LER

Met-Ed / GPU

Metropolitan Edison Company
Post Office Box 480
Middletown, Pennsylvania 17057

Writer's Direct Dial Number

May 11, 1981
LL2-81-0060

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JUN 15 1981
U.S. NUCLEAR REGULATORY COMMISSION

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

Dear Sir:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
LER 79-014/01L-0 Final Report

This letter transmits the final report for the TMI-2 accident of March 28, 1979 as required by Recovery Technical Specification 6.9.1.10. Also enclosed is the Licensee Event Report, Attachment A hereto.

The report is in the form of four attachments to this letter. These attachments correspond to the four categories of information set forth as required in the above Technical Specification, i.e.:

Attachment B A summary of Technical Specification violations related to events which occurred during and after* the transient of March 28, 1979.

Attachment C A summary of the cause(s) of the accident.

Attachment D A sequence of events which occurred during the transient.

Attachment E A summary of corrective actions (both short and long term) which were taken as a result of the March 28, 1979 accident.

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U.S. NUCLEAR REGULATORY COMMISSION

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*It is our understanding that "after", as used here, means until the date of the Order suspending our License to operate TMI-2 (July 20, 1979).

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We believe that this report is in compliance with the requirements of Technical Specifications 6.9.1.10 and, as such, makes it no longer necessary to submit the quarterly reports, which have also been required by the same Technical Specification, as discussed in our letter, LL2-81-0092 of April 8, 1981.

Sincerely,

G K Hovey

G. K. Hovey
Vice-President and
Director, TMI-2

GKH:RIN:vjf

Attachments

cc: L. Barrett, Deputy Program Director
Dr. B. J. Snyder, Program Director - TMI Office

Met-Ed / GPU

Metropolitan Edison Company
Post Office Box 480
Middletown, Pennsylvania 17057

Writer's Direct Dial Number

October 6, 1981
LL2-81-0228

RECEIVED
OCT 19 1981
U.S. NUCLEAR REG... COMMISSION

Office of Inspection and Enforcement
Attn: Mr. Ronald C. Haynes, Director
Region I
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, PA 19406

Dear Sir:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
Final Report of the Quarterly Report Series

In our submittal of the Final Report of the Quarterly Report Series (LER 79-014/01L) transmitted by LL2-81-0060 on May 11, 1981, page 48 of Attachment C, Sequence of Events was inadvertently omitted. Enclosed, please find a copy of that page for insertion into the Final Report. We regret any inconvenience this omission may have caused.

Sincerely,

G. K. Hovey
Vice-President and
Director, TMI-2

GKH:RBS:djb

Enclosure

cc: L. H. Barrett, Deputy Program Director
Dr. B. J. Snyder, Program Director, TMI Program Office

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s
1/1

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

ATTACHMENT A

CONTROL BLOCK: | | | | | | | (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

0	1	P A T M 1 2 (2)	0 0 - 0 0 0 0 0 - 0 0 (3)	4 1 1 1 1 (4)	(5)
7	8	9 LICENSEE CODE 14	15 LICENSE NUMBER 25	26 LICENSE TYPE 30	57 CAT 58

CON'T

0 1	REPORT SOURCE	L (6)	0 5 0 0 0 3 2 0 (7)	0 3 2 8 7 9 (8)	(9)
7 8		60	61 DOCKET NUMBER 68	69 EVENT DATE 74	75 REPORT DATE 80

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 Event description -- On Marcy 28, 1979 during full power operation the Unit exper-

0 3 ienced a loss of Feedwater. For further details see Attachment D to LL2-81-0060

0 4 Consequence of Event -- Gross building and equipment contamination. Off-site health

0 5 effects were minimal.

0 6 Releases - Liquid 0.3Ci I-131

0 7 Gas 14 Ci I-131, 9.7E6 Ci Xe 133, 1.0 E Ci Xe 133 M, 9.6E4 Ci Xe 135.

0 8

	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
0 9	C B (11)	X (12)	Z (13)	V A L V E X (14)	X (15)	B (16)

(17) LER/RO REPORT NUMBER	EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
	7 9	—	0 1 4	/	0 1	[illegible]	—	0

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
X (18)	X (19)	A (20)	C (21)	> 8 E 4	Y (23)	N (24)	N (25)	D 2 4 3 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 Cause Description -- See Attachment C to LL2-81-0060.

1 1 Corrective Actions -- Immediate -- See Attachment D to LL2-81-0060

1 2 Other -- See Attachment E to LL2-81-0060.

1 3

1 4

	FACILITY STATUS	% POWER	OTHER STATUS (30)	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION (32)
1 5	E (28)	0 9 7 (29)	N/A	A (31)	See Attachment D to LL2-81-0060

	ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
1 6	M (33)	M (34)	See Itme 10	R. B. to Atm. IWTS to River

PERSONNEL EXPOSURES

	NUMBER	TYPE	DESCRIPTION (39)
1 7	> E 3 (37)	B (38)	Estimated 5500 MANREM

PERSONNEL INJURIES

	NUMBER	DESCRIPTION (41)
1 8	0 0 0 (40)	N/A

LOSS OF OR DAMAGE TO FACILITY (43)

	TYPE	DESCRIPTION
1 9	B (42)	Core damage, gross plant contamination, water damage in containment

PUBLICITY

	ISSUED	DESCRIPTION (45)	NRC USE ONLY
2 0	Y (44)	Extensive, all media -- continuing	

8107020204

NAME OF PREPARER R. I. Newman PHONE: (717) 948-8461

TECHNICAL SPECIFICATION VIOLATIONS SUMMARY

The NRC performed an intensive review (reported in NUREG-0600 and NUREG-0760) of activities related to the March 28, 1979, accident and identified, in its letters of January 23, 1980, and January 27, 1981, those items it considered to be non-compliances. These letters included some items which the NRC, in its opinion, considered to be Technical Specification violations.

To make the record more complete, we have reviewed available records and interviewed people knowledgeable of events during and after the accident to develop a list of other non-compliances with Technical Specifications. The resulting list may not be complete, but it reflects all such Technical Specification non-compliance events we have been able to identify. It is important that this list be kept in the proper perspective in that many of the non-compliances were a consequence of the accident or resulted from subsequent inaccessability (because of high radiation levels) to equipment and instruments.

TMI-2 LIMITING CONDITIONS FOR OPERATION VIOLATED BUT NOT REPORTED IN NUREG-0600 OR 0760

Reorder Tech Spec. No.	Description/Discussion	Date
3.0.3	Shutdown Requirements - not in cold shutdown within 30 hours.	March 29
3.0.4	Mode changes - Various changes in operational modes occurred without meeting various LCO's.	Various
3.1.2.9 and 3.5.4	Borated Water Sources - Operating - The Borated Water Storage Tank remained below the minimum volume of 445,620 gallons for more than 12 hours.	April 1
3.3.3.8 - Table 3.3.-11	Fire Detection - Instrument - Reactor Building smoke detectors were inoperable.	March 28
3.4.1	Reactor Coolant Loop - Reactor Coolant Pumps in the B Loop were secured. Reactor Coolant Pumps were secured for about 3 hours in both loops.	March 28
3.4.4	Pressurizer - Pressurizer level exceeded 385"	March 28
3.4.5	Steam Generators - less than 18 inches	March 28
3.4.6.1	Reactor Coolant System Leakage - > 10 GPM identified leakage.	March 28
3.4.7	Chemistry - Reactor Coolant System chlorine was greater than 0.15 ppm.	March 28
3.4.8	Specific Activity - Primary coolant sample indicated high iodine activity ($\geq$ 1.0 μCi/gm Dose Equivalent) ($\geq$ 100/$\bar{E}$ μCi/gm)	March 29
3.4.9.1	Pressure/Temperature Limits - Exceeded maximum heatup/cooldown limits during the time frame following the reactor trip. (100° F/hr.)	March 28

Tech. Spec. No.	Description/Discussion	Date
3.4.9.2	Pressurizer - Exceeded maximum heatup/cooldown limits during the time frame following the reactor trip. (100° F/hr)	March 28
3.6.1.4	Internal Pressure - Primary containment pressure exceeded +3 psig.	March 28
3.6.1.5	Air Temperature - Primary containment temperature exceeded 130°F.	March 28
3.6.2.2	Spray Additive System - The NaOH tank remained below 13,275 gallons for greater than 72 hours.	April 1
3.6.5	Reactor Vessel Skirt Area Fans - The reactor vessel skirt area fans were inoperable.	March 28
3.21.4	Activity - Secondary Coolant System - $\geq$ 0.10 μCi/gm Dose Equivalent I-131	March 28
3.7.10.1	Fire Suppression Water System - The Unit 1 River Water Intake Motor Fire Pump was down with a worn impeller and the Unit 1 River Water Intake Diesel Fire Pump blew a head gasket.	April 28
3.7.10.2	Deluge/Sprinkler Systems Fuel Handling Building Exhaust Filter (AH-F-10 A/B) were isolated.	March 28
4.0.3.	Surveillance Time Interval - Exceed surveillance requirement on time interval - See pp. 4-10	Various
4.0.4	Operational Mode Changes - Mode Changes made without performing necessary surveillances.	Various

Description	TS No.
Incore Detector Channel Calibration	4.3.3.2.B
Seismic System Calibration	4.3.3.3.1 4.3.3.3.2
Pressurizer Level and Temperature Calibration	4.3.3.5
Reactor Coolant System Flow - NNI	4.3.3.5
Rx Building Sump Level & Rx Building Cooler Excess Condensate Level Switches Calibration	4.4.6.1.B 4.4.6.1.D
Safety Injection, RCS Pressure Low Channel Calibration	4.3.2.1.1
Reactor Building Isolation & Cooling Channel Calibration	4.3.2.1.1
FW Latching System Channel Calibration	4.3.2.1.1 4.3.2.1.2
H2 Recombiner Channel Calibration	4.6.4.2
4KV Bus 2-1E & 2-2E Undervoltage Relay Calibration	4.3.2.1
Emergency D.G. Load Sequence Relay Calibration	4.8.1.1.2(C.6) 4.8.1.2
Steam Generator Water Level	4.3.3.5 4.3.3.6
4KV Bus 2-3E & 2-4E Undervoltage Relay Calibration	4.3.2.1
Reactor Building Air Pressure	4.3.3.6
MU Storage Tank Level	4.3.3.5
RB Auto Sump Suction Calibration	4.3.2.1.1
FW Line Rupture Automatic Detection Calibration	4.3.2.1.1
CFT Level & Pressure Calibration	4.3.3.6
Low Pressure Injection Flow Channel Calibration	4.3.3.6
High Pressure Injection Flow	4.3.3.6
Secondary Coolant Specific Activity	4.7.1.4
Isotopic Analysis for Dose Equivalent Iodine	4.7.1.4

Description	TS No.
RCS Average Temperature	4.3.3.6
Steam Generator Pressure Calibration	4.3.3.6
RB Spray Pump Flow	4.3.3.6
Source Range Channel Functional Test	4.3.1.1.1
MU Pump Valve Functional Test	4.1.2.3 4.1.2.4 4.0.5.6
DHR Pump Functional Test and Valve Operability Test	4.1.2.5 4.5.2.F.2
Control Rod Movement	4.1.3.1.2 4.1.3.2.2
RPS Functional Tests	4.3.1.1.1
RB Pressure Hi Hi Channel Activation	4.3.2.1.1 Table 4.3-2(3a,b)
Logic Channel Functional Test	N/A
Area & Process Monitor - RMS Channel Functional Test	4.3.3.1
Containment Monitor - RCM Channel Functional Test	4.3.3.1 4.4.6.1.A&C
RB Cooling Unit Operation	4.6.2.3.A
H2 Purge Cleanup System	4.6.4.3.A
E.F. System Valve Lineup Verification and Operability Test; and Turbine Driven E. Feedpump Operability Test	4.7.1.2.A
FH Building Cleanup-Remote Start & Operability Check	4.9.12.A
Safety Injection - RCS Pressure Low Channel Functional Test	4.3.2.1.1 4.3.2.1.2
RB Isolation & Cooling/Safety Injection RB Pressure Hi Channel Functional Test	4.3.2.1.1
Reactor Building Spray Pump Functional Test and Valve Operability Test	4.6.2.1.b

TMI-2 Surveillance Not
Being Performed 4/14/79

ATTACHMENT B
Page 6

Description	TS No.
Boric Acid Pump Functional Test	4.1.2.7 & 4.1.2.6
NSCCW Pump Functional Test & Valve Operability Test	4.0.5
Rx Building Emergency Cooling Booster Pumps Functional Test and C/D Nuclear Services River Water Valve Operability Test	4.6.2.3.b 4.0.5.6
Spent Fuel Cooling Pump Functional Test	4.0.5
Safety Injection Manual Actuation & Actuation Logic Functional Test	4.3.2.1.1, 4.7.4.1([illegible] 4.5.2.e 4.7.3.1.[illegible]
R.B. Isolation & Cooling Manual Initiation and Actuation Logic Functional Test	4.3.2.1.1 4.6.2.2.6 4.6.2.3.[illegible] 4.6.3.1.2.a 4.7.7.1
4 KV ESF Bus Undervoltage Relays Channel Functional Test	4.3.2.1.1
H2 Mixing System Remote Start & Operability Test	4.6.4.4.a
Main Steam Isolation Valves A-Power	4.05, 4.6.3.1.3, 4.7.1.5
Main Steam Isolation Valves A-Power Operation/B Cold Shutdown	4.7.1.5
Hydrogen Recombiner Functional Test	4.6.4.2.A
In-Service Testing of Valves During Normal Plant Operations	4.0.5 4.6.3.1.3
In-Service Testing of HVAC Valves	4.0.5
Valve Operability Test During Cold Shutdown and Remote Indication Functional Test	N/A
Seismic Instrumentation Functional	4.3.3.3.1
RB Hatch Leakrate & Interlock Test	4.6.1.1.B 4.6.1.3.B&C
Control Rod Drop Times	4.1.3.5
RPS Response Times	4.3.1.1.3
ESFAS Response Time Testing	4.3.2.1.3
RCS Total Flow	4.2.5.2

Description	TS. No.
CF Tank Isolation Valve Alarm	4.5.1.D
D.H. Removal Isolation & Interlock Test and Valve Operability and Remote Indication Functional Tests	4.5.2.D.1 4.05 4.5.3
LP Injection System Leakage	4.5.2.D 3a & b
Hydrogen Recombiner Operational Test	4.6.4.2.B.2.3&4
H2 Purge System - Performance Analysis	4.6.4.3.B&d
Class IE Distribution System Source Transfer Test	4.8.1.1.1.B/ 4.8.1.2.1.B
Emergency Feedwater Valve Actuation	4.7.1.2.C.8
Rod Position Indication Functional Test	4.1.3.4
Class IE Distribution System Functional Test	4.8.1.1.2.C 4.8.1.1.2.C.5 4.8.1.1.2.C.2 4.8.1.2 4.3.2.1.1 4.0.5
Station Storage Batteries - Service Test	4.8.2.3.2 4.8.2.4.2
Pressurizer Code Safety Valve Check	4.4.2 4.4.3
Control Room Emergency Ventilation Performance Analysis	4.7.7.1.C 4.7.7.1.e.&3
Building Spray Stem Leakage	4.6.2.1.D
Main Steam Safety Valves	4.7.1.1
FH Building Air Cleanup Performance Analysis	4.9.12.B 4.9.12.d.1&3
Emergency Diesel Generator Loading Test	4.8.1.1.2.C.4 4.8.1.2
Control Room & FHB-RMS Channel Functional Test	4.7.7.1.e.2 4.9.12.d.2
Inaccessible Valve Remote Indication Functional Test	

Description	TS No.
Station Storage Batteries - Performance Discharge Test	4.8.2.3.2.E
RB Spray Nozzle	4.6.2.1.E
NAOH Solution Flow Rate Check	4.6.2.2.D
RCS Chemistry	4.4.7
RCS Specific Activity	4.4.8
Fuel Canal Boron Concentration	4.9.1.2
CF Tank Boron Concentration	4.5.1.B
ECCS Containment Emergency Sump Inspection	4.5.2.D.2
OTSG Eddy Current Testing	4.4.5.1 thru 4.4.5.5
Diesel Generator Inspection	4.8.1.1.2.C.1
RV Internals Vent Valve Inspection	4.4.10.1.b
Hydraulic Snubber Inspection	4.7.8.1
RV Material Surveillance Speciman Inspection	4.4.9.1.2
Reactor Coolant Pump Flywheel Inspection	4.4.10.1.A
Minimum Temperature for Criticality	4.1.1.4.B
Rod Program Surveillance	4.1.3.8.A
Power Level Cut-Off Xenon Equilibrium	4.1.3.9
Containment Integrity	4.5.2.c 4.6.1.1, 4.6.1.3A
RPS Setpoints vs. RCP Combinations	4.4.1
CR Emergency Ventilation System Operability	4.7.7.1.f,g
CR Emergency Ventilation System Charcoal Analysis	4.7.7.1.C.2.d
H2 Purge Charcoal Analysis	4.6.4.3.e,f
H2 Purge Charcoal Analysis	4.6.4.3.c

Description	TS. No
FH Building Air Cleanup System Performance Analysis	4.9.12.e,f
FH Building Air Cleanup System Charcoal Analysis	4.9.12.c
Reactivitity Anomaly	4.1.1.2
Power Distribution	4.2.2.1.2 4.2.3.1,2
Shutdown Margin Determination	4.1.1.11(d)(e)
Moderator Temperature Coefficient	4.1.1.3.2(a)(b)
Structural Integrity Test	4.6.1.6
Containment Integrated Leakage Testing	4.6.1.2 a,b,c,f
Reactor Building Local Leak Rate Testing	4.6.1.2(d)(f)(h)
Intermediate Range Channel	4.3.1.1.3
Manual Reactor Trip Channel Functional Test	4.3.1.1.1 Table 4.3-1, Item 1
Check Valve Operability Test During Cold Shutdown	4.0.5.2
Core Flooding System Check Valve Functional Test	4.0.5.2
Fuel Handling Bridge Hoist Test	4.9.6
Fuel Handling Building Service Crane Interlocks	4.9.7
Fire System Deluge/Sprinkler System Inspection	4.7.10.2B1&2
Fire Detection Circuit Operational Check	4.3.3.8.3
Fire System Detection Instrumentation Functional Test	4.3.3.8.1
Fire Detection Circuit Operational Test	4.3.3.8.2
Spare H2 Recombiner Channel Calibration	N/A
Spare Hydrogen Recombiner Functional Test	N/A
Spare Hydrogen Recombiner Operational Test	N/A
Reclaimed Boric Acid Tank Temperature	4.1.2.8 4.1.2.9

Description	TS No.
Incore Backup Recorder Calibration	3.2.1 thru 3.2.5
Boric Acid Mix Tank Temperature & Level	4.1.2.8.A 2&3 4.1.2.9.A 2&3 4.5.9
Reclaimed Boric Acid Tank Level & Temperature Calibration	4.1.2.8.A 2&3 4.1.2.9.A 2&3
Decay Heat Removal Temperature	4.4.9.2
Containment Air Temperature	4.6.1.5
Spent Fuel Storage/Fuel Storage/Spent Fuel Surge Tank Level Calibration	4.9.11
Reactor Coolant System Leakage	4.4.6.2
Diesel "DF-X-1A" (DF-X-1B) Generator Protective Relaying	N/A
Atmospheric Radiation Monitors Calibration	N/A
Letdown Flow Calibration	N/A
Liquid Radiation Monitor Calibration	N/A
Radiation Monitor Calibration G-M Tube Area Monitors	N/A
RTD Input Functional Calibration (Computer)	N/A
Thermocouple Input Functional Calibration (Computer)	N/A
Differential Pressure Gage Calibration	N/A
Pressure Gage Calibration	N/A
Pressure Transmitter Loop Calibration	N/A

N/A denotes surveillance activities not directly required by Technical Specifications but necessary for actions required to be carried out, i.e. "Implied Technical Specifications."

ATTACHMENT C

A SUMMARY OF THE CAUSE(S) OF THE ACCIDENT

A number of reports have been generated that have included discussions on the cause of the March 28, 1979 accident at TMI-2. These reports include The Report to the Commissioners and the Public by Mitchell Rogovin (NUREG/CR-1250), The President's Commission on the Accident at Three Mile Island (The Kemeny Report), the Office of Inspection and Enforcement Report Number 50-320/79-10 (NUREG 0600), and the findings of the TMI-2 Lessons Learned Task Force (NUREG 0578/0585). Corrective actions based on the findings of the above reports have been taken when appropriate. For the purpose of this report, the licensee feels the causes of the accident are included among the causes and potential causes discussed in the above mentioned reports.

