GPU 2066

Complete compendium attached

table of effected pgs only

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2202-1.1
Revision 6
10/25/78

2066

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-1.1
REACTOR TRIP

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5.0			30.0			55.0		
6.0			31.0			56.0		
7.0			32.0			57.0		
8.0			33.0			58.0		
9.0			34.0			59.0		
10.0			35.0			60.0		
11.0			36.0			61.0		
12.0			37.0			62.0		
13.0			38.0			63.0		
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23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

D P Jarrett Date 10/25/78

V- Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

D L Rhoades Date 10/25/78

Manager Generation Quality Assurance Approval NA Date —

TMI 1S-4 Rev 5/77

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06/08/78

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-1.2
UNANTICIPATED CRITICALITY

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6.0			31.0			56.0		
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23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

J.F. Hilbish mdb Date 6/3/7

V-Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendant Approval

J.J. Dieckamp Date 6/8/7

A

Manager Generation Quality Assurance Approval NA Date —

TMI 55-A Rev 8/77

2202-1.3
Revision 11
10/06/78

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-1.3

LOSS OF REACTOR COOLANT/REACTOR COOLANT SYSTEM PRESSURE

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20.0	12/30/77	3						
21.0	12/30/77	3						
22.0	12/30/77	3						
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Unit 1 Staff Recommends Approval

Approval N/A Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

N/A Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

R.P. Warren Date 10/6/78

V - Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

J.J. Barton Date 10/6/78

Manager Generation Quality Assurance Approval N/A Date —

TMI 55-A Rev 8/77

2202-1.6
Revision 2
11/15/77

THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-1.6

HIGH ACTIVITY IN REACTOR COOLANT

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23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

[signature] Date 11.14.77

V/ Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

[signature] Date 11/15/77

Manager Generation Quality Assurance Approval NA Date —

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Revision 2
11/15/77

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-1.7
EXCESSIVE RADIATION LEVELS

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14.0	08/13/75	0						
15.0	08/13/75	0						
16.0	11/15/77	2						
17.0	08/13/75	0						
18.0	08/13/75	0						
19.0	08/13/75	0						

Unit 1 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

Q.T. Dilworth Date 11/14/77
V/ Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

J. J. Seelinger Date 11/15/77

Manager Generation Quality Assurance Approval NA Date —

TMI 58-A Rev 9/77

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2202-1.4
Revision 8
02/08/79

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-1.4
LOSS OF RC FLOW/RC PUMP TRIP

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6.0			31.0			56.0		
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24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

R P Warren Date 2/7/79

Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

J. B. Logan Date 2/8/79

Manager Generation Quality Assurance Approval NA Date —

TMI 55-A Rev 8/77

2202-1.5
Revision 3
09/29/78
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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-1.5
PRESSURIZER SYSTEM FAILURE

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3.0	06/22/77	1	28.0			53.0		
4.0	11/17/77	2	29.0			54.0		
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6.0	03/11/77	0	31.0			56.0		
7.0	03/11/77	0	32.0			57.0		
8.0	03/11/77	0	33.0			58.0		
9.0	03/11/77	0	34.0			59.0		
10.0	03/11/77	0	35.0			60.0		
11.0	03/11/77	0	36.0			61.0		
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23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

J F Wilson Date 9/

Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

J F Wolfinger Date 9/

Manager Generation Quality Assurance Approval NA Date —

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01/05/78

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-1.8
LOSS OF DECAY HEAT REMOVAL

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4.0	01/05/78	2						
5.0	01/05/78	2						

Unit 1 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

J F Wilson Date 1-5-78
Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

J J Colburn Date 1/5/78

Manager Generation Quality Assurance Approval NA Date —

TMI 58-A Rev 8/77

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09/01/78

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-1.9
LOSS OF INTERMEDIATE CLOSED COOLING WATER

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24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

RP Warren Date 8/30/78

V - Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

J I Seelinger Date 9/1/78

Manager Generation Quality Assurance Approval NA Date —

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12/23/78

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-1.10
SHUTDOWN FROM OUTSIDE THE CONTROL ROOM

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23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

J. F. Hellosh Date 12-21
V. Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

J. B. Lynch Date 12/23

Manager Generation Quality Assurance Approval NA Date —

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06/21/78

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-2.1
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5.0	06/17/77	1						
6.0	12/30/77	3						
7.0	01/29/76	0						
8.0	03/30/78	6						

Unit 1 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

J. F. Wilson Date 6-20-78

Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

J. J. Colitz Date 6/21/78

Manager Generation Quality Assurance Approval NA Date —

TMI 55-A Rev 8/77

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2202-2.2
Revision 3
10/13/78

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-2.2
LOSS OF STEAM GENERATOR FEED

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4.0	10/13/78	3	29.0			54.0		
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23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

RP Warren Date 10/13/78
Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

J. Knee Jr. Date 10/13/78

Manager Generation Quality Assurance Approval NA Date —

TMI 1S-A Rev 8/77

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CENTRAL FILE

2202-2.3
Revision 3
02/10/78

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-2.3
LOSS OF INSTRUMENT AIR AND SERVICE AIR

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Unit 1 Staff Recommends Approval

Approval N/A Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

N/A Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

J. F. Wilson Date 2-8-78

Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

J. J. Seelinger Date 2/10/78

Manager Generation Quality Assurance Approval N/A Date —

TMI 55-A Rev 9/77

6/(m) 120

2203-2.3
Revision 5
10/06/78

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-2.3
STEAM SUPPLY SYSTEM RUPTURE

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22.0			47.0			72.0		
23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

R P Warren Date 10/6/78
V-Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

C. A. Mecklenburg Date 10/6/78

Manager Generation Quality Assurance Approval NA Date —

TMI 55-A Rev 8/77

Central file 6/(illegible) /77

2203-2.4
Revision 1
06/20/77

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-2.4
SECONDARY SERVICES CLOSED COOLING WATER FAILURE

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24.0			49.0			75.0		
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Unit 1 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —

Chairman of PORC

PORC comments of — included

(date)

By — Date —

Unit 2 PORC Recommends Approval

J A Marshall Jr Date 6-20-77

Chairman of PORC

PORC comments of — included

(date)

By — Date —

Approval H Hukkeller Date 6/20/77

Station Superintendent/
Unit Superintendent

TMI-6B-A 11-74

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2202-2.5
Revision 6
09/22/78

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 PLANT EMERGENCY PROCEDURE #2202-2.5

Station Blackout With Loss of Both Diesel Generators

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23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

N/A Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

J. F. Colvin Date 9/22/78
Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

J. F. Colvin Date 9/22/78

Manager Generation Quality Assurance Approval N/A Date —

TMI 55-A Rev 8/77

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2202-2.6
Revision 4
10/06/78

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-2.6

OTSG TUBE RUPTURE
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23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

N/A Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

R P Warren Date 10/4/7
V - Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

J. J. Colitz Date 10/4/7

Manager Generation Quality Assurance Approval N/A Date —

TMI 55-A Rev 3/77

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2202-2.7
Revision 2
02/08/79

THREE MILE ISLAND NUCLEAR STATION
EMERGENCY PROCEDURE 2202-2.7
HIGH CATION CONDUCTIVITY AND/OR SODIUM IN THE CONDENSATE AND/OR FEEDWATER SYSTEM
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23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

RP Warren Date 2/8/79

V- Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

[signature] Date 2/7/79

Manager Generation Quality Assurance Approval NA Date —

TMI 58-A Rev 8/77

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2202-2.8
Revision 1
07/19/78

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-2.8
LOSS OF ONE AUXILIARY TRANSFORMER

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24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

RP Date 7/18/78

Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

Date 7/19/78

Manager Generation Quality Assurance Approval NA Date —

TMI 55-A Rev 8/77

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05/19/78

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THREE MILE ISLAND NUCLEAR STATION
UNIT 2 EMERGENCY PROCEDURE 2202-2.9

PLANT RESPONSE TO PENETRATION OF THE PROTECTED AREA

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24.0			49.0			74.0		
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Unit 1 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

J. F. Gilbish Date 5/19

Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

D. L. Lindsey Date 5/19

Manager Generation Quality Assurance Approval NA Date —

TMI 55-A Rev 3/--

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Revision 4
10/13/78

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-3.1
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23.0			48.0			73.0		
24.0			49.0			74.0		
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Unit 1 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

N/A Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

R.P. Warren Date 10/6/78
V- Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

J. J. Colvin Date 10/6/78

Manager Generation Quality Assurance Approval N/A Date —

TMI 55-A Rev 8/

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09/12/78

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-3.3
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Unit 1 Staff Recommends Approval

Approval N/A Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

N/A Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

R P Warren Date 9/11

V- Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

[signature] Date 9/11

Manager Generation Quality Assurance Approval N/A Date —

TMI 55-A Rev 8/

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Revision 5
09/12/78

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-1.1
LOSS OF BORON (MODERATOR DILUTION)

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Unit 1 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

R P Warren Date 9/11/78
V- Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

J L Lee Jr Date 9/12/78

Manager Generation Quality Assurance Approval NA Date —

TMI 55-A Rev 8/77

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05/04/78

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-1.4
RCP & MOTOR EMERGENCIES

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Unit 1 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

N/A Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

J. F. Wilson Date 5/2/78
V. Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

R. F. Wilson Date 5/4/78

Manager Generation Quality Assurance Approval N/A Date —

TMI 55-A Rev 8/77

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Revision 2
09/22/78

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-1.2
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Unit 1 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

N/A Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

J F Ulrich Date 9/22/78
Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

G J Kuehnigen Date 9/22/78

Manager Generation Quality Assurance Approval N/A Date —

TMI 55-A Rev 8/77

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6/1(m)
138

2203-1.3
Revision 1
03/20/79

THREE MILE ISLAND NUCLEAR STATION

UNIT #2 ABNORMAL PROCEDURE 2203-1.3

CRD MALFUNCTION ACTIONS ASYMMETRIC ROD FAULT (6.5%)

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5.0	03/20/79	1	30.0			55.0		
6.0	05/12/77	0	31.0			56.0		
7.0	03/20/79	1	32.0			57.0		
8.0	05/12/77	0	33.0			58.0		
9.0	05/12/77	0	34.0			59.0		
10.0	05/12/77	0	35.0			60.0		
11.0	05/12/77	0	36.0			61.0		
12.0	05/12/77	0	37.0			62.0		
13.0			38.0			63.0		
14.0			39.0			64.0		
15.0			40.0			65.0		
16.0			41.0			66.0		
17.0			42.0			67.0		
18.0			43.0			68.0		
19.0			44.0			69.0		
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22.0			47.0			72.0		
23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval N/A Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

N/A Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

R P Warren Date 3/20/79

V. Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

C R Fairman Date 3/21/79

Manager Generation Quality Assurance Approval N/A Date —

TMI 58-A Rev 9/77

6/1/m; 1F3

2203-1.5
Revision 2
09/07/78

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-1.5
LOSS OF RC MAKEUP

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24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

N/A Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

R P Warren Date 9/1/78
V. Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

J. K. Kelley Date 9/7/78

Manager Generation Quality Assurance Approval N/A Date —

TMI 88-A Rev 8/77

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Revision 1
10/06/78

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-1.6
NUCLEAR SERVICE CLOSED COOLING WATER FAILURE

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23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

R.P. Jarrett Date 10/6/78

V- Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

J.J. Colitz Date 10/6/78

Manager Generation Quality Assurance Approval NA Date —

TMI 55-A Rev 3/77

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Revision 0
2203-1.7
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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2203-1.7
NUCLEAR SERVICES RIVER WATER FAILURE

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Unit 1 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

N/A Date —
Chairman of PORC

PORC comments of — (date) included

By — Date —

Unit 2 PORC Recommends Approval

Date —
Chairman of PORC

PORC comments of — (date) included

By — Date —

Approved [signature] Date 2-3-77
Unit Superintendent

TMI-83-A 11-77

5/1/79

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-1.8
VIBRATION AND LOOSE PARTS MONITORING SYSTEM

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24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval N/A Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

N/A Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

[signature] Date 10/25/78

V- Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

[signature] Date 10/25/78

Manager Generation Quality Assurance Approval N/A Date —

TMI 55-A Rev 5/77

6/1(m) 144

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06/08/78

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 ABNORMAL PROCEDURE 2203-2.1
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25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval N/A Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

N/A Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

J. F. Colburn Date 6/3/78

Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

J. J. Barton Date 6/7/78

Manager Generation Quality Assurance Approval N/A Date —

TMI 55-A Rev 8/77

G/1(m) 145

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10/25/78

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-2.2
TURBINE TRIP

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Unit 1 Staff Recommends Approval

Approval N/A Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

N/A Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

R P Warren Date 10/25/78

Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

R. J. Lieberman Date 10/25/78

Manager Generation Quality Assurance Approval N/A Date —

TMI 58-A Rev 3/77

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02/24/77

Central File

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-2.5
CONTROL ROOM HVAC FAILURE

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Unit 1 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —
Chairman of PORC

PORC comments of — included
(date)

By — Date —

Unit 2 PORC Recommends Approval

Norman W. Curtis Date 2-23-77
Chairman of PORC

PORC comments of — included
(date)

By — Date —

Approval [redacted] Date 2-24-77
Unit Superintendent

TMI-SS-A 11-7

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G/km
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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-2.6

POST ACCIDENT HYDROGEN CONTROL

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Unit 1 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

N/A Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

J. F. Fuller Date 6-22-78
V. Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

D. F. Lieberman Date 6/22/78

Manager Generation Quality Assurance Approval N/A Date —

TMI 55-A Rev 8/77

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Revision 6
02/13/78

THREE MILE ISLAND NUCLEAR STATION
STATION RADIATION EMERGENCY PROCEDURE 1670 CENTRAL FILE
LOCAL EMERGENCY PROCEDURE

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Unit 1 Staff Recommends Approval
Approval _____ Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval
Approval _____ Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval
H. Kunder Date 2/10/78
Chairman of PORC

Unit 2 PORC Recommends Approval
_____ Date 2.13.78
Chairman of PORC

Unit 1 Superintendent Approval
J. P. O'Hanlon Date 2/15/78

Unit 2 Superintendent Approval
_____ Date 2/13/78

Manager Generation Quality Assurance Approval
NA Date —

TMI 33-A Rev 8/77

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Revision 9
11/22/78

THREE MILE ISLAND NUCLEAR STATION
STATION RADIATION EMERGENCY PROCEDURE 1670.2
SITE EMERGENCY PROCEDURE

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Unit 1 Staff Recommends Approval

Approval N/A Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

[signature] Date 11/16/78

Chairman of PORC

Unit 2 PORC Recommends Approval

R P Warren Date 11/20/78

V- Chairman of PORC

Unit 1 Superintendent Approval

[signature] Date 11/22/78

Unit 2 Superintendent Approval

[signature] Date 11/20/78

Manager Generation Quality Assurance Approval NA Date —

TMI 15-A Rev 5/77

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Revision 5
02/13/78

THREE MILE ISLAND NUCLEAR STATION
STATION HEALTH PHYSICS PROCEDURE 1670.3
EMERGENCY PLAN, VOLUME 1
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3.0	02/13/78	6	28.0			53.0		
4.0	02/13/78	6	29.0			54.0		
5.0	02/13/78	6	30.0			55.0		
6.0	02/13/78	6	31.0			56.0		
7.0			32.0			57.0		
8.0			33.0			58.0		
9.0			34.0			59.0		
10.0			35.0			60.0		
11.0			36.0			61.0		
12.0			37.0			62.0		
13.0			38.0			63.0		
14.0			39.0			64.0		
15.0			40.0			65.0		
16.0			41.0			66.0		
17.0			42.0			67.0		
18.0			43.0			68.0		
19.0			44.0			69.0		
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21.0			46.0			71.0		
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23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

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Approval N/A Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

H. Kunder Date 2/15/78

Chairman of PORC

Unit 2 PORC Recommends Approval

[signature] Date 2.15.78

V. Chairman of PORC

Unit 1 Superintendent Approval

[signature] Date 2/15/78

Unit 2 Superintendent Approval

[signature] Date 2/15/78

Manager Generation Quality Assurance Approval N/A Date —

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2202-1.1
Revision 6
10/25/78

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-1.1
REACTOR TRIP

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10.0			35.0			60.0		
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24.0			49.0			74.0		
25.0			50.0			75.0		

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Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

D P Jarrett Date 10/25/78

V- Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

J. J. Colitz Date 10/25/78

Manager Generation Quality Assurance Approval NA Date —

THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-1.1

REACTOR TRIP

1.0 SYMPTOMS

1.1 Reactor trip may be due to any of the following:

a. Nuclear power > 105.5% of rated power.

b. Nuclear power > 1.05 times reactor coolant flow (%) minus power imbalance reduction.

c. Nuclear Power > 50.9% of rated power during operation with one R.C. Pump operating per loop.

d. Loss of two Reactor Coolant Pumps in one reactor coolant loop.

e. Reactor coolant pressure > 2355 psig.

f. Reactor coolant pressure < 1900 psig.

g. Reactor coolant pressure < (13.00 T_{out}-5887)

h. Reactor coolant outlet temperature > 619°F.

i. Reactor building pressure > 4 psig.

j. Manual reactor trip actuation.

1.2 Individual and group "In Limit" lights actuate.

1.3 Rapid decrease in neutron level as indicated by the Nuclear Instrumentation.

1.4 Main turbine master trip alarm energized.

1.5 Rapid decrease in Unit Load.

2.0 IMMEDIATE ACTION

2.1 Automatic Action

a. All rods insert (except Group 8)

b. Turbine trip, stop valves close

c. Generator breakers open

d. Turbine by-pass valves open

e. Source range nuclear instrumentation high voltage energizes at 5 X 10^{-10} amps on N1-3 or N1-4.

2.2 Manual Action

a. Manually TRIP the reactor

b. Verify all "in-limit" lights are actuated (except group 8)

c. Verify that turbine has tripped and generator breakers are open - if turbine has NOT tripped - manually TRIP the turbine, start lift pumps & turning gear oil pump.

d. CLOSE letdown isolation valve MU-V376.

e. START second make-up pump. Open MU-V-16B as necessary to maintain 100" in PZR.

f. Verify turbine bypass control valves are maintaining header pressure at 1010 psig.

g. If any feedwater stations are in hand, run stations back to maintain OTSG at 30".

h. Verify PZR Htrs are off @ 80" in PZR.

i. If PZR level gets to 20" open DH-V5B and start 3rd make-up pump and open MU-V16C/D to increase PZR level.

j. If any of the following ICS stations are in Hand (Steam Generator/Reactor Demand, either Feedwater Demand, Main or Startup Feedwater Valve Demand, Feedpump Speed, Reactor Master, and/or Diamond) runback the appropriate ICS stations.

3.0 FOLLOW-UP ACTION

3.1 Announce-on page system - Reactor Trip

3.2 Initiate emergency boration per 2203-1.1 (Loss of Boron) if reactor power is not below 10% in 1 minute.

3.3 Monitor make-up tank level and maintain level greater than 55" by using Waste Transfer Pumps (WDL-P5A/B) and feeding from an RCBT with boron concentration greater or equal to RCS concentration.

3.4 Verify that the pressurizer heaters and spray have returned RCS pressure to normal operating pressure of 2155 psig.

3.5 Reduce pressurizer level setpoint to 100" (25%).

3.6 Verify normal electrical lineup, i.e., no substation or in-plant distribution breakers are open (except generator breakers).

3.7 Check that all RMS channels are normal and that no unplanned or uncontrolled radioactive release is in progress.

3.8 Compute shutdown margin calculations per 2103-1.9 (Reactivity Balance Calculations). If shutdown margin is less than 1%, boron should be added to the RCS.

3.9 If reactor start-up is not intended within four hours raise OTSG level to 97-99% on the operating range using the feedwater valve bypass and tube sheet drains for level control.

3.10 Fill out a Reactor Trip report.

3.11 Notify H.P./Chemistry to sample R.C. Letdown for Dose Equivalent Iodine between 2 and 6 hours after power change of greater than 15% within one hour period per Surveillance Procedure #2304-302.

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-1.2
UNANTICIPATED CRITICALITY

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24.0			49.0			74.0		
25.0			50.0			75.0		

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Approval NA Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

J.F. Hilsh Date 6/3/78

V-Chairman of PORC MBD

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

J. J. Lichinger Date 6/8/78

Manager Generation Quality Assurance Approval NA Date —

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-1.2

UNANTICIPATED CRITICALITY

1.0 SYMPTOMS

1.1 Unanticipated increase in count rate.

1.2 Acutation of control rod out motion inhibit alarms (intermediate range set at 3 DPM, resets at 0.5 DPM; source range set at 2 DPM, resets at 0.5 DPM).

2.0 IMMEDIATE ACTION

2.1 Automatic Action.

a. Withdrawal of control rods stop (only if alarm actuated).

2.2 Manual Action.

a. Immediately start insertion of control rods.

b. Monitor count rate until shutdown by 1% $\Delta k/k$.

c. Trip the reactor, if necessary.

d. Ensure there is no dilution flow into the Reactor Coolant System.

3.0 FOLLOW-UP ACTION

3.1 The Shutdown Margin shall be $\geq$1% $\Delta k/k$. (T.S. 3.1.1.1) Emergency Borate if necessary per 2203-1.1.

3.2 Determine cause for unanticipated criticality.

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-1.3

LOSS OF REACTOR COOLANT/REACTOR COOLANT SYSTEM PRESSURE

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18.0	04/18/78	7						
19.0	12/30/77	3						
20.0	12/30/77	3						
21.0	12/30/77	3						
22.0	12/30/77	3						
23.0	06/22/77	1						

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Cognizant Dept. Head

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Cognizant Dept. Head

Unit 1 PORC Recommends Approval

N/A Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

R D Warren Date 10/6/78

V Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

G K Hovey Date 10/6/78

Manager Generation Quality Assurance Approval N/A Date —

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-1.3

LOSS OF REACTOR COOLANT/REACTOR COOLANT SYSTEM

PRESSURE

A. Leak or Rupture Within Capability of System Operation.

1.0 SYMPTOMS

1.1 Initial loss of reactor coolant pressure & decrease in pressurizer level becoming stable after short period of time.

1.2 Possible reactor building high radiation and/or temp. alarm.

1.3 Possible reactor building sump high level alarm.

1.4 Make-up tank level decreasing > 1" in 3 min.

1.5 Possible makeup line high flow alarm.

1.6 RB Fan Drip Pan Level Hi Alarms.

NOTE: The operator may distinguish between a loss of coolant inside containment, an OTSG tube rupture and a steam line break by the following symptoms which are unique to the aforementioned accidents.

1. Loss of coolant inside Rx Bldg. - particulate, iodine & gas monitor alarm on HP-R-227 "Reactor Building Air Sample".
2. OTSG tube rupture - gas monitor alarm on VA-R-748.
3. Steam line break
   (1) Low condensate storage tank level alarm - and or low hot well level alarm.
   (2) FW Latch System Actuation.

2.0 IMMEDIATE ACTION

2.1 Automatic Action:

2.1.1 MU-V17 will open to compensate for reduced pressurizer level.

2.1.2 Additional pressurizer heaters will come on in response to reduced reactor coolant pressure.

2.2 Manual Action

2.2.1 Verify MU-V17 open and pressurizer heaters on.

2.2.2 "CLOSE" MU-V376 letdown isolation valve, & "START" the backup MU pump, if required.

2.2.3 Reduce load at 10% minute & proceed with normal shutdown.

2.2.4 "LINE-UP" waste transfer pump from a R.C. Bleed Holdup Tank & pump to the makeup tank to maintain required level.

2.2.5 If for any reason the operator cannot maintain Make-up Tank and Pressurizer levels above their respective low level alarm setpoints, "TRIP" the reactor, "INITIATE" Safety Injection manually (push buttons on panel 3), & then "Close" MU-V12.

3.0 FOLLOW UP ACTION

3.1 Safety Injection Not Initiated.

3.1.1 Initiate unit shutdown & cooldown per 2102-3.1 and 2102-3.2 respectively.

3.2 Safety Injection Manually Initiated (HPI and LPI).

3.2.1 Verify that the Makeup Pumps & Decay Heat Removal Pumps start satisfactorily.

3.2.1.1 Close MU-V12 and MU-V18.

3.2.2 Bypass the SAFETY INJECTION by DEPRESSING the Group Reset Pushbuttons & "THROTTLE" MU-V16A/B/C/D as necessary to maintain 220" pressurizer level and not exceed 250 GPM/HPI flow leg.

3.2.3 If MU pump flow drops below 95 GPM, trip excess MU pumps.

NOTE: HPI String A flow is the sum of MU 23 FE1&2. HPI String B flow is the sum of MU23 FE 3&4.

3.2.4 Verify that Safety Injection equipment is in its ESF position as shown in Table A-1.

3.2.5 CAUTION: Continued operation depends upon the capability to maintain pressurizer level and RCS pressure above the 1640 PSIG Safety Injection Actuation setpoint.

1. If pressurizer level can be maintained above the low level alarm point and the RCS pressure above the Safety Injection Actuation point, then proceed to step 3.2.6.
2. If pressurizer level cannot be maintained above the low level alarm point and the RCS pressure above the Safety Injection Actuation point, then the plant has suffered a major rupture and operation should continue according to Part B - Leak or Rupture of Significant Size Such that Engineered Safety Features Systems are Automatically Initiated.

3.2.6 With the pressurizer level and RCS pressure being maintained within allowable limits, initiate plant shutdown and cooldown per 2102-3.1 and 2102-3.2, respectively.

NOTE: The HPI System is being used for makeup control and valves MU-V16A/B/C/D will have to be throttled to maintain pressurizer level. As RCS pressure decreases, it may be possible to return to the normal makeup flowpath and secure HPI. If MU pump flow drops below 95 GPM as a result of throttling, "Open" MU-V36 & 37 to provide MU pump recirculation path to MU Tank. Monitor MU Tank level and open MU-V12 as required.

3.2.7 At the time the DH System is to be brought on line for normal cooling only one DH string should be used for decay heat removal (i.e. - recirculation from the RC System). The other DH string should be maintained on standby for use in recirculating water from the RB sump to the RC system.

NOTE: Trip Reactor Coolant Pumps before R.C. Pressure decreases below pump NPSH (See figure 1 of 2102-3.1/3.2.

3.2.8 When the Borated Water Storage Tank level decreases to 12' as indicated on panel 8, Shift the MU/HPI pump(s) suction from the BWST to the RB sump if RCS Pressure is greater than 200 psig as follows: (assume DH string A(B) is being used for decay heat removal and DH string B(A) is being maintained on standby):

3.2.8.1 If not already done, "THROTTLE" HPI string(s) flow rate to at least 500 gpm each < 250 gpm per leg) using control valves MU-V16A/B/C/D (or MU-V17 if flow has been returned to the normal makeup flow path). Flow rate indication and valve control in control room on Panel 8 and 3, respectively.

3.2.8.2 "OPEN" valve DH-V7B(A) in crossover line from LPI String B(A) to HPI string B(A) (suction of HPI pumps)..

"REPOSITION" HPI flow control valves MU-V16A/B/C/D (or MU-V17 if flow has been returned to the normal makeup flowpath). HPI flow would increase because of increased HPI pump suction pressure.

3.2.8.3 When the BWST level decreases to 7', verify automatic transfer to the RB sump is initiated. Verify OPEN suction valve for string B(A), DH-V6B(A) from the RB sump.

3.2.8.4 When the suction valve from the RB sump DH-V6 B(A), is fully open, then "CLOSE" the ECCS suction valve, DH-V5B(A), from the BSWT (valve controls and position indication in control room). The ECCS B(A) string is now in "piggy-Back" operation providing makeup to the RCS from the RB sump as required.

3.2.9 After R.C.S. pressure decreases to ≈ 200 psig, throttle HPI discharge flow by throttling MU-V16A/B/C/D. Observe that LPI pumps now deliver water to RCS via DH-V4A/B.

3.2.10 When MU-V16A/B/C/D (HPI flow valves) are closed, stop the Hi pressure injection pumps & close DH-V7A & 7B from the LPI pump discharge. Injection flow path is now as follows:

Spill coolant to RB sump, RB sump to LPI pumps,
LPI pumps to RCS via DH-V4A/B.

3.2.11 Throttle DH-V12BA & B as required to maintain 220" pressurizer level and max. LPI pump flow of 3000-3300 gpm. Within about 24 hours, establish a long-term cooling circulation mode as described in 2104-1.3 and listed below.

Mode 1 Forced circulation using decay Heat drop line.

Mode 2 Gravity draining reactor coolant hot leg to the Reactor Building sump via the D.H. drop line.

Mode 3 Hot leg injection using Pressurizer Auxiliary Spray Line.

Mode 4 Reverse flow through the Decay Heat Drop line into "B" Reactor Coolant Loop Hot Leg.

3.2.12 Evaluate radiation levels & initiate action for Site Emergency as outlined in the TMI radiation emergency plan.

3.2.13 Reactor Building Isolation Initiated

1. Refer to Section B, 3.0 & complete all steps.

B. LEAK OR RUPTURE OF SIGNIFICANT SIZE SUCH THAT ENGINEERED SAFETY FEATURES SYSTEMS ARE AUTOMATICALLY INITIATED.

1.0 SYMPTOMS

1.1 Rapid continuing decrease of reactor coolant pressure.

(1) Lo alarm 2055 psig.

(2) Lo-Lo-alarm 1700 psig.

(3) Safety Injection actuation at 1640 psig.

1.2 Rapid continuing decrease of pressurizer level.

(1) Lo alarm 200".

(2) Lo-Lo alarm 80" (Interlock heater shutoff).

1.3 Hi radiation alarm in Reactor Building.

1.4 Reactor Building Ambient Temperature Alarm.

1.5 Hi Reactor Building Sump level.

1.6 Hi Reactor Building pressure (R.C.S or main steam line rupture).

1.7 Rapidly decreasing make-up tank level.

1.8 Both core flood tanks levels & pressures are decreasing.

NOTE: The operator may distinguish between a loss of coolant inside containment, an OTSG tube rupture and a steam line break by the following symptoms which are unique to the aforementioned accidents.

1. Loss of coolant inside Rx Bldg. - particulate, iodine gas monitor alarm on HP-R-227 "Reactor Building Air Sample."

2. OTSG tube rupture - Gas monitor alarm on VA-R-748.

3. Steam break inside Rx Bldg:

(1) Low condensate storage tank level alarm - and or low hot well level alarm.

(2) FW Latch System Actuation.

2.0 IMMEDIATE ACTION

2.1 Automatic Action.

2.1.1 Reactor trip 1900 psig.

2.1.2 Turbine Trip.

2.1.3 Safety Injection initiated @ 1600 psig R.C.S pressure, or 4 psig Reactor Building pressure.

2.1.4 Both Core Flood Tank levels & pressures may decrease depending upon rupture size and R.C.S. pressure. ($\leq$ 600 psig).

2.1.5 Reactor Building Isolation & Cooling initiated. (R.B. Press. $\geq$ 4 psig).

2.1.6 Reactor Building Spray if the Reactor Building pressure is greater than 30 psig.

2.2 Manual Action.

2.2.1 "CLOSE" MU-V12 and MU-V18.

2.2.2 Small Break LOCA Response

2.2.2.1 Within 2 minutes of the LOCA the CRO dedicated to recognition of a small break LOCA must complete the following:

a. Verify that small break LOCA with single failure symptoms exist.

Symptoms: 1. SFAS initiation and only one MUP started, or

2. SFAS initiation and loss of 2-1E or 2-2E

b. DISPATCH designated LOCA Response Primary A.O. to OPEN MUP Discharge Cross-connect.

c. PROCEED to MU-V16A & B or MU-V16C & D.

d. Within 5 minutes of the LOCA the MUP discharge cross connect valve must be opened off its closed seat and one of the MU-V16 valves on the side of the single failure must be opened 2 turns.

2.2.2.2 CRO at MU-V16A & B or MU-V16C & D must ESTABLISH communications with the Control Room.

2.2.2.3 Once in communication with the control room the CRO at MU-V16A & B or C & D continue to open the valves to establish 125 gpm per leg; while the control room CRO THROTTLES MU-V16C & D or A & B to prevent pump runout.

NOTE: If the LOCA was S.S. with loss of MUP all MU-V16 valves would initially be open. The operator at the controls must in this case throttle all 4 MU-V16 valves to 125 gpm/leg. Also in this case the CRO dedicated to Small Break LOCA would not need to go to the MU-V16 valves. He should go to the MUP discharge cross connect valves and assist the A.O. to speed up the opening of the MUP discharge cross connect.

SMALL BREAK LOCA ACTION TIMES

EVENT	TIME From Occurance
Recognition	$\leq$ 2 minutes
CRO to MU-16's	$\leq$ 4.5 minutes

SMALL BREAK LOCA ACTION TIMES (Cont'd)

EVENT	TIME From Occurance
AO to MUP Discharge X-Connects	≤ 3.5 minutes
Communications Established With CRO at MU-V16's	≤ 5.0 minutes
One of the Single Failure Side MU-V16's Open 2 turns	≤ 5.0 minutes
Discharge X-Connect OFF Closed Seat	≤ 5.0 minutes
Discharge X-Connect Open	≤ 10.0 minutes
MU-V16's Throttled to 125 gpm/leg	≤ 10.0 minutes

2.2.3 Verify Hi pressure injection is operating properly as evidenced by injection flow in all four legs. (MU-V16A/B/C/D). Flow indicated on MU23 FE1,2,3,4.

2.2.4 "TRIP" reactor coolant pumps before reaching 1200 psig.

2.2.5 Verify Reactor Building Cooling and Isolation is operating properly.

3.0 FOLLOW UP ACTION

3.1 Verify that all E.S.F. equipment is in its ESF position, by observing that all equipment status lights indicate as shown in Table 8-1.

3.1.1 Check locked valve status book and verify closed or close the following manual containment isolation valves MU-V330, MU-V364, CF-V114A, CFV114B, CF-V145, CF-V146, DH-V187, and DW-V28.

NOTE: Should any component not operate properly, attempt to actuate it at its remote switch in the Control Room. If it still does not operate, & the component has a local control station attempt to operate the component locally.

3.2 Notify Shift Foreman, who notifies all Station personnel over the cross-tied PA system that a site emergency has occured.

3.3 0 to 20 or 30 minutes past LOCA until sump recircualtion is initiated: Control Room operator continuously monitors the following:

3.3.1 Liquid levels in the:

1. Borated Water Storage Tank, DH-T1, (DH-3-LI 1/2).
2. Sodium Hydroxide Tank, DH-T2, (DH-7-LI).

3.3.2 Safety Features flow rate in each of the following:

1. Two Low Pressure (Decay Heat) Injection lines, DH-1-FI 1 and 2.
2. Four High Pressure (Makeup) Injection lines, MU-23-FI 1, 2, 3, and 4.
3. Two Reactor Building Spray injection lines, BS-1-FI and 2.
4. Four of five reactor building emergency cooling river water lines AH-FI-5620, 5621, 5522, 5623, or 5624 respectively.

3.3.3 Reactor Building environmental indications:

1. Temperature, recorder on Panel 25.
2. Pressure, recorder on Panel 3.

3.4 "DEFEAT" any <u>two</u> channels of Reactor Building Isolation and Cooling, then bypass all three Safety Injection Channels.

<u>CAUTION</u>: If normal power is lost while operating in the injection Mode from the BWST, RB Isolation and Cooling must be manually initiated, when either the BUS 2-1E or 2-2E Undervoltage alarm is received to ensure proper diesel generator load sequencing.

3.5 "THROTTLE" as required to prevent pump runout:

1. Hi press. inj. flow (MU-16A/B/C/D) 0-250 GPM/LEG.

<u>CAUTION</u>: If MU pump flow drops below 95 GPM trip the excess MU pumps.

2. Lo Press. Inj. Flow (DH-V128A/B) 3000-3300 GPM/PMP.

3. Building spray flow (BS-V1A/B) 1400-1700 GPM/PMP.

NOTE: Hi flow alarms should actuate as a warning to throttle flows.

CAUTION: The actions to be taken for switching suction from the BWST to the R.B. sump depend upon the number of operating ECCS injection strings and the delivered flowrates in these injection strings. Based upon the existing situation, in the ECCS, proceed as outlined below to perform switch over of suction to the RB sump:

	Situation	Go to Step
1.	Both LPI strings are operating and indicated flow in each is above 750 gpm.	3.6
2.	Both LPI strings are operating but indicated flow in each is below 750 gpm.	3.7
3.	One LPI string is inoperative.	3.8

NOTE: The main objective when switching suction from the BWST to the RB sump is to maintain ECCS flow through two flow paths.

3.6 Both LPI Strings are Operating and Indicated Flow in Each is Above 750 GPM.

3.6.1 When the BWST level reaches approximately 12', initiate the following steps:

3.6.2 If not already done "THROTTLE" LPI strings flow rates back to 3000 GPM each using control valves DH-V128 A &B (flow rate indication and valve control in control room).

3.6.2.1 If not already done, THROTTLE BS pump's flows back to 1600 gpm per pump. This must be done prior to taking suction from the RB sump.

3.6.3 "SHUT OFF" HPI pumps (pump control in control room).

3.6.4 Verify the ECCS suction valves DH-V6A & B from RB sump automatically open at BWST level of 7'.

3.6.5 When suction valves from RB sump are full open, "CLOSE the ECCS suction valves (DH-V5A & B) from the BWST.

3.6.6 "REPOSITION" LPI flow control valves (DH-V128 A&B) to obtain 3000 GPM each string if necessary. (Flow rate could change due to change in suction sources).

3.6.7 Proceed to step 3.9.

3.7 Both LPI Strings Are Operating But Indicated Flow In Each Is Below 750 GPM.

3.7.1 When the BWST level reaches approximately 12', initiate the following steps.

3.7.2 If not already done, "THROTTLE" HPI strings' flow rates back to 500 GPM per pump each using control valves MU-V16A, B, C, and D (flow rate indication and valve control in control room).

3.7.2.1 If not already done, THROTTLE BS pump's flows back to 1600 GPM per pump. This must be done prior to taking suction from the RB sump.

3.7.3 "OPEN" valves DH-V7A and B in crossover line from LPI line to suction of HPI pumps (valve control and position indication in control room). "Reposition" HPI flow control valves (MU-V16A,B,C, & D) to obtain 500 GPM each string. (HPI flow would increase because of increased HPI pump suction pressure).

3.7.4 VERIFY the ECCS suction valves (DH-V6A & B) from RB sump automatically open at BWST level of 7'.

3.7.5 When suction valves from RB sump are fully open, "Close" the ECCS suction valves (DH-V5A & B) from the BWST. The ECCS is now in "piggy-back" operation.

3.7.6 Proceed to step 3.9.

NOTE: Once the flow in each LPI string exceeds 750 gpm, the HPI pumps can be "SHUT OFF" and valves DH-V7A & B can be "CLOSED".

3.8 One LPI String is Inoperative

3.8.1 The BWST 7' automatic transfer to the RB sump is reached in approximately 55-80 minutes from initiation of ECCS injection, depending upon reason for string failure (i.e. - local LPI failure or diesel failure). Prior to actuation of the lo-lo level alarm, initiate the following steps.

3.8.2 Using the controls in the control room, attempt to "START UP" the non-operating LPI String. If successful, proceed to step 3.6. If not successful, proceed to step 3.8.3 below.

3.8.3 If step 2 was unsuccessful, initiate opening the DH cross-connect isolation valves (DH-V193 A & B) as follows:

3.8.3.1 ENSURE SN-V188 is closed then OPEN DH-V112 A & B to fill the inoperable string.

NOTE: If offsite power is lost DHV193 A & B must be manually opened.

3.8.3.2 Obtain the keys for the DH cross-connect isolation valves' (DH-V193A and DH-V193B) breakers from the shift supervisor.

3.8.3.3 Proceed to 480V MCC 2-32B and MCC 2-42B.

3.8.3.4 Remove the locks from the isolation valve breakers for the DH cross-connect line.

NOTE: Local control stations for DH-V193A and are located in the Aux Bldg at Elev. 280'6" near the DH vaults.

3.8.3.5 "OPEN" the DH cross-connect isolation valve (e.g. DH-V-193A(B)) next to the operating LPI String.

3.8.3.6 "OPEN" the second isolation valve (DH-V193A(B)).

3.8.3.7 While opening the second decay heat cross-connect isolation valve, "THROTTLE" either DH-V128A or DH-V128B in the control room as required to achieve essentially equal flow rates in both D.H. injection lines. (Approximately 1500 gpm per LPI string).

3.8.3.8 If flow is established at greater than 750 gpm through each LPI string, then proceed to step 3.8.5.

3.8.3.9 If flow cannot be established through each LPI string in excess of 750 gpm using the cross-connect line before ECCS suction must be switched to the RB sump, then proceed to step 3.9.4.

3.8.4 If opening the DH cross-connect line fails to provide flow in each LPI string in excess of 750 gpm, then place one HPI string in a modified "Piggy-Back" mode with the operating LPI string as follows (assume LPI string "A"(B) is the operating string):

3.8.4.1 If not already done, "THROTTLE" HPI string "A"(B) flow rate back to 500 GPM using control valve MU-V16A and V16B.

3.8.4.2 "THROTTLE" LPI string "A"(B) flow rate to 3000 GPM using control valve DH-V128A(B).

3.8.4.3 "OPEN" valve DH-V7A(B) in crossover line from LPI line to suction of the operating HPI pump.

3.8.4.4 "REPOSITION" HPI flow control valves MU-V16A&B to obtain 250 GPM per leg HPI flow and reposition LPI flow control valve DH-V128A to obtain 2500 GPM LPI flow. The LPI pump is pumping design flow of 3000 GPM (2500 GPM LPI plus 500 GPM HPI).

3.8.4.5 "SHUT OFF" HPI pump in HPI string "B"(A).

3.8.4.6 Proceed to step 3.8.6.

3.8.5 "SHUT OFF" HPI pumps (pump control in control room).

3.8.6 If not already done, THROTTLE BS pump's flows back to 1600 GPM per pump. This must be done prior to taking suction from the RB sump.

3.8.6.1 When the BWST level decreases to 7' VERIFY the ECCS suction valves (DH-V6A & B) from RB sump automatically open.

3.8.7 When suction valves from RB sump are full open (position indication in control room), "CLOSE" the ECCS suction valves (DH-V5A & B) from the BWST.

3.8.8 "REPOSITION" LPI flow control valve(s) (DH-V128A and/or B) as required to obtain proper string flowrates. (Flow rates could change due to change in suction sources).

3.9 When the Sodium Hydroxide Tank level reaches approximately 3 ft. "CLOSE" DH-V8A & B.

3.10 Actuate Environmental Barrier System by opening EB-V11.

3.11 Within 24 hours of ECCS initiation, establish one of the long-term cooling circulation modes described in 2104-1.3 - Decay Heat Removal System, and listed below:

Mode 1 Forced circulation using decay heat drop line.

Mode 2 Gravity draining reactor coolant hot leg to the Reactor Building sump via the D.H. drop line.

Mode 3 Hot leg injection using pressurizer auxiliary spray line.

Mode 4 Reverse flow through the decay heat drop line into "B" Reactor Coolant Loop Hot leg.

4.0 LONG TERM ACTION

4.1 Verify all previous actions and carry out additional actions as outlined below.

4.2 Evaluate symptoms and determine if possible the cause of the loss of coolant.

4.3 Secure turbine, feed water, and steam systems when time permits.

4.4 Monitor for H_2 buildup and assure actuation of H_2 recombiner per 2104-6.4, Hydrogen Recombiner Operations.

4.5 Monitor R.B. Sump for pH and add Sodium Hydroxide as required thru the decay heat removal system.

4.6 As conditions permit, evaluate unit conditions, and return all non-essential equipment to its normal line up.

NOTE: Refer to the following instructions and procedures for additional information as required.

1. Radiation emergency plan site emergency in the emergency plan.
2. 2104-5.4 - Control Building HVAC.

TABLE A-1

ESF EQUIPMENT - ESF POSITION IN THE CONTROL ROOM PANEL 13

The White light for each component should be lit to indicate that the component is in its ESF position, unless otherwise noted.

ACTUATION A

SAFETY INJECTION GP. 1 Equipment	ESF Position
DC-P-1A	ON
G2-12 (Note 1)	CLOSED
MU-P-1A (Note 2)	ON
NR-P-1A (Note 3)	ON
NS-P-1A (Note 4)	ON
TIE-2E2	OPEN
T3E-4E2	OPEN
T11E-21E2	OPEN
DC-V96A	CLOSED

SAFETY INJECTION GP. 2 Equipment	ESF Position
NR-V9A	CLOSED
DH-V4A	OPEN
DH-V5A	OPEN
DH-V8A	OPEN
DH-V100A	CLOSED
DH-V102A	OPEN
MU-P-1B (Note 2)	ON
NR-V40A	OPEN
NS-P-1C	ON
T12-22E-2	OPEN
MU-V28	CLOSED

ACTUATION B

SAFETY INJECTION GP. 1 Equipment	ESF Position
DC-P-1B	ON
G22-12 (Note 1)	CLOSED
MU-P-1B (Note 2)	ON
NR-P-1C (Note 3)	ON
NS-P-1B (Note 4)	ON
T2E-1E2	OPEN
T4E-3E2	OPEN
T21E-11E2	OPEN
DC-V96B	CLOSED

SAFETY INJECTION GP. 2 Equipment	ESF Position
NR-V9B	CLOSED
DH-V4B	OPEN
DH-V5B	OPEN
DH-V8B	OPEN
DH-V100B	CLOSED
DH-V102B	OPEN
MU-P-1C	ON
NR-V40B	OPEN
NS-P-1C (Note 4)	ON
T22E-12E-2	OPEN
HY-V55	CLOSED

TABLE A-1

ACTUATION A

SAFETY INJECTION GP. 3 Equipment	ESF Position
MU-V36	CLOSED
MU-V16A	OPEN
MU-V16B	OPEN
DH-P-1A	ON
NR-P-1B (Note 3)	ON
T31E-41E-2	OPEN
NS-V84B	CLOSED
NR-V42A	OPEN

ACTUATION B

SAFETY INJECTION GP. 3 Equipment	ESF Position
MU-V37	CLOSED
NS-V32	CLOSED
NS-V67	CLOSED
DH-P-1B	ON
MU-V16C	OPEN
MU-V16D	OPEN
NR-P-1D (Note 3)	ON
T41E-31E-2	OPEN
NS-V84A	CLOSED
NR-V42B	OPEN

PANEL 8

Equipment	ESF Position	Status Light Indicator
NS-V83A	OPEN	R
NS-V83B	OPEN	R
NS-V215	CLOSED	G
NS-V216	CLOSED	G

NOTE 1: Diesel Generator Breaker will only be closed if Normal Power is lost; otherwise status indication will be Open (white).

