SUMMARY OF CHANGES IN AMENDMENT NO. 23

1. A List of Effective Pages is provided.
2. Pages from Section No. 8.0 have been reissued since they were forwarded with Amendment No. 22 marked improperly as Amendment No. 6.
3. Information is provided concerning Reactor Coolant Sampling and Reactor Building Atmospheric Sampling.
4. Errors on the containment isolation tables are corrected.
5. Certain commitment dates have been deleted since the commitment dates have been provided in response to NUREG-0737 by letter dated January 23, 1981 (TLL-680).
6. Additional information justifying the Pressurizer Safety Valve position monitors is provided.
7. Certain revised information is provided concerning the GPU Nuclear organization.
8. Operator Guidelines for steam generator filling have been added to the response to Question No. 55 of Supplement No. 1, Part No. 1.
9. Revised analytical assumptions are added for Appendix 8A analyses.

TABLE OF EFFECTIVE PAGES
TMI-1 RESTART REPORT

*... ...dment numbers not noted on pages.

SECTION NO. 2 - FIGURES:

Figure No.	Am. No.
2.1-1	5
2.1-2	5
2.1-3	5
2.1-4	22
2.1-5	6
2.1-6	21
2.1-7	21
2.1-8	5
2.1-9	5
2.1-10	5
2.1-11	21
2.1-12	22
2.1-13	22
2.1-14	23
2.1-15	23
2.1-16	23

SECTION NO. 2 - APPENDIX 2A:

Description	Am. No.
1. BAW-1603	23
2. TDR-240, Rev. 0	23

SECTION NO. 2 - APPENDIX 2B:

Description	Am. No.
Cylindrical Source Shielding Equations Using Compact Functions and Including Buildup	15

SECTION NO. 3:

Page No.	Am. No.
3-1	5
3-2	5
3-3	5

SECTION NO. 3 - TABLES:

Table No.	Am. No.
3.1-1	5 (Page 1)
	15 (Page 2)
3.1-2	10 (10 Pages)

SECTION NO. 4:

Page No.	Am. No.
i	19
ii	19
iii	19
iv	19
v	19
vi	19
4-1	19
4.1-1	19
4.1-2	19
4.1-3	19
4.1-4	19
4.1-5	19
4.2-1	19
4.2-2	19
4.2-3	19
4.2-4	19
4.2-5	19
4.2-6	19
4.2-7	19
4.3-1	19
4.3-2	19
4.4-1	19
4.4-2	19
4.4-3	19
4.4-4	19
4.4-5	19
4.4-6	19
4.4-7	19
4.4-8	19
4.4-9	19
4.4-10	19
4.5-1	19
4.5-2	19
4.5-3	19
4.5-4	19
4.5-5	19
4.5-6	19
4.5-7	19
4.5-8	19

* Amendment numbers not noted on page.

SECTION NO. 4: (Cont'd.)

Page No.	Am. No.
4.5-9	19
4.5-10	19
4.5-11	19
4.5-12	19
4.5-13	19
4.5-14	19
4.5-15	19
4.5-16	19
4.5-16a	19
4.5-17	19
4.5-18	19
4.5-19	19
4.5-20	19
4.5-21	19
4.5-22	19
4.5-23	19
4.5-24	19
4.5-25	19
4.6-1	19
4.6-2	19
4.6-3	19
4.6-4	19
4.6-5	19
4.6-6	19
4.6-7	19
4.6-8	19
4.6-9	19
4.6-10	19
4.6-11	19
4.6-12	19
4.6-13	19
4.6-14	19
4.6-15	19
4.6-16	19
4.6-17	19
4.6-18	19
4.6-19	19
4.6-20	19
4.6-21	19
4.6-22	19
4.7-1	19
4.7-2	19
4.7-3	19
4.7-4	19
4.7-5	19
4.7-6	19
4.7-7	19

SECTION NO. 4: (Cont'd.)

Page No.	Am. No.
4.7-8	19
4.7-9	19
4.7-10	19
4.7-11	19
4.7-12	19
4.7-13	19
4.7-14	19
4.7-15	19
4.7-16	19
4.7-17	19
4.7-18	19
4.7-19	19
4.7-20	19
4.7-21	19
4.7-22	19
4.7-23	19
4.8-1	19
4.8-2	19
4.8-3	19
4.8-4	19
4.8-5	19
4.8-6	19
4.8-7	19
4.8-8	19
4.8-9	19
4.8-10	19
4.9-1	19
4.10-1	19
4.10-2	19
4.10-3	19

SECTION NO. 4 - FIGURES:

Figure No.	Am. No.
1	19
2	19
3	19
4	19
5	19
6	19
7	19
8	19
9	19
10	19

*Amendment number not noted on page.

SECTION NO. 4 - FIGURES: (Cont'd.)

Figure No.	Am. No.
11	19
12	19
13	19
14	19
15	19
16	19
17	19
18	19
19	19
20	19
21	19
22	19
23	19
24	19
25	19
26	19
27	19

SECTION NO. 4 - TABLES:

Table No.	Am. No.
1	19
2	19
3	19
4	19
5	19
6	19
7	19
8	19
9	19
10	19
11	19
12	19
13	19
14	19
15	19
16	19
17	19
18	19
19	19
20	19

SECTION NO. 4 - APPENDIXES:

	Am. No.
Appendix A	19
Appendix B	19 (25 pgs.)

SECTION NO. 4 - APPENDIXES: (Cont'd.)

	Am. No.
Appendix C	19 (31 pgs.)
Appendixes D - I	19 (1 pg.)

SECTION NO. 5:

Page No.	Am. No.
5.1-1	20
5.1-2	20
5.1-3	20
5.2-1	20
5.2-2	20
5.2-3	20
5.2-4	20
5.2-5	20
5.2-6	23
5.2-7	20
5.2-8	23
5.2-9	20
5.2-10	20
5.2-11	20
5.2-12	20
5.2-13	20
5.2-14	20
5.2-15	20
5.2-16	20
5.2-17	20
5.2-18	20
5.2-19	20
5.2-20	20
5.2-21	20
5.2-22	20
5.2-23	20
5.2-24	20
5.2-25	20
5.2-26	20
5.2-27	20
5.2-28	20
5.2-29	20
5.2-30	20
5.2-31	20
5.2-32	20
5.2-33	20
5.2-34	20
5.2-35	20
5.2-36	20
5.2-37	20
5.2-38	20
5.2-39	20

*Amendment number not noted on page.

SECTION NO. 5: (Cont'd.)

Page No.	Am. No.
5.2-40	20
5.2-41	20
5.2-42	20
5.2-43	20
5.2-44	20
5.2-45	20
5.2-46	20
5.2-47	20
5.2-48	20
5.2-49	20
5.2-50	20
5.2-51	20
5.2-52	20
5.2-53	20
5.2-54	20
5.2-55	20
5.2-56	20
5.2-57	20
5.2-58	20
5.2-59	20
5.2-60	20
5.2-61	20
5.2-62	20
5.2-63	20
5.2-64	20
5.2-65	20
5.2-66	20
5.2-67	20
5.2-68	20
5.2-69	20
5.2-70	20
5.2-71	20
5.2-72	20
5.2-73	20
5.2-74	20

SECTION NO. 5 - FIGURES:

Figure No.	Am. No.
5.2-1	20

SECTION NO. 5: (Cont'd.)

Page No.	Am. No.
5.3-1	20
5.3-2	20
5.3-3	20
5.3-4	20
5.3-5	20
5.3-6	20
5.3-7	20
5.3-8	20
5.3-9	20
5.3-10	20
5.3-11	20
5.3-12	20
5.3-13	20
5.3-14	20
5.3-15	20
5.3-16	20
5.3-17	20
5.3-18	20
5.3-19	22
5.3-20	22
5.3-21	22
5.3-22	22
5.3-23	22
5.3-24	22
5.3-25	22
5.3-26	22
5.3-27	20
5.3-28	20
5.3-29	20
5.3-30	20

SECTION NO. 5 - FIGURES:

Figure No.	Am. No.
5.3-1	23

SECTION NO. 5: (Cont'd.)

Page No.	Am. No.
5.4-1	20
5.4-2	20
5.4-3	20
5.4-4	20
5.4-5	20

*Amendment number not noted on page.

SECTION NO. 6:

Page No.	Am. No.
6-1	21
6-2	21
6-3	21
6-4	21
6-5	21
6-6	21
6-7	21
6-8	21
6-9	21
6-10	21
6-11	21
6-12	21
6-13	21
6-14	21
6-15	21
6-16	21
6-17	21

SECTION NO. 7:

Page No.	Am. No.
7-1	4
7-2	4
7-3	7
7-4	18
7-5	18
7-6	18
7-7	18
7-7a	18
7-8	4
7-9	23
7-9a	21
7-9b	21
7-10	5
7-11	4
7-11a	18
7-11b	8
7-12	13
7-13	4
7-14	23
7-15	16

SECTION NO. 7 - TABLES:

Table No.	Am. No.
7.3-1	5 (3 pgs.)

SECTION NO. 7 - FIGURES:

Figure No.	Am. No.
7-1	4
7-2	4
7-3	4
7-4	4

SECTION NO. 7 - APPENDIXES:

	Am. No.
Appendix 7A	17 (25 pgs.)
Appendix 7B	11 (6 pgs.)

SECTION NO. 8:

Page No.	Am. No.
8-1	22
8-2	22
8-3	11
8-4	11
8-5	11
8-6	6
8-7	16
8-8	6
8-9	16
8-10	16
8-11	7
8-12	22
8-13	22
8-14	22
8-15	22
8-16	22
8-17	22
8-18	22
8-19	22
8-19a	23
8-20	16
8-21	6

*Amendment number not noted on page.

SECTION NO. 8 - TABLES:

Table No	Am. No.
8-1	11
8-2	6

SECTION NO. 8 - APPENDIXES:

	Am. No.
Appendix 8A (Title Sheet)	6
Appendix 8A-1	10
Appendix 8A-2	10

SECTION NO. 8 - APPENDIX 8A - TABLES:

Table No.	Am. No.
8A-1	23
8A-2	23
8A-3	23
8A-4	23
8A-5	23
8A-6	23
8A-7	23
8A-8	23
8A-9	23
8A-10	23
8A-11	23
8A-12	23
8A-13	23
8A-14	23
8A-15	23
8A-16	23
8A-17	23
8A-18	23
8A-19	23
8A-20	23
8A-21	23
8A-22	23
8A-23	23
8A-24	23
8A-25	23
8A-26	23
8A-27	23
8A-28	23
8A-29	23
8A-30	23
8A-31	23

SECTION NO. 8 - APPENDIX 8A- FIGURES:

Figure No.	Am. No.
8A-1	12
8A-1a	13
8A-2	6
8A-2a	11
8A-3	*
8A-4	22
8A-5	6
8A-6	6
8A-7	6
8A-8	10
8A-9	10
8A-10	10
8A-11	10
8A-12	12
8A-13 (Missing)	
8A-14	12
8A-15	12
8A-16	15
8A-17	18
8A-18	22
8A-19	22
8A-20	22
8A-21	22
8A-22	22
8A-23	22
8A-24	22

SECTION NO. 9:

Page No.	Am. No.
i	21
ii	21
iii	21
iv	21
v	21
9-1	4

SECTION NO. 9 - TABLES:

Table No.	Am. No.
9.2-1	11

SECTION NO. 10:

Page No.	Am. No.
10-1	10
10-2	10

*Amendment number not noted on page

SECTION NO. 10:

Page No.	Am. No.
10-3	18
10-4	3
10-5	10
10-6	22
10-7	15
10-8	15
10-9	18
10-10	18
10-11	18

SECTION NO. 10 - APPENDIXES:

	Am. No.
Appendix No. 10A:	
1) Reactor Trip No. 11	*
2) Reactor Trip No. 12	*

SECTION NO. 11:

Page No.	Am. No.
11-1	21
11-2	21
11-3	21
11-4	21
11-5	21
11-6	21
11-7	21
11-8	21
11-9	21
11-10	21
11-11	21
11-12	21
11-13	21
11-14	21
11-15	21
11-16	21
11-17	21
11-18	21
11-19	21
11-20	21
11-21	21
11-22	21
11-23	21
11-24	21
11-25	21
11-26	21

SECTION NO. 11.2.1:

Page No.	Am. No.
(5 pgs.)	*

SECTION NO. 11.2.2:

Page No.	Am. No.
(9 pgs.)	21

SECTION NO. 11.2.3:

Page No.	Am. No.
(6 pgs.)	21

SECTION NO. 11.2.4:

Page No.	Am. No.
(4 pgs.)	*

SECTION NO. 11.2.5:

Page No.	Am. No.
(4 pgs.)	21

SECTION NO. 11.2.6:

Page No.	Am. No.
(5 pgs.)	21

SECTION NO. 11.2.7:

Page No.	Am. No.
(15 pgs.)	21

SECTION NO. 11.2.8:

Page No.	Am. No.
(5 pgs.)	16

*Amendment number not noted on page.

SECTION NO. 11.2.9:

Page No.	Am. No.
(3 pgs.)	21

SECTION NO. 11.2.10:

Page No.	Am. No.
(5 pgs.)	*

SECTION NO. 11.2.11:

Page No.	Am. No.
(2 pgs.)	*

SECTION NO. 11.2.12:

Page No.	Am. No.
(5 pgs.)	*

SECTION NO. 11.2.13:

Page No.	Am. No.
(4 pgs.)	16

*Amendment number not noted on page.

SUPPLEMENT NO. 1, PART NO. 1:

Page No.	Am. No.
i	16
ii	16
iii	16
iv	16
v	16

Question No.	Page No.	Am. No.
1	1	5
2	1	22
3	1	6
	2	6
	3	6
	4	6
4	1	5
5	1	5
6	1	5
7	1	5
8	1	13
9	1	5
10	1	16
10 - App. A		
1a	1	13
	2	13
1b	1	13
	2	13
2	1	13
	2	13
3	1	13
10a	1	5
10b	1	5
10c	1	5
10d	1	5
10e	1	5
10f	1	5
10g	1	5
10h	1	5
10i	1	18
	2	18
10j	1	18
10k	1	22
11	1	5
12	1	5
13	1	5
14	1	5
15	1	5
16	1	5

*Amendment number not noted on page.

SUPPLEMENT NO. 1, PART NO. 1: (Cont'd.)

Question No.	Page No.	Am. No.
17	1	6
	2	6
18	1	5
19	1	5
20	1	5
Table 20-1	1	12
	2	12
21	1	18
22	1	5
23	1	5
24	1	5
25	1	5
26	1	5
27a	1	10
27b	1	12
27c	1	12
27d	1	10
27e	1	5
28	1	5
29	1	18
30	1	5
31	1	5
32	1	5
33	1	5
34	1	5
35	1	5
36	1	5
36a	1	5
36b	1	5
Figure 1 to Q. 36	1	9
36c	1	5
37	1	5
38	1	5
39	1	5
39a	1	5
39b	1	5
40	1	5
41	1	5
42a	1	18
	2	18
	3	18
	4	18
42b	1	5
	2	5
42c	1	5
43	1	5
44	1	5

*Amendment number not noted on page.

SUPPLEMENT NO. 1, PART NO. 1: (Cont'd.)

Question No.	Page No.	Am. No.
45	1	15
Attachment to Q. 45		
1) Document No. 86-1105508-00	30	*
2) Document No. 69-1106001	43	*
46	1	5
47	1	5
48	1	5
49	1	5
50	1	5
51	1	5
52	1	5
53	1	5
54	1	5
55	1	23
	2	23
Attachment to Q. 55	13	23
56	1	5
57	1	5
58	1	5
59	1	5
60	1	5
61	1	5
62	1	9
63	1	9
64	1	9
65	1	5
66	1	22
67	1	9

* Amendment number not noted on page.

SUPPLEMENT NO. 1, PART NO. 2:

Page No.	Am. No.
i	16
ii	16
iii	16
iv	16
v	16
vi	16

Question No.	Page No.	Am. No.
1a/b	1	*
1c	1	22
2a/b	1	13
2c/d	1	9
	2	9
	3	12
	4	9
2e	1	9
Attachment to Q. 2e		
1) Document No. 32-1106898-00	49	*
3	1	13
	2	18
	3	18
Attachment to Q. 3		
1) An Evaluation of an Emergency Feedwater Flowrate of 500 gpm for 2568 MWT B&W Plants	9	10
2) Document No. 86-11-2587-00	11	6
4	1	22
5	1	7
	2	7
	3	7
Attachment to Q. 5		
1) Transducer/Computer Specification	2	*
6	1	18
Attachment to Q. 6	1	18
	2	18
	3	18
7	1	6

*Amendment number not noted on page.

SUPPLEMENT NO. 1, PART NO. 2: (Cont'd.)

Question No.	Page No.	Am. No.
8	1	6
9	1	6
10	1	6
11	1	6
12	1	12
	2	12
13	1	22
14	1	18
	2	18
Attachment to Q. 14		
1) Figure 1	1	18
2) Figure 2	1	18
3) Letter/Report	26	*
15	1	12
	2	12
	3	7
Table 15-1	1	7
Figure 15-1	1	7
Document 86-1102525-00	29	7
16	1	21
17	1	6
18	1	6
19	1	6
20	1	6
21	1	6
22	1	6
	2	6
23	1	6
24	1	6
25	1	9
	2	18
Attachment to Q. 25	5	*
26	1	6
27	1	6
28	1	6
29	1	6
30	1	6
31	1	7
32	1	10
33	1	6
34	1	6
35	1	9
36	1	6
37	1	6

*Amendment number not noted on page.

SUPPLEMENT NO. 1, PART NO. 2 (Cont'd.)

Question No.	Page No.	Am. No.
38	1	6
39	1	6
40		6
41		6
42		6
43		6
44	1	6
45	2	6
46	3	6
47		6
48		6
49		6
50		6
51	1	6
52	1	22
	2	22
	3	21
	4	21
	5	21
Attachment to Q. 52	3	21
53	1	18
	2	22
54	1	6
55	1	7
Attachment to Q. 55	8	6
56	1	6
Attachment to Q. 56	1	6
57	1	6
58	1	6
59	1	6
60	1	6
61	1	6
62	1	6
63	1	9
64	1	6
	2	6
65	1	6
66	1	6
	2	6
	3	6
	4	6
67	1	22
68	1	22
69	1	6
70	1	6
71	1	6
	2	6
	3	6
	4	6
	5	6
	6	6
	7	6

Am. 23

*Amendment number not noted on page.

SUPPLEMENT NO. 1, PART NO. 2 (Cont'd.)

Question No.	Page No.	Am. No.
72	1	6
73	1	6
74	1	6
75	1	6
76	1	6
77	1	6
78	1	6
79	1	6
	2	6
	3	6
80	1	6
81	1	6
82	1	6
	2	6
	3	
83	1	6
84	1	6
85	1	6
86	1	6
87	1	6
88	1	6
	2	6
	3	6
89	1	6
90	1	6
91	1	7
	2	7
Attachment to Q. 91	19	7
92	1	6
93	1	6
	2	6
94	1	6
Table 20-1	1	6
	2	6
95	1	21
Attachment to Q. 95 Document No. 86-1120838-00	21	*

*Amendment number not noted on page.

SUPPLEMENT NO. 1, PART NO. 3

Page No.	Am. No.
i	16

Question No.	Page No.	Am. No.
1	1	10
	2	13
	3	13
Attachment to Q. 1		
1) SDD-211A	i	18
	ii	18
	iii	18
	1	18
	2	18
	3	22
	4	18
	5	18
	6	18
	7	18
	8	18
	9	22
	10	18
	11	18
Figure 1	1	18
Figure 2	1	18
Figure 3	1	18
Table 1	1	18
Table 2	1	18
Table 3	1	18
Table 4	1	18
Table 5	1	18
Table 6	1	18
Letter TMI-80-070	7	*
Letter TMI-80-036	6	*
Letter TMI-79-208	5	*
2	1	10
3	1	10
Attachment to Q. 3	5	*
4	1	12
	2	12
Attachment to Q. 4		
Figure 1	1	12
Figure 2	1	12
Figure 3	1	12
Figure 4	1	12
Figure 5	1	12
5	1	12
6	1	12
7	1	12

*Amendment number not noted on page.

SUPPLEMENT NO. 1, PART 3 (Cont'd.)

SUPPLEMENT NO. 2

*Amendment number not noted on page.

contact isolators to preclude the propogation of faults into the Reactor Protection System (RPS). The reactor will be tripped through the existing RPS logic upon coincident signals from any two of the four channels.

A bypass arrangement will be provided in order to allow for power escalation, starting the main turbine and normal shutdown of the main turbine. The main turbine trip bypass will be automatically placed in effect when reactor power is less than 20%. The bypass will be automatically removed when the reactor power is increased above 20%. Bypass of the feedwater pump trip signal is automatically placed in effect when reactor power is less than 10%. It will be removed automatically when reactor power is raised above 10%. The bypass fnction will be accomplished individually in each of the four channels by means of bistables which monitor the power range nuclear instrumentation.

The additional modules required in the Reactor Protection System will be the same safety grade equipment type used in the original system. Wiring for redundant channels will be separated and run in Seismic I, safety grade raceways except in the turbine building. Since the turbine building is not Seismic I, the equipment and wiring therein cannot be classified as Seismic I. However, all wiring in the turbine building for this system will be run in conduit and redundant channgls will be routed separately to minimize the probability of disabling more than one channel due to damage to the turbine building. The system will be designed with normally closed contacts so that an open wire will represent a tripped condition. The signals from the turbine building will go through contact isolators in the Reactor Protection System to preclude the propogation of faults in the system.

