Met-Ed / GPU

Metropolitan Edison Company
Post Office Box 480
Middletown, Pennsylvania 17057
717 944-4041

Writer's Direct Dial Number

April 15, 1980
TLL 174

Office of Inspection and Enforcement
Attn: B. H. Grier, Director
Region I
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pa. 19406

Dear Sir:

Three Mile Island Nuclear Station, Unit II (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
TMI-II Quarterly Report No. 3

Enclosed please find the sixth followup report, the third quarterly report, which describes recovery related progress since the March 28, 1979 incident occurring at TMI-II. This report documents current progress information for the period January 1 thru March 31, 1980, and is submitted in accordance with section 6.9.1.10 of the TMI-II Recovery Technical Specifications. Also included in this report is the Radiation Safety Program Report as described in section 6.9.1.6 of the TMI-II Recovery Technical Specifications.

Sincerely,

/s/ G. K. Hovey

G. K. Hovey
Director, TMI-II

GKH:LWH:hah

Enclosure: TMI-II Recovery Quarterly Progress Report for the period January 1, 1980 thru March 31, 1980.

cc: Director of Nuclear Reactor Regulation
[redacted]
Light Water Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

J. T. Collins, Program Manager
TMI Program Office
U. S. Nuclear Regulatory Commission
c/o Three Mile Island Nuclear Station
Middletown, Pa. 17057

A001
S E
1/1

8004180338

TMI-II-RR-6

THREE MILE ISLAND UNIT 2
RECOVERY QUARTERLY PROGRESS
REPORT FOR THE PERIOD
ENDING MARCH 31, 1980

THREE MILE ISLAND UNIT 2 RECOVERY REPORT

Met-Ed / GPU

8004180339

TMI-II-RR-6

THREE MILE ISLAND UNIT 2
RECOVERY QUARTERLY PROGRESS
REPORT FOR THE PERIOD
ENDING MARCH 31, 1980

APPROVED BY:
G.R. Hovey
Director
TMI Unit 2

Prepared for the U.S. Nuclear Regulatory Commission, Director Region 1 Office, King of Prussia, Penna. in accordance with paragraph 6.9.1.10 of TMI Unit 2 Technical Specification.

Submitted to NRC Region 1 in April 1980.

METROPOLITAN EDISON COMPANY/
GENERAL PUBLIC UTILITIES
THREE MILE ISLAND
P.O. BOX 480
MIDDLETOWN, PA 17057

TABLE OF CONTENTS

TABLE OF CONTENTS (CONTINUED)

TABLE OF CONTENTS (CONTINUED)

SECTION

3

(continued)

TABLE OF CONTENTS (CONTINUED)

TABLE OF CONTENTS (CONTINUED)

TABLE OF CONTENTS (CONTINUED)

TABLE OF CONTENTS (CONTINUED)

TABLE OF CONTENTS (CONTINUED)

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (CONTINUED)

LIST OF FIGURES

LIST OF TABLES

SECTION 1

INTRODUCTION AND SUMMARY

1.0 GENERAL

This is the third quarterly progress report which documents salient recovery related work performed at the Three Mile Island Unit 2 nuclear facility (TMI-2) during the first quarter of calendar year 1980. Individual section summaries are included within this section under Summary of Current Progress; the more detailed treatise of each of these section summaries is presented within the body of this report.

2.0 SUMMARY OF OBJECTIVES

The TMI-2 quarterly recovery reports shall include available information concerning the cause of the March 28, 1979 incident, probable consequences of the incident, planned (short term and preliminary long term) corrective action and description of continuing activities related to the incident. Quarterly progress reports shall continue until a final report is issued, as previously commited in Metropolitan Edison Company letter GQL 0490 dated April 11, 1979. These reports shall include similar information as described above, as well as interim analysis results and evaluations which have become available. The final report shall also include a summary of Technical Specification Violations which occurred during and after the transient, a summary of the cause(s) of the incident, a sequence of events which occurred during the transient, and corrective actions (both short term and long term) which were taken as a result of the March 28, 1979 incident. Quarterly progress reports shall be prepared and submitted in lieu of REPORTABLE OCCURRENCE REPORTS, MONTHLY OPERATING REPORTS and SPECIAL REPORTS identified in the TMI-2 Technical Specification.

3.0 SUMMARY OF CURRENT PROGRESS

Recovery Engineering - Significant engineering efforts were expended in the design and construction of various liquid and solid radwaste handling systems. Detailed engineering and fabrication of components for the Submerged Demineralizer System (SDS) has continued, and preliminary results of analytical tests performed on actual Reactor Coolant System samples verified the design of that system. A Technical Evaluation Report for the SDS was drafted and reviewed in preparation for submittal to the NRC.

Hydrostatic testing, flushing and preoperational testing commenced during this quarter on the Mini Decay Heat Removal System. Development of design criteria, engineering design drawings and procurement of equipment for the jointly funded GPU/DOE Equipment Decontamination Facility continued.

Other work performed in the Recovery Engineering Department included continued surveillance and analysis of reactor conditions, containment building water level, evaluation of reactor cooling requirements and other analytical work in support of plant operations. Engineering work also continued in the evaluation and development of methods to solidify EPICOR II resins, in accordance with NRC requirements.

Operations and Maintenance - Plant Engineering section provided engineering support to the Reactor Building Purge effort, the Standby Pressure Control System Program, the Emergency Diesel Generator Maintenance program, the Waste Gas Compressor program, the Reactor Building and Steam Generator water level measurements, the Fire Protection program, the Submerged Demineralizer System program, and the Plant Chemistry and Radiochemistry programs. The Process Support group continued supporting the Fuel Pool Waste Storage System, EPICOR II system, the staging facilities for dewatered resins and evaporator bottoms and the Nuclear Sampling System efforts. The decontamination work effort within the Auxiliary and Fuel Handling Buildings continued. Plant maintenance continued on a scheduled and non-scheduled basis.

Radiological Controls - A comprehensive Management Plan for TMI-2 Radiological Control Program has been implemented. The Radiological Control Department was reorganized into five separate groups reporting to a manager that reports directly to a Senior Vice President. The Radiological Assessment Group was formed to independently monitor progress towards implementing and adhering to a strong Radiological Control Program. This group has the authority to stop work involving violations of sound radiological work practice. A comprehensive Radiation Protection Plan for TMI-2 was submitted to the NRC for approval. Radiological Control Department formalized training programs were implemented.

Special Projects - Design input and modifications to the hydrogen control system required to support TMI-2 reactor building purge has been completed. Expected dose rates have been calculated for personnel entering the reactor building and are hereinafter presented. Training of initial entry crew and testing of equipment for initial reactor building was completed. Entry into and radiation surveys were accomplished within the reactor building personnel airlock. Experiments were conducted to determine response characteristics of various gamma and beta/gamma survey instruments and diffusion rates through select wearing apparel subjected to a krypton environment.

Environmental Monitoring - Activities during this quarter centered around upgrading and improving the existing environmental radiation monitoring program. Upgraded detection systems included: infield placement of a new environmental TLD system acquisition of on-site TLD readout detection monitors, and air sampling equipment that will provide for grab sampling, one week continuous, and cryogenic analysis. In addition to improvements in instrumentation, numbers of sampling sites were increased to provide expanded coverage in the environment.

During this quarter, input and comments were provided on the design systems for recovery. Inputs were directed at ensuring compliance with plant operations, technical specifications, and regulatory statutes for purposes of protecting the health and safety of the general public.

Project Operations - The baseline engineering package for an interim waste staging facility has been completed and is currently under review. General arrangement drawings and a material handling study for the TMI-2 reactor building recovery service building have been completed and are currently under review. A supplement to the July 1979 reactor building decontamination report has been issued for review. In addition, an overall radwaste management study, a study on alternate methods for the disposal of tritiated water and preliminary evaluation of radwaste volume reduction techniques has been completed. A contract for the new TMI-2 administration building was awarded on February 27, 1980. Preliminary construction for this facility has been initiated. The ground water monitoring well system surrounding TMI-2 has been completed and is undergoing pre-operational testing.

Quality Assurance - The scope of the Quality Assurance Program for TMI-2 has been expanded to include all items and activities identified by engineering as important to safety. The new important to safety concept will encompass not only the items previsouly designated as safety related but also activities, systems, structures and components which may affect the capability of the unit to adequately protect the health and safety of the public. A separate Quality Assurance Plan is currently in progress which describes the program to be implemented during recovery. During this period the Quality Assurance staff has been reorganized to implement the requirements of the new plan.

Training - During this period, Auxiliary Operator, Reactor Operator (RO), and Senior Reactor Operator (SRO), and RO/SRO requalification training programs were continued. Maintenance training, Health Physics training, radwaste administration training, general employee training, emergency plan training, supervisory indoctrination into controlled substance training, reactor building re-entry team and purge system training and special training programs were continued.

Security - A computerized access control system was established for entrance into protected/vital areas. Security procedures have been completed and are currently in the review and approval cycle. A new badging system has been established and is being implemented.

__Appendix - 1__ - This report provides an Annotated Sequence of Events on the March 28, 1979, accident at Three Mile Island Unit 2 and is the result of a detailed analysis of reactimeter data, plant computer data, plant recorder charts, plant logs and operator interviews. The Report includes a chronology of plant events, the reference source of each entry in the chronology, and the information available to the operator regarding each event in the sequence. The "Information Available to the Operator" entries, addresses the type of information available, the form in which the information was presented, and the timeliness of the presentation of the information to the operator, relative to the time of occurrence of the event.

This report should be considered the final analysis on the Sequence of Events during the TMI-2 accident. Investigation and data analysis are still ongoing and continue to provide new insights. As this new information and/or understanding is developed, amendments or revisions to this document will be submitted as necessary.

SECTION 2

RECOVERY REPORTS AND TECHNICAL SERVICES

1.0 SCOPE

The objectives of this task are to ensure timely reporting of current TMI-2 recovery activities to the U.S. Nuclear Regulatory Commission, Region 1 Office. Technical Services include the coordination, compilation, illustration, review, approval, reproduction and distribution of the TMI-2 quarterly and final recovery progress reports.

Overall report direction and day-to-day administration will be provided under this task. Plana and controls will be established and maintained; periodic reviews will be held with principal contributors and the Commission; related correspondence and reports will be coordinated and day-to-day technical and administrative liason with cognizant recovery team personnel will be provided.

2.0 CURRENT ACTIVITIES

During this period a Technical Services Group was established to process TMI-2 recovery related reports and thereby alleviate this function from the Licensing Group. A more comprehensive reporting format was established. The third quarterly TMI-2 recovery progress report was completed. Project administration and day-to-day liason with cognizant contributors continued.

SECTION 3
RECOVERY ENGINEERING

1.0 SUBTASK A. PROJECT ENGINEERING

1.1 SCOPE

1.1.1 TMI-2 LOW LEVEL LIQUID WASTE PROCESSING SYSTEM

The low level liquid waste processing system for TMI-2 shall consist of the EPICOR I system, which is presently used for both TMI-1 and TMI-2, and will be relocated for exclusive use of TMI-2, to support recovery work.

This task requires the engineering, design and construction of foundations and weather protector for the EPICOR I system equipment, and the routing of system influent and effluent piping within existing plant structures and the yard north of the EPICOR II facility. It is intended that the system be employed to process liquids collected within the TMI-2 Containment Drain Tanks, prior to transfer of these liquids for disposal via the TMI-2 Evaporator Condensate Test Tanks.

1.1.2 SUBMERGED DEMINERALIZER SYSTEM

The Submerged Demineralizer System (SDS) is designed to be installed within the TMI-2 "B" spent fuel pool. The system utilizes the natural shielding capabilities of water to minimize personnel exposure while processing the water contained in the Reactor Building sump (RBS).

Approximately 700,000 gallons of water will be pumped from the RBS via pump WGP-1 through two (2) in-liner filters, to the tank farm. From the tank farm, the water is pumped through a train of three (3) zeolite beds which are designed to remove the majority of cesium 134 and 137, and strontium 89 and 90; the water is then pumped through a cation train and finally through a mixed bed polishing unit, where other trace elements are removed, and then to Monitor Tanks. The system has sampling capabilities and the program includes a full chemistry support program including a geli system and counting facilities. Current plans are to have the system operated by Chem Nuclear personnel under the direction of Met-Ed Operations.

1.1.3 GROUND WATER MONITORING

Develop the capability to monitor the ground water around the TMI-2 Reactor Containment Building.

1.1.4 PROCESSED WATER STORAGE TANKS

Provide two (2), 500,000 gallon capacity tanks, with associated ring foundations for storage of processed water from EPICOR II, SDS systems and future processing systems.

1.1.5 MINI DECAY HEAT REMOVAL SYSTEM (MDHRS)

The Mini Decay Heat Removal System consists of two (2), electric motor driven centrifugal pumps, two (2), shell and tube heat exchangers, piping, valves, and controls required to remove heat from the TMI-2 core. The system is located at the south end of the fuel handling building, at elevation 280'6" (MSL).

Shielding and air flow control and filtration equipment have been provided for system operation, to minimize personnel exposure to a level as low as reasonably achievable.

In addition to decay heat removal service, the system is also capable of providing intertie to the Submerged Demineralization System (reference section 1.1.2) for demineralized cleanup of the reactor coolant system.

1.1.6 EQUIPMENT DECONTAMINATION FACILITY

This jointly funded GPU/DOE project will consist of a facility which will demonstrate advanced equipment decontamination processes, capable of reducing occupational radiation exposures to workers in nuclear power plants. The decontamination equipment will be located in a fully detached, one story building of about 3000 square feet. Electropolish, freon spray, and vibratory media decontamination techniques will be employed.

1.1.7 EVAPORATOR/SOLIDIFICATION FACILITY

The Evaporator/Solidification Facility shall provide for the collection, treatment, storage and disposal of liquid radioactive wastes, generated during the decontamination of TMI-2.

1.1.8 PROCESSED WATER STORAGE & RECYCLE SYSTEM (PWST)

Provide pumps, piping, valves, controls, etc., for the PWST and interconnection with EPICOR II & SDS systems.

1.1.9 LAUNDRY FACILITIES

The scope of this task is to provide the means of laundering large quantities of protective clothing to be used during the recovery. Specific requirements and alternatives are being studied by Bechtel Power Corporation.

1.1.10 EPICOR II LINER SOLIDIFICATION FACILITY

The EPICOR II Liner Solidification Facility shall provide for the treatment, solidification and disposal of spent resins from EPICOR II water processing system.

1.1.11 SOLID WASTE STAGING FACILITY

The staging facility is a structure designed to store radioactive wastes (solidified or dewatered resins) until they can be shipped for burial. The structure will consist of six (6), modules. Each module consists of sixty (60), 84 inch diameter cells, embedded in concrete and capped with 3 feet thick concrete plugs. Each cell has a drain line to a sump which will serve three modules. The sump is designed to collect any leakage from lines installed in the cells, and meets the seismic requirements of USNRC Regulatory Guide 1.143.

1.1.12 INTERIM WASTE STAGING BUILDING

Provide an interim facility, to serve as a central point for staging solid or compacted waste, prior to transportation for disposal.

1.1.13 CONTAINMENT RECOVERY SERVICE BUILDING

This 25,000 square feet, one story building will be built directly adjacent to the TMI-2 containment and accomplish the following:

1. Provide contamination control and airborne particulate control envelope at the containment equipment hatch.
2. Provide for efficient personnel access to containment.
3. Allow passage of large pieces of equipment and bulk radwaste.
4. Provide a waste staging and temporary storage area.
5. Provide a decontamination area for equipment removed from containment.
6. Provide space to handle containment service systems.
7. Allow for maintaining a hot tool crib in vicinity of containment.

1.1.14 PERSONNEL ACCESS FACILITY/COMMAND CENTER

This two story building will be located directly adjacent to the containment recovery service building, to provide about 12,000 square feet of space and accomplish the following:

1. Provide efficient personnel access to the containment during all phases of containment decontamination and restoration.
2. Provide for personnel radiation monitoring and personnel decontamination.
3. Provide the necessary administrative spaces for processing radiation work permits, personnel briefing, and maintenance of records.
4. The Command Center, an integral part of the Personnel Access Facility, will provide a readout location for remotely monitored instruments, and a central location to direct the containment decontamination and recovery.

1.1.15 ADMINISTRATION BUILDING

Provide a two story building to house 350 to 400 staff personnel, document control center and others as required.

1.2 CURRENT ACTIVITIES

1.2.1 TMI-2 LOW LEVEL LIQUID WASTE PROCESSING SYSTEM STATUS

During the period of January through March, 1980, the following work has been accomplished:

1. A design criteria document has been issued to relocate EPICOR I from TMI-1 to TMI-2.
2. Bechtel Power Corporation has been assigned tasks of providing services, an enclosure structure, two (2), 10,000 gallon monitor tanks and pipe routing inside existing structures and the plant yard area, for system influent and effluent piping.
3. A letter to USNRC has been issued to notify the agency of the intent to move the system.

1.2.2 SUBMERGED DEMINERALIZER SYSTEM STATUS

During the period of January through March, 1980, the following activities have transpired:

1. Completion of design and engineering effort by AGNS has progressed to the point where 95% of the drawings have been released for construction.
2. The Cask Support Platform has been delivered to the site, with additional hardware currently 25% to 75% complete in the APCO fabrication shops. All major pieces of hardware are currently on order.
3. Interface, with plant operation groups and plant engineering personnel, has proceeded to the point where draft copies of operating procedures and chemistry/Health Physics procedures are being jointly reviewed. Plans are being formulated at this time to develop a cohesive training program for both the Chem-Nuclear and Met-Ed operators.
4. The first installation procedure (the cask support platform) has been completed and has been signed off by the PORC committee.

5. A schedule for the completion of operating, maintenance, and chemistry/Health Physics procedures has been developed.

6. An estimate of the required man hours and cost to install the system has been developed by Catalytic.

7. Project Engineering meetings, to identify and solve problems and to inform interested parties of on-going progress, were begun in January and currently are being conducted each Tuesday and Thursday. Minutes are published at each meeting and distributed to interested persons as a means of documenting progress.

8. Additional column testing was begun at Oak Ridge National Labs, utilizing the three (3), 1 liter samples of water which were taken from the reactor sump in October, 1979. To date, the water has been centrifuged and passed through a simulated SDS column set-up, and preliminary indication from Oak Ridge indicates acceptable performance, at least in the ability to remove cesium and strontium.

9. Filter loading and filter dewatering tests have been conducted at AGNS with satisfactory results.

10. The study, to determine the effects of gamma radiation on ion exchange resins and activated charcoal, has been completed by R.C. McFarland, Neely Nuclear Research Center, Department of Nuclear Engineering, Georgia Institute of Technology in February 1980, and is presented in Appendix A.

11. Work has been proceeding on the development of a solidification module to solidify the 10 cubic feet vessels. Currently a draft copy of a design criteria document is being reviewed. An assembly drawing has been prepared based on this document.

12. The Technical Evaluation Report for the SDS has been written and is in the final stages of preparation for presentation to the USNRC.

1.2.3 GROUND WATER MONITORING STATUS

Installation of eight (8), wells has been completed. The drillers demobilized on March 13, 1980. Preliminary sampling procedure has been reviewed. Pump tests to develop hydrogeological information commenced.

1.2.4 PROCESSED WATER STORAGE TANKS

1. Tanks PW-T1 and T2: Order placed with Pittsburgh Des-Moines on February, 1980.

2. Foundations PW-T1 and T2. Excavations complete, PW-T2, tank, concrete poured March 24, 1980.

1.2.5 MINI DECAY HEAT REMOVAL SYSTEM STATUS

During this period of January through March 1980, the following work has been accomplished:

1. Design changes have been issued to incorporate a demineralized water flush for MDHRS pump mechanical seals and seal water cyclone separater.

2. Design changes have been issued to incorporate piping and valves that will allow future inter-tie of the MDHRS and the Submerged Demineralizer System.

3. System hydrostatic testing, flushing, and preoperational testing commenced on March 8, 1980, and remains in process.

4. Final MDHRS operating dose, to equipment and instrumentation in proximity to the MDHRS piping and components, has been evaluated. No equipment movement is deemed to be necessary, within the following exceptions:

   a. A pressure indicator, employed for the Standby Pressure Control System, has been relocated to an area outside the shielding volume.

   b. Relocation of accelerometer equipment, used to detect earth/slab movement during a seismic event, will be accomplished to minimize operator exposures during equipment maintenance.

5. An assessment of boron concentrations within the reactor core due to MDHRS startup and operation, commenced in mid-March, 1980, and remains in process.

6. USNRC comments concerning information, a part of the MDHR system design criteria document, and the system description, were received, evaluated and disposition defined for incorporation into the above mentioned documents.

7. Revision 7, to the System design Criteria Document was issued. Final MDHRS, ALARA dose estimate were accomplished.

8. The PORC has initiated review and approval of the SOP for opening valves DHV-1 or DHV-171, the MDHRS operating procedure and maintenance requests for hydrostatic testing between DHV-3 and DHV-1 and 171.

9. Plant Engineering has begun, and is in the process of developing alarm response procedures and emergency procedures in support of system operation.

1.2.6 EQUIPMENT DECONTAMINATION FACILITY STATUS

1. Development of general arrangement of equipment.
2. Procurement of equipment.
3. Engineering design of systems.
4. Determination of building type and design.
5. Development of design criteria.

1.2.7 EVAPORATOR/SOLIDIFICATION FACILITY STATUS

1. Engineering in process.
2. Intermediate issues of layouts and flow diagrams reviewed.
3. Technical Evaluation Report preparation in progress.
4. Solidification system selected.

1.2.8 PROCESSED WATER STORAGE & RECYCLE SYSTEM (PWST) STATUS

P & ID for review and comments, ongoing.

1.2.9 LAUNDRY FACILITIES STATUS

A letter from Mr. R.F. Wilson, Director of TMI-II Recovery, dated March 4, 1980, authorized Bechtel Power Corporation to proceed in performing the studies necessary to define anti-contamination clothing, and laundry requirements during the TMI-II containment recovery.

1.2.10 EPICOR II LINE SOLIDIFICATION FACILITY STATUS

1. Studies completed for inliner and exliner solidification concepts, TDR's prepared for each.
2. Proposals for inliner and exliner concepts reviewed and evaluated.
3. Test program being developed with Hittman Nuclear Development Corporation, to ensure that EPICOR II resins can be solidified (inliner) with cement. Testa will use both lab samples (500 ml) and drum samples (300r 55 gallons).

1.2.11 SOLID WASTE STAGING FACILITY STATUS

Module A:

Coating applications completed.
Electrical and Piping work completed.
Gaskets installed.
Liners stored in 27 cells of A module.

Module B:

Construction commenced.
Base Mat poured with drain lines installed to sump.
Forms and wall steel for perimeter walls 60% complete.

Module C&D:

Bid packages in house for review and comment.

1.2.12 INTERIM WASTE STAGING BUILDING STATUS

Design criteria being developed.

1.2.13 CONTAINMENT RECOVERY SERVICE BUILDING STATUS

1. Development of general arrangement.
2. Preparation of design criteria.

3. Engineering design in progress.

1.2.14 PERSONNEL ACCESS FACILITY/COMMAND CENTER STATUS

1. Development of general arrangement.
2. Preparation of design criteria.
3. Engineering design in progress.

1.2.15 ADMINISTRATION BUILDING STATUS

1. A contractor has been selected for the entire construction engineering and construction effort, based on a competitive fixed price bidding.
2. Construction commenced (site cleaning and preparation for foundations).
3. Engineering is proceeding on building services.

2.0 SUBTASK B. SUPPORT ENGINEERING

2.1 SCOPE

2.1.1 PRESSURIZER (RCS) SAMPLE TIE-IN TO UNIT II TEMPORARY SAMPLE SINK

To verify the accuracy of the Reactor Coolant pump seal cavity pressure instrument.

2.1.2 OPENING DH-V1/DH-V-171 VALVES

To evaluate and write a procedure on the impact of operations with DH-V1 or DH-V171 and DHV2 being open.

2.1.3 REACTOR BUILDING WATER LEVEL

To monitor the water level in the Reactor Building on a continuous basis to determine if any gross changes of water inflow have occurred.

2.1.4 REACTOR COOLANT SYSTEM CHEMISTRY

To maintain continuous surveillance of the reactor coolant system chemistry parameters.

2.1.5 REACTOR COOLANT SYSTEM

To present a status of the reactor coolant system during the quarter.

2.1.6 REACTOR COOLANT SYSTEM BORON CONCENTRATION

To ensure boron concentration in the reactor coolant system remains above 3000 ppm.

2.1.7 SOLIDIFICATION

To present a status of the efforts expended to resolve problems of resin solidification.

2.1.8 DECREASING REACTOR COOLANT SYSTEM PRESSURE TO 100 PSIG

To provide guidance and support for lowering the reactor coolant system pressure.

2.2 CURRENT ACTIVITIES

2.2.1 PRESSURIZER (RCS) SAMPLE TIE-IN TO UNIT II TEMPORARY SAMPLE SINK STATUS

A memo was issued listing necessary action, responsible parties and time periods involved. A test was arranged to verify the accuracy of the reactor coolant pump, and seal cavity pressure instrument, which was to have been the means of Reactor Coolant System pressure indication. (The instrument failed on January 26, 1980). The results of the test were satisfactory but proved to be irrelevant.

Engineering support was provided for a construction procedure and operational procedure. The tie-in was completed on February 21, 1980.

2.2.2 OPENING DH-V1/DH-V171 VALVES STATUS

A procedure was drafted for this operation and forwarded to PORC. PORC comments on the first review are incorporated and the procedure resubmitted.

A work order was initiated to have the operator for DH-V1/DH-V171 rewired for step or staccato operation.

An evaluation of the impact of operations with DH-V1/DH-V171 and DH-V2 being open has been performed. DH-V1 or DH-V171 must be opened to operate the Mini Decay Heat Removal System.

2.2.3 REACTOR BUILDING WATER LEVEL STATUS

A direct measurement system using a manometer through penetration 401 was assembled and is being used to establish an accurate water level reading method with respect to penetration.

Statistical methods are being used to establish inflow rates for comparison with inflow leakage.

2.2.4 REACTOR COOLANT SYSTEM (RCS) CHEMISTRY STATUS

During this quarter, results of the sample analysis were recorded and graphed in order to provide long term trends. The graphs included values of boron, oxygen, hydrogen, nitrogen, chlorides, sodium, total gas, pH, tritium, strontium, and cesium.

Input was provided to answer question regarding:

1. The need for an alternate RCS buffer solution for pH control.
2. Increased RCS boron concentrations.
3. Erratic dissolved gas concentrations.

A review of alternate RCS pH control mechanisms (other than NaOH), was conducted due to processing problems encountered with ion exchange of high sodium fluids.

The reasons for a continually increasing RCS boron concentration were determined through a review of boric acid addition procedures, boron analysis accuracy, and control room log books.

Erratic dissolved gas concentrations were viewed in light of using the two different RCS pressure control systems and the possibility of sample air contamination.

A complete record of "Daily Plant System Sheets" and graphs of hourly values of critical RCS temperature have been, and continue to be, maintained.

2.2.5 REACTOR COOLANT SYSTEM STATUS

At noon on February 11, 1980, a compression type fitting connected to the discharge piping of Make-up pump 1B, failed. The resultant leakage of reactor coolant forced the reactor operator to secure reactor coolant makeup (seal injection) and let down. Loss of approximately 11 gpm make-up and let-down flows had a two-fold effect on the natural circulation cooling of the reactor core.

Approximately 200,000 BTU/hr., of direct cooling is provided by this flow, and the net migration of seal injection from the "B" loop reactor coolant pumps, over to the low point of an "A" loop cold leg, normally adds stability to natural circulation.

To compensate for the loss of MU/let-down flow, the reactor coolant temperature increased slightly (the largest temperature increase was 4° - 6°F) and more heat was rejected through the "A" steam generator to condenser. This latter change took the form of complex cyclic flows in the "A" loop, characterized by a regular set of temperature patterns in an 18 hour period. A decision to reactivate the seal injection mode was made in late February. On the third of March the 1B make-up pump was started and approximately 13 gpm of balanced reactor coolant pump seal injection and let down flow was re-instituted. A return to the reactor coolant patterns, observed prior to securing the pump in February, occurred. On March 20, seal injection and let down were once again secured. An identical pattern, seen on February 11th and 12th was repeated.

2.2.6 REACTOR COOLANT SYSTEM BORON CONCENTRATION STATUS

In accordance with the Technical Specifications, the Reactor Coolant System boron concentration must remain above 3000 ppm.

When the Mini Decay Heat System is started, a quantity of water with boron concentration of 2250 ppm will be injected into the Reactor Vessel. The question arose as to whether or not this would dilute the boron concentration below 3000 ppm in the Reactor Vessel.

Recovery Engineering analyzed the problem by calculating contentration changes over-time. The hydraulic dynamics of the entire system are complex; Recovery Engineering defined the specific design of the fluid system involved as a starting point for more advanced analysis that are currently being performed by GPU/Parsippany.

2.2.7 SOLIDIFICATION STATUS

In accordance with the directives of the USNRC, all waste generated during the TMI-II clean up will be solidified.

An extensive literature search on the current state of the art of radioactive waste solidification involving both spent organic ion exchange resins, and concentrated evaporator bottoms, did not produce satisfactory results. Information tended to indicate that most of the work done was unique and to a specific system. Results from various sources tended to conflict with one another and were incomplete.

A trip was undertaken to the Brookhaven National Laboratory in March allowing extensive discussions. We discovered that the answers to most of our questions do not yet exist in a satisfactory form. For example:

1. How to successfully solidify resins, etc.?
2. What are the failure mechanisms and corrective actions?
3. What methods are to be employed?
4. What quality assurance will be used?

A subsequent decision was made to proceed with demonstrator experiments as necessary to determine whether or not we can successfully and expediently use cement as the solidification agent for organic ion exchange resins, particularly those used in EPICOR system.

2.2.8 DECREASING REACTOR COOLANT SYSTEM PRESSURE TO 100 PSIG STATUS

Operational guidance was provided to Plant Operations for lowering RCS pressure. This involved listing prerequisite procedural outlines and alternatives to pressure reduction.

Continuing support is being provided for drafting a procedure. The procedure is completed and is being reviewed by the USNRC.

3.0 SUBTASK C. TECHNICAL PLANNING

3.1 SCOPE

To provide technical planning and support.

3.2 CURRENT ACTIVITIES

1. Reviewed the Phase I study and requested clarification of the recommendations regarding containment water level and return water chemistry.

2. Reviewed the Phase II study, made comments and coordinated GPU comments.

3. Continued the development of detail Flow Charts for Technical Planning.

4. Completed studies on tritiated water and submitted a near term plan to the USNRC.

5. Published TDR #137 "Water Quality" specifications for discharge of TMI-II waste water:

A. EPICOR II Solidification:

(1) Published TDR #122 - EPICOR II Resin Solidification Conceptual Design for Ex-liner Solidification. Wrote TDR #146 - EPICOR II Resin Solidification Conceptual Design for In-liner Solidification. Originated the Bechtel person-rem assessment. Solicited proposals for solidification demonstration.

(2) Closed out the feasibility of separating the evaporator solidification system from the evaporator facility as being non-feasible.

(3) Commenced a study of the total solidification projections.

B. Waste Management:

(1) Continued development of computer software to support solid waste management. Received Bechtel draft Waste Management study and initiated a Structural Waste Management Planning Document.

(2) Provided advice to DOE-TIO on waste management activities that are of generic value and may be of industry-wide interest.

4.0 SUBTASK D. BURNS & ROE ENGINEERING

4.1 SCOPE

To provide continuing support for all aspects of Recovery Engineering.

4.2 CURRENT ACTIVITIES

1. Prepared engineering change packages for the EPICOR II system to:

   A. Improve system venting.

   B. Allow for additional storage of processed water.

   C. Add instrumentation for improved system control.

   D. Improve access for system operation.

2. Continued engineering support for plant maintenance.

3. Prepared the following engineering change packages for the Mini-Decay Heat Removal System:

   A. Installation of radiation, television, and pump vibration monitoring instrumentation.

   B. Installation of pump oil supply tubing.

   C. Performed the finalizing of the system description.

   D. Supported studies concerning radiation level effects when the system is in operation.

4. Prepared engineering change packages to provide fire protection for recovery facilities. Performed initial engineering for a TMI-2 fire door alarm system.

5. Provided electrical engineering support for disconnecting the BOP diesel generators.

6. Performed engineering to develop the electrical power and distribution for the Submerged Demineralization System.

7. Provided general engineering and administrative support for recovery and plant engineering, by providing TMI-2 design information and engineering interface support to other organizations.

5.0 APPENDIX A - THE EFFECTS OF GAMMA RADIATION ON ION EXCHANGE RESINS AND ACTIVATED CHARCOAL

5.1 STATEMENT OF THE PROBLEM

Solid absorbents and ion exchange materials, used in the decontamination of high level liquid radioactive waste, will receive large radiation doses from the radioactive material which they remove from the waste stream. Radiation damage to the adsorbent material is responsible for the observed effects of loss of exchange capacity and chemical decomposition of the base material.

Since the actual decontamination process is relatively rapid, the loss of exchange capacity due to radiation damage is a minor consideration in the decontamination of liquid rad waste. On the other hand, chemical decomposition of the base material can be a major concern when the highly radioactive adsorbent materials are buried in sealed containers. The possibility that radiation decomposition of adsorbent materials could produce gases, in sufficient quantity to cause over pressurization of burial containers, led to this investigation. Specifically, the goals of this project were to determine the pressure buildup and gas composition as a function of gamma dose in burial canisters of the type being considered for use at Three Mile Island. The purpose of the first part of this investigation was to measure the pressure versus gamma radiation dose, in separate simulated burial containers holding organic cation resin, organic anion resin, and activated charcoal.

The first task in this investigation was to estimate the radiation dose to an absorbent material loaded with TMI high level radioactive waste and sealed in a burial container. The calculation of the estimated radiation dose is given in Appendix B. Once the magnitude of the total dose was determined, the gamma irradiator design was finalized. The details of the Co-60 irradiator and irradiation capsules are given in section 5.2. Finally, the simulated burial containers were irradiated in the Co-60 gamma irradiator in a 5 X 10^6 rad/hr., field until the total accumulated dose reached 5 X 10^9 rads or the capsule reached the pressure limit of 200 psig. Pressure versus gamma dose curves are given in section 5.3.

The second part of this investigation was to determine the gas composition inside the resin irradiation capsules at several different dose levels. The gas composition was determined using gas chromotography and the methods and results are discussed in Section 5.4.

5.2 IRRADIATION CAPSULE CONSTRUCTION AND IRRADIATOR DESIGN

In order to meet the time schedule of this project, it was decided that the pressure test and gas chromotograph test capsules should be constructed from commercially available fittings to the maximum extent possible. The construction material chosen was stainless steel in order to match the proposed burial containers as closely as possible. While stainless steel might influence the chemical reactions and be undesirable from a purely scientific point of view, it was a good compromise considering radiation resistance, pressure limitations, chemical reactivity, and the desire to simulate the actual burial containers. In addition, it was required that no organic materials, other than those being tested, be used in any of the units. At the high dose levels involved, organic O rings and bellows could breakdown and contaminate the gases or release the gaseous products. All stainless steel valves and pressure gauges were used in the construction of these containers.

Figure 3-1 shows the pressure test capsules. Specifications for the fittings and tubing are given in figure 3-5. The pressure test capsules required an internal stainless steel spacer plug (item 23) to raise the material into the radiation field and to provide the correct material to void ratio. From data supplied by Chem Nuclear Systems it was estimated that the burial containers would have a resin to total volume ratio of 0.875. The pressure test capsule shown in figure 3-1 has a resin to total volume ratio of 0.866. In order to get this ratio as close to the actual ratio as possible, the pressure gauges had to have a small internal volume and had to be connected to the capsule with a minimum 1/8" stainless steel tubing. To keep the tubing connections short, the gauges were located on a rack on the back wall of the hot cell about 5 feet from the capsules. The gauges were outside the most intense radiation field but the glass faces still had to be removed because of radiation darkening. The gauges were Ashcroft 0-200 psi gauges which were calibrated using a dead weight gauge tester. At the inside top of the pressure test capsule there was a stainless steel screen (10 micron openings) to prevent particles from entering the tubing leading to the gauge. Figure 3-2 shows the details of the top of the pressure capsule and the spacer plug.

To determine the gas composition as a function of the gas chromotograph, capsules shown in figure 3-3 were constructed. As with the pressure test capsules, these were also all stainless steel construction. The resin was weighed into these capsules which had an internal volume of 18.9 cc. The void volume was calculated assuming that the wet resin had a density of 1.1 gm/cc. After irradiation, the samples were removed for analysis using the gas chromotograph sample assembly shown in figure 3-4. The sample assembly was screwed onto the gas chromotograph capsule and evacuated with the capsule valve closed. After evacuation the sample assembly valve was closed and the capsule valve was opened. Samples could then be taken from the septum side arm for analysis by gas chromotography.

The Co-60 irradiator shown in figure 3-6 was assembled to irradiate the test capsules in the Georgia Tech hot cell. The Co-60 was in the form of four plates each containing 6.25 K Ci of Co-60 arranged as shown in figure 3-6. The outside spaces were 1 1/4" wide and the inside space was 1 1/2". The outside spaces held the pressure test capsules and the long term gas chromotograph capsules. Thermocouples were attached to the pressure capsules, the long term GC capsules and several other places on the irradiator. Compressed air was piped into the hot cell and used for cooling the capsules. Before starting irradiation, dose rates were measured at several positions inside the irradiation assembly. The dose rates were measured using Harshaw TL-800 lithium borate thermoluminescent dosimeter. These thermoluminescent dosimeters were calibrated against a Farmer dosimeter, model 2502/3, which had been calibrated using NBS Co-60 at M.D. Anderson Hospital, Houston, Texas. In the outside apace the dose rates were: at the top 3" above center, 4.46 X 10^6 rads/hr.; center, 5.87 X 10^6 rada/hr and 3" above bottom, 4.86 X 10^6 rads/hr. This gives an average of 5.0 X 10^6 rads/hr. The center apace for the G.C. capsules had dose rates of 4.3 X 10^6 rads/hr., at the top; 5.4 X 10^6 rada/hr., at the center, and 4.60 rada/hr., at the bottom. The average dose rate in the GC sample rack was 4.8 X 10^6 rads/hr.

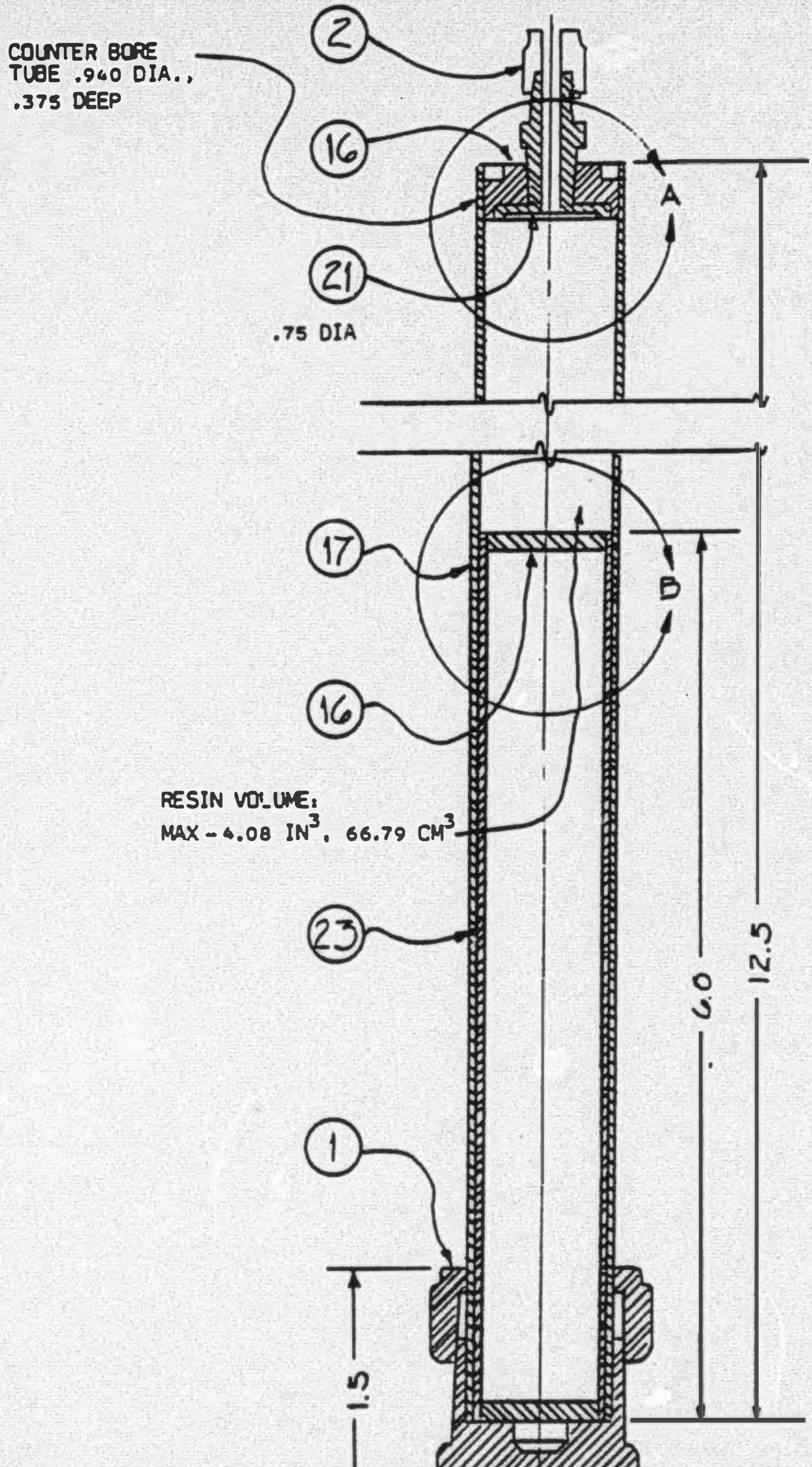


FIGURE 3-1 Section thru Resin Pressure Capsule

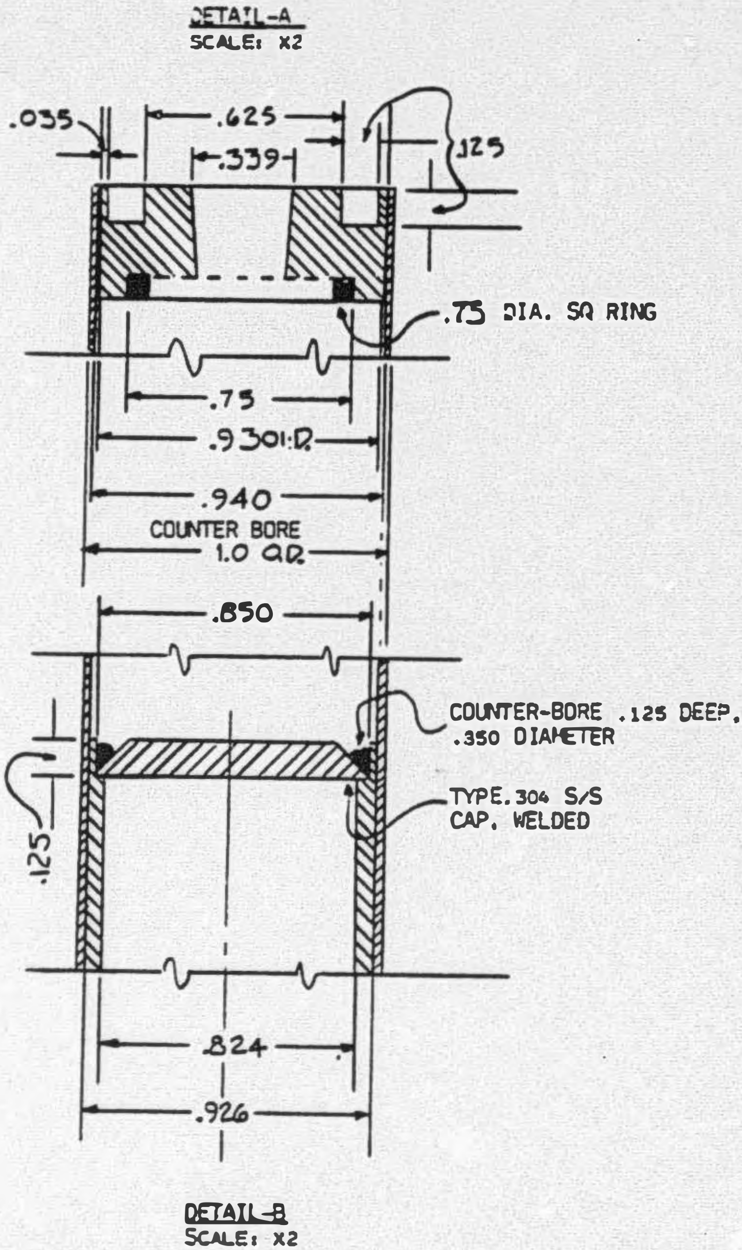


FIGURE 3-2 Detail "A" Enlarged

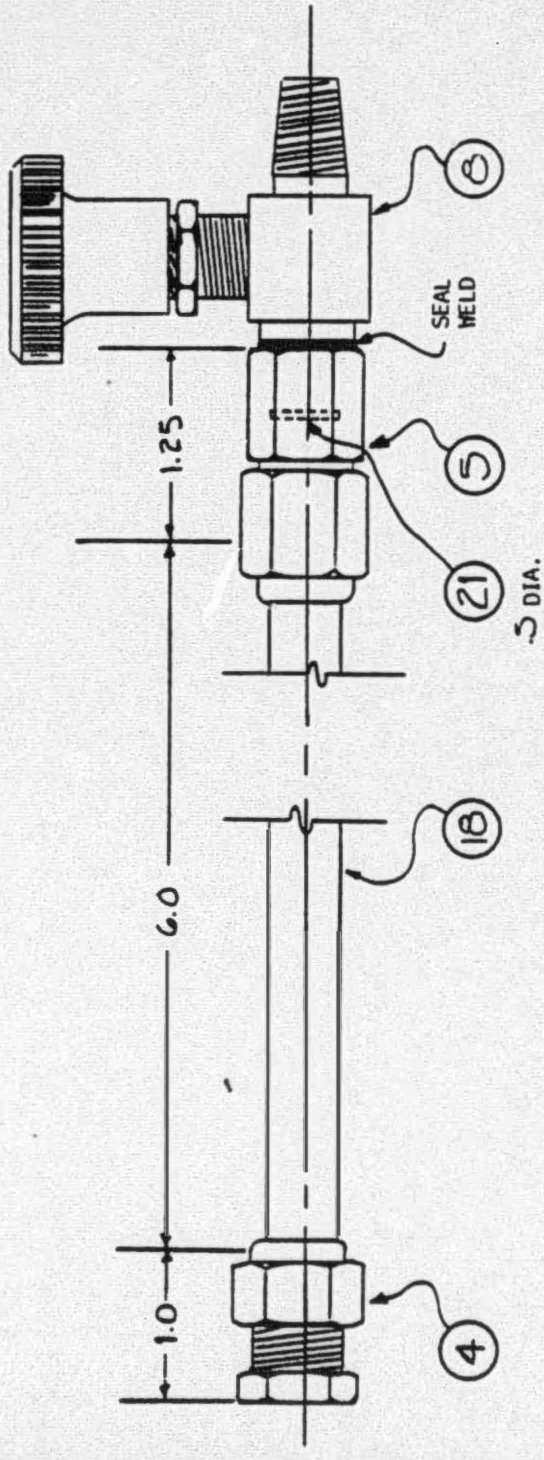


FIGURE 3-3 Gas Chromatograph Capsule

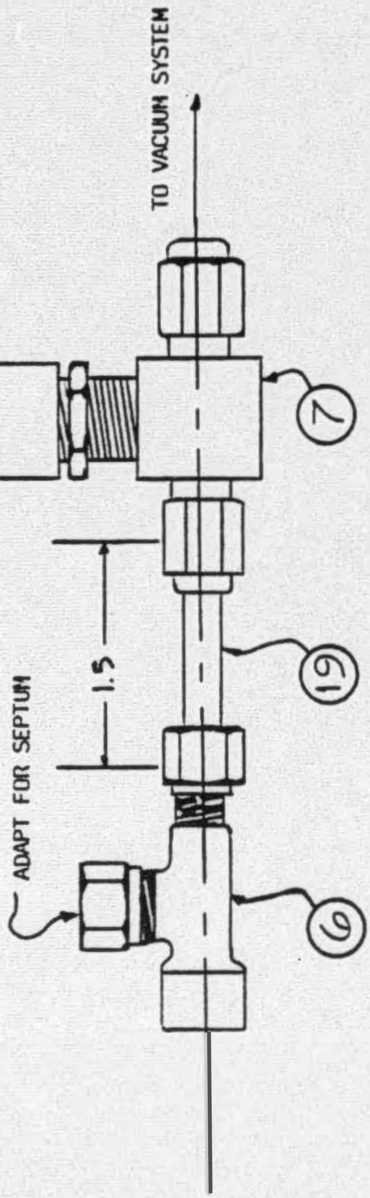


FIGURE 3-4 Gas Chromatograph Sample Assembly

ITEM	QUAN.	CAT PAGE	DESCRIPTION	SWAGELOK CODE
1	8 EA.	20	CAP-1"	SS-1610-C
2	8	7	MALE CONNECTOR 1/8 - 1/4	SS-200-1-2
3	8	10	FEMALE CONNECTOR 1/8 - 1/4	SS-200-7-4
4	10	20	CAP-1/2"	SS-810-C
5	10	10	FEMALE CONNECTOR 1/2T - 1/4P	SS-810-7-4
6	10	12	FEMALE RUN TEE 1/4"	SS-400-3-4TFT
7	9	NUPRO 7	H SERIES BELOWS VALVE	SS-4H
8	9	NUPRO 7	H SERIES BELOWS VALVE	SS-4H2
16	4 FT.		1" D. ROUND S/S ROD	
17	9 FT.		1" O.D. S/S TUBE 0.035 WALL SMLS	
18	6 FT.		1/2" O.D. S/S TUBE 0.035 WALL SMLS	
19	2 FT.		1/4" O.D. S/S TUBE 0.035 WALL SMLS	
20	40 FT.		1/8" O.D. S/S TUBE 0.032 WALL SMLS	
21	6 EA.		2" SQUARE, 10 MICRON SCREEN S/S	
22	7 EA.		2-1/2" D. PRESS. GAGE 0-200 PSI, S/S	
23	4 FT.		3/4" S/S PIPE, SCH. 40, TYP. 304	

FIGURE 3-5 Bill of Materials

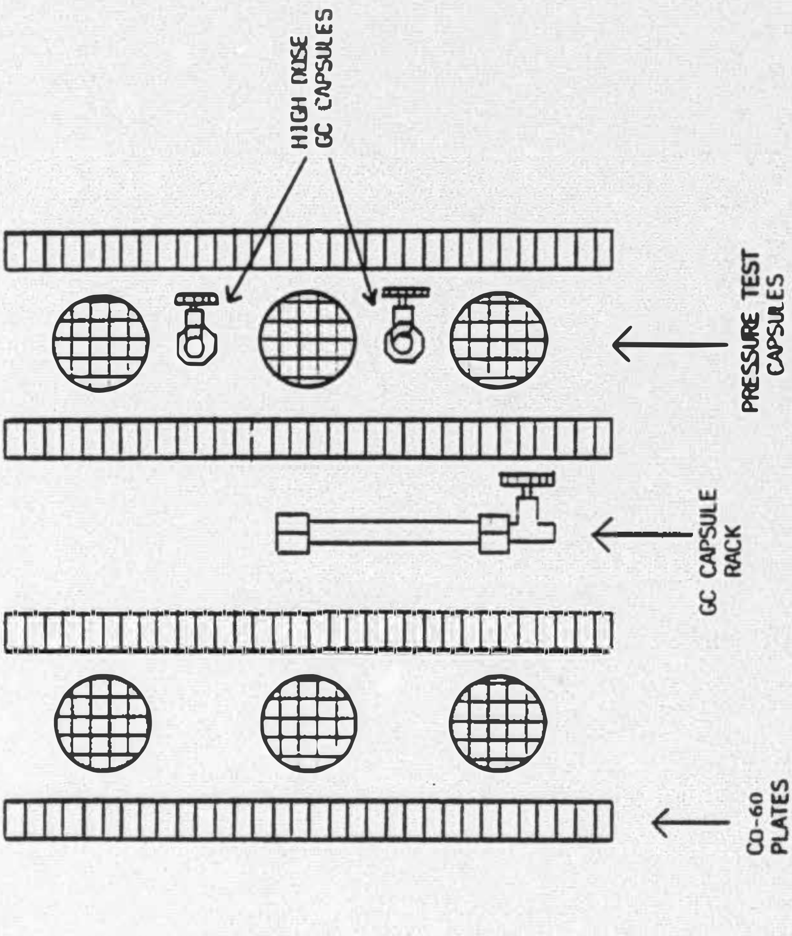


FIGURE 3-6 Co-60 Irradiator

5.3 CAPSULE PRESSURE AS A FUNCTION OF GAMMA DOSE

To simulate the decontamination of TMI liquid radwaste, samples of cation resin (DOW HCR-S), anion resin (DOW SBR-OH) and activated charcoal supplied by Chem Nuclear Systems, were converted to the sodium and borate dorms prior to irradiation. Sodium borate solution was passed through the resin samples until the pH of the effluent from the column was identical to the pH of the original solution. In the case of the activated charcoal, no simple indication of exhaustion could be found. To pretreat the activated charcoal, an amount of sodium borate solution, equal to the amount needed to convert the anion resin to the borate form, was passed through the charcoal sample.

After pretreatment with sodium borate solution, free liquid was removed by pulling air through the column for about 2 minutes. The adsorbent material was then transfered to the pressure test and GC irradiation capsules using a tap fill procedure. Adsorbent loading in the irradiation capsules matched the anticipated loading of the proposed TMI clean up canisters to within 10%.

The filled capsules were placed in the Co-60 irradiator as shown in figure 3-6 and irradiated at 5 X 10^6 rads/hr., in the Georgia Tech hot cell. The capsule pressure was monitored visually using a monocular sighting through the hot cell window in order not to disturb the irradiation. The capsule temperature was determined using thermocouples with the reader located outside the hot cell. The temperature stabilized within a few hours of the beginning of the irradiation and remained between 30°C and 45°C throughout the test. The pressure versus gamma ray dose curves for the anion resin, cation resin, and activated charcoal are presented in figures 3-7, 3-8, and 3-9, respectively. The pressure tests on the resin samples were terminated when the pressure reached the limit of the pressure gauge which was 200 psig. Duplicate pressure test capsules containing each of the adsorbent materials were prepared and irradiated. One of the anion pressure test capsules developed an interval leak in the spacer plug during the irradiation, and the results from that capsule were not reported. The close agreement between the duplicate cation resin tests indicate the reproducibility of the tests.

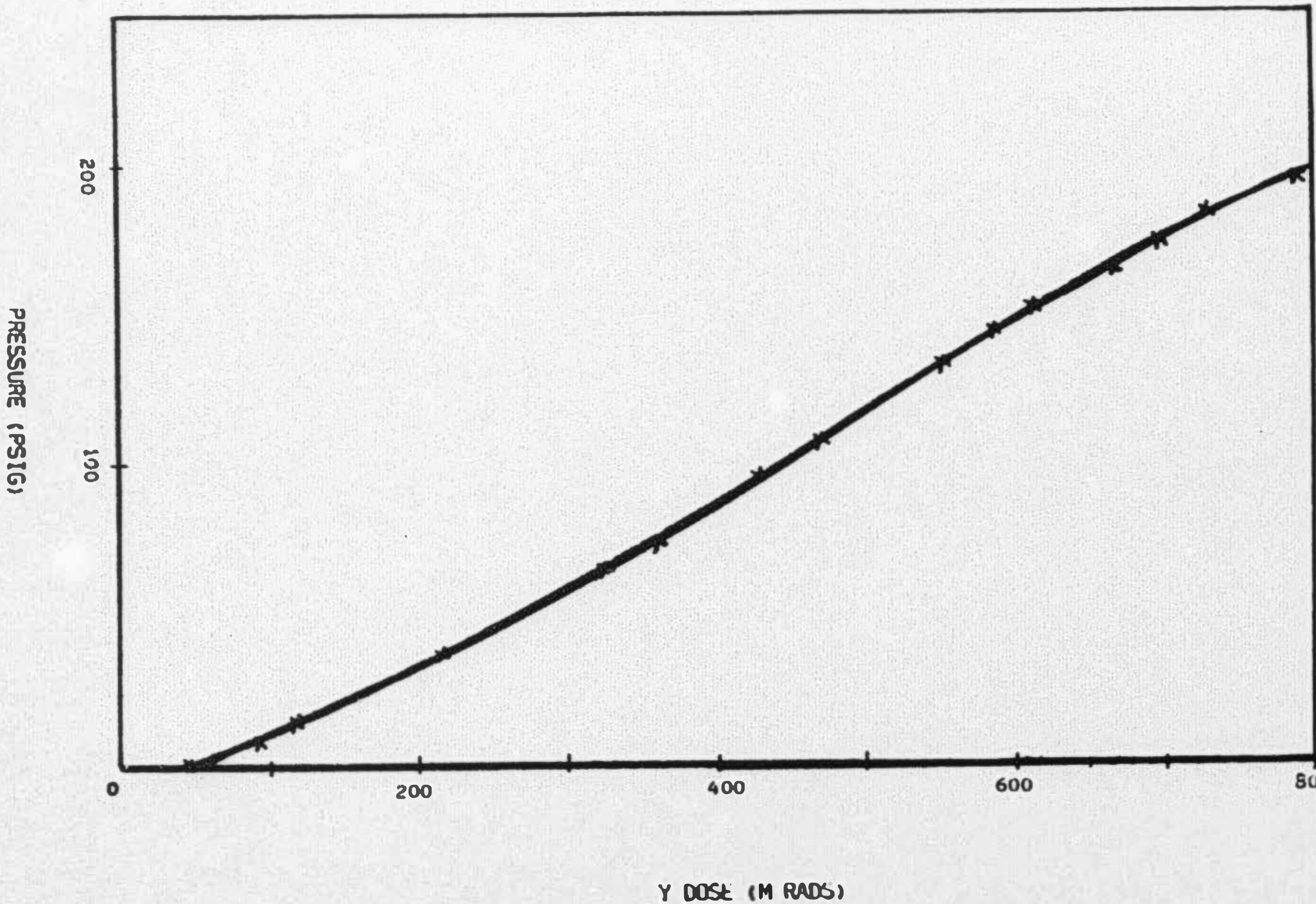


FIGURE 3-7 Anion Resin

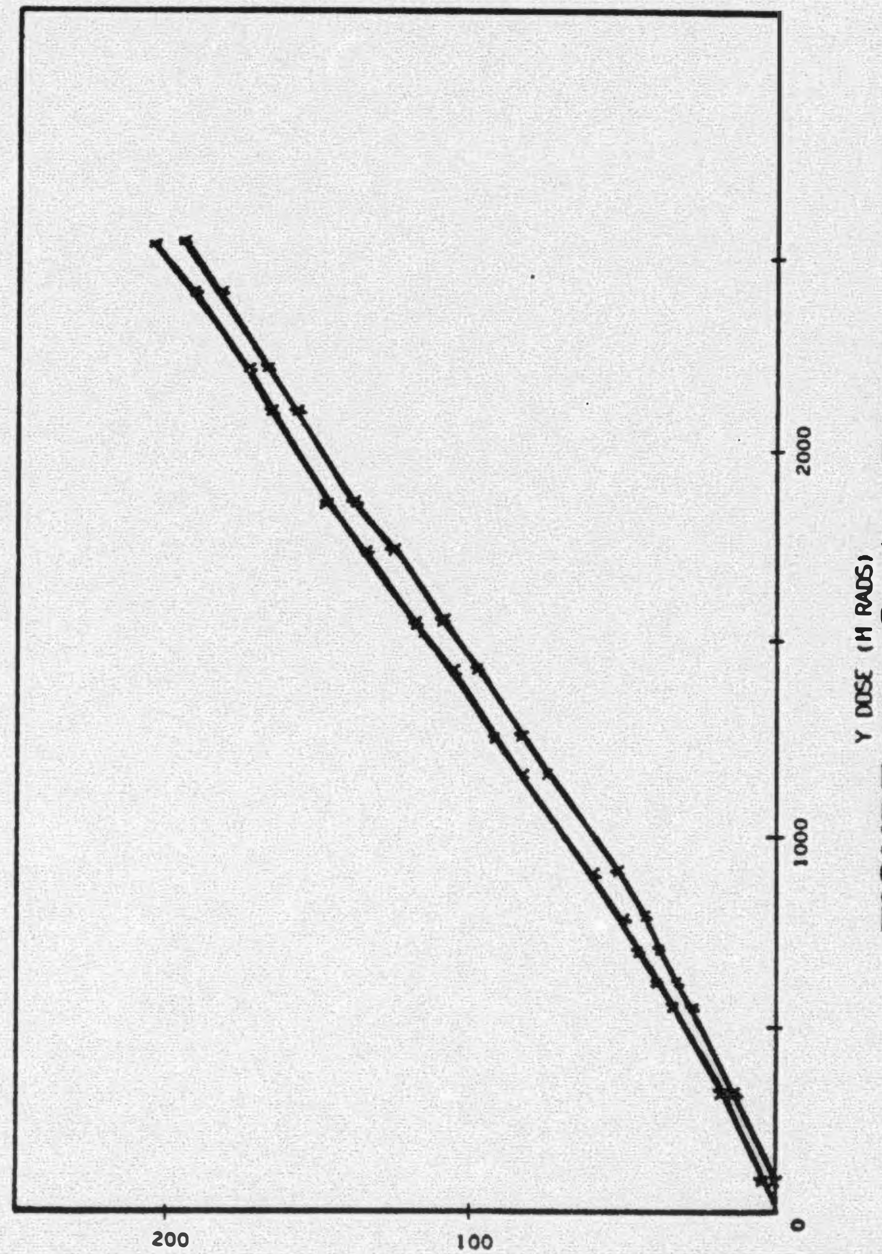


FIGURE 3-8 Cation Resin

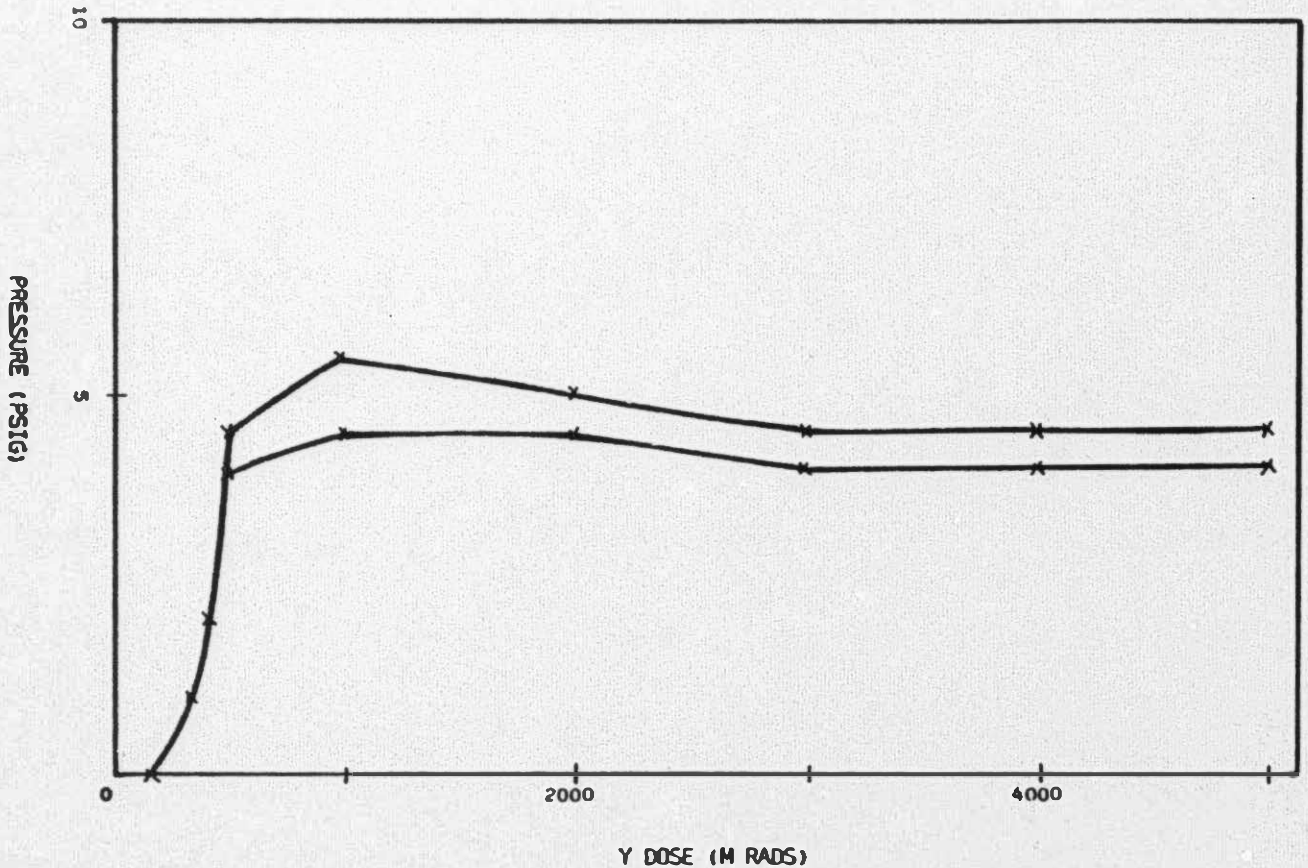


FIGURE 3-9 Charcoal

5.4 GAS COMPOSITION AS A FUNCTION OF GAMMA DOSE

The second part of this study was to determine the composition of the gas inside simulated resin burial containers at several different dose levels. Only the ion exchange resins HCR-S and SBR-OH were tested since it was anticipated that no significant pressure would develop inside the charcoal containers.

The gas chromotograph irradiation test capsules (figure 3-2) were filled with pretreated ion exchange resin using the same procedure as was used for the pressure test capsules. The loaded gas chromotograph sample capsules were sealed and placed in the Co-60 irradiation facility (figure 3-6) for the prescribed amount of time. Separate irradiations were performed to obtain each point since the sampling operation disturbed the resin to void volume ratio. After the irradiation the gas chromotograph sample assembly (figure (3-3) was screwed onto the GC sample capsule valve. With the capsule valve closed, the sample assembly was evacuated, the valve leading to the vacuum pump was closed, and the sample capsule valve was opened to allow the radiolysis gases to enter the sample assembly. Samples for gas chromotographic analysis were taken from the septum side arm of the sample assembly using gas tight syringes equiped with pressure lock valves. Only the gas phase of each sample was analyzed. In the high dose samples, there was a large liquid phase which approached one half of the total volume of the original resin sample. The total volume of the capsule and the interior of the valve was 18.9 cc. The free volume of the volume of the resin loaded capsule was estimated by assuming the wet resin had a specific gravity of 1.1, calculating the volume of the wet resin, and subtracting from the total. The pressure inside the capsule plus sample assembly was calculated using the pressure versus dose curve to obtain the pressure inside the capsule and allowing for expansion into the 3.1 cc sample assembly. Since the volume of sample taken for analysis could be determined from the syringe, and the pressure was calculated as above, the volume of sample could be corrected to standard pressure and quantitiative analysis could be accomplished using comparison standards.

The samples were analyzed for hydrocarbon gases using a Tracor 220 gas chromatograph with a flame ionzation detector. For the hydrocarbon analyses the column was 1/4" x 6" stainless steel packed with a chemically bonded support n-octane/porasil C. 100/120 mesh. Separations were performed isothermally with the column at 36°C and the inlet at 90°C. The carrier gas was nitrogen at a flow rate of 23cc/min. The chromotgraph was calibrated using mixed hydrocarbon gas standards obtained from Supelco Inc.

The major hydrocarbon gases detected in both the cation and anion capsules were the straight chain, saturated hydrocarbons methane, ethane, and propane. Small amounts of butane were detected in the cation capsules. Very small traces of an unidentified (possibly branched chain and/or unsaturated) hydrocarbon compound were detected in the high dose anion capsule. The results of the hydrocarbon analyses are given in figures 3-10, 3-11, 3-12, and 3-13. Samples from each of the capsules were analyzed for amine and nitrogen gases using the Tracor 220 gas chromatograph. The column waa 1/4 x 6" stainless steel packed with Pennwalt 223 amine packing. Significant amounts of amines were not detected in these samples.

Hydrogen was determined in each of the samples using a portable gas chromotograph (pre-production prototype) manufactured by Zethus Research Corporation. The column was 1/8" x 36" stainless steel packed with Porapak R and the Carrier was air at a flow rate of 9cc/min. The system was calibrated using hydrogen/nitrogen mixtures obtained from Supelco Inc. Figure 3-14 shows the results of the H_2 determinations.

