50-320

Miscellaneous documents from the Executive Management Team

3/28/79 - 4/10/79

7906130242

R

175 1017

EMT DIRECTORS

WATCH OUT FOR OVER-FATIGUE, THIS INCLUDES YOURSELF!

LET'S STICK TO THE ESTABLISHED SCHEDULE, SO THAT IF WE NEED TO "SURGE" FOR AN UNEXPECTED DEVELOPMENT, WE'LL ALL BE IN BEST POSSIBLE SHAPE. LVG

175 108

NOTICE -
HEW (BRH) has withdrawn
their 24 hr. rep at NRC Opns. Cntr.

175 1089

Note - This staffing will hold for "normal" operations, but will be adjusted as appropriate if planned events of a critical nature are scheduled to occur during a particular shift. All personnel will be on call in event of an emergency situation requiring augmented staffing.

Dudley Thompson, IE, Jess Crews and Ed Jordan, IE, have been designated Operations Status Officers. One or the other will be on duty at all times to assist the EMT in keeping informed on the current status at the Three Mile Island site, preparation and coordination of the "Preliminary Notices" that are being issued on TMI, and responding to Commission, Congressional, and Executive Branch agency queries.

10
175 1

Executive Management Team Staffing
NRC Operations Center

Until further notice the Director and NRR and IE members of the Executive Management Team (EMT) in the NRC Operations Center will be as shown below:

<u>Daily</u>

0600 - 0800 hours	Director, EMT - L. Gossick NRR - D. Eisenhut, or D. Davis, or B. Grimes IE - N. Moseley, or H. Thornburg, or M. Howard
0800 - 1600 hours	Director, EMT - L. Gossick NRR - E. Case IE - J. Davis
1600 - 2200 hours	Director, EMT - E. Case or J. Davis NRR - D. Eisenhut, or D. Davis, or B. Grimes IE - N. Moseley, or H. Thornburg, or M. Howard
2200 - 0600 hours	Director - To Be Designated by EMT Director of preceeding shift. NRR - D. Eisenhut, or D. Davis, or B. Grimes IE - N. Moseley, or H. Thornburg, or M. Howard

175 1101

TMI -- IE MANAGEMENT SCHEDULE

IE Director	Boyce Grier	8:00 am to 8:00 pm
	Karl Seyfrit	8:00 pm to 8:00 am
Operations	Rick Keimig	8:00 am to 8:00 pm
	Ebe McCube	8:00 pm to 8:00 am
Health Physics	Geo Smith	8:00 am to 8:00 pm
	Gen Roy	8:00 pm to 8:00 am

175 112

NRC SITE OPERATIONS CENTER
(Continuously Manned)

Stello/Vollmer

1. White House Line

2. 717-944-0301

NRC SITE TECH. REVIEW CENTER
(0600-2400)

Mattson/Ross

1. 717-944-0601

April 9, 1979

IE SHIFT ORGANIZATION

	0000-0800	0800-1600	1600-2400
IE Shift Supervisor	E. McCabe	R. Keimig	B. Warnick
Unit 2 CR Opns Surveillance	L. Bettenhausen	R. Conte	W. Lazarus
Unit 2 CR Communications	D. Hinckley	C. Brown	L. McGregor
Unit 2 Opns Procedure Review	B. Jorgensen	R. Wessman	J. Dyer
Lead-In Plant Health Physicist	G. Yuhas	B. Greger	M. Schumacher
In Plant Health Physicist	R. Thomas	L. Ewald	B. Axelson
In Plant Health Physicist	T. Tongue	G. Troup	L. Thonus
In Plant Health Physicist	L. Friedman	D. Collins	R. Curtis
In Plant Health Physicist		P. Clemons	
HP Procedure Review	J. Baird	R. Zavadoski	R. Miller
Lead-Environmental Surveys	D. Donaldson	D. Montgomery	D. Perrotti
Environmental Surveys	R. Paul	R. Woodruff	B. O'Neill
Environmental Surveys	D. Sreniawski	H. Young	W. Peery
Environmental Surveys		J. Glenn	
Environmental Surveys		N. Terc	
Sample Analysis - Mobile Lab	J. Everett	T. Jackson	J. Kottan

175 114

Case

Unit 2 Control Room
Manning by NRC Grier

2 Reactor Operations, IE
(1 on phone)
(1 surveillance)

1 H-P, IE

1 NRR, occasional access
from #2 T.B. office
turbine building

Grier & Mattson
proposal 4/6/79 1:00 p.m.

accepted by Arnold & Herbein
@ 1:45 p.m.

cc Stello, NRR
Mattson, NRR
Ross, NRR
Vollmer, NRR
Grier, IE
Denton, NRC
Arnold, Met Ed
Case, NRR

NRC COMMAND

	Shift	Start	End
H. Denton	day	3/30	
D. Mossburg	day	3/30	
Sue Lynd	night	4/2	
J. Cook	day	4/5	
S. Barnes	day	4/7	4/6
J. Johansen	day	4/2	

175 116

NRR Technical Review

Name	Function	Start	End
R. Mattson	Management	4/1	
D. Ross	Management, Tech. Coord.	3/30	
T. Novak	Review Team Leader Reactor Systems	3/30	
B. Siegel	Reactor Systems	4/6	
J.T. Beard	Instrumentation/Controls	4/6	
R. Fitzpatrick	Power Systems	4/6	
J. Wermiel	Auxiliary Systems	4/6	
P. T. Kuo	Structures	4/6	
F. Cherny	Mechanical Systems	4/6	
John Gilray	Quality Assurance	4/6	

NRR OPERATIONS

Name/Function	Shift	Start	End
Management			
V. Stello	day	3/30	
D. Vollmer	night	3/29	
Communication/Analysis			
J. Klingler	day	3/29	
A. Thadani	night	3/30	4/9
M. Taylor (RES)	night	3/30	4/9
Replacement	night	4/9	
Plant Procedures/Systems			
J. Mazetis	day	3/29	4/6
M. Williams	day	4/6	
G. Chipman	day	4/4	
W. Mills	day	4/5	
C. Berlinger	day	3/29	4/4
R. Chiramal	day	3/29	
J. Olshinsky	night	3/30	
S. Newberry	night	3/30	
F. Ashe	night	3/29	4/8
K. Lauben	day	3/30	4/6
H. Schierling	day	3/29	4/4
Effluents/Waste/HP			
B. Kreger	night	3/30	4/7
J. Cunningham (IE)	night	4/7	
C. Burke	day	4/4	
T. Murphy	day	3/30	
J. Collins	day	3/30	
E. Adensam	day	3/29	
F. Congel	day	4/4	
M. Roll	day	3/30	
V. Benarova	night	3/30	4/6
J. Donaldson	night		
Op. Licensing/Procedures			
J. Holman	day	3/30	4/8
R. Cambell	day	4/8	
B. Boger	night	3/30	4/8
K. Mahan	night	4/8	
J. Buzzy	day		4/8
R. Cooley	day	4/8	

175 118

4/10/79

Power Reactors

CY 78 (date inspection began)

Operating Power Reactor	68
Total items of noncompliance	1400
Total inspections	1959
Average noncompliance per reactor (Operating)	21.6
Average number of items of noncompliance per inspection	.7

Three Mile Island 2 (50-320)

36 inspections
17 items of noncompliance (Sev 1 = 0, Sev 2 = 14, Sev 3 = 3) 0.47 INC's/insp.

Three Mile Island 1 (50-289)

26 inspections
16 items of noncompliance (Sev 1 = 0, Sev 2 = 9, Sev 3 = 7)

175 1109

Date____________

SITUATION STATUS

TIME OF ENTRY	SITUATION
1/28 0715	NRC NOTIFIED
0835	EMT ACTIVATE IRC
1025	RI INSPECTORS ONSITE
025	DOE NEST/RAT STANDBY
10[illegible]	AMS MOVED INTO AREA ← ETA 1330
[illegible]30	RAT VOLUNTEERS ASSISTANCE
1200 PM	NRC REP AT EOC
[illegible]0 PM	EMT BRIEFING
[illegible]	NOTIFY UPDATES
24[illegible]	TOTAL 4 COPTERS IN AREA
	DOE - AMS
	AF - R1 TEAM
	CG - BNL RAT
	STATE POLICE
1/29	AIR/GROUND SURVEYS CONTINUE

175 120

General Description TO: JIM STONE AT DOE EOC

4:00 a.m. Unit at 98% power DRAFT- SUBJECT TO FURTHER REVIEW

slide 5 - Secondary pumps tripped due to a feedwater polishing problem. 13

slide 6- This resulted in a turbine trip and subsequent reactor trip on PAGES

High Reactor Coolant Pressure. FM: NRC OPCEN

~~slide~~ 7- A combination Feed Pump Operation and Pressurizer Relief-Steam

COMMENTS: Generator relief valve operation caused a RCS cooldown.

- At 1600 psig Emergency Safeguards Situation occurred.
- All ECCS components started and operated properly.
- Water level increased in the Pressurizer and Safety Injection was secured manually approximately 5 minutes after actuation.
- The RCS pumps were secured when low NPSH limits were approached.

7:00 a.m. High activity was noted in the RCS coolant sample lines. A site emergency was the declared.

7:30 a.m. General Emergency was declared.

7:45 a.m. Licensee notified Region of the incident.

8:10 a.m. Region Operations Response Center activated.

8:35 a.m. HQ Operations Center activated.

8:43 a.m. Region I Response Team dispatched to site.

10:05 a.m. Response Team arrives at site.

3:30 p.m. The plant is being slowly cooled down with RCS pressure at 450 psi,

slide 8 using normal letdown and makeup flow paths. The bubble has been collapsed in the A RC Loop hot leg, and some natural circulation cooling has been established. Pressurizer level has been decreased to the high range of visible indication, and some heaters are in operation.

175 121

slide 5

The secondary plant is being aligned to draw a vacuum in the main condenser and use the A Steam Generator for heat removal. The facility plans to continue a slow (3oF/hr) cooldoun, until the Decay Heat Removal System can be placed in operation at 350 psi RCS pressure, 350oF RCS temperature in 15-18 hours.

slide 7

~~As of 3:30 pm, a plume approximately 1/2 mile wide and reading generally 1 mr/hr~~ was moving to the ~~north of the plant. The ARM,s helicopter is being used to define the length of the plume. Airborne Iodine levels of up to 1 x 10-8 uCi/ml~~ have been detected in Middletown, Pennsylvania, which is located by plant personnel.

4:31 p.m. Decision was made to open the Electro Magnetic Relief Valve to depressurize.

4:47 p.m. Plan to put A-loop into service - produce steam - natural circulation and cool primary down to a point that the RHR can be put into service.

- Eary Miller - station super at site plus 6 of unit supers.

5:24 p.m. Just starting to steam on the A SG. Vacuum was established at the condenser of 15".

5:39 p.m. Valve between SG and condenser is not opening - steaming has not really started yet. Sending someone to investigate.

5:50 p.m. Started steaming from the "A" SG. SG level dropped slightly - using some makeup water.

6:02 p.m. Steaming in "A" SG continuing - Plan to raise primary system press to 2000 psig to collapse any existing air/steam bubbles.

2

175 1212

6:35 p.m. A-loop appears to have little natural circulation. Indications are that some of the bubbles are collapsing.

7:34 p.m. License has bumped the RCS pump - results appear to be successful - not positive confirmation yet.

7:36 p.m. RCS pump seemed to have pumped successfully for the 10 second bump SG pressure went from 20# to over 200#. Have to wait 15 minutes before bumping the motor again.

7:49 p.m. One RCS pump in A-loop has been reported as running - estimates are it will continue to run.

7:52 p.m. RCS pump continues to run, looks like normal cooldown with one pump will provide.

9:11 p.m. Mobile lab on site and is getting set up.

5:15 p.m. Tc-310 TH-? No bubble yet.
Pressure 1050 - Temperature 520°F
Problem with aux. boiler solved and vacuum back on "Hot Well".
Cool down going well and normal.

9:49 p.m. Licensee is now venting through the aux. bldg. vents.

175 123

10:00 p.m. Cooldown rate has now been established at 8°F/hr.

10:04 p.m. Licensee reports he now has PZR level indication and a bubble has been established.

11:04 p.m. Loss letdown flow. All waste tanks are full in both units.

11:13 p.m. Approximately 200,000 gallons of water in the aux. bldg. to get rid of.

12:01 a.m. PZR Pressure - 1010 psig
Tc-292
PZR - Temperature 550°F
No let down established yet.

12:29 a.m. PZR Pressure - 1066 psig
PZR Temperature - 554°F
Tc - 291°F
Planning to open bypass valve around the let-down valve.

4:30 a.m. PZR Pressure - 980 psig
TCA - 286°F
PZR Level - 363"
PZR Level - 545

4

Waste transfer started to Unit 1 neutralizing tanks.
Approximately 70,000 gallon capacity left in Bleed Holdup tanks.

5:45 a.m. PZR Pressure - 957 psig

TCA - 285p

PZR Level - 350"

PZR Level - 542

TCB - 285°

SG A Pressure - 33

SG B Pressure - 38

Attempts to open PZR spray valve has caused increase in PZR level.

Steam Generator A - Two Bypass valves are wide open.

Bleed holdup tanks have 75,000 gallon capacity.

Relief Valve Letdown flow - 25 gpm.

Incore Thermocouple readings 207-617° - Profile available.

5:45 a.m. PZR Pressure - 898 psig

TAC - 284

- 366

PZR Temperature - 535

5

TAB - 284

SG A Pressure - 32

SG B Pressure - 37

Tried spray, however, pressurizer level increased 17" in 15 to 20 seconds. Observed restriction in letdown flow.

6

4:00 a.m. The combination of RCS cooldown and loss of steam through the pressurizer relief valve resulted in shrinkage of the primary coolant until the indicated water dropped out of the pressurizer, and voiding occurred in ~~the balance~~ other points of the primary system.

slide 5

Apparently the reactor coolant pumps were stopped ~~before enough water had been pumped into the primary system (by the ECCS) to collapse the voids in order to prevent damage to recirculation pump seals~~ because of concern for NPSH.

slide 6

With the primary pumps stopped, ~~the~~ Apparently voids collected in the primary coolant loops at the highest points in the system, the tops of the steam generators.

These voids result in loss of natural circulation in the core and (probably inadequate cooling of the fuel and possible fule damage.

At some later time ECCS was restarted manually in order to improve cooling to the reactor core. Up to 500 gpm of water was pumped into the primary system through the High Pressure Coolant Injection (HPCI) pumps. (These are the same as the makeup pumps.)

slide 8

Steam venting was accomplished by opening the ~~electronic~~ electromatic relief valves on top of the pressurizer. This steam blows down to a tank in the containment. When the capacity of this tank is reached (after a very short time) the a rupture disc fails which steam is vented directly into the containment, where it is condensed. ~~the resulting water flows into the containment sump where it is available for recirculation to the primary system~~.

slide 7

The source of the HPCI water is the borated water storage tank which is required (by technical specification) to contain at least ~~3~~450,000 gallons of borated water Over 140,000 gallons of this water was pumped into the primary system.

Within Facility

Rx. Bldg. dome	20,000 Rem/hr.
Aux. Bldg.	1 to 10 Rem/hr.

Max dose to individual - at Northgate - outside of shelter <500 mrem annual limit for individual in population) estimated approx. 100-200 mrem

COLLECTIVE DOSE

Approx. 123,500 persons in three northern sections within 50 miles.
Approx. 2,000 man-rem COLLECTIVE DOSE-FIRST DAY.
Approx. 1% of annual COLLECTIVE dose due to natural bkg.

IODINE

One milk sample taken from plume area 21 PCi/I Iodine (MDA 14pCi/I)
FDA Protective Action Guides-
12,000 pCi/I - remove cows from pasture
120,000 pCi/I - control distribution of milk

Offsite measurements of radioactivity have been monitored continuously by federal (NRC and DOE) and state personnel since 9 a.m. on Wednesday, March 28, 1979. All of these measurements (air, water, soil and vegetation) indicate that there is no immediate threat to public health and safety.

The offsite airborne radioactivity is determined to be almost exclusively from noble gases - primarily Xenon-133. There has been small amounts of iodine detected in one of several milk samples. The offsite airborne radioactivity has resulted in minimal exposures to the public in a northerly direction from the plant. The exposures in the air, as measured by a helicopter, are about 0.1 - 0.5 milliroentgen per hour. Natural background is approximately .02 milliroentgen per hour. The highest ground level measurements offsite (about 12 milliroentgens per hour) were measured at about two miles north of the plant. These levels would result in exposures of only a small fraction of the Environmental Protection Agency's recommended protective action guidelines (1000 milli-roentgens).

There have been no known releases of liquid radioactivity.

* 21 picocuries/liter (MDA is 14 pCi/l; FDA protective action guides for peak levels - 12000 pCi/l, remove cows from pasture; 120,000 pCi/l - control distribution of milk)

10

Status of TMI 2 Incident - 6:00 a.m. 3/29/79 (L. Barrett)

No detectable radiodines in air samples. MDA $1x10^{-9}$ uci/ml

1 pci/L for 1 year = 15 mrem

State took 6 milk samples @ approx. 20:00 3/28. Samples were from around the site with the sample taken approx. 5 mi. NW indicating 21 pci/l I-131. Cow on stored feed and in barn. No detectable iodine in any other samples which was expected because plume was toward NNW-NNE.

Region I stated that no apparent steam generator leakage. Air ejector monitor reads background.

Unit 2 vent monitor off scale. Background at monitor 540 mr/hr makes monitor useless. Unit 1 vent monitor indicates low releases from Unit 1. Unit 1 release concentration $1x10^{-6}$ uci/cc iodine and 3×10^{-7} uci/ml particulates. 84,000 cfm exhaust rate.

ARMS information: 1st flight 16 mi. away approx. 1 mr/hr. NaI instrument calibration for Xn-133 is questionable. Best guess is 0.8-1 mr/hr @ 7 mi. from plant. Plume toward north.

2nd ARMs flight, 8:30 p.m. Harrisburg is few tenths of mrem. (0.1 mrem ?) No iodine detected by ARMs, only Xn-133

The Unit 2 Auxiliary Bldg fans were secured at 00:50 to minimize XN-133 release. This caused an increase in radiation levels in the plant. Unit 1 machine shop 40 r/hr, Unit 2 control room went to respirators approx. 5 mr/hr in control room. Counting room out, using Region I van for counting radiation protection samples. Offsite data @ approx. 1:00 0.1 mr/hr. Fans restarted at 3:30.

Its good to restart fans to prevent iodine exfiltration by pulling a negative pressure in the building and using charcoal filters to absorb the iodine. The Xn-133 with a 5 day half life would get out anyway.

11

175 1301

Light rain started at 1:00. Region to take rain samples to watch for iodine washout expected to occur. Rain ended at approx. 5:00. No data yet.

Weather forecast at 5:00. Wind from the southeast at 5 knots expected to come from the south at 10 knots later in the morning.

04:30 measurement of 20 Mrem /hr at Goldsboro due west. Most likely spike when started auxiliary building ventilation at 3:30. West boundry of Island 28 Mrem/hr at 0405 and 2 Mrem/hr at 0530. North gate (nearest residence) is 27 mrem/hr at 0425.

Apparently primary coolant has been transferred to the auxiliary building at 30 gpm most of the night. At 0400 (and maybe before) it was going to the CVCS bleed (holdup) tank. The bleed tank is vented to the waste gas decay tanks so that the XN-133 should be collected on the tanks for decay.

Failed fuel most likely 1% or could be more based on 0900 3/28 sample and core inventory. No firm containment radiation information.

Offsite Radiation Doses: Based on best available information is that no member of the public should have exceeded the 10 CFR 20 annual limits of 500 mrem over the last 24 hours. Worst location is homes outside of the North Gate. Best estimate is 100-200 mrem range total over the last 24 hours.

Consecrvatively estimated population doses to be 2,000 man-rems out to 50 miles over the last 24 hours. In the NNW, N, and NNE sectors . See J. Martin notes attached for method of calculation.

These doses are approximately 1% of the annual natural background radiation dose to the population. Asked for radioactivity concentration information in auxiliary building. Can't get radiation levels. Primary sample sink approx. 200 r/hr.

[Auxiliary Bldg sump]

12

175 132

Near & Onsite Monitoring Data
March 28, 1979

Time	Location	mrem/hr
0880	Site Perimeter	approx. 1
1000	North Parking Lot	3
	Base of Containment	50
1030	North Gate	approx. 1
1100	SE perimeter	2
1330	400 ft. above cooling towers	10
1400	North Parking Lot	15
	North Gate	3
	300 ft. above containment	20
1430	North Gate	34
1830	North Gate	approx. .05-26
1920	N.W. boundary	3-20

March 29, 1979

0100	North Gate	3.5
0425	North Gate	27
0430	West Boundary	28
	South Boundary	26

Significant Offsite Monitoring Date
March 28, 1979

1400	Middletown (2½mi. North of site)	1-1.5
1500	Harrisburg (approx. 8 miles North of West Site)	Background
	Middletown	approx. 1
1630	ARMS detected plume 16 miles North-North East of Site	.1-.3 Xenon 133
	at approx. 5 miles from site	.3-.8 Xenon 133
1800	Middletown	approx. 1
1930	2 Miles NNW of site	12
2045	ARMS-detected plume over Harrisburg-plume extends to	.1-.3 Xenon 133
	height of 3000 ft. & is 4-5 miles wide at approx. 3-5 miles form site	.3-.8 Xenon 133
2300	Highspere are approx. 6 miles NNW of site	3
	Olmstead Plaze approx. 3 miles NNW of site	5
	approx. 2 miles NNW of site	12

March 29, 1979

0130	3 readings from 2 to 7 miles N	approx. 0.1
0600	Goldsboro - 1 mile due west of site	20

175 133

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF PUBLIC AFFAIRS
WASHINGTON, D.C. 20555

No. 79-65
Contact: Frank L. Ingram
Tel. 301/492-7715

FOR IMMEDIATE RELEASE
(Wednesday, March 28, 1979)

NOTE TO EDITORS: The information below was issued at approximately 5 p.m. EST.

The Nuclear Regulatory Commission has received additional information from its inspectors at the Three Mile Island Power Plant in Pennsylvania where an accident occurred earlier today. This updates the NRC announcement made earlier today.

No injuries have been reported. Low levels of radiation have been measured off the plant site. The maximum confirmed radiation reading was about three milliroentgens per hour about one-third mile from the site. At one mile, a reading of one milliroentgen per hour was measured. It is believed that this is principally direct radiation coming from radioactive material within the reactor containment building, rather than from release of radioactive materials from the containment. Extensive efforts are continuing by the State of Pennsylvania, the Department of Energy, and the NRC to measure the amount of radioactive material which may have been released from the site. A helicopter with special instruments also is being used.

It now appears that the cause of the turbine shutdown at the plant early today was a reduction in flow of feedwater to the steam generators.

The sequence of events which led to the release of radioactivity to the reactor containment building has not been determined. There was a release of primary coolant water to the containment. Emergency core cooling systems are continuing to provide water to cool the fuel. The reactor is shut down. The pressure in the reactor system is being slowly reduced.

NRC has a team of six persons at the site. They will participate in the NRC investigation of the event. The results will be made public. An NRC team also is being formed to monitor the subsequent activities of the licensee, the Metropolitan Edison Company.

#

175 [illegible]

134

The Nuclear Regulatory Commission has received additional information from its inspectors at the Three Mile Island Power Plant in Pennsylvania where an accident occurred earlier today. No injuries have been reported.

Radiation levels have been measured off the plant site. The maximum confirmed radiation reading was about three milliroentgen per hour about one-third mile from the site. At one mile a reading of one milliroentgen per hour was measured and it is believed that this is principally direct radiation [illegible] from [illegible]

175 1345

2

material within the reactor containment building. Extensive efforts are continuing by the State, ~~[illegible]~~ the Department of Energy and the NRC to measure the amount of radioactive material which may have been released from the site. Typical background radiation from natural sources in the Harrisburg area is about 90 millirontgens, per ~~[illegible]~~ year.

Everything under control

is at areas near Dauphin

Small release to env

all safety systems functioning properly

[illegible] normal rad

release

CD alerted all counties

No rad leakage into env.

Low level radiation in [illegible]

Mr. Gilinsky
Mr B.

3/28/79

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The sequence of events which led to the release of radioactivity to the reactor containment building has not been determined. There was a release of primary coolant water to the containment. Emergency core cooling systems are continuing to provide water to cool the reactor fuel. The pressure in the reactor system is being slowly reduced.

NRC has a team of six persons at the site. They will participate in the NRC investigation of the accident. The results will be made public. An NRC team also is being formed to monitor the post-accident activities of the licensee.

175 138

The Nuclear Regulatory Commission has received preliminary information on an emergency situation at the Three Mile Island Nuclear Power Plant near Harrisburg, Pennsylvania. There ahs been a release of radioactivity inside the reactor containment system. Measurements are still being made to determine if there has been any radioactivity detected off the site. There is no indication of release off the site at this time.

Medtropolitan Edison Company, operator of the plant, has reported that the turbine of Unit 2 at the plant tripped off early this morning, closing off the steam flow from the reactor to the turbine. The cause is not known at this time.

There was a release of primary coolent water into the containment. The ~~emerging~~ emergency core cooling systems are being used to provide water to the reactor. The Unit 2 reactor is shut down. Unit 1 at the plant is shut down for refueling.

NRC ~~has~~ dispatched a team from the Regional Office at King of Prussia, Pa., and they will be at the site shortly. State and other Federal agencies have been notified.

175 1389

The Nuclear Regulatory Commission

reported tonight that Metropolitan Edison

Company has completed transfer of

radioactive water from the auxiliary

to holding tanks

Building (at its Three Mile Island Nuclear

Power Plant in Pennsylvania.

¶ The water is believed to have been

the source of a majority of the

radioactivity released offsite following

an accident yesterday. after the

transfer

175 ~~139~~
140

The Nuclear Regulatory Commission stated today that as a result of its inspection of activities at the Three Mile Island reactor in Pennsylvania, it has learned that

At about ______, the utility, the Metropolitan Edison Company, completed transfer of highly contaminated water from the Auxiliary Building to holding tanks. This water apparently was the source of the majority of radioactivity released offsite following an accident at the site yesterday.

~~Since~~ After the transfer had been completed offsite radiation surveys confirmed ~~a significant reduction in the offsite radiation~~ that the ~~[illegible]~~ release of radioactive material offsite has been terminated.

The fuel in the reactor is still being cooled by circulation of cooling water through ~~the~~ a steam generator, from which heat is removed by steam being condensed in the main condenser. The temperature and pressure in the reactor are stable at acceptable levels.

No assessment of the extent of any possible ~~[illegible]~~ damage to the reactor or associated equipment has been made at this time. ~~There is no evidence of~~ There have been no injuries, nor serious exposure of workers at the plant. No evidence is available to indicate that any member of the public has received radiation exposure above small fractions of NRC limits.

NBC

20-30 Goldsboro

175 1483

3/28/79
10:30 PM

The Nuclear Regulatory Commission said tonight that its inspectors at the Three Mile Island Nuclear Power Plant in Pennsylvania have reported that temperatures and pressures continue to drop in the reactor where an accident occurred earlier today. However, pressures and temperatures have not dropped far enough to activate the normal heat removal systems. When these systems are activated, emergency core cooling systems no longer will be needed.

Radiation levels within the containment building remain very high; one instrument indicates these levels are thousands of roentgens per hour at one area inside the containment ceiling.

There is a continuing release of detectable levels of radioactive material from the plant site. Measurements made thus far by a helicopter with special instruments indicate that these levels in the air were about one third of a milliroentgen per hour over the Harrisburg area. These levels are far below the 1000 milliroentgen level at which the Environmental Protection Agency reccomends protective action.

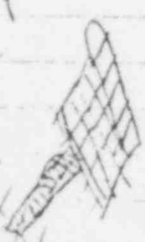
Metropolitan Edison Company, operator of the plant, estimates that as many as eight workers received radiation exposures of one half to one rem during the course of the day's activities. The annual exposure limit for radiation workers is 5 rem.

175 1404

The Nuclear Regulatory Commission has received additional preliminary information from its inspectors ~~here~~ at the ~~[illegible]~~ Three Mile Island Nuclear Power plant when an accident occurred earlier today.

The NRC inspectors have reported that the nuclear fuel in the reactor is being cooled

? miles fr plant



melt

fuel core

Metropolitan Edison company, operator of the plant, ~~reports~~ estimates that ~~[illegible]~~ As many as eight workers received radiation exposures of ~~[illegible]~~ one half to one rem during the course of the day's activities. The annual exposure limit for radiation workers is 5 rem.

175 146

The Nuclear Regulatory Commission said tonight that its inspectors at the Three Mile Island nuclear Power plant in Pennsylvania have reported that temperatures and pressures continue to drop in the reactor where an accident occurred earlier today. * However, pressures and temperatures have not dropped far enough to activate the normal heat removal systems. When these systems are activated, emergency core cooling systems no longer will be needed.

BM
at Harrisburg
airport

175 1467

Radiation levels within the containment building remain very high; one instrument indicates these levels are thousands of roentgens per hour at one area inside the containment ceiling.

~~[illegible]~~

There is a continuing release of ~~[illegible]~~ detectable levels of radioactive ~~[illegible]~~ material from the plant site. Measurements made thus far by a helicopter with special instruments ~~[illegible]~~ indicate that these levels in the air ~~were [illegible]~~ over the Harrisburg area, ~~[illegible]~~ about one third of a millirentgen per hour. ~~[illegible]~~

~~[illegible] reports [illegible] from monitoring [illegible]~~

~~[illegible]~~

~~[illegible]~~

~~These [illegible] levels are far below the~~

1000 to 5000 ~~millirentgen~~ levels ~~under~~ which protective ~~[illegible]~~

~~[illegible]~~

These levels are far below the 100 millirentgen level at which ~~[illegible]~~ the Environmental Protection Agency ~~[illegible]~~ recommends protective ~~[illegible]~~ action.

The Nuclear Regulatory Commission said ~~reported~~ tonight that its inspectors at the Three Mile Island Nuclear Power plant have reported that ~~[illegible]~~ normal heat removal systems now are being used to cool down the reactor x ~~[illegible]~~ where an accident occurred earlier Friday. Reactor coolant temperatures and pressures are decreasing in normal fashion. These systems -- known as decay heat removal systems -- were activated about _ p.m. ~~[illegible]~~ These systems were not used earlier because temperatures and pressures in the reactor were too high. ~~[illegible]~~ When these systems were activated, the use of emergency core cooling systems no longer was needed.

175 149
50

Bryan

Mile 9 Job L

4/7/79

PRELIMINARY INVESTIGATION/INSPECTION PLAN

I. Week of April 8, 1979.

A. Operational Group

1. Refine chronology of events to

a. Support activities of P. Check's group.

b. Identify answers to D. Eisenhut questions

c. Correctly identify sequence and nature of events during incident.

d. Identify key causal factors for subsequent detailed follow-up and evaluation.

2. Conduct site visits to:

a. Interview operators to clarify certain manipulation other actions taken which are not covered by hard data or prior interviews. This is to enlist their aid at their convenience in refining the chronology. Our

175 151

visit early in week (ASAP) of approximately one hour per man is deemed necessary. Later further interviews may be necessary.

b. Determine answers to questions on alignments, equipment details etc., which are unavailable elsewhere. Maximum use of existing IE site inspection personnel will be used. Some contact with Met Ed people unavoidable but will be kept to a practical minimum. (Might expect a second visit this week for this purpose).

B. Radiological Group

1. Organization and definition of assignments to members.
2. Determination of sources and locations of existing material.
3. Begin review of material already developed by other groups and start synthesis.
4. Establish contacts with site (NRC) but direct contacts with Met Ed unlikely.
5. No site visits expected.
6. Outline investigation methodology and program.

175 152

II. WEEK OF APRIL 15, 1979

A. Operations Group

1. Issue "final" chronology.

2. Start inspection of background of "causal" factors identified in chronology

3. Onsite time expected to be 2-4 inspector man days. Major portion would be records review with direct contact primarily for clarification of points.

B. Radiological Group

1. Implementation of investigation plan - start of limited site visits.

III. SUBSEQUENT WEEKS

A. Operations Group:

When plant is placed in cold shutdown condition and conditions adequately stabilized, a heavier inspection effort is anticipated. A total inspection load of up to 100 inspector-days is forecast

175 153

E. Radiological Group

Their activities are expected to involve larger manpower needs and be of longer duration. No estimates available at this time.

BDMartin

4/7/79

175 154

B. Radiological Group

Their activities are expected to involve larger manpower needs and be of longer duration. No estimates available at this time.

BDMartin

4/7/79

Office	Contact	Action	Est. Completion
IE	Moseley	IE Investigation	8/1/79
IE/NRR	Moseley	Evaluation of Licensee Responses to IEB 79-05	4/27/79
IE/NRR	Moseley	Evaluation of Licensee Responses to IEB 79-05A	5/4/79
IE/NRR	Moseley	Evaluation of Licensee Responses to IEB 79-06	5/10/79
IE	Davis/Donnelly	Emergency Supplemental Budget Request	5/1/79
IE	Davis	Accelerated/Expanded Resident Program Definition	6/15/79
IE/ADM	Thompson/Wallace	Operations Center Modifications	7/1/79
IE	Davis	IE HQ Organization for Operational Functions	7/1/79
IE/ADM	Thompson/Davis	Expanded Communications to sites and regions	7/1/79

Enclosure

175 156

Prompt report - TMI/1 aux. RI

Unit 1 turbine driven feed pump

MSV6 Steam regulator valve was found shut. U1 was in hot standby

v/v had been tagged for maintenance after maintenance, valve wasn't reopened found shut on 3/27, valve was reopened.

the 2 electric (50% cap ea) motor driven aux feed pumps were operable.

Don't know how long the valve stayed shut. This info expected in written report

T/S requires all 3 aux feed pumps to be operable in Hot standby.

rec'd per telecon with H. Kister

1030 4/11/79

CBBlackwood

175 150

243 54

LOSS OF PRESSURIZER LEVEL INDICATION

PURPOSE

The intent of this procedure is to track pressurizer level after a loss of normal level indication utilizing an empirically derived relationship between Make Up Tank level, pressurizer level, and RCS leakage. Periodic checks and updates will be made every 12 hours utilizing the test equipment installed on the pressurizer water space sample line and pressurizer temperature (RTD) voltage output to determine actual level.

1.0 Assumptions

1.1 Pressurizer level is being maintained at 250 ± 25 inches.

1.2 RCS pressure is being maintained between 900 and 1000 psig.

1.3 Summary B group trend including the following data points is being printed at 15 minute intervals.

COMPUTER PT.

389	Prz temp.	(Ensure control board selector switch is selected to TT-2)
1682	Prz level	
398	Press Loop A	
399	Press Loop A	
394	RCS Temp A	
400	Press Loop B	
347	MU Tank Level	

1.4 The pressurizer temperature transmitter selector switch on the control panel must be CAUTION tagged to require Shift-Supervisor permission to operate. After any readings taken on TT-1 the switch must be selected to TT-2. This will allow TT-2 to read out on the pen recorder and provide TT-1 as the point monitored by the computer.

1.5 Pressurizer heater interlocks disabled to prevent level failure from causing loss of heaters.

1.6 Temperature of RCS is between 160° and 280°F.

1.7 RC-V2, RC-2 and RC-V137 shut.

1.8 Primary leak rate is 2.6 gpm.

1.9 MUV 8 is aligned to the make-up tank.

175 158

MS

- Failed pressurizer level indication. Level instrument should fail to midscale, but may fail high or low.

2.2 Pressurizer level annunciator sounds.

2.3 Pressurizer level steady with changing plant parameters.

3.0 IMMEDIATE ACTION (First 5 Min.)

3.1. Do not secure spray if already initiated.

3.2. Do not alter pressurizer heater alignment.

3.3. Shut or check shut MUV 17 and 18. (With a loss of indication it will be necessary to go to shut on MUV 18 regardless of assumed position.

3.4. Check shut pressurizer vent valve RC V137 and relief valve block valve RC V2.

3.5. Check MU V-8 aligned to the make-up tank and not to the bleed tank.

3.6. Record the last valid pressurizer level from the pen recorder prior to malfunction of the instrument.

3.7. Check to ensure summary Group 8 is trending at 1 min. intervals with the data points as noted in 1.3 above.

3.8. Record the data listed in data sheet 1.

4.0 LONG-TERM ACTIONS (After 5 Min.)

4.1. Maintain RCS pressure 900-1000 psig. DO NOT EXCEED 1000 psig.

4.2. DO NOT VENT THE PRESSURIZER.

4.3. Maintain constant reactor coolant temperature during first 4 hours following loss of all pressurizer level indication. Subsequent shrink due to cooldown must [illegible] compensated for by periodic additions per section 5.4.

4.4. Maintain pressurizer level 225-275 inches utilizing the following method.

175 1589

CAUTION: Monitor pressurizer temperature and pressurizer heater current for evidence of uncovered pressurizer heaters. A superheated condition occurs rapidly if the heaters are uncovered. If either pressurizer temperature increases by 10°F or more in one minute, or if a marked decrease in pressurizer heater current occurs, secure all heaters and pump borated water to the RCS in accordance with Section 4.8.1.

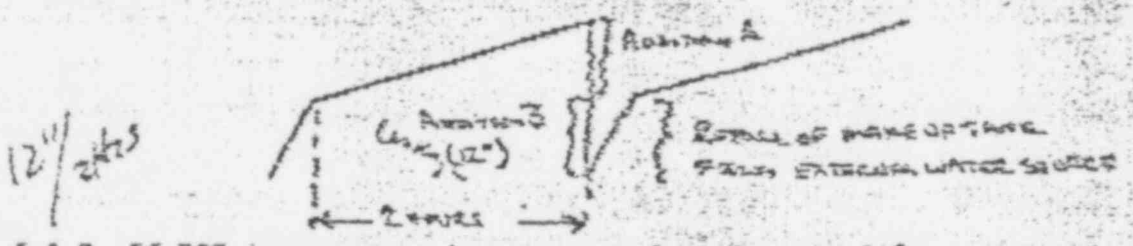
4.4.1 Compensate for calculated pressurizer level change every 2 hours with the following additions to the RCS.

NOTE: The following additions should be made in batches, i.e., open MUV17 and 18, add water to RCS then immediately close MUV 17 & 18.

ADDITION A
Add to the RCS from the make-up tank the amount that the make-up tank level increased during the previous two hour period as indicated by the pen recorder trace.

ADDITION B
Immediately following Addition A, add another 12 inches from the makeup tank to the RCS to account for the assumed RCS leak rate of 2.6 gpm. Refill the makeup tank 12 inches by an addition from an external water source (RC bleed tank, Demin. water, BA batching tank, etc).

Repeat Additions A and B every 2 hours. The makeup tank level trace should approximate the following:



4.4.2 If RCS temperature increases or decreases by 10°F compensate for system volume changes as follows:

For every 10° rise in RCS temperature, allow the make-up tank level to increase 16 inches from normal letdown, i.e., during the period that the M/U tank level rises or for every 10°F decrease in RCS temperature, lower the makeup tank level by 16 inches, by pumping from the makeup tank to the RCS.

These changes will be made in addition to Additions A & B.

A note is being added to prevent changes in system [illegible] due to this 10° criteria while attempting to establish natural circulation.

[illegible]

1750 [illegible]

4.5 Maintain a record of additions (other than Boldown) to the make-up tank by completing data sheet 2 whenever an addition is made. DO NOT use totalizer. Use the change in level as indicated on the pen recorder on the console.

4.6 Establish a plot of pressurizer level vs. time utilizing the following method: Update plot immediately following the additions made per section 4.4 every 2 hours.

4.6.1 Transfer the data collected initially following loss of LT-3 to data sheet 3 in the blanks provided under "TOF" (Time of failure).

4.6.2 Collect data for time "t_1" when last addition A+B were made.

4.6.3 Subtract values at TOF from values at t_1 to generate the values identified in equation 1 on data sheet 3. Maintain consistent sign convention throughout i.e. 5-6=-1.

4.6.4 Insert the values determined in 3 above into equation 1 and calculate pressurizer level at the time the addition A&B were made.

4.6.5 Plot this value on graph attached to this procedure.

4.7 Every 12 hours determine actual pressurizer level utilizing the test equipment installed on the pressurizer water space sample line and pressurizer temperature RTD as follows:

4.7.1 Immediately following the periodic addition secure spray flow and allow conditions to stabilize for approximately 1 hour.

4.7.2 Stabilize pressure between 900-1000 psig utilizing SCR controlled pressurizer heaters.

4.7.3 If a primary sample has been taken since the last level measurement verify that the line-up has been returned to normal by lining up per Z-107 to read both the heise gage and pressure transmitter.

If no sample has been taken, open or check open CRV3, CRV10, SR V11, 2, 4, & 5.

4.7.4 Start 1 min. trend interval on Summary B.

4.7.5 Establish communications between the control room and the cable spreading room.

4.7.6 Read and record the DVM readout (in the control room) and the Heise Gage (on the pressurizer water space for pressurizer water space pressure on data sheet 4.

175 1601

4.7.7 Read and record the DVM readout (in cable spreading room) for pressurizer water space temperature on data sheet 4.

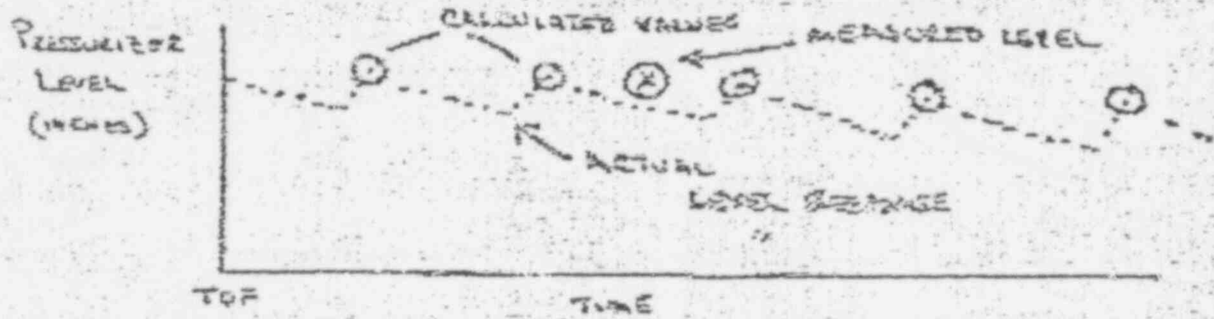
4.7.8 Re-perform step 4.7.5, 4.7.6, 4.7.7.

4.7.9 Complete data sheet 4 by performing the conversions listed on the bottom of the sheet.

4.7.10 Subtract P_{sat} from P_{dvm} and P ____ (Hefse) to obtain Δ P.

4.7.11 Determine actual level from the attached curve. Record this value on data sheet 4.

4.7.12 Plot value on pressurizer level plot. Plots over extended period are shown typically as below.



4.7.13 If measured pressurizer level is less than 140 inches or greater than 350 inches notify the shift supervisor and perform the following:

4.7.13.1 If level is < 140 inches make a batch addition to the pressurizer from the makeup tank with an amount calculated as follows:

$$\Delta \text{ MU Tank Level} = \frac{275 - \text{measured level}}{1.227}$$

4.7.13.2 If level is > 350 inches reduce the amount pumped back into the RCS at the next 2 hour addition, by the following amount.

$$\Delta \text{ MU Tank Level} = \frac{\text{measured level} - 275}{1.227}$$

4.7.13.3 These adjustments to level should not be added to equation 1 or data sheet 2.

4.7.14 If measured level is less than 275" but greater than 140"

or

If measured level is greater than 275" but less than 350"
Repeat the level measurement per steps 4.7.1 - 4.7.12, prior to the next periodic 2 hour addition. If the level is confirmed, make an adjustment per 4.7.13.
This addition should not be added to equation 1 or data sheet 2.

175 162

4.8 Loss of Pressurizer level control (indication of high or low level in the pressurizer)

4.8.1 Low pressurizer level: Indicated by increasing pressurizer temperature and/or marked decrease in pressurizer heater amps.

a. Secure all pressurizer heaters.

b. Open MU-V18 and close MU-V5.

c. Place MU-V17 in manual control and pump 30 inches of make-up tank level to the RCS.

d. Energize the Bank 1 heaters and observe pressurizer temperature.

e. If pressurizer temperature stabilizes, establish normal pressurizer level using MU-V5 and MU-V17.

f. If pressurizer temperature continues to increase above Tsat, secure the bank 1 heaters and perform steps b thru d again.

g. Restore pressurizer level to mid-scale (225-275 inches) by making several (5-6) additions (≈30 inches of make-up tank level) from the make-up tank to the RCS.

4.8.2 High pressurizer level: Indicated by increasing RCS pressure (greater than Psat). This indication only occurs if for existing PZR temperatures the pressurizer is near solid.

a. Take manual control of MU-V17 (if being used) and maintain the existing make-up flow rate. Use MU-V5 for pressure control.

b. Shut/check shut RC-V1, RC-V2 and RC-V137.

c. Secure all pressurizer heaters, record the RCS pressure and mark the pressure recorder chart.

d. Slowly increase the make-up flow rate to raise pressure to 50 psi above the value recorded in step c. Maintain the RCS pressure at this value to completely collapse the bubble and take the pressurizer to a solid water condition. Pressure should always be kept above 500 psi.

<u>NOTE</u>: The indication that the pressurizer is solid is a sudden increase in RCS pressure when making up at a constant rate.

