THIRD INTERIM REPORT ON THE

THREE MILE ISLAND NUCLEAR STATION

UNIT 2 (TMI-2) ACCIDENT

JULY 16, 1979

7907230633

METROPOLITAN EDISON COMPANY

542 001

CONTENTS

I. Sequence of Events

Included in this section is an expanded version of that contained in the May 15, 1979 report. This version has been formulated using information that was compiled subsequent to May 15 and was not available prior to this date. Since compilation and analysis is continuing, new information gathered subsequent to that contained in this report will be included in a future submittal.

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PRELIMINARY
ANNOTATED SEQUENCE OF EVENTS
MARCH 28, 1979

This report provides additional detail to the May 10, 1979 Revision 0 issue of the Preliminary Annotated Sequence of Events of the March 28, 1979 accident at Three Mile Island Unit 2 and is a result of a detailed analysis of reactimeter data, plant computer data, plant recorder charts, plant logs and operator interviews. Revision 1 of the report includes additional detail on the chronology, the reference source of each entry in the chronology, and the information available to the operator regarding each event in the sequence. The "Information Available to the Operator" entries address the type of information available, the form in which the information was presented, and the timeliness of the presentation of the information to the operator relative to the time of occurrence of the event. Information for which inference or corelations were required prior to providing useful knowledge to the operator is not included in the sequence of events, only that information which provides direct indication of the event is included.

The report should still be considered as preliminary since investigation and data analysis is ongoing and continues to provide new insights into the TMI-2 accident. As each new information and/or understanding is developed this report will be updated.

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

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

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ANNOTATED SEQUENCE OF EVENTS

LIST OF FIGURES

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ANNOTATED SEQUENCE OF EVENTS

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ANNOTATED SEQUENCE OF EVENTS

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ANNOTATED SEQUENCE OF EVENTS

List of References

1. Reactimeter Data (1)

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

3. Plant Stripcharts
   a. Area Gamma Monitors (HP-UR-1901)
   b. Area Gamma Monitors (HP-UR-1902)
   c. Atmospheric Radiation Monitors (HP-UR-2900)
   d. Liquid Radiation Monitors (HP-UR-3264)
   e. Atmospheric Radiation Monitors (HP-UR-3236
   f. Atmospheric Radiation Monitors (HP-UR-1907)
   g. Auxiliary Building Air Flow (AH-FR-5411 and AH-FR-5206)
   h. Fuel Handling Building Air Flow (AH-FR-5709 and AH-FR-5659)
   j. Reactor Building Temperature and Pressure
   k. Intermediate Range and Source Range Monitors
   m. Reactor Coolant System Flow Rate
   n. Reactor Coolant System Wide Range Pressure
   p. Reactor Coolant System Loop A and B, Hot Leg Temperatures (SC0043)

4. Plant Logs

5. Plant Drawings

6. Technical Specifications

7. FSAR

8. TMI Staff Interviews (1)
   a. Ken Bryan by Met-Ed/GPU
   b. Joe Beman by Met-Ed/GPU
   c. Craig Faust by Met-Ed/GPU
   d. Craig Faust by T. Van Witbeck, et.al.
   e. John Flint by Met-Ed/GPU
   f. C. Faust/E. Frederick by W. Marshall
   g. Jim Floyd by Met-Ed/GPU
   h. Ed Frederick by Met-Ed/GPU
   j. Juanita Gingrich by Met-Ed/GPU
   k. Dale Laudermilch by Met-Ed/GPU
   m. H. McGovern by W. Marshall
   n. Brian Mehler by Met-Ed/GPU
   p. Steve Mull by Met-Ed/GPU
   q. Fred Scheimann by Met-Ed/GPU
   r. Bill Zewe by Met-Ed/GPU
   s. Bill Zewe by GPUSC Investigation Team
   t. R. Dubiel, C. Miller and J. Seelinger by Met-Ed/GPU
   u. Sequence prepared by TMI Unit 1 and 2 Staff and Onsite B&W Personnel.
   v. Ed Frederick by T. Van Witbeck, et.al.
   w. Hugh McGovern by Met-Ed/GPU
   x. Don Miller by Met-Ed/GPU

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

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

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ANNOTATED SEQUENCE OF EVENTS

LIST OF SYMBOLS

Indications

SL	Electrical Status Light
ME	Meter
SC	Stripchart Recorder
AN	Annunciator
PL	Control Room Panel
AP	Alarm 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
EFP	Emergency Feedwater Pump
ESF	Engineered Safety Features
HPI	High Pressure Injection
EF	Emergency Feedwater
NI-1	Source Range Monitor
NI-3	Intermediate Range Monitor
RC-P	Reactor Coolant Pump
MU-P	Makeup Pump
FW-P	Feedwater Pump
DH-P	Decay Heat Pump

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This table in conjunction with Figure 54 "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.

July 16, 1979
Rev. 1

PRELIMINARY
ANNOTATED SEQUENCE OF EVENTS TMI 2 ACCIDENT OF MARCH 28, 1979

	Information Available to the Operator	Reference

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

PLANT STATUS

Prior to the accident Three Mile Island Unit Two was at 97% power with the Integrated Control System in full automatic. Rod groups one through five were fully withdrawn, rod groups six and seven were 95% withdrawn and rod group eight was 27% withdrawn. Reactor Coolant System total flow was approximately 138 million pounds per hour and the Reactor Coolant System pressure was 2155 psig. Reactor Coolant Makeup Pump B (MU-P-1B) was in service supplying makeup and Reactor Coolant Pump seal injection flow. Normal Reactor Coolant System letdown flow was approximately 70 gallons per minute. The Reactor Coolant System boron concentration was approximately 1030 parts per million. The Pressurizer Spray Valve (RC-V1) and the Pressurizer Heaters 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 Safety Valves discharge header thermocouples indicated values between 190F to 210F due to leakage through one of the Pressurizer Safety Valves (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.

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Information Available to the Operator | Reference

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

Table of Steam Generator Parameters[1]

	Steam Generator A	Steam Generator B
Loop Feedwater	5.7459 MPPH*	5.7003 MPPH*
Operating Level	56%	57.4%
Startup Level	158.8 Inches	163.6 Inches
Steam Pressure	910 psig	889.6 psig
Feedwater Temperature	462.7F	462.7F
Steam Temperature	595F	594F

* MPPH - Million Pounds Per Hour

[1] The differences between Steam Generator A and B parameters are typical of normal operation.

Steam Generator Feedwater Pumps (FW-P-1A and FW-P-1B), Condensate Booster Pumps (CO-P-2A and CO-P-2B) and Condensate Pumps (CO-P-1A and CO-P-1B) were in service. An attempt was being made to clear a clogged resin transfer line in the standby demineralizer of the Condensate Polishing System.

The Fuel Handling Building supply and exhaust fans were in service. The Auxiliary Building exhaust fans were in service. The status of the Auxiliary Building supply fans is not known.

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582 011

Time	Event	Information Available to the Operator	Reference
-00:00:01 (0400:36)	Condensate Pump A (CO-P-1A) tripped. The trip was a result of a breaker protection relay actuation. The cause of the relay actuation has not been determined.	Annunciator window (AN) at Panel 17 (PL17), meter (MR) indicating motor amperage (A) and electrical status lights (ST) at Panel 5 (PL5), alarm printer (AP) output of norm/trip and on/off (delay time between alarm printer output and real time approximately 0 seconds)	2a, 2b
00:00:00 (0400:37)	Feedwater Pumps (FW-P-1A and FW-P-1B) tripped at essentially the same time resulting in loss of feedwater flow to both Steam Generators.	AN at PL15 and PL17, speed and throttle valve position stripchart recorders (SC) at PL17, speed MR at PL4, pump discharge pressure (P_{DISCH}) MR at PL5, AP norm/trip (Delay $\gtrsim$ 0 seconds)	2a, 2b
00:00:00 (0400:37)	The Main Turbine and Main Generator tripped in accordance with plant design.	Turbine: AN at PL5 and PL17, various MR and ST at PL5, AP norm/trip (Delay $\gtrsim$ 0 seconds) Generator: AN at PL18, various MR and ST at PL6A, AP norm/trip (Delay $\gtrsim$ 0 seconds)	2a, 2b
00:00:00 (0400:37)	All three Emergency Feedwater Pumps (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 $\gtrsim$ 0 seconds)	2a, 2c
00:00:03 (0400:40) Approximate	The Electromatic Relief Valve (RC-R2) opened at the setpoint of 2255 psig.	ST at PL4	1
00:00:08 (0400:45)	The reactor tripped on high Reactor Coolant System pressure at 2345 psig. The setpoint is 2355 psig.	AN (Red/Green/Blue/Yellow) Two out of Four Logic AT, PL 8, Rod Position ST, and MR at PL14, Neutron Flux SC and MR at PL4. AP Two out of Four Logic (Delay $\gtrsim$ 0 seconds)	1, 2a, 2b

512 012

Time	Event	Information Available to the Operator	Reference
00:00:08 (04:00:45) Approximate	The operator placed the Pressurizer Spray Valve (RC-V1) and Pressurizer Heaters under automatic control. Pressurizer Heater Groups 1 through 5 were de-energized as a result of this action. The de-energize setpoint is 2125 psig for Groups 1 through 3 and 2140 psig for Groups 4 and 5 under increasing pressure. Note: There are a total of 13 Pressurizer Heater Groups.	Spray Valve: ST at PL4 Heaters: ST at PL4, AP norm/trip (Delay ≥ 0 seconds)	2a, 8b
00:00:09 (04:00:46)	Steam Generator levels were approximately 120 inches (Figure 34). Steam pressure had increased to 1065 psig in Steam Generator B and 1073 psig in Steam Generator A (Figure 31). Assuming the Steam Generator Safety Valves opened at the specified setpoints, then six of the Steam Generator B Safety Valves and eight of the Steam Generator A Safety Valves opened.	SG L: HR (Startup Range) at PL4, HR (Wide Range) at PL4 SC (Operate Range) at PL4 and PL5 SG P: HR at PL4, SC at PL17	1
00:00:10 (04:00:47) Approximate	The operator verified that all control and safety rods were tripped and fully inserted into the core.	ST at PL4 and PL14, AP norm/trip and yes/no (Delay ≥ 0 seconds)	8c,8h
00:00:12 (04:00:49)	The Reactor Coolant Pressurizer level reached a peak of approximately 225 inches (Figure 28).	SC at PL4, HR (uncompensated) at PL5	1
00:00:13 (04:00:50) Approximate	The operator attempted to start Reactor Coolant Makeup Pump A (MU-P-1A); however, he released the control switch before the required 2.5 seconds and the pump tripped. The operator opened High Pressure Injection Isolation Valve B (MU-V16B) and isolated letdown flow in anticipation of the expected Pressurizer level decrease which follows the initial increase in level after a loss of feedwater flow incident (Figure 28).	MU-P-1A: AN at PL8, ST and HR(A) at PL3, AP norm/trip (Delay ≥ 0 seconds) MU-V16B: ST at PL3, Injection flow HR at PL8 MU-V376: ST at PL3, letdown flow HR at PL3 PZR L: SC at PL4, HR (uncompensated) at PL5	1,2a,2d,8c,8h

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542 013

Time	Event	Information Available to the Operator	Reference
00:00:13 (0400:50)	The Condenser Hotwell low level alarm was received. The level was identified to be 21.77 inches at this time.	HR at PL5, AP low (22.5 inches)/norm/high (36 inches) (Delay ≥ 0 seconds)	2a
00:00:13 (0400:50) Approximate	The Electromatic Relief Valve (RC-R2) should have shut at about this time (closure setpoint of 2205 psig). The Electromatic Relief Valve position indication in the Control Room is a red lamp which illuminates when the Electromatic Relief Valve solenoid is energized. When the lamp is illuminated, i.e. solenoid energized, under normal circumstances the valve should be open. When the lamp is extinguished, i.e. solenoid de-energized, under normal circumstances the valve should be closed; however, it could be in any position as the lamp only indicates the solenoid is de-energized and not valve position. The solenoid on the Electromatic Relief Valve (RC-R2) de-energized at approximately 00:00:13 (0400:50) resulting in an implied "shut" indication in the Control Room. Although the plant operators did not know at the time subsequent events showed that the valve had failed to shut. The loss of reactor coolant through RC-R2 was not stopped until the Electromatic Relief Block Valve (RC-V2) was shut, approximately 2 hours and 19 minutes after the start of the transient.	ST at PL4	1

PLANT STATUS

The plant had just experienced a turbine/reactor trip. Reactor Coolant System pressure and Pressurizer level were decreasing rapidly after reaching peaks of 2365 psig and 256 inches respectively. Unknown to the plant operators the Electromatic Relief Valve (RC-R2) was not shut and was passing reactor coolant from the steam space at the top of the Pressurizer. Based on

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542 014

Time	Event	Information Available to the Operator	Reference
	Control Room indications, the Reactor Coolant System pressure and Pressurizer level were trending together and decreasing as was expected after a reactor trip. The Steam Generators water levels were at about 125 inches and decreasing at about 4 inches per second. The Steam Generator steam pressures were about 1060 psig and decreasing at 32 psig per second. The Turbine Bypass Valves and a number of Main Steam Relief Valves were open and relieving steam. All Emergency Feedwater Pumps had started but had not yet reached normal discharge pressure. The Steam Generator water levels had not yet reached the Integrated Control System setpoint of 30 inches for the programmed opening of the Emergency Feedwater Valves (EF-V11A and EF-V11B) which would admit feedwater to the Steam Generators. In addition, the Emergency Feedwater Block Valves (EF-V12A and EF-V12B) were shut which also prevented feedwater flow until they were opened eight minutes after the start of the transient. The reason for the block valves being shut is not known. The most likely explanation is that the valves were inadvertently left closed after performance of surveillance testing of the Emergency Feedwater System on the morning of March 26, 1979.		
00:00:14 (0400:51)	The Emergency Feedwater Pumps (EF-P-1, EF-P-2A and EF-P-2B) discharge pressure normal alarms were received indicating the pumps had achieved normal discharge pressure (Figure 49).	HR (P_{DISCH}) at PL4, AP low (setpoint = 875 psig)/norm (Delay ≃ 15 seconds)	7a
00:00:14 (0400:51)	Pressurizer Heater Groups 1 through 5 automatically energized as a result of reactor coolant pressure decreasing below the energize setpoint of 2105 psig for Groups 1 through 3 and 2120 psig for Groups 4 and 5.	ST at PL4, AP norm/trip (Delay ≃ 15 seconds)	7a

542 015

Time	Event	Information Available to the Operator	Reference
00:00:14 (0400:51) Approximate	Steam Generator levels were approximately 99 inches (Figure 34). Steam pressure was 1018 psig in Steam Generator B and 1012 psig in Steam Generator A (Figure 31).	SG L: HR (Startup Range) at PL4, HR (Wide Range) at PL4, SC (Operating Range) at PL4 and PL5 SG P: HR at PL4, SC at PL17	1
00:00:15 (0400:52) Approximate	"Water hammer" was noted in the Condensate Pump discharge piping by an auxiliary operator. The piping was displaced several feet according to the auxiliary operator.		8a
00:0018 (0400:55) Approximate	The operator announced on the Plant Page System that TMI Unit 2 turbine and reactor had tripped.		8c
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 31). Subsequently the operator reduced the control setpoint pressure to cooldown the Reactor Coolant System (Figure 31).	Turbine Bypass Valves: HR and ST at PL5 SG P: HR at PL4, SC at PL17	1
00:00:28 (0401:05) Approximate	Steam Generator A level reached the Integrated Control System setpoint of 30 inches at which the Emergency Feedwater Valve (EF-V11A) opens (Figure 34). Feedwater was not admitted to Steam Generator A because Emergency Feedwater Block Valve (EF-V12A) was shut. EF-V12A and EF-V12B are normally open.	SG L: AN (24 inches) at PL17, HR (Startup Range) at PL4, AP low (24 inches)/norm (Delay ≳ 10 seconds) EF-V11A and EF-V11B: MM at PL4 EF-V12A and EF-V12B: ST at PL4	1,8c
00:00:28 (0401:05)	Condenser hotwell level returned to normal at 26.44 inches.	MR at PL5, AP low (22.5 inches)/norm/high (36 inches) (Delay ≳ 10 seconds)	2a

542 016

Time	Event	Information Available to the Operator	Reference
00:00:30 (0401:07)	Pressurizer Safety Valve (RC-R1B) and Electromatic Relief Valve (RC-R2) discharge line temperature alarms were received and values of 203.5F and 239.2F, respectively printed out. The high temperatures in the discharge lines were a result of the high temperature steam flow through the Electromatic Relief Valve (RC-R2) during the reactor pressure transient. The Pressurizer Safety Valve (RC-R1B) did not open; the RC-R1B temperature alarm received was due to the back flow of steam in the common discharge header shared with the Electromatic Relief Valve (RC-R2). These alarms were expected by the operator.	SC at PL10, AP high (200F)/norm (Delay ≥ 30 seconds)	2a
00:00:32 (0401:09) Approximate	Steam Generator B level reached the Integrated Control System setpoint of 30 inches at which the Emergency Feedwater Valve (EF-V11B) opens (Figure 15). Feedwater was not admitted to Steam Generator B because Emergency Feedwater Block Valve (EF-V12B) was shut. As noted above EF-V12A and EF-12B should have been open.	SG L: AN (24 inches) at PL17, HR (Startup Range) at PL4 AP low (24 inches)/normal (Delay ≥ 30 seconds) EF-V11A and EF-V11B: HR 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). With Reactor Coolant Makeup Pumps A and B (MU-P-1A and MU-P-1B) operating, the Pressurizer level rate of decrease slowed (Figure 28).	MU-P-1A: AN at PL8, ST and MR(A) at PL3, AP norm/trip (Delay ≥ 45 seconds)	1,2a
00:01:00 (0401:37) Approximate	Pressurizer level started increasing (Figure 28). Reactor Coolant System hot leg and cold leg temperatures reached approximately 575F (Figure 6). The Reactor Coolant Drain Tank pressure was 12 psig and increasing (Figure 41).	PZR L: SC at PL4, HR (uncompensated) at PL5 RC T_C: SC at PL10 RC T_H: SC at PL4 and PL10, HR at PL4 RCDT P: HR at PL8A	1

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Time	Event	Information Available to the Operator	Reference
00:01:00 (0401:37)	The Pressurizer Safety Valve (RC-R1A) high discharge line 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).	SC at FL10. 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 was not expected as Reactor Coolant System pressure and Pressurizer level should normally trend together during a loss of feedwater flow transient. The deviation from expected behavior was due to: (1) the failure of the Electromatic Relief Valve (RC-R2) to reseat, which resulted in a lower Reactor Coolant System pressure, and (2) the reduction in the heat removing capability of the Steam Generators because of their low levels. Both of these conditions contributed to the expansion of the reactor coolant volume which forced reactor coolant from the Reactor Coolant System loops and Reactor Vessel into the Pressurizer via the surge line, thereby increasing the level of reactor coolant in the Pressurizer. Steam Generator A and B levels were 11 inches and 14 inches, respectively (Figure 34); however, Emergency Feedwater was not admitted to the Steam Generators because the Emergency Feedwater Block Valves (EF-V12A and EF-V12B) were shut. The Steam Generator pressures were being maintained by the Integrated Control System between 975 psig and 1020 psig (Figure 31). The Reactor Coolant Drain Tank pressure and temperature were increasing showing the effects of the continued discharge of reactor coolant through the Electromatic Relief Valve (RC-R2).

542 010

Time	Event	Information Available to the Operator	Reference
	Reactor Coolant Makeup Pumps (MU-P-1A and MU-P-1B) were in operation delivering water to the Reactor Coolant System at a rate in excess of 700 gallons per minute via the High Pressure Injection Valves (MU-V16A and MU-V16B) and the normal Makeup Valves (MU-V17 and MU-V18). The pumps were taking suction from the Borated Water Storage Tank.		9
00:01:13 (0401:50)	The Condenser Hotwell high level alarm was received. The level was 17.77 inches.	MR at PL5. AP low (22.5 inches)/norm/high (16 inches) (Delay ≥ 1 minute)	2a
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 have boiled dry at this time. This was indicated by a steadily decreasing Steam Generator pressure (Figure 8) while Reactor Coolant System hot leg and cold leg temperatures were increasing (Figures 19 and 24).	SG P: MR at PL4 and SG at PL5 RC T: SG at PL4 and PL10, MR at PL4	1
00:02:00 (0402:37) Approximate	The shift supervisor noted all Condensate Pumps, Condensate Booster Pumps and Steam Generator Feedwater Pumps were tripped.		Rs,9
00:02:02 (0402:39)	Engineered Safety Features actuation of High Pressure Injection occurred as Reactor Coolant System pressure reached 1640 psig.	ESF HPI: AN at PL13, SI at PL3 and PL13 AP norm/actuation (Delay ≥ 2 minutes)	2a,2b,7

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Time	Event	Information Available to the Operator	Reference
	Reactor Coolant Makeup Pump B (MU-P-1B) tripped automatically as a result of the actuation of High Pressure Injection. The Engineered Safety Features design is such that Makeup Pumps A and C are used for High Pressure Injection and if Makeup Pump B is running, it is tripped prior to actuation of Makeup Pumps A and C.	MU-P-1B: AN at PL8, ST and MR(A) at PL3 AP norm/trip (Delay ≳ 2 minutes)	
	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 actuation of High Pressure Injection.	DH-P-1A, 1B: ST at PL3 and PL13, MR(A) at PL1, MR(P_{DISCH}) at PL8, AP norm/low and on/off (Delay ≳ 2 minutes)	
00:02:04 (0402:41)	Reactor Coolant Makeup Pump C (MU-P-1C) started automatically.	MU-P-1C: AN at PL8, ST and MR(A) at PL3 AP norm/trip (Delay ≳ 2 minutes)	2a
00:03:12 (0403:49) Approximate	The Reactor Coolant Drain Tank Relief Valve (WDL-R1) lifted at 120 psig temporarily halting the Reactor Coolant Drain Tank pressure increase (Figure 41). The pressure increase was caused by the flow of reactor coolant from the Electromatic Relief Valve (RC-R2). The RCDT Relief Valve (WDL-R1) discharges to the Reactor Building Sump.	RCDT P: MR at PL8A	1
00:03:14 (0403:51)	The operator manually bypassed the High Pressure Injection portion of Engineered Safety Features to gain manual control of the Makeup Pumps and the High Pressure Injection Valves. Both Reactor Coolant Makeup Pumps A and C (MU-P-1A and MU-P-1C) were operating.	ESF HPI: AP norm/bypass (Delay ≳ 3 minutes)	1,2a
00:03:26 (0404:03)	The Reactor Coolant Drain Tank high temperature alarm was received at 127.2F.	RCDT T: MR at PL8A, AP high (120F)/norm/low (75F) (Delay ≳ 3 minutes)	2a

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Time	Event	Information Available to the Operator	Reference
00:04:18 (0405:15)	The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C). The indicated Pressurizer level was 160 inches and increasing rapidly (Figure 29).	MU-P-1C: AB at PL8, ST and MR(A) at PL3, AP note/trip (Delay ≥ 4 minutes) PZR L: AH (HI/HI - 315 inches, HI - 260 inches) at PL8, SC at PL4, MR (uncompensated) at PL5, AP low (200 inches)/norm/high (260 inches) (Delay ≥ 4 minutes)	1,2a
00:04:18 (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).	HPI F: MR at PL8	8b

PLANT STATUS

The Reactor Coolant System pressure was 1420 psig and steadily decreasing to the saturation pressure of the Reactor Coolant System hot leg temperature (Figure 3). The continued Reactor Coolant System depressurization was due to the failure of the Electromatic Relief Valve (RC-R2) to reseat, and reduction in High Pressure Injection flow rate. Engineered Safety Features, which actuated High Pressure Injection when pressure reached 1640 psig, had been bypassed by the operator to permit manual control of the Makeup Pumps and the High Pressure Injection Isolation Valves. As the Pressurizer level continued to increase, the operator stopped Reactor Coolant Makeup Pump C (MU-P-1C) and throttled the High Pressure Injection Isolation Valves (MU-16A and MU-V16B) in an attempt to control the Pressurizer level and not take the Pressurizer "solid" (Figures 3 and 29). The Reactor Coolant Drain Tank Relief Valve (WDL-R1) had opened at 120 psig and was discharging to the Reactor Building Sump (Figure 41). A Reactor Coolant Drain Tank

POOR ORIGINAL

Time	Event	Information Available to the Operator	Reference
	high temperature alarm had been received. 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 19, 24 and 32). 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. A level of 8 inches or less in a Steam Generator is considered indicative of a dry Steam Generator.		
00:04:52 (0405:29)	The operator started Intermediate Closed Cooling Water Pump A (IC-P-1A) in preparation of putting a second Letdown Cooler in service.	ZS at PL8, ST at PL8 and PL13, MR(P_{DISCH}) and MR(F) at PL8, AP on/off (Delay ≳ 5 minutes)	2a
00:04:58 (0405:35)	The operator initiated letdown flow at a rate greater than 160 gallons per minute in an attempt to reduce Pressurizer level to the normal range.	LD Fl: MR at PL3 AP (Range 0-160 gpm) (Delay ≳ 5 minutes)	2a,HI,Hr
00:05:06 (0405:43)	Pressurizer level momentarily stopped its sharp increase at 376 inches and began to decrease. It reached a minimum of 372 inches and again started to increase at 00:05:21 (0405:58) (Figure 29). Maximum Pressurizer level indication is 400 inches. Note: Due to the time scale, this event is difficult to identify on Figure 29.	PZR L: SC at PL4, MR (uncompensated) at PL5	1
00:05:08 (0405:45)	In an attempt to establish condensate flow, the operator started Condensate Pump 1A (CO-P-1A).	AN at PL17, MR(A) and SF at PL5, AP norm/trip and on/off (Delay ≳ 5 minutes)	2a

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542 022

Time	Event	Information Available to the Operator	Reference
00:05:15 (0405:52)	The operator attempted to start Condensate Booster Pump B (CO-P-2B). The pump tripped twice due to low suction pressure. The pump started on the third attempt at 00:05:27 (0406:04).	AN at PL17, HH(A) and SI at PL5, AP note/trip (Delay ≥ 5 minutes)	2a
00:05:50 (0406:27) Approximate	Reactor Coolant System pressure stopped its sharp decrease and began to increase. The minimum value reached was approximately 1340 psig (Figure 1). As the Reactor Coolant System pressure decreased and the Reactor Coolant System saturation temperature increased, the Reactor Coolant System hot legs reached a saturation temperature-pressure relationship at about 1350 psig which resulted in the formation of steam in the hot legs. The pressure turnaround is due to the existence of adequate decay heat within the Reactor Coolant System to generate steam at a rate which exceeds the loss of energy through the Electromatic Relief Valve and other Reactor Coolant System heat sinks.	RC P: HH and SC at PL4 RC T: SC at PL4 and 10, HH at PL4	1, 1a
00:05:56 (0406:33)	Pressurizer level increased beyond the range of the instrument indication.	PZR L: SC at PL4, HH (uncompensated) at PL5	1

POOR ORIGINAL

Time	Event	Information Available to the Operator	Reference
00:05:56 (0406:33) Approximate	The Reactor Coolant System hot leg temperature and pressure reached saturation conditions of 584F and 1351 psig as indicated by the reactimeter data and Control Room wide range Reactor Coolant System pressure stripchart respectively (Figure 3). The Reactor Coolant System flow rate decreased sharply, indicating a reduction in reactor coolant density (Figure 14). The increased reactor coolant volume resulting from the reactor coolant density decrease contributed to the Pressurizer level and pressure behavior at 00:05:50 (0406:31) and 00:05:54 (0406:31)	RC P: MR and SC at PL4 RC T: SC at PL4 and 10, MR at PL4 RC F: MR at PL4	1, 1a
00:06:24 (0407:01)	Condensate Booster Pump B (CO-P-2B) tripped.	AN at PL17, MR(A) and ST at PL5 AP norm/trip (Delay ≥ 5 minutes)	2a
00:06:29 (0407:06)	The operator started Condensate Booster Pump B (CO-P-2B).	AN at PL17, MR(A) and ST at PL5 AP norm/trip (Delay ≥ 5 minutes)	2a
00:06:54 (0407:31)	Letdown Cooler 1A (MU-C-1A) outlet high temperature alarm was received at a value of 139F.	AP norm/high (135F) (Delay ≥ 6 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, 8, 8
00:07:31 (0408:08)	Reactor Building Sump Pump A (WDL-P-2A) started on a high Reactor Building Sump level. The increased sump level was due to the discharge from the Reactor Coolant Drain Tank Relief Valve (WDL-R1) which had been open for approximately 4 minutes. The Reactor Building	AP on/off (Delay ≥ 6 minutes)	2a

Time	Event	Information Available to the Operator	Reference
	Sump Pumps generally started about once per shift. For this reason the pump start would not have been considered extraordinary by the operator.		
	Note: Each Reactor Building Sump Pump has a capacity of approximately 140 gallons per minute as measured from the Reactor Building Sump to the Miscellaneous Waste Holdup Tank. Records indicate the Miscellaneous Waste Holdup Tank level did not change during the incident and it is therefore believed the pump discharge was aligned to the Auxiliary Building Sump Tank. The operators however believe the pump's discharge was aligned to the Miscellaneous Waste Holdup Tank.		
00:08:00 (0408:37) Approximate	The operator discovered the Emergency Feedwater Block Valves (EF-V12A and EF-V12B) were shut. EF-V12A and EF-V12B was opened admitting emergency feedwater to the Steam Generators. Indicated Steam Generator levels were approximately 10 inches just prior to feedwater addition (Figure 35). A rapid rise in Steam Generator A and B pressure was observed when feedwater was admitted to the Steam Generators (Figure 12). Addition of feedwater was also confirmed by a decrease in the Emergency Feedwater Pumps discharge pressure and by "hammering" and "crackling" heard from the Loose Part Monitoring System which was aligned to monitor Steam Generator A (Figure 49).	EF-V12A,B: ST at PL4 SG L: MR at PL4 (startup level), SC at PL4 and PL5 (operating). AP low (24 inches)/norm (Delay ≥ 6 minutes) SG P: MR at PL4, SC (P_{MS}) at PL17 AP low (860 psig)/norm/high (960 psig) (Delay ≥ 6 minutes) EFP P: MR (P_{DISCH}) at PL4	1,26,86
00:08:15 (0408:52)	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 functional as a heat sink for the Reactor Coolant System (Figure 8).	RC T: SC at PL4 and PL10, MR at PL4 SG P: MR at PL4, SC (P_{MS}) at PL17 AP low (860 psig)/norm/high (960 psig) (Delay ≥ 6 minutes)	

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542 020

Time	Event	Information Available to the Operator	Reference
00:08:30 (0509:07)	The Reactor Coolant System pressure began to decrease, reflecting the decrease in Reactor Coolant System temperature (Figures 3 and 6).	RC P: RR and SC at PL4	1
00:08:59 (0509:36)	Condensate Pump 1A (CO-P-1A) tripped.	AN at PL17, RR(A) and ST at PL5 AP norm/trip and on/off (Delay ≥ 7 minutes)	2a
00:09:13 (0509:50)	Condensate Booster Pump suction header pressure low alarm was received. A value of 14.7 psig was recorded.	AP norm/low (15 psig) (Delay ≥ 7 minutes)	2a
00:10:00 (0510:37)	The Pressurizer level indication came on scale.	PZR L: SC at PL6, RR (uncompensated) at PL5	1
00:10:00 (0510:37) Approximate	The operators received Reactor Coolant Pump high vibration alarms.	AN at PL8, AN and RR at PL10	R1,R6

PLANT STATUS

The Reactor Coolant System pressure was near the saturation pressure of the reactor coolant hot leg temperature (Figure 5). This was the result of the Electromatic Relief Valve (RC-RV2) remaining open, high letdown flow rate, throttled High Pressure Injection Isolation Valves and emergency feedwater addition to the Steam Generators. Emergency feedwater flow was admitted to both Steam Generators which resulted in increased steam pressures and the recovery of both Steam Generators as Reactor Coolant System heat sinks as shown by the divergence of Reactor Coolant System hot leg and cold leg temperatures (Figures 6 and 8). Steam pressures were controlled by the Integrated Control System through modulation of the Turbine Bypass Valves. As heat was removed from the Reactor Coolant System, temperature and pressure decreased. The decreasing temperature in conjunction with the letdown flow

POOR ORIGINAL

Time	Event	Information Available to the Operator	Reference
	rate and the reduced High Pressure Injection flow rate resulted in a decrease in reactor coolant volume. The Pressurizer level indication came on scale. Reactor Coolant Makeup Pump A (MU-P-1A) was operating providing Reactor Coolant Pump seal water and makeup flow. Reactor Coolant System letdown flow rate was approximately 70 gallons per minute.		
00:10:19 (0410:56)	Reactor Building Sump Pump B (WDL-P-2B) started. The pump start setpoint is 4.416 feet from the bottom of the Reactor Containment Building Sump. The Reactor Building Sump Pump discharge was aligned to the Auxiliary Building Sump Tank or the Miscellaneous Waste Holdup Tank.	AP on/off (Delay ≥ 8 minutes)	2a,18
00:10:24 (0411:01)	Letdown Cooler 1A (MU-C-1A) outlet temperature returned to normal. A value of 123.7F was recorded.	AP norm/high (135F) (Delay ≥ 8 minutes)	2a
00:10:24 (0411:01)	The operator stopped Reactor Coolant Makeup Pump A (MU-P-1A).	AN at PLB, ST and MR(A) at PL3 AP norm/trip (Delay ≥ 8 minutes)	2a
00:10:27 (0411:04)	The operator started Reactor Coolant Makeup Pump A (MU-P-1A).	AN at PLB, ST and MR(A) at PL3 AP norm/trip (Delay ≥ 8 minutes)	2a
00:10:28 (0411:05)	The operator stopped Reactor Coolant Makeup Pump A (MU-P-1A).	AN at PLB, ST and MR(A) at PL3 AP norm/trip (Delay ≥ 8 minutes)	2a
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 A (MU-P-1A).	AN at PLB, ST and MR(A) at PL3 AP norm/trip (Delay ≥ 8 minutes)	2a

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512 027

Time	Event	Information Available to the Operator	Reference
00:13:13 (0413:50)	The operator stopped Decay Heat Removal Pumps (DH-P-1A and DH-P-1B).	ST at PL11 and PL3, MR(P_{DISCH}) at PCR MR(A) at PL3, AP on/off and norm/trip (Delay ≥ 11 minutes)	2a
00:13:27 (0414:03)	Condensate Booster Pump suction pressure returned to normal. A value of 17.0 psig was recorded.	AP norm/low (15 psig) (Delay ≥ 7 minutes)	2a
00:14:50 (0415:27)	The Reactor Coolant Drain Tank Rupture Diaphragm (WDL-U26) burst at about 190 psig (Figure 41). Design burst pressure is 200 ± 25 psig. The contents of the Reactor Coolant Drain Tank were released to the Reactor Building atmosphere.	RCDT P: AN at PL8A (125 psig), MR at PL8A	1
00:15:43 (0416:20)	Condensate Booster Pump low discharge pressure alarm was received at 187 psi.	AP norm/low (310 psig) (Delay ≥ 11 minutes)	2a
00:16:05 (0416:41)	The operator started Condensate Pump 1A (CO-P-1A).	MR(A) and ST at PL5, AP norm/trip and on/off (Delay ≥ 11 minutes)	2a
00:16:12 (0416:49)	Condensate Booster Pump suction header low pressure alarm was received at 14.9 psig.	AP norm/low (15 psig) (Delay ≥ 11 minutes)	2a
00:19:23 (0420:00) Approximate	The Reactor Building Purge Air Exhaust Duct A radiation monitor (HP-R-225) recorded an increase in radioactivity level. The level increased from 1 x 10^2 counts per minute to 5 x 10^2 counts per minute on the particulate channel. Additionally, there were slight radioactivity level increases noted on: (a) Reactor Building Purge Air Exhaust Duct B (HP-R-226) - particulate monitor	MR and SC on PL12	1c

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Time	Event	Information Available to the Operator	Reference
	(b) Reactor Building Purge Air Exhaust Duct B (HP-R-226) - gas monitor		
	(c) Auxiliary Building Purge Air Exhaust Duct (before filter) (HP-R-222) - gas monitor		
	(d) Auxiliary Building Purge Air Exhaust Duct (after filter) (HP-R-222) - particulate monitor		
00:24:58 (0425:45)	The operator requested the computer to print the outlet temperature (RC-10-TE1, RC-10-TE2 and RC-10-TE3) of the Electromatic Relief Valve (RC-R2) and the Pressurizer Safety Valves (RC-R1A and RC-R1B). Respective values of 285.4F, 263.9F and 275.1F were indicated. The operator attributed the temperature level 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.		2c, 8a, 8b
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. The alarms were believed to be the result of increased background radiation levels caused by the Reactor Coolant Drain Tank discharge of reactor coolant to the Reactor Building Sump.	HR and SC on PL12	8c

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Time	Event	Information Available to the Operator	Reference
00:29:23 (0430:00) Approximate	The Reactor Building Air Sample Line monitor (HP-R-227) gas channel count rate increased from 1×10^3 counts per minute to 5×10^4 counts per minute and then decreased to 1×10^3 counts per minute.	MR and SC on PL12	1e
00:32:23 (0433:00) Approximate	The following radiation monitor readings increased and 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 SC on PL12	11
00:32:36 (0433:13)	Incore thermocouple R-10 indicated signal was out-of-range (Range = 0 to 700 F). It is believed that the incore thermocouple failed.	AP T (Delay $\simeq$ 24 minutes)	2a
00:36:08 (0436:45)	Emergency Feedwater Pump 2B (EF-P-2B) was stopped after filling both Steam Generators to an indicated level of about 35 inches on the startup range (Figure 35).	ST, MR(P_{DISCH}) and MR(A) at PL4 AP on/off and low (875 psig)/norm (Delay $\simeq$ 24 minutes)	1,2a

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542 030

Time	Event	Information Available to the Operator	Reference
00:38:10 (04:38:47)	The auxiliary operator stopped Reactor Building Sump Pump A (WDL-P-2A). The operator believed the pump discharge was aligned to the Waste Holdup Tank and therefore, stopped the pump to prevent the overflow of the Waste Holdup Tank.	AP on/off (Delay ≃ 31 minutes)	2a, 8b
00:38:11 (04:38:48)	The auxiliary operator stopped Reactor Building Sump Pump B (WDL-P-2B). The operator believed the pump discharge was aligned to the Waste Holdup Tank and therefore, stopped the pump to prevent the overflow of the Waste Holdup Tank. NOTE: The two Reactor Building Sump Pumps had operated for 31 and 28 minutes, respectively. Based on the measured capacity of each pump, approximately 8260 gallons of water was transferred to the Auxiliary Building.	AP on/off (Delay ≃ 31 minutes)	2a, 8b
00:40:00 (04:40:37) Approximate	The operator monitored the Reactor Coolant Drain Tank parameters. The Reactor Coolant Drain Tank rupture disc had burst, therefore, the indicated temperature and pressure had reduced substantially and the plant operator did not associate the indication with leakage past the Electromatic Relief Valve (RC-R2), but rather with the initial opening of the relief valve.	RCDT P: HR and AN (125 psig) at PL8A RCDT T: HR at PL8A AP normal/high (Delay ≃ 33 minutes)	8r
00:46:23 (04:47:00) Approximate	Intermediate Cooling Letdown Cooler A monitor (IC-R-1092) increased from 1 x 10^3 counts per minute and eventually peaked at 2 x 10^4 counts per minute.	AN, HR and SC at PL12	14

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542 031

Time	Event	Information Available to the Operator	Reference
00:59:21 (0459:58)	Condensate high temperature alarm was received at 118.5F.	AP normal/high (Delay $\gtrsim$ 48 minutes)	2a
01:00:47 (0501:24)	The operator stopped Circulation Water Pump (CW-P-1C).	HR(A and P) and ST on PL17 AP on/off (Delay $\gtrsim$ 49 minutes)	2a
01:00:49 (0501:26)	The operator stopped Circulation Water Pump (CW-P-1B).	HR(A and P) and ST on PL17 AP on/off (Delay $\gtrsim$ 49 minutes)	2a
01:00:50 (0501:27)	The operator stopped Circulation Water Pump (CW-P-1A).	HR(A and P) and ST at PL17 AP on/off (Delay $\gtrsim$ 49 minutes)	2a
01:00:52 (0501:29)	The operator stopped Circulation Water Pump (CW-P-1D). The operator stopped the Circulating Water Pumps to activate a logic circuit which permits steam generator pressure control via the Power Operated Emergency Main Steam Dump Valves (MSV-3A and MSV-3B) from the control room. Steam Generator Pressure Control was maintained by intermittent use of MSV-3A until 08:30:00 (1230:37) when the valve was shut in response to concerns expressed by the State Government.	HR(A and P) and ST at PL17) AP on/off (Delay $\gtrsim$ 49 minutes)	2a, 8c
01:10:56 (0511:33)	A Reactor Building Air Cooling Coils Emergency Discharge Alarm was received.	HR(P) at PL25 AP normal/high (Delay $\gtrsim$ 59 minutes)	2a

PLANT STATUS:

Both Reactor Coolant System hot leg temperatures and pressures had decreased until they stabilized at a saturation temperature-pressure relationship of 552F and 1050 psig. The Reactor Coolant System loop flow rates had decreased from about 69 million pounds per hour to approximately 47 million pounds per hour and continued to decrease (Figure 15). Reactor Coolant

542 032

Time	Event	Information Available to the Operator	Reference
	Makeup Pump 1A (MU-P-1A) was operating. Letdown flow was in the normal range. Pressurizer level was approximately 352 inches. The Electromatic Relief Valve (RC-R2) was open. The Containment Building temperature and pressure had increased from 0 psig and 120 F to 2.5 psig and 130F, as a result of releasing the contents of the Reactor Coolant Drain Tank to the Reactor Containment Building atmosphere (Figure 45). The operator was having difficulty controlling the level of Steam Generator B. Emergency Feedwater Valves (EF-V11B and EF-V12B) were shut and the operator was admitting feedwater to Steam Generator B by cycling the Emergency Feedwater Crossconnect Valve (EF-V5B).		
01:13:29 (0514:00)	The operator stopped Reactor Coolant Pump 2B (RC-P-2B) to preclude the possibility of damage to the Reactor Coolant Pump from operation near minimum net positive suction head limits. Additional factors which contributed to the operator's decision were high pump vibration and an erratic reactor coolant flow rate.	RC-P-2B: ST, HR(A) and HR(F) at PL4, AN at PL8 AP norm/trip (Delay ≥ * minutes) RCP V: AN at PL8, AN and HR at PL10 RC F: HR and SC at PL4	7b, Ma, Mc, Md, [illegible]
01:13:42 (0514:13)	The operator stopped Reactor Coolant Pump 1B (RC-P-1B) to preclude the possibility of damage to the Reactor Coolant Pump from operation near minimum net positive suction head. Additional factors which contributed to the operator's decision were high pump vibration and erratic reactor coolant flow rate.	RC-P-2B: ST, HR(A) and HR(F) at PL4, AN at PL4 AP norm/trip (Delay ≥ * minutes) RCP V: AN at PL8, AN and HR at PL10 RC F: HR and SC at PL4	7b, Ma, Mc, Md, [illegible]

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

Time	Event	Information Available to the Operator	Reference
01:20:11 (0524:08)	The operator requested the computer print the outlet temperature (RC-10-TE, RC-10-TE7, and RC-10-TE8) of the Electromatic Relief Valve (RC-R2) and the Pressurizer Safety Valves (RC-R1A and RC-R1B). The recorded values were 283.0F, 211.1F, and 218.6F, respectively. The operator continued to believe that the Electromatic Relief Valve (RC-R2) was shut.	RC-R2: ST at PL4	2,
01:26:21 (0527:00)	Steam Generator B was isolated. Main Steam Isolation Valve (MS-V4B and MS-V7B), Emergency Feedwater Valves (EF-V5B, EF-V11B and EF-V12B) and Turbine Bypass Valves (MS-V25B and MS-V26B) were shut. The operator suspected a Steam Generator B to Reactor Building leak based on the large difference in steam pressure 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. After Steam Generator B was isolated the water level continued to increase which lead the operator to suspect a Reactor Coolant side to Feedwater side leak existed in Steam Generator B. An analysis of a water sample from the B Steam Generator taken at 01:21:23 (0724:00) supported this belief.	MS-V4B/7B: ST at PL15 EF-V5B: ST at PL4 EF-V11B: MR at PL4 EF-V12B: ST at PL4	4, [illegible]
01:30:00 (0530:37) Approximate	The reactor out-of-core Intermediate Range Channel (NI-3) indication increased from a minimum detectable indication of less than 1.0×10^{-11} amperes to approximately 1.6×10^{-11} amperes (Figure 46). Correspondingly, the out-of-core Source Range Channel (NI-1) indication increased from about 1.6×10^{4} to approximately 5.3×10^{4} counts per second (Figure 46). 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 core region.	NI-1: MR and SC at PL4 NI-3: MR and SC at PL4	11

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542 034

Time	Event	Information Available to the Operator	Reference
01:34:12 (0534:49)	Steam Generator A boiled dry (Figure 9). This was indicated by a steadily decreasing steam generator pressure while Reactor Coolant hot leg and cold leg temperatures were increasing. In addition Reactor Coolant differential temperature across the Steam Generator F approached zero, which indicated a lack of boiling in the steam generator.	SG P: HR at PL4 and SC at PL 17 SG L: HR (Startup Range) at PL4 AP Low (23.8 Inches) /norm (Delay ≥ *)	1
01:37:00 (0537:37) Approximate	The reactor out-of-core Intermediate Range Channel (NI-3) indication decreased from 2.5 $\times 10^{-11}$ amperes to a minimum detectable indication of 1.0 $\times 10^{-11}$ amperes (Figure 46). The out-of-core Source Range Channel (NI-1) indication had a step decrease from 5.2 $\times 10^{4}$ to 1.5 $\times 10^{3}$ counts per second (Figure 46). This indicated an increased moderator density as a result of liquid displacing steam in the Reactor Vessel.	NI-1: HR and SC at PL4 NI-3: HR and SC at PL4	41
01:40:00 (0540:37) Approximate	The operator started raising Steam Generator A level from 8 inches on the startup range to 50% on the operating range in preparation for establishing of natural circulation (Figures 35 and 38). Reactor Coolant System Loops A and B cold leg temperatures decreased (Figures 20 and 25).	SG L: HR (Startup Range) at PL4, HR (Wide Range) at PL4, SC (Operate Range) at PL4 and PL5 RC T_C: SC at PL10	1,[illegible]
01:40:37 (0541:14)	The Operator stopped the Reactor Coolant Pump 2A (RC-P-2A) to preclude the possibility of damage to the Reactor Coolant Pump from operation near net positive suction head limits. Additional factors which contributed to the Operator's decision were high pump vibration and erratic reactor coolant flow rate.	RC-P-2B: SC, HR(A) and HR(F) at PL4, AN at PL8 AP norm/trip (Delay ≥ * minutes) RCP V: AN at PL8, AN and HR at PL10 RC F: HR and SC at PL4	26

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

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542 035

Time	Event	Information Available to the Operator	Reference
01:40:45 (0541:22)	The Operator stopped Reactor Coolant Pump 1A (RC-P-1A) to preclude the possibility of damage to the Reactor Coolant Pump from operation near minimum net positive suction head limits. Additional factors which contributed to the Operator's decision were high pump vibration and erratic reactor coolant flow rate.	RC-P-2B: ST, HR(A) and HR(F) at PL4, AN at PL8 AP norm/trip (Delay ~ * minutes) RCP V: AN at PL8, AN and HR at PL10 RC F: HR and SC at PL4	26
01:41:00 (0541:37) Approximately	The operator manually initiated high pressure injection to supply additional cooling water to the reactor core. Makeup Pump 1C (MU-P-1C) started. Makeup Pumps, 1A and 1C (MU-P-1A and MU-P-1C) are operating.	HPI: AN at PL13, ST at PL3 and PL13 AP norm/actuation (Delay ~ * minutes)	

PLANT STATUS

The Reactor Coolant System had no forced Reactor Coolant System flow. All Reactor Coolant Pumps (RC-P-1A, RC-P-2A, RC-P-1B and RC-P-2B) had been stopped to preclude the possibility of damage from operation near net positive suction head limits. Additional factors which contributed to the Operator's decision were high pump vibration and an erratic reactor coolant flow rate. (Figure 15). The Reactor Coolant System average temperature and pressure were approximately 534F and 1000 psig, respectively (Figures 11 and 25). Letdown flow rate was in the normal range. Pressurizer level was 354 inches. Reactor Coolant Makeup Pumps 1A and 1B (MU-P-1B) were operating in the High Pressure Injection mode. The operator was attempting to establish natural circulation flow to cool the reactor core. Steam Generator B was isolated because of a suspected Steam Generator B to Reactor Building leak and a Reactor

POOR ORIGINAL

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

542 036

Time	Event	Information Available to the Operator	Reference
	Coolant side to feedwater side leak. Steam Generator B pressure and level were 190 psig and 95 inches respectively. Steam Generator A was steaming to the Atmosphere via the power operated Emergency Main Steam Dump Valve (MS-V3A). Steam Generator A pressure and level were 710 psig and 107 inches respectively.		
01:54:00 (0554:37) Approximate	The reactor out-of-core Intermediate Range Channel (NI-3) indication increased from less than 1.0×10^{-11} amperes to approximately 1.0×10^{-10} amperes (Figure 46). A corresponding increase was recorded on the reactor out-of-core Source Range Channel (NI-1) indication (Figure 46). 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 core region. 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.	NI-1: HR and SC at PL4 NI-2: HR and SC at PL4	11
01:54:00 (0554:37) Approximate	Reactor Coolant System loop A hot leg temperature began to increase, reflecting steam formation in the upper reactor core region (Figure 20).	SC at PL4 and PL10, HR at PL4	1

Time	Event	Information Available to the Operator	Reference
02:00:00 (06:00:37)	Steam Generator A level indication reached 50% on the operating range (Figure 19). This level was established by the operator to induce natural circulation.	SC at PL4 and PL5	1,8a,8b
02:00:00 (06:00:37)	Reactor Coolant System loop B hot leg temperature began increasing (Figure 26).	SC at PL4 and PL10, RR at PL4	1
02:00:00 (06:00:37)	A telephone conference call was made between Unit 2 Technical Superintendent (Unit 2 control room) and the Station Superintendent, Vice President of Generation and Babcock and Wilcox Resident Engineer which lasted approximately 20 minutes.		8a
02:10:55 (06:11:32)	Reactor Coolant System loop A hot leg temperature indication increased offscale, greater than 620F (Figure 21).	AN at PL8 (high at 612F), SC at PL4 and PL10, RR at PL5	1
02:14:23 (06:15:00) Approximate	The Reactor Building Air Sample (HP-R-227) particulate channel increased until it eventually went off scale.	AN, RR and SC at PL12	1c
02:17:53 (06:18:30)	The operator requested the computer print the outlet temperature (RC-10-TE1, RC-10-TE2 and RC-10-TE3) of the Electromatic Relief Valve (RC-R2) and the Pressurizer Safety Valves (RC-R1A and RC-R1B). The recorded values were 228.7F, 189.5F, and 194.2F, respectively.		
02:19:00 (06:19:37) Approximate	The operator shut the Electromatic Relief Block Valve (RC-V2), which stopped reactor coolant leakage through the Electromatic Relief Valve (RC-R2). The operator noted that the Reactor Building pressure started to decrease rapidly (Figure 45).	RC-V2: ST open/shut at PL4 RC-R2: ST open/shut command at PL4 RB Pr: SC at PL13	7c,11,1a,8a

POOR ORIGINAL

Time	Event	Information Available to the Operator	Reference
02:19:00 (06:19:37) Approximate	Reactor Coolant System Pressure started to increase from 680 psig to 2130 psig in the ensuing 41 minutes. Reactor Coolant System Pressure was then maintained at 2130 psig.	AM (Low-2055 psig and Low/Low - 1900 psig) at PL8 MR and SC at PL4	1a
02:24:21 (06:25:00) Approximate	The Reactor Building Air Sample (HP-R-227) gas channel increased until it eventually went off scale.	AM, MR and SC at PL12	1e
02:27:21 (06:28:00)	The alarm printer malfunctioned. The alarm printer function transferred to the utility printer. The alarm printer was 1 hour and 25 minutes behind in logging data (Figure 48).		2a,2b
02:28:21 (06:28:59)	Reactor Coolant Loop B hotleg temperature indication increased offscale, greater than 620F (Figure 26).	AM at PL8 (high at 612F), SC at PL4 and PL10, MR at PL4	1
02:30:00 (06:30:37)	The operator started increasing Steam Generator B level from 30 inches on the Startup Range to 50% on the Operating Range (Figure 19).	MR (Startup Range) at PL4, MR (Wide Range) at PL4 SC (Operate Range) at PL4 and PL5	1
02:31:21 (06:32:00) Approximate	The Incore Instrument Panel Area Monitor (HP-R-213) reading began to increase.	AM, MR and SC at PL12	1a
02:38:21 (06:39:00) Approximate	The Letdown Cooler A monitor (IC-R-1092) 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 Dome (HP-R-212)	AM, MR and SC at PL12	1a,1d

POOR ORIGINAL

542 039

Time	Event	Information Available to the Operator	Reference
02:40:00 (06:40:00) Approximate	The shift supervisor received the results of two boron analyses which indicated the boron concentration in the Reactor Coolant System was approximately 400 parts per million. This, in conjunction with increased neutron levels indicated on the source and intermediate range channels, prompted the shift supervisor to initiate emergency boration of the Reactor Coolant System. NOTE: The actual boron concentration in the Reactor Coolant System was in excess of 1000 parts per million. The samples are believed to have been diluted by distillation in the letdown system. This however was not known by the plant operators until several hours later.		[illegible]
02:44:23 (0645:00) Approximate	Incore Instrument Panel Area Monitor (HP-R-213) indication increased offscale high. The levels recorded on the following 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). The Reactor Building Air sample (HP-R-227) iodine channel indication increased off scale.	AN, HR and SC at PL12	[illegible]
02:45:00 (0645:37) Approximate	Several radiation alarms were received at the Control Room Radiation Monitor Panel.	AN, HR and SC at PL12	[illegible]

POOR ORIGINAL

542 040

Time	Event	Information Available to the Operator	Reference
02:45:00 (0645:37) Approximate	The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C).	AN at PL8, MR(A) and ST at PL3 AP norm/trip (Delay ≥ *) *See entry at time 02:47:31 (0648:08)	[illegible]
02:45:23 (0646:00) Approximate	Fuel Handling Building Monitor (HP-R-218) began to increase.	AN, MR and SC at PL12	[illegible]
02:46:23 (0646:00) Approximate	The operator attempted to start Reactor Coolant Pump 1A (RC-P-1A). The pump would not start.	ST, MR(A), MR(F) and SC(F) at PL4, AN at PL8, AP norm/trip (Delay ≥ *)	
02:47:31 (0648:08)	The operator initialized the alarm summary function to obtain current alarm data. As a result the alarm summary data from 01:13:22 (0513:59) to 02:47:31 (0648:08) was deleted.		2a
02:49:23 (0650:00) Approximate	The following radiation monitors indication were increasing steadily and by 03:20:23 (0721:00) all were off scale: (a) Gas channel of the Station Vent (HP-R-219) monitor (b) Fuel Handling Building Exhaust Duct (HP-R-221,A & B) (c) Hydrogen Purge Duct (HP-R-229) particulate & iodine channels. Condenser Vacuum Pump Exhaust Radiation Monitor (VA-R-748) increased from 1×10^2 to 8×10^5 counts per minute at 0639:23 (0700:00). Note: VA-R-748 is located in the Turbine Building at an elevation of 281'6".	AN, MR and SC at PL12	[illegible]
02:51:57 (0652:34)	The operator attempted to start Reactor Coolant Pump 2A (RC-P-2A). The pump would not start.	ST, MR(A), MR(F) and SC(F) at PL4, AN at PL8, AP norm/trip (Delay ≥ 3 minutes)	2a,[illegible]
02:52:26 (0653:03)	Condenser hotwell level indication returned to normal. A level of 35.94 inches was indicated.	MR(L) at PL5, AP low (22.5 inches)/normal/high (34 inches) (Delay ≥ 3 minutes)	2a
02:53:19 (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) *See entry at time 02:47:31 (0648:08).	2a,[illegible]

POOR ORIGINAL

542 041

Time	Event	Information Available to the Operator	Reference
02:54:09 (0654:46)	The operator started Reactor Coolant Pump 2B (RC-P-2B) and re-established forced Reactor Coolant System flow at a rate 10 million pounds per hour (Figure 16).	ST, RR(A), RR(F) and SC(F) at PL4, AN at PLR, AP norm/trip (Delay ≥ 5 minutes)	1,2a,4e,16
02:54:19 (0654:56)	Pressurizer heater groups 1 through 5 tripped.	ST at PL4 AP on/off (Delay ≥ 5 minutes)	2a
02:55:21 (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".		4e
02:54:33 (0655:10)	The following Incore Thermocouple temperatures were recorded in sequence on the Alarm Printer over the next seven minutes. After these values were recorded the Incore Thermocouples increased until the computer channel range upper limit of 700F was exceeded. 7B - 623.7F 10C - 599.0F 3H - 596.0F 2G - 623.8F 12O - 624.9F 13G - 670.6F 14H - 653.6F 7R - 619.7F 13C - 577.1F 5D - 699.5F 11L - 585.0F 3F - 681.9F	AP norm/bad (out of range 0-700F)(Delay ≥ 6 minutes)	2a
02:54:40 (0655:17) Approximate	Reactor Coolant system Pressure rapidly increased from approximately 1240 psig to approximately 2140 psig (Figure 4).	AN (Low-2055 psig and Low/Low-1900 psig) at PLR	4a
02:54:49 (0655:26)	High Pressure Injection actuation logic of the Engineer Safety Features reset on increasing Reactor Coolant System pressure. The set point is 1865 psig.	AN at PL13, ST at PL3 and PL13 AP norm/trip (Delay ≥ 6 minutes)	2a

POOR ORIGINAL

542 042

Time	Event	Information Available to the Operator	Reference
02:54:50 (0655:27)	The operator started Circulating Water Pump 1B (CW-P-1B).	MR(A and P), ST at PL17, and AP on/off (Delay ≃ 6 minutes)	2a
02:55:00 (0655:37) Approximate	A Site Emergency was declared upon receiving high level radiation alarms from the Condenser Vacuum Pump Exhaust Radiation Monitor (VA-R-748) and Reactor Building Monitor (HP-R-214). Notification of offsite authorities and organizations was initiated.	AN, MR and SC at PL12	7b, 8b
02:55:00 (0655:37) Approximate	The reactor out-of-core Intermediate Range Channel (NI-3) indication decreased sharply from about 8.0×10^{-11} amperes to less than 1.0×10^{-11} amperes (minimum detectable level) (Figure 46). The reactor out-of-core Source Range Channel (NI-1) indication showed a corresponding decrease which indicated the steam in the reactor core region was displaced by liquid. (Figure 46). The displacement of the steam in the reactor core region is attributed to the reestablishment of Reactor Coolant System flow.	NI-1: MR and SC at PL4 NI-3: MR and SC at PL4	8b
02:55:26 (0656:03)	Condenser hot well low level alarm was received. The level was 21.82 inches.	MR at PL5 AP low (22.5 inches)/norm/high (36 inches) (Delay ≃ 11 minutes)	2a
02:55:48 (0656:15)	The operator started Circulating Water Pump 1E (CW-P-1E).	MR (A and P) and ST on PL17 AP on/off (Delay ≃ 11 minutes)	2a
02:59:23 (0700:00) Approximate	Reactor Building Purge Unit Area Monitor (HP-R-3236) and the Fuel Handling Building Area Monitor (HP-R-3240) began to increase. Fuel Handling Building Air Supply Fans were turned off.	AN, MR and SC at PL12	1b, 8b
03:02:56 (0703:33)	Condenser hotwell low level alarm was received again. The level was 9.68 inches.	MR at PL5 AP low (22.5 inches)/norm/high (36 inches) (Delay ≃ 11 minutes)	2a

POOR ORIGINAL

542 043

Time	Event	Information Available to the Operator	Reference
03:03:39 (0704:16) Approximate	The operator isolated Steam Generator B for the second time. Turbine Bypass Valves (MS-V15B, MS-V25B and MS-V26B) were shut. Emergency Feedwater Valves (EF-V5B, EF-V11B, and EF-V12B) were shut. The operator suspected a Reactor Coolant side to Feedwater side leak. The Condenser Vacuum Pump Exhaust Radiation Monitor (VA-R-748) increased to 5 x 10^5 counts per minute.	MS-V25B/26B: ST at PL5 MS-V15B: ST at PL5 EF-V5B: ST at PL4 EF-V11B: MR at PL4 EF-V12B: ST at PL4	2a,8a
03:03:39 (0704:16) Approximate	The reactor out-of core Source Range Channel and Intermediate Range Channel (NI-1 and NI-3, respectively) indication increased approximately a quarter of a decade (Figure 46).	NI-1: MR and SC at PL4 NI-3: MR and SC at PL4	8
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)	Emergency Feedwater Pump 2A (EF-P-2A) was stopped. Both Steam Generators had levels of about 50% on the Operating Range (Figure 39).	EF-P-2A: ST, MR(A) and MR(P_{DISCH}) at PL4 SG L: SC (Operate Range) at PL4 and PL5	1,2a
03:11:10 (0711:47)	Condenser hotwell level returned to normal. The level was 23.07 inches.	MR at PL5 AP low (22.5 inches)/norm/high (36 inches) (Delay ≃ 15 minutes)	2a
03:12:28 (0713:05) Approximate	The operator opened the Electromatic Relief Block Valve (RC-V2) in an attempt to establish Pres ?izer level within the normal operating range. The reactor out-of-score Intermediate Range Channel NI-3 indication decreased sharply as a result of coolant flow through the reactor core (Figure 46).	RC-V2: ST at PL4 NI-3: MR and SC at PL4	1,11,8,16

POOR ORIGINAL

Time	Event	Information Available to the Operator	Reference
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, AM (trip) at PL8 AP norm/trip (Delay ≃ 16 minutes)	2b,3
03:14:23 (0715:00) Approximate	Intermediate Cooling Pump Area Monitor (HP-R-207) indicated an increased radiation level and at 03:20:23 (0721:00) the level stabilized at a value of 100 mR/hr.	AN, MR and SC at PL12	6a
03:18:13 (0718:00) Approximate	The reactor out-of-core Intermediate Range Channel (NI-3) indication increased from less than 1.0×10^{-11} amperes to about 2.7×10^{-10} amperes, which indicated steam was again formed in the reactor core region (Figure 46).	MR and SC at PL4	6b
03:19:45 (0720:22)	The operator manually initiated High Pressure Injection as a result of low Reactor Coolant System pressure (Figure 4). The High Pressure Injection automatic actuation setpoint is 1640 psig.	AM at PL13, ST at PL3 and 13 AP bypass/test/trip (Delay ≃ 19 minutes)	2a,9
03:20:13 (0720:50)	Reactor Coolant Makeup Pump C (MU-P-1C) started automatically. Reactor Coolant Makeup Pumps A and C (MU-P-1A and MU-P-1C) were operating. NOTE: During an Engineered Safety Features actuation, High Pressure 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,7

Time	Event	Information Available to the Operator	Reference
03:20:24 (0721:00) Approximate	The following radiation monitors registered increased radiation levels: (a) Primary Coolant Letdown HI (HU-R-720 HI) (b) Primary Coolant Letdown Lo (HU-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)	AN, HR and SC at PL12	[illegible]
03:21:00 (0721:37) Approximate	The reactor out-of-core Source Range Channel (NI-1) and reactor out of core Intermediate Range Channel (NI-3) indication decreased rapidly indicating flow was established thru the reactor core (Figure 46).	NI-1: MR and SC at PL4 NI-3: MR and SC at PL4	[illegible]
03:21:23 (0722:00) Approximate	The following radiation monitors indicated radiation levels exceeding the monitor range, and remained above range until the stripchart ended at 1100:00, 4/2/79. (a) Reactor Building Purge Air Exhaust Duct A (HP-R-225) - Particulate Monitor (b) Reactor Building Purge Air Exhaust Duct B (HP-R-226) - Particulate Monitor (c) Auxiliary Building Purge Air Exhaust (HP-R-222)	AN, HR and SC at PL12	[illegible]

Time	Event	Information Available to the Operator	Reference

(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.

