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Metropolitan Edison Company
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Writer's Direct Dial Number

May 15, 1979
GQL 0693

U. S. Nuclear Regulatory Commission
Director, Office of Inspection & Enforcement
Attn: Boyce Grier, Director
Region 1
631 Park Avenue
King of Prussia, Pennsylvania 19406

Dear Sir:

Three Mile Island Nuclear Station Unit 2 (TMI-2)
License No. DPR-73
Docket No. 50-320

By letter dated April 11, 1979 (GQL 0490) Metropolitan Edison Company committed to submit, by May 15, 1979, a followup report concerning the event which occurred on March 28, 1979 at TMI-2. Enclosed please find that report. Please note, this is a preliminary report, with updates to be submitted as discussed in the above-mentioned letter.

Sincerely,

J. G. Herbein
Vice President-Generation

JGH:RAL:dkf

Enclosure: TMI-2 Incident Report

cc: Director of Nuclear Reactor Regulation
Attn: Mr. S. A. Varga
Light Water Reactor Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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INTERIM REPORT ON THE
THREE MILE ISLAND NUCLEAR STATION
UNIT 2 (TMI-2)
ACCIDENT

MAY 15, 1979

Metropolitan Edison Company

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CONTENTS



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# I. SEQUENCE OF EVENTS

This interim report includes a detailed sequence of events spanning the first approximately twenty hours following initiation of the accident at 0400 on March 28, 1979. Subsequent reports will provide additional information on this time period, as well as information on principal developments during the ensuing period of establishing long term cooling. Later it is anticipated that this section will include discussion of principal millstones achieved in the recovery of TMI-2.

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May 10, 1979
Rev. 0

# PRELIMINARY ANNOTATED SEQUENCE OF EVENTS MARCH 28, 1979

This report provides additional detail to the April 16, 1979 issue of the sequence of events of the March 28, 1979 accident at Three Mile Island Unit 2. This 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 such new information and/or understanding is developed this report will be updated.

The figures included in this report represent the compilation of data from various installed instrumentation and recording sources. Future revisions of this report will identify those sources

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

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



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10 May 79
Rev. 0

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

For this chronology, a reference clock was established with the time of the turbine trip, 0400:37, defined as the time equal to zero. The 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

Three Mile Island Unit Two was at 97% power with the Integrated Control System in full automatic. Rod groups one thru 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 107.5% of design flow and the Reactor Coolant System pressure was 2155 psig. Reactor Coolant Makeup Pump 3 (MU-P-1B) was in service supplying makeup and Reactor Coolant Pump seal injection flow. Normal Reactor Coolant System letdown flow was approximately 70 gpm. 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 the pressurizer to equalize boron concentrations between the pressurizer and the remainder of the Reactor Coolant System. The pressurizer safety valves discharge header thermocouples were indicating 210 F to 230 F due to leakage thru one of the Pressurizer Safety Valves (RC-R1A and RC-R1B).

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Steam Generator parameters were as shown in the following table:

Table of Steam Generator Parameters

| | Steam Generator A | Steam Generator B |
|---|---|---|
| Loop Feedwater | 5.7459 MPPH* | 5.7003 MPPH* |
| Operating Level | 56% | 57.4% |
| Startup Level | 158.8 inches | 163.4 inches |
| Steam Pressure | 910 psig | 889.6 psig |
| Feedwater Temperature | 462.7 F | 462.7 F |
| Steam Temperature | 595 F | 594 F |

* MPPH - Million Pounds Per Hour

Steam Generator Feedwater Pumps (FW-P-1A and FW-P-1B) were in service, Condensate Booster Pumps (CO-P-2A, CO-P-2B and CO-P-2C) were in service, 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.

A loss of feedwater flow transient started when Condensate Pump (CO-P-1A) tripped resulting in both Main Feedwater Pumps (FW-P-1A and FW-P-1B) tripping. The Main Feedwater Pumps trip caused the main turbine to trip.

As was determined later, a small break loss of coolant accident was initiated when the Electromatic Relief Valve (RC-RV2) did not shut properly after opening to reduce Reactor Coolant System pressure. The high reactor coolant system pressure was a direct result of the loss of feed water flow transient which culminated in a reactor trip due to high reactor coolant system pressure.



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| -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. |
| -00:00:01 (0400:36) | Feedwater Pumps (FW-P-1A and FW-P-1B) tripped at essentially the same time resulting in a loss of feedwater flow to both steam generators. |
| 00:00:00 (0400:37) | The main generator and the main turbine tripped in accordance with plant design. |
| 00:00:00 (0400:37) | All three Emergency Feedwater Pumps (EF-P-1, EF-P-2A, EF-P-2B) started. |
| 00:00:03 (0400:40) Approximate | The Electromatic Relief Valve (RC-RV2) opened at the setpoint of 2255 psig. |
| 00:00:08 (0400:45) | The reactor tripped on high Reactor Coolant System pressure at 2345 psig. The setpoint is 2355 psig. |
| 00:00:08 (0400:45) Approximate | The operator placed the Pressurizer Spray Valve (RC-V1) and pressurizer heaters under automatic control. |
| 00:00:10 (0400:47) Approximate | The operator announced on Plant Page System that TMI Unit 2 had a turbine trip and a reactor trip. |
| 00:00:13 (0400:50) | The operator started Reactor Coolant Makeup Pump A (MU-P-1A), opened High Pressure Injection Isolation Valve A (MU-V16A) 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 1). |

00:00:13 (0400:50) Approximate — The Electromatic Relief Valve (RC-RV2) should have shut at about this time (closure setpoint of 2205 psig). The relief valve position indication in the control room is a red lamp which illuminates when the Electromatic Relief Valve solenoid is energized. The solenoid on the Electromatic Relief Valve (RC-RV2) deenergized resulting in a deenergized (i.e. and implied "closed") indication in the control room. Subsequent events showed that the valve had not reseated. The plant operators did not know at this time that the relief valve had not shut. The loss of reactor coolant through RC-RV2 was not stopped until the Electromatic Relief Block Valve (RC-V2) was shut, approximately 2 hours and 22 minutes after the start of the transient.

PLANT STATUS

The Unit had just experienced a Turbine/Reactor Trip. Reactor coolant system pressure and pressurizer level were decreasing rapidly after reaching peaks of 2345 psig and 256 inches respectively. Unknown to the plant operators the Electromatic Relief Valve (RC-RV2) was not shut and was passing reactor coolant from the steam space at the top of the Reactor Coolant System pressurizer. Based on control room indications, the 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/second. The steam generator steam pressures were about 1060 psig and decreasing at 32 psi per second. The turbine bypass valves and a number of main steam relief valves were open relieving steam. The Emergency Feedwater Pumps had started but had not reached normal discharge pressure. The steam generator

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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 8 minutes after the start of the transient. The reason for the block valves being shut is not known. The most likely explaination 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 Feed Pumps (EF-P1, EF-P-2A and EF-P-2B) achieved normal discharge pressure (Figure 49).

00:00:14 (0400:51) Approximate Steam generator levels were approximately 120 inches (Figure 34). Steam pressure had increased to 1065 psig in B Steam Generator and 1073 psig in A Steam Generator (Figure 31) Assuming the steam generator safety valves opened at the specified setpoints then 6 of the B Steam Generator safety valves and 8 of the A Steam Generator safety valves opened.

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.

00:00:20 (0400:57) Approximate The Steam Generator Safety Valves reseated and the Turbine Bypass Valves (MSV-25A, MSV-25B, MSV-26A and MSV-26B) modulated steam flow to the main condenser to control steam generator pressure at 1010 $\pm$ 10 psig (Figure 31).



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| 00:00:30 (0401:07) | Pressurizer Safety Valve (RC-RV13) and Electromatic Relief Valve (RC-RV2) discharge line temperature alarms printed out. The high temperatures in the discharge lines were a result of the high temperature steam flow thru the Electromatic Relief Valve (RC-RV2) during the reactor pressure transient. The Pressurizer Safety Valve (RC-RV13) did not open; the RC-RV13 temperature alarm received was due to the back flow of steam in the common discharge header shared with the Electromatic Relief Valve (RC-RV2). |
| 00:00:38 (0401:15) 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. |
| 00:00:39 (0401:16) | Reactor Coolant Makeup Pump A (MU-P-1A) tripped. |
| 00:00:40 (0401:17) Approximate | Steam Generator B level reached the Integrated Control System setpoint of 30 inches at which the Emergency Feedwater Valve (EF-V11B) opens (Figure 34). Feedwater was not admitted to Steam Generator B because Emergency Feedwater Block Valve (EF-V12B) was shut. |
| 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). |
| 00:01:00 (0401:37) Approximate | Pressurizer level started increasing. Reactor Coolant System hot leg and cold leg temperatures reached approximately 575F |

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(Figure 6). The Reactor Coolant Drain Tank pressure was 12 psig and increasing (Figure 41).

00:01:00 (0401:37) The Pressurizer Safety Valve (RC-RV1A) 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-RV2).

PLANT STATUS

The Reactor Coolant System was recovering from the initial loss of feedwater flow transient. The reactor coolant pressure was still decreasing and pressurizer level had begun to increase (Figures 1 and 3). This was not expected as reactor coolant system pressure and pressurizer level should trend together during a loss of feedwater flow transient. The deviation from expected behavior was due to the failure of the Electromatic Relief Valve (RC-RV2) to reseat and the reduction in heat removal capability of the steam generators because of their low level, which resulted in a lower Reactor Coolant System pressure and subsequent expansion of the reactor coolant volume. The Reactor Coolant Drain Tank pressure and temperature started to increase showing the effects of the continued discharge of reactor coolant thru the Electromatic Relief Valve (RC-RV2). The plant operators did not associate these indications with leakage past the relief valve but rather with the initial opening of the relief valve. 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 closed. The steam generator pressures were being maintained by the Integrated Control System between 975 psig and



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1020 psig (Figure 31). Reactor Coolant Makeup Pumps (MU-P-1A and MU-P-1B) were in operation delivering water to the Reactor Coolant System.

00:01:26 (0402:03) The Reactor Coolant Drain Tank temperature normal alarm was received at 35.5 F which indicated the Reactor Coolant Drain Tank temperature had returned to the normal range.

00:01:45 (0402:22) Approximate Steam Generators A and B 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).

00:02:01 (0402:38) Engineered Safeguards Actuation of High Pressure Injection (HPI) occurred as Reactor Coolant System pressure reached 1640 psig. Reactor Coolant Makeup Pump B (MU-P-1B) tripped automatically as a result of the Engineered Safeguards actuation of HPI. The ES design is such that Makeup Pumps A and C are used for HPI and if running, Makeup Pump B is tripped prior to actuation of Makeup Pumps A and C.

00:02:04 (0402:41) Reactor Coolant Makeup Pump C (MU-P-1C) started automatically.

00:02:12 (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-R72) (Figure 41).

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| 00:03:14 (0403:51) | The operator manually bypassed the High Pressure Injection portion of Engineered Safeguards. Both Reactor Coolant Makeup Pumps A and C (MU-P-1A and MU-P-1C) were operating. |
| 00:03:26 (0404:03) | The Reactor Coolant Drain Tank high temperature alarm was received at 127F. |
| 00:04:38 (0405:15) | The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C). The indicated pressurizer level was 360 inches and increasing rapidly (Figure 29). |
| 00:04:38 (0405:15) Approximate | In an attempt to gain control of the rapidly increasing pressurizer level the operator throttled the High Pressure Injection Isolation Valves (MU-V16A and MU-V16B). |

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. The continued Reactor Coolant System depressurization was due to the failure of the Electromatic Relief Valve (RC-RV2) to reseat, loss of steam generator water level (i.e. reactor heat sink), and reduction in HPI flow. Engineering Safeguards, which actuated High Pressure Injection when pressure reached 1640 psig, had been bypassed by the operator to permit manual control of the Make up Pumps and the HPI Isolation Valves. As the pressurizer level continued to increase, the operator stopped Reactor Coolant Makeup Pump (MU-P-1C) and throttled the HPI Isolation Valves (MU-16A and MU-V16B) in an attempt to control the pressurizer level and not take the pressurizer "solid" (Figure 28). The Reactor Coolant Drain Tank Relief Valve (WDL-R1) had opened at its setpoint of 120 psig and was discharging to the Reactor Containment Building Sump (Figure 41). The Reactor Coolant Drain Tank high temperature alarm had been received as the temperature 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.

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| 00:04:52<br>(0405:29) | Operator started Intermediate Closed Cooling Pump (IC-P-1A) in preparation of putting a second Letdown Cooler in service. |
| 00:04:58<br>(0405:35) | Received the first alarm indicating that Reactor Coolant System letdown flow had been secured. This conflicts with the operator's debriefing statement that letdown flow was isolated at the same ~~time~~ he started Makeup Pump A (MU-P-1A), about 13 seconds into the incident. |
| 00:05:06<br>(0405:43) | Presurizer level stopped its sharp increase at 376 inches and began to decrease. It reached a minimum of 372 inches and then started to increase at 5 minutes, 21 seconds into the transient (Figure 29). Maximum pressurizer level indication is 400 inches. NOTE: Due to the compressed time scale, this event is difficut to identify on Figure 29. |
| 00:05:15<br>(0405:15) | Condensate Booster Pump B (CU-P-2B) tripped. |
| 00:05:50<br>(0406:27)<br>Approximate | Reactor Coolant System pressure stopped its sharp decrease and began to increase. The minimum value reached was approximately 1350 psig (Figure 4). |

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00:05:54 (0406:31) Pressurizer level increased beyond the range of the instrument indication.

00:05:56 (0406:33) Approximate The Reactor Coolant System hotleg temperature and pressure reached saturation conditions of 584F and 1353 psig as indicated by the reactimeter data and control room wide range reactor coolant system pressure stripchart respectively (Figure 5). 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 density decrease contributed to the pressurizer level and pressure behavior at times 00:05:50 and 00:05:54. The reactor out-of-core Source Range Channel (NI-1) and Intermediate Range Channel (NI-3) nuclear instrumentation indication did not increase (Figure 46). Had steam been formed in significant quantities in the reactor core the intermediate range indication should have increased significantly.

00:06:58 (0407:35) Letdown flow of 71.4 gallons per minute was established in an attempt to reduce pressurizer level to the normal operating range (220 inches).

00:07:31 (0408:06) Reactor Building Sump Pump A (WDL-P-2A) started on a high reactor building sump level. The increased sump level was due to the discharge from the Reactor Coolant Drain Tank Relief Valve (WDL-R1) which had been open for approximately 4 minutes.

NOTE: The Reactor Building Sump Pumps generally started about once per shift. For this reason the pump start was not considered extraordinary by the operator.

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00:08:00 (0408:37) Approximate — The operator discovered the Emergency Feedwater Block Valves (EF-V12A and EF-V12B) were shut. He opened EF-V12A and EF-V12B admitting emergency feedwater to the steam generators. Indicated steam generator levels were approximately 10 inches just prior to feedwater addition (Figure 35).

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 became a heat sink for the Reactor Coolant System (Figure 8).

00:08:30 (0409:07 — The Reactor Coolant System pressure began to decrease, reflecting the decrease in Reactor Coolant System temperature (Figures 5 and 6).

00:10:00 (0410:37) — The Pressurizer level indication came on scale.

PLANT STATUS

The reactor coolant 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 established to both steam generators resulting in increased steam pressures (Figure 32). As heat was removed from the Reactor Coolant System, temperature and pressure decreased. The decreasing temperature resulted in a decrease in reactor coolant volume. The pressurizer level indication came on scale. Reactor Coolant System temperature and pressure decreased

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until they stabilized at approximately 550F and 1100 psig respectively (Figures 6 and 11). Reactor Coolant Makeup Pump A (MU-P-1A) was operating.

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 is believed to have been aligned to the Auxiliary Building Sump Tank.

00:10:24 (0411:01) Reactor Coolant Makeup Pump A (MU-P-1A) tripped.

00:10:27 (0411:04) The operator attempted to restart Reactor Coolant Makeup Pump A (MU-P-1A). The pump started and then tripped again.

00:10:40 (0411:25) The Reactor Building Sump high level alarm was received. Setpoint is 4.650 feet from the bottom of the Reactor Containment Building Sump.

00:11:40 (0412:17) The operator started Reactor Coolant Makeup Pump A (MU-P-1A).

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 Containment Building atmosphere.

00:24:58 (0425:35) The operator requested the computer print the Electromatic Relief Valve (RC-RV2) outlet temperature. A value of 285.4F was indicated. The operator attributed the temperature level to the normal cooldown of the discharge header following the initial



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opening and closing of the Electromatic Relief Valve (RC-RV2) and believed the Electromatic Relief Valve (RC-RV2) to be shut.

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| 00:25:00 (0425:37) Approximate | A high radiation alarm was received from Radiation Monitor IC-R-1092 Intermediate Cooling System at the Radiation Monitor Panel. This alarm is normally received following a reactor trip. The Intermediate Cooling System radiation monitor is physically located next to the Reactor Building Sump. |
| 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 (Figure 35). |
| 00:38:10 (0438:47) | The operator stopped Reactor Building Sump Pump A (WDL-P-2A) to prevent overflowing the Auxiliary Building Sump Tank. |
| 00:38:11 (0438:48) | The operator stopped Reactor Building Sump Pump B (WDL-P-2B) to prevent overflowing the Auxiliary Building Sump Tank. |
| 01:10:54 (0511:31) | Reactor Building Air Cooling Coils Emergency Discharge Alarm was received. |

PLANT STATUS

The reactor coolant was near a saturation temperature-pressure relationship at 542F and 1050 psig. The Reactor Coolant System loop flow rates had decreased from about 69 million pounds per hour to approximately 47 million pounds per hour and continued to decrease (Figure 15). The Containment Building temperature and pressure had increased from 0 psig and 120 F to 2.5 psig and 170F, as a result of releasing the contents of the Reactor Coolant Drain Tank to the Reactor Containment Building atmosphere (Figure 45). The Electromatic Relief Valve (RC-RV2) was open.

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Reactor Coolant Makeup Pump A (MU-P-1A) was operating. The operator was having difficulty controlling the level of Steam Generator B. Emergency Feed Water Valves (EF-V11B and EF-V12B) were shut and the operator was admitting feed water to Steam Generator B by cycling Emergency Feed Water Crossconnect Valve (EF-V5B).

01:13:29 (0514:06) Reactor Coolant Pump 2B (RC-P-2B) was stopped in accordance with Emergency Operating Procedure 2202-1.3 to preclude the possibility of damage to the Reactor Coolant Pump from operation near reactor coolant saturation temperature - pressure conditions.

01:13:42 (0514:19) Reactor Coolant Pump 1B (RC-P-1B) was stopped in accordance with Emergency Operating Procedure 2202-1.3 to preclude the possibility of damage to the Reactor Coolant Pump from operation near reactor coolant saturation temperature - pressure conditions.

01:20:31 (0525:37) The operator requested the computer print the Electromatic Relief Valve (RC-RV2) outlet temperature. A value of 283.0F was indicated. Based on this temperature the operator continued to believe that the Electromatic Relief Valve (RC-RV2) was shut.

01:26:23 (0527:00) Steam Generator B was isolated. Main Steam Isolation Valves (MS-V4B and MS-V7B) were shut. Emergency Feedwater Valves (EF-V5B, EF-V11B and EF-V12B) were shut. The operator suspected a Reactor Coolant System to Steam Generator leak based on the large difference in steam pressure between the two Steam Generators and the variations of flow and level experienced while controlling B Steam Generator. A radionuclide analysis of a sample from the B Steam Generator taken at 0724 supported this belief. The sample iodine 131 level was 0.491 microcuries per milliliter.

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| 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 $1.0 \times 10^{-11}$ amperes to approximately $1.6 \times 10^{-11}$ amperes. Correspondingly, the out-of-core Source Range Channel (NI-1) indication increased from about $1.6 \times 10^{4}$ to approximately $5.3 \times 10^{4}$ counts per second. 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. |
| 01:34:12 (0534:49) | Steam Generator A may have boiled dry (Figure 9). |
| 01:37:00 (0537:37) | 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 seconds (Figure 46). This indicated an increased moderator density as a result of the steam collapsing in the reactor vessel. |
| 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 (Figures 35 and 38). The intent was to establish natural circulation cooling within the Reactor Coolant System. Reactor Coolant System Loops A and B cold leg temperatures both decreased (Figures 20 and 25). |

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01:40:37 (0541:14) Reactor Coolant Pump 2A (RC-P-2A) was stopped in accordance with Emergency Operating Procedure 2202 - 1.3 to preclude the possibility of damage to the Reactor Coolant Pump from operation at reactor coolant saturation temperature - pressure conditions.

01:40:45 (0541:22) Reactor Coolant Pump 1A (RC-P-1A) was stopped in accordance with Emergency Operating Procedure 2202 - 1.3 to preclude the possibility of damage to the Reactor Coolant Pump from operation at reactor coolant saturation temperature - pressure conditions.

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) were stopped (Figure 15). The reactor coolant average temperature and pressure were approximately 530F and 1000 psig (Figures 11 and 25). Reactor Coolant Makeup Pump (MU-P-1A) was operating. The operator was attempting to establish natural circulation flow to cool the reactor core. Steam Generator B was isolated because of a suspected Reactor Coolant System to Steam Generator B leak.

01:54:00 (0554:37) Approximate The reactor out-of-core Intermediate Range Channel (NI-3) indication increased from less than $1.0 \times 10^{-11}$ 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 superheated steam spaces formed due to (1)



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increased reactor core temperatures, (2) the absence of reactor coolant system flow, (3) the decreased Reactor Coolant System pressure which resulted from the Electromatic Relief Valve (RC-RV2) being open and the increased reactor cooland density caused by filling Steam Generator A, and (4) the throttled reactor coolant makeup pump flow.

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| 01:54:00 (0554:37) Approximate | Reactor Coolant System Loop A hot leg temperature began to increase, reflecting the steam formation in the upper reactor core region (Figure 20). |
| 02:00:00 (0600:37) | Steam Generator A level indication reached 50% on the operating range (Figure 39). |
| 02:00:00 (0600:37) | Reactor Coolant System Loop B hot leg temperature began increasing (Figure 26). |
| 02:10:55 | Reactor Coolant System Loop A hot leg temperature indication increased offscale, greater than 620F (Figure 21). |
| 02:17:53 (0618:30) | The operator requested the computer print the Electromatic Relief Valve (RC-RV2) outlet temperature. A value of 228.7F was indicated. |
| 02:22:00 (0622:37) Approximate | The Electromatic Relief Block Valve (RC-V2) was shut, stopping reactor coolant leakage through the Electromatic Relief Valve (RC-RV2). The operator noted that the reactor building pressure started to decrease rapidly (Figure 45). |



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|---|---|
| 02:27:23 (0628:00) | The alarm printer malfunctioned. The alarm printer function transferred to the utility printer. The alarm printer was 1 hour and 25 minutes behind in logging data (Figure 48). |
| 02:28:21 (0628:58) | Reactor Coolant Loop B hotleg temperture indication increased offscale, greater than 620F (Figure 26). |
| 02:30:00 (0630:37) | The operator started increasing Steam Generator B level from 30 inches on the Startup Range to 50% on the Operating Range. |
| 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 ppm. 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.<br><br>NOTE: The actual boron concentration in the Reactor Coolant System was in excess of 1000 ppm. 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. |
| 02:45:00 (0645:37) Approximate | Several radiation alarms were received at the Control Room Radiation Monitor Panel. |
| 02:45:00 (0645:37) Approximate | The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C). |
| 02:45:00 (0645:37) Approximate | The operator opened Main Steam Isolation Valves (MS-V4B and MS-V7B) to return Steam Generator B to service prior to starting the Reactor Coolant Pump. |

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| | |
|---|---|
| 02:46:23 (0646:00) Approximate | The operator attempted to start Reactor Coolant Pump 1A (RC-P-1A). The pump would not start. |
| 02:47:31 (0648:08) | The alarm printer function was interrupted which resulted in the loss of alarm data from 0513:59 to 0648:08. |
| 02:51:57 (0652:34) | The operator attempted to start Reactor Coolant Pump 2A (RC-P-2A). The pump would not start. |
| 02:53:19 (0653:53) | The operator attempted to start Reactor Coolant Pump 1B (RC-P-1B). The pump would not start. |
| 02:54:09 (0654:46) | The operator started Reactor Coolant Pump 2B (RC-P-2B) and re-established forced Reactor Coolant System flow (Figure 16). |
| 02:54:49 (0655:26) | High Pressure Injection Engineered Safeguards actuation logic reset on increasing Reactor Coolant System pressure. The set point is 1845 psig. |
| 02:55:00 (0655:37) Approximate | A Site Emergency was declared. Notification of offsite authorities and organizations was initiated. |
| 02:55:00 (0655:37) Approximate | The reactor out-of-core Intermediate Range Channel (NI-3) indication decreased sharply from about $8.0 \times 10^{-11}$ amperes to less than $1.0 \times 10^{-11}$ amperes (minimum detectable level) (Figure 46). The reactor out-of-core Source Range Channel (NI-1) indication showed a corresponding decrease which indicated the collapse of the steam spaces in the reactor core region (Figure 46). |

| | |
|---|---|
| 02:56:19 (0656:56) Approximate | The operator isolated Steam Generator B for the second time. Main Steam Isolation Valves (MS-V4B and MS-V7B) were shut. Emergency Feedwater Valves (EF-V5B, EF-V11B, and EF-V12B) were shut. The operator suspected a Reactor Coolant System to Steam Generator B leak. The Condenser Vacuum Pump Exhaust Radiation Monitor (VA-R-748) increased to 5 x $10^5$ counts per minute. |
| 03:00:00 (0700:37) Approximate | Reactor Coolant System pressure increased to 2130 psig (Figure 12). |
| 03:03:39 (0704:16) Approximate | Steam Generator pressure control was shifted from the Turbine Bypass Valves (MSV-25A and B and MSV-26A and B) to the Power Operated Emergency Main Steam Dump Valves (MSV-3A and B). The reactor out-of-core Source Range Channel and Intermediate Range channel (NI-1 and NI-3, respectively) indicated an increase of approximately a quarter of a decade (Figure 46). |
| 03:10:27 (0711:040 | Emergency Feedwater Pump 2A (EF-P-2A) was stopped. Both Steam Generators had levels were about 50% in the Operating Range (Figure 39). |
| 03:12:28 (0713:05) Approximate | The operator opened the Electromatic Relief Block Valve (RC-V2) in an attempt to establish a normal operating pressurizer level. The reactor out-of-core and Intermediate Range Channel NI-3 indication decreased sharply indicating flow was established thru the reactor core (Figure 46). |
| 03:12:53 (0712:53) | The operator stopped Reactor Coolant Pump 2B (RC-P-2B) after he observed no flow indication and a pump running current of less than 100 amperes. Normal reactor coolant pump operating current |



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is approximately 600 amperes. A close examination of the flow recorder trace indicated a slight amount of reactor coolant flow.

03:18:13 (0718:00) Approximate — The reactor out-of-core Intermediate Range Channel (NI-3) indication increased from less than $1.0 \times 10^{-11}$ amperes to about $2.7 \times 10^{-10}$ amperes, which again indicated steam was formed in the reactor core (Figure 46).

03:19:45 (0720:22) — Engineered Safeguards actuation occurred as a result of low Reactor Coolant System pressure (Figure 4). The setpoint is 1640 psig.

03:20:13 (0720:41) — Reactor Coolant Makeup Pump C (MU-P-IC) started automatically. Reactor Coolant Makeup Pumps A and C (MU-P-1A and MU-P-1C) were operating.

NOTE: During an Engineered Safeguards actuation, High Pressure Injection utilizes Reactor Coolant Makeup Pumps A and C (MU-P-1A and MU-P-1C).

03:21:00 (0721:37) Approximate — The reactor out-of-core Source Range Channel (NI-1) and reactor excore Intermediate Range Channel (NI-3) indication decreased rapidly indicating flow was established thru the reactor core (Figure 46).

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. Superheated steam/gas was present in the reactor vessel head and Reactor Coolant System hot leg.

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Both Reactor Coolant System hot leg temperatures 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 System to steam generator leak. Steam Generator A pressure control was by means of the Power Operated Emergency Main Steam Dump Valve A (MSV-3A). An attempt was in progress to control Reactor Coolant System Pressurizer pressure and level with the Electromatic Relief Block Valve (RC-V2). This resulted in lower Reactor Coolant System pressure which caused an Engineered Safeguard actuation on low Reactor Coolant System pressure. Consequently, both reactor coolant Makeup Pumps A and C (MU-P-1A and 1C) were operating.

| | |
|---|---|
| 03:23:23 (0724:00) Approximate | A General Emergency was declared as a result of a radiation reading of 8 R/hr in the Containment Building. Notification of offsite authorities and organizations was initiated. |
| 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). |
| 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). |
| 03:37:00 (0737:37) | The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C) because pressurizer level was rapidly increasing. Pressurizer level indication was 350 inches (Figure 4). |
| 03:51:00 (0751:37) 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). |

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03:55:40 (0756:15) Intermediate Cooling Pump (IC-P-B) tripped.

03:55:39 (0756:16) Engineered Safeguards actuated on Reactor Building high pressure (Figure 45). The nominal setpoint is 4 psig; the tripped occured at3.58 psig.

03:55:39 (0756:16) The Reactor Building isolated automatically as part of Engineered Safeguards actuation from Reactor Building high pressure. Isolation occurred at the nominal setpoint of 4 psig.

03:56:04 (0756:41) Reactor Coolant Makeup Pump C (MU-P-1C) was started automatically by the Engineered Safeguard actuation.

03:59:23 (0800:00) Reactor Building Emergency Cooler B was shutdown.

03:59:53 (0800:30) Reactor Building Emergency Cooler B was started automatically by the Engineered Safeguards actuation.

04:08:37 (0809:14) The operator started Reactor Coolant Pump 1A (RC-P-1A) to re-establish Reactor Coolant Flow.

NOTE: During the previous run of Reactor Coolant Pump 2B (RC-P-2B) due to the indication observed,it was thought that the pump might not have started. For this reason it was decided to watch the starting current for another 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 slowly decreased to less than 100 amperes.

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.

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| | |
|---|---|
| 04:17:17 (0817:54) | Reactor Coolant Makeup Pump A (MU-P-1A) tripped. |
| 04:17:22 (0817:59) | Reactor Coolant Makeup Pump C (MU-P-1C) tripped. No Reactor Coolant Makeup Pumps were operating. |
| 04:18:17 (0818:54) | The operator attempted to start Reactor Coolant Makeup Pump A (MU-P-1A). The pump would not start. |
| 04:21:53 (0822:30) Approximate | The operator started Reactor Coolant Makeup Pump B (MU-P-1B). |
| 04:26:59 (0827:36) Approximate | The operator started Reactor Coolant Makeup Pump C (MU-P-1C); it tripped, and was restarted. |
| 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. |
| 04:30:45 (0831:22) Approximate | The operator opened the Power Operated Emergency Main Steam Dump Valve (MS-V3A) to induce natural circulation in Steam Generator A. Steam Generator B was still isolated. |
| 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. |
| 05:18:00 (0918:37) | The operator closed the Electromatic Relief Block Valve (RC-V2) in an attempt to compress the reactor coolant and collapse the superheated steam/gas. |

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PLANT STATUS

All reactor coolant pumps (RC-P-1A, RC-P-2A, RC-P-1B and RC-P-2B) were stopped. A superheated steam/gas space was present in the upper vessel 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% of 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 Safeguard 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 collapse the superheated steam/gas space in the Reactor Coolant System. This first attempt lasted for approximately 2 hours.

05:20:00 (0920:37) Reactor Coolant System pressure started to increase from 1250 psig.

Approximate The increasing trend in Reactor Coolant System pressure continued

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for approximately 45 minutes until pressure stabilized at about 2050 psig (Figure 12).

05:54:00
(0954:37)
Approximate

The operator commenced filling Steam Generator A to 95% of the operating range to induce natural circulation. A Steam Generator indicated level reached 100% at 07:30:00 (1130:37) (Figure 39).

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 12). This was done after the operator observed no evidence of natural circulation while the Reactor Coolant System pressure was above 2000 psig. The reduction in Reactor Coolant System pressure was also done to approach conditions which would allow the Decay Heat Removal Pumps P1A and 1B (DH-P-1A and DH-P-1B) to be put into service.

08:11:26
(1212:03)

The Core Flood Tank A high level alarm was received. The level was 13.32 feet.

08:30:00
(1230:37)

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.

08:31:06
(1231:43)

The operator started Decay Heat Removal Pumps 1A and 1B (DH-P-1A and DH-P-1B) in preparation for placing the Decay Heat System in service.

08:54:56

Core Flood Tank A normal level alarm was received. The level was 13.13 feet.

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09:04:18 (1304:55) The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C).

09:49:43 (1350:20) A hydrogen detonation occurred in the Reactor Building. Hydrogen gas from the reactor coolant collected in the pressurizer and was vented through the Electromatic Relief Valve (RC-RV2) to the Reactor Coolant Drain Tank and then 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 and detonated.

09:49:44 (1350:21) Engineered Safeguards 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) and Reactor Building Spray Pumps A and B (BS-P-1A and BS-P-1B) started automatically.

09:49:50 (1350:27) Reactor Building Spray Valves (BS-V1A and BS-V1B) opened.

09:49:58 (1350:35) Reactor Coolant Pumps 1A and 1B (RC-P-1A and RC-P-1B) inlet air temperature high alarms annunciated and Pressurizer Safety Valves (RC-R1A and RC-R1B) discharge line temperature high alarms annunciated.

09:50:24 (1351:01) The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C).

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.

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09:56:58 (1357:35) The operator stopped Decay Heat Pumps A and B (DH-P-1A and DH-P-1B). These pumps were run in anticipation of using the Decay Heat System. The Reactor Coolant System pressure was never low enough to use the Decay Heat System.

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/gas space in Loop A hot leg collapsing.

10:31:25 (1432:02) The operator started Reactor Coolant Makeup Pump C (MU-P-1C). Reactor coolant pressure was approximately 440 psig.

10:34:29 (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 the steam/gas space in Loop A hot leg re-established itself (Figures 22 and 27).

10:35:55 (1436:32) The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C).

10:39:29 (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 Injection to Loop A hot leg was re-established. The Loop A hot leg steam did not return.

11:06:00 (1406:37) Approximate Pressurizer level started decreasing from 370 inches to 180 inches over a period of 18 minutes (Figure 30).

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 natural circulation in Loop A (Figure 22).

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| | |
|---|---|
| 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. |
| 11:24:00 (1524:37) | Pressurizer level stopped decreasing at 180 inches and started increasing, going off scale during the next hour (Figure 30). |
| 11:28:12 (1528:49) | The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C). |
| 11:32:37 (1533:14) | The operator started Reactor Coolant Makeup Pump C (MU-P-1C) to raise the pressurizer level. |
| 11:35:48 (1536:25) | The operator stopped Reactor Coolant Makeup Pump C (MU-P-1C). |
| 11:36:00 (1536:37) Approximate | The operator commenced 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). |
| 12:48:00 (1648:00) Approximate | Pressurizer level indication came on scale (Figure 30). |
| 13:02:23 (1703:00) | The operator started Condenser Vacuum Pump 1C (VA-P-1C) in an attempt to re-establish vacuum. The auxiliary boiler had been returned to service and was supplying gland sealing steam to the main turbine. |
| 13:13:10 (1713:47) | The operator started Condenser Vacuum Pump 1A (VA-P-1A). |

PLANT STATUS

All reactor coolant pumps (RC-P-1A, RC-P-2A, RC-P-1B and RC-P-2B) were stopped. Superheated steam/gas existed in the vessel head and Loop B hot leg. The Loop A hot leg steam/gas 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.

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|---|---|
| 13:15:00 (1715:37) Approximate | The operator closed the Electomatic Relief Block Valve (RC-V2) in an attempt to collapse the remaining steam/gas voids by going to a high Reactor Coolant System pressure (Figure 12). |
| 13:23:04 (1723:41) | The operator started Reactor Coolant Makeup Pump C (MU-P-1C) to assist in increasing Reactor Coolant System pressure. |
| 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. |
| 14:54:00 (1854:37) Approximate | Reactor Coolant System pressure reached 2350 psig (Figure 12). |

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15:32:42 (1933:19) 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).

15:49:08 (1949:45) Reactor Coolant System Loop B hot leg temperature decreased to within the indication range of 572.6 F (Figure 27).

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 290 F and eventually stabilized at 250 F (Figure 22).

PLANT STATUS

22:15:00 (0215:37) Approximate Reactor Coolant System and Steam Generator conditions were:

Reactor Coolant System pressure was 1165 psig.

Pressurizer Temperature was 551F (pressurizer heaters maintaining temperature).

Pressurizer level indication was 397 inches.

Steam Generator A was steaming to the Main Condenser.

Steam Generator B was isolated.

Reactor Coolant Makeup Pump B (MU-P-1B) was operating to supply Reactor Coolant Pump seal injection flow.

Reactor Coolant System Loop A cold leg temperature was 256.4F.

Reactor Coolant System Loop B cold leg temperature was 252.4F.

Reactor Coolant System Loop A hot leg temperature was off scale low, i.e., less than 520.0F.

Reactor Coolant System Loop B hot leg temperature = off scale low, i.e., less than 520.0F.