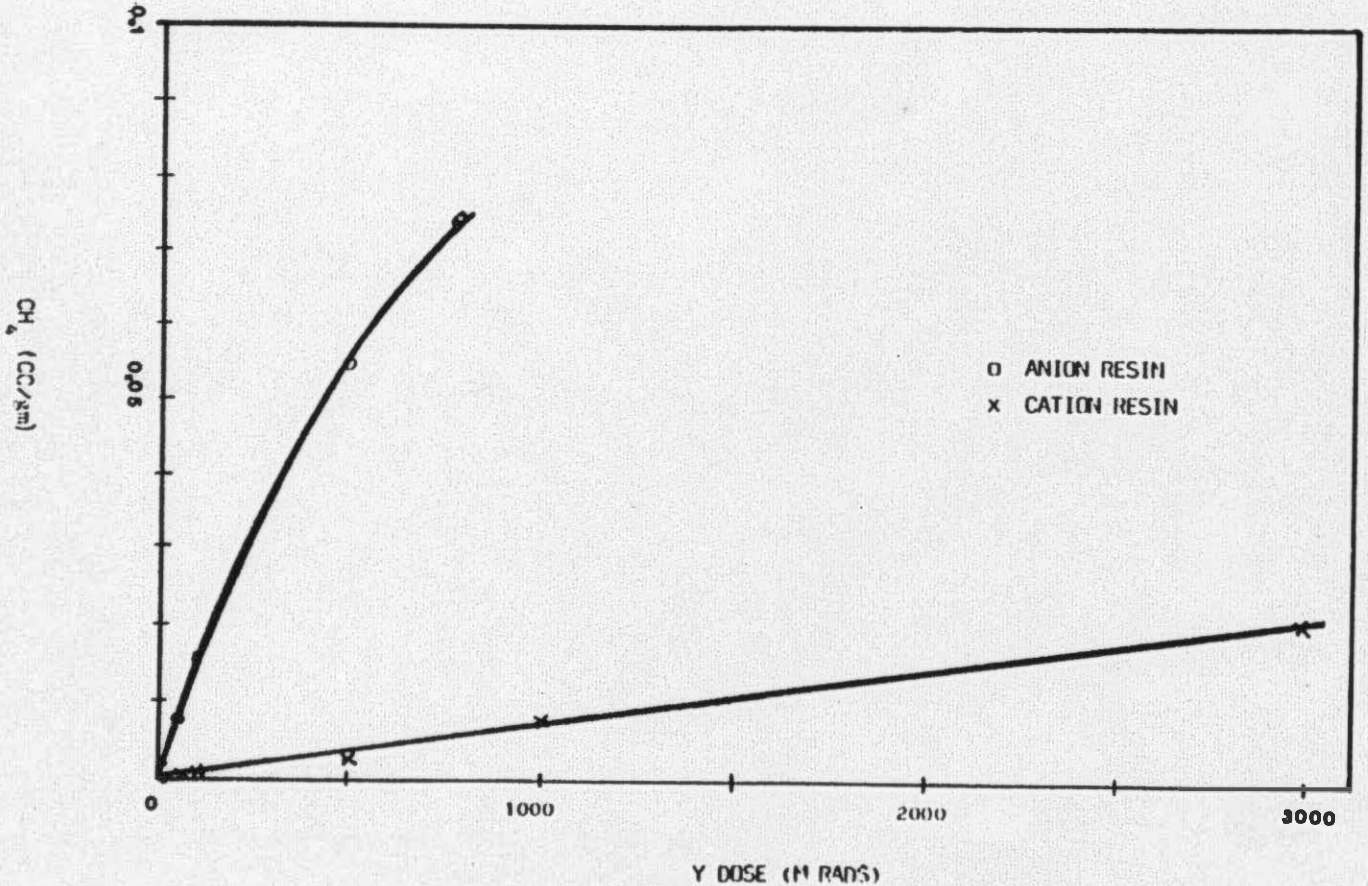


FIGURE 3-10 Methane Production

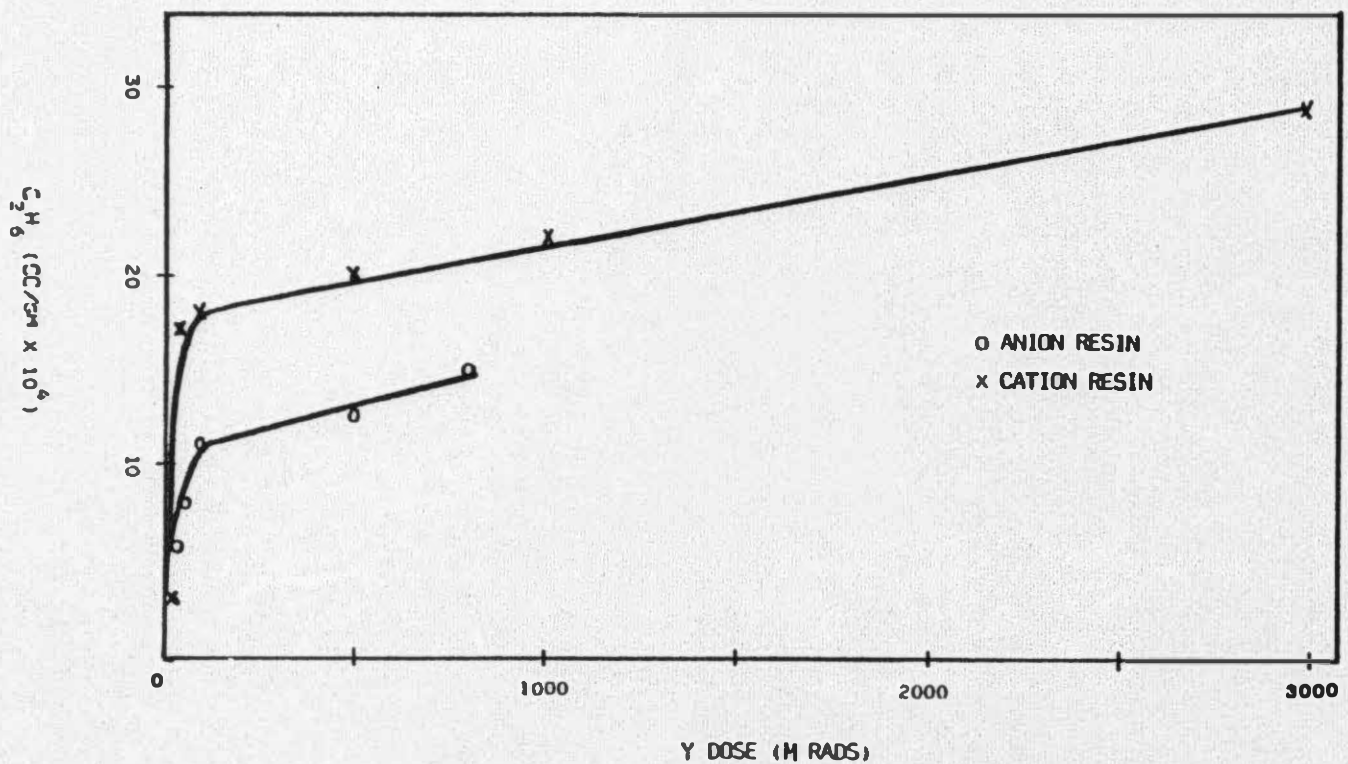


FIGURE 3-11 Ethane Production

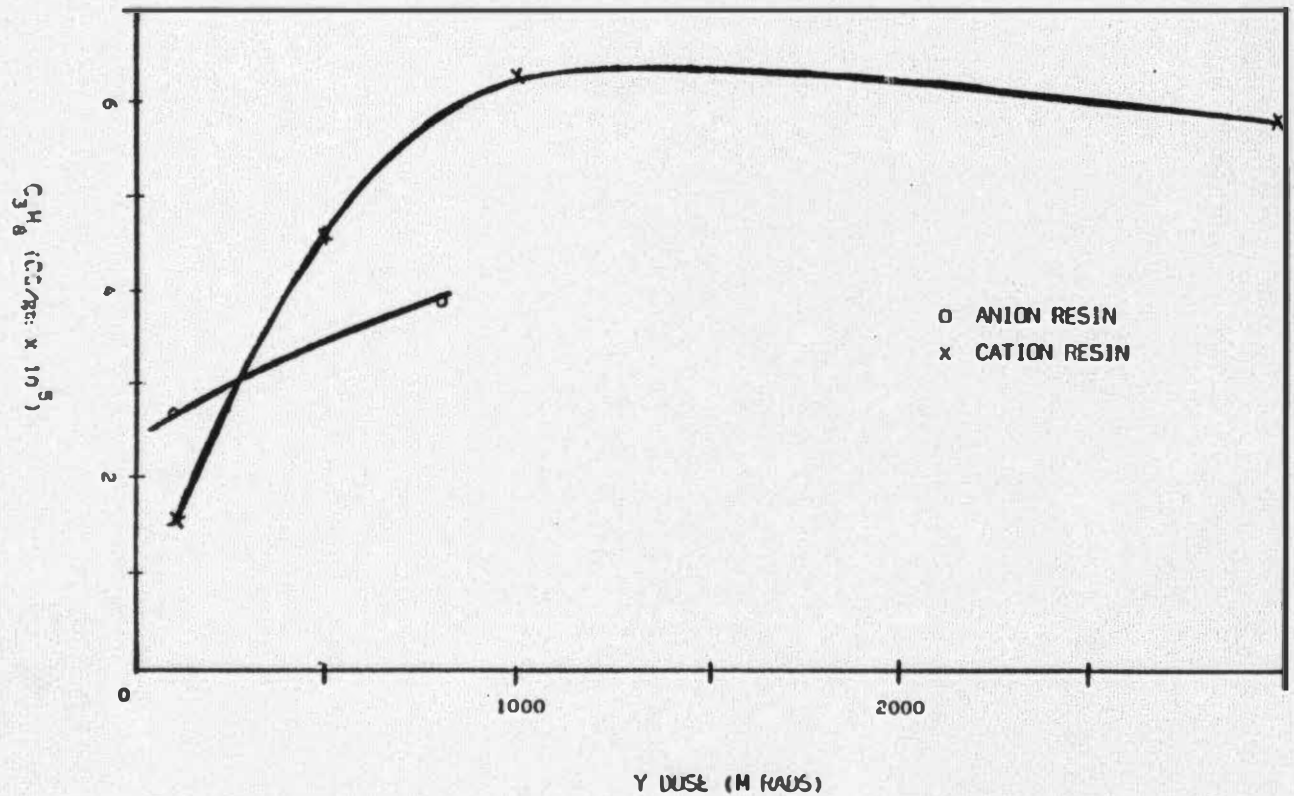


FIGURE 3-12 Propane Production

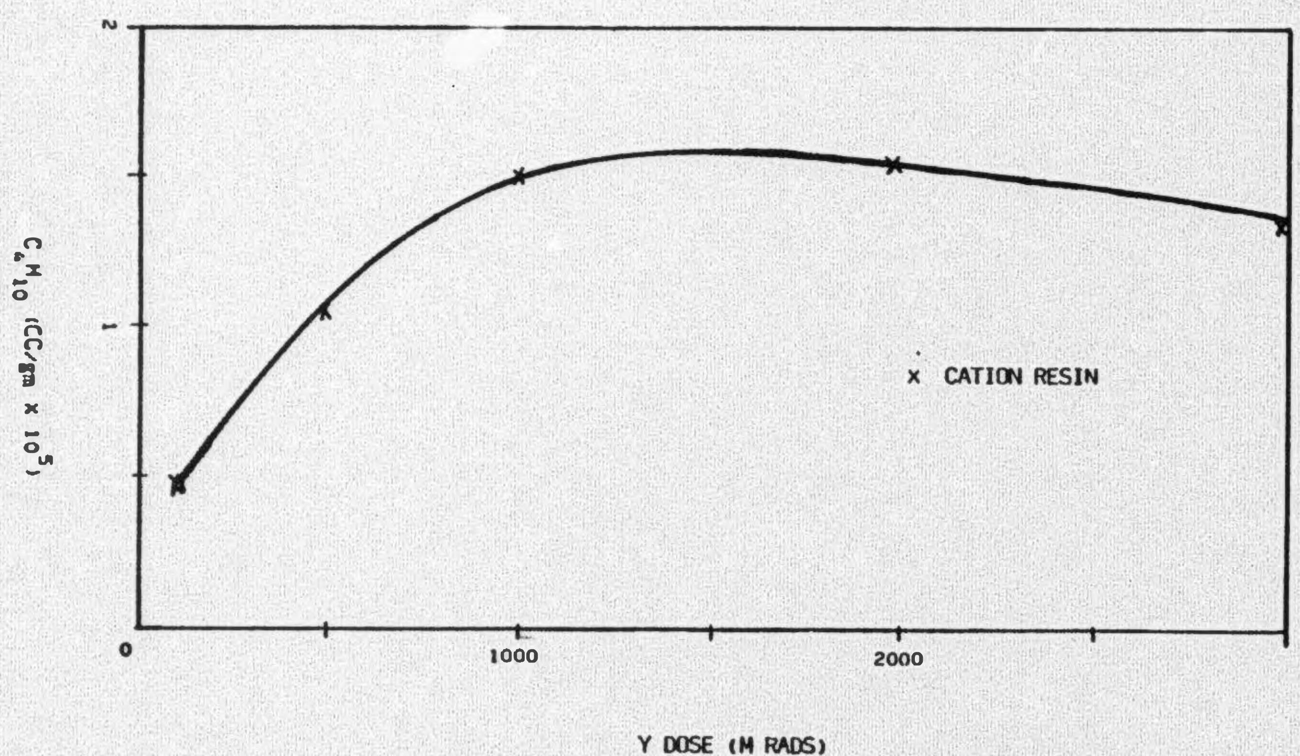


FIGURE 3-13 Butane Production

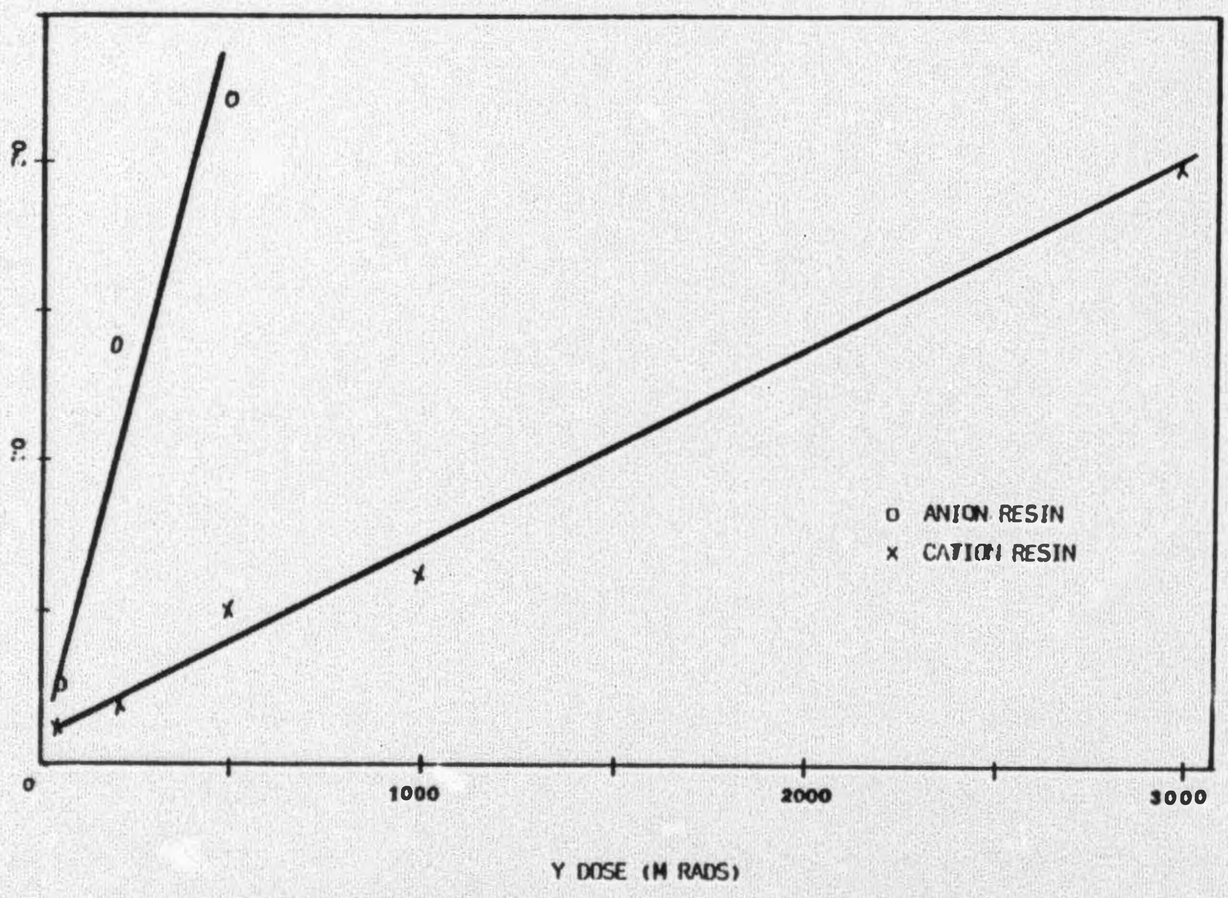


FIGURE 3-14 H_2 Production

6.0 APPENDIX B - CALCULATION OF DOSE TO BURIED ADSORBENT MATERIAL

Container Dimensions - h=136 cm, r=30.5 cm

Resin Load/Container - $10ft^3 \rightarrow 2.84 \times 10^5 cm^3$

Assumed Activity Loading (per container)

420 Ci Cs-137
50 Ci Cs-134
70 Ci Sr-90
120 Ci Ce-144

6.1 Beta Dose.

Dose Rate =

$$\frac{(Q\ Ci)(3.7 \cdot 10^{10} dps/Ci)(\bar{E}\ MeV)(1.6 \times 10^{-6} ergs/MeV)(3600\ sec/hr)}{(W gms)(100\ ergs/gm/rad)}$$

Assume:

Resin Density = 1gm/cc, or 2.84×10^5 gms Resin per Container

Dose Rate From Cs-137 β

β_1 1.18 MeV - 6.0%

β_2 0.514 MeV - 93.5%

0.557 MeV Average Maximum Energy (Weighted Average)
0.186 MeV Average Beta Energy (Branching Factor 1/3 X 0.557)

$$DR = \frac{(420Ci)(3.7 \times 10^{10} dps)(0.186\ MeV)(1.6 \times 10^{-6} ergs/MeV)(3600 sec/hr)}{(2.84 \times 10^5 gms)(100\ ergs/gm/rad)}$$

= 586 Rads/Hr

$\lambda = 0.0231\ yr^{-1}$ or $6.326 \times 10^{-5} day^{-1}$

$$\text{Ten Year Dose } D_{10} = \frac{(586 Rads/Hr)(24hr/day)(1-e^{-\lambda 3652.5})}{6.326 \times 10^{-5} day^{-1}}$$

= 4.60×10^7 Rads

$$D_\infty = \frac{(586\ Rads/Hr)(24hrs/day)}{6.326 \times 10^{-5}}$$ = 2.22×10^8 Rads

Dose Rate From Cs-134β

β_1 0.662 MeV 71%

β_2 0.089 MeV 28%

0.495 MeV Average Maximum Energy (Weighted Average)
0.165 MeV Average Beta Energy (Branching Factor 1/3 X 0.495)

DR = (50)(0.165)(7.50) = 62Rads/hr

$\lambda = 9.257 \times 10^{-4} \text{day}^{-1}$

$D_{10} = (62)(24)(1-e^{-\lambda 3652.5})$ $= \underline{1.55 \times 10^6 \text{Rads}}$

$D_\infty = \frac{(62)(24)}{9.257 \times 10^{-4}}$ $= \underline{1.6 \times 10^6 \text{Rads}}$

Dose Rate From Sr-90β

β_1 0.546 MeV 100%

β_2 2.27 MeV 100%

2.82 MeV Maximum Beta Energy

0.94 MeV Average Beta Energy

$DR = (70)(.94)(7.50)$ $= \underline{494 \text{ Rads/hr}}$

$D_{10} = (494)(24)(1-e^{-\lambda 3652.5})$ $= \underline{3.834 \times 10^7 \text{ Rada}}$

$D_\infty = \frac{(494)(24)}{6.778 \times 10^{-5}}$ $= \underline{1.75 \times 10^8 \text{ Rads}}$

Dose Rate From Ce-144β

β_1 0.33 MeV 75%

β_2 0.16 MeV 25%

β_{Daughter} 2.996 MeV 98%

3.284 MeV Maximum Average Energy

1.10 MeV Average Beta Energy

$DR = (120Ci)(1.10)(7.50)$ $= \underline{990 \text{ Rads/hr}}$

$\lambda = 2.44 \times 10^{-3} \text{day}^{-1}$

$D_{10} = (990)(24)(1-e^{-\lambda 3652.5})$ $= \underline{9.73 \times 10^6 \text{Rads}}$

$D_\infty = \frac{(990)(24)}{2.44 \times 10^{-3}}$ $= \underline{9.74 \times 10^6 \text{Rads}}$

6.2 Total Beta Dose

$D_{10} = 4.60 \times 10^7 + 1.55 \times 10^6 + 3.83 \times 10^7 + 9.73 \times 10^6 = \underline{9.56 \times 10^7 \text{ Rads}}$

$D_\infty = 2.22 \times 10^8 + 1.6 \times 10^6 + 1.75 \times 10^8 + 9.74 \times 10^6 = \underline{4.08 \times 10^8 \text{ Rads}}$

6.3 Gamma Dose

Dose Rate = C Γ g Rada/hr, Where

C = m Ci/Cm^3

Γ = gamma ray constant

g = average geometry factor

Cylinder r=30.5 cm, h=136 cm; g ∿150 assumes tissue eq ivalent.

Dose Rate From Cs-137 Γ

Γ = 3.3 Rads Cm^2/hr - mCi

$$DR = \frac{(3.3)(4.2 \times 10^5 \text{ mCi})(150)}{2.84 \times 10^5 Cm^3} = \underline{832 \text{ Rads/hr}}$$

D_{10} = $\underline{5.74 \times 10^7 \text{Rads}}$

$D\infty$ = $\underline{2.80 \times 10^8 \text{Rads}}$

Dose Rate From Cs-134 Γ

Γ = 8.7

$$DR = \frac{(5 \times 10^4 \text{mCi})(817)(150)}{2.84 \times 10^5 \text{ Cm}^3} = \underline{230 \text{ Rads/hr}}$$

D_{10} = $\underline{5.75 \times 10^6 \text{ Rads}}$

$D\infty$ = $\underline{5.95 \times 10^6 \text{ Rads}}$

Dose Rate From Sr-90 - Gamma Dose Is Zero

Dose Rate From Ce-144 Γ

Γ = 0.4

$$DR = \frac{(1.2 \times 10^5 \text{ mCi})(0.4)(150)}{2.84 \times 10^5 \text{ Cm}^3} = \underline{25.4 \text{ Rads/hr}}$$

D_{10} = $\underline{2.49 \times 10^5 \text{Rads}}$

$D\infty$ = $\underline{2.52 \times 10^5 \text{Rads}}$

6.4 **Total Gamma Dose**

$D_{10} = 5.74 \times 10^7 + 5.75 \times 10^6 + 0 + 2.49 \times 10^5 = \underline{6.34 \times 10^7 \text{Rads}}$

$D\infty = 2.80 \times 10^8 + 5.59 \times 10^6 + 0 + 2.52 \times 10^5 = \underline{2.86 \times 10^8 \text{Rads}}$

6.5 **Total Dose - Beta and Gamma**

$D_{10} = D_{10}\beta_{total} + D_{10}\gamma_{total}$

$= 9.56 \times 10^7 + 6.34 \times 10^7 = \underline{1.59 \times 10^8 \text{Rads}}$

$D\infty = D\infty\beta_{total} + D\infty\gamma_{total}$

$= 4.08 \times 10^8 + 2.86 \times 10^8 = \underline{6.94 \times 10^8 \text{Rads}}$

SECTION 4

OPERATIONS AND MAINTENANCE

1.0 SUBTASK A. PLANT ENGINEERING

1.1 SCOPE

1.1.1 REACTOR BUILDING PURGE

Provide engineering support to the Special Projects group for the purge program.

1.1.2 IN-PLACE RECOVERY SYSTEMS AND PRE-ACCIDENT PLANT SYSTEMS

Provide engineering support to operations and maintenance to solve problems encountered during operations and maintenance. Define required design modifications. Prepare operations and test procedures.

1.1.3 PLANT FIRE PROTECTION

Provide engineering support for plant fire protection systems and procedures. Interface with insurance organizations. Prepare design criteria for new systems, and modifications to existing systems.

1.1.4 RECOVERY SYSTEMS

Provide engineering support to Recovery Engineering during the development of design criteria, and a review of system designs from operations and maintenance viewpoint.

1.1.5 PLANT CHEMISTRY AND RADIOCHEMISTRY

Obtain liquid and gas samples from plant systems and facilities to support plant operations and special projects. Analyze samples chemically and perform radioactive counting. Evaluate and report results.

1.1.6 START-UP AND TEST

Provide start-up and test services for new recovery systems.

1.2 CURRENT ACTIVITIES

1.2.1 REACTOR BUILDING PURGE STATUS

Calibration procedures for the stack and purge line radiation monitors, to be utilized during the purge process were prepared. Containment isolation valve operability was confirmed. A revised procedure for sampling and analysis of the reactor building atmosphere was prepared.

1.2.2 IN-PLACE RECOVERY SYSTEMS AND PRE-ACCIDENT PLANT SYSTEMS STATUS

1.2.2.1 STANDBY PRESSURE CONTROL SYSTEM

This system is operational and has been used intermittently for RCS pressure control. The small variable charging pump has incurred periodic packing failure. A replacement pump of different design is being investigated. Provisions for taking total gas samples at the system interface with the makeup system are being made.

1.2.2.2 EMERGENCY DIESEL GENERATOR MAINTENANCE

Scheduled maintenance and testing of the in-plant emergency diesel generator was completed.

1.2.2.3 WASTE GAS COMPRESSOR

An evaluation of the previous failure of the Waste Gas Compressor was performed. As a result, operating procedure revisions were instituted.

1.2.2.4 REACTOR BUILDING WATER LEVEL MEASUREMENT

Installation of an alternate method for reactor building water level measurement (manometer) was completed and successfully tested. Results to date are consistent with the previous procedure (pressure measurement on the building sump recirculation line).

1.2.2.5 STEAM GENERATOR WATER LEVEL MEASUREMENT

The back-up steam generator water level measurement systems were modified to provide a more accurate reading and more useable indication.

1.2.2.6 HEATING AND VENTILATION SYSTEMS

DOP efficiency testing of the auxiliary building and fuel handling building HEPA filters was performed.

1.2.3 PLANT FIRE PROTECTION STATUS

An update to the Fire Protection Plan, considering present plant conditions, is currently in preparation. Installation of the fire protection system for the solid waste storage facility, south of the plant, is nearing completion. Engineering for an associated, remote fire alarm was completed. Evaluation of cracking in "Firewall 50" barriers continues.

1.2.4 RECOVERY SYSTEMS STATUS

1.2.4.1 MINI DECAY HEAT SYSTEM

Instrument calibration packages were completed. Operating and emergency procedures, and alarm responses were issued for review.

1.2.4.2 SUBMERGED DEMINERALIZER SYSTEM (SDS)

The instrument list was reviewed. Requirements, to ensure conformance of instrumentation hardware and software to existing plant standards, were reviewed with Chem Nuclear.

1.2.4.3 SYSTEM DESIGN CONCEPTS

Design concepts for power distribution to SDS, waste evaporation, and other temporary and permanent recovery facilities, were developed with Bechtel Corporation.

1.2.4.4 DIESEL GENERATORS AND 13.2 KV TRANSFORMERS

Technical evaluation report, revisions to blackout procedures, and load dispatching procedures were prepared, in support of disconnection and removal of the "BOP" Diesel Generators and 13.2 KV backup transformers.

1.2.5 PLANT CHEMISTRY AND RADIOCHEMISTRY STATUS

Construction is nearing completion on the expansion to the gamma spectroscopy facility on the turbine deck.

Engineering was completed for installation of an on-line oxygen monitor for reactor coolant sampling.

1.2.6 START-UP AND TEST STATUS

Start-up of the Standby Pressure Control System was completed, with turnover to operations. Testing of the Temporary Sampling System and Mini Decay Heat Systems were completed, with the exception of final leak testing, delayed by design modifications.

2.0 SUBTASK B. PROCESS SUPPORT

2.1 SCOPE

The Process Support group provides overall guidance to Plant Operations in the area of radioactive water processing. Within this responsibility, other more specific tasks are identified:

1. Water processing scheduling includes planning and coordination of contaminated water transfers, to accomplish cleanup expeditiously.
2. Shift Radwaste Engineering provides operations advice, on an around-the-clock basis for the Operations group. This includes data analysis of all water movement and processes.
3. Recovery Systems Engineering provides design, design review, construction management, operational review and performance evaluation, of systems associated with water processing and radioactive material handling.

4. Radioactive Material Shipment Engineering includea evaluating compliance requirements with NRC/DOT regulations for radwaste shipments. Radioactive shipments of samples and waste are prepared for transport. These shipments are coordinated and monitored to ultimate destination.

5. Respirators and anti-contamination clothing are processed for re-use by Support Services. These articles are picked up at predetermined check points, cleaned to required standards and returned to the check points.

2.2 CURRENT ACTIVITIES

2.2.1 FUEL POOL WASTE STORAGE SYSTEMS STATUS

2.2.1.1 SYSTEM FUNCTION AND DESIGN OBJECTIVES

This Fuel Pool Waste Storage System is used for temporary storage of liquid waste. These tanks add approximately 110,000 gallons to the present storage capacity of the plant, and are located within the "A" spent fuel pool. These tanks can be filled with liquid waste from the Reactor Building Sump and the Miscellaneous Waste Hold-Up Tank. This system enhances the capability of the plant to move and process radioactive waste.

2.2.1.2 SYSTEM DESCRIPTION

The system consiats basically of upper (4 at 15,000 gallons each) and lower (2 at 25,000 gallons each) tanks, forming two separate storage areas. Either storage area is capable of being filled from either the Reactor Building Sump or the Miscellaneous Waste Hold-Up Tank, and each has level indication. The tanks are protected from over-filling by automatically closing the feed valve when the storage area is nearly full. Provisions have been made to both flush the piping system after completion of the pumping operation, and to drain the piping system aa required.

The vents from the tanks and the stand pipes are directed through a dryer and a charcoal filter, to remove moisture and iodine before proceeding to the fuel pool ventilation system. The tanks and vent system is protected by a relief valve which vents through a parallel set of dryers and charcoal filters.

The tanks will be emptied as necessary by steam eductors. Two eductors are permanently installed in each stand pipe.

2.2.1.3 SYSTEM OPERATION

Water is transferred from the Reactor Building Sump of the Miscellaneous Waste Storage Tank to the tank farm. After either the lower set of tanks or upper set of tanks is full, the level controllers automatically close the air operated inlet valves.

Air forced from the tanks during the filling process is vented to a charcoal filter & dryer to remove moisture and iodine. This air is then piped to the Fuel Pool Ventilation System.

The steam eductors give the capability to transfer waste water from the tank farm to the Miscellaneous Waste Storage Tank or EPICOR II Rad Waste System, from the upper tanks to the lower tanks in the tank farm (or vice versa) or to recirculate the water in the tanks.

A high temperature alarm, and temperature switch to close the steam control valve, is installed in the tank vent line to prevent damage to the filter/dryer skids during use of the eductors.

2.2.1.4 SYSTEM STATUS

The system is being used to store 93,000 gallons from the Unit 2 Miscellaneous Waste System. The steam eductors have not been used, since no water has been pumped out of the tanks to date. The water in the tanks falls in the intermediate activity category. Shielding has proven highly adequate, and access above, and adjacent to the "A" Spent Fuel Pool, is unrestricted.

2.2.2 LIQUID RADIOACTIVE WASTE PROCESSING SYSTEM TITLED "EPICOR II" STATUS

2.2.2.1 SYSTEM FUNCTION AND DESIGN CRITERIA

The system is designed to cleanup radioactive liquids so as to produce water capable of being released from Three Mile Island. Cleanup includes removal of radioisotopes and chemical constitutents, to comply with Plant Technical Specifications for water releases to the Susquehanna River. The design is optimized with respect to ALARA considerations.

Instrumentation and controls are provided for monitoring system performance. Water flows are monitored where the values are critical to the process and/or system safety. Inline monitoring and comprehensive sampling system are provided, for thorough analyses of system water cleanup performance. Radiation and airborne monitoring equipment is provided for analysis of activity levels.

Shielding is being provided to minimize exposure related to the operation of this system.

An HVAC subsystem is utilized to cleanup and monitor any gases that might be released from the liquid processing system. It is the goal to minimize gas releases from the system, however, should they occur, they will be cleaned to reduce any releases to the environment. Monitoring of the air exhaust will continue to detect any potential radioactive gas. A slight negative pressure is maintained to ensure buildng inleakage is maintained. The system is being optimized with respect to ALARA considerations.

2.2.2.2 SYSTEM DESCRIPTION

2.2.2.2.1 LIQUID PROCESSING

The TMI Station Chemical Cleaning Building is used to house the system along with the existing tankage and sump existing in that building. Piping and pumps are provided for water movement through cleanup vessels. The system is composed of a pre-filter, two demineralizers and an after-filter. The pre-filter and demineralizers are designed for ease of hookup and disconnect to allow for quick installation and remote, reliable removal.

2.2.2.2.2 GAS PROCESSING

The primary components are a fan, an air cleanup filter train, and necessary ducting. The main HVAC components are located external to the Station Chemical Cleaning Building, but are enclosed in their own shelter.

2.2.2.3 SYSTEM OPERATION

The Auxiliary Building Emergency Liquid Cleanup System consists of a vendor supplied, liquid radwaste process system which is located in the Chemical Cleaning Building. The system is designed to decontaminate, by filtration and ion exchange, approximately 400,000 gallons of radioactive waste water contained in the Auxiliary Building of TMI-2. Contaminated water is being pumped from a connection located on the Miscellaneous Waste Holdup Tank (WDL-T-2), by a pump located on the Chemical Cleaning Building, through the yard and into the process system. Yard piping is enclosed within a guard pipe, the end of which terminates inside the Chemical Cleaning Building.

Decontaminated water is delivered to the Clean Water Receiving Tank (CC-T-2) for sampling and analysis, and pumped to the Liquid Waste Disposal System of TMI-2 for storage if within specs. Otherwise, it is transferred to the Off Spec Water Receiving Batch Tank (CC-T-1) for recycling through the process system. Capability also exists to discharge to a tank truck and the TMI-2 "B" Spent Fuel Pool. CC-T-1 may also be used for storage.

The Chemical Cleaning Building (CCB) has been made into a low leakage, confinement building, and provided with an exhaust ventilation system to maintain the building at a negative pressure. HEPA and charcoal filtering is provided on the ventilation system, which discharges to a local stack at the roof line of the CCB where all effluent air is monitored for radioactivity.

Normal operation of the processing system is by remote means except for infrequent operations, such as sampling and chemical addition. All remote system operations are controlled from the TV Monitor Control Building, located outside the northwest corner of the Chemical Cleaning Building.

Remote handling of spent resin containers, from their position inside the Chemical Cleaning Building to the transport cask and truck, are provided.

The system interfaces with the TMI-2 Radwaste Disposal Miscellaneous Liquids System, the TMI-2 Liquid Waste Disposal System, Demineralized Water System and the Service Air System.

2.2.2.4 SYSTEM STATUS

The system is operating successfully and has processed approximately 166,000 gallons. A total of thirty-four (34) spent resin liners have been used and are in the waste staging area. Processed water is stored in CC-T-1, Evaporator Condensate Test Tank B and Spent Fuel Pool "B".

During the month of January, the system underwent extensive testing to identify specific chemistry related problems. Upon completion of this testing, the system was started up and resumed processing. Due to information gathered during testing, system performance was significantly improved.

PROCESSING PERFORMANCE TABLE (BATCH 26)

Inlet (μci/ml)		Effluent (μci/ml)
Cs^{137}	31.65	$< 5.586 \times 10^{-6}$
Cs^{134}	5.974	$< 7.541 \times 10^{-6}$
Sr^{89}	0.25	2.79×10^{-6}
Sr^{90}	0.48	5.37×10^{-6}

Highly successful processing of the "A" Reactor Coolant Bleed Tank was completed. This was the highest activity water in the Auxiliary and Fuel Handling Building.

1. Processing rate —— over 2 gpm (Highest to date)
2. Man Rem exposure per gallon processed —— .021 (Lowest to date)

An outage was begun on March 14, for the purpose of system modification. The modification will enhance system reliability and provide greater processing flexibility.

2.2.3 STAGING FACILITIES FOR DEWATERED RESINS AND EVAPORATOR BOTTOMS STATUS

2.2.3.1 WG-21 - INTERIM SOLID WASTE STAGING FACILITY

2.2.3.1.1 SYSTEM FUNCTION AND DESIGN CRITERIA

Facilities are needed to stage dewatered radioactive resin and filters, generated by EPICOR I and EPICOR II until they can be shipped to a burial site. WG-21 provides space for this staging.

2.2.3.1.2 SYSTEM DESCRIPTIONS

The facility consists of 16-54" diameter cells and 12-84" diameter cells to receive 4' X 6' and 6' X 6' resin liners. The cells are installed in the Unit-2 cooling tower desilting basin, backfilled for shielding and capped with 3' thick concrete plugs.

2.2.3.1.3 SYSTEM OPERATION

Eight (8) EPICOR I Resin Liners, one (1) EPICOR I Prefilter, and one (1) smaller resin liners (used to remove trace activity and fluorescein dye) are staged in the facility. Sixteen (16) EPICOR II resin liners and one (1) Unit 1 used precoat liner are also staged in the facility.

2.2.3.1.4 SYSTEM STATUS

The interim solid waste staging facility is operational. Additional shielding (lead bricks) was installed along the interface, between the cell ver and facility top, to provide shielding due to streaming on some of the cells that are loaded. Readings are below the 5 mr/hr design criteria. All but one 4 X 4 and one 6 X 6 cell are filled. We have commenced shipment to the burial site from the cells.

2.2.3.2 WG-22 SOLID WASTE STAGING FACILITY

2.2.3.2.1 SYSTEM FUNCTION AND DESIGN CRITERIA

Facilities are required to stage the following radioactive wastes until they can be shipped to a burial site:

1. Dewatered radioactive resins from EPICOR I.
2. Dewatered radioactive resins from EPICOR II.
3. Dewatered radioactive resins or solidified evaporator bottoms from systems used to process water more radioactive than that processed by EPICOR I or EPICOR II.

The sump meets the seismic requirements of USNRC Regulatory Guide 1.143. Contact readings on the sides of the facility will be less than 0.5 mr/hr and less than 2.5 mr/hr on the top.

2.2.3.3 SYSTEM DESCRIPTION

The facility is designed as a modular one. Each module consists of 60" - 84" diameter cells imbedded in concrete capped with 3' thick concrete plugs. Each cell has a drsin line to a sump which will serve three modules. The sump is designed to collect any leakage from liners installed in the cells and meets the seismic requirements of USNRC Regulatory Guide 1.143.

2.2.3.4 SYSTEM OPERATION

Module A is complete with 27 cells in use.

2.2.3.5 SYSTEM STATUS

Module A is operational:

1. First liner in Module A - January 8, 1980.
2. As of March 31, 1980, there were thirty (30) liners in the A Module.

Radiation levels are as expected.

Work is underway on Module B:

1. Scheduling B Module is in progress; integrating expected liner generation rate with construction rate.

2.2.4 NUCLEAR SAMPLING SYSTEM STATUS

2.2.4.1 SYSTEM FUNCTION AND DESIGN OBJECTIVES

This nuclear sampling system is to be used as a temporary liquid waste sampling facility, to allow TMI-2 recovery operations to continue without interfering in the normal operations of TMI-1, when that unit is returned to service. It will provide a single controlled station, whereby fluid samples may be taken from tanks otherwise inaccessible for local sampling, and/or from tanks that require frequent sampling for analyses of chemical and radiochemical content. Included in the sampling scope will be capability for representative samples of TMI-2 Reactor Coolant from the pressurizer steam or water space or upstream of letdown coolers, and from the Mini-Decay Heat System; samples from the three TMI-2 Reactor Coolant Bleed Tanks, TMI-2 Miscellaneous Waste Hold-Up Tank and the Fuel Pool Waste Storage System, containing liquid waste from both the TMI-2 Reactor Building Sump and Miscellaneous Waste Hold-Up Tank. Provisions have also been provided in the system for monitoring of boron concentration in the reactor coolant.

2.2.4.2 SYSTEM DESCRIPTIONS

TMI-2 Sample Lines, which presently run into TMI-2 sampling area, shall be rerouted to a new sample sink which will be located in the Fuel Handling Building 305' elevation of TMI-2. In an adjacent room, the so-called "model room" a boronometer shall be installed.

The system shall provide for adequate recycle, purge and return of waste liquids. Purging of radioactive piping shall be performed prior to installation of new sample lines.

Drainage from the sample sink will be routed to the Fuel Pool Waste Storage System. A shielded bottle to collect drainage will also be provided.

All piping, valves and components of the sampling system will meet the design conditions of the system with which they are associated, or will meet 150 psig and 200°F. Primary coolant sampling points will have the design condition of 2500 psig and 670°F up to valve SNS-V-70.

Air exhausted from the sample hood will be filtered through charcoal and HEPA filters, and discharged to the Auxiliary Building ventiliation system exhaust ductwork.

2.2.4.3 SYSTEM OPERATION

Samples from the Reactor Coolant, Mini Decay Heat Removal and Fuel Pool Waste Storage Systems, can be collected by in-line sample containers. All samples may be taken as grab samples. In addition, a boronometer is provided to monitor the boron concentration in the primary coolant, circulating in the sample lines. A separate loop, with its own pump tank, is provided to limit fluid temperatures at the boronometer inlet, and to limit reactor coolant grab sample temperatures.

Sample lines from the Reactor Coolant System and the Mini Decay Heat Removal (MDHR) System, are purged either to the Bleed Hold-Up Tank or to the TMI-2 Miscellaneous Waste Hold-Up Tank. Sample Lines from the Fuel Pool Waste Storage System are purged first, on a continuous cycle within the storage system standpipes, and then on a batch basis to the sample sight bottles from which the water is returned to the waste storage system. Sample lines to the sample hood sink are purged through the sink to sample waste container SNS-T-1, which may be drained to the sample sight bottle by drawing a vacuum. From there, the water is returned to the Fuel Pool Waste Storage System.

The sample hood and room are ventilated by a filtered ventilation system which draws air from the hood and the room and exhausts to the Auxiliary Building ventilation system.

Containment isolation valves are operated from the TMI-2 Control Room (Containment Isolation Panel 15). For the reactor coolant and radwaste samples, some valve control is from Panel 329 in Unit 1 at the original nuclear sampling station. For the Fuel Pool Waste Storage

System, some local valve operation is necessary at the fuel pool. An intercom station operation is necessary at the fuel pool. An intercom station is provided in the sample room to coordinate sampling operations.

Valves, at or near the sample hood, are all manually operated. Pressure and temperature instruments and flow meters, at or near the hood, are read locally.

All liquid samples collected in containers are analyzed elsewhere, either in TMI-1 or offsite.

2.2.4.4 SYSTEM STATUS

The system design is essentially complete. Construction is in progress and will be completed in early 1980.

A start-up and test procedure was written to verify system construction and operation. Modifications were done to enhance system operability.

2.2.5 SOLID WASTE STATUS

Solid Waste, in the form of LSA boxes and LSA drums, continues to be generated and shipped as possible. The generation rates for the first quarter of 1980 are:

LSA Boxes	15	Boxes/Month
LSA Drums	85	Drums/Month

These continue to be stored onsite until shipment for disposal occurs.

Specifically tailored training of personnel involved in generating and handling this waste, has been implemented this quarter to reduce the volume of waste. Volume reduction of waste has been a key objective and continues to be a goal of the Waste Management Group.

Shipment of waste has not been allowed since mid February. Completion of training requirements is in progress to gain necessary NRC approval to reestablish shipment.

3.0 SUBTASK C. DECONTAMINATION

3.1 SCOPE

3.1.1 DECONTAMINATION OF AUXILIARY AND FUEL HANDLING BUILDING

The decontamination basis for the Auxiliary and Fuel Handling Building is to achieve less than 1000 DPM contamination levels in all areas and to reduce radiation levels to within design levels.

3.2 CURRENT ACTIVITIES

3.2.1 DECONTAMINATION STATUS

1. Decontamination of open areas (corridors, stairwells, etc.) is 89% complete. Contamination levels on the 328' and 305' elevations have been reduced to less than 1000 DPM, and general radiation levels are less than (1) mR/hr. Contamination of the 280' 6" elevation is less than 2000 DPM, and general radiation levels are less than one (1) mR/hr.

2. Decontamination of cubicles continued, with the following cubicles deconned to less than 1000 DPM:

   a. Spent resin A&B cubicles.
   b. Spent resin transfer pump cubicle.
   c. Waste gas compressor A&B cubicles.
   d. Waste gas filter cubicle.
   e. Valve rooms associated with the waste gas system.
   f. Deborated demineralizer cubicles.
   g. Valve rooms associated with the deborated demineralizer system.
   h. Spent fuel demineralizer cubicle.

Cubicle decontamination is seventy-five (75%) percent complete.

3. Fifty (50%) percent of the floor drain covers and drain bells were removed and the drain inlets deconned in preparation for hydrolasing the drain lines.

4. An additional one thousand eighty(1,080) gallons of decontamination solution has been solidified during this report

period. The generation of decontamination solution waste diminished to approximately one hundred fifty (150) gallons with the use of processed water and hydrolasing.

5. Four (4) tanks were inspected for sludge accumulation and were found to be clean. The tanks were:

   a. Evaporator condensate test tanks A&B.
   b. Contaminated drain tanks A&B.

6. The Auxiliary Building sump tank was sampled for sludge and was found to be free of same. A test was run on the inside walls of the tanks by hydrolasing areas inside the tank. It was determined by this test that the hydrolaser removed the contamination from the areas tested, indicating that the tank internals can be decontaminated utilizing this technique.

7. The area where the hydrogen recombiner was removed from TMI-2 was decontaminated. The control panel was released. The hydrogen recombiner is being decontaminated for removal.

4.0 SUBTASK D. PLANT MAINTENANCE

4.1 SCOPE

The objective of the TMI-2 Maintenance Department is to repair, replace, calibrate and maintian plant equipment in an operable condition.

To this end, Met-Ed personnel are disciplined in Electrical, Mechanical, Instrument and Controls, and Utility (general labor) Departments. These Departments are further divided into Corrective, Preventive and Layup Maintenance sections to address the varied facets of a complete maintenance program. Should additional manpower be required, the Departments are supplemented by outside contractor personnel.

4.2 CURRENT ACTIVITIES

4.2.1 AUXILIARY BUILDING VENTILATION SYSTEM STATUS

Three, of the four Auxiliary Building ventilation exhaust fans, were out of service at the beginning of the quarter. The objective was to have all four fans back in service.

The "A" train of the exhaust system is now in service. The "B" train is presently out of service while installing a refurbished fan motor.

4.2.2 EQUIPMENT LAYUP STATUS

Work continued on plans, procedures and schedules for the systematic layup of equipment that will be removed from service during the recovery period.

A detailed turbine layup schedule has been prepared. Plans are now being formulated for the balance of the secondary plant. No real implementation of the layup plans can be accomplished until an alternate means of core cooling is established.

4.2.3 NUCLEAR SERVICE RIVER WATER PUMP STATUS

The "B" and "D" pumps are presently out of service. These and all other deep well, vertical pumps have had problems generically on both units.

Currently, repairs cannot be made until replacement parts are received. Parts are due to ship in April. Rebalancing has been completed to the maximum extent possible without replacement parts. Engineering is evaluating the generic problems associated with these pumps.

4.2.4 WASTE GAS COMPRESSOR STATUS

The Waste Gas Compressor is currently out of service, awaiting ordered replacement parts for repair.

4.2.5 NEW PROJECT INSTRUMENTATION STATUS

Ongoing support, such as calibration and recording baseline data for Mini Decay Heat System instrumentation, EPICOR II instrumentation, and in plant radiation monitors, continued during this quarter.

4.2.6 PREVENTIVE MAINTENANCE STATUS

Work continued on developing and implementing programs for scheduled inspections and preventive maintenance of equipment, as recommended by vendors or from past experience, for specific equipment. Annual emergency diesel inspections have been completed, as have electrical inspections per industry bulletin , oil and grease changes, meggering of electrical equipment and IRD vibrational analyses of rotating equipment.

SECTION 5
RADIOLOGICAL CONTROLS

1.0 SCOPE

The objective of the Radiological Control Department is to develop and administer a strong, efficient Radiological Control Program in support of the recovery activities at TMI-2. The specific actions planned to achieve this objective are documented in the Management Plan for TMI-2 Radiological Control Program (Appendix A). The objectives for each group within the Radiological Control Department are presented below.

1.1 RADIOLOGICAL TECHNICAL SUPPORT GROUP

The Radiological Technical Support Group is responsible for revising all current Radiological Control procedures (Health Physics procedures) and generating new procedures for operations not previously addressed in a procedure. In addition, this group provides technical support for all Radiological Control functions and coordinates the ALARA program, and the effluent monitoring program.

1.2 RADIOLOGICAL FIELD OPERATION GROUP

The responsibility of this group is to perform radiological monitoring and to implement radiological controls for all work conducted at TMI-2.

1.3 RADIOLOGICAL TRAINING GROUP

This group is responsible for the development and administration of the Radiological Training Program for Radiological Control technicians and their foremen. In addition, this group is responsible for recommending the content and monitoring the effectiveness of the Radiological Control Training Programs presented to all personnel employed at TMI Unit 2.

1.4 DOSIMETRY GROUP

The Dosimetry Group develops and administers the personnel dosimetry programs. This responsibility includes selection, maintenance, issuance, and processing of TLD's, initiation and maintenance of dosimetry records, and the maintenance of environmental dosimetry devices.

1.5 RADIOLOGICAL SUPPORT SERVICES GROUP

This group is responsible for radiological instrumentation calibration and maintenance, the respiratory fitting and training program, bioassay program, the radiological laboratory, and radiation health activities.

2.0 CURRENT ACTIVITIES

The specific activities performed by the Radiological Control Department during this reporting period are outlined in the Management Plan for TMI-2 Radiological Control Program (Appendix A) and the Quarterly Progress Report on the Management Plan (Appendix D). An overview of the activity performed by the Radiological Control Department is presented below.

2.1 INITIAL REACTOR BUILDING RE-ENTRY PROGRAM

The Radiological Technical Support group has provided technical and ALARA support since the start of this program. Contributions include analysis and recommendation of dosimetry equipment, development of protective clothing, instrumentation selection and testing (both personnel and area), and procedure review.

2.2 RADIOLOGICAL CONTROL PROCEDURE REVISION

The Radiological Technical Support group is currently coordinating a major procedure revision program, intended to develop accurate procedures, which will allow and require verbatim compliance.

2.3 DOSIMETRY PROGRAM

The Radiological Technical Support and Dosimetry group has been involved with upgrading the dosimetry program procedures and bioassay program procedures. In addition, QA programs for the TLD systems and a computerized on-line exposure-tracking program is being developed.

2.4 TRAINING PROGRAM

The Training Group has developed a qualification program for Radiological Control Technicians and their foremen. Training, in accordance with this program, is currently underway. In addition, the Training Program for all TMI-2 workers has been developed and is currently in the review circuit.

2.5 ALARA PROGRAM

The ALARA review procedure which defines the current ALARA review process is currently developed and is approximately 95% complete.

2.6 INSTRUMENTATION

Radiological instrument selection is an on-going activity. Currently, instrumentation capable of measuring the beta levels associated with the post-accident primary coolant and radiological conditions expected in the reactor building are being tested and selected, based on the results of testing programs. In addition, the instrument calibration procedures are being revised to reflect current practices, and the calibration facility has been upgraded.

2.7 RADIATION PROTECTION PLAN

The Radiation Protection Plan, which outlines the philosophies and objectives of the TMI-2 Radiological Control Program has been prepared and submitted to the NRC for approval in January 1980.

3.0 APPENDIX A - MANAGEMENT PLAN FOR TMI-2 RADIOLOGICAL CONTROL PROGRAM

3.1 INTRODUCTION

The purpose of this plan is to identify the planned corrective actions in the TMI-2 management's commitment to eliminate the identified weaknesses in the TMI-2 Radiological Control Program, and to present a schedule for implementation and completion of these corrective actions. The schedule for implementing and completing the identified corrective actions will be closely monitored by the Manager of Radiological Controls. A quarterly status report on the progress made in each of the major areas of deficiency listed below will be prepared by the Manager of Radiological Controls for presentation to the Senior Vice President, Metropolitan Edison Company. The progress made implementing this plan will be independently monitored by Quality Assurance. The auditing group's findings will be reviewed monthly with the Senior Vice President, Metropolitan Edison Company and the TMI managerial personnel assigned responsibility for the specific corrective actions.

3.2 SPECIFIC PLAN OBJECTIVES

3.2.1 ESTABLISH MANAGEMENT'S COMMITMENT TO ACHIEVING A HIGH QUALITY RADIOLOGICAL CONTROL PROGRAM AT TMI-2

A morale and attitude problem existed in the radiation safety organization. Personnel within the organization felt they did not have the authority nor the management support to stop operations in the interest of radiological safety.

3.2.1.1 CORRECTIVE ACTIONS

1. The Senior Vice President, Metropolitan Edison Company, held meetings with all of TMI's managerial and line supervisory personnel to express Met-Ed's strong commitment to achieving a high quality Radiological Control Program. He pointed out at this meeting that the Radiological Control Department is

responsible for establishing and maintaining the Radiological Control Program, which includes the stoppage of any work not being conducted in a radiologically safe manner. It is, however, the responsibility of all personnel at TMI to ensure compliance with the Radiological Control Program.

Action due date: Policy statement session - complete

2. The Radiological Control Department was restructured under a manager reporting directly to the Senior Vice President. The reason for this reorganization was to remove the Radiological Control organizations from the direct management and influence of the operations organizations, and to provide a direct means of communication between the Radiological Control Department and the top management personnel at TMI-2.

Action due date: Complete

3. A Radiation Protection Plan, outlining the philosophy, basic policies and objectives of Metropolitan Edison Company and General Public Utilities Corporation concerning the TMI Unit 2 Radiological Control Program was initially drafted on December 7, 1979, revised on January 14, 1980, and is currently awaiting final NRC approval for issuance. This plan outlines the Radiological Program, and stresses the management's commitment to a high quality Radiological Control Program.

Action due date: Two weeks after receipt of NRC action on January 14 revision

4. A Radiological Assessment group, independent of the Radiological Control Department, was formed to independently monitor TMI's progress implementing and adhering to a strong Radiological Control Program within the concept of ALARA. This group also has the authority to stop any work that is not being conducted in accordance with sound radiological work practices. This group reports it's findings directly to the Senior Vice President and the Director of Recovery.

Action due date: Completed

5. To provide additional support and direction for radiological control technicians working in the field, all technician supervisors have been assigned solely to on-the-job supervisory duties. In the past, technician supervisors were assigned to tasks such as procedure writing and assignments as procedure review committee members. These duties distracted from the superviaor's main objective; support and direction for the radiological control technicians.

Action due date: Completed

6. Supervisory and Management Development Training programs will be utilized to upgrade supervisory and management skills.

Action due date: December 1980
Responsible Individual: Director of Reliability Engineering

3.2.2 FORMALIZE THE ORGANIZATIONAL STRUCTURE FOR THE TMI UNIT - 2 RADIOLOGICAL CONTROL PROGRAM

There existed a high degree of uncertainty regarding responsibilities, functions, assignments and lines of authority within the TMI organizations.

3.2.2.1 CORRECTIVE ACTIONS

1. The manager of Radiological Control reorganized the Radiological Control Department. The department reorganization identifies the chain of command and delineates areas of responsibility within the Radiological Control Department. All contractor controlled radiological functions are included within this organization along with Met-Ed/GPU employees under the management and direction of Met-Ed/GPU supervisory personnel.

Included in this reorganization, tasks previously performed by radiological control personnel which were considered not to be essential to the Radiological Control program were reassigned to operationally oriented groups. Radiochemistry operations are now the responsibility of a Chemistry Department and decontamination operations are performed by a Decontamination Department. The Radiological Control Department activities and functions were reorganized into five groups under the direction of management level personnel as follows:

a. Radiological Technical Support: Prepares procedures used in the Radiological Control Department. Provides technical support for all groups in the Radiological Control Department. Coordinates the ALARA program.

b. Radiological Field Operations: Performs radiological monitoring and implements radiological controls in the field.

c. Radiological Training: Trains radiological control technicians and their supervisors. Oversees the radiological control training of workers.

d. Dosimetry: Implements and administers dosimetry programs.

e. Radiological Support Services: Radiation instrument calibration and repair, respirator testing, bioassay, radiological laboratory, radiation health activities.

A recruiting program is currently in progress to fill the management positions for the above five groups with personnel possessing previous supervisor skills, in an attempt to upgrade the Radiological Control Department's capabilities to supervise its people and operations. Three of these positions have been filled to date.

Action due date: Reorganization - Complete

2. In the past, operations personnel were utilized to perform radiological control functions during outages to alleviate manpower shortages in the Radiological Control Organization. Henceforth, only radiological control technicians or their foremen who have been trained in accordance with the training program addressed in this plan will be utilized to provide radiological control coverage for work at TMI-2.

   Action due date: Training to be completed by June 30, 1980
   Responsible Individual: Supervisor, Radiological Control Training

3.2.3 INCREASE THE TECHNICAL DEPTH OF THE RADIATION SAFETY PROGRAM

Professional depth and input for the Radiation Safety Program at TMI-2 was apparently lacking.

3.2.3.1 CORRECTIVE ACTIONS

1. Sufficient professional depth has been available to TMI-2 throughout the initial phases of recovery through contractor organizations, however, their activities were not adequately coordinated or managed by Met-Ed supervision. The previously described reorganization is expected to provide the desired level of coordination and management for these activities.

   In addition, a recruiting program is currently in progress to place highly skilled managerial, professional and technical personnel within the structure of the TMI-2 Radiological Control Organization. Authorization has been received to add one management level, four technical/supervisory, six operations/supervisory, eleven technical/engineering, two training instructors, and eighteen technician personnel in addition to the existing Met-Ed and contractor personnel to reinforce the Radiological Control efforts. Completion of the reinforcement program is expected prior to initiation of major evolutions inside the reactor building following initial re-entry.

Action due date: August 1, 1980
Responsible Individual: Manager of Radiological Control

3.2.4 UPGRADE THE RADIOLOGICAL CONTROL TECHNICIAN AND RADIOLOGICAL WORKER TRAINING PROGRAMS

3.2.4.1 CORRECTIVE ACTIONS

1. As part of the reorganization of TMI-2 Radiological Control group, a Radiological Control Training group reporting directly to the Manager of Radiological Control has been established. The training group formalized the training program for technicians and foremen in December and began implementation of the program in January. This training program requires formal qualification of Junior Technicians, Senior Technicians, and Foremen. It states the required minimum acceptable knowledge, understanding, practical abilities and experience standards for qualification. Additionally, qualification is based upon satisfactory performance on a written examination, demonstration of practical abilities and satisfactory performance on an oral examination covering response to abnormal situations.

   A 40-hour training course is being given to technicians prior to their written examination. A course provides refresher training on radiological fundamentals; relates these fundamentals to the nuclides, instrumentation and procedures used at TMI-2; provides "thumb rules" to allow technicians to make rapid field evaluation of the abnormal conditions; familiarizes the technicians with TMI-2 systems, expected concentrations of liquid in these systems and associated radiation levels; informs the technicians of their responsibilities and expected response in routine and abnormal situations. Included in the course are problem-solving sessions which review past abnormal radiological situations and postulated potential situations with student participation in the initial analysis of radiological data, immediate and supplementary protective and corrective actions, taking of additional radiological measurements and a review of the radiological consequences of the postulated situation. To date, 16 technicians and foremen have completed this course and satisfactorily passed their written examinations.

Oral examinations have been administered to 3 foremen and 3 technicians, to date, who have satisfactorily demonstrated their knowledge, understanding and ability to handle unusual situations. The examinations are administered by senior technical, training, and radiological control operations personnel. The examinee is presented with a situation in which radiological data is given or an unusual situation is observed. The examinee must then state his actions and assessment of the situation. Additional data is provided based on the examinee's responses. The examinee is required to provide an assessment of the radiological consequences of the occurrence, based upon the data provided. The situations which are presented relate to high airborne activity, spread of surface contamination, liquid spills, contaminated injured personnel, and unusual gamma or beta level exposure. The oral examination board then evaluates the performance, critiques the performance with the examinee, and documents the results.

Practical ability demonstrations by technicians and foremen are required to assure that they understand the requirements for, and can satisfactorily perform routine surveys and operations required of technicians. Additionally, satisfactory performance in radiological spill drills are required as practical abilities. These practical abilities are witnessed and verified by individuals who are responsible for the direction and/or review of performance of persons demonstrating the practical ability, by training personnel, or by radiological control technical support personnel.

Two instructors and one training supervisor have been hired and are expected to report to work within the next several weeks. This brings the training staff up to our projected full-time instructor manning level.

We also intend to utilize others in the Radiological control Department on a part-time basis in their areas of technical competence. Although initial qualification will be a continuing effort; all radiological control technicians and foremen will be qualified, or shall be restricted in their assignments by June 30, 1980.

Action due date: June 30, 1980
Responsible Individual: Training - Supervisor Radiological Control Training
Restriction in assignments - Supervisor Radiological Control - Field Operations

2. Radiological Safety Training for all personnel employed at TMI-2 is being performed by the Met-Ed Training Department.

The responsibility to ensure that this training meets the minimum standards necessary to perform work in a radiological safe manner in the environment which exists and which will exist in TMI Unit 2, has been assigned to the Supervisor of Radiological Control Training. He has been directed to review, to change as necessary, and to approve course material, examinations, presentations, and practical factor performance tests. Implementation of the training program is expected to begin May 1, 1980.

Action due date: May 1, 1980
Responsible Individual: Supervisor, Radiological Control Training

3. In addition to Radiological Control Training described above, special training, such as mock-up training, walk-through exercises, and detailed worker briefings will be required for major evolutions, and those tasks which may result in encountering unusual or uncertain radiological environments. The determination for the necessity of this additional training will be made by the ALARA engineering personnel, based on the review of work procedures or task definition. The actual training/briefings will be conducted by technical operations personnel in conjunction with radiological control personnel. An example of this type task is the reactor building initial entry, for which training is underway.

The guidelines for making determination of which tasks require additional training will be established by the Radiological Technical Support Branch by July 1, 1980. In the interim, this determination will be made in consultation with the Manager of the Radiological Technical Support Group.

Action due date: July 1, 1980
Individual Responsible: Manager Radiological Technical Support

3.2.5 IMPROVE RESPONSE TO AND RESOLUTION OF AUDIT FINDINGS

3.2.5.1 CORRECTIVE ACTIONS

1. The responsibility for coordination of audit finding responses has been assigned to the Radiological Technical Support Group. A procedure describing the audit response process will be formulated and implemented by March 15, 1980. This procedure will include the following:

   a. All audit findings associated with Radiological Safety and operations will be distributed to the individual assigned corrective actions, and the Radiological Technical Support Group.

b. These findings will be reviewed by the Radiological Technical Support Group and evaluated to assess the overall program performance, and to identify trends which may indicate a decline in radiological control performance. These performance deficiencies may be further indication of generic procedural, operational, technical or managerial deficiencies.

c. A commitment record, documenting the suggested corrective action, action due date, and the individual responsible for the corrective action, will be distributed to the management individuals responsible for the area in which the deficiency was observed.

d. Upon notification of corrective action completion, the Radiological Technical Support Group will perform a follow-up inspection of the area of the deficiency, to determine if the corrective action performed is acceptable, prior to closing out the commitment record.

e. A monthly status report on all open action items will be made and presented to the Manager of Radiological Controls.

Although audits performed in radiological work areas customarily categorize the deficiencies as radiological deficiencies, the cause of the deficiencies and initiation of corrective actions is not entirely the responsibility of the Radiological Control Department. Many deficiencies are an indication of poor radiological work practices. It is the responsibility of the Radiological Control Department to evaluate radiological conditions, to initiate precautionary measures, and to correct deficiencies, if possible. The responsibility of all workers and their supervisors is to ensure that all work performed in a radiological work area is performed in accordance with the established radiological procedures and the concept of ALARA.

It is the intent of this procedure to identify responsibility for radiological deficiencies, and to hold the responsible groups accountable for implementation and completion of the corrective actions.

Action due date: Procedure for audit responses: March 15, 1980
Responsible Individual: Radiological Technical Support Manager

2. All previous NRC audit findings and the latest Quality Assurance audit (November 1979) findings are currently being reviewed by the Radiological Technical Support Group to assign responsibility for corrective action, to determine acceptability of intended corrective actions, and to assign action due dates for completion of corrective actions. Commitments for corrective actions will be assigned to the responsible individuals by February 15, 1980.

   Action due date: February 15, 1980
   Responsible Individual: Radiological Technical Support Manager

3. All audit findings, other than those indicated above, will be re-audited by the Quality Assurance audit group. Any open items will be reissued as new audit findings, and items which no longer apply due to changing conditions or operations will be closed out by the Quality Assurance audit group. A report, listing the current open audit items, will be forwarded to the Manager of Radiological Control.

   Action due date: March 1, 1980
   Responsible Individual: Supervisor, Quality Assurance Audit

4. An "in house" surveillance program is currently being developed. To identify weaknesses in radiological work practices in a more timely fashion and reduce the number of findings resulting from formal audits. It is the intent of this program to document all deficiencies noted, no matter how apparently insignificant, and whether or not they are immediately corrected, in order to monitor radiological work practices and to identify weaknesses in the Radiological Control Program. These deficiencies will be documented by any worker observing the condition. Copies of the deficiency reports will be distributed to the Radiological Technical Support Group for review and trend analysis. Recurring minor deficiencies in a particular area or category may indicate a large underlying problem area which necessitates identification on the commitment system described below.

   Action due date: April 15, 1980
   Responsible Individual: Radiological Technical Support Manager

3.2.6 REVISE AND IMPLEMENT PROCEDURES WHICH WILL ENSURE STRICT, VERBATIM COMPLIANCE, AND REVISE THE PROCEDURE REVIEW PRACTICES TO EXPEDITE IMPLEMENTATION OF RADIOLOGICAL CONTROL PROCEDURES.

3.2.6.1 CORRECTIVE ACTIONS

1. The format for all Radiological Control associated procedures is currently being restructured to achieve verbatim compliance. These procedures will be incorporated in a Radiological Control Procedure Manual, separate from the TMI site procedures. The revision for all radiological control procedures is estimated to require six months of work. A priority list for procedure revision has been

compiled, with procedures important to the continuing work effort given the highest priority. Following revision, all procedures will be field tested prior to formal implementation to ensure vertabim compliance is possible. Revisions to existing routine procedures applicable to the current recovery operations have been initiated. Revisions to the following procedures are expected to be issued, following field testing and training in their use, by April 1, 1980:

a. RWP Use,

b. Investigative Reports,

c. ALARA Review,

d. Administrative Procedure,

e. Administrative Exposure Guidelines.

All current radiological control procedures are expected to be revised, as required, and implemented following field testa and training in their use by December 1, 1980.

Action due date: Initial Revisions - April 1, 1980; Radiological Control Procedure Manual - December 1, 1980

Responsible Individual: Radiological Technical Support Manager

2. Action sign off steps will be added to all work procedures for work on major evolutions during the procedure review performed by ALARA engineers. The purpose of these sign-off atepa is to ensure a responsible individual verifies by signature that the radiological safety requirements have been satisfied, prior to continuing with the work

evolution. This format will be used for the Reactor Building re-entry procedure, and a procedure defining this practice and establishing the criteria for its use will be implemented by August 1, 1980.

Action due date: Procedure implementation - August 1, 1980
Responsible Individual: Radiological Technical Support Manager

3. The current procedure review circuit employed at TMI-2 prohibits the expedient implementation of procedures and procedure change. A revision to the TMI-2 Technical Specifications is currently being discussed with the NRC to expedite the review process for radiological control procedures, while maintaining compliance with the review requirements.

   Action due date: Pending resolution by the NRC
   Responsible Individual: TMI-2 Licensing Supervisor

3.2.7 IMPROVE THE EXTERNAL PERSONNEL DOSIMETRY PROGRAM.

TLD systems currently available to the industry do not adequately respond to energetic beta radiations that may be encountered during expanded recovery operations. The current quality assurance program does not include comparisons of results obtained from outside agencies; the present system of radiation exposure management does not allow the tracking of exposures received for specific work evolutions and there is insufficient technical expertise within the dosimetry group to permit comprehensive evaluations.

3.2.7.1 CORRECTIVE ACTIONS

1. Beta radiation associated with post accident primary coolant at TMI-2 presents a complication to personnel exposure monitoring. At present, there is no known, commercially available system in use within the industry, that possesses the capability of measuring the beta radiation exposures with the high degree

of accuracy desired by Met-Ed. At present, dosimetry systems under development and modifications to the presently-used Harshaw TLD system are being evaluated to determine their ability to more precisely measure beta exposures. In addition to the accuracy factor, these systems are being evaluated to determine their compatability with the existing reading equipment, in order to reduce the cost of system changeovers, if possible. All test data and cost study information, resulting from dosimetry system evaluations will be available, in time to permit a decision on system selection by July 1, 1980. System implementation will be accomplished by December 1, 1980.

Tests are currently being performed to determine the ability of existing dosimetry equipment to accurately monitor the exposures received by personnel performing the initial Reactor Building re-entry. The results of this analysis and subsequent recommendations will be made by February 15, 1980.

Beta radiation exposure determinations, made to date for personnel entering high beta radiation fields, have not relied solely on the existing TLD system results, due to known inaccuracies associated with beta radiation detection. Evaluations are performed by professional Health Physics personnel.

Action due date: Reactor building recommendations - February 15, 1980

System Modification Implementation - December 1, 1980

Responsible Individuals: Radiological Technical Support Manager/Supervisor, Dosimetry

2. Although technical expertise has been available to TMI-2, in the form of contracted advisors present at TMI and support agencies outside TMI, coordination and direction of this expertise was not achieved. Coordination of these efforts is assigned as the responsibility of the Radiological Technical Support Group. This responsibility will be documented in the Organization and Responsibilities Chapter of the Radiological Control Procedures Manual, scheduled for completion by February 1980. All dose assessments for non-routine exposure evaluations will be reviewed by the Radiological Technical Support Group.

   Action due date: February 1980
   Responsible Individual: Radiological Technical Support Manager

3. The current TLD Quality Assurance program is being expanded to include comparison with outside agencies and this program will be issued by April 30, 1980.

   Action due date: April 30, 1980
   Responsible Individual: Supervisor, Dosimetry

4. The current computerized exposure record system is designed to maintain accountability of personnel exposures, however, it does not provide a means of tracking exposures by the tasks performed. A study is presently being conducted to determine the best method to satisfy the above goal. By April 1, 1980, a system capable of tracking personnel exposures by work groups and by major tasks will be implemented. By December 31, 1980, a system capable of tracking exposures by specific individual tasks will be implemented.

   Action due date: Exposure tracking by work group and major task - April 1, 1980

   Exposure tracking by specific tasks - December 1, 1980

Responsible Individual: Supervisor, Dosimetry

3.2.8 IMPROVE THE INTERNAL DOSIMETRY PROGRAM

3.2.8.1 CORRECTIVE ACTION

1. Through contractor support, technical expertise has been available and utilized to evaluate results obtained from the internal exposure monitoring program. These efforts will be coordinated and managed by the Radiological Control Department reorganization described earlier in this plan.

   Action due date: Complete

2. The current bioassay program is being revised to formalize the basis for bioassays under the routine conditions, the follow-up assays of skin contamination or internal deposition, and evaluations for Sr-90 depositions. All dose assessments and evaluations of internally deposited radionuclides will be reviewed by the Radiological Technical Support Group.

   Action due date: April 1, 1980
   Responsible Individual: Supervisor, Dosimetry

3.2.9 UPGRADE THE RADIATION PROTECTION INSTRUMENTATION PROGRAM

3.2.9.1 CORRECTIVE ACTION

1. All instrumentation selection, installation, calibration and maintenance currently performed by contractor technicians will be coordinated and managed by the Radiological Control Department organization aa identified earlier in thia plan.

   Action due date: Complete

2. Instrumentation evaluation and subsequent selection is continuing at TMI Unit 2, to ensure that the most accurate and reliable instrumentation available is employed in the Radiological Control Program. As advances are made in instrumentation, or as the current instrumentation proves to be inadequate for conditions encountered, the search for, evaluation of, and subsequent selection of the most reliable instrumentation available to the industry will be made. There are currently two areas in which the current instrumentation used at TMI-2 proves to be less than desirable: high level energetic beta fields associated with undiluted post-accident primary coolant, and radiation measurements taken within the Kr-85 environment of the Reactor Building.

The search for a beta instrument capable of accurately measuring the beta fields associated with the post-accident primary coolant, is a difficult task complicated by the fact that accurate measurement of beta radiation has been neglected in the field of Health Physics. The Department of Energy and its contractors at the Idaho National Engineering Laboratory have been focusing on this problem for 2 to 3 years, and consequently have made significant steps towards improving the capability to monitor for, and to calibrate instruments for beta radiation. The Radiological Technical Support Group sent an instrumentation team to the Idaho National Engineering Laboratory during January, with newly-developed high-range and existing beta instruments, to perform calibration and comparative performance tests. The information obtained will be used to make a selection of the most suitable instrument for measuring beta radiation fields.

The second condition, in which the currently-used instrumentation proves to be ineffective, is radiation measurements performed within the Kr-85 environment contained in the Reactor Building as a result of the post-accident conditions. The Kr-85 apparently migrates through the instrumentation sealant materials, producing unreliable measurements. A search is currently being conducted by the Radiological Technical Support Group for instrumentation or sealant systems capable of producing reliable measurements under the conditions described.

Instrumentation selection for the Reactor Building re-entry program will be made, following resolution of the above-indicated problem areas, prior to March 1, 1980.

Action due date: Reactor Building Entry Instrumentation - March 1, 1980

Responsible Individual: Radiological Technical Support Manager

3. The Radiological Technical Support Branch is currently preparing a Quality Assurance Program for instrument calibration. This program is considered necessary to ensure that all portable instruments used at TMI-2 are properly calibrated. This program will cover calibration procedures performed at TMI and at off-site facilities.

   Action due date: July 1, 1980
   Responsible Individual: Radiological Technical Support Manager

4. The TMI instrument maintenance and calibration facility has been upgraded, and is considered adequate to meet our current needs. The Radiological Technical Support Group has presented design criteria to Bechtel Corporation, for design and construction of a new facility. This facility should be operational prior to expanding decontamination activities into the Reactor Building.

Action due date: December 31, 1980

Responsible Individual: Supervisor, Radiological Support Services

5. Rapid and accurate assessment of radiation and contamination levels in TMI-2 areas is essential to the Radiological Control Program. The present counting facilities for air activity and surface contamination levels are inadequate. There is a need for improved contamination analysis, isotopic analysis, and low energy beta analysis capabilities. Recommendations are being prepared by the Radiological Technical Support Group to upgrade the existing Health Physics Counting Laboratory.

   Action due date: Recommendations - February 15, 1980

   Implementation - June 1, 1980

   Responsible Individual: Radiological Technical Support Manager

6. In the past, area air quality measurements were performed with air sampling equipment installed in the filtered ventilation system. Although these units are still in use, area monitoring equipment has been installed throughout the TMI-2 auxillary and fuel handling buildings. This equipment is used to constantly monitor the airborne activity levels in the general work spaces and is placed in specific work areas for operations which have a potential for generation of airborne activity.

   Action due date: Complete

7. Previous portable air sampling practices for operations in radiological work areas were considered to be inadequate. Portable air sampling frequencies have been increased. Portable air samples are taken at the start,

during, and after work activities which are identified as having potential for generating airborne contamination.

Action due date: Complete

8. Radioiodine sampling and analysis performed prior to the accident at TMI Unit 2 was considered inadequate. Immediately following the accident, TMI-2's capabilities to monitor for radioiodine increased to an acceptable level. Although TMI-2 has maintained the capability of sampling and analyzing radioiodines, this practice is not performed routinely at present because radioiodines are no longer limiting factors for recovery operations.

Action due date: Complete

9. Survey frequencies followed under pre-accident conditions at TMI were considered to be inadequate. Survey frequencies increased during post-accident conditions. A new schedule for survey frequencies has been drafted in procedural format and is expected to be implemented by February 1, 1980.

Action due date: February 1, 1980
Responsible Individual: Supervisor, Radiological Field Operations

5.2.10 IMPROVE TMI-2 RADIOACTIVE MATERIAL SHIPPING AND LABELING PROCEDURE.

1. All procedures associated with the packaging, handling, shipping, and receipt of radioactive material at TMI-2 have been revised. In addition, the position of Radioactive Material Coordinator has been created and filled. The responsibility of this position is to ensure all shipments to or from TMI-2 are in accordance with the regulatory agency's requirements.

Action due date: Complete

2. Curie estimates performed on shipments of radioactive material during the pre-accident conditions were not consistent. All curie estimates performed since the accident have been coordinated by the Radioactive Material Coordinator, with the assistance of the Radiological Engineers currently assigned to Radiological Technical Support Group. The Radiological Technical Support Group is currently preparing guidelines for the performance of curie estimates by radiological control technicians for commonly-used containers (i.e. LSA boxes, 55 gal. drums), and the requirement to consult the Radiological Technical Support Group for other containers.

   Action due date: April 1, 1980
   Responsible Individual: Radiological Technical Support Manager

3.2.11 IMPROVE DECONTAMINATION PROCEDURES FOR EQUIPMENT AND TOOLS.

3.2.11.1 CORRECTIVE ACTIONS

During pre-accident conditions at TMI-2, decontamination operations were performed by the already overburdened radiological control personnel. This practice resulted in a backlog of work in the decontamination areas. This backlog resulted in tools and equipment being disposed of as waste, and in personnel performing their own decontamination operations. The decontamination operation is now being performed by a separate decontamination group, reporting to operations. No item is decontaminated unless it is tagged with the name and extension number of the individual to be contacted after decontamination. All items are surveyed for release by Radiological Control Technicians.

Action due date: Complete

3. It is the responsibility of all Met-Ed/GPU and contractor supervision to ensure that radiation exposures received by individuals reporting to them are as low as reasonably achievable. The reports, based on the computer returns described above and issued by the Radiological Technical Support group, should be used as an aid in tracking personnel exposures and identifying operations which may need ALARA engineering reviews to reduce exposures at TMI-2.

3.2.13 RADIOLOGICAL CONTROL PROGRAM FAMILIARIZATION

In order to achieve a strong Radiological Control Program, personnel must be made accountable for the actions they take. They must understand their responsibilities and expectations in achieving a sound radiological control program. Upper management, as indicated earlier in this plan, is committed to achieve a strong Radiological Control Program. The following corrective actions are being implemented to ensure that everyone at TMI understands their responsibilities to this comment.

3.2.13.1 CORRECTIVE ACTIONS

1. As indicated earlier in this plan, responsibility for corrective actions, performed to satisfy audit findings, will be delegated to supervising personnel responsible for the area or operations in which the deficiency occurred.

   Action due date: Continuing

   Responsible Individual: Radiological Technical Support Manager

2. Action items will be inserted in operational work procedures, to ensure proper attention is given to radiological considerations, prior to the performance of work steps. These action items shall be signed off by personnel responsible for, and cognizant of the work evolution, verifying the conditions have been satisfied prior to initiating the work step.

Action due date: August 1, 1980
Responaible Individual: Radiological Technical
Support Manager

3. Critiques will be conducted for unusual radiological occurrences, as defined by guidelines prepared by the Radiological Technical Support Group. The purpose of these critiques is to identify the cause and to determine corrective actions to prevent reccurrence, including disciplinary action if deemed necessary. A procedure presenting the guidelines and criteria for conducting a critique will be prepared by the Radiological Technical Support Group and incorporated in the Radiological Control Procedure Manual by December 1, 1980.

Action due date: December 1, 1980
Responsible Individual: Radiological Technical Support
Manager

4.0 APPENDIX B - MANAGEMENT PLAN SCHEDULE

4.1 PURPOSE AND SUMMARY

The following compendium describes the management plan corrective actions for improvements in the TMI-2 Radiological Control Program. Target dates for completion of the identified corrective actions are presented in the form of a bar chart, Figure 5-1. Proposed starting dates for major recovery activities, are included on the bar chart to present a comparative view of the program implementations, with the start of the major recovery actions. It should be noted that all programs are scheduled for completion prior to starting the Reactor Building decontamination operations. It is not considered necessary to complete all of the corrective actions identified in the management plan prior to the Reactor Building re-entry. This major recovery task has received a significant amount of managerial and technical attention since the start of the program. Radiological engineering (ALARA) has been following the procedure writing, and assisting in overviewing performance of special tests and evaluations, to analyze exposures, protective clothing, respiratory protection equipment, dosimetry equipment, and expected environment and radiological conditions for the personnel making the initial re-entry. All re-entry team members have undergone intensive medical examinations and special intensive training on equipment usage, operations to be performed, and the radiological conditions, requirements, and limits associated with the re-entry. Team members will also undergo intensive physical fitness training and practical factor training, utilizing the equipment selected for re-entry. The radiological control personnel selected for providing coverage during the re-entry, will receive special training prior to performing the re-entry.

MANAGEMENT PLAN SCHEDULE

1980: JAN FEB MAR APR MAY JUN JUL AUG SEP OCT NOV DEC — 1981: JAN FEB MAR APR MAY JUN JUL AUG

RESPONSIBLE INDIVIDUAL	ACTIVITY	REFERENCE
MANAGER OF RADIOLOGICAL CONTROLS	RADIATION PROTECTION PLAN	3.2.1.1.3
	DEPARTMENT STAFFING	3.2.3.1.1
TRAINING SUPERVISOR	COMPLETE RADIATION WORKER TRAINING	3.2.4.1.1
	COMPLETE QUALIFICATION TRAINING FOR CURRENT TECHS. & FOREMEN	3.2.4.1.1
QA AUDIT SUPERVISOR	RE-AUDIT OF DELINQUENT FINDING RESPONSES	3.2.5.1.3
OPERATIONS SUPERVISOR	REVISE SURVEY FREQUENCIES	3.2.9.1.9
RADIOLOGICAL TECHNICAL SUPPORT	ORGANIZATION & RESPONSIBILITY CHAPTER OF RAD CONTROL PROCEDURES MANUAL	3.2.[illegible].1.2
	REACTOR BUILDING RE-ENTRY TLD STUDY	3.2.7.1.1
	NRC & QA AUDIT REVIEW	3.2.5.1.2
	REVIEW HP COUNTING LAB NEEDS - MAKE RECOMMENDATIONS	3.2.9.1.5
	EVALUATE INSTRUMENTATION FOR R. B. RE-ENTRY	3.2.9.1.2
	AUDIT RESPONSE PROCEDURE	3.2.5.1.1
	INITIAL PROCEDURE REVISIONS	3.2.6.1.1
	CURIE CONTENT ESTIMATE GUIDE	3.2.10.1.2
	INTERNAL SURVEILLANCE PROGRAM	3.2.5.1.4
	H.P. COUNTING LAB REVISIONS	3.2.9.1.5
	MOCK-UP TRAINING GUIDELINES	3.2.4.1.3
	Q.A. PROGRAM FOR INSTRUMENT CALIBRATION	3.2.9.1.3
	PROGRAM FOR SIGN-OFF STEPS IN OPERATIONS PROCEDURES	3.2.6.1.2
	ALARA PROGRAM - EXPOSURE TRACKING	3.2.12.1.[illegible]
	RADIOLOGICAL CONTROL PROCEDURE MANUAL	3.2.6.1.1
	CRITIQUE GUIDELINES	3.2.13.1.[illegible]
	TLD SYSTEM MODIFICATIONS	3.2.[illegible].1.1
RADIOLOGICAL SUPPORT SERVICES MANAGER	INSTRUMENT CALIBRATION FACILITY	3.2.9.1.4
DOSIMETRY SUPERVISOR	BIOASSAY PROGRAM REVISION	3.2.[illegible].1.2
	COMPUTERIZED EXPOSURE TRACKING BY WORK GROUP & MAJOR TASK	3.2.7.1.4
	Q.A. PROGRAM FOR TLD'S	3.2.7.1.3
	EXPOSURE TRACKING FOR SPECIFIC JOBS	3.2.[illegible].1.4

Figure 5-1

5.0 APPENDIX C - MATRIX RELATING NUS AND NRC AUDIT FINDINGS WITH MANAGEMENT PLAN

5.1 PURPOSE AND SUMMARY

This appendix presents a matrix which illustrates how the corrective actions stated in this management plan address the NUS Corporation and the NRC Special Panel audit findings. The NUS Corporation audit findings are addressed in Table 5-1, and the NRC Special Panel findings are addressed in Table 5-2. The corrective actions listed adjacent to each major finding relate to the numbering system used to identify each objective presented in this management plan (Appendix A).