NOTE 2: MU-P-1B will be running if normal power is available for the Actuation, for the pump that it is selected to backup.
If normal power is lost, MU-P-1B will be running, if the pump that it is selected to backup fails to start or is inoperable.

TABLE A-1

NOTE 3: The NR pump in each neader selected for ES or standby will start if a pump is not operating in that header; otherwise the operating pump will remain in service.

NOTE 4: Normally NS-P-1A and 1B will start; however, NS-P-1C will start if either NS-P-1A or B (depending upon which pump it is selected to backup) fails to start or is inoperable.

TABLE B-1

ESF EQUIPMENT - ESF POSITION IN THE CONTROL ROOM
PANEL 13

The White light for each component should be lit to indicate that the component is in its ESF position, unless otherwise noted.

ACTUATION A

SAFETY INJECTION GP. 1	
Equipment	ESF Position
DC-P-1A	ON
G2-12 (Note 1)	CLOSED
MU-P-1A (Note 2)	ON
NR-P-1A (Note 3)	ON
NS-P-1A (Note 4)	ON
T1E-2E2	OPEN
T3E-4E2	OPEN
T11E-21E2	OPEN
DC-V96A	CLOSED

SAFETY INJECTION GP. 2	
Equipment	ESF Position
NR-V9A	CLOSED
DH-V4A	OPEN
DH-V5A	OPEN
DH-V8A	OPEN
DH-V100A	CLOSED
DH-V102A	OPEN
MU-P-1B (Note 2)	ON
NR-V40A	OPEN
NS-P-1C	ON
T12-22E-2	OPEN
MU-V28	CLOSED

ACTUATION B

SAFETY INJECTION GP. 1	
Equipment	ESF Position
DC-P-1B	ON
G22-12 (Note 1)	CLOSED
MU-P-1B (Note 2)	ON
NR-P-1C (Note 3)	ON
NS-P-1B (Note 4)	ON
T2E-1E2	OPEN
T4E-3E2	OPEN
T21E-11E2	OPEN
DC-V96B	CLOSED

SAFETY INJECTION GP. 2	
Equipment	ESF Position
NR-V9B	CLOSED
DH-V4B	OPEN
DH-V5B	OPEN
DH-V8B	OPEN
DH-V100B	CLOSED
DH-V102B	OPEN
MU-P-1C	ON
NR-V40B	OPEN
NS-P-1C (Note 4)	ON
T22E-12E-2	OPEN
HY-V55	CLOSED

TABLE 8-1

ACTUATION A		ACTUATION B	
SAFETY INJECTION GP. 3		SAFETY INJECTION GP. 3	
Equipment	ESF Position	Equipment	ESF Position
MU-V36	CLOSED	MU-V37	CLOSED
MU-V16A	OPEN	NS-V32	CLOSED
MU-V16B	OPEN	NS-V67	CLOSED
DH-P-1A	ON	DH-P-1B	ON
NR-P-1B (Note 3)	ON	MU-V16C	OPEN
T31E-41E-2	OPEN	MU-V16D	OPEN
NS-V84B	CLOSED	NR-P-1C (Note 3)	ON
NR-V42A	OPEN	T41E-31E-2	OPEN
		NS-V84A	CLOSED
		NR-V42B	OPEN
R.B. ISOLATION AND COOLING GP. 1		R.B. ISOLATION AND COOLING GP 1	
Equipment	Position	Equipment	Position
AH-V81	CLOSED	AH-E-11D	ON
AH-V101	CLOSED	AH-V80	CLOSED
AH-V102	CLOSED	AH-V103	CLOSED
AH-V105	CLOSED	AH-V104	CLOSED
AH-V107	CLOSED	AH-V106	CLOSED
CF-V144	CLOSED	AH-V108	CLOSED
DH-V3	CLOSED	CF-V115	CLOSED
NM-V52	CLOSED	DH-V2	CLOSED
NR-V51A	CLOSED	NM-V104	CLOSED
RR-V2A	OPEN	NR-V51B	CLOSED
RR-V2B			
RR-V5A	OPEN	RR-V2C	OPEN
RR-V5B	OPEN	RR-V2D	OPEN

TABLE B-1

ACTUATION A

R.B. ISOLATION & COOLING GP. 1

Equipment	Position
RR-V5C	OPEN
SV-V55	CLOSED
WDL-V1095	CLOSED
DC-V114	CLOSED

R.B. ISOLATION & COOLING GP. 2

Equipment	Position
AH-E-4A	ON
AH-E-11A	ON
RR-V25C	CLOSED
BS-V1A	OPEN
CA-V10	CLOSED
CA-V4A	CLOSED
CA-V9	CLOSED
RR-P-1B	ON
WDG-V199	CLOSED
WDL-V22	CLOSED
WDL-V1126	CLOSED
AH-D4092A & B	RECIRC
AH-D4092D & E	RECIRC
ED-4098	

ACTUATION B

R.B. ISOLATION & COOLING GP. 1

Equipment	Position
RR-V6C	OPEN
RR-V6D	OPEN
RR-V6E	OPEN
SV-V54	CLOSED
WDL-V1092	CLOSED
DC-V103	CLOSED
DC-V115	CLOSED

R.B. ISOLATION & COOLING GP. 2

Equipment	Position
AH-E-4B	ON
AH-E-11C	ON
RR-V25C	CLOSED
BS-V1B	OPEN
CA-V1	CLOSED
CA-V3	CLOSED
CA-V4B	CLOSED
CA-V8	CLOSED
CA-V6	CLOSED
RR-P-1D	ON
WDG-V2	CLOSED
WDL-V1125	CLOSED
WDL-V271	CLOSED
AH-D4092A & B	RECIRC
AH-D4092D & E	RECIRC
ED-4098	RECIRC

TABLE B-1

ACTUATION A		ACTUATION B	
R.B. ISOLATION & COOLING GP. 3		R.B. ISOLATION & COOLING GP. 3	
Equipment	Position	Equipment	Position
IC-V2	CLOSED	IC-V3	CLOSED
IC-V5	CLOSED	IC-V4	CLOSED
MU-V2A	CLOSED	MU-V376	CLOSED
MU-V2B	CLOSED	MU-V18	CLOSED
MU-V377	CLOSED	MU-V25	CLOSED
NS-V72	CLOSED	NS-V100	CLOSED
NS-V81	CLOSED	NR-P-2B	ON
NR-P-2A	ON	RR-P-1C	ON
NR-V144A	OPEN	RR-V25D	CLOSED
RR-P-1A	ON	RR-V25E	CLOSED
RR-V25A	CLOSED	AH-C-8A	ON
RR-V25B	CLOSED	AH-E-11E	ON
IC-P-1A	OFF	AH-P-1B	ON
AH-C-8A	ON	AH-V2A	CLOSED
AH-E-11B	ON	AH-V2B	CLOSED
AH-E-11C	ON	AH-V3A	CLOSED
AH-P-1A	ON	AH-V3B	CLOSED
AH-V1A	CLOSED	AH-V6	CLOSED
AH-V1B	CLOSED	AH-V61	CLOSED
AH-V4A	CLOSED	AH-V61	CLOSED
AH-V4B	CLOSED	AH-V63	CLOSED
AH-V5	CLOSED	AH-V71	CLOSED
AH-V60	CLOSED	IC-P-1B	OFF
AH-V102	CLOSED		
AH-V72	CLOSED		
* BS-P-1A	ON	* BS-P-1B	ON

* If RB Pressure >30 psig.

NOTE 1: Diesel Generator Breaker will only be closed if Normal Power is lost; otherwise status indication will be Open (white).

NOTE 2: MU-P-1B will be running if normal power is available for the Actuation, for the pump that it is selected to backup.
If normal power is lost, MU-P-1B will be running, if the pump that it is selected to backup fails to start or is inoperable.

NOTE 3: The NR pump in each neader selected for ES or standby will start if a pump is not operating in that header; otherwise the operating pump will remain in service.

NOTE 4: Normally NS-P-1A and 1B will start; however, NS-P-1C will start if either NS-P-1A or B (depending upon which pump it is selected to backup) fails to start or is inoperable.

PANEL 8

Equipment	ES Position	Indication
DH-V7A(1)	Close	G
DH-V7B(1)	Close	G
NS-V83A	Open	R
NS-V83B	Open	R
NS-V215	Close	G
NS-V216	Close	G
CF-V1A	Open	R
CF-V1B	Open	R

(1) This valve may have to be opened for "piggy-back" operation. Once, opened, the Position/Indication becomes Open/R.

PANEL 15

Equipment	ES Position	Indication
DH-V6A(1)	Close	G
DH-V6B(1)	Close	G
MU-V37B(2)	Open	R
MS-V4A(3)	Open	R
MS-V4B(3)	Open	R
MS-7A(3)	Open	R
MS-V7B(3)	Open	R

(1) This valve must be opened for sump-switchover. Once opened, the Position/Indication becomes OPEN/R.

(2) This valve should be closed at the operator's first chance. Once closed, the Position/Indication becomes CLOSE/W.

(3) These valves should be closed when the steam system is secured. Once closed, the Position/Indication becomes CLOSE/G.

PANEL 25

Equipment	ES Position	Indication
AH-E12A	Off	G
AH-E12B	Off	G
AH-E19A	Off	G
AH-E19B	Off	G

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2202-1.4
Revision 8
02/08/79

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-1.4
LOSS OF RC FLOW/RC PUMP TRIP

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24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

R P Warren Date 2/7/79

Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

J B Logan Date 2/8/79

Manager Generation Quality Assurance Approval NA Date —

TMI 55-A Rev 8/77

THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-1.4

LOSS OF RC FLOW/RC PUMP TRIP

1.0 SYMPTOMS

1.1 Annunciator Alarm for Auto Trip of R.C.P.

1.1.1 Instantaneous Phase Overcurrent

1.1.2 Time Delay Phase Overcurrent

1.1.3 Thermal Overload

1.1.4 Running Undervoltage

1.1.5 Phase Balance

1.1.6 Differential Overcurrent.

1.1.7 Loss of Seal Injection and I.C. Cooling Water.

1.2 R.C. Pump Light above Control Switch goes from red to green.

1.3 Conditions requiring manual trip of R.C. Pump.

1.3.1 Motor Guide bearing temperature exceed 185°F.

RCP	Computer Point	
	Upper Bearing	Lower Bearing
1A	0434	0438
2A	0435	0439
1B	0436	0440
2B	0437	0441

1.3.2 Motor thrust bearing temperature exceed 200°F.

RCP	Computer Point	
	Downthrust	Upthrust

1A	0426	0430
2A	0427	0431
1B	0428	0432
2B	0429	0433

1.3.3 Motor stator temp exceeds 302°F (150°C).

RCP	Computer Point
1A	0442
2A	0443
1B	0444
2B	0445

1.3.4 RCP seal staging water temp exceeds 185°F.

RCP	Computer Point
1A	0418
2A	0419
1B	0420
2B	0421

1.3.5 Air cooler leak detector alarms.

1.3.6 Shaft vibration exceeds:

a. 26 mils for 1 or 2 pump per loop operation or

b. 30 mils for first 4 hours of 1 pump per loop operation.

1.3.7 Either seal cavity pressure exceeds 2500 psig.

1.3.8 Loss of total seal injection and Intermediated Closed Cooling Flow.

1.3.9 Seal staging flow plus seal leakage flow exceeds 1.91 g.p.m.

1.3.10 Loss of cooling water to the motors.

1.3.11 Motor stand vibration > 3 mills.

2.0 IMMEDIATE ACTION

2.1 Automatic Action.

2.1.1 Loss of one (1) RC Pump or one (1) pump per loop will result in a runback to 75% or 45% neutron power respectively. If the ICS cannot respond quickly enough to avoid exceeding the RPS setpoints, a Reactor and Turbine trip will result.

2.1.2 Loss of two (2) RC Pumps in one loop or loss of more than two (2) pumps will result in a Reactor and Turbine trip initiated by the RPS.

2.1.3 Lift pumps and back-stop pumps Auto-start on loss of RC pump.

2.1.4 Emergency Feedwater Pumps Auto-Start and EFW flow occurs upon loss of four (4) RC Pumps. Steam Generator level will increase to 50% on operating range.

2.1.5 If the Reactor does not trip and both feedwater Control Stations are in Auto and not on low level limits, the feed flow will be ratioed according to the number of pumps still operating.

2.2 Manual Action.

2.2.1 Verify that the unit has runback to the Power Level allowable for the RC Pump Combination.

NOTE: If any ICS station in Hand, run station back in manual.

2.2.2 If a condition arises which necessitates manual tripping of a pump, reduce to appropriate power levels if possible prior to tripping pump, start the oil pumps and trip R.C. pump.

2.2.3 Verify Feedwater flows are rationed approximately 2.41 to 1 for loss of one RC pump.

2.2.4 Verify Lift Pumps and Back-Stop Pumps have Auto-Started, if not, start the oil pumps.

2.2.5 If any of the following ICS stations are in Hand (Steam Generator/ Reactor Demand, either Feedwater Demand, Main or Startup Feedwater Valve Demand, Feedpump Speed, Reactor Master, and/or Diamond) runback the appropriate ICS stations corresponding to plant conditions.

3.0 FOLLOW-UP ACTION

3.1 After pump reaches zero speed, shutdown lift pumps and Back-Stop pumps.

NOTE: Coastdown may take as long as 30 minutes on last pump shutdown. Time is shorter as number of pumps still in operation increases.

3.2 If all for (4) RC Pumps are tripped, proceed as outlined in 2202-2.1 "Blackout."

3.3 Adjust the nuclear overpower trip setpoint for the allowable pump combination per 2311-6.

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2202-1.5
Revision 3
09/29/78
G/Hm> 119

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-1.5
PRESSURIZER SYSTEM FAILURE

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Chairman of PORC

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-1.5

PRESSURIZER SYSTEM FAILURE

SECTION A Leaking Pilot Operated (electromatic) Relief Valve (RC-R2)

A.1 SYMPTOMS

1. Relief valve discharge line temperature exceeding the normal 130°F. Alarms on computer at 200°F.
2. RC drain tank pressure above normal on the control room radwaste disposal control panel and temperature above normal on the local radwaste disposal control panel.
3. RC System makeup flow above normal for the variable letdown flow and RC pump seal in-leakage conditions.
4. Boric Acid concentration continually increasing in the pressurizer.

A.2 IMMEDIATE ACTIONS

A. Automatic Actions

1. None.

B. Manual Actions

1. Close the Electromatic Relief Isolation Valve, RC-V2.

A.3 FOLLOW-UP ACTION

1. Repair during next shutdown.
2. Limit rate of change on ICS to less than 1% per minute while RCV is closed except for runbacks.

SECTION B Inoperative Pilot operated (electromatic) Relief Valve (RC-R2)

B.1 SYMPTOMS

1. RC System pressure is above 2255 psig and RC-R2 fails to open.
2. RC System pressure is below 2205 psig and RC-R2 fails to close.
3. RC-R2 discharge line temperature is above the 200°F alarm. Computer Point (402)
4. The RC drain tank pressure and temperature are above normal on the control room radwaste disposal control panel 8A.

B.2 IMMEDIATE ACTION

A. Automatic Action

1. For a failed closed RC-R2:
   a. Pressurizer heaters off at 2160 psig. Spray valve RC-V1 is open above 2205 psig.
   b. Reactor trip occurs at 2355 psig.
   c. Pressurizer code relief valves open at 2450 psig.
2. For a failed open RC-R2:
   a. All pressurizer heater banks on full below 2105 psig.
   b. Reactor trips at 1900 psig or variable pressure temperature.
   c. High Pressure Injection is actuated at 1600 psig.

B. Manual Action

1. For a failed close RC-R2:
   a. Shift spray valve RC-V1 to "MANUAL" and open further for additional spray flow.
   b. Insure all pressurizer heaters off above 2160 psig.

c. If reactor power is being changed (except for a runback) stop the power change until pressure is returned to normal.

d. Isolate RC-R2 by closing RC-V2.

2. For a failed open RC-R2:

a. Close Electromatic Relief Isolation Valve (RC-V2).

b. Insure all pressurizer heaters on below 2105 psig.

B.3 FOLLOW-UP ACTION

1. Return system pressure and temperature to normal.

2. Reduce ICS Rate of Change to less than 1% per minute (except for Runbacks.)

SECTION C Leaking Code Relief Valve (RC-R1A or RC-R1B)

C.1 SYMPTOMS

1. Code relief valve discharge line temperature(s) exceeding the computer normal 130°F. Computer alarms at 200°F. Computer Point (403) (404)
2. RC drain tank pressure and temperature above normal on the control room radwaste disposal control panel 8A.
3. RC System makeup flow is above normal for the variable letdown flow and RC pump seal in-leakage conditions.
4. Boric Acid Concentration continually increasing in the Pressurizer.

C.2 IMMEDIATE ACTION

A. Automatic Action

1. None.

B. Manual Action

1. Determine RC leakage according to 2301-303.

C.3 FOLLOW-UP ACTION

1. If RC system identified leakage is in excess of 10 gpm, reduce the leakage rate to within limits within 4 hours or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.
2. It will be necessary to recirculate the pressurizer through the spray valve to equalize Boron concentration.
3. Place Code Relief Discharge Line temperatures on Analog Trend Recorder.

SECTION D Inoperative Code Relief Valve (RC-R1A or RC-R1B)

D.1 SYMPTOMS

1. Code relief valve(s) fail to open when RC pressure is above 2450 psig.
2. Code relief valve(s) fail to close when RC pressure is below 2325 psig.
3. Code relief valve(s) discharge line temperature is above 200°F alarm.
4. The RC Drain Tank pressure and temperature are above normal on the control room radwaste disposal control panel.
5. RC system makeup flow is above normal for the variable letdown flow and the RC pump seal in-leakage conditions.

D.2 IMMEDIATE ACTION

A. Automatic Action

1. For a fail to open code relief valve:
   a. Reactor trip occurred at 2355 psig.
   b. Spray valve RC-V1 opened above 2205 psig.
2. For a fail to close code relief valve:
   a. Reactor trip occurs at 1900 psig or on variable P/T.
   b. Increased makeup flow.
   c. All pressurizer heaters energized.
   d. Safety Injection is actuated at 1600 psig.

B. Manual Action

1. For a fail to open code relief valve:
   a. Place pressurizer spray valve in "MANUAL" and open further for additional spray flow.

b. Verify pressurizer heaters are "OFF" at plant control panel.

2. For a fail to close code relief valve:

a. Turn all heaters "ON" at plant control panel.

b. Isolate letdown flow at plant control panel by "CLOSING" MU-V376.

c. Open DH-V5A. Start MU-P1A if necessary. Attempt to control pressurizer level using MU-V16B.

d. Manually initiate safety injection if required to maintain pressurizer level.

D.3 FOLLOW-UP ACTION

1. For a fail to open code relief valve:

a. Proceed with cooldown.

2. For a fail to close code relief valve:

a. Hold pressurizer, if possible, at or greater than 220 inches with Safety Injection.

b. Proceed with cooldown.

c. With no pressurizer code safety valve operable, immediately suspend all operations involving positive reactivity changes and place an operable DHR Loop into operation in the shutdown cooling mode.

d. With a pressurizer code safety valve inoperable, either:

1. Restore the inoperable valve to operable status within 15 minutes or

2. Be in Hot Shutdown within 12 hours.

SECTION E Inoperative Pressurizer Heaters

E.1 SYMPTOMS

1. Heater banks fail to energize or de-energize if RC pressure is at heater bank setpoint.

	Bank 1	Bank 2	Bank 3	Bank 4	Bank 5	Units
ON	2150	2145	2135	2120	2105	PSIG
OFF	2160	2155	2155	2140	2125	PSIG

NOTE: Banks 1, 2, and 3 are full on at "ON" setpoint.

2. Pressurizer level Lo-Lo alarm at 80 inches.
3. Pressurizer heater power supply ground alarm.
4. Abnormal console indicating lights for the heating groups.
5. High (2255 psig) or low (2055 psig) pressure alarms.

E.2 IMMEDIATE ACTION

A. Automatic Action

1. For energized heaters and rising pressure:
   a. Pressurizer spray valve (RC-V1) open (red and green console jog button lights).
2. For loss of heaters and decreasing pressure: None.

B. Manual Action

1. If control malfunction is suspected:
   a. Place heater controller in "MANUAL".
2. For energized heaters and rising pressure:
   a. Attempt to de-energize all heaters except Banks 1 or 2. (Groups 12 or 13 respectively).
3. For loss of heaters and decreasing pressures:
   a. Attempt to energize backup heaters from plant control panel.
   b. If a. is unsuccessful, begin reducing unit load.

E.3 FOLLOW-UP ACTION

1. For energized heaters and rising pressure:

   a. Open heater breakers in question at the pressurizer heater control centers except for Banks 1 or 2 (Groups 12 or 13 respectively).

   b. Control RC pressure at the normal 2155 psig set point with the pressurizer spray valve (RC-V1) in "MANUAL".

2. For loss of heaters and decreasing pressure:

   a. Determine cause.

   b. If pressure cannot be maintained with the remaining heaters, continue load reduction to shutdown and possibly cooldown condition.

   c. Close RC-V3 and reopen periodically to maintain spray line temperature greater than 540°F.

SECTION F Malfunction In Pressurizer Level Indication or Control

F.1 SYMPTOMS

1. Disagreement between the console recorder level readouts of more than 12 inches.
2. Rapid change in indicated/recorded level due to loss of compensation or loss of power or d/p cell failure or other malfunction.

F.2 IMMEDIATE ACTION

A. Automatic Action

1. If indication fails low:

   Pressurizer heaters trip @ 80 inches, makeup valve MU-V17 opens, and RC pressure increases.

2. If indication fails high:

   Makeup valve MU-V17 closes.

B. Manual Action

1. When any two of three console recorder level transmitter readouts disagree by more than 12 inches, take manual control of level and then select the third transmitter for indication.
2. Re-energize heaters if tripped due to malfunction.

F.3 FOLLOW-UP ACTION

1. If the switching level transmitters has not rectified the condition, switch to the alternate temperature detector.
2. If pressurizer level recorder indication is lost, select another transmitter or use the computer for level indication.

SECTION G Pressurizer Spray Valve Failure (RC-V1)

G.1 SYMPTOMS

1. Pressurizer spray valve (RC-V1) fails to open when the RC system pressure is greater than 2205 psig.
2. Pressurizer spray valve (RC-V1) is open when the RC System is less than 2155 psig.

G.2 IMMEDIATE ACTION

A. Automatic Action

1. RC system pressure greater than 2255 psig activates RC-R2 electromatic relief and the high pressure alarm.
2. RC-V1 failing open (in auto) causes RC system pressure to stablize at approximately 2100 psig with all heater "on".
3. Failure when manually opened beyond the automatic limit position causes continued pressure drop and alarm at 2055.

B. Manual Action

1. Control RC-V1 opening or closing in "MANUAL" with jog buttons.
2. If the spray valve has failed open, control pressure by closing the pressurizer spray isolation valve (RC-V3).

NOTE: If the pressurizer spray isolation valve (RC-V3) is closed, it must be periodically cycled to keep the spray line warm. Cycle RC-V3 is open as necessary to stay above RC pressurizer spray line temperature alarm of 540°F. (Computer point 0405).

CAUTION: Do not exeed a ΔT of 410°F between pressurizer temperature and reactor coolant hot leg temperature.

3. Reduce rate of ICS load change to less than 1% per minute.

G.3 FOLLOW-UP ACTION

1. Continue plant operation with reduced rate of load change.
2. Check thermal overload on RC-V1 and reset if necessary.

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-1.6

HIGH ACTIVITY IN REACTOR COOLANT

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Cognizant Dept. Head

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Unit 1 PORC Recommends Approval

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Chairman of PORC

Unit 2 PORC Recommends Approval

C. F. Terrell Date 11-14-77

V/ Chairman of PORC

Unit 1 Superintendent Approval

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Unit 2 Superintendent Approval

J. J. Seelinger Date 11/15/77

Manager Generation Quality Assurance Approval N/A Date —

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE #2202-1.6

HIGH ACTIVITY IN REACTOR COOLANT

1.0 SYMPTOMS

1.1 Alarm on R.C. Letdown Radiation Monitor MU-R-720.

1.2 High Gross Beta Gamma activity during steady state power operation as determined from 15 minute degassed Gross Beta Gamma sample analysis (higher than normal by a factor of two).

1.3 An increase in I^{131} activity by a factor of two during steady state power operation, as determined by gamma spectroscopy.

2.0 IMMEDIATE ACTION

2.1 Automatic Action

None

2.2 Manual Action

2.2.1 Notify the Shift Supervisor.

2.2.2 If due to MU R 720 Alarm, or high activity in routine analysis, sample and run gamma (DEI) spectrum analysis, and a 15 minute gross β-γ degassed activity analysis. Calculate the Dose Equivalent Iodine concentration.

NOTE: If major plant evolutions such as heatup, cooldown or abnormal pressure temperature transients are in progress, high activity levels can be experienced due to the release of irradiated corrosion products.

2.2.3 If no unit evolutions are in progress that would cause a crud burst and Hi activity is confirmed by a 15 minute gross activity >20 µCi/ml, reduce reactor power to less than 50%.

NOTE: If during a power transient, a gamma spectrum analysis indicates an increase in fission products (eg I^{131}) no immediate action need be taken. (I^{131} will change by a factor of 10 during normal operations).

3.0 FOLLOW-UP ACTION

3.1 If 15 minute degassed Gross Beta Gamma activity result is greater than 10uCi/ml immediately start $\bar{E}$ activity determination.

NOTE: An $\bar{E}$ activity determination consists of a quantitative measurement of 95% of radio nuclides in reactor coolant with half lives greater than 30 minutes.

3.2 If the Dose Equivalent I^{131} is >1.0 µCi/gram but within allowable limit of the Technical Specifications, operation may continue for up to 48 hours (T.S. 3.4.8).

3.3 If the Dose Equivalent I^{131} is >1.0 µCi/gram for more than 48 hours during one continuous time interval, or it exceeds the limit outlined in the Technical Specifications, shutdown the reactor and decrease Tave to <530°F within 6 hours. (T.S. 3.4.8).

3.4 If results of anaylsis indicate total activity due to nuclides with half lives longer than 30 minutes exceed or will exceed $100/\bar{E}$ µCi/ml, be in HOT STANDBY with Tave <530°F within 6 hours.

3.5 If the specific activity exceeds 1.0 µCi/gram Dose Equivalent I^{131} or $100/\bar{E}$ µCi/gram, analyze for Iodine, including I^{131}, I^{133}, and I^{135} once per 4 hours until the activity is restored to within its limits. (T.S. 3.4.8).

3.6 Increase sampling frequency as follows:

a. If the monthly radio chemical analysis results are greater than 10% of the allowable maximum activity for reactor operations ($100/\bar{E}$ µci/ml), the monthly radiochemical sample frequency must be increased to 5 times/week.

b. An $\bar{E}$ determination which is normally performed semi annually will be started when the 15 minute gross degassed activity analysis indicates greater than 10 µci/ml and will be redetermined for each 10 µci/ml increase in the 15 minute gross degassed activity analysis.

3.7 Attempt to determine whether fuel element failure exists, by using the I^{131} - I^{133} and Cs^{137} - Cs^{138} ratios and other radionuclide analysis, such as Gross Alpha activity.

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-1.7
EXCESSIVE RADIATION LEVELS

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NA Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

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V/ Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 PLANT EMERGENCY PROCEDURE

EP 2202-1.7

EXCESSIVE RADIATION LEVELS

22.1 SYMPTOMS

Symptoms of excessive or potentially excessive radiation levels are as follows: Note that these conditions may occur under both planned and unplanned radioactive releases. Planned releases as covered in HPP 1621 and HPP 1622 are exempt from this procedure.

a. Radiation Monitoring System trouble alarm on Control Room annunciator on panel 12.

b. Hi alarm (red indicator light) from any radiation monitor located on Control Room panel 12.

c. Radiation monitor ratemeter indicates in excess of Hi alarm setpoint.

d. Alert (yellow indicator light) alarm from any radiation monitor, located in Control Room panel 12.

e. Radiation Monitor ratemeter indicates radiation levels in excess of Alert Alarm Setpoint.

22.2 IMMEDIATE ACTION

22.2.1 Automatic Action

a. Receipt of an Alert Alarm

If applicable, direct personnel involved in maintenance, operation or sampling activities to immediately isolate the source of the radioactive release per the step specified in the applicable special operating procedure.

b. Receipt of a Hi Alarm:

Isolate flow paths in affected piping systems.

NOTE: Refer to Alarm Response for specific monitor alarming for details of automatic actions, general actions only are given below. Automatic interlocks associated with atmosphere and liquid high alarms are outlined in Enclosure I. A High Alarm on Gamma monitor HP-R-214 will actuate the station radiation alarm as described in the Emergency Plans and Procedures AP 1004.

22.2.2 Manual Action

a. If directed by Shift Supervisor/Shift Foreman, announce the following using the Intra Plant Communications System page channel: "Radiation Hi Alarm, (Location and alarming monitor).

b. If applicable, direct personnel involved in maintenance, operation or sampling activities to immediately isolate the source of the radioactive release per the step specified in the applicable special operating procedure.

c. Ensure that the appropriate control room charts (RMS and Meteorological) are marked with the date, time and occurrence.

22.2.3 Follow-up Action

a. Verify that automatic action (if any) as specified by Alarm Response has occurred.

b. Initiate the following action depending upon the type of monitor which is alarming and the type of alarm.

NOTE: Refer to specific Alarm Response for detailed manual action required. Below steps are generally applicable.

(1) Atmospheric Monitor:

(a) Alert Alarm:

- If applicable, take air sample in accordance with Appendix A.
- If applicable, analyze sample in accordance with Appendix B.
- Sample and analyze as specified in H.P.P. 1676.
- Evacuate affected area(s) if analyses indicate activity in excess of limits for which an RWP is required (3×10^{-10} µCi/cc).
- Determine source of radiation.
- Take appropriate actions to contain radioactivity and reduce levels.

(b) Hi-Alarm:

- Clear affected area of all unnecessary personnel.
- Re-enter affected area using air breathing apparatus to obtain air samples.
- Analyze air samples in accordance with Appendix B.
- If analyses equal greater than 3×10^{-10} µCi/cc, re-entry to area without breathing apparatus will be permitted only under RWP.

- Determine source of radiation.
- Take appropriate action to contain radioactivity and reduce levels.
- If HP-R-219 goes into the high alarm condition, perform the following:

1. Notify the Radiation Protection Department.
2. Give wind direction and specify location of site Fence Survey.
3. Insure that the Radiation Protection Department collects a gas, particulate and iodine sample and conducts a β-γ survey at the specified site fence location.
4. Clear affected area of all non-essential personnel.
5. Re-enter affected area using air breating apparatus to obtain air samples.
6. Obtain and analyze air samples in accordance with HPP 1676.
7. If the analysis indicates levels greater than 3×10^{-10} μCi/cc, re-entry to area without breathing apparatus will be permitted only under RWP.
8. Determine source of radiation.
9. Take appropriate action to contain radioactivity and reduce levels.

(2) Gamma Monitor

(a) Alert Alarm

- Survey affected area(s) using Eberline Rad-Owl, portable ion chamber, or equivalent, to verify alarm.
- Locate radiation source.
- Take appropriate action to reduce radiation levels.

(b) Hi Alarm:

- Survey affected area(s) using Eberline Rad-Owl, portable ion chamber, or equivalent, o verify alarm.
- If survey indicates radiation levels greater than 100 mr/hr, permit entry to area to locate the radiation source and take action to reduce the radiation levels only under RWP.

(3) Liquid Monitor:

(a) Alert Alarm:

- Obtain sample from affected system and count sample in accordance with Appendix C.
- Take action as necessary to contain radioactive liquid.
- Take appropriate action to reduce activity level.

(b) Hi Alarm:

- Obtain sample from affected system and count sample in accordance with Appendix C. If analysis indicates activity greater than alarm setpoint (5.6×10^{-5} µCi/cc) for closed cooling water systems, isolate affected portion of system if possible.
- Take action as necessary to contain radioactive liquid.
- Take appropriate action to reduce activity level.

(c) If source of radiation is a spill, refer to Emergency Plans and Procedures, AP 1004, Local Emergency Procedure.

(4) Complete the Operations Planned/Unplanned Release Report (Enclosure II)

- Ensure that Personnel Exposures are recorded.
- Ensure that Personnel Film Badges/TLD's are processed if required.
- Ensure that the release point cubicle or room air velocity flow is taken. As soon as reasonably possible, notify Instrumentation Department to obtain these readings.
- Decontaminate as necessary.
- Ensure that the calculations for H.P.P. 1676 Attachment A are completed. These calculations include 10 CFR 20.403 evaluation and airborne

concentrations that personnel may have been exposed to. In addition, these calculations:

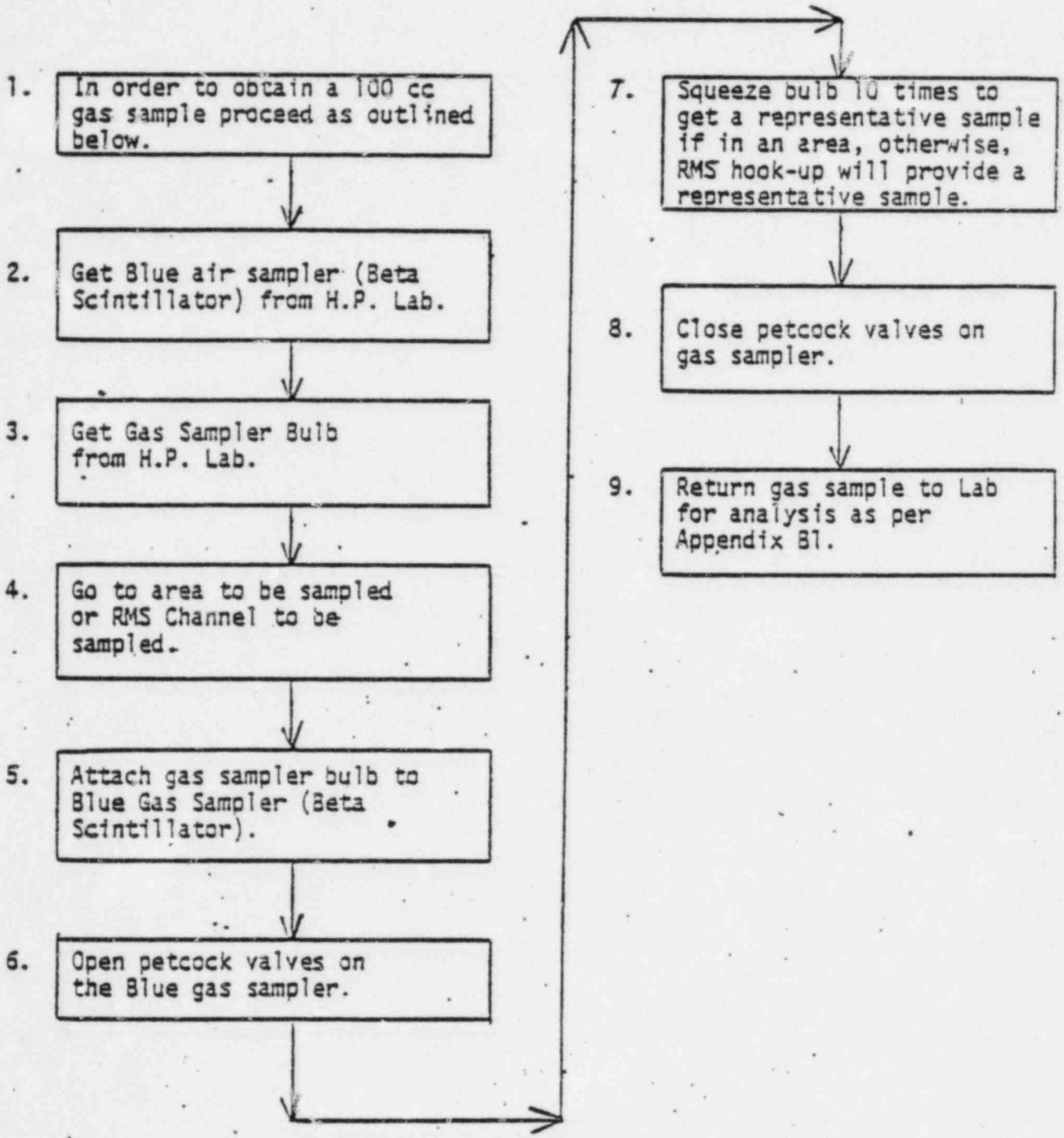
a. Compute Peak & Avg. I^{131} and Particulate release rate (if applicable).

b. Compute Peak & Avg. Noble Gas release rate.

c. Credit the release to the Semi-Annual activity inventory released to the environment.

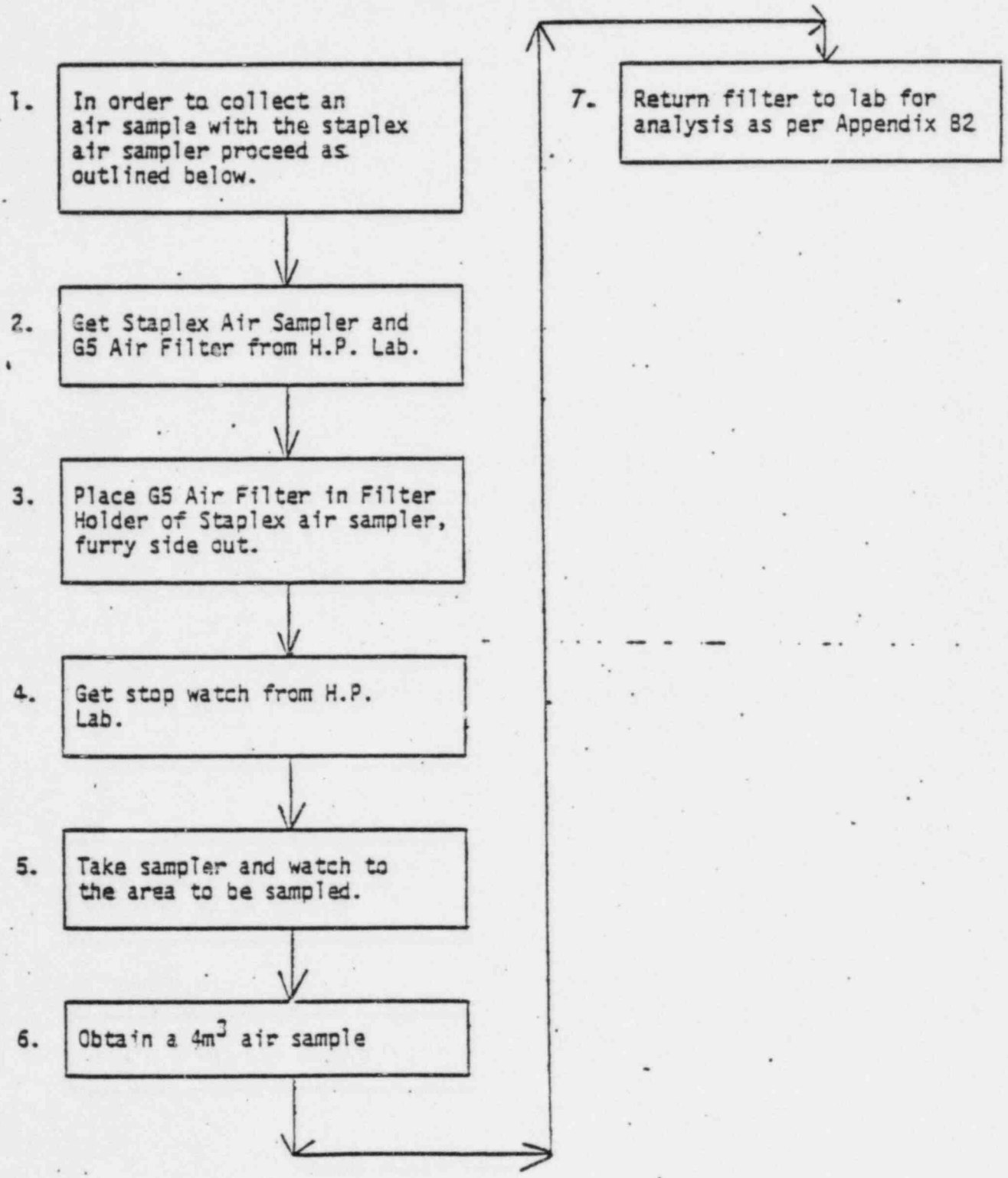
APPENDIX A1

GASEOUS COLLECTION



APPENDIX A2

PARTICULATE COLLECTION



APPENDIX B1

GASEOUS COUNT & CALCULATION

1. Connect BNC connector from output of Blue gas sampler to input of Ludlum Timer Scaler as per attached sketch B-2-1.

2. Turn on Pre-Amp Switch located on the Blue Gas Sampler.

3. Turn count time switch to 5 minutes.

4. Press count button to start counting.

5. After 5 minutes, divide accumulated counts by 5 and subtract background to obtain corrected counts per min. (cc/m).
   NOTE: Background is found on the OPS Scaler Data Sheet.

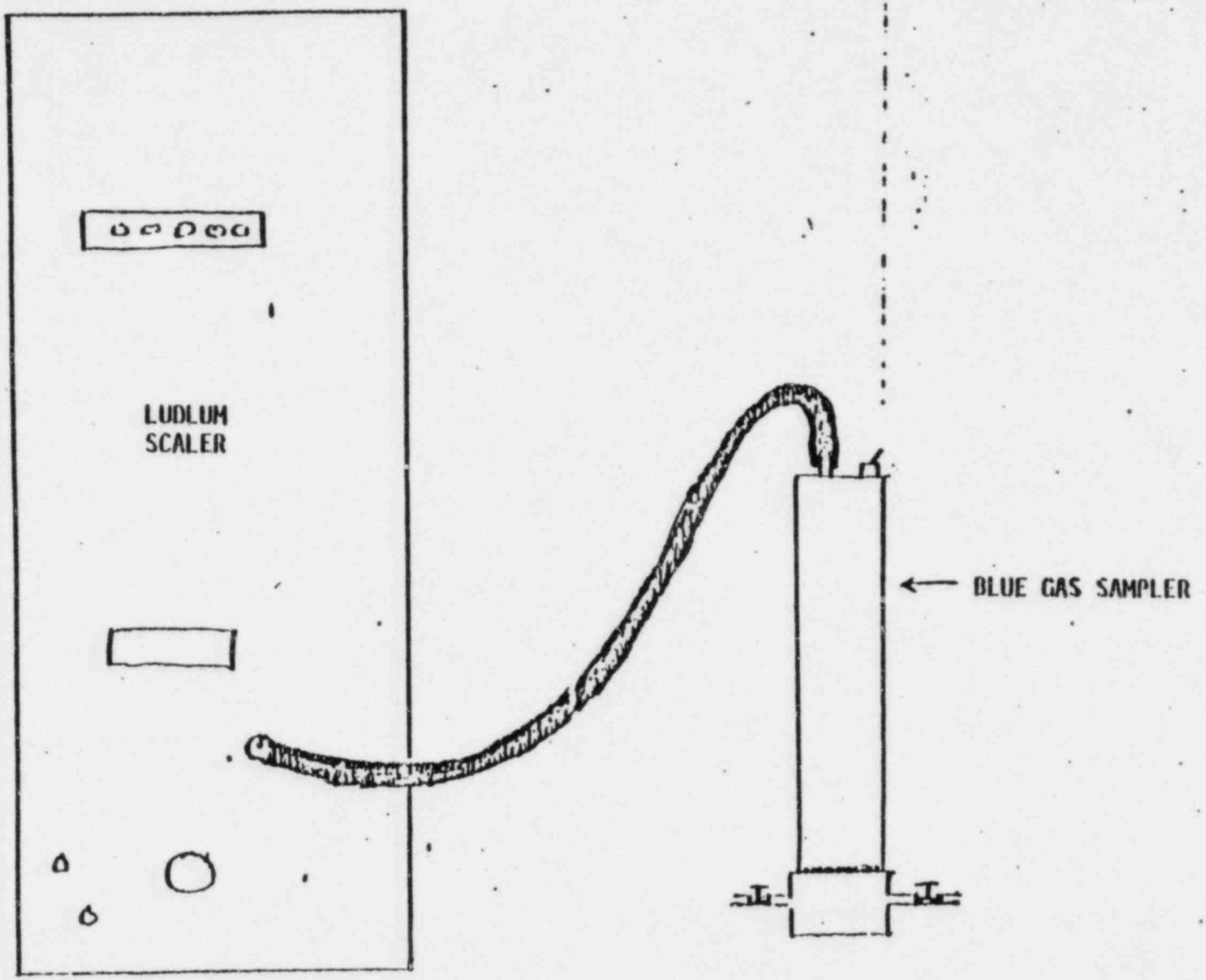
6. Calculate uCi/cc by using the following formula.

$$\frac{cc/min}{(Vol)\ (EFF)\ 2.22 \times 10^6}$$

NOTE: EFF is found on the OPS Scaler Data Sheet

7.