230 168

ANNOTATED SEQUENCE OF EVENTS

LIST OF FIGURES



230 169

# ANNOTATED SEQUENCE OF EVENTS

(CONTINUED)



230 170

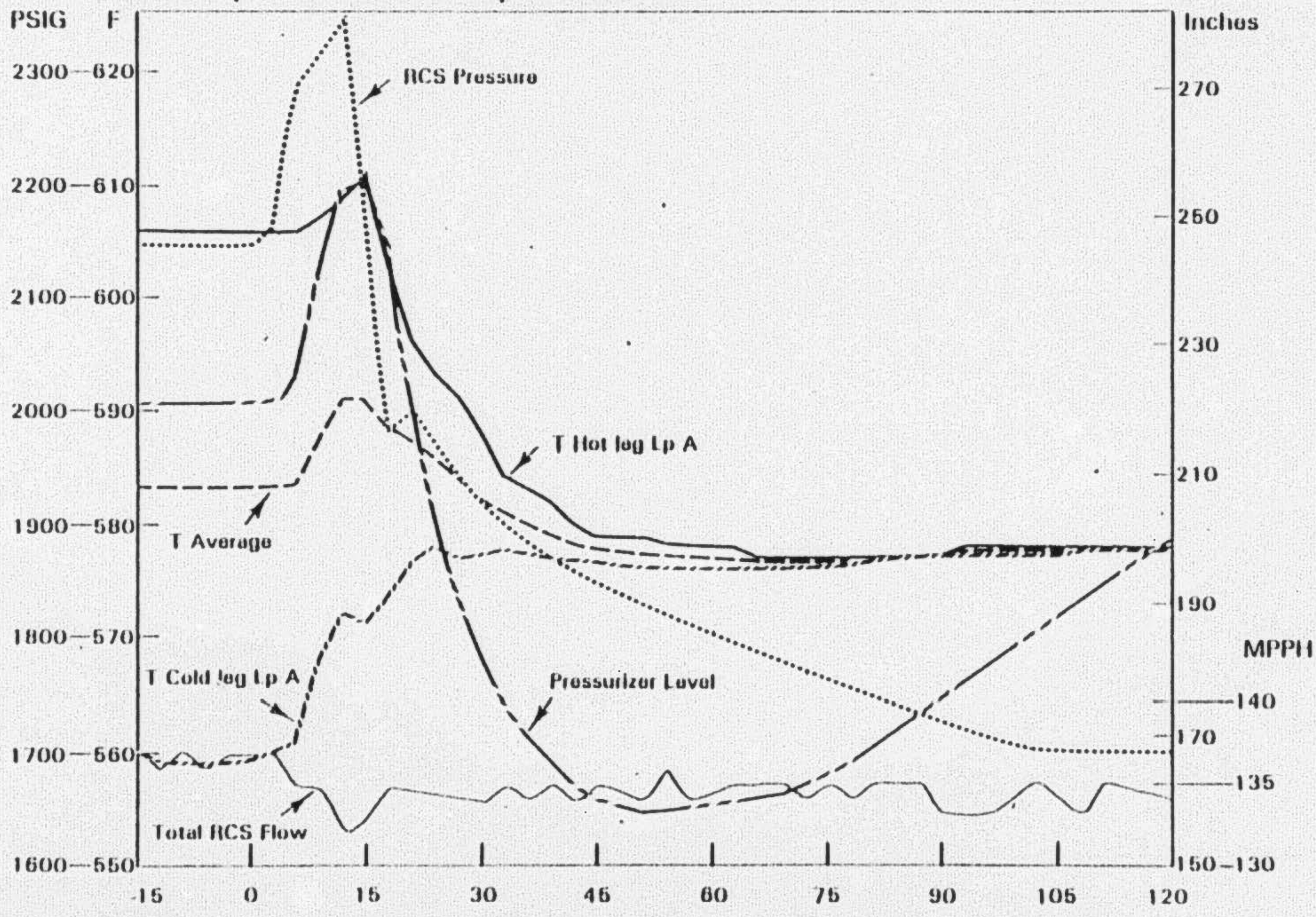
## ANNOTATED SEQUENCE OF EVENTS

(CONTINUED)



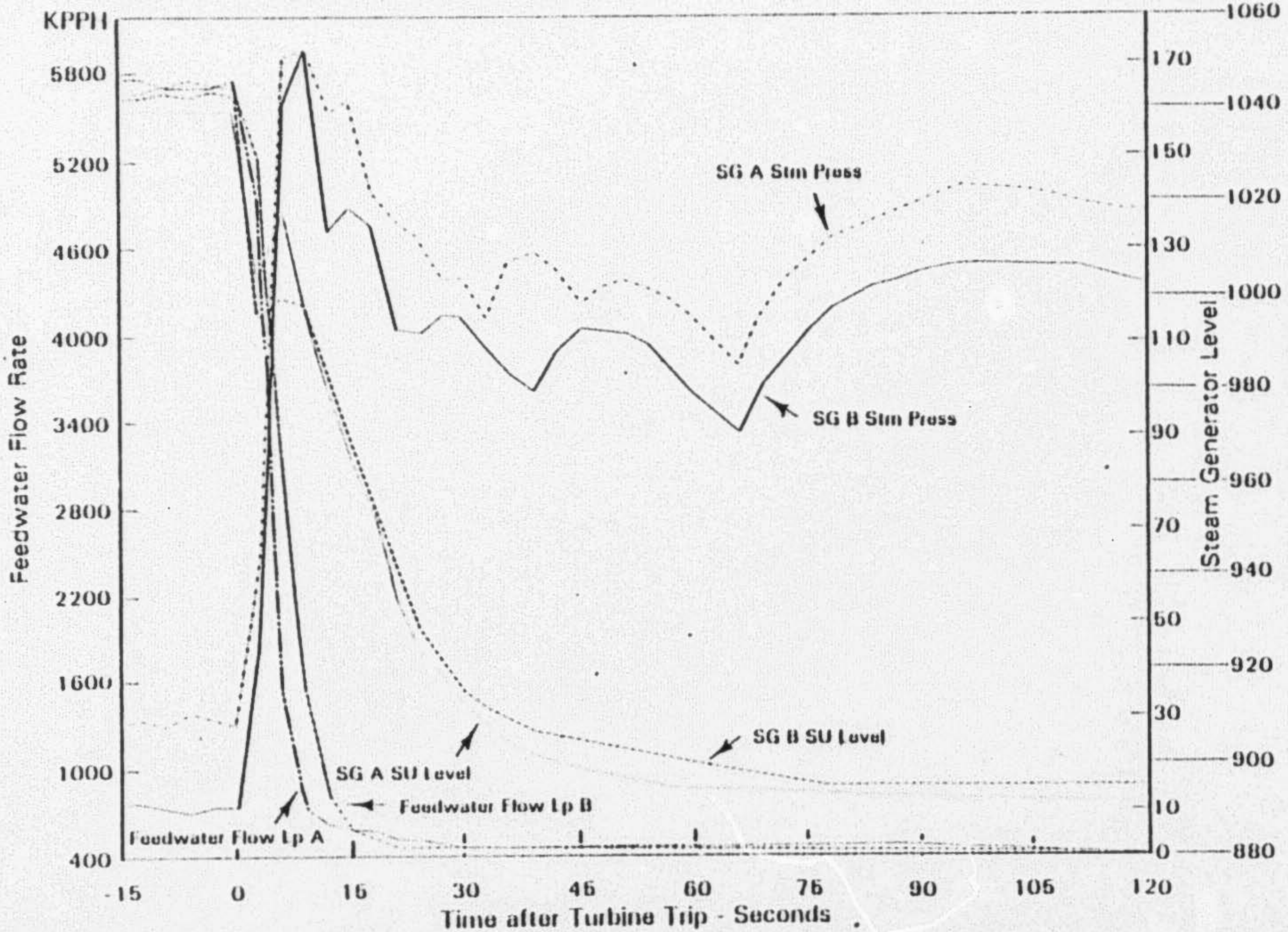
230 171

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



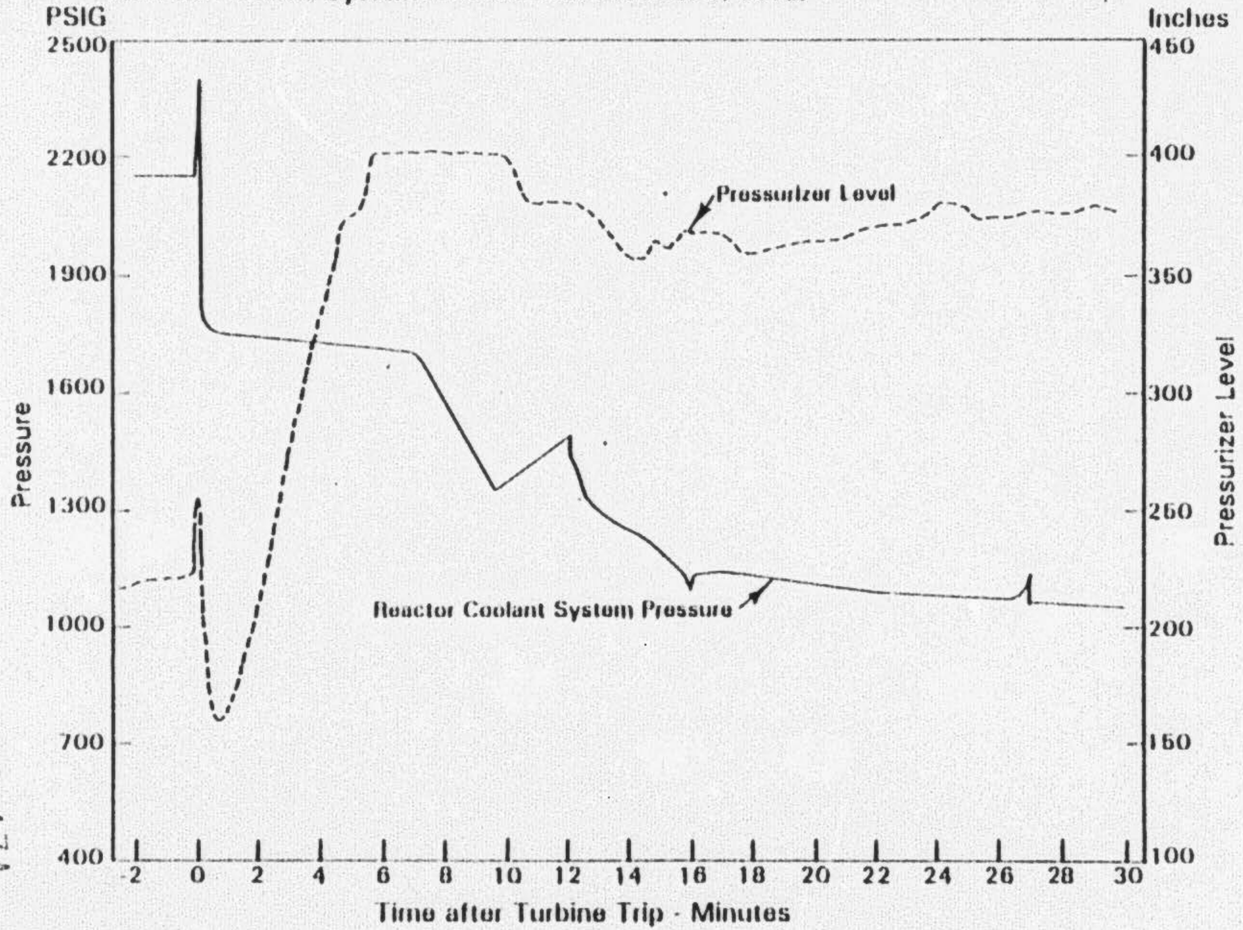

230 172

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



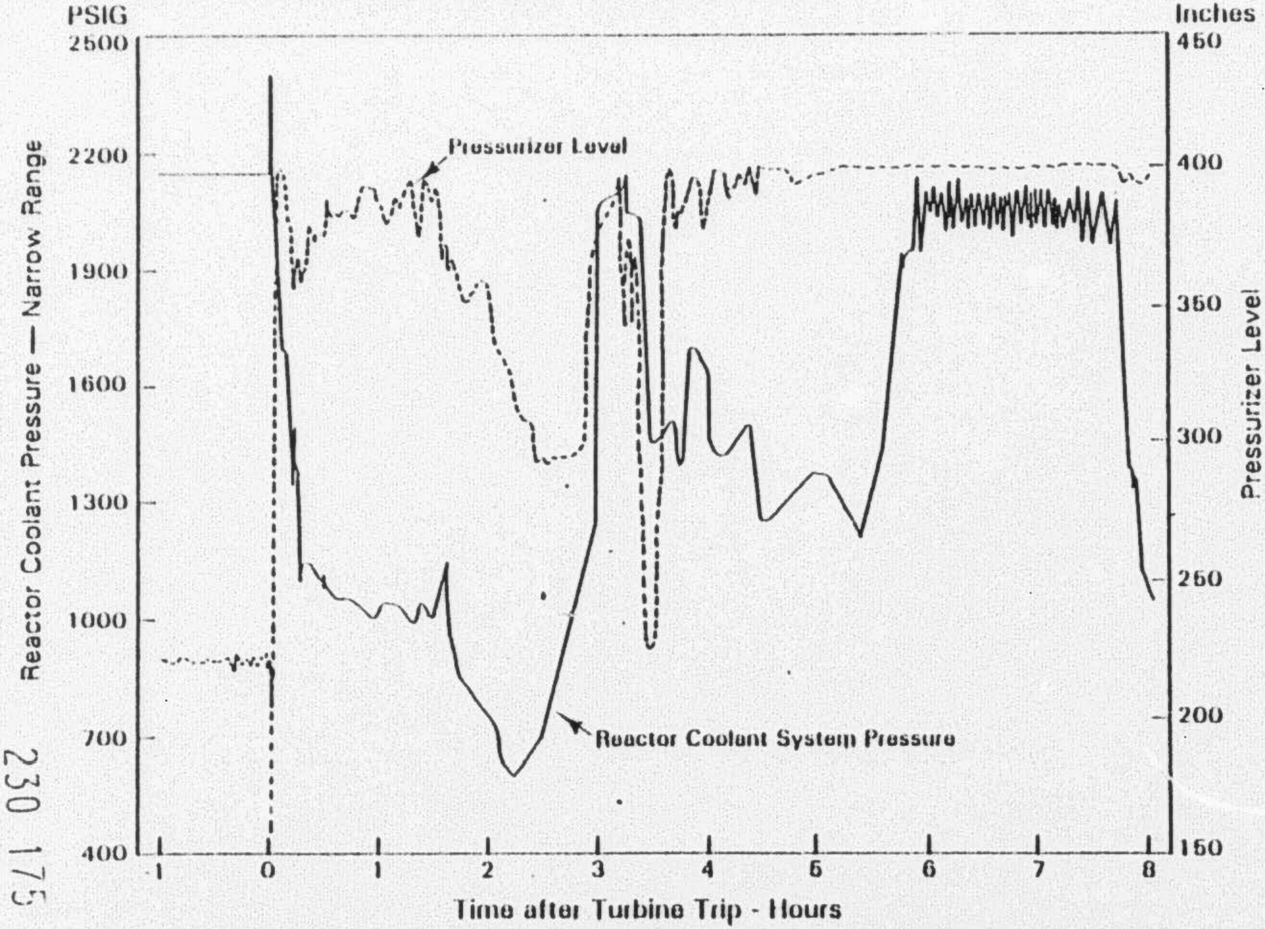

230 173

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



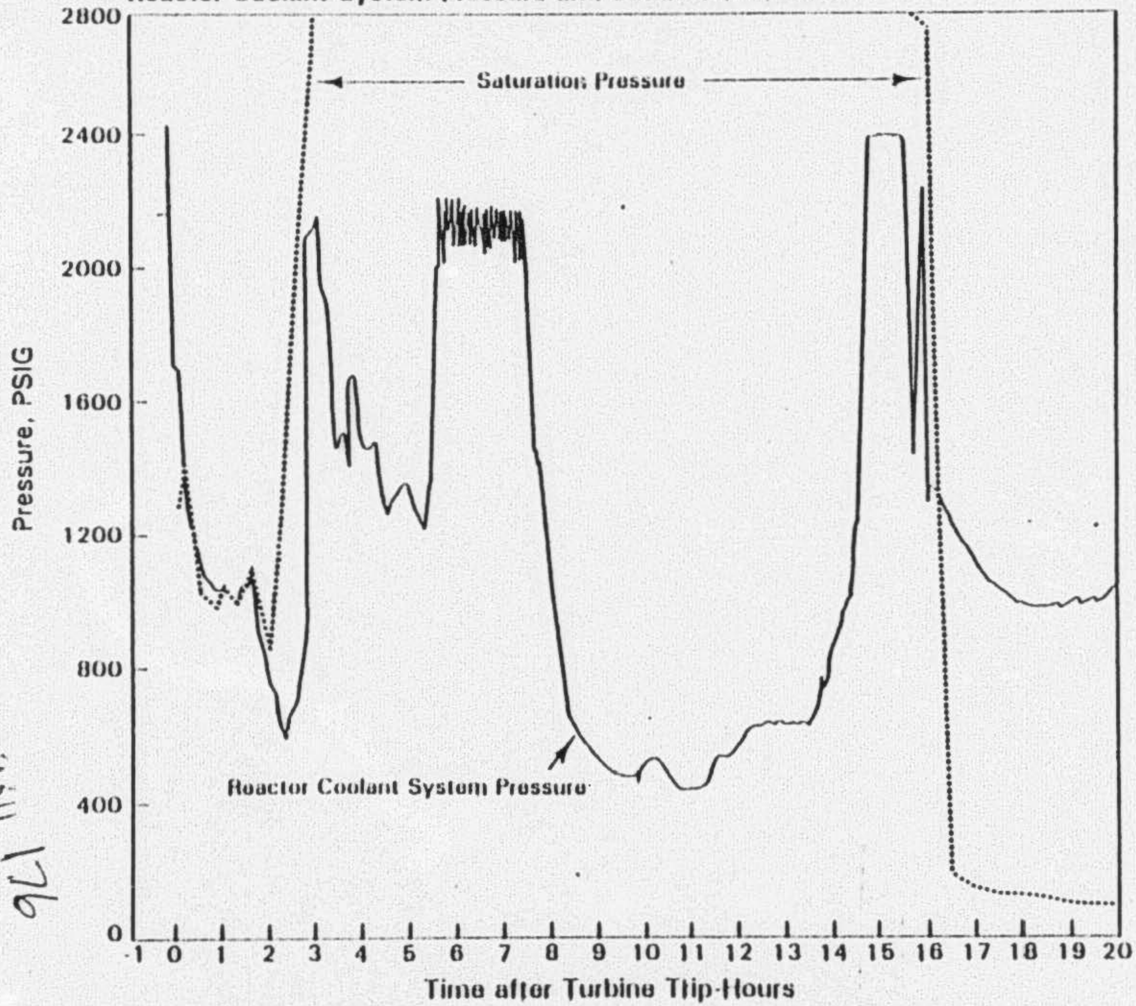

230 174

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



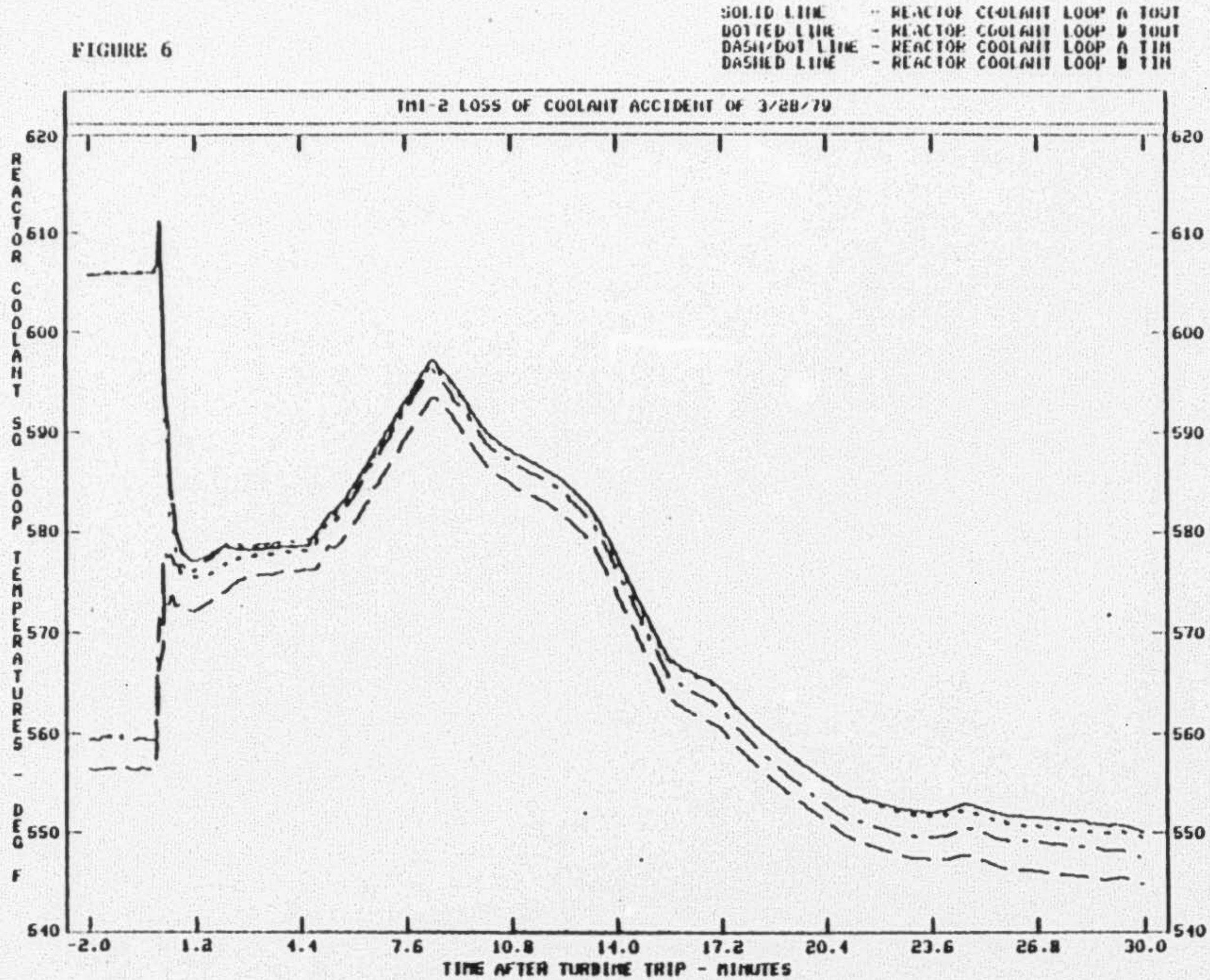

230 175

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




250 176

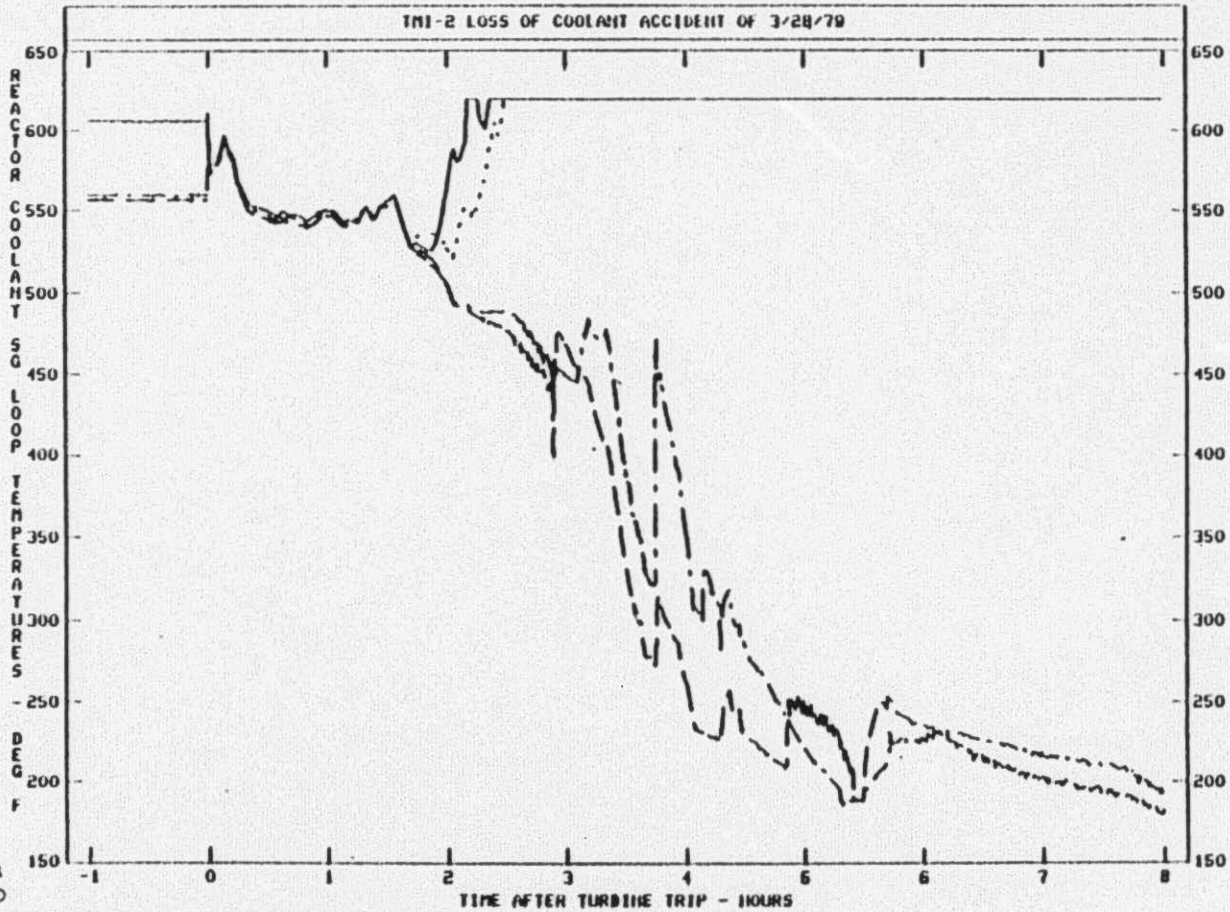
FIGURE 6




230 177

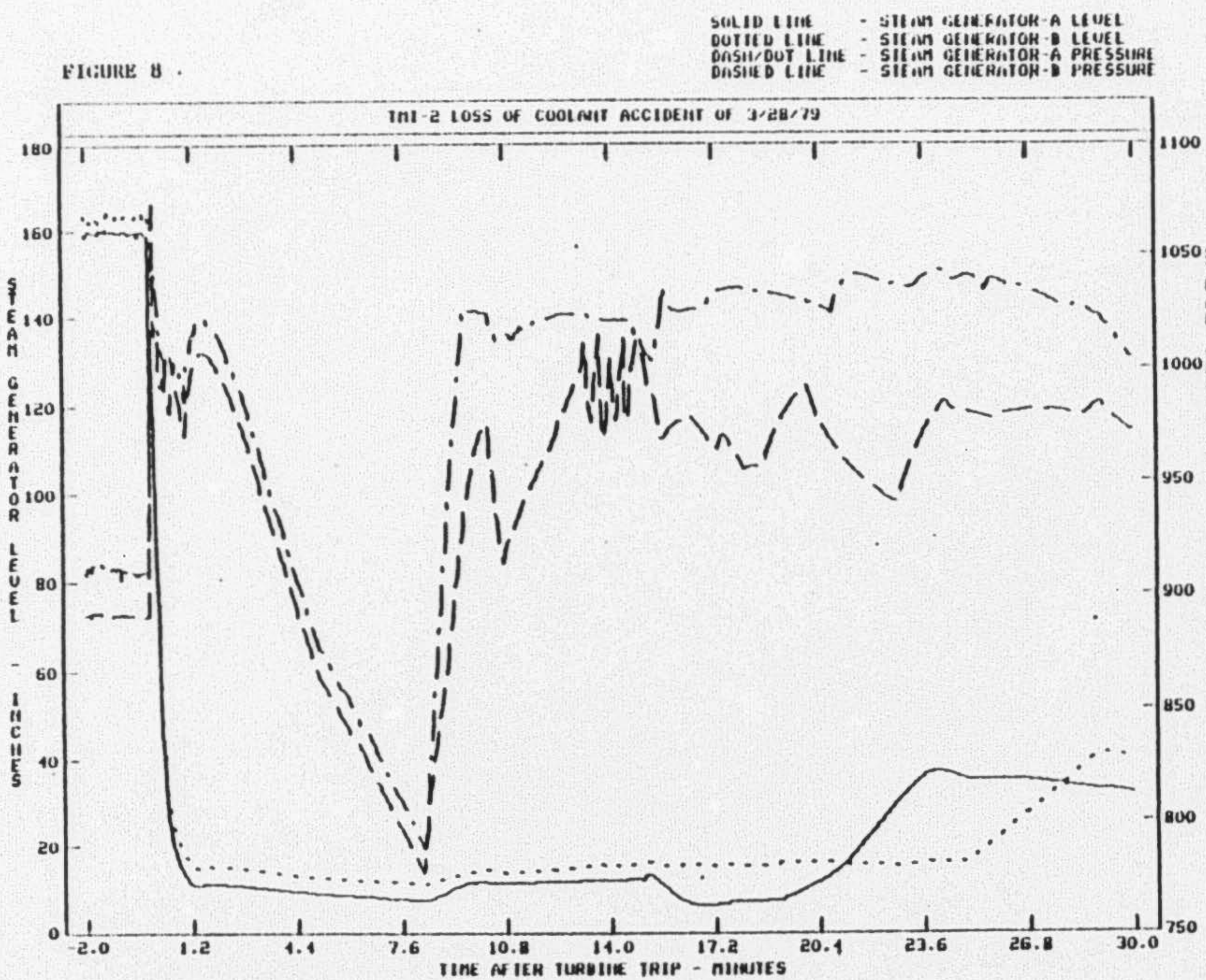
FIGURE 7

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



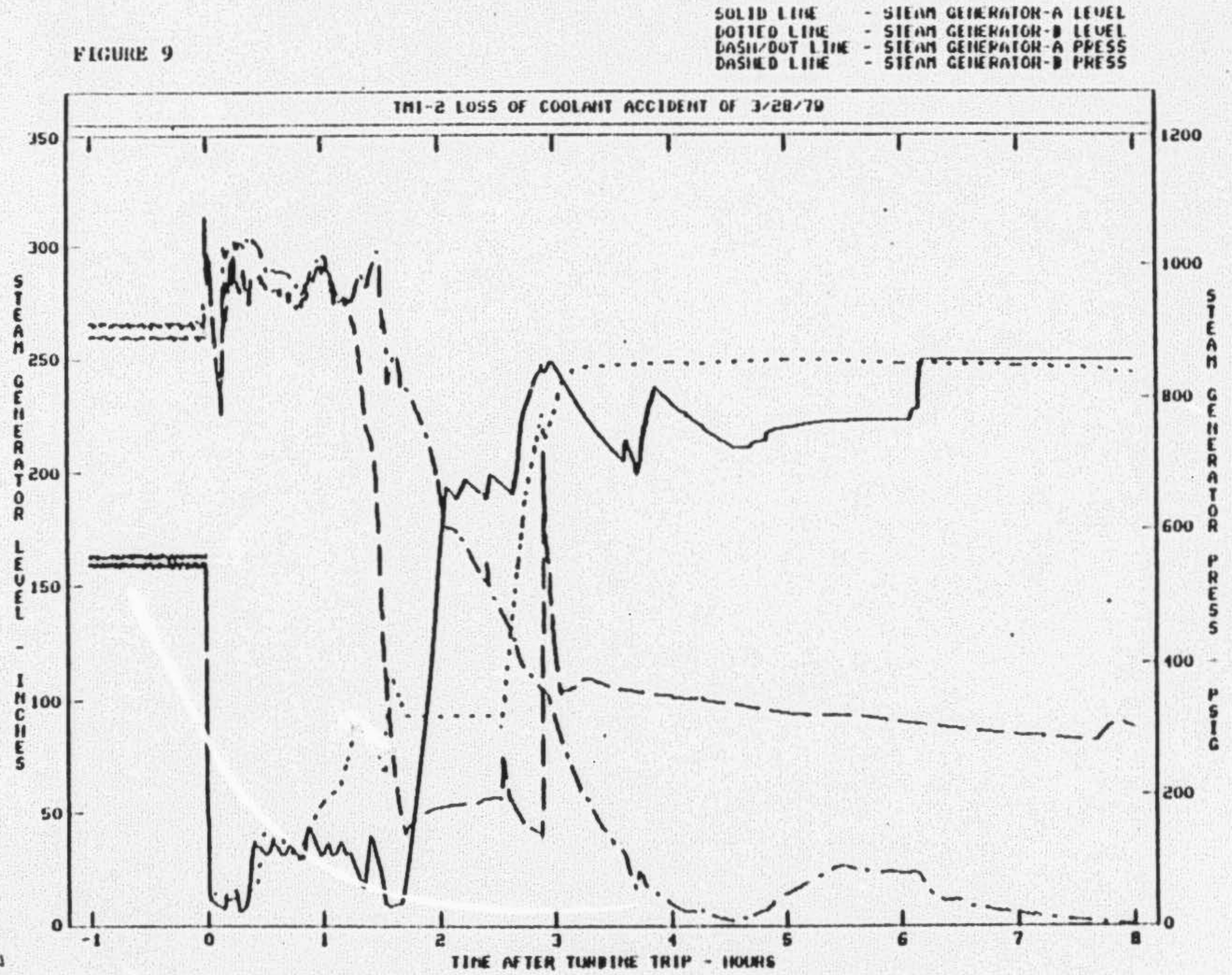

230 178

FIGURE 8 .



