decreases from 140 psig to approximately 50 psig. (T_{sat} approximately 330⁰F).</p>	1. Operating logs 2. Region I IRC tape 17-1 3. Alarm printer 4. Plant strip charts 5. Statement Plant Manager 6. (Utility typer)* 7. Interview 119 8. IRACT tapes
671	15 hrs. 52+ min.	Pressurizer level still offscale high.	1. Region I IRC tape 17-1

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
672		Operator places pressurizer temperatures, RCS temperatures and computed saturation temperature in computer display group.	1. Alarm typer 2. Utility typer
673	15 hrs. 55 min.	Operator secures decay heat pumps 1A & 1B.	1. Alarm printer
674	15 hrs 56 min.	Operator secures makeup pump 1C. Makeup pump 1B is still operating. It is assumed makeup pump 1A is still in pull-to-lock.	1. Alarm printer
675	16 hrs.	Note: At this point, the time frame for the Operational Sequence of Events is concluded due to the continued operation of RCP-1A.	
		PLANT STATUS	
676		RCP 1A is running with both RCS loop cold legs temperatures at about 290°F. Pressurizer level is still full scale with RCS pressure at about 1350 psig. "B" OTSG is isolated at about 97% level and 99 psig, while "A" OTSG is steaming to the main condenser at about 93% level and 76 psig. Makeup pump 1B is operating, supplying RCP seals and normal makeup, the latter at 95 gpm. Makeup pumps 1A & 1C are secured, as are the decay heat removal pumps 1A & 1B. Pressurizer temperature is about 150°F and operators are letting down in attempt to draw a bubble. The core flood tank isolation valves are open with breakers racked out. Operators are utilizing normal cooldown procedures.	1. Plant strip charts 2. Reactimeter 3. Region I IRC tapes 17-2 & 17A-1



APPENDIX I-B

OPERATIONAL ASPECTS POTENTIAL ITEMS OF NONCOMPLIANCE

The following items are under consideration as potential items of noncompliance.

A. Technical Specification 3.4.6.2.b requires Reactor Coolant System (RCS) leakage shall be limited to one gallon per minute (GPM) "Unidentified Leakage" during "Power Operations"; unless leakage rates above this limit are reduced to within limits in four hours or the plant be placed in "Hot Standby" in the next six hours.

Contrary to the above, RCS "Unidentified Leakage" remained above one GPM for the period March 22-28, 1979, during continuous "Power Operations", as a result of an incorrect calculational procedure (Section 1.2.3, Details I).

B. Technical Specification 6.8.1.a requires written procedures shall be implemented, covering the activities referenced as applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972. Appendix "A" of Regulatory Guide 1.33 recommends procedures for combating emergencies; such as, "Malfunction of Pressure Control System."

Unit #2 Emergency Procedure 2202-1.5, "Pressurizer System Failure," Revision 3, requires the closure of the Electromatic Relief Isolation Valve (RC-V2) for a leaking or failed open Electromatic Relief Valve (RC-R2) and the placing of Code Relief Valve (RC-R1A or RC-R1B) Discharge Line temperatures on the Analog Trend Recorder for a leaking Code Relief Valve.

1. Contrary to the above, RC-V2 was not closed and the Code Relief Valve Discharge temperatures were not placed on the Analog Trend Recorders, with leakage from RC-R2 and/or either or both RC-R1A and RC-R1B, on the morning of March 28, 1979, prior to 0400 hours. (Section 1.2.4, Details I).

2. Contrary to the above, RC-V2 was not closed on March 28, 1979, during the accident from 0400 hours until 0619 hours; allowing a significant loss of RCS inventory (Section 2.13, Details I).

C. Technical Specification 3.7.1.2 requires three independent steam generator emergency feedwater pumps and associated flow paths shall be operable during "Power Operations"; with one emergency feedwater system allowed to be inoperable if the system is restored to operability in 72 hours or the plant placed in "Hot Shutdown" within the next 12 hours.

Contrary to the above, the associated flow paths from all three emergency feedwater pumps were made inoperable on January 3, February 26 and March 26, 1979, during the routine performance of, and in accordance with an improper surveillance tests procedure (Section 1.3.2, Details I).

D. Technical Specification 6.5.1.7.b requires in part that the Plant Operations Review Committee (P.O.R.C) shall render determination in writing with regard to whether or not changes to those procedures required by Technical Specification 6.8 constitute an unreviewed safety question. Technical Specification 6.8 requires in part that procedures be established covering surveillance and test activities of safety related equipment.

Contrary to the above, the P.O.R.C. did not specifically address in their safety evaluations of Revision 4 of Unit 2 Surveillance Procedure

2303-M27A/B, dated August 30, 1978, and Revision 8 of Unit 2 Surveillance Procedure 2303-M14A/ B/C, dated November 21, 1978, the valve lineup modifications which result in complete isolation of all emergency feedwater flow paths to the steam generators during conduct of the procedures. Further, the P.O.R.C. did not render a written determination elsewhere that the valve lineup modifications did not constitute an unreviewed safety question (Section 1.3.2, Details I).

E. Technical Specification 6.10.1.d requires that records of surveillance activities, inspections, and calibrations required by these Technical Specifications shall be retained for at least five years.

Contrary to the above the complete records of the operations surveillance reviewed during this investigation were not maintained as required (Section 1.3.4, Details I).

F. Technical Specification 6.8.1.a. requires that written procedures shall be implemented covering the activities referenced as applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972, which recommends procedures for performance, administration, and schedule of surveillance activities.

Administrative procedure 1001, Revision 13, March 30, 1978, step 3.8.5 (Procedure Usage), requires the implementation of surveillance procedures including providing shift supervisor/shift foreman review of the results.

The failure to provide a review on a regular basis of the completed surveillance procedure results by the shift supervisor/shift foreman as required is an apparent item of non-compliance pursuant to Technical Specification 6.8.1 (Section 1.3.4, Details I).

G. 10 CFR 50, Appendix B, Criterion X requires a program for inspection of activities affecting quality shall be established and executed to verify conformance with documented instructions, procedures and drawings for accomplishing the activity. Three Mile Island Nuclear Station - Unit 2, Final Safety Analysis Report, Chapter 17.2.15, Section X requires the inspection program shall include random observation of operations and functional testing by individuals independent of the activity being performed. Metropolitan Edison Company Operational Quality Assurance Program, Procedure GP 4014, Revision 0, requires independent observation of activities affecting quality to verify conformance with established requirements utilizing both inspection and auditing techniques...for compliance with written procedures and the Technical Specifications.

Contrary to the above, the normal operations surveillance testing activities were not subject to, nor were they scheduled for, random and/or routine inspections by independent methods, as of March 28, 1979, (Section 1.3.5, Details I).

H. Technical Specification 6.9.1.9.b requires the licensee report in writing within 30 days the occurrence of events of the type leading to operations in a degraded mode permitted by a limiting condition for operation. Technical Specification 3.4.5 requires each steam generator shall be operable with a water level between 18 and 440 inches. Action statement 3.4.5.b requires that with one or more steam generators inoperable due to the water level being outside of limits, be in at least Hot Standby within 6 hours and in Cold Shutdown within the next 30 hours.

Contrary to the above, the licensee failed to report to NRC the occurrence of steam generator levels below 18 inches for about 4 minutes, a degraded mode allowed by Technical Specification, at about 2345 hours on November 3,

1978, following a turbine trip initiated by a loss of main feedwater (Section 1.5.2.2, Details I).

I. Technical Specification 6.8.1, in part, requires in part that written procedures shall be implemented covering activities referenced as applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972. Appendix "A" of Regulatory Guide 1.33 recommends general written procedures for maintenance, repair, replacement and modification work. Station Administrative Procedure 1021, Plant Modifications, Revision 5, paragraph 4.1.a requires that "the Supervisor of Maintenance receives a work request which represents a change/modification to a system or component".

Contrary to the above, no work request was issued for the modification of the Pressurizer Electromatic Relief Valve control logic and indicating circuit under ECM S-5934, performed on or prior to May 6, 1978, (Section 1.6, Details I).

J. 10 CFR 50, Appendix B, Criterion XVII, Quality Assurance Records, requires in part: "Sufficient records shall be maintained to furnish evidence of activities affecting quality." The Operational Quality Assurance Plan for Three Mile Island Nuclear Station, Revision 7, Section XVII, "Quality Assurance Records," defines "Quality Assurance Records" to include documentation of principal maintenance activities, results of tests, records required by Technical Specification 6.10, etc. Technical Specification 6.10.1 requires in part that records and logs be retained of unit operation; principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety; etc.

Station Administrative Procedure 1002, Rules for the Protection of Employees Working on Electrical and Mechanical Apparatus, Revision 14,

requires that on restoration of equipment to service, removed tags will have all required information entered thereon and then suitably stored, and the Shift Foreman shall approve equipment operation by signing the original tagging application.

Station Corrective Maintenance Procedure 1407-1, Revision 0, requires filing of the Job Package, following completion of work, that documents the corrective action taken (resolution description) and the satisfactory post maintenance test certification.

Station Administrative Procedure 1026, "Corrective Maintenance and Machinery History," Revision 0, requires the individual accomplishing the work adhere to the approved maintenance procedure.

Contrary to the above, documentation was either not generated or not retained which demonstrates that emergency feedwater isolation valves EF-V12A, 12B, 32A, 32B, 33A and 33B, were left in an operable state, following the performance of a maintenance check on or about January 20, 1979. Completed Station Preventative Maintenance Procedure E-2, "Dielectric Check of Insulation, Motors and Cables," which was attached to the original work request, did not contain the data that would indicate a "Meggar Check" was performed, although the work was certified as complete. The original work request, No. 756, dated January 16, 1979, did not contain certification of the completion of the required post maintenance valve cycling test by operations. The original application for tagging has not been completed indicating that tags have been removed and that the valves had been approved for equipment operation. The original tags have not been found. To satisfy Technical Specification 3.7.1.2 requirements for operability of emergency feedwater flowpaths, these valves had to be considered operable on or about January 30, 1979, prior to the plants return to "Power Operations" (Section 1.6.2.7, Details I).

K. Technical Specification 6.8.1.a requires written procedures shall be implemented covering activities referenced by applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972. Appendix A of Regulatory Guide 1.33 recommends written procedures covering "Loss of Feedwater" and "Turbine Trip". Unit 2 Emergency Procedures 2202-2.2, Loss of Feedwater, Revision 3, October 13, 1978, and 2203-2.2, Turbine Trip, Revision 7, October 25, 1978, require that the operator verify that the OTSG levels are being automatically controlled at 30 inches on the SU range indication contrary to the requirement to control level at 30 inches, the level control valves were placed in "manual" and throttled to control RCS temperature and pressure before the 30 inch level was established. The A and B OTSG levels remained below 30 inches from 0408 hours through 0420 hours and 0425 hours respectively. Moreover, the A OTSG level was below 30 inches between 0532 hours and 0543 hours. In both these examples, the affected OTSGs were taken to a dry condition as a result of this improper and/or inadequate manual control (Section 2.3, Details I).

L. Technical Specification 3.7.1.2 and its Action statement require as a minimum at least two operable independent steam generator emergency feedwater pumps and associated flow paths during "Power Operations".

Contrary to the above, steam generator emergency feedwater pump paths to both steam generators were in an isolated condition with feedwater isolation valves (EF-V12A and 12B) shut, prior to the Reactor trip at approximately 0400 on March 28, 1979, (Section 2.3, Details I).

M. Technical Specification 3.7.1.2 and its Action statement require as a minimum at least two operable independent steam generator emergency feedwater pumps and associated flow paths with average coolant temperature greater than 280°F and OTSG steam pressure greater than 800 psig.

Contrary to the above, only one steam generator emergency feedwater pump was operable as a result of operator action at 0436 hours on March 28, 1979, with average coolant temperature at about 540°F and OTSG pressures at about 965 psig (Section 2.3, Details I).

N. Technical Specification 6.8.1.a requires written procedures shall be implemented covering activities referenced by applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972. Appendix "A" of Regulatory Guide 1.33 recommends written procedures covering "Loss of Coolant". Unit 2 Emergency Procedure 2202-1.3, "Loss of Reactor Coolant/Reactor Coolant System Pressure", Revision 11, Section B, Step 2.2.4, requires reactor coolant pumps (RCPs) be tripped before reactor coolant system (RCS) pressure decreases below 1200 psig.

Contrary to the above, RCPs were not tripped before RCS pressure decreased below 1200 psig at about 0416 hours on March 28, 1979, (Section 2.5, Details I)

O. Technical Specification 3.3.2, Engineered Safety Feature Actuation System Instrumentation, Section 3.3.2.1 requires that the ESFAS instrumentation channels shall be OPERABLE as shown in Table 3.3-3, including the Safety Injection and Reactor Building Cooling and Isolation initiation from Reactor Building Pressure High with the minimum of two channels operable, in MODES 1, 2, and 3.

Contrary to the above requirement, at 0800 hours on March 28, 1979, two of three ESFAS channel in Train B were defeated, preventing the initiation of Reactor Building Cooling and Isolation and Safety Injection from Train B; and at 0820 hours on March 28, 1979, two of three ESFAS channels in Train A were defeated, preventing the initiation of Reactor Building Cooling and

Insulation and Safety Injection from Train A. The Train A and Train B Reactor Building Cooling and Isolation Actuation Trains remained defeated until 0924 hours on March 28, 1979, when the channels were automatically reset (Section 2.6, Details I).

P. Technical Specification 6.8.1.a requires written procedures shall be implemented, covering the activities referenced as applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33. This guide recommends procedures for combating emergencies; such as "Loss of Coolant". Unit 2 Emergency Procedure 2202-1.3, "Loss of Reactor Coolant/Reactor Coolant System Pressure", Revision 11, Section B, Step 2.2.3, requires that the operator verify the high pressure injection is operating properly as evidenced by injection flow in all four legs (MU-V16 A/B/ C/D), with flow indicated on MU 23-FE 1, 2, 3, 4.

Contrary to above procedure requirement, the high pressure injection flow was throttled to minimum flow conditions by 0405 hours and one of the two high pressure injection pumps, was stopped during the continuing reactor coolant system low pressure conditions.

Moreover, the high pressure injection system was routinely modified (flow throttled and makeup pumps stopped) throughout the day of March 28, 1979, during continuing low pressure conditions within the reactor coolant system following the period when the reactor coolant pumps were stopped and the high pressure injection system was the only certain mode available for the removal of core decay heat (Section 2.6, Details I).

Q. Technical Specification 6.8.1.a requires that written procedures shall be implemented, covering the activities referenced, as applicable, procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972.

Appendix "A" of Regulatory Guide 1.33 recommends procedures for normal plant operations; such as "Operation at Hot Standby".

Operating procedure 2102-3.3, Revision 5, March 17, 1978, Decay Heat Removal, Step 2.1.3, required that the RCS Th (not leg temperature) be maintained 30 degrees below the saturation temperature corresponding to pressurizer pressure in order to prevent boiling on the hot legs.

Contrary to the above, the system was placed onto the natural circulation mode at 950 psig RCS pressure at 528⁰F. The procedural requirements would indicate the pressure requirements to be about 1100 psig (saturation pressure for 558⁰F) (Section 2.10, Details I).

R. Technical Specification 6.8.1.a requires written procedures shall be implemented, covering the activities referenced as applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972. Appendix "A" of Regulatory Guide 1.33 recommends administrative procedures concerning "log entries". Unit 2 Administrative Procedure 1012, Shift Relief and Log Entries, Revision 8, October 4, 1977, establishes the requirements for shift relief and recording station operating activities in logs or other controlled documents on a shift basis (section 1.1) and describes the various shift records and logs involved and instructions required to maintain these records (Section 1.2, Details I).

Contrary to the above requirements numerous significant, safety-related log entries were not entered into the shift foreman and control room logs between 0400 hours and 1315 hours on March 28, 1979, (Section 2.12, Details I).

S. Technical Specification 6.8.1.a requires that written procedures shall be implemented, covering the activities referenced as applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972. Appendix "A" of Regulatory Guide 1.33 recommends procedures for combating emergencies; such as "Loss of Coolant". Unit 2 Emergency Procedure 2202-1.3, "Loss of Reactor Coolant/Reactor Coolant System Pressure", Revision 11, Section B, Step 3.1, requires the verification that all ES equipment is in its ESF position by observing all equipment status lights indicate as show in Table B-1; including the core flood tank valves (CF-V1A and V1B).

Contrary to the above, following the initiation of ES with continuing low reactor coolant pressure conditions, both core flood tank valves (CF-V1A and V1B) were closed at about 0600 hours on March 28, 1979, and remained closed for an undetermined period of time between about 0600 hours and 1300 hours (Section 4.6, Details I).

T. Technical Specification 6.8.1.a requires that written procedures shall be implemented, covering the activities referenced as applicable procedures, recommended in Appendix "A" of Regulatory Guide 1.33, November 1972. Appendix "A" of Regulatory Guide 1.33 recommends procedures for safety-related systems, including the emergency power sources. Unit 2 Operating Procedure (2104-6.2, Emergency Diesels and Auxiliaries, Revision 9, March 14, 1979) establishes the control of the emergency diesel generators. Procedure Step 2.2.3 requires that the Limiting Condition for Operation of Technical Specification 3.8.1.1 must be met, depending on the MODE of operation, unless these conditions cannot be met.

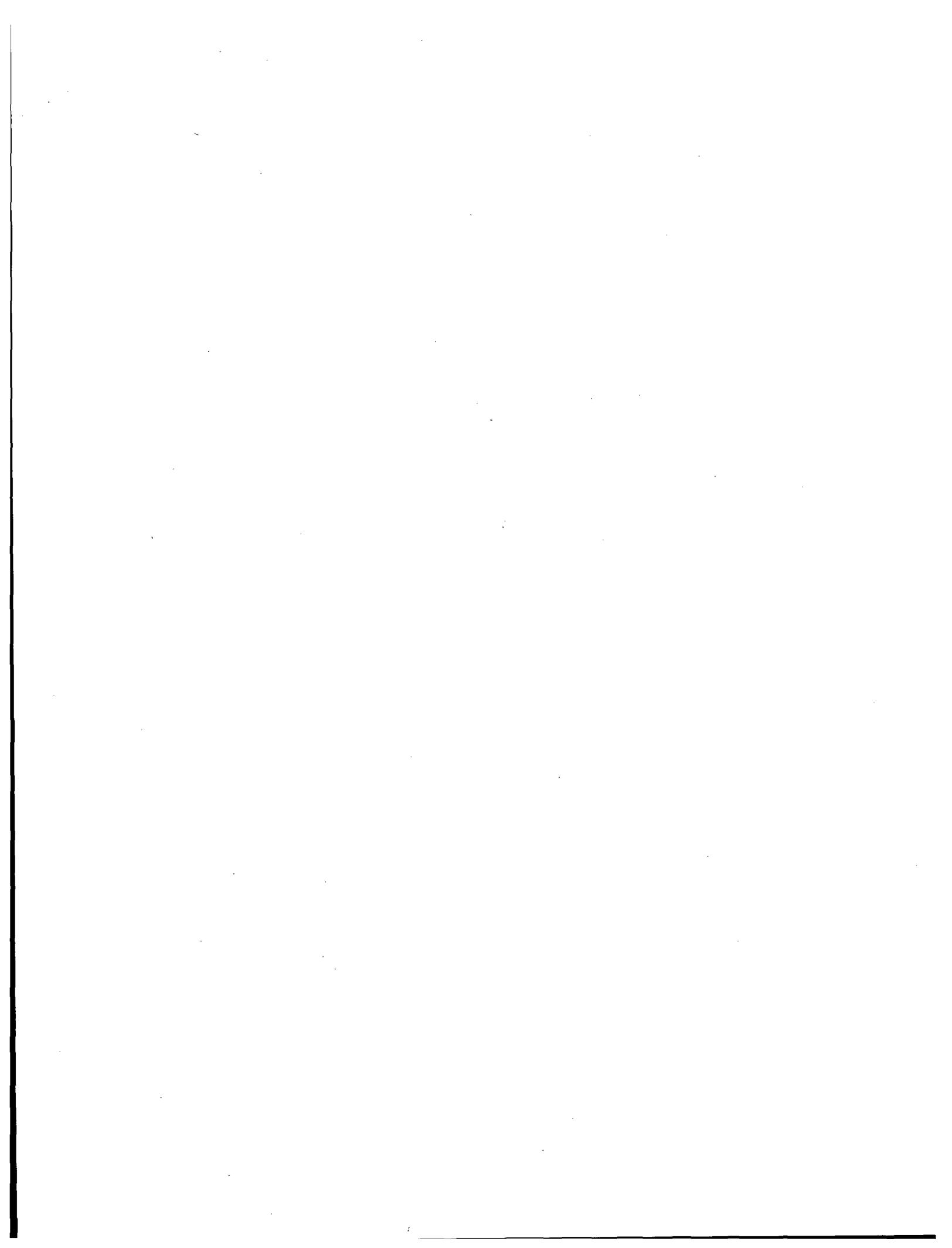
Procedure Sections 4.10, "Diesel Generator Automatic Start Upon Engineered Safety Features Actuation"; 4.6, "Diesel Generator Shutdown to Emergency Standby"; and 4.2, "Placing the Diesel Generator in Emergency Standby

Mode", specifically control the return of the emergency diesel generator to the EMERGENCY STANDBY mode unless the requirements of the Limiting Condition for Operation of Technical Specification 3.8.1.1 cannot be met.

Contrary to the above procedural requirements, the diesel generator fuel racks were tripped at about 0430 hours and remained tripped until about 0949 hours on March 28, 1979, rendering them incapable of being started by an automatic or manual start signal. Furthermore, the Unit 2 control room "Emergency Standby/Maintenance Exercise" switch was placed in the "Maintenance Exercise" positions on or about 0949 hours on March 28, 1979, enabling only manual starts of the diesel generators. Following the tripping of the fuel racks of the emergency diesels at about 0430 hours, the emergency diesel generators failed to respond to seven automatic initiation ESFAS signals between 0430 hours and 1950 hours on March 28, 1979 (Section 4.17, Details I).

DETAILS - II

RADIOLOGICAL ASPECTS



DETAILS II

TABLE OF CONTENTS

	<u>PAGE</u>
1. PREACCIDENT CONDITIONS	
1.1 ORGANIZATION AND RADIATION PROTECTION/CHEMISTRY STAFFING.....	II-1-1
1.2 TRAINING	II-1-5
1.2.1 Emergency Plan Training.....	II-1-5
1.2.2 Radiation Protection Training.....	II-1-12
1.3 EMERGENCY DRILLS.....	II-1-17
1.4 INSTALLED PLANT EQUIPMENT.....	II-1-22
1.4.1 Area, Process, and Atmospheric Monitors.....	II-1-22
1.4.1.1 Operational Status.....	II-1-22
1.4.1.2 Calibration.....	II-1-24
1.4.1.3 Setpoints.....	II-1-25
1.4.2 Radwaste Systems.....	II-1-26
1.4.2.1 Liquid Radwaste System.....	II-1-26
1.4.2.2 Gaseous Radwaste System.....	II-1-28
1.4.2.3 Heating and Ventilation Systems....	II-1-28
1.4.3 Communications Equipment.....	II-1-32

1.5	RADIATION PROTECTION EQUIPMENT AND SUPPLIES.....	II-1-34
1.5.1	Portable Radiation Survey Instruments.....	II-1-34
1.5.2	Radiochemistry Laboratory Equipment.....	II-1-36
1.5.3	Personnel Thermoluminescent Dosimetry.....	II-1-37
1.5.4	Self-Reading Dosimeters.....	II-1-38
1.5.5	Internal Dosimetry.....	II-1-39
1.5.6	Respiratory Protective Devices.....	II-1-39
1.5.7	Protective Clothing and Supplies.....	II-1-41
1.6	EMERGENCY EQUIPMENT.....	II-1-42
1.7	ENVIRONMENTAL MONITORING PROGRAM.....	II-1-44
1.7.1	Sampling Equipment.....	II-1-44
1.7.1.1	Air Samplers.....	II-1-44
1.7.1.2	Water Samplers.....	II-1-45
1.7.1.3	Environmental Thermoluminescent Dosimetry.....	II-1-46
1.7.2	Radiological Environmental Monitoring Program.....	II-1-47
1.7.3	Meteorological Monitoring Program.....	II-1-49
2.	INITIAL EMERGENCY RESPONSE	
2.1	DETECTION AND CLASSIFICATION.....	II-2-1
2.2	ORGANIZATION ACTIVATION.....	II-2-8
2.3	NOTIFICATIONS AND COMMUNICATIONS.....	II-2-21

2.3.1	Notifications.....	II-2-21
2.3.2	Communications.....	II-2-22
3. ASSESSMENT, CORRECTIVE, AND PROTECTIVE ACTIONS		
3.1	EFFLUENT MONITORING AND CORRECTIVE ACTIONS.....	II-3-1
3.1.1	Liquid and Gaseous Pathways.....	II-3-1
3.1.2	Determination of Release Rates and Amounts...	II-3-16
	3.1.2.1 Noble Gases.....	II-3-16
	3.1.2.2 Iodine.....	II-3-18
	3.1.2.3 Liquid Releases.....	II-3-22
3.2	IN-PLANT ASSESSMENT AND PROTECTIVE ACTIONS	II-3-24
3.2.1	Radiation Protection and Chemistry Prior to Declaration of The Emergency.....	II-3-25
3.2.2	Initial In-Plant Assessment and Protective Actions.....	II-3-29
3.2.3	Implementation of Emergency Plan.....	II-3-35
3.2.4	In-Plant Radiation Protection.....	II-3-46
3.2.5	Summary of Radiation Protection Findings.....	II-3-67
	3.2.5.1 Facilities.....	II-3-67
	3.2.5.2 Equipment.....	II-3-68
	3.2.5.3 Procedures.....	II-3-68

3.2.5.4	Training.....	II-3-68
3.2.5.5	Dosimetry.....	II-3-69
3.2.5.6	Air Sampling.....	II-3-69
3.2.5.7	Respiratory Protection.....	II-3-70
3.2.5.8	High Radiation Area Control.....	II-3-70
3.2.5.9	Management Control.....	II-3-70
3.3	ENVIRONMENTAL ASSESSMENT AND PROTECTIVE ACTIONS.....	II-3-71
3.3.1	Initial Offsite Dose Calculation.....	II-3-71
3.3.2	Initial Confirmatory Surveys.....	II-3-77
3.3.3	Offsite Radiation Measurements.....	II-3-79
3.3.3.1	Direct Radiation Measurements.....	II-3-80
3.3.3.2	Airborne Radioactivity Measurements.....	II-3-90
3.3.3.3	Radiological Environmental Monitoring Program Data.....	II-3-91
3.3.3.4	Assessment/Response to Environmental Measurement Results.....	II-3-94

TABLES

II-1-1	SUMMARY OF RADIATION PROTECTION TRAINING PROGRAM	II-1-12
II-1-2	SUMMARY OF RADIATION/CHEMISTRY TECHNICIAN TRAINING	II-1-15
II-1-3	AUXILIARY BUILDING RADWASTE CAPACITY	II-1-27

II-1-4	THERMOLUMINESCENT DOSIMETER LOCATIONS	II-1-48
II-2-1	EMERGENCY CONDITIONS AND TIMES MET	II-2-7
II-3-1	SUMMARY OF SURVEY DATA	II-3-31
II-3-2	OFFSITE DOSE CALCULATION RESULTS (ACTUAL) - PROCEDURE 1670.4	II-3-72
II-3-3	OFFSITE DOSE CALCULATION RESULTS (RECONSTRUCTED BY INVESTIGATORS) - PROCEDURE 1670.4	II-3-76
II-3-4	RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM DATA AVAILABLE TO THE LICENSEE BY MARCH 30, 1979	II-3-92

FIGURES

II-2-1	REACTOR COOLANT PRESSURE VS TIME	II-2-2
II-2-2	"NORMAL" EMERGENCY ORGANIZATION	II-2-9
II-2-3	EMERGENCY ORGANIZATION IN EFFECT 3/28/79, 0730-0900	II-2-12
II-2-4	EMERGENCY ORGANIZATION IN EFFECT 3/28/79, 0900-1100	II-2-13

II-2-5	EMERGENCY ORGANIZATION IN EFFECT 3/28/79, 1100-2000	II-2-15
II-3-1	SUMP PUMP DISCHARGES TO THE INDUSTRIAL WASTE TREATMENT SYSTEM	II-3-7
II-3-2	ON-ISLAND (GE SERIES) SURVEY POINTS AND NEAR-FIELD TLD LOCATIONS	II-3-73
II-3-3	DOME MONITOR, HP-R-214	II-3-75
II-3-4	TMI - GENERAL AREA MAP	II-3-82
II-3-5	ON-ISLAND (GE SERIES) SURVEY POINTS	II-3-83
II-3-6	PREDICTED PLUME LOCATION (1700-2400, MARCH 28)	II-3-86
II-3-7	PREDICTED PLUME LOCATION (0340-0540, MARCH 29)	II-3-87
II-3-8	PREDICTED PLUME LOCATION (0540-0910, MARCH 29)	II-3-88

REFERENCES

APPENDICES

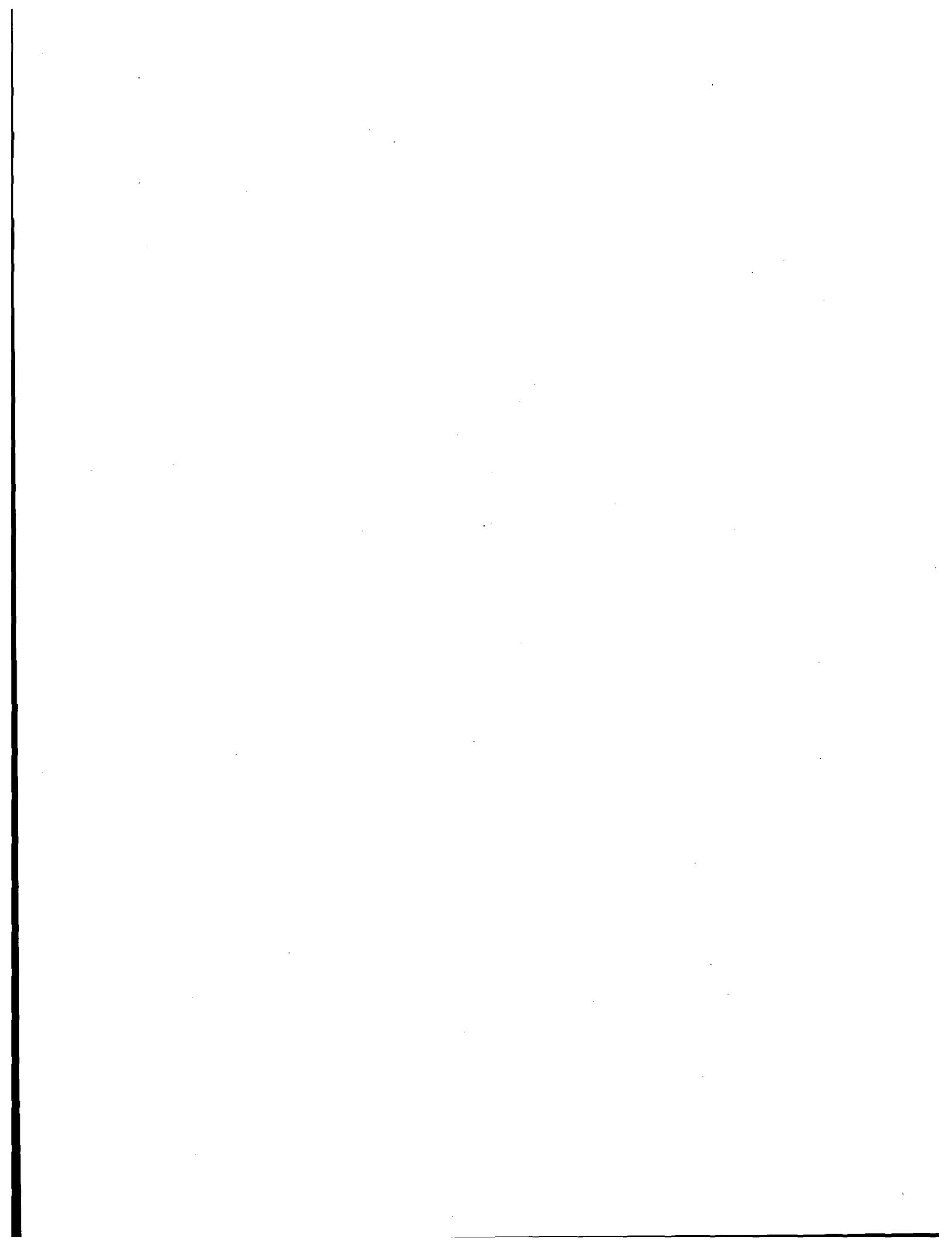
- II-A RADIOLOGICAL SEQUENCE OF EVENTS
- II-B OCCUPATIONAL DOSE CALCULATIONS
- II-C DIRECT RADIATION MEASUREMENTS MADE BY METROPOLITAN
EDISON COMPANY ON AND OFF THREE MILE ISLAND

II-D CAUTIONS REGARDING THE USE OF INSTALLED

RADIATION MONITORING SYSTEM DATA

II-E ACRONYMS, DEFINITIONS AND UNITS OF MEASURE

II-F POTENTIAL ITEMS OF NONCOMPLIANCE



1. PREACCIDENT CONDITIONS

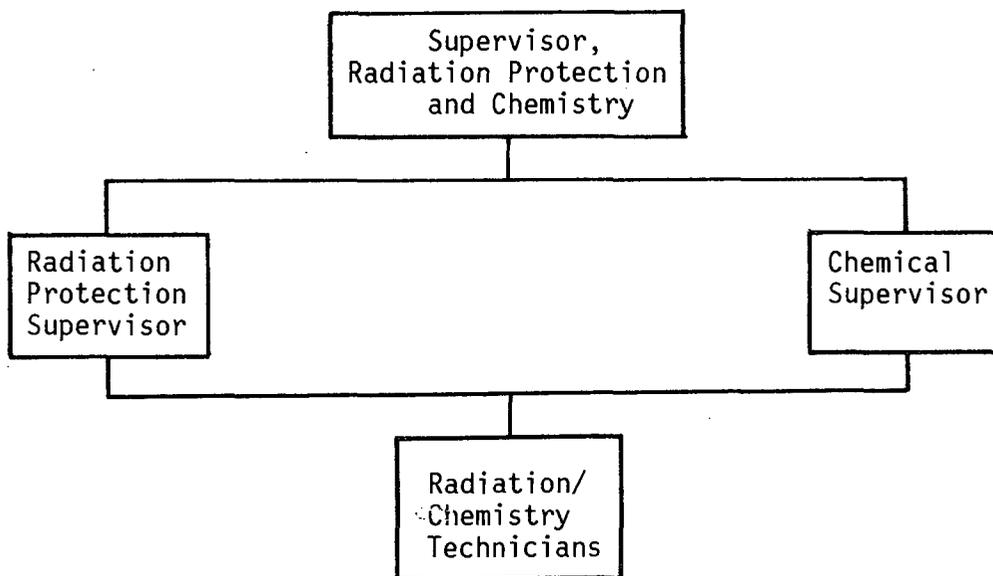
1.1 ORGANIZATION AND RADIATION PROTECTION/CHEMISTRY STAFFING

The investigators reviewed the licensee's organizational configuration in effect on March 28 to determine if it was consistent with the organizational configuration shown in Figure 6.2-1 of the licensee's Technical Specifications as required by Technical Specification 6.2.2. A comparison of the actual organization with the organization described in Figure 6.2-1 revealed the following discrepancies:

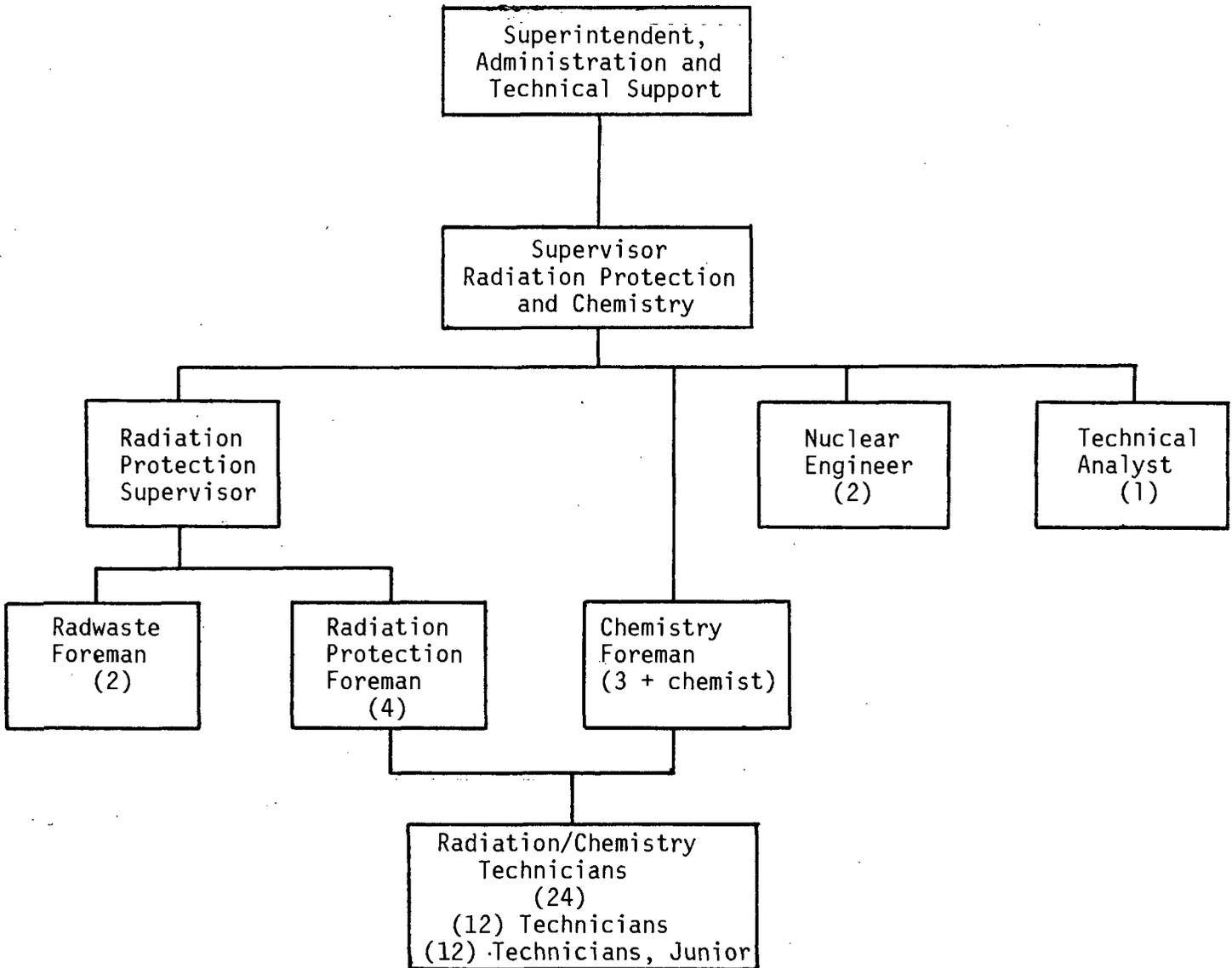
- a. A position titled "Superintendent of Administration and Technical Support" was added to the organization on September 18, 1978 and filled on March 1, 1979. This means the Supervisor, Radiation Protection and Chemistry reported through this new position rather than directly to the Station/Senior Unit Superintendent (Int. 94).
- b. On March 5, 1979, the "Station Superintendent" was promoted and his title changed to "Station Manager". In this new position, he reported directly to the Vice President, Generation rather than through the Manager, Generation Operations.
- c. The position of "Chemical Supervisor" has been vacant since the issuance of the Technical Specification (Int. 20).
- d. There are two "Supervisors of Maintenance" (one for each unit), not one, as shown in the Technical Specification (Int. 120).
- e. A position titled "Superintendent of Maintenance" has been added so that the Supervisors of Maintenance no longer report directly to the Station Superintendent (Station Manager)/Senior Unit Superintendent. The individual designated as the Superintendent of Maintenance also fills the position of Unit 1 Supervisor of Maintenance (Int. 120):

The licensee had discussed the promotion of the Station Superintendent to Station Manager and the change in reporting to the Vice President, Generation by telephone with the Region I Office of the NRC on March 5, 1979. In a followup letter to the Director of the NRC Region I Office, also dated March 5, 1979, the Vice President, Generation reiterated the changes discussed during the referenced telephone conversation (Ref. 1). The licensee did not, however, make application for or receive an amendment of the Technical Specifications prior to implementing the organizational changes described above. The effect of organizational discrepancies c, d, and e on the licensee's response to the March 28, 1979 incident is discussed further in Sections 2.2 and 3.2.

The organization of the radiation protection and chemistry staff as shown in Figure 6.2-1 of the Technical Specifications is shown as follows:



The actual organization of the radiation protection and chemistry staff prior to March 28, 1979 is shown as follows (Int. 94, 124, 133):



The resume for each member of the radiation protection staff was reviewed to determine compliance with the requirements of Technical Specification 6.3, which states:

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

ANSI N18.1-1971, Section 4.5.2 states:

"Technicians in responsible positions shall have a minimum of two years of working experience in their speciality. These personnel should have a minimum of one year of related technical training in addition to their experience."

The review and interviews with the technicians indicated five of the twelve radiation/chemistry technicians did not appear to have 1 year of related technical training in chemistry or radiation protection. Nine of the twelve radiation/chemistry technician juniors did not appear to have 1 year of related technical training.

No apparent item of noncompliance was identified since the term "should", as used in ANSI N18.1-1971, Section 2.2.1, denotes a recommendation, not a requirement.

The review indicated six radiation/chemistry technician juniors did not have 2 years of working experience in their speciality. The phrase "technician in responsible position" is not defined in any regulatory requirement. Based

on interviews with each technician with less than 2 years of experience, it appears that they acted in responsible positions during this emergency; however, no one of them acted alone or directed the activities of others during the period covered by this investigation (Int. 45, 49, 51, 75, 78, 96).

Section 6.2.2.d of the Technical Specifications states, "An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor." To meet this requirement at 0400 on March 28, three radiation/chemistry technicians and one junior technician were on site performing their routine assignments. The distribution of assignments placed two technicians performing radiation protection functions and two technicians performing chemistry functions (Int. 42).

1.2 TRAINING

1.2.1 Emergency Plan Training

The investigators examined training records and interviewed licensee personnel to determine if individuals had been trained in their emergency duties as required by the Emergency Plan (Ref. 2, 5).

Paragraph 6.1 of the licensee's Emergency Plan requires that:

- a. Individuals having emergency responsibilities be trained annually in these duties,
- b. Lesson plans be provided for each category of emergency training,
- c. Training include periodic examinations or assignments, and that
- d. Offsite agencies be invited to attend established training programs.

In January 1978, the licensee developed a written description of the detailed implementation of the emergency organization training program described in Paragraph 6.1 of the Emergency Plan. This description was incorporated into Procedure

1670.9, "Emergency Training and Emergency Exercise," Revision 4, dated January 16, 1978. This procedure specified by job title the assignment of emergency duties and the training to be provided as follows:

<u>Emergency Function</u>	<u>Persons Assigned</u>
Emergency Director	Station Superintendent Unit Superintendent Unit Superintendent-Technical Support Supervisor of Operations Shift Supervisor Shift Foreman
Accident Assessment (Group 1, Operational)	Supervisor of Operations Shift Supervisor Shift Foreman Control Room Operators
Accident Assessment (Group 2, Radiological)	Supervisor, Radiation Protection and Chemistry Radiation Protection Foremen Radiation/Chemistry Technicians Shift Supervisors Nuclear Engineers
Radiological Monitoring Team	Radiation/Chemistry Technicians Auxiliary "A" Operators
Repair Party Team	Maintenance Shift Workers
Division Support Personnel	Supervisor, Radiation Safety & Environmental Engineering Radiation Protection Specialist

Site Protection Personnel

Supervisor of Plant Security
Site Protection Sergeants
Site Protection Officers

Investigators compared Calendar Year 1978 and first quarter 1979 training attendance records with individual job titles and potential emergency duties to determine if all individuals were trained as required (Ref. 5). Selected individuals were interviewed to determine if the training had been provided and was of the scope and nature required by Procedure 1670.9.

This review indicated that representatives from the following offsite agencies had participated in training provided by the licensee: (Ref. 5)

- a. Pennsylvania Bureau of Radiological Health
- b. Pennsylvania State Council of Civil Defense (now called Pennsylvania Emergency Management Agency, PEMA)
- c. Dauphin County Civil Defense
- d. Pennsylvania State Police
- e. Londonderry Township Fire Company
- f. Goldsboro Fire Department
- g. York Haven Police Department.

This review indicated several discrepancies as discussed below.

During 1978, the following personnel had not received required training: two potential emergency directors (Shift Foremen D and A); one potential radiological accident assessment individual (Shift Supervisor E); eight potential Radiological Monitoring Team members (Auxiliary Operators A, C, G, and K and Radiation/Chemistry Technicians B, G, Q and V); and 37 of 38 potential Repair Party Team members (Ref. 6).

Of the eight potential Radiological Monitoring Team members who were not trained, Radiation/Chemistry Technicians B, G, Q, and V stated that 1977 entries listing SAM-2 (portable instrument used for measuring radioiodine in the environment) training and 1978 entries listing Radiation Monitoring Team training on their individual training records were in error because (1) they had not received training in the operation of the SAM-2 and (2) the Radiological Monitoring Team training in 1978 was not of the scope and content outlined in Procedure 1670.9 (Refs. 2, 4, 7, 8 and Int. 154, 156). They further stated that they had discussed their lack of understanding of their emergency duties (and in particular their inability to operate the SAM-2) with the Radiation Protection Supervisor in September 1978. Shortly thereafter, they were given a short briefing on emergency response actions during a backshift (the shift that they worked) but they were not instructed in operation of the SAM-2 until approximately 1 month after the March 28 incident. The investigators determined that it was the briefing on backshift response that had been entered into the radiation/chemistry technicians' training records as "Radiation Monitoring Team training." An interview of the Radiation Protection Supervisor verified that the training records were in error and that Radiation/Chemistry Technicians B, G, Q and V did not receive training in 1977 on SAM-2 operation. The interview further established that the 1978 training of Radiation/Chemistry Technicians B, G, Q and V did not meet the requirements of Procedure 1670.9 (Int. 172).

The only Repair Party Team training conducted in 1978 was one session held on October 10, 1978. Attendees were the Superintendent of Maintenance and five maintenance foremen. This session was intended to enable the attendees to train the workers they supervised who were assigned as potential members of the Repair Party Team (Refs. 2, 5).

By the end of 1978, site protection, accident assessment and operations personnel training were the only Emergency Plan training courses, which included "tests or assignments" as required by Procedure 1670.9 (Ref. 9).

The investigators noted that Procedure 1670.9 did not contain provisions for training individuals who would perform the duties of chemistry supervisor during emergencies.

On January 18, 1979, a Repair Party training session was held. Attendees were mechanical maintenance shift workers. The instructor was a maintenance foreman who had not been trained himself (Ref. 5).

A record of a telephone conversation between the Group Supervisor, Technical Training and the Supervisor of Mechanical Maintenance dated February 1, 1979, indicated that the Repair Party Team training requirement had been incorrectly interpreted to apply only to mechanical maintenance shift workers (Ref. 10). Electrical and instrument and control shift maintenance workers had therefore not been scheduled for training.

Four Repair Party training sessions were conducted in March 1979 by Shift Maintenance Foreman B. Maintenance Foreman B had not received any Repair Party Team training prior to conducting these sessions (Ref. 5). In a discussion with an investigator, Maintenance Foreman B stated that he did not fully understand the material he presented and that he was not sure that the training he provided was correct. Consequently, he decided not to submit the completed Training Program Administrative forms to the Training Department. He further stated that he had expressed his concerns to the Supervisor, Radiation Protection and Chemistry and to the Radiation Protection Supervisor and had requested additional guidance from them. On the Training Program Administrative Forms for these classes which he prepared but did not submit, Maintenance Foreman B wrote, "wish to reschedule more specific type training when received from [Supervisor, Radiation Protection and Chemistry] (Ref. 5)."

An investigator examined records of the Training Department's attempts to complete employee training in emergency duties. In a memorandum dated April 5, 1978, the Supervisor of Training (Acting) outlined to seven managers and supervisors

the Emergency Plan training requirements detailed in Revision 4 of Procedure 1670.9 (Ref. 11). On September 21, 1978, the Group Supervisor, Technical Training met with the Radiation Protection Supervisor and the Supervisor, Technical Training to discuss incomplete training (Ref. 12).

On November 28, 1978, the Group Supervisor, Technical Training sent a memo to the Superintendent of Maintenance, requesting a schedule of Repair Party Team training and reminding him that training had to be completed by the end of 1978 (Ref. 13). On November 29, 1978, the Group Supervisor, Technical Training sent a memo to Engineer F, who was assigned lead responsibility for emergency planning. The memorandum reiterated that lesson plans in nine areas were incomplete (Ref. 14). On December 20, 1978, the Group Supervisor, Technical Training, informed Engineer F that, if the lesson plans and training were not complete by the end of 1978, the licensee would be in noncompliance with regulatory requirements (Ref. 15). Nuclear Engineer F stated to investigators during an interview that the lesson plans had been completed but copies had not been sent to the Training Department (Int. 66). A written record of a telephone conversation on February 2, 1979 between the Supervisor, Radiation Protection and Chemistry, and the Group Supervisor, Technical Training, indicated that, once again, the subject of emergency plan training was discussed (Ref. 16). In this conversation, the Group Supervisor, Technical Training, outlined the training requirements of Procedure 1670.9 to the Supervisor, Radiation Protection and Chemistry and pointed out that the facility was in noncompliance with regulatory requirements in several areas related to the training program.

A record of a telephone conversation dated February 9, 1979 between the Group Supervisor, Technical Training and the Superintendent of Maintenance reiterated that all maintenance shift workers, not just mechanical maintenance workers, had to receive Repair Party Team training and that the licensee had been in noncompliance with regulatory requirements for over a month (Ref. 17).

On February 3, 1979, Revision 6 of Procedure 1670.9 was issued (Ref. 4). This change stated that only those individuals trained in specific duties could be assigned those duties in an emergency, thereby deleting the requirement that all individuals with specific job titles be trained in potential emergency duties. This change essentially substituted a requirement to limit persons assigned duties in an emergency for the training requirement.

In comparing the emergency duties of selected employees during the period March 28 to March 30, 1979, with training records, an investigator determined that several individuals who had not received Procedure 1670.9 training were assigned to Radiological Monitoring Teams and Repair Party Teams (Refs. 2, 5). (See Sections 2.2 and 3.2 for details.)

Interviews with individuals who were trained in accordance with Procedure 1670.9 indicated that the training provided was inadequate for them to properly perform their assigned emergency functions (Int. 154, 156, and 187). Several Radiation Monitoring Team members stated that they were unsure of whether to make radiation surveys with the meter window open or closed. Engineer C, who performed the initial offsite dose projection calculations on the morning of March 28, and Engineer D, who assisted him, both indicated that they did not understand how to properly read the containment dome monitor (HP-R-214) used as a basis for their calculations (Int. 174). The actions and resulting radiation exposures of other emergency workers demonstrated a general lack of knowledge of radiation protection precautions to be observed during emergencies. There was an admitted lack of understanding of how to effectively decontaminate skin contaminated with radioactive iodine. During an interview, the Supervisor, Radiation Protection and Chemistry stated, "In the four and a half years that I've been here... I don't think we've ever faced an iodine contamination problem... and personally, I don't think that I really... had the off-the-cuff knowledge to handle that. It took some time and some talking with people to start getting to a point where I felt comfortable with... iodine contamination and external contamination" (Int. 79). Users of emergency monitoring equipment demonstrated

and expressed a lack of knowledge regarding operation of the SAM-2 (Int. 154 and 156) and doubts about which of the elements of emergency organization teams were responsible for the functions of in-plant radiation control, emergency repair functions, and notifications to offsite agencies. (See Sections 2.2, 2.3, 3.2, and 3.3.1 for details relating to the situations discussed above.)

1.2.2 Radiation Protection Training

Radiation protection training required for individuals at Three Mile Island is described in Procedure 1690, "Training Requirements" Revision 6, dated March 22, 1978, which states that the Radiation Protection Department is responsible for the development and implementation of the program (Ref. 18). The program was developed to meet the needs of different plant groups. Listed below is a summary of training programs and the groups they served.

TABLE II-1-1

SUMMARY OF RADIATION PROTECTION TRAINING PROGRAM

<u>Program Title</u>	<u>Group Served</u>
1. Basic I	Temporary personnel on site less than one day.
2. Basic II (1 hour)	Temporary personnel working outside restricted areas.
3. Basic III (3 hours)	Permanent personnel working outside restricted areas. Temporary personnel in restricted areas for more than one day.

- | | |
|---|--|
| 4. Intermediate I (3 hours) | All radiation workers. All personnel under radiation work permits (RWP). |
| 5. Intermediate II (8 hours) | Maintenance personnel, engineers, supervisors, others requiring radiation work permit clearance. |
| 6. Advanced Radiation Protection (2 weeks) | Auxiliary operators, control room operators, senior licensed operators |
| 7. Comprehensive Radiation Protection (3 months) | Radiation/Chemistry Technicians |
| 8. General Employee Training (No time specified) | Selected temporary personnel (all permanent personnel once per year) |
| 9. Training for local fire, police and, civil defense departments (no time specified but details of training described in Procedure 1670.9) | |

The investigator reviewed the Generation Division, TMI Station Resume to determine if 50 selected individuals had received the appropriate radiation protection training described in Procedure 1690 (Ref. 7). This record showed that, for the individuals reviewed, all had received the appropriate radiation protection training.

Training of the Radiation Protection Department was reviewed in detail. Most radiation/chemistry technicians interviewed expressed dissatisfaction with the extent and quality of the radiation protection training they had received in recent years. (Int. 42, 47, 49, 50, 58, 68, 78, 82, 89, 96, 98, 99, and 106)

Section 13.2.1.5 of the Final Safety Analysis Report (FSAR), states that radiation/chemistry technicians will receive annually 40 hours of "Rad Chem Tech Training Program" and 8 hours of "general employee training/retraining" and "on-the-job training."

Amendment "A" to Procedure 1690, "Three Mile Island Comprehensive Training for Radiation Chemistry Technicians/JRS," Revision 6, dated March 22, 1978, contains documentation sheets which, when signed by an instructor, verify that an individual has successfully demonstrated adequate knowledge of a specific topic. Amendment A states that Section III training should be completed every 2 years. A review of Section III documentation sheets showed that only one radiation/chemistry technician had any items signed off as completed. However, some training had been provided to the radiation/chemistry technicians as documented on the Generation Division TMI Station Resume (Ref. 7).

Table II-1-2 summarizes the radiation protection and chemistry related training as documented in the Station Resume that had been provided for 22 of the 24 radiation/chemistry technicians. Two technicians are not included since they have been employed for a short time (less than 3 months). The table shows that for Calendar Year 1978, 19 of 22 technicians did not receive 40 hours of radiation protection and chemistry training. No technicians received the 8 hours of General Employee Training. All technicians appear to have received some on-the-job training. For Calendar Year 1977, 15 technicians did not receive 40 hours of radiation protection and chemistry training and 12 technicians did not receive 8 hours of General Employee Training. All but one technician appears to have received on-the-job training.

Technical Specification 6.4 states, "A retraining and replacement training program for the unit staff shall be maintained under the direction of the Director-Generation Training and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix A of 10 CFR Part 55."

TABLE II-1-2

SUMMARY OF RADIATION/CHEMISTRY TECHNICIAN TRAINING

	Radiation Protection	1978 Hours of Training				1977 Hours of Training				
		Chemistry	On Job Training	General Employee	Drills	Radiation Protection	Chemistry	On Job Training	General Employee	Drills
Rad. Chem. Tech. O	7	80*	24	4	6	324**		8		6
Rad. Chem. Tech. A	8		24	4	3	4		33		6
Rad. Chem. Tech. P	7	6	24	4	13	238**	80*	20		7
Rad. Chem. Tech. Q	17		24	4	5	4		32		8
Rad. Chem. Tech. B	9	1	24	4	3	4	88*	20	4	9
Rad. Chem. Tech. R	13		24	4	5	238**	80*	20		2
Rad. Chem. Tech. C	8		24	4		3	8	32	8	8
Rad. Chem. Tech. D	15	1	24	4	4	5	8	28	8	16
Rad. Chem. Tech. E	11	83*	24		6	7	8	30	4	7
Rad. Chem. Tech. F	9	1	24	4	5	3	8	29	8	5
Rad. Chem. Tech. M	11	85*	24		6	4		32	8	7
Rad. Chem. Tech. H	12	2	24		5	4	11	28	8	4
Rad. Chem. Tech. I	5		24	4	3	4		28	8	4
Rad. Chem. Tech. T	9		24	4	5	240**	86*	21	4	5
Rad. Chem. Tech. U	13	2	24	4		237**	80*	22	4	8
Rad. Chem. Tech. N	8	1	24	4	3	4		28	2	9
Rad. Chem. Tech. G	8	2	24	4	6	3	1	33	8	6
Rad. Chem. Tech. J	10	1	24	4	3	5**	8	20	4	11
Rad. Chem. Tech. V	12		24	4	5	239**	81*	20	4	11
Rad. Chem. Tech. W	14	81*	24	4	4	5		8		11
Rad. Chem. Tech. K	26	1	24	4	3	8	9	39	8	5
Rad. Chem. Tech. L	11	2	24	4	5	3	2			

NOTES:

- Hours of training have been rounded off to the highest whole number.
- *Indicated a two week vendor water chemistry school.
- **Indicates 6-8 week comprehensive Health Physics school provided on site by contractor.
- The 24 hours of on-the-job training for 1978 was described on the "Training Program Administrative Form: as "Unit II HP and Chem Startup." During interviews, no technician could describe this training. Item 7 on the "Training Program Administrative Form" titled "Instructor/Trainee's Evaluation" contained the following instructor's comment, "During the past 6 months, the techs have been getting good instructions in both HP and Chem aspects of Unit 2 Start-up."
- The only training provided in 1979 prior to the accident was a 4-hr chemistry class to six technicians.

Section 5.5 of ANSI N18.1-1971 (Ref. 19) states:

"5.5 Retraining and Replacement Training.

A training program shall be established which maintains the proficiency of the operating organization through periodic training exercises, instruction periods, and reviews covering those items and equipment which relate to safe operation of the facility and through special training sessions for replacement personnel. Means should be provided in the training program for appropriate evaluation of its effectiveness."

5.5.1 Retraining

A retraining program should include:

1. Plant startup and shutdown procedures;
2. Normal plant operating conditions and procedures;
3. Operational limitations, precautions, and set points;
4. Emergency plans and security procedures;
5. Abnormal operating procedures;
6. Emergency shutdown systems;
7. Changes in equipment and operating procedures;
8. General safety, first aid, and radiation safety;
9. Alarms and instrumentation signals; and
10. Operation of selected auxiliary systems important to overall plant safety."

In reviewing the retraining described by Procedure 1690 and from interviews with representatives of the Radiation Protection Department, an investigator determined Items 1, 2, 3, 5, 6, 7, 9 and 10 of Section 5.5.1 of ANSI N18.1-1977 were not included in the retraining program for radiation/chemistry technicians. One item of particular importance was the method for informing technicians of changes in procedures. According to the Radiation Protection Supervisor, a note was placed on the bulletin board stating that a procedure had been changed

(Int. 124). No acknowledgement sheet was routed to ensure that each technician read the change. Acknowledgement sheets were not used, according to the Radiation Protection Supervisor, because the technicians had refused to sign them.

The Supervisor Radiation Protection and Chemistry and the Radiation Protection Supervisor stated that there was no formalized retraining program for themselves or their foremen (Int. 20 and 124). Station records showed all had received some retraining but the training received did not include all of the areas specified in Section 5.5.1 of ANSI N18.1-1971 (Ref. 7).

Members of the radiation protection and chemistry staff provided the majority of radiation protection training for other members of the facility staff.

The licensee has not maintained a retraining program for members of the Radiation Protection Department that met or exceeded the requirements and recommendations of Section 5.5 of ANSI N18.1-1971.

1.3 EMERGENCY DRILLS

Paragraph 6.1.2 of the Emergency Plan requires the licensee to conduct, among others, an annual Site or General Emergency drill and an annual Repair Party Drill. The procedure to be followed in planning, conducting, and documenting drills is specified in Procedure 1670.9, "Emergency Training and Emergency Exercise."

An investigator reviewed available documentation and interviewed licensee employees to verify that an annual Site or General Emergency Drill and a Repair Party drill were performed during Calendar Year 1978, as required.

During Calendar Year 1978, the licensee conducted seven radiation emergency drills having a scope equivalent to Site/General Emergencies. One of these drills included Repair Party activities. The date and general scenario of each drill were as follows (Ref. 20):

<u>Date</u>	<u>Scenario</u>
October 23, 1978	Waste Gas Decay Tank Release (Unit 1)
October 25, 1978	Radioactive Liquid Release (Unit 2)
October 27, 1978	Once-Through Steam Generator (OTSG) Tube Rupture (Unit 1)
October 30, 1978	Loss of Water from Spent Fuel Pool with Fuel Raised (Unit 1)
November 2, 1978	Loss-of-Coolant with severe core damage, fuel melting and little core cooling, containment leak at maximum design rate (Unit 2)
November 6, 1978	Same as above, but Unit 1
November 8, 1978	OTSG Tube Rupture, Unit 1.

During each of the seven drills, licensee observers were positioned at various locations to evaluate the response organizations' activities. NRC representatives were present for and witnessed the conduct of the November 8, 1978 drill. (Results of this inspection were reported in IE Inspection Report Number 50-320/78-34.) Each of the seven drills was followed by a critique in which the discussion highlighted areas for further evaluation and possible improvement.

An investigator reviewed action item assignment sheets that were prepared pursuant to Procedure 1670.9 after the series of seven radiation emergency drills held in 1978. The purpose of this review was to determine the status of the items on March 28 (Ref. 21). This review indicated that the following action items were still open:

- a. Post RM-L7 sensitivity, alert, and high-alarm values adjacent to the No. 6 recorder in the Unit 2 control room,
- b. Ensure that newly revised site maps [in monitoring kits] replace those outdated ones,
- c. Ensure that emergency monitoring equipment has had an inventory and operational check out prior to use (especially SAM-2),
- d. Change accountability, Procedure 1670.7,
- e. Check out operation of Unit 2 450 MHz radio communications and black phone line circuit with offsite agencies,
- f. Remove labels from Radeco air samplers used in kits,
- g. Review radiation monitoring team communications with recommendations for improvement, and
- h. Establish routine surveillance program for GAI page system.

The following action items were identified as being closed:

- a. Check page system units for proper operations in both units; ensure merge-isolate switch is correct,
- b. Ensure all operations personnel know where to report and know their responsibilities during an emergency (especially auxiliary operators),
- c. Need additional status board in control room,
- d. Replace emergency kit air sampler,
- e. Change Emergency Plan for new telephone numbers,

- f. Ensure operations personnel read calculations Procedure 1670.4, especially liquid release calculations,
- g. Engineers should immediately relieve the operators in the ECC using communications hardware and the status board,
- h. Investigate walkie-talkie availability for emergency use only,
- i. Communications improvement ECC/ECS,
- j. Need communications hardware for green office building (west side),
- k. Check monitoring kits for high-range dosimeters,
- l. ECS dose assessment/RMS/ECS communications coordination,
- m. Ensure Site and General Emergency criteria are reviewed by all operations personnel,
- n. Ensure replacement of AC power cord in SAM-2 Kit No. 2,
- o. Check out SAM-2 instruments.

The investigator compared the list of "open" action items with events that took place on March 28 to determine if their being open led to any degradation of the licensee's response.

The "open" action item relating to inventory and operational check out of emergency equipment prior to use had a parallel occurrence on March 28. A radiation monitoring team left the site for Goldsboro at approximately 0730. They did not inventory/check their emergency monitoring kit before leaving. Upon arrival in Goldsboro at approximately 0830 they found that the SAM-2 in the kit would not operate.

With regard to the "open" action item regarding a change to Procedure 1670.7, the investigators noted that the licensee's actions on March 28 appeared to constitute noncompliance with Technical Specifications relative to the implementation of changes to Emergency Plan Implementing Procedures. As a result of the Calendar Year 1978 emergency drills, licensee management determined that the method for conducting accountability and assembly of personnel needed to be improved. The task to revise the applicable Emergency Plan Implementing Procedure 1670.7 was assigned to the Radiation Protection Supervisor (Ref. 21). On October 13, 1978 the Radiation Protection Supervisor issued a memorandum to all departments outlining a revised assembly and accountability method (Ref. 22).

Through interviews with site protection personnel, the investigator determined that the proposed method of assembly and accountability outlined in the October 13 memorandum was implemented during the response to the March 28 incident (Int. 72, 52, 32). The investigators determined that initially some members of the site protection force followed the accountability Procedure 1670.7, while others followed the method described in the October 13 memorandum. This created some degree of confusion and delayed the prompt attainment of full accountability.

The investigator also noted that Procedure 1670.5, "Onsite Radiological Monitoring," had not been revised to include the dispatch of an assembly area monitor to the North Warehouse (an additional assembly area established by the October 13 memorandum). During an interview of the Radiation Protection Supervisor, he acknowledged that the procedure should have been revised before the memorandum was issued. He also stated that Procedure 1670.7 had not been revised by March 28, 1979 (Int. 172).

The investigators determined that the use of the accountability and assembly method described in the Radiation Protection Supervisor's October 18 memorandum is an item of noncompliance with Technical Specification 6.8.2, which requires that Emergency Plan Implementing Procedures be reviewed and approved by the Plant Operations Review Committee (PORC) prior to implementation.

The investigator also compared the action items listed as "closed" with events that took place on March 28 to determine the adequacy of the corrective actions.

Of the items listed as "closed," two items appear to have recurred. Based on the demonstrated performance of individuals in recognizing and declaring the emergency, it appears that the review of the Site and General Emergency criteria with operations personnel was not adequate. Interviews with the shift operating crew of March 28 indicated some uncertainty regarding the applicability of certain conditions to Site and General Emergency criteria still remained (Int. 145) (see Section 2.1).

The check out of SAM-2s appeared to have corrected an operational problem existing at the time. However, since two of four SAM-2s were inoperable on March 28, it is apparent that the corrective action was not adequate to prevent recurrence of random inoperability (Ref. 23, Int. 82) (see Section 1.5).

The investigators noted that there were no explicit regulatory requirements pertaining to the verification of the timeliness of corrective action implementation or evaluation of the adequacy of corrective action.

1.4 INSTALLED PLANT EQUIPMENT

1.4.1 Area, Process, and Atmospheric Monitors

1.4.1.1 Operational Status

The investigators determined the operational status of area, process, and atmospheric monitors on March 28 by review of selected sections of records (Ref. 24-40), discussions with licensee staff, and observation of monitor operation.

The Out-of-Service Logs showed that the status of the monitors on March 28 was as follows (Ref. 41, 42):

<u>Description</u>	<u>Total Installed</u>	<u>Total in Service</u>
Unit 2 area monitors	21	20
Unit 2 atmospheric monitors	16	15
Unit 2 liquid monitors	12	12
Unit 1 area monitors	15	15
Unit 1 atmospheric monitors	10	8
Unit 1 liquid monitors	9	8

The following process, area and atmospheric monitors or monitor recorders were out-of-service on March 28 (for the reasons stated in the Out-of-Service Logs):

- a. Unit 2 makeup tank area radiation monitor (HP-R-206) - "bad cables" (since February 15, 1979)
- b. Unit 2 waste gas decay tank 1A gas monitor (WDG-R-1485) - pump inoperable (since February 16, 1979)
- c. Unit 2 liquid discharge monitor readout (RM-L7) - "signal to the RM-L7 recorder is false" (since March 13, 1979)
- d. Unit 1 miscellaneous sump discharge liquid monitor (RM-L8) - "decon liner" (since April 22, 1977)
- e. Unit 1 radiochemical laboratory and nuclear sampling room air monitor (RM-A12) - "wiring not hooked up" (since April 18, 1978)
- f. Unit 1 control tower air monitor (RM-A1) - pump inoperable (since February 22, 1979).

The record of the weekly surveillance check of RM-A12 conducted on March 17 shows the monitor was out-of-service. Records of monthly surveillance performed on March 18 and calibration on March 19, showed that the monitor was operational on March 19 (Ref. 35, 39, 43). The weekly surveillance check performed on March 24 showed the monitor out of service again (Ref. 40). Maintenance Foremen E and F, in discussions, and Radiation Protection Foreman B, in an interview, stated that there had been problems with the pump on the sampler in the past (Int. 21). However, Maintenance Foreman F thought the pump had been repaired. He said that he recalled that, when he examined the monitor after the calibration on March 19, the pump was operational. He stated that the monitor would very frequently alarm when samples were taken and therefore it was unplugged to stop the irritating noise from the alarm. He stated that he had discussed with Radiation Protection Foreman D the possibility of changing

the sampler alarm setpoints so that the sampler would be used, but this change had not been made before March 28. Because of the high dose rates in the area, the sampler had not been checked since March 28.

If the Unit 2 makeup tank area radiation monitor had been operational during the accident, it could have provided an accurate measure of increasing radiation levels in the auxiliary building. This monitor did respond to increasing radiation levels, but it did not respond accurately. The nuclear sampling room air monitor could have been used to collect air samples when reactor coolant samples were taken in this room. Because the control tower air monitor was inoperable, there was no real-time monitor of Unit 1 control room air radioactivity until the monitor was placed back in service on March 30 at 0600.

Unit 2 Technical Specifications 3.3.3.1 and 3.3.3.6; Unit 2 Environmental Technical Specifications 2.1.1.A, 2.1.2.A, and 2.1.2.B.1; and Unit 1 Environmental Technical Specifications 2.3.2.A and 2.3.1 require that certain radiation monitors be operable. The monitors required by these Technical Specifications were operable on March 28.

1.4.1.2 Calibration

The licensee is required by Unit 2 Technical Specifications 4.3.3.1 and 4.3.3.6, by Unit 2 Environmental Technical Specification 2.1.1, and by Unit 1 Technical Specification 4.1.1 to channel check, functionally test, and calibrate certain process, area, and atmospheric monitors at specified frequencies. Sections 11.4 and 12.1 of the Final Safety Analysis Report (FSAR) list other monitors installed in Unit 2 that are not required by technical specifications.

Investigators examined records of monitor calibrations, functional tests, and channel checks done in normal surveillance to determine if monitors were calibrated on March 28 (Ref. 24-36, 38-40, 44). All monitors required by Technical Specifications were in calibration, but certain discrepancies were identified. The preoperational test results of the atmospheric monitors, "Radiation System Test - Atmospheric Monitors," TP 360/1A, were incomplete. The tests showed that ventilation exhaust flow rates in the auxiliary and fuel

handling buildings were different than designed (Ref. 26). Engineer M stated in discussions that the air sampler flow rates had been adjusted for isokinetic sampling and that the test would be signed as complete after new sampler nozzles were installed and all sampler flow rates adjusted to be the same (Ref. 45).

The vent stack monitor (HP-R-219) sampler consisted of two sample probes and a flow measuring probe located at different elevations in the center of the stack (Ref. 44, 45, 46). The licensee considered, prior to startup of Unit 2, modifications to exhaust sampling systems in order to meet the recommendations of Regulatory Guide 1.21, "Measuring, Evaluation and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light Water Cooled Nuclear Power Plants," but did not make the modifications since Regulatory Guide 1.21 was not a requirement (Ref. 47).

Preoperational test results, "Radiation Monitoring System Test - Area Monitors" TP 360/1C, showed that although the output from the containment dome monitor (HP-PR-214) to the meter in the control room was calibrated, the output was not calibrated to the readings on the chart recorder (Ref. 24). The chart recorder scale read from 0.1 to 10^4 mR/hr but the output from the meter ranged from 0.1 to 10^7 mR/hr. The output from the meter read less than full-scale on the chart recorder (10^4 mR/hr) when the meter read full-scale (10^7 mR/hr). Work Request 2761 for installation of a different recorder was cancelled (Ref. 48). Maintenance Foreman G stated in discussions that an instrument technician had begun to calibrate the monitor output to the existing recorder but the work was not completed prior to the accident. There was no regulatory requirement that this monitor have a chart recorder. Any value for this monitor taken off the chart during the period March 28 to 30 would have been in error. A record of the readings from this monitor could have been useful for postaccident evaluation.

1.4.1.3 Set Points

In examining the bases for alarm setpoints for Unit 2 monitors, an investigator determined that the setpoints for air particulate monitors were incorrectly based on one-half of the Unit 1 Technical Specification release

rate limit of 0.63 $\mu\text{Ci}/\text{sec}$ rather than the one-half of Unit 2 Technical Specification limit of 0.3 $\mu\text{Ci}/\text{sec}$. The licensee had reduced the alarm setpoint by 25% to account for possible meter error (Ref. 49). Even though the basis for the setpoint was incorrect, the alarm setpoint was set less than the Unit 2 Technical Specification release rate limit.

An investigator also determined that the alarm setpoint for the reactor coolant letdown monitor was based on operation of the monitor without an attenuating shield, but Engineer M stated that the shield was in place (Ref. 44). If this monitor were used to determine the amount of radioactivity in the reactor coolant, it would result in an underestimation.

1.4.2 Radwaste Systems

1.4.2.1 Liquid Radwaste System

An investigator reviewed the Primary Auxiliary Operator's Log to determine the status of the radwaste system prior to the reactor trip at 0400 March 28 (Ref. 50). Liquid radwaste system tanks were filled to near 60% capacity (Refs. 50, 51). The Supervisor, Radiation Protection and Chemistry stated that Unit 2 depended on Unit 1 for radwaste processing support. Unit 1 was starting up from a refueling outage and generating radwaste that required processing in order to continue startup. A Unit 1 spill of 20,000 gallons of contaminated water from the fuel transfer canal into the reactor building generated more radwaste near the end of the outage than would normally be present (Int. 79). The investigator concluded that there were no known liquid radwaste problems existing at the time of the accident that caused the accident to be significantly worse than it otherwise would have been. Table II-1-3 shows the capacity of the Unit 2 radwaste system as recorded sometime between 2300 March 27 and 0400 March 28 (Ref. 50).

TABLE II-1-3
AUXILIARY BUILDING RADWASTE CAPACITY

Sump/Tank Name	Number Designation	Capacity ⁽¹⁾		Last Routine ⁽²⁾ Reading (ft)	Approximate Percent Capacity Remaining
		Gal	Ft		
Auxiliary Bldg Sump		7645	6+	3.8	37%
Auxiliary Bldg Sump Tank	WDL-T-5	3215	4.5+	3.4	24%
Miscellaneous Waste Storage Tank	WDL-T-2	20040	10.25+	6.6	33%
Neutralization Tank	WDL-T-8A	8780	10.75	10.5	< 1%
Neutralization Tank	WDL-T-8B	8780	10.75	10.0	< 1%
Contaminated Drain Tank	WDL-T-11A	2660	5.25+	4.0	23%
Contaminated Drain Tank	WDL-T-11B	2660	5.25+	1.25	76%
Reactor Coolant Bleed Holdup Tank	WDL-T-1A	82280	13	6.5	59%
Reactor Coolant Bleed Holdup Tank	WDL-T-1B	82280	13	9.0	39%
Reactor Coolant Bleed Holdup Tank	WDL-T-1C	82280	13	9.0	39%
Evaporator Condensate Test Tank	WDL-T-9A	11863	14	0.4	≈ 100%
Evaporator Condensate Test Tank	WDL-T-9B	11863	14	11.5	18%
Concentrated Waste Tank	WDS-T-2	9649	10	0	100%
Containment Spray Room Sump	1A	359	4	2.3	43%
Containment Spray Room Sump	1B	359	4	1.7	58%
Decay Heat Removal Sump	1A	359	4	2	50%
Decay Heat Removal Sump	1B	359	4	2.5	38%

(1) From FSAR and plant procedure 2104-4.1

(2) Primary Auxiliary Operator's Log for March 28 prior to 0400.

