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Revised by P.S. 12-2-83

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Washington Public Power Supply System
P.O. Box 968 3000 George Washington Way Richland, Washington 99352 (509) 372-5000

November 3, 1983
G02-83-1014

Docket No. 50-397

Director of Nuclear Reactor Regulation
Attention: Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
FSAR AMENDMENT NO. 33

The Washington Public Power Supply System herewith submits sixty (60) copies of Amendment No. 33 to its Final Safety Analysis Report for WNP-2.

Pursuant to 10CFR2.101, we are also, at this time, transmitting copies of the amendment to Mr. Nicholas D. Lewis, Chairman of the Energy Facility Site Evaluation Council, and the Board Chairman of the Benton County Commissioners.

Very truly yours,

GC Sorensen

G.C. Sorensen, Acting Manager
Nuclear Safety and Regulatory Programs

SIS
Enclosure

cc: R Auluck - NRC
WS Chin - BPA
A Toth - NRC Site

B001
1/60

STATE OF WASHINGTON)
)ss
COUNTY OF BENTON)

Subject: FSAR Am 33

I, G.C. SORENSEN, being duly sworn, subscribe to and say that I am the Acting Manager, Nuclear Safety and Regulatory Programs, for the WASHINGTON PUBLIC POWER SUPPLY SYSTEM, the applicant herein; that I have full authority to execute this oath; that I have reviewed the foregoing; and that to the best of my knowledge, information and belief, the statements made in it are true.

DATE 2 Nov, 1983

GC Sorensen
G.C. SORENSEN, Acting Manager
Nuclear Safety and Regulatory Programs

On this day personally appeared before me G.C. SORENSEN to me known to be the individual who executed the foregoing instrument and acknowledged that he signed the same as his free act and deed for the uses and purposes herein mentioned.

GIVEN under my hand and seal this 2nd day of November, 1983.

Nancy L. Skinner
Notary Public in and for the State of Washington

Residing at Richland, WA.
My Commission Expires 10/01/85

[illegible]

[illegible]

[illegible]

[illegible]

50-397 Superseded pages Per Amend 33 to FSAR
w/ ltr 83/11/03
8311230085
Revised by P.S. 12-2-83

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2.5-382	18
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2.5-391	18
2.5-392	18
2.5-393	18
2.5-394	18

CHAPTER 2 (Continued)

TEXT PAGES	AMENDMENT
2.5-395	18
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2.5-397	0
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2.5-399	0
2.5-400	27
2.5-401	27
2.5-402	27
Appendix 2.5A	0
Appendix 2.5B	0
Appendix 2.5C	0
Appendix 2.5D	0
Appendix 2.5E	0
Appendix 2.5F	0
Appendix 2.5G	0
Appendix 2.5H	0
Appendix 2.5I	0
Appendix 2.5J	18
Appendix 2.5K	18
Appendix 2.5L	18
Appendix 2.5M	18
Appendix 2.5N	18
Appendix 2.5O	18

FIGURES

2.1-1	0
2.1-2	0
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2.1-4	7
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2.1-13	20
2.1-14	20
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2.1-16	20
2.1-17	20
2.1-18	20
2.1-19	0
2.1-20	0

CHAPTER 2 (Continued)

FIGURES	AMENDMENT
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2.2-4	7
2.2-5	7
2.2-6	0
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2.3-2	0
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2.3-8	0
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2.3-9a	0
2.3-9b	0
2.3-10a	0
2.3-10b	0
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2.3-11b	0
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2.3-13b	0
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2.3-13d	0
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2.4-11	5
2.4-12	5
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2.4-14	0
2.4-15	0
2.4-16	0
2.4-17	0

CHAPTER 2 (Continued)

FIGURES	AMENDMENT
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2.4-21	13
2.4-22	13
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2.4-34	0
2.4-35	23
2.4-36	0
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2.5-3	18
2.5-4	18
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2.5-6b	18
2.5-6c	18
2.5-7	18
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2.5-8b	18
2.5-9a	18
2.5-9b	18
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2.5-12	18
2.5-13	18
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2.5-18	18
2.5-19	18
2.5-20	18
2.5-21	18
2.5-22	18
2.5-23	18

CHAPTER 2 (Continued)

FIGURES	AMENDMENT
2.5-24	18
2.5-25	18
2.5-26	18
2.5-27	18
2.5-28	18
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2.5-64	18
2.5-65	18
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2.5-67	18
2.5-68	18
2.5-69	18
2.5-70	18

CHAPTER 2 (Continued)

FIGURES	AMENDMENT
Appendix 2.5A	0
Appendix 2.5B	0
Appendix 2.5C	0
Appendix 2.5D	0
Appendix 2.5E	0
Appendix 2.5F	0
Appendix 2.5G	0
Appendix 2.5H	0,21
Appendix 2.5I	0
Appendix 2.5J	18
Appendix 2.5K	18
Appendix 2.5L	18
Appendix 2.5N	18
Appendix 2.5O	18

CHAPTER 3

TEXT PAGES	AMENDMENT
3-i	0
3-ii	12
3-iii	0
3-iv	0
3-v	0
3-vi	25
3-vii	0
3-viii	8
3-ix	27
3-ixa	27
3-x	27
3-xa	27
3-xi	0
3-xii	9
3-xiia	25
3-xiii	0
3-xiv	0
3-xv	9
3-xva	31
3-xvb	9
3-xvi	25
3-xvii	31
3-xviii	31
3-xix	9
3-xx	23
3-xxa	23
3-xxi	30
3-xxii	8
3-xxiia	1
3-xxiii	0
3-xxiv	12
3-xxv	0
3-xxvi	32
3-xxvia	8
3-xxvii	0
3-xxviii	0
3-xxix	8
3-xxx	9
3-xxxa	9
3-xxxi	0
3-xxxii	0
3-xxxiii	0
3-xxxiv	29
3-xxxv	32
3-xxxva	29
3-xxxvb	32

CHAPTER 3 (Continued)

TEXT PAGES	AMENDMENT
3-xxxvi	32
3-xxxvii	31
3-xxxviii	29
3-xxxix	29
3-xL	29
3-xLi	31
3-xLii	29
3-xLiii	29
3-xLiv	26
3-xLv	32
3-xLvi	32
3-xLvii	32
3-xLviii	29
3-xLix	29
3-L	27
3-La	27
3-Li	0
3-Lii	0
3-Liii	0
3-Liv	29
3-Liva	29
3-Lv	29
3-Lvi	29
3-Lvii	32
3-Lviii	31
3-Lix	32
3-Lx	0
3-Lxi	12
3-Lxii	23
3-Lxiii	31
3-Lxiiia	4
3-Lxiv	9
3-Lxiva	9
3-Lxv	27
3-Lxva	31
3-Lxvi	2
3-Lxvii	0
3-Lxviii	0
3-Lxix	0
3-Lxx	0
3-Lxxi	0
3-Lxxii	0
3-Lxxiii	9
3-Lxxiiia	9
3-Lxxiiib	9
3-Lxxiiic	9
3-Lxxiv	0

CHAPTER 3 (Continued)

CHAPTER 3 (Continued)

TEXT PAGES	AMENDMENT
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3.1-80	0
3.1-81	0
3.1-82	12
3.1-83	0

CHAPTER 3 (Continued)

TEXT PAGES	AMENDMENT
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3.2-33	31
3.2-33a	30
3.2-34	27
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3.2-36	0
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3.3-2	0
3.3-3	0
3.3-4	8
3.3-4a	8
3.3-4b	8
3.3-5	2

CHAPTER 3 (Continued)

TEXT PAGES	AMENDMENT
3.3-6	0
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3.3-8	0
3.3-9	0
3.4-1	13
3.4-2	5
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3.4-2b	5
3.4-3	0
3.4-4	21
3.4-5	5
3.4-5a	5
3.4-6	0
3.4-7	5
3.5-1	27
3.5-2	27
3.5-3	27
3.5-4	27
3.5-5	27
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3.5-11	27
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3.5-11b	27
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3.5-14b	23
3.5-14c	23
3.5-14d	23
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3.5-15a	27
3.5-16	9
3.5-17	32
3.5-17a	32
3.5-18	0
3.5-19	32
3.5-20	30
3.5-21	0
3.5-22	14

CHAPTER 3 (Continued)

TEXT PAGES	AMENDMENT
3.5-22a	14
3.5-22b	14
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3.5-25	27
3.5-25a	30
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3.5-27a	27
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3.5-29	13
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3.5-31	23
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3.5-38	27
3.5-39	27
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3.5-41	27
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3.6-3	0
3.6-4	25
3.6-5	0
3.6-6	9
3.6-6a	9
3.6-6b	9
3.6-6c	9
3.6-6d	9
3.6-6e	9
3.6-6f	9
3.6-6g	9
3.6-6h	9
3.6-6i	9
3.6-6j	9
3.6-6k	9
3.6-6l	9
3.6-6m	25
3.6-7	25

CHAPTER 3 (Continued)

TEXT PAGES	AMENDMENT
3.6-7a	25
3.6-8	25
3.6-9	25
3.6-10	25
3.6-11	25
3.6-12	25
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3.6-22b	9
3.6-22c	9
3.6-22d	9
3.6-22e	9
3.6-22f	25
3.6-22g	9
3.6-23	31
3.6-24	31
3.6-25	31
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3.6-28	30
3.6-28a	25
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3.6-31	31
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3.6-33	25
3.6-33a	25
3.6-34	31
3.6-35	25
3.6-36	0
3.6-37	31
3.6-38	31
3.6-39	0
3.6-40	0
3.6-41	0
3.6-42	31

CHAPTER 3 (Continued)

CHAPTER 3 (Continued)

TEXT PAGES	AMENDMENT
3.6-81	25
3.6-82	25
3.6-83	25
3.6-84	25
3.6-85	25
3.6-85a	25
3.6-85b	25
3.6-86	25
3.6-87	25
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3.6-90	9
3.6-91	9
3.6-92	9
3.6-93	25
3.6-94	25
3.6-95	9
3.6-96	9
3.6-97	9
3.6-98	9
3.6-99	9
3.6-100	9
3.6-101	25
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3.7-2	0
3.7-3	0
3.7-4	12
3.7-5	30
3.7-6	0
3.7-7	0
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3.7-9	30
3.7-10	23
3.7-11	30
3.7-11a	23
3.7-11b	23
3.7-12	30
3.7-13	8
3.7-13a	8
3.7-14	30
3.7-15	30
3.7-15a	25
3.7-16	30
3.7-17	8
3.7-18	30

CHAPTER 3 (Continued)

CHAPTER 3 (Continued)

CHAPTER 3 (Continued)

TEXT PAGES	AMENDMENT
3.8-27	14
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3.8-41	30
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3.8-42a	8
3.8-43	3
3.8-44	30
3.8-44a	3
3.8-44b	3
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3.8-46	8
3.8-46a	32
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3.8-59	0
3.8-60	30
3.8-61	1
3.8-62	13
3.8-63	30
3.8-64	0
3.8-65	12
3.8-65a	1
3.8-65b	1
3.8-66	13

CHAPTER 3 (Continued)

TEXT PAGES	AMENDMENT
3.8-67	0
3.8-68	0
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3.8-72	13
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3.8-81	0
3.8-82	0
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3.8-87	8
3.8-88	13
3.8-88a	8
3.8-89	8
3.8-89a	8
3.8-90	8
3.8-90a	8
3.8-91	8
3.8-91a	8
3.8-92	5
3.8-92a	0
3.8-93	2
3.8-94	30
3.8-95	5
3.8-96	21
3.8-97	0
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3.8-100	1
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3.8-102	0
3.8-103	9
3.8-103a	9
3.8-104	9
3.8-105	0
3.8-106	0

CHAPTER 3 (Continued)

TEXT PAGES	AMENDMENT
3.8-107	0
3.8-108	0
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3.8-110	0
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3.8-143	0
3.8-144	9
3.8-144a	8
3.8-145	0

CHAPTER 3 (Continued)

TEXT PAGES	AMENDMENT
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3.8-146a	12
3.8-147	9
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3.8-149	0
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3.8-184	26
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3.8-186	0
3.8-187	0
3.8-188	32
3.8-189	27

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CHAPTER 3 (Continued)

CHAPTER 3 (Continued)

TEXT PAGES	AMENDMENT
3.9-117	32
3.9-118	32
3.9-119	29
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3.9-159	29
3.9-160	29
3.9-161	29
3.9-162	29

CHAPTER 3 (Continued)

CHAPTER 3 (Continued)

TEXT PAGES	AMENDMENT
3.9-209	29
3.9-210	29
3.9-211	29
3.9-212	29
3.9-213	29
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3.9-215	29
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3.10-25	26
3.10-26	26
3.10-27	26
3.10-28	26

CHAPTER 3 (Continued)

TEXT PAGES	AMENDMENT
3.10-29	26
3.10-30	26
3.10-31	26
3.10-32	26
3.10-33	26
3.10-34	26
3.10-35	26
3.10A-1	0
3.10A-2	0
3.10A-3	0
3.10A-4	0
3.10A-5	0
3.10A-6	0
3.10A-7	0
3.10B-1	0
3.10B-2	0
3.10B-3	0
3.10B-4	0
3.10B-5	0
3.10B-6	0
3.10B-7	0
3.10B-8	0
3.10B-9	0
3.10B-10	0
3.10B-11	0
3.10B-12	0
3.10B-13	0
3.10B-14	0
3.10B-15	0
3.10B-16	0
3.10C-1	0
3.10C-2	0
3.10C-3	0
3.10C-4	0
3.10C-5	0
3.11-1	32
3.11-2	32
3.11-3	32
3.11-4	32
3.11-5	32
3.11-6	32
3.11-7	32
3.11-8	32
3.11-9	32
3.11-10	32

CHAPTER 3 (Continued)

CHAPTER 3 (Continued)

TEXT PAGES	AMENDMENT
3.12-31	31
3.12-32	31
FIGURES	
3.2-1	16
3.2-2	22
3.2-3	31
3.2-4	22
3.2-5	0
3.2-6	22
3.2-7	29
3.2-8	29
3.2-9	0
3.2-10	0
3.2-11	31
3.2-12	0
3.2-13	0
3.2-14	0
3.2-15	0
3.2-16	22
3.2-17	22
3.2-18	0
3.2-19	0
3.2-20	7
3.2-21	32
3.2-22	0
3.2-23	0
3.2-24	0
3.2-25	22
3.2-26	31
3.2-27	32
3.5-1	0
3.5-2	0
3.5-3	0
3.5-4	0
3.5-5	0
3.5-6	0
3.5-7	0
3.5-8	0
3.5-9	3
3.5-10	0
3.5-11	0
3.5-12	0
3.5-13	22

CHAPTER 3 (Continued)

FIGURES	AMENDMENT
3.5-14	0
3.5-15	0
3.5-16	0
3.5-17	0
3.5-18	0
3.5-19	0
3.5-20	0
3.5-21	0
3.5-22	0
3.5-23	0
3.5-24	0
3.5-25	0
3.5-26	0
3.5-27	0
3.5-28	0
3.5-29	0
3.5-30	0
3.5-31	0
3.5-32	0
3.5-33	23
3.5-34	23
3.5-35	23
3.5-36	0
3.5-37	0
3.5-38	0
3.5-39	0
3.5-40	0
3.5-41	4
3.5-42	0
3.5-43	0
3.5-44	0
3.5-45	0
3.5-46	0
3.5-47	0
3.5-48	0
3.5-49	0
3.5-50	7
3.5-51	4
3.5-52	4
3.5-53	23
3.6-1	0
3.6-2	0
3.6-3	27
3.6-4	31

CHAPTER 3 (Continued)

FIGURES	AMENDMENT
3.6-5	0
3.6-5a	0
3.6-5b	0
3.6-5c	0
3.6-5d	0
3.6-5e	0
3.6-6a	0
3.6-6b	0
3.6-6c	2
3.6-6d	0
3.6-6e	0
3.6-6f	0
3.6-6g	0
3.6-6h	2
3.6-6j	0
3.6-6k	0
3.6-7a	0
3.6-7b	0
3.6-7c	0
3.6-8	0
3.6-9a	2
3.6-9b	0
3.6-10	0
3.6-11	9
3.6-12a	0
3.6-12b	9
3.6-12c	9
3.6-13a	0
3.6-13b	9
3.6-13c	9
3.6-14a	9
3.6-14b	9
3.6-14c	9
3.6-15a	0
3.6-15b	9
3.6-15c	9
3.6-16a	0
3.6-16b	9
3.6-16c	9
3.6-16d	9
3.6-16e	9
3.6-17a	0
3.6-17b	9
3.6-17c	9
3.6-17d	9
3.6-17e	9

CHAPTER 3 (Continued)

CHAPTER 3 (Continued)

FIGURES	AMENDMENT
3.6-34b	9
3.6-34c	9
3.6-35a	31
3.6-35b	31
3.6-36a	2
3.6-36b	2
3.6-36c	2
3.6-37	25
3.6-38	2
3.6-39	2
3.6-40a	2
3.6-40b	2
3.6-41a	25
3.6-41b	25
3.6-41c	25
3.6-41d	25
3.6-41e	25
3.6-41f	25
3.6-41g	25
3.6-41h	0
3.6-42a	0
3.6-42b	0
3.6-42c	0
3.6-42d	0
3.6-42e	0
3.6-42f	0
3.6-42g	0
3.6-42h	0
3.6-43	0
3.6-44	0
3.6-45	0
3.6-46	0
3.6-47	0
3.6-48	0
3.6-49	0
3.6-50	0
3.6-51	0
3.6-52	0
3.6-53	22
3.6-54	0
3.6-55	0
3.6-56	0
3.6-57	0
3.6-58	0
3.6-59	0
3.6-60	0

CHAPTER 3 (Continued)

FIGURES	AMENDMENT
3.6-61	0
3.6-62	0
3.6-63	25
3.6-64	25
3.6-65	25
3.6-66	25
3.6-67	25
3.6-68	25
3.6-69	25
3.6-70	25
3.6-71	25
3.6-72	25
3.6-73	25
3.6-74	25
3.6-75	25
3.6-76	25
3.6-77	25
3.6-78	25
3.6-79	25
3.6-80	25
3.6-81	25
3.6-82	25
3.6-83	25
3.6-84	25
3.6-85	25
3.6-86	25
3.6-87	25
3.6-88	25
3.6-89	25
3.6-90	25
3.6-91	25
3.6-92	25
3.6-93	25
3.6-94	25
3.6-95	25
3.6-96	25
3.6-97	25
3.6-98	25
3.6-99	25
3.6-100	25
3.6-101	25
3.6-102	25
3.6-103	25
3.6-104	25
3.6-105	25
3.6-106	25

CHAPTER 3 (Continued)

CHAPTER 3 (Continued)

FIGURES	AMENDMENT
3.7-1	0
3.7-2	0
3.7-3	0
3.7-4	0
3.7-5	0
3.7-6	0
3.7-7	0
3.7-8	0
3.7-9	0
3.7-10	0
3.7-11a	0
3.7-11b	0
3.7-12a	0
3.7-12b	0
3.7-13	0
3.7-14a	0
3.7-14b	0
3.7-15	0
3.7-16	0
3.7-17	0
3.7-18	0
3.7-19	0
3.7-20	0
3.7-21	0
3.7-22	0
3.7-23	0
3.7-24	0
3.7-25	0
3.7-26	25
3.7-27	8
3.7-28	8
3.8-1	0
3.8-2	2
3.8-3	0
3.8-4	0
3.8-5	0
3.8-6	2
3.8-7	30
3.8-8	30
3.8-9	0
3.8-10	2
3.8-11	0
3.8-12	0
3.8-13	1

CHAPTER 3 (Continued)

FIGURES	AMENDMENT
3.8-14	1
3.8-15	1
3.8-16	0
3.8-17	0
3.8-18	2
3.8-19	0
3.8-20	0
3.8-21	0
3.8-22	0
3.8-23	0
3.8-24	0
3.8-25	0
3.8-26	0
3.8-27	0
3.8-28	0
3.8-29	0
3.8-30	0
3.8-31	0
3.8-32	0
3.8-33	2
3.8-34	0
3.8-35	0
3.8-36	0
3.8-37	0
3.8-38	0
3.8-39	0
3.8-40	0
3.8-41	0
3.8-42	5
3.8-43	0
3.8-44	0
3.8-45	0
3.8-46	0
3.8-47	0
3.8-48	8
3.8-49	0
3.8-50	1
3.8-51	1
3.8-52	1
3.8-53	1
3.8-54	2
3.8-55	2
3.8-56	7
3.8-57	3
3.8-58	3

CHAPTER 3 (Continued)

CHAPTER 3 (Continued)

CHAPTER 4

TEXT PAGES	AMENDMENT
4-i	0
4-ii	0
4-iii	0
4-iv	30
4-v	30
4-vi	30
4-vii	30
4-viii	30
4-ix	30
4-x	30
4.1-1	30
4.1-2	0
4.1-3	0
4.1-4	30
4.1-5	30
4.1-6	0
4.1-7	0
4.1-8	30
4.1-9	30
4.1-10	30
4.1-11	0
4.1-12	0
4.1-13	0
4.1-14	0
4.1-15	0
4.1-16	0
4.1-17	0
4.1-18	0
4.1-19	30
4.1-20	0
4.1-21	30
4.2-1	30
4.2-2	30
4.2-3	30
4.2-4	30
4.2-5	30
4.2-6	30
4.3-1	30
4.3-2	30
4.3-3	30
4.3-4	30
4.3-5	30

CHAPTER 4 (Continued)

TEXT PAGES	AMENDMENT
4.3-6	30
4.3-7	30
4.3-8	30
4.4-1	30
4.4-2	30
4.4-3	30
4.4-4	30
4.4-5	30
4.4-6	30
4.4-7	30
4.4-8	30
4.4-9	30
4.4-10	30
4.4-11	30
4.4-12	30
4.4-13	30
4.4-14	30
4.4-15	30
4.4-16	31
4.4-17	30
4.5-1	0
4.5-2	2
4.5-3	0
4.5-4	26
4.5-5	31
4.5-6	31
4.5-7	31
4.5-8	30
4.5-8a	30
4.5-9	27
4.5-10	0
4.6-1	23
4.6-2	12
4.6-3	13
4.6-4	0
4.6-5	0
4.6-6	0
4.6-7	0
4.6-8	0
4.6-9	21
4.6-10	21

CHAPTER 4 (Continued)

TEXT PAGES	AMENDMENT
4.6-11	21
4.6-12	29
4.6-13	29
4.6-13a	29
4.6-14	0
4.6-15	20
4.6-16	0
4.6-17	0
4.6-18	29
4.6-19	0
4.6-20	0
4.6-21	12
4.6-22	0
4.6-23	0
4.6-24	0
4.6-25	12
4.6-26	12
4.6-27	0
4.6-28	0
4.6-29	0
4.6-30	0
4.6-31	21
4.6-32	0
4.6-33	0
4.6-34	0
4.6-35	0
4.6-36	0
4.6-37	0
4.6-38	0
4.6-39	0
4.6-40	0
4.6-41	0
4.6-42	0
FIGURES	
4.1-1	0
4.1-2	0
4.2-1	30
4.2-2	30
4.2-3	30
4.3-1	30
4.3-2	30

CHAPTER 4 (Continued)

FIGURES	AMENDMENT
4.4-1	30
4.6-1	0
4.6-2	0
4.6-3	0
4.6-4	0
4.6-5a	22
4.6-5b	22
4.6-5c	22
4.6-6a	16
4.6-6b	22
4.6-6c	22
4.6-6d	16
4.6-6e	16
4.6-7	0
4.6-8	0

CHAPTER 4 (Continued)

CHAPTER 5

TEXT PAGES	AMENDMENT
5-i	23
5-ii	23
5-iii	0
5-iv	0
5-v	29
5-vi	12
5-vii	5
5-viii	5
5-ix	0
5-x	32
5-xi	0
5-xii	0
5-xiii	12
5-xiv	0
5-xv	12
5-xva	12
5-xvi	23
5-xvii	12
5-xviii	12
5-xix	12
5.1-1	0
5.1-2	0
5.1-3	0
5.1-4	0
5.2-1	0
5.2-2	0
5.2-3	30
5.2-4	32
5.2-5	30
5.2-5a	23
5.2-6	23
5.2-6a	23
5.2-7	30
5.2-7a	30
5.2-8	30
5.2-9	32
5.2-10	32
5.2-10a	32
5.2-11	32
5.2-12	30
5.2-13	32
5.2-14	30
5.2-15	32
5.2-15a	32

CHAPTER 5 (Continued)

TEXT PAGES	AMENDMENT
5.2-15b	32
5.2-15c	32
5.2-15d	32
5.2-15e	32
5.2-16	32
5.2-17	0
5.2-18	0
5.2-19	0
5.2-20	0
5.2-21	0
5.2-22	0
5.2-23	0
5.2-24	30
5.2-25	31
5.2-26	32
5.2-27	0
5.2-28	27
5.2-29	7
5.2-29a	7
5.2-30	7
5.2-31	7
5.2-32	30
5.2-33	30
5.2-34	7
5.2-34a	7
5.2-35	29
5.2-35a	29
5.2-36	8
5.2-37	13
5.2-38	27
5.2-39	21
5.2-39a	21
5.2-40	8
5.2-41	27
5.2-42	27
5.2-43	0
5.2-44	0
5.2-45	13
5.2-46	21
5.2-47	7
5.2-48	10
5.2-49	20
5.2-50	30
5.2-51	30
5.2-51a	30

CHAPTER 5 (Continued)

TEXT PAGES	AMENDMENT
5.2-52	29
5.2-52a	5
5.2-53	30
5.2-54	0
5.2-55	0
5.2-56	0
5.2-57	0
5.2-58	0
5.2-59	0
5.2-60	7
5.2-61	7
5.2-62	7
5.2-63	7
5.2-64	7
5.2-65	7
5.2-66	7
5.2-67	7
5.2-68	7
5.2-69	7
5.2-70	7
5.2-71	7
5.2-72	7
5.2-73	7
5.2-74	7
5.2-75	7
5.2-76	7
5.2-77	0
5.2-78	0
5.2-79	0
5.3-1	30
5.3-2	30
5.3-3	27
5.3-4	5
5.3-5	5
5.3-5a	5
5.3-5b	5
5.3-5c	5
5.3-6	30
5.3-7	30
5.3-8	5
5.3-9	5
5.3-10	30
5.3-11	5
5.3-12	5
5.3-13	5

CHAPTER 5 (Continued)

TEXT PAGES	AMENDMENT
5.3-14	30
5.3-15	0
5.3-16	0
5.3-17	13
5.3-18	13
5.3-19	30
5.3-20	0
5.3-21	0
5.3-22	0
5.3-23	30
5.3-24	30
5.3-25	30
5.4-1	0
5.4-2	13
5.4-3	8
5.4-4	0
5.4-5	4
5.4-6	0
5.4-7	29
5.4-8	0
5.4-9	0
5.4-10	0
5.4-11	0
5.4-12	12
5.4-13	0
5.4-14	13
5.4-15	0
5.4-16	0
5.4-17	30
5.4-18	32
5.4-18a	32
5.4-19	32
5.4-20	29
5.4-21	0
5.4-22	23
5.4-22a	8
5.4-23	32
5.4-24	30
5.4-25	20
5.4-26	0
5.4-27	21
5.4-28	8
5.4-29	0
5.4-30	0
5.4-31	32

CHAPTER 5 (Continued)

TEXT PAGES	AMENDMENT
5.4-32	32
5.4-33	32
5.4-34	0
5.4-35	0
5.4-36	32
5.4-36a	32
5.4-37	20
5.4-38	8
5.4-39	17
5.4-40	27
5.4-41	11
5.4-42	0
5.4-43	8
5.4-44	0
5.4-45	0
5.4-46	0
5.4-47	13
5.4-48	13
5.4-48a	13
5.4-49	13
5.4-49a	13
5.4-50	12
5.4-50a	12
5.4-51	0
5.4-52	30
5.4-53	0
5.4-54	12
5.4-55	0
5.4-56	0
5.4-57	30
5.4-58	30
5.4-59	14
5.4-60	30
FIGURES	
5.1-1	3
5.1-2	3
5.1-3a	16
5.1-3b	16
5.1-3c	16
5.2-1a	23
5.2-1b	23
5.2-2	0
5.2-3	23

CHAPTER 5 (Continued)

FIGURES	AMENDMENT
5.2-4	23
5.2-5	23
5.2-6	20
5.2-7	0
5.2-8	0
5.2-9	0
5.2-10	31
5.2-11	0
5.2-12	0
5.2-13	0
5.2-14	0
5.3-1	0
5.3-2	0
5.3-3	0
5.3-4	32
5.3-5	5
5.4-1	0
5.4-2a	16
5.4-2b	16
5.4-2c	16
5.4-3a	4
5.4-3b	0
5.4-4a	8
5.4-4b	8
5.4-5	4
5.4-6	0
5.4-7	0
5.4-8	0
5.4-9a	16
5.4-9b	16
5.4-10	16
5.4-11	0
5.4-12	0
5.4-13a	22
5.4-13b	22
5.4-14a	16
5.4-14b	22
5.4-14c	22
5.4-15a	5
5.4-15b	5
5.4-15c	5
5.4-16	16
5.4-17a	16
5.4-17b	16

CHAPTER 5 (Continued)

CHAPTER 6

TEXT PAGES	AMENDMENT
6-i	32
6-ii	0
6-iii	3
6-iv	30
6-v	21
6-va	11
6-vi	30
6-via	30
6-vii	26
6-viia	26
6-viii	0
6-ix	0
6-x	32
6-xi	9
6-xii	9
6-xiii	14
6-xiv	14
6-xv	14
6-xvi	12
6-xvii	32
6-xviia	31
6-xviii	17
6-xix	11
6-xx	31
6-xxa	31
6-xxb	31
6-xxc	31
6-xxi	31
6-xxii	31
6-xxiii	31
6-xxiv	31
6-xxv	31
6-xxvi	31
6-xxvii	31
6-xxviii	0
6.0-1	0
6.1-1	0
6.1-2	0
6.1-3	0
6.1-4	32
6.1-5	32
6.1-6	32
6.1-6a	32
6.1-6b	32

CHAPTER 6 (Continued)

TEXT PAGES	AMENDMENT
6.1-7	0
6.1-8	0
6.1-9	0
6.1-10	2
6.1-11	2
6.2-1	0
6.2-2	12
6.2-3	13
6.2-4	31
6.2-4a	31
6.2-5	0
6.2-6	3
6.2-7	20
6.2-8	0
6.2-9	0
6.2-10	0
6.2-11	0
6.2-12	0
6.2-13	0
6.2-14	0
6.2-15	0
6.2-16	0
6.2-17	0
6.2-18	12
6.2-19	31
6.2-20	0
6.2-21	0
6.2-22	0
6.2-23	0
6.2-24	0
6.2-25	0
6.2-26	0
6.2-27	32
6.2-28	32
6.2-29	29
6.2-30	21
6.2-30a	21
6.2-31	25
6.2-32	25
6.2-33	25
6.2-33a	25
6.2-33b	25
6.2-33c	25

CHAPTER 6 (Continued)

TEXT PAGES	AMENDMENT
6.2-33d	25
6.2-33e	25
6.2-34	0
6.2-35	12
6.2-36	30
6.2-36a	30
6.2-37	0
6.2-38	32
6.2-39	0
6.2-40	0
6.2-41	0
6.2-42	0
6.2-43	0
6.2-44	0
6.2-45	3
6.2-46	0
6.2-47	4
6.2-48	11
6.2-49	4
6.2-50	29
6.2-50a	26
6.2-50b	11
6.2-50c	11
6.2-50d	11
6.2-50e	11
6.2-51	11
6.2-51a	11
6.2-52	13
6.2-53	0
6.2-54	13
6.2-55	5
6.2-56	10
6.2-57	0
6.2-58	21
6.2-58a	29
6.2-59	5
6.2-60	27
6.2-61	27
6.2-61a	27
6.2-62	27
6.2-63	27
6.2-63a	27
6.2-64	25
6.2-64a	25
6.2-65	31
6.2-65a	31

CHAPTER 6 (Continued)

TEXT PAGES	AMENDMENT
6.2-66	30
6.2-66a	30
6.2-67	5
6.2-68	12
6.2-68a	5
6.2-69	5
6.2-70	26
6.2-70a	26
6.2-71	26
6.2-71a	26
6.2-72	26
6.2-72a	26
6.2-73	26
6.2-73a	26
6.2-74	26
6.2-74a	26
6.2-75	26
6.2-76	26
6.2-76a	26
6.2-77	26
6.2-77a	26
6.2-78	26
6.2-78a	26
6.2-79	26
6.2-80	13
6.2-80a	26
6.2-80b	32
6.2-80c	27
6.2-81	30
6.2-82	0
6.2-83	25
6.2-84	30
6.2-85	26
6.2-85a	26
6.2-86	25
6.2-86a	25
6.2-87	31
6.2-88	12
6.2-88a	11
6.2-89	0
6.2-90	31
6.2-91	31
6.2-92	31
6.2-93	31
6.2-94	31
6.2-95	0

CHAPTER 6 (Continued)

TEXT PAGES	AMENDMENT
6.2-96	17
6.2-97	0
6.2-98	0
6.2-99	0
6.2-100	31
6.2-101	0
6.2-102	0
6.2-103	31
6.2-104	0
6.2-105	0
6.2-106	0
6.2-107	26
6.2-108	12
6.2-109	16
6.2-110	16
6.2-111	16
6.2-112	16
6.2-113	16
6.2-114	16
6.2-115	16
6.2-116	16
6.2-117	30
6.2-118	0
6.2-119	27
6.2-120	27
6.2-121	29
6.2-122	27
6.2-123	29
6.2-124	25
6.2-125	27
6.2-125a	12
6.2-126	25
6.2-127	25
6.2-128	31
6.2-129	27
6.2-130	25
6.2-131	29
6.2-132	31
6.2-133	12
6.2-134	12
6.2-135	27
6.2-135a	27
6.2-136	29
6.2-137	3
6.2-138	5
6.2-139	29

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CHAPTER 6 (Continued)

TEXT PAGES	AMENDMENT
6.3-10	13
6.3-11	31
6.3-12	31
6.3-13	13
6.3-14	31
6.3-15	0
6.3-16	4
6.3-17	0
6.3-18	31
6.3-19	20
6.3-19a	31
6.3-20	31
6.3-21	31
6.3-21a	5
6.3-22	13
6.3-23	31
6.3-24	31
6.3-25	0
6.3-26	31
6.3-26a	31
6.3-27	8
6.3-27a	8
6.3-28	31
6.3-29	0
6.3-30	31
6.3-30a	31
6.3-31	31
6.3-32	31
6.3-33	0
6.3-34	13
6.3-35	20
6.3-36	31
6.3-37	31
6.3-38	31
6.3-38a	31
6.3-39	31
6.3-40	31
6.3-41	31
6.3-42	31
6.3-43	31
6.3-44	31
6.3-45	11
6.3-46	31
6.3-47	31
6.3-48	31
6.3-49	31

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CHAPTER 6 (Continued)

TEXT PAGES	AMENDMENT
6.4-9	9
6.4-9a	9
6.4-9b	9
6.4-9c	11
6.4-9d	11
6.4-9e	9
6.4-9f	9
6.4-10	13
6.4-10a	9
6.4-11	17
6.4-12	9
6.5-1	11
6.5-1a	11
6.5-2	0
6.5-3	11
6.5-4	26
6.5-5	0
6.5-6	5
6.5-7	5
6.5-8	5
6.5-9	13
6.5-9a	13
6.5-10	32
6.5-11	32
6.5-12	3
6.5-13	13
6.5-14	0
6.5-15	0
6.5-16	0
6.5-17	0
6.5-18	5
6.5-19	5
6.5-20	26
6.5-21	26
6.5-22	26
6.5-23	0
6.5-24	0
6.5-25	5
6.6-1	7
6.6-1a	7
6.6-2	30
6.6-3	30
6.6-4	7

CHAPTER 6 (Continued)

TEXT PAGES	AMENDMENT
6.6-5	7
6.6-6	7
6.6-7	7
6.6-8	7
6.6-9	7
6.6-10	7
6.6-11	7
6.6-12	7
6.6-13	7
6.6-14	7
6.6-15	7
6.7-1	0
6.7-2	5
6.7-3	14
6.7-4	14
6.7-5	8
6.7-6	0
6.7-7	0
6.7-8	0
6.7-9	0
6.7-10	0
6.7-11	5
6.7-12	0
6.7-13	0
6.7-14	14
6.7-15	14
6.7-16	14
6.7-17	14
6.7-18	14
6.7-19	14
6.7-20	14
6.7-21	0
FIGURES	
6.2-1	0
6.2-2	0
6.2-3	0
6.2-4	0
6.2-5	0
6.2-1	0
6.2-2	0
6.2-3	0

CHAPTER 6 (Continued)

CHAPTER 6 (Continued)

CHAPTER 6 (Continued)

CHAPTER 6 (Continued)

FIGURES	AMENDMENT
6.3-57a	31
6.3-57b	31
6.3-58a	31
6.3-58b	31
6.3-59a	31
6.3-59b	31
6.3-60a	31
6.3-60b	31
6.3-61	31
6.A.2-1	32
6.A.3-1	32
6.A.3-2	32
6.A.3-3	32
6.A.3-4	32
6.A.4-1	32
6.A.5-1	32
6.4-1	9
6.4-2	7
6.7-1	0
6.7-2	0

CHAPTER 7

TEXT PAGES	AMENDMENT
7-i	10
7-ii	10
7-iii	30
7-iv	23
7-iva	23
7-v	10
7-vi	23
7-vii	10
7-viii	10
7-ix	23
7-x	21
7-xi	23
7-xii	10
7-xiii	32
7-xiv	29
7-xv	29
7.1-1	10
7.1-2	10
7.1-3	10
7.1-4	12
7.1-5	12
7.1-6	10
7.1-7	27
7.1-8	10
7.1-9	10
7.1-10	27
7.1-11	27
7.1-12	10
7.1-13	12
7.1-14	10
7.1-15	10
7.1-16	10
7.1-17	32
7.2-1	10
7.2-2	23
7.2-2a	23
7.2-3	10
7.2-4	10
7.2-5	10
7.2-6	10
7.2-7	10
7.2-8	29
7.2-8a	29
7.2-9	10

CHAPTER 7 (Continued)

CHAPTER 7 (Continued)

TEXT PAGES	AMENDMENT
7.3-21	31
7.3-22	10
7.3-23	21
7.3-23a	21
7.3-24	10
7.3-25	12
7.3-25a	12
7.3-26	10
7.3-27	10
7.3-28	10
7.3-29	12
7.3-30	13
7.3-31	10
7.3-32	10
7.3-33	10
7.3-34	10
7.3-35	10
7.3-36	10
7.3-37	10
7.3-38	10
7.3-39	10
7.3-40	10
7.3-41	10
7.3-42	10
7.3-43	31
7.3-44	10
7.3-45	10
7.3-46	31
7.3-47	10
7.3-48	10
7.3-49	10
7.3-50	10
7.3-51	10
7.3-52	10
7.3-53	10
7.3-54	10
7.3-55	10
7.3-56	10
7.3-57	10
7.3-58	10
7.3-59	10
7.3-60	10
7.3-61	10
7.3-62	10
7.3-63	10
7.3-64	10

CHAPTER 7 (Continued)

TEXT PAGES	AMENDMENT
7.3-65	10
7.3-66	10
7.3-67	10
7.3-68	10
7.3-69	10
7.3-70	10
7.3-71	10
7.3-72	10
7.3-73	10
7.3-74	10
7.3-75	10
7.3-76	10
7.3-77	14
7.3-78	10
7.3-79	21
7.4-1	30
7.4-2	12
7.4-3	32
7.4-3a	32
7.4-4	10
7.4-5	32
7.4-6	13
7.4-6a	13
7.4-7	10
7.4-8	12
7.4-9	12
7.4-9a	12
7.4-10	30
7.4-10a	30
7.4-11	12
7.4-11a	12
7.4-12	10
7.4-13	12
7.4-14	12
7.4-14a	12
7.4-15	12
7.4-16	12
7.4-17	10
7.4-18	10
7.4-19	12
7.4-20	12
7.4-21	10
7.4-22	10
7.4-23	10

CHAPTER 7 (Continued)

TEXT PAGES	AMENDMENT
7.5-1	23
7.5-1a	23
7.5-2	23
7.5-2a	23
7.5-3	23
7.5-4	23
7.5-4a	23
7.5-5	23
7.5-6	23
7.5-7	23
7.5-8	23
7.5-9	23
7.5-10	23
7.5-11	23
7.5-12	23
7.5-13	23
7.5-14	23
7.5-15	23
7.5-16	23
7.5-17	23
7.5-18	23
7.5-19	23
7.5-20	23
7.5-21	23
7.5-22	29
7.5-23	29
7.5-24	23
7.5-25	23
7.5-26	23
7.5-27	23
7.5-28	23
7.5-29	29
7.5-30	23
7.5-31	23
7.5-32	29
7.5-33	23
7.5-34	23
7.5-35	23
7.5-36	23
7.6-1	10
7.6-2	21
7.6-3	27
7.6-4	10
7.6-5	27
7.6-6	10

CHAPTER 7 (Continued)

TEXT PAGES	AMENDMENT
7.6-7	10
7.6-8	29
7.6-8a	29
7.6-9	31
7.6-9a	21
7.6-10	21
7.6-11	12
7.6-13	10
7.6-14	21
7.6-15	21
7.6-15a	21
7.6-16	21
7.6-17	10
7.6-18	12
7.6-19	12
7.6-20	10
7.6-21	12
7.6-22	10
7.6-23	10
7.6-24	12
7.6-25	13
7.6-26	27
7.6-27	27
7.6-28	10
7.6-29	10
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7.6-33	10
7.6-34	10
7.6-35	10
7.6-36	10
7.6-37	10
7.6-38	10
7.6-39	21
7.6-40	10
7.6-41	10
7.6-42	21
7.6-43	10
7.6-44	10
7.6-45	10
7.6-46	10
7.6-47	10
7.6-48	10
7.6-49	21

CHAPTER 7 (Continued)

TEXT PAGES	AMENDMENT
7.7-1	23
7.7-2	10
7.7-3	10
7.7-4	10
7.7-5	10
7.7-6	10
7.7-6a	10
7.7-7	10
7.7-8	10
7.7-9	29
7.7-10	10
7.7-11	10
7.7-12	10
7.7-13	10
7.7-14	10
7.7-15	10
7.7-16	10
7.7-17	12
7.7-18	10
7.7-19	10
7.7-20	10
7.7-21	10
7.7-22	10
7.7-23	10
7.7-24	10
7.7-25	10
7.7-26	10
7.7-27	10
7.7-28	10
7.7-29	12
7.7-30	10
7.7-31	10
7.7-32	10
7.7-33	10
7.7-34	10
7.7-35	10
7.7-36	10
7.7-37	10
7.7-38	10
7.7-39	10
7.7-40	10
7.7-41	10
7.7-42	10
7.7-43	10
7.7-44	21
7.7-45	21

CHAPTER 7 (Continued)

TEXT PAGES	AMENDMENT
7.7-45a	21
7.7-46	21
7.7-46a	23
7.7-46b	23
7.7-46c	23
7.7-46d	30
7.7-46e	29
7.7-47	10
7.7-48	23
7.7-49	23
7.7-50	10
7.7-51	10
7.7-52	10

FIGURES	
7.2-1a	16
7.2-1b	16
7.2-1c	16
7.2-1d	16
7.2-2	10
7.2-3	0
7.2-4	0
7.2-5	0
7.2-6	0
7.2-7	0
7.2-8a	22
7.2-8b	22
7.2-8c	22
7.2-9	0
7.2-10a	10
7.2-10b	10
7.2-11	23
7.3-1	16
7.3-2	10
7.3-3	10
7.3-4a	16
7.3-4b	16
7.3-4c	16
7.3-5	0
7.3-6	0
7.3-7	16
7.3-8a	16
7.3-8b	16
7.3-8c	16

CHAPTER 7 (Continued)

FIGURES	AMENDMENT
7.3-9a	22
7.3-9b	22
7.3-9c	22
7.3-10a	16
7.3-10b	16
7.3-10c	16
7.3-10d	16
7.3-10e	16
7.3-10f	16
7.3-11	16
7.3-12a	16
7.3-12b	16
7.3-13a	16
7.3-13b	16
7.3-14a	16
7.3-14b	16
7.3-14c	16
7.3-14d	16
7.3-14e	16
7.3-15a	10
7.3-15b	10
7.3-15c	10
7.3-15d	10
7.3-15e	10
7.3-15f	10
7.3-15g	10
7.3-16a	16
7.3-16b	16
7.3-16c	16
7.3-17a	10
7.3-17b	10
7.3-17c	10
7.3-17d	10
7.3-17e	10
7.3-17f	10
7.3-17g	10
7.3-17h	10
7.3-17i	10
7.3-17j	10
7.3-17k	10
7.3-17l	10
7.3-17m	10
7.3-17n	10
7.3-17o	10
7.3-17p	10
7.3-17q	10

CHAPTER 7 (Continued)

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CHAPTER 7 (Continued)

FIGURES	AMENDMENT
7.7-15	10
7.7-16	23

CHAPTER 8

TEXT PAGES	AMENDMENT
8-i	31
8-ii	23
8-iii	31
8-iv	23
8-v	31
8-va	31
8-vb	31
8-vc	31
8-vd	31
8-ve	29
8-vi	23
8-vii	32
8-viii	31
8-ix	31
8-ixa	31
8-x	31
8-xi	0
8-xii	12
8-xiii	31
8-xiv	31
8-xv	31
8-xvi	23
8.1-1	31
8.1-2	0
8.1-3	0
8.1-4	0
8.1-5	0
8.1-6	0
8.1-7	0
8.1-8	0
8.1-9	0
8.1-10	0
8.1-11	0
8.1-12	0
8.1-13	0
8.1-14	23
8.1-15	23
8.1-16	23
8.1-17	0
8.2-1	31
8.2-2	31
8.2-2a	31
8.2-3	0
8.2-4	31

CHAPTER 8 (Continued)

TEXT PAGES	AMENDMENT
8.2-5	0
8.2-6	0
8.2-7	0
8.2-8	0
8.2-9	0
8.2-10	0
8.2-11	31
8.2-12	0
8.2-13	0
8.2-14	0
8.2-15	0
8.3-1	27
8.3-2	27
8.3-3	23
8.3-4	26
8.3-4a	27
8.3-5	23
8.3-6	27
8.3-6a	23
8.3-7	23
8.3-8	23
8.3-8a	23
8.3-9	23
8.3-10	31
8.3-11	23
8.3-12	32
8.3-13	27
8.3-13a	27
8.3-14	23
8.3-14a	23
8.3-15	27
8.3-15a	23
8.3-16	27
8.3-17	31
8.3-17a	23
8.3-18	23
8.3-19	23
8.3-20	31
8.3-21	31
8.3-22	23
8.3-23	23
8.3-24	31
8.3-24a	31
8.3-25	23

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TEXT PAGES	AMENDMENT
8.3-26	0
8.3-27	23
8.3-28	31
8.3-29	31
8.3-29a	23
8.3-30	9
8.3-31	0
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8.3-33	23
8.3-34	23
8.3-35	23
8.3-36	23
8.3-37	27
8.3-38	23
8.3-38a	23
8.3-39	27
8.3-40	27
8.3-41	23
8.3-42	23
8.3-43	30
8.3-44	30
8.3-45	27
8.3-46	23
8.3-47	23
8.3-48	27
8.3-48a	23
8.3-49	30
8.3-49a	23
8.3-50	23
8.3-51	23
8.3-52	23
8.3-52a	26
8.3-52b	23
8.3-53	31
8.3-54	31
8.3-55	31
8.3-56	31
8.3-56a	31
8.3-57	31
8.3-57a	31
8.3-57b	31
8.3-57c	31
8.3-57d	31
8.3-57e	31

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TEXT PAGES	AMENDMENT
8.3-81	0
8.3-82	31
8.3-83	31
8.3-84	31
8.3-85	0
8.3-86	0
8.3-87	0
8.3-88	2
8.3-89	23
8.3-90	0
8.3-91	1
8.3-92	14
8.3-93	14
8.3-94	23
8.3-95	1
8.3-96	0
8.3-97	0
8.3-98	23
8.3-99	31
8.3-100	31
8.3-101	31
8.3-102	23
8.3-103	31
8.3-104	31
8.3-105	31
8.3-106	31
8.3-107	23

FIGURES

8.1-1	31
8.1-1a	31
8.1-2	31
8.1-3	31
8.1-4	31
8.1-5	31
8.1-6	31
8.1-7	0
8.1-8	0
8.1-9a	22
8.1-9b	22
8.1-9c	22
8.1-9d	22
8.1-10	16

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CHAPTER 8 (Continued)

FIGURES	AMENDMENT
8.3-17b	30
8.3-17c	27
8.3-18a	27
8.3-18b	27
8.3-18c	23
8.3-19	22
8.3-20	31
8.3-21	31
8.3-22	31
8.3-23	0
8.3-24a	16
8.3-24b	16
8.3-24c	16
8.3-25a	0
8.3-25b	0
8.3-25c	0
8.3-25d	0
8.3-26a	0
8.3-26b	0
8.3-27	0
8.3-28	0
8.3-29a	31
8.3-29b	31
8.3-29c	31
8.3-29d	31
8.3-30	27
8.3-31	23
8.3-32	27
8.3-33	23
8.3-34	23
8.3-35	23
8.3-36	23
8.3-37	31
8.3-38	23
8.3-39	23
8.3-40	27
8.3-41	31
8.3-42	31
8.3-43a	27
8.3-43b	23
8.3-43c	27
8.3-43d	27
8.3-43e	23

CHAPTER 9

TEXT PAGES	AMENDMENT
9-i	27
9-ii	21
9-iii	0
9-iv	0
9-v	12
9-vi	0
9-vii	23
9-viia	23
9-viii	8
9-viiia	8
9-ix	0
9-x	32
9-xi	0
9-xii	0
9-xiii	32
9-xiv	30
9-xiva	30
9-xv	30
9-xvi	7
9-xvii	26
9-xviia	26
9-xviii	25
9-xviiia	12
9-xix	26
9-xixa	26
9-xx	12
9-xxi	31
9-xxia	8
9-xxii	12
9-xxiii	31
9.1-1	0
9.1-2	23
9.1-2a	23
9.1-3	13
9.1-4	29
9.1-4a	29
9.1-5	0
9.1-6	0
9.1-7	27
9.1-8	5
9.1-9	5
9.1-9a	5
9.1-10	0
9.1-11	0
9.1-12	12
9.1-13	0

CHAPTER 9 (Continued)

TEXT PAGES	AMENDMENT
9.1-14	1
9.1-15	23
9.1-15a	23
9.1-16	5
9.1-17	0
9.1-18	0
9.1-19	32
9.1-20	0
9.1-21	0
9.1-22	13
9.1-23	30
9.1-23a	30
9.1-24	30
9.1-24a	30
9.1-25	25
9.1-26	25
9.1-26a	30
9.1-26b	25
9.1-26c	30
9.1-27	30
9.1-28	30
9.1-29	1
9.1-30	0
9.1-31	0
9.1-32	0
9.1-33	0
9.1-34	0
9.1-35	0
9.1-36	0
9.1-37	0
9.1-38	0
9.1-39	1
9.1-40	0
9.1-41	0
9.1-42	13
9.1-43	32
9.1-44	0
9.1-45	0
9.1-46	0
9.1-47	0
9.1-48	0
9.1-49	30
9.1-50	0
9.1-51	0
9.1-52	0
9.1-53	0

CHAPTER 9 (Continued)

CHAPTER 9 (Continued)

TEXT PAGES	AMENDMENT
9.2-21a	32
9.2-22	32
9.2-23	30
9.2-24	0
9.2-25	0
9.2-26	8
9.2-27	30
9.2-27a	8
9.2-28	0
9.2-29	25
9.2-30	25
9.2-31	25
9.2-31a	21
9.2-32	0
9.2-33	0
9.2-33a	23
9.2-34	32
9.2-35	32
9.2-36	32
9.2-37	5
9.2-38	5
9.2-39	23
9.2-40	32
9.2-40a	32
9.2-41	32
9.2-41a	32
9.2-42	21
9.2-43	0
9.2-44	0
9.2-45	0
9.2-46	23
9.2-47	23
9.2-48	32
9.2-49	23
9.2-50	11
9.3-1	32
9.3-2	30
9.3-3	26
9.3-4	26
9.3-4a	26
9.3-5	26
9.3-6	26
9.3-7	0
9.3-8	0
9.3-9	0

CHAPTER 9 (Continued)

CHAPTER 9 (Continued)

CHAPTER 9 (Continued)

TEXT PAGES	AMENDMENT
9.4-52	0
9.4-53	21
9.4-53a	21
9.4-54	21
9.4-55	32
9.4-56	0
9.4-57	0
9.4-58	32
9.4-58a	32
9.4-59	27
9.4-60	11
9.4-61	32
9.4-61a	27
9.4-62	0
9.4-63	25
9.4-63a	25
9.4-64	0
9.4-65	32
9.4-66	0
9.4-67	0
9.4-68	32
9.4-68a	32
9.4-69	0
9.4-70	0
9.4-71	0
9.4-72	32
9.4-72a	32
9.4-73	0
9.4-74	0
9.4-75	32
9.4-75a	2
9.4-75b	2
9.4-75c	30
9.4-76	0
9.4-77	0
9.4-78	0
9.4-79	2
9.4-80	21
9.4-81	0
9.4-82	0
9.4-83	0
9.4-84	0
9.4-85	9
9.4-86	2
9.4-87	0
9.4-88	0

CHAPTER 9 (Continued)

TEXT PAGES	AMENDMENT
9.4-89	0
9.4-90	0
9.4-91	2
9.4-92	2
9.4-93	0
9.4-94	0
9.4-95	0
9.4-96	0
9.4-97	0
9.4-98	0
9.4-99	2
9.4-100	0
9.4-101	0
9.4-102	2
9.4-103	2
9.5-1	30
9.5-2	30
9.5-3	30
9.5-4	30
9.5-5	30
9.5-6	30
9.5-7	30
9.5-8	30
9.5-9	30
9.5-10	31
9.5-11	30
9.5-12	30
9.5-13	30
9.5-14	30
9.5-15	30
9.5-16	30
9.5-17	30
9.5-18	30
9.5-19	30
9.5-20	30
9.5-21	30
9.5-22	30
9.5-23	30
9.5-24	30
9.5-25	30
9.5-26	30
9.5-27	30
9.5-28	30
9.5-29	30
9.5-30	30

CHAPTER 9 (Continued)

CHAPTER 9 (Continued)

TEXT PAGES	AMENDMENT
9.5-69	30
9.5-70	30
9.5-71	30
9.5-71a	30
9.5-72	30
9.5-73	30
9.5-74	30
9.5-75	30
9.5-76	30
9.5-77	7
9.5-78	26

FIGURES	
9.1-1	0
9.1-2	0
9.1-3	0
9.1-4	30
9.1-5	0
9.1-6	0
9.1-7	0
9.1-8	0
9.1-9	0
9.1-10	0
9.1-11	0
9.1-12	0
9.1-13	0
9.1-14	0
9.1-15	0
9.1-16	0
9.1-17	5
9.2-1	22
9.2-2	22
9.2-3	22
9.2-4	22
9.2-5	22
9.2-6	16
9.2-7a	23
9.2-7b	23
9.2-7c	23
9.2-7d	23
9.2-7e	32
9.2-8	32
9.2-9	22

CHAPTER 9 (Continued)

FIGURES	AMENDMENT
9.2-10	22
9.2-11	0
9.2-12	0
9.2-13	0
9.3-1	31
9.3-2	32
9.3-3	0
9.3-4	0
9.3-5	22
9.3-6	22
9.3-7	22
9.3-8	22
9.3-9	22
9.3-10	22
9.3-11	22
9.3-12	22
9.3-13	22
9.3-14	4
9.3-15	0
9.3-16	0
9.4-1	22
9.4-2	22
9.4-3	22
9.4-4	22
9.4-5	22
9.4-6	22
9.4-7	30
9.4-8	22
9.4-9	22
9.4-10	22
9.4-11	22
9.4-12	22
9.4-13	30
9.5-1	22
9.5-2a	0
9.5-2b	0
9.5-3	11
9.5-4	27
9.5-5	32
9.5-6	31
9.5-7	31
9.5-8	31

CHAPTER 10

TEXT PAGES	AMENDMENT
10-i	12
10-ii	0
10-iii	21
10-iv	0
10-v	0
10-vi	13
10-vii	0
10.1-1	12
10.1-2	0
10.1-3	0
10.1-4	0
10.2-1	0
10.2-2	7
10.2-3	0
10.2-4	13
10.2-5	7
10.2-6	13
10.2-7	30
10.2-7a	7
10.2-8	13
10.2-8a	13
10.2-9	13
10.2-10	3
10.2-11	9
10.3-1	0
10.3-2	13
10.3-3	0
10.3-4	29
10.3-5	30
10.3-6	30
10.4-1	0
10.4-2	7
10.4-3	13
10.4-3a	7
10.4-4	30
10.4-4a	13
10.4-5	2
10.4-6	5
10.4-7	0
10.4-8	0
10.4-9	0
10.4-10	0

CHAPTER 10 (Continued)

CHAPTER 10 (Continued)

CHAPTER 11

TEXT PAGES	AMENDMENT
11-i	0
11-ii	0
11-iii	31
11-iv	31
11-v	0
11-vi	25
11-via	25
11-vii	29
11-viii	29
11-ix	0
11-x	0
11-xi	20
11-xii	20
11-xiia	20
11.1-1	0
11.1-2	0
11.1-3	0
11.1-4	0
11.1-5	0
11.1-6	0
11.1-7	0
11.1-8	0
11.1-9	0
11.1-10	0
11.1-11	0
11.1-12	0
11.1-13	0
11.1-14	0
11.1-15	0
11.1-16	0
11.1-17	0
11.1-18	0
11.1-19	0
11.2-1	5
11.2-2	0
11.2-3	32
11.2-4	0
11.2-5	0
11.2-6	0
11.2-7	0
11.2-8	0
11.2-9	0
11.2-10	0
11.2-11	0

CHAPTER 11 (Continued)

TEXT PAGES	AMENDMENT
11.2-12	0
11.2-13	0
11.2-14	0
11.2-15	0
11.2-16	0
11.2-17	0
11.2-18	0
11.2-19	0
11.2-20	0
11.2-21	0
11.2-22	0
11.2-23	0
11.2-24	0
11.2-25	0
11.2-26	0
11.2-27	0
11.2-28	0
11.2-29	0
11.2-30	0
11.2-31	0
11.2-32	0
11.2-33	0
11.2-34	0
11.2-35	0
11.2-36	0
11.2-37	0
11.2-38	0
11.2-39	0
11.2-40	0
11.2-41	0
11.2-42	0
11.2-43	0
11.2-44	0
11.2-45	0
11.2-46	0
11.2-47	0
11.3-1	0
11.3-2	27
11.3-3	30
11.3-3a	30
11.3-4	0
11.3-5	30
11.3-6	30
11.3-7	30
11.3-8	30
11.3-9	30

CHAPTER 11 (Continued)

TEXT PAGES	AMENDMENT
11.3-10	30
11.3-11	30
11.3-12	30
11.3-13	0
11.3-14	13
11.3-15	30
11.3-16	30
11.3-17	30
11.3-18	0
11.3-19	30
11.3-20	30
11.3-21	0
11.3-22	0
11.3-23	0
11.3-24	0
11.3-25	0
11.3-26	0
11.3-27	0
11.3-28	0
11.3-29	0
11.3-30	0
11.3-31	0
11.3-32	0
11.3-33	7
11.3-34	0
11.4-1	0
11.4-2	20
11.4-3	20
11.4-4	20
11.4-4a	20
11.4-5	20
11.4-6	20
11.4-7	20
11.4-8	0
11.4-9	20
11.4-10	20
11.4-10a	20
11.4-11	20
11.4-11a	25
11.4-11b	25
11.4-11c	25
11.4-11d	25
11.4-11e	25

CHAPTER 11 (Continued)

TEXT PAGES	AMENDMENT
11.4-12	20
11.4-13	20
11.4-14	20
11.4-14a	20
11.4-15	20
11.4-16	20
11.4-16a	20
11.4-17	20
11.5-1	0
11.5-2	2
11.5-3	0
11.5-4	0
11.5-5	29
11.5-6	29
11.5-7	29
11.5-7a	29
11.5-8	0
11.5-9	29
11.5-10	27
11.5-11	29
11.5-12	29
11.5-13	29
11.5-14	29
11.5-15	29
11.5-16	29
11.5-17	29
11.5-18	29
11.5-18a	29
11.5-19	29
11.5-20	0
11.5-21	29
11.5-22	0
11.5-23	29
11.5-24	29
11.5-25	29
11.5-26	29
11.5-27	0
11.5-28	0
11.5-29	0
11.5-30	0
11.5-31	0

CHAPTER 11 (Continued)

CHAPTER 12

TEXT PAGES	AMENDMENT
12-i	0
12-ii	5
12-iii	0
12-iv	5
12-v	0
12-vi	12
12-via	12
12-vii	5
12-viii	27
12-ix	31
12-x	31
12.1-1	20
12.1-1a	27
12.1-2	25
12.1-3	11
12.1-4	0
12.1-5	27
12.1-6	27
12.1-7	0
12.1-8	0
12.1-9	18
12.1-9a	20
12.1-10	11
12.1-11	0
12.1-12	11
12.1-13	11
12.2-1	0
12.2-2	16
12.2-3	16
12.2-4	16
12.2-5	12
12.2-6	16
12.2-7	16
12.2-8	16
12.2-9	0
12.2-10	16
12.2-11	4
12.2-12	0
12.2-13	0
12.2-14	16
12.2-15	0
12.2-16	0

CHAPTER 12 (Continued)

TEXT PAGES	AMENDMENT
12.2-16a	5
12.2-17	0
12.2-18	0
12.2-19	0
12.2-20	0
12.2-21	0
12.2-22	0
12.2-23	0
12.2-24	0
12.2-25	0
12.2-26	0
12.2-27	0
12.2-28	30
12.2-29	4
12.2-30	0
12.2-31	0
12.2-32	0
12.2-33	0
12.2-34	0
12.2-35	5
12.3-1	27
12.3-2	27
12.3-3	30
12.3-4	30
12.3-5	30
12.3-6	0
12.3-7	0
12.3-8	0
12.3-9	0
12.3-10	4
12.3-11	20
12.3-11a	20
12.3-12	27
12.3-13	27
12.3-13a	20
12.3-14	27
12.3-15	27
12.3-16	0
12.3-17	4
12.3-18	0
12.3-19	27
12.3-20	31
12.3-20a	31
12.3-21	29
12.3-21a	29

CHAPTER 12 (Continued)

TEXT PAGES	AMENDMENT
12.3-22	14
12.3-22a	14
12.3-22b	29
12.3-23	29
12.3-24	29
12.3-25	29
12.3-26	12
12.3-27	0
12.3-28	25
12.3-29	27
12.3-30	27
12.3-31	14
12.4-1	5
12.4-2	5
12.4-3	12
12.4-4	12
12.4-5	5
12.4-5a	5
12.4-6	5
12.4-7	12
12.4-8	5
12.4-9	5
12.4-10	5
12.4-11	5
12.4-12	5
12.4-13	5
12.4-14	5
12.4-15	5
12.4-16	5
12.4-17	5
12.4-18	5
12.4-19	5
12.4-20	5
12.4-21	5
12.4-22	5
12.4-23	5
12.4-24	5
12.4-25	5
12.5-1	13
12.5-2	4
12.5-3	31
12.5-3a	27
12.5-4	31

CHAPTER 12 (Continued)

TEXT PAGES	AMENDMENT
12.5-5	11
12.5-6	13
12.5-7	30
12.5-8	23
12.5-9	13
12.5-10	31
12.5-11	0
12.5-12	11
12.5-12a	11
12.5-13	0
12.5-14	4
12.5-15	4
12.5-16	27
12.5-17	4
12.5-18	31
12.5-19	0
12.5-20	4
12.5-21	0
12.5-22	27
12.5-23	27
12.5-24	27

FIGURES	
12.3-1	20
12.3-2	20
12.3-3	20
12.3-4	20
12.3-5	20
12.3-6	20
12.3-7	12
12.3-8	12
12.3-9	12
12.3-10	12
12.3-11	27
12.3-12	12
12.3-13	12
12.3-14	12
12.3-15	12
12.3-16	12
12.3-17	12
12.3-18	12
12.3-19	12
12.3-20	0
12.3-21	0

CHAPTER 12 (Continued)

CHAPTER 13

TEXT PAGES	AMENDMENT
13-i	31
13-ii	31
13-iii	31
13-iv	31
13-v	31
13-vi	31
13-vii	31
13.1-1	31
13.1-2	31
13.1-3	31
13.1-4	31
13.1-5	31
13.1-6	31
13.1-7	31
13.1-8	31
13.1-9	31
13.1-10	31
13.1-11	31
13.1-12	31
13.1-13	31
13.1-14	31
13.1-15	31
13.1-16	31
13.1-17	31
13.1-18	31
13.1-19	31
13.1-20	31
13.1-21	31
13.1-22	31
13.1-23	31
13.1A-i	32
13.1A-1	32
13.1A-2	32
13.1A-3	31
13.1A-4	32
13.1A-5	32
13.1A-6	31
13.1A-7	32
13.1A-8	31
13.1A-9	31
13.1A-10	31
13.1A-11	31
13.1A-12	31

CHAPTER 13 (Continued)

TEXT PAGES	AMENDMENT
13.1A-13	31
13.1A-14	31
13.1A-15	31
13.1A-16	31
13.1A-17	31
13.1A-18	31
13.1A-19	31
13.1A-20	31
13.1A-21	31
13.1B-i	31
13.1B-1	31
13.1B-2	31
13.1B-3	31
13.1B-4	31
13.1B-5	31
13.1B-6	31
13.1B-7	31
13.1B-8	31
13.1B-9	31
13.1B-10	31
13.1B-11	31
13.1B-12	31
13.1B-13	31
13.1B-14	31
13.1B-15	31
13.1B-16	31
13.1B-17	31
13.1B-18	31
13.1B-19	31
13.1B-20	31
13.1B-21	31
13.1B-22	31
13.1B-23	31
13.1B-24	31
13.1B-25	31
13.1B-26	31
13.1B-27	31
13.1B-28	31
13.1B-29	31
13.1B-30	31
13.1B-31	31
13.1B-32	31
13.1B-33	31
13.1B-34	31
13.1B-35	31

CHAPTER 13 (Continued)

TEXT PAGES	AMENDMENT
13.1B-36	31
13.1B-37	31
13.1B-38	31
13.1B-39	31
13.1B-40	31
13.1B-41	31
13.1B-42	31
13.1B-43	31
13.1B-44	31
13.1B-45	31
13.1B-46	31
13.1B-47	31
13.1B-48	31
13.1B-49	31
13.1B-50	31
13.1B-51	31
13.1B-52	31
13.1B-53	31
13.1B-54	31
13.1B-55	31
13.1B-56	31
13.1B-57	31
13.1B-58	31
13.1B-59	31
13.1B-60	31
13.1B-61	31
13.1B-62	31
13.1B-63	31
13.1B-64	31
13.1B-65	31
13.1B-66	31
13.1B-67	31
13.1B-68	31
13.1B-69	31
13.1B-70	31
13.1B-71	31
13.1B-72	31
13.1B-73	31
13.1B-74	31
13.1B-75	31
13.1B-76	31
13.1B-77	31
13.1B-78	31
13.1B-79	31

CHAPTER 13 (Continued)

TEXT PAGES	AMENDMENT
13.1B-80	31
13.1B-81	31
13.1B-82	31
13.1B-83	31
13.1B-84	31
13.1B-85	31
13.1B-86	31
13.1B-87	31
13.1B-88	31
13.2-1	31
13.2-2	31
13.2-3	31
13.2-4	31
13.2-5	31
13.2-6	31
13.2-7	31
13.2-8	31
13.2-9	31
13.2-10	31
13.2-11	31
13.2-12	31
13.2-13	31
13.2-14	31
13.2-15	31
13.2-16	31
13.2-17	31
13.2-18	31
13.2-19	31
13.2-20	31
13.2-21	31
13.2-22	31
13.2-23	31
13.2-24	31
13.3-1	31
13.4-1	31
13.4-2	31
13.4-3	31
13.5-1	31
13.5-2	31
13.5-3	31

CHAPTER 13 (Continued)

CHAPTER 14

CHAPTER 14 (Continued)

TEXT PAGES	AMENDMENT
14.2-23	12
14.2-24	0
14.2-25	7
14.2-26	7
14.2-26a	7
14.2-27	7
14.2-28	12
14.2-29	12
14.2-30	12
14.2-31	20
14.2-32	7
14.2-33	26
14.2-34	7
14.2-35	7
14.2-36	0
14.2-37	7
14.2-38	7
14.2-39	29
14.2-40	20
14.2-41	7
14.2-42	12
14.2-43	12
14.2-43a	12
14.2-43b	7
14.2-44	12
14.2-45	12
14.2-46	12
14.2-47	12
14.2-48	12
14.2-49	12
14.2-50	12
14.2-51	12
14.2-52	12
14.2-53	12
14.2-54	12
14.2-54a	20
14.2-55	20
14.2-56	12
14.2-57	13
14.2-58	12
14.2-59	12
14.2-60	12
14.2-61	12
14.2-62	13
14.2-63	31
14.2-63a	31

CHAPTER 14 (Continued)

TEXT PAGES	AMENDMENT
14.2-64	29
14.2-64a	12
14.2-65	12
14.2-65a	12
14.2-66	12
14.2-67	12
14.2-68	20
14.2-69	12
14.2-70	26
14.2-70a	20
14.2-71	7
14.2-72	12
14.2-73	12
14.2-74	20
14.2-75	20
14.2-76	12
14.2-77	12
14.2-78	12
14.2-79	0
14.2-80	12
14.2-81	12
14.2-82	20
14.2-83	12
14.2-84	12
14.2-85	20
14.2-85a	20
14.2-86	12
14.2-87	12
14.2-88	20
14.2-89	20
14.2-90	12
14.2-91	20
14.2-92	20
14.2-93	20
14.2-94	12
14.2-95	20
14.2-96	26
14.2-97	0
14.2-98	12
14.2-99	20
14.2-100	0
14.2-101	29
14.2-102	29
14.2-103	20
14.2-104	20
14.2-105	12

CHAPTER 14 (Continued)

TEXT PAGES	AMENDMENT
14.2-106	0
14.2-107	12
14.2-108	0
14.2-109	20
14.2-109a	20
14.2-110	7
14.2-111	0
14.2-112	12
14.2-112a	26
14.2-113	26
14.2-114	29
14.2-115	29
14.2-116	12
14.2-116a	20
14.2-117	7
14.2-118	26
14.2-119	20
14.2-119a	20
14.2-120	12
14.2-121	7
14.2-122	26
14.2-123	13
14.2-124	20
14.2-124a	20
14.2-125	13
14.2-126	12
14.2-127	0
14.2-128	20
14.2-129	7
14.2-130	12
14.2-131	0
14.2-132	20
14.2-132a	13
14.2-133	7
14.2-134	20
14.2-134a	20
14.2-135	12
14.2-136	13
14.2-137	13
14.2-138	20
14.2-138a	7
14.2-139	26
14.2-140	25
14.2-141	7
14.2-142	12
14.2-142a	12

CHAPTER 14 (Continued)

CHAPTER 15

TEXT PAGES	AMENDMENT
15-i	0
15-ii	0
15-iii	23
15-iv	23
15-v	0
15-vi	23
15-vii	0
15-viii	0
15-ix	0
15-x	0
15-xi	0
15-xii	0
15-xiii	0
15-xiv	0
15-xv	0
15-xvi	0
15-xvii	0
15-xviii	0
15-xix	0
15-xx	0
15-xxi	8
15-xxii	23
15-xxiia	8
15-xxiii	0
15-xxiv	0
15-xxv	0
15-xvi	0
15-xvii	0
15-xviii	0
15-xxix	0
15-xxx	0
15-xxxi	0
15-xxxii	0
15-xxxiii	0
15-xxxiv	0
15-xxxv	0
15-xxxvi	0
15-xxxvii	5
15-xxxviii	0
15-xxxix	0
15-xL	0
15-xLi	0
15-xLii	0
15-xLiii	0
15-xLiv	0
15-xLv	0

CHAPTER 15 (Continued)

TEXT PAGES	AMENDMENT
15-xLvi	0
15-xLvii	0
15-xLviii	0
15-xLix	0
15-L	0
15-Li	0
15-Lii	0
15-Liii	0
15-Liv	23
15-Lv	23
15-Lvi	23
15-Lvii	23
15-Lviii	0
15-Lix	23
15-Lx	23
15-Lxi	0
15-Lxii	0
15.0-1	0
15.0-2	0
15.0-3	0
15.0-4	0
15.0-5	0
15.0-6	0
15.0-7	0
15.0-8	23
15.0-8a	23
15.0-9	0
15.0-10	0
15.0-11	0
15.0-12	0
15.0-13	0
15.0-14	23
15.0-15	23
15.0-16	23
15.0-17	0
15.0-18	23
15.0-19	23
15.0-20	23
15.0-20a	23
15.0-20b	23
15.0-21	0
15.0-22	23
15.0-23	23