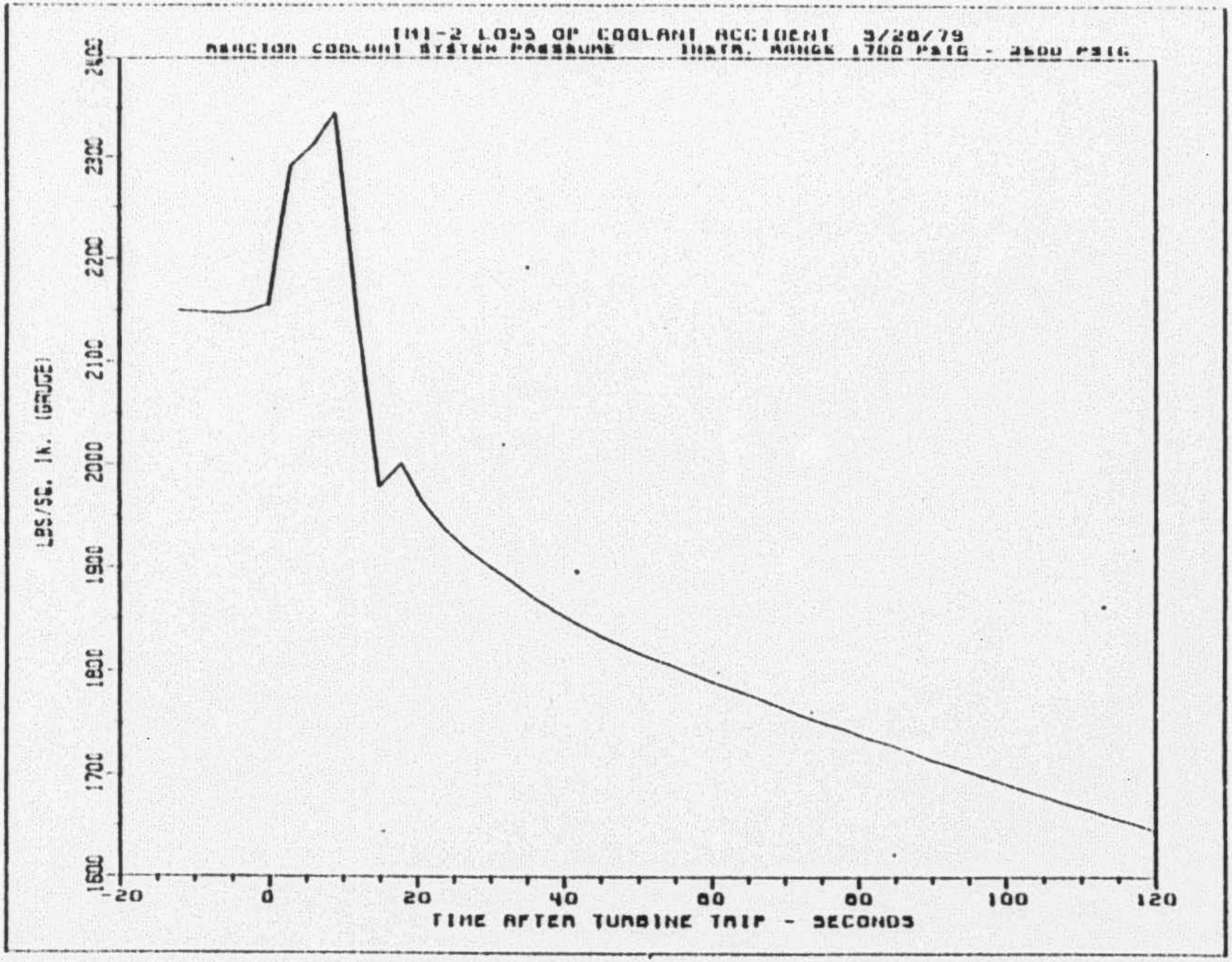

230 179

FIGURE 9



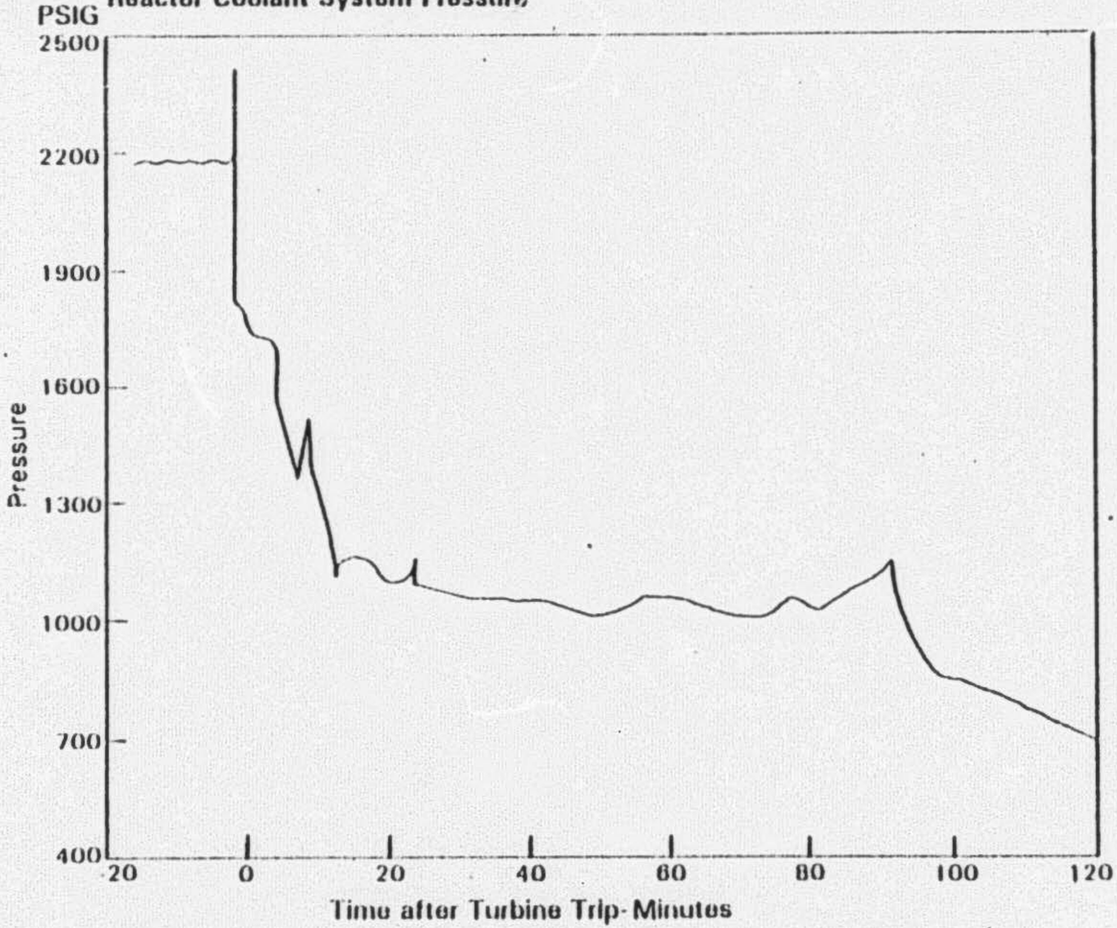

230 180

FIGURE 10



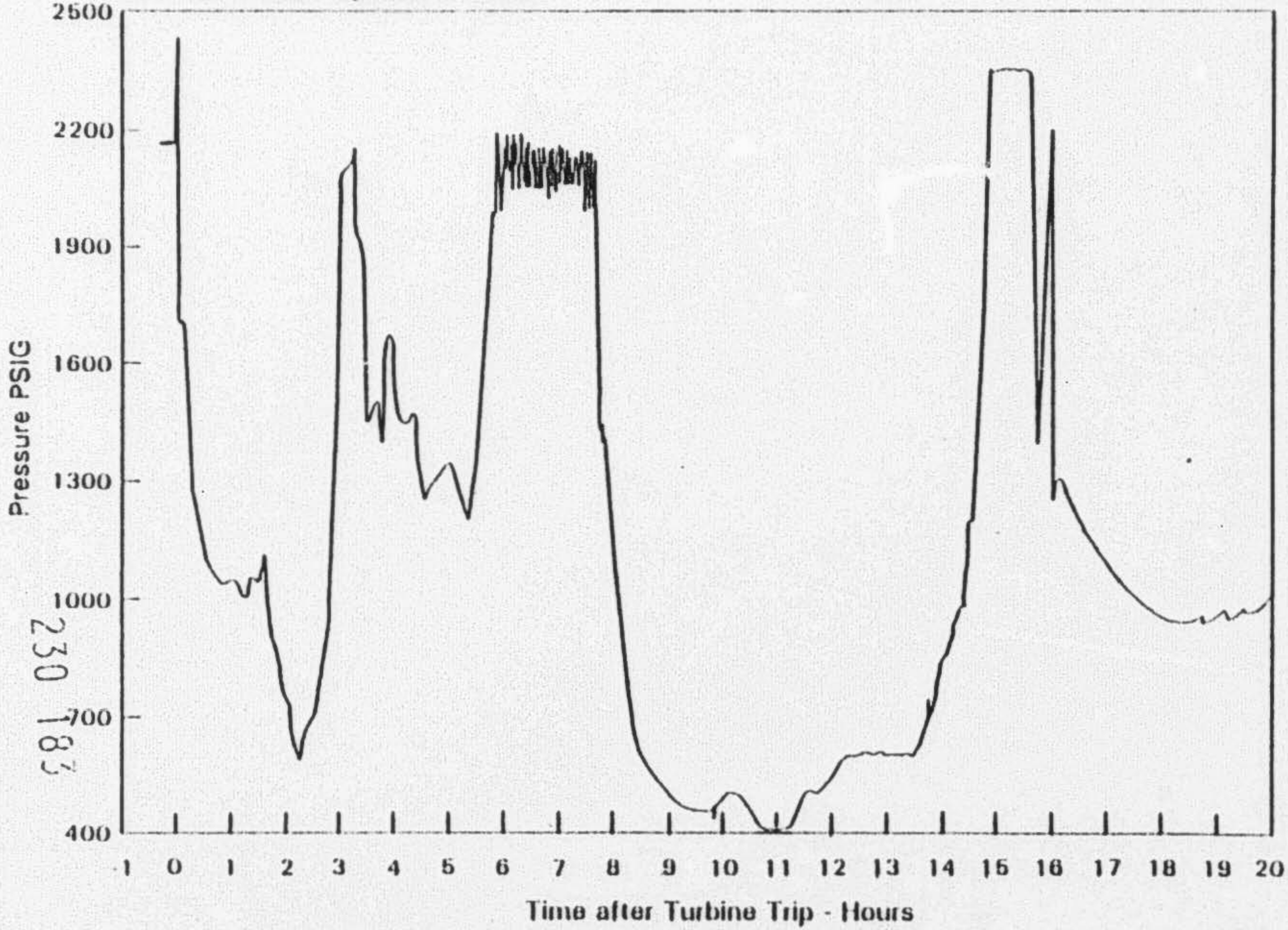

230 181

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



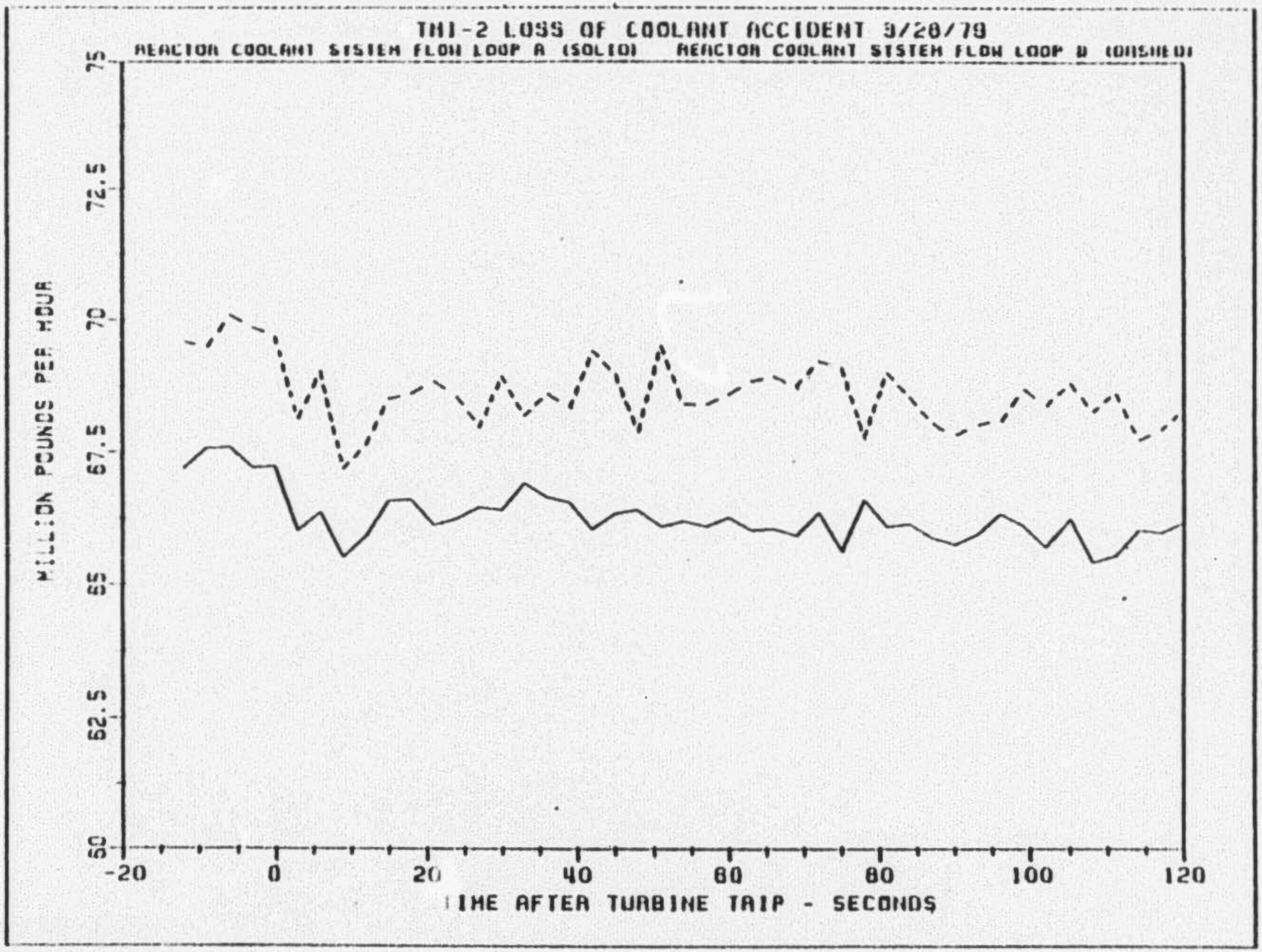

230 182

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



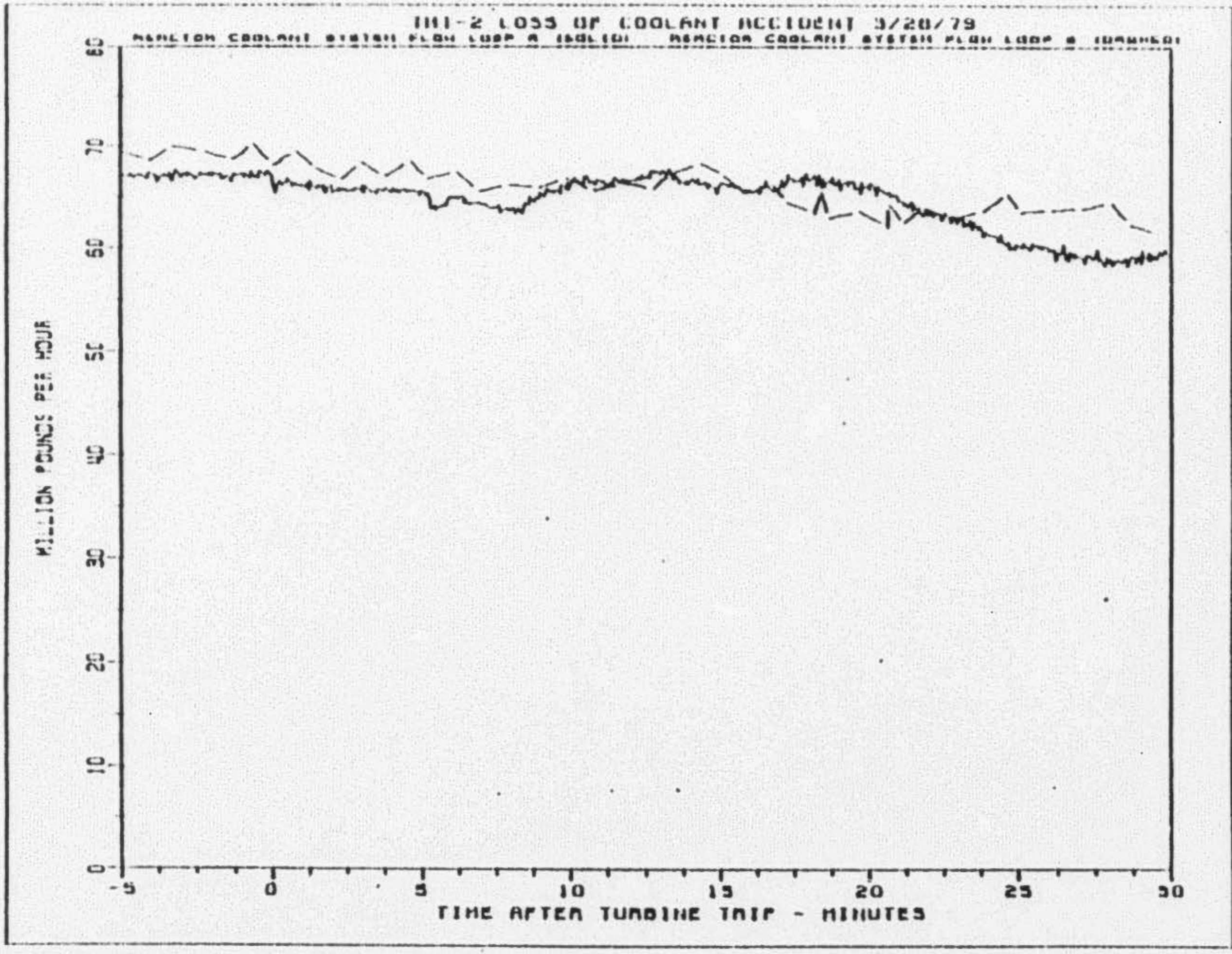

230 183

FIGURE 13



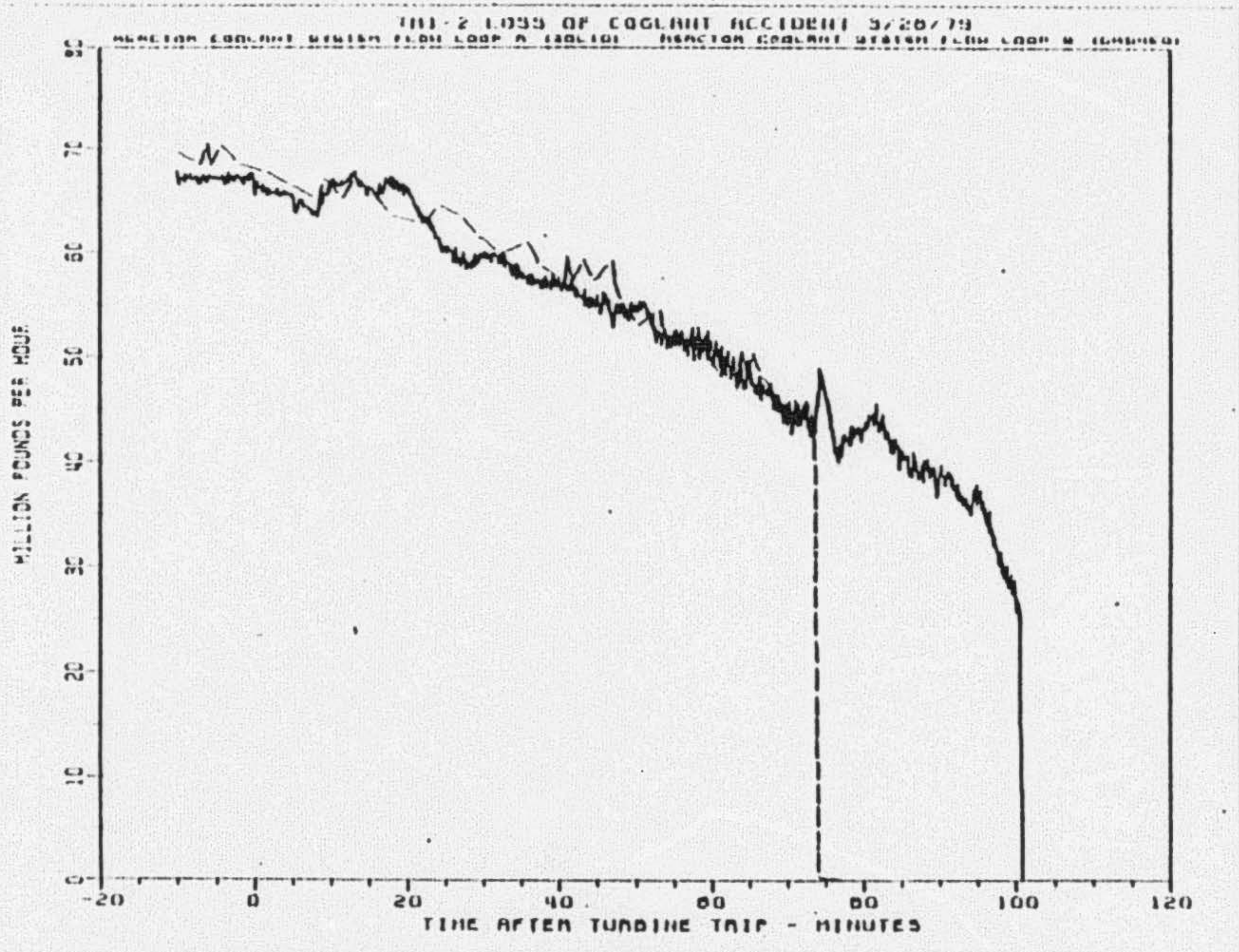

250 184

FIGURE 14



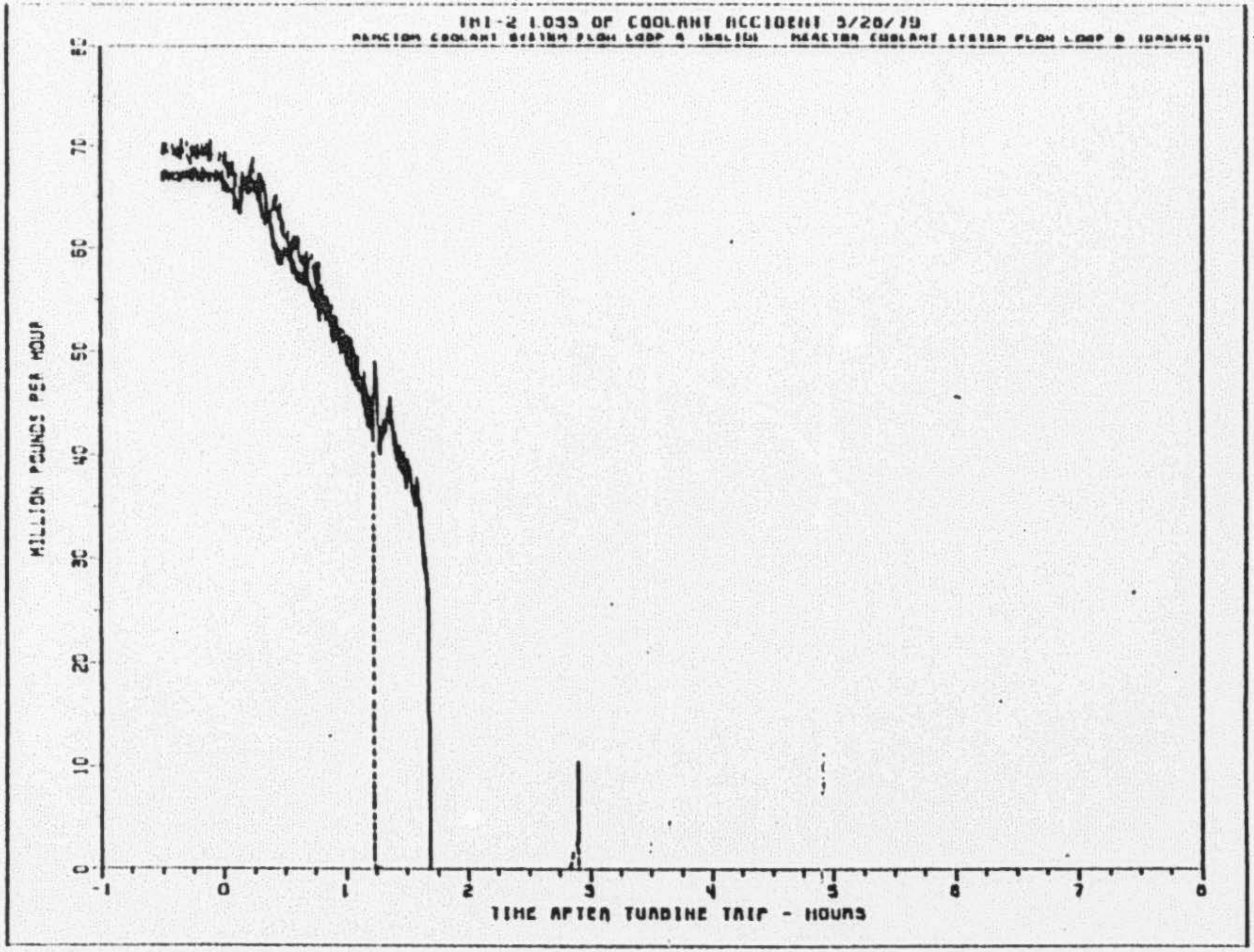

230 185

FIGURE 15



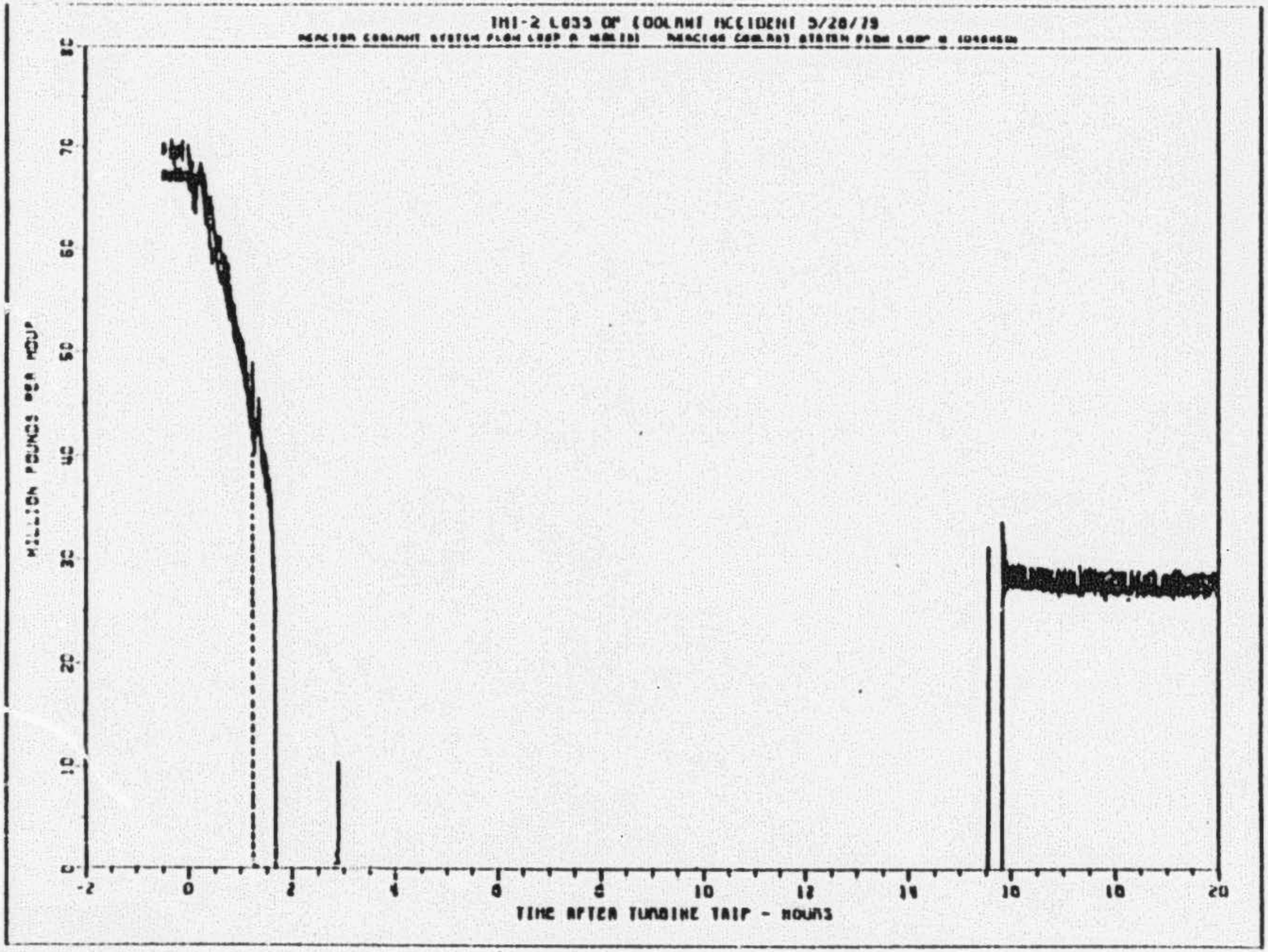

230 186

FIGURE 16



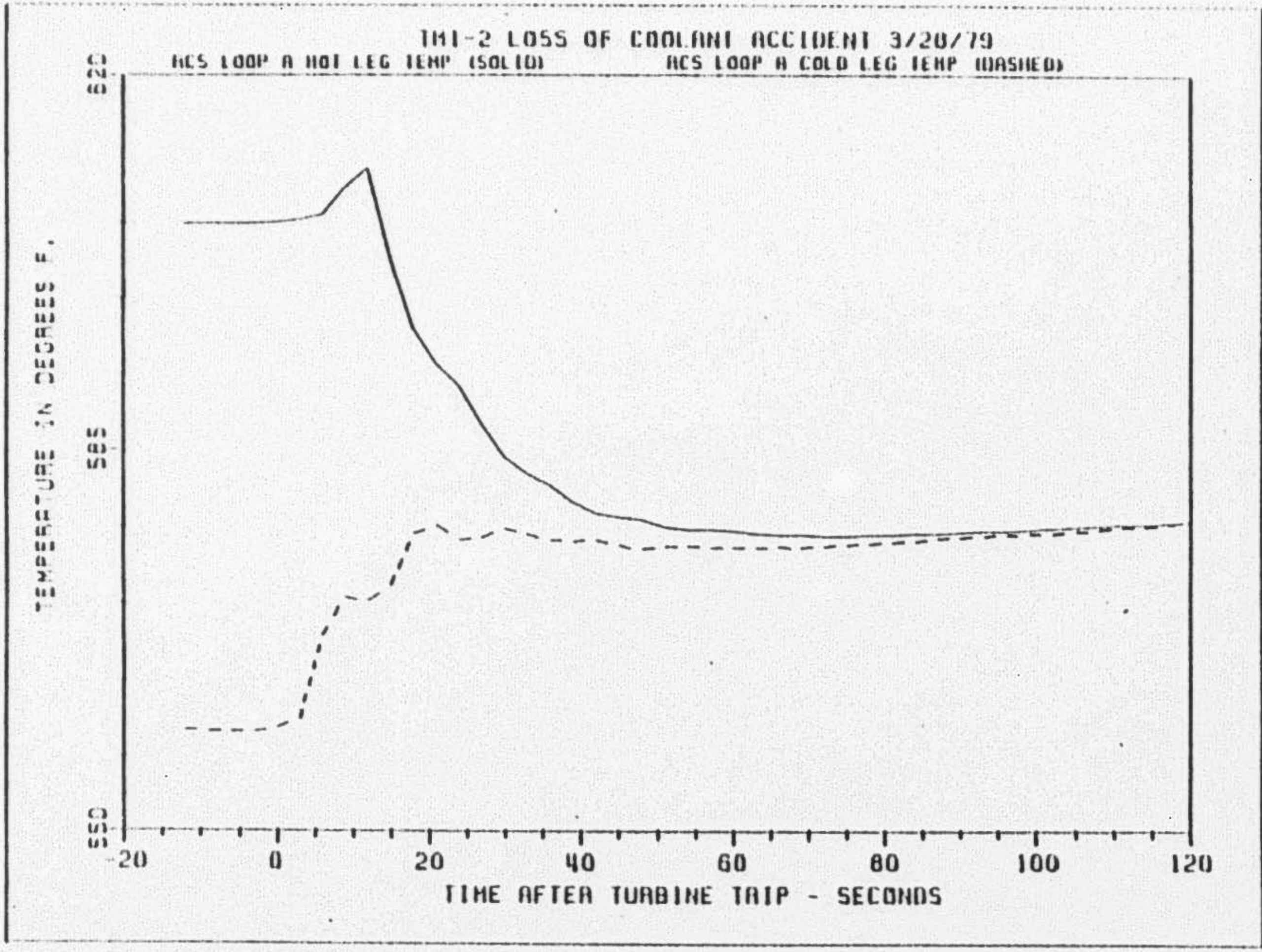

230 187

FIGURE 17



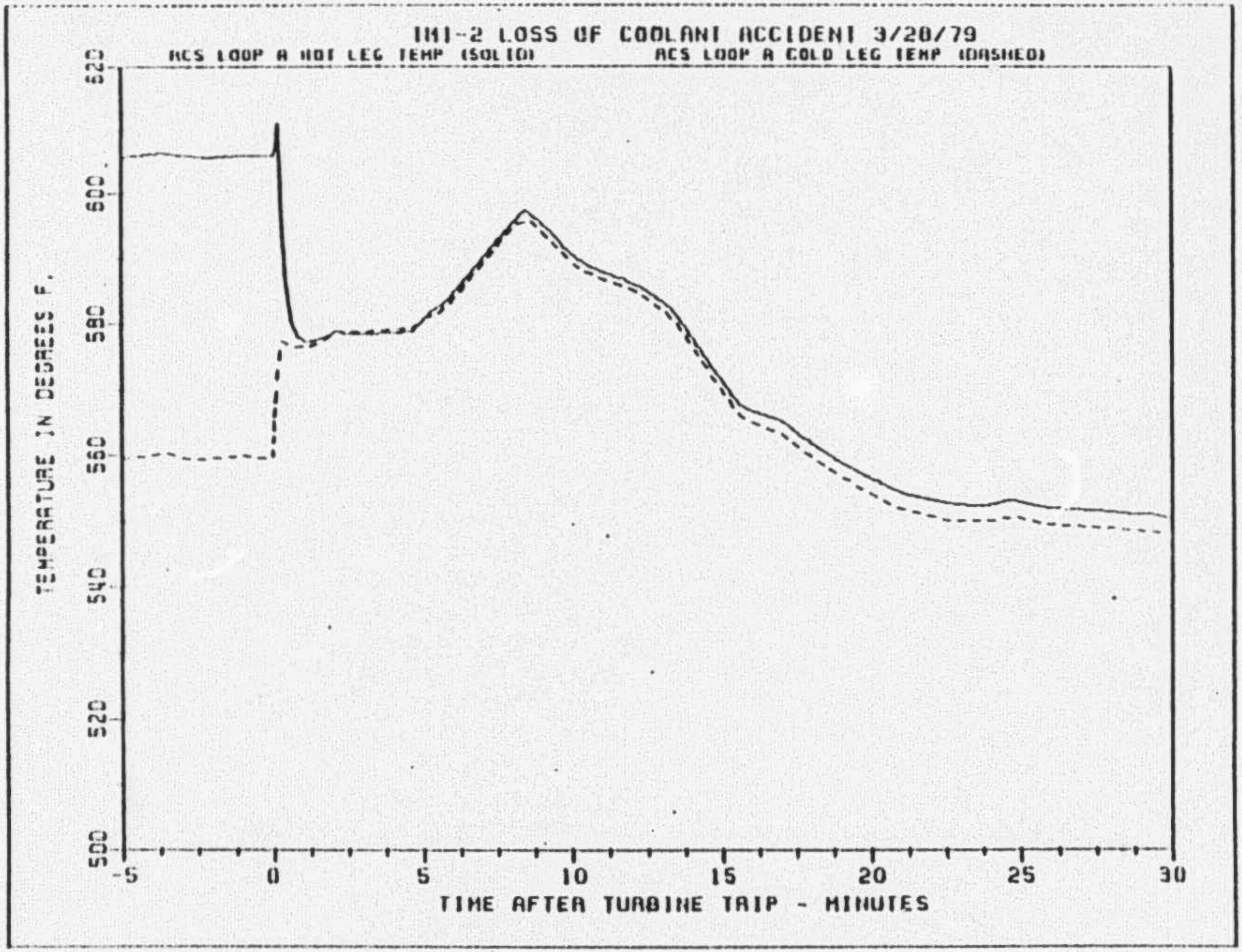

230 188

FIGURE 18

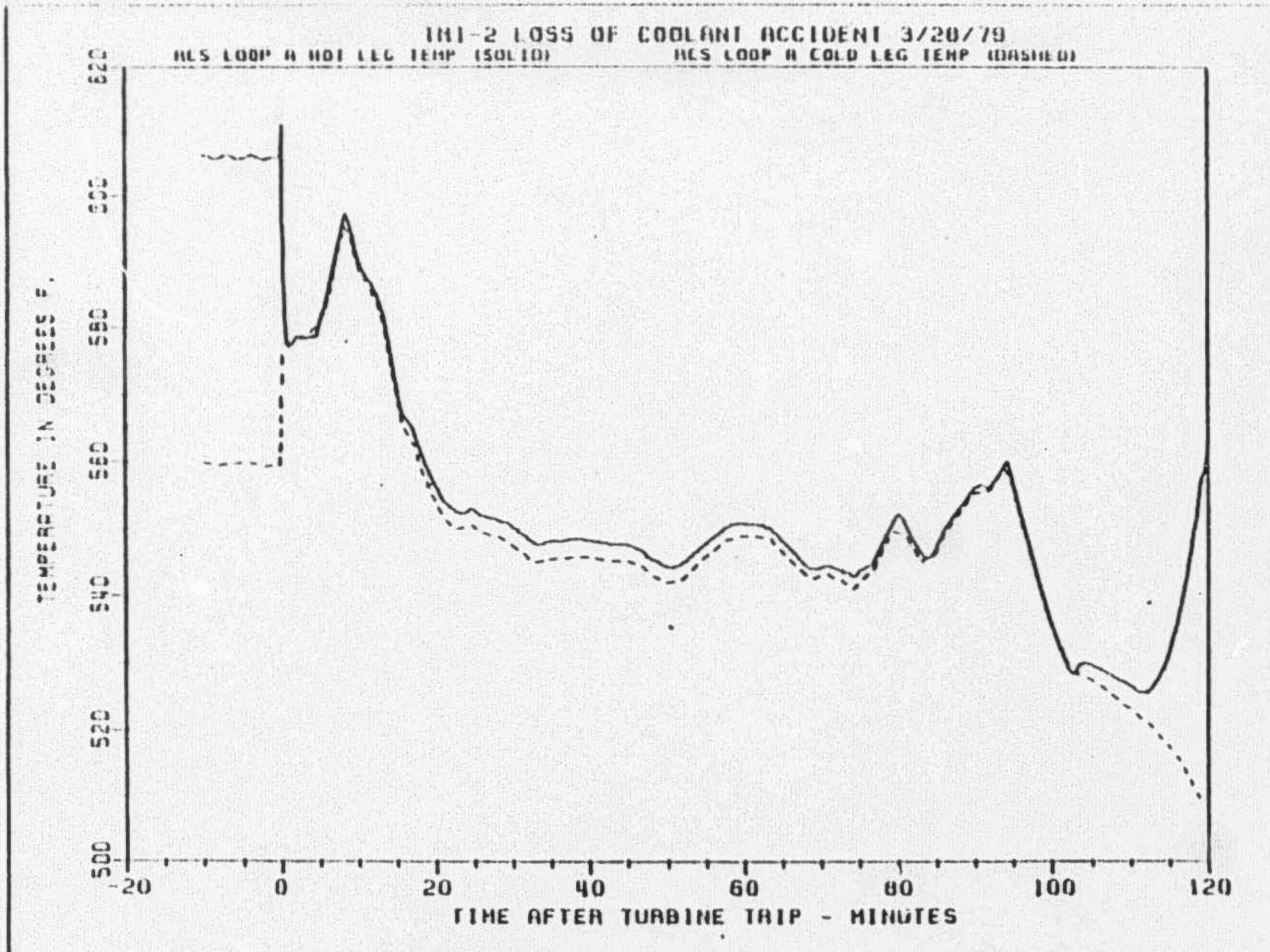



230 189

FIGURE 19

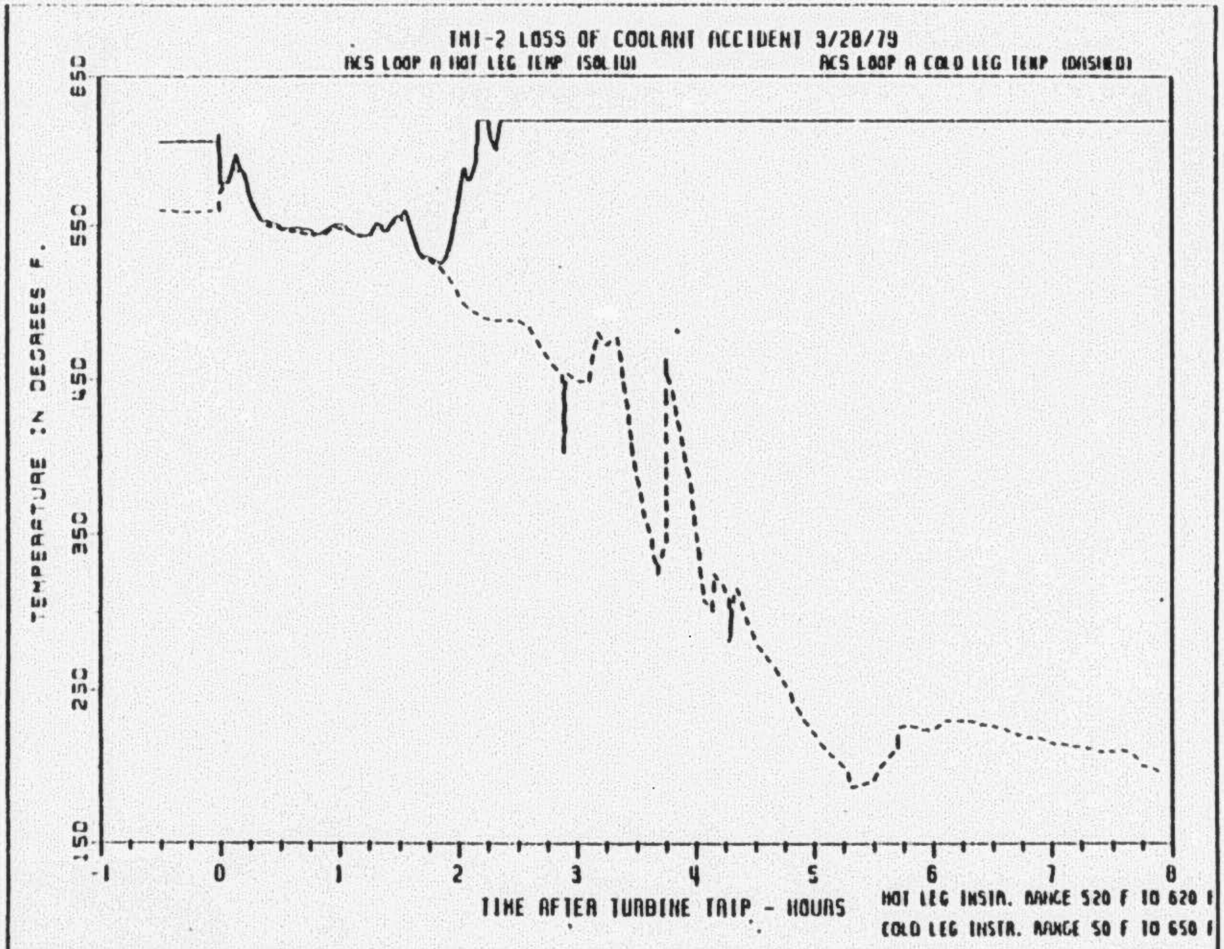



230 190


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

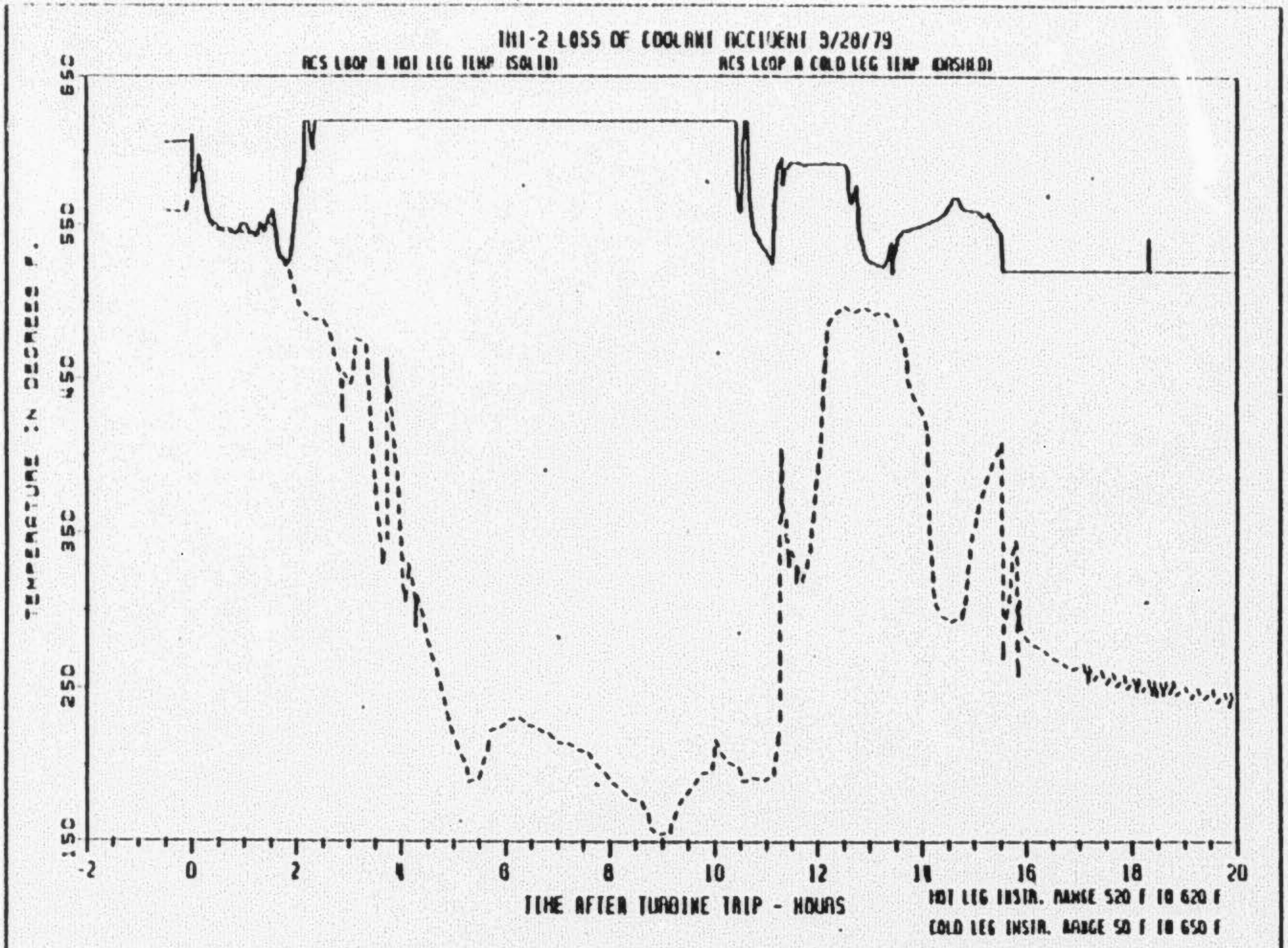