PLANT STATUS

The Reactor Coolant System was at minimum forced flow condition 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 and a pump running current of less than 100 amps, Reactor Coolant Pump 2B was stopped after 19 minutes. Steam was present in the Reactor Vessel head and Reactor Coolant System hot legs. Both Reactor Coolant System hot leg temperature were off scale high (i.e. greater than 620F). The Reactor Coolant System cold leg temperatures were 455F for loop A and 390F for loop B. Steam Generator B was isolated due to a suspected Reactor Coolant side to feedwater side leak. Steam Generator A pressure was controled was by means of the Power Operated Emergency Main Steam Dump Valve A (MSV-3A). An attempt was in progress to control Reactor Coolant System Pressurizer pressure and level with the Electromatic Relief Block Valve (RC-V2). This resulted in lower Reactor Coolant System pressure which prompted the operator to manually initiate High Pressure Injection. Consequently, both Reactor Coolant Makeup Pumps 1A and 1C (MU-P-1A and MU-P-1C) were operating.

542 047

Time	Event	Information Available to the Operator	Reference
03:23:16 (0723:51)	A High Pressure Injection actuation due to low Reactor Coolant System pressure was received. High Pressure Injection had previously been placed in service by the operator.		2a
03:23:23 (0724:00) Approximate	A General Emergency was declared as a result of an indicated radiation level of 8 R/hr on the Reactor Dome Radiation Monitor (HP-R-212). Notification of offsite authorities and organizations was initiated.	AN, HR and SC at PL12	8a
03:27:23 (0728:00) Approximate	The Radiation level indicated on the Auxiliary Building Access Control Corridor Radiation Monitor (HP-R-232) increased.	AN, HR and SC at PL12	8b
03:29:23 (0730:00) Approximate	The Fuel Handling Building Air Exhaust Fan(s) flow was to zero. Note: During the next 2-1/2 hours the exhaust fan(s) were turned on and off several times with run times of 30 to 60 minutes.		8b
03:30:00 (0730:37) Approximate	The operator shut the Electromatic Relief Block valve (RC-V2). The pressurizer level was 220 inches and the Reactor Coolant System pressure was 1480 psig (Figures 4 and 12).	RC-V2: ST at PL4 PZR L: SC at PL4, HR (uncompensated) at PL5 RC P: HR and SC at PL4	1, 31
03:35:08 (0735:43)	The operator started Emergency Feedwater Pump 2A (EF-P-2A). Steam Generator A level had been continuously falling from 68% to 44% of Operating Range during the previous 45 minutes (Figure 39).	EF-P-2A: ST, HR(P_{DISCH}) and HR(A) at PL4, AP on/off and P_{DISCH} low/normal (Delay $\gtrsim$ 40 minutes) SG L: SC (operating range) at PL4 and PL5	2a
03:37:00 (0737:37)	The Reactor Coolant Makeup Pump C (MU-P-1C) was stopped because pressurizer level was rapidly increasing. Indicated Pressurizer level was 350 inches (Figure 4).	MU-P-1C: ST and HR(A) at PL3, AN at PL8 AP norm/trip (Delay $\gtrsim$ 42 minutes) PZR L: ST at PL4, HR (uncompensated) at PL5	1, 2a

542 049

Time	Event	Information Available to the Operator	Reference
03:48:23 (07:49:00) Approximate	Makeup Tank Area Monitor (HP-R-206) advanced off scale. Levels in Fuel Handling Bridge S. (HP-R-210) and Reactor Dome (HP-R-214) stabilized at 1.5 x 10^{2} R/hr.	AN, PM and SC at PL12	1c
03:51:00 (07:51:37) Approximate	The operator opened the Electromatic Relief Block Valve (RC-V2) in an attempt to decrease pressurizer level, which had increased to 395 inches (Figure 4).	RC-V2: ST at PL4. PZR L: SC at PL4, RR (uncompensated) at PL5	1, 11, 30
03:55:38 (07:56:15)	The operator stopped Intermediate Cooling Pump 1B(IC-P-B).	AN, ST, MR(P_{DISCH}) and MR (F) at PL8 AP on/off (Delay ≳ 46 minutes)	2a
03:55:39 (07:56:16)	Engineered Safety Features actuated on Reactor Building high pressure at the setpoint of 3.58 psig (Figure 45).	AN at PL13, ST at PL3 and PL15, AP act/trip (Delay ≳ 46 minutes)	2a
03:55:39 (07:56:16)	The Reactor Building isolated automatically as part of the Engineered Safety Features actuation from Reactor Building high pressure. Reactor Building isolation occurred at the nominal setpoint of 4 psig.	AN at PL13, ST at PL3 and PL15, AP isolation/norm (Delay ≳ 46 minutes)	2a
03:55:46 (07:56:23)	The operator stopped Intermediate Cooling Pump 1A (IC-P-1A).	AN, ST, MR(P_{DISCH}) and MR(F) at PL8 AP on/off (Delay ≳ 46 minutes)	2a
03:55:46 (07:56:23)	Reactor Coolant Makeup Pump C (MU-P-1C) was started automatically by the Engineered Safety Features actuation.	AN at PL8, ST and MR(A) at PL3 AP norm/trip (Delay ≳ 46 minutes)	2a
03:59:23 (08:00:00)	Reactor Building Emergency Cooler B was shutdown.	ST at PL3	2a

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542 049

Time	Event	Information Available to the Operator	Reference
03:59:23 (0800:00) Approximate	Waste Gas Tank Discharge A (WDG-R-1485) monitor began to increase from 5 x 10^2 counts per minute until it reached 3 x 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.		6
03:59:53 (0800:30)	Reactor Building Emergency Cooler B was started automatically by the Engineered Safety Features actuation.	AP T (0-200F) (Delay ≃ 47 minutes)	2a
04:00:13 (0800:50)	The operator started Intermediate Cooling Pump 1B (IC-P-1B).	AN, ST, MR(P_{DISCH}) and MR(F) at PLB AP on/off (Delay ≃ 47 minutes)	2a
04:00:19 (0800:56)	The operator started Intermediate Cooling Pump 1A (IC-P-1A).	AN, ST, MR(P_{DISCH}) and MR(F) at PLB AP on/off (Delay ≃ 47 minutes)	2a
04:08:37 (0809:14)	The operator started Reactor Coolant Pump 1A (RC-P-1A) to re-establish Reactor Coolant System flow. NOTE: During the previous run of Reactor Coolant Pump 2B (RC-P-2B), due to the flow and current indication observed, it was thought that the pump might not have started. For this reason it was decided to observe the starting current during a Reactor Coolant Pump start. Reactor Coolant Pump 1A (RC-P-1A) was started and a correct starting current was observed by the operator. As before, the indicated pump current rapidly decreased to less than 100 amperes.	ST, MR(A), MR(F) and SC(F) at PL4, AN at PLB AP norm/trip (Delay ≃ 51 minutes)	2a
04:09:14 (0809:51)	The operator stopped Reactor Coolant Pump 1A (RC-P-1A) after observing a no-flow indication and a running current less than 100 amperes.	ST, MR(A), MR(F) and SC(F) at PL4, AN at PLB AP norm/trip (Delay ≃ 51 minutes)	2a

POOR ORIGINAL

542 050

Time	Event	Information Available to the Operator	Reference
04:10:10 (0810:47)	The operator stopped Intermediate Cooling Pump 1B (IC-P-1B).	AN, ST, MR(P_{DISCH}) and MR(F) at PL8 AP on/off (Delay ≳ 53 minutes)	2a
04:17:17 (0817:54)	The operator stopped Reactor Coolant Makeup Pump 1A (MU-P-1A).	ST and MR(A) at PL3, AN at PL8, AP norm/trip (Delay ≳ 55 minutes)	2a
04:17:22 (0817:59)	The operator stopped Reactor Coolant Makeup Pump 1C (MU-P-1C). No Reactor Coolant Makeup Pumps were operating.	ST and MR(A) at PL3, AN at PL8, AP norm/trip (Delay ≳ 55 minutes)	2a
04:18:17 (0818:56)	The operator placed Makeup Pump 1A (MU-P-1A) control switch in Pull-to-Lock position to prohibit further use of MU-P-1A due to the pump operating problems experienced.	Control switch handle position	2a,9
04:19:02 (0819:39)	The operator started Intermediate Cooling Pump 1B (IC-P-1B).	AN, ST, MR(P_{DISCH}) and MR(F) at PL8 AP on/off (Delay ≳ 55 minutes)	2a
04:21:53 (0822:39) Approximate	The operator started Reactor Coolant Makeup Pump 1B (MU-P-1B).	ST and MR(A) at PL3, AN at PL8, AP norm/trip (Delay ≳ 55 minutes)	2a
04:23:54 (0824:31)	Pressurizer heater groups 1 through 5 energized. All pressurizer heater groups were available at this time.	AN at PL8, ST at PL4, AP on/off (Delay ≳ 58 minutes)	2a
04:26:59 (0827:39) Approximate	The operator started Reactor Coolant Makeup Pump 1C (M[illegible] 1C).	ST and MR(A) at PL3, AN at PL8, AP on/off (Delay ≳ 58 minutes)	2a
04:30:30 (0831:07)	Pressurizer heater group 10 tripped and remained off for the remainder of March 28.	AN at PL8, ST at PL4, AP on/off (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,8a

POOR ORIGINAL

542 051

Time	Event	Information Available to the Operator	Reference
04:30:55 (0831:22) Approximate	The operator opened Power Operated Emergency Main Steam Dump Valve A (MS-V1A) to induce natural circulation in Steam Generator A. Steam Generator B was still isolated.	MR (Valve demand setpoint) at PL5	8f,8n,8o
04:44:23 (0845:00)	Intermediate Cooling Letdown Cooler A Radiation Monitor (IC-R-1092) indication decreased and went off scale. Note: Monitor failure was assumed.	MR and SC at PL12	6d
04:46:21 (0846:58)	Pressurizer heater groups 4 and 5 tripped and remained off for the remainder of March 28. There were 10 pressurizer heater groups available at this time.	AN at PL8, ST at PL4 AP on/off (Delay ≥ 63 minutes)	2a
04:49:23 (0850:00) Approximate	Condenser Vacuum Pump Exhaust radiation monitor (VA-R-748) decreased to 1×10^4 counts per minute.	MR and SC at PL12	6e
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-206) indications began to increase.	MR and SC at P12	6a
05:17:38 (0918:05)	The alarm printer was returned to service and the alarm function was transferred from the utility printer to the alarm printer.	AP (Delay ≥ 0 minutes)	2a
05:18:00 (0918:17) Approximate	The operator closed the Electromatic Relief Block Valve (RC-V2) in an attempt to compress the reactor coolant and condense the steam in the Reactor Coolant System.	ST at PL4	1j,3n,8o

PLANT STATUS

All Reactor Coolant Pumps (RC-P-1A, RC-P-2A, RC-P-1B and RC-P-2B) were stopped. Superheated steam/gas was present in the upper Reactor Vessel

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Time	Event	Information Available to the Operator	Reference
	and hot leg regions. Attempts to re-establish reactor coolant flow using Reactor Coolant Pump 1A (RC-P-1A) had not been successful. The Reactor Coolant hot leg temperature continued to read off-scale (i.e. greater than 620F). The Reactor Coolant cold leg temperatures were 180 F for Loop A and 225F for Loop B, and both were decreasing (Figures 21 and 26). Steam Generator A level was at 48% of the operating range (Figure 39). Steam Generator B was isolated, with a level at 66% on the operating range (Figure 39). Condenser vacuum was lost due to the auxiliary steam boiler tripping and loss of adequate main steam pressure. Steam Generator A was steaming through the Power Operated Emergency Main Steam Dump Valve (MS-V3A). Attempts to obtain a normal operating Pressurizer level of 220 inches of water and establish pressure control using the pressurizer were not successful. The Electromatic Relief Block Valve (RC-V2) was cycled to assist in this effort, resulting in increased Reactor Building pressure. The first Engineered Safety Features actuation on high Reactor Building pressure was received and, four minutes later, bypassed by the operator to re-establish cooling water to various plant equipment within the Reactor Building. The Reactor Building pressure continued to stay above the isolation trip setpoint for approximately 2.6 hours (Figure 45). 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/gas in the Reactor Coolant System. This first attempt lasted for approximately 2 hours.		
05:20:00 (0920:17) Approximate	The operator increased Reactor Coolant System pressure from 1250 psig to 2100 psig during the ensuing 45 minutes. Reactor Coolant System pressure was then maintained at 2100 psig (Figure 4).	AN (Low-2055 and Low/Low-1900) at PLA HR and SC at PLA AP (many clearing alarms) (Delay ≃ 100 minutes)	[illegible]

Time	Event	Information Available to the Operator	Reference
05:29:21 (0930:00) Approximate	Fuel Handling Building Radiation Monitor (HP-R-215) and Control and Service Building Corridor Radiation Monitor (HP-R-214) began to increase to between 50 and 70 counts per minute. The radiation levels indicated by the following monitors increased until 14:59:21 (1900:00) at which time all were almost off scale. (a) Auxiliary Building Access Corridor Radiation Monitor (HP-R-232) (b) Reactor Building Purge Unit Area Radiation Monitor (HP-R-3236) (c) Fuel Handling Building Exhaust Unit Area Radiation Monitor (HP-R-3240)	SC and HR at PL12	[illegible]
05:30:34 (0931:11)	Pressurizer heater group 3 tripped leaving 9 available pressurizer heater groups.	AH at PL8, ST at PL4 AP on/off (Delay ≥ 93 minutes)	2a
05:49:21 (0950:00) Approximate	Radiation levels as indicated by the Intermediate Cooling Pump Area Radiation Monitor (HP-R-207) and the Reactor Building Emergency Cooling Booster Pump Area Radiation Monitor (HP-R-204) peaked at 4×10^{1} mR/hr. All Control Room Intake Duct radiation monitor (particulate, iodine, gas) (HP-R-220) level increased. The particulate channel reached 1×10^{4} counts per minute while the iodine and gas channels reached 3×10^{3} counts per minute.	SC and HR at PL12	1c
05:56:00 (0956:37) Approximate	The operator commenced filling Steam Generator A to 95% on the operating range to induce natural circulation. Steam Generator A level reached 100% at 07:30:00 (1130:37) (Figure 39).	SG Lt HR (startup range) at PL4, HR (wide range) at PL4 SC (operating range) at PL4 and PL5	1
05:57:00 (0957:37)	The Reactor Coolant System pressure reached a value of 2200 psig. The operator opened the Electromatic Relief Block Valve (RC-V2) to stop the pressure increase and reduce pressure to about 2050 psig.	RC Pt HR and SC at PL4 RC-V2: ST at PL4 RB Pt SC at PL3	11, 1a, 6a

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Time	Event	Information Available to the Operator	Reference
	During the period 05:57:00 (0957:37) thru 07:45:00 (1145:37), 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).	RB T: SC at PL25	
05:59:23 (1000:00) Approximate	The Auxiliary Building Air Exhaust Fans stopped.	SC and ST at PL25	1g
06:10:00 (1010:37) Approximate	Airborne radioactivity levels in Unit 2 control room required evacuation of all but essential personnel.		4,8a
06:13:39 (1014:16)	Pressurizer heater groups 1 and 2 tripped. Seven pressurizer heater groups were available at this time.	AN at PL8, SF at PL4, AP on/off (Delay ≥ 116 minutes)	2a
06:14:06 (1014:43)	Pressurizer heater groups 1 and 2 re-energized making 9 heater groups available.	AN at PL8, ST at PL4, AP on/off (Delay ≥ 116 minutes)	2a
06:14:23 (1015:00) Approximate	The Auxiliary Building Air Exhaust Fans restarted.	SC and ST on PL25	1g
06:17:00 (1017:37) Approximate	Personnel in Unit 2 control room were required to wear respirators due to increased airborne radioactivity levels.		4a,8a
06:19:23 (1100:00) Approximate	The operator started the Fuel Handling Building Air Exhaust Fans. Control Room Intake Duct Radiation Monitor (HP-R-220) level decreased to below 100 counts per minute.	Fans: SC and ST at PL25; Radiation Monitor: SC and MN at PL12	1f, 8a

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Time	Event	Information Available to the Operator	Reference
07:30:00 (1130:37) Approximate	The Station Manager directed the operator to open the Electromatic Relief Block Valve (RC-V2) and the Pressurizer Spray Valve (RC-V1) to rapidly depressurize the Reactor Coolant System and actuate the Core Flood System while High Pressure Injection was maintained (Figure 17). This was done after the operator observed no evidence of natural circulation while the Reactor Coolant System pressure was above 2000 psig. The reduction in Reactor Coolant System pressure was also done to approach conditions which would allow the Decay Heat Removal Pumps 1A and 1B (DH-P-1A and DH-P-1B) to be put into service.	RC-V2: ST at PL4 RC-V1: ST at PL4	1, 6a,8a
07:41:37 (1142:14)	The Operator bypassed Engineered Safety Features channels A and B to prevent actuation of High Pressure Injection during the Reactor Coolant System depressurization.	ESF Bypass: ST at PN1, AP norm/test (Delay ≥ 144 minutes)	2a,9
07:43:44 (1144:21)	Pressurizer heater groups 1 and 2 tripped and re-energized after 2 seconds.	ST at PL4, AP norm/trip (Delay ≥ 145 minutes)	2a
07:44:23 (1145:00) Approximate	The Auxiliary Building Air Exhaust Fans stopped.		
07:50:16 (1150:53)	Pressurizer heater groups 1 and 2 tripped. Seven Pressurizer heater groups were available at this time.	ST at PL4 AP norm/trip (Delay ≥ 145 minutes)	2a
08:11:26 (1212:03)	Core Flood Tank 1A (CF-T-1A) high level alarm was received. The level was 13.32 feet.	AN and MP at PL8 AP norm/high (13.3 feet) (Delay ≥ 150 minutes)	2a

Time	Event	Information Available to the Operator	Reference
08:30:00 (12:30:37)	The Power Operated Emergency Main Steam Dump Valve (MS-V3A) was shut at the request of corporate management in response to concern expressed by the state government.	MR (valve demand setpoint) AT PL5	8a
08:31:06 (12:31:43)	The operator started Decay Heat Removal Closed Cooling Pumps 1A and and 1B (DH-P-1A and DH-P-1B) in preparation for eventually placing the Decay Heat System in service.	ST at PL3 and PL13, MR(P_{DISCH}) at PL8, MR(A) at PL3, AP norm/trip and on/off (Delay ≥ 155 minutes)	2a
08:37:00 (12:37:37) Approximate	Reactor Coolant System pressure reached 600 psig which is the nominal gas pressure maintained in the core flood tanks.	MR and SC at PL4	2a, 6a
08:56:56 (12:55:33)	Core Flood Tank 1A (CF-T-1A) normal level alarm was received. The level was 13.13 feet. This indicated that 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, 6a
09:04:18 (1306:55)	The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C) and returned the Makeup System to normal one pump operation.	AN at PL8, St and MR(A) at PL3, AP norm/trip and norm/off (Delay ≥ 150 minutes)	2a
09:15:00 (1314:37) Approximate	The operator shut the Electromatic Relief Block Valve (RC-V2).	ST at PL4	11
09:49:23 (1350:00) Approximate	The Auxiliary Building Air Exhaust fans restarted momentarily and then stopped.	SC at PL25	5c
09:49:43 (1350:20)	The operator opened the Electromatic Relief Valve (RC-R2) and a hydrogen detonation occurred in the Reactor Building. Hydrogen gas resident in the reactor coolant and hydrogen gas generated from the reaction between zirconium fuel cladding and the reactor coolant	ST at PL4	2b, 1, 8m, 8u, 9

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Time	Event	Information Available to the Operator	Reference
	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 eventually reached an explosive mixture. It is believed the detonation was a result of a set of RC-R2 contacts arcing when the operator opened the valve.		
09:49:44 (1350:21)	Engineered Safety Features actuation occurred on high-high reactor building pressure (Figure 45). The setpoint is 28 psig. This is the result of a 28 psig building pressure impulse from the hydrogen detonation. Reactor Building Isolation and Containment Spray were actuated. Reactor Coolant Makeup Pump C (MU-P-1C)), started at 09:48:46 (1350:23) and Reactor Building Spray Pumps A and B (BS-P-1A and BS-P-1B) started automatically.	ESF: AN at PL13, ST at PL3 and PL13 AP norm/act (Delay ≈ 159 minutes) MU-P-1C: AN at PL8, MM(A) and ST at PL3 AP norm/trip (Delay ≈ 159 minutes) BS-P-1A/1B: ST at PL13 and PL15 AP norm/trip (Delay ≈ 159 minutes)	2a,2b,11,6,7
09:49:50 (1350:27)	Reactor Building Spray Valves (BS-V1A and BS-V1B) opened.	ST at PL13 and PL15	7
09:49:58 (1350:35)	Reactor Coolant Pumps 1A and 1B (RC-P-1A and RC-P-1B) high inlet air temperature alarms annunciated and Pressurizer Safety Valves (RC-R1A and RC-R1B) high discharge line temperature alarms annunciated.	RC P: AP norm/high (122 F) (Delay ≈ 161 minutes) RC-R1A,B: AP norm/high (Delay ≈ 161 minutes)	2a
09:50:24 (1351:01)	The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C)	AN at PL8, ST and MM(A) at PL3 AP norm/trip (Delay ≈ 162 minutes)	2a
09:55:10 (1355:47)	Pressurizer heater group 8 tripped leaving 6 heater groups available.	ST at PL4 AP norm/trip (Delay ≈ 160 minutes)	2a

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542 008

Time	Event	Information Available to the Operator	Reference
09:55:30 (1356:07)	The operator stopped Reactor Building Spray Pumps A and B (BS-P-1A and BS-P-1B). BS-P-1A and BS-P-1B were operated for approximately 5 minutes and 40 seconds.	HR(A), HR(F) and ST at PL3, ST at PL14, AN at PL8	2a
		AP norm/trip (Delay ≥ 160 minutes)	2a
09:56:26	Core Flood Tank 1A (CF-T-1A) high water level alarm was received. The level recorded was 13.22 feet. One minute later, 09:57:26 (1358:03), a normal signal was received and a level of 12.05 feet recorded. This indicated a significant discharge of water from Core Flood Tank 1A to the Reactor Coolant System. Note: Over the next 150 minutes similar level changes in the Core Flood Tank 1A were noted indicating periodic discharge of water from Core Flood Tank 1A into the Reactor Coolant System.	AN norm/high (13.3 feet) (Delay ≥ 160 minutes)	
09:56:58 (1357:35)	The operator stopped Decay Heat Pumps A and B (DH-P-1A and DH-P-1B). The Reactor Coolant System pressure was never low enough to put the Decay Heat System in service.	ST at P13 and PL13, HR(A) at PL3 HR(P_{DISCH}) at PL8	2a
10:00:00 (1400:37) Approximate	The operator opened the Electromatic Relief Block Valve (RC-V2).	ST at PL4	11
10:05:25 (1406:02)	Pressurizer heater groups 1 and 2 returned to an operable condition.	ST at PL4 AP norm/trip (Delay ≥ 156 minutes)	2a
10:07:19 (1407:56)	Pressurizer heater groups 1 and 2 tripped.	ST at PL4 AP norm/trip (Delay ≥ 156 minutes)	2a
10:26:18 (1426:55)	Reactor Coolant System Loop A hot leg temperature decreased to within the instrumentation range (Figure 22). This was the result of the steam in Loop A hot leg condensing.	SC at PL4 and PL10, HR at PL4, AN at PL8	1

542 059

Time	Event	Information Available to the Operator	Reference
10:29:18 (1430:00) Approximate	At the request of the State government, the Vice President of Generation traveled to the Governor's office to report the Plant Status. He was accompanied by the Metropolitan Edison Station Superintendent and Unit 2 Technical Superintendent. The Emergency Director designee was directed to maintain the plant in a stable condition during their absence. The Station Superintendent carried a remote paging device to permit him to be signaled, if necessary.		2a
10:31:25 (1432:07)	The operator started Reactor Coolant Makeup Pump C (MU-P-1C). Reactor coolant pressure was approximately 440 psig.	MU-P-1C: AM at PLB, MR(A) and ST at PL3 AP norm/trip (Delay ≥ 173 minutes) RC P: MM and SC at PL4	2a, 3a
10:32:36 (1433:18)	Pressurizer heater groups 1 and 2 returned to an operable condition.	ST at PL4	2a
10:34:24 (1435:06)	The Reactor Coolant System Loop A hot leg temperature increased beyond the range of the instrumentation when High Pressure Injection was directed to Loop B hot leg (Figures 22 and 27).	SC at PL4 and PL10, MM at PL4, AM(high = 612 F) at PLB	1,4
10:35:55 (1436:37)	The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C).	MU-P-1C: AM at PLB, MR(A) and ST at PL3 AP norm/trip (Delay ≥ 113 minutes)	2a
10:39:12 (1439:54)	Pressurizer heater groups 1 and 2 tripped.	ST at PL4 AP norm/trip (Delay ≥ 105 minutes)	2a
10:39:24 (1440:06)	Reactor Coolant System Loop A hot leg temperature decreased to within the instrumentation range. This was the result of the steam in Loop A hot leg again collapsing when High Pressure	SC at PL4 and PL10, MM at PL4, AM(high = 612F) at PLB	1,4

542 060

Time	Event	Information Available to the Operator	Reference
	Injection to Loop A hot leg was re-established.		
10:39:51 (1440:28)	Pressurizer heater groups 1 and 2 returned to an operable condition.	ST at PL4	24
10:44:21 (1444:58) Approximate	The Auxiliary Building Air Exhaust Fans started and ran for approximately 30 minutes.	SC at PL25	24
11:06:00 (1506:37) Approximate	Pressurizer level started decreasing from 320 inches to 180 inches over a period of 18 minutes (Figure 30).	SC AT PL 4, AH (High/High = 315 inches, High = 209 inches, Low = 200 inches and Low/Low = 80 inches) at PL8.	1
11:09:23 (1510:00)	The airborne radioactivity level in Unit 2 Control Room had reduced to a level such that personnel were permitted to remove respirators.		4
11:12:00 (1512:37) Approximate	The operator shut the Electromatic Relief Block Valve (RC-V2).	ST at PL4	[illegible]
11:12:00 (1512:37) Approximate	Reactor Coolant System Loop A cold leg temperature started to increase from 200F to 400F indicating the occurrence of some natural circulation in Loop A (Figure 22)	SC at PL10	1
11:14:23 (1515:00) Approximate	The Auxiliary Building Air Exhaust Fans stopped.	SC AT PL25	[illegible]
11:18:34 (1519:11)	The operator started Reactor Coolant Makeup Pump C (MU-P-1C) to stop the rapid fall in the Pressurizer level.	AN at PL8, HR(A) and ST at PL3 AP norm/trip (Delay ≥ 84 minutes)	[illegible]
11:24:00 (1524:37)	Pressurizer level stopped decreasing at 180 inches and started increasing, going off scale during the next hour (Figure 30).	SC at PL4	[illegible]

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Time	Event	Information Available to the Operator	Reference
11:28:12 (1528:49)	The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C).	AN at PL8, HR(A) and ST at PL3 AP norm/trip (Delay ≥ 85 minutes)	2a
11:28:52 (1529:29)	Pressurizer heater group 1 and 2 tripped.	ST at PL4 AP norm/trip (Delay ≥ 85 minutes)	2a
11:32:37 (1533:14)	The operator started Reactor Coolant Makeup Pump C (MU-P-1C) to raise the pressurizer level.	AN at PL8, HR(A) and ST at PL3 AP norm/trip (Delay ≥ 86 minutes)	2a
11:35:48 (1536:25)	The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C).	AN at PL8, HR(A) and ST at PL3 AP norm/trip (Delay ≥ 83 minutes)	2a
11:36:00 (1536:37) Approximate	The operator started filling Steam Generator B to 97% on the operating Range to induce additional cooling of the Reactor Coolant System. This level was reached at 12:00:00 (1600:37) (Figure 39).	SG operate level at PL4 and PL5	1
11:45:17 (1545:54)	Pressurizer heater groups 1 and 2 returned to an operable condition.	ST at PL4	2a
12:14:23 (1615:00) Approximate	The Auxiliary Building Air Exhaust Fans restarted and remained on throughout the remainder of March 28.	SG at PL17	9p
12:36:00 (1636:37) Approximate	The operator opened the Electromatic Relief Block Valve (RC-V2).	ST at PL4	11
12:48:00 (1648:00) Approximate	Pressurizer level indication came on scale (Figure 30).	SG at PL4	

542 062

Time	Event	Information Available to the Operator	Reference
13:02:23 (1303:00)	The operator started Condenser Vacuum Pump 1C (VA-P-1C) in an attempt to re-establish vacuum. The auxiliary boiler had been returned to service and was supplying gland sealing steam to the main turbine.	ST at PL17 AP on/off (Delay ≥ 0 minutes)	2a
13:13:10 (1313:47)	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/gas existed in the vessel head and Loop B hot leg. The Loop A hot leg steam was collapsed and natural circulation flow established in this loop. Condenser vacuum was re-established after the auxiliary steam boiler was returned to service. Steam Generator A was steaming to the condenser and Steam Generator B was isolated. 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 a hydrogen concentration increase and subsequent denotation which caused a 28 psig Reactor Building pressure pulse. Attempts to use the Core Flood System to cool the core over the last six hours had resulted in limited success. The reactor core was being cooled by (1) Steam Generator A steaming, (2) High Pressure Injection flow into the Reactor Coolant System and then to the Reactor Building floor via the Electromatic Relief Valve (RC-R2) and (3) Core Flood Tank A partial discharge.

Time	Event	Information Available to the Operator	Reference
13:15:00 (1715:37) Approximate	The operator shut the Electromatic 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 PL4	1, 11, 6a, 6, 7a
13:21:04 (1721:41)	The operator started Reactor Coolant Makeup Pump C (MU-P-1C) to further increase Reactor Coolant System pressure.	MU-P-1C: AN at PL8, MR(A) and ST at PL3 AP norm/trip (Delay ≥ 0 minutes) RC P: MR and SC at PL4	2a, 6a
13:26:09 (1726:46)	Pressurizer heater groups 1 and 2 tripped.	SC at PL4 AP norm/trip (Delay ≥ 0 minutes)	2a
14:25:26 (1826:03)	Pressurizer heater groups 1 and 2 return to an operable condition.	ST at PL4	2a
14:43:15 (1843:52)	The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C) to slow the rapid increase in Reactor Coolant pressure.	MU-P-1C: AN at PL8, MR(A) and ST at PL3 AP norm/trip (Delay ≥ 0 minutes) RC P: MR and SC at PL4	2a
14:55:00 (1855:37) Approximate	Reactor Coolant System pressure reached 2350 psig (Figure 12).	MR and SC at PL4, AN (Low/Low = 1900 psig, Low = 2055 psig and High = 2500 psig) at PL8	6a
14:59:23 (1900:00) Approximate	The following radiation monitors indicated on scale and were 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-230) (d) Fuel Handling Building Exhaust Unit Area radiation monitor (HP-R-3240)	AN, MR and SC at PL12	8a

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Time	Event	Information Available to the Operator	Reference
	The Fuel Handling Building radiation monitor (HP-R-215) and Control and Service Building Corridor radiation monitor (HP-R-236) were steady about 10 counts per minute.		
15:32:42 (1931:29)	The operator started Reactor Coolant Makeup Pump C (MU-P-1C).	AN at PLR, HR(A) and ST at PL3 AP norm/trip (Delay ≥ 0 minutes)	2a
15:32:42 (1933:29)	The operator started Reactor Coolant Pump 1A (RC-P-1A) and after approximately 10 seconds stopped the pump. This was done to verify the pump starting current was correct. Reactor Coolant System pressure dropped from 2340 psig to 1440 psig and Loop A cold leg temperature decreased from 400F to 290F (Figures 12 and 22).	RC-P-1A: ST, HR(A) and HR(F) at PL6. AN (trip) at PLR AP norm/trip (Delay ≥ 0 minutes) RC T_c: SC at PL10 RC Pt HR and SC at PL4	1, 2a, 6a, 10, 5
15:38:52 (1939:19)	The operator stopped Reactor Coolant Makeup Pump 1C (MU-P-1C)	AN at PLR, HR(A) and ST at PL3 AP norm/trip (Delay ≥ 0 minutes)	2a
15:49:00 (1949:45)	Reactor Coolant System Loop B hot leg temperature decreased to within the indication range of 572.6F (Figure 22).	AN(high - 612F), at PLR, SC at PL4 and PL10	1
15:49:16 (1949:53)	The operator started Reactor Coolant Makeup Pump 1C (MU-P-1C).	AN at PLR, HR(A) and ST at PL3 AP norm/trip (Delay ≥ 0 minutes)	2a
15:50:09 (1950:46)	The operator started Reactor Coolant Pump 1A (RC-P-1A). Reactor Coolant pressure dropped from 2250 psig to 1380 psig and eventually stabilized at 1000 psig. The average Reactor Coolant temperature dropped to 290F and eventually stabilized at 250F (Figure 22).	RC-P-1A: ST, HR(A), HR(F) and SC(A) at PL4 AN (trip) at PLR AP norm/trip at PLR (Delay ≥ 0 minutes) RC T_c: SC at PL10 RC Pt HR and SC at PL4	1, 2a, 6a, 10, 5

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542 065

Time	Event	Information Available to the Operator	Reference
15:56:00 ([illegible])	The operator stopped Reactor Coolant Makeup Pump 1C (MU-P-1C).	AM at ICR, HR(A) and SI at PLJ AP mon/trip (Delay ≥ 0 minutes)	[illegible]
17:29:21 ([illegible])	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 placed in this tank.		[illegible]
18:36:23 ([illegible])	Reactor Coolant letdown flow is lost. This was due to suspected plugging of either the letdown coolers, orifices or purification filters.	HR(F) at PLJ	[illegible]

PLANT STATUS

Time	Event	Information Available to the Operator	Reference
20:00:00 ([illegible])	Reactor Coolant Pump 1A (RC-P-1A) was operating with flow to the core 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 280F and 1165 psig with the pressurizer level at 397 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 Reactor Coolant Pump Seal Injection Flow. Reactor Coolant letdown had been lost and an attempt to regain it was in progress. The Reactor Building had been isolated except for essential services and periodic sampling. The Auxiliary and Fuel Handling Buildings had airborne radioactive materials present. These were being released		

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Time	Event	Information Available to the Operator	Reference
	(through charcoal filters and absolute filters) to the environment through the station vent by their buildings ventilation system exhausts. To handle the water present in the Auxiliary Building Sump, pre-accident water present in the Auxiliary Building Neutralizer Tank (WDL-T-8B) was being transferred to Unit 1.		

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542 067

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

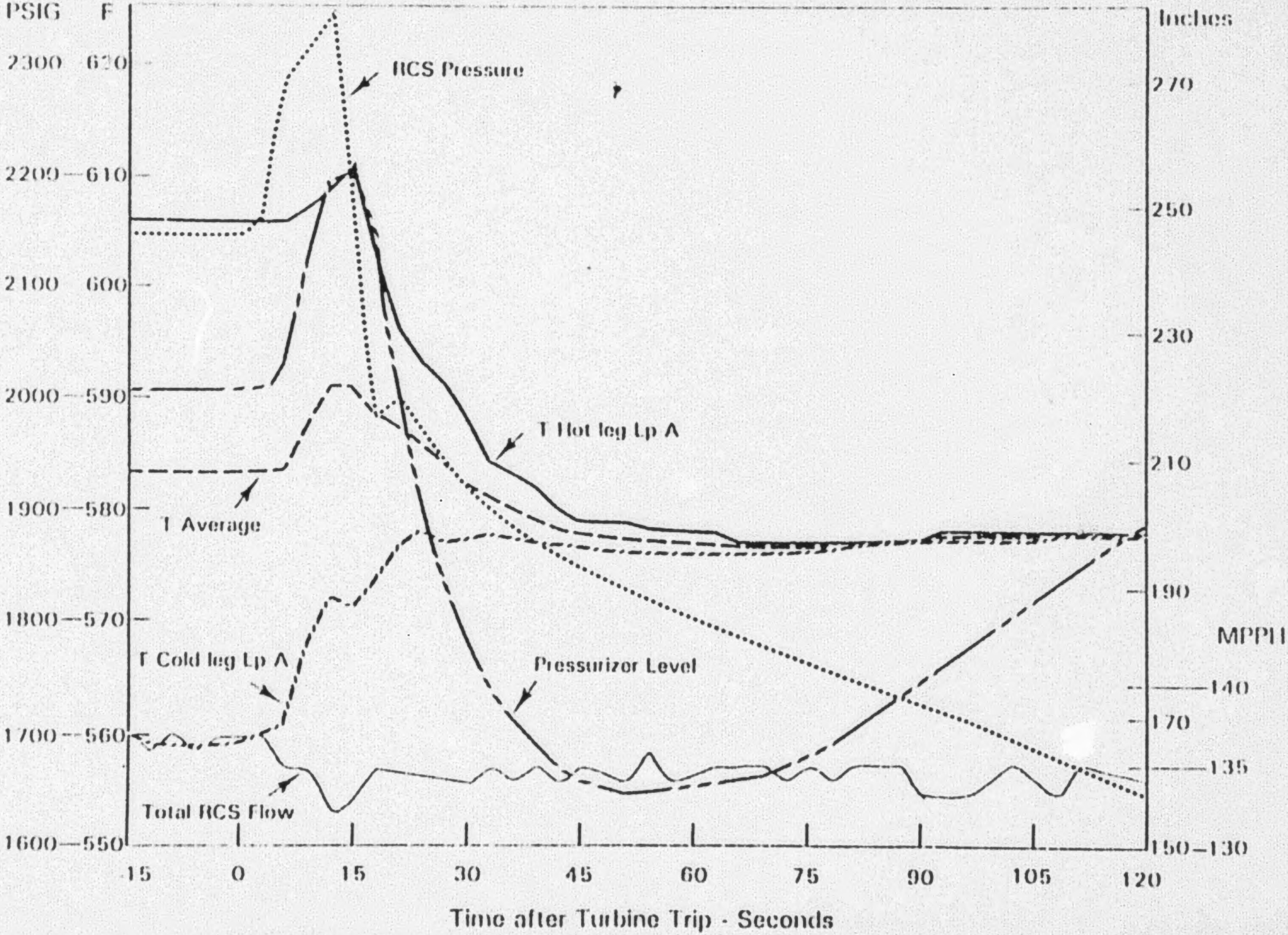

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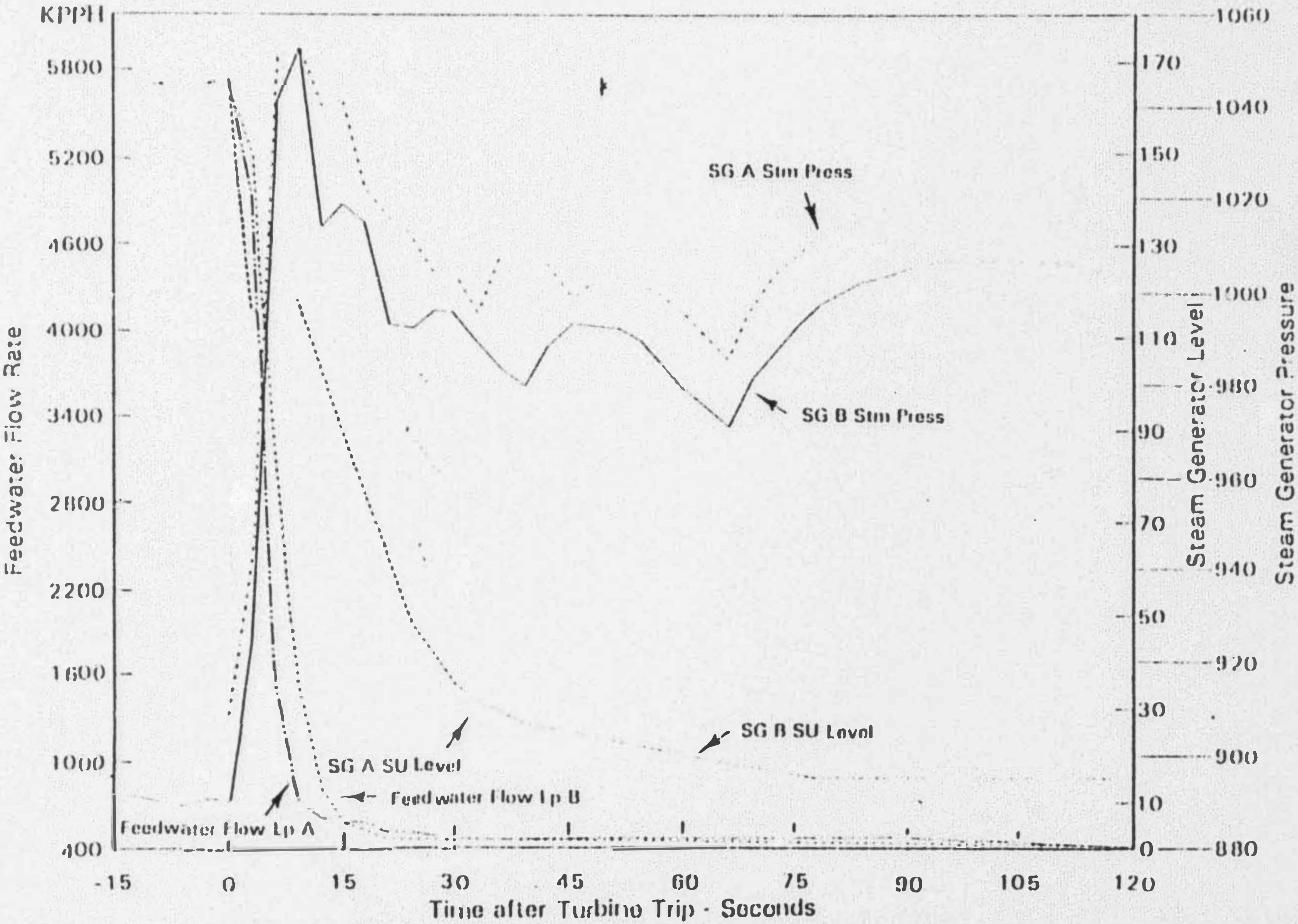
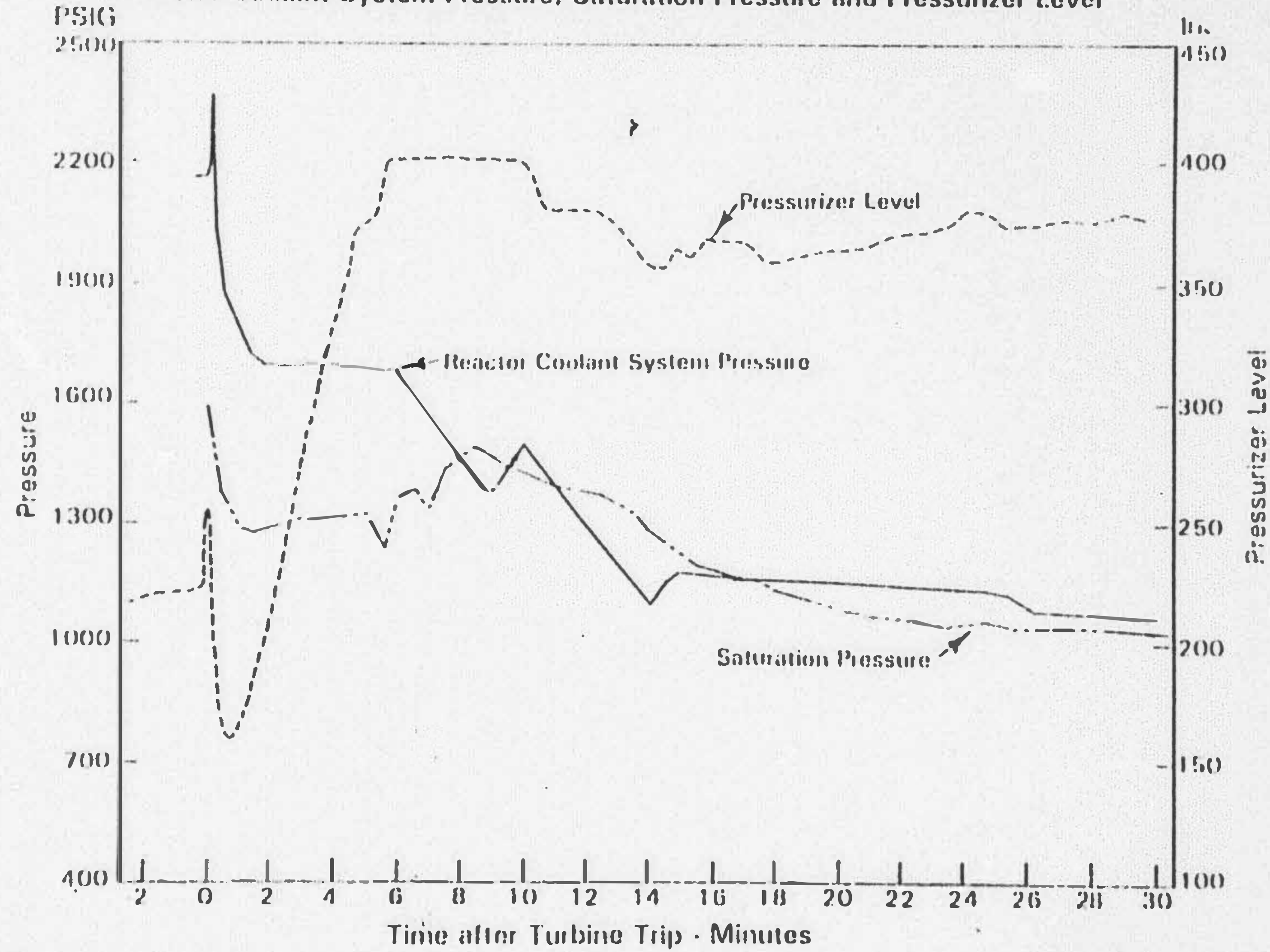
Figure 2
TMI-2 Loss of Coolant Accident 3/28/79
Summary of Steam Generator Parameters



542 069

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



542 070

Figure 4
TMI-2 Loss of Coolant Accident 3/28/79
Reactor Coolant System Pressure and Pressurizer Level

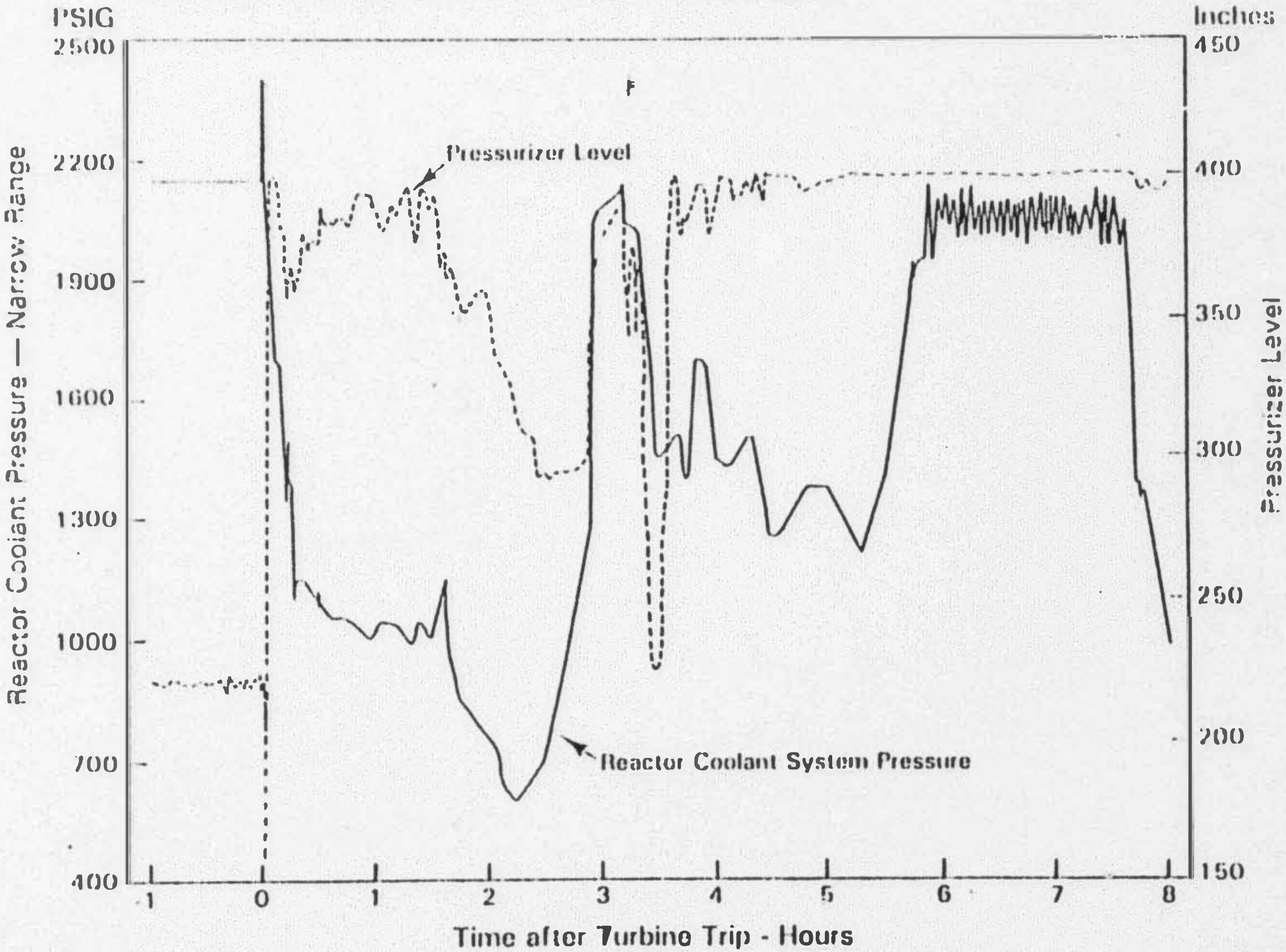
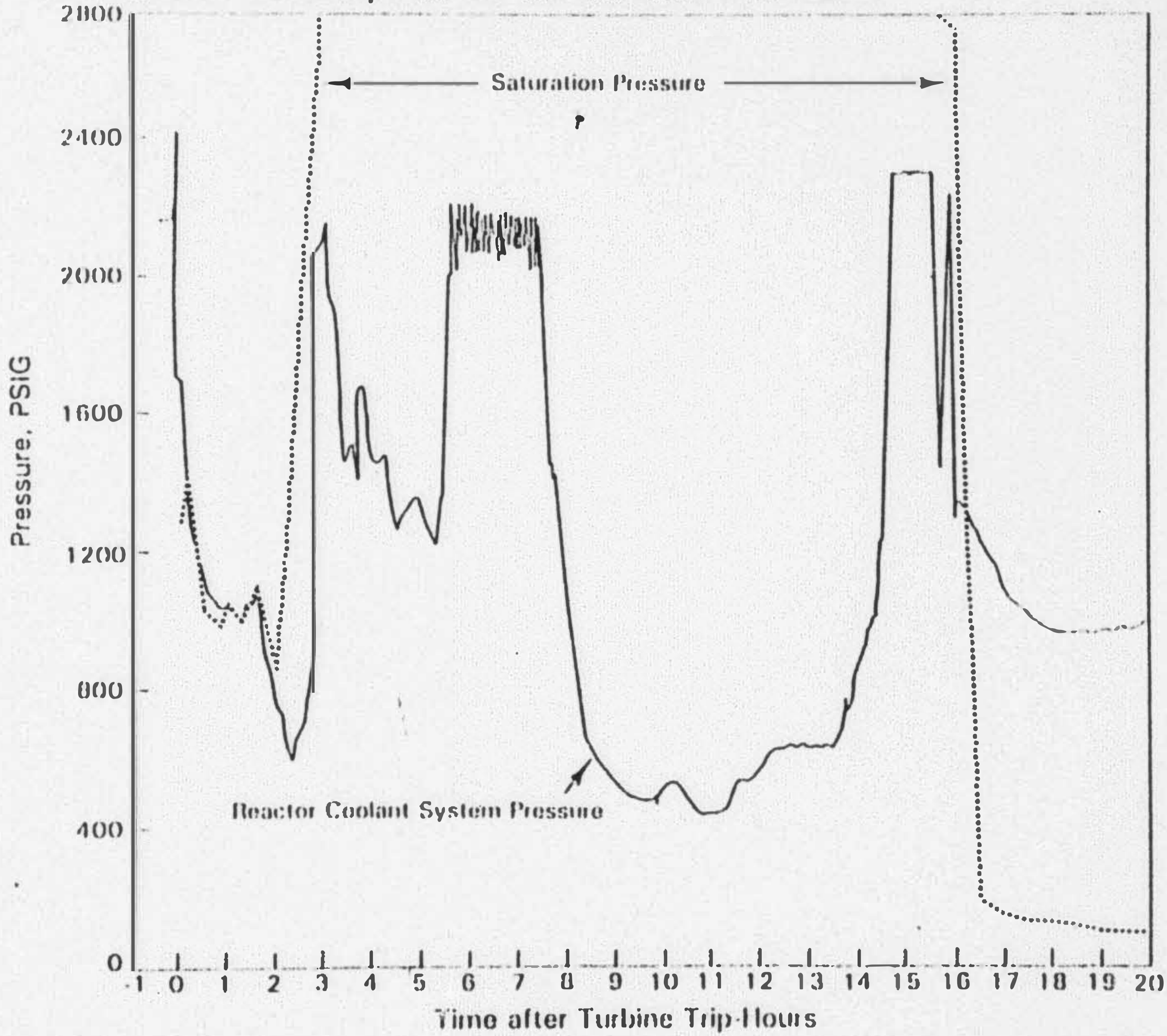