2.1.1.1.4 Design Evaluation

The Safety Grade Reactor Trip scheme provides an anticipatory trip to the reactor, reducing the number of reactor trips on high pressure and the number of challenges to the Pressurizer PORV and safety valves. (Also see Supplement 1, Part 2, Question 15.)

2.1.1.1.5 Safety Evaluation

The system is safety grade and meets the requirements of IEEE-279 including those for testability and single failure criterion. The modifications which will be required to the Reactor Protection System will not degrade the ability of that system to perform its design function. The design will result in an enhancement of nuclear safety.

2.1.1.1.6 Start-Up Testing

This system will be tested during installation to verify its operation prior to start-up.

conditions. Calculations have been made, using conservative assumptions, to demonstrate that a satisfactory signal will be generated when any of the valves open. Calculations have been made for saturated, liquid and two phase flow. A summary of these calculations is provided in Appendix 2A. The calculations demonstrate that a satisfactory signal is developed for flows as low as normal makeup flow for all plant conditions of interest. Tests run by B&W on the electromatic relief valve under reduced flow condiitons have confirmed the validity of this approach. Because of the straight-forward and well known relationships that exist between flow conditions and differential pressure across the elbow, the signal from one differential pressure transmitter can be confidently predicted for any flow conditions. For this reason it has been concluded that operating tests, which would be difficult since they involve opening the PORV and relief valves, will not be required.

Acoustic monitoring of the electromatic relief valve makes use of well proven equipment and techniques which have been used in the B&W Loose Parts Monitoring System. Tests run on this valve at the B&W Alliance facility demonstrated that the acoustic monitoring system gave satisfactory results.

2.1.1.2.5 Safety Evaluation

Instrument taps will be installed on elbows in the discharge piping of pressurizer code safety valves RC-RV1A and RC-RV1B and electromatic relief valve RC-RV2. This piping is classified as N2, Seismic I. Analysis has been performed to demonstrate that this modification will not degrade the integrity of the existing pipe. The pipe classification has been maintained up to and including the instrument root valves. The mounting of new equipment which will be located in the vicinity of safety related systems has been analyzed to ensure that no hazardous missiles will be generated in a seismic event. It has been concluded that this modification will not degrade any safety related systems. Shock loadings were considered in the original design of the PORV and safety valve discharge piping. It was also considered for the design of the elbow tap instrument since water loop seals are maintained on the safety valves to prevent contact of steam during normal operation and the PORV discharge lines may see slug flow under certain conditions. When the safety/relief valve opens a slug of water is discharged to the tailpipes. The small size of the elbow tap instrument lines compared to the 4 inch (PORV) tailpipe results in only a small portion of the pressure wave (due to slug flow) from effecting the elbow tap instrument. Any effect is further dampened by condensing pots in the instlrument lines close to the tailpipe elbow and due to the relatively long length of instrument sensing lines. The differential pressure cell is also designed to withstand full system pressure of 2500 psi across the diaphram compared to an operational requirement of 400 inches of water with no loss of accuracy or damage.

All of the equipment inside containment for Pressurizer PORV and safety valve detection will be seismically and environmentally qualified. Work is underway to upgrade the portion of the system outside containment. This involves specifying and procuring of additional equipment.

2.1.2 Long Term Modification

2.1.2.1 Post Accident Monitoring

2.1.2.1.1 System Description

Certain post accident monitoring capability will be provided in compliance with Reg. Guide 1.97, Rev.2 as discussed below. Pending the availability of appropriately qualified instrumentation and equipment, the following modifications will therefore be completed as soon as possible. The final design will be provided for NRC review.

Containment Pressure - Continuous containment pressure indication will be provided in the control room using a range from -5 psig to three times the design pressure of the containment. The pressure indication will be safety grade and will meet the design and qualification requirements of Reg. Guide 1.97. Redundant indication of pressure will be provided.

Containment Water Level - Continuous containment water level indication shall be provided in the control room. A safety grade wide range indicator from the bottom of containment to a level of 90 inches will be installed in accordance with the requirements of Reg. Guide 1.97. In addition, a narrow range indicator from the bottom to the top of the sump with continuous indication in the control room shall be installed which meets the requirements of Reg. Guide 1.89 and is capable of being periodically tested.

Containment Hydrogen Indication -Safety grade continuous indication of containment hydrogen will be provided in the control room. The range of indication will be 0-10% concentration assuming commercial availability over this range.

High Range Containment Radiation Monitor - Two safety grade containment radiation monitors that are physically separated shall be provided with recording display and continuous indicator presentation in the control room. The range of this monitor shall be 10^7 R/hr and shall detect photon radiation down to 60 Kev. The design of the radiation monitors shall be provided in accordance with Reg. Guide 1.97 Rev. 2.

High Range Effluent Monitor - High range effluent monitors intended as the Long-Term modification are planned for each normal gas release point.

The range of these monitors shall be as follows:

- Undiluted Containment Exhaust . . . . 10^5uCi/cc
- Main Steam Lines . . . . 10^2uCi/cc
- Auxiliary & Fuel Handling Building Exhaust . . . 10^3uCi/cc
- Condenser OFF GAS Exhaust . . . . 10^5uCi/cc

Regulatory Guide 1.97 Rev. 2 (Dec. 1979) will be followed for the design of range for the high range effluent monitors. Vital bus power shall be employed for each system's modular assembly with the normal power supplying the monitor pumps with diesel generations as back ups.

High Range Effluent Radio Iodine & Particulate Sampling Analysis - The existing sampling system will be expanded and will include the addition of silver zeolite cartridges. The system design and operation will both decrease the activity on the cartridges so they can be handled and will decrease the xenon to iodine ratio. Counting of the cartridges will be by use of NaI crystal connected to a single or dual channel analyzer with appropriate window and discrimination settings for th 364 Kev gamma of I-131, or by use of a GELI/MCA system. The expanded portion of the sampling system would be placed in service following an accident and will be located in an applicable area exhibiting low background.

Prior to incorporation of the expanded sampling system, procedures have been developed for the use of silver zeolite cartridges and normal particulated filters for sampling with a NaI detector and a single or dual channel analyzer for iodine and gross particulate release rate determination. Specific details to insure exposures are maintained as low as reasonably achievable are incorporated into the procedures.

2.1.2.2 RCS Venting

2.1.2.2.1 System Description

Power operated vents will be provided for the reactor coolant system in order to enhance natural circulation and adequate core cooling following an accident. The vents will be from the top of the pressurizer, the top of both hot legs using existing connections on the reactor coolant piping and from the Reactor Vessel Head. The discharge from the reactor vessel and hot leg vents will be directed to the containment atmosphere. The system is shown schematically in Figure 2.1.-11.

The hot leg vents will tie into existing hot leg vent piping inside the secondary shield wall. As part of this modification, remote operation of the vent valves in the existing vent line from the pressurizer to the reactor coolant drain tank will be provided and the system will retain the existing venting capability. Control and position indication for the power operated vent valves will be provided in the control room.

2.1.2.4 Post Accident Sampling Capability

2.1.2.4.1 Reactor Coolant Sampling

Reactor coolant post accident samples will be obtained and analyzed using the existing system and analysis equipment. The sampling system is shown on flow diagram C-302-671 and is located in the nuclear sampling room at Elevation 306'-0" of the control room tower as shown on general arrangement drawing E-015-015. The preparation of samples and chemical analysis will be performed in the radio-chemistry lab which is adjacent to the sampling room as shown on drawing E-015-015. The use of the existing equipment and facilities will be augmented with emergency plan procedures, use of long handled tools, lead shields and a shielded sample transport cart.

Reactor coolant following an accident will be sampled from the reactor coolant loop "B" cold leg letdown line via valves CA-V13 and CA-V2. Upon opening of these valves the reactor coolant is routed to sample cooler CA-C1 through the sample hood and sampled from valve CA-V16. From the sample hood the reactor coolant is then routed to a point upstream of the makeup purification system filters (MU-F1A/B) which discharge to the make up tank (MU-T1). The following table lists the valves which are operated to line up the sampling system as previously described:

Reactor Coolant Sampling System Valves

Valve Tag No.	Position	Operator	Location	Reference Flow Diagram
CA-V13	Open	Motor	Containment Building	C-302-671
CA-V2	Open	Motor	Auxiliary Building	C-302-671
CA-V25A	Open	Manual	Nuclear Sampling Room Sample Cooler	C-302-671
CA-V26A	Closed	Manual	"	C-302-671
CA-V25B	Closed	Manual	"	C-302-671
CA-V26B	Open	Manual	"	C-302-671
CA-V26C	Closed	Manual	"	C-302-671
CA-V25C	Closed	Manual	Nuclear Sampling Room Sample Hood	C-302-671

Reactor Coolant Sampling System Valves

Valve Tag No.	Position	Operator	Location	Reference Flow Diagram
CA-V33	Open	Manual	"	C-302-671
CA-V110	Open	Manual	"	C-302-671
CA-V34	Open	Manual	"	C-302-671
CA-V35	Open	Manual	"	C-302-671
CA-V16	Open	Manual	"	C-302-671
CA-V29	Open	Manual	"	C-302-671

Valves CA-V13 and CA-V2 may be operated from a local control panel in the nuclear sampling room or the ES (Engineered Safeguards) panel in the control room.

When the sampling system is used under post-accident conditions the nuclear sampling room and the areas through which the sample line runs will become high radiation areas. These areas will be administratively controlled during sampling evolutions to restrict personnel access to limit personnel radiation exposure. Figures 2.1-14 and 2.1-15 show the affected areas and the calculated dose rate based on the source strengths for T=0 (Reactor Trip) tabulated below.

The taking and handling of a post-accident reactor coolant sample will be done in accordance with emergency procedure. The procedure uses a team of three technicians in order to limit the exposure to any one individual. For the first sample, to be taken at T=45 min. the first technician will enter the nuclear sampling room to set up the equipment, perform a valve line-up and establish flow through the sampling system. Equipment set-up and valve line-up are expected to take 4 minutes with negligible exposure (3 MREM based on a 50 MREM/hr field). When flow is established the general area radiation level will increase to 180 REM/hr and result in a exposure of 3 REM to the technician based on a stay time of 1 minute. The second technician will then obtain the sample by opening valve CA-V16, filling the sample bottle (30 ml sample) from the CA-V16 drain line and then placing the sample bottle in the shielded cart, he will then close CA-V16. (CA-V2 and CA-V13 are closed from the control room). These operations are performed in the sample hood using long handled tools and with a stay time of 1.5 minutes results in an exposure of 4.5 REM (based on a dose rate of 180 REM/hr). The third technician will move the shielded cart to the chemistry lab. Moving the cart requires a stay of 1 minute resulting in an exposure of 3 REM (based on a dose rate of 180 REM/hr). The same procedure will be followed for a second sample assumed to be taken at T=8 hrs. The sampling evolution exposures are summarized below:

Reactor Coolant Sampling Dose Rates

	Stay Time	45 Min. Dose Rate	45 Min. Dose	8 Hr. Dose Rate	8 Hr. Dose
Technician 1	4 min.	* 55 mr/hr	3 REM	55 REM/hr	4.6 REM
	1 min.	180 REM/hr		55 REM/hr	
Technician 2	1.5 min.	180 REM/hr	4.5 REM	55 REM/hr	1.4 REM
Technician 3	1 min.	180 REM/hr	3 REM	55 REM/hr	.9 REM

* general area reading of the TMI-1 nuclear sampling room normally.

Note: The 45 minute sample is based on the decision to take a sample at T=0 (reactor trip). 45 minutes is to be the earliest time a sample can be taken.

Sample preparation and analysis will be done in accordance with emergency procedure. The procedure provides guidance to obtain analytical data on boron concentration, isotopic identification and chlorides. Upon transport of the shielded sample cart to the chemistry lab, the 30 ml sample will be transferred from the cart to a lead pig located in the chemistry hood. The hardware used for sample preparation and analysis will have been laid out per procedure prior to removing the sample bottle from the shielded cart.

Once a sample bottle is in the chemistry hood, 0.1 ml, 1.0 ml and 2.0 ml portions of the sample are pipeted into various containers for, (1) dilution in order to perform isotopic identification, (2) boron analysis and (3) chloride analysis, respectively. The estimated time to perform the dilution and analyses (boron) is 1.5 hours resulting in an exposure of 2.98 REM for first sample and .86 REM for the second sample to the technician. The dose takes into account the use of lead shields long handled tools and the handling of small or diluted samples. Samples prepare for isotopic analysis are transported to the count room located in the turbine building for counting on the Geli detector. There is no exposure associated with transporting and counting since the samples must read < 1 MR/hr prior to leaving the chemistry lab. The chloride sample will be obtained and analysis will be done off-site within 4 days.

The overall time to obtain and analyze the sample is tabulated below.

RCS Post Accident Sampling

Sequence	Time, hr	Cumulative Time, hr
1. Reactor trip & decision to take sample	0	0
2. Complete personnel and equipment preparation for obtaining sample	0.75	0.75
3. Obtain sample	0.50	1.25
4. Analyze sample	1.75	3.0

2.1.2.4.2 Containment Atmosphere Sampling

Containment atmosphere, post accident, will be analyzed by taking grab samples of the containment air and identifying the various isotopes and measuring hydrogen concentration. The sample will be obtained from a sample station located above the instrument air compressor (shown on drawing D-001-016) which is located in the southeast corner of the intermediate building. The sample system will use the existing penetrations for RM-A2 (Containment Air Monitoring System) as shown on drawing C-302-721 (Figure 2.1-16), to sample air inside containment. Using these penetrations, a sample can be drawn from two areas inside containment, the containment ventilation duct or the discharge of the containment cooling fans.

The sampling system is shown on Figure 2.1-16. The system ties into the sample lines for RM-A2 downstream of containment isolation valves CM-V3&4 and CM-V1 & 2 via the three way solenoid operated valves CM-V8 and CM-V7. Upon opening the containment isolation valves and positioning the three valves to divert flow to the sampling system, a jet pump is used to circulate the air from containment, through a sampler and back to containment. By proper positioning four way valve CM-V9 on the jet pump discharge, air can be drawn from the ventilation duct and returned to the discharge of the cooling fans or vice versa. The following table lists the valves which are operated to line up the sampling system as described:

Containment Atmosphere Sampling System Valves

Valve Tag No.	Position	Operator	Location
CM-V1	Open	Pneumatic	Intermediate Building
CM-V2	Open	"	"
CM-V3	Open	"	"
CM-V4	Open	"	"
CM-V7 (New)	Open to Sample System	"	"
CM-V8 (New)	"	"	"
CM-V9 (New)	Open	Motor	"
CM-V10 (New)	Open	"	"

* Valves are located inside the penetration cubicle next to and above the instrument air compressor cubicle in the southeast corner of the intermediate building.

The valves listed above will be operated from a new local control panel located in the corridor at Elevation 305'00" of the intermediate building. Once flow has been established from the control panel the technician proceeds to the sample station and withdraws a sample either through a septum on a 25cc sample bulb using a syringe or removes the entire sample bulb from the system. After obtaining the sample the system is then transported in a shield to the chemistry lab for preparation for analysis. The system can also be used to sample containment air using filtration or absorption devices.

Operation of the sampling system, under post accident conditions will result in a generally low area radiation level at the sampling station.

Of the evolutions described previously, the only significant exposure will be when the technician handles the sample prior to transport. The time to accomplish this evolution is 3 minutes resulting in an exposure of < 2.5 REM. All other sample system evolutions are either at the control panel which is located in an unrestricted area or after the system has been purged which reduces the general area radiation to a negligible level.

After the sample has been transported to the chemistry lab, a sample is either prepared for gamma spectrometry on the GELI detector or for hydrogen analysis on the gas chromatograph. If the 25cc sample bulb was used, a sample for gamma spectrometry is prepared by extracting 5cc using a syringe and injecting it into a 5cc glass vial. The glass vial is then transported to the count room. Hydrogen analysis is done by withdrawing 0.5cc and injecting into the gas partitioner which is located in the chemistry lab. Negligible exposure is associated with these evolutions because of the sample size, use of shields and the time to perform the above.

2.1.2.4.3 Reactor and Containment Atmosphere Post Accident Source Terms

The post accident source terms from which the dose rates are calculated were taken from the Midland Final Safety Analysis Report, Table 11.1-2, Total Core Fission Product Activity Versus Time in Equilibrium Cycle. The table in the Midland FSAR is based on an isotopic core inventory for 310 effective full power days in an equilibrium cycle at a power level of 2552 MWt. Source term information was taken from the Midland FSAR in the absence of comparable information for TMI-1. This information is slightly conservative since the TMI-1 power level is 2535 MWt.

The source strengths for reactor coolant and containment air following an accident are listed below. The reactor coolant source is based on the activity from 100% of the noble gases, 50% of the halogens and 1% of all other isotopes being diluted in the reactor coolant liquid volume. The containment air source is based on the activity from 100% of the noble gases and 25% of the halogens being dispersed within the air contained in the containment building free volume. The source strengths listed below are for times T=0 (reactor shutdown) and T=8 hrs. Personnel exposures for the various sampling evolutions are

based on the T=45 min. source strengths. Sampling at T=45 min. allows for a decay period of 4ᶜ min. which takes into account a time to fail fuel and to disperse the fission products since these events are not instantaneous processes.

Dose Rate Source Strengths

I Reactor Coolant

Time = 0		Time = 8 hrs	
Energy (MEV)	γ/cc-sec	Energy (MEV)	γ/cc-sec
.13	1.23×10^{10}	.094	6.59×10^{9}
.36	2.27×10^{10}	.32	8.19×10^{9}
.74	4.59×10^{10}	.61	8.02×10^{9}
1.23	1.19×10^{10}	1.23	3.04×10^{9}
1.74	6.51×10^{9}	1.73	1.19×10^{9}
2.20	8.09×10^{9}	2.28	7.05×10^{8}
2.57	1.64×10^{9}	2.56	6.85×10^{7}
3.52	1.44×10^{8}	3.19	9.72×10^{6}
4.11	2.33×10^{6}	4.68	1.34×10^{6}

II Containment Atmosphere

Time = 0		Time = 8 hrs	
Energy (MEV)	γ/cc-sec	Energy (MEV)	γ/cc-sec
.13	6.61×10^{7}	.095	3.56×10^{7}
.36	1.09×10^{8}	.31	2.75×10^{7}
.74	1.38×10^{8}	.61	2.25×10^{7}
1.24	3.79×10^{7}	1.25	9.02×10^{6}
1.75	2.64×10^{7}	1.72	2.31×10^{6}
2.19	4.42×10^{7}	2.28	3.75×10^{6}
2.57	9.29×10^{6}	2.57	1.39×10^{5}
3.52	4.08×10^{5}	3.52	1.14×10^{1}
4.09	6.42×10^{3}	4.09	1.80×10^{-1}

2.1.2.5 Reactor Coolant Pump Trip on HPI

2.1.2.5.1 System Description

The purpose of this proposed modification is to provide automatic trip of the Reactor Coolant Pumps when degraded primary system conditions associated with a LOCA have been detected. This will be accomplished by requiring that RCP trip be initiated when the Engineered Safeguards System has actuated Safety Injection and saturation margin has been lost.

The proposed logic will preclude RC pump trip during those events such as severe overcooling or very small breaks where maintenance

of forced cooling is very desirable. The conceptual design described in this section is being submitted for NRC review and comment and will be implemented subject to concurrence of the NRC staff.

2.1.2.5.2 Design Bases

See response to Question 11 of Supplement 1, Part 3.

2.1.2.5.3 System Design - To be provided later.

2.1.2.5.4 Design Evaluation - To be provided later

2.1.2.6 Auxiliary Feedwater System

Auto start of the emergency-feedwater (EFW) System is being implemented in two phases: 1. Control Grade Auto Start - This is a non-safety related initiation as described in paragraph 2.1.1.7 and it is a short-term approach, 2. Safety Grade Auto Start - This will be a long-term modification where the initiation will meet the requirements for Class 1E system and the system is functionally described below.

1. The safety grade EFW auto start when implemented will automatically initiate the system on presence of the following conditions with or without the availability of the off-site power:

   ° Loss of both normal feedwater pumps, or
   ° Loss of all four reactor coolant pumps, or
   ° Low differential pressure between the normal feedwater and main steam lines at either steam generator,

   The system initiation on low steam generator level will eventually be added. This will be done after the necessary analysis and engineering has been completed to insure that this signal will give a satisfactory actuation and will not interact with other plant functions. Loss of normal feedwater pumps is detected by differential pressure switches across each pump (two switches per pump, i.e., one switch per train).

   The model of differential pressure switches used for this application has been seismically tested. These switches have temperature limits of -60 to 200°F. Since they will be located in Turbine Building which is a non-seismic building, the switches will be tied into their respective EFW intiating circuits (Train A&B) through buffer devices and thus the switches will be treated as safety grade items to the extent possible. The buffer devices are relays similar to those described in the TMI-2 FSAR Section 7.3.2. This application is similar to the approved application described in that section.

2. All cables associated with the initiating logic will be qualified for Class 1E application and the initiations will be designed to meet single failure criteria. All circuits will meet the regulatory criteria for separation of Class 1E circuits.