ATTACHMENT D

SEQUENCE OF EVENTS
THREE MILE ISLAND
March 28, 1979

TABLE OF CONTENTS

2.0 LIST OF FIGURES

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 Hull 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 Byram 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 Mehler dated May 17, 1979
   F. John Donnachie dated May 17, 1979

(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.

the borated water storage tank was approximately 55 feet. The Pressurizer Spray Valve (RC-V1) and Heaters, except groups 6 and 7, were in manual 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 184F 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	5.7459 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	593F	594F

* MPPH – Million Pounds Per Hour

† The differences between Steam Generators 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

Time	Event	Information Available to the Operator	Reference
-00:00:05 (0400:32) Approximate	The Condensate Polisher outlet valves went shut simultaneously. The cause of their sudden closure has not been firmly established.	Local indication at the condensate Polishers Panel.	3g,3d,8b, 9B,9F
-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 3 (PL3).	3d
-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 PL3.alarm printer (AP) output of norm/trip and on/off (delay time between alarm printer output and real time approximately 8 seconds).	2a,2b,3d
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 PL13 and PL17, speed and throttle valve position stripchart recorders (SC) at PL17, speed MR at PL4, pump discharge pressure (P_{DISCH}) MR at PL3, AP norm/trip (Delay = 8 seconds)	2a,2b,3d
00:00:00 (0400:37)	The Main Turbine and Main Generator tripped in accordance with plant design.	Turbine: AN at PL3 and PL17, various MR and ST at PL3, AP norm/trip (Delay = 8 seconds) Generator: AN at PL1B, various MR and ST at PL4A, AP norm/trip (Delay = 0 seconds)	2a,2b,4c
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 = 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
00:00:10 (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 had elapsed and the pump tripped.	AR at PL8, ST and MR(A) at PL3, AP norm/trip (Delay = 0 seconds)	2a,2d,3e,8e, 8d,9e,9d
00:00:13 (0400:50)	The Condenser Hotwell low level alarm was received. The level was identified to be 21.73 inches.	MR at PL3, AP low (22.5 inches)/norm/high (30 inches) (Delay = 0 seconds)	2a
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 39).	MR ([illegible]) at PLA, 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 PLA, AP norm/trip (Delay = 15 seconds)	2a
00:00:15 (0400:52)	The Reactor Coolant System Pressurizer level reached a peak value of approximately 256 inches (Figure 31).	SC at PLA, MR (uncompensated) at PL3	1
00:00:15 (0400:52) Approximate	Steam Generator levels were approximately 87 inches (Figure 40). Steam pressure was 1010 psig in Steam Generator B and 1042 psig in Steam Generator A (Figure 31).	SG L: MR (Startup Range) at PLA, MR (Wide Range) at PLA, SC (Operating Range) at PLA and PL3; SG P: MR at PLA, 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 this time (closure setpoint of 2205 psig). The Electromatic	ST at PLA	1,6a

Time	Event	Information Available to the Operator	Reference
	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 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.		
00:00:18 (0400:55)	The Pressurizer Spray Valve (RC-V1) shut.	ST at PL4	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 PL3 OO F: MR at PL4, RC at PL17	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"	54,84,93,97

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: AM (24 inches) at PL17, MR (Startup Range) at PL4 AP low (24 inches)/normal (Delay = 30 seconds) EF-V11A and EF-V11B: MR at PL4 EF-V12A and EF-V12B: ST at PL4	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 delivering approximately 400 gallons per minute the Pressurizer level rate of decrease slowed (Figure 31).	MU-P-1A: AM 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,3c,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).	SG at PL4, MR (uncompensated) at PL3	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 PL4, MR (uncompensated) at PL3 RC T_C: MP at PL10 RC T_H: SC at PL4, MP at PL10 and MR at PL4 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 = 50 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
00:01:26 (0402:03)	A Reactor Coolant Drain Tank temperature normal alarm was received and printed out a temperature of 85.5F. 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
00:01:45 (0402:22) Approximate	Steam Generators A and B had 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 PL5 RC T: SC at PL4, MP at PL10 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 1B (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,7,8c,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 1B (MU-P-1B) tripped automatically as a result of the actuation of Safety Injection. The Engineered Safety Features design is such that Makeup Pumps 1A and 1C are normally used for High Pressure Injection. The normal minimum High Pressure Injection flow rate is 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 actuation. The	ESFSI: 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)	2a,2b,5c,6a,6c

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).	MD at PLA	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 Reactor Coolant 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
00:05:30 (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 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 existence 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: IR and SC at PLA RC T: SC at PLA, MP at PL10 and MR at PLA	1,2a,3a,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 PLA, 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)	2a,2d,0Z
00:06:56 (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 = 4 minutes)	2a
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 71.4 gallons per minute was recorded.	MR at PL3 AP Range 0 to 160 gpm (Delay = 6 minutes)	2a,8f,8r,9d

Time	Event	Information Available to the Operator	Reference
00:08:33 (0409:10)	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 5).	RC T: RC at PL4, MP at PL10 and MR at PL4 SG P: MR at PL4, RC (P_{SG}) at PL17 AP low (860 psig)/norm/high (960 psig) (Delay = 6 minutes)	1,5a
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 RC at PL4	1,2a,3a,5a
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).	AM 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 Condensate Booster Pump 2A Suction Valve (CO-V27A).	Unit 2 Control Room notified of leaking flange	5d,8k,8h,9b, 9I,9P
00:10:18 (0410:55)	The Reactor Coolant System Pressurizer level indication came on scale.	RC at PL4, MR (uncompensated) at PL5	1

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 = 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 = 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 = 8 minutes)	2a
00:12:00 (0412:37) Approximate	Condenser 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 = 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 = 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 resulted in a rapid increase in Reactor Containment Building Pressure (Figure 51).	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 PL3	5d,9a
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 = 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 = 13 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-U26) had burst. This conclusion was based on the existing high temperature and low pressure in the Reactor Coolant Drain Tank coupled with the low discharge pressure of the operating Leakage Transfer Pump(s) 9A and/or 9B (WDC-P-9A and/or 9B).	RCDT P: MR and AN (125 psig) at PL8A RCDT T: MR at PL8A	8c,9e,9K
00:24:58 (0425:35)	The Unit 1 Shift Supervisor requested the computer to 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). Respective values of 285.4F, 283.9F and 275.1F were indicated. The operator attributed the temperature levels to the normal cooldown 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.	MP (Delay = 8 minutes) AP high (200F)/norm (Delay = 21 minutes) MP at PL10	2c,8e,8c,9A
00:25:00 (0425:37) Approximate	The operator placed the Turbine Bypass Valves (MS-V25A, MS-V25B, MS-V26A and MS-V26B) under manual control and opened them slightly to promote additional cooling of the primary coolant.	MS-V25A/26A: MR and ST at PL8 MS-V25A/26B: MR and ST at PL5	8e
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	MR and MP on PL12	5b,8c,9e,9K

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,5b
00:32:36 (0433:13)	Incore Thermocouple R-10 signal indication went out-of-range (0F to 700F).	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_{DISCH}) 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,5a,8k,9b

Time	Event	Information Available to the Operator	Reference
00:59:21 (0459:58)	The Condensate high temperature alarm was received. A temperature of 110.5F was recorded.	AP norm/high (Delay = 40 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 control 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 the use of MSV-3B was terminated 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 condenser hotwell level control.	MR(A and F) and ST at PL17 AP on/off (Delay = 49 minutes) MS-V25A/26A: MR and ST at PL5 MS-V25B/26B: MR and ST at PL5	2a,8c,9a,9A
01:09:23 (0510:06) Approximate	A Radiation/Chemistry Technician drew a Reactor Coolant System sample for boron analysis after the reactor trip. The boron analysis results indicated a boron concentration of slightly over 100 parts per million.	Unit 2 Control Room notified of boron analysis	8c,9B
01:10:54 (0511:31)	The Reactor Building Air Cooling Coil E 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 have been a periodic malfunction in the computer input signal.	MR(F) at PL15 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.	MF (Delay = 0 minutes)	2c

Time	Event	Information Available to the Operator	Reference
	Feedwater Valve (FW-011B). Steam Generator Pressure Control was being accomplished using power operated Emergency Main Steam Dump Valves MSV-3A and MSV-3B.		
01:13:29 (0514:04)	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 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 ≈ 0 minutes) RCP V: AN at PL8, AN and MR at PL10 RC F: MR and SC at PLA	3b,4g,8e,8c, 8d,8r,9b,9k
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 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 ≈ 0 minutes) RCP V: AN at PL8, AN and MR at PL10 RC F: MR and SC at PLA	3b,4g,8e,8c, 8d,8r,9b,9k
01:14:13 (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 Pump 1B and 2B (RC-P-1B and RC-P-2B). Concurrent with this the water level in Steam Generator B started to rise. The rapid rise in Steam Generator B level was a result of a lower steaming rate in Steam Generator B and an unequal division of feed water flow between Steam Generators A and B. The unequal flow was caused by an imbalance in steam pressure between the steam generators with Steam Generator B having the lower steam pressure (Figure 10).	SG P: MR at PLA, SC at PL17 SG L: MR (Startup Range) at PLA	1,3e,8c

Time	Event	Information Available to the Operator	Reference
01:26:23 (0527:00)	The operator shut Steam Generator B Main Steam Isolation Valves (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-V23B/26B MR and ST at PL3	4b,8c,9c,9k, 9M
01:29:23 (0530:00) Approximate	The Unit 1 Shift Supervisor directed an auxiliary operator to energize Core Flood Tanks 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 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 02:38:57 (1139:34).	CF-V1A/1B Breaker Status: ST at PL8	8d,8k,9b,9d, 9M,9l,9T,9X
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 x 10^{-11} amperes to approximately 1.6 x 10^{-11} amperes (Figure 56). Correspondingly, the out-of-core Source Range Channel (NI-1) indication increased from about 1.6 x 10^{4} to approximately 2.0 x 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,5e

*See entry at time 02:47:31 (0448:08)

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 01:13:22 (0333:29) to 02:47:31 (0648:04). However, based on the sequence of events printout Makeup Pump 1C was stopped prior to 02:28:41 (0629:28).	MU-P-1C: AM at PLB, ST and MM(A) at PL3 AP mom/trip (Delay * minutes)	
01:41:00 (0341:37) 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 level 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 moderator density caused by liquid displacing steam in the reactor vessel.	NI-1: MR and SC at PLA NI-4: MR and SC at PLA	3k
01:41:23 (0342:00)	A Radiation/Chemistry Technician took a condenser vacuum pump exhaust sample for radiochemistry analysis per procedural requirements after a reactor trip. The results of the analysis indicated radioactivity levels were not above background. The results of the analysis are listed below. Potassium 40 — 4.860 E-06 mCi/ml Cobalt 58 — 1.555 E-06 mCi/ml Xenon 133 — 1.836 E-07 mCi/ml Total: — 6.585 E-06 mCi/ml	Unit 2 Control Room notified of Condenser Vacuum pump results.	4e,9F

*See entry at time 02:47:31 (0648:04)

Time	Event	Information Available to the Operator	Reference
01:22:23 (0543:00) 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 other chemist on shift performed an independent boron analysis of another sample and obtained similar results. The two values recorded were 402 and 407 parts per million boron.	Unit 2 Control Room notified of boron analysis results.	9c,9b
01:31:27 (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).	RC at PL4, NP at PL10 and NR at PL4	1,3a
01:34:00 (0554:37) Approximate	The reactor out-of-core Intermediate Range Channel (NI-4) indication increased from less than 1.0 x 10^{-11} amperes to approximately 1.0 x 10^{-10} amperes (Figure 36). A corresponding increasing trend was recorded on the reactor out-of-core Source Range Channel (NI-1) indication (Figure 36). 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 by (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: NR and SC at PL4 NI-3: NR and SC at PL4 NI-4: NR and SC at PL4	3b,3a

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 PLA, MP at PL10 and MR at PLA	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 PLA 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 reactor out of core Source and Intermediate Range channels in conjunction with the results of two boron analysis which indicated the boron concentration in the Reactor Coolant System to be 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,9m,9j,9A, 9C,9I,9P,9X
02:10:42 (0611:19)	Reactor Coolant System Loop A hot leg temperature indication increased offscale, greater than 620F (Figure 23).	AM at PL8 (high at 612F), SC at PLA, MP at PL10 and MR at PLA	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).	AM, MR and MP at PL12	3e
02:17:53 (0618:30)	The Unit 2 Relieving Shift Supervisor requested the computer print	MP (Delay " 0 minutes)	2c, 8m,9E

Time	Event	Information Available to the Operator	Reference
	radiation monitors began to increase. (a) Reactor Building Purge Air Exhaust Duct A (HP-R-225)-particulate (b) Reactor Building Purge Air Exhaust Duct B (HP-R-226)-particulate (c) Auxiliary Building Purge Air Exhaust (HP-R-222)-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).		
02:43:00 (0643:37) Approximate	Several radiation alarms were received at the Control Room Radiation Monitor Panel.	AB, MR and MP at PL12	8q,9a,9h
02:45:23 (0646:00)	A Radiation/Chemistry Technician took a reactor coolant sample. A gamma spectrum analysis was performed at 02:51:23 (0652:00) and indicated a 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.	4e

Krypton 85	1.437 E+01	mCi/ml
Rubidium 88	7.847 E+00	mCi/ml
Neodymium 95	6.277 E-02	mCi/ml
Neodymium 98	1.498 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 135	2.260 E+00	mCi/ml
Cesium 136	2.836 E-01	mCi/ml
Cesium 137	2.946 E-01	mCi/ml
Cesium 138	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) indicated radiation level began to increase.	MR and MP at PL12	9b,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(P) and SC(P) at PL4, AN at PL8, AP norm/trip (Delay = 0) *See entry at time 02:47:31 (0648:08)	8c,8r,9e

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 PL4, 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 PL4, AN at PL8, AP norm/trip (Delay ≈ 5 minutes)	1,2a,3m,8c, 9a,9j
02:54:15 (0654:52)	Steam Generator B steam pressure rapidly increases from approximately 140 psig to approximately 720 psig in the next two minutes.	MR at PL4, 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 PL4 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,8e
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 2O - 623.8F 12O - 624.9F 13G - 670.6F* 14H - 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
	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 in 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 and reduce Reactor Coolant System pressure.	PZR Spray: ST at PLA RC P: MM and SC at PLA	1,8g,84,8r
02:55:13 (0655:50)	The operator removed the bypass signal from the Safety Injection portion of Engineered Safety Features trains A and B.	AM 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.	MM 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 1E (CW-P-1E).	MM (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: MM at PLA 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 Physics Laboratory. Onsite and offsite radiation monitoring teams were formed to measure radiation levels.	Unit 2 Control Room notified of the Emergency Control Station status.	9g,9P

Time	Event	Information Available to the Operator	Reference
03:05:46 (0705:49)	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 ≈ 13 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.	MP 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
03:19:45 (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,9a, 9A
03:20:13 (0720:50)	Reactor Coolant Makeup Pump 1C (MU-P-1C) started automatically on the Engineered Safety Feature train A actuation. Reactor Coolant Makeup Pumps 1A and 1C (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 ≈ 19 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)	AM, MR and MP at PL12	3d,5b

Time	Event	Information Available to the Operator	Reference
	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 1×10^5 counts per minute. The gas channel of the Reactor Building Duct A Radiation Monitor indicated 5×10^5 counts per minute.		
03:31:23 (0722:00) 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.	BC and ST at PL 25	3f,3h,6c
03:22:00 (0722:37) Approximate	The operator stopped the boric acid addition to the Makeup Tauk (MU-T-1). This addition was associated with the emergency boration initiated at 02:10:00 (0610:37).	MU-V10:ST at PL3 CA-P-4A/B:ST at PL3 Boration: Batch Controller at PL3	3t,8k,8r,9d 9h,9h,9j,9A 9C,9I,9F,9X (1)

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

(1) Informal discussions with the operators to clarify their interviews.

Time	Event	Information Available to the Operator	Reference
	Generator B. He was unable to establish flow through the Steam Generator A sample line. A gamma spectrum analysis of the Steam Generator B sample was performed at 03:23:23 (0724:00) and indicated radioactivity levels were not above background. The results of the analysis are listed below. Potassium 40 — 4.840 E-06 mCi/ml Cobalt 58 — 1.555 E-06 mCi/ml Xenon 133 — 1.836 E-07 mCi/ml Total: — 6.585 E-06 mCi/ml The Control Room personnel questioned these results since they felt that Steam Generator B was contaminated. It was later determined that the steam generator sample steam lines leading 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 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 gamma spectrum analysis of these samples was not performed.		
03:25:56 (0726:33)	The Reactor Coolant System Pressurizer high level alarm cleared. A level of 238 inches was recorded.	AM 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.	AM at PL13, ST 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 (RP-R-232) increased.	AM, MB and MP at PL 12	3b

Time	Event	Information Available to the Operator	Reference
03:37:00 (0737:37)	The operator stopped Reactor Coolant Makeup Pump 1C (MU-P-1C) because pressurizer level was rapidly increasing. Indicated Pressurizer level was 373 inches (Figure 4).	MU-P-1C: ST and MR(4) at PL3, AM at PL8 AP norm/trip (Delay = 42 minutes)	1,2a,3a
03:39:23 (0740:00)	The operator isolated the Reactor Building Air Sample Line after it was reported to be blowing air into the Auxiliary Building.	ST at PL23	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,3a
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 205.2F 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, 6E, 6F*, 5O, 9G, 7L, 7B, 14M, 9O, 7K, 10H* and 8F). The increase temperatures were in response to stopping Reactor Coolant Makeup Pump 1C (MU-P-1C). Note: (*) This Incore Thermocouple temperature cycled near 700F until Safety Injection actuated on high Reactor Building pressure at 03:56:04 (0756:41).	AP norm/bad (out of range 0F-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 these fans stopped is unknown.	Exhaust Fans: SC and ST at PL25 Radiation Levels: MR and MP (Unit 1 Control Room)	3b,3a
03:44:23 (0745:20) 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 discussed with Operation.)	9a,9b

Time	Event	Information Available to the Operator	Reference
03:55:39 (0756:16)	Intermediate Cooling Pump 1B (IC-P-1B) was tripped automatically by the Engineered Safety Features train B actuation.	AM, ST, MR (P_{DISCH}) and MR(F) at PL8 AP on/off (Delay = 46 minutes)	2a,6a
03:55:39 (0756:16)	The Reactor Building isolated automatically as part of the Engineered Safety Features train B Actuation.	AM at PL13, ST at PL3 and PL15, AP isolation/norm (Delay = 46 minutes)	2a
03:55:39 (0756:16)	The Control Room Ventilation System should have aligned to internal recirculation upon actuation of Engineered Safety Features train B. In the recirculation mode the Control Room air exhaust flow is diverted to the supply duct and the supply flow is reduced to maintain a positive Control Room pressure. The exhaust flow recorder was out of service on March 28, 1979 and this flow diversion cannot be verified. However a reduction in the control room supply flow was experienced.	SC and ST at PL25	6c, 3v
03:55:46 (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 51).	AM at PL13, ST at PL3 and PL15 AP act/trip (Delay = 46 minutes)	2a,6a
03:55:46 (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
03:56:04 (0756:41)	Reactor Coolant Makeup Pump 1C (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
03:59:23 (0800:00) Approximate	Waste Gas Tank Discharge A (WDG-R-1485) monitor indication increased from 5×10^2 counts per minute to 3×10^3 counts per minute at approximately 05:00:00 (0900:37). Note: This monitor is located in the Auxiliary Building at a elevation of 305 feet.	MR and MP at PL12	3a,5b

Time	Event	Information Available to the Operator	Reference
04:07:01 (0807:38)	The operator removed the defeat signal from the Reactor Building Isolation and Cooling portion of Engineered Safety Features train A. Train B remained defeated.	AM at PL13, ST at PL3 and PL13 AP norm/defeated (Delay ≈ 52 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, MM(A), MM(F) and SC(F) at PL4, AM at PL8 AP norm/trip (Delay ≈ 51 minutes)	2a,9e
04:09:14 (0809:51)	The operator stopped Reactor Coolant Pump 1A (RC-P-1A) after observing zero flow indication and a running current of less than 100 amperes.	ST, MM(A), MM(F) and SC(F) at PL4, AM at PL8 AP norm/trip (Delay ≈ 51 minutes)	2a,9e
04:10:10 (0810:47)	The operator stopped Intermediate Cooling Pump 1B (IC-P-1B).	AM, ST, MM(P_{DISCH}) and MM(F) at PL8 AP on/off (Delay ≈ 53 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 filtered continuously.	SC and ST at PL25	3g,3h,8m,9J

Time	Event	Information Available to the Operator	Reference
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 ≥ * minutes)	3r
02:31:23 (0632:00) Approximate	The Incore Instrument Panel Area Monitor (HP-R-213) reading began to increase.	AN, 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 PL5	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)	AN, MR and MP at PL12	3a,3d,5b
02:39:23 (0640:00) Approximate	The Unit 1 hot machine shop 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 represenative 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 Panel Area Monitor (HP-R-213) indication increased offscale high. The levels indicated on the following	AN, 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
04:23:54 (0824:31)	The operator energized Pressurizer Heater Groups 1 through 5 by putting the heater controls in the automatic mode. Eleven pressurizer heater groups were available at this time.	AN 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, AN 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 was de-energized for the remainder of March 28, 1979. Ten pressurizer heater groups were available at this time.	AN 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: AN and SC at PL17	2a,8a,9x, 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 PL5	8f,8a,9x,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. 11L = 480.9F 9C = ***.*F 11K = ***.*F 8B = 511.7F 12K = ***.*F 7B = 436.7F 13H = ***.*F 6C = 641.9F 13G = 533.0F 5D = ***.*F	OP (Delay = 0 minutes) AP norm/bad (out of range 0F to 700F)(Delay = 60 minutes)	2c,9u,9v, 9w,9x

Time	Event	Information Available to the Operator	References
04:49:23 (0850:00) Approximate	Condenser Vacuum Pump Exhaust radiation monitor (VA-R-748) decreased to 1 x 10^4 counts per minute.	HR and BC 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,9a,9a,9B, 9B,9B,9F
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 from the surface of the 50-100 milliliter sample. A boron analysis was performed and a value of 248 parts per million boron was obtained. A gamma spectrum analysis was completed at 07:38:23 (1139:00) and indicated a total beta-gamma activity of 1554 uCi/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 uCi/ml Iodine 133 — 1.430 E+02 uCi/ml Xenon 133 — 4.493 E+02 uCi/ml Xenon 135 — 2.466 E+02 uCi/ml Rubidium 88 — 6.037 E+02 uCi/ml Krypton 85 — 2.886 E+02 uCi/ml Total: — 1.554 E+03 uCi/ml	Unit 2 Control Room notified of boron and gross beta-gamma analysis results	4e,4g,4h,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.	HR and BC 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
	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 plant equipment within the Reactor Building. The Reactor Building pressure continued to remain 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).	AB (Low-2055 and Low/Low-1600) at PL8 MR and BC at PL4 AP (many clearing alarms) (Delay ≈ 60 minutes)	3a
05:23:34 (0924:11)	The operator removed the defeat signal from the Reactor Building Isolation and Cooling portion of Engineered Safety Features train A.	AB at PL13, ST at PL3 and PL13 AP norm/defeated (Delay ≈ 60 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 3.50 psig (Figure 51).	AB at PL13, ST at PL3 and PL15 AP act/trip (Delay ≈ 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.	AB, ST, MR (F_{ICgpm}) and MR(F) at PL8 AP on/off (Delay ≈ 60 minutes)	2a
05:23:47 (0924:24)	The operator defeated the Reactor Building Isolation and Cooling portion of Engineered Safety Features train A.	AB at PL13, ST at PL3 and PL13 AP norm/defeated (Delay ≈ 60 minutes)	2a