8. Record the results on the Count Data Sheet.



10.1

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APPENDIX 82

PARTICULATE COUNT AND CALCULATION

1. If possible wait 15 minutes from end of collection to count Filter.

2. Place sample in Lead pig on top shelf-Furry side up using open GM-Tube.

3. Set count time to 5 minutes on Timer - Scaler.

4. Press Count Button to start counting.

5. After 5 minute count, divide accumulated counts by 5 and subtract background to obtain corrected counts per minute (cc/m).
   NOTE: Background is found on the OPS Data Sheet

6. Calculate the activity using the following formula.

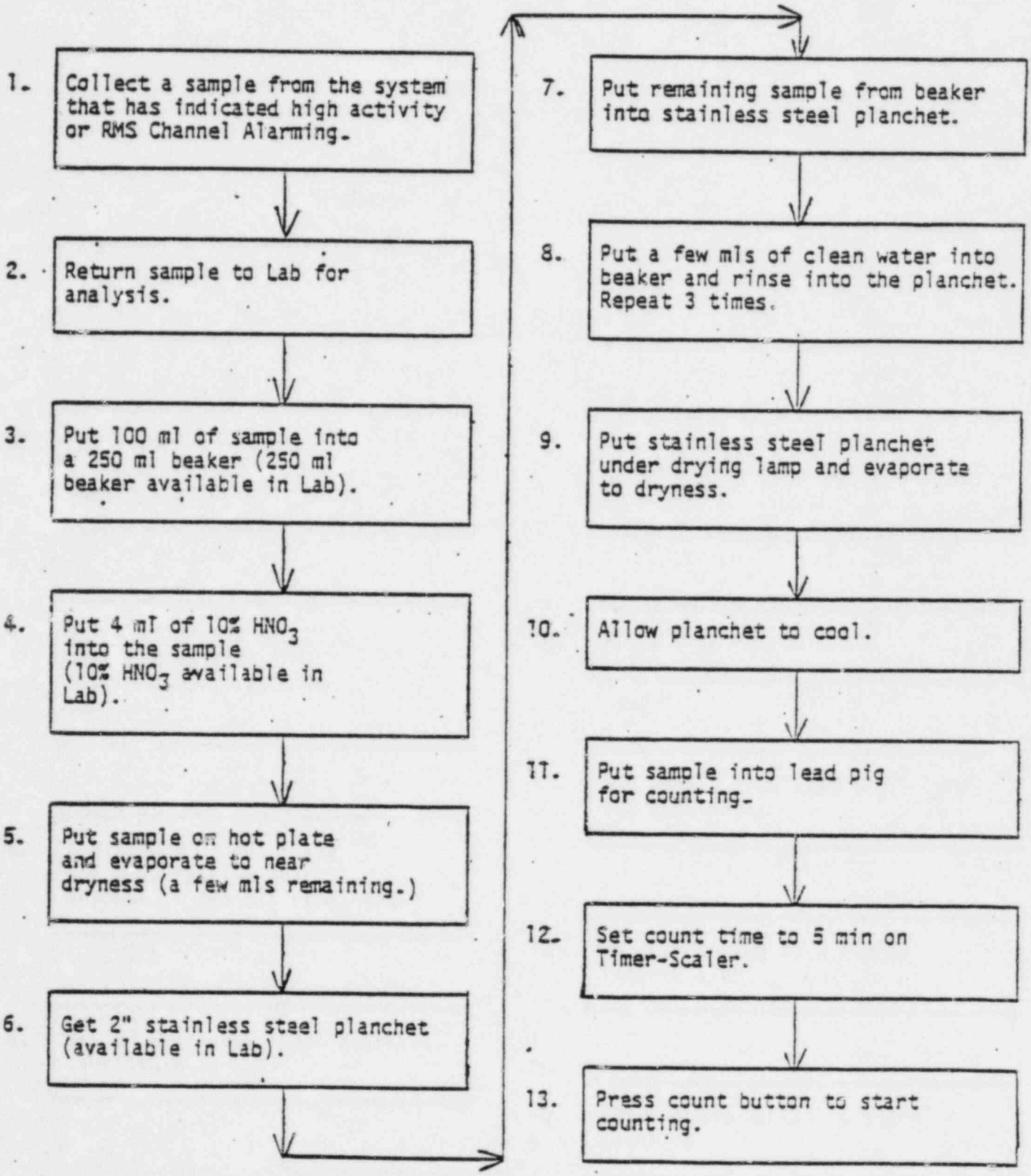
$$\frac{cc/m}{(Vol.)\ (EFF.)\ 2.22 \times 10^{6}}$$

NOTE: EFF. is found on the OPS Data Sheet. Vol is in cc's

7. Record the results on the OPS Data Sheet.

APPENDIX C

LIQUIDS



APPENDIX C (CONT'D)

LIQUIDS

14. After 5 minutes, divide accumulated counts by 5 and subtract background to obtain corrected counts per minute (cc/m).
NOTE: Background is found on the OPS Scaler Data Sheet.

15. Calculate the activity by using the following formula.

16. $$\frac{cc/m}{(Vol)\ (EFF)\ (2.22 \times 10^6)}$$
NOTE: Vol is 250, EFF is found on the OPS Scaler Data Sheet.

17. Record the activity on the OPS Scaler Data Sheet.

ENCLOSURE I

INTERLOCKS ASSOCIATED WITH RADIATION MONITORING SYSTEM HI-ALARMS

1.0 Interlock functions actuated by Hi Alarms from specific atmospheric radiation monitors are as follows:

1.1 Control Room Air Monitor, HP-R-220

a. Fans AH-E-4A and/or AH-E-4B are energized
b. The following dampers assume the position indicated:

(1)	D4092A	closes	
(2)	D4092B	opens	
(3)	D4091A	opens	(with AH-E-4A only)
(4)	D4091B	opens	(with AH-E-4B only)
(5)	D4092C	closes	
(6)	D4092D	closes	
(7)	D4092E	opens	
(8)	D4098	closes	

1.2A Reactor Building Purge Exhaust Monitor, HP-R-225

a. Purge Supply Fan AH-E-12A shuts down
b. The following dampers assume the position indicated:

(1)	D5129A	closes
(2)	D5129D	opens
(3)	D5128A	closes
(4)	D5218D	closes

1.2B Reactor Building Purge Exhaust Monitor, HP-R-226

a. Purge Supply Fan AH-E-12B shuts down
b. The following dampers assume the position indicated:

(1)	D5129B	closes
(2)	D5129C	opens
(3)	D5128B	closes
(4)	D5128C	closes

ENCLOSURE I (CONT'D)

1.3A Auxiliary Building Ventilation Exhaust Monitor (unfiltered), HP-R-222

a. The following dampers assume the position indicated:

(1)	D4020A	closes
(2)	D4020B	opens
(3)	D4020C	opens
(4)	D4020D	opens
(5)	D4020E	opens

1.3B Auxiliary Building Ventilation Exhaust Monitor (Filtered), HP-R-228

a. If operating, the following fans shut down:

(1) Aux. Bldg. Supply Fans, AH-E-7A & AH-E-7B
(2) Aux. Bldg. Exhaust Fans, AH-E-8A, AH-E-8B, AH-E-8C & AH-E-8D

b. The following dampers assume the position indicated:

(1)	D4002	closes
(2)	D4001A	closes
(3)	D4001B	closes
(4)	D4016A	closes
(5)	D4016B	closes
(6)	D4017A	closes
(7)	D4017B	closes

1.4A Fuel Handling Building Ventilation Exhaust Monitor, (unfiltered), HP-R-221A

a. The following dampers assume the position indicated:

(1)	D5683	closes
(2)	D5684A	opens
(3)	D5684B	opens
(4)	D5671A	opens
(5)	D5671B	opens

1.4B Fuel Handling Building Ventilation Exhaust Monitor (Filtered), HP-R-221B

a. If operating, the following fans shut down:

ENCLOSURE I (CONT'D)

(1) Fuel Handling Bldg. Supply Fans AH-E-9A & AH-E-9B
(2) Fuel Handling Bldg. Exhaust Fans - AH-E-10A, AH-E-10B, AH-E-10C & AH-E-10D.

b. The following dampers assume the position indicated:

(1) D5714 closes
(2) D5710 closes
(3) D5712 closes
(4) D5680 closes
(5) D5673 closes
(6) D5670 closes
(7) D5655 closes

1.5 Rad Waste Disposal - Gas Monitors WDG-R-1480, WDG-R-1485 & WDG-R-1486

a. If open, the following valves will close:

(1) WDG-V-30A
(2) WDG-V-30B

1.6 Station Vent Monitor, HP-R-219

a. The interlocks associated with the following monitors are also interlocked to HP-R-219.

(1) HP-R-225 Reactor Bldg. Purge Exhaust Monitor
(2) HP-R-226 Reactor Bldg. Purge Exhaust Monitor
(3) HP-R-228 Aux. Bldg. Ventilation Exhaust Monitor (Filtered)
(4) HP-R-221B Fuel Handling Bldg. Ventilation Exhaust Monitor (Filtered)
(5) WDG-R-1480
(6) WDG-R-1485 Waste Disposal - Gas Monitor
(7) WDG-R-1486

b. The following fans, if operating, will shut down:

(1) AH-E-23A & AH-E-23B, Health Physics Area Ventilation Exhaust Fans.

1.7 Waste Disposal - Liquid Monitor, WDL-R-1311

a. The following valves will colse:

(1) WDL-V99
(2) WDL-V100
(3) WDL-V93A
(4) WDL-V93B

b. The following pumps will be stopped if they are running;

(1) WDL-P8A or 8B
(2) WDL-P11A or 11B

ENCLOSURE I (CONT'D)

I.8 Station Discharge Line Monitor, RM-L7

a. The following valve will close:

(1) WDL-V99

ENCLOSURE II

OPERATIONS PLANNED/UNPLANNED RELEASE REPORT

(SHIFT SUPERVISOR: Complete Sections #1 through #11)

1. DATE: _____ 2. TIME: _____

3. RMS AND METEROLOGICAL CHARTS MARKED: Yes ____ No ____
4. Radiation Protection Dispatched for Environmental Samples Yes ____ No ____
5. DESCRIPTION AND CAUSE: _____

_____

_____

_____

6. IMMEDIATE CORRECTIVE ACTION: _____

_____

_____

_____

7. TOTAL TIME OF RELEASE: _____
8. SAMPLE ANALYSIS:
   a. Gas Sample Yes ____ No ____ Location _____
      Results: _____

      _____
   b. Iodine Sample Yes ____ No ____ Location _____
      Results: _____

      _____
   c. Particulate Sample Yes ____ No ____ Location _____
      Results: _____

      _____
   d. Radiation Survey Yes ____ No ____ Location _____
   e. Contamination Survey Yes ____ No ____ Location _____

9. PERSONNEL EXPOSURE: (List all personnel involved in incident)
Insure film Badges/TLD's are developed within 24 Hours

NAMES	EXPOSURE RESULTS (MREM)	TIME IN AREA OF POTENTIAL RELEASE

10. PERSONNEL CONTAMINATION: Yes ____ No ____

Names: _____

_____

11. Air Flow-Velocity Determination initiated Yes ____ No ____

12. RECOMMENDATIONS FOR CORRECTIVE ACTION: _____

_____

_____

_____

_____

13. FORM COMPLETED BY: _____
Shift Supervisor

(Forward the original form to the Unit Superintendent and copies to the Station Superintendent and the Health Physics Supervisor)
HP to include 20.403 and personnel exposure info. below per page 4.

14. ENGINEERING EVALUATION: _____

_____

_____

_____

_____

COMPLETED BY: _____

15. REVIEWED BY: _____

(Forward the completed form to the Unit Superintendent)

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2202-1.8
Revision 2
01/05/78
G/m) 122

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-1.8
LOSS OF DECAY HEAT REMOVAL

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5.0	01/05/78	2						

Unit 1 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

J F Wilbush Date 1-5-78
Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

J F Wilbush Date 1/5/78

Manager Generation Quality Assurance Approval NA Date —

TMI 55-A Rev 8/77

THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-1.8

LOSS OF DECAY HEAT REMOVAL SYSTEM

1.0 SYMPTOMS

1.1 Increasing Reactor Coolant Temperature, (DH-6-TI 1 & 2 located on Panel #8 indicates >140°F).

1.2 DH removal Loop A/B Flow Lo (DH-1-FI 1/2 located on Panel #8 indicates ≤1500 GPM, (Annun. 8.A22).

1.3 DH Removal Pump 1A or 1B
Suction Pressure LO. Local
Indicator DH-PI-2003 or 2004
Indicates ≤ 6.5 PSIG (Annunc. 8.A21 and/or 8.B21).

1.4 DH Removal Pump Suction Temperature Hi. DH-6-TI 1/2 located on Panel #8 indicates ≥ 300°F, (Annun. 8.C21).

1.5 Pressurizer Level Hi/Lo. RC-I-LI located on Panel #4 indicates ≤ 200" or ≥ 260" H_2O (Annun. 8.B26 and computer points 386, 387, 388).

1.6 DH Pump(s) Overload, (Annun. 8.D21), 4KV ES Bus 2-E1/2-E2 Voltage Loss, (Annunc. 18.A19), 4KV ES Bus Over Current, (computer point 3156).

1.7 D.H. Removal System valves closed or partially closed.

1.8 Loss of Decay Heat Closed Cooling or Nuclear Services River Water.

2.0 IMMEDIATE ACTION

2.1 Automatic Action

a. None

2.2 Manual Action

a. Shift to other string (start alternate Decay Heat Removal, Decay Heat Closed Cooling and Nuclear Service River Water Pumps).

b. Check that valves DH-V100A/B are open, attempt to reopen.

c. Check for leakage, loss of cooling water and valve lineup. If due to leakage isolate affected piping/components.

3.0 FOLLOW-UP ACTION

3.1 If failure exists in common Letdown (DH-V1, DH-V171, DH-V2, or DH-V3) and:

a. R.C.S. Pressure Boundary Intact

(1) Fill, vent (36" hot leg) and pressurize the R.C.S. as necessary to assure the loops are filled.

(2) Start a RC pump and use the steam generators to remove decay heat per Procedure 2102-3.3, Decay Heat Removal via OTSG, until the D.H.R.S. is available for operation.

(3) If conditions do not permit the starting of an R.C. Pump, use Steam Generators to achieve Natural Circulation to remove Decay Heat until the D.H.R.S. is available for operation by accomplishing the following.

(4) Start Electric Driven Emergency Feed Pump EF-P-2A or B, take suction from Condensate Storage Tanks, discharge through EF-V11A/12A & 11B/12B and raise OTSG's level to 50% on the operate range to flood feedwater nozzles.

(5) Dump steam to condenser or to atmosphere as plant conditions allow:

To condenser open MS-V15 A&B, 23 & 24 A/B, and 25 & 26 A/B.

To atmosphere open MS-V1A&B and 3A&B.

CAUTION: If a primary to secondary leak is detected, immediately close the affected OTSG's atmospheric dump valve isolation valve (MS-V1A or 1B) to eliminate the potential of leakage and an unmonitored release path.

3.2 If failure exists in common letdown (DH-V1, DH-V2 or DH-V3 and:

a. R.C.S. Boundary Not Intact and System Open to the Fuel Transfer Canal:

(1) Line up and fill the fuel transfer canal from the BWST as follows:

a. OPEN or insure open DH-V102 A/B. (DH suction from BWST/RB sump isolation valves)

b. VERIFY that DH-V100 A&B are closed, (RCS common letdown to DH pump suction isolation valves).

c. START the decay heat pumps and pump from the BWST via DH-V5A&B to the RCS via DH-V4A&B over flowing into the fuel transfer canal until the fuel transfer canal is full or the BWST Lo-Lo level, (12 ft.), is reached.

CAUTION: If BWST level decreases to 7 feet, DH-V6A/B will automatically open to take suction from the RB sump. If this is undesirable, OPEN DH-V6A/B's breakers from 2-11EA/2-21EA prior to draining the BWST to 7 feet.

(2) Lineup and pump from the fuel transfer canal via the decay heat system into the RCS as follows:

a. OPEN SF-V103, 104, & 105, (Fuel Transfer Canal Common Fill and Drain Isolation Valves).

b. OPEN DH-V110, 101A, & 101B (DH Pump suction from fuel transfer canal isolation valves).

c. START one decay heat pump 1A(1B) and throttle pump flow using decay heat cooler discharge valve DH-V128A(B) as necessary to prevent loss of suction to the DH pumps.

d. ADJUST DHCCW flow to the decay heat cooler by throttling DC-73A(B) to maintain RCS temperature <140°F.

NOTE: The second decay heat pump may be started if both pumps are required to maintain RCS temperature <140°F.

3.3 If failure exists in common Letdown (DH-V1, DH-V2 or DH-V3 and:

a. R.C.S. Pressure Boundary Open Outside Fuel Transfer Canal and a Drain Path to the RB sump is established.

(1) Take suction on B.W.S.T. through DH-V5A & B with Decay Heat Pumps and pump to R.C.S. via DH-V4A/B, overflowing to the Reactor Building Sump.

(2) When B.W.S.T. reaches 7 feet, DH-V6A&B will automatically open to shift suction to Reactor Building Sump. When DH-V6A&B are fully open, CLOSE DH-V5A&B.

(3) Throttle Pump Discharge (DH-V128 A/B) as necessary to prevent Pump Cavitation.

(4) Monitor RB Sump to ensure adequate level is being maintained.

3.4 If failure is on a Core Flood Injection Nozzle (Line Break) and the opposite DHR pump is not available:

Take suction on the B.W.S.T. through DH-V5A(B) with the operable DH pump and pump to the reactor vessel through the DHR cross-connect into nozzle. Use the following procedure.

1. VERIFY/CLOSE DH-V109, 110 (Fuel Transfer Canal Fill & Drain to DH Isolation Valves).
2. OPEN DH-V108A&B (DH cross-connect isolation valves).
3. CLOSE DH-V4A(B) (DH Disch. Isol. Valve) to isolate failure. Line in which failure occurred will be evidenced by hi D.H. injection flow alarm (3750 gpm).
4. VERIFY DH-V4A(B) (DH Disch. Isol. Valve) is open on the unaffected core flood nozzle.
5. When B.W.S.T. reaches 7 feet, DH-V6A&B will automatically open to shift pump suction to the RB sump. When DH-V6A&B are fully open, close DH-V5A&B.

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2202-1.9
Revision 4
09/01/78

G/I(m) 123

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-1.9
LOSS OF INTERMEDIATE CLOSED COOLING WATER

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25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

RP Warren Date 8/30/78

V - Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

J I Seelinger Date 9/1/78

Manager Generation Quality Assurance Approval NA Date —

TMI 55-A Rev 3/77

THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-1.9

LOSS OF INTERMEDIATE CLOSED COOLING WATER

1.0 SYMPTOMS

1.1 I.C. System Flow Low Alarm, 8.F14, 550 GPM.

1.2 I.C. System CRD Cooling Flow Low Alarm, 8.A15, 138 GPM.

1.3 I.C. Surge Tank Level Hi/Lo Alarm, 8.A14.

1.4 I.C. Cooler Outlet Temp. Hi Alarm, 8.E14.

1.5 CRD Stator Temp. Hi, Computer Points 625 through 693 Setpoint 160°F.

1.6 R.C. Pump Cooling Water Temp. Hi, Computer Point 451, 452, 453, & 454.

1.7 Any of the following valves closed due to E.S. testing, 4 pound E.S. Signal, or valve malfunction:

IC-V2, 3, 4, or 5.

1.8 Letdown isolation valve, MU-V376 closes on Hi Temperature of 135°F which secures letdown.

2.0 IMMEDIATE ACTIONS

2.1 Automatic Actions

2.1.1 Standby I.C. Pump starts on low flow (550 GPM) or breaker trip.

2.2 Manual Actions

2.2.1 Verify automatic start of Standby I.C. pump at a low flow of 550 g.p.m. or running pump breaker trip.

2.2.2 If standby I.C. pump has not started, verify an indicated level in the I.C. surge tank and attempt to start manually by turning control switch to "ON".

2.2.3 Check CRD pressure gages locally for CRD filter d/p. Change to standby filter if alarming at 10 psid. Note maximum d/p permitted is 12 psid. Utilize procedure 2104-1.6C for procedure to shift filters.

2.2.4 RCP operation may continue if normal seal injection is maintained.

2.2.5 Closely monitor RCP parameters to insure within the limits specified in 2103-1.4.

3.0 FOLLOW-UP ACTION

3.1 Try to isolate any leakage and/or while filling surge tank from DW-V-46 (Panel 8) isolate portions of system to keep flow in CRD's. Establish normal 18.5 inch surge tank level.

3.2 Verify on Panel 15 that IC-V-5 is open if CRD coolant flow is low. (Alarmed at 138 GPM).

3.3 Verify normal Nuclear Services River Water System Pressure of 17-21 psig on Panel #8, NR-PI 340 thru 343, NR pump discharge pressure gages.

3.4 At or before stator temperature 160°F assign one CRD stator temperature of the controlling CRD group to analog trend recorder.

3.5 If CRD Temp. is High (>170°F) on one rod, reduce power to 60%, drive rod in and de-energize.

3.6 If more than one CRD High Temp. exists (>180°F), reactor must be tripped.

3.7 Backwash IC cooler if High outlet temp. alarm (120°F) has actuated.

3.8 Check flow control and isolation valves for correct positions. (Should be secured in position and position logged on O.P. Checkoff List).

3.9 Re-establish at least the minimum 45 GPM letdown flow.

3.10 If letdown flow cannot be re-established, unit must be shutdown at 385 inches in pressurizer.

3.11 Place standby IC cooler in service for continued High outlet temp. or leakage.

3.12 If RC pump seal temperatures T_1 or T_4 are >180°F, or T_3 is >170°F, shut down and inspect the RCP seals.

6/1(80) 124

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2202-1.10
Revision 8
12/23/78

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-1.10
SHUTDOWN FROM OUTSIDE THE CONTROL ROOM

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24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

N/A Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

J. F. Hilbish Date 12-21-78
V. Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

J. B. Lynch Date 12/23/78

Manager Generation Quality Assurance Approval N/A Date —

TMI 55-A Rev 9/77

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-1.10
SHUTDOWN FROM OUTSIDE THE CONTROL ROOM

1.0 SYMPTOMS

1.1 Conditions such as fire or smoke make continued occupancy of the Control Room impossible.

NOTE: It is very improbable that any condition would prevent safe occupancy of the Control Room; however, if such a condition does occur and total evacuation of the Control Room becomes necessary, then the actions provided in this procedure will be followed to achieve hot standby.

2.0 IMMEDIATE ACTION

2.1 Automatic Action - None associated with this event.

2.2 Manual Action.

2.2.1 Make every effort to eliminate the cause for the Control Room evacuation. As soon as the cause has been eliminated, re-enter the Control Room and restore unit control from that location.

2.2.2 Before evacuating the Control Room, complete the following actions:

a) Depress reactor trip button on Panel No. 4.

b) Depress main turbine trip button on Panel No. 5.

c) Trip steam generator feedpump 1A and 1B from Panel No. 5.

NOTE: This will automatically start EF-P-1, the turbine driven emergency feed pump, and EF-P-2A and 2B, motor driven emergency feed pumps.

d) Place Feedwater Pump Turbine Turning Gear Control switches to ON.

e) TRIP all Reactor Coolant Pumps from Panel No. 4.

NOTE: This will introduce the 21' (50% on operate range) setpoint to the feedwater portion of the ICS.

f) CLOSE MU-V376 letdown block valve, from Panel No. 3.

g) OPEN DHV5A and 5B from Panel 8.

h) OPEN MU-V10, and MU-V127 (locally) set boric acid pump counter and batch controller to max and start the boric acid pumps.

NOTE: If the RBAT is the Tech. Spec. Tank, also start WDS-P-3 and open WDS-V111 from Panel 302.

3.0 FOLLOW-UP ACTION

3.1 If the Control Room was evacuated before any or part of the above immediate actions were completed, perform the immediate actions from local stations as indicated below:

Component	C.R. Action	Action Outside C.R. if Unable to Complete in C.R.	Location
Reactor	Trip	Trip A.C. Breakers NOTE: Tripping the Reactor by this method automatically trips the Turbine Generator.	Control Bldg. 305'
F.W. Pump FW-P-1A & 1B	Trip	Depress "Trip" Push Button on local station at FW Pump Turbines	Turbine Bldg. Elev. 281'-6"
Mtr. Driven Emerg. FW Pumps EF-P-2A & 2B	Start	Depress "START" Push Button on local station at Mtr. Driven Emerg. F.W. Pumps	Aux. Bldg. @ MCC 2-11EA 2-21EA

Component	C.R. Action	Action Outside C.R. if Unable to Complete in C.R.	Location
Reactor Coolant Pumps RC-P-1A, 1B 1C and 1D	Stop	"Trip" R.C. Pump Breakers at 6900V Swgr.	Turbine Bldg. Elev. 305
MU-V-376	Close	Close MU-V-376 from local control station	Aux. Bldg. Elev. 280'6", MU-V16A/B Room
Backup MU Pump	None	Place local CS to STOP	Aux. Bldg. Elev. 281'
MU-V127	None	Open Valve	FH Bldg. Elev. 305' Letdown Valve Alley
MU-V10	Open	Open Valve	" "
CA-V140	None	Check/Insure Valve Open	FH Bldg. Elev. 347' CA Area
CA-P-4A/B	Start	Start from Panel 303	FH Bldg. Elev. 347' CA Area

NOTE: If the RBAT is the Tech. Spec. Boron source WDS-P-3 must be started and WDS-V111 opened from pal. 302

3.2 Upon leaving the Control Room, immediately man the following locations with minimum shift personnel:

a) Shift Foreman and one Control Room Operator (CRO) - Patch Panel located in Control Bldg., Cable Room Elev. 305' to monitor unit parameters and direct the shutdown from outside the Control Room.

b) 1 Man - Auxiliary Building elevation 280'-6" for Reactor Coolant System inventory control and boration.

c) 1 Man - Control Building Area West Elevation 282'-6" for emergency feedwater control EF-P-1, 2A and 2B and/or EF-V32A and $V32_B$.

d) 1 Man - Control Building Area West elevation 282'-6" for atmospheric vent valve control, MS-V3A and 3B.

3.3 Cross tie the Nuclear and Turbine Plant channels of the M&I powered phone circuits in the Instrument Shop.

3.4 CLOSE MU-V154, Pressurizer level control valve isolation. Manually control pressurizer level with MU-V155, Pressurizer level control valve bypass.

CAUTION: During natural circulation cooldown do not exceed a makeup rate of 50 gpm or 1000 gallons for a 1 hour period.

3.5 Monitor Pressurizer Level and MU Tank Level. Open/Close DH-V5A and B as necessary to control MU Tank Level. Pressurizer Level Control can be accomplished by controlling Pressurizer Level MU Flow with MU-V155. If necessary to shift or start additional makeup pumps accomplish by using local control switches with suction from the MU Tank or the BWST.

3.6 VERIFY feedpump turning gears engage when the turbine reaches zero speed.

3.7 When the main turbine rotor has stopped rotating, verify turning gear engages.

NOTE: The reactor is in the hot standby condition when it is sub-critical by at least one percent $\Delta k/k$ and Tave is greater than $525^{o}F$.

3.8 If the Unit is to remain in hot standby condition, maintain the following parameters:

Parameter	Readout Range	Operating Point	Control Station
Pressurizer Level	0-400"	100"	MU-V155 MU Pump Discharge Valve Alley
Makeup Tank Level	0-100"	73"	MU-V119, MU-V319, MU-V8 Letdown Valve Alley or DH-V5A (B)(Aux Bldg 281')
RCS Pressure	0-2500 psig	2155 psig	Operate Pzr. Heater Breakers and Spray Valve RC-V1 (PB at Patch Panel) as required
RCS Temperature	50-650 to Loop A	525°F	MS-V3A
	50-650 to Loop B	525°F	MS-V3B CBA West
OTSG Level	0-100% Operating Range OTSG A	50%	EF-V32A (CBA-West) PB's or EF-V33A (CBA-East)
	0-100% Operating Range OTSG B	50%	EF-V32B (CBA-West) PB's or EF-V33B (CBA-East)

3.9 While maintaining 525°F, the CRO with the Shift Supervisor/Foreman will attempt to reman the Control Room and prove the ability to adequately control the parameters in step 3.8 above. If possible terminate this procedure and remain in Hot Standby with normal operating procedures.

3.10 If the unit is to be cooled down, continue per procedures 2103-3.1, Unit Shutdown, and 2102-3.2, Unit Cooldown, with the exception of controlling plant parameters from the control stations listed in Section 3.8 vice the Control Room and local controls as required.

NOTE 1: Safety Injection RCS pressure low channels can be bypassed as follows from outside the control room. Physically close the following relays by pressing against the relay's magnet:

1. Bistable auxiliary relay 62X/RC1A in Cabinet 133.
2. Bypass relay 43/RC1A in Cabinet 133.
3. Bistable auxiliary relay 62X/RC2A in Cabinet 198.
4. Bypass relay 43/RC2A in Cabinet 198.
5. Bistable auxiliary relay 62X/RC3A in Cabinet 199.
6. Bypass relay 43/RC3A in Cabinet 199.
7. Bistable auxiliary relay 62X/RC1B in Cabinet 201.
8. Bypass relay 43/RC1B in Cabinet 201.
9. Bistable auxiliary relay 62X/RC2B in Cabinet 202.
10. Bypass relay 43/RC2B in Cabinet 202.
11. Bistable auxiliary relay 62X/RC3B in Cabinet 203.
12. Bypass relay 43/RC3B in Cabinet 203.

NOTE 2: The "69" switches for the standby CO/CO Boostar pump pair (HD pump) must be placed to trip prior to tripping an operating CO pump pair (HD pump) to prevent the standby pump pair from auto starting.

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2202-2.1
Revision 7
06/21/78

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-2.1
STATION BLACKOUT

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Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —
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Unit 1 PORC Recommends Approval

NA Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

J. F. Hebdon Date 6-20-78
Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

J. F. Stolz Date 6/21/78

Manager Generation Quality Assurance Approval NA Date —

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE #2202-2.1
STATION BLACKOUT

1.0 SYMPTOMS

1.1 Separation from the 230KV system and Turbine Generator trip as indication by:

1.1.1 Zero volts on 230KV BUS voltmeters on the Electric Control Console Panel No. 6A.

1.1.2 Generator breakers open as indicated by alarm on Control Room Panel No. 18.

2.0 IMMEDIATE ACTION

2.1 Automatic Action

2.1.1 Reactor trips due to loss of voltage to the Control Rod Drive System.

2.1.2 Turbine trips.

2.1.3 Control Room DC lighting is energized.

2.1.4 Turbine Driven Emergency Feedpump starts and commences to feed the OTSG's (Once Through Steam Generators).

2.1.5 Atmospheric steam dump valves, MS-V3A and 3B, control main steam pressure 1010 at psig.

CAUTION: If a primary to secondary leak is detected, immediately close the affected OTSG's atmospheric dump valve isolation valve (MS-V1A or 1B) to eliminate the potential of leakage and an unmonitored release path.

NOTE: The Instrument Air Compressors will not restart automatically. Therefore, the atmospheric steam dump valves, emergency feedwater regulation valves, pressurizer level control valve, and seal injection flow control valve will only function as long as the

reserve air in the air receivers is available. The Instrument Air Compressors should be restarted manually as soon as possible.

2.1.6 Turbine Generator DC Emergency Seal Oil Pump starts.

2.1.7 Turbine Generator DC Turning Gear Oil Pump starts.

2.1.8 DC Emergency Bearing Oil Pump for the Main Feedpump Turbine starts.

2.1.9 The DC Oil Lift pumps for the RC pumps start.

2.1.10 The Emergency Diesel Generators start.
DH-V5A and 5B automatically open.

2.1.11 Both Motor Driven Emergency Feedpumps start.

2.1.12 Ventilation fans in the Control Room, Cable Room, Mechanical Equipment Room, River Water Pump House and Diesel Generator Building start.

2.1.13 Make-up pumps (MU-P-1A and 1C) start. (MU-P-1B may service as Backup to either MU-P-1A or 1C).

2.1.14 Nuclear Services River Water pumps start.

2.1.15 Nuclear Services Closed Cooling Water pumps start.

2.1.16 The Intermediate Closed Cooling Water pump starts.

2.2 Manual Action

2.2.1 Verify:

2.2.1.1 The Turbine Driven Emergency Feedpump starts.

2.2.1.2 Atmospheric Steam Dump valves open when steam pressure reaches 1010 psig.

2.2.1.3 The following DC pumps start.

2.2.1.3.1 Turbine Generator DC Emergency Seal Oil Pump.

2.2.1.3.2 Turbine Generator DC Turning Gear Oil Pump.

2.2.1.3.3 DC Emergency Bearing Oil Pump for the Main Feedpump Turbine.

2.2.1.3.4 DC Oil Lift pumps for the RC pumps.

2.2.1.4 Both Diesel Generators start and the generator voltage is 4160 volts and the frequency is 59-61 Hertz.

2.2.1.5 The voltage available lights Unit Substations 2-11E, 2-12E, 2-21E, 2-22E, 2-31E and 2-41E energize.

2.2.1.6 Verify the following equipment has started and the associated systems are operating properly:

2.2.1.6.1 Nuclear Services Closed Cooling Water Pumps.

2.2.1.6.2 Motor Driven Emergency Feedwater Pumps.

2.2.1.6.3 Nuclear Services River Water Pumps.

2.2.1.6.4 Make-Up Pumps and DH-V5A and B open.

2.2.2 "START" both Instrument Air Compressors per 2104-2.3A. If the Instrument Air Compressors do not have sufficient capacity to maintain the air pressure above 85 psia, "CLOSE" the Turbine Building Instrument Air Isolation valve, IA-V29.

2.2.3 Verify the Intermediate Closed Cooling Water pumps start.

2.2.4 Verify the Emergency Feedwater Regulation valves, EF-V11A and B, are functioning properly and "CLOSE" EF-V32A and B. If EF-V11A and B are not operable maintain OTSG level by manually throttling EF-V32A and B.

2.2.5 Verify the Turbine Driven Emergency Feedpump is functioning properly and "STOP" both Motor Driven Emergency Feedpumps.

2.2.6 Verify RCP seal temperatures did not exceed 185^{o}F, and that seal leakage plus seal return flow does not exceed 1.9 gpm. If either condition exists, the RCP seals should be examined before further RCP operation.

2.2.7 If any of the following ICS Stations are in Hand, Steam Generator Reactor Demand, either Feedwater Demand, Main or Startup Feedwater Valve Demand, Feedpump Speed, Reactor Master and/or Diamond Station, runback the applicable ICS Station

3.0 FOLLOW-UP ACTION

3.1.1 Attempt to restore one 230KV line and BUS based on direction from the Dispatcher. If at least one 230KV is re-energized, re-energize plant 6900V, 4160V and 480V switchgear through the auxiliary transformer to enable start of the Reactor Coolant Pumps and energization of the pressurizer heaters to enable a normal cooldown per 2102-3.2.

3.1.2 If neither 230KV BUS can be energized, continue with cooldown by natural circulation using the emergency feedpumps, the OTSG's and the atmospheric dump valves to control the cooldown rate. Remain within the limits of the curve for cooldown by natural circulation (Figure #1).

NOTE: Pressurizer spray and heaters are not available for pressure control due to power loss. Primary system pressure decrease will be dictated by Pressurizer heat losses to ambient which will cause approximately 100 to 200 psi/hr decrease in primary system pressure.

3.2.1 If (2) MU pumps are running.

3.2.2 Close DH-V5A/B as required to maintain MU Tank level. Trip a MU pump as required to maintain pressurizer level.

3.3 Borate the Reactor Coolant System using the Borated Water Storage Tank as follows:

NOTE: Neither the Boric Acid Mix Tank, the Concentrated Waste Tank nor the Reclaimed Boric Acid Tank are available as a source of borated water during a Loss of Normal Power situation.

3.3.1 "OPEN" DH-V5A and 5B.

3.3.2 Manually "OPEN" the RC Letdown to Bleed Hold-up Tank Isolation Valve WDL-V46.

3.3.3 Determine to which R.C. Bleed Hold-up Tank the RC Letdown will flow and manually "OPEN" the RC Hold-up Tank Inlet Valve, either WDL-V963, WDL-V964 or WDL-V965.

CAUTION: These valves are located in a 3000 mr/hr radiation zone and exposure time must be minimized.

3.3.4 Manually "POSITION" MU-V8 to the "BLEED" position.

3.3.5 "CLOSE" MU-V10.

3.3.6 "CLOSE" MU-V12.

3.3.7 Monitor RC Bleed Hold-up Tank and Borated Water Storage Tank levels.

3.4 Following the addition of boric acid, sample the Reactor Coolant for boron concentration and verify shutdown margin per 2103-1.9.

3.5 During cooldown by natural circulation observe the following limits:

3.5.1 Upper OTSG downcomer to cold leg maximum $\Delta T=25^{\circ}F$.

3.5.2 The maximum cooldown rate is $100^{\circ}F/hr$.

3.5.3 Pressure to temperature limits are dictated by the curve for cooldown by natural circulation, figure #1.

3.6 Manually "JACK" the Main Feedpump Turbines when they have stopped. The turbines should be jacked 180° every 20 minutes.

3.7 Manually "JACK" the Main Turbine when it has stopped. The Main Turbine should be turned 180° every 30 minutes.

3.8 "START" one Control Building Area Fan-coil unit, AH-C-50A or B.

CAUTION: Do not exceed a load of 3 MW on either Diesel Generator.

3.9 If the Reactor Building temperature exceeds $112^{\circ}F$ "START" one RB Normal Cooling Unit and three RB Cooling Fans per 2104-5.1.

3.10 If the RB penetration cooling air outlet temperatures exceed $140^{\circ}F$ "START" one RB Penetration Forced Air Cooling Unit, AH-C-48A or B, per 2104-1.7.

3.11 "START" one Control Building River Water Booster pump and one Control Building Chiller Unit if required.

3.12 Energize portions of the Heat Tracing System as required.

3.13 Between 1820 psig and 1600 psig RC System pressure, bypass all SFAS channels for Safety Injection.

3.14 At 700 psig RC System pressure "CLOSE" CF-V1A and B, and "OPEN" and tag the power supplies to these valves.

3.15 At 500 psig RC System pressure rack out RB Spray pumps BS-P-1A and 1B, breakers to prevent unnecessary actuation and cut in low pressure instrumentation. (RC-3A-PI2).

3.16 At 275°F RC System temperature place the NDTT "AUTO-OFF" control switch for the electromatic relief valve RC-R2 to "AUTO" to actuate the 500 psig relief set point.

3.17 When the RC System temperature is below 250°F shift cooldown from the atmospheric dump valves to the Decay Heat Removal System as follows:

CAUTION: Do not exceed 3 MW load on either Diesel Generator.

3.17.1 Line up the Nuclear Services River Water System for Decay Heat Service Cooler Operation per 2104-3.1F.

CAUTION: Do not start a second NSRW pump in the operating header to prevent overloading a Diesel Generator.

3.17.2 "START-UP" the Decay Heat Closed Cooling Water Loop associated with the Decay Heat Service Cooler placed in operation per step 17a above. Refer 2104-3.3.

3.17.3 "START-UP" the Decay Heat Removal Loop associated with the operating Decay Heat Closed Cooling Water Loop. Refer to 2104-3.3.

3.18 When the Decay Heat Removal System is operating properly and is capable of removing of all the decay heat generated in the Reactor, terminate heat removal with the OTSG's as follows:

3.18.1 "CLOSE" the Atmospheric Dump valves, MS-V3A and B.

3.18.2 Fill the OTSG's to the proper shutdown level.

3.18.3 "STOP" the Turbine Driven Emergency Feedpump and "CLOSE" EF-V11A and B or EF-V32A and B.

HEAT UP/COOLDOWN CURVE

Revision 5
03/30/78

CAUTION:
Operation below system pressure of 500 psig requires use of Figure and the use of the low range pressure instrument.

Point	Temp °F	Press Psig
A	82	155
B	157	476
C	275	476
D	280	1319
E	389	2225

Instrument Error 50 psig
-25 Assumed in TECH Spec -25 for installed recorder

2. Minimum RC pressure to maintain compression force on fuel (natural circulation) inst. error +50 psig = 12°F

3. Minimum RC pressure to maintain compression force on fuel (forced flow) inst. error +50 psig = 12°F

4. Minimum pressure for control rod drive operation inst. error +50 psig = 12°F

5. Minimum RC pressure for single pump in a loop NPSH. (240, 0/2, 27) inst. error +50 psig

6. Minimum RC pressure for two pumps in a loop NPSH. (240, 0/2) inst. error +50 psig

PRESSURE, psig

2300
2200
2100
2000
1900
1800
1700
1600
1500
1400
1300
1200
1100
1000
900
800
700
600
500
400
300
200
100

TEMPERATURE, °F

100 200 300 400 500 600

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Revision 3
10/13/78

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-2.2
LOSS OF STEAM GENERATOR FEED

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24.0			49.0			74.0		
25.0			50.0			75.0		

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Approval N/A Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

N/A Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

RP Warren Date 10/13/78
Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

J. K. Lee (Unit 2) Date 10/13/78

Manager Generation Quality Assurance Approval N/A Date —

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-2.2A

LOSS OF MAIN FEEDWATER FLOW TO BOTH OTSG'S

1.0 SYMPTOMS

1. Both OTSG levels decreasing.
2. Feedwater flow to both OTSGs decreasing.
3. Feed pump turbine speed decreasing or both main feed valves closing.
4. Both feed pump turbine trip alarms.
5. Increasing reactor coolant pressure and temperature.
6. Reactor - turbine runback.
7. Turbine trip if both feed pumps trip.

2.0 IMMEDIATE ACTION

A. AUTOMATIC ACTIONS

1. If loss of FW is due to loss of both feed pumps:
   a. Reactor/Turbine trip due to high RC pressure.
   b. Emergency feed pumps EF-P-1, EF-P-2A, and EF-P-2B start and maintain OTSG level at 30" (S/U range indication).
2. If loss of FW is due to valves closing, ICS trips to track due to FW X-Limits.

B. MANUAL ACTIONS

1. If loss of FW is due to loss of both feed pumps:
   a. Trip the Reactor
   b. Verify Turbine trip and stop valves closed.
   c. Verify EF-P-1, EF-P-2A, and EF-P-2B start as evidenced by pump discharge pressures.

d. Verify emergency feed water valves (EF-11A (B)) are in automatic and controlling OTSG level at 30 inches (S/U range indication).

2. Loss of feed water due to valves closing:

a. Verify or open main and startup feedwater block valves (FW-V-17A (B) FW-V-14A (B), FW-V-19A (B), FW-V-26A (B)).

b. Verify the MWe and reactor power are decreasing.

c. Attempt to open main feed reg. valves.

d. If reactor coolant temperature and pressure cannot be maintained, or if feedwater flow cannot be restored, or if the reactor trips, start the emergency feed pumps and maintain 30 inches in the steam generators (S/U range indication).

3.0 Follow-Up Action

1. Repair system as required and restore normal feed to OTSG.

2. If plant cooldown is required on loss of both main feedwater pumps, conduct cooldown using emergency feed system. Refer to 2102-3.3, Decay Heat Removal Via OTSG.

3. With flow through the emergency feedwater nozzles, there is a possibility of water accumulation in the OTSG steam annulus space, and the steam lines. Insure these areas are drained prior to startup.

THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-2.2B

LOSS OF MAIN FEEDWATER FLOW TO ONE OTSG

1.0 SYMPTOMS

1. Steam generator level decreasing on one OTSG.
2. Feedwater flow to one OTSG decreasing.
3. One main feedwater valve closing.
4. Increasing reactor coolant pressure and temperature.
5. Increasing temperature differential across reactor coolant cold legs.
6. Reactor - turbine runback.

2.0 IMMEDIATE ACTION

A. Automatic Action

1. ICS goes into track due to feedwater cross limits (FW error > + 5%).
2. ICS in tracking mode reduces generated MWe and reactor power to match available feedwater flow.

B. Manual Action

1. Verify that the ICS is in the tracking mode as indicated by both hand and auto lights lit on the unit load demand H/A station.
2. Verify the MWe and reactor power are decreasing.
3. If OTSG level on the affected OTSG is greater than 30 inches, go to manual on H/A station and increase the demand to the associated main and startup FW regulating valve and attempt to regain feedwater flow.

4. Verify that the associated feedwater block valves are open, (FW-V-14A(B), FW-V-17A(B), FW-V-19A(B) and FW-V-26A(B).

5. If any ICS stations are in hand, runback the appropriate ICS stations.

3.0 FOLLOW-UP ACTION

1. If able to restore FW flow prior to Reaching OTSG level of 15 inches, continue normal operation.

2. If unable to restore normal FW flow prior to reaching OTSG level of 15 inches, concurrently reduce reactor power to less than 8% full power and restore feedwater flow to the affected OTSG using emergency feedwater pump and EF-V11A(B).

   NOTE: In the event that an OTSG boils dry, establish feed flow using emergency feedwater pump through the emergency feed valves EF-V-11A(B) very slowly (2 inches per minute).

3. Raise the OTSG level to 25 inches using the emergency feedwater pump and EF-V-11A(B). Restore normal feedwater flow through FW-V-30A(B) amd/or FW-V-25A(B) and shutdown the emergency feedwater system. (stop emergency feedwater pump and close EF-V-11A(B)).

4. With flow through the emergency feedwater nozzles, there is a possibility of water accumulation in the OTSG steam annulus space, and the steam lines. Insure these areas are blown down before continued operation.