175 168

CAUTION: When operating in a solid condition, RCS temperature changes and/or net addition or removal of RCS water cause large changes in RCS pressure. A net addition or removal of 10 gallons results in a pressure change of approximately 100 psig. A RCS temperature change of 1°F results in a pressure change of approximately 130 psig.

e. Maintain pressure by varying make-up and/or letdown flow rate using MU-V5 and MU-V17.

f. If RCS pressure continues to increase with MU-V17 and MU-V18 shut and MU-V5 closed, jog open RC-V137 to decrease pressure back to the original control points (pressure in section 2 + 50 psi). When pressure returns to the value being maintained previously, shut RC-V137 and revert to varying make-up/letdown flow rate for pressure control. If continuous venting through RC-V137 is necessary, minimize the flow rate through the vent valve by keeping the make-up addition rate as low as possible. Adjust make-up tank level as required.

175 164

DATA SHEET 1

	Values at Time of Failure (TOF)
Console Pen Recorder RCS Pressure	
Console Pen Recorder PRZ Temperature	
Console Pen Recorder Th	
Console Pen Recorder Tc	
Console Pen Recorder MU Tank Level	
Console MU Tank Temperature	
Console MU Tank Press	
OSTG Operating Range A Level	
OSTG Operating Range B Level	
OSTG A Temperature	
OSTG B Temperature	
Upper Cavity Press RCP1A	
RCP1B	
RCP2A	
RCP2B	
Seal Leakage RCP 1A	
RCP 2A	
RCP 1B	
RCP 2B	
Seal Injection RCP 1A	
RCP 2A	
RCP 1B	
RCP 2B	

175 1605

RECORD OF ADDITIONS

DATA SHEET 2

TIME/ DATE	INITIAL MAKE-UP TANK LEVEL (INCHES)	NEW MAKE-UP TANK LEVEL (INCHES)	CHANGE IN MAKE-UP TANK LEVEL (INCHES)	CUMULATIVE TOTAL ADDITION TO MU TK SINCE TOF (INCHES)	INITIAL

Data Sheet 3

PRZ LEVEL CALCULATION

Observe all sign conventions, i.e., 5-6=-1

Time ____________ / Date ____________

	t_1	TOF		
Temp Pzr ^oF	______	- ______ values at TOF	= ______	ΔT PRZ (^oF)
Make-Up TK Level (utilize pen recorder) (inches)	______	- ______ values at TOF	= ______	ΔL MUT (inches)
T_c (^oF)	______	- ______ values at TOF	= ______	ΔT_c (^oF)

A - Total additions to MU TK since TOF (obtain from data sheet 2) ______ inches

L (PZR TOF) = ______ inches take from value recorded on data sheet 1.

Equation 1

$$L_{PZR} = L_{PZR_{TOF}} + 2.07\,(\Delta T_c) - 1.227\,(\Delta L_{MUT} - A) + .287\,(\Delta T_{PZR})$$

$$L_{PZR} = (\underset{L_{PZR}}{\underline{\quad\quad}}) + 2.07\,(\underset{\Delta T_c}{\underline{\quad\quad}}) - 1.227\,(\underset{\Delta L_{MUT}}{\underline{\quad\quad}} - \underset{A}{\underline{\quad\quad}})$$

$$+\ 0.287\,(\underset{\Delta T_{PZR}}{\underline{\quad\quad}}) + \underline{\quad\quad} = \text{Record and Plot}$$

Assumptions: T_c between 160 and 280°F

RCS pressure between 800 and 1000 psig.

175 1667

Data Sheet 4

Pressurizer Level from Δ Psat – Pressurizer Water Space Pressure

Date ______ Data Recorded by ______

Time	Description		Pressure			Temperature			ΔPsi	
	Level (inches)	Temp (°F)	Heise (psig)	DVM (Volts)	(psig)	DVM (Volts)	(°F)	P_{sat} (psig)	Heise	DVM
Clock	Computer	Computer	Reading	Reading	Calc	Reading	Calc	Calc	Calc	Calc
								Average		

Calculated Level from ΔP vs level curve ______ in

$\Delta Psi = P_{measured} - P_{sat}$

Pressure DVM Conversion:

$Pressure = (137.5)(Volts) - 275.0 \; ; \; (psig)$

Temperature DVM Conversion:

$Temperature = \left(\frac{0 + Volts}{10} \right) 700\,°F$

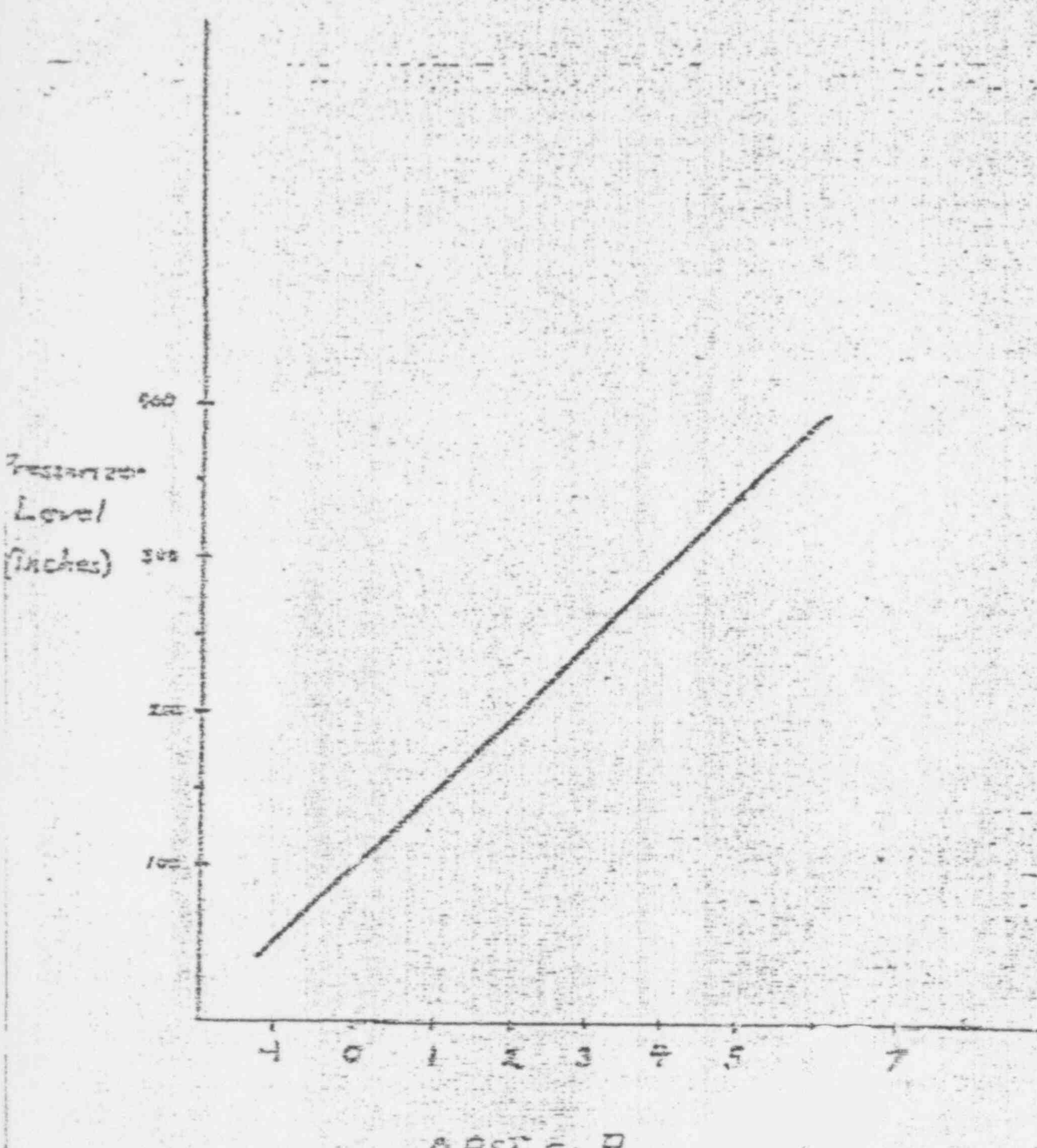
Saturation Pressure Calculation:

$P_{sat} = (37.3\ psi/5\,°F)(T_{sat} - 535) + 908.2 \; ; \; (psig)$

175 168

Figure 1

Pressurizer Level as calculated from Pressurizer Water Space Pressure Measurement



Pmeasured = - Water Space DVM or Heise Gage

175 1689

To: Instant Reco. Center

EP-32 - Loss of Operating RCP

Su cessful Natural Circulation

1.0 Purpose:

To provide adequate core cooling through natural circulation in the event of loss of the operating RCP.

2.0 References

Ref. No.	Description
Contingency Plan C-1,2,3	Loss of Reactor Coolant Pumps of 8 April 1979
Ref. A	Bill Lowe to (Illegible) Telecon Note of 0800, 4/12/79
Ref. #180	NDT Pressure/Temperature Curve
Ref. #10	Natural Circulation - Minimum RCS Pressure vs that curve with saturation curve added
Ref. #48	Preparation for an initiation of natural circulation.
Ref. #168	Effect of Partial Flow Blockage on N.C.
Ref. #156	RCP Startup Procedure (B&W procedure 15, Rev. 4)
Ref. #11	Loss of RC Flow/HPI
Ref. #(B&W) 77	Loss of RCP - Successful Natural Circulation (B&W 486, 4/19/79)
Ref. #TSG 071	Loss of Reactor Coolant Pump (R. Keaten 4/19/79)

3.0 Limitations and Precautions

3.1 Operational limits of RCP with backup RCP's available:

a. frame vibration exceeds 5 mils.

or

b. shaft vibration exceeds 30 mils

and

upper seal leakage & return flow increases to greater than 1.9 GPM.

3.2 Operational limits of RCP with no backup RCP available.
a. Shaft vibration > 70 mils
or
b. Upper seal leakage > MU system capability to maintain RC system water level.

4.0 Symptoms

4.1 RC Flow in the operating loop decreases or becomes erratic

4.2 Complete loss of RC flow in operating loop.

4.3 RC Pump Trip annunciator.

4.4 RCP limits (section 3) exceeded.

4.5 Indications the RCP has stopped as observed by no running current (amps) or vibration.

NOTE: Initial Condition

Heat removal through A OTSG in Steaming Mode
B OTSG H_2O/H_2O loop not yet ready.

Feedwater to OTSG supplied thru ~~auxiliary~~ MAIN 4/22/79 2130 FW nozzles.

5.0 Immediate Actions

5.1 Attempt to establish natural circulation

5.1.1 Immediately begin raising RCS pressure to approximately 900 psig ± 100 psig (to increase margin to saturation).

5.1.2 Trip the turbine

5.1.3 Without altering the feedwater lineup or flow rate allow OTSG "A" level to slowly increase to 430" on the wide range instrument. Secure feedwater flow, and allow OTSG "A" level to decrease to 400". Re-establish feedwater flow at the previous rate to increase level to 430". Repeat as necessary to maintain level between 400" - 430".

NOTE 1 - Following the above actions and when equilibrium conditions are established. RCS temperature T_c should stabilize at about 200 °F.

Note 2 - Natural Circulation is indicated by an increase in RCS ΔT to a new value greater than the approximately zero ΔT of forced circulation. This ΔT is expected to be approximately 10 to 35° when equilibrium conditions of flow have been achieved. Initially, however, ΔT will increase to greater values which could be as much as 15 to 40°F in the five to ten minute period after the loss of RCP, followed by a decrease to the equilibrium ΔT. The ΔT values in this paragraph are provided for information only and are a result of interpretation of analytical data. It should take about 25 to 35 min. to establish natural circulation.

Note 3 - The system will respond slowly to changes while in the natural circulation mode. The loop transport-time is about 20 minutes, therefore, changes in steam demand and feed rate should be made slowly and the system should be given time to equilibrate before additional changes are made.

5.1.4 Manually record and plot T_h, T_c and T_{stm} every twenty (20) minutes. Read and record all operable in-core thermocouples every ten (10) minutes.

5.1.5 If any of the (4) criteria provided below are exceeded, adequate Natural Circulation has not been established, therefore, proceed to step 5.2.

5.1.5.1 If T_H in the loop with the OTSG in the steaming mode exceeds 420°F, go to step 5.2.

5.1.5.2 If any thermocouple excess 1000°F go to step 5.2.

5.1.5.3 If any three (3) thermocouples have readings exceeding 800°F go to step 5.2.

5.1.5.4 At least 6 thermocouples must be below ~~570°F~~ 560°F, otherwise go to step 5.2. changed from 2/29 2130

NOTE: After natural circulation has been established it is expected that T_H and T_{stm} will be nearly equivalent. However, T_{stm} should not be less than T_c for natural circulation to occur. In making the above comparisons, an instrument error of up to ± 5°F must be considered.

175 172

5.2.4 Start AC Oil Lift and AC Backstop Pumps for RC-P-1B and 2B.

5.2.5 Verify intermediate and NSCCW is operating.

5.2.6 Verify Seal Injection Flow on RC-P-1B and 2B.

5.2.7 Open MU-V33 C & D. Verify RC Pump Seal staging by observing seal cavity pressure.

5.2.8 Verify RC Pump Seal Return Flow (< 1.91 gpm) on RC-P-1B and 2B.

5.2.9 ~~5.2.8~~ With steps 5.2.2 through 5.2.7 complete, start RC-P-~~1B~~ 2B

5.2.10 ~~5.2.10~~ If RC-P-~~1B~~ 2B start attempt unsuccessful, start RC-P-~~2B~~ 1B

5.2.11 Monitor the following during startup through the transient and thereafter for proper indication. Upper seal cavity pressure, lower seal cavity pressure, seal return temperature, upper seal leakage, seal return flow, pump shaft vibration and motor bearing temperatures.

5.2.12 Close MU-V33 on non-operating pumps.

5.2.13 Close seal injection valves on all but standby pump and readjust seal injection to minimize makeup.

5.2.14 Secure oil lift pumps on all non operating pumps.

5.2.15 If an RCP was successfully started, return RCS pressure and "A" OTSG level to the previous condition.

5.3 If no RC pumps can be started, attempt to establish natural circulation.

5.3.1 Verify RCS pressure is 900 $\pm$ 100 psig and "A" OTSG level 400-430". Adjust RCS pressure and OTSG level to reach andmaintain those parameters.

5.3.2 Allow temperatures to stabilize for 1 hour. During this period of time, read and record all operable thermocouples every ten minutes.

5.3.3 If any thermocouple exceeds 1200°F prior to successfully establishing Natural Circulation, go to step 5.4.

5.3.4 If any 3 thermocouples exceed 1000°F prior to successfully establishing Natural Circulation, go to step 5.4.

5.3.5 If at any time during stabilization or natural circulation attempts T_h in the loop with the OTSG in the steaming mode exceeds 500°F proceeed to step 5.4.

NOTE: Continue plotting T_H, T_C, and T_{stm} as in step 5.1.4.

5.4 If natural circulation is not established, go into HPI per EP33, starting with Step 3.2.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

April 11, 1979

IE Bulletin No. 79-06
~~Page 1 of 3~~

REVIEW OF OPERATIONAL ERRORS AND SYSTEM MISALIGNMENTS IDENTIFIED DURING THE THREE MILE ISLAND INCIDENT

As previously discussed in IE Bulletin 79-05 and 79-05A, the Three Mile Island Nuclear Power Plant, Unit 2 experienced significant core damage which resulted from a series of events initiated by a loss of feedwater transient and compounded by operational errors. Several aspects of the incident have generic applicability to all light water power reactor facilities, in addition to those previously identified as generic to Babcock and Wilcox reactors. This bulletin is to identify certain actions to be taken by all other light water power reactor facilities with an operating license.

Action to be taken by licensees:

For all light water power reactor facilities with an operating license except Babcock and Wilcox reactors:

1. Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 3/28/79 accident included in Enclosure 1 to IE Bulletin 79-05A.

   a. This review should be directed toward understanding: (1) the extreme seriousness and consequences of the simultaneous blocking of both auxiliary feedwater trains at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident; (2) the operational errors which led to the eventual core damage; and (3) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.

   b. Operations personnel should be instructed to: (1) not override automatic action of engineered safety features without careful review of plant conditions; and (2) not make operational decisions based on a single plant parameter indication when a confirmatory indication is available.

   c. All licensed operators and plant management and supervision with operational responsibilities shall participate in this review and such participation shall be documented in plant records.

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4. ~~7~~. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to cause containment isolation of all lines whose isolation does not degrade core cooling capability upon automatic initiation of safety injection.

5. ~~8~~. For pressurized water reactor facilities for which the reactor protection system does not initiate automatic starting of ~~auxiliary~~ the steam generator auxiliary feedwater systems, ~~develop~~ prepare and implement immediately procedures which require the stationing of an individual (with no other assigned concurrent duties and in direct and continuous communication with the control room) to promptly initiate auxiliary feedwater to the steam ~~generators~~ generator(s) for those transients ~~and~~ or accidents ~~requiring~~ the consequences of which can ~~be~~ be limited by such action.

6. ~~9~~. For all pressurized water reactors, prepare and implement immediately procedures which:

a. Identify those plant indications (such as valve discharge piping temperature, valve position indication, or valve discharge relief tank temperature or pressure indication) which plant operators may utilize to determine that pressurizer power operated relief valve(s) are open, and

175 176

1. Direct the plant operators to manually
Close the ~~relief~~ power operated relief block
Valve(s) ~~valve(s)~~ when reactor coolant system
pressure is reduced to the set point
for normal automatic closure of the
power operated relief valve(s).

2. For pressurized water reactor facilities review the actions required by your operating procedures for coping with transients and accidents, with particular attention to:

   a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability, especially natural circulation capability.

   b. Operator action required to prevent the formation of such voids.

   c. Operator action required to enhance core cooling in the event such voids are formed.

3. For pressurized water reactor facilities that use pressurizer water level coincident with pressurizer pressure for automatic initiation of safety injection into the reactor coolant system, instruct operators to manually initiate safety injection when the pressurizer pressure indication reaches the actuation set point whether or not the level indication has dropped to the actuation set point. Note that this recommendation has been made by Westinghouse to its reactor customers.

Insert items 4-5 attached

7. ~~4.~~ Review the action directed by the operating procedures and training instructions to ensure that:

   a. Operators do not override automatic actions of engineered safety features without careful review of plant conditions.

   b. Operators are provided additional information and instructions to not rely upon any one plant parameter but to also examine other related indications in evaluating plant conditions.

8 ~~5~~. Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance, ~~and~~ testing, plant and system startup, and supervisory periodic (daily/shift checks, etc.) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their ~~correct~~ proper positions during all operational modes.

9 ~~6~~. Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping, venting or other release of radioactive liquids and gases will not occur inadvertently.

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In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

a. Whether interlocks exist to prevent transfer when high radiation indication exists, and

b. Whether such systems are isolated by the containment isolation signal.

c. The basis on which continued operability of the above features is assured.

10 #7. Review and modify as necessary your maintenance and test procedures to ensure that they require:

a. Verification, by test or inspection per technical specifications, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

c. Explicit notification of involved reactor operating personnel whenever a safety-related system is removed from and returned to service.

11. #8. Review your prompt reporting procedures for NRC notification to assure very early notification of serious events. ~~Immediate notification of NRC shall be made following any incident where state~~

For all light water power reactor facilities with an operating license except Babcock and Wilcox reactors, respond to Items 1-7 within 14 days of the receipt of this Bulletin.

Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D.C. 20555.

For all other power reactors with an operating license or construction permit, this Bulletin is for information purposes and no written response is required.

Approved by GAO, B180225 (R0072); clearance expires 7/31/80. Approval was given under a blanket clearance specifically for identified generic problems.

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LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
78-05	Malfunctioning of Circuit Breaker Auxiliary Contact Mechanism-General Model CR105X	4/14/78	All Power Reactor Facilities with an OL or CP
78-06	Defective Cutler-Hammer, Type M Relays With DC Coils	5/31/78	All Power Reactor Facilities with an OL or CP
78-07	Protection afforded by Air-Line Respirators and Supplied-Air Hoods	6/12/78	All Power Reactor Facilities with an OL, all class E and F Research Reactors with an OL, all Fuel Cycle Facilities with an OL, and all Priority 1 Material Licensees
78-08	Radiation Levels from Fuel Element Transfer Tubes	6/12/78	All Power and Research Reactor Facilities with a Fuel Element transfer tube and an OL.
78-09	BWR Drywell Leakage Paths Associated with Inadequate Drywell Closures	6/14/79	All BWR Power Reactor Facilities with an OL or CP
78-10	Bergen-Paterson Hydraulic Shock Suppressor Accumulator Spring Coils	6/27/78	All BWR Power Reactor Facilities with an OL or CP

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LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
78-11	Examination of Mark I Containment Torus Welds	7/21/78	BWR Power Reactor Facilities for action: Peach Bottom 2 and 3, Quad Cities 1 and 2, Hatch 1, Monticello and Vermont Yankee
78-12	Atypical Weld Material in Reactor Pressure Vessel Welds	9/29/78	All Power Reactor Facilities with an OL or CP
78-12A	Atypical Weld Material in Reactor Pressure Vessel Welds	11/24/78	All Power Reactor Facilities with an OL or CP
78-12B	Atypical Weld Material in Reactor Pressure Vessel Welds	3/19/79	All Power Reactor Facilities with an OL or CP
78-13	Failures In Source Heads of Kay-Ray, Inc., Gauges Models 7050, 7050B, 7051, 7051B, 7060, 7060B, 7061 and 7061B	10/27/78	All general and specific licensees with the subject Kay-Ray, Inc. gauges
78-14	Deterioration of Buna-N Components In ASCO Solenoids	12/19/78	All GE BWR facilities with an OL or CP
79-01	Environmental Qualification of Class IE Equipment	2/8/79	All Power Reactor Facilities with an OL or CP

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LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
79-02	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	3/2/79	All Power Reactor Facilities with an OL or CP
79-03	Longitudinal Weld Defects In ASME SA-312 Type 304 Stainless Steel Pipe Spools Manufactured By Youngstown Welding and Engineering Co.	3/12/79	All Power Reactor Facilities with an OL or CP
79-04	Incorrect Weights for Swing Check Valves Manufactured by Velan Engineering Corporation	3/30/79	All Power Reactor Facilities with an OL or CP
79-05	Nuclear Incident at Three Mile Island	4/1/79	All B&W Power Reactor Facilities with an OL
79-05A	Nuclear Incident at Three Mile Island	4/5/79	All B&W Power Reactor Facilities with an OL

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(Draft letter to ~~light~~ pressurized water power reactor facilities other than B&W with an operating license.)

IE Bulletin No. 79-06

Addressee:

Enclosed is IE Bulletin No. 79-06, which requires action by you with regard to your ~~power~~ pressurized water reactor facility(ies) with an operating license.

Based on our current understanding of the Three Mile Island accident sequence, and discussion with the designer of your pressurized water reactor, we have reason to believe that pressurizer level indication in your facility may not provide reliable information regarding level in the reactor coolant system under certain transient or accident condition. You should immediately instruct your operating personnel accordingly. In addition you should consider this possibility in responding to the enclosed bulletin.

Should you have any questions regarding this Bulletin or the actions required by you, please contact this office.

Sincerely,

Signature
(Regional Director)

Enclosure:
IE Bulletin No. 79-06
with Enclosures

(Draft letter to light water power reactor facilities other than B&W with an operating license.)

IE Bulletin No. 79-06

Addressee:

Enclosed is IE Bulletin No. 79-06, which requires action by you with regard to your power reactor facility(ies) with an operating license.

Based on our current understanding of the Three Mile Island accident sequence, and discussion with the designer of your pressurized water reactor, we have reason to believe that pressurizer level indication in your facility may ~~lead to erroneous inferences of~~ not provide reliable information regarding level in the reactor coolant system under certain transient or accident condition. You should immediately instruct your operating personnel accordingly. In addition you should consider this possibility in responding to the enclosed bulletin.

Should you have any questions regarding this Bulletin or the actions required by you, please contact this office.

Sincerely,

Signature
(Regional Director)

Enclosure:
IE Bulletin No. 79-06
with Enclosures

Ok DR
for NRR

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

April 11, 1979

IE Bulletin No. 79-06

REVIEW OF OPERATIONAL ERRORS AND SYSTEM MISALIGNMENTS IDENTIFIED DURING THE THREE MILE ISLAND INCIDENT

As previously discussed in IE Bulletin 79-05 and 79-05A, the Three Mile Island Nuclear Power Plant, Unit 2 experienced significant core damage which resulted from a series of events initiated by a loss of feedwater transient and apparently compounded by operational errors. Several aspects of the incident have generic applicability to all light water power reactor facilities, in addition to those previously identified as ~~generic to~~ applicable to Babcock and Wilcox reactors. This bulletin is to identify certain actions to be taken by all other light water power reactor facilities with an operating license. Actions previously have been required of licensees with B&W reactors.

Action to be taken by licensees:

For all ~~light~~ water power reactor facilities with an operating license except Babcock and Wilcox reactors:

1. Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 3/28/79 accident included in Enclosure 1 to IE Bulletin 79-05A.

   a. This review should be directed toward understanding: (1) the extreme seriousness and consequences of the simultaneous blocking of both auxiliary feedwater trains at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident; (2) the apparent operational errors which led to the eventual core damage; and (3) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.

   b. Operations personnel should be instructed to: (1) not override automatic action of engineered safety features without careful review of plant conditions; and (2) not make operational decisions based on a single plant parameter indication when a confirmatory indication is available.

   c. All licensed operators and plant management and supervision with operational responsibilities shall participate in this review and such participation shall be documented in plant records.

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2. For pressurized water reactor facilities review the actions required by your operating procedures for coping with transients and accidents, with particular attention to:

   a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability, especially natural circulation capability.

   b. Operator action required to prevent the formation of such voids.

   c. Operator action required to enhance core cooling in the event such voids are formed.

3. For pressurized water reactor facilities that use pressurizer water level coincident with pressurizer pressure for automatic initiation of safety injection into the reactor coolant system, instruct operators to manually initiate safety injection when the pressurizer pressure indication reaches the actuation set point whether or not the level indication has dropped to the actuation set point. ~~Note that this recommendation has been made by Westinghouse to its reactor customers~~.

4. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to cause containment isolation of all lines whose isolation does not degrade core cooling capability upon automatic initiation of safety injection.

5. For pressurized water reactor facilities for which the ~~reactor protection system does not initiate automatic starting of the steam generator~~ auxiliary feedwater system is not automatically initiated, prepare and implement immediately procedures which require the stationing of an individual (with no other assigned concurrent duties and in direct and continuous communication with the control room) to promptly initiate auxiliary feedwater to the steam generator(s) for those transients or accidents the consequences of which can be limited by such action.

6. For all pressurized water reactors, prepare and implement immediately procedures which:

   a. Identify those plant indications (such as valve discharge piping temperature, valve position indication, or valve discharge relief tank temperature or pressure indication) which plant operators may utilize to determine that pressurizer power operated relief valve(s) are open, and

b. Direct the plant operators to manually close the power operated relief block valve(s) when reactor coolant system pressure is reduced to the set point for normal automatic closure of the power operated relief valve(s) and the valve(s) fail to close.

7. Review the action directed by the operating procedures and training instructions to ensure that:

   a. Operators do not override automatic actions of engineered safety features without careful review of plant conditions.

   b. Operators are provided additional information and instructions to not rely upon any one plant parameter but to also examine other related indications in evaluating plant conditions.

8. Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance, testing, plant and system startup, and supervisory periodic (daily/shift checks, etc.) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

9. Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping, venting or other release of radioactive liquids and gases will not occur inadvertently.

   In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

   a. Whether interlocks exist to prevent transfer when high radiation indication exists, and

   b. Whether such systems are isolated by the containment isolation signal.

   c. The basis on which continued operability of the above features is assured.

10. Review and modify as necessary your maintenance and test procedures to ensure that they require:

a. Verification, by test or inspection per technical specifications, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

c. Explicit notification of involved reactor operating personnel whenever a safety-related system is removed from and returned to service.

11. Review your prompt reporting procedures for NRC notification to assure very early notification of serious events.

For all ~~light~~ pressurized water power reactor facilities with an operating license except Babcock and Wilcox reactors, respond to Items ~~1-7~~ 1-11 within ~~14~~ 10 days of the receipt of this Bulletin.

Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D.C. 20555.

For all other power reactors with an operating license or construction permit, this Bulletin is for information purposes and no written response is required.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

April 11, 1979

IE Bulletin No. 79-06

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   b. Operations personnel should be instructed to: (1) not override automatic action of engineered safety features without careful review of plant conditions; and (2) not make operational decisions based on a single plant parameter indication when a confirmatory indication is available.

   c. All licensed operators and plant management and supervision with operational responsibilities shall participate in this review and such participation shall be documented in plant records.

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2. ~~For pressurized water reactor facilities~~ review the actions required by your operating procedures for coping with transients and accidents, with particular attention to:

   a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability, especially natural circulation capability.

   b. Operator action required to prevent the formation of such voids.

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3. For pressurized water reactor facilities that use pressurizer water level coincident with pressurizer pressure for automatic initiation of safety injection into the reactor coolant system, instruct operators to manually initiate safety injection when the pressurizer pressure indication reaches the actuation set point whether or not the level indication has dropped to the actuation set point. [~~Note~~ ? that this recommendation has been made by Westinghouse to its reactor customers.]

4. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to cause containment isolation of all lines whose isolation does not degrade core or cont ht removal or other ESF cooling capability upon automatic initiation of safety injection.

5. For pressurized water reactor facilities for which the reactor protection system does not initiate automatic starting of the steam generator auxiliary feedwater system, prepare and implement immediately procedures which require the stationing of an individual (with no other assigned concurrent duties and in direct and continuous communication with the control room) to promptly initiate auxiliary feedwater to the steam generator(s) for those transients or accidents the consequences of which can be limited by such action. must ?

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9. Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping, venting or other release of radioactive liquids and gases will not occur inadvertently.

   In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

   a. Whether interlocks exist to prevent transfer when high radiation indication exists, and

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   c. The basis on which continued operability of the above features is assured.

10. Review and modify as necessary your maintenance and test procedures to ensure that they require:

175 191

a. Verification, by test or inspection per technical specifications, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

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c. Explicit notification of involved reactor operating personnel whenever a safety-related system is removed from and returned to service.

11. Review your prompt reporting procedures for NRC notification to assure very early notification of serious events.

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Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D.C. 20555.

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April 11, 1979

apparently

IE Bulletin No. 79-06

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175 193

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April 2, 1979

POST SHUTDOWN EVENTS
(Next 6 months)

TO: Brian Grimes
From: John Austin
4/4/79

EVENT	EXPECTED RESPONSE	RELEASE AND TIME	WARNING TIME	EVACUATION SCENARIO
1. Significant breach of containment	Re-isolate containment within 1 hour	"Puff" Release		Stay inside 5 miles
Example: Penetration seal fails	Integrity uncertain	Significant continuous release		Evac. 2 mi stay inside 5 miles
2. Failure involving primary coolant water outside containment	Small leak, less than 1 gal/hour	Continuous release		Possible pre-cautionary evac. 2 mi, stay inside 5 miles
Examples: RHR pumps leak; RHR pipe rupture	Large leak, 50 gal/min.	Significant continuous release		Evac. 2 mi, stay inside 5 miles
3. Failure involving storage tanks (e.g. in auxiliary building)	Vent valve fails, reseats within one hour	"Puff" release		Stay inside 2 miles
	Vent valve fails, no reseat	Significant continuous release		Evac 2 mi, stay inside 5 miles

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4/5/79

EFFORTS RELATED TO EQUIPMENT SURVIVABILITY

I FEEDBACK ON RADIATION WITHSTAND CAPABILITY OF FOXBORO AND BAILEY TRANSMITTERS

<u>Naval Reactors</u> No experience with either Bailey or Foxboro transmitters (of Designs used in commercial plant).

<u>Sandia</u> No experience in weapons area. However, Sandia contacted Bailey and Foxboro. No additional information on Bailey's, but certain Foxboro pressure transmitters are equipped with hardened amplifiers (RC 22-PT1 through 8 which are used to measure reactor coolant pump seal cavity pressure). They have survived tests involving exposure to 2×10^8 R/hr. Therefore, should existing reactor coolant pump pressure transmitters fail, these hardened transmitters should be available as backup to measure reactor coolant pressure.

II ESTIMATES OF FAILURE TIMES FOR PRESSURE TRANSMITTERS AND OTHER VITAL EQUIPMENT DUE TO RADIATION

Approach: Knowing radiation withstand capability of equipment (based on test), calculate dose rate due to containment atmosphere and water at bottom of containment, and estimate equipment and lifetime.

Basis for Dose Rate Calculation:

(a) Assume containment air sample represents containment atmosphere.

(b) Assume water in containment has same constituents as coolant sample taken on 3-31-79.

Results: ORNL has calculated dose at location of pressure transmitters to be 1×10^4 R/hr.

Bailey transmitter BY is qualified to 1×10^5 R Using dose of 1×10^4 R/hr, one would expect this instrument to survive only about 10 hours. Since several of the Bailey transmitters continue to function, we are assuming that the dose calculation is grossly in error. Without a better estimate of the source terms, particularly a good estimate

175 198

of the activity of the water in the containment sump, it is impossible to estimate failure times of vital equipment.

III ASSESSMENT OF CRITICAL EQUIPMENT

- Have established design radiation level and location inside containment of all Foxboro, Bailey pressure and differential pressure transmitters and Rosemount Temperature sensors (see Enclosure 1).

- Have established location of all critical components of decay heat removal system (have not established design radiation levels).

- We are looking at reactor coolant pumps (Allis-Chalmers). We will provide additional information later.

- Fan cooler motors - we have established that they were tested to 10^9 Rads.

- Containment isolation valves - we don't believe external radiation is a problem. Radiation inside of valve is estimated to be 2000 R/hr. Assuming seat can take $\sim 10^6$ to 10^7, seat will last 22 days. However, radiation level should drop with time. Therefore, valve seat may last substantially longer.

- Effect of dumping 250 gpm of coolant on containment floor has been calculated (see Enclosure 2 and Enclosure 3) - times when various pieces of equipment would be flooded are enclosed.

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B+W INSTRUMENTS INSIDE CONTAINMENT (BACKBONE)

4-3-79

IM - INST MOUNTED BY ITSELF
IR - INST MOUNTED ON RACK

FOR PRESENT OPERATING MODE

PARAMATER	INST IDENT #	MOUNTING	TYPE	RAD LEVEL DESIGN TEST	RAD LEVEL AT EQUIP	ABOVE 272'6" LEVEL
S.G."B" PRESS	SPGB-PT1	IM 13	FOX E11GH	10^7		2'5"
RC FLOW LP A	RC 14A-DPT-3+4	IM 14+15	BAILEY BY	10^5-10^7		3'0"
PRZ LEVEL ~~PRESS~~	RC1-LT 1,2+3	IR-424+425	BAILEY BY	10^5-10^7		3'6"
SG "B" PRESS	SP6B-PT2	IR-428	FOX E11GH	10^7		
RC FLOW LP A	RC14A-DPT1+2	IR 425+426	BAILEY BY	10^5-10^7		
RC FLOW LP B	RC14B-DPT3+4	IR-429+430	BAILEY BY	10^5-10^7		
RC FLOW LP B	RC14B-DPT1+2	IM-12+13	BAILEY BY	10^5-10^7		
SG "A" LEVEL	SP1A-LT 2+3	IR-426	BAILEY BY	10^5-10^7		
SG "A" LEVEL (S.U.)	SP1A LT4+5	IR-426	BAILEY BY	10^5-10^7		
SG "B" LEVEL	SP1B LT1,2+3	IR-428	BAILEY BY	10^5-10^7		
SG "B" LEVEL (S.U.)	SP1B LT4+5	IR-428	BAILEY BY	10^5-10^7		
SG A LEVEL (	SP1A LT1	IR-426	BAILEY BY	10^5-10^7		5'2"
SG "A" PRESS	SP6A PT1+2	IR-426+424	FOX E11GH	10^7		
RC PRESS (W.R.)	RC3A PT3+4	IR 425+427	FOX E11GH	10^7		
RC PRESS (N.R.)	RC3A PT5	IR 424	FOX E11GH	10^7		
RC PRESS (W.R.)	RC3B PT3	IR 429	FOX E11GH	10^7		
RC TEMP A LOOP (NR T_c)	RC5A TE 2+4		ROSEMOUNT 177Y	10^7		
RC TEMP B LOOP (NR T_c)	RC5B TE 2+4		ROSEMOUNT 177Y	10^7		
RC TEMP A LOOP (WR T_H)	RC15A TE1					
RC TEMP A LOOP (WR T_c)	RC15A TE2+3					
RC TEMP B LOOP (WR T_H)	RC15B TE1					
RC TEMP B LOOP (NR T_c)	RC15B TE2+3					
PRZ TEMP	RC2 TE 1+2					

Enclosure 2

FLOOD TIMES

@ 250 GPM = 15000 GPH
12" = 74,330 gal
1" = 6194.2
15,000 G/HR = 2.42"/HR.

Starting Point = 2 feet

2 hrs 10 min — Steam Gen Press Loop B (1 of 2)

4 hrs 50 min — Reactor Coolant flow Loop A (2 of 4)

7 hrs 30 min — Press Level (3 of 3)
Steam Generator Press Loop B (2 of 2)
Reactor Cool Flow Loop A (2 of 4)
React Cool Flow Loop B (2 of 4)
(Loop A) (5 Total) { Steam Gen Lev (Operate Range) (2 of 5)
Steam Gen Lev (Start-up Range)(4 of 5) }
(Loop B) (5 Total) { Stm Gen Lev Full Range(1 of 5)
Stm Gen Lev (Operate)(3 of 5)
Stm Gen Level (Startup Range)(5 of 5) }

15 hrs 45 min — Stm Gen LevelA (full Range) 5 of 5) ==
Stm Gen Press Loop A(2 of 2)
React Cool Press Loop A (2 of 2) Wide Range
React Cool Press Loop A (1 of 1) Low Range
React Cool Press Wide (1 of 1) Loop B

Penetrations (Start) 39 hrs, 40 min

D.H Valves for shutdown cooling 49 hrs, 40 min

Reactor Coolant PUmps 8 day, 16 hrs, 50 min.

175 201

Enclosure 3

TRANSMITTER SUBMERGED STATUS

Elevation above ground level

2'-5" .-----------------------Steam Gen. Press. Loop B (1of2)

3'-0" -----------------------Reactor Coolant Flow Loop A (2of4)

3'-4" ---- ------------------Flood Level (Based on transmitted site calculations of 4/5/79)

3'-6" -----------------------Press. Level (3of3)

Stm. Gen. Press Loop B (2of2)

React Cool. Flow LoopA (2of4)

React. Cool. Flow Loop B (2of4)

Stm. Gen Lev. Loop A (operate range)(2of5)

" " "" " (start-up range)(4of5)

" "" " " B (Full range) (1of5)

" " " " B (operate range) (3of5)

" " " " B (start up range) (5of5)

React. Cool. Pump Seal Cavity Press. (8of8)

5'-2" -----------------------Stm. Gen. Lev. Loop A (Full range) (5of5)

Stm. Gen. Press. Loop A (2of2)

React. Cool. Press. Loop A (2of2)Wide range.

" " " " A (1of1)Low range.

" " " Loop B (1of1) wide range.

10'-6" -----------------------Bottom of the lowest Electrical Penetration

12'-0" -----------------------Bottom of the motor housing of the D.H.Valves for shutdown cooling.

46'-0'-----------------------Bottom of the motor housing of the Reactor Cooling Pumps.

175 202

Status of B&W Reactors - 4/2/79

Operating

	UNIT	STATUS	STATE	ARCHITECT/ENGINEER
1.	Arkansas Unit 1	Shut down 3/30/79 for reload	Arkansas	Bechtel
2.	Crystal River 3	Operating	Florida	Gilbert
3.	Davis Besse 1	Shut down for repair to inoperative relief valve	Ohio	Bechtel
4.	Oconee 1, 2, 3	Operating	South Carolina	Bechtel
5.	Rancho Seco	Operating	California	Bechtel
6.	Three Mile Island 1	Shut down because of Unit 2	Pennsylvania	Gilbert

Under Construction - Status as of April 3 (Blue Book/DPM list/NRC Caseload Forecast Panel)

Note: This information was obtained from Roger Boyd on PM of 4/3/79 from Caseload Forecast Panel files.

		FUEL LOAD DATE			
		Licensee	NRC Caseload Forecast Panel	STATE	ARCHITECT/ENGINEER
1.	Midland 1	11/80	11/81	Michigan	Bechtel
2.	Midland 2	11/81	11/82	Michigan	Bechtel
3.	Bellefonte 1	3/82	3/82	Alabama	TVA
4.	Bellefonte 2	12/82	12/82	Alabama	TVA
5.	WPPSS 1	6/83	6/83	Washington	United Engineers & Constructors
6.	WPPSS 4	12/84	12/84	Washington	United Engineers & Constructors
7.	North Anna 3	11/81	6/86	Virginia	Stone & Webster
8.	North Anna 4	12/82	12/86	Virginia	Stone & Webster

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	No CP	FUEL LOAD DATE Licensee	NRC Caseload Forecast Panel	STATE	ARCHITECT/ENGINEER
9.	Davis Besse 2	Company reevaluating on basis of need for power		Ohio	Bechtel
10.	Davis Besse 3			Ohio	Bechtel
11.	Erie 1	12/87	---	Ohio	Commonwealth Associates
12.	Erie 2	12/89	---	Ohio	Commonwealth Associates
13.	Pebble Springs 1	4/[illegible]	---	Oregon	Bechtel
14.	Pebble Springs 2	4/89	---	Oregon	Bechtel
15.	Greene County	7/86	---	New York	Stone & Webster
16.	Greenwood 2	7/88	---	Michigan	Bechtel
17.	Greenwood 3	7/88	---	Michigan	Bechtel
18.	Carolina 8	Postponed indefinitely		North Carolina	
19.	Carolina 9	Postponed indefinitely		North Carolina	
20.	Vandalia	Postponed indefinitely		Iowa	Bechtel

175 2014

PLANNING MEETING 4/7/79

Copies
Denton
Mattson
Ross
Vollmer

- BASE CASE SUMMARY
- FLOW CHART FOR BASE PLAN
- TASK LISTS
  - IAG
  - PLT OPS
  - TECH SUPPORT GROUP
  - WASTE MANAGEMENT GROUP
  - PLANT MODIFICATIONS GROUP
  - B+W

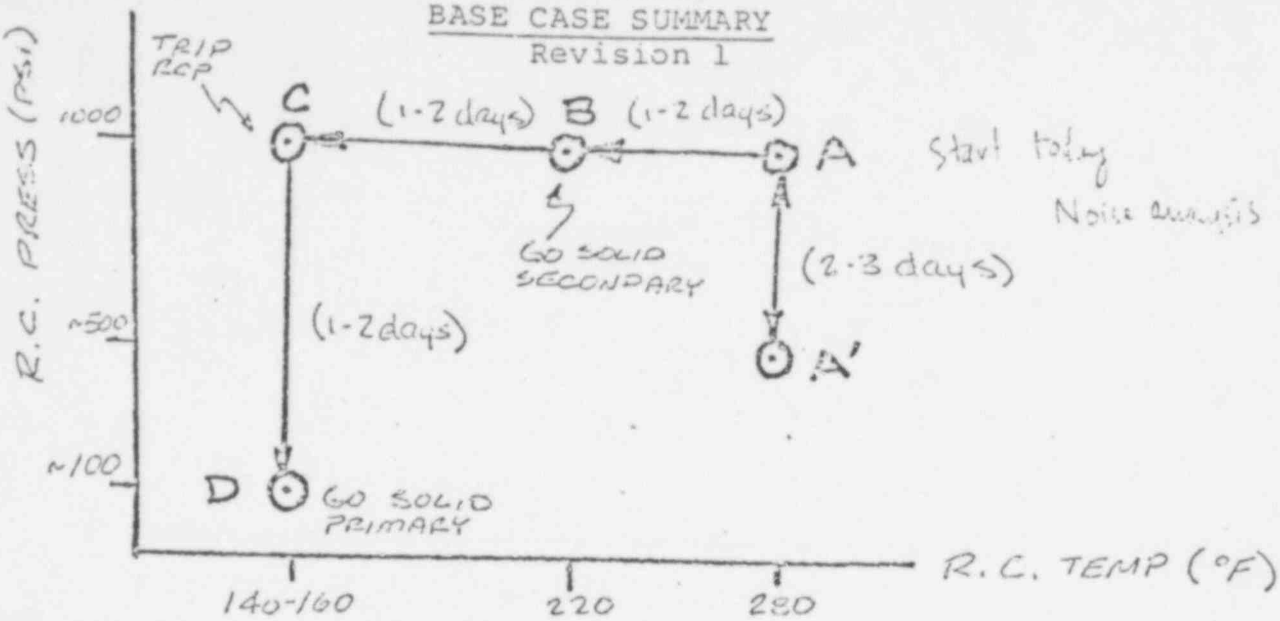
Gossick

Comments by noon today

175 205

4/6/79
1100 hrs.

TMI RECOVERY
BASE CASE SUMMARY
Revision 1



(1) Degas at A; Lower Pressure (A→A') while degassing, then return to A.

(2) Continue Design/Installation of static and active systems for primary makeup/pressure control and secondary cooling system for "B" S/G.

(3) Reduce temperature (A→B) by steaming on "A" S/G

(4) Take "A" S/G solid - drop primary temp. to minimum (B→C)

(5) Trip RC Pump "A" - Establish natural convection - Establish cooling to "B" S/G if available.
A and B if B is available.