Figure 5
TMI-2 Loss of Coolant Accident 3/28/79
Reactor Coolant System Pressure and Saturation Pressure



542 072

FIGURE 6

SOLID LINE - REACTOR COOLANT LOOP A TOUT
DOTTED LINE - REACTOR COOLANT LOOP B TOUT
DASH/DOT LINE - REACTOR COOLANT LOOP A TIN
DASHED LINE - REACTOR COOLANT LOOP B TIN

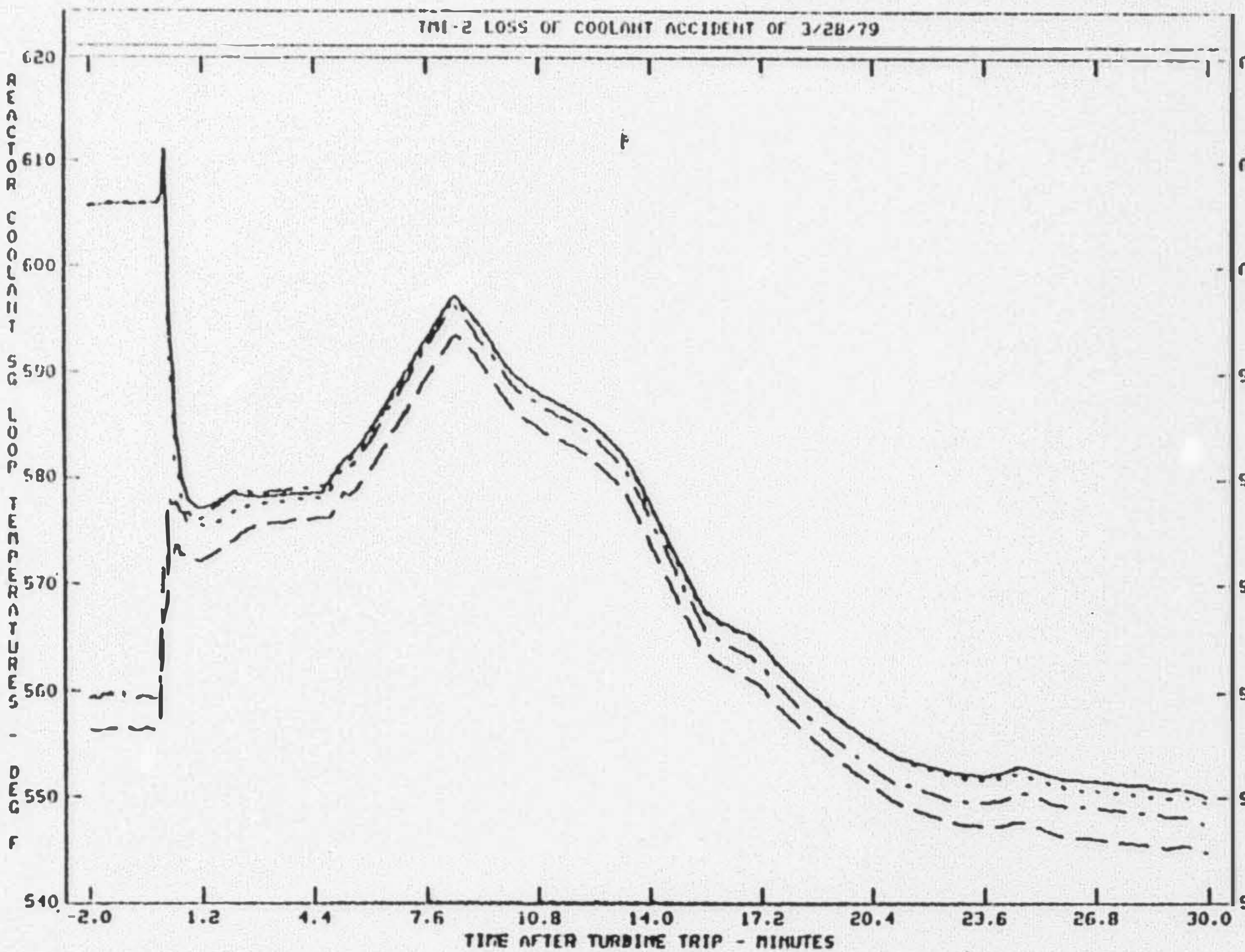


FIGURE 7

SOLID LINE - REACTOR COOLANT SYSTEM LOOP A TOUT
DOTTED LINE - REACTOR COOLANT SYSTEM LOOP B TOUT
DASH-DOT LINE - REACTOR COOLANT SYSTEM LOOP A TIN
DASHED LINE - REACTOR COOLANT SYSTEM LOOP B TIN

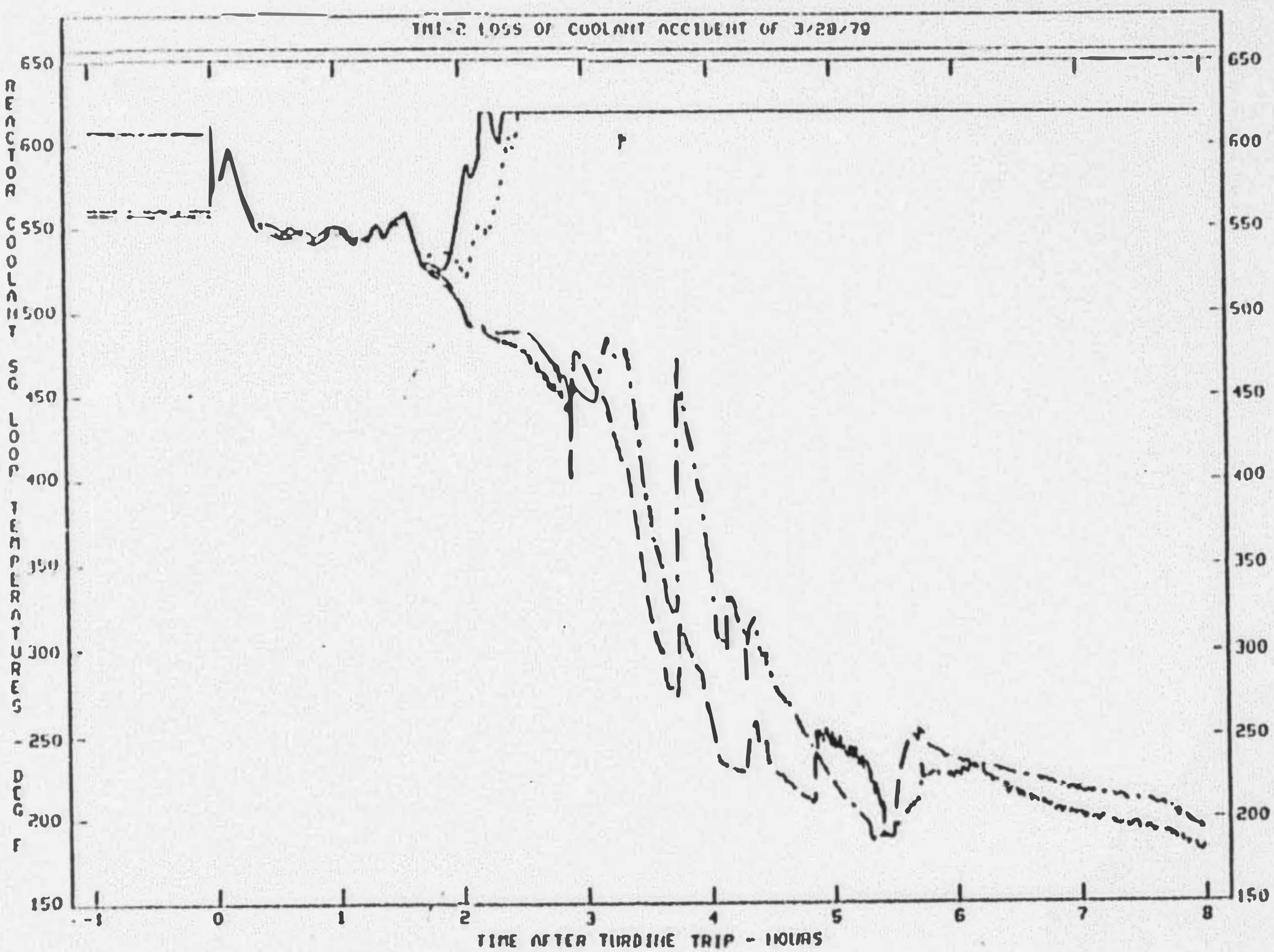
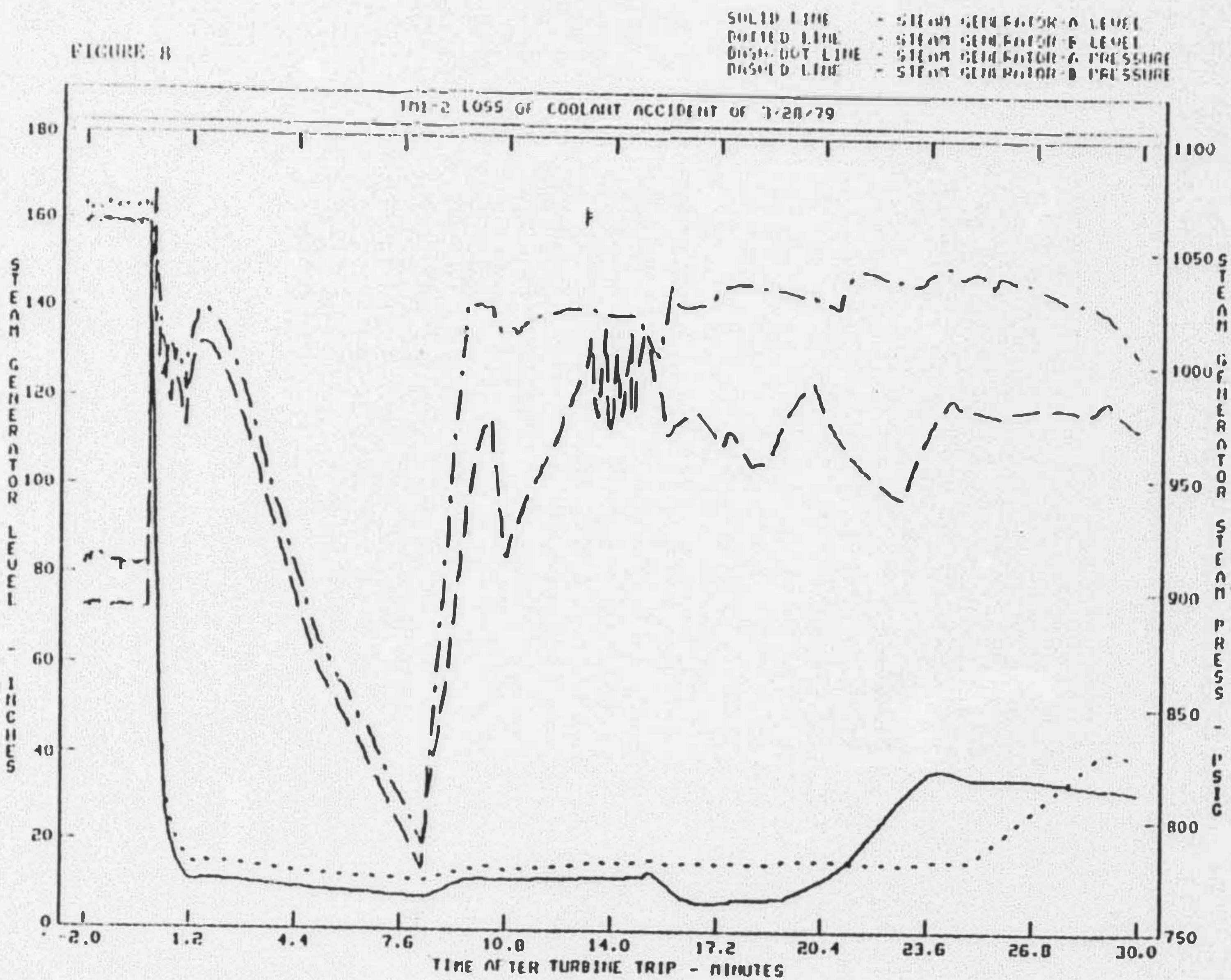


FIGURE 8



542 075

FIGURE 9

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

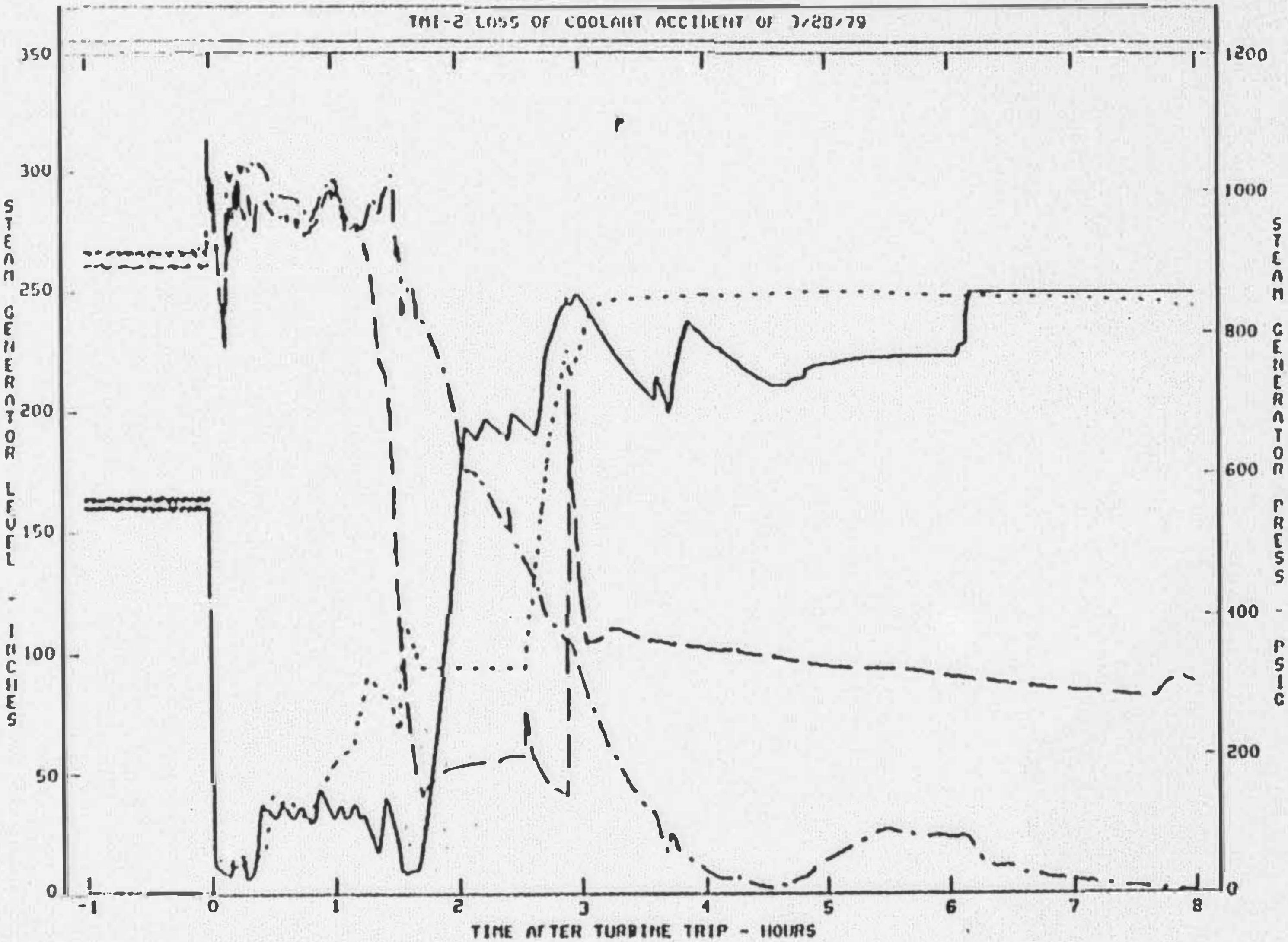


FIGURE 10

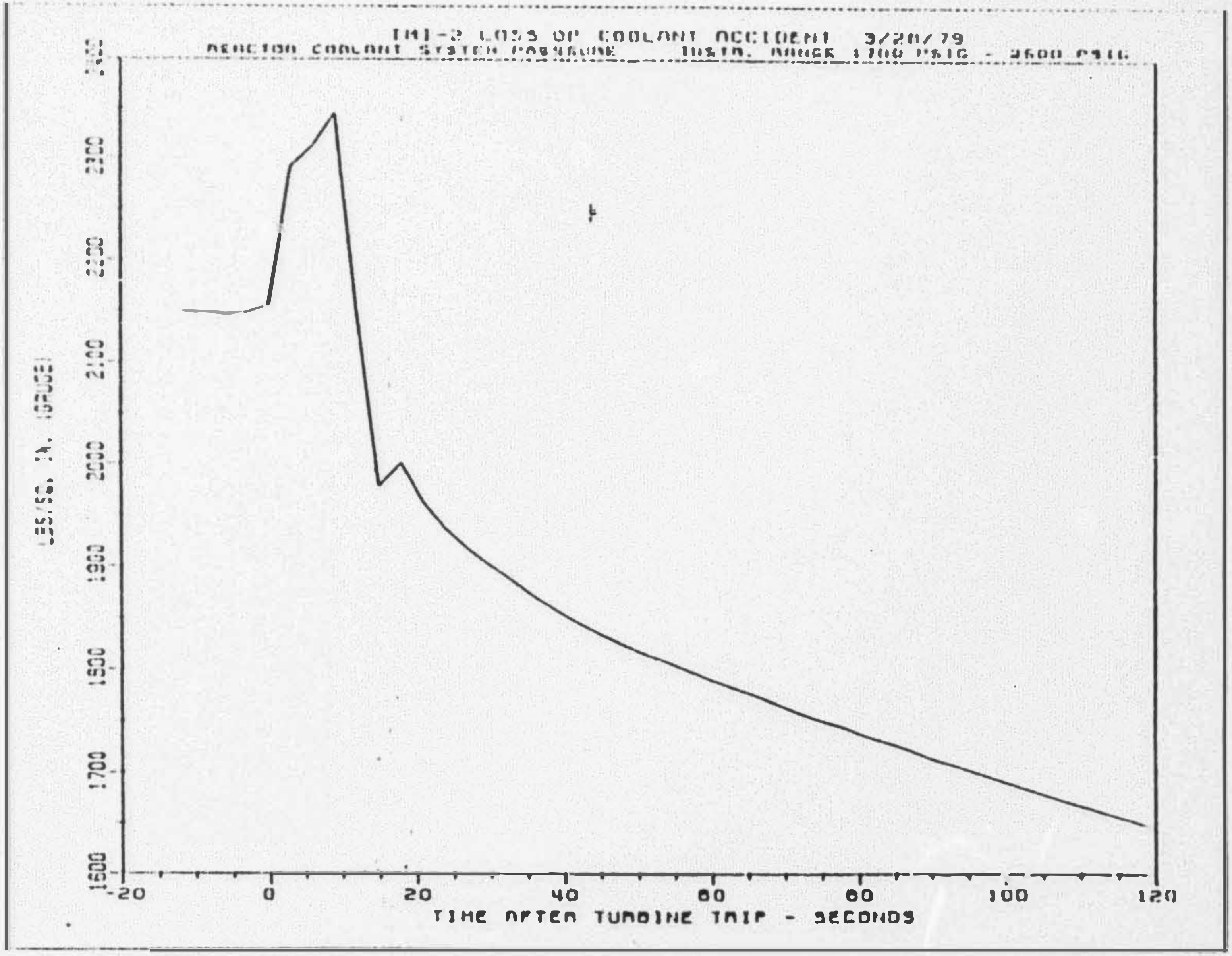
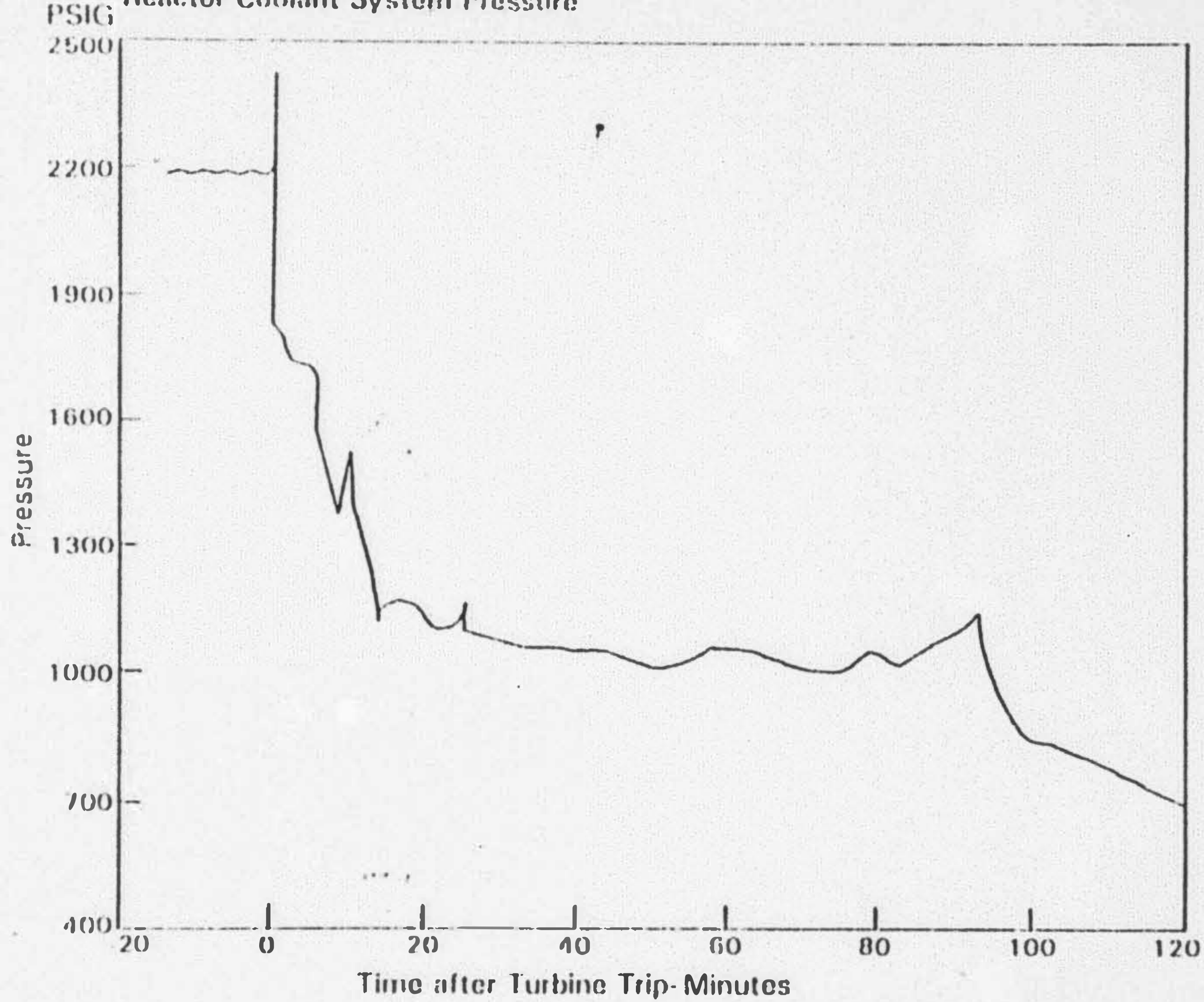

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
REACTOR COOLANT SYSTEM PRESSURE INSTR. RANGE 1700 PSIG - 2500 PSIG
LBS/SQ. IN. (GAUGE)
2300
2200
2100
2000
1900
1800
1700
1600
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - SECONDS

Figure 11
TMI-2 Loss of Coolant Accident 3/28/79
Reactor Coolant System Pressure



542 078

Figure 12
TMI-2 Loss of Coolant Accident 3/28/79
Reactor Coolant System Pressure

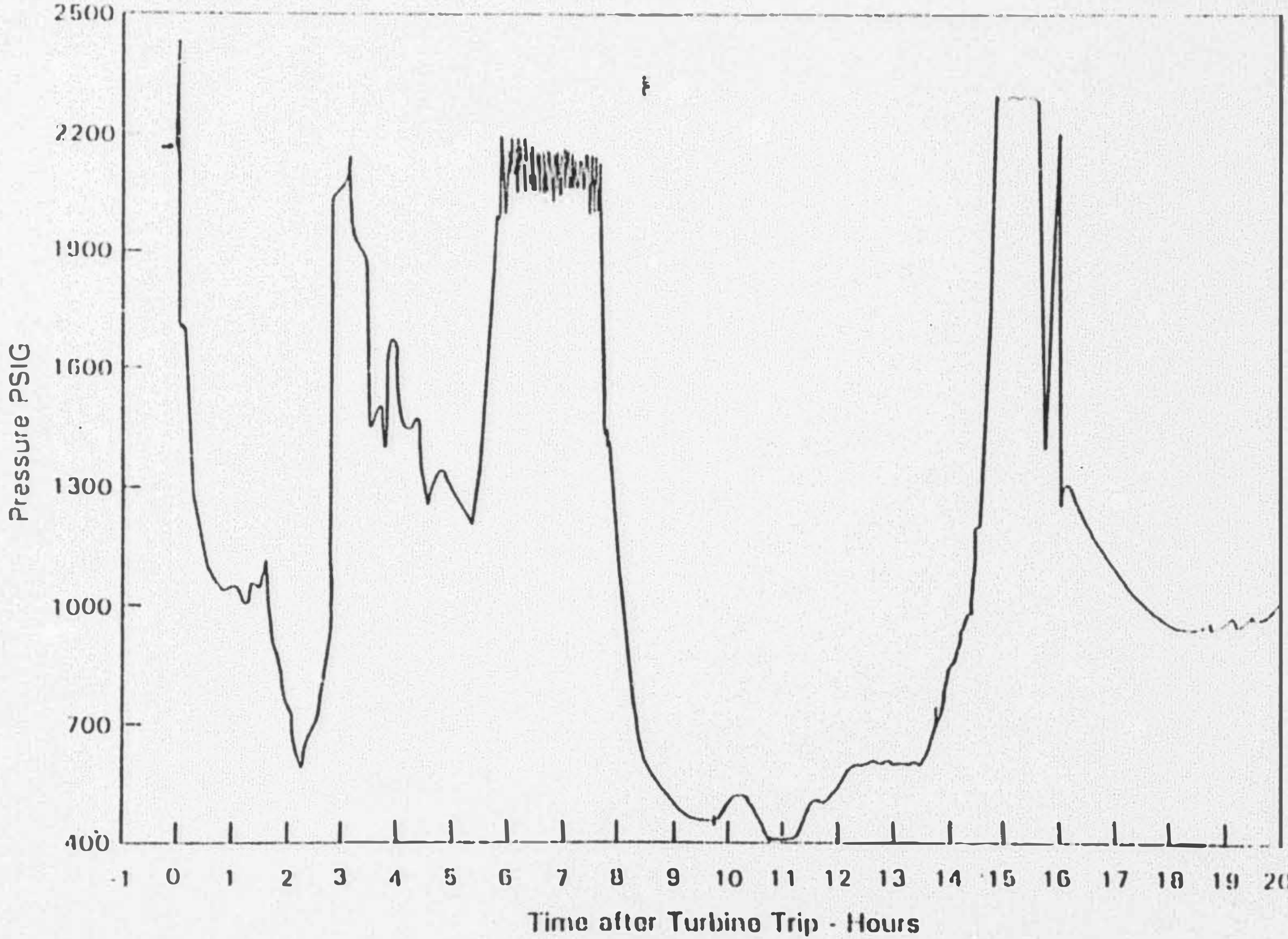


FIGURE 13

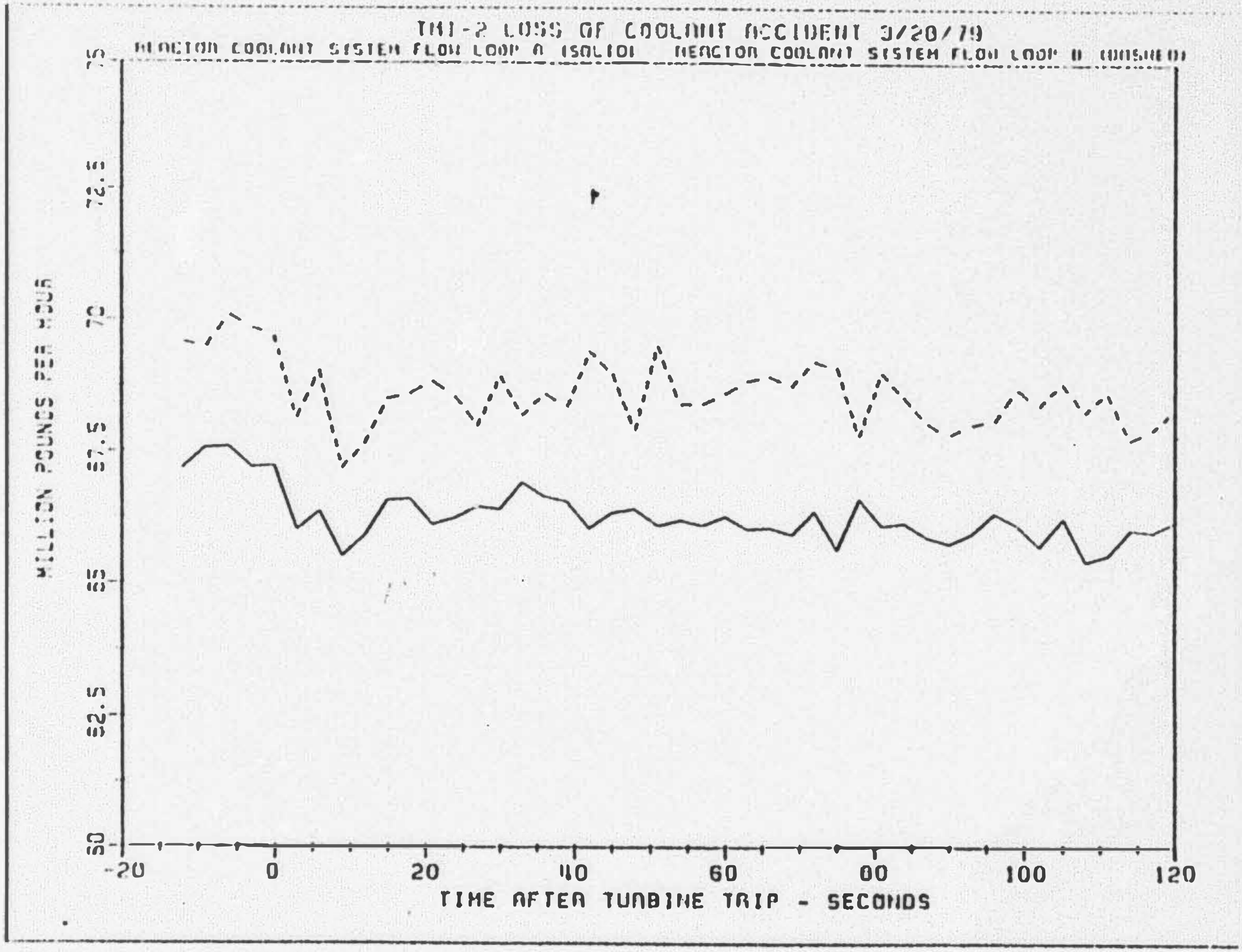

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
REACTOR COOLANT SYSTEM FLOW LOOP A (SOLID)
REACTOR COOLANT SYSTEM FLOW LOOP B (DASHED)
MILLION POUNDS PER HOUR
75
72.5
70
67.5
65
62.5
60
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - SECONDS

FIGURE 14

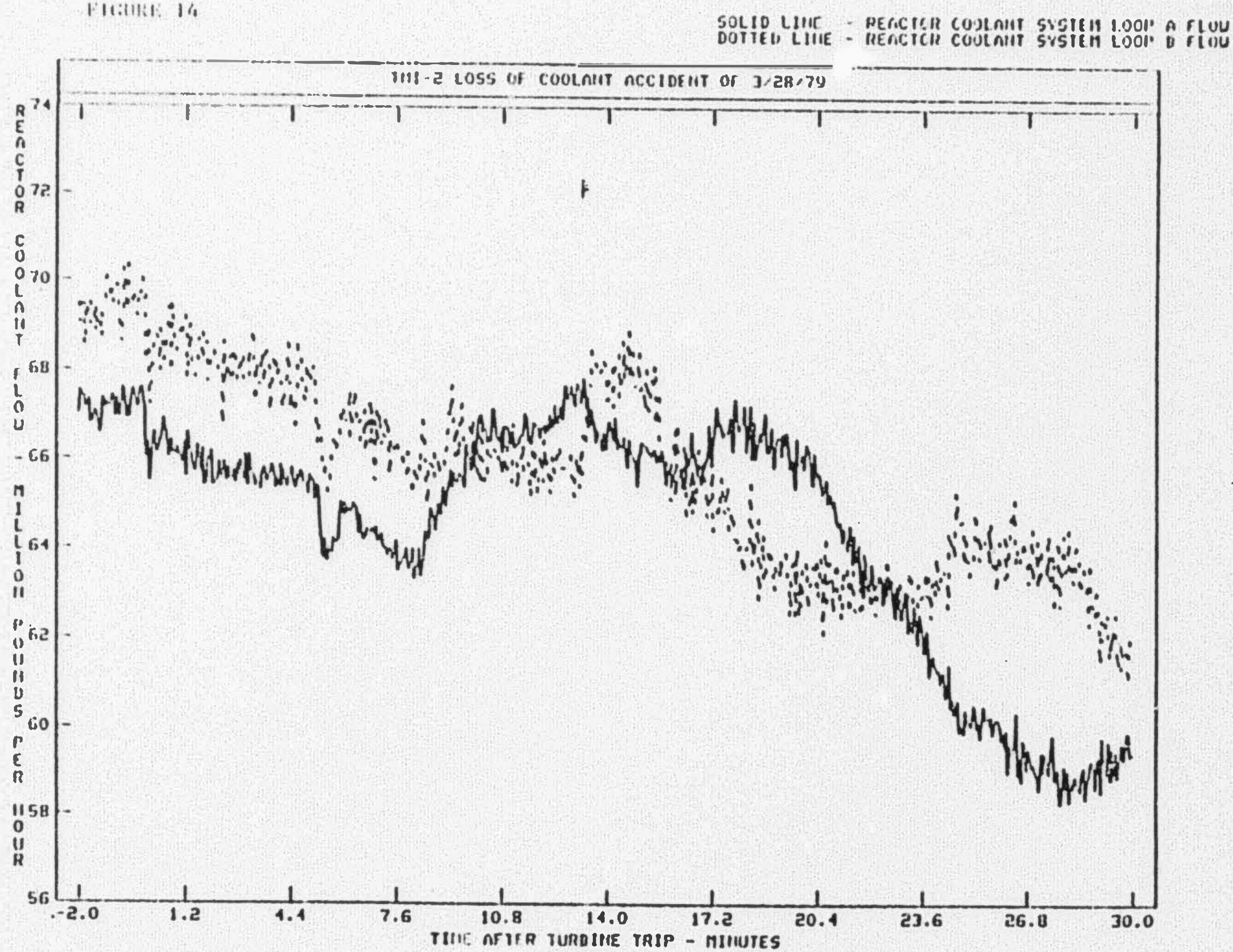

SOLID LINE - REACTOR COOLANT SYSTEM LOOP A FLOW
DOTTED LINE - REACTOR COOLANT SYSTEM LOOP B FLOW
TMI-2 LOSS OF COOLANT ACCIDENT OF 3/28/79
REACTOR COOLANT FLOW - MILLION POUNDS PER HOUR
74
72
70
68
66
64
62
60
58
56
-2.0
1.2
4.4
7.6
10.8
14.0
17.2
20.4
23.6
26.8
30.0
TIME AFTER TURBINE TRIP - MINUTES

FIGURE 15

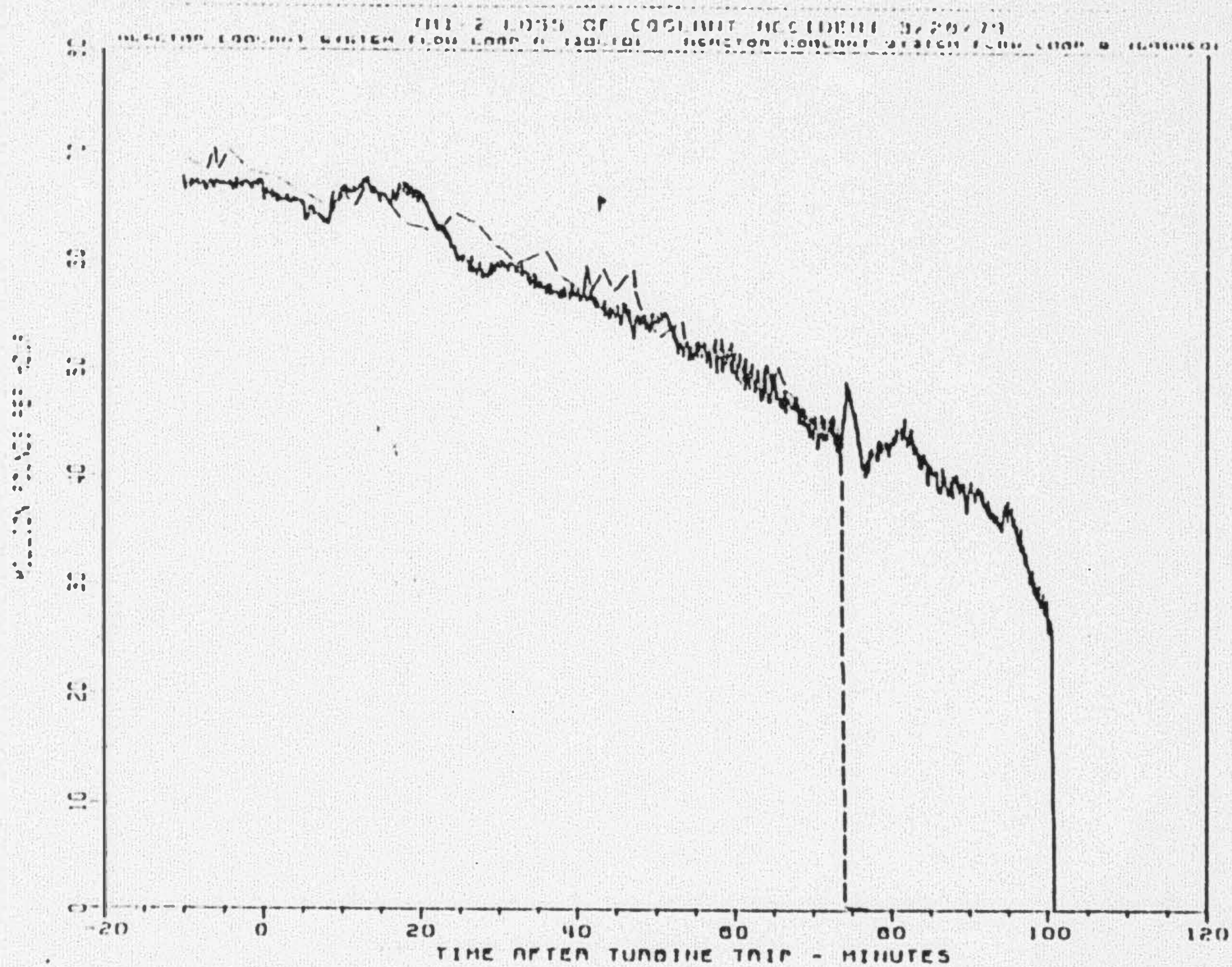
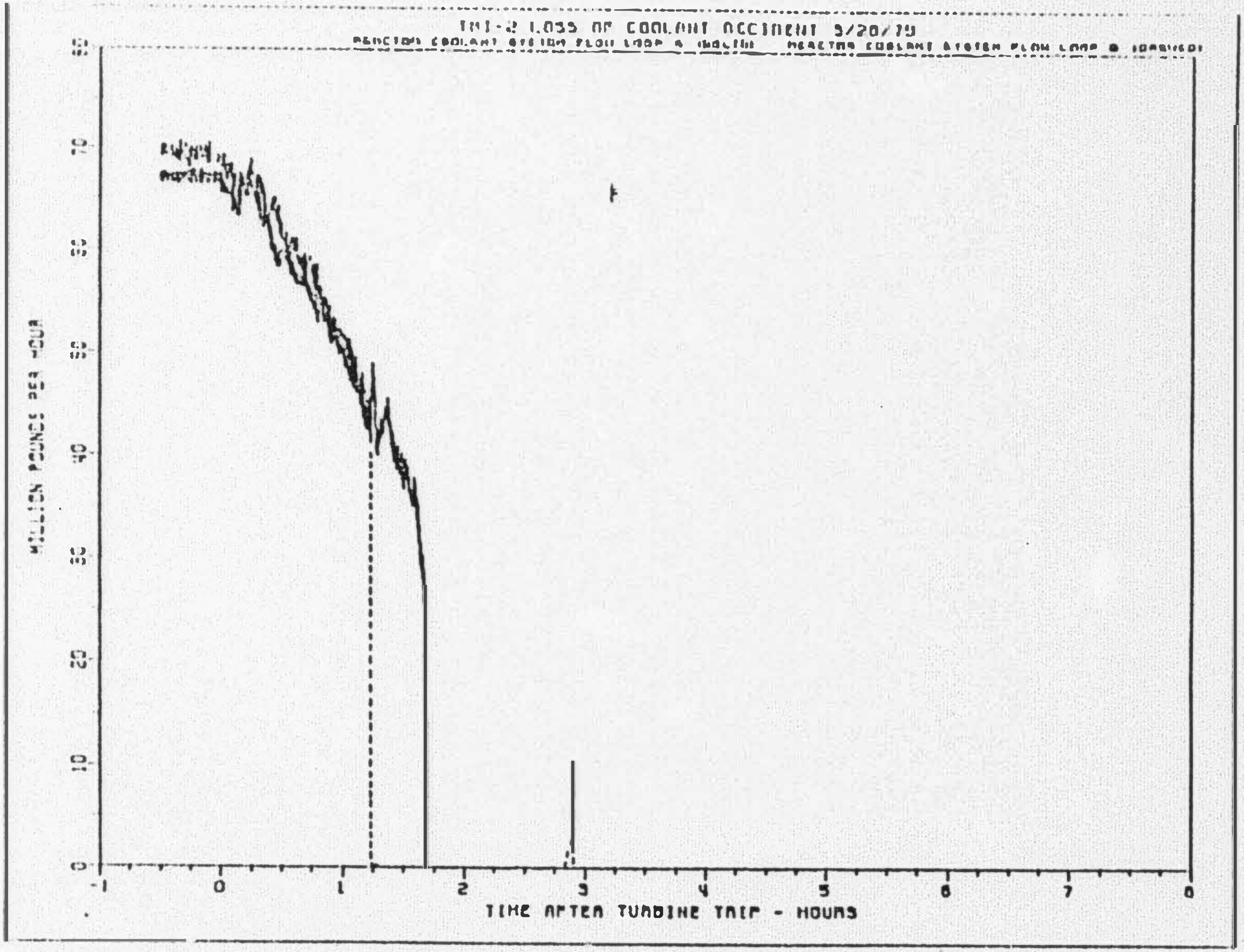
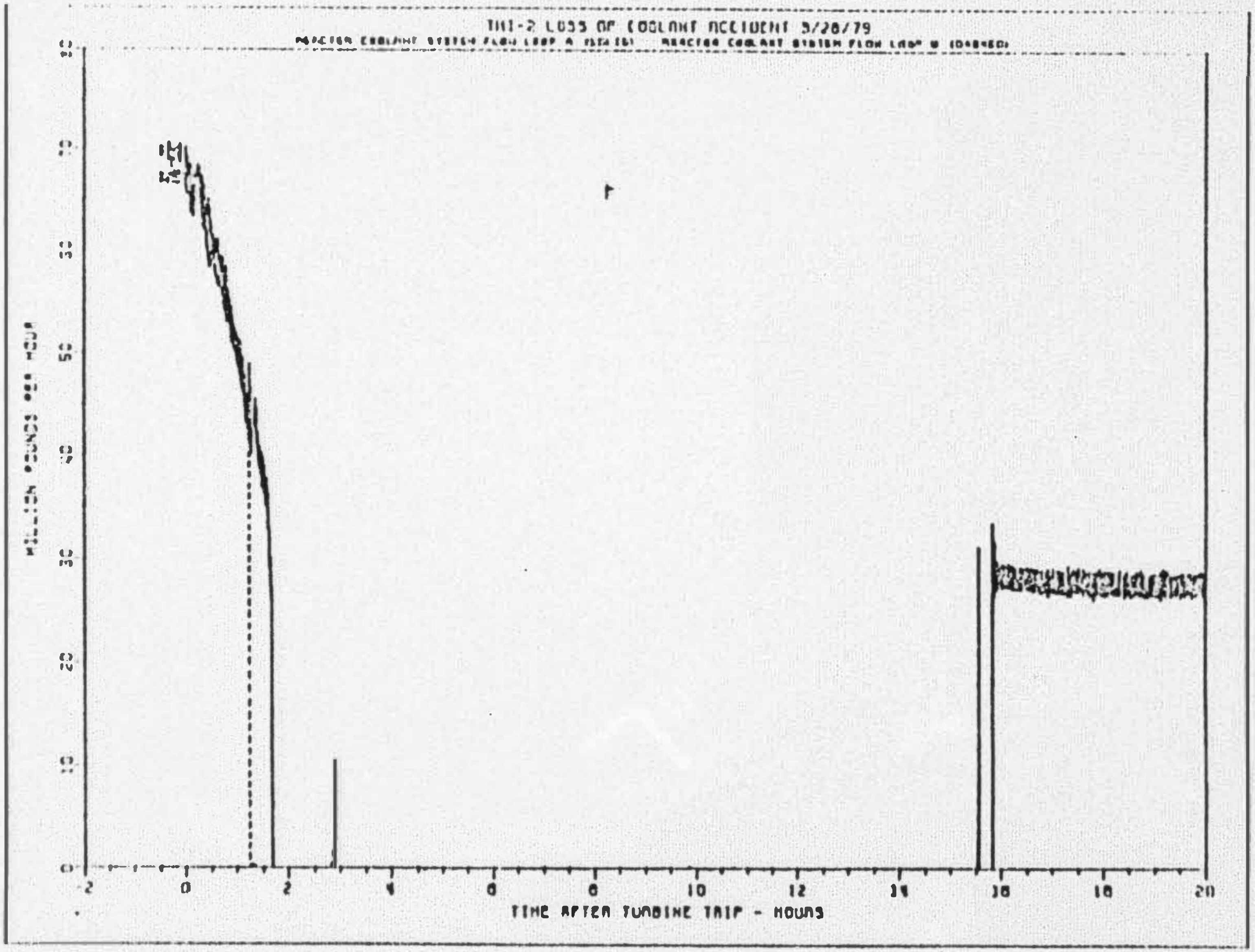


FIGURE 16



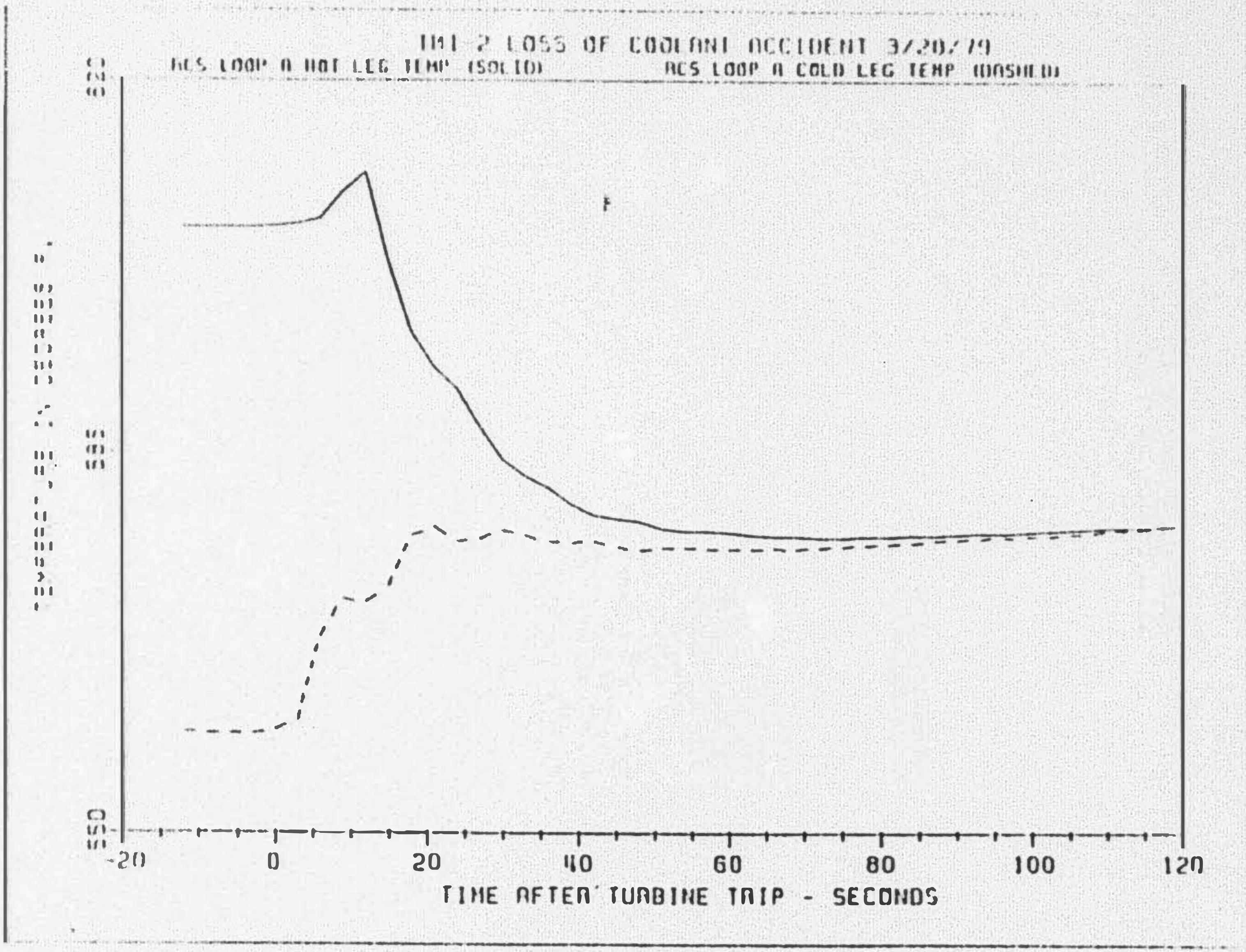
542 083

FIGURE 17



542 084

FIGURE 18



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

FIGURE 19

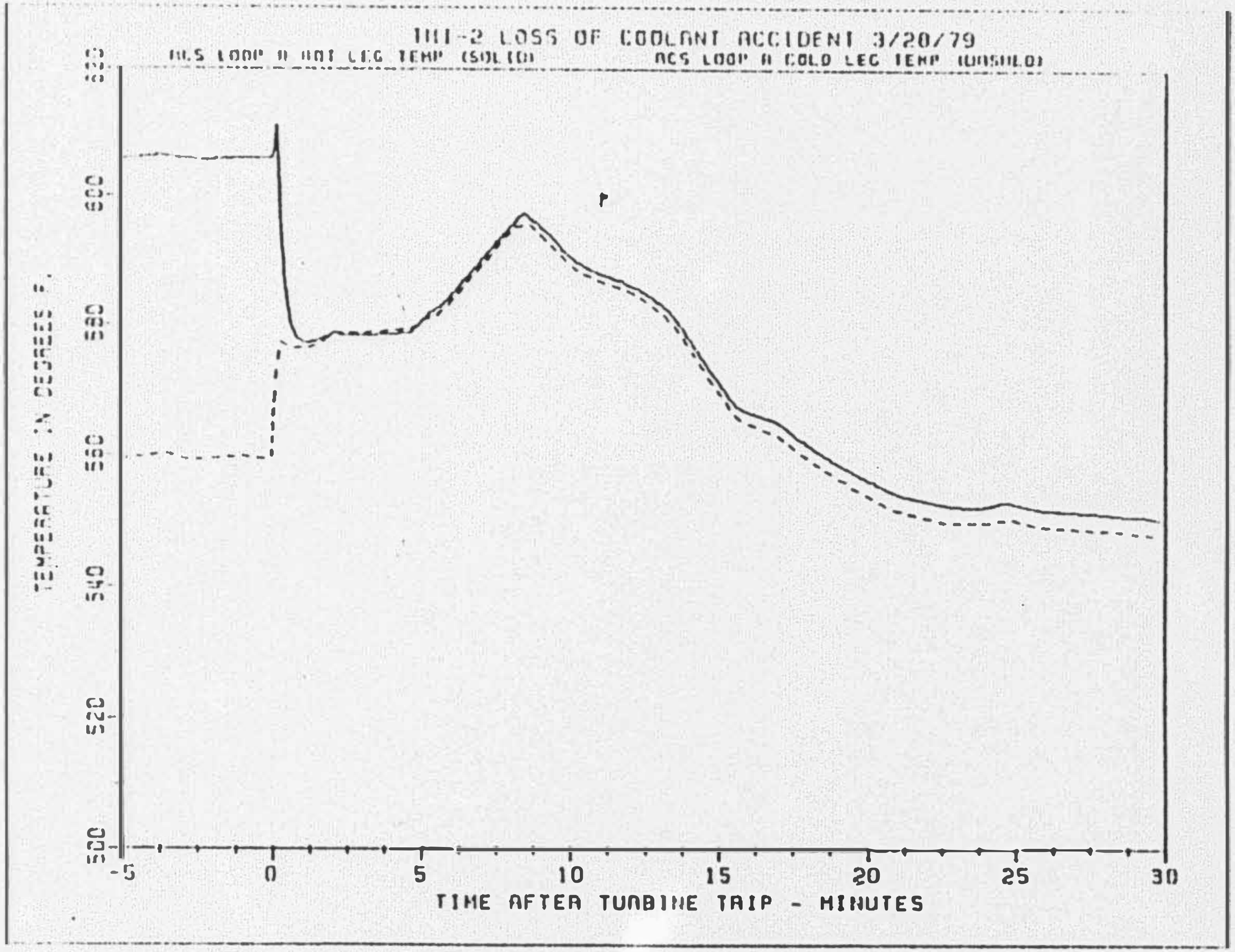

TMI-2 LOSS OF COOLANT ACCIDENT 3/20/79
RCS LOOP A HOT LEG TEMP (SOLID)
RCS LOOP A COLD LEG TEMP (DASHED)
TEMPERATURE IN DEGREES F.
620
600
580
560
540
520
500
-5
0
5
10
15
20
25
30
TIME AFTER TURBINE TRIP - MINUTES