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-2.3

LOSS OF INSTRUMENT AIR AND SERVICE AIR

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Cognizant Dept. Head

Unit 2 Staff Recommends Approval

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Cognizant Dept. Head

Unit 1 PORC Recommends Approval

N/A Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

D F Gilbish Date 2.8.78

Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

D F Seelinger Date 2/10/78

Manager Generation Quality Assurance Approval N/A Date —

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-2.3

LOSS OF INSTRUMENT AIR AND SERVICE AIR

1.0 SYMPTOMS

1.1 Instrument Air Compressor Trouble Alarm.

1.2 Service Air Compressors Trouble Alarm.

1.3 Low readings on Instrument Air and Service Air Receiver Pressure Indicators on Panel 17.

2.0 IMMEDIATE ACTION

2.1 Automatic Action

2.1.1 The SA & IA compressor in AUTO will start when IA/SA pressure drops to 100 psig.

2.1.2 SA-V358 closes at 70 psig isolating the Service Air System.

2.2 Manual Actions

2.2.1 Verify running or start all IA and SA compressors.

2.2.2 If Instrument Air pressure drops to 60 psig and continues to decrease:

2.2.2.1 Trip the Reactor and Turbine and carry out the immediate actions of 2202-1.1 and 2203-2.2.

CAUTION: Immediately isolate affected OTSG's atmospheric dump valve by closing the installed Manual valve (MS-V1A or MS-V1B) if a primary to secondary leak is detected.

2.2.2.2 Manually set MU-V32 for 40 gpm seal injection flow.

2.2.3 If, with all Instrument Air and Service Air compressors running, Instrument Air pressure is being maintained at ≈ 70 psig, CLOSE SA-V357 to isolate the Instrument Air System from the Service Air System.

NOTE: If Instrument Air pressure increases after closing SA-V357, the leak is in the Service Air System.

3.0 FOLLOW UP ACTION

3.1 Attempt to locate and isolate the leak in the air system.

3.2 If Instrument Air pressure has continued to decrease below 60 psig:

3.2.1 Manually OPEN MU-V224A and/or 224B, (Inlet to Makeup and Purificaiton demineralizer filters).

3.2.2 Manually OPEN VA-V4A, B and/or C to the running Condenser Vacuum pump.

3.2.3 Manually OPEN Feedwater and Main Steam Penetration Cooling valves (AH-V91, 92, 93, and 94).

3.2.4 Trip the Reactor Building Normal Cooling Water pumps (RB-P-1A and 1B).

3.2.5 Trip the Control Building River Water Booster pumps (NR-P-2A and 2B).

3.2.6 Initiate Reactor Building Emergency Cooling per 2104-5.1.

3.3 If Instrument Air pressure is being maintained >60 psig.

3.3.1 Locate and isolate the leak.

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2202-2.5
Revision 6
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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 PLANT EMERGENCY PROCEDURE #2202-2.5

Station Blackout With Loss of Both Diesel Generators

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1.0	09/22/78	6	26.0			51.0		
2.0	09/22/78	6	27.0			52.0		
3.0	09/22/78	6	28.0			53.0		
4.0	09/22/78	6	29.0			54.0		
5.0	09/22/78	6	30.0			55.0		
6.0	09/22/78	6	31.0			56.0		
7.0	09/22/78	6	32.0			57.0		
8.0	09/22/78	6	33.0			58.0		
9.0	03/17/78	4	34.0			59.0		
10.0			35.0			60.0		
11.0			36.0			61.0		
12.0			37.0			62.0		
13.0			38.0			63.0		
14.0			39.0			64.0		
15.0			40.0			65.0		
16.0			41.0			66.0		
17.0			42.0			67.0		
18.0			43.0			68.0		
19.0			44.0			69.0		
20.0			45.0			70.0		
21.0			46.0			71.0		
22.0			47.0			72.0		
23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval N/A Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

N/A Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

J. F. Colvin Date 9/22/78

Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

J. L. Seelinger Date 9/22/78

Manager Generation Quality Assurance Approval N/A Date —

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 PLANT EMERGENCY PROCEDURE #2202-2.5
Station Blackout with Loss of Both Diesel Generators

1.0 SYMPTOMS

1.1 Separation from the 230KV System and Turbine Generator trip as indicated by:

1.1.1 Zero volts on 230KV BUS voltmeter on the Electric Control Console Panel No. 6A.

1.1.2 Turbine Generator breakers open as indicated by alarms on Control Room Panel No. 18.

1.2 Failure of both Diesel Generators to energize the 4160v BUSES 2-1E, 2-3E, 2-2E, and 2-4E.

2.0 IMMEDIATE ACTION

2.1 Automatic Action

2.1.1 Reactor trips because of loss of voltage to the Control Rod Drive System.

2.1.2 Main Turbine Trips.

2.1.3 Control Room DC lighting is energized.

2.1.4 The turbine Driven Emergency Feedpump starts and commences to feed OTSG's through EF-V11A and B. I.C.S. controls OTSG Level at 50% in operating range.

2.1.5 Atmospheric Steam Dump valves, MS-V3A and B, control main steam pressure at 1010 psig. If pressure continues to increase the main steam reliefs will begin to open at 1050 psig.

CAUTION: If a primary to secondary leak is detected, immediately close the affected OTSG's atmospheric dump valve isolation valve (MS-V1A or 1B) to eliminate the potential of leakage and an unmonitored release path.

NOTE: The Atmospheric Dump valves will operate automatically only as long as the reserve air in the Instrument Air System is available, then the Atmospheric Dump valves must be operated manually.

NOTE: The Turbine Driven Emergency Feedpump Steam Pressure Regulator will only operate as long as the reserve air in the Instrument Air System and the Turbine Driven Emergency Feedpump Steam Pressure Regulator Accumulator (approximately 1 hour) is available, then the steam pressure regulator must be operated manually.

2.1.6 Turbine Generator DC Emergency Seal Oil Pump starts.

2.1.7 Turbine Generator DC Emergency Bearing Oil Pump starts.

2.1.8 The DC Emergency Bearing Oil Pump for the Main Feedpump Turbine starts.

2.1.9 The DC Oil Lift pumps for the RC pumps start.

2.1.10 Electromatic Relief will open if RCS pressure reaches 2255 psig.

2.1.11 Station batteries supply all five inverters.

3.0 MANUAL ACTION

3.1 Manually "CLOSE" the letdown block valve, MU-V376, and the seal return isolation valve MU-V377 to maintain water inventory in the primary system.

3.2 ENSURE the Diesel Generator Selector Switches are in "EMERGENCY STANDBY." The Diesel Generators should start and load automatically. If the Diesel Generators do not start automatically attempt to start them by depressing the start pushbutton on Panels No. 26 and

29. If either Diesel starts verify that the Diesel Generator Breaker closes automatically when frequency is up to 60 Hz and bus voltage is 4160V.

3.3 Verify:

3.3.1 The Turbine Driven Emergency Feedpump starts.

3.3.2 The following DC pumps start.

3.3.2.1 Turbine Generator DC Emergency Seal Oil Pumps.

3.3.2.2 Turbine Generator, DC Emergency Bearing. Oil Pump.

3.3.2.3 DC Emergency Bearing Oil pump for the Main Feedpump Turbine.

3.3.2.4 DC Oil Lift pumps for the RC pumps.

3.4 I.C.S. Maintaining an emergency high level of 50% on operating range in the OTSG's.

3.5 Atmospheric steam dump valves, MS-V3A and B, open when steam pressure reaches 1010 psig.

4.0 FOLLOW-UP ACTION

4.1 For Diesel start failure, check the following and take action as indicated:

4.1.1 Note alarm on Panesl No. 26 and 29 and follow procedure outlined in appropriate Alarm Response.

4.1.2 Observe annunciator panel window "OPERATING ENGINE ALARM RESET" to determine if the Shutdown Relay, SDR, is energized (If SDR is energized, window will alarm 26.C8, 29.C8).

4.1.3 Open engine mounted relay box door and determine if the Auxiliary Stopping Relay, SA, is energized. If energized, Tit will not protrude.

4.1.4 Depress the Reset pushbutton at the Diesel Generator.

4.1.4.1 If the SDR and 5A relays de-energize, correct the cause for the Diesel start failure and attempt to start the Diesel Generators.

4.1.4.2 If the SDR relay de-energizes and the 5A relay remains energized, check the 86 lockout at the 4160V Diesel Generator Breaker and in Diesel Generator Panels No. 308 and 309.

4.2 Attempt to correct problem and start the Diesel Generators.

CAUTION: If Diesel Generator(s) are restored to operation reset and manually open fuel oil pump suction valves. (These valves fail closed on loss of power)

4.3 Attempt to restore one 230KV line and BUS according to direction of the Dispatcher.

4.4 Control cooldown by natural circulation using the Turbine Driven Emergency Feedpump and EF-V11A and B to maintain 50% in operating range in OTSG's. Control cooldown rate by manual control of the Atmospheric Dump valves, MS-V3A and B.

CAUTION: Cooldown ONLY to maintain the RC System hot leg temperature, T_{HL} 30^{o} to 35^{o}F below Pressurizer temperature. The shrinkage due to cooldown will cause the Pressurizer to empty if the RC System T_{ave} reaches 515^{o}F. This will cause the bubble to shift from the Pressurizer to the hot leg and terminate natural circulation.

NOTE: Pressurizer spray and heaters are not available for pressure control. Primary system pressure decrease will be dictated by Pressurizer heat losses to ambient which will cause approximately 100 to 200 psi/hr decrease in primary system pressure.

NOTE: The Instrument Air Compressors are not available. When residual air in the system is used, the Atmospheric Dump valves must be operated manually. The Turbine Driven Emergency Feedpump Steam Pressure Regulator, MS-V14, is provided with an accumulator which will provide approximately one additional hour of automatic operation, after which MS-V14 must be manually operated.

4.5 During natural circulation cooldown (following a complete loss of AC with Diesel Generator failure) observe the following limits:

4.5.1 Upper OTSG downcomer to cold leg ΔT max. = $25^{\circ}F$.

4.5.2 Pressure and temperature limits associated with natural circulation cooldown curve. (Figure 1)

4.5.3 Maintain T_H between $30^{\circ}F$ and $35^{\circ}F$ below the Pressurizer temperature by utilizing manual control of the Atmospheric Dump valves MS-V3A and B.

CAUTION: Maintain T_H more than $30^{\circ}F$ below Pressurizer saturation temperature to prevent boiling in the hot leg.

CAUTION: Maintain T_H within $35^{\circ}F$ of the Pressurizer saturation temperature to minimize the shrinkage of the primary system due to cooldown. Since there is no make-up to the RC System, this ensures the maximum time available in which to restore power prior to loss of Pressurizer level.

4.5.4 The following table gives saturation temperature for given pressure, T_H max. to prevent hot leg boiling and primary shrinkage assuming an initial $T_{ave} = 582^{\circ}$ and a normal pressurizer water volume of 800 $ft.^3$.

Pressure (abs)	Tsat	Th Max.	RC System Shrinkage, ft^3
2155	646	616	-
2000	635	605	-
1800	621	591	-
1500	596	566	90
1200	567	537	495
1000	545	515	775
950	538	508	860

CAUTION: The above table points out that power must be restored to at least one make-up pump prior to RC temperature reaching 515°F.

4.6 If Pressurizer temperature has decreased to 545°F and power has not been restored, "STOP" cooldown as follows:

4.6.1 "CLOSE" the Atmospheric Dump valves, MS-V3A and B.

4.6.2 "CLOSE" the Turbine Driven Emergency Feedpump Steam Supply valve, MS-V207.

4.6.3 Break vacuum in the condenser and shutoff gland seal steam to the Main Turbine and the Main Feedpump Turbines.

4.6.4 Make every possible effort to restore power either to the 230KV transformers from the Dispatcher, by starting one Emergency Diesel Generator.

4.7 Once the Main Turbine has rolled to a stop manually turn the turbine shaft 180° every 30 minutes. One turn of the turbine shaft is equivalent to 701 hand rotations of the turning gear shaft.

4.8 Consider removing the following DC loads to maintain battery reserve when battery voltage decreases to 115V.

4.8.1 Station Battery 2-1SB.

4.8.1.1 Invert 2-5V (Computer).

4.8.1.2 DC High Pressure Oil Lift pumps for RC-P-1A and RC-P-1B (after the RC pumps have stopped rotating).

4.8.1.3 DC Emergency Bearing Oil pump for the Main Feedpump Turbine, FW-P-1A.

4.8.1.4 Purge the Main Generator with CO_2 and stop the Generator H_2 Seal Oil Pump SO-P-2.

4.8.2 Station Battery 2-2SB.

4.8.2.1 DC High Pressure Oil Lift pumps for RC-P-2A and RC-P-2B (after the RC pumps have stopped rotating).

4.8.2.2 DC Emergency Bearing Oil pump for the Main Feedpump Turbine, FW-P-1B.

4.8.2.3 Turbine Generator DC Emergency Bearing Oil pump (only if the Main Turbine cannot be hand jacked).

4.9 Loss of AC power for several hours to the 230KV breakers may result in low SF_6 gas pressure and the SF_6 gas pressure interlock would prevent closing the breakers. If the Diesel Generators are not yet available and enough air remains in the compressed air tank, the breakers may be closed as follows:

4.9.1 Verify that the line and bus on each side of the breaker are de-energized.

4.9.2 Co-ordinate closing of the breaker with the dispatcher and Lebanon Relay Department.

4.9.3 Pull the fuses for both Trip circuits and jumper out the closing interlock (under direction of the relay department).

4.9.4 "CLOSE" the breaker.

CAUTION: Do not trip the breaker after energizing until SF_6 gas pressure is restored.

4.9.5 Energize the line from the Middletown junction.

4.9.6 Restore 480V AC power to the breaker compressors. (Both air and SF_6 gas compressor).

4.9.7 When gas pressure is automatically restored in approximately one or two hours, replace the trip fuses and remove the jumper in the closing circuit.

FIGURE 1

HEAT UP/COOLDOWN CURVE

Revision 4
03/17/78

CAUTION:
Operation below system pressure of 500 psig requires use of Figure 1 and the use of the low range pressure instrument

Point	Temp °F	Press psig
A	82	155
B	157	476
C	275	476
D	280	1319
E	389	2225

Instrument Error 50 psig
-25 Assumed in TECH Spec -25
for installed recorder

2. Minimum RC pressure to maintain compression forces on seals (normal circulation) inst. error +50 psig -12°F

3. Minimum RC pressure to maintain compression forces on fuel (forced flow) inst. error +50 psig -12°F

4. Minimum pressure for control rod drive operation inst. error +50 psig -12°F

5. Minimum RC pressure for single pump in a loop NPSH (1/0, 1/1, 2/1) inst. error +50 psig

6. Minimum RC pressure for two pumps in a loop NPSH (2/0, 2/1, 2/2) inst. error +50 psig -12°F

RC PRESSURE, psig

TEMPERATURE, °F

2300
2200
2100
2000
1900
1800
1700
1600
1500
1400
1300
1200
1100
1000
900
800
700
600
500
400
300
200
100

100 200 300 400 500 600

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2202-2.6
Revision 4
10/06/78

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-2.6

OTSG TUBE RUPTURE
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6.0	04/18/78	3	31.0			56.0		
7.0			32.0			57.0		
8.0			33.0			58.0		
9.0			34.0			59.0		
10.0			35.0			60.0		
11.0			36.0			61.0		
12.0			37.0			62.0		
13.0			38.0			63.0		
14.0			39.0			64.0		
15.0			40.0			65.0		
16.0			41.0			66.0		
17.0			42.0			67.0		
18.0			43.0			68.0		
19.0			44.0			69.0		
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23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

RP Warren Date 10/6/78

V - Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

[signature] Date 10/6/78

Manager Generation Quality Assurance Approval NA Date —

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-2.6

OTSG TUBE RUPTURE

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1.0 SYMPTOMS

A. Initial loss of reactor coolant pressure and decrease in pressurizer level becoming stable after short period of time.

B. Make-up tank level decreasing

C. VA-R748 off gas monitor alarm.

D. Secondary sample analysis indicates activity in the secondary side.

2.0 IMMEDIATE ACTION

A. Automatic Action

1. MU-V17 will open to compensate for reduced pressurizer level.

B. Manual Action

1. Verify MU-V17 is maintaining pressurizer level at 220". Close MU-V376 and start an additional make-up pump if pressurizer level drops to 200".
2. Verify VA-R748 alarm.
3. Notify Unit I Control Room and isolate steam to Unit I by closing AS-V23.
4. Direct chemistry department to sample both OTSG's and analyze for radioactivity (β' γ, ^{3}H, I-133, Cs-137) to determine which OTSG has the leak.
5. Using an RM-14, or equivalent instrument, take a gamma reading in each steam line outside the Reactor Building. Attempt to take the readings on each steam line as close to the same relative position as possible.

6. Based on the readings taken in 4 above, attempt to determine which OTSG has the leak.

7. If make-up-tank level is decreasing at greater than 10 GPM (1 inch/3 minutes) and VA-R748 is in alarm, immediately begin reducing load at 10%/minute.

8. CLOSE the affected OTSG's atmospheric dump valve isolation valve (MS-V1A or B) to eliminate the potential of leakage and an unwarranted release.

9. If desired to aid in determining which OTSG is leaking, dump all MSR drains to the condenser per 2106-3.1.

C. Follow-Up Action

1. If make-up-tank level is not decreasing at greater than 10 GPM, calculate the RCS leak rate per Surveillance Procedure 2301-3D1. If the identified leak rate exceeds 10 GPM or primary to secondary OTSG tube leakage exceeds 1 GPM or unidentified leakage exceeds 1 GPM proceed with Technical Specification 3.4.6.2 Action Statement.

2. If make-up-tank level is decreasing at greater than 10 GPM and VA-R748 is in alarm proceed as outlined below.

   a. At 15% power place diamond in manual and fully insert the regulating groups 7, 6, and 5 (this is to reduce likelihood of lifting secondary safety valves).

   b. While reducing Rx Power to zero by inserting rod groups 7, 6, and 5, go to manual on the turbine bypass valves and open the turbine bypass valves to take steam from the turbine and thereby reduce load.

CAUTION: Do not increase OTSG Levels to 97-99% on the Operating Levels to avoid a possible R.C.S. Boron Dilution Accident during subsequent plant cooldown.

c. Trip the turbine. Then place the turbine bypass valves back to automatic and verify the turbine bypass valves maintain header pressure at 895 psig.

d. Follow up on chemistry analysis requested in Step 4 of manual action. Verify that gamma analysis has been performed in accordance with HP-1950 on samples taken from below listed points. For a large rupture (greater than 50 gpm leakage) the leaking OTSG level will hang up. For smaller leaks (Sodium-24) chemistry analysis will have to be used to determine which OTSG has the leak.

Steam Generator Inlet A

Steam Generator Inlet B

Main Steam A

Main Steam B

Secondary Side Stm. Gen A

Secondary Side Stm. Gen B

e. Isolate the effected OTSG by closing the following on the effected OTSG when RCS Th is <545°F.

Main FW Reg. Valve	FW-V30 A or B
Main FW Block Outlet Valve	FW-V17 A or B
S.U. FW Reg. Valve	FW-V25 A or B
S.U. FW Block Inlet Valve	FW-V26 A or B

S.U. FW Block Outlet Valve	FW-V19 A or B
Bypass FW S.V. Valve	FW-V66 A or B
Emerg. FW Reg. Valve	EF-V11 A or B
Emerg. FW Block Valve	EF-V12 A or B
Emerg. FW Reg. Valve Bypass	EF-V33 A or B
Emerg. FW Block Valve Bypass	EF-V32 A or B
Turbine Bypass Valves	MS-V25 A or B
	MS-V26 A or B
Turbine Bypass Block Valves	MS-V23 A or B
	MS-V24 A or B
Main Stm. Isolation Valves	MS-V4 A or B
	MS-V7 A or B
Stm. Gen. Reheat Stm. Isolation Valves	MS-V15 A or B

Assure the FW pump for the unaffected OTSG is running, or run the FW pumps off of Auxiliary steam.

CAUTION: The "A" FW Pump Turbine receives HP Steam from the "B" OTSG and the "B" FW Pump Turbine receives HP Steam from the "A" OTSG.

NOTE: Monitor isolated steam generator pressure, and if it approaches 1050 psig (MS relief valve setpoint) re-establish a flowpath to the condenser via the turbine bypass valves.

f. Take manual control of main and startup reg. valves and adjust uneffected OTSG level as needed to obtain a Tave of 532^{o}F.

g. Place Pressurizer Pressure Controller in manual and adjust output to maintain RCS Pressure at 1850 psig. Turn pressurizer heater banks 4 & 5 off.

NOTE: This will aid in lowering primary to sec. leak rate.

h. Open the MU pump suction valve from the BWST DH-V5 A and B and isolate MU-V12 if required, due to low make-up-tank level.

i. Begin a cooldown to cold shutdown as soon as possible in accordance with 2102-3.2 to minimize inleakage to the affected OTSG.

j. Open breakers for the following sump pumps to terminate any potential radioactive release:

Turbine Building Sump Pumps

SD-P-1A Unit 3B MCC 2-31A

SD-P-1B Unit 9C MCC 2-41A

Control Building Area Sump Pumps

SD-P-3A Unit 4B MCC 2-31C

SD-P-3B Unit 5C MCC 2-41C

k. CLOSE the following valves:

_____SD-V48 Control Building area sump discharge to non-radioactive waste treatment building sump.

____SD-V38 Turbine Building Sump discharge to non-radioactive waste treatment building sump.

_____WDL-V1160 Secondary Chem. Lab drain to control and service building sump.

l. OPEN or insure OPEN the following valves:

_____WT-V123A(B) Cond. Polisher reg. sump discharge to WDL misc.

CAUTION: Insure spool piece is in place between WDL-V403(404) and WDL-V361.

_____WDL-V403(404) Cond. Polisher reg. sump discharge to WDL Misc. spool piece isol.

_____WDL-V361 Con. polisher reg. sump discharge to WDL-F 8A/B.

_____SD-V29A Control Building area sump discharge to condensate polisher regeneration sump.

_____SD-V28A Turbine Building sump discharge to Cond. polisher reg. sump.

_____WDL-V1159 Secondary Chem Lab Drain to Cont. Drain Tanks.

m. Insure WT-R3894 and 3895 are operating properly to direct radioactive wastes to the WDL Misc. System.

n. CLOSE BREAKERS OPENED IN STEP h.

o. Notify Health Physics to begin surveys of the Control Building area and Trubine Building to determine the need for controlled areas. Inititate Emergency Plan if required, as a result of the surveys.

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2202-2.7
Revision 2
02/08/79

THREE MILE ISLAND NUCLEAR STATION
EMERGENCY PROCEDURE 2202-2.7
HIGH CATION CONDUCTIVITY AND/OR SODIUM IN THE CONDENSATE AND/OR FEEDWATER SYSTEM
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Approval NA Date —

Cognizant Dept. Head

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Approval NA Date —

Cognizant Dept. Head

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NA Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

RP Warren Date 2/2/79

V- Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

[signature] Date 2/1/79

Manager Generation Quality Assurance Approval NA Date —

TMI 55-A Rev 8/77

THREE MILE ISLAND NUCLEAR STATION

EMERGENCY PROCEDURE 2202-2.7

HIGH CATION CONDUCTIVITY AND/OR SODIUM IN THE CONDENSATE AND/OR FEEDWATER SYSTEM

1.0 SYMPTOMS

1.1 Any or all of the following Control Room Alarms

a. Panel 17, annunciator A.3 - "Cond. Polish. Sys. Trouble"

b. Panel 17, annunciator E.6 - "secondary Sampl. Sys. Trouble"

c. Panel 17, annunciator F.6 - "Cond./FW Chemistry Trouble"

1.2 The Control Room conductivity recorder indicates an increasing conductivity.

1.3 Increasing conductivity indication on recorders SS-CR-3153 and SS-CR-3154 at panel 310.

1.4 Condensate Polishing Control Panel 304 conductivity recorder indicates an increasing conductivity.

1.5 Sodium analyzer at condensate pump discharge, condensate booster pump discharge or final feedwater points are reading 5 ppb or greater.

2.0 IMMEDIATE ACTION

Automatic - None

2.1 Manual

2.1.1 Change the polisher with the most service time.

2.1.2 If conductivity is $\geq$ 1.0 umho, determine which side of the condenser is leaking as indicated by the conductivity recorders in chemistry lab, reduce power to 50%, and isolate the affected side by:

1. Stopping the pumps on the affected side.
2. Close C.W. Cross-Connect (CW-V3, CW-V19).
3. Close CW-V41A(B)
4. Close CW-V11A(B)

2.1.3 If sodium value at condensate pump discharge analyzer is 5 ppb or greater, determine which side of the condenser is leaking and isolate the affected side per 2.1.2. The following time schedule should be observed:

1. If Na 5-10 ppb, isolate within 24 hr.
2. If Na 10-20 ppb, isolate within 12 hr.
3. If Na greater than 20 ppb, isolate immediately.

2.1.4 Monitor Control Room Conductivity Recorded on Panel 17 point #3 (SS-6) - Feedwater Inlet to "A" OTSG point #4 (SS-7) - Feedwater Inlet to "B" OTSG <u>If conductivity reaches:</u>

a. 0.3-0.5 umhos, shutdown should be initiated within 2 weeks if uncorrected.

b. 0.5-1.0 umhos, shutdown should be initiated within 24 hrs. if uncorrected.

c. 1.0-1.5 umhos, shutdown should be initiated within 12 hrs. if uncorrected.

d.. 1.5-2.0 umhos, shutdown should be initiated within 6 hrs. if uncorrected.

*e. 2.0 umhos, shutdown should be initiated immediately.

*f. <u>5.0 umhos, Trip Turbine, Trip the reactor.</u>

<u>*NOTE:</u> It is assumed that these indications are not due to instrument malfunction. This can be qualitatively confirmed by a similar conductivity increase at point #1 (SS-2) Condensate pump discharge, or point #2 (SS-2) Condensate pump discharge, or point #2 (SS-3) Condensate Polisher Effluent.

2.1.5 If sodium value at analyzer connected to final feedwater sample point (SS-6 or SS-7) exceeds 5 ppb shutdown must be initiated according to the following schedule:

*1. If Na 5-10 ppb for two weeks initiate normal shutdown.

*2. If Na 10-20 ppb for 24 hr. initiate normal shutdown.

*3. If Na greater than 20 ppb. initiate normal shutdown.

*NOTE: A rough check of the analyzer calibration for the analyzer monitoring final feedwater during steady-operation can be made by analyzing a sample from either MSR belly drain sample point for sodium and dividing the result by 11. If this value agrees within 15-25% of final feedwater value, the instrument may be indicating correct values.

3.0 FOLLOW UP ACTION

3.1 If Control Room Alarm 17.A.3, "Condensate Polishing Sys. Trouble" is alarming (it is initiated by any alarm at Condensate Polishing Control Panel 304) and the alarm at panel 304 is due to High Conductivity, complete the Manual Action Required per the alarm responses for panel 304. Check A1, A2, A4, A8, A12, A14, A16, A18 and A20.

3.2 If Control Room alarm 17.E16 "Secondary Sampling System Trouble" or 17.F.6 "Condensate/Feedwater Chemistry Trouble" is alarming (Both are initiated by an alarm condition at panel 310 in the Secondary Sample Room) and the alarm at panel 310 is due to High Cond/FW conductivity, complete the Manual Action Required per the alarm responses for panel 310. Check A12, A14, A15, A16, A17, A18, A20 and B12.

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-2.8
LOSS OF ONE AUXILIARY TRANSFORMER

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5.0			30.0			55.0		
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23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

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Approval NA Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

RP Warren Date 7/18/78

Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

[signature] Date 7/19/78

Manager Generation Quality Assurance Approval NA Date —

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-2.8
LOSS OF ONE AUXILIARY TRANSFORMER

1.0 SYMPTOMS

1.1 The following alarms may annunciate on Panel 18:

1.1.1 Loss of Auxiliary Transformer 2A-

18.F11 Bus 4/8 Differential

18.B6 2A Aux. Transf. Sudden Pressure

18.C6 2A Aux. Transf. Voltage Loss

1.1.2 Loss of Auxiliary Transformer 2B-

18.F11 Bus 4/8 Differential

18.E6 2B Aux. Transf. Sudden Pressure

18.F6 2B Aux. Transf. Voltage Loss

1.2 All 4KV and 7KV busses have fast transferred to the energized Aux. Transf.

1.3 The 230KV Bus 4 or Bus 8 voltmeters on Panel 6 indicated zero volts.

2.0 IMMEDIATED ACTION

2.1 Automatic Action

2.1.1 The following pumps will trip and the starting of the appropriate back-up pumps will be blocked upon loss of either Aux. Transf. in accordance with Attachment 1.

2.1.2 The ICS will runback unit load to approximately 50% after a condensate booster/feedwater pump trip.

2.2 Manual Action

2.2.1 Verify that the appropriate pumps have tripped or have not started in accordance with Attachment I. If pumps have not tripped automatically, manually trip the appropriate pumps.

2.2.2 Verify that the ICS has initiated a runback to approximately 50% power. If not, manually initiate runback to 50% power.

3.0 FOLLOW-UP ACTION

3.1 Ensure that E.S. busses 2-1E/2-2E voltage is maintained at ≥ 3,700 volts as indicated on Panel 6 voltmeters. If necessary, trip an additional CWP or Heater Drain Pump and adjust the operating power level accordingly.

3.2 Attempt to restore the inoperative Aux. Transf. as soon as possible.

3.3 Comply with the T.S 3.8.1.1 or T.S.3.8.1.2 Action Statements.

ATTACHMENT I

Emergency Tripping of Pumps upon Loss of One Aux. Transf.

1. If CW-P-1A, CW-P-1B, and CW-P-1C are all running, CW-P-1A will trip after a time delay of 15 sec. after loss of one Aux. Transf.
2. If CW-P-1D, CW-P-1E, and CW-P-1F are all running, CW-P-1D will trip after a time delay of 15 sec. after loss of one Aux, Transf.
3. If CO-P-2A and CO-P-2C are running, CO-P-2A will be tripped and locked out. Also, the interlocked tripping of CO-P-1A will be blocked.
4. If CO-P-2B and CO-P-2C are running, CO-P-2B will be tripped and locked-out. Also, the interlocked tripping of CO-P-1B will be blocked.
5. If CO-P-2A and CO-P-2B are running, CO-P-2B will be tripped and locked-out. Also, the interlocked tripping of CO-P-1B will be blocked.
6. If HD-P-1A and HD-P-1C are running, HD-P-1A will be tripped and locked-out. Also, the interlocked starting of back-up pump HD-P-1B will be blocked.
7. If HD-P-1B and HD-P-1C are running, HD-P-1B will be tripped and locked-out. Also, the interlocked starting of back-up pump HD-P-1A will be blocked.
8. If HD-P-1A and HD-P-1B are running, HD-P-1B will be tripped and locked-out. Also the interlocked starting of back-up pump HD-P-1C will be blocked.

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THREE MILE ISLAND NUCLEAR STATION
UNIT 2 EMERGENCY PROCEDURE 2202-2.9

PLANT RESPONSE TO PENETRATION OF THE PROTECTED AREA

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19.0			44.0			69.0		
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23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

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Cognizant Dept. Head

Unit 2 Staff Recommends Approval

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Cognizant Dept. Head

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N/A Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

J. F. Gillish Date 5/19/78

Chairman of PORC

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N/A Date —

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J. J. Colitz Date 5/19/78

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THREE MILE ISLAND NUCLEAR STATION
UNIT 2 EMERGENCY PROCEDURE 2202-2.9
PLANT RESPONSE TO PENETRATION OF THE PROTECTED AREA

1.0 SYMPTOMS: (One or more of the following).

1.1 A penetration (forcible entry) of a protected area barrier defined by (Figure 1) has occurred based on visual, confirmed, observation, and the intruder's failure to halt, as reported by a Site Protection Officer/Sergeant to the Shift Supervisor or Shift Foreman.

1.2 Station Security Procedures have been implemented to apprehend the person(s) who penetrated (forced entry) the protected area barrier, and State Police have been notified that the protected area barrier has been penetrated.

1.3 Hostile groups or antinuclear demonstrators are present near the facility perimeter and have increased the potential protected area penetration threat level.

2.0 IMMEDIATE ACTION:

2.1 Automatic Action: None.

2.2 Manual Action

a. If the protected area barrier (Figure 1) has been penetrated and the person(s) accomplishing the penetration is not immediately physically apprehended as confirmed by a Site Protection Officer direct to the Shift Supervisor or Shift Foreman, orally by radio or walkie-talkie, then the Unit shall be immediately shutdown by tripping the reactor and carrying out the Reactor Trip Emergency Procedure (2202-1.1).

CAUTION: If it is clearly determined that the penetrator does not pose a real threat to the safety of the unit, the reactor should not be tripped.

b. Announce the reactor trip and the reason for the trip over the page and bring the Unit to the hot standby condition and establish stable plant conditions.

3.0 FOLLOWUP ACTIONS:

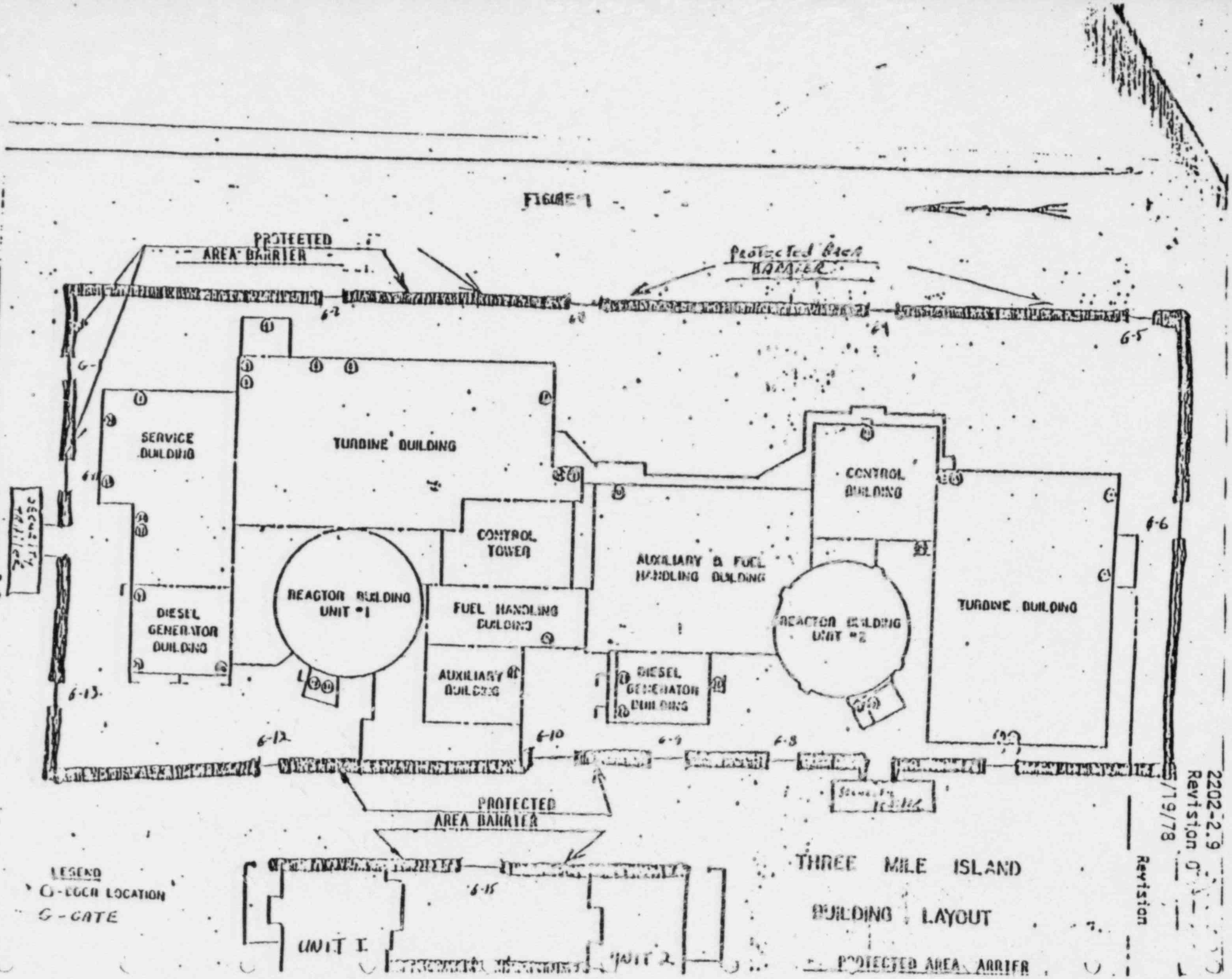
3.1 If the person(s) who have accomplished the penetration are not immediately physically apprehended, the security force with the help of other plant personnel shall ascertain the intruder's probable location and direct personnel to watch that area until the area can be safely entered.

3.2 Depending upon the nature and magnitude of the penetration and the resulting potential threat, other protective measures should be considered which may include, but not be limited to the following:

a. Locking the Control Room doors and Access Control point doors.

b. Close the Auxiliary Building or Fuel Handling Building concrete doors, if open, or station a guard at these locations.

c. Muster all station personnel by requesting personnel to call the Unit 2 Control Room.


PROTECTED AREA BARRIER
Protected Area BARRIER
SERVICE BUILDING
TURBINE BUILDING
CONTROL BUILDING
CONTROL TOWER
AUXILIARY & FUEL HANDLING BUILDING
REACTOR BUILDING UNIT #1
FUEL HANDLING BUILDING
DIESEL GENERATOR BUILDING
AUXILIARY BUILDING
REACTOR BUILDING UNIT #2
DIESEL GENERATOR BUILDING
TURBINE BUILDING
PROTECTED AREA BARRIER
UNIT 1
UNIT 2
LEGEND
G-ECCH LOCATION
G-GATE
THREE MILE ISLAND
BUILDING LAYOUT
PROTECTED AREA BARRIER
2202-2.9
Revision 0
5/19/78
Revision

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Revision 4
10/06/78

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-3.1
FIRE

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4.0	10/06/78	4	29.0			54.0		
5.0	06/13/77	0	30.0			55.0		
6.0	09/08/77	1	31.0			56.0		
7.0	09/08/77	1	32.0			57.0		
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12.0	06/13/77	0	37.0			62.0		
13.0	06/13/77	0	38.0			63.0		
14.0	06/13/77	0	39.0			64.0		
15.0	06/13/77	0	40.0			65.0		
16.0	06/13/77	0	41.0			66.0		
17.0	10/06/78	4	42.0			67.0		
18.0	02/10/78	2	43.0			68.0		
19.0	02/10/78	2	44.0			69.0		
20.0	02/10/78	2	45.0			70.0		
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N/A Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

R P Warren Date 10/6/78
V- Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

J L Seelinger Date 10/6/78

Manager Generation Quality Assurance Approval N/A Date —

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-3.1

FIRE

1.0 SYMPTOMS

Symptoms of a fire or potential fire emergency include any or all of the following:

a. Fire-Smoke alarm received on Panel 7 in the Control Room.

b. Temperature alarm received on Heating and Ventilation, Panel 25, Annunciator in the Control Room.

c. Indication of deluge system actuation on Panel 7.

d. Sight or smell of smoke or flames in any location.

2.0 IMMEDIATE ACTION

2.1 Automatic Action

a. Deluge system actuation for applicable area.

b. Sprinkler system actuation for applicable area.

c. Halon system actuation for fire in Air Intake Tunnel.

1. All three tunnel deluge systems actuate.

2. Air intake fire dampers AH-D-2A, 2B, 2C, 2D, 2E, and 2F close.

3. Air intake tunnel sump pumps (SD-P-4A and B) and air intake tunnel emergency pumps (SD-P-11A and B) start.

4. Ventilation systems respond (refer to Appendix D).

d. Fire pumps start on Low Fire Main pressure indicated below:

Motor Fire Pump Unit 1	(FS-P-2)	90 psig
Diesel Fire Pump Unit 2	(FS-P-1)	85 psig
Diesel Fire Pump Unit 1	(FS-P-3)	80 psig
Cool Tower Fire Pump Unit 1	(Also designated FS-P-1)	70 psig

e. Actuation of the Halon System in the Cable Room.

2.2 Manual Action

a. Announce the following using the page channel: "Fire, (Location), (Class, if known), Fire brigade report to Location".

b. Actuate fire alarm by depressing and releasing the pushbutton on Panel LATER in the Control Room.

c. If the fire main pressure decreases, verify the starting of the fire pumps by observing the illuminated red running indication lights on Panel 7. If the Diesel Fire Pump Unit 2 (FS-P-1) has not started, dispatch an Aux Operator to start the pump manually in accordance with the Fire Protection System procedure 2104-6.1. Coordinate with Unit 1 the starting of the Motor Fire Pump Unit 1 (FS-P-2), the Diesel Fire Pump Unit 1 (FS-P-3] and the Cool Tower Fire Pump Unit 1 (also designated FS-P-1),

NOTE 1: The announcement of any fire in an enclosed "Out Building" or any other area requiring admittance by a security guard shall require the roving security officer to report to the scene and provide the necessary entry.

NOTE 2: Refer to Appendix A for a fire in the Control Room. Refer to Appendix B for a fire in the Cable Room, and refer to Appendix C for a fire in the Reactor Building.

3.0 FOLLOW UP ACTION

3.1 Duties and Responsibilities.

3.1.1 Control Room Operator.

a. Perform required Manual Action (refer to paragraph 2.2).

b. Carry out Manual Action as specified in applicable alarm responses.

3.1.2 Shift Supervisor.

a. Report to the scene of the fire and coordinate the fire fighting activity.

b. Verify Fire Brigade personnel have reported to the scene of the fire with the equipment outlined in the following table:

Fire Brigade Personnel	Equipment	Remarks
Aux Operator #1	Ansul Portable Fire Ext.	Use fire ext. and/or fire hose to fight fire per direction of Shift Supervisor
Aux Operator #2	Ansul Portable Fire Ext.	Same as above
Aux Operator #3	Self Contained Breathing Apparatus	Take action to protect equipment and/or personnel as directed by Shift Supervisor
Aux Operator #4	Self Contained Breathing Apparatus	Same as above
Rad Chem Tech/Jr.	Battery Powered Air Sampler & a High Range Beta Gamma Instrument	Perform function of Rad Tech or provide other assistance as required by Shift Supervisor

c. Establish communications with the Control Room and inform the Shift Foreman if outside assistance is not needed.

d. INSURE that all unnecessary personnel are evacuated from the scene of the fire and organize efforts to rescue anyone who may be trapped in an area made hazardous by the fire.

e. Establish a Radiation Particulate Monitor downwind of any radiological fire as soon as possible.

f. Submit a written report containing an analysis of the fire emergency and recommendations for cleanup to the Station/Unit Superintendent.

3.1.3 Shift Foreman.

a. Report to the Control Room.

b. Direct Plant Operations to maintain safe conditions based on information supplied by the Shift Supervisor.

c. Call the Dauphin County Civil Defense Director, Telephone No. 236-7976, or call 911 for fire fighting assistance unless informed by the Shift Supervisor that outside assistance will not be needed.

d. Call in additional personnel as required.

e. Dispatch Aux Operator (if available) to monitor Unit 2 Fire Pump operation if the pump starts.

f. Establish communications with Unit 1 Control Room.

g. Notify Management personnel listed below:

1. Supervisor of Operations.
2. Supervisor of Maintenance.
3. Station/Unit Superintendent.
4. Station Safety Supervisor.

APPENDIX A

FIRE IN THE CONTROL ROOM

1.0 DISCUSSION

Fires in the Control Room will most probably be class A (Normal Combustible Matarials) or class C (Electrical) in nature. An electrical fire is the more probable type and would have more serious consequences. A class A fire would probably be of insignificant size such as trash in a waste basket.

2.0 PROCEDURE

2.1 Follow provisions of the main body of this Emergency Procedure.

2.2 Personnel in the Control Room shall, additionally, accomplish the following:

a. Determine class of fire and its exact location by Panel number, if possible.

b. Fight fire using the following readily available equipment:

1. Electrical fire, use portable Halon extinguishers first, then CO_2. Water in the form of "Fog" shall be used as a last resort.

2. Class "A" fire, use Ansul Dry Chemical Extinguisher.

CAUTION: Excessive use of CO_2 in a closed space may result in asphyxiation.

NOTE: If fire/smoke forces operators to leave the Control Room, shutdown plant according to 2202-1.10, Shutdown from Outside Control Room.

APPENDIX B

Fire Emergency in Cable Room

1.0 SYMPTOMS

a. Control Bldg Cable Room Fire Alarm on Panel 7.

b. Cable Room Temp. Hi Alarm on Panel 25.

c. Visual detection of smoke and/or flame in or near the Cable Spreading Room.

2.0 AUTOMATIC ACTION

a. After a fire signal is received, the Cable Room Halon System is activated.

3.0 MANUAL ACTION

a. A serious fire in the Cable Room as evidenced by symptoms 1a and 1b, or observance of a cable tray fire by a Shift Supervisor/ Foreman requires the following action:

1. Announce on the paging system:
   EVACUATE THE CABLE ROOM. THE HALON SYSTEM HAS BEEN ACTUATED.
2. Manual trip of the unit.
3. If automatic actuation of the Halon System has not occurred, manually activate from outside the Cable Room doors. In the event unit trip is required proceed as follows:
   (1) Trip the reactor by pressing the REACTOR TRIP pushbutton.
   (2) Press Turbine Trip.
   (3) Verify the Generator Breakers and Generator Field Breaker open.
   (4) Proceed with Reactor Trip Procedure.
   (5) Actuate the station fire alarm by pressing and releasing the pushbutton on panel LATER in the Control Room.