3. The initiating logic will include hardware for the following purposes:

° Latching mechanism to seal-in the actuation
° Manual Bypass Capability
° Testability of the initiating circuit

4. Indication will be provided in the control room to identify the source of the initiation.

5. Annunciation will be provided in the control room to alarm:

° Auto start of the EFW system. This will be a common alarm for both the trains.

° Initiating conditions being bypassed. This will be a common alarm for all initiating conditions associated with with the same train.

2.1.2.7 Increased Range of Radiation Monitors (2.1.8.b)

2.1.2.7.1 The existing Radiation Monitoring System provides in-line monitoring capability for effluents from:

a) Auxiliary and Fuel Handling Building (RM-A8)
b) Reactor Building Purge (RM-A9)
c) Condenser Off-Gas (RM-A5)

Discharge from Waste Gas Decay Tanks is monitored by RM-A7 prior to combination with other exhaust and after dilution by RM-A8. The Reactor Building Hydrogen Purge System discharge is monitored by the normal purge system monitor RM-A9.

The monitors, RM-A8 and RM-A9 are manufactured by Victoreen, Inc. and consist of:

a) A fixed filter particulate monitor; Beta scintillation detector; sensitivity approximately $1.5 \times 10^{10} \frac{cpm/min}{Ci/cc}$ based on SR-90; full range 1×10^6 cpm.

b) A Fixed Charcoal Filter Iodine Monitor; NaI detector with fixed window; Sensitivity approximately $1.3 \times 10^9 \frac{cpm/min}{Ci/cc}$ fill range 1×10^6 cpm.

c) A gross gaseous monitor; Beta Scintillation detector; Sensitivity approximately $4 \times 10^7 \frac{cpm}{Ci/cc}$; full range 1×10^6 cpm.

d) Air sampling pump with normal sample flow of approximately 1 cubic foot per minute.

Radiation monitor RM-A5 has only a gross gaseous monitor (c above) situated on the discharge of the condenser vacuum pumps, exhausting to the suction of the vacuum pumps. Flow through the monitor is regulated to maintain approximately 500 cc/min. All monitors have Control Room readout and recording.

2.1.2.7.2 Long Term Modifications

Increased range capabilities will be furnished for each of the effluent monitors described above (RM-A8, RM-A9, RM-A5) and the Main Steam lines. For the Long Term Modification additional monitoring ranges will be provided utilizing ionization chambers for the Reactor Building Purge Exhaust, the Condenser OFF-GAS Exhaust, the Main Steam Lines. The Auxiliary and Fuel Handling Building Exhaust will have extended monitoring ranges incorporating a G.M. device. The sensitivity of the individual units will be determined by standard volume source calculations.

The sensitivity will assure that release rate of:

5,600,000 Ci/sec from Auxiliary & Fuel Handling Building

2,300,000 Ci/sec from Reactor Building Purge.

1,400 Ci/sec from Condenser Off-Gas based on minimum flow rates from each release path.

2,500 Ci/sec from a single steam generator can be detected.

The installation of each monitor will include evaluation of the position of the monitor relative to other potential radiation sources and shielding necessary to minimize the effect of sources other than sample lines on the response of the monitor and recording.

For each of the monitors described, the following applies:

Each will be powered from vital power, thereby providing redundancy in power supply.

Establishing sensitivities will be correlated to solid source calibrations. Procedures defining calibration method and frequency will be written to assure proper response of the instruments.

Emergency procedures will be written to the use of the radiation instrumentation in conjunction with flow information to determine release rate.

Emergency Plan implementing procedures describe the dissemination of information obtained from monitors.

Procedures and evaluations will be available for NRC review prior to restart.

2.1.2.7.3 Short Term Modifications

Increased range capabilities will be furnished for the Reactor Building Purge Exhaust, the Condenser Off-Gas Exhaust, and the Main Steam Lines as a short term modification. This Short Term Modification will consist of G.M. Tubes or ionization chambers affixed to each of the effluent release paths described

in 2.1.2.1.1 (only one detection system will be provided for each OTSG). Remote readout will be provided to areas which are habitable during an accident. The Long Term Modification for the Auxiliary and Fuel Handling Building is projected to be complete before restart. If a Long Term Modification is not available by start-up, due to equipment delivery problems, a Short Term Modification utilizing a G.M. tube or ionization chamber will be incorporated. All devices will have necessary shielding if background effects are considered excessive.

The installation of each monitor will include evaluation of the position of the monitor relative to other potential radiation sources and shielding necessary to minimize the effect of sources other than sample lines on the response of the monitor.

The sensitivity will assure that release rates of:

5,600,000 Ci/sec from Auxiliary & Fuel Handling Bldg.

2,300,000 Ci/sec from Reactor Building Purge.

1,400 Ci/sec from Reactor Building Purge.

2,500 Ci/sec from a single steam generator can be detected.

The range of these monitors is identical to the range capability of the long term modification.

For each of the monitors described, the following applies:

Each will be powered from normal power with battery backup.

Established sensitivities will be correlated to solid source calibration. Procedures defining calibration methods and frequency will be written to assure proper response of the instruments.

Emergency procedures will be written to the use of radiation instrumentation in conjunction with flow information to determine release rate.

Emergency Plan implementing procedures describe the dissemination of information obtained from the monitors.

Procedures and evaluations for interim methods will be available for NRC review prior to criticality if the long-term modifications are not completed by that time.

THREE MILE ISLAND UNIT NO. 1

TABLE 2.1-1

List of Isolation Signal Override Capability

	Penetration No.	Reactor Trip	High Radiation	4 psig Building	30 psig Building	1600 psig (SFAS)	Line Break
			Isolation Signal				
Containment Air Sample	108	N/A	N/A	C	N/A	C	N/A
R.B. Sump	353	C	IB	C	N/A	N/A	N/A
RCDT	330,331	C	IB	C	N/A	N/A	N/A
RCS Sample	328	C	IB	C	N/A	N/A	N/A
R.B. Purge	336,423	K	K	K	N/A	N/A	N/A
RCS Makeup	323	N/A	N/A	C	N/A	C	N/A
RCS Letdown	309 (MU-V2A/B)	N/A	IB	C	N/A	C	N/A
	(MU-V3)	A	N/A	C	N/A	N/A	N/A
Demin Water	307	C	N/A	C	N/A	N/A	N/A
OTSG Sample	213, 214	C	IB	C	N/A	N/A	N/A
NSCCW	346, 347	N/A	N/A	N/A	NO	N/A	NO
ICCW	302, 333, 334	N/A	N/A	N/A	NO	N/A	NO
R.B. Air Coolers	421, 422	N/A	N/A	C	N/A	C	N/A
R.C. Pump Seal Return	329	N/A	N/A	NA	NO	N/A	N/A
Core Flood TK	348, 349	C	N/A	C	N/A	N/A	N/A

Legend
- C = Common Signal Override; initiating isolation condition may still exist.
- I = Individual isolation signal override capability; procedures governing override to be developed.
- IB = Individual isolation signal bypass capability
- A = Automatic isolation signal override.
- K = Common signal override with key interlock permissive.
- NO = No override or bypass capability; initiating condition must clear to allow reopening of valve.
- N/A = Not applicable.

Note: For combinations of initiating signals that are allowable, refer to Table 2.1-2.

THREE MILE ISLAND UNIT NO. 1
TABLE 2.1-2
LIST OF CONTAINMENT ISOLATION VALVES REQUIRING MODIFICATIONS

Penetration No.	Service	System	Valve Tag No.	Valve Type	Line Size, In.	Method of Actuation	Normal Valve Position	Post Accident Position Existing	Post Accident Position Modified	Valve Actual Position Indication	Valve Actuation Signal Source Existing	Valve Actuation Signal Source Modified	Notes
108	Containment Air Sample	RM	CM-V1	Ball	1	Air	Open	Closed	Closed	Yes	1,10	1,2,6,10	
			CM-V2	Ball	1	Air	Open	Closed	Closed	Yes			
			CM-V3	Ball	1	Air	Open	Closed	Closed	Yes			
			CM-V4	Ball	1	Air	Open	Closed	Closed	Yes			
213	Steam Generator Sample	CA	CA-V4A	Globe	3/8	EMO	Closed	Closed	Closed	Yes	1,10	1,4,5,6,10	No B&W recommendation
			CA-V5A	Globe	3/8	Air	Closed	Closed	Closed	Yes			
214	Steam Generator Sample	CA	CA-V4B	Globe	3/8	EMO	Closed	Closed	Closed	Yes	1,10	1,4,5,6,10	
			CA-V5B	Globe	3/8	Air	Closed	Closed	Closed	Yes			
302	Intermediate Cooling Water Outlet Line	IC	IC-V2	Gate	6	EMO	Open	Closed	Open/Closed	Yes	1,3,10	3,7,8,9,10	
			IC-V3	Gate	6	Air	Open	Closed	Open/Closed	Yes			
307	Demin. Water to Reactor Building	CA	CA-V189	Gate	2	Air	Open	Closed	Closed	Yes	1,10	1,5,10	
309	Letdown Line to Purification Demineralizers	MU	MU-V2A	Globe	2-1/2	EMO	Open	Closed	Open/Closed	Yes	1,10	1,2,4,6,10	
			MU-V2B	Globe	2-1/2	EMO	Open	Closed	Open/Closed	Yes	1,10	1,2,4,6,10	
			MU-V3	Gate	2-1/2	Air	Open	Closed	Open/Closed	Yes	1,10	1,5,6,10	
323	RC Makeup	MU	MU-V18	Gate	2-1/2	Air	Open	Closed	Closed	Yes	1,10	1,2,10	
328	Pressurizer and Reactor Coolant Sample Lines	CA	CA-V1	Globe	3/8	EMO	Closed	Closed	Closed	Yes	1,10	1,4,5,6,10	
			CA-V2	Gate	3/8	Air	Closed	Closed	Closed	Yes			
			CA-V3	Globe	3/8	EMO	Closed	Closed	Closed	Yes			
			CA-V13	Globe	3/8	EMO	Closed	Closed	Closed	Yes			
329	Reactor Coolant Pump Seal Return	MU	MU-V25	Globe	4	EMO	Open	Closed	Open/Closed	Yes	1,7,10	3,7,8,10	
			MU-V26	Gate	4	Air	Open	Closed	Open/Closed	Yes			
330	Reactor Coolant Drain Tank Vent	WDG	WDG-V3	Globe	2	EMO	Open	Closed	Closed	Yes	1,10	1,4,5,10	
			WDG-V4	Gate	2	Air	Open	Closed	Closed	Yes			
331	Reactor Coolant Drain Tank Pump Discharge	WDL	WDL-V303	Gate	4	EMO	Closed	Closed	Closed	Yes	1,10	1,4,5,10	
			WDL-V304	Gate	4	Air	Closed	Closed	Closed	Yes			
333	Intermediate Cooling Water Supply Line	IC	IC-V4	Gate	6	Air	Open	Closed	Open/Closed	Yes	1,3,10	3,7,8,9,10	
334	Intermediate Cooling to CRDM Cooling Coils	IC	IC-V6	Gate	3	Air	Open	Closed	Open/Closed	Yes	1,3,10	3,7,8,9,10	

THREE MILE ISLAND UNIT NO. 1
TABLE 2.1-2 (CONT'D)
LIST OF CONTAINMENT ISOLATION VALVES REQUIRING MODIFICATIONS

Penetration No.	Service	System	Valve Tag No.	Valve Type	Line Size, In.	Method of Actuation	Normal Valve Position	Post Accident Position Existing	Post Accident Position Modified	Valve Actual Position Indication	Valve Actuation Signal Source Existing	Valve Actuation Signal Source Modified	Notes
336	Reactor Building Outlet Purge Line	AH	AH-V1A	Butter-fly	48	Air	Closed	Closed	Closed	Yes	1,10	1,4,5,10	
			AH-V1B	Butter-fly	48	EMO	Closed	Closed	Closed	Yes			
346	Reactor Coolant Pump Motor Cooling Water Supply	NS	NS-V15	Gate	8	EMO	Open	Closed	Open/Closed	Yes	1,10	7,8,9,10,	
347	Reactor Coolant Pump Motor Cooling Water Return	NS	NS-V4	Gate	8	EMO	Open	Closed	Open/Closed	Yes	1,10	7,8,9,10	
			NS-V35	Gate	8	EMO	Open	Closed	Open/Closed	Yes	1,10	7,8,9,10,	
353	Reactor Building Sump Drain	WDL	WDL-V534	Gate	6	Air	Closed	Closed	Closed	Yes	1,10	1,4,5,10	
			WDL-V535	Gate	6	Air	Closed	Closed	Closed	Yes			
421	Reactor Building Normal Air Coolers Supply Line	RB	RB-V2A	Gate	8	EMO	Open	Closed	Open	Yes	1,10	1,2,10;	Add auto initiation of Emerg. R.B. cooling on 4 psig R.B. and 1600 psig R.C. pressure isolation signals.
422	Reactor Building Normal Air Coolers Return Line	RB	RB-V7	Gate	8	Air	Open	Closed	Open	Yes	1,10	1,2,10;	
423	Reactor Building Inlet Purge Line	AH	AH-V1C	Butter-fly	48	EMO	Closed	Closed	Closed	Yes	1,4,10	1,4,5,10	
			AH-V1D	Butter-fly	48	Air	Closed	Closed	Closed	Yes			
348,349	Core Flood TK. Sample and N_2 Fill Lines	CF	CF-V2A&B	Globe	1	EMO	Closed	Closed	Closed	Yes	1,10	1,5,10	
			-V19A&B	Gate	1	Air							
			-V20A&B	Gate	1	Air							

Valve Actuation Signal Source

1) 4 psig reactor building pressure isolation
2) 1600 psig (SFAS) isolation
3) Radiation alarm, operator action required
4) High radiation (non-safety) isolation
5) Reactor trip isolation
6) Override capability on individual valves
7) Classify line to Seismic Category I
8) 30 psig reactor building pressure isolation
9) Line break isolation signal
10) Remote manual control

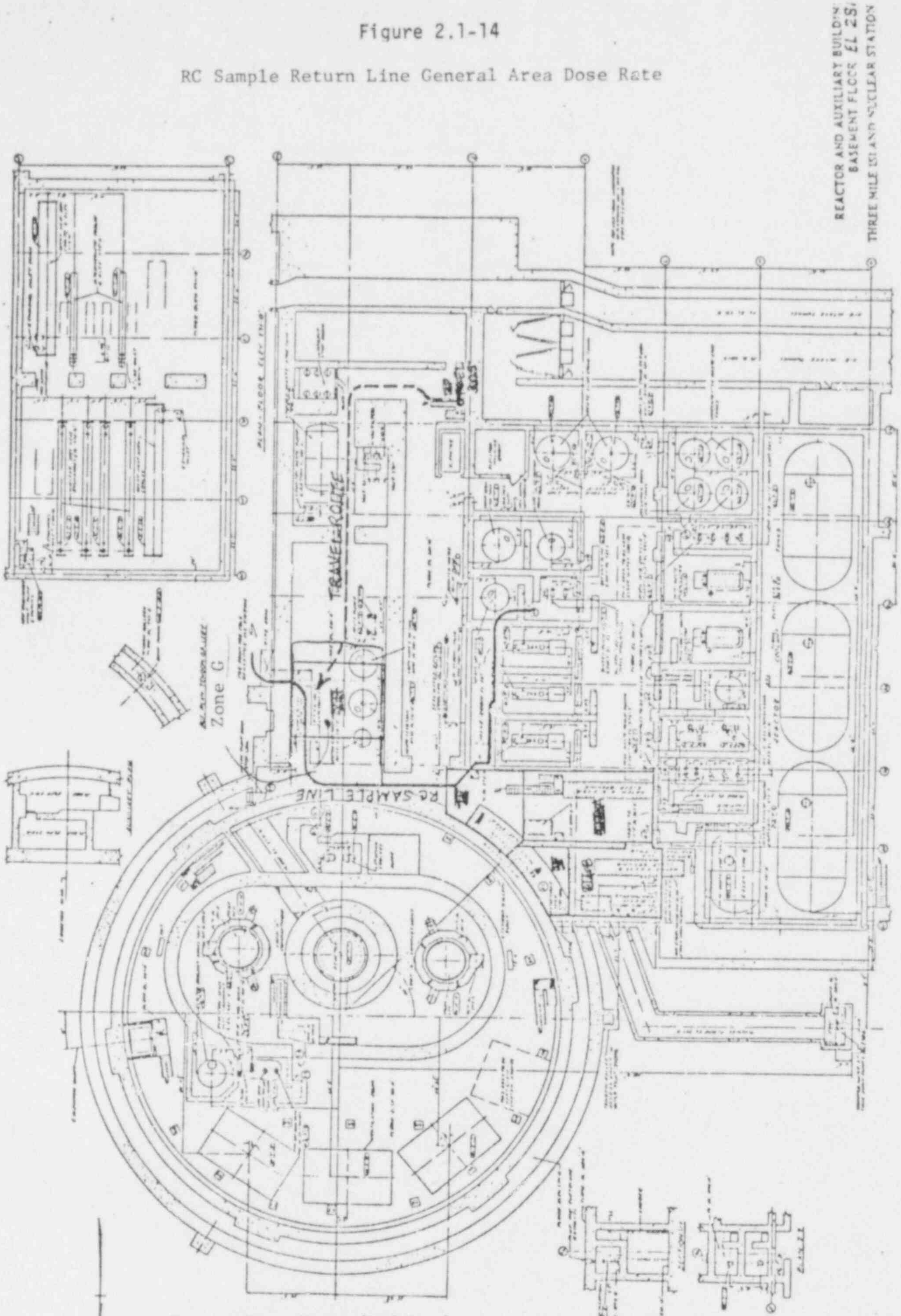
TABLE 2.1-3

LIST OF CONTAINMENT PENETRATIONS REQUIRING ISOLATION ON HI-RADIATION

Penetration No.	Service	System	Isolation Valve Tag No.	Radiation Detector Location	Type of Monitor
213 and 214	Steam Generator Sample	CA	CA-V4A -V5A -V4B -V5B	Locate the monitors outside the R.B. near the sampling line downstream of the containment isolation valve and upstream of connection for Turb. Plant sampling	Area Gamma Detectors (New)
309	Letdown Line to Purification Demineralizers	MU	MU-V2A -V2B	Utilize existing Rad. Monitor RM/L-1 located outside R.B.	Inline (Existing)
328	Pressurizer and Reactor Coolant Sample Lines	CA	CA-V1 -V2 -V3 -V13	Locate the monitor outside the R.B. between the isolation valve and the sample cooler.	Area Gamma Detector (New)
329	Reactor Coolant Pumps Seal Return	MU	MU-V33A -33B -33C -33D Or MU-V25 -V26	Locate the online radiation monitor downstream of the containment isolation valves outside of the R. B. for Alarm Operator action is required to close valves.	Area Gamma Detector (New)
330 and 331	Reactor Coolant Drain Tank Vent Reactor Coolant Drain Tank Pump Discharge	WDG WDL	WDG-V3 -V4 WDL-V303 -V304	Locate the monitor on the outside of the tank.R.B. Strap monitor onto vent and drain lines are near each other	Area Gamma Detector (New)
336 and 423	Reactor Building Outlet and Inlet Purge Lines	AH	AH-V1A -V1B -V1C -V1D	Utilize the existing purge outlet line Rad. Monitor RM/A-9 located outside of R.B.	Inline (Existing)
353	Reactor Building Sump Drain	WDL	WDL-V534 -V535	Locate an area radiation monitor in the R.B. Sump mounted inside a seismically supported pipe.	Sump Area Monitor (New)
302 333 and 334	Intermediate Cooling Supply & Return	IC	IC-V2,3 -V4,6	Locate the radiation monitor on the 6" IC return line between valve IC-V3 and the 2" pump recirc. line	Incline (Existing)