CHAPTER 15 (Continued)

TEXT PAGES	AMENDMENT
15.1-1	0
15.1-2	21
15.1-3	0
15.1-4	0
15.1-5	0
15.1-6	0
15.1-7	0
15.1-8	0
15.1-9	0
15.1-10	0
15.1-11	0
15.1-12	23
15.1-12a	27
15.1-13	23
15.1-13a	23
15.1-14	23
15.1-15	23
15.1-16	20
15.1-17	0
15.1-18	0
15.1-19	0
15.1-20	0
15.1-21	0
15.1-22	0
15.1-23	0
15.1-24	0
15.1-25	0
15.1-26	0
15.1-27	21
15.1-28	0
15.1-29	0
15.1-30	0
15.1-31	23
15.2-1	0
15.2-2	0
15.2-3	0
15.2-4	23
15.2-5	0
15.2-6	23
15.2-7	23
15.2-8	23
15.2-9	23
15.2-10	23
15.2-11	23
15.2-12	23

CHAPTER 15 (Continued)

TEXT PAGES	AMENDMENT
15.2-13	23
15.2-14	23
15.2-15	23
15.2-16	0
15.2-17	0
15.2-18	23
15.2-19	23
15.2-20	23
15.2-21	0
15.2-22	0
15.2-23	0
15.2-24	23
15.2-25	23
15.2-26	0
15.2-27	23
15.2-28	0
15.2-29	0
15.2-30	0
15.2-31	0
15.2-32	0
15.2-33	0
15.2-34	0
15.2-35	0
15.2-36	0
15.2-37	0
15.2-38	0
15.2-39	0
15.2-40	21
15.2-41	0
15.2-42	0
15.2-43	0
15.2-44	0
15.2-45	0
15.2-46	0
15.2-47	0
15.2-48	11
15.2-49	11
15.2-50	11
15.2-51	0
15.2-52	11
15.2-53	11
15.2-54	11
15.2-55	11
15.2-56	11
15.2-57	11
15.2-58	0

CHAPTER 15 (Continued)

TEXT PAGES	AMENDMENT
15.2-59	0
15.2-60	0
15.2-61	0
15.2-62	0
15.2-63	0
15.2-64	0
15.2-65	17
15.2-66	21
15.2-67	21
15.2-68	8
15.2-69	17
15.2-70	0
15.2-71	21
15.2-71a	8
15.2-72	8
15.2-73	8
15.2-74	8
15.2-75	8
15.2-76	8
15.2-77	0
15.2-78	0
15.2-79	0
15.2-80a	8
15.2-80b	8
15.2-80c	8
15.2-80d	8
15.2-81	0
15.3-1	0
15.3-2	0
15.3-3	0
15.3-4	0
15.3-5	0
15.3-6	0
15.3-7	0
15.3-8	0
15.3-9	0
15.3-10	0
15.3-11	0
15.3-12	0
15.3-13	0
15.3-14	0
15.3-15	0
15.3-16	0
15.3-16	0

CHAPTER 15 (Continued)

TEXT PAGES	AMENDMENT
15.3-18	0
15.3-19	0
15.3-20	0
15.3-21	0
15.3-22	0
15.3-23	0
15.3-24	0
15.3-25	0
15.3-26	0
15.3-27	0
15.3-28	0
15.3-29	0
15.3-30	0
15.3-31	0
15.3-32	0
15.4-1	0
15.4-2	0
15.4-3	0
15.4-4	0
15.4-5	0
15.4-6	0
15.4-7	0
15.4-8	5
15.4-9	0
15.4-10	0
15.4-11	0
15.4-12	0
15.4-13	0
15.4-14	0
15.4-15	0
15.4-16	0
15.4-17	0
15.4-18	0
15.4-19	20
15.4-20	0
15.4-21	0
15.4-22	0
15.4-23	0
15.4-24	0
15.4-25	0
15.4-26	0
15.4-27	0
15.4-28	0
15.4-29	0
15.4-30	0

CHAPTER 15 (Continued)

CHAPTER 15 (Continued)

TEXT PAGES	AMENDMENT
15.6-10	0
15.6-11	0
15.6-12	0
15.6-13	0
15.6-14	0
15.6-15	0
15.6-16	0
15.6-17	0
15.6-18	0
15.6-19	0
15.6-20	0
15.6-21	0
15.6-22	0
15.6-23	0
15.6-24	0
15.6-25	0
15.6-26	0
15.6-27	0
15.6-28	0
15.6-29	0
15.6-30	20
15.6-31	0
15.6-32	0
15.6-33	14
15.6-34	30
15.6-34a	11
15.6-35	0
15.6-36	0
15.6-37	17
15.6-38	0
15.6-39	10
15.6-40	9
15.6-41	10
15.6-42	9
15.6-43	0
15.6-44	10
15.6-45	10
15.6-46	10
15.6-47	10
15.6-48	9
15.6-49	0
15.6-50	0
15.6-51	0
15.6-52	0
15.6-53	0
15.6-54	0

CHAPTER 15 (Continued)

TEXT PAGES	AMENDMENT
15.6-55	0
15.6-56	0
15.6-57	0
15.6-58	0
15.6-59	0
15.6-60	0
15.7-1	0
15.7-2	0
15.7-3	0
15.7-4	0
15.7-5	0
15.7-6	0
15.7-7	0
15.7-8	0
15.7-9	0
15.7-10	0
15.7-11	0
15.7-12	0
15.7-13	0
15.7-14	0
15.7-15	0
15.7-16	0
15.7-17	0
15.7-18	0
15.7-19	0
15.7-20	0
15.7-21	0
15.7-22	0
15.7-23	0
15.7-24	0
15.7-25	0
15.7-26	0
15.7-27	0
15.7-28	0
15.7-29	0
15.7-30	0
15.7-31	23
15.7-32	23
15.7-33	23
15.7-34	23
15.7-35	23
15.7-36	23
15.7-37	23
15.7-38	23
15.7-39	0

CHAPTER 15 (Continued)

TEXT PAGES	AMENDMENT
15.7-40	0
15.7-41	0
15.7-42	0
15.7-43	0
15.7-44	0
15.7-45	0
15.7-46	0
15.7-47	0
15.7-48	0
15.7-49	0
15.7-50	0
15.7-51	0
15.7-52	0
15.7-53	0
15.7-54	0
15.7-55	0
15.7-56	0
15.7-57	0
15.7-58	0
15.7-59	0
15.7-60	0
15.7-61	0
15.7-62	23
15.7-63	23
15.7-64	23
15.7-65	23
15.7-66	23
15.7-67	23
15.7-68	23
15.7-69	23
15.7-70	23
15.7-71	23
15.8-1	0
15.8-2	0
15.8-3	12
15.A-i	0
15.A-ii	0
15.A-iii	0
15.A-iv	0
15.A-v	0
15.A-vi	0
15.A-vii	0
15.A-viii	0
15.A-ix	0

CHAPTER 15 (Continued)

TEXT PAGES	AMENDMENT
15.A-x	0
15.A-xi	0
15.A.1-1	0
15.A.1-2	0
15.A.1-3	0
15.A.2-1	0
15.A.2-2	0
15.A.2-3	0
15.A.2-4	0
15.A.2-5	0
15.A.2-6	0
15.A.2-7	0
15.A.2-8	0
15.A.2-9	0
15.A.2-10	0
15.A.2-11	0
15.A.2-12	0
15.A.2-13	0
15.A.2-14	0
15.A.2-15	0
15.A.2-16	0
15.A.2-17	0
15.A.2-18	0
15.A.2-19	0
15.A.2-20	0
15.A.3-1	0
15.A.3-2	0
15.A.3-3	0
15.A.3-4	0
15.A.3-5	0
15.A.3-6	0
15.A.3-7	0
15.A.3-8	0
15.A.3-9	0
15.A.3-10	0
15.A.3-11	0
15.A.3-12	0
15.A.3-13	0
15.A.3-14	0
15.A.4-1	0
15.A.4-2	0
15.A.4-3	0
15.A.5-1	0
15.A.5-2	0
15.A.6-1	0
15.A.6-2	0

CHAPTER 15 (Continued)

TEXT PAGES	AMENDMENT
15.A.6-3	0
15.A.6-4	0
15.A.6-5	0
15.A.6-6	0
15.A.6-7	0
15.A.6-8	0
15.A.6-9	0
15.A.6-10	0
15.A.6-11	0
15.A.6-12	0
15.A.6-13	0
15.A.6-14	0
15.A.6-15	0
15.A.6-16	0
15.A.6-17	0
15.A.6-18	0
15.A.6-19	0
15.A.6-20	0
15.A.6-21	0
15.A.6-22	0
15.A.6-23	0
15.A.6-24	0
15.A.6-25	0
15.A.6-26	0
15.A.6-27	0
15.A.6-28	0
15.A.6-29	0
15.A.6-30	0
15.A.6-31	0
15.A.6-32	0
15.A.6-33	0
15.A.6-34	0
15.A.6-35	0
15.A.6-36	0
15.A.6-37	0
15.A.6-38	0
15.A.6-39	0
15.A.6-40	0
15.A.6-41	0
15.A.6-42	0
15.A.7-1	0
15.A.8-1	0
15.A.9-1	0

CHAPTER 15 (Continued)

CHAPTER 15 (Continued)

FIGURES	AMENDMENT
15.A.6-40	11
15.A.6-41	0
15.A.6-42	11
15.A.6-43	11
15.A.6-44	11
15.A.6-45	11
15.A.6-46	0
15.A.6-47	0
15.A.6-48	0
15.A.6-49	0
15.A.6-50	0
15.A.6-51	11
15.A.6-52	11
15.A.6-53	11
15.A.6-54	0
15.A.6-55	0
15.A.6-56	0
15.A.6-57	0
15.A.6-58	0
15.A.6-59	0
15.A.6-60	0
15.A.6-61	0
15.A.6-62	0
15.A.6-63	0
15.A.6-64	0
15.A.6-65	0

CHAPTER 16

TEXT PAGES	AMENDMENT
16.0-1	0

CHAPTER 17

TEXT PAGES	AMENDMENT
17-i	30
17-ii	17
17-iii	17
17-iv	17
17-v	17
17.1-1	17
17.1-2	27
17.1-2a	27
17.1-2b	27
17.1-3	27
17.1-3a	27
17.1-3b	27
17.1-3c	27
17.1-4	29
17.1-4a	29
17.1-4b	29
17.1-5	27
17.1-6	27
17.1-7	20
17.1-7a	30
17.1-7b	20
17.1-7c	27
17.1-7d	20
17.1-7e	20
17.1-8	30
17.1-8a	20
17.1-9	27
17.1-9a	27
17.1-10	17
17.1-11	17
17.1-12	30
17.1-13	17
17.1-14	17
17.1-15	30
17.1-16	17
17.1-17	27
17.1-17a	30
17.1-18	17
17.1-19	20
17.1-19a	20
17.1-20	30
17.1-21	17
17.1-22	26
17.1-23	20
17.1-24	17

CHAPTER 17 (Continued)

APPENDIX A

TEXT PAGES	AMENDMENT
A-1	0
A-2	0
A-3	0
A-4	0
A-5	0
A-6	0
A-7	0
A-8	0
A-9	0
A-10	0
A-11	0
A-12	0
A-13	0
A-14	0
A-15	0
A-16	0
A-17	0
A-18	0
A-19	0
A-20	20
A-21	0
A-22	0

APPENDIX B

TEXT PAGES	
Title Page	17
B-i	17
B-ii	17
B-iii	17
B-iv	29
B.1-1	17
B.1-2	17
B.1-3	23
B.1-3a	23
B.1-3b	23
B.1-3c	23
B.1-4	17
B.1-5	32
B.1-5a	32
B.1-6	17
B.1-7	17
B.1-8	17
B.1-9	17

APPENDIX B

TEXT PAGES	AMENDMENT
B.1-10	32
B.1-11	27
B.1-12	27
B.1-13	17
B.1-14	23
B.1-14a	23
B.1-15	23
B.1-16	23
B.1-17	23
B.1-18	23
B.1-19	17
B.1-20	30
B.1-21	30
B.1-22	30
B.1-23	17
B.1-24	30
B.1-25	17
B.1-26	23
B.1-27	17
B.1-28	17
B.1-29	23
B.1-30	17
B.1-31	30
B.1-32	17
B.1-33	30
B.1-33	30
B.1-34	17
B.1-35	23
B.1-35a	30
B.1-35b	30
B.1-36	17
B.1-37	23
B.1-37a	23
B.1-38	23
B.1-39	23
B.1-40	17
B.1-41	23
B.1-42	17
B.1-43	23
B.1-44	23
B.1-45	17
B.1-46	17
B.1-47	17
B.1-48	31

TEXT PAGES	AMENDMENT
B.2-1	17
B.2-2	17
B.2-3	17
B.2-4	17
B.2-5	21
B.2-5a	21
B.2-5b	21
B.2-5c	21
B.2-5d	21
B.2-5e	21
B.2-6	17
B.2-7	17
B.2-8	17
B.2-9	17
B.2-10	17
B.2-11	30
B.2-12	23
B.2-13	17
B.2-14	17
B.2-15	17
B.2-16	23
B.2-16a	23
B.2-16b	23
B.2-16c	23
B.2-16d	23
B.2-17	31
B.2-17a	23
B.2-18	23
B.2-18a	23
B.2-19	23
B.2-20	23
B.2-21	17
B.2-22	17
B.2-23	17
B.2-24	17
B.2-25	17
B.2-26	17
B.2-27	17
B.2-28	25
B.2-29	17
B.2-30	17
B.2-31	17
B.2-32	17
B.2-33	17
B.2-34	17
B.2-35	17
B.2-36	17
B.2-37	29

TEXT PAGES	AMENDMENT
B.2-37a	29
B.2-38	17
B.2-39	17
B.2-40	17
B.2-41	29
B.2-41a	29
B.2-42	17
B.2-43	17
B.2-44	23
B.2-45	27
B.2-46	17
B.2-47	23
B.2-48	23
B.2-49	23
B.2-50	23
B.2-51	23
B.2-52	17
B.2-53	17
B.2-54	17
B.2-55	21
B.2-56	21
B.2-57	21
B.2-57a	21
B.2-57b	21
B.2-57c	21
B.2-58	23
B.2-59	23
B.2-60	23
B.2-61	23
B.2-62	23
B.2-62a	23
B.2-62b	23
B.2-62c	23
B.2-63	32
B.2-63a	32
B.2-64	17
B.2-65	23
B.2-66	23
B.2-67	21
B.2-68	21
B.2-69	17
B.2-70	23
B.2-71	23
B.2-72	17
B.2-73	23
B.2-74	23
B.2-75	23

TEXT PAGES	AMENDMENT
B.3-4i	31
B.3-4j	31
B.3-4k	31
B.3-4l	31
B.3-4m	31
B.3-4n	23
B.3-4o	31
B.3-4p	31
B.3-5	23
B.3-5a	23
B.3-6	23
B.3-6a	23
B.3-7	23
B.3-8	23
B.3-9	27
B.3-9a	31
B.3-10	17
B.3-11	17
B.3-12	17

APPENDIX C

TEXT PAGES	
C-i	0
C-ii	0
C-iii	14
C-iv	0
C-v	14
C-vi	0
C-vii	0
C.1-1	0
C.2-1	0
C.2-2	0
C.2-3	4
C.2-4	0
C.2-5	0
C.2-6	0
C.2-7	0
C.2-8	0
C.2-9	0
C.2-10	0
C.2-11	0
C.2-12	0
C.2-13	0

TEXT PAGES	AMENDMENT
C.2-14	30
C.2-15	0
C.2-16	4
C.2-17	0
C.2-18	13
C.2-19	13
C.2-20	13
C.2-21	0
C.2-22	0
C.2-23	0
C.2-24	0
C.2-25	0
C.2-26	0
C.2-27	0
C.2-28	0
C.2-29	0
C.2-30	0
C.2-31	0
C.2-32	0
C.2-33	0
C.2-34	0
C.2-35	0
C.2-36	0
C.2-37	0
C.2-38	0
C.2-39	0
C.2-40	27
C.2-41	27
C.2-42	0
C.2-43	30
C.2-44	21
C.2-45	21
C.2-46	13
C.2-47	0
C.2-48	30
C.2-49	0
C.2-50	0
C.2-51	0
C.2-52	30
C.2-53	0
C.2-54	0
C.2-55	13
C.2-56	0
C.2-57	0
C.2-58	0
C.2-59	0
C.2-60	0

TEXT PAGES	AMENDMENT
C.2-61	0
C.2-62	0
C.2-63	13
C.2-64	0
C.2-65	30
C.2-66	32
C.2-67	0
C.2-68	9
C.2-69	0
C.2-70	7
C.2-71	0
C.2-72	7
C.2-73	0
C.2-74	0
C.2-75	0
C.2-76	0
C.2-77	0
C.2-78	0
C.2-79	0
C.2-80	0
C.2-81	0
C.2-82	23
C.2-83	30
C.2-83a	30
C.2-84	30
C.2-85	0
C.2-86	14
C.2-87	0
C.2-88	0
C.2-89	0
C.2-90	32
C.3-1	0
C.3-2	0
C.3-3	0
C.3-4	0
C.3-5	5
C.3-6	0
C.3-7	0
C.3-8	0
C.3-9	13
C.3-10	0
C.3-11	30
C.3-12	0
C.3-13	30
C.3-14	12
C.3-15	0

TEXT PAGES	AMENDMENT
C.3-16	0
C.3-17	2
C.3-17a	2
C.3-18	13
C.3-19	0
C.3-20	0
C.3-21	31
C.3-22	5
C.3-23	29
C.3-24	30
C.3-25	8
C.3-26	27
C.3-27	0
C.3-28	23
C.3-29	0
C.3-30	0
C.3-31	31
C.3-32	8
C.3-33	8
C.3-34	8
C.3-35	11
C.3-36	0
C.3-37	0
C.3-38	0
C.3-39	27
C.3-40	0
C.3-41	21
C.3-41a	21
C.3-42	0
C.3-43	0
C.3-44	0
C.3-45	5
C.3-46	0
C.3-47	8
C.3-48	0
C.3-49	13
C.3-49a	9
C.3-50	0
C.3-51	13
C.3-52	0
C.3-53	0
C.3-54	8
C.3-55	13
C.3-56	27
C.3-57	26
C.3-58	0
C.3-59	7

TEXT PAGES	AMENDMENT
C.3-60	14
C.3-61	14
C.3-62	0
C.3-63	0
C.3-64	27
C.3-65	0
C.3-66	0
C.3-67	8
C.3-68	23
C.3-69	0
C.3-70	9
C.3-71	0
C.3-72	0
C.3-73	27
C.3-74	27
C.3-75	20
C.3-76	11
C.3-77	32
C.3-78	0
C.3-79	0
C.3-80	0
C.3-81	4
C.3-82	23
C.3-83	0
C.3-84	13
C.3-85	27
C.3-86	14
C.3-87	0
C.3-88	0
C.3-89	0
C.3-90	0
C.3-91	13
C.3-92	9
C.3-93	0
C.3-94	0
C.3-95	0
C.3-96	0
C.3-97	0
C.3-98	0
C.3-99	0
C.3-100	0
C.3-101	0
C.3-102	8
C.3-103	0
C.3-104	0
C.3-105	0
C.3-106	0

TEXT PAGES	AMENDMENT
C.3-107	8
C.3-108	30
C.3-109	0
C.3-110	0
C.3-111	27
C.3-112	14
C.3-113	27
C.3-114	27
C.3-115	23

APPENDIX D

TEXT PAGES	AMENDMENT
Title Page	27

APPENDIX E

TEXT PAGES	
Title Page	10
E.1-1	7
E.2-1	14
E.3-1	8
E.3-2	0
E.3-3	12
E.3-4	12
E.3-5	12
E.3-5a	3
E.3-6	0
E.3-7	0
E.3-8	0
E.3-9	3
E.3-10	5
E.7-1	13
E.7-2	13
E.7-3	13
E.7-4	13
E.7-5	13
E.8-1	0
E.8-2	0
E.9-1	8
E.9-2	5
E.9-3	8
E.9-3a	8
E.10-1	5
E.11-1	20
E.12-1	12
E.12-2	16
E.12-3	16

APPENDIX F

TEXT PAGES	
Title Page	7
F-i	19
F-ii	19
F-iii	31
F-iiia	24

TEXT PAGES	AMENDMENT
F-iv	24
F-v	31
F-vi	19
F-vii	19
F-viii	19
F-ix	19
F-x	19
F-xa	24
F-xi	24
F-xia	31
F-xii	31
F-xiii	19
F-xiv	19
F-xv	19
F-xvi	19
F-xvii	19
F-xviii	19
F-xix	19
F.1-1	19
F.1-2	19
F.1-3	19
F.1-4	19
F.1-5	19
F.1-6	19
F.1-7	19
F.1-8	19
F.1-9	19
F.1-10	19
F.1-11	19
F.1-12	19
F.1-13	19
F.2-1	19
F.2-2	19
F.2-3	19
F.2-4	19
F.2-5	19
F.2-6	19
F.2-7	24
F.2-8	24
F.2-8a	19
F.2-9	19
F.2-10	24
F.2-11	31
F.2-11a	31
F.2-12	24
F.2-13	31

TEXT PAGES	AMENDMENT
F.2-14	24
F.2-15	19
F.2-16	31
F.2-17	31
F.2-18	19
F.2-19	19
F.2-20	24
F.2-21	24
F.2-22	19
F.2-23	19
F.2-24	24
F.2-25	24
F.2-26	19
F.2-27	24
F.2-28	24
F.2-29	19
F.2-30	24
F.2-31	31
F.2-32	19
F.2-33	24
F.2-33a	31
F.2-34	31
F.2-35	19
F.2-36	24
F.2-37	24
F.2-38	19
F.2-39	24
F.2-39a	31
F.2-40	24
F.2-41	19
F.2-42	24
F.2-43	24
F.2-44	19
F.2-44a	24
F.2-45	24
F.2-46	19
F.2-47	24
F.2-47a	24
F.2-48	24
F.2-49	24
F.2-50	24
F.2-51	19
F.2-52	24
F.2-53	24
F.2-54	24
F.2-55	24
F.2-56	19
F.2-57	19

TEXT PAGES	AMENDMENT
F.2-58	24
F.2-58a	24
F.2-59	24
F.2-60	19
F.2-61	24
F.2-62	19
F.2-63	24
F.2-64	24
F.2-65	24
F.2-66	24
F.2-67	19
F.2-68	24
F.2-69	24
F.2-70	19
F.2-71	24
F.2-72	19
F.2-73	19
F.2-74	24
F.2-75	31
F.2-76	31
F.2-76a	24
F.2-77	24
F.2-77a	24
F.2-78	31
F.2-79	24
F.2-80	19
F.2-81	19
F.2-82	24
F.2-83	24
F.2-84	24
F.2-85	31
F.2-85a	31
F.2-86	24
F.2-86a	24
F.2-87	24
F.2-88	24
F.2-89	24
F.2-89a	24
F.2-90	24
F.2-91	24
F.2-91a	31
F.2-92	24
F.2-93	24
F.2-94	24
F.2-95	24
F.2-96	24
F.2-97	19

TEXT PAGES	AMENDMENT
F.2-98	24
F.2-99	31
F.2-99a	31
F.2-100	24
F.2-101	24
F.2-102	24
F.2-103	24
F.2-104	24
F.2-105	24
F.2-105a	24
F.2-106	24
F.2-107	24
F.2-108	24
F.2-109	31
F.2-110	24
F.2-111	24
F.2-111a	24
F.2-112	19
F.2-113	24
F.2-114	31
F.2-114a	31
F.2-115	31
F.2-115a	24
F.2-116	19
F.2-117	24
F.2-118	24
F.2-119	19
F.2-120	31
F.2-121	24
F.2-121a	24
F.2-122	24
F.2-123	24
F.2-124	24
F.2-124a	24
F.2-125	19
F.2-126	24
F.2-127	24
F.2-127a	24
F.2-128	24
F.2-129	24
F.2-129a	24
F.2-129b	24
F.2-129c	24
F.2-130	31
F.2-131	24
F.2-132	31
F.2-132a	31

TEXT PAGES	AMENDMENT
F.2-133	19
F.2-134	19
F.2-135	31
F.2-136	24
F.2-137	24
F.2-138	19
F.2-139	24
F.2-140	24
F.2-140a	24
F.2-141	24
F.2-141a	24
F.2-142	19
F.2-143	19
F.2-144	19
F.2-145	24
F.2-146	19
F.2-146a	31
F.2-146b	31
F.2-146c	31
F.2-147	19
F.2-148	19
F.2-149	19
F.2-150	19
F.2-151	19
F.2-152	19
F.2-153	24
F.2-154	19
F.2-155	19
F.2-156	19
F.2-157	19
F.2-158	19
F.2-159	19
F.2-160	19
F.2-161	19
F.2-162	24
F.2-163	24
F.2-164	19
F.2-165	24
F.2-166	24
F.2-167	19
F.2-168	19
F.2-169	19
F.2-170	24
F.2-171	19
F.2-172	19
F.2-173	19
F.2-174	19
F.2-175	19

TEXT PAGES	AMENDMENT
F.2-176	24
F.2-177	19
F.2-178	19
F.2-179	24
F.2-180	19
F.2-180a	19
F.2-181	19
F.2-182	19
F.2-183	19
F.2-184	24
F.2-184a	24
F.2-185	19
F.2-186	19
F.2-187	19
F.2-188	19
F.2-189	24
F.2-190	24
F.2-191	24
F.2-192	24
F.2-193	19
F.2-194	19
F.2-195	19
F.2-196	19
F.2-197	24
F.2-198	19
F.2-198a	24
F.2-198b	24
F.2-198c	24
F.2-199	24
F.2-200	19
F.2-201	31
F.2-202	31
F.2-203	24
F.2-204	24
F.2-205	24
F.2-206	24
F.2-206a	24
F.2-207	19
F.2-208	24
F.2-209	19
F.2-210	24
F.2-211	24
F.2-212	24
F.2-212a	24
F.2-213	19
F.2-214	24
F.2-215	19

TEXT PAGES	AMENDMENT
F.2-216	24
F.2-217	24
F.2-218	24
F.2-219	24
F.2-220	24
F.2-221	31
F.2-222	24
F.2-222a	24
F.2-223	24
F.2-223a	24
F.2-224	24
F.2-224a	24
F.2-225	24
F.2-226	19
F.2-227	24
F.2-228	24
F.2-228a	24
F.2-229	19
F.2-230	24
F.2-231	19
F.2-232	24
F.2-233	19
F.2-234	24
F.2-234a	24
F.2-235	19
F.2-236	24
F.2-236a	24
F.2-237	19
F.2-238	24
F.2-239	24
F.2-239a	24
F.2-240	19
F.2-241	24
F.2-242	24
F.2-243	24
F.2-243a	24
F.2-244	31
F.2-245	31
F.2-246	31
F.2-247	31
F.2-248	31
F.2-248a	19
F.2-249	31
F.2-249a	31
F.2-250	24
F.2-250a	31
F.2-251	24

TEXT PAGES	AMENDMENT
F.2-251a	24
F.2-252	19
F.2-252a	24
F.2-253	31
F.2-253a	19
F.2-254	31
F.2-254a	19
F.2-255	19
F.2-255a	19
F.2-256	19
F.2-256a	19
F.2-257	24
F.2-257a	24
F.2-258	24
F.2-259	31
F.2-260	24
F.2-261	19
F.2-262	19
F.2-263	19
F.2-264	24
F.2-264a	31
F.2-265	24
F.2-265a	31
F.2-266	19
F.2-267	31
F.2-267a	31
F.2-267b	31
F.2-268	31
F.2-268a	31
F.2-269	31
F.2-269a	31
F.2-270	24
F.2-270a	31
F.2-271	24
F.2-271a	31
F.2-272	19
F.2-272a	24
F.2-273	31
F.2-273a	31
F.2-274	31
F.2-274a	31
F.2-274b	31
F.2-275	31
F.2-275a	31
F.2-275b	31
F.2-275c	31
F.2-276	31
F.2-276a	31

TEXT PAGES	AMENDMENT
F.2-277	31
F.2-277a	31
F.2-278	31
F.2-278a	31
F.2-279	31
F.2-279a	31
F.2-280	31
F.2-281	24
F.2-281a	31
F.2-281b	31
F.2-282	24
F.2-283	31
F.2-283a	24
F.2-283b	24
F.2-283c	24
F.2-284	31
F.2-285	31
F.2-286	31
F.2-287	31
F.2-288	24
F.2-288a	19
F.2-289	31
F.2-290	24
F.2-290a	24
F.2-291	31
F.2-291a	31
F.2-291b	24
F.2-292	24
F.2-293	24
F.2-293a	31
F.2-294	24
F.2-294a	24
F.2-295	31
F.2-295a	24
F.2-296	19
F.2-297	19
F.2-298	19
F.2-299	19
F.2-300	19
F.2-301	24
F.2-302	24
F.2-302a	24
F.2-303	24
F.2-303a	24
F.2-304	24
F.2-304a	24

TEXT PAGES	AMENDMENT
F.2-305	31
F.2-305a	31
F.3-1	19
F.3-2	19
F.3-3	19
F.3-4	19
F.3-5	19
F.3-6	19
F.3-7	24
F.3-8	19
F.3-9	19
F.3-10	19
F.3-11	19
F.3-12	19
F.3-13	19
F.3-14	19
F.3-15	24
F.3-15a	24
F.3-16	32
F.3-17	19
F.3-18	19
F.3-19	19
F.3-20	19
F.3-21	19
F.3-22	19
F.3-23	19
F.3-24	19
F.3-25	19
F.3-26	19
F.3-27	19
F.3-28	19
F.3-29	29
F.3-30	29
F.3-31	29
F.3-32	29
F.3-33	29
F.3-34	19
F.3-35	19
F.3-36	19
F.3-37	29
F.3-38	19
F.3-39	19
F.3-40	19
F.3-41	19
F.3-42	19
F.3-43	19

TEXT PAGES	AMENDMENT
F.3-44	19
F.3-45	19
F.3-46	19
F.3-47	24
F.3-47a	19
F.3-48	19
F.3-49	24
F.3-49a	24
F.3-50	19
F.3-51	19
F.3-52	24
F.3-53	19
F.3-54	19
F.3-55	19
F.3-56	29
F.3-57	29
F.3-58	19
F.3-59	19
F.3-60	19
F.3-61	19
F.3-62	19
F.3-63	19
F.3-64	24
F.3-65	24
F.3-66	19
F.3-67	19
F.3-68	19
F.3-69	19
F.3-70	19
F.3-71	19
F.3-72	19
F.3-73	24
F.3-74	19
F.3-75	19
F.3-76	19
F.3-77	19
F.3-78	19
F.3-79	19
F.3-80	24
F.3-81	19
F.3-82	19
F.3-83	19
F.3-84	24
F.3-85	19
F.3-86	19
F.3-87	19
F.3-88	24

TEXT PAGES	AMENDMENT
F.3-89	19
F.3-90	24
F.3-91	19
F.3-92	24
F.3-93	24
F.3-93a	19
F.3-94	24
F.3-95	19
F.3-96	19
F.3-97	24
F.3-98	24
F.3-99	24
F.3-100	19
F.3-101	24
F.3-102	24
F.3-103	19
F.3-104	19
F.3-105	24
F.3-106	24
F.3-107	19
F.3-108	19
F.3-109	24
F.3-110	24
F.3-111	19
F.3-112	24
F.3-113	19
F.3-114	19
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F.3-120	19
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F.3-123	19
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F.3-125	19
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F.3-127	24
F.3-128	19
F.3-129	19
F.3-130	19
F.3-131	19
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F.3-134	19

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H.0-viia	12
H.0-viii	4
H.0-ix	4
H.0-x	4
H.0-xi	4
H.0-xii	4
H.0-xiii	4
H.1.1-1	4
H.1.2-1	12
H.1.2-2	4
H.1.2-3	4
H.1.2-4	21
H.1.2-4a	21

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H.2.1-2	4
H.2.1-3	4
H.2.1-4	30
H.2.1-5	4
H.2.1-6	4
H.2.1-7	4
H.2.1-8	4
H.2.1-9	4
H.2.1-10	4
H.2.2-1	4
H.2.2-2	4
H.2.2-3	29
H.2.2-4	4
H.2.3-1	4
H.2.3-2	4
H.2.3-3	4
H.2.3-4	4
H.2.3-5	12
H.2.3-5a	12
H.2.3-6	4
H.2.3-7	4
H.2.3-8	4
H.2.4-1	4
H.2.4-2	4
H.2.4-3	4
H.2.5-1	4
H.2.5-2	4
H.2.5-3	4
H.2.5-4	4
H.2.5-5	4
H.2.5-6	4
H.2.5-7	4
H.2.5-8	4
H.2.5-9	4
H.3.0-1	4
H.3.1-1	4
H.3.1-2	4
H.3.1-3	4
H.3.2-1	4
H.3.2-2	4
H.3.2-3	4
H.3.2-4	4
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TEXT PAGES	AMENDMENT
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H.3.3-6	4
H.3.3-7	4
H.3.3-8	4
H.3.3-9	4
H.3.3-10	4
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H.3.3-13	4
H.3.3-14	4
H.3.3-15	4
H.4.0-1	4
H.4.1-1	4
H.4.1-2	4
H.4.1-3	4
H.4.1-4	4
H.4.1-5	4
H.4.1-6	4
H.4.1-7	4
H.4.1-8	4
H.4.2-1	29
H.4.3-1	12
H.4.3-1a	12
H.4.3-2	4
H.5.1-1	4
H.5.1-2	4
H.5.1-3	4
H.5.1-4	4
H.5.1-5	4
H.5.1-6	4
H.5.1-7	4
H.5.1-8	4
H.5.1-9	4
H.5.2-1	4
H.5.2-2	4
H.5.2-3	4
H.5.2-4	4
H.5.2-5	4
H.5.2-6	4
H.5.2-7	4
H.5.2-8	4
H.5.2-9	4
H.5.2-10	4
H.5.2-11	4
H.5.2-12	4

TEXT PAGES	AMENDMENT
H.6.1-1	4
H.6.1-2	4
H.6.1-3	4
H.6.1-4	4
H.6.2-1	4
H.6.2-2	4
H.6.2-3	4
H.6.2-4	4
H.6.3-1	4
H.6.3-2	4
H.6.3-3	4
H.6.3-4	4
H.6.4-1	4
H.6.4-2	4
H.6.4-3	4
H.6.4-4	4
H.6.5-1	4
H.6.5-2	4
H.6.5-3	4
H.6.5-4	4
H.6.6-1	4
H.6.6-2	4
H.6.6-3	4
H.6.6-4	4
H.6.6-5	4
H.7.0-1	4
H.A.0-1	10
H.B.0-i	4
H.B.0-ii	4
H.B.1-1	4
H.B.1-2	4
H.B.1-3	4
H.B.2-1	4
H.B.2-2	4
H.B.2-3	4
H.B.2-4	4
H.B.2-5	4
H.B.2-6	4
H.B.2-7	4
H.B.2-8	4
H.B.2-9	4
H.B.2-10	4
H.B.2-11	4
H.B.2-12	4
H.B.2-13	4

TEXT PAGES	AMENDMENT
H.B.2-14	4
H.B.2-15	4
H.B.2-16	4
H.B.2-17	4
H.B.2-18	4
H.B.2-19	4
H.B.2-20	4
H.B.2-21	4
H.B.2-22	4
H.B.2-23	4
H.B.2-24	4
H.B.2-25	4
H.B.3-1	4
H.B.3-2	4
H.B.3-3	4
H.B.4-1	4
H.B.4-2	4
H.B.4-3	4
H.B.4-4	4
H.B.4-5	4
H.B.5-1	4
H.B.5-2	4
H.B.5-3	4
H.B.5-4	4
H.B.5-5	4
H.B.5-6	4
H.B.5-7	4
H.B.5-8	4
H.B.6-1	4
H.B.6-2	4
H.B.6-3	4
H.B.6-4	4
H.B.6-5	4
H.B.6-6	4
H.B.6-7	4
H.B.6-8	4
H.B.6-9	4
H.B.6-10	4
H.B.6-11	4
H.B.6-12	4
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APPENDIX I

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I-iii	23
I-iv	23
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I.1-1	23
I.1-2	23
I.1-3	23
I.2-1	23
I.2-2	23
I.2-3	23
I.2-4	23
I.2-5	31
I.2-6	23
I.2-7	23
I.3-1	23
I.3-2	23
I.3-3	23
I.3-4	23
I.3-5	23
I.3-6	23
I.3-7	23
I.3-8	32
I.3-9	23
I.3-10	29
I.3-11	29
I.4-1	32
I.4-2	23
I.4-3	23
I.4-4	23
I.4-5	23
I.4-6	23
I.4-7	23
I.4-8	23
I.4-9	23
I.5-1	23
I.5-2	23
I.5-3	23
I.5-4	23
I.5-5	23

TEXT PAGES	AMENDMENT
I.5-6	32
I.5-7	32
I.6-1	23
I.6-2	23
I.6-3	23
I.6-4	23
I.6-5	23
I.6-6	23
I.6-7	23
I.6-8	23
I.6-9	23
I.6-10	23
I.6-11	30
I.6-12	30
I.6-13	23
I.6-14	23
I.6-15	23
I.6-16	23
I.6-17	23
I.6-18	23
I.6-19	23
I.6-20	23
I.6-21	23
I.6-22	23
I.6-23	23
I.6-24	23
I.6-25	23
I.6-26	30
I.6-27	23
I.7-1	23
I.7-2	23
I.7-3	31
I.7-4	31
I.7-5	23
I.7-6	23
I.7-7	31
I.7-8	31
I.7-9	29
I.8-1	30
I.8-2	23
I.8-3	23
I.8-4	23
I.8-5	31

TEXT PAGES	AMENDMENT
I.8-6	23
I.8-7	29
I.8-8	23
I.8-9	23
I.8-10	23
I.8-11	23
I.8-12	23
I.8-13	23
I.8-14	32
I.8-15	32
I.8-16	23
I.8-17	23
I.8-18	31
I.8-19	23
I.8-20	23
I.8-21	23
I.8-22	23
I.8-23	23
I.8-24	23
I.8-25	23
I.8-26	23
I.8-27	23
I.8-28	23
I.8-29	23
I.8-30	23
I.8-31	23
I.8-32	23
I.8-33	27
I.8-34	27
I.8-35	23
I.8-36	23
I.8-37	32
I.8-38	23
I.8-39	23
I.8-40	23
I.8-41	23

APPENDIX J

TEXT PAGES	
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LIST OF NRC QUESTIONS

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
005.001	2	27
010.001	1	1
010.002	1	25
010.003	1	1
010.004	1	1
010.005	1	1
010.006	1	13
010.007	1	1
010.008	1	1
010.009	1	1
010.010	1	21
010.011	9	9
010.012	2	5
010.013	2	5,25
010.014	3	5
010.015	1	13
010.016	3	5,13
010.017	1	13
010.018	1	13
010.019	1	5

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
010.020	1	5
010.021	1	21
010.022	1	5
010.023	2	5
010.024	1	5
010.025	1	5
010.026	1	5
010.027	1	5
010.028	2	5,21
010.029	2	5
010.030	1	5
010.031	1	5
010.032	1	5
010.033	1	5
010.034	1	21
010.035	1	21
010.036	1	30
010.037	3	27
010.038	1	27
010.039	1	26
010.040	2	21

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
010.041	8	21,31
010.042	2	21
010.043	2	23
010.044	1	21
010.045	1	21
010.046	1	21
010.047	1	21
010.048	1	21
010.049	7	21,30
010.050	1	21
010.051	1	21
010.052	1	21
010.053	1	21
010.054	1	23
010.055	5	26
010.056	2	21
010.057	1	21
010.058	1	21
010.059	1	21
010.060	1	21
010.061	1	21
010.062	1	21
010.063	1	29
010.064	1	21

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
010.065	1	21
010.066	11	27,29
010.067	1	31
010.068	1	29
010.069	1	29
010.070	1	29
010.071	1	29
010.072	1	29
010.073	1	29
010.074	1	29
022.001	1	1
022.002	1	1
022.003	1	1
022.004	1	1
022.005	3	8
022.006	1	13
022.007	1	11
022.008	1	14
022.009	1	1
022.010	1	13
022.011	1	1
022.012	1	3
022.013	1	3

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
022.014	1	13
022.015	1	3
022.016	1	3
022.017	1	3
022.018	1	3
022.019	1	3
022.020	1	3
022.021	1	3
022.022	1	30
022.023	1	3
022.024	1	11
022.025	1	14
022.026	1	3
022.027	1	14
022.028	1	3
022.029	1	31
022.030	1	3
022.031	5	16,31
022.032	4	5
022.033	1	5
022.034	1	5
022.035	2	29
022.036	1	5
022.037	1	5

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
022.038	1	5
022.039	3	30
022.040	1	5
022.041	1	5
022.042	1	5
022.043	2	5,14
022.044	1	5
022.045	1	30
022.046	1	5
022.047	1	5
022.048	13	5,26,27
022.049	4	5
022.050	2	5
022.051	1	32
022.052	1	5
022.053	3	20
022.054	2	20
022.055	2	20
022.056	4	20
022.057	1	20
022.058	3	20

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
022.059	6	20
022.060	1	20
022.061	8	20
022.062	2	20
022.063	3	20
022.064	1	21
022.065	1	21
022.066	1	21
022.067	1	21
022.068	1	21
022.069	1	21
022.070	1	21
022.071	1	21
022.072	1	21
022.073	1	29
022.074	1	21
022.075	1	21
022.076	1	21
022.077	1	21
022.078	1	26
022.079	1	21

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
022.080	1	21
022.081	1	21
022.082	1	31
022.083	1	21
022.084	1	21
022.085	1	21
022.086	1	21
022.087	1	21
022.088	1	21
022.089	1	21
022.090	1	21
022.091	1	30
022.092	1	21
022.093	3	21
022.094	2	21
022.095	1	21
022.096	1	21
022.097	1	21
022.098	1	32
022.099	2	21
022.100	2	21

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
022.101	2	21
022.102	1	21
022.103	1	21
022.104	1	21
022.105	1	21
022.106	1	21
022.107	8	20
022.109	2	20
022.110	1	20
022.111	1	20
022.112	2	20
022.113	1	20
022.114	1	20
031.001(a)	1	0
031.001(b)	1	14
031.001(c)	1	14
031.001(d)	1	14
031.001(e)	1	0
031.001(f)	1	0
031.001(g)	1	14
031.001(h)	1	14

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.001(i)	2	0
031.001(j)	1	13
031.001(k)	1	13
031.001(l)	1	0
031.001(m)	1	32
031.001(n)	1	0
031.001(o)	1	0
031.001(p)	1	0
031.001(q)	1	0
031.001(r)	2	0,14
031.001(s)	1	0
031.001(t)	1	0
031.001(u)	1	14
031.001(v)	1	14
031.001(w)	1	14
031.001(x)	1	0
031.001(y)	1	14
031.001(z)	1	14
031.001(aa)	1	14
031.001(bb,cc)	1	14
031.001(dd)	1	14

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.001(ee)	1	14
031.001(ff)	1	14
031.001(gg)	1	0
031.001(hh)	1	14
031.001(ii)	1	14
031.002	1	0
031.003	1	0
031.004	1	0
031.005	1	14
031.006	1	32
031.007	1	0
031.008	1	0
031.009	4	0
031.010	4	0,14
031.011	1	14
031.012	1	0
031.013	1	14
031.014	1	30
031.015	3	0,14
031.016	2	0,14
031.017	2	0

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.018	2	0
031.019	1	14
031.020	1	0
031.021	2	14
031.022	1	14
031.023	1	25
031.024	1	0
031.025	2	0
031.026	2	29
031.027	1	14
031.028	1	14
031.029	1	0
031.030	2	0,14
031.031	1	14
031.032	1	0
031.033	1	0
031.034	2	0
031.035	1	14
031.036	1	14
031.037	1	14
031.038	2	14

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.039	2	0,14
031.040	1	14
031.041	1	14
031.042	1	0
031.043	1	0
031.044	2	0,14
031.045	1	14
031.046	1	14
031.047	1	14
031.048	3	0,14
031.049	1	0
031.050	3	0
031.051	1	0
031.052	1	14
031.053	2	0
031.054	2	14
031.055	1	32
031.056	1	10
031.057	1	10
031.058	3	3,11
031.059	5	3,10

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.060	1	3
031.061	1	3
031.062	1	3
031.063	1	3
031.064	1	3
031.065	2	3
031.066	2	3
031.067	1	3
031.068	1	3
031.069	1	14
031.070	5	3, 21
031.071	1	29
031.072	1	14
031.073	1	3
031.074	1	3
031.075	1	14
031.076	3	5,14,25
031.077	1	3
031.078	2	3,14
031.079	2	3
031.080	7	10,13

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.081	1	31
031.082	1	10
031.083	1	10
031.084	1	10
031.085	1	10
031.086	1	10
031.087	6	10
031.088	1	10
031.089	1	10
031.090	1	10
031.091	1	10
031.092	2	10
031.093	1	31
031.094	1	10
031.095	1	10
031.096	1	10
031.097	1	10
031.098	1	10
031.099	1	10
031.100	1	23
031.101	1	10
031.102	1	10
031.103	5	10
031.104	2	10

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.105	2	10
031.106	1	32
031.107	1	10
031.108	2	10
031.109	1	10
031.110	1	10
031.111	1	10
031.112	1	10
031.113	1	10
031.114	2	21
031.115	1	21
031.116	1	21
031.117	3	23
031.118	1	29
031.119	1	21
031.120	1	21
031.121	1	32
031.122	1	21
031.123	1	21
031.124	2	31
031.125	1	21
031.126	2	21
031.127	1	21
031.128	1	23

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.129	1	21
031.130	1	21
031.131	2	21
031.132	1	21
031.133	1	21
031.134	1	21
031.135	97	31
031.136	6	29
031.137	31	31
031.138	48	31
031.138A	27	31
031.139	10	27,30
031.140	1	30
031.141	1	30
040.001	2	0
040.002	1	0
040.003	1	0
040.004	1	1
040.005	1	0
040.006	1	0
040.007	1	0
040.008	1	0

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
040.009	1	32
040.010	2	0,31
040.011	1	0
040.012	1	0
040.013	1	0
040.014	1	0
040.015	1	26
040.016	1	0
040.017	1	26
040.018	1	26
040.019	1	26
040.020	7	0
040.021	1	0
040.022	1	0
040.023	2	26
040.024	2	0
040.025	1	0
040.026	2	26,27
040.027	1	0
040.028	1	0
040.029	1	0

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
040.030	1	0
040.031	1	0
040.032	1	26
040.033	1	0
040.034	33	26
040.035	1	26
040.036	2	7,25
040.037	1	29
040.038	1	7
040.039	4	7,11
040.040	1	7
040.041	1	7
040.042	1	7
040.043	1	14
040.044	2	7,32
040.045	6	26
040.046	1	26
040.047	3	26
040.048	1	7
040.049	1	7
040.050	1	26

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
040.051	1	7
040.052	1	26
040.053	2	7
040.054	1	26
040.055	1	7
040.056	1	26
040.057	1	7
040.058	1	7
040.059	1	26
040.060	2	7,14
040.061	1	26
040.062	1	7
040.063	1	7
040.064	1	7
040.065	1	7
040.066	1	7
040.067	1	7
040.068	1	7
040.069	1	7
040.070	1	14
040.071	1	7

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
040.072	1	7
040.073	1	7
040.074	1	7
040.075	50	31
040.076	4	31
040.077	3	31
040.078	3	31
040.079	14	31
040.080	2	25
040.081	2	31
040.082	3	21,26
040.083	1	26
040.084	4	26,27
040.085	1	26
040.086	1	32
040.087	1	21
040.088	3	26,31,32
040.089	1	21
110.001	1	9
110.002	1	25
110.003	1	27

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
110.004	1	25
110.005	1	9
110.006	1	9
110.007	1	9
110.008	1	9
110.009	1	9
110.010	1	9
110.011	1	9
110.012	1	9
110.013	1	9
110.014	1	9
110.015	1	9
110.016	1	9
110.017	1	25
110.018	1	9
110.019	5	9,32
110.020	1	9
110.021	1	9
110.022	1	25
110.023	1	9
110.024	1	9

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
110.025	1	9
110.026	1	31
110.027	1	30
110.028	2	9
110.029	1	9
110.030	2	32
110.031	1	9
110.032	1	9
110.033	1	9
110.034	1	21
110.035	1	9
110.036	2	9,30
110.037	2	9,31
110.038	2	30
110.039	3	30
121.001	4	5
121.002	8	5,31
121.003	1	5
121.004	1	29
121.005	1	5
121.006	1	5
121.007	1	13
121.008	12	5,10,27

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
121.009	1	5
121.010	3	7,31
121.011	2	23
121.012	1	23
121.013	1	23
121.014	1	23
121.015	2	30
121.016	1	23
121.017	1	23
121.018	1	23
121.019	1	30
130.001	1	1
130.002	1	1
130.003	1	1
130.004	1	1
130.005	1	1
130.006	1	1
130.007	1	1
130.008	1	1
130.009	1	13
130.010	1	8

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
130.011	1	8
130.012	1	8
130.013	1	23
130.014	1	8
130.015	1	8
130.016	1	8
130.017	1	8
130.018	1	8
130.019	1	8
130.020	1	23
130.021	1	8
130.022	1	8
130.023	1	8
130.024	5	23
130.025	1	8
130.026	1	8
130.027	1	8
130.028	1	8
130.029	1	8
130.030	1	8
130.031	1	8

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
130.032	1	8
130.033	1	8
130.034	1	8
130.035	4	8,25
130.036	1	8
130.037	1	8
130.038	2	8
130.039	7	8
130.040	1	8
130.041	1	8
130.042	8	8
130.043	1	8
130.044	1	8
130.045	7	0,12
130.046	6	12
130.047	1	12
130.048	2	12,23
130.049	3	12
130.050	1	23
130.051	2	23
130.052	1	23

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
130.053	4	23
130.054	1	23
130.055	9	23
130.056	9	23
130.057	1	23
130.058	1	23
130.059	1	23
130.060	11	23,30
130.061	2	23,30
130.062	3	23,30
130.063	1	30
130.064	2	23
130.065	1	23
130.066	2	23
130.067	4	23,30
130.068	3	23
130.069	2	23
130.070	2	23
130.071	3	23
130.072	6	23
130.073	13	23

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
130.074	1	23
130.075	1	23
130.076	1	23
130.077	1	23
130.078	1	23
210.001	1	0
211.002	1	8
211.003	1	8
211.004	1	14
211.005	1	8
211.006	1	21
211.007	1	14
211.008	2	8,14
211.009	1	30
211.010	2	21,23
211.011	1	8
211.012	2	8,23
211.013	1	8
211.014	2	8
211.015	1	8
211.016	2	29

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.017	2	8
211.018	1	21
211.019	4	23
211.020	1	8
211.021	2	8
211.022	1	8
211.023	2	27
211.024	1	8
211.025	2	23
211.026	1	8
211.027	4	8,30
211.028	2	8
211.029	1	8
211.030	1	8
211.031	10	23
211.032	1	21
211.033	4	30,31
211.034	1	8
211.035	1	8
211.036	1	8
211.037	1	17

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.038	3	8,27
211.039	2	8
211.040	2	8
211.041	1	8
211.042	1	8
211.043	1	8
211.044	1	8
211.045	1	8
211.046	1	8
211.047	1	8
211.048	2	8
211.049	1	23
211.050	2	11
211.051	19	11,23
211.052	1	11
211.053	1	11
211.054	1	23
211.055	1	11
211.056	2	11
211.057	1	32
211.058	3	11

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.059	1	11
211.060	1	11
211.061	2	11,31
211.062	1	11
211.063	1	32
211.064	1	11
211.065	2	11
211.066	2	11,27
211.067	1	11
211.068	1	31
211.069	1	11
211.070	1	31
211.071	1	11
211.072	3	23
211.073	1	11
211.074	1	30
211.075	1	11
211.076	2	11,30
211.077	4	11
211.078	1	11
211.079	4	11

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.080	2	11
211.081	2	11
211.082	3	11
211.083	1	11
211.084	1	23
211.085	4	11
211.086	1	11
211.087	6	0,21
211.088	2	11
211.089	5	11
211.090	1	11
211.091	2	11
211.092	3	23
211.093	1	27
211.094	1	11
211.095	1	11
211.096	1	20
211.097	1	11
211.098	1	11
211.099	1	23
211.100	1	31

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.101	2	11
211.102	1	11
211.103	1	11
211.104	1	11
211.105	3	11
211.106	1	11
211.107	1	21
211.108	1	21
211.109	1	21
211.110	1	21
211.111	1	21
211.112	1	21
211.113	2	21
211.114	1	21
211.115	1	20
211.116	3	21
211.117	1	21
211.118	1	21
211.119	1	21
211.120	2	21
211.121	1	21

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.122	1	21
211.123	1	21
211.124	1	20
211.125	1	21
211.126	1	21
211.127	2	20,30
211.128	2	21
211.129	1	23
211.130	1	21
211.131	1	21
211.132	2	21
211.133	1	21
211.134	1	21
211.135	1	21
211.136	1	23
211.137	1	20
211.138	1	21
211.139	2	20,30
211.140	1	20
211.141	1	20
211.142	1	20

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.143	2	21
211.144	2	21
211.145	1	20
211.146	1	23
211.147	1	20
211.148	3	23
211.149	1	20
211.150	1	21
211.151	1	20
211.152	1	21
211.153	2	20
211.154	1	20
211.155	2	20
211.156	2	20
211.157	2	20
211.158	1	20
211.159	1	23
211.160	1	20
211.161	1	20
211.162	1	20
211.163	1	20

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.164	2	21
211.165	2	20
211.166	2	20
211.167	1	20
211.168	1	27
211.169	2	20
211.170	1	20
211.171	4	21
211.172	1	20
211.173	1	21
211.174	1	32
211.175	1	31
211.176	1	20
211.177	1	20
211.178	1	20
211.179	1	20
211.180	1	21
211.181	1	20
211.182	1	21
211.183	1	20
211.184	1	20

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.185	1	21
211.186	1	20
211.187	1	20
211.188	1	20
211.189	2	21
211.190	1	20
211.191	1	20
211.192	1	20
211.193	1	20
211.194	1	20
211.195	2	20
211.196	1	21
211.197	1	23
211.198	2	30
211.199	2	21
211.200	1	20
211.201	1	21
211.202	2	21
211.203	1	21
211.204	1	21
211.205	2	20

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.206	11	21
211.207	2	20,21
211.208	1	21
211.209	1	30
211.210	1	20
211.211	1	21
211.212	1	32
211.213	2	21
212.001	1	3
212.002	1	3
212.003	3	27
212.004	1	3
221.001	2	7,32
221.002	1	7
221.003	1	7
221.004	2	32
221.005	1	32
221.006	1	7
221.007	1	32
221.008	1	32
221.009	3	7,32

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
221.010	3	7,32
221.011	1	7
221.012	3	7
221.013	1	32
222.001	1	8
222.002	2	8,25
222.003	1	8
222.004	1	8
231.001	1	32
231.002	3	3
231.003	1	32
231.004	2	20,32
231.005	7	20,32
231.006	1	29
232.001	1	3
232.002	1	32
232.003	1	29
232.004	1	29
232.005	2	30
271.001	3	21
271.002	269	21
271.003	3	21

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
271.006	1	21
271.007	1	29
271.008	1	29
271.009	1	29
271.010	1	32
271.011	1	29
271.012	1	29
271.013	1	29
271.014	1	29
271.015	1	29
271.016	1	29
272.001	7	30
272.002	2	30
281.001	2	21
281.002	1	21
281.003	1	30
281.004	1	21
281.005	1	21
281.006	1	21
281.007	1	21
281.008	1	21
281.009	2	21
281.010	1	21
281.011	1	21

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
281.012	1	21
281.013	1	21
281.014	1	21
311.001	1	23
312.001	1	1
312.002	1	1
312.003	1	1
312.004	1	1
312.005	1	1
312.006	1	1
312.007	1	1
312.009	1	8
312.010	1	1
312.011	1	1
312.012	2	1
312.013	1	1
312.014	1	1
312.015	1	8
312.016	1	5
312.017	1	14
312.018	1	11
312.019	1	5
321.001	1	14
321.002	1	1

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
321.003	1	13
321.004	1	21
321.005	4	5,20
331.001	1	13
331.002	2	13
331.003	2	1
331.004	1	1
331.005	1	1
331.006	1	1
331.007	1	1
331.008	1	1
331.009	1	1
331.010	1	1
331.011	1	1
331.012	1	1
331.013	1	1
331.014	1	1
331.015	2	5
331.016	1	5
331.017	1	5
331.018	1	5
331.019	1	32
331.020	1	5
331.021	1	13

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
331.022	1	5
331.023	1	5
331.024	1	5
331.025	2	20
331.026	2	20
331.027	1	20
331.028	2	20
331.029	1	20
360.001	1	1
360.002	1	1
360.003	1	1
360.004	15	0,10,29
360.005	6	10,29
360.006	2	21
360.007	3	21
360.008	4	21
360.009	7	21
360.010	3	21
360.011	2	21
360.012	2	21
360.013	3	21
360.014	16	23
360.015	7	23
360.016	6	23

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360.017	21	23
360.018	5	23
360.019	6	23
360.020	13	23
360.021	1	23
360.022	4	23
361.001	1	21
361.002	2	21
361.003	1	21
361.004	28	21
361.005	6	21
361.006	2	21,29
361.007	2	21
361.008	2	21
361.009	3	21
361.010	4	21
361.011	2	21
361.012	1	21
361.013	2	21
361.014	8	0,21
361.015	3	21

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
361.016	33	27
361.017	88	23,27
361.018	8	23
361.019	44	23
361.020	5	27
361.021	24	27
361.022	1	27
361.023	12	27
361.024	5	27
361.025	3	27
362.001	1	3
362.002	2	3
362.003	1	3
362.004	2	3
362.005	1	5
362.006	1	5
362.007	1	5
362.008	1	5
362.009	2	5
362.010	5	21
362.011	6	26

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
362.012	2	21
362.013	1	21
362.014	4	21
371.001	1	23
371.0002	1	1
371.003	1	1
371.004	1	1
371.005	1	1
371.006	1	5
371.008	1	23
371.009	1	5
371.010	1	5
371.011	1	5
371.012	1	5
371.013	1	5
371.014	1	5
371.015	5	23
371.016	5	23,27
371.017	2	23
371.018	4	23
371.019	2	23

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
371.020	1	23
371.03E	2	20
371.04E	1	20
371.05E	1	20
371.06E	1	20
371.07E	1	20
372.001	1	1
372.002	1	1
372.003	1	1
372.004	1	1
372.005	1	1
372.006	1	1
372.007	1	3
372.008	14	5,29
372.009	1	5
372.010	1	5
372.011	1	5
372.012	1	5
372.013	1	5
372.014	11	5
372.015	5	5

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
372.016	6	5
372.017	1	5
421.001	1	20
421.002	1	20
421.003	1	20
421.004	2	20
421.005	1	20
421.006	1	20
421.007	1	20
421.008	1	21
421.009	1	20
421.010	1	20
421.011	1	20
421.012	1	20
421.013	1	20
421.014	1	27
421.015	1	20
421.016	1	30
421.017	1	20
421.018	1	20
421.019	1	20

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
421.020	2	20
421.021	1	20
421.022	1	20
421.023	1	20
421.024	1	20
421.025	1	20
421.026	1	20
421.027	1	20
421.028	1	20
421.029	1	20
421.030	1	20
421.031	1	20
421.032	1	20
421.033	1	20
421.034	1	20
421.035	1	20
421.036	1	20
421.037	1	20
421.038	1	20
421.039	1	20
421.040	1	20

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
421.041	2	21
421.042	1	29
421.043	17	20,25,30,31
422.001	2	7
422.002	1	26
422.003	1	26
422.004	3	26
422.005	1	7
422.006	3	7
422.007	2	7
422.008	2	7
422.009	1	7
423.001	1	1
423.002	1	1
423.003	1	1
423.004	1	0
423.005	1	1
423.006	2	1
423.007	1	1
423.008	1	1
423.009	1	1

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
423.010	1	1
423.011	4	7
423.012	1	7
423.013	1	7
423.014	1	7
423.015	1	29
423.016	2	7
423.017	1	29
423.018	1	7
423.019	7	7
423.020	4	7
423.021	3	26
423.022	1	7
423.023	12	7,26,12
423.024	1	7
423.025	1	7
423.026	1	7
423.027	1	7
423.028	1	7
423.029	1	7
423.030	22	20,26,27

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
423.031	1	20
423.032	1	20
423.033	1	26
423.034	1	20
423.035	1	20
423.036	1	20
423.037	1	20
423.038	1	20
423.039	1	26
423.040	1	20
423.041	9	26,31
423.042	4	26
423.001	7	5
423.002	2	5
423.003	2	5
423.004	3	5
423.005	1	5
423.006	1	5
423.007	1	5
423.008	1	5
423.009	1	5

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
432.010	1	5
432.011	2	5
432.012	1	5
432.013	2	5
432.014	1	5
432.015	2	5
432.016	1	5
432.017	2	21
432.018	1	21
432.019	1	21
432.020	1	21
432.021	1	21
432.022	1	21
432.023	1	21
432.024	1	21
432.025	1	21
441.001	3	7
441.002	1	7
441.003	1	29
441.004	1	7
441.005	1	7

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
600.001	4	23
600.002	3	23
600.003	2	23
600.004	2	23
600.005	1	23
600.006	1	23
600.007	1	23
600.008	1	23
600.009	1	23

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361.006
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022.030	022.061	022.107
022.032	022.062	022.109
022.053	022.063	022.110
022.054	022.065	022.111
022.055	022.066	022.112
022.056	022.083	022.113
022.057	022.084	022.114
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010.062	010.029
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022.069	022.018
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022.070	022.006
	022.035
022.078	022.048
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031.058	031.001
031.059	031.001
031.060	031.001
031.061	031.039
031.062	031.001
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031.064	031.009
031.065	031.010
031.066	031.009
	031.018
031.067	031.025
031.068	031.032
031.071	031.001
031.077	031.021
	031.037
031.078	031.050
031.083	031.056
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031.090	031.032
031.094	031.033
031.097	031.026
031.099	031.030
031.106	031.006
031.140	031.139
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040.035	040.034
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Question	Question Referenced
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040.078	040.075
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130.052	220.003
130.053	220.004
130.054	220.005
130.055	220.006
130.056	220.007
130.057	220.008
130.058	220.009
130.059	220.010
130.060	220.011
130.061	220.012
130.062	220.013
130.063	220.014
130.064	220.015
130.065	220.016
130.066	220.017
130.067	220.018
130.068	220.019
130.069	220.020
130.070	220.021
130.071	220.022
130.072	220.023
130.073	220.024
130.074	220.025
130.075	220.026
130.076	220.027
130.077	220.028
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211.125	010.052
211.126	010.053
211.130	010.044
211.131	010.045
211.133	010.046
211.134	010.047
211.135	010.048
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Question	Question Referenced
211.212	212.003
221.013	221.002
312.016	022.048
371.001	371.016
371.008	371.016
371.016	371.001
	371.008
421.042	421.014
423.011	423.002
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423.024	423.023
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010.061	131.136
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010.063	040.034
022.031	040.040
022.036	040.041
022.039	040.042
022.050	040.044
022.051	110.038
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022.071	121.012
022.097	121.013
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TABLE 1.7-1(Continued)

Drawing No. (G.E. MPL No.)	Title	Revision No.	Revision Submittal Date
E31-1050 807E154TC	Leak Detection System Sheet 4A	3	Original
E31-1050 807E154TC	Leak Detection System Sheet 5	10	January 1979
E31-1050 807E154TC	Leak Detection System Sheet 6	11	February 1980
E31-1050 807E154TC	Leak Detection System Sheet 7	11	February 1980
E31-1050 807E154TC	Leak Detection System Sheet 8	7	Original
E31-1050 807E154TC	Leak Detection System Sheet 9	9	Original
E31-1050 807E154TC	Leak Detection System Sheet 10	6	Original
E31-1050 807E154TC	Leak Detection System Sheet 11	9	Original
E31-1050 807E154TC	Leak Detection System Sheet 12	10	January 1979
E51-1050 807E173TC	RCIC System - Sheet 1	13	December 1979
E51-1050 807E173TC	RCIC System - Sheet 1A	12	January 1979
E51-1050 807E173TC	RCIC System - Sheet 2	12	January 1979
E51-1050 807E173TC	RCIC System - Sheet 3	12	January 1979
E51-1050 807E173TC	RCIC System - Sheet 4	13	December 1979

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Drawing No. (G.E. MPL No.)	Title	Revision No.	Revision Submittal Date
E51-1050 807E173TC	RCIC System - Sheet 5	12	January 1979
E51-1050 807E173TC	RCIC System - Sheet 6	13	December 1979
E51-1050 807E173TC	RCIC System - Sheet 7	13	December 1979
G33-1050 807E175TC	Reactor Water Cleanup System - Sheet 1	6	Original
G33-1050 807E175TC	Reactor Water Cleanup System - Sheet 2	6	Original
G33-1050 807E175TC	Reactor Water Cleanup System - Sheet 3	6	Original
N64-1050 828E155TC	Off-Gas System - Low Temperature - Sheet 1	9	August 1979
N64-1050 828E155TC	Off-Gas System - Low Temperature - Sheet 2	7	January 1979
N64-1050 828E155TC	Off-Gas System - Low Temperature - Sheet 3	7	January 1979

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3.2.2 SYSTEM QUALITY GROUP CLASSIFICATIONS

System quality group classifications, as defined in NRC Regulatory Guide 1.26, Revision 3, are determined for each water, steam or radioactive waste containing component of those applicable fluid systems relied upon to:

a. Prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary,

b. Permit shutdown of the reactor and maintain it in safe shutdown condtion, and

c. Contain radioactive material

A tabulation of quality group classification for each component so defined is shown in Table 3.2-1 under the heading, "Quality Group Classification." Figure 3.2-1 is a diagram of the relative locations of these components along with their quality group classification.

System quality group classifications as indicated in Tables 3.2-1 and 3.2-2 meet the requirements of 10 CFR Part 50 and Regulatory Guide 1.26, Revision 3.

3.2.3 SAFETY CLASSIFICATIONS

Structures, systems and components are classified as Safety Class 1, Safety Class 2, Safety Class 3 or General Class G in accordance with the importance to nuclear safety. Recognizing that components within a system may be of differing safety importance, a single system may have components in more than one safety class. Supports shall be appropriate for the components supported, as defined by the ASME Code, Section III. Table 3.2-1, Equipment Classification, provides a summary of the safety classes for the principal structures, systems and components of the plant.

Design requirements for components of safety classes are also delineated in this section. Reference is made to accepted industry codes and standards which define design requirements commensurate with the safety function(s) performed.

3.2.3.1 Safety Class 1

3.2.3.1.1 Definition of Safety Class 1

Safety Class 1, SC-1, applies to components of the reactor coolant pressure boundary or core support structure whose failure could cause a loss of reactor coolant at a rate in excess of the normal makeup system, or to equipment in which a single failure could cause major fuel damage.

3.2.3.1.2 Design Requirements for Safety Class 1

The design requirements for SC-1 mechanical equipment (i.e., vessel, pipes, pumps and valves) are delineated in 3.9. The design requirements for the reactor vessel are delineated in 5.3.

The design requirements for SC-1 structures (i.e., reactor pedestal and supports of the reactor coolant pressure boundary as defined in 3.2.3.1.1) are specified in 3.8 and 3.9.

Safety Class 1 structures, systems and components are listed in Table 3.2-1.

3.2.3.2 Safety Class 2

3.2.3.2.1 Definition of Safety Class 2

Safety Class 2, SC-2, applies to those structures, systems and components, other than service water systems, that are not Safety Class 1 but are necessary to accomplish the safety function of:

a. Inserting negative reactivity to shut down the reactor

b. Preventing rapid insertion of positive reactivity

that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. All engineered safeguards fall within this category. All Quality Class I items meet the applicable provisions of 10 CFR Part 50 Appendix B.

b. Quality Class II - Any system, structure, subassembly, component or design characteristic which could cause a safety hazard to plant personnel, an extended reduction in unit output, an unscheduled unit trip, or equipment damage. Appropriate quality assurance requirements for these items are assigned in the purchase specifications.

c. Quality Class G - Any non-nuclear system, structure, subassembly, component or design characteristic to which quality assurance requirements are assigned in accordance with the consequences of failure, operating costs or procurement costs.

3.2.5 CORRELATION OF SAFETY CLASSES WITH INDUSTRY CODES

The design of plant equipment is commensurate with the safety importance of the equipment. Hence, the various safety classes have a gradation of design requirements. The correlation of safety classes with other design requirements are summarized in Table 3.2-3.

3.2.6 IDENTIFICATION OF SAFETY RELATED SYSTEMS AND COMPONENTS ON FLOW DIAGRAMS

The system classifications are shown on the flow diagrams using symbols code group A, B, C, D; Seismic Category I, II; and Quality Class I, II and G. These symbols correspond to Table 3.2-1 classification as follows:

Flow Diagram	Table 3.2-1
Code Group A	Quality Group A
Code Group B	Quality Group B
Code Group C	Quality Group C
Code Group D	Quality Group D

Flow Diagram	Table 3.2-1
Seismic Category I	Seismic Category I
Seismic Category II	Seismic Category II
Quality Class I	Quality Class I
Quality Class II	Quality Class II
Quality Class G	Quality Class G

Flow diagrams indicating the safety classes assigned to each piping system are shown in Figures 3.2-2 to 3.2-25. Table 3.2-1 references these figures for each system. The list of tables, which follows the Table of Contents, provides a cross-reference between FSAR figure numbers and Burns and Roe, Inc. drawing numbers, where applicable. These figures delineate the boundary of each safety class for each system. Figures 3.2-2 to 3.2-25 present as-built safety classifications. In some instances, these classifications reflect voluntary upgrades which may exceed Regulatory Guide 1.26, Revision 3 requirements. For inservice inspection requirements, the appropriate levels of inspection are given in 5.2.4 and 6.6.