Time	Event	Information Available to the Operator	Reference
05:36:44 (0937:11)	The Safety Injection logic of the Engineer Safety Features trains A and B reset automatically on increasing Reactor Coolant System Pressure. The setpoint is 1665 psig.	AN at PL13, ST at PL3 and PL13 AP trip/norm (Delay = 95 minutes)	2a
05:39:27 (0940:04)	The Pressurizer Safety Valve (RC-R1B) discharge line high temperature alarm reset. A value of 192.6F was recorded.	MP at PL10 AP high (200F)/norm (Delay = 97 minutes)	2a
05:41:06 (0941:43)	The operator removed the bypass signal from 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 = 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,9s,9u,10
05:43:27 (0944:04)	The Electromatic Relief Valve (RC-R2) discharge line high temperature alarm was received. A value of 214.7F was recorded.	MP at PL10 AP high (200F)/norm (Delay = 100 minutes)	2a
05:44:01 (0944:38)	Pressurizer Safety Valve (RC-R1B) discharge line high temperature alarm was received. A value of 205.4F was recorded.	MP at PL10 AP high (200F)/norm (Delay = 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 = 103 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 radioactivity 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,8h,8t, 9b,9f,9w
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 5 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 1 inspectors entered Unit 2 Control Room to evaluate the operational and radiological status of Unit 2.		4h,9m,9n
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, MR(A) and MR (P_{DISCH}) at PL4 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. 8H = ***.*F 6C = ***.*F 9H = ***.*F 5G = ***.*F	UP AP norm/bad (out of range 0F to 700F) (Delay = 11′ minutes)	2c,9f

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 on Reactor Coolant System pressure decreased to less than 1640 psig. However, initiation did not occur because of the manual bypass previously introduced.	AN at PL13, ST at PL3 and PL13 AP norm/actuation (Delay~ 144 minutes)	2a
07:43:44 (1144:21)	Pressurizer Heater Groups 1 and 2 tripped and re-energized after 2 seconds when the pressurizer heater control selector switch was placed in the manual position.	ST at PL4, AP norm/trip (Delay = 145 minutes)	2a
07:50:16 (1150:53)	The operator de-energized Pressurizer Heater Groups 1 and 2 to assist in lowering Reactor Coolant System pressure. Seven pressurizer heater groups were available at this time.	ST at PL4 AP norm/trip (Delay = 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: MR and SC at PL4	1,8c,8e
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 = 0 minutes) AP high (200F)/norm (Delay = 148 minutes) MP at PL10	2c
08:11:26 (1212:03)	Core Flood Tank 1A (CF-T-1A) high level alarms was received. The level was 13.32 feet.	AN and MR at PL8 AP norm/high (13.3 feet) (Delay = 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.	MP at PL10 AP high (200F)/norm (Delay = 154 minutes)	2a
08:22:58 (1223:35)	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 = 156 minutes)	2a

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 13.83 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, 3m
09:04:18 (1304:55)	The operator stopped Reactor Coolant Makeup Pump 1C (MU-P-1C) and returned the Reactor Coolant Makeup System to one pump operation.	AN at PL8, BE and MR(A) at PL3	2a,3c
09:04:23 (1303:00) Approximate	Personnel in Unit 1 Control Room were required to wear respirators due to increased airborne radioactivity levels. Personnel not essential to control room operations were moved to the Observation Center.	AN, MR and MP (Unit 1 Control Room) Unit 1 Control Room air sample data	4g,4h
09:07:24 (1308:01)	The operator secured Pressurizer Spray flow.	PZR SPRAY: ST at PLA RC P: MR and BC at PLA	1,8c,8e
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 shut the Electromatic Relief Block Valve (RC-V2).	ST at PLA	3j
09:30:18 (1330:55)	The operator started the plant computer on a two minute group	MP (Delay = 0 minutes)	2c

Time	Event	Information Available to the Operator	Reference
	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 Tank Rupture Diaphragm (WDL-U26) which had been breached. The hydrogen concentration in the containment eventually reached an explosive mixture. The 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 Centers 32A and 42A were lost. These motor control centers supply power to the Seal Water Pumps which furnishes seal water to many of the Radwaste System Pumps in the Auxiliary Building.	Local indication at the Radwaste panel	5b,8m,9J
09:49:44 (1350:31)	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 3.38 psig and 28 psig respectively. This was a result of the 28 psig Reactor Building pressure impulse from the hydrogen detonation. Reactor Building isolation, cooling and containment spray were actuated.	AN at PL13, ST at PL3 and PL13 AP norm/act (Delay = 159 minutes)	2a,2b,3J,6a
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(F_{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 1C (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

Time	Event	Information Available to the Operator	Reference
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
09:50:09 (1350:46)	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 = 162 minutes)	2a
09:50:09 (1350:46)	The operator started Intermediate Cooling Pump 1A (IC-P-1A).	AN, ST, MR (F_{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.	AN 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).	AN, ST, MR (F_{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 1C (MU-P-1C).	AN 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 178.6F were 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

Time	Event	Information Available to the Operator	Reference
09:59:26 (1400:03)	The following Incore Thermocouple temperatures decreased to less than 700F over the next minute. This was a result of Core Flood Tank (CF-T-1A) discharging into the reactor vessel. 6L - 687.2F 13L - 666.7F 10R - 461.9F 0M - 681.6F	AP norm/bad (out of range 0F to 700F) (Delay = 160 minutes)	2a
10:00:58 (1401:35) Approximate	The operator opened the Electromatic Relief Block Valve (RC-V2).	ST at PLA	3j
10:00:58 (1401:35)	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 208.0F were recorded.	MP at PL10 AP high (200F)/norm (Delay = 159 minutes)	2a
10:04:30 (1405:07)	The operator initiated pressurizer spray flow to assist in lower Reactor Coolant System pressure. Pressurizer spray flow was maintained until 12:05:54 (1606:31).	PZR SPRAY: ST at PLA RC P: MR and SC at PLA	1,8e
10:05:25 (1406:02)	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 PLA and PL10, MR at PLA, AM at PL8	1,9j,9D,9J, 9M
10:29:13 (1430:00) Approximate	At the request of the Pennsylvania State Government, the Metropolitan Edison Company Vice President of Generation traveled to	Annoucement made to personnel in Unit 2 Control Room	9k,9u,9z,10

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 PL6 AP norm/trip (Delay ≈ 103 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 computer sampling rate 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 11:01:36 (1502:13). 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 390 inches to 174 inches over a period of 13 minutes (Figure 33).	SC AT PL 4, AN (High/High = 315 inches, High = 300 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.	AN, 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 263F to 412F 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 1C (MU-P-1C) to stop the rapid fall in the Pressurizer level.	AN 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
11:35:48 (1536:25)	The operator stopped Reactor Coolant Makeup Pump 1C (MU-P-1C).	AP at PL8, MR(A) and ST at PL3 AP norm/trip (Delay = 83 minutes)	2a,3c
11:29:36 (1530:13)	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 11:52:06 (1552:43).	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 Building Exhaust Ventilation flow fluctuated during the next three hours. The reason for the flow fluctuation is unknown.	SC and ST at PL25	3h
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. 8H = ***.*F 5G = ***.*F 9H = 596.9F 5H = ***.*F 9G = ***.*F 5K = ***.*F 8F = ***.*F 6L = 562.1F 9E = ***.*F 7M = ***.*F 7F = ***.*F 8N = ***.*F 7E = ***.*F 9M = ***.*F 6G = ***.*F Note (***.*) indicates the signal was outside of the computer range (0F to 700F)	MP (Delay = 0 minutes) AP norm/bad (out of range 0F to 700F)(Delay = 30 minutes)	2c,9f

Time	Event	Information Available to the Operator	Reference
13:05:29 (1706:06)	Pressurizer Safety Valve (RC-R1B) discharge line high temperature alarm reset. A temperature of 192.8F was recorded.	MP at PL10 AP high (200F)/norm (Delay = 0 minutes)	2a
13:13:10 (1713:47)	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 steam in Loop A hot leg had been condensed and natural circulation flow had been 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 the Electromatic Relief Valve (RC-R2) to the Reactor Building resulted in an increase in hydrogen concentration. The subsequent hydrogen denotation caused a 28 psig pressure pulse in the Reactor Building. Attempts to use the Core Flood System to cool the core during the last six hours had 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,3m,4g, 9j,9e,9u,10

Time	Event	Information Available to the Operator	Reference
14:43:15 (1843:52)	The operator stopped Reactor Coolant Makeup Pump 1C (MU-P-1C) to slow the rapid increase in 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 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,9M
14:54:35 (1855:12) Approximate	Reactor Coolant System pressure reached 2332 psig (Figure 14).	MR and SC at PL4, AM (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 indication came 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-318) (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-234) indication was steady at about 10 counts per minutes.	AM, 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
15:33:07 (1933:44)	The operator manually bypassed the Safety Injection portion of Engineered Safety Features trains A and B.	AM 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.	AM 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).	AM at PL8, MM(A) and ST at PL3 AP norm/trip (Delay ≈ 0 minutes)	2a,5c
15:39:27 (1940:04)	The operator removed the Bypass signal from the Safety Injection portion of Engineered Safety Features trains A and B.	AM 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).	AM at PL8, MM(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 1380 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, MM(A), MM(F) and BC(A) at PL4 AM (trip) at PL8 AP norm/trip at PL8 (Delay ≈ 0 minutes) RC T_C: MP at PL10 RC P: MM and BC at PL4	1,2a,3e, 4b,5j,5o,5u, 5y,5M
15:50:08 (1950:45)	Reactor Coolant System Loop A hot leg temperature decreased from offscale high (greater than 620F) to offscale low (less than 520F) (Figure 29).	AM(high = 612F) at PL8, SC 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	AM at PL13, ST at PL3 and PL13 AP norm/bypassed (Delay ≈ 0 minutes)	2a

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.	MM(F) 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 Reactor Coolant System 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 to 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 this was due to plugging of either the letdown coolers, orifices or purification filters.	MM(V) at PL3	4b,4g

PLANT STATUS

Time	Event	Information Available to the Operator	Reference
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 1163 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		

Figure 1
TMI-2 Loss of Coolant Accident 3/28/79
Summary of Reactor Coolant System Parameters

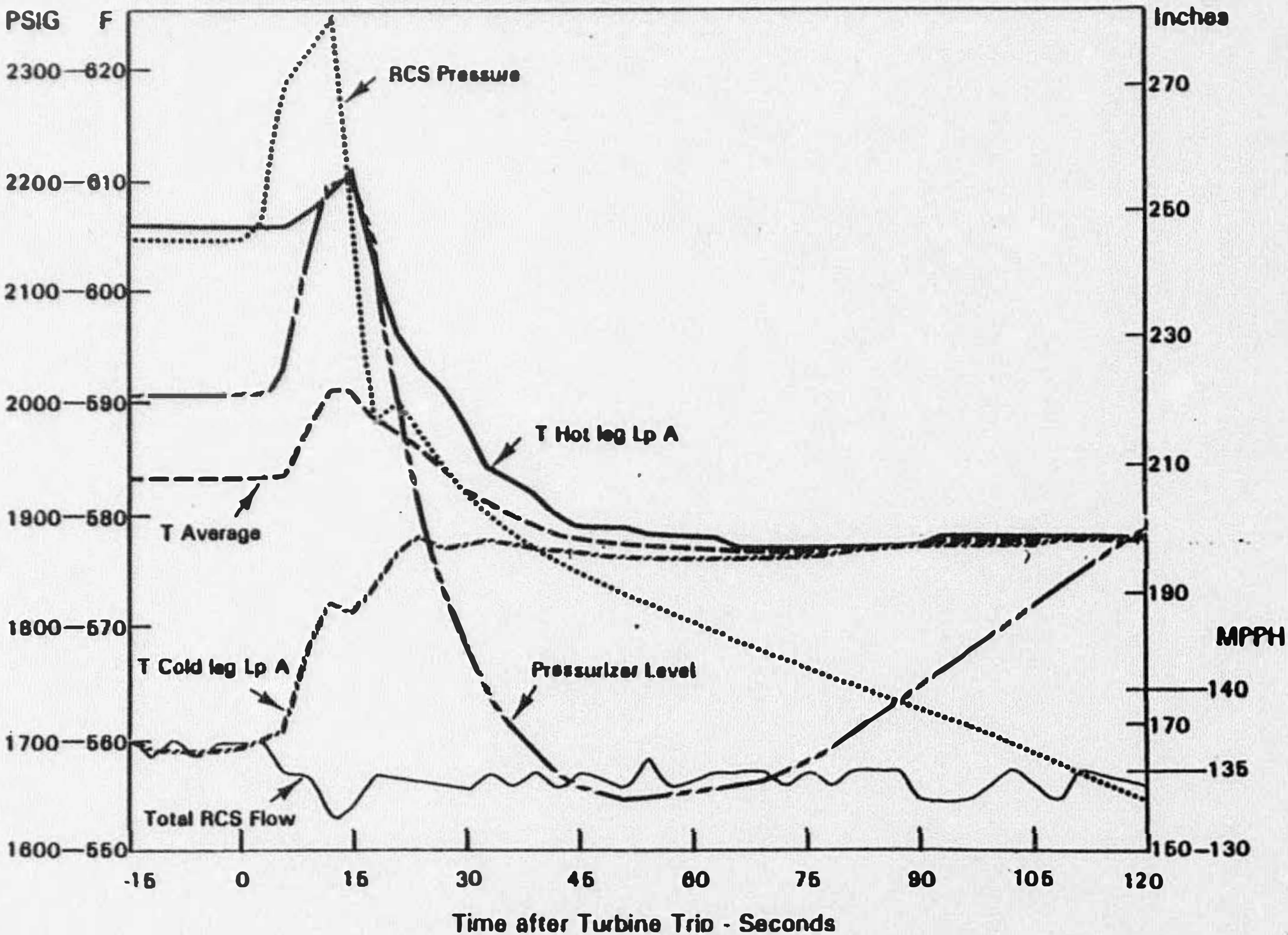


Figure 3
TMI-2 Loss of Coolant Accident 3/28/79
Reactor Coolant System Pressure, Saturation Pressure and Pressurizer Level

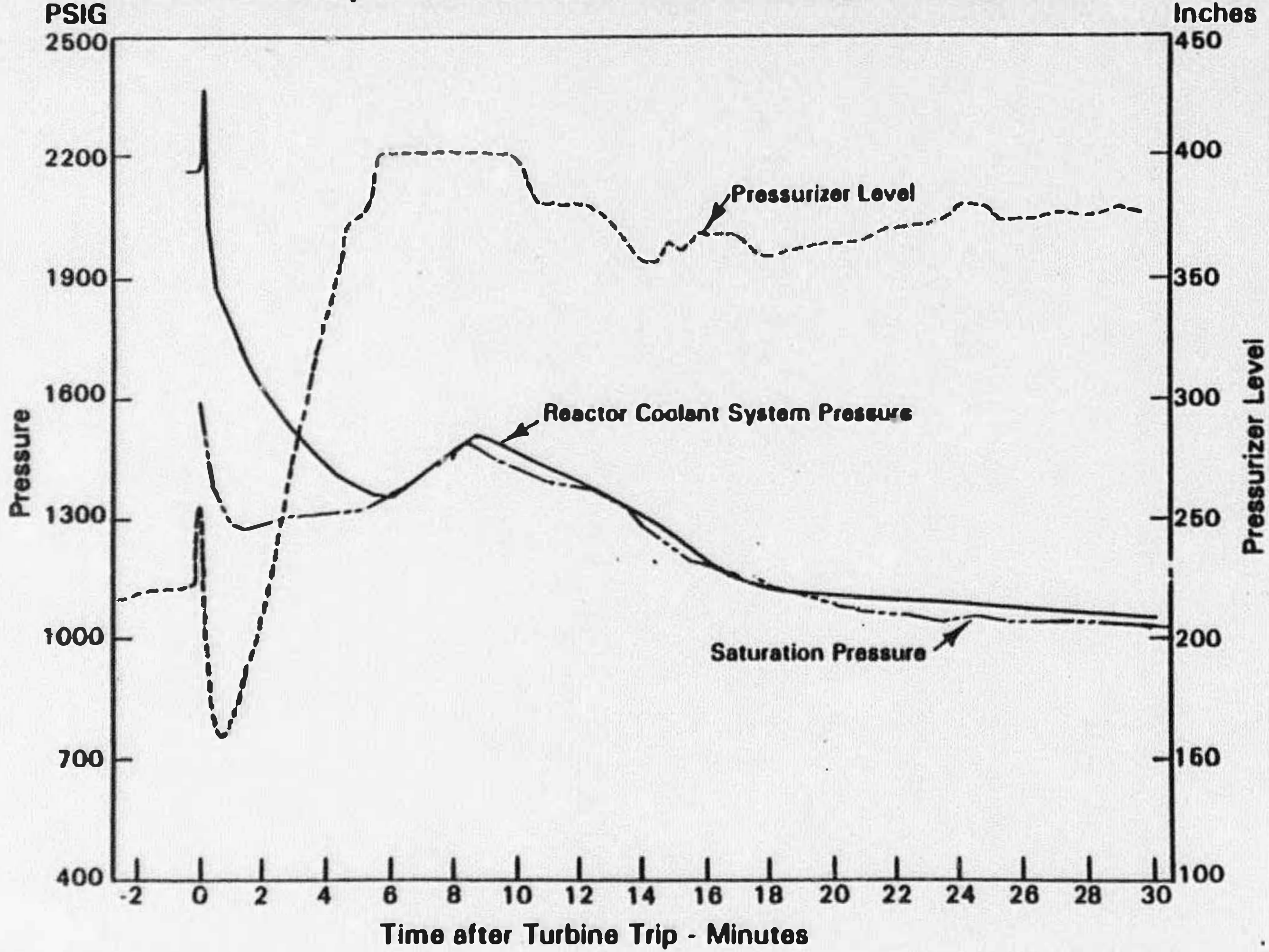


Figure 5
TMI-2 Loss of Coolant Accident 3/28/79
Reactor Coolant System Pressure and Saturation Pressure