1.4.2.2 Gaseous Radwaste System

An investigator reviewed United Engineers and Constructors, Inc., "Data Sheets, Testing Data of Piping Systems," MTX No. 124.3.1, 124.3.2, 124.3.3, 124.3.4, 124.3.5 and 124.3.6. These data sheets indicated that the gaseous radwaste system had been pressure tested by sections prior to final functional testing. "Waste Gas Disposal Functional Test," TP 231/3, had been completed and approved by March 21, 1979. This procedure contained documentation that several deficiencies noted during the functional test were addressed by either correcting the deficiency or changing operating procedures to compensate for the deficiency.

There have been a number of maintenance problems with the waste gas system since plant startup. Examples are:

Work Request 4979 Waste gas compressor, WDG-P-1B, overpressurized

Work Request 4985 Waste gas compressor, WDG-P-1A, makes loud noise.

Work Request 4262 Waste gas compressor, WDG-P-1A, no seal water level in tank, check level control pump operation.

These work requests were outstanding at the time of the accident. In addition, Control Room Operator J stated that makeup tank vent valve MU-V-13 was suspected to be leaking (Int. 57). Some of these problems may have caused releases to be larger than they would have otherwise been.

1.4.2.3 Heating and Ventilation Systems

An investigator reviewed the following documents pertaining to the auxiliary building and fuel handling building ventilation systems:

Auxiliary Building Ventilation Functional Test, TP 173/2

Fuel Handling Building Ventilation Functional Test, TP 177/2

Filter Efficiency Test, (SP-2)

Burns and Roe Field Questionnaire No. 590

Burns and Roe Field Questionnaire No. 2513

GPU Start-up Problem Report 2646

Amendment 3 to Operating License No. DPR-73, March 24, 1978

Attachment 2 to Operating License No. DPR-73, February 8, 1978

Technical Specifications 3.9.12 and 4.9.12.

In addition, an investigator discussed the history of these ventilation systems with the Unit 1 Maintenance Supervisor; GPU SC Project Site Manager, Superintendent of Maintenance; Supervisor, Radiation Protection and Chemistry; and Unit 2 Maintenance Supervisor.

The history of these filters prior to the accident may have had a significant impact on their performance during and after the accident.

During the testing of the auxiliary building and fuel handling building ventilation systems on February 13-16, 1978, leakage through bypass dampers AH-D-4020A on the auxiliary building and AH-D-5683 and AH-D-5890 on the fuel handling building caused the tests to be unacceptable (Ref. 52). The bypass dampers were temporarily sealed and acceptance retesting was performed on March 9, 1978. After retest, the filters met or exceeded the acceptance criteria of 99.97% removal of dioctylphthalate (DOP) in the HEPA filters and 99.95% removal of freon in the charcoal adsorbers (Ref. 52).

No removal efficiency was performed after the bypass lines were semipermanently sealed on April 7, 1978. No retesting has been performed since that date. There were no regulatory requirements to conduct retests. No record of any other alterations has been found at this time.

Investigators attempted to determine if the filters could have been poisoned by fumes from large scale painting operations. The GPUSC Project Site Manager stated that any painting performed in the fuel handling and auxiliary building after acceptance testing of the filter trains was done by a long-term contractor who is no longer onsite. The GPUSC Project Site Manager also stated that he knew of no way to establish how much painting was done, if any. During tours of the auxiliary building and fuel handling building by an investigator, there were no indications of recent large-scale painting operations. Several high radiation areas were not examined.

A review of testing requirements showed that there were no technical specification requirements for particulate and halogen collection efficiency determinations for the auxiliary building ventilation system filters.

Technical Specification 3.9.12 requires the fuel handling building air cleanup systems to be operable whenever irradiated fuel is in the pool. There has been no irradiated fuel in the pool. However, the Units 1 and 2 fuel pools are not physically divided for ventilation flow purposes and the Unit 2 fuel handling building exhaust ventilation draws from this common air space. There is irradiated fuel in the Unit 1 fuel pool.

Technical Specification 4.9.12.c requires laboratory analysis of a representative sample from the fuel handling building charcoal adsorbers after every 720 hours of operation. Item F.2 of Attachment 2 to operating license DPR-73 defers the applicability of Technical Specification 4.9.12.c until replacement of charcoal adsorbers during the first refueling outage.

The Supervisor, Radiation Protection and Chemistry stated that the charcoal halogen adsorbers were ordered by Burns and Roe before they learned that they (fuel handling building filters only) must meet Regulatory Guide 1.52, Revision 1,

July 1976 guidelines for methyl iodine removal efficiency. Testing of the ordered adsorber indicated that the removal efficiency was slightly less than the regulatory guidelines. According to discussion Item No. 6 of "Safety Evaluation by the Office of Nuclear Reactor Regulation supporting Amendment No. 3 to Facility Operating License No. DPR-73," the deferment of applicability for Technical Specification 4.9.12.c was intended to cover minor deviations from the specified chemical analysis requirements of Regulatory Guide 1.52, Revision 1, July 1976.

GPU Startup Problem Report 2646, February 14, 1978, discussed the problem of leaking bypass dampers and referenced Field Question No. 2513, March 30, 1978. The resolution was to "Block off the following dampers on either the upstream or downstream side with 18 gauge sheet metal making sure all joints are airtight: AH-D-5890, AH-D-5683, and AH-D-4020A. In terminal box IT2782 lift and tag wire "45" of cable IT 27826 in terminal box IT 3119, lift and tag wire "31" of cable IT 3120C. Wires should be tagged to indicate their removal per this ECM (ECM S-5915 dated March 24, 1978)." Step seven of Field Question No. 2513 indicates that ECM S-5915 was completed on April 7, 1978. Field Question No. 2513 was approved by the Plant Operations Review Committee on April 3, 1978. Subsequently, ECM S-5915 Revision 1 was initiated to purchase and install new bypass dampers of a type that would not permit leakage when closed. Proposed intallation was to be during the first refueling.

A "Burns and Roe Nuclear Safety Review" on ECM S-5915 was signed on March 24, 1978, and ECM S-5915 Revision 1 was signed on June 20, 1978. The review of ECM S-5915 Revision 1 does not appear to address adverse effects on the adsorber that might result from continuous use, although the system description stated that the purpose of the bypass system was to prolong the life of the filter trains (Ref. 53).

It was determined from discussions with the Unit 2 Maintenance Supervisor and the Supervisor, Radiation Protection and Chemistry that, since the completion of acceptance testing, (approximately 1 year prior to the accident) all ventilation flow from the fuel handling and auxiliary buildings had been through the filter banks. According to the Superintendent of Maintenance and Shift Supervisor F, there were no run-time records kept on these filter banks.

1.4.3 Communications Equipment

The March 28 availability and operability of the following normal and emergency communications equipment listed in paragraph 5.4 of the Emergency Plan were determined through discussions with licensee personnel and review of available records:

- a. Radiation Emergency Alarm;
- b. Public Address System;
- c. Dial telephone (Metropolitan Edison Tie Line);
- d. Direct current (battery) powered telephones;
- e. Pennsylvania Bell telephone;
- f. Dauphin County Radio Monitor;
- g. Metropolitan Edison System Radio (Utility Radio);
- h. Frequency modulation (FM) radio; and
- i. NAWAS (National Warning System, connected to Pennsylvania Emergency Operations Center and State Police Troop Headquarters).

An investigator determined that the only inoperative item was the Dauphin County radio monitor in the Unit 2 control room. This radio monitor is a frequency scanner and, prior to March 28, it would not receive certain emergency frequencies. Work Request No. C1406 was prepared on February 27, 1979 to correct the problem (Ref. 54). The set was still at an offsite repair facility and was not in service on the morning of March 28. The absence of this item had no effect on the licensee's ability to communicate or coordinate with the Dauphin County Civil Defense and other agencies since it was only intended to monitor radio communications. The licensee used telephone to communicate with Dauphin County Civil Defense and other officials during the emergency and had backup communications provisions through the utility radio.

A licensee evaluation of radio communications between onsite and offsite Radiation Monitoring Teams, the Emergency Control Station (ECS), and control rooms had been ongoing since shortly after a 1977 radiation emergency drill.

Areas under evaluation were: (1) the portable walkie-talkie transmit range, number available, and "dead spot" locations, (2) the audibility of the plant page system in all areas within the site boundary, (3) the procedure to ensure merging of the Unit 1 and Unit 2 communication systems during emergencies, (4) timely reporting and repair of communication system malfunctions, (5) installation of an override to the plant page external speaker timer, (6) and improvement of the Emergency Control Station Emergency Control Center communication link.

For the 9 listed available communications systems, the following were the methods in existence for testing the operability of the systems (Ref. 55):

- a. Radiation Emergency Alarm - tested every Friday at noon
- b. Public Address System - no surveillance testing, however, maintenance was to be performed when malfunctions were reported
- c. Dial Telephone (Metropolitan Edison Tie Line) - no surveillance testing, however, maintenance was to be performed when malfunctions were reported
- d. Direct Current (Battery) Powered Telephones - no surveillance testing, but to be checked periodically
- e. Pennsylvania Bell Telephone - no surveillance testing, but maintenance was to be performed when malfunctions were reported
- f. Metropolitan Edison System Radio (Lebanon Frequency) - the transmitter was to be checked yearly for frequency and modulation per Federal Communications Commission regulation; was to be checked weekly for proper operation
- g. Dauphin County Radio Monitor - to be tested yearly
- h. Frequency Modulation (FM) Radio - to be frequency calibrated every 6 months

- i. NAWAS - to be tested weekly via NAWAS roll call.

An investigator noted that, with the exception of the Radiation Emergency Alarm, there are no regulatory requirements in this communications area.

1.5 RADIATION PROTECTION EQUIPMENT AND SUPPLIES

1.5.1 Portable Radiation Survey Instruments

Section 12.3.2.3 of the FSAR states, "Portable radiation survey instruments are maintained under the responsibility of radiation protection personnel. A sufficient inventory is available to allow for periodic calibration, maintenance and repair. These include detection instruments for measuring alpha, beta, gamma, and neutron radiation. Monitoring instruments and count rate meters are located at the Access Control Point." Table 12.4-1 of the FSAR lists several types of instruments, the type of radiation detected, and their range and use."

An investigator examined Forms 1763-1 maintained pursuant to Health Physics Procedure 1763, "H.P Instrument Calibration/Status," Revision 1, dated December 1, 1977, to determine the availability of portable radiation protection equipment on site at 0400 March 28. Portable radiation survey instruments were considered available for use if they were operable and had been calibrated within the frequency specified in Health Physics Procedure 1749, "Portable/Lab Instrument Calibration Schedule," Revision 4, dated December 1, 1977. The results of this review are summarized below.

<u>Instrument</u>	<u>Description</u>	<u>Total Inventory</u>	<u>Available for Use on March 28</u>
Eberline Teletector	Wide range beta-gamma; maximum gamma 1000 R/hr	16	4
Eberline PIC-6A	Wide range beta-gamma; maximum gamma 1000 R/hr	14	4
Eberline RO-2	Intermediate range beta- gamma; maximum gamma 5 R/hr	12	5
Eberline E-520	Intermediate range beta- gamma; maximum gamma 2 R/hr	14	6

Eberline PNR-4	Intermediate range neutron; maximum neutron 5 rem/hr	2	0
Victoreen 808 Vamp	Intermediate range gamma; maximum gamma 100 mR/hr	5	0
Eberline PAC-4S	Alpha count rate maximum 200 Kcpm	2	0
Eberline RM-14 and HP-210	Beta-gamma count rate maximum 50 Kcpm	18	18
Radeco Grab Sampler	Air sample collection devices	14	14
Staplex Grab Sampler	Air sample collection devices	4	2
UNICO Grab Sampler	Air sample collection devices	1	0
MSA Model F	Air sample collection devices	5	5

The above table indicates that less than one-half of the inventory of portable radiation dose rate instruments was available for use at the time of the accident. The Radiation Protection Supervisor (Int. 124) and Radiation Chemistry Technicians C, I, and L (Int. 42, 58, and 82) indicated this shortage was not unusual considering that Unit 1 had just completed a refueling outage. These same individuals indicated that the availability of portable survey instruments had been a chronic problem.

Radiation survey instrument maintenance was the responsibility of the Instrument and Control Section. Calibration of survey instruments was the responsibility of the radiation/chemistry technicians. Radiation/Chemistry Technician O stated that a few days prior to the accident he and another technician attempted to calibrate over a dozen instruments (Int. 78). Three of the instruments could be calibrated. While performing an inventory of survey instruments, the technician reported finding "most of our instruments in the repair shop, and when they weren't in the shop being worked on, they were back in the parts cage, sitting, because they didn't have parts to fix it. That was the status of the instruments. Very few of them were available. When you wanted an instrument you almost had to fight for it." The Radiation Protection Supervisor stated

that they had repeatedly contacted the Instrument Section to repair survey instruments; but, because of apparent manpower shortages, the repairs never seemed to be done (Int. 124). In an effort to alleviate this problem, the licensee was evaluating the possibility of having an outside vendor perform radiation survey instrument repair and calibration service.

Interviews with Auxiliary Operator C and radiation chemistry/ technicians showed that the lack of radiation survey instruments directly contributed to unnecessary personnel exposure as discussed in Section 3.2 of this report (Int. 104).

Many additional survey instruments and air samplers were necessary to support the in-plant and environmental monitoring after the accident. The Supervisor, Radiation Protection and Chemistry stated that there was a shortage of equipment because plans had not considered that an emergency would require such frequent operations in very high exposure rate areas. No clear regulatory requirement or licensee commitment established minimum inventories for portable radiation survey instruments at this facility.

1.5.2. Radiochemistry Laboratory Equipment

Section 12.3.2.7 of the FSAR states, "The Unit 1 Radiochemistry Laboratory in conjunction with a separate shielded Counting Room is used for radioactivity analysis. Counting equipment such as G-M, scintillation and proportional counters is available."

The status of radiochemistry or health physics laboratory radiation measurement equipment was established by review of the "Laboratory Instrument Background and Efficiency Log Book," discussions with members of the radiation protection group, and direct observation of equipment status by the inspector.

<u>Unit 1</u>			
<u>Instrument</u>	<u>Description</u>	<u>Total Inventory</u>	<u>Available For Use</u>
Ortec 401A	Gross beta	2	2
Ludlum 2000	Gross beta	2	2

Ludlum 2000	Gross alpha	1	1
Packard Model 3320	Liquid scintillation spectrometer system	1	1
Beakman Wide Beta 11	Gross alpha/beta	1	0
NMC Proportional Counter PCC-iiT	Gross alpha/beta	1	0
Lithium Drifted Germanium, Ge(Li), Multi-Channel Analyzer	Gamma	1	1

Unit 2

Ludlum 2000	Gross beta	3	2
Beckman	Liquid scintillation	1	0
Lithium Drifted Germanium, Ge(Li) Multi-Channel Analyzer	Gamma	1	0

Interviews with Radiation Protection Foreman A (Int. 7) indicated that the Ge(Li) and Beckman Liquid Scintillator had never been operable since startup of Unit 2.

Although there appears to be more equipment than listed in the FSAR, during the accident the use of vital (multi-channel analyzer) counting equipment was lost because of the presence of airborne radioactive material in the equipment area. This directly affected operations of the facility as described in Section 3.2.

1.5.3 Personnel Thermoluminescent Dosimetry

Section 12.3.3.1.2 of the FSAR states, "All radiation workers at the station are issued film badges/TLDs and are required to wear such at all times while within the controlled areas. Special or additional film badges/thermoluminescent dosimeters (TLD) are issued as may be required under unusual conditions at the discretion of radiation protection personnel. These special or additional devices are processed as required. The film badge/TLD of any individual is processed whenever it appears that an overexposure may have occurred."

The licensee maintained an onsite dosimetry program using a Harshaw, Inc., TLD System. The Harshaw system used a natural lithium fluoride (LiF) two-chip dosimeter and Model 2271 reader. Health Physics Procedure 1642, "Operation and Calibration of the Thermoluminescent Dosimetry System," Revision 1, dated September 28, 1977, provided the information necessary for the radiation/chemistry technicians to operate the dosimetry function. Radiation/Chemistry Technician C and the Radiation Protection Supervisor stated that no one individual had been assigned specific responsibility for dosimetry (Int. 82). Each radiation/chemistry technician performed this function as a routinely assigned task. A radiation/chemistry technician stated that one might be expected to read TLD's twice a year. Radiation/chemistry technicians stated that extremity monitoring was available by taping a TLD badge to the extremity.

The operation and management of an acceptable dosimetry system requires specialized training and consistent attention to detail. This concept was addressed in a review titled "Evaluation of the Health Physics Chemistry Organization at Three Mile Island Nuclear Station, Units 1 and 2," performed in September 1977 and again in a review performed by a consultant in March 1979 (Ref. 56). Both reviews suggested the need to assign one individual the sole responsibility for the dosimetry program. Implementation of this suggestion had not been made prior to March 28, 1979.

No regulatory requirements or license commitments establish minimum standards for management of personnel dosimetry systems.

1.5.4 Self-Reading Dosimeters

Section 12.3.3.1.1 of the FSAR states, "Self-reading dosimeters are issued in addition to a film badge/TLD to individuals as required. Each individual is instructed in the necessity of reading his self reading dosimeter at frequent intervals while in radiation areas."

An investigator examined forms 1772-1 maintained pursuant to Health Physics Procedure 1772, "Dosimeter Calibration and Leak Test," Revision 3, dated November 22, 1977 to determine the availability of self-reading pocket

dosimeters on March 28. This review showed that a sufficient number of 0-200 mR dosimeters were available for use. Fifty 0-5 R and fifteen 0-20 R dosimeters were tested and available from the Unit 1 health physics office.

The lack of an adequate supply of 0-5 R dosimeters directly contributed to one individual being exposed to whole-body gamma radiation in excess of a regulatory limit. This is further discussed in Section 3.2.

1.5.5. Internal Dosimetry

Section 12.3.3.3 of the FSAR states, "whole-body counts will be taken on a special basis if the nature of exposure or suspected exposure indicates a need for a whole-body count. This procedure acts as a confirming check on the adequacy of the station contamination control practices. The TMI Unit 2 Radiation Protection Procedures require that all contractor personnel provide a urine sample, which is used as a baseline sample. Urinalyses are also run on contractor people on completion of the job. The Radiation Protection Supervisor has the option of ordering additional bioassay analysis and/or whole-body counts on any personnel should the need arise."

Procedure 1628, "Program for Medical and Bioassay Examination," Revision 3, dated September 30, 1977, implements the internal dosimetry program. The whole-body counting and urinalyses were performed by offsite vendors. A whole-body counter was on site on March 28. This program meets present regulatory requirements.

1.5.6. Respiratory Protective Devices

Section 12.3.2.5 of the FSAR states, "Respiratory protective devices may be required in any situation in which airborne radioactive contamination exists or is expected. In such cases, the airborne concentrations are monitored for concentration and type of airborne contaminants present. The necessary protective devices are specified accordingly. Respiratory devices available for use include:

- a. Half-face respirator (Filter Type)
- b. Full-face respirator (Filter Type)
- c. Full-face respirator (Supplied Air)
- d. Self-contained breathing apparatus.

A self-contained breathing apparatus will also be used in any situation involving oxygen deficient atmospheres.

The respiratory protective program is designed in accordance with the practices recommended by the American National Standards Institute (ANSI Z88.2-1969)."

An investigator examined Forms 1616-1 maintained pursuant to Health Physics Procedure 1616, "Use of Respiratory Protection Devices," Revision 13, dated April 21, 1979, to establish the availability of these devices. As of February 1979, there were 50 self-contained breathing devices on site. Six of these were of the emergency egress type. In addition, about 15 backup air supply bottles were available. Based on interviews with licensee representatives (Int. 18, 124, 133) about 150 full-face respirators with particulate cartridges and about 25 half-face respirators were available. No iodine absorbing cartridges were available on site, since they had not been approved for protection factors.

There was a breathing air compressor located in the west end of Unit 1 intermediate building. This equipment had been used to refill air supply bottles. During the accident this compressor was not used since the quality of air being compressed could not be verified as acceptable because of high airborne radioactivity in the compressor suction area.

Based on interviews with Auxiliary Operator C (Int. 104) and Radiation/Chemistry Technicians E and H (Int. 49, 50), it was determined that an adequate supply of readily accessible self-contained breathing devices was not available during the first 2 days of the accident.

1.5.7. Protective Clothing And Supplies

Section 12.3.2.4 of the FSAR states "All personnel entering a contaminated area are required to wear specified protective clothing. The nature of the work to be done and the level of contamination are the governing factors in the selection of protective clothing to be worn. The protective apparel available includes shoe covers, head covers, gloves, and coveralls or lab coats. Additional items of specialized apparel, e.g., plastic or rubber suits, face shields and respirators, are available for operations as required. In all cases, radiation protection personnel evaluate the radiological conditions and specify the required items of protective clothing to be worn."

Based on discussions with licensee representatives and vendors (Int. 155), it appears that an adequate supply of protective clothing was available on site, including: paper and cloth coveralls; hoods; cotton, plastic, rubber, and neoprene gloves; plastic shoe covers, rubbers and rubber boots; plastic suits, beta glasses and surgeon caps. In addition, it appears there was an ample supply of step-off-pad material, radiation warning signs, labels, boundary rope, masking tape, and plastic bags.

Paragraph 4.2.1 of the Three Mile Island Site Emergency Plan states, "Appropriate on-site assessment surveys performed by the Radiation Monitoring Teams include general in-plant and site β - γ surveys, airborne particulate and gas sampling, assisting repair and rescue teams, personnel monitoring at access control points, and manning the North or South Gate monitoring points or assembly areas."

From the preceding statement, in conjunction with a review of the emergency equipment set aside for emergency use (see Section 1.6), it is apparent that the "routine" complement of radiation protection instrumentation is intended to be the primary source of instrumentation to support in-plant radiological controls and measurements during emergencies.

The licensee's reliance on the routine complement of health physics instrumentation and respiratory protection equipment to support in-plant β - γ

surveys, airborne particulate, and gas sampling and for assisting repair teams degraded the response to in-plant radiological conditions resulting from the emergency. Significantly greater numbers and types of instruments and equipment were needed to support the in-plant efforts than were actually available. Teletectors, high-range survey meters, and air packs were all in short supply. The loss of the counting room capability (GeLi) further compounded the in-plant efforts to establish radiological controls and necessitated the set up of substitute methods having less reliability.

An interview of the Supervisor, Radiation Protection and Chemistry substantiated the observed shortcomings in the area of equipment (Int. 25). Planners had not conceived of the type and extent of the maintenance and operations activities that would have to be performed in areas of such high radiation dose levels as were encountered.

1.6 EMERGENCY EQUIPMENT

Paragraph 4.2.2 of the Emergency Plan and Procedure 1778, "Emergency Readiness Checklist," establish an inventory of emergency supplies and equipment to be maintained at specific locations throughout the facility (Ref. 57). This equipment consisted of radiation detection and measurement devices, protective clothing and supplies, and procedures. The emergency kit designations and their locations were:

<u>Emergency Kit</u>	<u>Number</u>	<u>Location</u>
Emergency Monitoring	4	North Search Facility (Processing Center)
Washdown Area	2	North Search Facility (Processing Center)
Ambulance	1	Machine Shop
Emergency Clothing	1	Unit 1 Storage Area outside Access Control Point
Control Room	2	One in Each Control Room
Medical Emergency Cabinet	1	Control Building

Paragraph 4.2.2 of the Emergency Plan states, "Emergency Monitoring kits contain all instrumentation, protective clothing and supplies needed for on/off-site field radiation monitoring." Procedure 1778 lists the following total inventory of radiation detection or measurement instruments, respiratory protection devices, and dosimetry to be reserved for emergency use in the kits:

<u>Item</u>	<u>Number</u>	<u>Location</u>
TLDs	20	5 in each of 4 monitoring kits
Air Samplers (SAM-2)	4	1 in each of 4 monitoring kits
Stabilized Detector (RD-19)	4	1 in each monitoring kit
High-Range Dosimeters	10	5 in each control room kit
Dosimeter Chargers	2	1 in each control room kit
High-Range Survey Meters	2	1 in each control room kit
Half-Face Respirators (particulate cartridge)	20	10 in each control room kit
RM-14	1	Health Physics Lab
GM Survey Meter	1	Health Physics Lab

This procedure requires that the emergency equipment kits be inventoried and the instruments operationally checked at least once each calendar quarter.

To determine the status of the licensee's emergency kits on the morning of March 28, an investigator reviewed records of an emergency equipment inventory and operational check performed on March 11, 1979, and interviewed radiation/chemistry technicians who were the initial users of the equipment on the morning of March 28 (Ref. 23, Int. 82). The review of the inventory and operational check records showed that one of the four SAM-2s in the monitoring kits at the North Search Facility (Processing Center) was out-of-service. Radiation/Chemistry

Technician C stated that the SAM-2 was still out-of-service on the morning of March 28 and not located in the north search facility. Radiation/Chemistry Technician C further stated that during a preoperational checks of the instruments by one of the survey team members on the morning of March 28, a second SAM-2 was found to be inoperable (Int. 82). Consequently, only two of four field counting systems were operational and available during the first several hours of the licensee's emergency response.

Paragraph 5.5.4 of the Emergency Plan states that "at least two company vehicles can be quickly equipped...for off-site monitoring." On the morning of March 28, Radiation/Chemistry Technician C, assigned to an emergency survey team, had to "commandeer" a company vehicle to use for transportation to Goldsboro to perform offsite surveys, and it was necessary for Radiation/Chemistry Technician N to use his personal vehicle. (Int. 47, 82).

1.7 ENVIRONMENTAL MONITORING PROGRAM

1.7.1 Sampling Equipment

1.7.1.1 Air Samplers

An investigator observed the operation of an environmental air sampler and discussed sampler operation with the Supervisor, Radiation Protection and Chemistry and Radiation Protection Consultant A. The samplers consist of a sample head, a pump, and a dry test meter, a rotameter, and a timer (for locations see Section 1.7.2).

An investigator discussed the calibration checks of environmental air samplers with licensee personnel and examined surveillance records to determine the status of the samplers on March 28. This item was originally discussed in IE Report Nos. 50-289/78-08 and 50-320/78-16. Unit 2 Environmental Technical Specification 5.5 requires that written instrument calibration procedures be developed and followed. Unit 2 Environmental Technical Specification 3.2.A outlines the detailed sampling and analysis program. Unit 1 Procedure 1302-5.24, "Environmental Monitor Calibration," Revision 3 dated December 19, 1974 requires annual air sampler calibration checks (Ref. 58).

Review of the surveillance records and discussions with Maintenance Foreman G, the Radiation Protection Supervisor, and the Supervisor, Radiation Protection and Chemistry, showed no Unit 2 procedure equivalent to Procedure 1302-5.24. In discussions, Technical Analyst A stated that the Surveillance Task Index showed that the Unit 1 calibration of the eight environmental air samplers had not been scheduled to be done since 1974 (Ref. 59).

In discussions, Technical Analyst B stated that he thought that Procedure 1302-5.24 had been withdrawn by PORC several years ago because it did not contain technical specification requirements. The central procedure files contained no record that the procedure had been withdrawn. An investigator noted that the procedure was in the active operating procedure file in the Unit 1 control room on June 15, 1979 and was listed as active on the procedure list dated July 6, 1979 (Ref. 60).

To determine if the samplers were operating on March 28, an investigator examined records of sample collections just prior to and during the period March 28-31 (Ref. 61, 62, 63, 64) and the Work Request Log for the period March 1 through April 15, 1979. The records completed by the individuals collecting the samples showed on March 22 the Falmouth substation air sampler was blowing air out of the filter rather than pulling air through the filter and, on March 29 the Observation Center air sampler flow meter was erratic.

1.7.1.2 Water Samplers

An investigator discussed the operation of environmental water samplers with the Supervisor, Radiation Protection and Chemistry, Radiation Protection Consultant A, and Radiation Protection Consultant E. The licensee had two automatic liquid samplers, one at the Columbia water works, sampling downstream drinking water, and the other at York Haven Dam. When operating, these samplers drew 10 ml of water 10 times an hour. In discussions, the Supervisor, Radiation Protection and Chemistry stated that individuals collecting the samples were to check the sample volume each time a sample was collected as a check of the proper operation of the sampler. The other samples routinely taken were grab samples or grab composite samples.

To determine if the samplers were operating on March 28 to 30, an investigator examined records (Ref. 61, 62) of sample collections for the period September 27, 1978 through March 30, 1979 and the Unit 1 Work Request Log (the samplers were under Unit 1 maintenance) for the same period. Investigators discussed the samplers with Consultant E who collected the samples on March 29 and 31, 1979. The records showed that the sampler at the York Haven dam was operational, but that grab samples, rather than a composite sample, had been taken at Columbia since January 31, 1979. Prior to the accident, weekly grab samples had been taken at Columbia since the composite sampler was not operating. Radiation Protection Consultant E stated that the composite sampler at Columbia was not operating on March 29 or 31. Radiation Protection Consultant A and the Supervisor, Radiation Protection and Chemistry stated that there had been operational difficulties with the sampler in the past and that it had been replaced after the accident.

1.7.1.3 Environmental Thermoluminescent Dosimetry

The primary dosimeter used by the licensee at 20 locations (15 indicators and 5 controls) is the Teledyne Isotopes environmental dosimeter (Int. 30, 74). A representative of Teledyne stated in discussions that this dosimeter contains $\text{CaSO}_4:\text{Dy}$ elements with a minimum sensitivity of 0.5 mR. These elements are sandwiched between two 500 mg/cm² copper sheets to flatten the energy response of the dosimeter. The TLD elements and shields are contained in a rectangular plastic holder. After a readout, annealing at 250°C for one hour, an irradiation, and a re-annealing step, the dosimeters are sent from the Teledyne facility at Westwood, NJ to Three Mile Island. In-transit controls are sent with the dosimeters. After arrival at the site, the dosimeters are reannealed at 250°C for 1-2 hours prior to placing them in the field.

Ten of the twenty locations have RMC TLDs in addition to the Teledyne TLDs as a quality control check (Int. 30, 160). Each location has two RMC Model UD-200S dosimeters, each containing two $\text{CaSO}_4:\text{Tm}$ phosphorus inside of a plastic and metal shield (Ref. 65).

Functional dosimeters were in place at all twenty locations at the time of the March 28 accident. Dosimeter locations are shown in Table II-1-4. In addition to the seven TLDs located on Three Mile Island as shown in Table II-1-4, the licensee also had distributed, as part of an informal program, 21 LiF TLDs of the type used in the personnel dosimetry program (see Section 1.4.3) at various locations immediately outside of the plant, most of which were on or near the plant security fence (Ref. 66).

1.7.2 Radiological Environmental Monitoring Program

The radiological environmental monitoring program, which includes the above-mentioned sampling and monitoring equipment (air and water samplers; TLDs), addresses the following environmental media:

<u>Medium</u>	<u>Number of Locations Routinely Monitored (Ref. 180)</u>	<u>Number of Locations Required by Tech Specs</u>
Air iodines	4 ind.*; 2 bkg.	4 ind.; 1 bkg.
Air particulates	5 ind.; 3 bkg.	5 ind.; 3 bkg.
Immersion dose (TLD)	15 ind.; 5 bkg.	15 ind.; 5 bkg.
Milk	5 ind; 1 bkg.	5 ind.; 1 bkg.
Surface & drinking water	5 ind.; 2 bkg.	5 ind.; 2 bkg.
Other**	-	-

*Ind. - indicator (near-field locations)
 Bkg. - background (control or far-field) locations

**The program also includes samples of fish, aquatic plants, sediment, green leafy vegetables, and precipitation. These sample types were not listed above because they were either unimportant as far as the March 28 incident was concerned or represented only an indirect pathway to man.

TABLE II-1-4

THERMOLUMINESCENT DOSIMETER LOCATIONS

<u>Licensee Designation</u>	<u>Location</u>	<u>Distances and Direction*</u>
1S2**	North Weather Station	0.4 mi N
2S2	North Bridge	0.7 mi NNE
4S2**	Top of Dike	0.3 mi ENE
5S2**	Top of Dike	0.2 mi E
9S2	South TMI	0.4 mi S
11S1**	Mech. Draft Cooling Tower	0.1 mi SW
14S2	Shelley Island***	0.4 mi WNW
16S1**	North Boat Dock	0.2 mi NNW
4A1	Laurel Road	0.5 mi ENE
5A1**	Ob. Center Bldg.****	0.4 mi E
16A1	Kohr Island***	0.4 mi NNW
10B1	Shelley Island***	1.1 mi SSW
12B1	Goldsboro Air Station	1.6 mi WSW
1C1	Middletown Substation	2.6 mi N
8C1**	Fallmouth Substation	2.3 mi SSE
7F1**	Drager Farm****	9 mi SE
4C1**	Rt. 241****	10 mi ENE
7G1	Columbia Water Plant	15 mi SE
9G1	York Med Ed Station	13 mi S
15G1**	West Fairview Substation****	15 mi NW

*Relative to a point midway between the two containment buildings.

**Location also has RMC TLD for quality control purposes.

***Island locations contained two Teledyne TLDs on 3/28/79.

****Location also has a dosimeter which is readout on a monthly basis.

Analyses of the above sample types are performed by Teledyne Isotopes on a frequency which varies from weekly (air iodine and particulates; water) to semi-monthly (milk during the grazing season; monthly otherwise) to quarterly (TLD). A representative of Porter-Gertz Consultants stated in discussions that, in addition to the routine analyses performed by Teledyne Isotopes, additional samples (4 air, 10 TLD, 1 milk, and 3 water) from the same locations have been analyzed by the Radiation Management Corporation as part of a quality assurance program.

Based on a selective review of this area by the investigator, it appears that the above-described program was in effect at the time of the March 28 incident.

1.7.3 Meteorological Monitoring Program

The Three Mile Island meteorological tower is located 2200 feet north of the Unit 2 reactor building. Monitoring instrumentation on the tower consists of redundant Teledyne/Geotech wind speed and direction indicators at the 100 ft. level, Rosemount temperature sensors at the 33 and 150 ft. levels. A temperature difference is measured between the 33 and the 150 ft. level (Ref. 67, Int. 60).

A calibration of the instrumentation on the tower was performed on October 26, 1978, in accordance with Technical Specification 4.3.3.4. The calibration consisted of the following (Int. 60, Ref. 68):

- a. A visual inspection of the wind vane and a test of its azimuth indication accuracy. The Crawford Plant west stack (a coal-fired plant upstream of TMI), having an azimuth of 336° , was used as a point of reference. Alignment was checked at 0° , 90° , 180° , and 270° .

The wind vane bearings were also checked to ensure that the starting torque was acceptable. One sensor was noted to have a torque of 2.0 gm-cm relative to a maximum acceptable value of 1.4. It was adjusted to 1.4 gm-cm.

- b. Anemometer cups previously wind tunnel tested and certified (Int. 60), were given a visual inspection for damage and none was apparently noted. A known torque was applied to the anemometer to ensure that the starting torque of the bearings corresponded to a wind speed of less than 1 mph. One of the anemometers was noted to have a starting torque of 1.5 gm-cm, relative to a maximum acceptable value of 0.74 gm-cm. It was adjusted to 0.14 gm-cm.
- c. Temperature sensors were placed in a 32.0°F ice bath and the response noted. Other temperatures, from -40°F to 100°F, were electronically simulated relative to the ice bath response. No problems were noted.
- d. The remainder of the system, up to and including the control room recorder, was electronically calibrated. No significant problems were noted.

Prior to the event, the meteorological tower had been operating reliably, providing better than 98% data recovery (Int. 60). The tower has continued to operate equally well post-event with no significant problems noted (Int. 60).

2. INITIAL EMERGENCY RESPONSE

2.1 DETECTION AND CLASSIFICATION

The Emergency Director was responsible for classifying a situation as an emergency in accordance with the conditions in Table 1 of the Emergency Plan and for initiating actions according to the Emergency Plan Implementing Procedures and his own best judgment. The Emergency Plan requires that the actions listed in Table 1 be considered for each type of emergency, but that these or other actions only be taken if they are appropriate.

The first conditions in Table 1 that appeared to have been met were those specified in Criterion (c) for a Site Emergency. These conditions were met by 0415 on the morning of March 28. This Site Emergency action level states: "Loss of primary coolant pressure coincident with high reactor building pressure and/or high reactor building sump level."

By 0415, reactor coolant system pressure had dropped from 2435 psig at the time of reactor trip (0400) to approximately 1275 psig (Figure II-2-1). This pressure (1275 psig) was below the reactor coolant low pressure trip setpoint (1940 psig) and the setpoint for emergency core cooling system initiation (1600 psig). At 0415, a pressure rise of about 1.4 psig inside the reactor building was detected. The duty shift supervisor was aware of the drop in reactor coolant system pressure and increase in the reactor building pressure. Initially, he evaluated these conditions in relation to the Emergency Plan and determined that they were not indicative of an emergency since the primary coolant system pressure had stabilized and there were no increased radiation levels either in or being released from the facility (Int. 145).

Information regarding the reactor building sump level is provided to the control room via printout on the alarm printer or by phone or page from an individual at the radwaste panel in the auxiliary building. Reactimeter data for March 28 indicated that a high reactor building sump level was reached at

**REACTOR COOLANT PRESSURE VS.
TIME ON MARCH 28, 1979,
AS DISPLAYED ON THE WIDE
RANGE PRESSURE RECORDER**

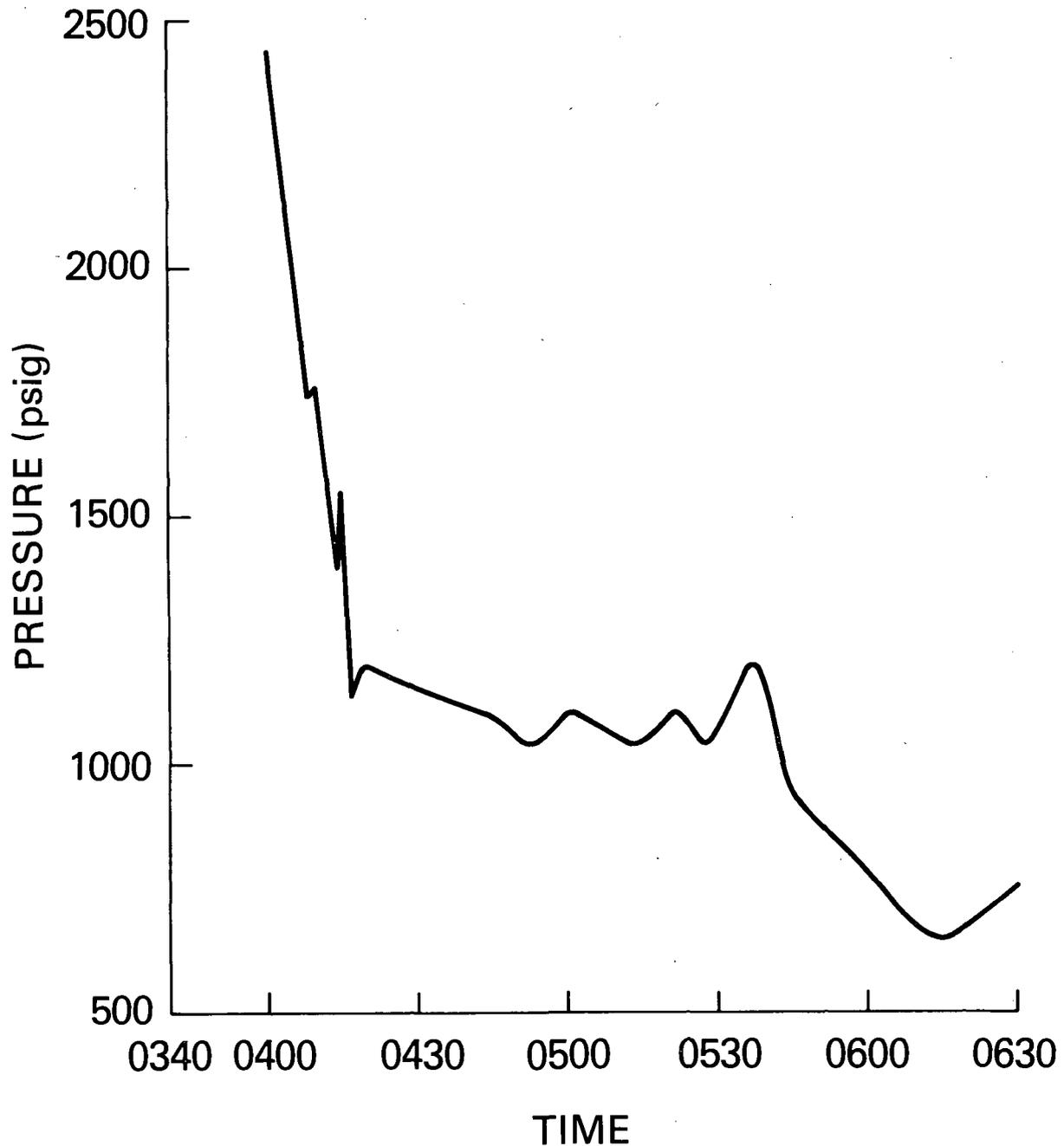


Figure II-2-1

0411. At this time, the alarm printer was printing numerous alarms and plant parameters. The high reactor building sump alarm printout went unnoticed until approximately 0430 when an auxiliary operator, located in the auxiliary building near the radwaste panel, noted that both reactor building sump pumps were running and that there was a high reactor building sump alarm on the radwaste panel. Upon leaving the auxiliary building, the individual also noted an alarm (500 cpm) of a count rate meter (RM-14/HP-210) located near the model room door. These conditions were reported to an operator in the control room. The operator directed the auxiliary operator to survey the area. This survey was performed using an E-520 instrument and indicated radiation levels of less than 0.1 mR/hr. Based on the survey results and the location (normally a radiation area), the significance of the RM-14/HP-210 alarm was discounted. The operator did verify, via the alarm printer in the control room, that the reactor building sump level was "high." This was reported to the shift supervisor who became concerned about available capacity in the liquid radwaste system and directed that the sump pumps be turned off (Int. 15).

Since the drop in primary system pressure had been stabilized by this time and there were still no alarms on the radiation monitors in the control room, the shift supervisor did not interpret the high reactor building sump level and the earlier noted conditions of decreased primary system pressure and increased reactor building pressure to meet the conditions for a Site Emergency (Int. 145). The investigator determined that the lack of specificity regarding the definition of the terms "loss of primary system pressure" and "high reactor building pressure" appeared to contribute to the failure to declare the Site Emergency by 0430.

At 0518, the reactor building air particulate monitor (HP-R-227(P)) passed through the alarm setpoint (50,000 cpm)(Ref. 102). The shift supervisor did not recall having received this alarm (Int. 145). Two intermediate cooling water let-down cooler radiation monitors (IC-R-1091 and IC-R-1092) alarmed at 0500 and 0518, respectively. The detectors for these monitors were located near the reactor building sump and were intended to monitor for leakage from the primary to secondary side of the letdown cooler. The alert setpoints were 1,000 cpm and the alarm setpoints were 5,000 cpm. The alarm setpoints were designed to be 10 times background

to provide positive identification of leakage of the letdown cooler (Ref. 3). Upon receipt of the intermediate cooling water letdown cooler radiation monitor alarms, operators verified that the intermediate closed cooling water surge tank level had not increased, indicating that there was no leakage from the primary to the secondary side of the letdown cooler. This is a specified action for alarm of these monitors.

Since the intermediate cooling water letdown cooler monitor and reactor building air sample particulate monitor alarms occurred in the same building, it appears that criterion (b) of Table 1 of the Emergency Plan, "Local Emergency," was met. Condition (b) states, "More than one radiation monitor in a single building reaches the alarm setpoint."

The shift supervisor evaluated these alarms (intermediate letdown cooling water letdown cooler), but did not interpret them to be applicable for serving as a basis for the classification and declaration of a Local Emergency (Int. 145). Based on their training and the wording of the action levels it was the shift supervisor's and operating crew's understanding that the action levels in Table 1 of the Emergency Plan relative to "radiation monitors" referred only to area radiation monitors. Since the intermediate cooling water letdown cooler monitors and reactor building air monitor were process monitors, their alarms were not thought to meet specified action levels. In addition, prior to this incident, the intermediate cooling water letdown cooler radiation monitors were often in alert due to background radiation. The significance of their alarm on March 28 was further discounted by the shift supervisor, based on his assumption that the alarms resulted from a "crud burst" through the letdown cooler. An investigator noted that Procedure 1670.1, Criterion 3.1 contained the same wording as Table 1, Criterion (b) of the Emergency Plan and did not specify whether the criterion pertained to area monitors, process monitors, or both.

The investigator did note, however, that relative to a Site Emergency, Criterion 3.6 of Procedure 1670.2, "Site Emergency Procedure," specifies area monitors, whereas condition (f) of Table 1 in the Emergency Plan, relative to

a Site Emergency, does not. An interview of the operating crew indicated that their training had not distinguished or made clear the applicability of area or process radiation monitors in relation to the action levels of the Emergency Plan.

The investigators have determined that the alarms of the intermediate cooling water letdown cooler radiation monitors were a result of increased radiation levels in the area of the detectors, i.e., the reactor building sump.

Review of strip chart records HP-UR-1901 indicates that alarm of the incore instrument panel area monitor (HP-R-213) should have occurred at about 0625. If the alarm occurred, it apparently went unnoticed.

Review of strip chart records HP-UR-1901 indicates that alert of the containment dome monitor (HP-R-214) should have occurred at about 0635. Alert of this monitor and its correlation with Site Emergency criterion (e) apparently went unnoticed. Criterion (e) of Table 1 states, "Reactor building high range gamma monitor alert alarm."

At approximately 0655, there were many indications of a sudden increase in in-plant radiation levels. Radiation/chemistry technicians located in the area of the Unit 1 primary sample station heard local alarms of area radiation monitors and noted increased readings on portable radiation survey instruments, while Unit 2 control room personnel noted remote alarms of various area radiation monitors on the area radiation monitor panel almost simultaneously. Radiation/chemistry technicians determined the source of these increased levels to be the reactor coolant letdown sample lines (Int. 20, 42, 145).

A Site Emergency was declared at 0655 on March 28 by the shift supervisor in coordination with the Unit 2 Superintendent, Technical Support based on the alarm of numerous area radiation monitors in the Unit 2 control room (Int. 145). This action level was consistent with the Emergency Plan, Table 1, criterion (f),

for a Site Emergency and with paragraph 3.6 of Station Radiation Emergency Procedure 1670.2, "Site Emergency Procedure", Revision 8, dated February 13, 1978. The 0655 time is based on the stated sequence that restart of the 2B reactor coolant pump (RCP) preceeded the sudden increase in radiation levels. Alarm printer data shows that the 2B RCP was restarted at 0654.

For the next 30 minutes, radiation levels on many of the area radiation monitors, including the containment dome monitor (HP-R-214), steadily increased.

Upon declaration of the Site Emergency, a calculation of projected offsite radiation exposure rate was performed. This calculation was completed by 0710 and indicated a whole body exposure rate of 40 R/hr in Goldsboro (Ref. 70, Int. 48). Apparently, no attempt was made to calculate a projected dose by factoring in the anticipated duration of the condition that precipitated the number. The initial projection of offsite exposure rate was believed to be overly conservative since pressure in the reactor building was only 1.4 psig, far below the 56 psig pressure on which the calculational model was based. At the time, there were no other data available to use in evaluating the validity of this calculation. Consequently, it would appear that the initial projected exposure rate should not have been discounted so readily. As a minimum it should have been evaluated in light of the general emergency criterion (d) which states, "Offsite projected doses downwind from the site boundary are greater than 25 Rem thyroid and/or 5 Rem whole body." (See Section 3.3.1 for a detailed discussion concerning this calculation and evaluation.)

At 0724, the Station Manager declared a General Emergency based upon a containment dome monitor reading of greater than 8 R/hr. This action was found to be consistent with paragraph 3.1 of Station Radiation Emergency Procedure 1670.3, "General Emergency Procedure", Revision 6, dated February 13, 1978.

The emergency classifications, respective criteria (action levels) and the times when the criteria were apparently met are shown in Table II-2-1.

TABLE II-2-1
EMERGENCY CONDITIONS AND TIMES MET ON MARCH 28

<u>Emergency Classification</u>	<u>Conditions</u>	<u>Time Met*</u>	<u>Notes</u>
Personnel Emergency or Local Emergency	a) Personnel injuries may involve contamination or excessive radiation exposures.	-	
	b) More than 1 radiation monitor in a single building reaches their alarm setpoint.	0520 0625	IC-R-1091 IC-R-1092 HP-R-227(P)
	c) Report of an unexpected increase in the level of radiation or airborne activity in a work area.	0655	Radiation/ Chemistry Technicians
	d) Report of a radioactive spill in a work area.	-	
	e) Flooding or localized fire, that could affect a release of radioactivity.	-	
Site Emergency	a) A unit vent gas monitor indicated 100 times the instantaneous release limit specified in the technical specifications.	0730-0740	HP-R-219 (P, I, and G)
	b) The radiation level at the station security fence is 125 mR/hr.	1720	140 mR/hr at GE-10
	c) Loss of primary coolant pressure, coincident with high reactor building pressure and/or high reactor building sump level.	0430	RC pressure 1275 psig RB pressure 1.4 psig
	d) Reactor building evacuation alarm from the source range instrumentation (Unit 1), manual initiation (Unit 2).	-	Hi RB sump
	e) Reactor building high range gamma monitor alert alarm.	0635	Unnoticed
	f) The high alarm of the radiation monitoring system in two separate buildings due to a single event.	0655	Site Emergency Declared
General Emergency	A General Emergency will be declared when a Site Emergency has been declared and one or more of the following conditions exists:		
	a) Reactor building high range gamma monitor high alarm.	0720	General Emergency declared
	b) The radiation level at the station boundary is >125 mR/hr.	1720	140 mR/hr, GE-10
	c) The liquid effluent radiation monitor indicates. >6.8 E-3 μCi/cc.	-	
	d) Offsite projected doses downwind from the site boundary are >25 rem thyroid and/or 5 rem whole body.	0710	10-40 R/hr (dose rate)

*Times based on an evaluation of information obtained from interviews, strip chart recorders, logs and reactimeter. Times reported were selected based on the relative reliability of the various sources and reconciliation with other activities or events occurring at the same approximate times.

2.2 ORGANIZATION ACTIVATION

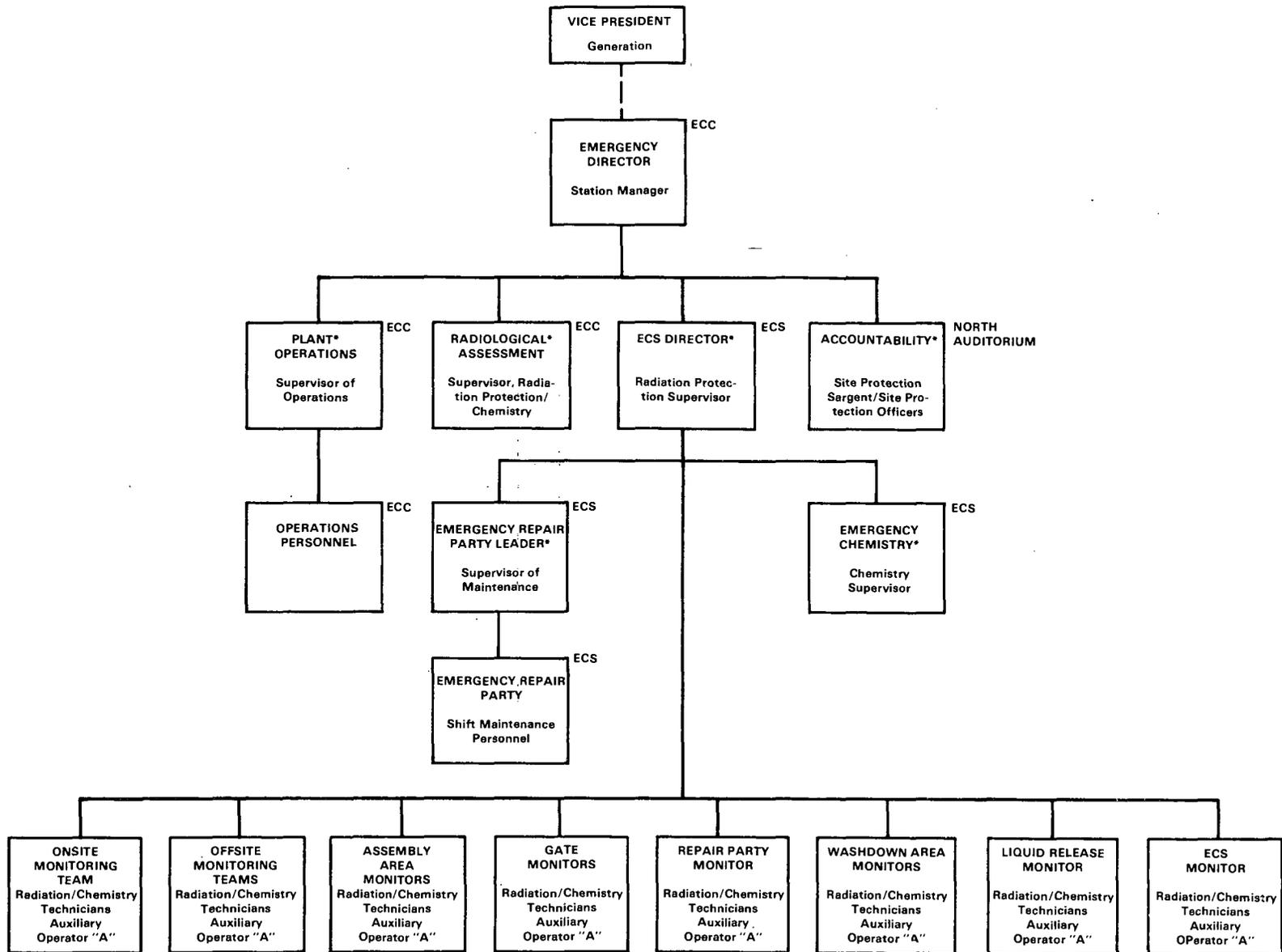
Shortly after the reactor tripped, workers on site initially phoned selected operations, chemistry, and radiation protection supervisors at home to notify them of the trip and request their assistance (Int. 20, 27, 120, 145). During the following 2 hours, as onsite personnel sought solutions to operational problems, other supervisors were alerted at home. Consequently, at 0655, when several area radiation monitors showed sharp increases, many key TMI employees were either on site or enroute. In addition, the remainder of the licensee's day shift employees were arriving for the beginning of their shift.

On declaration of the Site Emergency at 0655, TMI's response organization pursued actions along five parallel paths: (1) continued assessment of plant operating conditions, (2) assessment of actual and projected radiological conditions in the plant and in the environment, (3) notification of and communication with offsite support agencies and individuals, (4) accountability of TMI personnel, (5) and consideration of corrective actions to mitigate or terminate the emergency.

Initially, the emergency response organization approximated the "planned" organization established in the Emergency Plan and procedures which is shown in Figure II-2-2. Slight deviations from the "planned" organization occurred, and there was some confusion regarding who was performing and in charge of directing certain emergency actions. Initially, the Supervisor, Radiation Protection and Chemistry assigned Radiation/Chemistry Technician I to assume command of the Emergency Control Station until relieved by a radiation protection foreman or the Radiation Protection Supervisor (Int. 79). The emergency responsibilities of the individual in charge of the ECS are specified in Paragraph 4.2.7 of Procedure 1670.2 (Ref. 71).

At about 0700, Radiation Protection Foreman B arrived at the ECS and Radiation/Chemistry Technician I assumed that Radiation Protection Foreman B had taken control of the ECS. Radiation/Chemistry Technician I then began to function as a Radiation Monitoring Team member (Int. 42). According to Radiation

"NORMAL" EMERGENCY ORGANIZATION



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

Figure II-2-2

Protection Foreman B, during the time frame of about 0700 to 0800, he primarily performed in-plant (Radiation Monitoring Team) radiation surveys (Int. 21).

At approximately 0705, Radiation Protection Foreman D arrived at the ECS shortly after Radiation Protection Foreman B and, for a short time, assisted Radiation Protection Foreman B in getting onsite and offsite survey teams underway to the processing center. He then left the ECS and went to the Unit 1 control room (Int. 18).

Radiation Protection Foreman A stated that he arrived in the ECS at about 0715 and, observing that no one appeared to be in charge, he took charge. (Int. 7). At about 0735, the Radiation Protection Supervisor arrived and assumed control of the ECS (Int. 22).

According to the Emergency Plan and Emergency Plan Implementing Procedures, a single Repair Party would form at the Emergency Control Station (ECS), functioning under the Supervisor of Maintenance or a maintenance supervisory foreman who would also be located at the ECS. The general activities of the Repair Party would be coordinated by the Emergency Director, located in the Emergency Control Center (ECC), through the Radiation Protection Supervisor or his alternate, who would be in charge of the ECS. This method of coordination and control was designed to ensure that radiological conditions and controls were noted and observed prior to, during, and after each emergency repair job. It also established a single point of command and control of Repair Party activities, avoiding confusing, counterproductive, duplicative, or "unauthorized" efforts or requests.

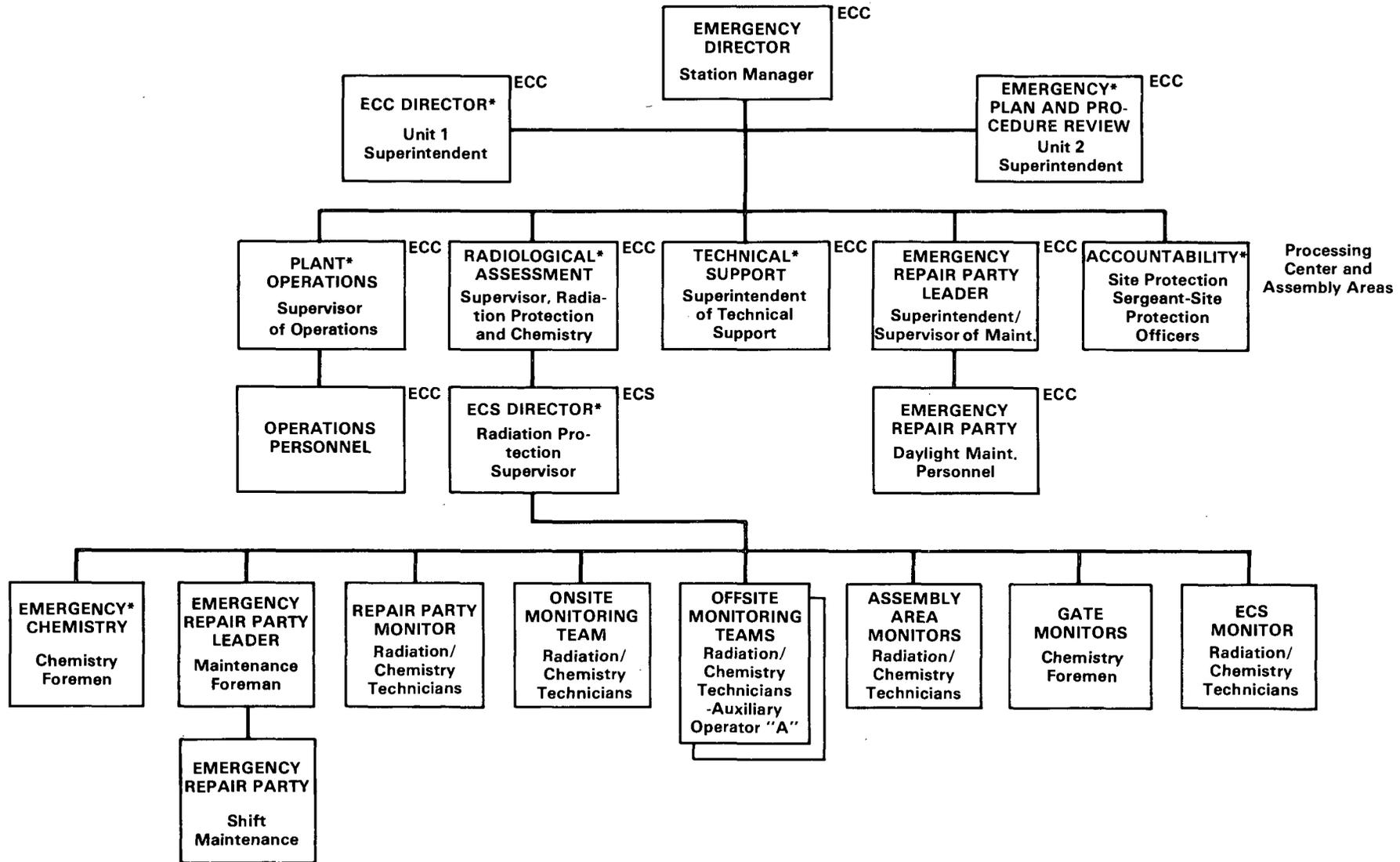
Procedures governing the assembly, direction, and functions of the Repair Party Team are covered in Paragraph 4.4 of Procedure 1670.2 and Procedure 1670.8.

After the Site Emergency was declared at 0655, a Repair Party composed of six maintenance shift workers was formed at the ECS under the control of Maintenance Foreman B (Int. 187). A second Repair Party, composed primarily of daylight instrument and control personnel, was formed in the Unit 2 control room under the direction of the Superintendent of Maintenance and the Unit 2 Supervisor of Maintenance. The Superintendent of Maintenance and Unit 2 Supervisor of Maintenance were aware that the assigned location for either of them during an emergency was the ECS where they would act as the Repair Party leader. However, to ensure prompt availability of their expertise, and since a Repair Party was already formed at the ECS under the direction of a maintenance foreman, they decided to remain in the control room (Int. 120). This decision was reinforced by the Station Manager. On assuming the position of Emergency Director, he announced that the Superintendent of Maintenance would be the one in the control room to be in charge of emergency repair functions (Ref. 72, Int. 120). Some time later, the Superintendent of Maintenance directed the Repair Party at the ECS to move to the Unit 2 control room. By 0800, all Repair Party personnel were assembled in the Unit 2 control room, separated from the ECS.

The emergency duties of the chemistry supervisor are outlined in Paragraph 4.2.8 of Procedure 1670.2. Since the TMI position of chemistry supervisor was vacant, Chemistry Foremen, A and C, filled this position in the emergency organization (Int. 79).

The licensee operated generally with the organizational configuration shown in Figure II-2-3 from approximately 0730 until 0900 when the ECS was evacuated to the Unit 2 control room. Since the Unit 2 shift supervisor's office, the planned alternate location of the ECS, already had a number of people in it, the Radiation Protection Supervisor established the ECS in the control room proper. He was careful to establish the location of the ECS activities in an area where they would not interfere with plant operations (Int. 172). The configuration of the emergency organization upon relocation of the Unit 2 control room is shown in Figure II-2-4.

EMERGENCY ORGANIZATION IN EFFECT 3/28/79, 0730 - 0900



ECC—Emergency Control Center, Unit 2 Control Room
 ECS—Emergency Control Station, Unit 1 Chemistry/Health Physics Lab Area
 *—Functional Title Added

Figure II-2-3

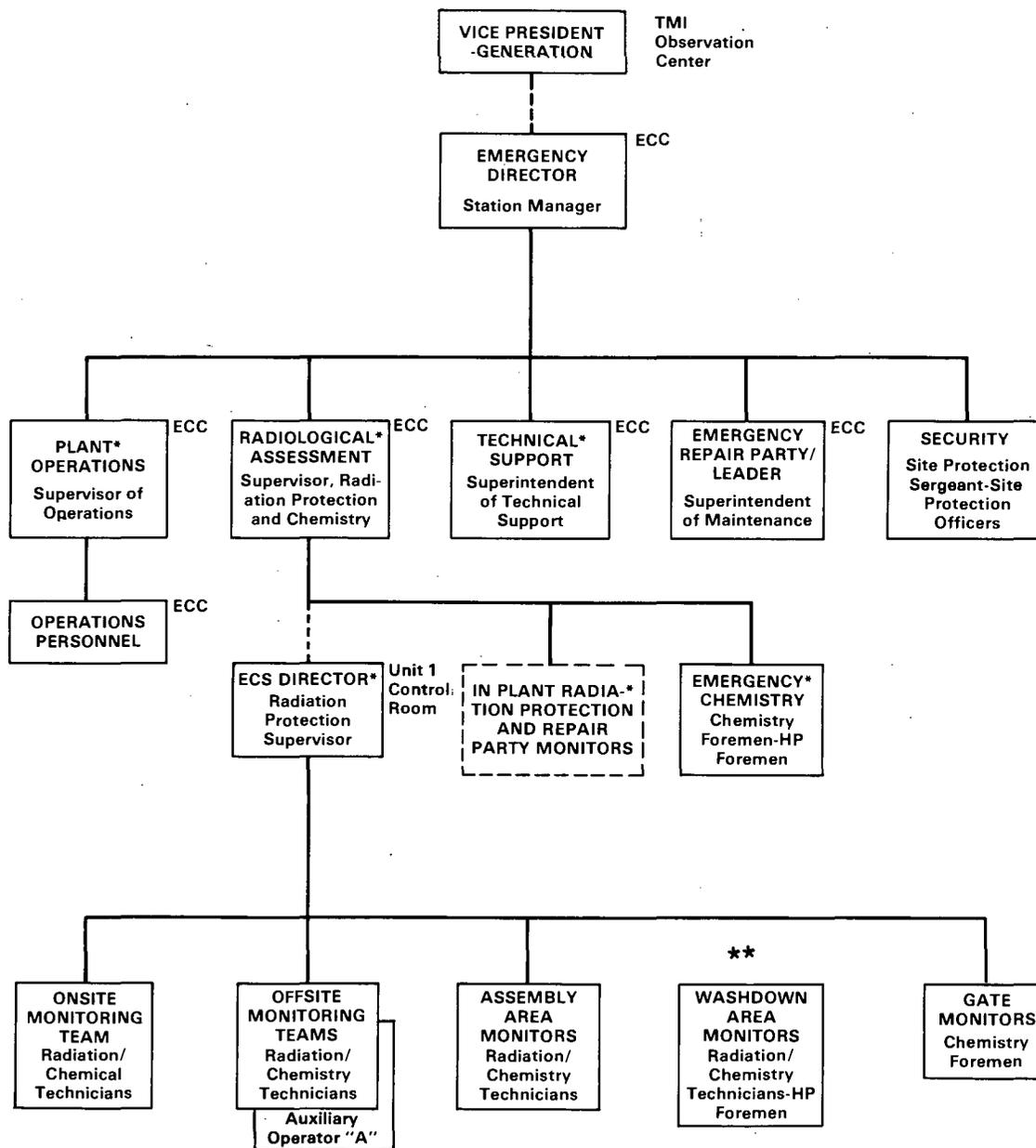
Shortly after relocation of the ECS to the Unit 2 control room, the Radiation Protection Supervisor and the Supervisor, Radiation Protection and Chemistry met to discuss the direction in which the organization should proceed (Int. 79, 172). The Supervisor, Radiation Protection and Chemistry determined that the Radiation Protection Supervisor had onsite and offsite monitoring activities fully implemented and under control. To better divide the workload, the Supervisor, Radiation Protection and Chemistry decided to assume direct control of all in-plant radiation protection functions. He determined that this was necessary because the Emergency Plan and procedures did not contain adequate provisions to deal with radiological controls under the type of conditions that existed in the plant. During an interview, the Supervisor, Radiation Protection and Chemistry stated, "The procedures themselves do not address in any great detail in-plant radiation protection support...or coordination of maintenance and operations...I can also say quite honestly, in my own mind, I never conceived of the type of maintenance and operations activities that would have to go on in areas of such tremendous radiation dose rate levels as we had" (Int. 79).

The ECS, which had limited its activities to directing onsite and offsite monitoring, began a relocation to the Unit 1 control room at about 1030, and shortly before 1100 was fully operational at its new location. Figure II-2-5 shows the organizational configuration from that point in time (Int. 20, 22, 25, 47, 79, 124).

Two chemistry foremen assumed the function of gate monitors shortly before evacuation of the island. The normal individuals designated to perform this function are radiation/chemistry technicians or auxiliary A operators. Neither chemistry foremen had received emergency plan training in this duty.

At about 1130, a washdown (decontamination) area was established at the 500 kV substation on the direction of Radiation Protection Foreman D to serve as a monitoring and decontamination point for individuals evacuating the island (Int. 18, 45, 51). The preplanned locations for the washdown areas were the north washdown area, 2 miles north of the north gate on route 441 at

EMERGENCY ORGANIZATION IN EFFECT 3/28/79, 1100-2000



ECC—Emergency Control Center
 *—Functional Title Added
 ** Line intentionally omitted

Figure II-2-5

the Kiwanis Field adjacent to the Swatara Creek, and the south washdown area, 3 miles south on route 441 at the Collins Substation. Procedures governing the selection, operation and manning of washdown areas are specified in Paragraph 2.3 of Procedure 1670.6. Prior to the accident, washdown areas were intended to be used for vehicle decontamination; but on March 28 the designated washdown area (500 kV substation) also served to decontaminate individuals. The washdown area was staffed by radiation/chemistry technicians, two of whom (Radiation/Chemistry Technicians S and X) were relatively new employees and had never been trained in Radiation Monitoring Team functions.

From 0705 to approximately 2000, the Station Manager (Emergency Director) maintained ultimate control and direction of site activities in coordination with the Vice President, Generation, who was located at the Observation Center by about 1145. At approximately 2000 on March 28, the nature of the relationship between the Station Manager and Vice President, Generation changed from one appropriate to the emergency to one more in line with the management interface that would be expected during a recovery operation. During an interview, the Station Manager stated that, in his mind they (the site) "had entered a recovery phase, since, for the first time, the plant appeared stable and that there was necessary flow and cooling to the core" (Int. 71). In prepared testimony, the Station Manager stated that "By 8:00 p.m., I was still in charge of the control room but I was essentially under the direction of Senior Management..." (Ref. 72).

An investigator reviewed this statement in light of the radiological and plant operational conditions existing at the time and compared them with the following policy set forth in Procedure 1670.15, "Post Accident Re-entry and Recovery Plan":

"...the immediate phases of responsive action include initial actions directed toward protection of the public and immediate and planned action directed toward termination of the incident, containment of effluents, establishment of incident boundaries, establishment of control and channeling of information, and protection of the facility and equipment.

After these actions have been effectively accomplished, emergency actions will be directed toward restoring the facility to its normal operating condition."

By about 2000, the operational elements of the emergency organization had generally "stabilized" the reactor to the point where their actions were generally not of an emergency response nature. The radiological elements of the emergency organization were, however, still responding to emergency conditions, attempting to reestablish routine radiological controls in plant (protective actions), continue in-plant, onsite, and offsite radiological monitoring (assessment actions), and implement corrective actions to terminate or reduce actual or potential releases of radioactive materials.

An example of the increasing influence being exerted by groups outside the licensee's emergency organization was evident in the taking of a reactor coolant sample on March 29 at about 1630. During an interview, the Supervisor, Radiation Protection and Chemistry indicated that there was concern regarding the boron concentration of the primary system and he was involved in a great deal of discussion concerning the necessity of taking a primary coolant sample. Regarding the source of the "order" to secure the sample, the Supervisor, Radiation Protection and Chemistry stated, "There were many individuals, and by many individuals it's kind of hard to put names or even companies. I think they spanned B&W, the NRC, Met Ed, and GPU. There was probably somebody from every organization." The Supervisor, Radiation Protection and Chemistry indicated that he felt the samples were necessary, but not critical, because of the concern that the core was still critical (Int. 79). The taking of this sample resulted in the overexposure of two individuals, neither of whom were informed that the operation would be performed on a voluntary basis. Examples of activities that were performed without adequate radiation protection support are described in Section 3.2.

The deviations and shortcomings listed above, i.e., separation of the Repair Party Team from its usual location under the Radiation Protection Supervisor; the lack of a clearly established element in the emergency organization responsible for in-plant radiation protection during emergencies; realignment of the emergency duties of the Radiation Protection Supervisor and Supervisor, Radiation Protection and Chemistry; and an apparent premature shift to a recovery mode appeared to contribute to a breakdown in radiation protection and Repair Party controls. This breakdown was evidenced by the statements, actions, and radiation exposures of emergency personnel. (See Section 3.2 for a detailed account of these statements, actions, and exposures.)

During an emergency, the site protection force is initially responsible for accountability of personnel on site. Once accountability is complete, they are to continue to maintain security in accordance with established security procedures (Ref. 71). The investigators reviewed applicable procedures and interviewed members of the licensee's site protection force to determine the manner in which the security force performed these responsibilities. (Int. 32, 52, 72)

Upon declaration of the Site Emergency, site protection officers closed the north gate access point to the owner-controlled area, and initiated accountability (Int. 32, 52, 72). Entry to the site was granted only on clearance received from the ECC. After declaration of the Site Emergency, personnel who were permitted entry to the site, with the exception of non-Metropolitan Edison personnel i.e., NRC and contractors, were not logged in from this point in time. Accountability was completed by 0830, 1 hour and 35 minutes after declaration of a Site Emergency (Int. 72).

The site protection force maintained their security posts until approximately 0900 when radiation protection personnel directed evacuation from certain areas inside the plant itself. At approximately 1015, the entire site

protection force shut down the security console, thereby defeating security controls for vital areas within the plant, and evacuated to the north and south gates of the owner-controlled area. Security checkpoints were established at both gates, with only the north gate used for access (Int. 72).

Withdrawal from the owner-controlled area to the north and south gate access control points was made without taking normal security compensatory measures to these new checkpoints. A five-man, armed force was, however, on standby at the Observation Center for deployment if needed. The nature of security operations during the emergency, was described by a site protection sergeant who stated, "We are directly under the direction of the Emergency Director. Had he, in fact, felt that it was required at the time to maintain the security posture we would have done so." The Supervisor of Generation Security stated, "We've operated on the premise that safety overrides security and that's what we were doing at this point" (Int. 72).

The investigators noted that the licensee's emergency planning provisions did not contain contingency security procedures for implementation under emergency conditions where the normal security procedures may be inappropriate. In regard to the planning for contingency security measures during emergencies, the Supervisor of Generation Security stated, "The only involvement that security had in any radiation emergency was strictly accountability. The planning it went into, contingency as you described, did not occur, to my knowledge, in the last five years," (Int. 72).

Consequently, when the normal security measures were abandoned to facilitate the response to the emergency and ensure the safety of the site protection force, contingency measures were developed on the spot and modified as the need arose.

During the period March 28 through the afternoon on March 30, personnel and deliveries of emergency supplies and equipment were processed through the owner-controlled access point at the north gate. Clearance for each entry was

obtained from the ECC, ECS, or Command Post at the Observation Center. Search requirements were dispensed with during this period, and once individuals had shown some form of identification and were cleared through the north gate they had unrestricted access to all areas on the site, to include vital areas and areas having radiation and/or contamination at significant levels. In addition to the suspension of search requirements, licensee employees were not logged on or off site, making accountability in the event of further problems impossible. During later periods, on March 30, a means of accountability based on dosimeter issuance was instituted (Int. 72).

Approximately two days after the event, on March 30, a roving patrol was instituted outside the protected area barrier. By March 31, four days after the incident, one site protection officer was stationed in the processing center. It was approximately nine days after the incident before the site protection force fully reentered the processing center and began reconstituting a return to more normal operations (Int. 72).