(6) Drop primary pressure to selected value (C→D)

(7) Take primary system solid - Control pressure & makeup with static or new active system

END POINT

Primary - Natural Circ. solid liquid, Long-term P/V Control

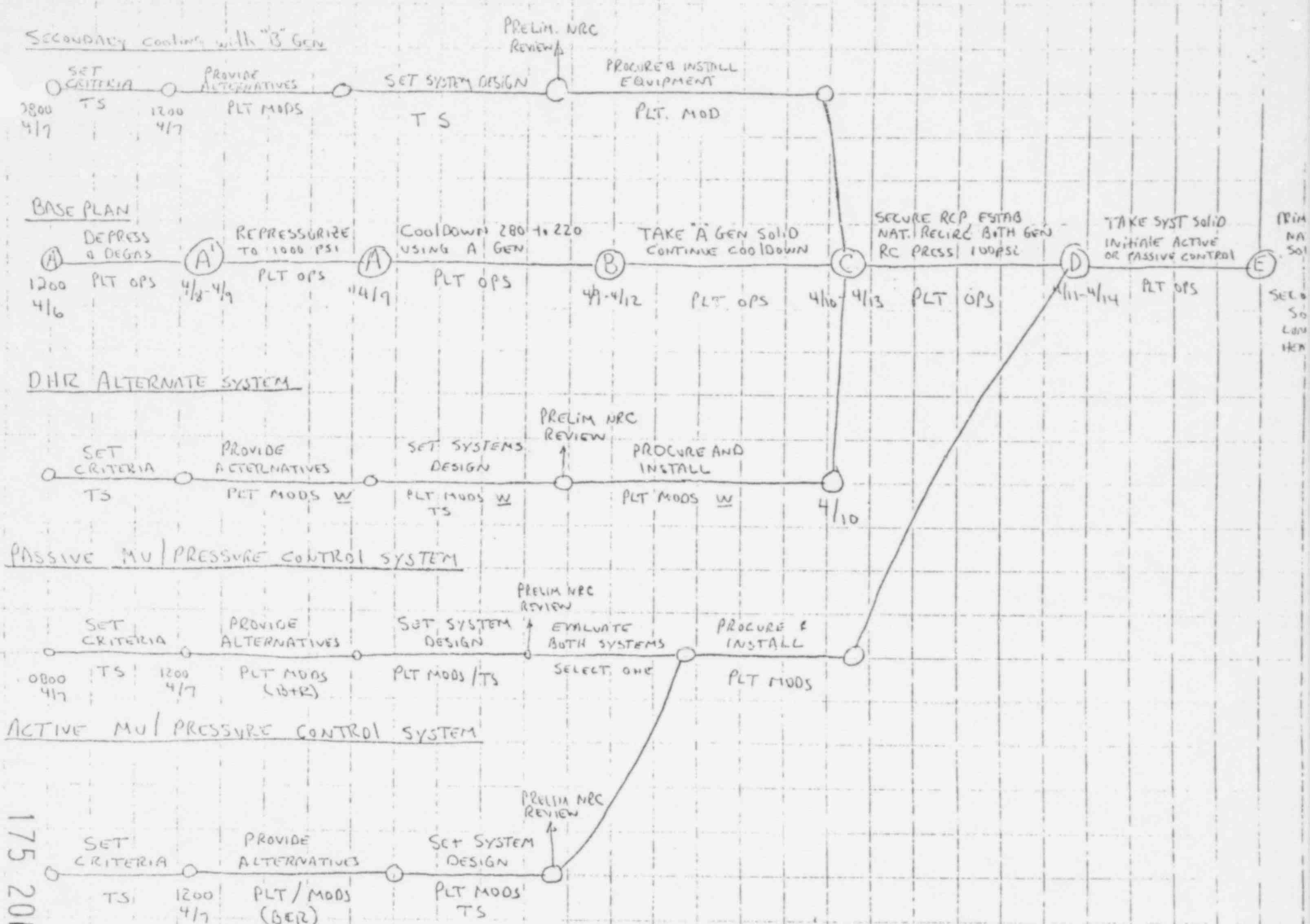
Secondary - Solid water, Long-term Heat Dump System

Approved for Issue:

R. Arnold

RA:clb

175 206


SECONDARY cooling with "B" GEN
SET CRITERIA
TS
2800 4/7
PROVIDE ALTERNATIVES
1200 4/7
PLT MODS
SET SYSTEM DESIGN
T S
PRELIM. NRC REVIEW
PROCURE & INSTALL EQUIPMENT
PLT. MOD
BASE PLAN
A
DEPRESS & DEGAS
PLT OPS
1200 4/6
A'
REPRESSURIZE TO 1000 PSI
PLT OPS
4/8-4/9
A
COOLDOWN 280 to 220 USING A GEN
PLT OPS
4/9
B
TAKE "A" GEN SOLID CONTINUE COOLDOWN
PLT OPS
4/9-4/12
C
SECURE RCP ESTAB NAT. RECIRC BOTH GEN RC PRESS 100PSI
PLT OPS
4/10-4/13
D
TAKE SYST SOLID INITIATE ACTIVE OR PASSIVE CONTROL
PLT OPS
4/11-4/14
E
DHR ALTERNATE SYSTEM
SET CRITERIA
TS
PROVIDE ALTERNATIVES
PLT MODS W
SET SYSTEMS DESIGN
PLT MODS W TS
PRELIM NRC REVIEW
PROCURE AND INSTALL
PLT MODS W
4/10
PASSIVE MU / PRESSURE CONTROL SYSTEM
SET CRITERIA
TS
0800 4/7
PROVIDE ALTERNATIVES
1200 4/7
PLT MODS (B&R)
SET SYSTEM DESIGN
PLT MODS / TS
PRELIM NRC REVIEW
EVALUATE BOTH SYSTEMS
SELECT ONE
PROCURE & INSTALL
PLT MODS
ACTIVE MU / PRESSURE CONTROL SYSTEM
SET CRITERIA
TS
PROVIDE ALTERNATIVES
1200 4/7
PLT / MODS (B&R)
SET SYSTEM DESIGN
PLT MODS TS
PRELIM NRC REVIEW

175 2017

Industry Advisory Group

Task	Description	Priority	Status	Lead Man
				Levenson
1	Recommend if Pri. sample worth exposure	H		
2	Provide recommendation for alternative methods of P/V control	H		
3	Evaluate fire in containment	H	Complete	

175 208

PLANT OPERATIONS

Task	Description	Priority	Status	Lead Man
	Procedure for reducing containment vacuum	H		
1D	Verify let-down valve alignment of make-up system	H		Miller
1E	Restore Pressurizer Heater	H		Shovlin
2A	Robot proceedure	M		Miller
2B	Determine urgency reqt. for primary sample			Herbein
3	Improve TLD methods limit exposures	H		Grayber/ Bachofer
4	Determine source of high Iodine-AB elevator	H		Miller
6	Repair fitting on make-up tank to reactor bldg.	H		Miller
11	Qualify 5 men to enter Aux. Bldg.	H		Limroth
14	Clear south end warehouse	M		Gunn
16	Design/Install filters at vacuum pump discharge	M		Gunn
19A	Control/room Island access lst	M		Limroth
B	Security	M		Stacy
C	Fire fighting readiness/proceedures	M		Miller
22A	Develop list of Plant changes	M		Miller
B	Establish control room change control log	M		Miller
23	Procedure for Plant condition upon evacuation Update emergency plan	H		Miller

— TANK

175 2089

Plant Operations

Task	Description	Priority	Status	Lead Man
15	Install portable IWT system	M		Gunn

<u>TECHNICAL SUPPORT GROUP</u>

TASK	DESCRIPTION	PRIORITY	STATUS	LEAD MAN
1.	Provide Additional boiler capacity			
2.	Develop procedure for limiting containment vacuum			
3.	Evaluate need for backup HPI pump (Hydrolaser)			
4.	Provide estimate of required HPI flow for 200 to 2500 psi (degenerated state)			
5.	Reconstruction of event			
6.	Incfements for pressure decrease	H	Complete	Devine
7.	How to measure rate of degas	H		Devine
8.	Increase Letdown flow	H		Devine
9.	Investigate the use of sample line to degas	H		Devine
10.	Calculate Reactor Coolant System spray flow	M		Wallace
11.	Radiation monitor system desensitization	M		Devine
12.	Construct brick wall at Unit 1 HX Vault			McGuoy
13.	Provide degeneration procedures			
	A. Fire in containment			
	B. Fire in Auxiliary			
	C. Fire in other areas			
	D. Evacuation of control room			
	E. Breach of waste systems			

175 2181

WASTE MANAGEMENT GROUP

LIQUID WASTE

Task	Description	Priority	Status	Lead Man
#2	AB&FHB Filter Trains	H	Underway	S. Kraft
#11	Tank Inventory Status	H	Underway	McGoey - Plant Opr.
#23	Assessment CAP-GUN system	H	Underway	McGoey - Tames
#14	Arrangement Study-RB Contaminated Water	M		
#18	Flush System for AB Components	M		
#8	Determine Leakage Paths from Unit 2 to Unit 1	L		
#16	D/C Liquid Wastes Processing System	Long Term		
#19	Additive to Primary Water	Long Term		
#21	Reactor Building Sump Level Measurement	Long Term		

GAS WASTE

Task	Description	Priority	Status	Lead Man
#1	AB&FHB Filter Trains	H	Underway	Hirst/Dorn
#4	Evaluate and Upgrade Gas Release Monitors	H	Underway	Yarborough
#5	Replace Charcoal Filters	H	Underway	Pavlick/Fitrell
#15	D/C Emergency RB Gas Purge Clean-Up System	H	Underway	B&R
#7	Condensor Off-Gas Discharge Filter	M	Underway	Hirst
#9	Preheaters to FHB Vent Filters	M		
#10	Preheaters to FHB Vent Filters	M		

GENERAL

Task	Description	Priority	Status	Lead Man
#20	Develop Waste Management Game Plan	Long Term		Palmer
#24	Organize An Integrated, QA'd Radiation Survey	H		Lee/Palmer

175 212

WASTE MANAGEMENT GROUP (CONT'D)

TASK	DESCRIPTION	PRIORITY	STATUS	LEADMAN
	Sample AB/FH Bldg. for filter replacement indicating acceptable operation.	H		McConnell
	Provide alternate set of filters	M		McConnell
	Determine best solution to be used in Aux. Bldg. to maintain acceptable iodine limits.	H		McConnell
	Design Shield Wall at condensate demineralizers	M		McConnell

175 213

Plant Modifications

Task	Description	Priority	Status	Lead Man
WG-1	Design new AB/FB filter/structure	H	Done	
WG-2	Instructions for decon Aux. Bldg. using cap-gun Ion exchange process			
TS-1	Recommend methods to improve reliability of implant electrical supply			
TS-2	Design package for secondary side cooling of S/GB	H		
TS-3	Design package for use of secondary services cooler			
TS-4	Design system for measuring water level in containment			
TS-5	Develop method for flooding containment with 10^6 ft^3 of water			
TS-6	Design system for pressure make-up control of RCS	H		
1063	Design/procure HEPA and charcoal filters for condenser VP discharge		Complete	
1064	Review S/G cooldown scheme for reliabilty		Complete	
1082	Recommend portable filters for Aux. bldg. (location, type, power source, etc.)		Complete	

175 214

PLANT MODIFICATIONS

Task	Description	Priority	Status	Lead Man
1085	Design temporary shielding covers for DHR pits		On schedule complete 4/7	
1103 (?)	Evaluate line-up to use one decay heat and one spray pump		On Hold	
1004	Get design for waste gas to Cont. Bldg.		Complete	
1108	Review B&W natural circulation cooldown proc.		Complete	
19	Determine Aux. Bldg. TV locations to monitor DHR components (Mark up General Arr.)		Complete	
39	Provide electrical power supply for cross connecting RB with FHB purge filters		80% on hold since not needed for 2 wks.	
45	Determine leakage paths Unit 2---Unit 1		Complete	
52	Design supports for Cond. H line to surface condenser H hot CO-C-IB to make it as seismically capable as feasible		John Lucena to arrive site 4/7 with sketches calcs	
53	Investigate supply of new charcoal trays for Aux. purge in fuel handling syst.		Complete	
56	Examine 1E diesel generator to determine if BOP loads can be added		Intiated 4/4	
64	Review alternate cooling source for secondary		Intiated 4/4	

175 215

PLANT MODIFICATIONS

Task	Description	Priority	Status	Lead Man
65	Design waste gas system for pump down of RB to fuel pool		Intiated 4/4	
63	Supports for H.S. system in Turbine bldg. when filled (related to #52)			
66	Location for secondary plant diesel		Assigned 4/4	
70	Max P&T for DHR downstream of valve DH-V3		Assigned 4/5	
73	Back-up Power Source for secondary plant loads		Assigned 4/5	
74	Review fire protection for charcoal filter		Complete	
	Design/ Fab/Install shield plugs at DH vaults	M		

175 216

B & W

Task	Description	Priority	Status	Lead Man
1	Analysis of gas conc. in Primary system	H		
2	Provide list of critical systems for present conditions	H		
3	Analyze In-core thermocouples during LOFON 4/6	H		
4	Provide minimum allowable RCS pressure for degassing	H		
5	Provide stress Analysis for generator (points BtoC)	M		
6	Determine minimum primary system pressure (point D, Base Plan)	M		
7	Provide noise anaylsis of pressure during degassing	H		
8	Document of sequence of Plant conditions in base plan	L		
9	Develop procedure to determine pressurizer level using Heise Gauge			Rogers
10	Develop procedure for cooldown using OTSG's on natural circulation			Rogers
11	Core Anaylsis Program A. Thermocouples from Incores B. Neutron signals from Incores c Noise Levels			Rogers

175 2117

4/6/79
1300 Hrs.

ACTION ITEMS
TASK MANAGEMENT/SCHEDUEL MEETING

0900 4/6/79

		Action Party
1.	Establish Data Bank; identify cognizant person-inform R. Arnold	Wilson
2.	Provide list of activities in progress to F. Stern	Palmer Wilson
3.	Identify planning coordinator	Wilson Palmer Cobean
4.	Support for Base Plan, Rev. 1	
	- Determine minimum degassing pressure (Point A)	McMillan
	- Continue design/installation of static & active make-up/pressure.	Cobean
	- Design & install cooling system for "B" Stm. Gen.	Cobean
	- Stress analysis for Stm. Gen. (Points B to C)	McMillan
	- Determine minimum primary system pressure (Point D)	McMillan
	- Alert Noise Analysis Group; Determine reporting.	Wilson McMillan Herbein
5.	Assure that at least one containment spray pump remains available for at least the next week - 10 days.	F. Stern
6.	Need criteria for additional waste gas storage facility.	Palmer
7.	Need criteria for upgrading electrical supply system.	Cobean
8.	Locate additional air compressors.	Cobean
9.	Write administrative procedure.	Cobean
10.	Confirm status of upgrading current RHR System.	Westinghouse (Cobean)
11.	Consider how to develop required contingency plans; "Flesh out" Degeneration List.	Stern Arnold
12.	Arrange for outside organization to do required Safety Analysis.	Stern Arnold
13.	Increase primary system boron concentration to 3,000 - 4,000 ppm.	Arnold Herbein

175 218

NOTIFICATIONS

CONTACT	PERSON NOTIFIED	TIME	CONTACT	PERSON NOTIFIED	TIME
CHAIRMAN	DORIE		OPERATIONS OFFICER	THOMPSON	
EDO	GOSSICK		IAT/IE	WARD	[illegible]
WHITE HOUSE	SIT RM	0915	IAT/NRR	MILLER	0856
COMMISSIONER	KENNEDY	8:50	IAT/NMSS	DAVIDSON	0842
COMMISSIONER	GILINSKY / SECY	9:10	PA	FOUCHARD	[illegible]
COMMISSIONER	AHEARNE	9 AM	CA	PINE	0859
COMMISSIONER	BRADFORD / THOMPSON	0855	SP	✓	0915
DIRECTOR, IE	DAVIS		GUARD OFFICE	✓	0912
DIRECTOR, NRR	DENTON		NRC OPERATOR	JONES	0907
DIRECTOR, NMSS	DIRCKS		DOE EOC	✓	9:00
ROI/IE	MOSELEY		EPA ORP	✓	9:05
RGI/IE			ELD	ENGELHARDT	9:32
FFMSI/IE	SNIEZEK		IP	HAUBER	0923
SI/IE	HOWARD		ADM	DONOGHUE	0930
NRR IRACT	STELLO		DCPA	Op Center	9:25
NMSS IRACT	T. CARTER	8:50	DOD NMCC		0930
R.I.	GRIER		USDA	C.II, III, IV	10 AM
[illegible]	[illegible]	[illegible]	WIND	OCD ONE	
TRAINING		905	DIA	FORWARD	0935

Original notification
3/28/79

175 2189

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 6 1979

NOTE TO: XOOS Staff

FROM: D. Thompson

SUBJECT: WEEKEND MANNING OF OPERATIONS CENTER

XOOS MANNING

4/7/79	IRACT Support	EMT
0001-0800	Ward Paulus	Crews
0800-1600	Weiss Gower	Jordan
1600-2400	Hegner	Thompson
4/8/79		
0001-0800	Baci	Crews
0800-1600	Hegner	Thompson
1600-2400	Weiss	Jordan
4/9/79		
0001-0800	Baci	Crews

Dudley Thompson
D. Thompson
Executive Office for Operations
Support

175 2~~19~~ 20

PLANNING MEETING 4/7/79

Copies
Denton
Mattson
Ross
Vollmer

- BASE CASE SUMMARY
- FLOW CHART FOR BASE PLAN
- TASK LISTS
  - IAG
  - PLT OPS
  - TECH SUPPORT GROUP
  - WASTE MANAGEMENT GROUP
  - PLANT MODIFICATIONS GROUP
  - B+W

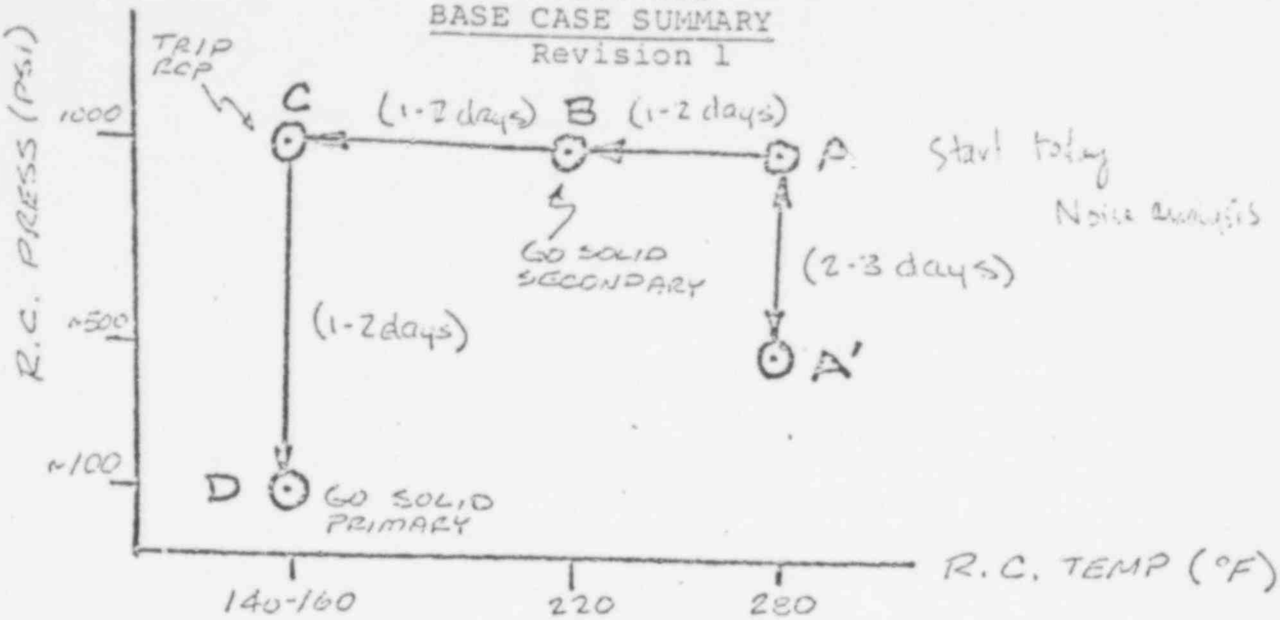
Gossick

Comments by noon today

175 2201

4/6/79
1100 hrs.

TMI RECOVERY
BASE CASE SUMMARY
Revision 1



(1) Degas at A; Lower Pressure (A→A') while degassing, then return to A.

(2) Continue Design/Installation of static and active systems for primary makeup/pressure control and secondary cooling system for "B" S/G.

(3) Reduce temperature (A→B) by steaming on "A" S/G

(4) Take "A" S/G solid - drop primary temp. to minimum (B→C)

(5) Trip RC Pump "A" - Establish natural convection - Establish cooling to "B" S/G if available.
A and B if B is available.

(6) Drop primary pressure to selected value (C→D)

(7) Take primary system solid - Control pressure & makeup with static or new active system

END POINT

Primary - Natural Circ, solid liquid, Long-term P/V Control

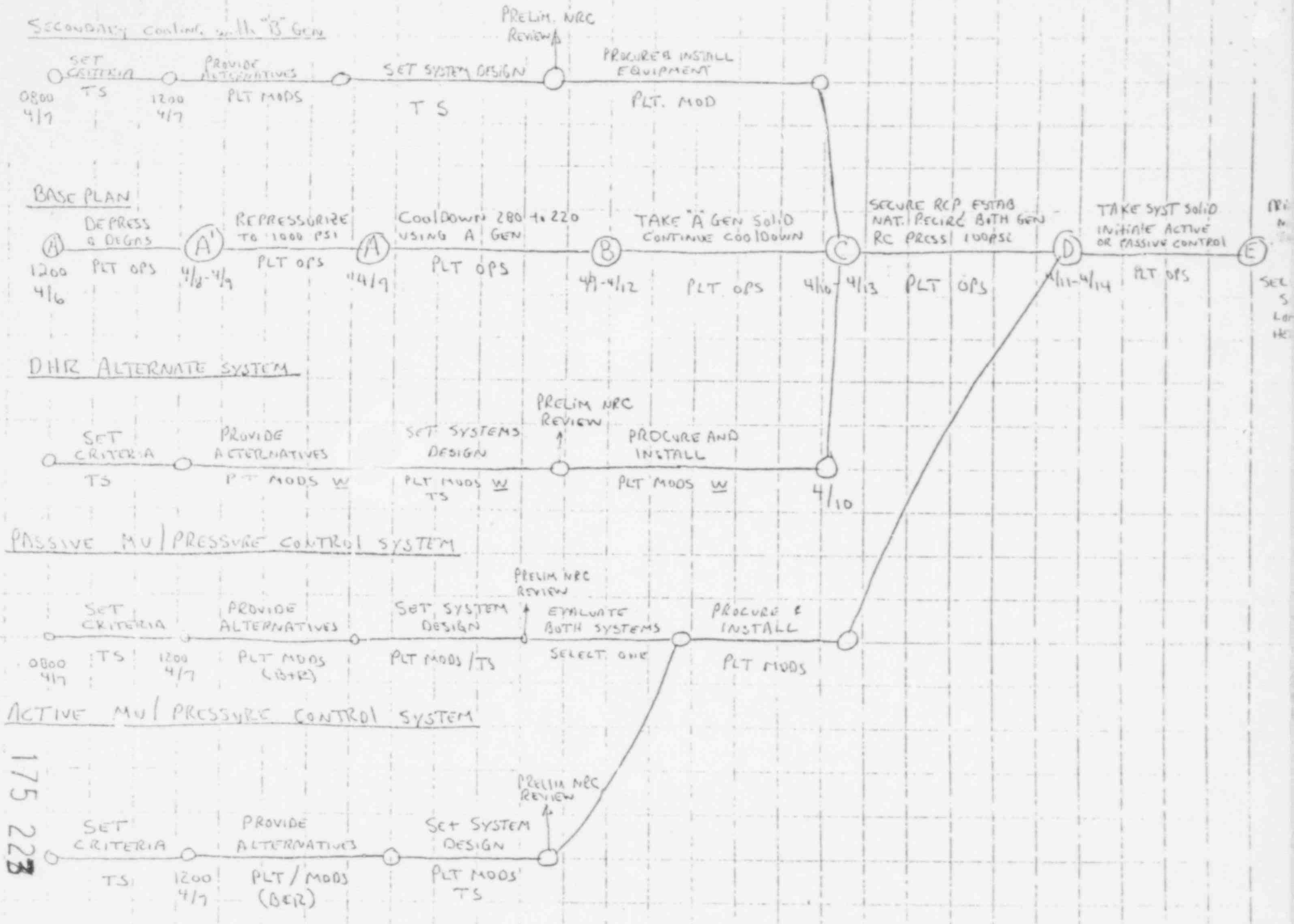
Secondary - Solid water, Long-term Heat Dump System

Approved for Issue:

R. Arnold

RA:clb

175 222


SECONDARY COOLING with "B" GEN
SET CRITERIA
TS
0800 4/7
PROVIDE ALTERNATIVES
PLT MODS
1200 4/7
SET SYSTEM DESIGN
T S
PRELIM. NRC REVIEW
PROCURE & INSTALL EQUIPMENT
PLT. MOD
BASE PLAN
A
DEPRESS & DEGAS
PLT OPS
1200 4/6
A'
4/8-4/9
REPRESSURIZE TO 1000 PSI
PLT OPS
A
4/9
COOLDOWN 280 to 220 USING A GEN
PLT OPS
B
4/9-4/12
TAKE "A" GEN SOLID CONTINUE COOLDOWN
PLT OPS
C
4/10-4/13
SECURE RCP ESTAB NAT. RECIRC BOTH GEN RC PRESS 100PSI
PLT OPS
D
4/11-4/14
TAKE SYST SOLID INITIATE ACTIVE OR PASSIVE CONTROL
PLT OPS
E
DHR ALTERNATE SYSTEM
SET CRITERIA
TS
PROVIDE ALTERNATIVES
PLT MODS W
SET SYSTEMS DESIGN
PLT MODS W TS
PRELIM NRC REVIEW
PROCURE AND INSTALL
PLT MODS W
4/10
PASSIVE MU/PRESSURE CONTROL SYSTEM
SET CRITERIA
TS
0800 4/7
PROVIDE ALTERNATIVES
PLT MODS (B&R)
1200 4/7
SET SYSTEM DESIGN
PLT MODS/TS
PRELIM NRC REVIEW
EVALUATE BOTH SYSTEMS
SELECT ONE
PROCURE & INSTALL
PLT MODS
ACTIVE MU/PRESSURE CONTROL SYSTEM
SET CRITERIA
TS
PROVIDE ALTERNATIVES
PLT/MODS (B&R)
1200 4/7
SET SYSTEM DESIGN
PLT MODS TS
PRELIM NRC REVIEW

175 223

Industry Advisory Group

Task	Description	Priority	Status	Lead Man
1	Recommend if Pri. sample worth exposure	H		Levenson
2	Provide recommendation for alternative methods of P/V control	H		
3	Evaluate fire in containment	H	Complete	

175 225

PLANT OPERATIONS

Task	Description	Priority	Status	Lead Man
	Procedure for reducing containment vacuum	H		
1D	Verify let-down valve alignment of make-up system	H		Miller
1E	Restore Pressurizer Heater	H		Shovlin
2A	Robot proceedure	M		Miller
2B	Determine urgency reqt. for primary sample			Herbein
3	Improve TLD methods limit exposures	H		Grayber/ Bachofer
4	Determine source of high Iodine-AB elevator	H		Miller
6	Repair fitting on make-up tank to reactor bldg.	H		Miller
11	Qualify 5 men to enter Aux. Bldg.	H		Limroth
14	Clear south end warehouse	M		Gunn
16	Design/Install filters at vacuum pump discharge	M		Gunn
19A	Control/room Island access lst	M		Limroth
B	Security	M		Stacy
C	Fire fighting readiness/proceedures	M		Miller
22A	Develop list of Plant changes	M		Miller
B	Establish control room change control log	M		Miller
23	Procedure for Plant condition upon evacuation Updace emergency plan	H		Miller

175 225

Plant Operations

Task	Description	Priority	Status	Lead Man
15	Install portable IWT system	M		Gunn

175 226

TECHNICAL SUPPORT GROUP

TASK	DESCRIPTION	PRIORITY	STATUS	LEAD MAN
1.	Provide Additional boiler capacity			
2.	Develop procedure for limiting containment vacuum			
3.	Evaluate need for backup HPI pump (Hydrolaser)			
4.	Provide estimate of required HPI flow for 200 to 2500 psi (degenerated state)			
5.	Reconstruction of event			
6.	Incfements for pressure decrease	H	Complete	Devine
7.	How to measure rate of degas	H		Devine
8.	Increase Letdown flow	H		Devine
9.	Investigate the use of sample line to degas	H		Devine
10.	Calculate Reactor Coolant System spray flow	M		Wallace
11.	Radiation monitor system desensitization	M		Devine
12.	Construct brick wall at Unit 1 HX Vault			McGuoy
13.	Provide degeneration procedures			
	A. Fire in containment			
	B. Fire in Auxiliary			
	C. Fire in other areas			
	D. Evacuation of control room			
	E. Breach of waste systems			

175 2217

WASTE MANAGEMENT GROUP

LIQUID WASTE

Task	Description	Priority	Status	Lead Man
#2	AB&FHB Filter Trains	H	Underway	S. Kraft
#11	Tank Inventory Status	H	Underway	McGoey - Plant Opr.
#23	Assessment CAP-GUN system	H	Underway	McGoey - Tomes
#14	Arrangement Study-RB Contaminated Water	M		
#18	Flush System for AB Components	M		
#8	Determine Leakage Paths from Unit 2 to Unit 1	L		
#16	D/C Liquid Wastes Processing System	Long Term		
#19	Additive to Primary Water	Long Term		
#21	Reactor Building Sump Level Measurement	Long Term		

GAS WASTE

Task	Description	Priority	Status	Lead Man
#1	AB&FHB Filter Trains	H	Underway	Hirst/Dorn
#4	Evaluate and Upgrade Gas Release Monitors	H	Underway	Yarborough
#5	Replace Charcoal Filters	H	Underway	Pavlick/Fitrell
#15	D/C Emergency RB Gas Purge Clean-Up System	H	Underway	B&R
#7	Condensor Off-Gas Discharge Filter	M	Underway	Hirst
#9	Preheaters to FHB Vent Filters	M		
#10	Preheaters to FHB Vent Filters	M		

GENERAL

Task	Description	Priority	Status	Lead Man
#20	Develop Waste Management Game Plan	Long Term		Palmer
#24	Organize An Integrated, QA'd Radiation Survey	H		Lee/Palmer

175 228

WASTE MANAGEMENT GROUP (CONT'D)

TASK	DESCRIPTION	PRIORITY	STATUS	LEADMAN
	Sample AB/FH Bldg. for filter replacement indicating acceptable operation.	H		McConnell
	Provide alternate set of filters	M		McConnell
	Determine best solution to be used in Aux. Bldg. to maintain acceptable iodine limits.	H		McConnell
	Design Shield Wall at condensate deminezalizers	M		McConnell

175 2219

Plant Modifications

Task	Description	Priority	Status	Lead Man
WG-1	Design new AB/FB filter/structure	H	Done	
WG-2	Instructions for decon Aux. Bldg. using cap-gun Ion exchange process			
TS-1	Recommend methods to improve reliability of implant electrical supply			
TS-2	Design package for secondary side cooling of S/GB	H		
TS-3	Design package for use of secondary services cooler			
TS-4	Design system for measuring water level in containment			
TS-5	Develop method for flooding containment with 10^6 ft^3 of water			
TS-6	Design system for pressure make-up control of RCS	H		
1063	Design/procure HEPA and charcoal filters for condenser VP discharge		Complete	
1064	Review S/G cooldown scheme for reliabilty		Complete	
1082	Recommend portable filters for Aux. bldg. (location, type, power source, etc.)		Complete	

175 2~~29~~ 30

PLANT MODIFICATIONS

Task	Description	Priority	Status	Lead Man
1085	Design temporary shielding covers for DHR pits		On schedule complete 4/7	
1103 (?)	Evaluate line-up to use one decay heat and one spray pump		On Hold	
1004	Get design for waste gas to Cont. Bldg.		Complete	
1108	Review B&W natural circulation cooldown proc.		Complete	
19	Determine Aux. Bldg. TV locations to monitor DHR components (Mark up General Arr.)		Complete	
39	Provide electrical power supply for cross connecting RB with FHB purge filters		80% on hold since not needed for 2 wks.	
45	Determine leakage paths Unit 2---Unit 1		Complete	
52	Design supports for Cond. H line to surface condenser H hot CO-C-IB to make it as seismically capable as feasible		John Lucena to arrive site 4/7 with sketches calcs	
53	Investigate supply of new charcoal trays for Aux. purge in fuel handling syst.		Complete	
56	Examine 1E diesel generator to determine if BOP loads can be added		Intiated 4/4	
64	Review alternate cooling source for secondary		Intiated 4/4	

175 230

PLANT MODIFICATIONS

Task	Description	Priority	Status	Lead Man
65	Design waste gas system for pump down of RB to fuel pool		Intiated 4/4	
63	Supports for H.S. system in Turbine bldg. when filled (related to #52)			
66	Location for secondary plant diesel		Assigned 4/4	
70	Max P&T for DHR downstream of valve DH-V3		Assigned 4/5	
73	Back-up Power Source for secondary plant loads		Assigned 4/5	
74	Review fire protection for charcoal filter		Complete	
	Design/ Fab/Install shield plugs at DH vaults	M		

175 232

B & W

Task	Description	Priority	Status	Lead Man
1	Analysis of gas conc. in Primary system	H		
2	Provide list of critical systems for present conditions	H		
3	Analyze In-core thermocouples during LOFON 4/6	H		
4	Provide minimum allowable RCS pressure for degassing	H		
5	Provide stress Analysis for generator (points BtoC)	M		
6	Determine minimum primary system pressure (point D, Base Plan)	M		
7	Provide noise anaylsis of pressure during degassing	H		
8	Document of sequence of Plant conditions in base plan	L		
9	Develop procedure to determine pressurizer level using Heise Gauge			Rogers
10	Develop procedure for cooldown using OTSG's on natural circulation			Rogers
11	Core Anaylsis Program A. Thermocouples from Incores B. Neutron signals from Incores c Noise Levels			Rogers

4/6/79
1300 Hrs.

ACTION ITEMS
TASK MANAGEMENT/SCHEDUEL MEETING

0900 4/6/79

		Action Party
1.	Establish Data Bank; identify cognizant person-inform R. Arnold	Wilson
2.	Provide list of activities in progress to F. Stern	Palmer Wilson
3.	Identify planning coordinator	Wilson Palmer Cobean
4.	Support for Base Plan, Rev. 1	
	- Determine minimum degassing pressure (Point A)	McMillan
	- Continue design/installation of static & active make-up/pressure.	Cobean
	- Design & install cooling system for "B" Stm. Gen.	Cobean
	- Stress analysis for Stm. Gen. (Points B to C)	McMillan
	- Determine minimum primary system pressure (Point D)	McMillan
	- Alert Noise Analysis Group; Determine reporting.	Wilson McMillan Herbein
5.	Assure that at least one containment spray pump remains available for at least the next week - 10 days.	F. Stern
6.	Need criteria for additional waste gas storage facility.	Palmer
7.	Need criteria for upgrading electrical supply system.	Cobean
8.	Locate additional air compressors.	Cobean
9.	Write administrative procedure.	Cobean
10.	Confirm status of upgrading current RHR System.	Westinghouse (Cobean)
11.	Consider how to develop required contingency plans; "flesh out" Degeneration List.	Stern Arnold
12.	Arrange for outside organization to do required Safety Analysis.	Stern Arnold
13.	Increase primary system boron concentration to 3,000 - 4,000 ppm.	Arnold Herbein

175 234

INCIDENT RESPONSE CENTER - THIS SCHEDULE WILL CONTINUE UNTIL FURTHER NOTICE

**Karen Jackson 8a.m. until 8 p.m.   **Donna Smith 8 p.m. until 8 .a.m.

*8 am until 4 pm	*4 p.m. until 12 Mid	*12 Mid until 8 a.m.
Claudine Deliso	Kevin Bohrer	Sue Lynn
Marie Jambor	Nancy Hobbes	Jean Cook

* This schedule will begin Monday at 8:00 a.m. April 2

** This schedule will begin Sunday at 8:00 a.m. April 1

Have a happy day!

Contacts: Karen & Donna

3/31/79

CRESS WORK SCHEDULE

Saturday, 3/31/79 - midnight to 9am

midnight - indef.	Beth Williams
10:30pm - indef.	Andrea Perkins
5:00am - 9am	Alice Werner (call her @ 4:20am to wake her up)

Sunday, 4/1/79

8am - Noon	Laverne Johnson
10am - 4pm	Eileen Chun
noon - 6pm	Jean Schmidt
4pm - midnight	Beth Williams -- will find someone to replace Beth
6pm - midnight	Joanne Johansen

Sunday, 4/1/79 - midnight to 8am

midnight - 6:30am Jeannette Kiminas

need to find one more person

Monday, 4/2/79

7:15am - 4:00pm Laverne Johnson
7:15am - 4:00pm Irene Suissa
bring CRESS work with you

4:00pm - midnight Jean Schmidt
4:00pm - midnight Andrea Perkins
bring CRESS work with you

midnight - 8am ? need to schedule

If you have time, you might want to call 492-8585 for a tape on the Chairman's latest information.

If you need sleep, you can use the cots out in the hall.

Have a pleasant evening,
Joanne

175 236

IVORY PURPOSE

NRC ESTIMATED COSTS (As of 5pm 4/4/79)

RI - 46 K (Travel 18 K; Supplies, comm, etc. 28K)
RII - 20 K
RIII - 16 K
RIV - 10 K
RV - 9 K
HQS - 25 K (Travel)
HQS(ADM) - 80 K (30K Publication + graphics)
16 K Transpt HVAC
20 K Telecommunications
24 K Miscell, Videk, cables, ceres, etc)

Total 206 K*

*This does not include salaries of individuals involved in work on Ivory Purpose. At least 300 NRC people working on this problem.

B.H. Weiss

Dudley,
Gave FDAA copy at
12:00 pm - 4/5/79

175 2317

NOTE FOR DUDLEY THOMPSON

SUBJECT: COMPENSATION

I spent some time researching NRC MC 4136 with a view to calling out possible avenues for compensation for those employees affected by TMI. Listed below are a few observations which you may want to pursue with OA.

1) Nonexempt employees are entitled to overtime pay. Under certain circumstances, they may request [illegible] comp. time.

2) No employee earning in excess of $47,500 gets any additional compensation of any sort.

3) No employee can be granted comp. time in excess of the amount which would increase his compensation in excess of the $47500 rate for any particular pay period (were the comp. time to be paid at the overtime rate.)

4) An employee, even an exempt employee, earning less than $22788 per annum must be paid overtime unless he specifically askes for comp. time.

5) An employee earning more than $22788 can only get comp time.

6) Any employee earning less than $47500 can get night premium pay (10% of base pay) if the night work is scheduled.

7) Any employee earning less than $47500 can get 25% Sunday pay for

scheduled Sunday work. They can get night (10%) plus Sunday (25%) for Sunday night work, a total of 35% premium pay. When such work is ~~premium~~- also overtime, they can also get comp time.

All of this is rather intricate, and there are questions as to whether ~~or not~~ our scheduled night work is scheduled, or whether any of this can be retroactive. It may take some extraordinary effort on the part of OA to pull it off. On the other hand, the affected employees have put forth extraordinary effort throughout this affair without particular regard to remuneration. I think they deserve some answers and a good faith effort to compensate them to the extent the law allows.

In the same vein, I believe that employees forced to the use of their automobiles by the weird h urs of this operation may be entitled to mileage. They certainly are in call back situations.

It would be most useful and stimulating if OA could get out a fact sheet regarding the above ASAP, particularly in view of rumblings about NTEU taking an interest in this matter.

Bud Ward

175 23[illegible]9

April 2, 1979

Received a call from Lt. Tom Nelson, intelligence officer,USN, Light Photography Squadron 306 (Tel. 433-2881). He said his unit has helped in photgraphing forest fires and other emergency situations. If we need help, we can contact the squadron commander:

Lt. Commander Osbourne
433-2881

L I Caltr

175 ~~239~~ 240

April 2, 1979

IMPACT OF TEAMSTER STRIKE/LOCKOUT

Per Greg Benoit, who contacted these nonunion companies directly, these companies are available to haul as needed in Harrisburg area.

Daily Express - only intra-state ---the biggest
Harrisburg
Joe Spandler   bus. 717-939-9861
home 717-564-3136

Keene Transport
Interstate only
John Jennings   717-243-6622

Thurston Transport
Interstate, but could get special permit promptly from State
Jim Hanks   717-238-0431

Wards
Interstate
Davle Meyers   717-761-1334

L D Cobb

175 240

175 ~~239~~

OCE - H'burg Airport,
ED Patterson - Incharge

Total 30 - (20* more due in from ORNL & ANL)
*mostly HP's

2 Helo pilots
4-5 chemists
1 meteorologist (Sweeney)
7-8 H.Ps
2-3 Instrument techs
Rest are Admin or General techs (super.)

---

5 telephone lines
Comms Pod (Nest 3) package
Counting lab set up, gas waste etc.
Iodine Reagents + Chem lab
Portable residence for 8-10 people
EG+G Van - Counting
EG+G Van - Computer
Helicopter instrument pods

ON THE WAY
3 HP's from DOD/DCPA

175 242

Dudley Thompson or Sam Bryan

Attached for your information is a schedule of the Inspector coverage that will be provided at Rancho Seco up through 4/12/79.

MANNING SCHEDULE FOR
RANCHO SECO

April 2 - 4	Lewis Miller
April 4 - 6	John Carlson
April 6 - 8	Harvey Canter
April 8 - 10	Phil Johnson
April 10 - 12	Al Johnson

NUCLEAR REGU...

UTILITY STAFF HAS
BEEN ROTATING
THROUGHOUT THE DAY.
SOME DAY SHIFT PEOPLE
ARE STILL THERE,
BUT FATIGUE ISN'T
A PROBLEM.

175 245

4/5/79

John,

Boyce called back about his HP needs at the site for about the next two weeks, at least.

On-Site HP needs - 23		
	2 HP's per shift per unit	12
	1 per shift for procedure review	3
	1 supervisor	1
	H-P Support	
	1 Sample control	1
	1 per shift - mobile lab	3
	1 Instrument technician	1
	1 Emergency plans	1
	1 Coordinator w/other agencies	1
		23

This assumes no resources for environmental work. If continue environmental work, will need 4 HP's per shift plus a supervisor. Total of 13 additional.

RI has a total of 20 HP's.

175 246

Boyce indicates that he needs a clear understanding of his responsibilities at the site in this transition and afterwards.

Boyce also wanted to point out that if the licensee gets in better shape and he thinks that is begining to happen, he may be able to go down to 1 HP per unit per ~~site~~ Shift (6 HP's) which would reduce need to about 17. However, do..'t think he sees this in the next few days.

Boyce had not had any discussions with EPA or DOE regarding our continuing role in the environment.

RHW

For: Edson Case

Description of Proposed Organization
for NRC Operation at TMI-2

175 248

For: Edson Case

Description of Proposed Organization for NRC Operation at TMI-2

There will be three principal organizations: NRR Operations, NRR Technical Review, and IE, as described in the attached chart.

The NRR operations function interacts with the NRR people in the TMI-2 control room area and with GPU's Task and Schedule Managerial Team (F. Stern, Chairman). The NRR operations function will be managed by Stello and Vollmer. The GPU managerial team will be prioritizing work and NRR will provide liaison.