TABLE 3.2-1 (Continued)

Principal Component (1)	Scope of Supply (2)	Safety Class (3)	Location (4)	Quality Group Classification (5)	Quality Class (6)	Seismic Category (7)	Comments
13. RCIC System (Figure 3.2-81)							
.1 Piping, within outermost isolation valves Reactor Coolant Pressure Boundary	P	1	C,R	A	1	1	(12)
.2 Piping, beyond outermost isolation valves, except as noted in 3	P	2	R	B	1	1	(12 & 23)
.3 Piping, return test line to condensate storage tank beyond second stop valve; drip pot discharge valve to condenser; condenser to vacuum tank and to the condensate pump discharge; and vacuum pump discharge to the outboard check valve break flange	P	G	R	D	11	11	(12 & 32)
.4 Pumps	GE	2	R	B/D	1	1	(39
.5 Water leg pumps	P	2	R	B	1	1	
.6 Valves, isolation and Coolant Pressure Boundary	P	1	C	A	1	1	(12)
.7 Valves, other	P	2	R	B/D	1	1	(13 & 39)
.8 Turbine	GE	2	R	N/A	1	1	
.9 Electrical modules, with safety function	GE	2	R	N/A	1	1	
.10 Cable, with safety function	P	2	R,W	N/A	1	1	
14. Fuel Service Equipment							
.1 Fuel preparation machine	GE	3	R	N/A	1	1	
.2 General purpose grapple	GE	3	R	N/A	1	1	
15. Reactor Vessel Service Equipment							
.1 Steam line plugs	GE	3	R	N/A	1	1	
.2 Dryer and separator sling and head strongback	GE	3	R	N/A	1	1	
16. In-Vessel Service Equipment							
.1 Control rod grapple	GE	3	C	N/A	1	1	

TABLE 3.2-1 (Continued)

Principal Component (1)	Scope of Supply (2)	Safety Class (3)	Location (4)	Quality Group Classification (5)	Quality Class (6)	Seismic Category (7)	Comments
17. Refueling Equipment							
.1 Refueling equipment platform assembly	GE	3	C	N/A	I	I	
.2 Refueling Bellows	P	G	C,R	D	II	I	(33)
18. Storage Equipment							
.1 Fuel storage racks	GE/P	3	R	N/A	I	I	
.2 Defective fuel storage container	GE	3	R	N/A	I	I	
19. Radwaste System (Figures 11.2-2, 3.2-9, 3.2-10, 11.2-3, 11.2-4a thru 11.2-4c, 11.4-1a, 11.4-1b)							
.1 Tanks, Atmospheric	GE/P	G	W	C	II	II	(16, 24 & 38)
.2 Heat exchangers	GE/P	G	W	C	II	II	(16 & 24)
.3 Piping and valves forming part of containment boundary	P	2	C,R	B	I	I	
.4 Piping, other	P	G	W	C	II	II	(16 & 38)
.5 Pumps	GE/P	G	W	C	II	II	(16, 24 & 38)
.6 Valves, flow control and filter system	GE/P	G	W	C	II	II	(16, 24 & 38)
.7 Valves, other	P	G	W	C	II	II	(12, 16, 24 & 38)
.8 Mechanical modules	GE/P	G	W	C	II	II	(16 & 24)
.9 Radioactive Equipment & Floor Drains and other radwaste piping and valves upstream of collector tanks	P	G	R,T,W	D	II	II	(32)
.10 Instrumentation and control boards	GE/P	G	W	N/A	II	II	
.11 Concentrator	GE	G	W	C	II	II	
.12 Plant discharge line	GE/P	G	W	D	II	II	(37)
20. Reactor Water Cleanup System (Figure 3.2-11)							
.1 Vessels, filter/demineralizer	GE	G	W	C	II	II	
.2 Heat exchangers	GE	G	W	C	II	II	
.3 Piping, within outermost isolation valves	P	1	C	A	I	I	(12)
.4 Piping, beyond outermost containment isolation valves	P	G	R,W	C	II	II	(12 & 32)
.5 Pumps	GE	G	R	C	II	II	(12 & 32)
.6 Valves, isolation valves Reactor Coolant Pressure Boundary	P/GE	1	C,R	A	I	I	(12)

Table 3.2-1 (Continued)

Principal Component (1)	Scope of Supply (2)	Safety Class (3)	Location (4)	Quality Group Classification (5)	Quality Class (6)	Seismic Category (7)	Comments
.7 Valves, beyond outermost containment isolation valves	GE/P	G	R,W	C	II	II	(12 & 32)
.8 Mechanical modules	GE	G	R,W	C	II	II	(32)
21. Fuel Pool Cooling and Cleanup System (Figure 3.2-12)							
.1 Vessels, filter/demineralizers	P	G	W	C	II	II	
.2 Vessels, other	P	G	W	C	II	II	
.3 Heat exchangers	P	3	R	C	II	II	(32 & 42)
.4 Piping	P	3	R,W	C	II	II	(32 & 43)
.5 Pumps	P	3	R	C	II	II	(32 & 42)
.6 Makeup System (normal)	P	G	R	C	II	II	(18 & 32)
.7 RHR Connection	P	3	R	C	I	I	
.8 Makeup System (emergency)	P	3	R	C	I	I	
.9 Piping, suppression pool to outer isolation valves	P	2	R	B	I	I	
22. Control Room Panels							
.1 Electrical modules with safety function	GE	2	W	N/A	I	I	
.2 Cable, with safety function	GE/P	2	W	N/A	I	I	
23. Local Panels and Racks							
.1 Electrical modules, with safety function	GE	2	R	N/A	I	I	
.2 Cable, with safety function	P	2	R	N/A	I	I	
24. Off-Gas System (Figure 11.3-2)							
.1 Tanks	GE	G	T,W	C	II	II	(16)
.2 Heat exchangers	GE	G	T,W	C	II	II	(16)
.3 Piping	P	G	T,W,O	C	II	II	(16)
.4 Pumps	GE	G	T,W	C	II	II	(16)
.5 Valves	P	G	T,W	C	II	II	(16)

TABLE 3.2-1 (Continued)

Principal Component (1)	Scope of Supply (2)	Safety Class (3)	Location (4)	Quality Group Classification (5)	Quality Class (6)	Seismic Category (7)	Comments
.6 Mechanical modules, with safety function	GE	G	T,W	C	II	II	(16 & 12)
.7 Pressure vessels	GE	G	T,W	C	II	II	(16)
25. Standby Service Water System (Figure 3.2-13)							
.1 Piping	P	3	P,R,DG,O	C	I	I	
.2 Pumps	GE	3	P	C	I	I	
.3 Pump motors	GE	3	P	N/A	I	I	
.4 Valves	P	3	P,R,DG,O	C	I	I	
.5 Electrical modules, with safety function	P	3	P,R,DG,O,W	N/A	I	I	
.6 Cable, with safety function	P	3	P,R,DG,O,W	N/A	N/A	I	
26. Turbine Plant Service Water (Figure 9.2-1)							
.1 Piping and Valves	P	G	T,R,O,P,W	D	II	II	(32)
.2 Pumps	P	G	P	D	N/A	II	
27. Reactor Building Closed Cooling Water System (Figure 3.2-14)							
.1 Heat Exchangers	P	G	R	D	II	II	(32)
.2 Pumps	P	G	R	D	II	II	(32)
.3 Tanks	P	G	R	D	II	II	(32)
.4 Piping and Valves Inside Containment	P	G	C	C	II	II	(32)
.5 Containment Isolation Valves and Associated Piping	P	2	C,R	B	I	I	
.6 Piping and Valves in Reactor Building	P	G	R	D	II	II	(32)
.7 Piping and Valves Other	P	G	W	D	II	II	(32)

Notes (Continued)

33. Although the refueling bellows are designed to withstand the SSE without rupture, they may be plastically deformed.

34. Inspection records shall be maintained as provided in the Technical Specifications.

35. This piping has been voluntarily upgraded from Safety Class 3 to Safety Class 2 and from Quality Group Classification C to Quality Group Classification B.

36. The following qualification is met with respect to the certification requirements:

    a. The manufacturer of the turbine stop valves, turbine governor valves, turbine bypass valves, and main-steam leads from turbine control valve to turbine casting utilized quality control procedures.

    b. A certification has been obtained from the manufacturer of these valves and steam leads that the quality control program so defined has been accomplished.

37. Up to and including the last stop valve, this line meets requirements of Quality Group C.

38. Equipment, piping, and valves which are part of the radwaste solids handling system are designed to Quality Group D standards.

39. Equipment classification is commensurate with the quality group classification of the associated piping.

40. These piping systems are liquid penetrant or magnetic particle examined to the acceptance standards of ASME Section III, Class 3.

41. The auxiliary piping systems on the engines are built to the guidelines of ANSI B31.1.

42. Heat exchangers required for cooling will be Quality Class 1 and Seismic Category I by the first refueling outage (nozzle reinforcement), as will the pumps, motors, and valves. Piping will be ASME Section III, Class 3, Code Group C, Seismic Category I.

TABLE 3.2-1 (Continued) Page 25 of 25

Notes (Continued)

43. Piping, excluding valves (see Note 42), is ASME Section III, Class 3, Code Group C, Seismic Category I for the cooling portion of the fuel pool cooling system.

44. Safety-related instrument and control systems are identified in Chapter 7, Table 7.1-1.

3.6.1.6.2.2 Steel Targets

The Ballistic Research Laboratories formula is used to determine perforation of a steel target. The thickness, T, of a steel target that will be just perforated by a missile is given as (Reference 3.6-17):

$$T^{3/2} = \frac{0.5MV^2}{17{,}400\ K^2 D^{3/2}} \qquad \text{(Eq. 3.6.1.6.2.2-1)}$$

where:

T = Steel wall thickness to just perforate (inches)

M = Mass of the missile (weight/g in $lb\text{-}sec^2/ft$)

V = Velocity of missile (ft/sec)

K = Constant depending on grade of steel and is usually $\cong 1$

D = Diameter of missile (inches). For irregularly shaped missiles, an equivalent diameter is used, taken as the diameter of a circle with the same area as the projected frontal area of the irregularly shaped missile

The recommendation in Reference 3.6-13 to increase the perforation thickness, T, obtained by the Ballistic Research Laboratories Formula by 25% to prevent perforation is observed; that is:

$$t_p = 1.25T \qquad \text{(Eq. 3.6.1.6.2.2-2)}$$

where:

t_p = Thickness of steel barrier required to prevent penetration (inches)

3.6.1.6.3 Overall Structural Response

3.6.1.6.3.1 General

In general, pipe break loads are considered in combination with other loads (see 3.6.1.6.6). Dead loads, live loads, operating thermal loads and earthquake loads may or may not be significant compared to the pipe break load, depending on the severity of the pipe break load. Thermal loadings due to pipe break have only skin effect and are not considered.

Pressure loads due to pipe break do not necessarily peak with pipe whip and jet impingement loads; however, in the analysis, they are considered to act simultaneously.

With regard to pipe break, when high energy pipes under pressure fail, a fluid jet is created. The associated jet impingement force on a target as well as the reaction force exerted on the piping by the fluid jet force have a time history qualitatively presented in Figure 3.6-118. This force is conservatively idealized as a step function load. For the fluid forces associated with these pipe failures, see Table 3.6-6.

To obtain a solution for the actual complex system, the structure is idealized by an equivalent single degree of freedom system (see Figure 3.6-119) following the procedures described by J. M. Biggs in Chapter 5 of "Introduction to Structural Dynamics" (Reference 3.6-1). The response of this mathematical idealization to a step function load (jet impingement) or to a step function load concurrently with an impact loading (due to whipping pipe) involves an energy transfer from the impacting object to the impacted structure. The following exposition on how this energy transfer is addressed makes use of procedures that have been presented by the Bechtal Corporation in its report on missile impact, Topical Report BC-TOP-9A, Revision 2 (Reference 3.6-13).

3.6.1.6.3.2 Structural Response to Whipping Pipe Missile Impact Load

a. Discussion

A method of energy-balance procedures is utilized in order to evaluate the structural response, when a missile impacts a target. The method utilizes the strain energy of the target at maximum response to counteract the residual kinetic energy of the target or target missile combination that results from the missile impact.

A missile of mass M_m is postulated to strike a spring-backed target mass, M_e, with a velocity, V_s. Since the actual coupled mass during impact varies, an estimated average effective target mass, M_e, is used to evaluate the inertia effects during impact. The impact of the missile is considered plastic. This assumes that the missile remains in contact with the target after impact.

If other loads are present on the target structure which act concurrent with missile impact loads, (see 3.6.1.6.5, 3.6.1.6.6 and Table 3.6-11), the maximum combined displacement is determined as follows:

Let:

$x' = x_e - x_o$ (see Figure 3.6-120) (ft.)

x_o = Displacement due to other loads (ft.)

x_e = Yield displacement (ft.)

x_m = Maximum combined displacement (ft.)

R_m = Plastic resisting force (lbs.)

k = Elastic spring constant (lbs./ft.)

Then:

$$E_s = \frac{k(x')^2}{2} + kx'(x_m - x_e)$$

(See Figure 3.6-120)

or:

$$x_m = \frac{E_s}{kx'} - \frac{x'}{2} + x_e$$

Substituting $x' = x_e - x_o$ in the above equation gives:

$$x_m = \frac{E_s}{k(x_e - x_o)} + \frac{x_e + x_o}{2} \quad \text{(Eq. 3.6.1.6.3.2-12)}$$

The required ductility ratio, μ_r, is obtained by dividing both sides of Equation 3.6.1.6.3.2-12 by x_e.

$$\mu_r = \frac{Es}{R_m(x_e - x_o)} + \frac{1 + x_o/x_e}{2} \quad \text{(Eq. 3.6.1.6.3.2-13)}$$

The values of μ_r should be less than the allowable ductility ratios, μ, given in Table 3.6-1.

3.6.1.6.3.3 Jet Impingement

Jet impingement loads are loads that emanate from a break in a high energy line. It is postulated that the characteristics of the jet are such that the jet exits from a break opening in the pipe equal in area to the cross sectional area of the pipe itself (see Figure 3.6-117). The jet is postulated to travel conforming to the configuration of the cross sectional area of the pipe for a distance of five pipe diameters and then to diverge at an angle of divergence of 10°. For the jet thrust forces at the postulated breaks, see Table 3.6-6. Jet loads impacting structures are treated as equivalent static loads. A dynamic load factor is applied to the jet force emanating from the pipe and the resulting load is modified by an appropriate load factor according to its use in combination with other loads. The structure impacted is then evaluated for structural capability.

3.6.1.6.4 Allowable Design Stresses and Strains

For allowable design stresses and strains for reinforced concrete and structural steel, see 3.8.4.5 and Tables 3.8-12 and 3.8-17, except as modified in 3.6.1.6.4.1 and 3.6.1.4.2.

3.6.1.6.4.1 Pipe Whip Loading With or Without Other Loads

The acceptability of pipe whip loading with or without other loads is considered from two aspects:

a. The overall structural response of the impacted structural element

b. The local damage sustained by the impacted structural element.

The overall structural response is considered acceptable if the ductility ratio resulting from the loading does not exceed the maximum allowable ductility ratios as given in Table 3.6-1. The determination of ductility ratios utilizes the procedures set forth in 3.6.1.6.3 and the loading combinations in 3.6.1.6.6. In using these procedures, the allowable limit on section strength, M , used in the determination of yield displacement X_e, (3.6.1.6.3.2e, Tables 3.6-9, 3.6-10 and Figure 3.6-120) is computed in accordance

with the strength design methods described in ACI 318-71 (Reference 3.6-12) and in the general practices of Part 2 of the AISC specifications (Reference 3.6-11), modified by the dynamic strength increase factors of Table 3.6-8.

The local damage is considered acceptable if the pipe whip impact does not cause spalling and excessive penetration in concrete, or perforation in steel, as determined by the procedures set forth in 3.6.1.6.2.

3.6.1.6.4.2 Pipe Break Loads (Excluding Pipe Whip) With or Without Other Loads

Pipe break loads (excluding pipe whip) with or without other loads are considered acceptable if the loading from the loading combinations in 3.6.1.6.6 does not result in stresses that exceed the allowable limits on section strength as given in Tables 3.8-12 and 3.8-17, modified by the dynamic strength increase factors in Table 3.6-8.

3.6.1.6.5 Loads, Definition of Terms and Nomenclature

For loads, definition of terms and nomenclature, see 3.8.4.3.

3.6.1.6.6 Load Combinations

3.6.1.6.6.1 Seismic Category I Concrete Structures

For load combinations for Seismic Category I concrete structures, see Table 3.8-15, load combinations 6, 7, and 8.

3.6.1.6.6.2 Seismic Category I Steel Structures

For load combinations for Seismic Category I steel structures, see Table 3.8-16, load combinations 6, 7 and 8.

3.6.1.7 Structural Design Loads

Structural elements are designed to withstand the loads generated by piping failures outside of primary containment in combination with other loads given in 3.6.1.6.6. Table 3.6-11 furnishes the design loads considered in the areas where piping failures occur.

3.6.1.8 Analysis of Load Reversal

Structural elements such as floors, interior walls, exterior walls and the building as a whole are analyzed for the effects of reversal of load due to the postulated pipe failure accident. They are also analyzed for rebound loads that accompany pipe break accidents. The analysis approach for rebound is set forth in Figure 3.6-122.

3.6.1.9 Modified Structures

New Openings or other modifications are not planned to be provided in existing structures, and, therefore, the capabilities of structures to carry the design loads due to modification need not be demonstrated.

3.6.1.10 Verification That Failure of Any Structure Does Not Preclude Safe Reactor Shutdown

Structures subjected to pipe whip and/or jet impingement loads are investigated and found not to fail under these loads in conjunction with the applicable load combinations, so that there are not cases of structural barriers failing and causing additional structural failures which would adversely affect the mitigation of the consequences of accidents and the capability to bring the plant to a cold shutdown condition.

3.6.1.11 Verification That Adequate Redundancy Exists for All Postulated Fluid Piping System Ruptures

3.6.1.11.1 Approach

The purpose of the study is to ensure that for all postulated ruptures of fluid piping systems, safe reactor operation and shutdown is not precluded. The basis of this approach is that adequate separation of redundant systems or components, required to shutdown and maintain the reactor in a cold condition, provides the level of protection required to ensure safe reactor operation and shutdown.

The input used for this study includes the routing of all cables, cable trays and conduit necessary to shutdown and maintain the reactor in a cold condition. The locations of all motor control centers, instrument racks, sensors and heating ventilation and air conditioning (HVAC) equipment necessary to shutdown and maintain the reactor in a cold condition are also included in the input of this study.

electrical division to which the component belongs; what the function of the component is; the various references, such as the drawings, in which the component is found; devices interconnecting the component and another system; and additional information of this type. This coding facilitates storage of the input for retrieval at any time.

Table 3.6-6 lists the high energy design basis break locations outside containment, the piping subsystems involved, the pipe diameter, the plan figure showing the piping subsystem, the maximum blowdown thrust or the thrust versus time figure, and the room or area containing the postulated pipe break. Figures 3.6-41a through 3.6-41h illustrate the high energy break locations outside containment.

Figures 3.6-12 through 3.6-36 illustrate and list the high energy break locations inside containment.

Moderate energy crack locations are postulated in accordance with Standard Review Plans 3.6.1 and 3.6.2.

3.6.1.11.2 Method of Analysis for Postulated High Energy Fluid System Ruptures

3.6.1.11.2.1 Effects of Postulated Passive Component Failures

Postulated pipe breaks in high energy fluid systems are investigated to determine their effects on the ability to bring the plant to a safe shutdown and to limit the offsite radiological consequences to an acceptable level as stated in 10CFR50.

On a case-by-case basis, the effects of pipe whip, jet impingement, and the resulting environmental conditions on safety-related equipment are evaluated. The effects of the postulated pipe break are dependent on the fluid properties of the system; the location and orientation of the pipe break, the proximity to safety-related systems, components, and structures, and the individual design limits of the safety-related systems, components, and structures.

Pipe breaks in high energy systems are postulated according to the criteria in 3.6.2.1. After identifying what equipment becomes inoperable, a single random active component failure is postulated in a system not effected by the postulated high energy fluid system rupture. Additionally, if the direct consequences of the postulated rupture results in a reactor or turbine trip, offsite power is assumed unavailable.

3.6.1.11.2.2 Analytical Procedure

After all the consequences of the postulated passive and active component failures are evaluated, an analysis determines if safe reactor control is maintained. The following guidelines are used in this analysis:

a. For postulated ruptures of fluid piping systems, ensure that core cooling and reactivity control is maintained.

b. Demonstrate that redundant components or systems necessary to safely shut down and cool the reactor are not involved in the postulated passive component failure.

c. Demonstrate that offsite radiological consequences do not exceed relevant standards.

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3.6.1.11.3 Method of Analysis for Postulated Moderate Energy Fluid System Ruptures

3.6.1.11.3.1 Approach

Postulated ruptures in moderate energy fluid systems do not generate pipe whip. The analysis investigates the effects of the environment which results from such a postulated rupture on safety-related equipment, including the effects of water spray.

The effects of the postulated moderate energy pipe cracks are dependent on the fluid properties, available fluid reservoir, drain systems, location of the safety-related equipment, components, and structures, and the individual design limits of the safety-related equipment, components, and structures.

Where moderate energy pipe cracks are postulated in close proximity to high energy systems, the environmental analysis compares the effects of both high and moderate energy pipe ruptures. The most limiting case is evaluated for safe cold shutdown.

Moderate energy pipe cracks are postulated according to the criteria in 3.6.2.1.

3.6.1.11.3.2 Method of Analysis

The locations of all postulated ruptures, resulting in through wall leakage cracks, are identified for later retrieval. The analysis assumes that the spray resulting from a postulated moderate energy rupture causes the malfunction of all equipment not enclosed by watertight compartments.

Additionally, the most damaging single random active component failure in a system not effected by the postulated passive component failure is postulated. If the direct consequences of the pasive component failure results in a turbine or reactor trip, then offsite power is assumed unavailable.

3.6.1.11.4 Summary of Analysis

The analyses discussed in 3.6.1.11.2 and 3.6.1.11.3 do not identify any location where a postulated passive component

failure in a high or moderate energy system precluded the safe shutdown and cooling of the reactor. Therefore, the ruptures in fluid piping systems, which are postulated, have no effect on the ability to bring the reactor to a cold shutdown condition.

This analysis by actual examination of the plant is undertaken to provide results based on as-built conditions.

Design drawings are used to supplement the study in cases where piping or equipment have not been installed. Prior to fuel load, a walkdown of the plant is performed to verify the results of the analysis and confirm that all design modifications have been implemented.

Piping layouts for areas containing high and moderate energy lines, whose failure can affect the performance of safety-related equipment, are presented as Figures 3.6-43 through 3.6-62, inclusive.

Section 3.6.1.11 discusses in detain the methods used to demonstrate that no single postulated passive component failure, in conjunction with a single active component failure, precludes safe shutdown of the plant.

The following should serve to further clarify the method of analysis:

a. The forces developed at each postulated high energy pipe break are determined by the methods of 3.6.2.2. The effects of the resultant pipe whip and jet impingement are evaluated. Credit is taken for automatic isolation and operator action to mitigate the consequences of the postulated pipe break, if the equipment required for this function is not affected by the break or included in 3.6.1.11.4(c) below.

b. As a first step, all equipment impacted by the whipping pipe or jet is assumed to fail. If the equipment is required for safe cold shutdown or accident mitigation, a detailed analysis is performed to determine if the equipment will actually fail. Structures contacted by the whipping pipe or jet are evaluated for structural adequacy by the methods of 3.6.2.2.

Impacted pipes of smaller nominal diameter than the impacting pipe are assumed to fail, regardless of wall thickness of impacted pipe. Impacted pipes of both larger nominal diameter and thinner wall thickness than the impacting pipe are assumed to develop through wall leakage cracks.

c. Additionally, a single random active component not affected by a) and b) is assumed to malfunction. Should a) or b) result in a turbine generator or reactor trip, then offsite power is assumed unavailable.

d. After a), b), and c) above have been evaluated, possible shutdown modes are analyzed. If shutdown is possible, the postulated passive component failure is not significant from a safety standpoint.

e. Should alternate shutdown modes not be available then:

   1. Reroute or relocate cable, pipe, or equipment to prevent loss of function.

   2. If (1) is not feasible, shield the adversely affected component(s) to prevent loss of function.

f. The flooding and environmental effects of moderate energy failure are evaluated to determine whether they are more severe than the high energy breaks and are addressed in 3.6.1.15.

The area temperature is evaluated by determining the limiting postulated pipe break and using RELAP4/MOD5 (Reference 3.6-21). The limiting pipe break for temperature analysis is that pipe break giving the highest energy release rate over the longest blowdown period.

The effects of flooding are evaluated by determining the limiting pipe break and calculating the effects of the fluid release. The limiting pipe break for flooding analysis is that pipe break with the highest mass flow rate over the longest blowdown period.

Peak differential pressure analysis results are provided in Table 3.6-12 and discussed in 3.6.1.20.

Refer to 3.6.1.13 for electrical equipment environmental qualifications.

3.6.1.12 Control Room Habitability

A postulated rupture of either the main steam or feedwater piping has no effect on the continued habitability of the control room, since the radiation dose that control room personnel receive as a result of a postulated rupture is below the allowable limits.

The nuclear steam supply system (NSSS) piping outside of primary containment within the reactor building is enclosed by the main steam tunnel. The main steam tunnel, provided with pressure-relieving blowout panels, is designed to withstand the worst postulated piping system rupture attributable to the NSSS within the steam tunnel.

The high energy piping in the main steam tunnel is provided with pipe whip restraints as described in 3.6.1.5. These restraints limit the motion of the free ends of the ruptured NSSS piping to preclude the impact of the NSSS piping with the main steam tunnel structure.

The remaining high energy piping outside the primary containment is not routed in the vicinity of the control room, or does not possess sufficient energy to adversely affect the structural integrity of the control room wall.

Additionally, a remote shutdown panel is provided to permit safe reactor shutdown to a cold condition in the event the control room must be evacuated.

3.6.1.13 Electrical Equipment Environmental Qualifications

All electrical systems, necessary for safe shutdown and necessary to maintain the plant in a safe shutdown condition, are designed to remain functional in the general area environment resulting from a high energy line break or from leakage cracks in moderate energy piping. Specific equipment is either:

a. Designed to remain functional as long as necessary in the general area environment.

b. Isolated from the general area environment in compartments capable of maintaining normal equipment operating conditions.

Certain rotating equipment cannot be designed to function in the more severe, local steam environment. However, due to physical separation, rotating equipment, of not more than one subsystem, is exposed to the local conditions which exceed the general area accident environment. Required redundancy is thus maintained for safety equipment.

Refer to 3.11 for a more complete description of environmental design of electrical equipment.

3.6.1.13.1 Identification of Equipment

Safety equipment required to mitigate the consequences of an accident and place the reactor in a cold shutdown condition is listed in Table 3.11-2. The table also indicates the required duration, following an accident, which equipment is required to operate.

3.6.1.13.2 Environmental Design

Refer to 3.11 for a discussion of environmental design and an analysis of safety-related electrical components. The section identifies the safety-related equipment that must operate in a hostile environment, and Table 3.11-2 indicates the postulated environmental envelop conditions for both the general and local accident areas.

3.6.1.13.3 Jet Impingement Barriers

For results of the steam system study, see 3.6.1.11.4. Analysis indicates jet impingement barriers are not required at WNP-2 since no postulated pipe break precluded reactor safe shutdown. Some room walls, floors, and ceilings act as jet impingement barriers, however.

3.6.2.3.2 Jet Impingement Effect

3.6.2.3.2.1 Physical Separation

The physical separation of different essential systems and components is used to ensure that the plant retains function of sufficient essential systems to assure safe shutdown in the event of a postulated LOCA, and subsequent generation of a jet stream together with an additional single random active component failure and the loss of offsite power.

Where physical separation cannot be used to protect systems, a detailed analysis is performed to determine the effects of jet impingement on their operability. If necessary, barriers are provided to protect structures, systems, and components required for a safe shutdown, to prevent offsite radiological consequences, and to mitigate the effects of a LOCA.

3.6.2.3.2.2 Jet Impingement Evaluation

The evaluation of the adequacy of physical separation included the inspection of all essential systems and their components that are necessary to start, operate, and control the essential systems required for safe shutdown. The evaluation included the following:

a. Review pipe break locations inside primary containment, to provide conservative jet stream orientation and geometry.

b. Review effected equipment by both design drawing examination and plant walkdown.

c. Review signals that result in the actuation of essential systems.

d. Review signals that are necessary to be returned to inside primary containment, to activate the shutdown systems.

e. Review availability of power that is required inside primary containment to operate the essential systems.

f. Review mechanical engineered safety systems required for safe shutdown.

For systems or parts thereof, that are necessary for a safe shutdown but cannot be protected by redundancy, a detailed analysis is performed to determine jet impingement effects on the operability of these systems. Barriers are provided where necessary.

3.6.2.3.2.3 Postulated Pipe Rupture Locations Inside Containment

The criteria used to define pipe rupture locations is described in 3.6.2.1 and is shown in Figures 3.6-12a through 3.6-17a, 3.6-18a, 3.6-18b, 3.6-19a through 3.6-34a and 3.6-35.

3.6.2.3.2.4 Signals from Primary Containment

For instrumentation located inside primary containment, sufficient redundancy is provided such that all signals necessary to cause actuation of essential systems, remain functional. Each system, that is required to bring the plant to a safe shutdown condition, is furnished with two or more sets of redundant instrumentation lines.

In this review, it is conservatively assumed that a jet stream or whipping pipe may damage one of these sets. The redundant system is shown to remain operational by physical separation and barriers, such as the RPV. An example of the above is the location of Sets "A" and "B" instrumentation lines for the HPCS. Set "A" and its redundant Set "B" are located at opposite sides of the RPV. Therefore, a jet stream or whipping pipe cannot damage both sets of instrumentation. Function of instrumentation inside primary containment necessary to result in the actuation of the HPCS system is thereby assured. These conditions, as discussed for the HPCS instrumentation lines, are typical for all instrumentation lines that support essential systems. The capabilities of redundant instrumentation is discussed in 7.3.

3.6.2.3.2.5 Signals to the Primary Containment

No instrumentation signal is necessary to return inside primary containment to operate any of the essential systems. Signals to the ADS valves are provided through their power supply as described in the following section.

3.6.2.3.2.6 Power Requirement Inside Primary Containment

The only essential system that required power, inside primary containment, is the automatic depressurization system (ADS).

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TABLE 3.6-1

MAXIMUM DUCTILITY RATIOS

STEEL STRUCTURAL COMPONENTS

Steel Beams (Lateral Load) (Note: To develop this ductility, the flanges must be thick enough to prevent local plastic buckling).	26
Steel Beams (Lateral and Axial Load)	8
Welded Portal Frames (Vertical Load)	6-16

REINFORCED CONCRETE STRUCTURAL COMPONENTS

Tension reinforced concrete beams and slabs, (flexure controls design)	$\frac{0.10}{p^*} \leq 10$
Doubly reinforced concrete beams and slabs, (flexure controls design)	$\frac{0.10}{p-p'^{**}} \leq 10$
Reinforced concrete columns, walls and other elements exhibiting brittle fracture, (compression controls design)	1.3

*p is the ratio of tensile reinforcement and must satisfy the limitations:

$$0.0025 \leq p = \frac{As}{bd} \leq 0.015$$

**p' is the ratio of compression reinforcement and must satisfy the limitations:

$$p' = \frac{A's}{bd} \geq 0.0025$$

TABLE 3.6-11

DESIGN LOAD IN AREAS WHERE PIPING FAILURES OCCUR

Pipe Break Nos.	Room	Elev. (ft.)	Differential Pressure (psi)	Differential Temperature °F Int. to Int.	Int. to Ext.	Live Load (psf)	Hung Loads (psf) From Floor	From Ceiling	Equip. Loads (Kips)
8-10	R 15	422	0.51	0°	40°	-	-	59	1.4k Pump
4	R 113	441	0.33	0°	40°	250	59	68	None
5-7	R 112	441	0.51	0°	40°	250	59	68	None
41-45	R 206	471	0.05	0°	40°	250	32	34	None
1, 2, 21, 22	R 313	510'-6"	0.48	0°	40°	250	40	30	None
20	R 408	522	1.0	0°	-	250	41	88	None
13-15, 30-34	R 406 & 407	522	15.0	0°	-	250	126	0	1.5k Pump
11, 12, 23-29, 32	R 409	535	11.0	0°	-	250	40	80	None
60, 61, 65	R 511	548	4.4	20°	-	400	80	55	None
16-18, 53-56, 62-64	R 510	548	1.8	20°	-	400	65	51	Heat Exchs. 16.2 & 29.5
19	R 509	548	2.1	20°	-	400	88	50	None
40, 47, 49, 70-82	R 604	572	0.03	0°	40°	250	15	36	Heat & Vent Unit 51K
90-94	R 504	548	0.00	0°	40°	400	59	15	None
6	R 308	501	0.41	0°	40°	1000	63	55	None
Steam Tunnel	R 310	501	20.0	20°	-	1000	277	41	None

NOTES: 1. For location of pipe break nos., see Figures 3.6-43 through 3.6-62.
2. For vertical and horizontal seismic factors, see 3.7.

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TABLE 3.6-12

SUMMARY OF SUBCOMPARTMENT PRESSURE ANALYSIS[a]

Compartment Where Break Occurs: Elevation (ft.)	Room Number	Description	Piping System: Line Designation	Differential Pressure: Maximum Differential (psi)	Differential Between the Rooms	Time of the Peak (sec)	Design Pressure (psi)
442	R14/R113	RHR Pump Rooms	4" RCIC (13)-4	0.33	R14, R113/R206	0.33	0.50
				0.33	R14, R113/R12, R114	0.33	0.50
				0.33	R14, R113/R15, R112	0.33	0.50
422	R15/R112	RCIC Pump Room	4" RCIC (13)-14	0.51	R15, R112/R206	0.53	0.76
				0.51	R15, R112/R14, R113	0.53	0.76
				0.51	R15, R112/R6, R116	0.53	0.76
471	R206	El. 471' Open Floor Area	4" AS (11)-2	0.05	R206/R103, R105, R106, R305, R308 R310, R306, R315	0.35	0.08
				0.05	R206/R114, R113, R112	0.35	0.08
				0.05	R206/R116, R115		
501	R308	TIP Room	4" RCIC (13)-4	0.32	R308/R305, R206, R313	0.03	0.50
501	R308	TIP Room	6" RWCU (2)-4	0.48	R308/R305, R206, R313	0.35	0.60
501	R313	El. 510' Valve Room	6" RWCU (2)-4	0.41	R313/R308, R408	0.35	0.60
522	R404	El. 522' Open Floor Area	8" CRD (12)-3	0.03	R404/R305, R504, R508	0.04	0.05

(a) Table applies to reactor building secondary containment, exclusive of the main steam tunnel, tunnel ventway, and tunnel extension.

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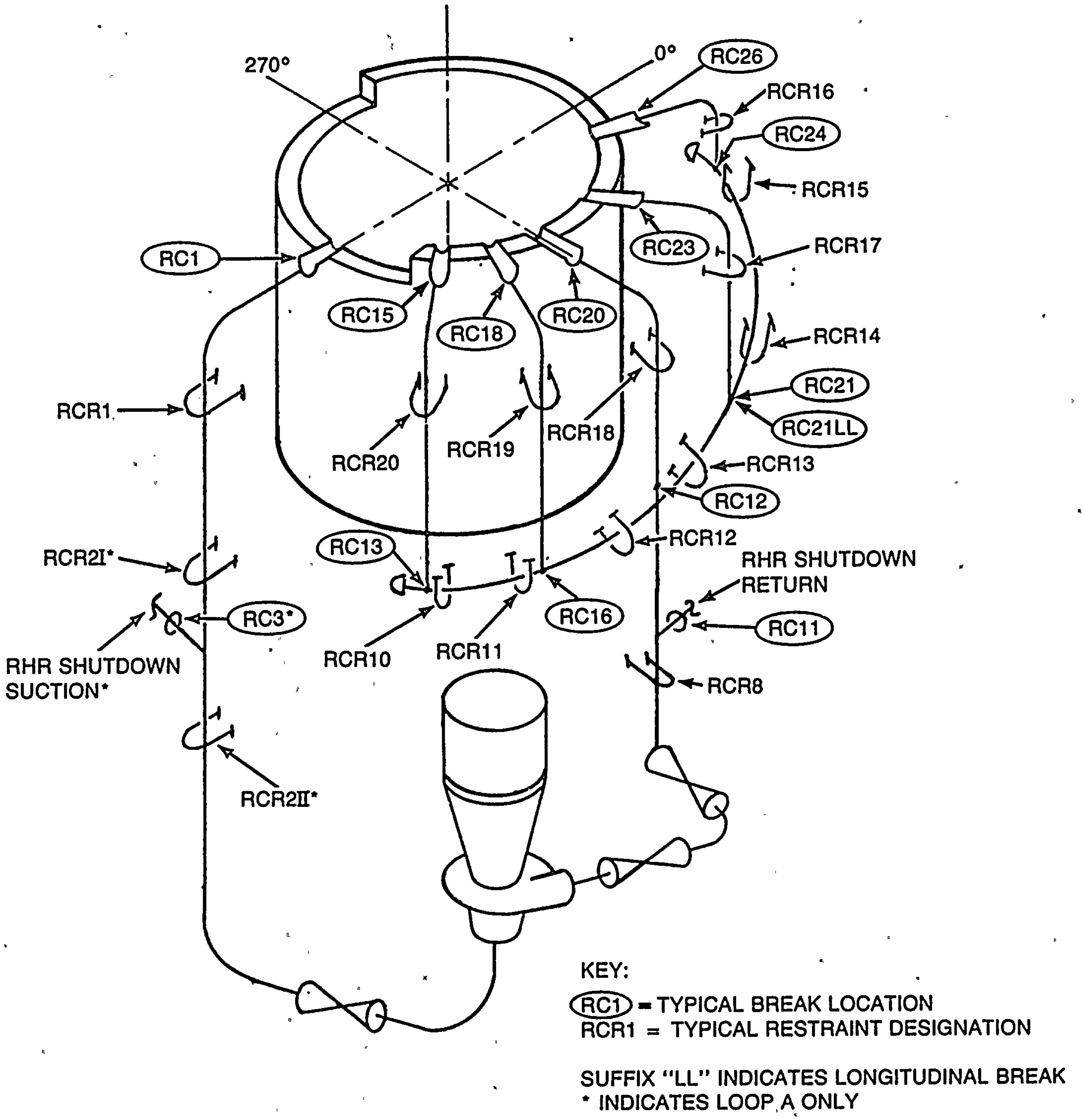

NOTES: (1) THIS FIGURE REPRESENTS LOOP A. LOOP B IS SIMILAR EXCEPT AS NOTED.
(2) SEE FIGURE 3.6-35b FOR RESTRAINT-BREAK LOCATION CORRELATION AND BREAK TYPES.
(3) ONLY THOSE RESTRAINTS THAT MAY ACT DURING THE POSTULATED BREAKS ARE SHOWN.

WASHINGTON PUBLIC POWER **SUPPLY SYSTEM** NUCLEAR PROJECT NO. 2	REACTOR RECIRCULATION PIPING SYSTEM	FIGURE 3.6-35a

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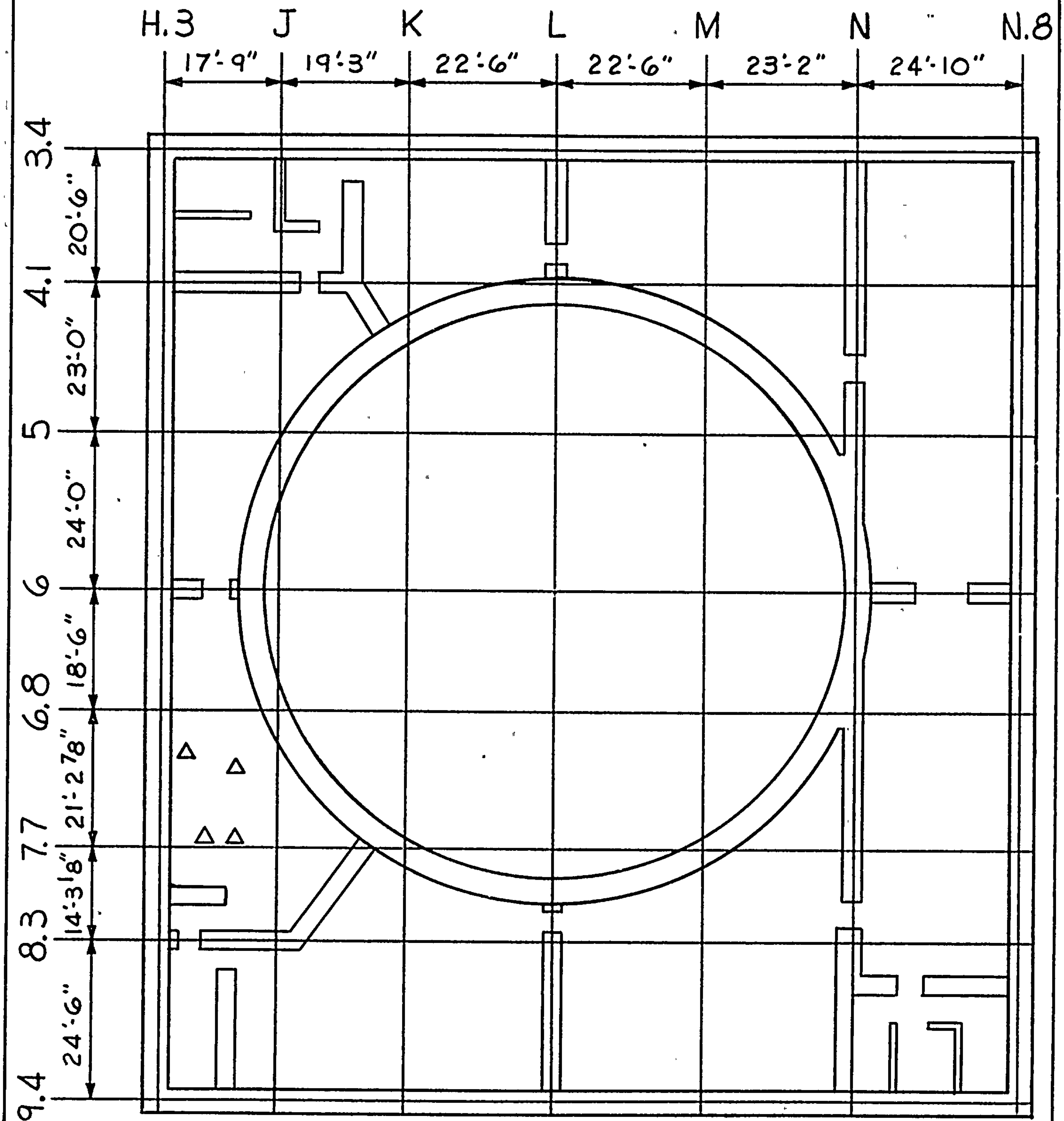

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH ENERGY FLUID PIPING SYS. RUPTURE LOC. PLAN @ EL. 422'-3"	FIGURE 3.6-41a

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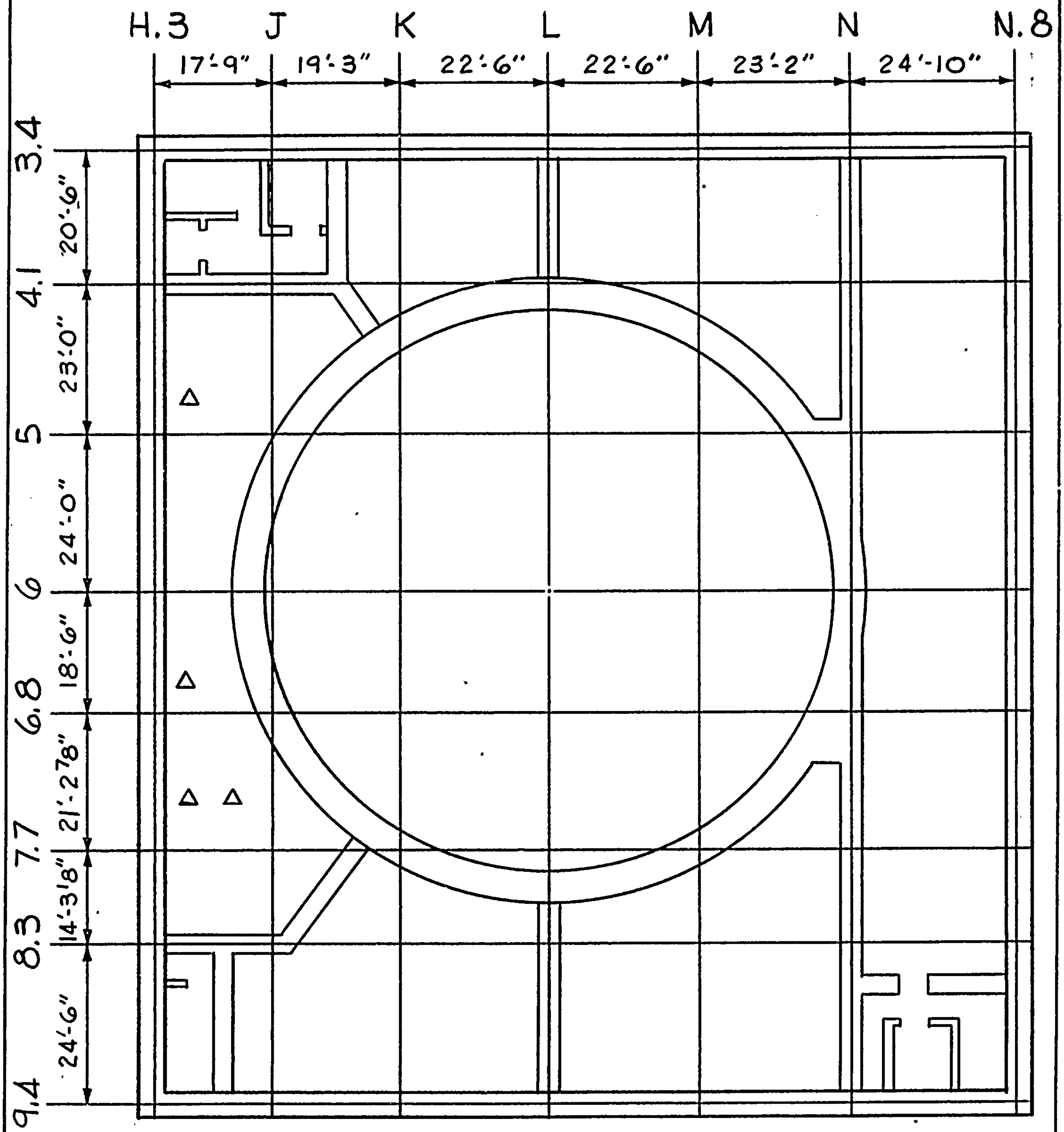

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH ENERGY FLUID PIPING SYS. RUPTURE LOC. PLAN @ EL. 441'	FIGURE 3.6-41b

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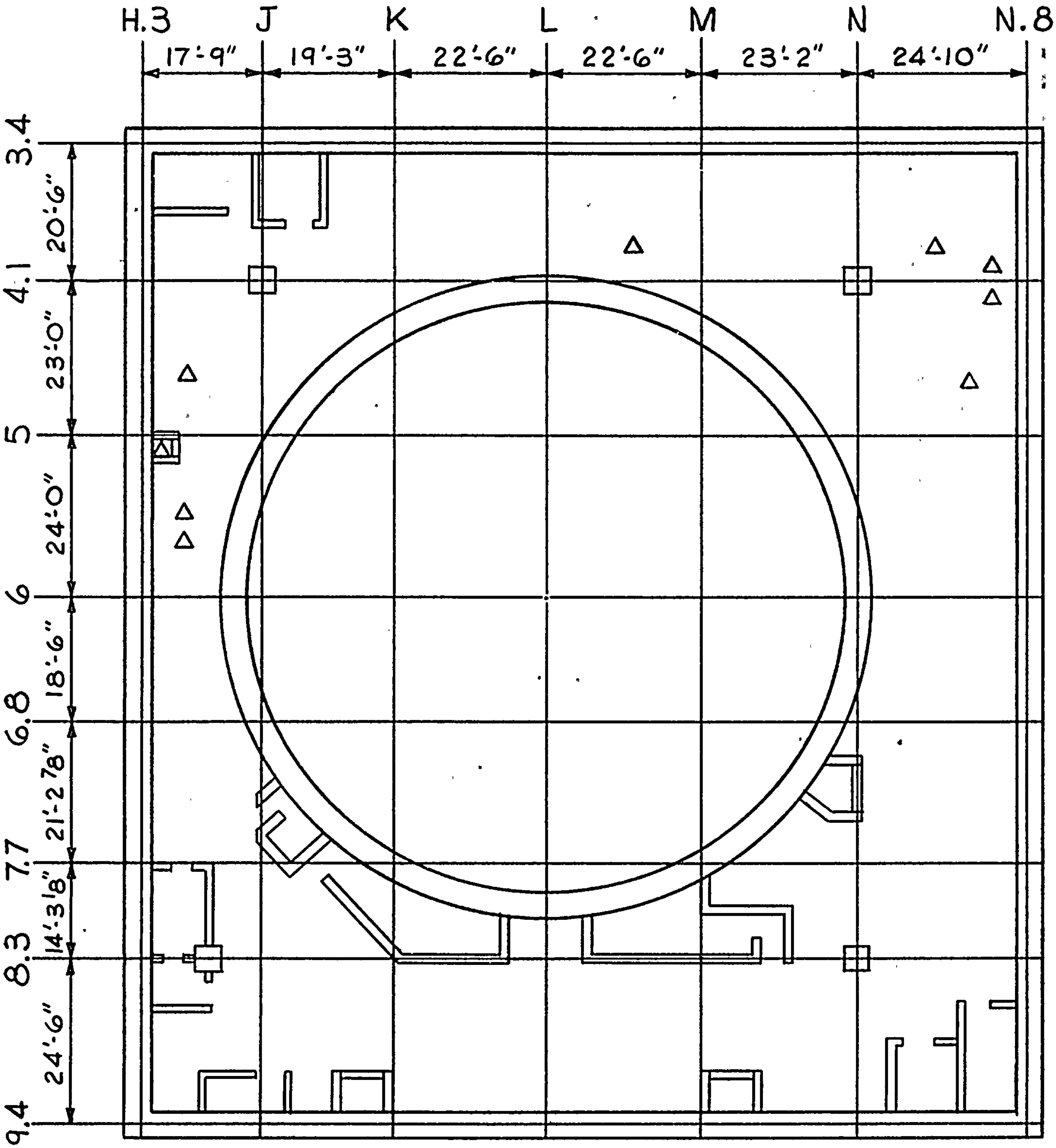
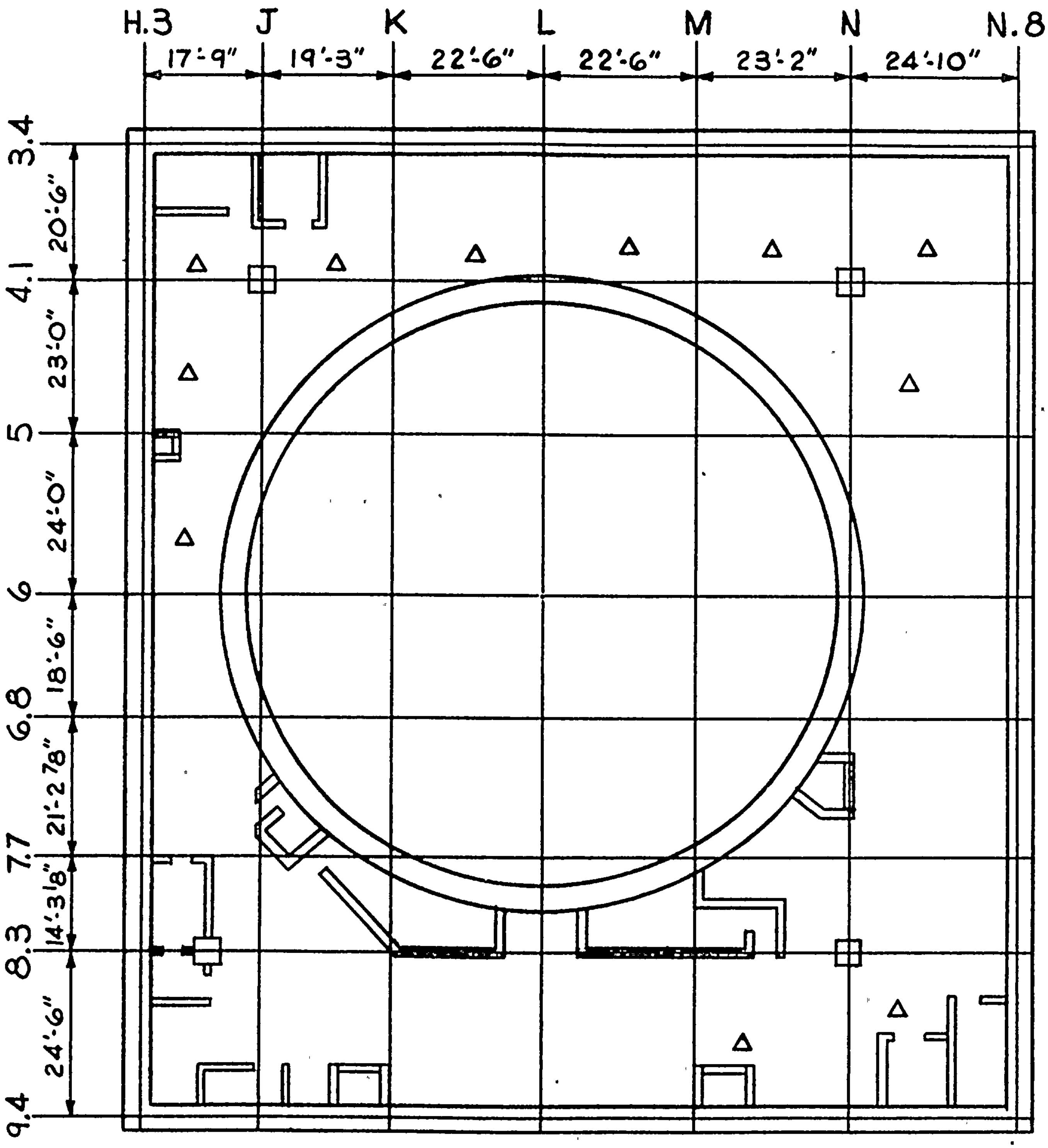
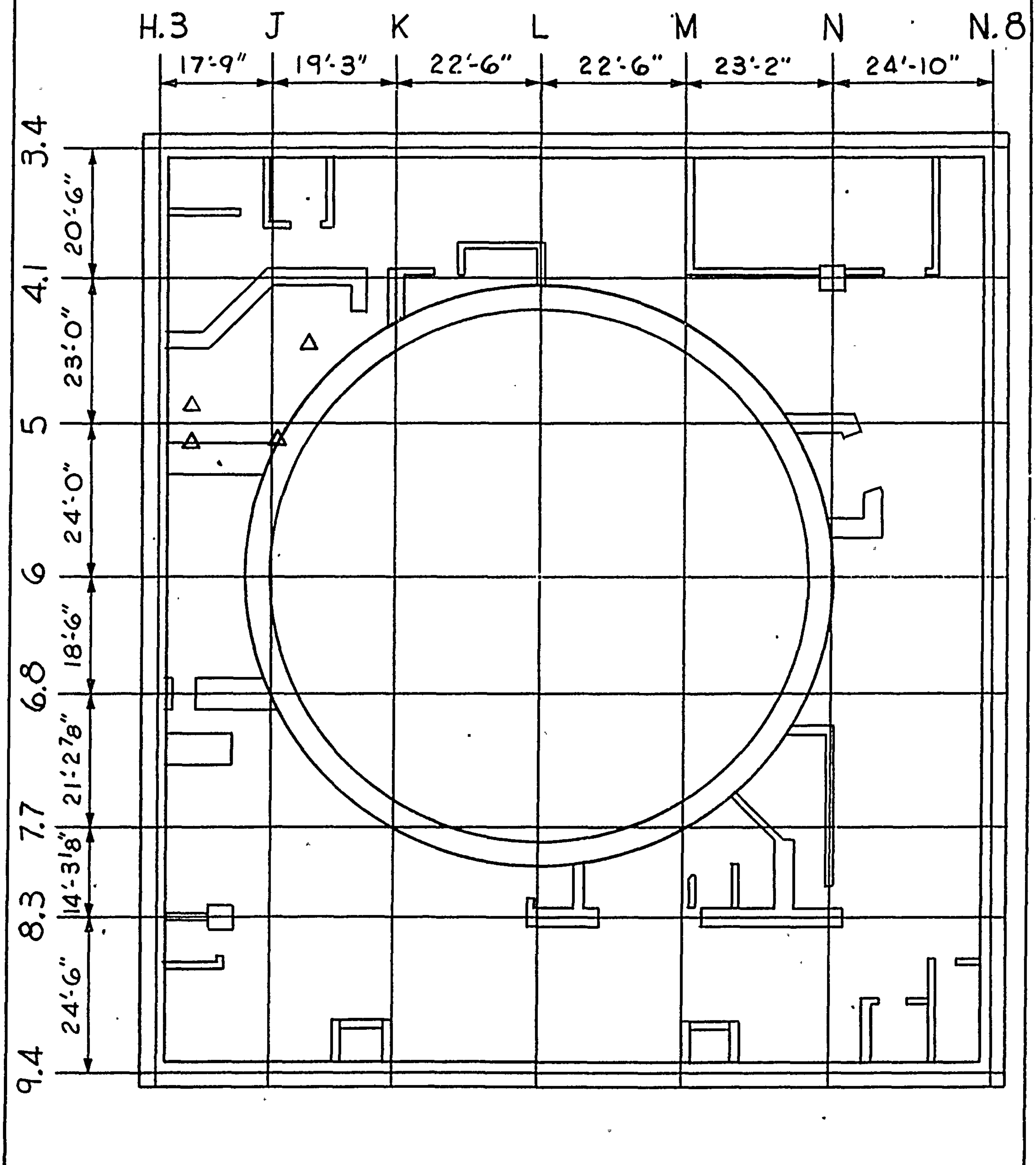


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH ENERGY FLUID PIPING SYSTEM RUPTURE LOC. PLAN @ EL. 471'	FIGURE 3.6-41c



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH ENERGY FLUID PIPING SYSTEM RUPTURE LOC. PLAN @ EL. 471'	FIGURE 3.6-41c

AMENDMENT NO. 25
June 1982



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH ENERGY FLUID PIPING SYS. RUPTURE LOC. PLAN @ EL. 501'	FIGURE 3.6-41d

AMENDMENT NO. 25
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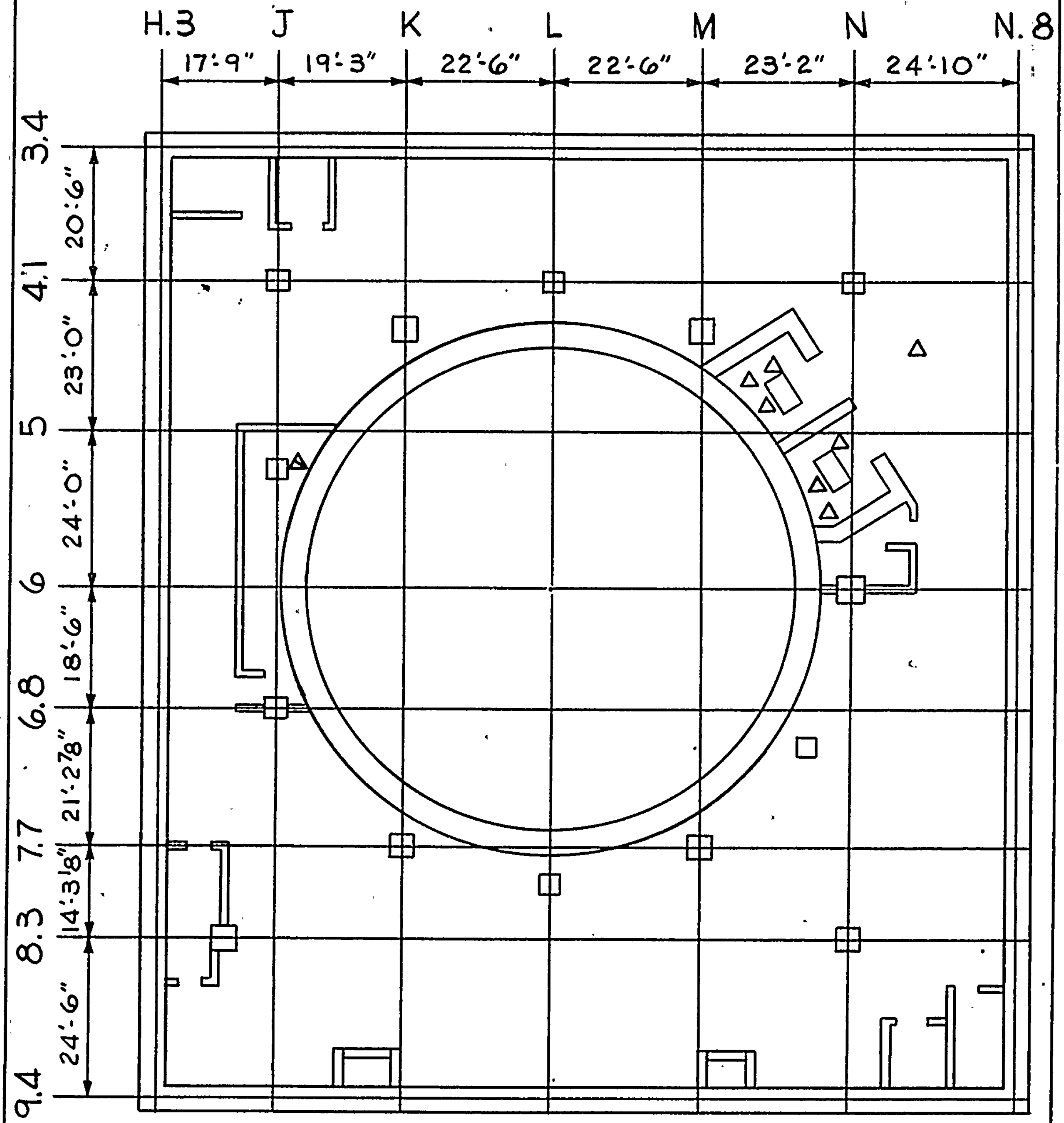

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH ENERGY FLUID PIPING SYSTEM RUPTURE LOC. PLAN @ EL. 522'	FIGURE 3.6-41e

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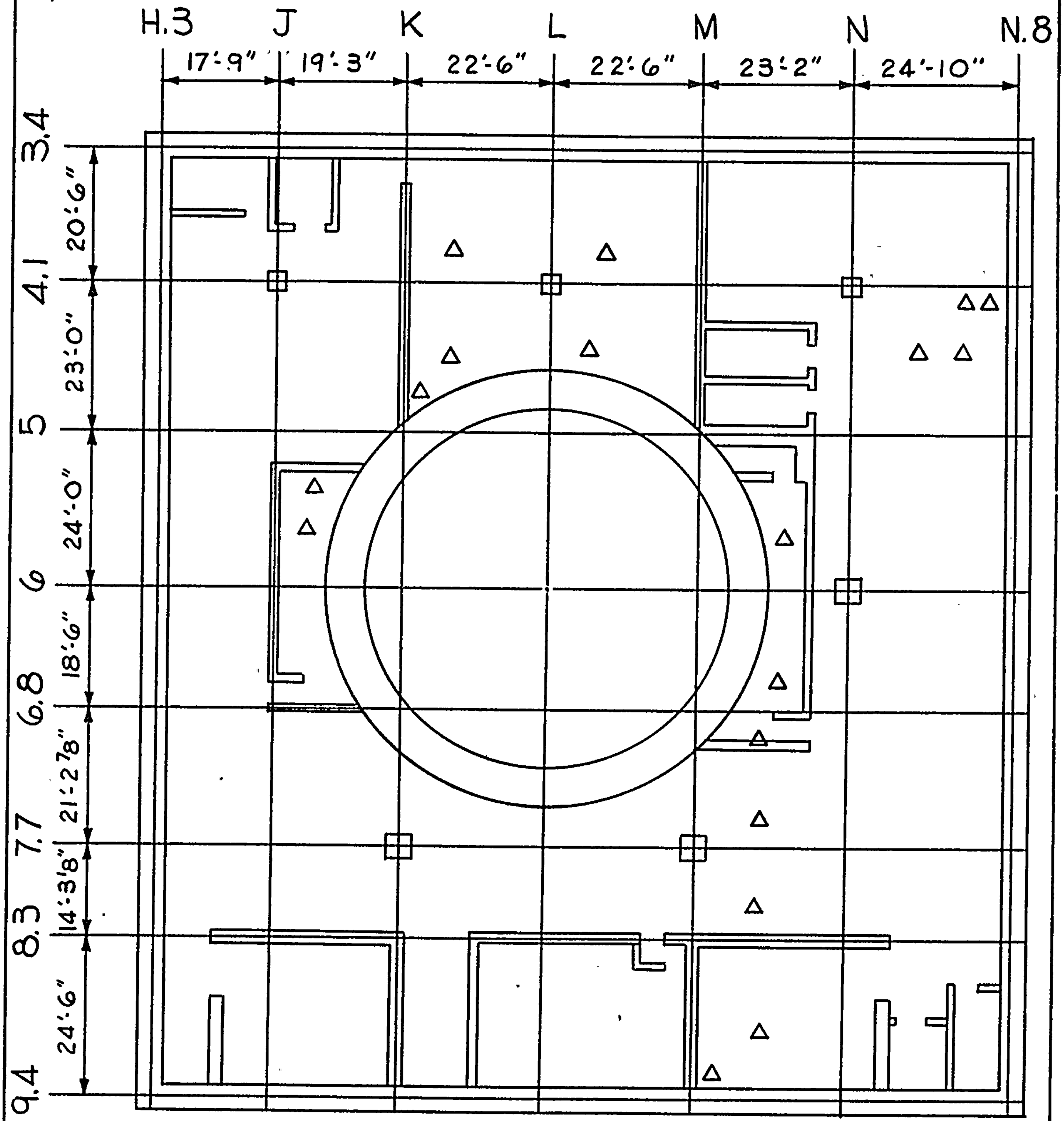
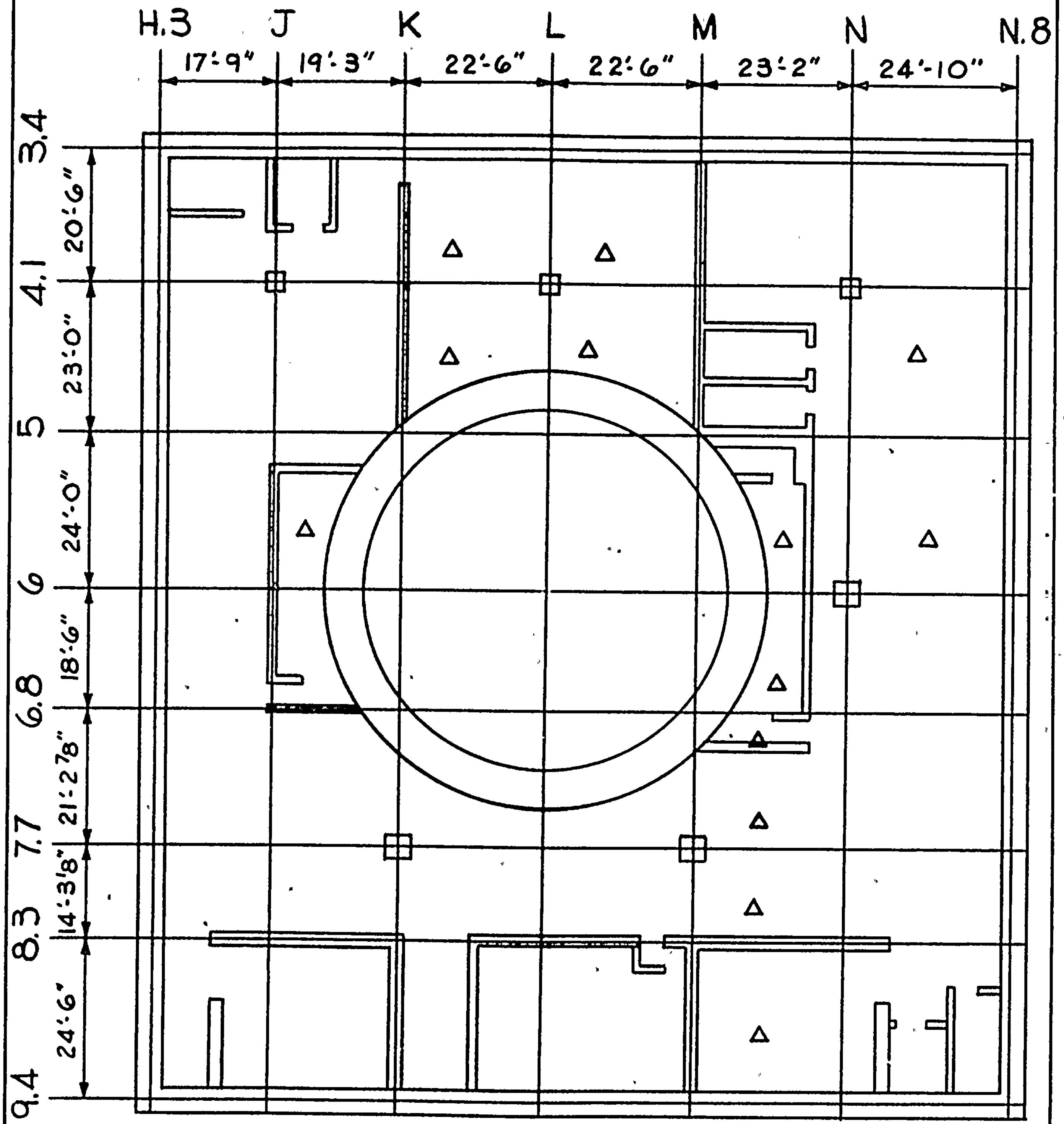
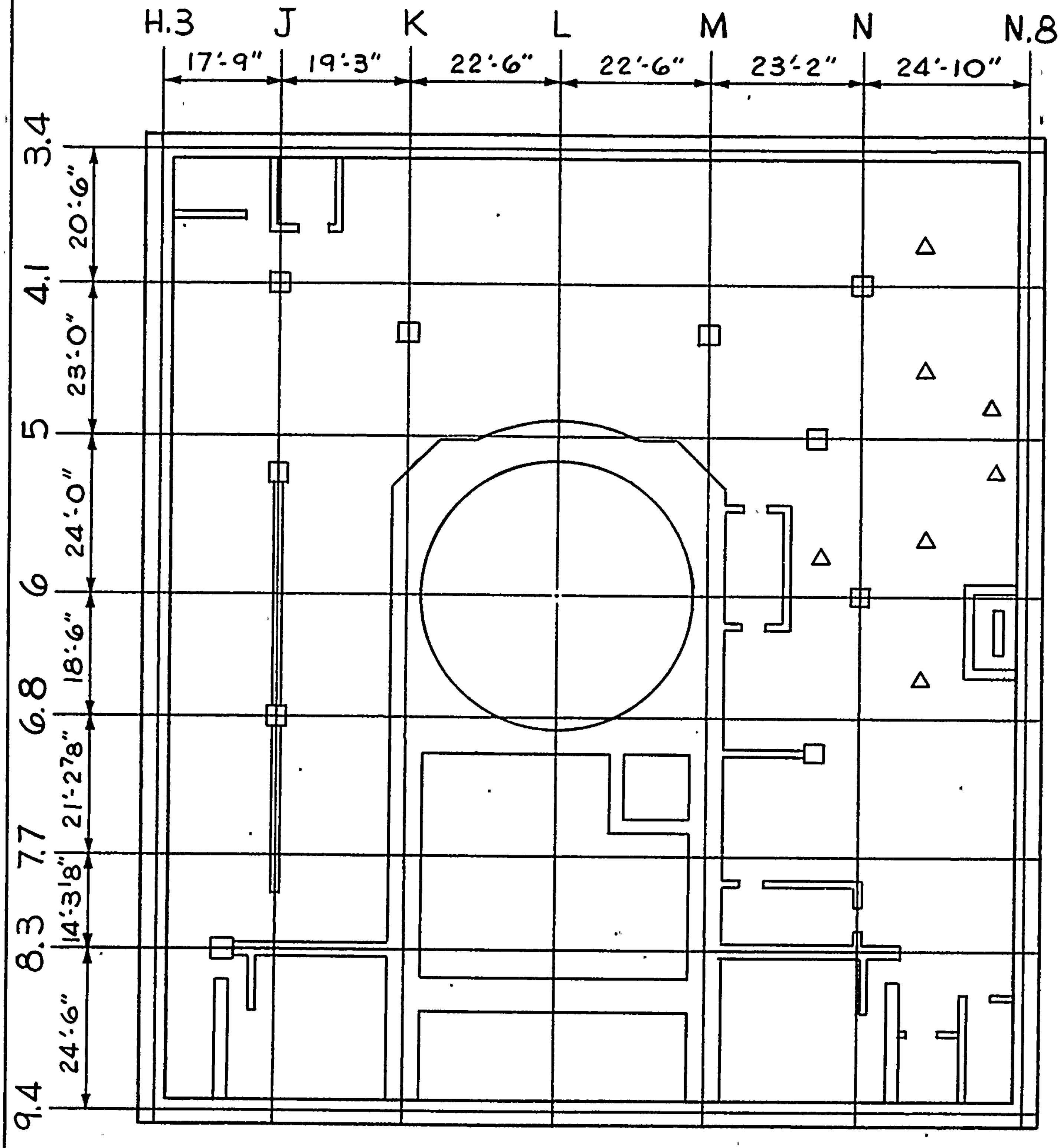


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH ENERGY FLUID PIPING SYS. RUPTURE LOC. PLAN @ EL. 548'	FIGURE 3.6-41f