The investigators determined that the abandonment of the normal security measures was appropriate under the circumstances existing at the time. However, the failure to have contingency security measures available for implementation during such conditions appeared to have the potential for jeopardizing the safety of individuals for several reasons. The failure to maintain records of entries and exits from the site would have made it impossible to ensure that all individuals could be accounted for in the event of an escalation of events on site requiring the evacuation of nonessential personnel. The Supervisor of Generation Security further confirmed that the security force could not have performed accountability because, "...our personnel... had not been respirator qualified...[and] would not have been able to go into various places in the plant to get muster sheets,"(Int. 72). In addition, the lack of positive security controls for certain vital areas left these areas open to possible sabotage or unauthorized manipulation of vital equipment. The potential for the occurrence of either such act could have seriously jeopardized the public

health and safety through the release of radioactive materials resulting from equipment damage or operation. This observation was of particular significance in light of the plant and core condition at the time.

The investigators noted that there were no regulatory requirements relative to minimum security measures during emergency situations wherein the normal methods and level of safeguards must be abandoned for reasons of safety.

2.3 NOTIFICATIONS AND COMMUNICATIONS

2.3.1 Notifications

Initial notifications that a Site Emergency had been declared were begun at 0702. By 0720, the licensee had made all applicable notification calls required by Station Radiation Emergency Procedure 1670.2, Figure 4. The results of these notifications were as follows (Ref. 73-78, Int. 70):

<u>Time</u>	<u>Agency/Individual</u>	<u>Persons Contacted</u>
0702	PEMA	Duty Officer
0704	NRC Region I	Answering Service
0705	Vice President Generation	*
0709	Dauphin County Civil Defense	Duty Officer
0709	Dept. of Energy-Radiological Assistance Program	Duty Officer
0709	Manager, Generation Operations	Manager, Generation Engineering
0713	Radiation Management Corp.	No answer
0718	Pennsylvania State Police	Dispatcher
0720	American Nuclear Insurers	No answer

*The Vice President, Generation was contacted earlier in the morning and participated in a conference call concerning the incident. The individuals making the initial notifications were not aware that the Vice President Generation had been contacted and could not be reached at the telephone number listed in the procedures (Int. 91).

On declaration of a General Emergency, each of the aforelisted agencies were recontacted to inform them of escalation of the emergency to the general category. The results of these notifications were as follows:

<u>Time</u>	<u>Agency/Individual</u>	<u>Persons Contacted</u>
0730	Dauphin County Civil Defense	Duty Officer
0730	Manager, Generation Operations	Manager, Generation Engineering
0735	PEMA	Duty Officer
0735	DOE-RAP	Duty Officer
0735	American Nuclear Insurers	No answer
0740	NRC: Region I	Answering Service
0740	Pennsylvania State Police	Dispatcher
0740	Vice President, Generation	*
0740	Radiation Management Corporation	Philadelphia Electric Company Load Dispatcher

2.3.2 Communications

Subsequent to the respective notifications, open telephone lines between the control room and the Pennsylvania Bureau of Radiological Health and control room and the NRC Region I Incident Response Center were established at 0740 and 0800, respectively.

An interview of individuals of the Pennsylvania Bureau of Radiological Health and discussions with the Director, Dauphin County Civil Defense and Director, Pennsylvania Emergency Management Agency (PEMA) indicated that the licensee's initial notifications were in accordance with existing agreements. The continued flow of updated information to the Pennsylvania Bureau of Radiological Health via the open telephone lines was characterized as good (Int. 46).

On the morning of March 30, at 0710, Shift Supervisor C directed Control Room Operator C to open MU-V-13 and vent the MUT. Shift Supervisor C did not believe this operation would result in a significant release to the environment. He did, however, believe that a small release was possible and contacted the ECS to suggest that a helicopter be dispatched if it was not already (Int. 173).

Venting of the MUT had been in progress for about 50 minutes when Shift Supervisor C became aware of a reading of 1200 mR/hr at an elevation of 130 feet above the Unit 2 auxiliary building. About this same time, the Emergency Director (Station Manager) and NRC Inspectors B and D arrived in the control room. Inspectors B and D were told by a control room operator that an uncontrolled release had occurred during transfer of gases from the MUT to the WGDTs. The reason for the release was described to be the lifting of relief valves due to overpressurization of the vent header and/or WGDTs. This information was, in turn relayed to NRC Headquarters at 0845 (Ref. 79). The Emergency Director instructed Shift Supervisor C to contact the ECS and have them inform PEMA and BRH.

Interviews of BRH personnel and a discussion with the Director, PEMA indicated that two simultaneous but conflicting phone calls were received from the site concerning this release. One call, from the Unit 2 Supervisor of Operations, was in an "excited" tone and recommended evacuation. The second caller, who was not identified, presented details of the release in a calmer tone and did not recommend protective action. The contradictory nature of the two calls did not create confusion. The BRH was in contact with the facility via the open telephone line established two days earlier. They were provided with current offsite dose rates, meteorological conditions and plant conditions which indicated that radiation levels were decreasing (Int. 46).

An interview of the Unit 2 Supervisor of Operations indicates that his recollection of his involvement in the events regarding this release are contradictory to the recollections of several other individuals (Int. 46, 63, 31, 173, 189, Ref. 80).

According to the Unit 2 Supervisor of Operations, his telephone call served "to check on the readiness condition of State Civil Defense to evacuate people...I needed the contingency of knowing the status of State Civil Defense." Upon further discussion, the Unit 2 Superintendent of Operations provided additional insight into the reason why he made the call directly to PEMA rather than having the ECS notify BRH and PEMA.

"It was not clear to me on the morning of the 30th whether the general emergency had ever been terminated or not. Rather than assume something in the wrong direction, I attempted to act in a conservative direction by renotifying them and if they were already in a general emergency with the State Civil Defense and Bureau of Rad Health, it would only be an information call. If they were not geared up, then it was anticipated that this call would, in fact, bring them to an operational readiness state" (Int. 157).

The investigator determined that the licensee's communications facilities and provisions generally were adequate to support implementation of the emergency plan with respect to initial notifications and the relay of updated information regarding radiological aspects of the response. Early in the event, however, it became apparent that additional telephones were needed to support the operational aspects of the emergency. (See Details I for specific information)

A potentially significant communications problem was averted early in the event. Except for a very few direct outside lines, all calls to the site must be manually routed through a main switchboard. When the switchboard is unattended, no more than one call at a time may be received through the main telephone number. Recognizing that the normal operator would not be available and realizing the potential difficulties that would occur, the Emergency Director located an individual who was able to operate the switchboard and assigned him to attend it.

3. ASSESSMENT, CORRECTIVE AND PROTECTIVE ACTIONS

3.1 EFFLUENT MONITORING AND CORRECTIVE ACTIONS

3.1.1 Liquid and Gaseous Pathways

At 0400 on March 28, TMI Unit 2 experienced a feedwater pump trip, followed by a turbine trip, and approximately 8 seconds later by a reactor trip (Ref. 81). Within 3 to 6 seconds of the turbine trip, the reactor coolant system pressure reached the electromatic relief valve (EMOV) opening setpoint (2255 psig) and within 13 seconds was back down to the EMOV closure setpoint (2205 psig) (Ref. 82, 83). The EMOV failed to reseal, however, and continued to allow reactor coolant to be discharged to the reactor coolant drain tank (RCDT). The pressure and temperature of the RCDT started up almost immediately and continued at a rapid rate for approximately 3 minutes, at which time it leveled off at approximately 120 psig (Ref. 84, 85). At this point, it appears that the RCDT relief valve opened and started releasing steam and water into the reactor building where water, including condensed steam, flowed into the reactor building sump. At 0411 the reactor building sump pumps activated and pumped as much as 8100 gallons of this low activity water to the auxiliary building sump tank (ABST). (There is some low probability that the valve lineup was to the miscellaneous waste holdup tank (MWHT) but evidence presented later strongly indicates that the lineup was to the ABST.) The first pump started at 0408 and the second pump started at 0411. Both pumps remained on until approximately 0438 when they were manually turned off. The RCDT rupture disc failed at approximately 0415 when the maximum RCDT pressure of 192 psig was reached. (Ref. 86, Int. 10).

The estimated flow of 8100 gallons represents a flow rate of 140 gpm per pump. This data was for single pump operation; with two pumps operating the flowrate was probably less than twice the flowrate for a single pump (Ref. 87). If the discharge from the sump was routed to the auxiliary building sump tank, as suspected, the flow would be further reduced since the line size reduces

from 4 inches to 2 inches (Ref. 88). Although normally aligned to the MWHT, it appears from plant records and from an observation by Auxiliary Operator B that the level in the MWHT did not change on March 28. This indicates that the sump pump discharge was aligned to the ABST (Ref. 50, 89, Int. 10). Details I of this report further states that the ABST level was observed to increase, apparently as a result of the RB sump pump operation. The last recorded reading prior to the accident indicated approximately a 700-gallon capacity remained in the auxiliary building sump tank (Ref. 50, 51). This tank would have overflowed into the auxiliary building sump causing water to backup through floor drain piping in the auxiliary and fuel handling buildings.

The amount of radioactivity transferred from the reactor building sump was relatively small, since the transfer was completed prior to fuel cladding failure (Ref. 90). Flooding of the auxiliary building sump created a situation where extremely contaminated reactor coolant could later travel into various parts of the auxiliary and fuel handling buildings via flooded floor drains. Investigators evaluated the possibility of siphoning from the sump to the auxiliary building and determined this to be unlikely because the maximum elevation of the piping above the water level in the reactor building exceeded the height to which the combination of atmospheric pressure and reactor building pressure could lift a column of water (Ref. 92-95).

Retracing the sequence of events to approximately 0415, investigators determined that high pressure in the reactor coolant drain tank just prior to failure of the rupture diaphragm caused a small release of radioactive gases, which was detected by the radiation monitoring system (Ref. 99). This provided the first indication of a leak in the waste gas vent header system that would later be the primary pathway for the release of gases.

The RCDT was connected to the vent header via two paths (Ref. 96). In the first pathway, a 2-inch vent line connects the RCDT to the reactor coolant bleed holdup tanks (RCBHT) through valve WDL-V-1905. This valve automatically closes at 10 psig to protect the RCBHTs but reopens at 6 psig to reestablish

communication with the RCBHTs (Ref. 97). The RCDT pressure exceeded 10 psig approximately 2 minutes after the reactor trip, which should have closed valve WDL-V-1095 (Ref. 84, 85). This valve would have reopened after the rupture diaphragm failed (at approximately 0415) and the pressure in the RCDT decayed to less than 6 psig (at approximately 0435). This would have provided an open line between the RCDT and the RCBHTs which would have remained open until reactor building isolation at approximately 0756 (Ref. 84, 85, 98). Except for a 4.5 psig spike at 0715 and a 12 psig spike at 0745, pressure was on the order of 0 psig to 3 psig in the RCDT from approximately 0500 until reactor building isolation at approximately 0756. The 12 psig spike spanned the time period during which reactor building isolation occurred.

The second pathway is a 1-inch vent line from the RCDT to the waste gas vent header, which is normally isolated from the vent header by valves WDL-V-126 and WDL-V-127 (Ref. 96). Operators have stated in discussions that these valves were open on the morning of March 28. Pressure in the RCDT would have pressurized the waste gas vent header and created a driving force as high as 192 psig which would have been reached just before the rupture disk failed (Ref. 84, 85). The pressurization of the RCDT to the rupture disk failure point coincides with a step increase in gaseous activity as measured at 0415 by the ventilation system monitors. A gradual decline of the monitor readings after the RCDT rupture disk failure indicates that the leak rate into the auxiliary building was dropping with the pressure in the RCDT (Ref. 84, 85, 99). The actual pressure on the vent header could have been as high as 150 psig, which is the reactor building vent header relief valve (WDG-R-3) setting (Ref. 100). WDG-R-3 relieves to the reactor building atmosphere so that its opening would not have resulted in a release to the environment. Flow via either of the two pathways just described would have eventually been released via the leaking vent header.

Early in the morning on March 28 and prior to major fuel cladding failure, the operators became concerned with the increasing pressure in the reactor building (Ref. 101). Reactor building pressure had increased rapidly by 1.4 psig when the rupture diaphragm on the RCDT failed at 0415 (Ref. 84, 85, 95). The operators

apparently were not aware of the failure of the rupture diaphragm on the RCDT, or they didn't realize that the leakage into the RCDT was large enough to pressurize the reactor building. In any event, B once through steam generator (OTSG) was isolated at approximately 0527 because the operators suspected a steam leak from B OTSG into the reactor building (Ref. 101). A primary to secondary leak was not suspected at that time. Isolation of B OTSG may have prevented significant releases from occurring through a primary to secondary leak that developed sometime prior to 0656 (Ref. 102). Prior to 0527, both OTSGs were using the power-operated emergency mainsteam dump valves MS-V-3A and MS-V-3B (atmospheric reliefs) because of a high hotwell level (Ref. 103).

After B OTSG was isolated, A OTSG continued to discharge through the atmospheric relief valve until approximately 0656 when the restarting of the circulating water pumps would have automatically shifted control to the turbine bypass mode of operation (Ref. 104). Records indicate that at approximately 0842 A OTSG was not steaming and it appears to have remained in this condition until approximately 0915. Between 0915 and 1100, steaming from the atmospheric relief on A OTSG was reinitiated. B OTSG remained isolated on the main steam side from 0527 until the main steam isolation valves were briefly opened (7 seconds) at 0656 (Ref. 104). It is possible that steam from the B OTSG was directed to the condenser when the OTSGs switched back to turbine by-pass operation. The B OTSG turbine by-pass valve was closed at 0704 (Ref. 106). The condenser vacuum pump discharge monitor, VA-R-748, alarmed at 0656 due either to the brief opening of the MSIVs or to the switching of OTSG pressure control back to the turbine bypass mode of operation. This was the initial release of detectable activity from the OTSGs (Ref. 102). This activity was primarily noble gases and was discharged to the vent stack (Ref. 107, 108).

Data indicate that B OTSG was effectively isolated and the secondary system, except for B OTSG itself, remained relatively free of contamination. Samples from condenser vacuum pump discharge taken at 0524 and from the condensate pump discharge taken at 1500 showed no iodine activity (Ref. 107, 109). Levels of

noble gas in the range of E-3 $\mu\text{Ci/ml}$ on the condensate pump discharge sample probably were due to background at the time of counting. An isotopic analysis of a sample taken from B OTSG at 1030 on March 30 showed 7.9 $\mu\text{Ci/ml}$ of iodine-131 and 2.1 $\mu\text{Ci/ml}$ of iodine-133 (Ref. 110). These levels are so high that it seems likely that any leakage out of B OTSG would have been easily detected at the condenser or condensate vacuum pump discharge. Cooling using the A OTSG was discontinued at approximately 0842 on March 28 but was initiated again sometime after 0915 (Ref. 111). When the atmospheric relief valve on the main steam line from A OTSG was reopened, a man was sent to the roof of the turbine building at approximately 1100 to survey near the release point (Int. 41). No radiation was detected by this survey. Cooling A OTSG using the atmospheric relief valve continued until approximately 1230 or possibly as late as 1315 (Ref. 112). No further use of the atmospheric relief valve was recorded.

NRC personnel performed radiation surveys in the visible steam plume from the atmospheric relief valves between 1130 and 1200. Radiation levels as high as 20 mR/hr were attributed to the plume (Ref. 113). Based upon data showing effective isolation of the B OTSG, investigators concluded that these readings were caused by the noble gas plume (which was invisible) coming from the Unit 2 vent stack.

The misidentification of OTSG sample lines that run to the primary sample room contributed to release of iodines through the industrial waste treatment system. In addition to the normal OTSG sample lines which are piped to the secondary system lab, sample lines are also piped to the primary sample room for the express purpose of taking OTSG samples when there is a primary to secondary leak. Since the unit had no history of primary to secondary leaks, the misidentification of the lines had not been discovered earlier. In an attempt to confirm which OTSG was leaking, samples from both OTSGs were taken at approximately 0724 using the Unit 2 OTSG sample station in the Unit 1 primary sample room (Ref. 108). When surveyed with a GM survey meter the A OTSG sample appeared contaminated and the B OTSG sample did not (Int. 23). The operations person-

nel requested another sample since they felt sure it was the B OTSG that had the leak. A chemistry foreman and the Supervisor, Radiation Protection and Chemistry decided that the sample lines might be mislabelled (Int. 23, 79, 130). The foreman then went to the normal (secondary side) steam generator sample station in Unit 2 which they knew to be piped correctly, and resampled the two OTSGs. This time the B OTSG was contaminated and A OTSG was not. While taking these samples, the sample lines were flushed until radioactivity was detected in the B OTSG sample. This flush water went into the control and service building sump where it was later pumped to the industrial waste treatment system (IWTS). A diagram of the sump discharge system is shown in Figure II-3-1. This was perhaps the first and most significant source of contaminated water to the IWTS during the first three days. It was not determined when the turbine building sump became contaminated or what contribution the turbine building sump made to activity levels found in the IWTS. It is known that the activity in the IWTS, which had been declining on March 30, increased on March 31 (Ref. 114). This coincided with the sampling of the B OTSG at 2045 on March 30 in the secondary sample room and also with the pumping of the turbine building sump to the IWTS (Ref. 110).

In summary, the EMOV was allowing reactor coolant to flow to the RCDT through the failed rupture diaphragm into the reactor building sump. The reactor building sump pumps were turned off at approximately 0438 after pumping as much as 8100 gallons of relatively low activity water to the ABST, which would have overflowed into the auxiliary building sump and caused the auxiliary building sump to overflow (Ref. 86, 87). The RCDT was open to the waste gas system through two pathways and had provided indication that the waste gas system vent header was already leaking (Ref. 96, 99). B OTSG developed a leak sometime prior to 0656 but did not become a significant contributor to the effluent releases because it was isolated early due to suspected (but nonexistent) steam leaks in the reactor building (Ref. 102).

Despite these plant conditions, there were still no significant radiological indications outside of the reactor building.

SUMP PUMP DISCHARGES TO THE INDUSTRIAL WASTE TREATMENT SYSTEM

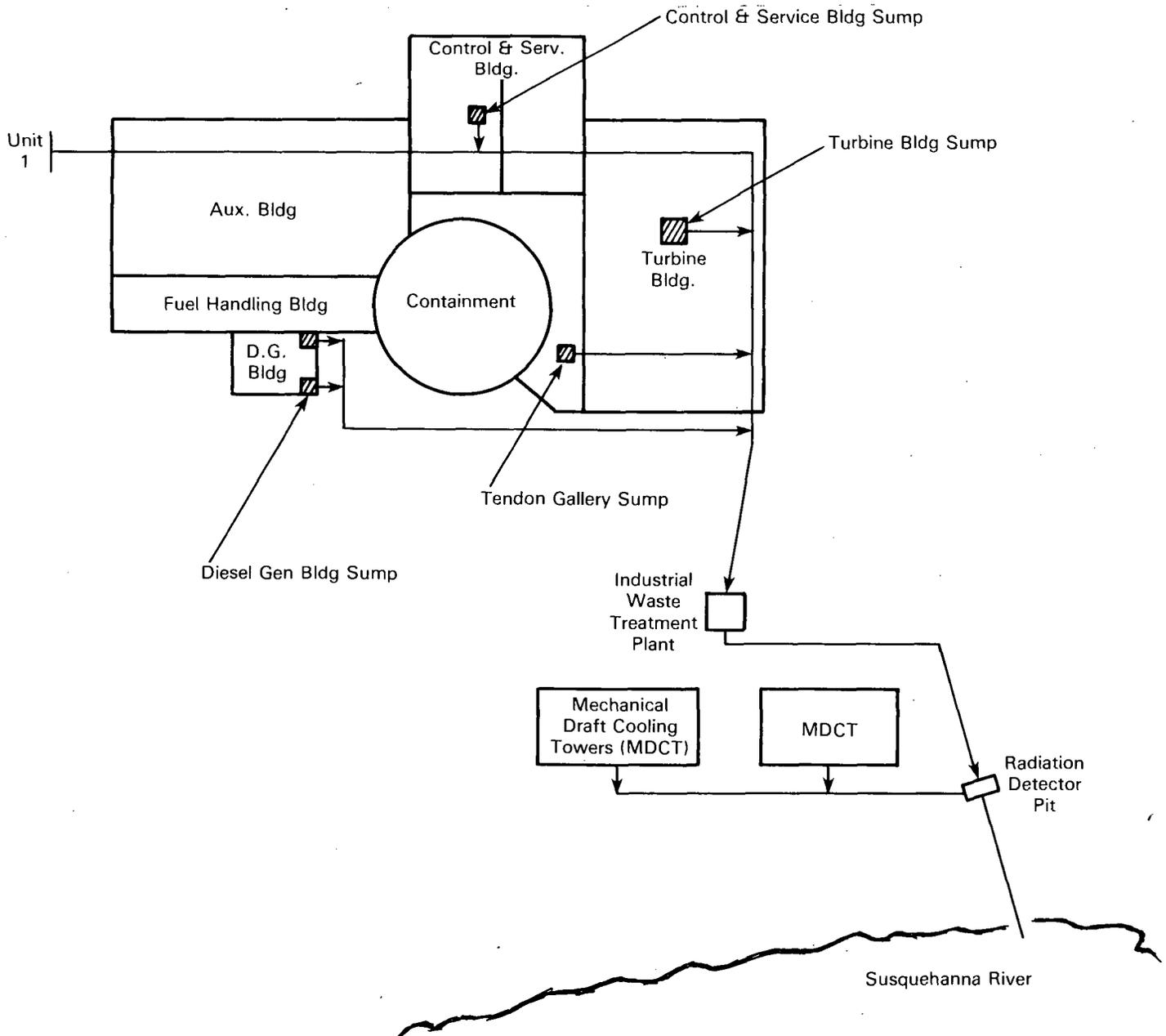


Figure II-3-1

At approximately 0622 on March 28, the first area radiation monitor response to fuel cladding damage was detected (Ref. 115). The incore instrument area monitor, HP-R-213, located on the 347-ft elevation in the reactor building began increasing at a rate that took it from approximately 1 mR/hr to greater than 10 R/hr (off scale) in 14 minutes. This monitor had previously dropped from 100 mR/hr to 1 mR/hr almost simultaneously with the reactor trip because of the rapid decay of nitrogen-16 in the primary coolant, indicating that the monitor could detect radiation in the primary coolant loops. It is thus possible that HP-R-213 was initially detecting the collection of fission product activity in the hot leg loops to the OTSGs, or it may have been detecting gases discharged into reactor building via the RCDT. In either case, this was the first clear indication of fuel cladding failure as detected by the radiation monitoring system. Data from HP-R-213 is supported by data from HP-R-214, the containment dome monitor. Monitor HP-R-214 is a heavily shielded ionization chamber mounted on the top of the reactor building elevator shaft (Ref. 44, 116). This monitor was increasing by 0627 at a rate comparable to HP-R-213.

Major fuel cladding damage was particularly significant from a radiological safety and environmental release standpoint since millions of curies of radioactive nuclides were present in the gap between the fuel pellets and the cladding. Very little of this activity normally escapes from the fuel cladding. With the huge inventory of gap activity released in such a short time, the activity per unit volume of liquids and gases in the reactor coolant system became so high that almost every leak, no matter how small in volume, became significant from a radioactivity standpoint. Gaseous radioactivity was detected in the auxiliary building by the atmospheric radiation monitor system within 15 minutes of the first indications of gap activity release as indicated by radiation monitors in the reactor building. (Ref. 99). Since the only sources of such high radioactivity were the reactor building and systems containing reactor coolant, one of these sources had to be leaking.

The reactor building was probably not a major contributor to releases to the fuel handling and auxiliary buildings since no detectable changes in radioactivity in these buildings occurred when the reactor building pressure went negative at 1530 on March 28 (Ref. 95, 118). The RCDT vented some highly radioactive gases to the waste gas system vent header which leaked into the auxiliary building and were detected by the atmospheric monitors (Ref. 99, 117). The RCDT pathway was closed by reactor building isolation at 0756 (Ref. 98).

Investigators have concluded that reactor coolant flowing through the makeup and purification system was the major source of releases both initially and for the duration of the accident. The reactor coolant makeup and purification system continually takes a portion of the water out of the reactor coolant system and passes the water through demineralizers then through filters and into the makeup tank (Ref. 119). The makeup pumps return water from the makeup tank to the reactor coolant system (RCS). The rate of return to the RCS is varied to provide the inventory control, with the inventory need being determined by pressurizer level. In addition, a small but important amount of the water is pumped to the reactor coolant seals. This system isolated automatically on reactor building isolation but was quickly returned to service in order to maintain seal water to the reactor coolant pumps and control pressurizer level (Ref. 120). The makeup tank is designed to operate with a gas space occupying about one third of the normal tank volume (Ref. 114). It was in this space that gases evolving from the cooled and depressurized reactor coolant collected. One indication that this tank was a source of gaseous leakage was a correlation of monitor readings with the makeup tank level. This correlation was considered reliable since the makeup tank is not normally vented and a change in level causes a corresponding change in pressure. If the makeup tank vent valve, MU-V-13, was closed (as it normally is) during this period, there was apparently leakage past its seat; however, the investigators were unable to definitely establish if it was closed.

Starting at approximately 0635 there appears to be a direct relationship between some of the atmospheric monitor responses and the level in the makeup tank (MU-T-1) (Ref. 118, 121). An example of a monitor which correlated well

with the makeup tank level was the gas channel for HP-R-228, auxiliary building exhaust duct B. The correlation is such that the monitor responded upward rapidly as the level was increased in the makeup tank and when the level in the makeup tank decreased the slope of the monitor response started decreasing. It appears that the vent header was pressurized on a rising tank level; however a falling tank level did not draw gas back out of the header since there is a check valve to prevent flow back to the tank (Ref. 96). The vent header pressure would then decrease because of a leak in the vent header or because the waste gas compressor (if working properly) was pumping the gas into a waste gas decay tank (Ref. 122).

Early on March 29, the buildup of noncondensable gases in the makeup tank caused a reduction in letdown flow due to the back pressure from the makeup tank (Int. 173, 189). Manual venting of the makeup tank to reduce pressure was initiated at 0710 on March 29 and continued on a periodic basis until sometime after March 30 (Int. 157, 173, 189). The venting process initially consisted of short periods of time when MU-V-13 was opened to vent makeup tank pressure into the waste gas decay tanks via the leaking vent header (Int. 173). The pressure buildup, however became too rapid to control with short venting periods and on the morning of March 30, the pressure in the makeup tank caused the liquid relief valve (MU-R-1) to open, forcing all of the water from the makeup tank into the reactor coolant bleed holdup tanks. Sometime after the opening of MU-R-1 (between 0600 and 0710 on March 30) the pressure in the RCBHTs was observed to be greater than 30 psig (Int. 173). This pressure exceeded the normal relief valve setpoint of 20 psig. Because of the potential operational problems created by the high pressure and resulting low water level in the makeup tank, two shift supervisors decided jointly to vent the makeup tank in an attempt to reduce pressure and reestablish a normal water level in the makeup tank. The makeup tank vent valve (MU-V-13) was opened at approximately 0710 and remained opened for at least 2 hours (Int. 157, 173, 189). Because of the high pressure in the RCBHTs, an uncontrolled release was likely already in progress via the relief valves on the RCBHTs.

A 1200 mR/hr beta-gamma exposure rate was measured from a helicopter 130 ft above the Unit 2 reactor building on March 30 at approximately 50 minutes after the opening of the makeup tank vent valve. This exposure rate was due to gas leakage from the waste gas vent header which was released to the environment via the auxiliary building ventilation exhaust.

The investigators concluded from the makeup tank level that the makeup tank relief valve was closed at the time of the measurement. This indicates that the RCBHT relief valves were closed. Manually closing the makeup tank vent valve, however, would likely have caused the makeup tank relief valve to reopen, causing the pressure in the RCBHTs to increase, resulting in the reinitiation of an uncontrolled release through the RCBHT relief valve. It is likely that for some period of time during this event, a release occurred over which operators had no practical control. Communications with the State and other actions of a less operational nature are discussed in section 2.3.2.

Discussions with licensee personnel involved in a continuing program to identify the sources of leaks which occurred during the accident indicate that the waste gas compressors were the most likely pathway for gases which leaked from the waste gas system. These compressors use a water seal and discharge a mixture of liquid and gases into a separator tank (Ref. 123). High pressures in the vent header and/or liquid in the vent header may have caused these pumps to lose the water seal and thus the ability to move the gas into the waste gas decay tanks (Ref. 123). Gas could have been discharged through a relief valve on the separator or possibly through the automatic liquid overflow to the auxiliary building sump tank (ABST) either as a gas or as a water-gas mixture. The rupture disc on the ABST had failed prior to March 28 and any gases going into the ABST would have escaped immediately to the auxiliary building. The release pathways of the gases from the auxiliary building to the environment are discussed in Section 3.1.2.

The amount of liquid released into the auxiliary building and the effect of this release on airborne radioactivity levels in the plant are not accurately known. However, liquid leaking from the reactor coolant system into the fuel handling and auxiliary buildings was cooled and depressurized, thus releasing entrained and dissolved gases. Evaporation of water leaking onto the floor probably contributed to airborne iodine and particulate activity inside the auxiliary and fuel handling buildings. Several pathways for liquid releases into fuel handling and auxiliary buildings from the makeup and purification system are considered possible. Relief valves for those systems containing highly radioactive material were evaluated. The following valves were considered:

- MU-R-1 Makeup tank relief valve; setpoint 80 psig; discharges to the reactor coolant bleed holdup tanks (Ref. 100, 124),

- MU-R-2 Upstream of seal return cooler; setpoint 150 psig; discharges to the makeup tank (Ref. 100, 124),

- MU-R-3 Letdown relief valve; setpoint 130 psig; discharges to the reactor coolant bleed holdup tanks (Ref. 100, 124),

- MU-R-4 Inlet to the core flood tanks, setpoint 700 psig; discharges to the floor drain (Ref. 100, 124),

- MU-R-5A/B Makeup and purification demineralizer relief valves; setpoint 150 psig; discharges to the floor drain system (Ref. 100, 124).

Of these valves, MU-R-5A and B were determined to be likely pathways to the auxiliary building sump. There were difficulties in maintaining letdown flow on March 28 and the block orifice MU-1-FE was bypassed in an attempt to increase letdown flow (Ref. 125). Letdown flow restrictions could have caused the pressure to exceed the setpoint of MU-R-5A and 5B (150 psig). An auxiliary operator stated that letdown flow was oscillating at times as if a relief valve

were opening and closing (Ref. 125). MU-R-3 may have been opening and closing, however this valve does not discharge to the floor drains (Ref. 124). It could have contributed to the inventory in the RCBHTs, increasing the potential for this tank to discharge liquid into the waste gas system vent header and out thru vent header liquid drains. A liquid release via the RCBHT vent line probably did not occur before March 30.

On March 30, makeup tank relief valve MU-R-1 lifted at approximately 0430 and stayed open until approximately 0715 (Int. 173, 189, Ref. 121). During this period, a large amount of water was pumped into the RCBHTs (Ref. 68, 158). Two of these tanks indicated a high level (full) and tank pressures of greater than or equal to 30 psig, which means water from the tanks could have flowed into the vent header (Int. 173). Water from the tanks could have flowed into the sump via the vent header liquid drains, the waste gas compressor drains, and waste gas decay tank drains (as discussed previously).

Relief valves MU-R-2 and MU-R-4 were not determined to represent significant potentials for release. MU-R-2 discharges to the makeup tank and MU-R-4 could only be a pathway during filling of the core flood tanks (Ref. 124).

Another potentially significant pathway developed at approximately 1352 on March 28 when seal water to various radwaste system pumps was lost. Seal water was lost because motor control centers (MCC) 2-32A and 2-42A became inoperable (Ref. 126). MCC 2-32A was made operable again at 0700 on March 29. Loss of seal water resulted in significant leakage from pumps WDL-P-5A & B, which take suction on the reactor coolant bleed tanks (Ref. 127). Because of letdown relief valve (MU-R-3) discharges and because of their use in the letdown flow path, these tanks contained high levels of radioactivity. A sample of RCBHT-C on March 30 showed an iodine-131 concentration of 109 $\mu\text{Ci/ml}$ (Ref. 128). Other radwaste pumps were of less significance since the pumps were taking suction on tanks containing pre-accident water.

There was known leakage from seals on the river water pumps (RR-P-1A, 1B, and 1C) which, although not radioactive, contributed to flooding around the auxiliary building floor drains on the 281-ft elevation. This input of clean water into the sump may have forced contaminated water out of the floor drains and into the control and service building sump and the diesel generator sump. These two sumps were eventually pumped to the industrial waste treatment system (Ref. 129). (See Figure II-3-1.)

Additional leaks of radioactivity probably occurred through system leaks in the letdown and makeup systems (Ref. 130). From maintenance records and discussions with licensee personnel, investigators determined that there were small liquid leaks from valves and instruments in the makeup and purification system (Ref. 130). The contribution from these sources to the activity released to the auxiliary building sump could not be determined. The following are examples of some of the potential pathways taken from maintenance work requests:

Letdown sample monitor (MU-R-720) sample pump had a leak at the pump suction union. This had created a radioactive hot spot. Work request 1806 dated March 22, 1979 (Ref. 130).

Makeup Pump 1C, casing drain plug leaked. Work request 1297 dated March 20, 1979 (Ref. 130).

Makeup valve 17, packing leaked. Work request 1825 dated March 22, 1979 (Ref. 130).

Flow Transmitter MU-7-FIT leaked downstream of MU-V-383D and MU-V-384D. Work request 1856 dated March 27, 1979 (Ref. 130).

There are numerous other examples similar to those above indicating leaks in the makeup system (Ref. 130). Discussions with licensee personnel indicated that such leaks ranged from a few drops per hour to perhaps a few tenths of a gpm. From a review of preaccident leak rate determinations required by Technical Specifications, an investigator estimated the total preaccident leakage from all reactor coolant system sources in the auxiliary building to be from 0.4 to 1.6 gpm. Leakage from other sources which developed during the accident would have added to this basic leak rate. There are some potential liquid release paths such as drain pots off of the waste gas vent header and automatic drains off of the waste gas compressors and waste gas decay tanks that have not been thoroughly evaluated.

Although liquid volume release rates appear to be small, the specific activity of the reactor coolant was so high that any leakage was significant (Ref. 131). Gases evolved from reactor coolant released into the fuel handling and auxiliary buildings contributed to airborne radioactivity inside these buildings. After March 30 this source of airborne radioactivity may have been of little significance for noble gases but could have been a contributor to the radioactive iodine which evolved more slowly through evaporation of liquid on the floors.

The following table lists the major nuclides for a sample of reactor coolant taken on March 29.

<u>Nuclide</u>	<u>Half life</u>	<u>Coolant Concentration</u> <u>μCi/cc</u>
iodine-131	8 d	1.3 E 4
iodine-133	20.8 h	4.6 E 4
cesium-134	2 y	6.3 E 1
cesium-136	13 d	1.8 E 2
cesium-137	30 y	2.8 E 2
barium-140	12.8 d	21.0 E 1
strontium-89/90	50 d/29y	5.3 E 0
gross α	—	3.6 E-4

*Reactor coolant sample taken at approximately 1700 on March 29.

3.1.2 Determination of Release Rates and Amounts

3.1.2.1 Noble Gases

This section is a summary of release rates calculated by the licensee using TLD data and actual X/Q values for the periods that the TLDs were exposed (Ref. 85).

In order to provide a perspective of the releases for the first three days of the accident, data for the period March 28 through April 30 is shown even though this investigation was generally limited to the period March 28 through March 30. These data represent approximately 99% of all of the noble gas releases.

The following noble gas release values are from Table IV-B-1 of the licensee's "Third Interim Report on Three Mile Island Nuclear Station Unit 2 (TMI-2) Accident, July 16, 1979" (Ref. 84, 85). The release values in this table were calculated by the licensee by applying atmospheric dispersion factors to environmental results. These calculations were not verified by the investigators. The licensee values are consistent with a preliminary assessment by the NRC staff which estimated a release of $1.3 \text{ E } 7 \text{ Ci}$ for the period March 28 through April 5.

PERIOD OF TLD EXPOSURE (MO, DAY, HR)

	03-28-07 to 03-29-16	03-29-17 to 03-31-16	03-31-17 to 04-03-15	04-03-16 to 04-06-13	04-06-14 to 04-03-24	TOTAL
Isotope						
Xe-133	4.9E6	2.1E6	1.1E6	2.7E5	1.5E4	8.3E6
Xe-133m	1.2E5	3.9E4	1.5E4	1.9E3	0	1.7E5
Xe-135	1.5E6	7.7E4	1.4E3	0	0	1.5E6
Xe-135m	1.4E5	1.3E3	0	0	0	1.4E5
Kr-88	6.1E4	0	0	0	0	6.1E4
TOTAL	6.6E6	2.2E6	1.1E6	2.7E5	1.5E4	1.0E7

*The last three weeks of the month are combined into one group since the contribution is less than 1% of the total.

Using this table, the noble gas releases can be compared to 10 CFR 20.106 limits. A footnote to Appendix B, 10 CFR 20 states that releases to unrestricted areas should be limited, such that:

$$\frac{C_A}{MPC_A} + \frac{C_B}{MPC_B} + \frac{C_C}{MPC_C} \leq 1$$

where C_A is the concentration of radionuclide A divided by the maximum permissible concentration of nuclide A as given in Appendix B to 10 CFR 20. If the quantities (Ci) of noble gases released between March 28 and April 30 were assumed to be the total releases for a year, the quantities could be divided by the number of seconds in a year to obtain an average annual release rate in curies per second. This yields:

xenon-133	$\frac{8.3 \text{ E } 6 \text{ curies/year}}{3.15 \text{ E } 7 \text{ seconds/year}}$	=	2.6 E-1 Ci/sec
xenon-133m	$\frac{1.7 \text{ E } 5 \text{ curies/year}}{3.15 \text{ E } 7 \text{ seconds/year}}$	=	5.4 E-3 Ci/sec
xenon-135	$\frac{1.5 \text{ E } 6 \text{ curies/year}}{3.15 \text{ E } 7 \text{ seconds/year}}$	=	4.8 E-2 Ci/sec
xenon-135m	$\frac{1.4 \text{ E } 5 \text{ curies/year}}{3.15 \text{ E } 7 \text{ seconds/year}}$	=	4.4 E-3 Ci/sec
krypton-88	$\frac{6.1 \text{ E } 4 \text{ curies/year}}{3.15 \text{ E } 7 \text{ seconds/year}}$	=	1.9 E-3 Ci/sec

Applying the dispersion factor (X/Q) given in the Environmental Technical Specifications, ($6.7 \text{ E-6} \frac{\text{sec}}{\text{m}^3}$) one obtains the following:

$$\text{xenon-133: } 2.6 \text{ E-1 } \frac{\text{Ci}}{\text{sec}} \times 6.7 \text{ E-6 } \frac{\text{sec}}{\text{m}^3} = 1.7 \text{ E-6 } \frac{\text{Ci}}{\text{m}^3}$$

$$\text{xenon-133m: } 5.4 \text{ E-3 } \frac{\text{Ci}}{\text{sec}} \times 6.7 \text{ E-6 } \frac{\text{sec}}{\text{m}^3} = 3.6 \text{ E-8 } \frac{\text{Ci}}{\text{m}^3}$$

$$\text{xenon-135: } 4.8 \text{ E-2 } \frac{\text{Ci}}{\text{sec}} \times 6.7 \text{ E-6 } \frac{\text{sec}}{\text{m}^3} = 3.2 \text{ E-7 } \frac{\text{Ci}}{\text{m}^3}$$

$$\text{xenon-135m: } 4.4 \text{ E-3 } \frac{\text{Ci}}{\text{sec}} \times 6.7 \text{ E-6 } \frac{\text{sec}}{\text{m}^3} = 2.9 \text{ E-8 } \frac{\text{Ci}}{\text{m}^3}$$

$$\text{krypton-88: } 1.9 \text{ E-3 } \frac{\text{Ci}}{\text{sec}} \times 6.7 \text{ E-6 } \frac{\text{sec}}{\text{m}^3} = 1.3 \text{ E-8 } \frac{\text{Ci}}{\text{m}^3}$$

Ci/m^3 equals $\mu\text{Ci}/\text{cc}$, therefore the above concentrations can be substituted into the summation equation as follows:

$$\begin{aligned} \sum_i \frac{\text{Ci}}{\text{MPC}_i} &= \frac{1.7 \text{ E-6}}{3 \text{ E-7}} + \frac{3.6 \text{ E-8}}{3 \text{ E-7}} + \frac{3.2 \text{ E-7}}{1 \text{ E-7}} + \frac{2.9 \text{ E-8}}{3 \text{ E-8}} + \frac{1.3 \text{ E-8}}{2 \text{ E-8}} \\ &= 11 \end{aligned}$$

This means that noble gas releases when averaged over a year are at least 11 times the limit in 10 CFR 20.

3.1.2.2 Iodine

The amount of radioactive iodine released was calculated based on continuous samples collected from the Unit 2 station vent. The samples were taken from monitor HP-R-219. A sample for the period 0400 on March 28 through 1900 on March 28 was lost prior to analysis. The amount of radioactivity released during this period was estimated by a licensee contractor based on samples from auxiliary building and fuel handling building ventilation exhausts.

Data shown below was extracted from TABLE IV-C-1 of the "Second Interim Report on the Three Mile Island Nuclear Station Unit-2 (TMI-2) Accident, "June 15, 1979, published by the licensee (Ref. 84).

TABLE IV-C-1

Smoothed Iodine Release Rate Data
Used in Dose Assessments

Start Date (Mo. Date Hr.)	¹³¹ I Release Rate
	<u>μCi/sec</u>
79032804	4.2 ¹
79032819	22.7
79033022	2.7
79040106	9.7
79040303	2.3
79040319	7.0
79040519	0.43
79040615	3.7
79040706	6.9
79040803	12.7
79040909	0.46
79041016	1.3
79041119	2.2
79041323	4.1
79041410	6.6
79041505	8.6
79041508	14.0
79041518	6.0
79041616	11.0
79041624	3.0
79041716	5.5
79041804	7.5
79041808	2.0
79041914	5.5
79042022	1.5
79042213	2.5
79042304	1.0
79042312	3.8
79042316	1.5
79042406	0.80
79042516	0.50
79042600	0.46
79042700	0.37
79042808	0.38
79042900	0.56
79043000	0.48

¹ Iodine release rates are roughly constant over the period from one start time to the next.

From this table, investigators estimated that the amount of iodine-131 released through April 30 was approximately 14 curies. This number differs with results of an analysis by a licensee contractor of samples taken on April 12 from the charcoal adsorber in the fuel handling building and auxiliary building ventilation systems. This analysis determined 26.84 Ci of iodine-131 were released from March 28 through April 12. The detailed data for the charcoal adsorber analysis are in a licensee contracted report by Nuclear Consulting Services, Inc., "Analysis of the Adsorbers and Adsorbents from Three Mile Island Unit 2," May 25, 1979. The method used for this analysis was not reviewed by the investigators.

If the release rate in curies per second is calculated for the 26.84 curies above, the release rate can be multiplied by the X/Q in the Environmental Technical Specifications to show that concentrations of iodine-131 (only) at the site boundary would have been less than 10 CFR 20 Appendix B limits if averaged over a year as allowed by 10 CFR 20.106.

The radiation monitoring stripcharts indicate that significant noble gas releases started at approximately 0700 on March 28 but significant iodine releases did not start until a few hours later (Ref. 99, 117). Unit 2 gas monitors rapidly went offscale, with the Unit 2 particulate and iodine monitors going offscale within 1-2 hours after the gas monitors. A review of data from the Unit 1 fuel handling building and auxiliary building ventilation system monitors which were monitoring leakage into Unit 1 from the Unit 2 fuel handling building shows the iodine channels increased and fell rather sharply back to near previous levels. This indicates that the monitors were responding to noble gases rather than iodine since iodine would have accumulated on the charcoal sample cartridge and caused a continuous increase in the reading (Ref. 69).

Neither particulate monitors nor iodine monitors on the effluent release paths provided any quantitative information because of the high noble gas background. Even though HP-R-219 particulate and iodine channels came back on scale

during the period under investigation, it is not known what fraction of the reading should be attributed to iodines and what fraction to noble gas. For one nine hour period on March 30, the iodine channel reading on HP-R-219 decreased instead of increased.

Since the iodine detector was measuring a buildup of iodine on a charcoal cartridge, the rate of increase (cpm/min) must be determined in order to establish a concentration. A declining monitor response would probably indicate that noble gases adsorbed on the filter were being removed at a rate exceeding the iodine buildup or the background radiation was declining at a rate greater than iodine buildup rate. Either situation means the monitor reading should not be used.

It is probable that most of the releases were via the Unit 2 auxiliary building and fuel handling building ventilation systems with possibly some short releases from the relief valve vent header directly to the vent stack (Ref. 122). There was some leakage into Unit 1 via the fuel handling buildings (Int. 79). Gaseous radioactivity in Unit 1 increased significantly whenever the Unit 2 ventilation systems were turned off (Ref. 69). The Unit 2 and Unit 1 fuel handling buildings are not physically separated in the spent fuel pool areas. Shift Foreman B stated that the Unit 2 ventilation system supply fans tripped and remained off because of high radiation levels, but the exhaust fans operated continuously except for a few brief periods when the ventilation systems were turned off in an attempt to reduce the release rates.

Securing the fuel handling building and auxiliary building ventilation systems early on March 28 and again on March 29 caused exposure rates to increase significantly in the Unit 2 auxiliary building, thus hampering emergency activities. Perhaps more important was the fact that control room airborne radioactivity levels started increasing when the ventilation systems were shutdown. Difficulty in maintaining control room isolation due to heavy personnel traffic into and out of the control room in conjunction with poor meteorological conditions allowed

gases escaping from the plant to be picked up in the ventilation system supply tunnel. This was the likely cause for gaseous activity increases in the control room. Because of the need to ensure habitability of the control room and to keep dose rates as low as possible in the auxiliary building to facilitate emergency activities, the ventilation systems were subsequently kept in operation.

3.1.2.3 Liquid Release

The only significant release of liquid occurred through the industrial waste treatment system (IWTS). The IWTS does not normally process radioactive fluids, and there are no radiation monitors in this system. The effluent, after dilution, is sampled continuously by plant discharge monitor RM-L7 just prior to being discharged into the Susquehanna River.

An investigator determined from discussions with Chemistry Foreman A that contaminated water probably was initially introduced into the industrial waste treatment system from the control and service building sump. This sump receives the discharges from the secondary sample system sink. The sample line from the contaminated B OTSG was flushed into the secondary sample sink for approximately 45 minutes commencing at 1000 March 28. Other sumps were known to contain radioactive water after March 30 as a result of the accident; but only the turbine building sump should have been contaminated because of the primary-to-secondary leak. It could have been a source of contamination to the IWTS during the first three days. How soon contaminated water was pumped to the IWTS could not be determined; but levels of iodine-131 as high as $E-4 \mu\text{Ci/cc}$ were detected in the IWTS as early as 1000 on March 29.

At 0400 on March 28, a discharge from the IWTS was in progress. This discharge continued until 0900, when it was stopped because of the problems in Unit 2 (Ref. 138). Approximately 29,100 gallons were released between 0400 and 0900. There is no evidence to suggest that this release was contaminated. A small increase in the reading on RM-L7 during this period was considered by

the investigator to be due to increasing background radiation from Unit 2 gaseous releases. This is supported by the fact that the reading on RM-L7 did not change after the liquid discharge was terminated.

On March 29, a release was started at 1315 but terminated at 1410 because of high levels of xenon reported in a sample of the IWTS water taken at 1225 (Ref. 138, 139). This analysis, performed in the NRC mobile laboratory (NRC R-I Van), was later determined by the laboratory personnel to be error due to the high noble gas background. This was reported to the licensee and the release was restarted at 1610 and continued until 1815. The release was terminated at 1815 on orders originating from the NRC executive management team at NRC Headquarters (Ref. 140). A sample of the IWTS water taken between 1000 and 1100 on March 29 and analyzed on March 31 contained iodine-131 at a concentration of $1.2 \text{ E-4 } \mu\text{Ci/ml}$ (Ref. 139). For estimating releases, it was assumed that $1.2 \text{ E-4 } \mu\text{Ci/ml}$ was the maximum value during the releases on March 29. A small increase in the reading on RM-L7 was noted at approximately the time the release began at 1610 (Ref. 102). The reading on RM-L7 remained relatively constant during the release, indicating that the radioactive characteristics of the release were not changing significantly during the release (Ref. 102). Releases from the IWTS on March 29 totaled 29,950 gallons containing an estimated 12.66 millicuries of iodine-131.

The IWTS remained shut down from 1815 on March 29 until 0300 on March 30. During this period of time, there were several communications between the NRC and various State and Federal agencies (Pennsylvania Bureau of Radiological Health, a representative of the State of Maryland, and Region III of the Environmental Protection Agency) regarding discharges from the IWTS (Ref. 141). The most significant communications were from the Pennsylvania Bureau of Radiological Health (BRH) to the NRC, requesting that the NRC stop the licensee from discharging liquid waste to the river. The last such BRH request was at 1825 on March 29 and asked the NRC to holdup discharges to the river. The BRH rescinded this request at 2205 on March 29. At approximately 0005 on March 30, the NRC

executive management team instructed NRC personnel onsite to inform the licensee that releases from the IWTS could resume (Ref. 141). The licensee was instructed to notify the NRC when releases were resumed (Ref. 141).

At 0300 on March 30, a release was initiated from the IWTS and continued for the remainder of the day (Ref. 138). During the period from 0300 to 2400 three samples were taken which, when analyzed on March 31, showed positive results for iodine-131 (Ref. 139). These results were used, in conjunction with the sample results from the March 29 sample, which also showed iodine-131, to estimate average concentrations over the period of the release (Ref. 139). It is estimated that the releases from the IWTS for March 30 totaled 206,400 gallons containing 59.91 millicuries of iodine-131.

In summary, it is estimated that releases from the IWTS for the period from 0400 March 28, until 2400 March 30, totaled 265,450 gallons containing 72.56 millicuries of iodine-131. Based on an annual average, the releases were below 10 CFR Part 20, Appendix B, limits for releases to an unrestricted area.

3.2 IN-PLANT ASSESSMENT AND PROTECTIVE ACTIONS

This section describes, in chronological order, the in-plant radiation protection actions taken by the licensee during the period 0400 March 28 to 2400 March 30.

The section is divided into five subsections. The first four subsections describe in-plant radiation protection during discrete phases of the emergency organization. The fifth subsection is a summary of the radiation protection findings.

3.2.1. Radiation Protection and Chemistry Prior to Declaration of the Emergency

Shortly after hearing the announcement that Unit 2 reactor had tripped, Radiation Chemistry Technicians L and P placed the Unit 2 reactor coolant sample system on recirculation in anticipation of performing an isotopic analysis for radioiodine as required by Technical Specification 4.4.8 (Int. 58, 96). About this same time (approximately 0430), Auxiliary Operator B noted an alarm (500 cpm) on the count rate meter (RM-14/HP-210) located near the model room access door and an indication that both reactor building sump pumps were running (Int. 10). The auxiliary operator adjusted the RM-14 meter from the X1 to the X10 scale, performed a survey with an exposure rate meter (E-520) in the area, and found less than 0.1 mR/hr. He informed Radiation/Chemistry Technician I of his findings. He also informed Control Room Operator D of his findings and concern.

About 0445, Radiation/Chemistry Technician L was requested to collect a reactor coolant sample and to analyze it to determine boron concentration (Int. 58). This analysis, performed about 0515, indicated approximately 700 ppm boron. This was a decrease of about 300 ppm from the 1026 ppm determined at 0330, prior to the reactor trip (Ref. 90). The technician stated that he did not understand why the boron concentration had decreased, since he had been informed that water from the borated water storage tank (BWST) had been added to the reactor coolant system. He informed Shift Foreman C of the boron analysis results. Shift Foreman C stated that he also did not understand the decrease and requested a second sample be analyzed.

During this same period of time, Radiation/Chemistry Technicians B and I sampled the main condenser vacuum pump discharge as required by Technical Specification Table 2.3-2, Note (3) (Int. 42, 106). This sample was analyzed at 0548 and did not appear to indicate a primary-to-secondary leak.

At 0545 the Supervisor, Radiation Protection and Chemistry arrived in the control room. His major concern at that time was that an engineered safeguards (ES) function had occurred resulting in the injection of sodium hydroxide into the primary coolant system. He did not address this concern at that time, since he was immediately approached by the Unit Superintendent, Technical Support and advised that there were indications of a leak inside the reactor building and that a sample of the reactor building atmosphere was needed (Int. 20).

Indication of a leak included both reactor building sump pumps starting automatically, followed by a reactor building sump high water level; an increase in reactor coolant drain tank pressure, burst of the rupture disk followed by a rise in reactor building pressure to 1.4 psig; increase of the intermediate closed cooling water coolant letdown monitor (IC-R-1092) to five times its alarm setpoint, followed by an alarm on the particulate channel of the reactor building air monitor (HP-R-227(P)) (Ref. 142). In addition, at 0521 an operator had requested from the computer several parameters which indicated the electromatic relief discharge pipe temperature was at least 64°F hotter than discharge pipe from the code safety valves (Ref. 143). These indications were consistent with a release from the pressurizer. Interviews with the Supervisor, Radiation Protection and Chemistry did not indicate that he was informed of these indications.

The Supervisor, Radiation Protection and Chemistry and Radiation/ Chemistry Technician I immediately proceeded to the 305-ft elevation of the auxiliary building to take a gas sample and remove the charcoal cartridge from monitor HP-R-227. As the technician loosened the wing nuts that secured the charcoal cartridge holder, water began to spray with enough force to propel the cartridge from the holder. The technician caught the cartridge and forced it back into the holder and secured the wing nut. The water spray contaminated his hand. He was later decontaminated (Int. 20, 42). The Supervisor, Radiation Protection and Chemistry returned to the control room and informed Unit 2 Superintendent, Technical Support of the unsuccessful attempt to obtain the air sample. The Unit 2 Superintendent, Technical Support informed the Supervisor, Radiation Protection and Chemistry of concern over the 700 ppm boron concentration. He requested that the analysis be verified.

Returning to the Unit 1 laboratory area, the Supervisor, Radiation Protection and Chemistry discussed the boron analysis results with Radiation/Chemistry Technician L and learned that a confirmatory sample was being taken, by Radiation/Chemistry Technician P (Int. 96). At about this same time (approximately 0620), the Supervisor, Radiation Protection and Chemistry was requested to make preparations for an entry into the reactor building. He asked Radiation/Chemistry Technician I to accompany the operator assigned to enter the reactor building. Radiation/Chemistry Technician I suggested that, since he and the other radiation/chemistry technicians were fatigued, the entry be delayed until the oncoming shift arrived in about 30 minutes. The Supervisor, Radiation Protection and Chemistry agreed with this suggestion, but asked that he and Radiation/Chemistry Technician B prepare to make the entry in the event a delay was impossible. The two technicians left for Unit 2 to make the preparations (Int. 106, 156).

At this time (approximately 0630), the Supervisor, Radiation Protection and Chemistry was informed that two independent boron samples had been completed (one indicated 402 ppm, the other 405 ppm) and that a sodium analysis indicated 0.153 ppm (Int. 20, 94). This concerned him, since he believed it indicated that deboration of the primary coolant system was taking place (Int. 20). He requested the control room be informed of the sample results. Radiation/Chemistry Technician P had measured the gross degassed beta-gamma activity of the reactor coolant and found that it had increased from 0.38 $\mu\text{Ci/ml}$ at 0330 to 4.0 $\mu\text{Ci/ml}$ at 0602. This information, with the boron results, was phoned to the control room (Int. 96).

A reactor coolant sample collected about 0643 indicated a gross gamma activity of 140.73 $\mu\text{Ci/ml}$ (Ref. 144). The Unit 1 nuclear sample room area radiation monitor RM-G3 alarmed (2.5 mR/hr) during collection of this sample. Radiation/Chemistry Foreman B surveyed the area and found 200 mR/hr on contact with the sample lines (Int. 77). These exposure rates were not recorded. The primary coolant recirculation flow was turned off. The nuclear sampling room air monitor, RM-A12, was inoperable during this period of time (Ref. 42). No other measurements of the concentration of radioactive materials in air to which these technicians were exposed were made.

As the Supervisor, Radiation Protection and Chemistry was about to leave the Unit 1 laboratory area for the Unit 2 control room, he heard an area radiation monitor alarm (Int. 20). This alarm, the Unit 1 hot machine shop area monitor (RM-G4), was set at 2.5 mR/hr (Ref. 145). A review of Unit 1 strip chart (RMR-2) showed that this monitor had been reading a steady 0.85 mR/hr until 0645 (Ref. 69). From 0645 to 0650, the reading increased from 0.85 mR/hr to 400 mR/hr. Radiation/chemistry technicians were returning from Unit 2 via the model room door and walking towards the catwalk that crosses over the railroad tracks in the fuel handling area, when they heard the hot machine shop radiation monitor (RM-G4) alarm (Int. 42). Their first action was to order a security guard stationed near the hot machine shop door to leave the area. Then, as one cut the lock off the door, the other returned with a survey instrument (R0-2). The Supervisor, Radiation Protection and Chemistry arrived on the scene and observed as the technicians surveyed the area. The survey indicated general area gamma exposure rates of 500 mR/hr and up to 1 R/hr several feet from the unshielded sample lines located in the hot machine shop. The technicians determined the source of radiation to be the sample lines.

Unit 2 FSAR Section 12.1.2.11, "Sample Station Shielding," states, "All sample and recirculation lines are shielded by placing them in shielded cubicles or pipe chases which are located at sufficient distances from normally occupied areas." An inspector observed that the reactor coolant sample lines were not shielded or placed in a pipe chase as they run from the Unit 2 fuel handling building through the normally occupied access pathways of Unit 1 fuel handling building through the hot machine shop into the nuclear sampling room.

Based on the survey, the Supervisor, Radiation Protection and Chemistry concluded Unit 2 had suffered severe fuel degradation (Int. 20). He contacted the Unit 2 Superintendent, Technical Support and relayed his findings. The Unit 2 Superintendent, Technical Support indicated to him that they were already doing everything they could to get cooling water and boron to the reactor cooling system (Int. 20). He was informed that a Site Emergency was being declared.

3.2.2. Initial In-Plant Assessment and Protective Actions

Day shift radiation/chemistry technicians were arriving in the laboratory area at this time. The Supervisor, Radiation Protection and Chemistry directed Radiation/Chemistry Technician O to proceed to Unit 2 to conduct surveys and warn personnel (Int. 78). He directed Radiation/Chemistry Technician U to call in the Radiation Protection Supervisor and a radiation protection foreman. He placed Radiation/Chemistry Technician I in charge of the Emergency Control Station until relieved by a foreman (Int. 6). He told the technician to start assembling other technicians into onsite and offsite survey teams.

At this point, the Supervisor, Radiation Protection and Chemistry ran to the Unit 2 control room to assume his Emergency Plan responsibilities. As he passed the Unit 1 condensate water treatment area, he heard the radiation emergency alarm (Int. 20).

Upon arrival in the control room, the Supervisor, Radiation Protection and Chemistry proceeded to the radiation monitoring system portion of the control panel. He noticed that the condenser off gas monitor (VA-R-748) was in alarm, and concluded that a primary to secondary leak had developed. He noted that other radiation monitors were increasing and that the containment dome monitor (HP-R-214) was in the alert and rising (Int. 20).

A review of multipoint recorder strip chart HP-UR-1901 indicates that the first area radiation monitor to increase was the incore instrument panel area monitor (HP-R-213) located on the 347-ft elevation of the reactor building (Ref. 115). At 0400, following the reactor trip, this monitor's reading decreased from 100 mR/hr to 1 mR/hr due to the rapid decay of the nitrogen-16 inventory in the reactor coolant system. At 0622, this monitor increased rapidly from 1 mR/hr to greater than the range of the instrument (10 R/hr) in 14 minutes. This may be indicative of clad failure and corresponding release of gap activity into the coolant system. The containment dome monitor (HP-R-214) responded in

a similar manner. The containment dome monitor increased rapidly from 1 mR/hr to 150 mR/hr. By 0652, the containment dome monitor had increased to 350 mR/hr. The first indications of increasing radiation levels recorded by the area radiation monitoring system in the Unit 2 auxiliary building were noted at 0638 by the makeup tank area monitor (HP-R-206). It increased from 0.7 mR/hr at 0636 to 100 mR/hr in 16 minutes. By 0700, this monitor was reading 1 R/hr. This monitor had been tagged out of service before the accident because it under-responded to radiation.

A review of multipoint recorder strip charts HP-UR-1907 and HP-UR-2900 indicated that at 0640 most of the auxiliary building and fuel handling building ventilation systems' monitors were indicating a rapid rise in airborne activity (Ref. 99, 117).

In the period 0645 to 0730, Radiation Protection Foreman B, several radiation/chemistry technicians and auxiliary operators were in the auxiliary building (Int. 21). These radiation/chemistry technicians were performing beta gamma general area surveys and collecting air samples. Radiation/Chemistry Technician J was at the Unit 2 health physics control point area counting air samples (Int. 51).

A review of pre-accident radiation surveys indicates that radiation levels were generally less than 1 mR/hr in all areas of the auxiliary building except in valve rooms and cubicles containing demineralizers, filters, and waste tanks (Ref. 179).

The results of the majority of these radiation surveys performed on March 28 after 0400 were not recorded. In some cases survey results were recorded but these records were not retained.

Table II-3-1 summarizes survey data from about 0700 to approximately 0730. The data are based on information obtained during interviews with several radiation/chemistry technicians and and Radiation Protection Foreman B (Int. 21, 42, 51, 78).

SUMMARY OF SURVEY DATA

<u>LOCATION</u>	<u>INSTRUMENT</u>	<u>TYPE OF RADIATION</u>	<u>RESULTS</u>
281 water on floor	Teletector	γ	0.2 mr/hr*
305 valve alley	R0-2	γ	100 mr/hr
305 door to makeup purification valve room	R0-2	γ	200 mr/hr
305 door to makeup purification valve room (re-survey)	R0-2	γ	5 R/hr
305 door to makeup tank room	R0-2	γ	1 R/hr
305 door to makeup tank room (re-survey)	E-520	γ	2 R/hr
305 hydrogen line just inside makeup tank room	Teletector	γ	10 R/hr
305 door to makeup tank room (re-survey)	R0-2	γ	5 R/hr at at 10 ft
305 Waste gas decay tank room	R0-2	γ	10-20 mR/hr
305 radwaste operating panel	R0-2	γ	50 mR/hr
305 HP-R-227	R0-2	γ	10 mR/hr
305 near containment personnel access hatch	Instrument unknown	γ	1-2 R/hr
328 near HP-R-219 a rapid series of measurements taken just prior to 0655	E-520	γ	10 mR/hr 50 mR/hr 75 mR/hr 175 mR/hr

*This survey was performed by Radiation Protection Foreman B who walked through the water on the floor without wearing protective clothing (Int. 21). He surveyed himself immediately on leaving the auxiliary building and did not find himself to be contaminated. Radiation Chemistry Technician I also toured this area shortly thereafter and reported water on the floor but no increase in radiation levels (Int. 42). He had worn a particulate respirator during his tour. He also was not contaminated.

Air samples were collected on the 305-ft elevation near the containment air monitor (HP-R-227) and on the 328-ft elevation near the stack monitor (HP-R-219) by Radiation/Chemistry Technician U at about 0650. These samples were given to Radiation/Chemistry Technician J for analysis. A record of the results was not maintained. Radiation/Chemistry Technicians O and U surveyed themselves after leaving the auxiliary building. Both found they were uniformly contaminated to levels of up to 30 mrad/hr. No record of this survey was maintained. While they were in the process of decontamination, the background radiation levels in the area of the Unit 2 health physics control point began to increase, causing the area to be evacuated to the hall between the control and auxiliary buildings. A step-off-pad was established in the hall (Int. 51, 78).

Air samples taken about 0700 near the entrance to the auxiliary building were remembered by Radiation/Chemistry Technician L to be in the range of E-8 $\mu\text{Ci}/\text{cc}$ of beta gamma particulate radioactivity (Int. 58).

At about this time, the auxiliary building and fuel handling building ventilation was secured. Radiation/Chemistry Technician U, at the health physics control point area, noted the ventilation had been secured and called the ECS to remind them to have someone close the model room door (Int. 78). When Unit 2 fuel handling building ventilation is off, Unit 1 fuel handling building ventilation will draw air from Unit 2 if the model room door is open. This appears to have occurred in this case since Unit 1 fuel handling building ventilation monitor (RM-A4) reached the gross particulate alarm set point of 1 E-8 $\mu\text{Ci}/\text{cc}$ at 0726 (Ref. 146). Ventilation of the Unit 2 auxiliary and fuel handling buildings was started at 0900 according to the notes of Control Room Operator J (Ref. 147). With ventilation secured, the airborne activity increased throughout the auxiliary and fuel handling buildings.

By 0730, a General Emergency had been declared, word had been passed via the paging system to evacuate the auxiliary buildings and the ECS had been established at the Unit 1 chemistry and health physics control points.

Health Physics Procedure 1670.8 "Emergency Re-entry for Repair or Rescue," Revision 1, dated January 16, 1978 and Procedures 1670.5 and 1670.2 provide discussion and guidance for in-plant radiological assessment and protective actions in a radiation emergency (Ref. 148). These procedures do not specifically establish what methods the licensee will use in controlling a sustained in-plant adverse radiological environment. The Supervisor, Radiation Protection and Chemistry stated that, in retrospect, the Emergency Plan Implementing Procedures do not adequately address an in-plant radiation emergency (Int. 25).

Section 12.3.2.10 of the FSAR states, "a radiation work permit (RWP) is required for all personnel working in radiation areas or posted RWP areas except in emergency situations where action is necessary for the protection of personnel or safety of the Unit. In such a case, the entry should be made by qualified personnel carrying radiation monitors. In the event such an entry is made, entry shall be documented."

The first Emergency Repair Party Team (ERT) action performed by the team located at the ECS was to close the model room door. The Repair Party Leader, Maintenance Foreman B, and two assistants were told by Radiation/ Chemistry Technician I that the exposure rate was 400 mR/hr in the work area (Int. 187). The Repair Party wore full face particulate respirators, appropriate dosimetry, and their street clothes. They were not accompanied by a Repair Party Team monitor and could not locate a radiation survey instrument so they took a count rate meter (RM-14/HP-210) with them.

This instrument is normally used for measuring contamination. The instrument deflected full scale prior to reaching the work location. The team continued, closing the door and exiting in about 10 minutes. On return to the ECS all were found to be contaminated. The individuals removed their clothing and were decontaminated by showering. Their entry was not under a RWP, it was not documented, no air activity measurements were made, no continuously indicating dose rate instrument was used in the high radiation area, and no record of the extent

or distribution of personnel contamination was maintained. The investigators review of emergency training provided to these individuals, as discussed in Section 1.2.1, indicates the training they had received was not consistent with that required by Procedure 1670.9. The performance of this Repair Party tends to substantiate the investigators evaluation of training.

In the time period from 0730 until 0807 (the time the model room door was closed) two independent surveys of the auxiliary building were performed (Ref. 149). One survey performed by Radiation Protection Foreman B was intended to determine the exposure rate on the roof near the stack (Int. 21). The measurements taken with an exposure rate meter (E-520, closed-window) read 4 mR/hr. The foreman descended from the roof through the auxiliary building. He did not wear any protective clothing. He attempted to return to ECS through the model room door, but found it closed. He did not perform any radiation measurement in the auxiliary building. On exiting at the newly established step-off-pad he was found to be uniformly contaminated to 20 mR/hr as measured with an exposure rate meter (E-520/HP-177) closed window probe. He removed his clothes and returned to Unit 1 for decontamination. No record of this entry or his skin contamination was made.

The other independent survey was performed by Radiation/Chemistry Technician I (Int. 6). On his own volition, he decided to make what he believed to be the first re-entry survey into the auxiliary building after evacuation. The protective measures he took included full protective clothing, high- and low- range pocket dosimeters, self-contained breathing apparatus, and a high-range exposure rate instrument (Teletector). He found the gamma radiation levels had increased to 1.5 to 2 R/hr on the 281-ft elevation. The general area exposure rates in the 305-ft elevation hallways were 1.5 to 2 R/hr, with 4 R/hr on the door to the makeup tank room and 1 to 2 R/hr near the reactor building air monitor (HP-R-227). He found the exposure rates the 328-ft elevation to be, on the average, less than 100 mR/hr. On exiting the area and removing the protective clothing, he surveyed himself and found that he was not contaminated. He made several notes of exposure rates but these notes were not maintained.

Based on the above examples, the initial entries into the auxiliary building to assess radiological conditions were not adequately planned. Preparations did not include consideration of appropriate protective clothing for entries nor did they include recognition of the need for surveys that could be used to plan future entries into the area. Air samples were not analyzed to identify isotopic constituents of the activity.

3.2.3 Implementation of Emergency Plan

The Radiation Protection Supervisor arrived on site about 0730 (Int. 22). He proceeded to the ECS and assumed control as called for in the Emergency Plan. His first efforts were to complete establishment of communications and assignment of survey teams. He repeatedly attempted to contact the Supervisor, Radiation Protection and Chemistry to determine the extent and nature of the emergency. He did not make this contact until the ECS was evacuated to the Unit 2 control room at 0910 (Ref. 150). In the period from 0745 to 0900, several samples were taken and analyzed to evaluate the extent of fuel degradation and primary system integrity.

The Radiation Protection Supervisor stated that the licensee did not have a "Chemistry Supervisor" and that this position was filled during the Emergency by Chemistry Foreman C (Int. 22). Interviews with chemistry foremen indicated that two of them, Chemistry Foremen A and C, assumed responsibility for filling the "Chemistry Supervisor" position described in the Emergency Plan (Int. 130, 132). A third chemistry foreman, Chemistry Foreman B, had arrived about 0700 and gone to the Unit 2 control room (Int. 23). There, he became aware of a possible primary to secondary leak from discussions with the shift foreman or control room operator. He proceeded to the Unit 2 secondary laboratory to position valves so that potentially contaminated samples from the steam generators (OTSG) could be collected in the Unit 1 primary sample room. Collection of OTSG samples was consistent with the requirements of Unit 2 Emergency Procedure 2202-2.6, "OTSG Rupture," Revision 4, dated October 6, 1978 (Ref. 151). Chemistry

Foreman B proceeded to the Unit 1 primary sample room and collected a liter sample from each steam generator. He surveyed each bottle with a count rate meter (RM-14/HP-210) and found that A OTSG sample caused the survey meter to peg full scale (50,000 cpm). He immediately informed the control room that the A OTSG showed high activity. Because of increasing air activity in the chemistry laboratory area, he did not perform any further analysis on those samples. However, sample records indicate that the B OTSG sample taken at 0724 was gamma scanned at 0745 and did not have indications of a primary-to-secondary leak.

The control room, when informed that A OTSG had a leak, told Chemistry Foreman B that they did not understand, since their indications were consistent with a leak in the B OTSG. Chemistry Foreman A had a preconceived notion that the sample points in the primary chemistry laboratory may have been labeled incorrectly. A confirmatory sample at the normal sample point in the Unit 2 secondary laboratory was taken at about 1000 by Chemistry Foremen A and B (Int. 23, 130).

The one liter sample from the B OTSG caused the survey meter (RM-14/HP-210) to rapidly peg full scale. The two chemistry foreman concluded that the sample lines in the primary sample room were in fact improperly labeled and that the B OTSG had developed the primary to secondary leak. This information was provided to Shift Foreman C. The samples were taken to the Unit 1 chemistry and health physics area (ECS) to be analyzed. The samples were not analyzed because of the high background radiation.

Chemistry Foreman A arrived on site at about 0800 (Int. 130). He proceeded to the ECS, mustered in, and reviewed what chemistry data was available at that time. He then went to the Unit 2 control room to discuss plant conditions with the Supervisor, Radiation Protection and Chemistry. Returning to Unit 1 ECS, he decided that a sample of the reactor coolant was necessary. He instructed two radiation/chemistry technicians to collect and analyze a reactor coolant

sample. Radiation/Chemistry Technician K, when told to collect the primary coolant sample, hesitated because of the information he had become aware of regarding the exposure rates (2 R/hr) measured on the primary sample lines (Int. 24). He was also concerned that collection of the sample would result in high airborne activity and radiation levels in the ECS since it adjoins the nuclear sample room. The spread of airborne activity from the nuclear sample room to other parts of Unit 1 control building, including the control room, had been observed in the past (Int. 11).

The Radiation Protection Supervisor (the ECS Director) was not informed that the chemistry foreman had directed a reactor coolant sample be taken (Int. 22). A review of the Emergency Organization as presented in Station Radiation Emergency Procedure 1670.2, "Site Emergency Procedure," Revision 9, dated November 22, 1978 does not specifically describe the need for the chemistry supervisor to coordinate his activities with the radiation protection foreman (Ref. 71). However, figure 4 of this procedure shows the chemistry supervisor operating under the direction of the Radiation Protection Supervisor.

Radiation/Chemistry Technicians K and I, told to collect the reactor coolant sample, did not take special precautions such as temporary shielding, use of shielded containers, remote handling tools, verification of ventilation, and special plans to dispose of liquid waste. Neither considered themselves as volunteers and neither discussed their planned actions with the Radiation Protection Supervisor. Both dressed in a complete set of protective clothing, including self-contained breathing device operated in the pressure demand mode, and a plastic rain suit. Both wore 0-200 mR, and 0-5 R pocket dosimeters and TLD badges. Neither wore extremity monitoring. No provisions for air samples were made (Int. 24, 42).

When reactor coolant sample recirculation flow was initiated at 0845 the hot machine shop area radiation monitor (RM-G4) increased from 200 mR/hr to 800 R/hr in a period of about 7 minutes. The nuclear sample room area radiation

monitor (RM-G3) increased from 300 mR/hr to 25 R/hr in about 7 minutes. The radiochemistry laboratory monitor (RM-G2) increased, starting a few minutes after RM-G4 and RM-G3, from 0.3 mR/hr to 8 mR/hr (Ref. 69).

One radiation/chemistry technician entered the nuclear sample room with a Teletector and measured an exposure rate of 200 R/hr at the sample hood. The second radiation/chemistry technician entered the room, opened the sample isolation valve for a few seconds and allowed about 50 ml to flow into the sample bottle. He estimated his total time in the sample room to be 10 seconds. A review of the Unit 1 effluent monitors, fuel handling building (RM-A4) and auxiliary building (RM-A5), indicated a rapid increase on all channels of these monitors (Ref. 152). The Unit 1 control room air monitor had been out of service since February 27, 1979. The technicians exited the nuclear sample room, leaving the sample in the hood (Int. 24, 42).

The Radiation Protection Supervisor at the ECS noted increasing radiation and levels of airborne activity in the ECS at this time (Int. 22). The hand and foot monitors and count rate survey meters (RM-14/HP-210) located in the ECS alarmed. Levels of radiation were recorded as 2 to 3 mR/hr at 0850 to 50 mR/hr at about 0900 in the ECS area (Ref. 153, Int. 22). The Radiation Protection Supervisor decided, based on increasing radiation and airborne activity levels ($9.36 \text{ E-11 } \mu\text{Ci/cc}$), to evacuate the ECS to Unit 2 control room. He was informed of the increasing levels on RM-G3 and RM-G4 but was not aware of the reactor coolant sample.

The reactor coolant sample recirculation flow was secured at 0900 (Ref. 150). This resulted in a decrease in radiation levels measured by monitor RM-G4 from 700 R/hr to 4 R/hr in about 10 minutes. The RM-A4 and RM-A5 monitors responded in a similar fashion, returning to their initial levels in about 30 minutes. This appears to indicate a gaseous release resulted from this sample.