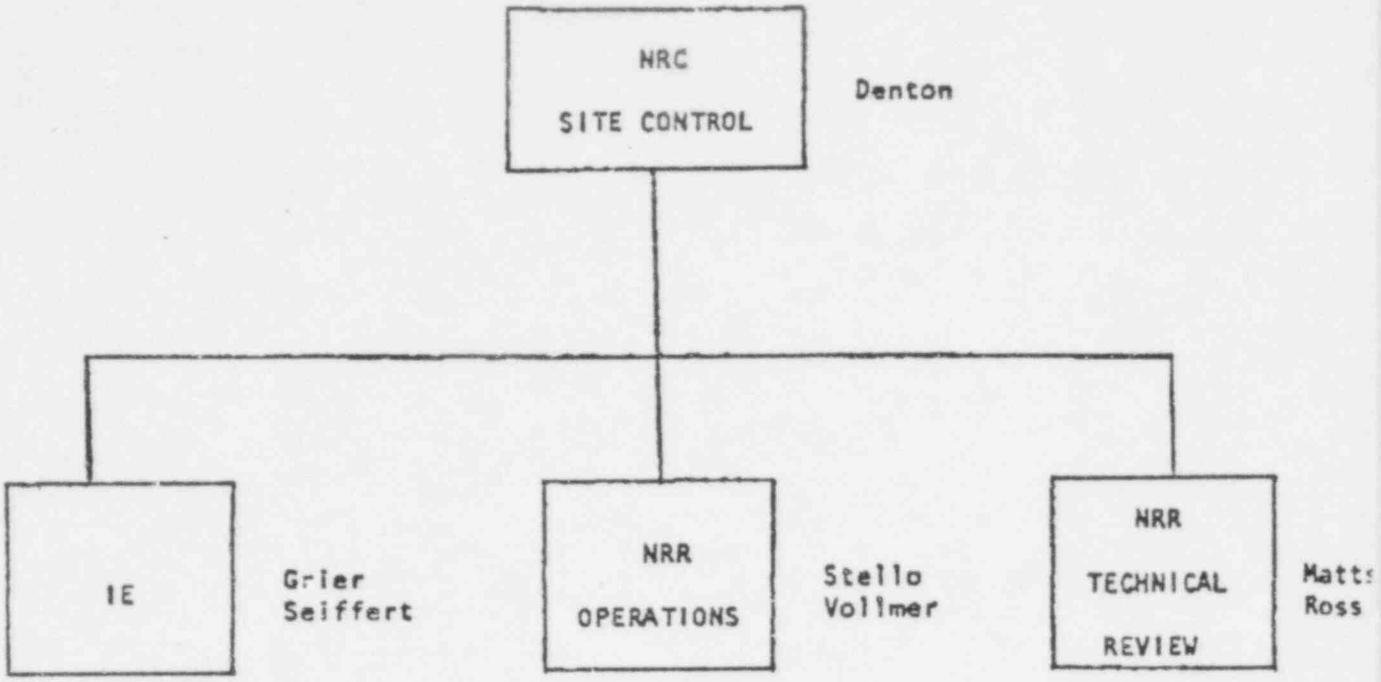
The NRR technical review function will have responsibility for reviewing plant modifications. In the GPU organization the modifications are the responsibility of the Plant Modification team headed by G. Cobean of Burns and Roe. Its scope includes the new RHR, the electrical power modifications, the primary system instrumentation alternatives, the radwaste modifications, and the modified secondary cooling system. NRR will review the adequacy and safety of these changes. The technical review organization will be composed of the required engineering disciplines to accomplish the reviews. Names are listed; where an asterisk (*) is provided, the branch chief is to provide a qualified reviewer. The technical review team will be managed by Mattson and Ross.

175 2419

In addition to these principal divisions of NRR labor, we have an interface with and membership on the GPU Technical Working Group (McMillan of B&W is Chairman). D. Ross is currently assigned as NRC liaison. One task already assigned to NRC by this group is to prioritize the input to and output from the government laboratories and consultants. This task will be the responsibility of L. Ybarrando of INEL who will report to Ross.

The question of core coolability in various cooling modes has been assigned by GPU to the Industry Advisory Group under the leadership of M. Levenson of EPRI. The liaison with this group will be handled by Ross and Mattson, aided by Nick Kaufman of INEL.

The IE site organization is also in an attached table.


NRC
SITE CONTROL
Denton
IE
Grier
Seiffert
NRR
OPERATIONS
Stello
Vollmer
NRR
TECHNICAL
REVIEW
Matt
Ross

175 250

NRC THREE MILE ISLAND ORGANIZATION

NRC Command

Denton (2) — One Shift
Secretary — 8 A.M. - 8 P.M.
(Trailer 1)

NRR Operations (16)

Stello 6 A.M. - 6 P.M.		Vollmer 12 P.M. - 12 A.M.
- Communications and Data (Trailer 1)	(2)	2 shifts of 1 person ea. (8 - 8)
- Plant Procedures (Trailer 2 & Unit #2 Turbine Bldg.)		2 shifts of 4 people ea. (8 - 8)
Systems	(6)	
Radiological	(2)	
- Effluent Control & Health Physics	(4)	1 shift from 8 A.M. to 8 P.M.
- Operator Licensing	(2)	2 shifts of 1 person ea.

NRR Technical Review (8)

Mattson/Ross 8 A.M. - 8 P.M.

- Reactor Systems	(1)	Novak
- Electrical	(1)	Tondi
- Mechanical	(1)	Bosnak*
- Aux. Systems	(1)	Benaroya
- Structural	(1)	Schauer*
- QA	(1)	Haass*

* This branch chief to provide one reviewer.

IE OPERATIONS (69)

Grier 8 A.M. - 8 P.M.	Seyfrit 8 P.M. - 8 A.M.
- Communications (5) Reactor Operations (Trailer)	2 shifts of 1 person each 3 shifts of 1 person each
- Operational Surv (12) Procedure Review Communications (Unit 2 CR)	3 shifts of 4 persons each
- In Plant Health Physics (18) Effluent Control Procedure Review Communications (Unit 1 CR)	3 shifts of 6 persons each
- Environmental Analysis (18) Offsite Env Surveys (Mobile Lab, Trailer Instrument Van)	3 shifts of 6 persons each
- HP Support (7) Sample Control Emergency Planning	8 A.M. - 8 P.M.
- Admin Support (7)	3 Shifts of 2 persons

EMT/XOOS - OPERATIONS STATUS OFFICER

	Tu 4/10	W 4/11	Th 4/12	Fr 4/13	Sa 4/14	Su 4/15
6am 2pm	Paulus* / Hegner	Paulus / Hegner	Paulus / Hegner	Paulus / Hegner	Paulus	Gower
2pm 10pm	Weiss	Weiss	Weiss	Weiss	Ward	Ward

	Mo 4/16	Tu 4/17	Wd 4/18	Th 4/20	Fr 4/21	Sa 4/22
6am 2pm	Gower	Gower				
2pm 10pm	Hegner	Hegner				

*Paulus - 6am to 8am; Hegner 8am to 2pm

Duties of Operations Status Officer:

1. AM shift pulls PN together, obtains reviews, appropriate concurrences and sees that it is dispatched promptly.
2. Provides TMI-2 status information as requested from legitimate outside inquiries, other NRC offices and foreign sources if arranged thru IP. All press inquiries for status information are to be referred to PA.
3. Maintains continuity of taping; assures tapes are changed and that used tapes are properly stored.
4. Will take action, if necessary, to recall EMT & IRACT back into full operation.
5. Expedite completion of high priority items.
6. Coordinate requests for support from other Federal agencies

Operations Status Officer calls are to be handled by IRACT from 10pm to 6am.

GAGLIARDO
HUNTER
SINKULE
BLACKWOOD
FASANO
MURRAY
STOHR
McCABE
BARBER

KISTER
LEDOUX

2031 Sand [illegible] [illegible]

Industrial waste –

Sewage –

Thunderstorm
& winds up to
80 mph

2061 — Rev. 14

W 90°
E 270°

TMI fuel @

what is average burnup in MWD/assembly
three mile Island.

$\boxed{33}$ Lauben

1300

$$\frac{90 \times 2785\ mw}{177} = \cancel{1416}\ \boxed{1410}\ mD/Ass.$$

Roy Woods

flow Charts and talked to operators

looks like about one minute - ~~on~~ temp.
~~temp~~ no

1:18 — 2m 13:

22- 45 loop A. 285-287
for 20 minutes to
half hour.

Tw 7% ~~13:15~~ 13:13- 13:15

175 256

	9-M	5-G	8-H
1519	343	373	464
1520	343		
21	343		
22	343		
23	343		
24	344		
25	344		
26	343		
27	343		
28	343		
	343		
	343		
	341		
	342		
	342		
	342		
	343	375	461

5-D

~~15~~ 295

//

295

175 2517

1A Coolant pump -

1447 Readings

9-M 5-6 8-9

Aux building Sump -

A. Haliburton

Halliburton

2029

2030 - lost intermediate cooling water

pump 2B shorting contacts alarm contacts.

Bill Zewick

Message from Bob Warnick @—
Bob Martin: Incore
~~No~~ set of readings at 98%
~~Incore~~
Believe that one is available for 85% power
and will get copy to Martin

Gen 1 ~~flow~~ close to secure, at 30psi
Component cooling flow lo
plus Lo flow to seals } Trip RCP

* B&W requesting to go to 4000 ppm
* Bleed and Feed to achieve from 2000 ppm:

B&W gave permission to come
down by 50 psi — 1000
to 950 ± 50

175 2589

322

Plant organization status

In Relation to DHL

inside containment valves – are they open or closed?

if open – is pressure interlock bypassed? if so how?

is there power to valves

What is status of pressure interlock

Discharge DHV4 A, B

V1 + V2 interlock defeated by jumper on P-S.

Suction DH V171
DH V1
DH V2
DH V3

bkrs shut

29th @ 1640 Still shut, Tagged, Pwr. avail

493 ~ 544 comp. Pts.

175 260

Preparing to start. Z-57 800

Kemick Rogers

Z-57 to go degas through pressure

Roger Madsen

9440301

feed Boric acid mix tank have to move Concrete door to Aux. building

Ger Gauge reads 800
90
1800 1743 →

Bob Martin: Procedure - RC System pressure calibration procedure
717 - 944 - 4144

Three Put on Recombiner.

DATA Accumulation

1. Put Special TC data on separate sheet and tabulate on separate Summary Sheet.

2. Retain previous core mapping on about a 2hr schedule - Probably should recopy the Summary Sheet starting 0000 4-8-79 to reduce the number of entries.

Special TC data is being taken at each 50psi step level.

PN: Start @ 10 — Ready @ 8:00 am

Unusual noise heard on press. level

cylinder #2

RC sample is OFF again.

#/ft³ ft/#

at 212

water

175 265

Milton Schater - Dept of Agriculture

Sheet or tube of Palladium

175 266

NUCLEAR REGULATORY COMMISSION

OFFICE OF INSPECTION AND ENFORCEMENT

WASHINGTON, D.C.

INCIDENT MESSAGE FORM

TO: H R DENTON TMI SITE

FROM: D THOMPSON

PER REQUEST FROM

F. INGRAM

CC: J FOUCHARD

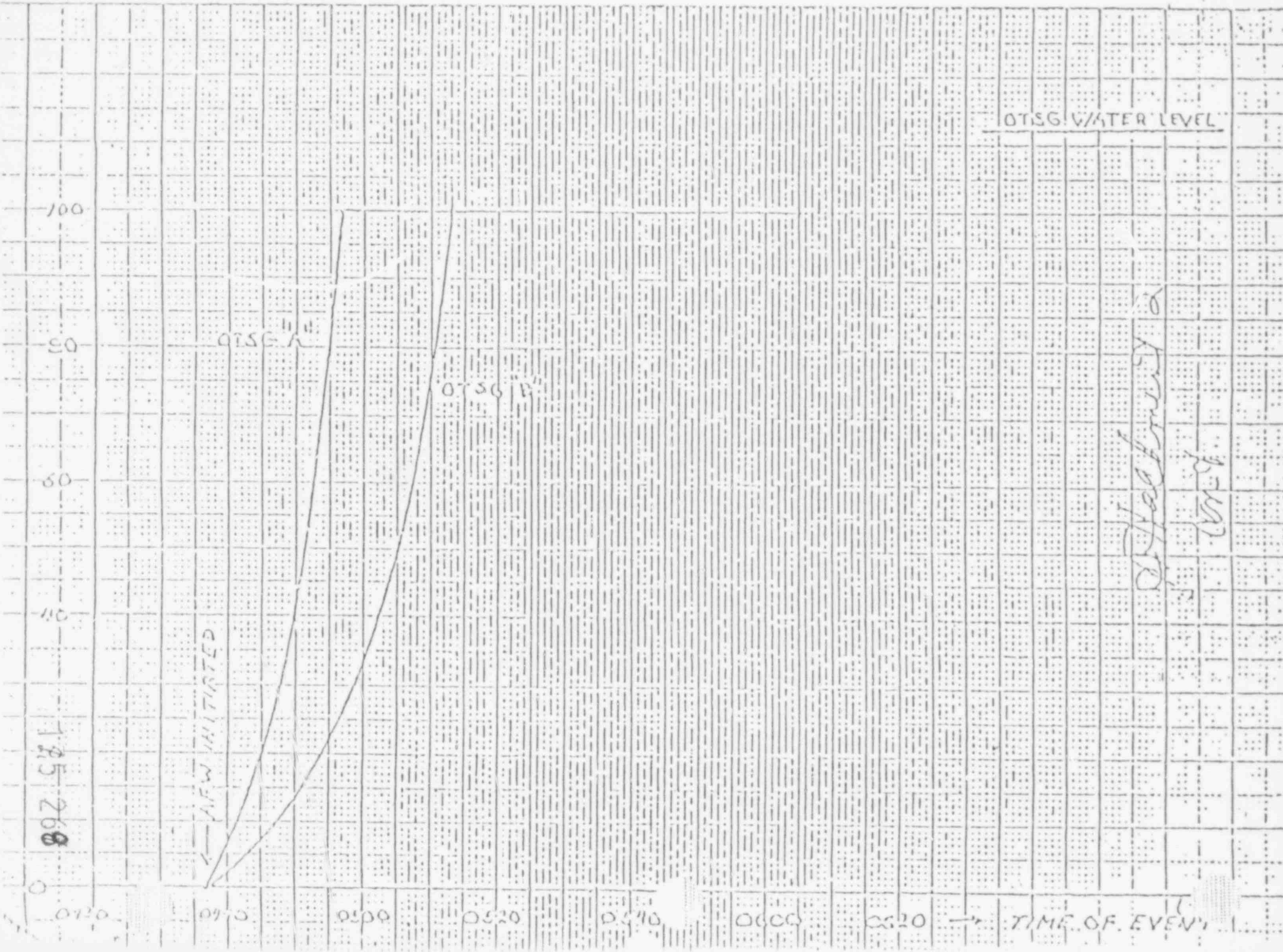
DATE: 4/3 TIME: 18:57

RAPID FAX SENT

8:20

175 2617

46 1470

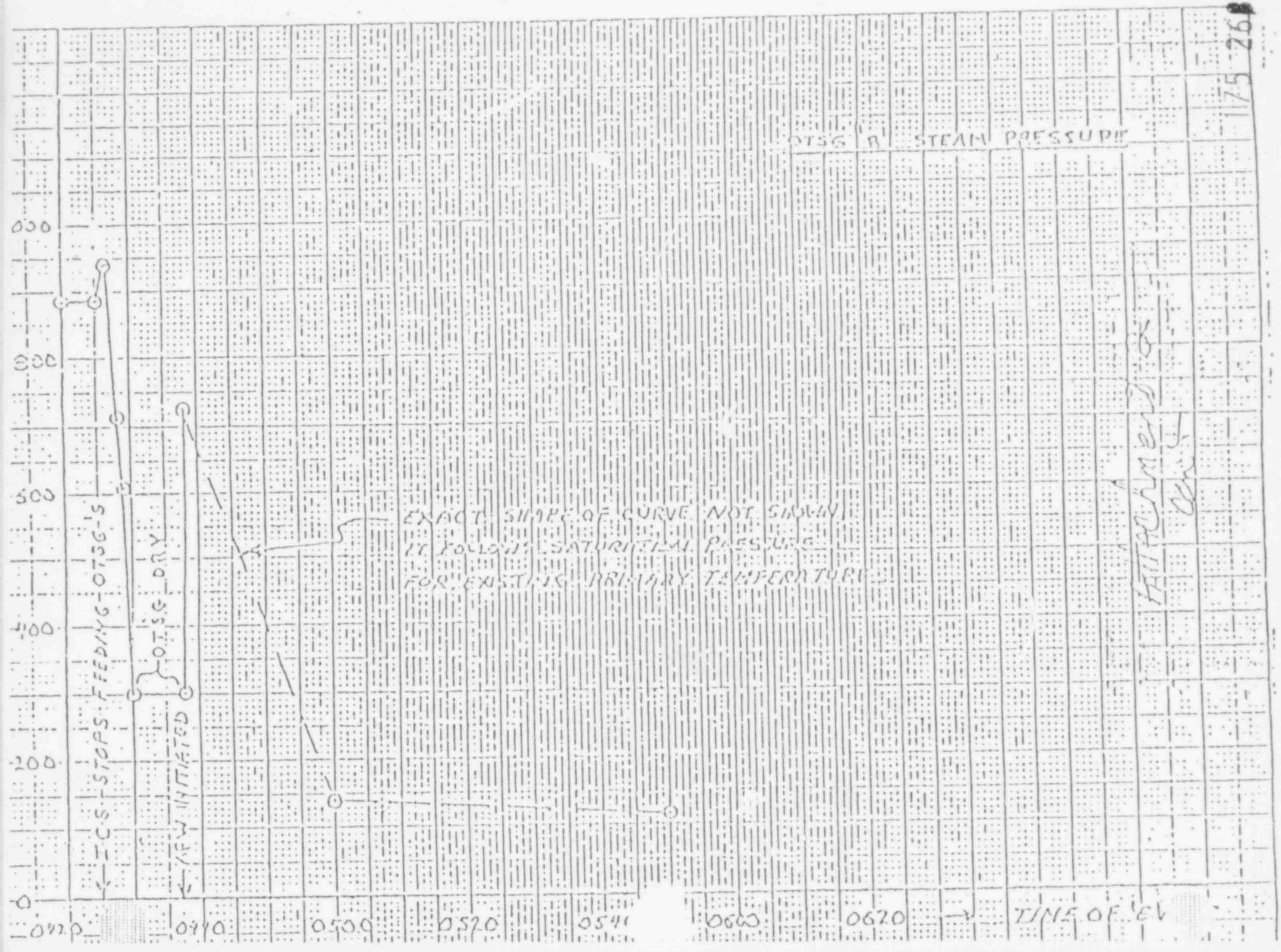


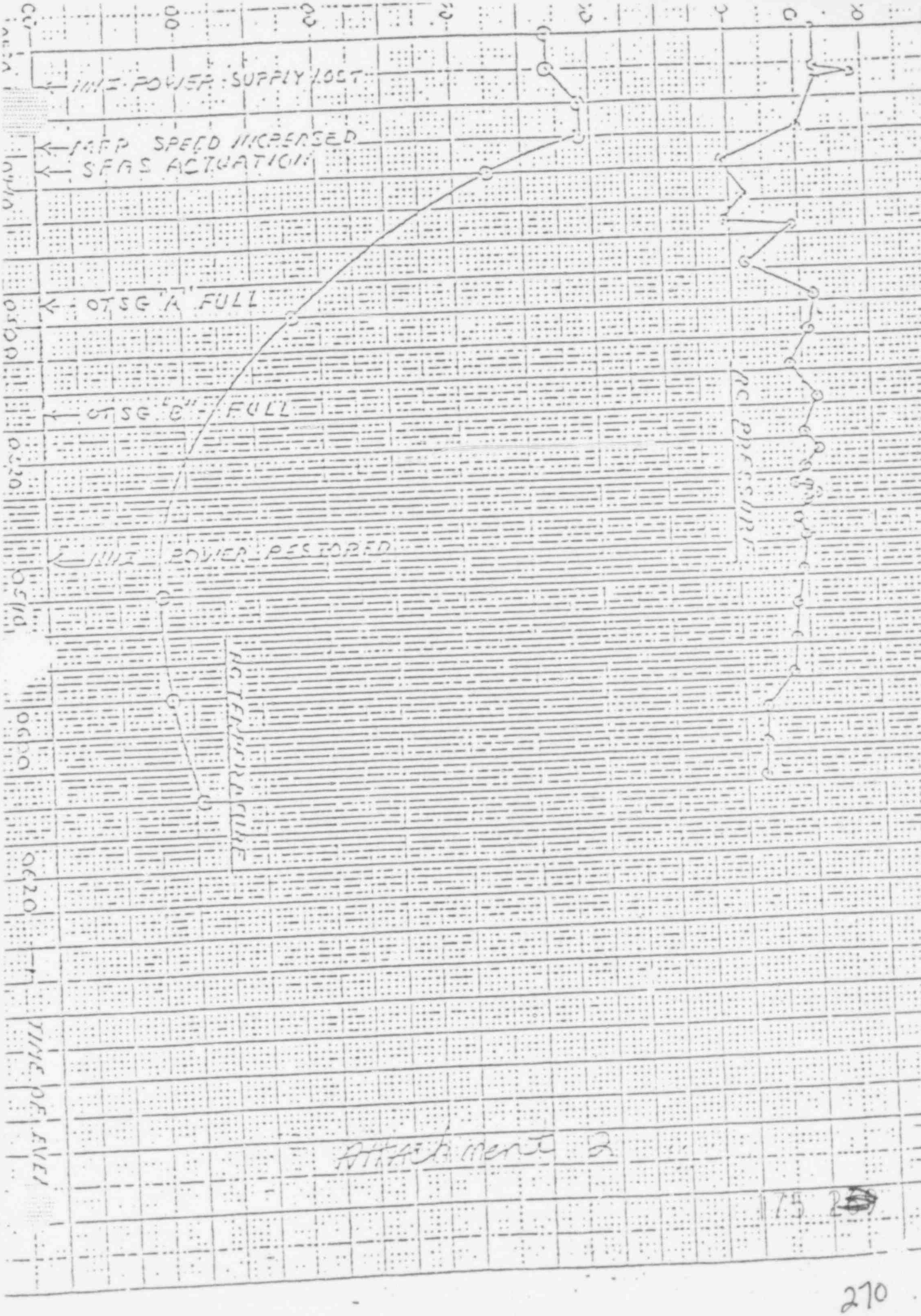
125 268

K+E 10 X 10 TO ½ INCH • 7½ X 10 INCHES KEUFFEL & ESSER CO.

46 1470

175 2619





INJ POWER SUPPLY LOST
MFP SPEED INCREASED
SFAS ACTUATION
OTSG 'A' FULL
OTSG 'B' FULL
INJ POWER RESTORED
RC PRESSURE
RC TEMPERATURE
0500
0520
0540
0600
0620
TIME OF EVENT
Attachment 2

46 1470

270

13:47 - OTSG "B" level - 599.1"

34 - Power restored to NNI cabinets 5,6,&7

$T_{ave} = 285^{\circ}F$

RCS Pressure ≈2000 psig

Both OTSG full level ranges pegged high

Operator begins to reduce RC pressure using pressurizer spray.

ICS closes turbine bypass valves to condenser.

Operator stops emergency FW flow.

Operator stops main FW pumps.

175 270

- Operator increases speed of a MFP and feeds "A" OTSG. This starts RCS on pressure and temperature decrease.

- RC pressure ≈1900 psi

:16 - SFAS actuation at 1600 psig

This starts HPI, LPI and initiates emergency feed. The emergency FW pump is started and the bypass emergency FW valves are opened to full open position. The system makes no automatic attempt to control steam generator water level.

- RC pressure at 1475 psig. It starts to recover from this point due to HPI. T_{ave} = 528°F.

3:56 - "A" HPI pump secured.

5:09 - LPI secured.

4 - "A" HPI initiated. From this point on, the operator started and stopped HPI pumps as necessary to maintain pressurizer level.

0 - Steam Line Failure Logic closes ICS-controlled start-up feed valves to each OTSG when the corresponding OTSG pressure falls below 435 psig.

1:25 - Secured RCP-D (T_{ave} ≈435°F)

This reduced #RCP's to three

7:27 - OTSG "A" water level - 599.7"

Speculate that ≈2 ft. of tubes are not flooded (at top) due to steam line arrangement.

0:00 - Hourly computer log print-out
Steam temp. 380°F (OTSG "B")
Steam pressure 171 psig (OTSG "B")

175 272

Assuming $T_{ave} = T_{sat} \Rightarrow T_{ave} = 380°F$

SEQUENCE OF EVENTS - SMUD 04:25 to 05:34 - MARCH 20, 1978

(Revision 1, 5/25/7?)

EVENT

:35 - Lost NNI power supply cabinets 5,6, & 7

- This caused a loss of valid signals to the ICS. BTU limits ran back feedwater, resulting in a partial loss of feedwater (actual Rx power was 72%).

- Probable opening of "B" turbine bypass valves to the condenser (timing uncertain).

:44 - Reactor trip on high pressure, turbine trip on interlock.

- Pressurizer code relief setting was known to be low (approximately 2225 psig). The electromatic relief was isolated due to previous leakage problems. The data indicates primary pressure went ≈2400 psig => code relief valve lifted.

- ICS closes main control and start-up feed valves and drive main feed pumps to minimum speed following trip.

- Decay heat and RC pumps energy removal accomplished through generators by inventory boil off and the addition of main feedwater.

6:25 - Pressurizer code relief valve reseats at approximately 2100 psig.

- Operator starts HPI pump "B".

8:23 - Operator stops HPI pump "B".

0. - OTSG "B" pressure reaches 435 psig set-point of Steam Line Failure Logic.

- OTSG "B" goes dry.

175 273

2. Given that the operator can determine that electrical power has been lost to all or part of the NNI, he should know the location of the power supply breakers, and have a procedure available to quickly regain power.

3. If the fault cannot be cleared (i.e. the breakers to the power supplies reopen), the operator should have a list of alternate instrumentation available to him, and he should be thoroughly trained in its use. Examples are:

   a. ESFAS panels
   b. RPS panels
   c. ECI (Essential Controls and Instrumentation)
   d. SRCI (Safety Related Controls and Instrumentation)
   e. Remote shutdown panels
   f. Local gages
   g. Plant computer

4. Recognizing that no procedure can cover all possible combinations of NNI failures, the operator's response should be keyed to certain variables. If the operator realizes that he has an instrumentation problem (as opposed to a LOCA or steam line break, for example), he can limit the transient by controlling a few critical variables:

   a. Pressurizer level (via HPI or normal Makeup Pumps)
   b. RCS pressure (via Pressurizer heaters, spray, E/M relief valves, etc.)
   c. Steam Generator level (via feed flow, feedwater valves, etc.)
   d. Steam Generator pressure (via turbine bypass system)

   The pressurizer level and RCS pressure assure that the Reactor Coolant System is filled; the Steam Generator level and pressure assure adequate decay heat removal.

Attachments 1 and 2 are provided to give a brief description of the events following this loss of NNI power at Rancho Seco. As can be seen by this transient, prompt precise operator action and the ability to recognize a loss of NNI power are critical factors in limiting the severity of a transient such as this.

If you have any questions or comments, please advise.

Yours truly,

Ivan D. Green

Ivan D. Green
Site Operations Manager

IDG:TPS:mlf
encl.
cc: See attached sheet.

175 274

1130 4/2 call from site

Wyn—

IWTS

discharging at 100-140 gpm into
normal plant discharge of 55,000 gpm

Friday a.m., unit 2 TB sump was
pumped to IWTS.

Reviewing the concentrations of Iodine
in IWTS, peak I-131 concentrations
in release after dilution could be
in neighborhood of 12 x MPC (3×10^{-7})
for < 1 day.

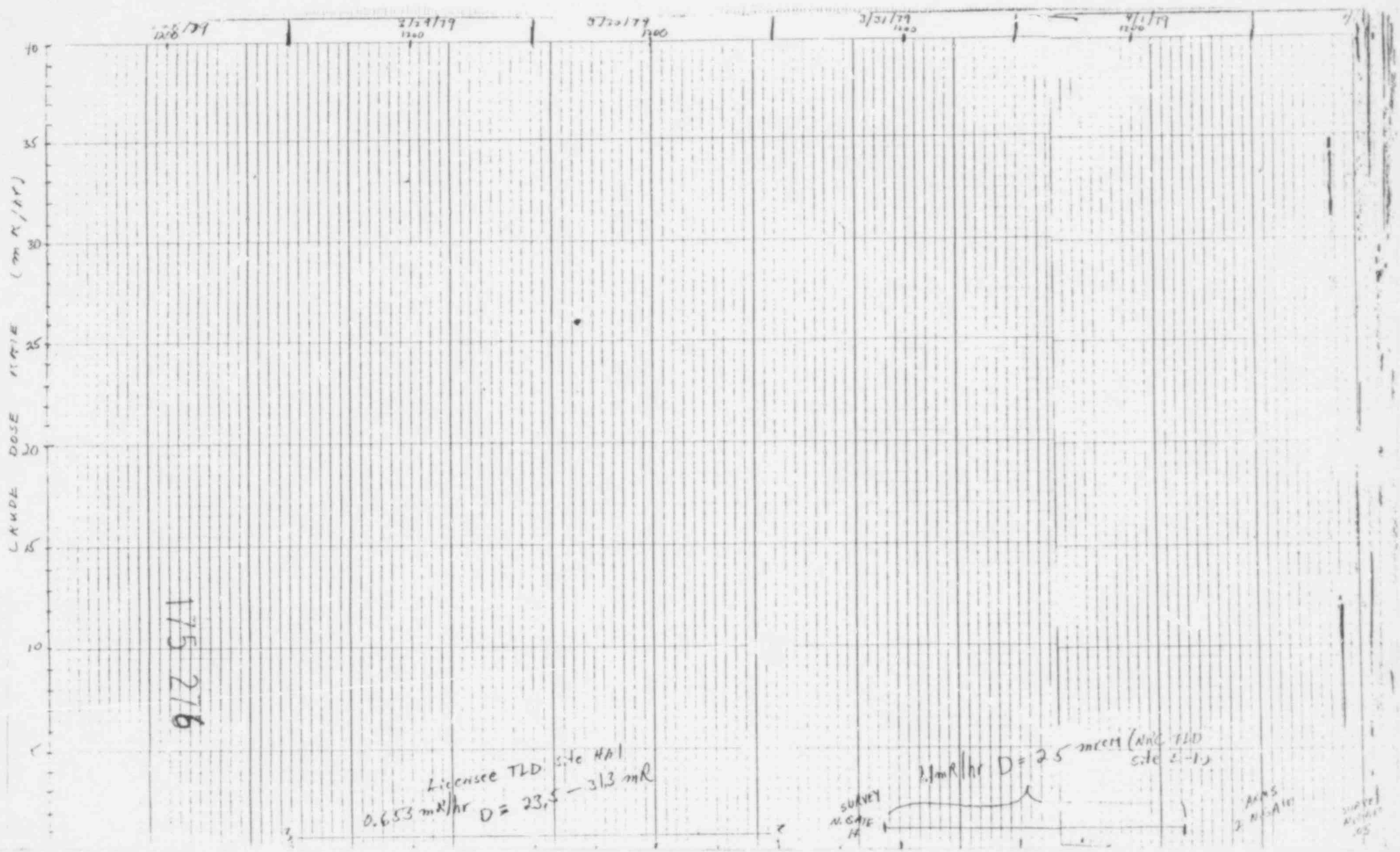
Concentrations in discharge have
been slowly decreasing, and are
now at about 4.5 x MPC.

~~State has~~
Discussing with licensee.
State has been informed.
Moseley informed and EMT informed.

175 275

rec 1:20 pm
Albert the Amato 1:25pm

557-7390


CRUDE DOSE RATE (m R/hr)
40
35
30
25
20
15
10
5
12/28/79
3/29/79 1200
3/30/79 1200
3/31/79 1200
4/1/79 1200
Licensee TLD site HA1
0.653 mR/hr D = 23.5 – 31.3 mR
1.1 mR/hr D = 25 mrem (NRC TLD site 2-1)
SURVEY N. GATE
ARMS N. GATE
.05

175 276

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C 20555

CLASS OF SERVICE:

X URGENT

_____ IMMEDIATE

_____ ASAP

FACSIMILE SERVICE REQUEST

DATE: 4/4

MESSAGE TO: Eysymontt (Name) Commissioner Assistant (Office/Division - Agency/Company)

STATE AND CITY: Washington DC

TELECOPY NUMBER: AUTO: YES NO

✓ (Rapifax - 50 secs./page)

(3-M - 4 Mins ____/6Mins ____)

(Xerox - 4 Mins ____/6Mins ____)

OTHER: (Transmission Mode: ____ Mins.)

VERIFICATION NUMBER: _____

NO. OF PAGES 2 EXCLUDING COVER SHEET RETURN COPIES - YES NO

MESSAGE FROM: John Davis (Name) IE (Office/Division - Agency/Company)

BUILDING: EAST WEST TOWERS OFFICE PHONE _____

TELECOPY NUMBER: 301 492-8187 - RAPIFAX - AUTOMATIC
301 492-7285 - 3M VRC AUTOMATIC
301 492-7264 - XEROX (4001) - MANUAL

VERIFICATION NUMBER: 301 492-7928 - (Bethesda, MD)

Received/Time -Date	Transmitted/Time Date
4/4 - 8:50 AM	10:00am - COMPLETED

175 2707

To Eysymontt
From Davis
cc Commissioners Assistants

Status of B&W Reactors - 4/2/79

(This info was checked with Roger Boyd NRR during PM of 4/3)

Operating

	UNIT	STATUS	STATE	ARCHITECT/ENGINEER
1.	Arkansas Unit 1	Shut down 3/30/79 for reload	Arkansas	Bechtel
2.	Crystal River 3	Operating	Florida	Gilbert
3.	Davis Besse 1	Shut down for repair to inoperative relief valve	Ohio	Bechtel
4.	Oconee 1, 2, 3	Operating	South Carolina	Bechtel
5	Rancho Seco	Operating	California	Bechtel
6.	Three Mile Island 1	Shut down because of Unit 2	Pennsylvania	Gilbert

Under Construction - Status as of April 3 (Blue Book/DPM list/NRC Caseload Forecast Panel)

Note: This information was obtained from Roger Boyd on PM of 4/3/79 from Caseload Forecast Panel files.

	CP	FUEL LOAD DATE Licensee	NRC Caseload Forecast Panel	STATE	ARCHITECT/ENGINEER
1.	Midland 1	11/80	11/81	Michigan	Bechtel
2.	Midland 2	11/81	11/82	Michigan	Bechtel
3.	Bellefonte 1	3/82	3/82	Alabama	TVA
4.	Bellefonte 2	12/82	12/82	Alabama	TVA
5.	WPPSS 1	6/83	6/83	Washington	United Engineers & Constructors
6.	WPPSS 4	12/84	12/84	Washington	United Engineers & Constructors
7.	North Anna 3	11/81	6/86	Virginia	Stone & Webster
8.	North Anna 4	12/82	12/86	Virginia	Stone & Webster

278

175 ~~277~~

	No CP	FUEL LOAD DATE Licensee	NRC Caseload Forecast Panel	STATE	ARCHITECT/ENGINEER
9.	Davis Besse 2	Company reevaluating on basis of		Ohio	Bechtel
10.	Davis Besse 3	need for power		Ohio	Bechtel
11.	Erie 1	12/87	---	Ohio	Commonwealth Associates
12.	Erie 2	12/89	---	Ohio	Commonwealth Associates
13.	Pebble Springs 1	4/86	---	Oregon	Bechtel
14.	Pebble Springs 2	4/89	---	Oregon	Bechtel
15.	Greene County	7/86	---	New York	Stone & Webster
16.	Greenwood 2	7/88	---	Michigan	Bechtel
17.	Greenwood 3	7/88	---	Michigan	Bechtel
18.	Carolina 8	Postponed indefinitely		North Carolina	
19.	Carolina 9	Postponed indefinitely		North Carolina	
20.	Vandalia	Postponed indefinitely		Iowa	Bechtel

279

Facility: Three Mile Island Unit 2
Middletown, Pennsylvania (DN 50-32)

Subject: RELEASE OF LIQUID WASTE TO THE SUSQUEHANNA RIVER

The NRC was notified by Metropolitan Edison Company, the Licensee, at about 2:30 p.m., March 29, 1979, that liquid effluent containing a small amount of radioactive material (approximately 10^{-3} to 10-4 microcuries per milliliter) is being released to the Susquehanna River in a controlled fashion. The radioactivity in the liquid waste consists primarily of noble gases, Xenon-133 and Xenon-135. Considering dilution in the cooling water the release is within the NRC limits for discharge of normal effluents to the environs.

The licensee reports that this release is necessitated because of the large amount of water being handled by the waste treatment system.

There is significant media interest at the present time. The Commonwealth of Pennsylvania and EPA have been informed.

80
175 2~~79~~

3/28/79

Information on the Three Mile Island Nuclear Electrical Generating Plant indicent as related by Bob Bores, NRC, Region 1, in King of Prussia, PA. (Dictated by Mr. McKool)

At approximately 4:00 a.m. the steam turbine generator tripped causing a reactor scram. There was simultaneous indication of malfunction of a secondary feed water pump. The reactor operating crew initiated a rapid cool-down procedure. Shortly thereafter they noted a loss of pressure in the primary cooling system. They also noted an indication that the pressurizer bubble had collapsed and they speculated that the bubble had been entrained and pumped into the reactor pressure vessel. They likewise speculated that there may have resulted some coolant starving of the reactor fuel assemblies. Shortly afterward there was an indication of radioactivity in the containment system and a pressure increase in the containment system to approximately 1 PSI. A sample of the primary coolant water indicated 140 microcuries per cc Beta Gamma gross activity. A radiation monitor located in the dome of the containment system indicated 200 R per hour at 8:00 a.m. At 8:30 a.m. the monitor reading had increased to 600 R per hour. At 8:45, the monitor indicated a level of 1,000 R per hour and at 9:00 a.m., the monitor indicated a level of 6,000 R per hour. Mr. Bores cautioned against any great confidence in the levels of the monitor inasmuch as the functioning and the calibration of the monitor became questionable sometime during the course of the incident. Other monitors peripheral to the containment system indicated levels of 100 MR per hour up to as high as 10 R per hour. At 8:00 a.m., there was a two-mile per hour wind at 90° which shifted at 11:47 this morning to 150° and 6 miles per hour. Radiation levels monitored outside the plant were as follows:

175 280

3-4 MR per hour east of the plant and directly across the river at the south end of Three Mile Island.

One grab sample made with portable instrumentation indicated iodine 131 at an approximate level of 3×10^{-8} microcuries per cc. Due West of the plant another grab sample indicated iodine 131 at a level of approximately 1.1×10^{-8} microcuries per cc.

The above information was relayed to General Bratton, Director of EACT, who in turn briefed John Deutch on this matter at 1:15 p.m.

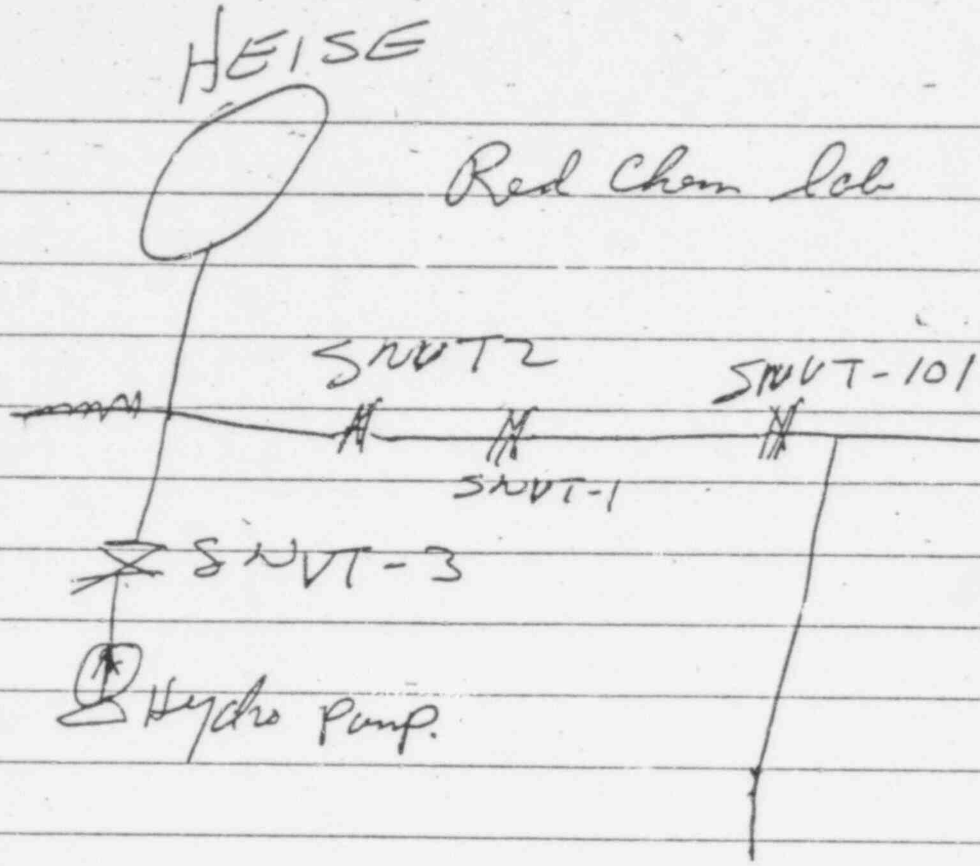
containment levels

probable 39.2" max 52.9"
min 35.8"

by GPU calc of inventory

4/5 1015

175 283



203-1 P&ID

SOP Z37 Rev.1

175 284

Venting not in Process. —

1450 Roger Zabadowski

when LD reestablished & flowrate

Bubble return

50 ft³ in coolant now

Filter inform - 1. how long operating

2. Conductivity

Smith Lozouth

1B - 5R/Hr

Mod. for gas venting tibs been completed and ~~pressurized~~ pressure tested - showed stat only in about an hour

Owen Shockellon

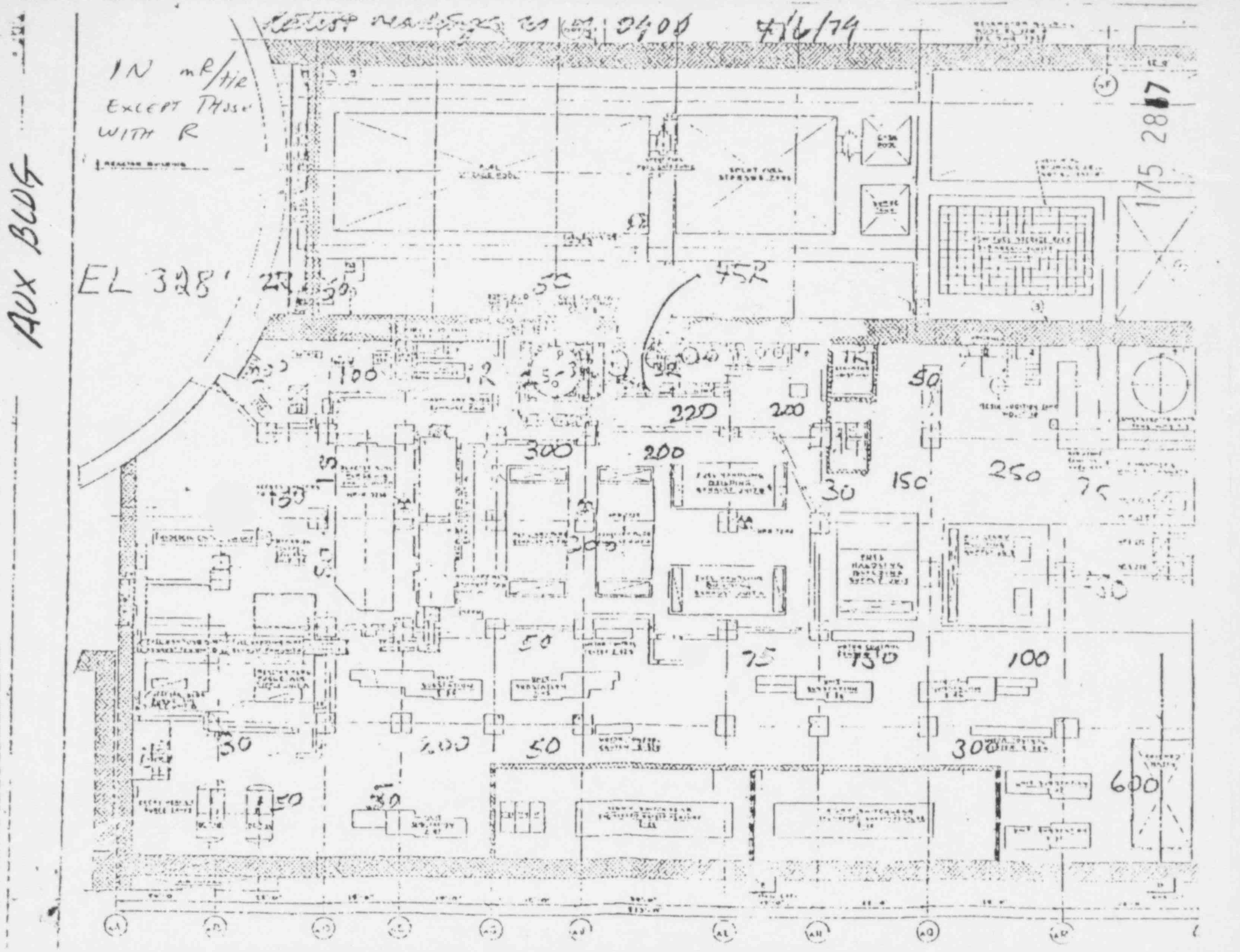
415 932 8301

175 285

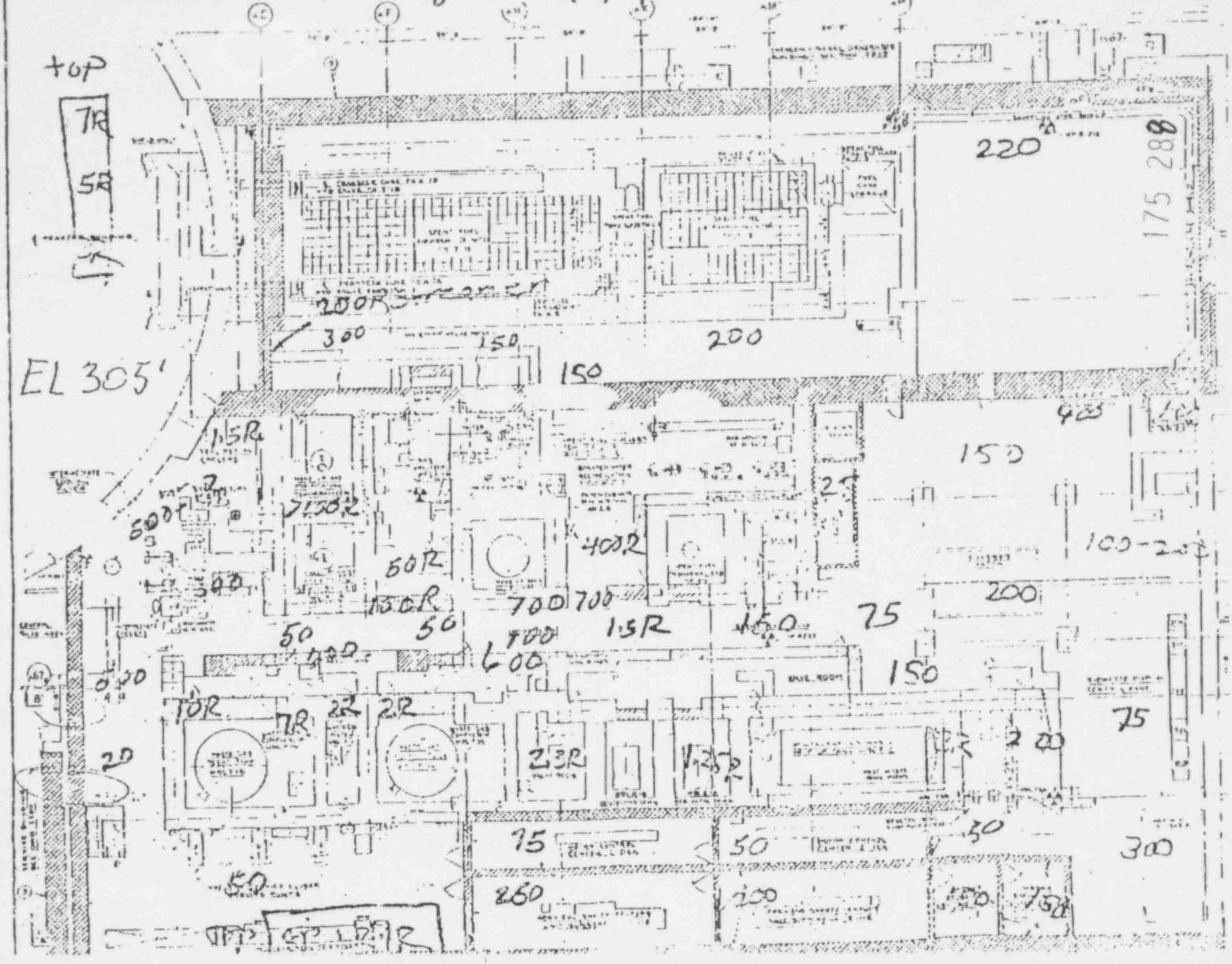
Telephone report from Montgomery at site reporting ARMS survey data, received 1120 hrs 4/5