Figure 2.1-14

RC Sample Return Line General Area Dose Rate

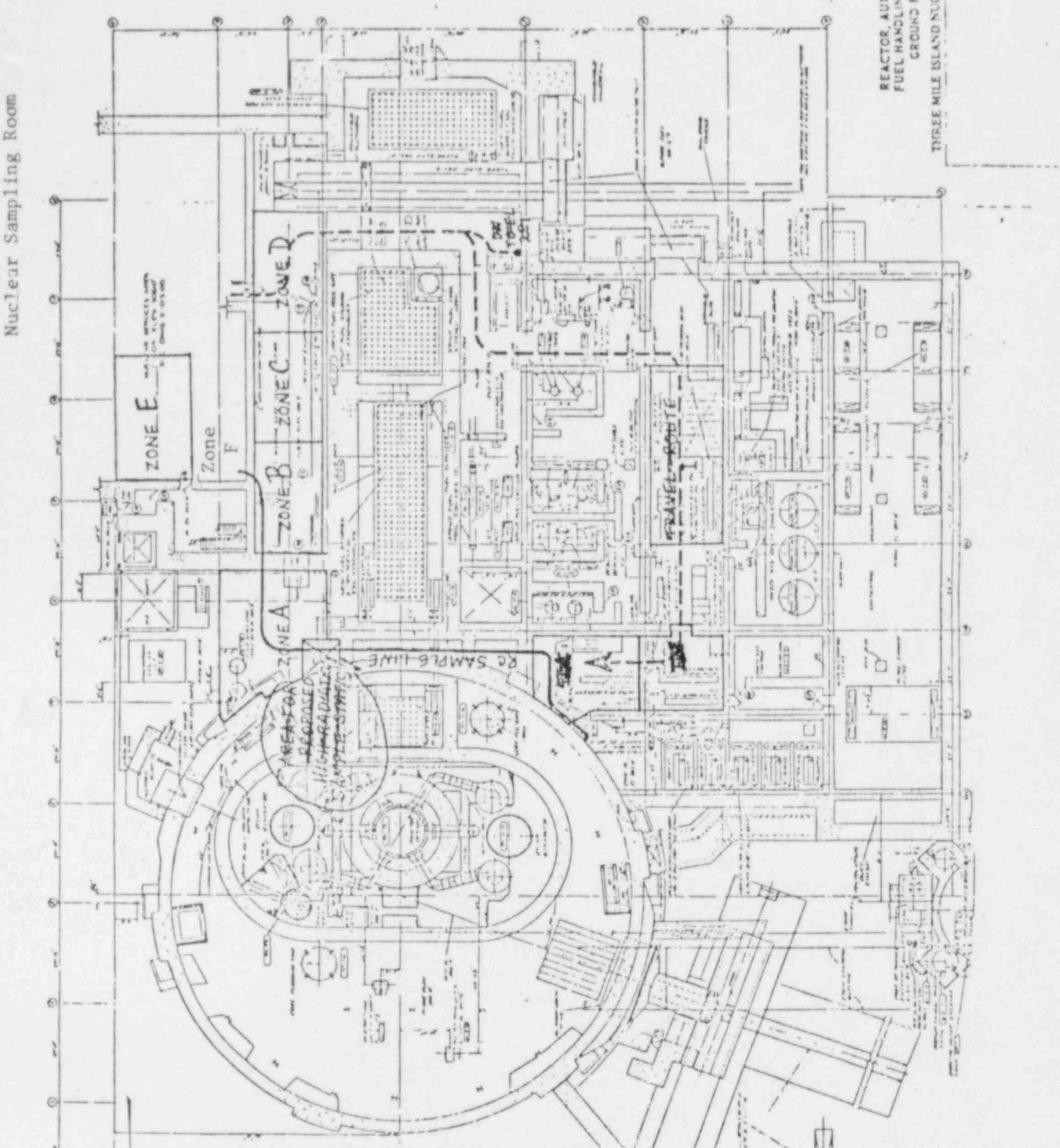


Zone	Dose Rate (T=8 hrs)
G	4.75 REM/hr

Figure 2.1-15

RC Sample Dose Rate

Line General Area



Zone	Dose Rate (T=8 hrs)
A	15.2 REM/hr
B	2.5 REM/hr
C	.08 REM/hr
D	.006 REM/hr
E	.06 REM/hr
F	55.0 Rem/hr

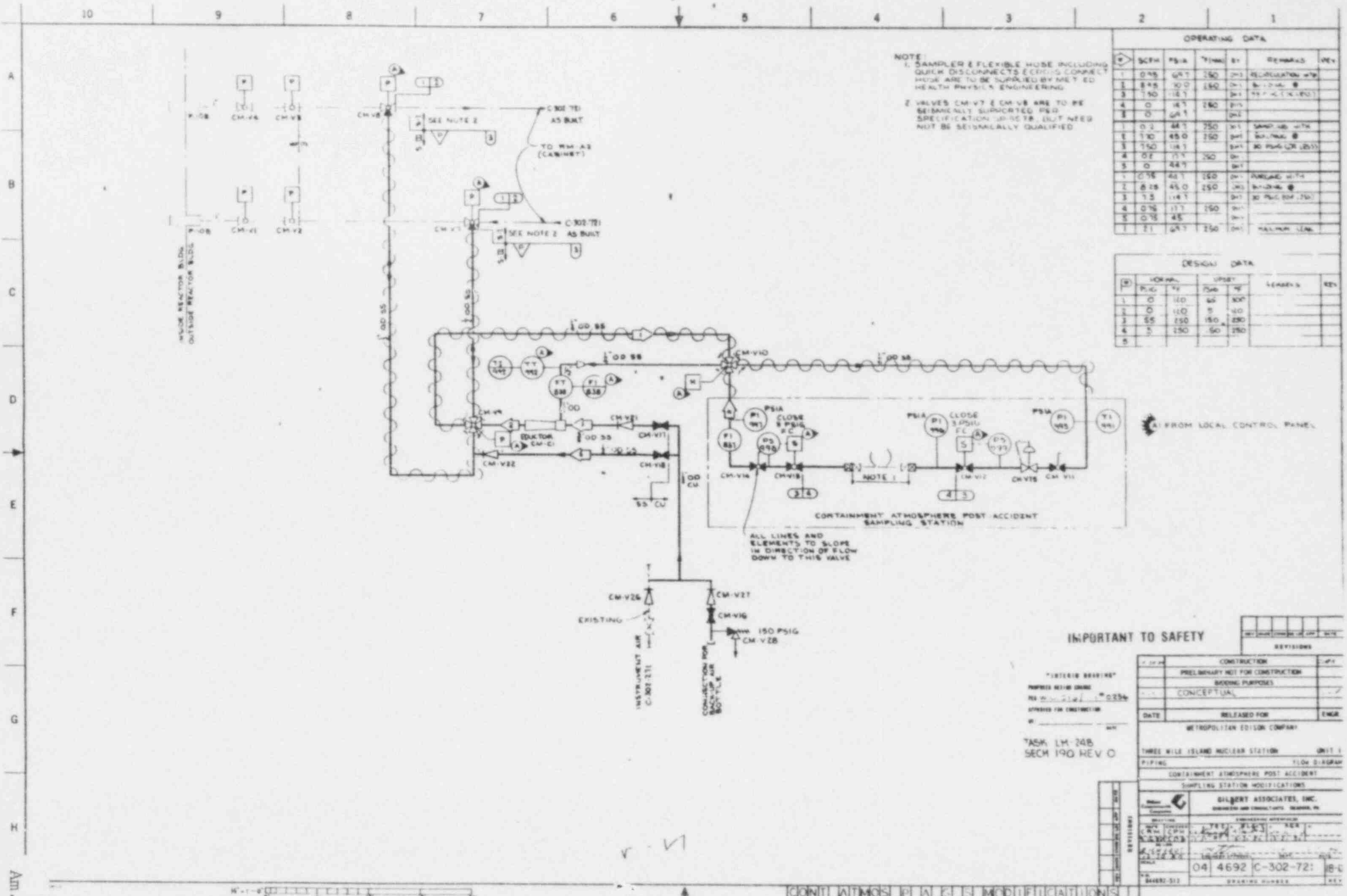


Figure 2.1-16

Am. 23

APPENDIX 2A

1. BAW-1603, "Pilot Operated Relief Valve (PORV) Monitor Test," (Proprietary Report - forwarding letter only attached).

2. TDR-240, "Analysis of Two-Phase Flow Induced Differential Pressure Across TMI-1 Pressurizer Safety Valve Line Elbow Taps," Rev. 0.

5.2.2 Operations and Maintenance Director, TMI-1

a. Function

The Director reports directly to the Vice President TMI-1 and assists him in the overall operation of TMI-1.

b. Responsibility

This position has direct responsibility for operating the unit in a safe, reliable and efficient manner; is responsible for off site radioactive discharges and bears the responsibility for compliance with the operating licenses and the rules and regulations of the Commonwealth of Pennsylvania; supervises the Operations Group and Maintenance Group and the Radioactive Waste Processing and Shipment Group.

c. Authority

The authority of the Director, to act on behalf of the Vice President, TMI-1, is inherent in the position and commensurate with the assigned responsibilities. It includes the authority to order the shutdown and cooldown of TMI-1 whenever the health and safety of the public is endangered or when in his judgement a shutdown is warranted. It also includes the authority to issue procedures, orders, and other directives required in the execution of the assigned responsibilities. Necessarily included in the responsibility for plant operation, compliance with Technical Specifications, is the authority to assign and prioritize requirements to the Plant Engineering, Training and Administration and Services Groups. Similarly, the authority of the Director includes the initiation and prioritization of corrective maintenance and preventative maintenance in the execution of his responsibilities. The Director may delegate his authority to the Manager, Plant Operation, TMI-1 or Shift Supervisor during absences. This delegation of authority extends to the issuance of standing orders and directives in support of the responsibilities assigned. In the absence or incapacitation of the Vice President TMI-1, this Director is delegated the authority of that office for the centralized control, supervision, coordination and planning of all aspects of TMI-1 Operations.

d. Minimum Qualifications

The Operations and Maintenance Director, TMI-1 shall possess a Bachelors degree in Science or Engineering and ten years of responsible power plant experience of which at least three years will be in nuclear power plant design, construction, startup, operation, maintenance, or technical services. A maximum of four years of the remaining seven may be fulfilled by academic training. This Director shall have acquired the experience and training normally required for examination by the NRC for a SRO license whether or not the examination is taken.

f. Interfaces

1. Offsite

The Director interfaces with company, corporate, local Commonwealth of Pennsylvania, and Federal Government organizations in fulfillment of responsibilities assigned, State and Federal regulations, and directives received.

5.2.3 Manager, Plant Operations TMI-1

a. Function

The Manager, Plant Operations TMI-1 has the responsibility for directing the actual day-to-day operation of the unit. He reports directly to the Operations and Maintenance Director, TMI-1. The Manager, Plant Operations TMI-1 coordinates operations and related maintenance activities with the Manager, Plant Maintenance TMI-1.

b. Responsibility

This position is responsible for the day-to-day direction of the Operations personnel to ensure compliance with the conditions of the plant operating license and the Technical Specifications. He is also responsible for the supervision of the TMI-1 Radioactive Waste Processing and Shipment Group.

c. Authority

The Manager, Plant Operations TMI-1 has the authority to shutdown and cooldown of TMI-1 whenever the health and safety of the public is endangered or when, in his judgement, a shutdown is warranted.

d. Minimum Qualifications

The Manager, Plant Operations TMI-1 will have a minimum of eight years of responsible power plant experience of which at least three years will be in nuclear power plant design, construction, startup, operations, maintenance, or technical services. A maximum of two years academic or related training may be included as part of the remaining five years of power plant experience. The Manager, Plant Operations TMI-1 shall hold a Senior Reactor Operators License.

e. Incumbent Qualifications

Education:	High School Graduate 1960
Military Service:	U.S. Navy - 1960-1968

FIGURE 5.3-1

GPU NUCLEAR CORPORATION

- OFFICE OF THE PRESIDENT
  - President
  - Executive Vice President
- Chairman - General Office Review Board
- Vice President Oyster Creek
  - Operation
  - Maintenance
  - Plant Engineering
  - Maintenance & Construction Oyster Creek
  - Forked River
- Vice President TMI-1
  - Operation
  - Maintenance
  - Plant Engineering
- Vice President TMI-2
  - Operation
  - Maintenance
  - Plant Engineering
  - Decontamination
- Vice President Technical Functions
  - Systems
  - Eng. & Design
  - Licensing
  - Proj. Engineering
  - Startup & Test
  - Engineering Services
- Vice President Nuclear Assurance
  - Qual. Assur.
  - Training
  - Nuclear Safety Assessment
  - Emergency Planning
  - System Lab.
  - IOSRG
- Vice President Administration
  - Fiscal Mgmt Information Sys.
  - Human Resources
  - Security & Facilities & Industrial Safety
  - Legal Services
- Vice President Communications
  - External Communication
  - Internal Communication
- Vice President Radiological & Environmental Controls
  - Hqtrs Radcon
  - Hqtrs Environment
  - Site Radcon
  - Site Environment
- Vice President Maintenance & Construction
  - Maint. & Constr. - Oyster Creek
  - Maint. & Constr. - TMI-1
  - Maint. & Constr. - TMI-2

Further contributing to the availability and security of the liquid waste system is the fact that all of the above equipment is located within Seismic Class I structures that have been hardened to withstand an aircraft impact. Within these structures, all equipment that is anticipated to become a significant radiation source is housed within 2 to 3 foot thick shield walls for the protection of plant personnel from radiation. The atmosphere of each of these shielded cubicles is maintained at a slightly lower pressure than that in surrounding areas to ensure that any radioactive gas leakage is away from plant personnel.

Based on the above indicated systems and equipment, the design basis waste liquid quantities generated annually are 49,000 μCi of mixed fission products (excluding tritium) and 5.02×10^8 μCi of tritium. With 17,500 gpm of the cooling tower effluent allocated to Unit 1, Unit 1 annual discharge volume for which dilution credit may be taken is 3.48×10^{10} liters. This results in an annual average concentration of mixed fission products (excluding tritium) and tritium in the plant effluent of 1.4×10^{-6} μCi/liter and 1.45×10^{-2} μCi/liter respectively. This annual average concentration of mixed fission products (including tritium) is within Appendix I to 10CFR 50 guidelines.

7.3.1.1.3 Epicor I Liquid Radwaste Treatment System

7.3.1.1.3.1 System Function and Design Objectives

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

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

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

The discharge of radioactive gases is minimized.

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

d. After each complete or partial replacement of charcoal adsorber bank by verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975.

7.3.3.3 Implementation Schedule

The testing schedule as previously described will be in place prior to the restart of Unit-1.

7.3.4 Nuclear Sampling

7.3.5 Nuclear Sampling Capabilities

7.3.5.1 Post-Accident Sampling

See section 2.1.2.4.1

7.3.5.2 Improved In-Plant Radioiodine Monitoring Instrumentation

See section 2.1.2.1.1.

7.4 Affect of TMI-2 Recovery on TMI-1 Operation

Activities in TMI-1 related to radwaste processing and activities in the common fuel handling building will not be affected by the TMI-2 recovery program. As demonstrated in Section 7.2.3, TMI-1 does not have to rely on any TMI-2 facilities for the processing of radwaste. Section 7.2.1 describes specifics to be taken to isolate the radwaste piping systems of the two units. Through the isolation of the piping system, interface between the two unit's radwaste systems will be eliminated. Waste processing activities related to TMI-2 will be performed in the fuel handling building during the recovery program. These activities will not affect activities in the auxiliary building because the areas will be separated by an environmental barrier (Section 7.2.2). Communication of the air spaces (TMI-1/TMI-2) of the fuel handling building will be minimized with appropriate modifications of the ventilation equipment in the building (See Supplement 1, Part 2, Question 52). Continuous access to the fuel handling building is not required for the safe operation of TMI-1 (with environmental barrier installed).

ii. Offsite doses are within the limits of 10CFR100.

iii. Radially averaged enthalpy should not be greater than 280 cal/gm at any axial location in any rod.

3. Mitigation

i. The power excursion is limited by the Doppler coefficient.

ii. The power excursion is terminated by reactor trip on high pressure or high flux.

4. Conclusions

The lower high pressure trip setpoint results in increased safety margins over the FSAR analysis. Improvements to the containment isolation signal (radiation +Rx trip) make release of fluid from the containment building less likely.

8.3.13 Feedwater Line Break Accident (TMI-2 FSAR, Section 15.1.8 S3-21.49, S2-21.43, Reference 2, Q.3 of Supplement 1, Part 2)

1. Description

This event has not been analyzed for TMI-1. The following description is based on FSAR analyses for TMI-2. A loss of feedwater flow results in a loss of heat sink, primary system heatup, increased pressurizer level and pressure, and reactor trip on high RCS pressure. The TMI-2 analysis assumes a complete loss of feedwater due to a break upstream of the first feedwater line check valves. No analysis of loss of feedwater due to pump trip or valve closures were analyzed. The loss of feedwater flow due to the postulated break is analyzed as an immediate loss of flow, which results in a bounding analysis for loss of feedwater events. The reactor is initially at 2772 Mw(t). Assumptions were made to provide two worst case scenarios-one for containment, and one for primary system conditions.

A double ended rupture (with a blowdown area limited by the feedwater header area) was analyzed; steam generators are assumed to have a fouled inventory of 62,500 lbs., and emergency feedwater is assumed to be at full flow within 40 seconds. The loss of feedwater is not directly calculated but taken as a conservative loss of heat demand (100-0% in 5 seconds for the affected generator and 100-0% in 20 seconds for the unaffected generator).

Reference 2 and Question 3, Supplement 1, Part 2 provide results for a loss of normal feedwater event. Table 8-2 compares the analysis assumptions to the plant design.

2. Acceptance Criteria

i. Core thermal power shall not exceed 112% of rated power.

ii. Reactor coo ant system pressure shall not exceed code allowable limits of 2750 psig.

iii. Pressurizer does not become water solid during a loss of feedwater transient.

3. Mitigation

i. Reactor coolant system trip on hig pressure.

ii. The secondary system heat sink is restored by initiation of emergency feedwater to full flow within 40 seconds. Heat removal is through the turbine bypass valves or main steam relief valves.

4. Conclusions

Results of the TMI-2 feedwater line break accident will become bounding for Unit 1 with the addition of a feedwater line break initiating signal. The addition of reactor trip on loss of feedwater increases the safety margin over the TMI-2 analysis. Lowering of the high pressure trip setpoint also increases safety margins since reactor trip will be initiated sooner. The RCS heatup is thus reduced. PORV operation was not assumed in the fe line break analysis, so that the increase in the valve set, : does not affect analysis results. The PORV would a..uate for the worst case feedline break accident analyzed in the TMI-2 FSAR.

Analyses are being performed by B&W to determine if a low level initiating signal would be suitable for auto EFW initiation during a feed line break accident. The B&W analysis will take into account the lower thermal power level of TMI-1, as well as modeling of the OTSG in a more complete manner so that the heat demand will not be estimated so conservatively. Furthermore, the back area will be based on the cross-sectional area of the feedwater injection nozzles rather than the cross-sectional area of the feedwater heater. This reduced blowdown area, along with the more detailed OTSG modeling is expected to reduce the severity of the feedwater line break substantially (refer to Appendix 8A, Figures 8A-21 and 22 for an example of expected results).

As demonstrated by Table 8-2 and Q3 Supp 1, Part 2, TMI-1 meets the acceptance criteria for a loss of Feedwater transient and the analysis bounds the TMI-1 plant design.

8.3.14 Waste Gas Decay Tank Rupture (FSAR Section 14.2.2.5)

1. Description

The rupture of a waste gas decay tank would result in radiological releases via the plant ventilation system. The tank contents as calculated assuming the activity evolved from degasing the primary coolant system after operation with 1% failed fuel.

2. Acceptance Criteria

Doses shall not exceed the limits of 10CFR100.

3. Mitigation

Elevated release of activity from the unit vent.

4. Conclusions

This analysis has not been changed as a result of any plant modifications.

8.3.15 Small Break Loss of Coolant Accidents (LOCA)

1. Description

Small break LOCA's are piping ruptures whose break areas range from as small as 0.005 $ft.^2$ to as large as 0.5 $ft.^2$. These LOCA's may or may not involve depressurization of the Reactor Coolant System (RCS).

2. Acceptance Criteria

i. Local fuel cladding oxidation (metal water reaction) shall not exceed 0.17 times the total cladding thickness, or .05 the overall cladding mass.

ii. Peak Cladding Temperature (PCT) shall not exceed 2200°F.

iii. A coolable geometry shall be maintained.

iv. Long term cooling shall be assured.

3. Mitigation

i. Inventory will be maintained by the high pressure injection system.

ii. Emergency Feedwater flow within 20 minutes of very small break LOCA's allows depressurization of the RCS and allows sufficient inventory addition by the HPI system to maintain core cooling.

4. Conclusion

Pursuant to NRC regulations (10CFR50.46) and 10CFR50 Appendix K) B&W performed generic LOCA analyses of their 177 fuel assembly lowered loop plants. Initially this work was performed to meet the Interim Acceptance Criteria (IAC) and documented in BAW-10052. Later, the analyses were revised to the Final Acceptance Criteria (FAC) using the approved Appendix K model (BAW-10104). The FAC analysis results were documented in BAW-10103.

The work performed for BAW-10052 was used as the basis for the small break LOCA lcoation and size sensitivity study and therefore no new work was performed for BAW-10103 other than analysis of three specific break sizes and locations (0.04 $ft.^2$, 0.44 $ft.^2$ and 0.5 $ft.^2$ break sizes).

In April 1978, B&W identified an error in their ECCS model. The error was also evident in the model used for the BAW-10052 sensitivity studies and therefore the basis for the acceptability of the small break analysis was eliminated. B&W performed additional small break studies using the corrected model. The revised analyses are documented in a letter from J. H. Taylor, B&W to S. A. Varga, NRC dated July 18, 1978. These analyses cover break sizes 0.04, .055, .07, .085, 0.1, 0.15, 0.2, 0.3, 0.13, and 0.17 $ft.^2$.

Key assumptions for the small break LOCA analyses versus the TMI-1 plant design are given below:

	BAW-10103 Generic	TMI-1
Reactor Power (MWt)	2772	2335
Reactor Trip (psig)	1900	1900
RC Pumps (LOOP)	Coastdown	Coastdown
AFW Available**	Yes-40 sec.	Yes****
ESFAS HPI (psig)	1600	1600
Operator Action	Yes-cross-connect	none***
HPI Distribution	70% to Core within 10 min.	70% to core from time zero***
HPI Flow (gpm)	450 at 600 psig	500 at 600 psig*****

** Amount assumed for generic analyses 550 gpm. The resonse to Supplement 1, Part 2 Question 4 demonstrates that 500 gpm is

the minimum EFW required for TMI-1. TMI-1 is capable of delivering this minimum under the worst case single failure. Results of Reference 2 demonstrate that EFW is not required before 20 minutes.