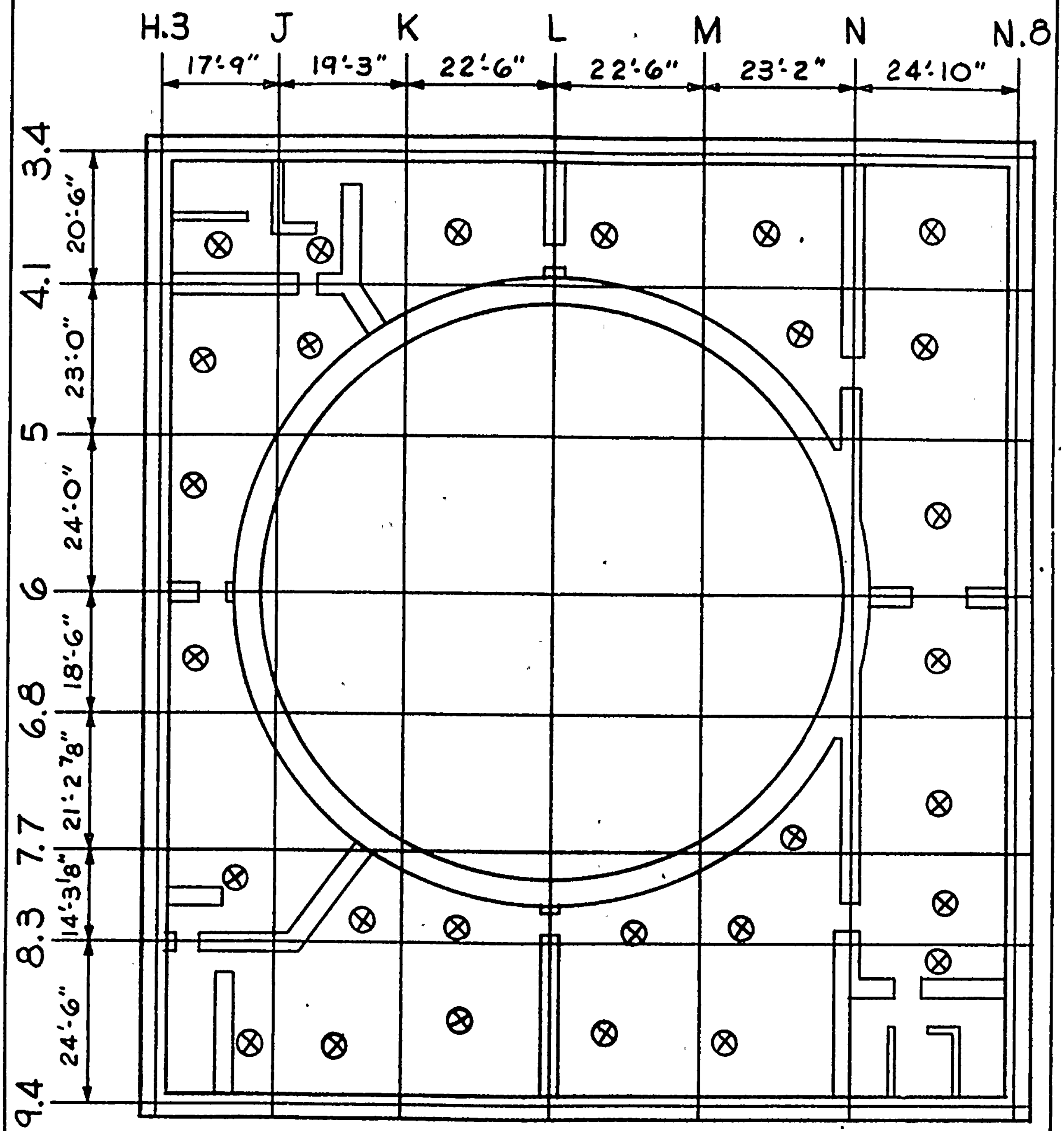


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH ENERGY FLUID PIPING SYS. RUPTURE LOC. PLAN @ EL. 548'	FIGURE 3.6-41f

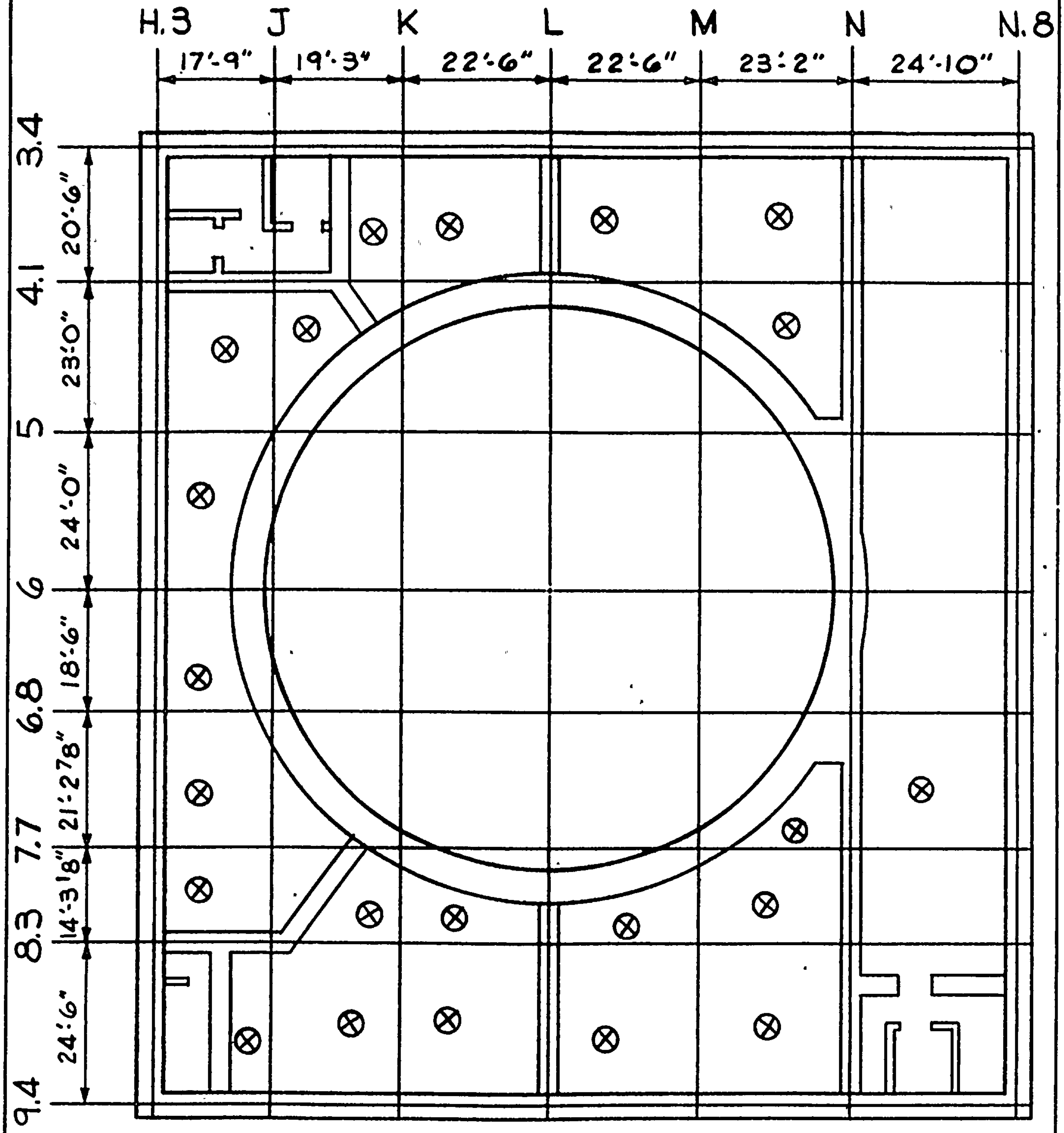
AMENDMENT NO. 25
June 1982



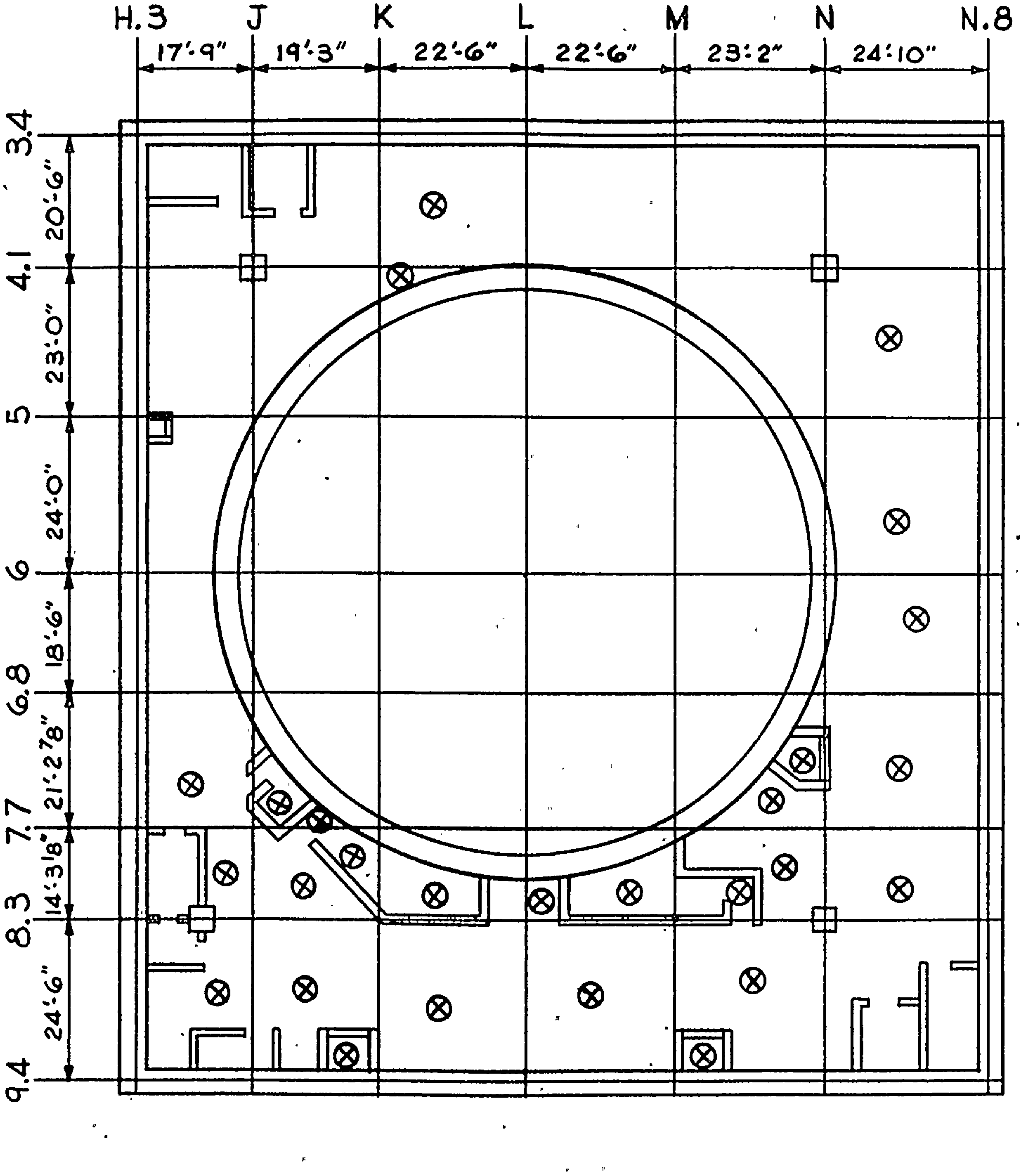
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH ENERGY FLUID PIPING SYSTEM RUPTURE LOC PLAN @ EL. 572'	FIGURE 3.6-41g



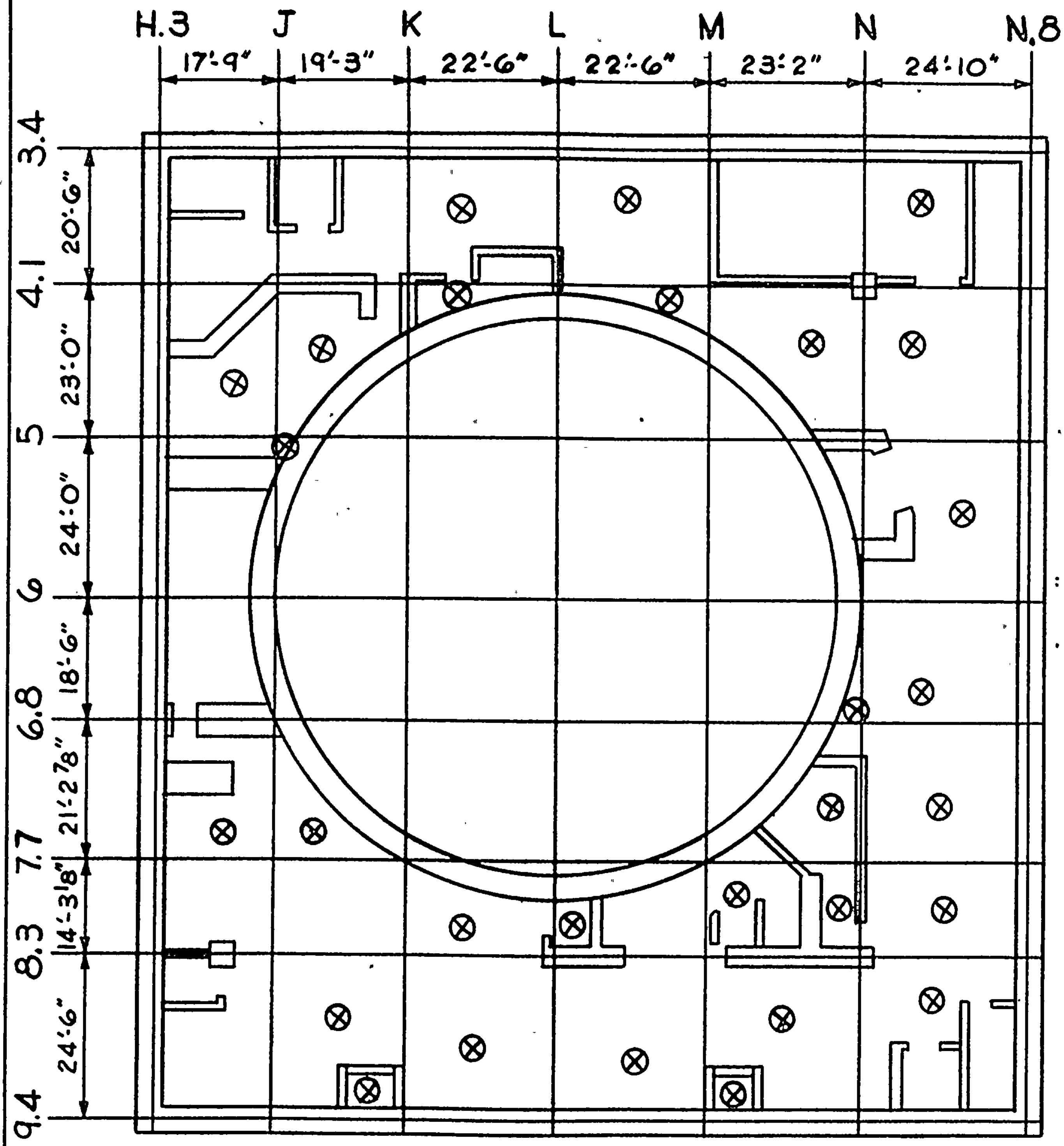
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MODERATE ENERGY FLUID PIPING SYSTEM RUPTURE LOC. PLAN @ EL. 422'-3"	FIGURE 3.6-42a



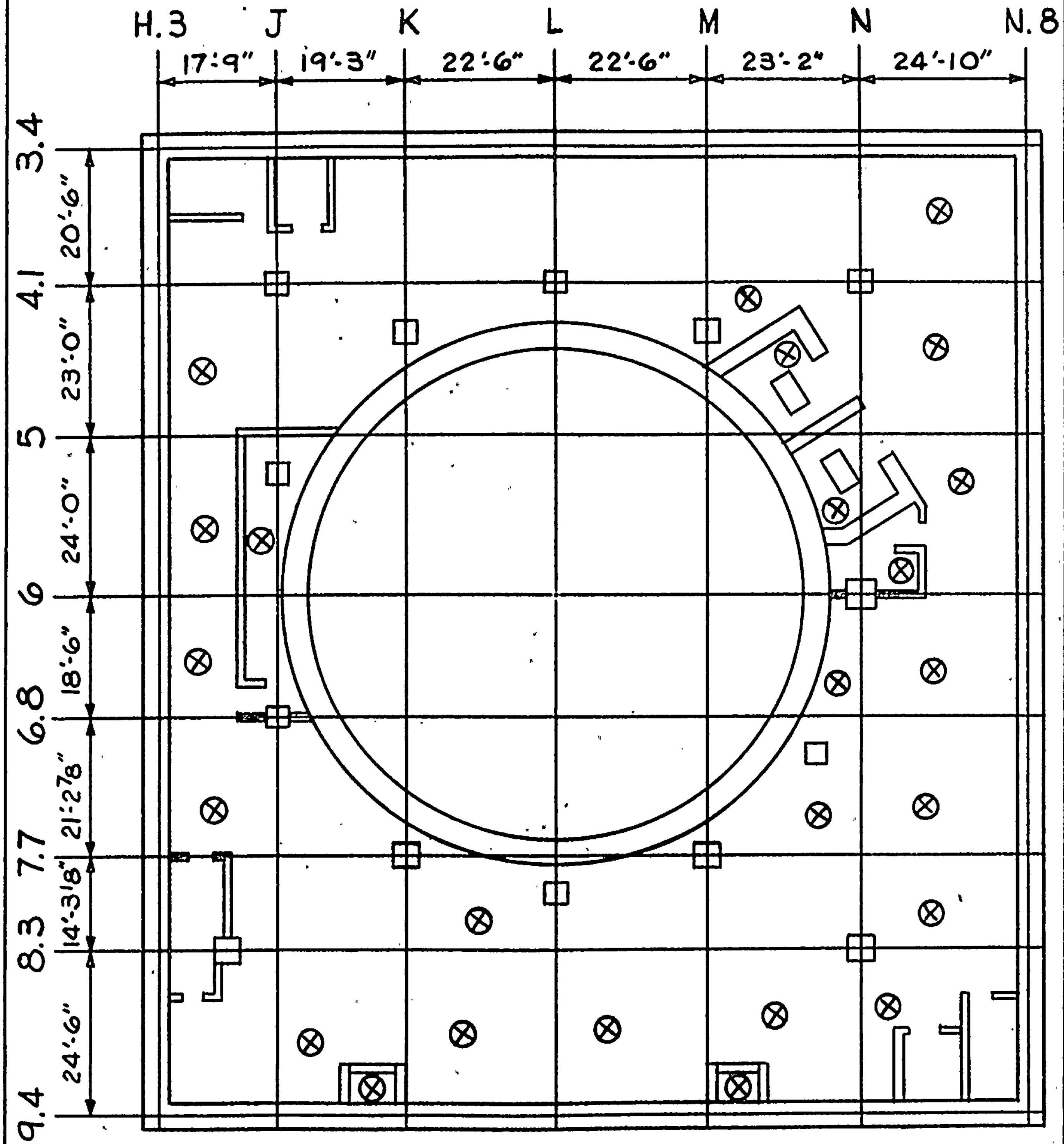
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MODERATE ENERGY FLUID PIPING SYSTEM RUPTURE LOC. PLAN @ EL. 441'	FIGURE 3.6-42b



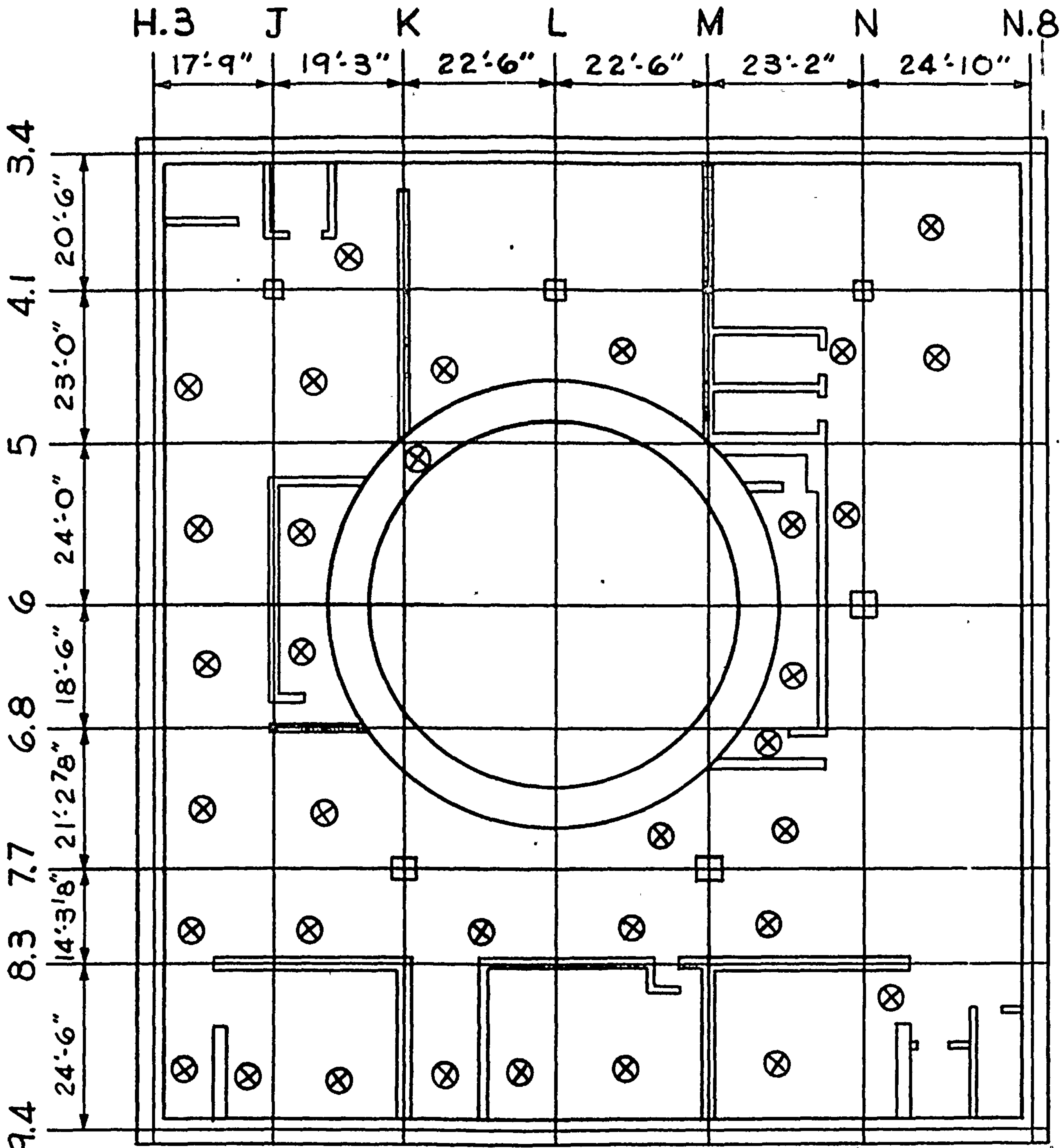
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MODERATE ENERGY FLUID PIPING SYSTEM RUPTURE LOC. PLAN @ EL. 471'	FIGURE 3.6-42c



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MODERATE ENERGY FLUID PIPING SYSTEM RUPTURE LOC. PLAN @ EL. 501'	FIGURE 3.6-42d



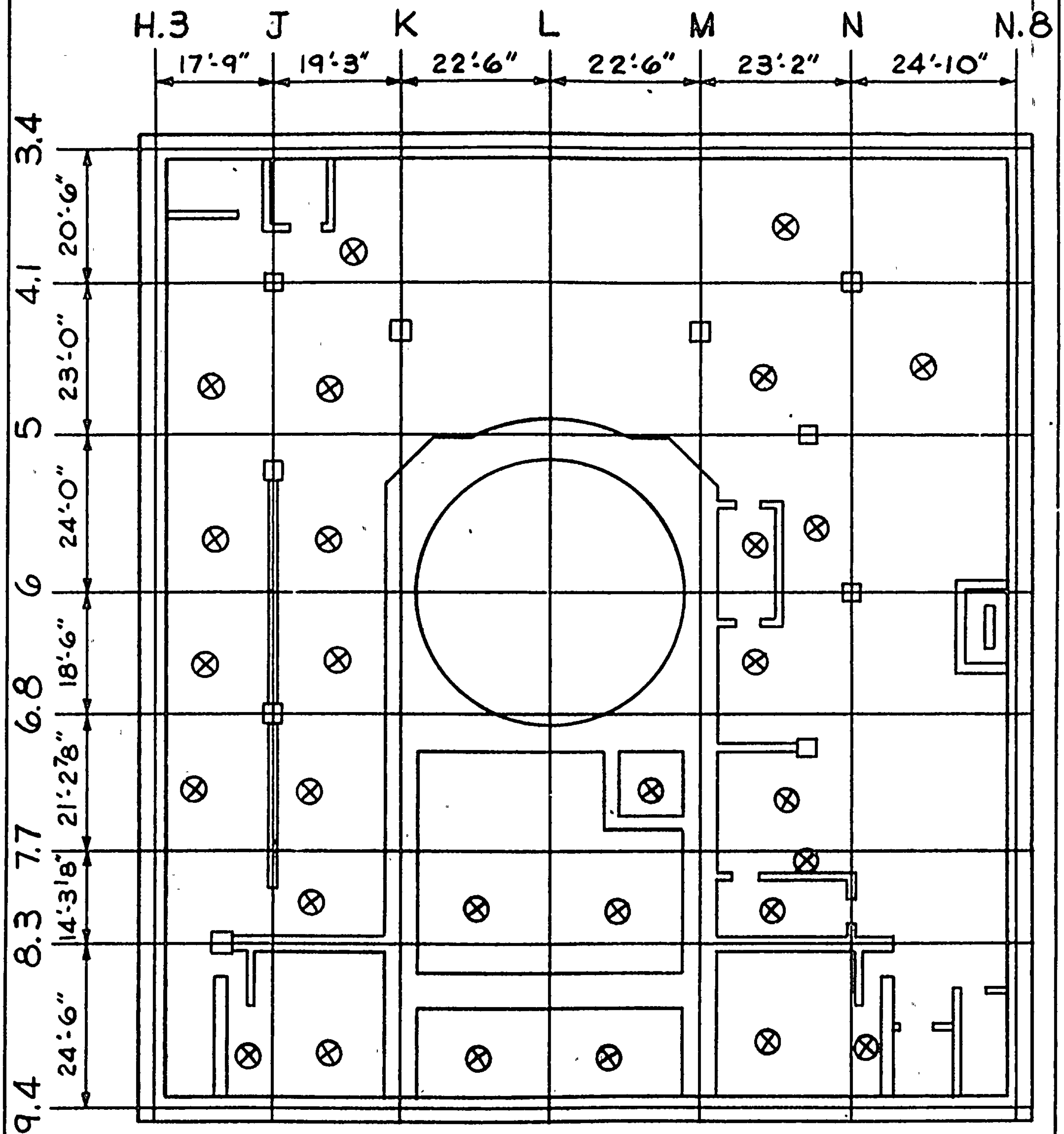
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MODERATE ENERGY FLUID PIPING SYSTEM RUPTURE LOC. PLAN @ EL. 522'	FIGURE 3.6-42e



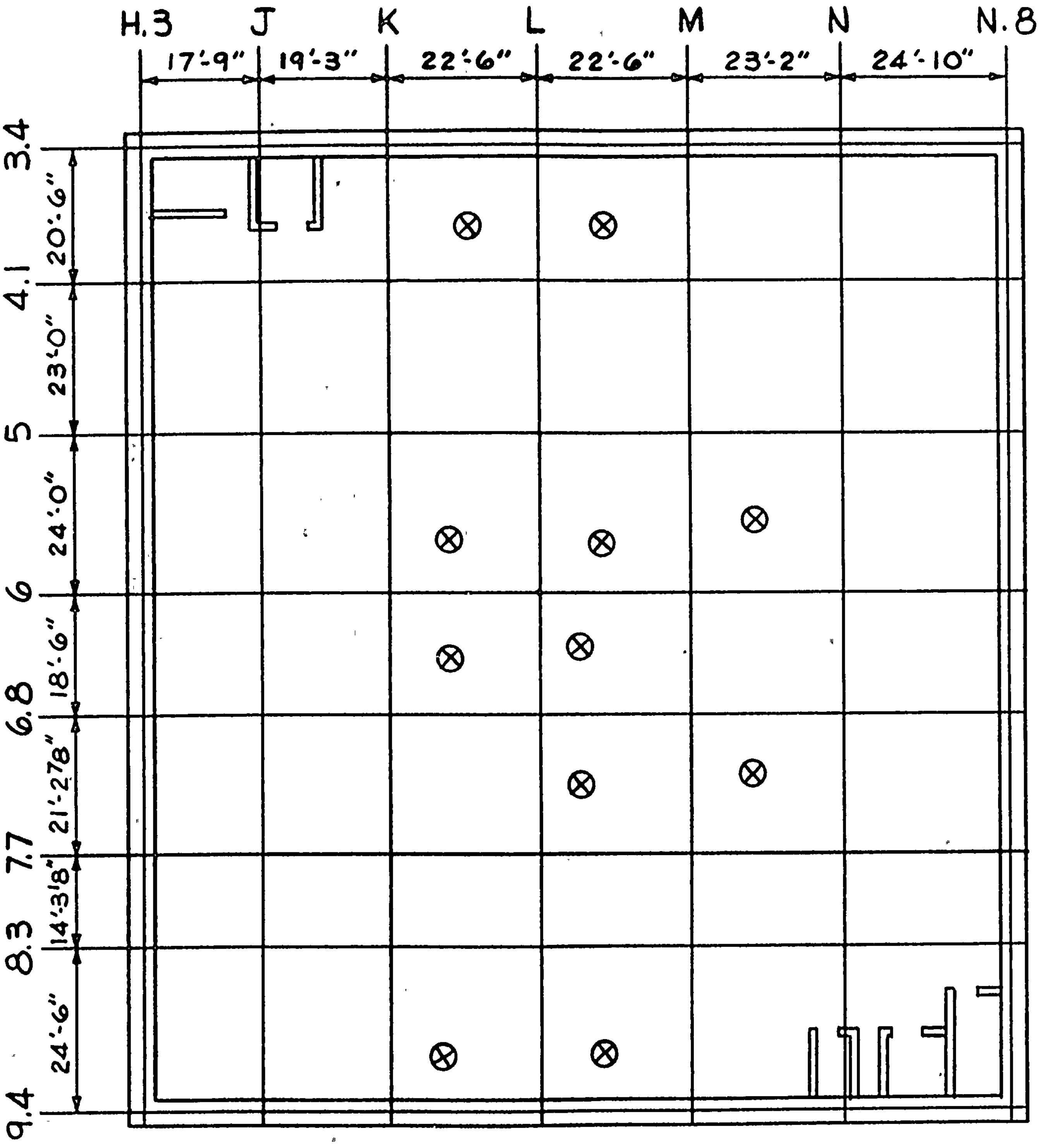
WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

MODERATE ENERGY FLUID PIPING SYSTEM
RUPTURE LOC. PLAN @ EL. 548'

FIGURE
3.6-42f



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MODERATE ENERGY FLUID PIPING SYSTEM RUPTURE LOC. PLAN @ EL. 572'	FIGURE 3.6-42g



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MODERATE ENERGY FLUID PIPING SYSTEM RUPTURE LOC. PLAN @ EL. 606'-10 1/2"	FIGURE 3.6-42h

d. The duration of the test was 10 seconds.

Upon completion of the above abnormal environmental transient test, the snubber was tested dynamically at a frequency within a specified frequency range. The snubber did operate normally during the dynamic test.

d. Rigid Supports

The design load on rigid supports includes those loads caused by dead weight, thermal expansion, primary or secondary forces, i.e., operating basis earthquake (OBE) and safe shutdown earthquake (SSE), system anchor displacements, and reaction forces caused by relief valve discharge, turbine stop valve closure, etc.

Rigid supports are designed in accordance with NF-3000 to be capable of carrying the design load for all operating conditions.

e. Bolting for Piping Supports and Pipe Mounted Equipment (Valves and Pump) Supports

The supports are hanger and snubber type (including clamps) linear standard components as defined by the ASME Code Section III, Subsection NF. The bolts used in these supports meet criteria of NF-3280 for Service Levels A and B and NF-3230 for Service Levels C and D. NF-3280 is applicable to bolting for Service Levels A and B.

For Service Levels C and D, XVII-2460 with factors indicated under XVII-2110 is applicable to the design requirements of bolting. The calculated stresses under these categories do not exceed the specified minimum yield stresses at temperature.

f. Operability Assurance of Snubbers

There are no hydraulic snubbers installed on safety-related systems at WNP-2. Mechanical snubbers are used exclusively. Because of normal construction problems (interferences, etc.), the list of snubbers is not yet complete. When the list is complete, a list of all

safety-related snubbers will be included in Section 7 (Plant Systems) of the Technical Specification.

3.9.3.4.2 Reactor Pressure Vessel Support Skirt

The RPV support skirt is designed as an ASME Code Class 1 plate and shell type component support per the requirements of ASME Boiler and Pressure Vessel Code Section III, Subsection NF. The loading conditions, stress criteria, calculated stresses and the allowable stresses in the critical support areas for various plant operating conditions are summarized in Table 3.9-2a.

3.9.3.4.3 NSSS Floor Mounted Equipment (Pumps, Heat Exchanger, and RCIC Turbine)

The RHR pump, high pressure core spray, low pressure core spray, RHR heat exchanger, reactor core isolation cooling, standby liquid control, and RCIC Turbine are all analyzed to verify the adequacy of their support structure under various plant operating conditions. In all cases, the stress loads in the critical support areas are within ASME Code allowables. The loading conditions, stress criteria, and allowable stresses in the critical support areas are summarized in Tables 3.9-2b, 3.9-2n, 3.9-2o, 3.9-2p, 3.9-2q, and 3.9-2r.

3.9.3.4.4 Supports for ASME Code Class 1, 2, and 3 Active Components

ASME Code Class 1, 2, and 3 active components are either pumps or valves. Since valves are supported by piping and not tied to building structures, pipe design criteria govern.

Seismic Category I active pumps supports are qualified for seismic and hydrodynamic loads by testing when the pump supports along with the pumps are fulfilling the following conditions:

a. Simulate actual mounting conditions;

b. Simulate all static and dynamic loadings on the pump;

c. Monitor pump operability during testing;

3.9.5.4 Design Bases

3.9.5.4.1 Safety Design Bases

The reactor core support structures and internals shall meet the following safety design bases:

a. Arrangement provides a floodable volume in which the core can be adequately cooled in the event of a breach in the nuclear system process barrier external to the reactor vessel.

b. Deformation is limited to assure that the control rods and core standby cooling systems can perform their safety functions.

c. Mechanical design of applicable structures assures that safety design bases a and b, above, are satisfied so that the safe shutdown of the plant and removal of decay heat are not impaired.

3.9.5.4.2 Power Generation Design Bases

The reactor core support structures and internals are designed to the following power generation design bases:

a. They provide the proper coolant distribution during all anticipated normal operating conditions up to full power operation of the core without fuel damage.

b. They are arranged to facilitate refueling operations.

c. They are designed to facilitate inspection.

3.9.5.4.3 Response of Internals Due to Inside Steam Break Accident

The maximum pressure loads acting on the reactor internal components result from an inside steam line break, and on some components the loads are greatest with operation at the minimum power associated with the maximum core flow. This has been substantiated by the analytical comparision of liquid versus steam breaks and by the investigation of the effects of core power and core flow.

It has also been pointed out that it is possible, but not probable that the reactor would be operating at the rather abnormal condition of minimum power and maximum core flow. More realistically, the reactor would be at or near a full power condition and thus the maximum pressure loads acting on the internal components would be less.

3.9.5.4.4 Stress, Deformation, and Fatigue Limits for Reactor Internals (Except Core Support Structure)

These limits are summarized in Table 3.9-2b.

Design Condition	SF_{min}
Normal	2.25
Upset	2.25
Emergency	1.5
Faulted	1.125

Elastic displacement is considered in the design of reactor internal components in which deflection can affect control rod insertability. No plastic deformation occurs in any permanent core support structure components or the reactor vessel. Radiation induced deformation can occur in the fuel channel over the core life. These effects are considered in control rod insertability tests. No fatigue analysis is required under the faulted conditions due to the low encounter frequency of faulted events and the low number of cycles. The forcing functions applicable to the reactor internals are discussed in 3.9.2.5.

3.9.5.4.5 Stress, Deformation, and Fatigue Limits for Core Support Structures

These limits are summarized in Tables 3.9-2a, 3.9-2b, and 3.9-2aa.

3.9.6 INSERVICE TESTING OF PUMPS AND VALVES

This section addresses the program for inservice testing for operational readiness of ASME Section III Class 1, 2, and 3 pumps and valves. The detailed program indicating valves and pumps to be tested and the test specifics was submitted to the NRC October 1, 1981 via transmittal letter GO2-81-322. Revision 1 to the program submitted October 7, 1982, incorporated the requirements contained in the 1980 Edition of the ASME Boiler and Pressure Vessel Code, Section XI, Subsections IWP and IWV with addenda through Winter 1980. Where compliance with certain Code requirements specified in 10CFR50.55a(g) are found to be impractical for WNP-2, requests

APPENDIX 3.11A

ENVIRONMENTAL QUALIFICATION OF SELECTED CLASS 1E EQUIPMENT

APPENDIX 3.11A

TABLE OF CONTENTS

APPENDIX 3.11A

LIST OF TABLES

APPENDIX 3.11A

3.11A ENVIRONMENTAL QUALIFICATION OF SELECTED CLASS 1E EQUIPMENT

3.11A.1 INTRODUCTION

Presented in Appendix 3.11A is environmental qualification data for selected Class 1E electrical equipment furnished for installation at WNP-2. The particular items discussed herein are as follows:

a. 4.16 kV Switchgear SM-7

b. 480 V motor control center MC-7A-A

c. DG control equipment

d. Cable spreading room cooling fan WMA-FN-52B

e. Cable spreading room/main control room cooling system intake air damper WOA-V-52C

f. SGTS logic equipment

g. MSIV solenoid valves

3.11A.2 QUALIFICATION STANDARDS

The applicable standards relating to environmental qualification of the Class 1E items indicated in 3.11A.1, based upon the dates of equipment purchase, are as follows:

a. IEEE Std 308-1971 - "Class 1E Electric Equipment for Nuclear Power Generating Stations"

b. IEEE Std. 323-1971 - "Qualifying Class 1E Electric Equipment for Nuclear Power Generating Stations"

IEEE Std. 308-1971 indicates the base requirements for environmental qualification of Class 1E equipment; IEEE Std. 323-1971 details the guidelines for compliance to that standard. The particular environmental parameters of concern are temperature, pressure, humidity, gas composition and

radiation (seismic vibration is covered in 3.10). All Class 1E equipment is required to demonstrate the capability to meet or exceed its performance specifications under all normal and postulated accident environments over the entire service life of the equipment.

3.11A.3 SPECIFICATION REQUIREMENTS

Table 3.11A-1 indicates the specifications governing the purchase of equipment indicated in 3.11A.1, and includes a listing of any special environmental requirements noted in the documents.

3.11A.4 DOCUMENTATION

With the exception of the MSIV solenoid valves, all equipment is located in plant areas outside of both the primary and secondary containments.

(Discussion of documentation relating to equipment qualification tests to be provided later.)

The solenoid valves for the MSIV's are not required to be Class 1E equipment because the results of an electrical failure do not prevent the MSIV's from closing to the safe position. Therefore, the Class 1E qualification does not apply for these solenoid valves.

TABLE 3.11A-1

SPECIFICATION REQUIREMENTS

Equipment	Location		Service	Purchase Spec.	Specification Requirements	
	Bldg.	Elev.			IEEE Conformance	Special
4.16kV SM-7	Radwaste and Control	467'	BOP	2808-47A	308-1971 323-1971	1) Max. Ambient - $40^{\circ}C$ 2) Maintain dielectric strength in high humidity and temp.
480V MC-7A-A	Diesel Generator	441'	BOP	2808-49	308-1971 323-1971	1) Max. Ambient - $40^{\circ}C$
DG1, DG2 Control Equip.	Diesel Generator	441'	BOP	2808-53	308-1971 323-1971	1) Max. Ambient - $40^{\circ}C$ (elec. equip. areas) - $49^{\circ}C$ (eng. exh. areas)
DG3 Control Equip.	Diesel Generator	441'	NSSS	2808-2	308-1971	None
WMA-FN-52B	Radwaste and Control	525'	BOP	2808-67	308-1971 323-1971	1) Max. Ambient - $40^{\circ}C$
WOA-V-52C	Radwaste and Control	525'	BOP	2808-216	323-1971	None
SGTS Logic	Radwaste and Control	501'	BOP	2808-218	308-1971 323-1971	1) Max Ambient - $40^{\circ}C$
SGTS Logic	Radwaste and Control	501'	NSSS	2808-2	See Tables 7.3-5 and 7.3-7	
MSIV Solenoid Valves	Reactor Building	501'	NSSS	2808-2	None	1) Max. Ambient - $5.75^{\circ}F$ 2) Max. Pressure - 1250 Psig 3) Max. RH - 100% 4) Incident Rad. - continuous 5) Gamma Rad - 15R/hr 6) Gamma & Neutron Rad - 25R/hr.

e. The General Electric thermal analysis basis, GETAB, is applied to assure that more than 99.9% of the fuel rods in the core are expected to avoid boiling transition for the most severe abnormal operational transient described in Chapter 15. The possibility of boiling transition occurring during normal reactor operation is insignificant.

f. Because of the large negative moderator density coefficient of reactivity, the BWR has a number of inherent advantages. These are the uses of coolant flow for load following, the inherent self-flattening of the radial power distribution, the ease of control, the spatial xenon stability, and the ability to override xenon, in order to follow load.

Boiling water reactors do not have instability problems due to xenon. This has been demonstrated by special tests which have been conducted on operating BWRs in an attempt to force the reactor into xenon instability, and by calculations. No xenon instabilities have ever been observed in the test results. All of these indicators have proven that xenon transients are highly damped in a BWR due to the large negative power coefficient of reactivity (Reference 4.1-1).

Important features of the reactor core arrangement are as follows:

a. The bottom-entry cruciform control rods consist of B_4C in stainless steel tubes surrounded by a stainless steel sheath.

   Rods of this design have been irradiated for more than eight years in the Dresden-1 reactor and have accumulated thousands of hours of service without significant failure in operating BWRs.

b. The fixed in-core ion chambers provide continuous power range neutron flux monitoring. A probe tube in each in-core assembly provides for a traversing ion chamber for calibration and axial detail. Source and intermediate range monitors are located in-core and are axially retractable.

The in-core location of the startup and source range instruments provides coverage of the large reactor core and provides an acceptable signal-to-noise ration and neutron-to-gamma ration. All in-core instrument leads enter from the bottom and the instruments are in service during refueling. In-core instrumentation is further discussed in 7.6.1.5 and 7.7.1.6.

c. As shown by experience obtained at Dresden-1 and other plants, the operator, utilizing the in-core flux monitor system, can maintain the desired power distribution within a large core by proper control rod scheduling.

d. The Zircaloy-4 reusable channels provide a fixed flow path for the boiling coolant, serve as a guiding surface for the control rods, and protect the fuel during handling operations.

e. The mechanical reactivity control permits criticality checks during refueling and provides maximum plant safety. The core is designed to be subcritical at any time in its operating history with any one control rod fully withdrawn.

f. The selected control rod pitch represents a practical value of individual control rod reactivity worth, and allows ample clearance below the pressure vessel between control rod drive mechanisms for ease of maintenance and removal.

4.1.2.1.2 Core Configuration

The reactor core is arranged as an upright circular cylinder containing a large number of fuel cells and is located within the reactor vessel. The coolant flows upward through the core. The core arrangement (plan view) and the lattice configuration, are shown in Figure 4.2-5.

4.1.2.1.3 Fuel Assembly Description

As can be seen from the referenced figures, the boiling water reactor core is composed of essentially two components--fuel assemblies and control rods. The fuel assmebly and control rod mechanical configurations (see Figures 4.2-1, 4.2-2, and Figure 1-1 of Reference 4.1-12.) are basically the same as used in Dresden-1 and in all subsequent General Electric boiling water reactors.

4.4.7 REFERENCES

4.4-1 "General Electric Standard Application for Reactor Fuel," (NEDE-24011, latest approved revision).

TABLE 4.4-1

THERMAL AND HYDRAULIC DESIGN CHARACTERISTICS OF THE REACTOR CORE

General Operating Conditions	
Reference design thermal output, Mwt	3,323
Power level for engineered safety features, Mwt	3,489
Steam flow rate, at 420°F final feedwater temperature millions lb/hr	14.30
Core coolant flow rate, millions lb/hr	108.5
Feedwater flow rate, millions lb/hr	14.26
System pressure, nominal in steam dome, psia	1020
System pressure, nominal core design, psia	1035
Coolant saturation temperature at core design pressure, °F	549
Average power density, kW/liter	49.15
Maximum Linear Heat Generation Rate kW/ft	13.4
Average Linear Heat Generation Rate kW/ft	5.4
Core total heat transfer area, ft^2	74,871
Maximum heat flux, Btu/hr-sq ft	361,500
Average heat flux, Btu/hr-sq ft	145,100
Design operating minimum critical power ratio (MCPR)	1.24 (see Table 15.0-3)
Core inlet enthalpy at 420°F FFWT, Btu/lb	527.6
Core inlet temperature, at 420°F FFWT, °F	533
Core maximum exit voids within assemblies, %	0.76
Core average void fraction, active coolant	0.418

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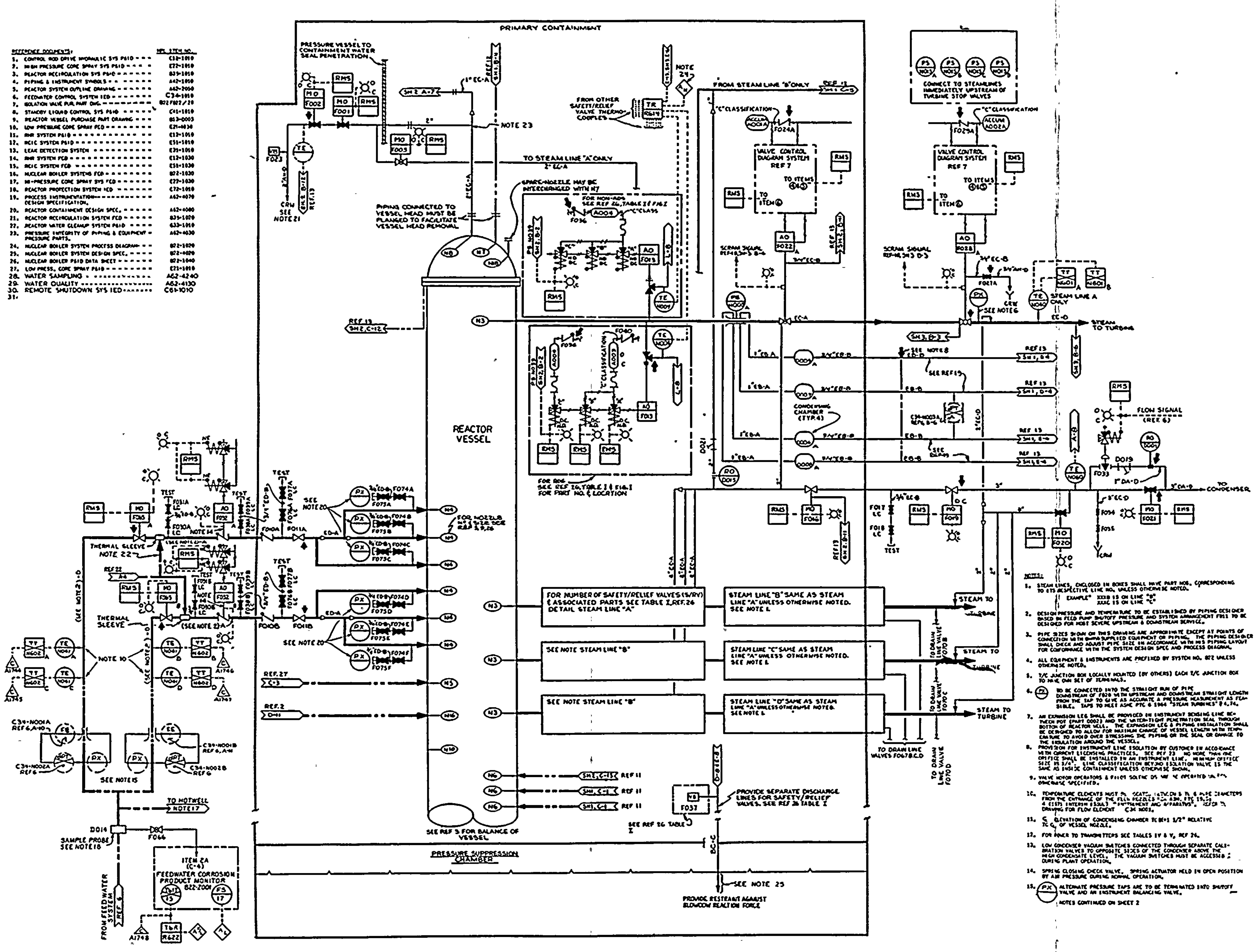

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

P & ID NUCLEAR BOILER SYSTEM

FIGURE 5.1-3a

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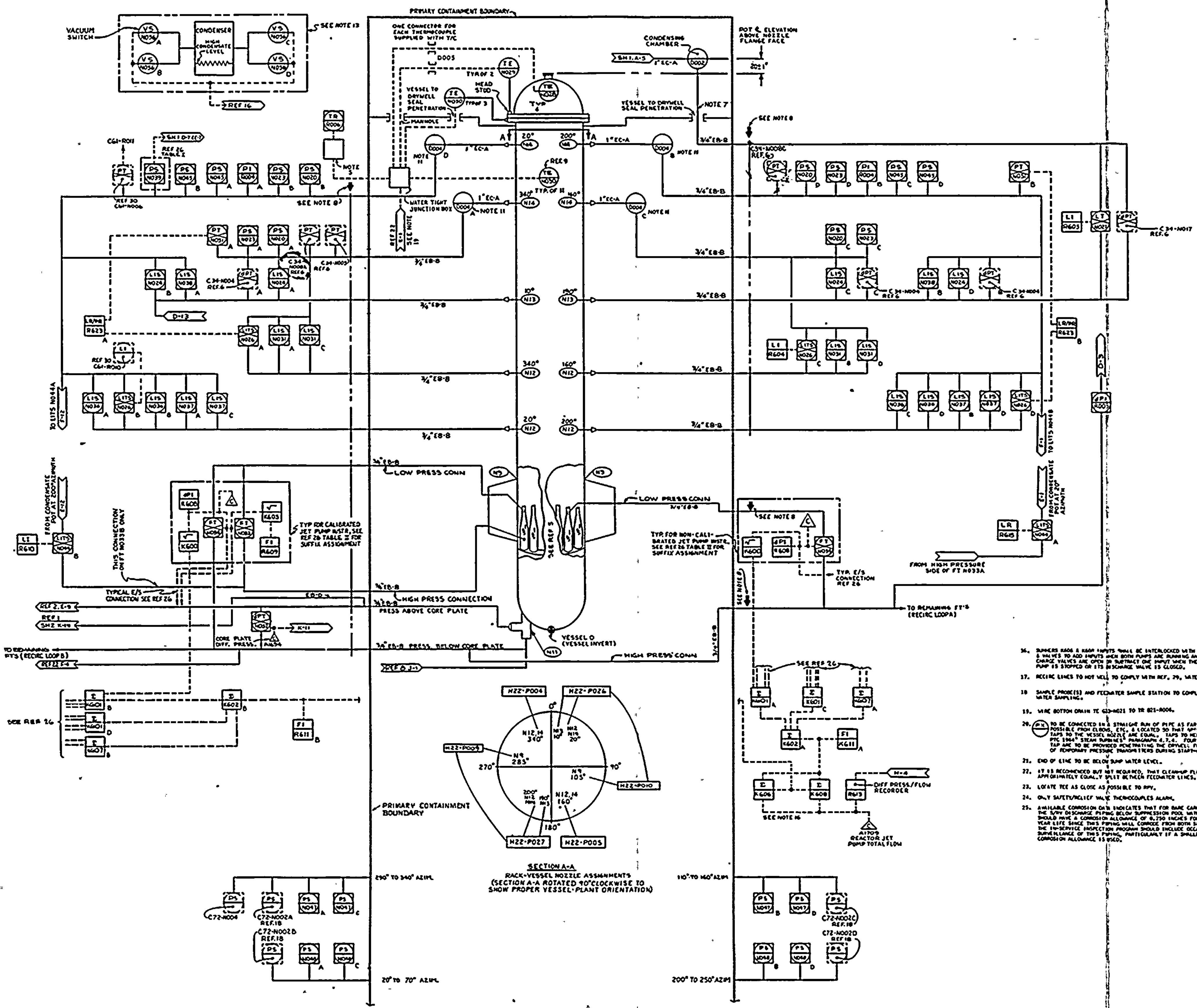


16. SUMMERS RAOG & RAOR INPUTS SHALL BE INTERLOCKED WITH RECIRC PUMP & VALVES TO ADD INPUTS WHEN BOTH PUMPS ARE RUNNING AND THEIR DISCHARGE VALVES ARE OPEN OR SUBTRACT ONE INPUT WHEN THE CORRESPONDING PUMP IS STOPPED OR ITS DISCHARGE VALVE IS CLOSED.

17. RECIRC LINES TO HOT WELL TO COMPLY WITH REF. 29, WATER QUALITY.

18. SAMPLE PROBE(S) AND FEEDWATER SAMPLE STATION TO COMPLY WITH REF. 28, WATER SAMPLING.

19. WIRE BOTTOM DRAIN TE G33-N021 TO TR B21-R006.

20. (PX) TO BE CONNECTED IN A STRAIGHT RUN OF PIPE AS FAR AS POSSIBLE FROM ELBOWS, ETC. & LOCATED SO THAT ΔP'S FROM THE TAPS TO THE VESSEL NOZZLE ARE EQUAL. TAPS TO MEET ASME PTC 1964" STEAM TURBINES" PARAGRAPH 4.7.4. FOUR WIRES PER TAP ARE TO BE PROVIDED PENETRATING THE DRYWELL FOR READOUT OF TEMPORARY PRESSURE TRANSMITTERS DURING START-UP.

21. END OF LINE TO BE BELOW SUMP WATER LEVEL.

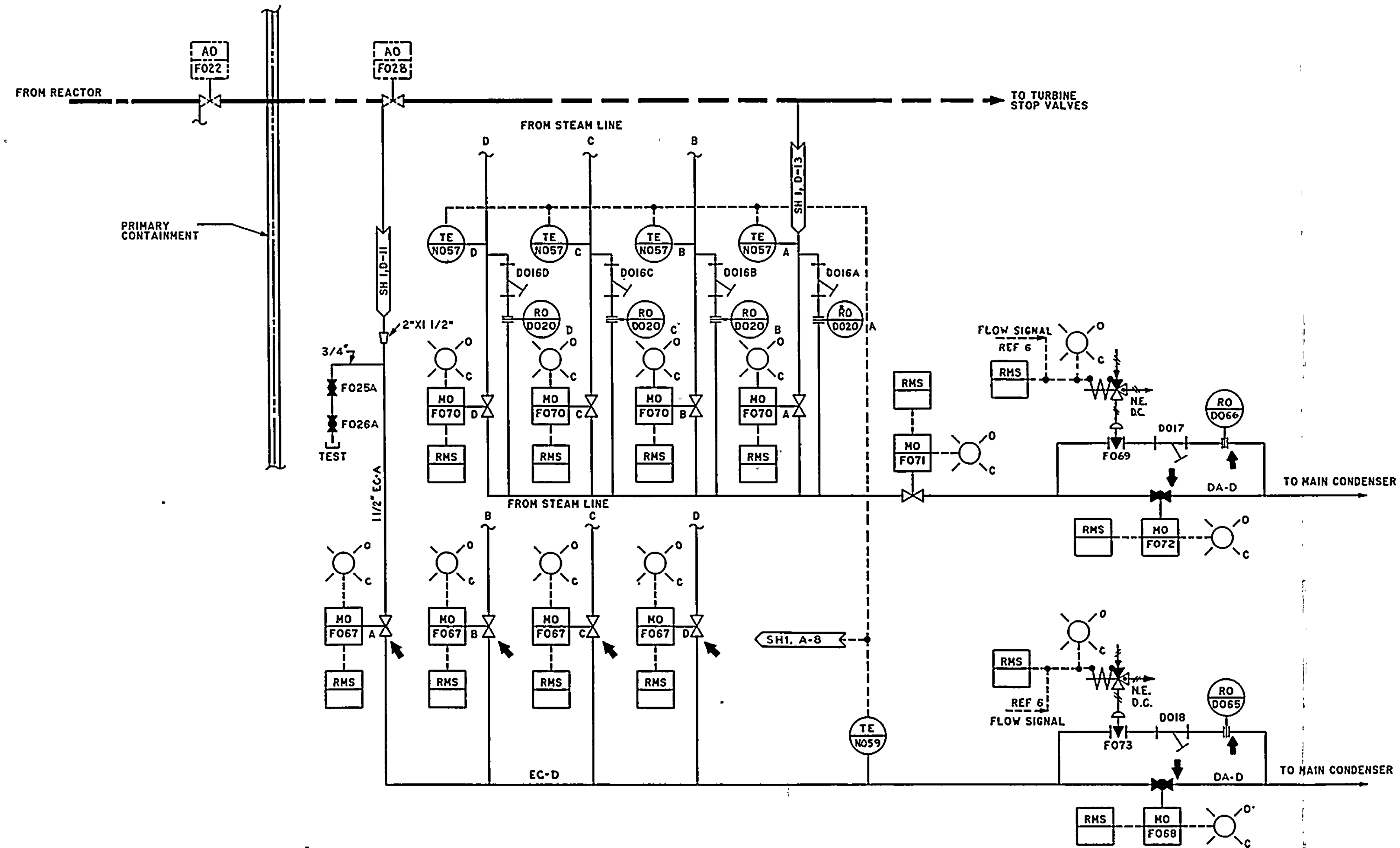
22. IT IS RECOMMENDED BUT NOT REQUIRED, THAT CLEAN-UP FLOW BE APPROXIMATELY EQUALLY SPLIT BETWEEN FEEDWATER LINES.

23. LOCATE TEE AS CLOSE AS POSSIBLE TO RPV.

24. ONLY SAFETY/RELIEF VALVE THERMOCOUPLES ALARM.

25. AVAILABLE CORROSION DATA INDICATES THAT FOR BARE CARBON STEEL THE S/RV DISCHARGE PIPING BELOW SUPPRESSION POOL WATER LEVEL SHOULD HAVE A CORROSION ALLOWANCE OF 0.250 INCHES FOR A 40 YEAR LIFE SINCE THIS PIPING WILL CORRODE FROM BOTH SIDES. THE IN-SERVICE INSPECTION PROGRAM SHOULD INCLUDE OCCASIONAL SURVEILLANCE OF THIS PIPING, PARTICULARLY IF A SMALLER CORROSION ALLOWANCE IS USED.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	P & ID NUCLEAR BOILER SYSTEM	FIGURE 5.1-3b



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	P & ID NUCLEAR BOILER SYSTEM	FIGURE 5.1-3c

plant water quality surveillance program is established providing assurance that off specification conditions will quickly be detected and corrected.

The sampling frequency established for primary coolant at normal conductivity levels is adequate for instrument checks and routine audit purposes. When specific conductance increases and higher chloride concentrations are possible, or when continuous conductivity monitoring is unavailable, sampling frequency is increased according to plant technical specifications.

The primary coolant conductivity monitoring instrumentation, ranges, accuracy, sensor and indicator locations are shown in Table 5.2-8. The sampling is coordinated in a reactor sample station especially designed with constant temperature control and sample conditioning and flow control equipment.

3. Water Purity During a Condenser Leakage

The condensate treatment system is designed to maintain the reactor water chloride concentration below 200 ppb during a condenser tube leak of 50 gallons per minute for greater than one hour.

To protect against a major condenser tube leak, ion exchange capacity of 50 percent of installed capacity is maintained during normal operation.

5.2.3.2.3 Compatibility of Construction Materials with Reactor Coolant

The materials of construction exposed to the reactor coolant consist of the following:

a. Solution annealed austenitic stainless steels (both wrought and cast) Types 304, 304L, 316 and 316L.

b. Nickel base alloys - Inconel 600 and Inconel 750X.

c. Carbon steel and low alloy steel.

d. Some 400 series martensitic stainless steel (all tempered at a minimum of 1100°F).

e. Colmonoy and Stellite hardfacing material.

All of these materials of construction are resistant to stress corrosion in the BWR coolant. General corrosion on all materials, except carbon and low alloy steel, is negligible. Conservative corrosion allowances are provided for all exposed surfaces of carbon and low alloy steels.

Contaminants in the reactor coolant are controlled to very low limits by the reactor water quality specifications. No detrimental effects will occur on any of the materials from allowable contaminant levels in the high purity reactor coolant. Radiolytic products in the BWR have no adverse effects on the construction materials.

The Supply System's response to the provisions of NUREG-0313, Revision 1, was contained in a letter sent to the NRC dated September 2, 1981.

Type 304 stainless steel has been replaced to the extent practical with type 316L stainless steel in the recirculation inlet line safe ends. The bypass lines and the control rod drive hydraulic return line were eliminated and nozzles capped. The core spray lines are fabricated of carbon steel. The piping components that do not comply with the requirements of the NUREG will be subjected to the augmented inspection requirements of NUREG-0313, Revision 1.

5.2.3.2.4 Compatibility of Construction Materials with External Insulation and Reactor Coolant

The materials of construction exposed to external insulation are:

a. Solution annealed austenitic stainless steels. (Types 304, 304L, and 316.)

b. Carbon and low alloy steel.

The design bases with respect to General Design Criteria 34, 55, 56, and 57 are provided in Chapter 3.

5.4.6.1.1.2 Isolation

Isolation valve arrangements include the following:

a. Two RCIC system lines penetrate the coolant pressure boundary for the reactor. The first is the RCIC steam line which branches off one of the main steam lines between the reactor vessel and the main steam isolation valve. This line has two automatic motor operated isolation valves. One is located inside and the other outside containment. The isolation signals noted earlier close these valves.

b. The RCIC pump discharge line is the other line that penetrates containment and connects to the reactor vessel head. This line has two testable check valves (one inside containment and the other outside primary containment). Additionally, an automatic motor operated valve is located outside containment.

c. The RCIC turbine exhaust line vacuum breaker system line has two automatic motor operated valves and two check valves. This line runs between the suppression pool air space and the turbine exhaust line downstream of the exhaust line containment isolation valve. Positive isolation shall be automatic via a combination of flow, reactor pressure, and high drywell pressure.

   The vacuum breaker valve complex is placed outside containment due to a more desirable environment. In addition, the valves are readily accessible for maintenance and testing.

d. The RCIC pump suction line, minimum flow pump discharge line, and turbine exhaust line both penetrate the primary containment and are submerged in the suppression pool. The isolation valves for the lines are all outside primary containment and require remote-manual operation except the minimum flow valves which actuate automatically. Additionally, the turbine gland seal system vacuum pump discharges beneath the suppression pool after penetrating the primary containment.

5.4.6.1.2 Reliability, Operability, and Manual Operation

5.4.6.1.2.1 Reliability and Operability (Also see 5.4.6.2.4)

The RCIC system as noted in Table 3.2-1 is designed commensurate with the safety importance of the system and its equipment. Each component is individually tested to confirm compliance with system requirements. The system as a whole is tested during both the startup and preoperational phases of the plant to set a base mark for system reliability. To confirm that the system maintains this mark, functional and operability testing is performed at predetermined intervals throughout the life of the reactor plant.

A design flow functional test of the RCIC system may be performed during normal plant operation by drawing suction from the condensate storage tank and discharging through a full flow test return line to the condensate storage tank. The discharge valve to the head cooling spray nozzle remains closed during the test, and reactor operation remains undisturbed. All components of the RCIC System shall be capable of individual functional testing during normal plant operation. System control shall provide automatic return from test to operating mode if system initiation is required. There are three exceptions: 1) Auto/manual initiation on the flow controller. This feature is required for operator flexibility during system operation. 2) Steam inboard/outboard isolation valves. Closure of either or both of these valves requires operator action to properly sequence their opening (see 4.4.3). An alarm sounds when either of these valves leaves the fully open position. 3) Major system component's inoperability or bypassing condition shall be automatically indicated in the control room at the system level. Other system components require manual operability status checking.

5.4.6.1.2.2 Manual Operation (Also see 5.4.6.2.5.2 and 5.4.6.2.5.3)

In addition to the automatic operational features, provisions have been included for remote-manual startup, operation, and shutdown of the RCIC System, provided initiation or shutdown signals have not been actuated.

After the RHR system is placed in the steam condensing mode, the operator will select the condensate discharge from the RHR steam condensing heat exchangers as the RCIC pump suction supply. The steam condensing mode of the RHR System is manually placed in operation. Once steam condensing has been established, water level in the RHR heat exchangers is automatically maintained by means of a regulating valve in

the condensate discharge line. Initially, the condensate discharge is directed to the suppression pool. After proper water quality is obtained, the condensate discharge may be directed to the RCIC pump suction. The level control for the RHR heat exchangers shall be independent from the RCIC control system. The operator selects the flow set point of the RCIC System to match the condensate flow rate from the RHR heat exchangers.

5.4.6.1.3 Loss of Offsite Power

The RCIC System power is derived from a highly reliable source that is maintained by either onsite or offsite power. (Refer to 5.4.6.1.1)

5.4.6.1.4 Physical Damage

The system is designed to the requirements of Table 3.2-1. commensurate with the safety importance of the system and its equipment. The RCIC is physically located in a different quadrant of the reactor building and utilizes different divisional power (and separate electrical routings) than its redundant HPCS system described in 5.4.6.1.1 and 5.4.6.2.4.

5.4.6.1.5 Environment

The system operates for the time intervals and the environmental conditions specified in 3.11.

5.4.6.2 System Design

5.4.6.2.1 General

5.4.6.2.1.1 Description

The Reactor Core Isolation Cooling System consists of a turbine, pump, piping, valves, accessories, and instrumentation designed to assure that sufficient reactor water inventory is maintained in the reactor vessel to permit adequate core cooling to take place. This prevents reactor fuel overheating during the following conditions:

a. Should the vessel be isolated and maintained in the hot standby condition.

b. Should the vessel be isolated and accompanied by loss of coolant flow from the reactor feedwater system.

c. Should a complete plant shutdown under conditions of loss of normal feedwater system be started before the reactor is depressurized to a level

where the shutdown coolant system can be placed into operation.

Following a reactor scram, steam generation will continue at a reduced rate due to the core fission product decay heat. At this time the turbine bypass system will divert the steam to the main condenser, and the feedwater system will supply the make-up water required to maintain reactor vessel inventory.

In the event the reactor vessel is isolated, and the feedwater supply is unavailable, relief valves are provided to automatically (or remote manually) maintain vessel pressure within desirable limits. The water level in the reactor vessel will drop due to continued steam generation by decay heat.

Upon reaching a predetermined low level, the RCIC System is initiated automatically. The turbine driven pump will supply demineralized make-up water from the condensate storage tank to the reactor vessel. The suction line from this source is provided with an in-line reserve with appropriate safety-related level instrumentation. In the event that the water supply from the condensate storage tank becomes exhausted, the level instrumentation in the in-line reserve initiates an automatic switchover to the suppression pool as the water source for the RCIC pump. The in-line reserve has sufficient volume to maintain the minimum required RCIC pump NPSH plus a two foot margin while the switchover occurs, thus assuring a water supply for continuous operation of the RCIC system. The turbine will be driven with a portion of the decay heat steam from the reactor vessel, and will exhaust to the suppression pool.

During RCIC operation, the suppression pool shall act as the heat sink for steam generated by reactor decay heat. This will result in a rise in pool water temperature. Heat exchangers in the Residual Heat Removal System are used to maintain pool water temperature within acceptable limits by cooling the pool water directly or by condensing generated steam prior to entering the suppression pool. When using the steam condensing mode, the condensate discharge from the heat exchangers may be used as RCIC pump suction supply.

5.4.6.2.1.2 Diagrams

The following diagrams are included for the RCIC Systems.

a. A schematic "Piping and Instrumentation Diagram" (Figure 5.4-9) shows all components, piping, points where interface system and subsystems tie

5.4.6.2.1.3 Interlocks

The following defines the various electrical interlocks:

a. There are four key locked valves namely F063, F008, F068, and F069 and two key locked resets namely the "isolation resets".

b. F031s limit switch activates when fully open and closes F010, F022, and F059.

c. F068s limit switch activates when fully open and clears F045 permissive so F045 can open.

d. F045s limit switch activates when F045 is not fully closed and energizes 15-second time delay for low pump suction pressure trip and also initiates startup ramp function. This ramp resets each time F045 is closed.

e. F045s limit switch activates when fully closed and permits F004, F005, F025, and F026 to open and closes F013 and F019.

f. The turbine trip throttle valve (part of C002) limit switch activates when fully closed and closes F013 and F019.

g. The combined pressure switches at reactor low pressure and high drywell pressure when activated closes F080 and F086.

h. RCIC FCD 110% overspeed, high turbine exhaust pressure, low pump suction pressure, or an isolation signal actuates and closes the turbine trip throttle valve. When signal is cleared, the trip throttle valve must be reset from control room.

i. 125% overspeed trips both the mechanical trip at the turbine and the trip throttle valve. The former is reset at the turbine and then the latter is reset in the control room.

j. An isolation signal closes F008, F063, F064, F076, and other valves as noted above in items f and h.

k. An initiation signal opens F010 if closed, F013 and F045; starts barometric condenser vacuum pump; and closes F022 and F059 if open.

l. High and low inlet RCIC steam line drain pot levels, respectively, open and close F054.

m. The combined signal of low flow plus discharge pressure open and with increased flow closes F019. Also see items e and f above.

n. the signal of in-line reserve tank low water level opens valve F031.

5.4.6.2.2 Equipment and Component Description

5.4.6.2.2.1 Design Conditions

Operating parameters for the components of the RCIC Systems, defined below, are shown on Figure 5.4-10.

a. One 100% capacity turbine and accessories

b. One 100% capacity pump assembly and accessories

c. Piping, valves, and instrumentation for:

   1. Steam supply to the turbine

   2. Steam supply to RHR condensing heat exchanger

   3. Turbine exhaust to the suppression pool

   4. Make-up supply from the condensate storage tank to the pump suction

   5. Make-up supply from the suppression pool to the pump suction

   6. Make-up supply from the RHR steam condensing heat exchangers

   7. Pump discharge to the head cooling spray nozzle, including a test line to the condensate storage tank, a minimum flow bypass line to the suppression pool, and a coolant water supply to accessory equipment.

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Steam Supply Isolation Valves (F008)	Open and/or close against full differential pressure of 1140 psi at a minimum rate of 12 inches per minute.
RHR Steam Supply Isolation Valves (F063/F064)	Open and/or close against full differential pressure of 1140 psi.
Cooling Water Pressure Control Valve (F015)	Downstream control valve capable of maintaining constant downstream pressure of 75 psia.
Pump Suction Relief Valve (F017)	100 psig relief setting; 10 gpm at 10 percent accumulation.
Cooling Water Relief Valve (F018)	Sized to prevent overpressurization piping, valves and equipment in the coolant loop in the event of failure of pressure control valve F015.
Pump Test Return Valve (F022)	Is capable of throttling against 1000 psi differential pressure.
Relief Valve Barometric Condenser (F033)	Relief valve is capable of retaining 10 inches of mercury vacuum at 140°F ambient, with a set pressure of 5-7 psig and a flow of 20 gpm at 10 percent accumulation.
Pump Suction Valve Suppression Pool (F031)	Is located as close as practical to the primary containment.
Pump Suction Condensate Storage Tank (F010)	Open and/or close against full differential pressure of 45 psi within 15 seconds.

Testable Check Valve (F065/F066)	System test mode bypasses this valve, and its functional capability shall be demonstrated separately. Therefore, valve test provisions are provided, including limit switches to indicate disc movement. The valve and valve associated equipment are capable of proper functional operation during high ambient conditions.
Warm-Up Line Isolation Valve (F076)	Valve will open and/or close against full differential pressure of 1140 psi at a minimum rate of 12 inches per minute.
Vacuum Breaker Valves (F080 & F086)	Valves will open and/or close against a differential pressure of 200 psi at a minimum rate of 12 inches per minute.
e. Rupture Disc Assemblies (D001/D002)	Utilized for turbine casing protection, shall include a mated vacuum support to prevent rupture disc reversing under vacuum conditions. Rupture pressure 150 psig $\pm$ 10 psig Flow capacity 60,000 lb/Rv @ 165 psig
f. Condensate Storage Requirements	Total reserve storage for RPV makeup is 135,000 gallons.

p. Follow steps n through s of 5.4.6.2.5.1.

5.4.6.2.5.3 Steam Condensing (Hot Standby) Operation

This mode of operation is manually initiated by the operator as follows:

a. Verification made in steps a through j of 5.4.6.2.5.1 shall be completed.

b. When the reactor water level is going to be maintained in the hot standby mode and the level starts to drop the RCIC system can be started by manually pushing the RCIC "Manual Initiation" push button. See 5.4.6.2.5.1(k) for RCIC subsequent starts. Concurrently, the RHR System water quality should be readied for vessel injection, see 5.4.7.2.6(b).

c. Adjust controller so it may be switched to manual mode and maintain same flow at pressure condition established by step b above. Then switch to manual mode.

d. Adjust flow controller set point as required to maintain desired reactor water level.

e. When RHR water is ready for vessel injection open RHR suction valve to RCIC System pump. During steam condensing operation if the RHR produces more condensate than required to maintain reactor level, the excess may be dumped to the suppression pool via the RHR system. Also, if more flow is required than supplied from the RHR head exchangers it will come from the condensate storage tank.

f. When steam condensing is completed and the RCIC system is no longer required, close the RHR suction valve, manually trip the RCIC system, and turn flow controller back to automatic.

g. Follow steps n through s of 5.4.6.2.5.1.