230 191

FIGURE 21



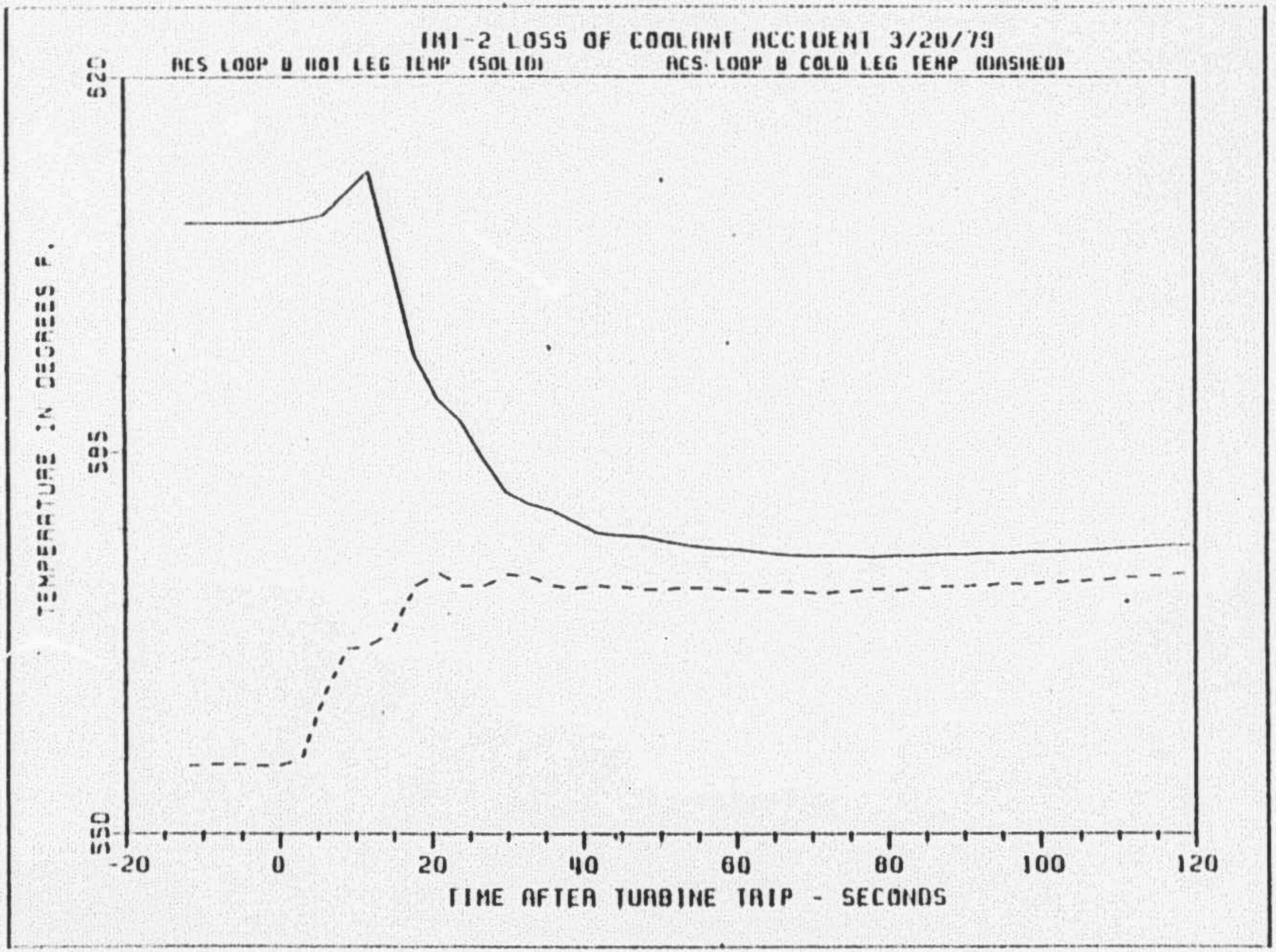

230 192

FIGURE 22



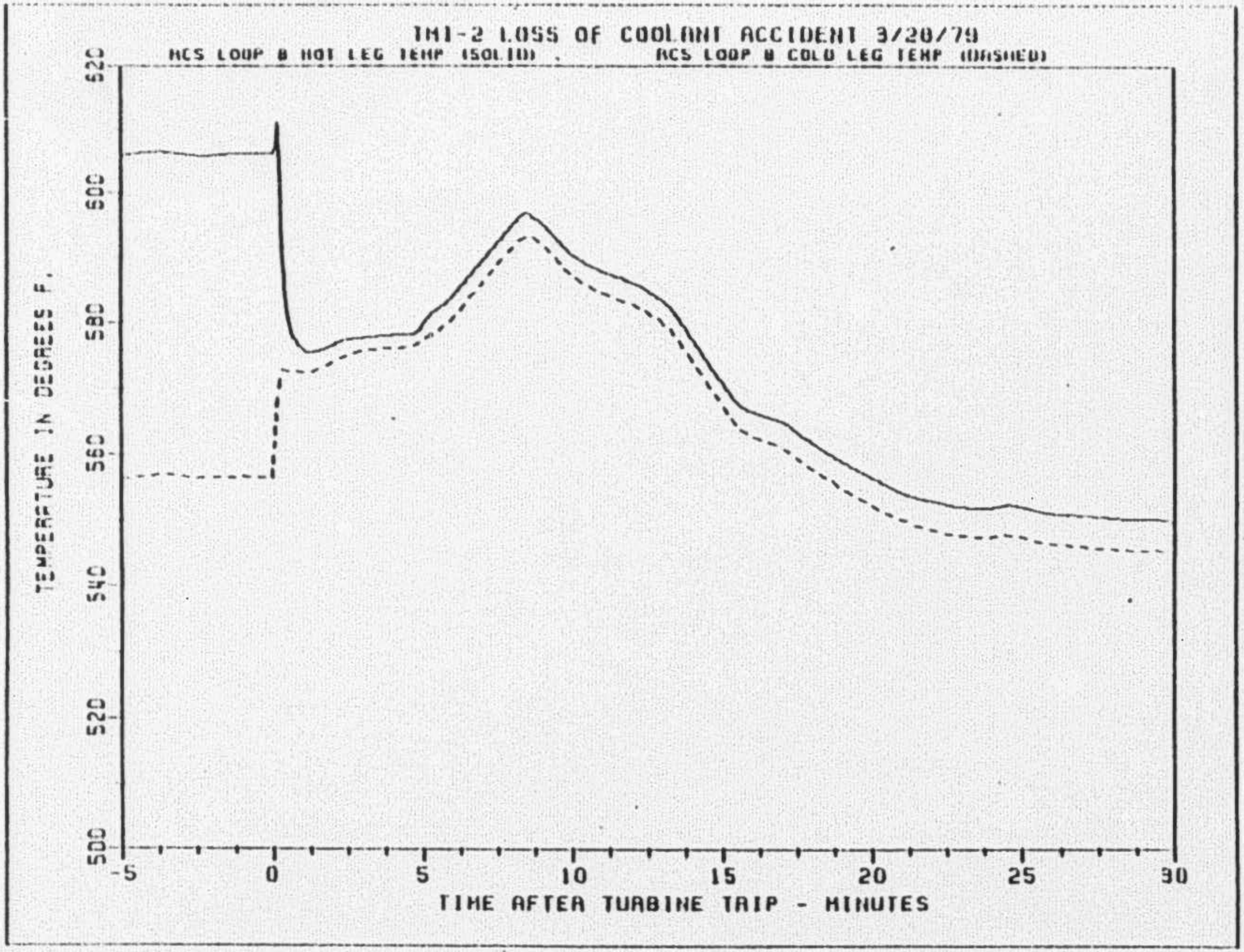

230 193

FIGURE 23




230 194

FIGURE 24




230 195

FIGURE 25

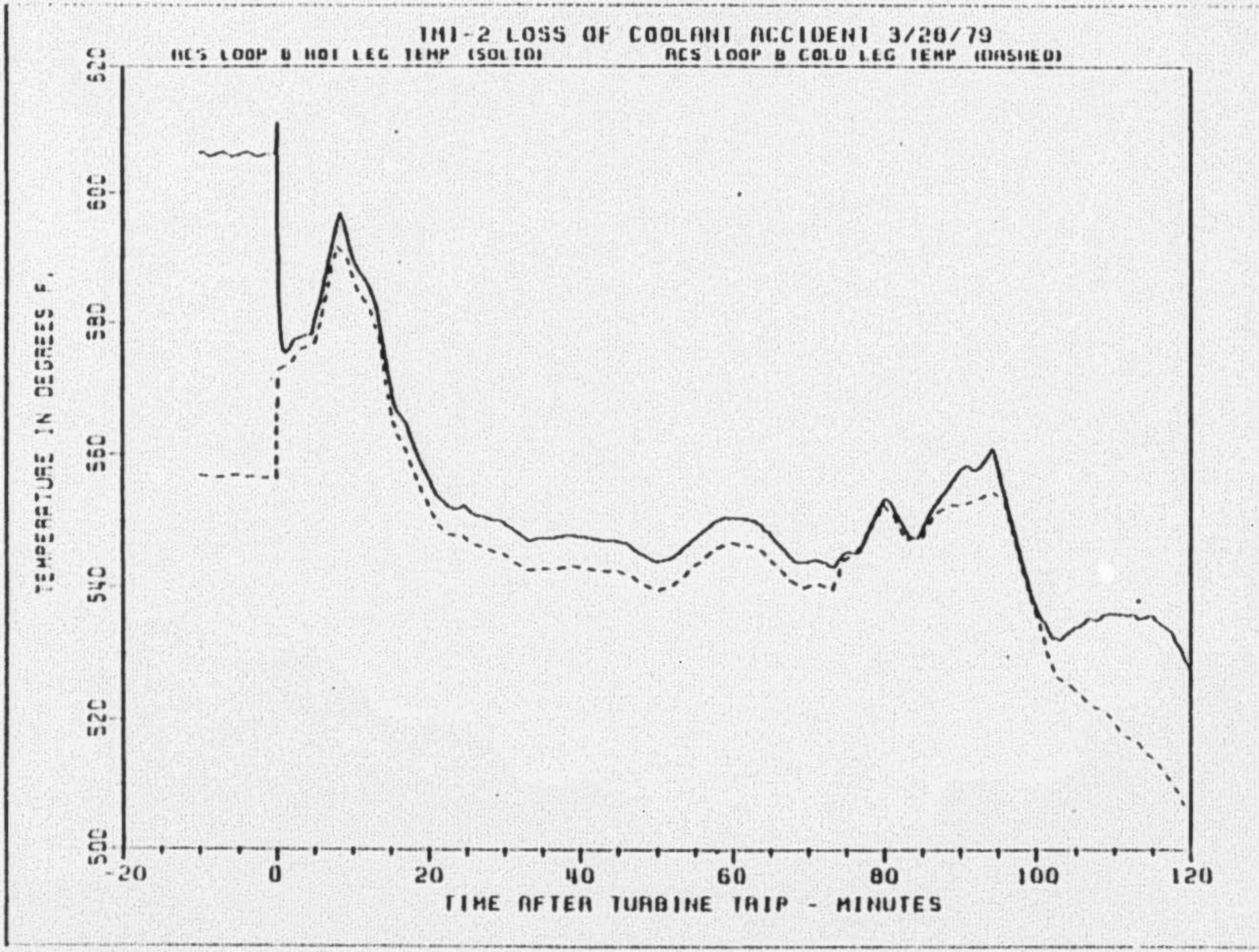
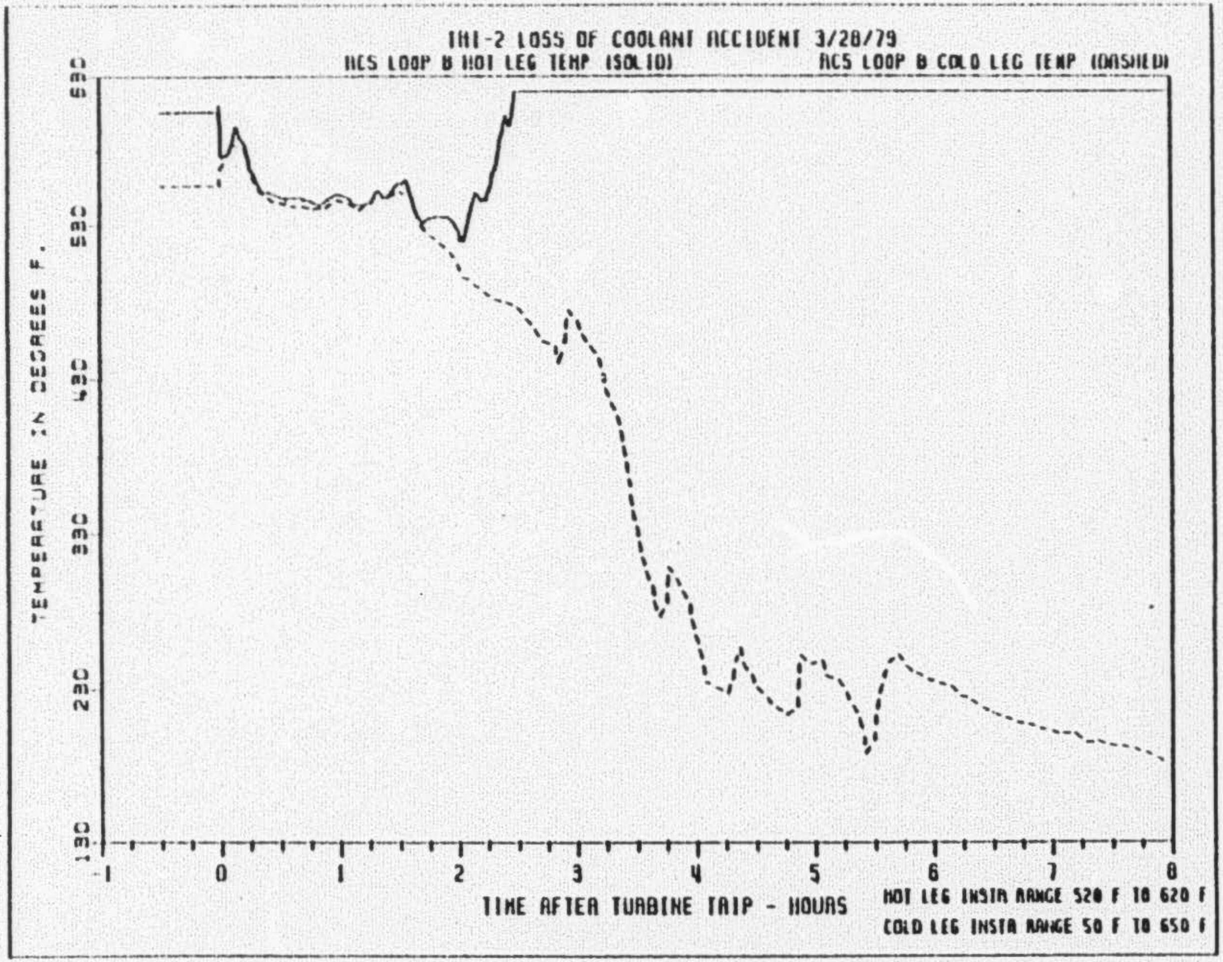
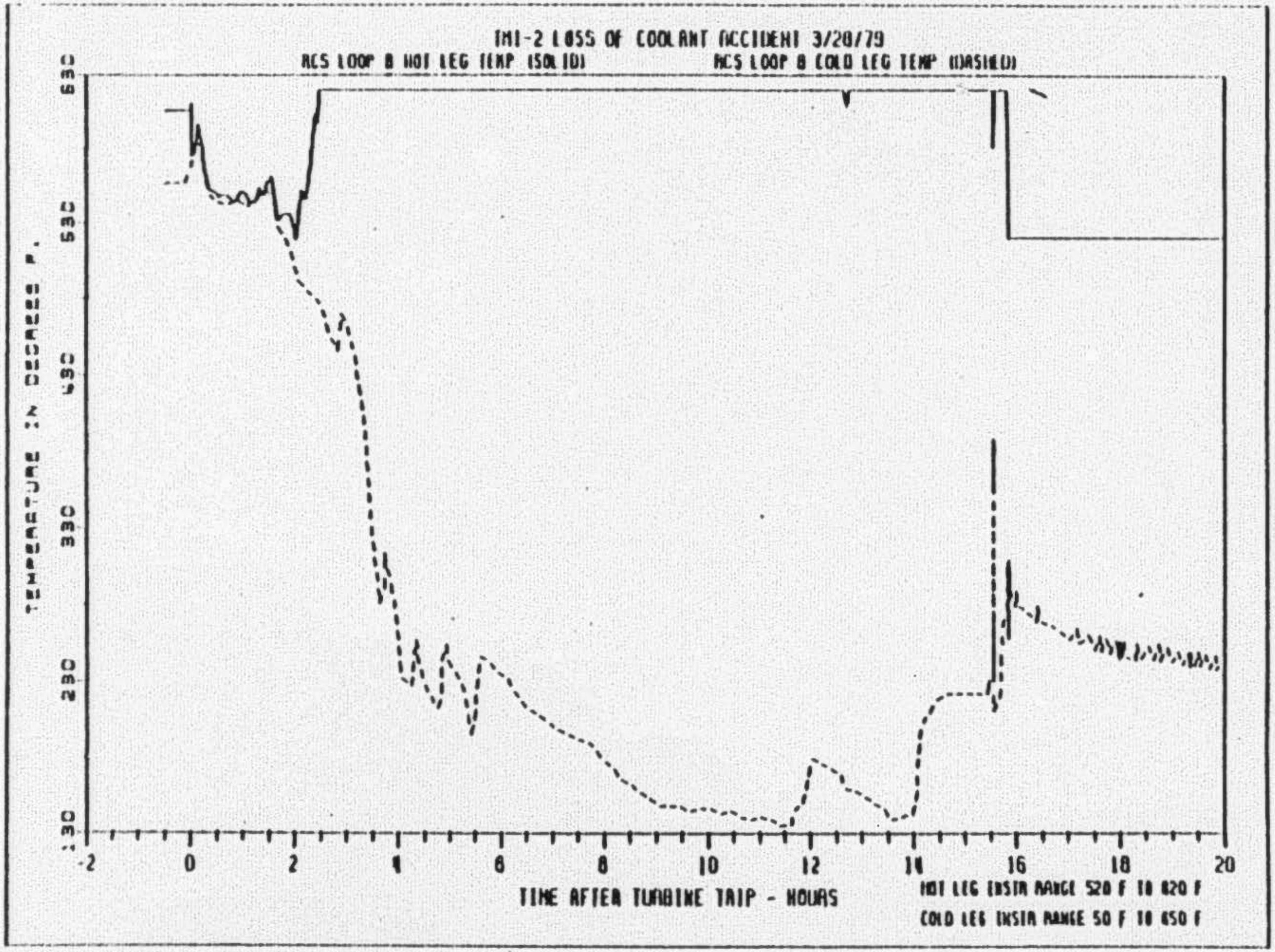

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
RCS LOOP B HOT LEG TEMP (SOLID)
RCS LOOP B COLD LEG TEMP (DASHED)
TEMPERATURE IN DEGREES F.
620
600
580
560
540
520
500
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - MINUTES

FIGURE 26



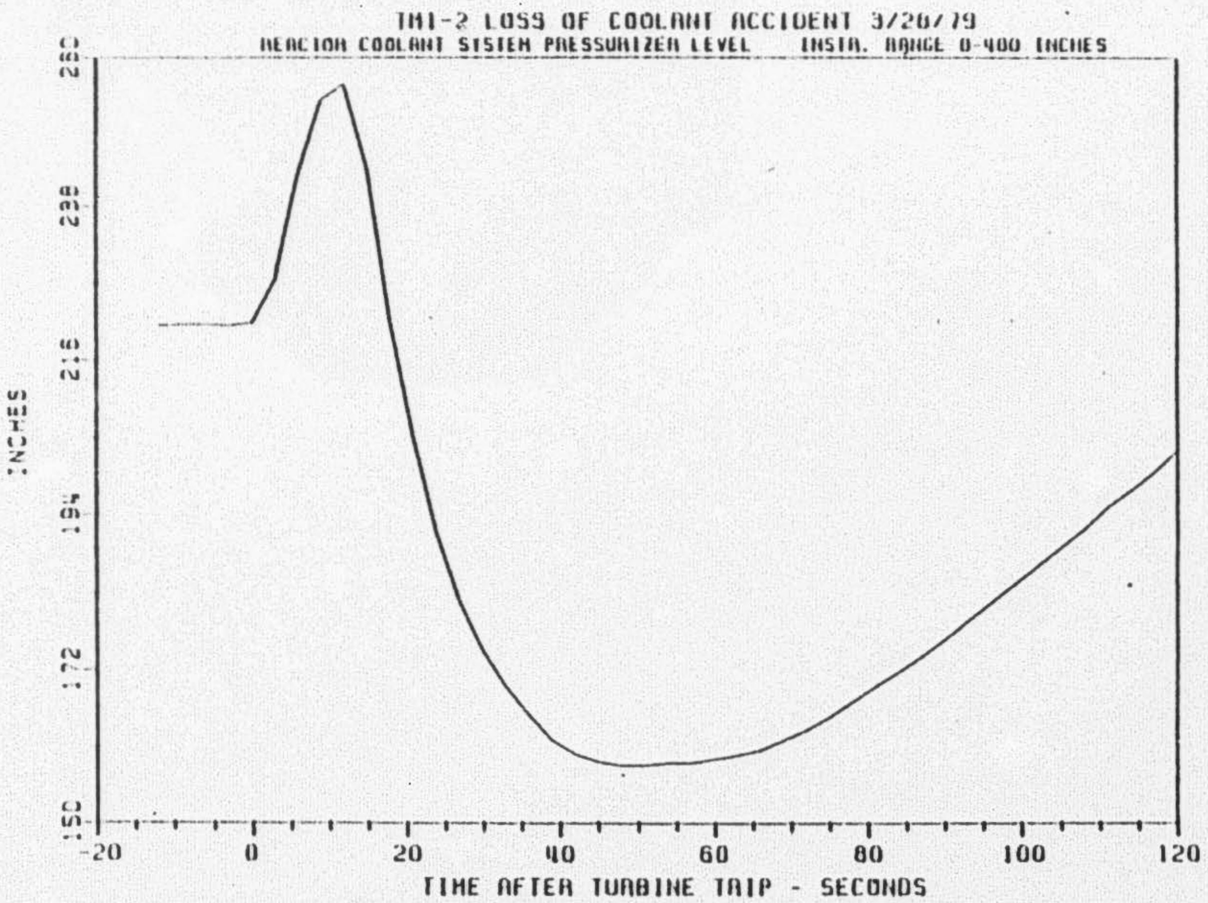

230 197

FIGURE 27



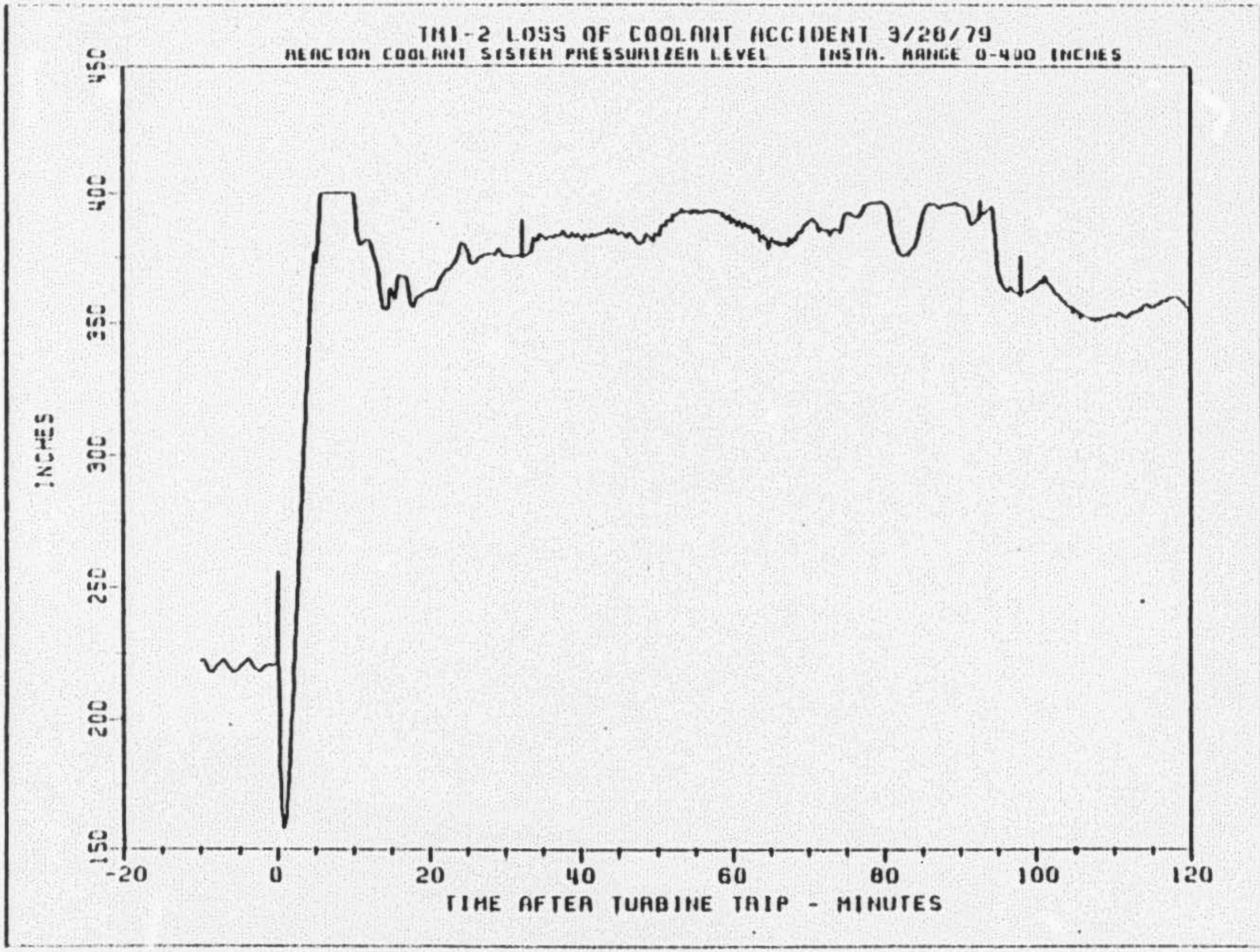

230 198

FIGURE 28



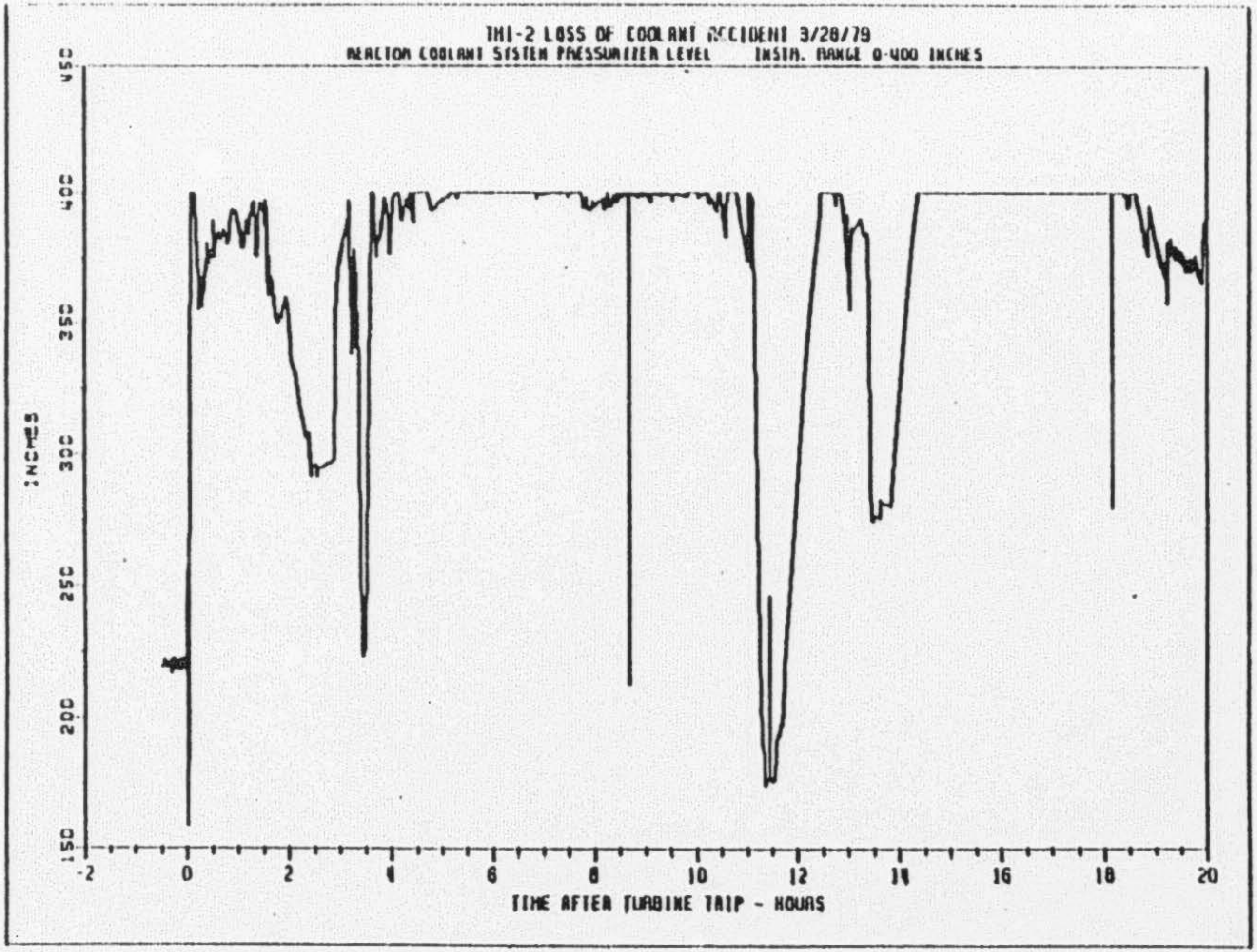
TMI-2 LOSS OF COOLANT ACCIDENT 3/20/79
REACTOR COOLANT SYSTEM PRESSURIZER LEVEL INSTR. RANGE 0-400 INCHES
INCHES
260
238
216
194
172
150
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - SECONDS