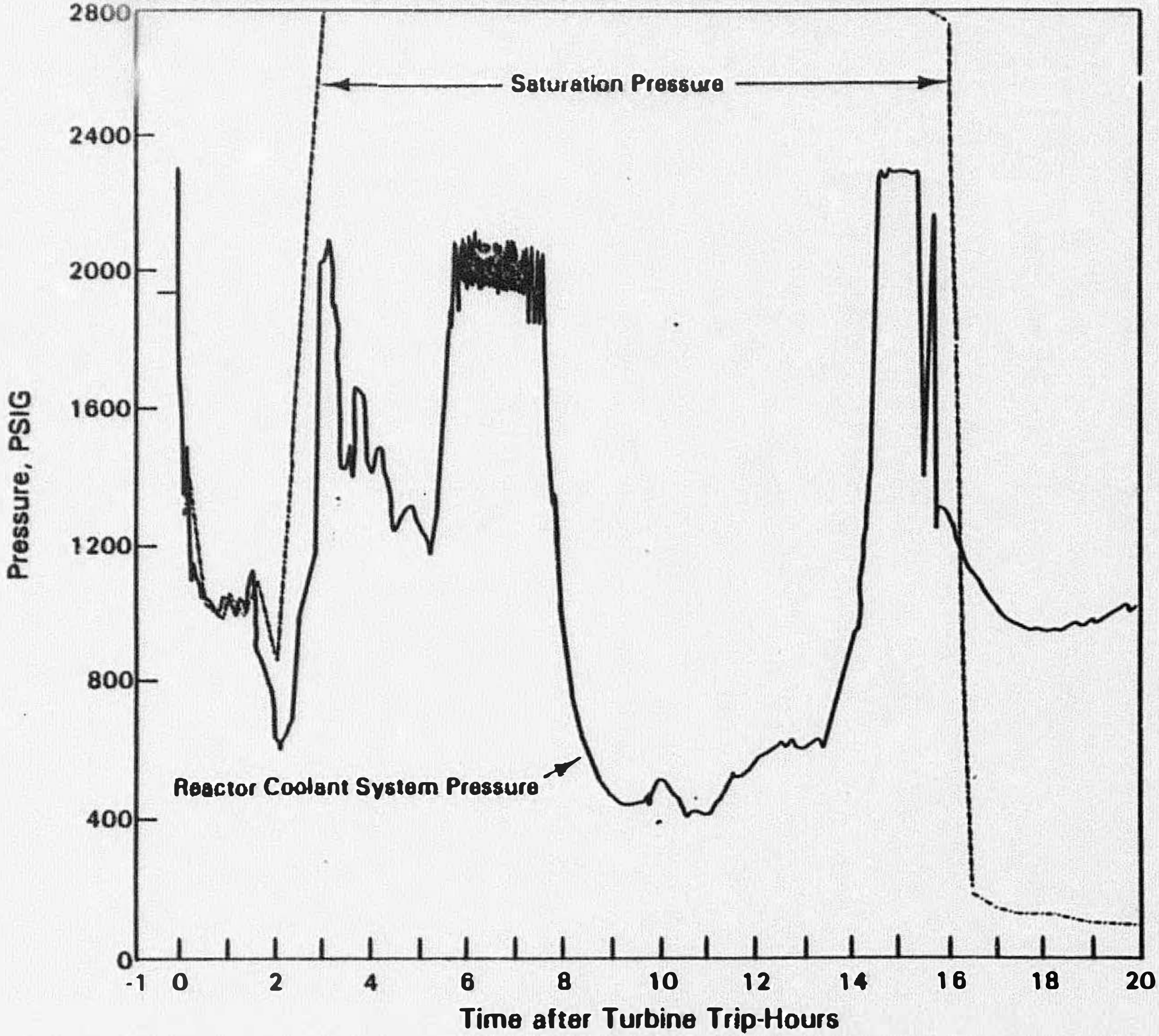


FIGURE 7

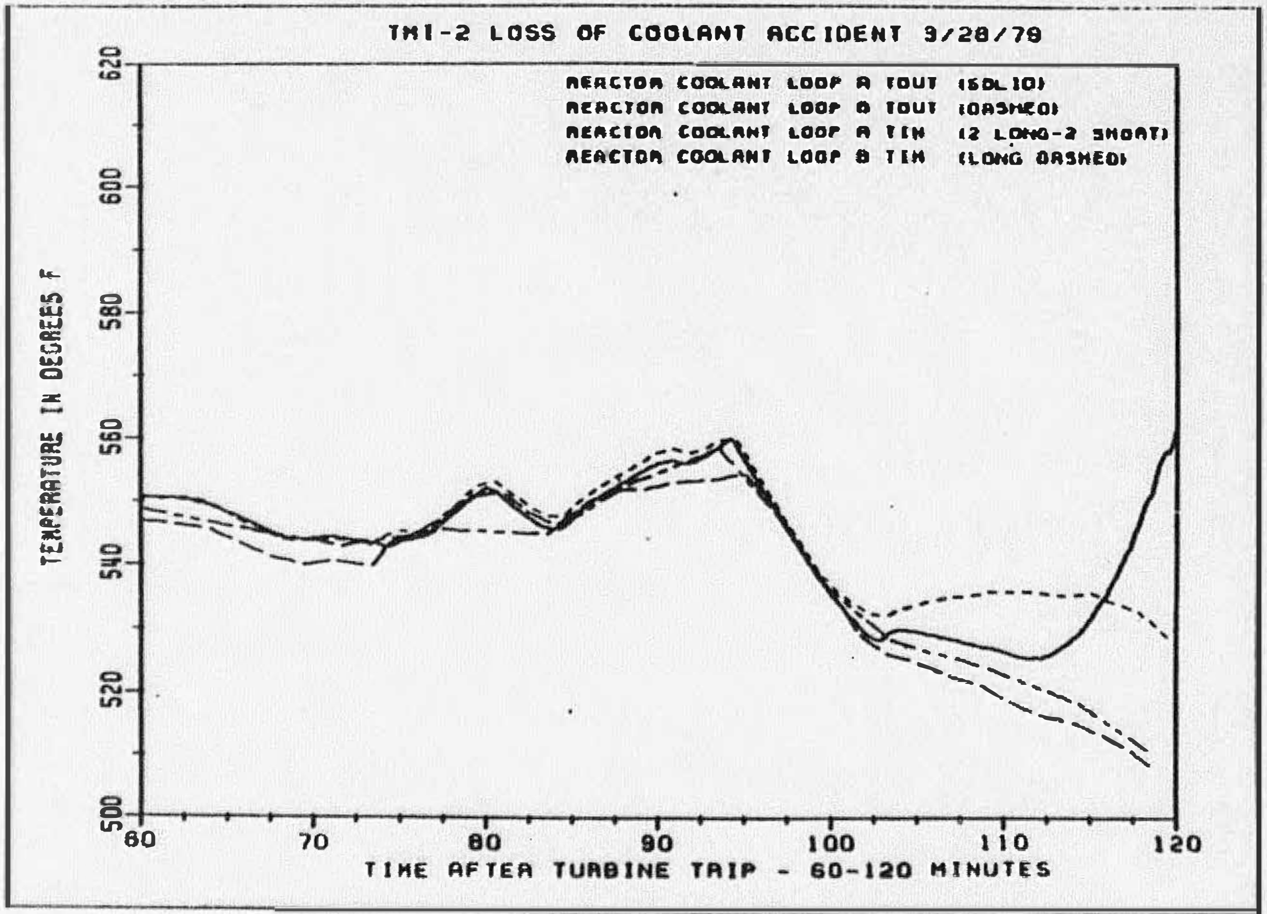
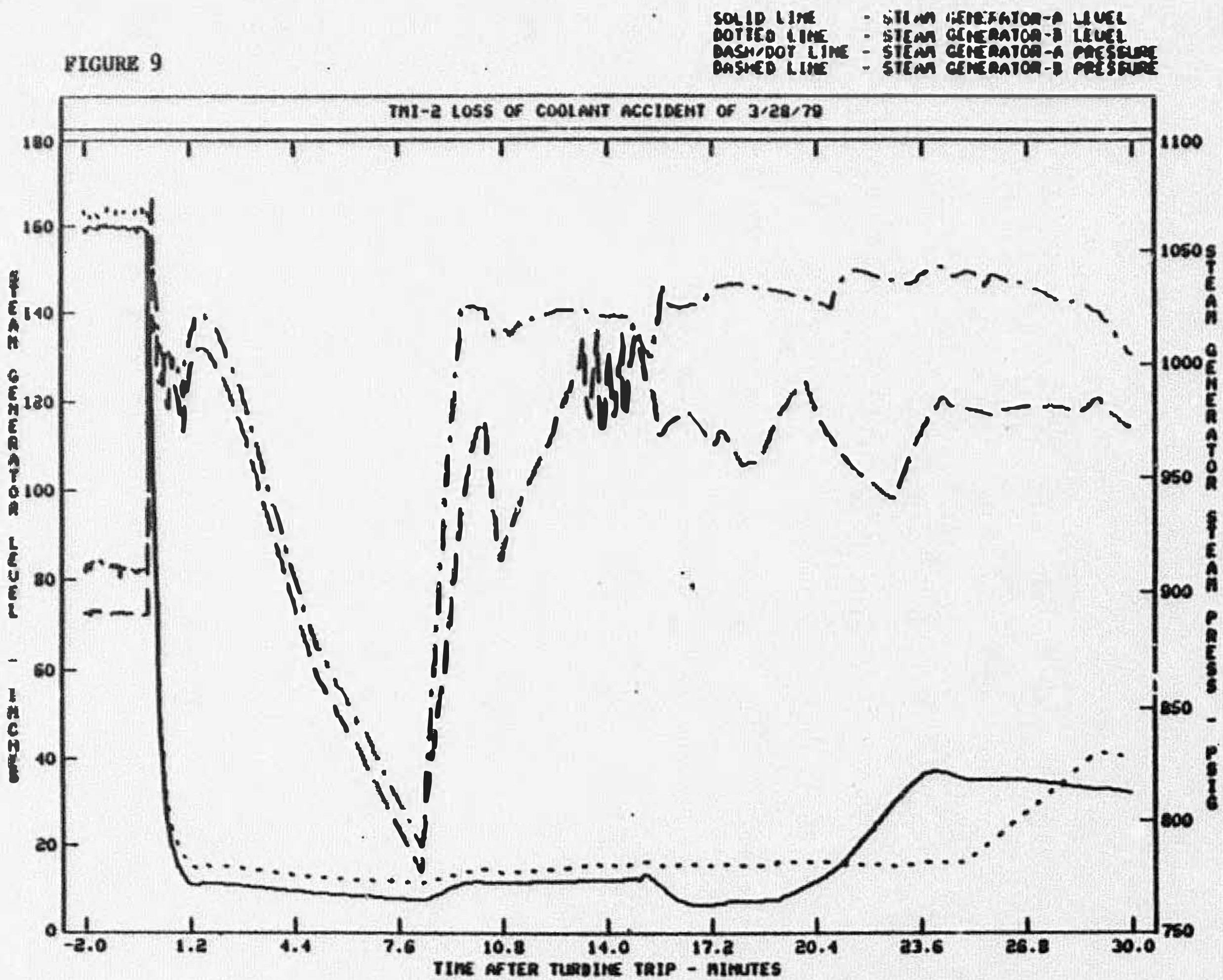

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
REACTOR COOLANT LOOP A TOUT (SOLID)
REACTOR COOLANT LOOP B TOUT (DASHED)
REACTOR COOLANT LOOP A TIN (2 LONG-2 SHORT)
REACTOR COOLANT LOOP B TIN (LONG DASHED)
TEMPERATURE IN DEGREES F
620
600
580
560
540
520
500
60
70
80
90
100
110
120
TIME AFTER TURBINE TRIP - 60-120 MINUTES

FIGURE 9



SOLID LINE - STEAM GENERATOR-A LEVEL
DOTTED LINE - STEAM GENERATOR-B LEVEL
DASH/DOT LINE - STEAM GENERATOR-A PRESSURE
DASHED LINE - STEAM GENERATOR-B PRESSURE
TMI-2 LOSS OF COOLANT ACCIDENT OF 3/28/79
STEAM GENERATOR LEVEL - INCHES
0
20
40
60
80
100
120
140
160
180
STEAM GENERATOR STEAM PRESS - PSIG
750
800
850
900
950
1000
1050
1100
-2.0
1.2
4.4
7.6
10.8
14.0
17.2
20.4
23.6
26.8
30.0
TIME AFTER TURBINE TRIP - MINUTES

FIGURE 11

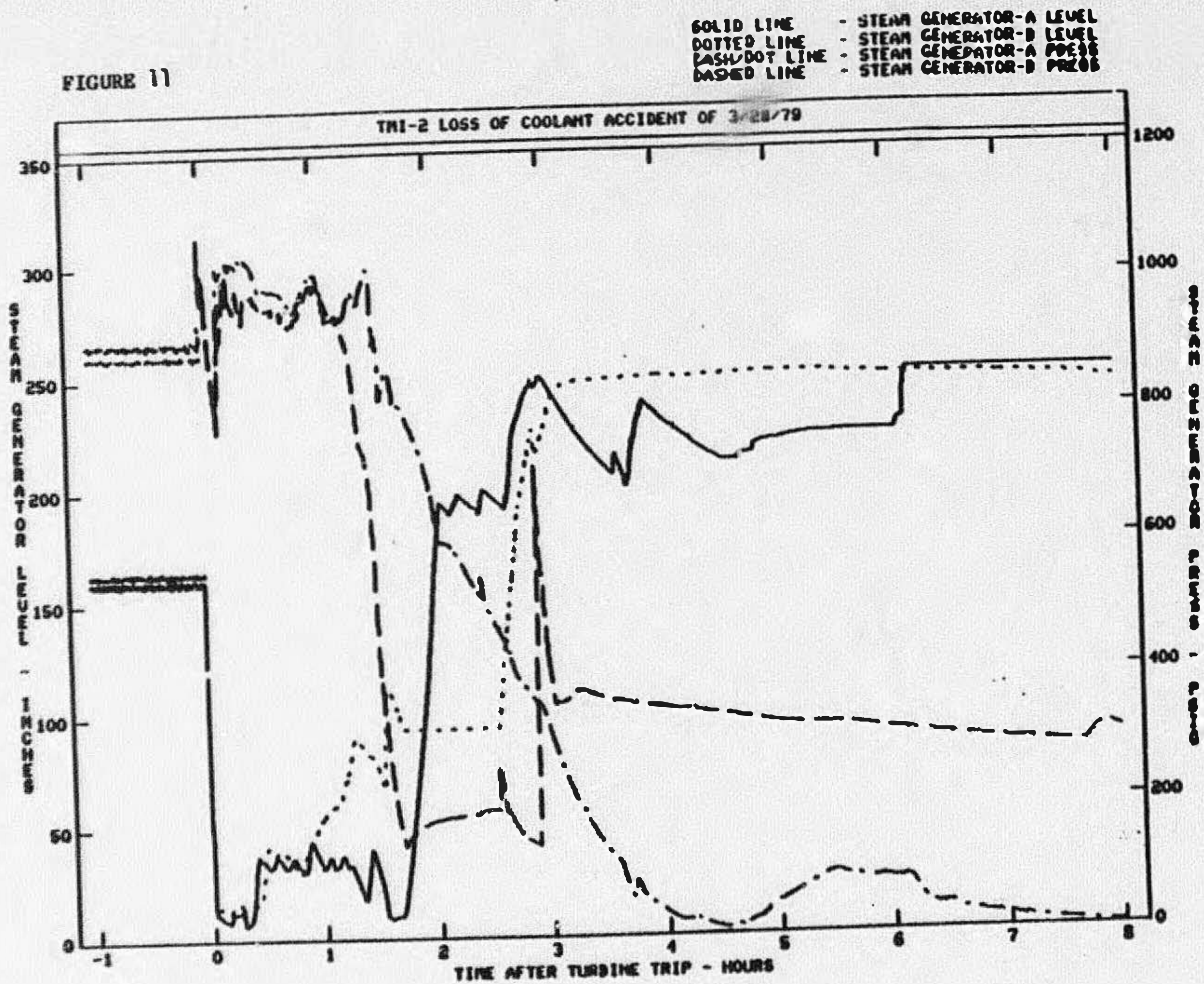

SOLID LINE - STEAM GENERATOR-A LEVEL
DOTTED LINE - STEAM GENERATOR-B LEVEL
DASH/DOT LINE - STEAM GENERATOR-A PRESS
DASHED LINE - STEAM GENERATOR-B PRESS
TMI-2 LOSS OF COOLANT ACCIDENT OF 3/28/79
STEAM GENERATOR LEVEL - INCHES
STEAM GENERATOR PRESS - PSIG
TIME AFTER TURBINE TRIP - HOURS
0
50
100
150
200
250
300
350
200
400
600
800
1000
1200
-1
1
2
3
4
5
6
7
8

Figure 13
TMI-2 Loss of Coolant Accident 3/28/79
Reactor Coolant System Pressure

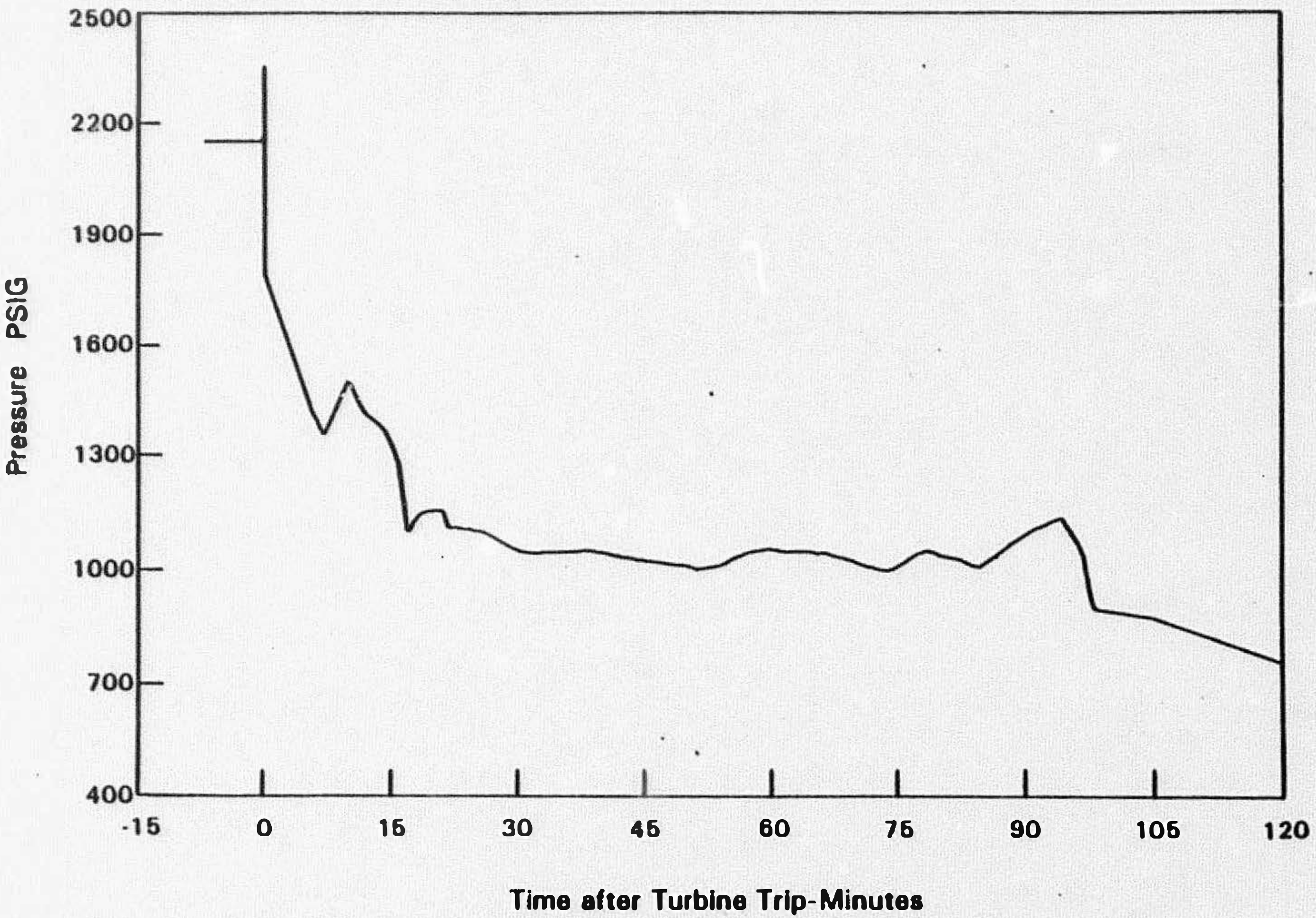
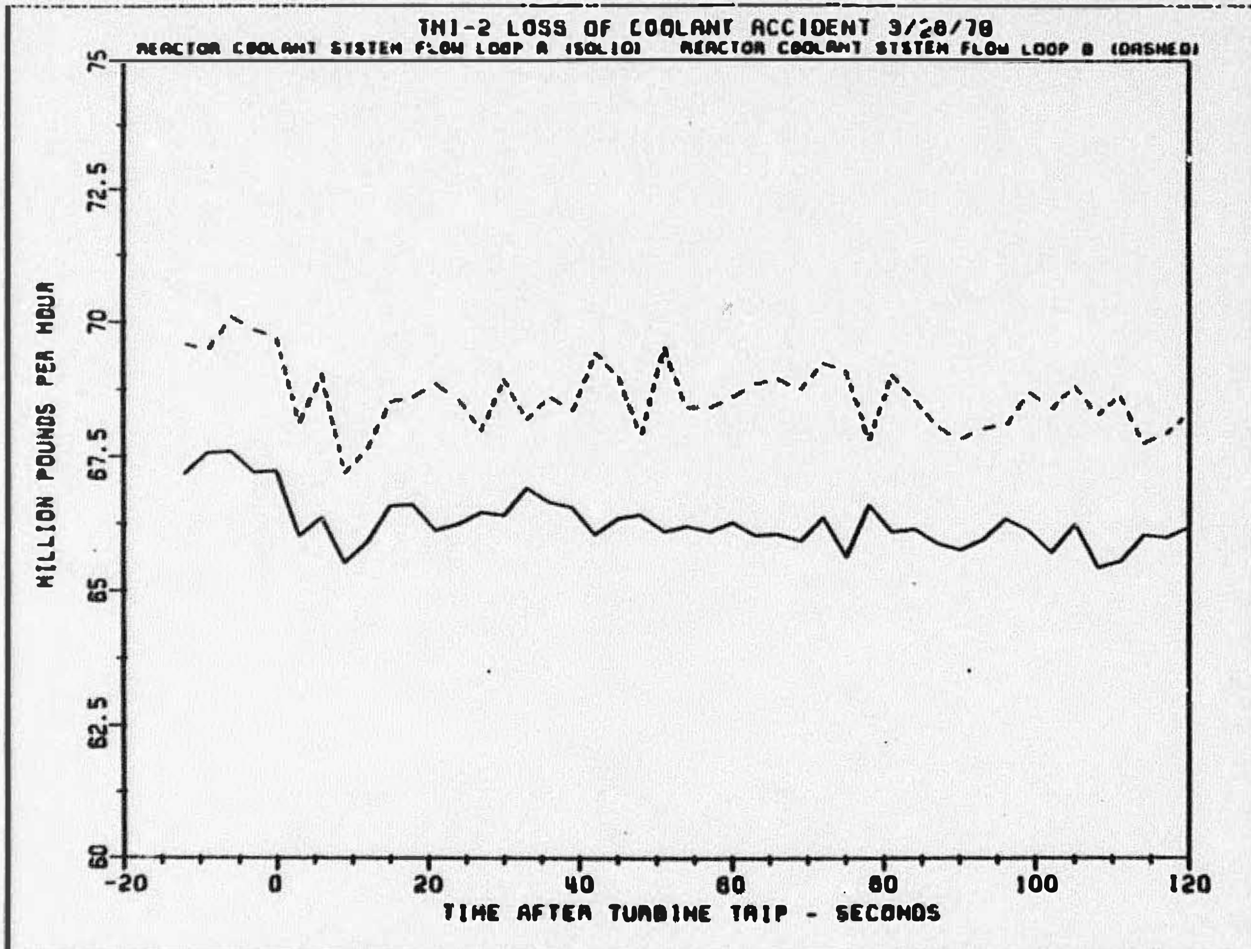
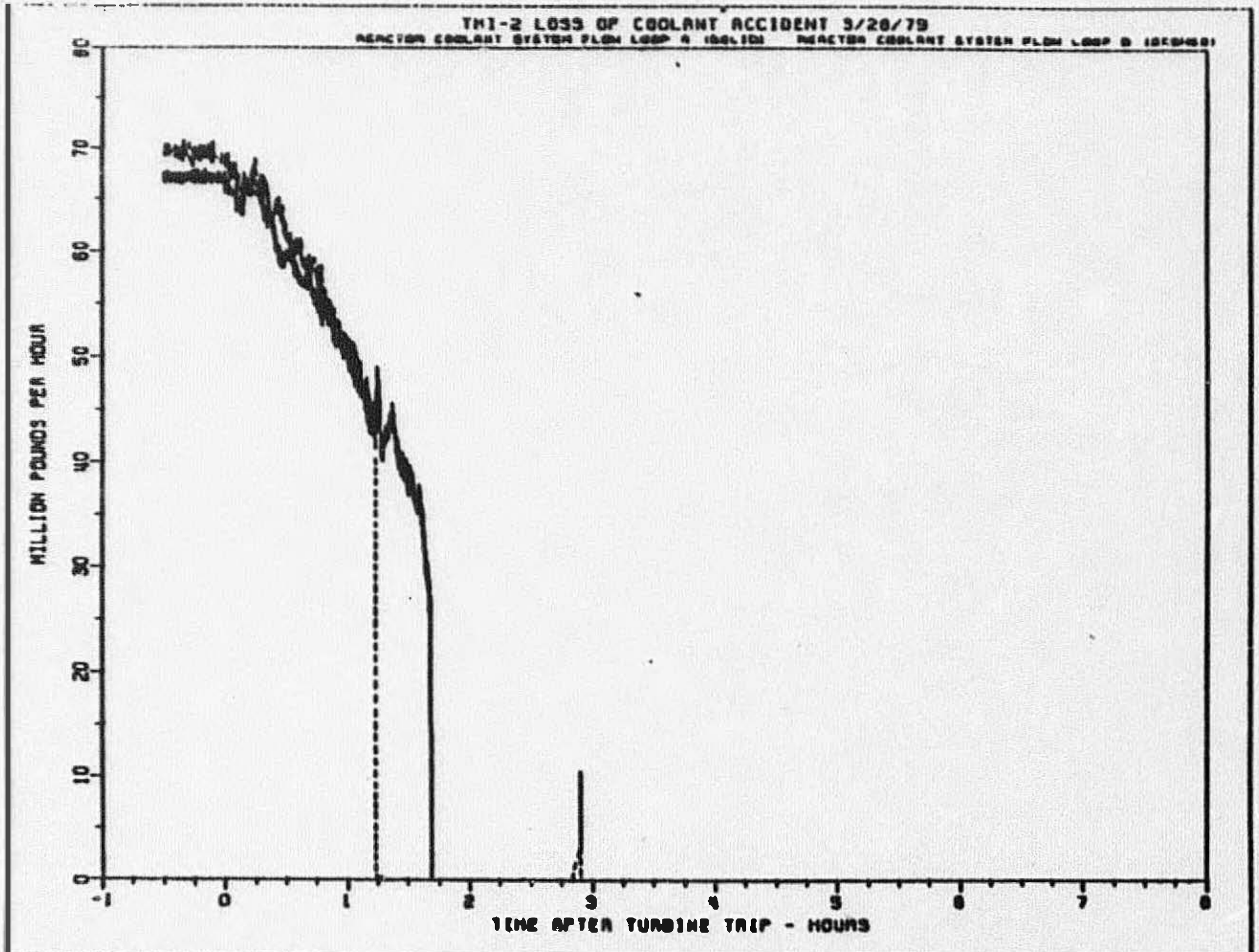