The radiation/chemistry technician measured the 50 ml sample with a Teletector and found it read 200 R/hr at 6 inches (Int. 42, 78). On exiting the area, one radiation/chemistry technician checked his pocket dosimeter and noted a dose of 200 mR had been received in the 10-second period. This technician was not surprised that airborne activity was increasing at the ECS. Both technicians removed their protective clothing, performed a rough check for personnel contamination and helped evacuate the ECS to Unit 2 control room. At 0917, the ECS was established in the Unit 2 control room. At 0925 and 1004, the ECC Log indicates a request to preform an air sample in the Unit 1 control room (Ref. 153). No air sample results were recorded; but at 1015 the log indicates Unit 1 control room ventilation was put in air recirculation mode. In the absence of air sample data, the control room area monitor RM-G1 strip chart was reviewed and did not indicate an increase in radiation levels from 0700 to 1200 (Ref. 154).

As the ECS was being evacuated, Radiation/Chemistry Technician U and Radiation Protection Foreman C stayed behind to complete the reactor coolant sample (Int. 43, 78). The foreman stationed himself outside the ECS door to control access.

Radiation/Chemistry Technician U dressed in protective clothing with a full-face particulate respirator, wearing a high-and low-range pocket dosimeter and TLD, prepared to enter the sample room. He had been informed that the dose rates were in the 200 R/hr range. No extremity monitoring was worn and no special precaution such as sheilds or handling tools were used. No air samples were taken in the nuclear sample room (Int. 78).

The radiation/chemistry technician entered the sample room, removed 1 ml of sample and diluted it in a waiting volumetric held by another radiation/chemsity technician in the primary sample laboratory. This sample was analyzed on the gamma multichannel analyzer. He reentered the nuclear sample room and withdrew 5 ml of coolant for a boron analysis. He checked the exposure rate

with a Teletector and remembered it to read about 2 R/hr. Radiation/Chemistry Technician U recalls a pocket dosimeter exposure of 400 mR for the sum of both entries. He remembered the sample boron concentration was 248 ppm (Int. 78). The activity of this sample indicated the following (Ref. 155):

50 $\mu\text{Ci/ml}$ iodine-132
181 $\mu\text{Ci/ml}$ iodine-131
144 $\mu\text{Ci/ml}$ iodine-133
449 $\mu\text{Ci/ml}$ xenon-133
247 $\mu\text{Ci/ml}$ xenon-135
604 $\mu\text{Ci/ml}$ rubidium-88
29 $\mu\text{Ci/ml}$ krypton-85.

It is important to note that this reactor coolant sample was initiated and taken without the cognizance of the Radiation Protection Supervisor. The sample was not recirculated for the normal 30 to 45 minutes, and the sample line was not flushed for 15 or 20 seconds as is normally done to remove the stagnant volume of water in the sample leg. Radiation/Chemistry Technician K stated that the sample was not representative of water in the reactor coolant system because the recirculation and flush were inadequate (Int. 24) This conclusion has since been supported by evaluation of radiation data from monitors RM-G2, RM-G3, and RM-G4 (Ref. 69, 154). These monitor data indicate the total activity of the reactor coolant sample was probably several times higher than the reactor coolant sample collected on March 29 at 1615.

Failure to take special precaution resulted in unnecessary exposure to this individual.

Another important result of this sampling effort was its effect on the emergency response capability of the facility staff. This highly radioactive sample caused an increase in airborne activity and radiation levels that resulted in evacuation of the ECS area. It also resulted in the loss of analytical equipment that included:

- a. Both Ortec 401A beta counting systems
- b. Packard Model 3320 Liquid Scintillator, and
- c. Multi-channel Analyzer

Loss of this equipment left the licensee without isotopic identification capability until the arrival of the NRC Mobile Laboratory at 1930 on March 28 (Int. 73). Loss of this capability resulted in the inability to confirm the presence of iodine-131 in air and water samples inplant and to verify, in a timely manner, the results of environmental samples.

At about 0800 to 0830, Maintenance Foreman G was directed by operations supervision to enter the 305-ft elevation of the auxiliary building to bypass a fire defeat switch that would allow starting of the ventilation fans. This Repair Party entry was coordinated through the ECC rather than the ECS (Int. 143).

The foreman was briefed by the Supervisor, Radiation Protection and Chemistry as to the possible radiation hazards. He proceeded to the Unit 2 health physics control point, which was manned by radiation/chemistry technicians. He dressed in full protective clothing and wore a self-contained breathing apparatus, TLD, and pocket dosimeter. He entered the area carrying a Teletector survey instrument. Just inside the entrance doors he recalled an exposure rate of 150 mR/hr. The fire detection system panel is located just about 20 to 25 ft beyond these doors. He recalled an exposure rate of 20 mR/hr at the panel. The job took about 5 minutes. On exit of the area, he was surveyed and found to be free of personal contamination. He returned to the control room and debriefed with the Supervisor, Radiation Protection and Chemistry.

While the primary coolant sample was being collected, Radiation/Chemistry Technician T, at the request of Radiation Protection Foreman B, donned a full-face particulate respirator and proceeded to the 328-ft elevation of the auxiliary building to change the station vent stack monitor (HP-R-219) particulate and charcoal samples (Int. 75). He wore high- and low-range pocket dosimeters and a TLD badge and carried a Telectector. The entry took place about 0830 to 0900. He changed the charcoal cartridge; however, the particulate filter was missing and was not replaced with a new particulate filter since the filter housing was missing. The technician recalled exposure rates of 2 R/hr in the area of monitor HP-R-219. He stated that it took 45 seconds to change the samples and about 2 minutes total time in the building. His dose by pocket dosimeter was 450 mR, by TLD it was 395 mrem. This indicates exposure rates higher than 2 R/hr were probably present in the auxiliary building at the time of this entry. Results of the charcoal cartridge analysis have not been identified.

The ECS personnel, including the Radiation Protection Supervisor, arrived in the Unit 2 control room and established the alternate ECS off to one side of the operating area. This resulted in about 40 to 75 people in the control room (Int. 7, 22). The ECS group had brought what equipment they could from Unit 1. This included portable survey instruments, Ludlum scalers, and portable radios. The Supervisor, Radiation Protection and Chemistry briefed the Radiation Protection Supervisor.

At 0800, the control room ventilation system was placed in a recirculation mode (Ref 147). This means the ambient control room air is recirculated through particulate filters and charcoal adsorbers. Fresh air is supplied to the control room in this mode through particulate filters and charcoal adsorbers to make up for air leakage from the room. The control room air monitor (HP-R-220) samples this air supply before filters and adsorbers, and thus is not representative of actual control room atmosphere in the recirculation mode. In this condition, the actual air activity in the control room should be less than the concentration measured by HP-R-220 as long as the control room doors are infrequently opened. Entry into the control room by contaminated personnel and the leaving of doors open result in introduction of radioactivity into the control room.

The monitor HP-R-220 particulate channel alarmed at 0948 (Ref. 99). A review of strip chart HP-UR-1907 indicates the monitor HP-R-220 particulate channel increased from 40 cpm at 0930 to 30,000 cpm at 1040 (Ref. 99). The gas channel increased from 30 cpm to 200 cpm in the same time period. However, the iodine channel increased from 1 cpm at 0945 to 3000 cpm at 1130. The alert alarm for the iodine channel is set by procedure at 100,000 cpm (Ref. 49). Using the calibration data from the license procedure, the table below is an indication of incoming concentrations of airborne activity for the period described above (Ref. 49).

Unit 2 Control Room Incoming Air Activity

(0930-1130, calculated from HP-R-220)

Gross particulate	2.2 E-8 $\mu\text{Ci/cc}$
Gaseous	5.6 E-5 $\mu\text{Ci/cc}$
Iodine 131	7.3 E-12 $\mu\text{Ci/cc}$

At 1010, the gas channel of monitor HP-R-220 reached the high alarm setpoint (Ref. 99). A decision was made to move the ECS personnel to the Unit 1 control room. The Radiation Protection Supervisor considered his main duty the control of onsite and offsite survey teams and that this would be performed from Unit 1 control room (Int. 22).

Some radiation/chemistry technicians remained in the Unit 2 control room in support of the Supervisor, Radiation Protection and Chemistry. At 1017, all personnel in the Unit 2 control room donned respiratory protective devices equipped with particulate filters (Ref. 156). A sample taken in the control room at 1030 indicated 1 E-7 $\mu\text{Ci/ml}$ gross beta-gamma particulate activity. No isotopic identification was possible because of high background radiation in radiochemistry laboratories. Several additional air samples were taken in the control room during the next 6 hours that particulate filter masks were worn. These samples indicated initial gross particulate activities of up to 1 E-5 $\mu\text{Ci/cc}$. Half-life determinations were made that indicated short-lived isotopes (Int. 24).

The radiation/chemistry technicians and the Supervisor, Radiation Protection and Chemistry discussed these findings and reviewed the monitor HP-R-220 indications (Int. 24). Records of the survey results were not maintained.

The charcoal cartridge for monitor HP-R-220 was put in service on March 13, 1979 and removed April 4, 1979. Sample results indicate an average iodine-131 activity $1.14 \text{ E-}12 \text{ } \mu\text{Ci/cc}$.

The strip chart for HP-UR-1901 was reviewed and indicated that the control room gamma radiation levels, as measured by monitor HP-R-201 were generally less than 0.5 mR/hr for the first three days, except for a 1-hour period from 0030 to 0130 on March 29 when the maximum reading was 1 mR/hr (Ref. 115). The TLD badge located in the control room from February 28 to May 31, 1979 indicated a total dose of 25 mrem gamma and no beta (Ref. 157).

The licensee's decision to wear respiratory protection in the control room for 6 hours was conservative from a radiation protection point of view and consistent with station Health Physics Procedure 1616, "Use of Respiratory Protective Devices," Revision 9, dated March 20, 1978 (Ref. 158).

The Supervisor, Radiation Protection and Chemistry stated that he pursued the following general course of action in implementing his Emergency Plan responsibilities (Int. 133):

- a. Establish communications, get the monitoring teams into action, and maintain a feed-back of survey data,
- b. Support entries into the auxiliary building, and
- c. Provide input to the operations staff based on his understanding of the radiation monitoring system.

By 1100, the ECS was reestablished in the Unit 1 control room and a flow of information had commenced (Ref. 153). From the time of his arrival in the Unit 2 control room, the Supervisor, Radiation Protection and Chemistry was

communicating directly with the Emergency Director, supervisor of operations, shift supervisors, and control room operators on matters relating to interpretation of radiation monitoring systems and data.

The Supervisor, Radiation Protection and Chemistry stated his plan for controlling the in-plant radiation hazard was to attempt to personally brief each individual as to the radiological conditions expected in the area, review ways to minimize exposure, and provide general precautions or recommendations.

During the period of the accident up to 2400 March 30, the Supervisor, Radiation Protection and Chemistry stated that no entries were made into areas for the protection of vital equipment that might have warranted an emergency dose. The planned dose commitment per entry was not expected to exceed 1.5 rem.

Because of increasing air activity, the radiation/chemistry technicians at the step-off-pad access to the auxiliary building evacuated the area and relocated to the Unit 2 control room about 1100 (Int. 51). They set up a count rate meter with (RM-14/HP-210), Ludlum beta scaler, and a couple of E-520s and RO-2s monitors on a table in the control room.

Evacuation of the technicians left no positive control over entrance to the auxiliary building.

About this same time all nonessential personnel were evacuated from the site. Radiation Protection Foreman A and several technicians proceeded to the north auditorium and surveyed the individuals assembled there for contamination prior to release (Int. 7, 51). Other individuals, including Chemistry Foreman B surveyed vehicles at the gates as they left (Int. 23). At about 1100, word was passed to have all vehicles and personnel proceed to the 500 kV station for survey and release. This assembly area was established at the direction of Radiation Protection Foreman D and was used to monitor and decontaminate

personnel (Int. 18). This represents a departure from the Emergency Plan since this location differs from the two "washdown areas" designated in implementing Procedure 1670.6.

3.2.4 In-plant Radiation Protection

This section presents a review of radiation protection, including examples, during the period 1100 March 28 to 2400 March 30. Information presented is based on interviews with each member of the radiation protection staff.

During this period, the RWP procedure was not used (Ref. 159). No log of each entry into the auxiliary building was made. No temporary pocket dosimeter dose accumulation system was put into effect for entries into high radiation areas such as the auxiliary building (Int. 133). Records of dose rate surveys in the auxiliary building were made at times, but these records were not maintained (Int. 133). Surveys of personnel contamination were made, but in only one instance were records maintained (Int. 23). High radiation areas were not controlled in accordance with Technical Specification 6.12.

Station Radiation Emergency Procedure 1670.2, "Site Emergency Procedure", Revision 9, dated November 22, 1978 defines Emergency Repair Party duties as follows:

"4.4.2.3 Specific duties of the ERP will depend upon the nature of the emergency. However, the goal of the ERP is to correct any malfunction and return the plant to normal operating condition as soon as possible."

Entries into the auxiliary building to operate valves, pumps, breakers; to perform inspections and maintenance; and to troubleshoot or repair are considered to have been Repair Party activities. In the absence of documentation, several of the entries will be discussed; however, the investigators may not be aware of all entries into the auxiliary building.

3.2.4.1 About 1100 March 28 Shift Foreman A entered the 305-ft elevation of the auxiliary building to close electrical breakers that permit operation of the decay heat suction valves (Int. 107). The entry was made alone. Protective clothing and a pocket dosimeter were not worn. This individual carried a survey meter, which he remembered indicated exposure rates in his work area of 2 R/hr. He stated that his entry took about 5 minutes, and on leaving the building he was found to be contaminated. Review of the individual's TLD results indicate that a dose of 550 mrem gamma and 660 mrem beta may have resulted from this entry (Ref. 160). No records of the survey of his personnel contamination were maintained.

3.2.4.2 Between 1100-1600 March 28 the Supervisor, Radiation Protection and Chemistry told Radiation/Chemistry Technician U to escort two individuals into the auxiliary building to check valves and close the containment building spray pump breakers (Int. 78). The two individuals were not shift maintenance workers, and neither had received emergency Repair Party Team Training as required by Procedure 1670.9.

They dressed in full protective clothing, which included cloth coveralls, paper coveralls, plastic and rubber boots, gloves, hood, and rainsuit. They wore self-contained breathing devices, TLDs, and 0-200 mR pocket dosimeters. They did not wear high range pocket dosimeters. The entry was not made under an RWP nor was it logged. The radiation/chemistry technician carried a Teletector.

The three individuals entered the auxiliary building on the 305-ft elevation where a survey of the seal injection filters was made. They spent about 10 minutes attempting to locate the decay heat valves of concern. Not finding these valves, they proceeded to the 328-ft elevation via the stairway near the elevator. Exposure rates increased rapidly as they entered the 328-ft elevation. The radiation/chemistry technician terminated the entry when the exposure rate reached 100 R/hr. The radiation/chemistry technician did not document his survey results, but remembered the following exposure rates:

<u>Area</u>	<u>Elevation (ft)</u>	<u>Exposure Rate</u>	<u>Preaccident Exposure Rate (mR/hr) Ref. 179</u>
Seal injection filters	281	50-75 R/hr at 1 ft	<1
Hallways	281	2 R/hr	<1
Decay heat room	281	50 mR/hr	<1
Top of stairwell by elevator just before door	281	1 R/hr	<1
Just inside door from well by elevator	328	5 R/hr	<1
Near fuel handling building supply unit	328	100 R/hr	<1
Near auxiliary building supply unit	328	>1000 R/hr	<1

On exit from the area, the individuals noted that their pocket dosimeters were off scale. Two individuals returned to Unit 2 control room, the radiation/chemistry technician went to Unit 1.

A review of TLD data indicates 560 mrem and 440 mrem gamma for the two individuals and about 1.8 rem gamma for the radiation/chemistry technician.

About 1100, USNRC Inspector H arrived in the Unit 2 control room (Int. 40). He was present in the control room until about 1800. In an interview with the investigator, he described conditions during this period. His description is summarized below:

- a. The control room was very busy.
- b. The door to the control room was continuously being left open.
- c. People in the control room were wearing protective clothing.

- d. The auxiliary building had high airborne and radiation levels.
- e. No positive control was established over entry into the auxiliary building.
- f. RWPs were not being used.
- g. An auxiliary building entry log was not being maintained.
- h. Survey results were not being documented.
- i. The Supervisor, Radiation Protection and Chemistry was involved with operations in evaluating valve lineups and their effect.
- j. The control room was not equipped with an operational continuous airborne activity monitor.
- k. The control room appeared to be part of the controlled area from a radiological point of view.
- l. Respiratory protective devices hindered communication.
- m. Exposure rates in the control room were measured as high as 10 mR/hr, with beta-gamma survey instruments.

3.2.4.3. Shift Supervisor C and Radiation/Chemistry Technician J entered the auxiliary building at about 1300 to transfer radioactive water (Int. 13, 51). Both dressed in full protective clothing and wore self-contained breathing

devices, high-range dosimetry, and TLDs. Upon entering the auxiliary building, their radiation survey meter (RO-2) pegged full scale (5 R/hr). Both withdrew to the health physics area and waited until a Teletector was available. When the Teletector arrived from Unit 1, the two individuals reentered the 305-ft elevation and proceeded to the radwaste operating panel. The general area dose rates were 8-10 R/hr. The shift supervisor was unsuccessful in starting pumps to transfer water. On exiting the area, a survey revealed both were contaminated. No records of this survey were maintained. Their high range pocket dosimeters indicated about 1 R of exposure.

3.2.4.4. In the afternoon on March 28, Radiation/Chemistry Technician H and an electrician entered the 328-ft elevation of the auxiliary building to make an electrical breaker lineup (Int. 49). No RWP, no log of entry, and no survey data were reviewed prior to the entry. They had difficulty locating self-contained breathing devices and could not find rainsuit hoods. They entered the building and proceeded to the 328-ft level to make the breaker alignment. The following survey data was taken with a Teletector and recalled by the technician:

<u>Area</u>	<u>Elevation (ft)</u>	<u>Exposure Rate (R/hr)</u>	<u>Preaccident Exposure Rate (mR/hr)</u>
Entrance to auxiliary building	305	5-7 R/hr	<1
Bottom of west stairs to 328-ft level	305	10 R/hr	<1
Top of west stairs	328	20 R/hr	<1
Motor control centers	328	50-100 R/hr	<1
Motor control centers average	328	20 R/hr	<1

They did not finish the job because they ran out of air. They left the building and returned to the Unit 1 decontamination area. The radiation/chemistry technician stated that he was so contaminated that he pegged the RM-14/HP-210

monitor (50,000 cpm) at a distance of 3 feet. He measured 35 mR/hr with an E-520 on contact with his head. Both men showered, but the radiation/chemistry technician had residual contamination of 1500 cpm on his hair. No dose rate or contamination survey records were made.

The electrician received a gamma dose of 1.33 rem, the Radiation/Chemistry Technician a dose of 1.76 rem by TLD data.

3.2.4.5. On the afternoon of March 29, Radiation/Chemistry Technician Q was instructed by a radiation protection foreman to lead a Repair Party Team consisting of about four auxiliary operators (Int. 142). According to the Emergency Plan, auxiliary operators are not designated as Repair Party Team members. This team was to place plastic sheeting over the water on the floors in the auxiliary building on the 281-ft elevation. Prior to entry, they did not have any radiation survey results to review. However, they were told that exposure rates were 10 R/hr near the sump drain and that 6 inches of highly contaminated water was on the floor. The team dressed in full protective clothing, including self-contained breathing devices operating in the demand mode. They wore high- and low-range pocket dosimeters and TLD badges. No extremity monitoring was worn on their feet or ankles.

They entered the 281-ft elevation by the elevator and found exposure rates of 2 R/hr and 10 R/hr near the sump drain. The floor had spots of standing water, some up to 6 inches deep. They covered most of the floor, using three to four, 80-pound rolls of plastic. The job took about 20 to 30 minutes.

No air samples were taken in the area during this work. On exit from the area and removal of protective clothing, no one was found to be contaminated. The radiation/chemistry technician did not log his pocket dosimeter reading for this entry. He recalls receiving a dose of about 1 rem, and stated that his TLD also indicated a dose of about 1 rem.

Review of this individual's TLD data for the period of March 1 to March 29 indicates a dose of 0.44 rem. The individual stated that all team members had received a dose of about 1 rem. Review of one known team member's TLD data indicates a dose for the period March 29 to March 30 of 1.5 rem gamma and 2.42 rem beta.

3.2.4.6 On the evening of March 28, two radiation/chemistry technicians decided, based on Unit 1 control room conversation, to enter the auxiliary building and pull the HP-R-219 charcoal and particulate samples for analysis and to collect a water sample from the floor of the 281-ft elevation of the auxiliary building (Int. 49 and 50). Radiation/chemistry technicians E and W were under the impression that the NRC wanted the samples. They announced to the Radiation Protection Supervisor that they would collect the samples. He was in agreement. They planned the job for about 1 hour, deciding to carry a lead shield with which to wrap the HP-R-219 cartridge.

They were unaware of previous entries into the building and did not review any survey information. They stated that at the time of their entry into the auxiliary building there was no control over entries, either in the form of a person or locked door to restricted access to the building. Prior to the entry, they went to the Unit 2 control room but could not locate the Supervisor, Radiation Protection and Chemistry or his alternate. They entered the area dressed in full protective clothing with the exception of rainsuit hoods, which they could not find. Neither wore extremity monitoring. One technician carried a Teletector and the other an E-520. They did not record their survey results; but recalled them as follows:

<u>Area</u>	<u>Elevation (ft)</u>	<u>Exposure Dose Rate (R/hr)</u>	<u>Preaccident Exposure Rate (mR/hr)</u>
HP Lab by hand and foot monitor	305	2	<1
Hallways	305	Up to 15	<1
On floor near HPR219	328	50-70	<1

Just inside the auxiliary building doors, the technicians split up. The one with the Teletector went to the 328-ft elevation to sample HP-R-219. The other technician's E-520 pegged full scale (2 R/hr) as he passed near HP-R-227 on the 305-ft elevation. He did not stop, and the meter remained off scale as he went down to the 328-ft elevation to scoop a water sample from the floor.

While on the 281-ft elevation, he found a Teletector. He checked the batteries to see if it worked but did not take an exposure rate measurement. He scooped a 100 ml sample and left the building.

They returned to the Unit 1 health physics area. An exposure rate on the bottle read 40 mR/hr at contact. A dose rate on one technician's head read 100 mR/hr. Both were contaminated to about the same extent on their heads and stated they were able to decontaminate themselves. No survey records were made.

Failure to leave the area when the individual's dose rate instrument pegged full scale represents an unsafe radiation protection practice, which can result in unnecessary exposure.

Radiation/Chemistry Technician E estimated his total time to change the charcoal cartridge (the particulate filter was still missing) was between 3.5 and 5.5 minutes (Int. 50). He stated his TLD indicated a dose of 870 mrem. The other technician stated his TLD read 720 mrem.

At 1930 March 28, the NRC Mobile Laboratory arrived at the Observation Center (Int. 72). Two in-plant air samples were analyzed for iodine activity between 1930 and 2400. These charcoal cartridges were counted with a Ge(Li) detector. Neither indicated above the minimum detectable activity for iodine-131 ($2 \text{ E-}8 \text{ } \mu\text{Ci/cc}$).

3.2.4.7. At about 2100 March 28, Auxiliary Operator I was directed by a Shift Supervisor B to repressurize the core flood tanks with nitrogen (Int. 114). He understood this action to be necessary but was not told it was for

the protection of vital equipment. No one told him how much dose was permitted for this entry. He stated no surveys were available for review, so he listened to radiation protection personnel discussing dose rates and concluded he might be entering fields of 100 to 150 R/hr. He did not specifically tell the radiation protection personnel he was going into the building and they did not ask. He did not have an RWP and an entry log was not being used.

The auxiliary operator dressed himself in protective clothing but did not wear a rainsuit. He did not have a high-range pocket dosimeter. He donned a self-contained breathing apparatus at the step-off-pad outside the entrance to the auxiliary building, picked up a Teletector, and entered the building. There was no one there to challenge his entry and no locked access to restrict entry into the building at that time.

He entered on the 305-ft elevation, noting an exposure rate of 20 R/hr, proceeded upstairs to the 328-ft elevation, then back through the fuel handling building to near the penetration room door where the core flood nitrogen valves are located. He recalls dose rates of 70, 100, 30, and finally 10 R/hr in the valve area. He estimates it took him between 5 and 10 minutes for the total job.

On leaving the building, he found his 0-200 mR pocket dosimeter was off scale. He returned to Unit 2 control room and, on entering, alarmed an RM-14/HP-210 count rate meter on the table. Assuming his clothing to be contaminated with gaseous activity, he decided to return to the auxiliary building and attempt to transfer water to Unit 1.

A radiation/chemistry technician observed Auxiliary Operator I alarm the RM-14/HP-210 probe and told him he would have to go to Unit 1 and decontaminate. This comment reaffirmed his decision to reenter the auxiliary building at that time before decontaminating himself since he thought he would become contaminated again. He told one shift foreman that his pocket dosimeter had gone off scale. Subsequently, he planned the liquid radwaste transfer with Shift Supervisor D but did not inform him of his previous exposure.

Shift Supervisor D was concerned about exposure for the transfer and discussed possible low background areas near the radwaste panel. Radiation protection personnel did not participate in these discussions.

The auxiliary operator searched out protective clothing but was unable to locate a high-range dosimeter either in Unit 1 or 2. He asked Radiation/ Chemistry Technician K for one and discussed his planned entry into the building.

At this point he rezeroed a 0-200 mR dosimeter, dressed in the same manner, and once again entered the auxiliary building unchallenged. This time he walked the 305-ft elevation through the switch gear room, hoping the exposure rates would be lowered. He remembered 30 R/hr as he passed the makeup tank room and 10 R/hr at the radwaste panel. The pump would not start so he called the control room to reset a trip. He then waited in the model room (3 R/hr) for 2 or 3 minutes before attempting to again start the pump. The pumps still did not energize, so he checked the breakers, found they were tripped, and left the building. He estimates about 10 minutes for this entry.

On exiting the area, he discovered his pocket dosimeter was again off scale. He undressed and returned to the Unit 2 control room. He again alarmed the RM-14/HP-210 probe. He informed the Shift Supervisor D that his dosimeter had gone off scale. The shift supervisor told him he could not be used any more and to return to Unit 1 and decontaminate. The auxiliary operator was about to shower when a radiation chemistry technician told him that, because of high airborne activity, he should wear a mask while showering. The technician did not survey the operator.

After decontamination, the operator's TLD was read, indicating a gamma dose of 3.170 rem. His Form NRC-5 equivalent indicated a total first quarter gamma dose of 3.87 rem. The licensee submitted a report pursuant to 10 CFR 20.405, dated May 1, 1979, indicating that this individual had received a first quarter whole body dose of 3.870 rem.

This entry exemplifies the lack of good health physics practices.

3.2.4.8. At about 2200 on March 28, Shift Foreman B requested Auxiliary Operator C to open the auxiliary spray valve DH-V-187 located on the 328-ft elevation of the auxiliary building in an elevated pipe run between the fuel handling and reactor buildings (Int. 104). In order to operate the valve, one must start from a platform and climb up a hand rail onto a pipe run which is located over an open space that drops to the 281-ft elevation. Based on interviews, it appears that this area was not surveyed prior to this entry. The following quotation is provided to describe execution of this Repair Party function:

"Then the foreman came up to me... and when we go into decay heat... we have an aux spray line we can set up from decay heat for the pressurizer. So my job then was to establish the lineup for auxiliary spray in the pressurizer....So I went down to our HP area and there was no Telectectors there at that time. There was E-520s but they were all pegged so they were useless. So I went over to Unit 1 HP and it was contaminated. There was no Teletector there. There was nothing there I could use. So, on the way back, I just made up my mind that I was not going to use a Teletector. I knew where I was going and I knew which valve I had to open. It was a locked valve. I had no intentions of entering any rooms; so I thought it'd be -- I had no idea at that time what the radiation levels were. As I got there again, an HP fellow came out of the aux building. He told me the radiation levels in the hallways were 100 R/hr, and I asked him at that time for his Teletector. But he still needed it and could not surrender it. So I just told myself; time, distance, and shielding. In other words, bag it through the hallways, and once I got up in the area of the valve.... it was pretty well shielded. I thought I'd be all right. Then the game was getting Scott Air Pacs. It was quite a few of them being used at that time, so I used the bottle that was half full. There was no

one there to assist us; no one there to help you. You were pretty well on your ownand I got myself dressed with what we had there and went in. I climbed up to the valve. I made the mistake of running through the hallway, though. And I ended up hyperventilating and suffering from claustrophobia as a result. But I got that under control after a bit and then got up to the valve.... got it open. My alarm is ringing. And then I left. Then I ran out of air, so I had to remove the mask just, oh, maybe 25 to 30 feet before the door where you get out. So I got out of there and got undressed. And everything was contaminated. There was no clean area that you could use. So I went upstairs and the same thing with me... as soon as I opened the door the monitor they had in there... the frisker in there went off. So I told them I had the valve open and left."

This individual wore a TLD and 0-200 mR pocket dosimeter. He had unchallenged access to the auxiliary building. He did not inform anyone as to the time he was entering the building.

Upon returning to the Unit 2 control room and opening the door, the contamination on him immediately caused the RM-14/HP-210 to peg full-scale. He proceeded to Unit 1 and showered. He could not estimate his contamination because the RM-14/ HP-210 there was alarming. He returned to the Unit 2 control room and found he was still contaminated but that it did not exceed 5000 cpm on the RM-14/ HP-210. He then told the shift foreman that he would not be of any further use, since he was "pretty sure he was overexposed." His TLD badge was read and he remembers a dose of 1.25 rem. A review of his TLD data sheet is not consistent with his recollection of this entry and needs to be resolved by the licensee. The TLD data printout of his exposure for the period of March 29 to March 30 indicates a dose of 0.655 rem. The licensee was informed of this finding.

This individual had not been designated as a Repair Party Team member and therefore did not receive Repair Party Team training. He had received "Advanced Radiation Training" according to Procedure 1690. His entry demonstrates a lack of implementation of basic radiation protection practices that could have resulted in serious consequences.

3.2.4.9. On the morning of March 29, a radiation chemistry technician made a survey of the auxiliary building. He prepared a survey map of his findings. These records were not maintained; however, the table below summarizes the Teletector readings he recalled.

<u>Area</u>	<u>Elevation (ft)</u>	<u>Exposure Rate</u>
Water on floor	281	50-60 mR/hr
Bleed tank door (closed)	281	35-40 R/hr
Bleed door (open)	281	300 R/hr
Seal injection filters	281	600 R/hr
B makeup pump room just inside door	281	100 R/hr
B spray vault	281	100 R/hr
A bleed tank (inside door)	281	200-300 R/hr
Streams (4 or 5)	281	1000 R/hr
General hallways	281	300-400 mR/hr
Makeup tank door	305	40-50 R/hr
Streamers (door to makeup and purification valve room)	305	1000 R/hr
Outside valve alley	305	15 R/hr
Door to makeup purification room	305	50-70 R/hr

This radiation/chemistry technician received a dose of 1.2 rem by TLD as a result of surveys he performed on the first 2 days.

On the afternoon of March 29, additional supplies of self-contained breathing devices, face masks, and combination particulate and charcoal cartridges began arriving on site (Int. 155). Radiation protection support personnel were arriving with their own instrumentation at the Observation Center (Int. 54). With few exceptions, these individuals were not used in-plant until after March 30.

A consultant's mobile laboratory arrived at the Observation Center about 1400 and began setting up. The mobile laboratory did not begin counting samples until 0030 on March 30, 1979. A review of all air samples counted during the period 0400 March 28 to 2400 March 30 by NRC and vendor mobile laboratories indicates the highest iodine-131 concentration to have been $5 \text{ E-}9 \text{ } \mu\text{Ci/cc}$ at 2203 on March 30 in the primary chemistry laboratory. A total of 48 air samples were gamma scanned (33 from both Unit 1 and 2 control rooms). The highest concentration in either control room was reported as $3 \text{ E-}10 \text{ } \mu\text{Ci/cc}$ in Unit 1 at 1255 on March 30. The remaining samples were taken in the control, service, and turbine buildings. The highest level of iodine-131 activity measured was $1 \text{ E-}9 \text{ } \mu\text{Ci/cc}$ in the Unit 2 access control area.

During the period 0800 March 28 through the afternoon of March 29, TLD badges were read on an as requested basis on site at the normal location. The TLDs were read by any radiation/chemistry technician or radiation protection foreman that was available at the time. In some instances, radiation/chemistry technicians acted as Repair Party monitors (e.g., see paragraph 3.2.4.5) and then read the TLDs worn by themselves and the other team members.

Procedure 1642 states that a TLD reader calibration test should be done on or near each badge exchange or reading period (Ref. 161, 181, 182). This test was not performed prior to reading TLD badges during the incident. The last recorded test was performed on February 28, 1979. That test indicates acceptable results.

Sometime on the afternoon March 29, the TLD reader and support equipment was moved from its normal location in a trailer on site to the Observation Center. The equipment was placed into operation and some TLD badges were read. No records indicate that a reader calibration test was performed.

The TLD system was operated until about 0700 March 31 by Radiation/Chemistry Technician C who had received two hours of on-the-job training on June 6, 1977, and had not operated the equipment in about a year and a half (Int. 82). He did not have a copy of the procedure for either operation or documentation of TLD results available. He performed this job for about forty-eight continuous hours without sleep.

His work included zeroing all the April TLD badges that had been stored on site in the TLD trailer and reading TLD badges as they were turned in. He indicated that he had little or no help through this period.

Review of the licensee's TLD exposure data (Form NRC 5 equivalents) dated May 21 and July 8, 1979 for 200 individuals indicated several errors. Of the 200 individuals, ten badged employees did not have entries in their dose records for the period March 1 - 31, 1979. Five of the 10 individuals would have received dose during this period by virtue of their job functions. In addition, doses recorded for four individuals appeared inconsistent with information provided in interviews (Int. 35, 49, 104, 114, Ref. 160). The review also indicated that changes to individuals' dose records were made, but no record of the required evaluation leading to the changes were maintained (Ref. 163, 164). The licensee was informed of these records discrepancies.

The licensee's method of evaluating skin dose during the period March 28 to 30 appears in error. No dose contribution from gamma radiation was indicated on the TLD data sheets provided to the NRC on May 21, 1979 (Ref. 163, 164). The licensee dosimetrist was informed of this finding. Corrections are being made to individuals' dose records as errors are identified.

Using the licensee's TLD data, an investigator estimated a total dose of 104 man rem for 270 individuals monitored during the period March 28 through March 31, 1979.

During the period of this investigation, numerous individuals in the auxiliary building were exposed to high concentrations of noble gas activity. On several occasions, the whole body exposure rate as measured by Teletector was remembered to be 100 R/hr. The majority of this radiation was probably due to xenon-133 and xenon-135. Review of data from individuals' TLDs and background TLD badges indicated several apparent inconsistencies.

One such inconsistency is discussed in Section 3.2.4.5 of this report. In that example, two individuals who were similarly exposed had different dosimetry results. According to TLD results one individual received 0.44 rem gamma while the other received 1.5 rem gamma and 2.42 rem beta dose. Since no records were maintained of how individuals were dressed or where their TLDs were worn, this investigation cannot accurately explain this inconsistency.

In other instances, some background TLD badges measured beta radiation while others, exposed in similar surroundings, measured none. Orientation of these badges in relation to the source of radiation is unknown.

A possible source of measurement error applicable to all cases involves loading of the TLD chips into the holder. When loaded into the TLD badge, the chips can easily be positioned such that the capability to measure beta radiation is lost and the response to low energy gamma radiation is diminished. This occurs when the TLD card is placed in the badges with chip G2 in front of the beta window or when the card is placed such that both chips are shielded by the plastic badge case.

An investigator discussed these problems with the Supervisor, Radiation Protection and Chemistry, who stated additional evaluations are in progress in an effort to resolve dosimetry questions.

3.2.4.10. The Supervisor, Radiation Protection and Chemistry instructed Radiation Protection Foreman D to take a reactor coolant sample on the afternoon of March 29 (Int. 18). Radiation Protection Foreman D and Chemistry Foreman B decided they would take the sample (Int. 23). They were aware of the need to determine the boron concentration of the coolant. They were not informed of any other specific analysis to be performed. They discussed the assignment for a period of less than one hour, deciding to use a three-man approach and temporary shielding to minimize their exposure. Chemistry Foreman C was recruited to perform the boron analysis (Int. 132).

The individuals dressed in protective clothing, which included several pair of coveralls, shoe covers, and three pair of gloves. One individual wore a rain suit. Two wore full face respirators with iodine adsorbing cartridges. The other wore a full face mask with particulate filter. The chemistry foreman's TLD and pocket dosimeters were worn inside the protective clothing. The high-range pocket dosimeter was not accessible to be periodically checked. No one wore extremity monitoring on their hands. The radiation protection foreman wore a high-range pocket dosimeter taped to his forearm. This was knocked off during collection of the sample. Prior to taking the sample, they built a lead shield in the primary sample lab and in the nuclear sampling room.

Prior to recirculation of the sample, the room was surveyed with a Teletector and found to have an average exposure rate of 8 R/hr. RM-G3, the nuclear sample room area monitor, indicated 1 R/hr at this time.

Just prior to initiation of recirculation flow, Chemistry Foreman B was asked to collect a Unit 1 reactor coolant sample for boron. He collected this sample but did not check his high-range pocket dosimeter prior to starting on the Unit 2 Sample.

The valve lineup was made for Unit 2. When flow was initiated at about 1545, RM-G4 the hot machine shop area monitor increased from 850 mR/hr to 2 R/hr. The RM-G3 monitor increased from 1 R/hr to 6 R/hr (Ref. 154). The sample was allowed to recirculate for about 45 minutes. Several times in that period

the "drag" valve had to be adjusted in order to maintain flow. The radiation protection foreman entered the nuclear sample room and attempted to collect a sample. When he opened the sample valve he observed an "alka seltzer-like" fluid. He closed the valve and checked the dose rate near the sample line and found it read 90 R/hr.

They reviewed the valve lineup and asked the control room to cycle the valves. They reentered the nuclear sample room and collected about 300 ml of sample in a one liter bottle. The one-liter sample bottle was handled unshielded at contact with their gloved hands. One hundred ml was poured into a graduated cylinder and the exposure rate measured. At contact, the graduated cylinder caused the Teletector to deflect full scale (1000 R/hr). The graduated cylinder was set on the floor and the exposure rate measured 400 R/hr at 1 foot and 10 to 15 R/hr at 3 feet. The graduated cylinder was returned to the sample hood. Meanwhile, Chemistry Foreman B had drawn an additional 40 ml of reactor coolant into a 100-ml breaker and carried it into the primary sample lab. The primary sample laboratory area radiation monitor (RM-G2), increased from 20 mR/hr to 800 mR/hr as he entered the room. The sample was placed behind a lead shield and pH adjusted by addition of hydrochloric acid. Five mls were given to Chemistry Foreman C for the boron analysis. Chemistry Foreman B returned to the nuclear sample room and poured the 100 ml sample back into the sample bottle. The radiation protection foreman took the 40-ml sample from the primary sample laboratory and dumped it down the nuclear sample room sink.

Radiation Protection Foreman D and Chemistry Foreman B removed their protective clothing. Each checked their high-range pocket dosimeters. The radiation protection foreman's read about 0.8 R, the chemistry foreman's read about 4 R. They left for the process center to be surveyed. On arrival, both alarmed the RM-14/HP-210. Someone surveyed them with an E-520 and found the radiation protection foreman's wrist read 200 to 250 mR/hr and the chemistry foreman's finger read 50 mR/hr at contact. Other areas of their bodies were also contaminated.

Both returned to Unit 1 to decontaminate. Chemistry Foreman C, after completing the boron analysis, concluded that he was contaminated. He immediately began decontamination in the shower prior to the return of the radiation protection foreman.

These three individuals attempted decontamination for the next several hours. They were periodically resurveyed at the 500 kV substation. All three went home with areas of their body having maximum fixed contamination of up to 50 mR/hr on a finger and 30 mR/hr on wrists. Each individual supplied urine samples the next day which were not analyzed by the licensee as of May 24, 1979.

The licensee submitted a report pursuant to 10 CFR 20.405 dated May 1, 1979 indicating that Chemistry Foreman B had received a whole body gamma dose of 4.115 rem and beta dose of 0.090 rem for the first quarter of 1979.

The NRC has evaluated the dose to the extremity and skin of each individual involved in this sampling effort. This evaluation is attached to this report as Appendix B. One chemistry foreman received an extremity exposure in the range of 50-147 rem and a skin dose of approximately 6 to 13 rem for the first quarter of 1979 and a skin dose of 4 to 19 rem for the second quarter of 1979. Radiation Protection Foreman D received an extremity dose in the range of 44 to 54 rem for the first quarter of 1979.

The following reactor coolant activity results were reported to the licensee (Ref. 165). The results were developed from an undiluted sample by a vendor laboratory.

iodine-131	1.3 E 4	μCi/cc
iodine-133	6.5 E 3	μCi/cc
strontium-89/90	5.3	μCi/cc
cesium-134	6.3 E 1	μCi/cc
cesium-136	1.8 E 2	μCi/cc
cesium-137	2.8 E 2	μCi/cc
barium-140	2.0 E 2	μCi/cc

In addition, the NRC reviewed the exposure to other personnel who handled this sample. The licensee was informed that several exposure evaluations were necessary. On the night of March 29 and the early morning of March 30, Radiation Protection Foreman B and Radiation/Chemistry Technician L removed a 1 ml sample for shipment to a vendor laboratory (Int. 21, 58). Radiation/Chemistry Technicians B and V diluted a 1-ml sample reading 11 R/h with a Teletector into 1 liter of demineralized water and found it read 1 R/hr (Int. 98, 106). One ml of this sample was diluted in 1 liter of demineralized water and that read 50-100 mR/hr. One ml of that sample diluted into 100 ml of demineralized water read 1 mR/hr. Five samples of this diluted coolant were packaged in vials for analysis on site (Int. 98).

During this operation no air samples were taken, no extremity monitoring was worn, no records of surveys were maintained, nor were records of personnel contamination made.

A review of the TLD data report issued July 5, 1979 does not indicate any dose results for the period March 1 through 31, 1979 for Radiation Chemistry Technician L who handled the sample.

3.2.4.11. The evening of March 29, Nuclear Engineers J and E, discussed with the Supervisor, Station Operations the importance of finding out if there were any major leaks in the auxiliary building (Int. 33, 35). They decided to tour the area since both were familiar with radwaste operations. They were briefed by Radiation Protection Foreman B, dressed appropriately for the degree of hazard, and were provided an E-520 and high-range gamma instrument.

Shortly after entering the 305-ft elevation of the building, the high-range gamma instrument failed. They continued on together noting that the E-520 frequently pegged full scale (2 R/hr). They operated a valve on the 305-ft elevation and then toured the 281-ft elevation. Again, the meter frequently pegged. On the 281-ft elevation, they split up, Engineer E going back up to the radwaste panel, Engineer J touring near the south access corridor (a previously unsurveyed area) and the bleed tank doors. Engineer J remembered his instrument was pegged

in these areas. On exit of the building, the individual that had the E-520 in the basement found his high-range pocket dosimeter read about 3 Rem. The individual that went to the radwaste panel recalled his pocket dosimeter read less than 1 R.

Both returned to the control room and were debriefed by the Supervisor, Radiation Protection and Chemistry and Supervisor, Station Operations. Their TLDs were read and indicated 3.14 rem and 0.170 rem. The 3.14 rem, when added to Engineer J's previous quarterly exposure resulted in a total first quarter whole-body exposure of 4.175 rem. The licensee submitted a report pursuant to 10 CFR 20.405 dated May 1, 1979, indicating that this individual had received a whole body gamma dose of 4.175 rem during the first quarter of 1979.

This entry demonstrated a lack of implementation of basic radiation protection training that could have resulted in serious consequences.

3.2.4.12. Not all Repair Party activities were handled as described above. On March 29, a Repair Party activity to install a temporary pipe permitting venting of the waste gas decay tank to the reactor building was installed. This job was planned in advance. The Supervisor Radiation Protection and Chemistry briefed the Repair Party Team leader (Maintenance Foreman G). The individuals were properly dressed and personnel pocket dosimetry records were kept. The job was completed within the dose allotment and no one became contaminated (Int. 143).

On March 30 about 0500, NRC Inspectors H and O met with the Superintendent, Technical and Administrative Services and Unit 1 Superintendent Technical Support to express concerns in regard to the radiation protection program (Int. 40). These concerns included: adequacy of personnel surveys, control of high radiation areas, documentation of surveys, contamination control, RWPs, and effluent monitoring.

On March 30, more control was established over in-plant radiation hazards. Radiation/protection technicians were being used on site to establish control points. Temporary dose accountability forms were implemented that evening at the site access point (Int. 106) The whole-body counter was in operation and individuals were being counted.

On the afternoon of March 30, the NRC became more actively involved in the radiation protection program. Five NRC radiation specialists were assigned to each of the three shifts providing around the clock coverage. In addition, two NRC health physicists provided technical guidance.

3.2.5. Summary of Radiation Protection Findings

3.2.5.1 Facilities

- a. Airborne radioactive materials spread from Unit 2 into Unit 1.
- b. High radiation areas were created in Unit 1 from unshielded Unit 2 reactor coolant sample lines.
- c. The combination of unshielded sample lines and poor ventilation resulted in the loss of:
 - a. Counting room equipment
 - b. The primary ECS
 - c. In-plant personnel decontamination facilities
- d. Airborne radioactivity monitoring devices were not available on site. These instruments were needed in the control rooms and laboratories. These devices would have helped to provide information necessary to make decisions on the need for respiratory protection.

3.2.5.2 Equipment

- a. Lapel air samplers were not available.
- b. Appropriate extremity monitors were not available.
- c. The number of wide range beta-gamma survey meters (0-1000 R/hr) available was insufficient. No beta-gamma survey meters with a range exceeding 1000 R/hr were available for persons entering radiation fields in that range.
- d. An adequate supply of high-range pocket dosimeters were not available.
- e. Alarming personnel dosimeters were not available.

3.2.5.3 Procedures

The Emergency Plan Implementing Procedures do not adequately address sustained in-plant radiation hazards.

3.2.5.4 Training

Actions of some workers, including members of the radiation protection and chemistry staff, did not reflect comprehension of the following:

- a. The need to know exactly when individuals entered and returned from areas of radiation hazard.
- b. The need to measure and document the airborne radioactivity to which workers are exposed.
- c. The need to perform detailed surveys of personnel contamination with appropriate instrumentation.

- d. The consequences of handling highly radioactive material.
- e. The necessity of appropriate personnel monitoring devices.
- f. The methods used to evaluate internal and skin dose.
- g. The limitation of portable survey instruments, including range, energy response, and significance of open and closed window readings.
- h. The plant systems and components.
- i. Concepts and techniques to minimize dose.
- j. The regulations, license conditions, and procedures.
- k. The emergency organization.

3.2.5.5 Dosimetry

The TLD dose printout for the first quarter of 1979 is not accurate or complete.

3.2.5.6 Air Sampling

Air samples collected in other than the auxiliary and fuel handling buildings did not indicate concentrations of iodine-131 in excess of the values listed in 10 CFR 20 Appendix B.

A review of the records for these samples does not indicate that air samples for particulate, iodine, or noble gas were collected in the auxiliary building breathing air zone from 0800 March 28 to 2400 March 30, a period in which entries were made.

3.2.5.7 Respiratory Protection

Since iodine adsorbing respiratory protective devices are not approved for use pursuant to 10 CFR 20.103(c), the licensee did not have any such devices on site until sometime on the afternoon of March 29.

The iodine-adsorbing cartridges would have been of value in reducing the intake of iodine if emergency egress of personnel had been necessary because of the development of an iodine release on site.

The quantity of self-contained breathing devices was not adequate, causing individuals to do without such devices at times when they should have been used. The use of a partially filled self-contained breathing device constituted an unnecessary risk for at least one individual who was alone and under stress in a hazardous environment, without backup search and rescue assistance.

Whole body count data were reviewed by the investigators. No apparent exposures in excess of the limits specified in 10 CFR 20.103 (a) were identified. However, the investigators are not aware of all entries to high airborne activity areas and therefore cannot conclude that no excessive exposures have occurred.

3.2.5.8 High Radiation Area Control

From about 1100 March 28 to the afternoon of March 30, control of access into the auxiliary buildings was not maintained to restrict access by unauthorized or unprepared individuals to hazardous radiation environments.

3.2.5.9 Management Control

During the period 0400 March 28, through midnight on March 29, licensee management did not make every reasonable effort to maintain radiation exposures to workers as low as was reasonably achievable considering the emergency conditions at the time and the equipment available for use. Specifically the following actions were not taken.

- a. Positive access control was not established to prevent entry of unprepared individuals into hazardous radiological environments.
- b. An effective method was not implemented to insure that all individuals entering hazardous radiological environments were fully briefed as to the hazard and the degree of urgency with which the task was to be performed.
- c. Equipment such as high range pocket dosimeters and survey meters were not controlled to insure that each individual entering high radiation areas was provided with the appropriate instruments.
- d. Individuals that became contaminated were not properly surveyed and decontaminated to insure their dose was minimized.
- e. Planning of those tasks which presented significant potential for radiation dose, such as reactor coolant sampling, was not reviewed by a knowledgeable management representative to insure that reasonable precautions were to be taken.

3.3 ENVIRONMENTAL ASSESSMENT AND PROTECTIVE ACTIONS

3.3.1 Initial Offsite Dose Calculation

Environmental assessment actions began at 0655 on March 28 with the declaration of a Site Emergency. Section 4.1.11 of Procedure 1670.2, "Site Emergency Procedure," requires the Emergency Director (Station Superintendent) to "Direct evaluation of off-site monitoring team data and project offsite dose consequences in accordance with 1670.4" (Ref. 166). Procedure 1670.4 produces estimates of projected thyroid and whole body doses at down wind locations as an end product (Ref. 167).

Nuclear Engineer C, working in the ECC (Unit 2 control room), completed the initial projection of environmental impact at 0710 on March 28 (Int. 48, 174). A whole-body exposure rate of 40 R/hr was calculated for location W-11 (Goldsboro), 1.2 miles west of the plant (Int 48, 174,; Ref 168). The location of survey point W-11 with respect to the plant and the onsite survey is shown in Figure II-3-2. The calculation was based on the following input parameters:

TABLE II-3-2

Offsite Dose Calculation Results (Actual) - Procedure 1670.4

Monitor reading	HP-R-214	300	R/hr
	HP-R-219	6 E+5	cpm (gas channel)
		*	cpm/min (iodine channel)
Noble gas release rate	HP-R-214	1325	Ci/sec
	HP-R-219	0.24	Ci/sec
I-131 release rate	HP-R-214	0.083	Ci/sec
	HP-R-219	*	Ci/sec
Atmospheric dispersion factor (χ/Q)		**	
Wind direction (toward)		270°	
Wind Speed		4	mph
Vent flow rates	AH-FR-5720	0	cfm
	AH-FR-5284	0	
	AH-FR-5286	14,000	
	AH-FR-5659	15,000	
	AH-FR-5063	0	
	AH-FR-5075	0	
Projected whole body dose rate due to noble gases		40,000	mrem/hr***
Projected thyroid dose rate due to iodine-131		8.5	mrem/hr***

*Not determined.

**The value used for the atmospheric dispersion factor is not known because the initial calculation sheet cannot be found (Ref. 168, Int. 48, 174). The χ/Q value (normalized to a wind speed of 1 mph) required to yield an exposure rate of 40 R/hr from a release rate of 1325 Ci/sec is $9.4 \text{ E-}4 \text{ sec/m}^3$, using the methods outlined in procedure 1670.4. This χ/Q value appears incorrect for two reasons: (1) the companion calculation for the offsite iodine concentration used a χ/Q value of $2.5 \text{ E-}4 \text{ sec/m}^3$ and (2) the specific value 9.4 represents an interpolation between the χ/Q isopleths (referenced in procedure 1670.4) that is unlikely to have been made. Had the same value ($2.5 \text{ E-}4 \text{ sec/m}^3$) been used for the noble gas calculation that was used in the iodine calculation, an exposure rate of 10 R/hr would have been calculated for Goldsboro.

***Dose rate is based on the addition of release rates determined from monitor reading.

ON ISLAND (GE SERIES) SURVEY POINTS
AND NEAR-FIELD TLD LOCATIONS

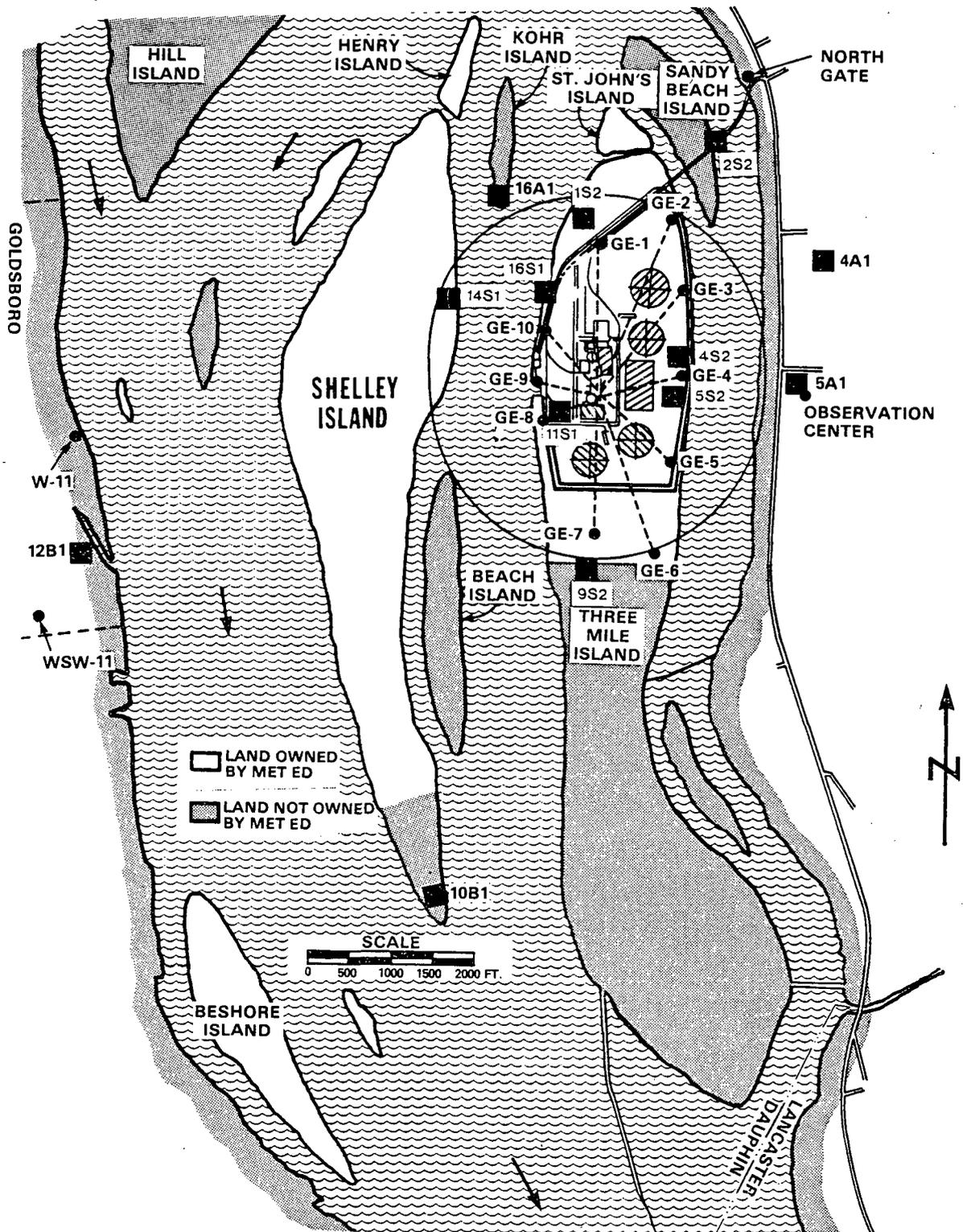


Figure II-3-2

The investigators have identified several probable errors in the preceding tabulation of input parameters employed in the offsite dose calculation, the largest of which was an error of three orders of magnitude (in the conservative direction) in the reading of HP-R-214. The containment dome monitor (HP-R-214) consists of a Victoreen 845 Series Area Monitor having an 8-decade response from 10^{-1} to 10^7 mR/hr (Ref. 44). The ion chamber detector is encased in a 2-inch lead shield in order to extend the range of the instrument. The monitor can be read out in two primary modes: (1) with the range selector switch on "all," the meter readout covers the entire eight decades or (2) with the range selector switch on 10^2 , 10^3 or 10^7 , any of six 3-decade expanded scale readouts are available (the switch position indicates the top decade of the 3-decade scale). The monitor readings contained in Table II-3-1 and in the text which follows have not been corrected for the attenuation provided by the lead shielding. Direct readings from the monitor were used in accordance with Procedure 1670.4.

Based on a review* of the chart recorder (HP-UR-1901) for the HP-R-214 monitor, the instrument was actually indicating an exposure rate of 400 mR/hr when the engineer read it at approximately 0655 on March 28, rather than 300 R/hr.** The investigator's reconstruction of this chart is shown in Figure II-3-3. The probable cause of this error is attributed to the fact that the engineers involved with the offsite dose calculations were unaware of the "expanded scale" feature of the instrument and misread it (Int. 174). The misinterpretation of this monitor reading was an indication of a weakness in the licensee's emergency plan training program (see Section 1.2.1).

*A discussion with Maintenance Foreman G indicated that during the period beginning about March 26, 1979, a measurements program was undertaken to calibrate the output signal from the HP-R-214 monitor so that the recording of this signal (eight decade response) on HP-UR-1901 (five decade recorder) could be interpreted after the chart was removed from the recorder. These measurements were part of the routine calibration program for these monitors. Although this program has not yet been completed, the calibration of the HP-R-214 signal was completed (after March 28) and was used by the investigators to interpret the chart from this monitor.

**A reading of 400 mR/hr on the expanded scale ($1-10^3$ range) was misinterpreted as 3×10^5 mR/hr by the individual reading the instrument.

DOME MONITOR HP-R-214

3/28/79

Note: The response of this monitor should be interpreted only as a relative indication of radiation level in containment. The actual level is not shown because the attenuation provided by the detector shield was not accounted for.

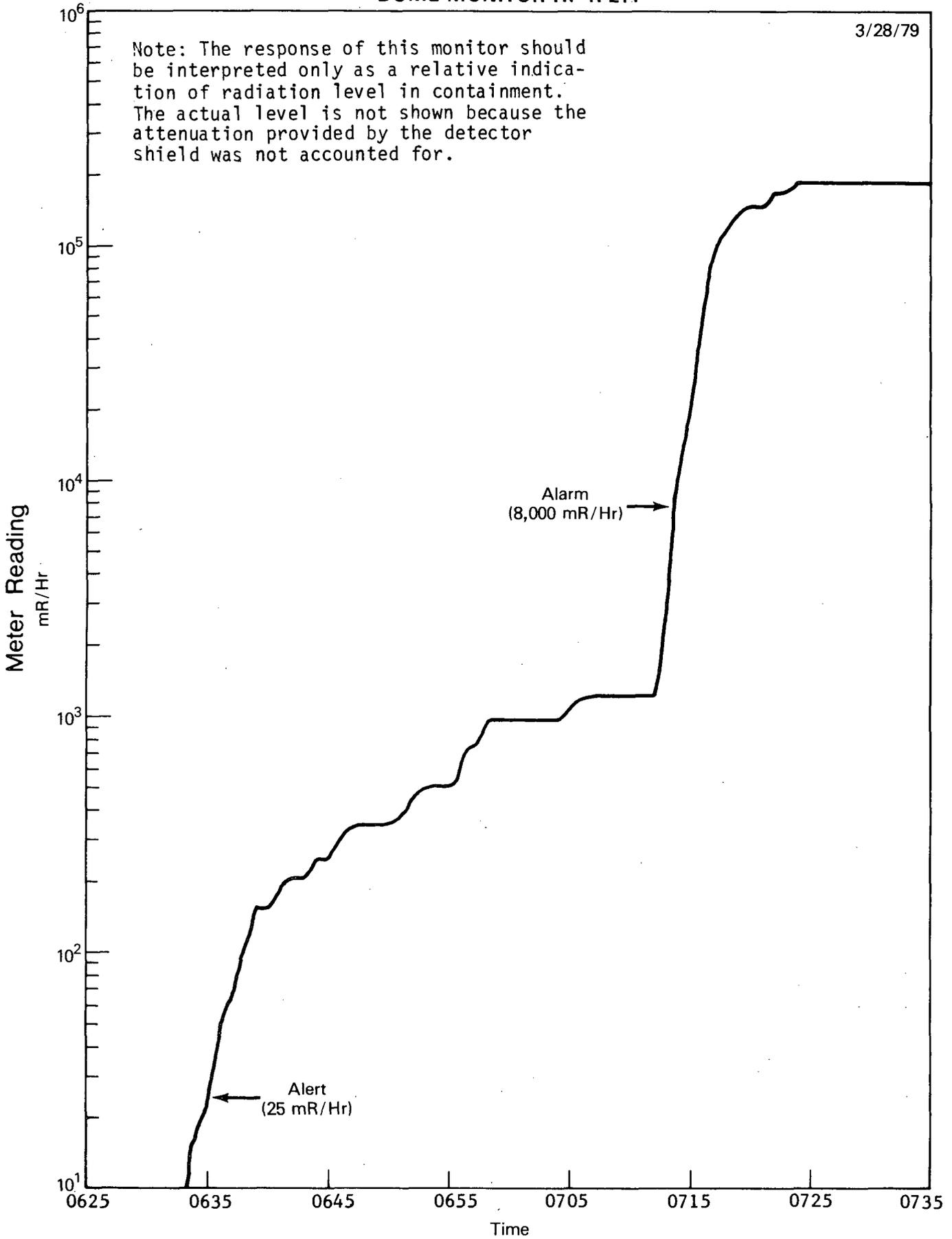


Figure II-3-3

The investigators have determined that if the licensee had read the instrumentation in the control room properly and had used the proper atmospheric dispersion factor for Goldsboro, the following would likely have resulted:

TABLE II-3-3

Offsite Dose Calculation Results
(Reconstructed by Investigators)
Procedure 1670.4

Monitor reading	HP-R-214*	400	mR/hr
	HP-R-219	500	cpm (gas channel)
		22	cpm/min (iodine channel)
Noble gas release rate	HP-R-214	1.8	Ci/sec
	HP-R-219	6 E-3	Ci/sec
Iodine-131 release rate	HP-R-214	6 E-4	Ci/sec
	HP-R-219	9 E-6	Ci/sec
Atmospheric dispersion factor**		4 E-4	sec/m ³
Wind direction (toward)		270°	
Wind speed		4	mph
Vent flow rates	AH-FR-5720	7000	cfm
	AH-FR-5284	0	
	AH-FR-5286	40,500	
	AH-FR-5659	43,000	
	AH-FR-5063	0	
	AH-FR-5075	0	
Projected whole body dose rate due to noble gases		2.2	mrem/hr***
Projected thyroid dose rate due to Iodine-131		0.1	mrem/hr***

*The release rates shown for the HP-R-214 monitor assume a reactor building pressure of 56 psig following a LOCA (Int. 48). The actual reactor building pressure at 0700 was about 1 psig; hence, the corresponding release rate was much lower than assumed in Procedure 1670.4.

**Value is for centerline of plume (270°). Location W-11 (Goldsboro) is slightly off the centerline (265°). The χ/Q estimated for this location is 6 E-5 sec/m³; the corresponding whole-body and thyroid dose estimates are 0.3 and 0.02 mrem/hr, respectively.

***Dose rate is based on the addition of release rates determined from each monitor reading.

In summary, by 0710 on March 28, the licensee had estimated that the whole-body exposure rate to an individual residing in Goldsboro was 40 R/hr. The investigators concluded that if the proper input parameter had been used, the licensee's procedure would have yielded an exposure rate of 2 mR/hr at Goldsboro. The basis for the 40 R/hr could not be determined by the investigators. The Supervisor, Radiation Protection and Chemistry later (prior to 0800) performed a cursory check of the calculation performed by the engineer and apparently concluded that no errors had been made (he also misread the HP-R-214 monitor). He recalled that this calculation resulted in an exposure rate of 10 R/hr rather than 40 R/hr. However, because of the very low reactor building pressure at the time (less than 1 psig), the supervisor felt that the offsite dose calculation was totally unreliable (Int. 133).

The State Bureau of Radiological Health (BRH) was notified at approximately 0725 that the estimated exposure rate in Goldsboro was 10 R/hr. This was apparently the same estimate that was made at 0710; it is not known why 10 rather than 40 R/hr was reported (Int. 46). The basis for the 10 R/hr is discussed in the footnote to Table II-3-2. Actions taken by the BRH at this point included (Int. 46) "...We then contacted Civil Defense [now the Pennsylvania Emergency Management Agency (PEMA)]... to inform them where the wind was blowing and to be prepared to evacuate on the west side of the river...". After being informed by Metropolitan Edison that there were no significant radiation levels detected outside of the plant, BRH "... notified the Civil Defense [PEMA] to hold tight. This was all before 8:00..."

3.3.2. Initial Confirmatory Surveys

Although a site emergency was declared at 0654, the first environmental radiation survey was not performed until 0748 at the greenhouse (between survey points GE-8 and GE-9 on the Island). This delay was apparently due to two factors: (1) a certain amount of time (estimated to be 10-15 minutes) was required to perform an operability check of each emergency kit, and (2) confusion existed at the time with respect to initial dispatching of teams.

The second point became apparent during interviews of radiation/chemistry technicians:

Radiation/Chemistry Technician A (Int. 99)

"When I got there [approximately 0700, March 28], it wasn't too long that we were assigned the job of the offsite team... we were ready to go ... where they needed some onsite readings right away... so in the haste, I radioed back and said we would take over the onsite team..."

Radiation/Chemistry Technician C (Int. 82)

"... I arrived at work approximately 7:00 a.m. [March 28] we walked into the processing center and everyone was standing around ...the first foreman we saw, he came up to me and said go out and check the radiation emergency kits... so I went out there and we got the SAMs out and made sure they responded... I guess it was approximately 8:30... [a radiation protection foreman] asked us if we were on our way to Goldsboro and I proceeded to tell him that we had never received any instructions where to go... so in about ten minutes I guess we had finally rounded up the vehicle and proceeded to Goldsboro..."

Radiation/Chemistry Technician N (Int. 47):

"...about 7:30 or 25 to 8. The [radiation protection] foreman... was trying to determine wind direction and other factors involved in setting up these monitoring teams. He told me... to get a radiation monitoring kit... and go to Goldsboro by vehicle... we got to Goldsboro about 8:45..."

Team alpha, consisting of two radiation/chemistry technicians (Int. 99), performed the survey at the screenhouse (between survey points GE-8 and GE-9) at 0748. The result was less than 1 mR/hr (the detection limit for the instru-

ment used). This survey confirmed that the significant offsite radiation levels calculated at 0710 (10 and 40 R/hr in Goldsboro) did not exist (Int. 127). Other confirmatory surveys were performed at 0832 and 0842 in Goldsboro (Ref. 153, 169). Results of both surveys were less than 1 mR/hr.

The calculated radioiodine concentration at Goldsboro, which corresponded to the initial thyroid dose rate projection of 8.5 mrem/hr (Table II-3-1) was $5 \text{ E-}6 \text{ } \mu\text{Ci/cc}$ (Ref. 168). Air samples (filter plus charcoal) were collected at 0900 at location GE-8 (on-island; 700 ft WSW of Unit 2 vent) and at location WSW-11 (South Goldsboro; 1.4 mi WSW of Unit 2 vent). Results (Ref. 169) for these samples, which were counted in the field with a SAM-2 with RD-19 detector, were $1.2 \text{ E-}8 \text{ } \mu\text{Ci/cc}$ for location GE-8 and $5.2 \text{ E-}9 \text{ } \mu\text{Ci/cc}$ for location WSW-11, both of which are very close to the minimum detectable activity (MDA) of $5 \text{ E-}9 \text{ } \mu\text{Ci/cc}$ applicable to the sample volume and background (Int. 22). These results were interpreted as iodine-131 activity in accordance with Procedure 1670.6, "Off-site Radiological Monitoring."

A radioiodine sample collected at 0940 at location W-11 (Goldsboro; 1.2 mi W) was transported to the Commonwealth of Pennsylvania Bureau of Radiological Health in Harrisburg for analysis on their Ge(Li) system (Int. 46). The results of that analysis completed at 1300 on March 28 indicated that most of the radioactivity on the charcoal was due to the 81 keV gamma rays from xenon-133 and the 250 keV gamma rays from xenon-135 (Ref. 170). The iodine-131 concentration was reported as less than $1.5 \text{ E-}11 \text{ } \mu\text{Ci/cc}$. The field-determined iodine-131 concentration for this sample was $3 \text{ E-}8 \text{ } \mu\text{Ci/cc}$ (Ref. 170). These results are discussed further in Section 3.3.3.2 which follows.

3.3.3. Offsite Radiation Measurements

The offsite radiation survey effort performed by the licensee during March 28 to 30 was directed by personnel in the ECS (Int. 22, 28, 76, 110).

The licensee's radiation survey effort consisted of the following types of measurements:

(a) Direct radiation measurements performed with portable radiation survey instruments by land-based and helicopter-based teams (Ref. 169). Instruments used (Int. 22, 28) were generally the PIC-6A* (an ion chamber type instrument having a range from 1 mR/hr - 1000 R/hr), the RO-2* (an ion chamber type instrument having a range from 0-5000 mR/hr with four scale multipliers) and the E-520* (a GM-type instrument having a range from 0-2 R/hr with five scale multipliers).

(b) Collection of short-term air samples (particulate and iodine) for field determination of radioiodine concentrations (primarily March 28). These samples were later (primarily after March 28) counted with a Ge(Li) system based in a mobile laboratory (Int. 73, 93).

(c) Collection of air, water, and milk samples and TLDs as part of the routine Radiological Environmental Monitoring Program (REMP) beginning on March 29 (Int. 19, 28).

3.3.3.1 Direct Radiation Measurements

Beginning with the initial surveys (discussed earlier) performed early on March 28, a radiation survey program continued beyond the period of this investigation (Ref. 169). As mentioned above, the offsite and onsite surveys were directed by the ECS. Teams consisted initially of two radiation/chemistry technicians and later (primarily March 29 and 30), as more manpower became available, teams consisted of three individuals (Int. 22, 182). Three teams were dispatched initially and up to six teams were later making surveys on March 29 and 30 (Int. 76, 82).

* Manufactured by the Eberline Instrument Co.

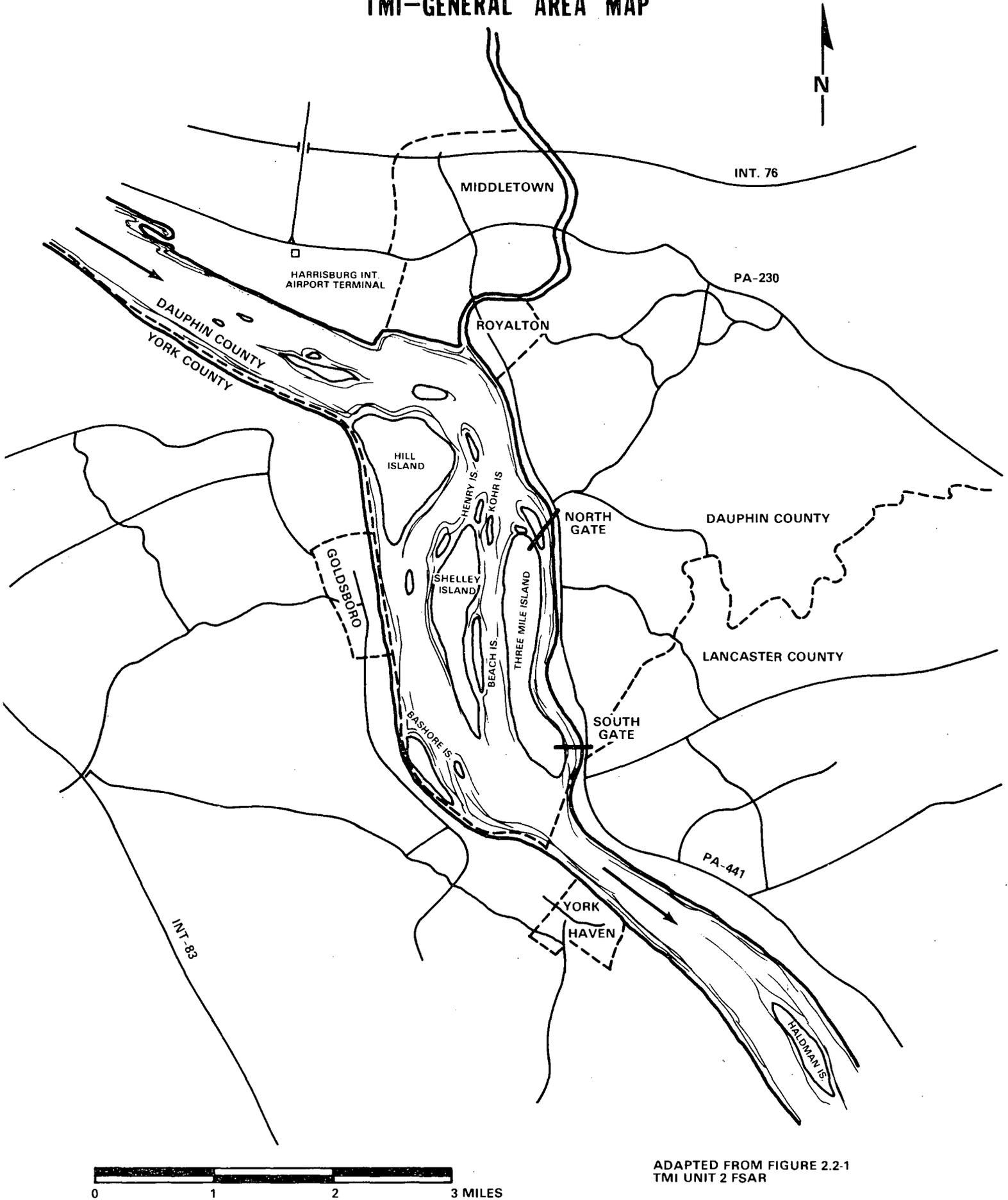
Offsite teams were dispatched by the ECS to specific, predetermined points (most were within 5 miles from the site) located on a 1:24,000 scale U.S. Geological Survey Quadrangle Map (Int. 48). A general area map is shown in Figure II-3-4. A wind speed and direction recorder is located in both control rooms; these parameters are measured at the site meteorological tower located 2200 feet north of the Unit 2 reactor building (Figure 2.3-22 TMI-2 FSAR). Plume dispersion at any given location was determined by persons in the ECS using the methods and atmospheric dispersion factor (χ/Q) isopleth overlays (also drawn to a 1:24,000 scale) referenced in Emergency Procedure 1670.4, "Radiological Dose Calculations" (Ref. 167). According to this procedure, a stable, neutral, or unstable atmospheric dispersion isopleth is used, depending whether the wind direction range (over a 20-minute interval) is $<45^\circ$, between 45° and 75° , or $>75^\circ$, respectively.

In general, teams were dispatched in the downwind direction to perform surveys at points that were inside the χ/Q isopleths (i.e., inside the known extent of the plume). Onsite teams, in general, were dispatched in the downwind direction to one of the GE-series locations shown in Figure II-3-5.

Radiation levels measured by land-based onsite and offsite teams first began to increase at 1020 on March 28, with an onsite team in the north parking lot reporting a radiation level of 3 mR/hr (Ref. 169). Onsite radiation levels generally continued to increase during the next 12 to 13 hours, peaking at 300 mR/hr (β, γ) near the north warehouse at 2130 on March 28 and 365 mR/hr (β, γ) at survey point GE-10 at 2325 on March 28.