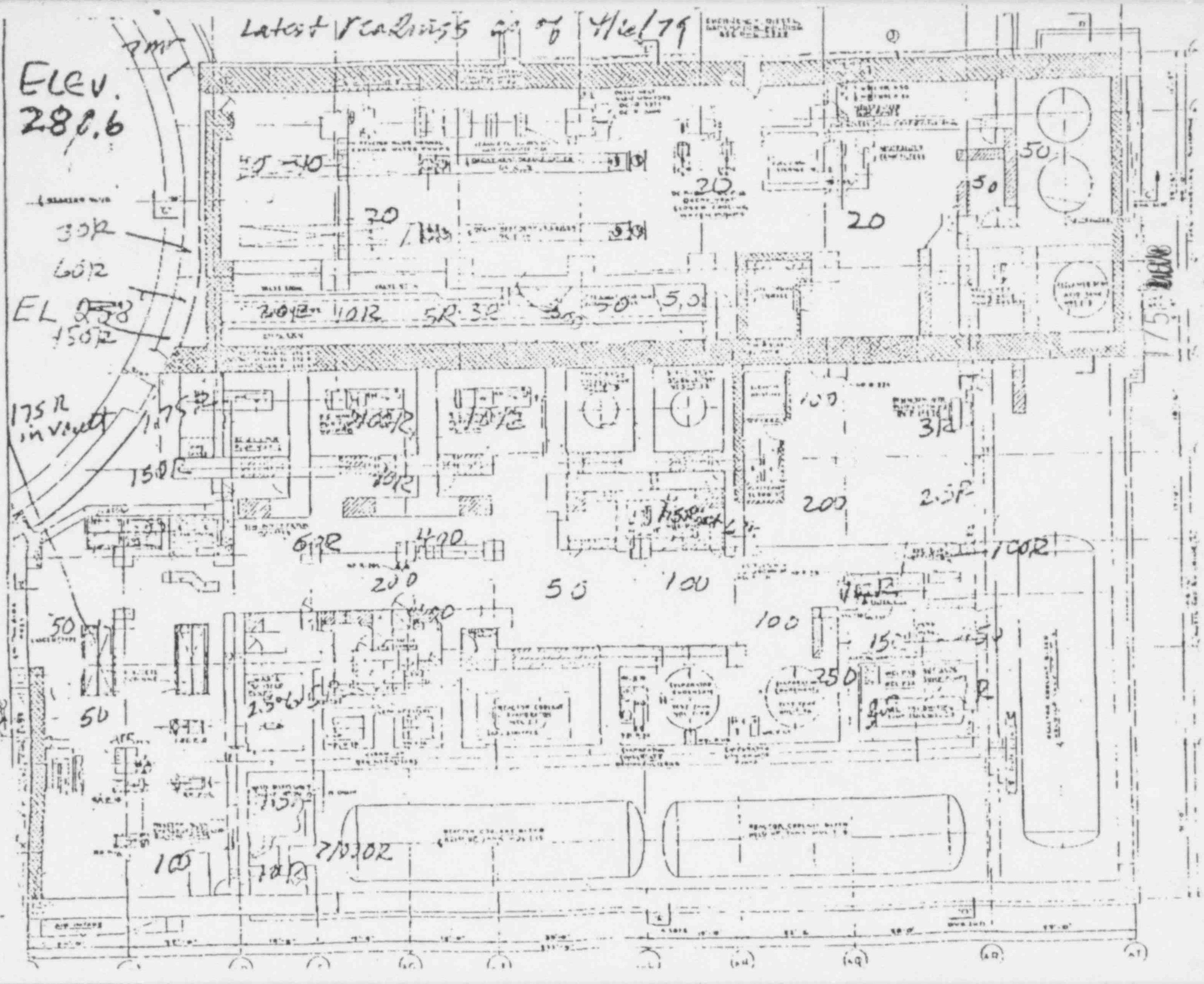
ARMS flight at 0950 4/5 identified plume in sector of 125° - 130°. Radiation measurements using portable gamma scintilator survey instrument were as follows:

at one mile - 0.3 mR/hr
at three miles - 0.05 mR/hr
at ten miles - 0.03 mR/hr


0400
IN mR/HR
EXCEPT THOSE
WITH R
AUX BLDG
EL 328'
45R
50
100
220
200
300
150
30
250
75
75
150
100
200
300
600

Latest readings as of 0900 4/6/79




Latest readings as of 4/6/79
Elev. 280.6
30R
60R
150R
175R in vault
175R
150R
10R
5R
50
20
70
100
200
50
100
250
50
100
50
100

289

4/4/79 . ①

MJW 06:00

Information from Bettis — analysis of condensate from second containment air sample

(0550 from Bettis)

I^{131}	1.9×10^{8} DPM/ml	$85.6\ \mu Ci/ml$	}	Condensate samples
I^{133}	8.7×10^{6} "	$3.9\ \mu Ci/ml$		
Cs^{134}	3.1×10^{4} "	$1.4 \times 10^{-2}\ \mu Ci/ml$		
Cs^{136}	6.3×10^{4} "	$2.8 \times 10^{-2}\ \mu Ci/ml$		
Cs^{137}	1.2×10^{5}	$5.5 \times 10^{-2}\ \mu Ci/ml$		

1.3 ml of condensate in a 19 ml gas sample

175 2~~89~~ 90

Ø733 - Rad Monitors
No indication
Open and Closed

~~0535 - Full set of parameters, Gas Comp.~~
~~0540 - Rccic started - sample~~

another Bulletin - other reactor West, CE,
perhaps GE.
- Roger - Terry Ross
- Cover down here

~~Retesting:~~
~~- Status of 4 psi gage testing~~

Radiation readings on sample

717-944-4144

175 290

IE

~~Region I~~

1, Grier in motion to Site by helicopter
ETA

2. Inspectors @ Site

- 3 Operations

-

3-4 HP's RI

2 RII

2 RIII

12
28 RI

~~Vollmer's Team~~
~~4 NRR~~
~~1 IE~~

175 292

RIII - All leaving O'Hare 1:10 pm
Due Harrisburg 3:44 pm

By charter aircraft

HP's	OPS
R. Dicey	W Little
J. Hiatt	T Harpster
D. Miller	D Boyd
T. Tongue	J Kohler
W. Grant	
W. Axelson	
R. Paul	
D. Sreniawski	

RIV On standby

RV 1 Ops } being dispatched to HQ for support.
1 HP }

175 292

Harrisburg

Command Post — started

R II

Greg G	Bill Peery	Dick W
Don	Jerry Troup	Frank J.
Don P	Jim Saphan	Tom M
Geo J.		Ted V
~~Hooz~~ Jackson		John D.

R II

A	B		C
Greg Gibson	George Jenkins	OPS	Dick Wessman
Don Perrotti	Larry Jackson	OPS	Frank Jape
Dan Montgomery	Jerry Troup		Tom McHenry
Bill Peery	Dick Woodruff		Ted Verdery
Pete McPhail	Herb Young	OPS	John Dyer

Group A to depart by charter flight within 30 min
Flight time 1 hr 45 min.
Aircraft will return to Atlanta and
pick up second group immediately.
Group C will travel commercially ASAP.

175 294

FOR SAM BRYAN

FAXED TO NRR

(MIRAGLIA, AYCOCK

HAD LEFT)

175 295

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 14, 1979

MEMORANDUM FOR: B. H. Grier, Director, Region I
J. P. O'Reilly, Director, Region II
J. G. Keppler, Director, Region III
K. V. Seyfrit, Director, Region IV
R. H. Engelken, Director, Region V

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, OIE

SUBJECT: IE BULLETIN 79-08, Events Relevant to Boiling Water Power Reactors Identified During Three Mile Island Incident

The subject IE Bulletin should be dispatched for action by April 14, 1979, to all BWR power reactor facilities with an operating license.

Subject bulletin and enclosures should also be dispatched for information to all other power reactor facilities with an operating license and to all power reactor construction permit holders.

The text of the Bulletin, and draft letters to the licensee are enclosed for this purpose. ~~Enclosure 1 which consists of the referenced Preliminary Notifications, should be added by the regional office. The letters to the licensee make the commitment to forward the continuing Preliminary Notifications of the incident. These should be forwarded as they are received.~~

Norman C. Moseley

Norman C. Moseley, Director
Division of Reactor Operations
Inspection
Office of Inspection and Enforcement

Enclosures:
1. Draft Transmittal Letters to B&W Licensees
2. Draft Transmittal Letter to all other power facilities
3. IE Bulletin No. 79-08
4. ~~Enclosure 2, to the Bulletin~~

CONTACT: D ~~C. Kirkpatrick~~, IE
~~49-28186~~
49-28019

BMT

175 296

(Draft letter to BWR ~~all~~ power reactor facilities with an operating license)

IE Bulletin No. 79-55 ~~07~~

Addressee:

Enclosed is IE Bulletin No. 79-55 ~~07~~ ~~05~~, which requires action by you with regard to your power reactor facility(ies) with an operating license~~, or a construction permit~~.

Should you have any questions regarding this Bulletin or the actions required by you, please contact this office.

Sincerely,

Signature
(Regional Director)

Enclosures:
1. IE Bulletin No. 79-55 ~~07~~ ~~05~~
~~and Enclosures~~
2. List of IE Bulletins
   Issued in the past
   12 months

175 2907

BWR facilities.
except ~~[illegible]~~

(Draft letter to all power reactor facilities with an operating license or a construction permit)

IE Bulletin No. 79-~~09~~ 08

Addressee:

The enclosed Bulletin 79-~~09~~ 08 is forwarded to you for information. No written response is required. If you desire additional information regarding this matter, please contact this office. ~~The Preliminary Notification of the subject incident (Enclosure 1) will continue to be issued periodically. These will be forwarded to you as they are issued.~~

Sincerely,

Signature
(Regional Director)

Enclosures:
1. IE Bulletin No. 79-~~09~~ 08
~~and Enclosures~~
~~2. List of IE Bulletins issued in the past 12 months~~

175 298

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

April 1, 1979

IE Bulletin No. 79-~~05~~ 08

~~NUCLEAR INCIDENT AT THREE MILE ISLAND~~ Events Relevant to Boiling Water ~~Reactors~~ Power Reactors Identified During Three Mile Island Incident.

Description of Circumstances:

On March 28, 1979 the Three Mile Island Nuclear Power Plant, Unit 2 experienced core damage which resulted from a series of events which were initiated by a loss of feedwater transient. Several aspects of the incident may have general applicability ~~in addition to generic applicability of~~ to operating ~~Babcock and Wilcox~~ Boiling water reactors. This bulletins ~~is provided to inform you of the nuclear incident and to~~ requests certain actions of licensees of operating boiling water reactors.

Actions to be taken by Licensees:

~~For all Boiling water reactor facilities with an operating license (the actions specified below, [illegible])~~ Complete

1. Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 3/28/79 accident included in Enclosure 1 to IE Bulletin 79-05A.

   a. This review should be directed toward understanding: (1) the extreme seriousness and consequences of the simultaneous blocking of both trains of a safety system at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident; (2) the apparent operational errors which led to the eventual core damage; ~~(3) that the potential exists, under certain accident or transient conditions, to have an acceptably high water level in the vessel simultaneously with the reactor core having insufficient~~ cooling ~~amount of water;~~ and (~~4~~ 3) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.

   b. Operational personnel should be instructed to (1) not override automatic action of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions (see Section ~~6a~~ 5 of this bulletin); and (2) not make operational decisions based solely on a single plant parameter

175 2989

indication when one or more confirmatory indications are available.

c. All licensed operators and plant management and supervisors with operational responsibilities shall participate in this review and such participation shall be documented in plant records.

2. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to ~~permit~~ initiate containment isolation, whether manual or automatic, of all lines whose isolation does not degrade needed safety features or cooling capability, upon automatic initiation of safety injection.

3. Describe the actions, both automatic and manual necessary for proper functioning of the auxiliary ~~core cor~~ heat removal systems (e.g., RCIC) that are used when the main feedwater system ~~is~~ is not operable. ~~Describe the time ... timing for ... sequence of.~~

~~Des~~

For any manual action necessary, describe in summary ~~form~~ the procedures by which this action is taken in a timely sense.

175 302

Auxiliary Cooling System

makeup of RCS inventory

3. For facilities for which ~~any safety system~~ is not automatically initiated, prepare and implement immediately procedures which require the stationing of an individual (with no other assigned concurrent duties and in direct and continuous communication with the control room) to promptly initiate this safety system for those transients or accidents the consequences of which can be limited by such action.

7

1

4. Describe all uses ~~of~~ and types of vessel level indication for both automatic and manual initiation of safety systems. Describe other redundant instrumentation which the operator might have to give the same information regarding plant status. Instruct operators to utilize other available information to initiate safety systems.

~~Desirable~~ Ensure promptly action.

an auxiliary system which provides ~~cooling in the~~ core cooling in the event of loss of main feedwater

175 303

5. Review the action directed by the operating procedures and training instructions to ensure that:

a. Operators do not override automatic actions of engineered safety features, unless continued operation of engineered safety features will result in unsafe plant conditions (e.g. vessel integrity). ~~For example, if continued operation of engineered safety features would threaten reactor vessel integrity then the ECCS should be secured.~~ Out

b. Operators are provided additional information and instructions to not rely upon vessel level indication alone for manual actions, but to also examine other plant parameters indications in evaluating plant conditions.

2 Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance, testing, plant and system startup, and supervisory periodic (e.g., daily/shift checks,) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

175 305

7. Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping, venting or other release of radioactive liquids and gases will not occur inadvertently.

In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

a. Whether interlocks exist to prevent transfer when high radiation indication exists, and

b. Whether such systems are isolated by the containment isolation signal.

c. The basis on which continued operability of the above features is assured.

8. Review and modify as necessary your maintenance and test procedures to ensure that they require:

a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

9. Review your prompt reporting procedures for NRC notification to assure that NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation. Further, at that time an open continuous communication channel shall be established and maintained with NRC.

175 308

10. Review operating modes and procedures to deal with significant amounts of hydrogen gas that may be generated during a transient or other accident that would either remain inside ~~this~~ the primary system or be released to the containment.

175 3089

11. Propose changes, as required, to those technical specifications which must be modified as a result of your implementing the items ~~1 through 14~~ above.

175 310

For all boiling water reactor facilities with an operating license, respond to Items 1-~~12~~ 10 within 10 days of the receipt of this Bulletin. Respond to item ~~13~~ 11 (Technical Specification Change proposals) in 30 days.

Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D.C. 20555.

For all other power reactors with an operating license or construction permit, this Bulletin is for information purposes and no written response is required.

Approved by GAO, B180225 (Roo72); clearance expires 7/31/80. Approval was given under a blanket clearance specifically for identified generic problems.

175 310

10:00 STATUS

Reactivity Control Systems

All full length control rods fully inserted into the core.

Reactor Coolant System Parameters -

T_{cold} = ~ 281°F

T_{hot} = 281 °F

Pressurizer Press = 863 psi (Bubble in Pressurizer)

Temp = 529 °F

Pressurizer Level = 348 inches

Reactor Coolant System Status -

- 1 Reactor Coolant Pump providing forced circulation through the core

Level 92% Press 30 psig

- Steam Generator A (in loop with operating pump) providing heat sink for reactor coolant system. Steam Generator B isolated (Press = 35 psig)
- Steam Generator A steaming to condenser. No steam being released to the atmosphere.

Makeup / Letdown

- Makeup flow approximately 31 gpm from makeup pump (same as High Pressure Injection ECCS). Letdown flowrate ~10 gpm.

Decay Heat Removal

System secured, System should be available when needed. Pump rooms have 4 inches and 1 inch of water each. Motor operated suction valves from the RCS hot leg inside containment should not be submerged.

175 312

Containment

Containment Building isolated. Approximately 250,000 gallons of water in containment sump, slight vacuum in containment.

Auxiliary Building

Approximately 10,000 gallons of water in the building due to apparent overflow reactor coolant bleed holdup tanks which were filled from reactor coolant drain tank.

175 312

Gas Decay Tanks

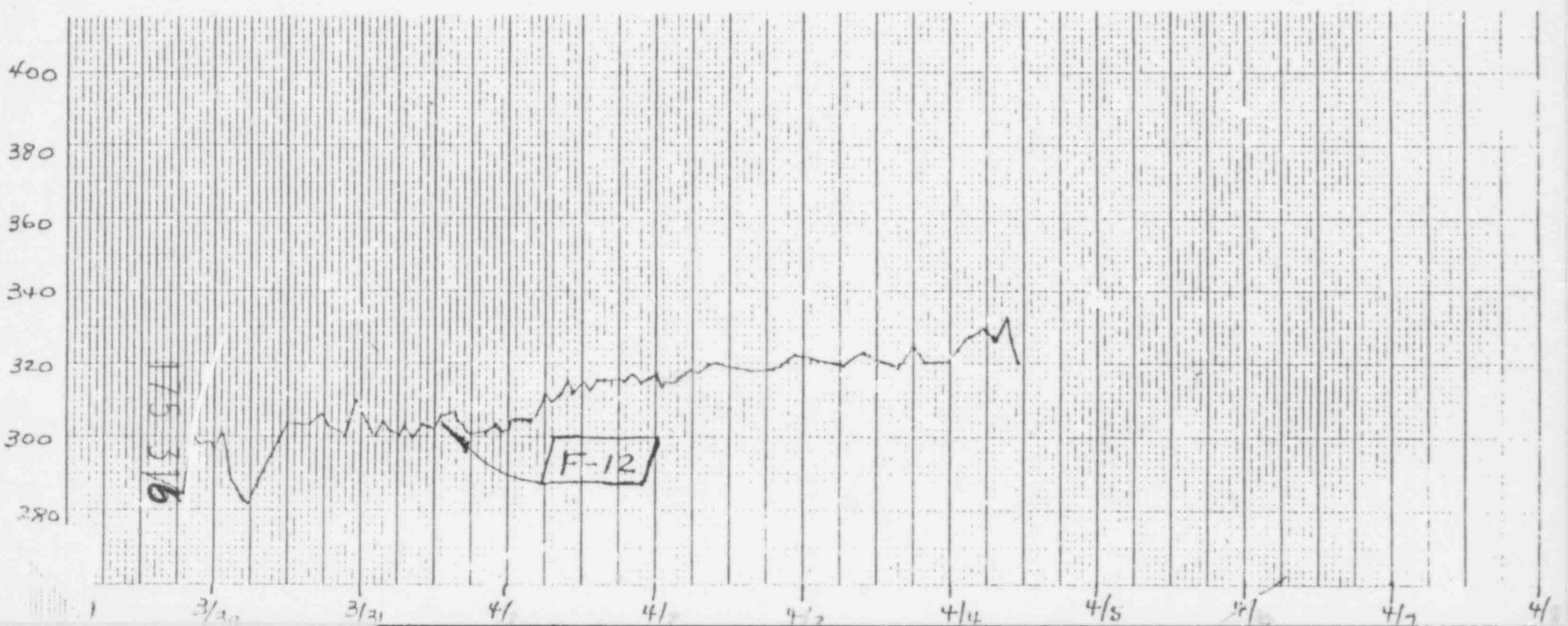
1. (Pressure)	Tanks crossconnected	80 psig
2. (Pressure)		80 psig

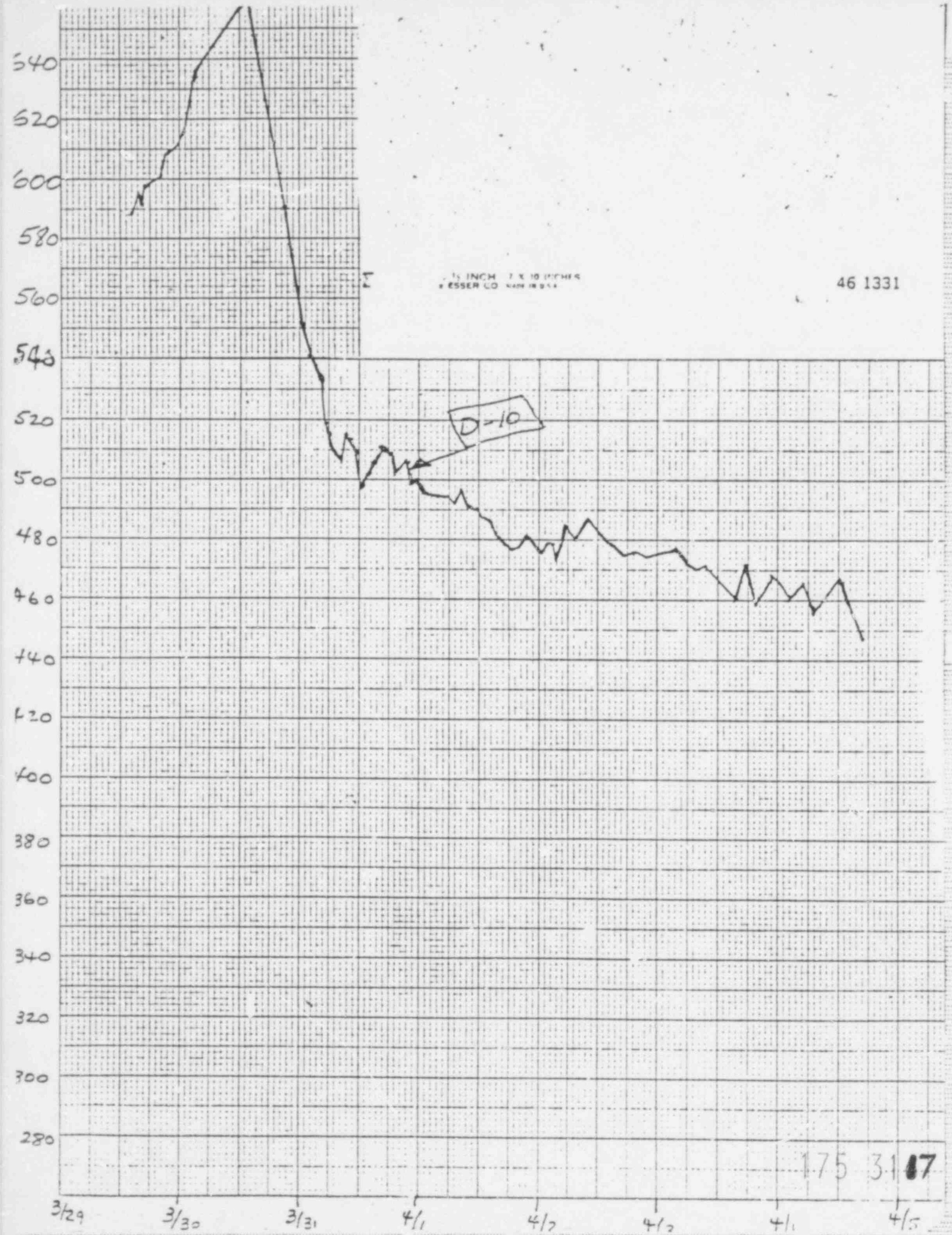
DG Generator (Operability)	Operable
Off-Site Power (Sources Avail)	2A, 2B xform operable
Instrument Power Supplies	4 inverters operable/on

RI EMERGENCY RESPONSE CENTER MANNING - (IF REQUIRED)

		In-Charge	RO&NS Section Chief	RO&NS Inspectors	FF&MS Branch
March 28	1700 - 2400	E. Brunner	R. Keimig	D. Haverkamp/Bettenhausen	L. Thonus/M. Slobodien
March 29	0001 - 0800	J. Allan	E. McCabe	W. Lazarus/G. Kalman	G. Yuhas/H. Crocker
	0800 - 1600	B. Grier	H. Kister	D. Beckman/T. Foley	J. White/R. Bores
	1600 - 2400	E. Brunner	D. Caphton	T. Stetka/W. Rekito	L. Thonus/M. Slobodie
March 30	0001 - 0800	J. Allan	R. Keimig	D. Haverkamp/Bettenhausen	G. Yuhas/H. Crocker
	0800 - 1600	B. Grier	H. Kister	W. Lazarus/G. Kalman	J. White/R. Bores
	1600 - 2400	E. Brunner	D. Caphton	D. Beckman/T. Foley	L. Thonus/M. Slobodie
March 31	0001 - 0800	J. Allan	R. Keimig	T. Stetka/ W. Rekito	G. Yuhas/H. Crocker
	0800 - 1600	B. Grier	H. Kister	D. Haverkamp/Bettenhausen	J. White/R. Bores
	1600 - 2400	E. Brunner	D. Caphton	W. Lazarus/G. Kalman	L. Thonus/M. Slobodie

175 315


400
380
360
340
320
300
280
175 316
F-12
3/2
3/21
4/
4/
4/11
4/5
4/7
4/


640
620
600
580
560
540
520
500
480
460
440
420
400
380
360
340
320
300
280
D-10
46 1331
3/29
3/30
3/31
4/1
4/2
4/5

175 3107

OPERABLE EQUIPMENT

~0300
4/1/79

RCS

1. Which Reacor Coolant Pump Oil Lift Pumps have been run and are considered operable — ✓ All oil lift pumps operable
2. Are Any Reactor Coolant Pumps inoperable for any reason (Except possible vapor binding) — All RCPs operable
3. Are all Pressurizer heaters available — All are operable
4. Are both Spray valves operable — Both operable

DHR

Last Tested

1. Have both Decay Heat Removal Pumps been checked out as operable (Test Run) — considering testing in near future.
~~2. Are all valves~~
2. Are any valves in either Train out of position or inoperable

Hi Head Injection

1. Are both Hi head injection pumps operable — All 3 are operable
2. Any Values in either Emergency Injection Train inoperable — Yes

Contain. Sprays

1. Are both Trains of Spray operable — yes

Containment Fan Cooling Units

1. How many operating — 4/5 operating ✓
2. How many available — 5 operable ✓

Nuclear Service water System

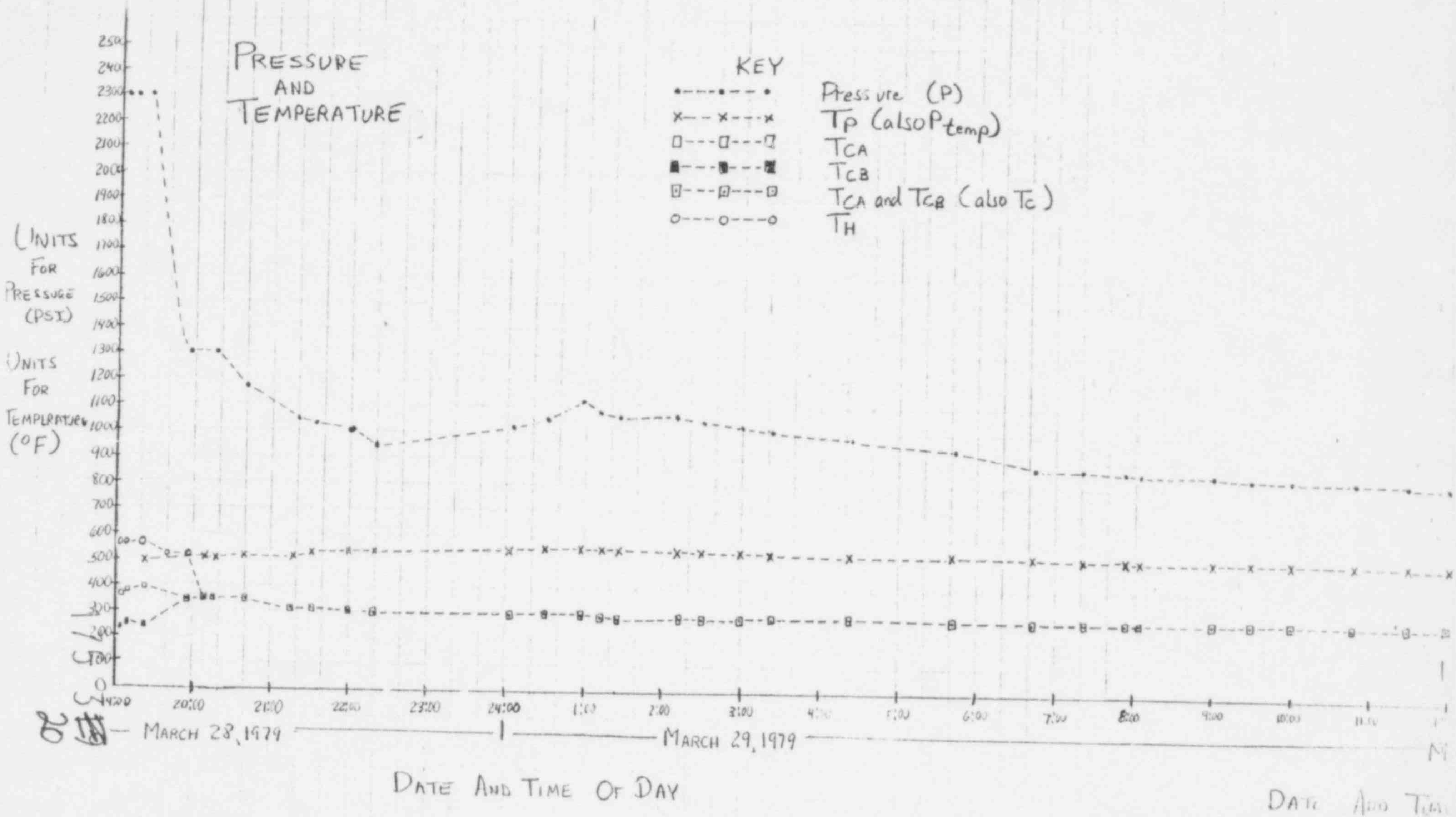
1. How many pumps operating — 2 operating
2. How many pumps Available — 4 available

~~175~~-318

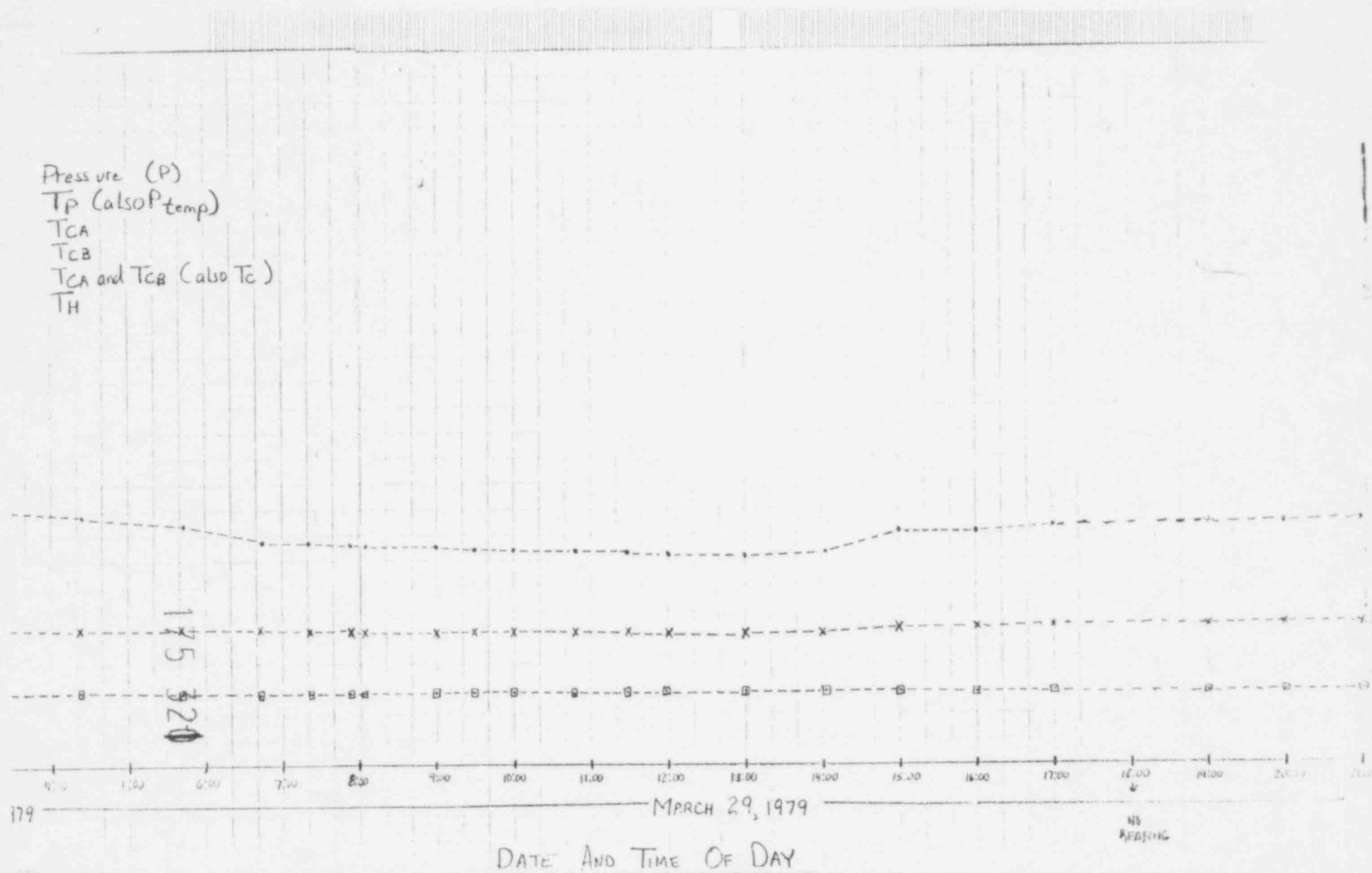
0300
4/1/79

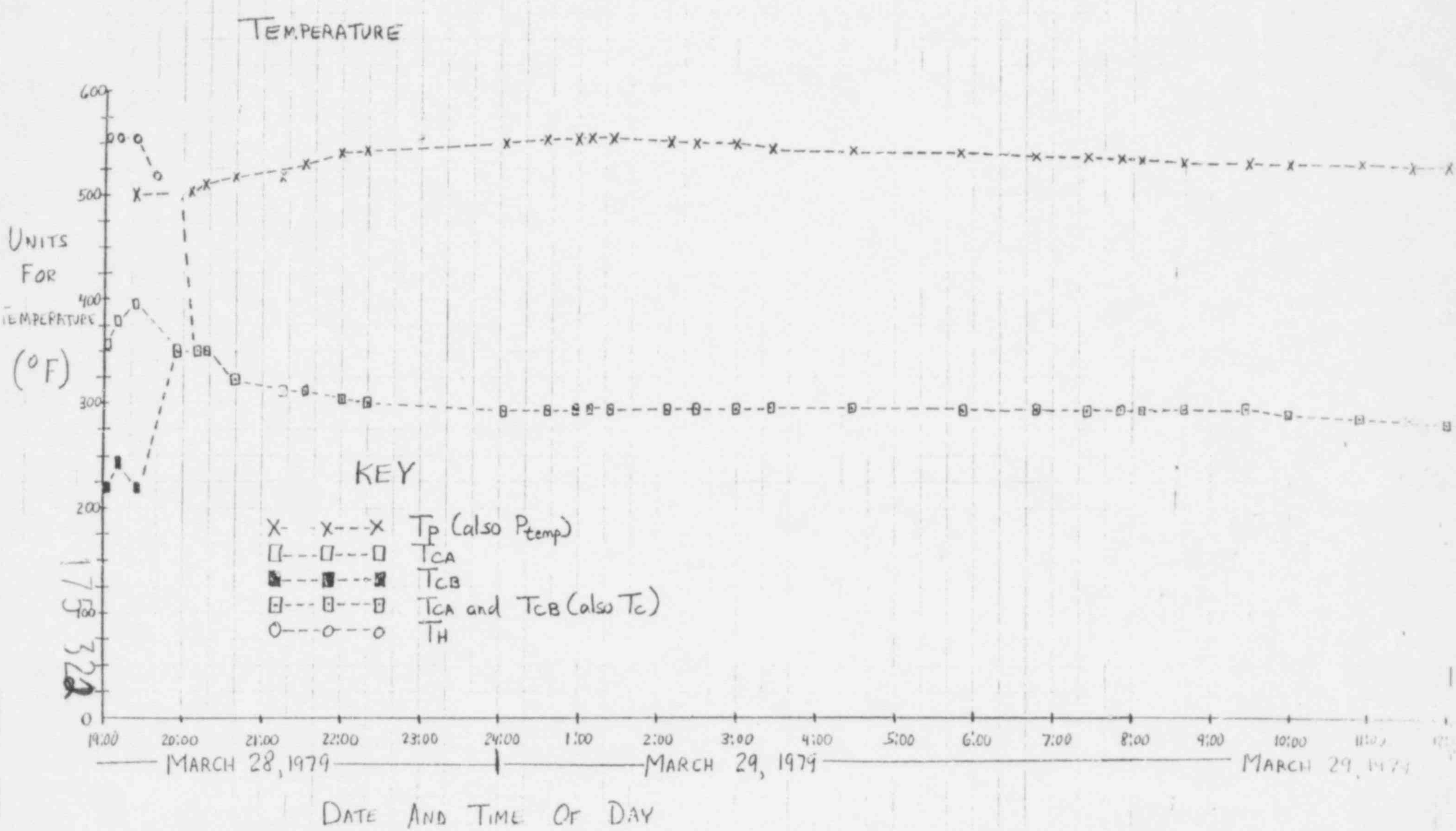
Nuclear Services Closed Cooling Water System	1. How many Pumps ~~are~~ running	2 1A, 1B
	2. How many available	~~4~~ 3
	~~3. How~~	
Aux. Feed Sys	1. Are both electric pumps Available	Yes
	2. Volume of water (gal) in Aux feed Tank min 27.5' in both tanks	A Tank 21' B Tank 21'
BWST	1. Volume in Borated Water Storage ~~system~~ Tank	36'
Diesels	1. How many Diesels Available (Unit 2)	2 diesels operable
	2. Can any of Unit 1 Diesels be made available to supply unit 2 emergency Busses — Cannot	
Off-Site Power	1. How Many sources of Off-Site Power are Available ~~2~~ now available	
Hydrogen Re-Combiner	1. Have both Recombiners been tested 1 tested Sat ok 3/31 1 being tested now.	
Unit #2 Spent Fuel Storage Pool	1. Level of Water Now in The Spent Fuel Pool 0 in Unit 1 drained to work on rack	
	2. Can it water in spent fuel pool be released to The environment (is it contaminated)	

175 3189



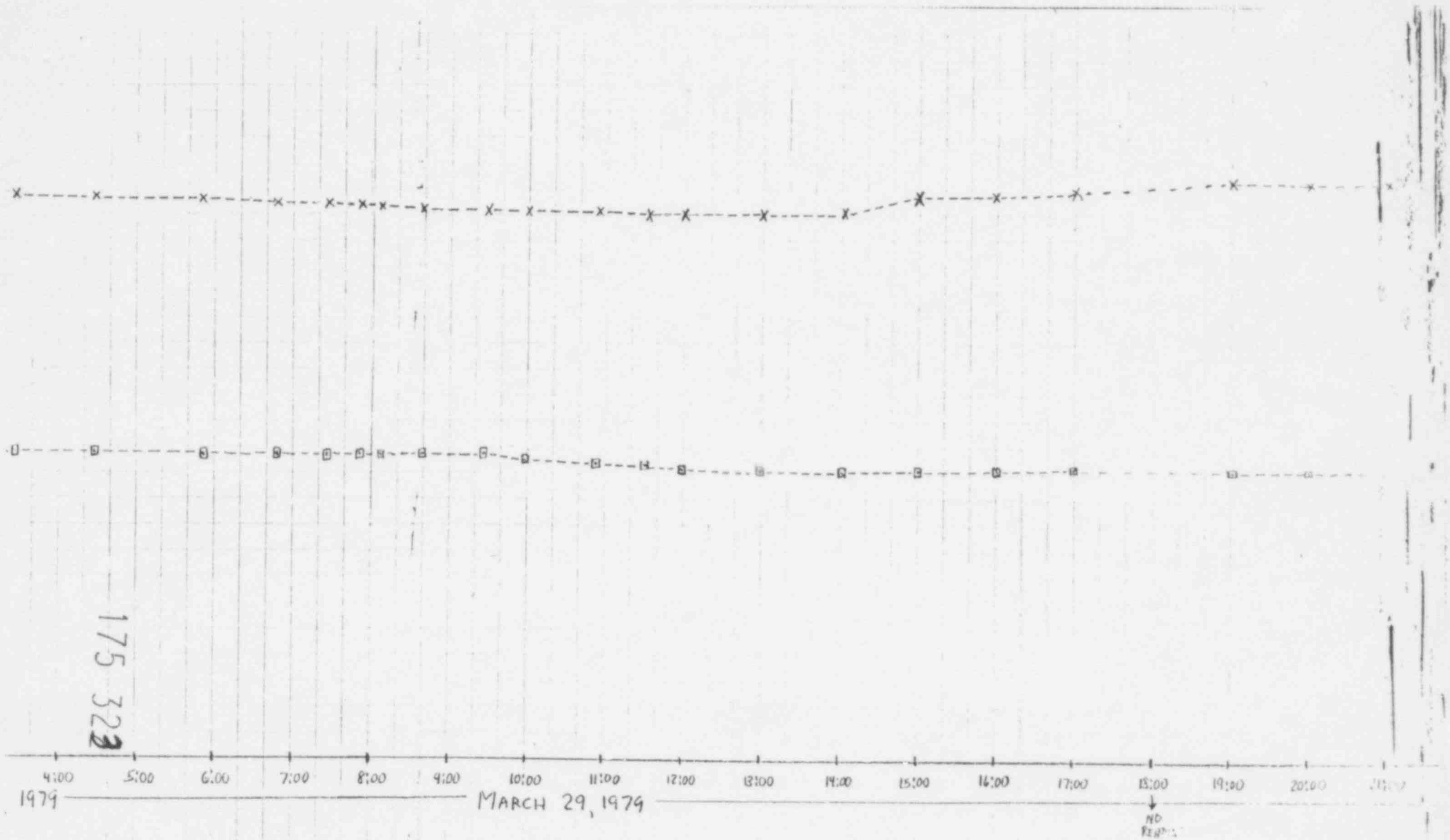
PRESSURE AND TEMPERATURE
KEY
Pressure (P)
TP (also Ptemp)
TCA
TCB
TCA and TCB (also TC)
TH
UNITS FOR PRESSURE (PSI)
UNITS FOR TEMPERATURE (°F)
2500
2400
2300
2200
2100
2000
1900
1800
1700
1600
1500
1400
1300
1200
1100
1000
900
800
700
600
500
400
300
200
100
0
20:00
21:00
22:00
23:00
24:00
1:00
2:00
3:00
4:00
5:00
6:00
7:00
8:00
9:00
10:00
MARCH 28, 1979
MARCH 29, 1979
DATE AND TIME OF DAY
DATE AND TIME


Pressure (P)
T_P (also P_{temp})
T_{CA}
T_{CB}
T_{CA} and T_{CB} (also T_C)
T_H
175 320
8:00
9:00
10:00
11:00
12:00
13:00
14:00
15:00
16:00
17:00
19:00
179
MARCH 29, 1979
NO READING
DATE AND TIME OF DAY