APPENDIX B

(6) Announce on the paging system:

FIRE IN THE CABLE ROOL, (type) FIRE BRIGADE REPORT TO CONTROL BUILDING 1ST FLOOR, 305' ELEVATION.

(7) Duties and responsibilities shall be as assigned in 3.0.

b. Either alarm 1a or 1b above or any other symptoms of fire required immediate investigation of the Cable Room, and action as directed by the Shift Supervisor.

APPENDIX C

Fire in the Reactor Building

1.0 SYMPTOMS

a. Reactor Building Fire Alarm on Panel 7.

b. Reactor Building High Temp. Alarm on Panel 25.

c. Visual detection of smoke and/or flame in the Reactor Building.

2.0 AUTOMATIC ACTION

a. Reactor Building purge, if in progress is secured (See Appendix D).

3.0 MANUAL ACTION

a. Announce the following using the page channel:
"FIRE IN THE REACTOR BUILDING, (Class, if known), FIRE BRIGADE REPORT TO THE REACTOR BUILDING PERSONNEL ACCESS HATCH".

b. If the severity of the fire has been established, proceed with step d.

c. If the severity of the fire is unknown, the Rad/Chem Tech and Aux Operator #2 will enter the Reactor Building to determine the severity of the fire under the direction of the Shift Supervisor/Foreman.

CAUTION: In an extreme emergency, only a Scott Air Pack or equivalent need be donned. The Rad/Chem Tech will carry a high range Beta-Gamma Dose Rate Meter when entering the Reactor Building. Even in an extreme emergency, every effort should be made to wear personnel dosimetery.

d. If time permits, as determined by the Shift Supervisor/Foreman, TLD's and pocket dosimeters will be issued to the Fire Brigade by Health Physics personnel. If not, proper Health Physics

APPENDIX C

coverage will be provided to insure that the relative exposures of the fire fighting personnel can be ascertained. Also, if time permits, protective clothing may be donned by the Fire Brigade personnel.

CAUTION: Under all circumstances, personnel entering the Reactor Building during a Fire Emergency will wear a Scott Air Pack or equivalent.

e. As determined by the Shift Supervisor/Foreman, the Fire Bridge will enter the Reactor Building to fight the fire.

f. When individuals leave the Reactor Building, they will be monitored and decontaminated in accordance with approved Health Physics procedures.

APPENDIX D

FIRE PROTECTION ZONE DETECTION SYSTEM

BUILDING	ELEVATION	ZONE	DETECTOR FS-XD	PANEL	FPFP-4 FUSE #	AUXILIARY FUNCTIONS
Auxiliary	258'-6 & 280'	2	6087 thru 6097 6083, 6084	711 Auxiliary Building 305' Elev Near South Stairwell	#1	Stops Fans AH-E-7A/B AH-E-8A, B, C, D AH-E-41
	305'	3	6085 6098-1 thru 6105			
	328'	4A	6080			
		4B	6081, 6082			
		4C	6086			
		4D	6107-1 thru 6110			
Chlorinator		5	6111-1 thru 6113	712 Chlorinator Bldg. Outside Main Entry	#2	Stops Fans AH-E-44, 45 46 & 47, Shuts Dampers D5236A, B, C, & D.
Circ. Water		6	6116-1 thru 6123-2	713 C.W. PH Near Main Entrance	#3	Stops Fans AH-E-22A, B, C, D, E, F, AH-E-29, Shuts Dampers D5081A, B,C,D,E,F & D5080A,B.
Coagulator		7	6130 thru 6131-6	714 Coag. Bldg. Near Entry to Turbine Bldg.	#4	Stops Fans AH-E-17A, B, AH-E-20
Control	280'	8A	6135, 6142-1 thru 6142-4	715 Control Bldg 331' Elev Outside Control Room	#5	Stops Fans AH-E-17A, B, AH-E-20
		8B	6136, 6143-1 thru 6143-4			
	Mezzanine	8C	6137, 6144-1 thru 6144-9			
	280'	8D	6138-1 thru 6138-4			
		8E	6139-1 thru 6139-4			
	351'	8F	6141			Stops Fans AH-E-17A/B,2
	305'	9	6140-1, 2 6147-2,4,6,8 10,12, TS-6147			Stops Fans AH-E-17A/B AH-E-20, Actuates Halon Fire Suppression System Only if One Detector From Both 9A & 9 are Actuated
		9A	6147-1,3,5,7 9,11,13			

APPENDIX D

FIRE PROTECTION ZONE DETECTION SYSTEM

BUILDING	ELEVATION	ZONE	DETECTOR FS-XD	PANEL	FPFP-4 FUSE #	AUXILIARY FUNCTIONS
Control (cont'd)	331'	10	6148-1 thru 6148-16, 6154 6161, TS-6153	715	#5	Stops Fans AH-C-16A/B AH-E-35, AH-E-4A/B AH-E-5, AH-E-25 Shuts Damper 04092C
	351'	11A	6149, 6150			
		11B	6155, thru 6160			Stops Fans AH-C-19A/B AH-E-6, AH-E-40
Control Bldg. A.	280'	12A	6180 thru 6182-9	716 Control Bldg Area Near Entrance to Control Building	#6	Stops Fans AH-C-50A/B AH-C-51
		12B	6183-1 thru 6183-12			Stops Fans AH-E-13 AH-58A thru J, AH-E-62
Diesel Generator	280'6"	13A	6210-1 thru 6210-10	717 Diesel Generator Bldg. 280' in N.E. Corner of East Side	#7	Stops AH-E-24A
		13B	6212			Stops AH-E-24A DF-P-1A/B, Closes DF-V2A
		13C	6217-1 thru 6217-8			AH-E-24B
		13D	6219			Stops AH-E-24B DF-P-1C/D Closes DF-V2B
	305'	14A	6213-1 thru 6213-9, 6215 FD-6214			Stops AH-E-24A DF-P-1A/B Closes DF-V2A
		14B	6220-1 thru 6220-8, 6222 FD-6221			Stops AH-E-24B DF-P-1C/D Closes DF-V2B
Fire Pump	312'	15	6230, 6231 FD-6232	718	#8	Stops AH-E-32
Fuel Handling	280'	16	6240-1 thru 6240-12, 6249	719 Fuel Hand. Building 305' Elev. Inside Main Entrance	#9	Stops AH-E-9A/B AH-E-10A/B/C/D.
	305'	17	6241-1 thru 6241-4			
	328'	18A	6242			
		18B	6243, 6244 6245			
	347'	19	6251-1 thru 6251-12			

APPENDIX D

FIRE PROTECTION ZONE DETECTION SYSTEM

BUILDING	ELEVATION	ZONE	DETECTOR FS-XD	PANEL	FPFP-4 FUSE #	AUXILIARY FUNCTIONS
MDCT		20	6551, 6552	NA	#13	Stops AH-E-60
Reactor	328'	22A	6270, 6271	720 Service Building 305' Elev. Near Entrance to Hot Inst. Decon Room	#10	Stops Fans AH-E-12A/B 19A/B Closes AH-V1A/B 4A/B
		22B	6272			Stops AH-E12A/B, AH-E-19A/B, AH-E-34 Closes AH-V1A/B AH-V4A/B
		22C	6273			
	305' & 347'	22D	6275-1 thru 6276-5, 6279-1 thru 6279-3			
	358'	22E	6277			Stops AH-E-63
	282'	22F	6278			Stops AH-E-12A/B AH-E-19A/B, AH-E-34 Closes AH-V1A/B, AH-V4A/B
RWPH	280'-312'	23	6290 thru 6291-9, 6294 thru 6297 6299-1 thru 6300-2	718 RWPH Near Main Entrance	#8	Stops AH-C-20A/B AH-E-13
Service	280'	24	6342-1 thru 6344-4	721 Service Building 305' Elev. Personnel Monitoring Area	#11	Stops AH-C-50A/B, AH-C-51
	305'	25	6319 thru 6339-2, 6348-1 thru 6348-4			Stops AH-E-23A/B AH-C-24, AH-C-25A/B, AH-E-25, AH-E-27, AH-E-28, AH-E-33 AH-C-36
	305', 322' & 333'	26A	6310			
		26B	6311, 6315-1 thru 6318, 6340-1 thru 6341-5			
		26C	6312			
		26D	6313			
		26E	6314			
Turbine	281'	27A	6360	722 Turbine Building 331' Elev. Near Entrance to Control Building	#12	Stops AH-E-1, 2, 3 AH-E-39, AH-E-61
		27B	6364, 6373-1 thru 6373-25			
	305'	28A	6362			
		28B	6363			
		30	6366-1 thru 6369-2			
	331'	29	6365, 6370 thru 6372-16			Stops AH-E-1, 2, AH-E-38

APPENDIX D
FIRE PROTECTION SYSTEMS

SYSTEM	MULTIMATIC VALVE	ROOT ISOLATION VALVE	CONTROL PANEL	POWER SUPPLIES	AUXILIARY FUNCTIONS
Main Transformer 2A	FS-V-403A	FS-V403B	701 Turb. Bldg. 305' Elevation East Wall	DC Dist. Pnl DCT-2A Bkr 20	None
Main Transformer 2B	FS-V402A	FS-V402B			
Auxiliary Transformer 2A	FS-V-401A	FS-V401B			
Auxiliary Transformer 2B	FS-V400A	FS-V400B			
Tubine Building East Wall	FS-V404A	FS-V404B			
Hydrogen Seal Oil Unit	FS-V405A	FS-V405B			
Main Turb. Oil Reservoir	FS-V408A	FS-V408B	702 Turb. Bldg. 305' Elevation Near Heater Drain Tank	DC Dist. Pnl DCT-2A Bkr 4	Stops Fans All-E-1, 2, 3, All-E-39 & All-E-61
S.G.F.P. Turb. Oil Res'v 1A	FS-V409A	FS-V409B			
S.G.F.P. Turb. Oil Res'v 1B	FS-V410A	FS-V410B			
Emergency Diesel Gen. Fuel Oil Tank	FS-V-425A	FS-V425B	704 Diesel Gen. Bldg. 280' Elevation	DC Dist. Pnl DCA-2A Bkr 19	Stops Fans All-E-24A, 24B Stops Pumps DP-P-1A,B,C,D Shuts Valves DP-V2A, 2B
Emergency Diesel Gen. Bldg. Air Intake	FS-V-426A	FS-V426B			
Plant Air Intake Tunnel Vert. Shaft	FS-V-429A	FS-V429B	705/705A Service Bldg. 280' Elevation	705 DC Dist. Pnl DCC-1B Bkr 9 705A -	Control Rm - shuts Dampers D-4092C, D-4092D, ED-4098 Opens Damper D-4092E Control Bldg. Mech. Room Stops Fan All-E-40 Shuts Damper D-4086 Control Bldg. Cable Room Shuts Dampers D-4073, ID-4075A, ED-4075, Open Damper ID-4075B

APPENDIX D
FIRE PROTECTION SYSTEMS

SYSTEM	MULTIMATIC VALVE	ROOT ISOLATION VALVE	CONTROL PANEL	POWER SUPPLIES	AUXILIARY FUNCTIONS
R.B. Purge Exhaust Charcoal Filter Unit AH-F-31B	FS-V422A	FS-V422B	708	DC Dist. Pnl DCA-2A	Stops Fans AH-E-12E AH-E-19B
F.H. Bldg. Exhaust Charcoal Filter Unit AH-F-14A	FS-V421A	FS-V421A	707 Auxiliary Bldg 328' Elevation Near Chem. Add. Control Panel	DC Dist Pnl DCA-2A Bkr 21	Stops Fans AH-E-9A, B AH-E-10A, B, C, D.
F.H. Bldg. Exhaust Charcoal Filter Unit AH-F-14B	FS-V420A	FS-V420B			
Aux. Bldg. Exhaust Charcoal Filter Unit AH-F-10A	FS-V419A	FS-V419B			Stops Fans AH-E-7A, B AH-E-8A, B, C, D.
Aux. Bldg. Exhaust Charcoal Filter Unit AH-F-10B	FS-V418A	FS-V418B			
Control Room Bypass Charcoal Filter Unit AH-F-5	FS-V413	FS-V507	709 Control Bldg. 351' Elevation North Wall	DC Dist. Pnl DCC-1B Bkr 8	Stops Fans AH-E-4A, B
Radwaste Gas Charcoal Filter Unit WDG-F-1	FS-V433A	FS-V433B	710 Auxiliary Bldg. 305' Elevation in WDG-F-1 Cubicle	DC Dist. Pnl DCA-2A Bkr 20	None.

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SYSTEM	MULTIMATIC VALVE	ROOT ISOLATION VALVE	CONTROL PANEL	POWER SUPPLIES	AUXILIARY FUNCTIONS
(Cont'd)					Control Bldg. Area - Stops Fans AH-E-13, AH-C-51 Shuts Damper D-5560 Aux. Bldg. - Stops Fans AH-E-23A, B, AH-E-8A, B, C, D Shuts Damps D-4002 Service Bldg. - Stops Fans AH-E-23A, B, AH-C-24, AH-E-26, AH-E-27, AH-E-28, AH-E-33, AH-C-36, Shuts Dampers D-5739, D-5742, SD-5739. Fuel Handling Bldg. - Stops Fans AH-E-9A, B, AH-E-10A, B, C, D. Reactor Bldg. - Stops Fans AH-E-12A, B, AH-E-19A, B. Shuts Dampers D-5128A, D-5128B. Air Intake Tunnel - Shuts Fire Dampers AH-E-2A, B, C, D, E, F.
Plant Air Intake Tunnel North of Structural Div.	FS-V430A	FS-V430B			
Plant Air Intake Tunner South of Structural Div.	FS-V431A	FS-V431B	706/705A	706-DC Dist Pnl DCC-1B Bkr 5	
Hydrogen Purge Exhaust Charcoal Filter Unit AH-F-34	FS-V424A	FS-V424B	708 Auxiliary Bldg. 328' Elevation Near South Stairway	DC Dist. Pnl DCA-2A Bkr 22	Stops Fan AH-E-34
R.B. Purge Exhaust Charcoal Filter Unit AH-F-31A	FS-V423A	FS-V423B			Stops Fans AH-E-12A, 19A

SYSTEM	MULTIMATIC VALVE	ROOT ISOLATION VALVE	CONTROL PANEL	POWER SUPPLIES	AUXILIARY FUNCTIONS
Main Turbine Bearings	FS-V412A	FS-V412B	703	DC Dist. Pnl DCT-2A Bkr 8	Stops AH-E-1, 2, 3 AH-E-39, AH-E-61
S.G.F.P. Turbine Bearings	FS-V411A	FS-V411B			
East Turb Bldg. Basement and Mezzanine	FS-V407A	FS-V407B	N/A	N/A	None
West Turb Bldg. Basement and Mezzanine	FS-V406A	FS-V406B			
Fire Pump Room	FS-V434A	FS-V434B			
East Emergency Diesel Generator Room	FS-V427A	FS-V427B			
West Emergency Diesel Generator Room	FS-V428A	FS-V428B			
N.D. Cooling Tower A	FS-V520 FS-V521 FS-V522 FS-V523 FS-V524 FS-V525	FS-V155 FS-V156 FS-V157 FS-V158 FS-V159 FS-V160		DCT-2A (Subfuse Pnl. FPFP-8)	
N.D. Cooling Tower B	FS-V526 FS-V527 FS-V528 FS-V529 FS-V530 FS-V531	FS-V161 FS-V162 FS-V163 FS-V164 FS-V165 FS-V166		DCT-1A (Subfuse Pnl. FPPP-7)	
MDCT	FS-V544A	FS-V544B		DCT-1A (Subfuse Pnl. PPPP-6)	Stops Pan 2-1
	FS-V545A	FS-V545B			Stops Pan 2-2
	FS-V546A	FS-V546B			Stops Pan 2-3

Appendix E

Oil Fire in the Turbine Building

1.0 DISCUSSION

Oil fires can be serious if not rapidly contained. For equipment protection, however, oil systems usually cannot be immediately shutdown upon the detection of a fire. Immediate action should be taken to contain the fire, and then as rapidly as possible proceed to plant conditions which allow the total shutdown of the oil system. This may include breaking vacuum, and/or dumping generator hydrogen gas to vent.

2.0 PROCEDURE

2.1 Main turbine generator oil fire.

2.1.1 Follow the provisions in the main body of this procedure.

2.1.2 Attempt to extinguish fire with on-hand equipment.

CAUTION: Do not trip the oil pumps unless absolutely necessary or bearing and seal damage may result.

2.1.3 If the fire is detected in the main lube oil reservoir, guard piping, or inside a bearing pedestal, the fire may be suffocated by stopping the vapor extractor LO-E-1 from panel 17.

CAUTION: With the vapor extractor stopped, oil may leak from bearing pedestal cover seals and present an additional fire hazard. Do not operate longer than necessary with the vapor extractor out of service.

2.1.4 If step 2.1.3 did not extinguish the fire, or if the fire is not in an enclosed space, attempt to contain the fire and proceed to shutdown the lube oil system as follows:

a. Trip the turbine and insure generator breakers open.

NOTE: Insure the TGP, SOP, and BL pumps are in AUTO and start to supply lube oil to the bearings.

b. Break vacuum by opening VA-V7A&B from panel 5.

c. Isolate H_2 from Unit 1 by closing HY-V24.

d. Dump H_2 to the atmosphere by opening generator gas valves 7 and 24.

e. Isolate LO system from the seal oil system by closing seal oil valve 265.

2.1.5 When the turbine is at zero speed, the lube oil system may be shutdown by placing the oil pumps panel 5 control switches in pull to lock.

CAUTION: When all oil pumps are stopped, the lube oil reservoir level will increase.

2.1.6 Extinguish fire.

2.2 Main turbine generator seal oil fire.

CAUTION: Do not trip all seal oil and lube oil pumps unless absolutely necessary before the turbine is at rest, and all H_2 vented. Stopping pumps early may result in Hydrogen leakage and an explosion hazard around the generator.

2.2.1 Follow the provisions in the main body of this procedure.

2.2.2 Attempt to extinguish the fire with on-hand equipment.

2.2.3 If the fire cannot be extinguished without securing the seal oil system, attempt to contain the fire, and proceed as follows:

a. Trip the turbine and insure generator breakers open.

b. Break vacuum by opening VA-V7A&B from panel 5.

c. Isolate H_2 from Unit 1 by closing HY-V24.

d. Dump Hydrogen to the atmosphere by opening generator gas valves 7 and 24.

e. Start the DC air side seal oil backup pump, and trip the AC air and hydrogen side seal oil pumps.

f. When the turbine speed has decreased to 500 RPM, inject CO_2 to dilute the H_2 by opening generator gas valves 5 and 60, and supplying CO_2 from Unit 1.

g. When the machine gas pressure decreases to 10 psig, the DC Seal Oil backup pump may be tripped.

h. If required to extinguish the fire, the lube oil system may be isolated from the seal oil system by closing seal oil valve 265.

i. If the turbine is at zero speed, the lube oil pumps may be tripped.

2.2.4 Extinguish the fire.

2.3 Feedpump Turbine Oil Fire.

CAUTION: Do not trip all oil pumps before the turbine is at rest, or damage to the bearings may occur.

2.3.1 Follow the provisions in the main body of this procedure.

2.3.2 Attempt to extinguish the fire with on-hand equipment.

2.3.3 If the fire cannot be extinguished, attempt to contain the fire and proceed as follows:

a. Trip feedpump turbine.

b. Start the DC Emergency oil pump, and trip the Main and Aux oil pumps.

c. Close the feed pump's inlet, outlet, and exhaust valves (CO-V 52 A/B, FW-V8 A/B, and MS-V44/45).

2.3.3 When the turbine is at zero speed, trip the DC Emergency oil pump.

2.3.4 Extinguish the fire.

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-3.2
FLOOD

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23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval NA (Cognizant Dept. Head) Date —

Unit 2 Staff Recommends Approval

Approval NA (Cognizant Dept. Head) Date —

Unit 1 PORC Recommends Approval

NA (Chairman of PORC) Date —

Unit 2 PORC Recommends Approval

J. F. Hilbish (Chairman of PORC) Date 1-5-78

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

J. L. Seelinger Date 1/5/78

Manager Generation Quality Assurance Approval NA Date —

TMI 55 A Rev 8/77

THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-3.2

FLOOD

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-3.2

FLOOD

1.0 REFERENCES

1.1 FSAR; Volume 1, Chapter 2, Site Charasteristics, Section 2.4.2 Floods.

1.2 Federal - State River Forecast Center
National Weather Service
228 Walnut Street
Box 1185
Harrisburg, Pa. 17108
PHONE: 717-782-3901
717-782-3488
717-782-2256 (unlisted)

If phone communications are out the Forecast Center may be reached through the Met-Ed Dispatcher and the Civil Defense system by radio.

2.0 BASIS FOR IMPLEMENTING THE FLOOD PROTECTION PROCEDURE

2.1 The Federal - State River Forecast Center will call the T.M.I. Station when a 36 hour river flow forecast at Harrisburg indicates 350,000 cfs.

2.2 Following this initial notification, or if it is apparent that the center might have failed to notify the station while a possible major flood may be forming, the River Forecast Center should be contacted to periodically obtain the current forecasts.

2.3 The following forecasts of river flow at Harrisburg form the basis for procedural action to be taken during a flood.

2.3.1 A 36- hour forecast of 350,000 cfs or greatar will initiate Section 3 of this procedure.

2.3.2 If a 36-hour forecast of 640,000 cfs or greater is received, a flood ALERT will be initiated by the Station/Unit Superintendent.

2.3.3 If a 36-hour forecast of 940,000 cfs. or greater is received, an EMERGENCY CLOSURE will be called by the Station/Unit Superintendent.

2.3.4 If the stage of the River Water Intake Structure of Unit 1 is 301 ft. corresponding to 950,000 cfs river flow the Station/Unit Superintendent will order a SHUT-DOWN ALERT

2.3.5 If the river stage reaches elevation 302 ft. corresponding to 1,000,000 cfs river flow, a SHUTDOWN order will be given by the Station/Unit Superintendent.

3.0 FLOOD PROTECTION PROCEDURE

3.1 Before the river flow reaches 350,000 cfs, the flap at the southesat drainage culvert should be checked for freedom of operation. Also, the inside stop gate should be readied for closing in case the flap gate does not hold tight when the river level on the outside of the southeast dike is higher than the level on the inside.

3.2 Initiate patrol and inspection of the dikes surrounding the site for signs of deterioration such as undermining or excessive seapage. In the event the inspection of the dikes indicates significant dike deterioration and the river flow is greater than 340,000 cfs a shutdown order will be given. Significant deterioration of dike is deemed to mean the design function of the dike has been or may be compromised.

3.3 When a flood ALERT is called, personnel should be called out to perform the following:

3.3.1 Start moving flood panels to placement location, clearing the imbedded bolt holes for the flood panels, and seal the following hatch cover:

Hatch Designation	Location (See Appendix I) Description	No.	Checked
(H-1)	Air Intake Structure	1	_____

3.3.2 Check floor drains and pumps listed in Appendix II to ensure proper operation.

3.3.3 The area in vicinity of the North Bridge will be inspected at least daily for buildup of debris. Any significant buildup of debris will be removed as soon as possible. As conditions deteriorate, the frequency of inspection will be increased accordingly.

3.3.4 Plastic sheeting and sand bags marked for flood use only will be available at the warehouse to help minimize in leakage to buildings if necessary.

3.3.5 Soundings of the intake screen house floor will be accomplished at least daily to detect sediment accumulation. Soundings will be taken in accordance with Appendix III.

3.4 When an EMERGENCY CLOSURE is announced, the following should be accomplished within a time period of 36 hours.

3.4.1 Call out all available personnel to place flood panels and for plant operations.

3.4.2 Provide for emergency food supplies.

3.4.3 Notify Manager - Generation Operations of the situation.

3.4.4 Install the following flood panels.

Gate Designation	Location (See Appendix I) Description	No.	Checked
(D-1, D-2)	North entrance to Diesel Building at 305 ft. elevation	2	_____ _____
(D-3)	South entrance to Diesel Generator Building at 305 ft. elevation	1	_____
(A-1,A-2)	West entrance at 305 ft elevation to access hatch to tendon Gallery	2	_____ _____
(C-1)	West entrance to the Control Building at 305 ft elevation	1	_____
(C-2)	East entrance to the control Building at 305 ft elevation	1	_____
(C-3)	South entrance to Control Building at 305 ft elevation	1	_____

3.4.5 Install door seal at the following Location:

Seal Designation	Location (See Appendix XI) Description	No.	Checked
S-1	Auxiliary Building shield door northeast corner at 305 ft elevation	1	_____

3.4.6 Verify closure of water tight doors at the following location:

Door Designation	Location (See Appendix I) Description	No.	Checked
(W-1, W-2, W-3)	Bulkhead doors on north wall of Turbine Building at 281 ft elevation.	3	_____ _____ _____

3.4.7 Increase makeup water to all storage tanks as much as possible and fill all outdoor tanks to at least elevation 312 to help prevent floatation in case of site flooding, i.e. - all tanks greater 7' level (305' grade).

3.4.8 Check and fill fuel storage tanks.

3.4.9 Arrange for an alternate supply of diesel fuel oil as follows:

a. Attach a hose from the top of the Fuel Handling to the Unit 2 fuel oil fill connection.

b. Procure 4 or more tanks of approximately 275 gallon capacity and capable of being slung under a helicopter.

c. Contact Keystone Helicopter, 215-644-4430, or Carson Helicopter, 215-249-3535. Either can supply a helicopter capable of lifting 3,500 lbs. Carson also has helicopter capable of lifting loads up to 8,000 lbs.

d. Arrange for a local, non-flooded fuel oil supply.

e. Arrange for the helicopter to set tanks down on the fuel handling building roof.

f. Remove the air ventilation grills, but not the fire dampers, in the air ducts between the Storage tank room and the Unit II diesel Room. This will allow entry into the Diesel Storage Tank Room during flooded conditions.

3.5 If a SHUTDOWN ALERT is called, preparations should be made to shut down the plant as required and notify executive level personnel of the situation.

3.6 If a SHUTDOWN is ordered, the following should be done:

3.6.1 The Station should be placed in HOT SHUTDOWN within the time frame specified in technical specifications section 3.7.6.1 and be in cold shutdown conditions as soon as possible in accordance with 2102-3.1. (Unit Shutdown), and 2102-3.2 (Unit cooldown).

3.6.2 Shutdown unnecessary systems as Unit Conditions will allow.

3.6.3 De-energize any non-essential electrical busses subject to flooding below 311' per 2107-1.1, BOP Electrical System.

NOTE: The above steps will place most all of the secondary plant systems which are non-essential in a shutdown condition. Pumps, motors, etc, can be moved or lifted to higher positions to preclude water damage as time permits.

The plant will be in a cold shutdown with all required nuclear safety related systems operable and protected from inundation.

4.0 POST FLOOD PROCEDURE

4.1 As the river recedes, provided that there is no precipitation forecast or the river flow is less than 640,000 cfs, a detailed evaluation of possible plant damage will be made. The Station/Unit Superintendent will make an evaluation as to the recall of personnel, and the method of access to the site (helicopter, boat, car). The Station/Unit Superintendent will make the determination on continued plant operation or return to power if the shutdown order was given, based on findings of detailed evaluation of possible plant damage, review of sounding readings, and Tech. Specs. requirements.

4.2 If the stop gate has been closed during the river rise, it should be opened as the river level on the outside of the southeast culvert opening goes below the level of the water inside the dike.

APPENDIX II

SUMP AND FLOOR DRAIN CHECK LIST

Auxiliary Building, 258 ft. elevation	Initials
Floor Drain Locations	
1. South RB Spray Pump Room	_____
2. North RB Spray Pump Room	_____
3. South D.H. Pump Room	_____
4. North D.H. Pump Room	_____
Sumps South RB Spray Pump Room	_____
North RB Spray Pump Room	_____
South DH Pump Room	_____
North DH Pump Room	_____
Sump Pumps WDL-P-17B	_____
WDL-P-17A	_____
WDL-P-16B	_____
WDL-P-16A	_____

Auxiliary Building, 280 ft. elevation	Initials
Floor Drain Locations	
5. Northeast - Bleed Holdup Tank Rooms	
a. Near WDL-T-1A (4)	_____
	_____
	_____
	_____
b. Near WDL-T-1B (2)	_____
	_____
c. Near WDL-T-1C (2)	_____
	_____

Auxiliary Building, 280 ft. elevation	Initials
6. Waste disposal liquid valve room	
7. Southeast, East of Reactor Building emergency cooling booster pump RR-P-1A	
8. West of Reactor Building emergency cooling booster pump RR-C-1B	
9. Waste Transfer Pump Room	
10. Clean-up demineralizer Room (2)	
11. Hallway outside clean-up demineralizer Room	
12. Reactor Coolant Evaporator Room	
13. Hallway outside Evaporator Room	
14. Evaporator Condensate Tank Room (2)	
15. Outside Evaporator Condensate Tank Room	
16. South of the Seal Water Pump Unit OW-P-2	
17. Outside Auxiliary Building Sump Filter Room	
18. Inside Auxiliary Building Sump Filter Room	
19. North-South Main Corriodr (3)	
20. Spent Resin Storage Tank Room (2)	

Auxiliary Building, 280 ft elevation	Initials
21. Makeup and Purification Pump Room MU-P-1A	
22. Makeup and purification Pump Room MU-P-1B	
23. Makeup and Purification Pump Room MU-P-1C	
24. Southwest corner near instrument rack #459	
Sump - Sump Tank Room, North	
Sump Pumps WDL-P-3B	
WDL-D-3A	

Auxiliary Building, 305 ft elevation	Initials
25. Northwest, Concentrated Liquid Waste Pump Room	
Northwest of elevator, around covered hatch (5)	
26. Along North Wall (2)	
27. Near Motor Control Center 2-21EB	
28. Near Motor Control Center 2-11EB	
29. At entrance to miscellaneous waste hold-up	
Tank Room and near miscellaneous waste tank pumps (4)	

Auxiliary Building, 305 ft. elevation	Initials
30. Miscellaneous Waste Hold-Up Tank Room	
31. Near Motor Control Center 2-11 EA	
32. Near Motor Control Center 2-11E	
33. Near Motor Control Center 2-21EA	
34. Near Motor Control Center 2-21E	
35. Near Nuclear Service Closed Cooling Water Pumps (2)	
36. East of Intermediate Coolers	
37. Between Intermediate Cooling Filters	
38. Near Seal Return Coolers	
39. Near Makeup and purification Demineralizer MU-K-13	
40. Near Makeup and purification demineralizer MU-K-1A	

Auxiliary Building, 305 ft. elevation	Initials
30. Miscellaneous Waste Hold-Up Tank Room	
31. Near Motor Control Center 2-11 EA	
32. Near Motor Control Center 2-11 E	
33. Near Motor Control Center 2-21 EA	
34. Near Motor Control Center 2-21 E	
35. Near Nuclear Service Closed Cooling Water Pumps (2)	
36. East of Intermediate Coolers	
37. Between Intermediate Cooling Filters	
38. Near Seal Return Coolers	
39. Near Makeup and purification demineralizer MU-K-113	
40. Near Makeup and purification demineralizer MU-K-1A	
41. Near Gas analyzer WDG-G-1 (2)	
42. Near Make-up filters (2)	
43. Near Make-up tank	
Near Spent Fuel Coolers (2)	
44. In Spent Fuel Demineralizer Room	
45. In Spent Fuel Filter Room	

Auxiliary Building, 305 Ft Elevation — Initials

46. Valve Room, West of the Miscellaneous Waste Hold-up Tank _____
47. Corridor Outside Spent fuel Demineralizer Room _____
48. In Deborating Demineralizer, WDL-K-1A Room _____
49. In Deborating Demineralizer, WDL-K-1B Room _____
50. In Valve Room South of the deborating demineralizers _____
51. In Waste Gas Compressor Room (North) _____
52. In Valve Room Near Waste Gas Compressor (North) _____
53. In Waste Gas Filter Room _____
54. In Waste Gas Compressor Room (South) _____
55. In Valve Room Near Waste Gas Compressor Room (South) _____
56. In Corridor Outside Drain 55 _____

Service Building 280 Ft Elevation _____

Floor Drains.

1. Near Service Air Receivers (2) _____ _____
2. West of instrument Air Compressor IA-P-1A _____
3. Near Service Air Receivers _____
4. West of Service Air Compressors _____
5. West of Stairwell _____
6. West of Sump _____
7. North of Service Building River Water Booster Pumps _____
8. Near Isolation Valve Water Storage Tank _____
9. In Contaminated Drain Tank Room _____

Sump-South Wall _____

Sump Pumps SD-P-9B _____

SD-P-9A _____

Control Building, 280 Ft. Elevation	Initials
Floor Drains	
1. Battery Room No. 2	_____
2. Battery Room No. 1	_____
3. DC Switchgear Room No. 2	_____
4. DC Switchgear Room No. 1	_____
5. Cable Tray Area	_____
Control Building Area 282 Ft. Elevation	**Initials**
6. Northwest Corner, West of Stariway	_____
7. In access area above tendon access	_____
Gallery, North of West Stairway (2)	_____
8. Northeast of Unit Substation 2-44	_____
9. Southeast of Unit Substation 2-44	_____
10. Southwest Corner, near AH-C-58F	_____
11. East of Drain 10.	_____
12. West of Instrument Rack #436	_____
13. East of Control Building Sump	_____
14. East of Drain 13, near AH-E-48B	_____
15. East of Drain 9, Near AH-E-48A	_____
Sump -South Wall	_____
Sump Pump SD-P-3A	_____
SD-P-3B	_____

Diesel Generator Building, 280 Ft. Elevation	Initials
Floor Drains	
1. East Room (6)	
2. West Room (6)	
Sumps - East Room	
West Room	
Sump Pumps SD-P-10A	
SD-P-10B	
SD-P-10C	
SD-P-10D	
Diesel Generator Building, 305 Ft. Elevation	**Initials**
3. Diesel Fuel Storage Tank Room (2)	
4. East-West Corridor on North Side (4)	

Diesel Generator Building, 305 Ft. Elevation	Initials
5. Diesel Generator 1B Room (6)	_____
	_____
	_____
	_____
	_____
	_____
6. Diesel Generator 1A Room (6)	_____
	_____
	_____
	_____
	_____
	_____
	_____

Tendon Access Gallery	Initials
Floor Drain Locations	
1. 8 Floor Drains	_____
Sump	_____
Sump Pumps SD-P-13A	_____
SD-P-13B	_____

Air Intake Tunnel	Initials
Floor Drain Locations	
None	
Sump	_____
Sump Pumps SD-P-11A	_____
SD-P-11B	_____
SD-P-4A	_____
SD-P-4B	_____

Fuel Handling Building, 280 Ft. Level	Initials
Floor Drains	
1. South, immediately adjacent the Reactor Building Wall (2)	
2. East Valve Room (4)	
3. East Access Corridor (2)	
4. West Valve Room (2)	
4. West Valve Room (2)	
5. Southeast of Chemical Feed Tank RB-T-2	
6. Between the Decay Heat Service Coolers (3)	
7. Outside Nutralizer Tank Room	
8. Inside Nutralizer Tank Room	
9. Outside Reclaimed Boric Acid Pump Room	
10. Inside Reclaimed Boric Acid Pump Room	
Fuel Handling Building, 305 Ft. Level	**Initials**
11. Makeup and Purification Valve Room (2)	

Fuel Handling Building, 305 Ft. Level	Initials
12. Along East Wall (4)	_____
	_____
	_____
	_____
13. Waste Storage Area (4)	_____
	_____
	_____
	_____

Reactor Building, 282 Ft. Elevation	Initials
Floor Drain Locations	
1. Inside O Ring (4)	_____
	_____
	_____
	_____
2. South Side at Re-Drain Tank WD-L-T3	_____
3. Southwest, Near Leakage Transfer Pumps	_____
4. East at Drain 3	_____
5. East, Near Elevator	_____
6. East of Steam Generator West Lay-up Recirc. Pump	_____
7. North, Near Letdown Coolers	_____
8. South at Drain 8.	_____
Sump - North End	

Sump Pumps - WDL-P-2A

WDL-P-2B

Reactor Building, 305 Ft. Elevation	Initials
9. In front of elevator	
10. North of Elevator	
11. Behind Elevator	
12. Southwest of Elevator	
13. South behind Reactor Building Coolers	
14. In front of Reactor Building Coolers, below Core Flooding Tanks	
15. West of 14, Southeast of the Equipment Hatch	

Reactor Building, 305 Ft. Elevation	Initials
16. Southeast of West Stairway (2)	
17. North of West Stariway, South of Instrument Mounting R10	
18. South of Instrument Mounting R10	
19. Northeast Near Instrument Mounting R11	
20. South of Drain 19	

APPENDIX III

Check of River Water Intake Structure for Sediment Accumulation.

1. Measurements to be taken at openings in front of each bar rake assembly.
2. Measuring device to consist of a Rope measuring 50 feet or longer with a suitable weight, such as a flat weight, attached to the end.
3. Procedure
   a. Standing on an operating level, lower the weight to the floor of the pump house.
   b. When the weight is resting on the botton, take a depth reading using the operating floor as the reference point. Ensure that the rope is taut and straight and that the weight is resting on the bottom.

      NOTE: The sediment may be soft so care should be used to note the point at which the weight first detects resistance.
4. Action Required.
   a. No accumulation (47' on tape) - Clean design condition - no action required.
   b. 0' - 4' accumulation (47' - 43' on tape) - Base line sediment acculumation no action required.
   c. 4' - 6' accumulation (43' - 41' on tape) - Increase surveillance to every 2 hours. Pumps should be observed for proper operation.
   d. 6' or greater accumulation (41' or less on tape) - Inform station Superintendent - Plant should be shut down and cooled down.

APPENDIX IV

A Gauge Reading Market Street Bridge, Harrisburg (Feet)	B River Elevation at TMI (Feet Above Sea Level)	C River Flow (Cubic Feet) per second)	D River Velocity (MPH)
4.3	278.7	20,000	.9
5.3	279.5	40,000	1.4
6.2	280.1	60,000	1.7
7.1	280.7	80,000	2.0
8.1	281.3	100,000	2.3
10.4	282.5	150,000	2.6
12.5	283.6	200,000	3.1
14.3	284.9	250,000	3.3
16.1	285.8	300,000	3.5
17.9	287.0	350,000	3.7
19.5	288.1	400,000	3.9
21.2	289.7	450,000	4.1
22.7	291.0	500,000	4.3
24.3	292.6	550,000	4.5
25.6	294.0	600,000	4.7
26.9	295.2	650,000	4.9
28.1	296.1	700,000	5.1
29.3	297.1	750,000	5.3
30.4	298.1	800,000	5.5
31.3	299.1	850,000	5.7
32.0	300.1	900,000	5.9
32.6	301.1	950,000	6.1
33.1	302.0	1,000,000	6.3

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2202-3.3
Revision 2
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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2202-3.3
EARTHQUAKE

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24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

N/A Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

R P Warren Date 9/11/78
V- Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

J. F. Colaccino Date 9/12/78

Manager Generation Quality Assurance Approval N/A Date —

TMI 55-A Rev 3/77

THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202.3.3

EARTHQUAKE

1.0 SYMPTOMS

1.1 "Seismic Trigger On" Alarm (8.E13) in the Control Room will illuminate when ground acceleration exceeds 0.025g horizontal or 0.025g vertical. The "Seismic Recorder On" Alarm (8.E12) will illuminate when a ground acceleration of 0.01g has been sensed.

1.2 The event Indicator on the front panel of the Strong Motion recording system in the Cable Room changes from black to white and the Yellow Event Alarm indicator lamp illuminates when the Strong Motion recording system has sensed a ground acceleration of 0.01g. The Yellow Event Alarm Indicator lamp will extinguish approximately ten seconds after the last acceleration above 0.01g.

1.3 Normal plant process instrumentation will indicate the effect of an earthquake on operating systems in the Unit.

1.4 Ground motions from the earthquake may be felt by personnel at the nuclear station.

2.0 IMMEDIATE ACTION

A. Automatic Action:

1. The strong motion recording system will record ground acceleration greater than 0.01g on a magnetic tape cassette. It will also record the response to this acceleration at the ring girder of the Reactor Building, on another magnetic tape cassette.

2. Abnormal conditions or faults resulting from seismic stress may, depending on severity, automatically trip components, or systems within the Unit and may initiate an automatic reactor trip.

NOTE: Structures, components, and systems required to permit safe shutdown of the Unit are designed to withstand the stresses which will result from the maximum hypothetical earthquake.

B. Manual Action.

1. After the strong motion recording system indicates that is has recorded a ground acceleration which has not exceeded:

0.025g horizontal acceleration and/or 0.025g vertical acceleration, evaluate the data contained on the strong motion recording system magnetic tape cassettes as soon as it is convenient to do so, and allow normal Unit operation to continue.

2. When ground acceleration of 0.025g horizontal and/or 0.025g vertical is exceeded, make an immediate analysis of the data contained on the strong motion recording system's magnetic tape cassette. (Refer to the Earthquake Emergency Plan in Site Emergency Plan Administrative Procedure AP 1004).

3. When data evaluation shows that an earthquake has occurred, but the peak amplitude of the earthquake does not exceed 0.06g horizontal and/or 0.04g vertical acceleration maintain Unit operation. Take whatever steps are necessary

to respond to the faults occurring in components or systems within the plant as a result of the earthquake.

4. When data evaluation shows that an earthquake has occurred, and the peak amplitude of the earthquake has exceeded 0.06g horizontal and/or 0.04g vertical acceleration take normal actions to initiate an orderly and controlled poer reduction to the hot Standby (Mode III) condition. Also take whatever steps are necessry to respond to the faults occurring in components or systems within the Unit as a result of the earthquake.

   NOTE: If steam generator level and/or pressure instrumentation in the Control Room is lost, dispatch an Aux Operator to monitor those levels from the Patch Panel in the Control Building at 305 feet elevation.

5. When data evaluation shows that an earthquake has occurred, and the peak amplitude of the earthquake has exceeded Safe Shutdown Earthquake limits (0.12g horizontal acceleration and/or 0.08g vertical acceleration) take all actions, necessary to affect a Unit cooldown to cold shutdown condition. Also take whatever steps are necessary to respond to the fault occurring in components, or systems within the Unit as a result of the earthquake.

3.0 FOLLOW UP ACTION

3.1 Any time data evaluation shows that an earthquake has occurred, the duty lead engineer will be notified of the event.

3.2 Anytime data evaluation shows that an earthquake has occurred, replace the magnetic tape cassettes from the strong motion recording system with new cassettes. Then place a record of calibration on the new magnetic tape cassettes.

3.3 If the peak amplitude of the earthquake was great enough to required Unit Shutdown, remove the magnetic erasure clips from the peak reading accelerographs, located in the Unit as shown in Table 1. Replace these clips with new clips and develop and evaluate the removed clips. (Perform these actions in the manner described in the Earthquake Emergency Plan in Site Emergency Plan Administrative Procedure AP 1004, Section 10). Also, evaluate the Triaxial Spectrum Response Recorder in accordance with the Manufacturers' Instruction Manual.

3.4 Submit a report to the NRC pursuant to Technical Specification 4.3.3.3.2.

TABLE 1

Peak Reading Accelerograph (PRA) Locations

PRA attached to RC Pump 1A in the Reactor Building.

PRA attached to RC-P-1A Cold Leg in the Reactor Building.

PRA attached to Switchgear 2-1E in the Auxiliary Building, 328' elevation.

Triaxial Spectrum Response Recorder Loaction

Tendon access gallery in Unit II.

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2203-1.1
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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-1.1
LOSS OF BORON (MODERATOR DILUTION)

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R P Warren Date 9/11/78
V- Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

Unit 2 Superintendent Approval

J. F. Goodman Date 9/12/78

Manager Generation Quality Assurance Approval N/A Date —

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 ABNORMAL PROCEDURE 2203-1.1

LOSS OF BORON (MODERATOR DILUTION)

1.0 SYMPTOMS

1.1 Chemistry analysis indicates a decreasing boron concentration in the Reactor Coolant System.

1.2 Neutron flux levels increasing or with the control rod drive system under automatic ICS control, the control rods are inserting.

1.3 The pressurizer and/or makeup tank levels are increasing.

1.4 "Feed and Bleed Enabled" alarm is actuated (at Panel 8).

1.5 With the reactor vessel head removed, the fuel transfer canal and/or spent fuel pool water levels are increasing.

2.0 IMMEDIATE ACTION

2.1 Automatic Action

None

2.2 Manual Action

2.2.1 For a case where there is a large low borated water inflow and/or the inflow is readily identified, immediately secure the appropriate pump(s) and/or valve(s) to stop the inflow.

2.2.2 Verify the proper makeup and pruification demineralizer is in service.

2.2.3 Commence RCS Boration to maintain Control Rod drives within the insertion limits if applicable.

3.0 FOLLOW UP ACTION

3.1 Initiate a chemistry sampling program for surveillance of boron concentration in order to determine where changes have occurred.

3.1.1 For the case where the reactor vessel head is in place and normal letdown and makeup are in progress, with the DH System secured, sampling should be initiated from the following

areas:

a. RC System via the letdown

b. Pressurizer

c. Makeup Tank

3.1.2 For the case where the reactor vessel head is removed or in place, but only the DH System is in operation, sampling should be initiated from the RC System via the DH System.

3.1.3 The purpose of this sampling is to both aid in locating the source for the loss of boron and to gauge the rate of this loss.

3.2 Determine the amount of shutdown margin available in accordance with 2103-1.9, Reactivity Balance Calculations.

3.2.1 If the shutdown margin is <1% $\Delta K/K$, with K_{eff} <1 or if <2% $\Delta K/K$ with $K_{eff} \geq 1$ immediately initiate and continue to emergency borate as follows until the required shutdown margin is restored.