Offsite radiation levels reported during this period, with the exception of 50 mR/hr measured at 1548 on PA Route-441 about 1500 feet south of the north gate (see Figure II-3-4 for location), were not above background until 2238 when a radiation level of 13 mR/hr was measured near Kunkel School (5.6 mi NNW) (Ref. 169). Several other radiation levels above background were noted in this general area prior to midnight. However, the 13 mR/hr value was the highest one measured until 30 mR/hr (β, γ) was measured in Goldsboro at 0600 on March 29.

TMI-GENERAL AREA MAP



ADAPTED FROM FIGURE 2.2-1
TMI UNIT 2 FSAR

Figure II-3-4

Several comments relative to the Kunkel School and Goldsboro measurements referenced above need to be made at this point:

(1) During the 7-hour period from 1700 to 2400 on March 28, the wind was blowing consistently toward the northwest at a speed of 5-15 mph; during the 2-hour period from 0340 to 0540 on March 29, the wind was blowing consistently toward the west-northwest at a speed of 4-8 mph; and during the 3 1/2-hour period from 0540 to 0910 on March 29 the wind was blowing consistently toward the west at a speed of 2-6 mph (Ref. 171). During these periods of time (12 1/2 hours total), the effluent plume from the plant should have been well defined.

(2) According to an analysis made by the licensee, during the 33-hour period from 0700 on March 28 to 1600 on March 29, $6.6 \text{ E}+6 \text{ Ci}$ of noble gases were released (Ref. 84, 85). This quantity corresponds to 75% of that released through the end of March and 66% of the total released from March 28 through April 30.

(3) The plume positions for the periods of time discussed in (1) above are shown in Figures II-3-6, -3-7, and -3-8. The plumes shown are for stable atmospheric conditions (wind direction range 45°) and, although they are somewhat idealistic (e.g., they do not depict the turbulence that might have been introduced by Hill Island - Figure II-3-6), they served as a reasonable means of directing the survey teams to the higher concentrations of radioactivity offsite (Ref. 172, Int. 22)

(4) From an examination of Figures II-3-6 through II-3-8 and the survey times contained in the ECS Radiation Survey Log (Ref. 169), it is apparent that:

- a. The licensee failed to perform radiation surveys in the first plume until 2238 (at Kunkel School) - more than 5 hours after the plume appeared to have been established;

- b. The licensee failed to perform any radiation surveys in the second plume - a period of 2 hours;
- c. Surveys were performed in the third plume (beginning at 0600) in Goldsboro with a resultant radiation level of 30 mR/hr (β , γ), 20 mR/hr (γ).

Offsite radiation levels measured by land-based teams during the remainder of March 29 were generally less than 1 mR/hr, with the maximum noted as 3 mR/hr in Royalton at 2355 (see Figure II-3-4 for location). Onsite radiation levels measured during March 29 were generally less than 10 mR/hr, with the maximum noted as 150 mR/hr (β , γ) and 100 mR/hr (γ) at location GE-9 at 0532 (see Figure II-3-5 for location).

Offsite radiation levels measured by land-based teams during March 30 were generally less than 1 mR/hr, with the maximum noted as 15 mR/hr (β , γ) and 5 mR/hr (γ) at standard survey point S-11 (approximately 1 mile south of the plant) at 0906. Onsite radiation levels measured during March 30 were generally less than 10 mR/hr, with the maximum noted as 110 mR/hr (β , γ) and 20 mR/hr (γ) at location GE-9 at 1445 (Ref. 169).

The numbers of radiation measurements made by the licensee's land-based teams during March 28 through March 30 at onsite and offsite locations is as follows (Ref. 169):

<u>Date</u>	<u>Shift</u>	<u>On-site (Outside)</u>	<u>Off-site</u>
March 28	0700-1500	20	19
	1500-2300	29	27
March 29	2300-0700	14	14
	0700-1500	15	8
	1500-2300	36	3
March 30	2300-0700	54	45
	0700-1500	68	36
	1500-2300	67	19
Totals		303*	171

*Includes standard survey locations GE-1 through GE-10 only (Figure II-3-5); estimated by the investigator to represent about 80% of the total effort on site (outside of buildings).

PREDICTED PLUME LOCATION (1700-2400, MARCH 28)

1700-2400
3/28/79

ATMOSPHERIC DISPERSION FACTOR (x/Q)
FOR A STABLE ATMOSPHERIC CONDITION
(WIND DIRECTION VARIABILITY $<45^\circ$),
NORMALIZED TO A WIND SPEED OF 1 MPH.

- PREDETERMINED RADIATION SURVEY POINT
- ⊙ SURVEY PERFORMED BY MET ED DURING STATED INTERVAL
- TLD LOCATION (ROUTINE PROGRAM)

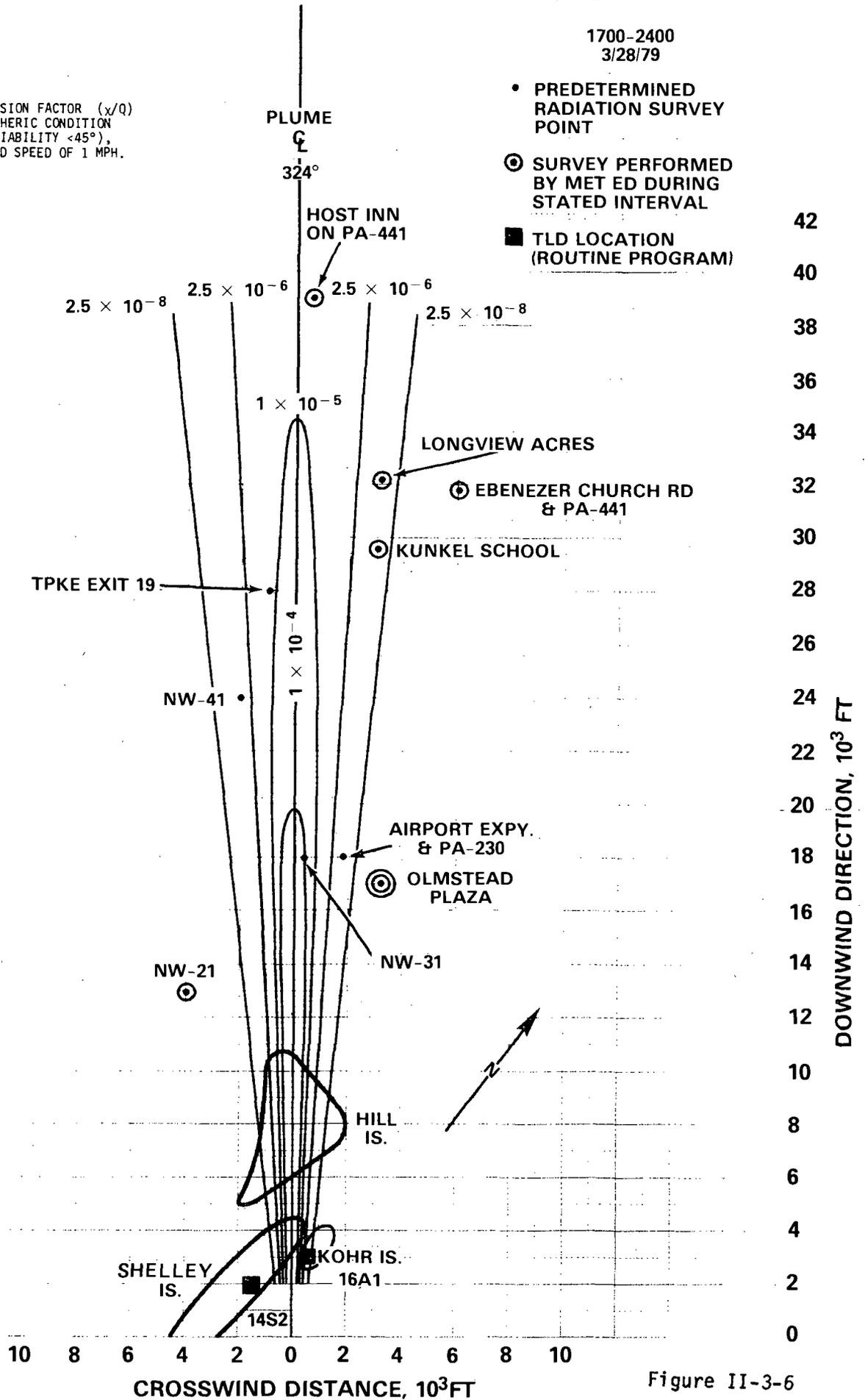


Figure II-3-6

PREDICTED PLUME LOCATION (0340-0540, MARCH 29)

ATMOSPHERIC DISPERSION FACTOR (χ/Q)
 FOR A STABLE ATMOSPHERIC CONDITION
 (WIND DIRECTION VARIABILITY $<45^\circ$),
 NORMALIZED TO A WIND SPEED OF 1 MPH.

0340-0540
 3/29/79

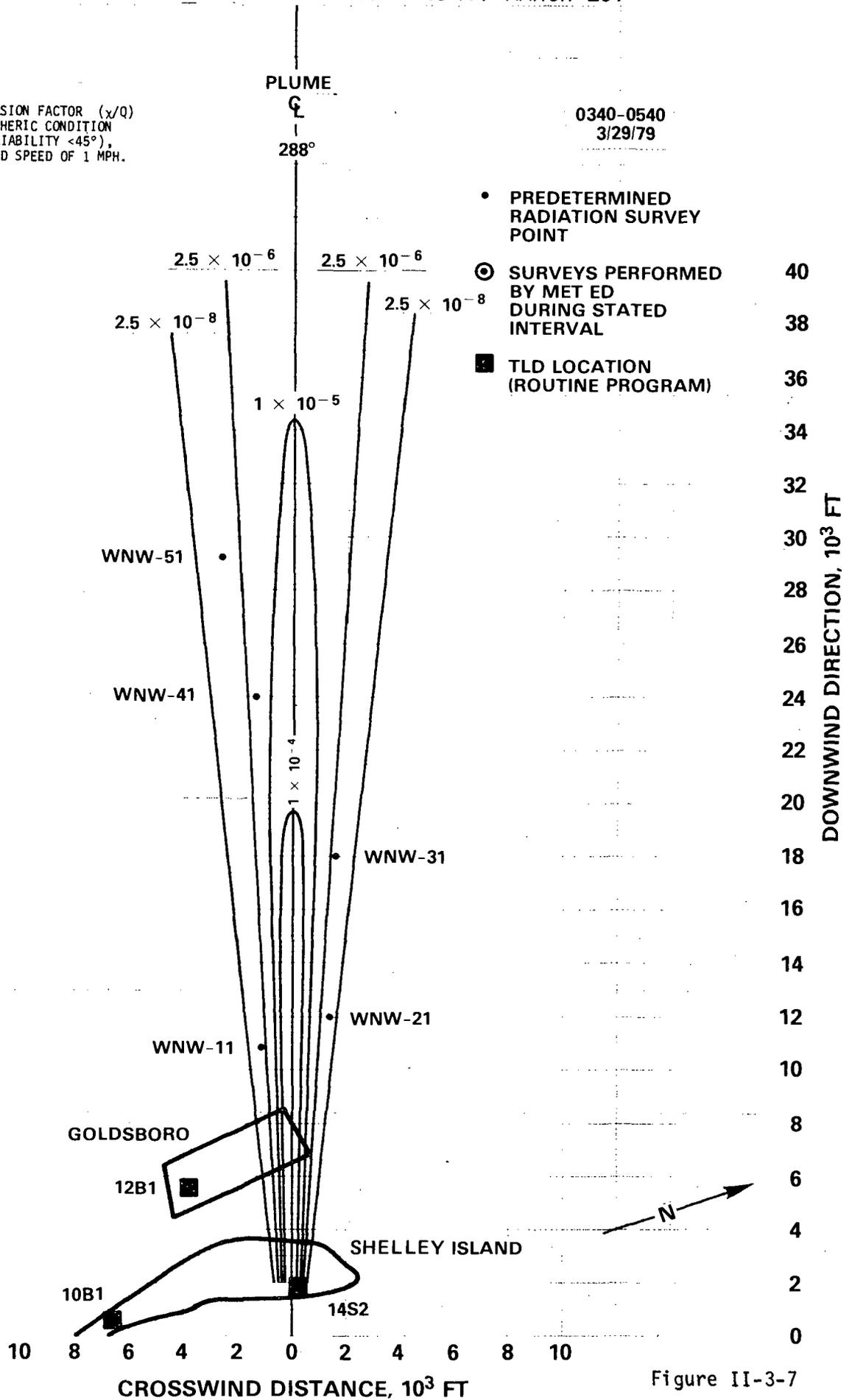
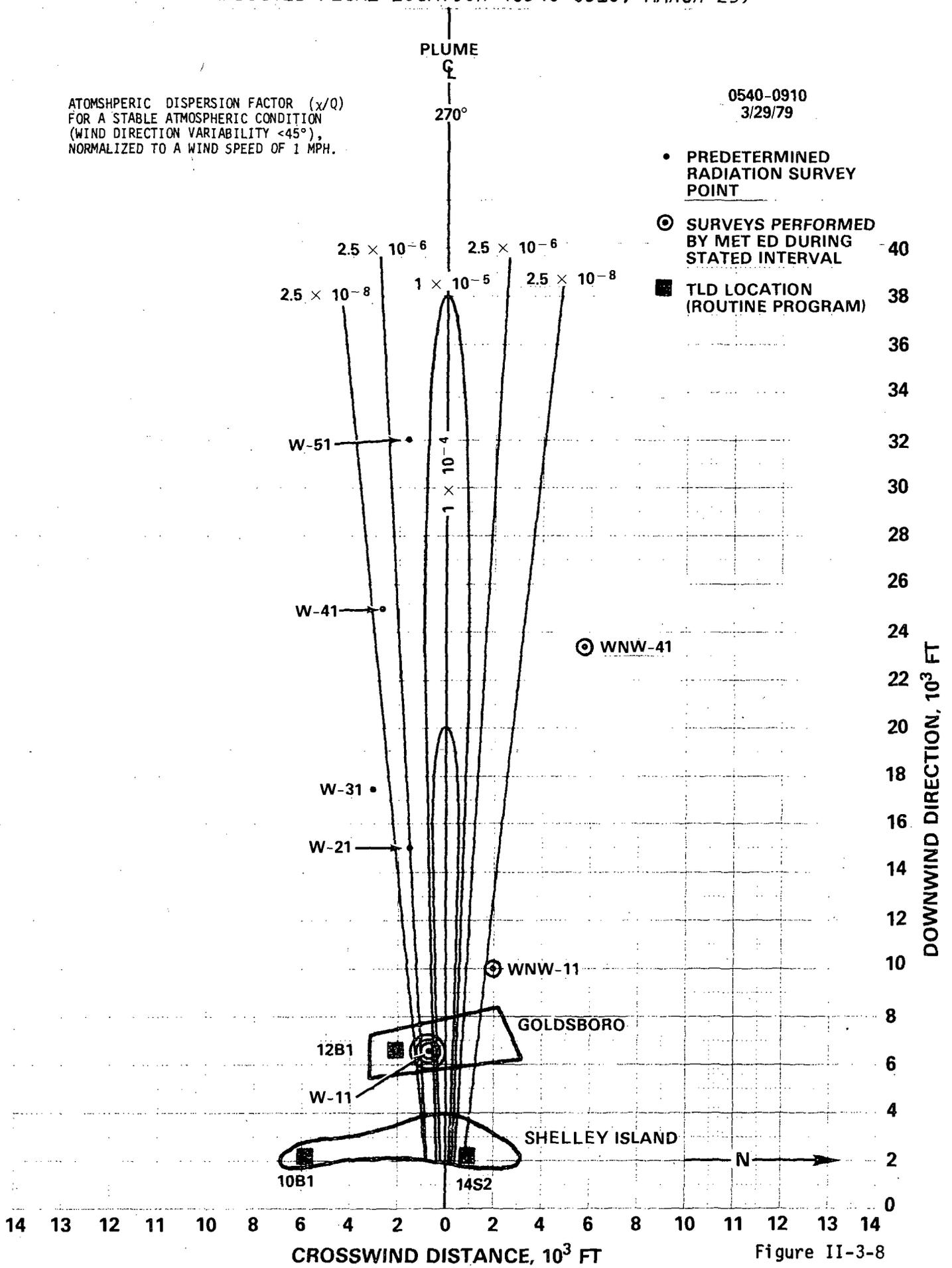


Figure II-3-7

PREDICTED PLUME LOCATION (0540-0910, MARCH 29)

ATMOSPHERIC DISPERSION FACTOR (x/Q)
 FOR A STABLE ATMOSPHERIC CONDITION
 (WIND DIRECTION VARIABILITY <45°),
 NORMALIZED TO A WIND SPEED OF 1 MPH.

0540-0910
 3/29/79



- PREDETERMINED RADIATION SURVEY POINT
- ⊙ SURVEYS PERFORMED BY MET ED DURING STATED INTERVAL
- TLD LOCATION (ROUTINE PROGRAM)

Figure II-3-8

The above radiation survey effort represents only those surveys performed by the licensee and those performed under the licensee's direction. Surveys performed by the Commonwealth of Pennsylvania, the Department of Energy and its contractors, the Nuclear Regulatory Commission, and other Federal agencies during March 28 through 30 are not included above since, with the exception of a few Commonwealth and NRC surveys, results were not made available to the licensee (Ref. 169, Int. 48). Details of these surveys appear in Appendix II-C, Tables II-C-1 through II-C-6.

Helicopter-based teams were used by the licensee during the period from March 28 through 30, although the heaviest usage during the period was on March 30. This type of survey capability is not addressed in the Emergency Plan (Int. 20); however, both the Supervisor, Radiation Protection and Chemistry and the Radiation Protection Supervisor indicated that such capability was very helpful (Int. 20, 22).

The helicopter-based teams were called upon to find the noble gas effluent plume and to make sufficient surveys to define it with reference to certain landmarks (Int. 25). Initially one helicopter belonging to the State Police, and later, three helicopters chartered by the licensee were used in this effort (Int. 25). A summary of these surveys follows (Ref. 169):

<u>Date</u>	<u>Number of Flights</u>	<u>Number of Radiation Measurements</u>	<u>Range of results - mR/hr (β, γ)</u>
March 28	1	5	<1 - 20
March 29	1	24	<0.1 - 3000
March 30	13	295	<0.1 - 1200

The maximum radiation level measured during each survey was generally directly above the plant. The 3000 mR/hr (γ component was 400 mR/hr) was measured approximately 15 ft above the Unit 2 plant at 1410; the 1200 mR/hr was measured approximately 130' (600' above sea level) above the Unit 2 reactor building at 0801.

3.3.3.2 Airborne Radioactivity Measurements

Including the three initial air samples discussed earlier, a total of 28 air samples were counted in the field with the SAM-2 during the period March 28 through 30 (24 on March 28 and two on both March 29 and March 30) (Ref. 169). Eleven of these samples were also counted by the Pennsylvania BRH Laboratory (Ref. 170); the Radiation Management Corporation's Fixed and Mobile Laboratories (Ref. 178, 173); and by the Nuclear Regulatory Commission's Region I Mobile Laboratory (Ref. 177). Radioiodine concentrations on these 11 samples were less than the minimum detectable activity (MDA) for each laboratory, which ranged from 2 E-11 to 6 E-10 $\mu\text{Ci/cc}$ (Int. 46, 73, 93). Licensee results for these same samples ranged from $<5 \text{ E-9}$ to $6.8 \text{ E-7 } \mu\text{Ci/cc}$ (Ref. 169). In summary, the samples collected and method of counting were as follows:

<u>Date</u>	<u>Number Counted with SAM-2 only</u>	<u>Number Counted with SAM-2 & GeLi</u>	<u>Number Counted with GeLi only</u>	<u>Total</u>
March 28	15	9	3	27
March 29	1	1	12	14
March 30	1	1	14	16

Radioiodine was not detected (i.e., was below the MDA) in any of the 40 samples counted by Ge(Li) Spectrometry (Ref. 170, 173, 177, 178); however, xenon-133 and xenon-135 were detected in many of them. These radioxenon's interfered with the ability of the SAM-2 to quantify radioiodine according to a Metropolitan Edison consultant who was present in the ECS at the time (Int. 19, 28):

".....I suspected that it might be what we called white out..... [there] was just absolutely so much xenon there that the crystal was saturated because there are small crystals in the SAM-2s....."

Another difficulty encountered with the use of the SAM-2 was the fact that several radiation/chemistry technicians stated that they either had not been trained in the use of the SAM-2 or that they were not confident of their ability to use the instrument because of the small amount and infrequent nature of the training that they did receive (Int. 82, 99, 156).

3.3.3.3 Radiological Environmental Monitoring Program Data

Samples from the routine Radiological Environmental Monitoring Program (REMP) were collected on March 29 after comparing the benefit of early sample collection and early knowledge of environmental impact with the benefits of later sample collection (and being able to integrate more of the early environmental impacts) (Int. 19, 28). Discussions on this matter were held between the Supervisor, Radiation Safety and Environmental Engineering and his staff and Radiation Protection Consultant A on March 28 (Int. 19, 28, 30, 160). Samples from the REMP collected on March 29 were analyzed by Teledyne Isotopes and the Radiation Management Corporation (telecon with representative of Porter-Gertz Consultants). The data available to the licensee prior to 2400 on March 30 are contained in Table II-3-3 (Ref. 175). After the March 29 collection, the licensee conducted the REMP on an increased regime basis, which included daily milk and water sampling, daily TLD changeout, and air sample changeout every three days (Int. 19, 28). Based on a telephone conversation with a representative of Porter-Gertz Consultants, Inc., air iodine measurements were also made at two additional locations, making a total of eight locations (ten results appear in Table II-3-4 because two locations have additional samples for quality control purposes).

With regard to the TLD data shown in Table II-3-4, two results for Kohr Island contained in the footnote (908 and 453 mR/6 mo.), need further elaboration at this point. During a telephone conversation with a representative of Teledyne Isotopes, an investigator determined that the Kohr Island TLD's were exposed for a six month period (ref. 103); however, most of that exposure was received during the last two days that the TLD was exposed (March 28 and 29). The background radiation exposure for these TLDs, using the 9G1, 7G1 and 15G1 locations as controls, was estimated to be 40 mR for the six month period. If this 40 mR is subtracted from the two dosimeters and the remaining exposure is divided by the exposure time after the event (29 hours from 0700 on March 28 until approximately 1200 on March 29), average exposure rates for Kohr Island are calculated to be 30 and 14 mR/hr. Since the radiation being measured is gamma radiation the simplifying assumption is made that a 1 mR exposure is approximately equal

to a dose of 1 mrem. Therefore, these exposure rates are equivalent to dose rates of about the same value. These dose rates are in excess of the 2 mrem/hr limit for unrestricted areas contained in 10 CFR 20.105(b).

Table II-3-4
Radiological Environmental Monitoring Program
Data Available to the Licensee by March 30, 1979

<u>Type</u>	<u>Location</u>	<u>Result (mR/qtr)</u>	<u>Available</u>
TLD	N. Bridge (1S2) - 0.4 mi N	37.5	3/30/79 @1230
	N.Wthr. Stn (2S2) -0.7 mi NNE	81.0	3/30/79 @1230
	N. Boat Dock (16S1) - 0.2 mi NNW	923.0	3/30/79 @1230
	Falmouth Subst. (7B3) - 1.6 mi SE	11.6	3/30/79 @1230
	Ob. Center* (5A1) - 0.4 mi E	15.6	3/30/79 @1230
	Goldsboro (12B1) - 1.6 mi WSW	15.1	3/30/79 @1230
	Middletown Substn. (1C1) -2.6 mi N	16.7	3/30/79 @1230
	Drager Farm* (7F1) - 9 mi SE	21.4	3/30/79 @1230
	N. York Subst. (9G1) - 13 mi S	19.3	3/30/79 @1230
	Columbia (7G1) - 15 mi SE	22.4	3/30/79 @1230
	W. Fairview* (15G1) - 15 mi NW	17.2	3/30/79 @1230
	S. Beach TMI (9S2) - 0.4 mi S	23.0	3/30/79 @1230
	Laurel Rd (4A1) - 0.5 mi ENE	17.0	3/30/79 @1230
	Kohr Is. (16A1) - 0.4 mi NNW	8.0**	3/30/79 @1230
	S. Shelley Is. (10B1) - 1.1 mi SW	7.0**	3/30/79 @1230
	Shelley Is. (14S1) - 0.4 mi WNW	6.9**	3/30/79 @1230

* Monthly TLDs were also read at these locations. Data were consistent with the quarterly TLD, except for the Observation Center which read only 0.5 mR/mo.

** Results for TLDs that were sent to TMI for changeout at the end of the fourth quarter; however, TLDs at these locations were not changed out because of weather conditions and remained in a lead container (i.e., were unexposed) for the first quarter of 1979. Results for these locations, which were exposed for the fourth quarter of 1978 and the first quarter of 1979, (two TLDs per location) were: 14S2-131 & 148 mR; 16A1-908 & 453 mR; and 10B1-40.6 & 36.6 mR (all values are mR/6 month interval). According to a telecon with a representative of Teledyne Isotopes, the factor of two difference between the Kohr Island TLDs cannot be explained.

Table II-3-4 Continued

Radiological Environmental Monitoring Program
Data Available to the Licensee by March 30, 1979

<u>Type</u>	<u>Location</u>	<u>Result</u>	<u>Available</u>
Air iodine	Middletown Substn (1C1) - 2.6 mi N*	0.05 pCi/m ³	3/30 @ 0945
	Drager Farm (7F1) - 9 mi SE*	<0.02	3/30 @ 0945
	N. Weather Stn (1S2) - 0.7 mi NNE	0.47	3/30 @ 1230
	Falmouth Substn (8C1) - 1.6 mi SE	<0.02**	3/30 @ 1230
	Observation Cntr (5A1) - 0.4 mi E	<0.02	3/30 @ 1230
	W. Fairview (15G1) - 15 mi NW	<0.03	3/30 @ 1230
	Drager Farm (7F1) - 9 mi SE	<0.04	3/30 @ 1230
	Middletown Substn (1C1) - 2.6 mi N	0.03	3/30 @ 1230
	Goldsboro (12B1) - 1.6 mi WNW	0.30	3/30 @ 1230
	N. York Substn. (9G1) - 13 mi S	<0.02	3/30 @ 1230
Water Iodine	Swarata Creek (upstream)	<10 pCi/l	3/30 @1230
	Steelton Water Works (upstream)	<10	3/30 @1230
	York Haven Gen. Stn. (downstream)	<10	3/30 @1230
	Columbia Water Plant (downstream)	<10	3/30 @1230
	York (downstream)	<10	3/30 @1230

*Additional sampler for quality control purposes.

**Result is suspect. Air sampler did not appear to be working properly.
(Section 1.7.1.1)

Data from 21 locations on the island having LiF TLDs (the same as those used to assess occupational exposure - see Section 1.4.3 and 1.6.1), although not available to the licensee during the period covered by this investigation (March 28 to 30), need to be briefly recognized at this point. A comparison of results for these dosimeters, which were generally changed out on a monthly basis (near the end of the month), with the environmental TLD results for March and April is as follows (Ref. 166):

(1) Environmental TLDs indicated that the highest offsite dose rates occurred during the first three days following the event, approaching background levels levels prior to mid-April (Ref. 84, 85).

(2) The twenty-one onsite LiF TLDs indicated dose rates during the first few days in April may have been higher than dose rates during March 28-30 if most of the April exposure occurred during the first few days of the month. If a longer exposure period is assumed, however, these TLDs appear to be consistent with the results of the environmental TLDs.

Further review of this area is beyond the scope of this investigation.

3.3.3.4 Assessment/Response to Environmental Measurement Results

Based on the environmental impact information gathered by the licensee during March 28 through March 30, as discussed in the previous sections, it appears to the investigators that:

1. The licensee's actions taken to confirm the initial prediction of 40 R/hr in Goldsboro could have been more timely. Even though the licensee believed the predicted exposure rate to be unrealistic, confirmation that such levels did not exist outside of the plant required 38 minutes (prediction completed at 0710; confirmatory survey made at onsite location GE-8, 720 ft. SW of vent, at 0748).

2. The principal radiation exposure pathway for individuals residing in the vicinity of the site was direct radiation from radioactive noble gases (primarily xenon-133) released to the atmosphere. Radioiodines and other nuclides released in gaseous and liquid effluents constituted a much less significant exposure pathway.

3. The methods employed by the licensee to assess doses received via the principal pathway were:

(a) Use of TLDs which had been deployed (prior to the event) at 15 indicator and 5 background locations as part of the routine monitoring program.

It appears that, on occasion, the plume centerline was located between TLD locations. The deployment of additional TLDs early in the event would have increased the probability of a given TLD being in the plume. Based on an examination of meteorological data, however, the time periods when the wind blew consistently into a given sector for several hours at a time were $\leq 30\%$ of the total 68-hour period examined during this investigation (Ref. 171). During the major portion of the 68-hour period examined, the plume was either not well defined or tended to meander (Int. 19, 22, 28, Ref. 171). This latter condition likely caused exposure rates at a given point to fluctuate considerably during much of the time (Ref. 169, Int. 22). Both of the above conditions (a well-defined plume in between TLD locations and an ill-defined, meandering plume) could have been evaluated better by the licensee had additional TLDs been deployed early in the response. This point is addressed in procedure 1670.6 (Ref. 176):

" . . . Place sufficient TLDs at either the continuous air monitor or at a convenient, representative location in the designated area to permit reading of TLDs every four hours during the emergency . . . "

The deployment of additional TLDs apparently had not been considered by the licensee during early stages of event (Int. 22, 25); however, even if this had been considered, according to Metropolitan Edison Corporate Environmental Scientist B the licensee was not prepared to implement this portion of the program (Int. 74):

"I was aware that [the deployment of additional TLDs] was in the procedure; however, I was going to change the procedure because the environmental TLDs cannot be picked up and read on site. They are special TLDs which have to be sent off to Teledyne Isotopes and Radiation Management Corporation to be read. So an offsite monitoring team could not collect those environmental TLDs and bring them here to be read."

b. Use of survey teams with portable radiation survey instruments to perform surveys at predetermined offsite locations and to track effluent plumes (primarily with the helicopter).

The data gathered by such teams were the licensee's primary means of monitoring the environmental impact of effluent releases until about mid-day on March 30 when TLD results (changed out March 29) were available. The survey instrument results were the only means of quickly assessing changes in environmental impact and were a necessary adjunct to the TLDs.

Results of offsite radiation surveys were evaluated by personnel in the Emergency Control Station. An environmental consultant was called upon to play a lead role in the evaluation. The consultant used the following criterion for assessing whether offsite radiation levels were increasing to the point where action might be required (Int. 19, 28):

". . . my trigger point is that if we're going to be giving people in the environment somewhere between 50-100 millirem, at that point we have to let the State know . . . we had a hot line to the State and every, practi-

cally every number that came in, we just relayed to the State so it was just a matter if it looked like the collective dose was up there between 50-100 mR then it was up to me to say to the Emergency Director, 'At this point, I think we better tell the state that it looks like the collective dose is reaching this point and so therefore, they need to think whether or not they want to take protective action.' But we never even got to that point"

As noted earlier, the investigators have identified problems with the surveys conducted by the licensee, i.e., the periods from 1700 to 2400 on March 28 and 0340 to 0910 on March 29 when an inadequate assessment of radiological conditions was made. In addition, it appears that original radiation survey records have been discarded (Ref. 194), leaving the only record of survey results being those radioed in and recorded by personnel in the ECS (Ref. 169, 174). The ECS survey log, however, does not contain (with few exceptions) any notation of the specific instrument used to perform the survey, making it very difficult to determine the correction factor (due to the 81 keV gamma ray from xenon-133) to be applied to the measurements (Ref. 183). The following were also partially or totally absent from this survey log: (a) mode in which the instrument was used, i.e., whether open or closed window; (b) orientation of the instrument with respect to the source being measure; (c) the duration of the measurement; and (d) name of the individuals performing the survey.

4. Radiation dose estimates for the hypothetical maximum exposed individual have been published previously by the licensee and by an Ad Hoc Committee consisting of NRC, EPA and HEW representatives (Ref. 65, 84, 85). Estimates of child and adult thyroid doses from the inhalation and ingestion (milk) pathways contained in these reports was 5 mrem or less for the highest exposed individual.

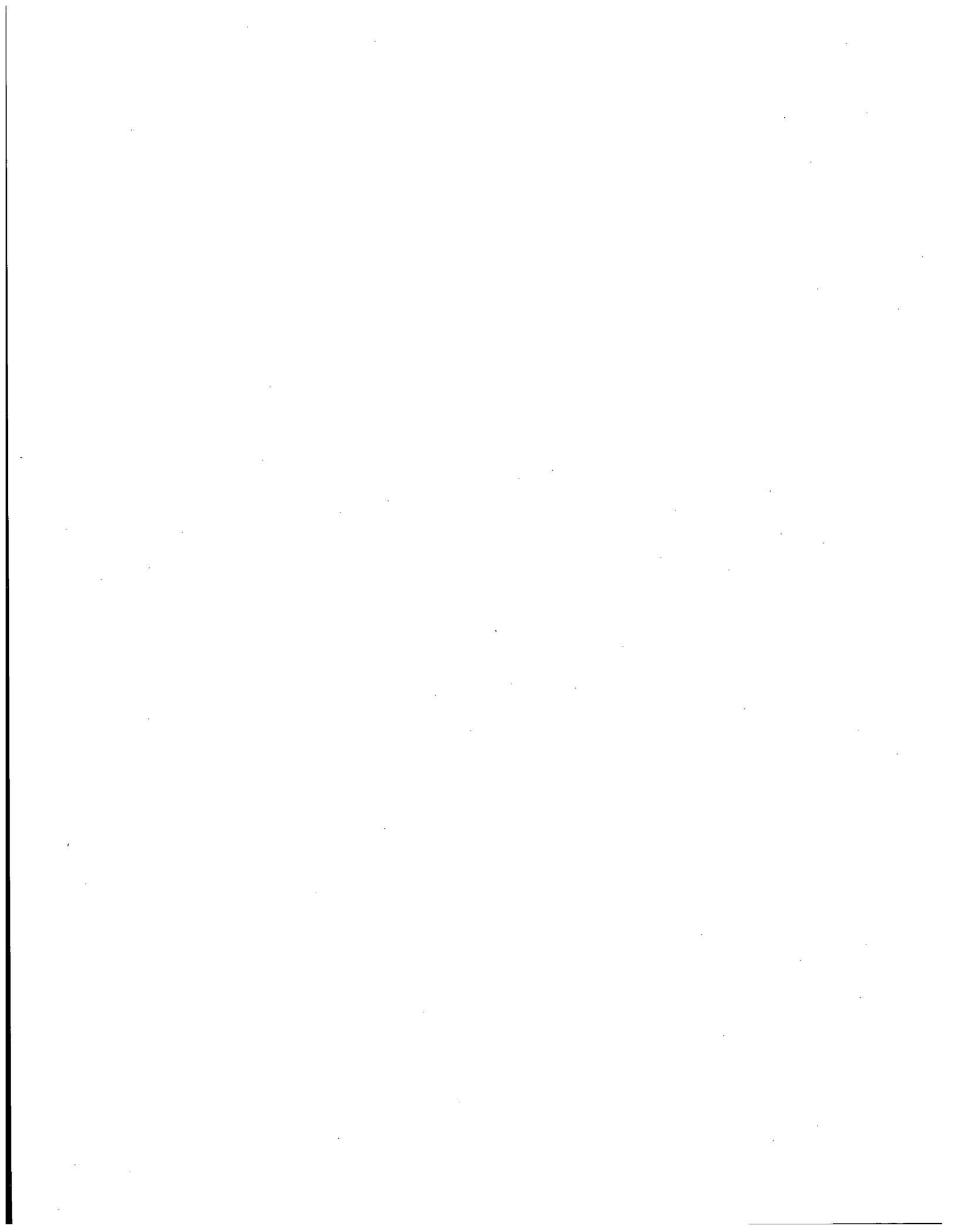
With respect to radioiodine releases to the atmosphere, results of fifty-seven airborne radioiodine measurements based on short-term samples and ten airborne radioiodine measurements based on long-term samples from the routine radiological environmental monitoring program (REMP) were available to the licensee during

the period March 28-30. Fifty of these samples (40 of 57 short-term and 10 of 10 REMP samples) were analyzed by Ge(Li) spectrometry for I-131. These analyses indicated that all results, with the exception of five REMP samples, were below the minimum detectable activity (MDA) which ranged from 2 E-14 to 6 E-10 $\mu\text{Ci/cc}$. The lower MDA values were associated with the larger volume samples from the REMP, while the higher MDA values were associated with the small volume (grab) samples collected by the survey teams. Results of the five REMP air iodine samples above the MDA ranged from 0.03-0.47 pCi/m^3 . The MPC for iodine-131 in air for unrestricted areas is 100 pCi/m^3 .

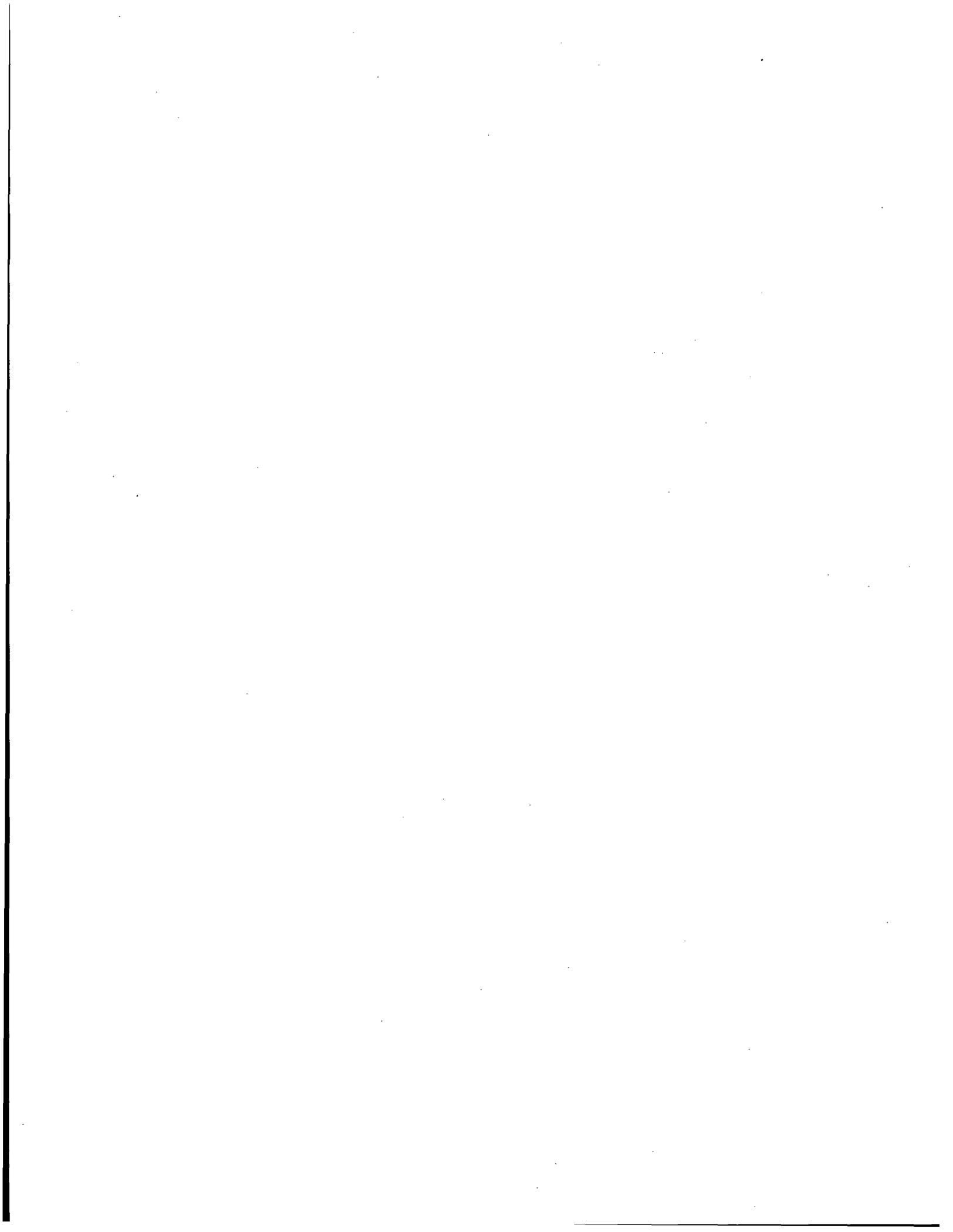
Doses received by way of the principal environmental pathway (noble gas releases to the atmosphere) were also included in each report (Ref. 65, 85). The licensee's estimate of doses to the maximum exposed individual was 75 mrem through April 30 for a location of 1200 meters NNE, while that of the Ad Hoc Committee was 83 mrem through April 7 for a location 800 meters ENE (Ref. 65, 84, 85). Although different time frames are referenced in these two estimates, they are fairly comparable (in time) since, based on estimates made by the licensee, greater than 99% of the noble gas releases through April 30 occurred by April 7 (Ref. 84, 85).

While the investigators did not perform a detailed, independent assessment of individual or population doses calculated by the licensee and by the Ad Hoc Committee, a review of environmental radiation levels due to noble gas releases was made (Ref. 65, 84, 85). Special attention was paid to those periods of time when the licensee did not, in the judgment of the investigators, perform a sufficient number of surveys in a given area and other data suggested that relatively high radiation levels were likely to be present. One of these time periods was from 1700-2400 on March 28 in the northwest and north-northwest sectors (see Section 3.3.3.1). The dose received by an individual residing continuously in this area, based on the isodose contours in Appendix A of the Ad Hoc Committee Report, was estimated to be 100-200 mrem during the period from March 28 to April 3 (Ref. 65). A significant fraction of this dose was

likely to have been received during the evening of March 28. This report (Ref. 65) also states that the dose rate measurements performed by DOE (which form the basis for the dose contours) are likely to be conservative by at least a factor of three, so that the actual dose was probably less than 100 mrem.



REFERENCES



REFERENCES

1. Letter dated March 5, 1979 from Metropolitan Edison to NRC
2. Computer Printout, Emergency Training/Qualification, Status as of 3-1-79
3. "Radiation Monitoring System Alarm Responses, Three Mile Island Nuclear Station, Unit 2," Revision 2, dated November 9, 1978
4. Procedure 1670.9, "Emergency Training and Emergency Exercise," Revision 6, dated February 3, 1979
5. Training Program Administrative Forms, 1978 and First Quarter 1979 - Emergency Plan Training
6. 1978 Listing of Shift Maintenance Personnel
7. Generation Division, TMI Station Resume
8. 1977 SAM-2 Training Form
9. Lesson Plans - EP Training
10. Record of Telephone Conversation Between the Group Supervisor - Technical Training and the Supervisor, Mechanical Maintenance, dated February 1, 1979
11. Memorandum from Supervisor of Training dated April 5, 1979, Subject: Preparation of Lesson Plans for Radiation Emergency Drill Training Programs - 1978
12. Memorandum of Meeting Between the Radiation Protection Supervisor, Supervisor, Technical Training and Group Supervisor, Technical Training dated September 21, 1978
13. Memorandum from Group Supervisor, Technical Training to Superintendent of Maintenance, dated November 28, 1978, Subject: REPAIR PARTY TEAM TRAINING
14. Memorandum from Group Supervisor, Technical Training to Nuclear Engineer dated November 29, 1978, Subject: HPP 1670.9
15. Memorandum from Group Supervisor, Technical Training to Nuclear Engineer, dated December 20, 1978, Subject: EMERGENCY TRAINING
16. Record of Telephone Conversation Between the Group Supervisor, Technical Training and the Supervisor, Radiation Protection and Chemistry, dated February 2, 1979

17. Record of Telephone Conversation Between Group Supervisor, Technical Training and the Superintendent of Maintenance, dated February 9, 1979
18. Procedure 1690, "Training Requirements," Revision 6 dated March 22, 1978
19. ANSI N18.1-1971 Section 5.5
20. Scenarios and Observer Sheets, Radiation Emergency Drills Conducted During 1978
21. Radiation Emergency Drill Critique Individual Action Item Assignment Sheets for 1978 Radiation Emergency Drills
22. Memorandum from Radiation Protection Supervisor to All Departments dated October 13, 1978, Subject: ACCOUNTABILITY DURING A RADIATION EMERGENCY
23. Surveillance Performance Form, Emergency Equipment and Inventory Control, dated March 11, 1979
24. Radiation Monitoring System Test - Area Monitors, TP 360/1C
25. Radiation Monitoring System Test - (Liquid), TP-360/1B
26. Radiation Monitoring System Test - Atmospheric Monitors, TP 360/1A
27. Surveillance Performance Form and Waste Gas Decay Tank Monitor Calibration Data Sheets for Procedure 2322-Q1 on February 9, 1979
28. Surveillance Performance Form and Liquid Effluent Radiation Monitor Data Sheets for Procedure 2322-2Q dated February 13, 1979
29. Surveillance Performance Form and Post Accident Monitoring Instrumentation Channel Check Data Sheets for Procedure 2301-M1 dated on February 28, 1979
30. Surveillance Performance Form and Waste Gas Unit Vent Functional Test Data Sheets for Procedure 2322-A1 dated September 8, 1978
31. RMS Instrumentation Channel Check Data Sheets for Procedure 2301-S1 dated March 22-28, 1979
32. Radiation Monitoring System Data Sheets for Procedure 2612-R5 dated March 20-26, 1979
33. Surveillance Performance Form and Radiation Monitoring System Atmospheric Monitor Calibration Data Sheets for Procedure 2612-R2 dated on March 4-5, 1979 and October 14, 1978

34. Surveillance Performance Form and Containment Monitor Functional Test Data Sheets for Procedure 2303-M9 dated on March 1, 1979
35. Surveillance Performance Form and Radiation Monitoring System Monthly Test Data Sheets for Procedure 1303-4.15 dated on March 18, 1979
36. Surveillance Performance Form and Radiation Monitoring System Weekly Surveillance Check Data Sheets for Procedure 1301-4.1 dated on March 17, 1979
37. Surveillance Performance Form and Radiation Monitoring System Data Sheets for Procedure 1302-3.1 dated on January 10, 1979
38. Surveillance Performance Form and Liquid Radiation Monitor Calibration Data Sheets for Procedure 2612-R3 dated March 4, 1979, October 20, 1978 and January 4, 1979
39. Surveillance Performance Form and Radiation Monitoring System Data Sheets for Procedure 1302-3.11 dated March 19, 1979
40. Surveillance Performance Form and Weekly Surveillance Check Data Sheets dated March 24, 1979
41. Unit 2 Out-of-Service Log
42. Unit 1 Out-of-Service Log
43. Work Request 0579
44. Operation, Maintenance Instructions and Parts Catalog for the Three Mile Island Nuclear Power Plant Unit 2, "Operating Instructions for Radiation Monitoring Systems," E-07582, 1976
45. Engineering Change Memo No. S-9137, Problem Report No. 2699 and Memo dated July 28, 1978
46. Engineering Change Memos 4722 R1, R2 and R3
47. Field Questionnaire 1597; Letter dated August 7, 1978 and Memo dated January 15, 1975, and Technical Memo 668 dated March 18, 1975
48. Work Request 2761
49. Procedure 2105-1.12, "Radiation Monitoring System Setpoints," Revision 6 dated November 9, 1978

50. Primary Auxiliary Operators Log dated March 29, 1979
51. Procedure 2104-4.1, "Miscellaneous Liquid Radwaste Disposal," Revision 2, dated March 14, 1979
52. Filter Efficiency Test (Ventilation Systems)
53. System Description #36 Heating and Ventilation System, Auxiliary Building

54. Work Request C1406 dated February 25, 1979
55. Information Provided by Met-Ed - Communications Systems and Maintenance
56. NUS Audit 3364 and cover letter HPS-79-053
57. Procedure 1778, "Emergency Equipment and Inventory Control," Revision 4, dated January 26, 1978
58. Procedure 1302-5.24, "Environmental Monitor Calibration," Revision 3, dated December 19, 1974
59. Surveillance Task Index dated June 19, 1978, Page 16
60. Unit 1 Procedure Index dated July 16, 1979
61. TMI Environmental Sample Collection Sample Sheet dated March 22, 1979
62. Form 1660-1, TMI - Radiological Environmental Monitoring Program - Surface Water dated March 1, 1979
63. Work and Station Status Sheet dated March 31, 1979
64. Work and Station Status Sheet dated March 29, 1979
65. NUREG-0558, "Population Dose and Health Impact of the Accident at the Three Mile Island Nuclear Station," Ad Hoc Population Dose Assessment Group (NRC, EPA and HEW), May 10, 1979
66. Metropolitan Edison Company, NRC Form-5 Request for Area TLD Badges, dated July 5, 1979
67. Three Mile Island Unit 2, Surveillance Procedure 2302-SA1, "Meteorological Instrumentation Calibration," Revision 1, 3/13/78
68. Metropolitan Edison Company, Generation Maintenance System, Manual Performance Form, Procedure No. 2302-SA1, dated 10/26/78

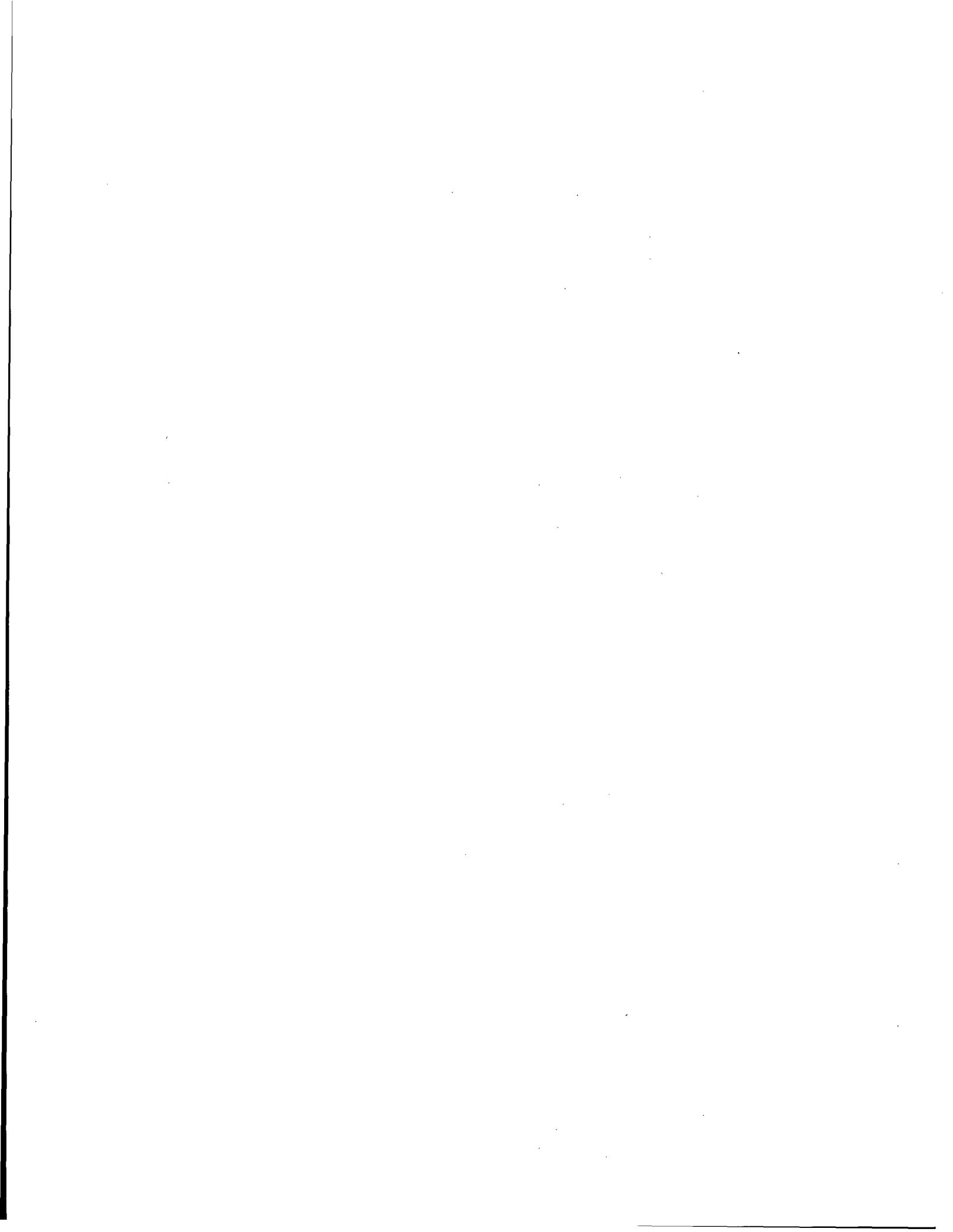
69. Unit 1 Radiation Monitor Strip Chart from Recorder RMR-2
70. Offsite Dose Calculation Logs (Enclosures 2, 3 and 4 of Procedure 1670.4)
71. Procedure 1670.2, "Site Emergency Procedure," Revision 9 dated November 22, 1978
72. Station Manager Testimony
73. Information Transcribed From Unit 2 Emergency Status Board
74. Telephone Message Forms, NRC Region I Answering Service, dated March 28, 1979
75. Bell of Pennsylvania Itemized Call Listing, Three Mile Island, dated April 13, 1979, for Calls Made on March 28, 1979
76. Dauphin County Civil Defense Draft Three Mile Island, Incident Log Covering the Period March 28, 1979 through April 2, 1979
77. Appendix II-A, Item 169
78. Notes of Commonwealth of Pennsylvania Nuclear Engineer
79. Appendix II-A, Item 286
80. Memo dated June 7, 1979, C. O. Gallina, Investigation Specialist, to A. F. Gibson, Leader, TMI Radiological Investigation Team, Subject: RADIOACTIVE GASES RELEASED FROM TMI ON THE MORNING OF MARCH 30, 1979
81. Appendix I-A, Time 0
82. Appendix I-A, Time 3-6 sec
83. Appendix I-A, Time 13 sec
84. Second Interim Report on the Three Mile Island Nuclear Station Unit 2 (TMI-2) Accident, June 15, 1979
85. Third Interim Report on the Three Mile Island Nuclear Station Unit 2 (TMI-2) Accident, July 16, 1979
86. Appendix II-A Item 8
87. Acceptance Test Data

88. Burns and Roe Drawing 2045, Revision 19
89. Informal Record of Tank Levels on March 29
90. Unit 2 Chemistry Log
91. Unit 2 Chemistry Log
92. Burns and Roe Drawing 2487
93. Burns and Roe Drawing 2485-WDL-551, Revision 0
94. Burns and Roe Drawing 2064, Revision 16
95. Strip Chart for Unit 2 Reactor Building Pressure
96. Burns and Roe Drawing 2632, Revision 9
97. System Description
98. Appendix I-A, Time 3 hrs. 56 min.
99. Unit 2 Radiation Monitor Strip Chart from Recorder HP-UR-1907
100. Valve List, Issue 6, dated 6/29/77
101. Appendix I-A, Time 87 min.
102. Radiation Monitor Strip Chart from Recorder HP-UR-3236
103. Appendix I-A, Time 60 min.
104. Appendix I-A, Time 2 hrs. 56 min.
105. Appendix I-A, Time 6 hrs. 53 min.
106. Appendix I-A, Time 3 hrs. 4 min.
107. Condensate Pump Discharge Sample Analysis Sheet 3/28 1500
108. Steam Generator A Sample Analysis Sheet 3/28 0724
109. Condensor Vacuum Pump Discharge Sample Analysis Sheet 3/28 0542
110. Steam Generator B Sample Analysis, 3/30, 1030
111. Appendix I-A, Time 5 hrs. 15 min.
112. Appendix I-A, Time 9 hrs. 15 min.
113. Appendix II-A, Item 177
114. Industrial Waste Treatment System Sample Results
115. Unit 2 Radiation Monitor Strip Chart from Recorder HP-UR-1901
116. Burns and Roe Drawing 2062

117. Unit 2 Radiation Monitor Strip Chart from Recorder HP-UR-2900
118. Unit 2 Radiation Monitor Strip Chart from Recorder HP-UR-1902
119. System Description #17, Reactor Coolant Makeup and Purification System
120. Appendix I-A, Time 4 hours
121. Unit 2 Strip Chart for Makeup Tank Level
122. Burns and Roe Drawing 2028
123. Nash Engineering Co. Inc., "Instructions for Operation of Size 623C Waste Gas Compressor"
124. Burns and Roe Drawing 2024, Revision 25
125. Appendix I-A, 4 hrs. 26 min.
126. Appendix I-A, 9 hrs. 51 min.
127. Burns and Roe Drawing 2027, Revision 24
128. Reactor Coolant Bleed Holdup Tank Samples Analysis Sheets
129. Burns and Roe Drawing 2496, Revision 10
130. Maintenance Work Request Log
131. Unit 2 Reactor Coolant Analysis Sheets
132. Sample Analysis Sheets for HP-R-219
133. Radiation Management Corporation Counting Room Log
134. Appendix II-A, Item 131
135. Appendix II-A, Item 194
136. "Analysis of the Absorber and Absorbents from Three Mile Island Unit 2" dated May 25, 1979, by Nuclear Consulting Services, Inc.
137. Not used
138. Industrial Waste Treatment System Log
139. Industrial Waste Treatment System Log Sample Result Sheets
140. Appendix II-A, Item 249
141. NRC RI Incident Message Forms
142. Appendix II-A, Item 25
143. Appendix I-A, Time 0521

144. Results for reactor coolant sample taken at 0643 on March 28, 1979
145. Appendix II-A, Item 33
146. Unit 1 Shift Foreman Log
147. Control Room Operator J's Notes
148. Procedure 1670.8, "Emergency Re-entry for Repair or Rescue," Revision 0 dated January 16, 1978
149. Appendix II-A, Item 21
150. Shift Supervisor C's Notes
151. Procedure 2202-2.6, "OTSG Tube Rupture," Revision 4 dated October 6, 1978
152. Unit 1 Radiation Monitor Strip Chart from Recorder Unit 1 RMR-5
153. Emergency Control Center (ECC) Log
154. Unit 1 Radiation Monitor Strip Chart from Recorder RMR-1
155. Results for reactor coolant sample taken at 0850 on March 28, 1979
156. NRC Incident Messageform C5
157. TLD Printout for 1-1 to 7-5-79 Background Badges
158. Procedure 1616, "Use of Respiratory Protection Devices," Revision 13 dated April 21, 1979
159. Radiation Work Permit Log
160. 1st Quarter Licensee TLD Printout
161. Procedure 1642, "Operation and Calibration of the Thermoluminescent Dosimetry System," Revision 1 dated September 28, 1977
162. Teletector Technical Manual for Total
163. NRC Memorandum: M. Slobodien (Radiation Specialist) to G. Smith (Director, Health Physics Operations), "Analysis of Personnel Dosimetry at TMI since March 28, 1979: dated May 16, 1979
164. TLD data sheets dated May 21, 1979
165. Bettis Sample Results
166. Three Mile Island Radiation Emergency Procedure 1670.2, "Site Emergency Procedures," Revision 9 dated November 22, 1978

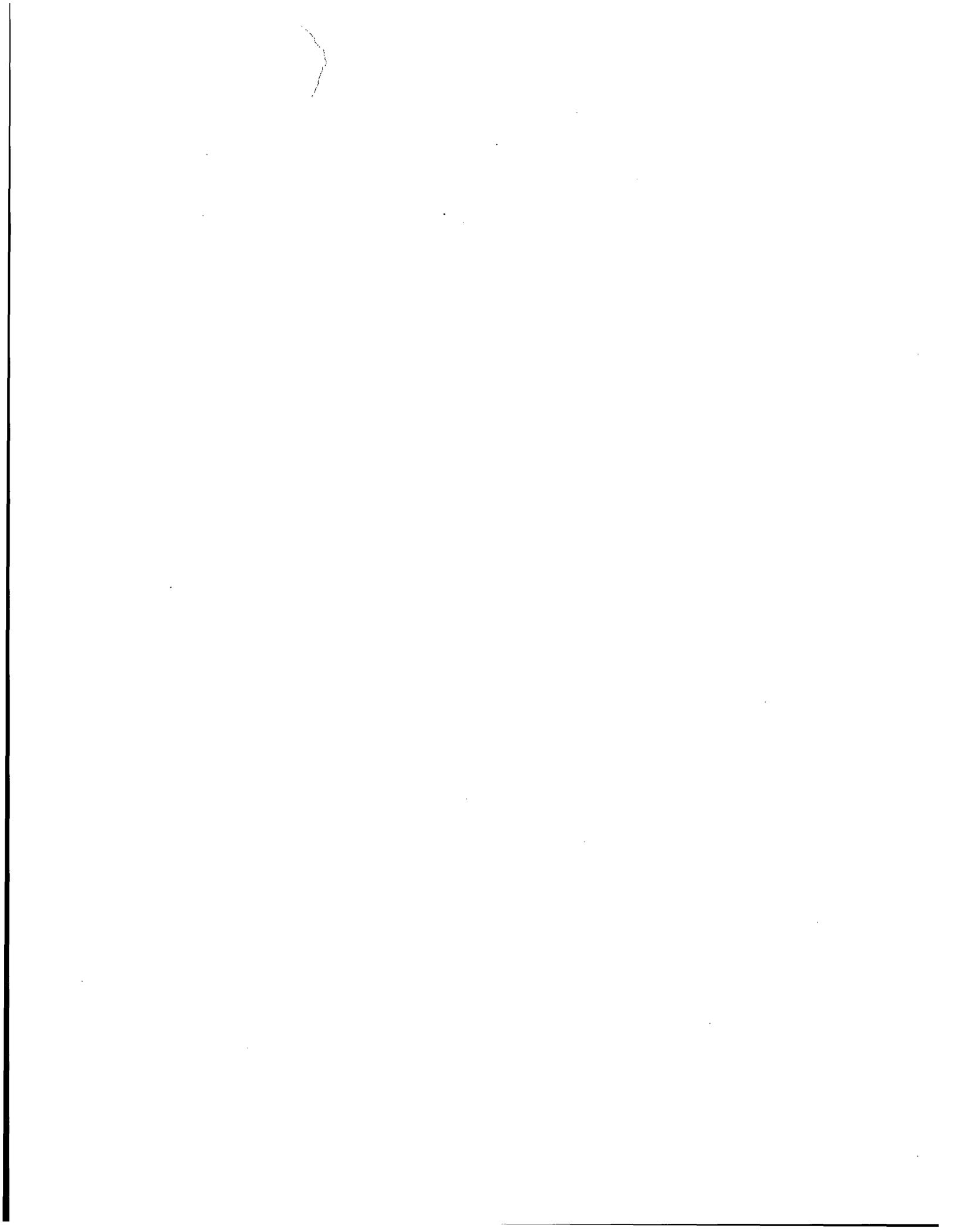
167. Three Mile Island Radiation Emergency Procedure 1670.4, "Radiological Dose Calculations," Revision 3 dated February 15, 1979
168. Offsite Dose Calculation Log (Enclosures 2, 3 and 4 of Procedure 1670.4)
169. Emergency Control Station (ECS) Survey Log
170. Pennsylvania Bureau of Radiological Health Sample and Laboratory Report Sheets for Samples Analyzed 3/28/79
171. TMI-1 Wind Speed and Direction Chart (Recorder #42B)
172. ECS Wind Data Summary Log
173. Radiation Management Corporation Mobile Laboratory Sample Analysis Sheets dated 3/28 - 3/30/79
174. Undated Memorandum from Radiation Protection Supervisor, Relative to Radiation Survey Records Following the March 28 Event
175. NRC Region I Incident Message Form
176. Three Mile Island Health Physics Procedure 1670.6, "Off-site Radiological Monitoring," Revision 2, dated January 16, 1978
177. NRC Region I Mobile Laboratory Sample Analysis Log - Analytical Results dated 3/28 - 3/30/79
178. Radiation Management Corporation/Philadelphia Laboratory Analytical Result dates 3/28/79 (received via telephone by Met-Ed)
179. Preaccident Unit 2 Auxiliary Building Surveys
180. Metropolitan Edison Company Radiological Environmental Monitoring Report - 1978 Annual Report (prepared by Teledyne Isotopes, Westwood, NJ)
181. Procedure 1643, "Quality Assurance Tests for TLD Dosimetry System," Revision 0 dated April 11, 1977
182. TLD Calibration Data
183. Memorandum NBS to NRC (in preparation)



Appendix II-A
Radiological Sequence
of
Events

Notes:

- (1) All events pertain
to Unit 2 unless
otherwise specified
- (2) Times are approximate



TMI-2 RADIOLOGICAL SEQUENCE OF EVENTS
AS OF JULY 11, 1979

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCES</u>
	3/28/79		
1	0400	The reactor tripped.	1. Alarm printer
2	0400-0618	Reactor coolant pressure dropped from 2435 psig just after the reactor trip to 1600 psig at 0402 resulting in emergency core cooling system initiation. The pressure continued to drop until it reached approximately 1015 psig at 0420 where it remained relatively constant until about 0530 when it decreased again reaching a low point of 660 psig at 0618.	1. Metropolitan Edison Interim Report Figure 12 2. RC wide range pressure recorder
3	0408	The reactor building sump pump (WDL-P-2A) started.	1. Alarm printer
4	0410	A second reactor building sump pump (WDL-P-2B) started.	1. Alarm printer
5	0411	There was a reactor building sump high water level alarm.	1. Alarm printer
6	0415	The reactor coolant drain tank rupture disk burst at approximately 190 psig. A pressure rise of 1.4 psig was measured in the reactor building.	1. Second Interim Report Figure 45
7	0426	The steam driven emergency feedwater pump was turned off.	1. NRC Operational Sequence of Events
8	0438	An operator manually turned off the reactor building sump pumps after up to 8,120 gallons had been pumped to the auxiliary building.	1. Alarm printer 2. Interview of Auxiliary

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		Operator B (Int. 10) 3. Primary Auxiliary Operator Log (OP 2104-4.1)

NOTE: The quantity 8120 gallons was calculated by multiplying the run time of each sump pump by 140 gpm (the capacity of each pump operating singularly as determined by preoperational testing). With two pumps operating, the total capacity should be less than 280 gpm. In addition, since the discharge of these pumps was probably to the auxiliary building sump tank instead of to the miscellaneous waste holdup tank, as it was during preoperational testing, the line size is reduced from 4 in. to 2 in. This would also result in a reduced flow rate. By procedure, the sump pumps are normally aligned to the miscellaneous waste holdup tank. The miscellaneous waste holdup tank level, however, did not appear to change significantly from 3/27 to 3/30. The primary auxiliary operator on shift at the time of the reactor trip stated that the reactor building sump pump discharge was aligned to the miscellaneous waste holdup tank; however, it appears likely that the discharge was actually aligned to the auxiliary building sump tank. Procedural guidance allows the sump pumps to be aligned to the auxiliary building sump tank if the activity of the sump is low, if the miscellaneous waste holdup tank is full, or if both sump filters (WDL-F-8A/B) are at high differential pressures.

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
9	0445-0515	A radiation/chemistry technician took a reactor coolant sample. The boron analysis result was approximately 700 ppm. He phoned the results to the control room. The control room asked for another sample and analysis. A radiation level survey of the sample was not made.	1. Interview of Radiation/Chemistry Technician L (Int. 58)
10	0450-0500	The reactor intermediate closed cooling water coolant letdown monitor (IC-R-1091) increased from 2000 cpm to 3000 cpm. NOTE: This indicates an increase in background radiation in the reactor building.	1. Chart recorder HP-UR-3264
11	0500	The Supervisor, Radiation Protection and Chemistry was phoned at home by the Unit 2 Superintendent, Technical Support, and told to report to the site.	1. Interview of Supervisor, Radiation Protection and Chemistry and Unit 2 Superintendent Technical Support (Int. 20 and 27)
12	0500	The circulating water pumps were turned off to switch the steam generators onto the atmospheric relief valves. Emergency feedwater pump suction was to the hotwell.	1. NRC Operational Sequence of Events

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
13	0515-0545	The intermediate closed cooling water monitor (IC-R-1092) increased from 3,500 cpm to 22,000 cpm. It reached the alarm setpoint (5000 cpm) at 0518.	<ol style="list-style-type: none"> 1. Chart recorder HP-UR-3236 2. Procedure 2105-1.12
		NOTE: This indicates an increase in background radiation in the reactor building.	
14	0518	The reactor building air sample particulate monitor (HP-R-227 (P)) reached the alarm setpoint (50,000 cpm).	<ol style="list-style-type: none"> 1. Chart recorder HP-UR-3236 2. Procedure 2105-1.12
		NOTE: This alarm corresponded to 9.4 E-8 uCi/cc. The licensee established this setpoint at 25% of an equilibrium concentration of krypton-88 and rubidium-88 expected to be present in the reactor building due to a 1 gpm reactor coolant leak with 0.1% failed fuel.	
15	0527	A control room operator isolated the B steam generator because operators suspected that a steam leak from this generator was causing an increase in reactor building pressure.	<ol style="list-style-type: none"> 1. Unit 2 Foreman Log
16	0540	The Supervisor, Radiation Protection and Chemistry arrived on site.	<ol style="list-style-type: none"> 1. Blue Badge Access List 2. Interview of Supervisor, Radiation Protection and Chemistry (Int. 20)

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
17	0542	The condenser vacuum pump exhaust was sampled. The results showed no radioactivity above background.	1. Licensee sample result
18	0545	Radiation/chemistry technicians took a reactor coolant sample. The analysis result showed a boron concentration of approximately 400 ppm. A radiation level survey of the sample was not made. The result was phoned to the shift foreman. Another radiation/chemistry technician analyzed a separate sample to confirm the results.	1. Interviews of Supervisor, Radiation Protection and Chemistry Technicians L and I (Int. 20, 58, 42)
19	0545	The Supervisor, Radiation Protection and Chemistry arrived at the Unit 2 control room and was requested to change the charcoal cartridge and particulate filter in the reactor building air sample monitor (HP-R-227).	1. Interview of Supervisor, Radiation Protection and Chemistry (Int. 20)
20	0550	A radiation/chemistry technician attempted to change the iodine cartridge and particulate filter in the reactor building air monitor (HP-R-227), but the charcoal cartridge and water blew out under pressure. The charcoal was reinstalled. This indicated the possibility of a steam environment in the reactor building atmosphere.	1. Interviews of Supervisor, Radiation Protection and Chemistry and Radiation/Chemistry Technician I (Int. 20, 42)

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
21	0602	<p>A radiation/chemistry technician took a reactor coolant sample for radio-activity analysis. A gross beta-gamma analysis showed 4.0 uCi/ml. Normal activity for this analysis should be around 0.4 uCi/ml. This result was phoned to an operator in the control room. A gamma isotopic analysis was not done on this sample. A sodium analysis showed 0.153 ppm.</p> <p>NOTE: The sodium results indicate that no appreciable amount of sodium hydroxide was present in the reactor coolant samples for which the boron results were much lower than expected.</p>	1. Interview of Radiation/Chemistry Technician L (Int. 58)
22	0615-630	A telephone conference call to discuss the conditions of the plant was placed between the Vice President, Generation; the Station Manager; the Unit 2 Technical Support Superintendent; and a Babcock and Wilcox representative.	1. Interview of Unit 2 Superintendent, Technical Support (Int. 27)
23	0618	<p>Leakage through the pressurizer electromatic relief valve was stopped by closing a block valve.</p> <p>NOTE: This stopped the loss of coolant. This valve continued to be cycled throughout the day to control pressurizer level.</p>	1. NRC Operational Sequence of Events
24	0622	The incore instrumentation area monitor (HP-R-213) appears to have detected the initial release of fuel cladding gap activity.	1. Chart Recorder HP-UR-1901
25	0624	The reactor building air particulate sample monitor (HP-R-227(P)) reached the alarm setpoint (50,000 cpm) for the second time since the incident.	<p>1. Chart recorder HP-UR-3236</p> <p>2. Procedure 2105-1.12</p>

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	NOTE: This corresponded to 9.4 E-8 uCi/cc . The monitor had been above the alarm setpoint for approximately 15 minutes from 0518 to 0533.	
26	0626	The area monitor in the reactor building on the 347 ft. elevation near the incore panel area (HP-R-213) reached the alarm setpoint (50 mR/hr). Also, the reactor containment dome monitor started to respond to the release of gap activity.	<ol style="list-style-type: none"> 1. Chart recorder HR-UR-1901 2. Procedure 2105-1.12
27	0630	The Supervisor, Radiation Protection and Chemistry, was asked to arrange for an entry into the reactor building. The entry was not made due to the shift change and subsequent increase in radiation levels.	<ol style="list-style-type: none"> 1. Interview of Supervisor, Radiation Protection and Chemistry and Radiation/Chemistry Technician I (Int. 20, 42)
28	0630	A radiation protection foreman arrived on site.	<ol style="list-style-type: none"> 1. Interview of Radiation Protection Foreman B (Int. 21) 2. Blue Badge Access List

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
29	0630-0700	A radiation/chemistry technician surveyed the Unit 2 auxiliary building using a portable beta-gamma survey meter (RO-2). He reported rapidly increasing levels of radiation during approximately twenty minutes it took to perform this survey. Initially, on the 305 ft. elevation the general area dose rates near the valve alley were 100 mR/hr, and the door to makeup purification valve room was 200 mR/hr. The door to the makeup tank room read 1 R/hr. Another radiation/chemistry technician, using a GM radiation level meter (Teletector), measured 10 R/hr just inside the makeup tank room on the nitrogen line. The dose rate in the waste gas decay tank room was 10 to 20 mR/hr, and at the radwaste operating panel area, the dose rate was 50 mR/hr. A second check on the makeup purification valve room door caused the portable beta-gamma survey meter (RO-2) to read off-scale (5 R/hr). The radiation/chemistry technician was advising personnel to leave the area when the site radiation emergency signal was sounded and an announcement was made to evacuate the Units 1 and 2 auxiliary buildings. As the technicians exited the building, the portable beta-gamma survey meter (RO-2) read off-scale at a distance of ten feet from the hallway door to the makeup tank room. This radiation/chemistry technician was contaminated to levels of 30 mR/hr.	1. Interview of Radiation/Chemistry Technician O (Int. 78)
30	0635	The containment dome monitor (HP-R-214) reached alert.	1. Chart recorder HP-UR-1901 2. Procedure 2105-1.12
31	0643	A reactor coolant sample was collected. The analysis results showed 140.73 uCi/ml gross gamma activity. The Unit 1 nuclear sample room, which contained Unit 2 reactor coolant sample lines, area radiation monitor (RM-G3) reached the alarm setpoint (2.5 mR/hr). The reactor coolant sample recirculation was stopped at the sample room by a radiation/chemistry technician.	1. Licensee sample result 2. Interview of Radiation/Chemistry Technician L (Int. 58)

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	NOTE: The alarm showed rapidly increasing radiation levels from the reactor coolant sample lines.	
32	0645	A radiation/chemistry technician was directed by the Supervisor, Radiation Protection and Chemistry to call in the Radiation Protection Supervisor and a radiation protection foreman. Both calls were completed.	1. Interview of Radiation/Chemistry Technician U (Int. 78)
33	0648	The Unit 1 hot machine shop area monitor (RM-G4) reached the alarm setpoint (2.5 mR/hr). Unshielded reactor coolant sample lines ran through this room. A survey identified 1.5 R/hr on the reactor coolant sample lines and 500 mR/hr general area dose rates in the hot machine shop. Unit 2 control room was notified.	1. Interviews of Supervisor, Radiation Protection and Chemistry, Radiation Protection Foreman B and Radiation/Chemistry Technician I (Int. 20, 21 and 42)
34	0648	The station vent stack particulate monitor (HP-R-219(P)) reached the alarm setpoint (1,300 cpm).	1. Chart recorder HP-UR-1907
		NOTE: This corresponded to 2.5 E-9 uCi/cc or approximately a release rate of 0.3 uCi/sec, the technical specification instantaneous release rate limit for iodine-131 and particulates.	