FIGURE 20

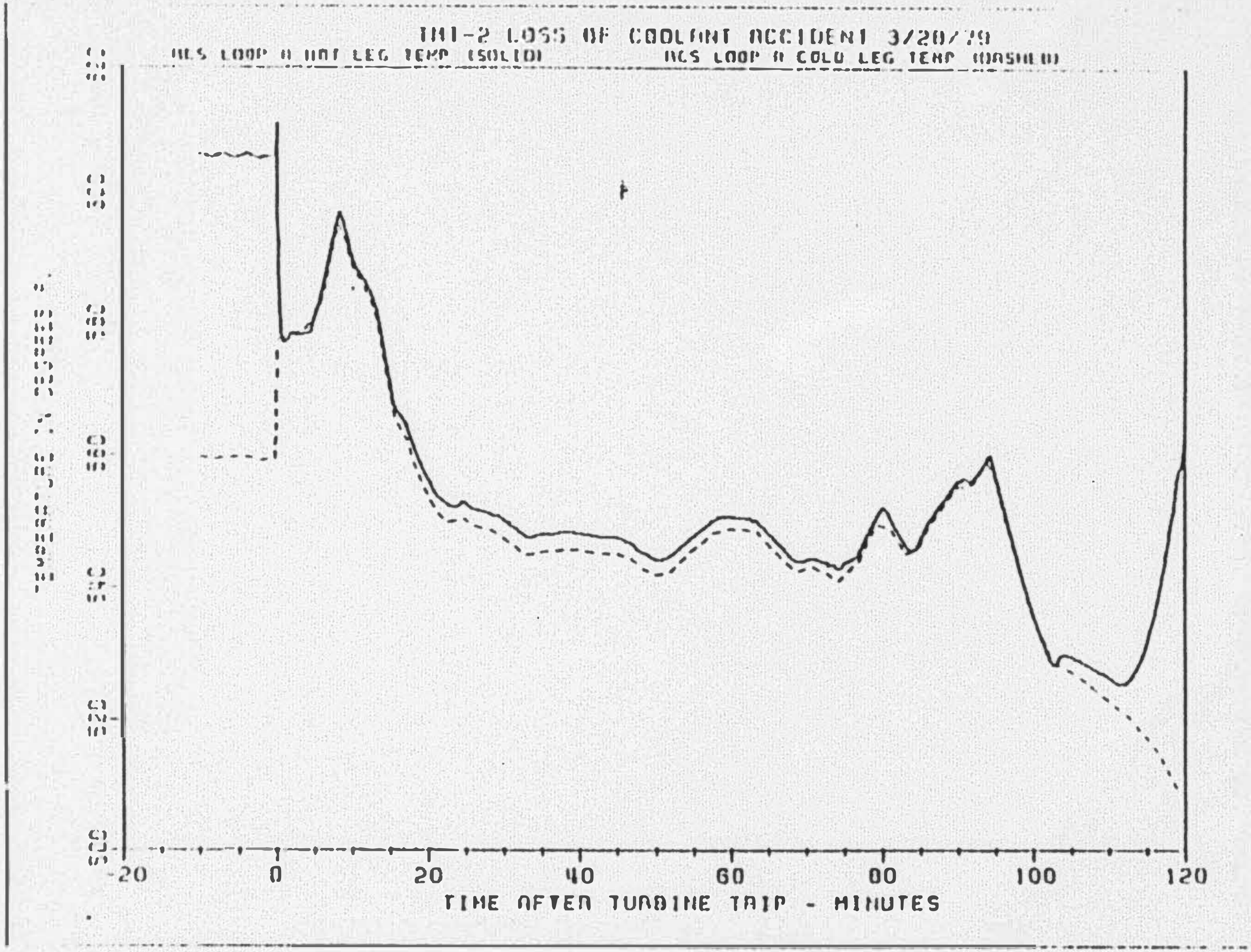

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
RCS LOOP A HOT LEG TEMP (SOLID)
RCS LOOP A COLD LEG TEMP (DASHED)
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - MINUTES

FIGURE 21

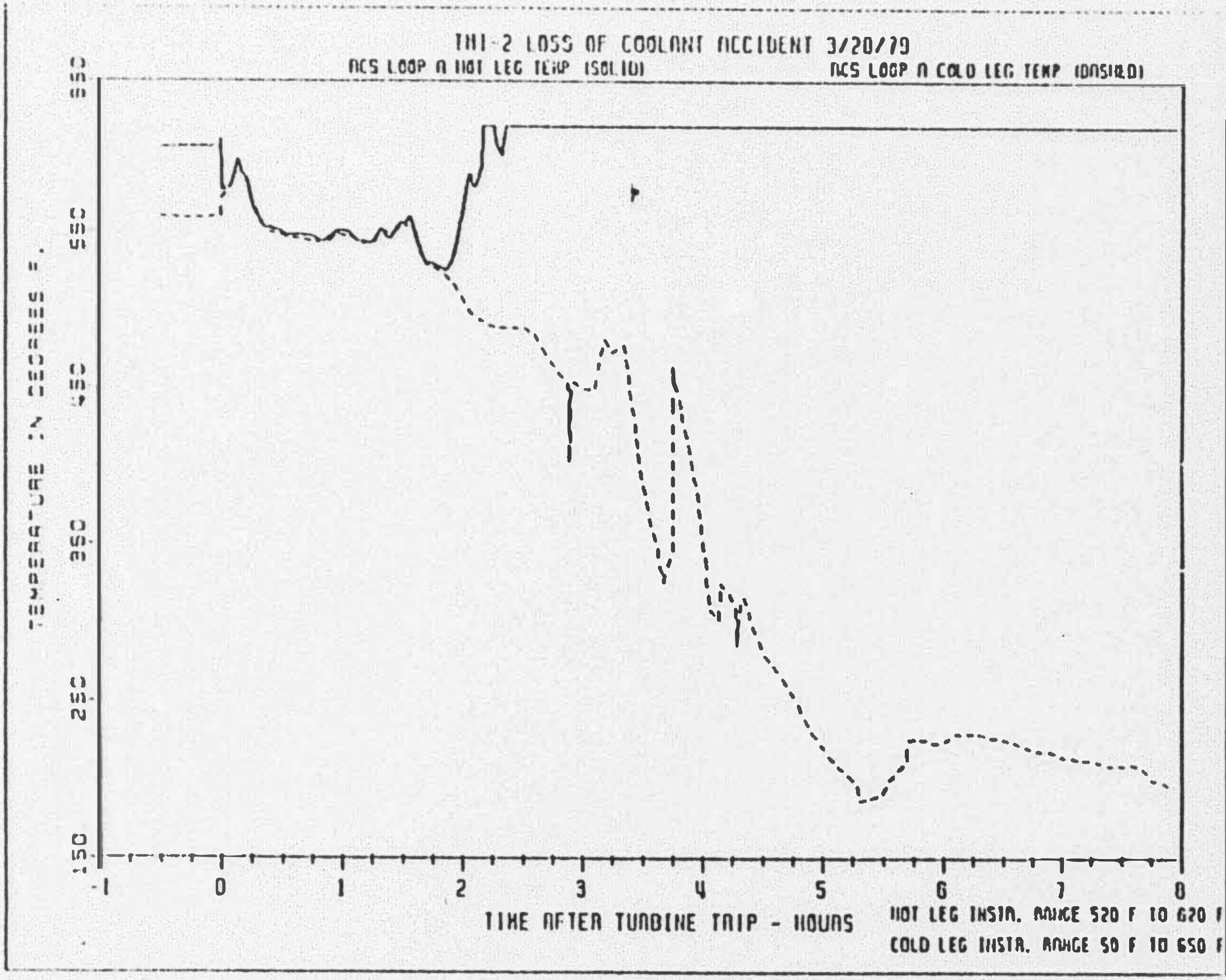
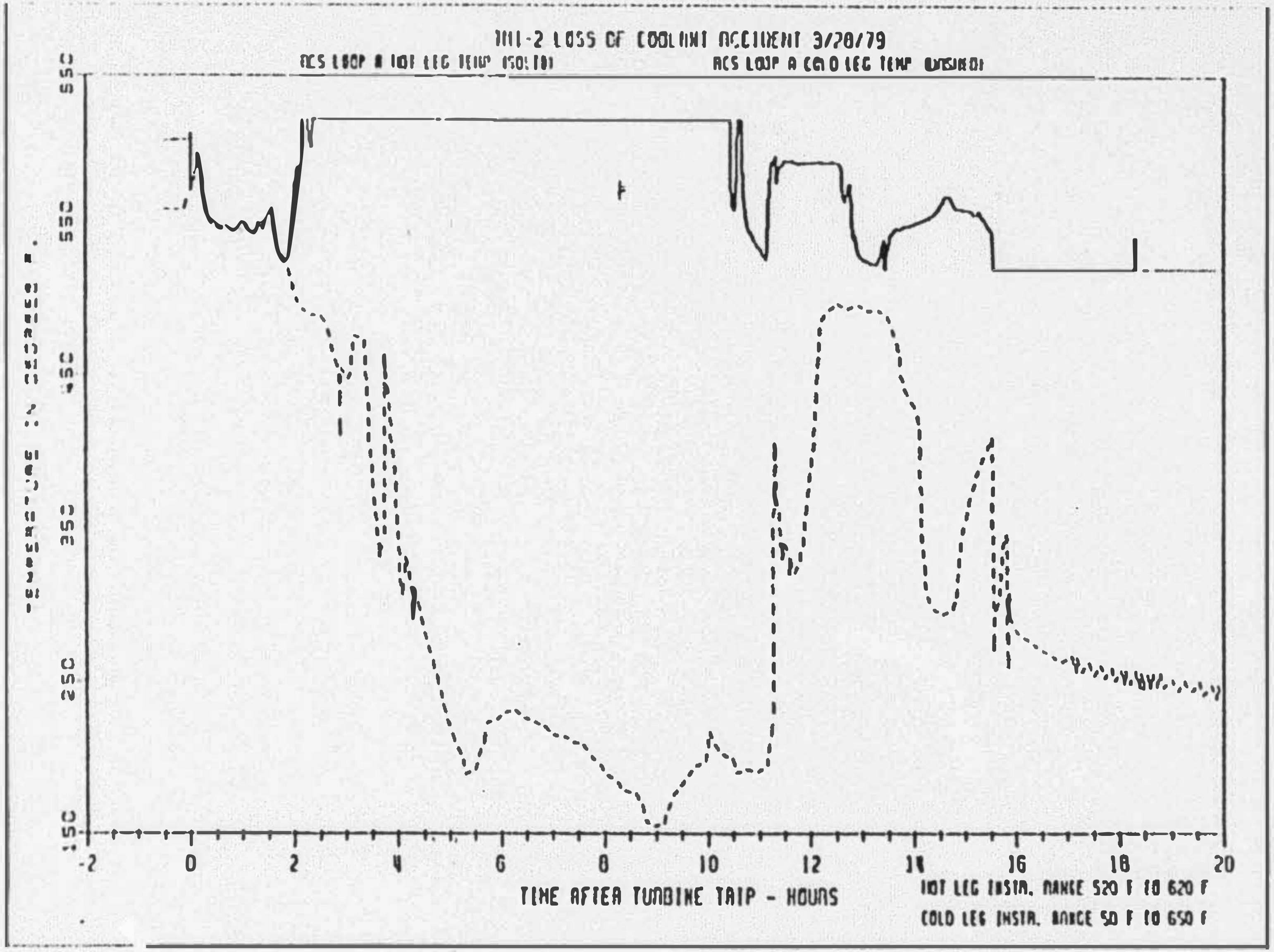

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
RCS LOOP A HOT LEG TEMP (SOLID)
RCS LOOP A COLD LEG TEMP (DASHED)
TEMPERATURE IN DEGREES F.
650
550
450
350
250
150
-1
0
1
2
3
4
5
6
7
8
TIME AFTER TURBINE TRIP - HOURS
HOT LEG INSTR. RANGE 520 F TO 620 F
COLD LEG INSTR. RANGE 50 F TO 650 F

FIGURE 22



542 087

FIGURE 23

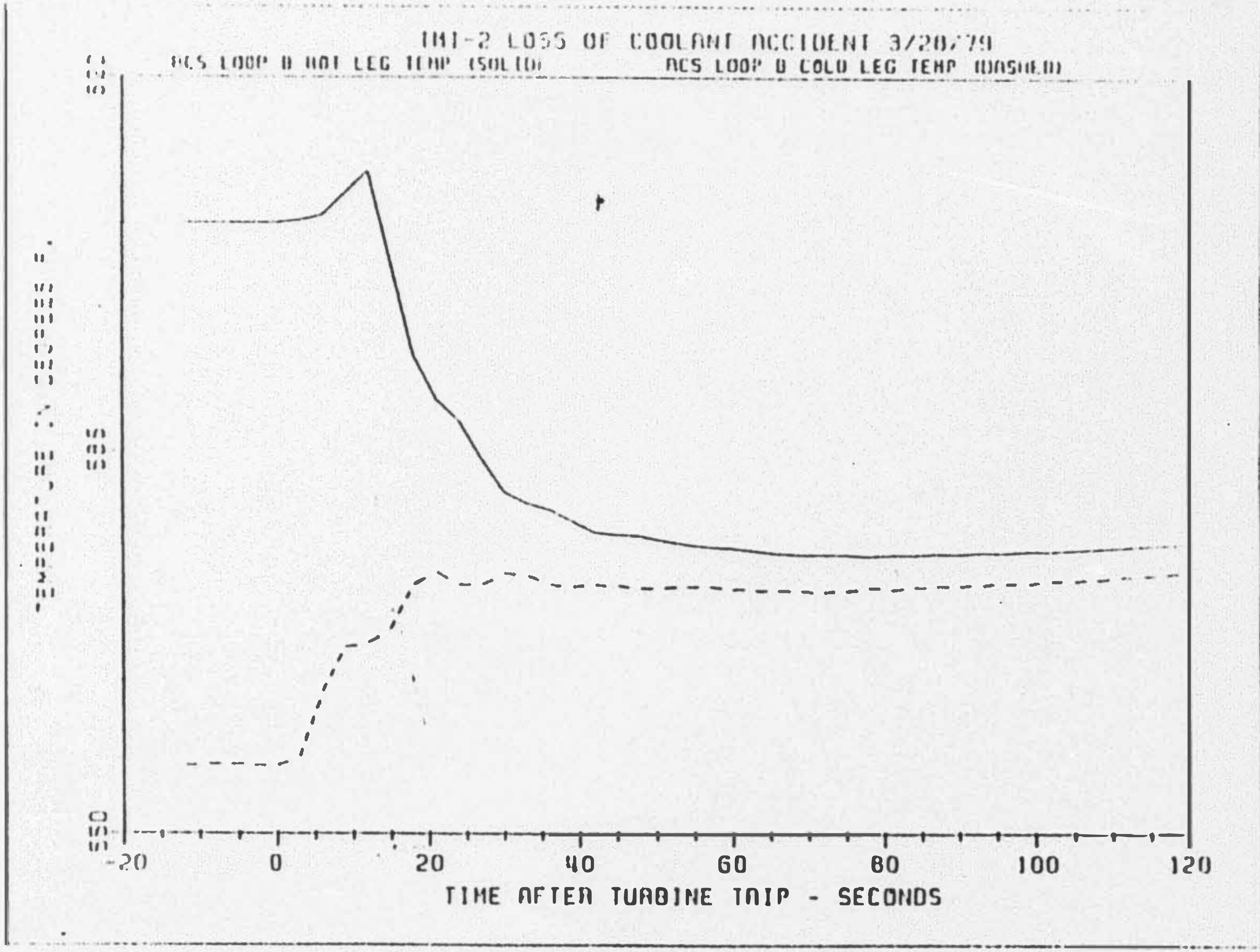
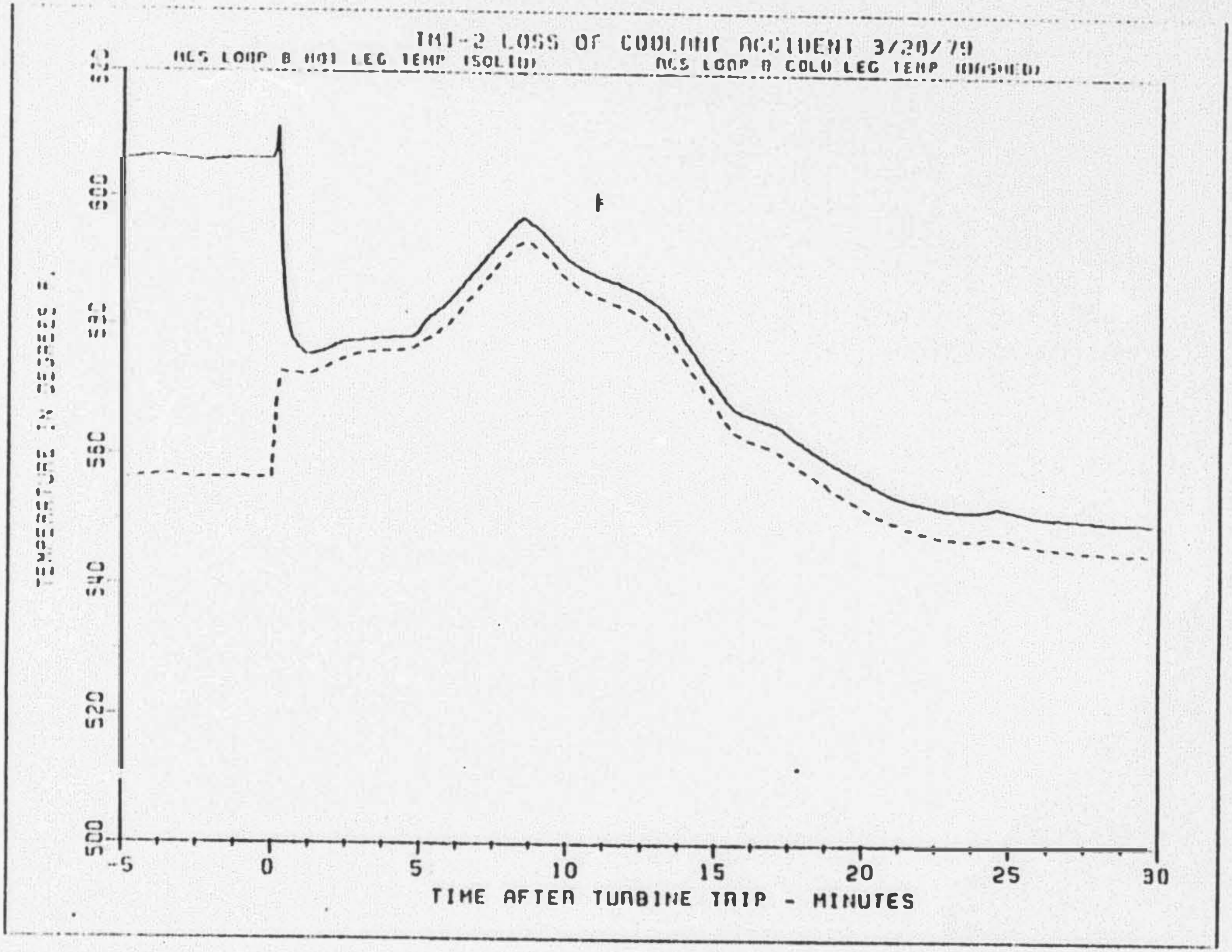

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
RCS LOOP B HOT LEG TEMP (SOLID)
RCS LOOP B COLD LEG TEMP (DASHED)
600
550
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - SECONDS

FIGURE 24



542 091

FIGURE 25

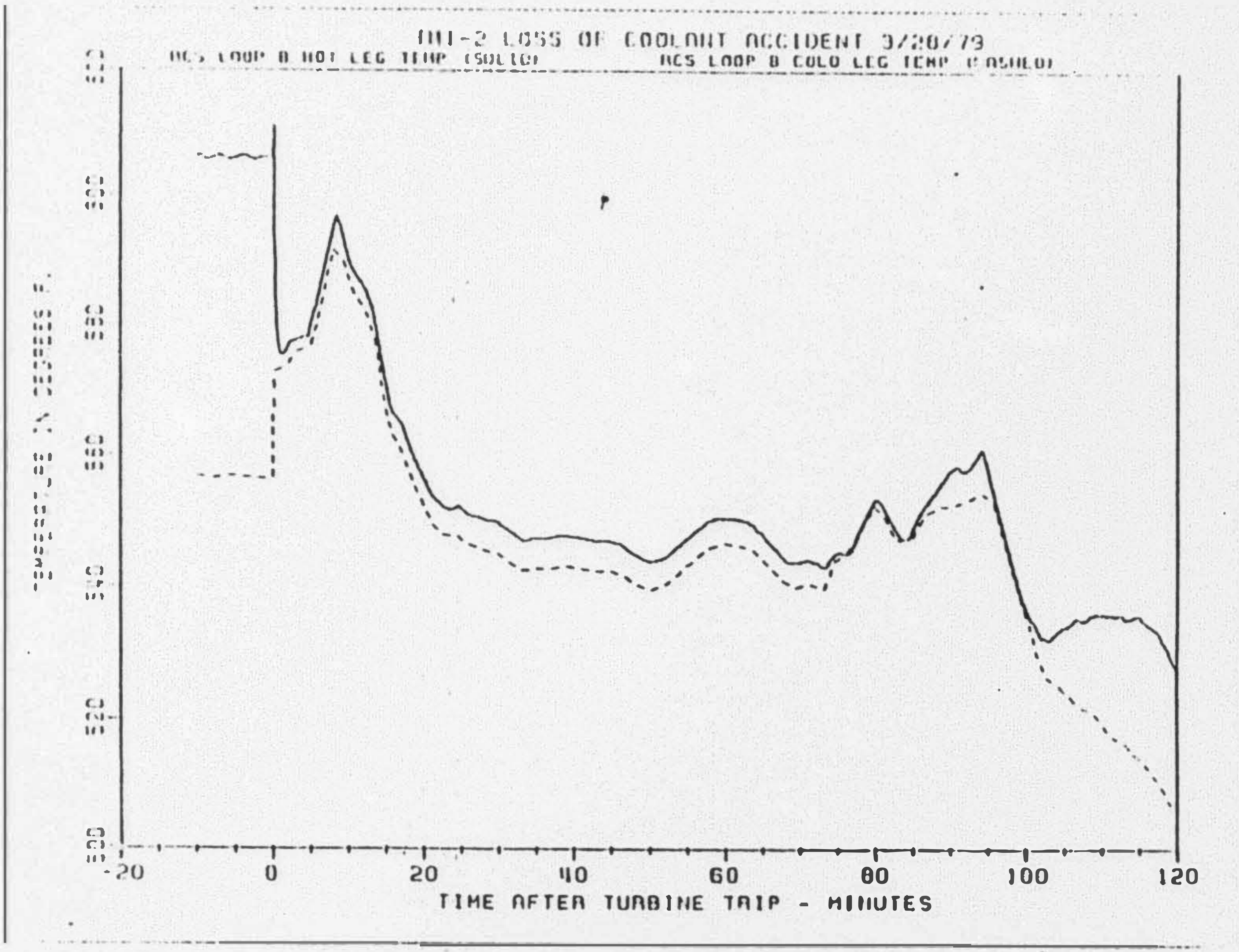
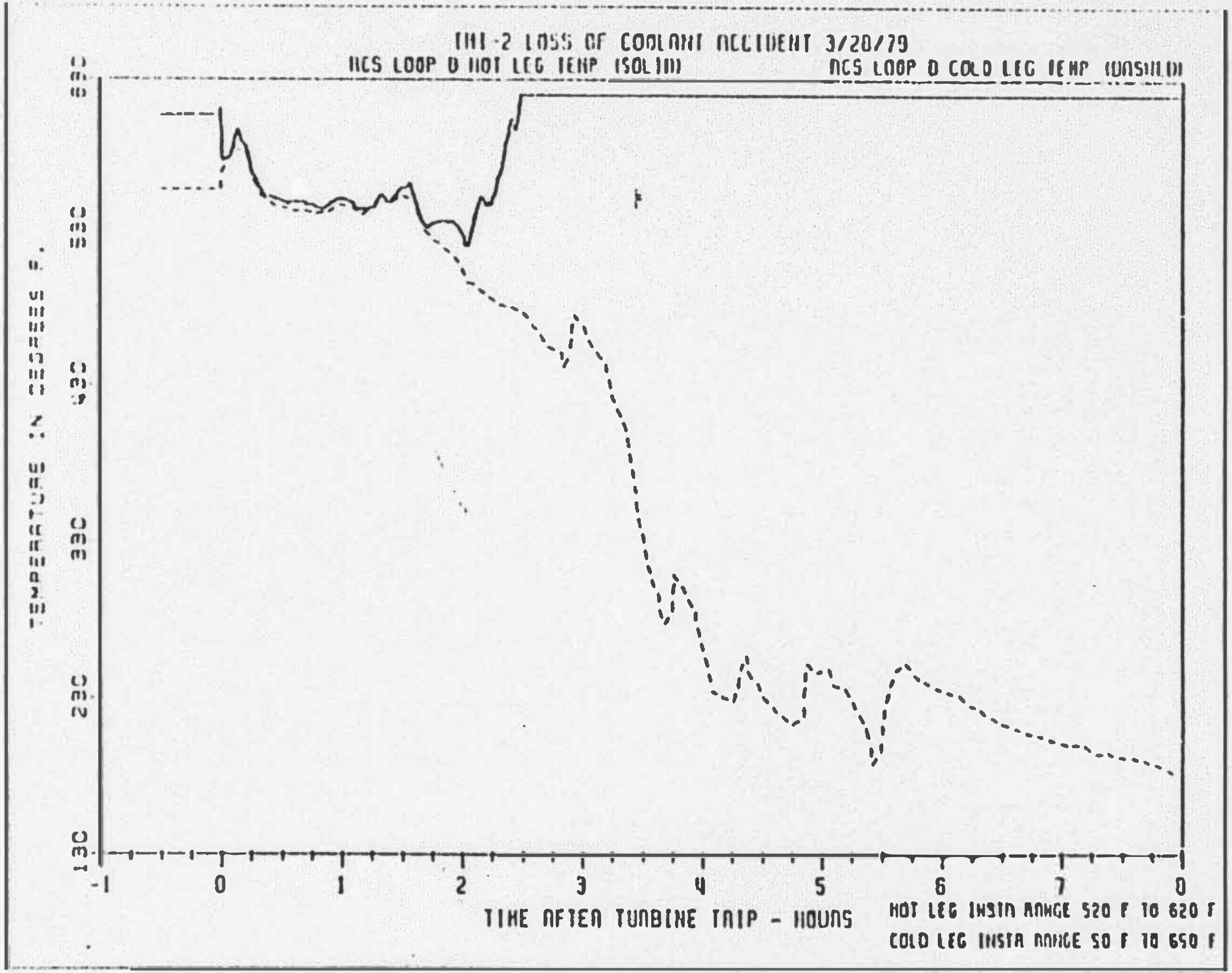
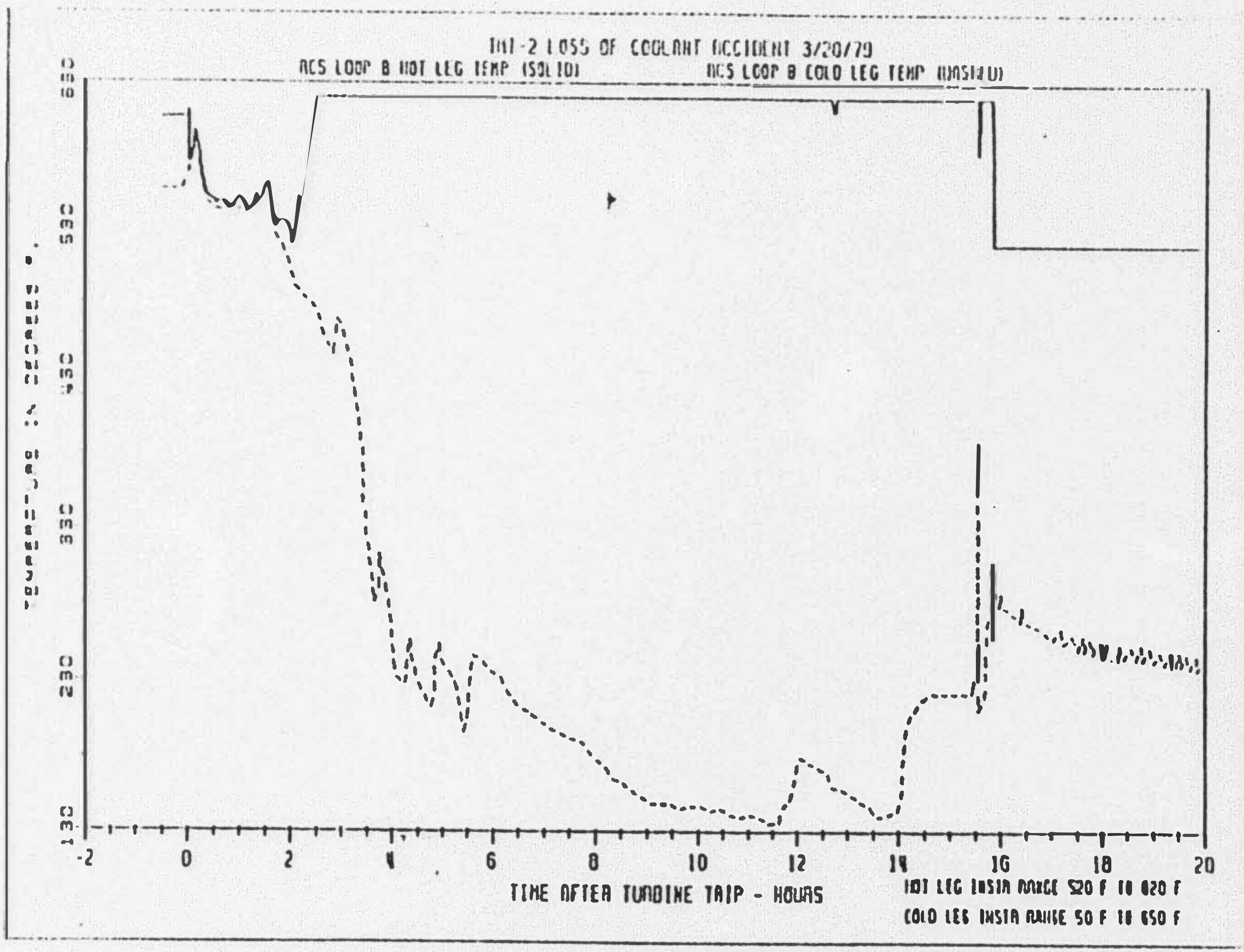

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
RCS LOOP B HOT LEG TEMP (SOLID)
RCS LOOP B COLD LEG TEMP (DASHED)
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - MINUTES

FIGURE 26



542 093

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
630
530
430
330
230
130
-2
0
2
4
6
8
10
12
14
16
18
20
TIME AFTER TURBINE TRIP - HOURS
HOT LEG INSTR RANGE 520 F TO 620 F
COLD LEG INSTR RANGE 50 F TO 650 F

FIGURE 28

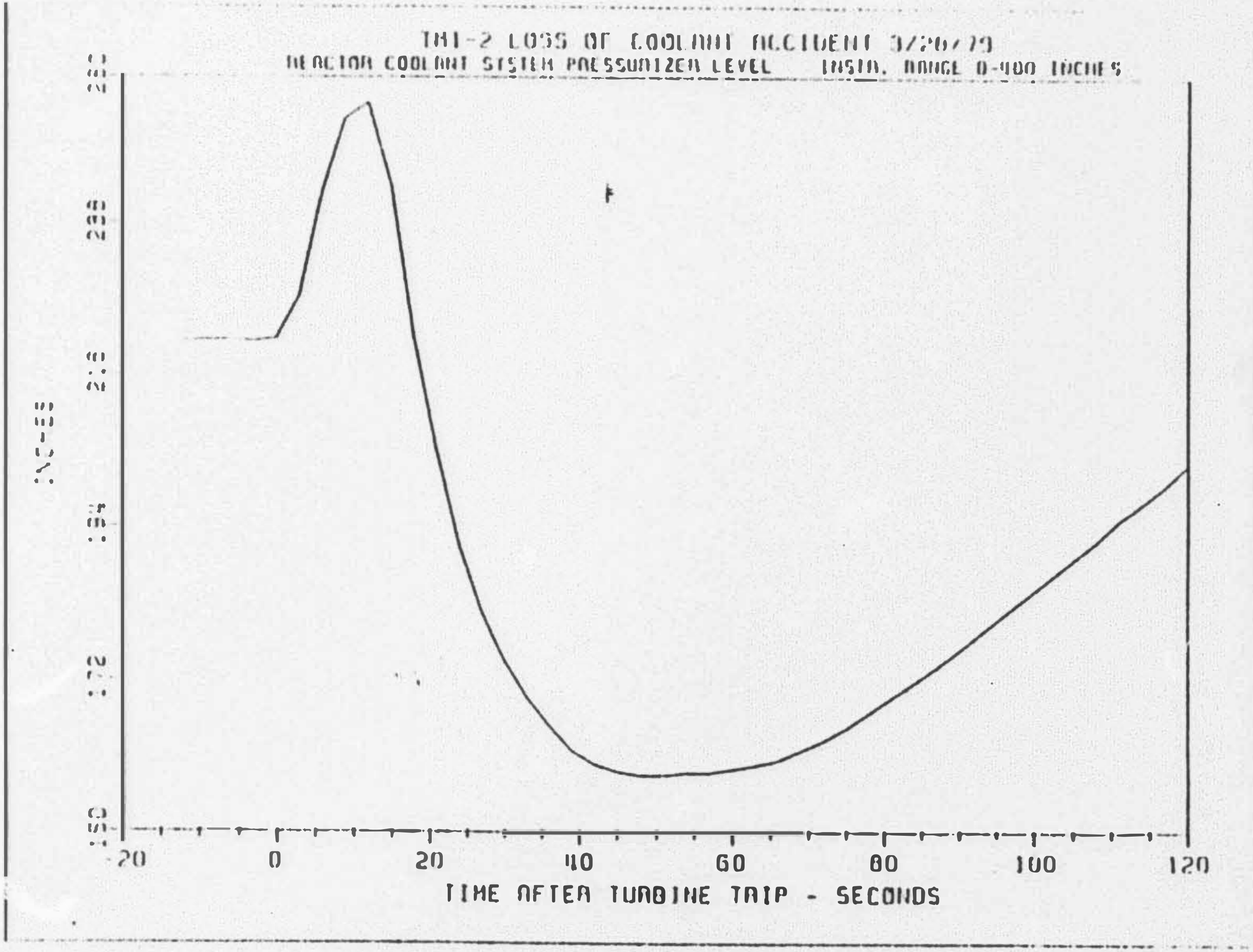

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
REACTOR COOLANT SYSTEM PRESSURIZER LEVEL   INSTR. RANGE 0-400 INCHES
INCHES
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - SECONDS

FIGURE 29

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
REACTOR COOLANT SYSTEM PRESSURIZER LEVEL   INSTR. RANGE 0-400 INCHES

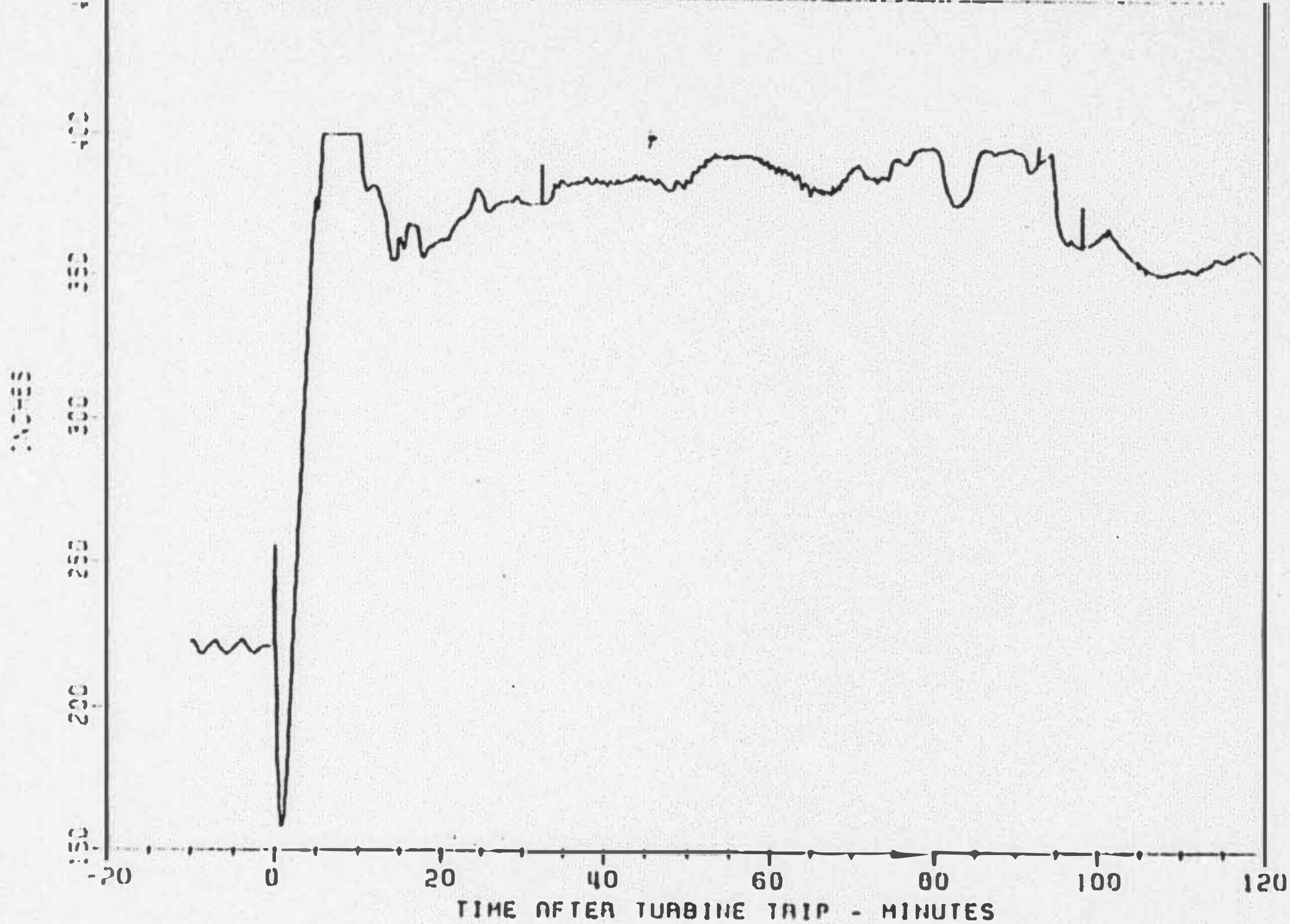


FIGURE 30

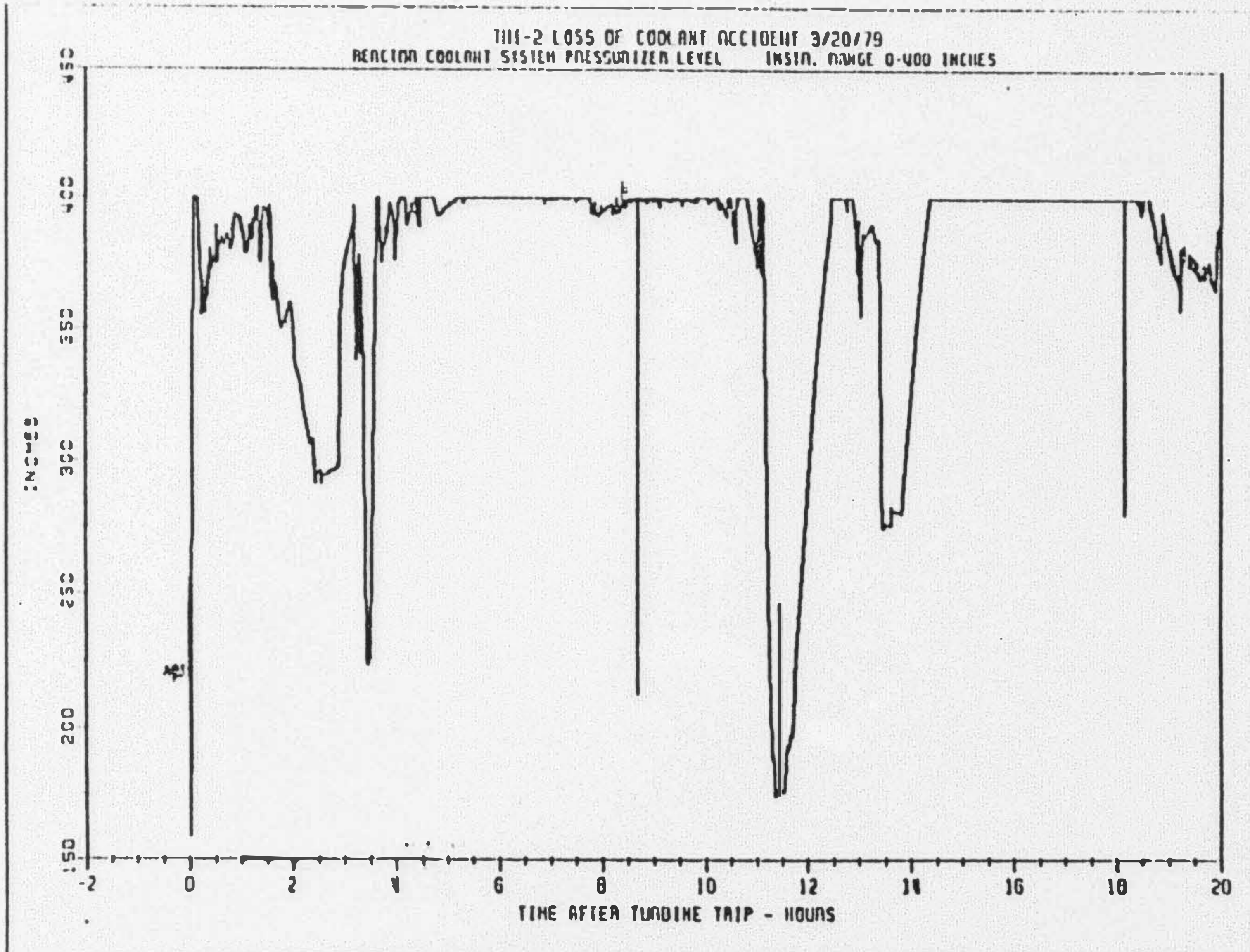

TMI-2 LOSS OF COOLANT ACCIDENT 3/20/79
REACTOR COOLANT SYSTEM PRESSURIZER LEVEL INSTR. RANGE 0-400 INCHES
INCHES
450
400
350
300
250
200
150
-2
0
2
4
6
8
10
12
14
16
18
20
TIME AFTER TURBINE TRIP - HOURS

FIGURE 31

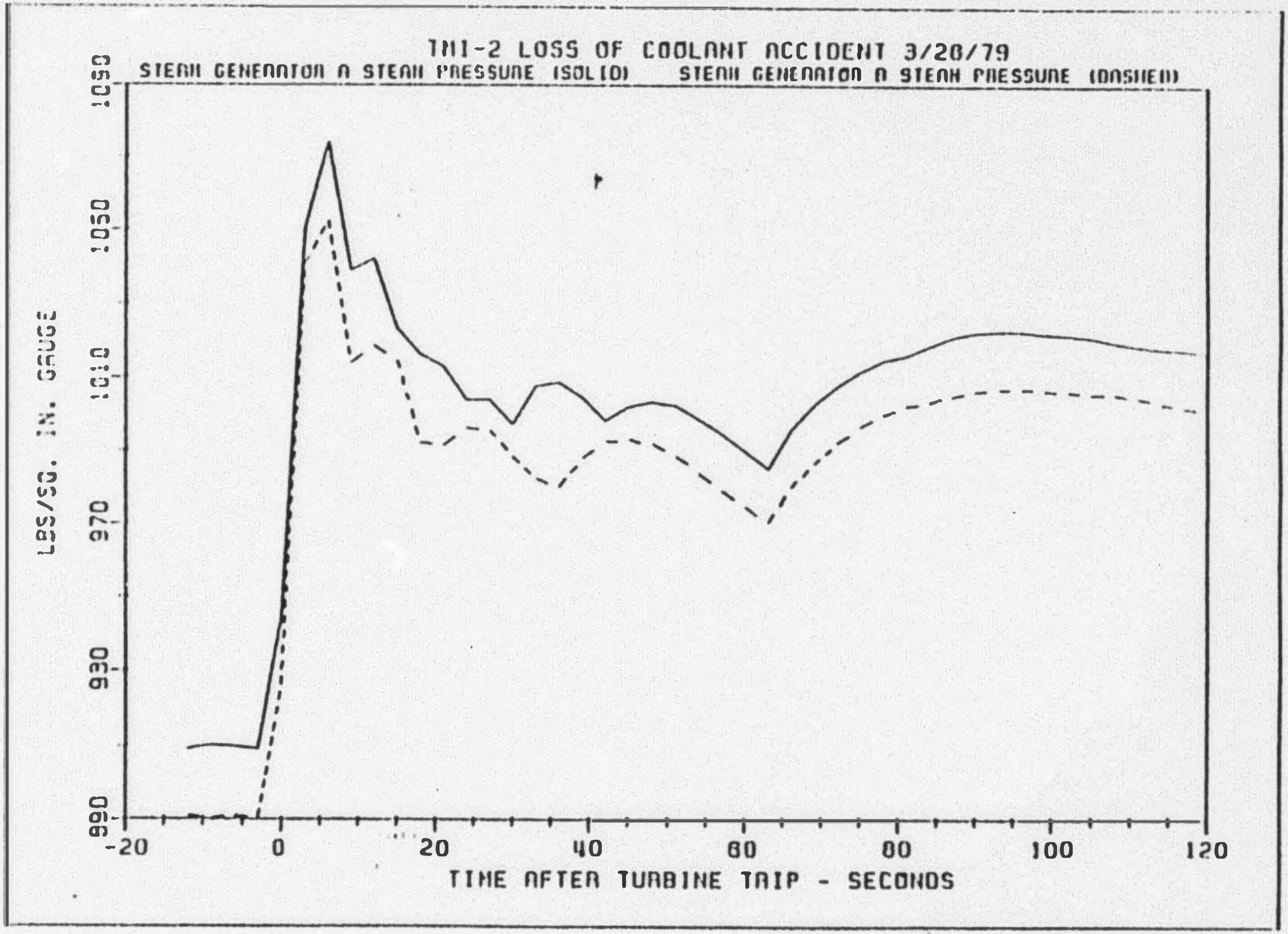
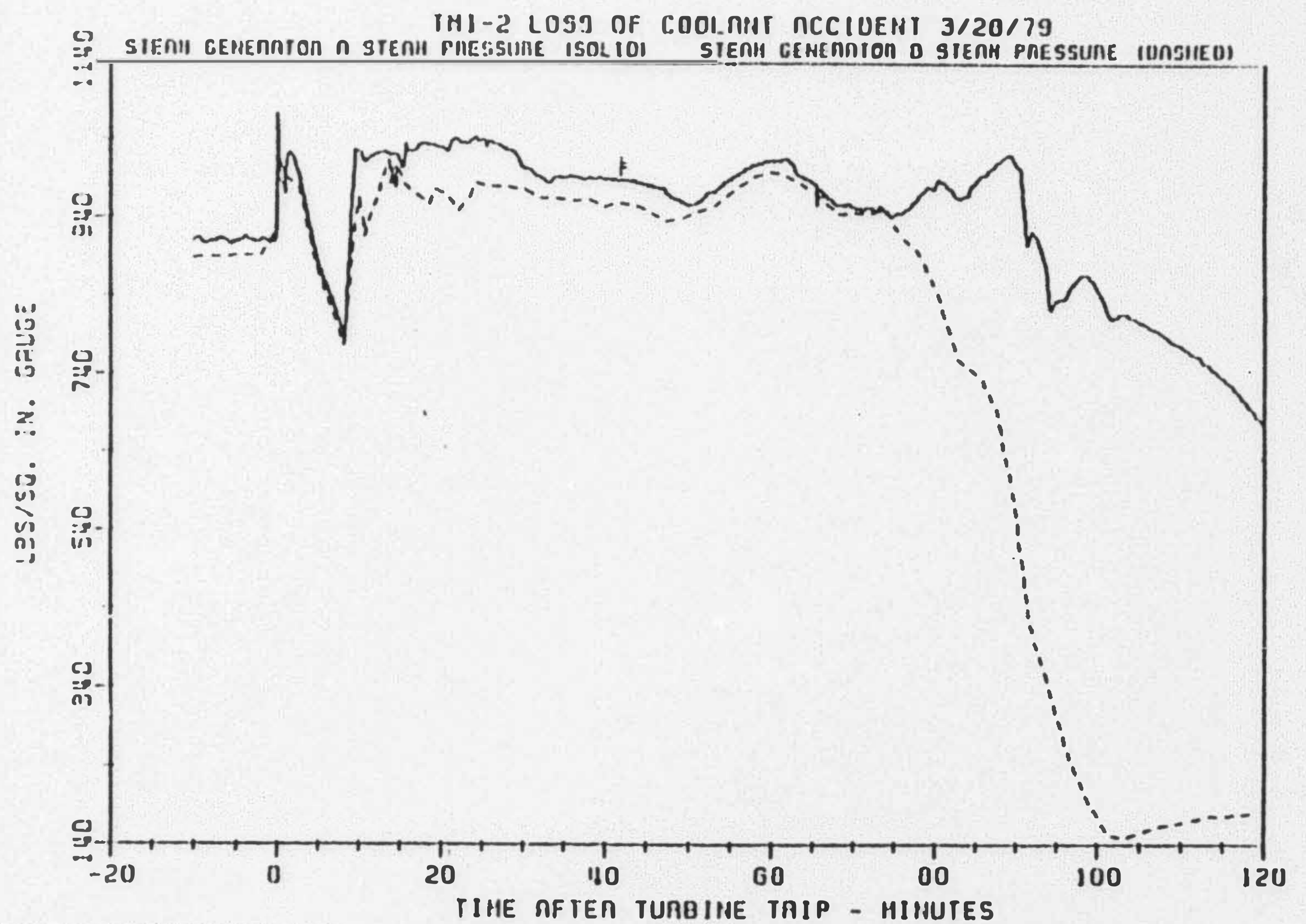
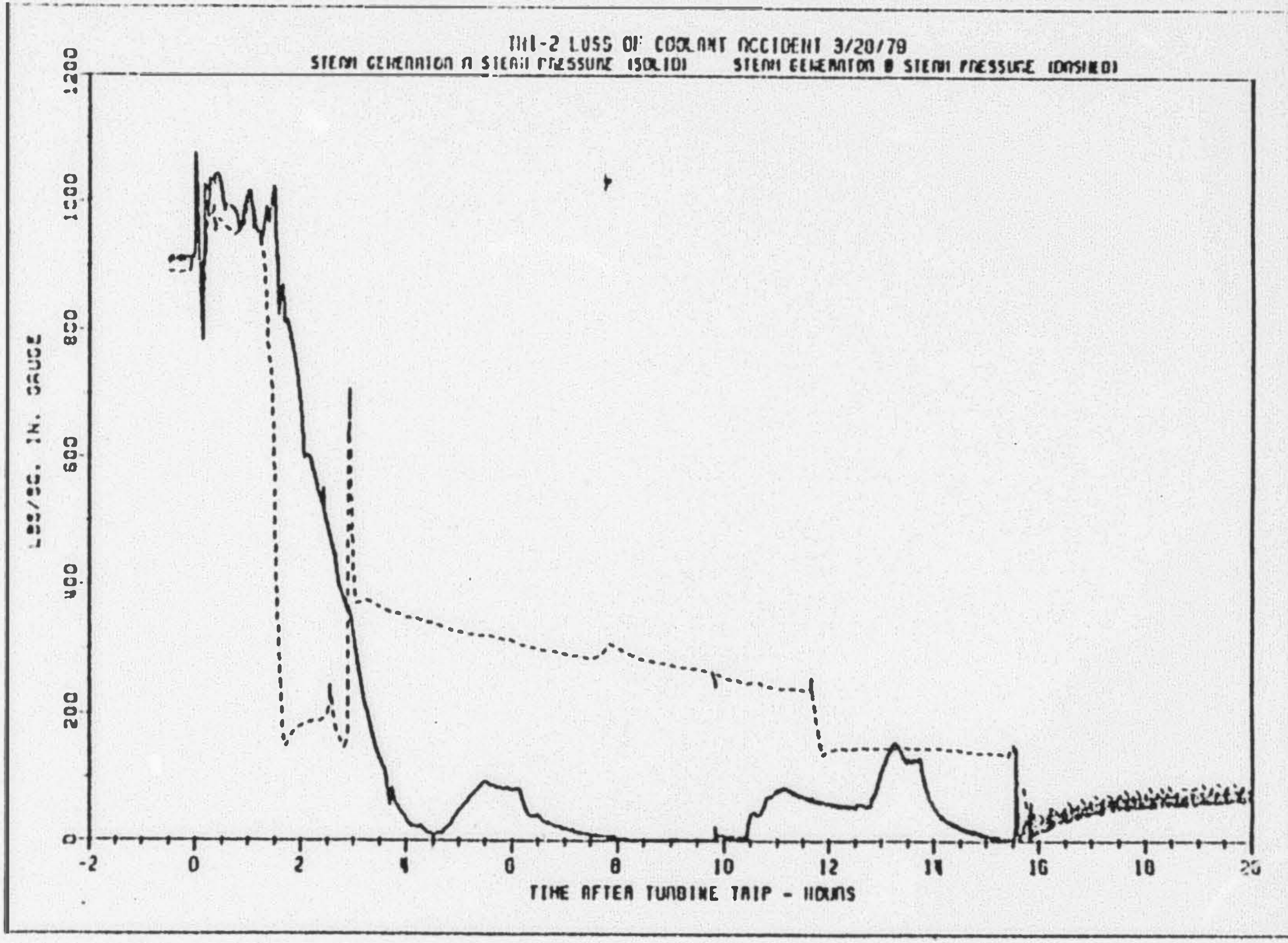

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
STEAM GENERATOR A STEAM PRESSURE (SOLID)
STEAM GENERATOR B STEAM PRESSURE (DASHED)
LBS/SQ. IN. GAUGE
890
930
970
1010
1050
1090
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - SECONDS

FIGURE 32



542 090

FIGURE 33



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
0
200
400
600
800
1000
1200
-2
0
2
4
6
8
10
12
14
16
18
20
TIME AFTER TURBINE TRIP - HOURS

FIGURE 34

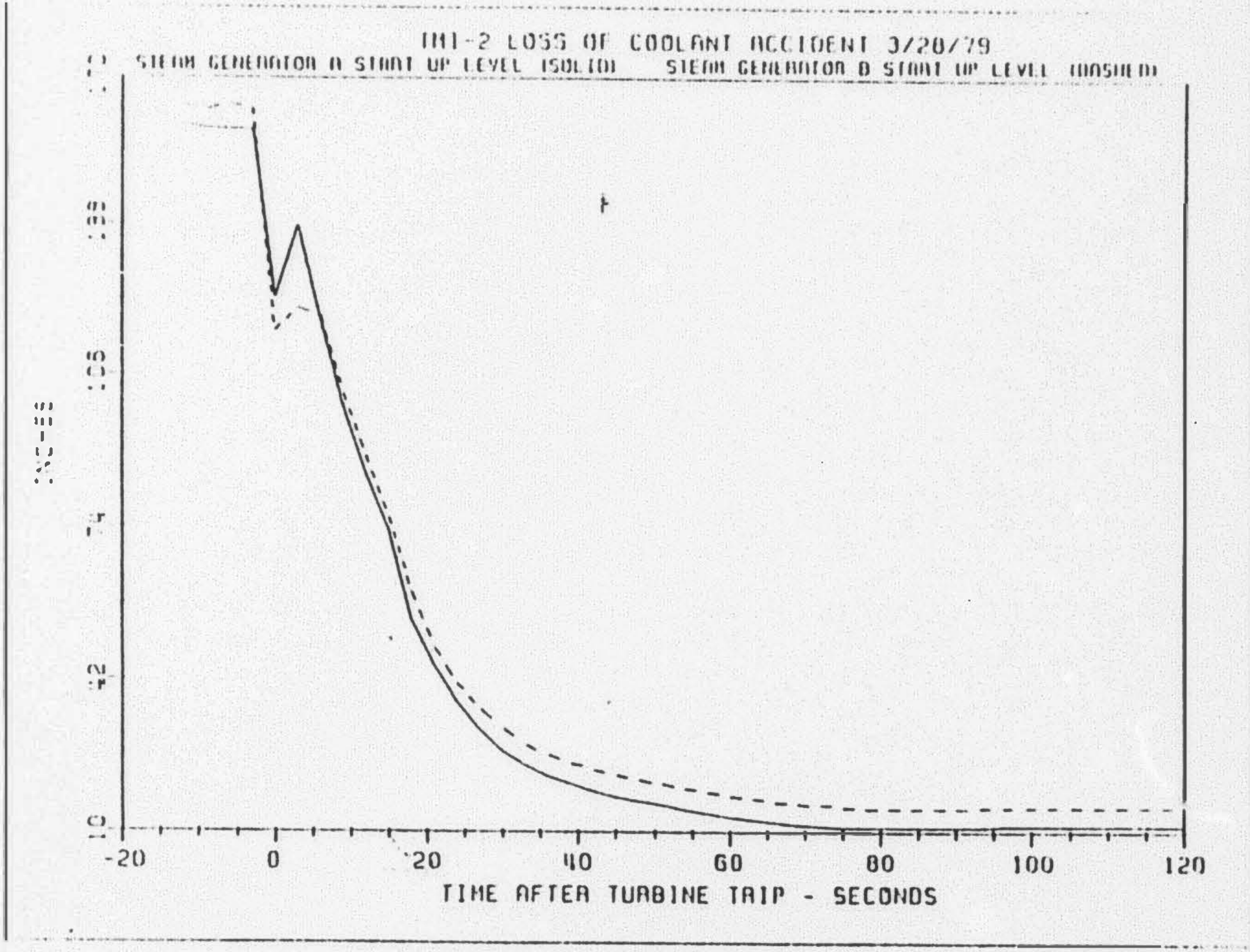

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
STEAM GENERATOR A START UP LEVEL (SOLID)
STEAM GENERATOR B START UP LEVEL (DASHED)
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - SECONDS