5.4.6.2.5.4 Limiting Single Failure

The most limiting single failure in the combined function of RCIC and HPCS systems is the failure of HPCS. If the capacity of RCIC System is adequate to maintain reactor water level, the operator follows 5.4.6.2.5.1. If however, the RCIC capacity is inadequate 5.4.6.2.5.1 applies, but additionally the operator may also initiate the ADS system described in 6.3.2.

5.4.6.3 Performance Evaluation

The analytical methods and assumptions used in evaluating the RCIC system are presented in Chapter 15, "Accident Analyses", and Appendix A to Chapter 15, "Plant Nuclear Safety Operational Analyses". The RCIC system provides the flows required by the analysis (see Figure 5.4-10) within a 30-second interval based upon considerations noted in 5.4.6.2.4.

5.4.6.4 Preoperational Testing

The preoperational and initial startup test program for the RCIC system is presented in Chapter 14, "Initial Test Program".

5.4.6.5 Safety Interfaces

The balance-of-plant/GE nuclear steam supply system safety interfaces for the reactor core isolation cooling system are: (1) preferred water supply from the condensate storage tanks; (2) all associated wire, cable, piping, sensors, and valves which lie outside the nuclear steam supply system scope of supply; and (3) air supply for testable check and solenoid actuated valve(s).

5.4.7 RESIDUAL HEAT REMOVAL SYSTEM

5.4.7.1 Design Bases

The RHR system is comprised of three independent loops. Each loop contains its own motor-driven pump, piping, valves, instrumentation, and controls. Each loop has a suction source from the suppression pool and is capable of discharging water to the reactor vessel via a separate nozzle, or back to the suppression pool via a full flow test line. In addition, the A and B loops have heat exchangers which are cooled by standby service water. Loops A and B can also take suction from the reactor recirculation system suction, and can discharge into the reactor recirculation discharge or to the suppression pool and drywell spray spargers. Spool-piece interties are provided to permit the RHR heat exchangers to be used to supplement the cooling capacity of the fuel pool cooling system. The A and B loops also have connections to reactor steam via the RCIC steam line and can discharge condensate to the RCIC pump suction or to the suppression pool. LaSalle 1 and 2, and Zimmer 1 are nuclear plants which employ similar RHR systems and which are in the process of being licensed.

separately to provide clarity.

5.4.7.1.1.1 Residual Heat Removal Mode (Shutdown Cooling Mode)

a. The functional design basis of the shutdown cooling mode is to have the capability to remove decay and sensible heat from the reactor primary system so that the reactor outlet temperature is reduced to 125°F 20 hours after the control rods have been inserted, to permit refueling when the maximum service water temperature is 95°F, the core is "mature" and the tubes are completely fouled. (See 5.4.7.2.2.6 for exchanger design details.) The capacity of the heat exchangers is such that the time to reduce the vessel outlet water temperature to 212°F corresponds to a cooldown rate of 100°F per hour with both loops in service. However, the flushing operation associated with shutdown prevents attaining 212°F coolant temperature at the minimum time.

   If 2 hours are used for flushing, the minimum time required to reduce vessel coolant temperature to 212°F is depicted by Figure 5.4-11.

b. The design basis for the most limiting single failure for the RHR system (Shutdown Cooling Mode) is that the shutdown line can be made usable by manual action (see 5.4.7.1.5) and the plant is then shutdown using the capacity of a single RHR heat exchanger and related service water capability. Figure 5.4-12 shows the minimum time required to reduce vessel coolant temperature to 212°F using one RHR heat exchanger and allowing 2 hours for flushing.

5.4.7.1.1.2 Low Pressure Coolant Injection (LPCI) Mode

The functional design bases for the LPCI mode is to pump a total of 7450 GPM of water per loop using the separate pump loops from the suppression pool into the core region of the vessel when there is a 26 psi differential between reactor pressure and the pressure of the suppression pool air volume. Injection flow commences at 225 psid vessel pressure above drywell pressure.

The initiating signals are: vessel level 1.0 feet above the active core or drywell pressure greater than or equal to 2.0 psig. The pumps will attain rated speed in 27 seconds and injection valves fully open in 40 seconds.

5.4.7.1.1.3 Suppression Pool Cooling Mode

The functional design basis for the suppression pool cooling mode is that it shall have the capacity to ensure that the suppression pool temperature immediately after a blowdown shall not exceed 170°F.

5.4.7.1.1.4 Containment Spray Cooling Mode

The functional design basis for the containment spray cooling mode is that there should be two redundant means to spray into the drywell and suppression pool vapor space to reduce internal pressure to below design limits.

5.4.7.1.1.5 Reactor Steam Condensing Mode

The functional design basis for the reactor steam condensing mode is that the heat exchanger in one loop of the RHR system, in conjunction with the RCIC turbine, shall be able to condense all of the steam generated after a reactor scram 1-1/2 hours after scram.

5.4.7.1.2 Design Basis for Isolation of RHR System from Reactor Coolant System

The low pressure portions of the RHR system are isolated from full reactor pressure whenever the primary system pressure is above the RHR system design pressure. See 5.4.7.1.3 for further details. In addition, automatic isolation may occur for reasons of vessel water inventory retention which is unrelated to line pressure rates. (See 5.2.5 for an explanation of the Leak Detection System and the isolation signals.)

The RHR pumps are protected against damage from a closed discharge valve by means of automatic minimum flow valves, which open when the main line flow is less than 550 gpm and close when the main line flow is greater than 550 gpm.

5.4.7.1.3 Design Basis for Pressure Relief Capacity

The relief valves in the RHR system are sized on one of three bases:

a. Thermal relief only

b. Valve bypass leakage only

c. Control valve failure and the subsequent uncontrolled flow which results.

Transients are treated by items a and c; item b above results from an excessive leak past isolation valves. E12-F055 and RHR-RV-95 are sized to maintain upstream piping at 500 psig and 10 percent accumulation with E12-F051 or E12-F087 fully open and a reactor pressure equal to the lowest Nuclear Boiler safety/relief valve spring set point. E12-F036 are sized to maintain upstream pressure at 75 psig and 10 percent accumulation with both PCV E12-F065 A&B failed open. E12-F005, F025, F088, and F030 are set at the design pressure specified in the process data drawing plus 10 percent accumulation. RHR-RV-98 is installed across E12-F009 to prevent thermal overpressurization between E12-F008 and E12-F009.

Redundant interlocks prevent opening valves to the low pressure suction piping when the reactor pressure is above the shutdown range. These same interlocks initiate valve closure on increasing reactor pressure.

A pressure interlock prevents connecting the discharge piping to the primary system whenever the pressure difference across the discharge valve is greater than the design differential. In addition a high pressure check valve will close to prevent reverse flow if the pressure should increase. Relief valves in the discharge piping shall be sized to account for leakage past the check valve.

5.4.7.1.4 Design Basis With Respect to General Design Criteria 5

The RHR system for this unit does not share equipment or structures with any other nuclear unit.

5.4.7.1.5 Design Basis for Reliability and Operability

The design basis for the Shutdown Cooling mode of the RHR system is that this mode is controlled by the operator from the control room. The only operation performed outside of the control room for a normal shutdown is manual operation of a local flushing water admission valve, which is the means of assuring that the suction line of the shutdown portions of the RHR system is filled and vented.

Two separate shutdown cooling loops are provided; and although both loops are required for shutdown under normal circumstances, the reactor coolant can be brought to 212°F in less than 20 hours with only one loop in operation. With the exception of the shutdown suction, shutdown return, and steam supply and condensate discharge lines, the entire RHR system is part of the ECCS and containment cooling systems, and is therefore designed with redundancy, flooding protection, piping protection, power separation, etc. required of such

systems. (See 6.3 for an explanation of the design bases for ECCS systems.) Shutdown suction and discharge valves are required to be powered from both offsite and standby emergency power for purposes of isolation and shutdown following a loss of offsite power. In the event that the outboard shutdown cooling suction supply valve (E12-F008) fails to open from the control room, an operator is sent to open the valve by hand. If the attempt to open the outboard valve (F008) proves unsuccessful, or the inboard shutdown cooling suction supply valve (E12-F009) fails to open, the operator will establish an alternate cooling path as described in the notes to Figure 15.2-11, Activity C1 or C2.

5.4.7.1.6 Design Basis for Protection from Physical Damage

The RHR system is designed to the requirements of Table 3.2-1. With the exception of the common shutdown cooling line, redundant components of the RHR system are physically located in different quadrants of the reactor building, and are supplied from independent and redundant electrical divisions. Further discussion on protection from physical damage is provided in Chapter 3.

5.4.7.2 Systems Design

5.4.7.2.1 System Diagrams

All of the components of the RHR system are shown in the P&ID Figure 5.4-13. A description of the controls and instrumentation is presented in 7.3.1.1.1, "Emergency Core Cooling Systems (ECCS) Instrumentation and Controls."

A process diagram and process data are shown in Figures 5.4-14a through 5.4-14c. All of the sizing modes of the system are shown in the process data. The FCD for the RHR system is provided in Chapter 7.

Interlocks are provided: (a) to prevent drawing vessel water to the suppression pool; (b) to prevent opening vessel suction valves above the suction line design pressure, or the discharge line design pressure with the pump at shutoff head; (c) to prevent inadvertant opening of drywell spray valves while in shutdown; (d) to prevent opening low pressure steam supply valve F087 when vessel pressure is above line design rating; and (e) to prevent pump start when suction valve(s) are not open.

5.4.7.2.2 Equipment and Component Description

a. System Main Pumps

The RHR main system pumps are motor-driven deep-well pumps with mechanical seals and cyclone separators. The pumps are sized on the basis of the LPCI mode (Mode A) and the minimum flow mode (Mode G) of the Process Data Figure 5.4-14b. Design pressure for the pump suction structure is 220 psig with a temperature range from 40°F to 360°F. Design pressure for the pump discharge structure is 500 psig. The bases for the design temperature and pressure are maximum shutdown cut-in pressures and temperature, minimum ambient temperature, and maximum shutoff head. The pump pressure vessel is carbon steel, the shaft is stainless steel. A comparison between the required NPSH (obtained from the pump characteristic curves provided in Figures 6.3-10a, b and c) and the NPSH needed in the Process Diagram Figure 5.4-14b (Note 8) demonstrates the required NPSH is adequate. Available NPSH is calculated per Regulatory Guide 1.1.

b. Heat Exchangers

The RHR system heat exchangers are sized on the basis of the duty for the shutdown cooling mode (Mode E of the Process Data). All other uses of these exchangers, including steam condensing, require less cooling surface.

Flow rates are 7450 gpm (rated) on the shell side and 7400 gpm (rated) on the tube side (service water side). Rated inlet temperature is 95°F tube side. The overall heat transfer coefficient is 195 BTU per hour square foot. The exchangers contain 7641 ft^2 of effective surface. Design temperature range of both shell and tube sides are 40°F to 480°F. Design pressure is 500 psig on both sides. Fouling factors are 0.0005 shell side and 0.002 tube side. The construction materials are carbon steel for the pressure vessel with stainless steel tubes and stainless steel clad tube sheet.

c. Valves

All of the directional valves in the system are conventional gate, globe, and check valves designed for nuclear service. The injection valves, reactor coolant isolation valves, and pump minimum flow valves are high speed valves, as operation for LPCI injection or vessel isolation requires. Valve pressure ratings are specified as necessary to provide the control or isolation function; i.e., all vessel isolation valves are rated as Class 1 nuclear valves rated at the same pressure as the primary system.

Steam pressure-reducing valves are designed to regulate steam flow into the heat exchangers from full reactor pressure to maintain downstream pressure at 200 psig.

d. ECCS Portions of the RHR System

The ECCS portions of the RHR system include those sections described through Mode A-1 of Figure 5.4-14a.

The route includes suppression pool suction strainers, suction piping, RHR pumps, discharge piping injection valves, and drywell piping into the vessel nozzles and core region of the reactor vessel.

Steam condensing components include steam supply piping and valves, heat exchangers, and condensate piping.

Suppression pool cooling components include pool suction strainers, suction piping, pumps, heat exchangers and pool return lines.

Containment spray components are the same as suppression pool cooling except that the spray headers replace the pool return lines.

5.4.7.2.3 Controls and Instrumentation

Controls and instrumentation for the RHR system are described in Chapter 7.

RHR system relief valve capacities and settings are given in 5.4.7.1.3. Discharge from the relief valves is directed to the suppression pool.

5.4.7.2.4 Applicable Codes and Classifications

Refer to 3.2.

5.4.7.2.5 Reliability Considerations

The Residual Heat Removal System has included the redundancy requirements of 5.4.7.1.5. Two completely redundant loops have been provided to remove residual heat, each powered from a separate emergency bus. With the exception of the common shutdown line, all mechanical and electrical components are separate. Either loop is capable of shutting down the reactor within a reasonable length of time.

5.4.7.2.6 Manual Action

a. Residual Heat Removal (Shutdown Cooling Mode)

In shutdown operation, when vessel pressure is 135 psig or less, the pool suction valve is closed for the B shutdown loop. Valve E12-F007 is opened to fill the suction line, after which it is closed. The testable check bypass valve may be opened in the shutdown return line and vessel water is permitted to enter the upper portion of the B loop to prewarm it and provide a nominal flush. Effluent is directed to radwaste via valves E12-F040 and F049 which are operated from the control room and a temperature element is used to control effluent temperature. The testable check bypass valve is closed and vessel suction valves are opened to allow prewarming of the lower half of the shutdown loop with effluent directed to radwaste as before. The radwaste effluent valves are closed. The service water pump is started and the service water valves are opened. Then the RHR pump is started at a regulated flow through return valve E12-F053 and cooldown of the vessel is in progress. Cooldown rate is subsequently controlled via valve E12-F053 (total flow) and E12-F048 (heat exchanger bypass flow). All operations are performed from the control room except for opening and closing of F007.

The manual actions required for the most limiting failure are discussed in 5.4.7.1.5.

b. Steam Condensing

The operator closes the heat exchanger inlet and outlet valves, starts the service water pumps, opens the service water valve, and actuates the drain valve logic, which opens the drain valve to the suppression pool. The heat exchanger water level drains to a preset value and the level controller shuts the outlet valve. The operator admits steam slowly to the heat exchangers by opening the steam supply valve partially. The automatic pressure regulator controls steam flow to maintain steam pressure in the exchanger. The operator regulates the opening of noncondensible relief valves to prevent a buildup of non-condensibles in the exchanger. When condensate quality attains the appropriate level, the operator switches condensate from the pool to RCIC pump suction. All operations are made from the control room.

5.4.7.3 Performance Evaluation

Thermal performance of the RHR heat exchangers is based on the residual heat generated at 20 hours after rod insertion, a 125^{o}F vessel outlet (exchanger inlet) temperature, and the flow of two loops in operation. Because shutdown is usually a controlled operation, maximum service water temperature less 10^{o}F is used as the service water inlet temperature. These are nominal design conditions; if the service water temperature is higher, the exchanger capabilities are reduced and the shutdown time may be longer or vice versa.

5.4.7.3.1 Shutdown With All Components Available

No typical curve is included here to show vessel cooldown temperatures versus time due to the infinite variety of such curves that may be due to: (1) clean steam systems that may allow the main condenser to be used as the heat sink when nuclear steam pressure is insufficient to maintain steam air ejector performance; (2) the condition of fouling of the exchangers; (3) operator use of one or two cooling loops; (4) coolant water temperature; and (5) system flushing time. Since the exchangers are designed for the fouled condition with relatively high service water temperature, the units have excess capability to cool when first cut in at high vessel temperatures. Total flow and mix temperature are

TABLE 5.4-3 (Continued)

Location	Active/ Inactive	Valve No.	Reference Figure
Flow Control	Inactive	B35F060	5.4-2b
Pump Discharge	Inactive	B35F067	5.4-2b
RCIC Vessel Head In	Active	E51F066	5.4-9a
	Active	E51F065	5.4-9a
	Active	E51F013	5.4-9a
HPCS In	Active	E22F005	6.3-1
	Active	E22F004	6.3-1
	Inactive	E22F038	6.3-1
LPCS In	Active	E21F006	6.3-5
	Active	E21F005	6.3-5
	Inactive	E21F051	6.3-5
Standby	Active	SLC-V-7	9.3-13
Liquid Control In	Active	SLV-V-4	9.3-13
	Active	SLC-V-6	9.3-13
	Inactive	SLC-V-8	9.3-13
Pump Description			
Recirculation Pump	Inactive	B35C001	5.4-2b

Active components are those whose operability is relied on to perform a safety function during the transients or accidents.

Inactive components are those whose operability (e.g., valve opening or closure, pump operation or trip) is not relied on to perform the system's safety function during the transients or accidents.

TABLE 5.4-4

SAFETY AND RELIEF VALVES FOR PIPING SYSTEMS CONNECTED TO THE RCPB

Safety and/or Relief Valve Identification	Description
B22F013A-H	Main Steam Line Safety/Relief Valves
B22F013J-N	Main Steam Line Safety/Relief Valves
B22F013P	Main Steam Line Safety/Relief Valves
B22F013R-S	Main Steam Line Safety/Relief Valves
B22F013U-V	Main Steam Line Safety/Relief Valves
E51F017	RCIC System Suction Line
E51F018	RCIC Lube Oil Cooler Supply Line
E51F033	RCIC Vacuum Tank
E12F055A,B	RHR Condensing Mode Steam Supply Line
RHR-RV-95A,B	RHR Condensing Mode Steam Supply Line
E12F036	RHR Condensing Mode Return Line to RCIC
E12F005	Shutdown Cooling Supply Line
E12F025A,B	Shutdown Cooling Return Line
E12F088A,B,C	Suppression Pool Supply for RHR
E12F030	RHR Flush Line
RHR-RV-1A,B	RHR Heat Exchanger (Shell side)
RWCU-RV-1	RWCU Regenerative Heat Exchanger (Shell side)
RWCU-RV-2	RWCU Non-Regenerative Heat Exchanger (Tube side)
RWCU-RV-3	RWCU Regenerative Heat Exchanger (Tube side)
G33F036	RWCU Blowdown to Radwaste System or Condenser
E22F014	High Pressure Core Spray Suction Line
E22F035	High Pressure Core Spray Pump Discharge Line
E21F018	Low Pressure Core Spray Pump Discharge Line
E21F031	Low Pressure Core Spray Suction Line
C41F029A,B	Standby Liquid Control Pump Discharge Line

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static loads, utilizing the appropriate dynamic load factors. The components stresses were found to be within the values specified in the appropriate Codes, however, after a LOCA, refueling a bulkhead would require requalification prior to use. This is considered acceptable since the refueling bulkhead does not perform a safety-related function and would not become a missile during the postulated LOCA.

The analyses for the annulus were reported in full detail in References 6.2-9 through 6.2-11. All potential pipe breaks within the sacrificial shield wall have been evaluated. The information is contained in References 3.8-5, 3.8-6, 3.8-7, and 3.8-24. These references have been previously submitted to the NRC. The result of the case of a 60-node model of the shield wall annulus for pressure transient calculation was confirmed by the NRC, and the analysis was considered acceptable for the shield wall base design and the design of the shield wall above the base, as stated in NRC letters (References 6.2-12 and 6.2-13).

Peak and transient loading used to establish the adequacy of the sacrificial shield wall, including the time/space dependent forcing functions are presented in References 6.2-9 through 6.2-11 and 3.8-24.

Subsequently, a more realistic approach was used in determining loads from postulated pipe breaks within the annulus area. These loads were used to produce response spectra for use in evaluating the secondary effects (the dynamic effects on piping systems, equipment, and components attached to the sacrificial shield wall of the RPV). Three principal changes were made in the assumptions used in the previous more conservative sacrifical shield wall analysis. Namely:

a. The volume in the annulus was utilized to receive the blowdown with the RPV installation volume conservatively assumed not to be available.

b. A finite time dependent blowdown was used for the recirculation break, utilizing NSSS supplier methodology. The effect of subcooling has been taken into account.

c. The feedwater pressurization analysis was developed utilizing blowdown values developed by detailed computer analysis rather than the previous hand calculation method.

Current state-of-the-art industry methods were used for these annulus pressurization calculations. These methods result in more realistic prediction of pressures as compared to the more conservative calculations discussed previously. Each of the three changes employed are described below:

a. Annular Volume

The current industry approach is to utilize the annular volume excluding the RPV insulation volume which is conservatively assumed not to be available. This approach is conservatively assumed not to be available. This approach is conservative but more realistic than previous analyses where only the annular volume on one side of the RPV insulation was available.

b. Finite Time Dependent Blowdown

The blowdown loading values given in Reference 6.2-11 were derived with the assumption that the pipe break would occur instantaneously and that the annulus area would see the maximum blowdown instantaneously. Actually, the full flow from the severed pipe can not be realized until the severed pipe ends separate a distance equal to one half (1/2) the pipe diameter. Movement actually occurs in a finite time and is a function of the stiffness characteristics of the pipe and the restraining capability of the pipe whip restraints.

Current industry practice was used to develop displacement versus time data for a finite break opening; the General Electric analytical method for determining the short-term mass and energy release was used. The analysis was utilized for the recirculation loop break, but not for the feedwater line since it was determined that the small precentage reduction for the feedwater would not warrant the additional calculations.

c. Feedwater Break Blowdown Data

The blowdown analysis for the postulated feedwater line break was based on a comprehensive model developed for the entire feedwater system from the condenser to the reactor vessel. This model, in conjunction with the RELAP4/MOD5 computer program (Reference 6.2-14) was used to calculate the transient and energy blowdown data.

6.2-11 Washington Public Power Supply System, Nuclear Project No. 2, Report No. WPPSS-74-2-R2-B, "Sacrificial Shield Wall Design Supplemental Information", August 19, 1975.

6.2-12 Letter from R. C. DeYoung of NRC to J. J. Stein of WPPSS, dated August 13, 1975. Subject: Sacrificial Shield Wall Design.

6.2-13 Letter from R. C. DeYoung of NRC to J. J. Stein of WPPSS, dated October 15, 1975. Subject: Sacrificial Shield Wall Design.

6.2-14 ANCR-NOREG-135, "RELAP4/MOD5 - A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactor and Related Systems Users Manual" - 3 volumes, September, 1976.

6.2-15 AEC-TR-6630, "Handbook of Hydraulic Resistance, Coefficients of Local Resistance and of Friction", by I. E. Idel'Chick, 1960.

6.2-16 Bilanin, W. J., "The General Electric Mark III Pressure Suppression Containment System Analytical Model," June 1974, (NEDO-20533).

6.2-17 "Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors", Licensing Topical Report NEDO-103 9, General Electric, April 1970.

6.2-18 A. K. Post and B. M. Johnson, "Containment Systems Experiment Final Program Summary", BNWL-1592, Battelle Northwest, Richland, Washington, July 1971.

6.2-19 J. G. Knudsen and R. K. Hilliard, "Fission Product Transport by Natural Processes in Containment Vessels", BNWL-943, Battelle Northwest, Richland, Washington, Jan. 1969.

6.2-20 R. K. Hilliard and L. F. Coleman, "Natural Transport Effects on Fission Product Behavior in the Containment Systems Experiment", BNWL-1457, Battelle Northwest, Richland, Washington, Dec. 1970.

6.2-21 R. K. Hilliard, "Removal of Iodine and Particles from Containment Atmospheres by Sprays -- Containment Systems Experiment Interim Report", BNWL-1244, Battelle Northwest, Richland, Washington, Feb. 1970.

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Pent. No.	FSAR Fig. Nos.	GDC	Code Gp. (12)	Valve No.	Valve Type	Loc.	Pwr. to Open (5)	Pwr. to Close (5)	Sig. (9)	Back Up	Norm Pos. (10)	Shut-down Pos.	Post LOCA	Fail. Pos. (6)	Vlv. Sz. (14)	Close. Time (7) (11)	Dist. to Pent.	Leads to ESF Sys.	Proc. Fld.	Leak Bar. (13)	Term. Zone (13)	Pot. Bypass Leak. (SCFH)	Notes
DW Service Line	92	9.2-4 6.2-31L	56	B	DW-V-157	Gate	I	Manual	Manual	–	–	LC	LC	LC	–	2	–		No	W	Vlvs.	S.B.	.13	
					DW-V-156	Gate	O	Manual	Manual	–	–	LC	LC	LC	–	2	–	5						
RHR Condensing Mode Steam Supply	21	3.2-8 6.2-31e	55	A	RCIC-V-63	MO Gate	I	AC	AC	K	RM	O	O/C	O/C	AS-IS	10	16	–	Yes	S	Vlvs.	R.B.	No	
					RCIC-V-76	MO Globe	I	AC	AC	K	RM	C	C	C	AS-IS	1	5	–						
					RCIC-V-64	MO Gate	O	DC	DC	X	RM	C	C	C	AS-IS	10	16	2						
RCIC Turbine Steam Supply	45	3:2-8 6.2-31e	55	A	RCIC-V-63	MO Gate	I	AC	AC	K	RM	O	O/C	O/C	AS-IS	10	16	–	No	S	Vlvs.	R.B.	No	
					RCIC-V-76	MO Globe	I	AC	AC	K	RM	C	C	C	AS-IS	1	5	–						
					RCIC-V-8	MO Gate	O	DC	DC	X	RM	O	O/C	O/C	AS-IS	4	Std	2						
RCIC Pump Minimum Flow	65	3.2-8 6.2-31h	56	B	RCIC-V-19	MO Globe	O	DC	DC	33	RM	C	C	O/C	AS-IS	2	5	7	No	W	Vlvs.	R.B.	No	22
RCIC Turbine Exhaust	4	3.2-8 6.2-31u	56	B	RCIC-V-68	MO Gate	O	DC	DC	35	RM	O	O	O/C	AS-IS	10	Std	10	No	S	Vlvs.	R.B.	No	22
			56	B	RCIC-V-40	Check	O	Process	Process	–	–	O	C	O/C	–	10	–	17	No	S	Vlvs.	R.B.	No	49
RCIC Turbine Exhaust Vacuum Breaker	116	3.2-8 6.2-31u	56	B	RCIC-V-110	MO Gate	O	DC	DC	N	RM	O	O	O/C	AS-IS	2	Std	9	No	A	Vlvs.	R.B.	No	17,49
					RCIC-V-113	MO Gate	O	DC	DC	N	RM	O	O	O/C	AS-IS	2	Std	5						
RCIC Vacuum Pump Discharge	64	3.2-8 6.2-31q	56	B	RCIC-V-69	MO Gate	O	DC	DC	36	RM	O	O	O/C	AS-IS	1-1/2	Std	3	No	W	Vlvs.	R.B.	No	22
			56	B	RCIC-V-28	Check	O	Process	Process	–	–	C	O	O/C	–	1-1/2	–	5	No	W	Vlvs.	R.B.	No	
RCIC Pump Suction from Suppression Pool	33	3.2-8 6.2-31n	56	B	RCIC-V-31	MO Gate	O	DC	DC	32	RM	C	O	O/C	AS-IS	8	Std	2	No	W	Vlvs.	R.B.	No	48
RPV Head Spray	2	3.2-8 6.2-31e	55	A	RCIC-V-66	Check	I	Process	Process	–	–	C	O	O/C	–	6	–	–	No	W	Vlvs.	R.B.	No	3
					RCIC-V-13	MO Gate	O	DC	DC	34	RM	C	O/C	O/C	AS-IS	6	15	21	No	W	Vlvs.	R.B.	No	
					RHR-V-23	MO Globe	O	DC	DC	L,U, M,R	RM	C	O/C	C	AS-IS	6	Std	28	Yes	W	Vlvs.	R.B.	No	
					RCIC-V-742	Globe	O	Manual	Manual	–	–	LC	LC	LC	–	3/4	–	3	No	W	Vlvs.	R.B.	No	

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TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Pent. No.	FSAR Fig. Nos.	GDC	Code Gp. (12)	Valve No.	Valve Type	Loc.	Pwr. to Open (5)	Pwr. to Close (5)	Iso. Sig. (9)	Back Up	Norm Pos. (10)	Shut-down Pos.	Post LOCA	Fail. Pos. (6)	Vlv. Sz. (14)	Close. Time (7) (11)	Dist. to Pent.	Leads to ESF Sys.	Proc. Fld.	Leak Bar. (13)	Term. Zone (13)	Pot. By-pass Leak. (SCFH)	Notes
Drywell Spray Loop A	11A	3.2-6 6.2-31g	56	B	RHR-V-16A	MO Gate	O	AC	AC	46	RM	C	C	O/C	AS-IS	16	10	26	Yes	W	Valves	R.B.	No	17, 24
					RHR-V-17A	MO Gate	O	AC	AC	46	RM	C	C	O/C	AS-IS	16	10	24						
Drywell Spray Loop B	11B	3.2-6 6.2-31g	56	B	RHR-V-16B	MO Gate	O	AC	AC	46	RM	C	C	O/C	AS-IS	16	10	12	Yes	W	Valves	R.B.	No	17, 24
					RHR-V-17B	MO Gate	O	AC	AC	46	RM	C	C	O/C	AS-IS	16	10	2						
LPCI Loop A	12A	3.2-6 6.2-31L	55	A	RHR-V-41A	Check	I	Process	Process	-	-	C	C	O/C	-	14	-	-	Yes	W	Valves	R.B.	No	3, 24
					RHR-V-42A	MO Gate	O	AC	AC	46	RM	C	C	O/C	AS-IS	14	27	21						
LPCI Loop B	12B	3.2-6 6.2-31L	55	A	RHR-V-41B	Check	I	Process	Process	-	-	C	C	O/C	-	14	-	-	Yes	W	Valves	R.B.	No	3, 24
					RHR-V-42B	MO Gate	O	AC	AC	46	RM	C	C	O/C	AS-IS	14	27	20						
LPCI Loop C	12C	3.2-6 6.2-31L	55	A	RHR-V-41C	Check	I	Process	Process	-	-	C	C	O/C	-	14	-	-	Yes	W	Valves	R.B.	No	3, 24
					RHR-V-42C	MO Gate	O	AC	AC	46	RM	C	C	O/C	AS-IS	14	27	20						
Shutdown Cooling Return Loop A	19A	3.2-6 6.2-31m	55	A	RHR-V-50A	Check	I	Process	Process	-	-	C	O	C	-	12	-	-	Yes	W	Valves	R.B.	No	3
					RHR-V-123A	MO Gate	I	AC	AC	F,L U,M, R	RM	C	O/C	C	AS-IS	1	STD	-						
					RHR-V-53A	MO Globe	O	AC	AC	M,L, U,R	RM	C	O	C	AS-IS	12	40	5						
Shutdown Cooling Return Loop B	19B	3.2-6 6.2-31m	55	A	RHR-V-50B	Check	I	Process	Process	-	-	C	O	C	-	12	-	-	Yes	W	Valves	R.B.	No	3
					RHR-V-123B	MO Gate	I	AC	AC	F,L U,M, R	RM	C	O/C	C	AS-IS	1	STD	-						
					RHR-V-53B	MO Globe	O	AC	AC	M,L, U,R	RM	C	O	C	AS-IS	12	40	2						
Shutdown Cooling Suction	20	3.2-6 6.2-31k	55	A	RHR-V-9	MO Gate	I	AC	AC	L,U, M,R	RM	C	O	C	AS-IS	20	40	-	Yes	W	Valves	R.B.	No	
					RHR-V-8	MO Gate	O	AC	AC	L,U, M,R	RM	C	O	C	AS-IS	20	40	14						
					RHR-V-209	Check	I	Process	Process	-	-	C	C	C	-	3/4	-	-	No	W	Valves	R.B	No	

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TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Pent. No.	FSAR Fig. Nos.	GDC	Code Gp. (12)	Valve No.	Valve Type	Loc.	Pwr. to Open (5)	Pwr. to Close (5)	Iso. Sig. (9)	Back Up	Norm Pos. (10)	Shut-down Pos.	Post LOCA	Fail. Pos. (6)	Vlv. Sz. (14)	Close. Time (7) (11)	Dist. to Pent.	Leads to ESF Sys.	Proc. Fld.	Leak Bar. (13)	Term. Zone (13)	Pot. By-pass Leak. (SCFH)	Notes
RHR Loop A:	47	3.2-6	56	B																				
pump test line		6.2-31p			RHR-V-24A	MO Globe	O	AC	AC	F, V	RM	C	C	C	AS-IS	18	Std	12	Yes	W	Valves	R.B.	No	2, 18, 24
discharge header relief					RHR-RV-25A	Relief	O	PP	Spring	-	-	C	C	C	-	2	-	33	Yes	W	Valves	R.B.	No	18, 19
heat exch. steam relief					RHR-RV-55A	Relief	O	PP	Spring	-	-	C	C	C	-	10	-	22	Yes	S	Valves	R.B.	No	18, 19
heat exch. condensate					RHR-V-11A	MO Gate	O	AC	AC	F, V	RM	C	O/C	C	AS-IS	4	-	18	Yes	W	Valves	R.B.	No	18
heat exch. condensate relief					RHR-RV-36	Relief	O	PP	Spring	-	-	C	C	C	-	8	-	20	Yes	W	Valves	R.B.	No	18, 20
pump minimum flow					RHR-FCV-64A	MO Globe	O	AC	AC	38	RM	C	C	O/C	AS-IS	3	15	22	Yes	W	Valves	R.B.	No	48
heat exch. thermal relief					RHR-RV-1A	Relief	O	PP	Spring	-	-	C	C	C	-	1-1/2	-	188	Yes	W	Valves	R.B.	No	18, 19
heat exch. vent					RHR-V-73A	MO Globe	O	AC	AC	39	RM	C	O/C	C	AS-IS	2	Std	175	Yes	A	Valves	R.B.	No	18
FDR system inter-tie					RHR-V-121	Gate	O	Manual	Manual	-	-	LC	C	LC	-	3	-	6	No	W	Valves	R.B.	No	
					RHR-V-120	Check	O	Process	Process	-	-	C	C	C	-	3	-	7	No	W	Valves	R.B.	No	
CAC system Loop A drain					RHR-V-134A	MO Gate	O	AC	AC	37	RM	C	C	O/C	AS-IS	2	Std	44	Yes	W	Valves	R.B.	No	18
pump A suction relief					RHR-RV-88A	Relief	O	PP	Spring	-	-	C	C	C	-	1	-	30	Yes	W	Valves	R.B.	No	18
RHR Loop B	48	3.2-6	56	B																				
pump test line		6.2-31p			RHR-V-24B	MO Globe	O	AC	AC	F, V	RM	C	C	C	AS-IS	18	Std	12	Yes	W	Valves	R.B.	No	2, 18, 24
discharge header relief					RHR-RV-25B	Relief	O	PP	Spring	-	-	C	C	C	-	2	-	30	Yes	W	Valves	R.B.	No	18, 19
heat exch. steam relief					RHR-RV-55B	Relief	O	PP	Spring	-	-	C	C	C	-	10	-	20	Yes	S	Valves	R.B.	No	18, 19
pump A&B suction relief					RHR-RV-5	Relief	O	PP	Spring	-	-	C	C	C	-	2	-	20	Yes	W	Valves	R.B.	No	18, 19

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Valve No.	Valve Type	Loc.	Pwr. to Open (5)	Pwr. to Close (5)	Iso. Sig. (9)	Back Up	Norm Pos. (10)	Shut-down Pos.	Post LOCA	Fail. Pos. (6)	Vlv. Sz. (14)	Close. Time (7) (11)	Dist. to Pent.	Leads to ESF Sys.	Proc. Fld.	Leak Bar. (13)	Term. Zone (13)	Pot. By-pass Leak. (SCFH)	Notes
heat exch. condensate	RHR-V-11B	MO Gate	O	AC	AC	F,V	RM	C	O/C	C	AS-IS	4	Std	15	Yes	W	Valves	R.B.	No	18
pump minimum flow	RHR-FCV-64B	MO Globe	O	AC	AC	38	RM	C	C	O/C	AS-IS	3	15	22	Yes	W	Valves	R.B.	No	18
flush line relief	RHR-RV-30	Relief	O	PP	Spring	-	-	C	C	C	-	2	-	34	Yes	W	Valves	R.B.	No	18, 19
heat exch. thermal relief	RHR-RV-18	Relief	O	PP	Spring	-	-	C	C	C	-	1-1/2	-	189	Yes	W	Valves	R.B.	No	18, 19
heat exch. vent	RHR-V-73B	MO Globe	O	AC	AC	39	Manual	C	O/C	C	AS-IS	2	Std	190	Yes	A	Valves	R.B.	No	18
CAC system Loop B drain	RHR-V-134B	MO Gate	O	AC	AC	37	Manual	C	C	O/C	AS-IS	2	Std	44	Yes	W	Valves	R.B.	No	18
pump B suction relief	RHR-RV-88B	Relief	O	PP	Spring	-	-	C	C	C	-	1	-	30	Yes	W	Valves	R.B.	No	18

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Pent. No.	FSAR Fig. Nos.	GDC	Code Gp. (12)	Valve No.	Valve Type	Loc.	Pwr. to Open (5)	Pwr. to Close (5)	Iso. Sig. (9)	Back Up	Norm Pos. (10)	Shut-down Pos.	Post LOCA	Fail. Pos. (6)	Vlv. Sz. (14)	Close. Time (7) (11)	Dist. to Pent.	Leads to ESF Sys.	Proc. Fld.	Leak Bar. (13)	Term. Zone (13)	Pot. By-pass Leak. (SCFH)	Notes
CAC Division 1 discharge to drywell	96	3.2-17 6.2-31g	56	B	CAC-V-2	MO Gate	O	DC	DC	37	RM	C	C	O/C	AS-IS	4	Std	4	Yes	A	Valves	R.B.	No	17
					CAC-FCV-2A	EHO Globe	O	AC	AC	37	RM	C	C	O/C	C	2-1/2	Std	6						
CAC Division 2 suction from drywell	97	3.2-17 6.2-31g	56	B	CAC-V-15	MO Gate	O	DC	DC	37	RM	C	C	O/C	AS-IS	4	Std	2	Yes	A,S	Valves	R.B.	No	17
					CAC-FCV-1B	EHO Globe	O	AC	AC	37	RM	C	C	O/C	C	2-1/2	Std	4						
CAC Division 2 discharge drywell	98	3.2-17 6.2-31g	56	B	CAC-V-11	MO Gate	O	DC	DC	37	RM	C	C	O/C	AS-IS	4	Std	8	Yes	A	Valves	R.B.	No	17
					CAC-FCV-2B	EHO Globe	O	AC	AC	37	RM	C	C	O/C	C	2-1/2	Std	10						
CAC Division 1 suction from drywell	99	3.2-17 6.2-31g	56	B	CAC-V-6	MO Gate	O	DC	DC	37	RM	C	C	O/C	AS-IS	4	Std	4	Yes	A,S	Valves	R.B.	No	17
					CAC-FCV-1A	EHO Globe	O	AC	AC	37	RM	C	C	O/C	C	2-1/2	Std	7						
CAC Division 1 discharge to wetwell	102	3.2-17 6.2-31g	56	B	CAC-V-4	MO Gate	O	DC	DC	37	RM	C	C	O/C	AS-IS	4	Std	3	Yes	A	Valves	R.B.	No	17
					CAC-FCV-4A	EHO Globe	O	AC	AC	37	RM	C	C	O/C	C	2-1/2	Std	5						
CAC Division 2 discharge to wetwell	103	3.2-17 6.2-31g	56	B	CAC-V-13	MO Gate	O	DC	DC	37	RM	C	C	O/C	AS-IS	4	Std	7	Yes	A	Valves	R.B.	No	17
					CAC-FCV-4B	EHO Globe	O	AC	AC	37	RM	C	C	O/C	C	2-1/2	Std	9						
CAC Division 2 suction from wetwell	104	3.2-17 6.2-31g	56	B	CAC-V-17	MO Gate	O	DC	DC	37	RM	C	C	O/C	AS-IS	4	Std	5	Yes	A,S	Valves	R.B.	No	17
					CAC-FCV-3B	EHO Globe	O	AC	AC	37	RM	C	C	O/C	C	2-1/2	Std	7						
CAC Division 1 suction from wetwell	105	3.2-17 6.2-31g	56	B	CAC-V-8	MO Gate	O	DC	DC	37	RM	C	C	O/C	AS-IS	4	Std	2	Yes	A,S	Valves	R.B.	No	17
					CAC-FCV-3A	EHO Globe	O	AC	AC	37	RM	C	C	O/C	C	2-1/2	Std	6						

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TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Pent. No.	FSAR Fig. Nos.	GDC	Code Gp. (12)	Valve No.	Valve Type	Loc.	Pwr. to Open (5)	Pwr. to Close (5)	Iso. Sig. (9)	Back Up	Norm Pos. (10)	Shut-down Pos.	Post LOCA	Fail. Pos. (6)	Vlv. Sz. (14)	Close. Time (7) (11)	Dist. to Pent.	Leads to ESF Sys.	Proc. Fld.	Leak Bar. (13)	Term. Zone (13)	Pot. Bypass Leak. (SCFH)	Notes
RB to Wetwell Vacuum Breakers	119	3.2-15 6.2-31q	56	B	CSP-V-9	AO Butfy	0	Spring	Air	40	RM	C	C	C	O	24	4	1	Yes	A	Vlvs	R.B.	No	17
					CSP-V-10	PC Check	0	Process	Process	-	RM	C	C	C	-	24	-	4	Yes	A	Vlvs	R.B.	No	26
RB to Wetwell Vacuum Breakers & Wetwell Ventilation Supply	66	3.2-15 3.2-26 6.2-31b 6.2-31q	56	B	CSP-V-5	AO Butfy	0	Spring	Air	40	RM	C	C	C	O	24	4	7	Yes	A	Vlvs	R.B.	No	17
					CSP-V-7	PC Check	0	Process	Process	-	RM	C	C	C	-	24	-	10	Yes	A	Vlvs	R.B.	No	26
					CSP-V-4	AO Butfy	0	Air	Spring	F,A, Z	RM	C	C	C	C	24	4	14	No	A	Vlvs	R.B.	No	
					CSP-V-3	AO Butfy	0	Air	Spring	F,A, Z	RM	C	C	C	C	24	4	17	No	A	Vlvs	R.B.	No	
					CSP-V-93	AO Gate	0	Air	Spring	F,A, Z	RM	O	C	C	C	1	Std	4	No	A	Vlvs	R.B.	No	
					CSP-V-98	AO Gate	0	Air	Spring	F,A, Z	RM	O	C	C	C	1	Std	6	No	A	Vlvs	R.B.	No	
RB to Wetwell Vacuum Breakers & Wetwell Ventilation Exhaust	67	3.2-15 6.2-31j 6.2-31q	56	B	CSP-V-6	AO Butfy	0	Spring	Air	40	RM	C	C	C	O	24	4	9	Yes	A	Vlvs	R.B.	No	17
					CSP-V-8	PC Check	0	Process	Process	-	RM	C	C	C	-	24	-	16	Yes	A	Vlvs	R.B.	No	26
					CEP-V-4A	AO Butfy	0	Air	Spring	F,A, Z	RM	C	C	C	C	24	4	10	No	A	Vlvs	R.B.	No	
					CEP-V-3A	AO Butfy	0	Air	Spring	F,A, Z	RM	C	C	C	C	24	4	12	Yes	A	Vlvs	R.B.	No	
					CEP-V-4B	AO Gate	0	Air	Spring	F,A, Z	RM	C	C	C	C	2	1	10	No	A	Vlvs	R.B.	No	
					CEP-V-3B	AO Gate	0	Air	Spring	F,A, Z	RM	C	C	C	C	2	1	12	No	A	Vlvs	R.B.	No	
Drywell Ventilation Supply	53	3.2-15 3.2-26 6.2-31b	56	B	CSP-V-2	AO Butfy	0	Air	Spring	F,A, Z	RM	C	C	C	C	30	4	1	No	A	Vlvs	R.B.	No	17
					CSP-V-1	AO Butfy	0	Air	Spring	F,A,	RM	C	C	C	C	30	4	4	No	A	Vlvs	R.B.	No	
					CSP-V-96	AO Gate	0	Air	Spring	F,A, Z	RM	O	C	C	C	1	Std	3	No	A	Vlvs	R.B.	No	
					CSP-V-97	AO Gate	0	Air	Spring	F,A, Z	RM	O	C	C	C	1	Std	5	No	A	Vlvs	R.B.	No	
Drywell Ventilation Exhaust	3	3.2-15 6.2-31j	56	B	CEP-V-1A	AO Butfy	0	Air	Spring	F,A, Z	RM	C	C	C	C	30	4	12	No	A	Vlvs	R.B.	No	17
					CEP-V-2A	AO Butfy	0	Air	Spring	F,A, Z	RM	C	C	C	C	30	4	8	No	A	Vlvs	R.B.	No	
					CEP-V-1B	AO Gate	0	Air	Spring	F,A, Z	RM	C	C	C	C	2	1	12	No	A	Vlvs	R.B.	No	
					CEP-V-2B	AO Gate	0	Air	Spring	F,A, Z	RM	C	C	C	C	2	1	8	No	A	Vlvs	R.B.	No	

AMENDMENT NO. 31
June 1983

LINE DESCRIPTION	Pent. No.	FSAR Fig. Nos.	GDC	Code Gp. (12)	Valve No.	Valve Type	Loc.	Pwr. to Open (5)	Pwr. to Close (5)	Iso. Sig. (9)	Back Up	Norm Pos. (10)	Shut-down Pos.	Post LOCA	Fail. Pos. (6)	Vlv. Sz. (14)	Close. Time (7) (11)	Dist. to Pent.	Leads to ESF Sys.	Proc. Fld.	Leak Bar. (13)	Term. Zone (13)	Pot. By-pass Leak. (SCFH)	Notes
RCC Inlet Header	5	3.2-14 6.2-31t	56	B	RCC-V-104	MO Gate	O	AC	AC	F,A	-	O	O	C	AS-IS	10	Std	5	No	W	Valves	R.B.	No	17
					RCC-V-5	MO Gate	O	AC	AC	F,A	-	O	O	C	AS-IS	10	Std	3						
RCC Outlet Header	46	3.2-14 6.2-31o	56	B	RCC-V-21	MO Gate	O	AC	AC	F,A	-	O	O	C	AS-IS	10	Std	3	No	W	Valves	R.B.	No	
					RCC-V-40	MO Gate	I	AC	AC	F,A	-	O	O	C	AS-IS	10	Std	-						
Suppression Pool Cleanup Suction	100	3.2-12 6.2-31l	56	B	FPC-V-153	MO Gate	O	AC	AC	F,A	RM	C	C	C	AS-IS	6	Std	2	No	W	Valves	R.B.	No	17,48
					FPC-V-154	MO Gate	O	AC	AC	F,A	RM	C	C	C	AS-IS	6	Std	7						48
Suppression Pool Cleanup Return	101	3.2-12 6.2-31o	56	B	FPC-V-156	MO Gate	O	AC	AC	F,A	RM	C	C	C	AS-IS	6	Std	3	No	W	Valves	R.B.	No	17,48
					FPC-V-149	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	6	-	41						48
RWCU From Reactor	14	3.2-11 6.2-31k	55	A	RWCU-V-1	MO Gate	I	AC	AC	A,J, E,W	RM	O	O	C	AS-IS	6	Std	-	No	W	Valves	Rad. W.	.35	
					RWCU-V-4	MO Gate	O	DC	DC	A,J, E,Y,W	RM	O	O	C	AS-IS	6	Std	4						
RRC Pump A seal Water	43A	3.2-3 6.2-31c	56	B	RRC-V-13A	Check	I	Process	Process	-	-	O	O	O	-	3/4	Std	-	No	W	Valves	R.B.	No	
					RRC-V-16A	MO Gate	O	AC	AC	45	RM	O	O	O	AS-IS	3/4	Std	2						
RRC Pump B seal water	43B	3.2-3 6.2-31c	56	B	RRC-V-13B	Check	I	Process	Process	-	-	O	O	O	-	3/4	Std	-	No	W	Valves	R.B.	No	
					RRC-V-16B	MO Gate	O	AC	AC	45	RM	O	O	O	AS-IS	3/4	Std	2						
RRC Sample Line	77Aa	3.2-3 6.2-31d	55	A	RRC-V-19	SO Globe	I	AC	Spring	A,C,	RM	C	C	C/O	C	3/4	<5	-	No	W.	Valves	T.B.	.05	
					RRC-V-20	SO Globe	O	AC	Spring	A,C,	RM	C	C	C/O	C	3/4	<5							

AMENDMENT NO. 27
November 1982

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Pent. No.	FSAR Fig. Nos.	GDC	Code Gp. (12)	Valve No.	Valve Type	Loc.	Pwr. to Open (5)	Pwr. to Close (5)	Iso. Sig. (9)	Back Up	Norm Pos. (10)	Shut-down Pos.	Post LOCA	Fail. Pos. (6)	Vlv. Sz. (14)	Close. Time (7) (11)	Dist. to Pent.	Leads to ESF Sys.	Proc. Fld.	Leak Bar. (13)	Term. Zone (13)	Pot. By-pass Leak. (SCFH)	Notes
Drywell Equipment Drain	23	3.2-9 6.2-31k	56	B	EDR-V-19	AO Gate	O	Air	Spring	F,A	RM	O	O	C	C	3	Std	2	No	W	Valves	R.B.	No	17
					EDR-V-20	AO Gate	O	Air	Spring	F,A	RM	O	O	C	C	3	Std	4						
Drywell Floor Drain	24	3.2-10 6.2-31k	56	B	FDR-V-3	AO Gate	O	Air	Spring	F,A	RM	O	O	C	C	3	Std	2	No	W	Valves	R.B.	No	17
					FDR-V-4	AO Gate	O	Air	Spring	F,A	RM	O	O	C	C	3	Std	3						
Decontamination Soltn. Supply Header	94	3.2-10	NA	B	-	-	-	-	-	-	-	-	-	-	-	4	-	-	-	-	Blanked Close	R.B.	No	
Decontamination Soltn. Return Header	95	3.2-10	NA	B	-	-	-	-	-	-	-	-	-	-	-	4	-	-	-	-	Blanked Close	R.B.	No	
CIA for Safety Relief Valve Accumulators	56	3.2-21 6.2-31c	56	B	CIA-V-21	Check	I	Process	Process	-		C	C	C	-	3/4	-	-	No	A	Valves	R.B.	No	
					CIA-V-20	MO Globe		AC	AC	41	RM	O	O	O	AS-IS	3/4	Std	10						
CIA Line A for ADS Accumulators	89B	3.2-21 6.2-31c	56	B	CIA-V-31A	Check	I	Process	Process	-		C	C	C	-	1/2	-	-	No	A	Valves	R.B.	No	
					CIA-V-30A	MO Globe		AC	AC	42	RM	O	O	O	AS-IS	1/2	Std	15						
CIA Line B for ADS Accumulators	91	3.2-21 6.2-31c	56	B	CIA-V-31B	Check	I	Process	Process	-		C	C	C	-	1/2	-	-	No	A	Valves	R.B.	No	
					CIA-V-30B	MO Globe		AC	AC	42	RM	O	O	O	AS-IS	1/2	Std	15						
CRD Insert Lines (185 separate lines)	9	3.2.4	56	B						See Note 4														
CRD Withdrawal lines (185 separate lines)	10	3.2.4	56	B						See Note 4														

TABLE 6.2-16 (Continued)

33. The RCIC minimum flow valve is open only between the time of system initiation and the time at which the system flow to the RPV exceeds 40 gpm. The valve is shut at all other times. RCIC-V-19 auto closes when the turbine throttle valve is closed following a turbine trip (see note 32). Should a leak occur when the valve is open, it will be detected by a high level alarm in the appropriate reactor building sump.

34. The RCIC injection valve is open only during RCIC turbine operation. Injection line check valves on either side of the containment provide immediate leak isolation, if required. RCIC-V-13 auto closes when the turbine throttle valve is closed following a turbine trip (see note 32).

35. The RCIC steam exhaust valve, RCIC-V-68, is normally open at all times. Should a leak occur, it would be detected and alarmed by the RCIC room high temperature leak detection system (see note 32).

36. The RCIC vacuum pump discharge valve, RCIC-V-69, is normally open at all times. The valve could be remote-manually closed by the operator upon indication that vacuum (annunciated in Main Control Room) can no longer be maintained in the barometric condenser.

37. System isolation valves are normally closed. System is placed in operation only if the hydrogen monitors detect hydrogen buildup after a LOCA.

    The four CAC lines comprising one process loop are all powered from the same 1E division, including both inner and outer isolation valves for each of these four lines. The four CAC lines serving the other hydrogen recombiner loop are powered from the alternate 1E division. Thus, the loss of either 1E division will affect only one CAC loop. A single electrical failure could conceivably cause both inner and outer isolation valves to open spuriously, but containment isolation would not be lost since the CAC System is a closed system outside containment per 6.2.4.3.2.2.3.1 and NRC question 022.035 (Appendix D).

TABLE 6.2-16 (Continued)

38. The minimum flow valve for an ECCS pump is open only between time of ECCS initiation and the time at which the system flow to the RPV exceeds 640 gpm. The valve is shut at all other times. Should a leak occur when the valve is open, it will be detected by a high level alarm in the appropriate reactor building sump.

39. Valve is open only during steam condensing mode. Valve position is provided in main control room to provide the operator confirmation of valve status.

40. Normally closed. Signalled to open if reactor building pressure exceeds wetwell pressure by 0.5 psid. Valves automatically reshut when the above condition no longer exists. Operator to use valve position indicator as confirmation of valve status.

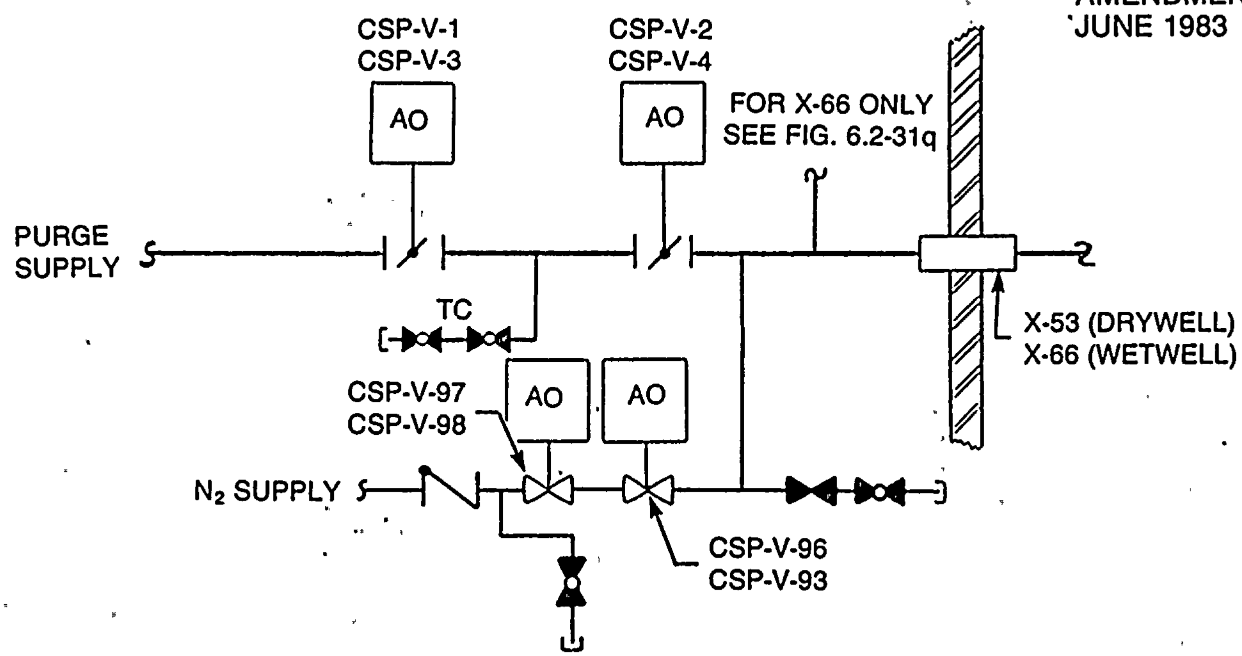
41. Indication of containment air compressor discharge header pressure and a low pressure alarm exist in the main control room. The operator can remote-manually shut valve CIA-V-20 should the containment air compressors become unavailable. The isolation check valve, CIA-V-21, provides immediate isolation.

42. Indication of nitrogen bottle header pressure and a low pressure alarm exist in the main control room. The operator can remote-manually shut valve CIA-V-30(A, B,) should the nitrogen bottle bank pressure decrease below the alarm setpoint. The isolation check valves, CIA-V-31(A, B) provide immediate isolation.

43. The operator's indication that remote-manual closure of the TIP shear valves is required, is failure of the TIP ball valves to close as monitored on Panel S.

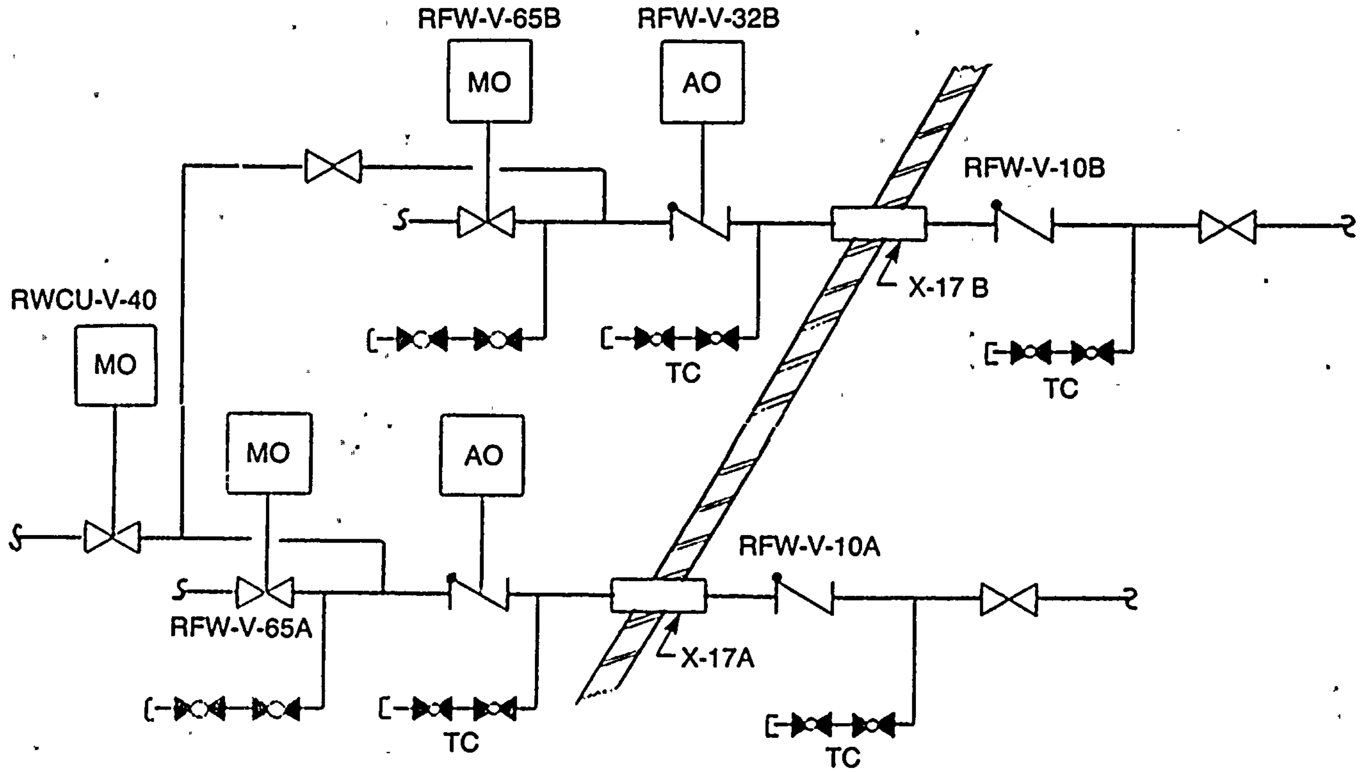
44. Normally closed. Opened only when testing wetwell-to-drywell vacuum breakers.

45. The isolation valve can be remote-manually closed upon indication that the CRD or the RRC pumps have been tripped. The isolation check valves, RRC-V-13 (A, B,), provide immediate isolation.



NOTE: SEE NOTE 4 ON FIG. 6.2-31a

X-53 DRYWELL PURGE SUPPLY & INERTING MAKEUP
X-66 WETWELL PURGE SUPPLY & INERTING MAKEUP



NOTE: SEE NOTE 1 ON FIG. 6.2-31a

REACTOR FEEDWATER LINES

WASHINGTON PUBLIC POWER **SUPPLY SYSTEM** NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-53, X-66, X-17A & X-17B	FIGURE 6.2-31b

6.4 HABITABILITY SYSTEMS

6.4.1 DESIGN BASIS

The main control room habitability systems are designed to ensure habitability inside the main control room during all normal and abnormal station operating conditions, including 30 days of habitability following a LOCA, in compliance with Criterion 19 of 10CFR50, Appendix A. Within the control room, sufficient food, water, medical supplies, sanitary and hygenic facilities are stored to sustain a crew as indicated in Table 13.1-1 for at least 5 days. Portable breathing apparatus is also provided in the control room for operating personnel protection in the event of a fire or a chemical (chlorine) spill on or off site. Redundant chlorine detectors with sensors are located in the common intake header. Controls are provided for the automatic isolation of the control room in the event that the chlorine concentration exceeds the sensor setpoints.

In the event of a LOCA, operating personnel within the control room are protected from airborne radioactivity by means of pressurizing the control room, with filtered air drawn from either of two separate remote fresh air intakes. Both intakes are physically remote from all plant structures. Redundant radiation monitors sensing the radiation level at each of the two remote intake headers are provided. The valves in each remote intake close automatically as the radiation level at the intake rises above the allowable level. Adequate shielding is also provided to protect operating personnel from radiation streaming. The control room doors are adequately designed to protect operating personnel from steam pipe break in the turbine generator building.

All components of the HVAC systems serving the control room that are required to ensure control room habitability and essential equipment operations are redundant, Seismic Category I and powered from Class 1E buses.

6.4.2 SYSTEM DESIGN

6.4.2.1 Definition of Main Control Room Envelope

The main control room envelope is shown in Figure 6.4-1. It is located on elevation 501 of the control building. Included in the control room envelope are all essential control equipment of the plant plus sanitary and kitchen facilities. The envelope includes the control room area, kitchenette, dining area, office area and computer peripherals. The control room and

office areas are continuously occupied. The computer peripherals, kitchenette and dining area are frequently occupied. The selection of spaces in the control room envelope is based on need during postulated emergencies. The HVAC equipment rooms (located on elevation 525' above the main control room) are not in the control room envelope and are not serviced by the control room habitability systems. The control room HVAC equpment and its associated ductwork in the HVAC equipment rooms are, however, included as part of the control room envelope. The enclosed volume of the control room envelope is approximately 204,000 cubic feet.

6.4.2.2 Ventilation System Design

A detailed description of the ventilation systems serving the control room and a listing of the design and performance parameters of the ventilation system equipment is given in 9.4.1. A schematic of the control room ventilation system is presented in Figure 9.4-1. The system is composed of two 100 percent capacity air conditioning systems, two 1000 cfm capacity emergency charcoal filter units and associated ductwork, dampers, controls, and isolation valves.

Suitable temperature conditions are maintained in the control room during both normal and emergency conditions by either of two air handling units. The air handling units of the air conditioning system normally cool and pressurize the control room by drawing 1000 cfm of fresh air from a local fresh air intake, while recirculating 21,000 cfm of room air through roughing filters and cooling coils. The local fresh air intake is fitted with two fail close isolation valves in series. During normal operation, chilled water is supplied to the air handling units from the radwaste building chilled water system. The chilled water system is the only portion of the control room habitability system which is not Seismic Category I. In the event that the radwaste chilled water system is not available, standby service water is supplied to the air handling units for emergency cooling. In this mode of operation the control room ambient temperature will be maintained at 104°F which is sufficient to ensure critical equipment operation and area habitability. (It is anticipated that with the chilled water system inoperative, temperatures within the control room would be maintained appreciably lower than 104°F. However, under the worst hypothesized conditions of standby service water temperature, one control room air handling unit inoperative (single failure) and maximum may design environmental conditions, a control room temperature rise to 104°F for short periods of time is possible.

All of the above components are mounted in an all welded steel housing.

There are 2268 pounds of charcoal in each of the two (2) adsorber units. The adsorbing capability of each unit is 2.5 milligrams of halogens per gram of charcoal or a total of 2,572 grams. The maximum theoretical accumulation of halogens on the SGTS adsorbers for a 30-day period after a LOCA (in accordance with Regulatory Guide 1.3, Revision 2, June 1974) is 940 grams.

Three independent deluge spray systems are provided for fire protection in each standby gas treatment system filter train. One deluge spray system is provided for protection of the prefilter and a deluge spray system is provided for each of the two charcoal filter beds.

Two full capacity, 4457 cfm, centrifugal fans are provided with each standby gas treatment system filter train. One fan is the units primary fan which starts automatically in the event of a LOCA. The second fan is a standby fan which operates only in the event of primary fan failure. The two fans of each unit are powered from separate emergency diesel buses. Automatic inlet vanes are provided on all fans for fan capacity control. Ductwork and butterfly valves on the discharge air side of each filter train are arranged in such a manner that either fan can draw air through the filter train and discharge it either out of the reactor building, via the reactor building elevated release duct, or recirculate it back into the reactor building. Provision is made to recirculate air back into the reactor building so that decay heat, which may be generated within the SGTS unit due to the collection of radioactive contaminants, is removed by recirculating air while a unit is in standby.

Ductwork and valving are arranged on the intake of each SGTS unit in such a manner that the units draw air from the reactor building in the immediate vicinity of the unit or from the primary containment drywell and/or wetwell via duct connections to the primary containment purge exhaust lines (See 9.4.2).

Both standby gas treatment system filter units are located on elevation 572 of the reactor building. A twelve-inch thick concrete partition wall, fourteen feet high, separates the

two units. The partition wall, which is of Seismic Category I design, serves as both a missile barrier and fire barrier between the two units.

During normal plant operation, both SGTS units are on standby. In this mode the only portions of the system which are operational are the strip heaters in the filter units, which are cycled on and off by thermostats set to maintain the filter plenum at 90°F to ensure that the relative humidity within the plenums never exceed 70%, thus protecting the charcoal adsorber from condensed moisture.

The maximum dew point temperature in the reactor building during normal plant operation is 75°F. When in standby, all isolation valves downstream of the unit fans are closed.

In the event that a purge of the primary containment is required, but radiation monitors within the containment indicate that radiation levels are too high for direct purge through the reactor building exhaust system, the purge exhaust can be performed through the SGTS. All controls for the SGTS are located in the main control room from where a control room operator starts the SGTS fans and opens all isolation valves between the containment and SGTS and atmosphere. Purge flow rate is recorded and adjusted by means of electronic flow recording controllers mounted in the control room which transmit control signals to the inlet vanes of the SGTS fans. The sensor for each flow recording controller is in the discharge duct of the fan controlled. Purge supply air to the primary containment is supplied from the reactor building supply air system (See 9.4.2). During primary containment purge both SGTS units can be operated if a purge rate greater than 4457 cfm is desired.

Upon completion of containment purge all primary containment isolation valves are closed and the SGTS inlet valve from the secondary containment is opened. Air from the secondary containment is then drawn through the SGTS unit and discharged to atmosphere. This operation is performed at a reduced flow rate (approximately 500 cfm) through a recycle timer to dissipate the decay heat from radioactive contaminants collected in the filters. The recycle timer is capable of energizing the SGTS unit fan for 1 to 15 minutes every 30 minutes to 3 hours. The cooling operation is terminated and the SGTS unit returned to the standby mode when the radiation level at the filters decays to background levels as determined by the use of portable radiation monitors.

All of the above components are mounted in an all welded steel housing.

There are 2268 pounds of charcoal in each of the two (2) adsorber units. The adsorbing capability of each unit is 2.5 milligrams of halogens per gram of charcoal or a total of 2,572 grams. The maximum theoretical accumulation of halogens on the SGTS adsorbers for a 30-day period after a LOCA (in accordance with Regulatory Guide 1.3, Revision 2, June 1974) is 940 grams.

Three independent deluge spray systems are provided for fire protection in each standby gas treatment system filter train. One deluge spray system is provided for protection of the prefilter and a deluge spray system is provided for each of the two charcoal filter beds.

Two full capacity, 4457 cfm, centrifugal fans are provided with each standby gas treatment system filter train. One fan is the units primary fan which starts automatically in the event of a LOCA. The second fan is a standby fan which operates only in the event of primary fan failure. The two fans of each unit are powered from separate emergency diesel buses. Automatic inlet vanes are provided on all fans for fan capacity control. Ductwork and butterfly valves on the discharge air side of each filter train are arranged in such a manner that either fan can draw air through the filter train and discharge it either out of the reactor building, via the reactor building elevated release duct, or recirculate it back into the reactor building. Provision is made to recirculate air back into the reactor building so that decay heat, which may be generated within the SGTS unit due to the collection of radioactive contaminants, is removed by recirculating air while a unit is in standby.

Ductwork and valving are arranged on the intake of each SGTS unit in such a manner that the units draw air from the reactor building in the immediate vicinity of the unit or from the primary containment drywell and/or wetwell via duct connections to the primary containment purge exhaust lines (See 9.4.2).

Both standby gas treatment system filter units are located on elevation 572 of the reactor building. A twelve-inch thick concrete partition wall, fourteen feet high, separates the

two units. The partition wall, which is of Seismic Category I design, serves as both a missile barrier and fire barrier between the two units.

During normal plant operation, both SGTS units are on standby. In this mode the only portions of the system which is operational are the strip heaters in the filter units, which are cycled on and off by thermostats set to maintain the filter plenum at 90°F to ensure that the relative humidity within the plenums never exceed 70%, thus protecting the charcoal adsorber from condensed moisture.

The maximum dew point temperature in the reactor building during normal plant operation is 75°F. When in standby, all isolation valves downstream of the unit fans are closed.

In the event that a purge of the primary containment is required, but radiation monitors within the containment indicate that radiation levels are too high for direct purge through the reactor building exhaust system, the purge exhaust can be performed through the SGTS. All controls for the SGTS are located in the main control room from where a control room operator starts the SGTS fans and opens all isolation valves between the containment and SGTS and atmosphere. Purge flow rate is adjusted by means of electronic flow controllers mounted in the control room which transmit control signals to the inlet vanes of the SGTS fans. The sensor for each flow controller is in the discharge duct of the fan controlled. Purge supply air to the primary containment is supplied from the reactor building supply air system (See 9.4.2). During primary containment purge both SGTS units can be operated if a purge rate greater than 4457 cfm is desired.

Upon completion of containment purge all primary containment isolation valves are closed and the SGTS inlet valve from the secondary containment is opened. Air from the secondary containment is then drawn through the SGTS unit and discharged to atmosphere. This operation is performed at a reduced flow rate (approximately 500 cfm) through a recycle timer to dissipate the decay heat from radioactive contaminants collected in the filters. The recycle timer is capable of energizing the SGTS unit fan for 1 to 15 minutes every 30 minutes to 3 hours. The cooling operation is terminated and the SGTS unit returned to the standby mode when the radiation level at the filters decays to background levels as determined by the use of portable radiation monitors.

Both SGTS filter units are automatically actuated in the event of any of the following three isolation signals:

a. High radiation in the reactor building ventilation exhaust duct

b. High pressure in the drywell

c. Reactor vessel low water level

When actuated the following sequence of events occur in each SGTS train:

a. The primary bank of electric blast coil heaters is energized

b. A time delay starts the primary fan and opens the isolation valves on the inlet and outlet of the SGTS 10 seconds after heater start

Fan start is delayed 10 seconds after energizing the heater so that the heaters reach a sufficiently high temperature to ensure that the air entering the charcoal bed does not exceed 70% relative humidity. Once the fan is started, the three stages of electric heat are controlled by humidistats sensing the relative humidity of the air entering the heater bank.

Both SGTS units are fully operational within 34 seconds after an emergency signal. The heaters are energized 20 seconds after the emergency diesel generators are given a start signal. The fans start and valves open 10 seconds after heater start with the valves stroked full open. in 4 seconds. The secondary containment pressure transient upon isolation of normal reactor building ventilation and startup of the SGTS following the postulated design basis LOCA is presented in Table 6.2-12. The secondary containment is normally maintined at a negative pressure by the reactor building ventilation system (See 9.4.2).

When started automatically by one of the above three isolation signals, the fans inlet vane position are automatically controlled by the secondary containment pressure control system. There are two, emergency powered, Seismic Category I, control systems. Each is composed of four differential pressure controllers which transmit a control signal to the fan inlet vanes to maintain the lowest of the above differential pressures at a minimum of 0.25 inches of water, negative.

In the event of primary fan failure a flow switch on the primary fan discharge automatically closes the isolation valve on the primary fan intake, deenergizes the primary heater bank, energizes the secondary heater, opens the secondary fan isolation valves, and starts the standby fan.

The plant operator may stop one of the SGTS units from the control room after start up is complete. In the event that the radiation monitors in the discharge duct indicate an unaccepatable radiation level in the system discharge air the operator starts the second unit and diverts the discharge air of the operating unit back into the reactor building to minimize off-site release of halogens, and/or for cooling of the charcoal bed.