FIGURE 15



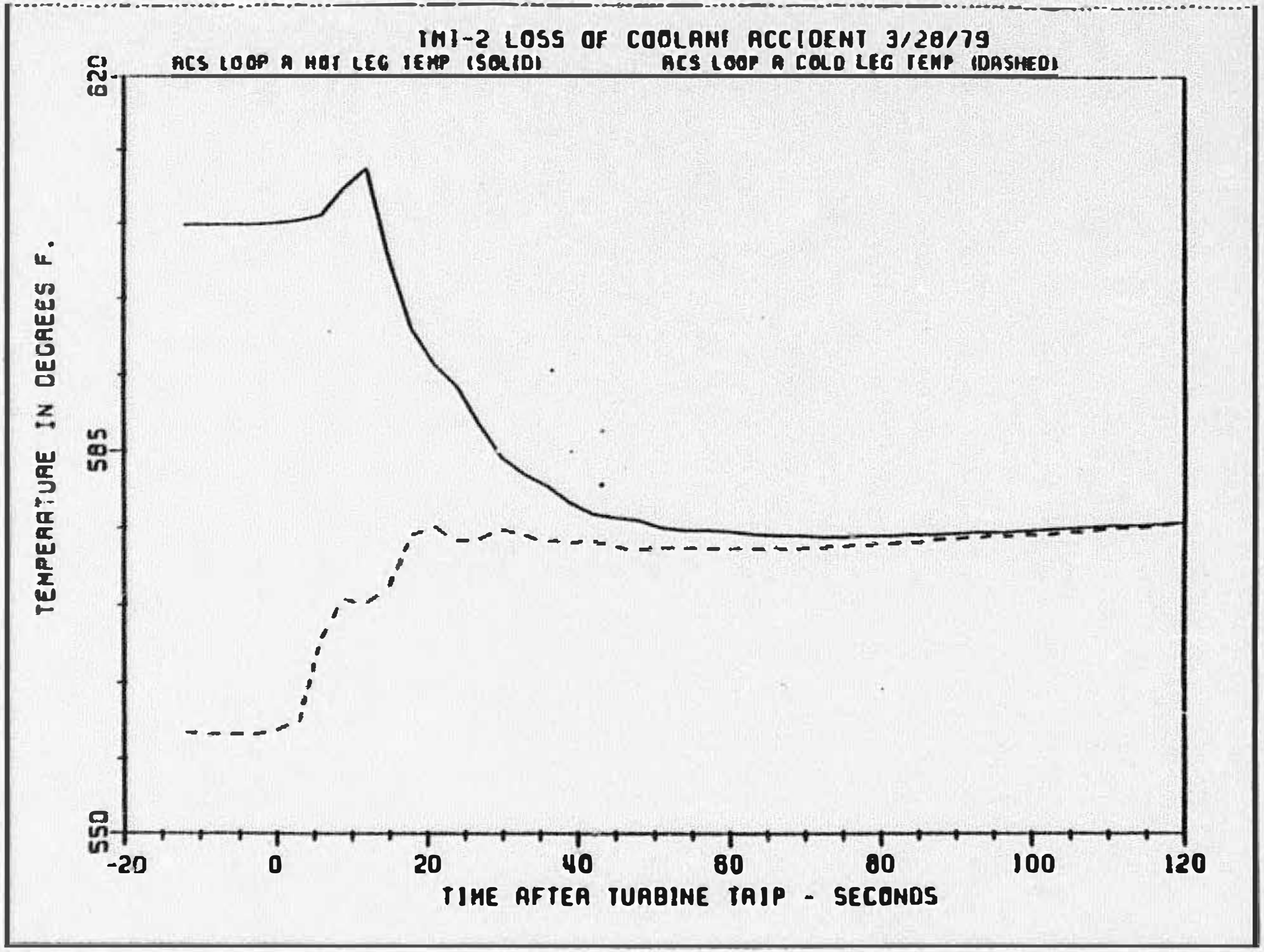
TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
REACTOR COOLANT SYSTEM FLOW LOOP A (SOLID) REACTOR COOLANT SYSTEM FLOW LOOP B (DASHED)
MILLION POUNDS PER HOUR
75
72.5
70
67.5
65
62.5
60
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - SECONDS

FIGURE 18



TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
REACTOR COOLANT SYSTEM FLOW LOOP A (SOLID) REACTOR COOLANT SYSTEM FLOW LOOP B (DASHED)
MILLION POUNDS PER HOUR
0
10
20
30
40
50
60
70
80
-1
0
1
2
3
4
5
6
7
8
TIME AFTER TURBINE TRIP - HOURS

FIGURE 20



TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
RCS LOOP A HOT LEG TEMP (SOLID)
RCS LOOP A COLD LEG TEMP (DASHED)
TEMPERATURE IN DEGREES F.
620
585
550
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - SECONDS

FIGURE 22

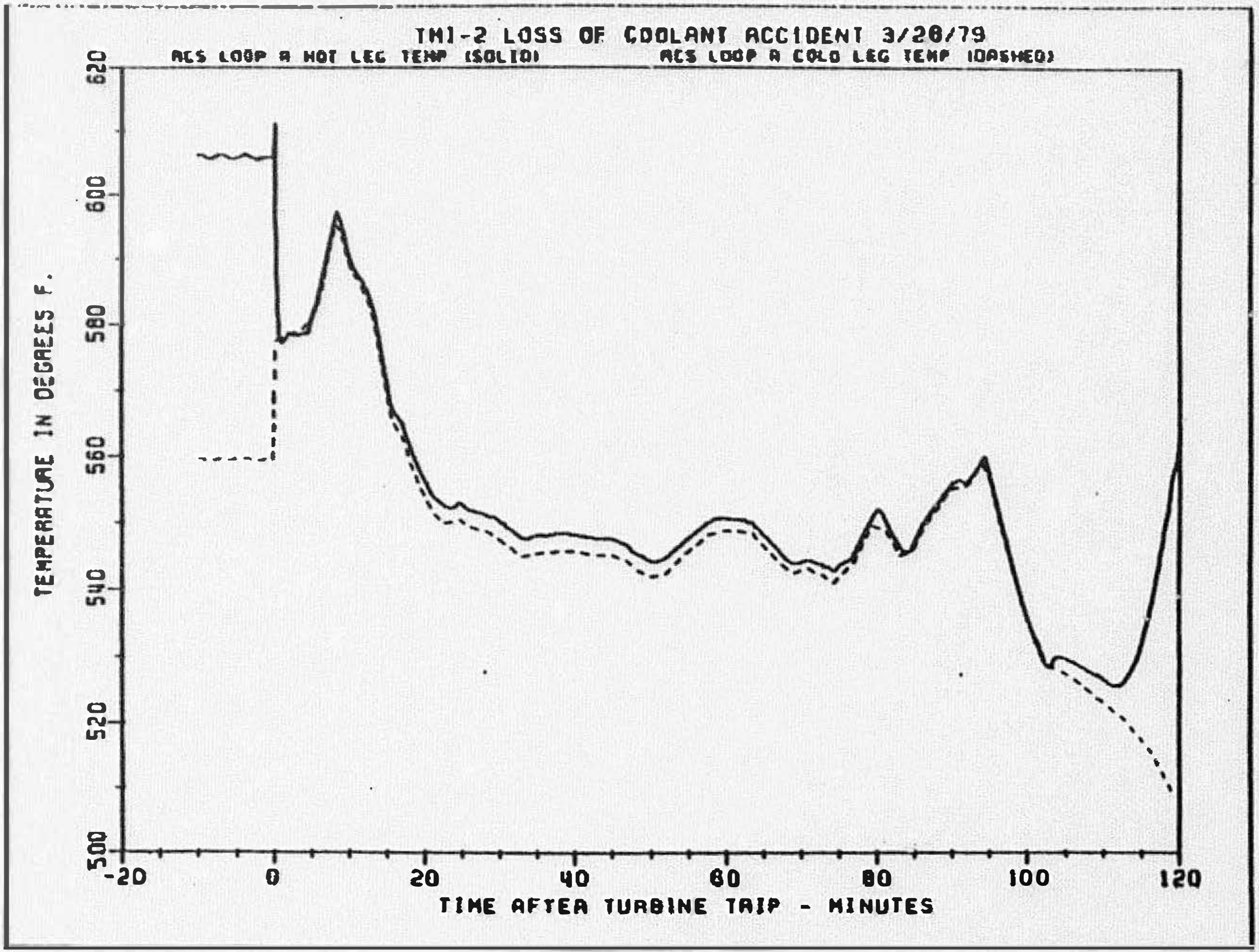

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
RCS LOOP A HOT LEG TEMP (SOLID)
RCS LOOP A COLD LEG TEMP (DASHED)
TEMPERATURE IN DEGREES F.
620
600
580
560
540
520
500
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - MINUTES

Figure 24
TMI-2 Loss of Coolant Accident 3/28/79
RC-2 Loop A Cold and Hot Leg Temperatures

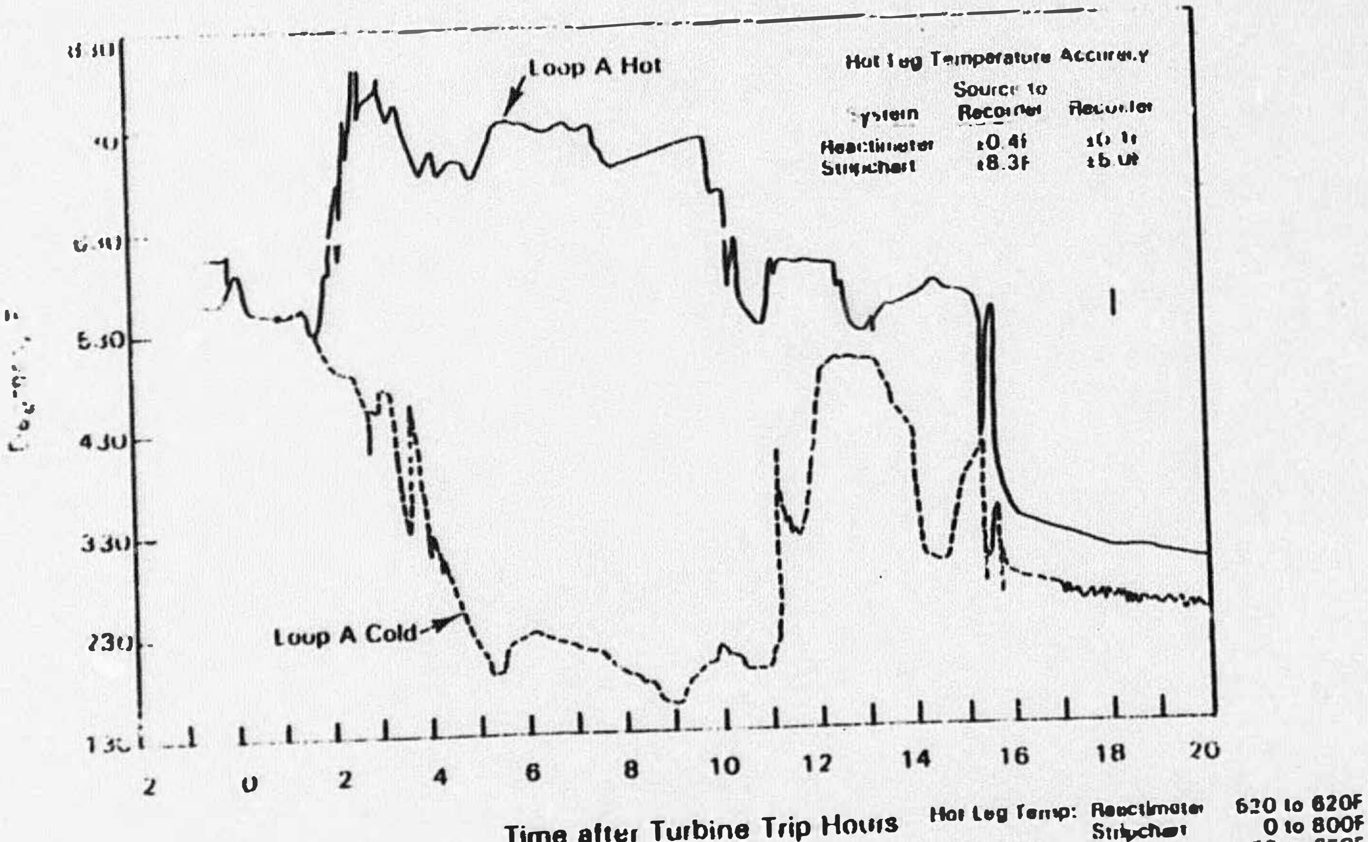
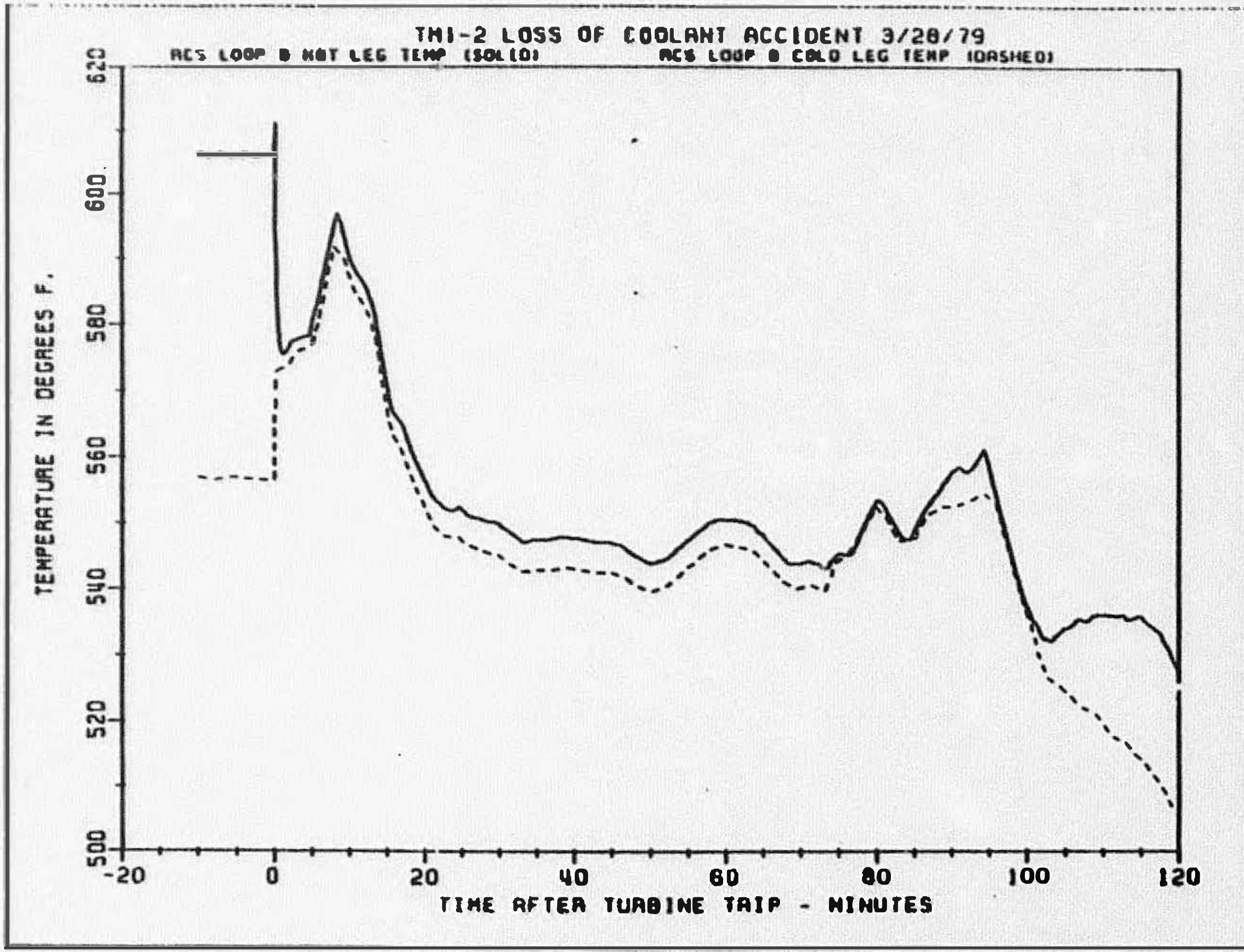


FIGURE 27



TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
RCS LOOP B HOT LEG TEMP (SOLID)
RCS LOOP B COLD LEG TEMP (DASHED)
TEMPERATURE IN DEGREES F.
620
600
580
560
540
520
500
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - MINUTES

Figure 28
TMI-2 Loss of Coolant Accident 3/28/79
RCS Loop B Cold and Hot Leg Temperatures