TEMPERATURE
Units For Temperature (°F)
600
500
400
300
200
100
0
KEY
X---X---X T_P (also P_{temp})
□---□---□ T_{CA}
■---■---■ T_{CB}
⊡---⊡---⊡ T_{CA} and T_{CB} (also T_C)
O---O---O T_H
19:00
20:00
21:00
22:00
23:00
24:00
1:00
2:00
3:00
4:00
5:00
6:00
7:00
8:00
9:00
10:00
MARCH 28, 1979
MARCH 29, 1979
DATE AND TIME OF DAY

175 322



4:00
5:00
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10:00
11:00
12:00
13:00
14:00
15:00
16:00
17:00
18:00
19:00
20:00
21:00
1979
MARCH 29, 1979
NO READING

175 322

TEMP CHANGES RESULTING FROM CHANGE RCP 1A TO RCP 2A

Element	Initial change	Change @ 1600	Change* @ 1800	Change @ 1847	Change @ 2000	Change @ 2207
B-7	+2	-1	-1	-1	0	-2
C-9	+1	-1		0	+1	-3
D-10	-117	-7		-7	-6	-4
D-14	-4					
E-7	+11	-7	-4	-3	-2	-3
E-9	-7	-1		-3	-3	-2
E-11	-105	-1		-3	0	-3
F-7	+9	+2	-2	-2	0	-1
F-8	+8	-1	-5	-5	+5	0
F-12	-29	-1		-1	0	-2
F-13	-16	-1		-1	-1	-1
G-2	+1	0	-1	-1	-1	-2
G-5	+48	+7		+2	0	-1
G-9	-5	-1		-2	+2	+1
G-11	-110	-5		-4	-1	-3
G-13	-28	-1		0	-1	-2
H-1	+1	0	0	-1	0	-2
H-5	+41	+1	-1	-1	-1	+21
H-8	+94	-7	-5	-8	+1	+2
H-9	-11	0		-1	0	-2
H-13	-29	-		-1	-1	-2
K-11	-34	-10		-1	0	-1
L-3	+2	-		+1	+1	-1
L-11	-44	+3		-6	+1	+1
M-9	+30	-15		-11	+2	+3
M-10	+17	-16		-9	+1	-1
N-8	+7	-7		-2	+1	0

* Only SW Quadrant TC Rdgs

Element	Initial ΔT	Rdg 1447	Rdg 1600	ΔT 1447 to 1600	Special Rdg 1800	ΔT 1600 to 1800	Rdg @ 1847	ΔT 1600 to 1847
B-7	+2	294	293	-1	292	-1	292	-1
C-9	+1	308	307	-1			307	0
D-10	-117	319	312	-7			305	-7
D-14	-4	219	—	—				
E-7	+11	313	312	-1	308	-4	309	-3
E9	-7	340	339	-1			336	-3
E11	-105	295	294	-1			291	-3
F7	+9	276	278	+2	276	-2	276	-2
F8	+8	222	221	-1	216	-5	216	-5
F12	-29	293	292	-1			291	-1
F13	-16	292	291	-1			290	-1
G2	+1	293	293	0	292	-1	292	-1
G5	+48	369	376	+7			378	+2
G9	-5	336	335	-1			333	-2
G11	-110	327	322	-5			318	-4
G13	-28	295	294	-1			294	0
H1	+1	292	292	0	292	0	291	-1
H5	+41	335	336	+1	335	-1	335	-1
H8	+94	468	461	-7	456	-5	453	-8
H9	-11	296	296	0			295	-1
H13	-29	292	—	—			291	-1
K11	-34	324	314	-10			313	-1
L3	+2	293	—	—			294	[illegible]
L11	-44	293	296	+3			290	-6
M9	+30	355	340	-15			329	-11
M10	+17	333	317	-16			308	-9
N8	+7	316	309	-7			307	-2

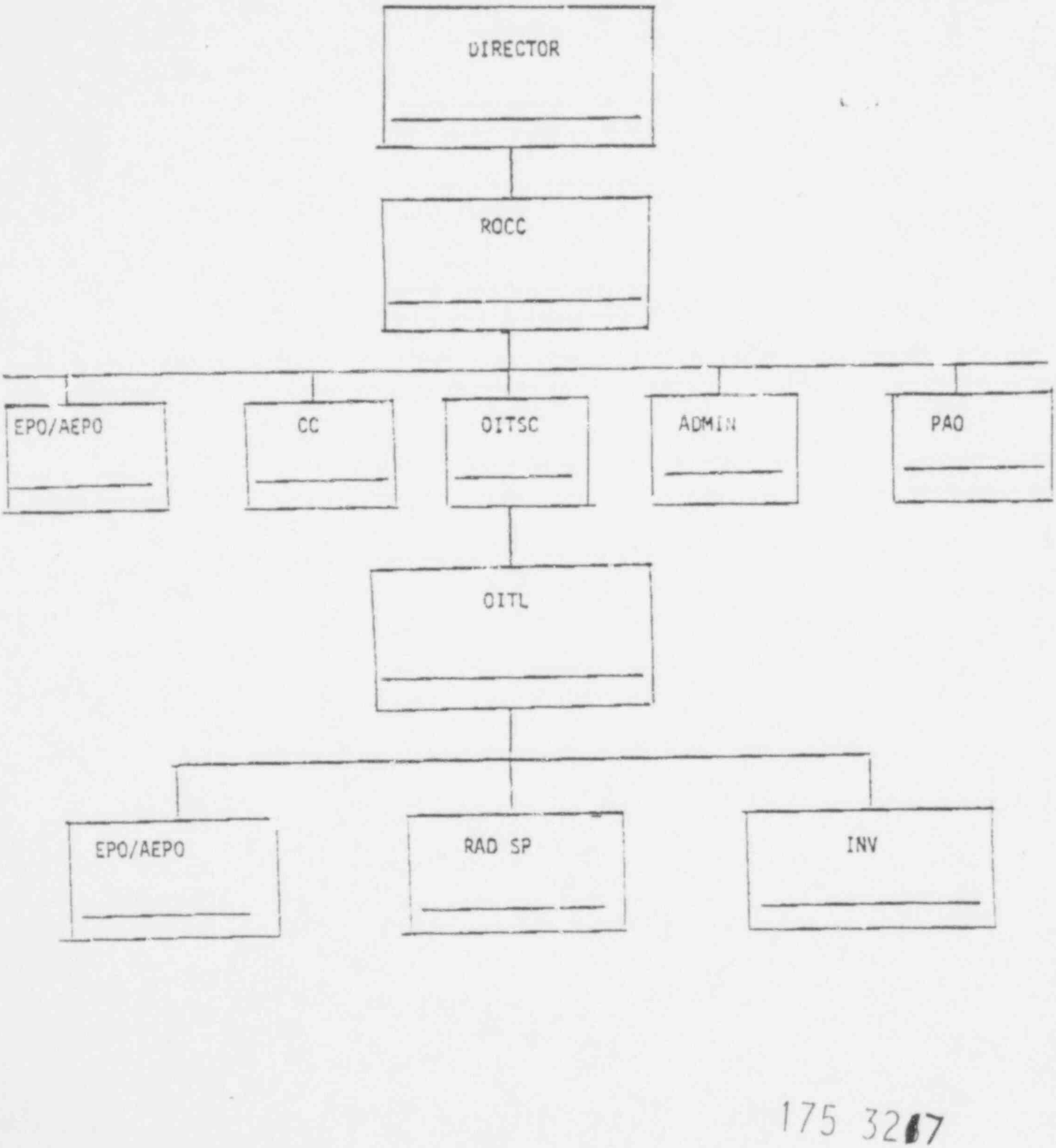
175 325

	Rdg @ 2000	ΔT 1847 to 2000	Rdg @ 2207	ΔT 2000 to 2207	Rdg @ 0001	Δt 2207 to 0001	Rdg @ 0210	Δt 0001 to 0210	Rdg @ 0400	Δt 0210 to 0400
B7	292	0	290	-2	289	-1	290	+1	291	+1
?9	308	+1	305	-3	~~292~~ 303	~~+5~~ -2	303	0	303	0
D10	299	-6	295	-4	293	-2	294	+1	294	+1
~~H4~~	–	—								
E7	307	-2	304	-3	304	0	~~304~~ 301	-2	301	-1
E9	333	-3	331	-2	329	-2	328	-1	326	-2
?11	291	0	288	-3	287	-1	172		172	
F7	276	0	275	-1	276	-1	273	-3	272	-1
F8	221	+5	221	0	223	+2	218	-5	214	-4
F12	291	0	289	-2	288	-1	290	+2	290	0
F13	289	-1	288	-1	287	-1	288	+1	289	+1
G2	291	-1	289	-2	289	0	290	+1	291	+1
G5	378	0	377	-1	377	0	377	0	377	0
G9	335	+2	336	+1	336	0	335	-1	334	-1
G11	317	-1	314	-3	315	+1	315	0	313	-2
G13	293	-1	291	-2	291	0	292	+1	292	0
H1	291	0	289	-2	289	0	291	+2	292	+1
H5	334	-1	355	+21	336	-19	339	+3	340	+1
H8	454	+1	456	+2	444	-12	455	+11	453	-2
H9	295	0	293	-2	292	-1	294	+2	291	-3
H13	290	-1	288	-2	288	0	289	+1	289	0
K11	313	0	312	-1	312	0	313	+1	314	+1
L3	295	?	294	-1	293	-1	294	+1	295	+1
L11	291	+1	292	+1	291	-1	291	0	291	0
M9	331	+2	334	+3	334	0	334	0	334	0
M10	309	+1	308	-1	312	+4	312	0	313	+1
N8	306	-1	306	0	306	0	307	+1	308	+1

JAY CUNNINGHAM

FOR TRANSMITTAL TO: TO: MR. GRIER FROM: REGION I USNRC

VIA HQ LINK TO INDEPENDENT MEAS VAN OR NRC TRAILER AT TMI SITE



175 32[illegible]7

Region II

Name	Section	Date to TMI	Date return to Office
Collins	Rad. Support	3/29	4/13
Zavodoski	"	3/29	4/13
Jackson	"	3/30	4/13
Jenkins	"	3/30	4/6
Troup	"	3/30	4/20
Ewald	"	4/4	4/20
Hosey	"	4/18	5/2
A. Gibson	"	4/4	4/20
Andrews	ES&P	3/30	4/6
G. Gibson	"	3/30	4/6
McPhail	"	3/30	?
Montgomery	"	3/30	4/11
Perrotti	"	3/30	4/13
Peery	"	4/4	4/20
Projanowski	"	4/11	4/25
Allen	"	4/11	4/25
Brown, Mat. Insp.	FFMS	4/11	4/25
Kahle, Fuel Fac.	"	4/11	4/25
Woodruff, Mat. Insp.	"	3/30	4/13
Young, Mat. Insp.	"	3/30	4/13

175 328

Recd 2235 [illegible]

[illegible]

DATE: 4-2-79

OFF-SITE DATA

Time (EST)

1950 NRC ground level gamma surveys at 15 checkpoints located on both sides of the river up to a distance of about 2.5 miles north and south again showed essentially background readings.

175 3209

3/31/79 22:30 (7) [illegible]

Bob Bores Region I

1° Coolant Sample

Gross alpha Data

2 samples

8.4×10^5 dpm/ml = 0.38 μCi

1.0×10^6 " = 0.45 "

Uranium Chemistry - NO uranium means < 250 ppm.

Maybe extent of core melt is not as extensive as originally estimated.

Containment samples - Not available ~~next couple hours.~~

next couple of hours

Population Centers within
10 miles of Three Mile Island

Center	Population (1970)	Distance*-miles	Direction
Goldsboro	576	1.0	W
Royalton	1040	2.0	N
Middletown	9080	2.5	N
Highspire	2947	4.0	NNW
Yorkhaven	671	4.0	S
Elizabethtown	8072	6.0	E
Manchester	2391	6.5	S
Steelton	8555	7.0	NW
New Cumberland	9803	9.0	WNW
Harrisburg	68061	9.0	NW
Hummelstown	4723	9.0	N
Hershey	4707	10.0	NNE

*references FES 1972

Mt. Wolfe	(2)	I	0.10	child
		Cs	0.0055	
		I	0.034	adult
		Cs	0.0018	
Mt Wolfe	(3)	I	0.12	child
		I	0.041	adult

Dose Finite Plume

1%/day containment Leakage

Xe - 133 - 675 µci/cc
~~I-131 - 0.063 µci/cc~~
inside containment

4/2/79

1700 hrs

Bivins - was requested to look at soils and the needs to protect against spills from the temporary rad waste tankage.

Need to look at venting of temporary tankage

Need to do dose assessment on failure of liquid temp waste tank, gas decay tank failure.

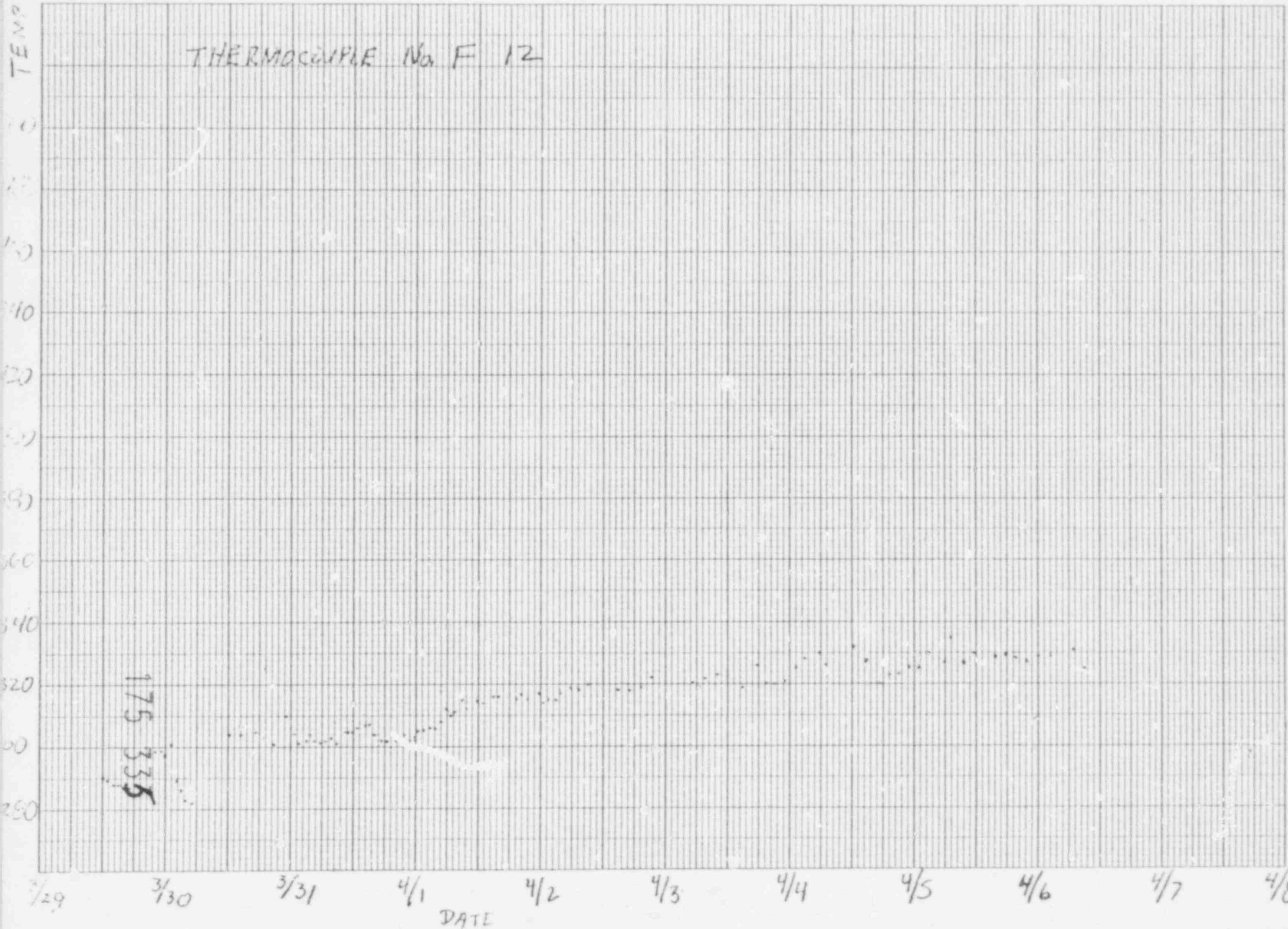
B&R — Wagner called give info on samples to Frank Patti - Paramus, N.J.
201-265-6717-
Chief Nuclear Engineer

60-65 Ci...
1605 2%
750 mcr/hr
90 mr/hr

Info is necessary for designing
2mR/hr
Gaseous & Liquid Waste Tank Farm for Met Ed.

175 334

K+E 10 X 10 TO ½ INCH 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A. 46 1331

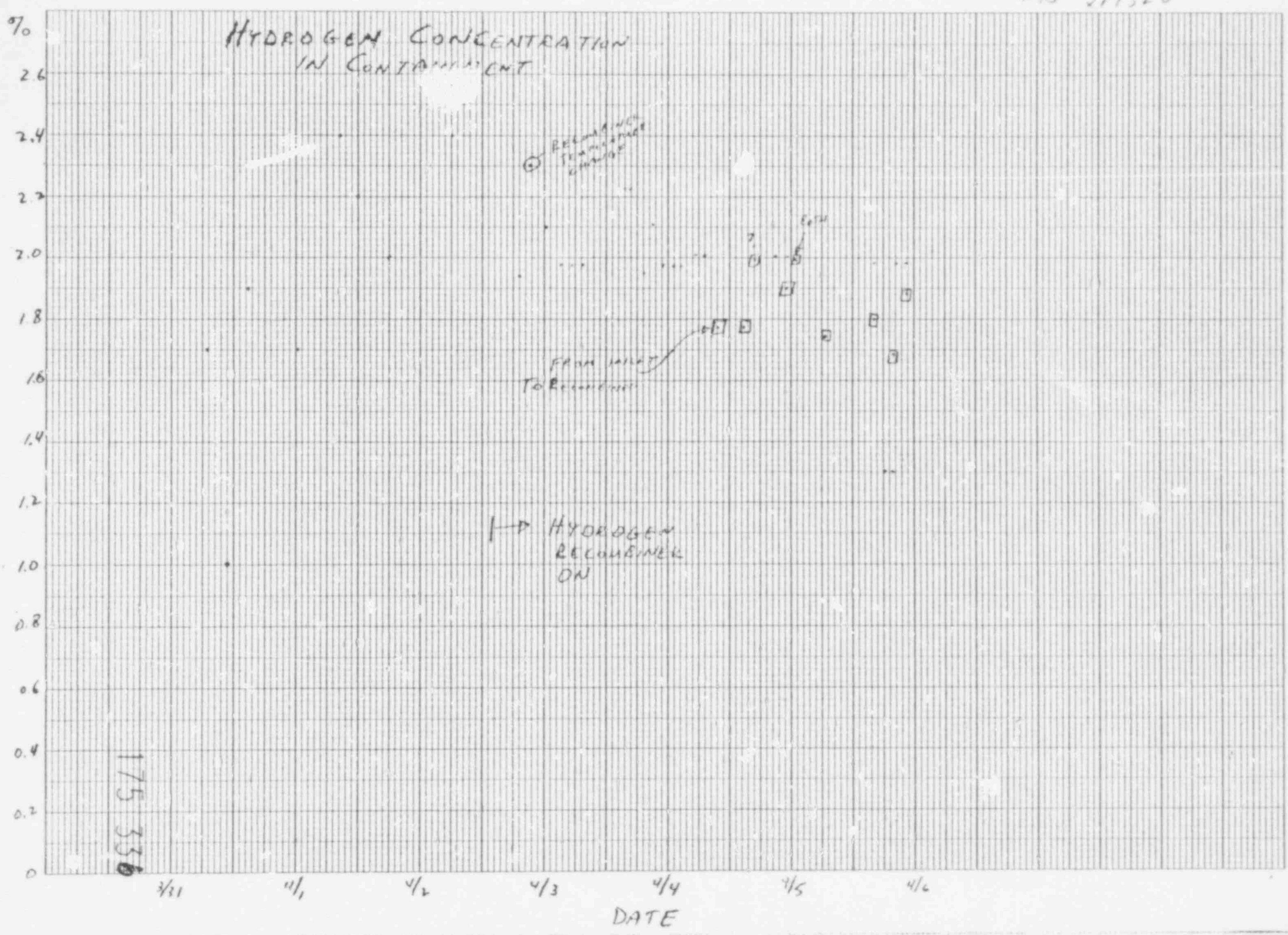


K-E 10 X 10 TO ½ INCH 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.

46 1331

60 cfm

R18 74320



DATA FROM
PHONE LOG + THERMOCOUPLE
DATA SHEETS

① H_2

H_2 CONCENTRATION IN CONTAINMENT

(NOTE - SOME ARE FROM CONTAINMENT SAMPLES OR RECOMBINER INLET SAMPLE)

DATE	TIME	% H_2
3/31	0700	1.7
	1045	1
	1500	1.9 (REPORTED 1845 - 3/31)
4/1	0050	1.7 (LOGGED 0500)
	0845	2.4
	1200	2.2 (REPORTED 1347)
	1800	2 (REPORTED 1915)
4/2	0200	2.3
	0630	2.4 (REPORTED 0800)
	1900	1.94 (FROM RECOMBINER TEMP.)
	2130	2.3 (FROM RECOMBINER)
4/3	0005	2.1
	0200	1.97
	0500	1.97
	0730	1.97
	1600	2.23
	1700	2.23
	1930	1.95
	2100	2.11
	2300	1.97

DATE	TIME	% H_2
4/4	0118	1.97
	0300	1.97
	0500	2.01
	0700	2.01
	1400	1.77 (FROM RECOMBINER INLET)
	1430	1.77 "
	1710	1.99 "
	2100	2.01
	2300	2.01 (RECOMBINER 1.90%)

TMI

175 33[illegible]7

(2)
H_2

H_2 CONCENTRATION IN CONTAINMENT
CON'T

DATE	TIME	% H_2
4/5	0030	1.95 (RECOMBINER) (1.98 CONT. SAMPLE)
	0630	1.74 "
	1600	1.80 (") (1.98 CONT. INCLINE)
	1830	1.3 (CONT. SAMPLE)
	2000	1.88 (RECOMB.) (1.3 INCLINE)
	2030	1.98 (INCLINE)
	2200	1.88 (") (1.98 ")
4/6	~~[illegible]~~ 0500	1.73 (RECOMB.) (1.98 INCLINE)

175 338

977-7535

IRACT SUPPORT SCHEDULE

APRIL 7 THROUGH APRIL 10

Shift

Shift	IRACT DIRECTOR			
8-4	Moseley	Blackwood**	Showe	Kirkpatrick**
4-12	Bryan	Whitt	Harmon	DeBevic
12-8	Woodruff*	McKee	Bemis	Stone
		Off Sunday	Off Tuesday	Off Saturday

*Woodruff begins 12-8s at 12:01 a.m. Sunday.

**Kirkpatrick and Wilber Blackwood to report at 6:00 a.m on Monday

B E Bryan for
N. C. Moseley

NRR CALL SHEET

Provide the following core/plant data to NRR contacts (Frank Miraglia to 12 midnite and Mike Aycock 6 AM to 4: pm) at Phone No. 28041. This info is in lieu of NRR man on duty.

Core / plant data

Date & Time

1. Rx or PZR Pressure
2. TA (cold leg) Temp
3. TPZR (HOT Leg) Temp
4. Let DOWN Flow
5. HOTEST TC (H8)
6. Average core TC (make estimate)
7. Hydrogen Co (Recombiner)
8. Containment Pressure
9. Containment Temp.

from Control Room Inspector

175 3~~38~~ 40

49 14394

Ed Blackwood

We need someone who lives near Goddard Space Flight Center to pick up some tapes each night at 2p.m. and bring them back to the operations center when you start your shift at 12 Mid.

The nights we need are Wednesday, Friday, Saturday and Sunday. If you can do this for us please leave me the message and let me know a time I can call you about it.

I get into the office at 9:15a.m. and leave at 6p.m.
Joe comes in from 2pm to 10pm.

Thank you and please let us know.

Donna

April 3, 1979
12:30 PM

TO ALL PERSONS WORKING ON THE TMI INCIDENT

Because of the constantly changing names and faces both within and outside the Operations Center, the following procedure is effective immediately:

- if you are working out of an office on the 3rd floor, please give that office phone number out for all callbacks:

- from now on, any phone call received in the Operations Center for individuals not assigned routine positions in the Center will be posted on the bulletin board behind the phone operators in Rm. 339.

Thanks.

Joe

Joe Hegner

175 342

4/4/79

PREPARATION OF PN's

PN's --- all input --- are to be <u>initially</u> drafted by HQ IRACT then sent to TMI CENTER for concurrence.

TMI CENTER must receive promptly a copy of the PN as issued by HQ IRACT.

J G Davis

175 343

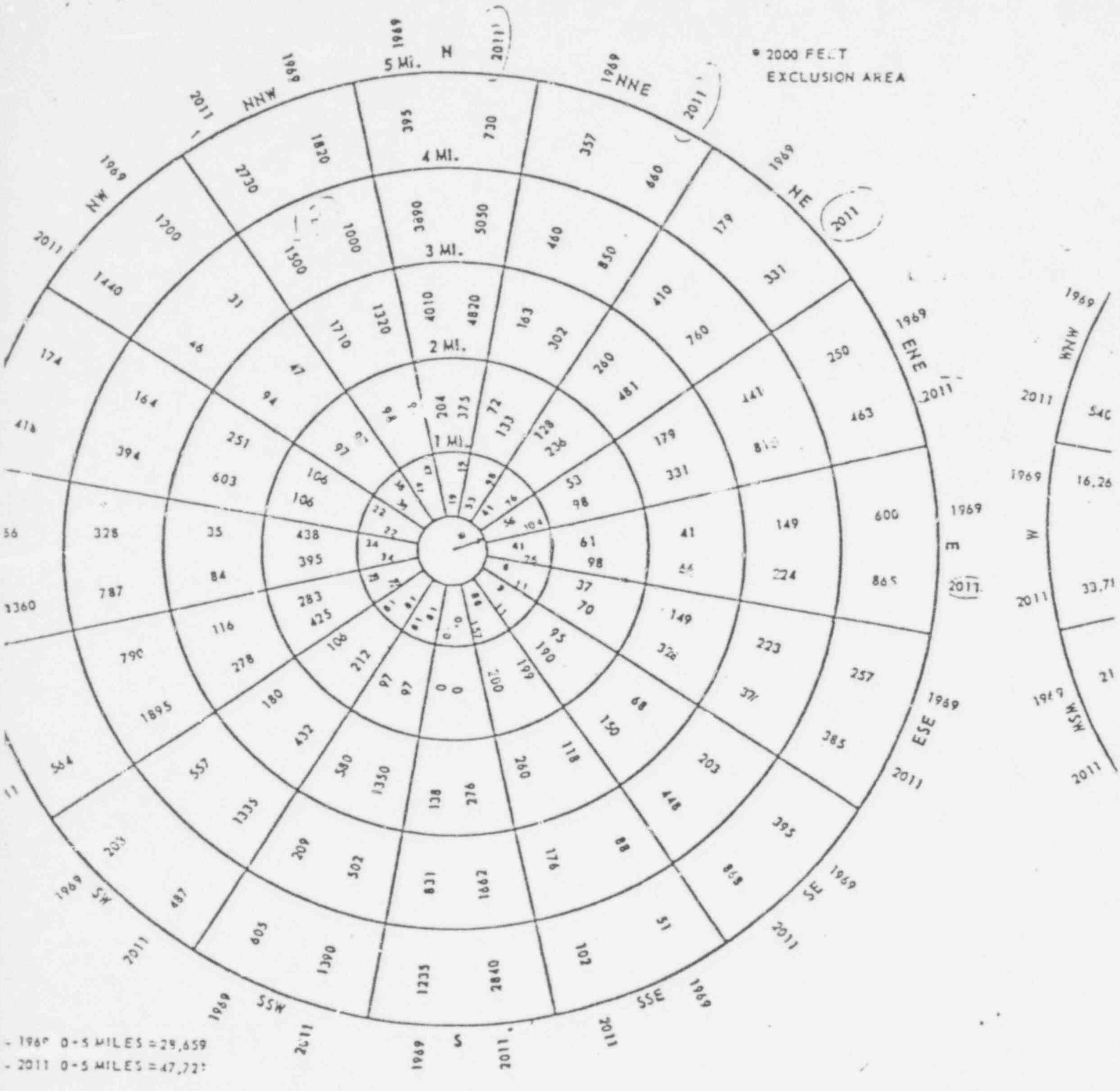
FDAA 4/2/79

NRC requests assistance in estimating the people within 5 mile radius of the Three Mile Island Plant site. Recognizing that some voluntary evacuation has taken place, it would be useful to have an approximation of the people remaining. The estimates need not be in precise measure. Due to the factors involved accuracy to a factor of 2 is adequate.

The population distribution, we are using is attached.

175 344

* 2000 FEET EXCLUSION AREA



- 1969 0-5 MILES = 29,659
- 2011 0-5 MILES = 47,721

175 345

DATE 4-4-79

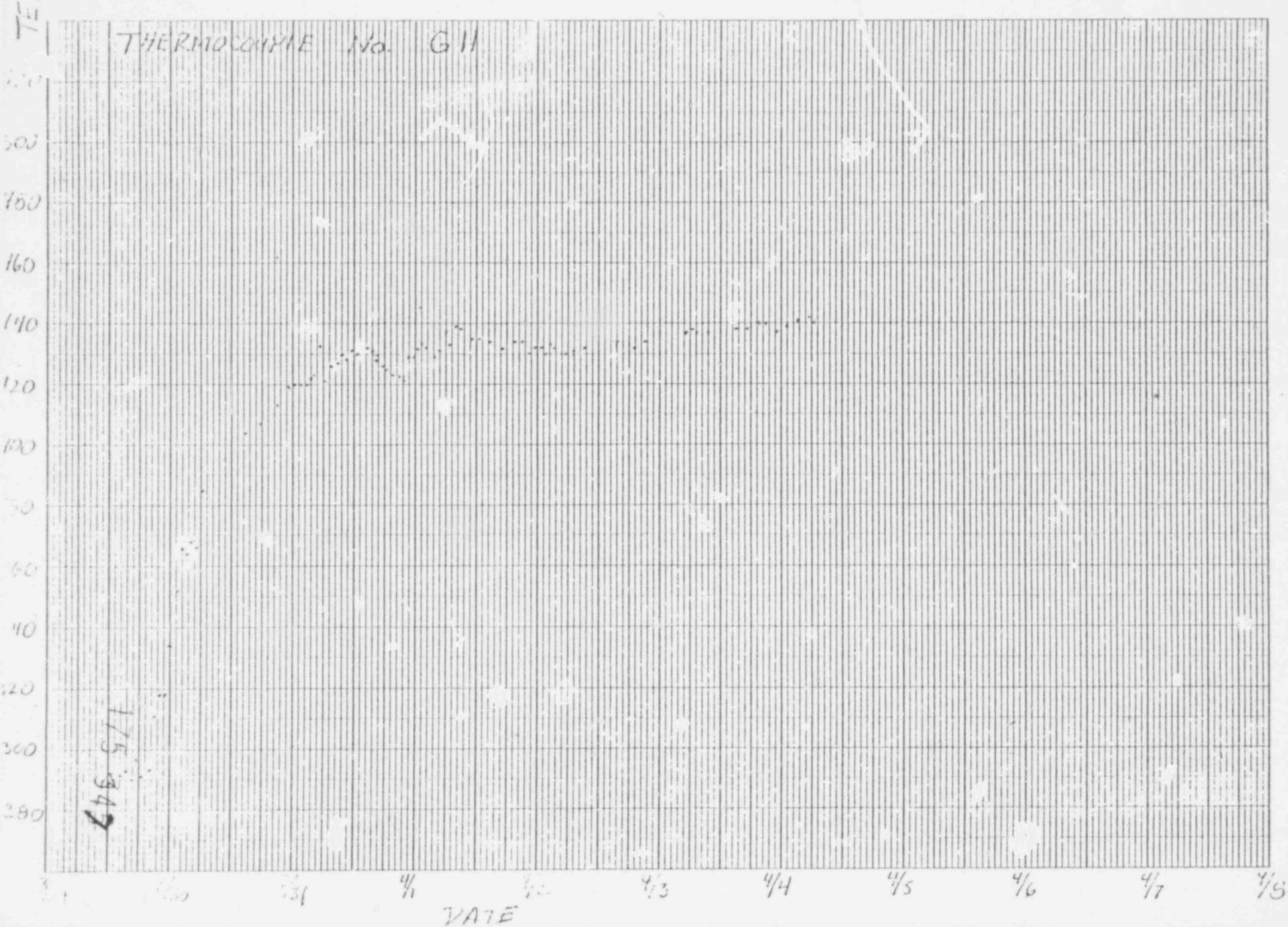
TIME 0628 HOURS

CONROLL ROOM	<.01 MR/HR
RC EVAP CONTROL PANEL	40
FUEL HANDLING BRIDGE	15
ANX BLDG ACCESS CORRIDOR COL	40
RX BLDG PURGE UNIT AREA	40
CABLE ROOM <	<.1
MAKEUP TANK AREA	3000
WASTE DISPOSAL STORAGE AREA	30
RX BLDG COOLING PUMP AREA	25
INTERWELL COOLING PUMP STORAGE AREA	40
~~CONTROL + SERVICE BLDG CORRIDOR~~	~~0.25~~
FUEL HANDLING BLDG EXHAUST	150

175 3416

K·E 10 X 10 TO ½ INCH 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.

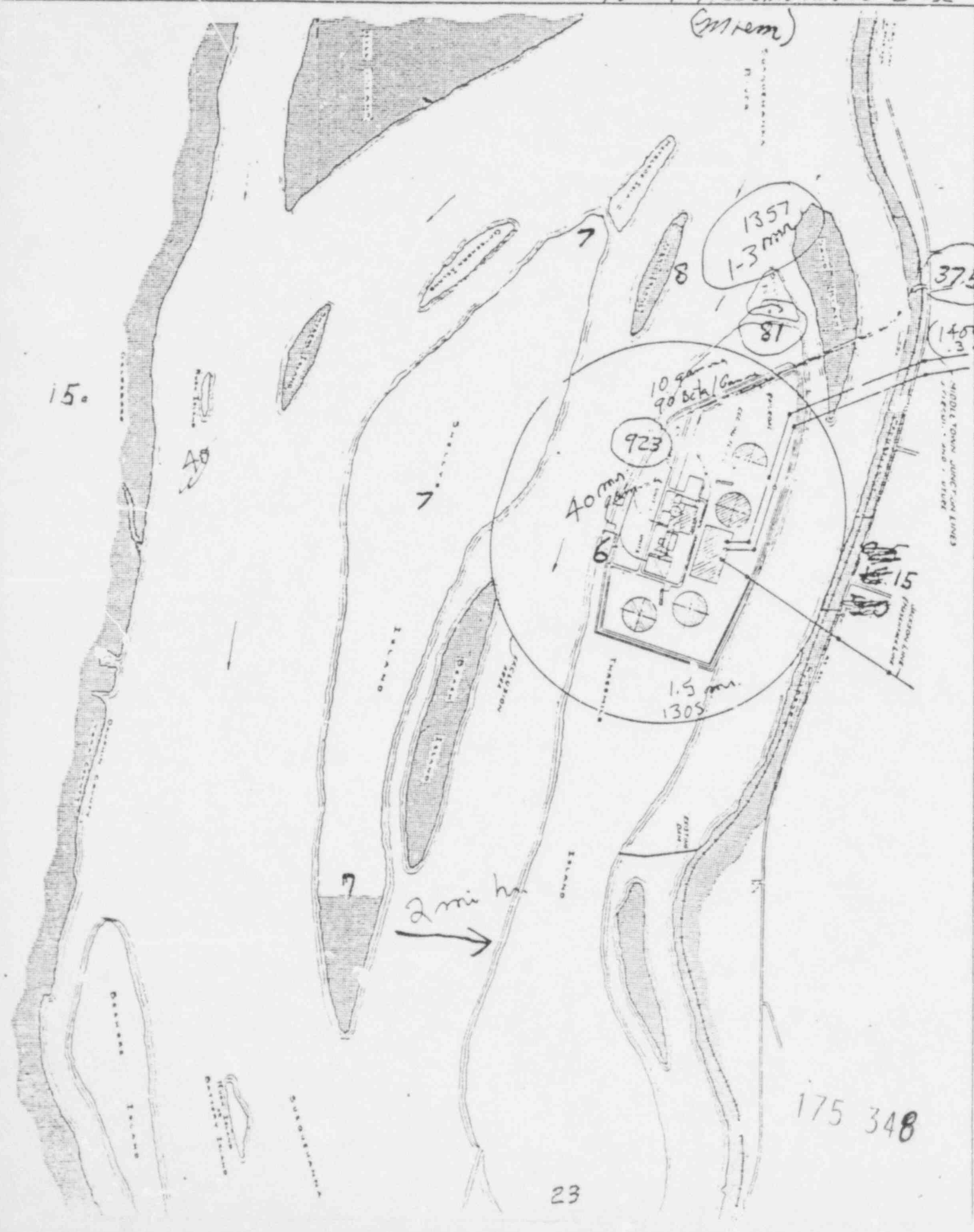
46 1531



TLD data pulled at 1300-1500 3/29/79 1

Background is ~22 mrem

Total Accumulated Dose (Mrem)



4/7/79

22:30 - 23:00

Release Survey

Wind at 22:10 = from 320°
Speed = 0-5 mph

at South end of Island = 6 in (open) = 0.3 mr/hr
(on highway 441 " closed = 0.04 "

Vent Monitor
particulate
Before = 3.0E+04 cpm
during = 4.0E+04 "

Iodine
before = 1.5 E+05 cpm
during = 1.6 E+05 cpm

Noble Gas
before = 9.0 E+05 cpm
After = off scale (at E+06 cpm)

ARMS Team in air at 22:47

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 6 1979

NOTE TO: XOOS Staff

FROM: D. Thompson

SUBJECT: WEEKEND MANNING OF OPERATIONS CENTER

XOOS MANNING

4/7/79	IRACT Support	EMT
0001-0800	Ward Paulus	Crews
0800-1600	Weiss Gower	Jordan
1600-2400	Hegner	Thompson
4/8/79		
0001-0800	Baci	Crews
0800-1600	Hegner	Thompson
1600-2400	Weiss	Jordan
4/9/79		
0001-0800	Baci	Crews

Dudley Thompson
D. Thompson
Executive Office for Operations
Support

175 350

NRR - SCHEDULE

4/6	B. Grimes	til 1900	
	D. Davis	1900	2400
4/7	D. Davis	0000	0200
	G. Lainis	0200	1200
	B. Grimes	1200	2400
4/8	D. Davis	0000	1200
	G. Lainas	1200	2400

175 3~~49~~ 51

GOVERNORS

PA The Honorable Richard Thornburgh
Governor of Pennsylvania
State Capitol
Harrisburg, PA 17120
~~(717) 787-2500~~

NY The Honorable Hugh L. Carey
Governor of New York
State Capitol
Albany, NY 12224
~~(518) 474-8390~~

NJ The Honorable Brendan T. Byrne
Governor of New Jersey
State House
Trenton, NJ 08625
~~(609) 292-6000~~

MD The Honorable Harry Hughes
Governor of Maryland
State House
Annapolis, MD ~~210~~ 21404
~~(301) 267-5901~~

WVa The Honorable John D. Rockefeller IV
Governor of West Virginia
State Capitol
Charleston, WV 25305
~~(304) 348-2000~~

175 352

State Health

cc's

PA Mr. Thomas M. Gerusky, Director,
~~Director~~ Bureau of Radiation Protection
Department of Environmental Resources
P.O. Box 2063
Harrisburg, PA 17120

NY Mr. Sherwood Davies, Director
Bureau of Radiological Health
State Department of Health
Empire State Plaza
Tower Building
Albany, NY 12237

NJ Mr. Eugene Fisher, Acting Chief
Bureau of Radiation Protection
Division of Environmental Quality
Depart. of Environmental Protection
380 Scotch Road
Trenton, NJ 08628

MD Mr. Robert E. Corcoran, Chief
Division of Radiation Control
Department of Health and Mental Hygiene
O'Conor Office Building
201 West Preston St.
Baltimore, MD 21201

WV Mr. William H. Aaroe, Director
Bureau of Industrial Hygiene
Radiological Health [illegible]
151 11th Ave
South Charleston, WV 25303

175 353

EMT/XOOS - OPERATIONS STATUS OFFICER

	Tu 4/10	W 4/11	Th 4/12	Fr 4/13	Sa 4/14	Su 4/15
6am 2pm	Paulus* Hegner	Paulus Hegner	Paulus Hegner	Paulus Hegner	Paulus	Gower
2pm 10pm	Weiss	Weiss	Weiss	Weiss	Ward	Ward

	Mo 4/16	Tu 4/17	Wd 4/18	Th 4/20	Fr 4/21	Sa 4/22
6am 2pm	Gower	Gower				
2pm 10pm	Hegner	Hegner				

*Paulus - 6am to 8am; Hegner 8am to 2pm

Duties of Operations Status Officer:

1. AM shift pulls PN together, obtains reviews, appropriate concurrences and sees that it is dispatched promptly.

2. Provides TMI-2 status information as requested from legitimate outside inquiries, other NRC offices and foreign sources if arranged thru IP. All press inquiries for status information are to be referred to PA.

3. Maintains continuity of taping; assures tapes are changed and that used tapes are properly stored.

4. Will take action, if necessary, to recall EMT & IRACT back into full operation.

5. Expedite completion of high priority items.

6. Coordinate requests for support from other Federal agencies

Operation Status Officer calls are to be handled by IRACT from 10pm to 6am.

175 3514

Marge Smith ... TV

FED's

DCPA

✓ Mr. Bardyl R. Tirana, Director

✓ Defense Civil Preparedness Agency

The Pentagon

Washington, D.C. 20301

202-697-4484

DOA

~~The~~ Honorable

✓ ~~Mr.~~ Bob Bergland, Secretary of Agriculture — Dear Mr. Secretary

~~Fourteenth St. and Independence Ave. S.W.~~

Washington, D.C. 20250

202-655-4000

HEW

The Honorable — Dear Mr. Secretary

✓ ~~Mr.~~ Joseph A Califano, Jr.

Secretary of HEW

200 Independence Ave. +

Washington, D.C. 20201

LABOR

The Honorable

~~Mr.~~ Ray Marshall, Secretary of Labor — Dear Mr. Secretary

✓ 200 Constitution Ave, NW +

Washington, D.C. 20210

202-523-8165

ENERGY

The Honorable — Dear Mr. Secy

✓ ~~Mr.~~ James R. Schlesinger, Secretary of Energy

Washington, D.C. 20545

202-376-4000

See EPA on reverse side

175 355

4/10/79

Dudley ?

Somehow we sent out the attached "FINAL VERSION" of the PRELIMINARY DESCRIPTION OF EVENTS AT THREE MILE ...

I got a call from TMI site (Warnick) on this.

SO, I sent the corrected "final - latest version" to TMI site - This version is also attached.

To prevent confusion I have dated (April 10, 1979) the latest revision.

We also sent this latest revision (several extra copies attached) to the Regions. Linda U. should check, and those who did not receive over night 4/10/79 copies should be sent a copy on 4/11/79. (Mike Wilber should know status as of 0800 4/11/79).

Hope this is clear.

Jess

PS: Master copy of document is in "Green book" _

175 3516

FINAL VERSION

PRELIMINARY

DESCRIPTION OF EVENTS
AT THE THREE MILE ISLAND 2
FACILITY ACCIDENT

Revised cee 4/10/79 revision version

The following is a summary of the significant events that occurred at the Three Mile Island No. 2 nuclear facility on March 28, 1979, and thereafter. Attached is a detailed chronology of these events listed with the times they each occurred.

At about 4:00 am on March 28, 1979, the secondary (nonnuclear) cooling system of the Three Mile Island facility suffered a malfunction. This system normally pumps water through the plant's steam generators where the water turns to steam which then flows to turn a turbine generator. The water is then condensed back to water, is pumped by a condensate pump through a clean up system, through a feedwater pump, and finally back to the steam generators, and continually flows around this loop.

A malfunction in the main feedwater system caused the feedwater pumps to turn off (trip), which in turn caused the turbine-generator to turn off and stop generating electricity. Since the steam generators were not removing heat due to the stoppage of feedwater flow, the reactor coolant system pressure increased and the pressurizer relief valve opened to reduce reactor pressure. Immediately, the reactor turned off by the rapid insertion of the plant's control rods (scrammed) as designed and the nuclear chain reaction stopped leaving behind only residual, or decay, heat. These events all occurred within the first 30 seconds following the event.

175 3517

Up to this point, this sequence is normal and the auxiliary feedwater system should startup and deliver secondary coolant to the plant's two steam generators to remove heat. In addition, the pressurizer relief valve should close as reactor pressure decreases.

All three of the auxiliary feedwater pumps started but were unable to deliver flow because their flow paths were blocked by closed valves. In addition, the pressurizer relief valve failed to close and therefore allowed the reactor coolant system pressure to continue to decrease.

As the reactor pressure reached a preset value (1600 psi), the plant's Emergency Core Cooling System (ECCS) started as designed and began to inject cold water into the reactor. It is at this point that an indication of a rapidly rising pressurizer level apparently led the plant operators to terminate the ECCS flow. At this point the Three Mile Island incident had been underway for 11-12 minutes.