The following is a comparison of the ESF filteration systems with each position detailed in Regulatory Guide 1.52, Rev. 2.

Article A - "Introduction"

The engineered safety feature filtration systems provided for WNP-2 are designed to the General Design Criterion referenced in Article A. Those systems which are designed to meet the criterion are as follows:

a. Standby Gas Treatment System (6.5.1)

b. Control Room Emergency Filter System (9.4.1)

Article B - "Discussion"

The above two systems are both classed as secondary systems and are not subjected to the drywell environment during any design basis accident and are not subjected to containment cooling sprays. Equipment design includes the ability to operate under all environmental conditions to which they can be subjected during accident conditions. The components of each control room filter unit are as described in this article except that no demisters are required and HEPA filters are not provided downstream of the charcoal adsorber section. The effects of aging, weathering and relative humidity have been considered in the design of these atmosphere cleanup systems, and are tested periodically to verify the required performance capability.

The effects of moisture on the charcoal adsorber media is minimized by the use of strip heaters for humidity control in the plenum of the charcoal adsorbers section of the SGTS units and by periodically circulating heated air through the control room filter units. Adequate space and accessibility for personnel has been incorporated in the filter unit designs to ensure maintainability and testability. Testing of all filters is performed to meet the objectives of Regulatory Guide 1.52, Rev. 2.

Article C-"Regulatory Position"

Table 6.5-2 provides an analysis of the engineered safety feature air filtration systems with respect to the regulatory positions of Regulatory Guide 1.52, Rev. 2.

6.5.1.3 Design Evaluation

A design evaluation of the control room emergency filter units is given in 9.4.1.

The standby gas treatment system is designed to prevent the exfiltration of contaminated air from the secondary containment following an accident or abnormal occurrence which could result in high airborne radiation in the secondary containment. All necessary equipment and surrounding structures are Seismic Category I. The engineered safety features buses supply power to the SGTS in the event of loss of normal a-c power. Two fully redundant equipment trains separated by a missile wall are provided to ensure that a single failure does not impair or preclude system operation. A standby gas treatment system failure analysis is presented in Table 6.5-3.

6.5.1.4 Tests and Inspections

The test and inspection program applicable to the control room emergency filter units is discussed in 9.4.1.

The standby gas treatment system and its components are thoroughly tested in a program consisting of the following classifications:

a. Predelivery tests and component qualification tests

b. Post-delivery acceptance tests

c. Post-operation surveillance tests.

Written test procedures establish acceptance criteria for all tests. Test results are recorded in performance records, thus enabling early determination of end life performance.

All predelivery, post-delivery and post-operation tests are performed to meet the objectives of Regulatory Guide 1.52, Rev. 2.

HEPA filters are factory tested to a minimum efficiency of 99.97% when measured with a 0.3 micron DOP aerosol.

Charcoal media qualification tests meet the objectives of Regulatory Guide 1.52, Rev. 2.

Once charcoal media is installed in the filter frames, at the site, each charcoal bed is leak tested with refrigerant 112. R-112 is injected upstream of the charcoal beds at a concentration of 20 ppm at rated flow. A gas chromatograph is used to sample the R-112 concentration downstream of the filter assembly. The maximum allowable concentration of R-112 downstream of the filter is 0.01 ppm. Concentrations greater than 0.01 ppm constitute failure of the filter and the filter will be repaired and retested. This test is performed semiannually at the same time the HEPA filters are tested.

Twelve 4 inch deep test canisters are installed in parallel with each of the charcoal adsorber sections. Once a year one test canister from each adsorber section is removed, with opening blanked off, and sent to a laboratory for testing. Each sample is tested with methyl iodide per RDT M16-1T as defined in Regulatory Guide 1.52, Rev. 2. In the event a sample fails to meet this test, the charcoal media in that adsorber section will be replaced.

All SGTS fans are factory tested in accordance with AMCA Standard 210 "Air Moving and Conditioning Association Test Code for Air Moving Devices". Once installed, fans are started once per month to ensure operability.

All valves associated with the SGTS are factory leak tested, bubble tight, at a pressure differential of 2 psig. All valves are factory tested to ensure that valve stroke time, full close to full open, does not exceed 4 seconds. Once installed, the valves of the SGTS are stroked at least once every six months to ensure operability.

6.5.1.5 Instrumentation Requirements

The instrumentation and control system of the control room emergency filter units are discussed in 7.3.1.1.7 and 9.4.1. The instrumentation and control system of the SGTS is discussed in 7.3.1.1.9.

All instrumentation and controls are designed to meet the objectives of Regulatory Guide 1.52, Rev. 2.

The following instrumentation is provided for each SGTS train in addition to that described in 6.5.1.2.

a. An indicating differential pressure gauge is provided across each element (excluding heaters) in the SGTS train. High differential pressure annunciates an alarm in the main control room and is permanently recorded by the computer.

b. Relative humidity detectors, with humidity indication in the main control room, are located before the electric blast coil heaters and before the charcoal adsorber banks. High humidity annunciates an alarm in the main control room and is permanently recorded by the computer.

c. Thermostats with sensors on either side of an adsorber section control strip heaters in both adsorber plenum sections. Two thermostats in parallel energize the heaters to maintain a minimum temperature of 90°F. Another thermostat deenergizes the heaters on a temperature rise to 110°F with a manual reset thermostat cutting out the heaters on a temperature rise to 125°F.

d. Temperature indication is provided in the main control room for air entering the electric blast coil heater section and the air leaving both banks of charcoal filters.

e. The deluge spray systems of the prefilter and both charcoal adsorber sections are each controlled separately by temperature switches.

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Temperature switch sensors are located on the down-stream side of the prefilter and adsorber sections. On a temperature rise to 210°F the temperature switch annunciates an alarm in the main control room. The control room operator determines the cause of the temperature rise and manually initiates the deluge spray system if a fire condition exists or is imminent.

6.5.1.6 Materials

The housings and all framing materials of the standby gas treatment system filter units are fabricated of steel alloys and as such are non-flammable. Following is a list of the materials used in the various components of the SGTS filter units.

Demisters - The demister (moisture separator) section of each SGTS unit consists of four assemblies of metal baffle plates and fiberglass separator pads. Each assembly has 3 two inch thick fiberglass pads and 1 four inch thick galvanized metal moisture eliminator with a nominal face area of 16 x 20 inches.

Prefilters - There are four 24" x 24" prefilters in each SGTS unit. The media is Farr Co. HP 100 filter cartridges which is a pleated, U.L. Class 1, fiberglass mounted on a metal retainer frame.

Absolute Particulate Filters - There are two banks, one before and one after the charcoal adsorber section, of HEPA filters on each SGTS filter unit. The filters are manufactured by the Farr Co. and consist of U.L. Class 1 fiberglass media in cadmium plated frames with aluminum separators. There are four 24" x 24" filters in each filter bank.

Charcoal Adsorber Media - Each charcoal adsorber section (two per SGTS unit) contains 40.5 cubic feet of North American 727 coconut base charcoal. At an approximate density of 28 lbs per cubic foot a total of 2,268 lbs of charcoal is provided in each SGTS unit.

The only material in the SGTS units which supports combustion is the charcoal which has a minimum ignition temperature of 340°C. The charcoal is provided with a deluge spray system to prevent fire. A twelve inch thick concrete partition wall is provided between the two SGTS units to protect them in the unlikely event of a fire in one unit.

The housing and framing of the control room emergency filter units are constructed of steel. Each contains one prefilter and one HEPA filter of identical design to those of the SGTS units and one tray type charcoal filter assembly. The charcoal filter assembly contains 4.5 cubic feet or approximately 126 pounds of North American 727 coconut base charcoal. The two control room emergency filter units are located outside the control room, are separated by a fire wall, and are furnished with deluge spray systems to prevent fire.

6.5.2 CONTAINMENT SPRAY SYSTEM

6.5.2.1 Design Bases

The containment spray system is capable of quickly reducing containment pressure during the post-accident period of a LOCA through condensation of steam in the drywell and through cooling of the non-condensable gases in the free volume above the suppression pool. Containment spray is not required to prevent overpressurization of the containment (see 6.2.1). The containment spray system is not used for fission product removal from the containment atmosphere.

6.5.2.2 System Design

The drywell spray consists of two independent loops and spray headers. The suppression chamber spray consists of one spray header supplied from two otherwise independent loops. Since the water source for all containment spray is the suppression pool, the spray system is a closed loop cooled by the RHR heat exchangers. The rated flows for drywell and suppression chamber sprays are 7450 gpm/loop and 450 gpm/loop, respectively. Containment spray is a subsystem of the RHR System (5.4.7).

The drywell spray valves are electrically interlocked to allow actuation of the drywell spray only when there is a high drywell pressure signal present. After a high drywell pressure signal is present, a second electrical interlock prevents actuation of either the drywell or the suppression chamber spray lines until the corresponding LPCI injection valve is shut.

A procedural restriction prohibits the operators during the first ten minutes following a LOCA from closing a LPCI injection valve and interrupting core cooling (Refer to 6.2.2.2). Containment spray must be initiated and secured by operator action. Hydrogen mixing which results from containment spray operation is discussed in 6.2.5.

The suppression chamber spray header location is shown in Figure 3.5-3. The lower drywell spray header location is shown in Figures 3.5-4, 3.5-21, 3.5-22, and 3.5-25 through 3.5-28. The upper drywell spray header is shown in Figures 3.5-16 through 3.5-18.

6.7 MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM (MSIV-LCS)

The MSIV-LCS controls and minimizes the release of fission products which could leak through the closed main steam isolation valves (MSIV's) after a LOCA. The leakage is directed through a bleed line into an area served by the standby gas treatment system (SGTS) for processing prior to release to atmosphere.

6.7.1 DESIGN BASIS

6.7.1.1 General Criteria

The following general design criteria represent system design requirements.

a. The offsite dose rate does not exceed the guidelines of 10 CFR Part 100 following a design basis LOCA resulting from a complete severance of the recirculation line.

b. The fission product release model is based upon TID 14844.

c. The MSIV-LCS does not prevent the SGTS from performing its function.

d. Steam discharge from the MSIV-LCS is directed such that it does not affect functioning of structures, systems, or components important to safety.

e. The MSIV-LCS has been designed in accordance with Regulatory Guide 1.96, Revision 1, with the exceptions noted in 6.7.5.1.1.1.3.4.

6.7.1.2 Specific Criteria

The following specific criteria delineate system performance requirements.

a. The MSIV-LCS and necessary subsystems are designed in accordance with Seismic Category I requirements.

b. The MSIV-LCS and necessary subsystems are capable of performing their safety function, when necessary, considering the design basis LOCA effects including: (1) internally generated missiles, (2) the dynamic effects associated with pipe whip and jet forces from the event and (3) normal operating and accident-caused local environmental conditions consistent with the event.

c. The MSIV-LCS is capable of performing its intended function following any single active component failure (including failure of any one of the main steam line isolation valves to close).

d. The MSIV-LCS is capable of performing its intended function following a loss of all offsite power coincident with the postulated design basis LOCA.

e. The MSIV-LCS is designed with sufficient capacity and capability to control the leakage from the main steam lines consistent with containment integrity under the conditions associated with the postulated design basis LOCA.

f. The MSIV-LCS is manually initiated and is designed to permit actuation at any time following the postulated design basis LOCA.

g. Instrumentation and controls necessary for the functioning of the MSIV-LCS are designed in accordance with standards applicable to nuclear plant safety-related instrumentation and control systems.

h. The MSIV-LCS controls are provided with interlocks actuated from appropriately designed safety systems or circuits to prevent inadvertent MSIV-LCS operation.

i. The MSIV-LCS is designed to permit testing of the operability of controls and actuating devices during power operation to the extent practical, and testing of the complete functioning of the system during plant shutdowns.

j. The MSIV-LCS is designed so that effects resulting from a sealing system single active component failure will not affect the integrity of the main steam lines or MSIV's.

6.7.1.3 Codes and Standards

The detailed design and construction criteria are provided by published codes, standards and regulatory guides. All piping systems and components for the MSIV-LCS comply with the applicable codes, addenda, code cases and errata in effect at the time the equipment is procured. Currently in effect is the ASME Boiler and Pressure Vessel Code - Section III, Nuclear Power Plant Components. The piping and components at the point of connection to the main steam line including the reactor pressure retaining system valves are Class 1. All other piping and components are Class 2. Subsections NA, NB, and NC of the Code apply to the MSIV-LCS.

The equipment and piping of the MSIV-LCS, in order to meet specified seismic capabilities, are designed to Seismic Category I requirements. This category includes all structures and equipment essential to the safe shutdown and isolation of the reactor, or whose failure or damage could result in undue risk to the health and safety of the public.

Refer to Table 5.2-4, Reactor Coolant Pressure Boundary Materials for a presentation of the specifications which generally apply to the selection of materials used in the MSIV-LCS. Nonmetallic materials such as lubricants, seals, packings, paints and primers, insulation, metallic materials, etc., are selected as a result of an engineering review and evaluation for compatibility with other materials in the system and the surroundings with concern for chemical, radiolytic, mechanical and nuclear effects.

Refer to 7.3.2 for a complete description of Standards and Regulatory Guides applicable to the MSIV-LCS instrumentation and controls.

6.7.2 SYSTEM DESCRIPTION

6.7.2.1 General Description

The MSIV-LCS is designed to minimize the release of fission products which could by-pass the standby gas treatment system (SGTS) after a LOCA. This is accomplished by directing leakage from the closed main steam isolation valves (MSIV's) through a bleed line and into an area served by SGTS. The flow is effected by a small blower which maintains the pressure in the steam lines negative with respect to atmosphere, thus ensuring that the MSIV leakage will pass through the blower and on into the SGTS prior to release to the atmosphere.

The flow diagram of the MSIV-LCS is shown in Figure 3.2-25. As indicated, two independent systems (an outboard system and an inboard system) are provided to accomplish the leakage control function. The inboard system receives power from one division and the outboard system from the other division of the two redundant critical electrical power supply divisions.

The outboard system is connected to the segments of the main steam lines between fast closing MSIV's outside containment and the downstream block valves. The bleed line from each main steam line connects to a bleed header. The bleed header outlet is provided with two valves in series to permit the main steam lines to be depressurized by venting, following a LOCA. A parallel set of valves is provided which are opened following depressurization to connect the blower sunction to the steam lines. Pressure sensors are also used for depressurization interlock control to prevent any accidental actuation of the system during normal reactor operation. Another pressure sensor is used for interlock control on the valves in the blower suction line to prevent accidental actuation when pressure is appreciably greater than atmospheric. Pressure indicators are provided for monitoring the pressure in the main steam lines between the fast closing MSIV's outside containment and the downstream block valves. The major flow to the blower suction is dilution air from the reactor building. This dilution air greatly reduces the temperature of the MSIV leakage as it passes through the blower.

The inboard system is connected to the segments of the main steam lines between the fast closing MSIV's outside and inside the containment. An individually controlled bleed line is provided for each main steam line. For each bleed line two bleed vlaves, which are of the motor operated gate type, are connected in series followed by a flow element and a motor operated bypass valve. Flow through the four flow elements passes to a common blower which discharges to a building volume served by the SGTS. Discharge through the flow element bypass valve is similarly routed to an area served by the SGTS. Pressure sensors are used for interlock control to prevent any accidental actuation of the system during normal reactor operation. In addition, the bleed valves and bypass valves are interlocked with their respective inboard MSIV to prevent their opening unless the MSIV is closed. Pressure indicators are provided for monitoring the pressure in the main steam lines between the fast closing MSIV's outside containment and those inside containment. Delay timers and pressure sensors are used for reclosing bleed valves after the inboard system is activated, in case of gross leakage through the inboard MSIV. Other timers, together with high flow limiters, are used to monitor and reclose the valves should the total leakages through both MSIV's exceed the high flow set point. Electric heaters are used, one at the low point of each bleed line, to boil off any condensate and pass it through the flow limiter. A flow indicator is provided to monitor dilution flow.

Manual switches are provided for functional testing of the bleed valves. The bleed line depressurization branches are terminated at a location in the vicinity of the SGTS inlet connections where steam can be discharged, while depressurizing the steam lines, without adversely affecting safety related equipment. The blower discharge lines also terminate at a location where the discharge flow will pass to the standby gas treatment system.

For each system, a dilution air flow indicator is provided to monitor blower flow rate. A timer is used to actuate a high steam line pressure alarm within a pre-set time period after system actuation, if a sub-atmospheric main steam line pressure is not established.

6.7.2.2 System Operation

The MSIV-LCS is actuated manually (both inboard and outboard system) after it has been ascertained that a LOCA has occurred and after pressure in the steam lines is below the pressure permissive interlock set point.

In the outboard system, the valves in the depressurization branch line are opened to permit the steam lines beyond the fast closing MSIV's outside containment to depressurize, and the blower is started. When the steam lines have depressurized to approximately atmospheric pressure, the valves in the branch line to the blower are automatically opened and the valves in the depressurization branch are automatically closed. This establishes a sub-atmospheric pressure in the steam lines and the MSIV leakage is routed to an area served by the SGTS. If a sub-atmospheric pressure is not established in the main steam lines within the time required to depressurize, the timer will actuate a high pressure alarm, indicating failure of the system to function.

Inboard system actuation automatically depressurizes the main steam lines through the flow limiter by-pass valves which reclose automatically to route the flow through the flow limiter. The electric heaters are turned on from the system actuation signal. The blower also starts from the system actuation signal and establishes a sub-atmospheric pressure in the main steam lines after the flow limiter by-pass valves reclose. MSIV leakage is routed to areas served by the standby gas treatment system during both depressurization and exhaust stages of operation.

The inboard system is designed to automatically reclose in the event of excessive MSIV leakage. Automatic reclosure capability is provided on the bleed system for each individual steam line. Each steam line has its own bleed valves, electric heater, flow limiter, flow limiter by-pass valve, timers and pressure instrumentation. Assuming that the leakage through the isolation valves is at the technical specification value, the depressurization of the steam line volume between the MSIV's can be computed as shown in Figure 6.7-1 for a typical BWR. The depressurization rate is varied as a function of the equivalent length of the bleed-off line flow resistance.

those valves and piping necessary for pressure boundary isolation. The remaining equipment is located outside the steam tunnel and is removed and shielded from such effects by the concrete walls of the pipe tunnel. The inboard and outboard systems are physically separated. The equipment is designed to operate under the expected LOCA environmental conditions appropriate to the equipment location.

c. The MSIV-LCS remains functional following any active component failure (including failure of any one MSIV to close) by virtue of two redundant systems. The systems are independently powered from different divisions of the critical power supply.

d. The use of the critical power source to power the components of the system ensures system operation during a loss of offsite power.

e. A discussion of the activity released to the environment by way of the SGTS is contained in the radiological evaluation of the design basis LOCA, 15.6.5.

f. The manual initiation of the system need not occur until the pressure of steam trapped between the isolation valves decreases to the containment vessel pressure and abnormally high radiation is present. The pressure decay between the MSIV's due to MSIV leakage at an estimated maximum rate of 11.5 scfh @ 29 psid, is such that the line pressure will exceed vessel pressure for about one hour at which time the vessel and trapped line pressure will be about 35 psia. There would be no need to actuate any portion of the MSIV-LCS when the pressure between the valves is higher than containment pressure since clean steam, trapped between the valves when they closed, would be leaking toward the containment. The critical electrical buses are well able to provide the estimated 15 kW required for system operation.

g. The instrumentation necessary for control and status indication of the MSIV-LCS is classified as essential and as such is designed and quali-

fied, per IEEE 344-1971, IEEE 279-1971 and IEEE 323-1971, to function under Seismic Category I and LOCA environmental loading conditions appropriate to its location. The control circuits are designed to satisfy the mechanical and electrical separation criteria.

h. The system detects high steam line pressure and prevents system actuation. It also detects high leakage and prevents excessive release of leakage to the SGTS.

i. Components of the system downstream of the system isolation valves may be tested at any time during plant operation. The isolation valves may also be operated sequentially at any time during plant operation. Simultaneous operation of the isolation valves and leak testing can be performed only during reactor shutdown in order not to interfere with normal plant operation.

j. Double series system isolation valves are provided to ensure the integrity of the main steam lines.

k. In addition, this system, by exhausting leakage steam and gases, does not expose the steam piping or valves to thermal or mass loadings different from that experienced in normal isolation valve service and, therefore, cannot affect or degrade the sealing ability of the MSIV's.

The maximum process loads imposed by the MSIV-LCS are 80 lbs of steam @ 35 psig saturated initial conditions vented to the reactor building volume served by the SGTS, followed by the continuous MSIV leakage flow. The initial discharge will have no significant effect on building pressure buildup. The continuous flow is considered negligible compared to the SGTS rated flow. The MSIV-LCS conditions the exhaust temperature and humidity to the requirements of the SGTS prior to delivery toward the SGTS inlet.

TABLE 7.1-2 (Continued)

Instrumentation and Controls (System)	Plants Applying for or Having Construction Permit or Operating License	Similarity of Design
(14) (RHRS) Reactor Shutdown Cooling Mode	Zimmer-1	Identical
(15) Fuel Pool Cooling and Clean Up System	None	----------
(16) Standby Gas Treatment System	None	----------
(17) Main Steamline Isolation Valve Leakage Control System	None	----------
(18) Safety-Related Display Instrumentation	Zimmer-1	Identical
(19) Containment Instrument Air System	None	----------
(20) Reactor Bldg. Closed Cooling Water System	None	----------
(21) (RHRS)-Containment Spray Cooling Mode	Zimmer-1	Identical
(22) Remote Shutdown System	Zimmer-1	Note 1
(23) Recirculation Pump Trip	Zimmer-1	Identical
(24) (RHRS) Suppression Pool Cooling Mode	Zimmer-1	Identical

Note 1

The number of valves controlled is slightly different due to differences in the necessary shutdown capability.

TABLE 7.1-3

CODES AND STANDARDS APPLICABILITY MATRIX

	RPS	RCRVICS	ECCS	NMS	PROCESS RAD MON.	MAIN CONTROL ROOM HVAC EMERG SWGR RM	SERVICE WATER SYSTEM	RCIC	SLCS	CONTAIN. ATMOS. MON.	LEAK DETEC. SYSTEMS	RHR SHUT-DOWN COOL. MODE	SFPCS	SGTS	MSIVLCS	SAFETY-RELATED DISPLAY	RHR CONTAIN. SPRAY COOL. MODE	REMOTE SHUTDOWN	RPT	RHR SUPP. POOL COOL. MODE
GDC 1	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
GDC 2	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
GDC 3	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
GDC 4	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
GDC 10	X			X																
GDC 12	X																			
GDC 13	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
GDC 15	X			X																
GDC 19																		X		
GDC 20	X			X	X															
GDC 21	X			X	X															
GDC 22	X			X	X															
GDC 23	X			X	X															
GDC 24	X			X	X															
GDC 25	X			X																
GDC 26									X											
GDC 28				X																
GDC 29	X			X	X				X											
GDC 33			X																	
GDC 34							X				X	X								X
GDC 35			X				X				X									
GDC 37			X				X													X
GDC 38		X	X				X										X			X
GDC 40			X														X			X
GDC 41														X						
GDC 43						X								X						
GDC 44							X													
GDC 46							X													
GDC 50																				X
GDC 54		X	X	X					X		X	X			X		X			X
GDC 55		X	X	X				X	X			X			X					X
GDC 56		X	X					X	X			X					X			X
GDC 57																				

The HPCS is initiated automatically by either reactor vessel low water level (Trip Level 2) or drywell high pressure. The system is designed to operate automatically for at least 10 minutes without any actions required by the control room operator. Once initiated the HPCS logic seals-in and can be reset by the operator only when the initial conditions return to normal. Refer to Figure 7.3-8 for a schematic representation of the HPCS System Initiation logic.

Reactor vessel water level (Trip Level 2) is monitored by four redundant differential pressure switches. The switch contacts are arranged in a one-out-of-two twice logic arrangement to assure that no single event can prevent the initiation of the HPCS.

Initiation diversity is provided by drywell pressure which is monitored by four redundant pressure switches. The switches are electrically connected in a one-out-of-two twice logic arrangement to assure that no single instrument failure can prevent the initiation of the HPCS.

The HPCS components respond to an automatic initiation signal as follows (actions are simultaneous unless stated otherwise):

1. The HPCS Diesel Generator is signalled to start and its protective relays are bypassed. Once the Diesel is started it signals the start of its cooling water pump. See 6.3.1.1.8.2.

2. The HPCS pump motor is signalled to start.

3. The normally open pump suction from the condensate storage tank valve MO F001, is signalled to open.

4. The test return valves MO F010, MO F011 and MO F023 are signalled closed.

5. The HPCS injection valve MO F004 is signalled to open.

The HPCS pump discharge flow and pressure are monitored by pressure switches. If pump discharge pressure is normal but discharge flow is low enough that pump overheating may occur the minimum flow return line valve MO F012 is signalled open. The valve is automatically closed if flow is normal.

If the water level in the condensate storage tanks falls below a predetermined level, the suppression pool suction valve MO F015 automatically opens. When MO F015 is fully open the condensate storage tank suction valve MO F001 automatically

closes. Two level switches mounted on a Seismic Category I standpipe in the reactor building are used to detect low water level in the condensate storage tanks. Either switch can cause automatic suction transfer. The suppression pool suction valve also automatically opens if high water level is detected in the suppression pool. Two level switches monitor suppression pool water level and either switch can initiate opening of the suppression pool suction valve. To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other closes.

The HPCS provides makeup water to the reactor until the vessel water level reaches the high level trip (Trip Level 8) at which time the injection valve MOF004 is automatically closed. The pump will continue to run on minimum flow recirculation. The injection valve will automatically reopen if vessel level again drops to the low level (Trip Level 2) initiation point.

The HPCS pump motor and injection valve are provided with manual override controls. These controls permit the reactor operator to manually control the system following automatic initiation.

7.3.1.1.1.2 Automatic Depressurization System (ADS) - Instrumentation and Controls

a. ADS System Function

The automatic depressurization system is designed to provide automatic depressurication of the reactor vessel by activating seven safety/relief valves. These valves vent steam to the suppression pool in the event that the HPCS cannot maintain the reactor water level following a LOCA. ADS reduces the reactor pressure so that flow from the low pressure ECCS, LPCI system and LPCS, can inject into the reactor vessel in time to cool the core and limit fuel cladding temperature. Refer also to 6.3.2.2.2.

b. ADS Operation

Schematic arrangements of system mechanical equipment is shown in Figure 7.3-9 (Nuclear Boiler P&ID). ADS component control logic is shown in Figure 7.3-10 (Nuclear Boiler FCD). Instrumentation specifications are listed in Tables 7.3-3 and 7.3-4. Plant Layout Drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 7.3-9 (Nuclear Boiler P&ID) and Figure 7.3-10 (Nuclear Boiler FCD).

4. The following normally closed valves are signalled closed to ensure proper system lineup:

   a) The RHR heat exchanger discharge to RCIC valves MO F026 A, B, and AO F065 AB,

   b) The RHR heat exchanger flush to suppression pool valves MO F011 A, B,

   c) The RHR heat exchanger steam pressure reducing valves AO F051 A, B,

   d) The RHR heat exchanger steam inlet isolation valves MO F052 A, B and MO F087 A, B,

   e) The test return line to the suppression pool valves MO F024 A, B and MO F021,

   f) The suppression pool spray valves MO F027 A, B.

5. The normally open heat exchanger bypass valves MO F048 A, B are signaled open. The open signal is automatically removed 10 minutes after system initiation to allow operator control of the valve for throttling purposes.

Each LPCI pump discharge flow is monitored by a differential pressure switch which, when the pump is running and following an 8-second time delay, opens the minimum flow return line valve MO F064 A, B, C if flow is low enough that pump overheating may occur. The valve is automatically closed if flow is normal. The 8-second time delay is provided to prevent reactor vessel inventory loss during the shutdown cooling mode of the RHRS (see 5.4.7.2.6(a)).

The three RHR pump suction from the suppression pool valves MO F004 A, B, C and the RHR heat exchanger inlet and outlet valves MO F047 A, B and MO F003 A, B have their control switches keylocked in the open position, and thus require no automatic open signal for system initiation.

The two series service water crosstie valves MO F093 and MO F094 have their control switches keylocked in the close position, and thus require no automatic close signal for system initiation.

The two series containment spray valves (MO F016 A, B and MO F017 A, B), the two series RHR heat exchanger vent valves (MO F073 A, B and F074 A, B), and the RHR shutdown cooling mode suction valves (MO F006 A, B) are all normally closed and thus require no automatic close signal for system initiation.

The LPCI pump motors and injection valves are provided with manual override controls. These controls permit the operator to manually control the system subsequent to automatic initiation.

7.3.1.1.2 Primary Containment and Reactor Vessel Isolation Control System (PCRVICS) - Instrumentation and Controls

a. PCRVICS Function

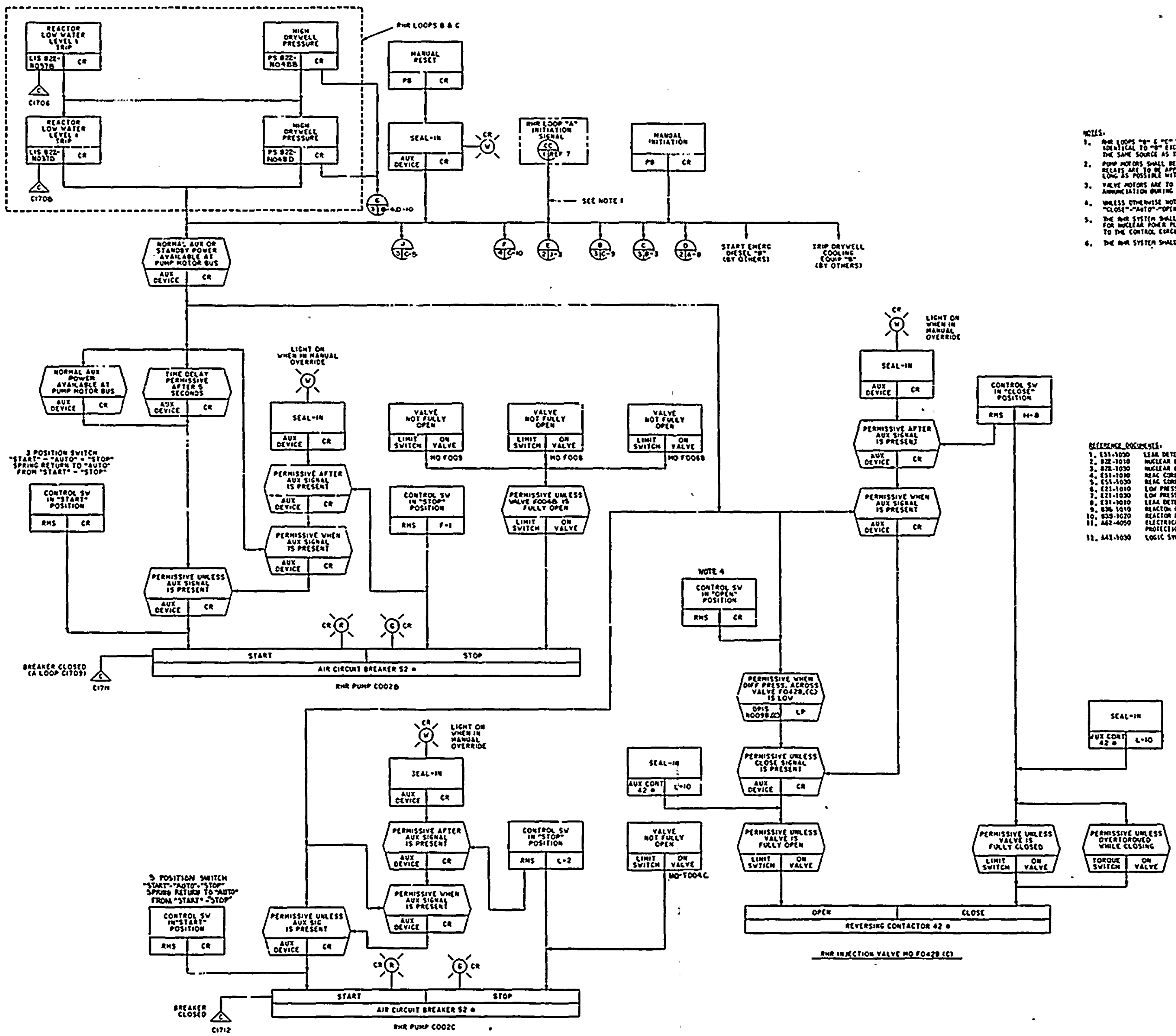
The PCRVICS includes the instrument channels, trip logics, and actuation circuits that automatically initiate valve closure providing isolation of the primary containment and/or reactor vessel, and initiation of systems provided to limit the release of radioactive materials.

See 6.2.4 and Table 6.2-16 for a complete description of primary containment and reactor vessel process lines and isolation signals applied to each.

b. PCRVICS Operation

Schematic mechanical arrangements of containment isolation valves and other components initiated by PCRVICS are shown in Figures 5.4-13 (RHR P&ID), 7.3-9 (Nuclear Boiler P&ID), 3.2-11 (RWCU P&ID), 3.2-3 (Reactor REURC P&ID), 11.2-2 (Equip. Drain Flow Diag.), 11.2-3 (Floor Drain Flow Diag.), and 3.2-16 (SGTS Flow Diag.). PCRVICS component control logic is shown in Figure 7.3-10 (Nuclear Boiler FCD), 7.3-14 (RHR FCD) and 7.3-1 (RWCU FCD). Instrument specification are listed in Tables 7.3-5 and 7.3-7. Plant layout drawings and electrical schematics are identified in 1.7. Operator information displays are shown in Figure 7.3-10 (Nuclear Boiler FCD). Refer also to Figure 7.3-9 (Nuclear Boiler P&ID).

During normal plant operation, the isolation control system sensors and trip logic that are essential to safety are energized. When abnormal conditions are sensed, instrument contacts open and de-energize the trip logic and thereby initiate isolation. Once initiated the PCRVICS trip logics seal-in and may be reset by the operator only when the initial conditions return to normal.



NOTES:

1. RHR LOOPS "B" & "C" LOGIC AND EQUIPMENT ARE SHOWN. RHR LOOP "A" IS IDENTICAL TO "B" EXCEPT CONTROL & MOTIVE POWER SHALL BE FROM THE SAME SOURCE AS THE LPCS SYSTEM (REF 7), AND AS NOTED.
2. PUMP MOTORS SHALL BE PROTECTED WITH OVERLOAD PROTECTION. PROTECTIVE RELAYS ARE TO BE APPLIED SO AS TO MAINTAIN POWER ON THE MOTOR AS LONG AS POSSIBLE WITHOUT IMMEDIATE DAMAGE TO EMERGENCY POWER SYSTEM.
3. VALVE MOTORS ARE TO BE PROVIDED WITH THERMAL OVERLOAD TRIPS AND ANNUNCIATION DURING TESTING AND LOSS OF POWER ANNUNCIATION.
4. UNLESS OTHERWISE NOTED, ALL RMS SHALL BE 3 POSITION SWITCHES "CLOSE"-"AUTO"-"OPEN". SPRING RETURN TO "AUTO" FROM "CLOSE" - "OPEN".
5. THE RHR SYSTEM SHALL BE DESIGNED IN ACCORDANCE WITH "PROPOSED CRITERIA FOR NUCLEAR POWER PLANT PROTECTION SYSTEMS IEEE 279", AS APPLICABLE TO THE CONTROL CIRCUITRY.
6. THE RHR SYSTEM SHALL BE DESIGNED IN ACCORDANCE WITH REF 11.

REFERENCE DOCUMENTS:

1. E31-1030 LEAK DETECTION SYSTEM (FCD)
2. B22-1010 NUCLEAR BOILER SYSTEM (PE10)
3. B22-1030 NUCLEAR BOILER SYSTEM (FCD)
4. E51-1010 REAC CORE ISLN CLG SYSTEM (PE10)
5. E51-1030 REAC CORE ISLN CLG SYSTEM (FCD)
6. E21-1010 LOW PRESS. CORE SPRAY SYS (PE10)
7. E21-1030 LOW PRESS. CORE SPRAY SYS (FCD)
8. E31-1010 LEAK DETECTION SYSTEM (IED)
9. B35-1010 REACTOR RECIRCULATION SYS (PE10)
10. B35-1020 REACTOR RECIRCULATION SYS (FCD)
11. A62-4050 ELECTRICAL EQUIPMENT SEPARATION FOR PROTECTION SYSTEMS
12. A42-1030 LOGIC SYMBOLS

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RESIDUAL HEAT REMOVAL SYSTEM FCD SHEET 1	FIGURE 7.3-14a

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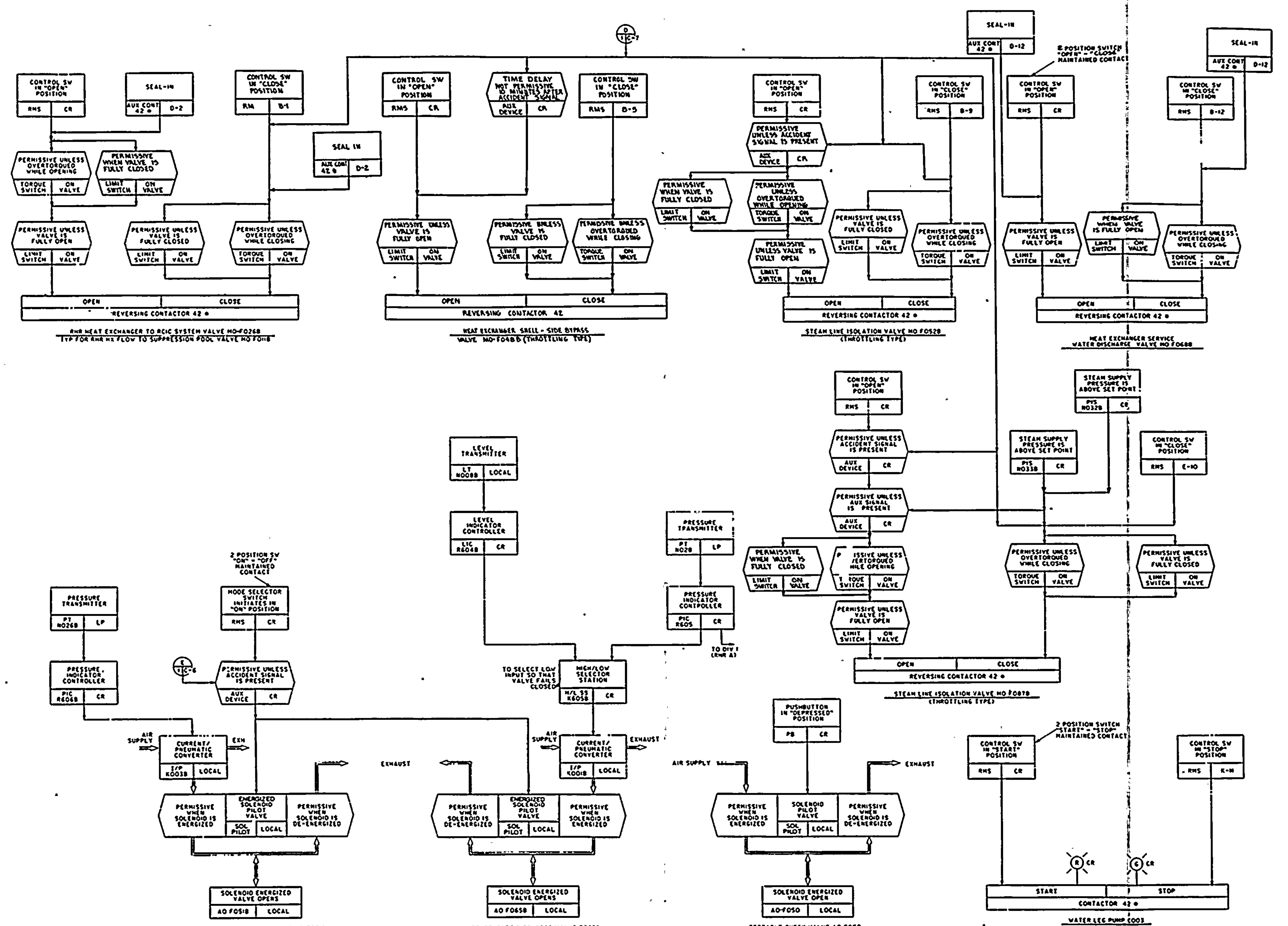
June 1981



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RESIDUAL HEAT REMOVAL SYSTEM FCD SHEET 2	FIGURE 7.3-14b

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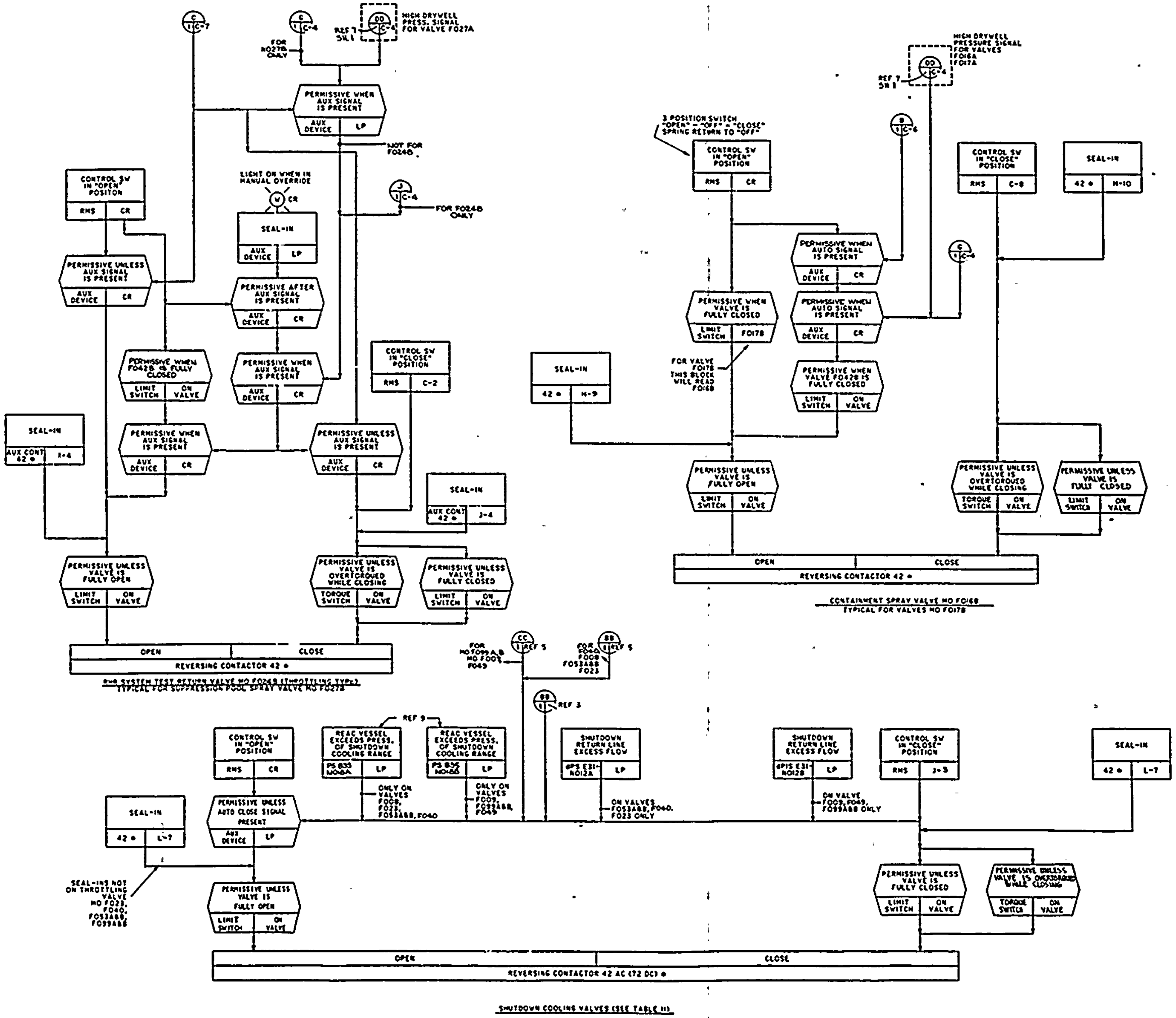
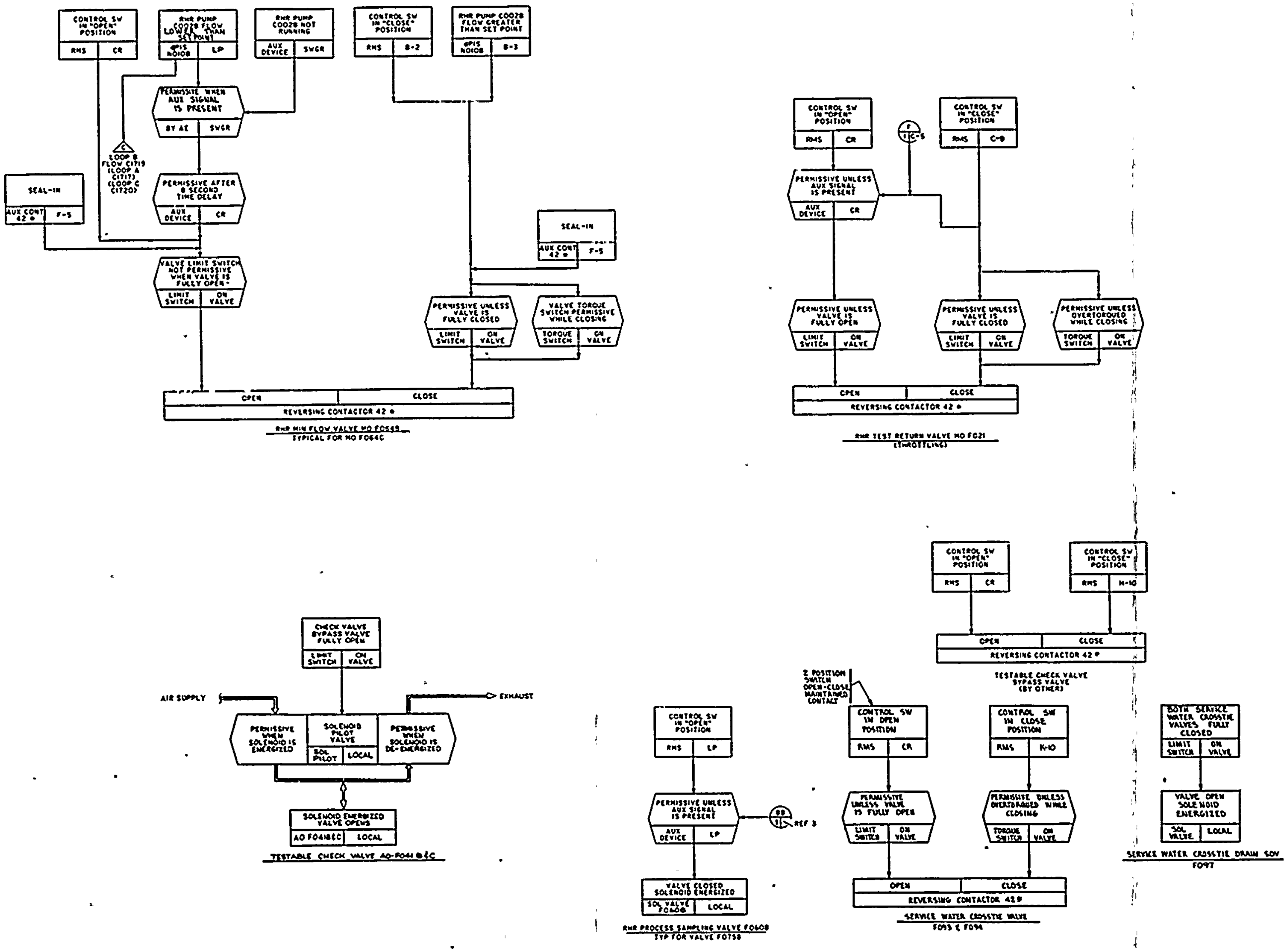


TABLE II

	VALVE NO.
SUCTION ISOLATION	MO F009
RADWASTE DISCHARGE	MO F040
INJECTION VALVE	MO F053A,B
RADWASTE DISCHARGE	MO F049
TEST CV BYPASS VALVE	MO F099A & B
HEAD SPRAY ISOLATION	MO F023
SUCTION ISOLATION	MO F008

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RESIDUAL HEAT REMOVAL SYSTEM FCD SHEET 3	FIGURE 7.3-14c

AMENDMENT NO. 16
June 1981



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

RESIDUAL HEAT REMOVAL SYSTEM FCD
SHEET 4

FIGURE 7.3-14d

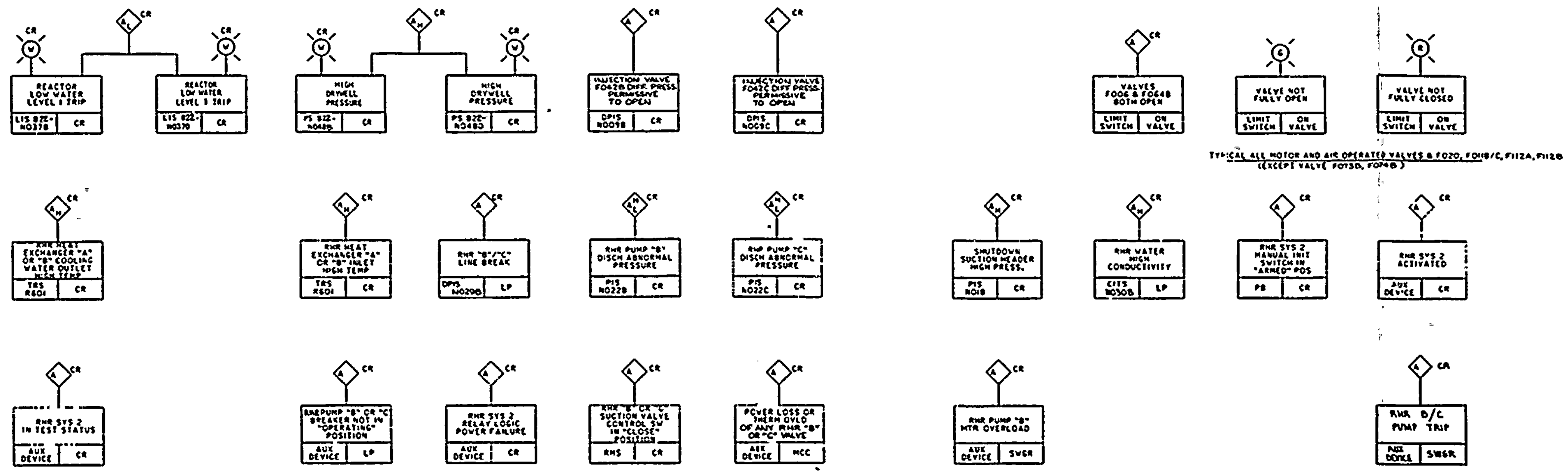
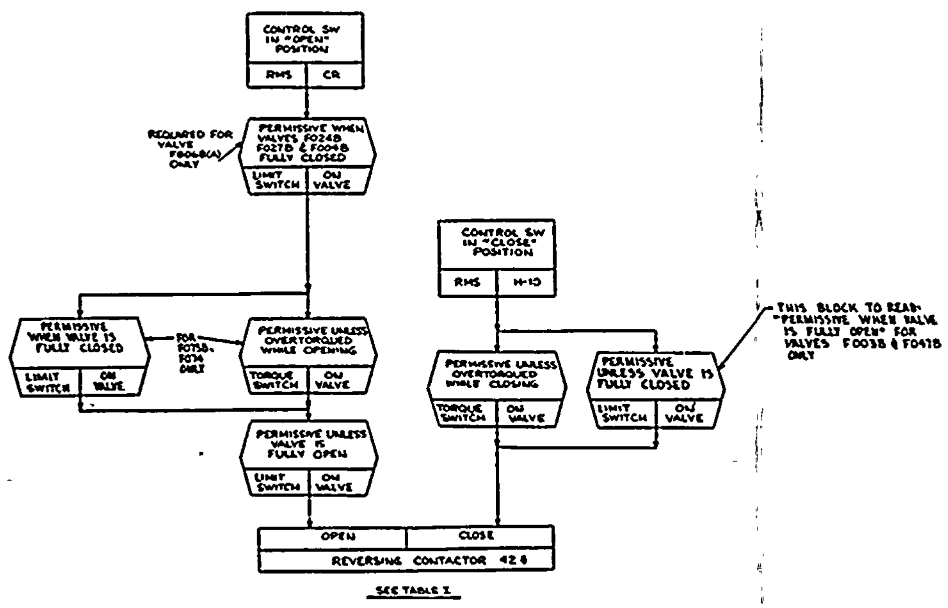


TABLE I

VALVE DESCRIPTION	VALVE NUMBER	SWITCH DESCRIPTION
Hx VENT VALVES	F073B & F074B	3 POS. SW "OPEN"-"STOP"-"CLOSE" MAINTAINED CONTACTS
SHUTDOWN COOLING SUCTION VALVE Hx SW INLET VALVE	F004B	2 POSITION SW "CLOSE"-"OPEN" MAINTAINED CONTACTS
RHR PUMP SUCTION VALVE Hx INLT VALVE	F004B & F004C F047B	2 POSITION SW "CLOSE"-"OPEN" MAINTAINED CONTACTS KEYLOCKED IN "OPEN" POSITION
Hx OUTLET VALVE	F003B	3 POSITION SW "CLOSE"-"NEUT"-"OPEN" SPRING RET TO NEUT



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

RESIDUAL HEAT REMOVAL SYSTEM FCD
SHEET 5

FIGURE 7.3-14e

7.4 SYSTEMS REQUIRED FOR SAFE SHUTDOWN

7.4.1 DESCRIPTION

This section discusses the instrumentation and controls of the following systems required for safe plant shutdown:

a. Reactor Core Isolation Cooling (RCIC) System

b. Standby Liquid Control System (SLCS)

c. RHR Shutdown Cooling Mode (RSCM)

d. Remote Shutdown System (RSS)

The sources which supply power to the safe shutdown systems originate from on-site AC and/or DC safety-related busses. Refer to Chapter 8 for a complete discussion of the safety-related power sources.

7.4.1.1 Reactor Core Isolation Cooling (RCIC) System

a. RCIC System Function

The reactor core isolation cooling system (see 5.4.6.2) instrumentation is designed to maintain or supplement reactor vessel water inventory during the following conditions:

1. Normal Operation. When the reactor vessel is isolated from its primary heat sink (the main condenser) and maintained in the hot standby condition;

2. Normal Operation. When the reactor vessel is isolated and accompanied by a loss of normal coolant flow from the reactor feedwater system;

3. When the plant is being shutdown and normal coolant flow from the feedwater system is stopped before the reactor is depressurized to a level where the reactor shutdown cooling mode of the RHR system can be placed into operation.

4. When required as a backup to the High Pressure Core Spray System to mitigate the consequences of the rod drop accident by automatically supplying cooling water to the reactor if vessel low water level is sensed.

b. RCIC System Operation

Schematic arrangements of system mechanical equipment is shown in Figure 7.4-1 (RCIC P&ID). RCIC system component control logic is shown in Figure 7.4-2 (RCIC FCD). Instrumentation Specifications are listed in Tables 7.4-1 and 7.4-2. Plant layout drawings, and electrical schematics are identified in 1.7. Operator Information displays are shown in Figure 7.4-1 (RCIC P&ID) and Figure 7.4-2 (RCIC FCD).

The RCIC System can be initiated either manually or automatically. The control room operator can initiate RCIC by operating the manual initiation pushbutton which simulates an automatic initiation or by activating each piece of equipment sequentially as required.

RCIC is automatically initiated by four redundant differential pressure switches, arranged in a one-out-of-two-twice logic configuration, which sense reactor vessel low water level (Trip Level 2).

The RCIC steam line isolation and the turbine steam exhaust motor-operated (MO) valves are keylocked in the open position, the turbine trip and throttle valve is normally open and require no change of position for automatic system initiation.

The RCIC system responds to an automatic initiation signal as follows (actions are simultaneous unless stated otherwise):

1. The pump suction from the condensate storage tanks valve MO F010 is signaled open,

2. To ensure pump discharge flow is directed to the reactor vessel only, the test return line to the condensate storage tanks valves MO F022 and MO F059 are signaled closed.

3. The turbine steam inlet and the turbine lube oil cooler cooling water supply valves MO F045 and MO F046 are signaled to open,

4. When the turbine steam inlet valve MO F045 starts to open, the RCIC pump discharge to reactor vessel valve MO F013 is signaled open, valve MO F013 is prohibited from opening or if open, automatically closes when MO F045 or the turbine trip and throttle valve is closed.

5. The barometric condenser vacuum tank vacuum pump is signaled to start,

7.4 SYSTEMS REQUIRED FOR SAFE SHUTDOWN

7.4.1 DESCRIPTION

This section discusses the instrumentation and controls of the following systems required for safe plant shutdown:

a. Reactor Core Isolation Cooling (RCIC) System

b. Standby Liquid Control System (SLCS)

c. RHR Shutdown Cooling Mode (RSCM)

d. Remote Shutdown System (RSS)

The systems discussed in this section are all capable of assisting the operator in achieving a safe plant shutdown. The Remote Shutdown and Standby Liquid Control Systems are manually operated backups to the Control Room and Reactor Manual Control Systems, respectively, only for special event conditions. Their use is not required to achieve a safe shutdown under normal, transient or accident conditions. The Shutdown Cooling Mode of RHR is one of several RHR modes available to the operator to remove residual heat. Loss of any of the systems will not impede safe shutdown of the plant, since alternate shutdown cooling modes exist. As such these systems are not required for safety, and have not been designed to meet safety system requirements.

The sources which supply power to the safe shutdown systems originate from on-site AC and/or DC safety-related busses. Refer to Chapter 8 for a complete discussion of the safety-related power sources.

7.4.1.1 Reactor Core Isolation Cooling (RCIC) System

a. RCIC System Function

The reactor core isolation cooling system (see 5.4.6.2) instrumentation is designed to maintain or supplement reactor vessel water inventory during the following conditions:

1. Normal Operation. When the reactor vessel is isolated from its primary heat sink (the main condenser) and maintained in the hot standby condition;

2. Normal Operation. When the reactor vessel is isolated and accompanied by a loss of normal coolant flow from the reactor feedwater system;

3. When the plant is being shutdown and normal coolant flow from the feedwater system is stopped before the reactor is depressurized to a level where the reactor shutdown cooling mode of the RHR system can be placed into operation.

4. When required as a backup to the High Pressure Core Spray System to mitigate the consequences of the rod drop accident by automatically supplying cooling water to the reactor if vessel low water level is sensed.

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b. RCIC System Operation

Schematic arrangements of system mechanical equipment are shown in Figure 5.4-9(a and b) (RCIC P&ID). RCIC system component control logic is shown in Figure 7.4-2 (RCIC FCD). Instrumentation Specifications are listed in Tables 7.4-1 and 7.4-2. Plant layout drawings, and electrical schematics are identified in 1.7. Operator Information displays are shown in Figure 5.4-9(a and b) (RCIC P&ID) and Figure 7.4-2 (RCIC FCD).

The RCIC System can be initiated either manually or automatically. The control room operator can initiate RCIC by operating the manual initiation pushbutton which simulates an automatic initiation or by activating each piece of equipment sequentially as required.

RCIC is automatically initiated by four redundant differential pressure switches, arranged in a one-out-of-two-twice logic configuration, which sense reactor vessel low water level (Trip Level 2).

The RCIC steam line isolation and the turbine steam exhaust motor-operated (MO) valves are keylocked in the open position, the turbine trip and throttle vale is normally open and require no change of position for automatic system initiation.

The RCIC system responds to an automatic initiation signal as follows (actions are simultaneous unless stated otherwise):

1. The pump suction from the condensate storage tanks valve MO F010 is signaled open,

2. To ensure pump discharge flow is directed to the reactor vessel only, the test return line to the condensate storage tanks valves MO F022 and MO F059 are signaled closed.

3. The turbine steam inlet and the turbine lube oil cooler cooling water supply valves MO F045 and MO F046 are signaled to open,

4. When the turbine steam inlet valve MO F045 starts to open, the RCIC pump discharge to reactor vessel valve MO F013 is signaled open. Valve MO F013 is prohibited from opening or if open, automatically closes when MO F045 or the turbine trip and throttle valve is closed. A one out of two twice

6. When valve MO F045 leaves the closed position the RCIC turbine is accelerated in speed until the automatic flow controller setpoint is reached and the system discharge flow is controlled by the turbine electronic governor mechanism.

During system operation if the barometric condenser vacuum tank water level becomes high the condenser condensate discharge pump is automatically started and the condensate returned to the RCIC pump suction. When the system is not operating excess tank water is discharged through isolation valves AO F004 and AO F005 to the clean condensate system.

In the event that the water level in the condensate storage tanks should become low the RCIC pump suction is automatically transferred from the condensate storage tank to the suppression pool by opening valve MO F031. Two level switches mounted on a Seismic Category I standpipe in the reactor building are used to detect low water level in the condensate storage tanks. Either switch can cause automatic suction transfer. Once valve F031 is fully open the condensate storage tank valve MO F010 is automatically closed.

The RCIC system includes design features which provide system equipment protection or accomplish primary containment isolation if certain types of abnormal events occur. The turbine is automatically shut down by closing the turbine trip and throttle valve if any of the following conditions are detected:

1. Turbine overspeed
2. High turbine exhaust pressure
3. RCIC isolation signal
4. Low pump suction pressure
5. Manual trip actuated by the control room operator.

To protect the RCIC pump from overheating during low flow conditions the pump discharge flow and pressure is monitored. If the pump discharge pressure switch indicates the pump is running and the pump discharge flow switch indicates low flow, the minimum flow return line valve MO F019 is automatically opened. The minimum flow valve is automatically closed when flow is normal, or when either the turbine trip and throttle valve or the steam inlet valve MO F045 is closed.

High water level in the reactor vessel indicates that the RCIC system has performed satisfactorily in providing makeup water to the reactor vessel. An automatic high water level trip is used to initiate closure of the steam supply valve shutting off steam to the RCIC turbine halting operation. The system will automatically restart if the water level subsequently decreases to the low level (initiation) trip point whereupon the steam supply valve is reopened.

reactor pressure is relieved through the relief valves to the suppression pool.

4. The reactor feedwater system which is normally available is also assumed to be inoperable. Reactor vessel water inventory is provided by the RCIC system.

5. Division 1 Emergency DC power is assumed to be available.

The RSS is required only during times of Main Control Room inaccessibility when normal plant operating conditions exist, i.e., no transients or accidents are occurring. For this reason the RSS function is not single failure proof and only the equipment which interfaces directly with safety-related equipment (RHR, RCIC, etc.) is required to be of a safety-related quality.

b. Remote Shutdown System Operation

Some of the existing systems used for normal reactor shutdown operation are also utilized in the remote shutdown capability to shutdown the reactor from outside the main control room. The remote shutdown capability is designed to control the required shutdown systems from outside the main control room irrespective of shorts, opens, or grounds in the control circuit in the main control room that may have resulted from an event causing an evacuation. The functions needed for remote shutdown control are provided with manual transfer switches which override controls from the main control room and transfer the controls to the remote shutdown control. Remote shutdown control is not possible without actuation of the transfer devices. All necessary power supplies and control logic are also transferred. Operation of the transfer devices causes an alarm in the main control room. Access to the remote shutdown panel is administratively and procedurally controlled. All system equipment (i.e., valves and pumps) necessary for proper system lineup and complete system control are located on the remote shutdown panel.

Manual activation of safety/relief valves and the initiation of Reactor Core Isolation Cooling (RCIC) system will maintain reactor water inventory and bring the reactor to a hot shutdown condition after scram. During this phase of shutdown, the suppression pool will be cooled by operating the Residual Heat Removal (RHR) system in the suppression pool cooling mode. Reactor pressure will be controlled and core decay and sensible heat rejected to the suppression pool by relieving steam pressure through the relief valves.

Manual operation of the relief valves will cool the reactor and reduce its pressure at a controlled rate until reactor pressure becomes so low that the RCIC system is unable to sustain operation. The RHR system will then be operated in the shutdown cooling mode using the RHR system heat exchanger to cool reactor water and bring the reactor to the cold low pressure condition.

The following RCIC System equipment/functions have transfer and control switches located on the remote shutdown control panel:

- F008 - Motor-operated valve (steam supply line isolation)
- F045 - Motor-operated valve (steam to turbine)
- F010 - Motor-operated valve (pump suction from condensate storage)
- F013 - Motor-operated valve (pump discharge to reactor)
- F019 - Motor-operated valve (pump bypass to suppression pool)
- C001 - Motor-operated valve (trip throttle valve)
- F046 - Motor-operated valve (cooling water supply valve)
- F064 - Motor-operated valve (RHR Cond. heat exch. steam line isolation valve)
- F063 - Motor-operated valve (steam supply isolation valve)
- F031 - Motor-operated valve (pump suction from suppression pool)
- F022 - Motor-operated valve (test bypass to condensate storage)
- P1 - Condensate pump from RCIC vacuum tank
- P2 - RCIC vacuum pump
- F069 - Motor-operated valve (vacuum pump discharge to suppression pool)
- F068 - Motor-operated valve (turbine exhaust to suppression pool)
- F059 - Motor-operated valve (test bypass to condensate storage)
- F076 - Motor-operated valve (steamline warmup line isolation
- F080 - Motor-operated valve (vacuum breaker isolation)
- F086 - Motor-operated valve (vacuum breaker isolation)

See Figure 7.4-1 (RCIC P&ID).

The following RCIC System instrumentation is provided on the remote shutdown control panel:

1. RCIC Flow Controller and indicator

2. RCIC Turbine Speed

3. Indicating lights are provided for:

   a) Turbine tripped

   b) Turbine bearing oil low pressure

   c) Turbine governor bearing oil temperature high

   d) Turbine coupling end bearing oil temperature high

The following RHR System equipment/functions have transfer and control switches located at the remote shutdown control panel:

F064B - Motor-operated valve (pump minimum flow)
F064B - Motor-operated valve (pump minimum flow)
F004B - Motor-operated valve (suppression pool pump suction)
F006B - Motor-operated valve (shutdown cooling pump suction)
C002B - Residual heat removal pump
F047B - Motor-operated valve (heat exchanger inlet)
F048B - Motor-operated valve (heat exchanger bypass)
F003B - Motor-operated valve (heat exchanger outlet)
F026B - Motor-operated valve (condensate discharge to RCIC suction line)
F087B - Motor-operated valve (steam reducing bypass)
F068B - Motor-operated valve (heat exchanger cooling water outlet)
F006A - Motor-operated valve (shutdown cooling pump suction)
F009 - Motor-operated valve (inboard shutdown isolation)
F008 - Motor-operated valve (outboard shutdown isolation)
F016B - Motor-operated valve (containment spray)
F027B - Motor-operated valve (suppression pool spray)
F042B - Motor-operated valve (LPCI injection)
F053B - Motor-operated valve (shutdown cooling injection)
F052B - Motor-operated valve (steam reducing isolation)
F024B - Motor-operated valve (test line)
F011B - Motor-operated valve (condensate discharge to suppression pool)
F049B - Motor-operated valve (discharge to radwaste)
F023 - Motor-operated valve (reactor head spray)
See Figure 7.3-13 (RHR P&ID).

The following RHR instrumentation is located on the remote shutdown control panel:

1. RHR flow indicator

The following nuclear boiler system equipment have transfer and control switches located on the remote shutdown control panel:

Three air operated relief valves, (non-ADS)

The following nuclear boiler instrumentation is provided on the remote shutdown control panel:

BLANK

1. Level indicator

2. Reactor pressure indicator

The following recirculation system valve has transfer and control switches located on the remote shutdown control panel:

F023A - Motor-operated valve (recirculation pump suction)

The following standby service water system (SSW) equipment/functions have transfer and control switches located at the remote shutdown control panel:

1. SSW pump 1B

2. SSW return to cooling tower, shut off valve SW-V-70B.

3. SSW return to cooling tower, shut off valve SW-V-69B.

4. SSW return to spray pond, valve SW-V-12B.

5. SSW pump 1B discharge, valve SW-V-2B.

6. RHR pump E12-C002B cooling water shut off, valve SW-V-24B.

7. RCIC pump room cooling coil SSW return, valve SW-V-34.

The following primary containment environmental parameters are indicated on the remote shutdown panel:

1. Drywell air temperature at several locations.

2. Drywell pressure.

3. Suppression chamber air temperature.

4. Suppression pool water temperature.

5. Suppression pool water level.

5. LOCA

The safe shutdown systems components located inside the drywell which are functionally required following a LOCA have been environmentally qualified to remain functional as discussed in 3.11 and indicated in Table 3.11-1.

6. Pipe Break Outside Secondary Containment

This condition will not affect the Safe Shutdown Systems. Refer to 3.6.

7. Missiles

Protection for safe shutdown systems is described in 3.5.

h. Minimum Performance Requirements

Minimum performance requirements for safe shutdown systems instrumentation and controls are provided in Chapter 16, "Technical Specifications".

7.4.1.6 Final System Drawings

The final system drawings including:

1. Piping and Instrumentation Diagrams (P&ID)
2. Functional Control Diagrams (FCD)

have been provided for the safe shutdown systems.

Functional and architectural design difference between the PSAR and FSAR are listed in Table 1.3-8.

7.4.2 ANALYSIS

The safe shutdown systems are designed such that loss of instrument air, a plant load rejection, or a turbine trip will not prevent the completion of the safety function. No abnormal operation is assumed for the RSS which, by itself, performs no safety-related function.

7.4.2.1 Conformance To 10 CFR 50 Appendix A - General Design Criteria

The following is a discussion of conformance to those general design criteria which apply specifically to the safe shutdown systems. Refer to 7.1.2.2 for a discussion of General Design Criteria which apply equally to all safety-related systems.

a. General Design Criteria 19 - Control Room

The remote shutdown system consists of equipment located outside the control room which is sufficient to provide and assure prompt hot shutdown of the reactor and to maintain safe conditions during hot shutdown. The equipment also provides capability for subsequent cold shutdown of the reactor.

b. General Design Criterion 26 - Reactivity Control System Redundancy and Capability

SLCS provides an independent reactivity control system redundant to manual control rod movement.

c. General Design Criterion 29 - Protection Against Anticipated Operational Occurrences

The SLCS maintains the reactor sub-critical by introducing poison into the reactor in the event the manual insertion of control rods cannot achieve subcriticality in the reactor.

d. General Design Criterion 34 - Residual Heat Removal

The reactor shutdown cooling mode of the residual heat removal system removes residual heat from the reactor when it is shutdown and the main steamlines are isolated, to maintain the fuel and reactor coolant pressure boundary within design limits. Redundant cooling routes are provided to meet the single failure criteria.

7.4.2.2 Conformance To IEEE Standards

The following is a discussion of conformance to those IEEE Standards which apply specifically to the Safe Shutdown Systems. Refer to 7.1.2.3 for a discussion of IEEE Standards which apply equally to all safety-related systems.

a. IEEE 279-1971

The reactor shutdown cooling mode of the residual heat removal system utilizes the same equipment used by the LPCI mode. Therefore, refer to 7.3.2 for the RSCM standards and regulatory compliance.

Since the remote shutdown system, by itself, does not perform any safety-related function, it does not fall within the criteria set by IEEE-279. This system interfaces with safety-related systems, such as RHR and RCIC, and during normal operation becomes part of those systems and meets the design criteria for those systems.