*** Prior to startup TMI-1 will install HPI injection leg cross connects and flow control devices to eliminate operator action to cross connect HPI and equalize flow in all four injection legs.

**** For worst case LOCA in which offsite power is lost, EFW is initiated by the loss of feedwater or by the loss of 4 reactor coolant pump signals.

***** Also refer to the response to supplement 1, Part 3 Questions 1, 2 and 3.

In all cases, TMI-1 plant specific information is as conservative or more conservative than the generic assumption.

Since the TMI-2 accident, greater focus has been placed on small break LOCA's and the capability of the ECCS to mitigate them. Problems such as those discussed in Reference 21 (where the pressurizer stays full due to the loop seal arrangement despite loss of RCS inventory) have been addressed. These studies are documented in B&W's "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant" May 7, 1979 (Reference 2). Breaks of 0.01, 0.02, and 0.07 ft.^{2} are analyzed utilizing varying assumptions on the availability and timing of AFW and HPI. These analyses use the same initial assumptions as used in BAW-10103 except that ESFAS is assumed to occur at 1350 psig. Therefore, they are also bounding assumptions for TMI-1 except for the distribution of HPI flow as discussed below. The analysis in Reference 2 also established that EFW flow is not required less than 20 minutes before any steam line break accident.

In Reference 2, credit is taken for operator action to initiate HPI or EFW. No mention is made as to whether whether operator action includes the time necessary to cross connect HPI as required in B&W's other small break accident analyses. TMI-1 will complete the installation of permanent cross connection of the HPI prior to startup, therefore, operator action will not be necessary. All of the B&W small break LOCA analyses assume essentially equal backpressure for all four HPI injection points. This assumption is the basis for the 70%/30% flow split of HPI (assuming a single failure of one HPI train) between the core and the break respectively, after cross connection is accomplished. Such an equal backpressure would not exist given an HPI line rupture. The back pressure on the broken HPI leg would be essentially zero and therefore the HPI loss out the break could be high resulting in inadequate injection to the core.

The criterion established by B&W for the small break analysis requires that 70% of the total flow for one HPI pump be injected

into the broken legs of the reactor coolant system. This criteria applies to a 2772 MW thermal 177 fuel-assembly plant. For TMI-1 with a licensed core power of 2535 MWt, the 70% - 30% criterion can be relaxed in direct proportion to the power reduction. This is justified based on the fact that the decay heat load following a small break LOCA is proportional to power and therefore cooling requirements will be directly proportional to the power at which the plant has operated. Therefore, for TMI-1, the acceptable flow split can be relaxed to 64% - 36%. The 64%/36% flow split would not be obtained for an HPI line break as explained above. Therefore, operator action would be required to isolate the ruptured HPI line. The need to isolate could be determined by observing the individual flow indicators for the HPI legs. The high flow leg would then be isolated. This action would be contrary to the operators instinct and would require considerable judgment since the initial flow imbalance may not be dramatic. Since too great a chance for operator error (error of omission) exists, cavitating venturis will be added to the injection legs to limit flow in the broken leg.

The venturis have been sized to limit flow in each leg to 137.5 gpm when only one high pressure injection pump is operating and Reactor Coolant System is at atmospheric pressure. The venturi design ensures that for the worst case HPI line break condition, the 64%/36% flow split can be achieved when Reactor System Pressure is less than 1500 psig. At RCS pressure conditions greater than 1500 psig, a flow split beyond the 64%/36% acceptance criteria will occur. B&W has reviewed this situation and judged the cavitating venturi performance is acceptable. This conclusion is based on the fact that under HPI line break conditions, the Reactor Coolant System will not expend significant time above 1500 psig and that during the time the RCS is above 1500 psig the cavitating venturi ensures that there is significant flow of high pressure injection into the RC system. B&W also notes that a much larger small break than a HPI line break sets the generic flow split criteria and therefore for a HPI line break the flow split criteria can be relaxed.

In addition to the benefits discussed above, the venturis provide two added benefits. First, they balance flow of the injection legs under all other small break conditions such that TMI-1 flow split will be within the bounds of the generic analysis (i.e., 70%/30% flow split). Secondly, the cavitating can be relaxed.

8.3.16 Large Break Loss of Coolant Accidents (Reference FSAR Section 14.2.2.3)

1. Description

Break sizes in the reactor coolant system (RCS) greater than 0.5 $ft.^2$ are classified as large break loss of coolant accidents (LOCA's). These breaks involve rapid depressurization of the RCS and are accompanied by rapid increases in containment pressure. Offsite doses are calculated from the design basis radioactivity release to containment, and the design basis containment leak rate.

2. Acceptance Criteria

i. Peak fuel clad temperature does not exceed 2200°F.

ii. The core is maintained in a coolable geometry.

iii. Local fuel cladding oxidation (metal water reaction) shall not exceed 0.17 times the total cladding thickness of .05 times the total cladding mass.

iv. Offsite doses are within the limits specified by 10 CFR 100.

3. Mitigation

i. Core flood tank actuation at 600 psig to establish water inventory.

ii. Low pressure injection system flow below 200 psig to establish core cooling for the remainder of the accident.

iii. Building spray addition to put iodine in solution with the containment water volume thus preventing release to the environment.

iv. Containment leak tightness to limit radioactivity releases.

v. Switchover of the decay heat removal system suction source to the containment building sump on low-low BWST level.

4. Conclusion

The calculated offsite dose resulting from the design basis LOCA will increase as a result of the deletion of sodium thiosulfate from the building spray system. Doses will still be within the limits of 10 CFR 100. Dose calculations performed for TMI-2 (see TMI-2 FSAR, Section 15 and Reference 5) demonstrate that design basis LOCA doses are within the limits of 10 CFR 100. The TMI-2 dose calculations were performed taking no credit for sodium thiosulfate. Since Unit 2 has a

slightly large thermal power level and allowable containment leak rate, then Unit 2 dose calculations conservatively bound the worst case LOCA dose for TMI-1.

Automated switchover of the BWST to the recirculation mode provides additional assurance that switchover will occur within the correct level band. Correct operator action had always been assumed in previous LOCA analyses. The automated switchover achieves the same function requirement by means of a safety grade control system.

8.4 SUMMARY AND CONCLUSIONS

Plant modifications to TMI-1 allow the plant analyses to bound the expected plant behavior (see below). In some cases, analysis for TMI-2 have been referenced because they either analyze events that are not in the TMI-1 FSAR (feedline break) or provide additional assurances of safety margins (steam line break).

1. Raising the PORV setpoint and lowering the high pressure trip setpoint affects all of the pressurization transients in the FSAR. Safety margins are improved since the high pressure trip setpoint has been lowered. No credit was taken for operation of the PORV, so that raising the valve setpoint has no effect on the FSAR analysis results.

   The combined effect of the PORV and RPS setpoint changes are to decrease the probability of PORV operation. The integrity of the primary coolant system will be challenged less frequently, so that this change is in the conservative direction. It should be noted that this modification could result in more frequent plant trips.

2. Reactor trip resulting from loss of feedwater results in improved safety margins for loss of feedwater events and does not degrade plant response for any accidents/transients.

3. Reactor trip as a result of turbine trip increases safety margins for the loss of feedwater or feed line break analyses. The effect of retaining or deleting plant features that permitted this event to occur without a reactor trip is being analyzed.

4. The addition of emergency feedwater initiating signals for the feedline break accident makes the TMI-2 feedwater line break accident analysis bounding and conservative for TMI-1. This event has additional safety margins beyond the TMI-2 analysis since both turbine and feedwater trips result in a reactor trip. This earlier reactor trip will result in a smaller heatup of the primary system.

reactor trip. This earlier reactor trip will result in a smaller heatup of the primary system.

5. Modifications to the high pressure injection system will allow adequate HPI flow for the spectrum of LOCA's. System performance is not degraded for any other accidents/transients in which HPI flow is initiated.

6. Upgrading of instrumentation inside containment assures that instrumentation will be functional in the postulated accident environments.

7. Automated switchover of the BWST to the recirculation mode provides additional assurance that switchover will occur within the correct level band. Correct operator action had always been assumed in previous LOCA analyses. The automated switchover achieves the same function requirement by means of a safety grade control system.

8. Dose calculation performed for TMI-2 demonstrate that the requirements of 10CFR100 are met even after sodium thiosulfate is deleted.

9. The transition to natural circulation following a complete loss of feedwater will be demonstrated by a startup test. Reference 2 documents natural circulation tests and natural circulation events at B&W designed reactors. These tests and events demonstrate that natural circulation is a reliable and effective means of core cooling.

10. An analysis of loss of all AC will be performed as part of the B&W Owners Group ATOG program to determine what specific actions would be required to bring the plant to a safe shutdown condition.

11. A PORV setpoint of 2450 psig does not result in unacceptable interactions between the PORV and the pressurizer safety valves, whose setpoint is 2500 psig.

TABLE 8A-1

PROPOSED RETRAN/GPU-01 ANALYSES OF TMI-1

I. Loss of Offsite Power (LOOP)

		Purpose	Results
Case 1:	1840 gpm EFW lowered OTSG setpoint	Show plant response to LOOP and transition to natural circulation.	Figure 8A-20
Case 2:	1840 gpm EFW and 1% decay heat OTSG level control at 12.5 ft. (max. cooldown case)	Examine overcooling potential with 200% EFW and minimum decay heat.	Figure 8A-3
Case 3:	Stuck open OTSG safety valves (17% of design flow)	Examine plant performance for first 10 minutes of secondary side depressurization using one OTSG model.	Figure 8A-16
Case 3a:	Same as Case 3, but with no EFW	Evaluate plant response to an interruption of natural circulation caused by a loss of heat sink.	Figure 8A-2a
Case 4:	500 gpm EFW using flow limitation	Examine long-term plant response with EFW flow limitation and LOOP from 100% power.	Figure 8A-8
Case 5:	1% decay heat with 1145 gpm EFW maximum and OTSG level control at 10 ft.	Examine effect of flow limiters in EFW system on max. cooldown case.	Figure 8A-4
Case 8:	EFW in superheat region	Evaluate effect on transition to natural circulation when model puts EFW in superheat region rather than downcomer. Shows more realistic plant response.	Figure 8A-14
Case 10:	No EFW	Evaluate plant response to loss of heat sink under natural circulation conditions.	Figure 8A-11
Case 15:	1% decay heat with 1840 gpm EFW and OTSG level control at 20 ft.	Show plant response with existing level setpoint and full EFW flow.	Figure 8A-13

TABLE 8A-1

PROPOSED RETRAN/GPU-01 ANALYSES OF TMI-1

(Continued)

		Purpose	Results
II. Station Blackout			
Case 1:	Base	Look for long-term plant response to event, including voiding in the RCS LOOP.	Figure 8A-7
Case 2:	1 gpm pressurizer leakage and letdown isolation at 20 minutes	Effect on plant response due to cooldown of pressurizer steam space. Maximizes expected cooldown.	Figure 8A-12
III. Loss of Feedwater			
Case 1:	460 gpm EFW flow and 1.2 ANS decay heat	Examine plant response to operation of only one motor-driven EFW pump.	Figure 8A-9
Case 1a:	460 gpm EFW flow and .85 ANS decay heat		Figure 8A-10
Case 1b:	460 gpm EFW flow and 1.0 ANS decay heat, no pressurizer spray, reactor trip on high pressure.		Figure 8A-24
Case 2:	EFW flow limitation	Look at plant transition to stable shutdown with EFW flow limited.	Figure 8A-6
Case 3:	Base case 1840 gpm EFW and no equipment failures	Show plant response to LOFW with no equipment failures and no operator actions.	Figure 8A-5
Case 4:	Failure of EFW level control and 1% ANS decay heat	Evaluate effect of continuous EFW flow causing overcooling of OTSG.	Figure 8A-1

TABLE 8A-1

PROPOSED RETRAN/GPU-01 ANALYSES OF TMI-1

(Continued)

III. Loss of Feedwater (Cont'd.)		Purpose	Results
Case 8:	No EFW	Look at time available before HPI must be initiated.	Figure 8A-19
Case 11:	Loss of feedwater with trip on high pressure	Quantify benefit of anticipatory reactor trip.	Figure 8A-15
Case 14:	460 gpm EFW and 1.0 ANS decay heat. EFW trip on low OTSG level, spray on to delay HP trip.		Figure 8A-17
IV. Feed Line Break Accident			
Case 1:	EFW initiation on feed/steam ΔP, 500 gpm	Demonstrate plant can tolerate the design basis feedline break accident.	Figure 8A-21
Case 2:	Delayed EFW	Demonstrate that delay of EFW for greater than 40 seconds still allows an acceptable plant response. Bounds analysis in which EFW is initiated on low OTSG level signal.	Figure 8A-22

TABLE 8A-2

LOSS OF OFFSITE POWER CASE 1

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 5 SEC
EFW MECH	:	AVAILABLE
EFW STEAM	:	AVAILABLE
EFW CAPACITY	:	1840 GPM
ATMOS DUMP	:	AVAILABLE @ 1025 PSIA
TURBINE BYPASS	:	AVAILABLE
SMALL SAFETIES	:	1060/986 PSIA
BANK 1	:	1065/990 PSIA
BANK 2	:	1070/995 PSIA
BANK 3	:	1072/997 PSIA
PRESS HEATERS	:	1 BANK
PRESS SPRAY	:	UNAVAILABLE
RC PUMPS	:	COAST DOWN BEGINNING @ T=O SEC
RPS TRIP	:	LOOP
RPS TRIP DEFEAT	:	NONE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	1
OFFSITE POWER	:	UNAVAILABLE
DECAY HEAT	:	1.0 X ANS 5.1, 1971
PORV	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF	:	-4.4027 E-3 $/F
PRESS MODEL	:	NON-EQUILIBRIUM

TABLE 8A-3

LOSS OF OFFSITE POWER CASE 1B

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 5 SEC
EFW MECH	:	AVAILABLE
EFW STEAM	:	AVAILABLE
EFW CAPACITY	:	1840 GPM
ATMOS DUMP	:	AVAILABLE @ 1025 PSIA
TURBINE BYPASS	:	AVAILABLE
SMALL SAFETIES	:	1060/986 PSIA
BANK 1	:	1065/990 PSIA
BANK 2	:	1070/995 PSIA
BANK 3	:	1072/997 PSIA
PRESS HEATERS	:	1 BANK
PRESS SPRAY	:	UNAVAILABLE
RC PUMPS	:	COAST DOWN BEGINNING @ T=0 SEC
RPS TRIP	:	LOOP
RPS TRIP DEFEAT	:	NONE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	1
OFFSITE POWER	:	UNAVAILABLE
DECAY HEAT	:	1.0 X ANS 5.1, 1971
PORV	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF	:	-4.4027 E-3 $/F
PRESS MODEL	:	NON-EQUILIBRIUM

TABLE 8A-4

LOSS OF OFFSITE POWER CASE 2

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 5 SEC
EFW MECH	:	AVAILABLE
EFW STEAM	:	AVAILABLE
EFW CAPACITY	:	1840 GPM
ATMOS DUMP	:	AVAILABLE @ 1025 PSIA
TURBINE BYPASS	:	AVAILABLE
SMALL SAFETIES	:	1060/1028 PSIA
BANK 1	:	1065/1033 PSIA
PRESS HEATERS	:	2 BANKS (126 KW)
PRESS SPRAY	:	UNAVAILABLE
RC PUMPS	:	COAST DOWN BEGINNING @ T=O SEC
RPS TRIP	:	LOOP
RPS TRIP DEFEAT	:	NONE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	UNAVAILABLE
DECAY HEAT	:	0.01 X ANS 5.1, 1971
[illegible]	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF	:	-4.4027 E-3 $/F
PRESS MODEL	:	NON-EQUILIBRIUM

TABLE 8A-5

LOSS OF OFFSITE POWER CASE 3

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 5 SEC
EFW MECH	:	AVAILABLE
EFW STEAM	:	AVAILABLE
EFW CAPACITY	:	1840 GPM
ATMOS DUMP	:	AVAILABLE @ 1025 PSIA
TURBINE BYPASS	:	AVAILABLE
SMALL SAFETIES	:	1060/590 PSIA
BANK 1	:	1065/590 PSIA
BANK 2	:	1070/995 PSIA
BANK 3	:	1072/997 PSIA
PRESS HEATERS	:	2 BANKS
PRESS SPRAY	:	UNAVAILABLE
RC PUMPS	:	COAST DOWN BEGINNING @ T=O SEC
RPS TRIP	:	LOOP
RPS TRIP DEFEAT	:	NONE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	UNAVAILABLE
DECAY HEAT	:	1.0 X ANS 5.1, 1971
PORV	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF	:	-4.4027 E-3 $/F

TABLE 8A-6

LOSS OF OFFSITE POWER CASE 3A

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 5 SEC
EFW MECH	:	AVAILABLE
EFW STEAM	:	AVAILABLE
EFW CAPACITY	:	0 GPM
ATMOS DUMP	:	AVAILABLE @ 1025 PSIA
TURBINE BYPASS	:	AVAILABLE
SMALL SAFETIES	:	1060/14.7 PSIA
BANK 1	:	1060/14.7 PSIA
BANK 2	:	1070/1038 PSIA
BANK 3	:	1072/1040 PSIA
PRESS HEATERS	:	2 BANKS
PRESS SPRAY	:	UNAVAILABLE
RC PUMPS	:	COAST DOWN BEGINNING @ T=O SEC
RPS TRIP	:	LOOP
RPS TRIP DEFEAT	:	NONE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	UNAVAILABLE
DECAY HEAT	:	0.85 X ANS 5.1, 1971
PORV	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 \$/F
MODERATOR COEF	:	-4.4027 E-3 \$/F

TABLE 8A-7

LOSS OF OFFSITE POWER CASE 4

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 0 SEC
EFW MECH	:	AVAILABLE
EFW STEAM	:	AVAILABLE
EFW CAPACITY	:	490 GPM
ATMOS DUMP	:	AVAILABLE @ 1025 PSIA
TURBINE BYPASS	:	AVAILABLE
SMALL SAFETIES	:	1060/1028 PSIA
BANK 1	:	1060/1033 PSIA
BANK 2	:	1070/1038 PSIA
BANK 3	:	1072/1040 PSIA
PRESS HEATERS	:	2 BANKS (126 KW)
PRESS SPRAY	:	UNAVAILABLE
RC PUMPS	:	COAST DOWN BEGINNING @ T=O SEC
RPS TRIP	:	LOOP
RPS TRIP DEFEAT	:	NONE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	UNAVAILABLE
DECAY HEAT	:	1.0 X ANS 5.1, 1971
PORV	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF	:	-4.4027 E-3 $/F

TABLE 8A-8

LOSS OF OFFSITE POWER CASE 5

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 0 SEC
EFW MECH	:	AVAILABLE
EFW STEAM	:	AVAILABLE
EFW CAPACITY	:	1140 GPM
ATMOS DUMP	:	AVAILABLE @ 1025 PSIA
TURBINE BYPASS	:	AVAILABLE
SMALL SAFETIES	:	1060/1028 PSIA
BANK 1	:	1060/1033 PSIA
BANK 2	:	1070/1038 PSIA
BANK 3	:	1072/1040 PSIA
PRESS HEATERS	:	3 BANKS
PRESS SPRAY	:	UNAVAILABLE
RC PUMPS	:	COAST DOWN BEGINNING @ T=0 SEC
RPS TRIP	:	LOOP
RPS TRIP DEFEAT	:	NONE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	UNAVAILABLE
DECAY HEAT	:	0.01 X ANS 5.1, 1971
PORV	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF	:	-4.4027 E-3 $/F

TABLE 8A-9

LOSS OF OFFSITE POWER CASE 5A

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 5 SEC
EFW MECH	:	AVAILABLE
EFW STEAM	:	AVAILABLE
EFW CAPACITY	:	1140 GPM
ATMOS DUMP	:	AVAILABLE @ 1025 PSIA
TURBINE BYPASS	:	AVAILABLE
SMALL SAFETIES	:	1060/986 PSIA
BANK 1	:	1060/1033 PSIA
BANK 2	:	1070/1038 PSIA
BANK 3	:	1072/1040 PSIA
PRESS HEATERS	:	1 BANK
PRESS SPRAY	:	UNAVAILABLE
RC PUMPS	:	COAST DOWN BEGINNING @ T=0 SEC
RPS TRIP	:	LOOP
RPS TRIP DEFEAT	:	NONE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	UNAVAILABLE
DECAY HEAT	:	0.01 X ANS 5.1, 1971
PORV	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF	:	-4.4027 E-3 $/F

TABLE 8A-10

LOSS OF OFFSITE POWER CASE 8

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 5 SEC
EFW MECH	:	AVAILABLE
EFW STEAM	:	AVAILABLE
EFW CAPACITY	:	1840 GPM
ATMOS DUMP	:	AVAILABLE @ 1025 PSIA
TURBINE BYPASS	:	AVAILABLE
SMALL SAFETIES	:	1060/1028 PSIA
BANK 1	:	1065/1033 PSIA
BANK 2	:	1070/1038 PSIA
BANK 3	:	1072/1040 PSIA
PRESS HEATERS	:	3 BANKS
PRESS SPRAY	:	UNAVAILABLE
RC PUMPS	:	COAST DOWN BEGINNING @ T=O SEC
RPS TRIP	:	LOOP
RPS TRIP DEFEAT	:	NONE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	UNAVAILABLE
DECAY HEAT	:	1.0 X ANS 5.1, 1971
PORV	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF	:	-4.4027 E-3 $/F