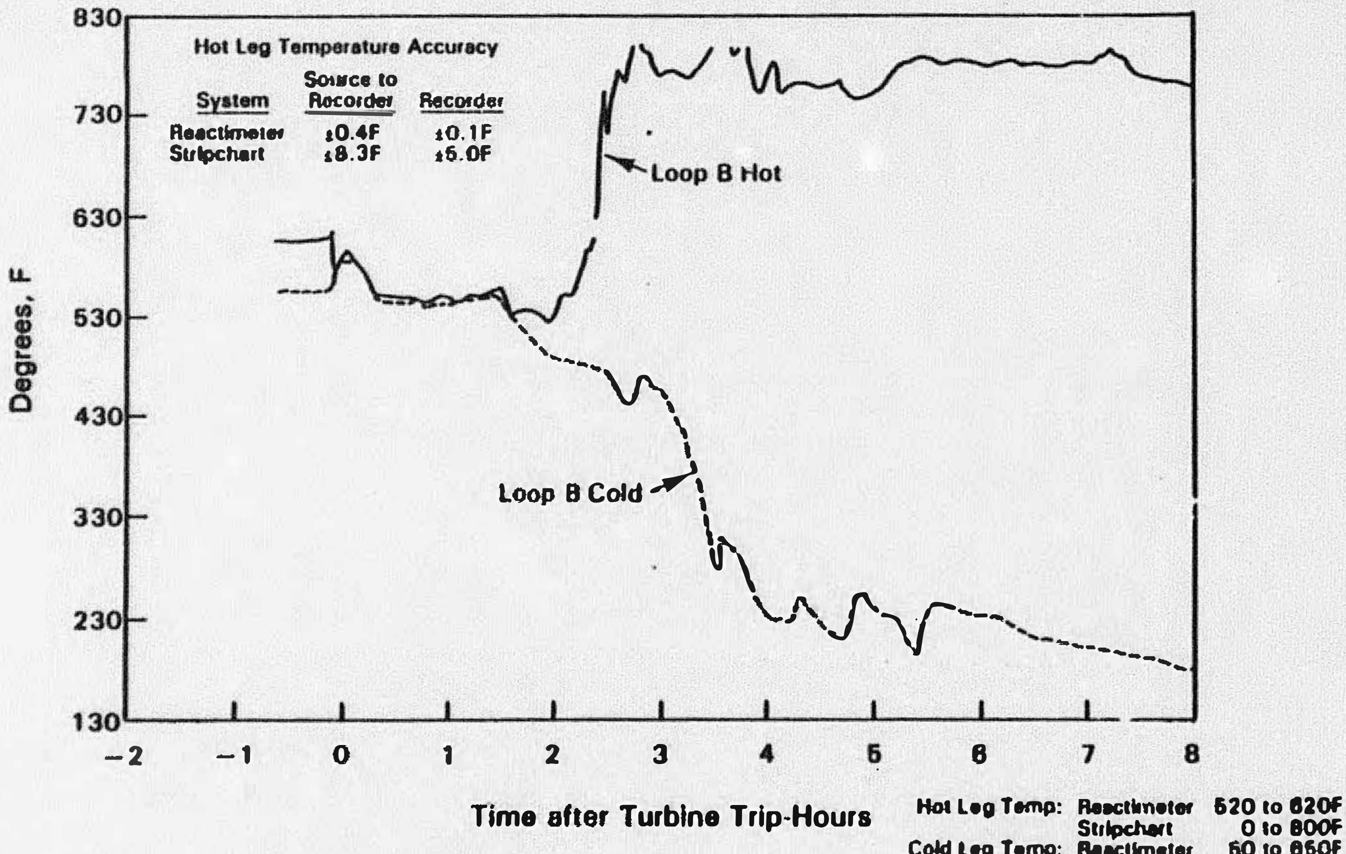


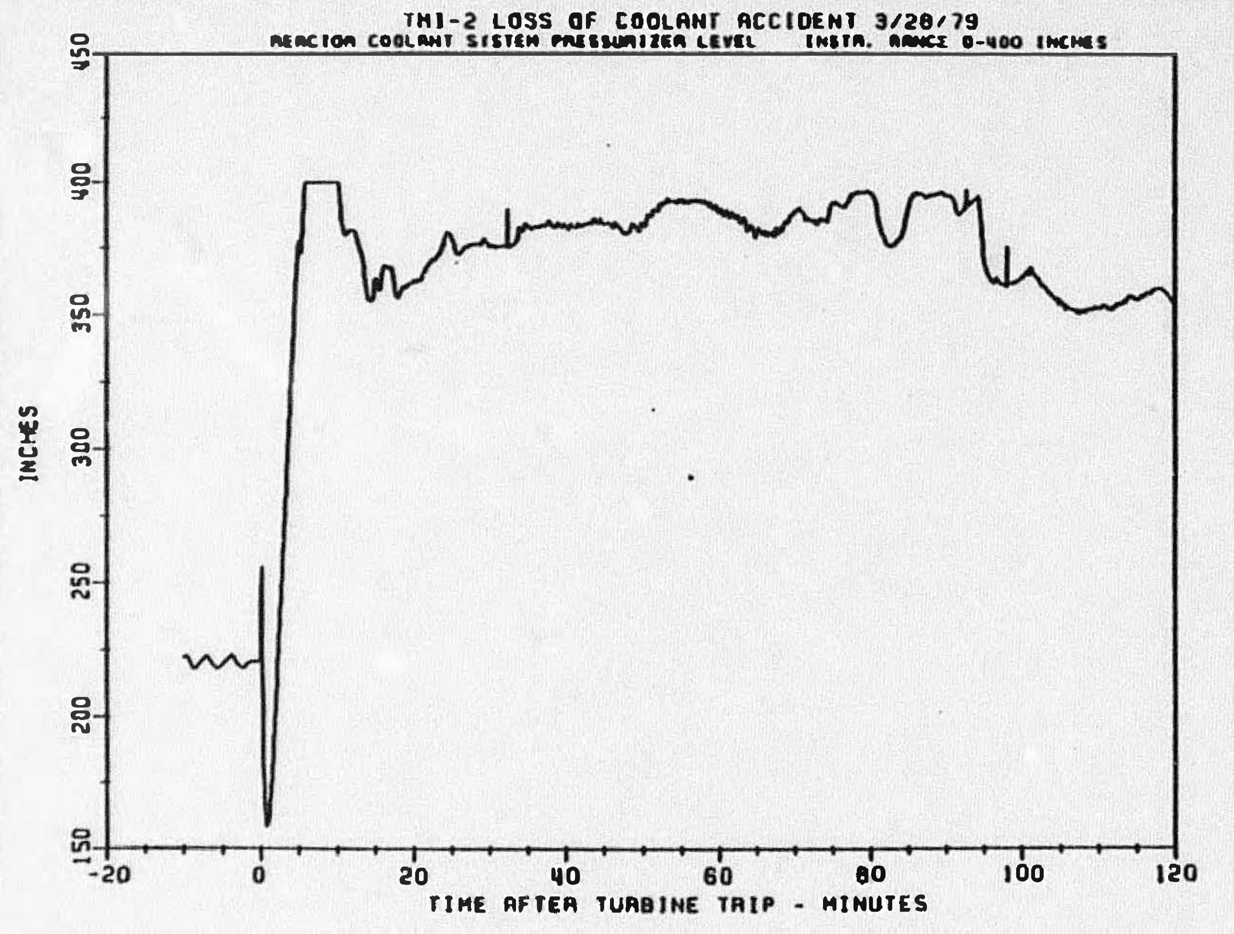
Figure 30

TMI-2 Loss of Coolant Accident 3/28/79
Reactor Coolant System Exit Fuel Assembly Temperature at Approximately 04:59:23 (0900:00)

	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					

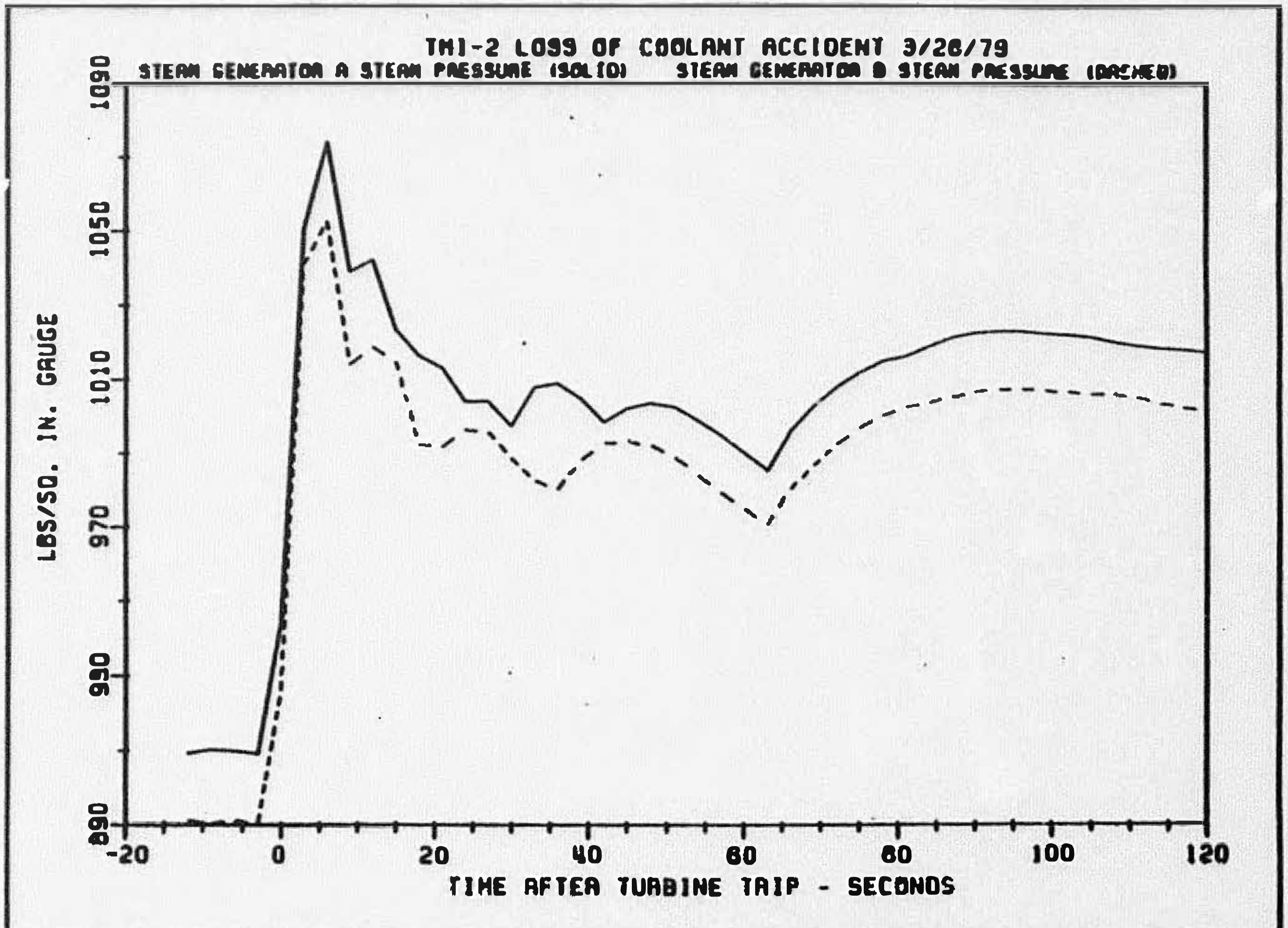
Note: These values do not include connections for the reference junction which was reading approximately 75F.

FIGURE 32



TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
REACTOR COOLANT SYSTEM PRESSURIZER LEVEL
INSTR. RANGE 0-400 INCHES
INCHES
450
400
350
300
250
200
150
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - MINUTES

FIGURE 34



TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
STEAM GENERATOR A STEAM PRESSURE (SOLID)
STEAM GENERATOR B STEAM PRESSURE (DASHED)
LBS/SQ. IN. GAUGE
1090
1050
1010
970
930
890
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - SECONDS

FIGURE 36

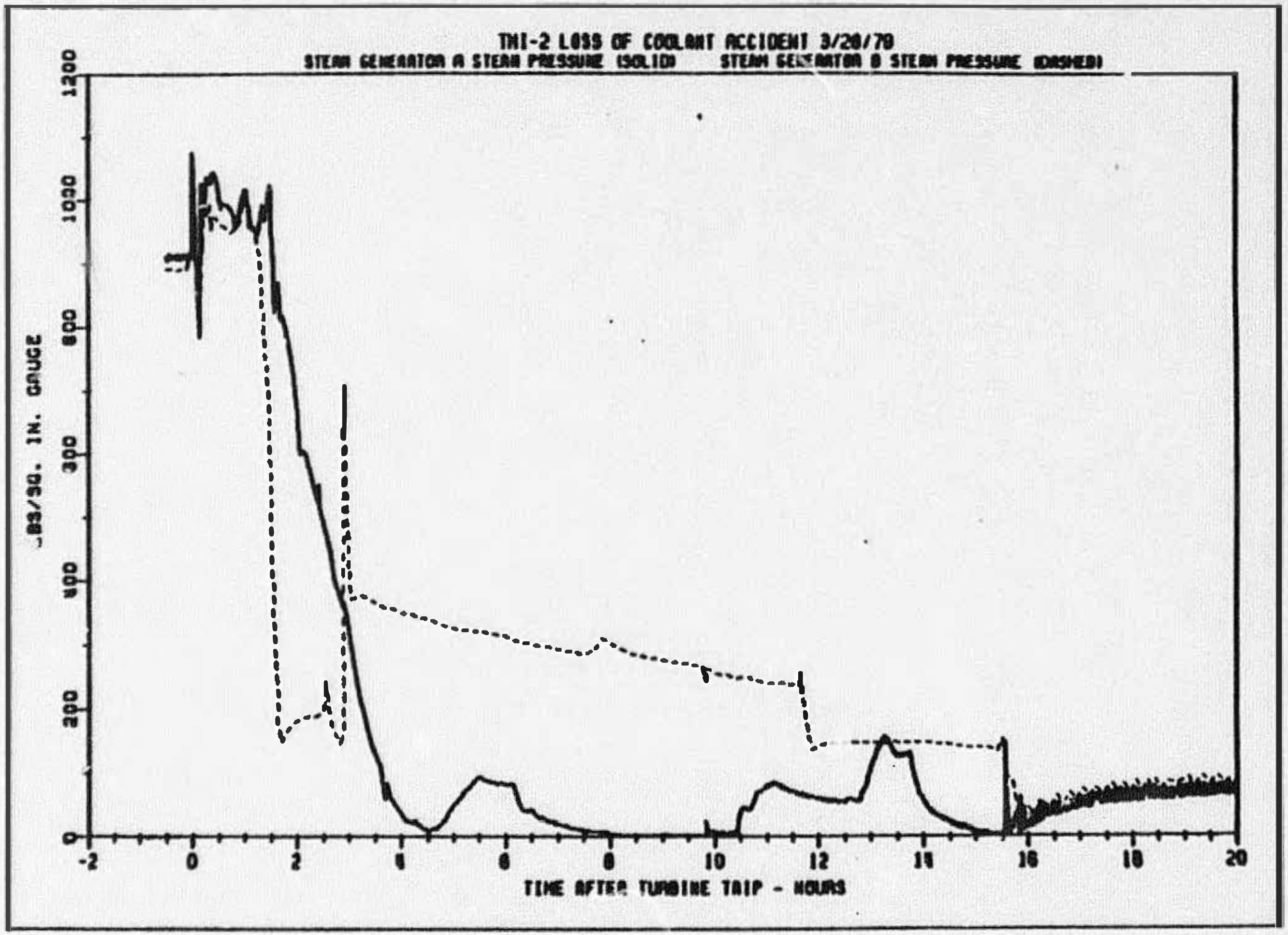

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
STEAM GENERATOR A STEAM PRESSURE (SOLID)
STEAM GENERATOR B STEAM PRESSURE (DASHED)
LBS/SQ. IN. GAUGE
TIME AFTER TURBINE TRIP - HOURS

Figure 38
TMI-2 Loss of Coolant Accident 3/28/79
Turbine Header Pressure Loop A and Loop B

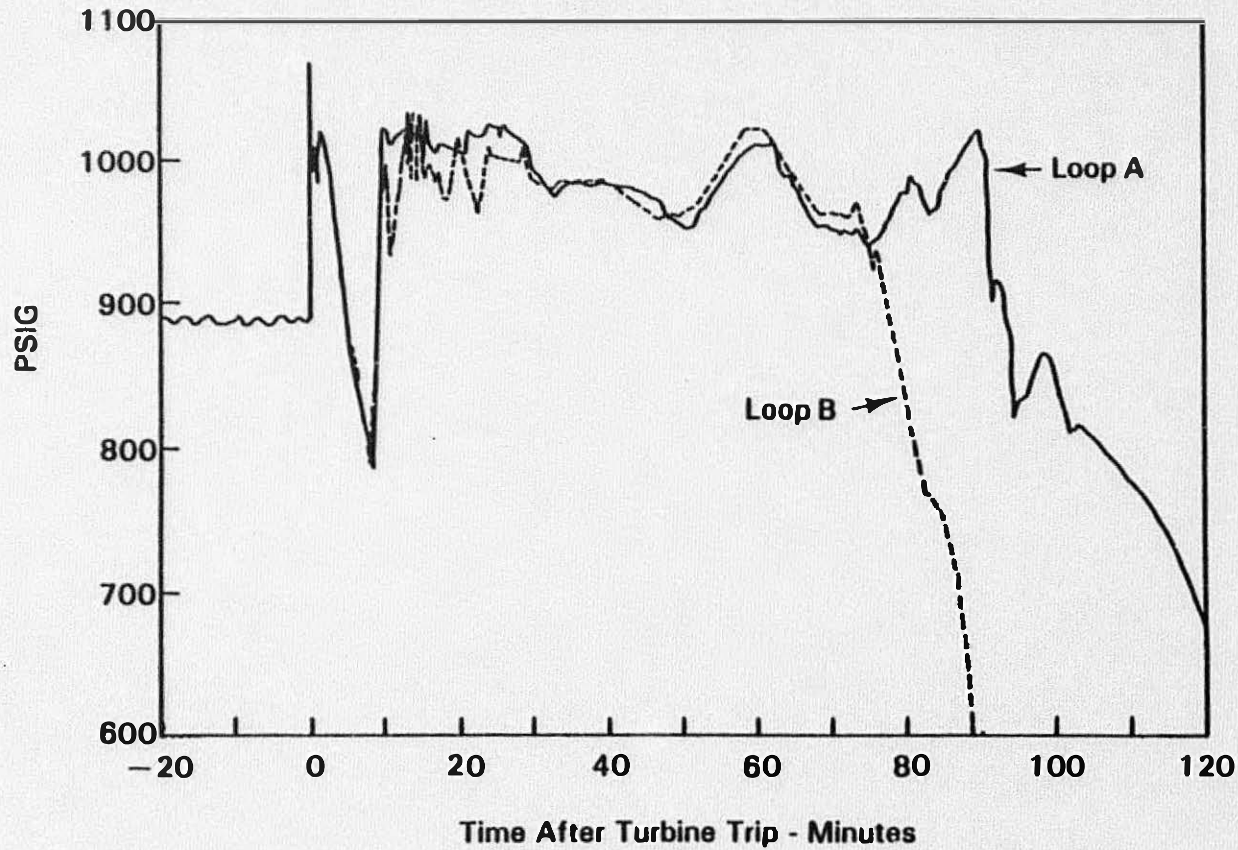


FIGURE 10

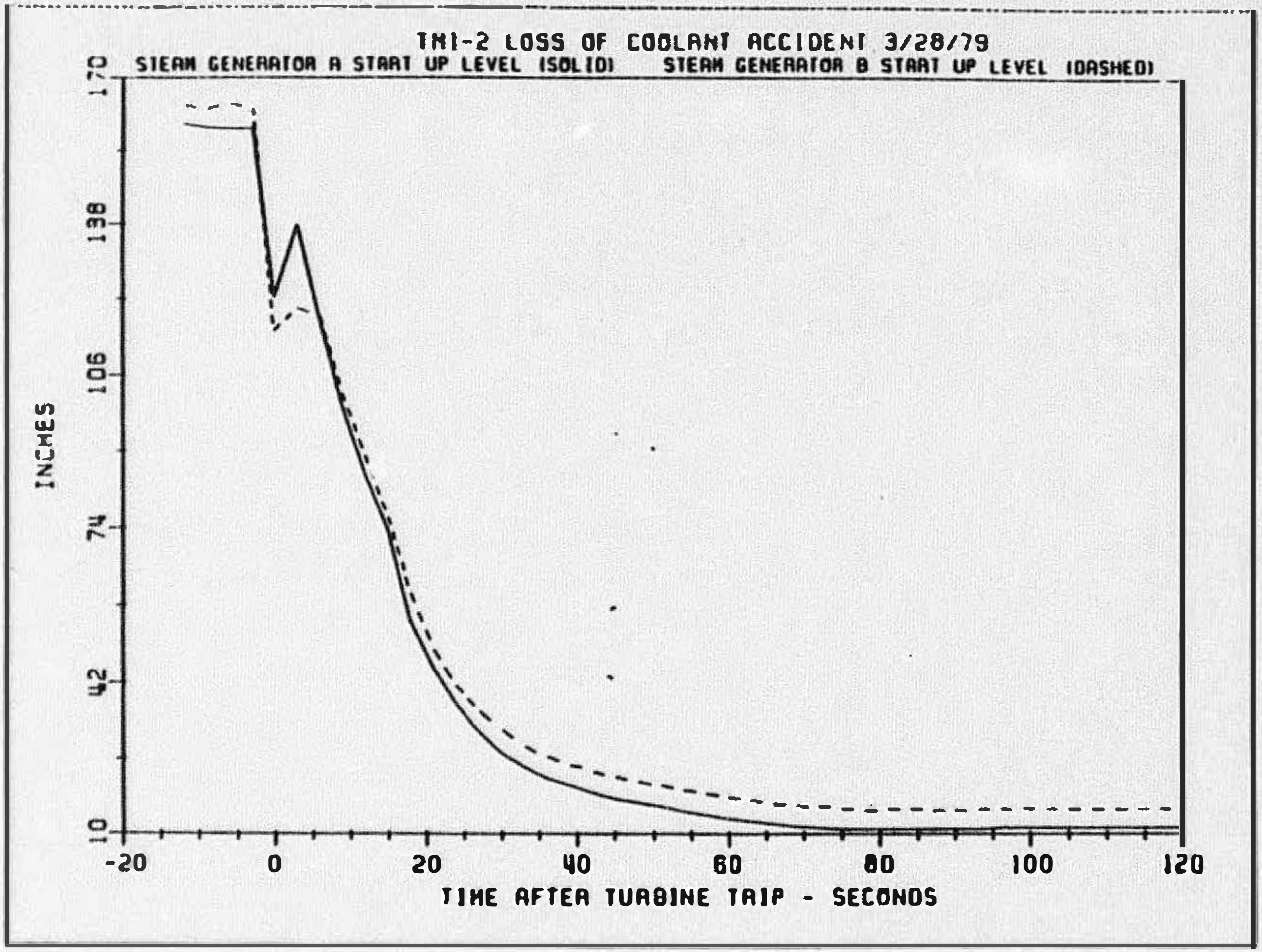

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
STEAM GENERATOR A START UP LEVEL (SOLID)
STEAM GENERATOR B START UP LEVEL (DASHED)
INCHES
170
138
106
74
42
10
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - SECONDS