FIGURE 35

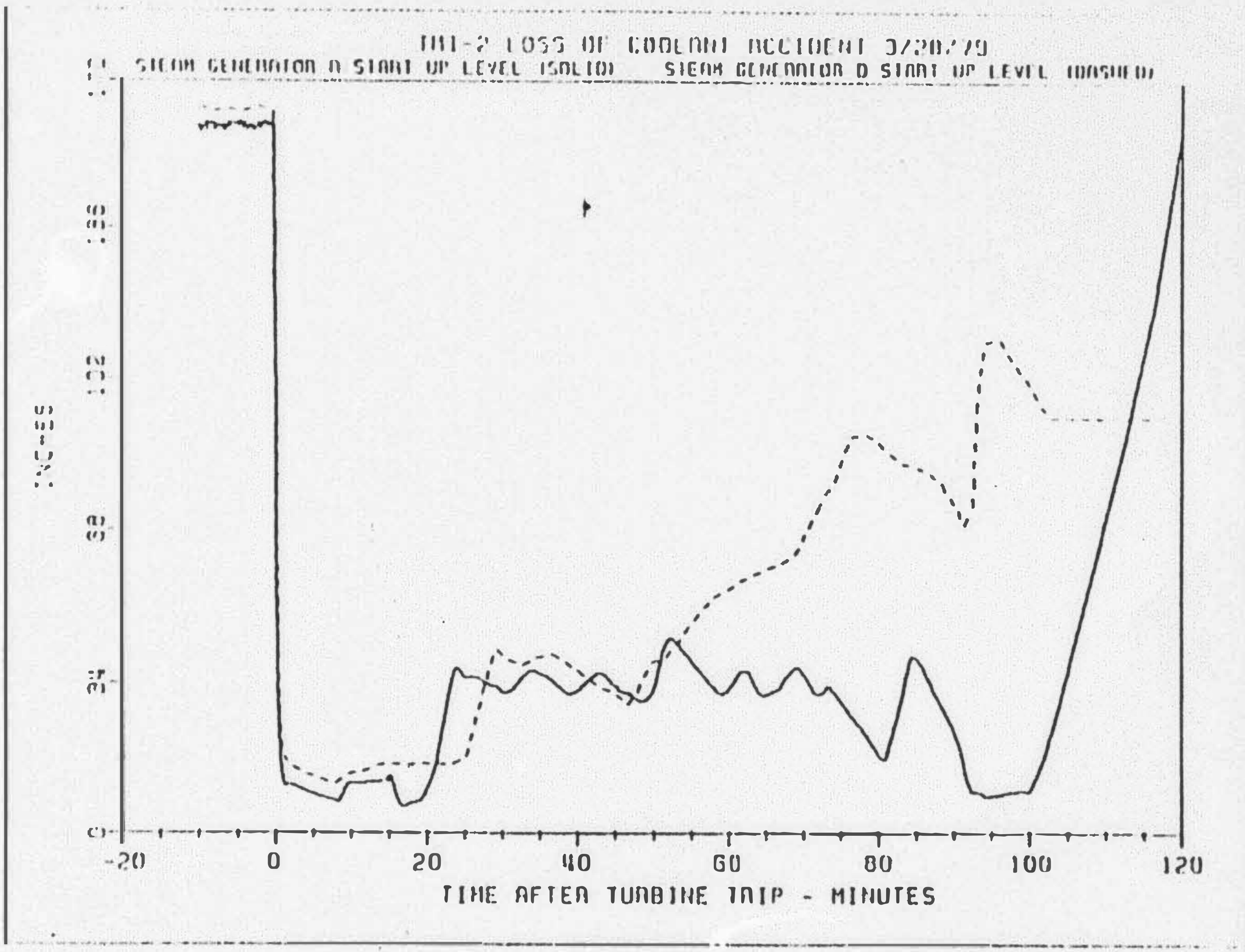
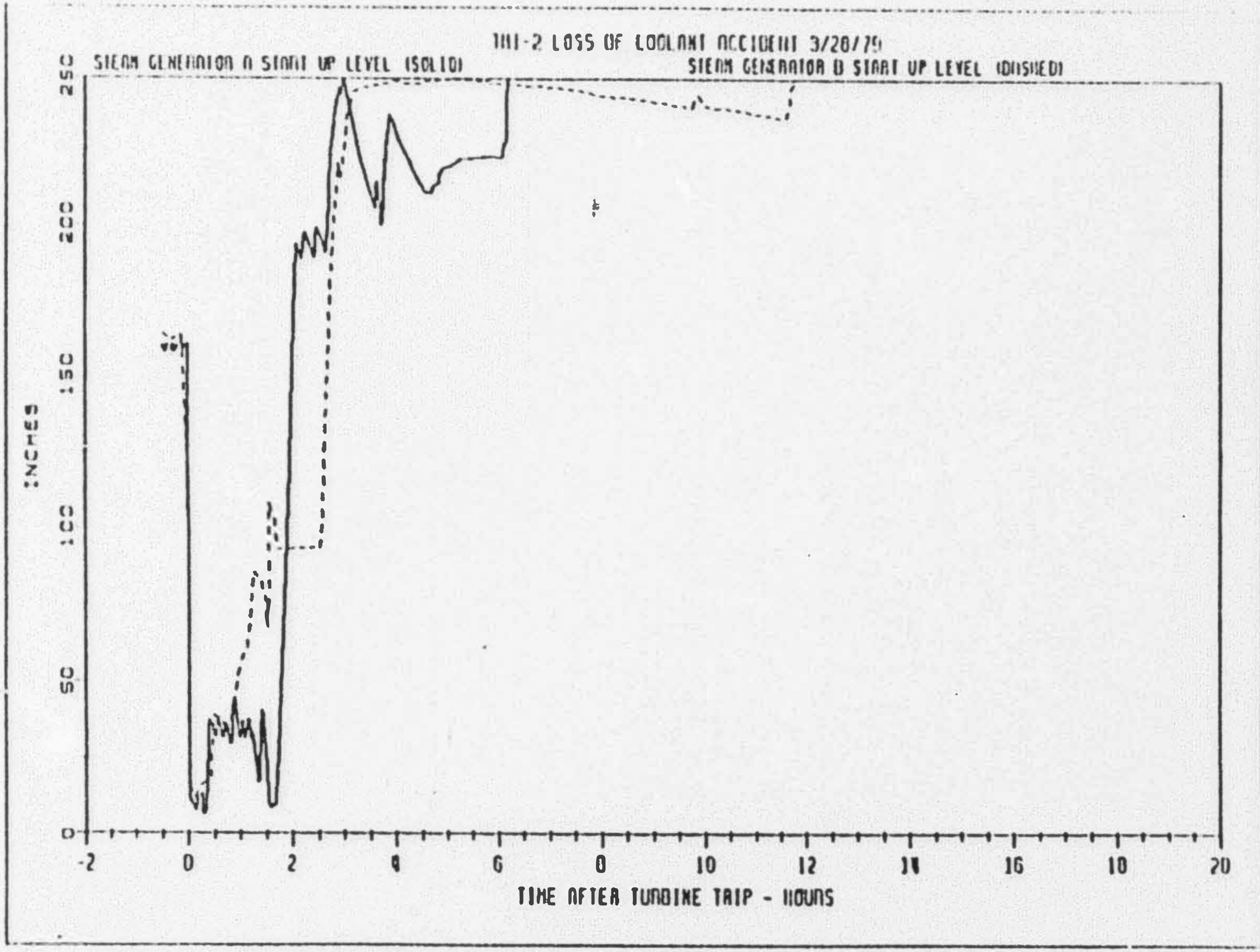
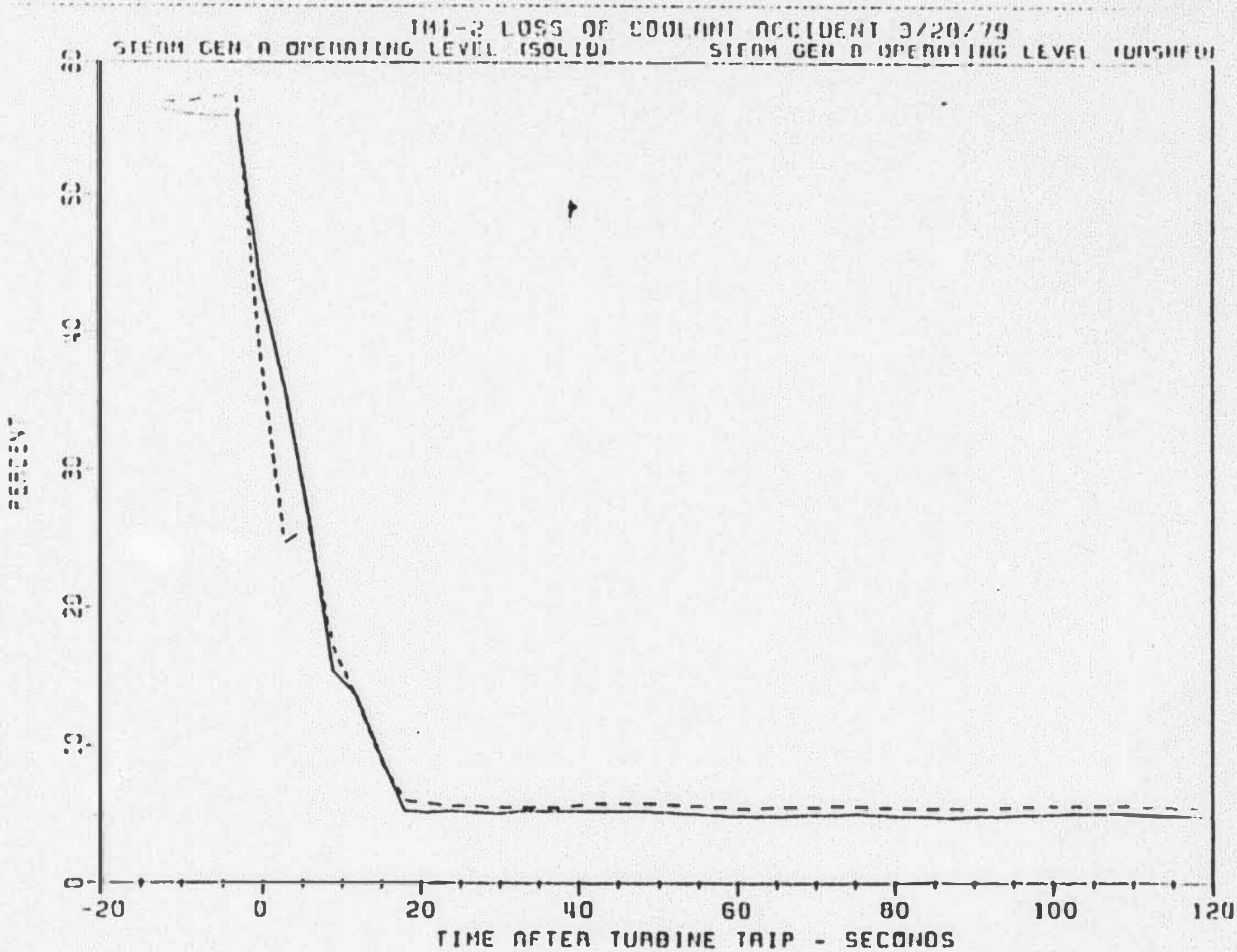

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
STEAM GENERATOR A START UP LEVEL (SOLID)
STEAM GENERATOR B START UP LEVEL (DASHED)
INCHES
0
40
80
120
160
200
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - MINUTES

FIGURE 36



542 103

FIGURE 37



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 - SECONDS

FIGURE 38

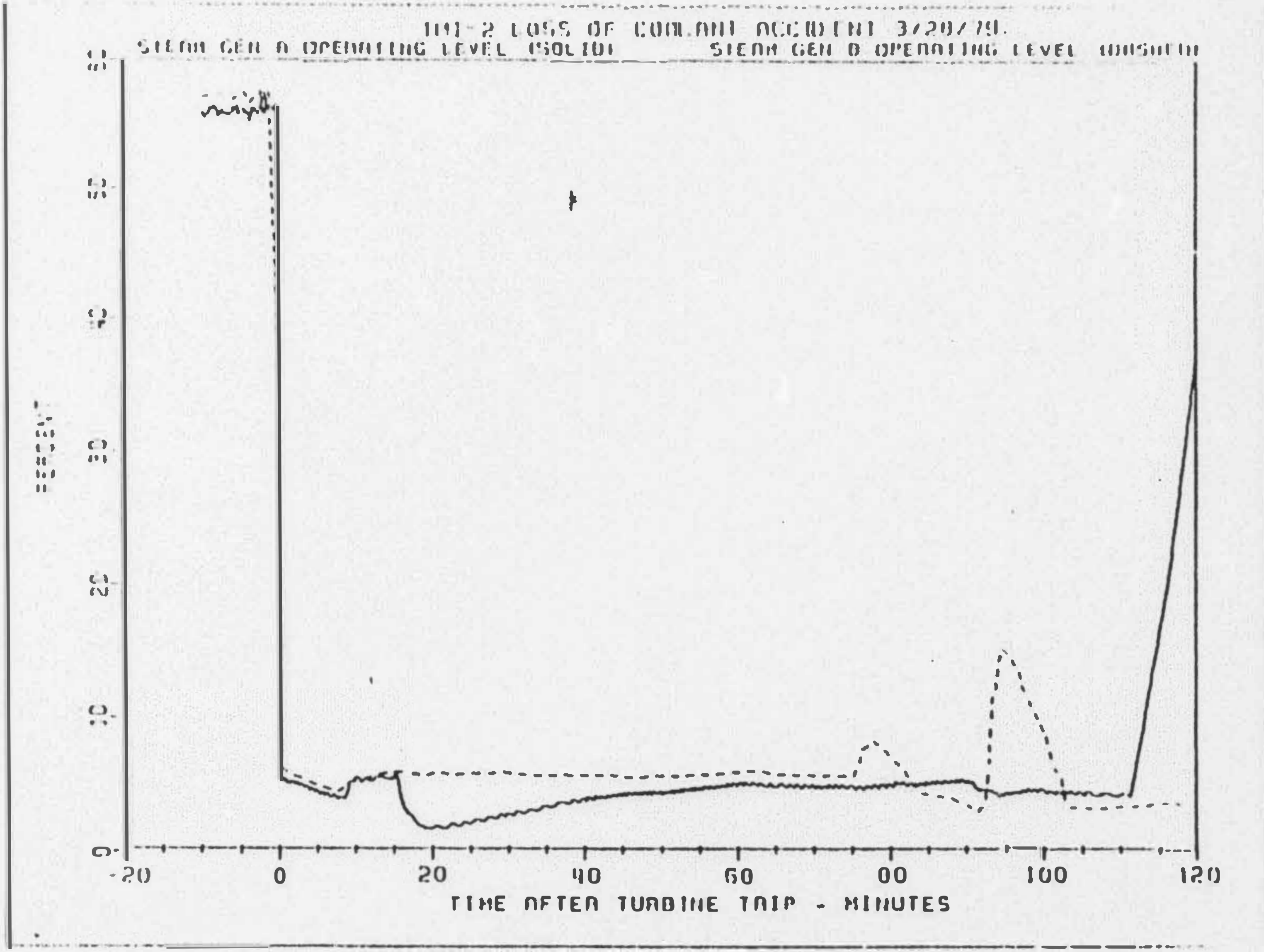

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79.
STEAM GEN A OPERATING LEVEL (SOLID)
STEAM GEN B OPERATING LEVEL (DASHED)
PERCENT
0
10
20
30
40
50
60
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - MINUTES

FIGURE 39

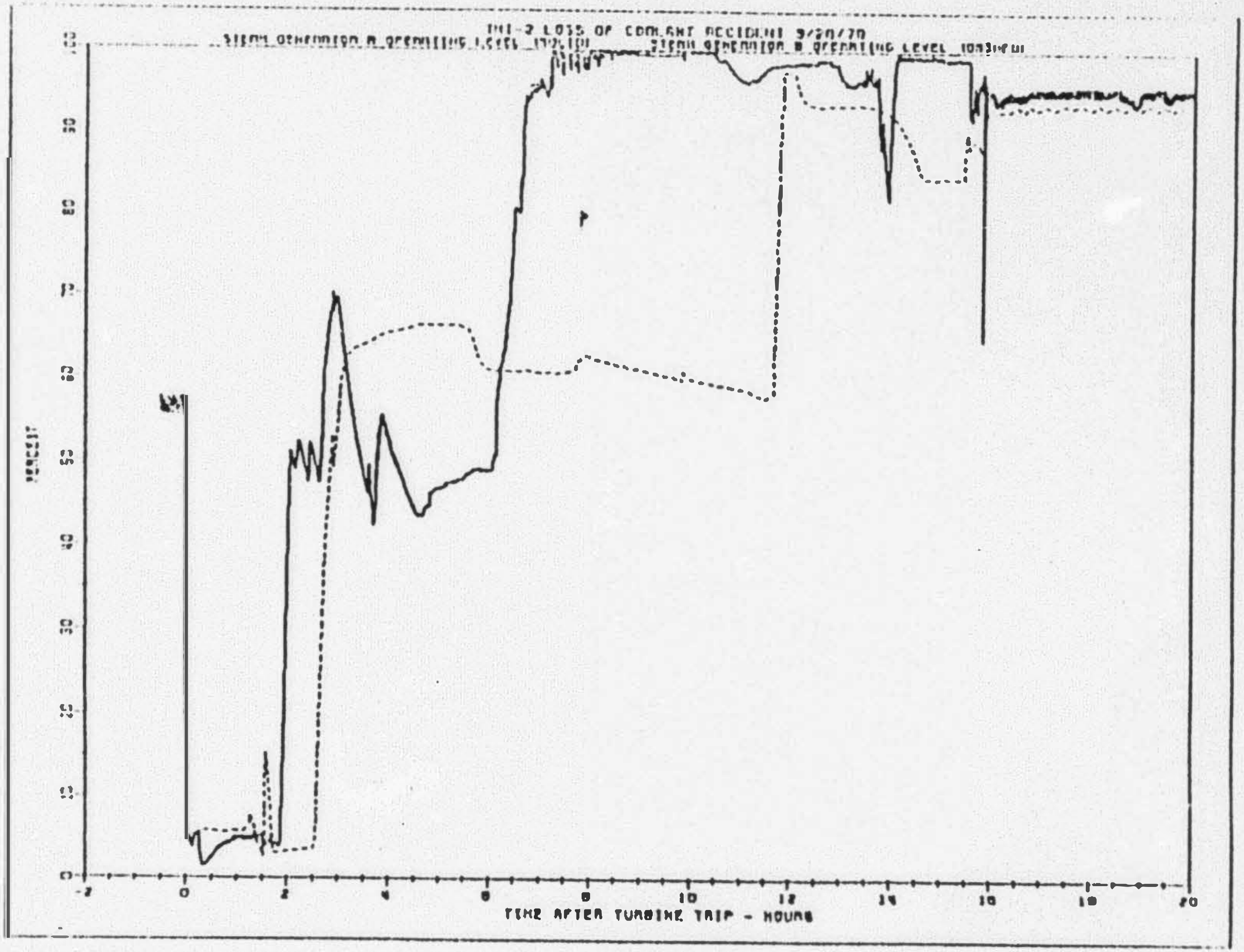

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
STEAM GENERATOR A OPERATING LEVEL
STEAM GENERATOR B OPERATING LEVEL
PERCENT
TIME AFTER TURBINE TRIP - HOURS
-2
0
2
4
6
8
10
12
14
16
18
20

FIGURE 40

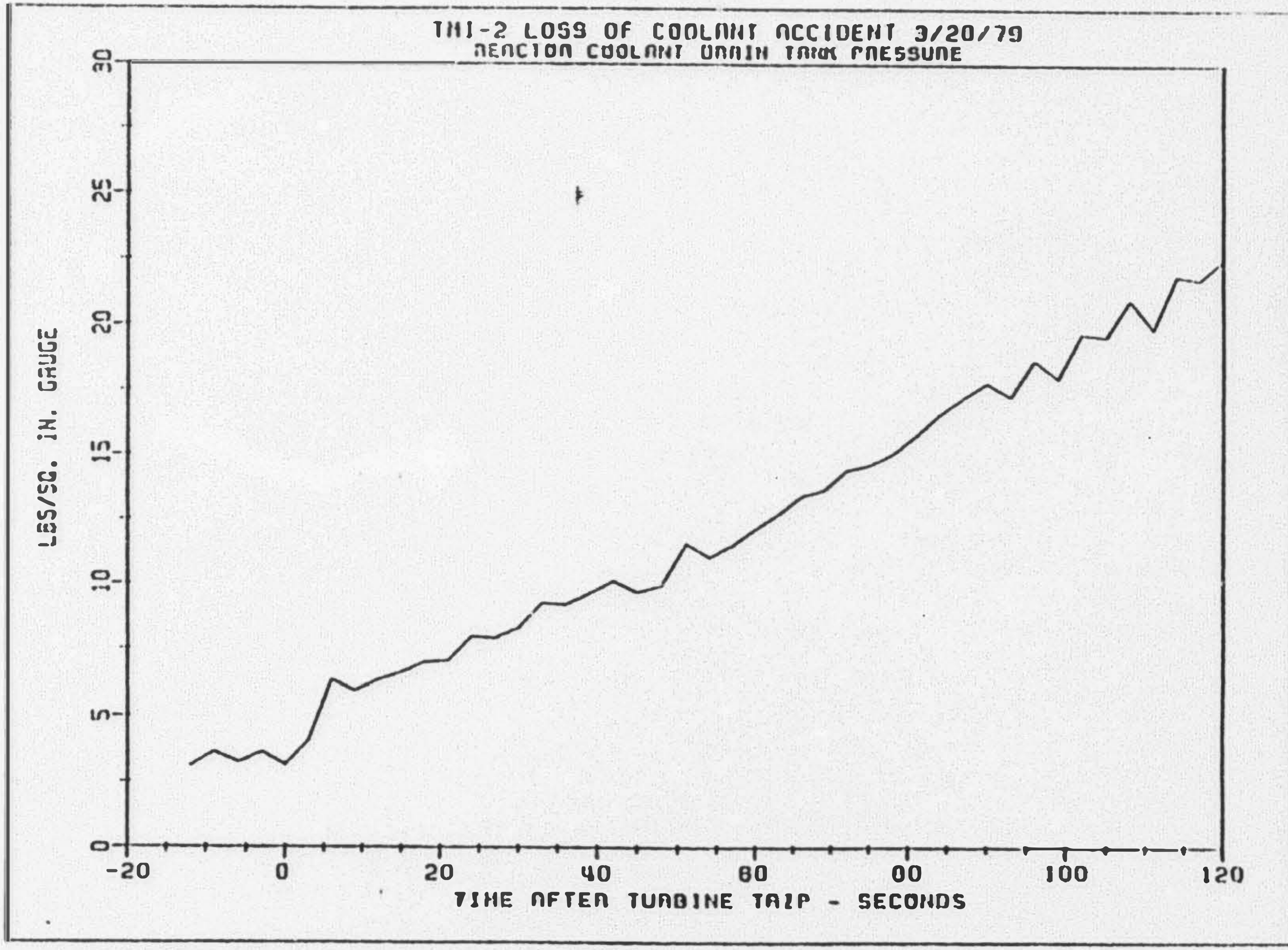

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
REACTOR COOLANT DRAIN TANK PRESSURE
LBS/SQ. IN. GAUGE
0
5
10
15
20
25
30
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - SECONDS

FIGURE 41

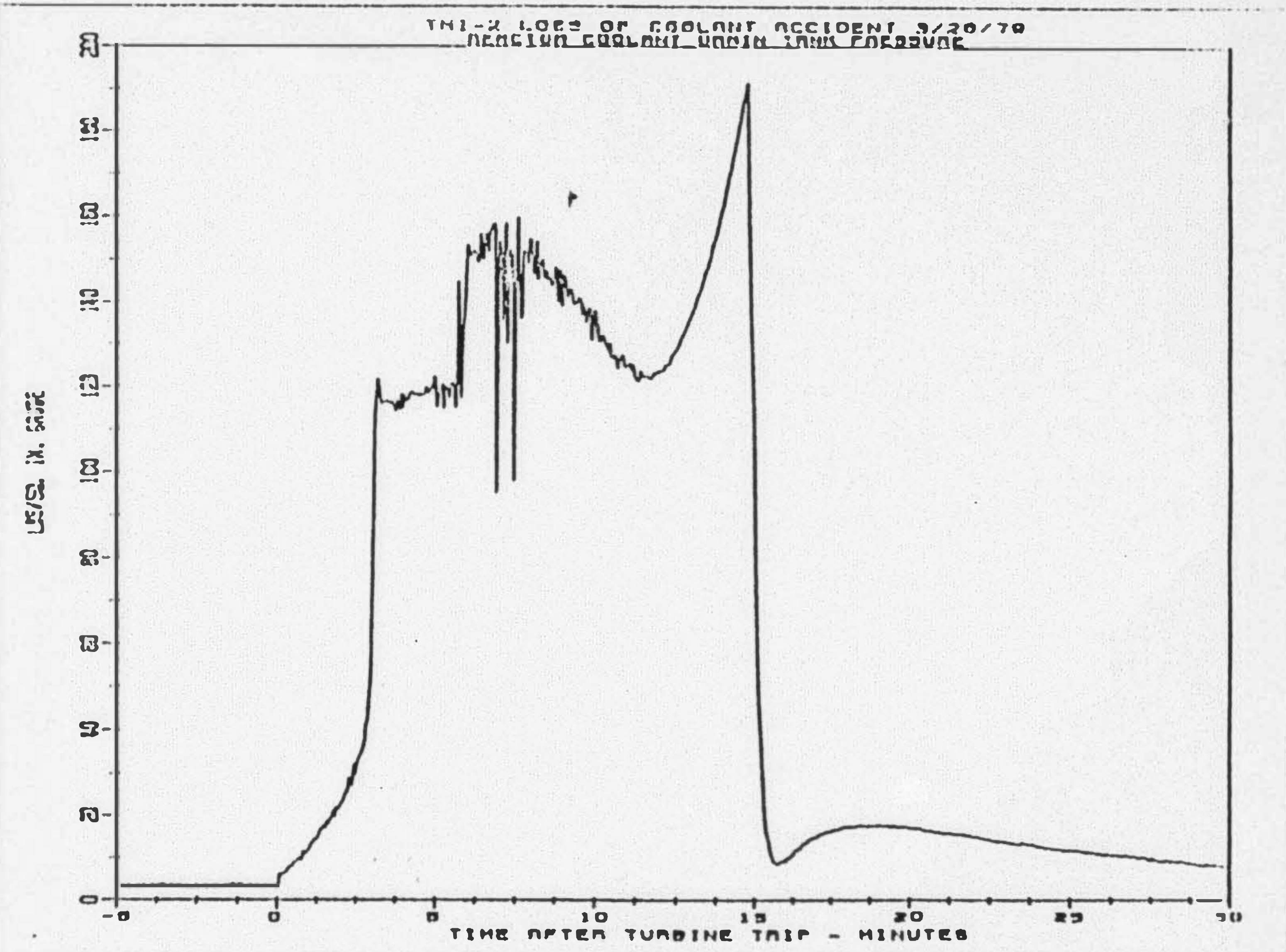

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
REACTOR COOLANT DRAIN TANK PRESSURE
-5
0
5
10
15
20
25
30
TIME AFTER TURBINE TRIP - MINUTES

FIGURE 42

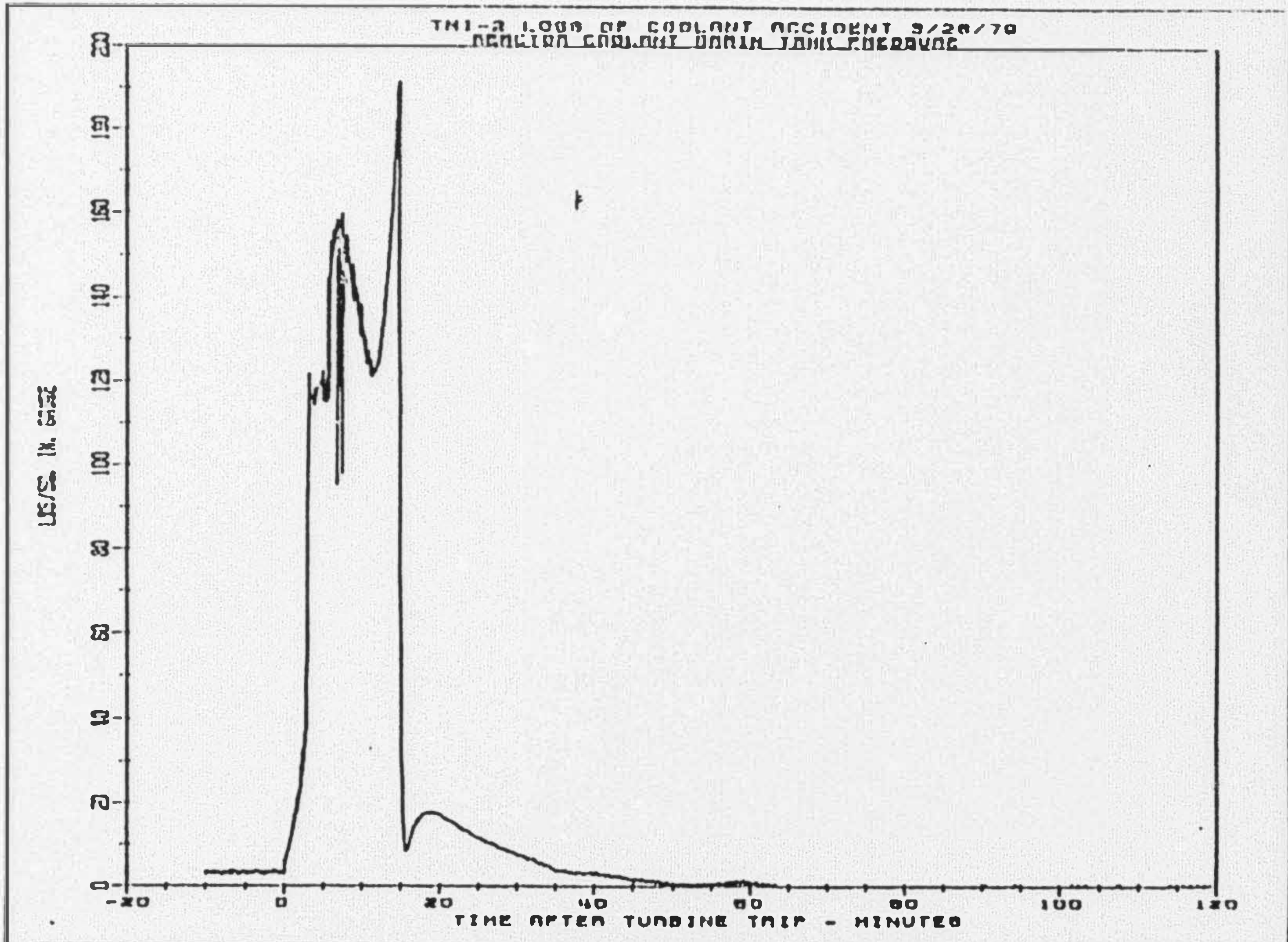

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
REACTOR COOLANT DRAIN TANK PRESSURE
200
180
160
140
120
100
80
60
40
20
0
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - MINUTES

FIGURE 43

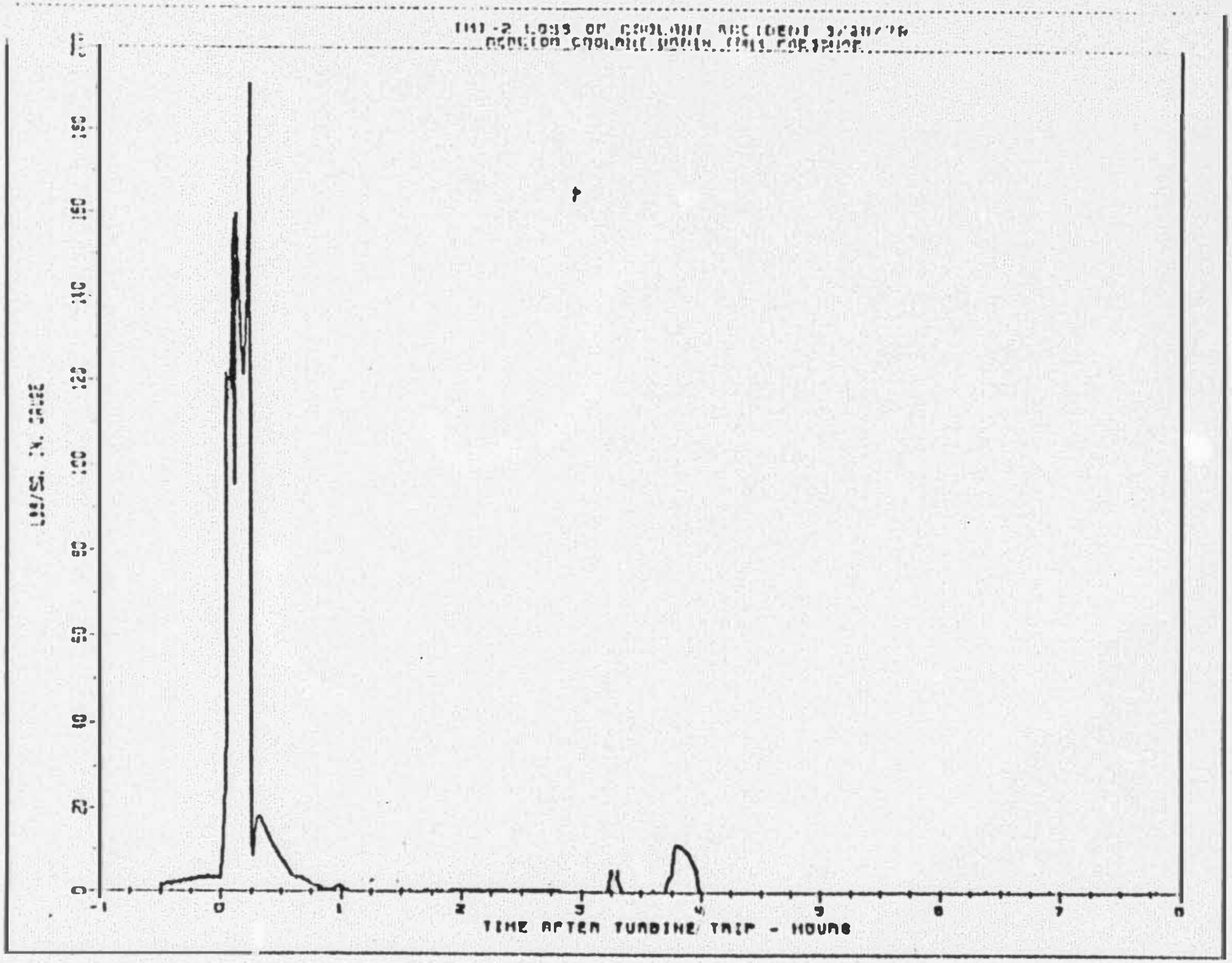
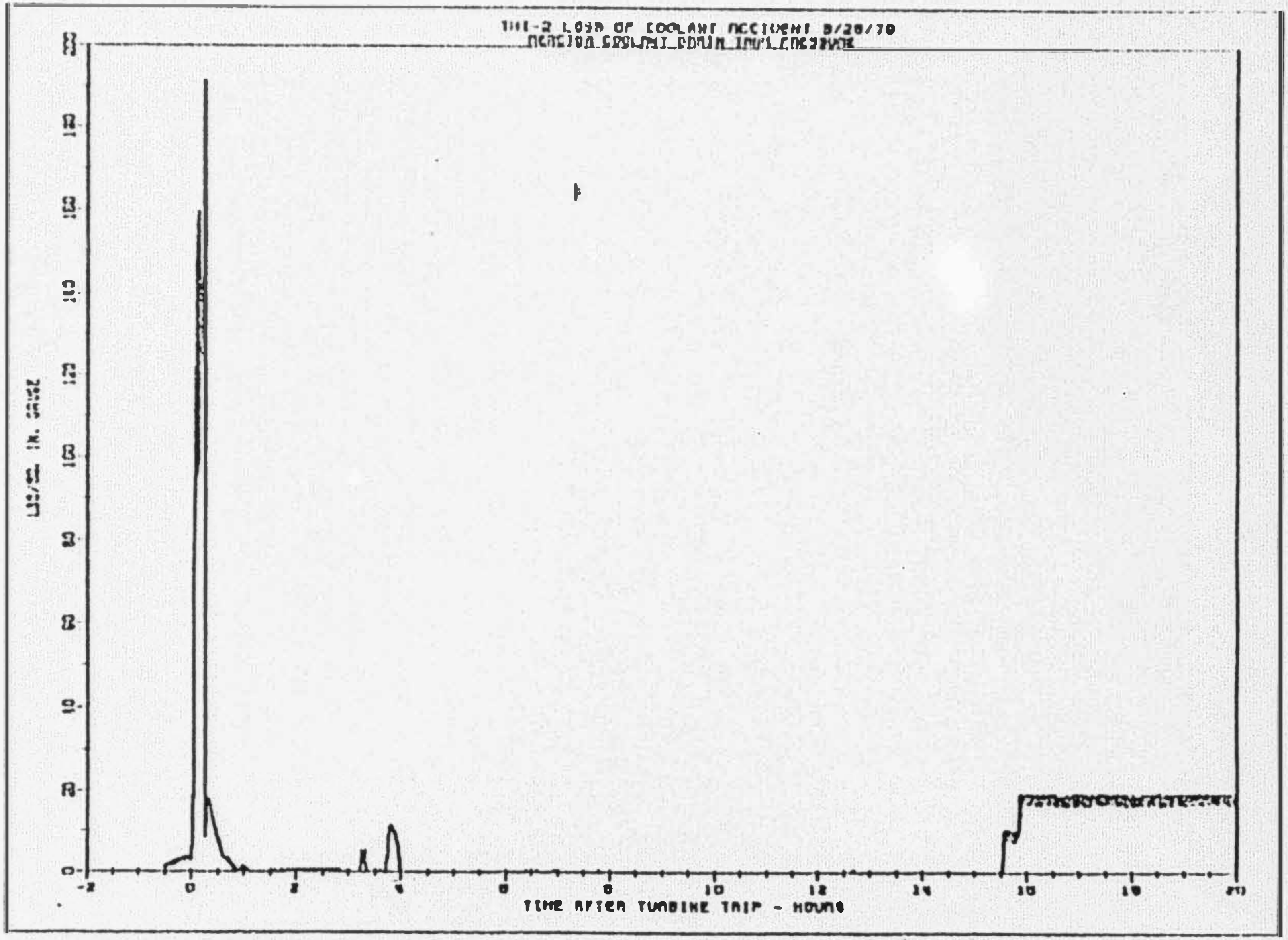


FIGURE 44



TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
REACTOR COOLANT DRAIN TANK PRESSURE
LBS/SQ. IN. GAUGE
TIME AFTER TURBINE TRIP - HOURS
-2
0
2
4
6
8
10
12
14
16
18
20

Figure 45
TMI-2 Loss of Coolant Accident 3/28/79
Reactor Building Temperature and Pressure

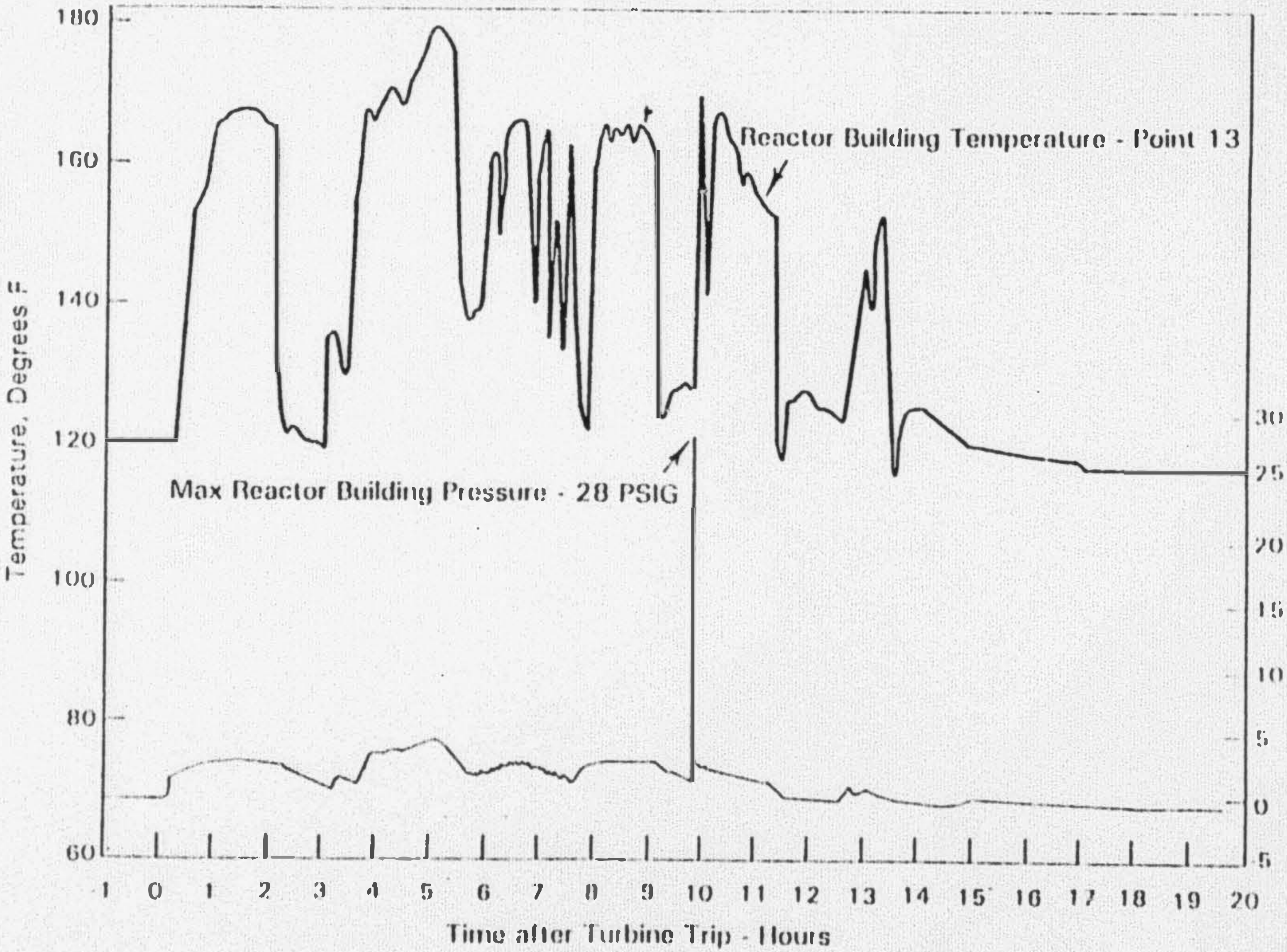


Figure 46
TMI-2 Loss of Coolant Accident of 3/28/79
Intermediate and Source Range Nuclear Instrumentation

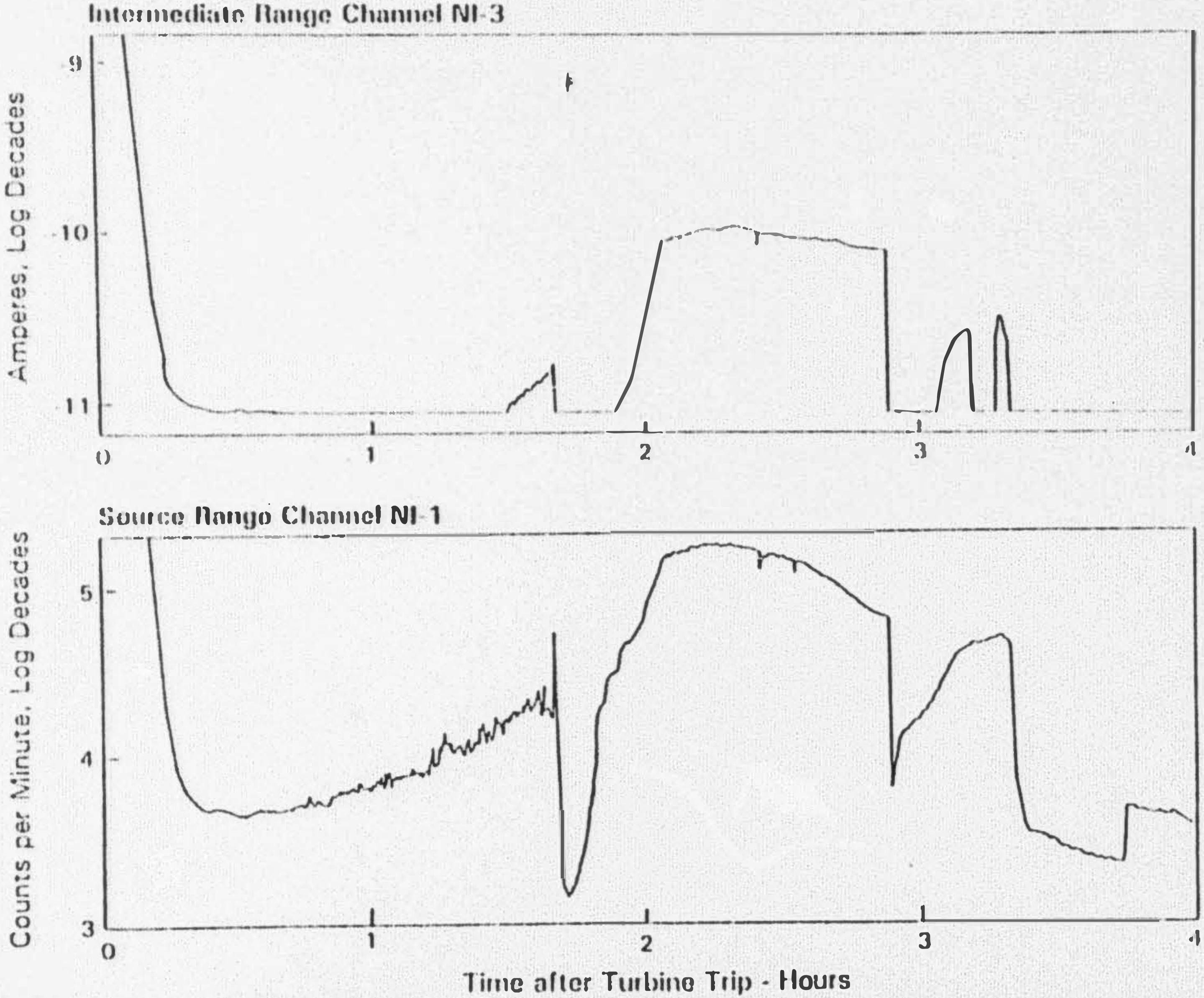


Figure 47
TMI-2 Loss of Coolant Accident of 3/28/79
Intermediate and Source Range Nuclear Instrumentation

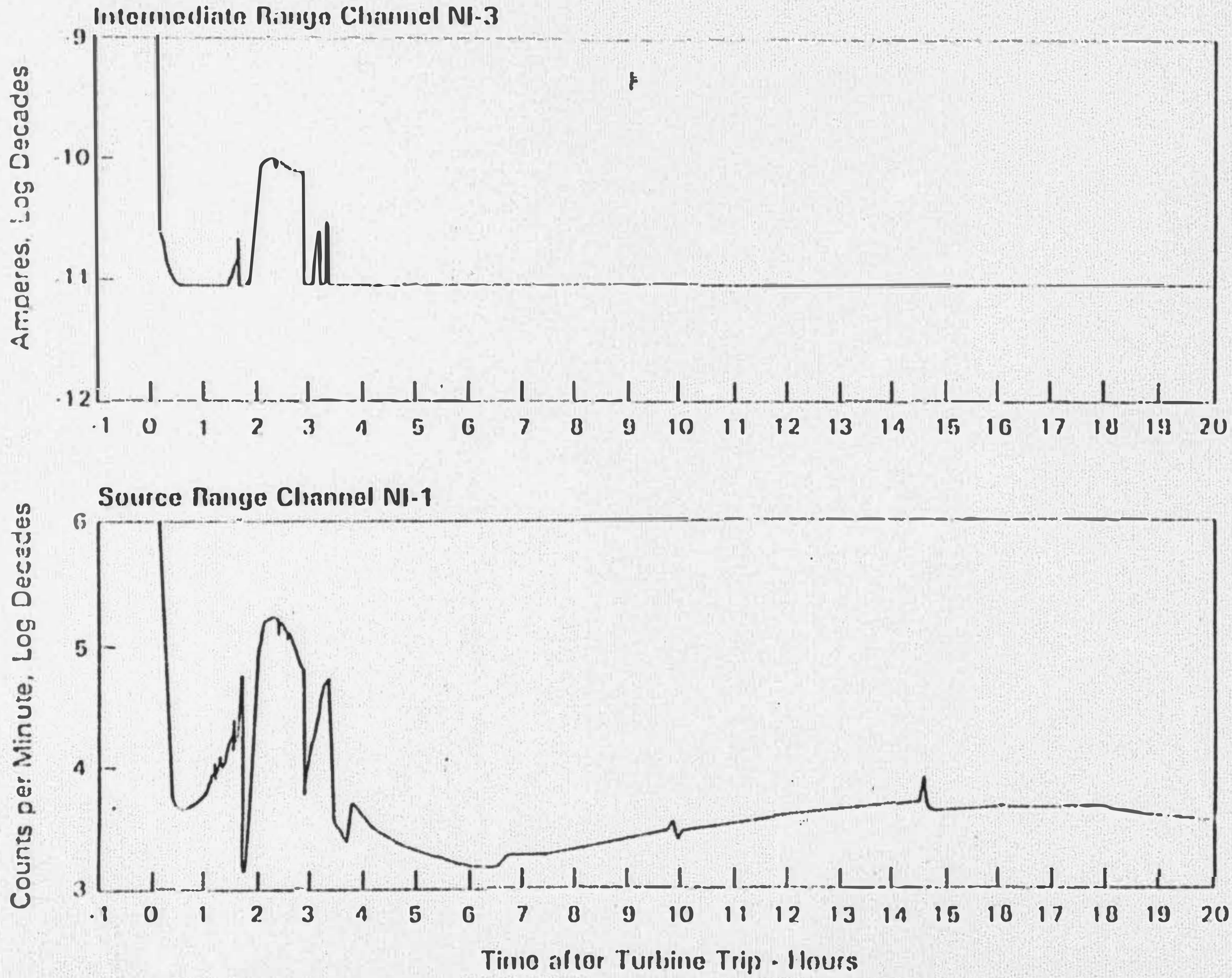


Figure 48
TMI-2 Loss of Coolant Accident 3/28/79
Computer Alarm Printer Lag Time

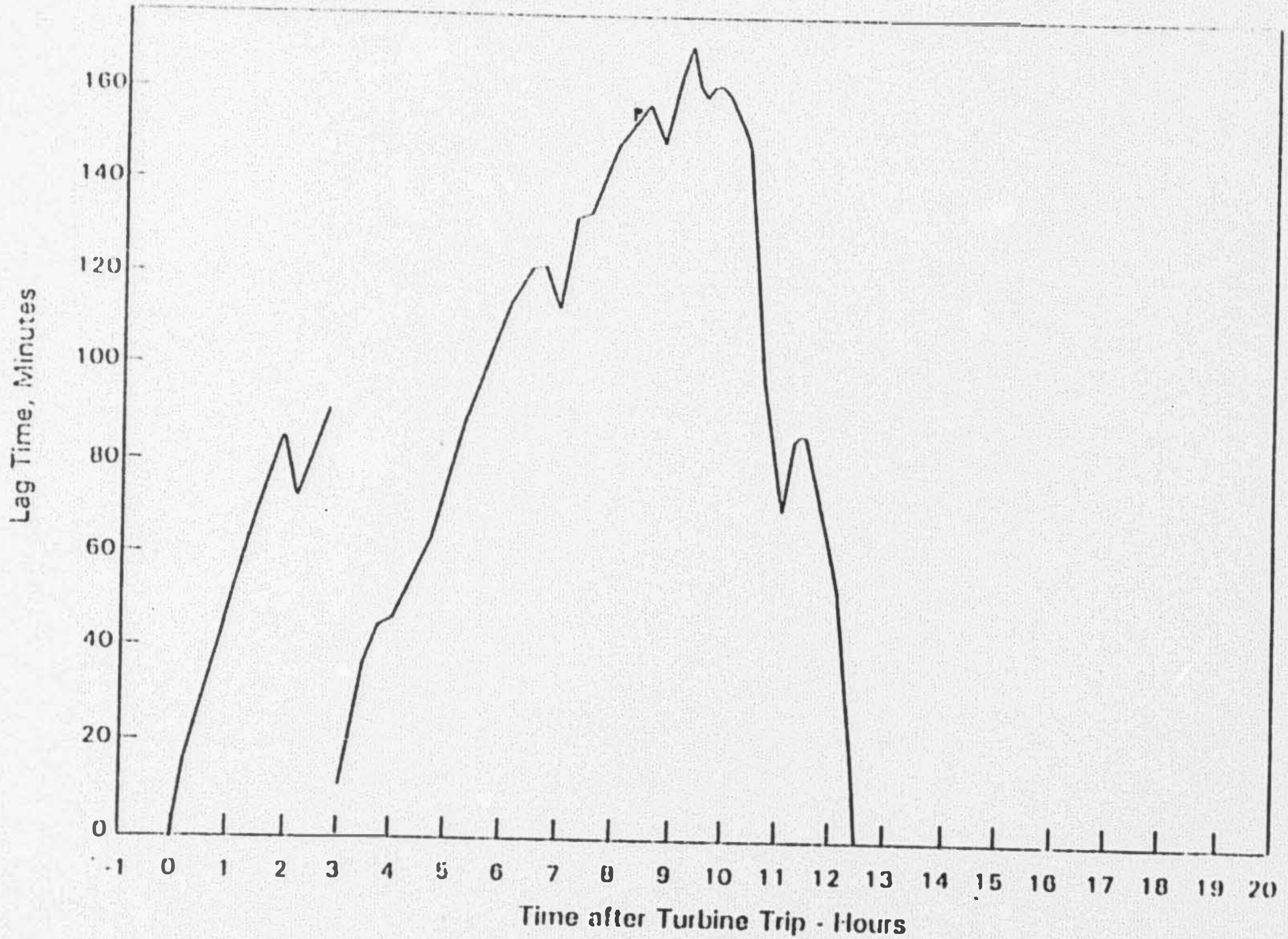


Figure 49
TMI-2 Loss of Coolant Accident 3/28/79
Emergency Feedwater Pump Discharge Pressures

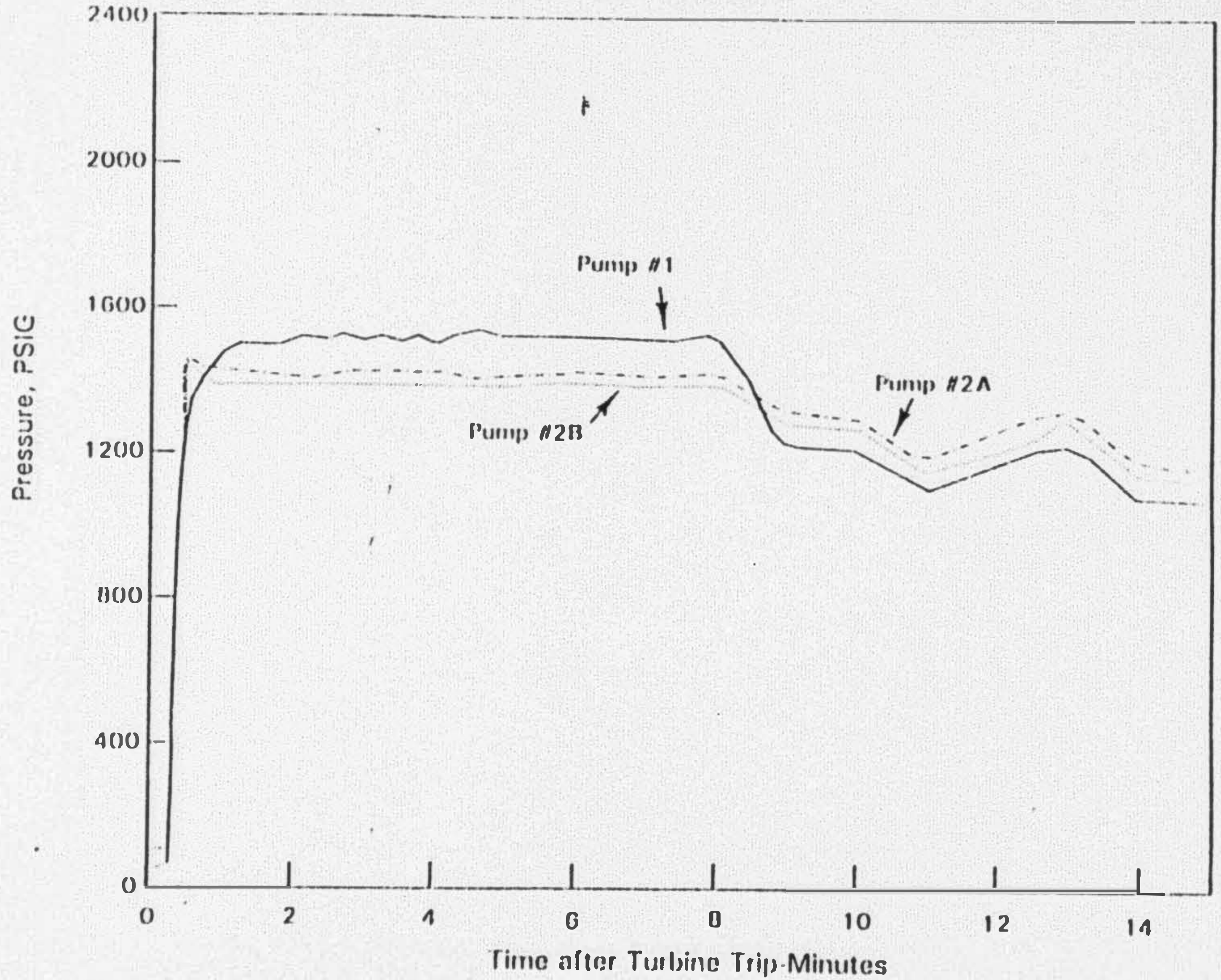


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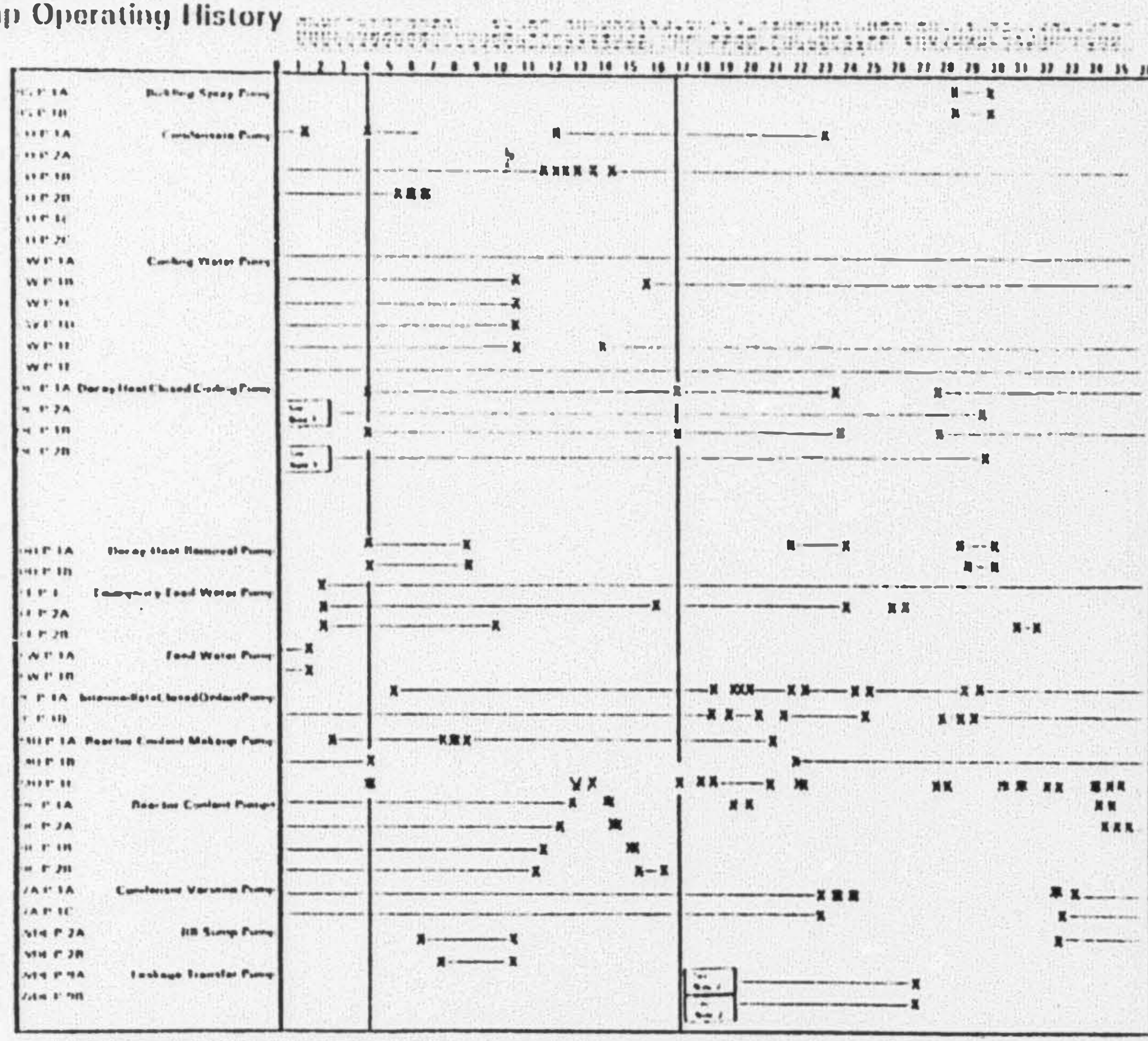
Figure 50
TMI-2 Loss of Coolant Accident 3/28/79
TMI-2 Pump Operating History