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
35	0650	<p>A radiation protection foreman surveyed the auxiliary building using a portable beta-gamma survey meter (R0-2). He entered through the model room door. The sample line to the reactor building air sample monitor (HP-R-227) read 10 mR/hr at contact. The makeup tank room door read 1 R/hr. The hall on 305 ft. and 281 ft. elevations had no unusual levels of radiation. The Unit 2 health physics control point, auxiliary building stairwell, decay heat vaults, isolated areas near drains and main halls had water 2 to 3 inches deep at drains. He waded through a few inches of water on the floor. When he exited the auxiliary building there was no contamination on his clothes or body.</p> <p>NOTE: This indicated that the water on the auxiliary building floor may not have been highly contaminated at this time.</p>	1. Interview of Radiation Protection Foreman B (Int. 21)
36	0651	<p>The auxiliary building exhaust duct B particulate monitor (HP-R-228 (P)) reached the alarm setpoint (4,000 cpm).</p> <p>NOTE: This corresponded to 7.5 E-9 uCi/cc or a release rate of 0.3 uCi/sec.</p>	1. Chart recorder HP-UR-2900 2. Procedure 2105-1.12
37	0654	Reactor coolant pump 2B was started.	1. NRC Operational Sequence of Events
38	0654	The condenser vacuum pump discharge monitor (VA-R-748) reached the alarm setpoint (2,000 cpm).	1. Chart recorder HP-UR-3236 2. Procedure 2105-1.12

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	NOTE: This corresponded to a release rate of 0.024 uCi/sec of iodine-131 and particulates, a technical specification quarterly average limit.	
39	0654	The Unit 2 Superintendent, Technical Support, directed that there be an emergency boration.	1. Interview of Unit 2 Superintendent of Technical Support (Int. 27)
40	0655	A Site Emergency was declared by the shift supervisor based on the alarms of process and area radiation monitors.	1. Interview of Unit 2 Superintendent Technical Support (Int. 27) 2. Unit 1 CRO Log 3. Emergency Status Board
41	0655	The Supervisor, Radiation Protection and Chemistry put a radiation/chemistry technician in charge of the Emergency Control Station and directed him to form teams per the Emergency Plan. The Emergency Control Station was established at the 305 ft. elevation in the Unit 1 control building at the Unit 1 health physics control point.	1. Interview of Supervisor, Radiation Protection and Chemistry (Int. 20)
42	0655-0800	The Supervisor, Radiation Protection and Chemistry directed a radiation/chemistry technician to go to the Unit 2 auxiliary building to take radiation measurements and air samples. The technician started an air sampler near the reactor building air sample monitor (HP-R-227) and noted radiation levels were increasing. He measure 2 R/hr at the hallway door to the	1. Interview of Radiation/Chemistry Technician U (Int. 78)

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	makeup tank room. While collecting an air sample near the station vent stack monitor (HP-R-219) on the 328 ft. elevation, the dose rates began rapidly increasing from 50 to 175 mR/hr. Several local radiation alarms sounded, and the technician left the area. At this point another radiation/chemistry technician returning from the 281' elevation reported 1 to 3 in. of water on the floor on that elevation. The radiation/chemistry technician who had taken the air samples was found to be contaminated.	
43	0655	An engineer arrived at the control room and began offsite dose calculations. The containment dome monitor (HP-R-214) reading at this time was recorded as 300 R/hr by the engineer.	<ol style="list-style-type: none"> 1. Interview of Engineer C (Int. 48, 127, 174) 2. Chart recorder HP-UR-1901 3. Procedure 2105-1.12
		NOTE: This monitor was later determined by the NRC to have been initially misread. The actual reading appears to have been approximately 400 mR/hr.	
44	0655	The Unit 1 liquid discharge from the B waste evaporator storage tank was stopped. The Unit 1 gaseous discharge authorized by permit 33-79-G was stopped.	<ol style="list-style-type: none"> 1. Unit 1 CRO Log
		NOTE: Both of these releases were started and stopped intermittently during the next few days. These releases were of materials that were accumulated prior to the Unit 2 accident and the releases did not contribute significantly to total release from the site.	

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
45	0656	The circulating water pumps were restarted. This should have switched steam generators back to the turbine by-pass mode of operation. B steam generator was unisolated for 7 seconds which was long enough to put the condenser vacuum pump discharge monitor into alarm.	1. NRC Operational Sequence of Events
46	0658	The Supervisor, Radiation Protection and Chemistry moved to the radiation monitoring system panel and observed that the condenser offgas monitor (VA-R-748) was in alarm and the containment dome monitor (HP-R-214) was in alert and increasing.	1. Interview of Supervisor, Radiation Protection and Chemistry (Int. 20)
		NOTE: The condenser offgas monitor in the alarm mode indicated a potential primary to secondary system leak. The containment dome monitor in the alert mode indicated degradation of the fission product boundary.	
47	0700	The area monitor near the personnel access hatch inside the reactor building (HP-R-211) reached the alarm setpoint (50 mr/hr).	1. Chart recorder HP-UR-1901 2. Procedure 2105-1.12
48	0700	Two Emergency Repair Party Teams were formed and standing by one at the Emergency Control Station and one outside the Unit 2 control room.	1. Discussion with Maintenance Foremen B and G
49	0700	A radiation/chemistry technician set up Emergency Control Station communications. He was subsequently relieved by a radiation protection foreman.	1. Interview of Radiation/Chemistry Technician I (Int. 42)

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
50	0700-0713	The reactor coolant letdown monitor (MU-R-720) gross activity increased from $4.5 \text{ E } 4$ cpm to greater than $1 \text{ E } 6$ cpm reaching the alarm setpoint of $5 \text{ E } 5$ cpm at 0712.	<ol style="list-style-type: none"> 1. Chart recorder HP-UR-3264 2. Procedure 2105-1.12
51	0700-0730	Because the air radioactivity increased to greater than $3 \text{ E } 10$ uCi/cc, the control point for entry into the auxiliary building was moved from the doors to the auxiliary building to the hallway in the control building.	<ol style="list-style-type: none"> 1. Interview of Radiation/Chemistry Technician I (Int. 42)
52	0701	The fuel handling building exhaust particulate monitor upstream of the filters (HP-R-221A (P)) reached the alarm setpoint (8000 cpm).	<ol style="list-style-type: none"> 1. Chart recorder HP-UR-1907 2. Procedure 2105-1.12
		NOTE: This corresponded to $1.5 \text{ E } 8$ uCi/cc or 0.3 uCi/sec. This monitor was probably measuring effluent and direct radiation in the area of the detector outside the duct.	
53	0701	The fuel handling building exhaust iodine monitor downstream of filters (HP-R-221B (I)) reached the alarm setpoint (200,000 cpm).	<ol style="list-style-type: none"> 1. Chart recorder HP-UR-1907 2. Procedure 2105-1.12
		NOTE: This corresponded to $5.1 \text{ E } 8$ uCi/cc-sec or 50% of the 0.30 uCi/sec technical specification release rate limit. This monitor was probably measuring effluent and direct radiation in the area of the detector outside the duct.	

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
54	0702	The shift supervisor phoned the Pennsylvania Emergency Management Agency (formerly the Pennsylvania State Council of Civil Defense) Duty Officer and informed him of the Site Emergency and requested that the Pennsylvania Bureau of Radiological Health be notified.	<ol style="list-style-type: none"> 1. Interview of Unit Superintendent (Int. 99) 2. Emergency Status Board 3. Pennsylvania Emergency Management Agency Log
55	0704	The reactor building purge exhaust duct A particulate monitor (HP-R-225 (P)) reached the alarm setpoint (10,000 cpm).	<ol style="list-style-type: none"> 1. Chart recorder HP-UR-2900 2. Procedure 2105-1.12
		NOTE: This corresponded to 1.9 E-8 uCi/cc or 0.3 uCi/sec. This monitor was probably measuring direct radiation in the area of the detector outside the duct.	
56	0704	An engineer phoned the NRC Region I Office. The answering service received the call. The answering service attempted to contact the duty officer at home. There was no answer. The answering service paged the duty officer on his beeper. While waiting for beeper response, the answering service recalled the duty officer at home and was informed that he was enroute to the office. The beeper call was not answered (not received by duty officer's beeper). The answering service then called the Deputy Director's home and was informed that he was enroute to the office. The answering service paged the Deputy Director. His beeper did not activate.	<ol style="list-style-type: none"> 1. Interview of Station Manager (Int. 41) 2. Emergency Status Board 3. NRC telephone message 4. Bell of PA itemized call listing dated 4/13/79

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
57	0705	The Station Manager arrived at the Unit 2 control room and assumed the position of Emergency Director.	1. Interviews of Station Manager and Supervisor, Radiation Protection and Chemistry (Int. 71, 41 and 20)
58	0705	An engineer attempted to phone the Metropolitan Edison Vice President, Generation. He could not be reached via any of the numbers listed in the emergency procedures.	1. Interview of Engineers I and A (Int. 70)
		NOTE: The Vice President of Generation had been contacted earlier in the morning in Philadelphia and had participated in a conference call concerning the incident.	
59	0706	The reactor building purge exhaust duct B particulate monitor (HP-R-226 (P)) reached the alarm setpoint (10,000 cpm).	1. Chart recorder HP-UR-2900 2. Procedure 2105-1.12
		NOTE: This corresponded to 1.9 E-8 uCi/cc or 0.3 uCi/sec . This monitor was probably measuring direct radiation in the area of the detector outside the duct.	

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
60	0709	The site phoned the Department of Energy Radiological Assistance Plan Office at Brookhaven National Laboratory and informed them of the Site Emergency.	<ol style="list-style-type: none"> 1. Interview of Plant Management and Unit 1 Superintendent (Int. 41) 2. Discussion with DOE 3. Bell of PA itemized call listing dated 4/13/79
61	0709	A senior engineer attempted to contact the Manager, Generation Operations, Metropolitan Edison. He was not in the office. Subsequently, the Manager of Generation Engineering was called and notified that a Site Emergency had been declared.	<ol style="list-style-type: none"> 1. Interview of Plant Manager and Unit Superintendent (Int. 41)
62	0709	An engineer called Dauphin County Civil Defense to notify the agency of the Site Emergency.	<ol style="list-style-type: none"> 1. Discussion with Civil Defense Director
63	0710	The first offsite dose calculations were completed by an engineer. The initial calculations indicated a total body exposure rate of 40 R/hr in Goldsboro (1.3 miles west of the plant). The calculations were verified shortly thereafter by the Supervisor, Radiation Protection and Chemistry but, together with the engineer, he concluded that the calculations were overly conservative because the reactor building pressure was only about 2 psig, and hence the leak rate was much lower than assumed	<ol style="list-style-type: none"> 1. Interview of Engineer C (Int. 48, 127, 174);

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	in the source term estimates (approximately 50 psig) used in the offsite dose calculation procedure (1670.4). Following these calculations, the licensee dispatched survey teams in the downwind direction to the west side of the island (where the initial survey was made at 0748) and to Goldsboro (where the initial survey was made at 0832).	
64	0710	A Pennsylvania Bureau of Radiological Health nuclear engineer was called by the Pennsylvania Emergency Management Agency and notified that TMI had declared a Site Emergency. The engineer was instructed to call the site to obtain details.	<ol style="list-style-type: none"> 1. State Engineers Notes 2. Discussion with State Engineer
65	0712	The station vent stack gas monitor (HR-P-219 (G)) reached the alarm setpoint (10,000 cpm).	<ol style="list-style-type: none"> 1. Chart recorder HP-UR-1907 2. Procedure 2105-1.12
		NOTE: This corresponded to 2.8 E-4 uCi/cc or approximately the quarterly average technical specification limit for xenon-133 releases. This monitor sampled the air going out the vent stack; however, since the monitor was located inside the auxiliary building on the 328 ft. elevation, the monitor reading probably reflected an increase in background radiation and an increase in sample activity.	
66	0712	The Pennsylvania Bureau of Radiological Health nuclear engineer attempted to call the site but was not able to get through the switchboard to the control room. He left his phone number with the switchboard operator and asked to have the control room call him.	<ol style="list-style-type: none"> 1. State Engineer's Notes

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
67	0713	An individual at the site phoned Radiation Management Corporation, the licensee's health physics consultant firm. The phone was answered by the Philadelphia Electric Company load dispatcher.	1. Interview of Engineers I and A (Int. 70)
68	0713	Reactor coolant pump 2B stopped.	1. Alarm printer
69	0714	The reactor building hydrogen purge duct iodine monitor (HPR-229(I)) reached the alarm setpoint (4 E 5 cpm). This monitor was adjacent to the vent monitor (HP-R-219) on the 328 ft. elevation and was probably responding to increasing background radiation. There was no known flow from this system.	1. Chart recorder HP-UR-1907 2. Procedure 2105-1.12
		NOTE: The strip chart for reactor building hydrogen purge flow showed intermittent flow of up to 15 cfm on the morning of 3/28. Discussions with licensee operating personnel and instrument and controls personnel indicate that the flow recorder was detecting pressure changes caused by varying flows in the auxiliary building and fuel handling building ventilation systems and was not indicating purge flow. For leakage to occur through this path, four butterfly valves would have had to leak. The radiation monitor samples this exhaust downstream of the four valves.	
70	0715	The shift supervisor returned the call to the Pennsylvania Bureau of Radiological Health nuclear engineer and informed him that a transient had occurred, the plant was shutdown, safeguards were operational, and there was a slight pressure increase in reactor building but no offsite releases.	1. Discussion with State engineer 2. Interview of the Plant Manager and Station Superintendent (Int. 99)

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
71	0715	A radiation/chemistry technician measured dose rates of 1 to 2 R/hr fifteen feet from the reactor building personnel access hatch.	1. Interview of Radiation/Chemistry Technician L (Int. 58)
72	0715	A rapid increase in readings of most radiation monitors on stripcharts HP-UR-1907, HP-UR-1901 and HP-UR-2900 is attributed to the EMOV being opened. The pathway was probably from the reactor coolant drain tank to the vent header and out through a leak in the vent header into the auxiliary building. NOTE: This is a normal access route. The high level exposure rate may have been due to noble gases.	1. NRC Operational Sequence of Events
73	0715	The Emergency Director assigned an auxiliary operator to attend the telephone switchboard.	1. Station Manager testimony
74	0715-0719	A radiation/chemistry technician toured the auxiliary building and informed emergency workers (Repair Party and Monitoring Team) in the building to evacuate. He phoned the control room to inform them that the auxiliary building had been evacuated.	1. Interview of Radiation/Chemistry Technician I (Int. 42)
75	0716	The auxiliary building exhaust duct B gas monitor (HR-P-228(G)) reached the alarm setpoint (20,000 cpm). NOTE: This corresponded to 6.8 E-4 uCi/cc or approximately the quarterly average technical specification limit for xenon-133 releases. The monitor was probably measuring effluent and direct radiation in the area of the detector outside the duct.	1. Chart recorder HP-UR-2900 2. Procedure 2105-1.12

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
76	0718	The reactor building purge exhaust duct B gas monitor (HP-R-226(G)) reached the alarm setpoint (80,000 cpm).	1. Chart recorder HP-UR-2900 2. Procedure 2105-1.12
		NOTE: This corresponded to 2.3 E-3 uCi/cc or approximately the technical specification limit for xenon-133 releases. The monitor was probably measuring direct radiation in the area of the detector outside the duct.	
77	0718	The Pennsylvania State Police was contacted by an individual in the control room.	1. Interview of Station Manager and Unit Super- tendent (Int. 41)
78	0719	The reactor building purge exhaust duct A gas monitor (HP-R-225(G)) reached the alarm setpoint (80,000 cpm).	1. Chart recorder HP-UR-2900 2. Procedure 2105-1.12
		NOTE: This corresponded to 2.3 E-3 uCi/cc or approximately the quarterly average technical specification limit for xenon-133 releases. This monitor was probably measuring direct radiation in the area of the detector outside the duct.	
79	0719	The reactor building purge unit area monitor (HP-R-3236) reached the alarm setpoint (20 mR/hr).	1. Chart recorder HP-UR-1902 2. Procedure 2105-1.12

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
80	0720	The site phoned American Nuclear Insurers. There was no answer to the phone call.	1. Interview of Station Manager and Unit Superintendent (Int. 41)
81	0720	The Unit 1 fuel handling building air particulate monitor (RM-A4) reached the alarm setpoint (1 E-8 uCi/cc). NOTE: This monitor was probably measuring radioactive materials spreading from Unit 2.	1. Unit 1 Foreman Log
82	0720	The reactor containment dome monitor (HP-R-214) increased to 8 R/hr. NOTE: This monitor is an environmentally protected ionization chamber located on top of the elevator shaft enclosure. It has a 2 inch thick lead shield to which the manufacturer has assigned an attenuation factor of 100. This means that an indicated 8 R/hr was actually 800 R/hr. Because of the shielding, the monitor does not measure the radiation from xenon-133, which has a very low energy gamma, 80 kev.	1. Chart recorder HP-UR-1901
83	0723	The fuel handling building exhaust gas monitor downstream of filters (HP-R-221B(G)) reached the alarm setpoint (30,000 cpm).	1. Chart recorder HP-UR-1907 2. Procedure 2105-1.12

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	NOTE: This corresponded to 8.5 E-4 uCi/cc or approximately the quarterly average technical specification limit for xenon-133 releases. This monitor was probably measuring effluent and direct radiation in the area of the detector outside the duct.	
84	0723	The fuel handling building exhaust unit area monitor (HP-R-3240) reached the alarm setpoint (20 mR/hr).	1. Chart recorder HP-UR-1902 2. Procedure 2105-1.12
85	0724	A General Emergency was declared by the Station Manager based on the greater than 8 R/hr reading on the reactor containment dome monitor (HP-R-214). This reading corresponded to 800 R/hr when corrected for shielding around the detector.	1. Emergency Status Board
86	0724	The Unit 2 A steam generator was sampled in the Unit 1 sample room; the results recorded as the B steam generator included: cobalt-58 - 1.5 E-6 uCi/ml xenon-135 - 1.8 E-7 uCi/ml The results were phoned to the control room. NOTE: It appears the A and B sample point identifications in the Unit 1 sample room were reversed.	1. Licensee sample result 2. Interviews of Chemistry Foreman B and Supervisor, Radiation Protection and Chemistry (Int. 39, 20)
87	0725	The fuel handling building exhaust gas monitor (HPR-221A(G)) upstream of filter reached the alarm setpoint (40,000 cpm).	1. Chart recorder HP-UR-1907 2. Procedure 2105-1.12

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	NOTE: This corresponded to 1.1 E-3 uCi/cc or approximately the quarterly average technical specification limit for xenon-133 releases. This monitor was probably measuring effluent and direct radiation in the area of the detector outside the duct.	
88	0728	The auxiliary building exhaust duct A gas monitor (HPR-222(G)) reached the alarm setpoint (20,000 cpm).	<ol style="list-style-type: none"> 1. Chart recorder HP-UR-2900 2. Procedure 2105-1.12
		NOTE: This corresponded to 5.6 E-4 uCi/cc or approximately the quarterly average limit for xenon-133 releases. This monitor was probably measuring effluent and direct radiation in the area of the detector outside the duct.	
89	0729	The reactor building purge exhaust duct A iodine monitor (HP-R-225(I)) reached the alarm setpoint (200,000 cpm).	<ol style="list-style-type: none"> 1. Chart recorder HP-UR-2900 2. Procedure 2105-1.12
		NOTE: This corresponded to $5.1 \text{ E-8 uCi/cc-sec}$ or the technical specification limit of 0.3 uCi/sec . This monitor was probably measuring direct radiation in the area of the detector outside the duct.	
90	0730	The first onsite survey team, designated as alpha, and consisting of two radiation/chemistry technicians, was dispatched to the west side of the island (downwind direction). A radiation survey was performed at 0748 near the screen house.	<ol style="list-style-type: none"> 1. ECC Log 2. Interview of Radiation/Chemistry Technician D (Int. 82)

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
91	0730	An engineer recalled the Dauphin County Civil Defense to inform them of the General Emergency.	1. Discussion with Civil Defense Director
92	0730	A plant liquid effluent sample was taken from sample point at the plant liquid effluent discharge monitor (RM-L7).	1. Interview of Radiation Protection Foreman B and Radiation/Chemistry Technician C (Int. 21 and 82)
		NOTE: This sample may not have been analyzed. No record is available.	
93	0730	While a radiation/chemistry technician held a survey meter on the sample line on the reactor building air sample monitor (HP-R-227) the reading went from 50 mR/hr to 1 R/hr. A radiation protection foreman phoned the control room and requested that the sample flow be turned off.	1. Interview of Radiation Protection Foreman B (Int. 21)
94	0730	Offsite survey team "charlie," consisting of a radiation/chemistry technician and an auxiliary operator, was dispatched to Goldsboro. The initial survey was performed at this location at 0832.	1. Interview of Radiation/Chemistry Technician C (Int. 82)
95	0730	An engineer was assigned to watch the station vent monitor (HP-R-219) and reactor containment dome monitor (HP-R-214). The engineer performed this duty until about 1100.	1. Interview of Engineer H (Int. 97)

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
96	0731	The auxiliary building access corridor area monitor (HP-R-232) reached the alarm setpoint (2 mR/hr).	<ol style="list-style-type: none"> 1. Chart recorder HP-UR-1902 2. Procedure 2105-1.12
97	0734	An onsite survey team was sent to survey the Unit 2 auxiliary building. The results reported were: <ol style="list-style-type: none"> 1. The spent fuel resin area located at the 305 ft. elevation had very high dose rates. The GM radiation level survey meter (Teletector) moved toward full scale on the 1000 R/hr scale. 2. Outside the makeup tank room the dose rates were 1 R/hr. 3. The spent resin storage tanks read 3 R/hr (normally the dose rates were 0.1 mR/hr). <p>NOTE: Since the resins had not been used, this indicated the radiation was not from the resin.</p> <ol style="list-style-type: none"> 4. The reactor building air sample monitor (HP-R-227) was alarming. 5. There was water on the floor of the 281 ft. elevation. 	<ol style="list-style-type: none"> 1. Interviews of Radiation Protection Foremen A and D (Int. 11 and 18) 2. ECC Log

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
98	0735	The site called the Department of Energy Radiological Assistance Plan Office to inform them of a General Emergency. The Department of Energy was subsequently requested by the Commonwealth of Pennsylvania at 1118 to provide assistance.	<ol style="list-style-type: none"> 1. Interview of Station Manager and Unit Superintendent (Int. 99) 2. Bell of PA phone list dated 4/13/79 3. Discussion with DOE
99	0735	The station vent stack iodine monitor (HP-R-219(I)) reached the alarm setpoint (160,000 cpm).	<ol style="list-style-type: none"> 1. Chart recorder HP-UR-1907 2. Procedure 2105-1.12
		NOTE: This corresponded to 2.8 E-7 uCi/cc-sec or the technical specification limit of 0.3 uCi/sec.	
100	0735	A shift supervisor contacted the Pennsylvania Emergency Management Agency and reported the escalation to a General Emergency. The supervisor reported failed fuel and a small offsite release.	<ol style="list-style-type: none"> 1. Pennsylvania Emergency Management Agency Log
101	0735	The Radiation Protection Supervisor arrived on site and assumed control of the Emergency Control Station.	<ol style="list-style-type: none"> 1. Interview of Supervisor, Radiation Protection (Int. 22)

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
102	0736	The Dauphin County Civil Defense was contacted by the Supervisor, Radiation Protection and Chemistry, to verify that the Emergency Director, Dauphin County Civil Defense, had been personally notified. He spoke with the Director, Dauphin County Civil Defense.	<ol style="list-style-type: none"> 1. Interview of Supervisor, Radiation Protection and Chemistry (Int. 20) 2. Discussion with Civil Defense Director
103	0737	A Metropolitan Edison representative from Reading, Pennsylvania, phoned NRC Region I again. He left word with the answering service that a General Emergency had been declared.	<ol style="list-style-type: none"> 1. NRC telephone message
104	0737	York Haven (hydroelectric plant) was called by an individual in the control room and instructed to turn on the radiation survey meter and to start taking readings.	<ol style="list-style-type: none"> 1. Interview of Station Manager and Unit Superintendent (Int. 41)
105	0737	The reactor building purge exhaust duct B iodine monitor (HPR-226(I)) reached the alarm setpoint (20,000 cpm).	<ol style="list-style-type: none"> 1. Chart recorder HP-UR-2900 2. Procedure 2105-1.12

NOTE: This corresponded to $5.1 \text{ E-8 uCi/cc-sec}$ or the technical specification limit of 0.3 uCi/sec . This monitor was probably measuring direct radiation in the area of the detector outside the duct.

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
106	0738	The NRC Region I duty officer was paged for a second time. The duty officer's beeper activated. He was approximately 5 minutes from the Region I office.	1. Discussion with NRC Duty Officer
107	0740	A representative of the Pennsylvania Bureau of Radiological Health called the control room on a direct outside line using a phone number which had been previously given to the Commonwealth for such events. The Unit 2 Superintendent, Technical Support, briefed Pennsylvania Bureau of Radiological Health on the situation. The phone line was left open.	1. Interview of Unit Technical Support Superintendent (Int. 27)
108	0740	The site phoned the NRC, Region I, to inform them that a General Emergency had been declared. The answering service informed the caller that they had been unable to contact designated Region I individuals, but the regional office would open the switchboard at 0745 to receive any messages.	1. NRC telephone message 2. Interview of Station Manager and Unit Superintendent (Int. 99) 3. Bell of PA itemized call list dated 4/13/79
109	0741	The Emergency Control Station established communications with the Unit 1 control room.	1. ECC Log
110	0748	The first out-of-plant radiation surveys were made on the island. The exposure rate at the screen house (near west shore of island) was less than 1 mR/hr. This measurement was used to revise the initial prediction of offsite radiation levels in Goldsboro.	1. Emergency Status Board

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
111	0750	The NRC Region I established telephone contact with the Unit 2 control room.	1. Discussion with NRC Branch Chief
112	0755	An offsite survey team reported less than 1 mR/hr at both the north gate and Route 441 at the Observation Center.	1. Licensee notes
113	0756	The reactor building was isolated by high reactor building pressure at 4 psig.	1. Alarm printer
114	0800	Director, Region I NRC notified Acting Director, Inspection and Enforcement of the incident.	1. Discussion with Director, Region I
115	0800	A radiation/chemistry technician toured the auxiliary building basement and found water on the floor around drains, but concluded it was not highly radioactive based on direct readings with a high range beta-gamma survey instrument (Teletector). The Teletector indicated less than 1 mR/hr on the water.	1. Interview of Radiation/Chemistry Technician I (Int. 42)
116	0800	The auxiliary building and fuel handling building exhaust fans were started.	1. Licensee notes
117	0800	The auxiliary building exhaust duct B iodine monitor (HPR-228(I)) reached the alarm setpoint (100,000 cpm).	1. Chart recorder HP-UR-2900 2. Procedure 2105-1.12

NOTE: This corresponded to $2.6 \text{ E-8 uCi/cc-sec}$ or 0.3 uCi/sec .

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
118	0800	A radiation protection foreman surveyed the Unit 2 auxiliary building vent for gamma radiation with a GM survey meter, E-520 (closed window). The stack read 4 mR/hr on contact with the vent pipe.	1. Interview of Radiation Protection Foreman B (Int. 21)
119	0800	The Region I NRC Incident Response Center was activated.	1. Discussions with NRC Branch Chief
120	0800	The open phone line established between the Unit 2 control room and NRC Region I was transferred to the Region I Incident Response Center. An open line was established between Unit 1 control room and Region I Incident Response Center.	1. Discussions with NRC Branch Chief
121	0800	A survey team was dispatched to Middletown to survey a train that had passed the site after the incident.	1. Licensee notes 2. Interview of Radiation/Chemistry Technicians X, J and F (Int. 51)
		NOTE: After the team was dispatched, an individual from the control room called the train yard and said the trains could be allowed to continue without a survey.	
122	0805	The NRC Headquarters Incident Response Center was activated.	1. Discussions with NRC Branch Chief
123	0807	The model room door between the auxiliary building and the fuel handling building was recorded as closed.	1. ECC Log

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	NOTE: When open, this door was a pathway for airborne radioactivity to move from Unit 2 to Unit 1.	
124	0810	The NRC Region I Incident Response Center was fully manned and activated.	1. Discussion with NRC Branch Chief
125	0812	The result of the first air sample collected on the island outside of the plant (counted in the field using a portable sodium iodine single channel analyzer, SAM-2) was less than minimum detectable activity of 5 E-9 uCi/cc (5,000 pCi/m ³) between GE-8 and GE-9. Locations GE-8 and GE-9 are respectively 700 feet west south west and 750 feet west northwest of the Unit 2 plant vent.	1. Emergency Status Board 2. Interview with the Radiation Protection Supervisor (Int. 20)
		NOTE: The 10 CFR 20, Appendix B, concentration for iodine-131 in unrestricted areas is 1 E-10 uCi/cc.	
126	0815	Recirculation of a sample line for a reactor coolant sample was begun. Dose rate at contact with the sample lines were measured to be 200 R/hr with a high range GM survey meter (Teletector).	1. Interview of Radiation/Chemistry Technician U (Int. 78)
127	0815	Two members of the emergency Repair Party Team and a foreman entered the auxiliary building to close the model room door through which air was flowing from Unit 2 to Unit 1. It took about ten minutes. The three individuals were found to be contaminated on leaving the auxiliary building.	1. Interview of Maintenance Foreman B (Int. 21) 2. ECC Log
		NOTE: There appears to be a long-standing problem with balancing of ventilation systems to prevent flow between Unit 1 and Unit 2.	

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
128	0819	The Unit 1 fuel handling building air particulate monitor (RM-A4) alarmed. NOTE: This corresponds to 1 E-8 uCi/cc.	1. ECC Log
129	0820	The radiation protection foreman returning from the roof of the Unit 2 auxiliary building through the 305 ft. elevation became contaminated to a level of 20 mR/hr as measured with a GM survey meter (E-520), closed window. He concluded that the contamination appeared to be due to airborne radioactive materials since he was evenly contaminated from head to foot.	1. Interview of Radiation Protection Foreman B (Int. 18)
130	0826	The letdown high temperature alarms were received. NOTE: This should have isolated the letdown and purification demineralizers.	1. Alarm printer
131	0830 - 0900	The station vent stack monitor (HP-R-219) charcoal cartridge was changed by a radiation/chemistry technician. The particulate filter and holder were missing. The general area dose rate near station vent stack monitor (HP-R-219) was 2 R/hr. The technician was alone and wore a respirator with a particulate filter. The change took 2 minutes and he received a dose of 395 mrem.	1. Interview of Radiation Chemistry Technician T (Int. 75)
132	0830	Survey team "bravo" was dispatched to Goldsboro.	1. Interview of Radiation/Chemistry Technician C (Int. 82)

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
133	0830	The Station Manager, the Unit 2 Technical Support Superintendent, and the Station Operations Superintendent discussed the status of the plant and the Emergency Plan implementation. They concluded that the Emergency Plan was properly implemented and actions were being taken consistent with protecting public health and safety.	1. Interview of Unit 2 Technical Support Superintendent (Int. 27)
134	0830	NRC Region I Mobile Laboratory was recalled from Millstone Point by the Region I Incident Response Center and directed to Three Mile Island.	1. Interviews of NRC Inspector N (Int. 137) 2. NRC Incident Response Log
135	0831	The power operated emergency mainsteam dump valve (MS-V-34) was opened to induce natural circulation in steam generator A.	1. Metropolitan Edison Second Interim Report
136	0832	Survey team "charlie" reported no detectable radiation (less than 1 mR/hr) at survey point W-11 (Goldsboro). This measurement, along with the survey made on the island at 0748, confirmed that the initial prediction (made at 0710) of 40 R/hr in Goldsboro was in error.	1. ECC Log
137	0833	The waste disposal storage area monitor (HP-R-218) reached the alarm setpoint (50 mR/hr).	1. Chart recorder HP-UR-1902 2. Procedure 2105-1.12

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
138	0835	A Pennsylvania State Police helicopter landed north of the parking lot.	1. Security Log 2. Interview of Radiation Protection Foreman D (Int. 18) 3. Discussions with pilot
139	0842	Neither A nor B steam generators were steaming to the atmosphere.	1. NRC Operational Sequence of Events
140	0843	The Unit 1 nuclear sampling room monitor (RM-G3) increased to 10 R/hr. This detector is located about five feet from the Unit 2 sample lines.	1. ECC Log
141	0846	The Unit 1 hot machine shop monitor (RM-G4) read 1000 R/hr. This detector is located about six feet under the Unit 2 sample lines.	1. ECC Log
142	0847	A 5 man NRC inspection team from Region I left King of Prussia for TMI. The team consisted of an investigator, 3 reactor health physicists, and a reactor operations inspector.	1. NRC Inspector D's Notes
143	0850	A 50 to 100 ml reactor coolant sample was collected in a 250 ml bottle. A radiation/chemistry technician took 1 ml for analysis. Five mls of this sample was added to 95 ml of demineralized water for boron analysis. The boron concentration was reported as 248 ppm. The sample as drawn read 200 R/hr at six inches from the sample bottle. The results of gamma analysis were:	1. Interviews of Radiation/Chemistry Technicians U, K, and I (Int. 78, 24, and 42)

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	50 uCi/ml iodine-132 81 uCi/ml iodine-131 144 uCi/ml iodine-133 449 uCi/ml xenon-133 247 uCi/ml xenon-135 604 uCi/ml rubidium-88 29 uCi/ml krypton-85	2. NRC Region I Incident Message-form R19 3. Licensee sample results
		NOTE: This was the first reactor coolant sample drawn that had extremely high radioactivity levels although it had already been established from the radiation monitoring system readings and various health physics surveys that high activity existed in the reactor coolant as early as 0630 and was a consideration in the decision to announce a Site Emergency (0655).	
144	0855	A two man backup team from NRC Region I was dispatched.	1. Interviews of NRC Inspector A (Int. 62)
145	0900	The reactor coolant sample recirculation was shut off. The hot machine shop monitor (RM-G4) started to decrease.	1. CRO J's notes

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
146	0900	A control room bypass supply fan (AH-E-4B) was started and the Unit 2 control room was placed on recirculation air. This was the first time the control room had been placed on recirculation.	1. CRO J's notes
147	0900	An air sample taken in the southern part of Goldsboro was less than 5 E-9 uCi/cc (minimum detectable activity for iodine-131).	1. ECS Log
148	0900	The industrial waste water treatment system (IWTS) was shutdown because Unit 2 Site Emergency.	1. IWTS Log
149	0910	The primary Emergency Control Station was evacuated from the Unit 1 health physics control point to the alternate Emergency Control Station, the Unit 2 control room.	1. Licensee notes (collected by Shift Supervisor C) 2. Interviews of Radiation Protection Supervisor, Radiation Protection Supervisor B and Radiation/Chemistry Technician I (Int. 20, 21, and 42)

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	<p>NOTE: The decision to evacuate the Emergency Control Station was based on increasing air activity and general area dose rates. The GM counter rate meter (RM-14 with a HP-210 probe) at the Emergency Control Station had increased from 100 counts per minute to 3000 counts per minute. The increasing background radiation resulted in a loss of counting capability, especially the lithium drifted germanium detectors. After the loss of the lithium drifted germanium detectors, the licensee had to count samples on a less accurate instrument.</p>	
150	0915	<p>The steam generators were reported isolated.</p> <p>NOTE: The term "isolated" may only mean that the steaming generators were not steaming through the atmospheric reliefs due to conditions in the steam generators. It may not mean that the steam generators had been isolated with the purpose of reducing radioactive releases.</p>	1. NRC Operational Sequence of Events
151	0917	The Emergency Control Station was established at the Unit 2 control room.	1. ECC Log
152	0922	<p>An air sample taken in Goldsboro indicated 1 E-8 uCi/cc iodine-131.</p> <p>NOTE: 10 CFR 20, Appendix B, concentration for iodine-131 in unrestricted is 1 E-10 uCi/cc (100 pCi/m³). This was the same location as the 0900 sample.</p>	1. NRC Region I Incident Message-form R7

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
153	0930	The Metropolitan Edison Vice President, Generation, was directed by the President, Metropolitan Edison, to leave Philadelphia for TMI.	1. Interview of Vice President, Generation (Int. 91)
154	0930	American Nuclear Insurers returned the licensee's phone call.	1. Interview of Station Manager and Unit Superintendent (Int. 41)
155	0945	The fuel handling bridge area monitor (HP-R-215) located in the fuel handling building, reached the alarm setpoint (20 mR/hr).	1. Chart recorder HP-UR-1902 2. Procedure 2105-1.12
156	0945	The control and service building corridor area monitor (HP-R-234) reached the alarm setpoint (2 mR/hr).	1. Chart recorder HP-UR-1902 2. Procedure 2105-1.12
157	0948	The control room intake particulate monitor (HP-R-220 (P)) reached the alarm setpoint (300 cpm).	1. Chart recorder HP-UR-1907 2. Procedure 2105-1.12

NOTE: This corresponded to 5.7 E-10 uCi/cc . The 10 CFR 20, Appendix B, concentration for strontium-90 in restricted areas is 1 E-9 uCi/cc .

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
158	0950	An air sample taken in the northern part of Goldsboro indicated 1.2 E-8 uCi/cc of iodine-131 (1.2 E 4 pCi/m ³).	<ol style="list-style-type: none"> 1. NRC Region I Incident Message-form R7 2. Offsite Dose Calculation Log
		NOTE: The 10 CFR 20, Appendix B, concentration for iodine-131 in unrestricted areas is 1 E-10 uCi/ml (100 pCi/m ³).	
159	0950	The control room air monitor (HP-R-220) particulate, iodine, and gas channels were increasing.	<ol style="list-style-type: none"> 1. Chart recorder HP-UR-1907
160	1000	A chemistry foreman sampled secondary water from the A and B steam generators at the Unit 2 sample station. The A steam generator sample indicated slight contamination based on a qualitative gross beta-gamma measurement with a GM count rate meter (RM-14 with a HP-210 probe). The B steam generator caused the GM count rate meter to go off scale (50,000 cpm). The shift foreman was informed of these results.	<ol style="list-style-type: none"> 1. Interviews of Chemistry Foreman B and A (Int. 21, 11)
161	1010	The control room gas monitor (HP-R-220 (G)) reached the alarm setpoint (260 cpm).	<ol style="list-style-type: none"> 1. Chart recorder HP-UR-1907 2. Procedure 2105-1.12
		NOTE: This corresponded to a concentration of 7.3 E-6 uCi/cc. The 10 CFR 20, Appendix B concentration for xenon-133 is 1 E-5 uCi/cc.	
162	1010	The NRC Region I inspection team arrived on site and went to the Unit 1 control room upon direction of Station Manager.	<ol style="list-style-type: none"> 1. Security Visitors Log 2. Licensee notes

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
163	1012	The Emergency Control Station moved from the Unit 2 control room to the Unit 1 control room to reduce number of people in Unit 2 control room.	1. Licensee notes
164	1017	Control room personnel donned respiratory protection. Face masks with particulate filters were worn.	1. NRC Region I Incident Message-form C5
165	1022	An open phone line between the Unit 1 control room and the NRC Region I was established.	1. Bell of PA list of calls dated 4/13/79
166	1030	An air sample collected in the control room indicated 1 E-7 uCi/cc of beta-gamma activity. Since the isotopic content had not been determined the licensee assumed the 10 CFR 20, Appendix B, concentration was 3 E-10 uCi/cc .	1. NRC Region I Incident Message-form R3
167	1034	A two man backup team consisting of a reactor inspector and an investigator arrived on site from NRC Region I.	1. Visitors Log
168	1053	The plant was steaming from the atmospheric reliefs.	1. NRC Operational Sequence of Events
169	1055	A State Police helicopter transported an air sample (taken in Goldsboro) to Holy Spirit Hospital for pickup by Pennsylvania Bureau of Radiological Health personnel.	1. Discussion with Pilot
170	1100 - 1130	NRC inspectors began making radiation measurements on site and in the environment to check the accuracy of the licensee's results.	1. Interview of NRC Inspectors I and J (Int. 146)

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
171	1100	A liquid sample from the plant effluent monitor point (RM-L7) showed iodine-131 less than 4 E-8 uCi/ml and iodine-133 less than 1 E-8 uCi/cc. NOTE: The 10 CFR 20, Appendix B, concentration for iodine-131 in water for unrestricted areas is 3 E-7 uCi/ml.	1. Licensee sample result
172	1104	The Unit 2 ventilation system was turned off. NOTE: The NRC expressed concern to the licensee that this action might lead to ground level releases.	1. NRC Region I Incident Message-form C6
173	1104	The gross beta-gamma airborne particulate activity in the Unit 1 shift supervisor's office was reported as 1 E-7 uCi/cc. The licensee assumed the 10 CFR 20, Appendix B, concentration to be 3 E-10 uCi/cc.	1. NRC Region I Incident Message-form C6
174	1110	The island was evacuated of all non-essential personnel.	1. Security Log
175	1110	The air sample from Goldsboro was delivered to the Pennsylvania Bureau of Radiological Health.	1. Discussion with State Officials
176	1130 - 1215	Two NRC inspectors arrived in Unit 2 control room and made preliminary assessment of radiological controls based on discussions with the Supervisor, Radiation Protection and Chemistry, and direct observations of licensee activities.	1. Interview of Inspector H (Int. 40)

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
177	1130 - 1200	An NRC inspector measured 20 mR/hr at ground level in the plume from the plant. The plume was visible because it contained steam from the atmospheric dumps.	<ol style="list-style-type: none"> 1. NRC Region I Incident Message-form C12 2. Interview of NRC Inspector J (Int. 146)
178	1200	A shift foreman, an engineer and a radiation/chemistry technician entered the auxiliary building to check on three decay heat valves in preparation for starting of decay heat and to close sump pump breakers for an auxiliary building water transfer. High range pocket dosimeters were not available for entry. A survey of the seal injection filters showed 50 and 75 R/hr at 1 foot using a GM radiation level survey meter (Teletector). The general area dose rate on the 281 ft. elevation was 2 R/hr. The decay heat removal pump vault floor was dry and the general dose rate was 50 mR/hr. After about 10 minutes the individuals could not locate the desired decay heat valves and left for the 328 ft. elevation. The dose rate at the door located at the west stair entrance to the 328 ft. elevation read 1 R/hr, 5 R/hr just inside the door, 100 R/hr near the air filter system, and 1000 R/hr 8-10 ft. from the equipment hatch. The individuals left the area without closing the sump pump breakers. These individuals received about 800 mrem each during this entry.	<ol style="list-style-type: none"> 1. Interview of Radiation/Chemistry Technician U (Int. 78)
179	1245	Route 441 was blocked to traffic by Pennsylvania State Police.	<ol style="list-style-type: none"> 1. NRC Region I Incident Message-form R11
180	1230	The A steam generator atmospheric relief valve was closed.	<ol style="list-style-type: none"> 1. NRC Inspector D's notes

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	NOTE: Steaming from the A steam generator atmospheric relief valve commenced after 0915 (See Item 150).	
181	1230	An NRC inspector reported to Region I that steam dumps were closed and radiation level readings appeared to be going down (2-4 mR/hr). NOTE: Investigators concluded based upon data showing effective isolation of the "B" OTSG that these readings were caused by the noble gas plume (which was invisible) coming from the Unit 2 vent stack.	1. NRC Region I Incident Message-form C14
182	1300	The Commonwealth of Pennsylvania Bureau of Radiological Health Lab completed the lithium drifted germanium analysis of the charcoal cartridge from Metropolitan Edison collected from location WSW-21 on 3/28/79 at 0940. The Pennsylvania Bureau of Radiological Health relayed the result to Metropolitan Edison shortly after 1300 (exact time unknown). Result: less than 1.5 E-11 uCi/cc of iodine-131. The Metropolitan Edison portable sodium iodide single channel analyzer (SAM-2) results had been 3 E-9 uCi/cc. Seven additional charcoal cartridges (all collected on 3/28) were sent to Pennsylvania Bureau of Radiological Health Lab; four were less than 1.5 E-11 uCi/cc (minimum detectable activity), and three were not analyzed because of higher priority samples.	1. Penna. Bureau of Radiological Health Laboratory Analysis Sheets
183	1310 - 1314	An NRC inspector left for Harrisburg to assist the Commonwealth with surveys. He began performing surveys at 1400.	1. Interview with NRC Inspector J (Int. 146)
184	1350	There was a reactor building pressure spike which isolated the containment. The pressure went up to 28 psig, then dropped to 1.5 psig. NOTE: This pressure spike resulted from a hydrogen burn in the reactor building.	1. Alarm printer 2. Unit 2 Foreman Log

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
185	1359	Unit 1 control room air activity was reported as 1 E-7 uCi/cc iodine and 9.4 E-8 uCi/cc particulate.	1. NRC Region I Incident Message-form C14
186	1400	The Metropolitan Edison Vice President, Generation; the Station Manager (Emergency Director); the Unit 2 Technical Support Superintendent; and others left the north gate to meet with the Lt. Governor.	1. Interview of Vice President, Generation (Int. 91)
187	1400-1430	An NRC inspector (dispatched 1310-1314) made surveys in Harrisburg and found no radiation levels above background.	1. NRC Inspector J's notes 2. Interview of NRC Inspectors I and J (Int. 146)
188	1415	The Department of Energy Aerial Monitoring System helicopter arrived and began tracking the plume.	1. Discussion with DOE
189	1430	The Metropolitan Edison Vice President, Generation; the Station Manager; and the Unit 2 Technical Support Superintendent met with the Lt. Governor	1. Interview of Vice President, Generation (Int. 91)
190	1600	The reactor building area radiation monitors located at the fuel handling bridge-north (HP-R-209), equipment access hatch (HP-R-212) and in-core instrument panel area (HP-R-213) were tagged out of service.	1. Out-of-Service Log 2. Out-of-service tag

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
191	1630	The Station Manager and the Unit 2 Superintendent, Technical Support returned to the plant after briefing the Lt. Governor.	1. Interview of Vice President, Generation (Int. 91)
192	1700	Hourly X/Q values for the period 0700-1500, 3/28/79, were provided by telephone to the Unit 2 control room by Pickard, Lowe, and Garrick (meteorological consultant). Similar data was provided on 3/29 and 3/30/79.	1. Interview of Meteorological Consultants A and B (Int. 60)
193	1900	A representative of Porter-Gertz, the licensee's environmental monitoring and analysis consulting firm, arrived at the Observation Center and reported to the Vice-President, Generation.	1. Interview of Radiation Protection Consultant A (Int. 19, 28)
194	1900	Two radiation/chemistry technicians entered the auxiliary building to change the charcoal cartridge on the station vent stack monitor (HP-R-219) and collect a water sample from the basement floor. The charcoal cartridge was changed. Auxiliary building dose rates were: 305 ft. elevation at entrance doors, 5-7 R/hr and 10 R/hr at the steps; 328 ft. elevation, 20 R/hr at the top of the steps; 50 R/hr at HP-R-219. Dose rates up to 100 R/hr at other locations. A water sample from the 281 ft. elevation floor was collected. The sample was taken to the Observation Center for analysis. The sample results are unknown, however a reading of 40 mR/hr was measured on contact with the 100 ml sample container.	1. Interview of Radiation/Chemistry Technicians W and H (Int. 49)
195	1900	While enroute to the TMI site from Region I, NRC inspectors made the following radiation surveys:	1. Interview of NRC Inspector N (Int. 137)

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
		<p><u>Location</u> <u>Met Ed Descriptor</u> <u>Exposure Rate (Open End Window GM)</u></p> <p>1 mile east of Exit 19 on Pennsylvania Turnpike NNW-41 2 mR/h</p> <p>Olmstead Plaza NNW-31 10-15 mR/h</p>	
196	1930	The NRC Region I Mobile Lab arrived at the TMI Observation Center, and informed the licensee that it was available to analyze samples. The licensee was also informed of the results of radiation surveys made by the NRC at 1900.	1. Interview of NRC Inspector N (Int. 137)
197	2000	The overall direction and control of the emergency organization was transferred from the Emergency Director (Station Manager), in the control room, to the Metropolitan Edison Vice President of Generation, in the Observation Center.	1. Interview of Station Manager (Int. 71)
198	2030	A representative of the licensee's environmental consultant firm arrived at the Unit 1 control room and reported by phone to the Emergency Director.	1. Interview of Radiation Protection Consultant A (Int. 19, 28) 2. Visitors Log
199	2100	An auxiliary operator was directed by a shift foreman to enter the auxiliary building and increase the nitrogen pressure on the core flood tanks from 400 to 600 psig. He could not find a high range pocket dosimeter so the entry was made with a 0-200 mr dosimeter. He entered the area alone and without a safety man. The dose rate just	1. Interview of Auxiliary Operator I (Int. 114)

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79	<p>inside the 305 ft. elevation entrance doors was 20 R/hr. He measured dose rates of 70 and 100 R/hr as he walked across the 328 ft. elevation. The dose rate near the nitrogen valve was 10 R/hr. The task took about 10 minutes. On exiting the building he found his pocket dosimeter had gone off-scale. On entering the control room he caused the GM counter (RM-14/HP-210 probe) to go off-scale (50,000 cpm). Since he had to decontaminate himself he decided to re-enter the auxiliary building and start a pump from the radwaste panel. He discussed this with a shift supervisor. He did not tell the shift supervisor about his previous entry. He re-zeroed his 0-200 mR pocket dosimeter and entered the auxiliary building and received a dose that caused his pocket dosimeter to go off-scale. His thermoluminescent dosimeter was read and the results indicated a dose of 3.2 rem.</p>	
200	2100	<p>Radiation surveys made by an NRC inspector, using an end-window GM survey meter, along PA Rte. 230, in the vicinity of the Harrisburg International Airport indicated an exposure rate of 12 mR/hr. The inspector made 4 passes through the plume.</p>	1. Interview of NRC Inspector J (Int. 146)
201	2130	<p>Unit 2 began transferring the neutralizer tank to the Unit 1 miscellaneous waste holdup tank. This was stopped on 3/29 at 0020.</p> <p>NOTE: The Unit 2 neutralizer tank contained preaccident water and was clean enough to be processed and released by Unit 1. Movement of this water from Unit 1 would provide capacity to start getting the water in the auxiliary building off the floor and into the tanks.</p>	1. Unit 2 CRO Log

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/28/79		
202	2150	Four health physics technicians from the Salem Nuclear Station arrived with monitoring equipment to assist in performing off-site surveys.	1. Visitors Log
203	2325	Radiation levels of 365 mR/hr beta-gamma and 50 mR/hr gamma were measured at GE-10 (1000 feet northwest of the Unit 2 plant vent) on the island. NOTE: This was the highest beta-gamma radiation level measured at ground level outside of the plant.	1. Observation Center (OC) Log
204	2400	Releases from industrial waste treatment system are estimated at 29,100 gallons released, with no activity.	1. NRC calculation based on licensee sample results and IWTS Log

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/29/79		
205	0001	A long range planning and logistical support organization composed of senior Metropolitan Edison and GPU management and various site personnel was established at the Observation Center.	1. Interview with Radiation/Chemistry Technician E (Int. 50)
206	0020	The transfer of liquid radwaste from the Unit 2 neutralizer tank (WDL-T-8B) to the Unit 1 miscellaneous waste holdup tank was stopped.	1. Unit 2 Foreman Log
207	0055	The auxiliary building and fuel handling building fans were stopped.	1. Unit 1 and Unit 2 Foreman Logs
208	0120	The miscellaneous waste holdup tank level was 7.5 feet. The level was 6.6 feet in this tank at the start of the accident. There were no known operations involving this tank during the period between the two readings.	1. Unit 2 CRO Log
209	0210	The auxiliary and fuel handling building ventilation systems were started.	1. Unit 2 Foremen and CRO Logs
210	0305	Although there was no liquid release being made, the liquid radiation monitor alarmed due to background radiation.	1. Region I Incident Messageform R42
211	0435	The makeup tank (MU-T-1) was vented to the vent header to the waste gas decay tanks.	1. Unit 2 CRO and Foreman Logs

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/29/79	NOTE: Operations personnel knew that the vent header leaked and that venting to the vent header resulted in releases to the auxiliary building exhaust system.	
212	0530	The Vice-President, Midwest Division, Radiation Management Corporation was phoned by the Philadelphia Office of Radiation Management Corporation and requested to report to the site as soon as possible to set up the Radiation Management Corporation lithium drifted germanium detector for sample counting and to ensure that the Radiation Management Corporation whole body counter (on site prior to the event because of the Unit 1 refueling outage) was functional.	1. Interview with Radiation Protection Consultant C (Int. 93)
213	0600-0700	Additional health physics technicians began arriving from Nuclear Support Services and Rad Services. NOTE: Nuclear Support Services provided 14 technicians on March 29 and an additional 4 technicians on March 30. Rad Services provided two instrument technicians and 7 tele-tectors and approximately 27 survey meters, and pocket dosimeters.	1. Interview of Protection Foreman A (Int. 11) 2. Discussion with Manager Instrument Services, Rad Services and letter from Nuclear Support Services
214	0830-1130	A nuclear engineer proposed changing makeup filters to increase letdown flow. A radiation/chemistry technician was requested by an engineer to survey the makeup filters. The survey instrument read full scale (1000 R/h) when placed through a port hole into the filter cubicle. The room outside the filter cubicle had general	1. Interview with Radiation/Chemistry Technician D (Int. 89)

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/29/79	<p>area dose rates of 2-5 R/hr and the reactor coolant letdown monitor (MU-R-720) read 90 R/hr. When discussing the dose rates on the make up filter with the operations staff, the radiation/chemistry technician expressed his concern about high radiation exposures which could be received during the proposed makeup filter change out. The technician was asked to resurvey the makeup cubicle. The technician reentered the area and escorted two repairmen to the 281 ft. elevation to repair a leak in the river water cooling system. The resurvey of the makeup filters again indicated greater than 1000 R/hr inside and 2 R/hr on top of the shielded cubicle. The technician advised the four mechanical maintenance personnel not to change the makeup filters. He also informed his foreman since the engineer who requested the work did not appear responsive to his warnings. The radiation/chemistry technician received 1.4 rem. The filters were not changed.</p>	
215	after 0700	<p>A radiation/chemistry technician surveyed the auxiliary building with the following results:</p>	<p>1. Interview of Radiation/Chemistry Technician I (Int. 42)</p>

ITEMDATE AND TIME
3/29/79EVENT DESCRIPTIONREFERENCE281 foot elevation

water on the floor, 50-60 mR/hr (the water may not have been the source of radiation).

35-40 R/hr - hall to bleed tanks.

300 R/hr - inside the door to bleed tanks.

500-600 R/hr - by the seal injection filter.

100 R/hr - inside the door by makeup pump "B".

100 R/hr - inside the south reactor building spray pump vault.

200 R/hr - inside the door of the "A" bleed tank cubicle.

1.5-2.5 R/hr - near the pit from the 305 ft. elevation at the north end of the building.

40-50 R/hr - at the makeup tank room door.

305 foot elevation

300-400 mR/hr - general halls.

1000 R/hr - localized radiation level on the door to the valve room containing the makeup filter bypass valve (MU-V-105).

15 R/hr - halls near the makeup filter bypass valve (MU-V-105) room.

50-70 R/hr - at the valve room door (which was locked).

328 foot elevation

20-30 mR/hr - in the area of the station vent stack monitor (HP-R-219).

The survey results were recorded and put in the control room.

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/29/79		
216	after 0700	A radiation/chemistry technician turned the reach rod to open a makeup system valve (MU-V-105) to bypass makeup and purification demineralizer filters to reduce pressure upstream of the filters. The valve was partially open prior to this action.	1. Interview of Radiation/Chemistry (Int. 42)
		NOTE: The letdown flow monitor in the control room showed no increase after the valve was opened.	
217	0715	The auxiliary building sump tank was pumped to a neutralizer tank (WDL-T-8B) to provide room in the sump tank for auxiliary building sump water.	1. Unit 2 CRO and Foreman Logs
218	0716	The makeup tank was bypassed and reactor coolant letdown was directed to reactor coolant bleed holdup tank B because of high pressure in the makeup tank.	1. Unit 2 CRO and Foreman Logs
		NOTE: The control room operators had noted that when the makeup tank was vented, the radiation levels in the auxiliary building increased.	
219	0815	Industrial waste treatment system and the industrial waste filter system sumps were sampled for activity prior to starting up the system.	1. IWTS Log
220	0835	Liberty Fire Company trucks arrived to refill bottles for self-contained breathing devices.	1. OC Log

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/29/79		
221	0845	A Unit 2 neutralizer tank (WDL-T-8A) was being transferred to Unit 1 miscellaneous waste holdup tank, moving preaccident water to Unit 1 to make room for highly radioactive water in the auxiliary building sump.	1. Unit 1 CRO and Unit 2 Foreman Logs
222	0847	Workers exiting the plant were advised by security personnel at the north gate to go to the 500 kV station to be surveyed for contamination.	1. OC Log
223	morning	A water sample from the plant effluent monitor (RM-L7 discharge from the mechanical draft cooling towers) was analyzed approximately 5/4/79 by Teledyne Isotopes for Porter-Gertz. Sample results showed 500 ± 90 pCi/l tritium ($5 \text{ E-}7$ uCi/ml).	1. Discussions with Radiation Protection Consultant A (Int. 19, 28)
224	0903	A radiation protection foreman and a radiation/chemistry technician brought the thermoluminescent dosimeter readers to the Observation Center. The background exposure rate at the normal thermoluminescent dosimeter reader location had increased to 40 mR/hr due to gaseous emissions.	1. OC Log 2. Interview of Radiation/Chemistry Technician A (Int. 99)
225	1000	Records showed the industrial waste treatment system was sampled at 1000 with the following results: iodine-131, $1.2 \text{ E-}4$ uCi/ml; iodine-133, $1.1 \text{ E-}4$ uCi/ml. The sump was not being discharged.	1. Licensee sample result
226	1032	The NRC reported to the Emergency Control Station that there was no iodine in the Unit 2 control room air samples collected by the licensee.	1. OC Log

<u>ITEM</u>	<u>DATE AND TIME</u> 3/29/79	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
227	1100	<p>The industrial waste treatment system and industrial waste filtration system were resampled. A radiation protection foreman informed the operator that there might be a xenon problem and did not know if it was one or both systems.</p> <p>NOTE: This was later determined to be xenon background on the counter. The sample point was not identified on the results sheet, but samples were normally taken at sample point 107 for the industrial waste treatment system and 104 for the industrial waste filtration system. These two points correspond to the discharge of the respective systems prior to the effluent entering the discharge from the mechanical draft cooling tower. One or two early samples for the industrial waste treatment system were dipped directly from the sump.</p>	1. IWTS Log
228	afternoon	A radiation/chemistry technician accompanied an electrician to the 328 ft. elevation of the auxiliary building to operate switch gear. They approached from the east side of the building and found dose rates in the range of 2-3 R/hr.	1. Interview with Radiation/Chemistry Technician R (Int. 75)
229	1215	The floor surface of the 281 ft. elevation of the auxiliary building was covered with plastic sheeting to reduce the release rate of radioactive gases from water on the floor into the auxiliary building atmosphere.	1. NRC Region I Incident Response Messageform C46
230	afternoon	The routine environmental samples (thermoluminescent dosimeters, air, water, milk) were collected. The decision to collect the samples at the time was made by a Metropolitan Edison radiation safety and environmental engineer and a licensee's environmental consultant.	1. Interview of Radiation Protection Consultant A (Int. 19, 28)

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/29/79		
231	1223	The Lt. Governor arrived at the north gate and went to the Observation Center for a briefing from Metropolitan Edison.	1. OC Log
232	1240	The turbine building, control building, and control and service building sump pumps were turned off due to a high level in the industrial waste treatment system.	1. Unit 2 Foreman Log
233	1315	The industrial waste treatment system was started at chemistry department instructions. The sump level was greater than 100 percent. The sump was overflowing to a yard drain which drains to a settling pond called the east dike drainage area. The east dike drainage area was containing most of the water but did have a small (undetermined) amount of leakage. The eventual destination of this leakage was the river.	1. IWTS Log 2. Discussion with IWTS operator
234	1328-1340	The supervisor, technical training, phoned the Department of Environmental Resources, Pennsylvania Bureau of Radiological Health concerning the industrial waste treatment system release to the river.	1. OC Log
235	1330	The industrial waste treatment system discharge was sampled. The results showed iodine-133 concentrations to be 3.6 E-4, but no iodine-133 was recorded.	1. IWTS Log
		NOTE: Because of the presence of iodine-133 with no iodine-131, this result was investigated and found to have been in error. Neither isotope of iodine was present in measurable amounts.	
236	1400	A representative of Radiation Management Corporation (a health physics consulting firm employed by the licensee) arrived at the Observation Center and began to set up the whole body counter and the mobile counting laboratory.	1. Interview of Radiation Protection Consultant C (Int. 93)

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/29/79		
237	1410	A helicopter measured 3 R/hr, beta-gamma, and 400 mR/hr, gamma, at 15 feet above the stack.	1. OC Log 2. Interview of Radiation/Chemistry Technician C (Int. 82)
238	1410	An industrial waste treatment system operator was informed by the shift supervisor to terminate industrial waste treatment system release due to xenon in the sample. The release had been approximately 90 gpm for 40 minutes (4950 gallons). NOTE: The xenon was later determined to be counter background and not in the sample.	1. IWTS Log
239	1455	The transfer of water from Unit 2 neutralizer tank WDL-T-8A to Unit 1 miscellaneous waste holdup tank was completed.	1. Unit 2 Foreman Log
240	1500	The Supervisor, Radiation Protection and Chemistry directed a radiation protection foreman and a chemistry foreman to take a reactor coolant sample within 2 hours. They were told the exposure rate from a 100 ml sample would be approximately 800 to 1000 R/hr. They took approximately 30 minutes to plan the job.	1. Interview of Radiation Protection Foreman D (Int. 18)
241	1535	The Radiation Management Corporation mobile laboratory arrived at the Observation Center.	1. OC Log
242	1600	The auxiliary building sump tank was pumped to a neutralizer tank (WDL-T-8A), to provide space for water from the auxiliary building sump.	1. Unit 2 CRO and Foreman Logs

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/29/79		
243	1610	The industrial waste tank system processing was started (at approximately 200 gpm) after receiving authority from the Supervisor, Radiation Protection and Chemistry. The sump level was greater than 100 percent.	1. IWTS Log
244	1615	A 100 ml reactor coolant sample was taken by a radiation protection foreman and a chemistry foreman. The sample read greater than 1000 R/hr on contact, 400 R/hr at 1 foot and 10-15 R/hr at 3 feet. They did not wear extremity dosimeters on their hands, and there were no air samples taken. The chemistry foreman received 4.1 rem dose to his whole body. The radiation protection foreman had nonremovable contamination of 150 mR/hr on his forearm, the chemistry foreman had nonremovable contamination of 25 mR/hr on his hands.	1. Interviews of Radiation Protection Foreman D and Chemistry Foreman B and C and Radiation Chemistry Technician X (Int. 18, 23, 138, 151) 2. Form 5 Printout 3. Unit 2 CRO Log
245	1616	One health physics supervisor, one health physics foreman and four health physics technicians from Oyster Creek Nuclear Station arrived with 3 high range GM survey meters (Telectors) and self-contained breathing devices.	1. OC Log 2. Discussion with Oyster Creek HP Supervisor
246	1652	The Radiation Management Corporation whole body counter was put into operation at the 500 kV Station.	1. OC Log

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/29/79		
247	1700	An industrial waste treatment system sample was taken at the discharge of the industrial waste treatment system prior to mixing with any other samples. A note in the log stated that samples were to be taken every 2 hours. The sample point on the industrial waste treatment system sump pump discharge line can be used either during industrial waste treatment system recirculation or discharge.	1. IWTS Log
248	1740	Two health physics supervisors and two health physics technicians from Salem Nuclear Plant arrived with an emergency van with 5-10 survey instruments, pocket dosimeters, respirators and protective clothing.	1. OC Log
249	1755	The NRC Executive Management Team directed the licensee to stop dumping all water. Region I notified headquarters that stopping the dumping would cause water to backup into the turbine building. The Region I Director personally discussed this order with the Station Manager.	1. NRC Region I Incident Messageform C49 2. Discussions with NRC Region I Director
250	1815	The industrial waste treatment system release was stopped after approximately 25,000 gallons were released. This release started at 1410.	1. IWTS Log
251	1850	Four health physics supervisors and ten health physics technicians arrived from Peach Bottom Nuclear Station. They brought approximately 6 air samplers, 36 radiation survey meters, respirators and protective clothing.	1. OC Log 2. Discussion with Peach Bottom HP Supervisor
252	1859	One ml of the reactor coolant sample collected at 1615 read 4 R/hr on contact. It was later diluted for boron analysis.	1. OC Log

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/29/79		
253	after 1900	A radiation/chemistry technician and a shift supervisor took a water sample from the auxiliary building basement floor.	1. Interview of Radiation/Chemistry Technician I (Int. 42)
		NOTE: Analytical results for this sample have not been located.	
254	2020-2030	An operator began to depressurize the makeup tank (MU-T-1) to the waste gas vent header through the sample system. The attempt was stopped.	1. Unit 2 CRO Log
255	2045	The makeup tank (MU-T-1) was vented to the waste gas header to lower the pressure to approximately 55 psig. Operators had noted increases in radiation monitor readings and concluded there was a leak in the waste gas system and that the leak rate increased as pressure in the vent header increased.	1. OC and Unit 2 CRO Log
256	2100	The auxiliary building sump tank was pumped to a neutralizer tank (WDL-T-8A) and the auxiliary building sump to the sump tank.	1. Unit 2 Foreman Log
257	2133	Respiratory protection was required on the island from the north bridge to the warehouse.	1. OC Log
258	2204-2213	The contaminated chemistry foreman was whole body counted and sent home with one hand reading 25 mR/hr. The radiation protection foreman also had fixed contamination on his forearm, however, records of the levels were not maintained.	1. OC Log

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/29/79		
259	2300	Two engineers entered the auxiliary building with a high range and a low range survey meter to look for water leaks and to check radwaste system status. The high range survey meter failed and they entered several areas with radiation levels above the range of the other meter (2 R/hr), including the area near the reactor coolant bleed holdup tank room. One engineer received 3.140 rem.	1. Interviews of Engineers E and J (Int. 33, 35)
260	2350	The makeup tank (MU-T-1) was vented to the waste gas vent header (WDG-T-1B).	1. Unit 2 CRO Log
261	2400	The Radiation Management Corporation lithium drifted germanium detector system was ready to begin counting samples.	1. Interview of Radiation Protection Consultant C (Int. 93)
262	2400	Releases from industrial waste treatment system for 3/29 (only) were estimated to be 12.65 millicuries. A total of 29,950 gallons were released from the industrial waste treatment system on 3/29.	1. Licensee sample results 2. IWTS Log
		NOTE: Since 10 CFR 20 allows releases to be averaged over a year, 10 CFR 20 limits would not have been exceeded.	

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/30/79		
263	0130	A nuclear engineer at the Emergency Control Station stated that he notified a representative of the NRC in the Unit 1 control room and the Commonwealth of Pennsylvania of a planned venting of the makeup tank to the vent header at approximately 0200-0300.	1. Interview of Engineers C and D (Int. 48, 127, 174)
		NOTE: The nuclear engineer stated that he notified the Commonwealth of Pennsylvania and NRC each time he knew of a vent to the vent header.	
264	0150	The makeup tank was vented to the B waste gas decay tank (WDG-T-1B). The A waste gas decay tank was at 50 psig; B waste gas decay tank, 75 psig.	1. OC Log
265	0152	A helicopter was sent to the south end of the island to monitor while Unit 2 vented.	1. OC Log
266	0155	The transfer of liquid from the auxiliary building sump tank to the neutralizer tank (WDL-T-8A) was stopped.	1. Unit 2 CRO Log
		NOTE: A log entry at 1600, 3/29/79 indicated that this transfer was initiated at or before 1600. A log entry at 2100 indicated that during this period water was also being pumped into the sump tank from the auxiliary building sump. The exact times and sequences of this operation can not be determined.	
267	0213-0215	The helicopter reported no radiation readings south of the plant. The Emergency Control Station directed the helicopter to the east side of the island.	1. OC Log

<u>ITEM</u>	<u>DATE AND TIME</u> 3/30/79	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
268	0215	All sump pumps in turbine building and control building areas were shut off.	
269	0215	The venting of the makeup tank to a waste gas decay tank was stopped. This venting had been in progress since 0150.	1. Unit 2 CRO Log
270	0300	An industrial waste treatment system discharge began at 190 gpm. The sump level was 100%.	1. IWTS Log
271	0315	The control building area sump was pumped to the turbine building sump with temporary pump.	1. Unit 2 Foreman Log
272	0330	An operator began venting the makeup tank to the waste gas vent header.	1. Unit 2 CRO Log
273	0350	The venting of the makeup tank was stopped. The pressure in the A waste gas decay tank was 50 psig. The pressure in the B waste gas decay tank was 80 psig. Venting had been in progress since 0330. NOTE: The relief valves for the waste gas decay tank were set at 120 psig. The relief valve for the makeup tank was set at 80 psig.	1. Unit 2 CRO Log
274	0430	An industrial waste filtration system discharge was started to the river via the discharge line from the mechanical draft cooling tower blowdown. NOTE: The amount of radioactivity discharged from the industrial waste filtration system was insignificant for the period 3/28-3/30.	1. Unit 2 CRO and IWTS Logs

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/30/79		
275	0435	<p>The liquid pressure relief valve (MU-R-1) on the makeup tank opened draining the makeup tank to a reactor coolant bleed holdup tank. The makeup tank level dropped straight down to zero. An unsuccessful attempt was made to pump B reactor coolant bleed holdup tank (RCBHT) to the makeup tank but the transfer pumps would not pump against the pressure in the makeup tank. The loss of level caused the operations personnel to switch the suction of the makeup pumps from the makeup tank to the borated water storage tank. Operation in this mode caused operators concern about the loss of water inventory in the borated water storage tank that might be needed for future core cooling and for reactivity control and prompted them to vent the makeup tank to establish normal makeup path. Venting of makeup tank (see 0710) caused an increase in gaseous discharges. The pressure in the makeup tank was 80-84 psig. Because of the discharges to the RCBHT's via relief valve MU-R-1 on the makeup tank, the pressure in the RCBHT's was offscale at > 30 psig which is 1 1/2 times the design pressure of these tanks. It's probable that the relief valves on the RCBHT's were opening at least periodically.</p>	<ol style="list-style-type: none"> 1. Unit 2 CRO Log 2. MU-T-1 level chart recorder 3. Interview of Station Operations Supervisor (Int. 157) 4. Interviews of Shift Supervisors C and E (Int. 173, 189)
276	0455	<p>NRC inspectors met with the Superintendent, Technical and Administrative Services, to discuss concerns with the radiation protection program. The following areas were discussed:</p> <ol style="list-style-type: none"> 1. Access control. No positive control was established over entries into high radiation areas such as the auxiliary building. 2. Exposure control. The radiation work permit procedure was not being followed and equivalent measures were not being taken. 	<ol style="list-style-type: none"> 1. Interview of NRC Inspector H (Int. 40) 2. NRC Region I Incident Messageform R87

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/30/79	3. Effluent monitoring. The installed samplers were not being changed out in a timely manner.	
277	0630	The industrial waste treatment system flow dropped from 190 to 150 gpm (0300 hrs to 0630 hrs at 190 gpm equaled 39,900 gal).	1. IWTS Log
278	0650	A representative of Pickard, Lowe, and Garrick, the licensee's environmental consultant firm, went on site.	1. OC Log
279	0700	The A reactor coolant bleed holdup tank sample was 3.22 E-2 uCi/ml and the C reactor coolant bleed holdup tank was 109 uCi/ml. NOTE: This indicated that very little reactor coolant had been put in A reactor coolant bleed holdup tank. Since the tanks were separated only by a loop seal it also indicated the reactor coolant bleed holdup tank probably did not overflow.	1. Licensee sample result
280	0700-0800	One ml of the reactor coolant sample taken on 3/29 was removed from the sample container by a radiation protection foreman and placed in a 6 ml glass vial. No air samples were taken and no extremity dosimeters were worn.	1. Interview of Radiation Protection Foreman B (Int. 21)
281	0710	An operator opened the makeup tank vent valve (MU-V-13) and started venting the makeup tank to the vent header. At least two shift supervisors and one operator were involved in this decision. They decided this step was necessary in order to reduce pressure in the makeup tank so that water could be transferred from the reactor coolant bleed holdup tank to the makeup tank and so that the feed from the borated water storage tank could be stopped.	1. Unit 2 Foreman Log 2. Interviews of the Unit Supervisor, Station Operations and Engineer C

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/30/79		(Int. 157, 127, 174)
282	0744	The Unit Supervisor, Operations phoned the Emergency Control Station to have survey teams and the helicopter dispatched.	1. OC Log
283	0744-0845	An engineer at the Emergency Control Station notified the NRC representatives in the Unit 1 shift supervisor's office and the Commonwealth of Pennsylvania via telephone that makeup tank had been vented and that increased releases were expected. He later notified the NRC and Commonwealth that the vent would last longer than originally anticipated.	1. Interview of Engineers C and D (Int. 48, 174)
284	0756-0801	Radiation levels of 1000 and 1200 mR/hr (beta-gamma) were measured at 0756 and 0801 respectively, from a helicopter at an elevation of 600 feet. This was 130 feet above the Unit 2 auxiliary building. The helicopter decreased altitude in order to better define the source of radiation. As the helicopter decreased altitude, the radiation level decreased to 600 mR/hr. The helicopter returned to the location of the original 1000 and 1200 mR/hr readings, but no further readings in that range were found.	1. Unit 1 CRO and Foreman Logs 2. Interviews of the Unit Super- visor, Station Operations and Radiation/ Chemistry Technician J (Int. 157, 51)
285	0834	The Unit Supervisor, Station Operations, called Civil Defense to report that there was a release of radioactive materials and stated that the	1. Civil Defense Logs

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/30/79	<p>site was prepared to evacuate, that the site had its own buses and recommended that the State prepare to evacuate. At the same time, another call was received from the site which provided details concerning the release. The caller explained that an evacuation of the site was not planned.</p> <p>NOTE: The Unit Supervisor, Station Operations, stated that there were two calls, one he placed and one confirmatory call back from Civil Defense. He said that these calls contained routine emergency wording which would include the request of the status of the Civil Defense evacuation plans, the availability of buses to provide transportation, and the status of the plant's evacuation planning.</p>	<ol style="list-style-type: none"> 2. Discussions with Civil Defense Personnel 3. Interviews of Unit Supervisor, Station Operations (Int. 157) 4. NRC Region I Incident Messageforms C56 and C57
286	0845	<p>An NRC inspector in control room reported to NRC headquarters that the seal return to the makeup tank was causing excessive gas pressure in the makeup tank which was directed to the waste gas decay tanks which were full. The waste gas tanks were being released to the stack. This was resulting in an increased stack release rate. He said the Civil Defense and Commonwealth of Pennsylvania were being notified by the licensee.</p>	<ol style="list-style-type: none"> 1. NRC Region I Incident Messageform C56

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/30/79		
287	0900-1300	A radiation protection foreman accompanied a sample of reactor coolant which had been collected at 1615 on 3/29/79 to the airport. The sample was flown to an outside laboratory for analysis.	<ol style="list-style-type: none"> 1. Unit 2 CRO Log 2. Interview of Radiation Protection Foreman B (Int. 21) 3. Licensee sample results
		The sample results were:	
		iodine-131 - 1.3 E+4 uCi/cc	
		iodine-133 - 6.5 E+3 uCi/cc	
		cesium-134 - 6.3 E+1 uCi/cc	
		cesium-136 - 1.8 E+2 uCi/cc	
		cesium-137 - 2.8 E+2 uCi/cc	
		strontium-89/90 - 5.3 uCi/cc	
		ruthenium-106 - 2 E+2 uCi/cc	
		barium-140 - 2 E+2 uCi/cc	
		tellurium-132 - 2 E+2 uCi/cc	
288	0945	Air iodine results for two locations (2.6 mi north and 9 mi southeast) sampled weekly (off 3/29) were telephoned to the Metropolitan Edison Corporate office. Results were 0.05 and less than 0.02 pCi/m ³ respectively.	<ol style="list-style-type: none"> 1. Telecons with representatives of Porter-Gertz Consultants and Teledyne Isotopes

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/30/79		
289	1000	The A and B waste gas decay tank pressures were 79 psig.	1. Unit 2 Foreman Log
290	1020	Water iodine results for five locations (2 upstream and 3 downstream) were telephoned to the Metropolitan Edison Corporate office. The results were less than the minimum detectable activity (approximately 10 pCi/l).	1. Telecons with representatives of Porter-Gertz Consultants and Teledyne Isotopes
291	1030	The Unit 2 B steam generator was sampled with the following results indicating a primary to secondary leak: iodine-131 - 7.9 uCi/ml iodine-133 - 2.1 uCi/ml iodine-135 - 8.5 E-2 uCi/ml cesium-136 - 6.8 E-3 uCi/ml	1. Licensee sample results
292	1045	The area radiation monitors (HP-R-231, HP-R-233, HP-R-3238) located at the auxiliary building sump tank filter room on the 281 ft. elevation, in the access corridor on the 305 ft. elevation and at the building exhaust unit were tagged out of service because they did not source check.	1. Out-of-service Log
293	1220	The transfer of Unit 2 miscellaneous waste holdup tank to Unit 1 was started.	1. Unit 2 CRO Log
294	1230	The first industrial waste filtration system sample was taken at National Pollution Discharge Elimination System point 104. This samples point was prior to the discharge into the blowdown from the mechanical draft cooling tower. No activity above background was observed.	1. IWTS Log

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
295	1230 3/30/79	<p>The Metropolitan Edison Corporate Office received results of the samples collected on the afternoon of 3/29. The results were:</p> <ul style="list-style-type: none"> - Preliminary results for offsite thermoluminescent dosimeters were reported as less than 25 mrem/quarter. The maximum result reported for thermoluminescent dosimeters located on the island (location 16 S 1) was 923 mrem/quarter. (The final result for this location was 1044 mrem/quarter). <p>NOTE: Results from dosimeters located on Kohr and Shelley Islands were not available at this time. The highest value subsequently received for these locations was 908 mrem/quarter for Kohr Island (location 16A1).</p> <ul style="list-style-type: none"> - All iodine air samples were less than 0.03 pCi/m^3 except one with a 0.47 pCi/m^3 at 1S2 (north-northeast) on the island. The MPC for unrestricted areas is 100 pCi/m^3. - surface water samples from one upstream and four downstream location were less than minimum detectable activity for iodine (0.5 pCi/l). The MPC for unrestricted areas is 300 pCi/l. 	<p>1. Interview of Radiation Protection Consultant A (Int. 19, 28)</p>
296	1440	A waste decay gas valve was opened to transfer B waste gas decay tank into the reactor building.	<p>1. Unit 2 CRO Log</p>

<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/30/79		
297	1500	The pressures in A and B waste gas decay tank were 80 psig.	1. Unit 2 CRO Log
298	1545	Representatives of NRC, NRR, arrived in the Unit 1 control room.	1. OC Log
299	1900	The station vent stack monitor (HP-R-219) charcoal cartridge was changed. The results were iodine-131, 1.4 E-6 uCi/cc; and iodine-133, 5.2 E-7 uCi/cc.	1. Licensee sample results
300	2045	The Unit 2 B steam generator was sampled using the normal Unit 2 sample point. The results were indicative of a primary to secondary leak: iodine-131 - 4.2 uCi/ml iodine-133 - 1.4 uCi/ml xenon-135 - 2.7 E-1 uCi/ml cesium-136 - 2.2 E-2 uCi/ml cesium-137 - 3.8 E-2 uCi/ml	1. Licensee sample results
301	2128	The pressurizer was vented through vent valve (RC-V-137) to the reactor coolant drain tank.	1. Unit 2 Foreman Log
302	2200	After arrival on site at 2000, the Science Application, Incorporated mobile lab performed operability checks and calibrations and were ready to accept the first samples for counting. They did not receive samples until 3/31/79.	1. Interview of Radiation Protection Consultant B (Int. 69)

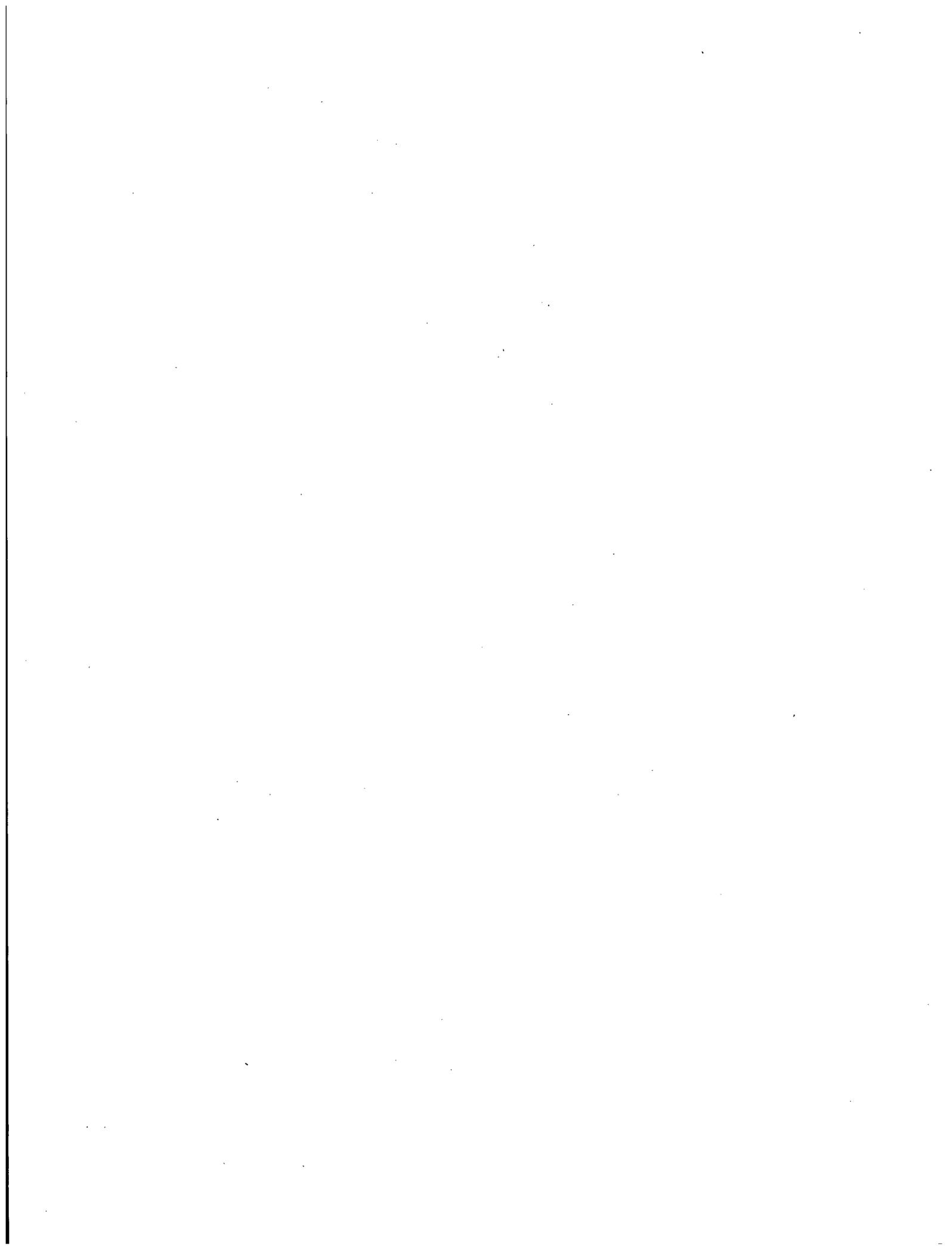
<u>ITEM</u>	<u>DATE AND TIME</u>	<u>EVENT DESCRIPTION</u>	<u>REFERENCE</u>
	3/30/79		
303	2217	The reactor coolant system was vented through vent valve RC-V-137 to the reactor coolant drain tank.	1. Unit 2 CRO Log
304	2229	The miscellaneous waste holdup tank contents were being transferred to Unit 1.	1. Unit 2 CRO Log
305	2310	The reactor coolant system was vented via vent valve RC-V-137 to the reactor coolant drain tank.	1. Unit 2 CRO Log
306	2400	The transfer to Unit 1 from the miscellaneous waste holdup tank was stopped.	1. Unit 2 CRO Log
307	2400	The releases from industrial waste treatment system for 3/30 (only) are estimated at 59.91 millicuries.	1. NRC Calcula- tion

The total releases from the industrial waste treatment system for the period 3/28 through 3/30 is estimated at 72.56 millicuries.