FIGURE 29



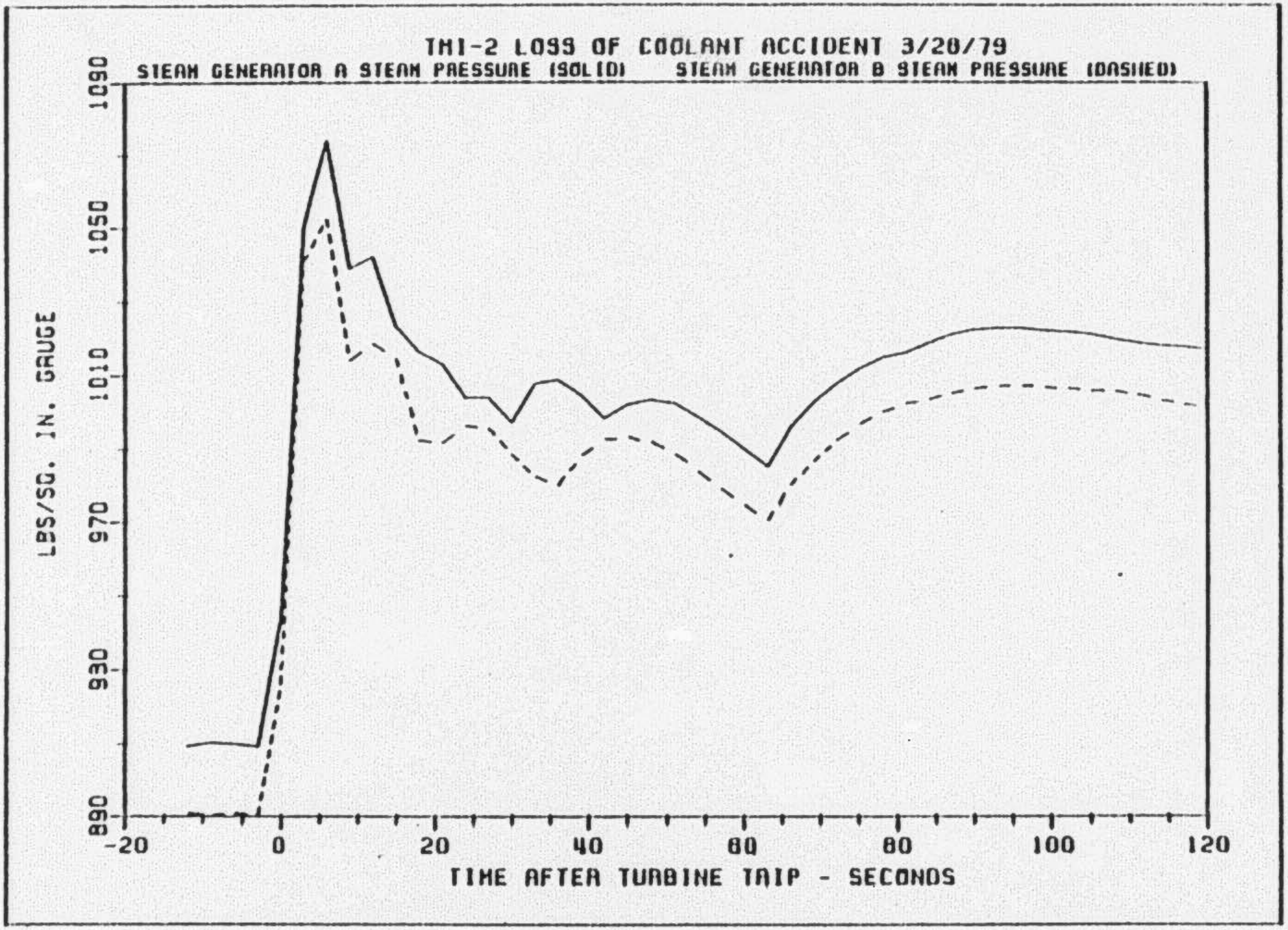

230 200

FIGURE 30




230 201

FIGURE 31




230 202

FIGURE 32

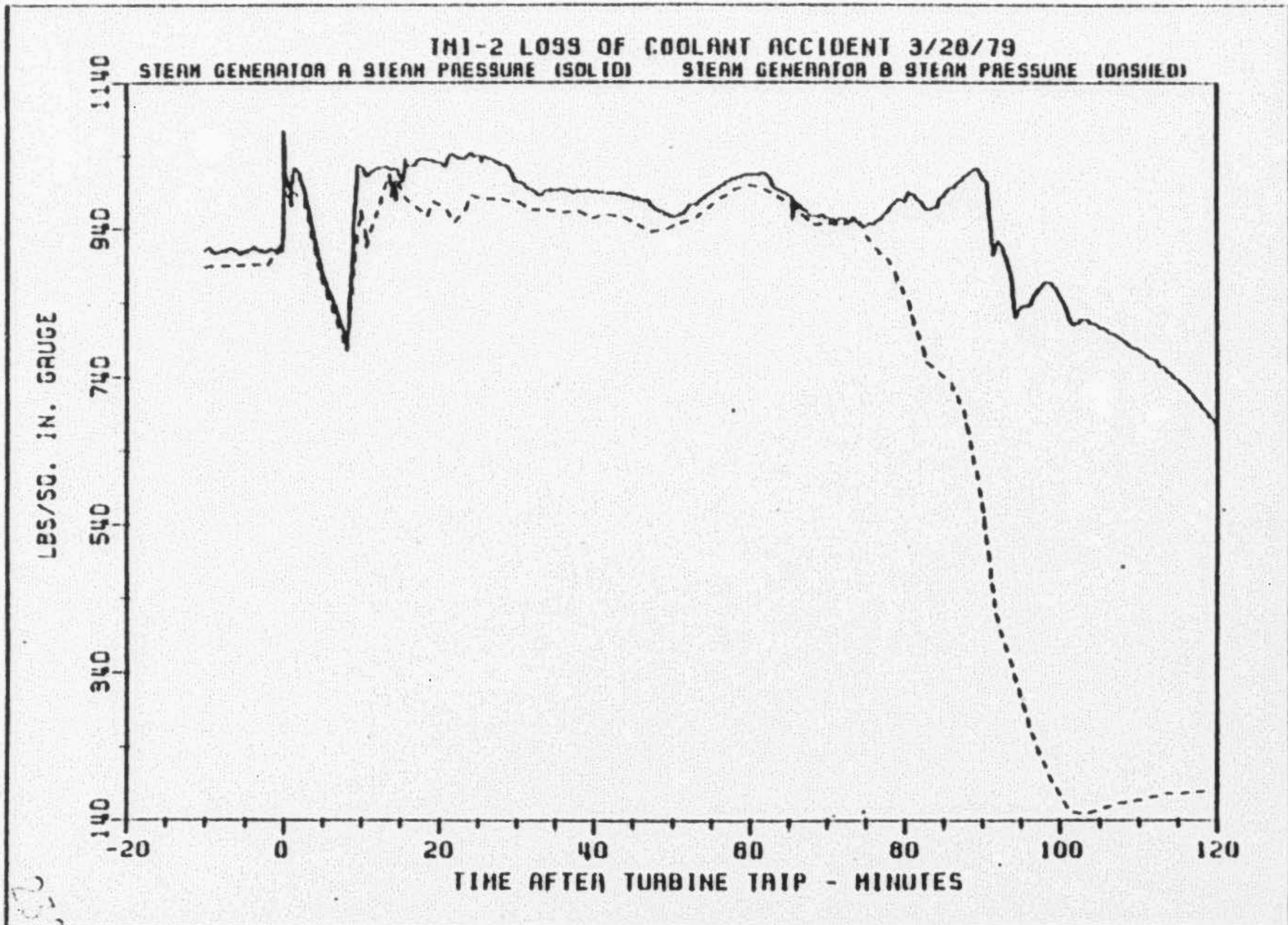
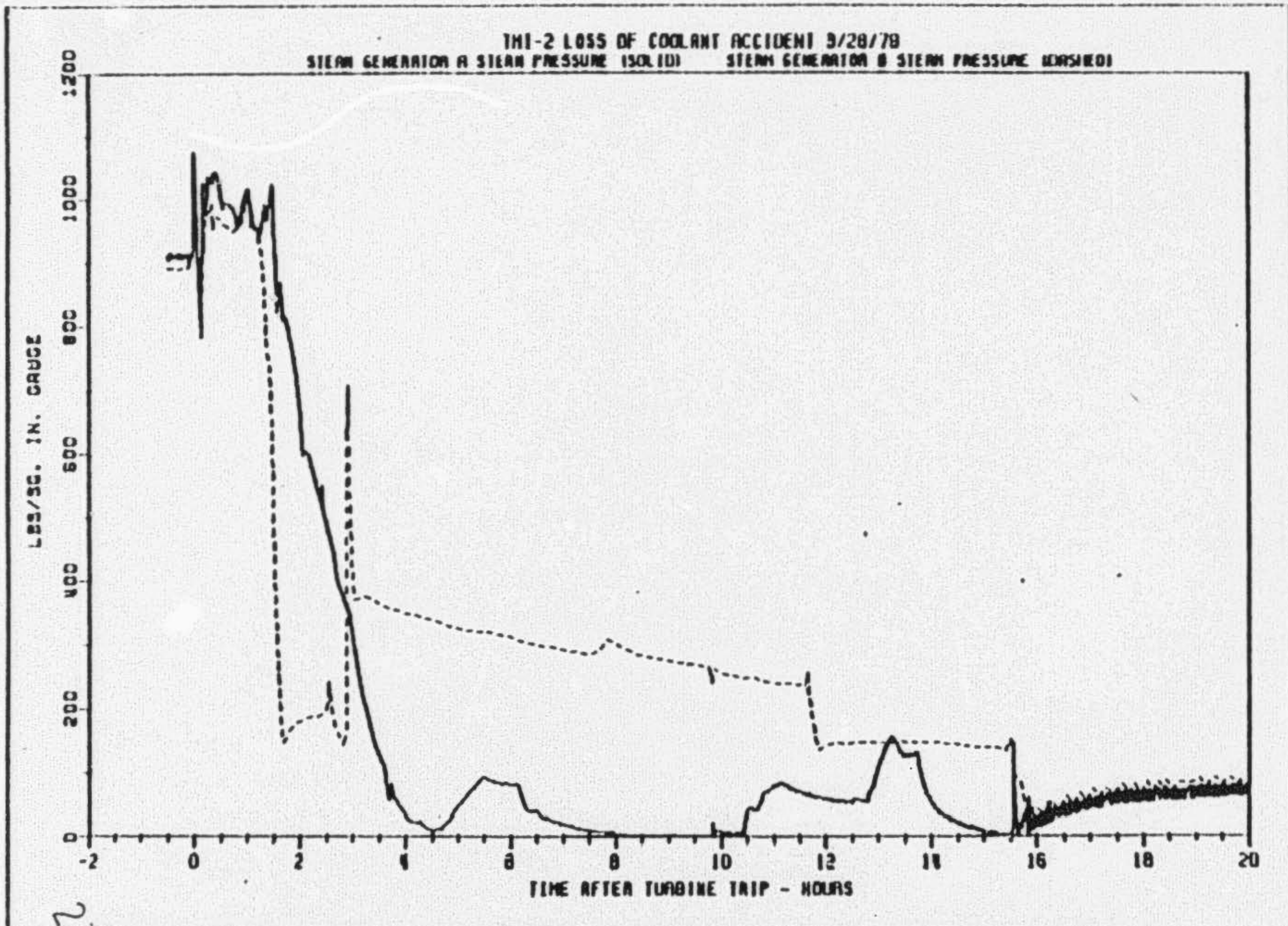
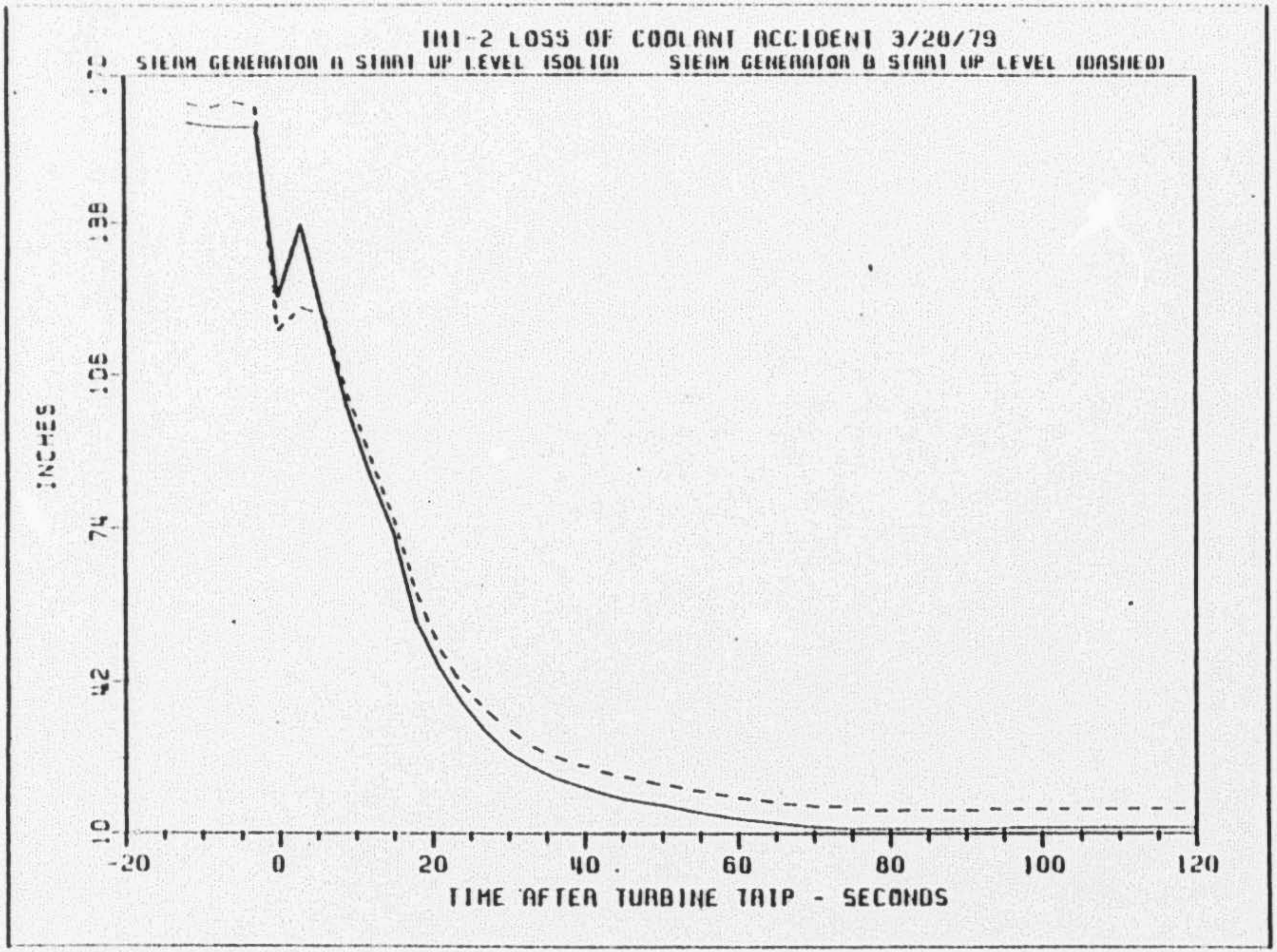

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
STEAM GENERATOR A STEAM PRESSURE (SOLID)
STEAM GENERATOR B STEAM PRESSURE (DASHED)
LBS/SQ. IN. GAUGE
1140
940
740
540
340
140
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - MINUTES