TABLE 5-1. NUS AUDIT FINDINGS

AUDIT FINDINGS/RECOMMENDATIONS	MANAGEMENT PLAN CORRECTIVE ACTION PARAGRAPH
1. Organization	
A. Separate health physics from chemistry group	3.2.2.1-1.
B. Assign an engineer to radwaste packaging	3.2.10.1-1.
C. Remove radwaste functions from health physics	3.2.2.1-1.
D. Remove decontamination functions from health physics	3.2.2.1-1.
E. Remove RWP filing from health physics	3.2.2.1-1.
F. Assign dosimetry responsibilities to an analyst	3.2.2.1-1.
G. Auxiliary operators should not be used as H.P. Technicians	3.2.2.1-2.
2. Health Physics Credibility	
A. Inadequate technician training	3.2.4.1-1.
B. Decisions made by health physics technicians are overridden	3.2.1.1-5.
3. Communications	
A. Operations performed without health physics notification	3.2.1.1-3.,-4.; 3.2.6.1-1.,-2.
B. Information not being passed down to Technicians	3.2.1.1-5.; 3.2.4.1-1.
C. Technicians not notified of procedure changes	3.2.1.1-5.; 3.2.4.1-1.
D. Procedures are not referred to during performance of work	3.2.6.1-1., -2.
4. Personnel Dosimetry	
A. TLD calibration discrepancies	3.2.7.1-3.
B. Improper review of personnel dosimetry records	3.2.2.1-1.; 3.2.7.1-4.
C. TLD's should be processed by same technician	3.2.2.1-1.
5. Radiological Surveillance	
A. Continuous airborne radioactivity monitoring from ventilation plenums	3.2.9.1-6.
B. Schedules for portable air sampling should be reevaluated	3.2.9.1-7.
C. No program for radioiodine sampling	3.2.9.1-8.
D. Use portable CAM's for monitoring specific work areas	3.2.9.1-6.
E. Schedules for radiation and contamination should be reevaluated	3.2.9.1-9.
6. Tool and Equipment Control	
A. Individuals perform their own decontamination	3.2.11.1
B. Many items left for decontamination are disposed of as solid radioactive waste	3.2.11.1
7. Radioactive Shipment	
A. Improper packaging	3.2.10.1-1.
B. Inconsistent curie content estimates	3.2.10.1-2.

TABLE 5-2. NRC SPECIAL PANEL FINDINGS

AUDIT FINDINGS/RECOMMENDATIONS	MANAGEMENT PLAN CORRECTIVE ACTION PARAGRAPH
1. Management Deficiencies	
A. Management commitment in support of Radiation Safety Program	3.2.1.1 , 3.2.1.2
B. Organizational structure	3.2.1.2 , 3.2.2.1
C. Technical depth of Radiation Safety program	3.2.3.1-1.
D. Training	3.2.4.1-1. , -2. , -3.
E. Resolution of audit findings	3.2.5.1-1. , -2. , -3. -4.
F. Preparation and implementation of procedures	3.2.6.1-1. , -2. , -3.
2. Technical Deficiencies	
A. External personnel dosimetry	3.2.7.1-1. , -2. , -3. , -4.
B. Internal dosimetry	3.2.8.1-1. , -2.
C. Instrument program	3.2.9.1-1. , -2. , -3. , -4. , -5.
D. Radiation control	3.2.12.1-1. , -2. , -3.

6.0 APPENDIX D - CURRENT PROGRESS RELATED TO THE MANAGEMENT PLAN.

6.1 PURPOSE AND SUMMARY

This appendix summarizes action initiated, in progress, or completed since the last progress report was issued. This appendix will primarily address the open action items committed for completion by April, 1980, and will be presented in the chronological and alpha-numeric sequence used in the Management Plan (Appendix A).

6.2 CORRECTIVE ACTION STATUS FOR COMMITMENTS DUE ON APRIL 1, 1980

1. Action Item 3.2.1.1.3: The Radiation Protection Plan for TMI Unit 2 is still awaiting NRC approval. This plan is expected to be issued two weeks after receipt of NRC approval.
2. Action Item 3.2.5.1.1: This procedure was submitted to the review and approval cycle by the committed date of April 1, 1980. This procedure is expected to be implemented following final approval.
3. Action Item 3.2.5.1.1: A list of the open audit findings was received by the Radiological Technical Support Group. Assignments for corrective action and action due dates have been assigned to all of these items. This action is complete.
4. Action Item 3.2.6.1.1: The committed corrective action was to complete revision on five procedures prior to April 1, 1980, and to completely revise the remaining Radiological Control Procedures by December 1, 1980. The status of the initial five procedures is as follows:
   a. RWP Use Procedure: This procedure completed the review cycle and all comments were incorporated by the committed date of April 1, 1980. This procedure will be implemented following final approval.

b. Investigative Report Procedure: This procedure completed the "in-house" review cycle and was submitted to the NRC on March 27, 1980. This proce'ure is expected to be implemented within two weeks following the NRC review.

c. ALARA Review Procedure: This procedure will be submitted to the review and approval cycle by April 30, 1980.

d. Administrative Procedure: This procedure has been submitted to the "in-house" review cycle; all comments generated were incorporated by the committed date of April 1, 1980. This procedure will be implemented following final approval.

e. Administrative Exposure Guidelines: This procedure was submitted to the "in-house" review cycle; all comments generated were incorporated by April 1, 1980. This procedure will be implemented following final approval.

5. Action Item 3.2.7.1.1: The committed action was to make recommendations on personnel dosimetry for use during the reactor building re-entry prior to February 15, 1980, and perform system modification by December 1, 1980. A report, describing the results of test performed in the containment building through penetration number 626 using Harshaw TLD's, self-reading pocket dosimeters and film badges, was issued prior to January 15, 1980. This report also made preliminary recommendations on TLD usage for the reactor building re-entry program. A letter, describing the specific dosimetry equipment and its placement, was issued on March 7, 1980. The format described by this letter was issued during the dress rehearsals of the reactor building re-entry.

6. Action Item 3.2.7.1.2: The organization and responsibilities chapter of the Radiological Control Procedures Manual was reviewed by PORC on March 6, 1980, and is currently awaiting approval.

7. Action Item: 3.2.7.1.4: The computer system used for personnel dosimetry is currently undergoing a major conversion from a data bank system to an on line retrieval system. This system will allow immediate input and retrieval of dosimetry information via installed CRT's. Due to this system change, program modifications to the existing system are behind schedule, however, the commitment established in the Management Plan is expected to be achieved on schedule.

8. Action Item 3.2.8.1.2: The bioassay program procedure is currently being written. The draft completion date for this procedure has slipped to April 30, 1980, with implementation expected within 30 days of submittal to the review and approval cycle.

9. Action 3.2.9.1.2: The testing of instrumentation within the reactor building atmosphere, was delayed unit NRC approval of the operating procedures for the conductance of tests, using the R626 penetration glove box. The instrumentation tests were performed during the week ending March 8, 1980.

10. Action Item 3.2.9.1.5: Recommendations for counting facility improvements have been submitted, and are currently being reviewed by the Manager of Radiological Technical Support, prior to initiation of purchase requisitions.

11. Action Item 3.2.9.1.9: A memo was issued in January, establishing the survey frequency schedule currently in use. A survey procedure was submitted for review and approval by the committed date of April 1, 1980. This procedure will be implemented following final approval.

12. Action Item 3.2.10.1.2: Guidelines for the performance of curie estimations were presented to PORC in the form of a procedure change request on February 29, 1980. As of this date, PORC has not reviewed this procedure. This procedure change is expected to be implemented by May 1, 1980, as scheduled.

6.3 CORRECTIVE ACTION STATUS FOR COMMITMENTS DUE DURING THE SECOND QUARTER 1980

1. Action Item 3.2.2.1.2: The committed action is to complete radiological controls training for all current radiological control technicians and foremen. The classroom phase of this training program, which includes a written examination, was completed for all of the committed technicians and foremen as of March 14, 1980. By the commitment date of June 30, 1980, the remaining phases of the qualification program will be completed.

2. Action Item 3.2.4.1.1: The committed action for this item was the same as the action required for Item No. 2a above.

3. Action Item 3.2.4.1.2: The procedure for radiological control training for all personnel employed at TMI-2 has been prepared by the Supervisor of Radiological Control Training, reviewed within the Radiological Control Department, and returned with comments which are currently being resolved. This procedure is expected to be issued before the commitment date of May 1, 1980.

4. Action Item 3.2.5.1.4: A procedure describing the "in house" surveillance program has been prepared, submitted to PORC, and returned with comments which are currently being resolved. This procedure will be re-submitted for final review and approval by the April 15, 1980, commitment date.

5. Action Item 3.2.7.1.3: The QA program for the TLD system is currently being prepared by the Radiological Technical Support Group. This procedure will be submitted for review and approval by the committed date of April 30, 1980.

6.4 ADDITIONAL ACTIVITY IN PROGRESS

1. Action Item 3.2.3.1.1: The recruiting program to place highly skilled managerial, professional and technical personnel within the TMI-2 Radiological Control Department is a current and continuing activity. To date, there are 24 individuals with college degrees, and 3 individuals with Health Physics certifications in the Radiological Control Program.

   In addition, three technician foremen, two training instructors, one training administrator, one training supervisor, two engineering supervisors, four engineers, one dosimetry supervisor, one support services supervisor, three radiological control technicians, and all but one management level positions have been filled.

2. Action Item 3.2.6.1.2: Action sign off steps have been added to the Reactor Building air lock and initial entry procedures, to ensure precautions and requirements have been satisfied prior to continuing with the action. The procedure for TMI-2 wide use has not been developed as of this date. The commitment date for this action is August 1, 1980.

3. Action Item 3.2.6.1.3: A revision to the TMI-2 technical specifications, which would provide appropriate and timely review and approval for radiological control procedures, has been submitted to the NRC, and is currently awaiting approval.

4. Action Item 3.2.9.1.3: The Radiological Technical Support group is developing a QA program for instrument calibration. This program is expected to be implemented by the commitment date of July 1, 1980.

5. Action Item 3.2.9.1.4: The TMI instrument calibration and maintenance facility has been upgraded as indicated

in the Management Plan; construction of a new facility is no longer necessary, since the facility presently in use is considered to be adequate for the present and the immediate future.

6. Action Item 3.2.13.1.3: Critiques for unusual radiological occurrences are currently being conducted with the Manager of Radiological Control, the individual(s) involved and their supervision, as a minimum, to determine the cause and to initiate corrective actions. The procedure formalizing this practice has not been developed, however, it is expected to be implemented on schedule.

SECTION 6
SPECIAL PROJECTS

1.0 SCOPE

Special Projects is responsible for accomplishing a cleanup of the Reactor Building atmosphere and for conducting the initial entries into the Reactor Building. To support these activities, Special Projects conducts Reactor Building air samples and measurements and experiments designed to measure radiation and contamination levels in the Reactor Building.

Special Projects' tasks and task objectives are:

1. Hydrogen control system modifications - make changes to the Reactor Building hydrogen control system to allow use of this system for purging the Reactor Building.
2. Ante-Room modifications - make changes to the ante-room area around Reactor Building personnel airlock number 2 to all use of this area for the initial Reactor Building entries.
3. Airborne activity samples - determine the particulate, gaseous and iodine activity levels of the Reactor Building atmosphere.
4. Gamma radiation readings through the equipment hatch - determine the isotopic identity and magnitude of plate-out on the 305' elevation.
5. Gamma radiation readings through the inner flange of penetration R605 (approximately 2 feet above the sump water level, near the basement of the Reactor Building) - determine sump level and specific activity of the contamination in the sump.

6. Sump water sample from penetration R401 (approximately 2 feet above the sump water level) - perform an activity analysis of the water.

7. Gamma radiation readings through the inner metal flange of penetration R626 (at the 347' elevation approximately 11 feet above the Reactor Building operating floor) - determine general area radiation levels and determine the isotopic identity and magnitude of plateout on the 347' elevation operating floor.

8. Radiation mapping of the number 2 personnel airlock - determine airborne activity level and radiation reading inside the airlock.

9. Analysis of the hydrogen recombiner inlet spool piece - determine what plateout exists on the spool piece as a result of the several days of flow through the hydrogen recombiner which occurred within the first month after the accident.

10. Remote TV camera and radiation surveys through penetration R626 - obtain an initial visual assessment of the damage that may have been done by the accident and obtain the first <u>direct</u> radiation measurement readings inside the building.

11. Airlock entry - obtain better information on the 305' elevation radiation levels and the 305' elevation plate-out source.

2.0 CURRENT ACTIVITIES

2.1 HYDROGEN CONTROL SYSTEM SUPPORT

All modifications of the hydrogen control system required to support Reactor Building purge were completed on March 15. Final system testing began on March 31. All initial operator training to support the Reactor Building purge was completed on March 4. Final training can only be completed when NRC approval of the purge procedure is obtained.

2.2 REACTOR BUILDING ENTRY SUPPORT

All modifications to the ante-room area in support of Reactor Building entry were completed on March 20. Final estimates of dose rates expected in the Reactor Building have been calculated and are shown in Tables 6-1 and 6-2. Training of the initial entry crew and testing of equipment to be used during the initial entry were completed on March 20. This training included dry run/rehearsal of the entry in the TMI Unit 1 Reactor Building. All equipment tested satisfactorily. A request for permission to enter the Reactor Building was submitted to the NRC on March 17.

2.3 REACTOR BUILDING PERSONNEL AIRLOCK RADIATION SURVEY

Entry into and survey of the Reactor Building personnel airlock number 2 was completed on March 14. Prior to airlock entry, airborne activity was 3×10^{-3} μ Ci/ml krypton 85 and 2×10^{-9} μ Ci/ml particulate. This activity was purged into the plant ventilation system. The maximum surface contamination level found in the airlock was 460 dpm/100 cm^2. General area radiation levels at the inner door of the airlock were 40 mR/hr. Calculations of general area dose rate (based on inner airlock door readings) on the 305' elevation of the Reactor Building, and final analysis of the Ge(Li) scan of the inner door have not yet been completed.

2.4 RADIATION INSTRUMENT AND WEARING APPAREL STUDIES

Experiments were conducted in penetration R626 to determine the response of various gamma and beta/gamma survey instruments in a krypton atmosphere. Based on these experiments, the instruments to be used by the Reactor Building entry team are a teletector (gamma detection) and an RO-7 (beta detection). Experiments were also conducted to determine the rate of krypton diffusion through a Viking dry suit (to be worn during initial Reactor Building entry). The experiment showed that less than 10% diffusion can be expected during the twenty minute entry into the Reactor Building.

TABLE 6-1. WHOLE BODY DOSE RATES (a)

LOCATION	WITHOUT PURGE OF REACTOR BUILDING (RAD/HR)	WITH PURGE OF REACTOR BUILDING (RAD/HR)
305' ELEVATION		
- Krypton	0.900	—
- Plateout	0.2	0.2
- Sump Water	1.500	1.5
TOTAL	2.6	1.7
347' ELEVATION		
- Krypton	1.2	—
- Plateout	0.4	0.4
TOTAL	1.6	0.4
STAIR NUMBERS 1 and 2		
- Krypton	1.15	—
- Plateout	0.2	0.2
- Sump Water	9.0	9.0
TOTAL	10.35	9.2
AIRLOCK (DURING ENTRY)		
- Krypton	0.941	0
ANTE-ROOM (DURING EXIT)		
- Krypton	0.101	0

(a) General Area Only- Does not Include Hot Spots

TABLE 6-2 BETA SKIN DOSE RATES (a)

LOCATION	WITHOUT PURGE OF REACTOR BUILDING (RAD/HR)	WITH PURGE OF REACTOR BUILDING (RAD/HR)
305' ELEVATION		
- Krypton	9.0	0
- Plateout	1.0	1.0
TOTAL	10.0	1.0
347' ELEVATION		
- Krypton	9.0	0
- Plateout	1.5	1.5
TOTAL	10.5	1.5
STAIR NUMBERS 1 and 2		
- Krypton	9.0	0
- Plateout	1.0	1.0
TOTAL	10.0	1.0

(a) General Area Only- Does Not Include Hot Spots

SECTION 7
ENVIRONMENTAL MONITORING

1.0 SCOPE

The stated objective of the Environment Impact Assesment Group is to establish and maintain environmental surveillance programs (radiological and nonradiological) necessary to minimize effects on the general population and the environment due to TMI-2 recovery activities. Specifically, the following programs apply:

1. Determine procedures and technical specifications necessary to monitor and maintain effluent concentrations as low as practicable in accordance with existing regulations.

2. Identify and monitor critical radionuclides and their potential pathways to the population.

3. Report results of environmental surveillance in accordance with technical specifications, federal and state regulations and permits and company agreements.

4. Review, follow and upgrade the existing surveillance programs.

5. Specify radioactivity source terms.

6. Coordinate environmental surveillance activities.

2.0 CURRENT ACTIVITIES

2.1 ENVIRONMENTAL IMPACT STATEMENTS

2.1.1 SYSTEM REVIEWS

Engineering systems designed for TMI-2 recovery operation, that warrant assessment of potential off-site impacts, were reviewed and commented on as needed. The review process is intended to assure system compliance with unit Technical Specification as well as regulatory statutes, with a stated overall purpose of protecting the health and safety of the general public.

During this period environmental reviews were prepared for the following recovery systems:

SYSTEM	PRIMARY AREAS OF REVIEW
Submerged Demineralization System (SDS)	Nonradiological environmental impacts Off-site dose calculations.
Interim Waste Storage Facility	Potential off-site radiation dose levels.
Evaporator Crystallizer System	Nonradiological environmental impacts. Potential off-site radiation dose levels.
Process Water Storage Tanks	Potential off-site radiation levels.
Groundwater Monitoring	Groundwater radionuclide content.

2.2 ENVIRONMENTAL MONITORING PROGRAMS

Radiation environmental monitoring programs were upgraded and expanded to insure adequate off-site radiation detection capabilities in response to TMI-2 recovery operations.

Several environmental programs were upgraded in response to recovery operation. The environmental Thermo Luminescent Dosimetry (TLD) program was expanded to include seventy-three (73) sites, from the original twenty (20), along with infield placement of new TLD monitors with an expanded detection capability for β radiation.

The entire environmental TLD program was upgraded with the acquisition of on-site readout and data processing. In addition to the expanded TLD programs, monitoring equipment has been obtained to permit direct sampling of air, in both a continuous and grab mode, and direct readout of radiation levels in the field.

The need for increased environmental monitoring with respect to TMI-2 recovery operation was greatly enhanced by the acquisition of a mobile monitoring van. The van is being equipped with instrumentation for infield monitoring and sampling.

A groundwater monitoring program has been initiated around TMI-2 containment with an established sampling regime to be followed throughout the recovery phase.

2.3 SUMMARY OF ACCOMPLISHMENTS

The overall effort expended during this period was directed toward improving, upgrading, and providing input for environmental monitoring programs. Equipment acquisitions were directed toward increasing the on-site capabilities for environmental radiation monitoring during and after TMI-2 recovery. Inputs into recovery system design were made to insure the health and safety of the general public and compliance with radiological technical specifications and regulatory statutes.

SECTION 8
PROJECT OPERATIONS

1.0 SCOPE

The scope of work for the Project Operations organization includes all activities carried out by Bechtel Power Corporation. Included is engineering design, procurement, construction, construction management and associated support activities required for containment decontamination, defueling, plant reconstruction, and the associated systems and facilities required to support these activities. Project Operations' tasks include such activities as preparation of drawings and specifications, various engineering planning studies, preparation of cost and schedule estimates, purchasing and contracting for equipment and services required to support TMI-2 recovery, management of construction and decontamination and other technical publications, and assistance in the acquisition and analysis of radiochemical data regarding the status of TMI-2.

2.0 CURRENT ACTIVITIES

Estimates of gamma and beta dose rates for use by re-entry teams have been completed and drawings depicting isodose profiles have been prepared for use by the initial entry teams.

A baseline engineering package for an interim waste staging facility to accomodate 55 gallon drums and LSA boxes has been completed and is under review by the General Public Utility Service Corporation (GPUSC). General arrangement drawings and a material handling study for the containment recovery service building have been completed and are being reviewed by GPUSC.

The ground water monitoring well system has been completed and is now undergoing pre-operational testing.

An evaluation directed to optimizing the Engineering Change Memorandum procedure has been completed and is now undergoing internal review.

A supplement to the July 1979 report on containment decontamination has been issued in draft form for internal review. In addition, an overall radwaste management study, a study on alternate methods for the disposal of tritiated water, and a preliminary evaluation of radwaste volume reduction techniques have been completed.

Cost and schedule estimates have been prepared for potential alternates to the submerged demineralizer system, the equipment decontamination system building, the fluorocarbon adsorption system alternate to containment purge, a new sewage collection and treatment system for the TMI-2 site, and the EPICOR II resin solidification facility.

A set of survey control monuments has been installed at various locations around the TMI-2 site. Land has been cleared, graded, and compacted for a new laydown yard south of the dike on the TMI-2 site. Scrap metal and miscellaneous tankage has been relocated to allow for future expansion of the TMI-2 parking lot.

A contract for the new TMI-2 administration building was awarded on February 27, 1980. Preliminary construction activities for this new building are now underway.

SECTION 9

QUALITY ASSURANCE

1.0 SCOPE

The Quality Assurance Department provides quality assurance and quality related services in support of the recovery effort for all items and activities identified by engineering as "important to safety". The department provides and maintains the QA plan which describes the QA program on a project-wide basis and reviews "important-to-safety" implementing procedures from other departments to assure overall program compliance with regulatory requirements and commitments. The department also conducts an independent audit program to assess program adequacy and to verify compliance.

During the design process, quality engineering reviews of specifications are performed to assure that appropriate quality requirements have been incorporated. During shop manufacturing and modification or construction on site, the department provides inspection to verify compliance with specification and regulatory requirements and independently monitors the "important to safety" activities of the recovery staff to assure compliance with approved procedures, administrative controls and regulatory requirements. The Quality Assurance Organization has the authority to stop work or discontinue further processing when nuclear safety considerations warrant this action. Periodic reports are issued to upper management regarding the implementation and effectiveness of the Quality Assurance Program during this recovery effort.

2.0 CURRENT ACTIVITIES

2.1 PROGRAM DEVELOPMENT

The quality assurance department has been working closely with other TMI-2 organizations to prepare and issue the TMI-2 Recovery Plan. When issued, this plan will be the basic document which describes the overall approach to quality being implemented at TMI-2. A primary consideration during preparation has been to produce a workable, useful document which meets all applicable regulatory requirements. The plan is organized by functional area to facilitate its utilization.

The scope of the QA Program as described in the new Plan has been enlarged beyond the existing "safety-related" concept. The new Program will encompass all items and activities "important to safety". Engineering has the responsibility for classification and the Quality Assurance Department has worked with engineering to define and clarify the new concept.

A project-wide effort to update the procedures necessary to implement the QA Program is underway and is being assisted by the Quality Assurance Organizations.

2.2 DESIGN AND PROCUREMENT

Quality Assurance Department personnel have participated, to a limited extent, in the design and procurement activities necessary to support recovery. During this period, the department has been active in review of design and procurement documents pertaining to the submerged demineralizer and the radwaste evaporation systems. It is expected that implementation of the Recovery QA Plan and issuence of the New Quality Classification List will lead to greater participation of the department due to the increased scope under the quality program. Shop inspection of vendor activities is also expected to increase as production of additional Unit 2 facilities increases to support recovery.

2.3 SITE QC INSPECTION

Quality Assurance personnel perform inspection of modification and construction activities within the scope of the QA program. All work authorization documents are reviewed to ensure inclusion of applicable inspection and notification requirements. During this period, inspection activities on Unit 2 continued at levels up to 200 inspections per month including receipt inspection and inspections of maintenance activities.

2.4 MONITORING SITE ACTIVITIES

During this period, the existing QA surveillance program on site is being restructured. The new program is based on an independent monitoring by QAD personnel of activities "important to safety". The program is designed to provide management with an adequate confidence level that activities are being conducted in accordance with regulatory and administrative requirements. Program development is continuing, including procedure preparation and indoctrination of personnel to the new concept. Surveillance activities have continued in support of TMI-2 Recovery during this period. Operation of EPICOR II, construction of radwaste facilities and special projects such as initial entry of the airlock have been subjects of recent surveillances. Increased activity is expected in future months under the new monitoring program.

2.5 AUDITS

The Quality Assurance Department audit program has remained on schedule, with both on-site and off-site audits being conducted. Attention has been focused on reducing the backlog of open audit findings and ensuring timely response for corrective actions. Additional audit emphasis during this period was applied to the Radiological Controls Programs.

SECTION 10

TRAINING

1.0 SUBTASK A. OPERATOR TRAINING

1.1 SCOPE

1.1.1 AUXILIARY OPERATOR TRAINING

This program is applicable to all personnel who successfully bid or are hired as an Auxiliary Operator. Program objectives are to:

1. Provide training to Auxiliary Operators to progress from new hire, Auxiliary C, to Auxiliary A.
2. Provide a qualified Auxiliary Operator as an input to the replacement operator program as vacancies dictate.

The training program for Auxiliary Operator progression consists of a two year program of classroom instruction, in-plant training and experience, and written and practical examinations. Successful completion of the first year results in automatic promotion from Auxiliary [illegible] to Auxiliary B; successful completion of the second [illegible]s in automatic promotion to Auxiliary A. After a [illegible] period of performing as an Auxiliary A, and as ope[illegible] become available, the operator is eligible for consideration for entry into the Control Room Operator training program.

1.1.2 LICENSED REACTOR OPERATOR TRAINING

This program is applicable to all Auxiliary Operators A. The objectives of the courses are:

1. Prepare an Auxiliary Operator A to achieve the position of Control Room Operator.

2. Prepare a candidate to successfully complete the NRC License Examination.

The training program consists of the following:

a. Specific study assignments
b. Oral checkouts in practical factors
c. Written tests
d. Oral comprehensive examinations
e. Classroom lessons
f. Simulator training

The program is conducted over a period of nine months, with the classroom phase consisting of approximately six weeks.

1.1.3 SENIOR REACTOR OPERATOR TRAINING

This program is applicable to all Shift Foreman candidates who are licensed TMI Control Room Operators.

The objectives of the program are to provide the following:

1. Senior Reactor Operator License
2. Supervisory Development

The program consists of the following:

a. SRO License training program
b. PWR simulator S/U certification course (if not completed during Operator Licensing Program)
c. Licensed operator requalification program
d. PWR simulator refresher training
e. Supervisory training
f. First aid
g. Fire fighting

h. Switching and Tagging
i. Safety for Supervisors
j. SRO Decision Analysis
k. Other Related Training

1.1.4 RO/SRO REQUALIFICATION TRAINING

This program is applicable to all Licensed Reactor Operators and Senior Reactor Operators.

The objective of the program is to maintain the qualification of Licensed Reactor Operators and Senior Reactor Operators. The requalification Program consists of four interrelated segments which run concurrently. These segments are:

1. Operational Review Lecture Series (OR)
2. Fundamentals and System Review Program (FSRP)
3. On-The-Job Training
4. Annual Evaluation Examinations

The OR Series is a classroom lecture presentation which provides licensed personnel with the details of operational information related to the Three Mile Island Station. As part of the OR Series, selected FSR topics are presented. FSR topics are selected in areas where annual operator and senior operator written examinations indicate that emphasis in scope and coverage is needed. OR lectures are scheduled for a minimum of 60 hours per year.

On-the-job training is designed to ensure that all licensed personnel operate reactor controls and participate in major unit evolutions. Records of all on-shift performance are maintained and periodically reviewed by supervisory personnel.

The annual evaluation examinations simulate the written and oral examinations administered by the Nuclear Regulatory Commission. Performance on these annual evaluation examinations determine the extent of the FSR program during the twelve month requalification period.

1.2 CURRENT ACTIVITIES

1.2.1 AUXILIARY OPERATOR TRAINING ACCOMPLISHMENTS

There are currently ten (10) Auxiliary Operator "B's" enrolled in the program. All training conducted was on-the-job.

1.2.2 LICENSED REACTOR OPERATOR TRAINING ACCOMPLISHMENTS

During this period, one person completed the training which includes simulator certification at B&W Training Center, Lynchburg, Va.; a company administered audit examination, and licensing examination by NRC for a Reactor Operator License. Four other personnel are enrolled in the program. Their training was on-the-job.

1.2.3 SENIOR REACTOR OPERATOR TRAINING ACCOMPLISHMENTS

There are currently four people enrolled in the program. During this period a company audit examination was administered; they attended simulator training at Lynchburg, Va., as part of the Licensed Operator Requalification Program, and they were adminiatered licensing examinations by the NRC.

1.2.4 RO/SRO REQUALIFICATION TRAINING ACCOMPLISHMENTS

There are nineteen (19) licensed Reactor Operators/Senior Reactor Operators enrolled in the requalification program. Annual requalification examinations were administered and all operators attended simulator training during this period.

2.0 SUBTASK B. MAINTENANCE TRAINING

2.1 SCOPE

Maintenance training is provided in the following four phases, each for a six month duration:

1. Phase I: General Site and Maintenance Fundamentals Training
2. Phase II: Skills Systems Maintenance Training
3. Phase III: Skills Specialty Training
4. Phase IV: Applications Training

The program is applicable to the following classifications:

a. I&C Technicians
b. Electrical Maintenance Technicians
c. Mechanical Maintenance Technicians
d. Utility Technicians

2.2 CURRENT ACTIVITIES

During this period, Phase I training was conducted continuously for maintenance personnel. Personnel attendance was by shift rotation, with each shift receiving the training one week out of six weeks. There are 150 individuals enrolled in this program.

3.0 SUBTASK C. HEALTH PHYSICS TECHNICIAN TRAINING

3.1 SCOPE

The training for Health Physics Technicians is separated into the following categories:

1. Met-Ed Employees

   a. Newly hired Technician training

   b. Health Physics Technician/Foremen training

2. Contractor Personnel Training

TMI Unit 2 is currently utilizing Nuclear Support Services (NSS) contractor Radiological Control Personnel. The report contains the scope of that training. At such time as Met-Ed system employees are hired by Unit 2 for Radiological Control Positions, the training program for them will be described. The contractor personnel training consists of:

a. Training of personnel in TMI-2 procedures

b. Qualification Testing

3.2 CURRENT ACTIVITIES

During this period, the program has been conducted continuously in order to provide training to the incumbent NSS Technicians. Approximately 50 NSS personnel were qualified to perform duties in Unit 2 Radiological Control Positions.

4.0 SUBTASK D. GENERAL EMPLOYEE TRAINING

4.1 SCOPE

This program is applicable to all personnel on site. In addition, the program was expanded to include practical factors training during this period.

4.2 CURRENT ACTIVITIES

All new personnel reporting to the site during this period received this training. The new employees who will be permitted to enter

Radiological Control areas unescorted, began the expanded training commencing February 18, 1980. Approximately 18 people per day who received the radiological control training prior to February 18, are being retrained in the new practical factors training. It is anticipated that all personnel will have received the additional training prior to June 1, 1980.

5.0 SUBTASK E. RADWASTE MANAGEMENT TRAINING

5.1 SCOPE

5.1.1 RADWASTE ADMINISTRATION TRAINING

This program is applicable to all personnel who control and prepare radwaste for shipment. Its objective is to train personnel in the regulations for radwaste shipment.

5.1.2 RADWASTE REDUCTION TRAINING

This program is applicable to all personnel who operate radwaste systems or who are involved through their work in the production of radwaste.

The primary objective of the program is reduction in the amount of the radioactive waste produced.

5.2 CURRENT ACTIVITIES

5.2.1 RADWASTE ADMINISTRATION TRAINING ACCOMPLISHMENTS

A series of four lectures were conducted during this period, in order to train all personnel involved in radwaste shipment. Approximately twenty-seven (27) supervisory personnel attended these lectures. Fifty-seven (57) technicians were trained during this period.

5.2.2 RADWASTE REDUCTION TRAINING ACCOMPLISHMENTS

The principle of radwaste reduction has been incorporated in the General Employee Training which was conducted after February 18, 1980. In addition, a backfit program is being conducted continuously to indoctrinate all personnel in the principles of radwaste reduction. This will continue until all personnel on site receive this training; completion is estimated to be June 1, 1980. Approximately 18 individuals each day receive this indoctrination as part of radiological controls practical factors training program.

6.0 SUBTASK F. SPECIAL TRAINING

6.1 SCOPE

6.1.1 REACTOR BUILDING RE-ENTRY TEAM TRAINING

This training is applicable to the personnel forming the Reactor Building Re-entry Team and to the management personnel involved in the control of the re-entry. Training consists of the following:

1. Radiological fundamental
2. Equipment familiarization
3. Task familiarizations and practices
4. Physical and psychological examinations and preparations
5. Radiological condition
6. Communications familiarizations
7. Management Briefings

6.1.2 REACTOR BUILDING PURGE SYSTEM TRAINING

This training is applicable to all TMI-2 personnel who operate the system. The objectives are to familiarize operators with the system construction, controls, and operations.

6.1.3 EMERGENCY PLAN TRAINING

This program is applicable to all Unit 2 personnel. It provides indoctrination in the emergency plan, and provides for drill in the various casualties that require implementation of the plan.

6.1.4 SUPERVISORY INDOCTRINATION IN CONTROLLED SUBSTANCE TRAINING

This program is applicable to all supervisors. The training consisted of a lecture on a Description of Controlled Substances and recognition of the use of controlled substances by personnel.

6.2 CURRENT ACTIVITIES

6.2.1 REACTOR BUILDING RE-ENTRY TEAM TRAINING ACCOMPLISHMENTS

During this period, Items 1 through 7 were completed and Item 1 was repeated to account for newly assigned personnel.

6.2.2 REACTOR BUILDING PURGE SYSTEM TRAINING ACCOMPLISHMENTS

During this period, a total of six (6) lectures were conducted to provide training for the operation shifts. In addition, one make-up lesson was given for personnel missing any shift lecture.

6.2.3 EMERGENCY PLAN TRAINING ACCOMPLISHMENTS

During this period, a series of six drills were conducted on specific casualities. The various emergency plans response personnel were provided training prior to conducting these drills. A station-wide drill was conducted in March. Three Radiological Control Drills were conducted for Radiological Control Personnel.

6.2.4 SUPERVISORY INDOCTRINATION IN CONTROLLED SUBSTANCE TRAINING ACCOMPLISHMENTS

The training was provided three times during March. Approximately 50 supervisors attended.

7.0 SUBTASK G. CHEMISTRY TECHNICIAN TRAINING

7.1 CURRENT ACTIVITIES

This program is under development and will be included in subsequent reports.

SECTION 11
SECURITY

1.0 SCOPE

The scope of the Security Department of TMI Unit II is to institute a Security system separate from TMI Unit I to enable the Security Force to address the unique security requirements in connection with Recovery Operations. The department must be able to allow the flow of employees necessary for a recovery and at the same time provide high assurance that unauthorized personnel and/or equipment will be denied access into the protected/vital areas. The department must also be able, in the event of intrusion into the protected area, to neutralize the intruders or deny them entry to vital areas until assistance arrives.

2.0 CURRENT ACTIVITIES

2.1 ORGANIZATION

The organization of the security department for TMI Unit II became effective January 1, 1980. The department consists of a security supervisor, a site protection sergeant in the capacity of administration/training, site protection sergeants in the capacity of shift commanders (Duty Sergeants), an administrative assistant and sufficient site protection officers to man the necessary posts.

2.2 SECURITY SYSTEM

A new system for access control into the protected/vital area went into effect the latter part of January, 1980. This is a computerized system that simplifies record keeping and expedites the entry of authorized personnel. It also assists the security department in exercising positive access control into the vital/protected area.

2.3 SECURITY PROCEDURES

New Security procedures for TMI Unit II have been ninety percent completed and are now in the review and approval cycle.

2.4 BADGING SYSTEM

A new and separate badging system for TMI Unit II employees is being implemented and upon completion will assist the employee in needed access, and enable the Security Department to have more positive control of the personnel entering the vital/protected area.

APPENDIX 1 - ANNOTATED SEQUENCE OF EVENTS

TABLE OF CONTENTS

1.0 INTRODUCTION

This report is a sequence of the events for the 20 hour period following the March 28, 1979 loss of coolant accident at Three Mile Island, Unit 2 Nuclear Station. The report is a culmination of 4 man years of detailed analysis of recorded plant parameter data, plant logs and operator interviews. The results of the analysis are presented in an annotated sequence of events which includes a chronology of events, a reference to the source of each entry in the chronology and the information available to the operators regarding each event. The type of information available to the operator, the form in which the information was presented and the timeliness of the presentation of the information relative to the occurrence of the events are also included.

The sequence of events focuses on the operator actions and the plant response during the 20 hours following the Unit 2 Turbine trip at 0400:37 a.m. The report does not address the extent of nuclear fuel damage, the pathways of radioactive material releases nor implementation of the Emergency Plan. Conclusions and recommendations are also considered to be outside the scope of the report.

The analysis and preparation of the sequence of events was performed by an integrated group whose members were drawn from the staff of Metropolitan Edison Company, Jersey Central Power and Light Company, Energy Incorporated and Babcock and Wilcox. Significant contributions were also made by General Public Utility Service Corporation, Pickard, Lowe and Garrick and the Nuclear Regulatory Commission.

2.0 LIST OF FIGURES

3.0 LIST OF REFERENCES

1. TMI Unit 2 Reactimeter Data (1)

2. TMI Unit 2 Computer Data
   a. Alarm Summary
   b. Sequence of Events Review
   c. Utility Printer
   d. TMI Station Log
   e. Post Trip Review

3. TMI Plant Stripcharts
   a. Area Gamma Monitors (HP-UR-1901)
   b. Area Gamma Monitors (HP-UR-1902)
   c. Atmospheric Radiation Monitors (HP-UR-2900)
   d. Liquid Radiation Monitors (HP-UR-3264)
   e. Atmospheric Radiation Monitors (HP-UR-3236)
   f. Atmospheric Radiation Monitors (HP-UR-1907)
   g. Auxiliary Building Air Flow (AH-FR-5313 and AH-FR-5286)
   h. Fuel Handling Building Air Flow (AH-FR-5709 and AH-FR-5659)
   j. Reactor Building Temperature and Pressure
   k. Intermediate Range and Source Range Monitors
   m. Turbine Header Pressure Loop B Steam Pressure
   n. Reactor Coolant System Wide Range Pressure
   p. Reactor Coolant System Loop A and B, Wide Range Hot Leg Temperatures (YM-TR-7167)
   q. Condensate Polisher Flows
   r. Self Power Neutron Detector Backup Incore Recorders
   s. Condenser Hot Well Level
   t. Atmospheric Radiation Monitors (RM-R-6)
   u. Atmospheric Radiation Monitors (RM-R-5)
   v. Control Room Air Flow (AH-FR-5191 and AH-FR-5192)

4. TMI Plant Logs
   a. TMI Unit 1 Shift Supervisor/Control Room Operator Logs
   b. TMI Unit 2 Shift Supervisor/Control Room Operator Logs
   c. TMI Unit 2 Emergency Control Station Log
   d. TMI Unit 2 Primary Radwaste Auxiliary Operator Log
   e. TMI Unit 2 Chemistry Log/Sample Results
   f. TMI Unit 2 Emergency Control Center Log
   g. NRC Region 1 Response Center Tapes
   h. NRC Region 1 Incident Message forms
   j. TMI Unit 2 Hand Recorded Exit Fuel Assembly Temperatures

5. Technical Data Reports with General Public Utility Service Corporation (1)
   a. Accident Transient Modeling Analysis, TDR 045
   b. Source and Pathways of TMI 2 Releases of Radioactive Materials, TDR 055
   c. TMI 2 Makeup Pump Operation, TDR 118
   d. Loss of Feedwater Flow Leading to the Accident of March 28, 1979, TDR 136

6. Technical Manuals
   a. TMI Unit 2 Technical Specification
   b. TMI Unit 2 Final Safety Analysis Report
   c. TMI Unit 2 System Descriptions

(1) This information was not available to the Operator on March 28, 1979.

7. T. Van Witbeck memorandum regarding TMI Unit 2 Operating Staff and PORC Sequence of Events Review Meeting (1)

8. TMI Staff Interviews Conducted By Met-Ed/GPU (1)
   a. Ken Bryan dated April 26, 1979
   b. Joe Deman dated April 25, 1979
   c. Craig Faust dated March 30, 1979 and April 6, 1979
   d. Ed Frederick dated March 30, 1979 and April 6, 1979
   e. John Flint dated April 20, 1979
   f. Craig Faust and Ed Frederick dated March 29, 1979
   g. Jim Floyd dated April 20, 1979
   h. Don Miller dated March 30, 1979
   j. Juanita Gingrich dated March 30, 1979
   k. Dale Laudermilch dated March 30, 1979
   m. Hugh McGovern dated March 29, 1979 and May 4, 1979
   n. Brian Mehler dated April 25, 1979
   p. Steve Mull dated March 30, 1979
   q. Frederick Scheimann dated March 30, 1979
   r. Bill Zewe dated March 30, 1979 and April 6, 1979
   s. Dick Dubiel, Gary Miller and Jim Seelinger dated April 12, 1979

9. TMI Staff Interviews Conducted by NRC (1)

   a. Craig Faust dated April 21, 1979
   b. Terry Daugherty dated April 22, 1979
   c. Frederick Scheimann dated April 23, 1979
   d. Ed Frederick dated April 23, 1979
   e. Bill Zewe dated April 23, 1979
   f. John Flint dated April 23, 1979 and July 2, 1979
   g. Joe Deman dated April 24, 1979
   h. Dick Dubiel dated April 24-25, 1979, May 8, 1979 and May 22, 1979
   j. Mike Ross dated April 25, 1979 and May 19, 1979
   k. George Kunder dated April 26, 1979, May 17, 1979, May 23, 1979 and July 11, 1979
   m. James Higgins dated May 1, 1979
   n. Donald Neely dated May 2, 1979 and May 5, 1979
   p. Michael Janouski dated May 2, 1979
   q. Michael Benson and Howard Crawford dated May 3, 1979, May 22, 1979 and June 6, 1979
   r. Thomas Leach dated May 3, 1979
   s. Lee Roger dated May 4, 1979
   t. David Zeiten dated May 5, 1979
   u. Gary Miller dated May 7, 1979
   v. Richard Benner and Michael Kuhn dated May 8, 1979
   w. Jim Seelinger dated May 8, 1979
   x. Lynn Wright dated May 9, 1979
   y. Joe Logan dated May 9, 1979
   z. Jack Herbein dated May 10, 1979
   A. Ken Bryan dated May 16, 1979 and July 11, 1979
   B. Tom Davis, Jr. dated May 16, 1979
   C. Scott Wilkerson dated May 16, 1979
   D. Walter Marshall dated May 17, 1979
   E. Brian Mehlen dated May 17, 1979
   F. John Donnachie dated May 17, 1979

(1) This information was not available to the Operator on March 28, 1979.

TMI Staff Interviews Conducted by NRC (1) (cont'd)

G. Ronald Fountain dated May 17, 1979
H. Carl Guthrie dated May 18, 1979
I. Dale Laudermilch dated May 18, 1979
J. Hugh McGovern dated May 18, 1979 and June 20, 1979
K. Patrick Shannon dated May 19, 1979
L. Frank Telenko dated May 19, 1979
M. Joe Chwastyk dated May 21, 1979
N. Ivan Porter, Jr. dated May 21, 1979 and July 2, 1979
P. Don Miller dated May 21, 1979
Q. Steve Null dated May 22, 1979
R. Theodore Illjes dated May 23, 1979
S. Douglas Weaver, Jr. dated May 24, 1979
T. Ed Fredrick, Craig Faust, Fred Schelmann, Bill Zewe and Mike Ross dated May 29, 1979
U. Thomas Wright dated June 15, 1979
V. Instrument Man B dated June 20, 1979
W. Barry Kalchevitch dated June 26, 1979
X. Ed Fredrick, Craig Faust, Fred Schelmann and Bill Zewe dated June 28, 1979
Y. Harold Farst dated July 13, 1979
Z. Charles Mell dated July 14, 1979

10. Sequence of Events prepared by TMI Staff and Onsite B&W Resident Engineer titled: (1) "TMI Station, March 28, 1979 Events Unit Number 2, G. P. Miller Station Manager."

(1) This information was not available to the Operator on March 28, 1979.

4.0 LIST OF SYMBOLS

Indications

ST	Electrical Status Light
MR	Meter
SC	Stripchart Recorder
AN	Annunciator
PL	Control Room Panel
AP	Alarm Printer
MP	Multipoint Recorder
UP	Utility Printer

Parameters

T	Temperature
P	Pressure
L	Level
F	Flow
A	Amperage
V	Vibration

Plant Identifiers

RC	Reactor Coolant
PZR	Pressurizer
C	Loop Cold Leg
H	Loop Hot Leg
SG	Steam Generator
MS	Main Steam
RB	Reactor Building
RCDT	Reactor Coolant Drain Tank
LD	Letdown
ESF	Engineered Safety Features
EF	Emergency Feedwater
NI-1	Source Range Monitor
NI-3	Intermediate Range Monitor
NI-4	Intermediate Range Monitor
RC-P	Reactor Coolant Pump
MU-P	Makeup Pump
FW-P	Feedwater Pump
DH-P	Decay Heat Pump
EF-P	Emergency Feedwater Pump

This table in conjunction with Figure 65 "TMI Unit 2 Control Room Layout", is provided as a guide to understanding the entries under the "Information Available to the Operator" column in the Annotated Sequence of Events.

5.0 ANNOTATED SEQUENCE OF EVENTS

Information Available to the Operator	Reference

For this chronology, an elapsed time clock was established with the time of the turbine trip, 0400:37, defined as elapsed time equal to zero. The elapsed time of each event in the sequence is given as the number of hours, minutes and seconds relative to 0400:37, followed in parenthesis by the real time using a 24-hour clock. For example, 1:52:43 p.m. on March 28 would be written 9:52:06 (1352:43). Depending upon the accuracy of the source of data for each event, the times appear alone or with the notation "approximate."

Annotations included with the chronology of events, in addition to providing periodic assessments of the plant status, represent input culled from interviews with the operating staff.

In cases where direct action was taken by the plant operating staff the term "the operator or the shift supervisor" is used in the sequence of events.

.LANT STATUS

Prior to the accident Three Mile Island Unit Two was at 97% power with the Integrated Control System in full automatic. Rod groups one through five were fully withdrawn, rod groups six and seven were 95% withdrawn and rod group eight was 27% withdrawn. Reactor Coolant System total flow was approximately 138 million pounds per hour and the Reactor Coolant System pressure was 2155 psig. Reactor Coolant Makeup Pump 1B (MU-P-1B) was in service supplying makeup and Reactor Coolant Pump seal injection flow. Normal Reactor Coolant System Makeup flow was approximately 70 gallons per minute. Reactor Coolant System boron concentration was approximately 1030 parts per million. The water level in

Information Available to the Operator — **Reference**

the borated water storage tank was approximately 55 feet. The Pressurizer Spray Valve (RC-V1) and Heaters, except groups 8 and 7, were in normal control while spraying reactor coolant into the Pressurizer to equalize boron concentrations between the Pressurizer and the remainder of the Reactor Coolant System. The Pressurizer Relief Valves discharge header thermocouples indicated values between 186F and 200F due to leakage through one of the three Pressurizer Relief Valves (RC-R2, RC-R1A or RC-R1B). An RC-R1B high temperature alarm had been received at -2:37 (0123) and was reset at -2:28 (0132). Temperatures recorded were 200F for the alarm and 192.4F for the alarm reset.

The following table lists Steam Generator parameters prior to the accident.

Table of Steam Generator Parameters+

	Steam Generator A	Steam Generator B
Loop Feedwater	5.7458 MPPH*	5.7003 MPPH*
Operating Level	56%	57.4%
Startup Level	158.8 inches	163.4 inches
Steam Pressure	910 psig	889.6 psig
Feedwater Temperature	462.7F	462.7F
Steam Temperature	595F	594F

* MPPH - Million Pounds Per Hour

+ The differences between Steam Generator A and B parameters are typical of normal operation.

Steam Generator Feedwater Pumps 1A and 1B (FW-P-1A and FW-P-1B), Condensate Booster Pumps 2A and 2B (CO-P-2A and CO-P-2B) and Condensate Pumps 1A and 1B (CO-P-1A and CO-P-1B) were in service. Two heater drain pumps were on line to

Information Available to the Operator	Reference
recovering heater drain water and supplementing main feedwater pump suction requirements. The auto/manual switch which provides certain start/stop features to the condensate, condensate booster and main feedwater pump control circuitry was in the manual mode. An attempt was being made to clear a clogged resin transfer line in an isolated standby demineralizer of the Condensate Polishing System.	
The Fuel Handling Building supply fans (AH-E-9A and AH-E-9B) and exhaust fans (AH-E-10A and AH-E-10B or AH-E-10C and AH-E-10D) were in service.	
The Auxiliary Building exhaust fans (AH-E-8C and AH-E-8D) were in service.	
The Auxiliary Building supply fans (AH-E-7A and AH-E-7B) were not of service.	

Time	Event	Information Available to the Operator	Reference
-00:00:05 (0400:32) Approximate	Condensate Polisher outlet valves went shut simultaneously. The cause of their sudden closure has not yet been firmly established.	Local indication at the condensate Polishers Panel).	3q,5d,8h, 9O,9P
-00:00:01 (0400:36) Approximate	Condensate Booster Pumps 2A and 2B (CO-P-2A and CO-P-2B) tripped on low suction pressure. No computer printout occurred because the auto/manual switch for the condensate system was in the manual position.	Annunciator window (AN) at Panel 17 (PL17), meter (MR) indicating motor amperage (A) and electrical status lights (ST) at Panel 5 (PL5).	5d
-00:00:01 (0400-36)	Condensate Pump 1A (CO-P-1A) tripped. This was the result of a wiring error in the 4160 volt switchgear bus control circuit which tripped condensate pump 1A (CO-P-1A) when condensate booster pump 2A (CO-P-2A) tripped with the auto/manual switch in the manual position. Condensate pump 1B (CO-P-1B) apparently remained on line.	AN at PL17, MR(A) and ST at PL5.alarm printer (AP) output of norm/trip and on/off (delay time between alarm printer output and real time approximately 0 seconds).	2a,2b,5d
00:00:00 (0400:37)	Feedwater Pumps 1A and 1B (FW-P-1A and FW-P-1B) tripped on low suction pressure caused by the loss of condensate booster pumps 2A and 2B (CO-P-2A and CO-P-2B). This resulted in a loss of feedwater flow to both steam generators.	AN at PL15 and PL17, speed and throttle valve position stripchart recorders (SC) at PL17, speed MR at PL4, pump discharge pressure (P_{DISCH}) MR at PL5, AP norm/trip (Delay $\simeq$ 0 seconds)	2a,2b,5d
00:00:00 (0400:37)	The Main Turbine and Main Generator tripped in accordance with plant design.	Turbine: AN at PL5 and PL17, various MR and ST at PL5, AP norm/trip (Delay $\simeq$ 0 seconds) Generator: AN at PL18, various MR and ST at PL4A, AP norm/trip (Delay $\simeq$ 0 seconds)	2a,2b,6c
00:00:00 (0400:37)	All three Emergency Feedwater Pumps 1, 2A and 2B (EF-P-1, EF-P-2A and EF-P-2B) started.	All EF-P's: ST and MR(P_{DISCH}) at PL4 EF-P's 2A and 2B: MR(A) at PL4, AP on/off (Delay $\simeq$ 0 seconds)	2a,8c
00:00:04 (0400:41) Approximate	The Electromatic Relief Valve (RC-R2) opened at the setpoint of 2255 psig.	ST at PL4	1

Time	Event	Information Available to the Operator	Reference
00:00:04 (0400:41)	The Reactor Coolant Drain Tank Pressure began to increase.	MR at PL8A	1
00:00:05 (0400:42) Approximate	Turbine Bypass Valves (MS-V25A, MS-V25B, MS-V26A and MS-V26B) opened at 1010 ± 10 psig to control steam generator pressure, (Figure 34).	MS-V25A/26A: MR and ST at PL3 MS-V25B/26B: MR and ST at PL3	1
00:00:08 (0400:45)	The reactor tripped on high Reactor Coolant System pressure at 2345 psig. The setpoint is 2355 psig.	AN (Red/Green/Blue/Yellow) Two out of Four Logic at PL8, ST and MR at PL14, Neutron Flux SC and MR at PL4. AP Two out of Four Logic (Delay ≃ 0 seconds)	1,2a,2b
00:00:08 (0400:45) Approximate	The operator shut Letdown Isolation Valve (MU-V3 6 o stop letdown flow in anticipation of the expected pressurizer level decrease which follows the initial increase in pressurizer level after a loss of feedwater flow incident (Figure 31).	ST at PL3, letdown flow MR at PL3	8c,8d,9a
00:00:08 (0400:45) Approximate	The operator placed the Pressurizer Spray Valve (RC-V1) and Pressurizer Heaters under automatic control. Pressurizer Heater Groups 1 through 5 were de-energized as a result of this action. The setpoint at which the pressurizer heater de-energized is 2125 psig for Groups 1 through 3 and 2140 psig for Groups 4 and 5 under increasing pressure. Note: There are a total of 13 Pressurizer Heater Groups. Heater Group 6 and 7 were not available on March 28, 1979.	Spray Valve: ST at PL4 Heaters: ST at PL4, AP norm/trip (Delay ≃ 0 seconds)	1,2a,8d,9d
00:00:09 (0400:46)	Steam Generator levels were approximately 120 inches (Figure 34). Steam pressure had increased to 1053 psig in Steam Generator B and 1074 psig in Steam Generator A (Figure 34). Assuming the Steam Generator Safety Valves opened within the specified setpoints range, then four of the Steam Generator B Safety Valves and eight of the Steam Generator A Safety Valves opened.	SG L: MR (Startup Range) at PL4, MR (Wide Range) at PL4 SC (Operate Range) at PL4 and PL5 SG P: MR at PL4, SC at PL17	1,6a

Time	Event	Information Available to the Operator	Reference
00:00:10 (0400:47) Approximate	The operator verified that all control and safety rods were tripped and fully inserted in the Reactor core.	ST at PL14, AP norm/trip and yes/no (Delay ≅ 0 seconds)	8c,8h
00:00:13 (0400:50) Approximate	The operator attempted to start Reactor Coolant Makeup Pump A (MU-P-1A); however, he released the control switch before the required 2.5 seconds and the pump tripped.	AM at PL8, ST and MR(A) at PL3, AP norm/trip (Delay ≅ 0 seconds)	2a,2d,5c,8c, 8d,9a,9d
00:00:13 (0400:50)	The Condenser Hotwell low level alarm was received. The level was identified to be 21.72 inches at this time.	MR at PL3, AP low (22.5 inches)/norm/high (36 inches) (Delay ≅ 0 seconds)	2a

PLANT STATUS

The plant had just experienced a turbine/reactor trip. Reactor Coolant System pressure and Pressurizer level were decreasing rapidly after reaching peaks of 2345 psig and 256 inches respectively. The Electromatic Relief Valve (RC-R2) was open and was passing reactor coolant from the steam space at the top of the Pressurizer. Based on Control Room indications, the Reactor Coolant System pressure and Pressurizer level were trending together and decreasing as was expected after a reactor trip. The Steam Generators water levels were at about 96 inches and decreasing at about 4 inches per second. The Steam Generators steam pressures were about 1060 psig and decreasing at 32 psig per second. The Turbine Bypass Valves and a number of Main Steam Relief Valves were open and relieving steam. All Emergency Feedwater Pumps had started. The Steam Generators water levels had not yet reached the Integrated Control System setpoint of 30 inches for the programmed opening of the Emergency Feedwater Valves (EF-V11A and EF-V11B) which would admit feedwater to the Steam Generators. In addition, the Emergency Feedwater Block Valves (EF-V12A and EF-V12B) were shut which also prevented feedwater flow until they were

Time	Event	Information Available to the Operator	Reference
	opened eight minutes after the start of the transient. The reason for the block valves being shut is not known. The last documented operation of these valves was during the performance of surveillance testing of the Emergency Feedwater System on the morning of March 26, 1979.		
00:00:14 (0400:51)	The Emergency Feedwater Pumps (EF-P-1, EF-P-2A and EF-P-2B) normal discharge pressure alarm was received (Figure 59).	MR (P_{DISCH}) at PL4. AP low (setpoint = 875 psig)/norm (Delay ≅ 15 seconds)	2a
00:00:14 (0400:51)	Pressurizer Heater Groups 1 through 5 automatically energized as a result of reactor coolant pressure decreasing below the energize setpoints of 2105 psig for Groups 1 through 3 and 2120 psig for Groups 4 and 5.	ST at PL4, AP norm/trip (Delay ≅ 15 seconds)	2a
00:00:15 (0400:52)	The Reactor Coolant System Pressurizer level reached a peak of approximately 256 inches (Figure 31).	SC at PL4, MR (uncompensated) at PL3	1
00:00:15 (0400:52) Approximate	Steam Generator levels were approximately 87 inches (Figure 40). Steam pressure was 1018 psig in Steam Generator B and 1042 psig in Steam Generator A (Figure 31).	SG L: MR (Startup Range) at PL4, MR (Wide Range) at PL4, SC (Operating Range) at PL4 and PL3 SG P: MR at PL4, SC at PL17	1
00:00:15 (0400:52) Approximate	The Unit 2 Shift Supervisor announced on the Plant Page System that TMI Unit 2 turbine and reactor had tripped.	Announcement made on Plant Page System	8r
00:00:15 (0400:52) Approximate	The Electromatic Relief Valve (RC-R2) should have shut at about this time (closure setpoint of 2205 psig). The Electromatic Relief Valve position indication in the Control Room is a red lamp which illuminates when the Electromatic Relief Valve solenoid is energized. When the lamp is illuminated, i.e. solenoid energized	ST at PL4	1,6a

Time	Event	Information Available to the Operator	Reference
	under normal circumstances the valve should be open. When the lamp is extinguished, i.e. solenoid de-energized, under normal circumstances the valve should be closed; however, the valve could be in any position as the lamp only indicates the electrical status of the solenoid and not valve position. The solenoid on the Electromatic Relief Valve (RC-R2) de-energized at approximately 00:00:15 (0400:52) resulting in an implied "shut" indication in the Control Room. Although the plant operators did not know at the time subsequent events showed that the valve had failed to shut.		
00:00:18 (0400:55)	The Pressurizer Spray Valve (RC-V1) shut.	ST at FLA	1
00:00:20 (0400:57) Approximate	The Steam Generator Safety Valves reseated and the Turbine Bypass Valves (MS-V-25A, MS-V-25B, MS-V-26A and MS-V-26B) modulated steam flow to the Main Condenser to control Steam Generator pressure at 1010 ± 10 psig (Figure 34).	Turbine Bypass Valves: MR and ST at FL3 SG P: MR at FLA, SC at FL1?	1
00:00:25 (0401:02) Approximate	A "Water Hammer" was noted in the condensate pumps discharge piping by an Auxiliary Operator. The piping was displaced approximately 2.5 to 3.0 feet according to the Auxiliary Operator. The pipe movement caused a leak in the flange joint on condensate booster pump (CO-P-2A). It also severed an instrument air line which caused reject inhibit valve CO-V57 to fail shut.	Unit 2 Control Room notified of "Water Hammer"	54,86,91,97

Time	Event	Information Available to the Operator	Reference
00:00:28 (0401:05)	The condenser hotwell normal level alarm was received. The level was 26.44 inches and increasing rapidly. Note: After the Polisher Outlet Valves and the Reject Inhibit Valve (CO-V57) failed shut, hotwell water could not be transferred to either condensate storage tank 1A or 1B (CO-T-1A or 1B). This caused the hotwell level to increase until the Polisher Bypass Valve CO-V12 was opened and the air supply was restored to the Reject Inhibit Valve (CO-V57).	MR at PL3. AP low (22.5 inches)/norm/high (36 inches) (Delay ~ 30 seconds)	2a,54
00:00:30 (0401:07)	Pressurizer Safety Valve (RC-R1B) and Electromatic Relief Valve (RC-R2) discharge line high temperature alarms were received and values of 203.5F and 230.2F, respectively printed out. The high temperatures in the discharge lines were a result of the high temperature steam flow through the Electromatic Relief Valve (RC-R2) during the reactor pressure transient. The Pressurizer Safety Valve (RC-R1B) did not open since the Reactor Coolant System Pressure did not reach the pressurizer safety valve lift setpoint of 2450 psig. The RC-R1B temperature alarm received was due to the back flow of steam in the common discharge header shared with the Electromatic Relief Valve (RC-R2). These alarms were expected by the operator.	MP at PL10, AP high (200F)/norm (Delay ~ 30 seconds)	2a
00:00:31 (0401:08) Approximate	Steam Generator A level reached the Integrated Control System setpoint of 30 inches at which the Emergency Feedwater Valve (EF-V11A) opens (Figure 40). Feedwater was not admitted to Steam Generator A because Emergency Feedwater Block Valve (EF-V12A) was shut. EF-V12A is normally open.	SG L: AR (24 inches) at PL17, MR (Startup Range) at PL4, AP low (24 inches)/norm (Delay ~ 30 seconds) EF-V11A and EF-V11B: MR at PLA EF-V12A and EF-V12B: ST at PLA	1,8c

Time	Event	Information Available to the Operator	Reference
00:00:35 (0401:12) Approximate	Steam Generator B level reached the Integrated Control System setpoint of 30 inches at which the Emergency Feedwater Valve (EF-V11B) opens (Figure 40). Feedwater was not admitted to Steam Generator B because Emergency Feedwater Block Valve (EF-V12B) was shut. EF-V12B is normally open.	SG L: AN (24 inches) at PL17, MR (Startup Range) at PLA AP low (24 inches)/normal (Delay ≃ 30 seconds) EF-V11A and EF-V11B: MR at PLA EF-V12A and EF-V12B: ST at PLA	1,8c
00:00:41 (0401:18)	The operator started Reactor Coolant Makeup Pump A (MU-P-1A). and opened High Pressure Injection Valve (MU-V16B). With Reactor Coolant Makeup Pumps A and B (MU-P-1A and MU-P-1B) operating and deliverying approximately 400 gallons per minute, the Pressurizer level rate of decrease slowed (Figure 31).	MU-P-1A: AN at PL8, ST and MR(A) at PL3, AP norm/trip (Delay ≃ 45 seconds) MU-V16B: ST at PL3, injection flow MR at PL 8	1,2a,5c,8d,9a,9d,9e
00:00:54 (0401:31)	The Reactor Coolant System pressurizer Level reached an indicated minimum level of approximately 158 inches (Figure 31).	SC at PLA, MR (uncompensated) at PL5	1
00:00:57 (0401:34) Approximate	Pressurizer level started increasing (Figure 31). Reactor Coolant System hot leg and cold leg temperatures reached approximately 577F (Figure 6). The Reactor Coolant Drain Tank pressure was 11 psig and increasing (Figure 47).	PZR L: SC at PLA, MR (uncompensated) at PL5 RC T_C: MP at PL10 RC T_H: SC at PLA, MP at PL10 and MR at PLA RCDT P: MR at PL8A	1
00:01:00 (0401:37)	The Pressurizer Safety Valve (RC-R1A) discharge line high temperature alarm was received. This alarm was expected and resulted from back flow in the common discharge header shared with the Electromatic Relief Valve (RC-R2).	MP at PL10 AP high (200F)/normal (Delay ≃ 30 seconds)	2a

PLANT STATUS

The Reactor Coolant System was recovering from the initial loss of feedwater flow transient. The Reactor Coolant System pressure was decreasing and the Reactor Coolant Pressurizer level had begun to increase (Figures 1 and 3). The divergence of Reactor Coolant System pressure and Pressurizer level

Time	Event	Information Available to the Operator	Reference
	was not expected as Reactor Coolant System pressure and Pressurizer level should normally trend together during a loss of feedwater flow transient. The deviation from expected behavior was due to: (1) the failure of the Electromatic Relief Valve (RC-R2) to reseat, which resulted in a lower Reactor Coolant System pressure, and (2) the reduction in the heat removing capability of the Steam Generators because of their low levels. Both of these conditions contributed to the expansion of the reactor coolant volume which forced reactor coolant from the Reactor Coolant System loops and Reactor Vessel into the Pressurizer via the surge line, thereby increasing the level of reactor coolant in the Pressurizer. Steam Generator A and B levels were 11 inches and 14 inches, respectively (Figure 34); however, Emergency Feedwater was not admitted to the Steam Generators because the Emergency Feedwater Block Valves (EF-V12A and EF-V12B) were shut. The Steam Generator pressures were being maintained by the Integrated Control System between 975 psig and 1020 psig (Figure 31). The Reactor Coolant Drain Tank pressure and temperature were increasing showing the effects of the continued discharge of reactor coolant through the Electromatic Relief Valve (RC-R2). Reactor Coolant Makeup Pumps (MU-P-1A and MU-P-1B) were in operation delivering water to the Reactor Coolant System at a rate of approximately 400 gallons per minute via the High Pressure Injection Valve (MU-V16B) and the normal Makeup Valve (MU-V17).		
00:01:13 (0401:38)	The Condenser Hotwell high level alarm was received. The level was 37.77 inches.	[illegible] at PL3. AP low (22.5 inches)/norm/high (36 inches) (Delay ≈ 1 minute)	2e

Time	Event	Information Available to the Operator	Reference
00:01:26 (0402:03)	A Reactor Coolant Drain Tank temperature normal alarm was received and printed out a temperature of 83.5F. This indicated the Reactor Coolant Drain Tank temperature was increasing and had reached the normal range.	MR at PL8A AP high (120F)/norm/low (75F) (Delay ≈ 1 minute)	2a
00:01:43 (0402:22) Approximate	Steam Generators A and B have boiled dry at this time. This was indicated by a steadily decreasing Steam Generator pressure (Figure 9) while Reactor Coolant System hot leg and cold leg temperatures were increasing (Figures 21 and 26).	SG P: MR at PL4 and SC at PL3 SG T: SC at PL4, MP at PL18 and MR at PL4	1,5a
00:02:00 (0402:37) Approximate	The Unit 2 shift supervisor noted all Condensate Pumps, Condensate Booster Pumps and Steam Generator Feedwater Pumps were tripped. Note: It is believed that condensate pump (CO-P-1B) continued to operate throughout the first hour. This is based on a lack of any computer alarm printout for a pump trip or low condensate pump discharge header pressure.	FW-P-1A/1B: AM at PL15 and PL17 CO-P-1A/1B: AM at PL17 CO-P-2A/2B: AM at PL17	5d,7,8r,9D,9I,9K
00:02:02 (0402:39)	The Safety Injection portion of Engineered Safety Features trains A and B actuated as Reactor Coolant System pressure reached 1640 psig. Reactor Coolant Makeup Pump B (MU-P-1B) tripped automatically as a result of the actuation of Safety Injection. The Engineered Safety Features design is such that Makeup Pumps A and C are normally used to deliver a High Pressure Injection rate of 1000 gallons per minute. If Makeup Pump B is running, it is automatically tripped when Safety Injection actuation occurs. Decay Heat Removal Pumps (DH-P-1A and DH-P-1B) Decay Heat Closed Cooling Water Pumps (DC-P-1A and DC-P-1B) and the Emergency Diesels (DF-X-1A and DF-X-1B) also started automatically on Engineered Safety Features trains A and B	RSFBI: AM at PL13, ST at PL3 and PL13 AP norm/actuation (Delay ≈ 2 minutes) MU-P-1B: AM at PL8, ST and MR(A) at PL3 AP norm/trip (Delay ≈ 2 minutes) DH-P-1A, 1B: ST at PL3 and PL13, MR(A) at PL3, MR (P_{DISCH}) at PL8, AP Norm/low and on/off (Delay ≈ 2 minutes) DC-P-1A,1B: AM, MR (P_{DISCH}) and ST AT PL 8 AP on/off (Delay ≈ 2 minutes)	2a,2b,5c,6a,6c

Time	Event	Information Available to the Operator	Reference
	actuation. The Decay Heat Removal Pumps operated in a recirculation mode and did not injecting water into the Reactor Coolant System.	DF-P-1A, 1B: AM, ST, MR (Amps, volts, Hertz, KW/KVARS) AT Falt/norm (Delay $\simeq$ 2 minutes)	
00:02:06 (0402:41)	Reactor Coolant Makeup Pump C (MU-P-1C) started automatically.	AM at PL8, ST and MR(A) at PL3 AP norm/trip (Delay $\simeq$ 2 minutes)	2a,5c
00:03:15 (0403:52) Approximate	The Reactor Coolant Drain Tank Relief Valve (WDL-R1) lifted at 122 psig temporarily halting the Reactor Coolant Drain Tank pressure increase (Figure 47). The pressure increase was caused by the flow of reactor coolant from the Electromatic Relief Valve (RC-R2). The RCDT Relief Valve (WDL-R1) discharges to the Reactor Building Sump.	MR at PL8A	1
00:03:14 (0403:51)	The operator manually bypassed the Safety Injection portion of both Engineered Safety Features trains A and B to gain manual control of the Makeup Pumps and the High Pressure Injection Valves. Both Reactor Coolant Makeup Pumps A and C (MU-P-1A and MU-P-1C) were operating.	AP norm/bypass (Delay $\simeq$ 3 minutes)	2a,9d,9e
00:03:26 (0404:03)	The Reactor Coolant Drain Tank high temperature alarm was received. A temperature of 127.7F was recorded.	MR at PL8A, AP high (120F)/norm/low (75F) Delay $\simeq$ 3 minutes)	2a
00:03:28 (0404:05)	The Reactor Coolant System Pressurizer high level alarm was received. A level of 266.1 inches was recorded.	AM (high = 260 inches) At PL8 AP low (200 inches)/norm/high (260 inches) (Delay $\simeq$ 3 minutes)	2a
00:04:38 (0405:15)	The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C). The indicated Pressurizer level was 360 inches and increasing rapidly at a rate of approximately one inches second (Figure 32).	MU-P-1C: AM at PL8, ST and MR(A) at PL3, AP norm/trip (Delay $\simeq$ 4 minutes) PZR L: AM (high/high = 315 inches, high = 260 inches) at PL8, SC at PL4, MR (uncompensated) at PL5,	1,2a,5e,9c,9d

Time	Event	Information Available to the Operator	Reference
00:04:38 (0405:15) Approximate	In an attempt to gain control of the rapidly increasing pressurizer level the operator throttled the High Pressure Injection Isolation Valves (MU-V16A and MU-V16B).	MR at PLB	8d,9c,9e

PLANT STATUS

The Reactor Coolant System pressure was 1420 psig and steadily decreasing to the saturation pressure of the Reactor Coolant System hot leg temperature (Figure 3). The continued Reactor Coolant System depressurization was due to the failure of the Electromatic Relief Valve (RC-R2) to reseat, and reduction in High Pressure Injection flow rate. Engineered Safety Features, which actuated High Pressure Injection when pressure reached 1640 psig, had been bypassed by the operator to permit manual control of the Makeup Pumps and the High Pressure Injection Isolation Valves. As the Pressurizer level continued to increase, the operator stopped Reactor Coolant Makeup Pump C (MU-P-1C) and throttled the High Pressure Injection Isolation Valves (MU-16A and MU-V16B) in an attempt to control the Pressurizer level and not take the Pressurizer "solid" (Figures 3 and 32). The Reactor Coolant Drain Tank Relief Valve (WDL-R1) had opened at 120 psig and a high temperature alarm had been received as the temperature and pressure of the tank continued to increase. The Steam Generators had boiled dry as indicated by a continuously decreasing steam pressure while Reactor Coolant System hot leg and cold leg temperatures increased (Figures 21, 26 and 35). This was due to the Emergency Feedwater Block Valves (EF-V12A) and EF-V12B) being closed. The operator did not realize EF-V12A and EF-V12B were shut. The Steam Generators startup level indication remained at approximately 10 to 14 inches. In accordance with operating procedures a level of 8 inches or less in a Steam Generator was considered indicative of a dry Steam Generator.