3.2.1.1 If BAMT is T.S. source and is lined up thru MU Pumps (2301-M2 Att 2A).

a. Start CA-P-4A or 4B

b. Open MU-V10

c. If MU-V10 fails to open, open MU-V127.

d. Start MU-P-1A, B, or C if not already operating.

e. If there is a bubble in the pressurizer maintain level constant by controlling RCS letdown valve MU-V376. If the pressurizer is solid maintain RCS pressure below 500 psig using MU-V376.

3.2.1.2 If RBAT is T.S. source and is lined up thru MU Pumps (2301-M2 Att 2B)

a. Open WDS-V111

b. Start WDS-P-3

c. Start either CA-P-4A or 4B

d. Open MU-V10

e. If MU-V10 fails to open, open MU-V127

f. Start MU-P-1A, B or C if not already operating.

g. If there is a bubble in the pressurizer maintain level constant by controlling RCS letdown valve MU-V376. If the pressurizer is solid maintain RCS pressure below 500 psig using MU-V376.

3.2.1.3 If BWST is T.S. source and is lined up thru DHR Pumps (2301-M2 Att 3)

3.2.1.3.1 If DH-P-1A is operating:

a. Close breaker for DH-V5A at MCC 2-11 EA and open valve

b. Stop DH-P-1A

c. Close DH-V100A

d. Open DH-V102A

e. Start DH-P-1A

3.2.1.3.2 If DH-P-1B is operating

a. Close breaker for DH-V5B at MCC2-21EA and open valve

b. Stop DH-P-1B

c. Close DH-V100B

d. Open DH-V102B

e. Start DH-P-1B

3.2.1.4 If BAMT is T.S. source and is lined up thru MU Pump Bypass (2301-M2 attach 4A)

a. Start either CA-P-4A or 4B

b. Open MU-V10. If MU-V10 fails to open, open MU-V127.

3.2.1.5 If RBAT is T.S. source and is lined up thru MU Pump Bypass (2301-M2 Att 48)

a. Open WDS-V111

b. Start WDS-P-3A

c. Start CA-P-4A or 4B

d. Open MU-V10. If MU-V10 fails to open, open MU-V127.

3.3 Maintain the RC System boron concentration as directed by the Shift Supervisor/Shift Foreman in accordance with 2103-1.2, Soluble Poison Concentration Control.

3.4 If the cause of the loss of boron (moderator dilution) is still not known, refer to the below listed enclosures for guidance on locating the possible cause.

3.4.1 Enclosure 1

Loss of Boron Checklist (RV Head in Place with Makeup System Operational and DH System secured.

3.4.2 Enclosure 2

Loss of Boron Checklist (RV Head Removed or In Place but only DH System in operation).

3.5 Initiate the appropriate action to eliminate the problem.

3.6 When the loss of boron (moderator dilution) has been stopped, cease the sampling program initiated in Step A above and return the systems to their normal conditions for the mode of operation which the unit is in.

3.7 Determine the shutdown margin available in accordance with 2103-1.9, Reactivity Balance Calculations.

Enclosure 1 - Loss of Boron Checklist (RV Head in Place With Makeup System Operational and DH System Secured)

	Possible Cause		Corrective Action
1.	The Makeup System is introducing a fluid of lower boron concentration into RC System.	1.	Close, or verify closed, MU-V10 (Boric Acid & DW Addition Isolation Valve) MU-V294, (DW Addition Isolation Valve) and MU-V290 (MU Pump DW Suction Isolation).
2.	If a batch deboration process is or was in progress, failure of valves MU-V10, (Boric Acid & DW Addition Isolation Valve) and MU-V294, (DW Addition Isolation Valve), to close tight could cause a deboration of RC System.	2.	Close, or verify closed, MU-V10, (Boric Acid & DW Addition Isolation Valve) and MU-V294, (DW Addition Isolation Valve). Use manual verification if necessary.
3.	If a mixed flow batch process is or was in progress, freezing of the boric acid addition line could cause a diluent to flow into the RC System.	3.	Check for level decrease in RBAT or BAMT. Check the heat tracing on the boric acid addition line to verify operation. Close, or verify closed, MU-V10, Boric Acid & DW Addition Isolation Valve. If required, use alternate boric acid addition line for any further boration.
4.	If a feed and bleed deboration process is or was in progress, failure of valves MU-V10, (Boric Acid & DW Addition Isolation Valve) and MU-V294, (DW Addition Isolation Valve), to close tight could cause further RC System deboration.	4.	Close, or verify closed, MU-V10, Boric Acid & DW Addition Isolation Valve and MU-V294, DW Addition Isolation Valve. Use manual verification if necessary.
5.	If a feed and bleed mixed flow process is or was in progress, freezing of the boric acid addition line could cause a diluent to flow into the RC System.	5.	Check for level decrease in RBAT or BAMT. Check the heat tracing on the boric acid addition line to verify operation. Close, or verify closed, MU-V10 Boric Acid & DW Addition Isolation Valve. If required, use alternate boric acid addition line for any further boration.

Enclosure 1 - continued

Possible Cause	Corrective Action
6. If a flow path is inadvertently lined up to the deborating demineralizer(s), boron concentration could drop until saturation is attained.	6. Close, or verify closed, MU-V10, (Boric Acid & DW Addition Isolation Valve) and WDL-V45 (Deborating Demineralizers Inlet Valve). If required, also close deborating demineralizer isolation valves WDL-V81A/B and WDL-V118A/B.
7. Other means of diluents entering the RC System are through the following MU System valves: MU-V285A/B (Makeup & Purification Demineralizers N_2/DW Supply Valves) MU-V292A/B (Makeup & Purification Demineralizers DW Backflush Valves).	7. Close, or verify closed, MU-V285A/B, (Makeup & Purification Demineralizers N_2/DW Supply Valves) and MU-V292A/B, (Makeup & Purification Demineralizers Backflush Valves). If required, remove spool piece from demineralized water supply line to Makeup & Purification Demineralizers (MU-K1A/B).

Enclosure 2 - Loss of Boron Checklist (RV Head is Removed, or in Place but Only the DH System is in Operation)

Possible Cause	Corrective Action
1. Decay Heat Pump(s) (DH-P-1A/B) taking suction from the following sources which may have lower boron concentrations than the RC System:	1. Cease suction from these sources as follows:
a) from SF Cooling System	a) Close, or verify closed, valves DH-V110 and DH-V101A/B, DH Suction from SF System Valves (Manual Valves)
b) from SF Cooling System purification	b) Close, or verify closed, valves SF-V107 and SF-V125, SF Cooling & Purification Loops Isolation Valves. Also verify that the demineralized water supply to the SF Demineralizer (SF-K1) is secured by verifying closed SF-V186, SF Demineralizer N_2/DW Supply Valve. If required, secure the SF purification loop and isolat from the DH System by closing SF-V122, SF Purification to DH System Isolation Valve.
c) from NaOH Storage Tank	c) Close, or verify closed, valves DH-V8A/B, NaOH Storage Tank Isolation Valves. If this is considered to be the source for boron dilution, also consult chemistry department as to effects of NaOH in RC System.
2. During refueling when the transfer tubes are open, the possible inflow of lower boron concentration SF Pool water into the Fuel Transfer Canal could cause RC System boron dilution.	2. Verify that SF Pool and Fuel Transfer Canal water levels are the same and check for possible flow through the transfer tube from the SF Pool to the Fuel Transfer Canal. Adjust system flow as necessary and close FH-V1A/B, Transfer Tube Isolation Valves, if required while corrective action is underway.

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Revision 2
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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-1.2
CRD EQUIPMENT FAILURE

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5.0			30.0			55.0		
6.0			31.0			56.0		
7.0			32.0			57.0		
8.0			33.0			58.0		
9.0			34.0			59.0		
10.0			35.0			60.0		
11.0			36.0			61.0		
12.0			37.0			62.0		
13.0			38.0			63.0		
14.0			39.0			64.0		
15.0			40.0			65.0		
16.0			41.0			66.0		
17.0			42.0			67.0		
18.0			43.0			68.0		
19.0			44.0			69.0		
20.0			45.0			70.0		
21.0			46.0			71.0		
22.0			47.0			72.0		
23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

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Approval N/A Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

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Unit 1 PORC Recommends Approval

N/A Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

J F Ulrich Date 9/22/78
Chairman of PORC

Unit 1 Superintendent Approval

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Unit 2 Superintendent Approval

J J Kelliher Date 9/22/78

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TMI 55-A Rev 3/77

THREE MILE ISLAND NUCLEAR STATION

UNIT #2 ABNORMAL PROCEDURE 2203-1.2

CRD EQUIPMENT FAILURE

MOTOR FAULT

1.0 SYMPTOMS

A. Motor fault lamp on the CRD operators console (Panel 4).

B. Position indication shows in or out motion with no command or out motion with In command for one or more groups of rods (Panel 14).

C. Unexpected increase in reactor power, temperature, and pressure (Panel 4).

D. Possible sequence inhibit indication (Panel 4).

2.0 IMMEDIATE ACTION

A. Automatic Action

1. The CRD control panel reverts to manual.

B. Manual Action

1. Verify that the CRD operators console is in manual
2. Select jog speed.
3. If out motion continues, place the Manual Command Switch in the insert position.

3.0 FOLLOW UP ACTION

A. If uncontrolled out motion continues, trip the reactor.

B. If placing manual command switch in the insert position stops rod withdrawal, deenergize the programmer motor as follows:

1. Locate and remove the fuse on the 120V AC distribution panel that feeds the programmer for the affected group. The 120V AC distribution cabinet is located in the secondary DC hold power supply cabinet.

Group	Fuse
5	F7
6	F8
7	F9
8	F10
Aux	F11

NOTE: Removing the above fuses deenergizes the main programmer lamps as well as the power to the programmer drive motors. The secondary lamps must be operational to avoid tripping that group of rods.

C. If a sequence inhibit has occurred:

1. Select the affected group on the Group Select Switch (Panel 4).
2. Press sequence bypass.
3. Press transfer reset.
4. Continue to hold the manual command switch in the insert position.

D. If the motor fault occurred without excessive out motion, check the DC Brake Power Supply fuse (F2) on the Programmer Control Assembly..

E. If the power supply is faulty, transfer the group to the auxiliary supply and continue operation using the aux supply as the group supply.

NOTE: If the aux supply has a motor fault, the fuse (F-11) on the 120V AC Distribution Panel must be used to synchronize the supply to make the transfer.

STATOR HIGH TEMPERATURE

1.0 SYMPTOMS

A. Computer printout of the affected mechanism giving the stator temperature (Annunciator Group A10).

B. Possible low CRD cooling flow indicated on Panel 8, Annunciator A15.

C. Possible IC cooling CRD filter differential pressure high on Panel 8, Annunciator B15.

D. Possible CRD temperature high on Panel 8 Annunciator C15.

2.0 IMMEDIATE ACTION

A. Automatic Action

B. Manual Action

1. Close MU-V1A, B if CRD outlet temp is > 160°F as indicated by Annunciator Group A10 from computer.
2. Check the IC system for Δ p across the filter for the CRD system.
3. Start standby IC pump and/or valve in standby CRD filter to provide continued CRD cooling, if required.

3.0 FOLLOW UP ACTION

A. Place the temperature input of the affected mechanism on an analog trend recorder of the computer and reset the alarm point for that computer input to 170°F.

B. Observe the stator temperature on analog trend recorder. If the temperature is between 160°F and 170°F and constant, continue normal operation.

C. If the stator temperature is above 170°F and rising, deenergize the mechanism as follows:

1. Reduce power to 60% of the thermal power allowed for the RC pump combination.
2. Bypass the asymmetric rod alarm for the mechanism using the S2 switch and the PI amplifier for the affected CRD.
3. Transfer the rod to the auxiliary supply, drive it to the in limit and deenergize the mechanism by removing the programmer lamp fuses for the aux supply. (Fuse F1 and F3 on the programmer control assembly). Observe Tech. Spec limits on tilt (T.S.3.2.4) and assymetric rods (TS 3.1.3.1).

   CAUTION: Do not pull the motor output fuse on energized phases. Severe flashing may occur. If necessary, check with a voltmeter that each phase is deenergized.

4. Pull the motor output fuses located in the transfer cabinets for the mechanism and replace the programmer lamp fuses.

D. If the stator temperature reaches 180°F, immediately deenergize the mechanism as follows:

1. If not already on the auxiliary power supply, transfer the rod to the auxiliary supply and deenergize the mechanism by removing the programmer lamp fuses for the auxiliary supply. (Fuse F1 and F3 on the programmer control assembly,) Observe Tech. Spec. Limits on tilt (T.S. 3.2.4) and asymmetric rods (T.S.3.1.3.1).

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 ABNORMAL PROCEDURE 2203-1.3

CRD MALFUNCTION ACTIONS ASYMMETRIC ROD FAULT (6.5%)

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12.0	05/12/77	0	37.0			62.0		
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19.0			44.0			69.0		
20.0			45.0			70.0		
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23.0			48.0			73.0		
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N/A Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

R P Warren Date 3/20/79

V. Chairman of PORC

Unit 1 Superintendent Approval

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Unit 2 Superintendent Approval

O R Tomlinson Date 3/20/79

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 ABNORMAL PROCEDURE 2203-1.3

CRD MALFUNCTION ACTIONS

ASYMMETRIC ROD FAULT (6.5%)

1.0 SYMPTOMS

A. Asymmetric rod fault light on the CRD operators console. (Panel 4).

B. One or more fault lamps lit on the Position Indication Panel. (Panel 14).

C. CRD pattern asymmetrical alarm above PI Panel on Panel 14.

D. Possible ICS runback alarm. (ICS Trouble Alarms - Panel 4).

E. Rod mis-alignment indicated by the Position Indication meters (Panel 14) and computer.

F. Possible Group 6/7 fault on CRD operators console. (Panel 4).

2.0 IMMEDIATE ACTION

A. Automatic Action.

1. If rod control is in automatic, the ICS runs the unit back to 55% power on any of the following conditions:

(1) Loss of any Safety group out-limit.

(2) Asymmetric rod with a Group 5 in-limit.

(3) Asymmetric rod with Group 5 $\geq$ 25% and Group 6 or 7 in-limit.

B. Manual Action.

1. Manually reduce power to less than 60%. If runback has occurred, verify that MWe, neutron power and feed flow are decreasing and that neutron power and feed flow level off at the values corresponding to 55% power (527 MWe).

2. If any ICS stations are in manual, run the station in manual back to correspond to 60% of the power allowed for the RC Pump combination.

NOTE: Compatible values of neutron power, MWe and feedwater flow are found on the nomogram located on Panel 4.

3.0 Follow Up Action.

A. Following runback, verify rod group overlap between rod groups 5 and 6/7 is 25 ± 5%, if applicable.

B. If Group 6/7 fault light is lit and the runback has concluded, select manual, sequence bypass and adjust Group 6/7 to obtain the correct overlap, after the group asymmetric condition has been corrected.

C. If quadrant power tilt exceeds limits of T.S. 3.2.4, insure the applicable action statements are followed.

D. Confirm the position and operability of the asymmetric rod by a 2% exercise of the affected group. Check the indicated position with the zone reference lamps in System Logic Cabinet 7 and 8 by transferring the affected rod to the auxiliary supply and driving it in to the nearest zone reference position.

E. If the rod and its position indication are functional, realign the rod with its group average position.

F. If the rod cannot be exercised and is in a safety or regulating group or if both the absolute and relative positions are inoperable, perform the following per Tech. Spec. 3.1.3.1 and 3.1.3.3.

1. Maintain power at 60% of the thermal power allowed for the RC Pump combination.

NOTE: If less than four RC pumps are running, power must be manually reduced from the runback setpoint of 55% to less than 60% of the thermal power allowed for the RC pump combination.

2. Initiate an immediate evaluation to verify existance of two percent ΔK/K hot shutdown margin within one (1) hour. Continue to verify this 2% ΔK/K SDM at least once per 12 hours per 2311-1 until the rod is again declared operable.

3. A shutdown margin of 2% must be present when the worth of an inoperable rod is combined with the highest worth operable rod or the reactor must be brought to hot standby within 6 hours.

4. During the shutdown margin evaluation, boration may be initiated to compensate for the worth of the inoperable rod or until the regulating rods are withdrawn to the limit of the rod index.

5. Reduce the Overpower Trip setpoint to $\leq$ 70% of the Thermal Power allowable for the reactor coolant pump combination within 4 hours.

6. If a shutdown margin greater than 2% is verified within one hour of the determination that a rod is inoperable and if the rod is in a regulating group, and if the control rod is not inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, trim the remaining rods in

the group to the position of the inoperable rod and continue normal operation. Maintain the inoperable rod within the limits of the group average position and maintain proper overlap between groups.

G. Obtain a power distribution map from the incore detectors and verify that FQ and $F_{\Delta}^{N}H$ are within their limits within 72 hours.

J. If more than one rod (other than in Group 8) is inoperable, shut down the reactor.

I. With one or more control rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SDM is $\geq$ 2% $\Delta K/K$ within one hour and be in at least Hot Standby within 6 hours.

J. Perform a re-evaluation of each accident analysis of Table 3.1-1 of the Technical Specifications within 5 days. This re-evaluation shall confirm that the preveiously analyzed results of these accidents remain valid for the duration of operation under these conditions.

DROPPED ROD

1.0 SYMPTOMS

A. Unexpected in-limit indication. (Panel 14).

B. CRD Pattern Asymmetrical Alarm. (Panel 4).

C. Absolute Position Indication shows one or more rods at 0% on both the individual meters (Panel 14) and the computer.

D. Flux tilt as indicated by incore and out-of-core detectors. (Panel 4 and 14).

E. Reduction in reactor power level with accompanying fluctuations in reactor coolant temperature, reactor coolant pressure, and pressurizer level. (Panel 4).

F. Possible Group 6/7 fault indication on CRD operator console. (Panel 4).

2.0 IMMEDIATE ACTION

A. Automatic Action.

1. If rod control is in automatic, the ICS runs the plant back to 55% power.

2. Possible low reactor pressure trip.

B. Manual Action

1. Manually reduce power to less than 60%. If runback has occurred, verify that MWe, neutron power and feed flow are decreasing and that neutron power and feed flow level off at the values corresponding to 55% power (527 MWe).

2. If any ICS station is in manual, run the station back to correspond to 60% of the power allowed for the RC Pump combination.

NOTE: Compatible values of neutron power, MWe and feedwater flow are found on the nomogram located on Panel 4.

3.0 FOLLOW UP ACTION

A. For a Group 6 or Group 7 dropped rod condition, the Group 6/7 fault light will be energized.

B. Do not re-adjust Group 6 or 7.

C. If quadrant power tilt exceeds limits of T.S. 3.2.4, insure the applicable action statements are followed.

D. Confirm the position and operability of the asymmetric rod by a 2% exercise of the affected group. Transfer the dropped rod to the Auxiliary supply. Reset the RPI associated with the dropped rod. Relatch the dropped rod. Check the indicated position with the zone reference lamps in System Logic Cabinet 7 and 8 by driving the rod out to the nearest zone reference position.

E. If the rod and its position indication are functional, realign the rod with its group average position and transfer the rod to the Group Supply.

F. If the rod cannot be exercised and is in a safety or regulating group or if both the absolute and relative positions are inoperable, perform the following per Tech. Spec. 3.1.3.1 and 3.1.3.3.

1. Maintain power at 60% of the thermal power allowed for the RC Pump combination.

NOTE: If less than four RC pumps are running, power must be manually reduced from the runback setpoint of 60% to less than 60% of the thermal power allowed for the RC pump combination.

2. Initiate an immediate evaluation to verify existence of two percent ΔK/K hot shutdown margin within one (1) hour. Continue to verify this 2% ΔK/K SDM at least once per 12 hours per 2311-1 until the rod is again decleared operable.
3. Reduce the Overpower Trip setpoint to ≤ 70% of the Thermal Power allowable for the reactor coolant pump combination within 4 hours.
4. If the rod cannot be re-latched, then have Electrical Maintenance check or replace the fuse in each of the 6 stator phases.
5. To verify that the problem is mechanical and not an electrical problem, have current reading taken of each stator coil. The phase current should be approximately 15 amperes.
6. If the rod cannot be re-latched, and the Tech Spec limitations for power operation cannot be met, then cooldown as required and repair the CRDM.

G. Obtain a power distribution map from the incore detectors and verify that FQ and $F_{\Delta}^{N}H$ are within their limits within 72 hours.

J. If more than one rod (other than in Group 8) is inoperable, shutdown the reactor.

I. With one or more control rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SDM is ≥ 2% ΔK/K within one hour and be in at least Hot Standby within 6 hours.

J. Perform a re-evaluation of each accident analysis of Table 3.1-1 of the Technical Specifications within 5 days. This re-evaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

ROD EJECTION

1.0 SYMPTOMS

A. Asymmetric rod fault alarm as indicated on PI Panel (7")(Panel 14) or console (9") (Panel 4).

B. Possible unexpected out-limit indication (Panel 14).

C. Unexpected increase in reactor power (Panel 4).

D. Possible increase in primary system temperature (Panel 4).

E. Reactor Coolant Pressure will increase or decrease (Panel 4).

F. LOCA symptoms e.g. increase in reactor building pressure (Panel 8 Annunciator A1).

2.0 IMMEDIATE ACTION

A. Automatic Action.

1. If a rod is ejected, the RPS will automatically trip the reactor, and the SFAS will be activated by:

   a) Underpressure of reactor coolant.

   b) Overpressure of reactor building.

   NOTE: Reactivity excursions initiated by uncontrolled rod withdrawal at a fixed rate will be safely terminated by the RPS. In order for reactivity to be added to the core at a more rapid rate, physical failure of the pressure boundary component must occur. The power excursion due to a rapid increase in reactivity is limited by the doppler effect and terminated by reactor protection system trips.

B. Manual Action

1. Manually trip the reactor.

2. Initiate Safety Injection.

3.0 FOLLOW UP ACTION

A. Proceed with the following:

1. Loss of RC/RC Press - Procedure No. 2202-1.3 as applicable.

STUCK ROD

1.0 SYMPTOMS

A. Failure of the rod to respond to group commands.

B. CRDM may be de-energized with no in-limit indication and no zero position indication (Panel 14).

2.0 IMMEDIATE ACTION

A. Automatic Action.

1. None.

B. Manual Action.

1. Transfer CRDCS to manual mode.
2. Transfer the stuck rod to the Auxiliary Power Supply per 2105-1.9.
3. Attempt to move the rod by first inserting the rod, at jog speed, a short distance (about 20 seconds) and then withdrawing it in the out direction.
4. If the rod can be moved, then align the rod with the group and transfer the rod from Auxiliary Supply to the Group Supply.
5. If the rod cannot be moved reduce power to 60% of allowable power for RCP combination.

3.0 FOLLOW UP ACTION

A. If quadrant power tilt exceeds limits of T.S. 3.2.4, insure the applicable action statements are followed.

B. If the stuck rod cannot be moved with the Auxiliary Supply, then have Electrical Maintenance check or replace the fuse in each of the 6 stator phases. Fuse blocks are located in front of the CRDCS Transfer Cabinets 1 through 8.

C. To verify that the problem is mechanical and not an electrical problem, have current reading taken of each stator coil. The phase current should be approximataly 15 amperes.

D. If step B above shows that the problem is mechanical, then again try to exercise the rod.

E. Investigate the possibility of an inoperable API channel. By inserting the affected rods RPI indication to the nearest zone reference lamp area. If it is confirmed that the rod has defective API indication, proceed with action statements of 3.1.3.3.

F. With one or more control rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the shutdown margin requirements of Specification 3.1.1.1 is satisfied within one hour and be in at least hot standby within 6 hours per T.S. 3.1.3.1.

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-1.4
RCP & MOTOR EMERGENCIES

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Approval N/A Date —

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Unit 2 Staff Recommends Approval

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Unit 1 PORC Recommends Approval

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Chairman of PORC

Unit 2 PORC Recommends Approval

J. F. Wilson Date 5/2/78

V. Chairman of PORC

Unit 1 Superintendent Approval

N/A Date —

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J. J. Colitz Date 5/4/78

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 ABNORMAL PROCEDURE 2203-1.4

RCP & MOTOR EMERGENCIES

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 ABNORMAL PROCEDURE 2203-1.4

RCP & MOTOR EMERGENCIES

1.0 LOSS OF SEAL INJECTION TO PUMPS

1.1 Symptoms

1.1.1 ALARM: "Reactor Coolant Pump Loop Seal Inject Flowlow" (8:B.27).

1.1.2 ALARM: "Reactor Coolant Pump Total Injection Loop Flow/Hi/Lo" (8:A.27).

1.1.3 Seal Inlet Flow (individual) decreasing (panel 3).

1.1.4 RC Pump Total Seal Inlet Flow decreasing (panel 3).

1.2 Immediate Action

1.2.1 Automatic Action

a. None

1.2.2 Manual Action (Loss of Injection).

a. If seal injection flow to operating pumps is lost, the pumps may continue operation provided intermediate cooling water flow is maintained and seal temperatures remain within the following limits.

T_1 <185°F

T_2 <170°F

T_4 <185°F

If seal injection flow to idle reactor coolant pumps is lost the associated seal return isolation valve (MU-V33 A,B,C or D) must be closed within one minute to prevent seal damage.

b. During loss of injection, monitor seal bleedoff temperature Pump Bearing Outlet temperature, seal bleedoff flow, and

seal leakage flow carefully. If seal leakage plus seal returns exceeds 1.9 gpm, trip the pump, cooldown, and inspect the seals.

c. Close the injection flow control valve (MU-V32) to prevent high injection flow upon recovery of injection.

d. The seal bleedoff flow control valves (MU-V33 A, B, C, D) should remain open as long as the motors remain energized.

1.2.3 Follow Up Action (Recovery of Injection)

a. When injection fluid is available, reestablish injection flow as follows:

1. Slowly open the injection flow control valve (MU-V32) to establish a flow of 2 gpm per pump. Continue at this flow rate for a period of 2 minutes.

2. Increase injection flow to 5 gpm per pump for a 2 minute period.

3. If seal return valves, MU-V33 A, B, C, or D has closed, reopen them at this time.

4. Increase injection flow to normal flow of about 10 gpm per pump.

1.3 Follow Up Action

1.3.1 In the event the pump must be shutdown during a loss of injection mode, the following steps should be taken:

NOTE: The pump should not be shutdown when injection flow is lost unless it is absolutely necessary.

a. Reduce power to within RPS limits.

b. Trip the RCP. The appropriate seal return isolation valve (MUV33 A, B, C or D) must be closed within one

minute after pump shutdown. Shutdown Oil Lift System at zero speed.

c. Continue flow of intermediate cooling water at normal flow until Reactor Coolant System temperature is less than 200°F and pressure is below 200 psig.

d. Do not restart pump until injection flow is available. Recover injection flow per section 1.2.3 above.

e. If RCP seal temperatures T_1 or T_4 have exceeded 185°F, or if seal leakage plus seal return exceeds 1.9 gpm, do not restart the pump. Cooldown and inspect the seals for damage.

2.0 SEAL FAILURE

2.1 Symptoms

2.1.1 Abnormal Pump Seal Cavity Pressure

2.1.2 Excessive Pump vibration

2.2 Immediate Action.

2.2.1 Automatic Action.

a. None.

2.2.2 Manual Action.

a. The affected RCP must be tripped if the seal cavity pressures exceed 2500 psig.

b. If the sum of seal bleedoff flow and seal leakoff flow exceeds 3.0 gpm - TRIP THE RCP.

2.3 Follow Up Action

2.3.1.a Determine degree of seal failure as follows:

a.1 Failure of No. 1 seal is indicated by Pump Seal Cavity pressure between seal 1 and 2 at approximately system pressure.

a.2 Failure of No. 2 seal is indicated by Pump Seal Cavity pressure between seal 2 & 3 at approximately ½ system pressure.

a.3 Failure of No. 3 seal is indicated by Pump Seal Cavity pressure between seal 2 & 3 at approximately 50 psig.

a.4 Failure of No. 1 and No. 2 seals is indicated by Pump Seal Cavity pressure between seal 1 & 2 and 2 & 3 at approximately system pressure.

a.5 Failure of No.2 and No. 3 seals is indicated by Pump Seal Cavity pressure between seal 1 & 2 at approximatley 50 psig.

a.6 Failure of No. 1 and No. 3 seals is indicated by pump seal cavity pressure between seal #1 and 2 at approximatley system pressure, and pressure between seal #2 and 3 at approximately 50 psig.

NOTE: All values are approximate. Destaging may cause only partial ΔP cross a seal.

2.3.2 If two seals are leaking, the pump may be operated, however, a planned unit shutdown should be scheduled.

2.3.3 In the event the pump must be shutdown, the following steps should be taken:

a. Reduce power to within RPS limits.

b. Trip the RCP.

c. When pump reaches zero speed, Lift Pump and Back Stop Pumps should be stopped.

d. Continue flow of Cooling Water and Seal Injection until RC System Pressure is below 200 psig, and temperature is below 200°F.

e. Do not attempt to restart pump until seal problem has been fixed.

3.0 PUMP AND MOTOR VIBRATION

3.1 Symptoms

3.1.1 Excessive shaft vibration on IRD System greater than 30 mils for first four hours of one pump per loop operation. Greater than 25 mils for one or two pump per loop operation.

3.2 Immediate Action.

3.2.1 Automatic Action

a. None.

3.2.2 Manual Action.

a. Reset shaft vibration alarm on IRD System Panel 10.

b. If vibration alarms reoccur immediately, reduce power to within RPS limits and trip affected RCP per normal procedure.

3.3 Follow Up Action.

3.3.1 Determine cause and repair.

4.0 REVERSE ROTATION

4.1 Symptoms.

4.1.1 Excessive vibration due to a backstop failure.

4.2 Immediate Action.

4.2.1 Automatic Action.

a. None.

4.2.2 Manual Action.

a. Start oil lift system on affected pump.

b. Commence a normal plant shutdown and cooldown of the R.C.S. per 2101-3.2.

4.3 Follow Up Action.

4.3.1 Determine cause and repair.

5.0 LUBE OIL EMERGENCY

5.1 Symptoms.

5.1.1 Low lift pump discharge oil pressure alarm (1800 PSIG) on computer.

5.1.2 High bearing temp. alarms Radial 185°F, 8.831, Thrust 200°F 8.E31 for Upthrust temp Hi and F31 for downthrust temp Hi.

5.1.3 Hi/Lo Levels alarms on upper and lower oil pots annunciated on computer.

Upper Oil Pot - Hi 12.2 INC.

Lo 8.2 DEC.

Lower Oil Pot - Hi 13.8 INC.

Lo 9.8 DEC.

5.1.4 RCP lube oil flow Lo. (7.0 gpm on computer).

5.1.5 RCP Backstop oil flow low. (0.24 gpm on computer).

5.2 Immediate Action.

5.2.1 Automatic Action

5.2.2 Manual Action.

a. Start D.C. oil lift pump.

b. In the event of high motor bearing temps, (Radial>185°, thrust >200°F), reduce reactor power to 75% and trip affected pump.

5.3 Follow Up Action

5.3.1 If low exists on lift and/or backstop system, start lift and/or backstop pumps on affected RCP.

5.3.2 If Hi/Lo levels exist in oil pots, reduce reactor power to 75% and secure affected pump.

5.3.3 A possibility of oil fire exists if the oil systems have ruptured. To prevent oil fire, the unit will be shutdown and cooled down to 50°F below the Flash point of the oil used in the system. The flash point of the oil is 450°F.

5.3.4 Determine cause and repair.

6.0 PUMP MOTOR SEPARATION - DROPPED IMPELLER

6.1 Symptoms.

6.1.1 Low motor current.

6.1.2 Low R.C.S. flow.

6.1.3 Hi vibration:

a. Motor stand Vibraswitch alarms.

b. Shaft - IRD System Panel 10.

6.2 Immediate Action.

6.2.1 Automatic Action.

a. Possible reactor trip on power to flow.

6.2.2 Manual Action.

a. Reduce power and trip reactor.

b. Secure affected RCP.

c. Commence shutdown and cooldown of the RCS.

6.3 Follow Up Action.

6.3.1 Determine cause and repair.

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-1.5
LOSS OF RC MAKEUP

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R P Warren Date 9/1/78
V- Chairman of PORC

Unit 1 Superintendent Approval

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 ABNORMAL PROCEDURE 2203-1.5

LOSS OF RC MAKEUP

1.0 SYMPTOMS

1. Operating MU Pump tripped and standby MU Pump not running or providing sufficient makeup.
2. Makeup flow indication abnormal.
3. MU Pump discharge header pressure high (2850 psig) or low (2400 psig) as indicated on Panel 3.
4. Low seal injection flow to RCP seals as indicated on Panel 3.
5. Decreasing pressurizer level with no change in RCS temperature or letdown flow.

2.0 IMMEDIATE ACTION

A. Automatic Action

1. Decreasing flow caused MU-V32 RCP seal injection flow control valve to open and decreasing pressurizer level will open MU-V17.

B. Manual Action

1. CLOSE MU-V376, Letdown Isolation.
2. Determine cause for loss of R.C. Makeup.
   a. Operating and Standby MU pumps tripped.
   b. MU-V17 closed.
   c. Low Level in MU Tank.

3.0 FOLLOW-UP ACTION

1. MU Pumps Tripped
   a. Verify IC flow to RCP's by flow on IC-3-FI between 650 - 1000 gpm.

b. Place MU-V17 and 32 control stations to manual and close.

NOTE: The MU system is normally line-up such that the "B" Pump is running powered from BUS 2-1E with the "A" Pump as its backup, or with the "B" Pump running powered from BUS 2-2E with the "C" Pump as its backup, as determined by the Backup Start Selector and System Valve Line-up.

c. Attempt to restart one of the tripped MU Pumps. If unsuccessful open DH-V5A (B) and maintain pressurizer level with the High Pressure Injection MU Pump controlling flow with MU-V16A, B, C or D.

d. Slowly re-establish RCP Seal Injection Flow per 2203-1.4 and place MU-V32 control station to auto.

NOTE 1: Limit RC Pump radial bearing temperature decrease to not more than $1^{o}F$/min.

NOTE 2: If RC pump seal temperatures T_1 or T_4 exceeded $180^{o}F$, if T_3 is $>170^{o}F$, or if seal leakage plus seal return is >1.9 gpm, shutdown and inspect the RCP seals.

e. Re-establish pressurizer level and place MU-V17 control station to auto.

f. Restore letdown flow by opening MU-V376.

2. MU-Pump Operating, MU-V17 Failed Closed

a. Shift MU-V17 MU flow control station to manual and restore pressurizer level.

b. If MU-V17 control station has failed, take local control of valve and restore makeup flow. MU-V17 bypass valve MU-V155 may also be used.

c. Restore letdown flow by opening MU-V376.

3. Loss of MU Due to Low Level in MU Tank

NOTE: Loss of reactor coolant may exhibit some similiar symptoms. If any doubt exists, follow 2202-1.3, Loss of Reactor Coolant/RCS Pressure.

a. Trip the operating Makeup Pump and close MU-V32.

b. Verify IC flow to RCP's by flow on IC-5-FI between 650 - 1000 gpm.

c. Fill the MU Tank from the applicable RC Bleed/Holdup Tank per 2104-1.2.

d. If unable to establish MU Tank level in step "c" proceed as follows:

1. Close MU-V12 and establish suction from the BWST to a MU pump by opening DH-V5A (B), (DH-V147A (B) are locked open).

2. Start the Backup Make Up Pump. Restore makeup and slowly open MU-V32 to re-establish seal injection to the RCP seals per 2203-1.4.

3. If RC pump seal temperatures T_1 or T_4 exceeded 180°F, if T_3 is >170°F, of if seal leakage plus seal return is >1.9 gpm, shutdown and inspect the RCP seals.

e. Restore letdown flow by opening MU-V376.

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-1.6
NUCLEAR SERVICE CLOSED COOLING WATER FAILURE

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Cognizant Dept. Head

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Unit 1 PORC Recommends Approval

NA Date —

Chairman of PORC

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R.P. Jarvis Date 10/6/78

V- Chairman of PORC

Unit 1 Superintendent Approval

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J.J. Doll Date 10/6/78

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 ABNORMAL PROCEDURE 2203-1.6

NUCLEAR SERVICE CLOSED COOLING WATER FAILURE

1.0 SYMPTOMS

1.1 Low Nuclear Service Pump Suction Pressure Alarm.

1.2 Nuclear Service Pump Trip Alarms.

1.3 Low Nuclear Services Surge Tank Level Alarm.

1.4 Decreasing Nuclear Service Pump Suction or Discharge Header Pressure.

1.5 Nuclear Service Cooler Discharge Temperature High Alarm.

2.0 IMMEDIATE ACTION

2.1 Automatic Action

2.1.1 Standby Nuclear Service Pump will start automatically if running pump trips.

2.1.2 Nuclear Services Closed Cooling Water Head Tank level control will modulate valve DW-V50 to maintain proper level in the head tank.

2.1.3 Operating make-up pump will trip on loss of cooling water. Standby make-up pump will not start (non-ES only).

2.1.4 Instrument air compressor(s) will trip on reaching high discharge air temperature.

2.1.5 BS-P1A&/or B will trip on low cooling flow if running and 30 psig ES signal is not actuated.

2.2 Manual Action

2.2.1 Verify that the standby Nuclear Services Closed Cooling Pump has started if an operating pump has tripped. If not, manually start the standby pump.

2.2.2 If the Nuclear Services Closed Cooling Water Head Tank has alarmed low, verify that make-up valve DW-V50 is open to increase tank level. If not, open manual bypass valve DW-V53. Check Demin Water Tank level and that DW-P-1A/B are running.

2.2.3 Attempt to locate and isolate source of leakage.

3.0 FOLLOWUP ACTION

CAUTION: The Reactor Coolant and Make-up Pump have the following temperature limitations. If these limits are reached, pumps should be shut down. Loss of Reactor Coolant Pumps will necessitate natural circulation.

	R. C. PUMP	MAKEUP PUMP
Thrust Bearing Temp, ^{o}F Computer Point	200	180 MU-P-1A - 352 MU-P-1B - 353 MU-P-1C - 354
Stator Temp, ^{o}F Computer Point	302 ($150^{o}C$) RC-P-1A - 1670 RC-P-2A - 1672 RC-P-1B - 1673 RC-P-2B - 1671	257 ($125^{o}C$) MU-P-1A - 1663 MU-P-1B - 1664 MU-P-1C - 1665
Time without cooling water (including coast-down time)	10 Min.	

3.1 If non-isolable leakage exists in excess of normal NSCCW make-up capacity, commence plant shutdown and cooldown, start standby Nuclear Services River Water Pumps and open NSCCW - NSRW crosstie valves NS-V154, NS-V191, NS-V193.

3.2 If leakage exists in Make-up, (MU) Building Spray (BS) or Reactor Building Emergency Cooling Booster (RR) systems, crosstie with Nuclear Services River Water for cooling per the following valve line-up.

	NS Valves (close)	NR Valves (open)
Building Spray (BS)	NS-V192, 195	NS-V193, 194
Make-Up (MU)	NS-V190, 197	NS-V191, 196
Emergency Booster (RR)	NS-V155, 156	NS-V154, 157

3.3 When cooling water flow is established, restart a Make-up Pump.

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 EMERGENCY PROCEDURE 2203-1.7
NUCLEAR SERVICES RIVER WATER FAILURE

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Cognizant Dept. Head

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Unit 1 PORC Recommends Approval

_____ Date _____
Chairman of PORC

PORC comments of _____ included
(date)

By _____ Date _____

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PORC comments of _____ included
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By _____ Date _____

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 PLANT EMERGENCY PROCEDURE 2203-1.7

NUCLEAR SERVICES RIVER WATER FAILURE

1.0 SYMPTOMS

1.1 "NSRW Header Pressure Low" alarm @ 20 psig decreasing on annunciators 8.A5 or 8.C5.

1.2 "2A/2B or 2C/2D NSRW Pump Trip" alarm on annunciator 8.34 or 8.04.

1.3 NSRW Pump Discharge Indicators on Panel 8 indicator low discharge pressure.

2.0 IMMEDIATE ACTION

A. Automatic Action

1. The standby NSRW pump in the operating header will start if the running NSRW pump tripped and power is available.
2. The NSRW selfcleaning strainer automatically backwashes on high ΔP.

B. Manual Action

1. Verify the standby pump starts, if not manually start the standby pump.
2. If the standby NSRW pump in the operating header cannot be started
    a. Start a NSRW pump in the nonoperating header.
    b. Shift NSCCW, ICCW and DHCCW Coolers from the affected NSRW header to the operating NSRW header.
3. If NSRW is completely lost (both NSRW headers are out of service)

a. Reduce plant load, to 15%.

b. Trip the reactor if CRD stator temperatures read 180°F.

c. Trip the reactor and stop the reactor coolant pumps, if the RC pump motor stator temperatures exceed 150°C.

d. Stop letdown flow by closing MU-V376.

e. Stop SG hot draining if in progress.

f. Reduce heat load on the NSCCW system as follows

1) Isolate NSCCW to RC waste evaporator by locally closing NS-V32.

3.0 FOLLOW-UP ACTION

1. Verify the following valves on the operating header are fully open.

a. NSRW pump discharge valve associated with the operating pump; NR-V2A, 2B, 2C or 2D.

b. NSRW header isolation valves; NR-V4A + NR-V5A or NR-V4B + NR-V5B.

c. ICCW cooler inlet valves; NR-V51A or 51B.

d. NSCCW cooler inlet valves; NR-V6A or 6B if valves are not open, open manually.

2. Place NSRW self-cleaning strainer in the operating header in operation.

3. As NSCCW and ICCW temperatures increase, the CRD stator temperatures and RC pump stator and motor bearing temperatures will increase.

4. Shutdown non-essential equipment served by the NSCCW system as operating conditions permit (e.g. spent fuel coolers, waste gas compressors and instrument air compressors).

5. The makeup pumps must be stopped if any of the below limiting conditions are reached, if two of the three makeup pumps become inoperable, the reactor must be tripped.

   a. Radial bearing temperature exceeds 170°F.

   b. Pump gear bearing temperature exceeds 165°F.

   c. Thrust bearing temperature exceeds 170°F.

   d. Motor stator temperature exceeds 125°C.

   e. Motor bearing temperature exceeds 180°F.

6. When the plant has reached 15% power fill the pressurizer to the high end of the normal operating band and shutdown the makeup pumps to avoid reaching limiting conditions on the makeup pumps.

   CAUTION: Do not perform any operations which would cause a change in pressurizer level without starting a makeup pump.

   CAUTION: The makeup pumps are limited to two successive starts, each, after which the motor stator temperature must be reduced to 15°F below operating temperature before restart is possible.

7. If the Reactor is tripped, maintain the plant in the Hot Shutdown to preclude continuous use of the makeup pumps. Add boron as required to achieve Hot Shutdown Boration and fill the pressurizer to the high end of the normal band. Stop the makeup pumps and restart as necessary to maintain pressurizer level.

a. After steady state temperature has been reached in the NSCCW System and all makeup pumps are stopped, the NSCCW pumps may be stopped to prevent adding pump heat to the NSCCW System.

CAUTION: Prior to restarting a makeup pump NSCCW flow must be established.

8. If "NSRW Self-Cleaning Strainer High ΔP" alarm failed to clear, proceed to the River Water Pump House and accomplish the following:

   a. Start a NSRW pump in the affected header.

   b. Verify high ΔP on the local ΔP gage.

   c. Verify the strainer is backwashing and the backwash valve is open.

   d. Backwash until high ΔP alarm clears.

   e. If the NSRW self-cleaning strainer is not backwashing properly, manually initiate backwash and open the backwash valve.

   f. If the high ΔP alarm still fails to clear, stop the NSRW pump in the affected header and refer to corrective maintenance procedures for the NSRW self-cleaning strainer (Zurn Ind. Manual 52.00).

9. If NSRW prelube pumps started due to low normal lubrication pressure, proceed to the River Water Pump House and clean the strainer in the normal lubrication line. Refer to Byron Jackson Cuno Filter Manual 97.00.

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-1.8
VIBRATION AND LOOSE PARTS MONITORING SYSTEM

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N/A Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

[signature] Date 10/25/78
V- Chairman of PORC

Unit 1 Superintendent Approval

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 - ABNORMAL PROCEDURE 2203-1.8

Vibration and Loose Parts Monitor System

1.0 SYMPTOMS

a) Alarm lights in monitor cabinet on 305' elev. of Control Bldg. (Cable Room) Vibration or Loose Parts Indicator energized for one or more channels.

(Channels 1 thru 12)

Nuclear Noise Indicator energized for one or both channels.

(Channels 13 thru 14)

b) Abnormal noises heard over local (in cabinet) or remote (Panel 8, Control Room) speakers.

c) Sudden increase in amplitude (a change which exceeds two times the background trace) noted on the chart recorders for one or more channels.

(Channels 1 thru 12) Determine increase in amplitude in accordance with Appendix I of 2105-1.13.

2.0 IMMEDIATE ACTION

A. Automatic Action - None

B. Manual Action

1) Report alarm and affected channels to Shift Foreman/ Supervisor.

2) Attempt to reset the alarm lights by pressing the reset pushbutton at the loose parts monitor panel.

3.0 FOLLOW UP ACTION

1. Record the alarming channel or channels in accordance with 2105-1.13.

2. Determine type and degree of noise.

   a. If the noise is loud on more than one channel (ie heard repeatedly)

      1. TRIP THE REACTOR

      2. Using the cassette recorder obtain one minute of data for each sample using the order below:

	Recorder Channel Selection			
Sample	1	2	3	4
1	1	3	5	9
2	2	4	6	10
3	7	11	8	12
4	13	14	5	6

      NOTE: Record tape footage from recorder meter for each sample, date and time.