FIGURE 33



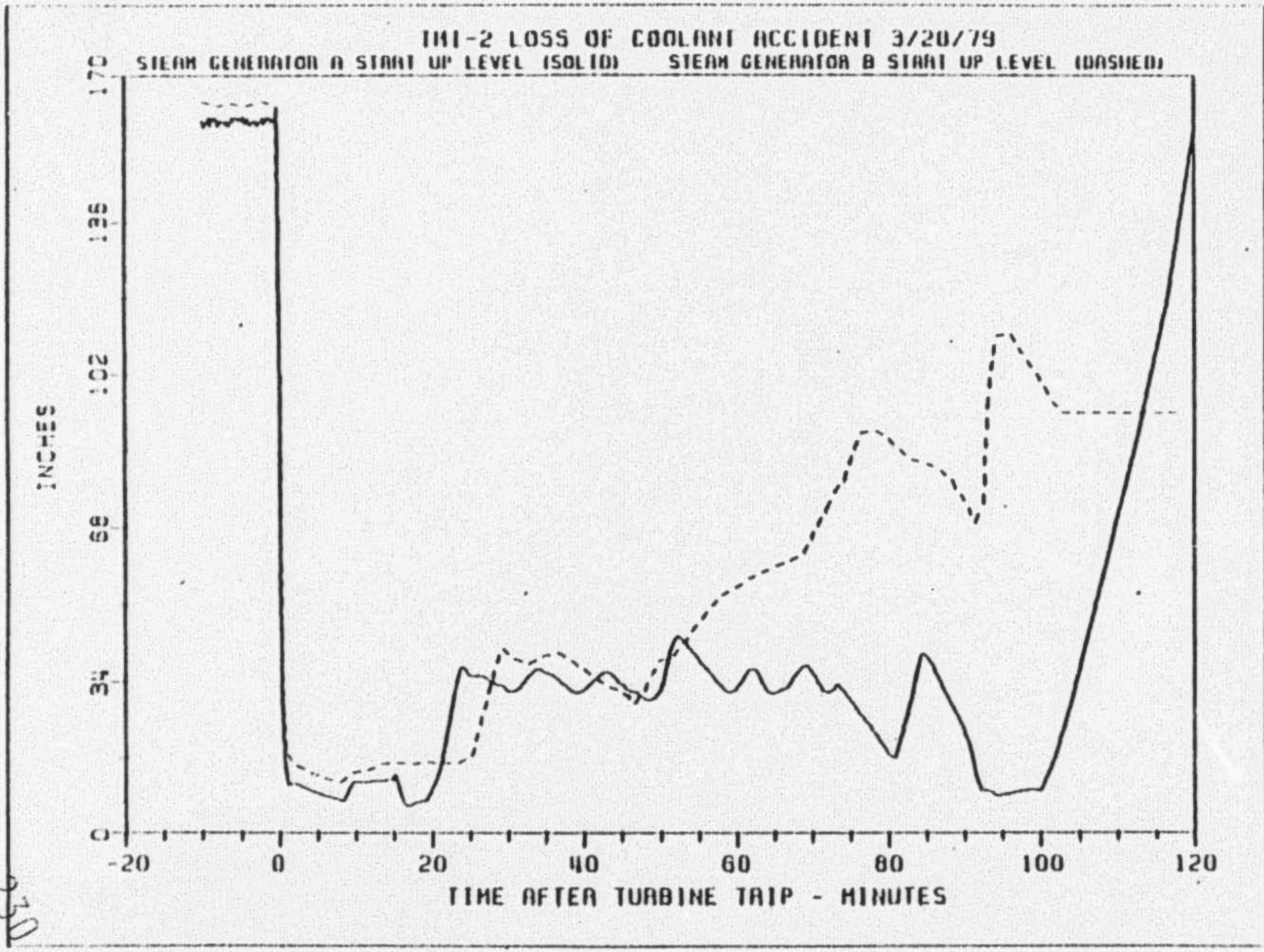

230 204

FIGURE 34



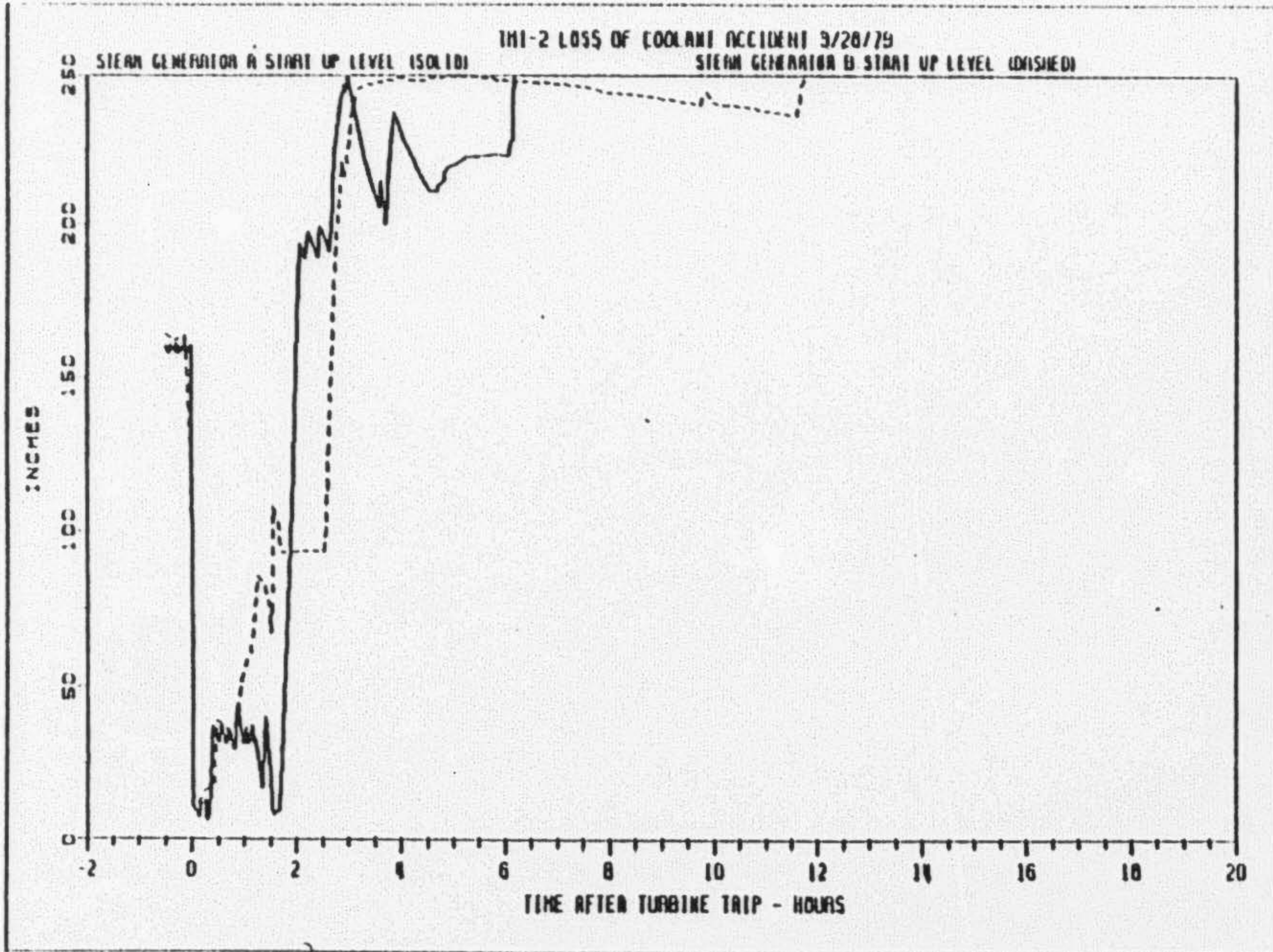

230 205

FIGURE 35



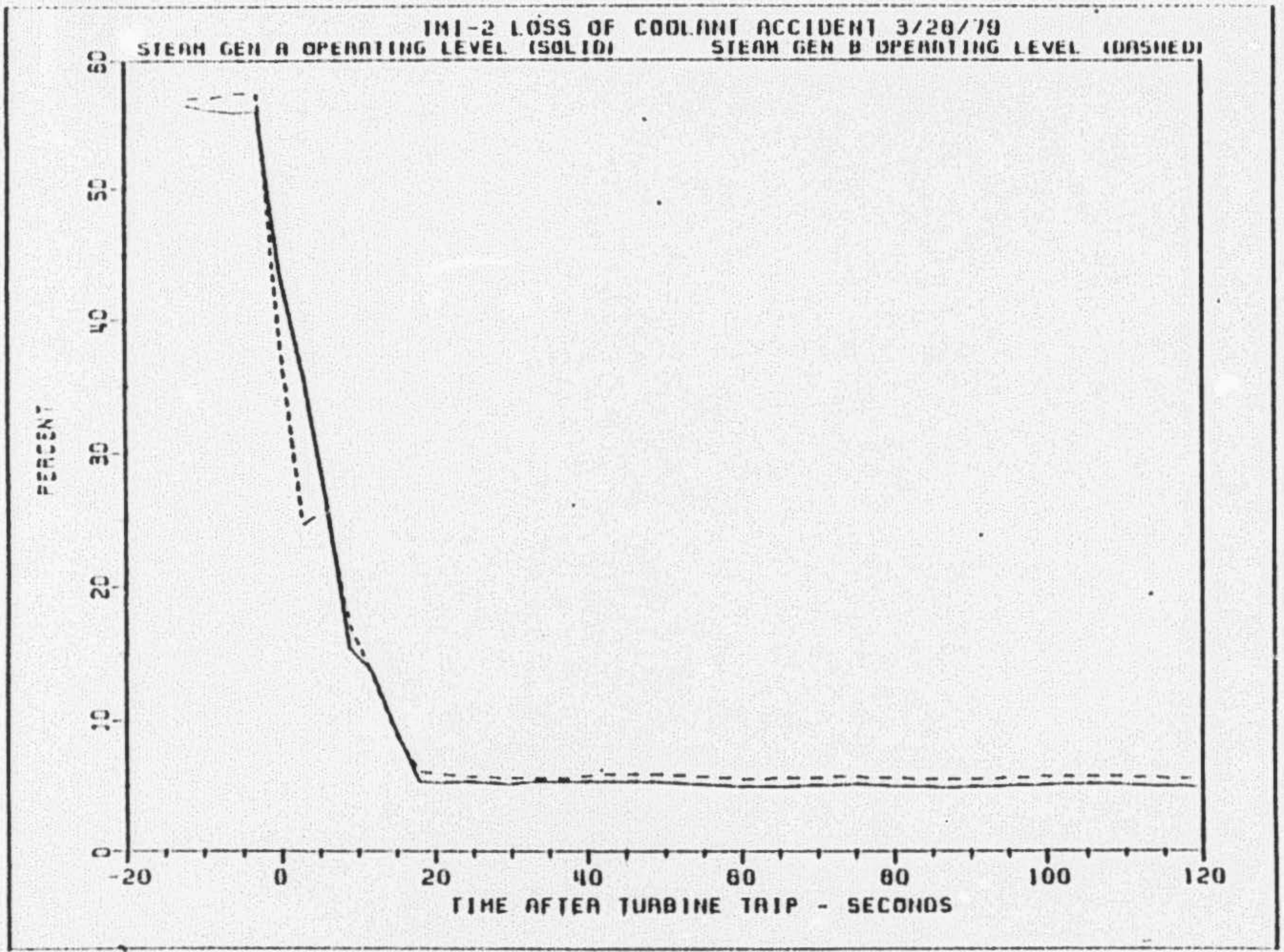

230 206

FIGURE 36



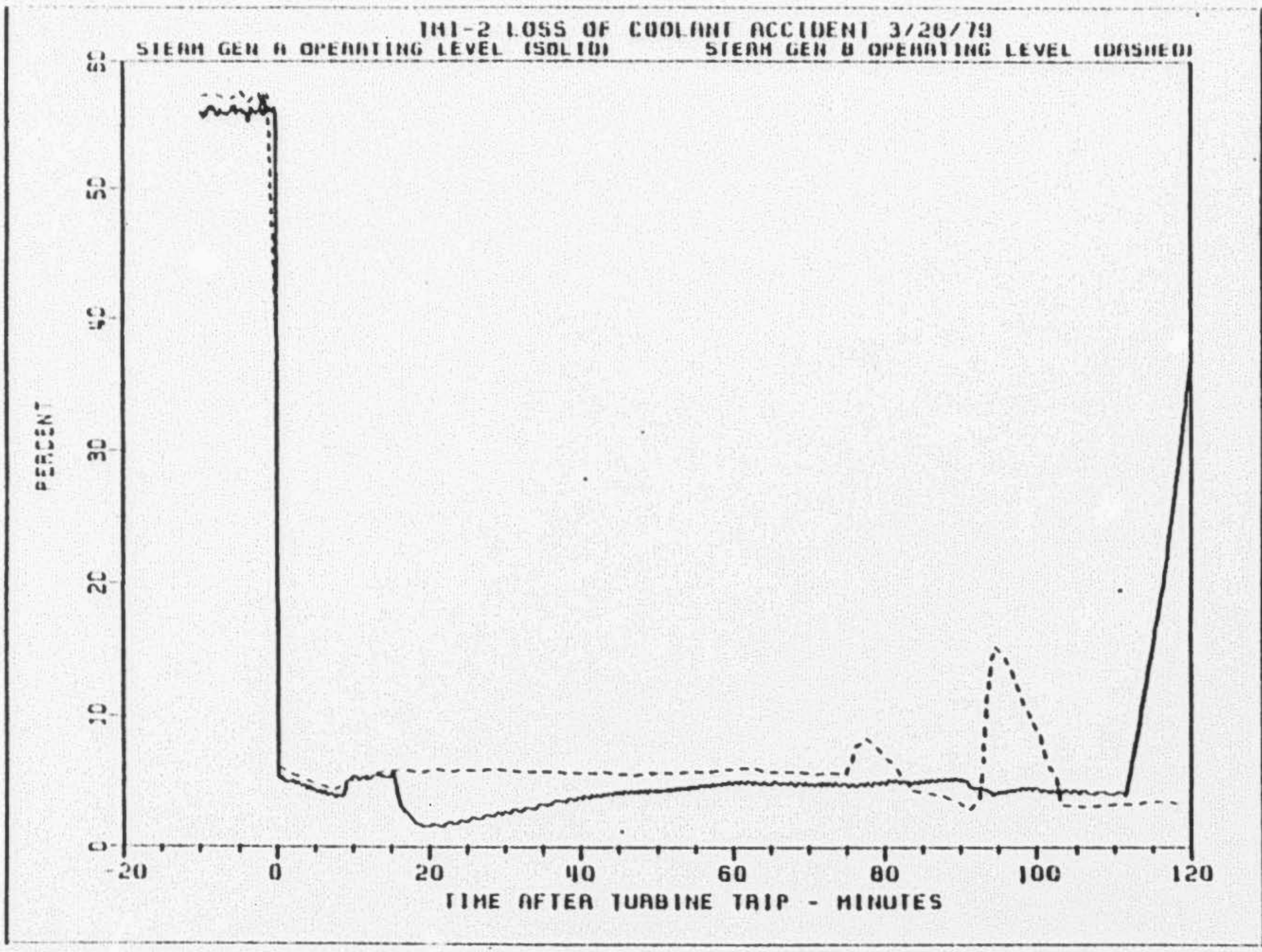

230 207

FIGURE 37



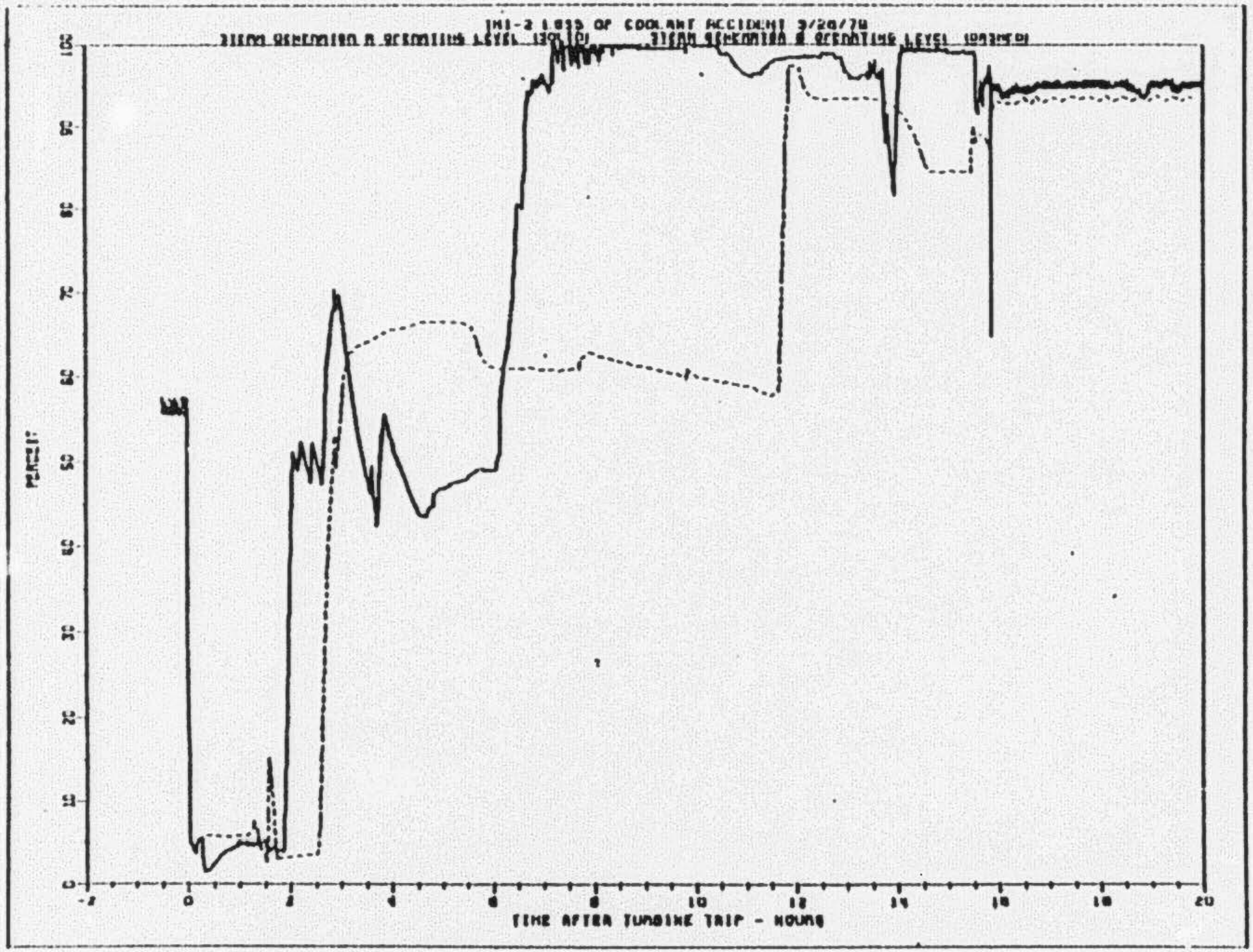

230 208

FIGURE 38



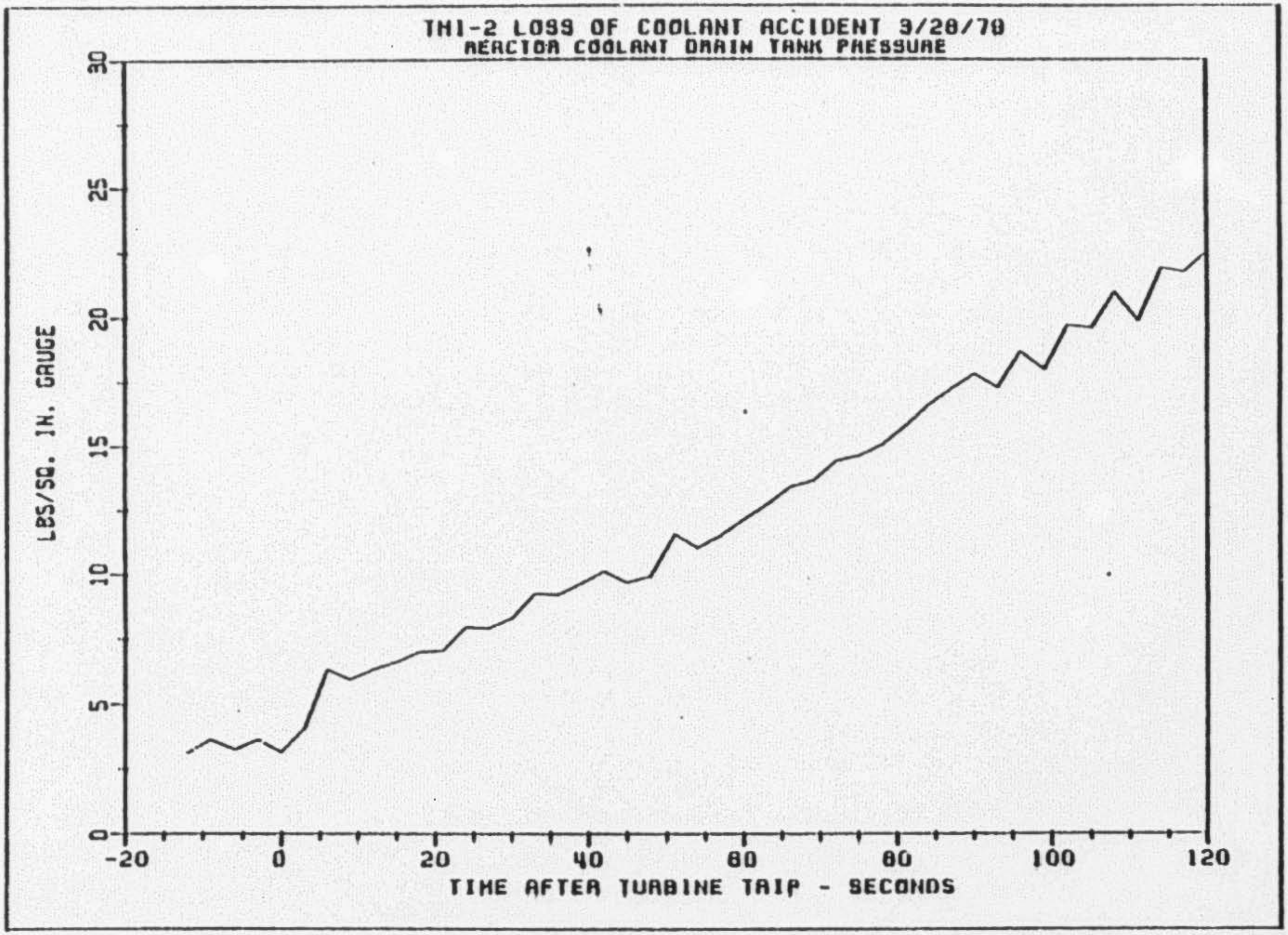

230 209

FIGURE 39



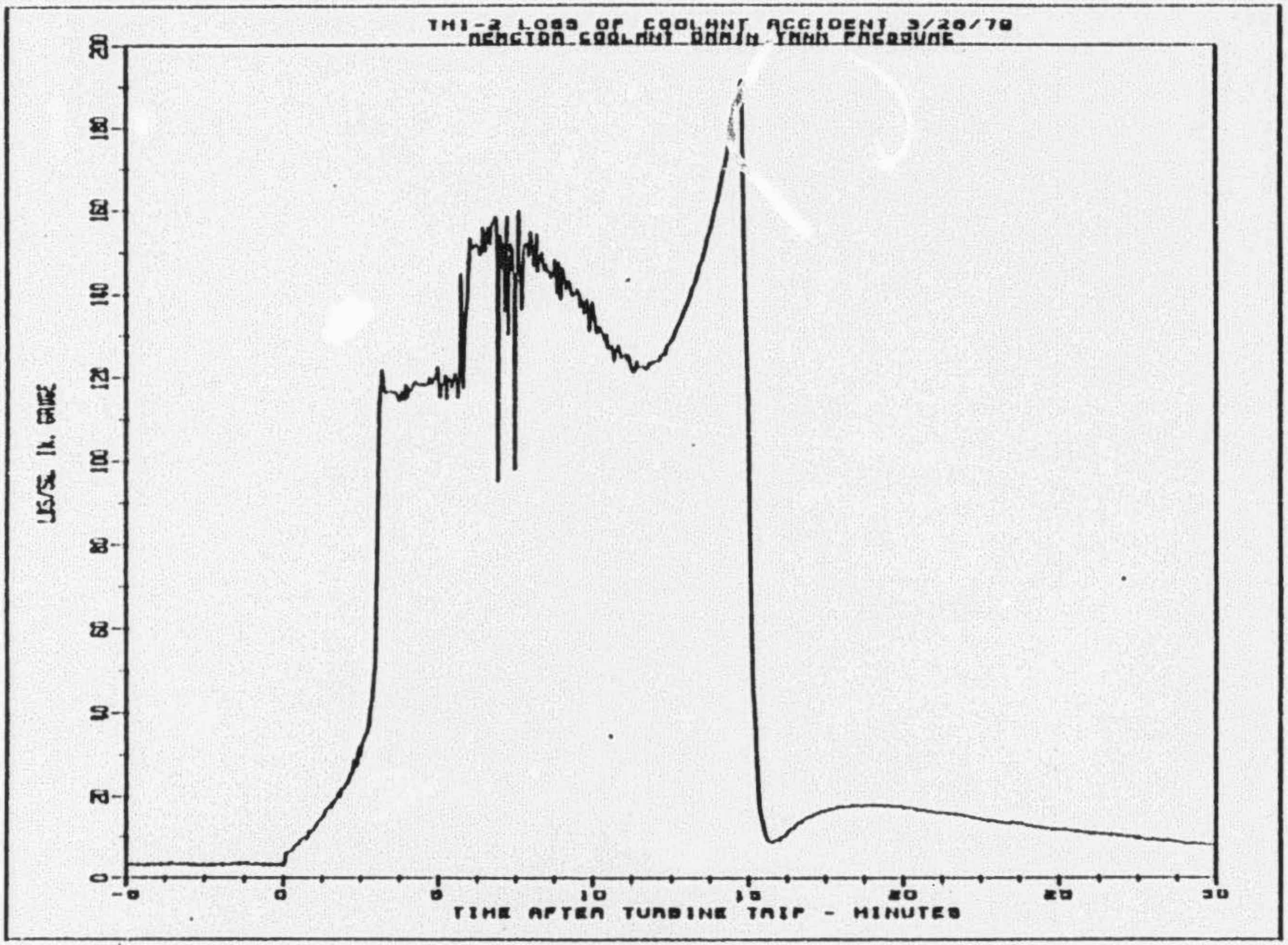

230 210

FIGURE 40




230 211

FIGURE 41




230 212

FIGURE 42

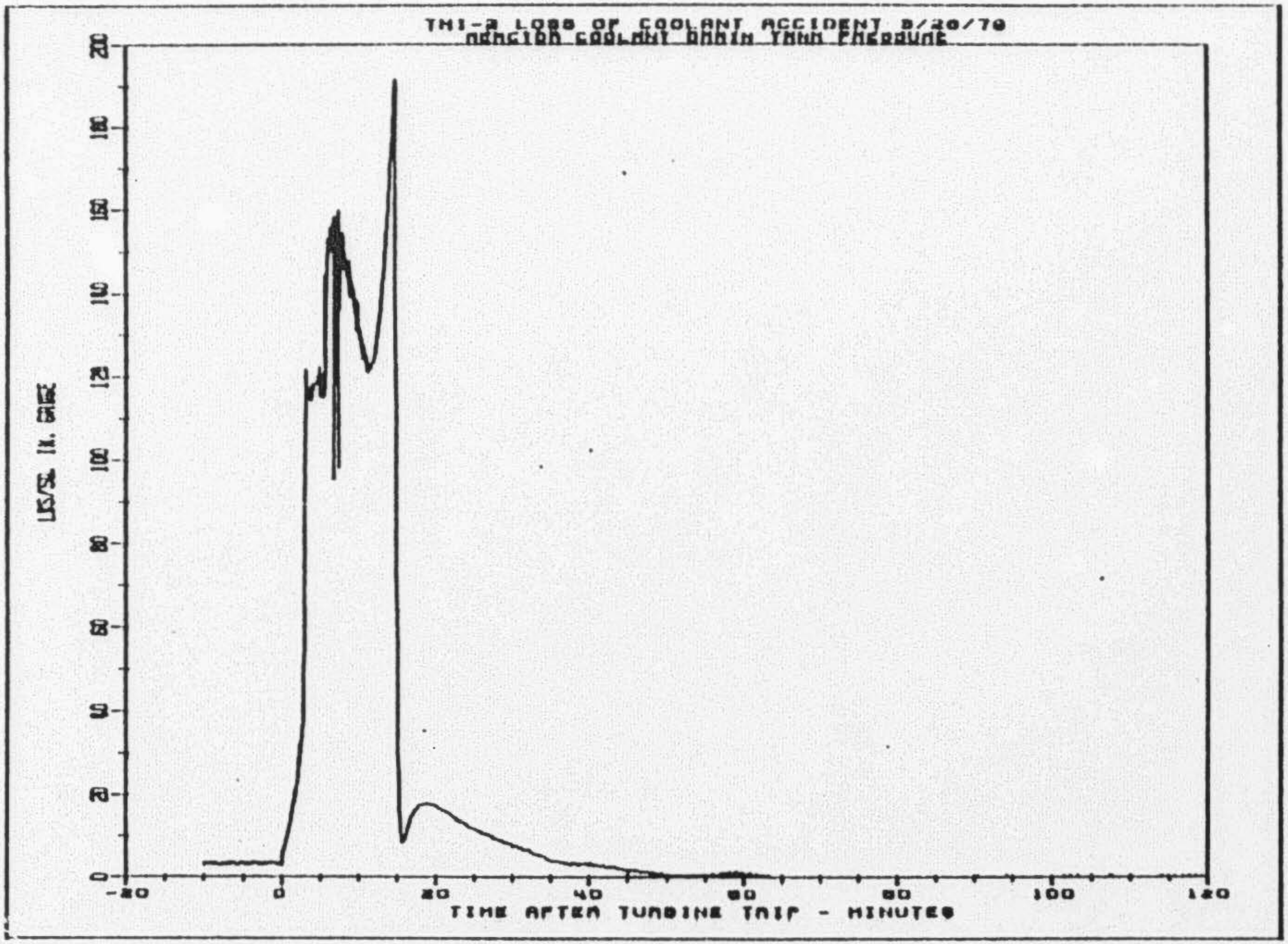
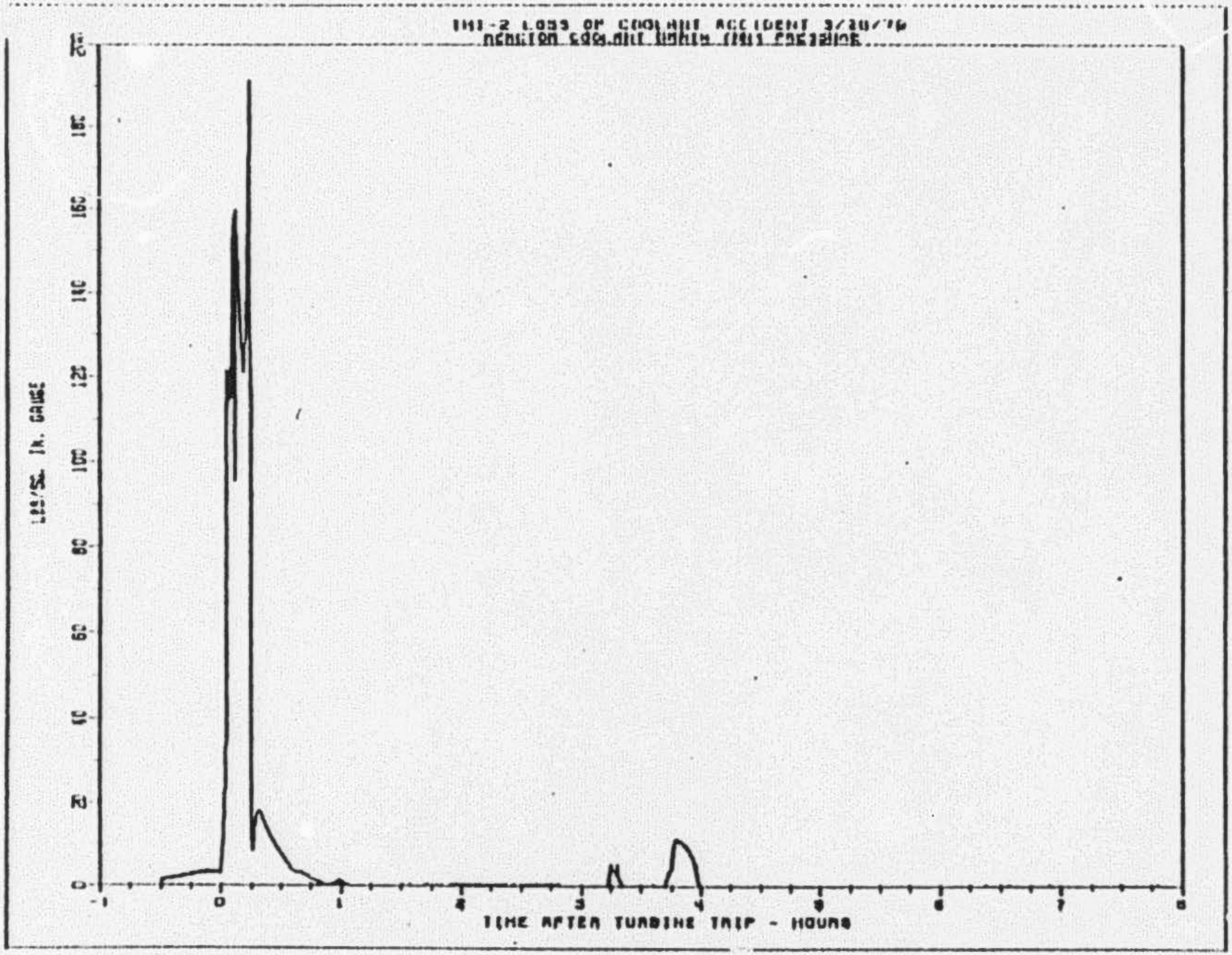
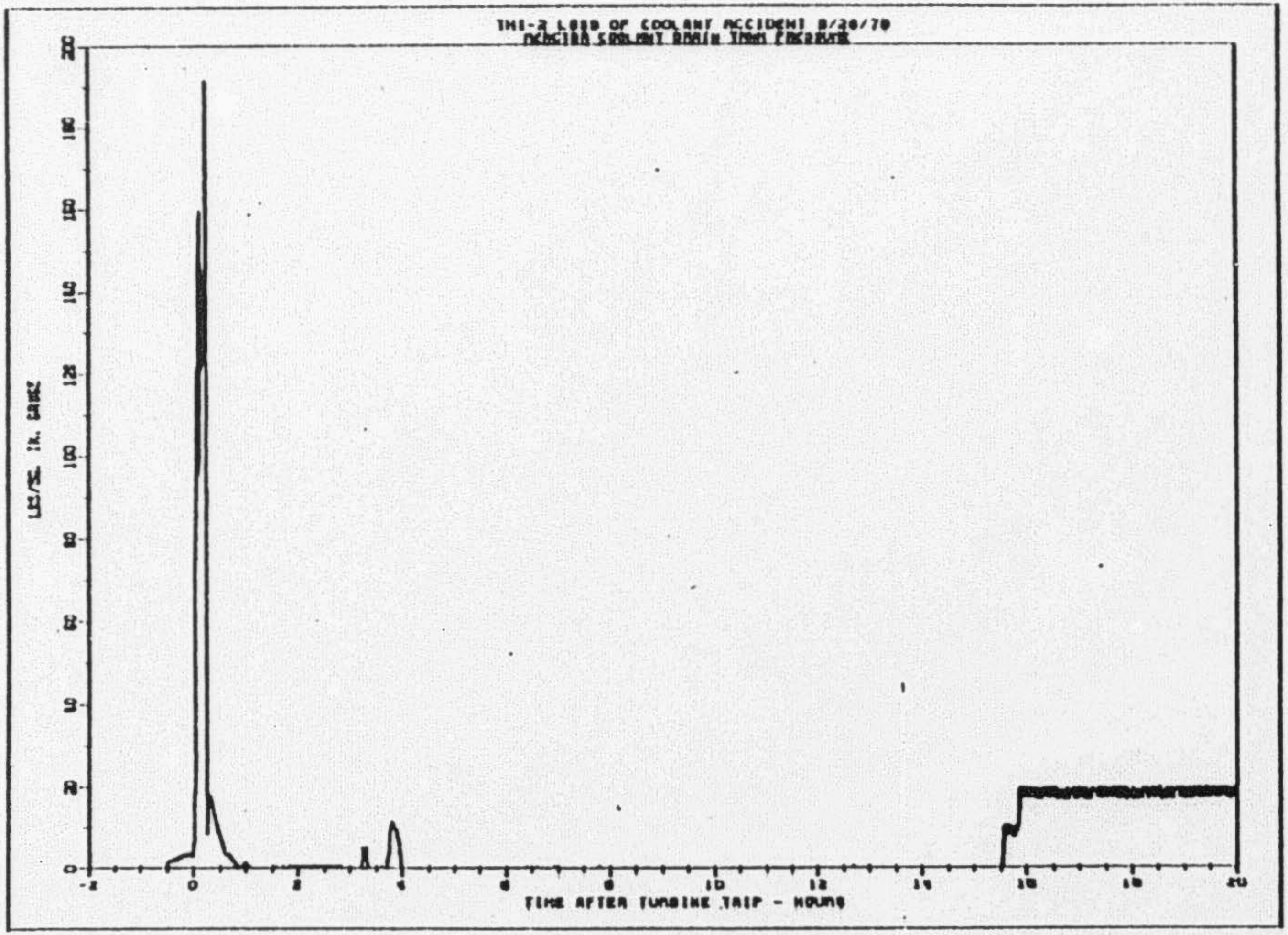

TMI-2 LOSS OF COOLANT ACCIDENT 3/28/79
REACTOR COOLANT DRAIN TANK PRESSURE
LBS/SQ. IN. GAGE
0
20
40
60
80
100
120
140
160
180
200
-20
0
20
40
60
80
100
120
TIME AFTER TURBINE TRIP - MINUTES

FIGURE 43



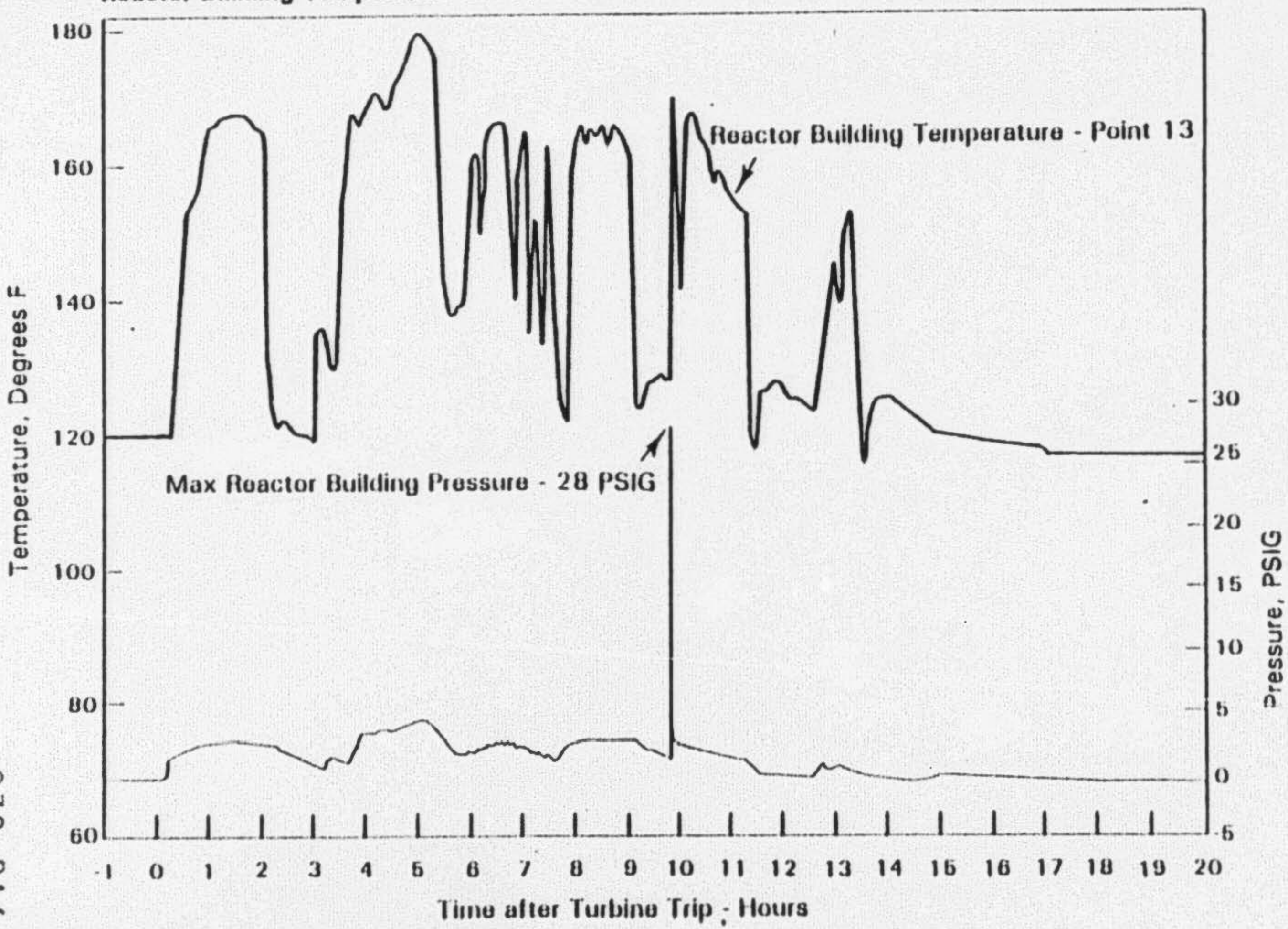

230 214

FIGURE 44




230 215

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




230 216

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

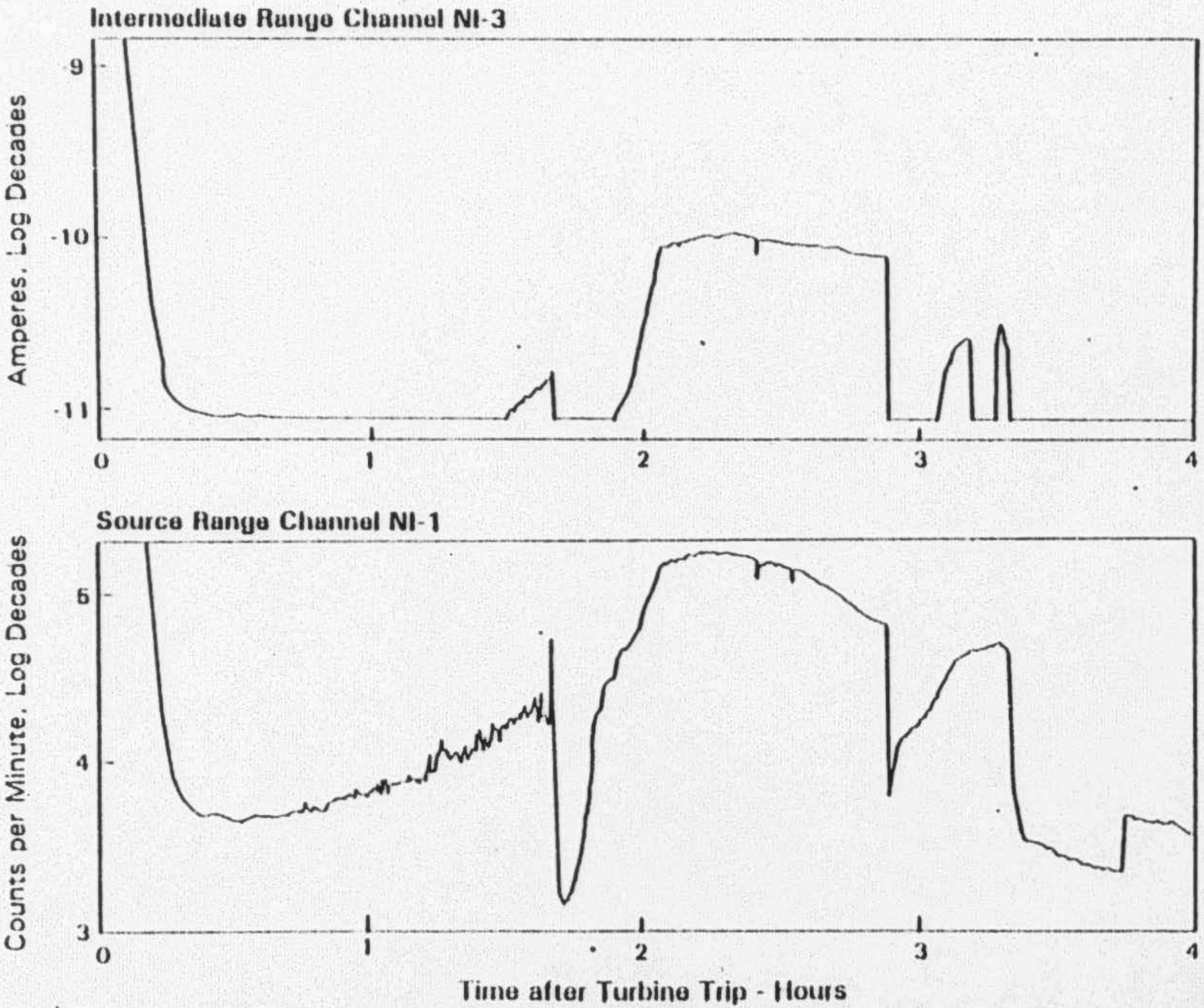
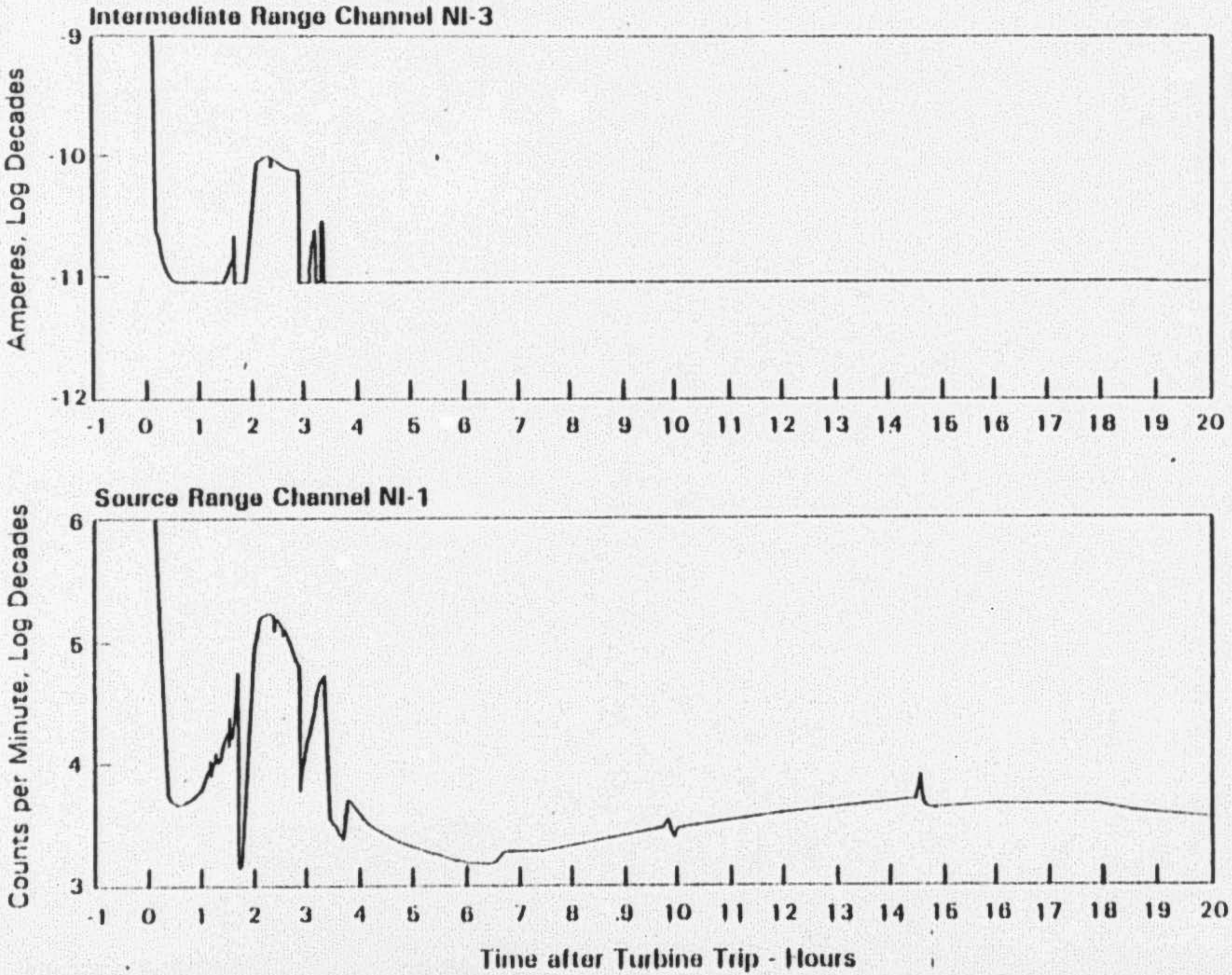
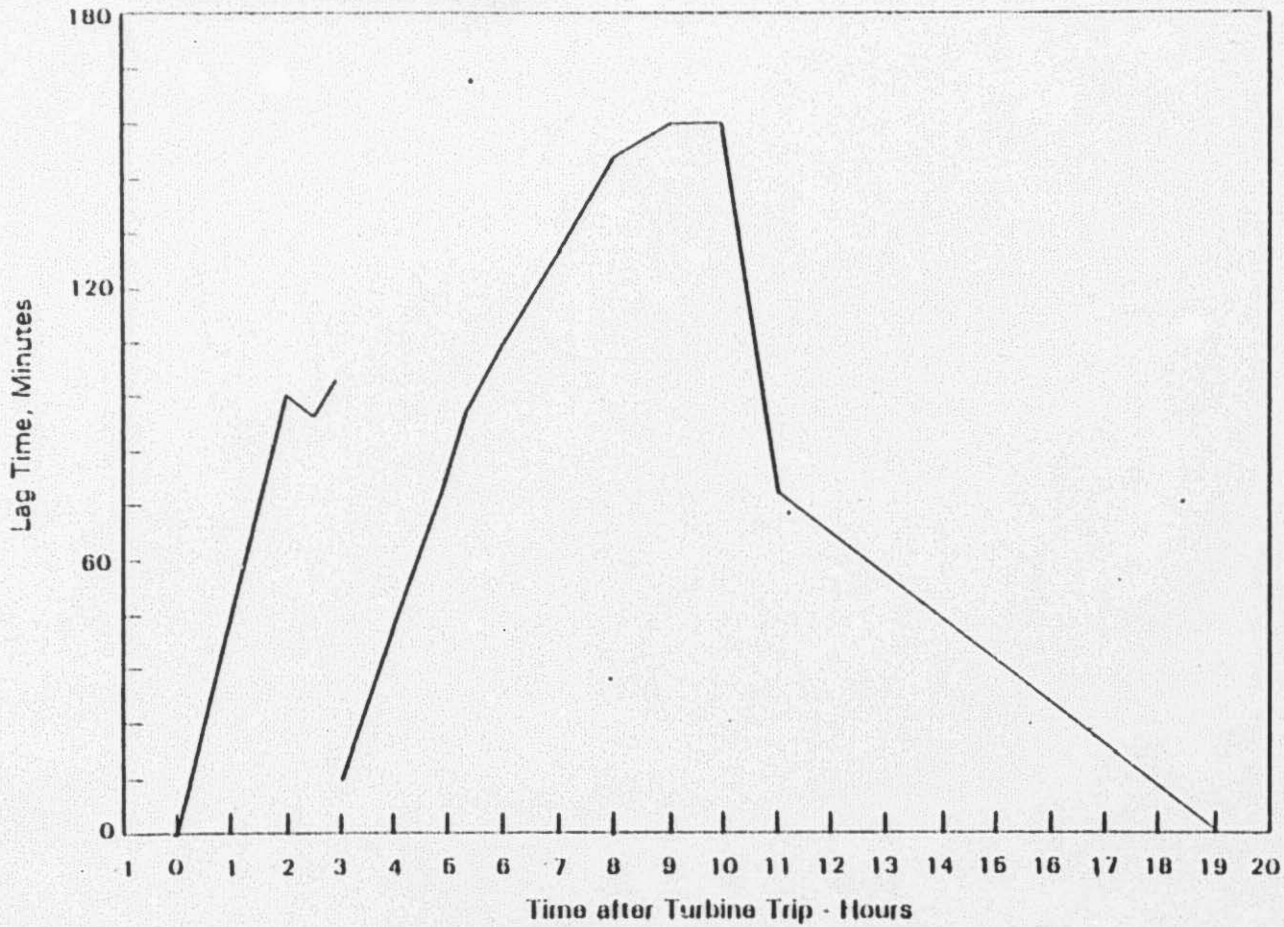


Figure 47
TMI-2 Loss of Coolant Accident of 3/28/79
Intermediate and Source Range Nuclear Instrumentation



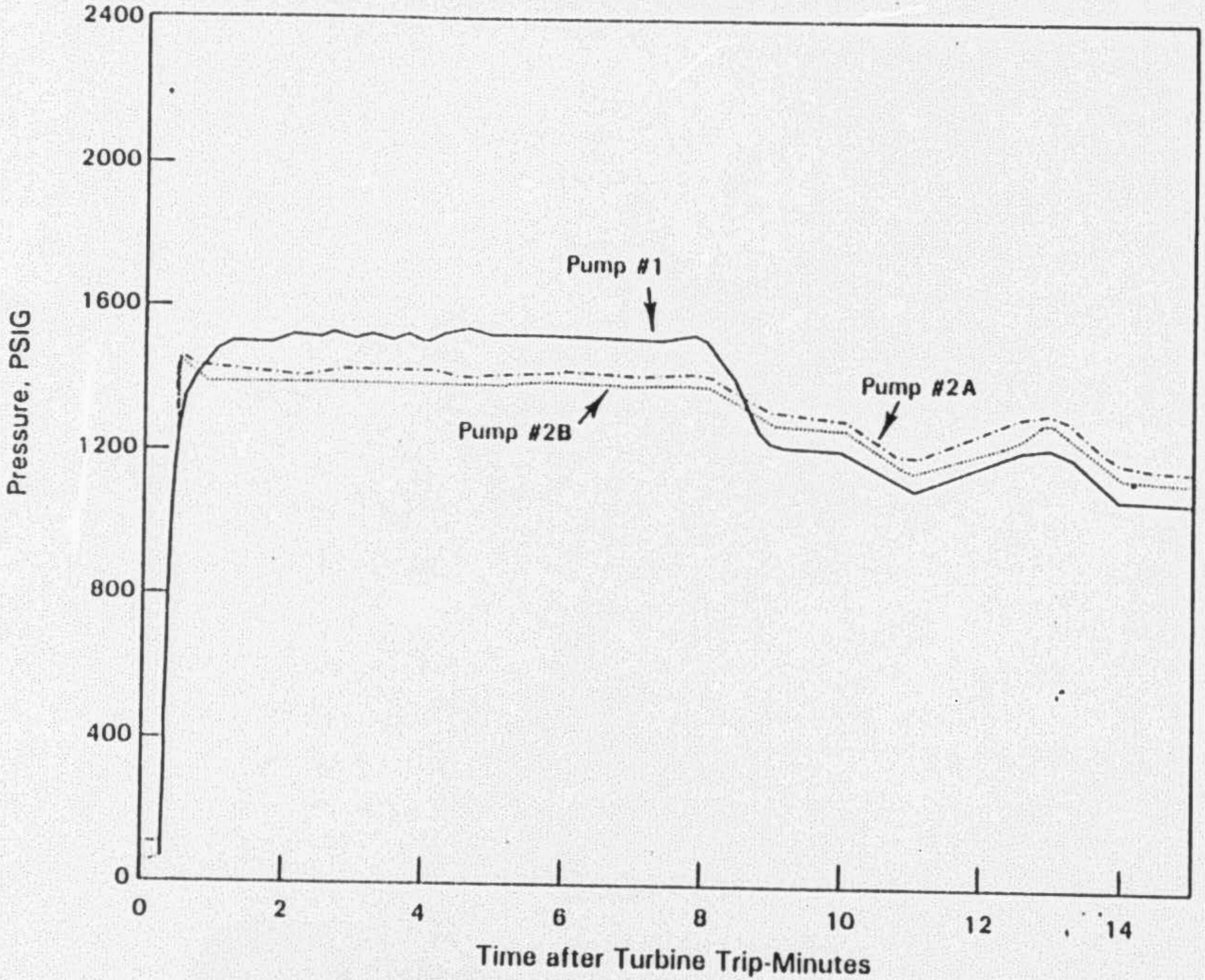

230 218

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




230 219

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




230 220

## II. RECOVERY ORGANIZATION

On Monday, April 2, 1979, the TMI Unit 2 Recovery Organization was initiated. Due to the constraints of the crisis, it was recognized then that the organization would be continually evaluated in light of the conditions that would exist and the tasks at hand, and that refinements and modifications would take place as appropriate.

The Recovery Organization consisted of an integration of GPU personnel with senior, experienced people from other utilities and nuclear industry organizations across the country.

The Recovery Organization focused on the following priorities:

A) Maintaining the current plant operations in the safest conditions.

B) Containing the release of radioactivity to minimize exposure to the public and onsite personnel.

C) Making a reliable safe transition to a benign and reliable long-term cooling mode for the plant.

D) Reinforcing the capability of the plant to assure long-term cooling.

Mr. Herman Dieckamp, President of General Public Utilities, established the Three Mile Island Unit 2 (TMI-2) Recovery Organization.

Mr. Robert C. Arnold, Vice President-Generation, GPU Service Corporation, was designated as the GPU Operations Manager with responsibility for the overall management of all onsite and near site capabilities and resources related to the recovery effort for TMI-2. The major near term objectives of the Recovery Organization were established as follows:

- Maintain the unit in a stable condition
- Control and manage the volumes of existing radioactivity
- Develop an overall waste management plan for liquid, gas, and solids
- Develop a strategy to reach cold shutdown safely and expeditiously
- Modify the necessary procedures, facilities, and equipment to accomplish the above; and,
- Establish the plan for accomplishing a transition into the organization necessary to proceed with the longer term recovery efforts.

The Recovery Organization established is shown in Exhibit 1. The Organization was divided into the following major groups:

Industry Advisory Group, Technical Support, Met-Ed Plant Operations, Waste Management Group, Plant Modifications Group, Task Management/Schedule Group, Administrative & Logistics, and Public & Government Affairs.

230 22

The functions for which each group has been responsible are as follows:

1. Technical Support Group - has provided engineering criteria and support, technical planning and analysis. procedure support, technical support to the control room, support to licensing requirements, and data reduction and management.

2. Plant Modification Group - has provided the engineering, design, materials and construction necessary to complete the plant modifications to equipment or structures.

3. Waste Management Group - was established to safely and effectively manage the quantities of radioactive gases, liquids and solids during the initial phases of the recovery operation. They are responsible for the development and implementation of short term plans to manage and process contaminated gases, liquids and solids; identification of the status of Auxiliary Building systems, establishment of processing priorities that are based on plant needs and decontamination of the Auxiliary Building.

4. Industry Advisory Group (I.A.G.) - was established as a "think tank" to function in parallel with all ongoing activities. The group was not to be part of the implementation structure. The group would of its own initiative look into potential problems of any kind, maintain a current awareness of the perceived status of the core, and provide assessments based on experience and judgement as opposed to detailed engineering review and calculations.

5. Plant Operations Group - consisted of the Met-Ed TMI Plant Staff with substantial augmentation from other organizations whose immediate objectives were:

   - Perform all plant operations and maintenance activities required
   - Limit personnel exposure
   - Stop off-site uncontrolled releases
   - Return the plant to a benign status
   - Ensure the plant's ability to respond to any future emergencies.

6. Task Management/Scheduling Group - was formed to coordinate and monitor the overall tasks and priorities, plans, schedules and work progress of all groups. They were aware of information that the groups required to perform their tasks and assisted them in obtaining the required information.