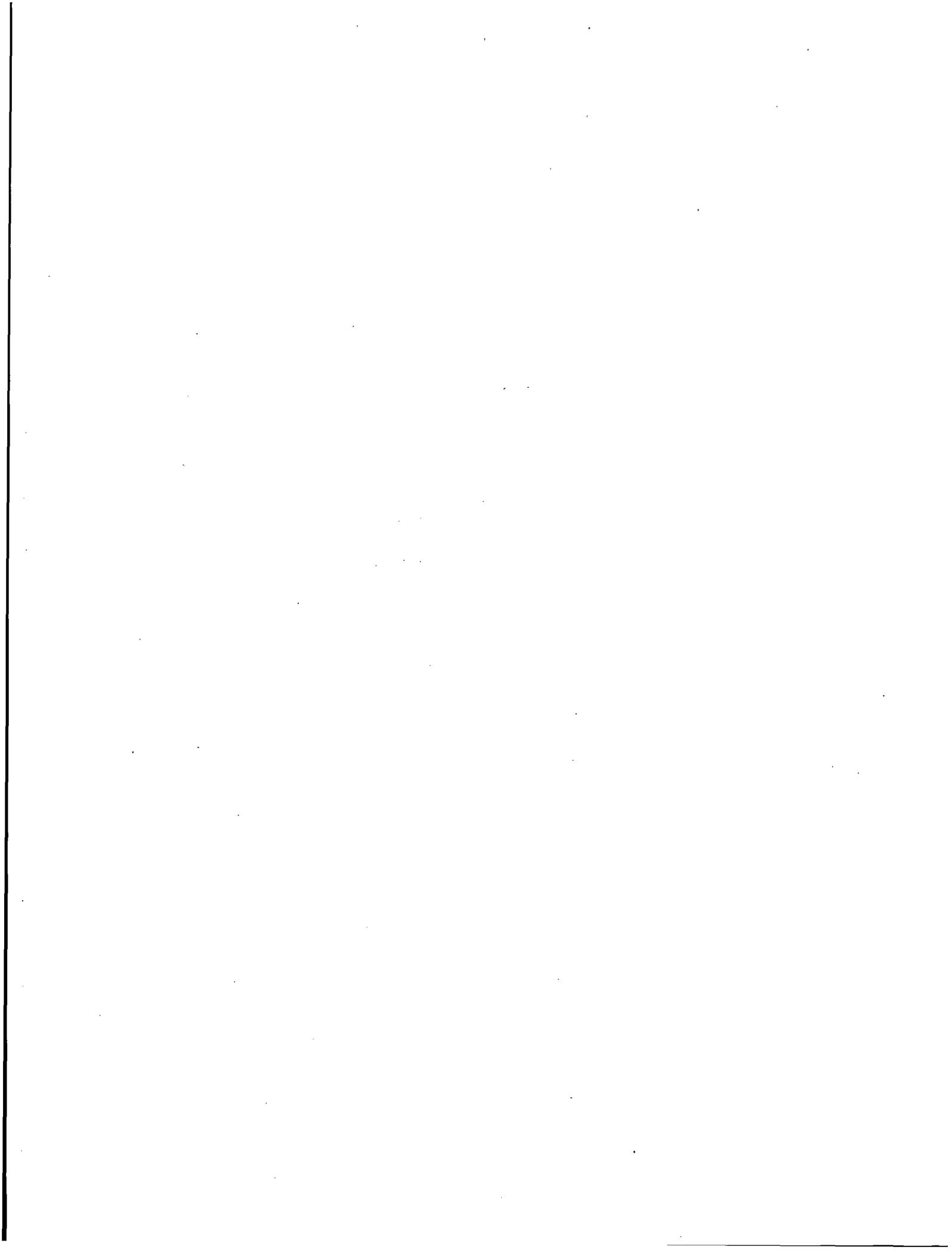
A total of 206,400 gallons were released on 3/30.

A total of 265,450 gallons were released from 3/28 at 0400 to 3/30 at 2400.

Based on an annual average, no limits would have been exceeded.



APPENDIX II-B
DOSIMETRY EVALUATIONS
POTENTIAL OVEREXPOSURES





UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

MEMORANDUM FOR: A. Gibson, NRC:IE Investigation Team
FROM: M. J. Slobodien, Radiation Specialist
SUBJECT: DOSIMETRY EVALUATIONS POTENTIAL OVEREXPOSURES

I have reviewed six potential overexposures identified by the Investigation Team. Two individuals appear to have been exposed in excess of regulatory requirements. Tables II-B-1 and II-B-2 summarize these overexposures giving lower and upper bounds. The detailed analyses of these overexposures are given in Tables II-B-3 and II-B-4 to this memorandum. Four of the six individuals identified during the investigation as being potentially overexposed were not exposed in excess of regulatory limits.

Michael J. Slobodien
Michael J. Slobodien
Radiation Specialist

Enclosures:
As Stated

TABLE II-B-1

EXPOSURES TO SUBJECT A IN 1ST QTR 1979

<u>Area of Body</u>	<u>Lower Bound</u>	<u>Upper Bound</u>	<u>Reg Limit</u>
Whole Body	0.88 rem (calc.)	0.89 rem (TLD)	3 rem
Extremities			
Fingers	17	39	18.75 rem
Hand (palm)*	17	39	18.75 rem
Forearm (skin)*	44 rem	54 rem	18.75 rem

* The estimate of dose of fingers, hand and palm are the same as it was not possible to make accurate estimates of the various finger/hand configurations. The forearm is taken as the area around the wrist.

TABLE II-B-2

EXPOSURES TO SUBJECT B IN 1ST QTR 1979

<u>Area of Body</u>	<u>Lower Bound</u>	<u>Upper Bound</u>	<u>Reg Limit</u>
Whole Body	4.0 rem (calc.)	4.1 rem (TLD)	3 rem
Extremity			
Fingers	144 rem	147	18.75 rem
Skin (Top of Head)*	6 rem*	13 rem*	7.75 rem

These values will increase since only I-131 contamination was considered.

- + The licensee reported the 4.1 rem W/B dose only on May 1, 1979 pursuant to 10 CFR 20.405(b), no extremity or skin doses were reported.

EXPOSURES TO SUBJECT B IN 2ND QTR 1979

Skin (Top of Head)*	19.75 rem	7.75
---------------------	-----------	------

- * To this value must be added the dose due to contamination (Cs-134, Cs-137, I-131) which was present from the time of contamination until the first whole body count.

Details for the data on this table are on page II-B-29 and II-B-30.

TABLE II-B-3

Subject - A

Circumstances of Exposure

A is a health physics foreman. On March 29, 1979 he assisted a chemistry foreman (subject B) in taking and measuring a sample of primary coolant from the letdown system sampler located in the Unit 1/2 nuclear sampling room. As a result, he handled high level radioactive sources with gloved hands, worked in high radiation areas, and sustained significant forearm skin contamination.

The following outline summaries A's actions on March 29, 1979 to the present:

March 29, 1979

<u>TIME</u>	<u>ACTION</u>
~ 15:00	A and chemistry foreman (B) proceeded to Unit 1/2 sampling room to prepare to take a sample of letdown.
15:00-15:30	A and B reviewed the piping runs.
15:30-15:45	Wearing full protective clothing including a MSA full face respirator equipped with a particulate/iodine canisters A and B entered the Nuclear Sampling room. A radiation measurement with a teletector indicated ambient radiation levels of 6-8 R/hour. B proceeded to the sample cooler, aligned the valves and placed the letdown system line for Unit 2 on Recirc. A and B exited.
16:30	45 minutes were allowed for sample recirculation. A entered the sample room and measured radiation levels of 90 R/hour at the "drag" valve. He measured 17-20 R/hour ambient radiation levels. A moved into the sample room, placed a 400 ml beaker in the sample sink, opened the letdown valve one full turn and took 30-100 ml. The sample was very gassy. A closed the valve and left the sample in the sink. A exited to discuss the situation with B.

B confirmed the valve alignment. B may have entered the sample room to do this, or may have confirmed valve locations from a diagram. It is not clear from available data.

B re-entered the sample room, opened the letdown line valve at the sink and drew a 300 milliliter sample while holding the bottle with his left hand. As the bottle filled A stood behind the valve control panel east of the sample sink. B closed the letdown valve. A and B exited.

B entered and poured 100 ml into a small polyethylene graduated cylinder then withdrew 5 ml then took a 40 ml sample and exited with it. A hand carried the graduated cylinder to the southeast corner of the sample room and placed it on the floor. A used a teletector to measure greater than 1000 R/hour on contact with the beaker, 400 R/hour at one foot and 40 R/hour at three feet.

A returned the graduated cylinder to the sample sink and exited the sample room. As he left the area, A ripped his wet suit on the valve.

In the primary chemistry lab A assisted B by picking up a 40 ml sample of primary coolant and dumping it down the sample sink and by returning the sample of 260-300 ml of primary coolant to the shield in the storage cabinet.

16:30-17:00

Upon removing protective clothing and frisking with an Eberline E-520 geiger counter with an HP-177 side window probe, A identified a 5-10 cm diameter spot on the left forearm which indicated 115 mR/hour on contact.

3/30/79

At 24:00 the contaminated spot on A's forearm indicated 40 mR/hr. This was after 6 hours of intensive attempts at decontamination.

4/2/79

At 16:00 on 3/30/79 the contaminated spot indicated 2 mR/hour.

4/6/79

At 16:00 on 4/6/79 the contaminated spot indicated background. At 16:07 A received an initial whole body count which indicated 329 nCi. I-131 as the principle contaminant with most material on the skin.

4/11/79

A was whole body counted indicating 70 nCi primarily as skin contamination 18 nCi in the thyroid.

DOSE EVALUATION
1ST QTR 1979
SUBJECT A

Action	Duration	Dose Rates* (R/sec)			Dose (rem)		
		Whole Body	Hand/Forearm	Fingers	Whole Body	Hand/Forearm	Fingers
1. Initial Survey Nuclear Sample Room	12 sec	2.2×10^{-3}	2.2×10^{-3}	2.2×10^{-3}	2.6×10^{-2}	2.6×10^{-2}	2.6×10^{-2}
2. Entry to sample sink	3 sec	4.7×10^{-3}	4.7×10^{-3}	4.7×10^{-3}	1.4×10^{-2}	1.4×10^{-2}	1.4×10^{-2}
3. Place beaker in sink, open valve take 30-100 ml sample	5 sec	4.7×10^{-3}	2.5×10^{-2}	2.5×10^{-2}	2.3×10^{-2}	1.25×10^{-1}	1.25×10^{-1}
4. Closes sample valve	2 sec	4.7×10^{-3}	2.5×10^{-2}	2.5×10^{-2}	9.4×10^{-3}	5×10^{-2}	5×10^{-2}
5. Exits area	3 sec	4.7×10^{-3}	4.7×10^{-3}	4.7×10^{-3}	1.4×10^{-2}	1.4×10^{-2}	1.4×10^{-2}
6. Reentry and place- ment of 1 liter in sample sink	5 sec	4.7×10^{-3}	4.7×10^{-3}	4.7×10^{-3}	2.3×10^{-2}	2.3×10^{-2}	2.3×10^{-2}
7. Stands behind con- trol panel while bottle fills	60 sec	4.7×10^{-3}	4.7×10^{-3}	4.7×10^{-3}	2.8×10^{-1}	2.8×10^{-1}	2.8×10^{-1}

DOSE EVALUATION
1ST QTR 1979
SUBJECT A

<u>Action</u>	<u>Duration</u>	<u>Dose Rates* (R/sec)</u>			<u>Dose (rem)</u>		
		<u>Whole Body</u>	<u>Hand/Forearm</u>	<u>Fingers</u>	<u>Whole Body</u>	<u>Hand/Forearm</u>	<u>Fingers</u>
8. Picks up the 100ml graduated cylinder filled w/primary coolant and carries it to southeast corner of lab	6 sec	1.2×10^{-2}	2.5×10^{-1}	7×10^{-1}	7.2×10^{-2}	1.5	4.2
9. Measures exposure rates standing at 6' from the cylinder	8 sec	1.1×10^{-2}	1.1×10^{-2}	1.1×10^{-2}	8.8×10^{-2}	8.8×10^{-2}	8.8×10^{-2}
10. Picks up 100 ml cylinder returns it to sample sink	6 sec	1.2×10^{-2}	2.5×10^{-1}	7×10^{-1}	7.2×10^{-2}	1.5	4.2
11. Exits Nuclear Sampling Room	3 sec	1.1×10^{-2}	1.1×10^{-2}	1.1×10^{-2}	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}
12. Picks up 40 ml coolant in 100 ml beaker (drawn by subject B) and dumps into sample sink	3 sec	4.4×10^{-2}	2.8×10^{-1}	2.8×10^{-1}	1.3×10^{-1}	8.4×10^{-1}	8.4×10^{-1}

DOSE EVALUATION
1ST QTR 1979
SUBJECT A

<u>Action</u>	<u>Duration</u>	<u>Dose Rates* (R/sec)</u>		<u>Dose (rem)</u>			
		<u>Whole Body</u>	<u>Hand/Forearm</u>	<u>Fingers</u>	<u>Whole Body</u>	<u>Hand/Forearm</u>	<u>Fingers</u>
13. Picks up 260 ml primary coolant in 1l bottle carries to cabinet. Distance to WB=30', fingers on contact, hand, 3"	6 sec	1.7×10^{-2}	0.6	1.2	1.0×10^{-1}	3.6	7.2
14. Forearm skin contamination LEFT FOREARM 3/29/79 @ 16:00 to 3/31/79 @ 24:00	1384 uCi-hrs	N/A			N/A	36-46 rem	N/A
				TOTAL #	0.884 rem	8.1 rem direct; 36-46 rem contamination 44-54 total, contamination & direct	17.1 rem

The dose rates in this calculation for contact doses are based on the method of calculating a photon flux with $E=.44\text{MeV}$ based on the assumed fission product inventory. This method appears in pages II-B-16 - II-B-22 of this report. Dose rates other than those on contact are based on teletector measurements. The skin dose calculations are found on pages II-B-23 - II-B-25 of this report.

BASIS FOR DOSE RATES

Action No.

Basis

1. Dose rates measured by subject 6-8 R/hr selected 8 R/hr for W/B.
2. Dose rates measured by subject 15-17 R/hr after sample circulated.
3. Dose rates measured by subject using teletector.
4. Dose rates measured by subject using teletector, 90 R/hr on contact with value.
5. Dose rates measured by subject using teletector.
6. Dose rates measured by subject using teletector.
7. Dose rates measured by subject using teletector.
8. W/B dose rates measured by subject using teletector, extremity dose rates calculated for source.
9. Dose rates measured by subject using teletector.
10. W/B: dose rates measured by subject using teletector. Extremity from calculation as shown (after Kase et. al.) pages 17-21 of this report.
11. Subject measured dose rates.
12. See No. 10.
13. See No. 10.
14. MIRD calculations, done in accordance with Phamplits 10, 11 and ICRP-23 contamination values checked w/HP-177 probe and E-520 on disc source whole body count data, see pages 23-25 of this report.

DOSE EVALUATION SUBJECT A
2ND CALCULATION

I. 1st QTR 1979

A. Exposure prior to 3/29/79

Whole Body .09 rem
Extremity .09 rem

B. Data available for 3/29/79

1. TLD read 0.89 for W/B.
2. No extremity dosimetry worn.
3. A measurement of 100 ml of primary coolant with a teletector indicated:
 - a. > 1000 R/hr on contact.
 - b. 400 R/hr @ 12" (the distance was determined using a floor tile).
 - c. 40 R/hr @ 3' (the distance was determined using 3 floor tiles).
4. Primary coolant analysis (see data sheet), pages II-B-20 through II-B-22.

$$7.53 \times 10^8 \text{ photons sec}^{-1} \text{ ml}^{-1}$$

$$E = 0.44 \text{ MeV}$$

C. Calculations

1. Dose rate from 100 ml to deep tissue.

By assuming a point source and using the measurements made by the subject.

$$\times 1 \text{ inch} = \left(400 \frac{\text{R}}{\text{hr}}\right) \left(\frac{12^2}{1}\right) = 57600 \text{ R/hr}$$

$$\times 30 \text{ inch} = \left(400 \frac{\text{R}}{\text{hr}}\right) \left(\frac{12^2}{30^2}\right) = 64 \text{ R/hr}$$

$$\times 30 \text{ inch} = \left(40 \frac{\text{R}}{\text{hr}}\right) \left(\frac{36^2}{30^2}\right) = 57.6 \text{ R/hr}$$

$$\times 30 \text{ inch} = \frac{64 + 57.6}{2} = 60.8$$

2. Time Distance Study

DOSE EVALUATION
 1ST QTR 1979
 SUBJECT A
 2ND CALCULATION

Action	Duration	Dose Rates* (R/sec)			Dose (rem)		
		Whole Body	Hand/Forearm	Fingers	Whole Body	Hand/Forearm	Fingers
1. Initial Survey Nuclear Sample Room	12 sec	2.2×10^{-3}	2.2×10^{-3}	2.2×10^{-3}	2.6×10^{-2}	2.6×10^{-2}	2.6×10^{-2}
2. Entry to sample sink	3 sec	4.7×10^{-3}	4.7×10^{-3}	4.7×10^{-3}	1.4×10^{-2}	1.4×10^{-2}	1.4×10^{-2}
3. Place beaker in sink, open valve take 30-100 ml sample	5 sec	4.7×10^{-3}	2.5×10^{-2}	2.5×10^{-2}	2.3×10^{-2}	1.25×10^{-1}	1.25×10^{-1}
4. Closes sample valve	2 sec	4.7×10^{-3}	2.5×10^{-2}	2.5×10^{-2}	9.4×10^{-3}	5×10^{-2}	5×10^{-2}
5. Exits area	3 sec	4.7×10^{-3}	4.7×10^{-3}	4.7×10^{-3}	1.4×10^{-2}	1.4×10^{-2}	1.4×10^{-2}
6. Reentry and place- ment of 1 liter in sample sink	5 sec	4.7×10^{-3}	4.7×10^{-3}	4.7×10^{-3}	2.3×10^{-2}	2.3×10^{-2}	2.3×10^{-2}
7. Stands behind con- trol panel while bottle fills	60 sec	4.7×10^{-3}	4.7×10^{-3}	4.7×10^{-3}	2.8×10^{-1}	2.8×10^{-1}	2.8×10^{-1}

DOSE EVALUATION
 1ST QTR 1979
 SUBJECT A
 2ND CALCULATION

Action	Duration	Dose Rates* (R/sec)		Dose (rem)			
		Whole Body	Hand/Forearm	Fingers	Whole Body	Hand/Forearm	Fingers
8. Picks up the 100ml graduated cylinder filled w/primary coolant and carries it to southeast corner of lab	6 sec	1.2×10^{-2}	1.1×10^{-1}	1.33 @ 1"	7.2×10^{-2}	6.6×10^{-1}	8.0
9. Measures exposure rates standing at 6' from the cylinder	8 sec	1.1×10^{-2}	1.1×10^{-2}	1.1×10^{-2}	8.8×10^{-2}	8.8×10^{-2}	8.8×10^{-2}
10. Picks up 100 ml cylinder returns it to sample sink	6 sec	1.2×10^{-2}	2.5×10^{-1}	1.33 @ 1"	7.2×10^{-2}	1.5	8.0
11. Exits Nuclear Sampling Room	3 sec	1.1×10^{-2}	2.7×10^{-3}	1.1×10^{-2}	8.3×10^{-3}	3.3×10^{-2}	3.3×10^{-2}
12. Picks up 40 ml coolant in 100 ml beaker (drawn by subject B) and dumps into sample sink	3 sec	4.4×10^{-2}	2.8×10^{-1}	0.3×10^{-1} @ 1"	1.3×10^{-1}	8.4×10^{-1}	1.6

DOSE EVALUATION
 1ST QTR 1979
 SUBJECT A
 2ND CALCULATION

Action	Duration	Dose Rates* (R/sec)		Hand/Forearm	Fingers	Dose (rem)		
		Whole Body	Whole Body			Whole Body	Hand/Forearm	Fingers
13. Picks up 260 ml primary coolant in 1l bottle carries to cabinet. Distance to WB=30', fingers on contact, hand, 3"	6 sec		1.7×10^{-2}	0.6	3.5 @ 1"	1.0×10^{-1}	3.6	21
14. Forearm skin contamination LEFT FOREARM 3/29/79 @ 16:00 to 3/31/79 @ 24:00	1384 uCi-hrs	N/A		36-46 rem		N/A	36-46 rem	N/A
						.860	7.2 rem direct 36-46 rem contamination 44-54 total, contamination & direct	~ 39 rem

*The dose rates for extremities in this calculation are based on teletector data. For fingers, the radiation sources were considered as line sources. Dose rates for a line source at 1 inch are estimated from teletector data to be $\frac{400 \text{ R hr}}{100\text{m}} \frac{12''}{1''} = 4800 \text{ R hr} = 1.33 \text{ R sec}^{-1}$ at 1"

BASIS FOR x VALUES SELECTED
IN 2ND CALCULATION

<u>Action No.</u>	<u>Basis</u>
1.	Subject indicated general area exposure rates 6- ~ 8 R/hr (2.2×10^{-3} R/sec).
2.	Subject measured exposure rates @ sample sink of 15-17 R/hr; 16 R/hr = 4.7×10^{-3} R/sec.
3.	As in #2 above for W/B. Extremity dose rates based on subjects measurement of valve dose rate of 90 R/hr = 2.5×10^{-2} R/sec.
4.	As in #3 above.
5.	As in #2 above.
6.	As in #3 above.
7.	As in #2 above.
8.	30" W/B based on correction from 40 R/hr @ 3' assuming a line source. Extremity x based on 400 R/hr @ 1 foot and assuming hand was at one foot.
9.	As in #8 above.
10.	30" x based on 400 R/hr @ 12". 1" x based on 400 R/hr @ 12" and assuming deep tissue.
11.	Subject measurement of 40 R/hr @ 3' corrected to 6'.
12.	As in #10 above.
13.	As in #10 above.
14.	As in #2 above.
15.	MIRD 10, 11, ORNL 5000 assuming uniform deposition throughout skin.

ASSUMPTIONS AND CALCULATIONS

1. The exposure times are estimated from data provided by subjects A and B in investigation interviews.
2. The dose rate from a right circular cylinder containing the primary coolant sample is based on the material inventory determined by sample analysis of a 3/29/79 sample at the Westinghouse - Bettis Laboratory. The calculation which follows is for a one liter bottle. The dose rates from the 100 ml sample would be ~ 10% of that from a one liter sample. Thus, the contact dose rate is 0.7 rad/sec (2480 rad/hr). This may be low by a factor of 3 depending on the accuracy of extrapolation to a=0 in accompanying figure on page II-B-19 of this report. Note this is not in agreement with point source assumption used in the "2nd calculation". The 1st and 2nd calculation would indicate lower and upper extremity dose estimates respectively. The estimate of 2480-3500 R/sec is in agreement with the assumption of a the graduated cylinder being a line source at distances < 12". In that case the dose rate at one inch would be approximately:

$$\{400\} \frac{R}{hr} \frac{\{12''\}}{1''} = 4800 \frac{R}{hr}$$

This is reasonably close to the value determined in the following calculation:

CALCULATION OF DOSE RATE FROM RIGHT CIRCULAR CYLINDER

- Sources: 1. Concepts of Dosimetry K. Kase and W. Nelson, Pergamon Press, New York, 1978 pp 118-140.
2. Engineering Compendium of Radiation Shielding, Vol. 1, Jaeger, ed., Springer-Verlag, New York, 1968 Chapter 6 pp 374-387.

$$D = \frac{(1.6 \times 10^{-8} \text{ rad gram}) (\emptyset) (S) (V) (E) \mu_{en}}{\text{MeV} \cdot p}$$

MeV

where

\emptyset = photon flux photon $\text{cm}^{-2} \text{sec}^{-1}$ normalized to unit activity

S_v = volume activity concentration

V = volume

E = average photon energy

μ_{en}/ρ = mass attenuation coefficient for tissue

$$\emptyset = \frac{S_v R_o^2 (F(\theta_1, b_2) + F(\theta_2, b_2))}{4(a+Z)}$$

where

S_v = photon/ cm^3

R_o = radius of sample container

a, z = distance to absorbing medium

F = the volume integral taken from reference 2

for the case in point \emptyset is taken from the figure which follows. \emptyset has

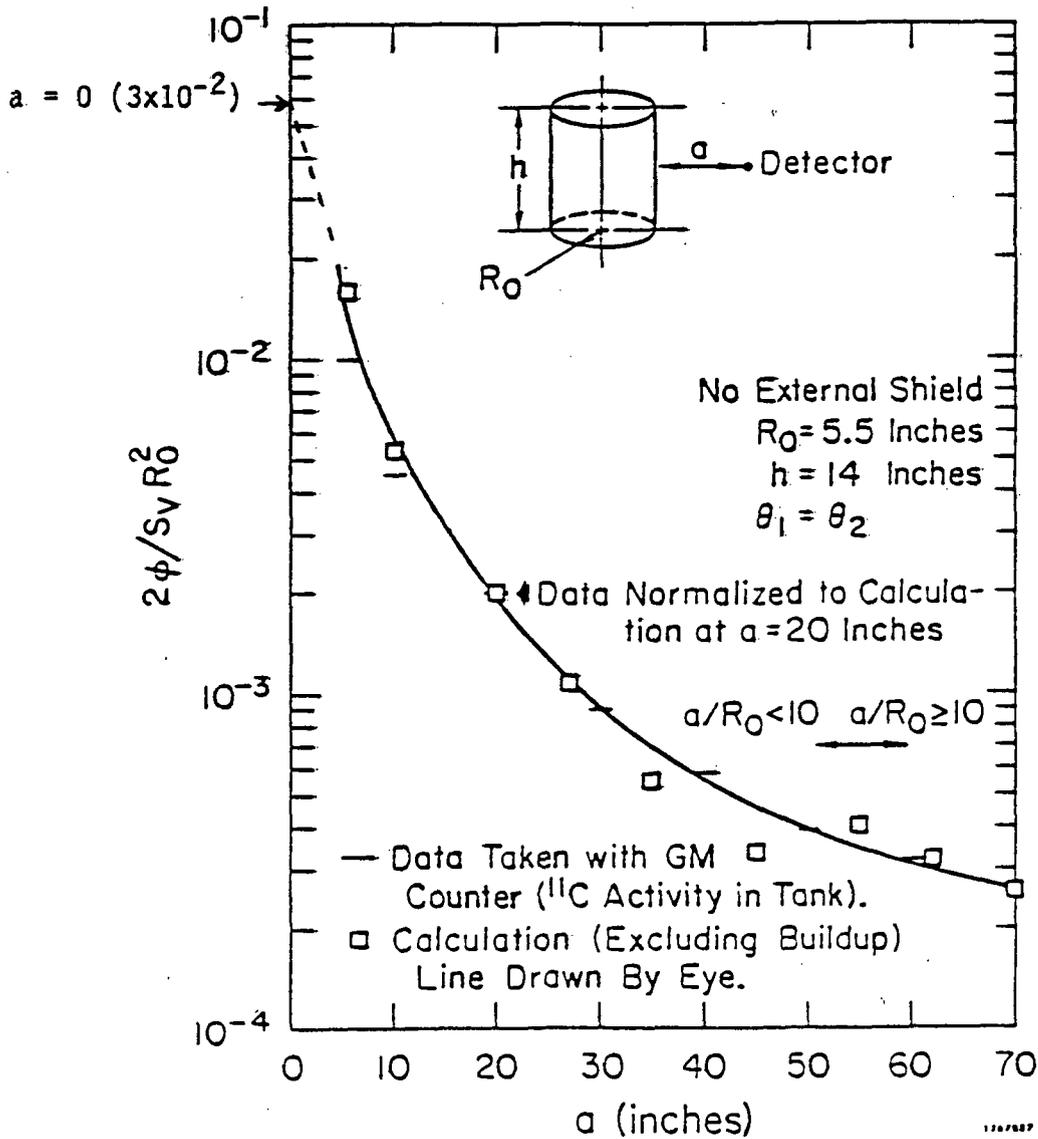
$$\text{units } \frac{\text{photons } \text{cm}^{-2} \text{sec}^{-1}}{\text{disintegrations } \text{sec}^{-1}} \quad S_v R_o^2$$

\emptyset is calculated from the curve by taking the ratio of R_o for the one liter

$S_v R_o^2$ bottle used to that of the cylinder used in the example. $R_o^1/R_o^2 = 1.75/5.5 = 0.32$.

Note that the dose calculated here is a photon dose only. The beta dose in this situation is considered to be negligible due to self absorption in solution, absorption in container walls, and protective clothing. Bremsstrahlung will be neglected.

EVALUATION OF PHOTON FLUX (ϕ) AT THE SURFACE OF A RIGHT CIRCULAR CYLINDER



For $R/R_0 = 0.32$, $\phi = \frac{2.6 \times 10^{-2}}{2} = 1.3 \times 10^{-2}$ photons/second

PHOTONS FROM PRIMARY COOLANT

<u>i</u>	<u>E</u> (MeV)	<u>f</u>	<u>Source</u>	<u>dps/ml</u>	$\frac{N}{\text{photons sec}^{-1} \text{ ml}^{-1}}$	<u>(E)(N)</u>
1.	.080	.026	I-131	4.8×10^8	1.25×10^7	1×10^6
2.	.284	.054	I-131	4.8×10^8	2.6×10^7	7.4×10^6
3.	.364	.82	I-131	4.8×10^8	3.94×10^8	1.43×10^8
4.	.637	.068	I-131	4.8×10^8	3.26×10^7	2.08×10^7
5.	.723	.016	I-131	4.8×10^8	7.68×10^6	5.55×10^6
6.	.530	.87	I-133	2.4×10^8	2.1×10^8	1.11×10^8
7.	.875	.036	I-133	2.4×10^8	8.7×10^6	7.61×10^6
8.	1.24	.016	I-133	2.4×10^8	3.86×10^6	4.79×10^6
9.	1.29	.019	I-133	2.4×10^8	4.6×10^6	5.93×10^6
10.	.856	.01	I-133	2.4×10^8	2.4×10^6	2.06×10^6
11.	.66	.01	I-133	2.4×10^8	2.4×10^6	1.6×10^5
12.	.57	.23	Cs-134	2.33×10^6	5.36×10^6	3.05×10^5
13.	.61	.98	Cs-134	2.33×10^6	2.28×10^6	1.4×10^6
14.	.79	.99	Cs-134	2.33×10^6	2.30×10^4	1.82×10^6
15.	1.03	.01	Cs-134	2.33×10^6	2.33×10^4	2.4×10^4
16.	1.17	.019	Cs-134	2.33×10^6	4.43×10^4	5.18×10^5
17.	1.36	.034	Cs-134	2.33×10^6	7.93×10^5	1.08×10^4
18.	.067	.11	Cs-136	6.66×10^6	7.33×10^5	4.91×10^4
19.	.086	.06	Cs-136	6.66×10^6	4.0×10^6	3.4×10^5
20.	.16	.36	Cs-136	6.66×10^6	2.4×10^6	3.8×10^5
21.	.273	.18	Cs-136	6.66×10^6	1.2×10^6	3.28×10^6
22.	.340	.53	Cs-136	6.66×10^6	3.5×10^6	1.19×10^6
23.	.818	1.0	Cs-136	6.66×10^6	6.66×10^6	5.44×10^6
24.	1.05	.82	Cs-136	6.66×10^6	5.46×10^6	5.73×10^6
25.	1.25	.20	Cs-136	6.66×10^6	1.33×10^6	1.66×10^6

<u>i</u>	<u>E</u> (MeV)	<u>f</u>	<u>Source</u>	<u>dps/ml</u>	<u>photons</u> $\frac{N}{\text{sec}^{-1} \text{ ml}^{-1}}$	<u>(E)(N)</u>
26.	.662	.85	Cs-137	1.03×10^7	8.8×10^6	5.83×10^6
27.	.512	.21	Ru-106 - Rh-106	7.4×10^6	1.55×10^6	7.94×10^5
28.	.622	.11	Ru-106 - Rh-106	7.4×10^6	8.14×10^5	5.06×10^5
29.	1.05	.015	Ru-106 - Rh-106	7.4×10^6	1.11×10^5	1.17×10^4
30.	1.13	.005	Ru-106 - Rh-106	7.4×10^6	3.7×10^4	4.18×10^4
31.	1.55	.002	Ru-106 - Rh-106	7.4×10^6	1.48×10^4	2.29×10^4
32.	.03	.11	Ba-140	7.4×10^6	8.14×10^5	2.44×10^4
33.	.163	.06	Ba-140	7.4×10^6	4.44×10^5	7.24×10^4
34.	.305	.06	Ba-140	7.4×10^6	4.44×10^5	1.35×10^5
35.	.438	.05	Ba-140	7.4×10^6	3.7×10^5	1.62×10^5
36.	.537	.34	Ba-140	7.4×10^6	2.52×10^5	1.35×10^5
37.	.053	.17	Tc-132	7.4×10^6	1.26×10^6	6.67×10^4
38.	.23	.90	Tc-132	7.4×10^6	6.66×10^6	1.53×10^6

$$\bar{E} = \frac{\sum N_i E_i}{\sum N_i}$$

$$\sum N_i = 7.53 \times 10^8 \text{ photons sec}^{-1} \text{ ml}^{-1}$$

$$\sum E_i N_i = 3.32 \times 10^8 \text{ MeV photon sec}^{-1}$$

$$\bar{E} = 0.44 \text{ Mev}$$

@ 0.44 MeV μ_{en}/ρ for tissues $\cong \mu_{en}/\rho$ for \bar{H}_2O

$$\mu/\rho \cong 0.1 \frac{\text{cm}^2}{\text{g}} \text{ from Radiological Health Handbook p. 133}^*$$

$$D = k \left(\frac{0}{R_0 V S_V} \right) (S_V)(V)(E)(\mu_{en}/\rho)$$

$$\dot{D} = \frac{(1.6 \times 10^{-8} \text{ rad MeV}^{-1} \text{ g})}{\text{photon}} (1.3 \times 10^{-2} \text{ cm}^{-2} \text{ sec}^{-1} \text{ dis}^{-1} \text{ sec}) (7.53 \times 10^8 \text{ photon sec}^{-1} \text{ ml}^{-1}) (1 \times 10^3 \text{ ml}) (0.44 \text{ MeV}) (0.1 \frac{\text{cm}^2}{\text{g}})$$

$$= 6.89 \text{ rad/sec}$$

$$= 24800 \text{ rad/hr}$$

* The value μ/ρ is the mass attenuation coefficient. If the linear coefficient is used ($.03 \text{ cm}^2/\text{gm}$) the dose rate from a one liter sample would be approximately 7440 rad/hr which is in better agreement with Teletector data. The principle difference between the two coefficients is that the mass attenuation coefficient includes compton, photoelectric and scattering interactions while the linear attenuation coefficient does not include scatter.

ASSUMPTIONS AND CALCULATIONS (Continued)

3. The skin dose (beta and gamma components) is evaluated using the technique of Walter Snyder et. al. in Absorbed Dose Per Unit Cumulated Activity For Selected Radionuclides and Organs, Society for Nuclear Medicine, New York, October, 1975.

The skin activity was estimated from the measurements made by subject A on March 29, 1979. A measured skin dose rates with a side window geiger tube. On June 6, 1978 a Ba-133 source set up to simulate I-133 was assayed with a side window geiger counter similar to that used by A. The source was a disc 4.7 cm in diameter with a 7 mg/cm² mylar cover. It had a I-131 equivalent activity of 0.19 μCi. The count rate was 0.2 mR/hr closed window and 0.4 mR/hr open window. For calculation purposes the open window data was used.

Subject A indicated that the skin on left forearm was contaminated in a spot about 2 inches diameter (5.08 cm). Therefore, the 4.7 cm Ba-133 disc simulates well the contamination geometry.

To obtain the total skin dose in the period 3/29-31/79, the integrated activity time is required. Figure 1 is a curve of activity in the contaminated area as a function of time. The integral of the curve is μCi-hrs.

Using the formulation of Snyder et. al. the dose to the skin may be obtained by assuming uniform concentration throughout the volume contaminated. The dose to the affected area is given by

$$D = \frac{s \int^t A(t) dt}{f}$$

where s = absorbed dose per μCi-hr. for entire skin organ assuming uniform concentration

$\int A(t) dt$ is the graphically determined μCi-hr. contamination

f = The fraction of the entire organ which is contaminated. This fraction is required to correct for the fact that the Snyder calculation averages the dose to the entire organ from uniform contamination. Here we assume that there is a linear relationship between activity per unit area and dose.

From ICRP 23, Report of the Task Group on Reference Man Pergamon Press, New York, 1975, the following skin data was taken:

Surface area of typed adult male	=	$1.8 \times 10^4 \text{ cm}^2$
Surface area fore subject		
by height = 188 cm		$2.5 \times 10^4 \text{ cm}^2$
by weight = 72 kg		$1.6 \times 10^4 \text{ cm}^2$

Since subject is thin, the values of $1.65 \times 10^4 \text{ cm}^2$ will be used.

The contaminated area was estimated by the subject to be a circle about 5-10 cm diameter. The area would be 19.6-78.5 cm^2 . For calculational purposes a 100 cm^2 area will be used. From this area f may be determined.

$$f = \frac{1 \times 10^2}{1.65 \times 10^4} = 6.1 \times 10^{-3}$$

From Snyder et. al. (MIRD No. 11 p. 185, 1975) $S = 1.6 \times 10^{-4} \text{ rad}/\mu\text{Ci-hr.}$ for I-131. Since I-131 was the major contaminant, we assume it was responsible for all skin dose. This is a nonconservative assumption, since we know that I-131, C_s -134, C_s -137 and Co-60 were also present between the onset of contamination and the second whole body count.

Therefore, the forearm skin dose is estimated:

$$D_{\text{skin}} = \frac{(1.6 \times 10^{-4} \frac{\text{rad}}{\mu\text{Ci-hr}})(1.38 \times 10^3 \mu\text{Ci-hr})}{6.1 \times 10^{-3}}$$

$$= 36.2 \text{ rad} = 36.2 \text{ rem}$$

Since the area contaminated is probably less than 100 cm^2 , this dose represents a lower limit. If an area of 78.5 cm^2 (10 cm diameter) is used, the dose will be 46.1 rem. This would represent the upper bound of skin dose.

Corroborating Information

The body worn TLD for subject A indicated a dose of .89 rem for 3/29/79. This is essentially the same value as calculated in the time/distance study. While this may be fortuitous, it is nevertheless an indicator of the reasonableness of the

time/distance study. Also dose rates for the 100 ml graduated cylinder when treated as a line source at distance less than 12" and by calculation as a photon source of a right circular cylinder indicate compatible dose rates of 2400-4000 R/hr at contact - 1". This agrees with teletector measurements. Since it forms the basis for other whole body dose rates and since there is close agreement with the whole body TLD, credence is lent to the calculation.

Stay times for the cases of A and B were based on interviews with subject A and re-enactments of mockups of the area by the inspector.

TABLE II-B-4

DOSE ESTIMATE FOR SUBJECT B

Circumstances of Exposure

B is the chemistry foreman identified as subject B in Enclosure 3.

The following outline summarizes B's actions on March 29, 1979 to present.

March 29, 1979

<u>TIME</u>	<u>ACTION</u>
~15:00	B and A proceeded to the Unit 1/2 sampling room in preparation for taking a letdown sample.
15:00-15:30	B and A review pipe runs. B enters room. Lineup and collects a Unit 1 letdown sample.
15:30	B and A entered sampling room for initial survey. Stay time 12 sec, dose rate 6-8 R/hr. B aligned valves at sample cooler and started letdown sample recirculation. Stay time ~180 sec. Dose rate W/B and extremity 10 R/hr. B exited area 3 sec 6-8 R/hr. B entered area, adjusted letdown drag valve to establish flow 120 sec @ W/B 17-20 R/hr hand 90 R/hr. B took a 300 ml sample in a polyethelene bottle (2.1 rad/sec contact). He then poured 100 ml into a 100 ml graduated cylinder. B also took a 40 ml aliquot capped the bottle and left room. Sample handling time 60 sec contact (W/B dose ~ 3.33×10^1 rad/sec). 120 sec handling time @ .28 rad/sec w/40 ml aliquot in the primary sample lab. B entered sample room and shut off sample cooler. B exited. Pocket dosimeter was > 5R. One spot on finger was 50 mR/hr W/E-520, w/HP-177. B was whole body counted on 3/29/79. At that time, surface contamination was noted.

On 3/29 a WBC indicated:

I-131	=	6875 nCi
I-133	=	3228 nCi
Cs-134	=	293 nCi
Cs-137	=	372 nCi
Co-60	=	172 nCi

The position scan is strongly indicative of surface contamination. Approximately 50% of the initial activity was on the hands and 20% on the surface of the head. Initially, a small fraction (less than 5%) was internal.

By April 3, the hand contamination was negligible. Surface contamination on the head was essentially all I-131 and was approximately 600-977 nCi. One finger remained contaminated.

A WBC on April 24 indicated 70 nCi on the surface of the head. The data is plotted in figure 1.

The area of the head estimated to be contaminated is 50-100 cm².

An evaluation of B's dose for March 29 and dose due to skin contamination follows in Table 1.

TABLE 1
EVALUATION OF B's DOSE ON 3/29/79 AND SUBSEQUENT
DOSE DUE TO SKIN CONTAMINATION

<u>ACTION</u>	<u>DURATION</u>	<u>DOSE RATES * (R/sec)</u>		<u>DOSE (rem)</u>		
		<u>WHOLE BODY</u>	<u>EXTREMITIES (fingers)</u>	<u>WHOLE BODY</u>	<u>EXTREMITY (fingers)</u>	<u>SKIN OF HEAD</u>
1. B and A enter sample room for initial survey	12 sec	2.2×10^{-3}	2.2×10^{-3}	2.6×10^{-2}	2.6×10^{-2}	2.6×10^{-2}
2. B aligns valves on sample cooler drag valves	180 sec	2.8×10^{-3}	2.8×10^{-3}	5×10^{-1}	5×10^{-1}	5×10^{-1}
3. B exits area	3 sec	2.2×10^{-3}	2.2×10^{-3}	6.6×10^{-3}	6.6×10^{-3}	6.6×10^{-3}
4. Entry to drag valves	12 sec	5.6×10^{-3}	5.6×10^{-3}	6.7×10^{-2}	6.7×10^{-2}	6.7×10^{-2}
5. Adjust drag valves	120 sec	5.6×10^{-3}	2.5×10^{-2}	6.7×10^{-1}	3.0	6.7×10^{-1}
6. Collect 300 ml primary coolant in 1 liter poly bottle	60 sec	1.9×10^{-2}	1.4	1.17	84	1.17
7. Draw 40 ml sample	6 sec	5.6×10^{-3}	2×10^{-1}	3.4×10^{-2}	1.2	3.4×10^{-2}

TABLE 1
EVALUATION OF B's DOSE ON 3/29/79 AND SUBSEQUENT
DOSE DUE TO SKIN CONTAMINATION

ACTION	DURATION	DOSE RATES *		DOSE (rem)			
		WHOLE BODY	EXTREMITIES (fingers)	WHOLE BODY	EXTREMITY (fingers)	SKIN OF HEAD	
8. Pour 100 ml into graduated cylinder from 300 ml bottle	16 sec	1.9×10^{-2}	1.4	3×10^{-1}	22.4	3×10^{-1}	
9. Dump 100 ml back into 1 l bottle	6	6×10^{-3}	4.6×10^{-1}	3.6×10^{-2}	2.7	3.6×10^{-2}	
10. Work with 40 ml sample in primary lab	120 sec	1.2×10^{-2}	2.8×10^{-1}	1.44	33.6	1.44	
11. Skin contamination on head	1ST QTR 1979 2ND QTR 1979					1.8 - 8.8 rem 3.9 to 19.7 rem	
				TOTAL	4.25	147	**6 - 13 rem in the 1st qtr 4 - 19 rem 2nd qtr

*Dose rates are based on teletector data. For extremities the sources are considered to be line sources at less than 12" and point at distances greater than 12"

**This is the sum of the whole body dose and the dose to the skin of the head.

For step 8 the doses may have been somewhat reduced if the 40 ml sample had been taken from the 300 ml aliquot drawn earlier. In that case whole body dose rate would have been about 1.6×10^{-2} R/sec and the extremity dose rate would have been 1.2 R/sec. The net effect would be to reduce the calculated whole body dose to 4 rem and the finger dose to 144 rem. The value of 4.0 rem for W/B is used in Table II-B-2 for the lower bound for whole body dose for the finger dose in Table II-B-2.

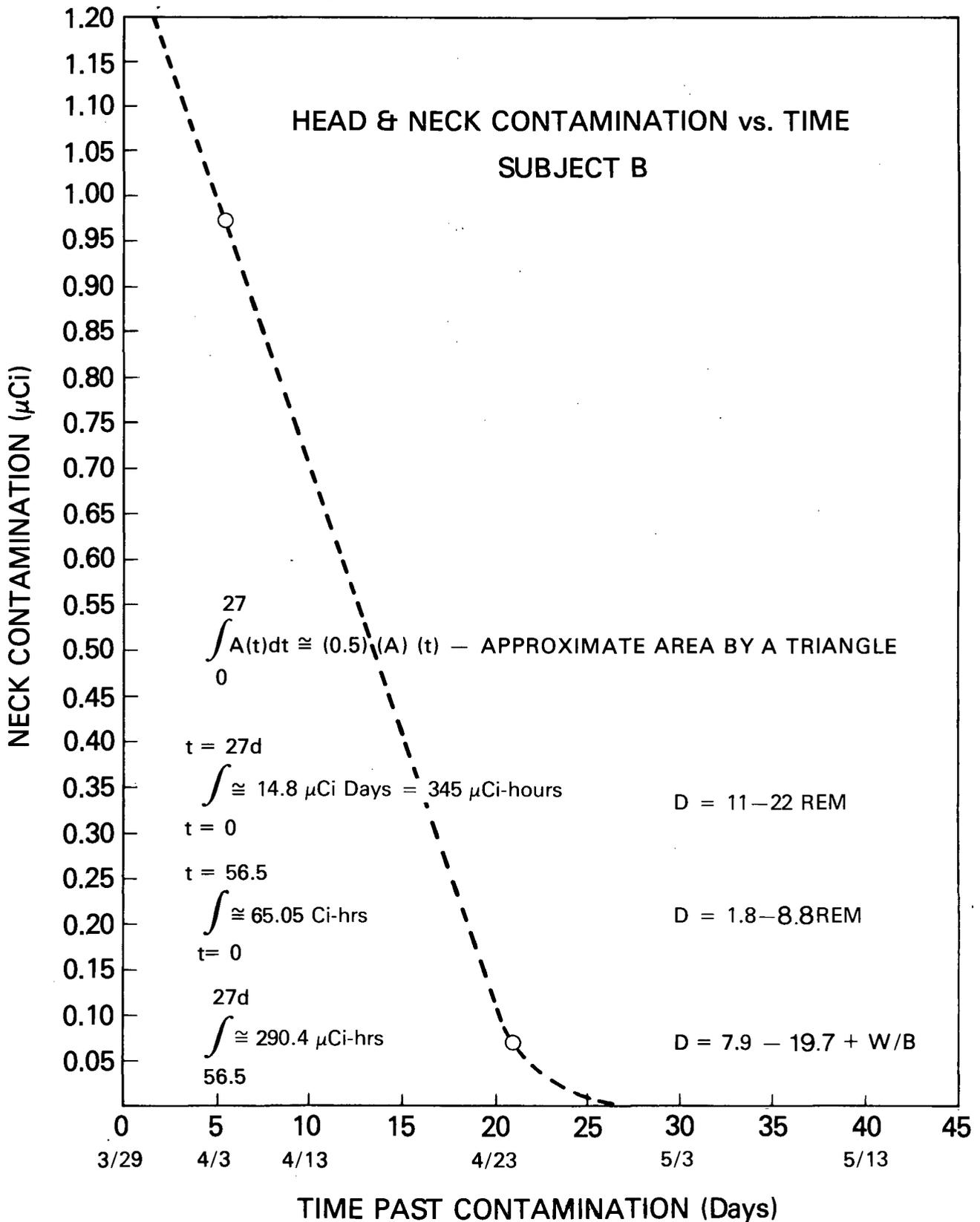


Figure II-B-2

EVALUATION OF SKIN DOSE DUE TO CONTAMINATION AND DIRECT RADIATION

SUBJECT: B, Chemistry Foreman

Assumptions

1. The area contaminated is uniformly contaminated.
2. All body skin is of uniform thickness. This is a slightly conservative assumption if the thickness selected is that of trunk.

The I-131 is assumed to be deposited within the skin. This is apparent since portions of the originally identified contamination were removed upon washing, scrubbing and chemical (KMnO_4) treatment. Following such a regimen residual contamination would be within the tissue.

3. The mass of any skin section is in proportion to its area fraction of the total body.
4. Contamination of the hair resulted in a minimum of 50% being deposited on the skin of the scalp, however for determining the upper bound to skin dose 100% deposition will be assumed.

Evaluation of HP-210 Counting Efficiency for I-131 on Skin.

I-131 (0.156 μCi) was spread over a circular section of fresh, shaved pig skin.

The area contaminated was $(\pi(2.5\text{cm}^2)) = 19.63 \text{ cm}^2$.

An HP-210 probe coupled with an RM-14 rate meter was used to count the skin on contact. The count rate was 2600-3000 cpm. The area contaminated is somewhat larger than the area of the HP-210 pancake probe.

HP-210 Efficiency:

$$\frac{2800 \text{ cpm}}{.156 \mu\text{Ci}/19.63 \text{ cm}^2} = \frac{3.52 \times 10^5 \text{ cpm}}{\mu\text{Ci cm}^{-2}}$$

Data

1. HP-210/RM-14 counting efficiency for I-131 as measured on pig skin is $3.5 \times 10^5 \text{ cpm}/\mu\text{Ci}/\text{cm}^2$.
2. Whole Body Counts (WBC).
 - a. The WBC of 3/29/79 identified a total of 6875 μCi I-131 as both internal and external. No thyroid shield was used. In addition, I-133, Cs-134, Cs-137 and Co-60 were identified.
 - b. A WBC on 3/29/79 with a thyroid shield in place indicated 7045 nCi I-131 total body.
 - c. Analysis of the positional WBC and scan data from self monitoring indicated about 20-25% of activity on head and neck with remainder on hands.
 - d. WBC of 4/3/79 indicated only I-131 and 10 nCi Cs-134. The very rapid elimination of Cs-137 and Co-60 is strongly indicative of skin contamination.
 - e. WBC of 4/3/79 indicated I-131 and 10 nCi Cs-134 only. Major location is at head. The failure to see Co-60 or Cs-137 and great reduction in Cs-134 is very strongly indicative of almost all external.

2. Evaluation of Contamination on Head, Neck, Hands w/HP-210*

a.	<u>DATE</u>	<u>Δt</u>	<u>HEAD</u>	<u>NECK</u>	<u>LEFT WRIST AND FOREARM</u>
	3/29/79	0d	$> 5 \times 10^4$ cpm	3×10^4 cmp	$> 5 \times 10^4$ cpm
	4/4/79	5d	3×10^3	bkg	7×10^3
	4/11/79	12d	3×10^3	bkg	2.5×10^3

$$* \text{MDA} = \frac{1.1 \times 10^{-2} \mu\text{Ci}}{20 \text{ cm}^2} = 11 \text{ nCi}/20\text{cm}^2$$

- b. The WBC for 4/3/79 indicates on the positional scan approximately 90% of activity on the head. The hands had been shielded by 4/3/79. The GSM of the neck indicated background. The only other head area indicating contamination with a GSM was the top of the head, or the temple.

3. Dose to Head Skin

Figure 1 plots the WBC data which indicates a major contamination on the head. These are the data from the 4/3 and 4/24/79 whole body counts. The positional counts indicate that this activity was largely on the head.

CALCULATION METHOD

1. The area under the curve in figure 1 is calculated in terms of $\mu\text{Ci-hrs.}$ for the first quarter then for the second quarter.

For the first quarter, the area is approximated by a rectangle (y axis at $t = 0 \cong 1.17 \text{ uCi.}$ $\Delta t = t = 0 \text{ to } t = 56.5 \text{ hrs}$)

$$\int_0^{55.6} A(t)dt \cong (1.17 \mu\text{Ci})(55.6 \text{ hrs}) = 65.05 \mu\text{Ci-hrs.}$$

For the second quarter, the integrated contamination time is from t=55.6 hrs. to t=23 days (55.6 hrs. to 552 hrs.).

$$\int_{55.6}^{552} A(t)dt \text{ is the area of a triangle} = \left(\frac{1}{2}\right) (\text{Activity}) (\Delta t)$$

$$= (0.5)(1.17 \mu\text{Ci})(552-55.6) = 290.4 \mu \text{ Ci-hrs.}$$

2. The skin contamination on the head is estimated conservatively to be 100% of the total. The area of contamination $\leq 20 \text{ cm}^2$.
3. From ICRP 23 we estimate the skin area of this individual to be $1.7 \times 10^4 \text{ cm}^2$.
4. From MIRD II the dose to the skin organ from uniformly deposited I-131 is $1.6 \times 10^{-4} \text{ rad per } \mu\text{Ci-hr}$.
5. The skin dose is calculated as follows for the 1st quarter.

$$D = \frac{\int_0^{56.5 \text{ hr}} A(t)dt (f_s) (1.6 \times 10^{-4} \frac{\text{rad}}{\mu\text{Ci-hr}}) (A_b)}{A_c}$$

where $\int A(t)dt$ is the total activity time for the period t=0 to t=24:00 3/31/79.

f_s is the fraction of contamination which was on the hair and skin which is estimated to be on the skin only (1.0 for skin of temple).

A_B is the total body skin area = $1.7 \times 10^4 \text{ cm}^2$.

A_C is the area of contamination = $2 \times 10^1 \text{ cm}^2$ (temple).

$$\text{1st qtr} \dots D = \frac{(65.05 \mu\text{Ci-hrs})(1.6 \times 10^{-4} \text{ rad}) (1.7 \times 10^4 \text{ cm}^2)}{\text{mCi-hr} (2 \times 10^1 \text{ cm}^2)}$$

$$= 8.8 \text{ rem}$$

This is the maximum dose due to skin contamination on the head (temple) for the first quarter 1979. To this the first quarter whole body dose must be added. That was 4.11 rem as measured by licensee TLD, 4.25 rem by calculation.

Therefore, the total quarter dose to the head is

$$D_{\text{contamination}} + D_{\text{direct}} = 8.8 + 4.1 = 13.3 \text{ rem.}$$

If the area of the skin contamination is 100, 50, or 20 cm² the following skin doses result:

$$\frac{(17000 \text{ cm}^2 \text{ total body})(65 \text{ } \mu\text{Ci-hrs.})(1.6 \times 10^{-4} \text{ rad})}{100 \text{ cm}^2 \text{ contaminated } \mu\text{Ci-hrs.}} = 1.8 \text{ rad in the 1st qtr}$$

$$\frac{(17000 \text{ cm}^2 \text{ total body})(65 \text{ } \mu\text{Ci-hrs})(1.6 \times 10^{-4} \text{ rad})}{50 \text{ cm}^2 \text{ contaminated } \mu\text{Ci-hrs.}} = 3.6 \text{ rad in 1st qtr}$$

$$\frac{(17000 \text{ cm}^2 \text{ total body})(65 \text{ uCi-hrs})(1.6 \times 10^{-4} \text{ rad})}{(20 \text{ cm}^2 \text{ contaminated}) \text{ uCi-hrs}} = 8.8 \text{ rad in the 1st qtr}$$

1.8 rem represents the lower bound for head dose, due to contamination of the skin.

8.8 represents the upper bound due to contamination of the skin.

Subject B's chest worn TLD was read out on 3/30/79. It indicated a whole body gamma dose of 4.1 rem. This value is within 15% of the value calculated herein for whole body dose and lends credence to the calculated value. Further supporting data is indicated by the pocket dosimeter which read 5R at the end of the activity on March 28.

6. The maximum skin dose due to skin contamination of the head for the second quarter is calculated as follows:

$$D = \frac{648}{56.5} \left(\int A(t) dt \right) (f_s (1.6 \times 10^{-4} \frac{\text{rad}}{\mu\text{Ci-hr.}})) \frac{A_b}{A_c}$$

$$= (290.4 \mu\text{Ci-hrs.})(0.5)(1.6 \times 10^{-4} \frac{\text{rad}}{\mu\text{Ci-hr.}})(1.7 \times 10^4) = 19.75 \text{ rem}$$

Here f_s is estimated at 0.5 as B was not positive of the fraction of long term contamination on the temple.

For a contaminated area of 100 cm^2 the skin doses would be $\sim 4 \text{ rem}$. To this the whole body dose for the second quarter must be added.

In the first quarter, the skin dose will be higher than calculated here due to contamination by I-133, Cs-134, Cs-137 and Co-60.

The licensee should make additional evaluations of the dose due to contamination for the first and second quarter 1979. In addition, to the dose due to I-131, the dose due to I-133, Cs-134, Cs-137 and Co-60 needs to be evaluated. The evaluation here is, therefore non-conservative in that it intentionally neglects dose contributions from 4 isotopes. The purpose here was to demonstrate that if, in the first quarter, a significant skin dose could be realized from I-131 alone, then the contributions of other contaminants added in would surely be a significant dose.

APPENDIX II-C

DIRECT RADIATION MEASUREMENTS MADE BY METROPOLITAN EDISON COMPANY ON AND OFF THREE MILE ISLAND

- NOTES:
1. See Figure II-3-2 for location of GE series survey points.
 2. A "+" in the offsite survey table indicates that the survey was made in the indicated sector but at a greater distance from the plant.
 3. A "+" in the onsite survey table indicates that the survey was made at a location other than the GE series survey point.
 4. A "x/y" indicates a $\beta\gamma/\gamma$ measurement. If only a single value without a diagonal (/) is shown, it is not known whether the measurement is $\beta\gamma$ or γ only. A dash (-) indicates that the $\beta\gamma$ or γ component was not measured.
 5. All values are in mR/hr.
 6. N-01, -11, -21, etc. refers to 0.5-1, 1-2, and 2-3 miles, respectively north of the site.