FIGURE 42

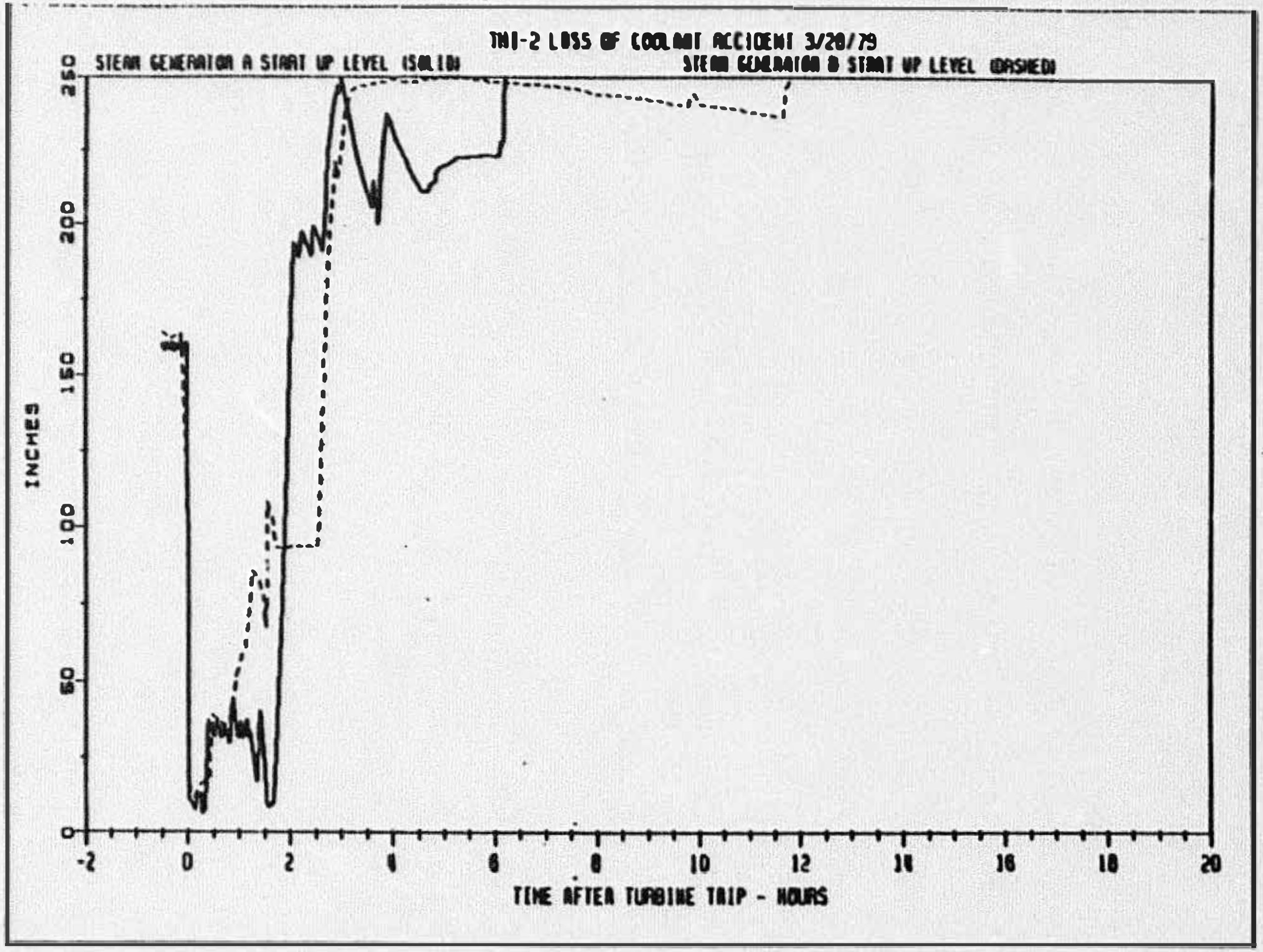

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
STEAM GENERATOR A START UP LEVEL (SOLID)
STEAM GENERATOR B START UP LEVEL (DASHED)
INCHES
0
50
100
150
200
250
-2
0
2
4
6
8
10
12
14
16
18
20
TIME AFTER TURBINE TRIP - HOURS

FIGURE 44

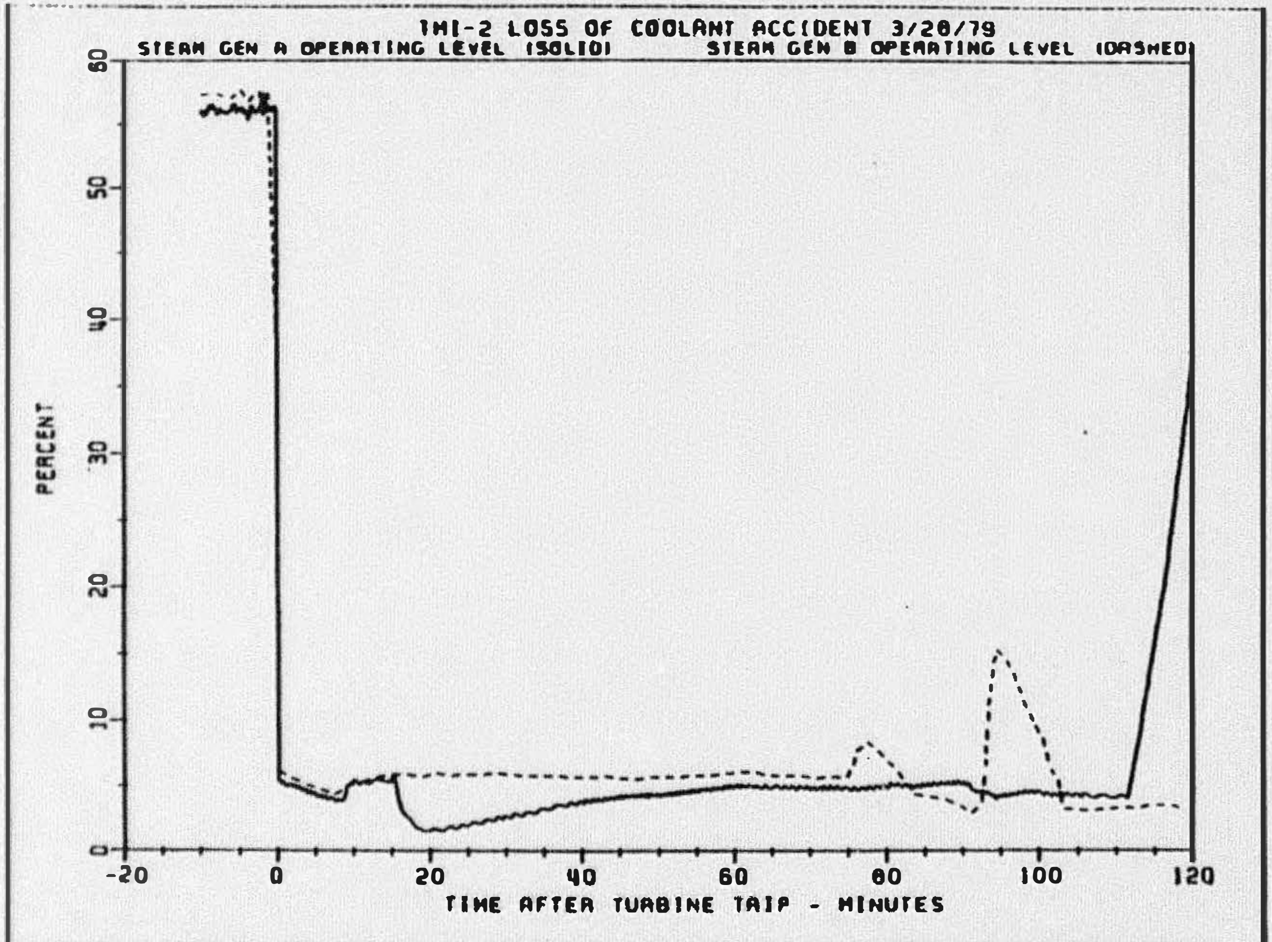
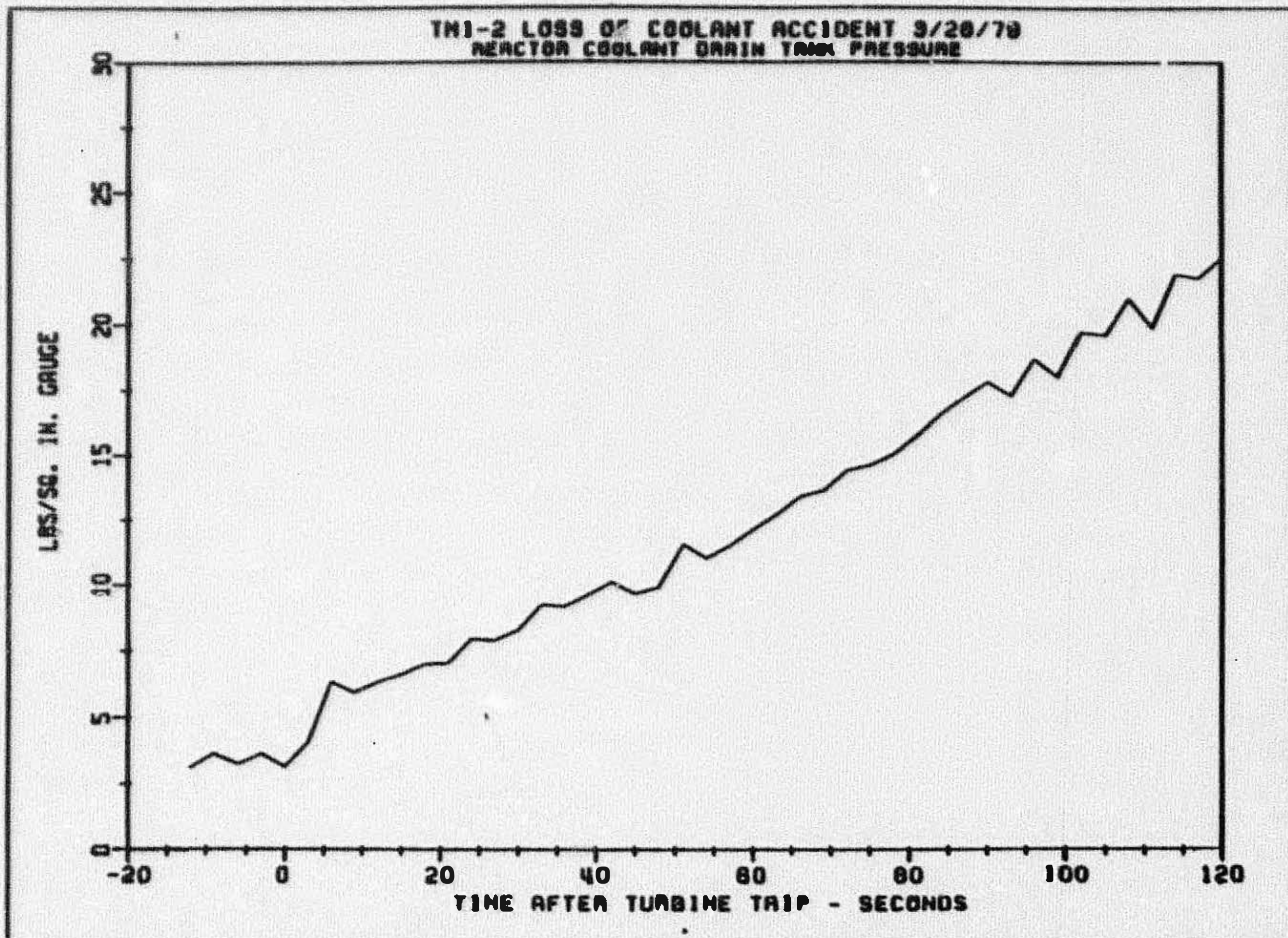

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
STEAM GEN A OPERATING LEVEL (SOLID)
STEAM GEN B OPERATING LEVEL (DASHED)
PERCENT
0
10
20
30
40
50
60
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - MINUTES

FIGURE 46



TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
REACTOR COOLANT DRAIN TANK PRESSURE
LBS/SQ. IN. GAUGE
30
25
20
15
10
5
0
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - SECONDS

FIGURE 48

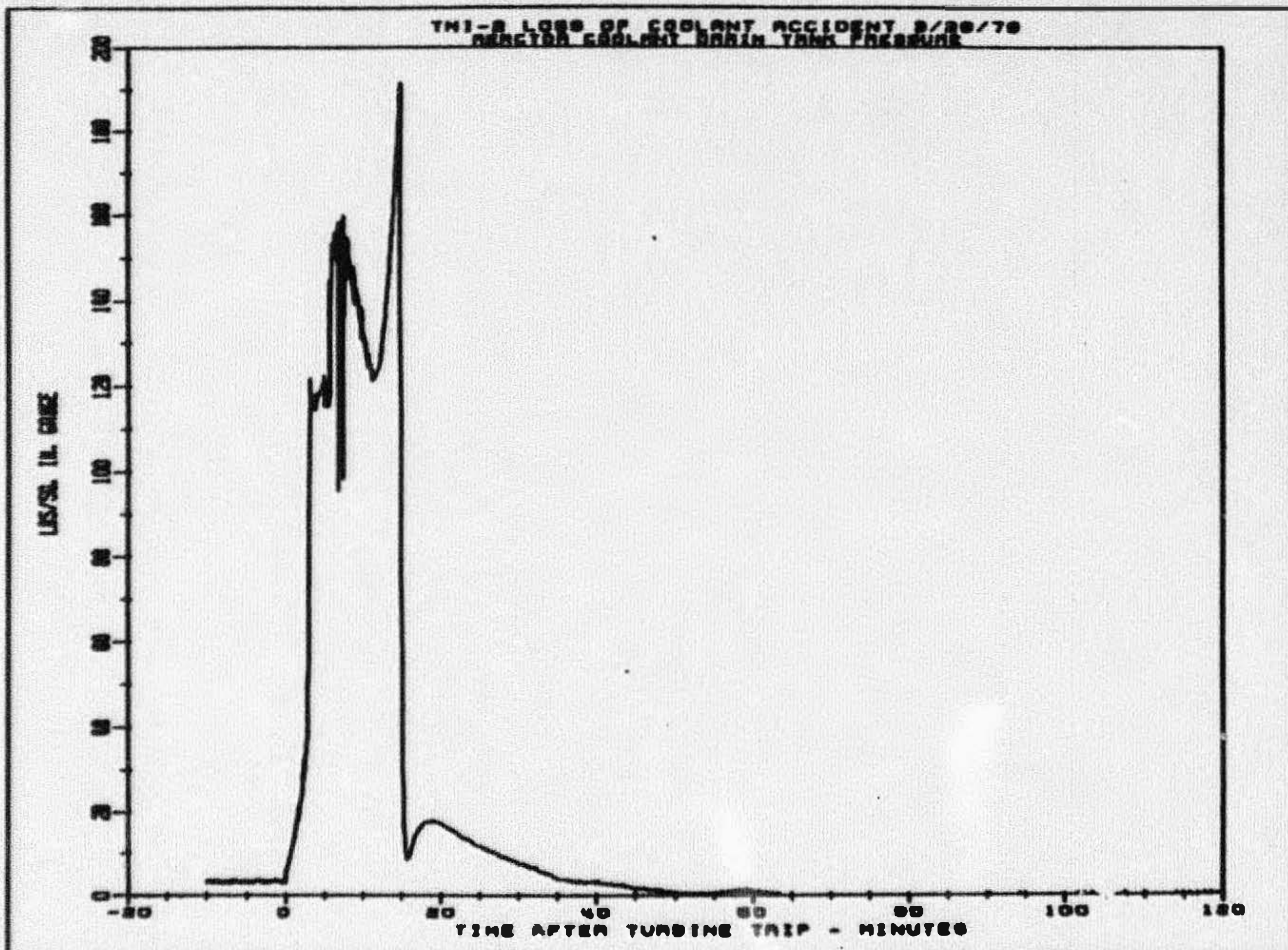
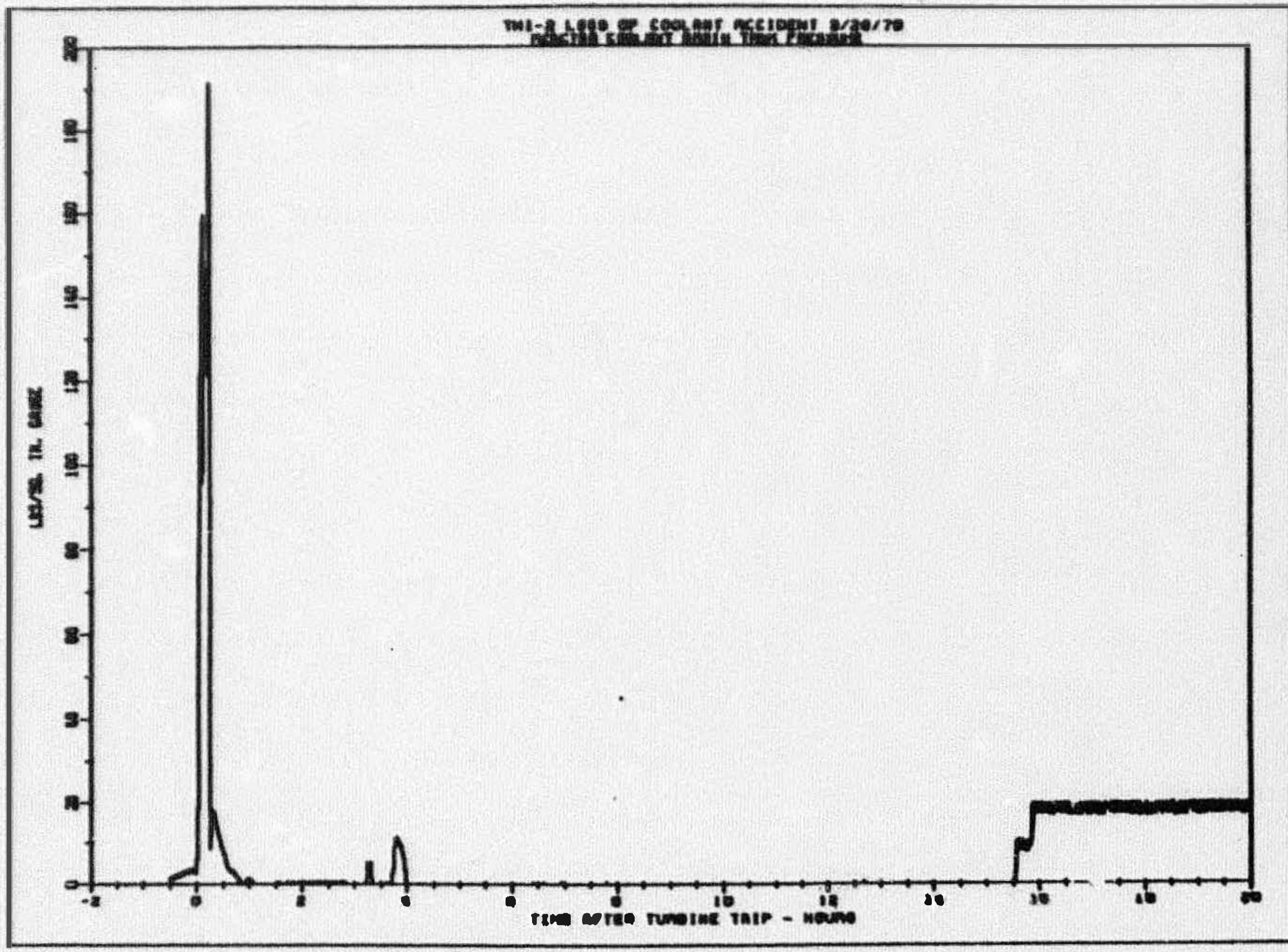

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
REACTOR COOLANT DRAIN TANK PRESSURE
LBS/SQ. IN. GAUGE
0
20
40
60
80
100
120
140
160
180
200
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - MINUTES

FIGURE 50



TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
LBS/SQ. IN. GAUGE
TIME AFTER TURBINE TRIP - HOURS

Figure 52
TMI-2 Loss of Coolant Accident 3/28/79
R_x Building Temperature Recorder
AH-YMTR-5017, Panel 25