TABLE 8A-11

LOSS OF OFFSITE POWER CASE 9

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 5 SEC
EFW MECH	:	UNAVAILABLE
EFW STEAM	:	AVAILABLE
EFW CAPACITY	:	490 GPM
ATMOS DUMP	:	AVAILABLE @ 1025 PSIA
TURBINE BYPASS	:	AVAILABLE
SMALL SAFETIES	:	1060/1028 PSIA
BANK 1	:	1065/1033 PSIA
BANK 2	:	1070/1038 PSIA
BANK 3	:	1072/1040 PSIA
PRESS HEATERS	:	2 BANKS
PRESS SPRAY	:	UNAVAILABLE
RC PUMPS	:	COAST DOWN BEGINNING @ T=O SEC
RPS TRIP	:	LOOP
RPS TRIP DEFEAT	:	NONE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	UNAVAILABLE
DECAY HEAT	:	1.0 X ANS 5.1, 1971
PORV	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF	:	-4.4027 E-3 $/F

TABLE 8A-12

LOSS OF OFFSITE POWER CASE 10

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 0 SEC
EFW MECH	:	UNAVAILABLE
EFW STEAM	:	UNAVAILABLE
EFW CAPACITY	:	0 GPM
ATMOS DUMP	:	AVAILABLE @ 1025 PSIA
TURBINE BYPASS	:	AVAILABLE
SMALL SAFETIES	:	1060/1028 PSIA
BANK 1	:	1060/1033 PSIA
BANK 2	:	1070/1038 PSIA
BANK 3	:	1072/1040 PSIA
PRESS HEATERS	:	2 BANKS
PRESS SPRAY	:	UNAVAILABLE
RC PUMPS	:	COAST DOWN BEGINNING @ T=O SEC
RPS TRIP	:	LOOP
RPS TRIP DEFEAT	:	NONE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	UNAVAILABLE
DECAY HEAT	:	1.0 X ANS 5.1, 1971
PORV	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF	:	-4.4027 E-3 $/F

TABLE 8A-13

LOSS OF OFFSITE POWER CASE 11

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T - 5 SEC
EFW MECH	:	UNAVAILABLE
EFW STEAM	:	UNAVAILABLE
EFW CAPACITY	:	0 GPM
ATMOS DUMP	:	AVAILABLE @ 1025 PSIA
TURBINE BYPASS	:	UNAVAILABLE
SMALL SAFETIES	:	1060/1028 PSIA
BANK 1	:	1065/990 PSIA
BANK 2	:	1070/995 PSIA
BANK 3	:	1072/997 PSIA
PRESS HEATERS	:	1 BANK
PRESS SPRAY	:	UNAVAILABLE
RC PUMPS	:	COAST DOWN BEGINNING @ T=0 SEC
RPS TRIP	:	LOOP
RPS TRIP DEFEAT	:	NONE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	UNAVAILABLE
DECAY HEAT	:	1.0 X ANS 5.1, 1971
PORV	:	OPEN @ T=0 SEC, CLOSE @ 600 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 \$/F
MODERATOR COEF	:	-4.4027 E-3 \$/F

TABLE 8A-14

LOSS OF OFFSITE POWER CASE 11A

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 5 SEC
EFW MECH	:	UNAVAILABLE
EFW STEAM	:	UNAVAILABLE
EFW CAPACITY	:	0 GPM
ATMOS DUMP	:	AVAILABLE @ 1025 PSIA
TURBINE BYPASS	:	UNAVAILABLE
SMALL SAFETIES	:	1060/1028 PSIA
BANK 1	:	1065/990 PSIA
BANK 2	:	1070/995 PSIA
BANK 3	:	1072/997 PSIA
PRESS HEATERS	:	1 BANK
PRESS SPRAY	:	UNAVAILABLE
RC PUMPS	:	COAST DOWN BEGINNING @ T=0 SEC
RPS TRIP	:	LOOP
RPS TRIP DEFEAT	:	NONE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	1
OFFSITE POWER	:	UNAVAILABLE
DECAY HEAT	:	1.0 X ANS 5.1, 1971
PORV	:	OPEN @ T=0 SEC, CLOSE @ 600 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF		-4.4027 E-3 $/F

TABLE 8A-15

LOSS OF OFFSITE POWER CASE 15

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 5 SEC
EFW MECH	:	AVAILABLE
EFW STEAM	:	AVAILABLE
EFW CAPACITY	:	1840 GPM
ATMOS DUMP	:	AVAILABLE @ 1025 PSIA
TURBINE BYPASS	:	UNAVAILABLE
SMALL SAFETIES	:	1060/1028 PSIA
BANK 1	:	1065/990 PSIA
BANK 2	:	1070/995 PSIA
BANK 3	:	1072/997 PSIA
PRESS HEATERS	:	1 BANK
PRESS SPRAY	:	UNAVAILABLE
RC PUMPS	:	COAST DOWN BEGINNING @ T=O SEC
RPS TRIP	:	LOOP
RPS TRIP DEFEAT	:	NONE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	1
OFFSITE POWER	:	UNAVAILABLE
DECAY HEAT	:	1.0 X ANS 5.1, 1971
PORV	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF	:	-4.4027 E-3 $/F

TABLE 8A-16

FEEDWATER LINE BREAK CASE 1

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 0 SEC
EFW MECH	:	UNAVAILABLE LOOP A
		AVAILABLE LOOP B
EFW STEAM	:	UNAVAILABLE LOOP A
		AVAILABLE LOOP B
EFW CAPACITY	:	0 GPM LOOP A
		500 GPM LOOP B
ATMOS DUMP	:	AVAILABLE
TURBINE BYPASS	:	AVAILABLE @ 1025 PSIA
SMALL SAFETIES	:	1060/1028 PSIA
BANK 1	:	1065/1033 PSIA
BANK 2	:	1070/1038 PSIA
BANK 3	:	1072/1040 PSIA
PRESS HEATERS	:	5 BANKS
PRESS SPRAY	:	AVAILABLE (2220/2170 PSIA)
RC PUMPS	:	AVAILABLE
RPS TRIP	:	HIGH PRESSURE
RPS TRIP DEFEAT	:	VARIABLE LOW PRESSURE & TURBINE
SFAS TRIP	:	1600 PSIG RCS PRESSURE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	AVAILABLE
DECAY HEAT	:	1.2 X ANS 5.1, 1971
PORV	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF	:	-4.4027 E-3 $/F

TABLE 8A-17

FEEDWATER LINE BREAK CASE 1A

MAKEUP :	AVAILABLE
LETDOWN :	ISOLATED @ T = 5 SEC
EFW MECH :	UNAVAILABLE LOOP A
	AVAILABLE LOOP B
EFW STEAM :	UNAVAILABLE LOOP A
	AVAILABLE LOOP B
EFW CAPACITY :	0 GPM LOOP A
	500 GPM LOOP B
ATMOS DUMP :	AVAILABLE
TURBINE BYPASS :	AVAILABLE @ 1025 PSIA
SMALL SAFETIES :	1060/1028 PSIA
BANK 1 :	1065/1033 PSIA
BANK 2 :	1070/1038 PSIA
BANK 3 :	1072/1040 PSIA
PRESS HEATERS :	5 BANKS
PRESS SPRAY :	AVAILABLE (2220/2170 PSIA)
RC PUMPS :	AVAILABLE
RPS TRIP :	TURBINE
RPS TRIP DEFEAT :	NONE
SFAS TRIP :	1600 PSIG RCS PRESSURE
SFAS TRIP DEFEAT :	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS :	2
OFFSITE POWER :	AVAILABLE
DECAY HEAT :	1.2 X ANS 5.1, 1971
PORV :	2450/2400 PSIG
PRESS SAFETY :	2500/2475 PSIG
DOPPLER COEF :	-8.4047 E-4 $/F
MODERATOR COEF :	-4.4027 E-3 $/F

TABLE 8A-18

FEEDWATER LINE BREAK CASE 1B

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 5 SEC
EFW MECH	:	UNAVAILABLE LOOP A
		AVAILABLE LOOP B
EFW STEAM	:	UNAVAILABLE LOOP A
		AVAILABLE LOOP B
EFW CAPACITY	:	0 GPM LOOP A
		550 GPM LOOP B
ATMOS DUMP	:	AVAILABLE
TURBINE BYPASS	:	AVAILABLE @ 990 PSIA
SMALL SAFETIES	:	1060/1028 PSIA
BANK 1	:	1065/1033 PSIA
BANK 2	:	1070/1038 PSIA
BANK 3	:	1072/1040 PSIA
PRESS HEATERS	:	5 BANKS
PRESS SPRAY	:	AVAILABLE (2220/2170 PSIA)
RC PUMPS	:	AVAILABLE
RPS TRIP	:	TURBINE
RPS TRIP DEFEAT	:	NONE
SFAS TRIP	:	1600 PSIG RCS PRESSURE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	AVAILABLE
DECAY HEAT	:	1.0 X ANS 5.1, 1971
PORV	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF	:	-4.4027 E-3 $/F

TABLE 8A-19

FEEDWATER LINE BREAK CASE 1C

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 0 SEC
EFW MECH	:	UNAVAILABLE LOOP A
		AVAILABLE LOOP B
EFW STEAM	:	UNAVAILABLE LOOP A
		AVAILABLE LOOP B
EFW CAPACITY	:	0 GPM LOOP A
		550 GPM LOOP B
ATMOS DUMP	:	AVAILABLE
TURBINE BYPASS	:	AVAILABLE @ 1025 PSIA
SMALL SAFETIES	:	1060/1028 PSIA
BANK 1	:	1065/1033 PSIA
BANK 2	:	1070/1038 PSIA
BANK 3	:	1072/1040 PSIA
PRESS HEATERS	:	5 BANKS
PRESS SPRAY	:	AVAILABLE (2220/2170 PSIA)
RC PUMPS	:	AVAILABLE
RPS TRIP	:	HIGH PRESSURE
RPS TRIP DEFEAT	:	VARIABLE LOW PRESSURE & TURBINE
SFAS TRIP	:	1600 PSIG RCS PRESSURE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	AVAILABLE
DECAY HEAT	:	1.2 X ANS 5.1, 1971
PORV	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF	:	-4.4027 E-3 $ F

TABLE 8A-20

FEEDWATER LINE BREAK CASE 1D

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 5 SEC
EFW MECH	:	AVAILABLE
EFW STEAM	:	AVAILABLE
EFW CAPACITY	:	720 GPM
ATMOS DUMP	:	AVAILABLE
TURBINE BYPASS	:	AVAILABLE @ 1000 PSIA
SMALL SAFETIES	:	1060/1028 PSIA
BANK 1	:	1065/1033 PSIA
BANK 2	:	1070/1038 PSIA
BANK 3	:	1072/1040 PSIA
PRESS HEATERS	:	5 BANKS
PRESS SPRAY	:	AVAILABLE (2220/2170 PSIA)
RC PUMPS	:	AVAILABLE
RPS TRIP	:	TURBINE
RPS TRIP DEFEAT	:	NONE
SFAS TRIP	:	1600 PSIG RCS PRESSURE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	AVAILABLE
DECAY HEAT	:	1.0 X ANS 5.1, 1971
PORV	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF	:	-4.4027 E-3 $/F

TABLE 8A-21

FEEDWATER LINE BREAK CASE 1F

MAKEUP :	AVAILABLE
LETDOWN :	ISOLATED @ T = 5 SEC
EFW MECH :	UNAVAILABLE LOOP A
	AVAILABLE LOOP B
EFW STEAM :	UNAVAILABLE LOOP A
	AVAILABLE LOOP B
EFW CAPACITY :	0 GPM LOOP A
	550 GPM LOOP B
ATMOS DUMP :	AVAILABLE
TURBINE BYPASS :	AVAILABLE @ 1025 PSIA
SMALL SAFETIES :	1060/1028 PSIA
BANK 1 :	1065/1033 PSIA
BANK 2 :	1070/1038 PSIA
BANK 3 :	1072/1040 PSIA
PRESS HEATERS :	5 BANKS
PRESS SPRAY :	AVAILABLE (2220/2170 PSIA)
RC PUMPS :	AVAILABLE
RPS TRIP :	TURBINE
RPS TRIP DEFEAT :	NONE
SFAS TRIP :	1600 PSIG RCS PRESSURE
SFAS TRIP DEFEAT :	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS :	2
OFFSITE POWER :	AVAILABLE
DECAY HEAT :	1.0 X ANS 5.1, 1971
PORV :	2450/2400 PSIG
PRESS SAFETY :	2500/2475 PSIG
DOPPLER COEF :	-8.4047 E-4 $/F
MODERATOR COEF :	-4.4027 E-3 $/F

Am. 23

TABLE 8A-22

FEEDWATER LINE BREAK CASE 2A

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 0 SEC
EFW MECH	:	UNAVAILABLE
EFW STEAM	:	UNAVAILABLE LOOP A
	:	UNAVAILABLE LOOP B
EFW CAPACITY	:	250 GPM LOOP A
		0 GPM LOOP B
ATMOS DUMP	:	AVAILABLE
TURBINE BYPASS	:	AVAILABLE @ 1025 PSIA
SMALL SAFETIES	:	1060/1028 PSIA
BANK 1	:	1065/1033 PSIA
BANK 2	:	1070/1038 PSIA
BANK 3	:	1072/1040 PSIA
PRESS HEATERS	:	5 BANKS
PRESS SPRAY	:	AVAILABLE, OFF AT REACTOR TRIP
RC PUMPS	:	AVAILABLE
RPS TRIP	:	TURBINE
RPS TRIP DEFEAT	:	NONE
SFAS TRIP	:	1600 PSIG RCS PRESSURE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	AVAILABLE
DECAY HEAT	:	1.2 X ANS 5.1, 1971
PORV	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF	:	-4.4027 E-3 $/F

TABLE 8A-23

LOSS OF NORMAL FEEDWATER CASE 1

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 5 SEC
EFW MECH	:	1 MDP AVAILABLE
EFW STEAM	:	AVAILABLE
EFW CAPACITY	:	460 GPM
ATMOS DUMP	:	AVAILABLE
TURBINE BYPASS	:	AVAILABLE @ 1025 PSIA
SMALL SAFETIES	:	1060/1028 PSIA
BANK 1	:	1065/1033 PSIA
BANK 2	:	1070/1038 PSIA
BANK 3	:	1072/1040 PSIA
PRESS HEATERS	:	5 BANKS
PRESS SPRAY	:	AVAILABLE (2220/2170 PSIA)
RC PUMPS	:	AVAILABLE
RPS TRIP	:	TURBINE
RPS TRIP DEFEAT	:	NONE
SFAS TRIP	:	1500 PSIG RCS PRESSURE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	AVAILABLE
DECAY HEAT	:	1.2 X ANS 5.1, 1971
PORV	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF	:	-4.4027 E-3 $/F

TABLE 8A-24

LOSS OF NORMAL FEEDWATER CASE 1A

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 5 SEC
EFW MECH	:	1 MDP AVAILABLE
EFW STEAM	:	UNAVAILABLE
EFW CAPACITY	:	460 GPM
ATMOS DUMP	:	AVAILABLE
TURBINE BYPASS	:	AVAILABLE @ 1025 PSIA
SMALL SAFETIES	:	1060/1028 PSIA
BANK 1	:	1065/1033 PSIA
BANK 2	:	1070/1038 PSIA
BANK 3	:	1072/1040 PSIA
PRESS HEATERS	:	5 BANKS
PRESS SPRAY	:	AVAILABLE (2220/2170 PSIA)
RC PUMPS	:	AVAILABLE
RPS TRIP	:	TURBINE
RPS TRIP DEFEAT	:	NONE
SFAS TRIP	:	1600 PSIG RCS PRESSURE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	AVAILABLE
DECAY HEAT	:	0.85 X ANS 5.1, 1971
PORV	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 \$/F
MODERATOR COEF	:	-4.4027 E-3 \$/F

TABLE 8A-25

LOSS OF NORMAL FEEDWATER CASE 1C

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 5 SEC
EFW MECH	:	AVAILABLE
EFW STEAM	:	AVAILABLE
EFW CAPACITY	:	460 GPM
ATMOS DUMP	:	AVAILABLE
TURBINE BYPASS	:	AVAILABLE @ 1025 PSIA
SMALL SAFETIES	:	1060/986 PSIA
BANK 1	:	1065/990 PSIA
BANK 2	:	1070/995 PSIA
BANK 3	:	1072/997 PSIA
PRESS HEATERS	:	5 BANKS
PRESS SPRAY	:	AVAILABLE @ T = 600 SEC
RC PUMPS	:	AVAILABLE
RPS TRIP	:	HIGH PRESSURE (2405 PSIA)
RPS TRIP DEFEAT	:	TURBINE TRIP
SFAS TRIP	:	1600 PSIG RCS PRESSURE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	AVAILABLE
DECAY HEAT	:	1.0 X ANS 5.1, 1971
PORV	:	2900/400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF	:	-4.4027 E-3 $/F

TABLE 8A-26

LOSS OF NORMAL FEEDWATER CASE 2

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 0 SEC
EFW MECH	:	AVAILABLE
EFW STEAM	:	UNAVAILABLE
EFW CAPACITY	:	460 GPM
ATMOS DUMP	:	AVAILABLE
TURBINE BYPASS	:	AVAILABLE @ 1025 PSIA
SMALL SAFETIES	:	1060/1028 PSIA
BANK 1	:	1065/1033 PSIA
BANK 2	:	1070/1038 PSIA
BANK 3	:	1072/1040 PSIA
PRESS HEATERS	:	5 BANKS
PRESS SPRAY	:	AVAILABLE (2220/2170 PSIA)
RC PUMPS	:	AVAILABLE
RPS TRIP	:	TRUBINE
RPS TRIP DEFEAT	:	NONE
SFAS TRIP	:	1500 PSIG RCS PRESSURE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	AVAILABLE
DECAY HEAT	:	1.0 X ANS 5.1, 1971
PORV	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF	:	-4.4027 E-3 $/F

TABLE 8A-27

LOSS OF NORMAL FEEDWATER CASE 3

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 0 SEC
EFW MECH	:	AVAILABLE
EFW STEAM	:	AVAILABLE
EFW CAPACITY	:	460 GPM
ATMOS DUMP	:	AVAILABLE
TURBINE BYPASS	:	AVAILABLE @ 1025 PSIA
SMALL SAFETIES	:	1060/1028 PSIA
BANK 1	:	1065/1033 PSIA
BANK 2	:	1070/1038 PSIA
BANK 3	:	1072/1040 PSIA
PRESS HEATERS	:	5 BANKS
PRESS SPRAY	:	AVAILABLE (2220/2170 PSIA)
RC PUMPS	:	AVAILABLE
RPS TRIP	:	TRUBINE
RPS TRIP DEFEAT	:	NONE
SFAS TRIP	:	1500 PSIG RCS PRESSURE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	AVAILABLE
DECAY HEAT	:	1.0 X ANS 5.1, 1971
PORV	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 \$/F
MODERATOR COEF	:	-4.4027 E-3 \$/F

TABLE 8A-28

LOSS OF NORMAL FEEDWATER CASE 4

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 5 SEC
EFW MECH	:	AVAILABLE
EFW STEAM	:	AVAILABLE
EFW CAPACITY	:	1145 GPM
ATMOS DUMP	:	AVAILABLE
TURBINE BYPASS	:	AVAILABLE @ 1025 PSIA
SMALL SAFETIES	:	1060/986 PSIA
BANK 1	:	1065/990 PSIA
BANK 2	:	1070/995 PSIA
BANK 3	:	1072/997 PSIA
PRESS HEATERS	:	5 BANKS
PRESS SPRAY	:	AVAILABLE (2220/2170 PSIA)
RC PUMPS	:	AVAILABLE
RPS TRIP	:	TRUBINE
RPS TRIP DEFEAT	:	NONE
SFAS TRIP	:	1600 PSIG RCS PRESSURE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	AVAILABLE
DECAY HEAT	:	0.01 X ANS 5.1, 1971
PORV	:	2450/2400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF	:	-4.4027 E-3 $/F

TABLE 8A-29

LOSS OF NORMAL FEEDWATER CASE 8

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 50 SEC
EFW MECH	:	UNAVAILABLE
EFW STEAM	:	UNAVAILABLE
EFW CAPACITY	:	0 GPM
ATMOS DUMP	:	1065/1035 PSIA
TURBINE BYPASS	:	1065/1035 PSIA
SMALL SAFETIES	:	1065/1012 PSIA
BANK 1	:	1080/1026 PSIA
BANK 2	:	1090/1036 PSIA
BANK 3	:	1117/1061 PSIA
PRESS HEATERS	:	5 BANKS
PRESS SPRAY	:	AVAILABLE
RC PUMPS	:	AVAILABLE
RPS TRIP	:	TRUBINE
RPS TRIP DEFEAT	:	NONE
SFAS TRIP	:	1600 PSIG RCS PRESSURE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	AVAILABLE
DECAY HEAT	:	0.65 X ANS 5.1, 1971
PORV	:	2470/2245 PSIG
PRESS SAFETY	:	2450/2325 PSIG
DOPPLER COEF	:	-7.100 E-4 $/F
MODERATOR COEF	:	-3.040 E-3 $/F

TABLE 8A-30

LOSS OF NORMAL FEEDWATER CASE 11

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 5 SEC
EFW MECH	:	AVAILABLE
EFW STEAM	:	AVAILABLE
EFW CAPACITY	:	1840 GPM
ATMOS DUMP	:	AVAILABLE
TURBINE BYPASS	:	AVAILABLE @ 1025 PSIA
SMALL SAFETIES	:	1060/986 PSIA
BANK 1	:	1065/990 PSIA
BANK 2	:	1070/995 PSIA
BANK 3	:	1072/997 PSIA
PRESS HEATERS	:	5 BANKS
PRESS SPRAY	:	AVAILABLE (2220/2170 PSIA)
RC PUMPS	:	AVAILABLE
RPS TRIP	:	HIGH PRESSURE
RPS TRIP DEFEAT	:	TURBINE
SFAS TRIP	:	1600 PSIG RCS PRESSURE
SFAS TRIP DEFEAT	:	4 PSIG CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	AVAILABLE
DECAY HEAT	:	1.0 X ANS 5.1, 1971
PORV	:	2450/2500 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.404 E-4 $/F
MODERATOR COEF	:	-3.4027 E-3 $/F

TABLE 8A-31

LOSS OF NORMAL FEEDWATER CASE 14

MAKEUP	:	AVAILABLE
LETDOWN	:	ISOLATED @ T = 5 SEC
EFW MECH	:	AVAILABLE
EFW STEAM	:	AVAILABLE
EFW CAPACITY	:	500 GPM
ATMOS DUMP	:	AVAILABLE
TURBINE BYPASS	:	AVAILABLE
SMALL SAFETIES	:	1060/986 PSIA
BANK 1	:	1065/990 PSIA
BANK 2	:	1070/995 PSIA
BANK 3	:	1072/997 PSIA
PRESS HEATERS	:	5 BANKS
PRESS SPRAY	:	AVAILABLE, OFF ON REACTOR TRIP
RC PUMPS	:	AVAILABLE
RPS TRIP	:	HIGH PRESSURE
RPS TRIP DEFEAT	:	TURBINE
SFAS TRIP	:	1600 PSIG RCS PRESSURE
SFAS TRIP DEFEAT	:	4 [illegible] CONTAINMENT PRESSURE
DIESEL GENERATORS	:	2
OFFSITE POWER	:	AVAILABLE
DECAY HEAT	:	1.2 X ANS 5.1, 1971
PORV	:	2900/400 PSIG
PRESS SAFETY	:	2500/2475 PSIG
DOPPLER COEF	:	-8.4047 E-4 $/F
MODERATOR COEF	:	-3.4027 E-3 $/F

QUESTION:

55. Bulletin 05B Item 1

Your procedure 1102-16, Natural Circulation, includes anticipatory filling of the OTSG prior to securing the reactor coolant pumps. Submit the analysis performed to provide guidance as to the expected system response.