7. Technical Working Group - is a group which includes the heads of each of the groups described in 1. through 6., and representatives from Babcock and Wilcox Co. and the Office of Nuclear Reactor Regulation.

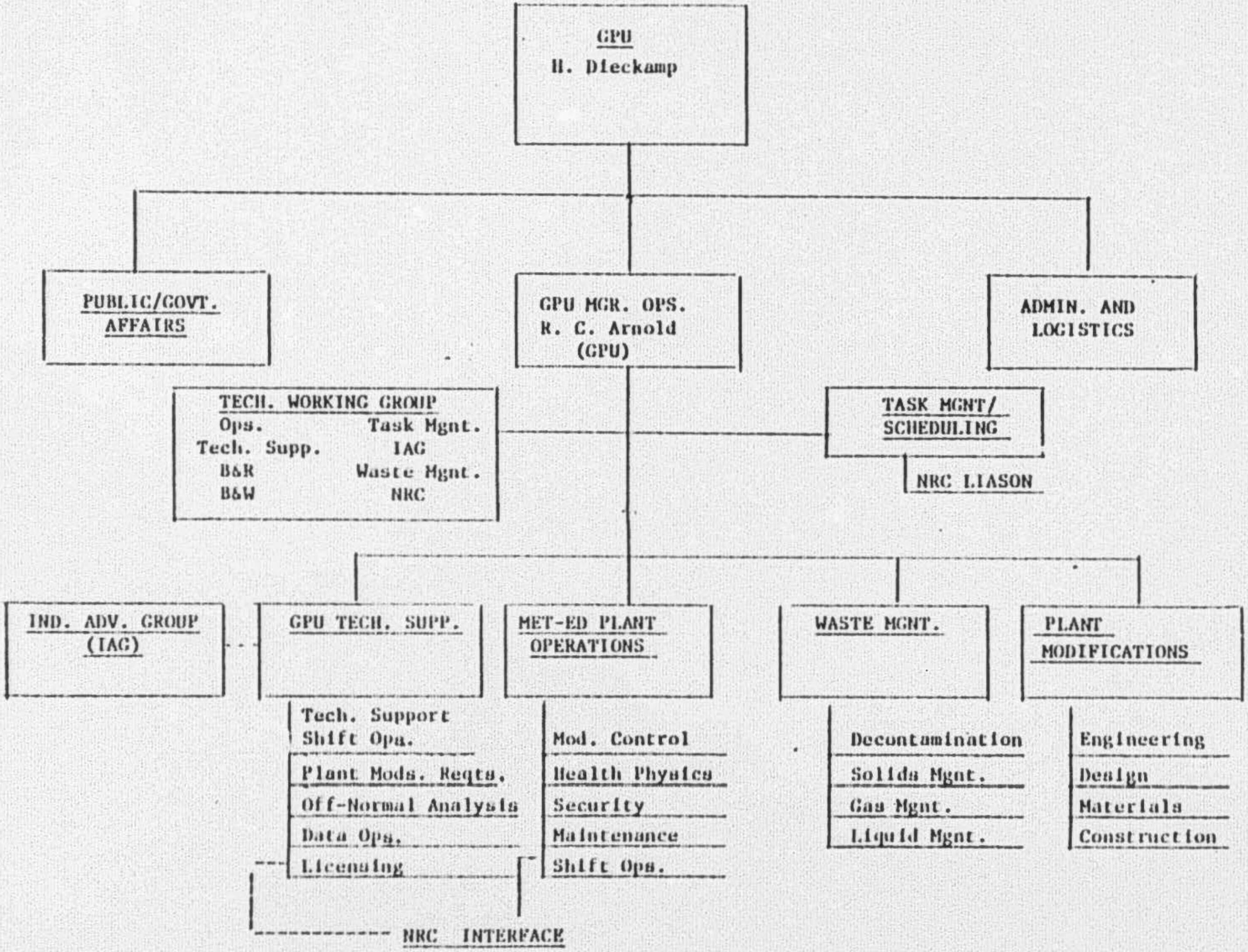
230 222

8. Administration & Logistics - was formed to handle the necessary administrative logistics requirements such as communications, manpower, transportation, maintenance and commissary arrangements.

9. Public and Government Affairs - was established to coordinate the interface requirements needed with the public and governmental bodies.

Due to the nature of this section of the report, only changes to the Recovery Organization will be presented in subsequent reports.

EXHIBIT 1




230 224

## III. PLANT MODIFICATIONS

Following the incident which occurred on March 28, 1979 at TMI-2, several modifications to TMI-2 systems were made or considered. These modifications were undertaken in order to augment the existing systems for both the containment of radioactivity, and the control of plant conditions during the establishment of long-term cooling.

This section includes a discussion of the various modifications which have been made. In addition, a book of drawings/diagrams pertaining to these modifications is enclosed for your reference.

A. Hydrogen Recombiners

1.0 System Function and Design Objectives

In anticipation of having to process substantial amounts of hydrogen to prevent a hydrogen explosion in the reactor building, and because of the uncertainity of the quantity of hydrogen being generated, the available hydrogen recombiner capacity was increased.

Operations shall not permit an uncontrolled release of reactor building atmosphere to the environment.

2.0 System Description

A thermal type hydrogen recombiner is installed in the fuel handling building at the spent fuel pool operating floor and is connected to the reactor building ventilation and purge system as originally intended (see FSAR Figure 6.2-30). In addition, a skid-mounted, thermal-type hydrogen recombiner has been installed next to, and has been connected in parallel with the first hydrogen recombiner. The integrity of the system is as originally installed and will ensure that there is no uncontrolled radioactive release to the environment.

3.0 System Operation

The hydrogen recombiners are only operated as required to control reactor building hydrogen concentrations. Periodic samples are taken from the containment atmosphere to monitor hydrogen concentration levels.

Recombiner operation is monitored and controlled manually from a local panel. A recombiner "trouble" alarm is annunciated in the control room.

Testing is performed in accordance with established procedures to ensure proper functioning.

4.0 Status

The Hydrogen Recombiners are installed and functional.

230 226

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 systems.

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 detailed description of the system's operation is not yet available. This description will be included in a subsequent report.

4.0 System Status

Engineering 80% Complete
Construction 75% Complete

Two (2) trains of four (4) are in operating with the stack uncapped. Balance of two (2) trains being readied for operation.

System description, flow diagrams, operating procedures, operating and failures modes analysis being prepared.

230 227

C. Condenser Air Extraction Filtration System

1.0 System Function and Design Objectives

In order to permit continued operation, the Condenser Air Extraction Filtration (CAEF) system has been installed to limit radioactive discharge to the station vents.

The system capacity will be sufficient to process normal condenser air extraction flow required to maintain vacuum.

2.0 System Description

The CAEF system is installed in the basement of the turbine building downstream of the condenser vacuum pumps. The system discharges to the auxiliary building ventilation system. Existing radiation monitors upstream and downstream of the CAEF system are used to monitor performance. (See FSAR Figures 10.1-5 and 9.4-4).

The system includes an air preheater at the filter train inlet, a fan at the train discharge, a bypass line, and a supplementary intake from the turbine building atmosphere. The filter train consists of (in the direction of flow) a prefilter, high-efficiency particulate air (HEPA) filter, an activated carbon filter, and a second HEPA filter. Since filter and fan capacity (2000 CFM) exceeds the normal condenser extraction flow, the difference in flow is made up from turbine building air. the CAEF system is designed to filter out 95% of the iodine in the extraction air. The filter bypass is provided to permit filter maintenance or rapid condenser draw down (hogging) when radiation levels are normal.

Fire suppression is normally provided by the fire water supply in the turbine building. In addition, a manually-initiated water supply is provided over the charcoal filter, should the filter bed overheat.

3.0 System Operation

During normal operation, all extraction air is directed through the filter train. The filter will be monitored for radiation to minimize exposure when temporarily shutting down the CAEF system and filters. Filter efficiency can be determined during operation by drawing and analyzing air samples from upstream and downstream of the charcoal filters. The filter will also be monitored to ensure that the preheater remains operational, that the flow is adequate, and that filter differential pressure remains within allowable limits.

230 228

The filter train will be bypassed when the condenser air extraction system is placed in the hogging mode only if extraction air radioactivity levels are within allowable limits. Furthermore, radiation levels will be continuously monitored and an operator available for bypass valve closure while operating in this mode.

During shutdown the filter exhaust fan will not be stopped until all of the condenser vacuum pumps are taken out of service. If maintenance is required, all operations will be performed in accordance with existing health physics requirements.

Normally the filter exhaust fan will draw more air than that being drawn from the condenser. However, should the fan trip, the condenser vacuum pumps could pass some lesser flow through the filter and cause leakage to the turbine building through the inlet gravity damper. If the exhaust air radiation levels are within acceptable levels, then the bypass line can be opened and control will be provided as in the hogging mode. However, if radiation levels are unacceptable, then both the filter inlet and outlet valves will be closed and the condenser vacuum pumps will be secured until the situation is corrected.

Effluent quality is normally monitored from the control room using existing radiation monitor HP-12-228 and other instrumentation. Any monitoring and action associated with changing CAEF system operation will manually be performed at the filter. The following indications and alarms are provided at the filter:

a. Preheater Operation
b. Air Temperature Indication
c. Low Fan Discharge Pressure Alarm
d. High Charcoal Filter Temperature Alarm
e. Fan Operation
f. Prefilter Differential Pressure Indication
g. HEPA Filter Differential Pressure (each filter) Indication

## 4.0 Status

The Condenser Air Extraction Filtration System is installed and operational.

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 lowering an eductor into the stand pipes for processing or transporting the radioactive waste.

3.0 System Operation

A detailed description of the system's operation is not yet available. This description will be included in a subsequent report.

4.0 System Status

The six tanks and the two stand pipes have been placed into position in the "A" spent fuel pool. The piping in the fuel handling building is approximately 20% complete.

230 230

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, remote venting of the pumps, and to permit remote feeding of oil to the bearings.

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.

230 231

4.0 Status

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

The installation and testing of bearing oil tanks and piping, vault shielding, and vibration and loose parts monitoring system, as well as the final positioning of TV monitoring equipment, and the preparation of DHR system cold operation test procedure remains to be completed.

230 232

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 will be 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 will be connected to the existing feed-water piping downstream of the main feedwater pumps, and the heat exchanger inlet piping will be 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 not yet available. This description will be included in a subsequent report.

4.0 System Status

Design is essentially complete. The heat exchanger and pump have been installed. The surge tank is being fabricated as well as the piping spool pieces.

The balance of piping fabrication, erection, and testing will proceed to completion.

G. Portable Disposable Demineralizer System

1.0 System Function and Design Objectives

Steam Generator "B" is presently contaminated with radioactive (fission) products. To ensure that the environment is not contaminated, this fluid must be cleaned up before the closed loop cooling system is placed into service. This cleanup capability will be provided by the Portable Disposable Demineralizer (PDD) system. After the initial cleanup is completed, water quality will be maintained by passing closed loop cooling system flow through additional portable demineralizers.

2.0 System Description

The PDD system will be located indoors near the northwest corner in the turbine building basement. The system will include disposable demineralizers each approximately 18 inches in diameter, 30 inches in height, and having a 1.5 cubic foot resin capacity. These demineralizers will be connected to the steam generator "B" closed loop cooling system, and will receive process water from the new closed loop pump discharge while returning the effluent to the pump suction. The number of demineralizers to be used will depend on the water quality and cleanup rate required.

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

The demineralizers are housed in portable shielded casks. 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 not yet available. This description will be included in a subsequent report.

4.0 Status

The demineralizers and shield casks have been fabricated.

Design will continue to completion. Piping connections are being installed in the steam generator "B" closed loop cooling systems.

230 234

H. Nuclear River Water System

1.0 System Function and Design Objectives

The river is the ultimate heat sink for the alternate decay heat removal (ADHR) system and the steam generator "B" closed loop cooling system.

To ensure system reliability, the nuclear services river water system was selected to supply the water.

The ADHR system requires approximately 3500 gpm, and the secondary services closed cooling water system that services the new steam generator "B" closed loop heat exchanger will require approximately 7000 gpm. These flow requirements will not be simultaneous.

2.0 System Description

Connections from the existing nuclear services river water supply and discharge headers are to be made. These connections will be made in the river water pump house and in the nuclear services river water piping between the river water pump house and fuel handling building. The former connection is for supply of river water to the "B" generator closed loop cooling scheme and the latter is to supply river water to the alternate decay heat removal system (ADHRS).

A jumper connection to supply nuclear services river water to the secondary services river water system will be made in the river water pump house. The connection will be made between valves NR-V3 and NR-V197 on the river water header and will be in accordance with ASME Section III requirements up to and in accordance with ASME Section III requirements up to and including the second isolation valve (two isolation valves are provided to segregate the safety class nuclear services river water system and the secondary services river water system ). The jumper connection will be made to the secondary services river water pump header downstream of valves SR-V2A, B, and C (see FSAR Figures 9.2-1 and 10.1-3).

3.0 System Operation

A detailed description of the system's operation is not yet available. This description will be included in a subsequent report.

4.0 Status

The connections for the alternate decay heat removal system to the nuclear services river water system have not been made and exact locations for the connections are not chosen.

The connection for the nuclear services river water system to the secondary services river water system have been designed and located. Installation has not started.

230 2--

## I. Secondary Services Closed Cooling Water System

### 1.0 System Function and Design Objectives

An intermediate closed cooling system is required to transfer heat from the new heat exchanger, in the steam generator "B" closed loop cooling system, during steam generator "B" closed loop cooling. This is necessary to prevent fouling which could result from river water passing through the shell side of the new DHR heat exchanger. The secondary services closed cooling water (SSCCW) system will be modified to satisfy this requirement.

### 2.0 System Description

No new major components will be added to the SSCCW system. Piping connections will be made between the closed cooling water supply and return headers and the new DHR heat exchanger. After modifications are completed, the SSCCW can be controlled and operated as originally intended during normal plant operations. However, during steam generator "B" closed loop cooling the SSCCW system will only serve the new steam generator "B" closed cooling loop heat exchanger, condensate pump motor bearings, and the service air compressors. All other components will be valved out. This operating mode will not exceed the performance limitations of the originally installed system.

### 3.0 System Operation

A detailed description of the system's operation is not yet available. This description will be included in a subsequent report.

### 4.0 Status

Design work is proceeding and piping sections are being fabricated.

Work yet to be completed is as follows:

Install cooling water pipe sections. Provide temporary cooling water supplies to the condensate pumps and air compressors during SSCCW system shutdown. Develop operating procedures for providing cooling water to the new steam generator "B" closed loop cooling heat exchanger.

230 236

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 detailed description of this system's operation is not yet available. Should the system be constructed, the operations description will be provided in a subsequent report.

4.0 System Status

Design is approximately 95% completed. The procurement and fabrication have been placed on hold, as the primary means of long-term decay heat removal will be by steaming in steam generator "A".

Design will proceed to completion. No further construction work is anticipated at this time.

K. Feedwater Bypass Lines

1.0 System Function and Design Criteria

A feedwater bypass line is required to ensure a continuous supply of feedwater to each of the steam generators while the respective long term cooling systems were being installed.

Each bypass line shall be sized to pass sufficient flow required to remove decay heat plus the heat generated by one reactor coolant pump during steam generator steaming.

The installation of the bypass lines shall not disrupt the flow of feedwater to the respective steam generator during steaming.

2.0 System Description

The bypass lines are sized to pass 50 gpm of condensate to each steam generator. The bypass lines to steam generators "A" and "B" are routed around feedwater heaters FW-J-6A and FW-J-6B respectively and tied into the feedwater system between existing valve vent and drain connections to permit a continuous supply of feedwater.

3.0 System Operation

The feedwater bypass lines may be used as long as the respective steam generator is in the steaming mode. The lines must be isolated or removed if the steam generator is placed in the water-solid, closed-loop cooling mode. No additional provisions for monitoring bypass line performance are necessary.

The feedwater bypass piping has been hydrostatically tested prior to service. There are no operational testing requirements for the feedwater bypass lines.

4.0 Status

The Feedwater Bypass Lines are installed and functional.

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. Therefore, this description will be provided in a subsequent report.

## 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 on site and assembly started. Motor control centers are on site. The valve vault excavation is completed and the control trailer purchased and delivered.

Temporary ADHR system shield building, permanent building and provision of electrical power and service water support is on hold. Tie-in of ADHR system to existing plant DHR system is on hold.

Piping supports are being designed and fabricated on site, control panels are being assembled and checked out at the vendor, the control trailer is being wired, air conditioned and insulated off site. Vault design is in progress. The DHCCW system skid will be brought on site and demonstration and rehearsal of fuel building wall penetration technique is in progress

230 240

## 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 functions:

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 changes in the RC system inventory.

### 2.0 System Description

The SRCPC system will include both makeup and letdown sub-systems. The makeup portion ties into the existing reactor coolant makeup pumps discharge header (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 the 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 of the RC system when operating in a water-solid mode. The letdown portion is provided to prevent RC system overpressurization. Overpressurization 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 undersirable and will only be considered as a last resort).

3.0 System Operation

A detailed description of this systems operation is not yet available. Therefore, it will be provided in a subsequent report.

4.0 Status

Conceptual design of the SRCPC makeup system is substantially complete. The surge tanks, charging water storage tank, borated water batching tank and transfer pump, and charging pumps are on site. Piping is being fabricated.

The following items have not yet been completed: Install and test the makeup system. Select a method for alternate letdown and begin design.

N. 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 should be completely independent of the existing Class IE busses.

The BOPEP busses should be loaded on a "manual only" basis in accordance with emergency operating procedures.

Modifications of power supplies associated with steam generator "A" cooling systems should be given priority of installation with respect to those for the steam generator "B" cooling systems.

The testing requirements for the BOPEP systems should 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 DHR 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. There is a cross-connection between the two busses which is only to be used for one buss to supply power to the other if there is a failure in one system and the diesel in the other system can handle the additional load demand.

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 air 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 2A and 2B, have been modified to connect the 2-3 (gray) and 2-4 (white) busses to their respective switchgear. Relays are provided at the circuit breakers 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 will have a 10 second delay.

* 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 auxiliaries, since sufficient power is not available from the diesel generators. This line is powered by a new 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 pumps 1E and 1F are 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. The cross-connection between the "gray" and "white" busses is normally locked out.

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 with similar control and monitoring capability locally at the diesel generators. The following instrumentation and controls are added in the control room:

a. Low lube oil temperature alarm (the diesel will start automatically on low lube oil temperature)

b. General "trouble" alarm for each diesel generator set

c. Generator output voltage indication

d. Generator line current indication

e. Generator power output indication

f. Diesel generator availability indication

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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 following work has not yet been completed:

Final approval of operating procedures

Determination of requirements for fuel oil heating

Detailing the provisions for the fire protection

Detailing the provisions for the supplementary fuel oil reserve

Installation of controls and instrumentation in the control room.

D. 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 constitutents 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. In line 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 being designed to cleanup and monitor any and all 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 controlled. The system is being optimized with respect to ALARA considerations.

Instrumentation and controls will be provided for monitoring system performance. Continuous activity monitoring of exhaust air will be provided.

2.0 System Description

Liquid Processing

The TMI Unit 1 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 will be a fan, an air cleanup filter train, and necessary ducting. The main HVAC components will be located external to the Unit 1 Chemical Cleaning Building but will be enclosed in its own housing.

230 247

3.0 System Operation

Operation of the system has not yet commenced because the system has not been totally fabricated.

4.0 Status

The design is essentially complete and fabrication and construction is approximately 45% complete.

P. Trash Compactor

1.0 System Function and Design Criteria

Additional compaction facilities were required due to the amount of compactible waste being generated. Therefore a system was needed to compact low level solid waste into 55 gal. drums for storage or shipment.

The system shall compact waste into 55 gal. Department of Transportation (DOT) drums meeting requirements for shipping LSA material.

2.0 System Description

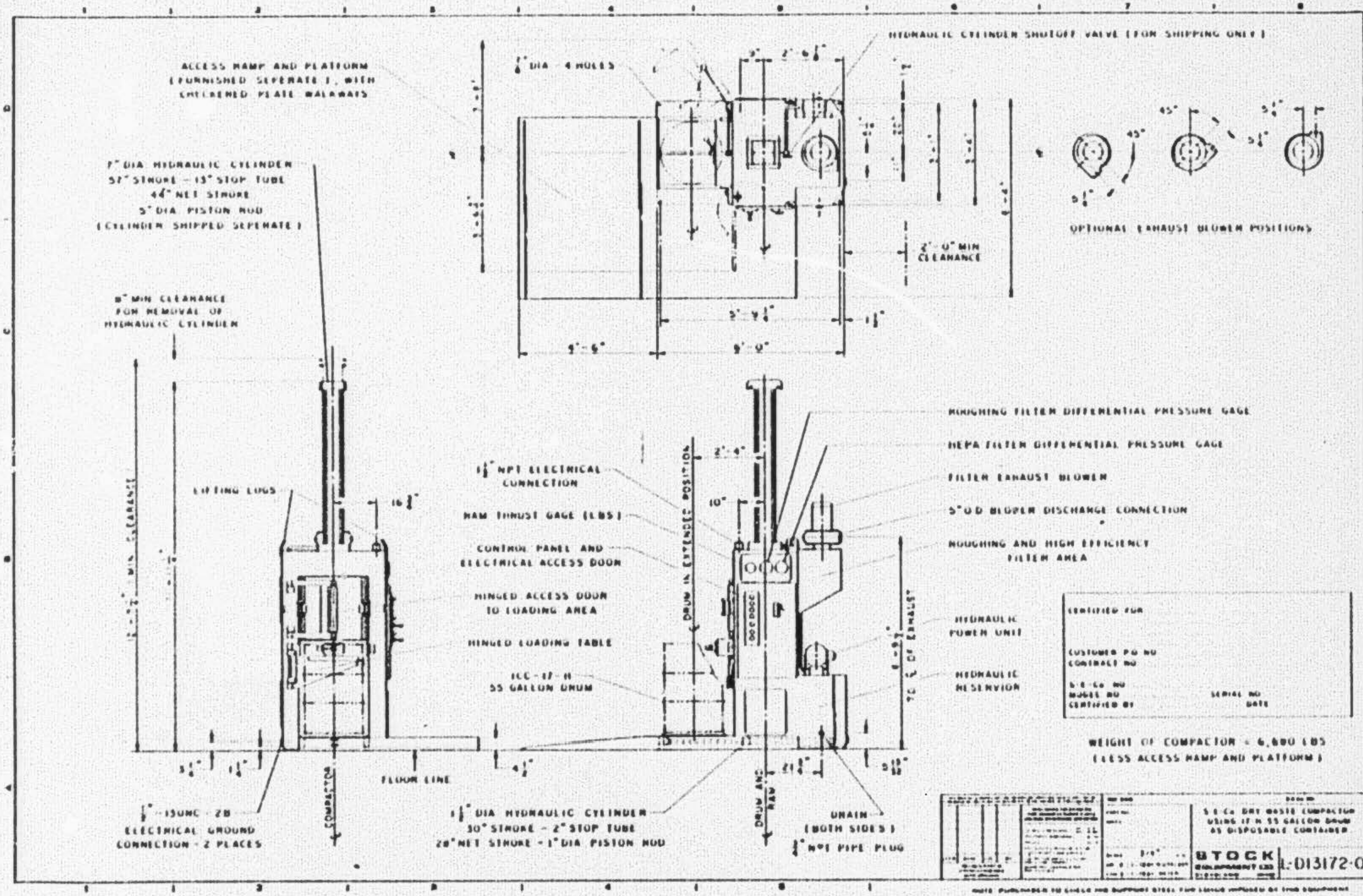
A Stock Equipment Company Model 2407 compactor was installed in the Unit 1 waste drumming area. A drawing of the unit is attached. The unit includes roughing and HEPA filters. The discharge of the unit is vented to the Aux. Building ventilation system which contains charcoal filters.

3.0 System Operation

Trash is compacted into 55 gal. DOT approved drums. Standard plant operating procedures have been revised to include the use of the new compactor. Use is limited to compactible dry waste only. No wood, metal or liquids are permitted. All bags of trash are surveyed prior to compaction; bags in excess of 500mr are not compacted.

4.0 Status

The trash compactor has been installed and is operational.


HYDRAULIC CYLINDER SHUTOFF VALVE (FOR SHIPPING ONLY)
ACCESS RAMP AND PLATFORM (FURNISHED SEPERATE), WITH CHECKERED PLATE WALKWAYS
DIA - 4 HOLES
7" DIA HYDRAULIC CYLINDER
57" STROKE - 13" STOP TUBE
44" NET STROKE
5" DIA PISTON ROD
(CYLINDER SHIPPED SEPERATE)
OPTIONAL EXHAUST BLOWER POSITIONS
2'-0" MIN CLEARANCE
8" MIN CLEARANCE FOR REMOVAL OF HYDRAULIC CYLINDER
LIFTING LUGS
NPT ELECTRICAL CONNECTION
RAM THRUST GAGE (LBS)
CONTROL PANEL AND ELECTRICAL ACCESS DOOR
HINGED ACCESS DOOR TO LOADING AREA
HINGED LOADING TABLE
ICC-17-H 55 GALLON DRUM
DRUM IN EXTENDED POSITION
ROUGHING FILTER DIFFERENTIAL PRESSURE GAGE
HEPA FILTER DIFFERENTIAL PRESSURE GAGE
FILTER EXHAUST BLOWER
5" O.D BLOWER DISCHARGE CONNECTION
ROUGHING AND HIGH EFFICIENCY FILTER AREA
HYDRAULIC POWER UNIT
HYDRAULIC RESERVIOR
FLOOR LINE
COMPACTOR
DRUM AND RAM
-13UNC-2B ELECTRICAL GROUND CONNECTION - 2 PLACES
DIA HYDRAULIC CYLINDER
30" STROKE - 2" STOP TUBE
28" NET STROKE - 1" DIA PISTON ROD
DRAIN (BOTH SIDES)
NPT PIPE PLUG
CERTIFIED FOR
CUSTOMER P.O NO
CONTRACT NO
SERIAL NO
DATE
WEIGHT OF COMPACTOR = 6,600 LBS
(LESS ACCESS RAMP AND PLATFORM)
DRY WASTE COMPACTOR USING 17 H 55 GALLON DRUM AS DISPOSABLE CONTAINER
STOCK
L-D13172-0

230 250

## Q. Staging Facilities for Dewatered Resins and Evaporator Bottoms

### 1.0 System Function and Design Criteria

This facility provides a staging area for dewatered spent resins and evaporator bottoms awaiting shipment offsite.

### 2.0 System Description

The concept is to provide shielded storage cells for containers per the attached sketch. The facility will be located in the south-east corner of the site, inside the dike. Design is also underway to provide a shielded container for use during transfer of liners from the truck to the storage cell.

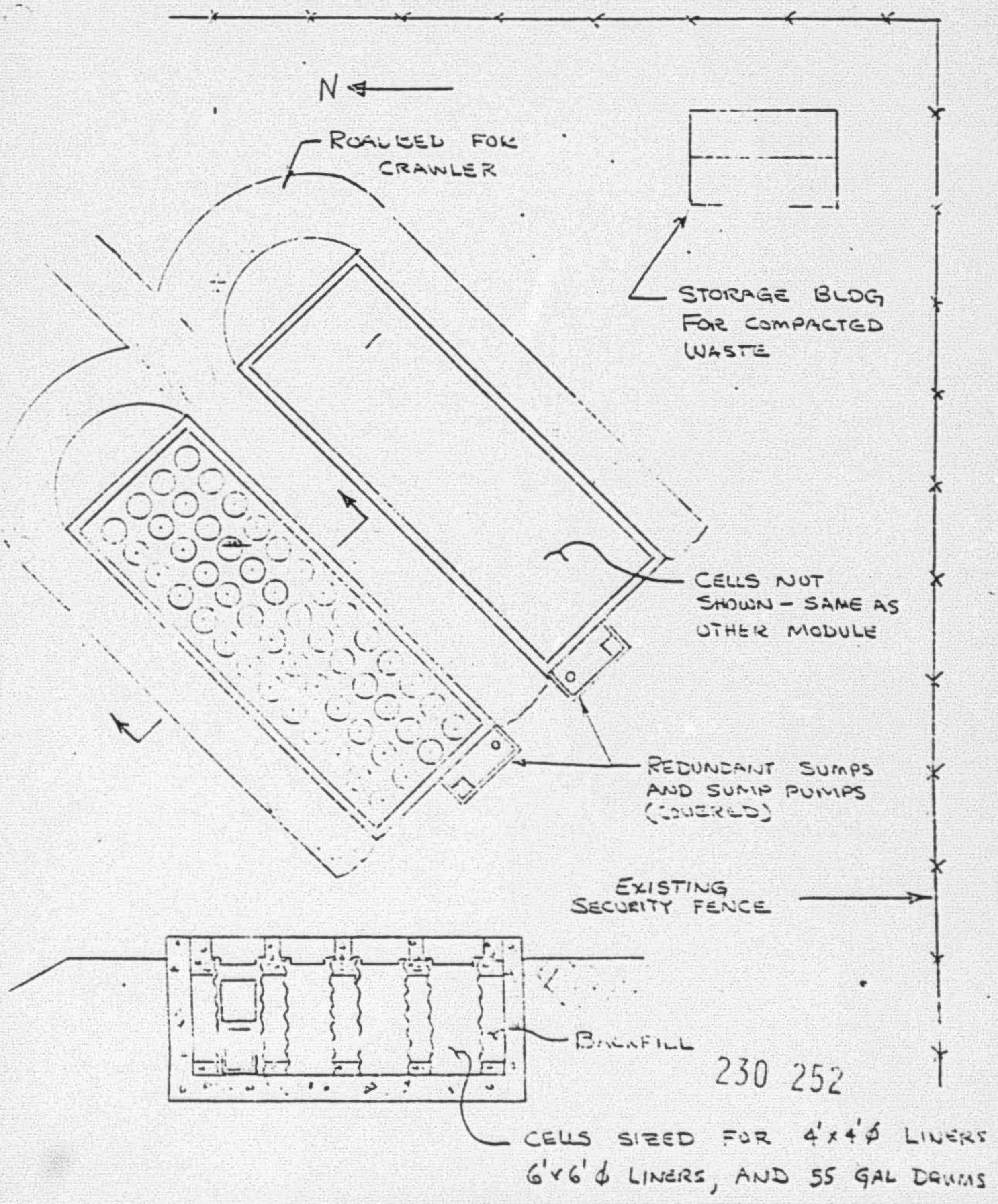
### 3.0 System Operation

The facility has not been constructed as of this report.

### 4.0 Status

The staging facility is in the design phase.

SKETCH OF PROPOSED ON-SITE STAGING AREA




230 252

R. Epicor-1 Liquid Rad Waste Treatment System

1.0 System Function and Design Objectives

Existing plant equipment was not designed to process the quantity or radioactivity of the waste generated subsequent to the Three Mile Island Unit 2 incident. A temporary custom-built externally located liquid radwaste treatment system, designated Epicor-1, was installed to supplement the station's existing system.

The temporary system is designed to remove suspended and dissolved radioactive contaminants from liquid waste. Treatment is achieved through filtration and demineralization.

Environmental protection is maintained by the use of features that provide leak and/or overflow protection.

The discharge of radioactive gases is minimized.

The system facilitates assembly and is flexible enough to conform to plant requirements and layout.

2.0 Description of the System

A. The Epicor-1 system consists of a demineralizer, prefilter and auxiliary hoses pumps and tanks. The process vessels are designed for disposal after they have been expended. The system takes its suction from either the Unit 1 Auxiliary Building Sump, Unit 1 Neutralizer tanks or the Unit 1 Fuel Handling Building Sump. Particulates are removed in a prefilter and dissolved contaminates are removed in the demineralizer. Water is returned to either of the Unit 1 Waste Evaporator Condensate Storage Tanks.

3.0 System Operation

The prefilter and the demineralizer are operated by filling them with the liquid waste and removing the treated effluent via internal laterals. The system is started up by actuating either CG-P-1A or B (feed pumps) to provide flow from any of the three Unit 1 sources to the prefilters. When the liquid level in the prefilter reaches the high level, level switch CG-P-2 (prefilter decant pump) is manually started to begin filling the demineralizer. Inlet and outlet flows are balanced to maintain a constant level in the prefilter. When the liquid level in the demineralizer reaches the high level, level switch CG-P-3 (demineralizer decant pump) is manually started. The demineralizer outlet flow is adjusted to maintain a constant level in the demineralizer. The effluent from CG-P-3 flows to one of the two effluent tanks labelled HAL 1 or 2.

The operation of the system is continued until:

a. There is a lack of source water

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b. The effluent quality is poor

c. The radiation level on the exterior of a process vessel exceeds a predetermined value. (Based on shipping requirements)

d. The effluent tank is full, or

e. There is a system malfunction.

Liquid stored in the effluent tanks may be reprocessed through the demineralizer (to provide additional treatment) or routed to the Unit 1 Waste Evaporator Condensate Storage Tanks in preparation for discharge to the environment.

Shutdown is initiated by stopping the feed pump and partially dewatering the process vessles. Reclaimed water from the Unit 1 Auxiliary Building is then used to flush the process hoses. Flush water is then displaced by blowing air through the hoses.

Process vessels are completely dewatered prior to being removed from service. This is done by operating either of the decant pumps (CG-P-2 or 3) until they loose suction.

In the event of a leak, high-high level alarm, or other system malfunction the main air supply to the pumps is terminated immediately causing all flow to stop.

Precautions have been taken to minimize the possibility of spilling radioactive liquid and to contain any spills if they occur. Primary emphasis has been given to creating and maintaining a leak-tight system. All fittings and hoses installed have pressure ratings that exceed the maximum discharge pressure of the pumps used. All discharge hoses have a pressure rating of 600 psig or greater. All hoses and fittings are hydrostatically tested prior to use. Pump diaphragms are designed to rupture at pressures greater than 125 psig. The maximum available air pressure to drive the pumps is 100 psig (thus protecting diaphragm integrity).

Any leaks that may occur will be contained in a multi barrier containment system. All hose connections are taped and wrapped with plastic to contain drips from fittings. Leaks within the Unit 1 Auxiliary Building would be contained in the floor drain and sump system. Small leaks external to the building would be contained in a plastic lined hose tray. A series of block containment dikes around the process vessels would contain leakage in the vicinity of the process vessles. Leaks from the process vessel would be contained in a circular steel container in which the vessel is set prior to operation.

## 4.0 Status

The Epicor-1 Liquid Rad Waste Treatment System is installed and operational.

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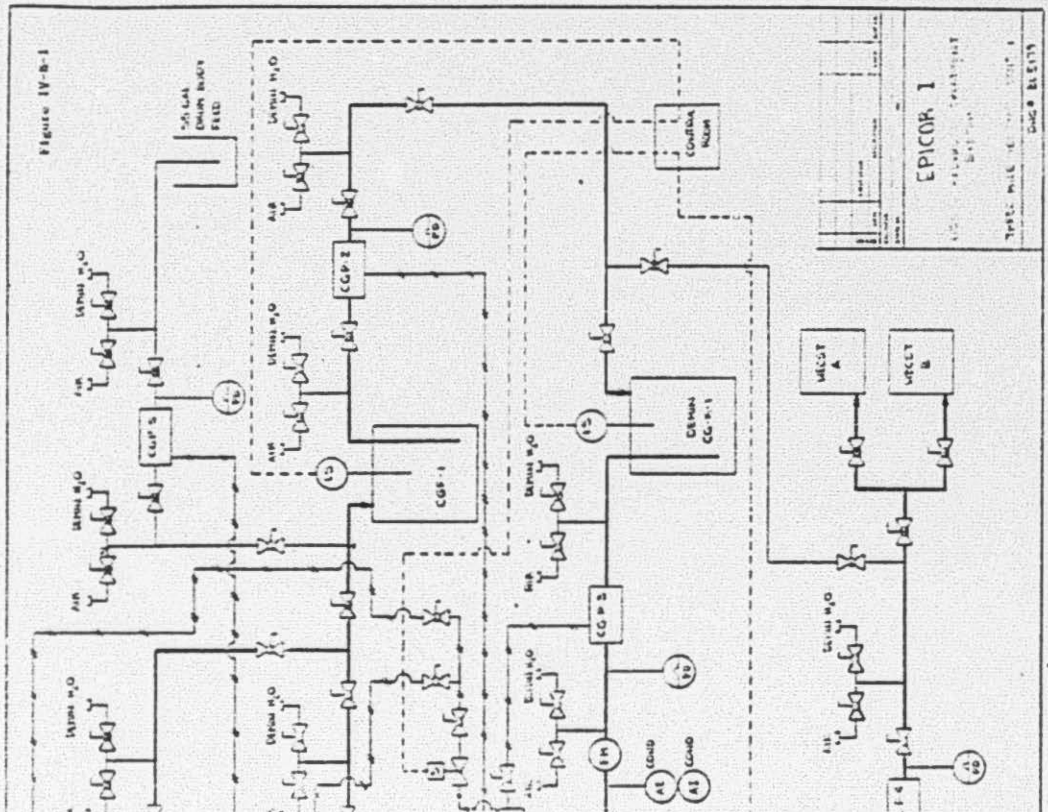


230 255

## IV. Radiological Monitoring

Due to the nature of this section of the report, only changes and additions to the information presented here will be included in next months report. The Final Report, however, will provide a comprehensive review of the Radiological Monitoring done following the March 28, 1979 incident at TMI-2.