Time	Event	Information Available to the Operator	Reference
00:04:52 (0405:29)	The operator started Intermediate Closed Cooling Water Pump A (IC-P-1A) in preparation of putting a second Letdown Cooler in service.	AN at PL8, ST at PL8 and PL13. MR(P_{DISCH}) and MR(F) at PL8. AP on/off (Delay $\gtrsim$ 5 minutes)	2a
00:04:58 (0405:35)	The operator initiated letdown flow at a rate greater than 160 gallons per minute in an attempt to reduce Pressurizer level to the normal range.	MR at PL3 AP (Range 0-160 gpm) (Delay $\gtrsim$ 5 minutes)	1,2a,8f,8r,9c 9d,9e
00:05:03 (0405:40)	Pressurizer level momentarily stopped its sharp increase at 377 inches and began to decrease. It reached a minimum of 373 inches and again started to increase at 00:05:23 (0406:01) (Figure 3). Maximum Pressurizer level indication is 400 inches.	SC at PL4, MR (uncompensated) at PL5	1
00:05:08 (0405:45)	The Unit 2 Shift Supervisor cleared the trip signal from Condensate Pump 1A (CO-P-1A).	AN at PL17, MR(A) and ST at PL5. AP norm/trip and on/off (Delay $\gtrsim$ 5 minutes)	2a,5d
00:05:13 (0405:50)	The Unit 2 Shift Supervisor started Condensate Pump 1A (CO-P-1A) in an attempt to establish condensate flow. It was not recognized that a blockage existed in the condensate to booster pump flow path due to the polisher outlet valves being shut. In addition it was also not realized that condensate pump 1B (CO-P-1B) was running dead headed.	AN at PL17, MR(A) and ST at PL5 AP norm/trip and on/off (Delay $\gtrsim$ 5 minutes)	2a,5d,9X
00:05:15 (0405:52)	The Unit 2 Shift Supervisor attempted to start Condensate Booster Pump 2B (CO-P-2B). The pump tripped immediately due to a low suction pressure.	AN at PL17, MR(A) and ST at PL5 AP norm/trip (Delay $\gtrsim$ 5 minutes)	2a,5d,9X,9Y
00:05:20 (0405:57)	The Unit 2 Shift Supervisor again attempted to start Condensate Booster Pump 2B (CO-P-2B). The pump tripped immediately due to a low suction pressure.	AN at PL 17, MR(A) and ST at PL5 AP norm/trip (Delay $\gtrsim$ 5 minutes)	2a,5d,9X,9Y

Time	Event	Information Available to the Operator	Reference
00:05:50 (0406:27) Approximate	Reactor Coolant System pressure stopped its sharp decrease and began to increase. The minimum value reached was approximately 1350 psig (Figure 3). As the Reactor Coolant System pressure decreased and the Reactor Coolant System saturation temperature increased, the Reactor Coolant System hot legs reached a saturation temperature-pressure relationship at about 1350 psig which resulted in the formation of steam in the hot legs. The pressure turnaround is due to the existance of adequate decay heat within the Reactor Coolant System to generate steam at a rate which exceeded the loss of energy through the Electromatic Relief Valve and other Reactor Coolant System heat losses.	RC P: MR and SC at PL4 RC T: SC at PL4, MP at PL10 and MR at PL4	1,2e,3m,5a
00:05:54 (0406:31)	Reactor Coolant System Pressurizer level increased beyond the range of the instrument indication (i.e. greater than 400 inches).	SC at PL4, MR (uncompensated) at PL3	1
00:06:24 (0407:01)	The Unit 2 Shift Supervisor again attempted to start Condensate Booster Pump 2B (CO-P-2B). The pump tripped immediately due to low suction pressure. Further attempts to start this pump were then abandoned.	AM at PL17, MR(A) and ST at PL3 AP norm/trip (Delay ≈ 5 minutes)	2e,5d,9X
00:06:54 (0407:31)	The letdown Cooler 1A (MU-C-1A) outlet high temperature alarm was received. A temperature of 139F was recorded.	AP norm/high (135F) (Delay ≈ 6 minutes)	2m
00:06:58 (0407:35)	The operator reduced Letdown flow in response to the Letdown Cooler 1A high temperature alarm and low Reactor Coolant Pressure. The Letdown flow returned to normal. A flow rate of 71.6 gallons per minute was recorded.	MR at PL3 AP Range 0 to 160 gpm (Delay ≈ 6 minutes)	2n,8f,8r,9d

Time	Event	Information Available to the Operator	Reference
00:07:29 (0408:06)	Reactor Building Sump Pump A (WDL-P-2A) started on a high Reactor Building Sump level. The increased sump level was due to the discharge from the Reactor Coolant Drain Tank Relief Valve (WDL-R1) which had been open for approximately 4 minutes. The Reactor Building Sump Pumps generally started about once per shift. For this reason the pump start would not have been considered extraordinary by the operator. Note: Each Reactor Building Sump Pump has a measured capacity of approximately 140 gallons per minute when there discharge is directed to the Miscellaneous Waste Holdup Tank (WDL-T-2).	AP on/off (Delay $\gtrsim$ 6 minutes)	2a
00:08:00 (0408:37) Approximate	The operator discovered the Emergency Feedwater Block Valves (EF-V12A and EF-V12B) were shut. EF-V12A and EF-V12B was opened admitting emergency feedwater to the Steam Generators. Indicated Steam Generator levels were approximately 10 inches just prior to feedwater addition (Figure 33). A rapid rise in Steam Generator A and B pressure was observed when feedwater was admitted to the Steam Generators (Figure 32). Addition of feedwater was also confirmed by a decrease in the Emergency Feedwater Pumps discharge pressure and by "hammering" and "crackling" heard from the Loose Part Monitoring System which was aligned to monitor Steam Generator A (Figure 59).	EF-V12A,B: ST at PLA SG L: MR at PLA (startup level), SC at PLA and PL5 (operating). AP low (24 inches)/norm (Delay $\gtrsim$ 6 minutes) SG P: MR at PLA, SC (P_{MS}) at PL17 AP low (860 psig)/norm/high (960 psig) (Delay $\gtrsim$ 6 minutes) EFP P: MR (P_{DISCH}) at PLA	1,2a,8c,9a,9e, 9A,9T,9X
00:08:12 (0408:49) Approximate	Reactor Building pressure began to increasing slowly from a value of -0.5 psig (Figure 51).	SC and MR at PL3	3j

Time	Event	Information Available to the Operator	Reference
00:08:33 (0409:19)	The Reactor Coolant System hot leg and cold leg temperatures began to decrease as a result of the feedwater added to the Steam Generators (Figure 6). Steam Generator pressure increased as the Steam Generators again functioned as a heat sink for the Reactor Coolant System (Figure 9).	RC T: SC at PL4, MP at PL10 and MR at PL4 SG P: MR at PL4, SC (P_{MS}) at PL17 ΔP low (860 psig)/norm/high (960 psig) (Delay ≃ 6 minutes)	1,3a
00:08:33 (0409:10) Approximate	The Reactor Coolant System pressure began to decrease, reflecting the decrease in Reactor Coolant System temperature (Figures 3 and 6).	MR and SC at PL4	1,2e,3a,3a
00:08:59 (0409:36)	Condensate Pump 1A (CO-P-1A) tripped. It is believed this pump trip was the result of an unsuccessful attempt to start Condensate Booster Pump 2A (CO-P-2A).	AN at PL17, MR(A) and ST at PL3 AP norm/trip and on/off (Delay ≃ 7 minutes)	2a,5d
00:09:05 (0409:42) Approximate	The operator recognized the condensate reject flow path was blocked and suspected the condensate polishing demineralizers to be the source of blockage. He attempted to establish the condensate flow by opening the condensate polishers bypass valve (CO-V12). The valve did not respond.	ST at PL17	5d,8c,8r,8h,9a
00:09:13 (0409:50)	The condensate Booster Pump suction header low pressure alarm was received. A pressure of 14.7 psig was recorded.	AP norm/low (15 psig) (Delay ≃ 7 minutes)	2a
00:10:00 (0410:37) Approximate	An Auxiliary Operator discovered a leaking flange in the Booster Pump Suction piping. After reporting this to the Control Room, he then closed Suction Valve (CO-V27A) for Condensate Booster Pump 2A (CO-P-2A).	Unit 2 Control Room notified of leaking flange	5d,8k,8b,9b, 9I,9P
00:10:18 (0410:55)	The Reactor Coolant System Pressurizer level indication came on scale.	SC at PL4, MR (uncompensated) at PL5	1

Time	Event	Information Available to the Operator	Reference

PLANT STATUS

The Reactor Coolant System pressure was near the saturation pressure of the reactor coolant hot leg temperature (Figure 5). This was the result of the Electromatic Relief Valve (RC-R2) remaining open, high letdown flow rate, reduced High Pressure Injection flow rate and emergency feedwater addition to the Steam Generators. Emergency feedwater flow was admitted to both Steam Generators which resulted in increased steam pressures and the recovery of both Steam Generators as Reactor Coolant System heat sinks as shown by the decrease in both Reactor Coolant System hot leg and cold leg temperatures (Figures 6 and 9). Steam pressures were controlled by the Integrated Control System Manipulation of the Turbine Bypass Valves. As heat was removed from the Reactor Coolant System, temperature and pressure decreased. The decreasing temperature in conjunction with the letdown flow rate and the reduced High Pressure Injection flow rate resulted in a decrease in reactor coolant volume. The Pressurizer level indication came on scale. Reactor Coolant Makeup Pump 1A (MU-P-1A) was operating providing Reactor Coolant Pump seal water and makeup flow. Reactor Coolant System letdown flow rate was approximately 70 gallons per minute.

Time	Event	Information Available to the Operator	Reference
00:10:19 (0410:18)	Reactor Building Sump Pump 2B (WDL-P-2B) started. The pump start setpoint is 4.616 feet from the bottom of the Reactor Containment Building Sump. The Reactor Building Sump Pump discharge was aligned to the Miscellaneous Waste Holdup Tank (WDL-T-2).	AP on/off (Delay ≈ 8 minutes)	2a,8a
00:11:20 (0411:01)	The letdown Cooler 1A (MU-C-1A) outlet temperature returned to normal. A value of 123.7F was recorded.	AP norm/high (135F) (Delay ≈ 8 minutes)	2a

Time	Event	Information Available to the Operator	Reference
00:10:24 (0411:01)	The operator stopped, restarted and again stopped Reactor Coolant Makeup Pump 1A (MU-P-1A) during the next four seconds.	AM at PL8, ST and MR(A) at PL3 AP norm/trip (Delay $\simeq$ 8 minutes)	2a,5c
00:10:48 (0411:25)	The Reactor Building Sump high level alarm was received. Setpoint is 4.650 feet from the bottom of the Reactor Building Sump.	AP norm/high (4.65 feet) (Delay $\simeq$ 8 minutes)	2a
00:11:43 (0412:20)	The operator started Reactor Coolant Makeup Pump 1A (MU-P-1A) after an unsuccessful attempt to start the pump at 00:11:40 (0412:17).	AM at PL8, ST and MR(A) at PL3 AP norm/trip (Delay $\simeq$ 8 minutes)	2a
00:12:00 (0412:37) Approximate	Condensate Hotwell level indication increased offscale high (greater than 50 inches).	SC at PL17	3r
00:13:13 (0413:50)	The operator stopped Decay Heat Removal Pumps 1A and 1B (DH-P-1A and DH-P-1B).	ST at PL13 and PL3, MR(P_{DISCH}) at PL8 AP on/off and norm/trip (Delay $\simeq$ 11 minutes)	2a,9d
00:13:27 (0414:04)	Condensate Booster Pump suction header pressure returned to normal. A pressure of 17.0 psig was recorded.	AP norm/low (15 psig) (Delay $\simeq$ 7 minutes)	2a
00:14:51 (0415:28)	The Reactor Coolant Drain Tank Rupture Diaphragm (WDL-U26) burst at about 192 psig (Figure 47). Design burst pressure is 200 ± 25 psig. The contents of the Reactor Coolant Drain Tank were released to the Reactor Containment Building atmosphere. This results in a rapid increase in Reactor Containment Building Pressure (Figure 91).	RCDT P: AM at PL8A (125 psig), MR at PL8A	1
00:15:40 (0416:17) Approximate	The operator stopped the two operating Heater Drain Pumps. These pumps had been maintaining the pressure in the condensate system.	AM at PL17, ST at PL5	5d,9e
00:15:43 (0416:20)	The condensate Booster Pump low discharge pressure alarm was received. A pressure of 307 psig was recorded.	AP norm/low (310 psig) (Delay $\simeq$ 13 minutes)	2a
00:15:57 (0416:34)	The Feedwater Pump low suction header pressure alarm was received. A pressure of 289.4 psig was recorded.	AP norm/low (Delay $\simeq$ 13 minutes)	2a

Time	Event	Information Available to the Operator	Reference
00:16:04 (0416:41)	The operator reset Condensate Pump 1A (CO-P-1A) trip circuit.	MR(A) and ST at PL5. AP norm/trip and on/off (Delay ≈ 13 minutes)	2a,5d
00:16:12 (0416:49)	The condensate Booster Pump suction header low pressure alarm was received. A pressure of 14.8 psig was recorded.	AP norm/low (15 psig) (Delay ≈ 13 minutes)	2a
00:19:23 (0420:00) Approximate	The Reactor Building Purge Air Exhaust Duct A radiation monitor (HP-R-225) recorded increase in radioactivity level. The level increased from 1×10^2 counts per minute to 5×10^2 counts per minute on the particulate channel. Additionally, there were slight radioactivity level increases indicated on: (a) Reactor Building Purge Air Exhaust Duct B (HP-R-226) - particulate monitor (b) Reactor Building Purge Air Exhaust Duct B (HP-R-[illegible]) - gas monitor (c) Auxiliary Building Purge Air Exhaust Duct (before filter) (HP-R-222) - gas monitor (d) Auxiliary Building Purge Air Exhaust Duct (after filter) (HP-R-222) - particulate monitor	MR and MP on PL12	3c,5b
00:21:59 (0422:36)	Reactor Coolant Pump 2A (RC-P-2A) full speed alarm was received intermittently.	AP norm/low (Delay ≈ 10 minutes)	2a
00:22:17 (0422:54)	The operator depressed the reactor trip pushbutton to confirm that the reactor tripped.	Reactor trip button at PL4 AP norm/trip (Delay ≈ 10 minutes)	2a
00:22:44 (0423:21)	The steam Generator A low level alarm cleared. A level of 26.6 inches was recorded.	AR at PL17 AP norm/low (23.8 inches) (Delay ≈ 20 minutes)	2a

Time	Event	Information Available to the Operator	Reference
00:24:00 (0424:37) Approximate	The Unit 2 Shift Supervisor reviewed the Reactor Coolant Drain Tank parameters and concluded that the drain tank rupture diaphragm (WDL-U20) had burst. This conclusion was based on the existing high temperature and low pressure in the drain tank coupled with a low discharge pressure on the Reactor Building Leakage Transfer Pump(s) 9A and/or 9B (WDL-P-9A and/or 9B).	RCDT P: NR and AR (125 psig) at PL8A RCDT T: NR at PL8A	8r,9a,9Z
00:24:58 (0425:35)	The Unit 2 Shift Supervisor requested the computer to print the outlet temperatures (RC-14-TE1, RC-14-TE2 and RC-14-TE3) of the Electromatic Relief Valve (RC-R2) and the Pressurizer Safety Valves (RC-R1A and RC-R1B). Respective values of 285.4F, 283.9F and 275.1F were indicated. The operator attributed the temperature levels to the normal residual of the discharge header following the initial opening and closing of the Electromatic Relief Valve (RC-R2) and believed the Electromatic Relief Valve (RC-R2) to be shut.	TP (Delay ≅ 8 minutes) AP high (200F)/horn (Delay ≅ 21 minutes) NP at PL10	3c,8a,8r,9a
00:25:00 (0425:37) Approximate	The operator placed the Turbine Bypass Valves (MS-V25A, MS-V25B, MS-V26A and MS-V26B) in manual and cracked them slightly open to promote additional cooling of the primary coolant.	MS-V25A/26A: NR and ST at PL8 MS-V25B/26B: NR and ST at PL3	8c
00:25:00 (0425:37) Approximate	High radiation alarms were received at the Radiation Monitor Panel from Intermediate Cooling Letdown Coolers A and B Radiation Monitors (IC-R-1091 and IC-R-1092). This alarm is periodically received because of its low alarm setpoint and sensitivity to background radiation. The Intermediate Cooling Letdown Cooler Radiation Monitors are physically located next to the Reactor Building Sump. It is believed that the alarms were the result of increased background radiation levels caused by the discharge of	NR and NP on PL12	5b,8r,9a,9Z

Time	Event	Information Available to the Operator	Reference
	reactor coolant from the Reactor Coolant Drain Tank to the Reactor Building sump or a small crud burst which sometimes accompanies a reactor trip.		
00:25:44 (0426:21)	The operator stopped Emergency Feedwater Pump 1 (EF-P-1).	MR (P_{DISCH}) @ PL4 AP low (setpoint = 875 psig)/norm (Delay $\simeq$ 22 minutes)	2a,9I
00:25:52 (0426:29)	Reactor Coolant Pump 1B (RC-P-1B) full speed alarm was received intermittently.	AP norm/low (Delay $\simeq$ 22 minutes)	2a
00:26:46 (0427:23)	The Steam Generator B low level alarm cleared. A level of 26.6 inches was recorded.	AB at PL17 AP norm/low (23.7 inches) (Delay $\simeq$ 25 minutes)	2a
00:29:23 (0430:00) Approximate	The Reactor Building Air Sample Line monitor (HP-R-227) gas channel count rate increased from 1×10^3 counts per minute to 3×10^4 counts per minute and then decreased to 1×10^3 counts per minute. This increase was not large enough to activate an alarm.	MR and UP on PL12	3a,5b
00:29:56 (0430:33)	The operator stopped Diesel Generators 1A and 1B (DF-X-1A and 1B) by manually tripping the fuel racks. They had been continuously running unloaded since the Engineer Safety Features actuation at 00:02:02 (0402:39).	AB, ST, MR (Amps, Volts, Hertz, KW/KVARS) at PL 26 and PL29 AP FALT/norm (Delay $\simeq$ 28 minutes)	2a,9X
00:30:12 (0430:49)	The plant staff requested the computer to printout the Memory Trip Review. The review consists of selected plant parameter 30 minutes prior to and after the transient.	UP (Delay $\simeq$ 0 minutes)	2a,9C
00:31:00 (0431:37)	The plant staff requested the computer to printout the Sequence of Events Review.	UP (Delay $\simeq$ 0 minutes)	2b,9C

Time	Event	Information Available to the Operator	Reference
00:32:23 (0433:00) Approximate	The following radiation monitor readings increased and then leveled off. (a) Gas channel of the Station Vent (HP-R-219) monitor (b) Iodine channel of the Fuel Handling Building Exhaust Duct (before filter) (HP-R-221A) monitor (c) Particulate channel of the Fuel Handling Building Exhaust Duct (before filter) (HP-R-221A) monitor (d) Iodine channel of the Fuel Handling Building Exhaust Duct (after filter) (HP-R-221B) monitor. (e) Particulate channel of the Fuel Handling Building Exhaust Duct (after filter) (HP-R-221B) monitor (f) Gas channel of the Fuel Handling Building Exhaust Duct (after filter) (HP-R-221B) monitor (g) Particulate Channel of the Hydrogen Purge Duct (HP-R-229) monitor (h) Iodine Channel of the Hydrogen Purge Duct (HP-R-229) monitor	MR and MP at PL12	3f,3b
00:32:36 (0433:13)	Incore thermocouple R-10 signal indication went out-of-range (Range = 0°F to 700°F).	AP T (Delay ≈ 24 minutes)	2a
00:36:08 (0436:45)	The operator stopped Emergency Feedwater Pump 2B (EF-P-2B) after filling both Steam Generators to an indicated level of about 38 inches on the startup range (Figure 41).	ST, MR($P_{D(DCH)}$) and MR(A) at PL4 AP on/off and low (875 psig)/norm (Delay ≈ 24 minutes)	1,2a
00:38:10 (0438:47)	The auxiliary operator stopped Reactor Building Sump Pump 2A (WDL-P-2A) to prevent overflowing the Miscellaneous Waste Holdup Tank (WDL-T-2).	AP on/off (Delay ≈ 31 minutes)	2a,3a,8a,9b

Time	Event	Information Available to the Operator	Reference
00:38:11 (0438:48)	The auxiliary operator stopped Reactor Building Sump Pump 2B (WDL-P-2B) to prevent overflowing the Miscellaneous Waste Holdup Tank (WDL-T-2). NOTE: The two Reactor Building Sump Pumps had operated for 31 and 28 minutes, respectively. Based on the measured capacity of each pump (approximately 140 gallons per minute for single pump operation), a maximum of 8260 gallons of water was transferred to the Auxiliary Building.	AP on/off (Delay $\simeq$ 31 minutes)	2a,5b,8k, 9b
00:43:34 (0444:11)	The operator requested the computer print pressurizer level differential pressure sensors 1, 2 and 3 in an attempt to determine if level indication was in error. Based on the close agreement of these three values the operator believed that the pressurizer level indication was correct.	UP (Delay $\simeq$ 0 minutes)	2c,9h
00:46:23 (0447:00) Approximate	Intermediate Cooling Letdown Cooler A monitor (IC-R-1092) increased from 1×10^3 counts per minute and eventually peaked at 2×10^4 counts per minute.	AB, MR and MP at PL12	3d,5b
00:55:00 (0455:37) Approximate	The operator shut Emergency Feedwater Valve (EF-V11B) after attempts to throttle the valve failed to stop the increasing water level in Steam Generator B.	EF-V11B: ST at PL4 SG L: MR at PL4 (startup range), SC at PL4 and PL3 (operating range)	8c,9a,9c
00:59:00 (0459:37) Approximate	The Auxiliary Operator manually opened Polisher Bypass Valve (CO-V12). The time required to open the valve was greater than expected due to the location of the valve, a missing handwheel and difficulty in the actual opening of the valve.	ST at PL17	5d,8k,8r,8h, 9b,9l,9P,9X
00:59:12 (0459:49)	Condensate Booster Pump suction header pressure returned to normal. A pressure of 80.2 psig was recorded.	AP norm/low (15 psig) (Delay $\simeq$ 48 minutes)	2a

Time	Event	Information Available to the Operator	Reference
00:59:21 (0459:58)	Condensate high temperature alarm was received. A temperature of 118.5F was recorded.	AP norm/high (Delay ≅ 48 minutes)	2a
01:00:47 (0501:24)	The operator stopped Circulation Water Pumps 1B, 1C, 1D and 1E (CW-P-1B, 1C, 1D and 1E) to activate a logic circuit which transferred steam generator pressure from the Turbine Bypass Valves (MS-V25A, MS-V25B, MS-V26A and MS-V27B) to the Power Operated Emergency Main Steam Dump Valves (MSV-3A and MSV-3B) This was done to stop steaming to the condenser which was increasing Hotwell Level. Steam Generator Pressure Control was then maintained by intermittent use of MS-V3A and MS-V3B until MSV-3B was shut at 1:26:23 (0527:00) when Steam Generator B steam line was isolated and until MSV-3A was shut at 02:54:50 (0655:27) after regaining condensor hotwell level control.	MR(A and F) and ST at PL17 AP on/off (Delay ≅ 49 minutes) MS-V25A/26A: MR and ST at PL3 MS-V25B/26B: MR and ST at PL3	2a,8c,9e,9A
01:09:23 (0510:06) Approximate	A Radiation/Chemistry Technician drew a Reactor Coolant System sample for boron analysis per procedural requirement after a reactor trip. The boron analysis results indicated a boron concentration of slightly over 700 parts per million boron.	Unit 2 Control Room notified of boron analysis	9t,9B
01:10:54 (0511:31)	The Reactor Building Air Cooling Coil B Emergency Discharge Temperature signal indication cycled in and out of computer range (Range 0F to 200F) within the 30 second scan time. This trend continued intermittently for the remainder of March 28, 1979 and is believed to be attributed to a periodic malfunction in the complete input signal.	MR(F) at PL33 AP bad/norm (0F to 200F) (Delay ≅ 39 minutes)	2a
01:12:11 (0512:48)	The operator requested the computer to print the current alarm conditions relative to the Reactor Coolant Pumps. The following alarms were received.	UP (Delay ≅ 8 minutes)	2c

Time	Event		Information Available to the Operator	Reference
	RC-P-1A:	Oil Lift Pump Discharge Pressure		
		Full Speed		
		Backstop Oil Flow		
	RC-P-2A:	Seal Leak Tank Level		
		Oil Lift Pump Discharge Pressure		
		Full speed		
		Backstop Oil Flow		
	RC-P-1B:	Oil Lift Pump Discharge Pressure		
		Full Speed		
		Backstop Oil Flow		
	RC-P-2B:	Oil Lift Pump Discharge Pressure		
		Backstop Oil Flow		

PLANT STATUS

Both Reactor Coolant System hot leg temperatures and pressures had decreased until they stabilized at a saturation temperature-pressure relationship of 542F and 1050 psig. The Reactor Coolant System loop flow rates had decreased from about 69 million pounds per hour to approximately 47 million pounds per hour and continued to decrease (Figure 17). Reactor Coolant Makeup Pump 1A (MU-P-1A) was operating. Letdown flow was in the normal range. Pressurizer level was approximately 352 inches. The Electromatic Relief Valve (RC-R2) was open releasing reactor coolant to the Reactor Containment Building via the Reactor Coolant Drain Tank (WDL-T-3). The Containment Building pressure and temperature (Point 13) had increased from -0.5 psig and 120F to 2.5 psig and 170F, as a result of releasing the contents of the Reactor Coolant Drain Tank to the Reactor Containment Building atmosphere (Figure 51). The operator was having difficulty controlling the level of Steam Generator B and had shut Emergency

Time	Event	Information Available to the Operator	Reference
	Feedwater Valve (FW-V11B). Steam Generator Pressure Control was being accomplished using power operated Emergency Main Steam Dump Valves MSV-3A and MSV-3B.		
01:13:29 (0514:06)	The Unit 2 Shift Supervisor stopped Reactor Coolant Pump 2B (RC-P-2B) to preclude the possibility of damage to the Reactor Coolant Pump from operation near the minimum net positive suction head limits. Additional factors which contributed to the operator's decision were high pump vibration and erratic reactor coolant flow rate.	RC-P-2B: ST, MR(A) and MR(F) at PLA, AN at PL8 AP norm/trip (Delay ≅ * minutes) RCP V: AN at PL8, AN and MR at PL10 RC F: MR and SC at PLA	2b,4g,8a,8c, 8d,8r,9h,9A
01:13:42 (0514:19)	The Unit 2 Shift Supervisor stopped Reactor Coolant Pump 1B (RC-P-1B) to preclude the possibility of damage to the Reactor Coolant Pump from operation near the minimum net positive suction head limits. Additional factors which contributed to the operator's decision were high pump vibration and erratic reactor coolant flow rate.	RC-P-2B: ST, MR(A) and MR(F) at PLA, AN at PLA AP norm/trip (Delay ≅ * minutes) RCP V: AN at PL8, AN and MR at PL10 RC F: MR and SC at PLA	2b,4g,8a,8c, 8d,8r,9h,9A
01:14:15 (0514:51)	Steam Generator B steam pressure rapidly decreased from approximately 950 psig to approximately 145 psig over the next 28 minutes. This was in response to reduced heat transfer in Loop B as a result of stopping reactor coolant pumps 1B and 2B (RC-P-1B and RC-P-2B). Concurrent with this the water level in Steam Generator B starts to rise. It is believed that the rise in water level in Steam Generator B was the result of an imbalance in heat flow between Steam Generators A and B.	SG P: MR at PLA, SC at PL17 SG L: MR (Startup Range) at PLA	1,3a,8c

*See entry at time 02:47:31 (0648:08)

Time	Event	Information Available to the Operator	Reference
01:14:23 (0515:00)	The Unit 1 Shift Supervisor received the results of the first boron analysis which indicated that the boron concentration in the Reactor Coolant System was slightly over 700 parts per million. This value was felt to be questionable since the boron concentration was in excess of 1000 parts per million Boron prior to the trip and high pressure injection had been from the borated water storage tank. Based on these factors another boron analysis was requested. Note: The actual Boron concentration in the Reactor Coolant System was in excess of 1000 parts per million. The sample is believed to have been diluted by distillation in the Letdown System. This however, was not known by the Plant Operators until several hours later.	Unit 2 Control Room notified of boron analysis.	8r,8a,9b,9p, 9t,9B
01:17:00 (0517:37) Approximate	The operator closed Emergency Feedwater Valve (EF-V12B) to halt the rise in Steam Generator B water level which had reached 90 inches. As level continued to gradually trend upward, the operator also closed Emergency Feedwater Crossconnect Valve (EF-V3B). Based on these indications the operator suspected that a Reactor Coolant side to Feedwater side leak might exist in Steam Generator B.	EF-V12B: ST at FLA EF-V3B: ST at FLA SG L: MR (Startup Range) at FLA	1,8c,9a,9c
01:20:31 (0521:08)	The Unit 1 Shift Supervisor requested the computer print the outlet temperature (RC-10-TE, RC-10-TE2, and RC-10-TE3) of the Electromatic Relief Valve (RC-R2) and the Pressurizer Safety Valves (RC-R1A and RC-R1B). The recorded values were 283.0F, 211.3F, and 218.6F, respectively. The operator continued to believe that the Electromatic Relief Valve (RC-R2) was shut.	UP (Delay ≅ 0 minutes) AP high (200F)/norm (Delay ≅ * minutes) MP at PL 1B	2c,9h

*See entry at time 02:47:31 (0648:08)

Time	Event	Information Available to the Operator	Reference
01:26:23 (0527:00)	The operator shut Steam Generator B Main Steam Isolation Valve (MS-V4B and MS-V7B). He suspected a Steam Generator B to Reactor Building leak based on the large difference in steam pressure of approximately 300 psig between the two Steam Generators, the variations of flow and level experienced while controlling Steam Generator B and the increased Reactor Building pressure and temperature. Steam Generator B was isolated completely at this time.	MS-V4B/7B: ST at PL15 MS-V25B/26B MR and ST at PL3	4b,8c,9c,9k, 9X
01:29:23 (0530:00) Approximate	The Unit 1 Shift Supervisor directed an auxilliary operator to energize Core Flood Tank 1A and 1B (CF-T-1A and CF-T-1B) Breaker to give the control room the capability to close isolation valves (CF-V1A and CF-V1B). There are no records which indicate the Core Flood Tanks were ever isolated. It was felt that the system was solid since the pressurizer level was high and the tendency was therefore to letdown as much as possible and not to add makeup water. The Core Flood Tanks were later floated on the Reactor core by depressuring the Reactor Coolant System at 07:38:57 (1139:34).	CF-V1A/1B Breaker Status: ST at PL8	8a,8k,9h,9A, 9E,9I,9T,9Z
01:30:00 (0530:37) Approximate	The reactor out-of-core Intermediate Range Channel (NI-4) indication increased from a minimum detectable indication of less than 1.0×10^{-11} amperes to approximately 1.6×10^{-11} amperes (Figure 56). Correspondingly, the out-of-core Source Range Channel (NI-1) indication increased from about 1.6×10^{4} to approximately 2.0×10^{4} counts per second (Figure 56). The indicated increase was not due to reactor core neutron flux level increases but rather an increase in neutron leakage from the reactor core as a result of the formation of steam in the reactor vessel.	NI-1: MR and SC at PLA NI-4: MR and SC at PLA	3j,3o

*See entry at time 02:47:31 (0648:08)

Time	Event	Information Available to the Operator	Reference
01:32:03 (0532:40) Approximate	Steam Generator B Water Level rapidly increased from 67 to 113 inches on the startup range in approximately three minutes. It is believed that the Steam Generator B water level increase was due to operator action and not the Result of a Reactor Coolant side to Feedwater side leak.	SG L: MR (Startup Range) at PLA	1,8c
01:32:03 (0532:40) Approximate	Steam Generator A boiled dry (Figure 11). This was indicated by a steadily decreasing steam generator pressure while Reactor Coolant Loop A hot leg and cold leg temperatures were increasing.	SG P: MR at PLA and SC at PL 17 SG L: MR (Startup Range) at PLA AP Low (23.8 inches) /norm (Delay $\simeq$ *)	1,5a,9T
01:34:02 (0534:49) Approximate	The operator commenced raising Steam Generator A level from 8 inches on the startup range to 50% on the operating range in preparation for establishing of natural circulation (Figures 41 and 44). This was indicated by a rapid pressure increase in Steam Generator A with a corresponding decrease in Reactor Coolant Loop A hot leg and cold leg temperatures.	SG L: MR (Startup Range) at PLA, MR (Wide Range) at PLA, SC (Operate Range) at PLA and PL5 RC T_C: MP at PL10	1,8c,8r,9a
01:40:37 (0541:14)	The Operator stopped the Reactor Coolant Pump 2A (RC-P-2A) to preclude the possibility of damage to the Reactor Coolant Pump from operation near the minimum net positive suction head limits. Additional factors which contributed to the Operator's decision were high pump vibration and erratic reactor coolant flow rate.	RC-P-2B: St, MR(A) and MR(F) at PLA, AN at PL8 AP norm/trip (Delay $\simeq$ * minutes) RCP V: AN at PL8, AN and MR at PL10 RC F: MR and SC at PLA	2b,4g,8c,9d, 9e,9k
01:40:45 (0541:22)	The Operator stopped Reactor Coolant Pump 1A (RC-P-1A) to preclude the possibility of damage to the Reactor Coolant Pump from operation near the minimum net positive suction head limits. Additional factors which contributed to the Operator's decision were high pump vibration and erratic reactor coolant flow rate.	RC-P-2B: St, MR(A) and MR(F) at PLA, AN at PL8 AP norm/trip (Delay $\simeq$ * minutes) RCP V: AN at PL8, AN and MR at PL10 RC F: MR and SC at PLA	2b,4g,8c,9d, 9e,9k
01:41:00 (0541:37) Approximate	The operator manually initiated the Safety Injection portions of Engineered Safety Feature trains A and B to supply additional	SI: AN at PL13, ST at PL3 and PL13 AP norm/actuation (Delay $\simeq$ * minutes)	7,8d,9d

*See entry at time 02:47:31 (0648:08)

Time	Event	Information Available to the Operator	Reference
	cooling water to the reactor core. Makeup Pump C (MU-P-1C) started automatically. Makeup Pumps A and C (MU-P-1A and MU-P-1C) are now operating. Note: The duration of this manual Safety Injection is not known because of the loss of alarm printer data during the period from 01:13:22 (0513:29) to 02:47:31 (0648:08). However based on the sequence of events printout Makeup Pump 1C was stopped prior to 02:28:41 (0629:28).	MU-P-1C: AN at PL8, ST and MR(A) at PL3 AP norm/trip (Delay $\gtrsim$ * minutes)	
01:41:00 (0541:37) Approximate	The reactor out-of-core Intermediate Range Channel (NI-4) indication rapidly increased from 1.6×10^{-11} to 2.3×10^{-11} amperes and then rapidly decreased to a minimum detectable indication of 1.0×10^{-11} amperes (Figure 56). The Reactor Out-of-Core Source Range Channel (NI-1) indication showed a corresponding rapid increase from 2.0×10^4 to 5.2×10^4 counts per second and then rapid decreased to 1.5×10^3 counts per second. After decreasing to 1.5×10^3 counts per second the Reactor Out-of-Core Source Range Channel (NI-1) immediately started increasing. These responses are attributed to changes in moderation density caused by liquid displacing steam in the reactor vessel.	NI-1: MR and SC at PL4 NI-4: MR and SC at PL4	3k
01:41:23 (0542:00)	A Radiation/Chemistry Technician took a condenser vacuum pump exhaust sample for Germanium Lithium analysis per procedural requirements after a reactor trip. The Germanium Lithium indicated radioactivity levels were not above background. The results of the analysis are listed below. Potassium 40 — 4.840 E-06 mCi/ml Cobalt 50 — 1.555 E-06 mCi/ml Xenon 135 — 1.836 E-07 mCi/ml Total: — 6.585 E-06 mCi/ml	Unit 2 Control Room notified of Condenser Vaccuum pump results.	4e,97

*See entry at time 02:47:31 (0648:08)

Time	Event	Information Available to the Operator	Reference
PLANT STATUS	The Reactor Coolant System had no forced Reactor Coolant System flow. All Reactor Coolant Pumps (RC-P-1A, RC-P-2A, RC-P-1B and RC-P-2B) had been stopped to preclude the possibility of damage from operation near net positive suction head limits. Additional factors which contributed to the decision to stop the Reactor Coolant Pumps were high pump vibration and an erratic reactor coolant flow rate. (Figure 17). The Reactor Coolant System average temperature and pressure were approximately 534F and 1000 psig, respectively (Figures 13 and 27). Letdown flow rate was in the normal range. Pressurizer level was 354 inches. Reactor Coolant Makeup Pumps 1A and 1C (MU-P-1A and MU-P-1C) were operating in the High Pressure Injection mode. The operator was attempting to establish natural circulation flow to cool the reactor core. Steam Generator B was isolated because of a suspected Reactor Coolant side to Feedwater side leak. Steam Generator B pressure and level were 190 psig and 95 inches respectively. Steam Generator A was steaming to the atmosphere via the power operated Emergency Main Steam Dump Valve (MS-V3A). Steam Generator A pressure and level were 730 psig and 107 inches respectively.		
01:44:23 (0545:00) Approximate	The Unit 2 Superintendent of Technical Support requested that a sample be taken of the Reactor Containment Building atmosphere. The normal sampling point on Unit 2 is from radiation monitor HP-R-227. Upon removing the charcoal filter to obtain the sample, water poured out of the monitor. The Radiation Technical immediately replaced the filter. The sample could not be taken under these conditions. It was speculated that a steam environment existed in Unit 2 Reactor Containment Building.	Unit 2 Control Room notified of difficulty in obtaining a Reactor Containment Building atmosphere sample.	8a,5b,9h,9i,5p

*See entry at time 02:47:31 (0648:00)

Time	Event	Information Available to the Operator	Reference
01:46:23 (0545:00) Approximate	A Radiation/Chemistry Technician drew a reactor coolant system sample for analysis of the boron concentration per the request of the Unit 1 Shift Supervisor. The boron analysis yielded a value of approximately 400 parts per million Boron. The chemist questioned this result and asked the other chemist on shift to run an independent analysis on another sample. This chemist also obtained similar results. The two values recorded were 402 and 407 parts per million Boron.	Unit 2 Control Room notified of boron analysis results.	9t,93
01:51:27 (0552:04) Approximate	Reactor Coolant System Loop A hot leg temperature began to increase, reflecting steam formation in the upper reactor core region (Figure 22).	SC at PLA, MP at PL10 and MR at PLA	1,5a
01:54:00 (0554:37) Approximate	The reactor out-of-core Intermediate Range Channel (NI-4) indication increased from less than 1.0×10^{-11} amperes to approximately 1.0×10^{-10} amperes (Figure 56). A corresponding increasing trend was recorded on the reactor out-of-core Source Range Channel (NI-1) indication (Figure 56). The indicated increase was not due to incore neutron flux level increases but rather an increase in neutron leakage from the reactor core as a result of the steam formed in the reactor vessel. The formation of steam was contributed to (1) increased reactor core temperatures, (2) throttled Reactor Coolant Makeup Pump flow, (3) the absence of Reactor Coolant System flow, and (4) the decreased Reactor Coolant System pressure which resulted from the open Electromatic Relief Valve (RC-R2) and the increased Reactor Coolant System cold leg density caused by filling Steam Generator A. After the	NI-1: MR and SC at PLA NI-3: MR and SC at PLA NI-4: MR and SC at PLA	3h,5a

*See entry at time 02:47:31 (0648:08)

Time	Event	Information Available to the Operator	Reference
	reactor out-of-core intermediate range channel (NI-4) reached approximately 2.5×10^{-11} amperes the other reactor out-of-core intermediate range channel (NI-3) increased from 1.0×10^{-11} amperes to approximately 4.0×10^{-11} amperes.		
01:59:23 (0600:00) Approximate	The relieving Unit 2 Shift Supervisor directed an operator to inspect the pressurizer heater breaks locally at their control cabin. Upon entering the area, the operator encountered hot and humid conditions which he attributed to tripped ventilation fans. After restarting these fans he then inspected the pressurizer heater and found four of the thirty-nine breakers tripped which he then reset. The pressurizer heater breakers had a history of tripping which made it necessary to normally check breaker status each shift.	Heaters: ST at [illegible] AP norm/trip (Delay $\approx$ 8 minutes)	9K,9M,9T
02:00:00 (0600:37) Approximate	A telephone conference call was made between Unit 2 Superintendent Technical Support (Unit 2 control room) and the Manager-Generating Station Nuclear, the Metropolitan Edison Company Vice President of Generation and the Babcock and Wilcox Resident Engineer. The telephone conversation lasted approximately 20 minutes.	Conference call conducted in Unit 2 Control Room.	9u,9w,9z,10
02:01:23 (0602:00)	A Radiation/Chemistry Technician drew a reactor coolant system sample for Germanium Lithium analysis after observing an increasing radiation level in the chemistry sampling room. A gross beta-gamma and sodium analysis yielded 4.0 mCi/cc and 0.153 ppm respectively. A gross beta gamma analysis performed prior to the reactor trip yielded a value of 0.4 mCi/cc. However, a Germanium Lithium analysis of this sample was not performed.	Unit 2 Control Room notified of gross beta-gamma and sodium analysis results.	4o,9h,9t,9B

*See entry at time 02:47:31 (0648:00)

Time	Event	Information Available to the Operator	Reference
02:02:00 (0602:37)	Condenser Hotwell Level indication came back on scale (less than 50 inches).		3r
02:02:26 (0603:03) Approximate	Reactor Coolant System Loop B hot leg temperature began increasing (Figure 28).	SC at PL4, MP at PL10 and MR at PL4	1
02:03:42 (0604:19)	Steam Generator A level indication reached 50% on the operating range (Figure 45). This level was established by the operator to induce natural circulation.	SC at PL4 and PL3	1,8c,8r
02:10:00 (0610:37) Approximate	The Unit 2 Shift Supervisor directed the operator to initiate emergency boration of the Reactor Coolant System via both makeup addition valves (MU-V10 and MU-V127) using both boric acid transfer pumps 4A and 4B (CA-P-4A and CA-P-4B). This was done in response to an increased neutron flux indication on the source and intermediate range channels in conjunction with the results of two boron analysis which indicated the boron concentration in the Reactor Coolant System at approximately 400 parts per million. Based on these indications it was believed that a reactor restart was in progress. Emergency boration continued until it was terminated at approximately 03:22:00 (0722:37).	MU-V10: ST at PL3 CA-P-4A/4B: ST at PL3 Boration: Batch Controller at PL3	3k,8h,8r,9d, 9k,9b,9j,9A, 9C,9I,9V,9X
02:10:42 (0611:19)	Reactor Coolant System Loop A hot leg temperature indication increased offscale, greater than 620F (Figure 23).	AN at PL8 (high at 612F), SC at PL4, MP at PL10 and MR at PL4	1
02:14:23 (0615:00) Approximate	The Reactor Building Air Sample (HP-R-227) particulate channel increased. It eventually went off scale, at 02:34:23 (0635:00).	AN, MR and MP at PL12	3a

*See entry at time 02:47:31 (0648:08)

Time	Event	Information Available to the Operator	Reference
02:17:53 (0618:30)	The Unit 2 Relieving Shift Supervisor requested the computer print the outlet temperatures (RC-10-TE1, RC-10-TE2 and RC-10-TE3) of the Electromatic Relief Valve (RC-R2) and the Pressurizer Safety Valves (RC-R1A and RC-R1B). The recorded values were 228.7F, 189.5F, and 194.2F, respectively.	UP (Delay ≈ 0 minutes) AP high (200F)/norm (Delay ≈ * minutes) MP at PL10	2c, 8a,9E
02:19:00 (0619:37) Approximate	The Unit 2 Relieving Shift Supervisor directed the operator to shut the Electromatic Relief Block Valve (RC-V2), which stopped the reactor coolant leakage through the Electromatic Relief Valve (RC-R2). The operator noted that the Reactor Building pressure decreased rapidly (Figure 51)	RC-V2: ST open/shut at PLA RC-R2: ST open/shut command at PLA RB P: SC at PL13	2c,3j,3a,8a, 9c,9E
02:19:00 (0619:37) Approximate	Reactor Coolant System pressure increased from 680 psig to 2130 psig during the next 41 minutes. Reactor Coolant System Pressure was then maintained at 2130 psig.	AM (Low-2055 psig and Low/Low - 1900 psig) at PL8 MR and SC at PLA	3a
02:19:59 (0620:36)	The Unit 2 Relieving Shift Supervisor requested the computer print the outlet temperature (RC-10-TE1) of the electromatic relief valve (RC-R2) three times. The recorded values were 227.6F at 02:19:59 (0620:36), 226.6F at 02:20:36 (0621:13) and 220.1F at 02:23:32 (0624:09).	UP (Delay ≈ 0 minutes) AP high (200F)/norm (Delay ≈ * minutes) MP at PL10	2c,9E
02:24:23 (0625:00) Approximate	The Reactor Building Air Sample (HP-R-227) iodine channel increased and eventually went off scale at 02:37:23 (0638:00).	AM, MR and MP at PL12	3a,5b
02:27:23 (0628:00)	The alarm printer malfunctioned. The alarm printer function transferred to the utility printer. The alarm printer was 1 hour and 25 minutes behind in logging data (Figure 58).	Paper feed problem in typewriter AP (Delay ≈ 0 minutes)	2a,2b,9d

*See entry at time 02:47:31 (0648:08)

Time	Event	Information Available to the Operator	Reference
02:29:21 (0629:58)	Reactor Coolant Loop B hotleg temperature indication increased offscale, greater than 620F (Figure 28).	AR at PL8 (high at 612F), SC at PL4, MP at PL10 and MR at PL4	1
02:30:00 (0630:37) Approximate	Self powered neutron detectors reading on backup incore detector recorders A and B came back on scale and started to rise rapidly. These recorders monitor 36 of the 364 incore detectors available.	MP at PL14 AP bad/norm (0 to 2000nA) (Delay ≥ 8 minutes)	3r
02:31:23 (0632:00) Approximate	The Incore Instrument Pool Area Monitor (HP-R-213) reading began to increase.	AR, MR and MP at PL12	3a,5b
02:33:02 (0633:39)	The operator started increasing Steam Generator B level from 93 inches on the Startup Range to 60% on the Operating Range (Figure 45).	MR (Startup Range) at PL4, MR (Wide Range) at PL4 SC (Operate Range) at PL4 and PL3	1,9a
02:38:23 (0639:00) Approximate	The Letdown Cooler A monitor (IC-R-1092) indicated radiation level increased offscale. The levels indicated on the following radiation monitors began to increase: (a) Makeup Tank Area Monitor (HP-R-206) (b) Fuel Handling Building S. (HP-R-210) (c) Reactor Building Dome (HP-R-214)	AR, MR and MP at PL12	3a,3d,5b
02:39:23 (0640:00) Approximate	The Unit 1 hot machine stop area monitor (RM-G4) reached the alarm setpoint (2.5 mR/hr). An inspection of the area by Radiation/Chemistry Technicians revealed that the Unit 2 Reactor coolant system sample lines were the source of the increased radiation levels. It takes approximately forty minutes for a representative sample on Unit 2 to reach this area due to the sample line length and flow rates employed.	Unit 2 Coolant Room Notified of Increasing Radiation Levels on Unit 2 Sample Line.	5b,9h,9p,10
02:44:23 (0645:00) Approximate	Incore Instrument Pool Area Monitor (HP-R-213) indication increased offscale high. The levels indicated on the following	AR, MR and MP at PL12 *See entry at time 02:47:31 (0648:08)	3a,3c,3e,5b

Time	Event	Information Available to the Operator	Reference
	radiation monitors began to increase. (a) Reactor Building Purge Air Exhaust Duct (HP-R-225)-particulate (b) Reactor Building Purge Air Exhaust Duct B (HP-R-226)-particulate (c) Auxiliary Building Purge Air Exhaust (HP-R-223)-particulate, gas, and iodine (d) Auxiliary Building Heating & Ventilation monitor gas channel (indication was off scale within 30 minutes). (e) Reactor Building Air sample (HP-R-227) gas channel (indication was off scale within 10 minutes).		
02:45:00 (0645:37) Approximate	Several radiation alarms were received at the Control Room Radiation Monitor Panel.	AN, MR and MP at PL12	8q,9a,9h
02:45:23 (0646:00)	A Radiation/Chemistry Technician took a reactor coolant sample. A Germanium Lithium analysis was done at 02:51:23 (0652:00) and indicated a total gross beta-gamma activity of 140.7 mCi/ml. The results of the analysis are listed below.	Unit 2 Control Room notified of gross beta-gamma analysis results	4a

Krypton 85	1.437 E+01 mCi/ml
Rubidium 88	7.847 E+00 mCi/ml
Molybdenum 95	6.277 E-02 mCi/ml
Molybdenum 99	1.496 E-01 mCi/ml
Iodine 131	1.731 E+01 mCi/ml
Iodine 132	2.294 E+01 mCi/ml
Iodine 133	3.302 E+01 mCi/ml
Iodine 134	9.846 E+00 mCi/ml
Iodine 135	2.147 E+01 mCi/ml
Xenon 133	5.807 E+00 mCi/ml
Xenon 133	2.260 E+00 mCi/ml
Cesium 136	2.856 E-01 mCi/ml
Cesium 137	2.944 E-01 mCi/ml
Cesium 136	5.121 E+00 mCi/ml
Total:	1.407 E+02 mCi/ml

Time	Event	Information Available to the Operator	Reference
02:45:23 (0646:00) Approximate	Fuel Handling Building Monitor (HP-R-218) began to increase.	MR and MP at PL12	3b,5b
02:46:23 (0646:00) Approximate	The operator attempted to start Reactor Coolant Pump 1A (RC-P-1A). The pump would not start.	ST, MR(A), MR(F) and DC(F) at PL4, AN at PL8, AP norm/trip (Delay * a) *See entry at time 02:47:31 (0648:08)	8c,8r,9a

Time	Event	Information Available to the Operator	Reference
02:47:33 (0648:08)	The operator initialized the alarm summary function to obtain current alarm data. As a result the alarm summary data from 01:13:22 (0513:59) to 02:47:31 (0648:08) was deleted.	AP (Delay ≈ 0 minutes) Current Alarm station printed out.	3a
02:47:31 (0648:08)	The current alarm status indicated many of the self-powered neutron detectors were going offscale between levels 3 through 7 (i.e. approximately 3 feet to the top of the core). This has been attributed to the high temperatures experienced in the core.	AP bad/norm (0 to 2000 nA) (Delay ≈ 0 minutes)	3a
02:49:23 (0650:00) Approximate	The following radiation monitors indication were increasing steadily and by 03:34:23 (0735:00) all were either off scale or cycling near their upper range. (a) Gas channel of the Station Vent (HP-R-219) monitor (b) Fuel Handling Building Exhaust Duct (HP-R-221,A & B) (c) Hydrogen Purge Duct (HP-R-229) particulate & iodine channels.	AB, MB and MP at PL12	3e,3f,3h
02:49:23 (0650:00) Approximate	The Unit 2 Shift Supervisor declared a Site Emergency upon receiving high level radiation alarms from the Reactor Building Monitor (HP-R-214). Notification of offsite authorities and organizations was initiated.	AB, MB and MP at PL12	6f,8d,8r,8s, 9E,99
02:50:00 (0630:37) Approximate	The air supply was restored to condensate reject valve (CO-V57 which had previously failed closed at 00:00:15 (0400:52) when its instrument air line was severed due to a "water hammer" in the condensate piping. This was done to allow water in the Hotwell to be transferred to condensate storage tanks 1A and 1B (CO-T-1A and 1B). A rapid decrease in hotwell level is experienced.	SC at PL5	3r,99
02:52:57 (0652:34)	The operator attempted to start Reactor Coolant Pump 2A (RC-P-2A). The pump would not start.	ST, MB(A), MB(F) and SC(F) at PL4, AP or PL8, AP norm/trip (Delay ≈ 3 minutes)	2a,8c,9a

Time	Event	Information Available to the Operator	Reference
02:52:26 (0653:03)	Condenser hotwell level indication returned to normal. A level of 34.94 inches was indicated.	MR(L) at PL5, AP low (22.5 inches)/normal/high (36 inches) (Delay ≃ 3 minutes)	2a
02:53:16 (0653:53)	The operator attempted to start Reactor Coolant Pump 1B (RC-P-1B). The pump would not start.	ST, MR(A), MR(F) and SC(F) at PLA, AN at PL8, AP norm/trip (Delay ≃ 4 minutes)	2a,8c,9a
02:54:09 (0654:46)	The operator started Reactor Coolant Pump 2B (RC-P-2B). A reactor coolant system flow rate of 10 million pounds per hour was experienced for approximately 5 seconds.	ST, MR(A), MR(F) and SC(F) at PLA, AN at PL8, AP norm/trip (Delay ≃ 5 minutes)	1,2a,3a,8c, 9a,9j
02:54:15 (0654:52)	Steam Generator B steam pressure rapidly increases from approximately 160 psig to approximately 728 psig in the next two minutes.	MR at PLA, SC at PL17	1
02:54:19 (0654:56)	The operator de-energized Pressurizer Heater Groups 1 through 5. Eleven pressurizer heater groups were available at this time.	ST at PLA AP norm/trip (Delay ≃ 5 minutes)	2a
02:54:23 (0655:00) Approximate	Waste Gas Discharge Monitor (WDG-R-1480) increased and went off scale. Note: This monitor is located in the Auxiliary Building at an elevation of 305'0".	MR and MP at PL12	3a,5b
02:54:23 (0655:00)	Unit 1 control room was notified of the site emergency in effect in Unit 2.	Announcement made on Plant Page System	4a,4f,8a
02:54:33 (0655:10)	The following Incore Thermocouple Temperatures decreased to less than 700F over the next seven minutes. The increased reactor core cooling was a result of Reactor Coolant Pump 2B (RC-P-2B) operation. 7B = 623.7F 10C = 599.0F 1H = 596.0F 2G = 623.0F 12O = 624.9F 13G = 670.6F 14M = 653.6F 7R = 679.7F	AP norm/bad (out of range 0F to 700F) (Delay ≃ 6 minutes)	2a

Time	Event	Information Available to the Operator	Reference
	19C = 577.1F 5D = 699.5F*		
	11L = 584.0F 3F = 681.9F*		
	NOTE: (*) This Incore Thermocouple Temperature cycled near 700F until Reactor Coolant Pump 2B (RC-P-2B) was stopped.		
02:54:40 (0653:17) Approximate	Reactor Coolant system pressure rapidly increased from approximately 1260 psig to approximately 2140 psig (Figure 4).	AM (Low-2055 psig and Low/Low-1900 psig) at PL8	3a
02:54:50 (0653:27)	Safety Injection logic of the Engineered Safety Features trains A and B reset on increasing Reactor Coolant System pressure.	AM at PL13, ST at PL3 and PL13 AP norm/trip (Delay ≈ 6 minutes)	2a
02:54:50 (0653:27)	The operator started Circulating Water Pump 1B (CW-P-1B). Steam pressure control was automatically transferred from the power operated Emergency Main Steam Dump Valves (MS-V3A and MS-V3B) to the Turbine Bypass Valves (MS-V25A, MS-V25B, MS-V26A and MS-V26B) which were in manual.	MR(A and P), ST at PL17 AP on/off (Delay ≈ 6 minutes) MS-V25A/26A MR and ST at PL8 MS-V25B/26B MR and ST at PL8	2e,8e,9j
	Note: Two circulating water pumps had been placed in service during the time period from 01:13:22 (0513:29) to 02:47:31 (0648:08) when no alarm printer data was available.		
02:54:56 (0655:31) Approximate	Condenser Vacuum Pump Exhaust Radiation Monitor (VA-R-748) increased rapidly from 1×10^2 to 8×10^3 counts per minute at 03:03:40 (0700:17).	AM, MR, and MP at PL12	1,3e,5b
	NOTE: VA-R-748 is located in the Turbine Building at an elevation of 281'6".		
02:55:00 (0655:37) Approximate	The reactor out-of-core Intermediate Range Channels (NI-3 and NI-4) indications decreased rapidly to less than 1.0×10^{-11} amperes (minimum detectable level) (Figure 56). The reactor out-of-core	NI-1: MR and SC at PLA NI-3: MR and SC at PLA NI-4: MR and SC at PLA	3k,5a

Time	Event	Information Available to the Operator	Reference
	Source Range Channel (NI-1) indication showed a corresponding marked decrease which indicated the steam in the reactor core region was displaced by liquid (Figure 56). The steam was displaced from the reactor vessel when Reactor Coolant System flow was established. After the Reactor out-of-core Source Range Channel (NI-1) indication rapidly decreased, it immediately started increasing.		
02:55:12 (0655:46)	The operator initiated pressurizer spray flow to stop the rapid rise in Reactor Coolant System pressure. Pressurizer spray flow was maintained until 03:13:27 (0714:04) in an effort to cool the Reactor Coolant System down and reduce Reactor Coolant System pressure.	PZR Spray: ST at PL4 RC P: MR and SC at PL4	1,8g,8d,8r
02:55:13 (0655:50)	The operator cleared the Safety Injection portion of Engineered Safety Features trains A and B.	AN at PL13, ST at PL3 and PL 13 AP norm/Bypassed (Delay ≃ 10 minutes)	2a
02:55:26 (0656:03)	Condenser hot well low level alarm was received. The level was 21.82 inches.	MR at PL5 AP Low (22.5 inches)/norm/high (36 inches) (Delay ≃ 11 minutes)	2a
02:55:48 (0656:15)	The operator started Circulating Water Pump 1B (CW-P-1B).	MR (A and P) and ST on PL17 AP on/off (Delay ≃ 11 minutes)	2a,8m,9J
02:56:07 (0656:44)	The operator opened Main Steam Isolation Valves (MS-V-4B and 7B) and steamed to the steam chest and main steam lines for approximately 12 seconds to stop the rapid increase in Steam Generator B pressure.	MS-V4B/7B: ST at PL15 SG P: MR at PL4 and SC at PL17 AP open/closed (Delay ≃ 11 minutes)	2a,9J
02:57:23 (0658:00)	The Emergency Control Station was established in Unit 1 Health Physic Laboratory. Onsite and offsite radiation monitoring teams are formed in preparation to measure radiation levels.	Unit 2 Control Room notified of the Elementary Control Station status.	9g,9F

Time	Event	Information Available to the Operator	Reference
02:59:23 (0700:00) Approximate	Reactor Building Purge Unit Area Monitor (HP-R-3236) and the Fuel Handling Building Area Monitor (HP-R-3240) began to increase.	MR and MP at PL12	3b,5b
03:00:56 (0701:33)	Condenser hotwell water level indication went off scale low. The Auxiliary Operator throttled the Reject Isolation Valve (CO-V68).	MR at PL3 AP bad (out of range) (Delay ≃ 11 minutes)	2a,3r
03:01:11 (0701:48)	Condensate Storage Tank 1B norm level alarm was received. A level of 20.31 feet was recorded.	AP low (20 ft)/norm/high (29 ft) (Delay ≃ 11 minutes)	2a
03:02:58 (0703:35)	Condenser hotwell low level alarm was received again. The level was 9.68 inches.	MR at PL3 AP low (22.5 inches)/norm/high (36 inches) (Delay ≃ 11 minutes)	2a
03:03:39 (0704:16) Approximate	The operator isolated Steam Generator B for the second time. Turbine Bypass Isolation Valve (MS-V15B) was shut. Emergency Feedwater Valves (EF-V5B, EF-V11B, and EF-V12B) were shut. The operator suspected a Reactor Coolant side to Feedwater side leak. The Condenser Vacuum Pump Exhaust Radiation Monitor (VA-R-748) indication had increased from 1×10^2 to 5×10^5 counts per minute.	VA-R-748: AR, MR and MP at PL12 MS-V15B: ST at PL3 EF-V5B: ST at PLA EF-V11B: MR at PLA EF-V12B: ST at PLA VA-R-748: AR, MR and MP at PL12	2a,8r,9E,0J
03:04:05 (0704:42)	The reactor out-of-core Intermediate Range Channels (NI-3 and NI-4) indications increased from less than 1×10^{-11} amperes to approximately 1.2×10^{-11} and 2.5×10^{-11} amperes respectively (Figure 56). A corresponding increase was recorded on the reactor out-of-core Source Range Channel (NI-1) indication (Figure 56).	NI-1: MR and SC at PLA NI-3: MR and SC at PLA NI-4: MR and SC at PLA	2a,3k,5a
03:04:23 (0705:00) Approximate	The Manager-Generating Station Nuclear arrives in the Unit 2 control room and assumes the responsibility of Emergency Director.	Announcement made on Plant Page System	9u,10

Time	Event	Information Available to the Operator	Reference
03:05:44 (0702:49)	Steam Generator B level indication reached 60% on the operating range (Figure 45). This level was maintained during the next 8.5 hours.	SC (operate range) at PL4 and PL5	1
03:06:40 (0707:17)	Condensate Storage Tank 1B low level alarm was received. A level of 19.96 feet was recorded.	AP low (20 ft)/norm/high (29 ft) (Delay ≈ 13 minutes)	2a
03:10:27 (0711:04)	The operator stopped Emergency Feedwater Pump 2A (EF-P-2A). Both Steam Generators had levels of about 60% on the Operating Range (Figure 45).	EF-P-2A: ST, MR(A) and MR(P_{DISCH}) at PL4 SG L: SC (Operate Range) at PL4 and PL5	1,2a
03:11:10 (0711:47)	Condenser hotwell level returned to normal. The level was 23.07 inches.	MR at PL5 AP low (22.5 inches)/norm/high (36 inches) (Delay ≈ 15 minutes)	2a
03:12:28 (0713:05) Approximate	The operator opened the Electromatic Relief Block Valve (RC-V2) to reduce reactor coolant system pressure and pressurizer level after attempts to reduce the pressure by using the pressurizer spray flow were unsuccessful.	RC-V2: ST at PL4	1,3j,3m,8q,8r
03:12:28 (0713:05)	The Electromatic Relief Valve (RC-R2) discharge line high temperature alarm was received. A temperature of 247.7F was recorded.	SP at PL10 AP high (200F)/norm (Delay ≈ 15 minutes)	2a
03:12:35 (0713:12) Approximate	The reactor out-of-core Intermediate Range Channels (NI-3 and NI-4) indications decreased to less than 1.0×10^{-11} amperes (Figure 56).	NI-3: MR and SC at PL4 NI-4: MR and SC at PL4	3k
03:12:53 (0713:30)	The operator stopped Reactor Coolant Pump 2B (RC-P-2B) based on indicated zero flow and motor running current of less than 100 amperes. Normal Reactor Coolant Pump operating current is approximately 600 amperes. Later a close examination of the flow recorder trace indicated a small amount of reactor coolant flow had existed.	ST, MR(A), MR(F) and SC(F) at PL4, AN (trip) at PL8 AP norm/trip (Delay ≈ 16 minutes)	2b

Time	Event	Information Available to the Operator	Reference
03:13:36 (0714:11)	The following Incore Thermocouple Temperatures increased to greater than 700F during the next seven minutes (14D, 12O, 14M, 13C, 7B, 2G, 10C, 13C and 7B). The increased temperatures were in response to stopping Reactor Coolant Pump 2B (RC-P-2B).	AP norm/bad (out of range 0F-700F) (Delay ≈ 17 minutes)	2a
03:13:58 (0714:35)	The pressurizer safety valves (RC-R1A and RC-R1B) discharge line high temperature alarms were received. Respective temperatures of 202.6F and 205.8F were recorded.	MP at PL10 AP high (200F)/norm (Delay ≈ 17 minutes)	2a
03:14:23 (0715:00) Approximate	Intermediate Cooling Pump Area Monitor (HP-R-207) indicated an increased radiation level and at 03:20:23 (0721:00) the level stabilized at a value of 100 mR/hr.	AR, MR and MP at PL12	3a
03:18:05 (0718:42)	The reactor out-of-core Intermediate Range Channel (NI-4) indication increased from less than 1.0×10^{-11} amperes to about 3.2×10^{-10} amperes (Figure 56). However, this increase was not observed on either the Source Range Channel (NI-1) or the other Intermediate Range Channel (NI-3).	MR and SC at PLA AP norm/bad (out of range 10^{11} to 10^{-3} amps) (Delay ≈ 19 minutes)	2e,3k
03:19:23 (0720:00) Approximate	The first offsite dose calculations were completed by Plant Engineers. The initial calculations indicated a total body exposure rate of 40 R/hr in Goldsboro (1.3 miles west of the plant). Based on the Reactor Building pressure being much lower than that assumed in the source term calculation (56.2 psig) it was felt that the above value was overly conservative since it incorporated a higher leak rate. Following these calculations, onsite and offsite radiation teams were dispatched in the downwind direction (west side of the Three Mile Island and Goldsboro, Pennsylvania).	Calculation performed in Unit 2 Control Room by Plant Engineers and discussed with operational personnel.	6a,9h,9q

Time	Event	Information Available to the Operator	Reference
03:19:45 (0720:22)	The operator manually initiated the Safety Injection portions of Engineered Safety Feature trains A and B as a result of low Reactor Coolant System pressure (Figure 4). The Safety Injection automatic actuation setpoint is 1640 psig.	AM at PL13, ST at PL3 and 13 AP bypass/test/trip (Delay ≈ 19 minutes)	2a,7,8q,9e, 9A
03:20:13 (0720:50)	Reactor Coolant Makeup Pump C (MU-P-1C) started automatically on the Engineered Safety Feature train A actuation. Reactor Coolant Makeup Pumps A and C (MU-P-1A and MU-P-1C) were operating. NOTE: During an Engineered Safety Features actuation Safety Injection utilizes Reactor Coolant Makeup Pumps 1A and 1C (MU-P-1A and MU-P-1C).	ST and MR(A) at PL3, AM at PL8 AP norm/trip (Delay ≈ 18 minutes)	2a,5c,6c
03:20:23 (0721:00) Approximate	The following radiation monitors registered increased radiation levels: (a) Primary Coolant Letdown Hi (MU-R-720 HI) (b) Primary Coolant Letdown Lo (MU-R-720 LO) (c) Intermediate Cooling Letdown Cooler B (IC-R-1091) (d) Intermediate Cooling Letdown Cooler A (IC-R-1092) (e) Intermediate Cooling Letdown Cooler Outlet (IC-R-1093) (f) Plant Effluent Unit II (WDL-R-1311) (g) Decay Heat Closed A Loop (DC-R-3399) (h) Decay Heat Closed B Loop (DC-R-3400) (i) Nuclear Service Closed Cooling (NS-R-3401) (j) Spent Fuel Cooling (SF-R-3402)	GR, MR and SP on PL12	3d,3b

Time	Event	Information Available to the Operator	Reference
03:20:33 (0721:10)	The following Incore Thermocouple Temperatures decreased to less than 700F over next fourteen minutes. The increased reactor core cooling was a result of the manual high pressure injection actuation. 10C = 637.9F 4E = 698.4F* 7R = 630.5F 6F = 678.2F 13G = 682.1F 2L = 668.7F 7B = 627.7F 8B = 617.2F 3G = 630.3F 10M = 689.0F 13D = 623.6F 6C = 631.9F 14M = 624.2F 7E = 692.7F 14D = 477.9F* 8F = 685.8F 13C = 685.7F 5O = 693.5F 3D = 645.0F Note: (*) This Incore Thermocouple Temperature cycled near 700F until Reactor Coolant Makeup Pump C (MU-P-1C) was stopped.	AP zero/end (out of range 00-700F)(Delay ≃ 20 minutes)	2a
03:21:00 (0721:37) Approximate	The reactor out-of-core Intermediate Range Channel (NI-4) indication decreased to less than 1×10^{-11} amperes. The Reactor out-of-core Source Range Channel (NI-1) indication showed a corresponding marked decrease which indicated that the steam in the reactor vessel was displaced by liquid. Both Intermediate Range Channels (NI-3 and NI-4) remained offscale (less than 1×10^{-11} amperes) for the remainder of March 28, 1979.	NI-1: NR and SC at PLA NI-4: NR and SC at PLA	3b

Time	Event	Information Available to the Operator	Reference
03:21:23 (0722:08) Approximate	The following radiation monitors indicated radiation levels exceeding the monitor range, and remained above range until the stripchart ended at 1100:00, 4/2/79. (a) Reactor Building Purge Air Exhaust Duct A (HP-R-225) - Particulate Monitor (b) Reactor Building Purge Air Exhaust Duct B (HP-R-226) - Particulate Monitor (c) Auxiliary Building Purge Air Exhaust (HP-R-222) (d) Auxiliary Building Heating & Ventilation Radiation Monitor gas channel The Reactor Building Purge Air Exhaust Duct Iodine Monitors indicated 1 x 10^5 counts per minute. The gas channel of the Reactor Building Duct A Radiation Monitor indicated 5 x 10^5 counts per minute.	AB, CB and HP at PL12	3c,5b
03:21:23 (0722:08) Approximate	Fuel Handling Building Exhaust Filter Outlet Radiation Monitor (HP-R-221B) and Unit Vent Stack Radiation Monitor (HP-R-219) alarmed high. As a result the fuel handling building supply fans (AH-E-9A and AH-E-9B) stopped automatically.	RC and UT at PL 25	3f,3h,6c

PLANT STATUS

The Reactor Coolant System was at minimum forced Reactor coolant flow with all Reactor Coolant Pumps (RC-P-1A, RC-P-2A, RC-P-1B, and RC-P-2B) stopped. After attempts to establish natural circulation failed, the operator started Reactor Coolant Pump 2B (RC-P-2B). However, based on a no flow indication (Figure 18) and a pump running current of less than 100 amps, Reactor Coolant Pump 2B was stopped after 19 minutes. Steam was present

Time	Event	Information Available to the Operator	Reference
	in the Reactor Vessel head and Reactor Coolant System hot legs. Both Reactor Coolant System hot leg temperature were off scale high (i.e. greater than 620F). The Reactor Coolant System cold leg temperatures were 455F for Loop A and 390F for Loop B. Steam Generator B was isolated due to a suspected Reactor Coolant side to Feedwater side leak and Steam Generator A was steaming to the condenser. An attempt was in progress to establish Reactor Coolant System Pressurizer level (Figure 33) by opening and closing the Electromatic Relief Block Valve (RC-V2). This resulted in a lower Reactor Coolant System pressure which prompted the operator to manually initiate Safety Injection. Consequently, both Reactor Coolant Makeup Pumps 1A and 1C (MU-P-1A and MU-P-1C) were operating.		
03:23:16 (0723:33)	A Safety Injection actuation due to low Reactor Coolant System pressure was received. Safety Injection had previously been placed in service by the operator.	AR at PL13, ST at PL3 and PL13 AP core/actuation (Delay ≈ 36 minutes)	2a
03:23:23 (0724:00) Approximate	The Manager-Generation Station Nuclear declared a General Emergency as a result of an indicated radiation level of 8 R/hr on the Reactor Building Dome Radiation Monitor (HP-R-212). Notification of offsite authorities and organizations was initiated.	AR, MR and MP at PL12	10
03:23:23 (0724:00) Approximate	A Radiation/Chemistry Technician attempted to take a feedwater sample on Steam Generators A and B from the Plant primary chemistry laboratory. The normal sampling location is the Unit 2 secondary chemistry laboratory, but because of the suspected Reactor coolant side to feedwater side leak to Steam Generator B the valve alignment was altered so as not to contaminate this area. The chemist was able to obtain a sample from Steam Generator B. He was unable to establish flow through the Steam Generator A sample line.	Unit 2 Control Room notified of steam generators A and B sample results	4e,9h,9r

Time	Event	Information Available to the Operator	Reference
	A Germanium Lithium analysis of the Steam Generator B sample was performed at 03:25:23 (0726:00) and indicated radioactivity levels were note above background. The results of the analysis is listed below. Potassium 40 — 4.840 E-06 uCi/ml Cobalt 58 — 1.555 E-06 uCi/ml Xenon 133 — 1.836 E-07 uCi/ml Total: — 6.585 E-06 uCi/ml The Control Room personnel questioned these results since they felt that Steam Generator B was contaminated. It was later determined that the sample lines leading over to the Plant primary chemistry laboratory were reversed. The determination of the level of radioactive materials in both steam generators was made by re-aligning the sample line discharge back to the Unit 2 secondary chemistry laboratory and monitoring the sample from each Steam Generator with a frisker (RM-14). The results indicated that only Steam Generator B was contaminated. A Germanium Lithium analysis of these samples were not performed.		
03:25:56 (0726:33)	The Reactor Coolant System high pressurizer level alarm c eared. A level of 330 inches was recorded.	AN at PL8 AP low (200 inches)/norm/high (260 inches) (Delay ≃ 37 minutes)	2a
03:26:33 (0727:10)	The operator bypassed the Safety Injection portion of Engineered Safety Features trains A and B.	AN at PL13, S. at PL3 and PL13 AP norm/bypassed (Delay ≃ 37 minutes)	2a
03:27:23 (0728:00) Approximate	The Radiation Level indicated on the Auxiliary Building Access Control Corridor Radiation Monitor (HP-R-232) increased.	AN, MR and MP at PL 12	3b
03:27:23 (0728:00)	The first onsite radiation monitor team was dispatched to measure the radiation level at detectors GE8 and GE9. AT 03:45:23 (0746:00)	Unit 2 Control Room notified of radiation levels at detectors GE8 and GE9.	4c

Time	Event	Information Available to the Operator	Reference
	a measure value of less than 1 mR/hr is reported. Detectors G18 and G19 are located on the east side of Three Mile Island between Unit 2 and Goldsboro, Pennsylvania. Onsite radiation monitoring continued during the remainder of March 28, 1979.		
03:30:00 (0730:37) Approximate	The operator shut the Electromatic Relief Block valve (RC-V2) to stop the rapid fall in reactor coolant system pressure. The pressurizer level was 226 inches and the Reactor Coolant System pressure was 1480 psig (Figures 4 and 14).	RC-V2: ST at PLA PZR L: SC at PLA, MR (uncompensated) at PL3 RC P: MR and SC at PLA	1,33
03:30:13 (0730:50)	The Borated Water Storage Tank low level alarms were received at 53.03 feet and 53.06 feet. The Borated Water Storage Tank level was reduced by the operation of Reactor Coolant Makeup Pumps during normal and Safety Injection conditions.	AN and MR at PL8 AP low (53.0 feet)/norm (Delay $\simeq$ 38 minutes)	2a
03:30:58 (0731:35)	Pressurizer Safety Valve (RC-R1A) discharge line high temperature alarm reset. A temperature of 192.4F was recorded.	MP at PL18 AP high (200F)/norm (Delay $\simeq$ 38 minutes)	2a
03:32:26 (0733:13)	The Reactor Coolant System high pressurizer level alarm was received. A level of 271 inches was recorded.	AN at PL8 AP low (200 inches)/norm/high (260 inches) (Delay $\simeq$ 39 minutes)	2a
03:34:28 (0735:05)	Pressurizer Safety Valve (RC-R1B) high discharge line temperature reset. A temperature of 192.6F was recorded.	MP at PL 18 AP high (200F)/norm (Delay $\simeq$ 40 minutes)	2a
03:35:06 (0735:43)	The operator started Emergency Feedwater Pump 2A (EF-P-2A). Steam Generator A level had decreased from 68% to 64% of Operating Range during the previous 45 minutes (Figure 45).	EF-P-2A: ST, MR(P_{DISCH}) and MR(A) at PLA, AP on/off and P_{DISCH} low/normal (Delay $\simeq$ 40 minutes) SG L: SC (operating range) at PLA and PL3	2a

Time	Event	Information Available to the Operator	Reference
03:37:00 (0737:37)	The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C) because pressurizer level was rapidly increasing. Indicated Pressurizer level was 373 inches (Figure 4).	MU-P-1C: ST and MR(A) at PL3, AB at PL8 AP norm/trip (Delay ≅ 43 minutes)	1,2a,5c
03:39:23 (0740:00)	The operator isolated the Reactor Building air Sample Line after it was reported as blowing air into the Auxiliary Building.	ST at PL25	4c
03:40:00 (0740:37)	The operator opened the Electromatic Relief Block Valve (RC-V2) in an attempt to decrease Pressurizer level, which had increased offscale (greater than 400 inches).	RC-V2: ST at PL4	1,3j,3m
03:40:28 (0741:05)	The Pressurizer Safety Valves (RC-R1A and RC-R1B) discharge line high temperature alarms were received. Respective temperatures of 201.6F and 205.3F were recorded.	MP at PL10 AP high (200F)/norm (Delay ≅ 43 minutes)	2a
03:44:03 (0744:40)	The following Incore Thermocouple Temperatures increased to greater than 700F during the next three minutes (5D, 4E, 6P*, 5O, 9C, 2L, 7B, 14M, 9G, 7E, 10M* and 8F). The increase temperatures were in response to stopping Reactor Coolant Makeup Pump C (MU-P-1C). Note: (*) This Incore Thermocouples Temperature cycled near 700F until Safety Injection actuation on high Reactor Building pressure.	AP norm/bad (out of range OF-700F) (Delay ≅ 43 minutes)	2a
03:44:23 (0745:00) Approximate	The Fuel Handling Building Exhaust Fans (AH-E-10A and AH-E-10B or AH-E-10C and AH-E-10D) stopped. Airborne radioactive contamination levels in Unit 1 Fuel Handling Building and Auxiliary Building started to increase. The reason for these fans stopping is unknown.	Exhaust Fans: SC and ST at PL15 Radiation Levels: MR and MF (Unit 1 Control Room)	3h,3u
03:44:23 (0745:00) Approximate	Using a resistance bridge and conversion tables, Plant Staff determined the Reactor Coolant System Loop A hot leg temperature	Measurement performed in Unit 2 Control Room by Plant Engineer and discusses with Operational	3e,9W

Time	Event	Information Available to the Operator	Reference
	to be 718F. Later on March 28, 1979 an additional resistance bridge was connected to Loop B hot leg sensor and gave results similar to Loop A.		
03:45:00 (0745:37)	The following self-powered neutron detectors were going offscale between Levels 1 through 7 which corresponds to the total length of the core (6P, 7R, 14M, 7E, 5G, 9H, 10D, 5O, 8D, 5K, 10M, 11K, 9E and 9M). This has been attributed to the high temperatures expe ienced in the core.	AP bad/norm (0 to 2000 mA) (Delay $\simeq$ 44 minutes)	2a
03:45:00 (0745:37) Approximate	The reactor out-of-core source range channel (NI-1) indication increased sharply from about 2.2×10^3 counts per second to 4.9×10^3 counts per second. The reactor out-of-core intermediate range channel (NI-3 and NI-4) indications were off scale low and did not respond.	NI-1: MR and SC at PLA NI-3: MR and SC at PLA NI-4: MR and SC at PLA	3a
03:45:27 (0746:04)	The operator initiated pressurizer spray flow to assist in lower Reactor Coolant System pressure which had rapidly increased from approximately 1400 psig to 1700 psig (Figure 4). Pressurizer spray flow was maintained until 04:21:48 (0822:25).	PZR SPRAY: ST at PLA RC P: MR and SC at PLA	1,8q,8r
03:48:23 (0749:00) Approximate	Makeup Tank Area Monitor (HP-R-206) advanced off scale. Levels in Fuel Handling Bridge S. (HP-R-210) and Reactor Dome (HP-R-214) stabilized at 1.5×10^2 R/hr.	AN, MR and MP at PL12	3e
03:55:39 (0756:16)	The Reactor Building Isolation and Cooling portion of Engineered Safety Feature train B actuated on Reactor Building high pressure. The setpoint is 3.98 psig (Figure 51).	AD at PL13, ST at PL3 and PL15 AP act/trip (Delay $\simeq$ 46 minutes)	2a,6a
03:55:39 (0756:16)	Intermediate Cooling Pump 1B (IC-P-1B) tripped automatically by the Engineered Safety Features train B actuation.	AN, ST, MR (P_{DISCH}) and MR(F) at PL8 AP on/off (Delay $\simeq$ 46 minutes)	2a,6a

Time	Event	Information Available to the Operator	Reference
03:33:39 (0756:16)	The Reactor Building isolated automatically as part of the Engineered Safety Features train B Actuation.	AM at PL13, ST at PL3 and PL13, AP isolation/norm (Delay ≃ 46 minutes)	2a
03:33:39 (0756:16)	The Control Room ventilation system should have aligned to internal recirculation mode upon actuation of Engineered Safety Features train B. In the Recirculation mode the Control Room air exhaust flow is diverted to the supply flow and the supply flow is reduced to maintain a positive room pressure. The exhaust flow recorder was out of service on March 28, 1979 and this flow diversion can not be verified. However a reduction in the control room supply flow was experienced (see reference 3V).	SC and ST at PL25	3r, 6c
03:33:46 (0756:23)	The Reactor Building Isolation and Cooling portion of Engineered Safety Features train A actuated on Reactor Building high pressure. The setpoint is 3.58 psig (Figure 31).	AM at PL13, ST at PL3 and PL15 AP act/trip (Delay ≃ 46 minutes)	2a,6a
03:33:46 (0756:23)	Intermediate Cooling Pump 1A (IC-P-1A) tripped automatically by the Engineering Safety Features train A actuation.	AM, ST, MR (P_{DISCH}) and MR(F) at PL8 AP on/off (Delay ≃ 46 minutes)	2a
03:34:04 (0756:41)	Reactor Coolant Makeup Pump C (MU-P-1C) was started automatically by the Engineered Safety Features train B Actuation.	AM at PL8, ST and MR(A) at PL3 AP norm/trip (Delay ≃ 46 minutes)	2a
03:39:23 (0802:00) Approximate	Waste Gas Tank Discharge A (WDG-R-1485) monitor began to increase from 3 x 10^2 counts per minute until it reached 3 x 10^3 counts per minute at approximately 05:00:00 (0922:37). Note: This monitor is located in the Auxiliary Building at a elevation of 305 feet.	MR and MP at PL12	3e,5b
04:00:13 (0822:50)	The operator defeated the Reactor Building Isolation and Cooling portions of Engineered Safety Features trains A and B.	AM at PL13, ST at PL3 and PL13 AP norm/defeated (Delay ≃ 47 minutes)	2a