RESPONSE:

The attached analysis provides guidance regarding system response to anticipatory steam generator fill transients. The analysis provides:

1. Calculated OTSG fill rate as a function of main feedwater or EFW flow. Feedwater temperature is accounted for as a function of equivalent primary system heat load.

2. The response of various B&W 177 facilities to OTSG fill transients.

The analysis shows a strong dependence of equivalent primary system heat load on feedwater temperature (as would be expected). The attached analyses provide guidance as to the expanded systems response to anticipatory filling of the OTSG. The following TMI-1 operator guidelines were developed using these analyses:

OPERATOR GUIDELINES FOR FILLING THE OTSG'S TO 50%
ON THE OPERATE RANGE LEVEL WITH THE MAIN FEEDWATER SYSTEM

1. When main feedwater is used to fill the steam generator to 50% with forced RCS flow, feed only through the startup valve. Control the rate of flow to maintain OTSG pressure within 100 psi of the control setpoint.

2. If feedwater is below 100°F, fill one OTSG at a time.

3. If forced RCS flow is lost before reaching 50% in both steam generators, feed with emergency feedwater using the EFW guidelines.

OPERATOR GUIDELINES FOR FILLING THE OTSG'S TO 50%
ON THE OPERATE RANGE LEVEL WITH THE EMERGENCY FEEDWATER SYSTEM

1. When emergency feedwater is used to fill the steam generators to 50% with forced RCS flow, then the rate of fill must be controlled in order to maintain the OTSG pressure within 100 psi of the control setpoint. Throttle EFW flow as required to control OTSG pressure.

2. Use either two (2) motor driven pumps or one (1) turbine driven pump.

PAGE NO. 2 TO QUESTION NO. 55

3. If forced RCS flow is lost before reaching 50% level in the OTSG's, observe the following restrictions:

    A. Steam generator level should always be increasing.

    B. EFW flow should be continuous until the setpoint is reached.

    C. EFW flow should be turned on full if natural circulation stops.

ATTACHMENT TO SUPPLEMENT NO. 1, PART NO. 1,
QUESTION NO. 55

ANALYSIS OF STEAM GENERATOR ANTICIPATORY FILL TRANSIENTS

The analysis of the anticipatory fill transient was divided into two sections. The first evaluated the influence of fast fill rates on the RC system using main feedwater, and the second section examined the effect of cooling the primary loop when filling steam generators with the emergency feedwater system.

In the first section, changes in primary loop conditions were obtained for main feedwater with various temperatures and flow rates. Predictions were substantiated by several instances of recorded 177 fuel assembly plant responses to steam generator fills using the main feedwater system.

Figure 1 shows the relationship of MFW flow rate per steam generator for observed fill rates using the start-up level indication. If the main feedwater system start-up valves are used, assuming a maximum flow rate of 1.5 million lbs/hour, then fill rates greater than 60 inches per minute could occur in each steam generator. The effect of MFW temperature was included and covered a range of temperature between 200 and 400°F.

Direct heating of the main feedwater by aspirated steam was assumed and in some instances the flow rate of steam condensed was approximately 35% of the MFW flow rate. If the heat transferred from the primary loop to the secondary loop is sufficient, then an identical amount of steam will be vaporized from the volume of water to restore the condensed steam and maintain a nearly constant steam pressure. The heat required to exactly replace the condensed steam has been calculated and is shown in Figure 1 as a percentage of the full rated power. As a comparison, a typical upper limit to decay heat plus RC pump power is shown as a heavy line at approximately 5%. For MFW temperatures of 300 to 400°F, a fill rate of 60 inches per minute (as indicated by the start-up level) would lead to a negligible change in steam pressure if sufficient decay heat is available. As MFW temperature decreases toward 200°F, more heat is required from the steam and the primary loop and smaller fill rates are necessary in order to maintain proper steam pressure. For very low MFW temperatures, it would be preferable to fill only one steam generator at a time.

Figures 2 and 3 display nearly identical plant response at two similar B&W plants for large rates of steam generator filling with the main feedwater system. In each case, main feedwater flow was not terminated following the reactor trip and both steam generators were filled above the 20 foot level before MFW flow was manually terminated. The flow rate of MFW is quite large, being either 1.5 or 3.0 million lbs/hour, and each steam generator was filled at rates of approximately 120 inches per minute.

The important fact to notice is that steam pressure dropped from roughly 1000 psig to 950 psig or higher during the fill operation and the effect on primary loop cooldown was very small. If the MFW temperature had been lower than 435 $\pm$ 20°F, a larger decrease in steam pressure would have been anticipated. (Actual plant data for low MFW temperature is not available.)

Figure 4 exhibits the transient behavior of the reactor coolant system, namely RC pressure and pressurizer level, for the two events that occurred at the Oconee Nuclear Station. At approximately one minute of filling the two steam generators with main feedwater, minimum values of RC pressure and pressurizer level were reached. This corresponds to the time that the desired level was achieved and main feedwater flow rates were reduced to zero.

The calculated values shown in Figure 1 appear to be conservative in comparison with real plant response. Since the calculations did not account for the heat stored in the metal of the steam generators, larger fill rates than predicted should be possible before a decrease in steam pressure would occur which would lead to cooling in the RC system.

In the second section of the analysis, a similar evaluation of steam generator performance during an emergency feedwater fill transient was performed. Both calculations and plant data reveal that steam pressure responds immediately to any imbalance between the direct heating of the near ambient temperature EFW by the steam and the heat supplied from the primary loop.

Figure 5 presents a comparison of actual changes in steam generator pressure due to selected fill rates using either main or emergency feedwater systems. Though all the plant data for EFW was recorded with all RC pumps shutdown, the effect of an additional 1/2 to 1% heat load due to running RC pumps would change the slope of the curve. However, the situation of RC pumps running and the operator selecting EFW to fill the steam generator has occurred very infrequently compared to either no RC pumps (and MFW pumps) and EFW system operation or RC pumps running and the MFW system used to fill both steam generators.

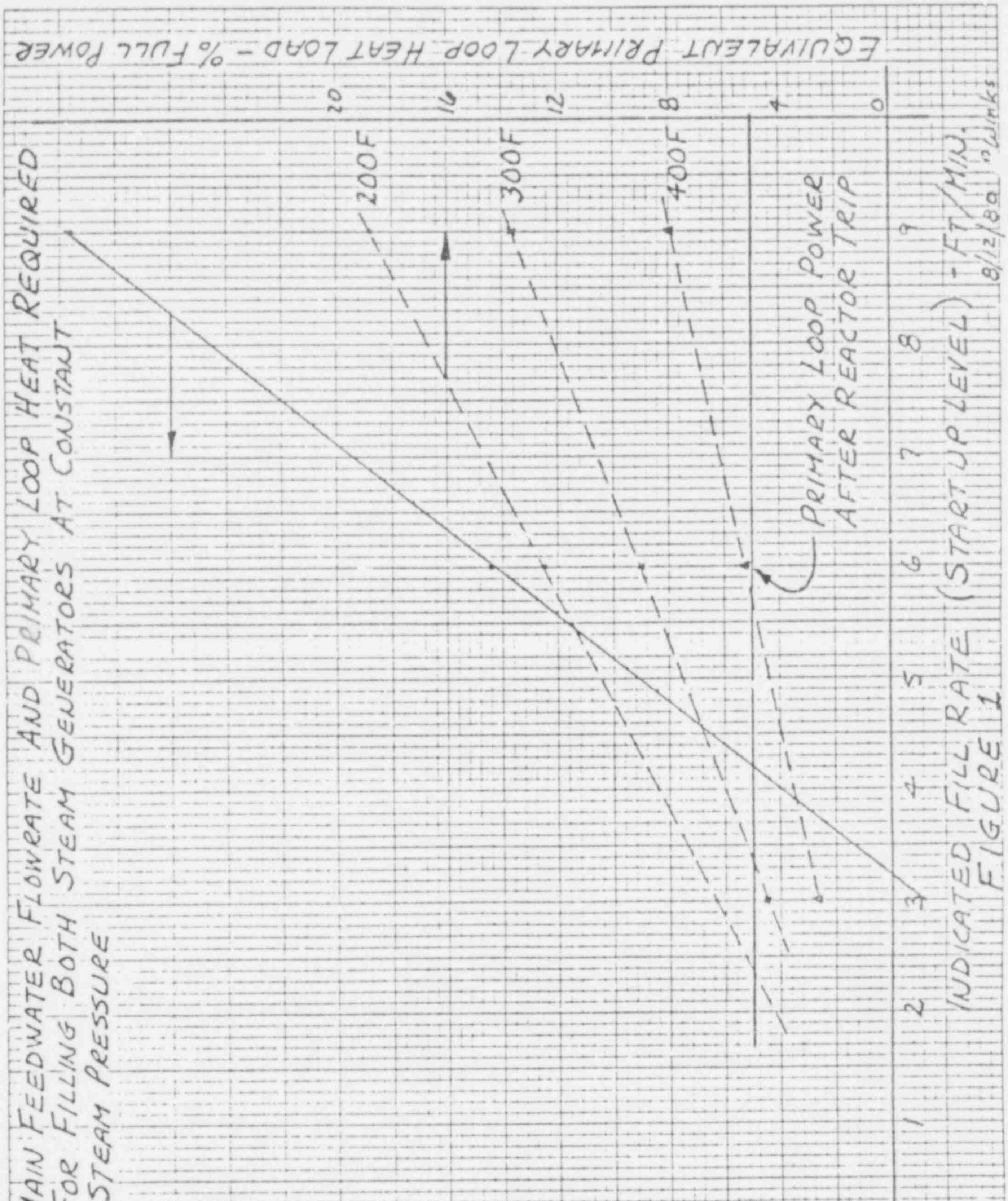
Figure 6 shows the severe impact that excessive EFW flow into two steam generators simultaneously has on the reactor coolant system. This was an unexpected loss of station power while the plant was operating at 40% power level. The EFW system was initiated and filled each steam generator at approximately 1200 gpm to a 100 to 120 inch level. In four minutes, the pressurizer level had decreased below a zero indication and the operators terminated the wide open filling of the second steam generator prior to reaching a 120 inch level.

Figure 7 shows a similar loss of power event that was conducted at TMI-2 on April 22, 1978 as a test. The EFW system was used to fill both steam generators approximately equally with a flow rate of about 500 gpm to each generator. Though the fill operation was not complete at eight minutes into the test, minimum values of RC pressure and pressurizer level were reached at that time. The severity of the impact on the RC system was quite small but may have been aided by reduced letdown flow and increased makeup system flow, actions usually taken by the operator after a reactor trip.

Figure 8 shows the EFW flow rate to one steam generator corresponding to observed fill rates on the start-up level. For the assumption of steam pressure at 900 psig and steam directly heating the EFW to a saturation temperature liquid, the relationship of fill rate and EFW flow rate is presented and compared to the 500 gpm flow rate limited by the cavitating venturis. As calculated before for MFW, the primary loop heat required to replace all condensed steam and maintain constant steam pressure is also

shown in Figure 8. The dotted lines show that a total heat load of 3% full rated power is required to maintain steady conditions within both steam generators when the total EFW flow is 1000 gpm. The effect of variable EFW temperature is small since the expected range of EFW temperature is only about 50°F. If the operating history of the reactor is low at the time of the reactor trip, then the decay heat will be low and it will be insufficient to match the "wide open" flow rate of EFW of 500 gpm. Thus, each decay heat level establishes a band of allowable EFW flow rates that can be used for filling the steam generators to the desired level. Since these are not known at the time of reactor trip, the operator should manually control EFW flow to maintain steam pressure greater than 100 psi below the control setpoint for steam generator pressure. Then satisfactory conditions will be maintained in the RC system.

If there is a choice between using MFW or EFW to achieve 20 feet of water in each steam generator before tripping the last RC pumps operating, the operator should use main feedwater. There are fewer restrictions and less manual control required than for the emergency feedwater system.