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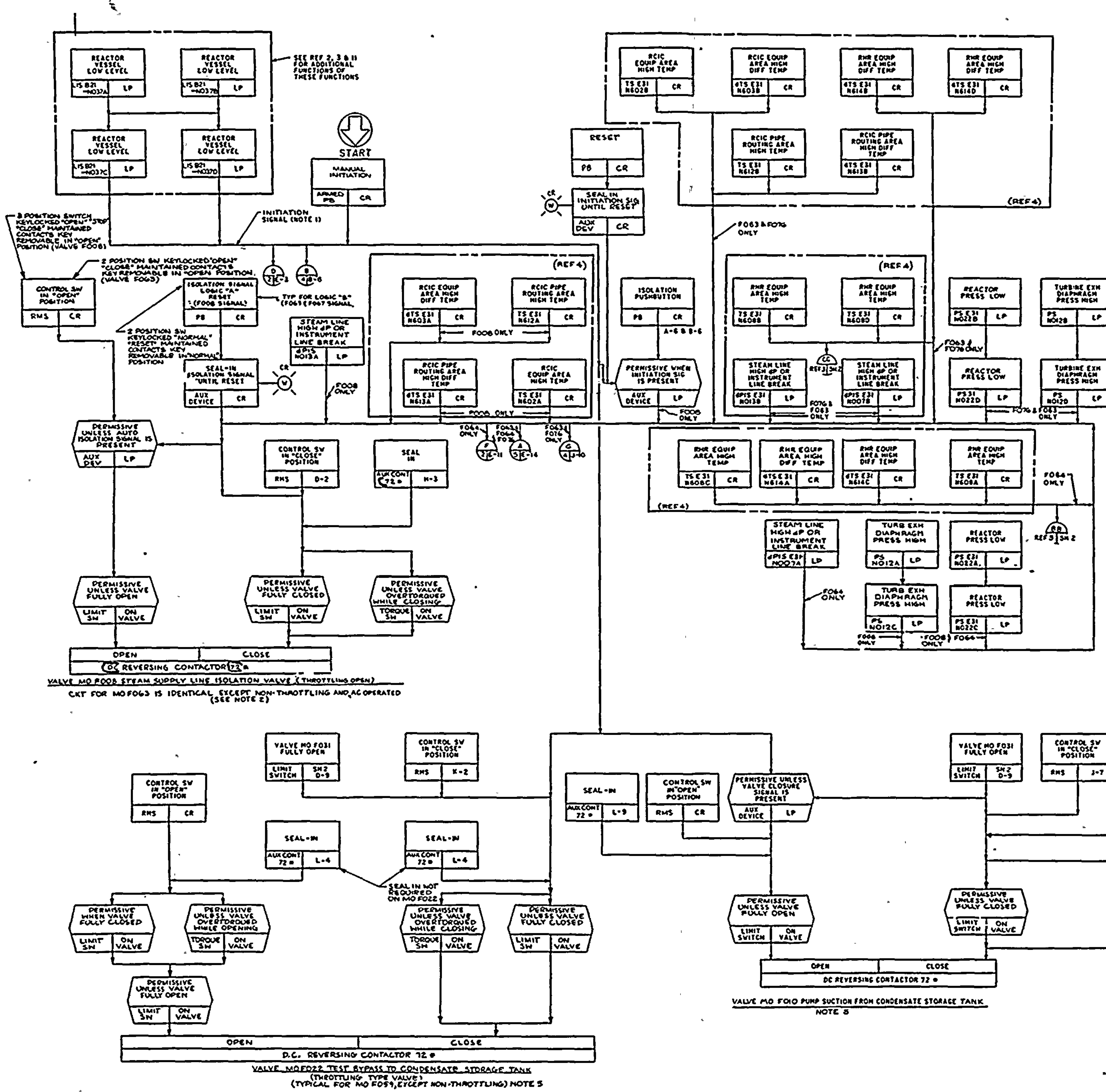
FCF: 239X248AD G1 (E51-1050)



NOTES:

1. THE RCIC SYS IS ARRANGED FOR TEST OF PUMP AT FULL FLOW AT ANY TIME EXCEPT WHEN INITIATION SIGNAL OR AUTO ISOLATION SIGNAL IS ACTIVATED. IN EVENT THE INITIATION SIGNAL OCCURS WHILE TEST IS UNDERWAY, THE SYS AUTOMATICALLY RETURNS TO STARTUP MODE.
2. ALL POWER FOR OPERATION OF D.C. VALVE MOTORS SHALL ORIGINATE FROM A PLANT D.C. BUS. POWER FOR AC OPERATED VALVES SHALL ORIGINATE FROM AN EMERGENCY A.C. BUS.
3. ALL EQUIPMENT & INSTRUMENT PREFIXED BY SYS NO. (E51) UNLESS OTHERWISE NOTED.
4. ISOLATION SIGNAL SWITCHES SHALL BE OF THE TYPE THAT CLOSE CONTACTS FOR THE SPECIFIED ISOLATION EVENT. WHERE AUXILIARY RELAYS ARE USED IN THE ISOLATION CHANNELS THEY SHALL BE POWERED FROM STATION BATTERIES. (REF 10)
5. AUXILIARY RELAYS & DEVICES NOT SHOWN ON FUNCTIONAL CONTROL DIAGRAMS EXCEPT WHERE REQUIRED TO CLARIFY FUNCTION.
6. FURNISHED WITH TURBINE.
7. THE RCIC SYS SHALL BE DESIGNED IN ACCORDANCE WITH REF 10.
8. THE RCIC SYSTEM SHALL BE DESIGNED IN ACCORDANCE WITH "PROPOSED CRITERIA FOR NUCLEAR POWER PLANT PROTECTION SYSTEM, IEEE-279", INSOFAR AS PRACTICAL.
9. RCIC SYSTEM PUMP MOTORS SHALL BE PROVIDED WITH A TRIP (WHICH WILL BE ANNUNCIATED) AND AN ALARM ON OVERLOAD D.C. VALVE OPERATOR MOTORS SHALL TRIP ON OVERLOAD (TRIPS SHALL BE ANNUNCIATED.) AC VALVE OPERATOR MOTORS SHALL ALARM AND TRIP ON OVERLOAD. TRIP SHALL BE SET HIGHER THAN ALARM.

REFERENCE DOCUMENTS:	MPL ITEM NO.
1. NUCLEAR BOILER P&ID	B21-1010
2. NUCLEAR BOILER SYSTEM FCD	B21-1030
3. RHR SYSTEM FCD	E12-1030
4. LEAK DETECTION SYSTEM P&ID	E31-1010
5. HPCS SYSTEM FCD	E22-1030
6. RCIC SYSTEM P&ID	E51-1010
7. LOGIC SYMBOLS	A42-1030
8. TURBINE CONTROL SYS & ELECTRIC WIRING	YPF 2767-7 (14)
9. LEAK DETECTION SYSTEM DESIGN SPEC	E31-4010
10. ELECTRICAL EQUIP. SEPARATION FOR SAFEGUARD SYSTEMS	A62-4050
11. LOW PRESSURE CORE SPRAY SYSTEM FCD	E21-1030
12. INSTRUMENT DATA SHEET	E51-3050

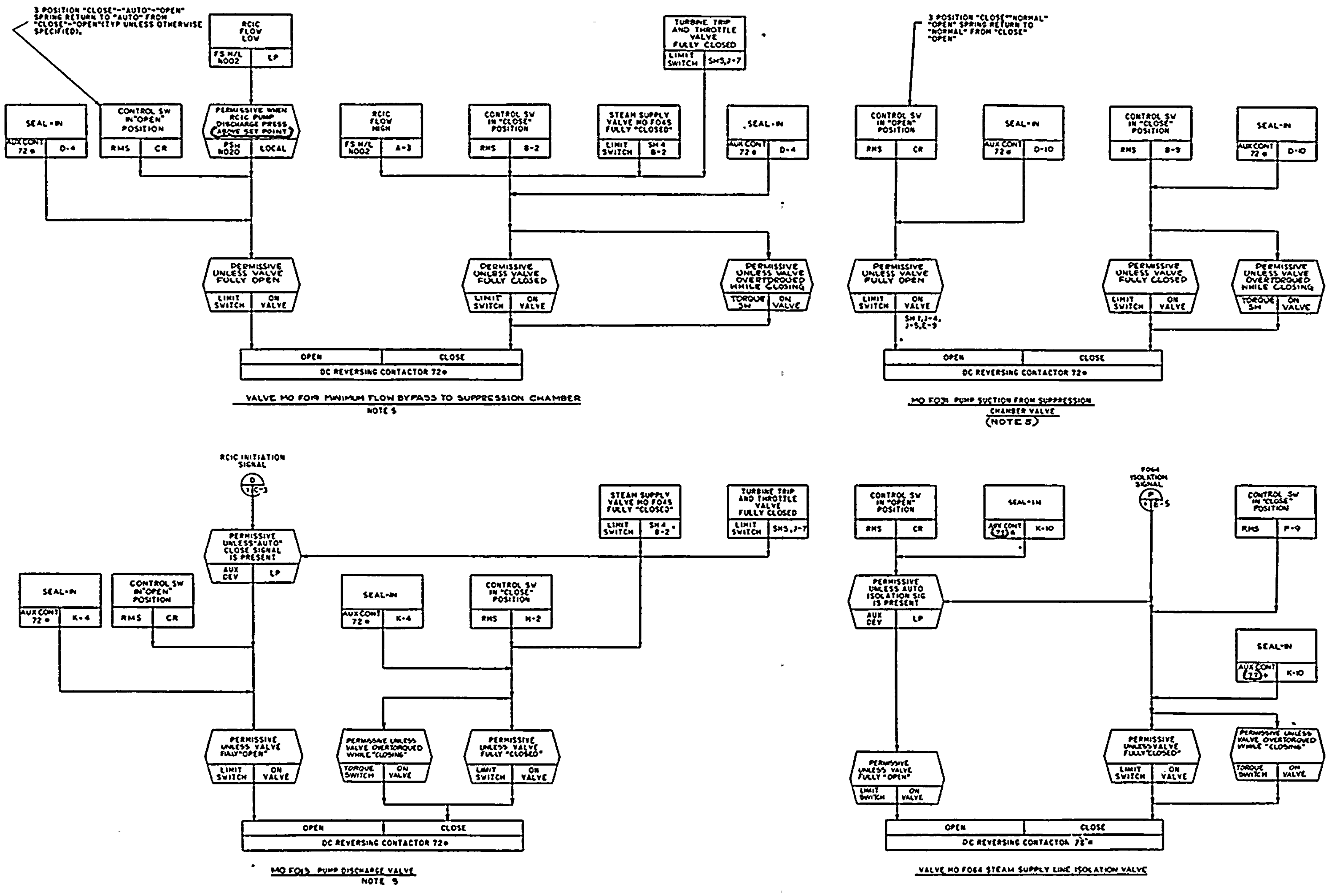
LEGEND:

● = SWITCHGEAR DEVICE FUNCTION NUMBERS ANSI SPEC C37.2

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

RCIC SYSTEM (FCD)

FIGURE 7.4-2a



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RCIC SYSTEM FCD	FIGURE 7.4-2b

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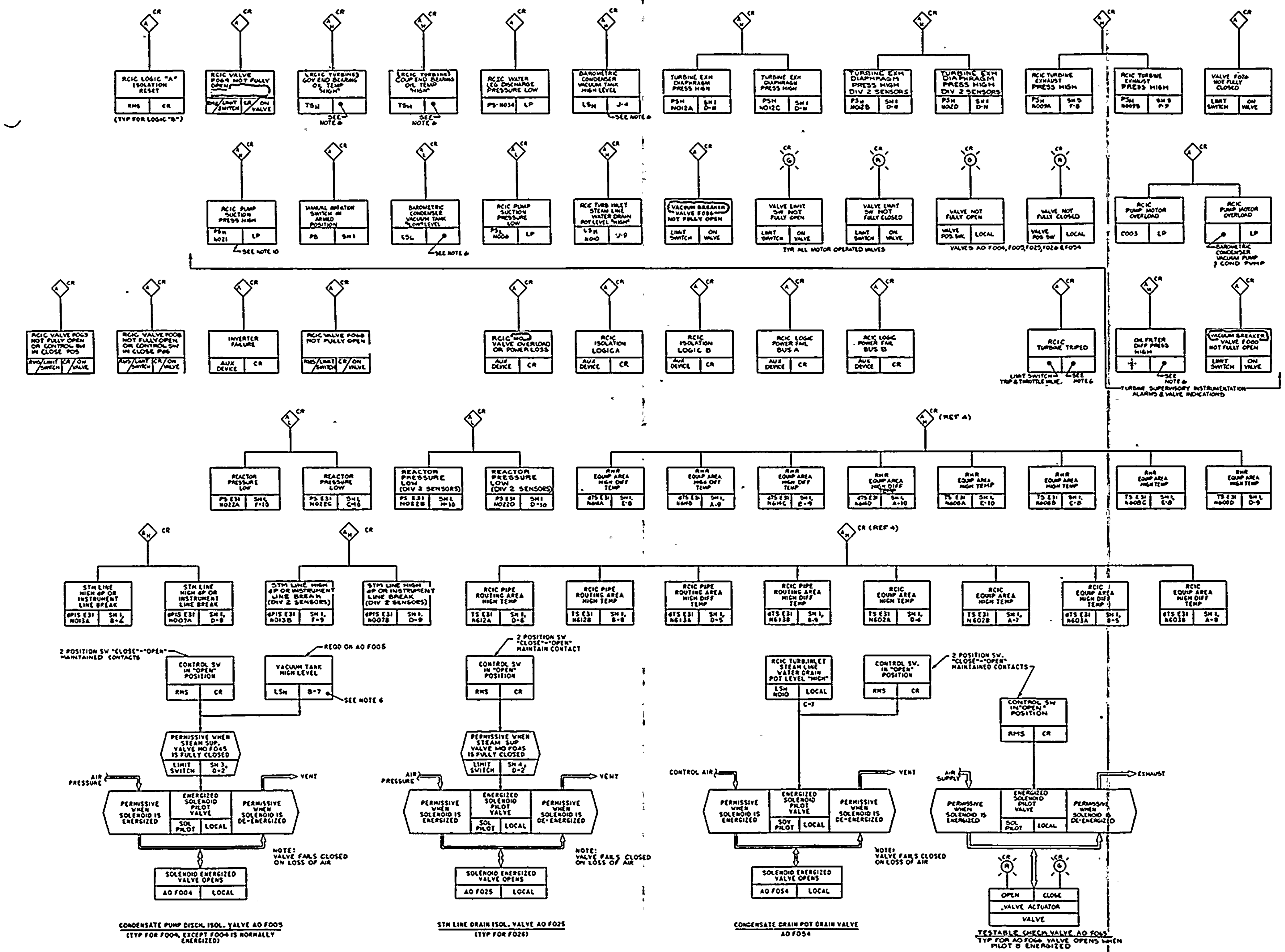

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

RCIC SYSTEM FCD

FIGURE 7.4-2c

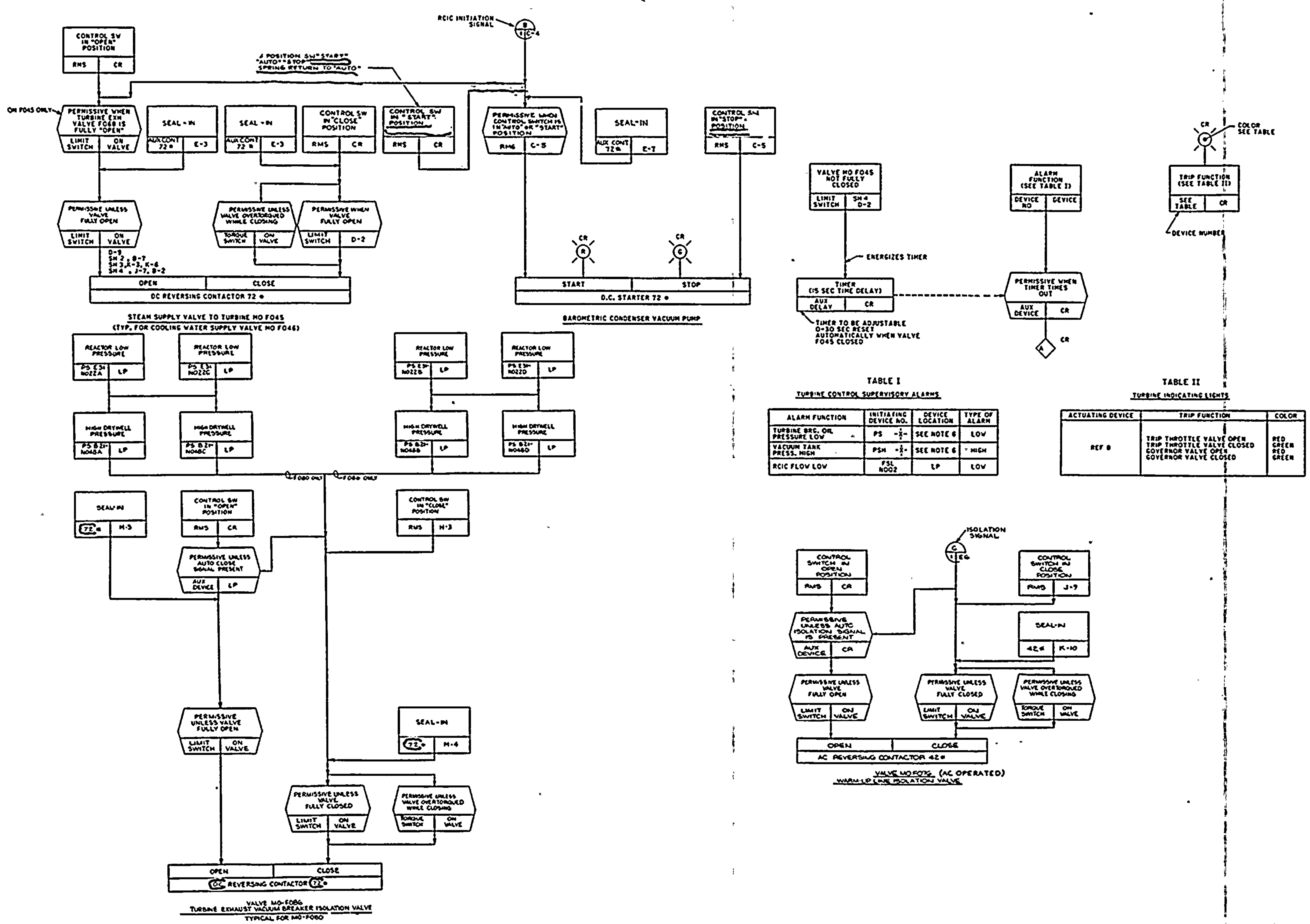

TABLE I

TURBINE CONTROL SUPERVISORY ALARMS

ALARM FUNCTION	INITIATING DEVICE NO.	DEVICE LOCATION	TYPE OF ALARM
TURBINE BRG. OIL PRESSURE LOW	PS -X-	SEE NOTE 6	LOW
VACUUM TANK PRESS. HIGH	PSH -X-	SEE NOTE 6	HIGH
RCIC FLOW LOW	FSL N002	LP	LOW

TABLE II

TURBINE INDICATING LIGHTS

ACTUATING DEVICE	TRIP FUNCTION	COLOR
REF 8	TRIP THROTTLE VALVE OPEN TRIP THROTTLE VALVE CLOSED GOVERNOR VALVE OPEN GOVERNOR VALVE CLOSED	RED GREEN RED GREEN

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RCIC SYSTEM FCD	FIGURE 7.4-2d

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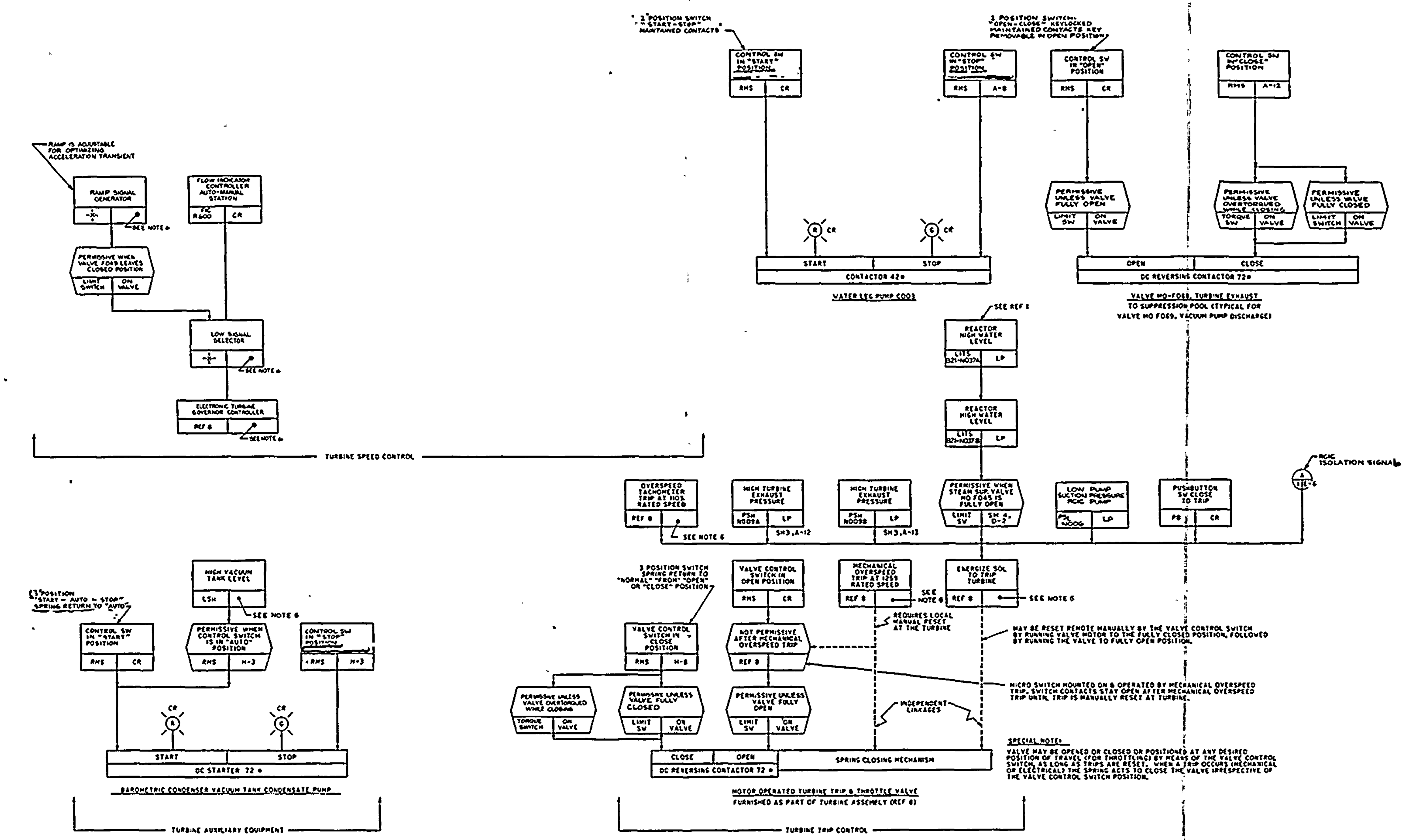

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RCIC SYSTEM FCD	FIGURE 7.4-2e

46. Post-Accident Sampling System

Onsite and/or offsite facilities are provided to analyze primary coolant and containment air grab samples for variables and ranges listed in Regulatory Guide 1.97.

TABLE 7.5-1

SAFETY-RELATED DISPLAY INSTRUMENTATION

	Design Criteria	Type Readout	Number of Channels	Range	Type & Category[1]	Display Instrument Accuracy	Location
1.	Reactor Vessel Pressure	Recorder	2	0-15500 psig	A, 1	± 2% FS	CR
2.	Reactor Vessel Water Level	Recorder	2	-150"/0/+60"	A, 1	± 2% FS	CR
		Recorder	2	-117" - -317"	A, 1	± 2% FS	CR
3.	Neutron Flux Power Level (SRM)	Recorder	4	10^{-7} - 10^{-3}% Power[2] 10^{-3} - 10% Power[3]	A	± 2%	CR
4.	Main Steam Line Flow	Indicator	4	0 - 4.25 x 10^6 lb/hr	---	± 2% FS	CR
5.	RCIC Flow	Indicator	1	0 - 700 GPM	D, 2	± 2% FS	CR
6.	RCIC Discharge Pressure	Indicator	1	0 - 150 psig	D, 2	± 2% FS	CR
7.	HPCS Flow	Indicator	1	0 - 8000 GPM	D, 2	± 2% FS	CR
8.	HPCS Discharge Pressure	Indicator	1	0 - 1500 psig	---	± 2% FS	CR
9.	LPCS Flow	Indicator	1	0 - 10,000 GPM	D, 2	± 2% FS	CR
10.	Drywell Atmos & Suppress Atmos Temps	Indicator Print Out	2	50 - 400°F	D, 2	± 2% FS	CR
11.	LOCA Radiation High Range Area Monitors	Recorder	2 2	10^0 - 10^7 R/hr	C, E, 2	± 2% FS	CR
12.	Leak Detection Radiation Monitors	Recorder	2 2	10^0 - 10^6 CPM 10^0 - 10^6 CPM	--- ---	± 2% FS ± 2% FS	CR CR

Notes: 1 The instruments meet the recommendations required by the Category type as described in Regulatory Guide 1.97, Revision 2.
2 Inserted
3 Withdrawn

TABLE 7.5-1 (Cont'd)

	Design Criteria	Type Readout	Number of Channels	Range	Type & Category	Display Instrument Accuracy	Location
13.	Primary Containment Hydrogen Concentration	Recorder	2	0 - 30%	C,1	±2% FS	CR
14.	Primary Containment Oxygen Concentration	Recorder	2	0 - 10%	C,1	±2% FS	CR
15.	Suppression Chamber Pressure	Recorder	2	0 - 60 psig	D,2	±2% FS	CR
16.	Suppression Pool Temperature	Print Out Indicator	2	30 - 230°F	D,2	±2% FS	CR
17.	Suppression Pool Water Level	Recorder	2	+25"/0/-25"	C,1	±2% FS	CR
18.	Bldg. Gaseous Release Monitor	Recorder	3	10^1 - 10^6 CPM	E,C,2	±2% Span	CR
			3	10^1 - 10^6 CPM	E,C,2	±2% Span	CR
			2	10^{-2} - 10^4 R/hr	C,2	±2% Span	CR
19.	Containment Instrument Air	Indicator	2	0-150 psig	D,2	±2% FS	CR
20.	Wind Speed	Recorder	1	0 - 67 MPH	E,3	±2% FS	CR
	Wind Direction	Recorder	1	0° - 360°	E,3	±2%	CR
21.	Temperature Differential	Recorder	1	±15°F	E,3	±2%	CR
22.	Radiation Exposure Rate	Recorder	3	10^{-2} - 10^4 R/hr	C,2	±2% Span	CR
23.	SRV Position Indication	Indicator	18	Full Closed to Full Open	D,2	-----	CR
24.	Power Supply Monitoring	Voltmeter	6	0-5.25 kVAC	D,2	±2% FS	CR
			4	0-600 VAC	D,2	±2% FS	CR
			3	0-300 VAC	D,2	±2% FS	CR
			5	0-150 VAC	D,2	±2% FS	CR
			4	±30 VAC	D,2	±2% FS	CR
			29	DC Ammeters of various ranges			CR

TABLE 7.5-1 (Cont'd)

	Design Criteria	Type Readout	Number of Channels	Range	Type & Category	Display Instrument Accuracy	Location
25.	Feed Water Flow	Indicator	2	0 - 8.5 X 10^6 #/hr	D,3	±2% FS	CR
26.	CST Level Indicator	Indicator	2	0 - 35 ft.	D,3	±2% FS	CR
27.	RHR Flow (LPCI and Shut-down Cooling)	Indicator	3	0 - 10,000 GPM	D,2	±2% FS	CR
	(Head Spray)		1	0 - 600 GPM	D,2	±2% FS	CR
28.	RHR HX Outlet Temperature	Recorder	2	0 - 600°F	D,3	±2% FS	CR
	RHR Service Water Flow	Indicator	2	0 - 10,000 GPM	D,2	±2% FS	CR
29.	SLCS Flow Rate	Indicator	2	0 - 50 GPM	D,2	±2% FS	CR
30.	SLCS Tank Level	Indicator	2	0 - 5000 Gal	D,2	±2% FS	CR
31.	SSW System Pump Discharge Line Pressure	Indicator	2	0 - 300 psig	---	±2% FS	CR
			1	0 - 100 psig	---	±2% FS	CR
32.	SSW System Flow Rate	Indicator	2	0 - 12,000 GPM	D,2	±2% FS	CR
	HPCS SS Flow Rate	Indicator	1	0-1320 GPM	D,2	±2% FS	CR
33.	SSW Pond. Water Level	Indicator	4	0 - 20 ft.	---	±2% FS	CR
34.	Spent Fuel Pool Cooling	Indicator	3	0 - 212°F	---	±2% FS	CR
35.	Main Control Room Temper-ature	Indicator	2	50 - 100°F	---	±2%	CR
36.	SGTS Flow Rate	Indicator	4	0 - 6000 CFM	---	±2%	CR
37.	CAC System Flow Rate	Indicator	4	0 - 300 CFM	---	±2%	CR

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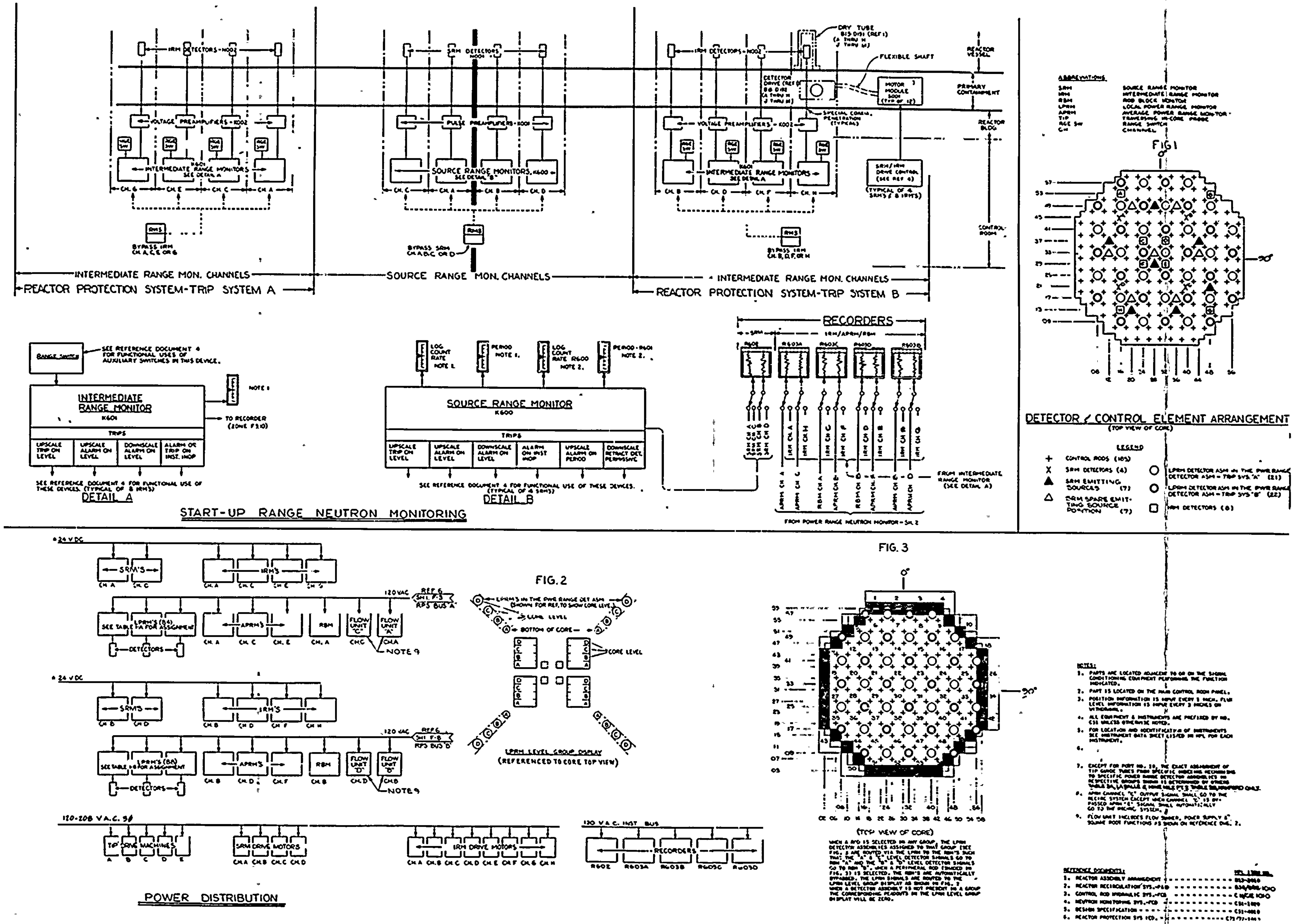
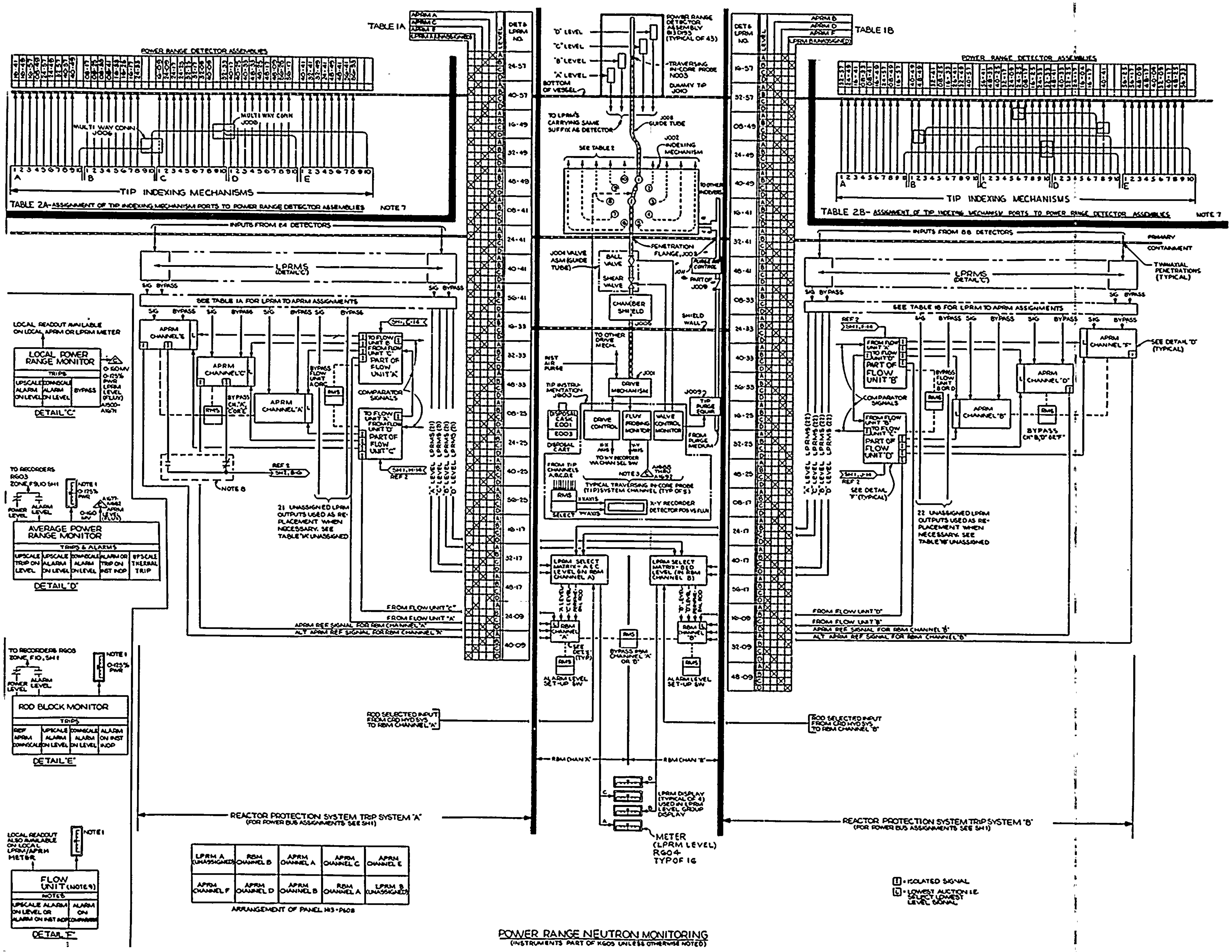
January 1982



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NEUTRON MONITORING SYSTEM IED
FIGURE	7.6-4a

AMENDMENT NO. 22
January 1982



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NEUTRON MONITORING SYSTEM IED	FIGURE 7.6-4b

than 0.95 for both normal and abnormal storage conditions. Normal conditions exist when the fuel storage racks are covered with a normal depth of water (about 25 feet above the stored fuel) for radiation shielding, and with the maximum number of fuel assemblies or bundles in their design storage position. An abnormal condition may result from damage caused by accidental drop of a fuel assembly or stuck fuel assembly during attempted withdrawal.

The criticality analyses of the normal condition included several conservative assumptions as well as the effect of uncertainties in calculation method and geometric and material variations of the fuel storage rack. The following conservative assumptions were used in the calculation:

a. Fresh fuel of 3.25 weight per cent U-235 enrichment.

   Initially the maximum enrichment will be much lower than this, but could approach this value if an 18-month fuel cycle is used. The enrichment selected is higher than the average enrichment of any fuel expected to be stored in the spent fuel pool. It was chosen because the fully loaded rack of fuel with this enrichment gives a more reactive condition than any presently foreseen.

b. Uniform planar array of 3.25 weight per cent enrichment fuel

   Calculations have shown that this is conservative compared to the realistic, planar distributed enrichments within an assembly.

c. Spent Fuel Pool Bulk Water Temperature $68^{\circ}F$

   This is considerably lower than expected. Nevertheless, a calculation was done to determine the increase in reactivity due to a decrease in pool temperature to $32^{\circ}F$. The results showed the effect to be negligible.

d. Fuel racks are infinite in three dimensions.

e. Fixed neutron poisons in the fuel assembly are neglected.

The majority of the calculations were performed with methods commonly used in light water reactor design; i.e., 4-group diffusion theory cell calculations using PDQ-7 (Reference 9.1-2). Cross sections for these calculations are generated with NUMICE-2, (Reference 9.1-3) the NUS Corporation version of the Westinghouse LEOPARD code. This code uses the same cross section library tape and calculational techniques as LEOPARD. Selected cases were checked and the final design multiplication factors were verified with Monte Carlo calculations using KENO-IV (Reference 9.1-4), with a 123-group cross section library generated from a basic GAM-THERMOS library using two subroutines, NITAWL and XSDRNPM, in the AMPX1 (Reference 9.1-5) code package. Both the PDQ-7 and the KENO-IV calculation methods, as described above, have been benchmarked. These calculation methods, as described, were used for the WNP-2 calculation and do not contain any significant modifications.

Under normal conditions, for a center-to-center spacing of 6.5 inches between fuel assemblies with B_4C plates surrounding each stored fuel assembly, the k_{eff}, as determined using KENO, is 0.851. With the void space between the B_4C plates and the stainless steel box flooded with water, the KENO calculation yielded a lower k_{eff}. Calculation uncertainties were determined from comparison between calculation and experiments using KENO and a statistical evaluation of Monte Carlo runs. The results indicated a calculational uncertainty for the former of 0.013 Δk and for the latter 0.010 Δk at a 95% confidence level; this represents a total calculation uncertainty of 0.023 Δk. Mechanical spacing and tolerances acting in a direction close to the water gaps between adjacent racks result in a slight reactivity increase of 0.002 Δk. Production tolerances of B_4C plates result in a reactivity increase of 0.003 Δk.

To determine the effect of reduced or missing neutron absorbing material, it was assumed that one out of every twenty-five neutron absorber plates was missing. This case is extremely unlikely but shows poison variation sensitivity. The results of the calculation was an increase in reactivity of 0.015 Δk. A temperature decrease in pool water temperature to 32°F was included, i.e., 0.004 Δk. Adding the total calculational uncertainties of 0.023 Δk and the total geometric and material uncertainties of 0.024 Δk to the nominal k_{eff} results in a k_{eff} of 0.897 with a confidence level of 95%. This is well below the design basis of equal to or less than 0.95 for the normal wet condition.

was performed using the horizontal floor response spectra (damping 1/2% of critical).

The fundamental frequency of vertical vibration of the rack was also determined using the STARDYNE computer program. The same model replacing lateral mass with vertical mass was utilized. In this case, since the fuel rests on the base frame, the entire mass of the fuel was lumped at the base grid. Since the calculated frequency was 50.9 Hz and the vertical floor response spectra (damping 1/2% of critical) showed constant acceleration at frequencies in excess of 18 Hz, the effects of the vertical accelerations were considered using the zero-period acceleration in a static analysis. The lateral and vertical loads were considered to be acting simultaneously.

In the general seismic/structural analysis of the fuel racks, the mass of a fuel assembly is assumed to be uniformly distributed along the length of each of the fuel storage cans. Since a maximum gap on the order of 3/8" exists between the side of a fuel assembly and the can (when the fuel is not encased in a channel), the fuel will actually move within the can during a seismic event and cause impact loads to be transmitted to the fuel rack restraints. The effects of this fuel can interaction are determined using a simplified finite element model of the rack and fuel. A nonlinear dynamic analysis is performed utilizing the ANSYS computer program. Details of this analysis are given in NUS Corporation Technical Report #2060, entitled "Fuel-Can Interaction Analysis," October, 1977.

Using the given loads, load combinations and analytical methods, stresses were calculated at critical sections of the rack and compared to the structural acceptance criteria. In all cases, the calculated stress did not exceed the allowable stress.

To assure the integrity of the spent fuel storage racks, specially designed control samples, consisting of B_4C plates sealed in storage tube stainless steel material and fabricated using the same procedures employed for the production of the fuel racks, will be placed in a readily accessible position in the spent fuel pool. These samples are subjected to periodic visual examination and neutron attenuation tests, if visual examination indicates evidence of corrosion.

Sectopm 15.7.4 of the FSAR presents an analysis (for radiological considerations) of how many fuel rods would fail in the event a fuel bundle (700# including channel) was accidently dropped on the core. The worse case accident for this

scenario would cause 124 fuel rods to fail based on 250 ft. lbs. of energy being required to cause compressive cladding failure in a single rod. The total kinetic energy required to cause the failure of the 124 fuel rods would be 124 times 250 ft. lbs. or 31,000 ft. lbs. This total kinetic energy was used as the limit for the spent fuel pool accidental drop. Loads were calculated that if dropped from: (a) the pool surface, (b) 5 feet above the surface, and (c) 20 feet above the pool surface would yield 31,000 ft. lbs. on impact with the spent fuel assemblies being stored in the high density racks. The fuel assemblies in the rack would be the first structure encountered by a falling object since the BWR assemblies protrude above the racks by approximately 4-1/2 inches. The maximum weights of objects dropped from the three aforementioned conditions are 1510 lbs., 1210 lbs., and 760 lbs., respectively. The maximum weight of 1510 lbs. was chosen as the maximum allowable load that would be handled over the spent fuel pool surface. While the 1510 lbs. is slightly greater than twice the weight of a fuel assembly, this value was chosen as a maximum credible number for analysis purposes concerning criticality and structural aspects of the fuel racks. Any loads less than this value dropped from the same height would yield a lower kinetic energy value and therefore result in less impact on fuel assemblies being stored in the fuel racks. Since the kinetic energy due to the maximum weight of 1510 lbs. is equal to that considered in the fuel handling accident of 15.7.4 of the FSAR, any kinetic energy less than this amount being dissipated will result in a lesser amount of cladding failures in fuel rods. Therefore, the effects from dropping of an object of less weight than a spent fuel assembly which is being handled over the surface over the spent fuel pool will be less than that described in 15.7.4.

The high density fuel rack designer, NUS Corporation, analyzed the fuel racks from both a structural and criticality standpoint concerning a 1510 lbs. object dropped from the surface of the fuel pool. The results indicated that none of the fuel rack damage that might occur in this situation would lead to a criticality problem. Details of these analysis are given in NUS Corporation Technical Reports 5326-FA-01 and G-RA-17 entitled, "Structural Analysis of the WNP-2 Rack and Fuel Assemblies for an Accidental Object Drop Loading Condition," and "Criticality Analysis of Dropped Object Accident for WNP-2 Spent Fuel Storage Racks," respectively.

9.2.6.3 Safety Evaluation

The condensate storage facilities are not required to assure any of the following:

a. The integrity of the reactor coolant pressure boundary.

b. The ability to shut down the reactor and maintain it in a safe shutdown condition.

c. The ability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures in excess of the guideline exposure of 10 CFR Part 100.

Although a minimum of 135,000 gallons is maintained in the condensate storage tanks as a source of water for the RCIC and HPCS pumps, the supply of water in the suppression pool is the emergency source of water for these pumps. The reserve of water is maintained by monitoring the level in the condensate storage tank and by preventing condensate transfer when this reserve level is reached. The RCIC and HPCS pumps are gravity fed from the condensate storage tanks.

The condensate storage tanks are Seismic Category II; however, they are located inside a Seismic Category I concrete dike which is designed to retain the condensate from both tanks. Drainage from the dike is routed to the radwaste system for processing. During precipitation, drainage from the dike is sampled and analyzed for radioactivity before being discharged and is monitored during discharge to the storm drain header. (See 9.3.3)

The evaluation of radiological considerations for the condensate supply system is presented in 2.4.13.3. Since the dike is designed to contain any condensate from a postulated tank rupture in conjunction with the heaviest recorded precipitation, the resulting offsite dose rate from this occurrence would be no greater than the value presented in 2.4.13.3.

For corrosion protection, the tanks are made of carbon steel with a 1/16 inch corrosion allowance and lined with a modified phenolic coating. Quality control for the application of the coating is in accordance with ANSI N101.4. The interior surfaces of the tanks are blasted to white metal in accordance with SSPC-SP-5 prior to the application of a minimum of 10 mils dry film thickness of Plasite 7155.

9.2.6.4 Testing and Inspection Requirements

The components are inspected and cleaned prior to installation into the system. The condensate storage tanks have all side wall, side wall to bottom, and nozzle joints examined in accordance with ASME Section III, Subsection NC-5000. The tank thickness is tested by magnetic means in accordance with ASTM E376. Underground portions of the HPCS and RCIC piping welds are 100% radiographed to ensure the integrity of the piping.

Instruments are calibrated during testing and automatic controls are tested for actuation at the proper set points. Alarm functions are checked for operability and limits during preoperational testing.

Automatic actuation of system components is tested periodically. The system is operated and tested initially with regard to flow paths, flow capacity, and mechanical operability as discussed in Chapter 14.

9.2.6.5 Instrumentation Requirements

Condensate storage tank level is monitored in the main control room. High and low-level alarms are provided to prevent overflow and to prevent the water level from dropping below the required reserve level for RPV makeup. Level switches provide low-low annunciation and interlock with the HPCS and RCIC systems. Building condensate supply pumps are provided with automatic controls for maintaining condensate supply pressure in the system headers so that condensate is immediately available for system process services.

The following parameters apply to the condensate storage tanks.

Reserve capacity for RPV makeup is provided between the set points of the low level switch (Elev. 453'-2") and the low-low level switch (Elev. 447'-4"). The elevation differential provides a reserve capacity of approximately 67,500 gallons per tank. (For 45 ft diameter tank, volume is approximately 11,900 gal/ft of height).

The air compressors take suction from the room through filter silencers which are 95% efficient in filtration of particles as fine as three microns. They discharge to the air receivers through intercoolers and aftercoolers for cooling and moisture separation. Moisture condensed in the coolers is discharged through float traps. Cooling water is supplied to the coolers by the plant service water system. Service and control air is directed from the air receivers to the system headers.

Service air is distributed from the header to quick-disconnect hose connections where it is used for pneumtic service equipment and maintenance throughout the plant. Service air is also distributed for plant services such as demineralizer resin mixing and filter and demineralizer backwashing (see Figure 9.3-1). In order to ensure the availability of the control air system for turbine and reactor plant operation, an air operated isolation valve is provided to isolate the service air system when the pressure in the service air header drops to 75 psig.

Control air for station instrumentation and controls is directed from the system header and is processed through one of two 100% capacity prefilters, an air dryer, and one of two 100% after filters prior to distribution for use. The filters are of the removable cartridge type and are arranged in parallel pairs to allow for replacement of a prefilter or an afterfilter without interruption of air flow. The twin tower air dryer contains a regenerable desiccant and is operated so that one section is in use while the other is being regenerated (four hours in service, four hours reactivation).

Table 9.3-1 presents the major characteristics of the air compressors, receivers, and the air dryer for the control and service air system.

9.3.1.2.2 Containment Instrument Air System

The containment instrument air system is primarily a nitrogen system which is shown in Figure 9.3-2. The major equipment characteristics of its air compressor components are presented in Table 9.3-1. The location of this system in the Seismic Category I reactor building is shown in Figures 1.2-3 and 1.2-5. With the installation of the containment inerting system discussed in 6.2.5.7, nitrogen is available to meet the requirements of the pneumatic components in containment from a 11,000 gallon (1 million scf) cryogenic storage tank. Thus, the system's two 100% capacity air compressors, associated coolers, a twin tower air dryer, filters, and an air receiver are normally in standby status. In addition, two nitrogen gas bottle banks and associated piping are provided as a backup to

either the cryogenic nitrogen supply or the compressor supplied air for seven of the main steam relief valves which perform the ADS function.

In the event the cryogenic nitrogen supply is unavailable, the compressed air components are placed into service. The compressors located in the reactor building take suction from the building atmosphere through intake filter-silencers which are 98% efficient in filtration of particles as fine as five microns. The air is then discharged through an aftercooler, a prefilter, a dryer, an afterfilter and air receiver to deliver dry, clean, pressurized air to the pneumatic control systems of the following valves inside the primary containment vessel:

a. Four main steam isolation valves and their accumulators,

b. Eighteen main steam safety/relief valves and their accumulators.

The two independent nitrogen bottle bank subsystems are provided to deliver apressurized nitrogen to seven of the safety/relief valves and accumulators. These seven valves perform the ADS function, if required, during postulated LOCA conditions. These nitrogen banks provide a 30-day supply of nitrogen for the ADS function. In addition, two remote nitrogen connections are provided (one for each bank) to allow supplementing of the existing 30-day supply to ensure the ADS function is operable indefinitely during isolation of the compressor loop. The remote connections are located outside of the reactor building to ensure access during post accident conditions. The in-place nitrogen for three main steam safety/relief valves and accumulators, and a second bank of 19 bottles supplying the remaining four main steam safety/relief valves and accumulators (see Figure 9.3-2).

The nitrogen bottles are located in the railroad lock of the reactor building to facilitate access. Under normal operating conditions, the controlled leakage boundary of the reactor building is maintained above the railroad lock so access is available to the bottles for recharging if required. The bottles are standard, commercially available units pressurized to 2490 psig. Each bottle has a capacity of 257 SCF. However, the bottles are mounted in accordance with Seismic Category I, Quality Class I requirements. The required quantity of bottles for each bank was conservatively based on providing a 30-day supply to the ADS valves to satisfy the long term post-LOCA demand based on the following:

solution, the storage tank is located in an area where the minimum environmental temperature is 70°F. Additionally, a tank heater is provided which turns on when the temperature drops below 75°F.

Cooldown of the nuclear system requires a minimum of several hours to remove the thermal energy stored in the reactor, cooling water, and associated equipment. Use of the main condenser and various shutdown cooling systems requires 10 to 24 hours to lower the reactor vessel to room temperature (70°F); this is the condition of maximum reactivity and, therefore, the condition that requires the maximum concentration of boron.

The specified boron injection rate is limited to the range of 6 to 25 ppm per minute. The lower rate assures that the boron is injected into the reactor in approximately two hours. This resulting negative reactivity insertion is considerably quicker than the reactivity increase caused by the cooldown. The upper limit injection rate assures that there is sufficient mixing so that boron does not recirculate through the core in uneven concentrations that could possibly cause reactor power to rise and fall cyclically.

The SLC system is required to be operable in the event of a station power failure; therefore, the pumps, heaters, valves, and controls are powered from the standby a-c power supply. The pumps and valves are powered and controlled from separate buses and circuits so that a single failure will not prevent system operation.

The SLC system and pumps have sufficient pressure margin, up to the system relief valve setting of approximately 1400 psig, to assure solution injection into the reactor above the normal pressure in the bottom of the reactor. The nuclear system relief and safety valves begin to relieve pressure above approximately 1100 psig. Therefore, the SLC system positive displacement pumps cannot overpressurize the nuclear system.

Only one of the two standby liquid control pumps is needed for system operation. If a redundant component (e.g., one pump) is found to be inoperable, reactor operation can continue during repairs. The time during which one redundant component upstream of the explosive valves may be out of operation is determined by the following: the probability of failure of both the control rod shutdown capability and the alternate

component in the SLC system and the fact that nuclear system cooldown takes several hours while liquid control solution injection takes approximately two hours. Since this probability is small, considerable time is available for repairing and restoring the SLC system to an operable condition while reactor operation continues. Assurance that the system will still fulfill its function during repairs is obtained by demonstrating operation of the operable pump.

The SLC system is evaluated against the applicable General Design Criteria as follows:

Criterion 2:

The SLC system is located in the area outside of the primary containment and below the refueling floor. In this location, it is protected by walls from external natural phenomena such as earthquakes, tornadoes, hurricanes and floods and also from the effects of internal postulated accident events.

Criterion 4:

The SLC system is designed for the expected environment in the compartment in which it is located. In this compartment, it is not subject to the conditions postulated in this criterion such as missiles, whipping pipes, and discharging fluids.

Criteria 20, 21, 23, and 25:

These criteria are applicable to protection systems only. The SLC system is a backup reactivity control system and is evaluated against Criteria 26, 27, 28, and 29 (see below).

Criterion 26:

The SLC system is a backup reactivity control system for the normal reactivity control systems. The requirements of this criterion do not apply to the SLC system.

Criterion 27:

This criterion is applicable to the SLC system. See the General Design Criteria Section for discussion of combined capability.

Criterion 28:

This criterion is not applicable to the SLC system.

Criterion 29:

The SLC system pumps and valves outboard of the isolation valves are redundant. Two suction valves, two pumps, and two injection valves are arranged and cross-tied such that operation of any one pair results in successful operation of the system. The SLC system also has test capability. A special test tank is supplied for providing test fluid for the yearly injection test. Pumping capability and suction valve operability may be tested at any time. A trickle current continuously monitors continuity of the firing mechanisms of the injection squib valves.

The SLC system is evaluated against the applicable regulatory guides as follows:

R.G. 1.26R2:

Because the SLC system is a reactivity control system, all mechanical components are at least Quality Group B. Those portions which are part of the Reactor Cooling Pressure Boundary are Quality Group A. This is shown in Table 3.2.1.

R.G. 1.29R1:

All GE supplied components of the SLC system which are necessary for injection of neutron absorber solution into the reactor are Seismic Category I. This is shown in Table 3.2.1.

Since the SLC system is located within its own compartment within the reactor building, it is protected from flooding, tornadoes, and internally and externally generated missiles. SLC system equipment is protected from pipe break by providing adequate distance between the seismic and non-seismic SLC system equipment where such protection is necessary. In addition, appropriate distance is provided between the SLC system and other piping systems. (See 3.6.2.5.4).

It should be noted that the SLC system is not required to provide a safety function during any postulated pipe break event. This system is only required during an extremely low probability event when all of the control rods are assumed to be inoperable while the reactor is at normal full power operation. Therefore, the protection provided is considered over and above that required to meet the intent of APCSB 3-1 and MEB 3-1.

This system is used in special plant capability demonstration events cited in Appendix A of Chapter 15. Specifically, Events 51, 52, and 53, which are extremely low probability non-design basis postulated incidents. The analyses given there are to demonstrate additional plant safety consideration far beyond reasonable and conservative assumptions.

A system-level, qualitative-type failure mode and effects analysis relative to this system's ability to meet single failure criterion is discussed in 15.A.6.6.

9.3.5.4 Testing and Inspection Requirements

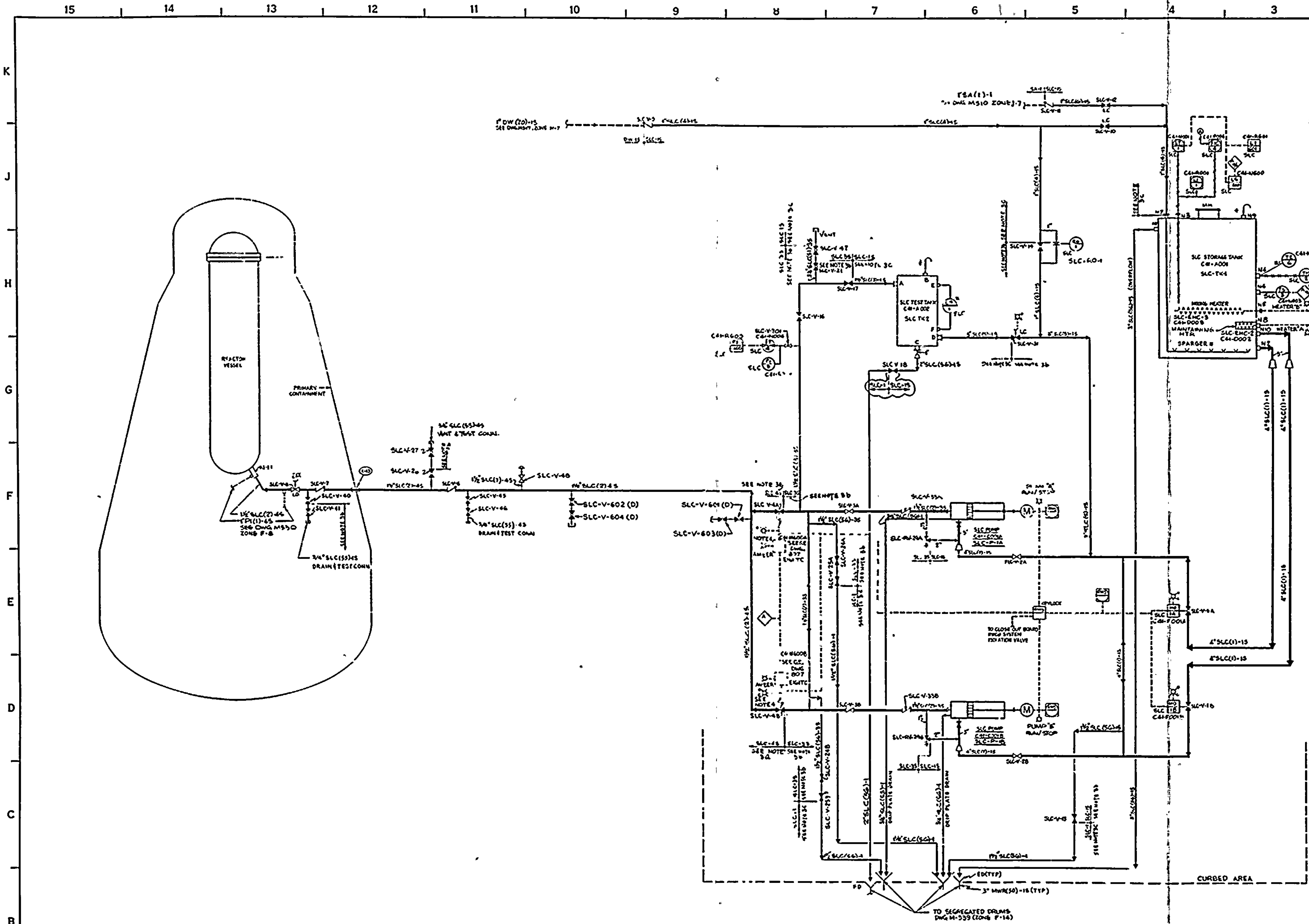
Operational testing of the SLC system is performed in at least two parts to avoid inadvertently injecting boron into the reactor.

With the valves from the storage tank closed and the valves to and from the test tank opened, demineralized water in the test tank can be recirculated by locally starting either pump.

The injection portion of the system can be functionally tested by valving the suction line to the test tank and actuating the system from the main control room. Both injection valves open on actuation. System operation is indicated in the main control room.

After functional tests, the injection valve shear plugs and explosive charges must be replaced and all the valves returned to their normal positions.

The test tank contains demineralized water for approximately 3 minutes of pump operation. Demineralized water from the makeup system or the condensate storage is available for refilling or flushing the system.



NOTES:

1. ALL INSTRUMENT ROOT VALVES NOT LABELED SHALL BE 3/4" GLOBE VALVES UNLESS SPECIFICALLY NOTED OTHERWISE.
2. ALL ITEMS MARKED # ARE FURNISHED WITH ASSOCIATED EQUIPMENT.
3. PIPING, VALVES AND ASSOCIATD COMPONENTS ON THIS DWG. SHALL BE CLASSIFIED AS FOLLOWS:
   a. ALL PIPING, VALVES AND COMPONENTS DESIGNATED AS SLC-45 EXCEPT AS NOTED.
      SEISMIC CATEGORY — I
      QUALITY CLASS — I
      CODE GROUP — A
   b. ALL PIPING, VALVES AND COMPONENTS DESIGNATED BY THE NOTE "SEE NOTE 3b.
      SEISMIC CATEGORY — II
      QUALITY CLASS — II
      CODE GROUP — B
   c. ALL PIPING, VALVES AND COMPONENTS DESIGNATED BY THE NOTE "SEE NOTE 3c" OR THE SYSTEM DESIGNATION SLC (56).
      SEISMIC CATEGORY — II ●
      QUALITY CLASS — II
      CODE GROUP — D
      ● SEE SECTION 15Q, CONTRACT 215.
4. THESE VALVES WILL BE INCLUDED IN THE "ISOLATION VALVE POSITION DISPLAY PANEL."

LEGEND

1. ALL VALVES SUFFIXED WITH A (V) EQUAL A 3/4" VENT VALVE
2. ALL VALVES SUFFIXED WITH A (D) EQUAL A 3/4" DRAIN VALVE

AMENDMENT NO. 22
January 1982

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	STANDBY LIQUID CONTROL SYSTEM REACTOR BLDG.	FIGURE 9.3-13

j. Fire hydrants with 200 ft. of 2-1/2-inch hose, emergency light, axe, and accessories as required by NFPA No. 24 placed in adjacent protected houses are located around the main plant buildings along the fire protection loop. Hydrants with 200 ft. of 2-1/2-inch hose are provided between the remote office warehouses and the cooling towers, and at each standby service water pumphouse. Two hydrants with hose houses are on either side of the five warehouses. These two hydrants have 500 ft. length hoses.

k. A series of 12-inch lines lead from the 12-inch plant fire protection loop to various building standpipes. Each line contains an outside 12-inch isolation gate valve with a post indicator. These lines are listed below:

(1) 12-inch to radwaste building standpipe RWB-2 (Stairway A-8)

(2) 12-inch to radwaste building standpipe RWB-1 (Stairway A-7)

(3) 12-inch to reactor building standpipe RB-2 (Stairway A-6)

(4) 12-inch to reactor building standpipe RB-1 (Stairway A-5) and service building standpipe SB-2 (Stairway A-10)

(5) 12-inch to service building standpipe SB-1 (Stairway A-9) and turbine building standpipe TGB-1 (Stairway A-1) and on to a 10-inch line to standpipe TGB-4 (Stairway S-1)

(6) 12-inch to turbine generator building standpipe TGB-2 (Stairway A-3)

(7) 12-inch to transformer yard (1st branch)

(8) 12-inch to turbine generator building standpipe TGB-3 (Stairway A-4) and on to a 6-inch line to standpipe TGB-5 (Stairway S-2)

(9) 12-inch to transformer yard (2nd branch)

(10) 12-inch to diesel generator corridor

l. Most standpipes are located in protected stairways. Except for: two open stairways in the turbine generator building, one open stairway in the diesel generator builing and two open stairways in the Reactor Building. Each contain a drain valve (angle valve), hose racks on each landing, takeoffs to sprinkler or other water fire protection systems where applicable, and an air vent valve and pressure gauge at the top of each standpipe. Water is piped to various systems from the standpipes. To insure availability of primary and secondary fire protection, the following standpipes have been interconnected: TGB-1 and TGB-2, TGB-5 and RWB-1, DG-1 and 12" branch line to RWB-1, and RB-1 and RB-2. Standpipes and hose racks are located at approximate intervals so that all portions of the protected areas are within 30 ft. of a nozzle when attached to a hose.

m. Each hose rack station is UL listed and provided with a 2 1/2" hose valve 150 ft. of 1 1/2" hose in non safety related areas and 100 ft. of 1 1/2" hose in safety related areas.

n. A 6" line extends to the field office buildings.

o. Two lines (one 3" and one 8") extend to the warehouse.

The 50 gpm jockey pump maintains a closed system pressure of 125 to 138 psig. A drop in system pressure below 110 psig will cause the first electric motor driven fire pump to automatically start. The sequential starting of the second electric motor driven pump and the diesel driven pump are controlled by a time delay relay. One or both of these pumps will start if the first electric driven pump cannot maintain system pressure. Alarms are provided within the annunciator for each of the three fire pumps in the event of failure of a pump to start.

The secondary pumping capacity in pumphouse no. 3 would automatically start if system pressure dropped below 100 psig. The secondary diesel pump is also controlled by a time delay relay. During outage of the main jockey pump in the CWPH, the secondary jocky pump would maintain system pressure of 115-125 psig. Each electric motor driven fire pump controller contains manual start - stop controls with provision for

Operation of the fuel storage tank transfer pump is controlled manually when fuel is being transferred through the interconnecting line from storage tank "A" to day tank "B" or from storage tank "B" to day tank "A". High and low level annunciation of the day fuel levels will provide a warning of overfilling or depleting the day tank when the transfer pump is on manual control.

The fuel oil supply from the day tanks to each diesel engine being served consists of two mutually redundant systems. Either system is capable of supplying fuel oil to the engine. Each system contains a fuel supply line, strainer, fuel oil pump, duplex filter, pressure gage, and relief and check valves. Separate fuel return lines from the relief valves to the day tanks are provided for each system on diesel generators 1A and 1B. The HPCS diesel utilizes a common return line to the day tank.

One of the fuel supply pumps is mechanically driven by the engine and is normally used during engine operation. The other supply pump is driven by a 120 volt DC motor and is used to fill the fuel oil system and fuel header prior to initial operation and after maintenance has been performed on system piping and components. The DC motor driven pump is also available for engine operation in the event fuel supply through the engine driven pump system fails.

The fuel pumps are located 2.3 feet higher than the suction pipes inside the day tank. The fuel pumps are designed to operate at a slight negative suction pressure.

The fuel oil supply and return piping is not exposed to ignition sources such as open flames or hot surfaces. The transfer lines between the storage and day tanks are buried and the lines between the day tanks and engines are routed through trenches in the diesel generator rooms.

The fuel oil day tank is located in a separate ventilated room which is sized to contain the full contents of the tank should a leak develop. For discussion of fire protection see 9.5.1.

The fuel oil storage tanks are provided with Seismic Category I individual fill and vent lines which are protected against the entry of contaminants. The fill lines are provided with screwed caps and the vent lines are provided with flame arrestors. The fill and vent lines terminate at 3.25 and 6.0 feet, respectively, above plant grade, which prevents direct seepage of any ground water into the storage tanks. The fuel oil is sampled periodically to detect the presence of water

or contamination before it could present a problem. Missile protection is not necessary since, in the event of a fill line rupture due to a missile, fuel oil suction would be switched to the other storage tank (diesel engine generator 1A and 1B). In the event that fill line damage due to a missile, the pump-out connection which is protected by a metal encloure, located at ground level, may be utilized for the fuel oil filling operation. The seven-day operation provided by one storage tank will permit both diesel generators to operate for minimum of 3-1/2 days. Within the 3-1/2 days the ruptured line will be repaired.

During filling of the storage tanks and until the time required for sediment to settle in the bottom of the storage tank, fuel oil supplied to the day tanks will be transferred from the other storage tank.

Diesel fuel oil conforming with ASTM Standard 0975-74 Grade 2-D will be provided for operation of the emergency diesel generators. This grade of diesel fuel complies with the engine manufacturer's requirements and is available from local distribution sources as discussed in 9.5.4.3.

Equipment design characteristics for the fuel oil supply system are shown in Table 9.5-6.

9.5.4.3 Safety Evaluation

The entire diesel oil supply system is located within the confines of a Seismic Category I building except for the buried storage tank. Each subsystem oil storage tank transfer pump, day tank, and diesel generator set is physically separated within separate concrete enclosures designed to protect against missiles, in compliance with 3.5, and to provide fire protection. No high or moderate energy piping is present in the diesel generator building. The oil storage tanks are buried for protection so that storage tank failure is completely contained within the soil at a level below any building penetration or access opening. Each storage and day tank is provided with a vent directly to the outside atmosphere. In addition, the enclosures are provided with exhaust ventilation to the outside atmosphere to ensure that any diesel fuel vapors are maintained well below the combustible limit. The enclosures are automatically monitored by temperature detectors which initiate the pre-action sprinkler system in the event of fire (FSAR Appendix F, Page F.2-37). All storage and day tank vents are equipped with flame arrestor devices.

Instrumentation and controls regulate pump recirculation flow rate for the condensate pumps, condensate booster pumps, and reactor feed pumps. Measurements of pump suction and discharge pressures are provided for all pumps in the system. Sampling means are provided for monitoring the quality of the final feedwater, refer to 9.3.2. Temperature measurements are provided for each stage of feedwater heating and these include measurements at the inlet and outlet on both the steam and water sides of the heaters. Steam-pressure measurements are provided at each feedwater heater. Instrumentation and controls are provided for regulating the heater drain flow rate to maintain the proper condensate level in each feedwater heater shell and heater drain tank. High-level alarm and automatic dump-to-condenser on high level are provided.

10.4.7.3 Safety Evaluation

During operation, radioactive steam and condensate are present in the feedwater heating portion of the system which includes the extraction steam piping, feedwater heater shells, heater drain piping, and heater vent piping. Shielding and controlled access are discussed in Chapter 12. The condensate and feedwater system is designed to minimize leakage with welded construction utilized throughout the piping system. Feedwater heater shell side relief valve discharges and operating vents are routed to the condenser.

The condensate and feedwater system is not required to effect or support the safe shutdown of the reactor or perform safety functions.

If it is necessary to remove a component such as a feedwater heater, pump, or control valve from service, continued operation of the system is possible by use of the multistream arrangement and the provisions for isolating and bypassing equipment and sections of the system.

The analysis of both the condensate and feedwater individual component failures is bounded by the feedwater component system failure analysis. These analyses are provided in 15.1.1, "Loss of Feedwater Heating", 15.1.2, "Feedwater Controller Failure", and 15.2.7, "Loss of Feedwater Flow". The effects of equipment malfunction on the reactor coolant system are provided in 15A. Included also in 15.6.6, "Feedwater Line Break", are the isolation provisions that minimize release of radioactivity to the environment.

Criteria for feedwater isolation of the reactor coolant system is presented in 6.2.4.

10.4.7.4 Tests and Inspections

Each feedwater heater, heater drain tank, condensate pump, condensate booster pump, reactor feedwater pump, and system valves are shop hydrostatically tested at 1-1/2 times their design pressure. All pumps are shop performance tested. All tube joints of feedwater heaters are shop leak tested. Prior to initial operation, the completed condensate and feedwater system receives a field hydrostatic test in accordance with Chapter 14.

Pressure, temperature, conductivity, and flow instrumentation are provided to monitor system performance during operation. Inservice inspection of applicable reactor feedwater piping is presented in 5.2.4.

10.4.7.5 Instrumentation

Pressure, temperature, conductivity, and flow instrumentation are provided to monitor system performance. The operation of the hotwell makeup and high level dump valves is controlled by the hotwell level controller (Figure 10.4-5).

10.4.8 STEAM GENERATION BLOWDOWN SYSTEM

This section is not applicable to a BWR.

10.4.9 AUXILIARY FEEDWATER SYSTEM

This section is not applicable to a BWR.

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LIST OF FIGURES

Chemical wastes collected in the chemical waste tank are from the following sources:

a. Detergent drains
b. Shop decontamination solutions
c. Reactor and turbine building decontamination drains
d. Low purity wastes from either the equipment or floor drain subsystems
e. Filter demineralizer element chemical cleaning solutions
f. Battery room drains
g. Chemical system overflows and tank drains
h. Laboratory drains

The quantities of these wastes are summarized in Table 11.2-4. These chemical wastes are of such high conductivity as to preclude treatment by ion exchange and the radioactivity concentrations are variable. These wastes are processed by a waste concentrator (evaporator).

The evaporator concentrates are processed by the solid waste management system, and the distillate is routed to the distillate tank. After analysis, the distillate may be routed through a polishing demineralizer to further reduce impurities, recycled through the evaporator, or sent directly to condensate storage for plant reuse. As with the other subsystems, when high purity water storage capacity is exceeded, some liquid is discharged via the blowdown line.

11.2.2.2.4 Shared Equipment

Other than serving as mutual backup, main process equipment normally is not shared between subsystems. Auxiliary equipment not in the direct process stream is shared between subsystems. Shared equipment includes:

a. The waste precoat tank and waste precoat pump are shared between the waste collector filter and the floor drain filter.
b. The waste filter aid tank is shared between the waste collector filter and floor drain filter.
c. The resin addition tank is shared between the waste demineralizer, floor drain demineralizer and polishing demineralizer.

d. The chemical addition tanks, caustic and acid, and associated pumps are shared between the waste collector tank, floor drain collector tank, detergent tanks and chemical waste tanks.

11.2.2.2.5 Surge Capacities

The radwaste system process data is the basis for sizing of the equipment. Tables 11.2-2, 11.2-3 and 11.2-4 list startup flows, daily flows and maximum flows for the equipment drain subsystem, floor drain subsystem and chemical waste subsystem respectively. Anticipated operational occurrences such as startup operations, equipment malfunction and shutdown operations are accounted for in these tabulations. The bases for these types of values are presented in NEDO-10951, "Releases from BWR Radwaste Management Systems", July, 1973.