      3. If noise continues selectively TRIP Reactor Coolant Pumps until noise stops and record Unit status. (ie. RCP combination running and/or other abnormal conditions). Trip remaining Pumps.

      NOTE: If Reactor Coolant Pumps were selectively tripped immediately notify the Unit Superintendent. Limit time of reduced pump operation in accordance with B & W recommendations.

      4. Cooldown per 2202-1.1 and 2102-3.3.

   b. If the noise is intermittant or periodic on more than one channel

      1. Determine location based on loudest channel.

2. Using the cassette recorder obtain one minute of data for each sample using the order below:

Sample	Recorder 1	Channel 2	Selection 3	4
1	1	3	5	9
2	2	4	6	10
3	7	11	8	12
4	13	14	5	6

NOTE: Record tape footage from recorder meter for each sample, date and time.

3. Check the following Unit parameters for abnormalities:

1. R.C. Flow
2. R.C. Pressure
3. R.C. Pump Seal Flow
4. R. C. Pump Vibration
5. R. C. Pump and Motor Bearing Temperatures
6. Power Range Power level
7. CRDM Drive Temperatures
8. MU-R-720 Readings
9. Letdown Filter dP
10. Letdown Sample (Upstream of Filter)
11. Secondary Sample
12. Feedwater Control Valves dP's
13. VA-R-748 Readings

NOTE: Some power levels have much more background noise than others. (There is considerably more background noise at 15% power.)

If significant abnormalities exist in condition 3.1.-3.13 record condition and TRIP the Reactor.

4. If the noise appears to be in one of the Steam Generators reduce Reactor Power and selectively TRIP on appropriate Reactor Coolant Pump. (ie. If noise is loudest or A(B) Steam Generator start by tripping RC-P-1A(RC-P-1B).

   NOTE: If Reactor Coolant Pumps were selectively tripped immediately notify the Unit Superintendent. Limit time of reduced pump operation in accordance with B & W recommendations.

5. If noise persists reduce Reactor Power and trip a Reactor Coolant Pump in the other loop (if two operating).

6. If noise continous TRIP the REACTOR and remaining Reactor Coolant Pumps until noise stops.

7. Cooldown per 2202-1.1 and 2102-3.3.

c. If noise is either a soft thumping, grinding or popping on more than one channel, or a loud noise on only one channel.

1. Determine location based on noise observed.

2. Using the cassette recorder obtain one minute of data for each sample using the order below:

Sample	Recorder Channel Selection 1	2	3	4
1	1	3	5	9
2	2	4	6	10
3	7	11	8	12
4	13	14	5	6

NOTE: Record tape footage from recorder meter for each sample, date and time.

3. Check the following Unit parameters for abnormalities:

   1. R.C. Flow
   2. R.C. Pressure
   3. R.C. Pump Seal Flow
   4. R.C. Pump Vibration
   5. P.C. Pump and Meter Reading Temperatures
   6. Power Range Power Level
   7. CRDM Drive Temperature
   8. MU-R-720 Readings
   9. Letdown Filter dP
   10. Letdown Sample (upstream of filter)
   11. Secondary Sample
   12. Feedwater Control Valves dP's
   13. VA-R-748 Reading

   NOTE: Some power levels have much more background noise than others. (There is considerably more background noise at 15% power.)

   Additional checks to be made, if deemed necessary by the Shift Foreman/Shift Supervisor.

   1. 3D Core Power Maps
   2. Primary and Secondary Samples
   3. Entry into the Reactor Building to confirm abnormal sounds by local observation.

4. If abnormalities exist record condition and Shift Supervisor/foreman will evaluate whether to continue operation or TRIP

Reactor. If all conditions are normal the Supervisor of Operations, Unit Superintendent-Technical Support and Unit/Station Supertendent will evaluate continued operation.

3. Strip Charts (LPV Monitor) should be checked for time of occurrence, so that alarm may be correlated with a possible large electrical impulse or maintenance personnel action which may produce spurious alarm.

4. Should all observations and plant parameters indicate normal operation, but LPV monitor system still indicates abnormal conditions, the system will be thoroughly checked for equipment failure. If the LPV Monitor System is found to be operating properly, further action will be based on Shift Supervisor, Supervisor of Operations, Station/Unit Superintendent and Technical Service Personnel decisions.

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-2.1
LOAD REJECTION

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5.0			30.0			55.0		
6.0			31.0			56.0		
7.0			32.0			57.0		
8.0			33.0			58.0		
9.0			34.0			59.0		
10.0			35.0			60.0		
11.0			36.0			61.0		
12.0			37.0			62.0		
13.0			38.0			63.0		
14.0			39.0			64.0		
15.0			40.0			65.0		
16.0			41.0			66.0		
17.0			42.0			67.0		
18.0			43.0			68.0		
19.0			44.0			69.0		
20.0			45.0			70.0		
21.0			46.0			71.0		
22.0			47.0			72.0		
23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

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Cognizant Dept. Head

Unit 2 Staff Recommends Approval

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Unit 1 PORC Recommends Approval

N/A Date —

Chairman of PORC

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J. F. Hellesch Date 6/3/78

Chairman of PORC

Unit 1 Superintendent Approval

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Unit 2 Superintendent Approval

J. J. Colitz Date 6/2/78

Manager Generation Quality Assurance Approval N/A Date —

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 ABNORMAL PROCEDURE 2203-2.1

LOAD REJECTION

1.0 SYMPTOMS

1.1 Outgoing 500KV line breakers open.

1.2 Reactor power decreasing.

1.3 Increase in Pressurizer level and Reactor pressure.

1.4 Load decreases to station service, 50-100MW, if all outgoing 230KV and 500KV lines are lost.

1.5 If 230KV substation outgoing lines and the auto transformer tie is available, the reactor and turbine generator remain on line supplying reduced load.

1.6 Possible high turbine header pressure, lifting of main steam safety valves, which will cause a reduction in pressurizer level and decrease in Tave.

1.7 Possible increase in generator frequency.

2.0 IMMEDIATE ACTION

A. Automatic Action

1. The ICS trips to track because the turbine control trips to MANUAL on turbine header pressure error.
2. ICS ULD runs back in tracking mode to a Reactor Power corresponding to the reduced load on the unit.
3. Main steam safety valves open.
4. Turbine bypass valves open (If load rejection is great enough).
5. Pressurizer relief valve, RC-R2 may open.

B. Manual Action

1. Verify Feedwater flow and reactor power are being reduced to supply one of the following:

   a) For loss of outgoing 500KV lines with the auto transformer and 230KV system in service; unit load reduces to approximately 866 MWe or less.

   b) For loss of both the 500KV and 230KV systems unit load will reduce to less than 100 MWe depending upon the status of Unit 1.

   NOTE: These steps are applicable only for loss of both the 230KV and 500KV systems.

2. If diamond power or reactor master is on HAND, insert rods manually to ≈15% neutron power. Rate of rod insertion should be that required to maintain reactor coolant system pressure between 2055 and 2255 psig.

3. If any feedwater stations are on hand, total feedwater should be reduced manually to that corresponding to 15% neutron power (approximately 1.82×10^6 lb/hr). Rate of feedwater reduction should be that required to maintain RCS pressure between 2055 and 2255 psig.

4. If any of the following ICS stations are in hand (Steam Generator/Reactor Demand, either Feedwater Demand, Main or Startup Feedwater Valve Demand, Feedpump Speed, Reactor Master, and/or Diamond) runback the appropriate ICS stations.

3.0 FOLLOW UP ACTION

A. Loss of 230KV and 500KV Systems.

1. Monitor pressurizer level, RC pressure, RC temperature, steam generator level, and steam header pressure.
2. Utilize pressurizer heaters and spray to control RC pressure at 2155 psig. With steam header pressure setpoint at 885 psig, header pressure should be 935 psig. Adjust makeup and letdown flows to control pressurizer level at 240". Adjust feed flow to control steam generator level at 30".
3. Perform Surveillance Requirement 4.8.1.1.2 a.4, verifying the diesel starts from ambient conditions, for both diesel generator and at least once per 8 hours thereafter, unless at least one of the inoperable offsite sources is restored to OPERABLE status.
4. Verify main steam safety valves and the electromatic relief valve RC-R2 on the pressurizer have closed.
5. While carrying auxiliary load, monitor turbine generator for vibration, evidence of rubbing, high exhaust temperature (>175^{o}F) or other conditions that might require turbine generator shutdown.
6. Reduce Reactor Power manually to match generated MW.
7. Check the water level in high and low pressure feedwater heaters at the local sight glass to ensure proper heater drain valve operation.

B. Loss of the 500KV System.

1. Reduce load on the unit as directed by the system dispatcher to prevent damage to the 500/230KV auto-transformer as directed by the System Dispatcher.

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-2.2
TURBINE TRIP

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9.0			34.0			59.0		
10.0			35.0			60.0		
11.0			36.0			61.0		
12.0			37.0			62.0		
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23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

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Unit 1 PORC Recommends Approval

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Chairman of PORC

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R P Jarrett Date 10/25/78

Chairman of PORC

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J. J. Colitz Jr. Date 10/25/78

Manager Generation Quality Assurance Approval N/A Date ——

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 ABNORMAL PROCEDURE 2203-2.2

TURBINE TRIP

1.0 SYMPTOMS

1. ICS in the tracking mode.
2. Turbine throttle, intercept and reheat stop valves closed.
3. Generator breakers open and megawatts electric to zero.
4. Reactor power decreasing.
5. Any one of the following cause a turbine trip:
   a. Generator trip
   b. Reactor trip
   c. Both feed pumps tripped
   d. Loss of all three Secondary Closed Cooling Water Pumps
   e. Vacuum <18 - 22" Hg
   f. Loss of 2 of 3 TURBINE SPEED SIGNALS
   g. Overspeed (1998 RPM)
   h. Thrust bearing failure trip >75 - 80 psig pressure
   i. Low bearing oil pressure trip <5-7 psig
   j. EHC loss of D.C. power
   k. High vibration (greater than 14 mils. for 10 Sec.)
6. Pressurizer level increasing above 240", steam pressure increasing above 885 psig, and T avg. increasing above 582^{o}F.

2.0 IMMEDIATE ACTION

A. Automatic Action

1. Turbine throttle, govenor, intercept and reheat stop valves close and Generator breakers open.

2. Turbine bypass valves or atmospheric relief open to control 885 PSI header pressure.
3. Pressurizer Power Operated Relief Valve open.
4. ICS trips to track and run back at 20%/min. or if the turbine trip is from loss of feedwater pump(s), the run back is 50%/min. to 15% power.
5. If both the main feed pumps have tripped, the steam driven emergency feed pump & two motor driven emergency feed pumps will start.
6. The feed demand will control O.T.S.G. at minimum level (30").
7. The Seal Oil Backup Pump, Bearing Oil Lift Pumps and Turning Gear Oil Pump will start as turbine speed decays.
8. Extraction valves on all feedwater heaters close, and extraction line drain valves open.

B. Manual Action

1. Verify that the Turbine Stop Valves are closed, generator breakers and field breakers are open. Verify the start of the Seal Oil Backup Pump, the Turning Gear Oil Pump, the Bearing Lift Pumps, and verify closed extraction steam valves:

3 Stage EXV 23 A/B and 22 A/B	8 Stage EXV 29 A/B
11 Stage EXV 6 A/B	10 Stage EXV 1 A/B
13 Stage EXV 16 A/B and 11 A/B	

2. If any feedwater stations are in hand, total feedwater should be reduced manually to that corresponding to 15% neutron power.

3. If Diamond Power or Reactor Master is in hand, run rods in manually to 15% neutron power.

4. If any of the following ICS stations are in Hand (Steam Generator/Reactor Demand either Feedwater Demand, Main or Startup Feedwater Valve Demand, Feedpump Speed, Reactor Master, and/or Diamond) runback the appropriate ICS stations.

3.0 FOLLOW-UP ACTION

1. Monitor pressurizer level, R.C. pressure, R.C. temperature, OTSG level, and steam header pressure.

2. Utilize pressurizer heaters and spray control R.C. pressure @2155 PSIG and steam header pressure setpoint @885 PSIG to control T avg. @582°F. Adjust makeup and letdown flows to control pressurizer level @240". Adjust feed flow to control OTSG levels #30".

3. If vacuum is lost in main condenser, verify atmospheric reliefs are controlling header pressure at 885 PSI and reduce reactor power to zero. (Hot Standby) Add required makeup to makeup tank as T avg. is reduced to 532°F.

4. If turbine trip is due to loss of both feed pumps, verify emergency feed pumps started and are delivering water to the OTSG's. Control EF-V11A/B to maintain OTSG levels at 30".

5. If Low Heater Drain Tank Level Alarm is received, STOP Heater Drain Pumps.

6. If possible, vacuum should be maintained until the unit coasts down to approximately 10% of rated speed.

7. OPEN/Insure open the drain valves (MS-V281A/B, MS-V282A/B, MS-V283, MS-V284, MS-V285A/B, and MS-V286A/B) immediately and keep them open until the turbine metal parts and piping are cool.
8. Place Turning Gear Control Switch in Auto and verify engagement of turning gear unless low bearing or low lift oil pressure exists.
9. Manually throttle cooling water thru seal oil coolers (SC-V21 or SC-V25) to maintain seal oil temperature 90^{o}-110^{o}F. Colder oil may seriously damage seal rings and allow a H_2 leak.
10. Notify H.P./Chemistry to sample RC Letdown for Dose Equivalent Iodine between 2 and 6 hours after a power change of greater than 15% power in a one hour period.

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-2.3
STEAM SUPPLY SYSTEM RUPTURE

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8.0	09/27/77	3	33.0			58.0		
9.0	09/27/77	3	34.0			59.0		
10.0	09/27/77	3	35.0			60.0		
11.0	09/27/77	3	36.0			61.0		
12.0	09/27/77	3	37.0			62.0		
13.0	09/27/77	3	38.0			63.0		
14.0	09/27/77	3	39.0			64.0		
15.0			40.0			65.0		
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25.0			50.0			75.0		

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Approval NA Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —

Chairman of PORC

Unit 2 PORC Recommends Approval

R P Warren Date 10/6/78

V-Chairman of PORC

Unit 1 Superintendent Approval

NA Date —

Unit 2 Superintendent Approval

C H Mehlenbacher Date 10/6/78

Manager Generation Quality Assurance Approval NA Date —

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 ABNORMAL PROCEDURE 2203-2.3

Steam Supply System Rupture

1.0 SYMPTOMS

1.1 Rapid decrease of secondary steam pressure. (Both OTSG's start to blow down).

1.2 Electrical load reducing rapidly.

1.3 Decrease in pressurizer level, R.C. Pressure, and cold leg temperature.

1.4 For a rupture inside the Reactor Building; Indication of increasing building pressure and temperature. (Possible high Radioactivity Levels on HP-R-227 if a tube leak exists).

1.5 For a rupture outside the Reactor Building; Noise may be heard in Control Room or a report made from personnel outside the Control Room.

1.6 Decrease in main condenser hotwell level or condensate storage tank level.

2.0 IMMEDIATE ACTION

2.1 Automatic Action

2.1.1 Reactor trips.

2.1.2 Turbine trips

1. Main Throttle valves close.
2. Generator breakers (B2-02 and B2-2602) open.
3. Generator Field breaker opens.

2.1.3 Low steam header pressure (<585 PSIG) on affected OTSG results in Feedwater Latching System closing feedwater valves and main steam isolation valves as follows:

Valves Closed	Affected OTSG A	B
Main Feedwater Control Valve	FW-V30A	FW-V30B
Startup Feedwater Control Valve	FW-V25A	FW-V25B
Main F.W. Block Valves	FW-V17A	FW-V17B
Startup F.W. Block Valves	FW-V19A	FW-V19B
Main Steam Isolation Valves	MS-V4A MS-V7A	MS-V4B MS-V7B

NOTE: Once the feedwater has been stopped to the affected OTSG, the affected OTSG will blow dry.

2.1.4 Pressurizer Heaters on.

2.1.5 Possible High Pressure Injection due to low R.C. Pressure of 1600 psig or 4 psig Reactor Building Pressure.

2.1.6 If rupture was inside Reactor Building, containment isolation will occur if building pressure exceeds 4 psi.

2.2 Manual Action

2.2.1 Manually trip the reactor if not already tripped automatically.

2.2.2 Verify the Turbine has tripped (Main throttle valves closed).

2.2.3 Manually terminate letdown flow by closing MU-V376.

2.2.4 Verify that the Feedwater Latching System has actuated (steam header pressure < 585 psig) and closed the appropriate feedwater and main steam isolation valves for the affected OTSG as listed in step 2.1.3 above.

2.2.5 Determine which OTSG has suffered the rupture. If the leak is upstream of the MS Isolation Valves (MS-V4A & B and MS-V7A & B), the ruptured one will have low pressure (less than 885 psig) and low water level (less than 30"). If the leak is downstream of MS-V4A,B,7A, B, the pressure in both OTSG's

will return to normal as soon as the MS Isolation valves close.

2.2.6 1. Open Block valves (FW-V17A(B) & 19A(B) on unaffected OTSG to re-establish FW flow.

2. (Go to HAND on the Turbine Bypass Valves for the unaffected OTSG and maintain Turbine Header Pressure @ 1010 psig.)

3. Go to manual on FW-V30 & 25's and runback demand to zero on affected OTSG.

2.2.7 If any of the following ICS stations are in Hand (Steam Generator/ Reactor Demand, either Feedwater Demand, Main or Startup Feedwater Valve Demand, Feedpump Speed, Reactor Master, and/or Diamond) runback the appropriate ICS stations.

3.0 FOLLOW-UP ACTIONS

3.1 Select header pressure for the unaffected OTSG to use for indication and control by depressing the TURBINE THROTTLE PRESSURE selector switch for the OTSG (SP-10-MS) on Panel 4.

CAUTION: Auxiliary operators should be warned that high pressure steam leaks can be extremely dangerous and that care should be taken when searching the various compartments for the steam leak.

3.2 If reactor building pressure is not increasing, the control room operator announces that a steam line break has occurred outside the reactor building and then dispatches auxiliary operators to the following locations with instructions to report break locations:

3.2.1 Control Building Area.

3.2.2 Turbine Building.

3.3 Depending on the break location, close the appropriate valves as indicated below to isolate the break.

Break Location	Valves to Close A OTSG	B OTSG
Between turbine stop valves and steam isolation valves	MS-V4A and MS-V7A	MS-V4B and MSV7B
Atmospheric dump valve failure	MS-V1A(1)	MS-V1B(1)
Steam line to emergency feedwater pump	MS-V11A and	MS-V11B
Steam line to turbine bypass valves or to main feedwater pumps or turbine bypass valve failure	MS-V15A	MS-V15B

(1) Manually operated valve.

3.4 If pressurizer level has fallen below 20 inches, reactor coolant system pressure continues to decrease below 1600 psig or flux is rising and the high pressure injection system has not automatically initiated, then "INITIATE" high pressure injection flow from the Borated Water Storage Tank by depressing the Emerg. Core Cooling Manual Actuation Pushbutton (PB1/RCA or PB1/RCB) for actuation A or B on Panel 3.

3.5 If Reactor Building pressure has increased to >4 psig as indicated on the control room Panel 3, verify that all E.S.F. equipment is in its E.S.F. position by observing that all equipment status lights indicate as shown in Table I.

NOTE: Should any component not operate properly, attempt to actuate it at its remote switch in the control room. If it still does not operate, and the component has a local control switch, attempt to actuate the component from its Local Control Station.

3.6 Initiate cooldown per 2102-3.2, "Unit Cooldown", using the unaffected OTSG with the following exception to 2102-3.2.

3.6.1 If affected OTSG is dry, the cooldown rate should be limited so that the maximum temperature difference between the shell temperature of the dry OTSG and the RC temperature in the loop with the dry OTSG is $\leq 100^{\circ}F$ in order to avoid excessive tube and shell temperature differential.

3.6.2 FW valves closed per step 2.1.3 and/or 2.2.5 of this procedure remain closed.

3.6.3 Steam valves closed to isolate steam break per step 3.3 of this procedure remain closed.

3.7 Secure High Pressure Injection when pressurizer level is greater than 100" by defeating the SFAS signals and placing the make-up pump control switches to trip.

NOTE: If pressurizer level drops below 20", manually initiate HP Injection.

3.8 Secure Building Spray Pumps if actuated provided there is no high radioactivity indicated on the reactor building radiation monitoring system HP-R-227 on Panel 12 in the Control Room.

3.9 When pressurizer level has been restored to >100", run reactivity balance to determine shutdown margin by using 2103-1.9 Reactivity Balance Calculation.

TABLE 1

ESF EQUIPMENT - ESF POSITION IN THE CONTROL ROOM PANEL 13

The White light for each component should be lit to indicate that the component is in its ESF position, unless otherwise noted.

ACTUATION A

SAFETY INJECTION GP. 1	
Equipment	ESF Position
DC-P-1A	ON
G2-12 (Note 1)	CLOSED
MU-P-1A (Note 2)	ON
NR-P-1A (Note 3)	ON
NS-P-1A (Note 4)	ON
T1E-2E2	OPEN
T3E-4E2	OPEN
T11E-21E2	OPEN
DC-V96A	CLOSED

SAFETY INJECTION GP. 2	
Equipment	ESF Position
NR-V9A	CLOSED
DH-V4A	OPEN
DH-V5A	OPEN
DH-V8A	OPEN
DH-V100A	CLOSED
DH-V102A	OPEN
MU-P-1B (Note 2)	ON
NR-V40A	OPEN
NS-P-1C	ON
T12-22E-2	OPEN

ACTUATION B

SAFETY INJECTION GP. 1	
Equipment	ESF Position
DC-P-1B	ON
G22-12 (Note 1)	CLOSED
MU-P-1B (Note 2)	ON
NR-P-1C (Note 3)	ON
NS-P-1B (Note 4)	ON
T2E-1E2	OPEN
T4E-3E2	OPEN
T21E-11E2	OPEN
DC-V96B	CLOSED

SAFETY INJECTION GP. 2	
Equipment	ESF Position
NR-V9B	CLOSED
DH-V4B	OPEN
DH-V5B	OPEN
DH-V8B	OPEN
DH-V100B	CLOSED
DH-V102B	OPEN
MU-P-1C	ON
NR-V40B	OPEN
NS-P-1C (Note 4)	ON
T22E-12E-2	OPEN

TABLE 1

ACTUATION A

SAFETY INJECTION GP. 3	
Equipment	ESF Position
MU-V36	CLOSED
MU-V16A	OPEN
MU-V16B	OPEN
OH-P-1A	ON
NR-P-1B (Note 3)	ON
T31E-41E-2	OPEN
NS-V84B	CLOSED
NR-V42A	OPEN

ACTUATION B

SAFETY INJECTION GP. 3	
Equipment	ESF Position
MU-V37	CLOSED
NS-V32	CLOSED
NS-V67	CLOSED
OH-P-1B	ON
MU-V16C	OPEN
MU-V16D	OPEN
NR-P-1D (Note 3)	ON
T41E-31E-2	OPEN
NS-V84A	CLOSED
NR-V42B	OPEN

PANEL 8

Equipment	ESF Position	Status Light Indicator
NS-V83A	OPEN	R
NS-V83B	OPEN	R
NS-V215	CLOSED	G
NS-V216	CLOSED	G

NOTE 1: Diesel Generator Breaker will only be closed if Normal Power is lost; otherwise status indication will be Open (Green).

NOTE 2: MU-P-1B will be running if normal power is available for the Actuation, for the pump that it is selected to backup.
If normal power is lost, MU-P-1B will be running, if the pump that it is selected to backup fails to start or is inoperable.

TABLE 1

NOTE 3: The NR pump in each neader selectad for ES or standby will start if a pump is not operating in that header; otherwise the operating pump will remain in service.

NOTE 4: Normally NS-P-1A and 1B will start; however, NS-P-1C will start if either NS-P-1A or B (depending upon which pump it is selected to backup) fails to start or is inoperable.

TABLE 1

ESF EQUIPMENT - ESF POSITION IN THE CONTROL ROOM PANEL 13

The White light for each component should be lit to indicate that the component is in its ESF position, unless otherwise noted.

ACTUATION A

SAFETY INJECTION GP. 1	
Equipment	ESF Position
DC-P-1A	ON
G2-12 (Note 1)	CLOSED
MU-P-1A (Note 2)	ON
NR-P-1A (Note 3)	ON
NS-P-1A (Note 4)	ON
T1E-2E2	OPEN
T3E-4E2	OPEN
T11E-21E2	OPEN
DC-V96A	CLOSED

SAFETY INJECTION GP. 2	
Equipment	ESF Position
NR-V9A	CLOSED
DH-V4A	OPEN
DH-V5A	OPEN
DH-V8A	OPEN
DH-V100A	CLOSED
DH-V102A	OPEN
MU-P-1B (Note 2)	ON
NR-V40A	OPEN
NS-P-1C	ON
T12-22E-2	OPEN

ACTUATION B

SAFETY INJECTION GP. 1	
Equipment	ESF Position
DC-P-1B	ON
G22-12 (Note 1)	CLOSED
MU-P-1B (Note 2)	ON
NR-P-1C (Note 3)	ON
NS-P-1B (Note 4)	ON
T2E-1E2	OPEN
T4E-3E2	OPEN
T21E-11E2	OPEN
DC-V96B	CLOSED

SAFETY INJECTION GP. 2	
Equipment	ESF Position
NR-V9B	CLOSED
DH-V4B	OPEN
DH-V5B	OPEN
DH-V8B	OPEN
DH-V100B	CLOSED
DH-V102B	OPEN
MU-P-1C	ON
NR-V40B	OPEN
NS-P-1C (Note 4)	ON
T22E-12E-2	OPEN

TABLE 1

ACTUATION A

SAFETY INJECTION GP. 3	
Equipment	ESF Position
MU-V36	CLOSED
MU-V16A	OPEN
MU-V16B	OPEN
DH-P-1A	ON
NR-P-1B (Note 3)	ON
T31E-41E-2	OPEN
NS-V848	CLOSED
NR-V42A	OPEN

R.B. ISOLATION AND COOLING GP. 1	
Equipment	Position
AH-V81	CLOSED
AH-V101	CLOSED
AH-V102	CLOSED
AH-V105	CLOSED
AH-V107	CLOSED
CF-V144	CLOSED
DH-V3	CLOSED
NM-V52	CLOSED
NR-V51A	CLOSED
RR-V2A	OPEN
RR-V5A	OPEN
RR-V5B	OPEN

ACTUATION B

SAFETY INJECTION GP. 3	
Equipment	ESF Position
MU-V37	CLOSED
NS-V32	CLOSED
NS-V67	CLOSED
DH-P-1B	ON
MU-V16C	OPEN
MU-V16D	OPEN
NR-P-1C (Note 3)	ON
T41E-31E-2	OPEN
NS-V84A	CLOSED
NR-V42B	OPEN

R.B. ISOLATION AND COOLING GP 1	
Equipment	Position
AH-E-11D	ON
AH-V80	CLOSED
AH-V103	CLOSED
AH-V104	CLOSED
AH-V106	CLOSED
AH-V108	CLOSED
CF-V115	CLOSED
DH-V2	CLOSED
NM-V104	CLOSED
NR-V51B	CLOSED
RR-V2C	OPEN
RR-V2D	OPEN

TABLE 1

ACTUATION A

R.B. ISOLATION & COOLING GP. 1

Equipment	Position
RR-V5C	OPEN
SV-V55	CLOSED
WDL-V1095	CLOSED
DC-V114	CLOSED

R.B. ISOLATION & COOLING GP. 2

Equipment	Position
AH-E-4A	ON
AH-E-11A	ON
RR-V25C	CLOSED
BS-V1A	OPEN
CA-V10	CLOSED
CA-V4A	CLOSED
CA-V9	CLOSED
RR-P-1B	ON
WDG-V199	CLOSED
WDL-V22	CLOSED
WDL-V1126	CLOSED
AH-04092A & B	
AH-04092D & E	
ED-4098	

ACTUATION B

R.B. ISOLATION & COOLING GP. 1

Equipment	Position
RR-V6C	OPEN
RR-V6D	OPEN
RR-V6E	OPEN
SV-V54	CLOSED
WDL-V1092	CLOSED
DC-V103	CLOSED
DC-V115	CLOSED

R.B. ISOLATION & COOLING GP. 2

Equipment	Position
AH-E-4B	ON
AH-E-11C	ON
RR-V25C	CLOSED
BS-V1B	OPEN
CA-V1	CLOSED
CA-V3	CLOSED
CA-V4B	CLOSED
CA-V8	CLOSED
CA-V6	CLOSED
RR-P-1D	ON
WDG-V2	CLOSED
WDL-V1125	CLOSED
WDL-V271	CLOSED
AH-04092A & B	
AH-04092D & E	
ED-4098	

TABLE 1

ACTUATION A

R.B. ISOLATION & COOLING GP. 3	
Equipment	Position
AH-C-8A	ON
AH-E-11B	ON
AH-P-1A	ON
AH-V1B	CLOSED
AH-V1A	CLOSED
AH-V4A	CLOSED
AH-V5	CLOSED
AH-V60	CLOSED
AH-V62	CLOSED
AH-V72	CLOSED
IC-V2	CLOSED
IC-V5	CLOSED
MU-V2A	CLOSED
MU-V2B	CLOSED
MU-V377	CLOSED
NS-V72	CLOSED
NS-V81	CLOSED
NR-P-2A	ON
NR-V144A	CLOSED
RR-P-1A	ON
RR-V25A	CLOSED
RR-V25B	CLOSED
IC-P-1A	OFF
AH-V48	CLOSED
* BS-P-1A	ON

ACTUATION B

R.B. ISOLATION & COOLING GP. 3	
Equipment	Position
AH-C-8B	ON
AH-V171	CLOSED
IC-P-1B	OFF
RR-V25D	CLOSED
RR-V25E	CLOSED
IC-V3	CLOSED
AH-E-11E	ON
IC-V4	CLOSED
AH-P-1B	ON
AH-V2A	CLOSED
MU-V376	CLOSED
AH-V2B	CLOSED
MU-V18	CLOSED
AH-V3A	CLOSED
MU-V25	CLOSED
AH-V3B	CLOSED
NS-V100	CLOSED
AH-V6	CLOSED
NR-P-2B	ON
AH-V61	CLOSED
NR-V144B	CLOSED
AH-V63	CLOSED
RR-P-1C	ON
* BS-P-1B	ON

* If RB Pressure >30 psig.

NOTE 1: Diesel Generator Breaker will only be closed if Normal Power is lost; otherwise status indication will be Open (Green).

NOTE 2: MU-P-1B will be running if normal power is available for the Actuation, for the pump that it is selected to backup.
If normal power is lost, MU-P-1B will be running, if the pump that it is selected to backup fails to start or is inoperable.

NOTE 3: The NR pump in each header selected for ES or standby will start if a pump is not operating in that header; otherwise the operating pump will remain in service.

NOTE 4: Normally NS-P-1A and 1B will start; however, NS-P-1C will start if either NS-P-1A or B (depending upon which pump it is selected to backup) fails to start or is inoperable.

PANEL 8

Equipment	ES Position	Indication
DH-V7A(1)	Close	G
DH-V7B(1)	Close	G
NS-V83A	Open	R
NS-V83B	Open	R
NS-V215	Close	G
NS-V216	Close	G
CF-V1A	Open	R
CF-V1B	Open	R

(1) This valve may have to be opened for "piggy-back" operation. Once, opened, the Position/Indication becomes Open/R.

PANEL 15

Equipment	ES Position	Indication
DH-V6A(1)	Close	G
DH-V6B(1)	Close	G
MU-V37B(2)	Open	R
MS-V4A(3)	Open	R
MS-V4B(3)	Open	R
MS-7A(3)	Open	R
MS-V7B(3)	Open	R

(1) This valve must be opened for sump-switchover. Once opened, the Position/Indication becomes OPEN/R.

(2) This valve should be closed at the operator's first chance. Once closed, the Position/Indication becomes CLOSE/W.

(3) These valves should be closed when the steam system is secured. Once closed, the Position/Indication becomes CLOSE/G.

PANEL 25

Equipment	ES Position	Indication
AH-E12A	Off	G
AH-E12B	Off	G
AH-E19A	Off	G
AH-E19B	Off	G

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Revision 1
06/20/77

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-2.4
SECONDARY SERVICES CLOSED COOLING WATER FAILURE

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Unit 1 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —
Chairman of PORC

PORC comments of — included
(date)

By — Date —

Unit 2 PORC Recommends Approval

J.A. Mackey Jr. Date 6-20-77
Chairman of PORC

PORC comments of — included
(date)

By — Date —

Approval [signature] Date 6/20/77
Station Superintendent/
Unit Superintendent

TMI-68-A 11-74

THREE MILE ISLAND NUCLEAR STATION

UNIT #2 ABNORMAL PROCEDURE 2203-2.4

SECONDARY SERVICES CLOSED COOLING WATER FAILURE

1.0 SYMPTOMS

1.1 Secondary Services Closed Cooling Water Pumps Low Suction Alarm at at 8 psig (decreasing) - Alarm 8.A8.

1.2 Secondary Services Closed Cooling Water Pump Trip - Alarm 8.D8.

1.3 Secondary Services Closed Cooling Water Head Tank High/Low Level Alarm at 76 inches (high) and 25 inches (low) - Alarm 8.B8.

1.4 Secondary Services Closed Cooling Water Differential Pressure High/Low Alarm at 51 psi (increasing) or 34 psi (decreasing) - Alarm 8.C8.

1.5 High temperatures from computer for components served by SSCCW.

1.6 Turbine trip on loss of secondary service closed cooling water - Alarm 8.E3.

2.0 IMMEDIATE ACTION

2.1 Automatic Action

2.1.1 Standby Secondary Services Closed Cooling Pump will start automatically if a running pump trips.

2.1.2 Secondary Services Closed Cooling Water Head Tank level control will modulate valve DW-V6 to maintain proper level in the head tank.

2.1.3 Differential pressure regulating valve SC-V129 will modulate to maintain proper system differential pressure.

2.1.4 Main Turbine - Generator will trip upon loss of all three Secondary Closed Cooling Water Pumps.

2.1.5 Service Air Compressors will stop upon reaching high discharge air temperature (475°).

2.2 Manual Action

2.2.1 Verify that the standby Secondary Services Closed Cooling Pump has started if an operating pump has tripped. If not, manually start the standby pump.

2.2.2 If it is impossible to establish normal Secondary Services Cooling Water Flow shutdown the unit as follows:

2.2.2.1 Reduce load at maximum rate, to 15% reactor power or as required by Generator, Exciter or Isolated Phase Bus Duct Temperature manually TRIP the turbine.

2.2.2.2 Verify that reactor power is in a run back at 20% per minute.

3.0 FOLLOWUP ACTION

3.1 If the Secondary Services Cooling Water Head Tank alarms low (25 inches), verify that makeup valve DW-V6 is open to increase level. If not, open manual bypass valve DW-V9.

3.2 Monitor pump bearing temperatures on the operating condensate/condensate booster pump and main feed pump. Monitor the following computer points on the CRT display for the operating pumps.

CO-P1A	1521	(max = 210°F)
CO-P1B	1525	(max = 210°F)
CO-P1C	1529	(max = 210°F)
CO-P2A	1538, 1539	(max = 210°F)
CO-P2B	1545, 1546	(max = 210°F)
CO-P2C	1552, 1553	(max = 210°F)

FW-P1A	1635	(max = 210°F)
FW-P1B	1644	(max = 210°F)

3.3 If the measured temperatures exceed the maximum allowable, trip the reactor, and all the pumps in the operating feed string.

3.4 If one normal feed string can be maintained, ensure the ICS controls steam generator level at the low level limit when at 15% power.

3.5 Attempt to isolate any leakage and/or, while filling the SSCC Surge tank, isolate portions of the system to keep continued supply of SSCC water to operating components of the secondary plant.

3.6 Start additional SSCC pumps and valve in additional coolers as needed to maintain adequate cooling to all components in service.

3.7 Place the standby SSCC water heat exchanger in service for continued hi SSCC water heat exchanger outlet temp and backwash the heat exchanger.

3.8 Shutdown in accordance with 2102-3.1 Unit Shutdown.

C

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Revision 2
02/24/77

Central File

THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-2.5
CONTROL ROOM HVAC FAILURE

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Unit 1 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval NA Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

NA Date —
Chairman of PORC

PORC comments of — included
(date)

By — Date —

Unit 2 PORC Recommends Approval

Norman W. Lewis Date 2-23-77
Chairman of PORC

PORC comments of — included
(date)

By — Date —

Approval [redacted] Date 2-24-77
Unit Superintendent

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THREE MILE ISLAND NUCLEAR STATION

UNIT #2 ABNORMAL PROCEDURE 2203-2.5

Control Room HVAC Failure

2.5.1 High Radiation

1.0 SYMPTOMS:

1.1 RMS trouble alarm - Panel No. 12.

1.2 Control Room Intake Duct Radiation Monitos, HP-R-220 indicates high radiation - Panel No. 12.

2.0 IMMEDIATE ACTION:

2.1 Automatic

2.1.1 One Control Room Bypass Supply fan, AH-E-4A or 4B starts.

2.1.2 Control Room Ventilation System is shifted to minimum outside air recirculation.

2.2 Manual

2.2.1 Verify AH-E-4A or 4B starts and the associated flow light is energized.

2.2.2 Verify the following damper positions:

AH-D4092A Closed

AH-D4092B Opens

AH-D4092D Closed

AH-D4092E Open

AH-ED4098 Closed

3.0 FOLLOW-UP ACTION

3.1 Start one chiller unit AH-C-8A or 8B per 2104-5.4.

3.2 Attempt to locate and reduce radiation source level. Keep the system in the minimum outside air recirculation mode until radiation levels are reduced and HP-R-220 can be reset and ventilation system returned to normal per 2104-5.4.

3.3 If radiation levels become high enough to force evacuation of the Control Room refer to 2202-2.2 Shutdown Outside the Control Room.

3.4 Restore the system to normal per 2104-5.4 when possible.

2.5.2 Air Intake Tunnel Alarms

1.0 SYMPTOMS:

One or more of the following alarms.

1.1 Air Intake Tunnel Chlorine Monitor AH-ClS-548 Chlorine Danger Alarm (25.C8).

1.2 Air Intake Tunnel Vertical Shaft Fire (7B.G1).

1.3 Air Intake Tunnel North of Divider Fire (7B.G2).

1.4 Air Intake Tunnel South of Divider Fire (7B.G3).

1.5 Air Intake Tunnel Combustible Hydrocarbons (7B.H1).

1.6 Control Building Air Duct and Chlorine Monitor Chlorine Danger (25.C7).

2.0 IMMEDIATE ACTION:

2.1 Automatic

2.1.1 High Chlorine Gas - One Control Room Bypass Supply Fans, AH-E-4A or 4B, starts and the Ventilating System shifts to 100% Recirculation.

2.1.2 Fire - Control Room Supply Damper AH-04092C closed and the Ventilating System shifts to 100% Recirculation.

2.1.3 High Combustible Hydrocarbons - None.

2.2 Manual Action:

2.2.1 High Temperature - Shift the Control Room Ventilating System to recirculation and start a chiller unit per 2104-5.4.

2.2.2 High Chlorine Gas

2.2.2.1 Verify a Control Room Bypass Supply fan has started.

2.2.2.2 Verify the following dampers positions:

D4092A Closed

D4092B Open

D4092C Closed

D4092D Closed

D4092E Open

EP4098 Closed

2.2.3 Fire - Verify the following damper positions:

D4092A Closed

D4092B Open

D4092C Closed

D4092D Closed

D4092E Open

ED4098 Closed

2.2.4 High Combustible Hydrocarbons - Start a control building bypass supply fan AN-E-4A or 4B. The ventilating system will automatically shift to the minimum outside air recirculation mode.

3.0 FOLLOW-UP ACTION

3.1 If there is a Fire or Combustible Vapors in the Air Intake Tunnel refer to 2202-3.1, Fire.

3.2 If chlorine gas exists in the Air Intake Tunnel find and isolate the source of the gas.

CAUTION: Chlorine Gas is poisonous. Protective clothing and breathing apparatus must be worn before approaching the source of the leak.

3.3 If there is high temperature in the Air Intake Tunnel determine the cause and correct.

3.4 Start a chiller unit per 2104-5.4 if one is not already operating.

3.5 As soon as the trouble is cleared return the system to normal per 2104-5.4.

2.5.3 Engineered Safety Features

1.0 SYMPTOMS:

1.1 SFAS Channel Trip Alarm associated with Reactor Building Isolation and Cooling. (4 psig RB Pressure).

2.0 IMMEDIATE ACTION

2.1 Automatic

2.1.1 One Control Room Bypass Supply fan AH-E-4A or 4B will start.

2.1.2 The Control Room ventilating system will shift to the minimum outside air recirculation mode.

2.1.3 Both Control Building River Water Booster Pumps, NR-P-2A and 2B, both Control Building Chilled Water Pumps, AH-P-1A and 1B, and a Chiller Unit AH-C-8A or 8B start.

2.2 Manual

2.2.1 Verify one Control Room Bypass Supply Fan starts.

2.2.2 Verify the following damper positions:

D4092A Closed

D4092B Open

D4092C Open

D4092D Closed

D4092E Open

ED4098 Closed

2.2.3 Verify the following:

Both Control Building RW Booster pumps,

NR-P-2A and 2B, start

Both Control Building Chilled Water pumps,

AH-P-1A and 1B, start

One Control Building Liquid Chiller Unit,

AH-C-8A or 8B, starts.

3.0 FOLLOW-UP ACTION

3.1 Refer to 2202-1.3 Loss of RC/Loss of RC Pressure.

2.5.4 Station Blackout

1.0 SYMPTOMS:

1.1 Loss of all station power.

2.0 IMMEDIATE ACTION

2.1 Automatic

2.1.1 All Control Room fans and the Control Building Chiller Unit trip.

2.1.2 After power is restored by the Diesel Generators, the Control Room fan coil unit and exhaust fans restarts.

2.2 Manual

2.2.1 Refer to 2202-2.1 Site Blackout.

2.2.2 Restart a Control Building Chiller Unit if required per 2104-5.4.

2.5.5 Control Room HVAC Failure

1.0 SYMPTOMS:

1.1 Control Room Fan Coil Unit trip.

1.2 Control Room Bypass Supply fan trip.

1.3 Control Room Exhaust fan trip.

1.4 Control Building Liquid Chiller Unit trip.

2.0 IMMEDIATE ACTION

2.1 Automatic

2.1.1 If a control room fan coil unit trips the standby fan will start.

2.1.2 If a control room bypass supply fan trips the standby fan will start.

2.1.3 If the control room exhaust fan trips the exhaust fan bypass damper will open.

2.2 Manual

2.2.1 If a control room fan coil unit trips verify the standby fan starts.

2.2.2 If a control room bypass supply fan trips verify the standby fan starts.

2.2.3 If the control room exhaust fan trips verify the exhaust fan bypass damper opens.

2.2.4 If a control building chiller unit trips start the standby unit per 2104-5.4.

3.0 FOLLOW-UP ACTION

3.1 If a control room fan coil unit trips and the standby fan fails to start automatically,

3.1.1 Attempt to start the standby fan manually.

3.1.2 If the standby fan cannot be started manually open dampers D4096A and B and start a control room bypass supply fan, AH-E-4A or 4B.

3.1.3 Determine and correct the cause of the trip and return the ventilation system to normal as soon as possible.

3.2 If a control room bypass supply fan trips and the standby fan fails to start:

3.2.1 Attempt to start the standby fan manually.

3.2.2 If the standby fan cannot be started shift the Control Room Ventilating System to 100% recirculation by closing AH-D4092C. Verify the following damper positions;

D4092A	Open
D4092B	Closed
D4092C	Closed
D4092D	Closed
D4092E	Open
ED4098	Closed

3.2.3 Determine and correct the cause of the trip and return the ventilating system to normal as soon as possible.

3.3 If the control room exhaust fan trips and the exhaust fan bypass damper fails to open.

3.3.1 Manually open the exhaust fan bypass damper AH-D4092G.

3.3.2 Determine and correct the cause of the trip.

3.4 If the standby control building liquid chiller unit cannot be started.

3.4.1 Shift the control room ventilation system to 100% outside air operation per 2104-5.4.

3.4.2 Provide additional means for cooling equipment as necessary (portable fans, etc.)

3.4.3 Determine and correct the cause of the trip and start a chiller unit as soon as possible.

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THREE MILE ISLAND NUCLEAR STATION
UNIT #2 ABNORMAL PROCEDURE 2203-2.6

POST ACCIDENT HYDROGEN CONTROL

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Unit 1 Staff Recommends Approval

Approval _____ Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval _____ Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

_____ Date —
Chairman of PORC

Unit 2 PORC Recommends Approval

_____ Date 6.22.78
V. Chairman of PORC

Unit 1 Superintendent Approval

_____ Date —

Unit 2 Superintendent Approval

_____ Date 6/22/78

Manager Generation Quality Assurance Approval _____ Date —

TMI 55-A Rev 3/77

THREE MILE ISLAND NUCLEAR STATION

UNIT #2 ABNORMAL PROCEDURE 2203-2.6

POST ACCIDENT HYDROGEN CONTROL

1.0 SYMPTOMS

1.1 Samples drawn and analyzed per HP and Chemistry Procedures show an increasing concentration of hydrogen gas in the Reactor Building.

2.0 IMMEDIATE ACTION

2.1 Automatic Action

None

2.2 Manual Action

2.2.1 Following an accident, samples for hydrogen should be taken daily after the containment pressure reduces to normal and the activity level is reduced. The sampling frequency may be adjusted as required after the concentration rate of increase is established. The samples are taken per HP 1630 and analyzed per Chemistry Procedure 1958.2 and 1958.3.