Note 1 Normally one pump (IC P 2A or 2B) is in continuous operation the computer print out shows information only that both were tripped at 0919

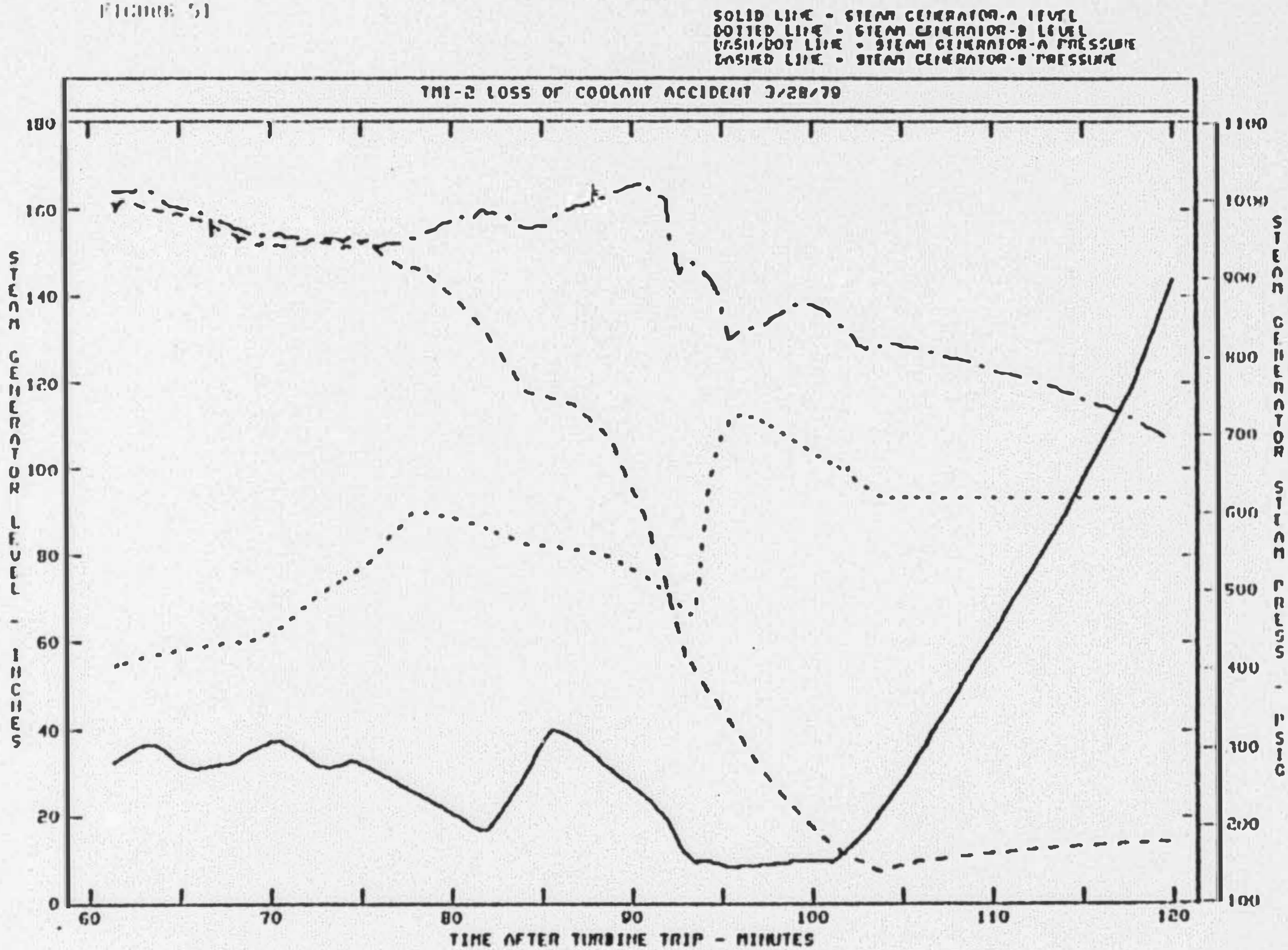
Note 2 Computer print out shows 1 trip for pumps WDL P 9A and 9B. In conversation with control room operator it was stated that these pumps were started and stopped 5 to 6 times between 0500 and 1145

X Would not start

X X Ran for short period of time (less than 15 minutes)

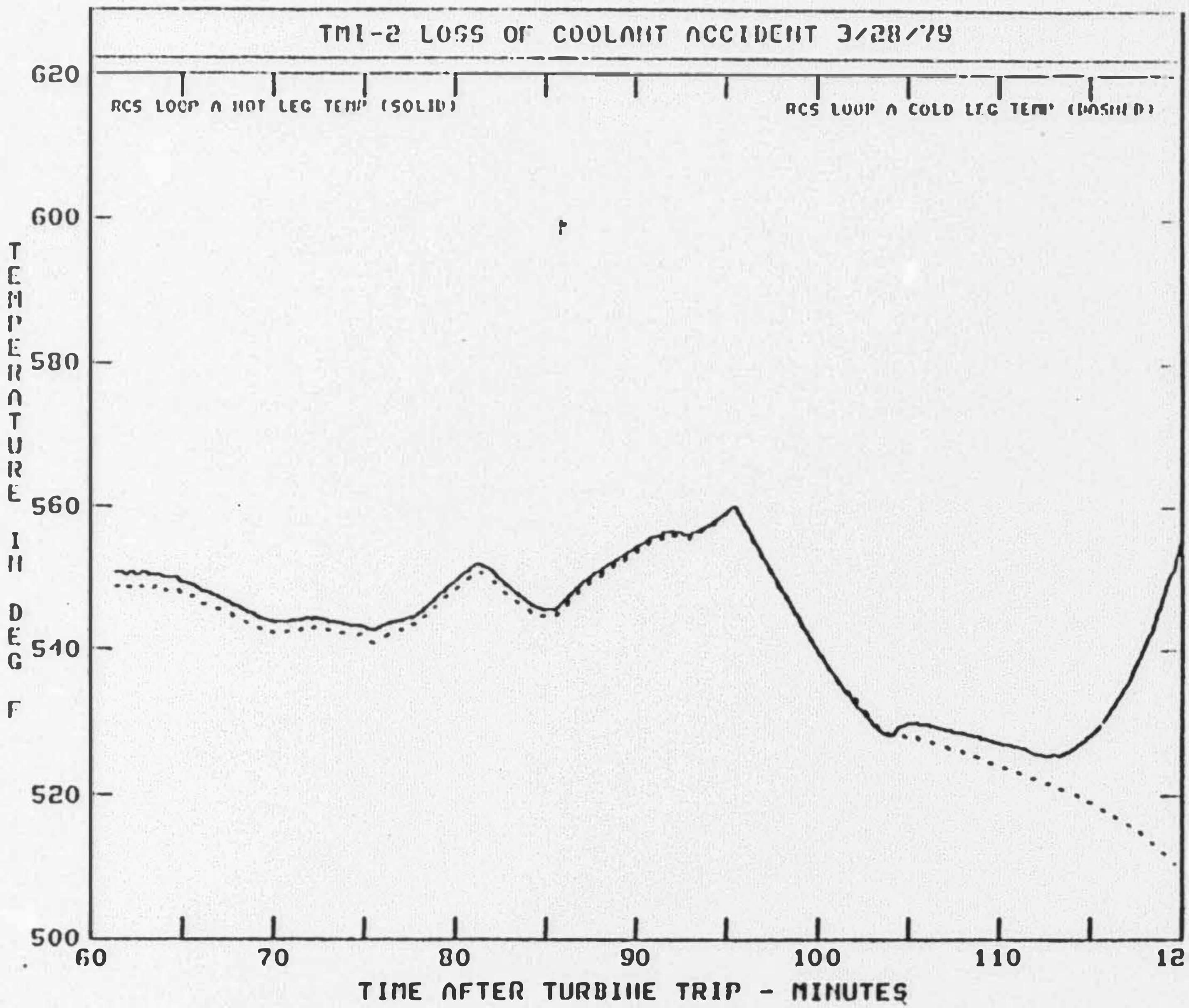
X——X
Start Run Stop

FIGURE 51



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 3/28/79
STEAM GENERATOR LEVEL - INCHES
0
20
40
60
80
100
120
140
160
180
STEAM GENERATOR STEAM PRESS - PSIG
100
200
300
400
500
600
700
800
900
1000
1100
60
70
80
90
100
110
120
TIME AFTER TURBINE TRIP - MINUTES

FIGURE 52



TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
RCS LOOP A HOT LEG TEMP (SOLID)
RCS LOOP A COLD LEG TEMP (DASHED)
620
600
580
560
540
520
500
TEMPERATURE IN DEG F
60
70
80
90
100
110
120
TIME AFTER TURBINE TRIP - MINUTES

FIGURE 51

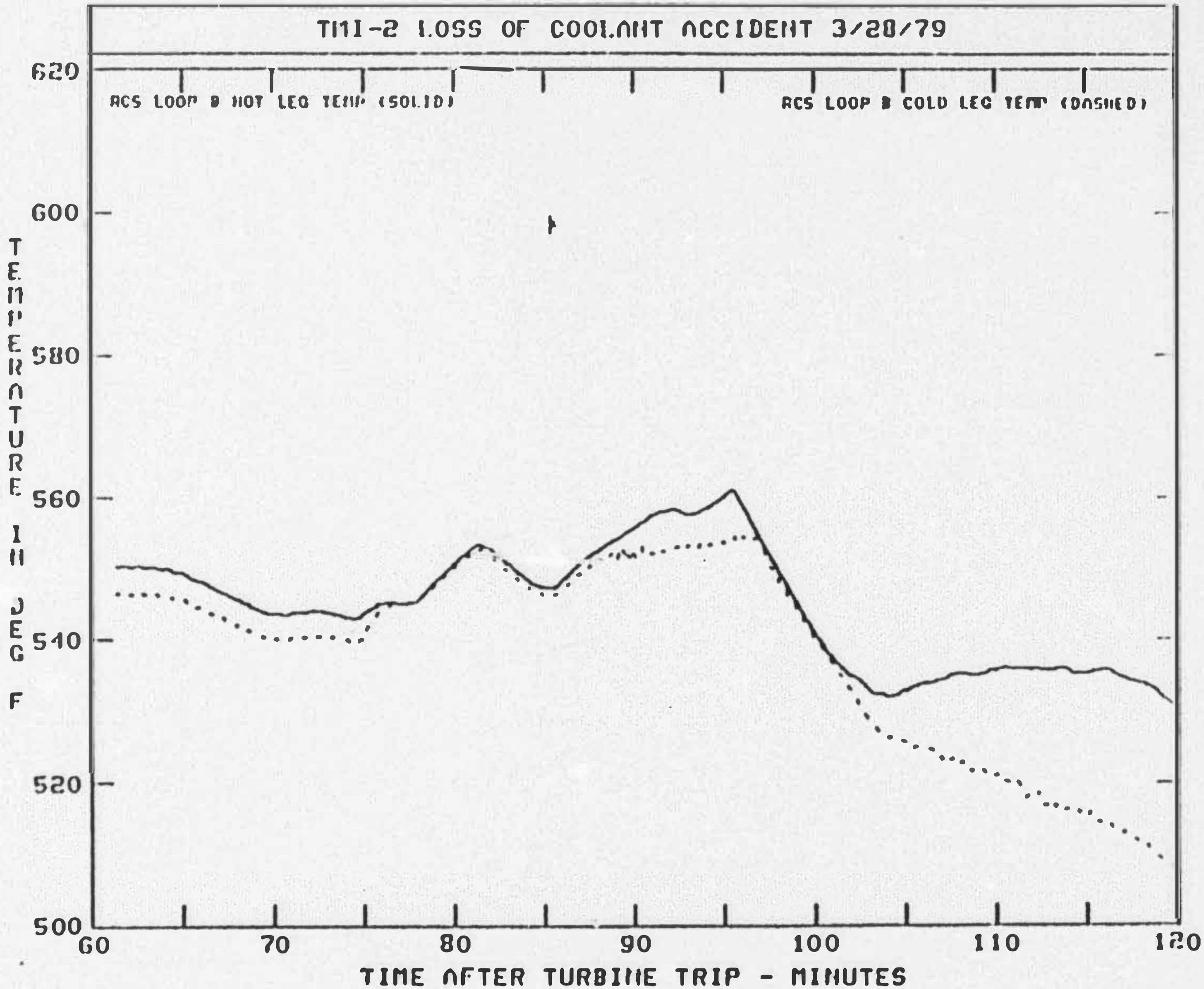

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
RCS LOOP B HOT LEG TEMP (SOLID)
RCS LOOP B COLD LEG TEMP (DASHED)
TEMPERATURE IN DEG F
620
600
580
560
540
520
500
60
70
80
90
100
110
120
TIME AFTER TURBINE TRIP - MINUTES

Figure 54
TMI-2 Loss of Coolant Accident 3/28/79
Control Room Panel Layout

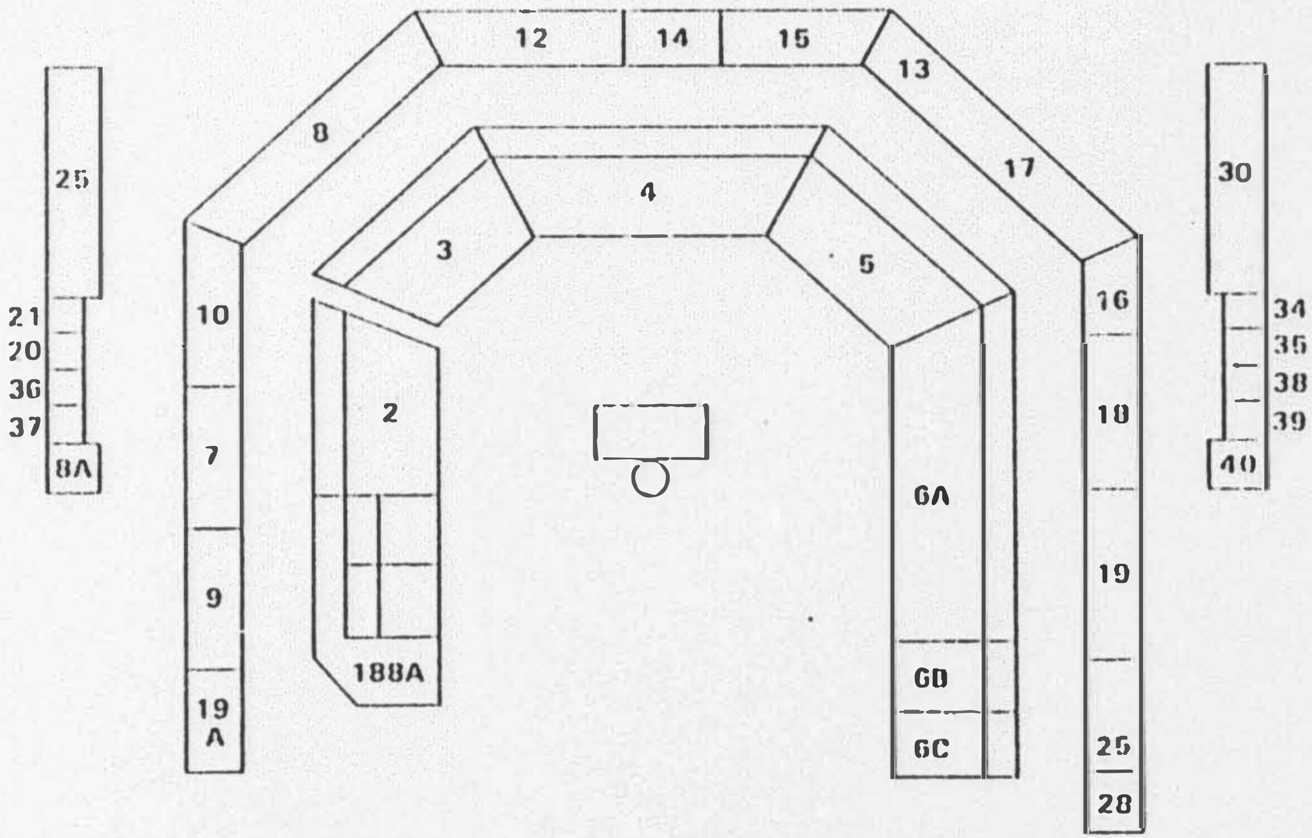


FIGURE 55

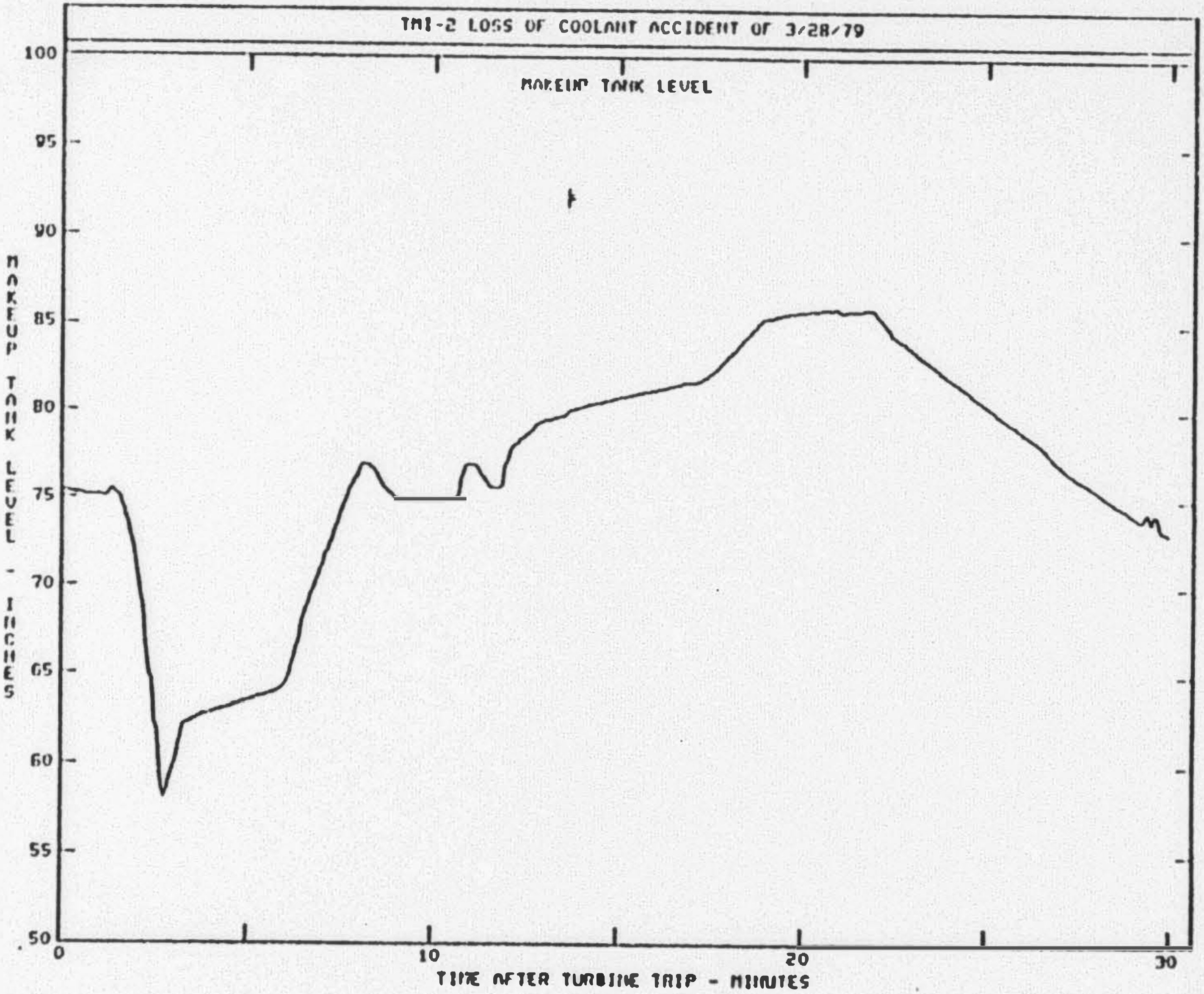


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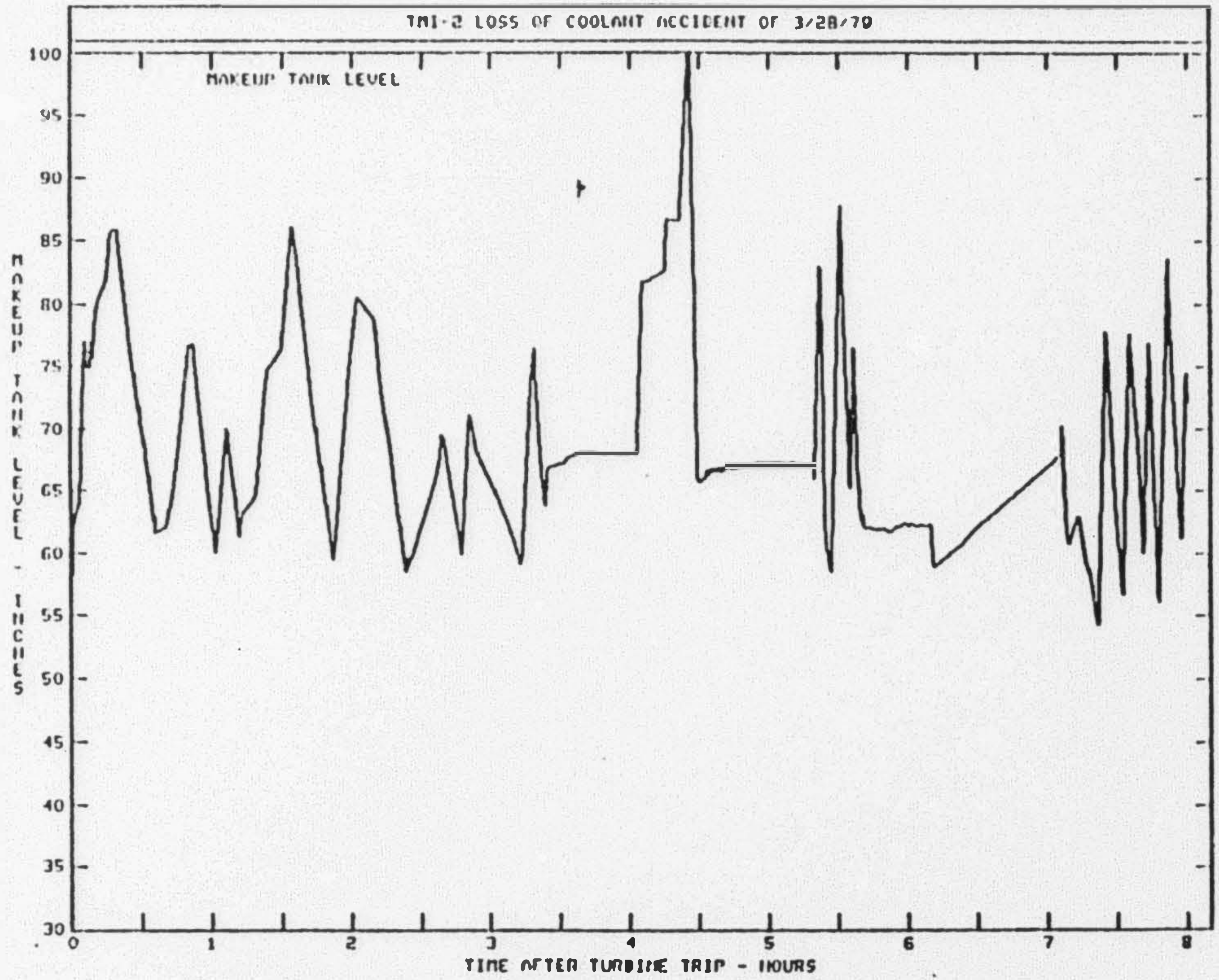
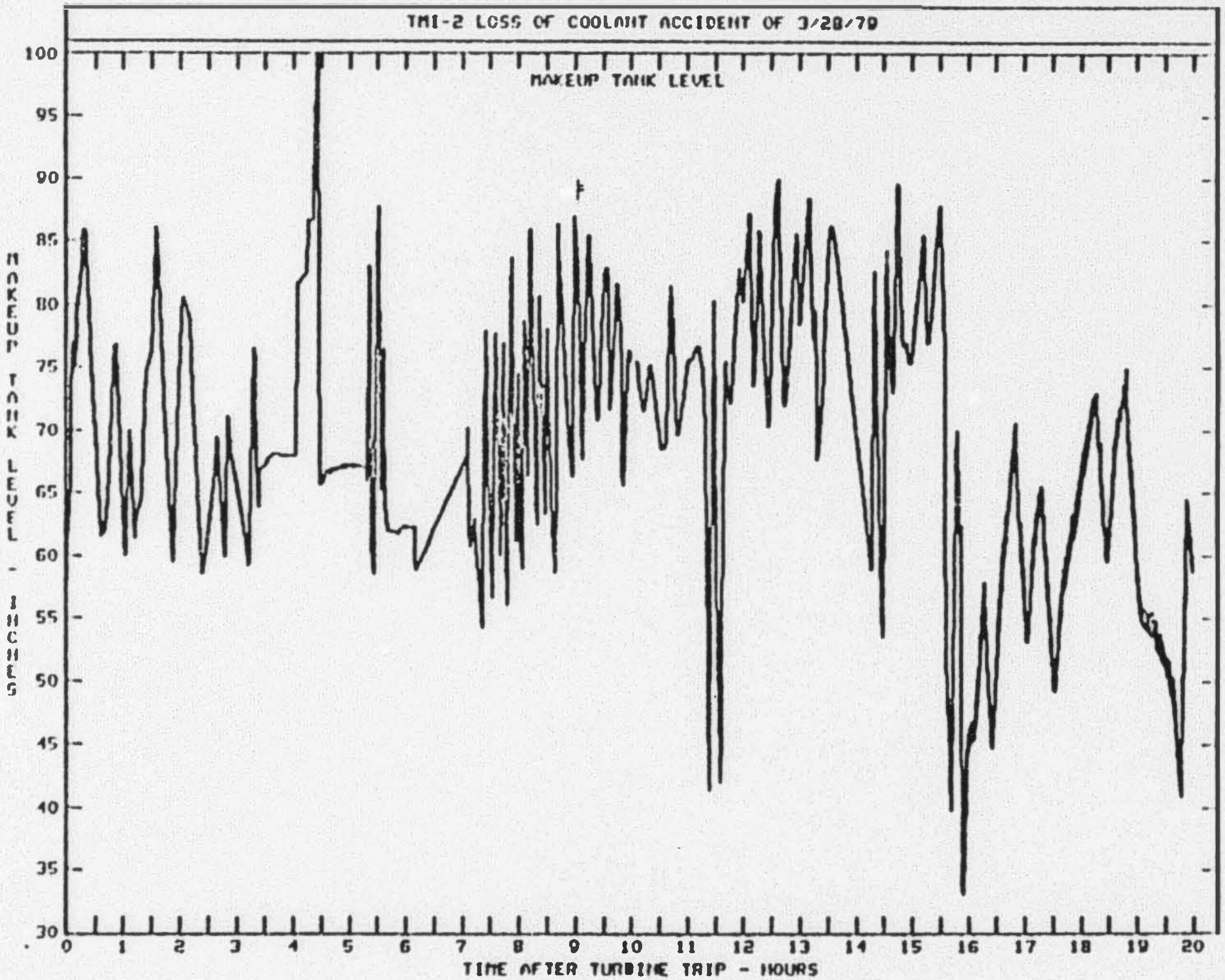
FIGURE 56



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 57



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II. RECOVERY ORGANIZATION

Included in this section is updated information concerning the Waste Management Activities (WMA) Organization.

For the period of June 1, 1979 through June 30, 1979 the following changes have been incorporated within the TMI Unit 2 Recovery Organization.

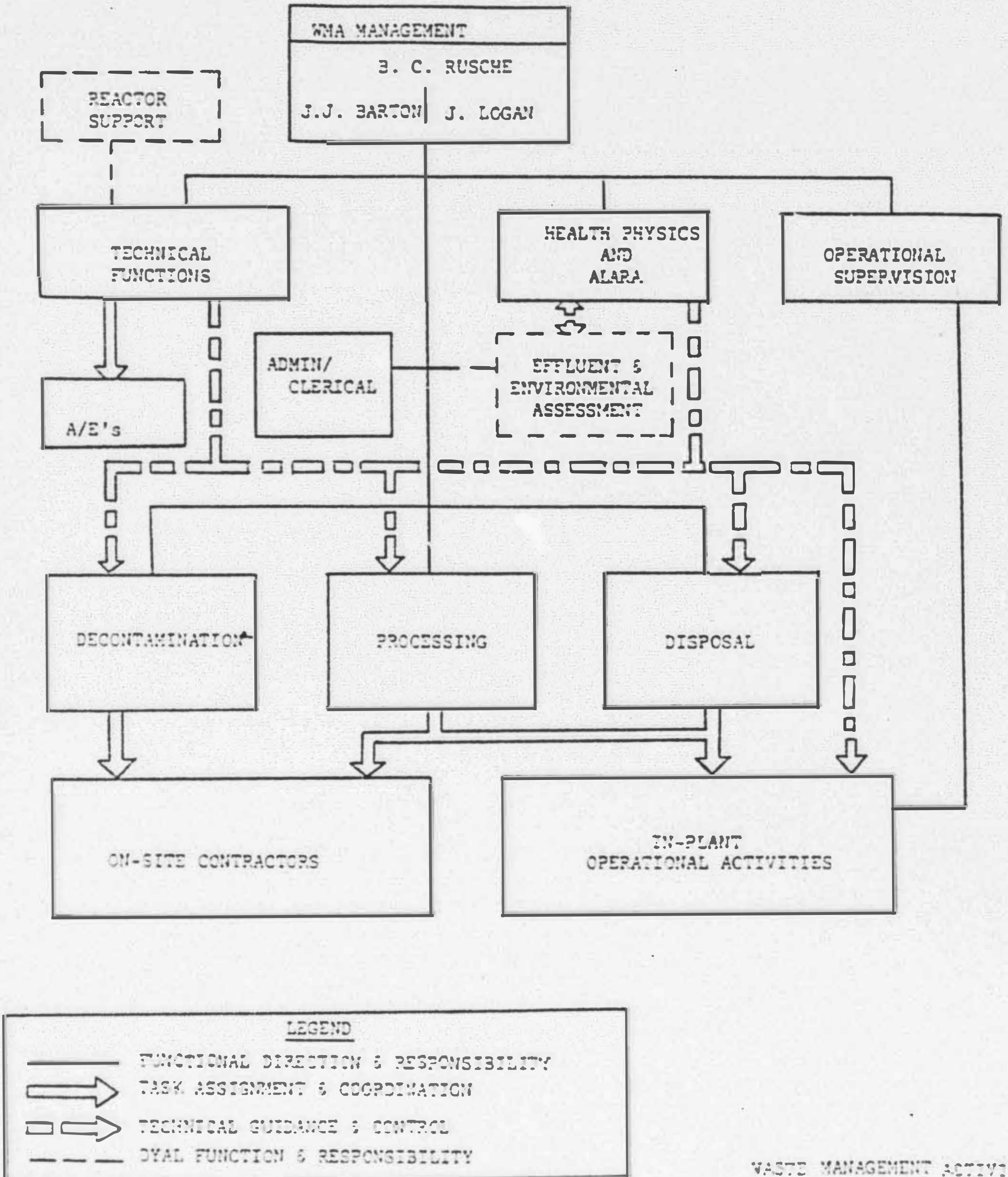
THREE MILE ISLAND RECOVERY ORGANIZATION - JUNE 1979

As a result of the increased emphasis on the radioactive waste accumulation on-site, the Waste Management Activities Group has been reorganized. Three distinct functional groups are now in effect. The Decontamination Group has immediate responsibility for cleanup of surface contamination in the Auxiliary and Fuel Handling Buildings. The Processing Group consolidates the existing liquid and gas processing teams and is responsible for processing waste to the point of an acceptable product for disposal. The Disposal Group is responsible for packaging, transportation, on-site staging, off-site shipping, and disposal of all forms of radioactive waste generated on-site.

The Technical Functions Group of Waste Management remains intact, with overall responsibility for technical planning and integration of the three new groups. This group is also responsible for the collection, analysis, and dissemination of all technical data pertaining to waste activities.

Exhibit 1 is the new Waste Management Activities organizational chart.

The remainder of the Recovery Organization remains as reported last month.



WASTE MANAGEMENT ACTIVITIES
ORGANIZATION CHART

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Rev. 2

III. PLANT MODIFICATIONS

Included in this section are updated and amended subsections from the June 18, 1979 Second Interim Report. Changes from the previous report are denoted by change bars in the right hand margin and Rev. 2 on the bottom right hand corner of the page. Subsections from the June 18, Second Interim Report which have not had any changes are not included in this report.

B. Auxiliary and Fuel Handling Building Supplementary Air Filtration Systems

1.0 System Function and Design Objectives

Radioactive iodine, released from the Reactor Coolant System during the TMI Unit 2 accident, was transferred into the Unit 2 Auxiliary and Fuel Handling Buildings. Immediate change out of the Auxiliary and Fuel Handling Building charcoal filter trains was not feasible because of the high radiation and contamination levels in the filter areas. As a consequence of the I-131 release rate, it was decided to construct a supplementary air filtration system to reduce off-site releases.

The function of the system is to filter radioactive particles and absorb iodine which has passed through the normal filtration system in the building ventilation system.

2.0 System Description

The system interfaces with the Auxiliary Building HVAC System, Fuel Handling Building HVAC System, and the Service Building HVAC System.

Discharge monitoring for the supplementary system is provided at each discharge point.

3.0 System Operation

A description of the system's operation is completed. Existing plant system's component functionality is being assessed. The impact of this program on the system's operation will be addressed and any changes in the system's operation will be included in a subsequent report.

4.0 System Status

Engineering Complete

Construction Complete

System description, flow diagrams, operating procedures, are complete. An operating and failure modes analysis has been prepared.

All four (4) trains are operable. The stack is capped. Present operation is with three (3) trains.

The operating procedure, which reflects the system operation description, is being reviewed by the NRC Site Staff.

D. Fuel Pool Waste Storage System

1.0 System Function and Design Objectives

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

2.0 System Description

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

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

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

3.0 System Operation

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

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

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

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

4.0 System Status

The system is essentially complete with the exception of the steam eductors. The eductors are undergoing design modifications to insure safe, reliable operation. Electrical submersible pumps are being fabricated for each stand pipe should a backup means of pumping water from the tanks be required.

E. Upgraded Decay Heat Removal System

1.0 System Function and Design Objectives

Future operation of the existing decay heat removal (DHR) system may result in radiation levels possibly ranging up to 500 Rads per hour in the vicinity of the system fluid components. This condition would severely limit personnel access for routine surveillance, operation, and maintenance. The upgraded DHR system consists of a program intended to identify, evaluate, and implement modifications necessary to ensure the integrity and reliability of the system in a radiation environment, substantially exceeding the original design basis, for up to one year of operation.

2.0 System Description

Proposed DHR system modifications include additional decay heat vault shielding, a remote TV monitoring system, modified DHR pump and motor bearing oilers, a vibration monitoring system, and associated operating and testing procedures.

Vault shielding will be provided by lead bricks assembled in a steel support frame. This will reduce the ambient personnel radiation exposure levels to "as low as reasonably achievable" (ALARA) in the accessible area above the vault. Radiation surveys will be made during initial DHR system operation and periodically thereafter to determine shield effectiveness.

The TV monitoring system will provide remote surveillance capability for DHR system operation and maintenance. Two independent systems are provided, one for each vault. Each system includes a radiation-tolerant, closed-circuit television with remote controls. Specific operations to be monitored include pump and motor bearing oil level, pump packing leak-off, remote oil fill, and pump venting.

DHR pump and motor bearing oiler modifications will provide for increased oil storage capacity, a means for remotely reading oil levels, and to permit feeding of oil to the bearings.

Provision for remote venting of the pumps is also provided.

Provisions will be made for monitoring pump vibration and loose parts in the system. This is intended to provide early indication of pump and motor degradation, loose parts in the system (particularly at the heat exchanger tube inlet), and changes in flow patterns due to partial line blockages.

Monitoring and control for these modifications will be provided from the fan room at elevation 322 in the service building.

3.0 System Operation

These modifications to the DHR will not appreciably alter system operation.

4.0 Status

The TV monitoring system, the bearing oil tanks and piping, and pump venting arrangement are installed and operational. The shielding bricks and support materials are on site. An operating test plan for the DHR system has been developed and is under review.

The installation of vault shielding is completed, except that the cover is not on since access is still necessary.

F. Steam Generator "B" Closed Loop Cooling System

1.0 System Function and Design Objectives

In order to provide a high pressure, closed cooling loop for water-solid steam generator "B", a system utilizing new equipment must be installed. The closed loop must remove the decay heat from the core plus the added heat load from one reactor coolant pump. To minimize the possibility for contamination of the closed loop, the system must be operated at a higher pressure than the reactor coolant system. The heat transferred to the closed loop will ultimately be rejected to the river. The system is intended to provide backup decay heat removal capability should the present steaming from steam generator "A" be discontinued.

2.0 System Description

The system consists of a new heat exchanger, pump, surge tank, piping and valves. The hot water leaving the steam generator will pass through the tube side of the new heat exchanger and return to the steam generator via the new pump. A pressurizer surge tank will maintain the steam generator secondary side pressure above the primary coolant system pressure.

The shell side of the heat exchanger is supplied with cooling water from the secondary services closed cooling water system which, in turn, will be cooled by water from the nuclear services river water pumps piped to the turbine building via the secondary services river water piping.

The new pump discharge piping is connected to the existing feedwater piping downstream of the main feedwater pumps, and the heat exchanger inlet piping is connected to the drain pot on the main steam line between the main steam isolation valve and main turbine stop valves.

3.0 System Operation

A detailed description of the system's operation is given in the operating procedure for Long Term OTSG "B" Cooling System.

A procedure has been completed to fill the "B" Steam Generator using the condensate pumps. An additional procedure to flush and vent the emergency water line has been completed as part of the fill procedure for the OTSG.

4.0 System Status

The system is installed and the preservice testing is completed.

The flushing and venting of the feedwater line is ready to be started upon approval of procedure.

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G. Portable Disposable Demineralizer

1.0 System Function and Design Objectives

Steam Generator "B" is presently contaminated with radioactive (fission) products. To minimize exposure to personnel and minimize the potential for contamination of the turbine building, this fluid must be cleaned up before the closed loop cooling system is placed into long term service. This cleanup capability will be provided by a portable disposable demineralizer (PDD) sub system. After the initial cleanup is completed, water quality can also be maintained by passing the closed loop cooling system flow through portable demineralizers.

2.0 System Description

The PDD sub system is located along the north wall of the turbine building basement. The system includes a disposable demineralizer approximately 18 inches in diameter, 30 inches in height, and having a 1.5 cubic foot resin capacity. The demineralizer will be connected to the steam generator "B" closed loop cooling system, and receives process water from the new closed loop pump discharge while returning the effluent to the pump suction. The number of demineralizer changes that will be required will depend on the water quality and activity.

The design pressure of the available demineralizers is 30 psig. Therefore, in order to protect the vessels, the PDD sub system also includes a pressure reduction valve, a pump, and safety relief facilities necessary to process the fluid while minimizing the potential for radioactive release to the environment.

The demineralizer is housed in a portable shielded cask. All operation, maintenance, and demineralizer removal and replacement will be performed in accordance with existing health physics requirements.

3.0 System Operation

A detailed description of the system's operation is given in the operating procedure for Long Term OSTG "B" Cooling System.

4.0 Status

The demineralizers and shield casks have been fabricated and installed.

The piping system and pump have been installed and is operational.

It is currently being used for wet layup of Long Term "B" Cooling System.

J. Steam Generator "A" Closed Loop Cooling System

1.0 System Function and Design Objectives

In order to provide a high pressure, closed cooling loop for water-solid steam generator "A", a cooling system utilizing new equipment has been proposed. The closed loop would remove the decay heat from the core plus the added heat load from one reactor coolant pump. To minimize the possibility for contamination of the closed loop, the system would be operated at a higher pressure than the reactor coolant system. The heat transferred to the closed loop would be rejected to the river. The system would be intended to provide primary decay heat removal capability redundant to the steam generator "B" closed loop cooling system.

2.0 Description

The system will consist of a new heat exchanger, pump, surge tank, and piping and valves. The hot water leaving the steam generator would be cooled in the shell side of the heat exchanger and returned to the steam generator by a new pump. A pressurized surge tank would maintain the steam generator secondary side at a minimum pressure greater than the primary coolant system pressure.

The tube side of the heat exchanger would be supplied with cooling water from the nuclear services river water pumps piped to the turbine building via installed secondary services river water piping.

The new pump discharge piping would be connected to the existing feedwater piping downstream of the main feedwater pumps. The heat exchanger inlet process piping would be connected to the main steam turbine bypass line between the isolation valve and the control valve at the condenser.

3.0 System Operation

A description of the system's operation is available. Should the system be constructed, the operations description will be provided in a subsequent report.

4.0 System Status

Design is completed. The pump and heat exchanger have been purchased and are on site. No piping, except the two tie-in pieces, have been fabricated. The procurement, fabrication, and installation have been placed on hold, and no further construction is anticipated.

L. Alternate Decay Heat Removal System

1.0 System Function and Design Objectives

The proposed Alternate Decay Heat Removal (ADHR) system augments the two existing DHR systems and the proposed water solid secondary/ natural circulation system as backup to steam generator "A" steaming. An integral Decay Heat Closed Cooling Water (DHCCW) system is included to transport heat from the ADHR cooler and the ADHR pump seal coolers to the nuclear services river water system. Connection points are also provided outside the fuel handling building to connect other dedicated liquid waste processing systems.

The specific function of the ADHR system is to remove decay heat such that the reactor coolant system can be brought to and maintained at a cold shutdown condition. With the exception of gross core flow restrictions, this system is intended to provide sufficient core flow to maintain reactor coolant subcooled.

2.0 System Description

The two ADHR pumps and a new heat exchanger will be mounted on a skid located outside the west wall of the fuel handling building. Three pipe runs will be installed from the existing DHR system piping within the fuel handling building and penetrate the fuel handling building west wall of a valve vault. The pipe runs will terminate in the valve vault by capping each line. Hook-up to the ADHR skid will be made later if needed. In addition, three capped taps will be provided on the ADHR piping installed outside the fuel handling building. These taps may be used later to connect other dedicated liquid waste processing systems.

Motor control centers and I&C panels for operation of all ADHR system pumps and motor operated valves will be mounted in a control trailer located near the ADHR skid.

The DHCCW system provides cooling water to the ADHR system heat exchanger and pump seal coolers. It utilizes a closed loop system to provide a double barrier between the ADHR system and the river water to prevent the direct release of radioactivity to the environment. A radiation detector is provided to monitor the level of radioactivity in the DHCCW system at the outlet of the DHR cooler. A radiation level indicator with high radiation level alarm is located in the ADHR system remote control room. If radioactivity is detected, operation of the decay heat removal loop and its associated DHCCW loop can be halted and the affected decay heat removal cooler isolated. The DHCCW system is mounted on a second skid and consists of the DHCCW pump, heat exchanger, and surge tank. Both skids will be located outdoors at grade level near the west wall of the fuel handling building and adjacent to each other.

3.0 System Operation

A detailed description of this system is not yet available; it is expected by July 30, 1979.

4.0 System Status

The piping for the ADHR system has been designed, fabricated, and received on site. The skid for the ADHR System with its components, two pumps, heat exchanger, valves and piping is near completion. Motor control centers are on site. The valve vault excavation is completed, shoring is in progress, and piping installation should start this week (July 9, 1979). The electrical trailer is very near completion.

Electrical power and service water connections will not be made until the system is put into service. Tie-in of the ADHR system to the existing plant DHR system is on hold until tie-in authorization.

Piping supports are being designed and fabricated on site. The control trailer wiring, air conditioning and insulation is finished. Piping construction is in progress in the Fuel Handling Building, and is 50% complete. Wall penetrations have been completed. Valve pit design, modified compatable with shoring and piping, should be done by July 13, 1979.

M. Standby Reactor Coolant Pressure Control System

1.0 Systems Function and Design Objectives

High radiation levels and flooding in the reactor building have or could potentially render much of the reactor coolant (RC) system electrical equipment and instrumentation inoperable. With much of the instrumentation inoperable, the RCS should be maintained water "solid". An alternate system of pressure control is required to ensure safe and reliable cooling of the reactor core, should control of the existing system become unmanageable. The standby reactor coolant pressure control (SRCPC) system will ensure reliable core cooling by performing the following function:

a. Maintain the RC system in a water-solid condition for natural circulation core cooling.

b. Maintain sufficient available NPSH should RC pump operation be required

c. Control the quality of the makeup fluid

d. Maintain pressure within control limits while accomodating thermal and volumetric contractions in RC system inventory.

2.0 System Description

The SRCPC system ties into the existing High Pressure Injection lines (see FSAR Figure 9.3-6). RC system pressure is maintained by three surge tanks arranged in series with a pressurized nitrogen blanket over the last tank. A fluid inventory of approximately two thirds of the total tank capacity is sufficient to maintain RC system pressure during sudden RC system inventory reduction transients. A level control valve at the tanks' discharge will prevent nitrogen from entering the RC system.

Long term makeup will be provided by the charging pump taking suction from an atmospheric storage tank. Makeup fluid conditions are adjusted by chemical addition and heating to meet RC system water quality requirements.

The RC system pressure will normally be maintained between 100 and 750 psig during the intended cooldown process. As of April 30, the RC system pressure must be maintained at 900 psig in order to provide letdown flow equal to th RC pump seal injection flow to the system so that the RC pumps can be operable.

The SRCPC makeup system will be operated manually from a local panel during initial operation and from the control room after system automation is complete. Makeup is provided in response to decreasing pressure in the RC system. An alarm will annunciate at the control station when the pressure differential between the RC and SRCPC makeup system reaches or exceeds 50 psi.

The SRCPC makeup system will prevent gross depressurization on the RC System when operating in a water-solid mode. Over-pressurization protection can be provided by increased letdown resulting directly from RC system pressure increase, letdown with concurrent termination of RC pump seal injection or makeup, opening the pressurizer vent valve, opening the pressurizer electromatic safety relief block valves, or lifting the pressurizer safety relief valves (the latter two methods are undesirable and will only be considered as a last resort).

3.0 System Operation

A preliminary description of this systems operation is now available.

TITLE: Preliminary System Description Task TS-6B Standby Reactor Coolant Pressure Control System, Revision 1, dated May 23, 1979.

4.0 Status

Phase I of the SRCPC makeup system is completed and has been hydrostatically tested and operated in recirculation mode. The Phase I will allow local manual operation of the system. The design work to ultimately convert the system to control room operation is being implimented.

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X. BOP Electrical Power System

1.0 System Function and Design Objectives

In the event of failure of normal off-site power sources to the BOP busses, the BOP Electrical Power (BOPEP) system provides an alternate source of power to serve existing components, which previously did not require loss-of-offsite power backup protection and new components that are planned to be used or may be used for decay heat removal from the primary system.

The BOPEP system is completely independent of the existing Class IE busses.

The BOPEP busses are loaded on a 'manual only' basis in accordance with emergency operating procedures.

Modifications of power supplies associated with Steam Generator "A" cooling systems have been given priority of installation with respect to those for the Steam Generator "B" cooling systems.

The testing requirements for the BOPEP systems are to be similar to those of the Class IE systems.

The BOPEP system shall supply power to the following components and associated auxiliaries at one time or another depending upon the specific situations:

a. Supplementary Air Compresser

b. Circulating Water Pumps

c. Condensate Pumps
   Steam Generator "A" Long Term Cooling Pumps*
   Steam Generator "B" Long Term Cooling Pump

d. New Decay Heat Removal Pump

e. Secondary System Closed Cooling Water Pumps

f. Alternate DHR System Pumps*
   Secondary Services River Water Strainer

g. Pressure and Volume Control System Charging Pumps

h. Chemical Cleaning Building Ventilation Equipment

i. Pressurizer Heaters

*Indicates components not currently planned to be put in service.

j. Alternate RHR System Closed Cooling Water Pump*
   Temporary Auxiliary and Fuel Handling Building HVAC

k. Fuel Handling Building HVAC Fans, Filters and Heaters

l. Auxiliary Building HVAC Fans, Filters and Heaters

m. Condenser Vacuum Pumps

n. Instrument and control power for above systems.

2.0 System Description

The BOPEP system includes two independent power block busses (2-3 and 2-4), each fed by a 2500 kw rated diesel generator, and two circulating water pump busses (2-5 and 2-6) fed by one 13.2 kv line. The loads associated with cooling steam generator "A" are connected to odd numbered busses. Correspondingly, loads associated with cooling steam generator "B" are connected to even numbered busses. The odd and even busses are powered by the gray and white diesel generators respectively and are, therefore, designated as the "gray" and "white" busses.

The diesel generators and associated auxiliary systems are located outdoors just south of the turbine building. Each diesel is a skid-mounted package complete with starting system, fuel injection equipment, and associated instrumentation and controls. The permanently installed fuel oil storage and supply system provides sufficient reserve for one day of rated load operation. In addition, there will be sufficient on-site fuel oil reserve to operate both diesel generators at rated load for the normal time required to obtain fuel resupply plus a four-day margin.

Suitable fire protection will be provided for the diesel generators and auxiliary systems. This may include a fire wall separating the two fuel oil tanks and diesels or a fire suppression system.

Existing circuit breakers, previously used for condensate booster pumps 3A and 3B, have been modified to connect the 2-3 (gray) and 2-4 (white) busses to their respective switchgear. Relays are provided at the busses to shed all loads on loss-of-offsite power. The existing bus transfer schemes that provide continuity of power supply by fast-transfer to the other transformer, have been left intact. To accommodate this, the new undervoltage detection schemes include a 10 second delay.

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* Indicates components not currently planned to be put in service.

The 13.2 kv line supplies power to the circulating water pumps and their associated auxilliaries. This line is powered by a 115 kv network which is backed by combustion turbines capable of being energized independently of the 230 kv network. The 13.2 kv line has sufficient capacity to start a second circulating water pump while one pump is still operating. However, only one pump is normally required.

Circulating water pump II is disconnected so their breakers can be used to connect the new power supply to busses 2-5 and 2-6 respectively. Buss 2-5 serves pumps 1A and 1C which are associated with steam generator "A" cooling systems. Correspondingly, buss 2-6 serves pumps 1B and 1D which are associated with steam generator "B" cooling systems.

3.0 System Operation

The BOPEP system normally provides standby power capabilities and is not operating. On loss-of-offsite power, the offsite power supply breakers will open and the diesel generators will be started and connected to their respective busses automatically.

Loading on the diesel generators, connection of the 13.2 kv line, and startup of the circulating water pumps will be performed manually from the control room in accordance with established procedures for the various potential plant conditions. For the "gray" and "white" busses, return to normal power is accomplished manually by first opening the diesel breaker and then closing the offsite supply breaker. For the 13.2 kv line, a return to normal power will be controlled manually by closing the normal supply breaker before opening the new supply breaker (hot transfer).

The primary control center for the BOPEP system is the control room control and monitoring capability exists locally for the diesel generators.

POOR ORIGINAL

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Initial startup testing will verify proper system and component operability, the adequacy of operating procedures, and ensure adequate performance capabilities of the BOPEP System. Periodic testing will be performed in accordance with procedural requirements and any additional testing and maintenance requirements by the component manufacturers. Periodic testing will verify proper breaker actuation, diesel starting and synchronizing, fuel oil quality, and breaker positions.

4.0 System Status

The work for the upgraded BOP electrical power system is approximately 90% complete.

The following work has not yet been completed:

The fire protection engineering and construction are approximately 50% complete.

Automatic lube oil system for "white" diesel is currently being designed.

G. Liquid Radioactive Waste Processing System Title "EPICOR II"

1.0 System Function and Design Criteria

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

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

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

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

2.0 System Description

Liquid Processing

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

Gas Processing

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

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3.0 <u>SYSTEM OPERATION</u>

The Auxiliary Building Emergency Liquid Cleanup System consists of a vendor supplied liquid radwaste process system which is located in the Chemical Cleaning Building. The system is designed to decontaminate by filtration and ion exchange between 150,000 and 200,000 gallons of radioactive waste water contained in the Auxiliary Building of TMI Unit 2. Contaminated water will be pumped from a connection located on the Miscellaneous Waste Holdup Tank (WDL-T-2) by a pump located in the Chemical Cleaning Building through the yard and into the process system. Yard piping will be enclosed within a guard pipe, the end of which terminates inside the Chemical Cleaning Building.

Decontaminated water will be delivered to the Clean Water Receiving Tank (CC-T-2) for sampling and analysis and pumped to the Liquid Waste Disposal System of TMI Unit I or Unit II for discharge if within specs, or transferred to the Off Spec Water Receiving Batch Tank (CC-T-1) for recycling through the process system. Capability also exists to discharge to a tank truck.

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

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

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

The system interface with the TMI Unit 2 Radwaste Disposal Miscellaneous Liquids System, the TMI Unit 1 Liquid Waste Disposal System, Demineralized Water System and the Service Air System.

4.0 <u>Status</u>

The system is essentially complete while undergoing an operability review and testing program. Operator training and qualification is proceeding.

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G. Staging Facilities for Dewatered Resins and Evaporator Bottoms

A. WG 21 - Interim Solid Waste Staging Facility

1.0 System Function and Design Criteria

Facilities are needed to stage dewatered radioactive resin generated by Epicor I and Epicor II until they can be shipped to a burial site. WG-21 will provide space for interim staging until WG-22, Solid Waste Staging Facility, is complete. Contact readings on the surface of the facility will be less than 5mr/hr.

2.0 System Description

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

3.0 System Operation

The facility has not been constructed as of this report.

4.0 System Status

Construction of the interim solid waste staging facility is nearing completion. Setting of the cell liners, backfilling, and asphalt installation is complete. Caulking of the cell seams has been accomplished, and curing of the caulking will permit application of the cell coating beginning July 2, 1979. Drip pans have been received. Concrete covers have been completed and are being held by the manufacturer until needed. It is anticipated the facility will be operational by July 6, 1979.

The Manitowac crane for handling Epicor liners is on-site and assembled. Metropolitan Edison utility operators are undergoing training on the operation of the crane.

B. WG-22-Solid Waste Staging Facility

1.0 System Function and Design Criteria

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

1.1 Dewatered radioactive resins from Epicor I.
1.2 Dewatered radioactive resins from Epicor II.
1.3 Dewatered radioactive resins or solidified evaporator bottoms from systems used to process water more radioactive than that processed by Epicor I or Epicor II.

B. WG-22 - Solid Waste Staging Facility (continued)

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

2.0 System Description

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

3.0 System Operation

The facility has not been constructed as of this report.

4.0 System Status

The design criteria for the solid waste staging facility has been approved by the waste management activity, Metropolitan Edison, and the NRC. Design is proceeding and should be completed by mid-July.

An ECM has been issued to construction for excavation. This will permit construction to start so the schedule of mid-October completion of the facility can be maintained.

All cell liners for the facility have been received. In addition, some of the drain line material has been received; the balance should be received during July. The concrete covers (28 of which will be used on the interim facility) are complete and are being held by the manufacturer until needed.

Design drawings for the transfer cask have been reviewed and approved by the technical support group. Fabrication of the cask has started. The promised delivery date for the cask is July 23, 1979.