Between about 1 and 2 hours into the transient, the operators turned off the four large pumps which circulate the reactor coolant through the reactor. It is following this action that we believe the severe damage to the nuclear fuel began. For the next several hours there was a very large temperature difference across the nuclear core indicating little flow of coolant through the core.

During this several hour period, when severe fuel damage is occurring, primary coolant from the reactor primary coolant system was being dumped onto the reactor containment floor from flow out of the pressurizer relief valve and through the drain tank. This coolant, which contained radioactivity, was partially pumped from the reactor containment building floor to tanks in the auxiliary building. The tanks overflowed permitting radioactivity to be vented from the auxiliary building. This situation lasted until about 9:00 am when the reactor containment was sealed (isolated).

During this time, from about 6:00 am until 8:00 pm, the licensee tried to depressurize the reactor coolant system sufficiently to be able to turn on the residual heat removal system. Since his attempts failed, it was decided to repressurize the system.

After repressurization, one of the main reactor coolant pumps was restarted and flow through the reactor core was re-established.

Since feedwater was being provided to the steam generator, heat was being removed and the reactor system was slowly cooled.

Reactor cooling has essentially been in this mode since that time.

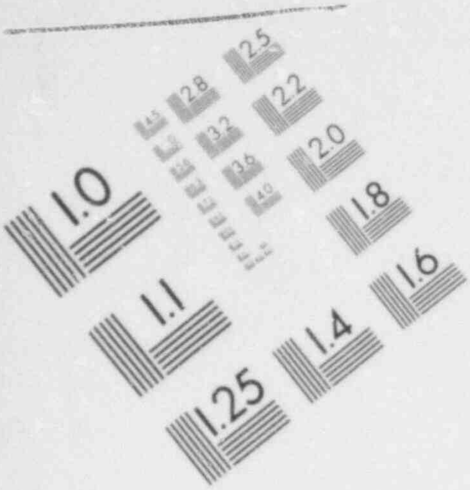
PRELIMINARY CHRONOLOGY OF THE MARCH 28, 1979 ACCIDENT AT THREE MILE ISLAND

Time (approximate)	Discussion of Events
Before 4:00 am	TMI operator working on Feedwater System.
4:00 am	The loss of all (main and auxiliary) feedwater flow occurred while the reactor was operating at 98% power. The transient was initiated by a loss of condensate pumps. The turbine tripped.
3-6 sec later	An electromatic relief valve opened to relieve pressure in the RCS* (2255 psi).
9-12 sec later	The Reactor tripped on high RCS pressure (2355 psi) to terminate the nuclear reactor and reduce power generation to decay heat alone.
12-15 sec later	The RCS pressure decayed to the point (2205 psi) where the relief valve should have reclosed. The RCS continued to depressurize for about the next two hours.
15 sec later	The temperature in the RCS hot leg peaks at about 610°F with a pressure of about 2150 psi.
30 sec later	The auxiliary feedwater pumps in both safety trains (1 turbine driven pump and 2 electrically driven pumps) were started and were running at pressure ready to inject water into the steam generators and remove the residual heat produced in the reactor core. No water was injected since the discharge valves were closed.

*Throughout, RCS denotes "reactor coolant system."

Time (approximate)	Discussion of Events
4:01 am	The pressurizer level indication began to rise rapidly. The steam generators, A and B, had low levels of water and were drying out.
4:02 am	The ECCS was initiated as the RCS pressure decreased to 1600 psi.
4:04-4:11 am	The pressurizer level indication went offscale high and the operator manually tripped the first HPI pumps at about 4:04:30 and the second at about 4:10:30.
4:06 am	Water in the RCS flashed to steam as the pressure bottoms out at 1350 psi. The hog leg temperature was about 585°F.
4:07-4:08 am	The Reactor building sump pump came on.
4:08 am	The operator opened the valves at the discharge of the auxiliary feedwater pump allowing water to be injected into the steam generators.
4:11-4:12 am	The operator restarted the ECCS to inject water into the RCS to control pressurizer level.
4:11 am	The pressurizer level indication comes back on scale.
4:15 am	The RC Drain (Quench) tank rupture disk blew at 190 psig due to continued discharge of the relief valve that had failed to open.
4:20-5:00 am	The RCS parameters stabilized at a saturated condition of about 1015 psi and 550°F.
5:15 am	The operator tripped both RC pumps in Loop B.
5:40 am	The operator tripped both RC pumps in Loop A.

Time (approximate)	Discussion of Events
5:45-6 am	The reactor core began a heatup transient. The RCS hot leg temperature went offscale at 620 degrees F within 14 minutes and the cold leg temperature dropped to near the temperature o, high pressure injection water (150 degrees F).
6:20 am	The failed open relief valve was isolated by the operator by closing a block valve. The operator also isolated steam generator B to prevent leakage of radioactive secondary water from leaking S.B. tubes.
7:00 am	The RCS pressure had increased to 2150 psi and the relief valve was opened to relieve RCS pressure.
7:15 am	A pressure spike of 5 psig occurred in the RC drain tank due to steam from the relief valve.
7:45 am	A pressure spike of 11 psig occurred in the RC drain tank and the pressure in the RCS was at 1750 psi.
9:00 am	The pressure in containment peaked at 4.5 psig.
9:00-11:00 am	The RCS pressure increased from 1250 psi to 2100 psi.
11:30 am	The operator opened the pressurizer relief valve to depressurize the RCS in an attempt to initiate RHR cooling at 400 psi.
12:00 am - 1:00 pm	The RCS pressure decreased to about 500 psi and the core flooding tanks partially discharged. The relief capacity was not sufficient to vent enough to reach 400 psi.
2:00 pm	The pressure in the containment spikes at 28 psig causing containment sprays to be initiated. The operator stopped the spray pumps after about 2 minutes of operation.



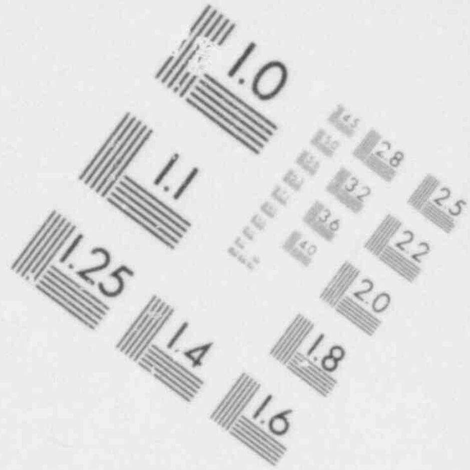
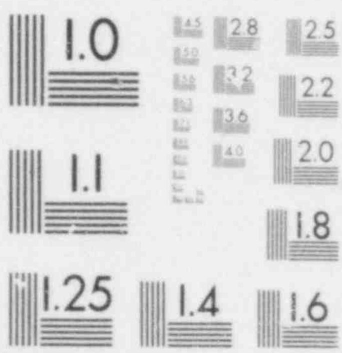
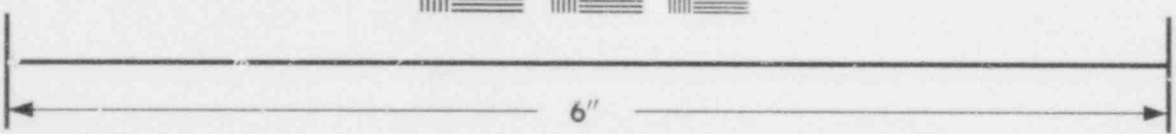
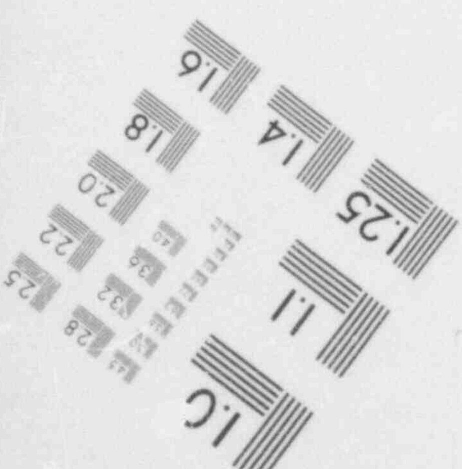


IMAGE EVALUATION TEST TARGET (MT-3)







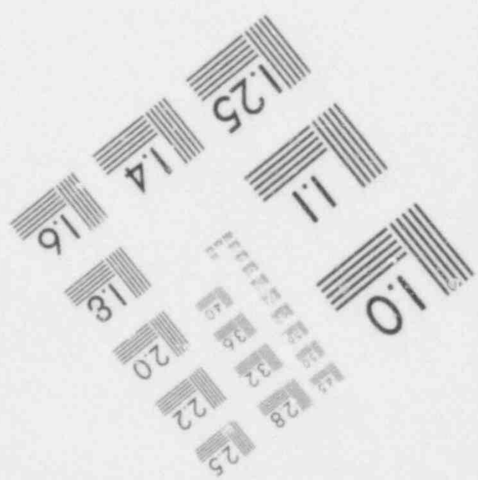
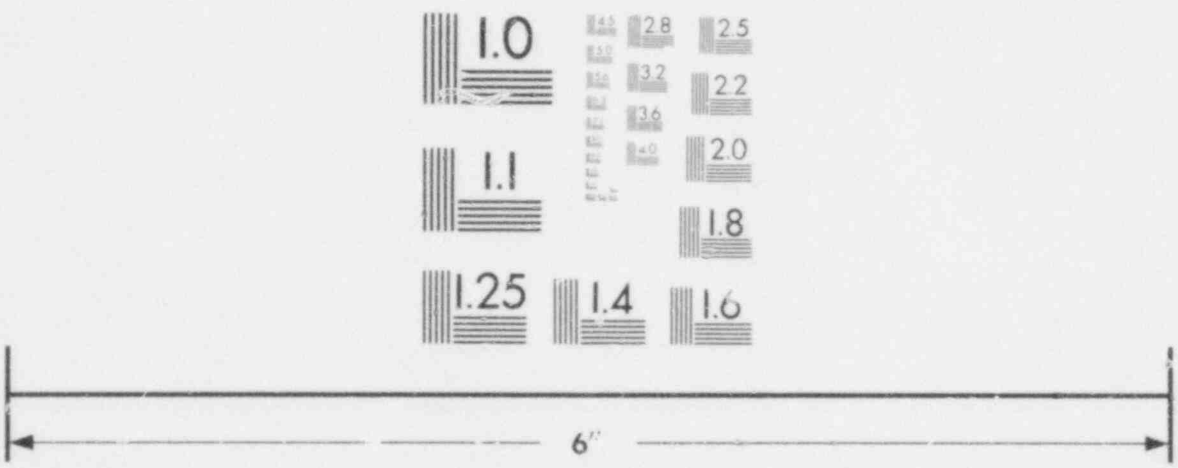


IMAGE EVALUATION
TEST TARGET (MT-3)



Time (approximate)	Discussion of Events
5:30 pm	The pressurizer relief valve was closed in order to repressurize the reactor coolant system.
5:30 - 8 pm	The RCS pressure increased from 650 psi to 2300 psi.
8 pm	RC pump in Loop A was started at which time the hot leg temperature decreased to about 560 degrees F and the cold leg temperature increased to 400 degrees F, indicating flow through the steam generator. Thereafter, the reactor was being cooled by reestablishing condenser vacuum and steaming to the condenser by steam generator A with the RCS cooled to about 280 degrees F and 1000 psi.

March 29

The RCS temperature and pressure was stablized at about 280 degrees F and 840 to 1020 psi. The maximum reading on the incore thermocouples was 612°F, but several were not with range for computer readouts (printing "?") which was subsequently found to indicate greater than 700 degrees F.

March 30

The RCS temperature and pressure was stable at nearly 280 degrees F and between about 1000 to 1060 psi. Several incore thermocouples were beyond the range for computer readout, the maximum indicated reading was 659 degrees F. The NRR staff estimated the bubble size in RCS to be about 1200 ft^3 and requested the licensee to refine their calculation of the bubble size.

March 31

The RCS temperature and pressure remained stable at about 280°F and 1000 psi. Slight drop in pressurizer level 251-191". Temperatures in the core as measured from the incore thermocouples were gradually decreasing (maximum indicated about 500°F). The hydrogen recombiner was in an operable status but additional shielding was needed and was being obtained. Two samples of containment atmosphere were analyzed which showed a hydrogen concentration of 1.7% and 1.0%. Licensee calculated bubble size to be about 620 ft^3 @ 875 psig.

April 1

No substantial change in RCS temperature and pressure
Incore thermocouples continue to show decreased trend.

Licensee continued hookup of hydrogen recombiners and addition of shielding. Licensee calculated valves of bubble size varied. Containment air samples indicate 2.3% hydrogen.

April 2

Reactor pressure stable at about 1000 psi. Incore thermocouples continued to show a decrease with all measurements below 475°F. Inlet and outlet temperatures were still about 280°F. One hydrogen recombiner was put in operation.

Analysis indicated that the oxygen generation rate in reactor less than originally estimated. Measurements indicated that the bubble was being significantly reduced.

April 3

Reactor pressure and temperature stable at 1000 psi and 280°F, respectively. Thermocouple readings analyzed- maximum 477°F, only 3 thermocouples were above 400°F. Gas bubble size much reduced. Containment about 1.9% hydrogen. One pressurizer level indicator failed.

April 4

Reactor pressure and temperature stable at 1000 psi and 280°F, respectively. Thermocouple maximum temperature was 466°F. Gas bubble size decreasing. Vent valve on pressurizer intermittently opened and degassing continues through letdown system.

April 5

Reactor pressure and temperature stable at 1000 psi and 280°F, respectively. Maximum thermocouple reading is 462°F. Pressurizer level responding normally to pressure changes indicating a completely full system.

Containment atmosphere indicates 2% hydrogen. One recombiner operating, one in standby. Pressurizer vented to containment about 15 minutes every 6-8 hours.

April 6

Reactor pressure stable at about 1000 psi and temperature about 285°F.

At approximately 1:25 pm, reactor coolant pump 1A tripped and reactor coolant pump 2A was started within about 2 minutes. Shift in thermocouple readings. The three thermocouples previously reading about 400°F are presently reading between 285°F and 315°F. Central thermocouple increased from 375°F to 425°F and is the only one reading about 400°F.

Containment measurements indicate about 2% hydrogen. Pump-back system for pumping waste gas decay tank volume to containment began.

April 7

Reactor pressure and temperature stable at about 1000 psi and 280°F, respectively.

At about 8 pm, the licensee began to slowly lower reactor system pressure. The slow decrease will end when reactor pressure reaches 500 psi. This is a step toward cold shutdown and includes degasification to prevent bubble formation as pressure and temperature decreases.

Hydrogen concentration in the containment is about 1.9%.

PRELIMINARY

April 10, 1979

DESCRIPTION OF EVENTS
AT THE THREE MILE ISLAND 2
FACILITY ACCIDENT

The following is a summary of the significant events that occurred at the Three Mile Island No. 2 nuclear facility on March 28, 1979, and thereafter. Attached is a detailed chronology of these events listed with the times they each occurred.

At about 4:00 am on March 28, 1979, the secondary (nonnuclear) cooling system of the Three Mile Island facility suffered a malfunction. This system normally pumps water through the plant's steam generators where the water turns to steam which then flows to turn a turbine generator. The water is then condensed back to water, is pumped by a condensate pump through a clean up system, through a feedwater pump, and finally back to the steam generators, and continually flows around this loop.

A malfunction in the main feedwater system caused the feedwater pumps to turn off (trip), which in turn caused the turbine-generator to turn off and stop generating electricity. Since the steam generators were not removing heat due to the stoppage of feedwater flow, the reactor coolant system pressure increased and the pressurizer relief valve opened to reduce reactor pressure. Immediately, the reactor turned off by the rapid insertion of the plant's control rods (scrammed) as designed and the nuclear chain reaction stopped leaving behind only residual, or decay, heat. These events all occurred within the first 30 seconds of the accident.

176 004

Up to this point, this sequence is normal and the auxiliary feedwater system should startup and deliver secondary coolant to the plant's two steam generators to remove heat. In addition, the pressurizer relief valve should close as reactor pressure decreases.

All three of the auxiliary feedwater pumps started but were unable to deliver flow because their flow paths were blocked by closed valves. In addition, the pressurizer relief valve failed to close and therefore allowed the reactor coolant system pressure to continue to decrease.

As the reactor pressure reached a preset value (1600 psi), the plant's Emergency Core Cooling System (ECCS) started as designed and began to inject cold water into the reactor about 2 minutes after the event started. An indication of a rapidly rising pressurizer level apparently led the plant operators to terminate the ECCS flow. At this point the Three Mile Island accident had been underway for 10-11 minutes.

Between about 1 and 2 hours into the accident, the operators turned off the four large pumps which circulate the reactor coolant through the reactor. It is following this action that we believe the severe damage to the nuclear fuel began. For the next several hours there was a very large temperature difference across the nuclear core indicating little flow of coolant through the core.

During this several hour period, when severe fuel damage was occurring, primary coolant from the reactor primary coolant system was being dumped onto the reactor containment floor from flow out of the pressurizer relief valve and through the drain tank. This coolant, which contained radioactivity, was partially pumped from the reactor containment building floor to tanks in the auxiliary building. The tanks overflowed permitting radioactivity to be vented from the auxiliary building. This situation lasted until about 9:00 am when the reactor containment was sealed (isolated).

From about 6:00 a.m. until 3:00 p.m., the licensee tried to depressurize the reactor coolant system sufficiently to be able to turn on the residual heat removal system. Since his attempts failed, it was decided to repressurize the system.

After repressurization, one of the main reactor coolant pumps was restarted and flow through the reactor core was re-established.

Since feedwater was being provided to the steam generator, heat was being removed and the reactor system was slowly cooled. Core temperatures decreased over the next several days and stabilized. Reactor cooling has essentially been in this mode since that time.

PRELIMINARY CHRONOLOGY OF

THE MARCH 28, 1979 ACCIDENT

AT THREE MILE ISLAND

Time (approximate)	Discussion of Events
Before 4:00 a.m.	TMI operator working on Feedwater System
4:00 a.m.	The loss of all (main and auxiliary) feedwater flow occurred while the reactor was operating at 98% power. The transient was initiated by a loss of condensate pumps. The turbine tripped.
3-6 sec later	An electromatic relief valve opened to relieve pressure in the RCS* (2255 psi).
9 sec after start of event	The Reactor tripped on high RCS pressure (2355 psi) to terminate the nuclear reactor and reduce power generation to decay heat alone.
12-15 sec after start of event	The RCS pressure decayed to the point (2205 psi) where the relief valve should have reclosed. The RCS continued to depressurize for about the next two hours.
14 sec after start of event	The auxiliary feedwater pumps in both safety trains (1 turbine driven pump and 2 electrically driven pumps) were started and were running at pressure ready to inject water into the steam generators and remove the residual heat produced in the reactor core. No water was injected since the discharge valves were closed.
15 sec after start of event	The temperature in the RCS hot leg peaks at about 610°F with a pressure of about 2150 psi.

*Throughout, RCS denotes "reactor coolant system."

Time (approximate)	Discussion of Events
4:01 a.m.	The pressurizer level indication began to rise rapidly. The steam generators, A and B, had low levels of water and were drying out.
4:02 a.m.	The ECCS was initiated as the RCS pressure decreased to 1600 psi.
4:06 a.m.	The pressurizer level indication went offscale high.
4:04-4:11 a.m.	The operator manually tripped the first HPI pumps at about 4:05:15 and the second at about 4:11:01.
4:06 a.m.	Water in the RCS flashed to steam as the pressure bottoms out at 1350 psi. The hog leg temperature was about 584°F.
4:07-4:08 a.m.	The Reactor building sump pump came on.
4:08 a.m.	The operator opened the valves at the discharge of the auxiliary feedwater pump allowing water to be injected into the steam generators.
4:12-4:13 a.m.	The operator restarted the ECCS to inject water into the RCS to control pressurizer level.
4:11 a.m.	The pressurizer level indication comes back on scale.
4:15 a.m.	The RC Drain (Quench) tank rupture disk blew at 190 psig due to continued discharge of the relief valve that had failed to close.
4:20-5:00 a.m.	The RCS parameters stabilized at a saturated condition of about 1015 psi and 550°F.
5:14 a.m.	The operator tripped both RC pumps in Loop B and one pump in Loop A.
5:27 a.m.	Operator isolated "B" Steam generator.
5:41 a.m.	The operator tripped the second RC pump in Loop A.

Time (approximate)	Discussion of Events
5:45-6 a.m.	The reactor core began a heatup transient. The RCS hot leg temperature went offscale at 620 degrees F within 14 minutes and the cold leg temperature dropped to near the temperature of high pressure injection water (150 degrees F).
6:20 a.m.	The failed open relief valve was isolated by the operator by closing a block valve.
7:00 a.m.	The RCS pressure had increased to 2150 psi and the relief valve was opened to relief RCS pressure.
7:15 a.m.	A pressure spike of 5 psig occurred in the RC drain tank due to steam from the relief valve.
7:45 a.m.	A pressure spike of 11 psig occurred in the RC drain tank and the pressure in the RCS was at 1750 psi.
9:00 a.m.	The pressure in containment peaked at 4.5 psig.
9:00-11:00 a.m.	The RCS pressure increased from 1250 psi to 2100 psi.
11:30 a.m.	The operator opened the pressurizer relief valve to depressurize the RCS in an attempt to initiate RHR cooling at 400 psi.
12:00 a.m.-1:00 p.m.	The RCS pressure decreased to about 500 psi and the core flooding tanks partially discharged. The relief capacity was not sufficient to vent enough to reach 400 psi.
2:00 p.m.	The pressure in the containment spikes at 28 psig causing containment sprays to be initiated. The operator stopped the spray pumps after about 2 minutes of operation.

Time (approximate)	Discussion of Events
5:30 pm	The pressurizer relief valve was closed in order to repressurize the reactor coolant system.
5:30 - 8 pm	The RCS pressure increased from 650 psi to 2300 psi.
8 pm	RC pump in Loop A was started at which time the hot leg temperature decreased to about 560 degrees F and the cold leg temperature increased to 400 degrees F, indicating flow through the steam generator. Thereafter, the reactor was being cooled by reestablishing condenser vacuum and steaming to the condenser by steam generator A with the RCS cooled to about 280 degrees F and 1000 psi.

March 29

The RCS temperature and pressure was stabilized at about 280 degrees F and 840 to 1020 psi. The maximum reading on the incore thermocouples was 612°F, but several thermocouples were not within range for computer readouts, i.e., the temperatures were higher than about 700 degrees F.

March 30

The RCS temperature and pressure were stable at about 280 degrees F and about 1000 to 1060 psi. Several incore thermocouples were beyond the range for computer readout, the maximum indicated reading was 659 degrees F. The licensee estimated the size of a bubble of non-condensible gas in the RCS to be about 1200 ft^3 at 875 psig.

March 31

The RCS temperature and pressure remained stable at about 280°F and 1000 psi. Slight drop in pressurizer level 251-191". Temperatures in the core as measured from the incore thermocouples were gradually decreasing (maximum indicated about 500°F). The hydrogen recombiner was in an operable status but additional shielding was needed and was being obtained. Two samples of containment atmosphere were analyzed which showed a hydrogen concentration of 1.7% and 1.0%. Licensee estimated the bubble size to be about 620 ft^3 @ 875 psig.

April 1

No substantial change in RCS temperature and pressure
Incore thermocouples continue to show decreased trend.

Licensee continued hookup of hydrogen recombiners and addition of shielding. Licensee calculated valves of bubble size varied. Containment air samples indicate 2.3% hydrogen.

April 2

Reactor pressure stable at about 1000 psi. Incore thermocouples continued to show a decrease with all measurements below 475°F. Inlet and outlet temperatures were still about 280°F. One hydrogen recombiner was put in operation to decrease the hydrogen gas concentration in the containment building.

Analysis indicated that the oxygen generation rate in reactor less than originally estimated. Measurements indicated that the bubble was being significantly reduced by degassing operations.

April 3

Reactor pressure and temperature stable at 1000 psi and 280°F, respectively. Thermocouple readings analyzed- maximum 477°F, only 3 thermocouples were above 400°F. Gas bubble size much reduced. Containment about 1.9% hydrogen. One pressurizer level indicator failed.

April 4

Reactor pressure and temperature stable at 1000 psi and 280°F, respectively. Thermocouple maximum temperature was 466°F. Gas bubble size decreasing. Vent valve on pressurizer intermittently opened and degassing continues through letdown system.

April 5

Reactor pressure and temperature stable at 1000 psi and 280°F, respectively. Maximum thermocouple reading is 462°F. Pressurizer level responding normally to pressure changes indicating a completely full system.

Containment atmosphere indicates 2% hydrogen. One recombiner operating, one in standby. Pressurizer vented to containment about 15 minutes every 6-8 hours.

April 6

Reactor pressure stable at about 1000 psi and temperature about 285°F.

At approximately 1:25 pm, reactor coolant pump 1A tripped and reactor coolant pump [illegible] was started within about 2 minutes. Shift in thermocouple readings. The three thermocouples previously reading about 400°F are presently reading between 285°F and 315°F. Central thermocouple increased from 375°F to 425°F and is the only one now reading above 400°F.

Containment measurements indicate about 2% hydrogen. Pump-back system for pumping waste gas decay tank volume to containment began.

April 7

Reactor pressure and temperature stable at about 1000 psi and 280°F, respectively.

At about 8:00 pm, the licensee began to slowly lower reactor system pressure in increments of 50 psig. The slow decrease ended when reactor pressure reached 500 psi. This intentional pressure reduction expanded gasses trapped in control rod drive housings above the vessel head so that they could be dissolved or entrained and then be gassed through pressurizer venting and letdown at higher pressures. This degasification process is designed to prevent bubble formation as pressure and temperature decrease during the placement of the reactor cooling system in a long term, shutdown cooling mode.

Hydrogen concentration in the containment is about 1.7%.

4/7/79.

Dudley,

Following are items of interest, underway pending, etc:

1. Met.Ed. needs charcoal filters @ TMI in anticipation of Iodine breakthrough of present Aux Bldg filters.

Filters (4 @ 26,000 lbs each) have been located in Richland, Wa - WPPSS project -

May require NRC cost authorization for Air Force to fly equip, per FDAA. Twenty-two (22) AF C-5A or equivalent aircraft -

Filters need @ TMI site 8:00 hr Monday 4/9/79. (See ~~attachment~~).

2. Visit by Japanese delegation Today - Sat. (See attachment).

3. Site reported borating TMI-2 to ~~4000~~ 3000 ppm - Reason ~~unknown~~ ~~as yet - other than it~~ is B&W recommendation prior to chg in cooling mode -

176 013

4. TI 2595/2 dispatched to
Regs 1, 4 & 5.

Regs. II & III to call in today
when ready to receive —

[illegible]

176 014

In PUD-79-67E, ~~it was reported that licensee~~ ~~TLD~~ thermoluminescent dosimeter (TLD) data for seventeen fixed positions was reported with the highest reading being 81 mr at a 0.4 miles north of the reactor. The licensee had 18 fixed locations with the highest reading being 921 mr at 0.2 miles NNW, ~~[illegible]~~ on Three Mile Island. This is a correction.

These TLDs were in place until 3pm on March 29. ~~during the first days~~

176 015

OPERATIONS CENTER SHIFT ASSIGNMENTS

OPS SUPPORT	8 a.m. - 8 p.m.	8 p.m. - 8 a.m.
	Weiss	Paulus
	Hegner	Ward
	Jackson	F. Cox

HP	2 p.m. - 2 a.m.	2 a.m. - 2 p.m.
	Sniezek	Higginbotham
	Flack	Cunningham
	L. Cohen - Days	

6504-1493

REACTOR OPERATIONS	2 a.m. - 2 p.m. A	2 p.m. - 2 a.m. B
	N. Moseley	M. Howard
	H. Thornburg	E. Jordan
	R. Woodruff	K. Whitt
	M. Wilbur	P. Harmon
	S. Showe	A. Oxfurth
	E. Blackwood	D. Kirkpatrick
	P. McKee	C. DeBevec
	S. Bryan	

STATE PROGRAMS

Day - Defayette
Evening - Gaut
Night - Collins

NRR	Thurs. Mid - Friday Noon	Fri. Mid - Sat. Noon
	R. Mattson	B. Grimes
	Roy Woods	T. Novak
	Fri. Noon - Fri. Mid	Sat. Noon - Sat. Mid
	D. Eisenhut	Don Davis
	T. Marsh	S. Israel

176 016

4/2/79

Central Laboratory Telephone Sources - 24 Hour Availability

LAB		FTS		Commercial
BNL		666-2238		516-345-2238
SANDIA		475-3155		505-264-3155
INEL		583-1515		208-526-1515
SRL		239-2117		803-725-2117
LLL		532-7222		415-422-7222
			Backup	415-828-7475
LASL	(1)	843-2125		505-672-1547
	(2)	843-2020		505-672-9019
	(3)	843-5037		505-672-1302
				505-672-9102

ORNL - 624-6606 [Lab Shift Sup has call list]

→ Phone Nos.
→ Chronology

176 017

Incomprehensible
Documents

0000 - 0600 - 2 opr
1 HP

0600 - 1800 - Ox Shift Off (0600 - 1800)
- 2 opr
- 1 HP
- 1 Shm

1800 - 2400 - Ox Shift (1800 - 2200)
3 opr
1 HP
1 Sh (1800 - 2200)

Impact 2 opr
Ox 1 HP
12 hr Stress

06 - 08 - Stabe

04 - 12 PM Shafer

176 018

Navy exp.
Evans min
~~[illegible]~~
[illegible] → [illegible]

Fw 10 pm —
Lo 4 pm —
Gilinsky

Review Bulletin Replys

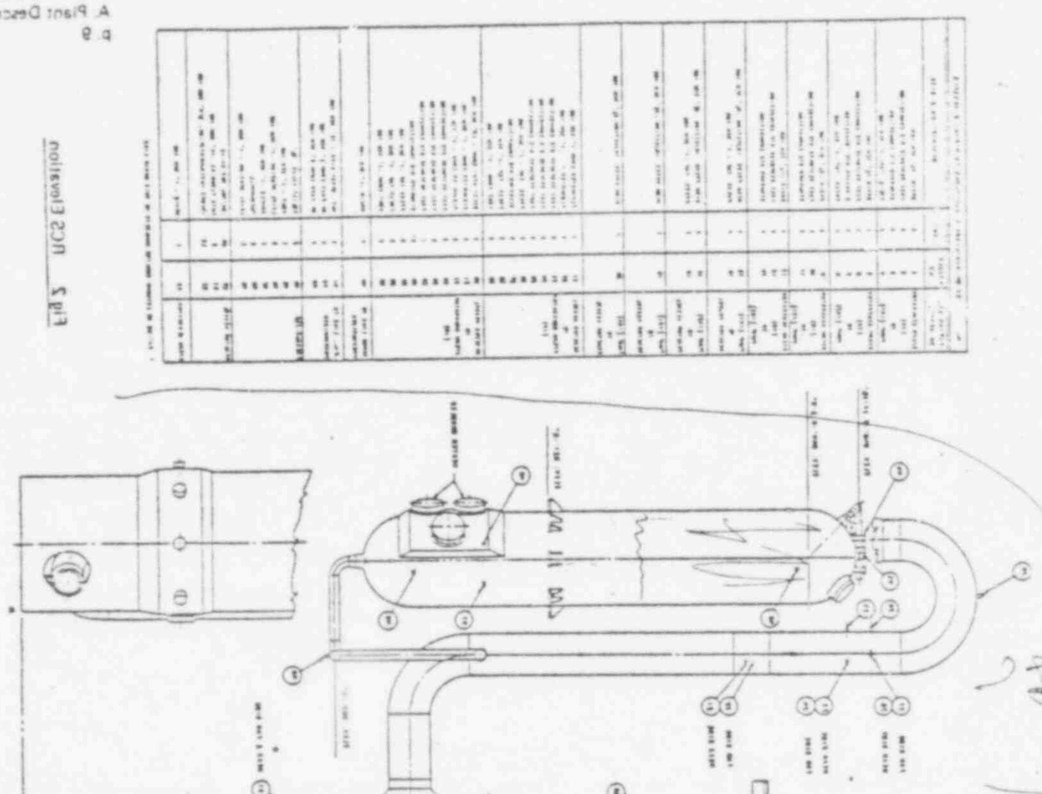
Marine Yambu >

[illegible] >

176 019

Vol. PWR-1
Oconee 1, 2 & 3
A. Plant Descri
p. 9

Fig 2 RCS Elevation



176 020

Vent Header Release resulted in MakeUp Tank Overpr

Leak in relief discharge of MakeUp Tank

Vent ~~thru~~ aux Bld thru

Vent Header direct to stacks

Stable now * — EW) 7-4406

485-1326 488-1080

Women who may be

pregnant should not

remain with 5 Miles of plant

FTS
637-7859

4350 EW Highway

Phone Edwards Nav
456-2281 2-7010

Monica Von
498-2899

176 021

Cook
Newsweek

1. Was there a leak between the primary and secondary loops thru the steam generator. - radioactivity level

Wed am
2. ECCS on; primary pumps ~~off~~ on or off
When did trains of ECCS & turn on the primary pumps on?

Don't Know?

3. When did they discover bubble in reactor vessel? Approx 1/2 am Thurs morn. when attempting to depressurize. Pr not responding level went up - expansion

4. Metal - water reaction - is ~~[illegible]~~ explosive force caused by steam?

204
3590

5. Was - water released through relief valve at top of pressurizer? yes.
1st during transient
2nd to control P when it went solid

1. Polishing system
2. caused main feedwater pump(s) to trip
3. causes turbine stops to close
4. reactor trip
unsure of whether happened P went up

feedwater both pumps?
Turb Trip

176 022

2500 psi at 80°F
2300 psi at 150°F

EP. 21 - 2500 @ 150
1300 @ 100

? for Governor

time to melt down

176 025

FLYING BLIND

START HPI to 1000 psi
POV set point to get solid
on last Pressurizer ~~level~~
REDUCE PRES ~~to 0~~
BY 200 psi increment
every 12 hour, ~~raising~~
maintain solid by bumbing
POV setpoint with HPI
At 400 psi stop - get
SG's ready for natural
circ.
After SG ready, trip
pump - see if Nat'l circ
if so ~~stop~~ stay & depress.
slowly
if not nat'l circ, go
RHR
if no RHR - ~~get~~ go ~~up~~
HPI up in press. and
go on sump

176 024

FLYING BLIND

START HPI to 1000psi

POV set point to get solid
on last pressurizer ~~level~~

REDUCE PRES ~~TO~~

BY 200 psi increment

every 12 hour, ~~raising~~

maintain solid by bumbing

POV setpoint with HPI

At 400 psi stop - get

SG's ready for natural

circ.

After SG ready, trip

pump - see if Nat'l circ

if so ~~stop~~ stay & depress.

slowly

if not nat'l circ, go

RHR

if no RHR - ~~get~~ go ~~up~~

HPI up in press. and

go on sump

176 025

Mills — 6AM Thurs 4/4
Cunningham — 6AM Sat. 4/6
Williams 6AM

$$1 \text{ watt} = \frac{3.12 \times 10^{10} \text{ fissions}}{\text{sec}} \times \frac{200 \text{ MeV}}{\text{fission}}$$

$$1 \text{ watt} = 6.24 \times 10^{12} \text{ MeV/sec}$$

$$\text{Ci} \times \frac{3.4 \times 10^{10} \text{ disint}}{\text{sec}/\text{Ci}} \times 0.5 \text{ MeV/disint} \times \frac{\text{watt}}{6.24 \times 10^{12} \text{ MeV/sec}}$$

No. of ci × 3×10^{-3}

If 10^8 ci, then 3×10^5 watts or 300 kw
if 10^7 ci, then 30 kw
if 10^6 ci, then 3 kw

$\sim 2 \times 10^7$ ci at 0.5 Mev

$2 \times 10^7_{ci} \times 3 \times 10^{10} \frac{\text{disint/sec.}}{ci} \times 0.5$ Mev/ci

3×10^{17} Mev/sec.

~~3×10~~

Bob Bernero Called
~15:10 4/5

Hart & Kennedy will hold
hearings next week on
decontamination/decommissioning
of TMI. Bob to be
witness. FYI.

Luke

Gave Bob misc. info.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

?1. H_2 Recombiner Readay

?2. Cont. penetration Adequacy
Type
100#

?3 RCP emerg. purp.

4 Coordinate with NRC
Site

5. L.O. to all RCP - OK

?6. SGA status

?7. H_2 Evolution [illegible]

8. Interim [illegible]

9. Lead [illegible]

10 Boron [illegible]

11 C/B [illegible]

176 029

103.5

NUCLEAR REGULATORY COMMISSION

OFFICE OF INSPECTION AND ENFORCEMENT

WASHINGTON, D.C.

INCIDENT MESSAGE FORM

Letters to other agencies

TO: Chmn - OK

FROM:

Bradford - problem with letters —

Gilinsky - OK

Kennedy - OK as is

90.9

DATE: ________________ TIME: ____________ __________

100.51

176 030

from D. Thompson (from Keppler Reg III)

(site)

Dunning, NRR asked Pratt to examine suitability of valves in accident environment.

36" butterfly Henry Pratt Valves — 2×10^7 RADS, 286°F — SPEC.

in "purge isolation" Contain a plastic part that didn't get tested to specs. Failure could cause valve failure. — 4×10^5, 200°F — TEST

~~Working on pressurizer~~

176 031

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 1, 1979

MEMORANDUM FOR: B. H. Grier, Director, Region I
J. P. O'Reilly, Director, Region II
J. G. Keppler, Director, Region III
K. V. Seyfrit, Director, Region IV
R. H. Engelken, Director, Region V

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, OIE

SUBJECT: IE BULLETIN 79-05, NUCLEAR INCIDENT AT THREE MILE ISLAND

The subject IE Bulletin should be dispatched for action by April ~~2~~ 5 1979, to all ~~B&W~~ power reactor facilities with an operating license.

Subject bulletin and enclosures should also be dispatched for information ~~to all other power reactor facilities with an operating license and to~~ To all power reactor construction permit holders.

The text of the Bulletin, Enclosures ~~1~~ drafts and draft letters to the licensee are enclosed for this purpose. ~~Enclosure 1 which consists of the referenced Preliminary Notifications, should be added by the regional office. The letters to the license make a commitment to forward the~~ The continuing Preliminary Notifications of the incident~~. These~~ should ~~be~~ continue to be forwarded as they are received ~~for~~ in accordance with the transmittal memorandum for IE Bulletin 79-05.

for

Norman C. Moseley, Director
Division of Reactor Operations Inspection
Office of Inspection and Enforcement

Enclosures:
1. Draft Transmittal Letter to ~~B&W~~ Licensees all operating
2. Draft Transmittal Letter to ~~all other power facilities~~ CP Holders
3. IE Bulletin No. 79-05 A (w/enclosures - 2)

CONTACT: D.C. Kirkpatrick, IE
49-23180

176 032

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, DC 20555

DRAFT

APRIL 5, 1979

IE Bulletin 79-05A

NUCLEAR INCIDENT AT THREE MILE ISLAND - SUPPLEMENT

Description of Circumstances:

Preliminary information received by the NRC since issuance of IE Bulletin 79-05 on April 1, 1979 has identified six potential human, design and mechanical failures which resulted in the core damage and radiation releases at the Three Mile Island Unit 2 nuclear plant. The information and actions in this supplement clarify and extend the original Bulletin and transmit a preliminary chronology of the TMI accident through the first 16 hours (Enclosure 1).

1. At the time of the initiating event, loss of feedwater, both of the auxiliary feedwater trains were valved out of service.

2. The pressurizer ~~power-operated~~ electromatic relief valve (PORV), which opened during the initial pressure surge, failed to close when the pressure decreased below the actuation level. ~~Operator action to close the backup valve required 2.3 hours.~~

3. Following rapid depressurization of the pressurizer, the pressurizer level indication may have lead to erroneous inferences of high level in the reactor coolant system. The pressurizer level indication apparently led the operators to prematurely terminate high pressure injection flow, even though substantial voids existed in the reactor coolant system.

4. Because the containment ~~did~~ does not isolate on high pressure injection (HPI) initiation, the highly radioactive water from the PORV discharge was pumped out of the containment by the automatic initiation of a transfer pump. This water entered the radioactive waste treatment system in the auxiliary building where some of it overflowed to the floor. Outgassing from this water and discharge through the auxiliary building ventilation system and filters was the principal source of the offsite release of radioactive noble gases.

5. Subsequent action by plant operators based largely upon pressurizer level indication apparently led to a gradual primary coolant inventory reduction due to premature securing of the high pressure injection, ~~and failure to isolate the PORV (Item 2)~~

176 033

6. Tripping of reactor coolant pumps during the course of the transient, to protect against pump damage due to ~~cavitation~~ pump vibration, lead to fuel damage since voids in the reactor coolant system prevented natural circulation.

Actions To Be Taken by Licensees:

A. For all Babcock and Wilcox pressurized water reactor facilities with an operating license (the actions specified below replace those specified in IE Bulletin 79-05):

1. (This item clarifies and expands upon item 1. of IE Bulletin 79-05.)

   In addition to the review of circumstances described in Enclosure 1 of IE Bulletin 79-05, review the enclosed preliminary chronology of the TMI-2 3/28/79 accident. This review should be directed toward understanding the sequence of events to ensure against such an accident at your facility(ies).

2. (This item clarifies and expands upon item 2. of IE Bulletin 79-05.)

   Review any transients similar to the Davis Besse event (Enclosure 2 of IE Bulletin 79-05) and any others which contain similar elements from the enclosed chronology (Enclosure 1) which have occurred at your facility(ies). If any significant deviations from expected performance are identified in your review, provide details and an analysis of the safety significance together with a description of any corrective actions taken. Reference may be made to previous information provided to the NRC, if appropriate, in responding to this item.

3. (This item clarifies item 3. of IE Bulletin 79-05.)

   Review the actions required by your operating procedures for coping with transients and accidents, with particular attention to:

   a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability, especially natural circulation capability.

   b. Operator action required to prevent the formation of such voids.

   c. Operator action requir to enhance core cooling in the event such voids are formed.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, DC 20555

DRAFT

APRIL 5, 1979

IE Bulletin 79-05A

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3. Following rapid depressurization of the pressurizer, the pressurizer level indication may have lead to erroneous inferences of high level in the reactor coolant system. The pressurizer level indication apparently led the operators to prematurely terminate high pressure injection flow, even though substantial voids existed in the reactor coolant system.

4. Because the containment did not isolate on high pressure injection (HPI) initiation, the highly radioactive water from the PORV discharge was pumped out of the containment by the automatic initiation of a transfer pump. This water entered the radioactive waste treatment system in the auxiliary building where some of it overflowed to the floor. Outgassing from this water and discharge through the auxiliary building ventilation system and filters was the principal source of the offsite release of radioactive noble gases.

5. Subsequent action by plant operators based largely upon pressurizer level indication apparently led to a gradual primary coolant inventory reduction due to premature securing of the high pressure injection and failure to isolate the PORV.

176 035

6. Tripping of reactor coolant pumps during the course of the transient, to protect against pump damage due to cavitation, lead to fuel damage since voids in the reactor coolant system prevented natural circulation.

Actions To Be Taken by Licensees:

A. For all Babcock and Wilcox pressurized water reactor facilities with operating license (the actions specified below replace those specified in IE Bulletin 79-05):

1. (This item clarifies and expands upon item 1. of IE Bulletin 79-05.)

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   b. Operator action required to prevent the formation of such voids.

   c. Operator action required to enhance core cooling in the event such voids are formed.

4. (This item clarifies and expands upon item 4. of IE Bulletin 79-05.)

Review the actions directed by the operating procedures and training instructions to ensure that:

a. Operators do not override automatic actions of engineered safety features without sufficient cause for doing so.

b. Operating procedures currently, or are revised to, specify that if the high pressure injection (HPI) system has been actuated because of low pressure condition it must remain in operation until either:

(1) Both low pressure injection (LPI) pumps are in operation and flowing at a rate in excess of 1000 gpm each and the situation has been stable for 20 minutes, or

(2) Verification has been made by evaluation of pressure as well as level indications that the primary water has returned to, and stabilized at, normal levels and all hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing RCS pressure. If 50 degree subcooling cannot be maintained after HPI cutoff, the HPI shall be reactivated.

c. Operating procedures currently, or are revised to, specify that in the event of HPI initiation, with reactor coolant pumps (RCP) operating, at least one RCP per loop shall remain remain operating unless there is clear evidence that pump damage is imminent.

d. During transients, operators do not rely upon pressurizer level indication alone, but also examine pressurizer pressure and other plant parameter indications, in evaluating plant conditions, e.g., water level in the reactor core.

5. (This item revises item 5. of IE Bulletin 79-05.)

Review all safety-related valve positions and positioning requirements to assure that valves are positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance and testing, to ensure that such valves are returned to their correct positions following necessary manipulations.

Review, the containment isolation initiation design and procedures, and prepare and implement promptly all changes necessary to cause

6. ~~Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the~~ containment isolation of all lines whose isolation does not degrade core cooling capability to be actuated upon automatic initiation of safety injection.

7. For manual valves or manually-operated motor-driven valves which could defeat or compromise the flow of auxiliary feedwater to the steam generators, prepare and implement procedures which:

   a. require that such valves be locked in their correct position; or

   b. require other similar positive position controls.

8. Prepare and implement immediately, procedures which assure that two independent steam generator auxiliary feedwater flow paths, each with 100% flow capacity, are operable at any time when heat removal from the primary system is through the steam generators. When two independent 100% capacity flow paths are not available, the capacity shall be restored within 72 hours or the plant shall be placed in a cooling mode which does not rely on steam generators for cooling within the next 12 hours.