2.2.2 Sampling for I-131 and Kr-85 insofar as hydrogen purging is concerned is done to determine which isotope, I-131 or Kr-85, controls the purge rate and the corresponding allowable I-131 or Kr-85 release rate.

3.0 FOLLOW-UP ACTION

3.1 As the hydrogen concentration increases, place the Hydrogen Recombiner into operation per 2104-6.4. A spare H_2 recombiner is located in the Unit 2 south warehouse and may be placed in operation per 2104-6.4 section 4.4.

3.2 If the hydrogen concentration cannot be maintained lower than 3.5%, initiate purging per 2104-6.5.

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1670.1
Revision 6
02/13/78

THREE MILE ISLAND NUCLEAR STATION
STATION RADIATION EMERGENCY PROCEDURE 1670 CENTRAL FILE
LOCAL EMERGENCY PROCEDURE

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Approval _____ Date —
Cognizant Dept Head

Unit 2 Staff Recommends Approval

Approval _____ Date —
Cognizant Dept Head

Unit 1 PORC Recommends Approval

_____ Date 2/10/78
Chairman of PORC

Unit 2 PORC Recommends Approval

_____ Date 2.13.78
Chairman of PORC

Unit 1 Superintendent Approval

_____ Date 2/15/78

Unit 2 Superintendent Approval

_____ Date 2/13/

NA

Manager Generation Quality Assurance Approval _____ Date —

TMI 55-A Rev 8/77

THREE MILE ISLAND NUCLEAR STATION

STATION RADIATION EMERGENCY PROCEDURE 1670.1

LOCAL EMERGENCY PROCEDURE

1.0 DISCUSSION

A Local (or Personnel) Emergency involves an occurrence which could effect the safety of personnel or equipment with an off-site hazard unlikely. This incident may require evacuation of one or more buildings on site. Once a report of an emergency has been received in the Control Room, the Shift Supervisor/Shift Foreman is responsible for determining the extent of the emergency.

2.0 OBJECTIVE

To outline action required in the event that a Local Emergency is declared.

3.0 EMERGENCY ACTION LEVELS

3.1 One or more radiation monitors in a single building reach their high alarm setpoint. More than 1 radiation monitor reaches the low alarm setpoint.

3.2 Report of a significant increase in the level of radiation or airborne activity in a work area.

3.3 Report of a radioactive spill* in a work area (at least 2 mr/hr @ ½" or 10^{-2} uCi/cc over 25 square feet).

4.0 IMMEDIATE ACTION

4.1 Emergency Director (Shift Supervisor/Foreman)

4.1.1 Report to Control Room.

4.1.2 Evaluate emergency conditions and directs appropriate announcement over paging system.

4.1.3 Notify the Radiation Protection Supervisor/Foreman of the incident. A Rad/Chem Tech or Auxiliary Operator will be sent to the affected area to perform a survey and take the action outlined in Health Physics Procedure 1695, "Occurrence of Unsuspected High Radiation Levels", or Health Physics Procedure 1681 (1681.2 Unit 2), "Control of Contaminated Spills".

4.1.4 Evaluate the results of the radiation survey.

a. If the results of the survey show radiation contamination levels to be normal, permit routine entry to the area. Announce over the paging system that the Local Emergency no longer exists and that the area is cleared for routine entry.

b. If the results of the radiation survey show radiation levels above normal in the local area, ensure that the affected area is roped off and approximately marked as to specific radiological conditions.

4.1.5 Determine the cause of high radiation levels and take appropriate steps to return condition to normal.

4.2 Operations Personnel (Control Room Operators)

4.2.1 Notify the Shift Supervisor/Shift Foreman.

4.2.2 Make an appropriate announcement over the paging system, if directed by the Shift Supervisor/Shift Foreman, that all personnel should evacuate the affected area and assemble at the ECS for monitoring.

4.2.3 As directed by the Shift Supervisor/Shift Foreman, take appropriate operator action to limit, contain, or correct the condition.

4.3 Radiation Monitoring Teams (Rad Chem Technicians and Auxiliary Operators "A")

4.3.1 Perform radiation survey personnel monitoring and decontamination as necessary.

4.4 Personnel in the Area

4.4.1 Notify the Control Room of the conditions in the area and suspected cause of the emergency, and implement the Medical Emergency Procedure as necessary.

4.4.2 If possible, take steps to isolate and/or confine the source of radiation without being subject to unnecessary Radiation exposure.

4.4.3 Leave the area if directed to do so by the Radiation Protection Personnel, Shift Supervisor or Auxiliary Operator. Take appropriate steps while leaving the area to prevent and minimize the spread of Radioactive Contamination. Leave the area and report to the ECS.

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1670.2
Revision 9
11/22/78

THREE MILE ISLAND NUCLEAR STATION
STATION RADIATION EMERGENCY PROCEDURE 1670.2
SITE EMERGENCY PROCEDURE

CENTRAL FILE

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Unit 1 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —
Cognizant Dept. Head

Unit 1 PORC Recommends Approval

[signature] Date 11/16/78
Chairman of PORC

Unit 2 PORC Recommends Approval

R P Warren Date 11/20/78
V- Chairman of PORC

Unit 1 Superintendent Approval

[signature] Date 11/22/78

Unit 2 Superintendent Approval

[signature] Date 11/20/78

Manager Generation Quality Assurance Approval NA Date —

TMI 1S-A Rev 3/77

THREE MILE ISLAND NUCLEAR STATION

STATION RADIATION EMERGENCY PROCEDURE 1670.2

SITE EMERGENCY PROCEDURE

1.0 DISCUSSION

A Site Emergency involves an occurrence of a major accident which could result in an uncontrolled release of radioactive material to the immediate environment. Such an emergency may require evacuation of non-essential personnel. This emergency is potential off-site hazard which could result in an off-site radiological dose.

2.0 OBJECTIVE

To outline action required in the event that a Site Emergency is declared.

3.0 EMERGENCY ACTION LEVELS

3.1 Any Vent Gas Monitor (RMA 8 or 9 or HP-R-219) is Off Scale or 100 times High Alarm Set Point.

3.2 The radiation level at the site security fence is 125 mR/hr.

3.3 Loss of primary coolant pressure coincident with a high Reactor Building Pressure and/or high Reactor Building Sump Level.

3.4 Reactor Building evacuation alarm.

3.5 Unexpected Reactor Building high range gamma monitor Alert Alarm.

3.6 The high alarm of the Radiation Monitoring System Area Monitors in two separate buildings due to a single event.

4.0 IMMEDIATE ACTION

NOTE: Designated alternates for essential personnel are on Figure 5.

4.1 Emergency Director (Station Superintendent)

4.3 Radiation Monitoring Teams (Rad Chem Technicians and Auxiliary Operators "A")

4.3.1 Perform radiation survey personnel monitoring and decontamination as necessary.

4.4 Personnel in the Area

4.4.1 Notify the Control Room of the conditions in the area and suspected cause of the emergency, and implement the Medical Emergency Procedure as necessary.

4.4.2 If possible, take steps to isolate and/or confine the source of radiation without being subject to unnecessary Radiation exposure.

4.4.3 Leave the area if directed to do so by the Radiation Protection Personnel, Shift Supervisor or Auxiliary Operator. Take appropriate steps while leaving the area to prevent and minimize the spread of Radioactive Contamination. Leave the area and report to the ECS.

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Revision 9
11/22/78

THREE MILE ISLAND NUCLEAR STATION
STATION RADIATION EMERGENCY PROCEDURE 1670.2
SITE EMERGENCY PROCEDURE

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R P Warren Date 11/20/78

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Manager Generation Quality Assurance Approval N/A Date —

TMI 55-A Rev 3/77

THREE MILE ISLAND NUCLEAR STATION

STATION RADIATION EMERGENCY PROCEDURE 1670.2

SITE EMERGENCY PROCEDURE

1.0 DISCUSSION

A Site Emergency involves an occurrence of a major accident which could result in an uncontrolled release of radioactive material to the immediate environment. Such an emergency may require evacuation of non-essential personnel. This emergency is potential off-site hazard which could result in an off-site radiological dose.

2.0 OBJECTIVE

To outline action required in the event that a Site Emergency is declared.

3.0 EMERGENCY ACTION LEVELS

3.1 Any Vent Gas Monitor (RMA 8 or 9 or HP-R-219) is Off Scale or 100 times High Alarm Set Point.

3.2 The radiation level at the site security fence is 125 mR/hr.

3.3 Loss of primary coolant pressure coincident with a high Reactor Building Pressure and/or high Reactor Building Sump Level.

3.4 Reactor Building evacuation alarm.

3.5 Unexpected Reactor Building high range gamma monitor Alert Alarm

3.6 The high alarm of the Radiation Monitoring System Area Monitors in two separate buildings due to a single event.

4.0 IMMEDIATE ACTION

NOTE: Designated alternates for essential personnel are on Figure 5.

4.1 Emergency Director (Station Superintendent)

4.1.1 Report to the affected Unit's Control Room and supervise implementation of Site Emergency Procedures.

NOTE: If both Unit Superintendents are on-site they will report to their respective Unit's Control Room.

4.1.2 Assess the conditions and direct protective and corrective actions.

4.1.3 Direct an appropriate announcement over the paging system (ensure page is in merge to reach both Units).

4.1.4 Contact the Civil Defense Duty Officer as soon as possible (within 5 minutes if practical) after determination of the emergency. Establish communications with off-site support groups as necessary. See Figure 4.

4.1.5 Order assembly and accountability in accordance with 1670.7.

4.1.6 Establish communications with ECS and unaffected Unit's Control Room. (A suggested communications scheme is presented in Figure 1).

4.1.7 Evaluate RMS and meterological data.

4.1.8 Maintain status board in the Control Room.

4.1.9 Dispatch on-site and off-site Radiation Monitoring Teams downwind for surveys in accordance with 1670.5 and 1670.6.

4.1.10 As necessary, initiate notification of Met-Ed personnel in accordance with AP 1014, Recall of Standby Personnel and 1670.14, Emergency Contact List.

4.1.11 Direct evaluation off-site monitoring team data and project off-site dose consequences in accordance with 1670.4.

4.1.12 Evaluate the emergency's impact on the unaffected Unit and determine the need to change its operational status, and take appropriate action.

4.1.13 Institute site evacuation of all non-essential personnel if the level of radioactive gas in the station vent exceeds its high alarm setpoint and two area radiation monitors exceed 10 R/hr and/or dose rates at security fence exceed 125 mR/hr. Evacuation may also take place at the discretion of the Station/Unit Superintendent. Evacuation will be carried out as per Radiation Emergency Procedure 1670.7.

4.2 Accident Assessment Personnel

4.2.1 Shift Supervisor

4.2.1.1 The Shift Supervisor will also be required to assume the appropriate duties, depending on the emergency of the following people:

a. Supervisor of Operations
b. Supervisor of Maintenance
c. Supervisor of Radiation Protection and Chemistry/Radiation Protection Supervisor

4.2.1.2 Report to the affected Unit's Control Room.

4.2.1.3 Ensure that if conditions warrant, the site emergency has been announced over the paging system and the Radiation Emergency Alarm has been sounded. Also, ensure that all non-essential personnel have been directed to report to their assembly areas.

4.2.1.4 Assume key duties of Station Superintendent/Unit Superintendent until properly relieved by him or the Duty Section Head.

4.2.1.5 Direct Auxiliary Operators as required, to assume the duties of the following:

a. Radiation Monitoring Team

b. Emergency Repair Party

4.2.2 Shift Foreman (Affected Unit)

4.2.2.1 Implement the Emergency Procedures (1200 Series-Unit 1 or 2200-Unit 2).

4.2.2.2 Assist the Station Superintendent/Unit Superintendent/Shift Supervisor.

4.2.3 Supervisor of Operations (Affected Unit)

4.2.3.1 Assist Station Superintendent/Unit Superintendent in the supervision and implementation of the Emergency Procedures.

4.2.3.2 Serve as back-up to Station Superintendent/Unit Superintendent.

4.2.3.3 Establish communications with unaffected Unit's Control Room.

4.2.3.4 Provide technical assistance and advice as necessary.

4.2.4 Unaffected Unit - Supervisor of Operations, Shift Foreman, Control Room Operators

4.2.4.1 Maintain Unit's status unless otherwise directed by Station Superintendent/Unit Superintendents/Shift Supervisor.

4.2.4.2 Establish communications with affected Unit's Control Room via Supervisor of Operations.

4.2.5 Nuclear Engineer

4.2.5.1 Provide technical assistance as needed in the Control Room.

4.2.5.2 Check and record readings on the Radiation Monitoring System and relay the readings to the Radiation Protection Supervisor in the Emergency Control Station. Also relay data of plant status.

4.2.5.3 Assist as directed by the Station Superintendent/Unit Superintendent to determine condition of the Unit.

4.2.6 Supervisor of Radiation Protection and Chemistry

4.2.6.1 Report to Affected Unit's Control Room.

4.2.6.2 Evaluate RMS and meterological data as well as survey data to project off-site radiological doses in accordance with 1370.4.

4.2.6.3 Provide recommendations and status reports to the Station Superintendent/ Unit Superintendent based on dose projections and information from the Radiation Protection Supervisor.

4.2.6.4 Insure dose rates and release rates and other pertinent information are relayed to off-site agencies as necessary.

4.2.6.5 Provide direction to the Radiation Protection Supervisor at the ECS for off-site monitoring team locations and monitoring.

4.2.7 Radiation Protection Supervisor

4.2.7.1 Report to the Emergency Control Station.

4.2.7.2 Establish communications with the affected Unit's Control Room. (See Figure 1)

4.2.7.3 Direct the muster of personnel at the ECS and forward lists to the North Assembly Area.

4.2.7.4 Maintain Emergency Status Board and Emergency Job Board at ECS (See Figure 2 and 3).

4.2.7.5 Assign personnel to Radiation Monitoring Team.

4.2.7.6 Dispatch on-site and off-site Radiation Monitoring Teams in downwind direction as directed by Control Room.

4.2.7.7 Establish and maintain communications with on-site and off-site teams.

4.2.7.8 Establish personnel monitoring for all personnel arriving at the ECS from other areas within the plant.

4.2.7.9 Authorize ingress and egress to Three Mile Island.

4.2.8 Chemistry Supervisor

4.2.8.1 Report to the Emergency Control Station.

4.2.8.2 Supervise performance of chemical and radio-chemical analysis as required.

4.2.8.3 Provide assistance to the Radiation Protection Supervisor as necessary.

4.2.9 Lead Engineers - Electrical, Mechanical, I&C - of Affected Unit

4.2.9.1 Report to Affected Unit's Control Room.

4.2.9.2 Provide technical support and assistance as required.

4.2.10 Lead Engineers - Electrical, Mechanical, I&C - of Unaffected Unit

4.2.10.1 Report to Emergency Control Station.

4.2.10.2 Provide technical support and assistance as required.

4.3 Radiation Monitoring Teams (RMT)

4.3.1 Provide Radiation Monitoring of areas as directed by the Radiation Protection Supervisor in accordance with HPP 1670.5 and 1670.6.

4.4 Repair Party Team

4.4.1 Supervisor of Maintenance

4.4.1.1 Report to the Emergency Control Station.

4.4.1.2 Determine if emergency repairs are required.

4.4.1.3 Supervise actions of the Emergency Repair Party.

4.4.2 Emergency Repair Party (ERP)

4.4.2.1 Report to the Emergency Control Station.

4.4.2.2 Perform emergency repairs as directed by the Supervisor of Maintenance.

4.4.2.3 Specific duties of the ERP will depend upon the nature of the emergency. However, the goal of the ERP is to correct any malfunction and return the plant to normal operating condition as soon as possible.

4.5 Operations Personnel

4.5.1 Control Room Operators (Affected Unit)

4.5.1.1 Notify the Shift Supervisor/Foreman that a Site Emergency exists.

4.5.1.2 As directed by the Shift Supervisor/Foreman, sound the Radiation Emergency Alarm.

4.5.1.3 On orders of the Shift Supervisor/Foreman, announce the following applicable message over the GAI-Tronics paging system in both units.

a. If Emergency is a drill:

"ATTENTION ALL PERSONNEL, THIS IS A DRILL, THIS IS A DRILL. A RADIATION EMERGENCY IN UNIT _____ HAS BEEN DECLARED. ALL NON-ESSENTIAL PERSONNEL IN UNIT #1 REPORT TO THE NORTH ASSEMBLY AREA IN THE SERVICE BUILDING AUDITORIUM. ALL NON-ESSENTIAL PERSONNEL IN UNIT #2 REPORT TO ASSEMBLY AREA AT THE WEST END OF THE UNIT #2 TURBINE BUILDING, 305' ELEVATION. ALL MEMBERS OF THE RADIATION EMERGENCY TEAM REPORT TO THEIR STATIONS. THERE WILL BE NO SMOKING, DRINKING OR EATING UNTIL FURTHER NOTICE. THIS IS A DRILL.

b. If Emergency is not a drill:

"ATTENTION ALL PERSONNEL, ATTENTION ALL PERSONNEL. A RADIATION EMERGENCY IN UNIT _____ HAS BEEN DECLARED. ALL NON-ESSENTIAL PERSONNEL IN UNIT #1 REPORT TO THE NORTH HOLDING AREA IN THE SERVICE BUILDING AUDITORIUM. ALL NON-ESSENTIAL PERSONNEL IN UNIT #2 REPORT TO ASSEMBLY AREA AT THE WEST END OF UNIT #2 TURBINE BUILDING, 305' ELEVATION. ALL MEMBERS OF THE RADIATION EMERGENCY TEAMS REPORT TO THEIR STATIONS. THERE WILL BE NO SMOKING, DRINKING OR EATING UNTIL FURTHER NOTICE".

Repeat above message.

4.5.1.4 Take appropriate operating action in accordance with Emergency Procedures (1200 or 2200 series).

4.5.1.5 Turn on the Met-Ed system short wave radio transmitter and establish contact with the Lebanon Office Dispatcher to ensure that back-up means of communication are available.

4.6 Security Personnel

4.6.1 North and South Gate and Service Building entrance guards remain on post.

4.6.2 Security Specialist/Security Sergeant reports to North Assembly Area.

4.6.3 Direct accountability in accordance with 1670.7.

4.6.4 Maintain site security in accordance with Security Procedures.

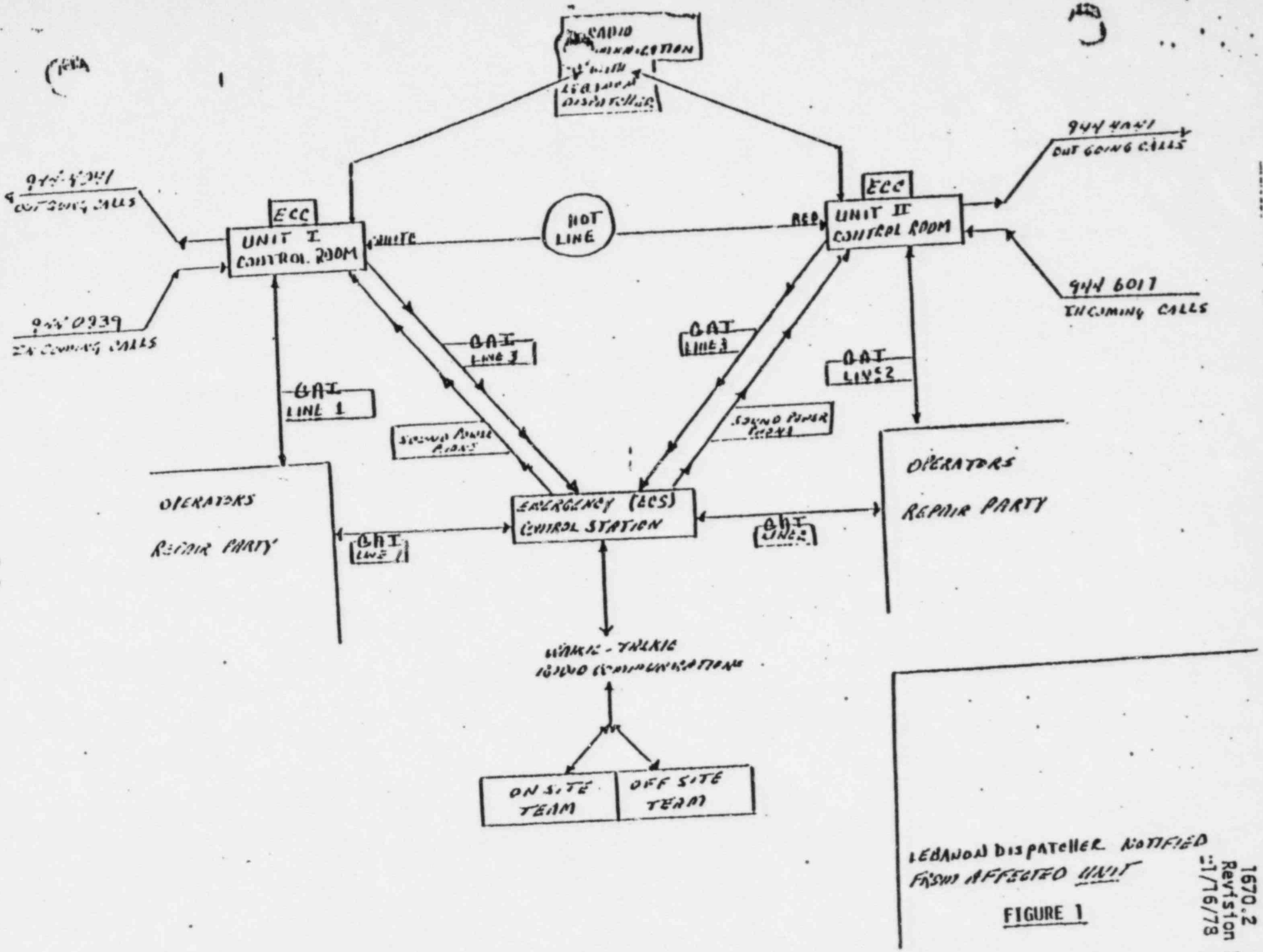


FIGURE 1

1670.2
Revision 7
2/16/78

EMERGENCY STATUS BOARD

PENNA. BUREAU OF RAD. HEALTH QUESTIONS

1. WHAT TYPE ACCIDENT HAS OCCURRED?
2. HAS THE REACTOR TRIPPED?
3. DID THE EMERGENCY SAFEGUARDS SYSTEMS ACTUATE?
   IF SO, WHICH ONES?
   a. HIGH PRESSURE INJECTION — YES ___ NO ___
   b. LOW PRESSURE INJECTION — YES ___ NO ___
   c. CORE FLOOD — YES ___ NO ___
   d. REACTOR BUILDING ISOLATION — YES ___ NO ___
   e. REACTOR BUILDING COOLING — YES ___ NO ___
4. WHAT IS THE STATUS OF THE PLANT?
   a. AT POWER
   b. HOT STANDBY
   c. HOT SHUTDOWN
   d. COOLING DOWN
   e. COLD SHUTDOWN
5. IS OFF SITE POWER AVAILABLE?
6. ARE THE DIESEL GENERATORS OPERABLE?
7. WHAT IS THE APPROXIMATE RADIOACTIVE DISCHARGE RATE FROM THE PLANT?
   a. NOBLE GASES _____ CI/SEC
   b. IODINES _____ CI/SEC
8. WHAT IS APPROXIMATE WIND SPEED AND DIRECTION?
9. WHAT IS THE APPROXIMATE DOSE RATE TO THE PUBLIC?
10. IS EVACUATION OF DOWNWIND AREAS RECOMMENDED?
11. HAVE ANY PERSONNEL INJURIES OCCURRED? IF SO, WHAT ACTION HAS BEEN TAKEN? WHAT ARE APPROXIMATE RADIATION AND/OR CONTAMINATION LEVELS?
12. ESTIMATED TIME OF RELEASE _____

Radiation Monitoring Survey

Area monitor readings

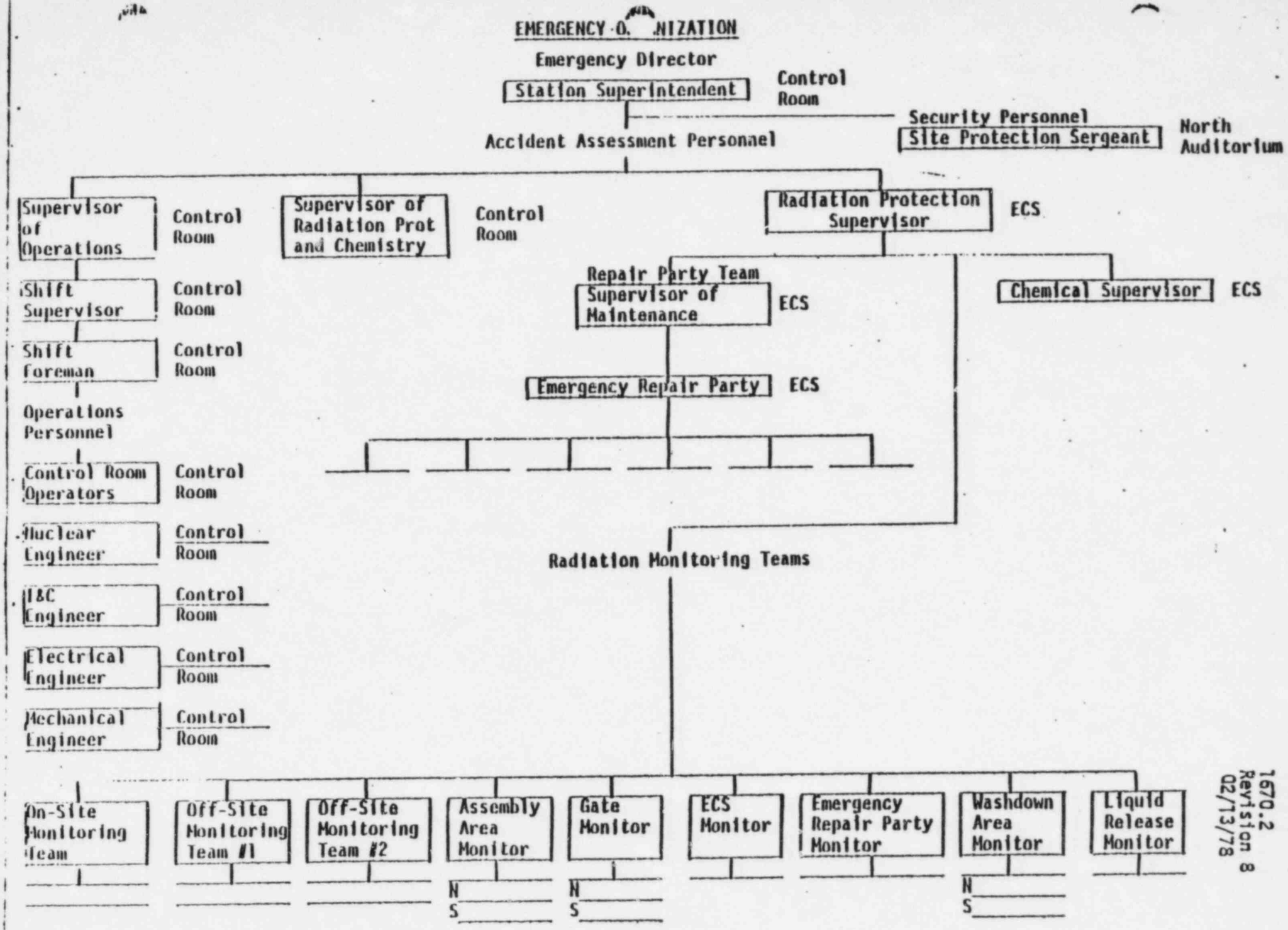
Liquid monitor readings

Air monitor readings

ADDITIONAL INFORMATION

1670.2
Revision 7
1/15/78

Figure 02


EMERGENCY ORGANIZATION
Emergency Director
Station Superintendent
Control Room
Security Personnel
Site Protection Sergeant
North Auditorium
Accident Assessment Personnel
Supervisor of Operations
Control Room
Supervisor of Radiation Prot and Chemistry
Control Room
Radiation Protection Supervisor
ECS
Repair Party Team
Supervisor of Maintenance
ECS
Chemical Supervisor
ECS
Shift Supervisor
Control Room
Shift Foreman
Control Room
Emergency Repair Party
ECS
Operations Personnel
Control Room Operators
Control Room
Nuclear Engineer
Control Room
I&C Engineer
Control Room
Electrical Engineer
Control Room
Mechanical Engineer
Control Room
Radiation Monitoring Teams
On-Site Monitoring Team
Off-Site Monitoring Team #1
Off-Site Monitoring Team #2
Assembly Area Monitor
N
S
Gate Monitor
N
S
ECS Monitor
Emergency Repair Party Monitor
Washdown Area Monitor
N
S
Liquid Release Monitor
1670.2
Revision 8
02/13/78

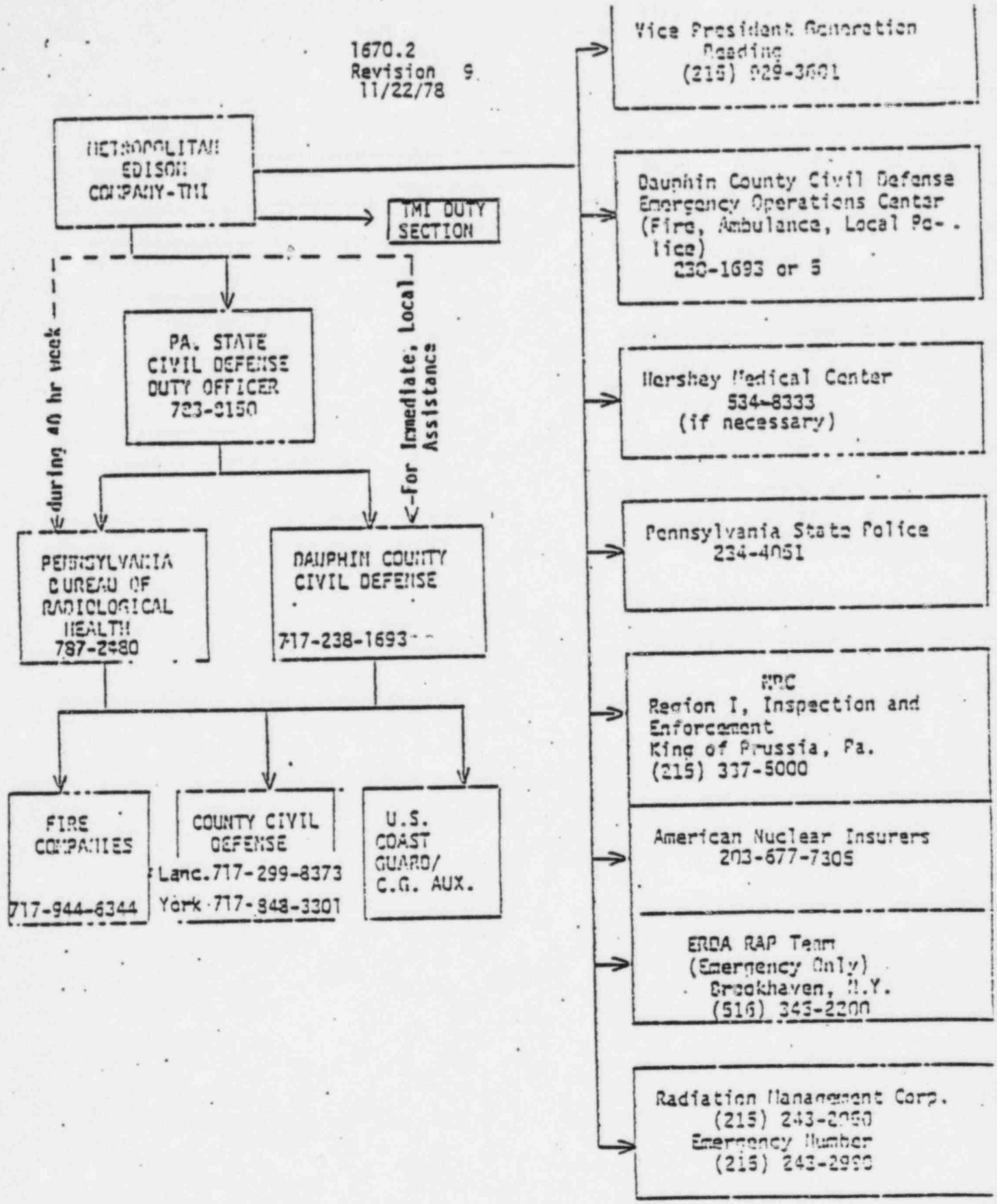

METROPOLITAN EDISON COMPANY-TMI
TMI DUTY SECTION
during 40 hr week
For Immediate, Local Assistance
PA. STATE CIVIL DEFENSE DUTY OFFICER 783-8150
PENNSYLVANIA BUREAU OF RADIOLOGICAL HEALTH 787-2480
DAUPHIN COUNTY CIVIL DEFENSE 717-238-1693
FIRE COMPANIES 717-944-6344
COUNTY CIVIL DEFENSE Lanc. 717-299-8373 York 717-848-3301
U.S. COAST GUARD/ C.G. AUX.
Vice President Generation Reading (215) 929-3601
Dauphin County Civil Defense Emergency Operations Center (Fire, Ambulance, Local Police) 230-1693 or 5
Hershey Medical Center 534-8333 (if necessary)
Pennsylvania State Police 234-4051
NRC Region I, Inspection and Enforcement King of Prussia, Pa. (215) 337-5000
American Nuclear Insurers 203-677-7305
ERDA RAP Team (Emergency Only) Brookhaven, N.Y. (516) 343-2200
Radiation Management Corp. (215) 243-2050 Emergency Number (215) 243-2900

FIGURE 1

FIGURE 5

ALTERNATES FOR PERSONNEL WITH EMERGENCY RESPONSIBILITIES

The following list indicates alternates to personnel assigned specific responsibilities in Site Emergency Procedure 1670.2 if those personnel are not on site.

A. Emergency Directors

A1) Primary: Station Superintendent
Alternates:
1) Unit Superintendents (1&2)
2) Unit Superintendents-Technical Support (1&2)
3) Supervisor of Operations
4) Shift Supervisor
5) Shift Foreman

B. Accident Assessment Personnel

B1) Primary: Supervisor Radiation Protection and Chemistry
Alternates:
1) Radiation Protection Supervisor
2) Nuclear Engineer
3) Shift Supervisor

B2) Primary: Radiation Protection Supervisor
Alternates:
1) Radiation Protection Foreman
2) Radiation/Chemistry Technicians

B3) Primary: Supervisor of Operations
Alternate:
1) Shift Supervisor

B4) Primary: Lead Engineers (Nuclear, I&C, Electrical, Mechanical)
Alternate:
1) Shift Supervisor

B5) Primary: Chemistry Supervisor
Alternate:
1) Radiation/Chemistry Technician

C. Repair Party Team

C1) Primary: Supervisor of Maintenance
Alternate:
1) Shift Maintenance Foreman

D. Radiation Monitoring Teams

D1) Primary: Radiation/Chemistry Technicians/Jr.
Alternate:
1) Auxiliary Operators

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Revision 5
02/13/78

THREE MILE ISLAND NUCLEAR STATION

CENTRAL FILE

STATION HEALTH PHYSICS PROCEDURE 1670.3
EMERGENCY PLAN, VOLUME 1
EMERGENCY PROCEDURE - GENERAL EMERGENCY PROCEDURE

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19.0			44.0			69.0		
20.0			45.0			70.0		
21.0			46.0			71.0		
22.0			47.0			72.0		
23.0			48.0			73.0		
24.0			49.0			74.0		
25.0			50.0			75.0		

Unit 1 Staff Recommends Approval

Approval N/A Date —

Cognizant Dept. Head

Unit 2 Staff Recommends Approval

Approval N/A Date —

Cognizant Dept. Head

Unit 1 PORC Recommends Approval

H. Kunder Date 2/15/78

Chairman of PORC

Unit 2 PORC Recommends Approval

[signature] Date 2.15.78

V. Chairman of PORC

Unit 1 Superintendent Approval

[signature] Date 2.15.78

Unit 2 Superintendent Approval

[signature] Date 2/15/78

Manager Generation Quality Assurance Approval N/A Date —

TMI 55-A Rev 8/77

THREE MILE ISLAND NUCLEAR STATION

STATION HEALTH PHYSICS PROCEDURE 1670.3

EMERGENCY PLAN, VOLUME 1

EMERGENCY PROCEDURE - GENERAL EMERGENCY PROCEDURE

1.0 DISCUSSION

A General Emergency is an occurrence with definite off-site protective actions required to minimize the off-site radiological dose. In addition to the off-site responses, it may be necessary to evacuate personnel. Off-site agencies will be involved extensively for both assessment and protective actions.

2.0 OBJECTIVE

To outline specific actions required in the event of a General Emergency.

3.0 EMERGENCY ACTION LEVELS

3.1 Reactor Building high range gamma radiation monitor indicates >8 R/hr.

3.2 Radiation levels at the site boundary are >125 mR/hr.

3.3 The liquid effluent radiation monitor indicates >6.8×10^{-3} uCi/cc.

3.4 Off-site projected doses down wind from the site boundary are >25 Rem thyroid and/or 5 Rem whole body.

4.0 IMMEDIATE ACTION

4.1 Emergency Director (Station Superintendent)

4.1.1 The initial actions and responsibilities are delineated as in the Site Emergency Procedure (1670.2).

4.1.2 Declare a General Emergency when any or all emergency action levels listed in 3.0 exist.

4.1.3 Contact off-site support groups as required.

4.1.4 The State of Pennsylvania Radiation Protection Guide values for probable evaculation of LPZ are greater than 5 Rem whole body or greater than 25 Rem to the thyroid avoidable by protective action.

4.1.5 In the event Three Mile Island and/or off-site agency monitoring teams project that the off-site radiation doses to the public exceed doses outlined in the State of Pennsylvania Radiation Emergency Plan (Enclosure 1), recommend to State of Pennsylvania representatives that affected portions of the Low Population Zone be evacuated.

4.1.6 Request the Civil Defense Duty Officer to ensure air, rail and coast guard authorities are notified to prevent transport vehicles from entering the affected area.

4.1.7 In the event the Bureau of Radiological Health cannot be contacted and the situation required immediate protective action, the Station Superintendent/Senior Unit Superintendent, will recommend to County Civil Defense to immediately evacuate the affected area.

4.2 Duties of the following personnel during a General Emergency are outlined in the Site Emergency Procedure.

4.2.1 Accident Assessment Personnel

a. Shift Supervisor

b. Shift Foreman

c. Supervisor of Operations

d. Nuclear Engineer

e. Supervisor Radiation Protection and Chemistry

f. Radiation Protection Supervisor

g. Chemical Supervisor

h. Lead Engineers

4.2.2 Radiation Monitoring Teams

4.2.3 Repair Party Team

4.2.4 Operations Personnel

4.2.5 Security Personnel

EMERGENCY ORGANIZATION

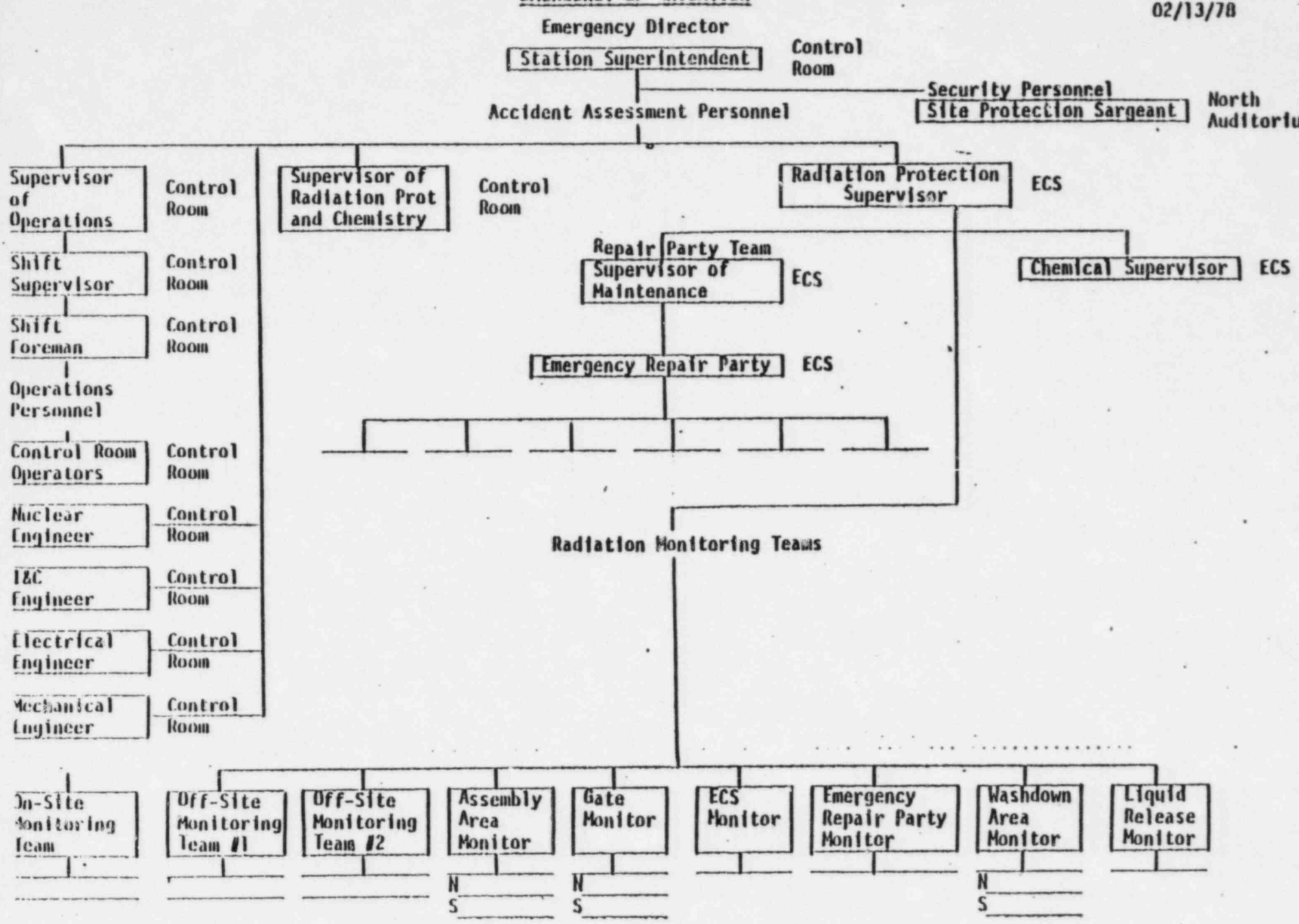


FIGURE 2

EMERGENCY STATUS BOARD

FLORIDA BUREAU OF RAD. HEALTH QUESTIONS

1. WHAT TYPE ACCIDENT HAS OCCURRED?
2. HAS THE REACTOR TRIPPED?
3. DID THE EMERGENCY SAFEGUARDS SYSTEMS ACTUATE?
   IF SO, WHICH ONES?
   a. HIGH PRESSURE INJECTION YES___ NO___
   b. LOW PRESSURE INJECTION YES___ NO___
   c. CORE FLOOD YES___ NO___
   d. REACTOR BUILDING ISOLATION YES___ NO___
   e. REACTOR BUILDING COOLING YES___ NO___
4. WHAT IS THE STATUS OF THE PLANT?
   a. AT POWER
   b. HOT STANDBY
   c. HOT SHUTDOWN
   d. COOLING DOWN
   e. COLD SHUTDOWN
5. IS OFF SITE POWER AVAILABLE?
6. ARE THE DIESEL GENERATORS OPERABLE?
7. WHAT IS THE APPROXIMATE RADIOACTIVE DISCHARGE RATE FROM THE PLANT?
   a. NOBLE GASES _____ CI/SEC
   b. IODINES _____ CI/SEC
8. WHAT IS APPROXIMATE WIND SPEED AND DIRECTION?
9. WHAT IS THE APPROXIMATE DOSE RATE TO THE PUBLIC?
10. IS EVACUATION OF DOWNWIND AREAS RECOMMENDED?
11. HAVE ANY PERSONNEL INJURIES OCCURRED? IF SO, WHAT ACTION HAS BEEN TAKEN? WHAT ARE APPROXIMATE RADIATION AND/OR CONTAMINATION LEVELS?

12. ESTIMATED TIME OF RELEASE _____

Radiation Monitoring Survey

Area monitor readings

Liquid monitor readings

Air monitor readings

ADDITIONAL INFORMATION

FIGURE

ROTECTIVE ACTION LEVELS DEFINED WITHIN THE COMMONWEALTH OF PENNSYLVANIA RADIATION EMERGENCY PLAN ARE AS FOLLOWS:

ASSIFICATION	WHOLE BODY DOSE		THYROID DOSE	ACTION
rotection Action Guide I	<.5 Rem	or a	<3 Rem	a. Increased Radiological environmental surveillance will be conducted. b. Eating and drinking local foods and water will be controlled. c. Control of access to affected area will be maintained.
rotection Action Guide II	greater than .5 Rem but less than 5 Rem	or a	greater than 3 but less than 25 Rem	a. Direction to the public to take cover will be initiated. b. Possible evacuation of selected individuals may be conducted.
rotection Action Guide III	greater than 5 Rem	or a	greater than 25 Rem	a. Actions planned include increased Rad. Env. surveillance. b. Probable evacuation of high risk individuals. (i.e., those in poor health and more susceptible to effects of radiation) and probable evacuation of general population in the affected area.

1670.3
Revision 6
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