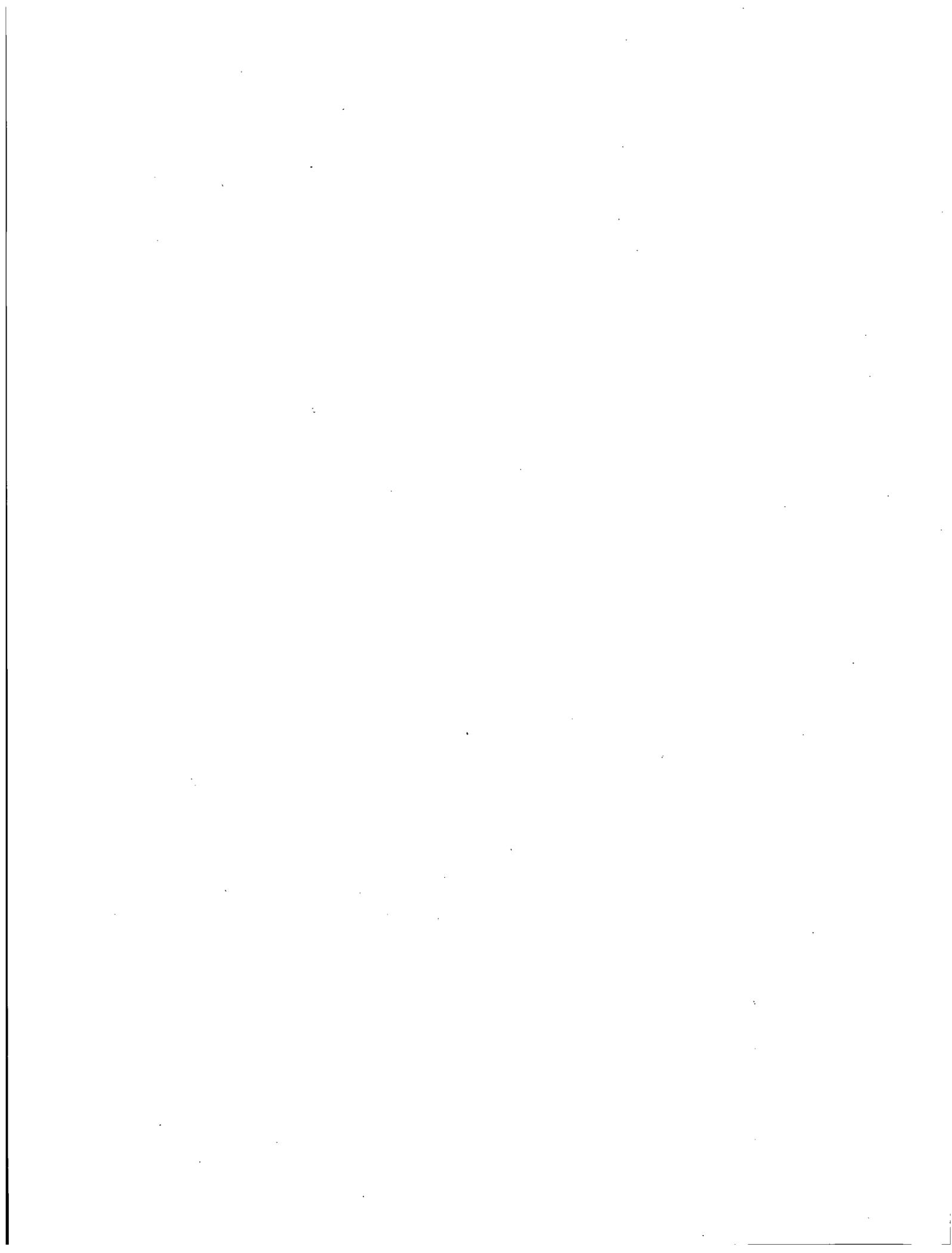


TABLE II-C-1

Direct Radiation Measurements Made by Metropolitan Edison Off Three Mile Island

(Locations shown are closest survey points in each sector)

Date	Time	N-11	NNE-01	NE-01	ENE-01	E-01	ESE-01	SE-01	SSE-01	S-01	SSW	SW-11	WSW-11	W-11	WNW-11	NW-21	NNW-21
3/28/79	Prior to 0700																
	0700		<1				<1										
	0800												<1	<1	<1		
	0900																
	1000		<1	+			<1										+
	1100	<1	2		3	<1	2.5 <1 <1	<1	<1								
	1200																
	1300	1															
	1400	++ +															
	1500	3 <1	20 50														

II-C-1

TABLE II-C-1 (Continued)

Direct Radiation Measurements Made by Metropolitan Edison Off Three Mile Island

(Locations shown are closest survey points in each sector)

Date	Time	N-11	NNE-01	NE-01	ENE-01	E-01	ESE-01	SE-01	SSE-01	S-01	SSW	SW-11	WSW-11	W-11	WNW-11	NW-21	NNW-21
3/28/79	1600	+	<1 <1														
	1700		<1 <1														
	1800		<1 + 0.5 <.05 <1 <1 <.05	<1	<1	<1	<1										
	1900		<.05														
	2000																+
	2100																
	2200															12	+
	2300		<1														+
																	+
																	+

II-C-2

TABLE II-C-2

Direct Radiation Measurements Made by Metropolitan Edison Off Three Mile Island

(Locations shown are closest survey points in each sector)

Date	Time	N-11	NNE-01	NE-01	ENE-01	E-01	ESE-01	SE-01	SSE-01	S-01	SSW	SW-11	WSW-11	W-11	WNW-11	NW-21	NNW-21
3/29/79	0000	+	3.5 3.5													+	+
	0100															++	+
	0200																
	0300																
	0400		27														
	0500																
	0600														30/20		
	0700														1/1 2/2	6/3 +	
	0800																
	0900		<0.5														
	1000																

II-C-3

TABLE II-C-2 (Continued)

Direct Radiation Measurements Made by Metropolitan Edison Off Three Mile Island

(Locations shown are closest survey points in each sector)

Date	Time	N-11	NNE-01	NE-01	ENE-01	E-01	ESE-01	SE-01	SSE-01	S-01	SSW	SW-11	WSW-11	W-11	WNW-11	NW-21	NNW-21
	1100																
	1200		<.5/<.5							+							
	1300																-/<.05
	1400																
	1500																
	1600																
	1700																
	1800									+							<1/-
	1900									+							
	2000																
	2100																
	2200																
	2300	+++															<.1/<.1
																	0.5 +

II-C-4

TABLE II-C-3

Direct Radiation Measurements Made by Metropolitan Edison Off Three Mile Island

(Locations shown are closest survey points in each sector)

Date	Time	N-11	NNE-01	NE-01	ENE-01	E-01	ESE-01	SE-01	SSE-01	S-01	SSW	SW-11	WSW-11	W-11	WNW-11	NW-21	NNW-21
3/30/79	0000	+		+			.15 .2/<.1										
		<.1 ++	.2/-				0.2/-		0.2/-								
	0200						2.0 0.4	+									
	0300				+		2.0/- 2.5/- .1-.2 .1-.15										
	0400	+	.02				.1/- .5										.03/.02
	0500																
	0600	++++ .6 .75 2	.5 +	<.1			.2						.15/.05 +	.1/.05			
	0700						.35						.15/.15 +	.15/.05			

11-C-5

TABLE II-C-3 (Continued)

Direct Radiation Measurements Made by Metropolitan Edison Off Three Mile Island

(Locations shown are closest survey points in each sector)

Date	Time	N-11	NNE-01	NE-01	ENE-01	E-01	ESE-01	SE-01	SSE-01	S-01	SSW	SW-11	WSW-11	W-11	WNW-11	NW-21	NNW-21
	0800									+		.2/.05		1.5/.4			
	0900						10/.4 1.0	+		++ +++ 8/4 ++ 3.0/- +							
	1000						1.2/.5	5.5/2.0						.1/.02			
	1100						1.0/.4	4.0/1.8									
	1200		2/1 2/.5														
	1300																
	1400																
	1500													2/.7			
	1600													6/- 1/-			

II-C-6

TABLE II-C-3 (Continued)

Direct Radiation Measurements Made by Metropolitan Edison Off Three Mile Island

(Locations shown are closest survey points in each sector)

Date	Time	N-11	NNE-01	NE-01	ENE-01	E-01	ESE-01	SE-01	SSE-01	S-01	SSW	SW-11	WSW-11	W-11	WNW-11	NW-21	NNW-21
	1700													.1/.05			
														1.2/.2			
														6/1			
														.6/.1			
														.3/.05			
														.06/.01			
														.02/.01			
	1800													.04/.02	+		
	1900																
	2000													<.01/<.01			
	2100	+														+	+
	2200														+	+	+
	2300		5/2														<.01/-

II-C-7

TABLE II-C-4

Direct Radiation Measurements Made by
Metropolitan Edison On Three Mile Island

Time Interval	GE-1	GE-2	GE-3	GE-4	GE-5	GE-6	GE-7	GE-8	GE-9	GE-10	Surveys at Other Locations On Three Mile Island
3/28/79 Prior to 0700											
0700											+
0800											
0900											+
1000				7							+
											+
1100					2						+
											+
1200		<1								8	+
1300		3									+++ +++ +

NOTE: GE number = sample location on Three Mile Island

II-C-8

TABLE II-C-4 (Continued)

Direct Radiation Measurements Made by
Metropolitan Edison On Three Mile Island

Time Interval	GE-1	GE-2	GE-3	GE-4	GE-5	GE-6	GE-7	GE-8	GE-9	GE-10	Surveys at Other Locations On Three Mile Island
1400		30									++
1500		70 8 12									+
1600		50									+ ++
1700	40									140	+ +
1800		26									+
1900	<1								3	10	+++ ++
2000											++
2100											+

6-C-II

TABLE II-C-4 (Continued)

Direct Radiation Measurements Made by
Metropolitan Edison On Three Mile Island

Time Interval	GE-1	GE-2	GE-3	GE-4	GE-5	GE-6	GE-7	GE-8	GE-9	GE-10	Surveys at Other Locations On Three Mile Island
2200											
2300	30/20							1	-/10	365/50	+

01-C-10

TABLE II-C-5

Direct Radiation Measurements Made by
Metropolitan Edison On Three Mile Island

Time Interval	GE-1	GE-2	GE-3	GE-4	GE-5	GE-6	GE-7	GE-8	GE-9	GE-10	Surveys at Other Locations On Three Mile Island
3/29/79											
0000											
0100											
0200											
0300									30/20	1	
0400							26	28	50	25	
0500									150/100	3/2	
0600										0.5/0.5	
0700								6/3.5			
0800			<0.5	0.5/-	0.5/-			5/1	11/5 11/5	<1/<1	+

II-C-11

TABLE II-C-5 (Continued)

Direct Radiation Measurements Made by
Metropolitan Edison On Three Mile Island

Time Interval	GE-1	GE-2	GE-3	GE-4	GE-5	GE-6	GE-7	GE-8	GE-9	GE-10	Surveys at Other Locations On Three Mile Island
0900	<0.5										++++ +++
1000	<0.5/<0.5	<0.5/<0.5		1/<.5	0.8/-	10/10 3/3 1.5/<0.5	1.5/.8 4/3	5/2	5/3	2.5/1.5	+
1100			<0.5/<0.5	.5/<.5	0.5/<0.5 0.5/<0.5	<0.5/<.5	1/0.5	3/<0.5 4/2	1/<0.5 5/3.5	5/3	
1200	3/3	0.5/<0.5									
1300									3/1	10/3	+
1400	1/<1	8/4	10/4					<1/<1	2/1 2/1	10/4	+
1500											
1600								9/4			+

II-C-12

TABLE II-C-5 (Continued)

Direct Radiation Measurements Made by
Metropolitan Edison On Three Mile Island

Time Interval	GE-1	GE-2	GE-3	GE-4	GE-5	GE-6	GE-7	GE-8	GE-9	GE-10	Surveys at Other Locations On Three Mile Island
1700								7/3			
1800						0.4/0.3	5/1.8	2/1	0.6/0.25		
1900	0.04/0.01	0.05/0.025				1.4/0.9	1.4/0.9	0.4/0.3	0.2/0.2	0.15/0.1	++
2000			.04/.01	.05/.03	.3/.14						
2100	.02/.01	.02/.02			<.01/<.01	.08/.08	13/7	0.4/0.2	0.2/0.1	0.05/0.04	
2200	<.01/<.01	.02/<.01		.03/.01	<.01/<.01	.16/.02	.2/.08	.2/.1	1.8/0.4	0.9/0.2	
2300				.03/.01							

II-C-13

TABLE II-C-6

Direct Radiation Measurements Made by
Met-Ed On the Island

Time Interval	GE-1	GE-2	GE-3	GE-4	GE-5	GE-6	GE-7	GE-8	GE-9	GE-10	Surveys at Other Locations On Three Mile Island
3/30/79											
0000		0.1/0.03									
0100	0.4/0.1		.03/.01	.05/.03		.04/.01 .05/.04	.2/.05 .01/<.01 .1/.05	3/0.8	.7/.25	.3/.15	
0200							.8/.2 .7/- .25/- .3/-				
0300	.3/.08	.3/.1	.25/.07	1.4/0.4	.3/.1	.2/.06	.3/.07 14/3.5 4/1 5/1	.3/.1	.3/.13	.2/<.01	
0400			.06/.04	.1/.03	.14/.08	.7/.2	2/0.5	<.01 <.01	.1/.1		+
0500	0.03/0.01	0.01/<.01								<.01/<.01	
0600	2.5/0.5	.01/.01	.4/.1	.4/.08	.65/.06	.15/.05	.2/.05	.4/.1	2.0/0.5	3.5/1.0	+

II-C-14

TABLE II-C-6 (Continued)

Direct Radiation Measurements Made by
Met-Ed On the Island

Time Interval	GE-1	GE-2	GE-3	GE-4	GE-5	GE-6	GE-7	GE-8	GE-9	GE-10	Surveys at Other Locations On Three Mile Island
0700											+
0800							0.3/- 8/4 30/9	14/8 25/8			
0900					4/1	10/-		0.3/0.1	0.5/0.2		++ ++
1000	.3	0.2/-	0.3/-	0.3/<0.1			9/3	13/6	7/4 19/7 25/9		
1100	1.7/0.7 8/1	4/1 7/2.5 8/2.5			.8/.2 7/3			8/3.5 15/5 10/3 11/3	6/4 5/1.5 7/1 6/1		+
1200	6/1 4/.7	2/0.3	5.5/1.5 5/1.5 3/1.2				4/0.5 1.5/0.5 4/1.8	18/5 18/3.5 3/0.4	-/3.5 17/5 5/2		+
1300	6.5/2.5						3/0.5	3.5/1		12/2 6.5/2.5 17/6	+

II-C-15

TABLE II-C-6 (Continued)
 Direct Radiation Measurements Made by
 Met-Ed On the Island

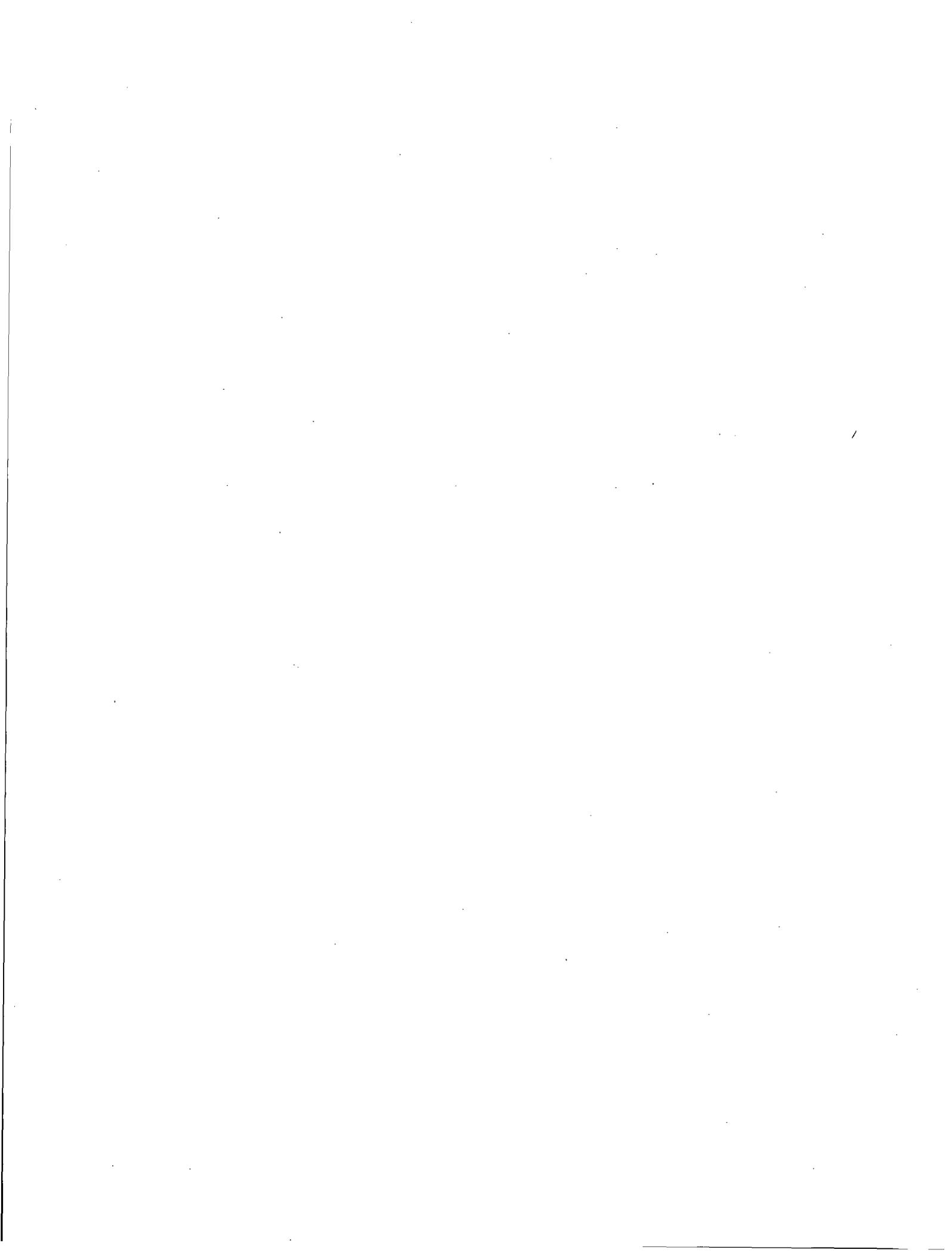
Time Interval	GE-1	GE-2	GE-3	GE-4	GE-5	GE-6	GE-7	GE-8	GE-9	GE-10	Surveys at Other Locations On Three Mile Island
1400	0.8/0.1 1.3/.4	3/1.2	0.8/0.1	4/0.3	4/0.5				7/1 110/20 5/3	6/1	+
1500									90/10 40/25		+
1600								2.5/-	40/24 50/10 20/-	8/2 30/	+
1700	<0.1/-							11/3 5/- .15/.05	100/ 30/- 80/15	.6/.2	+
1800									20/10 14/6 60/7	0.9/0.1 3/0.6	++ ++ +
1900	.1/.03	<.01/<.01			.02/.01	.01/.01	.01/.01	.4/.01	10/1.2	10/1.8	
2000	.15/.03	.04/.01	.04/.01 .01/.01	.07/.01 .02/.01	.01/.01	<.01/<.01	<.01/<.01	.3/.1	8/1	5/0.5	

91-C-II

TABLE II-C-6 (Continued)
 Direct Radiation Measurements Made by
 Met-Ed On the Island

Time Interval	GE-1	GE-2	GE-3	GE-4	GE-5	GE-6	GE-7	GE-8	GE-9	GE-10	Surveys at Other Locations On Three Mile Island
2100									20/-	20/-	++
2200				2/-		.15/-	.05/-	.25/-	12/-		
2300			0.1/-	0.2/-	0.1/-				12/- 2.5/-	8/2	

II-C-17



APPENDIX II-D

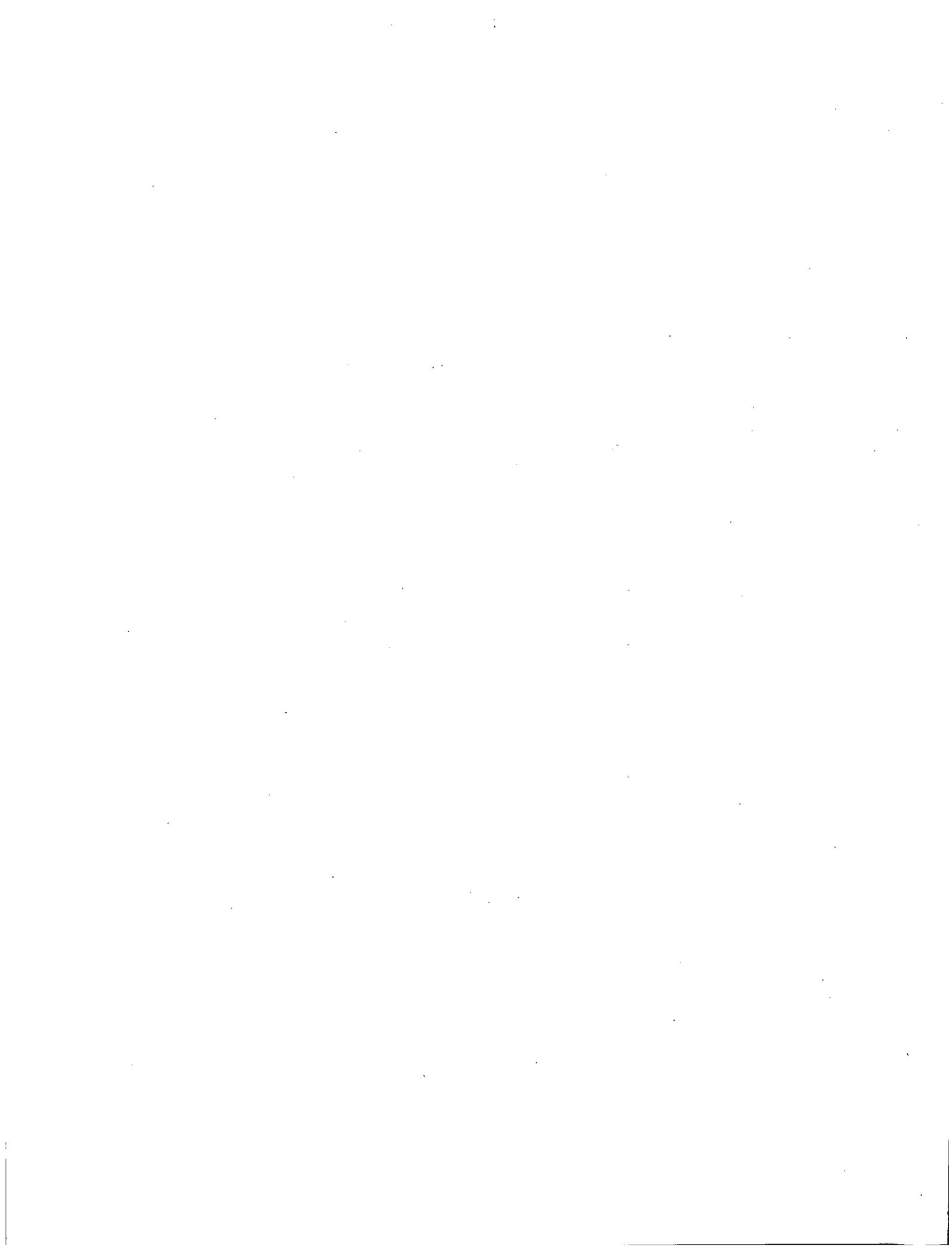
CAUTIONS REGARDING THE USE OF INSTALLED RADIATION MONITORING SYSTEM DATA



CAUTIONS REGARDING THE USE OF INSTALLED RADIATION MONITORING SYSTEM DATA

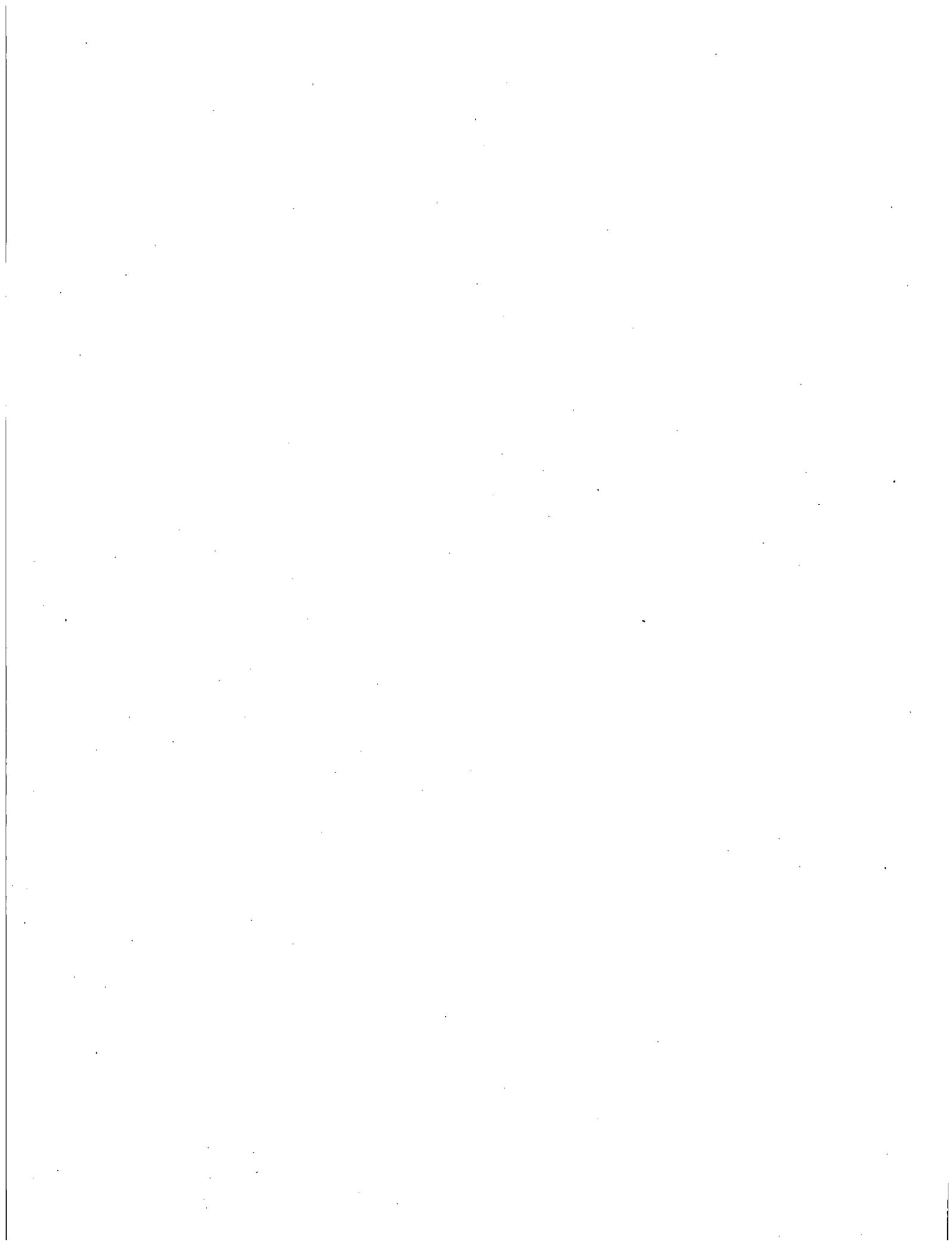
When using radiation monitoring system data, it should be remembered that the radiation monitoring system provides very little quantitative data. During the accident, atmospheric and process radiation monitors were measuring high background radiation from radioactive gases in the auxiliary building and high levels of direct radiation from the reactor coolant system letdown piping and other sources such as ventilation ducts. Area radiation monitors were, in some cases, measuring both gases around the monitor and direct radiation from certain process components. The influence of a high gaseous activity around the monitors made it impossible to quantify radiation being measured. In addition, it is not always possible to rely on the time indications on the strip chart because they were not clearly and regularly marked and/or because the paper did not advance properly on the recorder. Certain of the monitor traces were impossible to identify because the recorder was not inking properly. Since many of the monitors were responding simultaneously, their traces overlapped and were difficult to follow.

Certain monitors can, however, be used to establish times with reasonable assurance. The times may be established based on the monitors' general response in relation to certain operational occurrences. An example of these would be HP-213, Incore Instrumentation Area Monitor, and IC-R-1091, Intermediate Closed Cooling Water Letdown Monitor. Both of these monitors responded downward almost instantaneously on reactor trip, probably due to the decay of Nitrogen 16 which has a 7 second half life.



APPENDIX II-E

ACRONYMS, DEFINITIONS, AND UNITS OF MEASURE



APPENDIX II-E

ACRONYMS, DEFINITIONS, AND UNITS OF MEASURE

1. ACRONYMS

ABST	Auxiliary building sump tank
AMS	Aerial monitoring system
ANI	American Nuclear Insurers
BRH	Bureau of Radiological Health
BWST	Borated water storage tank
CaSO ₄ :Dy	Calcium Sulfate: Dysprosium
CaSO ₄ :Tm	Calcium Sulfate: Thulium
CRO	Control room operator
DCCD	Dauphin County Civil Defense
DER	Department of Environmental Resources
DOE	Department of Energy
DOP	Diocetylphythalate
ECC	Emergency Control Center
ECS	Emergency Control Station
EMOV	Electromatic relief valve
EPA	Environmental Protection Agency
ES	Engineered safeguard
FM	Frequency Modulation

FSAR	Final Safety Analysis Report
GAI	Plant page system
Ge(Li)	Lithium Drifted Germanium
GM	Geiger Mueller
GPU	General Public Utilities
GPUSC	General Public Utilities Service Corporation
HEPA	High efficiency particulate air (filters)
HEW	Department of Health, Education, and Welfare
HP	Health physics
IWFS	Industrial waste filtration system
IWTS	Industrial waste treatment system
LiF	Lithium fluoride
LOCA	Loss of coolant accident
MDA	Minimum detectable activity
MET ED	Metropolitan Edison
MUT	Makeup tank
MWHT	Miscellaneous Waste Holdup Tank
MWST	Miscellaneous Waste Storage Tank
NAWAS	National Warning System
NPDES	National Pollution Discharge Elimination System
NRC	Nuclear Regulatory Commission
OTSG	Once-through steam generator
PEMA	Pennsylvania Emergency Management Agency
PORC	Plant Operations Review Committee
PSE&G	Public Service Electric and Gas
RAP	Radiological Assistance Plan

RB	Reactor Building
RCBHT	Reactor coolant bleed holdup tank
RCDT	Reactor coolant drain tank
REMP	Radiological environmental monitoring program
RCP	Reactor coolant pump
RMC	Radiation Management Corporation
RMS	Radiation monitoring system
RWP	Radiation work permit
SAI	Science Application Incorporated
SAM-2	Stablilized assay meter
SCBA	Self-contained breathing apparatus
SOP	Step-off-pad
TLD	Thermoluminescent dosimeters
TMI	Three Mile Island
WECST	Waste evaporator condensate storage tank
WGDT	Waste gas decay tank

2. DEFINITIONS

Assessment actions - those actions taken during an accident to obtain and process information that is necessary to make decisions to implement specific emergency measures, to include protective actions.

Corrective actions - those emergency measures taken to ameliorate or terminate an emergency situation at or near the source of the problem in order to prevent an uncontrolled release of radioactive material or to reduce the magnitude of a release, e.g., shutting down equipment, repair and damage control.

Protective actions - emergency measures taken after an uncontrolled release of radioactive material has occurred for the purpose of preventing or minimizing radiological exposures of persons that would be likely to develop if the actions were not taken.

Recovery actions - those actions taken after the emergency to restore the plant or facility as nearly as possible to its preemergency condition.

Protective Action Guides (PAGs) - projected radiological doses or dose commitment values to individuals in the general population that warrant protective action following a release of radioactive material. Protective actions would be warranted provided the reduction in individual dose expected to be achieved by carrying out the protective action is not

offset by excessive risks to individual safety in taking the protective action. The PAG does not include the dose that has already occurred prior to the assessment.

Action levels - radiological dose rates; specific contamination levels of airborne, waterborne, or surface deposited concentrations of radioactive materials; or specific instrument indications (including their rates of change) that may be used as thresholds for initiating such specific emergency measures as designating a particular class of emergency, initiating a notification procedure, or initiating a particular protective action.

3. UNITS OF MEASUREMENT

cc = cubic centimeter

cfm = cubic feet per minute

Ci = curie

cpm = counts per minute

gpm = gallons per minute

l = liter

mg/cm² = milligram/square centimeter

ml = milliliter

mrem = millirem

mR = milliroentgen

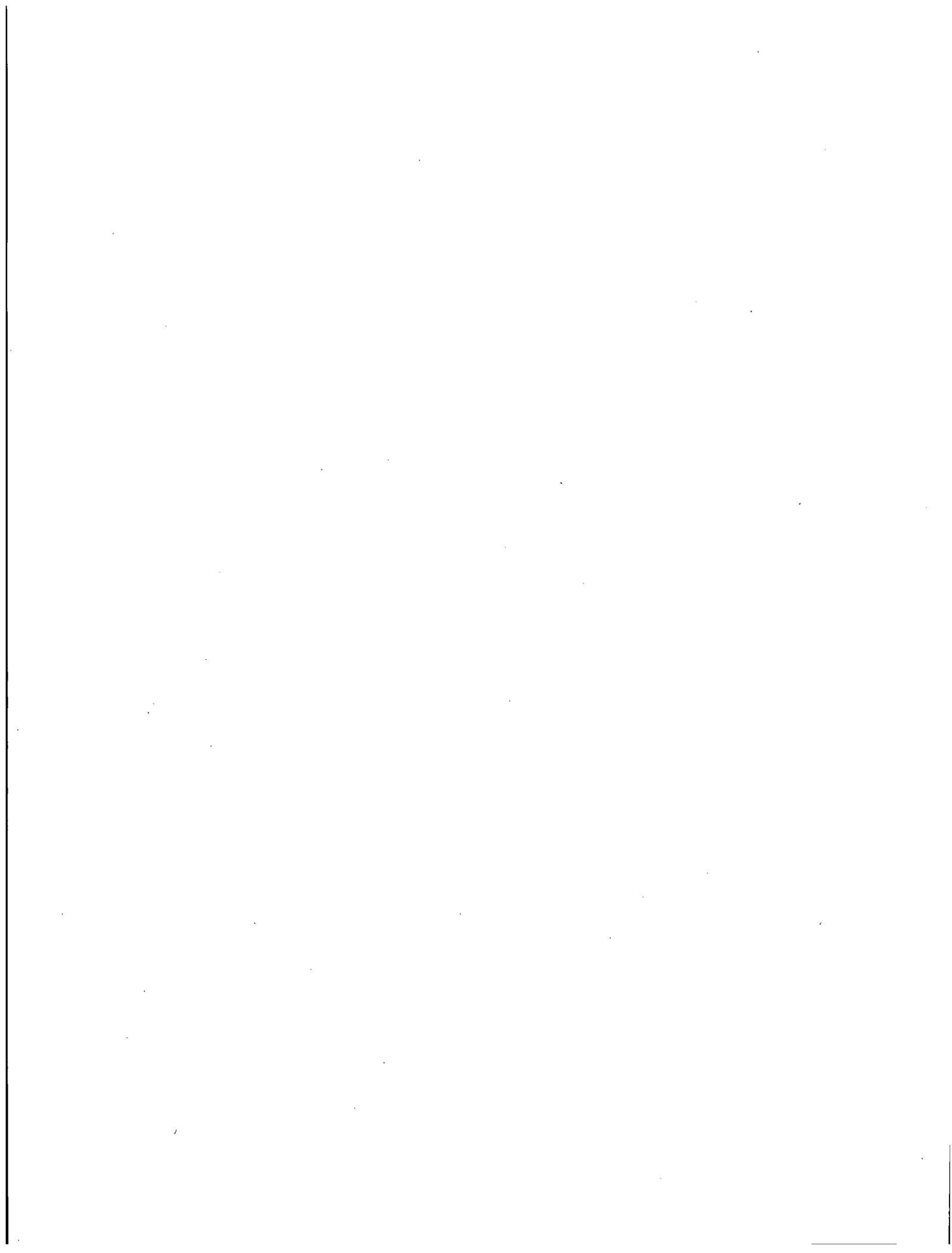
ppm = parts per million

pCi = picocurie

R/hr = roentgen per hour

uCi = microcurie

APPENDIX II-F
POTENTIAL ITEMS OF
NONCOMPLIANCE



APPENDIX II-F
POTENTIAL ITEMS OF NONCOMPLIANCE

Listed below are items being evaluated as potential items of noncompliance with requirements and commitments. These matters will be handled through the normal enforcement channels of the Office of Inspection and Enforcement.

- A. Technical Specification 6.2, "Organization," states in Sections 6.2.1 and 6.2.2 that the unit organization and the organization of the corporate technical support staff shall be as shown on Figure 6.2-1.

Contrary to the above, on March 28, 1979, the unit and corporate technical support staff was different from that specified in that:

1. A position titled, "Superintendent of Administration and Technical Support" was added to the organization on September 18, 1978 and filled on March 1, 1979, such that the "Supervisor, Radiation Protection and Chemistry," reported to this new position rather than directly to the "Station Superintendent/Senior Unit Superintendent;" and
2. On March 5, 1979, the position "Station Superintendent" was changed to "Station Manager," which reports directly to the "Vice President, Generation," rather than to the "Manager, Generation Operations;" and
3. The position of "Chemical Supervisor" has been vacant since the issuance of the Technical Specifications; and
4. There are two "Supervisor of Maintenance" positions, one for each unit, rather than one; and

5. A position titled "Superintendent of Maintenance" has been added such that the "Supervisors of Maintenance" report to this new position rather than directly to the "Station Superintendent (Station Manager)/ Senior Unit Superintendent." (See Details II, Section 1.1)
- B. Technical Specification 6.8, "Procedures," requires in Section 6.8.1 that written procedures shall be established, implemented and maintained for implementing the Site Emergency Plan.

Three Mile Island Nuclear Station Health Physics Procedure 1670.9, "Emergency Training and Emergency Drills," Revision 4, dated January 16, 1978:

1. Identifies in Section 3.1, the on-site emergency job categories and requires that training programs for these categories will be conducted on an annual (calendar year) basis; and
2. Describes in Sections 3.1.1 through 3.1.9, the training program for all on-site emergency job categories; and
3. Requires in Section 3.1.3.2 that Radiological Monitoring Teams be composed of Radiation/Chemistry Technicians and Auxiliary "A" Operators who have received the appropriate training; and
4. Requires in Section 3.1.5.2 that Repair Party Teams be composed of Maintenance Shift workers who have received the appropriate training.

Contrary to the above, during calendar year 1978, all individuals having emergency responsibilities were not trained in that two Emergency Directors, one Accident Assessment individual, eight Radiological Monitoring Team Members, and 37 Repair Party Team Members had not received the training

specified in Procedure 1670.9. On March 28, 1979, during an emergency, at least four individuals assigned to Radiological Monitoring Teams and seven individuals assigned to Repair Party Teams performed these emergency duties without having been trained as specified in Procedure 1670.9. (See Details II, Section 1.2.1)

- C. Technical Specification 6.4, "Training," requires that a retraining and replacement training program for the unit staff be maintained that meets or exceeds the requirements and recommendations of Section 5.5 of ANSI N18.1-1971.

ANSI N18.1-1971 recommends in Section 5.5.1, "Retraining," ten topics to be included in a retraining program.

Contrary to the above, as of March 28, 1979, a retraining program meeting or exceeding ANSI N18.1-1971 recommendations has not been maintained for members of the Radiation Protection and Chemistry staff in that only 2 of the 10 topics specified were included in the program. (See Details II, Section 1.2.2)

- D. Technical Specification 6.8, "Procedures," requires in Section 6.8.2 that procedures which implement the Emergency Plan be followed, and that changes thereto shall be reviewed by the Plant Operations Review Committee and approved by the Unit Superintendent prior to implementation.

Contrary to the above, an improper change to Station Health Physics Procedure 1670.7, "Emergency Assembly, Accountability and Evaluation," was implemented on March 28, 1979, without the required review and approval in that:

1. By a memorandum dated October 13, 1978, from the Radiation Protection Supervisor to all departments an additional assembly area was designated and the method to be used to perform personnel accountability was modified without subsequent change to Procedure 1670.7; and
2. On March 28, 1979, in response to an emergency, some licensee personnel followed the approved procedure for accountability while others followed the accountability guidance in the October 13, 1978 memorandum, creating some confusion and delaying prompt attainment of full accountability. (See Details II, Section 1.3)

- E. Environmental Technical Specification 5.7 requires that detailed written procedures for instrument calibration be prepared and followed.

Three Mile Island Nuclear Station Surveillance Procedure 1302-5.24, Revision 3, dated December 19, 1974, specifies the method of calibration and requires that it be performed annually.

Contrary to the above, as of March 29, 1979, eight environmental air samplers had not been calibrated since 1974. (See Details II, Section 1.7.1.1)

- F. Technical Specification 6.8, "Procedures," requires in Section 6.8.1 that written procedures shall be established, implemented and maintained for implementing the Emergency Plan.

Three Mile Island Nuclear Station Administrative Procedure 1004, Section 2, "Three Mile Island Emergency Plan," Revision 2, dated February 15, 1978:

1. Requires in Section 2.1, that the "Station Superintendent/Senior Unit Superintendent, Unit Superintendent/Shift Supervisor/Unit Superintendent-Technical Support in the Control Room will, after reviewing the emergency conditions, classify the emergency as one of the following:

- a. Personnel or Local Emergency,
- b. Site Emergency, and
- c. General Emergency.

He will make this classification according to the condition of Table 1 of this Plan, and initiate actions according to the Emergency Plan Implementing Procedures, and according to his own best judgment."

2. States in Table 1 of Section 2.1 that a Site Emergency exists when there is a:

- a. Loss of primary coolant pressure, coincident with high reactor building pressure and/or high reactor building sump level (Condition c); or
- b. Reactor building high range gamma monitor alert alarm (Condition e).

Contrary to the above, the shift supervisor:

- a. Failed to declare a Site Emergency at 0415 hours on March 28, 1979, at which time Condition "c" of Table 1 of Section 2.1 of the Three Mile Island Emergency Plan had occurred; and

b. Failed to declare a Site Emergency at 0635 on March 28, 1979, at which time Condition "e" of the Three Mile Island Emergency Plan had occurred. (See Details II, Section 2.1)

G. 10 CFR 20.106, "Radioactivity in effluents to unrestricted areas," requires in Section (a) that a licensee not release to an unrestricted area radioactive material in concentrations which exceeds the limits specified in Appendix B, Table II of 10 CFR 20 when averaged over a period not greater than one year. For mixtures of radionuclides, the limiting value is determined as follows:

$$\frac{C_a}{MPC_a} + \frac{C_b}{MPC_b} + \frac{C_c}{MPC_c} \leq 1$$

Where C_a , C_b , and C_c are the concentrations of radionuclides a, b, and c, and MPC_a , MPC_b , and MPC_c are the maximum permissible concentrations of nuclides a, b, and c established in Appendix B.

Contrary to the above:

The release of xenon-133, xenon-133m, xenon-135, xenon-135m and krypton-88 during the period 0700 on March 28, 1979, through 2400 April 30, 1979, produced an average annual release rate approximately eleven times greater than is permitted. (See Details II, Section 3.1.2.1)

H. 10 CFR 20.401, "Records of surveys, radiation monitoring, and disposal," requires in Section (a) that each licensee maintain records showing the radiation exposure for all individuals for whom personnel monitoring is required on a Form NRC-5 or equivalent and in Section (b) requires that each licensee maintain records of the results of surveys required by 10 CFR 20.201(b).

Contrary to Section (a) of the above:

During the period March 1-31, 1979, records of radiation exposure for at least 5 individuals for whom personnel monitoring was provided were not maintained on a Form NRC-5 or equivalent; and,

Contrary to Section (b) of the above:

- a. Records made by offsite survey teams during March 28-30, 1979, of results of approximately 500 surveys performed in unrestricted areas bordering Three Mile Island Nuclear Station were not maintained. These results were transmitted by radio and telephone to the Emergency Control Station where they were recorded in the Emergency Control Station Survey Log, which was maintained; however, results recorded in this log are incomplete (e.g., specific instrument used, whether window open or closed) in nearly every instance. (See Details II, Section 3.3)
 - b. During March 28-30, 1979, records of surveys of at least ten individuals who were found to be contaminated with radioactivity were not maintained. (See Details II, Section 3.2)
 - c. During March 28-30, 1979, records of radiation surveys necessary to assess dose of at least 5 individuals who handled samples of highly radioactive primary coolant were not maintained. (See Details II, Section 3.2)
- I. Technical Specification 6.12, "High Radiation Area," requires that each area in which the intensity of radiation is greater than 1000 mrem/hr be provided with locked doors to prevent unauthorized entry into the area and that, any individual entering the area be equipped with a continuously indicating dose rate monitoring device.

Contrary to the above:

- a. From 1100 on March 28 until the afternoon of March 30, 1979, the doors to the auxiliary building were not locked and access was not otherwise controlled even though the intensity of radiation in the building was known to be a high radiation area with radiation levels much greater than 1000 mrem/hr during this period; and,
 - b. From the evening of March 28 to the evening of March 29, 1979, at least three entries into the auxiliary building were made by individuals who were either not equipped with a radiation monitoring device which continuously indicated the dose rate, or were equipped with an instrument with insufficient range such that the instrument "pegged" or read full scale in some areas. (See Details II, Section 3.2)
- J. 10 CFR 20.103; "Exposure of individuals to concentrations of radioactive materials in air in restricted areas," requires in Section (a)(3) that the licensee make suitable measurements of the concentration of radioactive materials in air for detecting and evaluating airborne radioactivity in restricted areas for the purposes of determining compliance with the regulation in 10 CFR 20.103(a)(1).

Contrary to the above, no measurements were made of the concentrations of airborne radioactive materials in the Unit 2 auxiliary building for periods during which individuals were exposed from 1100 on March 28 through midnight March 30, 1979, nor in the Unit 1 nuclear sample room and primary chemistry laboratory for periods during which individuals were exposed from 0400 March 28 through 0800 March 30, 1979. (See Details II, Section 3.2)

Contrary to the above, no measurements were made of the concentrations of airborne radioactive materials in the Unit 2 auxiliary building for periods during which individuals were exposed from 1100 on March 28 through midnight March 30, 1979, nor in the Unit 1 nuclear sample room and primary chemistry laboratory for periods during which individuals were exposed from 0400 March 28 through 0800 March 30, 1979. (See Details II, Section 3.2)

- K. FSAR Section 12.3.2.10 states that a Radiation Work Permit (RWP) is required for all personnel working in radiation areas except in emergency situations. In emergency situations the entry should be made by qualified personnel carrying radiation monitors and each entry must be documented.

Contrary to this commitment, during the period March 28-30, 1979, an emergency situation, individuals were permitted to enter high radiation areas with radiation levels much greater than 1000 mrem/hr and with unknown levels of airborne activity, without an RWP, and these entries were not documented. (See Details II, Section 3.2.1)

- L. FSAR Section 12.1.2.11 states that all sample and recirculation lines are shielded by placing them in shielded cubicles or pipe chases which are located at sufficient distances from normally occupied area.

Contrary to this commitment, the primary coolant sample lines are not shielded or located in pipe chases at sufficient distances as they leave the Unit 2 fuel handling building and instead, pass through the normally occupied areas in the Unit 1 fuel handling building and hot machine shop. (See Details II, Section 3.2.2)

- M. 10 CFR 20.201, "Surveys," requires in Section (b) that each licensee shall make or cause to be made such surveys as may be necessary to comply with the regulations 10 CFR 20. Also, in Section (a), 10 CFR 20.201

defines a "survey" to mean an evaluation of the radiation hazards incident to the production, use, release, disposal, or presence of radioactive materials or other sources of radiation under a specific set of conditions, and that when appropriate, such evaluation includes a physical survey of the location of materials and equipment and measurements at levels of radiation or concentrations of radioactive material present.

1. 10 CFR 20.101, "Exposure of individuals to radiation in restricted areas," limits the dose to the whole body, to the extremities, and to the skin of the whole body, to values therein.

Contrary to the above, surveys to assure compliance with 10 CFR 20.101 were not made when:

- a. On March 29, 1979, an Auxiliary Operator was permitted to enter areas of the auxiliary building where an exposure rate of 30 R/hr was measured, without first performing a survey of his quarterly accrued dose. The individual's whole body dose was measured on return from the auxiliary building and when added to his previous dose resulted in a whole body dose in excess of the specified limit; and, (See Details II, Section 3.2.4)
- b. On March 29, 1979, a Nuclear Engineer entered an area of the auxiliary building where the radiation level was greater than that which could be measured by his portable survey instrument (2R/hr). Failure to perform a survey of the exposure rate in this area resulted in the individual receiving a whole body dose in excess of the specified limit; and, (See Details II, Section 3.2.4)

- c. On March 29, 1979, a chemistry foreman was permitted to repeatedly enter high radiation areas and handle highly radioactive reactor coolant samples without performing a survey of his accumulated dose. The whole body dose received by the foreman during this sample handling exceeded the specified limit; and,
(See Details II, Section 3.2.4)
 - d. On March 29, 1979, a chemistry foreman and a radiation protection foreman were permitted to handle a highly radioactive reactor coolant sample without first performing a survey of hand and forearm exposure rates. Handling of this sample resulted in a dose to the hand of the chemistry foreman and forearm of the radiation protection foreman in excess of the specified limit; and,
(See Details II, Section 3.2.4)
 - e. On March 28 and March 29, several individuals received skin contamination of the head and other parts of the body sufficient to cause exposure rates, in the range of 20-100 mR/hr when measured with a hand held survey instrument and no survey of the dose to the skin of these individuals has been made.
(See Details II, Section 3.2.4)
2. 10 CFR 20.105, "Permissible levels of radiation in unrestricted areas," limits radiation levels in unrestricted areas.

Contrary to the above, from 1700-2238 on March 28, 1979, and from 0340-0540 on March 29, 1979, surveys to assure compliance with 10 CFR 20.105 were not made in that measurements of the levels of radiation were not made in areas where the radioactive effluent

plume from the plant stack was predicted to have been and which could reasonably be expected to cause radiation levels in excess of the specified limit. (See Details II, Section 3.3)

- N. 10 CFR 20.101, "Exposure of individuals to radiation in restricted areas," requires that no licensee possess, use, or transfer licensed material in such a manner as to cause any individual in a restricted area to receive in any period of one calendar quarter a dose in excess of three rem to the whole body, or 18 3/4 rem to the hands and forearms, or 7 1/2 rem to the skin of the whole body.

Contrary to the above, during the first calendar quarter of 1979:

1. Two individuals working in restricted areas received doses to the whole body of 4.175 rem and 3.870 rems as measured by personnel dosimetry devices; and,
 2. One individual working in a restricted area received a dose to the whole body of 4.115 rem as measured by personnel dosimetry devices, and a calculated dose to the hands and forearms of from 50 to 147 rem; and
 3. One individual working in a restricted area received a calculated dose to the hands and forearms of from 44 to 54 rem. (See Details II, Section 3.2.4.5)
- O. 10 CFR 20.202, "Personnel Monitoring," requires that the licensee supply appropriate personnel monitoring equipment and require its use for each individual who enters a restricted area and is likely to receive a dose in excess of 25 percent of the applicable value specified in 10 CFR 20.101.

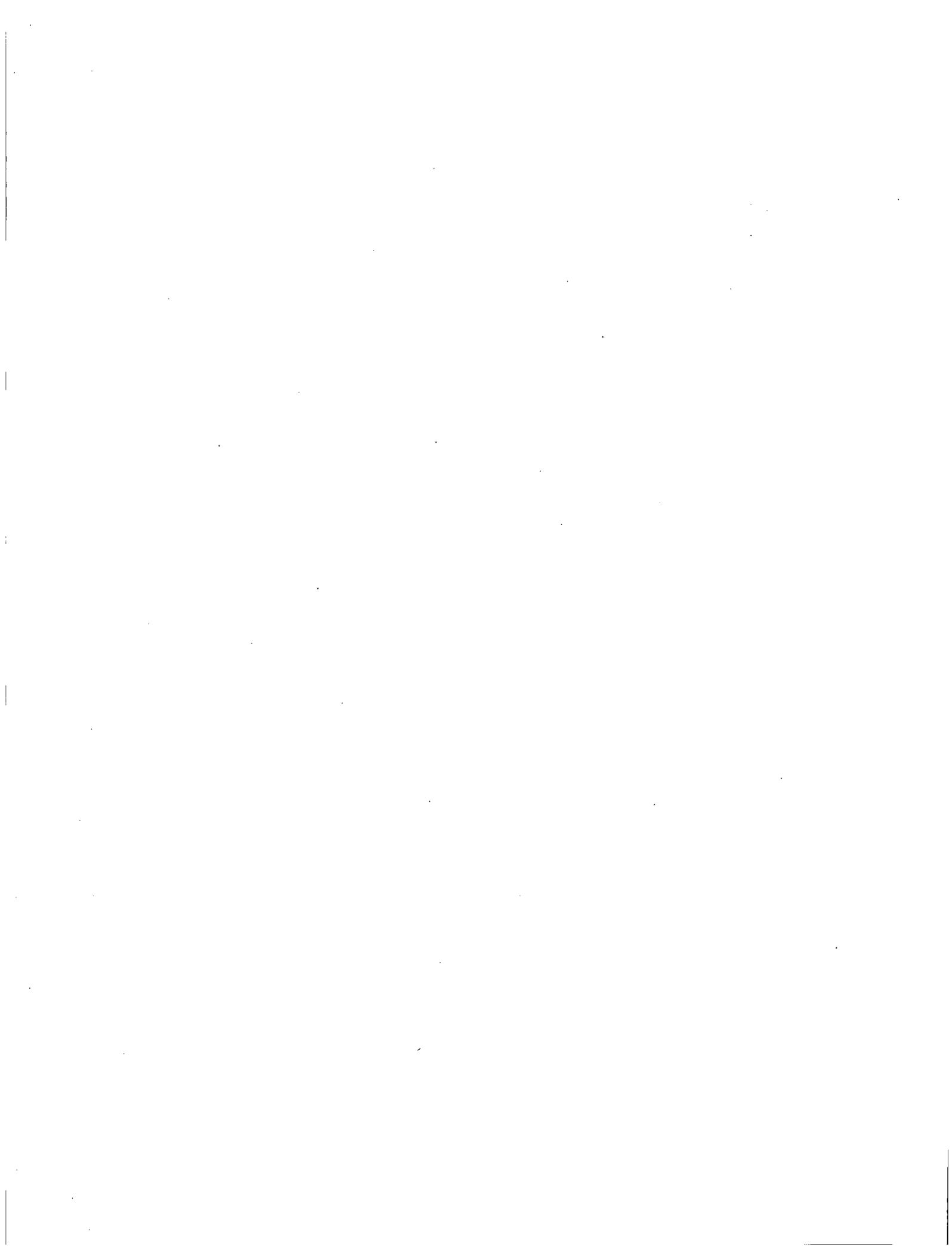
Contrary to the above, on March 29, 1979 the licensee did not provide personnel monitoring devices to measure the dose to:

1. The hands of at least two individuals received doses in excess of the limit specified in 10 CFR 20.101 during collection and handling radioactive samples of reactor coolant; nor to
 2. The hands or feet of the members of a repair party team who placed plastic over the radioactive water on the floor of the auxiliary building. (See Details II, Section 3.2.4.5)
- P. 10 CFR 20.105(b), "Permissible levels of radiation in unrestricted areas," requires in Section (b) that no licensee shall possess, use or transfer licensed material in such a manner as to create, in any unrestricted area, radiation levels which, if an individual were continuously present in the area, could result in his receiving a dose in excess of two millirems in any one hour.

Contrary to the above, licensee dosimetry shows that radioactivity released from Three Mile Island Nuclear Station caused average radiation levels in unrestricted areas on Kohr and S. Shelley Islands, which were about 30 mrem/hr and 3.1 mrem/hr for approximately a twenty-nine hour period on March 29 and 30, 1979, respectively. These levels would have resulted in an individual present receiving a dose in excess of two millirems in an hour. (See Details II, Section 3.3)

ENCLOSURE 1

This Enclosure contains the memoranda establishing the Office of Inspection and Enforcement investigation into the Three Mile Island Unit 2 accident of March 28, 1979.





UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 20, 1979

MEMORANDUM FOR: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Bradford
Commissioner Ahearne

THRU: Lee V. Gossick
Executive Director for Operations

FROM: John G. Davis, Acting Director
Office of Inspection and Enforcement

SUBJECT: INVESTIGATION OF THREE MILE ISLAND ACCIDENT BY THE
OFFICE OF INSPECTION AND ENFORCEMENT

This is to confirm our discussion of April 13, 1979, concerning the Three Mile Island accident investigation underway by the Office of Inspection and Enforcement (IE).

The investigation performed by IE has two basic goals:

1. To establish, in a comprehensive manner, the facts concerning the Three Mile Island accident. The parameters of this effort are further described in Attachment A.
2. To evaluate the performance of the licensee in association with the Three Mile Island accident as a basis for corrective action or enforcement action as appropriate.

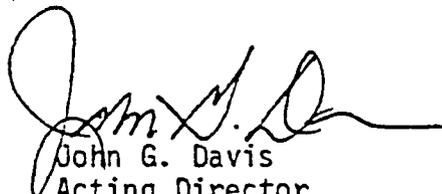
The Office of Inspection and Enforcement investigation does not include, under our current plans, the following:

1. Any evaluation of the actions of the NRC or any of its organizational components during the course of this accident or recovery period.
2. Any evaluation of the actions of other agencies during the course of the accident, or during the recovery period of the accident.

3. Any review and evaluation of the NRC regulatory process as it relates to the Three Mile Island accident for "lessons learned." IE is not collecting information concerning nor evaluating:

- Legislative authority of the NRC
- Rules and regulations of the NRC
- Safety research
- Licensing process
- Inspection and enforcement process

The Office of Inspection and Enforcement will expand its investigation as directed by the Commission or the EDO. In the absence of such direction, IE is proceeding with its investigation as outlined in this memorandum.


John G. Davis
Acting Director
Office of Inspection
and Enforcement

Enclosure:
IE/TMI Conceptual Outline

OFFICE OF INSPECTION AND ENFORCEMENT
THREE MILE ISLAND ACCIDENT INVESTIGATION

Conceptual Outline

1. The IE investigation of the Three Mile Island accident is directed toward:
 - a. Within the time period of the investigation, establishing the facts concerning the immediate causes of the accident and the actions of the plant and the licensee staff during the course of the accident.
 - b. Within the time period of the investigation, establishing the facts concerning the actions of the licensee, the NRC, other Federal agencies, and appropriate state agencies.
 - c. Evaluating the performance of the licensee during and in response to the Three Mile Island accident as a basis for corrective action or enforcement action as appropriate.
2. The investigation consists of two parts conducted in parallel:
 - a. Operational - The inplant, reactor operations situation. This will cover the time period from the closing of the auxiliary Feedwater System valves (or other earlier immediate cause of the accident) to the restart of reactor coolant pump 1A (about 8:00 p.m., March 28, 1979).
 - b. Radiological - The inplant and environmental radiological conditions. This will cover the time period from the beginning of the accident until about midnight on March 31, 1979.
3. The Office of Inspection and Enforcement will develop factual information concerning the condition of the plant and the environment and the performance of the licensee for the time periods subsequent to those described in Item 2, above. However, this information will not be included in the IE Three Mile Island Accident Investigation. (Note that this does not include development of information concerning NRC activities. The information will include the licensee's reaction and response to NRC activities but will not develop, for example, how an NRC position or recommendation was formulated.)

4. The investigation will include:

a. Sequence of Events

(1) Operational

Identify the sequence of events in relation to facility operation. Determine licensee actions or lack of actions related to operation of the facility within licensed parameters, constraints and limits. Determine the participation of licensee supervision, management and engineering support. Determine causes of the event.

(2) Radiological

Identify the sequence of events in relation to licensee activities in radiation control on site and off site. Determine licensee actions or lack of actions related to controlling and monitoring on site exposures, protection of workers, and control and monitoring of off site releases.

b. Immediate Cause of Accident

(1) Equipment

Trace the performance and maintenance history of important equipment which malfunctioned at the beginning or during the early phases of the incident. Define the signals or other intelligence provided to operators concerning serviceability and availability of equipment. Identify the serviceability of equipment required for operation.

(2) Procedures

Determine the requirements contained in surveillance and maintenance procedures and the appropriateness of procedures. Identify failures to follow procedures. Determine appropriateness of communications between plant groups on plant and equipment status. Determine whether emergency operating procedures were appropriate and were followed.

(3) Staff Performance

Describe the performance of the operators and other licensee personnel during the accident. Review training of operators particularly training concerning response to off-normal limits.

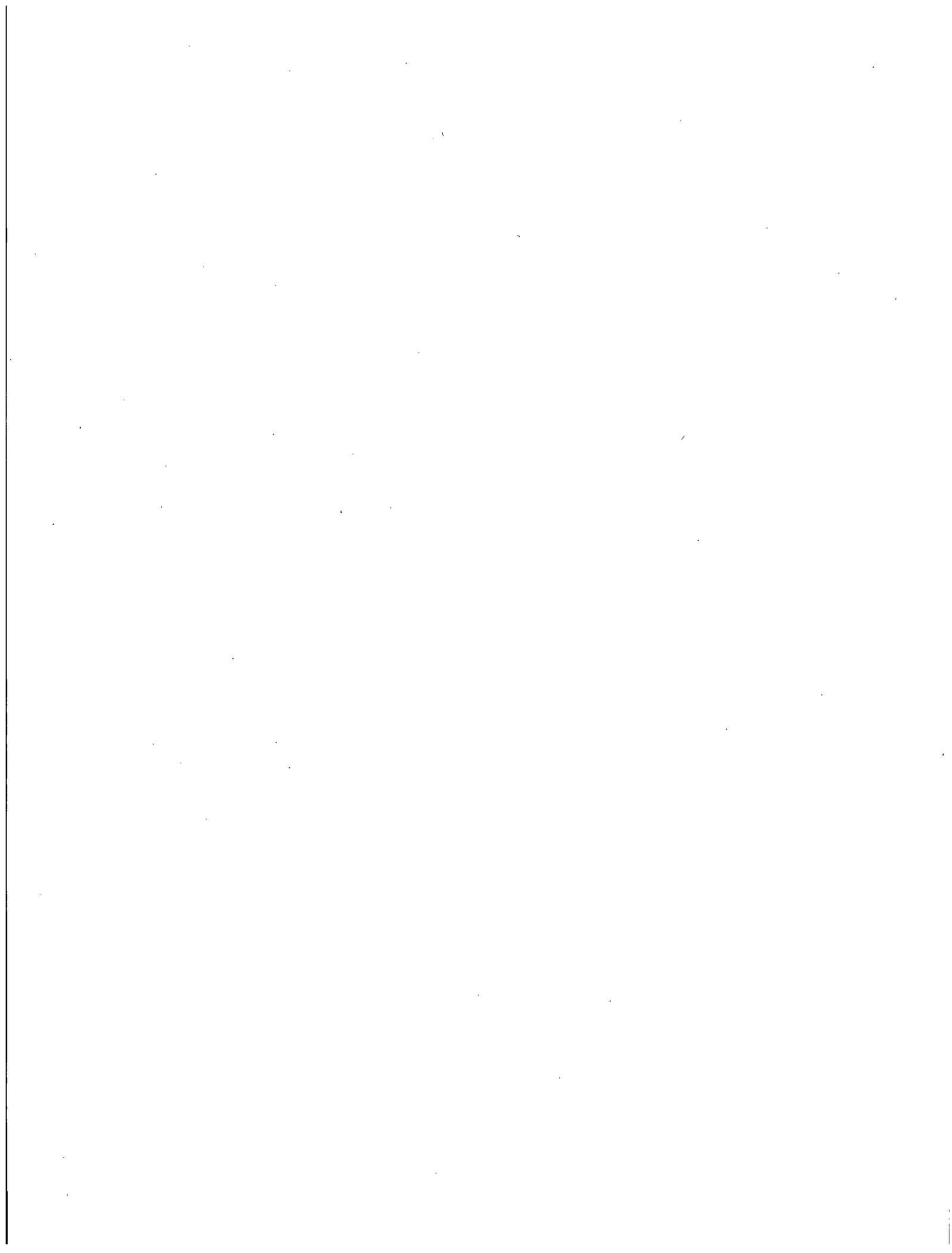
c. Licensee Management of Accident

Describe the response of the licensee from the standpoint of management and supervision of the accident. Identify the engineering and radiological support requested and received and its source. Determine the licensee's reaction to and analysis of the accident as it unfolded with particular emphasis on engineering analysis of alternative modes of accident recovery.

d. Emergency Plan Activation

Examine the licensee's emergency plan implementation to include preplanning and tests. Develop a detailed chronology of the implementation of the emergency plan with particular emphasis on timeliness of notification.

5. This investigation is being managed by the NRC Region I Office.





UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUN 6 1979

MEMORANDUM FOR: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Bradford
Commissioner Ahearne

THRU: Lee V. Gossick (Signed) Lee V. Gossick
Executive Director for Operations

FROM: John G. Davis, Acting Director
Office of Inspection and Enforcement

SUBJECT: INVESTIGATION OF THREE MILE ISLAND ACCIDENT
BY THE OFFICE OF INSPECTION AND ENFORCEMENT

Reference is made to my memorandum of above subject dated April 20, 1979.

The investigation by the Office of Inspection and Enforcement of the accident and of the licensee's actions during the course of the accident has been underway for approximately seven weeks. During the conduct of this investigation, IE has interviewed approximately 110 people and has examined other information sources in its efforts to determine what transpired, the proximate causes, and licensee actions before and during the incident proper.

As a result of a current review of the progress of the investigation, I believe the following matters should be specifically called to your attention:

1. The investigation is complex, involving extensive interplay of mechanical and control systems with human actions. This complexity has been further compounded by the need to release information prior to the completion of the investigation, by other concurrent investigations and by the need to reply to specific questions prior to completion of the IE investigation. This may lead to IE being unable to meet its projected August 1, 1979 date for the investigation report. We are currently examining our schedules to account for anticipated additional investigatory work. We will inform you by June 25, 1979, of any necessary rescheduling of the projected date for the investigation report.

cc: E. Kevin Cornell, EDO
 Harold R. Denton, NRR
 Saul Levine, RES
 Robert B. Minogue, SD
 William J. Dircks, NMSS
 James J. Cummings, OIA
 Robert G. Ryan, SP
 James R. Shea, IP
 Harold D. Thornburg, IE
 Norman C. Moseley, IE
 E. Morris Howard, IE
 James H. Sniezek, IE
 Boyce H. Grier, IE
 James P. O'Reilly, IE
 James G. Keppler, IE
 Karl V. Seyfrit, IE
 Robert H. Engelken, IE
 Dudley Thompson, IE
 Leonard I. Cobb, IE
 James M. Atlan, IE

DISTRIBUTION:

L. V. Gossick
 T. A. Rehm
 J. G. Davis
 EDO RF
 IE RF
 SECY (3)
 MPA

OFFICE →	IE AD	ELD	EDO			
SURNAME →	JGB [initials] s:sc		LVGossick			
DATE →	6/6/79	6/ /79	6/ /79			

ENCLOSURE 2

This Enclosure contains a listing of the interviews conducted during the course of this investigation.

ENCLOSURE 2

INTERVIEW LIST

<u>INTERVIEW NO.</u>	<u>POSITION TITLE</u>
1	Station Manager & Unit 2 Superintendent
2	Shift Foreman C
3	Control Room Operator D
4	Control Room Operator C
5	Shift Supervisor E
6	Radiation/Chemistry Technician I
7	Radiation Protection Foreman A
8	Engineer H
9	Control Room Operator C
10	Auxiliary Operator B
11	Radiation Protection Foreman A
12	Shift Foreman B
13	Shift Supervisor C
14	Shift Foreman C
15	Control Room Operator D
16	B&W Test Coordinator
17	Shift Supervisor E
18	Radiation Protection Foreman D
19	Radiation Protection Consultant A
20	Supervisor, Radiation Protection & Chemistry

<u>INTERVIEW NO.</u>	<u>POSITION TITLE</u>
21	Radiation Protection Foreman B
22	Radiation Protection Supervisor
23	Chemistry Foreman B
24	Radiation/Chemistry Technician K
25	Supervisor, Radiation Protection & Chemistry
26	Unit 1 Supervisor, Station Operations
27	Unit 2 Superintendent, Technical Support
28	Radiation Protection Consultant A
29	Radwaste Foreman
30	Corporate Technical Analyst
31	US NRC Inspector D
32	Site Protection Sergeant B
33	Engineer E
34	Unit 2 Supervisor, Station Operations
35	Engineer J
36	Auxiliary Operator G
37	Shift Foreman C
38	Auxiliary Operator H & Shift Supervisor E
39	Radiation Protection Foreman B
40	US NRC Inspector H
41	Met Ed Group Discussion- Emergency Planning

<u>INTERVIEW NO.</u>	<u>POSITION TITLE</u>
42	Radiation/Chemistry Technician I
43	Radiation Protection Foreman C
44	US NRC Inspector K
45	Radiation Protection Technician S
46	Pennsylvania Bureau of Radiation Protection
47	Radiation/Chemistry Technician N
48	Engineers C and D
49	Radiation/Chemistry Technicians H and W
50	Radiation/Chemistry Technician E
51	Radiation/Chemistry Technicians F, J
52	Site Protection Sergeant C
53	B&W Site Manager B
54	Supervisor Contract Radiation Protection
55	Region 1, US NRC, Operations Control Center
56	Preliminary Inquiry Interviews
57	Control Room Operator J
58	Radiation/Chemistry Technician L
59	US NRC Inspector H
60	Meteorological Consultants A and B
61	Control Room Operators C and Auxiliary Operator D
62	US NRC Inspector A
63	US NRC Inspector B
64	US NRC Inspector E

<u>INTERVIEW NO.</u>	<u>POSITION TITLE</u>
65	US NRC Inspector G
66	Engineer F
67	B&W Operations Engineer
68	Radiation/Chemistry Technician M
69	Radiation Protection Consultant-B
70	Engineers A and I
71	Station Manager
72	Supervisor, Generation Security & Site Protection Sergeant A
73	US NRC Inspector F
74	Corporate Environmental Scientist, B
75	Radiation/Chemistry Technicians R and T
76	Corporate Supervisor C
77	Unit 1 Superintendent
78	Radiation/Chemistry Technicians O and U
79	Supervisor, Radiation Protection & Chemistry
80	Group Interview of Training Staff
81	Control Room Operator I
82	Radiation/Chemistry Technician C
83	Unit 2 Superintendent
84	B&W Principle Engineer B
85	B&W Service Engineer
86	B&W Principle Engineer A
87	B&W Manager D
88	B&W Manager A

<u>INTERVIEW NO.</u>	<u>POSITION TITLE</u>
89	Radiation/Chemistry Technicians D and I
90	Vice President of Generation
91	Vice President Generation
92	B&W Senior Engineer
93	Radiation Protection Consultant C
94	Superintendent Administration and Technical Support
95	Shift Supervisor A
96	Radiation/Chemistry Technician P
97	Engineer H
98	Radiation/Chemistry Technician V
99	Radiation/Chemistry Technician A
100	Consultant A
101	US NRC Inspector M
102	Engineer K
103	Daupin County Civil Defense Director and Supervisor, Radiation Protection and Chemistry
104	Auxiliary Operator C
105	Shift Supervisor D
106	Radiation/Chemistry Technician B
107	Shift Foreman A
108	Environmental Scientist A
109	Auxiliary Operator E
110	Unit 1 Superintendent, Technical Support
111	Control Room Operator J

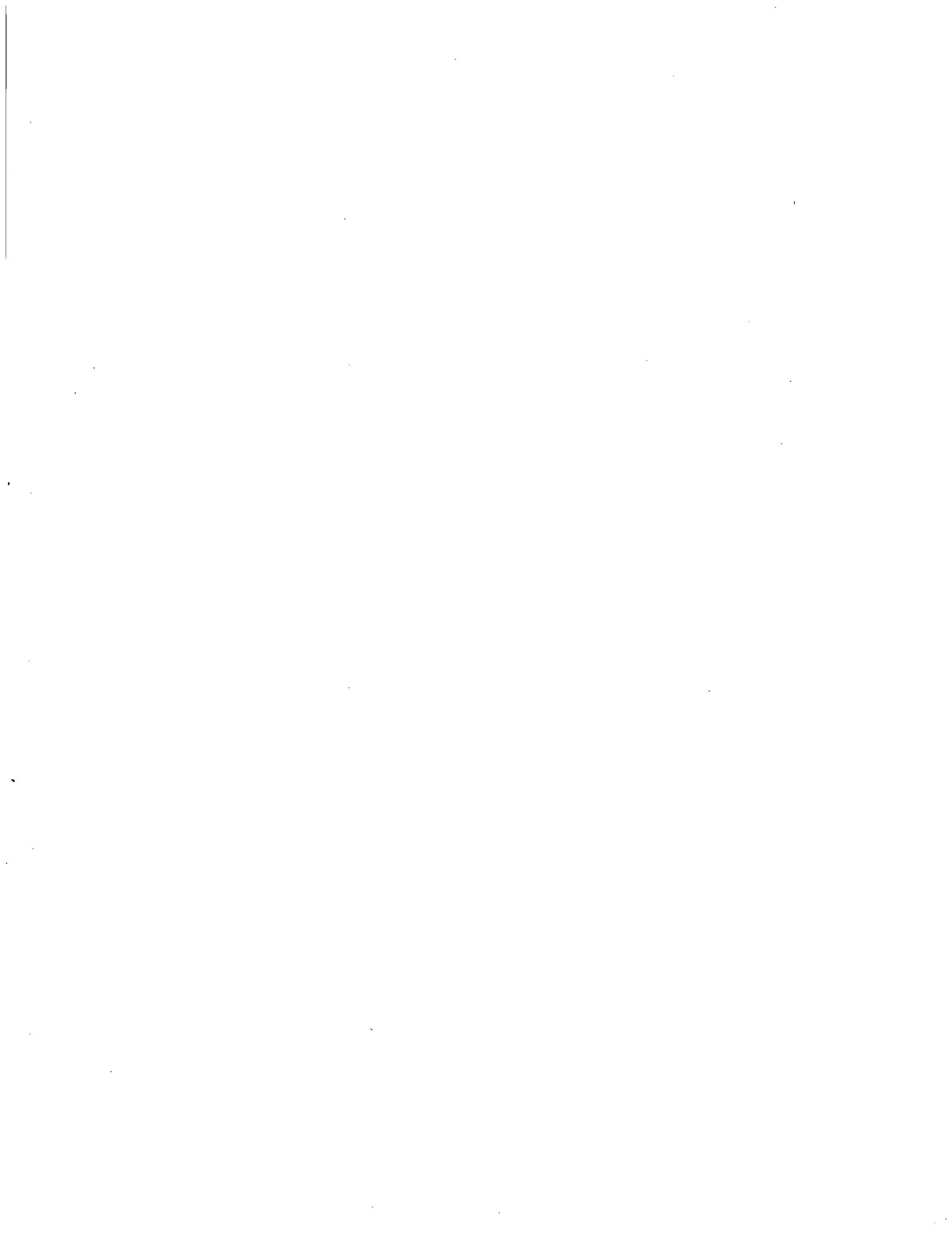
<u>INTERVIEW NO.</u>	<u>POSITION TITLE</u>
112	Maintenance Foreman C
113	Unit Supervisor Station Operations
114	Auxiliary Operator I
115	Auxiliary Operator J
116	Control Room Operator F
117	Control Room Operator A
118	Control Room Operator H
119	Shift Supervisor B
120	Maintenance Superintendent
121	Engineer B
122	Staff Chemist A
123	Auxiliary Operator G
124	Radiation Protection Supervisor
125	Auxiliary Operator H
126	Maintenance Foreman D
127	Engineer C
128	Auxiliary Operator D
129	Unit 2 Superintendent Technical Support
130	Chemistry Foreman A
131	Control Room Operator B
132	Chemistry Foreman C
133	Supervisor, Radiation Protection & Chemistry
134	Control Room Operator E
135	Unit 2 Superintendent Technical Support

<u>INTERVIEW NO.</u>	<u>POSITION TITLE</u>
136	Auxiliary Operator F
137	US NRC Inspector N
138	Maintenance Foreman E
139	Control Room Operator G
140	Shift Supervisor C
141	Chemistry Foreman B
142	Radiation Chemistry Technician Q
143	Maintenance Foreman G
144	Radiation Protection Consultant A
145	Group Interview-Control Room Staff
146	US NRC Inspectors I&J
147	Shift Supervisor E
148	Unit 2 Supervisor, Station Operations
149	B&W Test Coordinator
150	Shift Supervisor D
151	Control Room Operator D
152	Shift Supervisor A
153	Control Room Operator C
154	Radiation Chemistry Technicians Q and V
155	Radiation Protection Consultant D
156	Radiation Chemistry Technicians B and G
157	Unit 2 Supervisor, Station Operations
158	Corporate Supervisor A
159	Manager Generation Engineering
160	Corporate Supervisor B

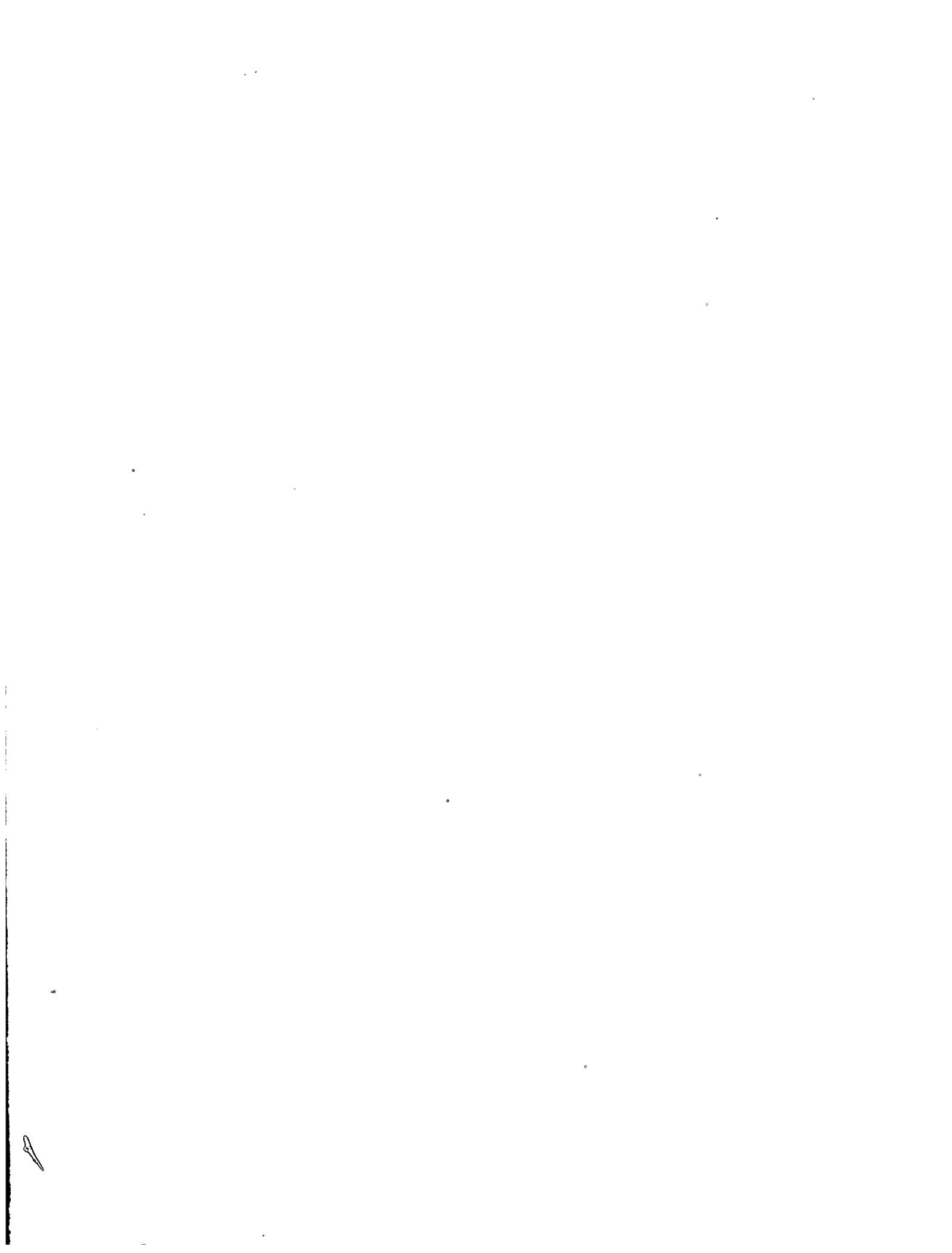
<u>INTERVIEW NO.</u>	<u>POSITION TITLE</u>
161	Corporate Environmental Scientist A
162	GPU Manager G
163	GPU Manager A
164	GPU Director Technical Functions
165	GPU Manager C
166	GPU Senior Engineer B
167	GPU Senior Engineer A
168	GPU Manager B
169	GPU Manager E
170	GPU Engineer B
171	GPU Manager D
172	Radiation Protection Supervisor
173	Shift Supervisor C and Control Room Operator A
174	Engineers C and D
175	Radiation Protection Foreman D
176	GPU Manager A
177	GPU Manager F
178	GPU Engineer A
179	GPU Mechanical Engineer
180	Unit 1 Superintendent Technical Support
181	Instrument Man A
182	US NRC Inspector C
183	Maintenance Foreman A
184	Control Room Operator J
185	B&W Manager C

<u>INTERVIEW NO.</u>	<u>POSITION TITLE</u>
186	Instrument Man B
187	Maintenance Foreman B
188	Auxiliary Operator A
189	Group Interview Control Room Staff
190	Shift Foreman B
191	B&W Test Coordinator
192	Engineer B
193	Unit 2 Maintenance Supervisor
194	Repairman A Repairman B
195	Engineer I
196	Shift Supervisor A
197	Unit 2 Superintendent Technical Support
198	Maintenance Foreman
199	Auxiliary Operator
200	Shift Foreman B
201	Control Room Operator K
202	Control Room Operator L
203	UC and M Man A UC and M Man B

NRC FORM 335 (7-77)		U.S. NUCLEAR REGULATORY COMMISSION BIBLIOGRAPHIC DATA SHEET		1. REPORT NUMBER <i>(Assigned by DDC)</i> NUREG-0600	
4. TITLE AND SUBTITLE <i>(Add Volume No., if appropriate)</i> INVESTIGATION INTO THE MARCH 28, 1979 THREE MILE ISLAND ACCIDENT BY OFFICE OF INSPECTION AND ENFORCEMENT Investigative Report No. 50-320/79-10				2. <i>(Leave blank)</i>	
7. AUTHOR(S)				5. DATE REPORT COMPLETED MONTH YEAR July 1979	
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS <i>(Include Zip Code)</i> Office of Inspection and Enforcement U.S. Nuclear Regulatory Commission Washington, D. C. 20555				- DATE REPORT ISSUED MONTH YEAR July 1979	
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS <i>(Include Zip Code)</i>				6. <i>(Leave blank)</i>	
13. TYPE OF REPORT				8. <i>(Leave blank)</i>	
15. SUPPLEMENTARY NOTES				10. PROJECT/TASK/WORK UNIT NO.	
16. ABSTRACT <i>(200 words or less)</i> <p>On March 28, 1979, the Three Mile Island Unit 2 Nuclear Power Plant experienced the most severe accident in U.S. commercial nuclear power plant operating history. This report sets forth the facts concerning the events of the accident determined as a result of an investigation by the NRC Office of Inspection and Enforcement. The IE investigation, which is based on the information available at this time, is limited to two aspects of the accident:</p> <ol style="list-style-type: none"> 1. Those related operational actions by the licensee during the period from before the initiating event until approximately 8:00 p.m., March 28, when primary coolant flow was re-established by starting a reactor coolant pump, and 2. Those steps taken by the licensee to control the release of radioactive material to the off-site environs, and to implement his emergency plan during the period from the initiation of the event to midnight, March 30. 				11. CONTRACT NO.	
17. KEY WORDS AND DOCUMENT ANALYSIS				14. <i>(Leave blank)</i>	
17b. IDENTIFIERS/OPEN-ENDED TERMS				17a. DESCRIPTORS	
18. AVAILABILITY STATEMENT Unlimited				19. SECURITY CLASS <i>(This report)</i> Unclassified	
20. SECURITY CLASS <i>(This page)</i>				21. NO. OF PAGES	
22. PRICE S					







UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, \$300

POSTAGE AND FEES PAID
U.S. NUCLEAR REGULATORY
COMMISSION