A purchase order has been issued for the shield cask transport. The shield cask transport consists of two (2) concentric concrete sewer pipes (total concrete thickness of 15 inches) mounted on a lowboy. This transport will be used to transport liners from the Epicor 2 facility to the staging facilities. The promised delivery date for the shield cask transport is July 13, 1979.

5. Nuclear Sampling System

1.0 System Function and Design Objectives

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

2.0 System Description

Unit 2 Sample Lines which presently run into Unit 1 sampling area shall be rerouted to a new sample sink to be located in the Fuel Handling Building 305' elevation of Unit 2. In an adjacent room, the so-called "model room" a boronometer shall be installed.

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

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

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

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

3.0 System Operation

A detailed description of the systems operation is not yet available as design changes are still being made. This description shall be incorporated in a subsequent report.

4.0 System Status

The system design is essentially complete. Construction and material procurement is in progress.

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IV. Radiological Monitoring

This section includes discussion of information compiled for the period of March 28, 1979 through May 31, 1979. The assessment for the period March 28 through April 30 is described in detail in the report submitted June 18, 1979. The results of that assessment are merely summarized and, if necessary, updated or corrected in this report. Changes from the previous report are indicated by change bars in the right hand margin and Rev. 2 on the bottom right hand corner of the page.

EXECUTIVE SUMMARY
RADIOLOGICAL MONITORING

EXECUTIVE SUMMARY
RADIOLOGICAL MONITORING
MARCH 28 - May 31, 1979

The results of an assessment of radiation doses to the public due to releases from March 28 through May 31, 1979 from the Three Mile Island Unit 2 accident, based on verified release data and verified radiological environmental data are summarized in the attached table. Virtually all of this dose resulted from releases in the period March 28, 1979 through April 30, 1979. Contributions from May releases were negligible. Doses from radioactivity released in liquid effluents were extremely low. No individual received more than a fraction of a millirem, and the population dose was much less than one person-rem. Doses from airborne effluents are due to noble gas isotopes which deliver whole body doses and iodine isotopes which deliver thyroid doses. Measurements indicate that the maximum individual whole body dose from noble gases was 75 millirem, and the noble gas whole body dose to the 50 mile population was calculated to be about 3300 person-rem. (The figure of 75 millirem for the maximum individual dose differs slightly from the previously reported 83 millirem due to a refinement in the estimate of natural background radiation contribution to the measured dose.) Doses from iodine isotopes in airborne effluents result from inhalation of iodine in air and ingestion of iodine in milk. Calculations and measurements show that no individual received more than about 10 millirem to the thyroid from inhalation, and the calculations indicate the thyroid inhalation dose to the 50 mile population was about 160 person-rem. (The estimate of the maximum individual exposure based on measured air concentrations of Iodine-131 has been revised from 2.7 to 3.7 millirem.) Measurements indicate that the maximum individual thyroid dose from ingestion of Iodine-131 in cow milk was 1.5 millirem. (This dose was incorrectly reported as 2.3 millirem in the previous report.) Calculations show that the population dose from Iodine-131 in milk produced within 50 miles was about 900 person-rem. Average doses to individuals in the population from any isotope in any pathway were very low, less than 2 millirem. The maximum doses to any individual is 75 millirem which is comparable to the difference in natural background radiation dose between Harrisburg, PA and Denver, CO over the period of one year.

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TABLE

Summary of Radiation Doses Due to TMI Unit 2 Accident

March 28 - May 31, 1979

Release Mode	Pathway	Estimated Integrated Dose[a] Maximum Individual (mrem)	Organ	Population Dose (person-rem)	Organ
Liquid	Drinking Water (^{131}I)[b]	< .04	Thyroid	< 1	Thyroid
	Fish Ingestion (^{131}I)[c]	.01	Thyroid	<< 1	Thyroid
	Swimming, Boating & Shoreline[b]	0	Thyroid	0	Thyroid
Airborne	Noble gases in plume	75	Whole Body	(3300)	Whole Body
	Iodine Inhalation	(6.3) 3.7	Thyroid	(160)	Thyroid
	Iodine uptake through cow milk ingestion	1.5	Thyroid	(900)	Thyroid

a. Doses are based on TLD measurements or measured isotopic concentrations in environmental samples except for those in parentheses. Doses in parentheses are based on release data and transport models.

b. No Iodine 131 activity detected in more than 95% of river water samples. Concentration in those samples are assumed to be minimum detectable. (366 water samples 3/28-4/30, 420 samples 5/1 - 5/31)

c. No Iodine 131 activity detected in any fish samples. Concentration assumed to be minimum detectable

Rev. 2

IV. A. OFFSITE LIQUID RELEASES AND DOSES

1. Releases

The releases of radionuclides in liquid effluents to the Susquehanna River have been within expected values as a result of the refueling outage of Unit 1. From March 28, 1979 to April 30, 1979, 10.7 curies of tritium and about 0.3 curies of various activation and corrosion products have been released to the river from both units. (Table IV-A-1). From May 1, 1979 through May 31, 1979, 4.7 curies of tritium and about 3.7 E-2 curies of various activation and corrosion products have been released to the river from both units (Table IV-A-1).

The only radionuclide released in significant concentrations and quantities as a result of the accident on March 28, 1979 has been Iodine-131. The total quantity of Iodine-131 released through April 30 is approximately 0.24 curies. Detailed data and discussion regarding the release of Iodine-131 during the period March 28 through April 30 were included in the June 18, 1979 submittal. The total quantity of ^{131}I released during the period May 1, 1979 through May 31, 1979 is approximately 5.05 E-3 curies. The source of most of the Iodine-131 released in May is the Industrial Waste Treatment System, the major source of earlier releases. This system is discussed in the June 18 submittal. May releases are a factor of about 47 lower than releases from March 28, 1979 through April 30, 1979 (Table IV-A-1). Detailed Iodine-131 data from each liquid release in May, 1979 are included in Table IV-A-2. Daily Iodine-131 release quantities are plotted in Figure IV-A-1. Data in this figure for the period March 28 through April 30 were presented in the June 18 report, but are plotted here in a different format. Although the release of Iodine-131 in liquid effluents did exceed normal levels, the levels did not exceed either the Technical Specification release rate limits, or concentration limits in 10 CFR 20.106, averaged over one day.

2. Environmental Measurements

The Radiological Environmental Monitoring Program conducted by Metropolitan Edison Company includes analyses of river surface water, downstream drinking water from treatment plants and aquatic biota. Except for seven samples collected on March 31, April 1 and 2, and May 12, 13, 24 and 25 at station 7G1, the Columbia Water Plant intake, which showed very low levels of Iodine-131 (0.4, 0.72, 0.66, 1.3, 0.57, 0.56, and 1.7 pCi/l) and one sample on April 27 at location 7G2, the Wrightsville Water Treatment Plant which showed 0.49 pCi/l of Iodine-131 no gamma emitting isotopes other than low levels of naturally occurring isotopes were detected. The concentrations of Iodine-131 listed above are only slightly greater than the minimum detectable concentrations. Data from the monitoring program are included in Attachment 1 to the June 18 submittal and are supplemented in Attachment I of this submittal.

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3. Estimated Offsite Exposures

Radiation doses estimated from the measurements described above are extremely low, a few hundredths of a millirem for a person drinking water, eating fish from the river, or using the river for swimming, boating, or shore line activities. Detailed radiation doses are shown in Appendix C of Attachment 1 to this report.

TABLE IV-A-1

SUMMARY OF RADIONUCLIDES
RELEASED TO THE SUSQUEHANNA RIVER

Radionuclide	3/28/79 - 4/30/79 Activity (Ci)	5/1/79 - 5/31/79 Activity (Ci)
^{3}H	10.670	4.7
^{51}Cr	3.5E -4	1.64E -3
^{54}Mn	4.11E -4	1.57E -4
^{58}Co	0.022	1.24E -2
^{60}Co	6.9E -3	1.41E -3
^{95}Nb	1.82E -4	5.17E -4
^{95}Zr	4.83E -5	7.22E -5
^{110m}Ag	1.25E -3	9.37E -4
^{131}I*	0.235	5.05E -3
^{131m}Xe	--	7.25E -4
^{132}I	3.44E -4	--
^{133}I	1.4E -4	1.43E -5
^{133}Xe	0.012	7.5E -5
^{134}Cs	2.11E -3	2.18E -3
^{136}Cs	2.7E -4	1.3E -3
^{137}Cs	5.61E -3	4.83E -3
^{140}Ba	5.99E -4	5.43E -3
^{140}La	1.29E -3	4.09E -3

* ^{131}I is the only radionuclide of significance released to the river from the Unit 2 accident of 3/28/79. Other isotopes result from routine releases from Unit 1.

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TABLE IV-A-2

LIQUID EFFLUENT RELEASES

[illegible]

(All data refers to I-131)

Start	Stop	Tank	(µCi/cc) [1]Concentration at Station Discharge (dilution calc.)	(µCi/cc) [2]Concentration at Station Discharge (grab samples)	µCi Discharged	Cumulative µCi Discharged
3/28 0400	3/28 0900	IWTS	1.6×10^{-7}	4×10^{-5} at 1100 hrs.	6469 [5]	6,469
3/28 0320	3/28 0655	WECST-B	None		None [3]	6,469
3/29 0015	3/29 1215	WECST-B	None		None	6,469
3/29 1315	3/29 1420	IWTS	7.5×10^{-10}		7.5	6476.5
3/29 1610	3/29 1815	IWTS	7.5×10^{-10}	5.4×10^{-10}* at 1700 hrs.	17.0	6493.5
3/30 0020	3/30 0753	SEC. NEUT.	None		None [4]	6493.5
3/30 1200	3/30 1630	IWTS	1.2×10^{-9}		135.8 [6]	6629.3
3/30 0300	3/30 1600	IWTS	6.7×10^{-10}		153	6782.3
3/30 2020	3/30 2253	SEC. NEUT.	None		None	6782.3
3/31 0140	3/31 0430	IWTS	2.5×10^{-8}		241.3	7023.6
3/30 1600	3/30 2400	IWTS	3.8×10^{-8}		6,981	14004.6
3/31 0001	3/31 2400	IWTS	2.7×10^{-7}		92,112	106116.6
3/31 0240	3/31 0710	WECST-A	3.9×10^{-9}		140	106256.6
3/31 2230	4/1 1035	SEC. NEUT.	None		None	106256.6
4/1 0630	4/1 15[illegible]	WECST-B	None		None	106256.6

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TABLE IV-A-2

LIQUID EFFLUENT RELEASES (Cont'd)

3/28/79 to 4/30/79

(All data refers to I-131)

Start	Stop	Tank	[1] Concentration (μCi/cc) at Station Discharge (dilution calc.)	[2] Concentration (μCi/cc) at Station Discharge (grab samples)	μCi Discharged	Cumulative μCi Discharged
4/1 0001	4/1 2400	INTS	1.49×10^{-7}	6.2×10^{-8}* 2020 hrs.	54673	160934.6
4/1 0130	4/1 0534	INTS	2.5×10^{-8}		346.3	161280.9
4/1 1521	4/1 1915	INTS	2.5×10^{-8}		396	161676.9
4/2 0001	4/2 1850	INTS	1.08×10^{-7}	1.5×10^{-8}* 1130 hrs.	27854	189530.9
4/2 1650	4/2 1850	SEC. NEUT.	None		None	189530.9
4/2 0515	4/2 1110	INTS	2.5×10^{-8}		394.6	189925.5
4/3 1025	4/3 1915	SEC. NEUT.	None	7.3×10^{-10}* 1540 hrs.	None	189925.5
4/5 1813	4/6 0750	WECST-A	1×10^{-8}	2×10^{-10}* 1815 hrs. 1.0×10^{-10} 4/5 at 1305 hrs.*	1010	190935.5
4/5 1800	4/6 0003	SEC. NEUT.	None		None	190935.5
4/6 0310	4/6 0400	INTS	1.9×10^{-7}		480.1	191415.6
4/6 0615	4/7 2230	INTS	7.9×10^{-8}	3.9×10^{-9}* 4/6 at 1640 hrs.	16668	208083.6
4/6 1930	4/7 0430	INTS	5.1×10^{-8}		930.9	209014.5
4/7 0235	4/7 1430	SEC. NEUT.	None	$<6\times10^{-8}$ 6.3×10^{-9}* 4/7 at 1045 hrs.	None	209014.5

TABLE IV-A-2

LIQUID EFFLUENT RELEASES (cont'd)

[illegible]

(All data refers to I-131)

Start	Stop	Tank	(uCi/cc) [1]Concentration at Station Discharge (dilution calc.)	(uCi/cc) [2]Concentration at Station Discharge (grab samples)	uCi Discharged	Cumulative uCi Discharged
[illegible]	4/8 [illegible]	WECST-B		<4x10^{-8} <2 x 10^{-10}" 4/8 at 1100 hrs.	1160	210194.5
[illegible]	4/9 0945	SEC. NEUT.	None	<4x10^{-8}	None	210194.5
[illegible]	4/9 0945	WECST-B		<4x10^{-8} <2 x 10^{-10}" 4/9 at 1910 hrs.	443	210637.5
[illegible]	4/10 1410	IWFS	5.7 x 10^{-9}	<6.5x10^{-8} 4.2 x 10^{-9}" 4/10 at 1710 hrs.	80.6	210718.1
[illegible]	4/10 1900	WECST-A		<5.5x10^{-8}	129.0	210847.1
[illegible]	4/11 1600	SEC. NEUT.	None	<1.1x10^{-8} 1.4 x 10^{-9}" 4/11 at 1620 hrs.	None	210847.1
[illegible]	4/12 [illegible]	WECST-B		<3.9x10^{-9} 1.5 x 10^{-9}" 4/12 at 2000 hrs.	200.0	211047.1
[illegible]	4/13 [illegible]	WECST-A	None	<3.6x10^{-9}	None	211047.1
[illegible]	4/13 1150	SEC. NEUT.	None	<3x10^{-9}	None	211047.1
[illegible]	4/13 0449	WECST-B		<5.1x10^{-8}	60.0	[illegible]
[illegible]	4/13 [illegible]	IWFS	1.1 x 10^{-8}	<2.3x10^{-8}	6,208	217315.1
4/13 [illegible]	4/14 0117	IWFS	[illegible] x 10^{-8}	<1.6x10^{-8} 1.2 x 10^{-9}" 4/13 at 2230 hrs.	1,956	219271.1
[illegible]	[illegible]	WECST-A	None	<5.1x10^{-8}	None	219271.1
[illegible]	[illegible]	IWFS	[illegible] x 10^{-8}	<2.2x10^{-8} 1.9 x 10^{-9}" [illegible]	1325	[illegible]
[illegible]	[illegible]	WECST-A		<5.6x10^{-8} [illegible] [illegible]	[illegible]	[illegible]
[illegible]	[illegible]	IWFS	[illegible] x 10^{-8}	<3.0x10^{-8} <2 x 10 [illegible]	102.6	[illegible]

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TABLE IV-A-2

LIQUID EFFLUENT RELEASES (Cont'd)

[illegible] 79 to [illegible] 79
(All data refers to [illegible])

Start	Stop	Tank	(µCi/cc) Concentration at Station Discharge (dilution calc.)	(µCi/cc) Concentration at Station Discharge (grab samples)	µCi Discharged	Cumulative µCi Discharged
4/17 [illegible]	4/17 1610	INTS	9.5 x 10^{-9}	7.5 x 10^{-8} 7.8 x 10^{-9}* 4/17 at 1810	21.5	222310.2
4/17 [illegible]	4/18 1043	WECST-A		6.5 x 10^{-8}	2112	224422.2
4/18 1522	4/18 1945	WECST-B	4.6 x 10^{-9}	7.78 x 10^{-8} 4.8 x 10^{-9}*	23	22444[illegible].2
4/18 1000	4/18 2400	INTS	6.1 x 10^{-9}	5.1 x 10^{-8} 4.5 x 10^{-9}* 4/18 at 1930	4.5	224449.7
4/19 2400	4/23 0430	INTS	3.5 x 10^{-8}	3.5 x 10^{-8}	2910.88	227360.5[illegible]
4/19 0155	4/19 2005	WECST-B		3.7 x 10^{-8} [illegible] x 10^{-9}* Ci/cc 4/19 at 2150 hrs.	315	227675.53
4/20 1535	4/20 2125	INTS	3.1 x 10^{-8}	3.1 x 10^{-8} 3.75 x 10^{-9}* Ci/cc 4/20 at 2010 hrs.	77	227752.58
4/20 1530	4/21 0255	WECST-A		1.39 x 10^{-8} 1.65 x 10^{-9}* Ci/cc 4/21 at 1535 hrs.	862	22861[illegible].58
4/23 2300	4/24 0230	WECST-A		6.9 x 10^{-10}	1520	230134.58
4/23 0510	4/23 1636	SEC. NEUT. TANKS	None	3.1 x 10^{-8} 4.2 x 10^{-9}*	None	230134.5[illegible]
4/23 1910	4/24 0310	INTS	3.1 x 10^{-8}	3.1 x 10^{-8} 2 x 10^{-10}*	108	230242.58
4/24 0950	4/25 0630	WECST-B		1.7 x 10^{-8} 3 x 10^{-10}*	1200	231442.58
4/24 1043	4/25 0753	SEC. NEUT. TANKS	None	2 x 10^{-8}	None	231442.58
4/25 1741	4/26 1300	WECST-A		[illegible] x 10^{-8} 9.5 x 10^{-10}	43.7	2314[illegible].28
4/26 [illegible]	4/27 1113	INTS	2.8 x 10^{-8}	2.5 x 10^{-8}	261.67	2317[illegible].95
4/26 [illegible]	4/27 1813	WECST-B		1.[illegible] x 10^{-8} 1.5 x 10^{-9}*	1480	233227.95
4/27 [illegible]	4/28 2015	INTS	3.[illegible] x 10^{-7}	3.1 x 10^{-8} [illegible]	[illegible]	233235.55

TABLE IV-A-2

[illegible]

(3/[illegible]/79 to 4/30/79)

(All data refers to I-131)

Start	Stop	Tank	(µCi/cc) [1] Concentration at Station Discharge (dilution calc.)	(µCi/cc) [2] Concentration at Station Discharge (grab samples)	µCi Discharged	Cumulative Ci Discharged
[illegible]	[illegible]	Sec Neut Tanks	None	3.1×10^{-8} 1.1×10^{-9}*	None	[illegible]
[illegible]	[illegible]	WECST-A		3.1×10^{-8}	592	[illegible]
4/28 [illegible]	4/29 0820	Sec Neut Tanks	None	3.1×10^{-8} 4.0×10^{-10}*	None	[illegible]
[illegible]	4/30 [illegible]	Sec Neut Tanks	None	3.1×10^{-8}	None	[illegible]
[illegible]	4/30 [illegible]	IWTS	1.9×10^{-9}	3.1×10^{-8} 1.7×10^{-9}*	688.6	234516.65

NOTES

1 Calculated based on average tank sample and known dilution factor (df) data during the period of time that the tank was being released. Discharges for IWTS are averaged over a 24 hour period.

2 Calculated by averaging the station discharge (RML-7) grab samples taken during the time the tank was being released. If the number appearing in this column is a "less than" (<) number, all the numbers averaged were less than MDA numbers and the MDA's were used for the purpose of averaging. This calculation is conservative in that it overestimates the actual I-131 concentration at the station discharge. (See Attachment 7 for RML-7 grab samples)

3 WECST Tank releases are controlled by procedure HP 1621 which limits the release concentration to 0.1 MPC. The HP 1621 permit takes the specific activity of all the isotopes in the tank, assumes a dilution factor from MDCT flow and calculates a release rate so that 0.1 MPC is not exceeded while discharging. (See Attachments 4, 5 and 6)

TABLE IV-A-2

NOTES (continued)

4 The source of water into the Secondary Neutralizing Tank is from the regeneration of the Illinois Water Treatment System demineralizers. The Illinois Water Treatment System produces demin water by taking pretreated river water from upstream of the station discharge and sending it through demineralizers. The water from the regeneration of these demineralizers goes to the Secondary Neutralizing Tank. All isotopic samples on this tank after 4/2/79 showed no detectable I-131 or I-133.

Since the input to this tank is essentially river water from upstream of the station discharge, it is reasonable to assume that from 3/28 at 0400 to 4/2 at 1850, no I-131 was released from this tank.

5 See Attachment 2 for IWTS calculations/assumptions

6 See Attachment 3 for IWTS calculations/assumptions

POOR ORIGINAL

TABLE IV-A-2

LIQUID EFFLUENT RELEASES

4/30/79 to 5/31/79

(All data refers to I-131)

Start	Stop	Tank	(Ci/cc) Concentration at Station Discharge (Dilution Calc.)	(Ci/cc) Concentration at Station Discharge (Grab Samples)	Ci Discharged	Cumulative Ci Discharged
4/30 1810	5/1 1123	WECST-A	$3.[illegible] \times 10^{-9}$	3.1×10^{-8}	1960	236476.65
5/1 1500	5/2 0420	WECST-B	1.07×10^{-9}	3.1×10^{-8} 2.2×10^{-9}* @2345 hrs	196	236672.65
5/1 1801	5/2 0620	U1 Sec Neut	None	3.1×10^{-8} 2.2×10^{-9}* @2345 hrs	None	236672.65
5/1 2119	5/2 0300	IWTS	$< 3.1 \times 10^{-8}$	3.1×10^{-8} 2.2×10^{-9}* @2345 hrs	None	236672.65
5/2 1700	5/2 2318	WECST-A	2.7×10^{-9}	2.7×10^{-9}* @1850 hrs	182	236854.65
5/3 2155	5/4 1430	U-1 Sec. Neut Tank	None	3.1×10^{-8} 4.0×10^{-10}* @1900 hrs.	None	236859.35
5/4 0800	5/4 1320	IWTS	$< 3.1 \times 10^{-8}$	3.1×10^{-8} 3.0×10^{-10}* @1420 hrs	-.29	236863.83
5/4 0900	5/5 0415	WECST-B	4×10^{-9}	4.3×10^{-8} 1.4×10^{-9}* @1925	194	237057.83
5/5 0945	5/6 0005	WECST-A	3.2×10^{-9}	3.1×10^{-8}	182	237239.83
5/6 0013	5/6 0950	U-1 Sec. Neut	None	3.1×10^{-8} 4.0×10^{-10}* @1345 hrs	None	237239.83
5/6 1315	5/7 1130	WECST-A	2.15×10^{-9}	2.45×10^{-8} 4.0×10^{-10}* @1345 hrs	422	237661.83
5/7 0323	5/7 1040	U-1 Sec. Neut	None	2.45×10^{-8} 3.0×10^{-10} @1525 hrs.	None	237661.83
5/7 1000	5/8 1200	IWTS	$< 3.3 \times 10^{-11}$	2.45×10^{-8} 3.0×10^{-10} @1525 hrs	.0177	237661.83
5/8 1422	5/8 2305	WECST-A	3.8×10^{-9}	2.45×10^{-8} 3.0×10^{-10} @1525 hrs	269	237930.83
5/9 [illegible]	5/9 1140	WECST-B	[illegible]	2.35×10^{-8} $[illegible] \times 10^{-10}$* [illegible]	131	238061.83

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TABLE IV-A-2

LIQUID EFFLUENT RELEASES

(4-30-79 to 5-31-79)

(All data refers to I-131)

Start	Stop	Tank	(Ci/cc) Concentration at Station Discharge (Dilution Calc.)	(Ci/cc) Concentration at Station Discharge (Grab Samples) **	Ci Discharged	Cumulative Ci Discharged
5/9 1135	5/9 1840	U-1 Sec. Neut	None	$< 2.35 \times 10^{-8}$ 6.7×10^{-10}* @1835 hrs	None	[illegible]1.63
5/9 1225	5/11 1010	IWTS	1.56×10^{-9}	$< 2.63 \times 10^{-8}$ 6.7×10^{-10}* @1835 hrs	205	238266.83
5/10 0735	5/10 1230	IWTS	$< 2.83 \times 10^{-8}$	2.72×10^{-8} 5.0×10^{-10}* @1825 hrs	2.8	238269.63
5/10 1030	5/10 2215	U-1 Sec. Neut.	None	2.72×10^{-8} 5.0×10^{-10}* @1825 hrs	None	238269.63
5/10 1925	5/11 0750	WECST-A	3.7×10^{-9}	2.61×10^{-8} 5.0×10^{-10} @1825 hrs	383	238652.63
5/11 1605	5/11 2110	U-1 Sec Neut	None	$< 3.65 \times 10^{-8}$ 7.2×10^{-9}* @2130	None	238652.63
5/11 1715	5/13 1000	U-1 Sec Neut	None	$< 3.59 \times 10^{-8}$ [illegible]	None	238652.63
5/14 09-[illegible]	5/14 23-3	IWTS	3×10^{-8}	$< 3.1 \times 10^{-8}$ 3.0×10^{-9}	9.16	238661.81
5/14 1925	5/14 2235	WECST-B	4.2×10^{-9}	$< 3.1 \times 10^{-8}$	120	238781.81
5/15 1105	5/15 1535	IWTS	7.3×10^{-10}	$< 2.92 \times 10^{-8}$ [illegible] } **	10.4	238792.21
5/15 1510	5/16 0355	U-1 Sec Neut	None	$< 2.59 \times 10^{-8}$ $< 3 \times 10^{-10}$ } **	None	238792.21
5/16 0425	5/18 2130	IWTS	1.09×10^{-10}	$< 3.64 \times 10^{-8}$ $< 3 \times 10^{-10}$ } **	79.81	238872.02
5/18 0030	5/18 0725	U-1 Sec Neut	None	$< 1.69 \times 10^{-8}$ } **	None	238872.02
5/18 0645	5/19 1740	IWTS	1.4×10^{-8}	1.44×10^{-8} } **	2.89	238874.91
5/19 [illegible]	5/19 04-5	U-1 Sec Neut	None	$< 1.2 \times 10^{-8}$ $< 3 \times 10^{-10}$ } **	None	238874.91
5/19 [illegible]	5/19 1510	WECST-[illegible]	$1.[illegible] \times 10^{-8}$	$< 1.2 \times 10^{-8}$ } **	39.6	238914.51

**Manual 24 Hr. compositing was started 5/15/79 for Station discharge I-131.

TABLE IV-A-2

LIQUID EFFLUENT RELEASES

([illegible] to 5/31/79)

(All data refers to I-131)

Start	Stop	Tank	(μCi/cc) Concentration at Station Discharge (Dilution Calc.)	(μCi/cc) Concentration at Station Discharge Composite Sample	μCi Discharged	Cumulative μCi Discharged
5/19 2110	5/20 0334	WECST-B	3.1×10^{-9}	2.1×10^{-9}* $<2.02 \times 10^{-8}$	62	238986.51
5/21 0205	5/21 1335	WECST-A	1.10×10^{-9}	$<1.4 \times 10^{-8}$ $<4 \times 10^{-10}$*	105	239091.51
5/21 0835	5/21 1220	IWFS	5.19×10^{-11}	$<1.4 \times 10^{-8}$	1.18	239098.69
5/21 0845	5/25 0135	IWTS	3×10^{-10}	$<1.5 \times 10^{-8}$	7.05	239105.74
5/22 2145	5/23 0730	U-1 Sec Neut	None	$<1.5 \times 10^{-8}$ $<6 \times 10^{-10}$*	None	239105.74
5/24 1755	5/25 0040	WECST-B	3.24×10^{-9}	$<1.5 \times 10^{-8}$ 4.4×10^{-10}* **	131	239236.74
5/25 0330	5/25 1610	IWFS	1.12×10^{-9}	$<1.5 \times 10^{-8}$	0.2	239236.94
5/25 2115	5/26 0300	WECST-B	1.17×10^{-9}	$<1.34 \times 10^{-8}$ 1.9×10^{-9}*	110	239346.94
5/26 1200	5/27 0316	U-1 Sec Neut	None	$<1.39 \times 10^{-8}$ $<4 \times 10^{-10}$*	None	239346.94
5/27 1700	5/28 0004	WECST-A	1.34×10^{-9}	$<1.34 \times 10^{-8}$ $<5 \times 10^{-10}$*	91.6	239438.54
5/28 0020	5/31 0730	IWTS	1.16×10^{-8}	$<2.13 \times 10^{-8}$	19	239457.54
5/28 0045	5/28 1615	WECST-B	3.4×10^{-10}	$<2.13 \times 10^{-8}$	24	239481.54
5/29 0246	5/29 1120	WECST-B	7.20×10^{-10}	$<2.13 \times 10^{-8}$	8.5	239490.0
5/29 0735	5/29 1700	IWFS	4.18×10^{-11}	$<2.13 \times 10^{-8}$	3.16	239493.2

** Automatic composite unit obtaining a 24 Hr. sample was started 5/24/79 for Station Discharge 1051.

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TABLE IV-A-2

LIQUID EFFLUENT RELEASES

(5/30/79 to 5/31/79)

(All data refers to I-131)

Start	Stop	Tank	(μCi/cc) [1]Concentration at Station Discharge (Dilution Calc.)	(μCi/cc) [2]Concentration at Station Discharge Composite Sample	Ci Discharged	Cumulative Ci Discharged
5/30 1000	5/31 0008	U-1 Sec Neut	None	$<3.00 \times 10^{-8}$	None	239,493.2
5/30 1000	5/30 1900	WECST-B	1.06×10^{-10}	$<3.00 \times 10^{-8}$ [illegible] $\times 10^{-9}$*	8.09	239,501
5/31 0645	5/31 2030	IWTS	4.2×10^{-11}	$<3.19 \times 10^{-8}$ [illegible] $\times 10^{-9}$*	4.03	239,505.32

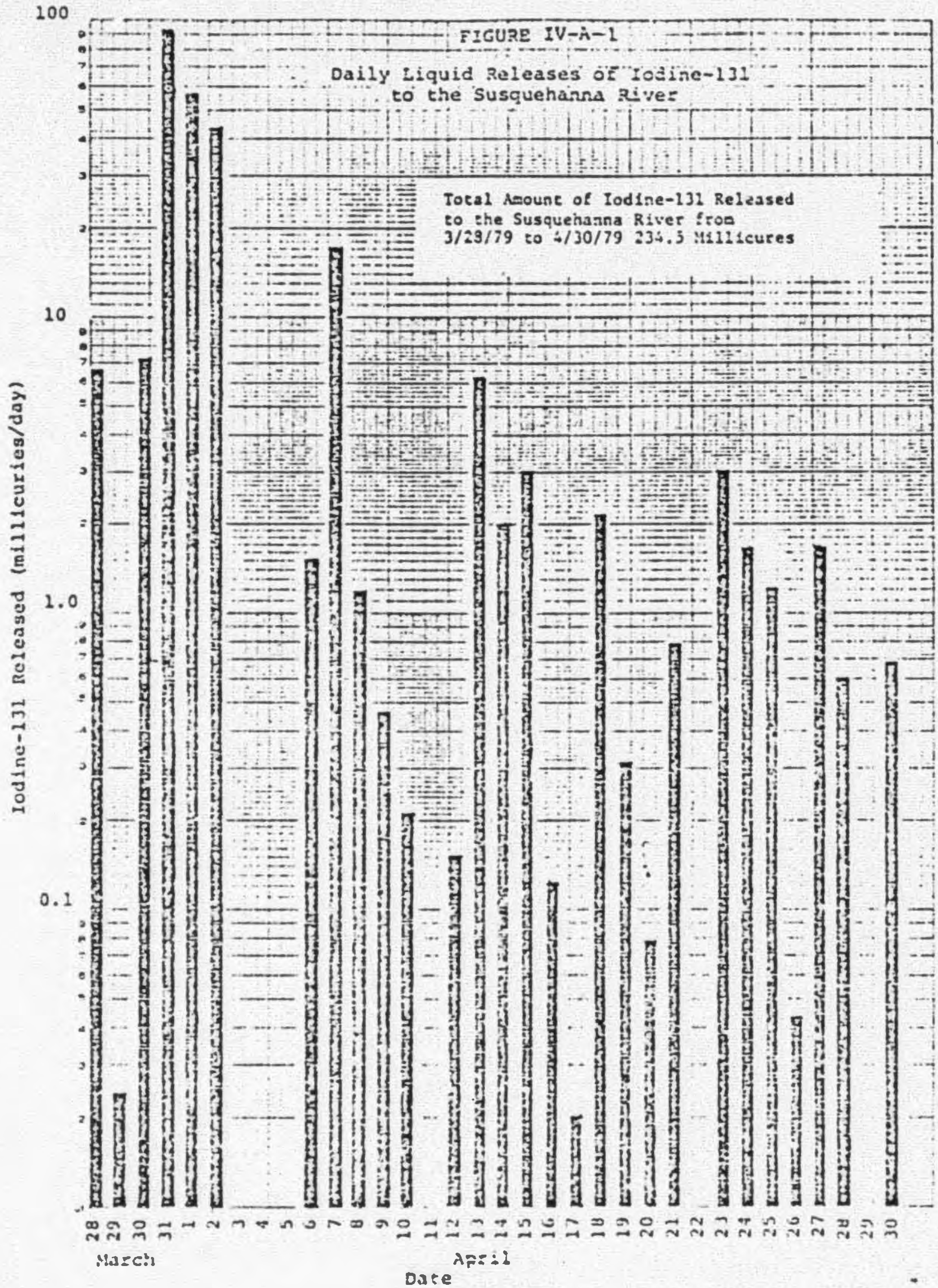
NOTES

1. Calculated value based on average tank sample concentration and known dilution factor (df) data during the period of time the tank was being discharged.

2. Calculated by averaging the station discharge (RML-7) grab samples taken during the period of time that the tank was being discharged. Starting on 5/15/79 samples were being manually composited into a 24-hour sample. Starting on 5/24/79 an automatic liquid compositor was put into operation for the purpose of obtaining a more representive sample. If the number appearing in this column is a "less than" (<) number, the MDA's were taken as real number for the purpose of averaging. This calculation is conservative in that it over-estimates the actual ^{131}I concentration at the station discharge.

3. WECST releases are controlled by procedure HP1621 and RML-6 which limits the release concentration to 0.1 $(MPC)_w$. The HP1621 permit takes into consideration the concentration (specific activity) for all the radionuclides in the tank, assumes a df from the MDCT flow. Calculations are made to restrict the release rate so that 0.1 $(MPC)_w$ is not exceeded during discharge.

4. The source of water into the Secondary Neutralizing (Sec. Neut.) Tank is from the regeneration of the Illinois Water Treatment System demineralizers. The Illinois Water Treatment System produces demin water by taking pretreated river water from upstream of the station discharge and sending it through demineralizers. All isotopic analysis of samples on this tank showed no detectable ^{131}I and ^{133}I.



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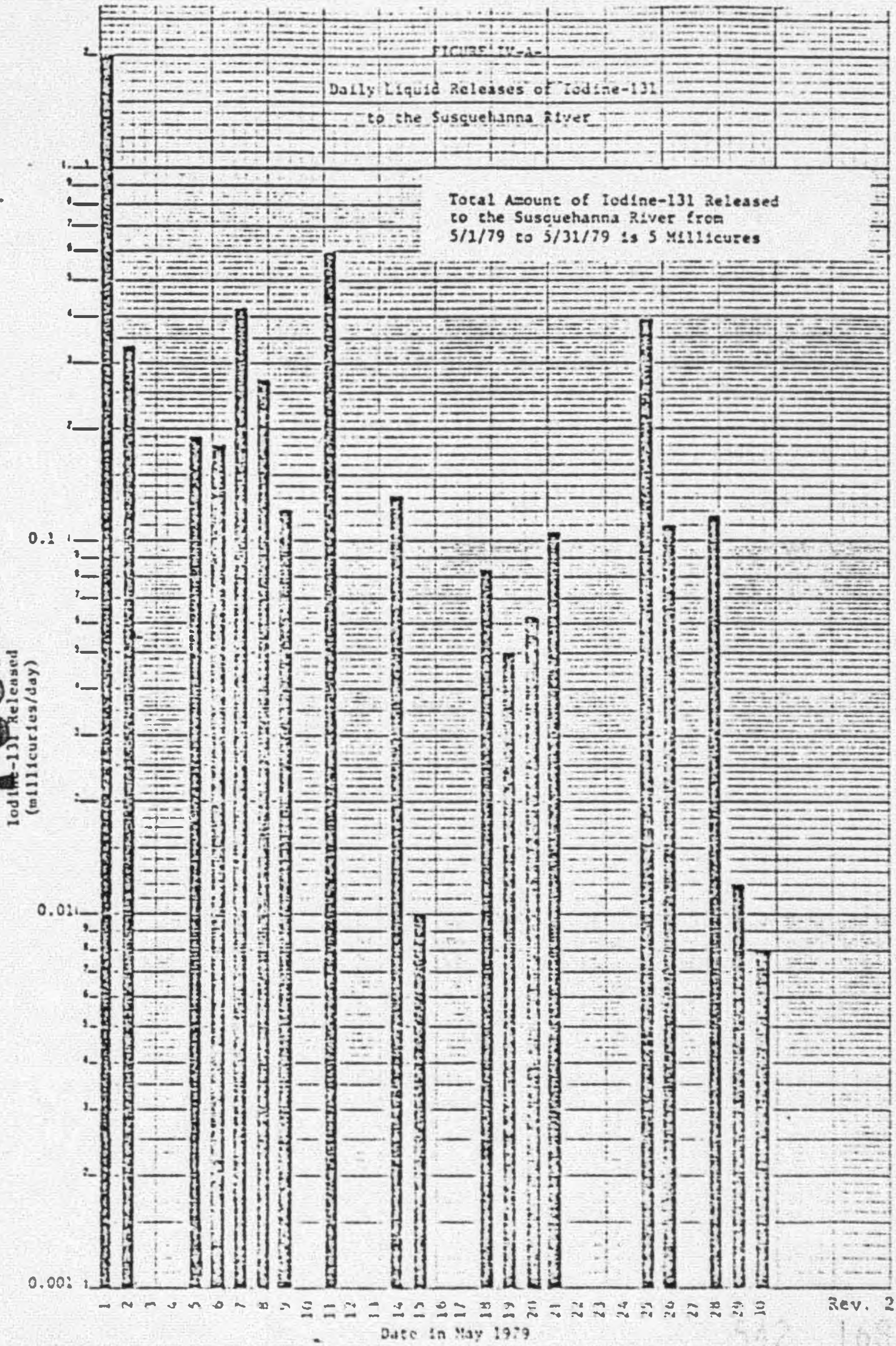

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IV. B. OFFSITE NOBLE GAS RELEASES AND DOSES

1. Estimated Releases

During the period March 28, 1979 through April 30, 1979, about 10 million curies of noble gases were released to the environment via the ventilation system of Units 1 and 2. This estimate is down slightly from 12 million curies reported in the June 18 submittal due to refinements in the data used with the analytical methodology described in the June 18 submittal. Revised data and results are presented in Table IV-B-1 and Figure IV-B-1 and IV-B-2. All of this release was a result of the Unit 2 accident of March 28, 1979.

During the period May 1, 1979 through May 31, 1979, approximately 1366 curies of noble gases (^{131m}Xe, ^{133}Xe and ^{135}Xe) were released to the environment via the ventilation system of Unit 2. The release of noble gases was determined by obtaining daily "grab" gas samples for gamma (GeLi) spectroanalysis to determine isotopic concentrations and an evaluation of strip chart records for this period. Daily noble gas release quantities are plotted in Figure IV-B-3. Releases in May did not represent a significant increment to releases for the period March 28 through April 30, 1979.

2. Environmental Measurements

The Radiological Environmental Monitoring Program (REMP) conducted by Metropolitan Edison Company is described in Attachment 1 to the June 18, 1979 submittal. Additional data are included in Attachment 1 to this submittal. The doses from noble gases released in May were not distinguishable from doses from naturally occurring background radiation. The highest measured exposure offsite for the period March 28 through April 30, reported in the June 18 submittal as 83 mR has been revised downward slightly to 75 mR based on a new assessment of the contribution of natural background radiation to the dose measured at that location, Station 4A1, 1200 meters NNE from the site.

3. Estimated Offsite Whole Body Doses

The maximum offsite whole body dose to an individual, based on the revised measurement result discussed above, is 75 millirem for the period March 28 through April 30. The estimated aggregate whole body dose to the two million persons from noble gases released from March 28 through April 30 is about 3300 person-rems, as reported previously. The addition of dose contributions from releases in May do not change these values significantly.

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Table IV-B-1

Estimated Quantities (Ci) of Each Noble Gas Isotope for Release Periods Corresponding to TLD Measurements 3/28/79-4/30/79

Isotope	3/28 @ 0700- 3/29 @ 1600	3/29 @ 1700- 3/31 @ 1600	3/31 @ 1700- 4/3 @ 1500	4/3 @ 1600- 4/6 @ 1300	4/6 @ 1400- 4/30 @ 2400*	Total
Xe-133	4.9E6	2.1E6	1.1E6	2.7E5	1.5E4	8.3E6
Xe-133m	1.2E5	3.9E4	1.5E4	1.9E3	0	1.7E5
Xe-135	1.5E6	7.7E4	1.4E3	0	0	1.5E6
Xe-135m	1.4E5	1.3E3	0	0	0	1.4E5
Kr-88	6.1E4	0	0	0	0	6.1E4
	6.6E6	2.2E6	1.1E6	2.7E5	1.5E4	1.0E7

*The last three weeks of the month are combined into one group since the contribution is less than 1% of the total. The estimated quantity released during this period is based on effluent measurements.

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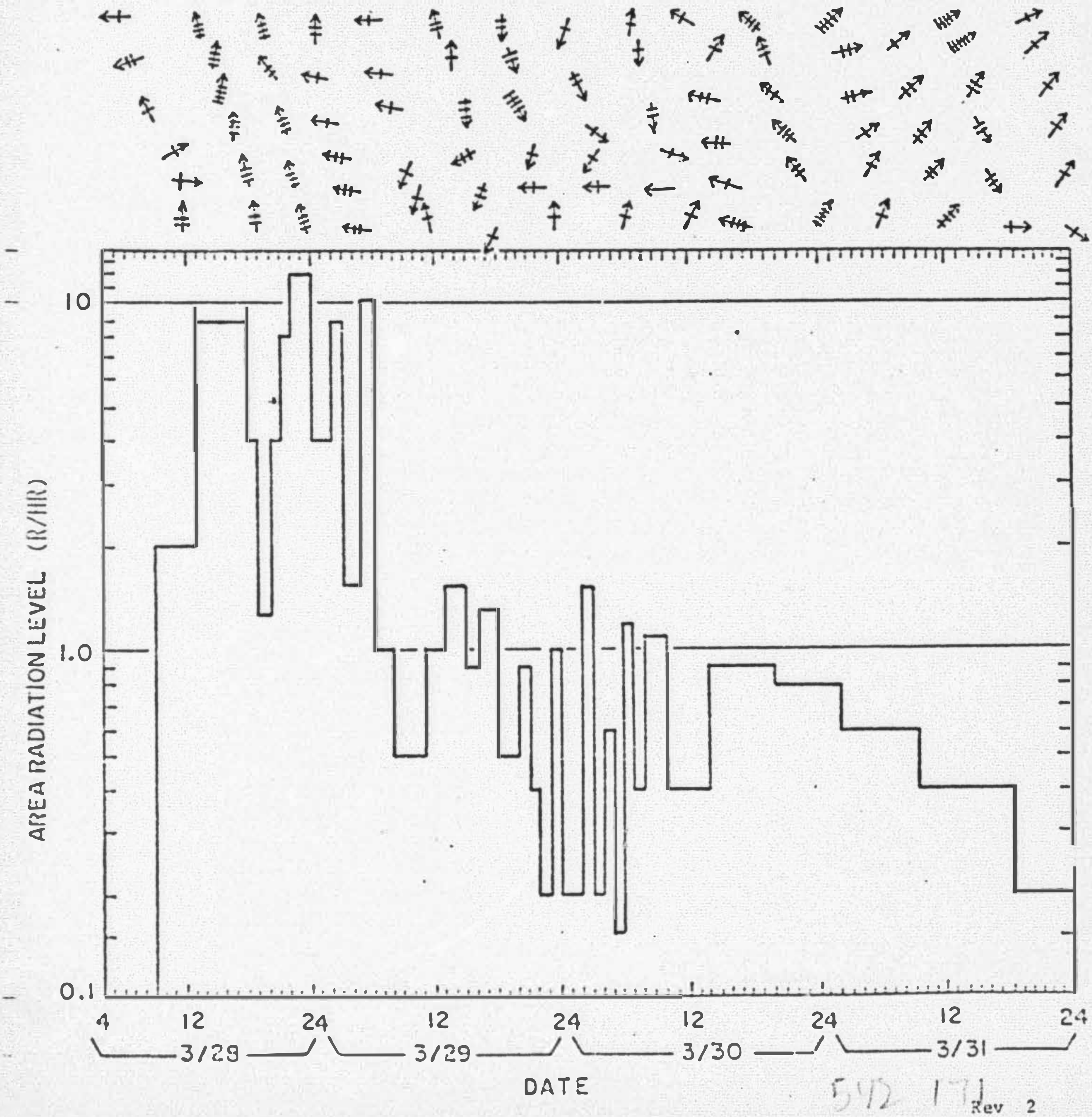
Figure IV-B-1

TREND OF AUXILIARY AND FUEL HANDLING BUILDING AREA RADIATION MONITORS

ARROWS INDICATE HOURLY WIND DIRECTION, EACH BAR ON ARROW INDICATES 3MPH WIND SPEED

↑ INDICATES WIND TOWARD NORTH



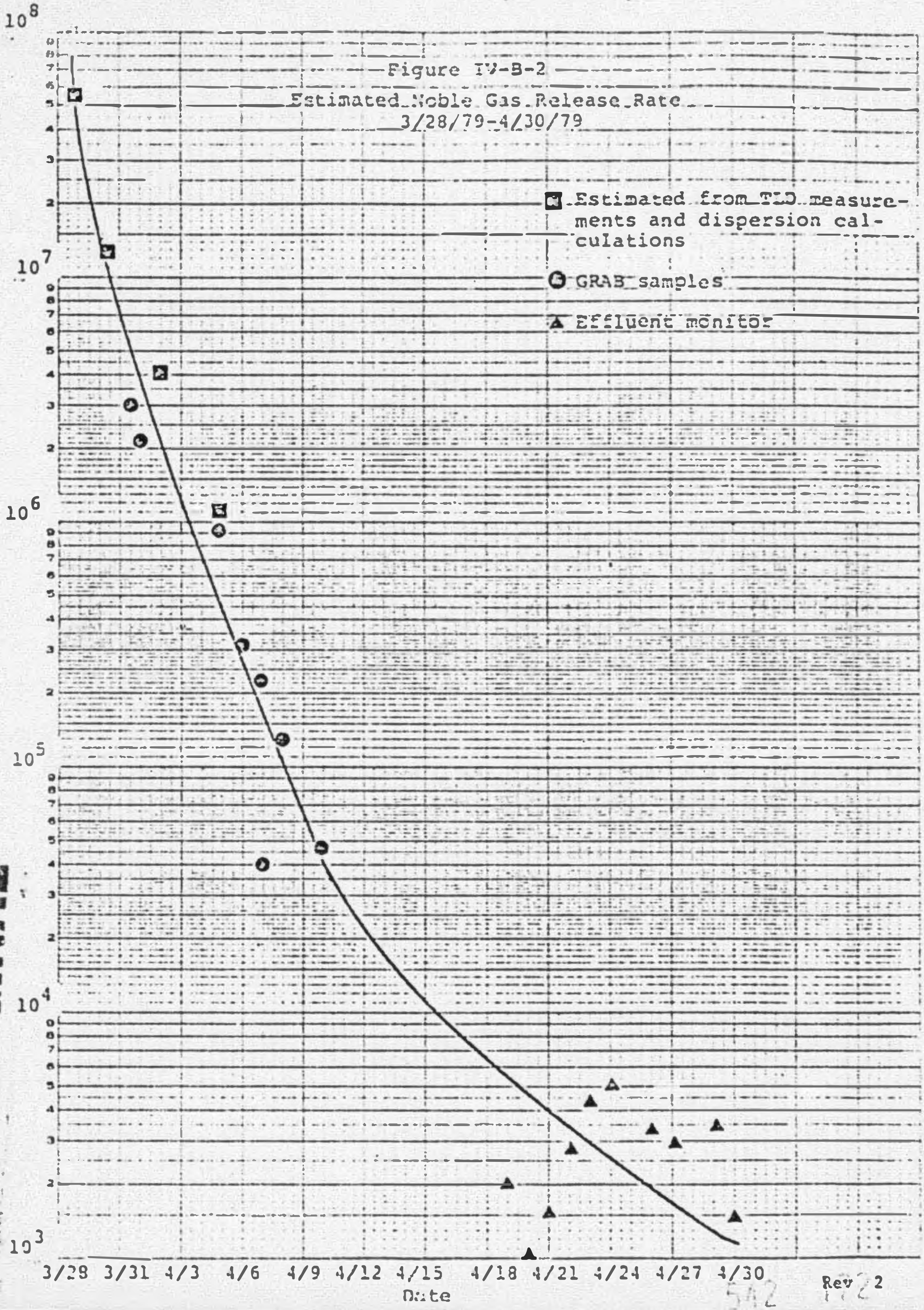


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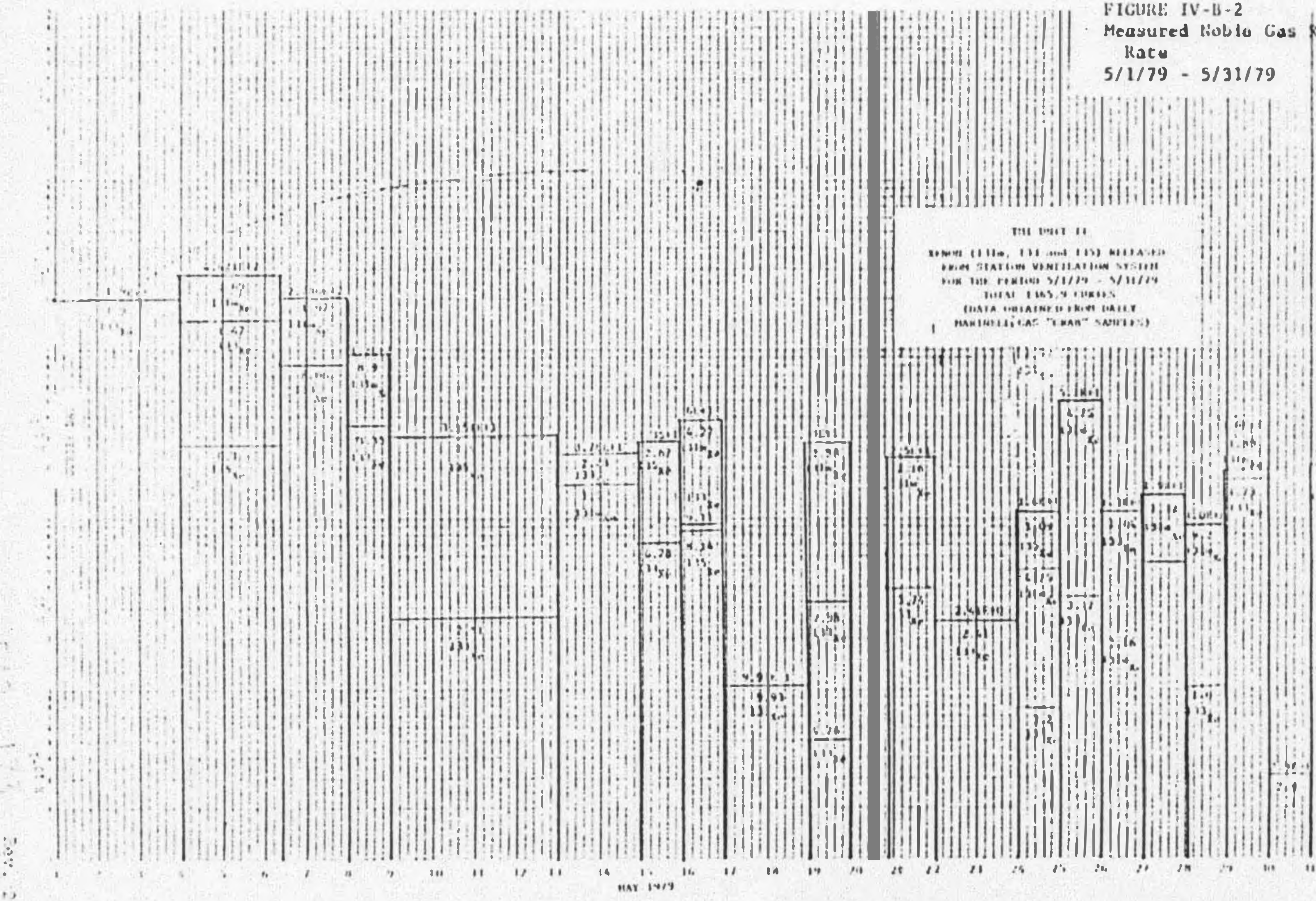
FIGURE IV-B-2
Measured Noble Gas Release Rate
5/1/79 - 5/31/79



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IV. C. OFF-SITE IODINE AND PARTICULATE RELEASES AND DOSES

1. Releases

During the period March 28, 1979 through April 30, 1979 about 14.1 curies of Iodine-131 and 2.6 curies of Iodine-133 were released to the environment via the ventilation systems of Unit 1 and 2. These releases were discussed in the June 18 submittal. All of these releases resulted from the Unit 2 accident of March 28, 1979.

During the period of May, 1979 through May 31, 1979 approximately 7.8E-2 curies of Iodine 131 were released to the environment via the ventilation system of Unit 2. This estimate is based on periodic collection and analysis of continuous effluent samples. The Unit 2 station vent was capped on about May 20, 1979 and since that time the effluent air has been released through at least three of four newly installed filter trains which back up normal plant air cleaning systems. Table IV-C-1 and Figure IV-C-1 summarizes the release of Iodine-131 for the period May 1, 1979 through May 31, 1979. Releases in May, 1979 represent a negligible increment to Iodine-131 releases during the period March 28, 1979 through April 30, 1979.

Preliminary evaluations of measured particulate concentrations and releases indicate these isotopes are not significant in off-site dose assessments. These evaluations are continuing and results will be presented in a later report.

2. Environmental Measurements

In support of routine plant operations, Metropolitan Edison Company conducts an Radiological Environmental Monitoring Program. Included are continuous air samples for iodine and particulate isotopes, vegetation samples and milk samples. This program has continued with a higher sampling frequency since the accident. Attachment 1 of the June 18 submittal includes a tabulation of these data and a brief discussion of the program. Attachment 1 to this report includes supplementary information. Results indicate that Iodine-131 was the only iodine or particulate isotope released in significant quantities. This isotope was detected in air and milk, as discussed below, and was also detected in some grass samples in the period March 28 through April 30. In early May, low concentrations of Iodine-131 were detected in some milk samples and in one air sample. By the last week of the month concentrations of Iodine-131 had fallen below detectable levels in all samples.

3. Estimated Off-Site Exposures

A summary of off-site doses from radioactive iodines is given in the Executive Summary Table.

The methodology and results of the assessment of doses from atmospheric releases of iodine isotopes were included in the June 18 submittal. Estimates of maximum individual doses from inhalation of air containing Iodine-131 and ingestion of milk containing Iodine-131, based on environmental measurements, have been revised slightly from 2.7 to 3.7 millirem for inhalation and from 1.2 to 1.5 millirem for milk ingestion. Results of other dose assessments for iodine isotopes released to the atmosphere remain as reported in the June 18 submittal.

TABLE IV-C-1

Ci/Day from Charcoal

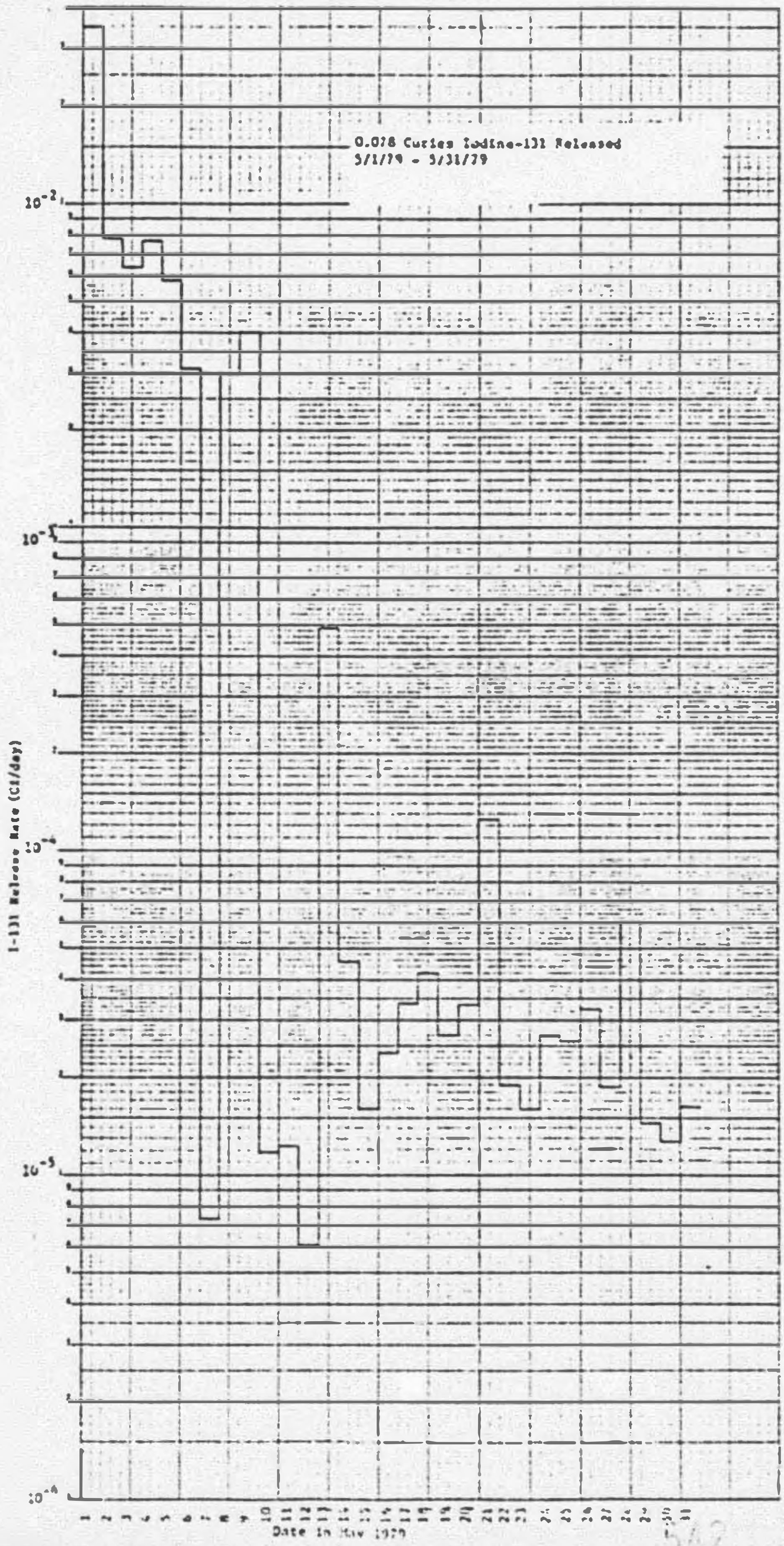
^{131}I Releases

5/1/79 - 5/31/79

Date	Total
5/1/79	3.50E-2
5/2/79	7.83E-3
5/3/79	6.30E-3
5/4/79	7.70E-3
5/5/79	5.81E-3
5/6/79	3.11E-3
5/7/79	7.26E-6
5/8/79	3.01E-3
5/9/79	4.08E-3
5/10/79	1.18E-5
5/11/79	1.21E-5
5/12/79	6.08E-6
5/13/79	4.92E-4
5/14/79	4.49E-5
5/15/79	1.99E-5
5/16/79	2.39E-5
5/17/79	3.37E-5
5/18/79	4.20E-5
5/19/79	2.70E-5
5/20/79	3.34E-5
5/21/79	1.25E-4
5/22/79	1.89E-5
5/23/79	1.61E-5
5/24/79	2.70E-5
5/25/79	2.56E-5
5/26/79	3.25E-5
5/27/79	1.86E-5
5/28/79	6.04E-5
5/29/79	1.46E-5
5/30/79	1.28E-5
5/31/79	1.[illegible]2E-5
	7.397E-2

Figure IV-C-1

Release of I-131 to Atmosphere from Unit 2



ATTACHMENT 1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

The information in this attachment supplements information provided in the submittal dated June 18, 1979, which also includes a description of the program and information to aid in interpreting results (maps, etc.)