Time	Event	Information Available to the Operator	Reference
04:00:13 (0000:50)	The operator started Intermediate Cooling Pump 1B (IC-P-1B).	AR, ST, MR(P_{DISCH}) and MR(F) at PLB AP on/off (Delay ≃ 47 minutes)	2a
04:00:19 (0000:56)	The operator started Intermediate Cooling Pump 1A (IC-P-1A).	AR, ST, MR(P_{DISCH}) and MR(F) at PLB AP on/off (Delay ≃ 47 minutes)	2a
04:01:23 (0002:00)	The first offsite radiation monitor team was dispatched to measure the radiation level at Detector W-11. At 04:31:23 (05:32:00) a measured value of less than 1 mR/hr is reported. Detector W-11 is located in Goldsboro, Pennsylvania. Offsite radiation monitoring continued during the remainder of March 28, 1979.	Unit 2 Control Room notified of Radiation Levels at Detection W-11.	4c
04:02:03 (0002:40)	The trend in Incore Thermocouple Temperatures increasing above 700F stopped. The increased reactor core cooling was a result of the manual High Pressure Injection actuation. The following Incore Thermocouple Temperatures decreased below 700F during the next twelve minutes. 8M = 680.5F 6R = 23.7F 13G = 694.0F 5M = 694.7F 4E = 619.3F 10H = 693.2F 12P = 694.9F 12F = 588.5F	AP norm/bad (out of range 0F-700F)(Delay ≃ 49 minutes)	2a
04:05:23 (0006:00) Approximate	The Auxiliary Building Exhaust Fans (AH-E-8C and AH-E-8D) tripped upon receiving a fire alarm in the Auxiliary Building. An Instrument Foreman was send to the Auxiliary Building to defeat the alarm.	SC and ST at PL25	3g,9S
04:06:23 (0007:00)	The Model Room door which connects the Unit 1 and Unit 2 Auxiliary Building was shut locally to minimize the spread of airborne radioactive materials to Unit 1.	Unit 2 Control Room Notified of Model Room door status	4c,9h,9W

Time	Event	Information Available to the Operator	Reference
04:07:01 (0807:38)	The operator cleared the Reactor Building isolation and cooling portion of Engineered Safety Features train A. Train B remained defeated.	AB at PL13, ST at PL3 and PL13 AP norm/defeated (Delay $\simeq$ 32 minutes)	2a
04:08:37 (0809:14)	The operator started Reactor Coolant Pump 1A (RC-P-1A) to re-establish Reactor Coolant System flow. NOTE: During the previous run of Reactor Coolant Pump 2B (RC-P-2B), due to the flow and current indication observed, it was thought that the pump might not have started. For this reason it was decided to observe the starting current during a Reactor Coolant Pump start. Reactor Coolant Pump 1A (RC-P-1A) was started and a correct starting current was observed by the operator. As before, the indicated pump current rapidly decreased to less than 100 amperes.	ST, MR(A), MR(F) and SC(F) at PL4, AB at PL8 AP norm/trip (Delay $\simeq$ 51 minutes)	2a,9a
04:09:14 (0809:51)	The operator stopped Reactor Coolant Pump 1A (RC-P-1A) after observing a zero flow indication and a running current less of than 100 amperes.	ST, MR(A), MR(F) and SC(F) at PL4, AB at PL8 AP norm/trip (Delay $\simeq$ 51 minutes)	2a,9a
04:10:10 (0810:47)	The operator stopped Intermediate Cooling Pump 1B (IC-P-1B).	AB, ST, MR(P_{DISCH}) and MR(F) at PL8 AP on/off (Delay $\simeq$ 33 minutes)	2a
04:16:23 (0817:00) Approximate	The Auxiliary Operator started the Fuel Handling Building Exhaust Fans (AH-E-10A and AH-E-10B or AH-E-10C and AH-E-10D) and the Auxiliary Building Exhaust fans (AH-E-8C and AH-E-8D). He also placed Control Room Bypass Filter Fan 4B (AH-E-4B)in service. NOTE: When the Control Room Bypass Filter Fan is running the Control Room atmosphere is continuously being filtered.	SC and ST at PL25	3g,3h,8m,9J

Time	Event	Information Available to the Operator	Reference
04:17:17 (0817:54)	The operator stopped Reactor Coolant Makeup Pump 1A (MU-P-1A).	ST and MR(A) at PL3, AM at PL8, AP norm/trip (Delay ≃ 55 minutes)	2a,3c,9T
04:17:22 (0817:59)	The operator stopped Reactor Coolant Makeup Pump 1C (MU-P-1C). No Reactor Coolant Makeup pumps were operating.	ST and MR(A) at PL3, AM at PL8, AP norm/trip (Delay ≃ 55 minutes)	2a,3c,9a,9T
04:18:17 (0818:54)	Following an unsuccessful start attempt, the operator placed Makeup Pump 1A (MU-P-1A) control switch in the Pull-to-Lock position to prohibit further use.	Control switch handle position	2a,2c,3c,8c, 8r,9a
04:19:02 (0819:39)	The operator started Intermediate Cooling Pump 1B (IC-P-1B).	AM, ST, MR(P_{DISCH}) and MR(F) at PL8 AP on/off (Delay ≃ 55 minutes)	2a
04:19:05 (0819:42)	The Reactor Building Isolation and Cooling portion of Engineered Safety Features train A actuated on Reactor Building High Pressure. The setpoint is 3.58 psig. (Figure 51).	AM at PL3, ST at PL3 and PL13 AP act/trip (Delay ≃ 55 minutes)	2a,6a
04:19:06 (0819:43)	Decay Heat Removal Pump 1A (DH-P-1A) started automatically by the Engineered Safety Features train A actuation	AM and MR (P_{DISCH}) at PL8, ST at PL3 and PL13 MR(A) at PL3 AP norm/low and on/off (Delay ≃ 55 minutes)	2a
04:19:06 (0819:43)	Intermediate Cooling Pump 1A (IC-P-1A) tripped automatically by the Engineered Safety Features train A actuation.	AM, ST, MR (P_{DISCH}) and MR(F) at PL8 AP on/off (Delay ≃ 55 minutes)	2a
04:19:24 (0820:01)	The operator defeated the Reactor Building Isolation and Cooling portion of Engineered Safety Features train A.	AM at PL13, ST at PL3 and PL13 AP norm/defeated (Delay 55 minutes)	2a
04:19:29 (0820:06)	The operator started Intermediate Cooling Pump 1A (IC-P-1A).	AM, ST, MR (P_{DISCH}) and MR(F) at PL8 AP on/off (Delay ≃ 55 minutes)	2a
04:21:53 (0822:30) Approximate	The operator started Reactor Coolant Makeup Pump 1B (MU-P-1B). This Reactor Coolant Makeup Pump ran for the remainder of March 28, 1979.	ST and MR(A) at PL3, AM at PL8, AP norm/trip (Delay ≃ 55 minutes)	2a,3c,9a,9T

Time	Event	Information Available to the Operator	Reference
04:23:54 (0824:31)	The operator energized Pressurizer Heater Groups 1 through 5 by placing heater control into automatic. Eleven pressurizer heater groups were available at this time.	AM at PL8, ST at PL4, AP on/off (Delay ≃ 58 minutes)	2a
04:26:09 (0826:46)	The Reactor Coolant Makeup Tank level increased offscale high (greater than 100 inches) and remained offscale for 42 seconds (Figure 54).	SC at PL3	1
04:27:02 (0827:39) Approximate	The operator started Reactor Coolant Makeup Pump 1C (MU-P-1C). after an unsuccessful attempt to start the pump at 04:26:59 (0827:36).	ST and MR(A) at PL3, AM at PL8, AP norm/trip (Delay ≃ 58 minutes)	2a,5c,9a
04:30:30 (0831:07)	Pressurizer Heater Group 10 tripped due to a ground fault and remained de-energized for the remainder of March 28, 1979. Ten pressurizer heater groups were available at this time.	AM at PL8, ST at PL4, AP norm/trip (Delay ≃ 58 minutes)	2a
04:30:45 (0831:22)	The operator stopped Condenser Vacuum Pumps 1A and 1C (VA-P-1A and VA-P-1C) and broke Main Condenser vacuum after experiencing difficulty with the operation of the Auxiliary Boiler.	Pumps: ST at PL17, AP on/off (Delay ≃ 60 minutes) Vacuum: AM and SC at PL17	2a,8a,9z, 10
04:30:45 (0831:22) Approximate	The operator opened Power Operated Emergency Main Steam Dump Valve A (MS-V3A) to induce natural circulation in Steam Generator A. Steam Generator B was still isolated.	MR (Valve demand setpoint) at PL3	8f,8a,9z,10
04:33:30 (0834:07)	The plant staff requested the computer to print the following Incore Thermocouple Outlet Temperatures. The following values were recorded.	UP (Delay ≃ 0 minutes) AP norm/bad (out of range 0F to 700F)(Delay ≃ 60 minutes)	2c,9s,9u, 9v,9W

11L = 480.9F	8C = ***.*F
11K = ***.*F	8B = 511.7F
12K = ***.*F	7B = 436.7F
13B = ***.*F	6C = 641.9F
13C = 533.0F	5D = ***.*F

Time	Event	Information Available to the Operator	Reference
	12F = 437.1F 4E = ***.*F 11E = ***.*F 3F = 309.7F 10C = 403.1F 2G = 518.7F Note: ***.*) indicates that the signal was outside of the computer range (Range = 0F to 700F).		
04:42:14 (0842:51)	The operator stopped Emergency Feedwater Pump 2A (EF-P-2A).	ST, MR(A) and MR (P_{DISCH}) at PL4 AP on/off and low (875 psig)/norm (Delay ≃ 61 minutes)	2a
04:44:23 (0845:00)	Intermediate Cooling Letdown Cooler A Radiation Monitor (IC-R-1092) indication decreased and went off scale. It was assumed the monitor failed.	MR and SC at PL12	3d,5b
04:46:07 (0846:44)	The plant staff requested the computer to print the following Incore Thermocouple Outlet Temperatures. The following values were recorded. 13F = ***.*F 9C = ***.*F 12F = 393.1F 8B = 418.6F 11G = ***.*F 7B = 377.6F 11E = ***.*F 6C = 561.4F 10D = 582.7F 5D = ***.*F 10C = 398.9F Note: (***.*) indicates that the signal was outside of the computer range (Range = 0F to 700F).	UP (Delay ≃ 0 minutes) AP norm/bad (out of range 0F to 700F)(Delay ≃ 62 minutes)	2c,9a,9u,9v, 9B
04:46:21 (0846:58)	Pressurizer Heater Groups 4 and 5 tripped due to ground faults and remained de-energized for the remainder of March 28, 1979. Eight pressurizer heater groups were available at this time.	AM at PL8, ST at PL4 AP norm/trip (Delay ≃ 63 minutes)	2a

Time	Event	Information Available to the Operator	Reference
04:49:23 (0850:00) Approximate	Condenser Vacuum Pump Exhaust radiation monitor (VA-R-748) decreased to 1 x 10^4 counts per minute.	MR and SC at PL12	3a,5b
04:59:23 (0900:00) Approximate	Incore Thermocouple readings obtained by the Plant Staff using a resistance bridge and a conversion table placed fuel assembly exit temperatures in the range of 217F to 2580F (Figure 30).	Unit 2 Control Room notified of measured fuel assembly exit temperatures	4j,9o,9u,9R, 9S,9U,9V
04:59:23 (0900:00)	A Radiation/Chemistry Technician drew a reactor coolant system sample. A radiation level of 200 R/hr (beta-gamma) was measured at six inches. A boron analysis was performed immediate and yielded a value of approximately 240 parts per million boron. A Germanium Lithium analysis was performed later at 07:38:23 (1139:00) and indicated a total beta-gamma activity of 1354 mCi/ml. No further samples were taken for the remainder of March 28, 1979. The results of the analysis is listed below. Iodine 131 8.149 E+01 mCi/ml Iodine 133 1.438 E+02 mCi/ml Xenon 133 4.493 E+02 mCi/ml Xenon 135 2.464 E+02 mCi/ml Rubidium 88 6.037 E+02 mCi/ml Krypton 85 2.884 E+01 mCi/ml Total: 1.354 E+03 mCi/ml	Unit 2 Control Room notified of boron and gross beta-gamma analysis results	4a,4g,6b,9v
04:59:23 (0900:00)	Intermediate Cooling Pump Area monitor (HP-R-207) and the Reactor Building Emergency Cooling Booster Pump Area monitor (HP-R-204) indications began to increase.	MR and SC at P12	3a,5b
05:11:23 (0912:00)	The Emergency Control Station was moved from Unit 1 Health Physics Laboratory to Unit 2 Control Room after experiencing increased levels in airborne radioactive materials.	Unit 2 Control Room notify of relocation of the Emergency Control Station	4c,4g,8a,9q

Time	Event	Information Available to the Operator	Reference
05:17:38 (0918:05)	The alarm printer was returned to service and the alarm function was transferred from the utility printer to the alarm printer.	AP (Delay ≅ 0 minutes)	2a
05:18:00 (0918:37) Approximate	The operator closed the Electromatic Relief Block Valve (RC-V2) in an attempt to compress the reactor coolant and condense the steam in the Reactor Coolant System.	ST at PLA	3j,3m,10
05:[illegible]:47 (0919:[illegible])	The operator stopped Decay Heat Removal Pump 1A (DH-P-1A). This pump did not start automatically on the next Engineered Safety Features train A Actuation, therefore the pump control switch must have been placed in pull to lock position at this time.	AM and MR (P_{DISCH}) at PL8, ST at PL3 and PL13 MR(A) at PL3 AP norm/low and on/off (Delay ≅ 86 minutes)	2a

PLANT STATUS

All Reactor Coolant Pumps (RC-P-1A, RC-P-2A, RC-P-1B and RC-P-2B) were stopped. Superheated steam and gas were present in the upper Reactor Vessel [illegible] Reactor Coolant System hot leg regions. Attempts to re-establish reactor coolant flow using Reactor Coolant Pump 1A (RC-P-1A) had not been successful. The Reactor Coolant System hot leg temperature continued to indicate off-scale (i.e. greater than 620F). The Reactor Coolant system cold leg temperatures were 180 F for Loop A and 225F for Loop B, and both were decreasing (Figures 21 and 26). Steam Generator A level was at 48% of the operating range (Figure 39). Steam Generator B was isolated, with a level at 60% on the operating range (Figure 39). Condenser vacuum was lost due to the auxiliary steam boiler tripping and loss of adequate main steam pressure. Steam Generator A was steaming through the Power Operated Emergency Main Steam Dump Valve (MS-V3A). Attempts to obtain a normal operating Pressurizer level of 220 inches of water and establish pressure control using the Pressurizer were not successful. The Electromatic

Time	Event	Information Available to the Operator	Reference
	Relief Block Valve (RC-V2) was cycled to assist in this effort, resulting in increased Reactor Building pressure. The first Engineered Safety Features actuation on high Reactor Building pressure was received and, four minutes later, bypassed by the operator to re-establish cooling water to various plant equipment within the Reactor Building. The Reactor Building pressure continued to stay above the isolation trip setpoint for approximately 1.4 hours (Figure 51). The Station Manager made the decision to maintain continuous High Pressure Injection and increase Reactor Coolant System pressure in an attempt to condense the superheated steam and gas in the Reactor Coolant System. This first attempt lasted for approximately 2 hours.		
05:20:00 (0920:37) Approximate	The operator increased Reactor Coolant System pressure from 1250 psig to 2100 psig during the ensuing 45 minutes. Reactor Coolant System pressure was then maintained at 2100 psig (Figure 4).	AN (Low-2055 and Low/Low-1900) at PL8 MR and SC at PL8 AP (many clearing alarms) (Delay $\simeq$ 88 minutes)	3a
05:23:34 (0924:11)	The operator cleared the Reactor Building Isolation and Cooling portion of Engineered Safety Features train A.	AN at PL13, ST at PL3 and PL13 AP norm/defeated (Delay $\simeq$ 90 minutes)	2a
05:23:34 (0924:11)	The Reactor Building Isolation and Cooling portion of Engineered Safety Features train A Actuated on Reactor Building high pressure. The setpoint is 3.58 psig (Figure 51).	AN at PL13, ST at PL3 and PL15 AP act/trip (Delay $\simeq$ 90 minutes)	2a,6a
05:23:34 (0924:11)	Intermediate Cooling Pump 1A (IC-P-1A) tripped automatically on the Engineered Safety Features train A Actuation	AN, ST, MR (P_{DISCH}) and MR(F) at PL8 AP on/off (Delay $\simeq$ 90 minutes)	2a
05:23:47 (0924:24)	The operator defeated the Reactor Building Isolation and Cooling portion of Engineered Safety Features train A.	AN at PL13, ST at PL3 and PL13 AP norm/defeated (Delay $\simeq$ 90 minutes)	2a

Time	Event	Information Available to the Operator	Reference
03:23:37 (0924:34)	The operator started Intermediate Cooling Pump 1A (IC-P-1A).	AM, ST, MR (P_{DISCH}) and MR(P) at PL8 AP on/off (Delay ~ 90 minutes)	2a
03:24:00 (0924:37)	During the next six minutes the operator removed the defeat signal from the Reactor Building Isolation and Cooling portion of Engineered Safety Features train A.	AM at PL13, ST at PL3 and PL13 AP norm/defeated (Delay ~ 90 minutes)	2a
03:29:23 (0930:00) Approximate	The operator secured the Fuel Handling Exhaust Fans (AH-E-10A and AH-E-10B or AH-E-10C and AH-E-10D) and the Auxiliary Building Exhaust Fans (AH-E-8C and AH-E-8D). Airborne contamination levels in Unit 1 Fuel Handling and Auxiliary Building, and Unit 2 Control Room began to increase.	Exhaust Fans: SC and ST at PL25 Radiation Levels: MR and MP (Unit 1 Control Room) MR and MP at PL12	3g,3h,3t,3u
03:29:23 (0930:00) Approximate	Fuel Handling Building Radiation Monitor (RP-R-215) and Control and Services Building Corridor Radiation Monitor (RP-R-234) began to increase to between 60 and 70 counts per minute. The radiation levels indicated by the following monitors increased until 14:59:23 (1900:00) at which time all were almost off scale. (a) Auxiliary Building Access Corridor Radiation Monitor (RP-R-232) (b) Reactor Building Purge Unit Area Radiation Monitor (HP-R-3236) (c) Fuel Handling Building Exhaust Unit Area Radiation Monitor (RP-R-3240)	MR and MP at PL12	3h,3b
03:30:34 (0931:11)	Pressurizer Heater Group 3 tripped due to a ground fault. Seven pressurizer heater groups were available at this time.	AM at PL8, ST at PL4 AP norm/trip (Delay ~ 93 minutes)	2a
03:34:37 (0935:34)	The Electromatic Relief Valve (RC-R2) and the Pressurizer Safety Valve (RC-R1A) discharge line high temperature alarm reset. Respective values of 192.0F and 192.9F were recorded.	MP at PL10 AP high (200F)/norm (Delay ~ 94 minutes)	2a

Time	Event	Information Available to the Operator	Reference
05:36:44 (0937:11)	Safety Injection logic of the Engineer Safety Features trains A and B reset on increasing Reactor Coolant System Pressure. The setpoint is 1665 psig.	AN at PL13, ST at PL3 and PL13 AP trip/norm (Delay $\simeq$ 95 minutes)	2a
05:39:27 (0940:04)	Pressurizer Safety Valve (RC-R1B) discharge line high temperature alarm reset. A value of 192.4F was recorded.	MP at PL10 AP high (200F)/norm (Delay $\simeq$ 97 minutes)	2a
05:41:06 (0941:43)	The operator cleared the safety injection portion of Engineered Safety Features trains A and B. At this time all Engineered Safety Features are in an armed condition.	AN at PL13, ST at PL3 and PL13 AP norm/defeated (Delay $\simeq$ 100 minutes)	2a
05:43:09 (0943:46) Approximate	The operator opened the Electromatic Relief Block Valve (RC-V2) to stop the Reactor Coolant System pressure increase. During the period 05:43:09 (0943:46) thru 07:38:57 (1139:34), the operator attempted to condense the steam in the Reactor Coolant System by maintaining high pressure injection and controlling the Reactor Coolant System pressure at approximately 2100 psig by cycling RC-V2 (Figure 4). The Reactor Building pressure and temperature reflected the cycling of RC-V2 (Figure 45).	RC P: MR and SC at PL4 RC-V2: ST at PL4 RB P: SC at PL3 RB T: MP at PL25	1,3j,3m,9j, 9k,9o,9m,10
05:43:27 (0944:04)	The Electromatic Relief Valve (RC-R2) discharge line high temperature alarm was received. A value of 214.7 was recorded.	MP at PL10 AP high (200F)/norm (Delay $\simeq$ 100 minutes)	2a
05:44:01 (0944:38)	Pressurizer Safety Valve (RC-R1B) discharge line high temperature alarm was received. A value of 203.4F was recorded.	MP at PL10 AP high (200F)/norm (Delay $\simeq$ 101 minutes)	2a
05:46:27 (0947:04)	Pressurizer Safety Valve (RC-R1A) discharge line high temperature alarm was received. A value of 205.9F was recorded.	MP at PL10 AP high (200F)/norm (Delay $\simeq$ 103 minutes)	2a

Time	Event	Information Available to the Operator	Reference
05:49:23 (0930:00) Approximate	Radiation levels as indicated by the Intermediate Cooling Pump Area Radiation Monitor (HP-R-207) and the Reactor Building Emergency Cooling Booster Pump Area Radiation Monitor (HP-R-204) peaked at a value 4×10^3 mR/hr. All Control Room Intake Duct radiation monitor (particulate, iodine, gas) (HP-R-220) level increased. The particulate channel reached a value 1×10^4 counts per minute while the iodine and gas channels reached a value 3×10^3 counts per minutes.	SC and MP at PL12	3a,5b
05:59:54 (1000:31) Approximate	The operator commenced filling Steam Generator A level from approximately 49% to 95% on the operating range in an attempt to induce natural circulation Steam Generator A level reached 95% on the operating range at 06:49:04 (1049:43).	MC (operating range) at PLA and PL3	1
06:03:23 (1004:00)	The Unit 1 Control Room ventilation system was placed in a recirculation mode.	SC and ST (Unit 1 Control Room)	4c
06:09:23 (1010:00) Approximate	Airborne radioactivity levels in Unit 2 control room required evacuation of all but essential personnel.	AM, MR and MP at PL12 (air supply) Unit 2 Control Room Air Sample data	4c,4g,9m, 9?,10
06:09:23 (1010:00) Approximate	The Nuclear Regulatory Commission Region I inspection team arrived on site and went to the Unit 1 Control Room upon direction of the Manager Generating Station Nuclear.	Unit 2 Control Room notified of Arrival of Nuclear Regulator Commission inspection team	9m,9n
06:13:39 (1014:16)	Pressurizer Heater Groups 1 and 2 de-energized automatically. Five pressurizer heater groups were available at this time.	AM at PL8, ST at PLA, AP norm/trip (Delay $\gtrsim$ 116 minutes)	2a
06:14:06 (1014:43)	Pressurizer Heater Groups 1 and 2 energized automatically. Seven pressurizer heater groups were available at this time.	AM at PL8, ST at PLA, AP norm/trip (Delay $\gtrsim$ 116 minutes)	2a

Time	Event	Information Available to the Operator	Reference
06:14:23 (1015:00) Approximate	The Emergency Control Station was moved from Unit 2 Control Room to the Unit 1 Control Room due to increased airborne radioactive levels.		8a
06:17:00 (1017:37) Approximate	Personnel in Unit 2 control room were required to wear respirators due to increased airborne radioactivity levels.		4c,4g,8b,8t, 9b,9f,9u
06:19:23 (1020:00) Approximate	The operator started the Fuel Handling Building Exhaust Fans (AH-E-10A and AH-E-10B or AH-E-10C and AH-E-10D) and the Auxiliary Exhaust Fans (AH-E-8C and AH-E-8D). These fans ran for approximately 3 minutes at which time they stopped.	SC and ST at PL25	3g,3h
06:47:23 (1048:00) Approximate	The operator started the Fuel Handling Building Exhaust Fans (AH-E-10A and AH-E-10B or AH-E-10C and AH-E-10D) and the Auxiliary Building Exhaust fans (AH-E-8C and AH-E-8D). Airborne contamination levels in Unit 1 Fuel Handling and Auxiliary Buildings began to decrease.	SC and ST and PL25	3g-3h,3t,3u, 4g
07:06:23 (1107:00) Approximate	Two Nuclear Regulatory Commission Region I inspectors arrive in Unit 2 Control Room to evaluate the operational and radiological status of Unit 2.		4h,8m,9u
07:08:31 (1109:08)	The operator started Emergency Feedwater Pump 2A (EF-P-2A) to increase Steam Generator A level from 95% to 100% on the operating range.	ST, WD(A) and WR (P_{DISCH}) at PLA AP on/off and low (875 psig)/norm (Delay ≈ 116 minutes)	2a
07:09:41 (1110:18)	The plant staff requested the computer to print the following Incore Thermocouples Outlet Temperatures. The following values were recorded. 8B = ***.°F 6C = ***.°F 9B = ***.°F 5C = ***.°F	UP AP norm/bad (out of range 0F to 700F) (Delay ≈ 116 minutes)	2c,9f

Time	Event	Information Available to the Operator	Reference
	9D = ***.*F 5B = ***.*F 9F = ***.*F 5E = ***.*F 9H = ***.*F 6L = 341.9F 7F = ***.*F 7H = ***.*F 7E = ***.*F 8B = ***.*F Note: (***.*) indicates that the signal was outside of the computer range (Range = 0F to 700F).		
07:10:12 (1110:49)	Steam Generator A Level indication reached 100% on the operating range (Figure 43).	SC (operating range) at PL4 and PL3	1
07:17:01 (1117:38)	The operator stopped Emergency Feedwater Pump 2A (EF-P-2A) after increasing Steam Generator A level from approximately 95% to 100% on the operating range.	ST, MR(A) and MR (P_{DISCH}) at PL4 AP on/off and low (875 psig)/norm (Delay $\simeq$ 116 minutes)	1,2a
07:38:57 (1139:34) Approximate	The Manager-Generating Station Nuclear directed the operator to open the Electromatic Relief Block Valve (RC-V2) to rapidly depressurize the Reactor Coolant System and actuate the Core Flood System while High Pressure Injection was maintained (Figure 12). This was done after the operator observed no evidence of natural circulation while the Reactor Coolant System pressure was above 2000 psig. The reduction in Reactor Coolant System pressure was also done to approach conditions which would allow the Decay Heat Removal Pumps 1A and 1B (DH-P-1A and DH-P-1B) to be put into service.	RC-V2: ST at PL4	1,3a,9j,9a, 9e,10
07:41:37 (1142:14)	The operator bypassed the Safety Injection portion of Engineered Safety Features trains A and B to prevent initiation during the Reactor Coolant System depressurization.	AM at PL13, ST at PL3 and PL13 AP norm/bypassed (Delay $\simeq$ 141 minutes)	2a,7

Time	Event	Information Available to the Operator	Reference
07:43:01 (1143:38)	The Safety Injection portion of Engineered Safety Features trains A and B actuated as Reactor Coolant System pressure decreased to less than 1640 psig. However, initiation did not occur because the manual bypass previously introduced was still on.	AM at PL13, ST at PL3 and PL13 AP norm/actuation (Delay ~ 166 minutes)	2a
07:43:44 (1144:21)	Pressurizer Heater Groups 1 and 2 tripped and re-energized after 2 seconds due to placing pressurizer heater control selection in the manual position.	ST at PL4. AP norm/trip (Delay $\simeq$ 145 minutes)	2a
07:50:16 (1150:53)	The operator de-energized Pressurizer Heater Groups 1 and 2 to assist in lower Reactor Coolant System pressure. Seven pressurizer heater groups were available at this time.	ST at PL4 AP norm/trip (Delay $\simeq$ 145 minutes)	2a
07:58:15 (1158:52)	The operator initiated pressurizer spray flow to assist in lowering Reactor Coolant System pressure. Pressurizer spray flow was maintained until 09:07:24 (1308:01).	PZR SPRAY: ST at PL4 RC P: VR and RC at PL4	1,8c,2a
07:59:37 (1200:14)	The operator requested the computer to print the outlet temperature (RC-10-TE3) of Pressurizer Safety Valve (RC-R1B). A temperature of 206.1F was recorded.	UP (Delay $\simeq$ 0 minutes) AP high (200F)/norm (Delay $\simeq$ 148 minutes) MP at PL10	2c
08:11:26 (1212:03)	Core Flood Tank 1A (CF-T-1A) high level alarm was received. The level was 13.32 feet.	AM and MR at PL3 AP norm/high (13.3 feet) (Delay $\simeq$ 150 minutes)	2a
08:16:58 (1217:35)	Pressurizer Safety Valve (RC-R1A) discharge line high temperature alarm reset. A temperature of 192.9F was recorded.	AP at PL10 AP high (200F)/norm (Delay $\simeq$ 154 minutes)	2a
08:22:58 (1223:35)	Pressurizer Safety Valve (RC-R1B) discharge line high temperature alarm reset. A temperature of 193.9F was recorded.	MP at PL10 AP high (200F)/norm (Delay $\simeq$ 156 minutes)	2a

Time	Event	Information Available to the Operator	Reference
08:30:00 (1230:37) Approximate	The Power Operated Emergency Main Steam Dump Valve (MS-V3A) was shut at the request of corporate management in response to concern expressed by the Pennsylvania State Government.	MR (valve demand setpoint) AT PL5	4g,4h,9x,9E, 10
08:31:06 (1231:43)	The operator started Decay Heat Removal Closed Cooling Pumps 1A and and 1B (DC-P-1A and DC-P-1B) in preparation for placing the Decay Heat System in service.	ST at PL3 and PL13, MR(P_{DISCH}) at PL8, MR(A) at PL3. AP norm/trip and on/off (Delay ≈ 155 minutes)	2e,9e
08:37:00 (12:37:37) Approximate	Reactor Coolant System pressure reached 600 psig which is the nominal gas pressure maintained in the Core Flood Tanks.	MR and SC at PL4	2e,3e,4e
08:40:15 (1240:52)	The plant staff requested the computer to print the following Incore Thermocouples Outlet Temperatures. The following values were recorded.	UP (Delay ≈ 8 minutes) AP norm/bad (out of range 0F to 700F)(Delay ≈ 132 minutes)	7c,9F

[illegible] = ***.*F	10N = 394.0F
[illegible] = ***.*F	11L = 370.5F
[illegible] = ***.*F	11R = 589.3F
[illegible] = ***.*F	12K = ***.*F
7F = ***.*F	13[illegible] = ***.*F
7G = ***.*F	13[illegible] = 304.7F
[illegible] = ***.*F	13F = ***.*F
[illegible] = ***.*F	12F = 321.6F
[illegible] = ***.*F	11G = 327.5F
[illegible] = ***.*F	11E = ***.*F
[illegible] = 653.2F	10D = 482.9F
[illegible] = ***.*F	10C = 582.5F
[illegible] = ***.*F	

Note: (***.*) indicates that the signal was outside of the computer range (Range = 0F to 700F).

Time	Event	Information Available to the Operator	Reference
08:54:56 (1255:33)	Core Flood Tank 1A (CF-T-1A) normal level alarm was received. The level was 13.13 feet. This indicated the Core Flood System injected a small amount of water into the Reactor Coolant System.	AN and MR at PL8. AP norm/high (13.3 feet) (Delay ≃ 150 minute)	2a, 3a
09:04:18 (1304:55)	The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C) and returned the Reactor Coolant Makeup System to one pump operation.	AN at PL8, St and MR(A) at PL3 AP norm/trip and norm/off (Delay ≃ 150 minutes)	2a,3c
09:04:23 (1305:00) Approximate	Personnel in Unit 1 Control Room were required to wear respirators due to increased airborne radioactivity levels. Excess personnel were moved to the Observation Center.	AN, MR and MP (Unit 1 Control Room) Unit 1 Control Room air sample data	4g,4h
09:14:23 (1315:00) Approximate	The operator shut the Electromatic Relief Block Valve (RC-V2).	ST at PLA	3j
09:16:58 (1317:35)	The Electromatic Relief Valve (RC-R2) discharge line high temperature alarm reset. A temperature of 192.7F was recorded.	MP at PL18 AP high (200F)/norm (Delay ≃ 170 minutes)	2a
09:20:28 (1321:05) Approximate	The operator opened the Electromatic Relief Block Valve (RC-V2).	ST at PLA	3j
09:20:28 (1321:05)	The Electromatic Relief Valve (RC-R2) discharge line high temperature alarm was received. A temperature of 220.4F was recorded.	MP at PL19 AP high (200F)/norm (Delay ≃ 167 minutes)	2a
09:30:00 (1330:37) Approximate	The operator closed the Electromatic Relief Block Valve (RC-V2).	ST at PLA	3j
09:30:18 (1330:55)	The operator started the computer on a two minute group trend of the following plant parameters. This trend was maintained for the duration of March 28, 1979.	GP (Delay ≃ 0 minutes)	2c

Time	Event	Information Available to the Operator	Reference
	Reactor Coolant Drain Tank Temperature		
	Reactor Coolant Pressurizer Water Level		
	Reactor Coolant Loop A Outlet Temperature		
	Reactor Coolant Pressurizer Temprature		
	Steam Generator B Shell Temperature		
	Reactor Coolant Loop A Pressure		
	Steam Generator B Steam Temperature		
	Steam Generator B Upper Downcomer Temperature		
	Steam Generator A Downcomer Temperature		
	Steam Generator A Steam Pressure		
	Steam Generator A Shell Temperature		
	Steam Generator A Upper Downcomer Temperature		
09:31:28 (1332:05)	The Electromatic Relief Valve (RC-R2) discharge line high temperature alarm reset. A temperature of 192.2F was recorded.	MP at PL10 AP high (200F)/norm (Delay $\simeq$ 157 minutes)	2a
09:48:58 (1349:35) Approximate	The operator opened the Electromatic Relief Block Valve (RC-V2). However upon receiving the Reactor Building pressure spike of 28 psig at 09:49:44 (1350:21), the operator immediately closed (RC-V2).	ST at PL4	9a,9D
09:48:58 (1349:35)	The Electromatic Relief Valve (RC-V2) discharge line high temperature alarm was received. A temperature of 225.7 was recorded.	MP at PL10 AP high (200F)/norm (Delay $\simeq$ 160 minutes)	2a
09:49:43 (1350:20)	A hydrogen detonation occurred in the Reactor Building. Hydrogen gas resident in the reactor coolant and hydrogen gas generated from the reaction between zirconium fuel cladding and the reactor coolant had collected in the Pressurizer. This gas had been vented through the Electromatic Relief Valve (RC-R2) to the Reactor Coolant Drain Tank and released to the Reactor Building through the Drain	RB P: MR and SC at PL3 RB T: SC at PL25	7b,3j,8a,8h, 9a

Time	Event	Information Available to the Operator	Reference
	Tank Rupture Diaphragm (WDL-U26) which had been breached. The hydrogen concentration in the containment eventually reached an explosive mixture This detonation resulted in a Reactor Building pressure spike of 28 psig with a corresponding rapid increase in Reactor Building air temperature.		
09:49:43 (1350:20) Approximate	Motor Control Center 37A and 42A were lost. These supply power to the seal water pump which furnishes seal water to many of the Radwaste Pumps in the Auxiliary Building.	Local indication at the Radwaste panel	5b,8m,9J
09:49:44 (1350:21)	The Reactor Building Isolation and Cooling portion of Engineered Safety Features trains A and B actuated on high and high-high Reactor Building Pressure (Figure 51). The setpoints are 3.58 psig and 28 psig respectively. This was a result of the 28 psig Reactor Building pressure impulse from the hydrogen denonation. Reactor Building isolation, cooling and containment spray were actuated.	AN at PL13, ST at PL3 and PL13 AP norm/act (Delay ≈ 159 minutes)	2a,2h,3j,4a
09:49:44 (1350:21)	Decay Heat Removal Pumps 1A and 1B (DH-P-1A and 1B) started and intermediate cooling pumps 1A and 1B (IC-P-1A and 1B) tripped automatically on the Engineered Safety Features Trains A and B actuations.	DH-P-1A/1B: ST at PL13 and PL3, MR(P_{DISCH}) at PL8 M(A) at PL3, AP on/off and norm/trip (Delay ≈ 159 minutes) IC-P-1A/1B: AN, ST, MR (P_{DISCH}) and MR (F) at PL8 AP on/off (Delay ≈ 159 minutes)	2a
09:49:47 (1350:23)	Reactor Coolant Makeup Pump C (MU-P-1C) started automatically by the Engineering Safety Feature Train A actuation.	AN at PL8, MR(A) and ST at PL3 AP norm/trip (Delay ≈ 159 minutes)	2a,5c
09:49:49 (1350:26)	Reactor Building Spray Pumps 1A and 1B (BS-P-1A and 1B) started automatically upon actuation of Engineered Safety Features trains A and B.	ST at PL13 and PL15 AP norm/trip (Delay ≈ 160 minutes)	2a
09:49:58 (1350:35)	Reactor Coolant Pumps 1A and 1B (RC-P-1A and RC-P-1B) inlet air high temperature alarms and Pressurizer Safety Valves (RC-R1A and RC-R1B) discharge line high temperature alarms were received.	RC-P-1A/1B T: AP norm/high (122F)(Delay ≈ 161 minutes) RC-R1A/1B: AP norm/high (Delay ≈ 161 minutes)	2a

Time	Event	Information Available to the Operator	Reference
09:50:09 (1350:46)	The operator defeated the Reactor Building Isolation and Cooling portion of Engineered Safety Features train A.	AM at PL13, ST at PL3 and PL13 AP norm/defeated (Delay ≃ 162 minutes)	2a
09:50:09 (1350:46)	The operator started Intermediate Cooling Pump 1A (IC-P-1A).	AM, ST, MR (P_{DISCH}) and MR(F) at PL8 AP on/off (Delay ≃ 162 minutes)	2a
09:50:11 (1350:48)	The operator defeated the Reactor Building Isolation and Cooling portion of Engineered Safety Features train B.	AM at PL13, ST at PL3 and PL13 AP norm/defeated (Delay ≃ 162 minutes)	2a
09:50:11 (1350:48)	The operator started Intermediate Cooling Pump 1B (IC-P-1B).	AM, ST, MR (P_{DISCH}) and MR(F) at PL8 AP on/off (Delay ≃ 162 minutes)	2a
09:50:24 (1351:01)	The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C)	AM at PL8, ST and MR(A) at PL3 AP norm/trip (Delay ≃ 162 minutes)	2a,5c,9N
09:51:58 (1352:35)	The Electromatic Relief Valve (RC-R2) and the Pressurizer Safety Valve (RC-R1A) discharge line high temperature alarm reset. Respective temperatures of 180.5F and 178.4F was recorded.	MP at PL10 AP high (200F)/norm (Delay ≃ 162 minutes)	2a
09:52:28 (1353:05)	The Electromatic Relief Valve (RC-R2) discharge line high temperature alarm was received. A temperature of 208.8F was recorded. It is believed that this was a result of the operator cycling the Electromatic Relief Block Valve (RC-V2).	MP at PL10 AP high (200F)/norm (Delay ≃ 161 minutes)	2a
09:52:55 (1353:32)	The operator, during the next minute, removed the defeat signal from the Reactor Building isolation and cooling portion of Engineered Safety Features trains A and B.	AM at PL13, ST at PL3 and PL13 AP norm/defeated (Delay ≃ 161 minutes)	2a
09:54:28 (1355:05)	The Electromatic Relief Valve (RC-R2) discharge line high temperature alarm reset. A temperature of 177.3F was recorded.	MP at PL10 AP high (200F)/norm (Delay ≃ 161 minutes)	2a

Time	Event	Information Available to the Operator	Reference
09:55:10 (1355:47)	Pressurizer Heater Group 8 tripped due to a ground fault. Six pressurizer heater groups were available at this time.	ST at PL4 AP norm/trip (Delay $\tilde{=}$ 160 minutes)	2a
09:55:28 (1356:05)	Pressurizer Safety Valve (RC-R1B) discharge line high temperature alarm reset. A temperature of 190.3F was recorded.	MP at PL10 AP high (200F)/norm (Delay $\tilde{=}$ 160 minutes)	2a
09:55:30 (1356:07)	The operator stopped Reactor Building Spray Pumps A and B (BS-P-1A and BS-P-1B). BS-P-1A and BS-P-1B were operated for approximately 5 minutes and 60 seconds.	AR at PL8, MR(A), MR(F) and ST at PL3, ST at PL13, AP Norm/trip (Delay $\tilde{=}$ 160 minutes)	2a,9e
09:56:26 (1357:03)	Core Flood Tank 1A (CF-T-1A) high water level alarm was received. The level recorded was 13.72 feet. One minute later, 09:57:26 (1358:03), a normal signal was received and a level of 12.85 feet recorded. This indicated a discharge of water from Core Flood Tank 1A to the Reactor Coolant System. Note: Over the next 150 minutes similar level changes in the Core Flood Tank 1A were noted indicating periodic discharge of water from Core Flood Tank 1A into the Reactor Coolant System. No core flood tank 1B (CF-T-1B) level alarms were received.	MR(L) at PL8 AP norm/high (13.3 feet) (Delay $\tilde{=}$ 160 minutes)	
09:56:58 (1357:35)	The operator stopped Decay Heat Pumps A and B (DH-P-1A and DH-P-1B). The Reactor Coolant System pressure was never low enough to put the Decay Heat System in service.	ST at PL3 and PL13, MR(A) at PL3, MR(P_{DISCH}) at PL8 AP on/off and norm/trip (Delay $\tilde{=}$ 160 minutes)	2a
09:59:26 (1400:03)	The following Incore Thermocouple temperatures decreased to less than 700F over the next minute. The increased reactor core cooling was a result of Core Flood Tank (CF-T-1A) discharge into the reactor vessel. 6L = 687.2F 13K = 666.7F 10M = 661.9F 8M = 681.6F	AP norm/bad (out of range 0F to 700F) (Delay $\tilde{=}$ 160 minutes)	2a

Time	Event	Information Available to the Operator	Reference
10:00:58 (1601:35) Approximate	The operator opened the Electromatic Relief Block Valve (RC-V2).	ST at PLA	33
10:00:58 (1601:35)	The Electromatic Relief Valve (RC-R2) and the Pressurizer Safety Valve (RC-R1B) discharge line high temperature alarms were received. Respective temperatures of 206.7F and 200.0F were recorded.	MP at PL10 AP high (200F)/norm (Delay ≅ 159 minutes)	2a
10:04:30 (1605:07)	The operator initiated pressurizer spray flow to assist in lower Reactor Coolant System Pressure. Pressurizer spray flow was maintained until 12:05:54 (1606:71).	PZR SPRAY: ST at PLA RC P: MD and SC at PLA	1,8a
10:05:25 (1606:02)	The operator energized Pressurizer Heater Groups 1 and 2. Six pressurizer heater groups were available at this time.	ST at PLA AP norm/trip (Delay ≅ 156 minutes)	2a
10:07:19 (1407:56)	The operator de-energized Pressurizer Heater Groups 1 and 2. Six pressurizer heater groups were available at this time.	ST at PLA AP norm/trip (Delay ≅ 156 minutes)	2a
10:23:58 (1424:35)	Pressurizer Safety Valve (RC-R1B) discharge line high temperature alarm reset. A temperature of 192.9F was recorded.	MP at PL10 AP high (200F)/norm (Delay ≅ 152 minutes)	2a
10:26:18 (1426:55)	Reactor Coolant System Loop A hot leg temperature decreased to within the instrumentation range (Figure 24). This was the result of the steam in Loop A hot leg condensing.	SC at PL4 and PL10, MR at PL4, AR at PL8	1,93,9D,9J,9M
10:29:13 (1430:00) Approximate	At the request of the State government, the Metropolitan Edison Company Vice President of Generation traveled to the Governor's office to report the TMI Plant status. He was accompanied by the Manager-Generating Station Nuclear and Unit 2 Superintendent Technical Support. The Emergency Director designee was directed	Announcement made to personnel in Unit 2 Control Room	9k,9u,9x,10

Time	Event	Information Available to the Operator	Reference
	to maintain the plant in a stable condition during their absence. The Manager-Generating Station Nuclear carried a remote paging device to permit him to be contacted, if necessary.		
10:31:25 (1432:07)	The operator started Reactor Coolant Makeup Pump C (MU-P-1C) after an unsuccessful attempt to start the pump at 10:31:20 (1432:02). Reactor coolant pressure was approximately 440 psig.	AB at PL8, MR(A) and ST at PL3 AP norm/trip (Delay ≈ 123 minutes)	2a,3a
10:32:36 (1433:13)	The operator energized Pressurizer Heater Groups 1 and 2. Six pressurizer heater groups were available at this time.	ST at PL4 AP norm/trip (Delay ≈ 120 minutes)	2a
10:35:37 (1436:14)	The Reactor Coolant System pressure reached a minimum value of 409 psig.	HP (Delay ≈ 0 minutes)	2c
10:35:[illegible] (1436:[illegible])	The Reactor Coolant System Loop A hot leg temperature increased beyond the range of the instrumentation when High Pressure Injection was directed to Loop B Hot Leg (Figures 24 and 29).	SC at PL4, HP at PL10 and HR at PL4, AD(high = 612 F) at PL8	1,4b,9j,9D, 9J,9M
10:35:55 (1436:32)	The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C).	AB at PL8, MR(A) and ST at PL3 AP norm/trip (Delay ≈ 113 minutes)	2a,5c
10:38:57 (1439:34)	The operator de-energized Pressurizer Heater Groups 1 and 2. Six pressurizer heater groups were available at this time.	ST at PL4 AP norm/trip (Delay ≈ 105 minutes)	2a
10:39:30 (1440:07)	Reactor Coolant System Loop A hot leg temperature decreased to within the instrumentation range. This was the result of the steam in Loop A hot leg again collapsing when High Pressure Injection to Loop A hot leg was re-established.	SC at PL4, HP at PL10, HR at PL4, AD(high = 612F) at PL8	1,4b,9j,9D, 9J,9M

Time	Event	Information Available to the Operator	Reference
10:39:51 (1440:28)	The operator energized Pressurizer Heater Groups 1 and 2. Six pressurizer heater groups were available at this time.	ST at PL4 AP norm/trip (Delay ~ 182 minutes)	2a
10:53:33 (1454:10)	The Reactimeter Data Acquisition System monitoring tape was removed and another tape started by plant staff personnel. This operation was required daily due to the recording time of three seconds and the tape storage capability. The reactimeter was off line between 10:53:33 (1454:10) to 10:57:23 (1458:00).	Unit 2 control room notified of reactimeter tape changeout.	1
11:00:13 (1501:10)	The alarm printer malfunctioned between 11:00:13 (1501:10) to 11:01:38 (1502:15). During this period the alarm printer function was performed by the utility printer.	AP (Delay ~ 0 minutes) Paper feed problem in printer	2a,2c
11:07:35 (1508:12)	Pressurizer level started decreasing from 398 inches to 174 inches over a period of 13 minutes (Figure 33).	RC HT PL 4, AM (High/High = 315 inches, High = 209 inches, Low = 200 inches and Low/Low = 80 inches) at PL8.	1
11:09:23 (1510:00)	The airborne radioactivity level in Unit 2 Control Room reduced to a level such that personnel were permitted to remove respirators.	AM, MR and MP at PL12 (air supply) Unit 2 control room air sample data	4b,4g
11:12:00 (1512:37) Approximate	The operator shut the Electromatic Relief Block Valve (RC-V2).	ST at PL4	3j,8m
11:15:47 (1516:24)	Reactor Coolant System Loop A cold leg temperature started to increase from 215°F to 415°F indicating the occurrence of some natural circulation in Loop A (Figure 24)	MP at PL10	1,9j,9M
11:18:34 (1519:11)	The operator started Reactor Coolant Makeup Pump C (MU-P-1C) to stop the rapid fall in the Pressurizer level.	AM at PL8, MR(A) and ST at PL3 AP norm/trip (Delay ~ 84 minutes)	1,2a,5c,9M

Time	Event	Information Available to the Operator	Reference
11:21:35 (1522:12)	Pressurizer level stopped decreasing at 174 inches and started slowly increasing, going off scale during the next hour (Figure 33).	SC at PL4	1
11:24:31 (1525:08)	The Electromatic Relief Valve (RC-R2) discharge line high temperature alarm reset. A temperature of 191.9F was recorded.	MP at PL10 AP high (200F)/norm (Delay ≃ 84 minutes)	2a
11:26:33 (1527:10)	The operator requested the computer to print the outlet temperature (RC-10-TE1, RC-10-TE2 and RC-10-TE3) of the Electromatic Relief Valve (RC-R2) and the Pressurizer Safety Valves (RC-R1A and RC-R1B). Respective values of 198.7F, 170.6F and 174.7F were indicated.	UP (Delay ≃ 0 minutes) AP high (200F)/norm (Delay ≃ 84 minutes) MP at PL10	2c
11:28:12 (1528:49)	The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C).	AB at PL8, MR(A) and ST at PL3 AP norm/trip (Delay ≃ 85 minutes)	2a,5c
11:28:52 (1529:29)	The operator de-energized Pressurizer Heater Groups 1 and 2 due to the low pressurizer level. Six pressurizer heater groups were available at this time.	ST at PL4 AP norm/trip (Delay ≃ 85 minutes)	2a,9M
11:29:23 (1530:00) Approximate	The Auxiliary Building exhaust fans (AH-E-8C and AH-E-8D) stopped. Airborne contamination levels in Unit 1 Fuel Handling and Auxiliary Building, and Unit 2 Control Room began to increase. The reason for these fans stopping is unknown.	Exhaust Fans: SC and ST at PL23 Radiation Levels: MR and MP (Unit 1 control room) MR and MP at PL12	3g,3a
11:32:37 (1533:14)	The operator started Reactor Coolant Makeup Pump C (MU-P-1C) to raise the pressurizer level.	AN at PL8, MR(A) and ST at PL3 AP norm/trip (Delay ≃ 84 minutes)	2a,5c,9M
11:33:44 (1534:21)	The operator started Emergency Feedwater Pump 2B (EF-P-2B) to increase Steam Generator B level.	ST, MR(A) and MR (P_{DISCH}) at PL4 AP on/off and low (875 psig)/norm (Delay ≃ 85 minutes)	2a

Time	Event	Information Available to the Operator	Reference
11:33:48 (1534:25)	The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C).	AN at PL8, MR(A) and ST at PL3 AP norm/trip (Delay ≃ 83 minutes)	2a,5c
11:38:38 (1539:15)	The operator started filling Steam Generator B from 57% to 97% on the operating range in an attempt to induce natural circulation (Figure 39). The 97% level was reached at 11:52:04 (1552:41).	SC (operating range) at PL4 and PL5	1
11:45:17 (1545:54)	The operator energized Pressurizer Heater Groups 1 and 2. Six pressurizer heater groups were available at this time.	ST at PL4 AP norm/trip (Delay ≃ 84 minutes)	2a
11:49:23 (1550:00) Approximate	The Fuel Handling exhaust flow fluctuated during the next three hours. The reason for the flow fluctuation is unknown.	SC and ST at PL25	3b
11:52:04 (1552:41)	Steam Generator B level indication reached 97% on the operating range (Figure 45). The operator stopped Emergency Feedwater Pump 2B (EF-P-2B).	SC (operating range) at PL4 and PL5 ST, MR(A) and MR (P_{DISCH}) at PL4 AP on/off and low (875 psig)/norm (Delay ≃ 85 minutes)	1,2a
12:10:55 (1611:32)	The plant staff requested the computer to print the following Incore Thermocouple Outlet Temperatures. The following values were recorded.	UP (Delay ≃ 0 minutes) AP norm/bad (out of range 0F to 700F)(Delay ≃ 50 minutes)	2c,9f

[illegible] = ***.*F	5G = ***.*F
9H = 396.9F	5H = ***.*F
9G = ***.*F	5E = ***.*F
8F = ***.*F	6L = 362.1F
9E = ***.*F	7K = ***.*F
7F = ***.*F	[illegible] = ***.*F
7E = ***.*F	[illegible] = ***.*F
8G = ***.*F	

Note (***.*) indicates that the signal was outside of the computer range (Range = 0F to 700F)

Time	Event	Information Available to the Operator	Reference
12:34:29 (1633:06) Approximate	The operator opened the Electromatic Relief Block Valve (RC-V2).	ST at PL4	3j,2c
12:29:23 (1630:00) Approximate	The operator started the Auxiliary Building exhaust fans (AH-E-8C and AH-E-8D). Airborne contamination levels in Unit 1 Fuel Handling and Auxiliary Buildings, and Unit 2 Control Room began to increase.	Exhaust Fans: SC and ST at PL25 Radiation Levels: MB and MP (Unit 1 Control room) MB and MP at PL12	3g, 3m
12:34:29 (1635:06)	The Electromatic Relief Valve (RC-R2) discharge line high temperature alarm was received. A temperature of 233.0F was recorded.	MP at PL10 AP high (200F)/norm (Delay $\simeq$ 0 minutes)	2a
12:34:59 (1635:36)	Pressurizer Safe y Valve (RC-R1B) discharge line high temperatere alarm was received. A temperature of 203.2F was recorded.	MP at PL10 AP high (200F)/norm (Delay $\simeq$ 0 minutes)	2a
12:36:29 (1637:06)	Pressurizer Safety Valve (RC-R1A) discharge line high temperature alarm was received. A temperature of 201.4F was recorded.	MP at PL10 AP high (200F)/norm (Delay $\simeq$ 0 minutes)	2a
12:43:00 (1643:37) Approximate	The operator closed the Electromatic Relief Block Valve (RC-V2).	ST at PL4	3j
12:44:53 (1645:30)	Pressurizer level indication came on scale (Figure 33).	SC at PL4	1
12:52:00 (1652:37) Approximate	The operator opened the Electromatic Relief Block Valve (RC-V2).	ST at PL4	3j
12:58:33 (1659:10)	Pressurizer Safety Valve (RC-R1A) discharge line high temperature alarm reset. A temperature of 192.2F was recorded.	MP at PL10 AP high (200F)/norm (Delay $\simeq$ 0 minutes)	2a
13:02:23 (1703:00)	The operator started Condenser Vacuum Pump 1C (VA-P-1C) in an attempt to re-establish vacuum. The auxiliary boiler had been returned to service and was supplying gland sealing steam to the main turbine.	ST at PL17 AP on/off (Delay $\simeq$ 0 minutes)	2a,4g

Time	Event	Information Available to the Operator	Reference
13:05:29 (1706:00)	Pressurizer Safety Valve (RC-R18) discharge line high temperature alarm reset. A temperature of 192.8F was recorded.	UP at PL10 AP high (200F)/norm (Delay ~ 0 minutes)	2a
13:13:10 (1713:67)	The operator started Condenser Vacuum Pump 1A (VA-P-1A).	ST at PL17 AP on/off (Delay ~ 0 minutes)	2a

PLANT STATUS

All Reactor Coolant Pumps (RC-P-1A, RC-P-2A, RC-P-1B and RC-P-2B) were stopped. Steam and gas existed in the Reactor vessel head and Loop B hot leg. The Loop A hot leg steam had been collapsed and natural circulation flow established in this loop. Condenser vacuum was established after returning the auxiliary steam boiler to service. Steam Generator A level was 97% of the operate range (Figure 39). Steam Generator B was isolated, with a level at 93% of the operate range (Figure 39). The Electromatic Relief Block Valve (RC-V2) was open, keeping the Reactor Coolant System depressurized to 650 psig (Figure 12). Venting through Electromatic Relief Valve (RC-R2) to the reactor building resulted in a hydrogen concentration increase and subsequent denotation which caused a 28 psig pressurize pulse in the Reactor Building. Attempts to use the Core Flood System to cool the core over the last six hours had resulted in limited success. The reactor core was being cooled by (1) High Pressure Injection flow into the Reactor Coolant System and then to the Reactor Building Floor via the Electromatic Relief Valve (RC-R2) and (2) Core Flood Tank A partial discharge.

Time	Event	Information Available to the Operator	Reference
13:15:00 (1715:37) Approximate	The operator shut the Electomatic Relief Block Valve (RC-V2) in an attempt to condense the remaining steam in the Reactor Coolant System by increasing Reactor Coolant System pressure (Figure 12).	ST at PLA	1,3j,3a,4g,9j,9a,9u,10

Time	Event	Information Available to the Operator	Reference
13:23:04 (1723:41)	The operator started Reactor Coolant Makeup Pump C (MU-P-1C) to further increase Reactor Coolant System pressure.	MU-P-1C: AM at PL8, MR(A) and ST at PL3 AP norm/trip (Delay ≈ 0 minutes) RC P: MR and SC at PLA	2a,3a,5c,9j
13:24:59 (1725:36)	The Electromatic Relief Valve (RC-R2) discharge line high temperature alarm reset. A value of 192.9F was recorded.	MP at PL10 AP high (200F)/norm (Delay ≈ 0 minutes)	2a
13:26:09 (1726:46)	The operator de-energized pressurizer Heater Groups 1 and 2. Six pressurizer heater groups were available at this time.	SC at PLA AP norm/trip (Delay ≈ 0 minutes)	2a
13:44:23 (1743:00)	The operator started steaming Steam Generator A to the condenser using Turbine Bypass Valves (MS-V25A and MS-V25B).	MS-V25A/25B: at PL 5	4b
14:25:26 (1826:03)	The operator energized Pressurizer Heater Groups 1 and 2. Six pressurizer heater groups were available at this time.	ST at PLA AP norm/trip (Delay ≈ 0 minutes)	2a
14:25:31 (1826:08)	The plant staff requested the computer to print the following Incore Thermocouple Outlet Temperatures. The following values were recorded. 9B = ***.*F 9C = ***.*F 9E = ***.*F 8F = ***.*F Note: (***.*) indicates that the signal was outside of the computer range (Range 0F to 700F)	UP (Delay ≈ 0 minutes)	2c,9f
14:32:01 (1832:38)	Safety Injection logic of the Engineer Safety Features trains A and B reset on increasing Reactor Coolant System pressure. The setpoint is 1665 psig.	AM at PL13, ST at PL3 and PL13 AP norm/trip (Delay ≈ 0 minutes)	2a
14:34:30 (1835:07)	The operator cleared the Safety Injection portion of Engineered Safety Features trains A and B. At this time all Engineered Safety Features are in an armed condition.	AM at PL13, ST at PL3 and PL13 AP norm/bypassed (Delay ≈ 0 minutes)	2a

Time	Event	Information Available to the Operator	Reference
14:43:15 (1843:53)	The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C) to slow the rapid increase in Reactor Coolant System pressure.	MU-P-1C: AN at PL8, MR(A) and ST at PL3 AP norm/trip (Delay ≃ 0 minutes) RC P: MR and SC at PL4	2a,5c
14:48:22 (1848:59)	The alarm printer malfunctioned due to paper feed problems. As a result the alarm summary data from 14:48:22 (1848:59) to 15:09:52 (1910:29) was deleted.	AP (Delay ≃ 0 minutes) Paper feed problem in printer	2a,9W
14:54:35 (1853:12) Approximate	Reactor Coolant System pressure reached 2332 psig (Figure 14).	MR and SC at PL4, AN (Low/Low = 1900 psig Low = 2055 psig and High = 2500 psig) at PL8	1,4g
14:59:23 (1900:00) Approximate	The following radiation monitors indicated on scale and continued decreasing. (a) Reactor Building Purge Unit Area radiation monitor (HP-R-3236) (b) Auxiliary Building Access Corridor radiation monitor (HP-R-232) (c) Waste Disposal Storage Area radiation monitor (HP-R-218) (d) Fuel Handling Building Exhaust Unit Area radiation monitor (HP-R-3240) The Fuel Handling Building radiation monitor (HP-R-215) and Control and Service Building Corridor radiation monitor (HP-R-236) indication was steady about 10 counts per minutes.	AN, MR and MP at PL12	3b,5b
15:11:22 (1911:59)	The operator requested the computer to print a summary of Reactor Coolant Pumps and Makeup Pumps parameter status.	UP (Delay ≃ 0 minutes)	2c

Time	Event	Information Available to the Operator	Reference
15:32:33 (1933:10)	The operator started Reactor Coolant Pump 1A (RC-P-1A) and after 10 seconds stopped the pump. This is the normal procedure on a routine loop refill. Reactor Coolant System pressure decreased from 2320 psig to 1440 psig, Loop A cold leg temperature decreased from 410F to 270F and Loop A hot leg temperature decreased from 545F to offscale low (i.e. less than 520F) (Figures 14 and 24).	RC-P-1A: ST, MR(A) and MR(F) at PLA, AN (trip) at PL8 AP norm/trip (Delay $\simeq$ 0 minutes) RC T_C: MP at PL10 RC P: MR and SC at PLA	1,2a,3m,3n 4b,4g,9j,9e, 9y,9M
15:32:46 (1933:23)	The Safety Injection portion of Engineered Safety Features trains A and B actuated as Reactor Coolant System pressures fell below 1640 psig.	AN at PL13, ST at PL3 and PL13 AP norm/actuation (Delay $\simeq$ 0 minutes)	2a
15:32:46 (1933:23)	The operator attempt to start Reactor Coolant Makeup Pump 1C (MU-P-1C) due to the falling reactor coolant system pressure.	AN at PL8, ST and MR(A) at PL3 AP norm/trip (Delay $\simeq$ 0 minutes)	2a,5c
15:32:47 (1933:24)	Decay Heat Removal Pumps 1A and 1B (DH-P-1A and 1B) started automatically upon Engineered Safety Features trains A and B Actuation.	AN and MR (P_{DISCH}) at PL8, ST at PL3 and PL13 MR(A) at PL3 AP norm/low and on/off (Delay $\simeq$ 0 minutes)	2a
15:32:52 (1933:29)	The operator started Reactor Coolant Makeup Pump 1C (MU-P-1C).	AN at PL8, ST and MR(A) at PL3 AP norm/trip (Delay $\simeq$ 0 minutes)	2a,5c
15:33:03 (1933:40)	The following Incore Thermocouple Temperatures decreased to less than 700F over the next minute. The increased Reactor Core Cooling was a result of starting Reactor Coolant Pump 1A (RC-P-1A) for 10 seconds. 9R = 524.6F 5D = 491.0F 5K = 604.0F 3M = 348.9F 8H = 585.3F 13L = 251.1F	AP norm/bad (out of range 0F to 700F) (Delay $\simeq$ 0 minutes)	2a

Time	Event	Information Available to the Operator	Reference
15:33:07 (1933:44)	The operator manually bypassed the Safety Injection portion of Engineered Safety Features trains A and B.	AN at PL13, ST at PL3 and PL13 AP norm/bypassed (Delay ≃ 0 minutes)	2a
15:35:18 (1935:55)	Safety Injection actuation logic of the Engineer Safety Features trains A and B reset on increasing Reactor Coolant System pressure. The setpoint is 1665 psig.	AN at PL13, ST at PL3 and PL13 AP norm/trip (Delay ≃ 0 minutes)	2a
15:38:42 (1939:19)	The operator stopped Reactor Coolant Makeup Pump 1C (MU-P-1C)	AN at PL8, MR(A) and ST at PL3 AP norm/trip (Delay ≃ 0 minutes)	2a,5c
15:39:27 (1940:04)	The operator cleared the Safety Injection portion of Engineered Safety Features trains A and B.	AN at PL13, ST at PL3 and PL13 AP norm/bypassed (Delay ≃ 0 minutes)	
15:49:16 (1949:53)	The operator started Reactor Coolant Makeup Pump 1C (MU-P-1C).	AN at PL8, MR(A) and ST at PL3 AP norm/trip (Delay ≃ 0 minutes)	2a,5c
15:49:59 (1950:36)	The operator started Reactor Coolant Pump 1A (RC-P-1A); Reactor Coolant pressure decreased from 2190 psig to 1300 psig. Loop A cold leg temperature decreased from 345F to 260F and Loop A hot leg temperature remained offscale low (i.e. less than 520F) (Figures 14 and 24).	RC-P-1A: ST, MR(A), MR(F) and BC(A) at PL4 AN (trip) at PL8 AP norm/trip at PL8 (Delay ≃ 0 minutes) RC T_C: MP at PL10 RC P: MR and BC at PL4	1,2a,3a, 4b,9j,9a,9e, 9y,9M
15:50:08 (1950:45)	Reactor Coolant System Loop B hot leg temperature decreased from offscale high (greater than 620F) to offscale low (less than 520F) (Figure 29).	AN(high = 612F) at PL8, BC at PL4 and MP at PL10	1
15:50:13 (1950:50)	The operator began to bypass the Safety Injection portion of Engineered Safety Features trains A and B because Reactor Coolant System.	AN at PL13, ST at PL3 and PL13 AP norm/bypassed (Delay ≃ 0 minutes)	2a

Time	Event	Information Available to the Operator	Reference
	pressure was near the initiation setpoint of 1640 psig. He succeeded in defeating train A, but train B did initiate. After 10 seconds train B was bypassed.		
15:50:33 (1951:10)	The following Incore Thermocouple Temperatures decreased below 700F during the next four minutes. This was in response to Reactor Coolant flow through the Reactor core as a result of starting Reactor Coolant Pump 1A (RC-P-1A). 9H = 468.3F 13H = 546.3F 7M = 465.9F 12K = 671.0F 5D = 364.8F	AP norm/bad (out of range 0F to 700F) (Delay ≃ 0 minutes	2a
15:55:03 (1955:40)	The operator stopped Decay Heat Removal Pumps (DH-P-1A and 1B).	AB and MR(P_{DISCH}) at PL8, ST at PL3 and PL13 MR(A) at PL3 AP norm/low and on/off (Delay ≃ 0 minutes)	2a
15:56:06 (1956:43)	The operator stopped Reactor Coolant Makeup Pump 1C (MU-P-1C).	AB at PL8, MR(A) and ST at PL3 AP norm/trip (Delay ≃ 0 minutes)	2a,3c
16:56:59 (2057:36)	The plant staff requested the computer to print the following Incore Thermocouple Outlet Temperatures. The following values were recorded. 8H = ***.*F 5H = ***.*F 9H = 408.9F 5K = 368.6F 9C = 498.1F 7H = 377.9F 8F = ***.*F 8H = 326.8F 9E = 637.6F 9H = 560.4F 7F = 249.7F 11L = 211.5F 7E = 509.8F 12L = 332.1F 8G = 347.0F 13H = ***.*F 5G = ***.*F	UP (Delay ≃ 0 minutes)	2c,9f
	Note: (***.*) indicates that the signal was outside of the computer range (Range = 0F to 700F).	UP	

Time	Event	Information Available to the Operator	Reference
16:59:23 (2100:00)	The operator increased the nitrogen cover gas pressure on Core Flood Tanks 1A and 1B (CF-T-1A and 1B) to 600 psig.	MR(P) at PL8	4b,9K
17:24:23 (2125:00)	The operator opened Decay Heat Valve (DH-V-187) in preparation for placing the Decay Heat System in service. It was later decided not to use this mode of core cooling.	Unit 2 control room authorized the valve to be opened.	4b
17:29:23 (2130:00)	The operator started transferring the contents of the Auxiliary Building Neutralizer Tank (WDL-T-8B) pre-accident water, to Unit 1. This was done to allow water in the Auxiliary Building Sump to be transferred in this tank.	Local indication at the Rad Waste Panel	4b,4g,9h,9w
18:34:23 (2235:00)	Reactor Coolant Letdown flow was lost. It is suspected plugging of either the letdown coolers, orifices or purification filters.	MR(P) at PL3	4b,4g

PLANT STATUS

20:00:00 (0000:37)	Reactor Coolant Pump 1A (RC-P-A) was operating. Reactor coolant flow to the core had been re-established. The steam present in loops A and B had been condensed; however, a non-condensible gas space still existed in the Reactor Vessel head. The existence of the gas space was not known by the operators. Reactor Coolant temperature and pressure were stable at approximately 240F and 1165 psig with the pressurizer level at 389 inches. Decay heat was being removed by steaming Steam Generator A to the Main Condenser. Steam Generator B was isolated and was believed to have a reactor coolant side to feedwater side leak. Reactor Coolant Makeup Pump 1B (MU-P-1B) was operating supplying		

Item	Event	Information Available to the Operator	Reference
	Reactor Coolant Pump Seal Injection Flow. Reactor Coolant Letdown had been lost and an attempt to regain it was in progress. The Reactor Building had been isolated except for essential services and periodic sampling. Airborne radioactive materials were present in the Auxiliary and Fuel Handling Buildings. These radioactive materials were being released (through charcoal filters and absolute filters) to the environment via the station vent by their buildings ventilation system exhausts. To provide for storage of water present in the Auxiliary Building Sump, pre-accident water in the Auxiliary Building Neutralizer Tank (WDL-T-8B) was being transferred to Unit 1.		