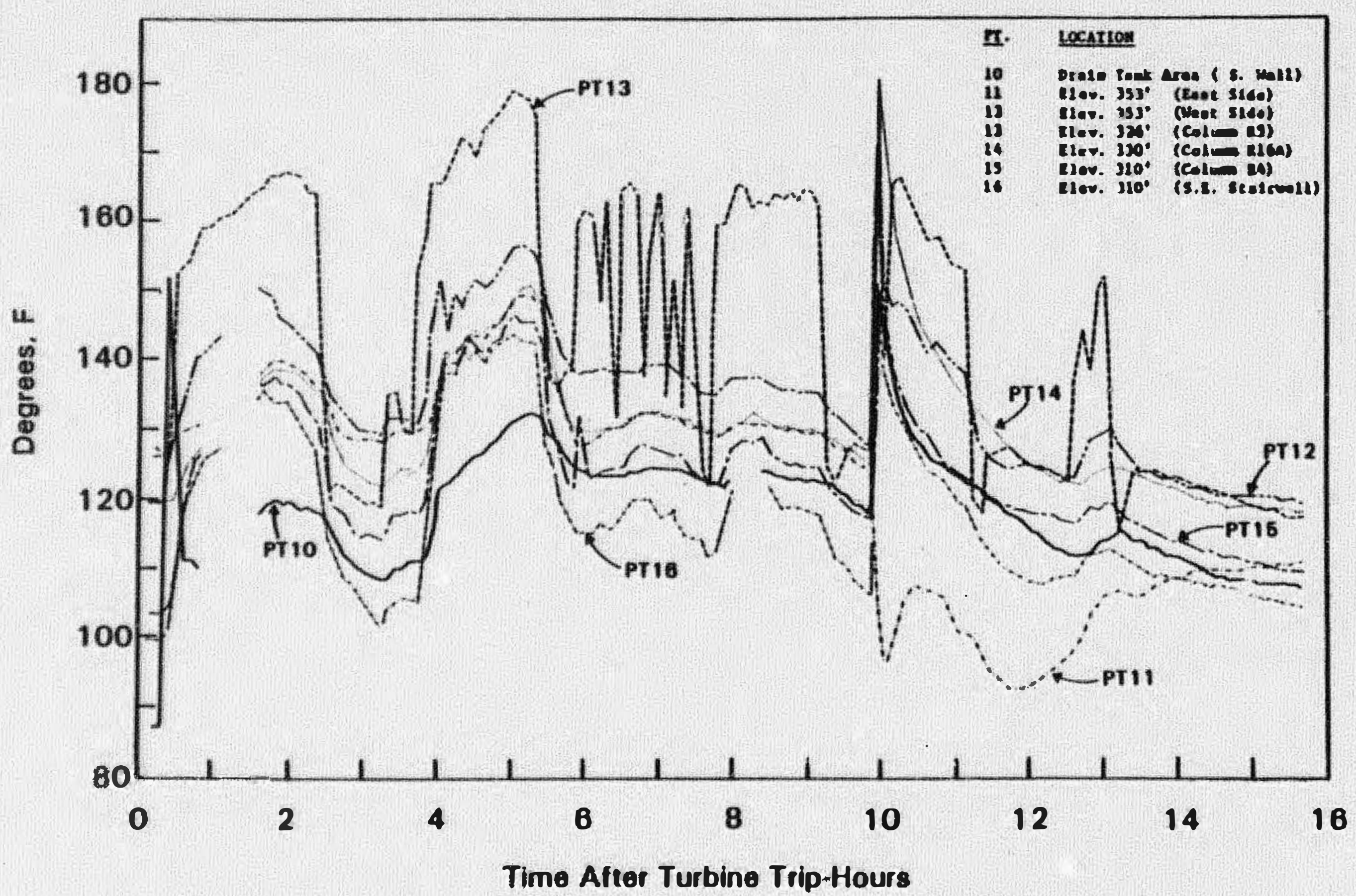
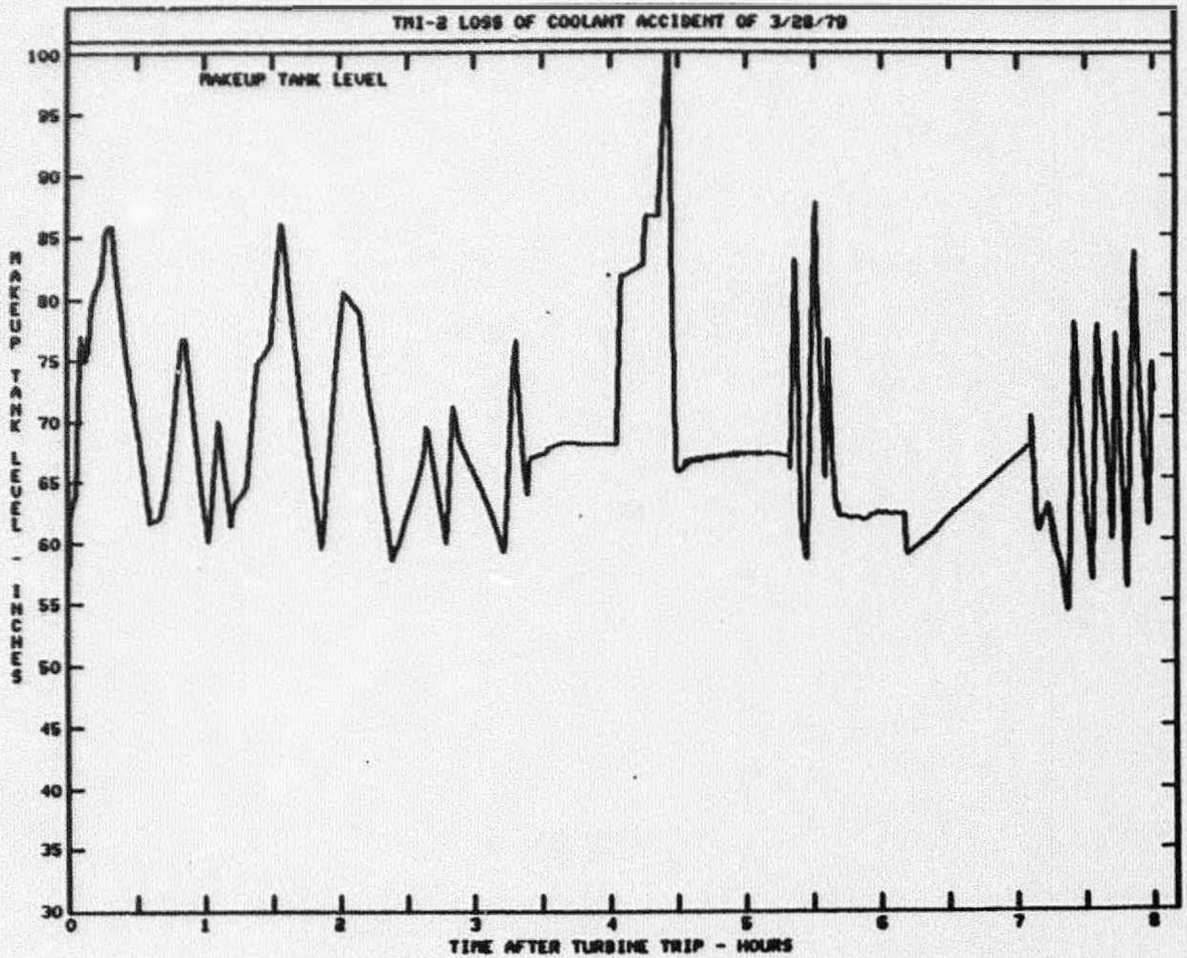


FIGURE 54



TMI-2 LOSS OF COOLANT ACCIDENT OF 3/28/79
MAKEUP TANK LEVEL
MAKEUP TANK LEVEL - INCHES
100
95
90
85
80
75
70
65
60
55
50
45
40
35
30
0
1
2
3
4
5
6
7
8
TIME AFTER TURBINE TRIP - HOURS

Figure 56
TMI-2 Loss of Coolant Accident of 3/28/79
Intermediate and Source Range Nuclear Instrumentation

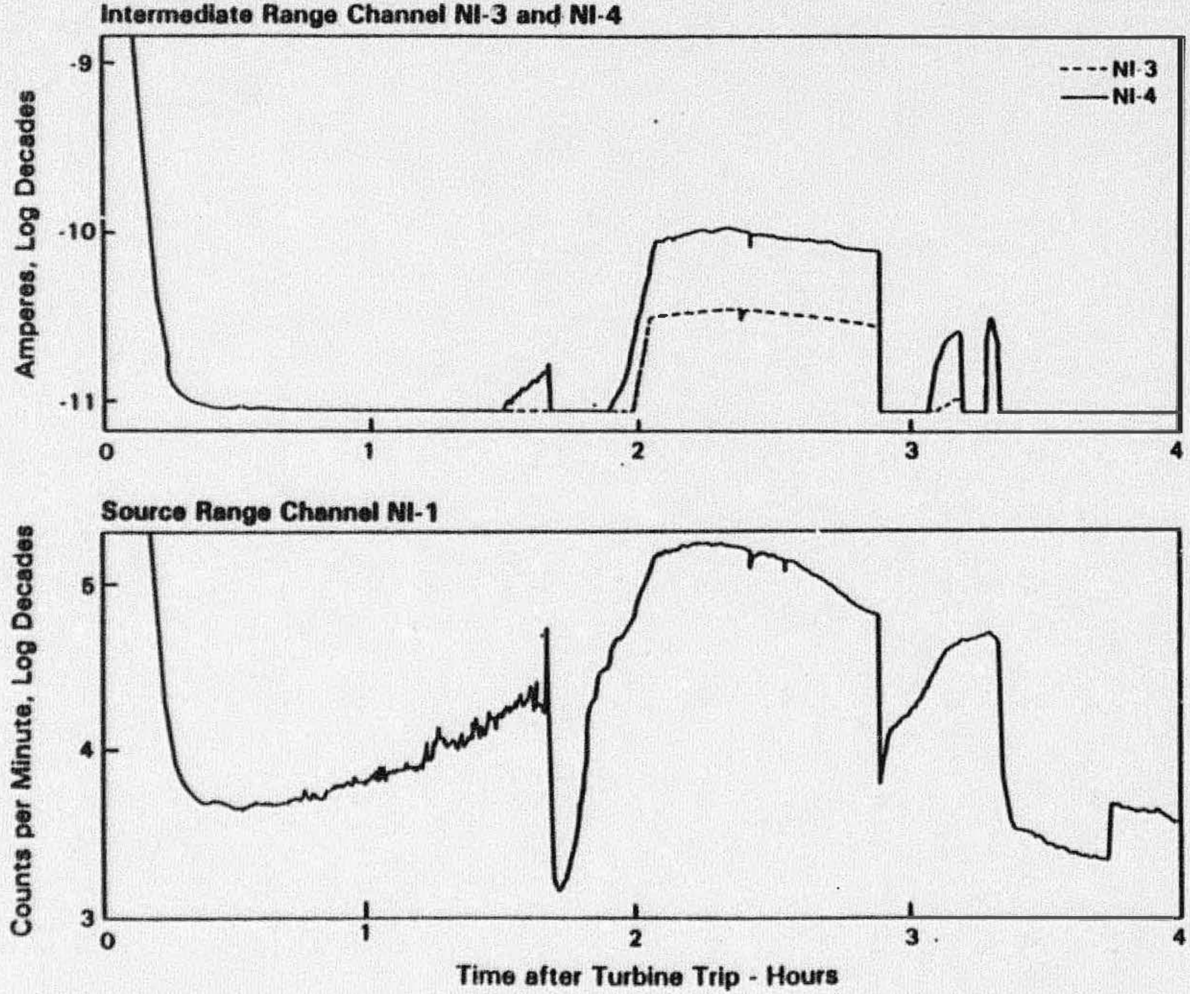


Figure 58
TMI-2 Loss of Coolant Accident 3/28/79
Computer Alarm Printer Lag Time

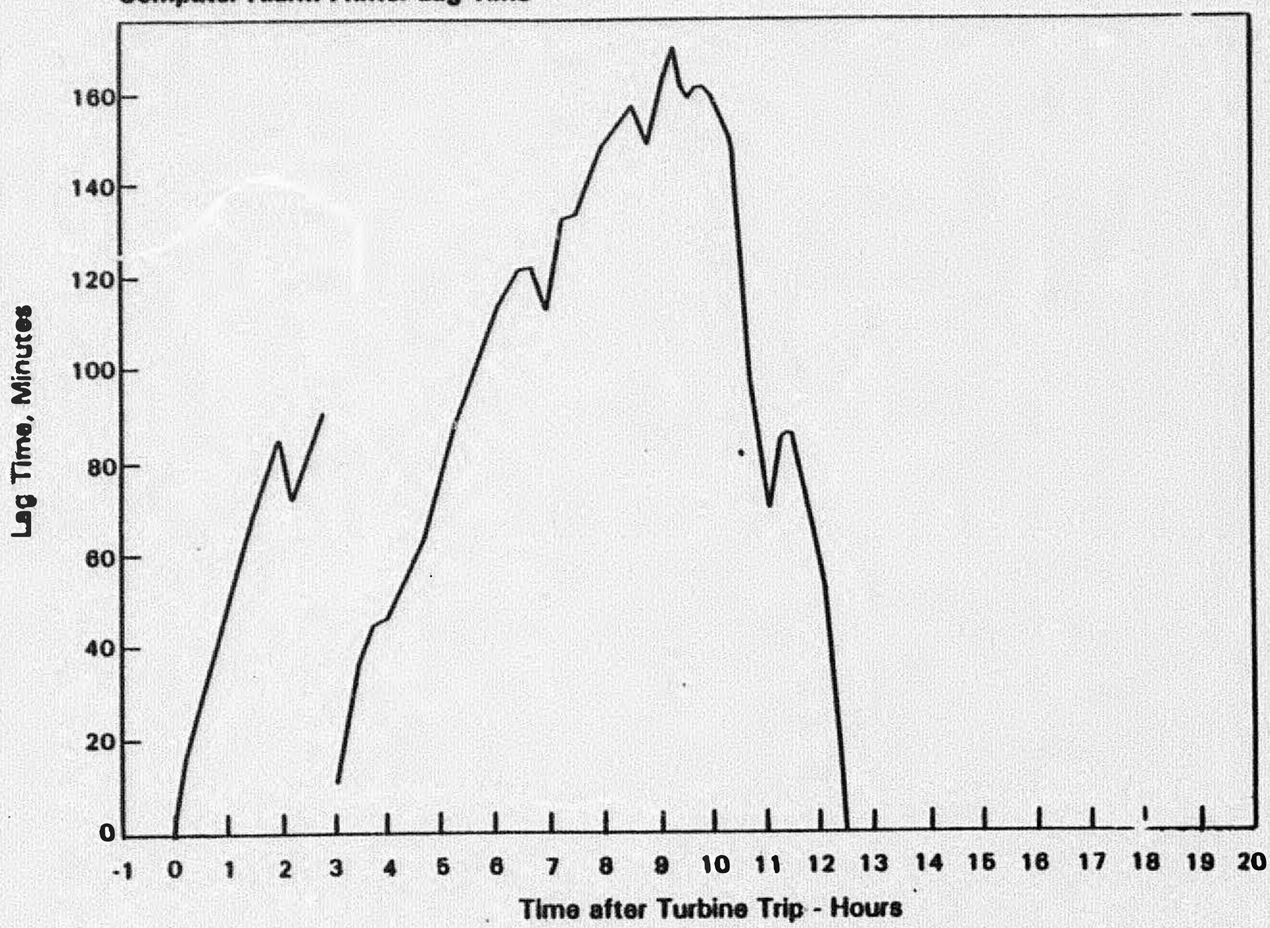
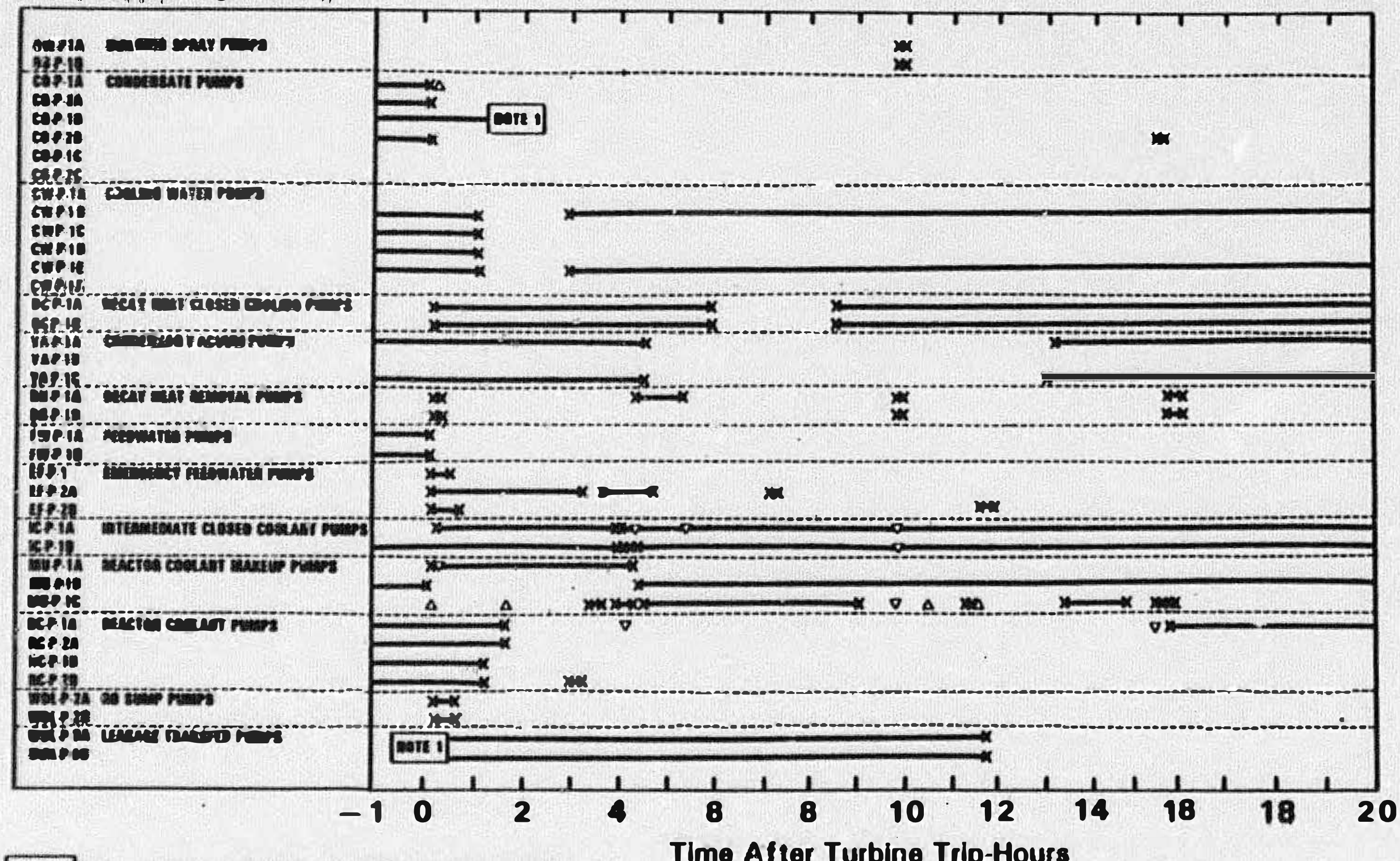


Figure 80
TMI-2 Loss of Coolant Accident 3/28/79
Pump Operating History



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

TMI-2 Loss of Coolant Accident 3/28/79
Electromatic Relief Block Valve (RC-V2) Position

Figure 83

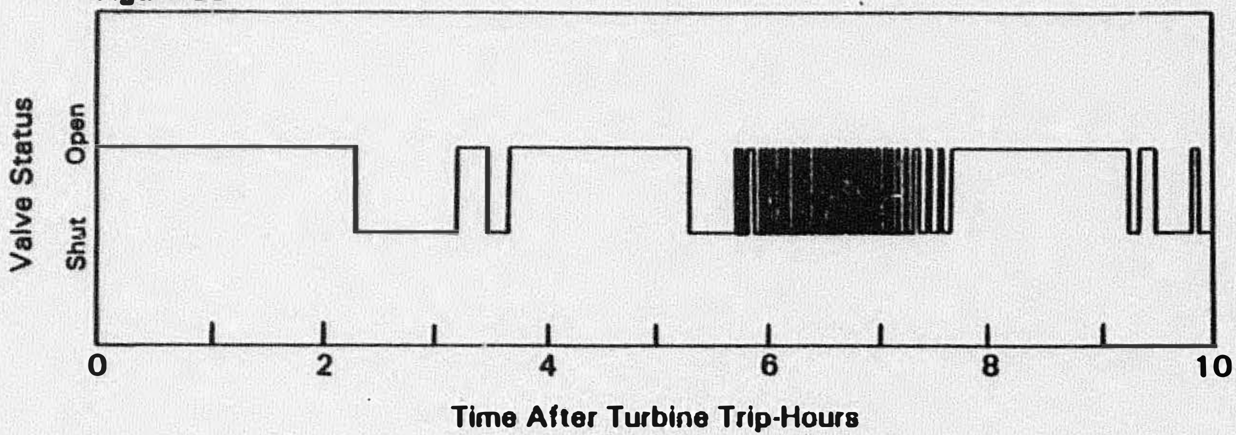
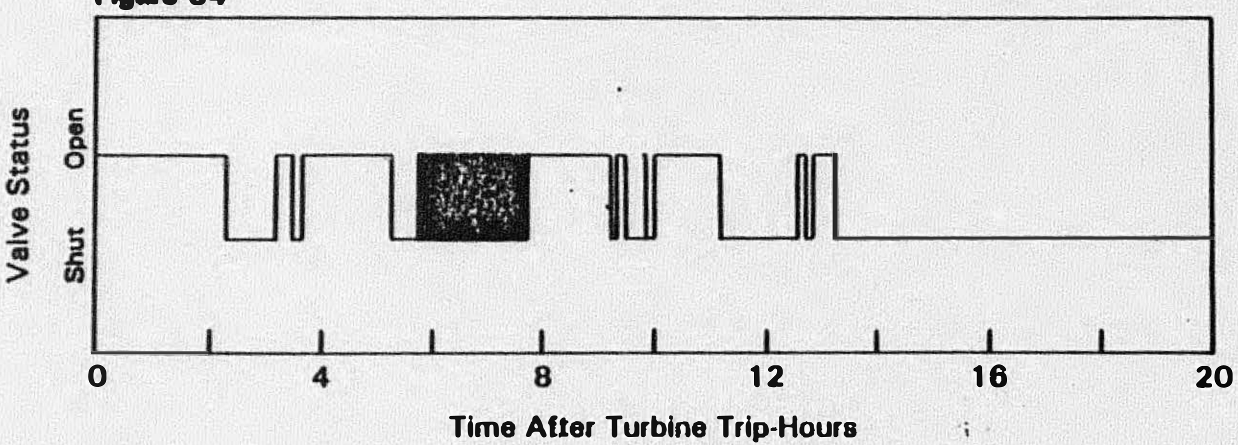


Figure 84



A SUMMARY OF CORRECTIVE ACTIONS

The NRC made a very intensive study of procedural activities carried out prior to and for a short time following the March 28, 1979 accident and published its findings in NUREG-0600, also known as "Office of Inspection and Enforcement Investigation Report Number 50-320/79-10." This report led to the identification of a number of items of concern as set forth in the NRC letter of October 25, 1979. Succeeding correspondence, which addressed corrective actions, included:

*Met Ed letter of December 5, 1979

*NRC letter of January 23, 1980

*Met Ed letter of May 19, 1980

*NRC letter of November 21, 1980

*Met Ed letter of December 15, 1980

*Met Ed letter of January 9, 1981

This series of letters set forth:

*NRC concerns

*Licensee's proposed corrective actions, addressing NRC's concerns

*NRC's response to the proposed corrective actions, indicating acceptability of the latter to the NRC.

An additional corrective action is in the form of a new TMI-2 Emergency Plan, transmitted to the NRC on December 31, 1981 via TLL 700.

We believe that these referenced documents provide the required information on "Corrective Actions." Those corrective actions (long and short term) which uniquely deal with an operating reactor will be addressed at an appropriate time should the decision be made to propose the restart of TMI-2.