   When at least one 100% capacity flow path is not available, the facility shall be made subcritical within one hour and the facility placed in a shutdown cooling mode which does not rely on steam generators for cooling within 12 hours.

B For all power reactor facilities with an operating license:

2. (This item revises item 6 of IE Bulletin 79-05.)

   Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping of radioactive liquids and gases will not occur inadvertently.

   Insert Item 1

   In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

   a. Whether interlocks exist to prevent transfer when high radiation indication exists, and

   b. Whether such systems are isolated by the containment isolation signal.

176 038

3. Review and modify as necessary your maintenance and test procedures to ensure that they require:

   a. Verification, by inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

   b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

   c. A means of notifying involved reactor operating personnel whenever a safety-related system is removed from and returned to service.

   In addition, all operating and maintenance personnel should be made aware of the extreme seriousness and consequences of the simultaneous blocking of both auxiliary feedwater trains at the Three Mile Island Unit 2 plant.

4. Review your prompt reporting procedures for NRC notification to assure very early notification of serious events.

For Babcock and Wilcox pressurized water reactor facilities with an operating license, respond to Items A.1, 2, 3, 4.a and 5 by April 11, 1979. Since these items are substantially the same as those specified in IE Bulletin 79-05, the required date for response has not been changed. Respond to Items A.4.b through A.4.b through A.4.d, A.6 through A.8, and B by April 16, 1979.

For all other power reactor facilities with an operating license, respond to Item B by April 16, 1979.

Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, DC 20555.

For all reactors under construction, this Bulletin is for information purposes and no written response is required.

Approved by GAO, B 180225 (R0072); clearance expires 7-31-80. Approval was given under a blanket clearance specifically for identified generic problems.

Enclosures:

1. Preliminary Chronology of TMI-2 3/38/79 Accident Until Core Cooling Restored.

2. List of IE Bulletins issued in last 12 months.

The Nuclear Regulatory Commission's ~~formal~~ investigation of the Three Mile Island accident is actively underway. Of course, our most immediate concern has been dominated by the operational considerations of limiting further releases and returning the plant to a safe and secure shutdown condition. Consistent with our efforts in that direction and to the extent we could do so without interfering with the recovery operations, ~~operational response~~, our investigators have gathering ~~been~~ at work information. ~~even during the early days of the recovery phase, to gather evidence to be used in the formal investigation.~~ We have begun our review of plant records associated with the accident and have conducted some preliminary interviews. We are continuing a more thorough examination of logs and preparing for interviews in greater depth.

~~We have now reached the stage that we can, and have, begun our formal investigation of the accident sequence and the licensee's response to it. The NRC investigation team is presently at the Three Mile Island site in the field phase of its work, following several days of review of plant records and charts furnished to our Operations Center in Bethesda.~~

As you are aware, Mr. Chairman, the NRC deliberated very carefully concerning the possibility of serious generic concerns that might require some sort of drastic action at other nuclear power plants designed by Babcock and Wilcox, the nuclear steam system supplier for the Three Mile Island plant. The results of our review in this area led us to the conclusion that although neither shutdown nor power reduction were warranted at other B&W plants, it was important to require these other utilities to make some procedural changes and to complete some supplementary training of their operators.

When will results of the investigation be available?

Primary attention has been devoted to assuring the safe status of the plant and detailed investigation has been delayed if this would in any way interfere with the primary effort.

Preliminary interviews with selected plant personnel (~~including operators~~) and a partial review of plant records have been ~~completed~~ initiated.

A detailed plan for carrying out a ~~full~~ and complete investigation is current being ~~finalized~~ developed by the Office of Inspection and Enforcement.

Given the scope and complexity of the investigation, we expect it will be several months before it is completed and a report of the investigation issued.

LVG:

Delivered by courier shortly before 11:00 pm, 4/7.

Recommend vigorous nonconcurrence, at the very least; screaming and shouting at the middle level of response; and a complete tantrum at the preferred level.

If this were to be used, the Chairman would break his own breastbone, and that is not an appropriate posture to cast him in. Mea culpa is the last thing he should do.

In addition, the abbreviated chronology they have him present needs substantial improvement.

DG

I agree

EV

176 042

Index to Tab 9 – General Q & A's

1. Complete list of Gen Q & A's with brief ~~[illegible]~~ answers

2. Answers to Specific Gen Questions from above list.

   Quest 2 – Precautions at other B&W Plants.

   Quest 3 – Effect of TMI with NRC inspector present.

   ~~Quest 4~~ Quest 5 – What NRC does to assure compliance by licensees.

   Quest 6 – General public Radiation Exposures

   Quest 7 – Commission Recommendation – Evacuation

   Quest 10, ~~[illegible]~~ – Respective Roles of NRC, State & Local Govs.

   Quest 11 – Basis for Gov of PA recommendation to evacuate pregnant women and preschool children – 5 mi radius
   Answer incomplete.

   Quest 12 – Precautionary Actions by NRC Regarding other B & W plants (See answer to Quest 2.)

176 043

Quest 17 - With risk of core meltdown and/or hydrogen explosion, why no evacuation?

Quest 18 - When will results of the investigation be available?

Quest 19 - Info on cause of TMI accident.

Quest 24 - What caused radiation release?

Quest 25 - Status of other B&W Plants -

Quest-27 - Whose responsibility to order evacuation?

Quest-29 - Agencies monitoring in environment at TMI

Quest 30 - Estimated Personnel Exposures Offsite.

Quest-31 - Delay in Licensee reporting to NRC

Quest-32 - Changes for future in dissemination of information to general public.

(1:05 pm. [illegible])
Corrected copy

CHRONOLOGY

OF

NRC RESPONSE

TO

THREE MILE ISLAND INCIDENT

(FOR PERIOD MARCH 28 - APRIL 1, 1979)

NRC OPERATIONS CENTER
(Draft of 9pm - April 6, 1979)

176 045

Introduction

Below is a selection of highlights taken from the more detailed Combined Chronology which follows it. Both the highlights and the full chronology emphasize notification actions involving the NRC and, for that reason, the entries for the first hours following knowledge of the incident are more extensive than for subsequent periods.

The Combined Chronology is a compendium of information received and actions taken by the NRC related to events at the Three Mile Island nuclear facility during the period March 28/9 through April 1, 1979. It draws upon a number of sources -- identified for each entry -- and reflects the factual situation as known to the cited source at the indicated time. The information contained in the chronology should be treated as preliminary in nature and subject to later confirmation or clarification.

Highlights

AM	Wednesday, March 28
7:02	Penn. Emergency Mgmt. Agency notified by licensee.
7:45	Licensee notifies Region I, NRC
8:00-02	Region I notifies IE, NRC Hq
8:05-10	NRC Incident Response Center activated
8:45	Region I team leaves for site
8:50	Open line from Region I to Plant Control Room established
9:15	White House Situation Room contacted ~~Phone contact with~~
9:27	Phone contact with Defense Civil Preparedness Agency
10:05	Region I Response Team arrives at site

176 046

10:16	Conference call: Commissioners and IE Director
Highlights	
space AM	Wedn~~s~~esday, March 28
space 10:20	Phone contact with PA State Rad Health Dept
10:40	Phone contact with PA Civil Defense
PM 6:15	PA Rad Health Dept notifie~~d~~s NRC that they will keep Governor informed
AM	THURSDAY, MARCH 29
9:30	Phone contact with Food and Drug Adm
11:05	Phone contact with ~~NY~~ NJ Dept of Health
11:45	Phone contact with Delaware Rad Health Dept.
PM 12:10	Phone Contact with W.VA Rad Health Dept
2:40	Phone contact with Fish and Wildlife Service
3:00	Licensee pulls dosimeters from 17 fixed positions in 15 mile radius: two above normal.
3:01	Phone contact with Maryland Rad. Health Dept.
5:55	NRC directs licensee to stop dumping all water
6:10	Licensee notifies NRC: stopping discharge
4:00	Report on briefing for Senators Hart, Heinz, Simpson and Ertel
5:30	Report on briefing for Senator Schweiker, Congressmen Gooding, McCormack, Waller, Weidler
10:12	Verification that Industrial Waste Discharge off
AM	FRIDAY, MARCH 30
8:00	Status report to EPA
8:20	Status report to FDA
9:15	Phone contact with PA Civil Defense re potential evacuation

176 047

10:30	State advises residents with 10 miles to stay indoors
10:47	NRC decision to dispatch H. Denton to site
<u>PM</u>	
12:03	NRC Chairman recommends to PA Governor to evacuate pregnant women and pre-school children in 5-mile radius
1:25	NRC Chairman meets with President; NSC convenes afterward
2:00	Denton and 12 staff arrive at site by helicopter; confers with President.
2:30	NRR Operations Center near site: notifications to President and Governor
3:30	83 NRC personnel on site.
<u>AM</u>	<u>SATURDAY, MARCH 31</u>
9:20	Phone contact with NY Radiation Health Bureau
12:00	Phone contact with FDA re supplies of potassium iodine.
<u>PM</u>	
10:45	NRC informed of reported planned sabotage attempt.
<u>AM</u>	<u>SUNDAY, APRIL 1</u>
11:10	97 NRC personnel on site
<u>PM</u>	NRC establishes 37 rad monitors at distances 1 to 12 miles from plant
	All licensees with B&W reactors contacted; inspectors dispatched
2:15 -27	President Carter on-site Unit 2 Control Room

COMPANIES AND AGENCIES REPRESENTED ON SITE

Angotors, Inc.

Argonne Lab

B&W

Bechtel

Bell of PA

Bisco

Boston Edison

Bureau of Land Mgmt

Calgon

Capalupo & Gundal

CAI

Catalytic

Chem. Nuclear

Combustion Engineering

Commonwealth Edison

DOE

Duke Power

EG & G (ARMS)

Energy, Inc.

Endochem

Florida Power & Light

L.H. Focht & Son

General Dynamics

combine pages 5,6,7 into a single page. single space the list and use 2 columns if necessary. change subsequent page nrs.

GE

GPU

BPUSC

Gilbert Commonwealth

Gilbert Associates

Halliburton Services

Harshaw

HEW

Hartford Steam Boiler

IBM

IBM

JCP&L

Keystone Helicopter

Lucking Brothers

Mitre Corp

Modesto

NASA-DOE

Nuclear Support Services

NUS

Oak Ridge National Laboratory

Ontario Hydro

Penelec

PP&L

Penn State

Penn National Guard

Philadelphia Electric

Pickard Lowe & Garrick

Porter

Public Service Electric & Gas

Radiation Management Corp Radiation Services

Rockwell International

Self Photo One

Science Applications

Stone & Webster Eng. Corp

SAI

Technology for Energy

Tri State Laundries

United Engineers

Union Carbide

USAF

USDA Forrest Service

U.S. Dept of Interior

United Telephone Co.

Vitro Services

WPS

Washington Power Service

Walters Septic Service

Westinghouse

EML(HAZ)

EPA

Date/Time	Source	Activity
Wednesday, March 28		
AM		
4:00	None	Incident sequence begins.
6:50	Operations Center (OC) tape	Licensee declares site emergency.
7:02	SP Followup w/PA	Licensee notifies PEMA of site emergency.
7:10 - 7:45	Region I	Licensee attempts to contact Region I. Duty officer and Dep. Director officer enroute to office when beeper sounds.
7:24 ~~7:20 - 7:30~~	OC tape ~~Region I~~	~~Burner~~ Licensee declares general emergency.
7:45	Reg. I	Upon opening switchboard, Region I receives message from licensee
7:50	Reg. I	Region I contacts TMI control room; maintains open line.
7:55	Reg. I	Region I classifies event as a Level 1 severity incident in accordance with Region I incident response plan.
8:00	Reg. I	Region I ~~NRC Headquarters~~ Incident Response Center activated; John Davis, HQs., notified by Boyce Grier and also by Public Affairs (Fouchard notified by Region I Public Affairs).
8:05	Various	John Davis orders Hqs Operations Center activated. Region I State Liaison attempts to contact PA Bureau of Rad. Health.
8:20	Reg. I	Region I contacts independent measurement van (at Millstone site) and orders it dispatched to TMI site.

176 052

Date/Time	Source	Activity
8:23	~~Operations~~ OC tape	Phone call from Weiss and Moseley to ~~Center (OC)~~ Grier; John Davis calls Gossick, ~~Tape~~ who is not in.
8:30	Reg. I	PA State Police informed that NRC emergency vehicle would be en route.
8:25	Incoming Tel. Log	Gossick calls Davis
8:31	OC Tape	Moseley notifies Stello & Eisenhut; Stello says he will send radiological experts to OC
8:32	OC Tape	John Davis calls L.V. Gossick
8:34	OC Tape	John Davis calls Denton's office; speaks to Case.
8:36	OC Tape	Davis calls Hendrie's office; speaks to Bill Dorie. Several calls made to reach Hendrie; finally contacted by _____ at (place) (____AM).
8:40	OC Tape	Mike Wilbur calls Boyce Grier, obtains technical information.
8:45	Reg. I	Five inspectors (including HPs) ~~types)~~ with radiation monitoring equipment leave for site.
8:46	OC Tape	Davis calls Dorie, asks for Comm. Gilinsky in Chairman's absence; Gilinsky not in yet.
8:48	OC Tape	Dudley Thompson notifies Tom Carter, NMSS. No NMSS action required.
8:49	OC Tape	Ward notifies J. Davidson, NMSS, to make IAT notification.
8:50	Reg. I Log	Licensee calls Reg. I with current status report. ~~(Open line from Region I to Control Room established ____ am)~~

Date/Time	Source	Activity
8:50	Reg. I	State Liaison contacts PA Bureau of Rad Health
8:56	OC Tape	Gossick attempts to reach Gilinsky through Bill Dorie and through Gilinsky's office. Gossick talks to John Austen and suggests that Dorie personally notify the Chariman.
8:57	OC Tape	Davis notifies Commission Ahearne
8:59	OC Tape	Bernie Weiss calls DOE Emergency Operations Center
9:00	OC Tape	Bill ~~Waten~~ Ward informs Randy Pine (CA). Randy Pine indicates that she will inform local Congressmen (Heinz & Schweiker) and Rep. __.
9:00	Reg. I	Second vehicle leaves Region for site (investigator plus inspector); Region I contacts RAP who has already been notified; two teams organized and standing by.
9:02	OC Tape	Weiss notifies EPA (Floyd Galpin)
9:08	SP notes	Joe Fouchard calls Carl Abraham, Reg I Public Affairs
9:10	OC Tape	Gossick calls Congressional Affairs Office. Randy Pine informs Gossick that CA had received several inquiries from local Congressmen.
9:00	Bob Ryan	Ryan notified by SP Region I personnel. Arrives IRACT 9:35am

Date/Time	Source	Activity
9:10 to 9:30	OCA	OCA places call to majority and minority staffs of House of Subcommittee on Energy and Environment, House Subcommittee on Energy and Powe. and Senate Subcommittee on Nuclear Regulation as well as Senators Heinz and Schweiker and Representatives Walker and Ertel to adivse of declaration of site emergency at Three Mile Island
9:06	OC Tape	Bill Ward informs Communications Branch
9:10	OC Tape	Grier calls Moseley to explain technical aspects of incident.
9:11	Incoming Log	Gilinsky calls Gossick and Davis
9:16	OC Tape	Fouchard notifies DOE Public Affairs Office (Bob Dulin).

AFTER 9:15 AM

Other NRC personnel began arriving at Incident Response Center (OC Tapes not yet scanned for times after 9:16 am)

Date/Time	Source	Activity
9:15	OC Notifications	White House Situation Room Log contacted
9:27	Incoming Log	Defense Civil Preparedness Agency (DCPA) Calls Joe Hegner
10:05	PNO-79-67	Reg. I response team arrives at site
10:05	Reg. I	Reg I contacts EPA Hqs. (Also attempts to contact EPA:III)
10:05	Reg. I	Reg I contacts EPA Hqs. (also attempt- 'o contact EPA:III)
10:05 - 10:10	Reg. I	Discussion w/RAP re ARMS aircraft survey. ARMS people put on standby

Date/Time	Source	Activity
10:15	Reg. I	Onsite team informs PA Rad. Health that they are available for questioning.
10:16	Incoming Log	Conference call: All Commissioners and Davis
10:20	SP Log	Telephone to PA State Rad Health Dept. (Gerusky not available). Call returned at 10:45 (1st liaison established by Hqs) subsequent calls every hour or two, starting about 5:25 pm to discuss status of sampling and monitoring.
10:30 to 11:30	OCA	OCA calls principal oversight committees (including appropriations subcommittees) and Pennsylvania representatives from vicinity of site to advise of release of radioactive materials.
10:30	Reg. I	Req. I contacts Delaware
10:40	Reg. I	Reg I contacts PA Governor's action center
10:45	Reg. I	Reg I contacts NY State Energy Office
10:30	PR#79-64	First Press Release based on Preliminary Notification
10:40	SP Log	Press Conference (from where?) patched through to PA Rad Health Dept.
10:55	SP Log	Telephone call to PA Civil Defense (second call to Operations Officer at 11:30 am).
10:58	Incoming Log	White [illegible] se Duty Officer to Weiss
11:00	Reg I	Second leam arrives at site and in control room

Date/Time	Source	Activity
11:35	Reg I	FPA Region III contacted
11:45	Reg I	NJ Dept of Energy contacted
11:55	Reg I	State of MD, Power Plant Siting council contacted
PM		
12:04	Reg. I	EPA Region III contacted
12:10	Reg I	DOE (Valley Forge Office) contacted
12:30	OCA	OCA calls principal oversight committees and PA representatives regarding latest information (In response to request from Henry Myers for technical information, arranged for briefing by Mr. Stello)
1:00	Reg I	Third vehicle departs for site
1:00	Reg I	Reg I contacts MD Health Dept.
1:01	Incoming Log	Stello calls Henry Myers
1:12	SP Log	SP calls VA Civil Defense ~~re rumors~~
1:30	DOE Logs	DOE advance party establishes command post at Capitol City Airport
2:15	DOE Logs	ARMS helicopter arrives at site and begins tracking.
2:30	DOE Logs	BNL RAP arrives Capitol City Airport; sampling begins
2:45	Reg I	State liaison contacts Governor's office Connecticut
3:00	SP Log	Proposed NRC Press Release cleared with PA Rad Health Dept
3:45	PN 79-67	NRC

176 057

Date/Time	Source	Activity
4:00	OCA	Telephone briefing for Senate Subcommittee on Nuclear Regulation and Senator Heinz
5:00	PR 79-65	Second press release issued
5:15	Reg I	NRC mobile lab arrives at site
6:00	Reg I	Fourth Reg I vehicle leaves for site
7:55	SP Log	Call to PA Rad Health Dept (ARMS data shows count is up, status of BNL monitoring activities).
6:15	SP Log	PA Rad Health Dept acknowledges receipt of ARMS data and says they are keeping Governor informed.
8:17	Reg. I Log	Reg I notified of decision to send NRR team to site; arrival expected next AM.
8:30	Reg I	Fourth vehicle arrives on site
8:30 - 9:45	Reg I	NRC, State, RAP Team brief ~~9:45~~ Lt. Governor Scranton
9:00	Region I Log	Reg I notified that Salem providing equipment
9:00	SP Log	SP verifies (how?) that FAA has not been notified.
10:00 - 11:00	Reg. I	Reg I participates in Lt. Governor's press conference
11:30 (PM) - ~~AM~~ 12:30 (AM)	Reg I	Team briefs Governor Thornburgh

~~Thursday March 29~~

176 058

~~Thursday, March 29~~

AM

12:15	PR #79-66	Press Release
1:00	SP Log	SP notifies Defense Civil Preparedness Agency and reads press release.
2:00	SP Log	SP telephones PA Emergency Management Agency, reads press release
~~PM~~ 2:10	SP Log	SP tries to telephone Health Center for Disease Control (Atlanta) (PA radiological Health says they will try again in morning).
8:30	SP Log	SP calls H. Calley, EPA, to read press release and suggests he call Gerusky and offer assistance
9:00	Reg I	Two additional vehicles (6 people) dispatched to site
9:30	SP Log	Food and Drug Administration calls and offers to have Baltimore Field Office provide assistance in looking at food pathways
10:25	PN-79-67A ~~Issued~~	Issued
11:00	SP Log	SP tries to contact NY Bureau of Radiological Health (call completed at 12:17 pm)
11:05	SP Log	SP calls NJ Dept. of Health.
11:30	OCA	Chairman Hendrie and NRC staff brief members of Subcommittee on Energy and Environment, other Members of Congress, and Congressional staff on status of incident.
11:45	SP Log	SP calls Delaware Rad-Health Dept.
PM		
12:05	SP Log	SP returns call to Gov. Ray's (Washington) assistant.

Date/Time	Source	Activity
12:10	SP Log	SP calls W.VA Rad Health contact
12:15	SP Log	SP calls Va. Radiological Health (call finally completed at 1:15).
Approx. Noon	unverified	Strasma (Region III PA liaison) on site; Vollmer +7 (NRR) arrive on site.
12:00 - 1:00	Reg I	Vehicles 6 and 7 on site (total IE personnel: 17) use color
2:15	Reg I	Congressional group (Hart, Udall, Heinz, et al) arrive observation center; receive briefing.
2:40	SP Log	SP telephones Fish and Wildlife Service.
3:00	PN-79-67E	Licensee pulls thermoluminescent dosimeters from 17 fixed positions located within a 15 mile radius of site. Dosimeters had been in place for three months and had been exposed for about 32 hours after incident. Only two dosimeters showed exposures above normal levels.
3:01	SP Log	SP briefs (by phone) Md. Radiological Health on status of samples.
3:20	Reg I	State liaison contact with Vermont
3:15 - 6:30	Reg I	NRC representatives are called to State Capital to brief Governor and Lt. Governor prior to Governor's 5:00 p.m. press Conference.
3:35	SP Log	In response to inquiry from Minnesota Rad Health, SP gives status briefing.

176 060

Date/Time	Source	Activity
5:55	Reg I Log	The Executive Management Team directs the licensee to stop dumping all water. RI notifies HQ that stopping the dumping will cause backing of water into the Turbine Building. (Late entry - the licensee has been dumping water to the river -- the water is within TS limits WRT contamination.)
6:00	Log???	PA Rad Health says NRC can make decision on rad. water dump without checking further with them.
6:10	Reg I Log	Licensee notifies NRC that he is stopping the discharge
5:30	Reg I Log	Briefings provided by Met. Edison to Senator Schweiker, Congressmen Gooding (York), Mike McCormack; Waller (Lancaster), Weidler (L.I.,NY)
6:00	PN-67B	NRC requests Met. Edison to terminate release of slightly contaminated industrial waste. Permission to resume release granted at 12:15 am and coordinated with State. State press release issued.
4:50 - 7:00	Reg. I	Multiple attempts to reach EPA:III concerning industrial waste dumps; finally contacted at 7:00 p.m.
6:35	SP Log	PA Rad Health calls back and asks NRC to hold river dump because of Governor's concern. SP provides update on ARMS data.
7:05	SP Log	Offer from DOE Emergency Assistance contact (Joe Deal) from Harrisburg airport.
8:30	Reg I	Call to Governor's Aide to inform of core damage

176 061

(under activity in ~~first~~ column

Date/Time	Source	Activity
6:00 – 8:30	SP Log	Updated status reports to MD, WVA and US Bureau of Rad. Health
Unknown	Reg. I Log	

Reg I Log Periodic ARMS flights begin;continue at   hour intervals

~~Date/Time~~	~~Source~~	~~Activity~~
10:05	SP Log	PA Rad Health says "go" if NRC wants to have water dumped.
10:12	Reg I Log	Industrial Waste Discharge verified to be off.
Friday, March 30		
AM 12:05	Reg I Log	EMT asks IE/site to relay to Met. Ed that NRC says OK to release industrial waste. Notify NRC when release commences.
2:06	Reg I Log	Industrial Waste Tank over-flowing onto ground
5:35	Reg I Log	Fire in Unit One Aux. Building Basement - (picked up from intercom). Fire in ventilation system.
8:00	SP Log	Status report to EPA.
8:20	SP Log	Status report ot FDA, Bureau of Radiological Health
8:30	SP Log	Status report to Md. Rad Health
8:50	SP Log	Call from FPA requesting status
9:00	Reg I	Control room personnel (Unit 1) hear announcement that evacuation in a 10 mile radius around plant has been recommended by NRC.

Date/Time	Source	Activity
9:05	Reg. I	Onsite inspector calls Region to verify that above was true. Were told recommendation was not official.
9:25	Reg I Log	State notified of release; evacuation rumored; site does not plan to call for evacuation.
9:15 - 10:10	SP Log	SP calls PA Civil Defense re potential evacuation
9:50	PN 79-67B	~~PN 67B~~ Issued
10:00	Reg I Log	Some confusion exists because State evidently has recommended evacuation of Middletown (Doc. Collins). Plant/NRC has not recommended evacuation.
10:25	Reg I Log	Communication Lost with Unit 2 Control room
10:30 - 10:45	Reg I Log	State has advised residents within 10 miles to go inside and shut windows. NRC position remains: no evacuation/ unnecessary to take any special precautions.
10:47	Cmsr. Tel. Log	Conference call between Gossick and Commissioners: decision to send Denton to site.
11:00	Reg. I	Unit 1 Control Room becomes aware that an evacuation <u>was</u> suggested by Governor.
11:40	Cmsr. Tel. Log	Hendrie and Governor discuss evacuation.
11:45	Reg I Log	Release at 11:09 for about 15 secs.
12:03	Reg I Log	Chairman of NRC recommends that Governor of PA evacuate 5 mi. radium.

Date/Time	Source	Activity
12:07	Reg I Log	EPA, Region III advised of evacuation recommendation.
12:30	Reg I Log	The NRC evacuation recommendation is changed or is clarified as follows: pregnant women and preschool children in the 5 mi. radius should be evacuated. This recommendation to Gov. PA - not public. Dr. Langford of EPA is notified of this change.
1:00	SP Log	Another conversation with FDAA re Governor's recommendation for President to call National Security Council meeting at 1:30.
1:15 - 1:30	SP Log	Calls to MD, Delaware, NY, NJ VA, W.VA Rad Health regarding Governor's recommendation.
1:25	Cmsr. Tel. Log	Hendrie conference at White House followed by 1:30 p.m. NSC meeting.
1:30	Reg I	Another vehicle (2 HPs) departs for site.
2:00	Mossburg, Gossick Notes	Denton +12 arrive by helicopter at site; National Security Council called, President wants to talk to Denton.
2:20	Mossburg, Gossick Notes	NRR Operations Center established at nearby residence; notifications to President Carter and Governor Thornburgh.
2:30	Reg I	Director and Branch Chief plus 2 HPs dispatched by helicopter.
4:00	Reg. I	Helicopter arrives at site
(About 3:30	Unverified	Wayne Kerr +5 arrive At site to assist IE Health Physics.

~~(By this time, 83 NRC personnel are on site and in vicinity: 51 IE, 4 SP, 3 PA, 25 NRR)~~

By this time, 83 NRC personnel are on site and in vicinity (51 IE, 4 SP, 3 PA, 25 NRR).

6:30	PR No. 79-67	Press release "no imminent danger of core melt; technical experts (Denton et al) on site"

Saturday, March 31

Date/Time	Source	Activity
8:45	Reg I Log	(Madden) acting as Administrative Officer reports the following: - Trailer, manned by NRR (Denton) and the White House Communications Group is now behind the Observation Center and wired for use. - Additional Trailer by 2145 will be wired with 6 telephones. - Boyce Grier is downtown in Harrisburg at a Press Conference w/Stello, Denton, Governor Thornburgh. - Air National Guard Unit at Harrisburg Int. Airport available for assistance.
AM 1:25	SP Log	Coordinating meeting held at Capital City Airport (EPA, DOE, PA Dept. of Environmental Resources, FDA, NRC).
3:50	SP Log	Telephone call from Bettis, Radiological Assistance Team at Command Post at Capital City Airport; analytical equipment in airport hangar.
6:00	Reg I Log	Oak Ridge man believes he can use the Loose Parts Monitoring to tell the size of the bubble in the vessel.
6:25	SP Log	Call from PA Civil Defense; status update.
About 9:00		Public Affairs Center activated at site; limited operation until April 1.

176 065

Date/Time	Source	Activity
About 8:30	SP Log	Trailer moves just outside plant gate; NRR operations center in full force.
9:17	SP Log	Call from HEW asking what their role would be if evacuation necessary; NRC says PA Civil Defense has lead; status update.
9:25	SP Log	Call from CEQ; status report
9:20	SP Log	Call from NY Rad Health Bureau; status report.
10:00	SP Log	Call from Defense Civil Preparedness Agency; status report.
PM		
12:00	SD Log	Conversations with FDA-Bureau of 12:30 Radiological Health re supplies of potassium iodine.
2:00	SP Log	PA Rad Health Dept. agrees to refer all calls relating to health matters to NRC.
3:26	Cmsr. Tel. Log	Commission meets in Bethesda at Operations Center.
4:25	Cmsr. Tel. Log	Hendrie and Governor confer via telephone on status.
5:00	SP Log	NRC informed that Governor, W. Va. had activated State Radiological Assistance Team.
10:45	PN-67G	NRC representative (Stello?) at facility informed that sabotage attempt would be made during the night. FBI, PA State Police and licensee notified.

176 066

Date/Time	Source	Activity
Sunday April 1		
1:30	SP Log	SP calls DOE Command Center (they do not know where EPA people are); also calls PA Rad. Health; EPA lab is next door but no one is there.
8:13	SP Log	SP calls PA Rad Health (DOE will collect all data and transmit results to NRC). Meeting scheduled at 8:30 to set up coordination.
9:36	SP Log	Contact established by Lubenau/Vaden at PA Rad. Health offices.
11:00	Reg I Log	NRC Personnel on site (65 I&E; 27 NRR; 5 others).
	PN-67H	NRC establishes 37 TLD stations at distances from 1 to 12 miles from plant.
	PN-67H	All utilities with an operating B&W reactor are sent an NRC Bulletin to: (1) provide information on TMI-2 incident (2) require a prompt review of their plant conditions, (3) take action to prevent such an incident. NRC inspectors are being sent to each licensed B&W reactor to provide increased inspection coverage.
2:15 - 2:27	Reg I Log	President Carter is on site in Unit 2 Control Room.
8:40	Gossick Notes	NRC calls DOE/EOC to request cleanup of AUS Building... General Public Utilities requested help.
Monday April 2 AM		
3:35	Reg I Log	HQ requests licensee to send sample of containment air (2330 sample) to Bettis.

Date/Time	Source	Activity
5:50	Reg I Log	Phone link drops out/HQ will try to re-establish conference call.
6:25 P.M.	Reg I Log	Phone link re-established
Midnight	Gossick notes	Denton briefs Governor of PA.

Q. Describe the fuel damage that may have occurred during this accident.

7
1-3

A. Based on ~~the~~ a preliminary evaluation of RCS pressure and temperature during the accident, fuel assembly outlet thermocouple readings, ~~and~~ signals from the fixed incore self-powered neutron detectors acting as thermionic elements at temperatures above 700°F, and (estimates of hydrogen evolution) it is estimated that at least the upper five feet of the core was uncovered (during three ~~sim~~ periods) for a total time in the range of two to four hours. ~~It is estim~~ a preliminary estimate is that 15% to 30% of total Zircaloy inventory is oxidized, but that little or no fuel pellet melting occurred. Fuel assembly structural components, such as the control rod guide tubes, and the control rods remain intact. Additional details are provided in the attached memorandum

176 069

1400
4/3/79

The attached is a retyped version of ~~instructions~~ recommendations which were sent by Westinghouse to licensees with W reactors. It was faxed to HQ from Region I.

176 070

ATTACHMENT 1

PRELIMINARY RECOMENDATIONS

RECOMMENDATIONS:

(1) Verify that the auxiliary feedwater system is properly aligned and operable (including automatic actuation.) In the event of a loss of all "in feed flow, auxiliary" feed flow is essential for core cooling (ECCS is not intended for this condition in a Westinghouse Plant.)

(2) Verify operating procedures for failure of a relief of safety valve to close, failure of a pressurizer relief valve to reclose is considered an ASME upset condition. There procedures should recognize that the pressu' zer will fill with water and that water could be vented if containment in the pressurizer relief tank failure disc. These procedures should recognize the following points:

(a) The isolation motor operated valve should be left to stop RCS blowdown through a power-operated relief valve when RCS pressure returns to a pre-relief valva actuation pressure.

(b) Pressurizer steam bubble will continue, pressurizer will be water solid and waste relief will result, and this is to be expected.

(c) ECCS maintaining pressure, ECCS flow is necessary to maintain RCS pressure well above that corresponding to saturation temperature in hot leg or core cutlet.

(d) Heat removal and cooldown by steam generator is needed.

(e) ECCS opration should continue until cold shutdown (below 200 degrees F) reached with further heat removal by RWR.

(3) Recheck procedures for containment isolation and pumping from containment building sump to auxiliary building lielic waste storage tanks, I.E., sump pump operation.

(4) Procedures should assure minimized accessibility in auxiliary building equipment in the event of radioactive water in auxiliary building systems.

(5) Review the plant procedure regarding control of hydrogen in containment.

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TMI DOCUMENTS

DOCUMENT NO: TM-0184

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Wilda R. Mullinix, NRC

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THREE MILE ISLAND NUCLEAR STATION

STATION CHEMISTRY PROCEDURE 1810.2
NPDES - OPERATIONS RESPONSIBILITY

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Unit 1 Staff Recommends Approval Approval R.W. Dubiel Date 6/19/78 Cognizant Dept. Head	Unit 2 Staff Recommends Approval Approval R.W. Dubiel Date 6/19/78 Cognizant Dept. Head
Unit 1 PORC Recommends Approval N/A Date — Chairman of PORC	Unit 2 PORC Recommends Approval N/A Date — Chairman of PORC
Unit 1 Superintendent Approval J.P. O'Hanlon Date 6-21-78	Unit 2 Superintendent Approval J.L. Seelinger Date 6/23/78

Manager Generation Quality Assurance Approval NA Date —

STATION CHEMISTRY PROCEDURE 1810.2

NPDES - OPERATIONS RESPONSIBILITY

1.0 PURPOSE

The purpose of this procedure is to outline the Operations Department requirements for compliance with the National Pollutant Discharge Elimination System (NPDES) Permit.

2.0 DISCUSSION

Operations responsibility in compliance with the NPDES is in the area of monitoring and reporting of the following parameters for the effluent points discussed in the permit: Flow, Temperature, and Heat Rejection. The Operations Department is also responsible for proper operation of the systems which effect the above parameters as well as the water quality of the process fluid effluent from the plant. Unit I operations shall be responsible for everything in this procedure except; The Unit II Neutralizing Tank and the Unit II Mechanical Draft Cooling Tower.

3.0 REFERENCES

3.1 NPDES Permit 000992

3.2 OP 1104-18

3.3 OP 1104-37

3.4 OP 1104-40

3.5 OP 2104-2.11

3.6 OP 2104-3.8

3.7 OP 2104-2.5

4.0 EQUIPMENT

None Required

5.0 PROCEDURE

5.1 Monitoring Requirements

NOTE: See Figure 1810.2-3

5.1.1 001 Combined-Mechanical Draft Cooling Tower Blowdowns

5.1.1.1 Flow - Flow is measured by a continuous strip chart recorder FR-146 on panel PLF in the Unit 1 control room. An integrator is built into the recorder.

5.1.1.2 Station Temperature - Station Temperature is recorded on TR-896 on panel PLF in the Unit 1 control room. (Red Pen)

5.1.1.3 Heat Rejection - This parameter has no requirement for routinely reporting; this limit cannot be exceeded without exceeding the temperature limits of plant effluent.

5.1.2 002, 003, 004 Emergency Outfalls

5.1.2.1 Flow will be calculated every 2 hours when any change in river water pump combination is made via the calculational method given in OP 1104-37 Mechanical Draft Cooling Tower and 2104-3.8 Mechanical Draft Cooling Tower.

5.1.2.2 Temperature will be monitored using a thermometer immersed in the effluent stream 5 times per day as directed in OP 1104-37 and OP 2104-3.8.

5.1.3 101 Treated Sewage Effluent - Not presently operational; flow will be measured via integrator and flow meter.

5.1.4 103 Preoperational Cleaning and Flushing Settling Basin - Flow is measured using manufacturer's pump curves. Discharge is in accordance with Special Operating Procedures.

5.1.5 104 - Matz - When Waste Treatment facility is complete, no direct discharges are expected. Until such time this discharge point is controlled by interium measures to limit impact of discharge. Flow is estimated with "stop watch and bucket".

5.1.6 105 Neutralizing Tank Discharge - Flow is monitored by measuring the difference in level when the tank is drained using LI-166 on the IWT panel.

5.1.7 107 Waste Treatment Facility Effluent - Flow will be monitored using a flow meter. (FM 342)

5.1.8 108 Waste Neutralizing Tank Discharge (Unit 2) - Flow is monitored by a local flow indicator FI - Later.

5.2 Compliance With Discharge Limitations - Implementation and Reporting

NOTE: Implementation of measures required to assure compliance with the NPDES permit are contained in both operating procedures and response to alarms.

5.2.1 001 Combined Mechanical Draft Cooling Tower Blowdowns - OP 1104-37 Mechanical Draft Cooling Tower and 2104-3.8 Mechanical Draft Cooling Tower are the implementing procedures to assure operational compliance relative to not exceeding the temperature of $87^{\circ}F$ discharge. Reporting of flow and temperature is in accordance with the "Monthly Data Report" Form 1810.2-1. Assurance that the operator will be aware of the high temperature ($87^{\circ}F$) condition is via an alarm in the control room.

5.2.2 002, 003, 004 Emergency Outfalls - OP 1104-37, Mechanical Draft Cooling Tower, and 2104-3.8 Mechanical Draft Cooling Tower implement the requirement to calculate flow and record temperature for this point. Since this discharge point is expected to be rarely used reporting data sheets will be devised at the time as the need arises.

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5.2.3 101 Treated Sewage Effluent - An operating procedure will be developed for this equipment prior to its operation.

5.2.4 103 Pre-Operational Cleaning and Flushing Settling Basin - Special operating procedures control the proper discharge from this point and specify flows to be recorded.

5.2.5 104 - Matz - When Waste Treatment facility is complete, no direct discharges are expected. Until such time this discharge point is controlled by interium measures to limit impact of discharge. Flow is estimated with "stop watch and bucket".

5.2.5.1 Turbine Building Sump Discharge - OP 1104-40 governs the operation of the turbine building sump pumps SD-P5. Reporting of flow is via the "Monthly Data Report" Form 1810.2-2.

5.2.5.2 <u>105 Waste Neutralizing Tank Discharge</u> - The discharge of the Neutralizing Tank is in accordance with OP 1104-18 "Discharge of Turbine Plant Neutralizing Tank". Reporting of flows is via the "Monthly Data Report" Form 1810.2-4.

5.2.5.3 <u>107 Waste Treatment Facility Discharge</u> - Flows are reported using Form 1810.2-2.

5.2.5.4 108 Waste Neutralizing Tank Discharge (Unit 2) - The Discharge of the neutralizing tank is in accordance with 2104-2.11. Reporting of flows is via the "Montlhy Data Report Form 1810.2-4.

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NPDES MONTHLY REPORT - OPERATIONS OUTFALL
<u>001 PLANT EFFLUENT</u>

DAY	FROM CRO LOG Plant Effluent ① INTEGRATOR READING	② DIFFERENCE
	Total ③	

MAX. Column 2 ______________ Gal/Day x $\frac{\text{M.Gal.}}{10^6 \text{ gal.}}$ = ______________ MGD

MIN. Column 2 ______________ Gal/Day x $\frac{\text{M. Gal.}}{10^6 \text{ gal.}}$ = ______________ MGD

Average = Last Day Month (Col. 1 bottom) = ______________ (00000)
Last Day Prev. Mo. (Col. 1 top) = ______________ (00000)

Average = $\frac{\text{Difference 3}}{\text{\# Days in Mo.}}$ ______________ = ______________ $\frac{\text{gal.}}{\text{day}}$ x $\frac{\text{M. Gal.}}{10^6 \text{ gal.}}$ = ______ MGD

cc: Administrator - TMI
NPDES File

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OUTFALL 001 PLANT EFFLUENT

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4 Max. and Min. Discharge Temperature:

Pick off max. and min. "Station Effluent Temp." from previous month's 24 hr. daily log computer printouts

MAX = ________ °F

MIN = ________ °F

Average Discharge Temperature:

$$\text{Average} = \frac{\text{Sum of ALL "Station Effluent Temp" Readings for Entire Month}}{\text{\# Days in Month X 24}} = \frac{\_\_\_\_\_\_\_\_}{\_\_\_\_\_\_\_\_ \text{ days X 24}} = \_\_\_\_\_\_\_\_ \text{ °F}$$

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176 079

UNIT #1
TURBINE
BUILDING
SUMP PUMP
DISCHARGE.
#107
Data Sheet
1810.2-2

DATE	SD-P5 INTEGRATOR READING	DIFFERENCE	IWTS INTEGRATOR READING	DIFFERENCE	TOTAL GAL. PER DAY
Diff. for Mo.				TOTAL	

Total IWTS Gal. based on Integrator Difference ______________.

MIN: (from Col. 2) ____________ $\times \frac{\text{M Gal.}}{10^6}$ = ____________ MGD

MAX: (from Col. 2) ____________ $\times \frac{\text{M Gal.}}{10^6}$ = ____________ MGD

AVERAGE = $\frac{\text{Tot. Gal. IWTS 3}}{\text{Days in mo.}}$ = ____________ $\times \frac{\text{M. Gal.}}{10^6}$ = ____________ MGD

7.0

FLOW AND TEMPERATURE MONITORING FREQUENCY - NPDES PERMIT

Form 1810-2.3

OUTFALL	FLOW	TEMPERATURE	HEAT REJECTION
001	Continuous Monitor	Continuous Record (2) (3)	Calculated
002 (1)	Continuous Calculated	Immersion Stabilization 5/Day (3)	
003 (1)	Continuous Calculated	Immersion Stabilization 5/Day (3)	
004 (1)	Continuous Calculated	Immersion Stabilization 5/Day (3)	
101	Measured 2/Month		
103	Measured 2/Month		
104	Measured 2/Month		
105	Measured 2/Month		
107	Continuous		
108	Measured 2/Month		

(1) EPA must be notified within 48 hours after discharging from these outfalls.

(2) Heat rejected to river shall not exceed 758 X 10^6 BTU/Hour or a maximum effluent temperature of 87^oF.

(3) The discharge 001, 002, 003, or 004 shall not cause a rise in the river temperature of more than 5^oF above the ambient or a maximum of 87^oF whichever is less; not to be changed by more than 2^oF during any one hour period.

8.0

1176 081

#105 - NEUTRALIZED WASTE TANK DISCHARGE

Data Sheet # 1810.2-4

# Days	Discharge Date	From OP1104-18 Data shs. from IWT Operators Desk Neut. Tank Level ① Empty Tk. Level (ft.)	② Full Tk. Level (ft.)	② Minus ① = Diff. ③ ③ Difference (ft.)	③ x 3396 $\frac{gal}{ft.}$ = ④ gallons Discharg. ④ Gals. Discharged	⑤ Time start	⑥ Time end	⑦ Diff. (min.)
				⑩		⑧	Total	

0.6

Max. & Min. Flows:

Based on Max. Time to Drain Tank

MIN. FLOW per day from Col. 4 = Min. Gal. in any day = _____gal. x $\frac{M.\ Gal.}{10^6\ Gal.}$ = _____MGD

MAX. FLOW per day from Col. 4 = Max. Gal. in any day = _____gal. x $\frac{M.\ Gal.}{10^6\ Gal.}$ = _____MGD

Average Flow:

Average = $\frac{\text{Total gals. disch. (10)}}{\text{Total calendar days in which discharges occurred.}}$ = _____________ $\frac{gal.}{days}$ x $\frac{M.\ Gal.}{10^6\ Gal.}$ = _____________MGD

Total time discharge occurred (from 8 total) = __________Minutes

cc: TMI ADMINISTRATOR
NPDES FILE

1810.2
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DISCHARGE 108 UNIT 2 NEUTRALIZED WASTE TANK DISCHARGE

Form 1810.2-5

Discharge Date	From Release Permit OP 2104-2.11 ① Tank Level Start (Gal.)	② Tank Level Stop (Gal.)	① minus ② = ③ ③ Gallons Disch.	OP 2104-2.11 ④ Time Started	⑤ Time Ended	⑥ Difference (Min.)
		TOTAL ⑦			TOTAL ⑧	

Max. & Min. Flows:

Based on Max. MIN. FLOW per day from Col. 3 = Min. Gal. in any day = ____gal. x $\frac{\text{M. Gal.}}{10^6 \text{ Gal.}}$ = _______MGD

Gallons Discharged MAX. FLOW per day from Col. 3 = Max. Gal. in any day = ____gal. x $\frac{\text{M. Gal.}}{10^6 \text{ Gal.}}$ = _______MGD

Average Flow:

Average = $\frac{\text{Total gals. disch. (7)}}{\text{Total calendar days in which discharges occurred.}}$ = __________ $\frac{\text{gal.}}{\text{days}}$ x $\frac{\text{M. Gal.}}{10^6 \text{ Gal.}}$ = _______________MGD

Total time discharge occurred (from 8 total) = _________Minutes

cc: TMI ADMINISTRATOR
NPDES FILE

1810.2
Revision 2
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10.0

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