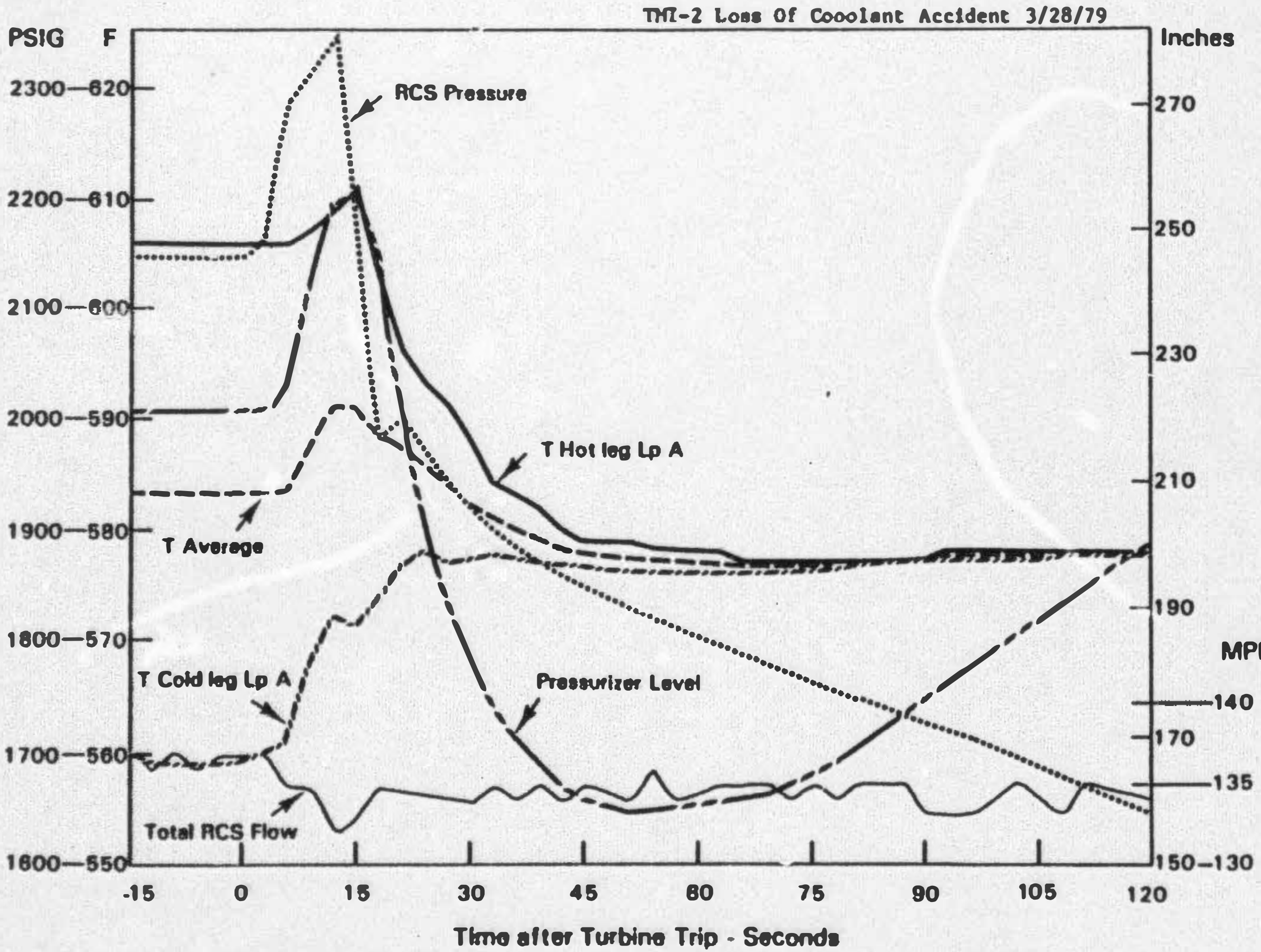


Figure 1 Summary of Reactor Coolant System Parameters

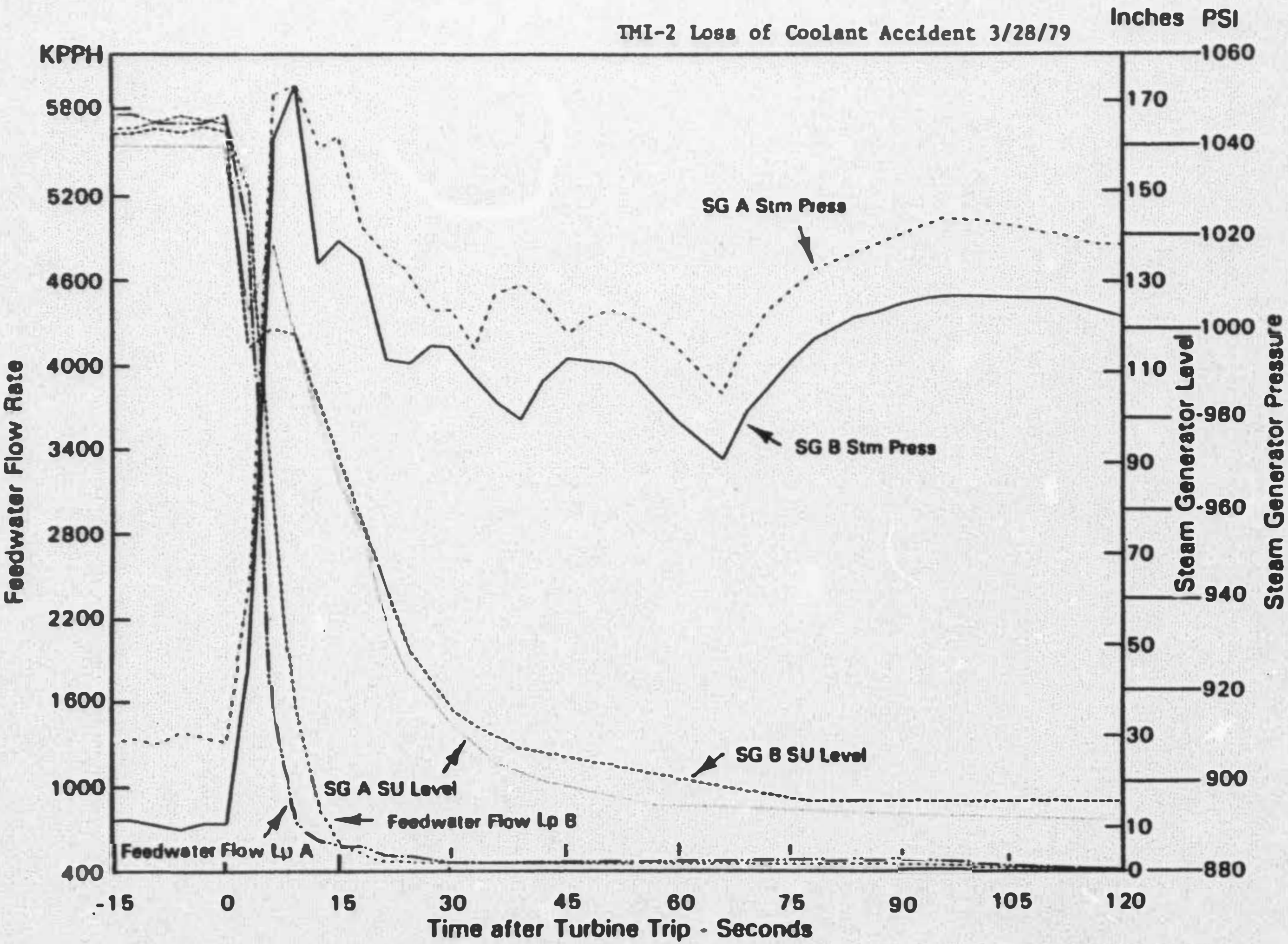


Figure 2 Summary of Steam Generator Parameters

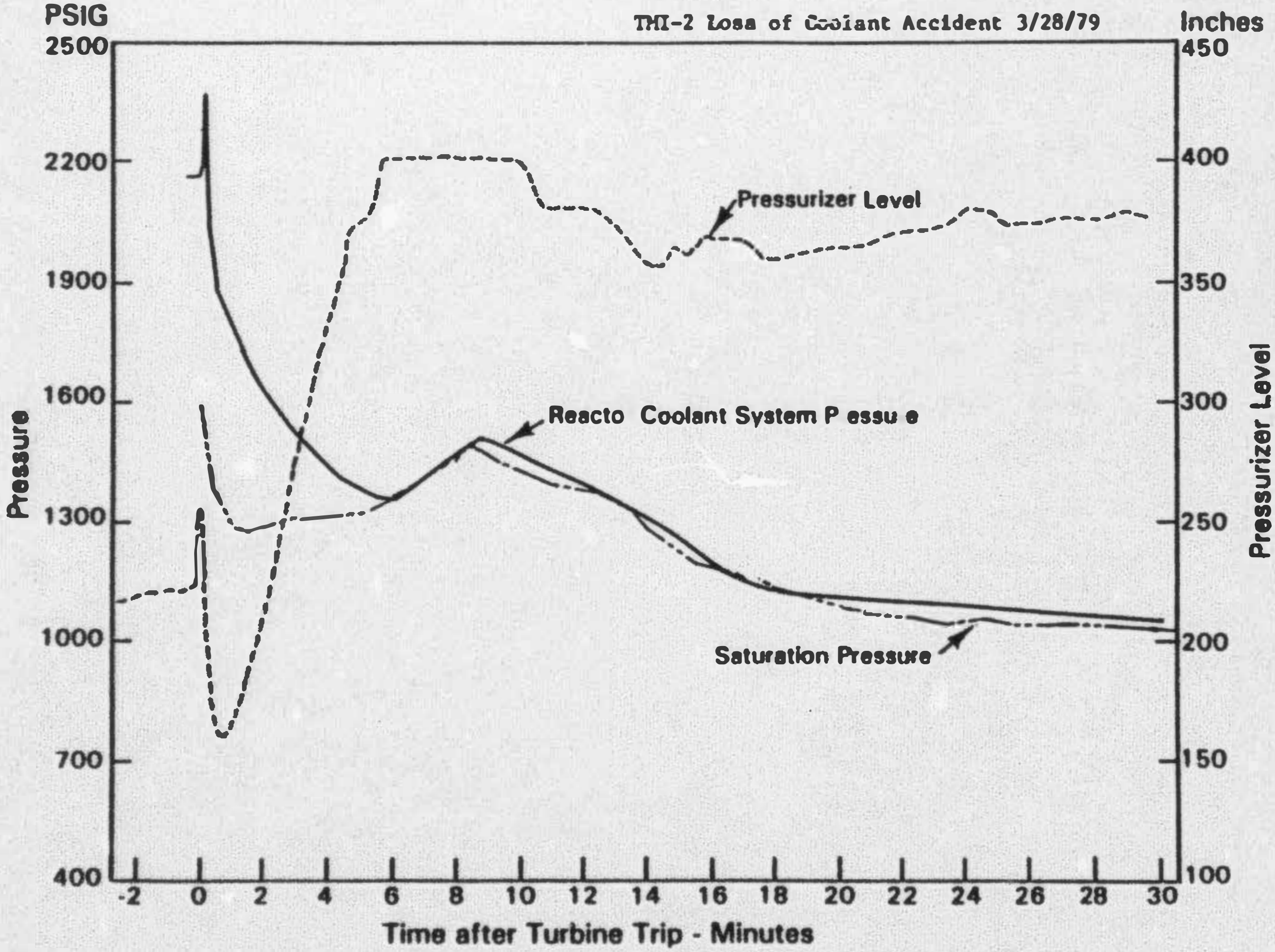


Figure 3 Reactor Coolant System Pressure, Saturation Pressure and Pressurizer Level