## IV. Radiological Monitoring

### A. Introduction

#### 1. Sources of Data

This report uses data collected in monitoring programs operated by Metropolitan Edison. At this writing, an extensive amount of data collected by others has not yet been evaluated in detail. However, the Metropolitan Edison program is comprehensive enough and sensitive enough to make an accurate assessment of radiological impact.

a. Measured Releases

Metropolitan Edison operates a monitoring program designed to permit identification and quantification of radioactive isotope releases in liquid and gaseous effluents from the station. These programs include automatic continuous monitoring of effluent streams using equipment with control room alarms. This monitoring is complemented by effluent sampling and laboratory analysis. Grab sampling is used for batch liquid releases, and continuous sampling with periodic laboratory analysis is used for radioiodines and particulates in gaseous effluent. Generally speaking, the automated monitoring equipment is considered most useful because of its early warning features but is, in most cases, less accurate and/or less sensitive than sampling and analysis for quantifying releases. For these reasons, sampling and analysis usually serves as the basis for most quantitative assessments of releases of radioactivity in effluents. Effluent measurements are described more extensively later in this section.

b. Estimated Release Rates

When data on releases were not available, estimates were made taking into consideration meteorological data, data from the radiological environmental monitoring program, and the data from area radiological monitors inside the Unit 2 Auxiliary and Fuel Handling Buildings.

c. Meteorological Data

Metropolitan Edison maintains a meteorological tower located at the north end of the island to support normal plant operation. Data from this tower have been continuously available via redundant sensors since the accident. These data were used to estimate noble gas releases and to compute population doses.

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d. Radiological Environmental Monitoring Program

For about five years, Metropolitan Edison has operated a monitoring program designed to evaluate the radiological impact of TMI station operations by sampling and analyzing media from the aquatic, terrestrial and atmospheric environments in the vicinity of the station (within 5 to 10 miles). The program was intensified immediately following the accident, in accordance with emergency response plans.

2. Summaries of Analyses

Available data were used to assess the radiation doses received by individuals and affected populations off-site, in the period following the start of the accident.

In addition, comprehensive dose analyses have been performed. Aquatic environmental monitoring data have been used to assess the whole body and thyroid doses to maximum exposed individuals and the population consuming water and fish from the Susquehanna River. Maximum individual doses have been assessed for recreational use of the river (swimming, boating, etc.). Environmental monitoring data have also been used to assess maximum individual whole body doses from noble gases released to the atmosphere.

These data were also used to estimate the release rates which were used in conjunction with meteorological data and population distribution data to assess the population whole body dose from noble gases released to the atmosphere, and individual whole body doses at locations other than those monitored. Environmental monitoring data have also been used to assess maximum individual thyroid doses from inhalation of air and consumption of milk containing iodine-131. In addition, measured release rates were used in conjunction with meteorological data, population distribution data, and cow distribution data to estimate the population thyroid doses from inhalation of air and consumption of milk containing iodine-131.

B. Off-Site Liquid Releases and Exposures

1. Environmental Measurements

Review of sampling data suggests that a total of about 0.25 Ci of Iodine-131 has been released as a result of the accident, most of it from March 31, 1979 through April 2, 1979.

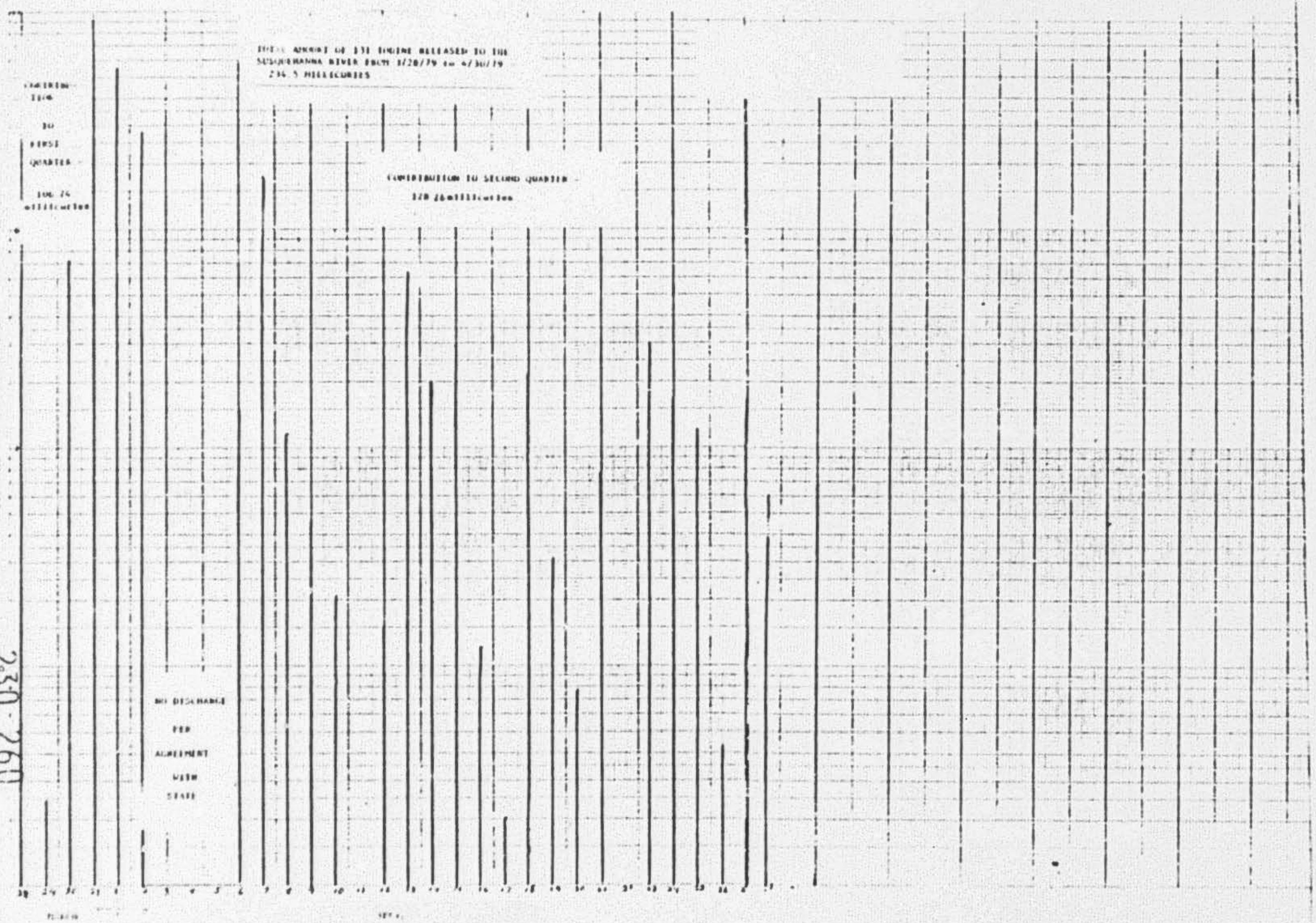
The radiological environmental monitoring program conducted by Metropolitan Edison Company includes analysis of river surface water, finished water from treatment plants, and aquatic biota. Except for three samples collected, 3/31, 4/1 and 4/2 at station 7G1, the Columbia Water Plant, which showed levels of iodine (9.4, 0.72 and 0.66 pCi/l) slightly above minimum detectable concentrations, no gamma-emitting isotopes other than low levels of naturally occurring potassium-40 and radium-226 were detected.

Tritium and gross beta concentrations were within normal ranges. The low concentrations measurable at Station 7G1 are consistent with estimated iodine release rates for the same period, shown in Figure IV-B-1, assuming liquid effluent has been fully mixed in the river prior to sampling downstream.

2. Estimated Off-Site Exposures

Radiation doses resulting from these concentrations are extremely low, a few thousandths of one millirem for a person drinking water or eating fish from the river or using the river for swimming, boating, or shoreline activities.

230 259


TOTAL AMOUNT OF 131 IODINE RELEASED TO THE SUSQUEHANNA RIVER FROM 3/28/79 to 4/30/79
234.5 MILLICURIES
CONTRIBUTION TO FIRST QUARTER - 106.24 millicuries
CONTRIBUTION TO SECOND QUARTER - 128.26 millicuries
NO DISCHARGE PER AGREEMENT WITH STATE

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C. Off-Site Noble Gas Releases and Exposures

This section discusses the analyses done to estimate off-site doses due to releases of noble gas fission products. The section is organized into two subsections dealing respectively with environmental measurements, and off-site exposures.

1. Environmental Measurements

The source term used in the following assessments was arrived at based on information obtained from the environmental monitoring TLD's. The source term determined in this fashion was then used to assess the dose received by the population in unmonitored areas.

Metropolitan Edison Company conducts a routine environmental radition monitoring program including use of stationary thermoluminescent dosimeters (TLD's) which measure integrated dose. They are in place at all times at 20 locations as shown on Figure IV-C-1. Eight of these locations are on site. The other twelve are off-site, including locations in Goldsboro and Middletown. Most are within several miles of the plant, but a few are located up to 15 miles away. These dosimeters were in place in the field at the time the accident occurred. Dosimeters in the field were replaced with fresh dosimeters every one to three days following the accident, and the collected dosimeters were evaluated to determine trends for dose rate as well as the dose accumulated since the beginning of the accident. These data represent a comprehensive measurement of doses due to noble gas releases at the locations monitored. Table IV-C-1 provides a complete summary of TLD measurements through April 30, 1979.

Shortly after the declaration of an emergency, mobile monitoring teams were dispatched by Metropolitan Edison. A helicopter was used to assist the monitoring teams since the on-site meteorological tower indicated winds toward the west over the river.

These teams were equipped with instruments which measured dose rates from airborne radioactive material (primarily noble gases with Xe-133 dominant) and with air samples which were capable of collecting airborne radioactive materials other than noble gases for later laboratory analysis. (Aside from noble gases, the only isotope contributing significantly to off-site dose has been Iodine-131. Its contribution results from concentration in the thyroid gland due to transport from air to grass to milk to thyroid and to some extent, from inhalation.)

During the first few days following the accident, release rates fluctuated and frequent wind shifts often occurred which caused radiation levels to fluctuate with time at any single location. The emergency response survey teams had to move from place to place following the transport of airborne radioactivity. Because of the fluctuations in radiation levels and the short monitoring periods at any one location, data collected by survey teams are not the best available for the determination of cumulative doses and they were not used in this assessment except in attempts to determine when noble gas releases were significantly higher than baseline values. This work is continuing and will be augmented using DOE helicopter data when it becomes available.

2. Estimate Off-Site Exposures

Mathematical models for estimating doses to individuals and populations normally use isotope release rates and meteorological parameters. Estimated release rates were used to estimate doses to individuals at locations not monitored by TLD's and to the population witinn 50 miles of the plant. For these calculations the atmospheric dispersion model which had been previously used only for estimates at TLD locations close to the plant was extended to a distance of 50 miles in each of 16 direction sectors. The population dose was computed using the dispersion model and site meteorological data to compute the whole body dose each hour at 10 locations downwind in the sector in which the wind was blowing. Doses were estimated for the period after the accident extending to April 30, 1979 and multiplied by the population in each of these 10 distances for each direction. The results of this analysis indicate that the aggregate whole body dose of the population within 50 miles (about 2 million people) was about 2500 person-rems from noble gases released through 4/30/79, not considering the effect of shielding due to housing or other structures which could reduce dose estimates by a factor of 2 to 3. The population estimated for 1980 was used.

The highest off-site integrated whole body dose measured at any TLD location through April 28, is about 43 millirems above background at a location about 1200 meters NNE from the plant. The accumulated doses measured at the Goldsboro and Middletown TLD monitoring stations over the same period were less than 10 millirem above background. The latter figure is an upper limit. The best estimates of about 5 millirem at each location, has some uncertainty because of the difficulty of separating normal fluctuations in background dose from accident contributions of similar or smaller magnitude. (This uncertainty does not affect the maximum dose of 43 mrem for which the accident contribution was substantially greater than fluctuations in natural background for the measurement period.

By contrast, whole body dose calculations at these same locations using the estimated noble gas releases resulted in doses in 30 mrem 1200 meters NNE: 4.0 mrem at Goldsboro; and 10.0 mrem at Middletown.

It should be noted that the maximum dose from noble gases, about 43 millirem, is less than the incremental dose a resident of Harrisburg would get by moving to Denver, Colorado for one year because the natural background dose rates in Denver are greater than those in Harrisburg. The average dose received by population was about 1 mrem, one-third of the dose received in a transcontinental subsonic plane flight.

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TABLE IV-C-1

METROPOLITAN EDISON - THREE MILE ISLAND

TLD DATA

(Period of Exposure 79032807 - 79040613)*

| Map No. | Station | Distance (M) | Direction | Total Measured Dose ** (rem) | Total Calc. Dose (rem) |
|---|---|---|---|---|---|
| 2 | 1S2 | 640 | N | 0.095 | 0.10 |
| 26 | 1C1 | 4180 | N | 0.006 | 0.010 |
| 3 | 2S2 | 1130 | NNE | 0.043 | 0.030 |
| 5 | 4S2 | 380 | ENE | 0.102 | 0.070 |
| 13 | 4A1 | 800 | ENE | 0.039 | 0.020 |
| 37 | 4G1 | 16100 | ENE | 0.004 | 0.001 |
| 6 | 5S3 | 320 | E | 0.072 | 0.045 |
| 14 | 5A1 | 640 | E | 0.015 | 0.015 |
| 34 | 7F1 | 14500 | SE | 0.010 | 0.0006 |
| 38 | 7G1 | 24100 | SE | 0.012 | 0.0004 |
| 28 | 8C1 | 3700 | SSE | 0.008 | 0.008 |
| 8 | 9S2 | 640 | S | 0.023 | 0.040 |
| 39 | 9G1 | 20900 | S | 0.005 | 0.003 |
| 23 | 10B1 | 1770 | SSW | 0.005 | 0.004 |
| 9 | 11S1 | 160 | SW | 0.010 | 0.020 |
| 24 | 12B1 | 1270 | WSW | 0.004 | 0.004 |
| 10 | 14S2 | 640 | WNW | 0.137 | 0.100 |
| 40 | 15G1 | 24100 | NW | 0.003 | 0.003 |
| 11 | 16S1 | 320 | NNW | 1.033 | 0.630 |
| 17 | 16A1 | 640 | NNW | 0.901 | 0.400 |

*Includes all TLD readings from the radiological environmental monitoring program.

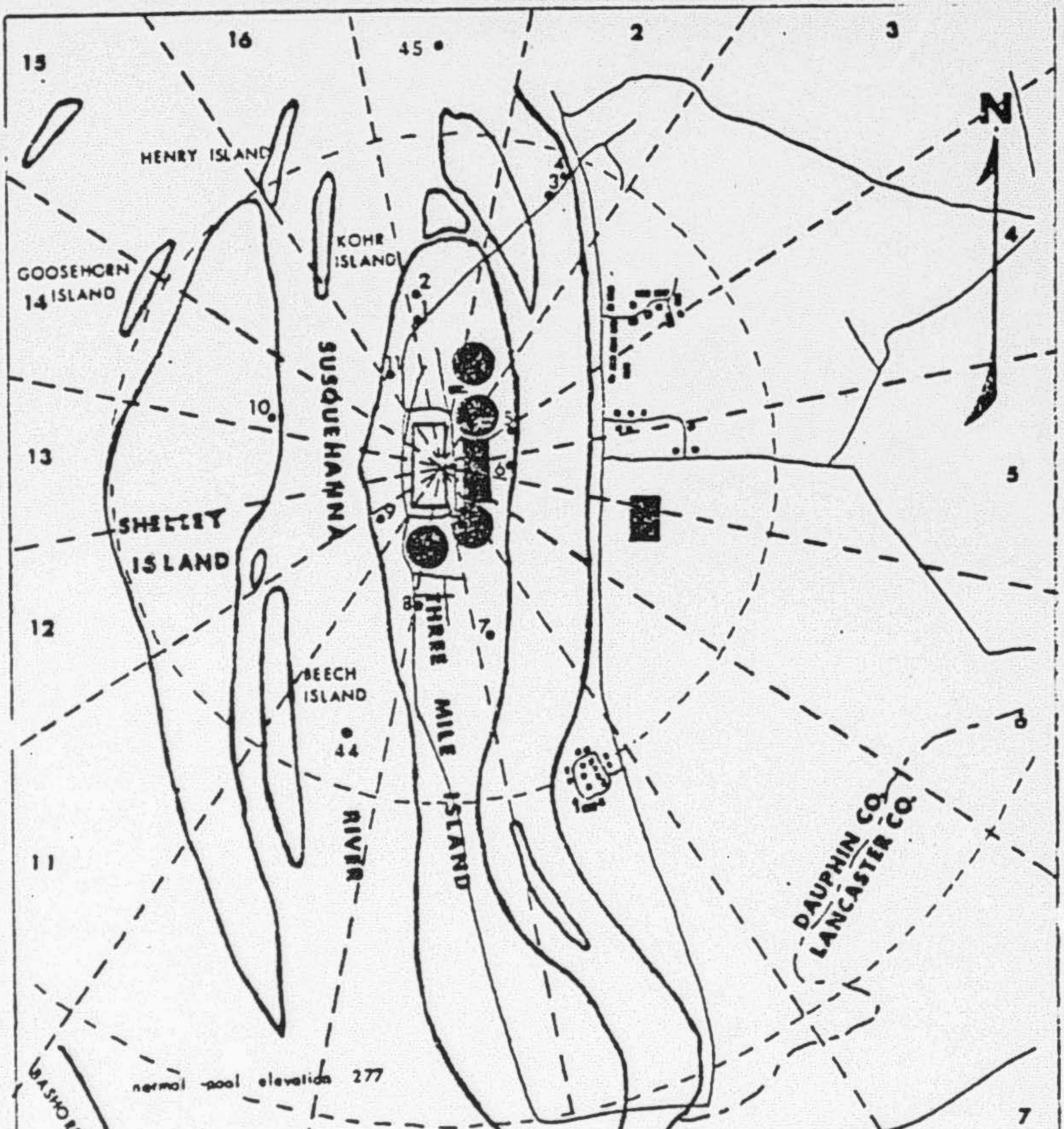
**Background radiation assumed to be 0.007 mrem/hour has been subtracted. Values less than about 10 mrem have substantial uncertainty because gross values exceed background by only a small margin. It is certain, however, that 10 mrem is an upper limit for these cases.

TABLE IV-C-2

(Ci)
Estimated Quantities of Each Isotope for
Release Periods Corresponding to TLD Measurements

| Isotope | 79032807-79032917 | 79032918-79033117 | 79033118-79040315 | 79040316-79040613 | 79040614*-79043024 | Total Ci |
|---|---|---|---|---|---|---|
| Xe-133 | 5.2 E6 | 2.7 E6 | 1.0 E6 | 2.8 E5 | 1.5 E4 | 9.2 E6 |
| Xe-133m | 7.6 E5 | 3.0 E5 | 6.8 E4 | 8.3 E3 | 0 | 1.1 E5 |
| Xe-135 | 1.4 E6 | 6.5 E4 | 0 | 0 | 0 | 1.5 E6 |
| Xe-135m | 1.7 E5 | 0 | 0 | 0 | 0 | 1.7 E5 |
| Kr-88 | 5.5 E4 | 0 | 0 | 0 | 0 | 5.5 E4 |
| | 7.6 E6 | 3.1 E6 | 1.1 E6 | 2.8 E5 | 1.5 E4 | 1.1 E7 |

*The last three weeks of the month are combined into one group since the contribution is less than 1% of the total.



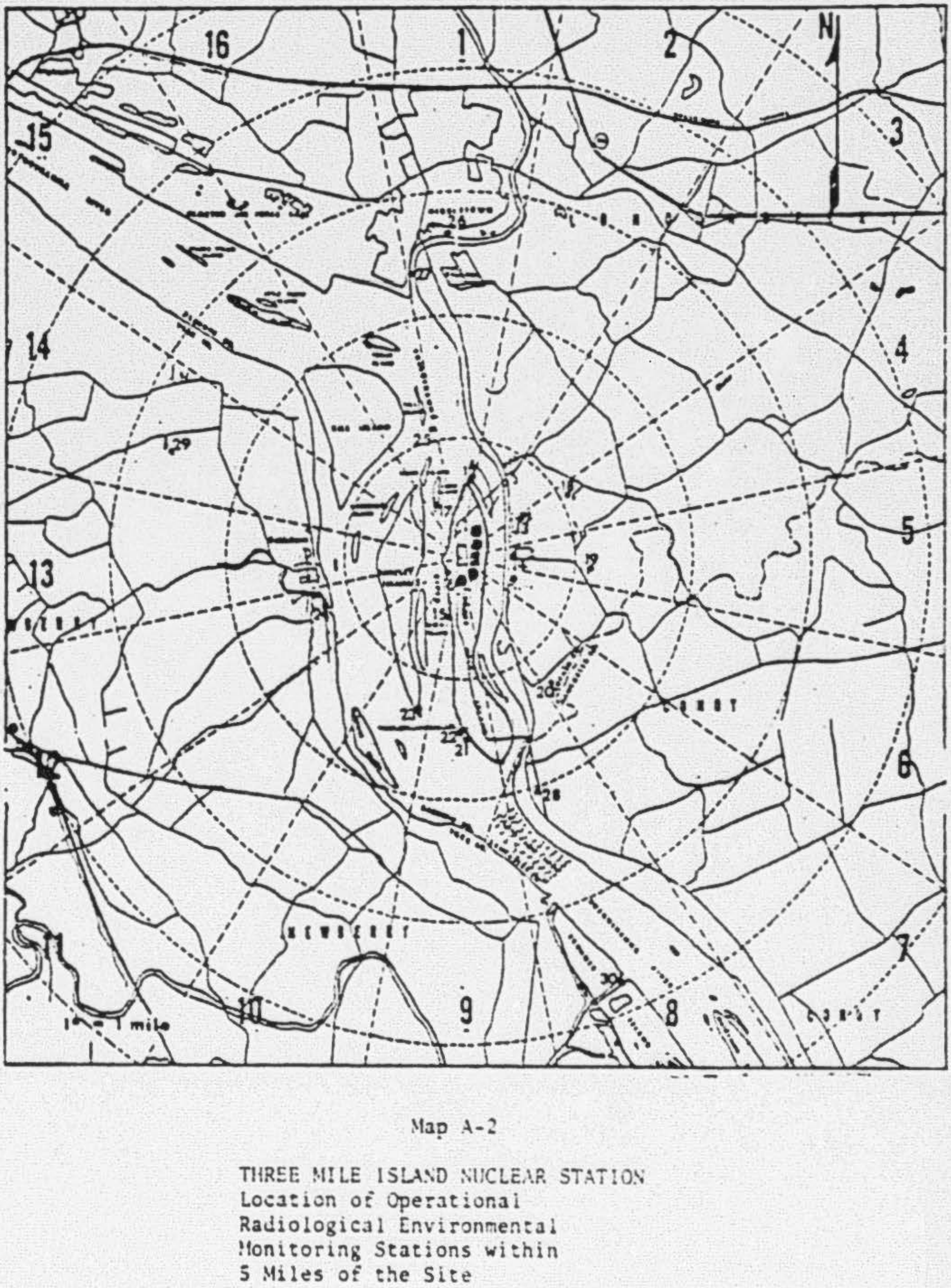

2.5" = 1 mile

Map A-1

THREE MILE ISLAND NUCLEAR STATION
Location of Operational
Radiological Environmental
Monitoring Stations within
the Site Boundaries

230 265

Figure IV C-1.1




Map A-2

THREE MILE ISLAND NUCLEAR STATION
Location of Operational
Radiological Environmental
Monitoring Stations within
5 Miles of the Site

230 266

TABLE A-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM SAMPLING LOCATIONS

| Location Code | Map No. | Description |
|---|---|---|
| 1S2 | 2 | 0.4 miles N of site, N. Weather Station |
| 2S2 | 3 | 0.7 miles NNE of site, on light pole in middle to North Bridge |
| 4S2 | 3 | 0.3 miles ENE of site on top of dike, East Fence |
| 5S3 | 6 | 0.2 miles E of site on top of dike, East Fence |
| 9S2 | 8 | 0.4 miles S of site on South Beach of Three Mile Island |
| 11S1 | 9 | 0.1 miles SW of site, west of Mech. Draft Towers on dike |
| 14S2 | 10 | 0.4 miles WNW of site at Shelly's Island picnic area |
| 16S1 | 11 | 0.2 miles NNW of site at gate in fence on W. side of TMI |
| 1A2 | 12 | 0.7 miles N of site at north tip of Three Mile Island |
| 4A1 | 13 | 0.5 miles ENE of site on Laurel Rd., Met-Ed, Pole #668-OL |
| 5A1 | 14 | 0.4 miles E of site on north side of Observation Center Bldg. |
| 9A2 | 15 | 0.5 miles S of site below Discharge Pipe |
| 11A1 | 16 | 0.2 miles SW of site off Discharge Pipe |
| 16A1 | 17 | 0.4 miles NNW of site on Kohr Island |
| 4B1 | 18 | 1.1 miles ENE of site, west of Gringrich Road |
| 5B1 | 19 | 1.0 miles E of site on Peck Road |
| 7B3 | 20 | 1.6 miles SE of site on east side of Conewago Creek |
| 9B1 | 21 | 1.5 miles S of site, above York Haven Dam |
| 9B2 | 22 | 1.4 miles S of site, north of York Haven Dam |
| 10B1 | 23 | 1.1 miles SSW of site on south beach of Shelly's Island |
| 12B1 | 24 | 1.6 miles WSW of site adjacent to Fishing Creek |
| 16B1 | 25 | 1.1 miles NNW of site below Fall Island |
| 1C1 | 26 | 2.6 miles N of site at Middletown Substation |
| 1C3 | 27 | 2.3 miles N of site at Swatara Creek |
| 8C1 | 28 | 2.3 miles SSE of site |
| 14C1 | 29 | 2.7 miles WNW of site near intersection of Rts. 262 and 392 |
| 8E1 | 30 | 4.1 miles SSE of site at Brunner Island |
| 1F1 | 31 | 6 miles N of site at Hummelstown Substation on Fiddler's Elbow Rd. |
| 5F1 | 32 | 9 miles of site on East Ridge & Greentree Rds. |
| 5F2 | 33 | 5.8 miles E of site at Masonic Home |
| 7F1 | 34 | 9 miles SE of site at Drager Farm off Engle's Tollgate Rd. |
| 15F1 | 35 | 8.7 miles NW of site at Steelton Municipal Water Works |
| 2G1 | 36 | 2 miles NNE of Hershey on Rt. 39 Hummelstown |
| 4G1 | 37 | 10 miles ENE of site at Lawn - Met-Ed Pole #J1813 |
| 7G1 | 38 | 15 miles SE of site at Columbia Water Treatment Plant |
| 9G1 | 39 | 13 miles S of site in Met-Ed York Load Dispatch Station |
| 15G1 | 40 | 15 miles NW of site at W. Fairview Substation |

*All distances are measured from a point that is midway between the Reactor Buildings of Units 1 and 2

230 267

Figure IV C-1.3

D. Off-Site Iodine and Particulate Releases and Doses

1. Environmental Measurements

In support of routine plant operations, Metropolitan Edison conducts an environmental monitoring program which involves collection of several different types of samples. Included are continuous air samples of iodines and particulates, vegetation samples and milk samples. This program has continued with a higher sampling frequency since the accident. 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.

2. Estimated Off-Site Exposures

Estimates of off-site exposures to iodines are accomplished using the atmospheric dispersion model and the measured effluent release rates along with on-site meteorological data. Of the particulate and iodine isotopes released, only I-131 is of significance in off-site dose calculations.

The dose from I-131 results from concentration of this isotope in the thyroid gland following intake from inhalation of air containing iodine and ingestion of milk containing iodine. The iodine in milk results from deposition on pasture grass after release.

An analysis using measured release rates and meteorological data has been made to estimate the average I-131 concentration at all off-site locations near the plant. Results show the highest average concentration from March 28, 1979 to April 30, 1979 to be $6.3 \times 10^{-12}$ uCi/cc in the SE direction. If an adult had resided there through the accident, the inhalation dose would have been about 6.3 mrem. No other age group would be more than 50% higher. A series of five short term calculations were made to estimate peak iodine concentrations off-site. Results showed the highest concentration was $3.7 \times 10^{-11}$ µCi/cc at 600 meters to the NNW averaged over the initial 24 hours of the accident. The highest off-site instantaneous concentration was determined to be $8.0 \times 10^{-10}$ µCi/cc on April 15, located 600 meters to the SE using a centerline dispersion formula. A comparison between calculated air concentration values and air samples for three different periods at several locations shown in Table IV -E-2 indicates generally good agreements (i.e., within a factor of about 3 in 70% of the cases).

Population doses through ingestion of milk produced in the site region has been estimated using atmospheric dispersion model consistent with Reg. Guide 1.111 out to 50 miles.

230 268

Detailed cow inventories were available out to a distance of 5 miles. However, beyond 5 miles, county milk production rates were relied upon to estimate cow populations assuming each cow produces 34 pounds of milk per day. The milk production rates for the 50-mile radius suggest a population of about 300,000 dairy cows with a population density in sectors to the ENE, E, ESE, and SE approximately 2.5 times the density in other sections (about 75 cows per square mile versus 30 cows per square mile). There is evidence from cow population surveys within five miles that stored feed is an important fraction of the dairy cattle diet. Furthermore, data on land use in three counties near the plant indicate that only 5 to 10 percent of the land is used to support dairy cows. At the yield specified in Regulatory Guide 1.109, that pasture land could only provide about 20% of the diet for 300,000 cows. In addition, warnings had been issued to keep cows in barns during the period following the accident. For these reasons, it has been assumed that pasture grass accounted for 30% of the average cow's diet. In addition, the portion of iodine that was released in organic form, which does not deposit on grass, was measured periodically and found to be less than 30 percent on the average, and has been taken into account for this analysis. Iodine concentrations in milk were determined using the models which serve as a basis for Regulatory Guide 1.109 relationships and parameter values given in same guide. The population dose was estimated by calculating the average concentration in milk produced within 50 miles, accounting for dilution due to atmospheric dispersion. All milk produced was assumed to be consumed.

Results of the above calculations indicated the potential for population thyroid doses to be 900 person-rems.

The above dose estimates have been made independent of measured iodine concentrations in air and milk. Measurement results indicate peak iodine levels in milk to be less than 110 pCi/l with an average from March 28 through April 27 at any one sample location of 29 pCi/l. These figures apply to goat milk collected at location 1B1 about one mile north of the plant. The comparable values for cow milk are 21 pCi/l peak and 2.8 pCi/l average at location 7B1, 1.4 mi. SE. If an infant had been consuming this milk through April 27, 1979, his dose is estimated to be 1.2 millirems from cow milk or 11 millirems from goat milk. However, the goat milk is not now being used for human consumption.

Airborne sample results indicate that average airborne iodine concentrations at any location through April 24, 1979 were 2.9 pCi/m$^3$, which would result in an inhalation dose of 2.7 millirems.

The doses from releases of radioactive iodine have been very low. As a matter of perspective, a round-trip transcontinental plane flight results in an incremental dose of 3 millirems and a resident of the area in the vicinity of the plant normally receives about 8 millirems each month from naturally-occurring radiation.

TABLE IV D-1

Comparison of Measured and Calculated Concentrations
of Airborne Iodine for Three Time Periods

Measurement Period 3/28 - 3/29

| Station | Distance (m) | Direction | Calculated I-131 in Air ($pCi/m^3$) | Measured I-131 in Air ($pCi/m^3$) |
|---|---|---|---|---|
| 1S2 | 640 | N | 8.1 | 3.0 |
| 1C1 | 4180 | N | 3.1 | 2.2 |
| 12B1 | 1270 | W, WSW | 15.0, 5.7 } 10.3 | 7.1 |

Measurement Period 3/29 - 3/31

| Station | Distance (m) | Direction | Calculated I-131 in Air ($pCi/m^3$) | Measured I-131 in Air ($pCi/m^3$) |
|---|---|---|---|---|
| 1S2 | 640 | N | 2.3 | 22.6 |
| 1C1 | 4180 | N | 3.3 | 12.7 |
| 5A1 | 640 | E | 0.54 | 20.3 |
| 8C1 | 3700 | SSE | 8.2 | 20.1 |
| 12B1 | 1270 | W, WSW | 26.0, 0.0 | 23.9 |

Measurement Period 4/15 - 4/18

| Station | Distance (m) | Direction | Calculated I-131 in Air ($pCi/m^3$) | Measured I-131 in Air ($pCi/m^3$) |
|---|---|---|---|---|
| 5A1 | 640 | E | 2.5 | 8.4 |
| 8C1 | 3700 | SSE | 1.1 | 0.057 |
| 7F1 | 14500 | SE | 0.31 | 0.39, 0.17 |

TABLE IV D-2

SMOOTHED IODINE RELEASE RATE DATA USED
IN DOSE ASSESSMENTS

| Start Date (Yr. Mo. Da. Hr.) | I-131 Release Rate µCi/sec |
|---|---|
| 79032804 | 4.2 [1)] |
| 79032819 | 22.7 |
| 79033022 | 2.7 |
| 79040106 | 9.7 |
| 79040303 | 2.3 |
| 79040319 | 7.0 |
| 79040519 | 0.43 |
| 79040615 | 3.7 |
| 79040706 | 6.9 |
| 79040803 | 12.7 |
| 79040909 | 0.46 |
| 79041016 | 1.3 |
| 79041119 | 2.2 |
| 79041323 | 4.1 |
| 79041410 | 6.6 |
| 79041505 | 8.6 |
| 79041508 | 14.0 |
| 79041518 | 6.0 |
| 79041616 | 11.0 |
| 79041624 | 3.0 |
| 79041716 | 5.5 |
| 79041804 | 7.5 |
| 79041808 | 2.0 |
| 79041914 | 5.5 |
| 79042022 | 1.5 |
| 79042213 | 2.5 |
| 79042304 | 1.0 |
| 79042312 | 3.8 |
| 79042316 | 1.5 |
| 79042406 | 0.80 |
| 79042516 | 0.50 |

1) Iodine release rates are roughly constant over the period from one start time to the next

CORRECTED GRAPH

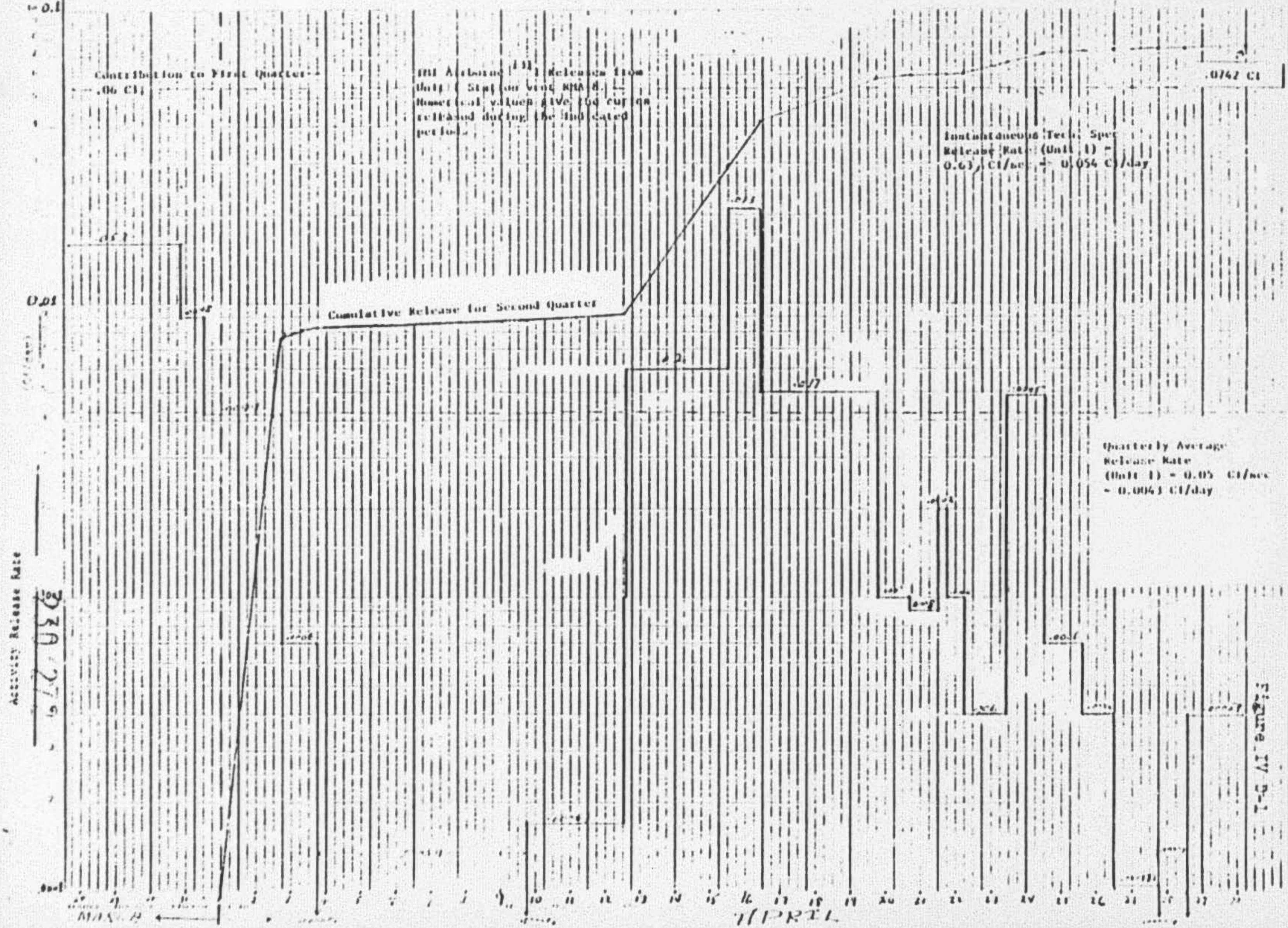



Figure IV D-1

830 279