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Summary

Specific radioanalytical results are found in Appendix B of this and the second Interim Report and the dosimetric applications of these findings are found in Appendix C of this report.

Waterborne Pathways

Surface & Drinking Water

All water samples were analyzed for radioiodine, tritium, and gross beta activities, as well as by gamma spectroscopy. Thirteen samples, 4 upstream and 9 downstream, had very low positive results for radioiodine (all <2.0 pCi/l) while all other samples had no detectable radioiodine. Tritium and gross beta activities were at normal ambient levels for all samples and no reactor produced radionuclides were found.

Effluent Water

Tritium levels ranged from 100 to 3690 pCi/l through April 28, 1979; iodine-131 levels ranged from 0.1 to 62 pCi/l through May 29, 1979; no gamma emitters other than iodine-131 and on 2 occasions cobalt-58 were found. The levels of radioactivity found, based on the surface and drinking water results, had no discernible effect offsite.

Fishes - Aquatic Sediment - Aquatic Plants

Analysis of fishes found only naturally occurring potassium-40 and occasional low levels of fallout cesium-137. Analysis of sediment samples found normal levels of naturally occurring radionuclides and on occasion low levels of cobalt-58, cesium-134, and manganese-54. No aquatic plants were found.

Airborne Pathways

Gross beta analyses of airborne particulates found typical background activities at all locations at all times. Radioiodine analyses found activities ranging from 0.02 to 23.9 pCi/m^3 within 3 days post incident. The distribution of these values was such that locations closest to Three Mile Island had the highest activities. No radioiodine was detected in any samples after May 3.

Terrestrial Pathways

Milk

Analyses of cow's milk noted radioiodine levels ranging from 0.1 to 21 pCi/l and normal background levels of cesium-137 and potassium-40. The higher radioiodine results were immediately post incident and have been decreasing such that no radioiodine has been detected since May 20. Analysis of goat's milk found radioiodine levels ranging from <0.3 to 110 pCi/l and normal background levels of cesium-137 and potassium-40. It should be noted that most to all goat's milk production in April was used to suckle newborn kids and thus there was little to no human exposure via this pathway. Radioiodine levels in goat's milk in May ranged from <0.3 to 49 pCi/l; no radioiodine was detected after May 24.

Rainwater

Tritium, gross beta, and gamma spectrometric analyses through April 27 found normal ambient activities and naturally occurring radionuclides only. Radioiodine analyses found detectable activities (2.1 and 1.2 pCi/l) in 2 indicator samples for the period March 31 through April 5, 1979; no other samples had detectable levels of radioiodine.

Other Samples

Two of 6 grass samples had low but detectable levels of radioiodine (0.033 and 0.063 pCi/g); no radioiodine or reactor produced radionuclides were found in soil, poultry, beef, eggs, or game.

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Teledyne Results in milli-roentgens
NRC (Q) Results in milli-rads
The Total Error is 1 sigma

TLD'S
NET EXPOSURES
INCLUDES BACKGROUND

Map #	Page 1 of 1		4/28-5/5	5/5-5/12	5/12-5/19	5/19-5/26	5/26-6/2		
2	N. Weather Station	1S2	0.8 ± 0.9	1.1 ± 0.4	1.0 ± 0.3	1.3 ± 0.3	0.8 ± 0.3		
2	N. Weather Station	1S2Q	1.49±0.12	1.57 ± 0.18	1.36 ± 0.09	1.85± 0.15	1.48 ± 0.28		
3	N. Bridge	2S2	0.4 ±0.7	0.8 ± 0.3	0.8 ± 0.3	1.0 ± 0.3	0.5 ± 0.3		
5	Top of Dike	4S2	0.5 ± 0.7	1.0 ± 0.3	1.0 ± 0.3	1.0 ± 0.2	0.7 ± 0.4		
5	Top of Dike	4S2Q	1.49±0.20	1.49±0.14	1.44 ± 0.15	1.01 ± 0.24	1.34 ± 0.10		
6	Top of Dike	5S2	0.6 ± 0.7	1.0 ± 0.3	0.9 ± 0.3	1.5 ± 0.4	0.9 ± 0.3		
6	Top of Dike	5S2Q	1.46±0.11	-	1.44 ± 0.26	1.75± 0.22	1.24 ± 0.10		
28	Falmouth-Collins Sub	8C1	0.3 ± 0.7	1.0 ± 0.3	0.7 ± 0.3	0.9 ± 0.2	1.1 ± 0.3		
28	Falmouth-Collins Sub	8C1Q	0.96±0.18	1.23±0.21	1.13 ± 0.11	1.31 ± 0.09	0.98 ± 0.17		
8	S. TMI	9S2	0.6 ± 0.7	1.1 ± 0.3	1.2 ± 0.3	1.2 ± 0.2	0.9 ± 0.3		
9	EDCT	11S1	1.1 ± 0.7	1.4 ± 0.3	1.2 ± 0.3	1.4 ± 0.2	1.1 ± 0.3		
9	EDCT	11S1Q	1.85±0.16	1.67±0.24	1.72 ± 0.30	2.14 ± 0.32	1.52 ± 0.11		
11	N. Boat Dock	16S1	1.2 ± 0.7	1.5 ± 0.3	1.4 ± 0.3	1.6 ± 0.2	1.1 ± 0.3		
11	N. Boat Dock	16S1Q	1.86±0.16	1.80±0.11	1.52 ± 0.11	2.20 ± 0.44	1.72 ± 0.32		
10	Shelley	14S2	0.4 ± 0.7	0.8 ± 0.3	0.7 ± 0.3	0.9 ± 0.2	1.1 ± 0.3		
13	Laurel Rd	4A1	0.6 ± 0.7	1.0 ± 0.3	1.0 ± 0.3	1.2 ± 0.2	0.7 ± 0.3		
14	Observ. Center	5A1	0.6 ± 0.7	1.0 ± 0.3	0.9 ± 0.3	1.2 ± 0.2	0.8 ± 0.3		
14	Observ. Center	5A1Q	1.40±0.18	1.49±0.12	1.23 ± 0.18	1.61 ± 0.09	1.34 ± 0.22		
17	Kohr Island	16A1	0.5 ± 0.1	1.2 ± 0.3	1.0 ± 0.3	1.0 ± 0.2	-		
23	S. End Shelley	10B1	0.8 ± 0.7	1.6 ± 0.3	1.2 ± 0.3	1.4 ± 0.2	0.6 ± 0.1		
24	Goldsboro Air Station	12B1	0.4 ± 0.7	1.0 ± 0.4	0.7 ± 0.3	0.9 ± 0.2	0.6 ± 0.3		
26	Middletown Sub	1C1	0.6 ± 0.7	1.2 ± 0.9	0.8 ± 0.3	1.0 ± 0.2	0.6 ± 0.3		
34	Drager Farm	7F1	1.3 ± 0.7	1.6 ± 0.3	1.6 ± 0.3	1.8 ± 0.2	1.4 ± 0.4		
34	Drager Farm	7F1Q	2.06±0.08	2.07 ± 0.29	1.87 ± 0.21	2.24 ± 0.39	1.72 ± 0.35		
37	RTE 241	4G1	0.7 ± 0.7	1.4 ± 0.2	1.1 ± 0.3	1.3 ± 0.2	1.5 ± 0.3		
37	RTE 241	4G1Q	1.60±0.16	1.80 ± 0.14	1.71 ± 0.17	1.77 ± 0.40	1.25 ± 0.19		
39	N. York Sub	9G1	0.9 ± 0.7	1.2 ± 0.3	1.2 ± 0.3	1.3 ± 0.2	1.1 ± 0.4		
40	W. Fairview	15G1	0.9 ± 0.7	1.2 ± 0.3	1.1 ± 0.3	1.3 ± 0.2	1.1 ± 0.3		
40	W. Fairview	15G1Q	1.38±0.14	1.5 ± 0.13	1.56 ± 0.29	1.71 ± 0.16	1.49 ± 0.36		
38	Columbia	7G1	1.3 ± 0.7	1.9 ± 0.4	1.6 ± 0.3	1.8 ± 0.2	1.7 ± 0.3		

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POOR ORIGINAL

TLD'S

EXPOSURE TIMES

Map #	Sampling Location		Time Collect 4/28	Elpsed Hours	Time Collect 5/5	Elpsed Hours	Time Collect 5/12	Elpsed Hours	Time Collect 5/19	Elpsed Hours	Time Collect 5/26	Elpsed Hours	Time Collect 6/2
2	N. Weather Station	1S2	1440	172.53	1912	166.80	1800	163.58	1335	166.78	1222	169.30	1340
2	N. Weather Station	1S2Q											
3	N. Bridge	2S2	1557	171.88	1950	167.62	1927	163.47	1455	166.50	1325	169.53	1457
5	Top of Dike	4S2	1550	171.83	1940	167.67	1920	163.42	1445	166.50	1315	169.53	1447
5	Top of Dike	4S2Q											
6	Top of Dike	5S2	1553	171.83	1943	167.67	1923	163.42	1448	166.50	1310	169.53	1450
6	Top of Dike	5S2Q											
28	Falmouth-Collins Sub	8C1	0800	174.83	1450	167.50	1420	164.25	1035	166.83	0925	169.75	1110
28	Falmouth-Collins Sub	8C1Q											
8	S. TMI	9S2	1457	171.97	1855	167.50	1825	163.45	1352	166.80	1240	169.30	1358
9	MDCT	11S1	1447	172.52	1918	166.83	1808	163.57	1342	166.80	1230	169.32	1349
9	MDCT	11S1Q											
11	N. Boat Dock	16S1	1444	172.52	1915	166.83	1805	163.57	1339	166.77	1225	169.37	1347
11	N. Boat Dock	16S1Q											
10	Shelley	14S2	1055	165.22	0808	168.78	0855	168.22	0908	167.37	0830	169.75	1045
13	Laurel Rd	4A1	0745	166.83	0635	168.50	0705	168.75	0750	167.17	0700	176.00	1500
14	Observ. Center	5A1	0755	166.58	0630	168.37	0652	168.88	0745	167.17	0655	174.25	1310
14	Observ. Center	5A1Q											
17	Kohr Island	16A1	1050	165.22	0803	168.78	0850	168.42	0915	167.17	0825	169.75	1010
23	S. End Shelley	10B1	1105	165.67	0845	168.30	0903	167.95	0900	167.62	0837	169.80	1025
24	Goldsboro Air Station	12B1	1030	165.33	0750	169.50	0920	168.17	0930	167.50	0900	168.75	0945
26	Middletown Sub	1C1	1305	170.92	1600	167.33	1520	162.83	1010	166.83	0900	169.58	1035
34	Drager Farm	7F1	1812	164.38	1435	165.50	1205	168.33	1225	167.50	1155	163.17	0705
34	Drager Farm	7F1Q											
37	RTE 241	4G1	1415	171.17	1725	166.58	1600	163.67	1140	166.67	1020	169.75	1205
37	RTE 241	4G1Q											
39	N. York Sub	9G1	0905	168.42	0930	169.00	1030	168.08	1035	166.67	0915	166.92	0810
40	W. Fairview	15G1	1155	163.17	0705	168.75	0750	169.42	0915	167.00	0815	169.25	0930
40	W. Fairview	15G1Q											
38	Columbia	7G1	0825	173.58	1400	165.75	1145	168.33	1205	167.42	1130	163.92	0725

POOR ORIGINAL

Rev. 1

(milli-roentgens) (0.955) = milli-rads

TLD'S

mr/hr

INCLUDES BACKGROUND

Regular Stations - Results in milli-roentgens/hour
Q Stations - Results in milli-rads/hour

Map #		Page 1 of 1		4/28-5/5	5/5-5/12	5/12-5/19	5/19-5/26	5/26-6/2
2	1	N. Weather Station	1S2	0.005	0.007	0.006	0.008	0.005
2	2	N. Weather Station	1S2Q	0.009	0.009	0.008	0.011	0.009
3	3	N. Bridge	2S2	0.002	0.005	0.005	0.006	0.003
5	4	Top of Dike	4S2	0.003	0.006	0.006	0.006	0.004
5	5	Top of Dike	4S2Q	0.009	0.009	0.009	0.011	0.008
6	6	Top of Dike	5S2	0.003	0.006	0.006	0.009	0.005
6	7	Top of Dike	5S2Q	0.008		0.009	0.010	0.003
28	8	Falmouth-Collins Sub	8C1	0.002	0.006	0.004	0.005	0.006
28	9	Falmouth-Collins Sub	8C1Q	0.005	0.007	0.007	0.008	0.006
8	10	S. TMI	9S2	0.003	0.007	0.007	0.007	0.005
9	11	MDCT	11S1	0.006	0.008	0.007	0.008	0.006
9	12	MDCT	11S1Q	0.011	0.010	0.011	0.013	0.009
11	13	N. Boat Dock	16S1	0.007	0.009	0.009	0.008	0.006
11	14	N. Boat Dock	16S1Q	0.011	0.011	0.009	0.013	0.010
10	15	Shelley	14S2	0.002	0.005	0.004	0.005	0.006
13	16	Laurel Rd	4A1	0.004	0.006	0.006	0.007	0.004
14	17	Observ. Center	5A1	0.004	0.006	0.005	0.007	0.005
14	18	Observ. Center	5A1Q	0.008	0.009	0.007	0.010	0.008
17	19	Kohr Island	16A1	0.003	0.007	0.006	0.006	-
23	20	S. End Shelley	10B1	0.005	0.010	0.007	0.008	0.004
24	21	Goldsboro Air Station	12B1	0.002	0.006	0.004	0.005	0.004
26	22	Middletown Sub	1C1	0.004	0.007	0.005	0.006	0.004
34	23	Drager Farm	7F1	0.008	0.010	0.010	0.011	0.009
34	24	Drager Farm	7F1Q	0.013	0.013	0.011	0.013	0.010
37	25	RTE 241	4G1	0.004	0.008	0.007	0.008	0.009
37	26	RTE 241	4G1Q	0.009	0.011	0.010	0.011	0.007
39	27	N. York Sub	9G1	0.005	0.007	0.007	0.008	0.007
40	28	W. Fairview	15G1	0.006	0.007	0.006	0.008	0.006
40	29	W. Fairview	15G1Q	0.008	0.009	0.009	0.010	0.009
38	30	Columbia	7G1	0.007	0.011	0.010	0.011	0.010
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Note: Blank indicates result is less than LLD pCi/g

Parameter	Station	Collect. Date	Wet/Dry	Item	I-131	Cs-137	K-40	Ra-226	Th-232	Be-7	Th-228				
FPF	1B1Q	4/3	W	Chicken	<0.05		3.0								
SOL	1B1Q	4/5	W	Soil	<0.1	1.0	110	0.72	1.0						
VGT	1B1Q	4/5	W	Grass	<0.14	0.32	2.9			3.4					
SOL	4B1Q	4/5	W	Soil	<0.1	0.58	8.7	0.89	0.92						
VGT	4B1Q	4/5	W	Grass	<0.11		3.6			2.3					
SOL	7B3Q	4/5	W	Soil	<0.01	0.72	110	1.0	1.1						
VGT	7B3Q	4/5	W	Grass	<0.2	0.18	3.5			4.0					
FPG	1B1Q	4/5	W	Kid	<0.05		1.5								
FPF	6A1Q	4/8	W	Chicken	<0.13		4.0								
MEF	14C3Q	4/9	W	Pig	<0.09		1.9								
FPE	6B1Q	4/8	W	Eggs	<0.13										
FPB	5G1Q	4/6	W	Beef	<0.15		2.9								
FPF	5G2Q	4/16	W	Chicken	<0.06		3.4								
FPE	5G2Q	4/16	W	Eggs	<0.09										
FPF	14D1	4/5	W	Duck	<0.05		3.09								
FPB	4E1	3/30	W	Beef	<0.09		2.27								
FPF	6B1	4/8	W	Chicken	<0.1										
E	2G1	4/5	D	Soil	<0.2	0.456	19.9	2.49			1.59				
E	14D1	4/5	D	Soil	<0.3	0.677	110				1.57				
E	14C1	4/5	D	Soil	<0.1	1.38	14.3	1.95			1.36				
E	2G1	4/5	W	Grass	<0.01										
E	14D1	4/5	W	Grass	0.033										
E	14C1	4/5	W	Grass	0.063										
FPV	1B1	5/13-1[illegible]	W	Asparagus	<0.0[illegible]		2.14								

POOR ORIGINAL

PRECIPITATION
pCi/ℓ
GROSS BETA, IODINE 131, TRITIUM

	LOCATION		STATION	2/28-3/29	2/28-3/31	3/31-4/5	4/5-4/27	4/27-5/31
40	W. Fairview	GROSS BETA	15G1		10.0	93.0	5.2	
40	W. Fairview		15G1Q	20.9		3.1	<3.3	2.35
28	Falmouth-CollinsSub		8C1		19.0	5.4	6.2	
28	Falmouth-CollinsSub		8C1Q	15.5		2.8	<3.3	4.37
14	Obs Cntr.		5A1		2.9	3.7	3.1	
34	Drager Farm		7F1		14.0	8.1	4.1	
40	W. Fairview	IODINE 131	15G1		<0.2	<0.2		
40	W. Fairview		15G1Q	<0.4		<0.6		
28	Falmouth-CollinsSub		8C1		<0.2	2.1		
28	Falmouth-CollinsSub		8C1Q	<0.3		<0.7		
14	Obs. Cntr.		5A1		<0.4	1.2		
34	Drager Farm		7F1		<0.4	<0.2		
40	W. Fairview	TRITIUM	15G1		<150	<110		
40	W. Fairview		15G1Q	201		<307		
28	Falmouth-CollinsSub		8C1		160	160		
28	Falmouth-CollinsSub		8C1Q	<248		<307		
14	Obs. Cntr.		5A1		160	<120		
34	Drager Farm		7F1		140	100		

POOR ORIGINAL

Note: * = <LLD

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RIVER SEDIMENT
pCi/gm (DRY)

POOR ORIGINAL

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Para.	Stat.	Collect. Date	I-131	Cs-137	K-40	Th-228	Sr-89	Sr-90	Co-58	Cs-134	Ra-226	Mn-54	Be-7	All Other
AQS	1A2	4/10	*	0.085	7.49	0.81	<0.01	0.006	*	*	*	*	*	*
AQS	10B1	4/10	*	0.066	4.88	0.61	<0.009	0.003	*	*	*	*	*	*
AQS	11A1	4/10	*	0.44	7.35	1.29	<0.01	0.007	0.54	0.21	*	*	*	*
AQS	7A1	4/10	*	0.28	10.9	1.15	<0.01	<0.002	*	*	2.25	*	*	*
AQS	9B1	4/10	*	0.18	9.47	1.00	<0.01	<0.002	0.19	*	*	*	*	*
AQS	1A2	4/5	*	0.13	7.22	0.98			*	*	*	*	*	*
AQS	10B1	4/5	*	0.19	6.32	1.27			*	*	*	*	*	*
AQS	11A1	4/5	*	0.24	9.08	1.09			0.57	*	1.56	0.12	*	*
AQS	7A1	4/5	*	0.52	10.4	1.47			*	*	3.00	*	*	*
AQS	9B1	4/5	*	0.43	12.7	1.52			0.37	*	*	*	*	*
AQS	10B1	4/12	*	0.12	5.82	0.72			*	*	1.39	*	*	*
AQS	1A2	4/12												
AQS	7A1	4/12												
AQS	9B1	4/12												
AQS	11A1	4/12												
AQS	7A1	4/16	*	0.284	9.51	1.16			*	*	*	*	1.00	*
AQS	9B1	4/16												
AQS	10A1	4/16	*	0.295	8.96	0.825			0.531	*	*	*	*	*
AQS	1A2	4/16												
AQS	10B1	4/16												
AQS	11A1	4/16												
AQS	1A2	4/23	*	0.222	7.35	0.745			*	*	1.15	*	*	*
AQS	7A1	4/23	*	0.245	9.59	1.16			*	*	1.79	*	*	*
AQS	9B1	4/23	*	0.457	14.7	1.63			0.200	*	2.39	*	*	*
AQS	10B1	4/23	*	0.109	6.63	0.709			*	*	*	*	*	*
AQS	11A1	4/23	*	0.206	7.13	0.926			0.580	*	*	0.072	*	*
AQS	1A2	4/30	*	*	5.07	1.08			*	*	*	*	*	*
AQS	7A1	4/30	*	0.178	8.56	1.27			*	*	*	*	*	*
AQS	9B1	4/30	*	0.525	11.5	1.42			0.329	0.280	2.60	*	*	Co-60= 0.185
AQS	10B1	4/30	*	*	8.67	1.43			*	*	2.01	*	*	*
AQS	11A1	4/30	*	0.310	11.1	1.00			0.385	*	1.31	0.122	*	*
AQS	1A2	5/7	*	0.130	7.45	0.978			*	*	1.41	*	*	*
AQS	7A1	5/7	*	0.525	12.8	1.10			*	*	1.47	*	*	*
AQS	9B1	5/7	*	0.424	12.4	*			0.286	*	*	*	*	*
AQS	10B1	5/7	*	0.095	5.52	0.606			*	*	1.44	*	*	*
AQS	11A1	5/7	*	0.354	8.97	1.43			0.741	*	*	*	*	*

Note: ■ = <LLD

RIVER SEDIMENT
pCi/gm (DRY)

Farm.	Stat.	Date	I-131	Cs-137	K-40	Th-228	Sr-89	Sr-90	Co-58	Cs-134	Ra-226	Mn-54	Be-7	Co-60	All other
AQS	1A2	5/15	■	0.226	9.00	1.18			■	■	■	■	2.09	■	■
AQS	1A2	5/21	■	0.222	9.87	1.29			■	■	■	■	■	■	■
AQS	7A1	5/15	■	0.319	7.74	1.03			■	■	2.06	■	■	■	■
AQS	7A1	5/21	■	0.255	9.83	1.04			■	■	1.61	■	■	■	■
AQS	9B1	5/15	■	0.965	15.8	1.78			0.497	0.303	2.22	■	■	0.193	■
AQS	9B1	5/21	■	0.746	15.2	1.64			0.324	■	2.97	■	■	0.160	■
AQS	10B1	5/15	■	0.113	7.47	0.789			■	■	■	■	■	■	■
AQS	10B1	5/21	■	0.049	5.67	0.660			■	■	1.16	■	■	■	■
AQS	11A1	5/15	■	0.540	10.4	1.46			1.19	■	■	0.353	■	0.148	■
AQS	11A1	5/21	■	0.252	11.7	1.40			0.548	■	■	■	■	■	■
AQS	1A2	5/29	■	0.134	7.85	1.22			■	■	2.33	■	2.10	■	■
AQS	7A1	5/29	■	0.235	12.0	1.32			■	■	■	■	■	■	■
AQS	9B1	5/29	■	0.317	8.73	1.32			0.116	■	■	■	1.56	■	■
AQS	10B1	5/29	■	0.145	7.95	1.13			■	■	1.74	■	■	■	■
AQS	11A1	5/29	■	0.313	10.1	1.32			0.190	■	■	■	■	■	■

POOR ORIGINAL

Rev. 1

X = result by radiochemistry
(X)= result by gamma spec.
40 gm/gal $NaHSO_3$ added to each sample.

Milk
Iodine - 131 (pCi/l)

	Page 1 of 4	Sample	4/22	4/23	4/24	4/25	4/26	4/27	4/28	4/29	4/30	5/1	5/2	5/3
18	Alwine Farm	4B1	<0.2	0.19	<0.2	<0.4	<0.2	<0.2	<0.4	<0.4	<0.3	<0.3	<0.2	< 0.2
20	Becker Farm	7B3	0.57	0.72	<0.4	0.42	0.86	0.49	<0.5	3.6	4.7	3.6	2.9	2.7
20	Becker Farm	7B3Q	1.7	<0.8	<2.2	0.5	<0.5	<0.7	<0.4	4.5	5.2	5.4	2.2	1.6
-	Fisher Farm	14D1	0.31	0.40	1.0	0.77	2.1	-	.89 <.5	<0.3	-	<0.4	<.[illegible] <.2	-
36	Oellig Farm	2G1	<0.2	<0.2	<0.2	<0.2	<0.2	<0.3	<0.5	<0.5	<0.3	<0.3	<0.2	<0.1
45	Hardison Farm	1B1	-	-	-	110. (97.6)	57. (60.8)	39. (46.8)	22. (28.9)	37. (50.8)	26. (21.6)	49.	46.	37.

POOR ORIGINAL

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X = result by radiochemistry
(X) = result by gamma spec.

40 gm/gal NaHSO$_3$ added to each sample.

Milk
Iodine - 131 (pCi/ℓ)

	Page 2 of 4	Sample	5/4	5/5	5/6	5/7	5/8	5/9	5/10	5/11	5/12	5/13	5/14	5/15
18	Alwine Farm	4B1	<0.3	<0.2	<0.5	<0.3	<0.3	<0.3	<0.3	<0.4	<0.4	<0.4	<0.3	<0.3
20	Becker Farm	7B3	1.9	1.4	4.7	1.8	1.9	2.3	0.86	1.4	1.4	1.1	1.0	1.0
20	Becker Farm	7B3Q	1.7	1.6	5.3	3.7	2.3	2.3	1.5	1.5	1.8	1.4	1.8	0.65
-	Fisher Farm	14D1	<0.2	<0.2	<0.2	<0.3	0.72	0.65	0.34	<0.4	0.43	<0.4	<0.3	<0.4 | 0.7
36	Oellig Farm	2G1	<0.3	<0.2	<0.3	<0.5	<0.3	<0.3	<0.3	<0.5	<0.4	<0.5	<0.4	<0.4
45	Hardison Farm	1B1	35.0	-	19.0	14.0	13.0	9.7	16.0	17.0	16.0	6.9	6.4	7.1

POOR ORIGINAL

Milk
Iodine - 131 (pCi/ℓ)

40 gm/gal NaHSO$_3$ added to each sample.

Map Loc	Page 4 of 4	Sample	5/28	5/29	5/30	5/31
18	Alwine Farm	4B1	<0.3	<0.2	<0.3	<0.5
20	Becker Farm	7B3	<0.2	<0.2	<0.4	<0.2
20	Becker Farm	7B3Q	<3.56	<0.2	<1.2	<0.2
-	Fisher Farm	14D1	<0.2	<0.3	<0.5	<0.4
36	Oellig Farm	2G1	<0.2	<0.4	<0.2	<0.3
45	Hardison Farm	1B1	<0.5	<0.3	<0.4	<0.4

Note: Composite samples are taken at 8E1 and 8C2
*Indicates Finished (Treated) Water

Water
pCi/ℓ

IODINE-131

X = result by radiochemistry
(X) = " " Gamma spec

Page 1 of 1

	Sample	5/1	5/2	5/3	5/4	5/5	5/6	5/7	5/8	5/9	5/10	5/11	5/12
Swatara Creek	1C1	<0.4	<0.3	<0.4	<0.6	<0.5	<0.2	<0.4	<0.2	<0.3	<0.2	<0.2	<0.4
Swatara Creek	1C3Q	<0.4	<0.5	0.7	0.7	0.5	0.5	0.6	<0.5	0.6	<0.2	<0.4	<0.4
Brunner Island	8E1	<0.3	<0.5	<0.3	<0.4	<0.3	<0.3	<0.3	<0.5	<0.5	<0.2	<0.3	<0.5
Brunner Island	8E1*	<0.2	<0.4	<0.4	<0.4	<0.4	<0.4	<0.4	<0.5	<0.5	<0.3	<0.5	<0.5
Columbia Water Plant	7G1	<0.4	<0.5	<0.3	<0.3	<0.3	<0.4	<0.4	<0.4	<0.5	<0.4	<0.3	1.3
Columbia Water Plant	7G1Q	<0.3	<0.3	<0.3	<0.4	<0.3	<0.3	<0.4	<0.3	<0.5	<0.3	<0.3	1.5
Steelton Water Works	15F1*	<0.2	<0.5	<0.4	<0.2	<0.3	<0.3	<0.4	<0.4	<0.3	<0.3	<0.5	0.7
Steelton Water Works	15F1Q*	<0.3	<0.3	<0.2	<0.3	<0.3	<0.4	<0.4	<0.5	<0.4	<0.2	<0.6	0.8
YHGS	8C2	<0.3	<0.4	<0.4	<0.5	<0.4	<0.5	<0.4	<0.5	<0.5	<0.4	<0.5	<0.4
YHGS	8C2Q	<0.5	<0.3	<0.2	<0.3	<0.4	<0.3	<0.3	<0.4	<0.3	<0.4	<0.6	<0.6
Discharge Pit	10S1	2.2	2.7	<0.4	<0.3	1.4	<0.4	<0.3	0.43	0.67	0.5	7.2	<0.4
Discharge Pit	10S1Q	2.7	3.9	<0.4	<0.5	2.0	<0.5	<0.5	<0.5	0.7	<0.7	6.2 (5.4)	1.4
York	9G2*	<0.5	<0.4	<0.4	<0.3	<0.3	<0.3	<0.3	<0.3	<0.3	<0.4	<0.4	<0.4
York	9G2Q*	<0.3	<0.3	<0.3	<0.3	<0.3	0.6	<0.4	<0.3	<0.3	<0.4	<0.6	<0.7
Wrightsville	7G2	<0.5	<0.4	<0.4	<0.3	<0.3	<0.5	<0.3	<0.3	<0.3	<0.4	<0.5	<0.4
Wrightsville	7G2*	<0.3	<0.4	<0.3	<0.3	<0.3	<0.5	<0.3	<0.4	<0.2	<0.4	<0.5	<0.3

Note: Composite samples are taken at 8E1, 8C2, 7G1 (starting 5/17) and 10S1 (starting 5/15)
*Indicates Finished (Treated) Water

Water
pCi/l

IODINE-131

		Sample	5/13	5/14	5/15	5/16	5/17	5/18	5/19	5/20	5/21	5/22	5/23	5/24
27	Swatara Creek	1C3	<0.3	<0.2	<0.2	<0.4	<0.5	<0.4	<0.3	<0.5	<0.8	<0.5	<0.3	<0.5
27	Swatara Creek	1C3Q	<0.5	<0.2	<0.3	<0.3	<0.5	<0.6	<0.4	<0.3	<0.4	<0.5	<0.2	<0.4
30	Brunner Island	8E1	<0.3	<0.4	<0.3	<0.5	<0.4	<0.5	<0.5	<0.7	<0.4	<0.4	<0.5	<0.4
30	Brunner Island	8E1*	0.5	<0.4	<0.3	<0.5	<0.4	<0.4	<0.4	<0.4	<0.3	<0.4	<0.5	<0.4
38	Columbia Water Plant	7G1	0.6	<0.3	<0.3	<0.4	<0.5	<0.5	<0.5	<0.3	<0.6	<0.5	<0.5	0.6
38	Columbia Water Plant	7G1Q	<0.7	<0.4	<0.3	<0.4	<0.5	<0.3	<0.3	<0.4	<0.4	<0.6	0.6	0.9
15	Steelton Water Works	15F1*	0.4	<0.3	<0.4	<0.5	<0.5	<0.4	<0.5	<0.3	0.96	<0.5	<0.7	<0.6
15	Steelton Water Works	15F1Q*	<0.7	<0.4	<0.4	<0.2	<0.5	<0.4	<0.3	<0.5	<0.5	0.8	<0.5	<0.4
18	YHGS	8C2	<0.3	<0.3	<0.3	-	<0.3	<0.4	<0.4	<0.5	<0.5	<0.5	<0.5	<0.7
18	YHGS	8C2Q	<0.5	<0.3	<0.3	<0.4	<0.5	<0.5	<0.4	<0.2	<0.3	<0.8	<0.4	<0.5
	Discharge Pit	10S1	14.0	2.1	0.48	<0.5	<0.4	<0.4	<0.5	<0.6	<0.4	0.7	<0.6	<0.6
	Discharge Pit	10S1Q	16.0	2.0	<5.9	<0.7	0.5	<0.5	<0.5	2.1	<0.7	1.6	<0.4	0.7
	York	9G2*	<0.5	<0.5	<0.3	<0.4	<0.4	<0.4	<0.5	<0.4	<0.5	<0.5	<0.5	<0.6
	York	9G2Q*	<0.5	<0.4	<0.4	<0.4	<0.4	<0.4	<0.4	<0.4	<0.4	<0.2	<0.3	<0.5
	Wrightsville	7G2	<0.2	<0.5	<0.5	<0.4	<0.5	<0.5	<0.4	<0.4	<0.3	<0.6	<0.5	<0.6
	Wrightsville	7G2*	<0.3	<0.5	<0.5	<0.3	<0.5	<0.5	<0.5	<0.7	<0.8	<0.5	<0.5	<0.4

Note: Composite samples are taken at 8E1, 8C2, 7G1, 10S1 Water IODINE-131
*Indicates Finished (Treated) Water pCi/ℓ

Map Location	Page 1 of 1	Sample	5/25	5/26	5/27	5/28	5/29	5/30	5/31					
7	Swatara Creek	1C1	< 0.4	<0.3	<0.3	<0.2	<0.5	<0.5	<0.2					
7	Swatara Creek	1C3Q	<0.2	<0.4	<0.3	<0.4	<0.4	<0.5	<0.4					
10	Brunner Island	8E1	<0.4	<0.6	<0.5	<0.3	<0.7		<0.5					
10	Brunner Island	8E1*	<0.5	<0.4	<0.7	<0.3	<0.5							
8	Columbia Water Plant	7G1	1.7	<0.3	<0.5	<0.3	<0.4		0.76					
8	Columbia Water Plant	7G1Q	2.5	<0.5	0.3	<0.4	<0.3	<0.5	0.78					
5	Steelton Water Works	15F1*	<0.5	<0.5	<0.6	<0.4	<0.3							
5	Steelton Water Works	15F1Q*	<0.4	<0.4	<0.4	<0.3	<0.4	<0.5	<0.1					
8	YRGS	8C2	<0.6	<0.8	<0.6	<0.6	<0.4							
8	YRGS	[illegible]2Q	<0.4	<0.4	<0.3	<0.2	<0.3	<0.5	<0.1					
	Discharge Pit	10S1	0.64	0.50	<0.5	<0.4	<0.5							
	Discharge Pit	10S1Q	<0.5	1.9	<0.4	<0.5	<0.4	1.41	1.08					
	York	9G2*	≤ 0.7	<0.5	<0.5	<0.5	<0.3							
	York	9G2Q*	<0.5	<0.3	<0.4	<0.3	<0.3	<0.1	<0.1					
	Wrightsville	7G2	<0.8	<0.3	<0.5	<0.7	<0.7							
	Wrightsville	7G2*	<0.5	<1.0	<0.5	<0.6	<0.6							

Note: Composite samples are taken at 8E1 and 8C2
*Indicates Finished (Treated) Water

Water
pCi/ℓ

GROSS BETA

	Page 1 of 1	Sample	5/1	5/2	5/3	5/4	5/5	5/6	5/7	5/8	5/9	5/10	5/11	5/12
7	Swatara Creek	1C1	2.6	2.6	9.7	3.6	3.5	2.4	2.5	1.9	2.6	2.5	2.7	4.6
7	Swatara Creek	1C1Q	<3.1	<3.3	<3.3	2.2	2.2	3.4	3.4	2.1	2.9	2.7	<3.2	5.0
0	Brunner Island	8E1	3.0	2.9	3.1	8.6	3.6	3.0	3.1	2.2	3.2	3.1	2.4	2.1
0	Brunner Island	8E1*	2.5	<1.0	2.3	<1.0	<1.0	<1.0	3.0	<1.0	2.2	1.9	1.6	<1.0
8	Columbia Water Plant	7G1	2.5	1.9	2.4	4.2	-	3.0	2.5	5.2	2.6	2.3	2.6	4.8
8	Columbia Water Plant	7G1Q	<3.3	<3.3	<3.3	2.2	2.7	3.4	4.0	<3.0	<3.0	2.5	3.1	6.7
5	Steelton Water Works	15F1*	1.6	2.5	1.8	2.7	3.0	2.0	2.3	1.8	1.3	1.9	<1.0	2.2
5	Steelton Water Works	15F1Q*	<3.1	<3.3	<3.3	<3.0	3.6	3.9	2.6	<3.0	<3:0	2.2	7.7	<2.7
8	YHGS	8C2	2.7	1.9	1.4	2.8	1.8	1.4	<1.0	1.3	<1.0	1.6	1.4	1.5
8	YHGS	8C2Q	<3.1	<3.3	<3.3	<3.0	2.2	4.2	2.7	<3.0	<3.0	3.3	2.2	2.1
	Discharge Pit	10S1	6.5	7.7	2.3	2.2	7.7	2.2	2.1	2.7	2.0	1.9	2.8	2.2
	Discharge Pit	10S1Q	<3.3	6.1	3.1	2.9	10.9	2.6	2.6	<3.0	<3.0	3.4	6.8	2.0
	York	9G2*	1.4	2.3	1.4	1.9	2.2	1.7	2.1	1.9	2.6	2.1	2.7	1.8
	York	9G2Q*	<3.3	<3.3	<3.3	<3.0	<3.0	<3.4	3.8	<3.0	<3.0	<3.2	<3.2	2.5
	Wrightsville	7G2	2.9	2.8	2.9	2.5	3.3	4.1	3.4	2.1	2.8	2.8	1.9	3.8
	Wrightsville	7G2*	2.4	2.7	2.2	2.3	2.8	2.3	2.7	2.4	2.7	3.2	4.0	2.6

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Note: Composite samples are taken at 8E1, 8C2, 7G1 (starting 5/17/79), and 10S1 (starting 5/15).
*Indicates Finished (Treated) Water.

Water
pCi/ℓ

GROSS BETA

Map Location	Location	Sample	5/13	5/14	5/15	5/16	5/17	5/18	5/19	5/20	5/21	5/22	5/23	5/24
27	Swatara Creek	1C3	3.6	3.3	3.1	2.7	3.0	3.7	5.3	3.1	2.9	3.3	3.2	—
27	Swatara Creek	1C3Q	3.9	2.3	3.1	2.7	2.1	2.4	<3.3	3.7	2.9	4.0	3.7	2.4
30	Brunner Island	8E1	2.8	3.4	2.7	3.5	3.7	3.7	3.0	3.8	3.7	3.4	2.2	
30	Brunner Island	8E1*	<1.0	<0.9	1.0	1.0	3.0	<0.9	1.1	6.7	<1.0	<1.0	2.0	
38	Columbia Water Plant	7G1	2.2	1.9	2.4	2.2	4.4	3.5	3.4	2.2	2.6	1.4	2.4	
38	Columbia Water Plant	7G1Q	<2.7	3.2	3.4	<2.8	<2.8	<3.3		3.0	2.5	4.1	3.3	2.8
35	Steelton Water Works	15F1*	1.7	2.2	1.6	2.7	2.1	2.3	1.7	2.7	1.9	2.1	1.5	
35	Steelton Water Works	15F1Q*	<2.7	<3.4	<3.4	2.3	<2.8	<3.3	<3.3	3.4	2.3	2.8	3.6	2.0
18	YHGS	8C2	1.5	2.2	2.0	2.3	2.0	1.9	2.3	<1.0	1.7	1.1	1.6	2.0
18	YHGS	8C2Q	2.5	<3.4	<3.4	<2.8	<2.8	<3.3	<3.3	<3.2	<3.2	4.5	2.2	2.8
	Discharge Pit	10S1	3.3		2.9	2.6	2.1	3.0	3.3	5.1	8.0	4.0	3.1	
	Discharge Pit	10S1Q	1.2	5.0	<3.4	<2.0	<2.8	<3.3	3.3	3.8	4.8	5.9	4.8	2.8
	York	9G2*	1.7	2.2	2.0	2.0	3.1	1.7	2.4	2.2	1.9	1.8	2.1	2.2
	York	9G2Q*	<2.7	<3.4	<3.4	2.1	<2.8	4.7	<3.3	<3.2	<3.2	3.7	3.7	2.0
	Wrightsville	7G2	2.3	2.2	3.5	3.2	3.8	4.0	4.3	3.3	3.2	2.4	2.4	5.4
	Wrightsville	7G2*	2.6	3.1	3.4	2.8	2.4	3.7	2.4	3.7	2.4	3.8	2.5	3.6

Note: Composite samples are taken at 8E1, 8C2, 7G1, 10S1
*Indicates Finished (Treated) Water

Water
pCi/ℓ

GROSS BETA

	Location	Sample	5/25	5/26	5/27	5/28	5/29	5/30	5/31					
7	Swatara Creek	1C3		6.9	5.9	3.9	4.7							
7	Swatara Creek	1C3Q	3.8	2.0	<3.1	4.3	2.7	2.5	4.0					
10	Brunner Island	8E1		8.3	7.4	7.8	4.2							
10	Brunner Island	8E1*		2.4	4.5	1.9	<0.9							
8	Columbia Water Plant	7G1		3.8	3.4	2.6	2.6							
8	Columbia Water Plant	7G1Q	3.3	3.1	<3.1	2.6	3.9	2.3	3.4					
5	Steelton Water Works	15F1*	2.1	3.5	2.3	1.7	1.8							
5	Steelton Water Works	15F1Q*	2.2	2.2	<3.1	<2.6	2.9	2.2	2.3					
8	YHGS	8C2	2.1	3.6	5.1	4.2	2.6							
8	YHGS	8C2Q	<2.6	3.0	2.5	3.2	2.6	3.5	<3.1					
	Discharge Pit	10S1		16.0	6.3	13.1	9.2							
	Discharge Pit	10S1Q	5.8	4.1	2.2	11.0	6.7	3.6	<3.1					
	York	9G2*	2.9	2.8	3.8	2.5	2.4							
	York	9G2Q*	2.8	<3.1	2.6	2.7	3.1	<3.1	<3.1					
	Wrightsville	7G2	8.9	7.5	7.1	5.0	3.4							
	Wrightsville	7G2*	3.6	3.7	4.0	3.5	3.4							

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POOR ORIGINAL

Air Particulates - Air Iodine
(Gross Beta) (Iodine-131)
pCi/m^3

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			Station	4/30-5/3	5/3-5/6	5/6-5/9	5/9-5/12	5/12-5/15	5/15-5/18	5/18-5/21	5/21-5/24	5/24-5/27	5/27-5/30	5/30-6/2
2	AP-GROSS BETA	North Weather Station	1S2	0.058	0.050	0.067	0.040	0.033	0.046	0.030	0.043	0.041	0.053	0.064
28		Falmouth Sub.	8C1	0.052	0.037	0.074	0.039	0.031	0.04	0.026	0.036	0.027	0.049	0.034
16		Observation Cntr.	5A1	0.036	0.030	0.075	0.037	0.027	0.053	0.020	0.020	0.025	0.027	0.032
40		West Fairview	15G1	0.060	0.052	0.054	0.062	0.034	0.036	0.032	0.037	0.026	0.042	0.042
34		Drager Farm	7F1	0.11	0.093	0.070	0.086	0.089	0.14	0.047	0.089	0.059	0.068	0.10
34		Drager Farm	7F1Q	0.053	0.054	0.039	0.053	0.027	0.039	0.029	0.042	0.039	0.034	0.065
26		Middletown	1C1	0.058	0.053	0.060	0.048	0.086	0.059	0.039	0.041	0.029	0.057	0.055
26		Middletown	1C1Q	0.046	0.044	0.055	0.030	0.037	0.041	0.034	0.038	0.030	0.035	0.057
4		Goldsboro Air Station	12B1	0.057	0.048	0.180	0.050	0.032	0.054	0.034	0.035	0.027	0.043	0.053
39		North York Sub.	9G1	0.051	0.057	0.080	0.045	0.049	0.045	0.028	0.039	0.026	0.042	0.040
2	AI-IODINE 131	North Weather Station	1S2	<0.081	<0.10	<0.10	<0.09	<0.07	<0.08	<0.09	<0.07	<0.09	<0.08	<0.04
28		Falmouth Sub.	8C1	<0.081	<0.047	<0.09	<0.10	<0.04	<0.09	<0.05	<0.05	<0.05	<0.08	<0.08
16		Observation Cntr.	5A1	0.077	<0.075	<0.07	<0.07	<0.05	<0.06	<0.06	<0.05	<0.06	<0.05	<0.06
40		West Fairview	15G1	<0.087	<0.057	<0.05	<0.06	<0.08	<0.05	<0.09	<0.07	<0.08	<0.09	<0.08
34		Drager Farm	7F1	<0.13	<0.073	<0.05	<0.08	<0.10	<0.10	<0.06	<0.05	<0.07	<0.10	<0.10
34		Drager Farm	7F1Q	<0.057	<0.056	<0.050	<0.056	<0.050	<0.056	<0.033	<0.059	<0.057	<0.067	<0.079
26		Middletown	1C1	<0.062	<0.10	<0.07	<0.07	<0.10	<0.08	<0.07	<0.06	<0.06	<0.10	<0.10
26		Middletown	1C1Q	<0.057	<0.057	<0.052	<0.056	<0.069	<0.041	<0.052	<0.081	<0.050	<0.069	<0.080
4		Goldsboro Air Station	12B1	<0.085	<0.057	<0.07	<0.08	<0.08	<0.08	<0.07	<0.05	<0.07	<0.09	<0.10
		North York Sub.	9G1	<0.070	<0.057	<0.05	<0.07	<0.07	<0.06	<0.06	<0.05	<0.06	<0.08	<0.08

Note: Composite samples are taken at 8E1 and 8C2
*Indicates Finished (Treated) Water

Water TRITIUM
pCi/ℓ

Rev. 1

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Map Location	Location	Sample	4/7	4/8	4/9	4/10	4/11	4/12	4/13	4/14	4/15	4/16	4/17	4/18
27	Swatara Creek	1C1	<100	100	100	180	100	220	270	110	100	220	200	310
27	Swatara Creek	1C1Q	<286	<292	<216	205	<273	<262	<283	<268	<268	<256	<207	440
30	Brunner Island	8E1	110	150	150	<120	100	320	110	100	130	280	210	160 | 150
30	Brunner Island	8E1*	100	100	100	120	<110	110	<100	180	130	270	190	180 | 100
38	Columbia Water Plant	7G1	270	160	100	110	100	120	100	100	<120	200	190	160
38	Columbia Water Plant	7G1Q	<286	190	<216	<264	<273	<262	<283	<268	<309	<256	139	<305
15	Steelton Water Works	15F1*	180	120	100	110	110	150	<100	150	160	190	140	120
15	Steelton Water Works	15F1Q*	<286	<292	<216	<273	<273	<283	<283	<268	<309	<256	<207	<305
18	YHGS	8C2	110	110	110	140	190	210	150	160	120	260	110	<100
18	YHGS	8C2Q	<286	<292	<216	<273	<262	<283	<244	<268	<309	<256	<207	<305
	Discharge Pit	10S1	<130	270	140	510	100	120	2920	220	3690	170	2770	170
	Discharge Pit	10S1Q	<292	<264	181	506	183	<244	2880	<309	3990	<256	2440	<250
	York	9G2*	<130	<110	120	<110	140	160	240	150	<110	130	<110	160
	York	9G2Q*	<286	142	<216	<273	<262	<283	<283	<268	<309	<256	<207	<305

POOR ORIGINAL

Note: Composite samples are taken at 8E1 and 8C2
*Indicates Finished (Treated) Water

Water
pCi/l

TRIT

POOR ORIGINAL

Rev. 1

		Sample	4/19	4/20	4/21	4/22	4/23	4/24	4/25	4/26	4/27	4/28	4/29	4/30
27	Swatara Creek	1C3	220	140	280	170	230	190						
27	Swatara Creek	1C3Q	<305	<287	<287	<312	<270	<220	<220	<279	<226	<272	<235	<262
30	Brunner Island	8E1	300	270	<100	90	260	130				100	320	
30	Brunner Island	8E1*	270	390	190	<110	160	160				220	170	
38	Columbia Water Plant	7G1	100	270	200	410	360	400	310			270		
38	Columbia Water Plant	7G1Q	<250	<287	<250	<312	<270	<220	<220	<279	<226	<272	244	<262
35	Steelton Water Works	15F1*	210	210	240	260	150	310	390			210	170	
35	Steelton Water Works	15F1Q*	<250	<287	<250	<312	180	<220	<252	<279	<226	<272	<235	<262
48	YHGS	8C2	<110	200	190	140	100	100	100					
48	YHGS	8C2Q	<250	<287	<250	<312	233	<220	<252	<279	<226	<272	<235	<262
	Discharge Pit	1081	370	2240	450	240	2290	2580	<120		1570	1660		
	Discharge Pit	1081Q	<250	2440	<262	235	1980	1920	<252	<279	1280	1400	<243	1130
	York	962*	120	180	110	290	310	240	100			160	330	
	York	962Q*	<250	<287	<250	<312	<270	<243	<252	<279	<226	<272	166	<262
	Wrightsville	7G2				200	260	180	280				170	
	Wrightsville	7G2*				260	<110	200	110			170		

Note: Composite samples are taken at 8C1 and 8C2
*Indicates Finished (Treated) Water

Water
pCi/l

TRITIUM

Map Location	Page 3 of 5	Sample	5/1	5/2	5/3	5/4	5/5	5/6	5/7	5/8	5/9	5/10	5/11	5/12
27	Swatara Creek	1C1												
27	Swatara Creek	1C1Q	220	<268	<242	<273	<304	<271	218	<277	<277	<256	<309	<277
30	Brunner Island	8C1												
30	Brunner Island	8E1*												
38	Columbia Water Plant	7G1												
38	Columbia Water Plant	7G1Q	<268	<268	160	<273	<304	<271	<300	<277	<253	<256	<309	<277
35	Steelton Water Works	15F1*												
35	Steelton Water Works	15F1Q*	<243	<268	<242	<273	<304	<271	<264	<277	<253	<256	<309	<277
48	YHGS	8C2												
48	YHGS	8C2Q	<243	<268	<242	<273	<304	<271	<264	<277	<253	<256	<309	<277
	Discharge Pit	10S1												
	Discharge Pit	10S1Q	2300	4110	<266	<266	1820	<300	<300	<253	<256	<309	<309	<308
	York	9G2*												
	York	9G2Q*	<268	<242	<266	<273	<304	<271	<264	<277	<253	213	<309	<277
	Wrightsville	7G2												
	Wrightsville	7G2*												

POOR ORIGINAL

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Map Location

Note: Composite samples are taken at 8E1, 8C2, 761 (starting 5/17), and 1051 (starting 5/15)
*Indicates Finished (Treated) Water

Water TRITIUM
pCi/l

Map Location	Page 4 of 5	Sample	5/13	5/14	5/15	5/16	5/17	5/18	5/19	5/20	5/21	5/22	5/23	5/24
27	Swatara Creek	1C1												
27	Swatara Creek	1C3Q	<308	<211	<211	<197	<197	<196	<278	147	<276	<264	<211	<188
30	Brunner Island	8E1												
30	Brunner Island	8E1*												
38	Columbia Water Plant	761												
38	Columbia Water Plant	761Q	<308	<211	<274	<197	<269	<196	<278	147	<200	<264	<211	<188
35	Steelton Water Works	15F1*												
35	Steelton Water Works	15F1Q*	<308	<211	<274	<197	182	<196	<278	<194	<200	<264	<211	141
48	YNGS	8C2												
48	YNGS	8C2Q	<308	<211	<274	<269	<269	< 196	<270	<194	<200	<264	<211	<138
	Discharge Pit	1051												
	Discharge Pit	1051Q	202	717	<267	<269	<269	<278	<278	461	853	<211	<211	<267
	York	962*												
	York	962Q*	<308	<211	<274	<197	<269	<196	<270	<194	<200	<264	<211	<188
	Wrightsville	762												
	Wrightsville	762*												

Note: Composite samples are taken at 8E1, 8C2, 761, 1081 *Indicates Finished (Treated) Water

Water pCi/l TRITIUM

Map Location	Page 5 of 5	Sample	5/25	5/26	5/27	5/28	5/29	5/30	5/31
27	Swatara Creek	1C1							
27	Swatara Creek	1C1Q	<188	<197	<197	<266	196	<179	<268
30	Brunner Island	8E1							
30	Brunner Island	8E1*							
38	Columbia Water Plant	761							
38	Columbia Water Plant	761Q	<267	<197	<280	<266	131	<179	<268
35	Steelton Water Works	15F1*							
35	Steelton Water Works	15F1Q*	<267	<197	<280	<266	<180	<179	<268
40	YHGS	8C2							
48	YHGS	8C2Q	<267	<197	<280	204	<180	<179	<268
	Discharge Pit	10S1							
	Discharge Pit	10S1Q	1100	381	<182	472	925	142	1042
	York	9G2*							
	York	9G2Q*	173	<197	<280	<266	<180	192	<268
	Wrightsville	762							
	Wrightsville	762*							

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Appendix C

Information On Potential Radiological Doses For The Noted Pathways And Periods

Waterborne Pathways

Location	Period	Nuclide and Average Concentration*
TM-SW-8E1	3/29-4/29	H-3 = 164 pCi/l
	3/29-5/29	I-131 = None detected
	3/29-4/13	γ = No reactor produced radionuclides detected
TM-SW-7G1	3/29-4/28	H-3 = 194 pCi/l
	3/29-5/29	I-131 = 0.38 pCi/l
	3/29-4/13	γ = No reactor produced radionuclides detected
TM-SW-7G2	4/24-4/29	H-3 = 152 pCi/l
	4/22-5/29	I-131 = 0.44 pCi/l
	Not available	γ =
TM-SW-9G2	4/1-4/29	H-3 = 176 pCi/l
	4/1-5/29	I-131 = None detected
	4/1-4/13	γ = No reactor produced radionuclides detected
Susquehanna River South of TMI	3/29-4/29	H-3 = 166 pCi/l
	3/29-5/29	I-131 = 0.36 pCi/l
	3/29-4/12	γ = No reactor produced radionuclides detected

*For purposes of averaging values at or below the detection limit were considered to be the value of the detection limit.

Waterborne Pathways (continued)

Pathway	Location	Radionuclide	Organ	Dose: mrem/period Maximum Individual	Dose: mrem/period Average Individual	Man-rem
Drinking Water	TM-SW-8E1	H-3	Whole body	1.1E-3	6.4E-5	1.6E-5
	TM-SW-7G1	H-3	Whole body	1.3E-3	7.6E-4	7.6E-3
	TM-SW-7G1	I-131	Thyroid	8.9E-2	5.3E-2	-
	TM-SW-7G2	H-3	Whole body	2.0E-4	1.2E-4	3.7E-4
	TM-SW-7G2	I-131	Thyroid	6.5E-2	3.9E-2	-
	TM-SW-9G2	H-3	Whole body	1.1E-3	6.4E-4	8.0E-2
Eating Fish	River	H-3	Whole body	2.5E-5	2.9E-6	-
	River	I-131	Thyroid	3.3E-2	3.7E-3	-
Swimming	River	H-3	Whole body	0	0	-
	River	I-131	Whole body	4.1E-6	1.7 E-7	-
Boating	River	H-3	Whole body	0	0	-
	River	I-131	Whole body	2.1E-6	8.5E-8	-
Shoreline	River	H-3	Whole body	0	0	-
	River	I-131	Whole body	1.3E-5	5.5E-8	-

Milk Pathway (cow's milk only)

Location	Radionuclide and Average Concentration	Period	Organ	Dose: mrem/period
TM-M-7B3	I-131 = 1.8 pCi/l	3/29-5/29	Infant Thyroid	1.5
All locations	I-131 = 0.8 pCi/l	3/29-5/29	Infant Thyroid	0.7

Inhalation of Radioiodine

Location	Average Concentration	Period	Organ	Dose: mrem/period
TM-AI-5A1	1.70 pCi/m^3	3/22-5/27	Adult Thyroid	3.7
TM-AI-1C1	0.76 pCi/m^3	3/22-5/27	Adult Thyroid	1.7
TM-AI-12B1	1.32 pCi/m^3	3/22-5/27	Adult Thyroid	2.9