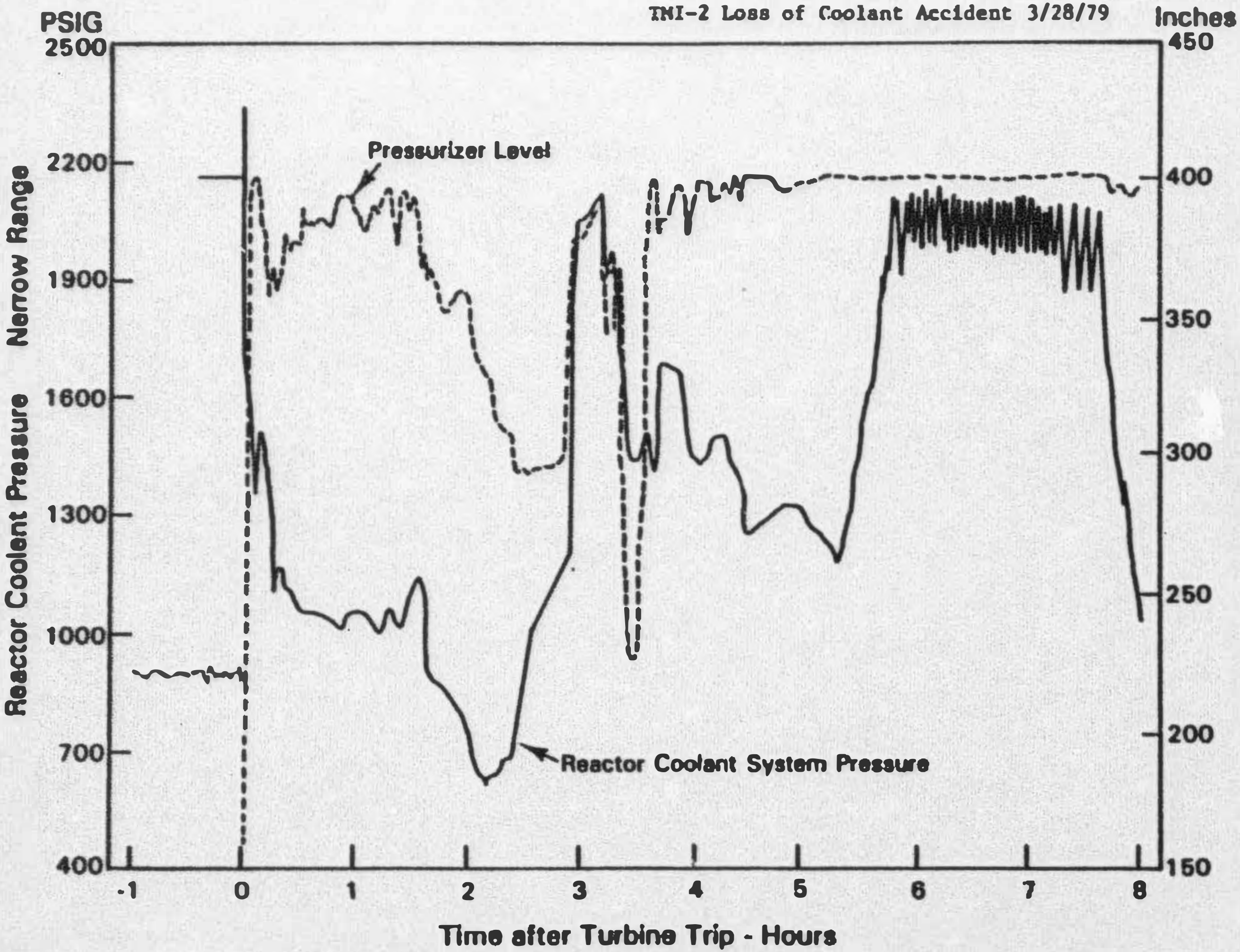


Figure 4 Reactor Coolant System Pressure and Pressurizer Level

THI-2 Loss of Coolant Accident 3/28/79

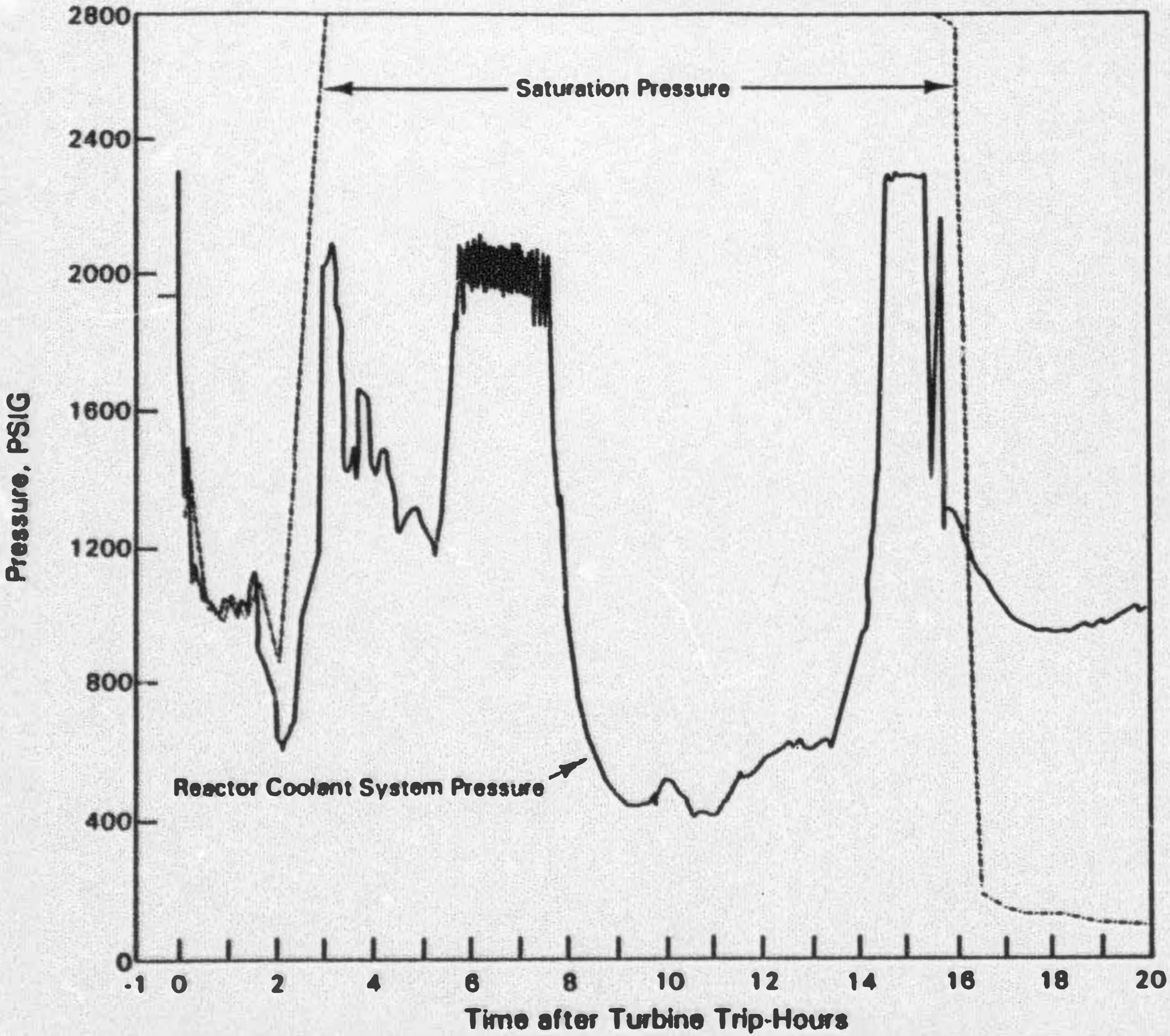


Figure 5 Reactor Coolant System Pressure and Saturation Pressure

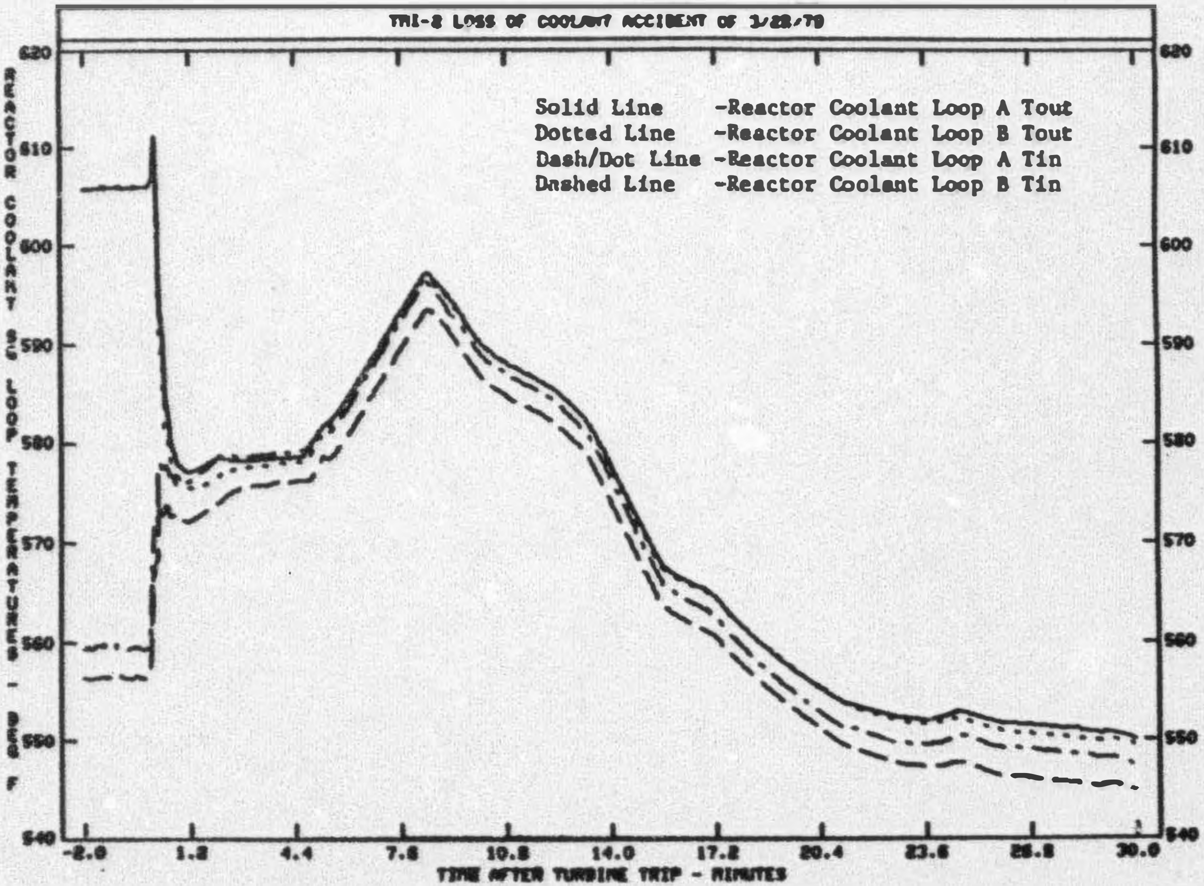


Figure 6 RCS Loop Temperature Vs Time After Turbine Trip

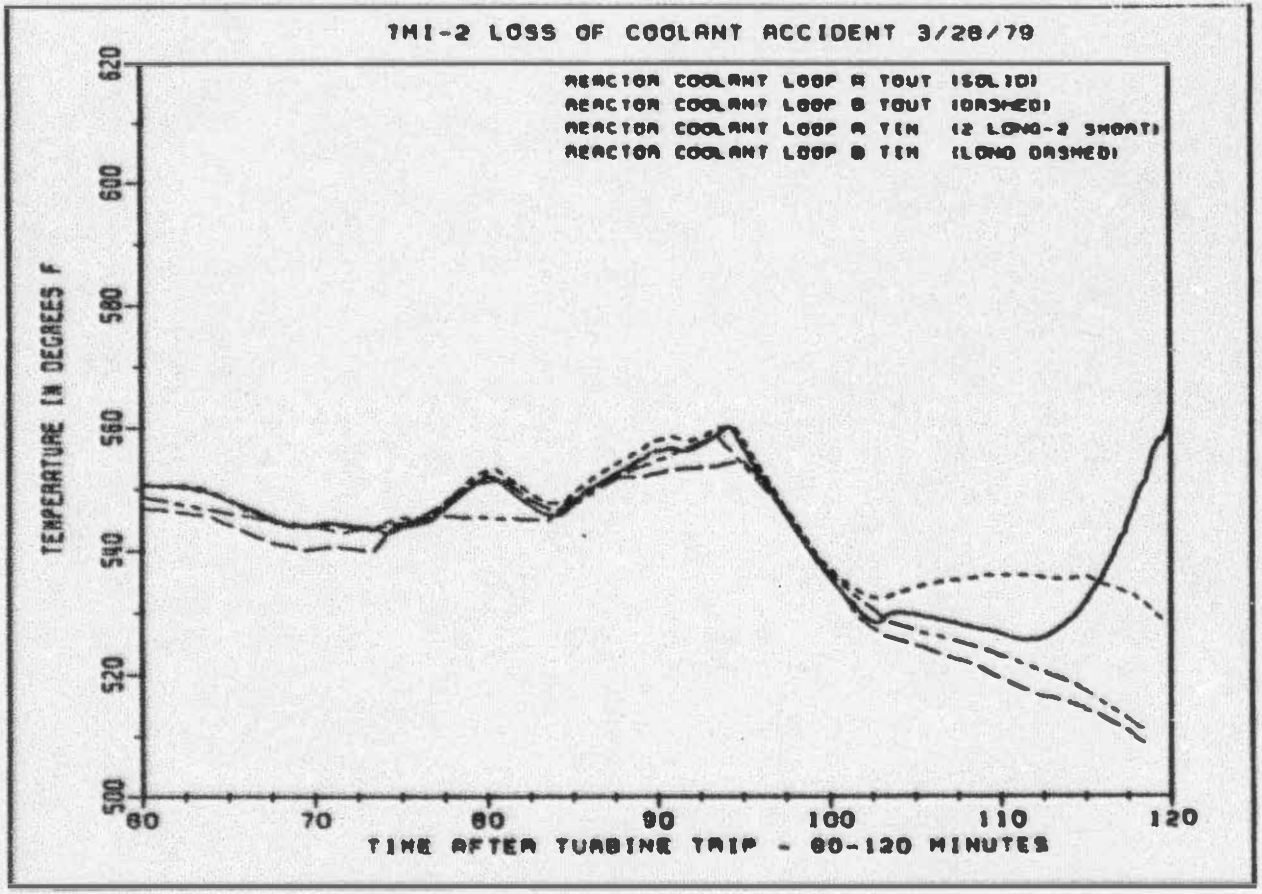


Figure 7 RCS Loop Temperature Vs Time after Turbine Trip

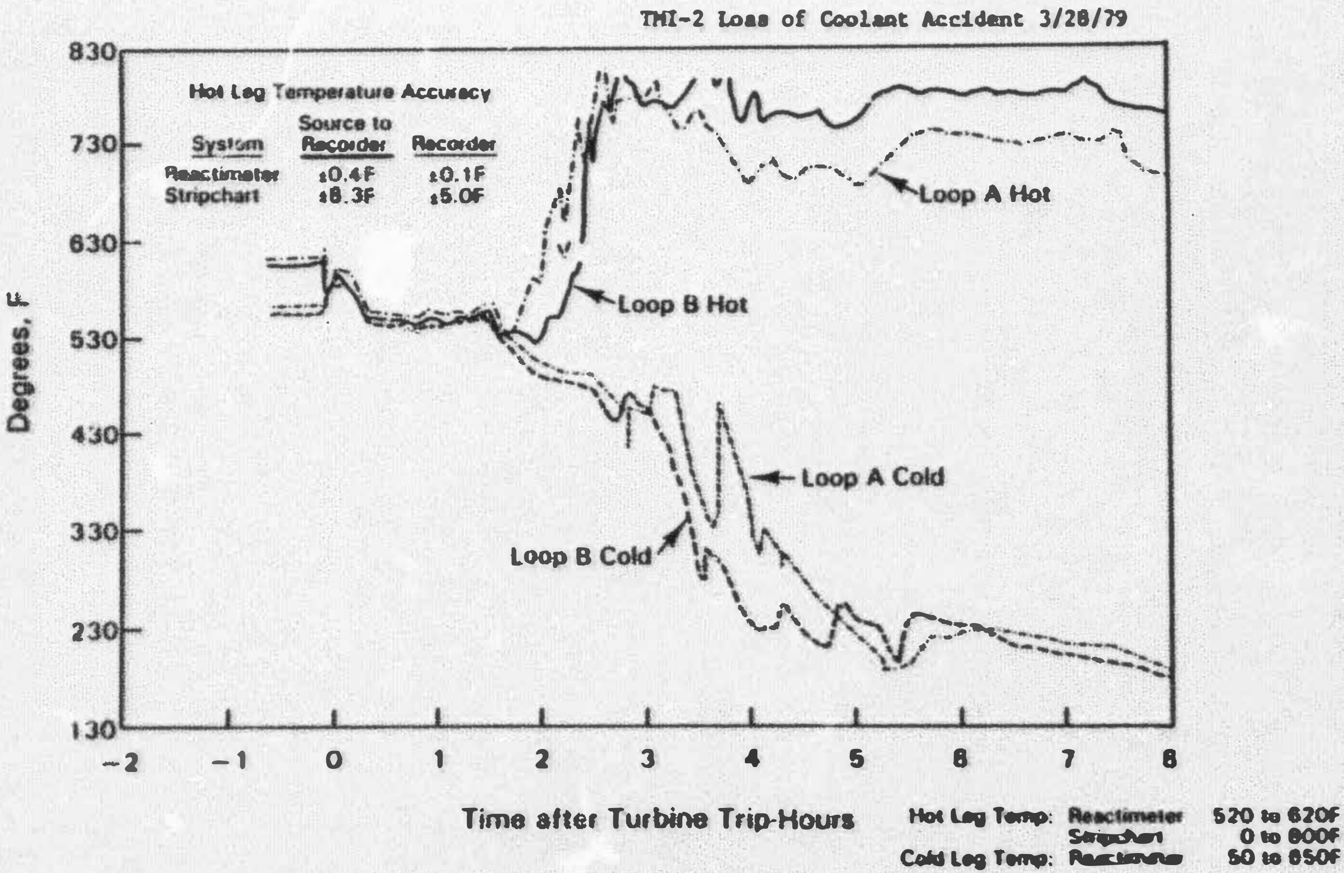


Figure 6 RCS Loop A and B Cold and Hot Leg Temperatures

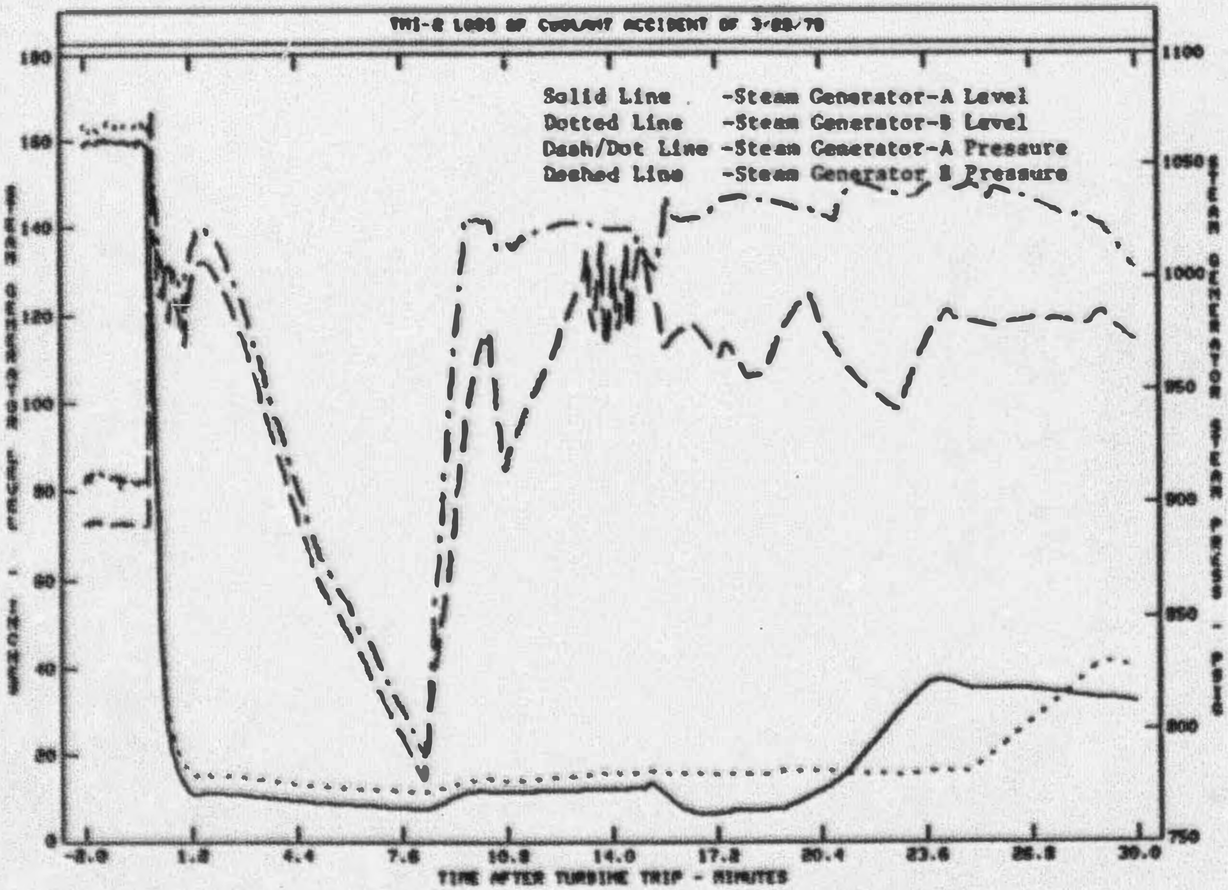


Figure 9 Steam Generator Level and Pressure Vs Time After Turbine Trip

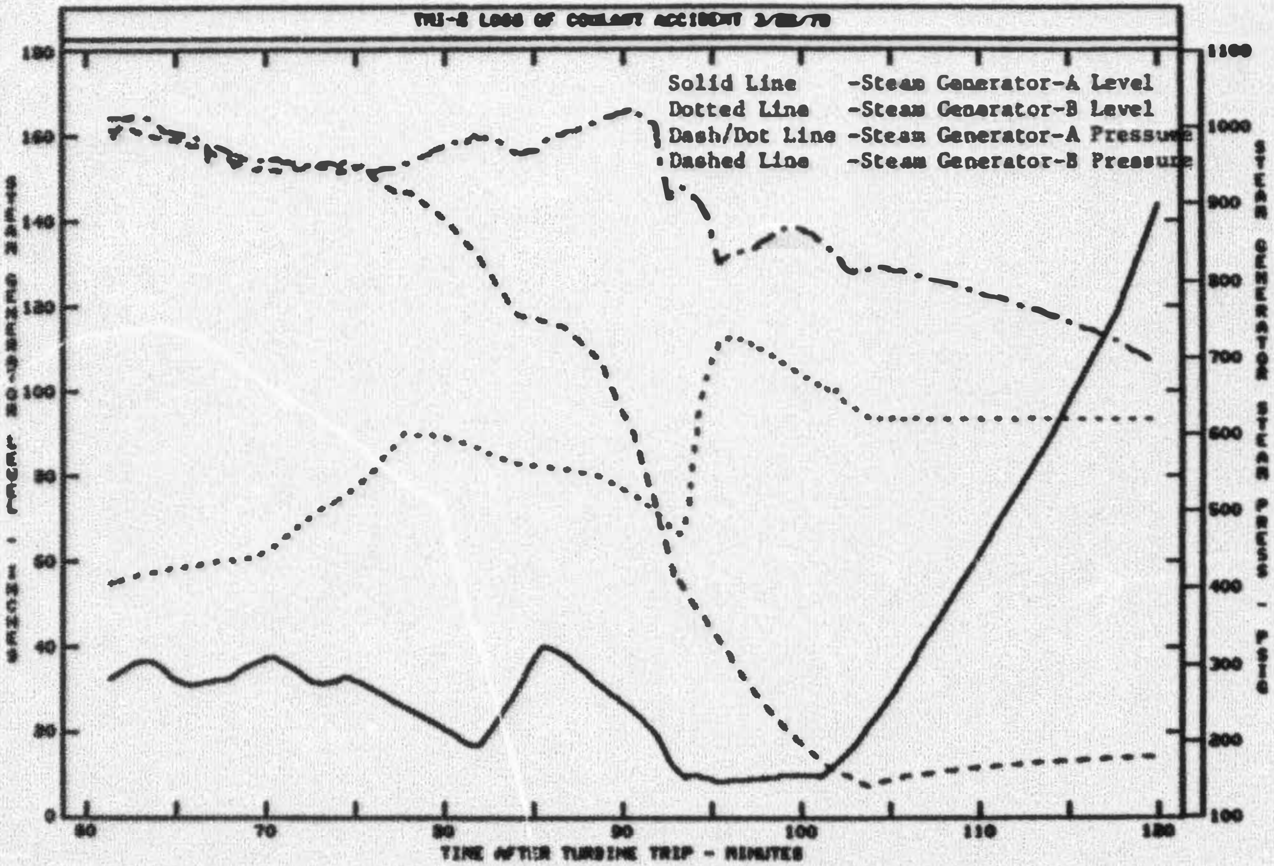


Figure 10 Steam Generator Level and Pressure Vs Time after Turbine Trip

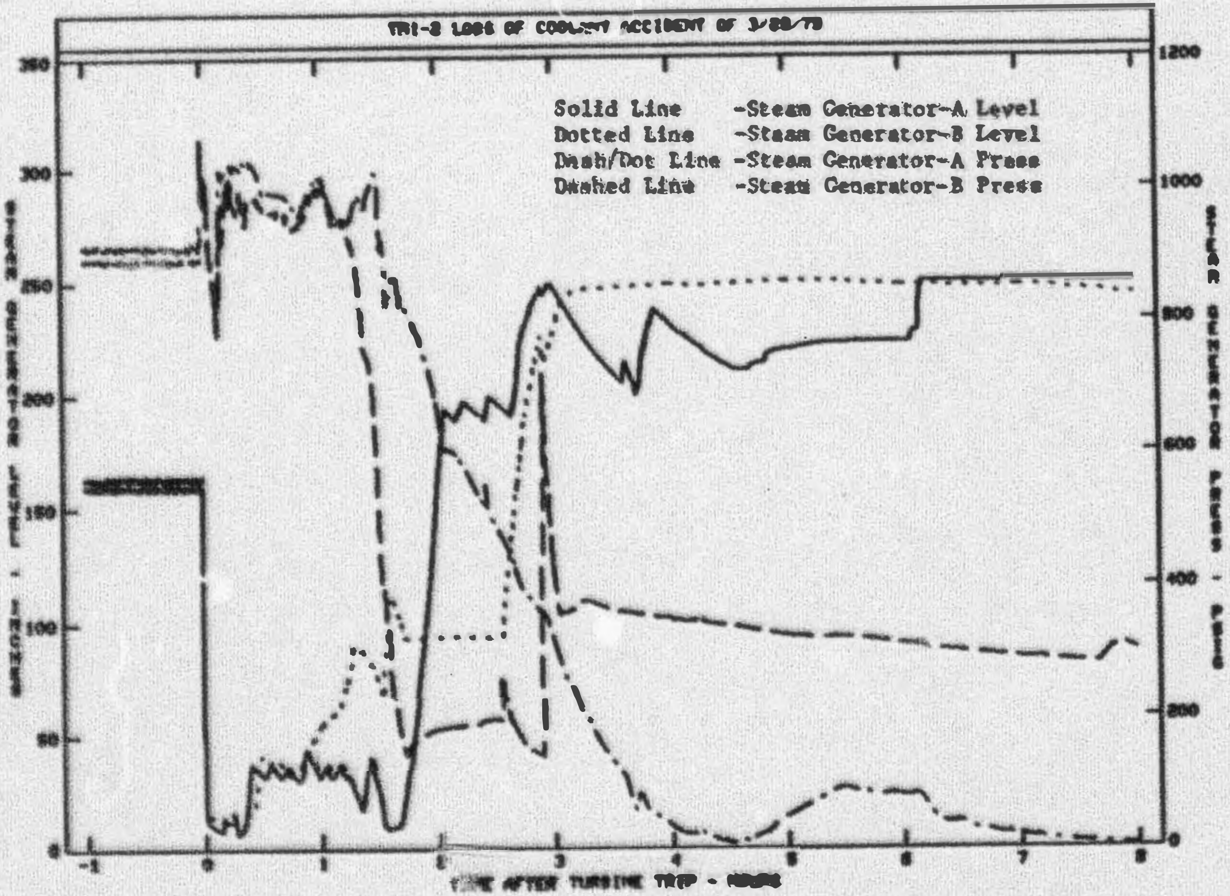


Figure 11 Steam Generator Level and Pressure Vs Time after Turbine Trip

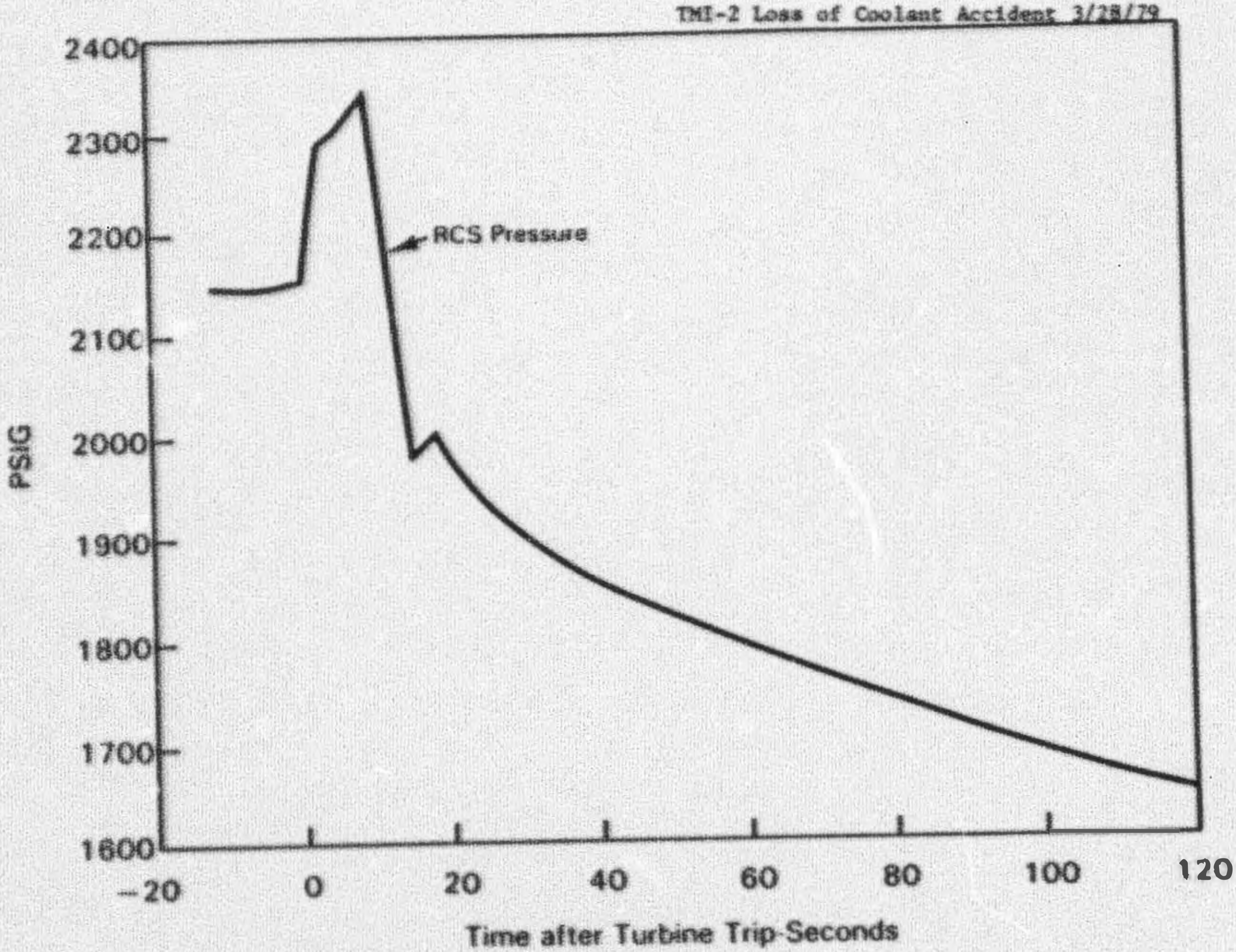


Figure 12 Reactor Coolant System Pressure

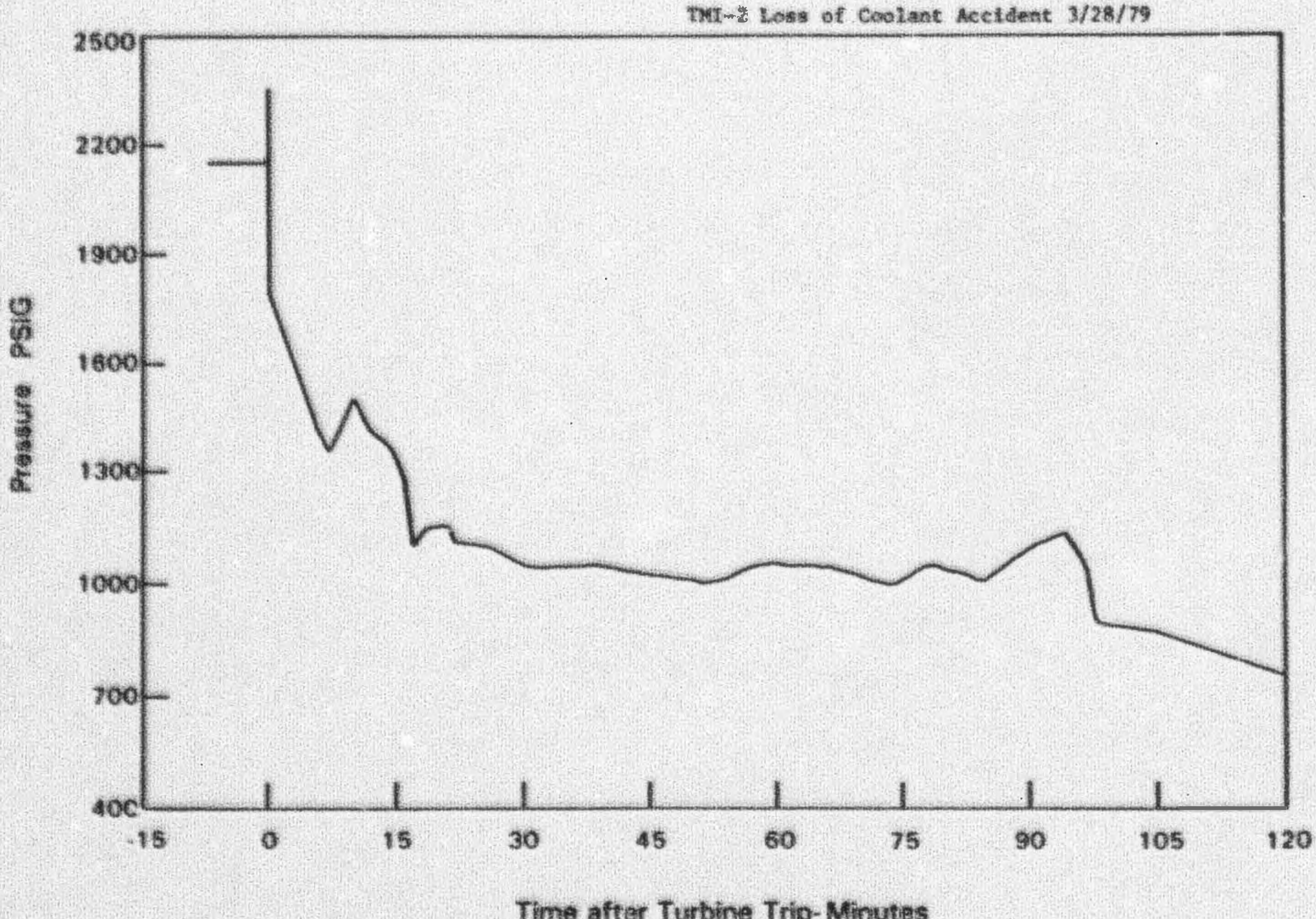


Figure 13 Reactor Coolant System Pressure

TMI-2 Loss of Coolant Accident 3/28/79

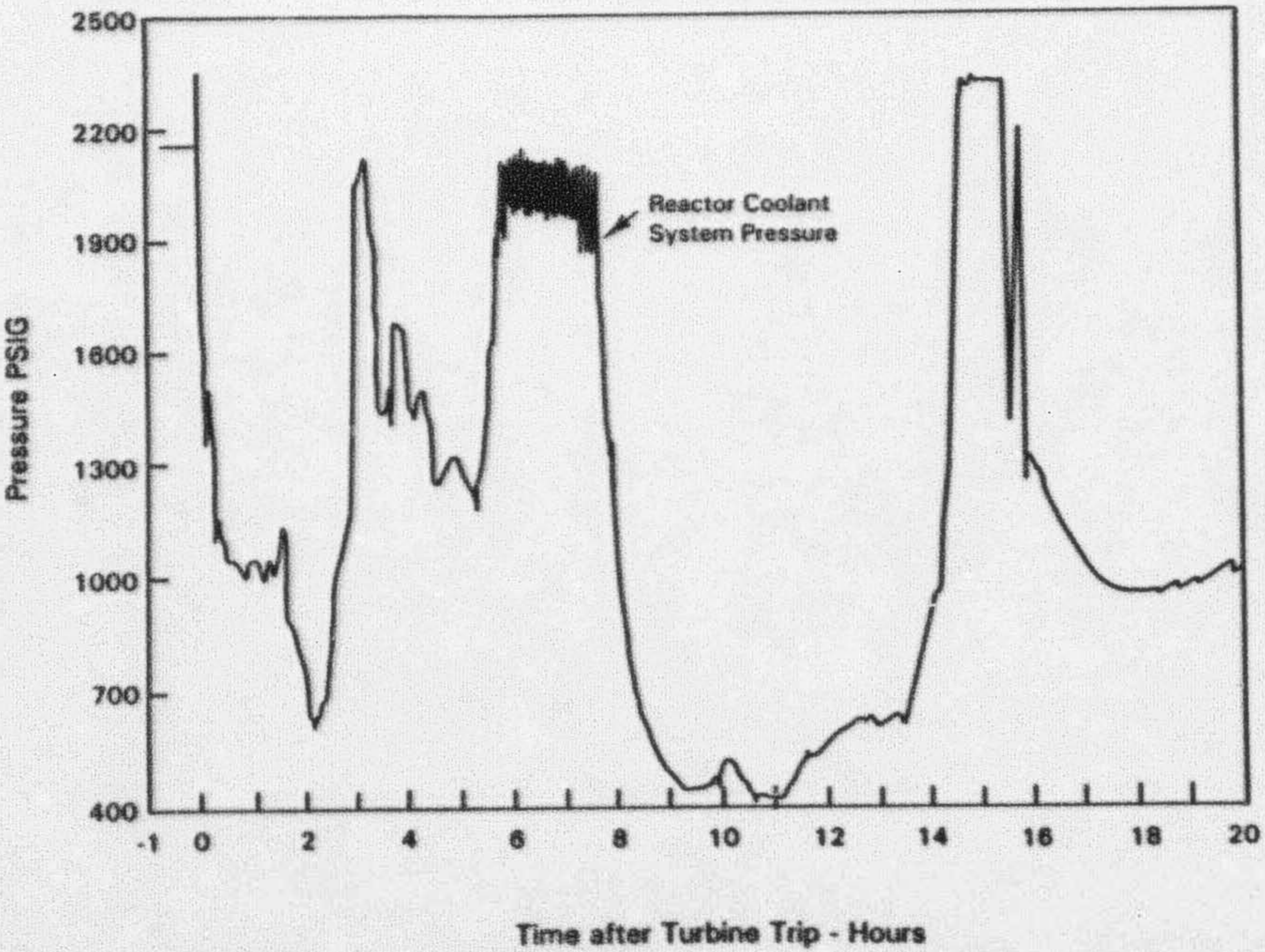


Figure 14 Reactor Coolant System Pressure

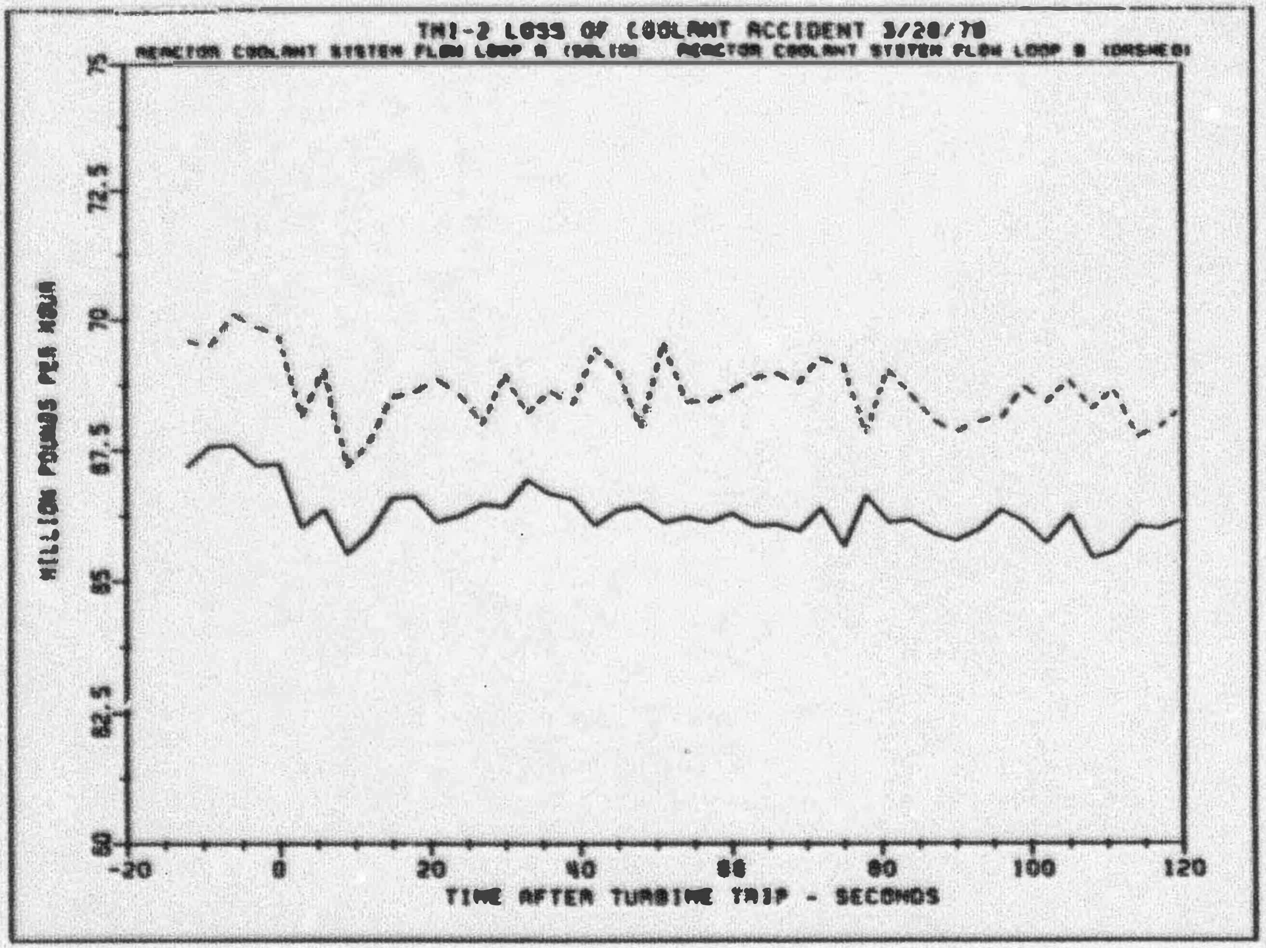


Figure 15 RCS Flow Rate Vs Time after Turbine Trip

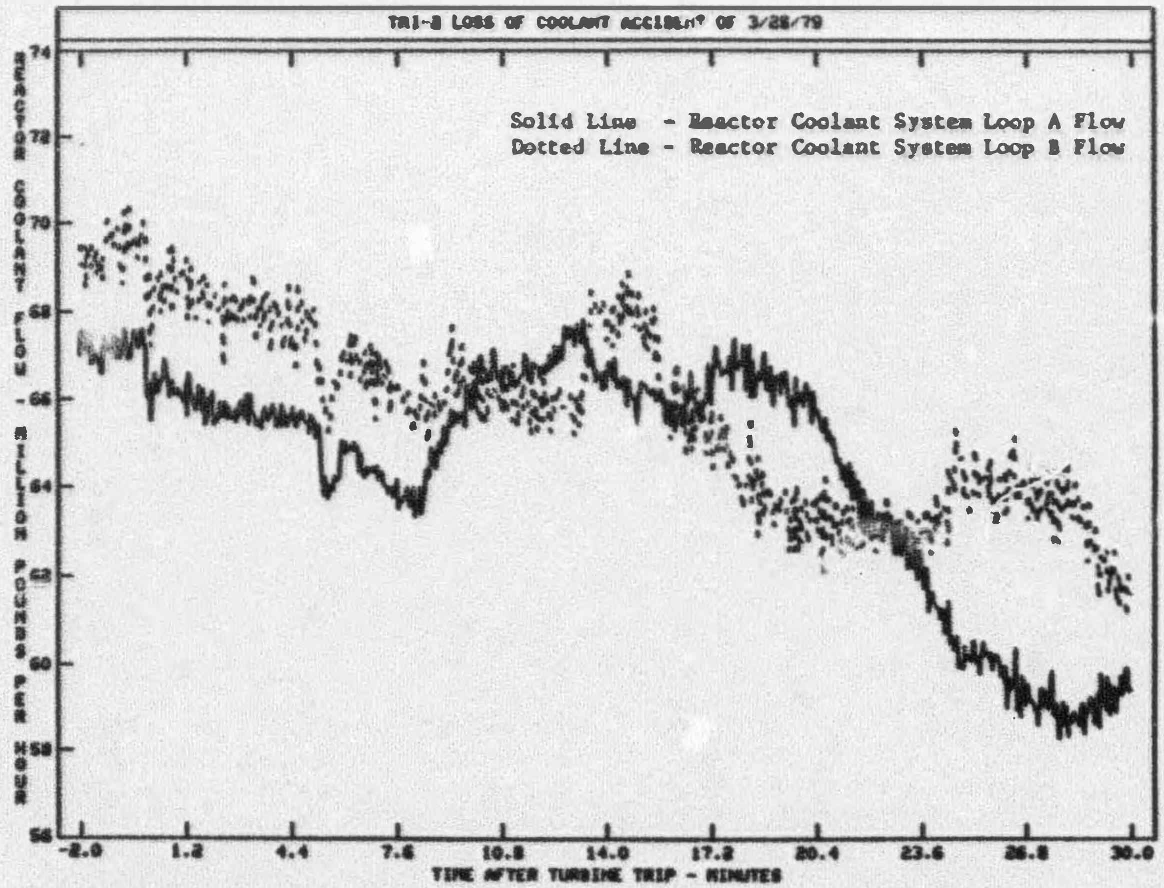


Figure 16 RCS Flow Rate Vs Time after Turbine Trip

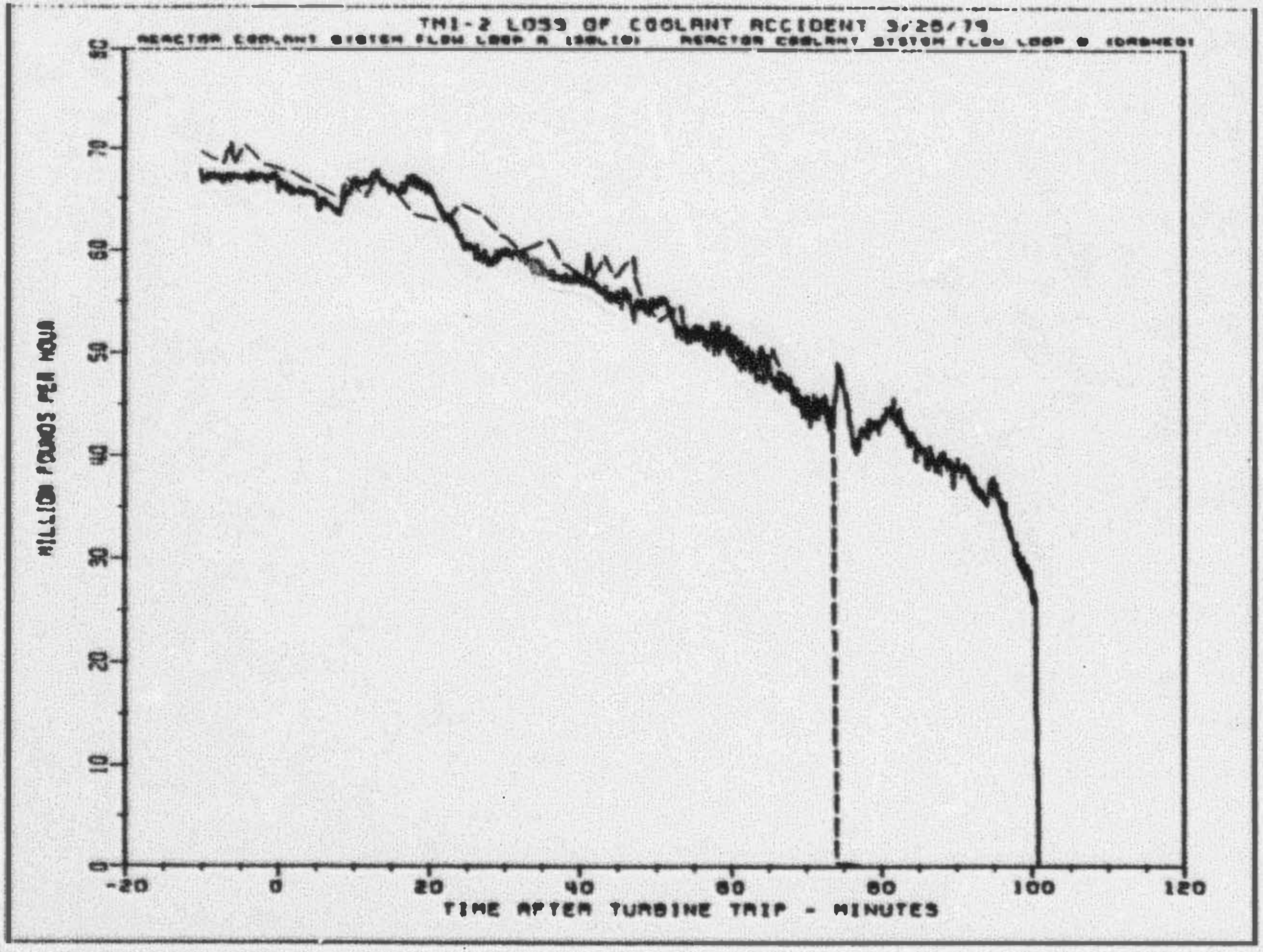


Figure 17 RCS Flow Rate Vs Time after Turbine Trip

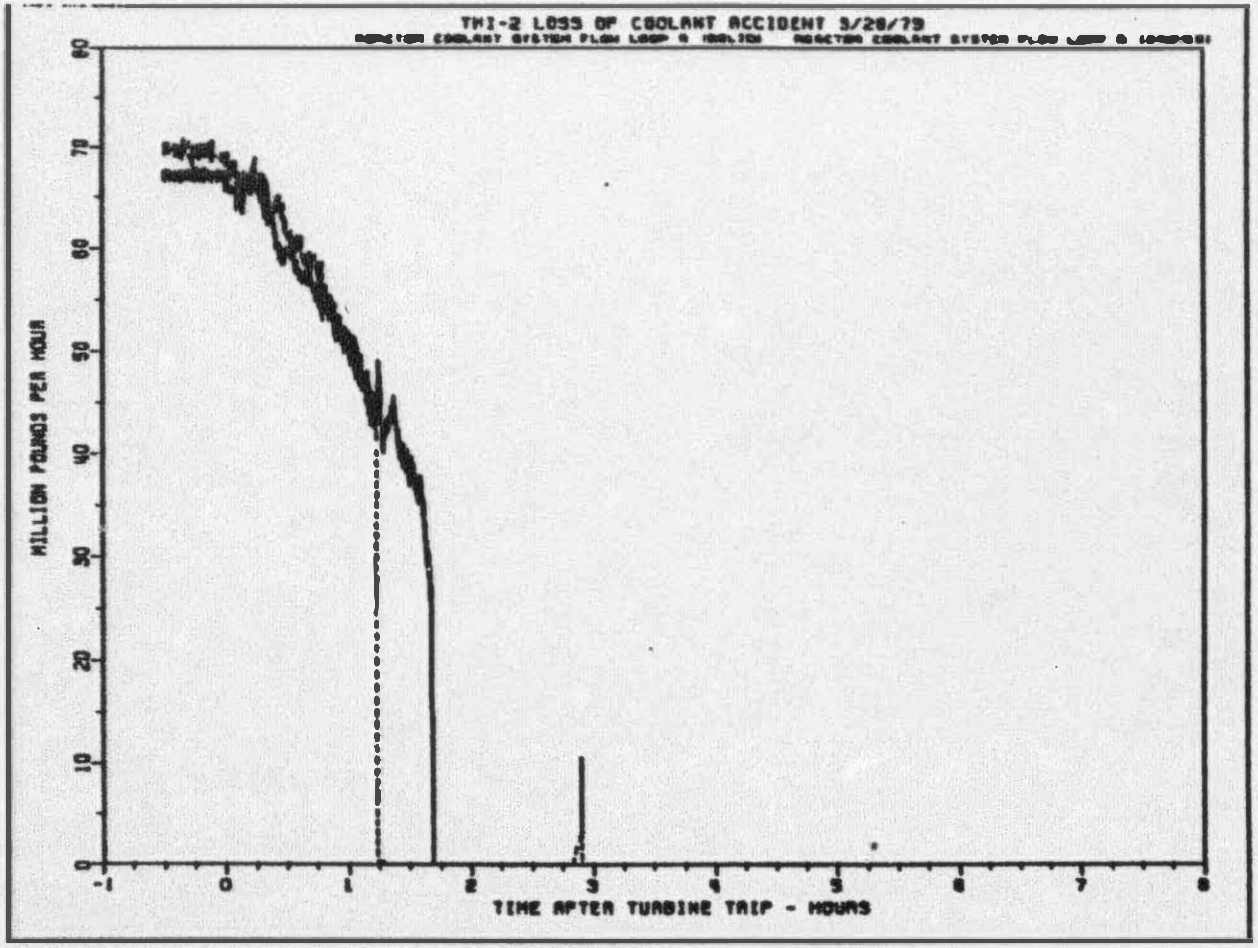


Figure 18 RCS Flow Rate Vs Time after Turbine Trip

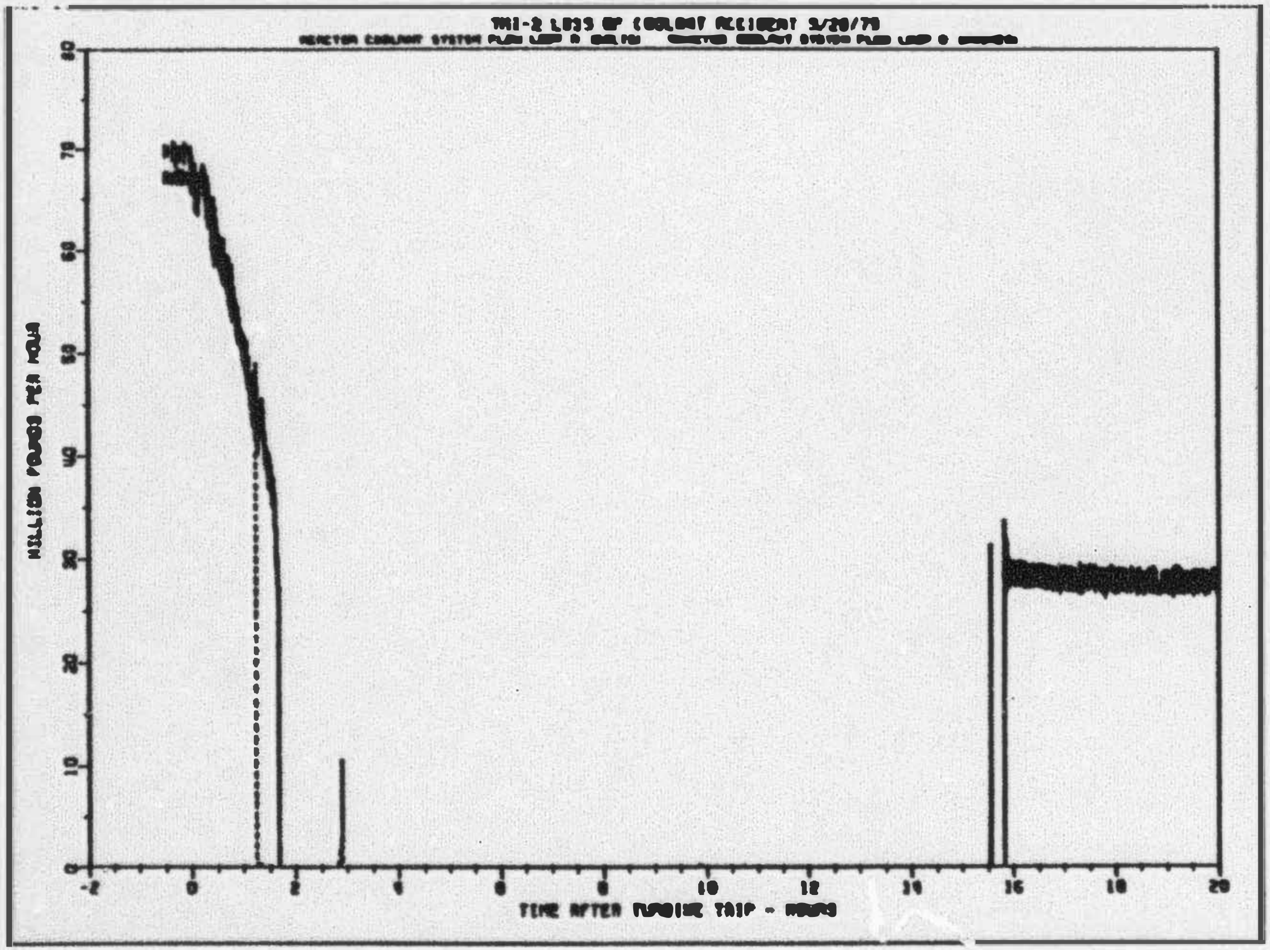


Figure 19 RCS Flow Rate Vs Time after Turbine Trip

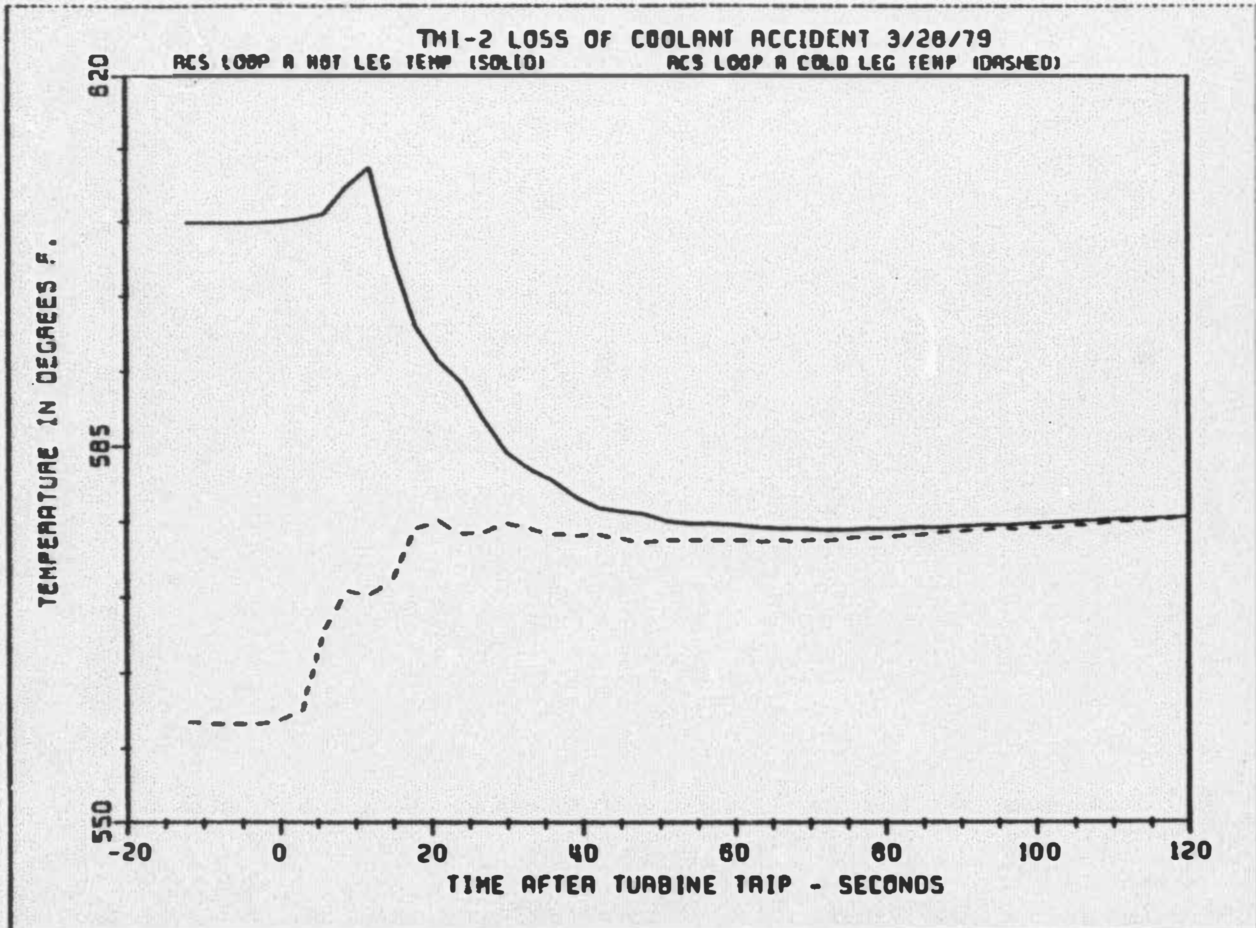


Figure 20 RCS Loop A Temperature Vs Time after Turbine Trip

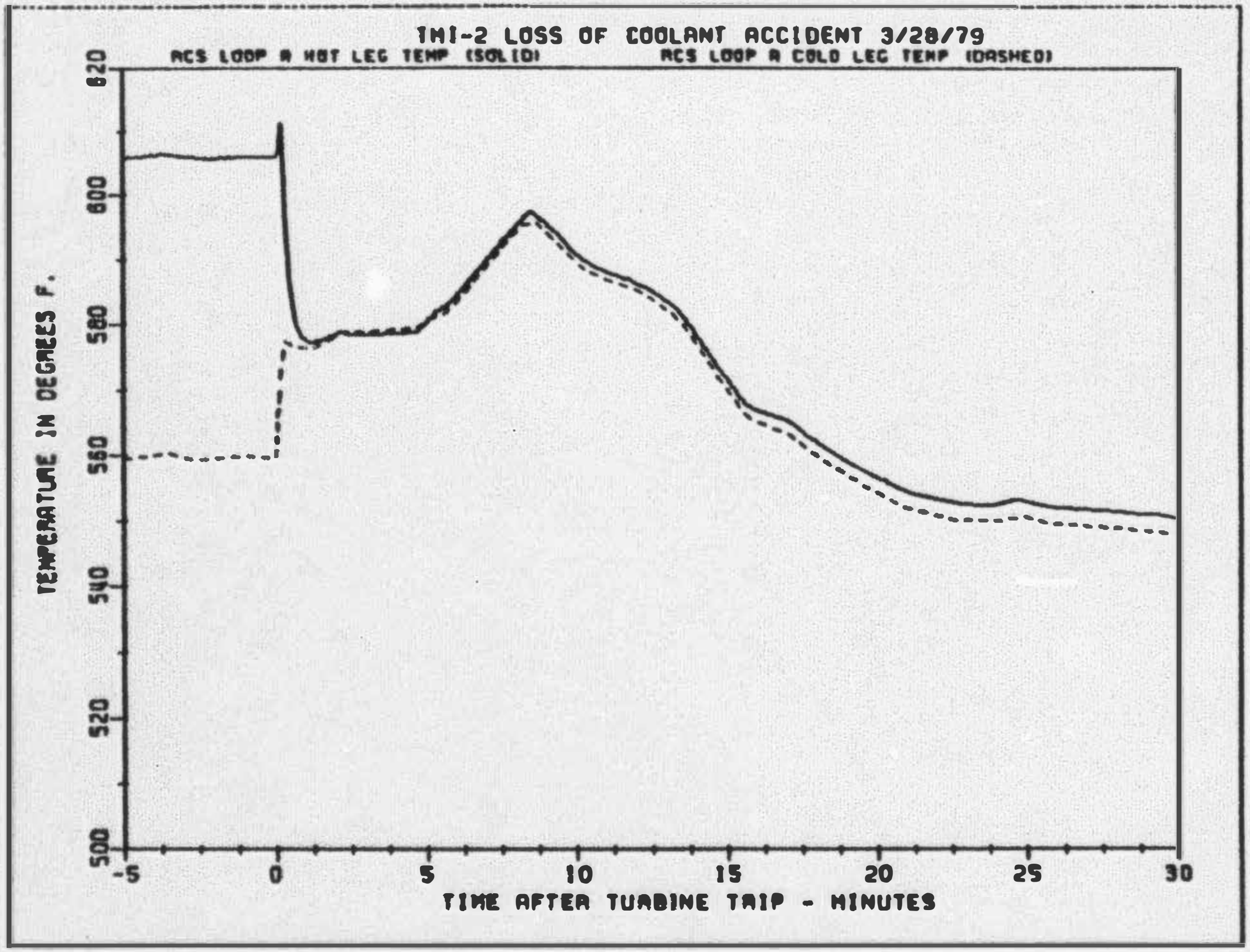


Figure 21 RCS Loop A Temperature Vs Time after Turbine Trip

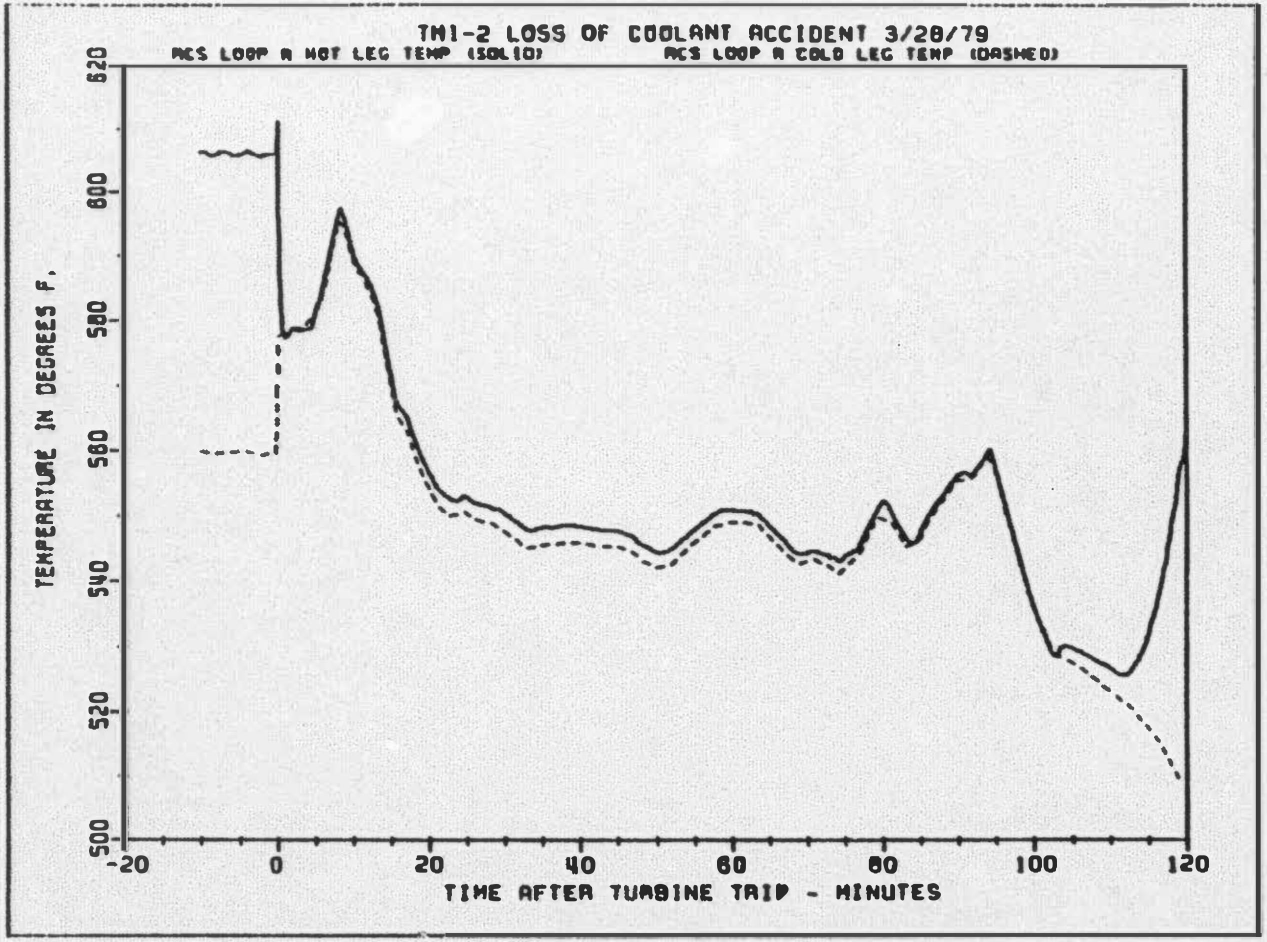


Figure 22 RCS Loop A Temperature Vs Time after Turbine Trip

TMI-2 Loss of Coolant Accident 3/28/79

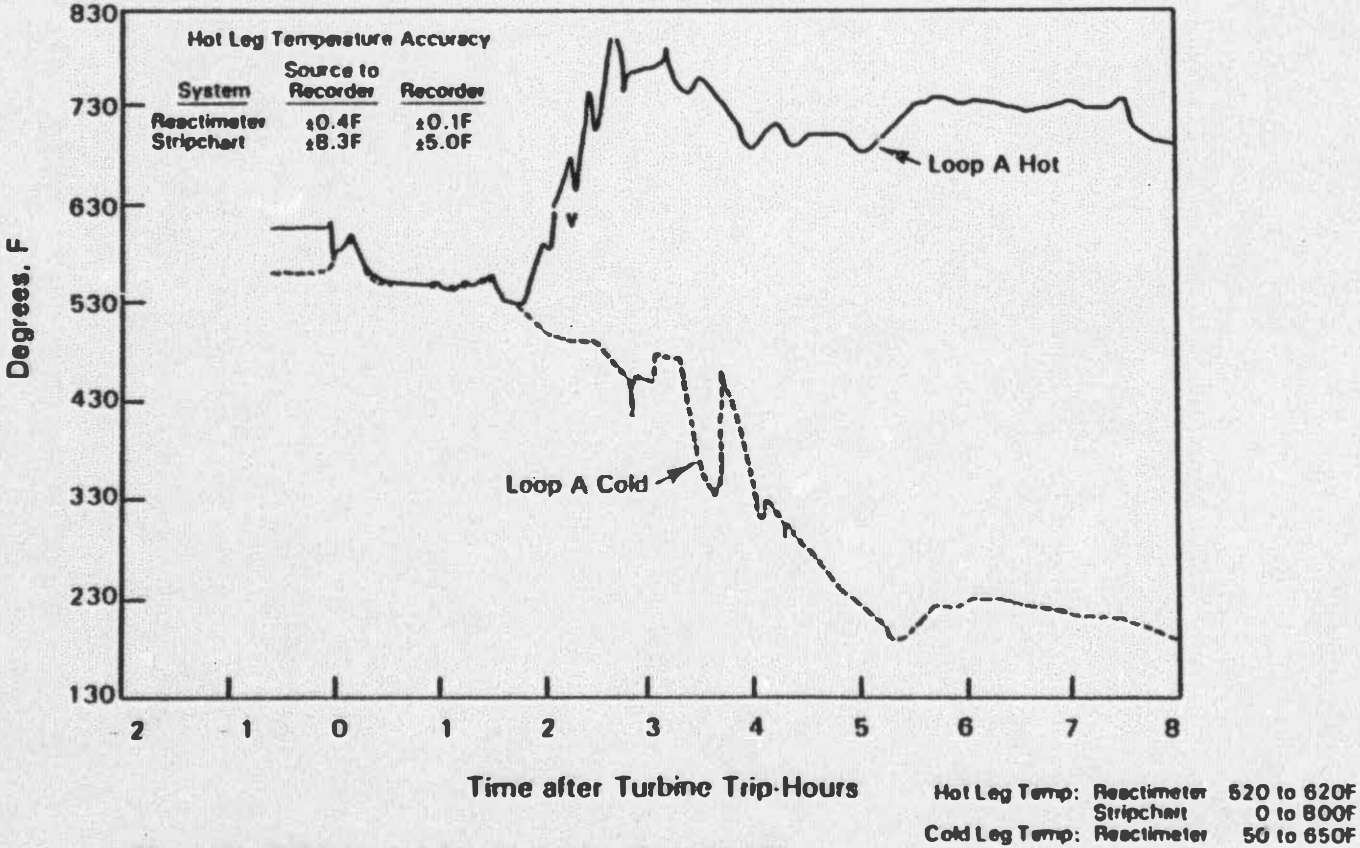


Figure 23 RCS Loop A Cold and Hot Leg Temperatures

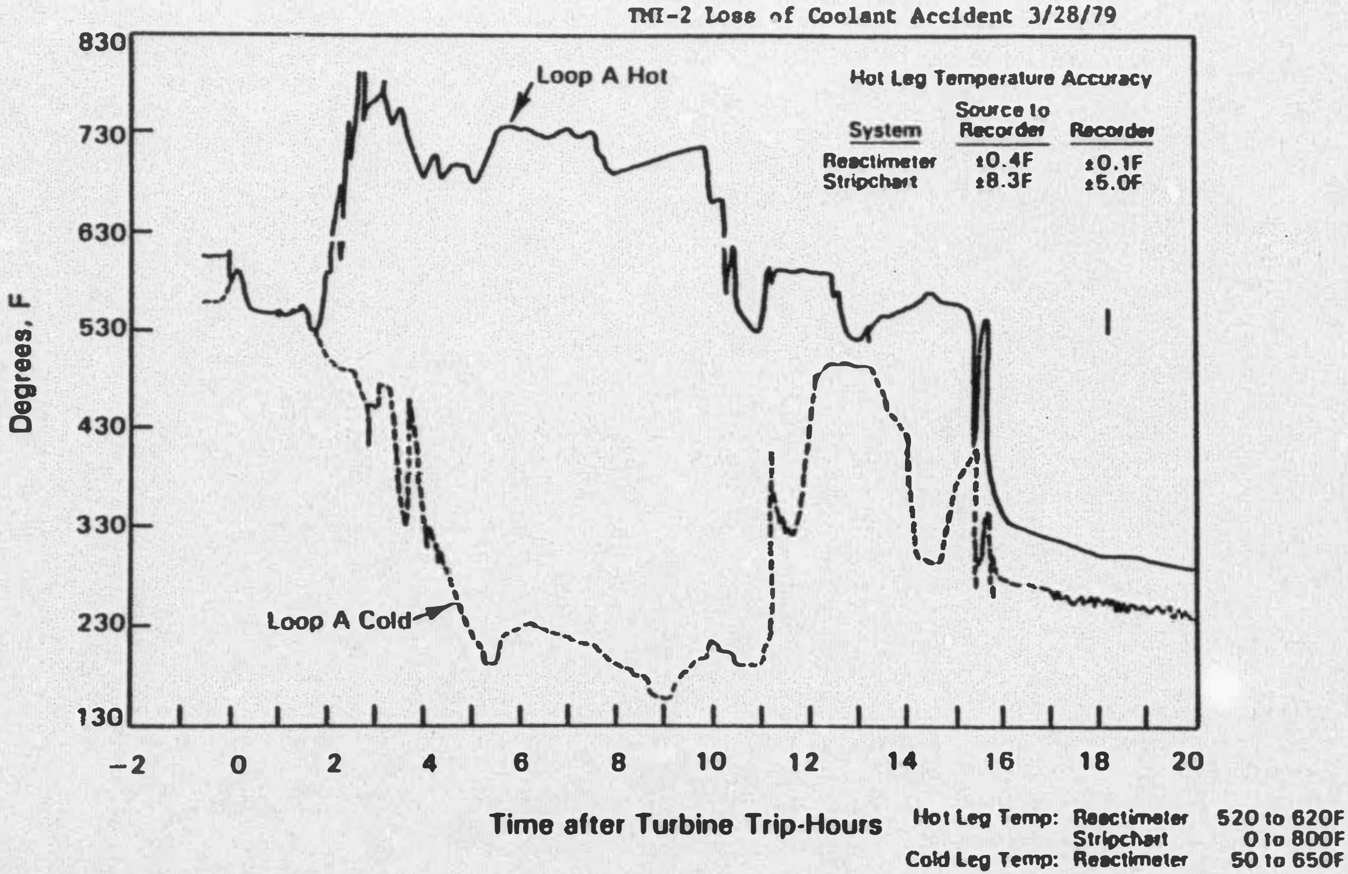


Figure 24 RCS Loop A Cold and Hot Leg Temperatures

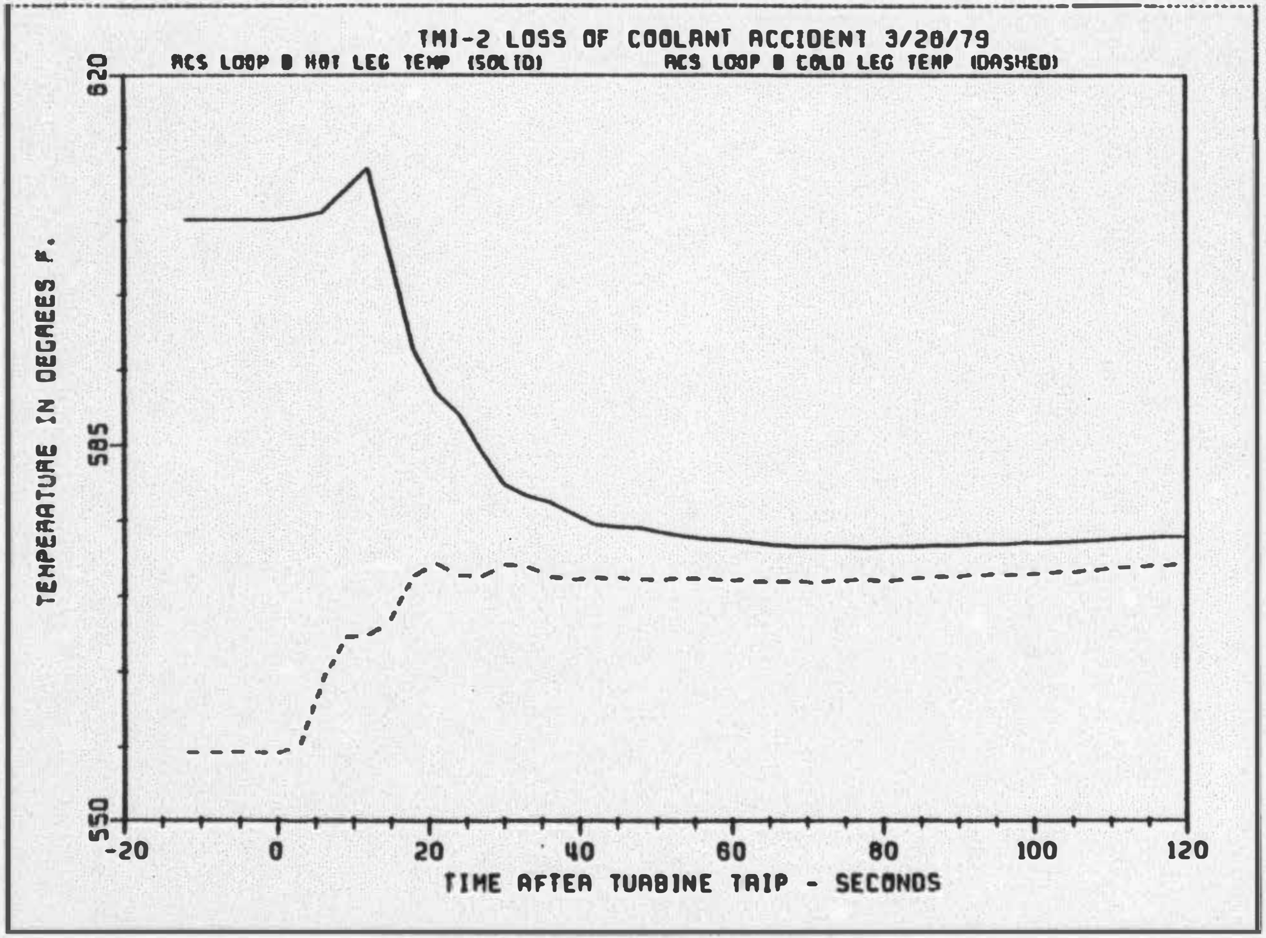


Figure 23 RCS Loop B Temperature Vs Time after Turbine Trip

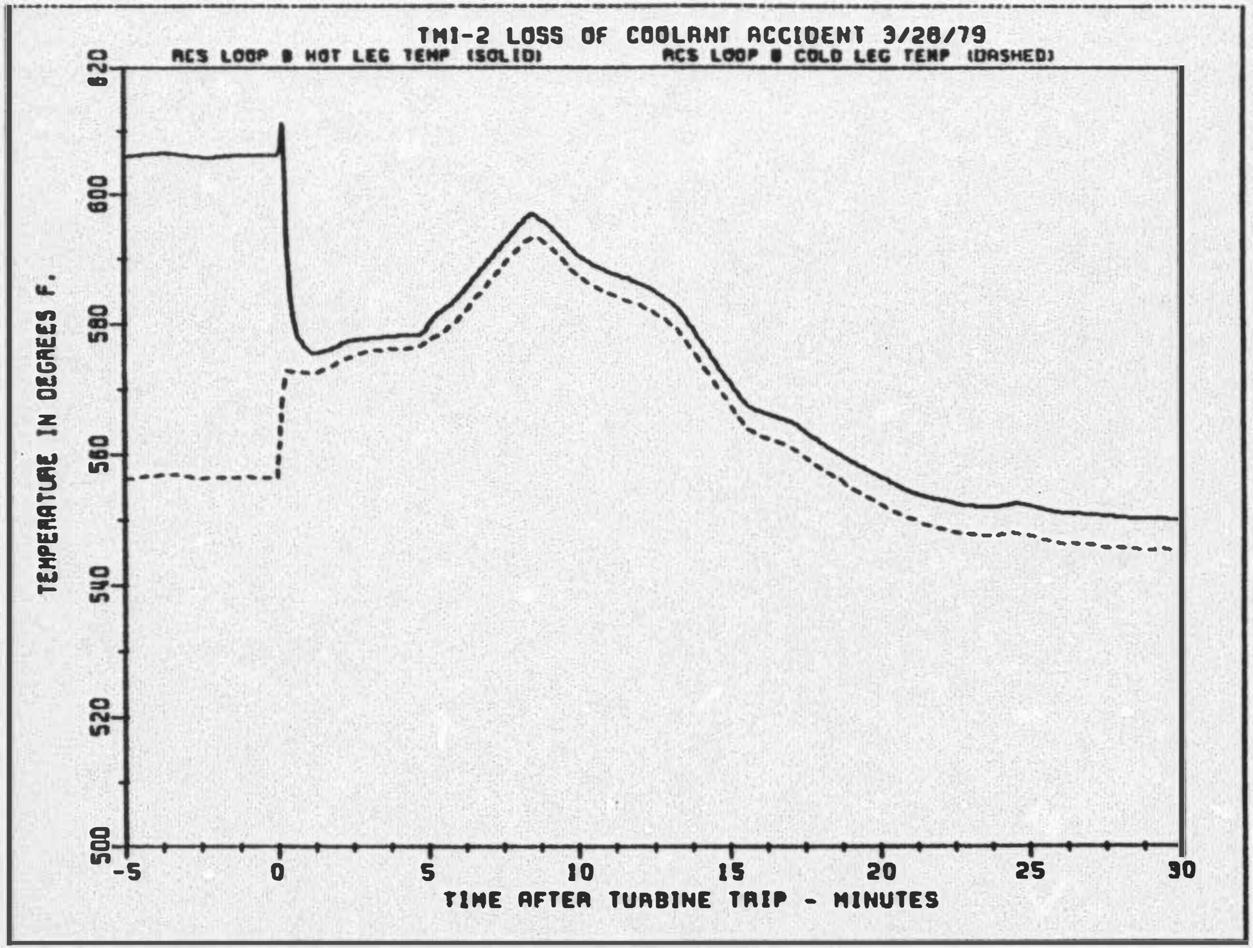


Figure 26 RCS Loop B Temperature Vs Time after Turbine Trip

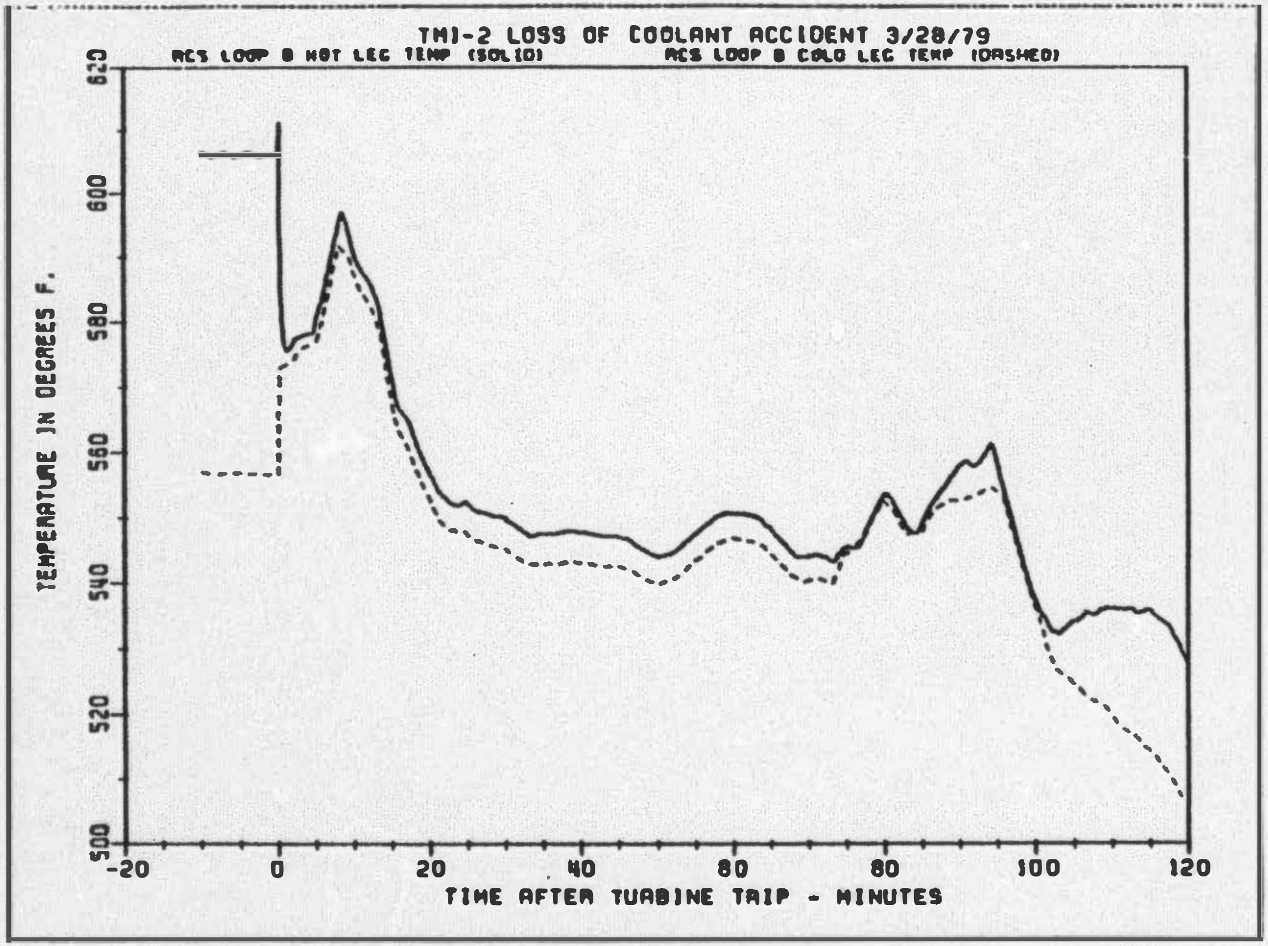


Figure 27 RCS Loop B Temperature Vs Time after Turbine Trip

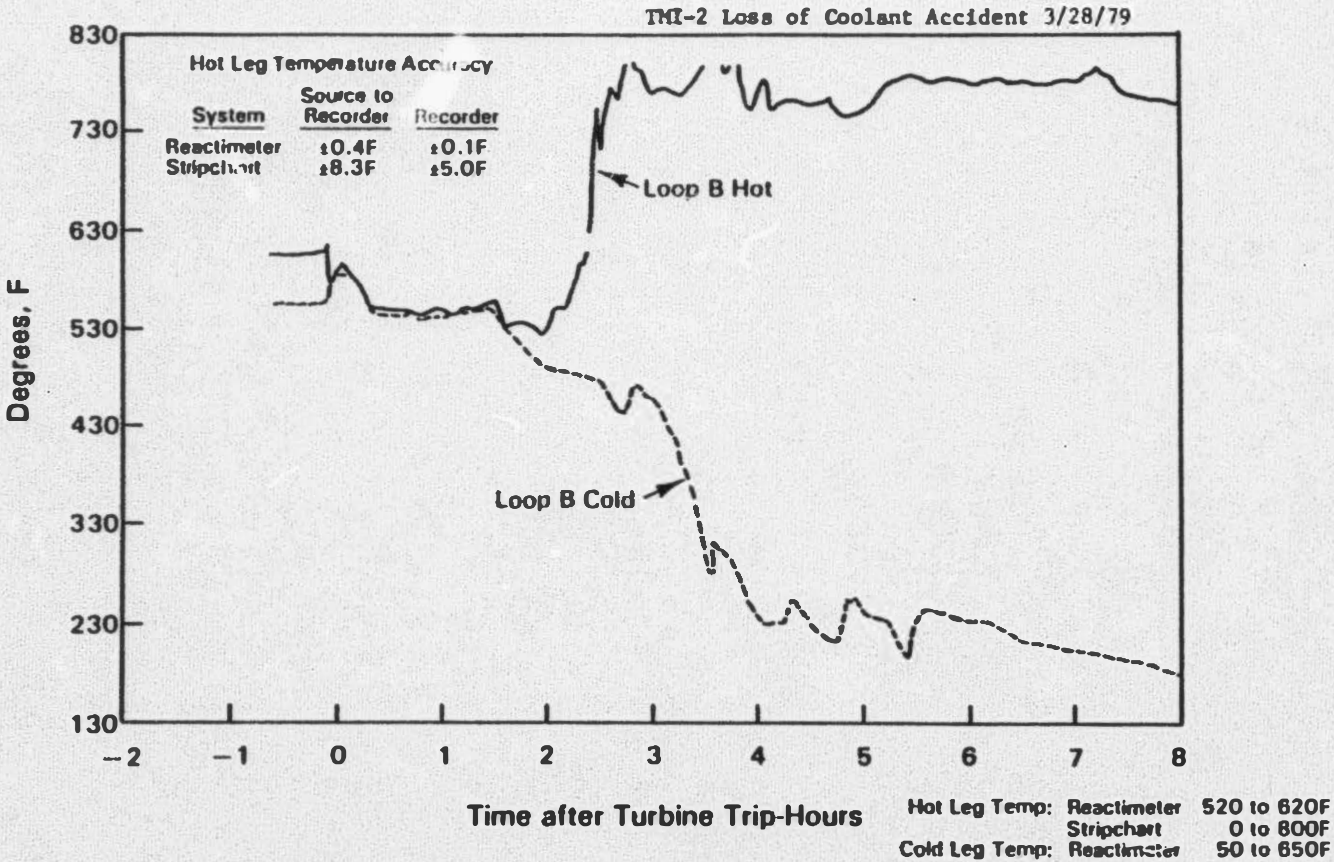


Figure 28 RCS Loop B Cold and Hot Leg Temperatures

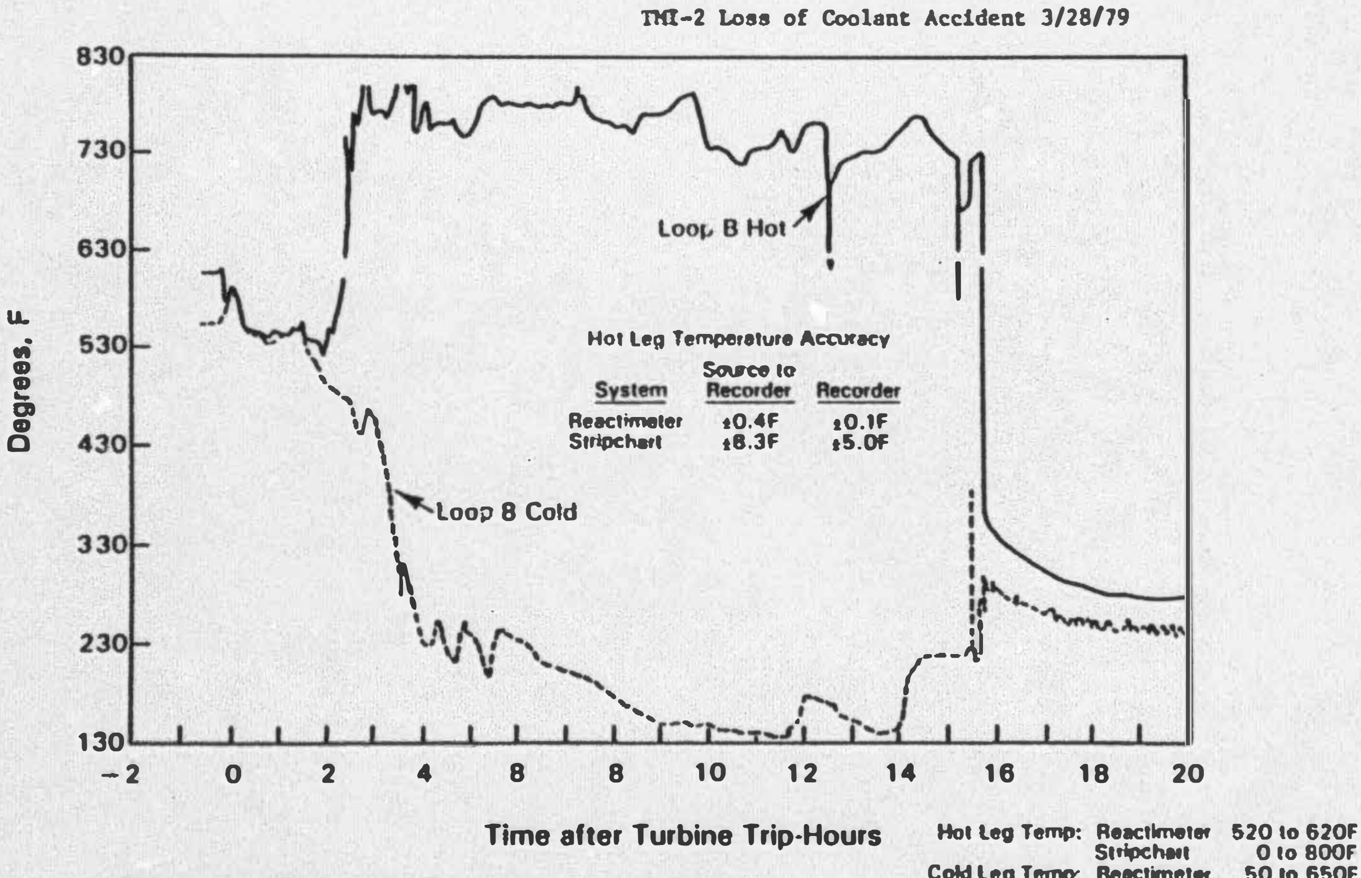


Figure 29 RCS Loop B Cold and Hot Leg Temperatures

TMI-2 Loss of Coolant Accident 3/28/79

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A															
B							281	317							
C						469			957	315					
D					1196					500				217	
E				599			1926		2582		326				
F			86				2367	2377				326	554		
G		348			2272	1774			1806		1875		234		
H	260				2452			1295	2176				1852		
K					1181						632	1760			
L		372	1566			382					296		1774		
M			323				2171		2327	349				252	
N				415				579	2167						
O					356	372				1137		310			
P						292									
R							352			476					

Note: These values do not include connections for the reference junction which was reading approximately 75F.

Figure 30 Reactor Coolant System Exist Fuel Assembly Temperature at Approximately 04:59:23 (0900:00)