MAIN FEEDWATER FLOWRATE AND PRIMARY LOOP HEAT REQUIRED FOR FILLING BOTH STEAM GENERATORS AT CONSTANT STEAM PRESSURE

FIGURE 1

POOR ORIGINAL

86-1120427-01

STEAM GENERATOR FILL AFTER MARCH 14, 1980
REACTOR TRIP AT OCONEE UNIT #3

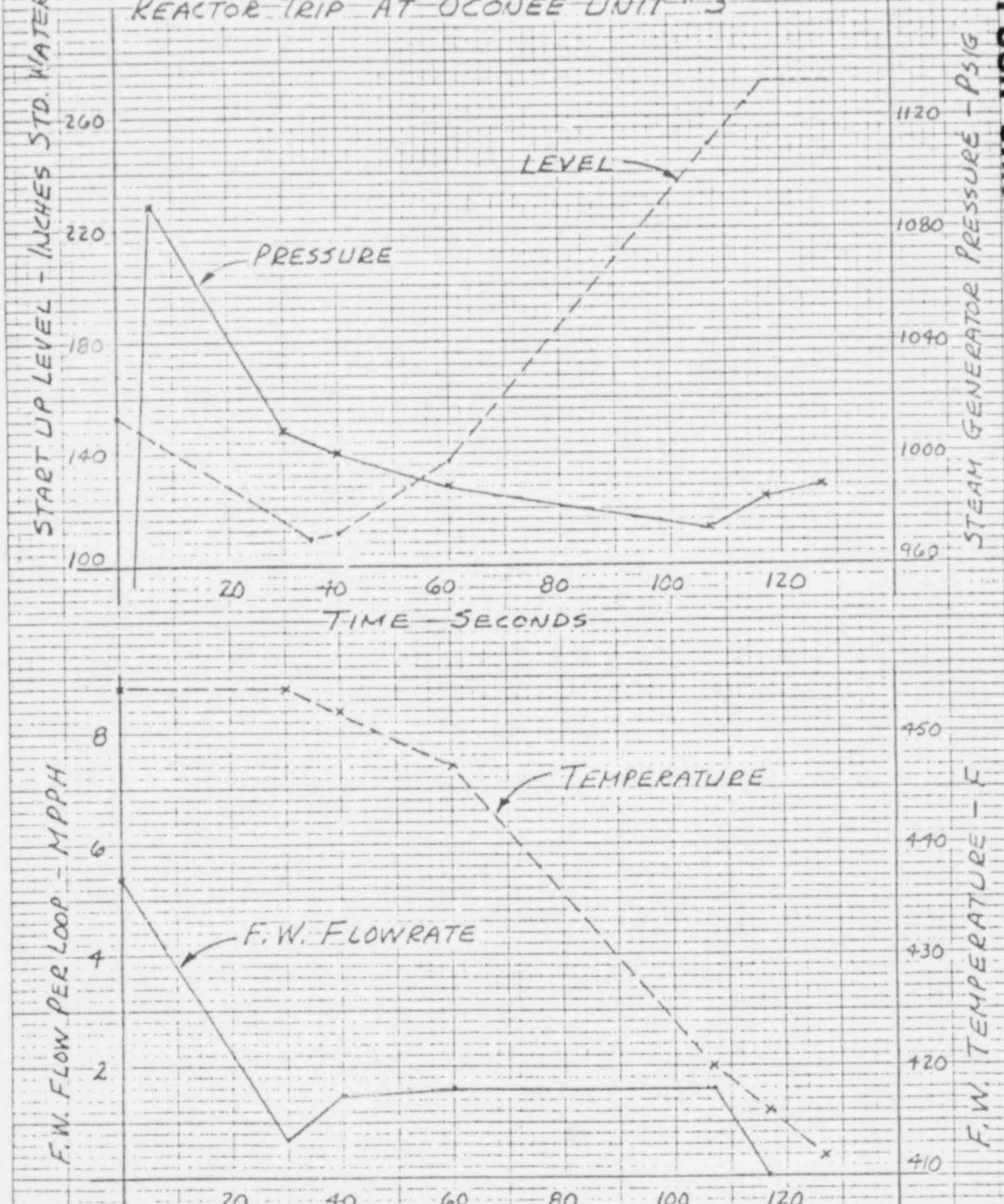


FIGURE 2

8/12/80 R. WINKS

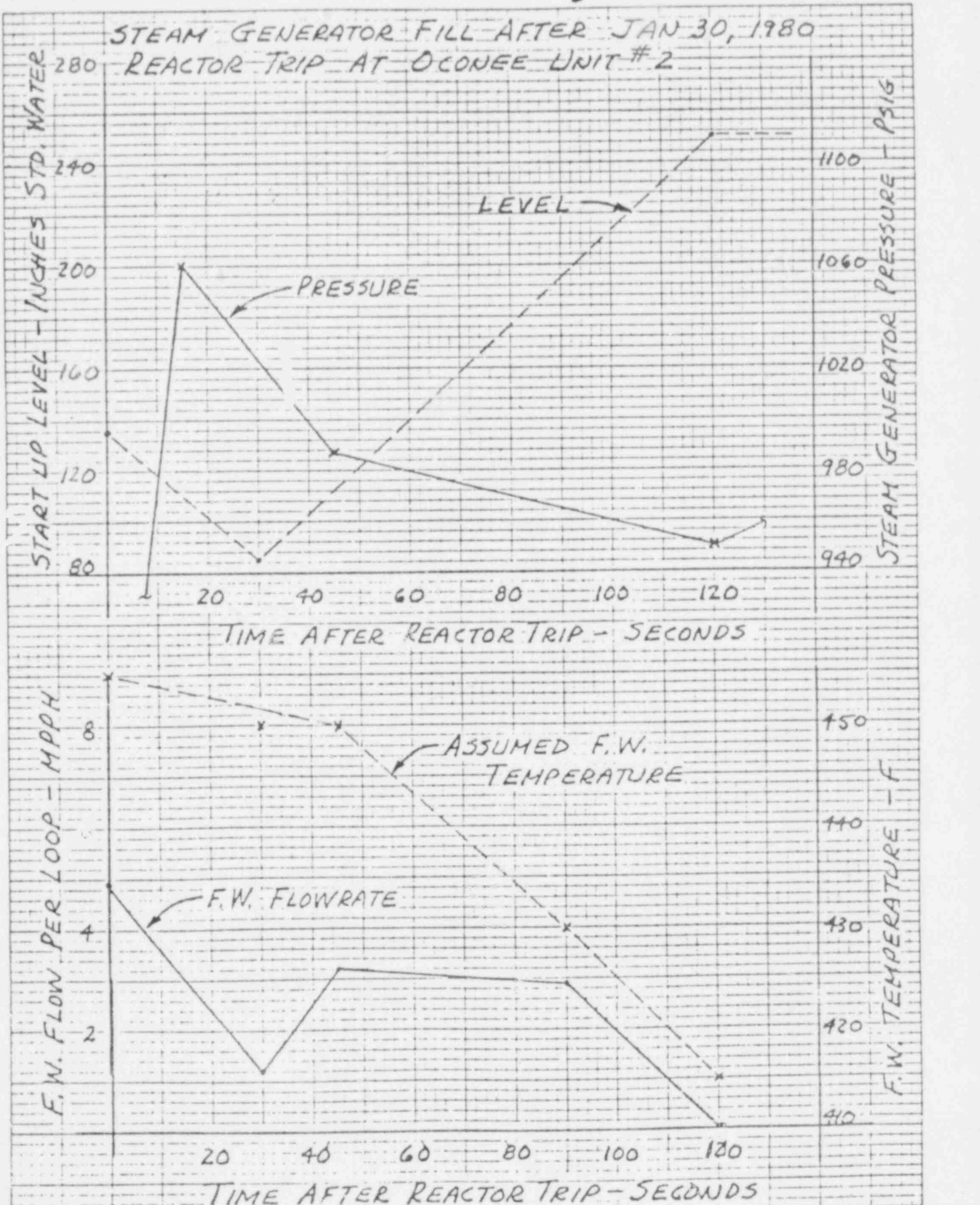


FIGURE 3

8/12/80 R WINKS

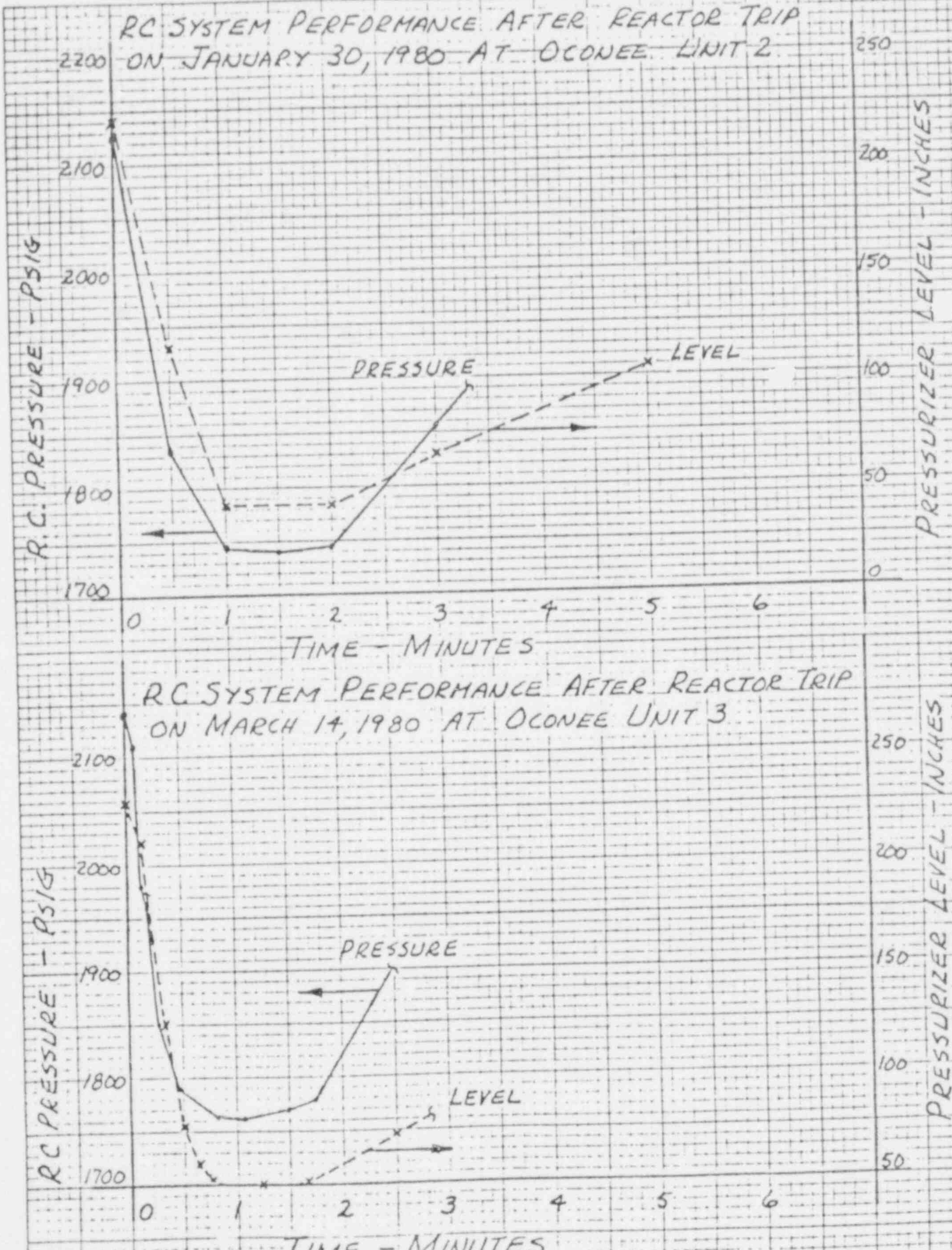


FIGURE 4

8/12/80 E. WINKS

Am. 23

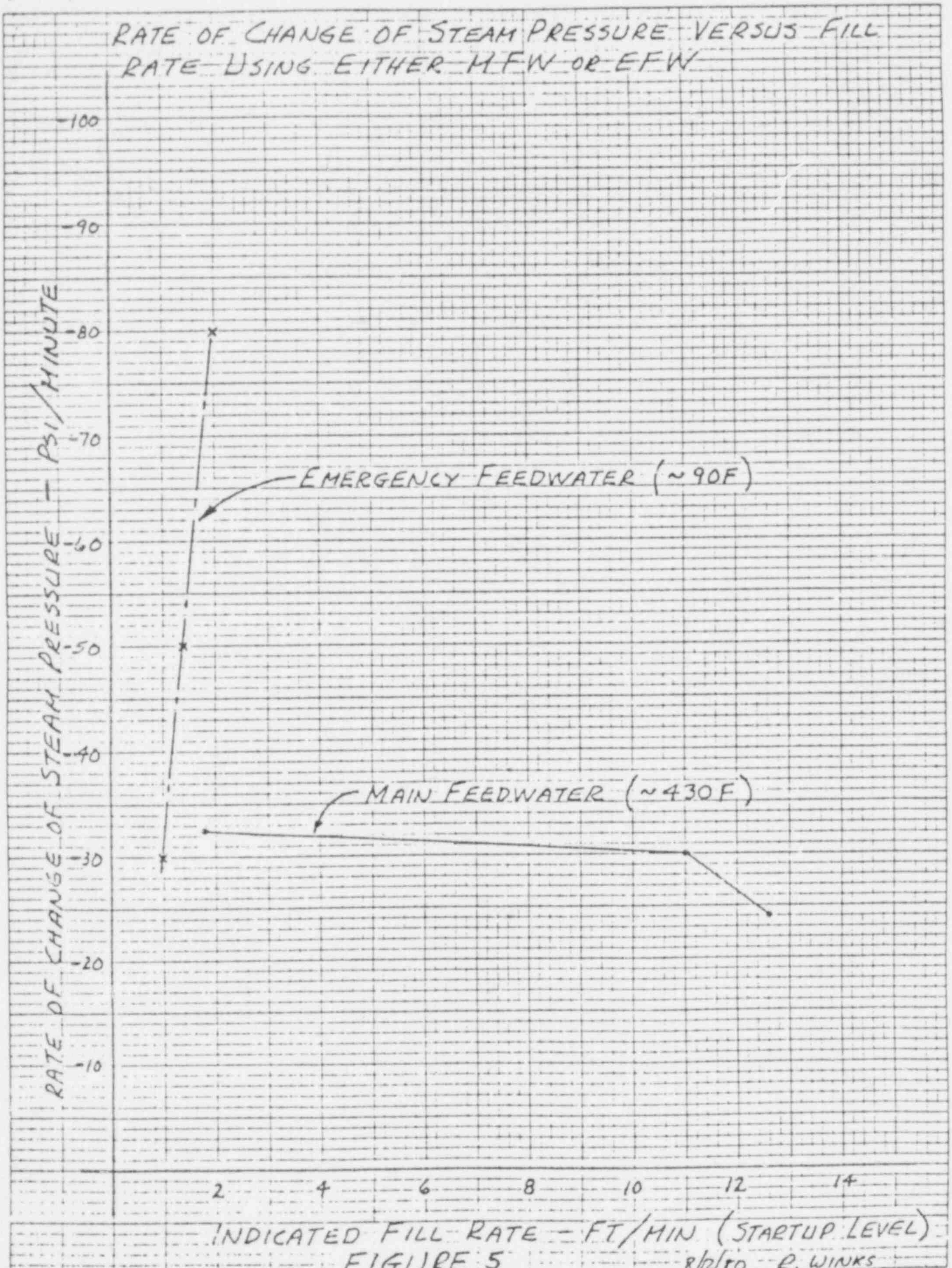

RATE OF CHANGE OF STEAM PRESSURE VERSUS FILL RATE USING EITHER MFW OR EFW
RATE OF CHANGE OF STEAM PRESSURE – PSI/MINUTE
-100
-90
-80
-70
-60
-50
-40
-30
-20
-10
EMERGENCY FEEDWATER (~90F)
MAIN FEEDWATER (~430F)
2
4
6
8
10
12
14
INDICATED FILL RATE – FT/MIN (STARTUP LEVEL)
8/12/80 R. WINKS

FIGURE 5

86-1120427-01

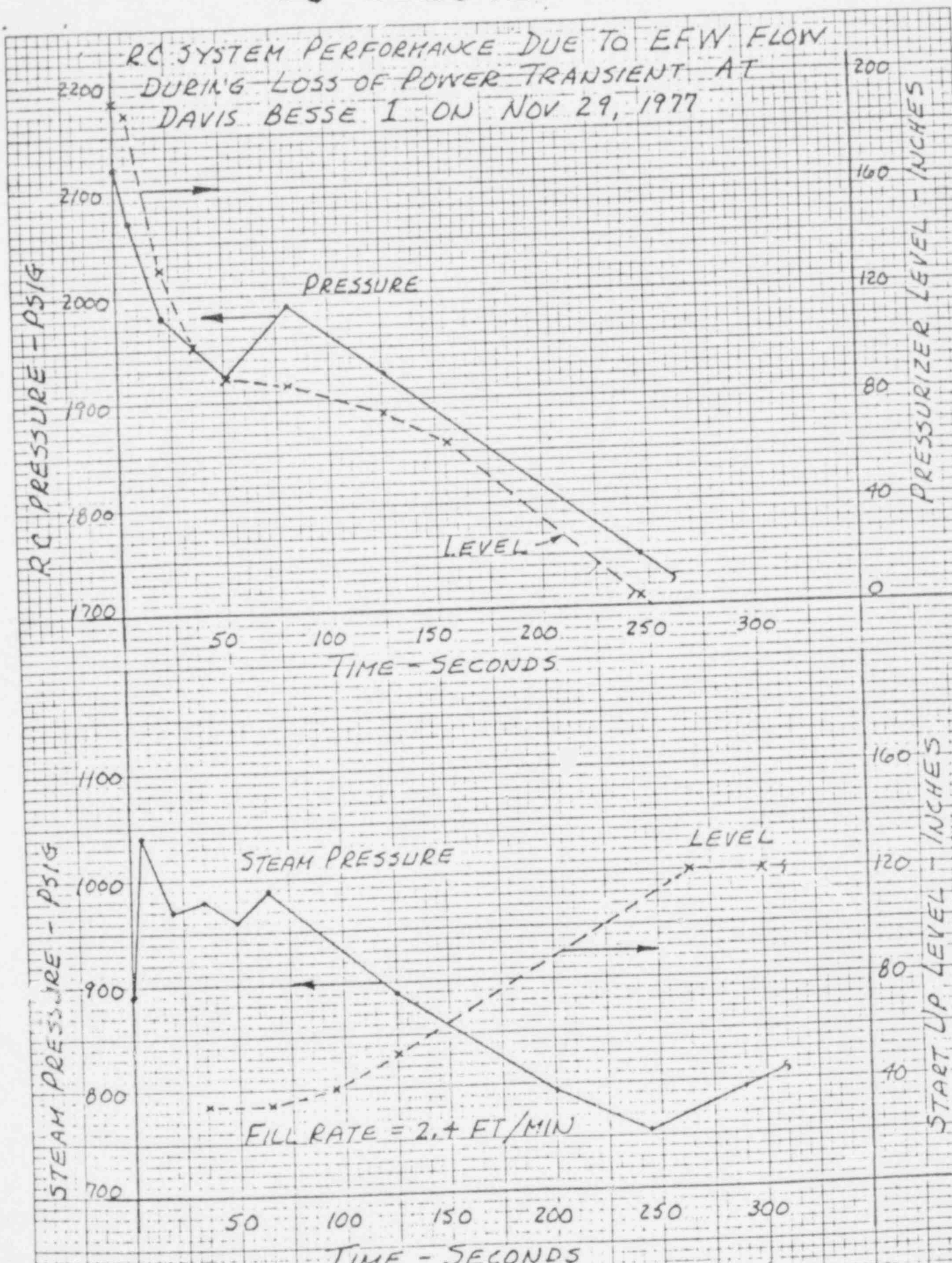


FIGURE 6

8/12/80 R. Winks

86-1120427-01

POOR ORIGINAL

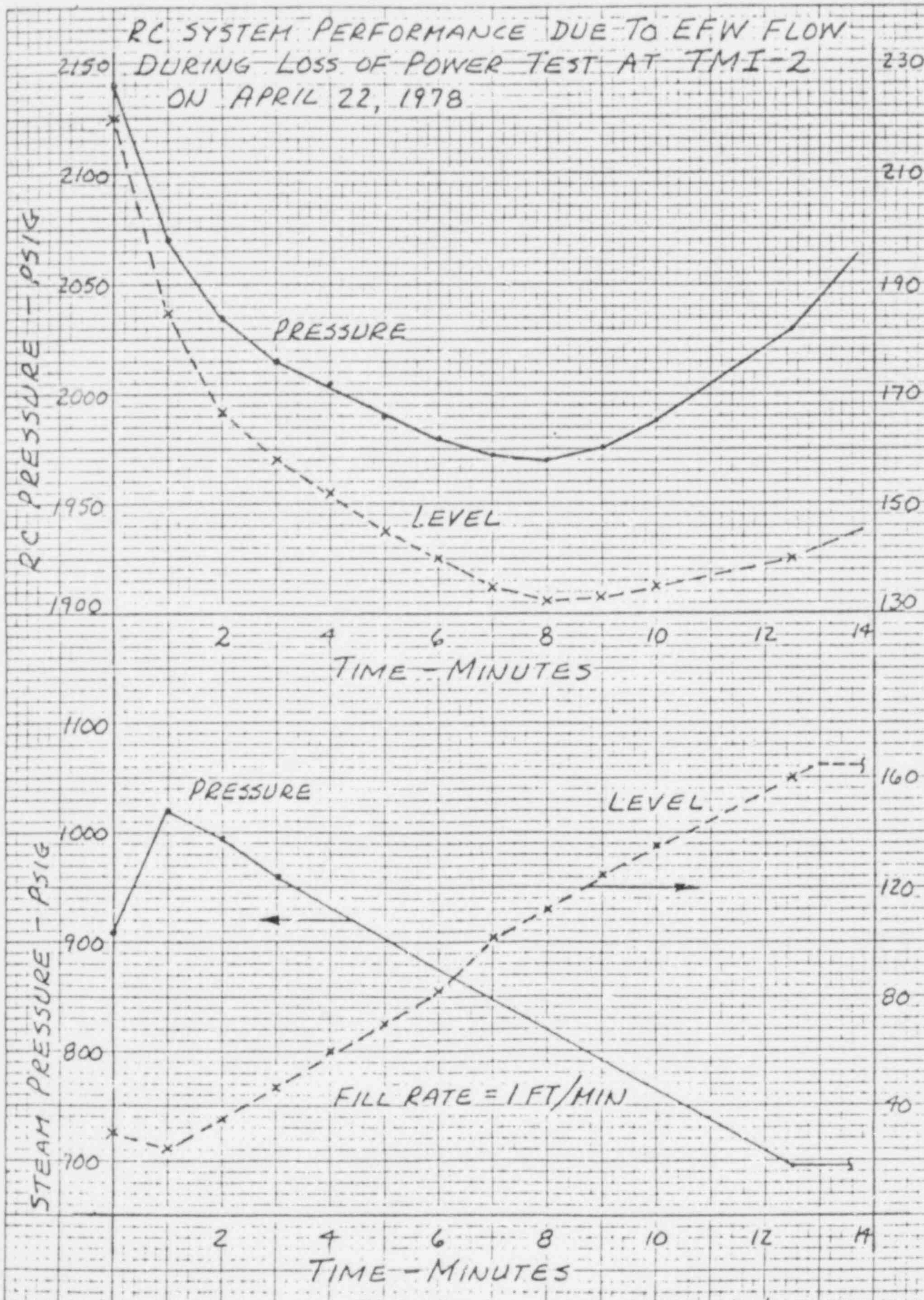
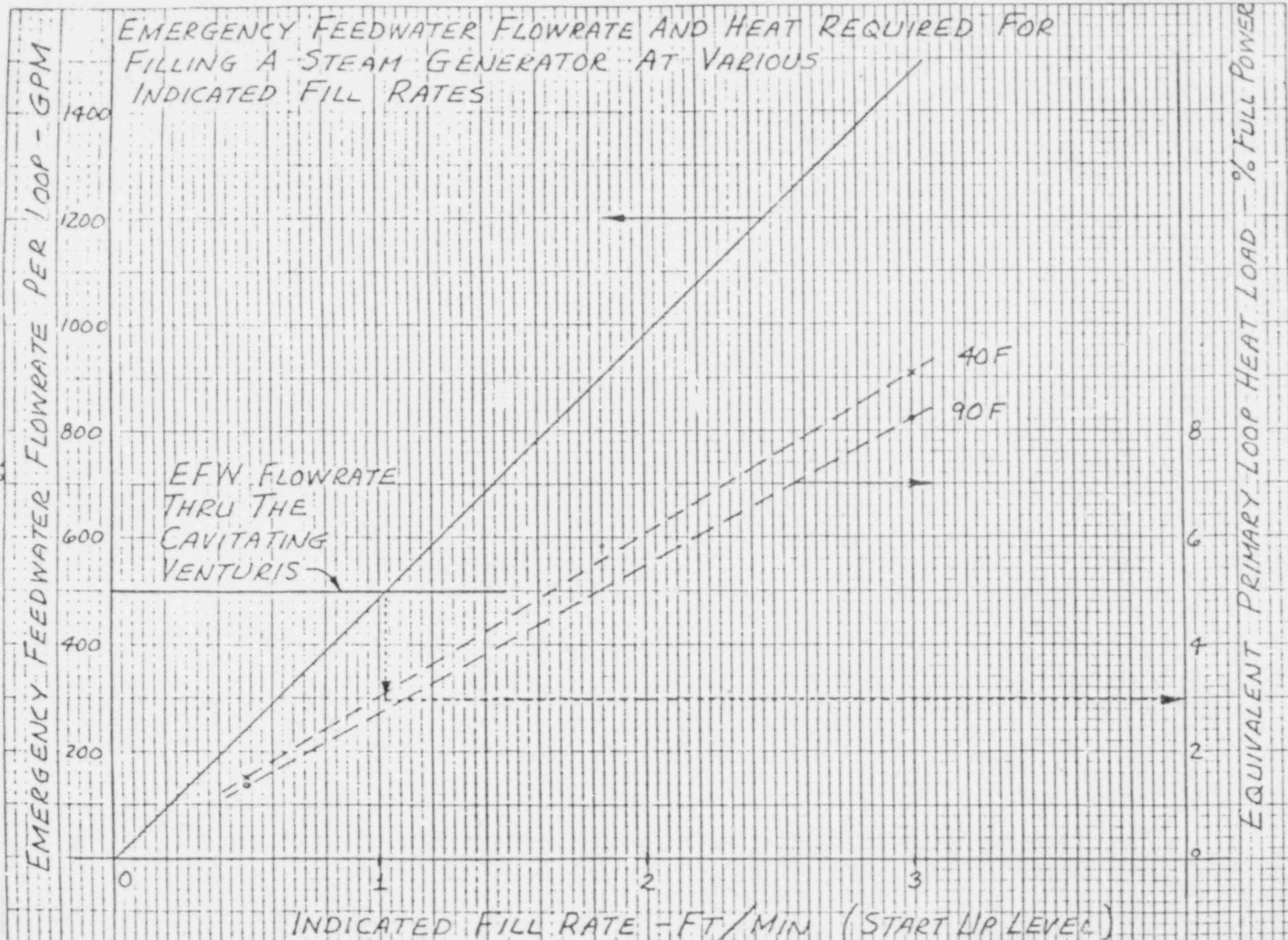


FIGURE 7

8/12/80 R. Winks

86-1120427-01


EMERGENCY FEEDWATER FLOWRATE AND HEAT REQUIRED FOR FILLING A STEAM GENERATOR AT VARIOUS INDICATED FILL RATES
EMERGENCY FEEDWATER FLOWRATE PER LOOP - GPM
1400
1200
1000
800
600
400
200
EQUIVALENT PRIMARY LOOP HEAT LOAD - % FULL POWER
8
6
4
2
0
40F
90F
EFW FLOWRATE THRU THE CAVITATING VENTURIS
0
1
2
3
INDICATED FILL RATE - FT/MIN (START UP LEVEL)
8/12/80 R. WINKS
FIGURE 8