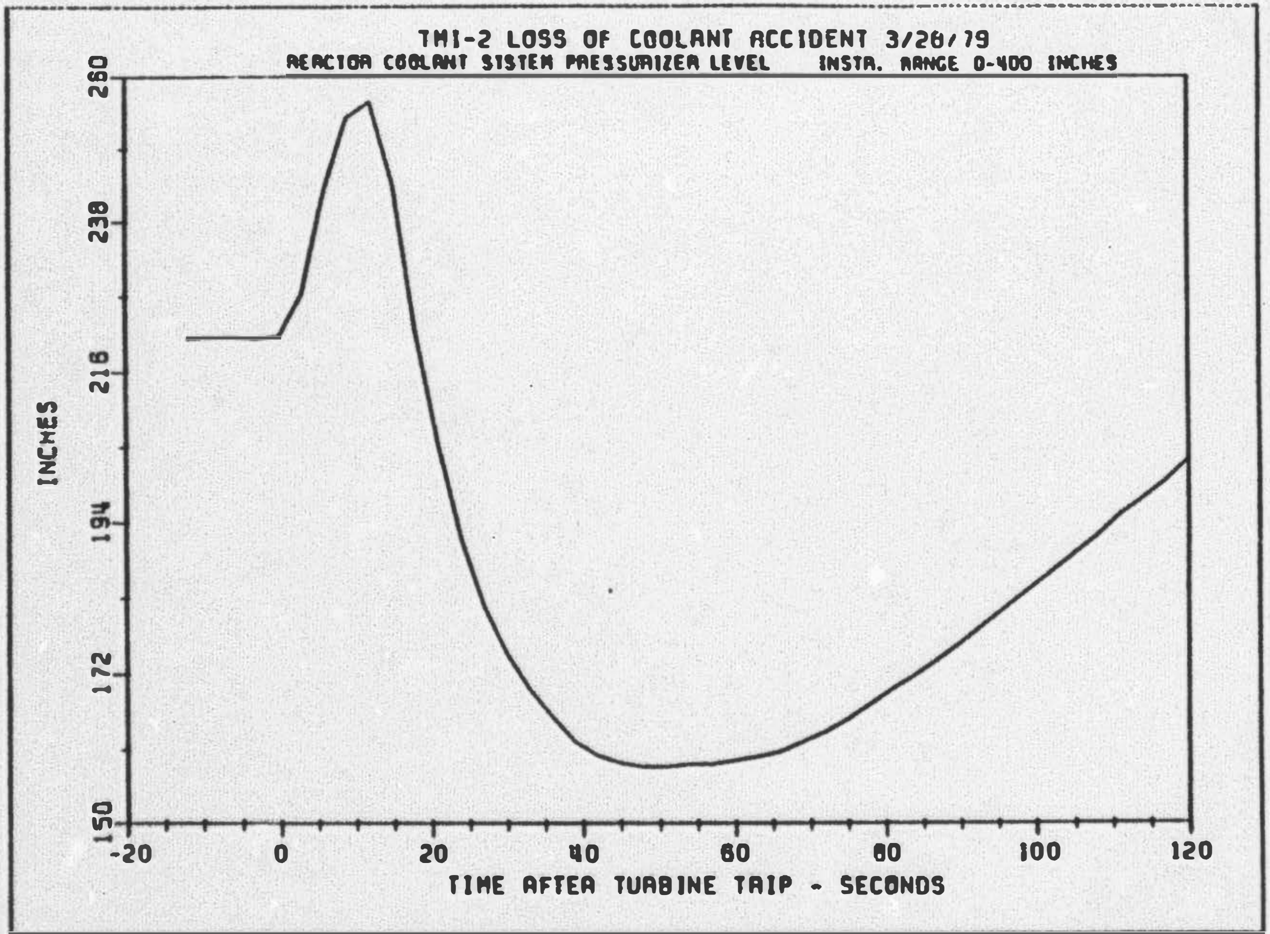


Figure 31 RCS Pressurizer Level Vs Time after Turbine Trip

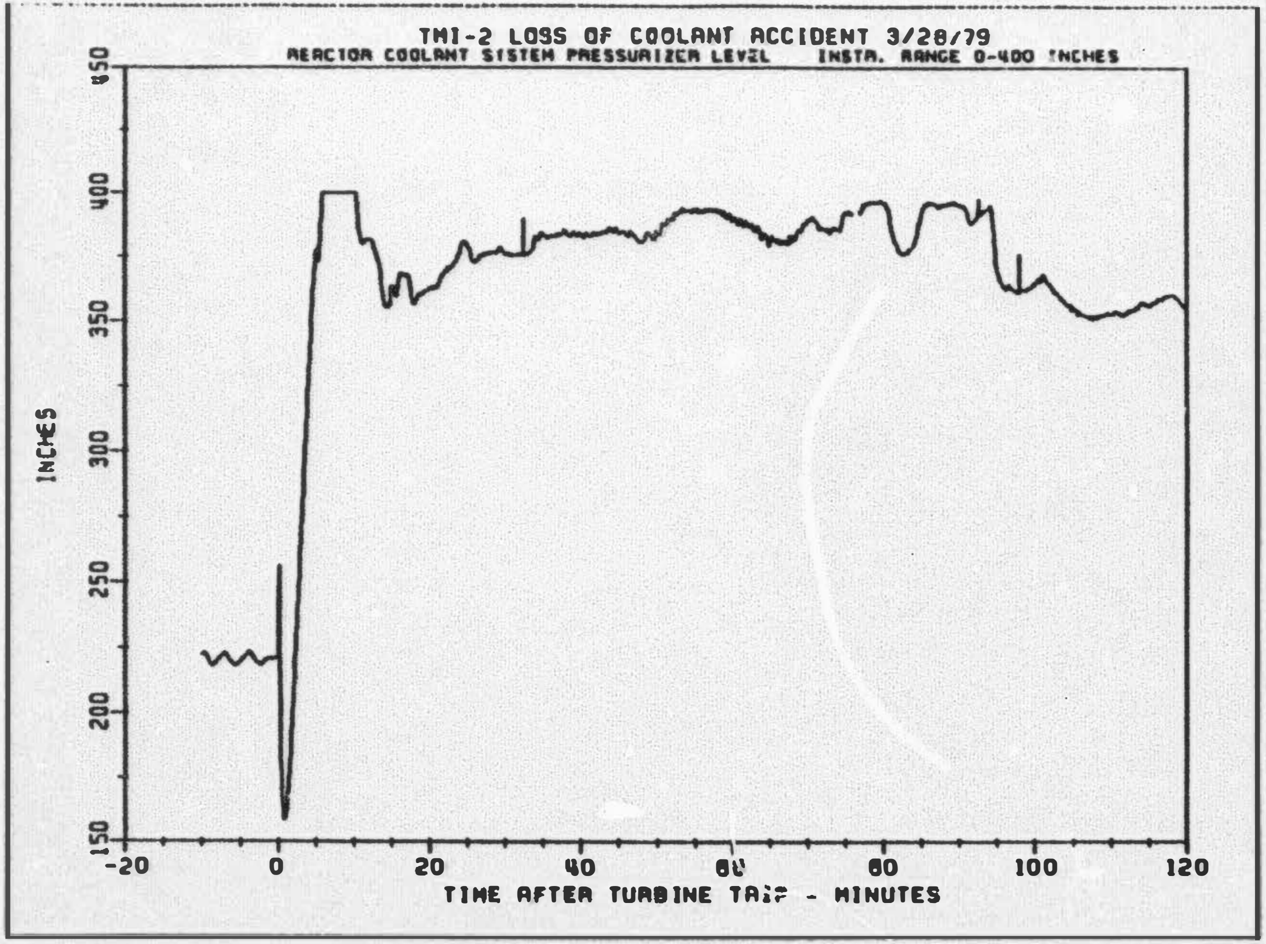


Figure 32 RCS Pressurizer Level Vs Time after Turbine Trip

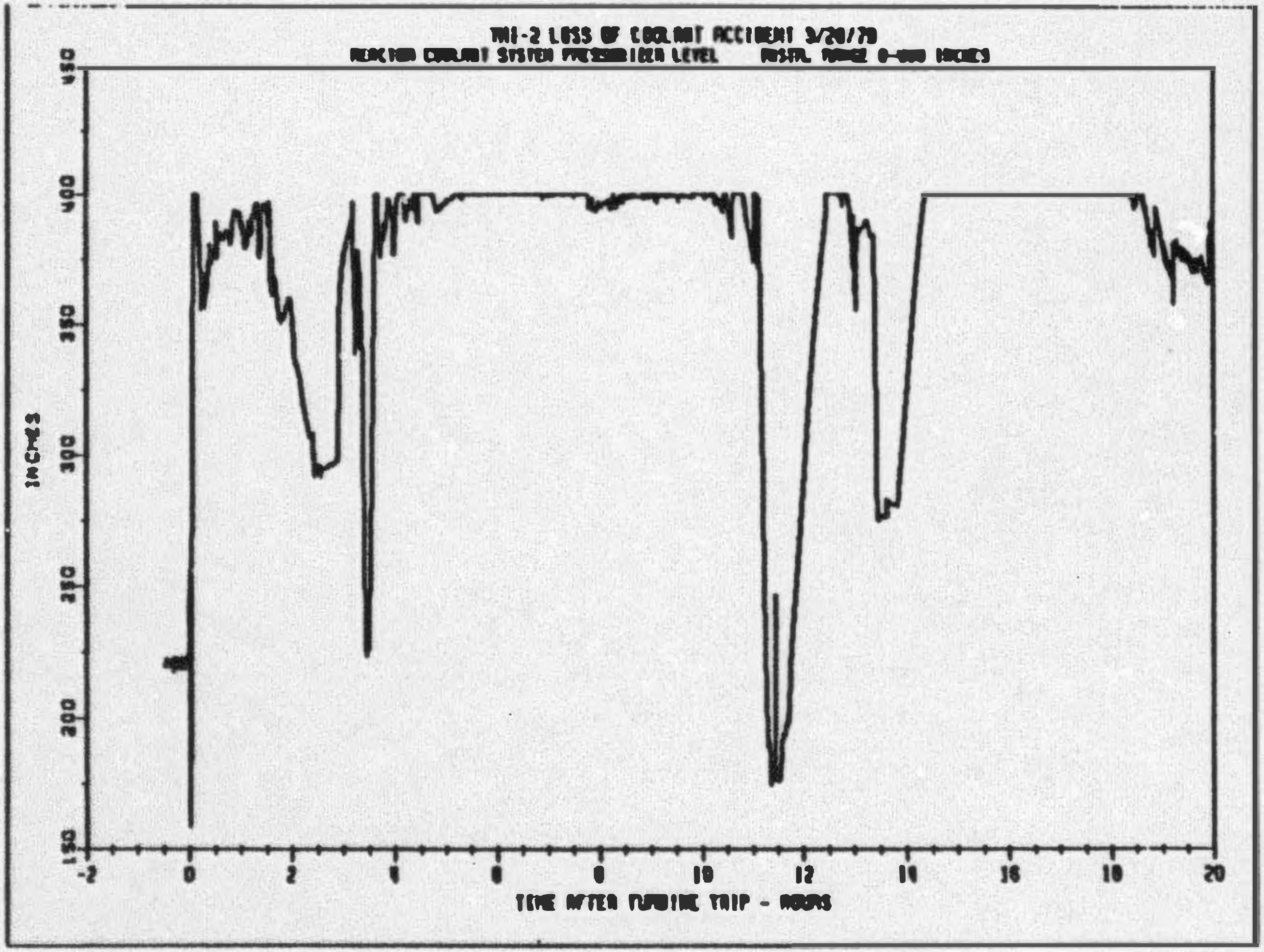


Figure 33 RCS Pressurizer Level Vs Time after Turbine Trip

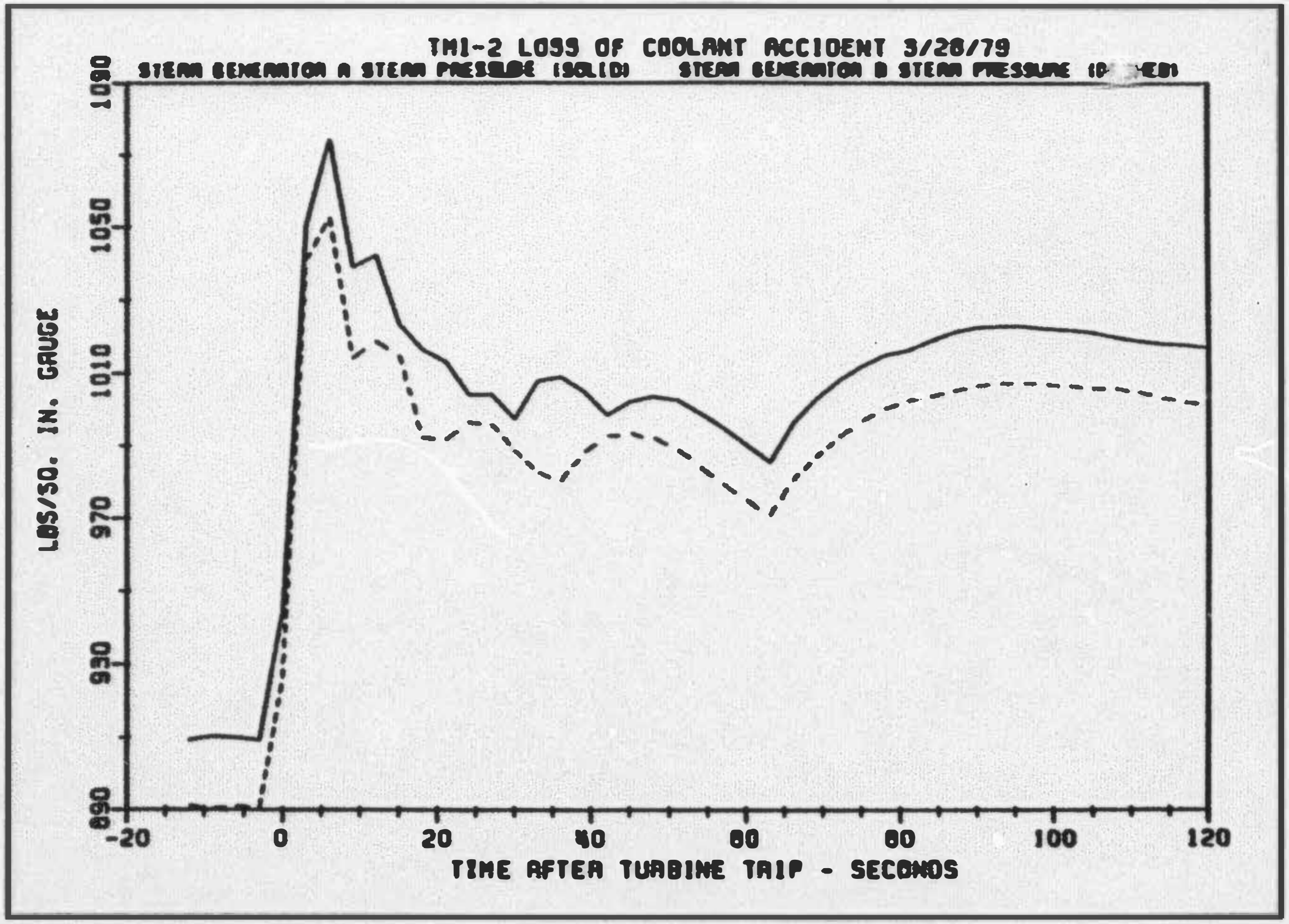


Figure 34 Steam Generator Steam Pressure Vs Time after Turbine Trip

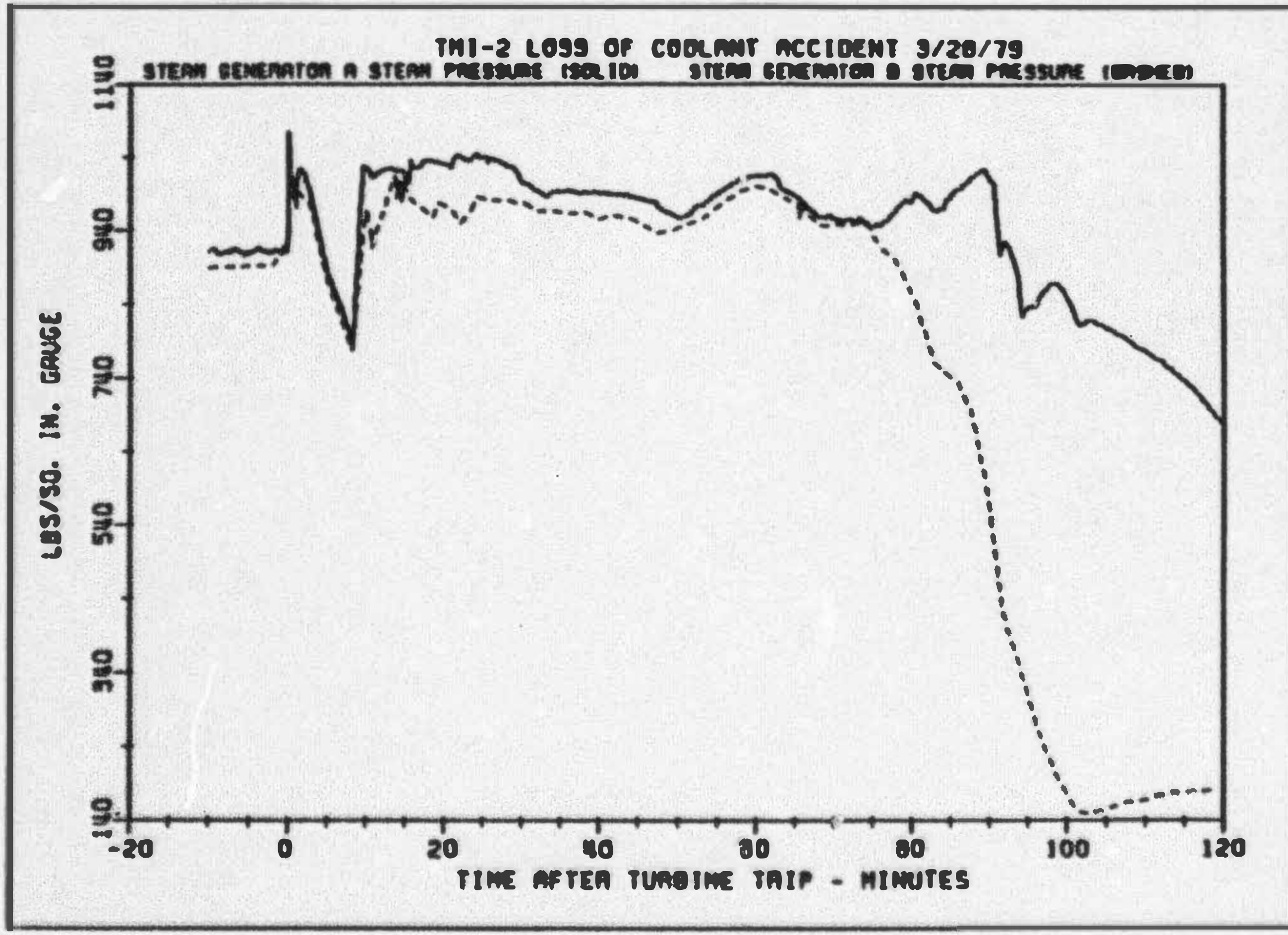


Figure 35 Steam Generator Steam Pressure Vs Time after Turbine Trip

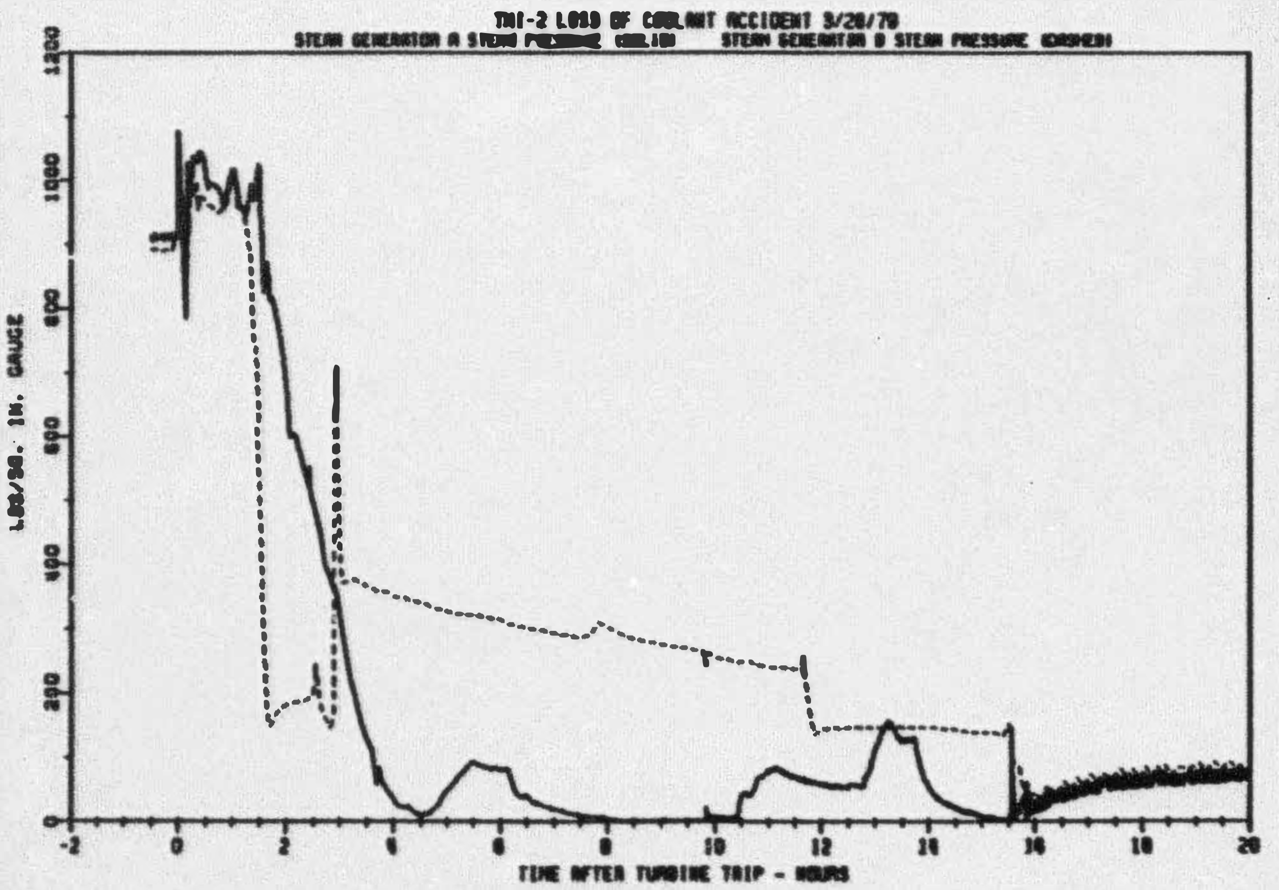


Figure 36 Steam Generator Steam Pressure Vs Time after Turbine Trip

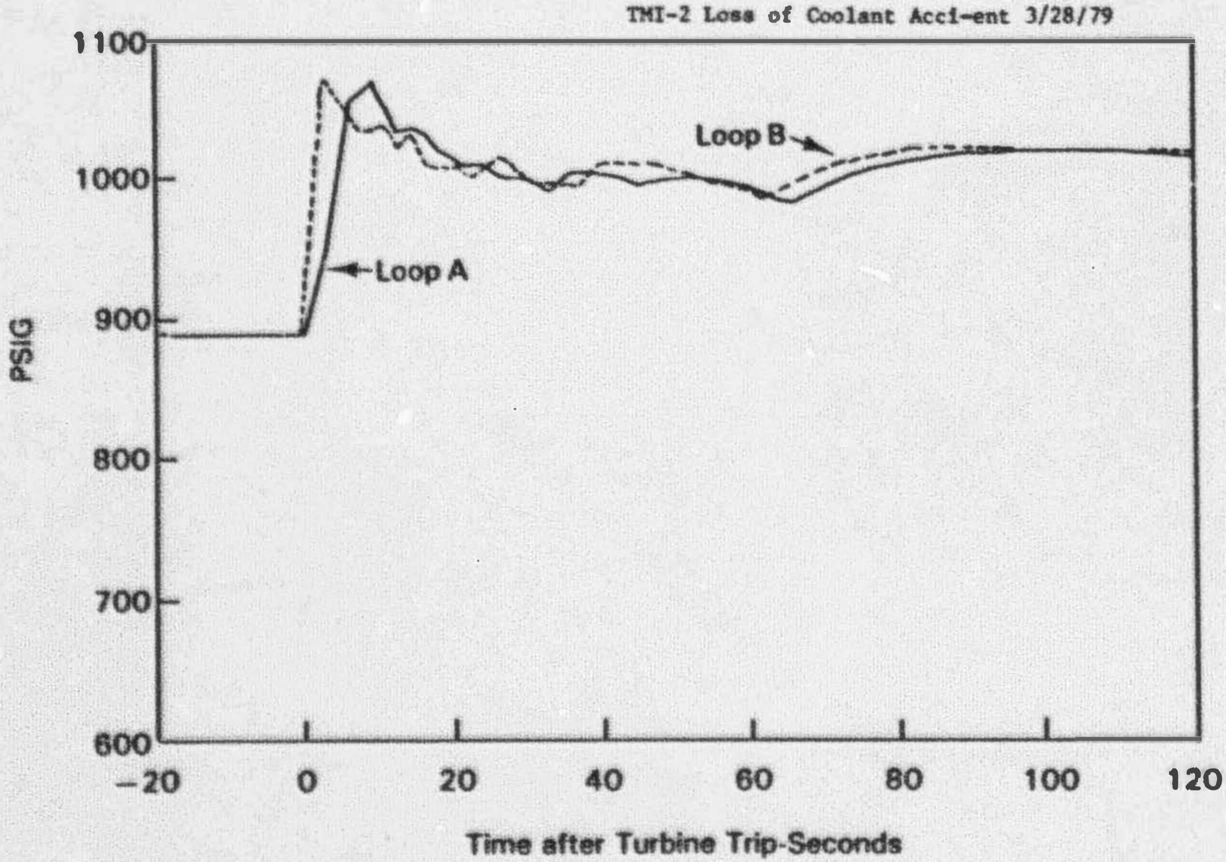


Figure 37 Turbine Header Pressure Loop A and Loop B

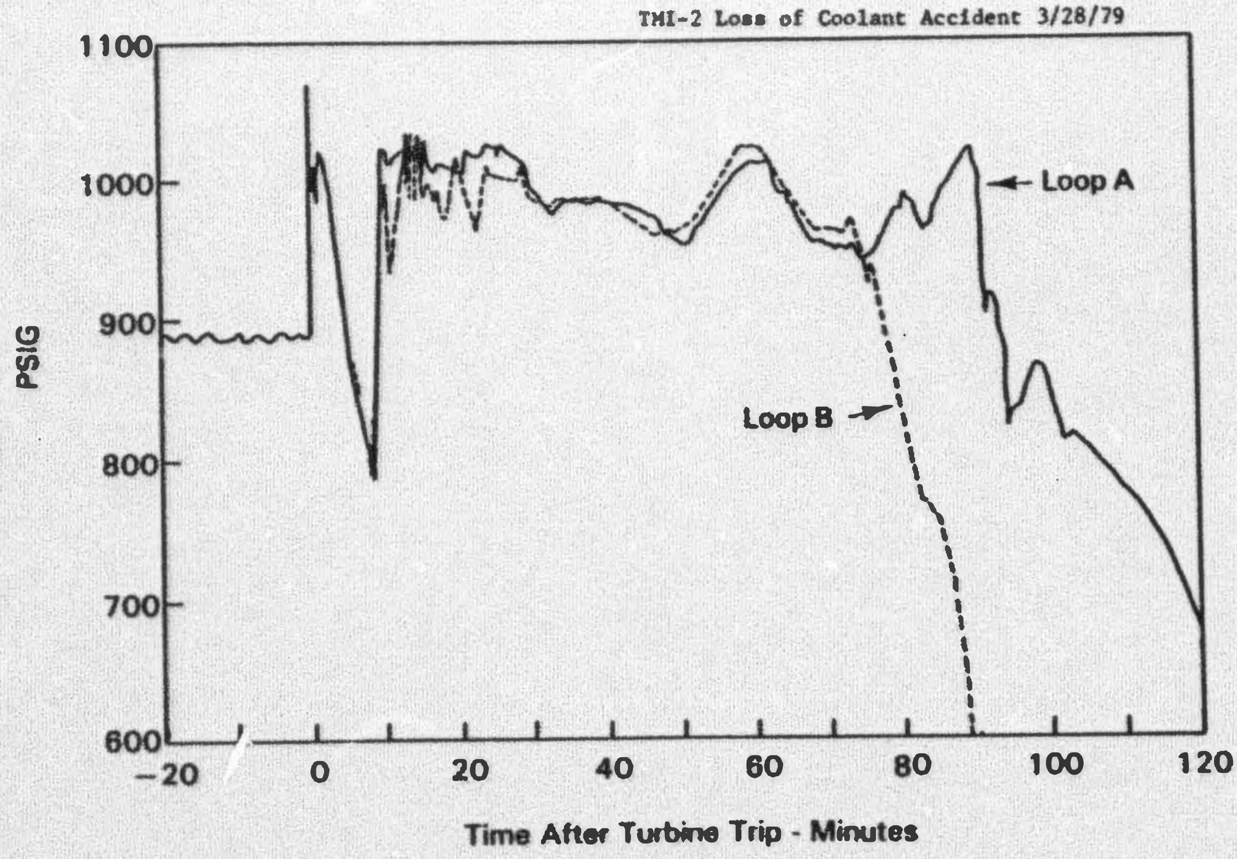


Figure 38 Turbine Header Pressure Loop A and Loop B

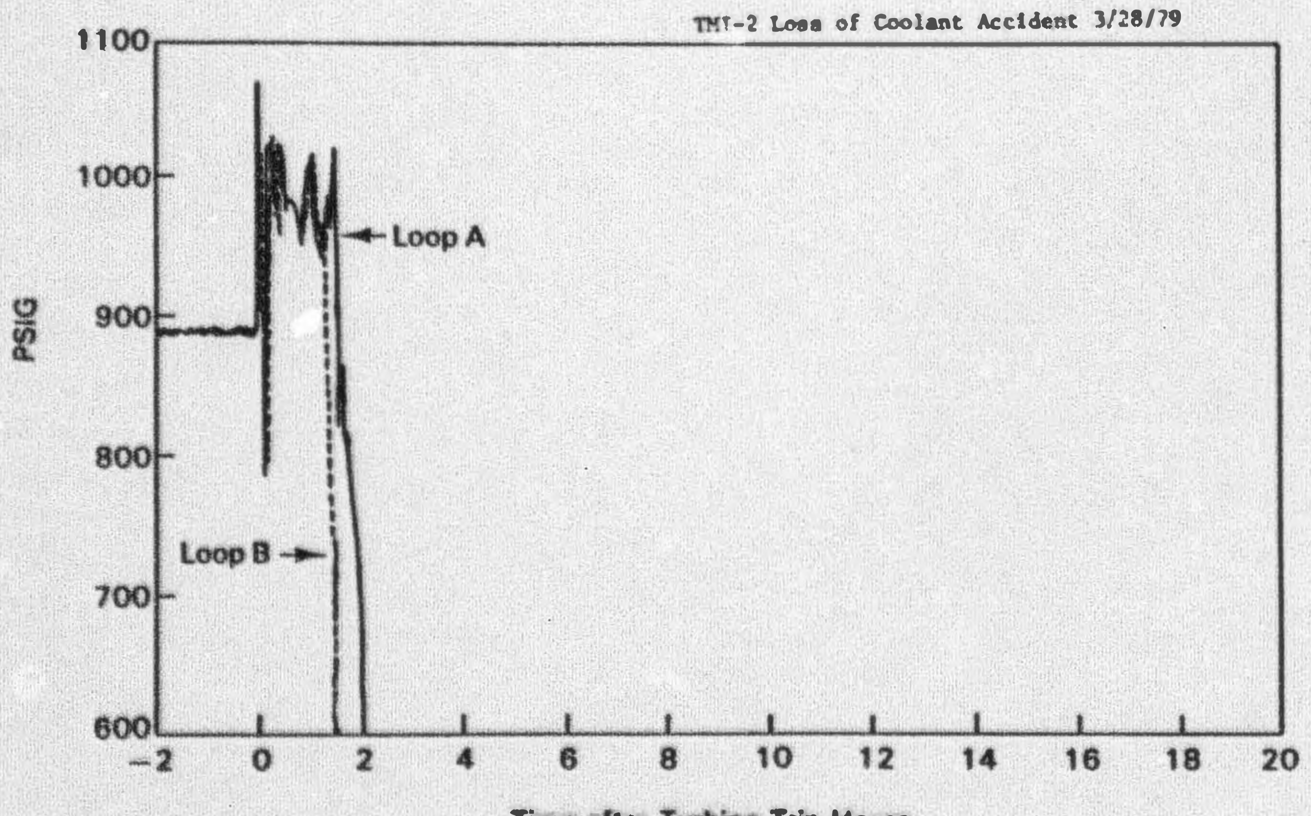


Figure 39 Turbine Header Pressure Loop A and Loop B

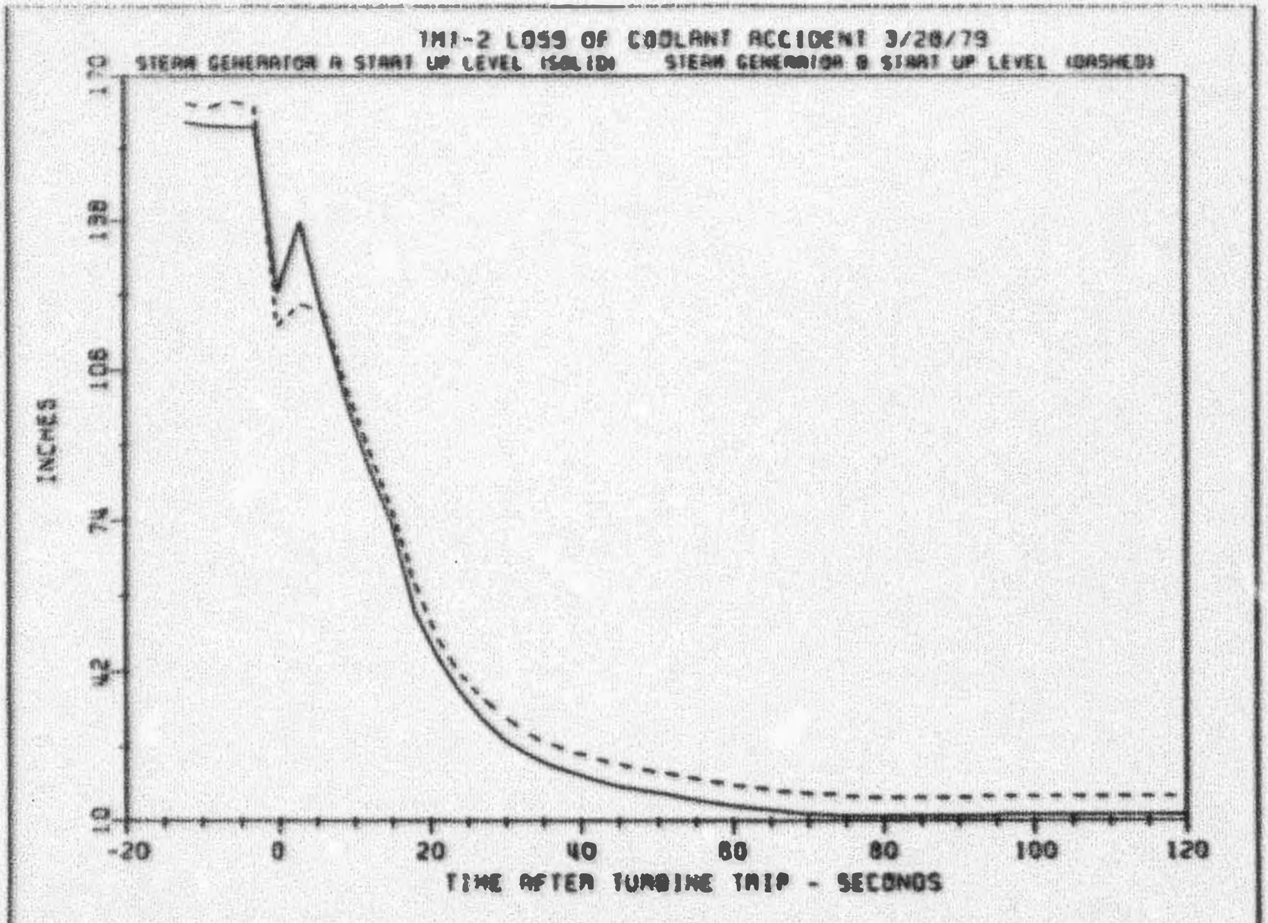


Figure 40 Steam Generator Level Vs Time after Turbine Trip

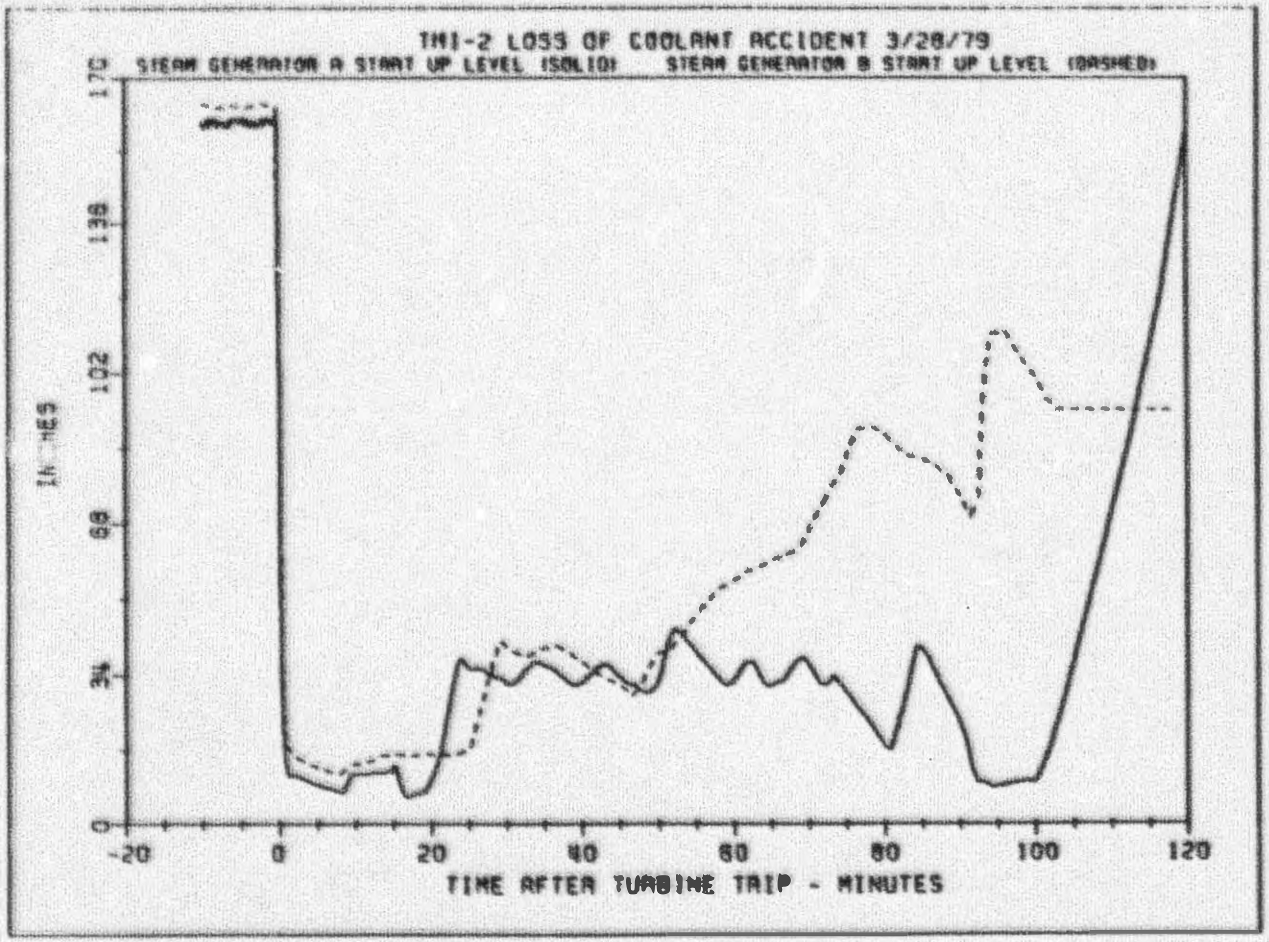


Figure 41 Steam Generator Level vs Time after Turbine Trip

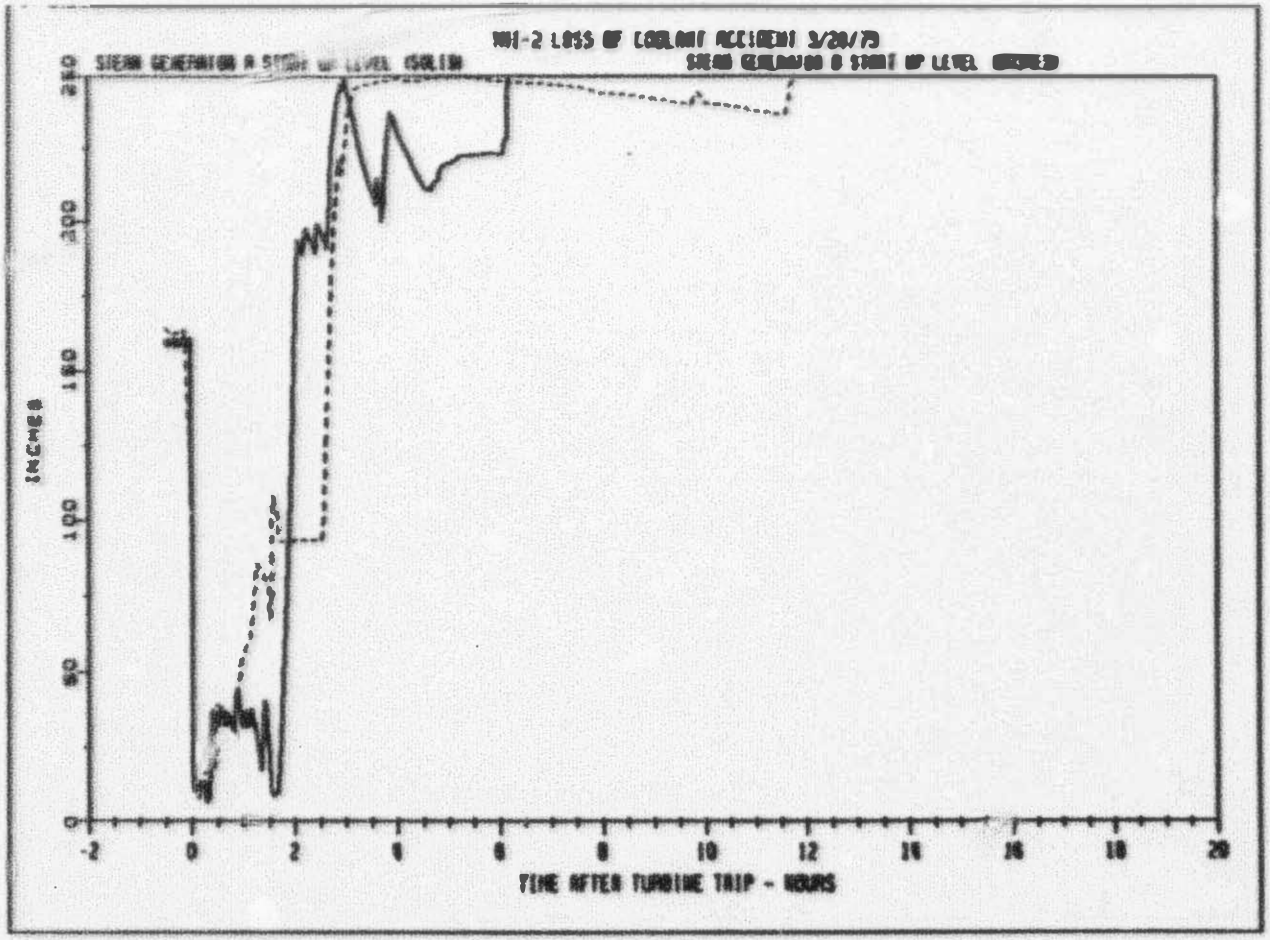


Figure 42 Steam Generator Level Vs Time after Turbine Trip

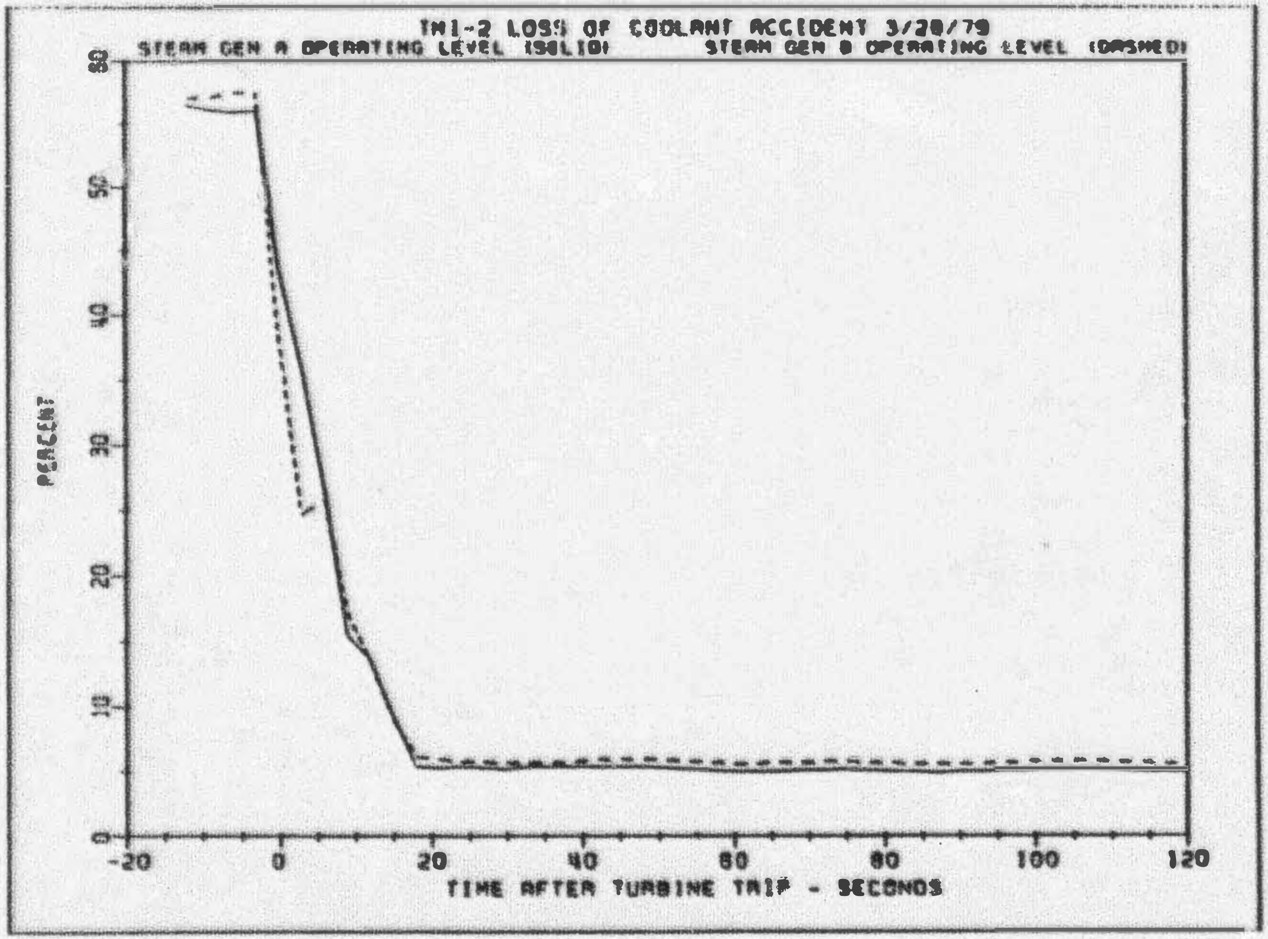


Figure 43 Steam Generator Level Vs Time after Turbine Trip

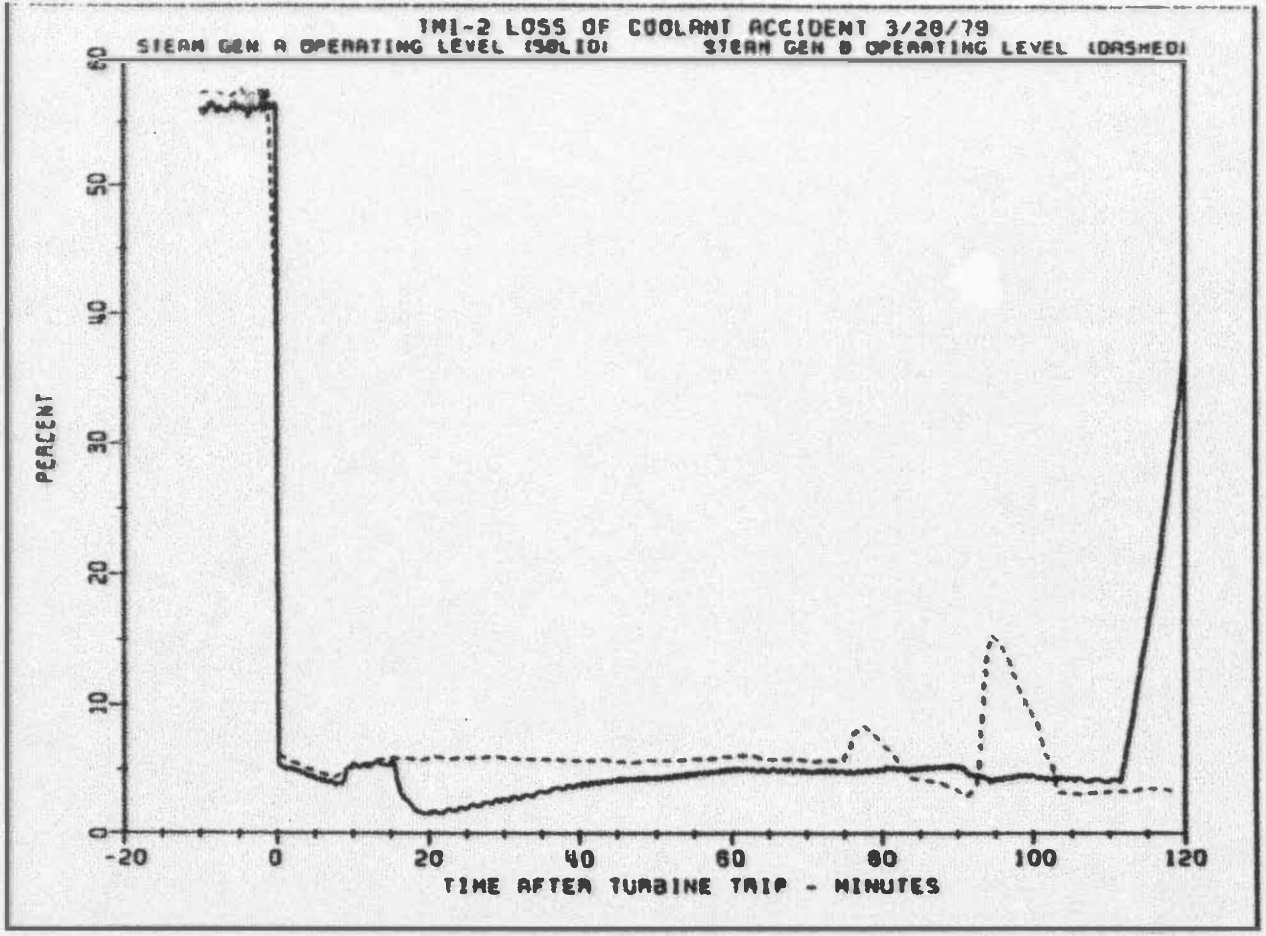


Figure 44 Steam Generator Level Vs Time after Turbine Trip

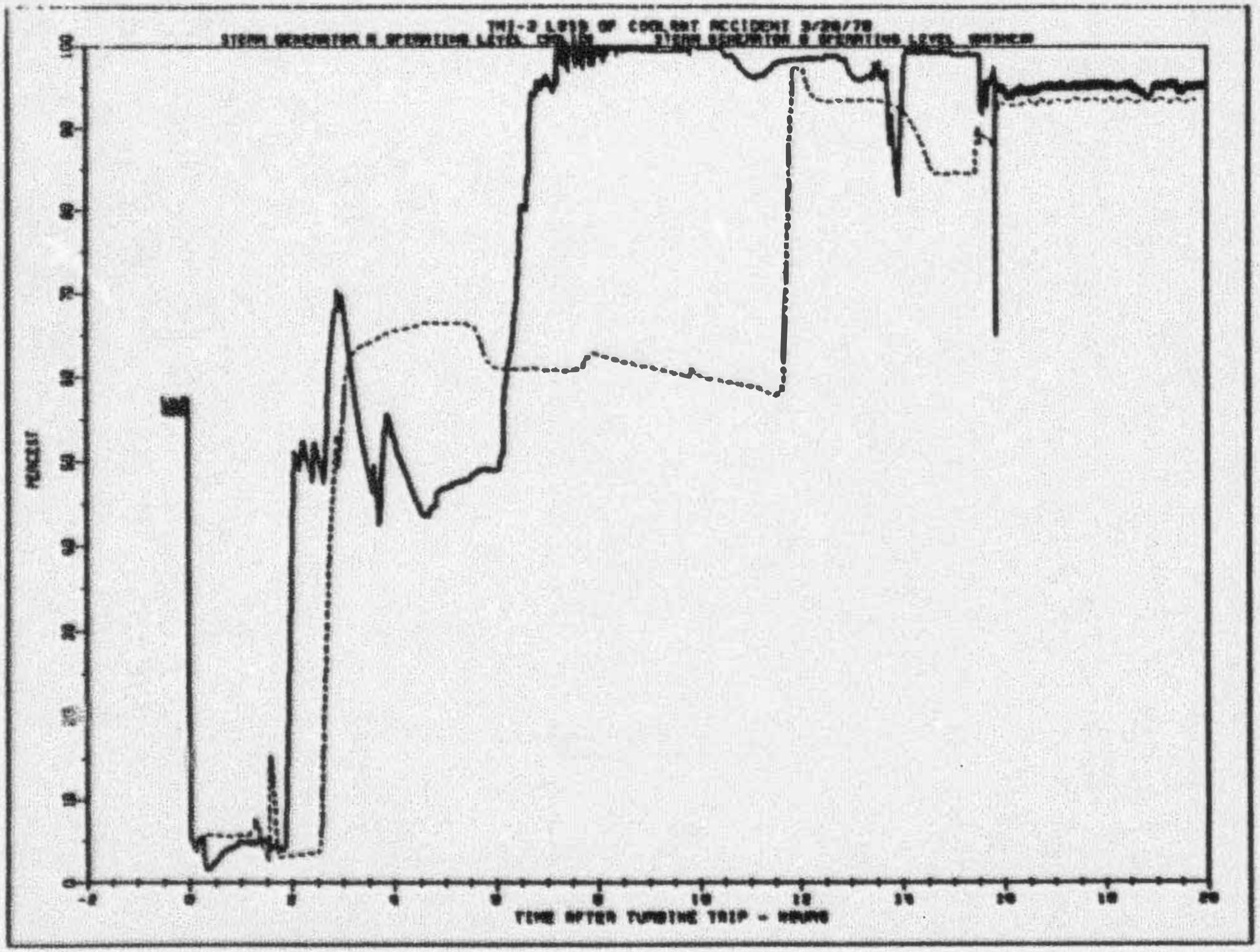


Figure 45 Steam Generator Level Vs Time after Turbine Trip

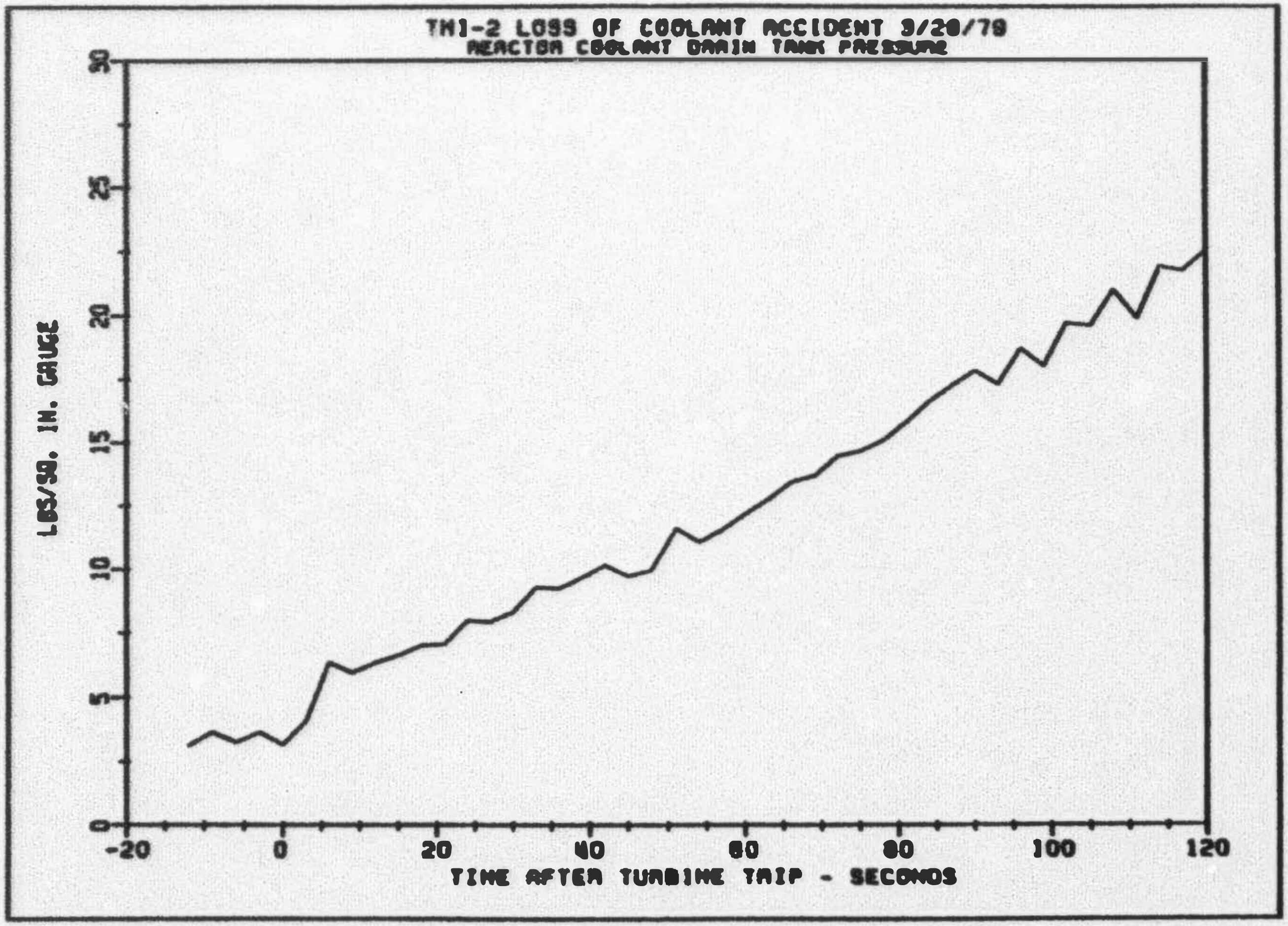


Figure 46 RC Drain Tank Pressure Vs Time after Turbine Trip

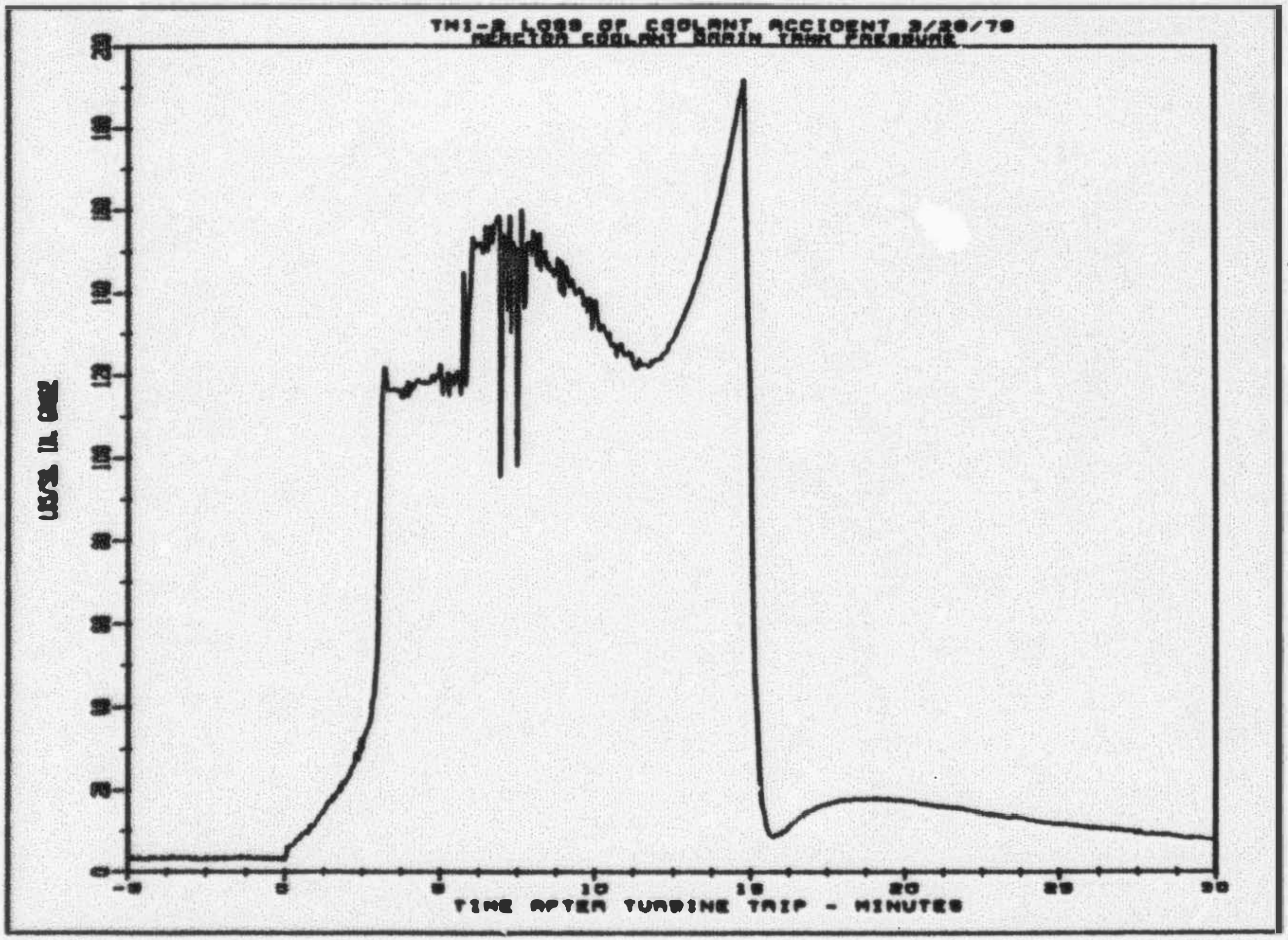


Figure 47 RC Drain Tank Pressure Vs Time after Turbine Trip

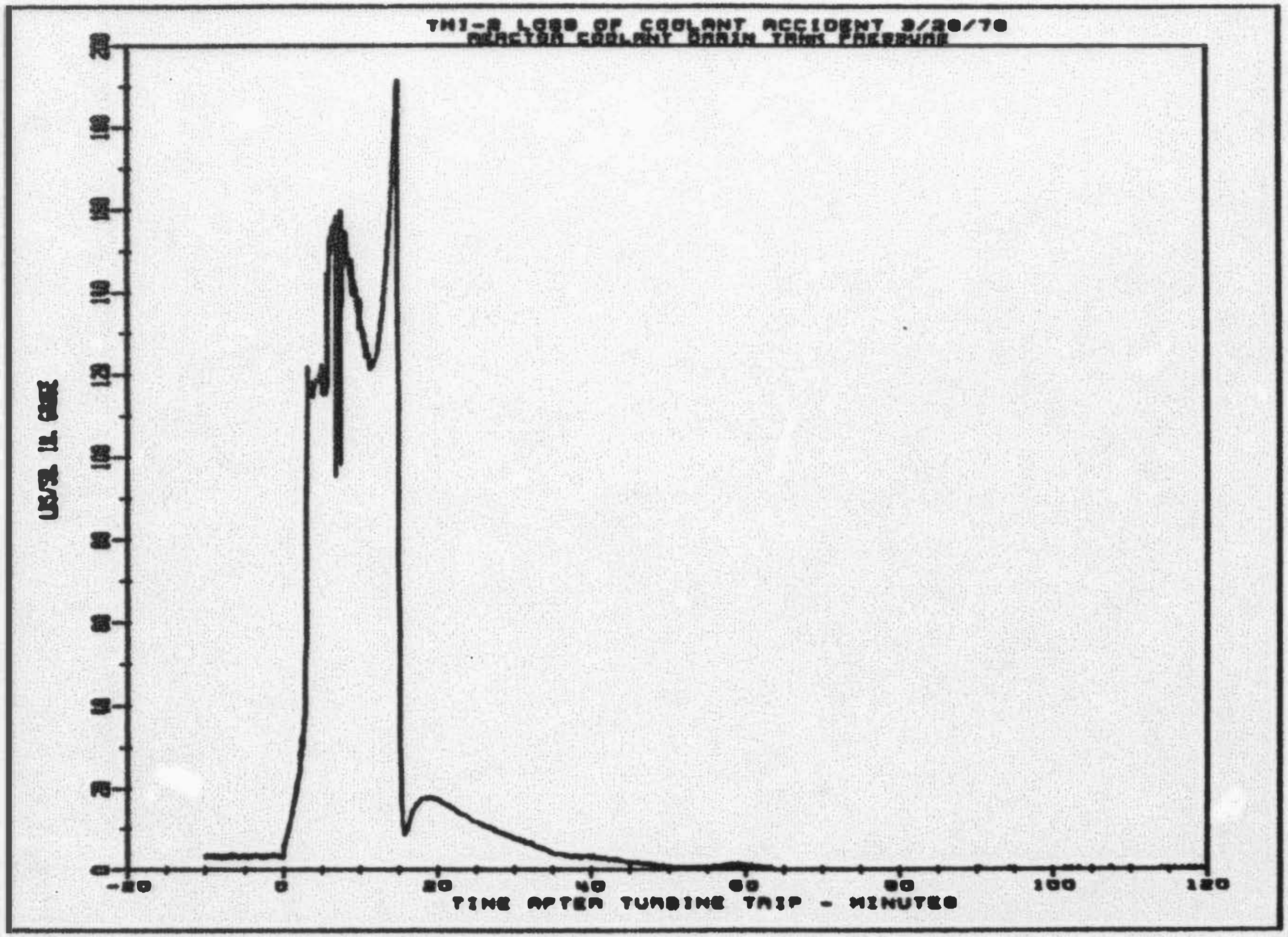


Figure 48 RC Drain Tank Pressure Vs Time after Turbine Trip

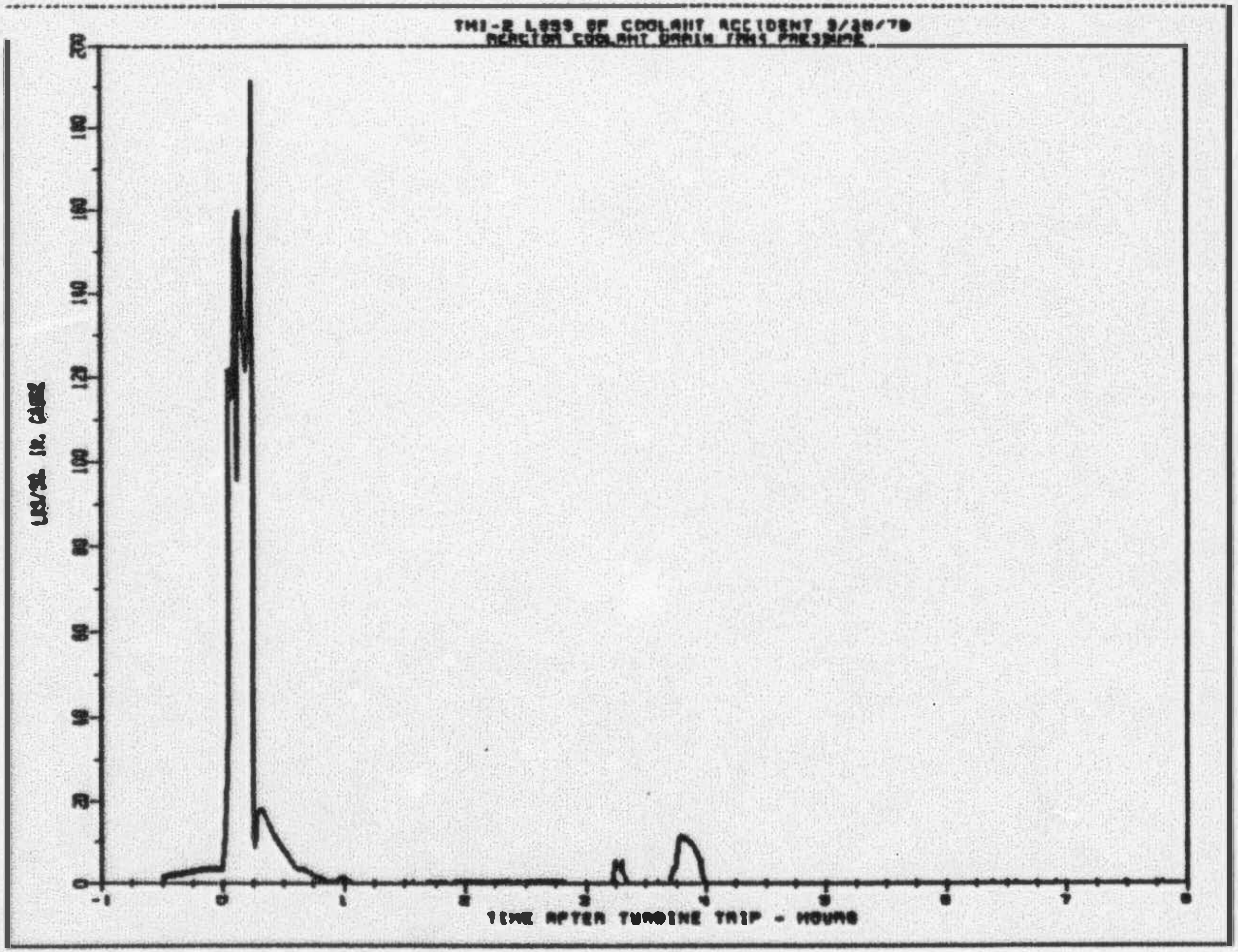


Figure 49 RC Drain Tank Pressure Vs Time After Turbine Trip

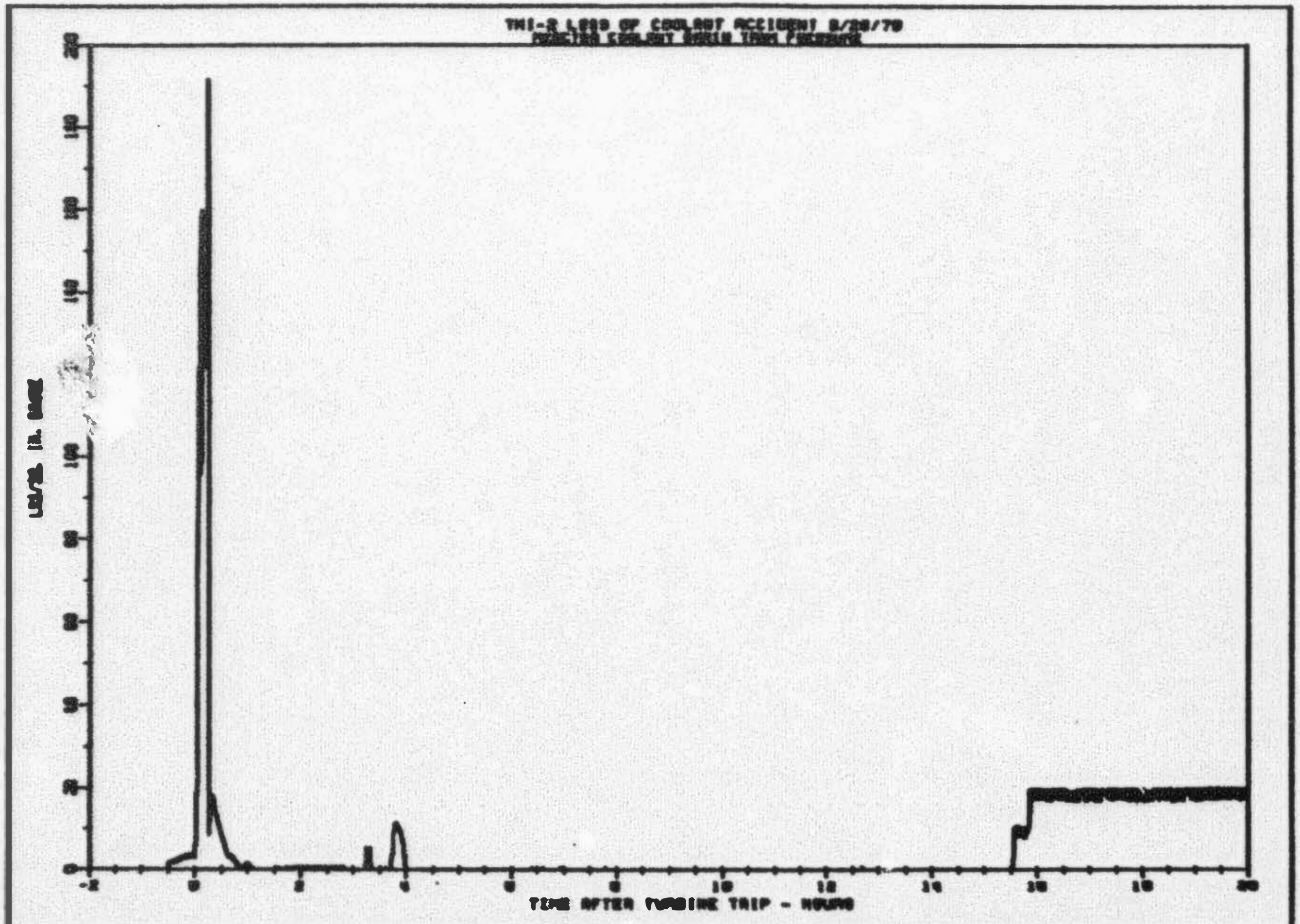


Figure 50 RC Drain Tank Pressure Vs Time after Turbine Trip

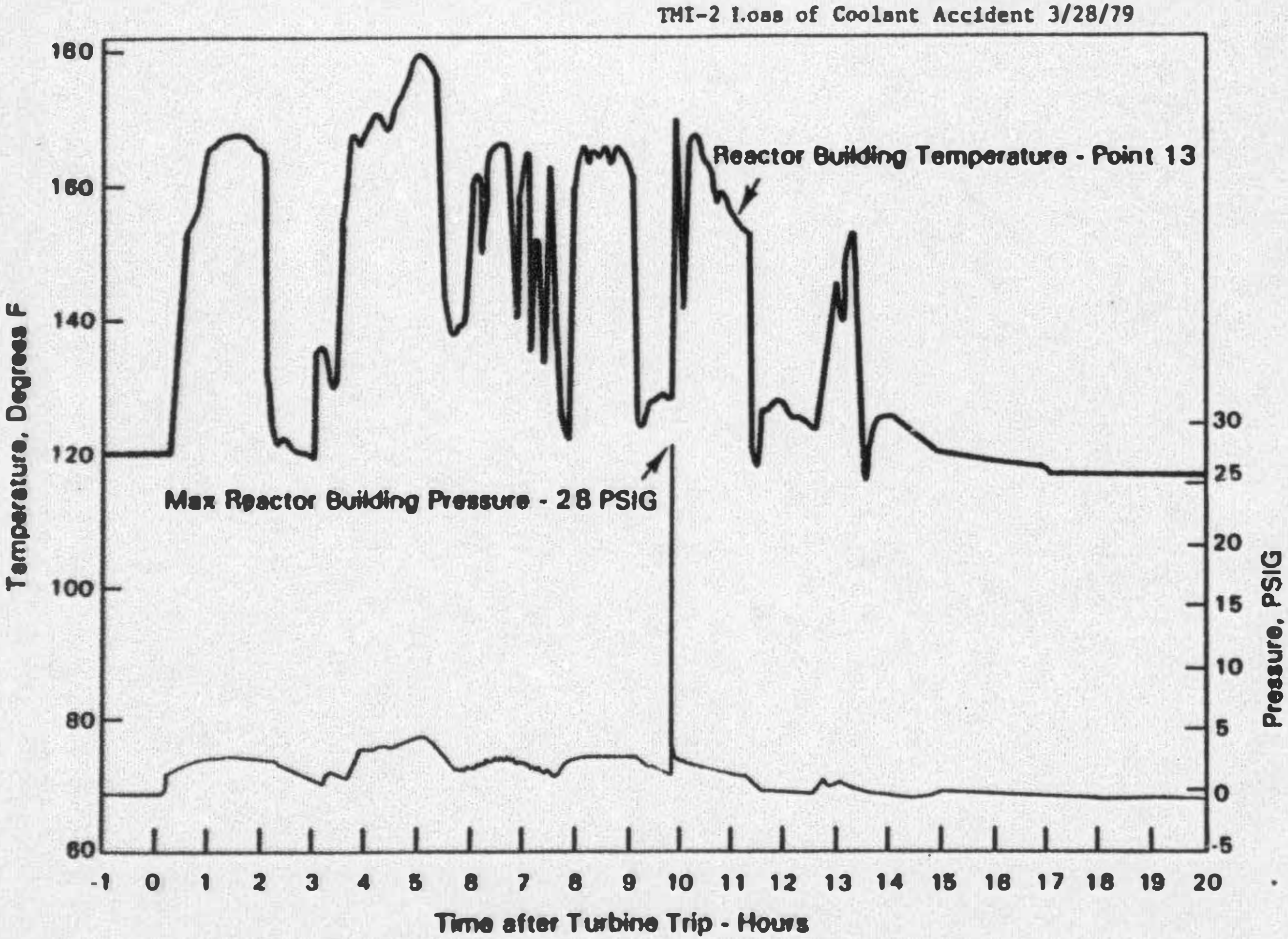


Figure 51 Reactor Building Temperature and Pressure

TMI-2 Loss of Coolant Accident 3/28/79

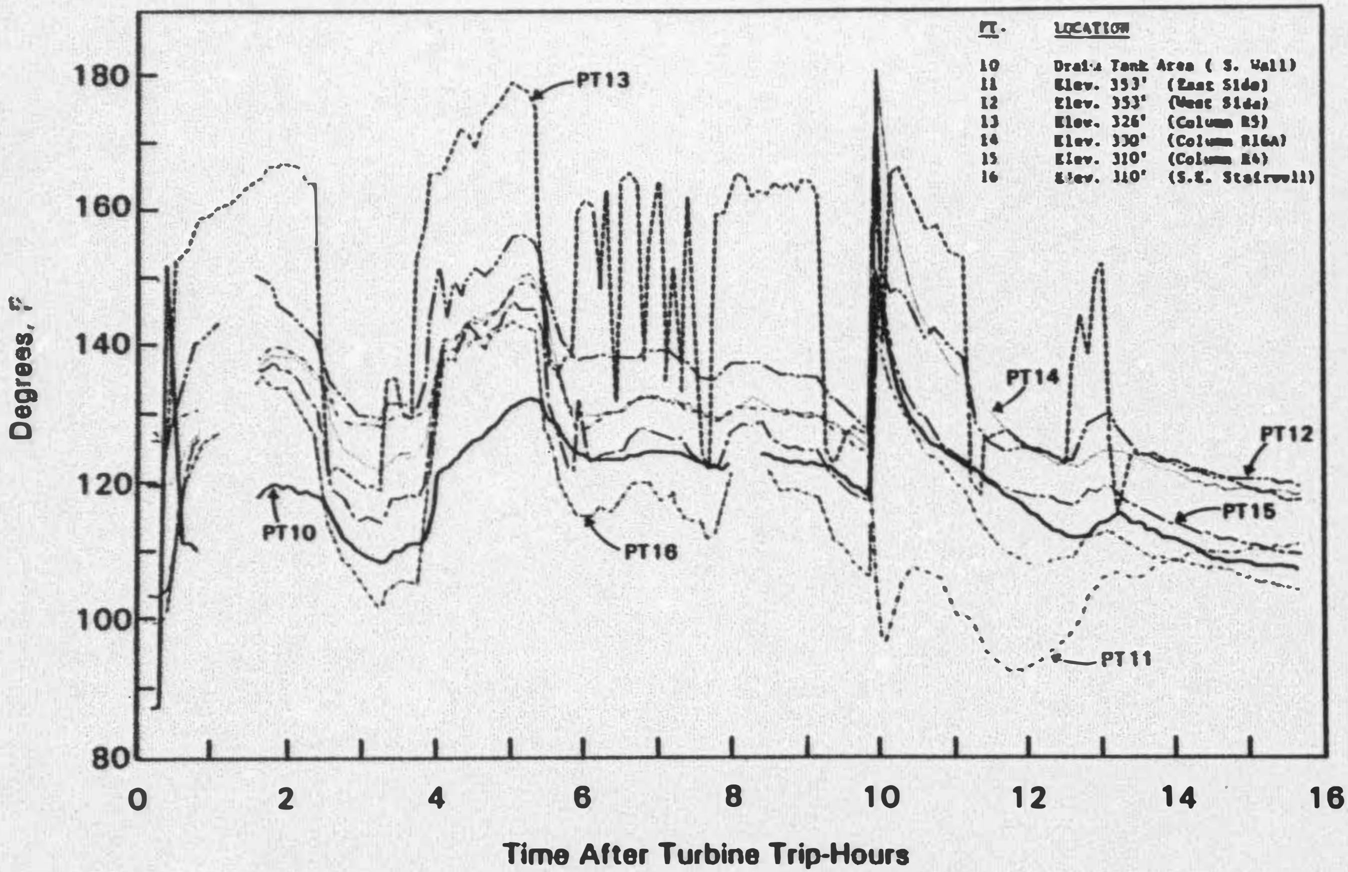


Figure 52 R_x Building Temperature Recorder AH-YMTR-5017, Panel 25

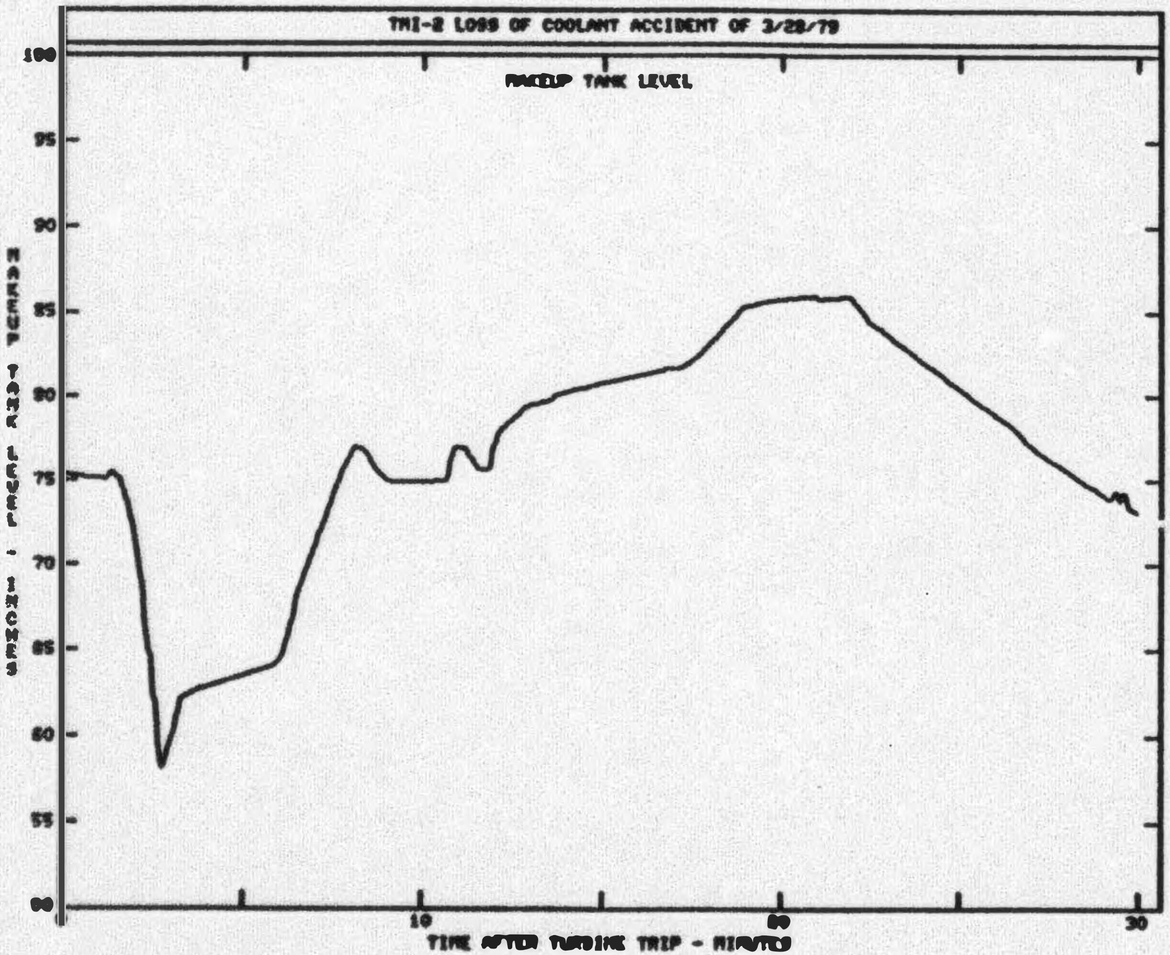


Figure 53 Makeup Tank Level Vs Time after Turbine Trip

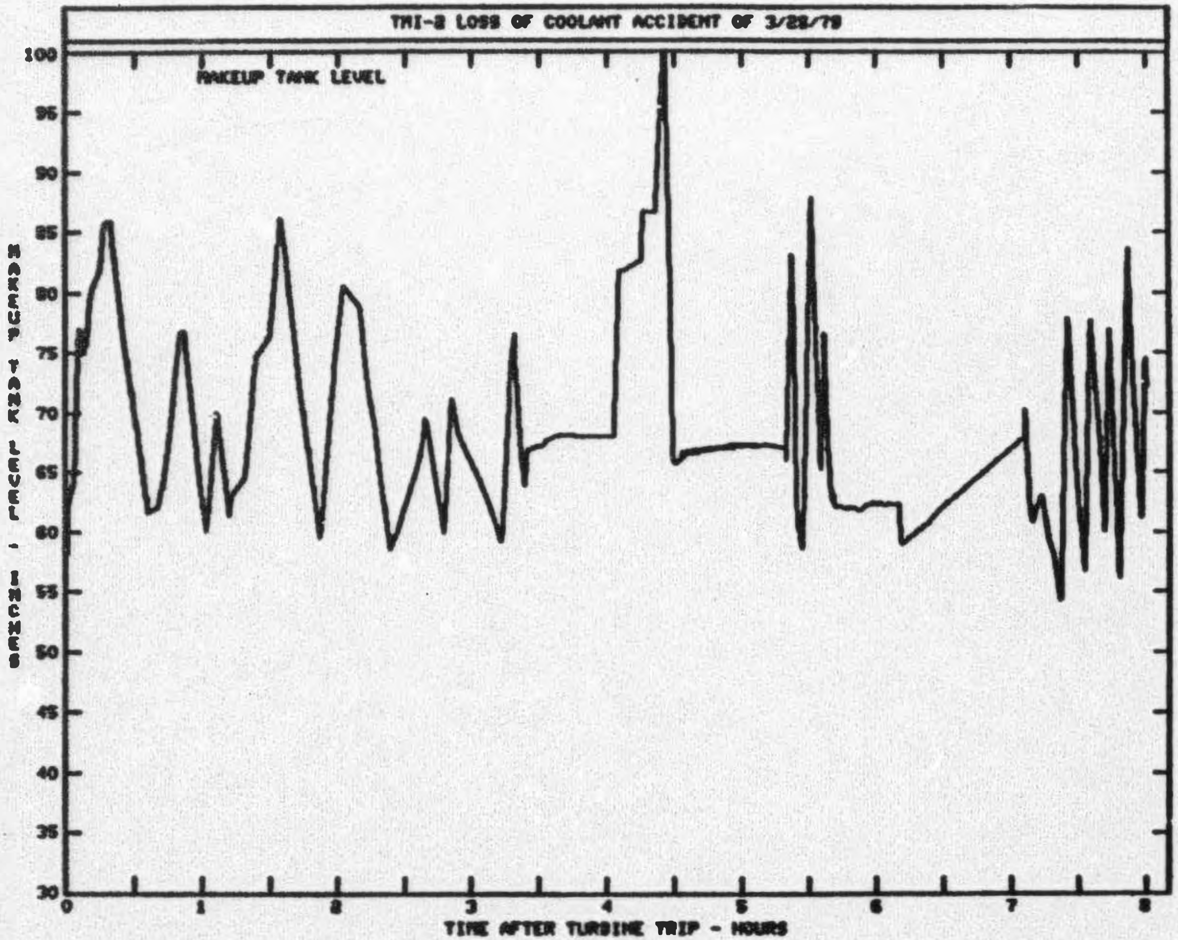


Figure 54 Makeup Tank Level Vs Time after Turbine Trip

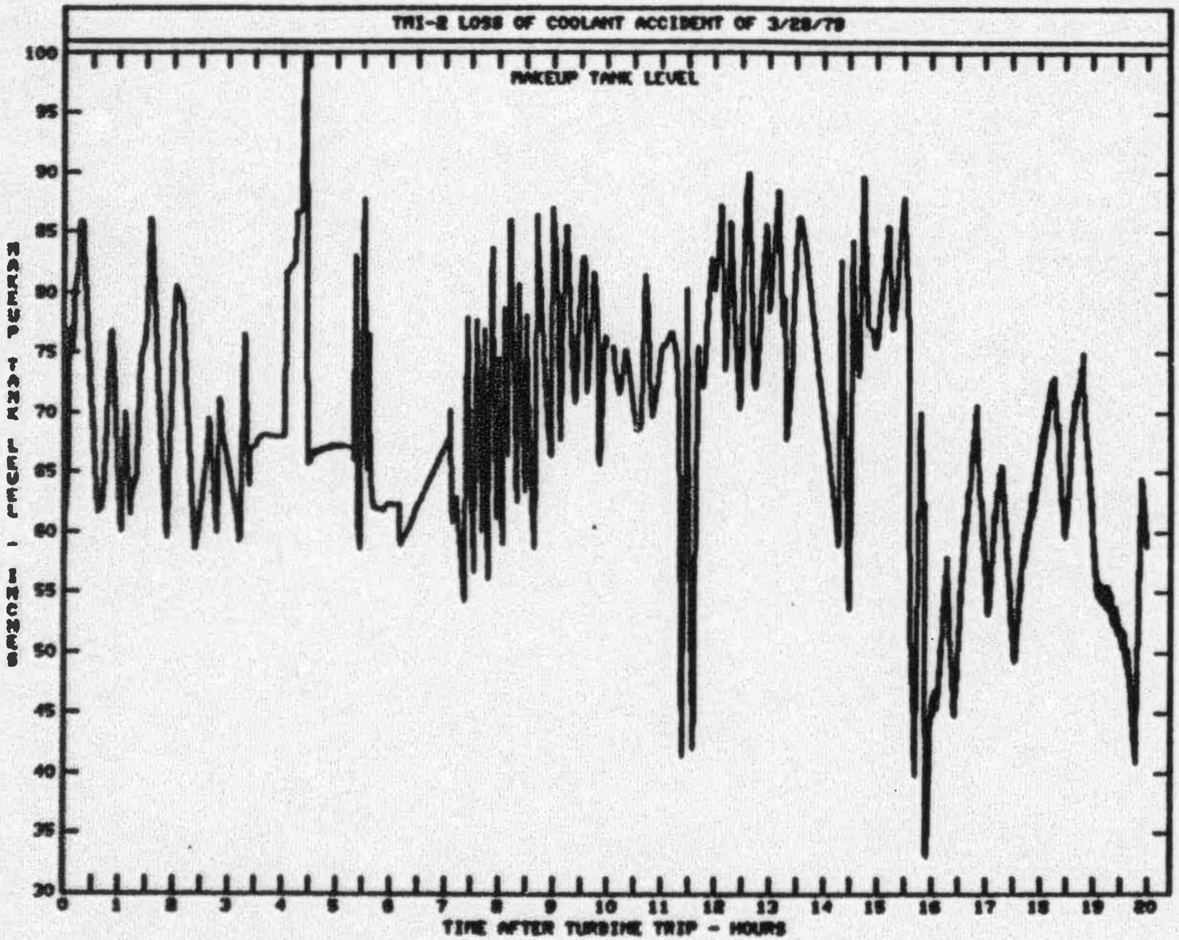


Figure 55 Makeup Tank Level Vs Time after Turbine Trip

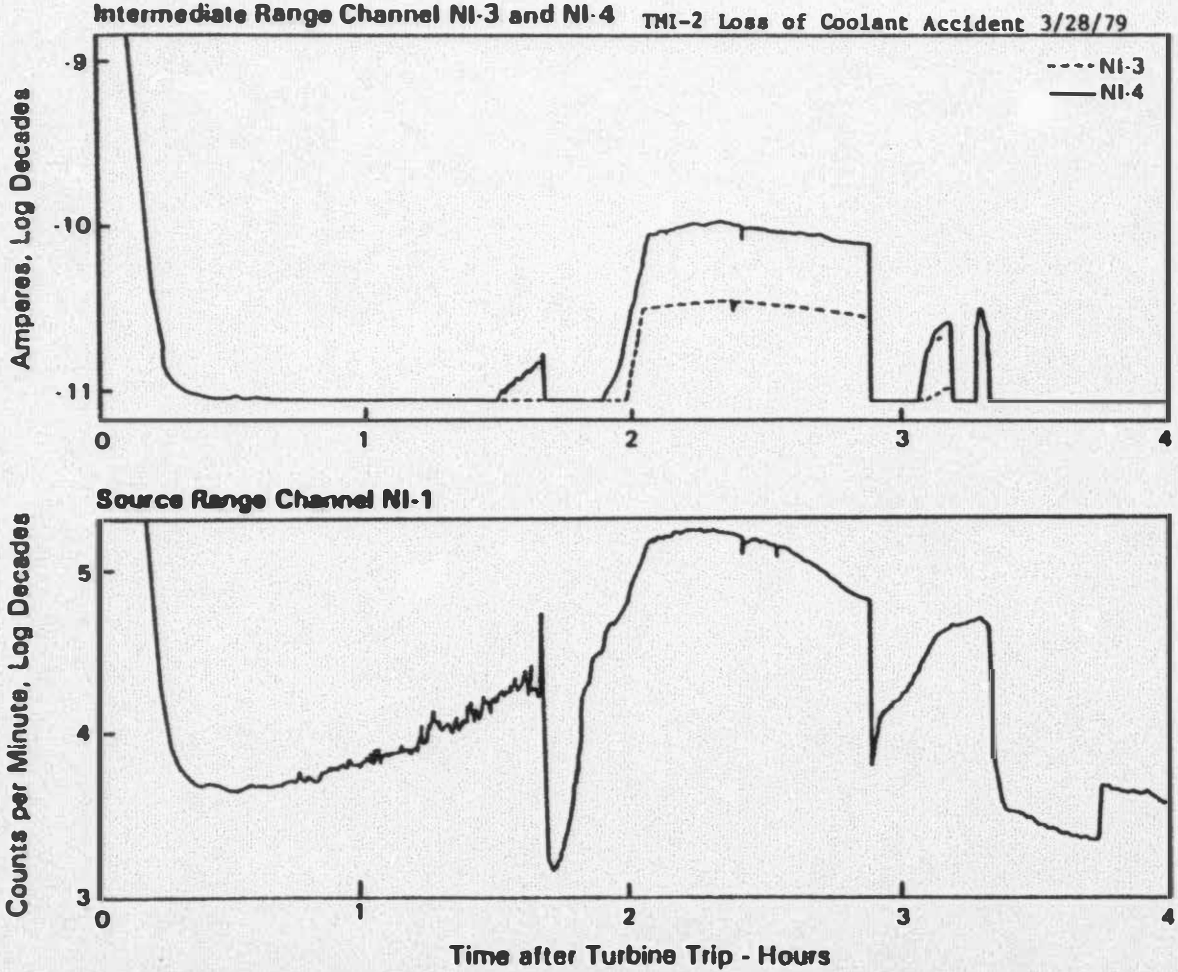


Figure 56 Intermediate and Source Range Nuclear Instrumentation

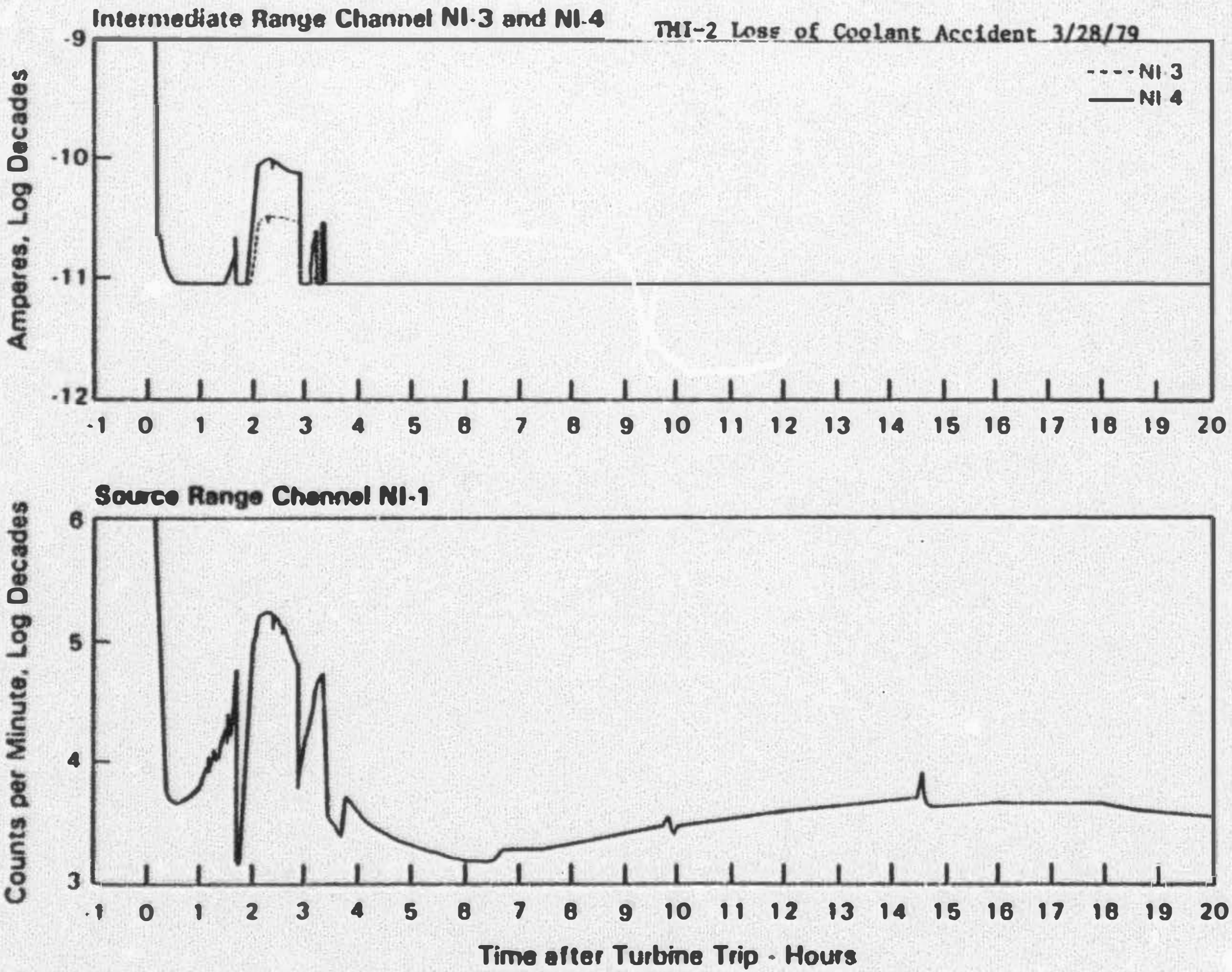


Figure 57 Intermediate and Source Range Nuclear Instrumentation

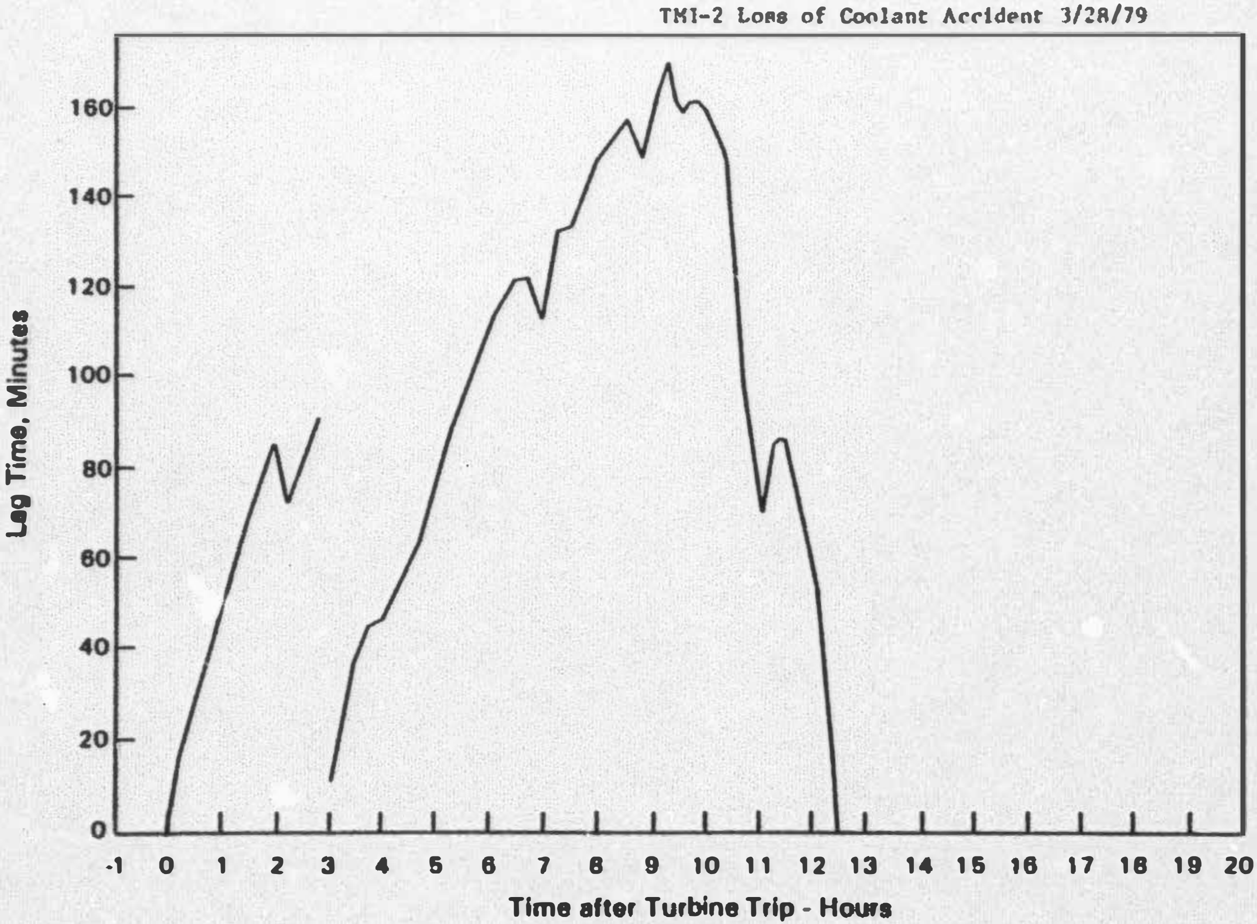


Figure 58 Computer Alarm Printer Lag Time

TMI-2 Loss of Coolant Accident 3/28/79

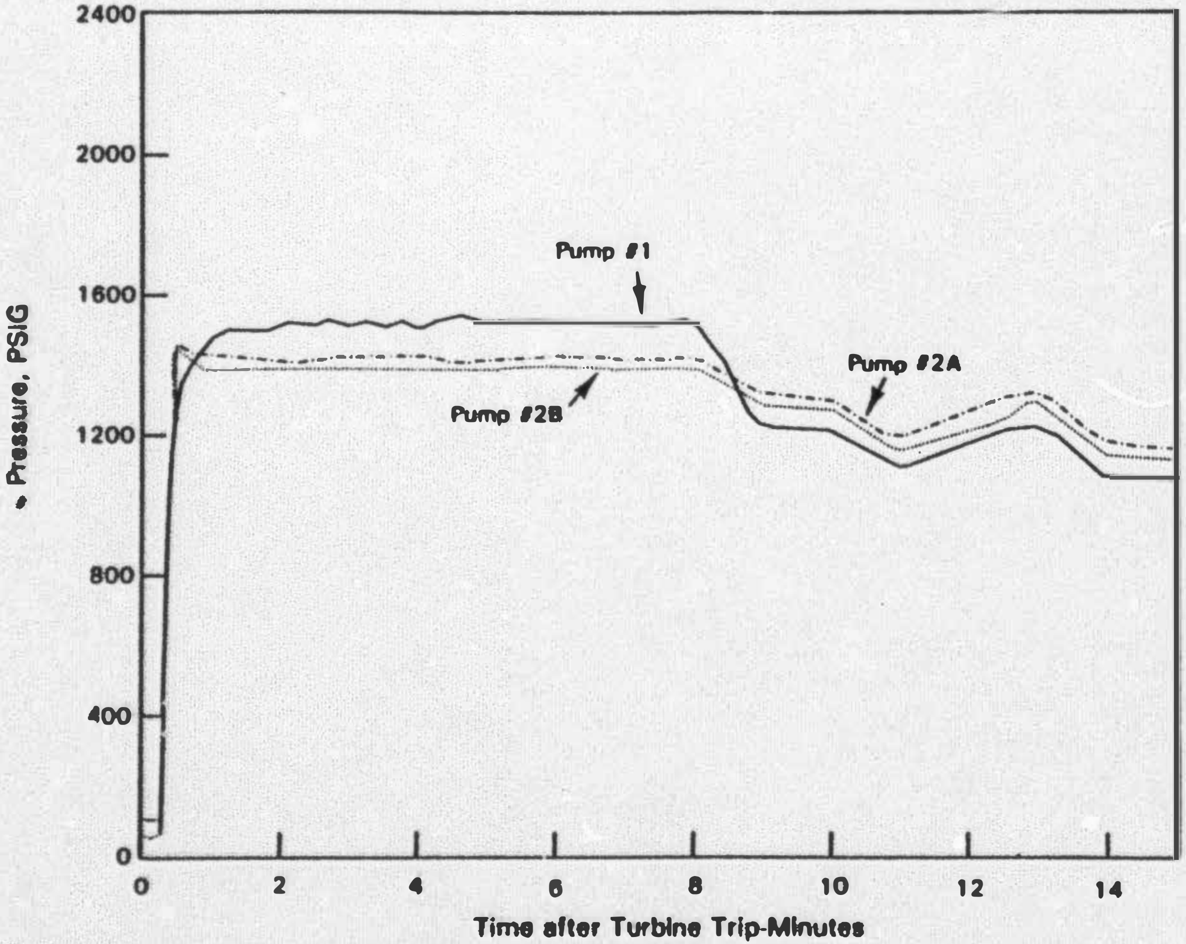
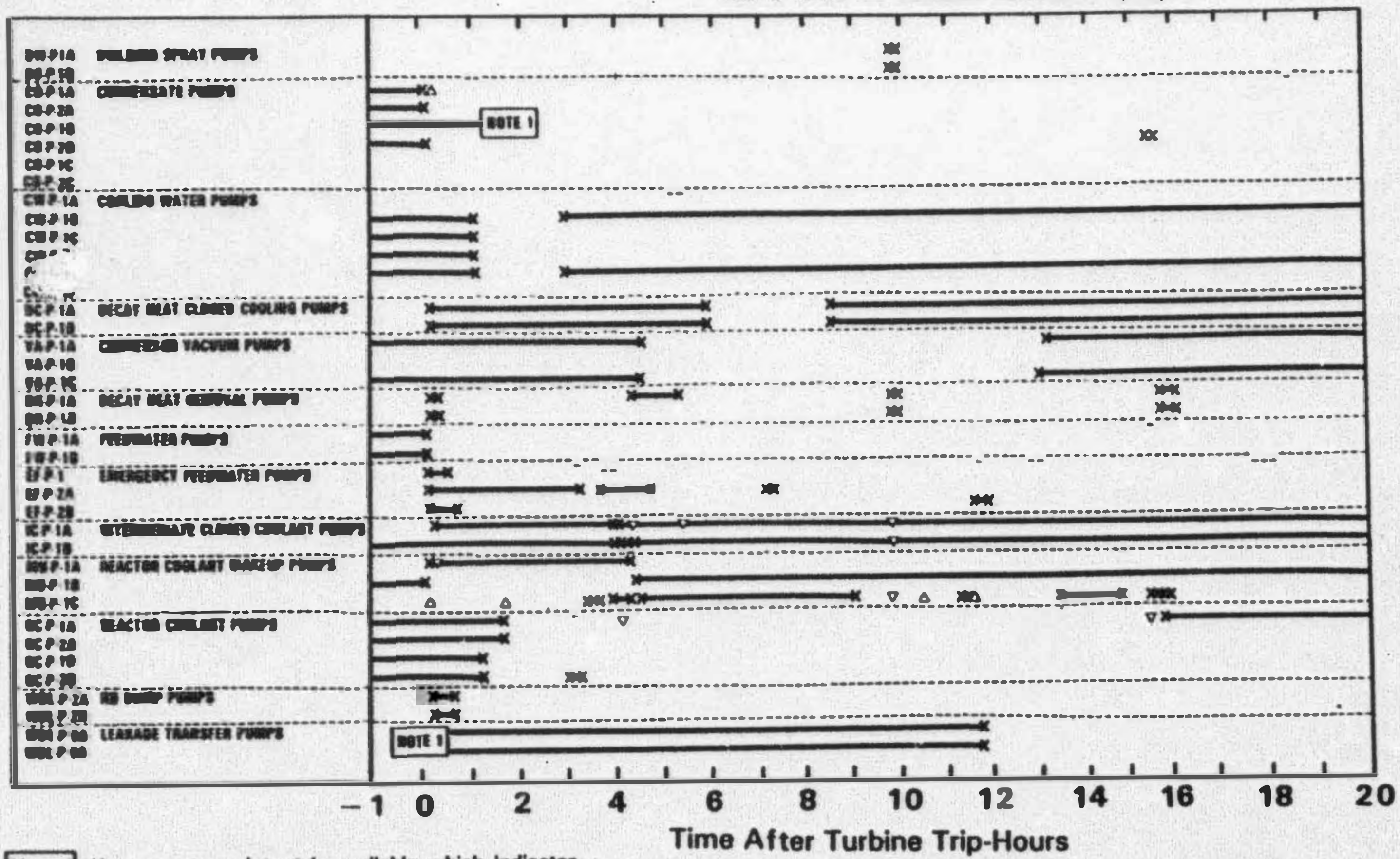


Figure 59 Emergency Feedwater Pump Discharge Pressures

TMI-2 Loss of Coolant Accident 3/28/79



Note 1: No computer printout is available which indicates when condensate pump 1B was secured or when leakage transfer pumps 9A and 9B were started.

△ Ran for short period (less than 5 minutes)

▽ Stop/start in less than 1 minute

Start Run Stop

Figure 60 Pump Operating History

TMI-2 Loss of Coolant Accident 3/28/79

Figure 61

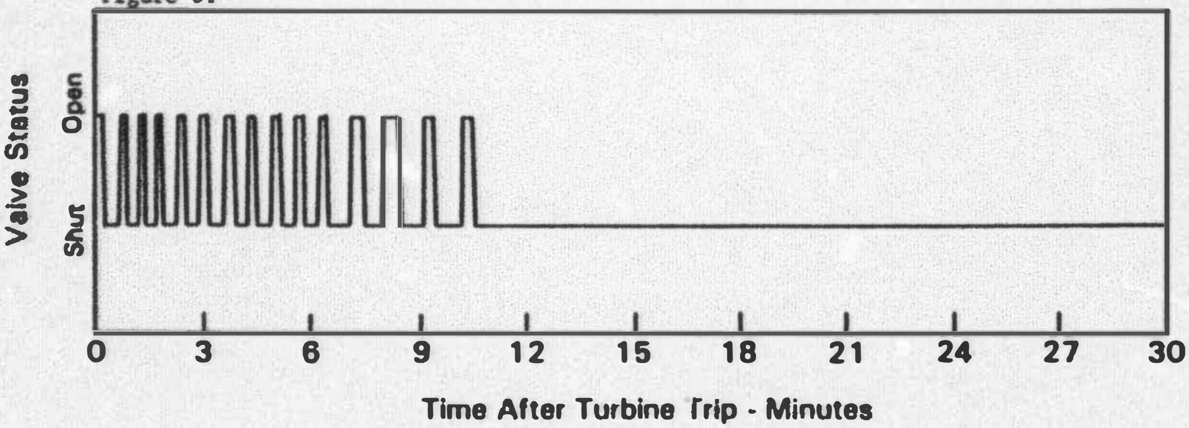
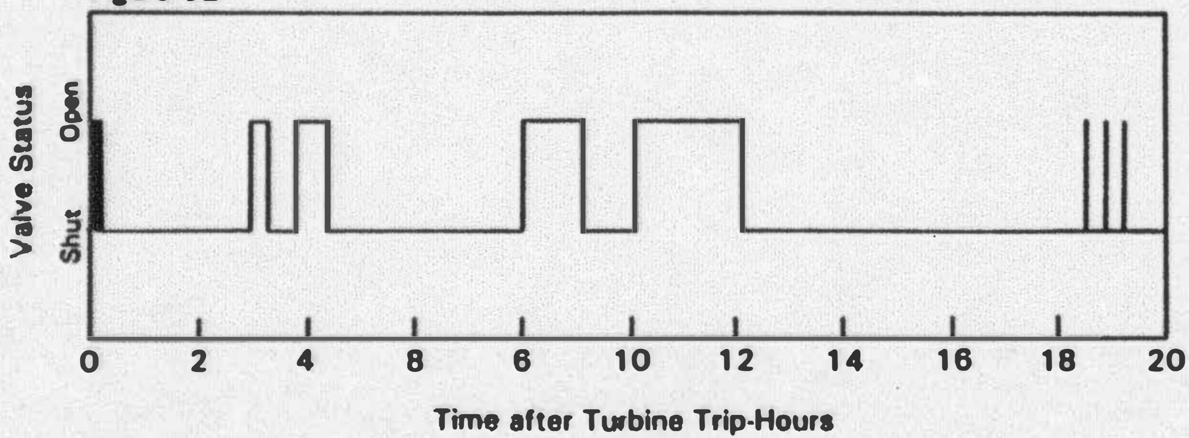


Figure 62



Figures 61 & 62 Pressurizer Spray Valve (RC-V1) Position

TMI-2 Loss of Coolant Accident 3/28/79

Figure 63

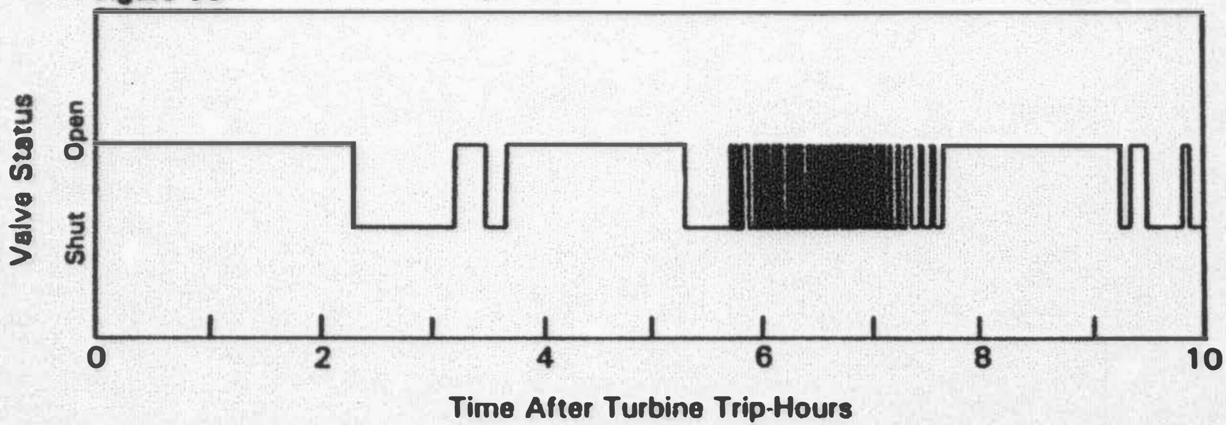
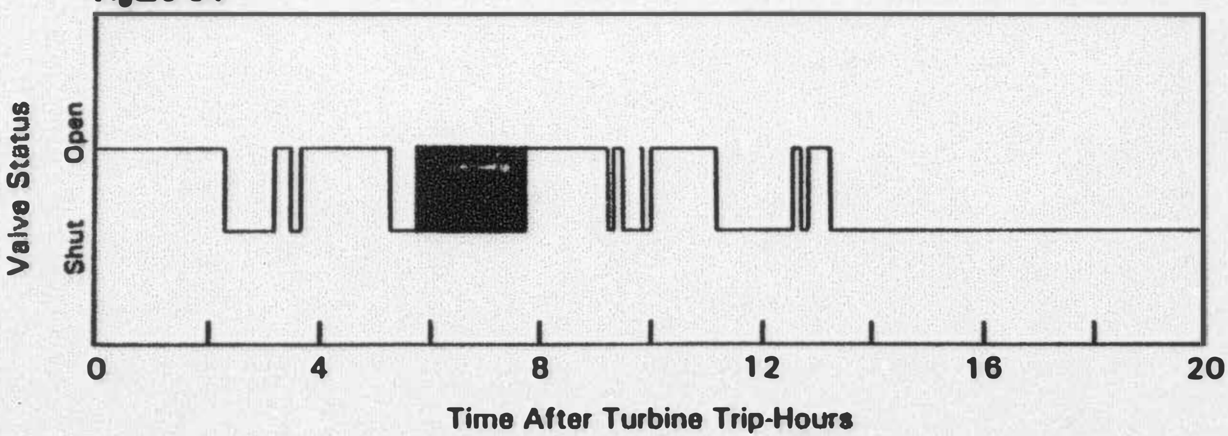
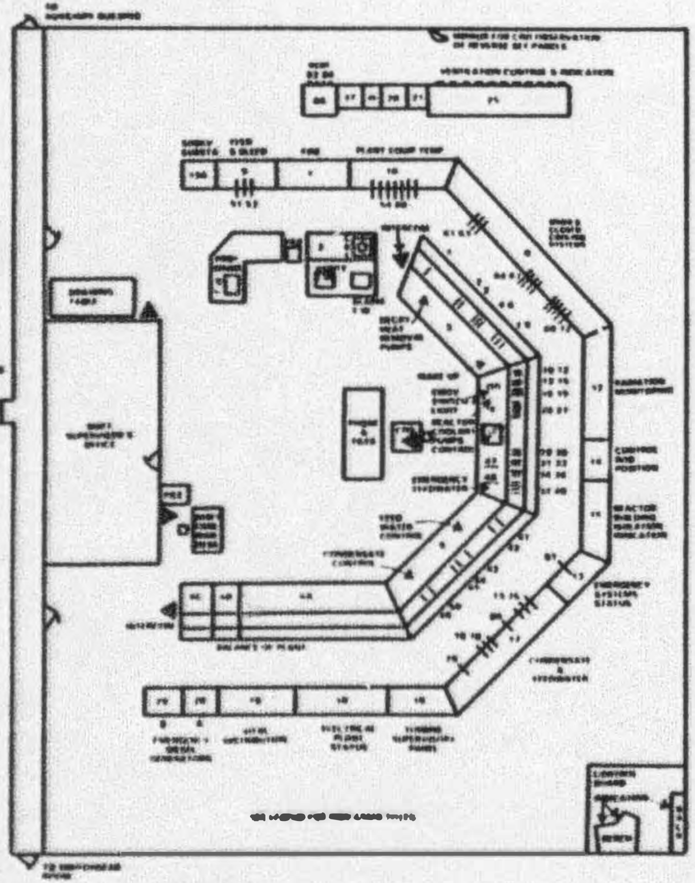


Figure 64



Figures 63 & 64 Electromatic Relief Block Vavle (RC-V2) Position



LEGEND FOR CONTROL ROOM

INDICATION

Panel	No.	Indication
3	1.	Rx Bldg Spray Flow A&B [illegible] gpm
	*2.	Rx Bldg Pressure WR [illegible], NR [illegible]
	3.	RCP Seal Water Inlet Flow 0-15 gpm
	4.	RC Makeup Hdr Pressure 0-4000#
	5.	RC Makeup Flow 0-1000 gpm
	6.	Letdown Flow and Temp 0-1000 gpm, 0-200°F
	*7.	Makeup Tank Level 0-100%-0-100°F
	8.	Makeup Tank Pressure 0-100#
	9.	Makeup Tank Temp. 0-200°F
4	*10.	Pressurizer Temp. 0-700°F
	*11.	Pressurizer Level 0-400"
	12.	Pressurizer Pressure Lo Rng 0-500#
	*13.	Wide Range Pressure 0-2500#
	*14.	Narrow Range Pressure 'B' (1700-2500#)
	*15.	Narrow Range Pressure 'A' (1700-2500#)
	16.	RCS Inlet Temp 'A & B' NR (520-620°F)
	17.	RCS Inlet Temp 'A & B' WR (50-650°F)
	18.	Rx Inlet Temp. NR (520-620°F)
	19.	RCS 'A&B' [illegible]
	20.	RC Outlet Temp A&B NR (520-620°F)
	*21.	RC Outlet Temp A&B WR (520-620°F)
	22.	RC T A&B 0-70°F
	23.	RC Average T 0-70°F
	24.	RC Tave A&B NR (520-620°F)
	*25.	RC Tave NR (520-620°F)
	*26.	RC Flow 0-100%
	27.	RC Flow A&B 0-110%
	*28.	Turbine Throttle Pressure [illegible]
	29.	Steam Generator Outlet Pressure A&B 0-1200#
	30.	Main Steam Temp. A&B (100-650°F)
	31.	Steam Generator Downcomer Temp Lower A&B 0-600°F
	32.	S/G Level Startup A&B 0-250"
	33.	S/G Level Full Scale A&B 0-600"
	*34.	S/G A Lvl Operating Range 0-100%
	*35.	S/G B Lvl Operating Range 0-100% [illegible]
	*36.	Feedwater Flow A 0-6.5x10⁶ lb/hr
	*37.	Feedwater Flow B 0-6.5x10⁶ lb/hr
	38.	Main Feedwater Flow A&B 0-6.5x10⁶ lb/hr
	39.	Startup Feedwater Flow 0-1.5x10⁶
	40.	Feedwater Temp. 0-500°F
5	41.	Uncompensated Pzr Lvl 0-400"
	42.	S/G Startup Lvl 0-250"
	*43.	S/G Operating Lvl 0-100% [illegible]
	44.	Uncompensated Pzr Lvl 0-400"
	45.	Main Steam Temp. [illegible]
6	46.	Emer. Feedwater B Disch. Press. 0-2000#
	47.	Emer. Feedwater A Disch. Press. 0-2000#
	*48.	Computer Alarm and Utility Typewriters
8	49.	Hotwell Level 0-50"
	50.	Gland Steam Supply Press.
8	51.	RC Bleed Holdup Tk A Lvl [illegible]
	52.	RC Bleed Holdup Tk B Temp [illegible]
	53.	RC Bleed Holdup Tk C Press [illegible]
10	*54.	RCS Temps. 24 pt. Recorder [illegible]
	*55.	RC Leakage Recovery Sys. Temps. [illegible] pts
	*56.	RCP Temps 10 pts [illegible]
	*57.	Control Rod Drive Mechanism Temp [illegible] pt
	*58.	Turbine Generator Temps. [illegible]
	*59.	Outside Air Temps. [illegible]
	*60.	Wind Speed and Direction 0-100 mph
8	61.	Intermediate Cld. Cooling Sys. Temp. 0-200
	62.	Intermediate Cld. Cooling Sys. Flow 0-150
	63.	
	64.	Core Flood Tk Lvl A1 & B1 0-14'
	65.	Core Flood Tk Lvl A2 & B2 0-14'
	66.	Core Flood Tk Lvl A1 & B1 0-800#
	67.	Core Flood Tk Lvl A2 & B2 0-800#
	68.	Decay Heat Removal Cooler Temp 0-300°F
	69.	Decay Heat Removal Inlet Temp 0-350°F
	70.	BWST Temp. and Level 0-200°, 0-50'
	71.	RC Makeup Press. 0-1000#
	72.	Emergency Flow 0-300 gpm
17	*73.	Condenser C Inlet Temp. 0-100°
	*74.	Condenser B Outlet Temp. 0-100°
	*75.	Condenser Vacuum 0-30" [illegible]
	76.	Main Steam Press. A&B [illegible] (2 records)
	77.	Service Air Receiver Pressure 0-120#
	78.	Instru. Air Receiver Pressure 0-120#
	*79.	Turbine First Stage Pressure 0-1000#
	80.	Condensate Polisher Bypass [illegible]
12	81.	Emergency Systems [illegible]
8a	82.	RC Drain Tank Lvl [illegible]
	83.	RC Drain Tank Press. 0-250#
	84.	RC Drain Tank Temp. 0-250°

*denotes a recorder

Number to left of bracket denotes panel location

Figure 65 Unit 2 Control Room Locations

DISTRIBUTION LIST

Mr. H.B. Grier
Director Office of Inspection
and Enforcement Region 1 USNRC
631 Park Avenue
King of Prussia, PA 19406

Mr. W.R. Gibson
Babcock & Wilcox
P.O. Box 1260
Lynchburg, VA 24505

Mr. T.F. Hartley, Jr.
March & McLennan, Inc.
1221 Avenue of the Americas
New York, NY 10020

Mr. E.L. Blake, Jr. (4)
Shaw, Pittman, Potts, &
Trowbridge
1800 M St. NW
Washington, D.C. 20036

Mr. R. Sanacore
American Nuclear Insurers
The Exchanger-Suite 245
270 Farmington Avenue
Farmington, CT 06032

Mr. A.S. Dam (Unit 2 only)
Burns & Roe, Inc.
P.O. Box 667
650 Winters Avenue
Paramus, NJ 07652

Mrs. Pat Higgins
Edison Electric Institute
90 Park Avenue
New York, NY 10016

J.T. Collins, Jr
Deputy Program Manager
USNRC
Box 311
Middletown, PA 17057

Mr. Steven M. Gertz PHD. (2)
Portar-Gertz Consultants, Inc.
76 Rittenhouse Place
Ardmore, PA 19003

R.C. McFarland
Neely Nuclear Research Center
Georgia Institute of Technology
900 Atlantic Dr.
Atlanta, GA 30332

Mr. Bill Woods, MGR.
Licensing Information Services
NUC Corporation
2536 Countryside Blvd.
Clearwater, Fla. 33515

Mr. Robert Reimers
EDS Nuclear
220 Montgomery St.
San Francisco, CA 94104

Director of Nuclear Reactor Regulation
ATTN: S.A. Varga
Light Water Reactors Branch No. 4
USNRC
Washington, D.C. 20555

Mr. D. Woodard
Pickard, Lowe & Assoc.
Suite 612
1200 18th St. NW, Wash., D.C. 20036

Mr. M.R. Buring
PP&L A-3-3
2 North 9th Street
Allentown, PA 18101

Nuclear Safety Analysis Center
Elec. Power Research Institute
ATTN: W. Lazallee
P.O. Box 10412
Palo Alto, CA 94303

Mr. George Kulynich
Babcock & Wilcox
P.O. Box 1260
Lynchburg, VA 24505

Harrisburg Public Library
Administration Office
101 Walnut Street
Harrisburg, PA 17101
ATTN: K. Myers

Middletown Public Library
20 N. Catherine Street
Middletown, PA 17057
ATTN: Sue Neiman