The surge storage and process capacities can be envisioned by comparing the normal and maximum daily volumes, listed in Table 11.2-6, with the design flow rates of pumps and tank volumes listed in Table 11.2-5. Alternate processing, rather than bypass operations, are used during equipment downtime. The equipment and floor drain subsystems are sized such that with either subsystem inoperative; the remaining subsystem is capable of processing the maximum expected volume of both subsystems. Additionally, the waste surge tank provides reserve storage capacity. The chemical waste subsystem incorporates two parallel processing paths. Cross-connections allow individual components from either process path to serve as a substitute in the other process path. The parallel path processing and adequate storage capacity ensures that inoperability of any component in this subsystem will not limit plant operation.

11.2.2.2.6 Design Data

11.2.2.2.6.1 Design Parameters

The design pressures and temperatures for individual components are listed in Table 11.2-5. Collection and storage tanks are designed for atmospheric pressure. The mixed bed demineralizer units, precoat filter units, and concentrators are pressure vessels. The quality classification for the system is Quality Group C as defined in 3.2.

11.2.2.2.6.2 Design Features

Chapter 12 discusses the design features incorporated in the system to maintain occupational exposure as low as reasonably achievable (ALARA). As depicted on the general arrangement drawings in 1.2, the radwaste processing equipment is located in shielded rooms and cells. Process lines which penetrate shield walls are routed to prevent a direct radiation path from the tanks and equipment to normally occupied areas.

11.3 GASEOUS WASTE MANAGEMENT SYSTEMS

11.3.1 DESIGN BASES

11.3.1.1 Design Objective

The objective of the gaseous waste management system is to process and control the release of gaseous radioactive effluents to the site environs so as to maintain as low as reasonably achievable the exposure of persons in unrestricted areas to radioactive gaseous effluents (Appendix I to 10 CFR 50, May 5, 1975). This is to be accomplished while maintaining occupational exposure as low as reasonably achievable and without limiting plant operation or availability.

11.3.1.2 Design Criteria

The gaseous waste management systems are designed to limit the dose to off-site persons from routine station releases to significantly less than the limits specified in 10 CFR 20 and to operate within the emission rate limits established in the technical specifications.

As a design basis for this system, an annual average noble radiogas source term (based on 30-minute decay) of 100,000 μCi/sec of the "1971 Mixture" as discussed in 11.1 is used. Table 11.3-1 indicates the design basis noble radiogas source terms referenced to 30-minute decay.

The annual average exposure at the site boundary during normal operation and anticipated operational occurrences from gaseous effluents does not exceed the dose objectives of Appendix I to 10 CFR 50. The radiation dose design basis for the treated off-gas is to delay the gas until the required fraction of the radionuclides has decayed and the daughter products are retained by the charcoal and the HEPA filters.

The gaseous radwaste equipment is selected, arranged and shielded to maintain occupational exposure as low as reasonably achievable. The design of the system was accomplished prior to the issuance of Regulatory Guide 8.8. However, the system incorporates substantially the guidance provided in this regulatory guide. The gaseous effluent treatment system is designed to the requirements of General Design Criteria as follows:

General Design Criteria 60

The system has sufficient capacity to reduce the off-gas activity to permissible levels for release during normal operation, including anticipated operational occurrences, and to alleviate any termination of releases or limitation of plant operation due to unfavorable site environmental conditions.

General Design Criteria 64

Continuous monitoring of activity levels in the system upstream of the delay line provides advance notice of any potentially significant increase in releases. Continuous monitoring of the system effluent, with automatic isolation at activity levels corresponding to administrative release limits and annunciation at lower levels, along with continuous monitoring of the reactor building elevated release duct, radwaste building ventilation exhausts and turbine generator building ventilation exhausts, provide assurance that activity releases to the environment will in all events be maintained within established limits.

11.3.1.3 Equipment Design Criteria

A list of the off-gas system major equipment items which includes materials, process conditions, and number of units supplied, is provided in Table 11.3-2. Equipment and piping is designed and constructed in accordance with the requirements of the applicable codes as given in Tables 3.2-1 and 3.2-2.

The quality group classifications of the various systems are shown in Table 3.2-1. Seismic category, safety class, quality assurance requirements, and principal construction codes information is contained in 3.2. The system is designed to Quality Group Classification C.

The reactor building, turbine generator building, and radwaste building contain radioactive gas sources. The design bases and characteristics for the ventilation systems for these three buildings are described in 9.4.

Equipment and components used to collect, process, or store gaseous radioactive waste are not designed as Seismic Category I. Conservative analyses similar to those presented in Reference 11.3-5 demonstrate that equipment failure will not result in offsite doses exceeding 0.5 Rem. The failure of the off-gas system, the related failure of the steam-jet air ejector, lines and the gland sealing system are analyzed in 15.7.1.

11.4 SOLID WASTE MANAGEMENT SYSTEM

11.4.1 DESIGN BASIS

11.4.1.1 Design Objective

Power plant operation results in various solid radioactive wastes that require disposal. These wastes are in the form of powdered ion exchange resins from filter demineralizers, expended bead resins from deep bed demineralizers, bottoms from the decontamination solution concentrator, and miscellaneous dry materials such as paper, rags and laboratory wastes.

The objective of the solid waste management system is to collect, monitor, process, and package these waste products in a suitable form for off-site shipment and burial. The system redundancy and capacity is sufficient to prevent impact on plant operation and availability due to anticipated operational occurrences. The system is designed to minimize exposure to operating personnel and to prevent undue exposure to persons outside of the restricted area.

11.4.1.2 Design Criteria

In designing the system to meet the stated objective, the following criteria were applied:

The system has the capacity to handle the volumes of waste from normal operation and anticipated operational ocurrences. The expected annual volumes of wet solid wastes are shown in Table 11.4-4.

The system is designed to process the quantity of waste and concentration of radionuclides listed in Table 11.4-3 while maintaining occupational exposure as low as reasonably achievable (ALARA). The design of the system was accomplished prior to the issuance of Regulatory Guide 8.8. However, the system does incorporate substantially the guidance provided in this regulatory guide. This is done by controlling the pipe run locations for shielding and exposure considerations, placing the process equipment in shielded areas and by providing remote operating stations. Shielding for the solidification area is designed for the highest radioactivity source, reactor water cleanup sludge. The equipment layout is shown on general arrangement drawings in 1.2.

In keeping with ALARA and Appendix I to 10 CFR Part 50 the solid waste management system's contribution to off-site doses is minimized by venting the process equipment to a filtered ventilation exhaust system and by directing the ventilation air flow from areas of low airborne contamination to areas of higher airborne contamination. The filtered ventilation radioactive releases are discussed in 11.3.

The solid waste management system operations and procedures are designed to limit the dose to off-site persons from station operations to significantly less than the limits specified in 10 CFR Part 20. Water separated in processing is returned to the liquid waste management system for treatment as described in 11.2.2.1.5 and shown in Figure 11.4-1.

The system can accommodate a variety of shipping container sizes and shapes with and without shields. Provisions are made for the remote detection and removal of loose surface contamination on the waste containers. The radiation level of the waste mixing tank contents is monitored so that provisions can be made to ensure that shipping regulation radiation levels are not exceeded. Compliance with applicable regulations, e.g., 10 CFR Part 71 and 49 CFR Part 173 is discussed in 11.4.2.10.

The solid waste management system upstream of and including the centrifuges is designed to Quality Group C as defined in 3.2. The solid waste management system downstream of the centrifuges is designed to Quality Group D. Table 11.4-1 lists capacities, design pressure, and design temperature of the equipment.

The solid waste management system is not designed to Seismic Category I. It is located in the Seismic Category I portion of the radwaste building. The seismic classification of the radwaste building is discussed in 3.8.4.1.2.

General Design Criteria 63

Area radiation monitors and waste mixing tank radiation monitors ensure that excessive radiation levels associated with the solid waste management system are detected and alarmed.

11.4.2 SYSTEM DESCRIPTION

11.4.2.1 General

The sources of the various radioactive solid waste inputs to the system are shown on Figure 11.2-1. Table 11.2-6 shows the expected frequency of input, the quantities of solids generated, the radioactivity level of the solids after accumulation, and the volume of liquid utilized in sluicing accumulated solids to the solidification equipment. The excess liquid is subsequently returned to the liquid waste management system. These values are based on experience from operational BWR nuclear power stations. Figure 11.4-1 shows the waste packaging portion of the solid waste management system. The phase separation and concentration portions of the system are shown on Figures 3.2-11, 10.4-4, 11.2-2, 11.2-3, and 11.2-4. Tanks containing radioactive waste are provided with overflow connections which direct any overflow to drain sumps.

The solid waste processing areas are located in the radwaste building. Both wet and dry solid wastes are processed. Wet solid wastes include backwash sludge from the reactor water cleanup system, the condensate filter demineralizer system, the fuel pool filter demineralizers, the floor drain filter and the waste collector filter; spent resin from the floor drain demineralizer, the waste demineralizer, the distillate polishing demineralizer; and concentrated bottoms from the decontamination solution concentrators. Dry solid wastes include items such as rags, paper, small equipment parts and laboratory wastes.

11.4.2.2 Radwaste Disposal System for Reactor Water Cleanup Sludge

The backwash discharge from the cleanup filter demineralizers is collected and concentrated in two 4500 gallon cleanup phase separators which are located below the cleanup demineralizers in the radwaste building. After several backwashes are accumulated, the concentrated waste is transferred to either the centrifuges or waste mixing tanks for dewatering.

The cleanup phase separators are designed to concentrate the sludge from 0.5% by weight solids to 5% by weight solids by sedimentation and decantation of the slurry. While the working separator is filling, the other previously filled tank is held isolated to allow additional decay of sludge activity.

After each backwash batch is received by the working separator, the batch is allowed to settle for a period of time and the decantate is then transferred by pumping to the waste collector tank. When sufficient sludge has accumulated, the working separator is isolated and allowed to stand for a period to permit radioactive decay. At the end of this decay period (1 to 2 months) the sludge is fluidized to the 5% weight slurry and transferred by pumping to either the centrifuges or waste mixing tanks for dewatering.

11.4.2.3 Radwaste Disposal System for Condensate Demineralizer Sludge

The backwash discharge from the condensate filter demineralizers is collected in the condensate backwash receiving tank which is located below the condensate filter demineralizers in the radwaste building. After collection, the waste is transferred by pumping to one of the two condensate phase separators for processing.

Operation of the condensate phase separators is similar to that for the cleanup phase separators. Backwash sludge is received at 0.5% by weight solids and concentrated to 5% by weight solids, allowed to stand for a period of radioactive decay and then decanted and transferred by pumping to either the centrifuges or waste mixing tanks for dewatering.

11.4.2.4 Radwaste Disposal System for Fuel Pool, Floor Drain and Waste Collector Filter Sludge

Backwash sludge wastes from the fuel pool filter demineralizers, floor drain filter, and waste collector filter are backwashed to the waste sludge phase separator tank which is located in the radwaste building. The waste sludge phase separator is designed to concentrate the sludge from 0.5% by weight solids to 5% by weight solids by sedimentation and decantation.

After each backwash batch is received by the separator it is allowed to settle for a period of time and the decantate is then transferred by pumping to the floor drain collector tank.

When a predetermined quantity of waste sludge has been accumulated, the sludge is fluidized to a 5% by weight slurry and transferred by pumping to either the centrifuges or waste mixing tanks for dewatering.

radwaste control room. This level transmitter also drives a level indicator on the local control panel and provides control functions for the decant pump, the sludge discharge pump, and the phase separator inlet selector valve. Sludge level indication is accomplishd by a pair of ultrasonic probes positioned in the phase separator.

11.4.2.13.2 Condensate Phase Separator Instrumentation

The condensate phase separators level instrumentation is the same as that described for the reactor water cleanup phase separators.

11.4.2.13.3 Waste Sludge Phase Separator Instrumentation

The waste sludge phase separator has total liquid level indication. It uses an air bubbler and a pressure sensing level transmitter. In addition to the level gage and high level alarm in the radwaste control room, the level transmitter provides control inputs to the decant pump, the stop and flush circuit on the sludge discharge pump, and the discharge valves from the waste collector and floor drain collector tanks to the waste sludge phase separator.

11.4.2.14 Spent Resin Tank Instrumentation

Level indication for the spent resin tank is essentially the same as that described for the cleanup phase separators utilizing an air bubbler and level transmitter for total liquid level and ultrasonic probes for resin level.

11.4.2.15 Concentrated Waste Measuring Tank Instrumentation

This tank is equipped with a level transmitter that drives a level indicator in the radwaste control room.

11.4.2.16 Waste Mixing Tank Instrumentation

The waste mixing tanks are equipped with ultrasonic level detectors that drive indicators, a recorder and high level alarms on the solid waste control panel. They also provide control signals to stop the centrifuges on high level. The waste mixing tanks are also provided with radiation detectors. These monitors which have a range of 10 mR/hr to 100 R/hr drive a recorder and alarms on the solid waste control panel.

11.4.3 PROCESS CONTROL PROGRAM

11.4.3.1 Objective

The objective of the process control program is to assure the complete solidification of all wet wastes. To meet this objective the process control program has incorporated the recommendations set forth in NUREG-0800, Branch Technical Position - ETSB 11-3 and NUREG-0473.

11.4.3.2 Process Control Program

The cement-sodium silicate solidification process is designed to produce a freestanding solid with essentially no free liquid. Due to the latitude of waste and cement proportions that will solidify under the influence of sodium silicate, the solidification system can be operated with mixing ratios that assure solidification occurs even with nominal waste stream variations.

To assure that the system will produce an acceptable solidified product, the following process control elements have been incorporated:

a. Process control systems

b. Process control interlocks

c. Process control logic

d. Setpoints and operating limits control

e. Laboratory verification of formulations

f. Preoperational testing

g. Maintenance, calibration, and formulation control

h. Unanticipated wastes

11.4.3.3 Process Control Systems

The processing and material handling equipment is fully instrumented and the entire operation from mixing and filling to placing containers in storage is monitored and controlled from the solid waste system control panels.

The levels of waste and solidification materials are monitored at key points in the system using ultrasonic sensors. Pressure switches on the discharge of each proportioning pump give positive indication of pump operation and the flow of process materials. A flow sensor provides positive indication of cement flow. The waste mixing tank temperature is monitored and any temperature outside of present limits is annunciated at the control panel. The level of waste-cement mixture, in the disposable container, is monitored with an ultrasonic sensor and the flow of waste-cement mixture is automatically stopped upon sensing high level by the ultrasonic signal.

The flow monitoring system provides a permanent record of the quantity of waste and solidification agents in each container by means of a four-pen recorder.

The solidification process is selected, initiated, and monitored at the solid waste system control panel. The control panel contains a graphic display with system control switches, indicators, and readouts arranged in mimic tracings for ease and accuracy of operation. The control panel graphic display includes valve position indicating lights, motor operation indicating lights, level indicators for storage tanks and the waste container under the fillport, alarm annunciators, process select and master control switches, closedcircuit television monitors, indication monitor readouts, and controls and indicators for the disposable container handling equipment.

11.4.3.4 Process Control Interlocks

Process control interlocks prevent system operation if components malfunction or inadvertent lineups are made. These interlocks ensure that the system operates to solidify wet waste only if the following conditions are met:

a. A waste container is in place under the fillport.

b. The fillport seal plate is down.

c. The waste container is not full.

d. The waste mixing tank and piping heat tracing is energized and above the minimum required temperature.

e. The waste tank mixer is operating.

f. Cement aeration blower and bag filter are operating.

g. Waste tank level is above minimum.

h. Cement storage tank and feed hopper levels are above minimum.

i. Sodium silicate day tank level is above minimum.

j. Cement is actually flowing.

k. Sodium silicate is actually flowing.

l. Waste is actually flowing.

m. Waste-cement mixture is actually flowing.

The process selector switch and master start switch are interlocked such that the initial process would continue even if the process selector position were changed and/or the master start switch were depressed again.

The ultrasonic level monitors on the bulk storage tanks prevent system startup if either insufficient cement, waste, or sodium silicate is available for a complete process run.

11.4.3.5 Process Control Logic

The solidification system contains a logic control unit that controls the sequence and duration of process operations. The control unit contains logic control steps designated to perform internal checks of system conditions prior to initiation of subsequent process operations. Continuation of the process is dependent upon satisfying the conditional set/points. Any time during a process run that a setpoint is exceeded the process is automatically stopped and annunciated. Conditional setpoints will be determined based on preoperational test results.

11.4.3.6 Setpoints and Operating Control

The laboratory verification of solidification formulations and the confirming data developed by full scale preoperational testing will determine the setpoint values for each component in the system. Setpoints for parameters that are critical to the solidification process are preset to assure operation at the required conditions. The critical setpoint conditions are segregated in a locked subpanel in the rear of the main process control panel. This provides for direct administrative control of access to and adjustments of the control system setpoints.

The present values for setpoints and the preset ratios of waste, cement, and sodium silicate are different for each particular type of waste. The proper values are automatically selected by an operator-controlled master switch with positions labelled according to the type of waste to be processed.

11.4.3.7 Laboratory Verification of Formulations

The design of the solidification system is based on laboratory, pilot plant, and full-scale system studies of each type of waste. Laboratory verification allows setpoint adjustments to compensate for plant variations from typical formulations.

Successful solidification of wet wastes is assured by development of solidification blends based on plant-waste composition coupled with laboratory solidification studies.

11.4.3.8 Preoperational Testing

The preoperational testing program will be designed to functionally test the solidification equipment under all modes of operation. The test program, in conjunction with the laboratory verification program, will determine the optimum setpoints and operating parameters necessary to insure the solidification of all wet waste matrices. In addition, the test program will also define the minimum and maximum parameter boundaries that still produce a freestanding solid with essentially no free liquid.

The solidified waste containers will be sectioned and the contents examined for homogeneity and the absence of free liquid. The results of this inspection will be documented and compared to the laboratory verification program results. If the results from both tests indicate the presence of a freestanding solid with essentially no free liquid, this information will then be recorded and used for future reference. Discrepancies between the laboratory verification program results and the actual results will be documented and resolved.

11.4.3.9 Maintenance, Calibration, and Formulation Control

Control of solidification parameters is assured by a maintenance, calibration, and formulation control program. The Operation and Maintenance Manual, supplied by the vendor, recommends specific maintenance and calibration frequencies for the various system components. The manual also recommends that a periodic verification of the quality and condition of the cement and sodium silicate in the plant storage tanks be performed. Laboratory verification of the effectiveness of the solidification formulations are performed to determine if system setpoint adjustments are required to maintain optimum product quality.

11.4.3.10 Unanticipated Wastes

From time to time, it will become necessary to solidify wet wastes which have never been verified by laboratory formulation. Examples of such wastes are: decontamination solutions and laundry detergents.

When this occurs the solidification requirements must be determined on a case-by-case basis using the laboratory verification of formulation program. Records and results of such testing will be retained for future use.

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TABLE 11.4-1

SOLID WASTE MANAGEMENT SYSTEM MAJOR EQUIPMENT ITEMS

Cleanup Phase Separators - 2 Required

Construction: Stainless steel shell and internals. Atmospheric design pressure. 250°F design temperature. Capacity - 4500 gallons/each.

Cleanup Sludge Discharge Mixing Pump - 1 Required

Construction: Steainless steel. Design pressure - 150 psig. Design temperature - 150°F. Capacity - 210 gpm at 170 feet TDH.

Cleanup Decant Pump - 1 Required

Construction: Stainless steel. Design pressure - 150 psig. Design temperature - 150°F. Capacity - 53 gpm at 50 feet TDH.

Condensate Backwash Receiving Tank - 1 Required

Construction: Steainless steel shell and internals. Atmospheric design pressure. 150°F design temperature. Capacity - 19,000 gallons.

Condensate Backwash Transfer Pump - 1 Required

Construction: Steainless steel. Design presure - 150 psig. Design temperature - 150°F. Capacity - 450 gpm at 50 feet TDH.

Condensate Phase Separator - 2 Required

Construction: Epoxy-coated carbon steel shell, stainless steel internals. Atmospheric design pressure. 250°F design temperature. Capacity - 23,500 gallons/each.

Condensate Sludge Discharge Mixing Pump - 1 Required

Construction: Stainless steel. Design pressure - 150 psig. Design temperature - 150°F. Capacity - 420 gpm at 160 feet TDH.

Condensate Decant Pump - 1 Required

Construction: Stainless steel. Design pressure - 150 psig. Design temperature - 150°F. Capacity - 450 gpm at 50 feet TDH.

TABLE 11.4-1 (Continued)

Waste Sludge Phase Separator Tank - 1 Required

Construction: Epoxy-coated carbon steel, stainless steel internals. Atmospheric design pressure. 150°F design temperature. Capacity - 13,000 gallons.

Waste Decant Pump - 1 Required

Construction: Stainless steel. Design pressure - 150 psig. Design temperature - 150°F. Capacity - 53 gpm at 50 feet TDH.

Waste Sludge Discharge Mixing Pump - 1 Required

Construction: Stainless steel. Design pressure - 150 psig. Design temperature - 150°F. Capacity - 210 gpm at 105 feet TDH.

Spent Resin Tank - 1 Required

Construction: Stainless steel shell and internals. Atmospheric design pressure. 150°F design temperature. Capacity - 1200 gallons.

Spent Resin Pump - 1 Required

Construction: Steainless steel. Design pressure - 150 psig. Design temperature - 150°F. Capacity - 21 gpm at 105 feet TDH.

Decontamination Solution Concentrated Waste Tank - 2 Required

Construction: Steainless steel shell and internals. Atmospheric design pressure. 150°F design temperature. Capacity - 700 gallons/each.

Decontamination Solution Concentrate Waste Pump - 1 Required

Construction: Stainless steel. Design pressure - 150 psig. Design temperature - 150°F. Capacity - 30 gpm at 70 feet TDH.

Concentrated Waste Measuring Tank - 1 Required

Construction: Stainless steel. Atmospheric design pressure. 150°F design temperature. Capacity - 400 gallons.

TABLE 11.4-1 (Continued)

Centrifuge - 2 Required

Type - Solid bowl, horizontal, continuous feed. Removal efficiency of solids - 98%
Solids discharge - 40% to 60% by weight.

Waste Mixing Tank - 2 Required

Construction: Stainless steel. Capacity - 80 cubic feet. Equipped with mixer and spray header.

Waste Feed Pump - 2 Required

Construction: Stainless steel. Capacity - 10 to 18.7 gpm at 20 psig.

Dewatering Pump - 2 Required

Construction: Stainless steel. Capacity - 40 gpm at 15 psig.

Sample Pump - 2 Required

Construction: Stainless steel. Capacity - 7.5 gpm at 25 psig.

Sodium Silicate Storage Tank - 6 Required

Construction: Aluminum. Capacity - 550 gallons/each.

Sodium Silicate Day Tank - 1 Required

Construction: Carbon Steel. Capacity - 250 gallons.

Chemical Addition - 2 Required

Construction: Polypropylene. Capacity - 100 gallons/each.

Sodium Silicate Transfer Pump - 1 Required

Construction: Carbon Steel. Capacity - 8 gpm at 20 psig.

Sodium Silicate Feed Pump - 1 Required

Construction: Stainless steel. Capacity - 1 to 2 gpm at 20 psig.

TABLE 11.4-1 (Continued)

Chemical Addition Pump - 2 Required

Construction: Stainless steel. Capacity - 0.05 to 2.0 gpm at 50 psig.

Waste Mixing Pump - 1 Required

Construction: Stainless steel. Capacity - 30 gpm at 10 psig.

Bulk Cement Storage Silo - 1 Required

Construction: Carbon Steel. Capacity - 1000 cubic feet. Equipped with pneumatic transporter system to convey cement from storage silo to cement day tank.

Cement Day Tank - 1 Required

Construction: Carbon steel. Capacity - 50 cubic feet. Equipped with vibrating screw feeder to waste mixing pump.

Transfer Dolly - 1 Required

Track riding dolly for transfer to 50 cubic foot containers between processing stations.

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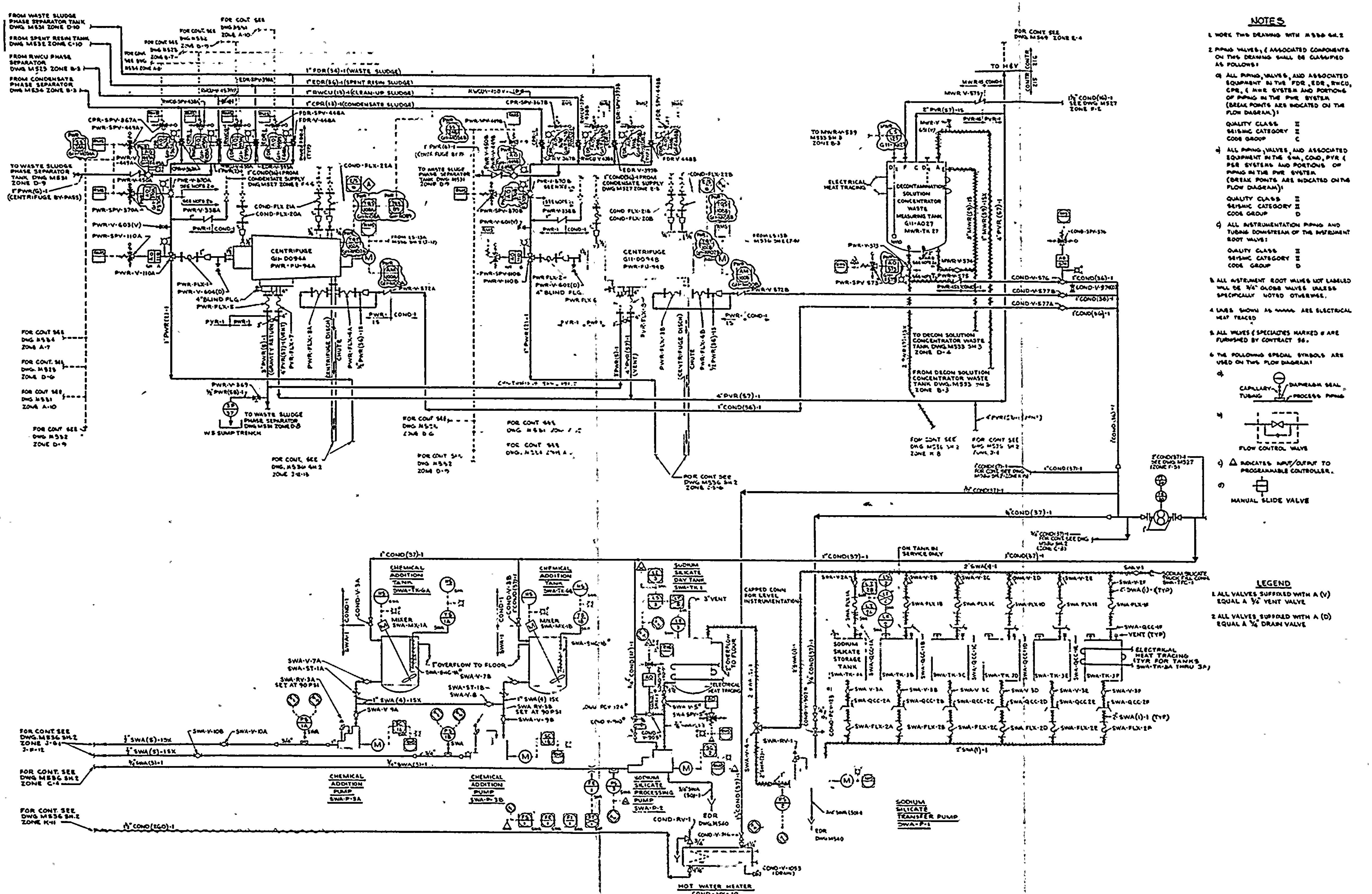
November 1982



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

FLOW DIAGRAM RADIOACTIVE WASTE DISPOSAL SOLIDS HANDLING SYSTEM

FIGURE 11.4-1a

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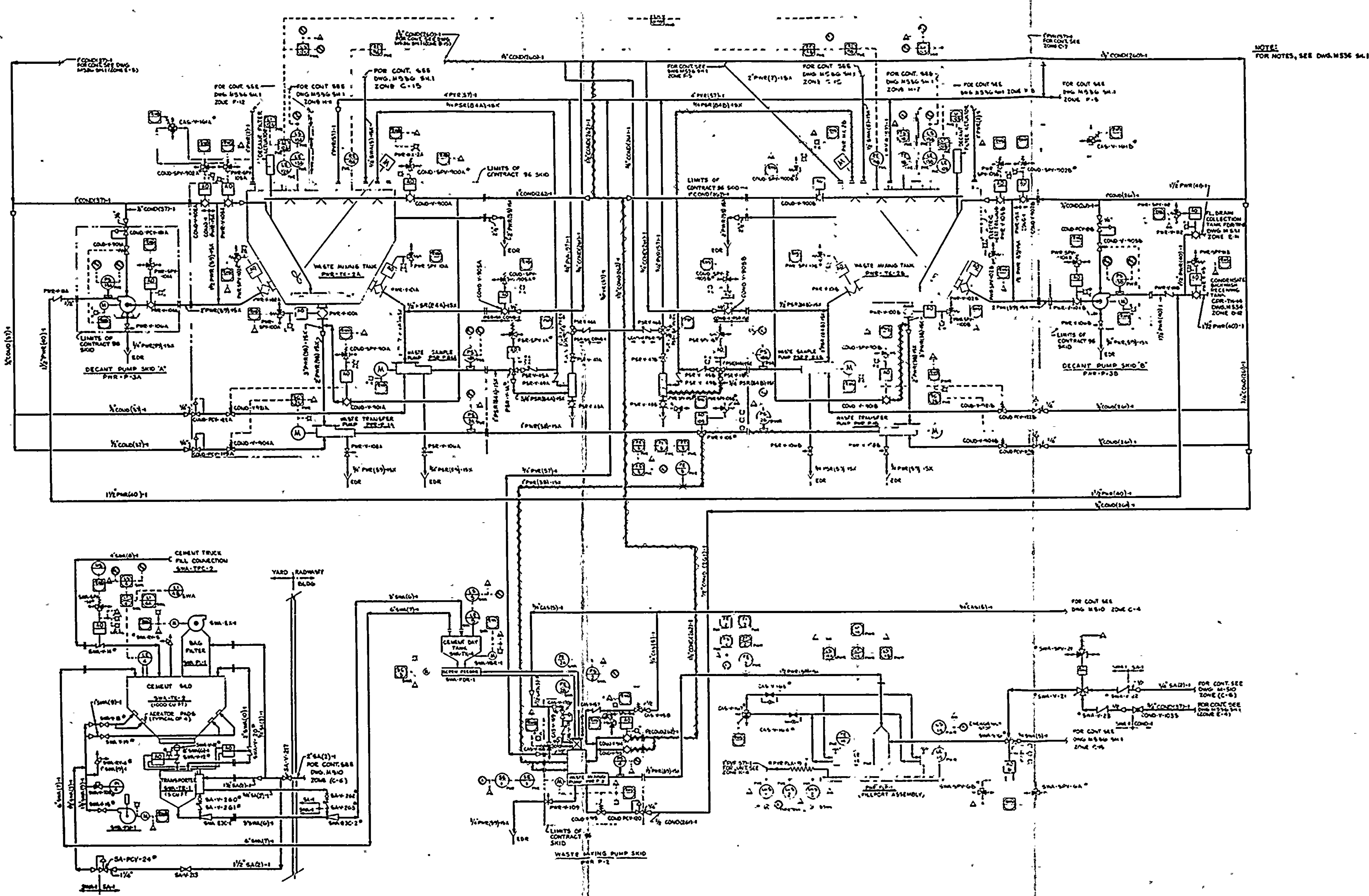
November 1982



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FLOW DIAGRAM RADIOACTIVE WASTE DISPOSAL SOLIDS HANDLING SYSTEM	FIGURE 11.4-1b

TABLE 11.5-1

PROCESS AND EFFLUENT RADIATION MONITORING SYSTEM (GASEOUS AND AIRBORNE MONITORS)

A. SAFETY-RELATED SYSTEMS

Monitor	Detector Location (No. of Channels)	Type	Sensitivity	Range (Scale)	Principle Radio-nuclides Measured	Expected Activity	Upscale Set Points Alarms	Upscale Set Points Trips
Main Steam Line Radiation Monitor	Adjacent to steam lines	γ-Ion Chamber	3×10^{-10} Amp/R/h	10^{0} -10^{6} mR/h (6 dec. log)	Coolant activation gases	Steam line activity defined in Table 11.1-4	Above full power back-ground, below trip	Tech. Spec.
Reactor Building Exhaust Plenum Radiation Monitor	In-line (4)	GM[b]		10^{-2} -10^{2} mR/hr (4 dec. log)		Reactor Bldg. activity defined in Table 11.3-7	Above back-ground, below trip	Tech. Spec.
Control Room Fresh Air Intake	Off-line (4)	β-Scint Part. Filter	10^{-5} μCi/cc	10^{1} -10^{5} cmp (5 dec. log)	Xe-133[a]	Within Monitor Range	Above back-ground, below low trip	Tech. Spec.

B. SYSTEMS REQUIRED FOR PLANT OPERATION

Monitor	Detector Location (No. of Channels)	Type	Sensitivity	Range (Scale)	Principle Radio-nuclides Measured	Expected Activity	Upscale Set Points Alarms	Upscale Set Points Trips
Off-Gas Pretreatment Radiation Monitor	Off-line, adjacent to sample chamber (1)	γ-Ion Chamber	3×10^{-10} Amp/R/h	10^{0} - 10^{6} mR/h (6 dec. log)	Noble gas Fission Products	Off-gas activity defined in Table 11.3-1	Above back-ground	Not Applica-ble
Off-Gas Post-Treatment Radia-tion Monitor	Off-line, (2)	GM Part. Filter Iodine Filter	250cpm/ pCi/cm^3	10^{1} - 10^{5} cpm (5 dec. log)	Kr-85[a]	Off-gas activity defined in Table 11.3-7	Above back-ground	Tech. Spec.
Charcoal Bed Vault Radiation Monitor	Charcoal bed vault (1)	GM		10^{0} - 10^{6} mR/h (6 dec. log)	Noble gas	Charcoal Bed Inven-tory defined in 11.3	Above back-ground	Not Applica-ble

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B. SYSTEMS REQUIRED FOR PLANT OPERATION (Continued)

Monitor	Detector Location (No. of Channels)	Type	Sensitivity	Range (Scale)	Principle Radio-nuclides Measured	Expected Activity	Upscale Set Points Alarms	Upscale Set Points Trips
Mechanical Vacuum Pump Discharge	In-line (1)	GM		10^{-2}- 10^{2} mR/hr (4 dec. log)	Xe-133	Within Monitor Range	Above back-ground	Tech. Spec.
Reactor Bldg. Elevated Discharge Radiation Monitor	Off-line (1)	GM Part. Filter Iodine Filter	50cmp/ pCi/cm^3	10^{1} -10^{5} cpm (5 dec. log)	Kr-85[a]	Reactor Bldg. activity defined in Table 11.3-7	Above back-ground	Tech. Spec.
		β -Scint		10^{-2} -10^{3} μ Ci/cc Xe -133 (5 dec. log)	Xe-133[a]	LOCA (DBA) activity as defined in Table 15.6-18		
Turbine Bldg. Ventilation Exhaust Radiation	Off-line (2)	β Scint Part. Filter Iodine Filter	50cmp/ pCi/cm^3	10^{1} -10^{5} cpm (5 dec. log)	Kr-85[a]	Turbine Bldg. activity defined in Table 11.3-7	Above back-ground	Tech. Spec.
		β Scint		10^{-2} -10^{3} μ Ci/cc Xe-133 (5 dec. log)	Xe-133[a]	LOCA mixture of F.P. activity see Table 15.6-18		
Radwaste Bldg. Ventilation Exhaust Radiation	Off-line (2)	β-Scint Part. Filter Iodine Filter	50 cmp/ pCi/cm^3	10^{1} -10^{5} cpm (5 dec. log)	Kr-85[a]	Radwaste Bldg. activity defined in Table 11.3-7	Above back-ground	Tech. Spec.
		β Scint		10^{-2} -10^{3} μ Ci/cc Xe-133 (5 dec. log)	Xe-133[a]	LOCA mixture of F.P. activity see Table 15.6-18		

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TABLE 12.3-1
AREA MONITORS

Station Number	Location And Title	Range
1	Reactor Building Fuel Pool Area	10^2-10^6mR/hr
2	Reactor Building Fuel Pool Area	1-10^4mR/hr
3	Reactor Building New Fuel Area	10^2-10^6mR/hr
4	Reactor Building Control Rod Hyd Equipment Area E	1-10^4mR/hr
5	Reactor Building Control Rod Hyd Equipment Area W	1-10^4mR/hr
6	Reactor Building S 589' Level	1-10^4mR/hr
7	Reactor Building Neutron Monitor System Drive Mech. Area	1-10^4mR/hr
8	Reactor Building STGS Filters Area	1-10^4mR/hr
9	Reactor Building Northwest RHR Pump Room	1-10^4mR/hr
10	Reactor Building Southwest RHR Pump Room	1-10^4mR/hr
11	Reactor Building Northeast RHR Pump Room	1-10^4mR/hr

TABLE 12.3-1 (Continued) Page 2 of 3

Station Number	Location And Title	Range
12	Reactor Building RCIC Pump Room	$1-10^4$mR/hr
13	Reactor Building HPCS Pump Room	$1-10^4$mR/hr
14	Turbine Building Turbine Front Standard	$1-10^4$mR/hr
15	Turbine Building Entrance	$1-10^4$mR/hr
16	Turbine Building Reactor Feed Pump Area 1A	$1-10^4$mR/hr
17	Turbine Building Reactor Feed Pump Area 1B	$1-10^4$mR/hr
18	Turbine Building Condensate Pump Area	$1-10^4$mR/hr
19	Main Control Room	$1-10^4$mR/hr
20	Radwaste Building Valve Room E	$1-10^4$mR/hr
21	Radwaste Building Valve Room W	$1-10^4$mR/hr
22	Radwaste Building Sample Area	$1-10^4$mR/hr
23	Reactor Building North CRD Pump Area	$1-10^4$mR/hr
24	Reactor Bldg. North 478' Level	$1-10^4$mR/hr

TABLE 12.3-1 (Continued) Page 3 of 3

Station Number	Location And Title	Range
25	Radwaste Bldg. Hot Machine Shop	$1-10^4$ mR/hr
26	Radwaste Bldg. Contaminated Tool Room	$1-10^4$ mR/hr
27	Radwaste Bldg. Waste Surge Tank Area	$1-10^4$ mR/hr
28	Radwaste Bldg. Tank Corridor Area North	$1-10^4$ mR/hr
29	Radwaste Bldg. Tank Corridor Area South	$1-10^4$ mR/hr
30	Radwaste Bldg. Radwaste Control Room	$1-10^4$ mR/hr
31	Reactor Bldg. New Fuel Area #2	10^2-10^6 mR/hr
		Range (6 decades)
32	Reactor Bldg. 471' Level N.E.	$10-10^7$ mR/hr
33	Reactor Bldg. 501' Level N.W.	$10-10^7$ mR/hr
34	Reactor Bldg. 606' Level East	$10-10^7$ mR/hr

Note: Alarm settings for all of the above monitors will be selected to provide indication of any abnormal increase in radiation levels while minimizing false alarms.

e. The Health Physics/Chemistry Manager's office is located in the service building. The Health Physics Supervisor and Health Physics/Chemistry Technicians are located in the radwaste and service buildings adjacent to the locker change rooms. These locations provide for ready access to the radiation protection staff by other plant workers and an area to generate and store all records.

f. A hot machine shop and a hot instrument shop are provided in the radwaste building for work on contaminated equipment under controlled conditions.

g. A laboratory complex is provided in the radwaste building consisting of a sample room, hot radiochemistry laboratory, and a counting room where radioactive samples will be qualitatively and/or quantitatively analyzed.

h. Facilities for calibration of all plant health physics instrumentation are provided at the: plant, emergency operations facility, central calibration facility, or a qualified contractor. Calibration equipment includes: pulse generators, electronic test equipment, and a source calibration unit described in 12.3.4.3 (with traceability to the National Bureau of Standards). Survey instruments are calibrated every six (6) months. Instruments used to monitor radiographic operations are calibrated quarterly. Periodic maintenance and checks are performed on all test equipment. Calibration records are maintained to ensure recalibration at specified intervals. Plant instrument maintenance facilities include both normal and hot shops. Portable instrument storage is located at selected points throughout the facility for their ready use. Storage areas include: two health physics areas, cabinets inside various buildings, the operations control room, the onsite technical support center, and the emergency relocation centers.

12.5.2.3 Equipment

Health physics equipment, other than instrumentation, is described below:

a. Protective clothing and accessories are provided for personnel required to work on contaminated areas. Clothing requirements for a particular task or area are prescribed by the radiation protection staff based upon the actual or potential conditions. Clothing available includes:

    (1) Coveralls and laboratory coats

    (2) Gloves - rubber and/or cotton

    (3) Head covers

    (4) Foot protection

    (5) Plastic suits - with or without supplied air.

b. Respiratory protection equipment is provided and required for personnel when levels of airborne radioactive materials approach or exceed applic able limits or when a potential for this con dition exists. The respiratory protection pro gram is conducted within the requirements of 10CFR20.103 and exposure is limited to average concentrations less than the values specified in Appendix B, Table 1, column 1, of 10CFR20. Allowance is made for use of respiratory protec tive equipment, as prescribed in 20.103, in determining an individual's inhalation of airborne radioactive materials. The following types of equipment are used:

It is intended that a minimum of five (5) additional cold licenses (RO) will be obtained. As far as possible, these five (5) licenses will be held within the initial complement of EOs as discussed in Section 13.2.1.6, Contingency Training.

It shall be WNP-2 policy to maintain an adequate number of personnel in the Shift Manager, Control Room Supervisor, Shift Support Supervisor, Shift Technical Advisor (if required), Control Room Operator, and Equipment Operator positions such that the use of overtime is not routinely required to compensate for inadequate staffing.

13.1.2.3.2 Shift Responsibility for Radiation Protection

Two Health Physics/Chemistry Technicians are assigned to each operating shift to provide radiological surveillance/control and chemistry/radiological chemistry services, and other related tasks as necessary during an assigned shift.

One of the technicians being of journeyman status (ANSI/ANS 3.1 (1978) qualified) having completed the necessary training as indicated by the Supply System "Health Physics/Chemistry Technical Training Manual".

Also, all shift personnel shall be instructed in the fundamentals of health physics such as; implementing radiation protection procedures, radiation and contamination surveys, use of protective barriers and signs, use of protective clothing and breathing apparatus, radiation monitoring, and accumulated dose.

Shift personnel are responsible for immediately informing the on duty Shift Manager if conditions develop that exceed or are likely to exceed preestablished radiation levels or exposure limits or if they believe that unsafe or hazardous conditions exist. The Shift Manager will evaluate the situation and if a radiological condition exists that warrants attention and investigation, the appropriate personnel in the Health Physics Department will be called for assistance.

13.1.2.3.3 Shift Maintenance Support

Craftsmen and technicians, as required, will be assigned to each operating shift for the purpose of providing maintenance support and surveillance testing in the areas of instrumentation and controls, and mechanical and electrical equipment.

13.1.2.3.4 Shift Fire Brigade

A Shift Fire Brigade, consisting of a minimum of five (5) members of the nominal shift complement, shall have advanced fire training and be equipped for fire fighting. This select group on each operating shift will have primary response capabilities and will respond to emergencies involving fire and/or emergencies where life threatening danger exists. Brigade certifications for all members is accomplished upon satisfactory completion of a twenty-four and one-half (24.5) hour training program.

The Fire Brigade shall consist of the following personnel:

a. Shift Support Supervisor - Brigade Leader
b. Plant Equipment Operators (2)
c. Shift Health Physics/Chemistry Technician (1)
d. Maintenance Support Personnel (1)

The Fire Brigade leader is dedicated to stay with the brigade during a fire emergency.

13.1.3 QUALIFICATIONS OF NUCLEAR PLANT PERSONNEL

The minimum educational and experience qualifications for the onsite plant personnel are detailed in 13.1.3.1. These minimum requirements are based on Regulatory Guide 1.8, Revision 1 (1977), "Personnel Selection and Training". If an individual is placed in a discipline who does not meet the minimum qualification criteria, it will be specifically pointed out and justification or explanation provided. Our personnel qualification and training programs are under continual review and modification to reflect the changes following TMI. The minimum qualification requirements identified in 13.1.3.1 will be revised accordingly.

13.1.3.1 Minimum Qualification Requirements

13.1.3.1.1 Plant Management

<u>Plant Manager/Assistant Plant Manager</u>

At the time of initial core loading or appointment to the active position, the Plant Manager/Assistant Plant Manager shall have ten (10) years of responsible power plant experience of which a minimum of three (3) years shall be nuclear power plant experience. A maximum of four (4) years of the remaining seven (7) years of experience may be fulfilled by

14.2.12.3.16.4 Criteria

Level 1

a. The reactor recirculation pumps shall not be started nor flow increased unless the coolant temperatures between the steam dome and bottom head drain are within 100°F (56°C).

b. The recirculation pump in an idle loop must not be started unless the loop suction temperature is within 50°F (28°C) of the steam dome temperature.

Level 2

During two pump operation at rated core flow, the bottom head temperature as measured by the bottom drain line thermocouple should be within 30°F (17°C) of the recirculation loop temperatures.

14.2.12.3.17 Test Number 17 - System Expansion

14.2.12.3.17.1 Purpose

The purpose of this test is to verify that piping systems are free and unrestrained in regard to thermal expansion and that suspension components are functioning in the specified manner. This test also verifies that all accessible snubbers installed on safety-related systems whose normal operating temperature is greater than 250°F have adequate swing clearance to accommodate system thermal expansion.

The test also provides data for calculation of stress levels in nozzles and weldments and confirmation that hot pipe containment penetrations are adequately designed.

14.2.12.3.17.2 Prerequisites

Necessary Preoperational Tests have been completed. The pre-heatup examination program relating to component supports as contained in the WNP-2 Preservice Inspection Program Plan has been completed. The POC has reviewed, and the Plant Manager has approved the test procedure and initiation of testing. Instrumentation has been installed and calibrated.

14.2.12.3.17.3 Description

During the Power Ascension Testing (PAT) Program, hot condition piping support performance and settings will be verified in accordance with System Test Instruction (STI), "Piping System Expansion and Vibration Tests." Hanger positions of major equipment and piping in the nuclear steam supply system and auxiliary systems will be prerecorded during the initial thermal cycle and after shakedown has taken place (normally about three cycles). At specified temperature intervals during initial system heatup and following cooldown, visual inspections are made by personnel certified to Level II or Level III for VT-3 and VT-4 to assure components are free to move as designed. Adjustments are made as necessary. Snubber thermal movement and swing clearance is verified during each inspection. For systems inspected that do not attain operating temperature, swing clearance for projected snubber movement will be verified by observation and/or calculation.

Devices for measuring continuous pipe deflections are mounted on main steam, recirculation, feedwater, RCIC, and selected safety/relief valve discharge lines. Motion measured during heatup is compared with calculated values.

14.2.12.3.37 Test Number 71 - Residual Heat Removal System

14.2.12.3.37.1 Purpose

The purpose of this test is to demonstrate the ability of the Residual Heat Removal (RHR) System to: 1) remove heat from the reactor system so that the refueling and nuclear system servicing can be performed and 2) condense steam while the reactor is isolated from the main condenser.

14.2.12.3.37.2 Prerequisites

The Preoperational Tests have been completed, the POC has reviewed and the Plant Manager has approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

14.2.12.3.37.3 Description

With the reactor at power, the condensing mode of the RHR system will be demonstrated. Condensing heat exchanger performance characteristics will be demonstrated. Final demonstration of the condensing mode will be done from an isolated condition. This test will optimize the controls for this mode of operation.

During the first suitable reactor cooldown, the shutdown cooling mode of the RHR system will be demonstrated. Unfortunately, the decay heat load is insignificant during the startup test period. Use of this mode with low core exposure could result in exceeding the 100°F/hr cooldown rate of the vessel if both RHR heat exchangers are used simultaneously, therefore, the demonstration is limited by the cooldown rate.

14.2.12.3.37.4 Criteria

Level 1

The transient response of any system-related variable to any test input must not diverge.

Level 2

The RHR system shall be capable of operating in the steam condensing, suppression pool cooling and shutdown cooling modes (with both one and two heat exchangers). System-related variables may contain oscillatory modes of response. In these cases, the decay ratio for each controlled mode of response must be less than or equal to 0.25.

The time to place the RHR heat exchangers in the steam condensing mode and commence operation shall be one-half hour or less.

safe operation and the management function of the shift supervisor is to provide for assuring safety.

d. The administrative duties of the shift supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room.

Clarification

The table attached provides clarification to the above position.

WNP-2 Position

The administrative duties of the shift manager will be reviewed; inappropriate functions will be delegated to other personnel including the shift support supervisor. The shift support supervisor will assist the shift manager by directing personnel assigned to perform balance-of-plant operating functions and by performing shift administrative duties.

WNP-2 procedures will be reviewed to ensure that the shift manager, control room supervisor, shift support supervisor, and operator functions are defined adequately to establish the shift manager as the commanding authority for plant operations relative to other plant management. The shift manager is to ensure the safe operation of the plant under all conditions. During an emergency, the responsibility for directing and controlling the actions of the operating crew to place and maintain the plant in a safe condition rests with the shift manager. During accident conditions, the shift manager will be in the control room at all times until properly relieved. He may after an evaluation, decide to relocate to the immediate site of the emergency.

This principle will be reinforced by a management directive issued from the office of the Director of Generation that emphasizes that the shift manager's primary responsibility is the safe operation of the plant under all conditions.

The shift manager's administrative duties will be reviewed annually by the operations manager to ensure that administrative responsibilities do not interfere with the primary responsibility.

Appropriate documentation will be available onsite for review by NRC I&E Branch.

safe operation and the management function the shift supervisor is to provide for assuring safety.

d. The administrative duties of the shift supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room.

Clarification

The attachment provides clarification to the above position.

WNP-2 Position

The administrative duties of the shift supervisor have been reviewed. Inappropriate functions were delegated to other personnel.

WNP-2 procedures were reviewed to ensure that the shift manager, control room supervisor, and operator funcions are defined adequately to establish the shift manager in the control room as the commanding authority for plant operations relative to other plant management.

This principle is reinforced by a management directive issued annually from the office of the Director of Generation that emphasies that the shift manager's primary responsibility is the safe operation of the plant under all conditions.

The shift manager's administrative duties are reviewed annually by the plant manager to ensure that administrative responsibilities do not interfere with the primary responsibility.

Appropriate documentation is available onsite for review by NRC I&E Branch.

TABLE I.C.3-1

SHIFT SUPERVISOR RESPONSIBILITY (2.2.1.A)

NUREG-0578 POSITION (POSITION NO.)	CLARIFICATION
Highest Level of Corporate Management (1.)	V. P. For Operations
Periodically Reissue (1.)	Annual Reinforcement of Company Policy
Management Direction (1.)	Formal Documentation of Shift Personnel, All Plant Management, Copy to IE Region
Properly Defined (2.0)	Defined in Writing in a Plant Procedure
Until Properly Relieved (2.8)	Formal Transfer of Authority, Valid SRO License, Recorded in Plant Log
Temporarily Absent (2.C)	Any Absence
Control Room Defined (2.C)	Includes Shift Supervisor Office Adjacent to the Control Room
Designated (2.C)	In Administrative Procedures
Clearly Specified	Defined in Administrative Procedures
SRO Training	Specified in ANS 3.1 (Draft) Section 5.2.1.8
Administrative Duties (4.)	Not Affecting Plant Safety
Administrative Duties Reviewed (4.)	On Same Interval as Reinforcement: i.e., Annual by V. P. for Operations

This requirement shall be met prior to July 1982. See NUREG-0578, Section 22.1a, Item 4 and NRC letters of September 27 and November 9, 1979.

equipment for maintenance or surveillance testing or return-to-service may be delegated to an on-shift SRO, provided provisions are made to ensure that the shift supervisor is kept fully informed of system status.

c. Work permits involving tagging for maintenance or surveillance testing are verified by the Shift Manager (or his designee) for correct implementation of control measures. Independent verification by qualified individuals is made for installation or removal of temporary modifications such as jumpers, lifted leads or bypass lines. Routine independent verification of equipment status at the location of the equipment will be performed for return-to-service activities of all important safety-related equipment having no control room status indications. These verifications will be by qualified equipment operators.

d. Equipment control procedures should include assurance that control room operators are informed of changes in equipment status and the effects of such changes.

e. For the return-to-service of equipment important to safety, a second qualified operator should verify proper systems alignment unless functional testing can be performed without compromising plant safety, and all equipment, valves, and switches involved in the activity are correctly aligned.

NOTE: A licensed operator possessing knowledge of the systems involved and the relationship of the systems to plant safety would be a "qualified" person. The staff is investigating the level of qualification necessary for other operators to perform these functions.

For plants that have or will have automatic system status monitoring as discussed in Task Action Plan item I.D.3, NUREG-0660, the extent of human verification of operations and maintenance activities will be reduced. However, the need for such verification will not be eliminated in all instances.

WNP-2 Position

WNP-2 will prepare or revise procedures as necessary to implement an effective system for verification of operating

activities important to safety. These procedures will be implemented prior to fuel load. The preparation of these procedures will be guided by ANS 3.2 Section 5.2.6 and the following supplemental provisions:

a. ANS 3.2 Section 5.2.6 will be applied to both maintenance and surveillance.

b. The shift manager will have the designated responsibility for implementing procedures for release of systems and equipment for maintenance or surveillance testing and for return-to-service. This responsibility may be delegated to any other licensed senior reactor operator (SRO) stationed within the control room. The shift manager will remain informed by reviewing records and receiving turnover.

c. Work permits involving tagging for maintenance or surveillance testing are verified by the shift manager (or his designee) for correct implementation of control measures. Independent verification is also made for installation or removal of temporary modifications such as jumpers, lifted leads, or bypass lines. Routine independent verification of equipment status at the location of the equipment will be performed for return-to-service activities of all important safety-related equipment having no control room status indications.

d. Equipment control procedures are implemented through the control room such that control room personnel are aware of changes being made in equipment status and the effects of such changes.

e. Activities for the return-to-service of equipment important to safety are verified by the shift manager (or his designee) for correct implementation of control measures. Independent verification is also made for installation or removal of temporary modifications such as jumpers, lifted leads, or bypass lines. Routine independent verification of status at the location of the equipment is limited to return-to-service activities performed prior to startups following refueling or long-term outages in accordance with ALARA concept to limit accumulation of personnel radiation exposures.

I.D.2 PLANT SAFETY PARAMETER DISPLAY CONSOLE

Position

In accordance with Task Action Plan I.D.2, Plant Safety Parameter Display Console (NUREG-0660), each applicant and licensee shall install a safety parameter display system (SPDS) that will display to operating personnel a minimum set of parameters which define the safety status of the plant. This can be attained through continuous indication of direct and derived variables as necessary to assess plant safety status (NUREG-0737).

Clarification

These requirements for the SPDS are being developed in NUREG-0696, which is scheduled for issuance in November 1980.

WNP-2 Position

FSAR 7.5.1.5, 7.5.1.6, and 7.7.1.15 describe the SPDS and supporting technical data aquisition system to be implemented on WNP-2 in response to this issue. WNP-2 is working with the BWR Owners' Group to develop the emergency response information system (ERIS) as the BWR utility position responding to the concerns of Item I.D.2. The combination of these descriptions and the implementation of the ERIS concept adequately answers the concerns of Item I.D.2.

systems described in NUREG-0696. The Emergency Response Information System will be based on the needs of the personnel using it, as determined by the emergency procedures, the design of BWR systems, and human factors analysis.

A complete design description of the WNP-2 system will be available for NRC review prior to April 1982.

a) Documentation of Plant Safety

b) Feedback/Confirmation of Anticipated Results

B. Test Phase Instruction Performed by Test Director on a Shift Basis (during testing)

1) Review of the Immediate Test Schedule

2) Discussion of the Impending Tests: Procedures, Anticipated results, Precautions

3) Review/Disseminate Plant Response Data from Previous Shift(s)

C. Post-STP Completion Instruction Performed by Test Director (following testing)

1) Review of the Actual STP Results vs. Anticipated Results

2) Review Plant Design Changes/System Modifications Required

II. Development and Performance of a Special Test Subprogram

A. Additional RCIC System Tests

1) RCIC Operation Following Loss of AC Power to the System

2) RCIC Operation to Prove DC Separation

B. Integrated Reactor Vessel Level Instrumentation Functional Test

C. Integrated Containment Pressure Instrumentation Functional Test

D. Simulated Loss of Control and Instrument Air Test

E. Repetition of Some Normal STP Tests, for example:

1) Feedwater Pump Trip/Recirc Runback Demonstration

2) Turbine Trip/Generator Load Rejection Within Bypass Valve Capacity

3) Pressure Regulator Setpoint Changes

4) Recirculation Pump Trips

5) RHR Steam Condensing Mode Operation

6) Feedwater Level Setpoint Changes

III. Utilization of the STP Data

A. Refine the WNP-2 Simulator Response Models, as appropriate

B. Incorporate a Major Plant Transient Response Section in Operator Training Program, as appropriate

C. Update License Program Training and Requalification Material, as appropriate.

It is anticipated that every participating member of the operations staff will obtain valuable knowledge and experience through participation in the WNP-2 Startup Test Program. Each will receive appropriate classroom instruction and through judicious scheduling of tests, most will be exposed to a variety of plant/system transient responses (or review of results thereof). The training received will be continually re-enforced through normal requalification program refinements. Future license candidates will also benefit from the training material upgrades resulting from the STP experience.

With this program outline, the Supply System is meeting the intent of NUREG-0660, Item I.G.1. Specific details of the training program, additional test procedures, and documentation methods have been developed and were made available for onsite NRC I&E review.

Additionally, the Supply System will review the results of the simulated loss of all AC power test to be performed at the LaSalle and Grand Gulf Power Stations, subject to satisfactory safety evaluations. The results and merits of performing these tests will be reviewed, an analysis performed, and recommendations forwarded to the NRC as to whether or not the test, or some portion of it, should be repeated at WNP-2.

Based on the Supply System recommendations and the benefits realized on other BWR plants conducting the subject tests, WNP-2 and the NRC will determine the scope of those portions of the tests requiring performance at WNP-2.

d. Instrumentation ranges shall overlap to cover the entire range of effluents from normal (ALARA) through accident conditions.

The design description shall include the following information:

1. System description, including:

   (i) instrumentation to be used, including range or sensitivity, energy dependence or response, calibration frequency and technique, and vendor's model number, if applicable;

   (ii) monitoring locations (or points of sampling), including description of methods used to assure representative measurements and background correction;

   (iii) location of instrument readout(s) and method of recording, including description of the method or procedure for transmitting or disseminating the information or data;

   (iv) assurance of the capability to obtain readings at least every 15 minutes during and following an accident; and,

   (v) the source of power to be used.

2. Description of procedures or calculational methods to be used for converting instrument readings or release rates per unit time, based on exhaust air flow and considering radionuclide spectrum distribution as a function of time after shutdown.

WNP-2 Position

WNP-2 is in the process of having extended range noble gas effluent monitors installed. Instrument ranges will be 10^{-6} to 10^{3} μCi/cc for the radwaste and turbine generator building and 10^{-6} to 10^{4} μCi/cc for the reactor building (Xe-133 equivalent). The power supplies will be from uninterruptible power. The instruments will be environmentally qualified for operation in the post-accident environment.

Each elevated release duct, turbine building exhaust and radwaste building exhaust is monitored for radioactive effluent gases by off-line systems. In addition, the elevated release duct has an in-line monitoring system.

The off-line systems draw samples from each exhaust duct through isokenetic probes. The sample passes through particulate and charcoal filters, a sample flow control system, and a radioactive gas monitor and is returned to the original exhaust duct. The sample flow rate is automatically adjusted to compensate for effluent flow changes. The system is equipped with local flow rate indication and remote flow rate trouble alarms to the control room. Each monitoring system has two separate detectors and instrument loops. The ranges of these detectors are:

a. low range (10^{-16} µCi/cc to 10^{-1} µCi/cc Xe-133)

b. medium range (10^{-2} µCi/cc to 10^{3} µCi/cc Xe-133)

The detectors are set in lead shielding to reduce the unwanted background. The low-range detector is a beta-sensitive scintillator. The medium-range detector is a G. M. tube which measures the gamma radiation from the sample chamber.

The in-line monitor provides high-range detection of 6 decades up to 10^{5} Ci/cc Xe-133. This is an ion chambr set into the elevated release duct. Each instrument loop contains a detector, a power supply, a ratemeter, alarm modules, and a recorder. The low-range channel is equipped with a radioactive test source while the medium-range channel has a built-in electronic test circuit.

accept flow control devices which have the capability of maintaining isokinetic conditions with variations in stack or duct design flow velocity of + 20%. Further departure from the isokinetic condition need not be considered in design. Corrections for non-isokinetic sampling conditions, as provided in Appendix C of ANSI 13.1-1969, may be considered on an ad hoc basis.

d. Effluent streams which may contain air with entrained water, e.g., air ejector discharge, shall have provisions to ensure that the adsorber is not degraded while providing a representative sample, e.g., heaters.

WNP-2 Position

Three low flow particulate/iodine sampling systems are being installed to monitor the post-accident exhausts from the reactor building, radwaste building, and turbine building. Each sample system consists of a shielded filter holder, a running time meter and a low volume positive displacement sample pump which draws a sample from the effluent duct. The sample is drawn through the particulate and iodine filter assembly at a rate of about 0.1 cfm and then returned to the effluent duct. The pump is automatically started when the high-high level alarm on the associated noble gas monitor is activated. The pump will continue to run until manually reset. The sample filter holder has a quick-release allowing it to be removed and handled with remote tools reducing any potential personnel exposure.

To protect personnel, a shadow shield of lead is positioned in front of the filter holder. This will reduce the dose rate from the filter following a reactor accident condition if there is a high level release of particulates and iodines.

TABLE II.F.1-2

SAMPLING AND ANALYSIS OR MEASUREMENT OF HIGH-RANGE RADIOIODINE PARTICULATE EFFLUENTS IN GASEOUS EFFLUENT STREAMS

EQUIPMENT - Capability to collect and analyze or measure representative samples of radioactive iodines and particulates in plant gaseous effluents during and following an accident. The capability to sample and analyze for radioiodine and particulate effluents is not required for PWR secondary main steam safety valve and dump valve discharge lines.

PURPOSE - To determine quantitative release of radioiodines and particulates for dose calculation and assessment.

DESIGN BASIS SHIELDING ENVELOPE - 10^2 μCi/cc of gaseous radioiodine and particulates, deposited on sampling media; 30 minutes sampling time, average gamma energy (E) of 0.5 MeV.

SAMPLING MEDIA

- Iodine > 90% effective adsorption for all forms of gaseous iodine

- Particulats > 90% effective retention for 0.3 micron (μ) diameter particles.

SAMPLING CONSIDERATIONS

- Representative sampling per ANSI N13.1-1969.

- Entrained moisture in effluent stream should not degrade adsorber.

- Continuous collection required whenever exhaust flow occurs.

- Provisions for limiting occupational dose to personnel incorporated in sampling systems, in sample handling and transport, and in analysis of samples.

ANALYSIS

- Design of analytical facilities and preparation of analytical procedurs shall consider the design basis sample.

f. HPCS service water pumps starts.

g. HPCS room cooler fan starts.

The operator can manually initiate the HPCS and RCIC Systems from the control room before the level 2 automatic initiation level is reached. The operator has the option of manual control after automatic initiation. The operator can verify that these systems are delivering water to the reactor vessel by:

a. Verifying reactor water level increases when systems initiate.

b. Verifying systems flow using flow indicators in the control room.

c. Verifying system flow is to the reactor by checking control room position indication of motor-operated valves. This assures no diversion of system flow to other than the reactor.

Therefore, the HPCS and RCIC can maintain reactor water level at full reactor pressure and until pressure decreases to where low pressure systems such as the Low Pressure Core Spray (LPCS) or Low Pressure Coolant Injection (LPCI) can maintain water level.

Containment Cooling:

After reactor scram and isolation and establishment of satisfactory core cooling, the operator would start containment cooling. This mode of operation removes heat resulting from safety/relief valve (SRV) discharge to the suppression pool. This would be accomplished by placing the Residual Heat Removal (RHR) System in the containment/suppression pool cooling mode, or the suppression pool spray mode, i.e., RHR suction from and discharge to the suppression pool. A summary of the operator actions is given below:

a. Start the associated RHR standby service water (SW) pump, if not already running.

b. Open the SW pump discharge valve, if not already open.

c. Open the SW loop return valve, if not already open.

d. Start the associated RHR pump.

e. Close the associated RHR heat exchanger bypass valve.

f. Adjust system flow by adjusting the RHR test return valve if in the suppression pool cooling mode.

g. Open the suppression pool spray valve if in the spray mode.

The Operator could verify proper operation of the RHR system containment cooling function from the control room by:

a. Verifying RHR and Service Water (SW) system flow usinng system control room flow indicators.

b. Verifying correct RHR and SW system flow paths using control room position indication of motor-operated valves.

c. On branch lines that could divert flow from the required flow paths, closing the motor-operated valves and noting the effect on RHR and SW flow rate.

Steam Condensing Mode:

The RHR system is started in steam condensing mode immediately after the isolation of the primary system from the main condenser. The RHR heat exchangers are used to condensate reactor steam, thereby transferring reactor decay heat to the service water. The operator actions necessary to place the RHR system in the steam condensing mode are as follows:

a. Close the associated RHR heat exchanger inlet and outlet valves.

b. Start the associatd RHR SW pump and establish SW flow through the associated RHR heat exchanger.

c. Open the associated RHR heat exchanger condensate flow valve to the suppression pool.

d. Verify the associated RHR heat exchanger level controller is set at the required value and in automatic.

e. Open the associated RHR heat echanger vent valves.

f. Verify the associated RHR heat exchanger pressure controller is set to maintain the air-operated steam pressure reducing valve closed.

g. Slowly open the associated steam supply valve ahead of the steam pressure control valve.

h. Switch mode selector switch to the "ON" position. This energizes the associated solenoids for the steam pressure reduction valve and the condensate discharge valve.

i. Gradually open the associated air-operated steam line pressure reducing valve and increase pressure to the required value, using the pressure controller manual control.

j. Verify that associated RHR heat exchanger pressure and both the inlet and outlet temperature are increasing.

k. When the associated RHR heat exchanger pressure reaches the required value switch the RHR heat exchanger pressure controller to automatic.

l. As steam pressure increases, slowly adjust the liquid level in the heat exchanger by regulating the level controller until the optimum differential temperature for the SW flow to the heat exchanger is obtained.

m. Verify the RCIC turbine is operating.

n. Verify that the RHR heat exchanger outlet temperature and condensate water quality have reached their acceptable limits.

o. Open the associated RHR heat exchanger condensate flow valve to RCIC.

p. Close associated RHR heat exchanger condensate flow valve to the suppression pool.

q. Monitor RHR heat exchanger level and pressure for stable operation.

The RHR steam condensing mode is now in service transferring reactor vessel heat to the atmosphere via the standby service water system.

Extended Core Cooling:

When the reactor has been depressurized, the RHR system can be placed in the long-term shutdown cooling mode. The operator manually terminates the containment cooling mode of one of the RHR loops and places the loop in the shutdown cooling mode as follows:

a. Trip the RHR pump to be used for shutdown cooling,

b. Close associated motor-operated valve in the suppression pool suction and LPCI discharge line to the vessel,

c. Open shutdown cooling suction valves from and discharge valves to the reactor vessel, and

d. Restart the RHR Pump.

In this operating mode, the RHR system can cool the reactor to cold shutdown. Proper operation and flow paths in this mode can be verified by methods similar to those described for the containment cooling mode.

In conclusion, the WNP-2 plant design is fully adequate to meet the intent of the requirements of auxiliary heat removal when the main feedwater system is inoperable.

II.K.3.13 Separation of HPCI and RCIC System Initiation Levels

a. Analysis
b. Modify

Position

Currently, the reactor core isolation cooling (RCIC) system and the high pressure coolant injection (HPCI) system both initiate on the same low water level signal and both isolate on the same high water level signal. The HPCI system will restart on low water level but the RCIC system will not. The RCIC system is a low-flow system when compared to the HPCI system. The initiation levels of the HPCI and RCIC systems should be separated so that the RCIC system initiates at a higher water level than the HPCI system. Further, the RCIC system, initiation logic should be modified so that the RCIC system will restart on low water level. These changes have the potential to reduce the number of challenges to the HPCI system and could result in less stress on the vessel from cold water injection. Analyses should be performed to evaluate these changes. The analyses should be submitted to the NRC staff and changes should be implemented if justified by the analyses (NUREG-0737).

a. Documentation provided results of evaluation and proposed modifications (if necessary) to staff by October 1, 1980. Provide sufficient supporting analysis to demonstrate that the systems, as modified, would not degrade proper system functions.

b. Modifications shall be completed (if necessary) by April 1, 1981.

See letter September 5, 1980, Enclosure 2, pg. 7 (Reference 33).

Clarification

None

WNP-2 Position

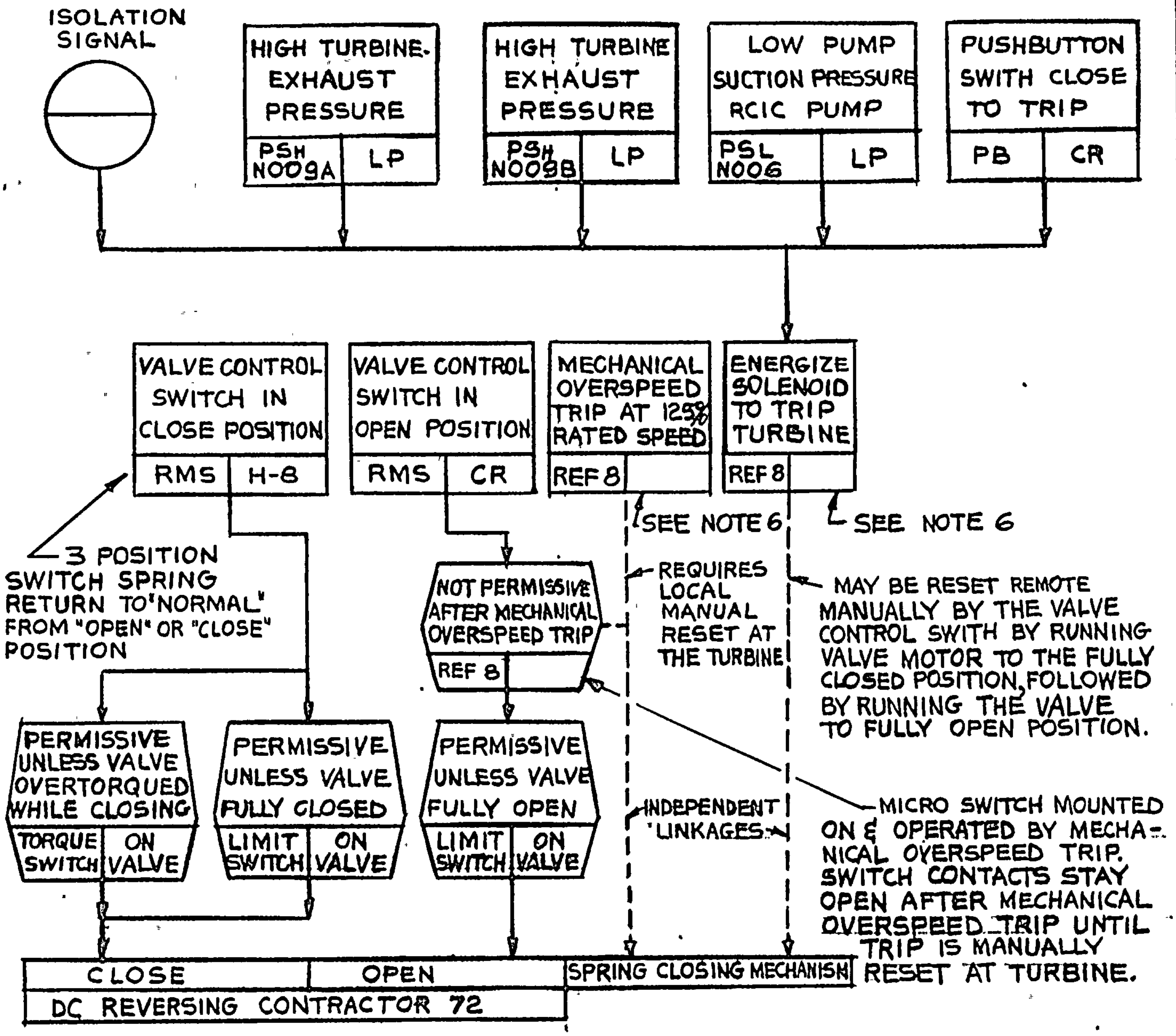
At WNP-2 the HPCS and RCIC are both initiated at low-water level 2 (489.5 inches above vessel zero).

As a generic item, the possible separation of initiation levels for RCIC and HPCS was studied by GE for the BWR Owners' Group. The results of that study were provided to the

Commission in a GE letter of October 1, 1980 from R. A. Buchholz to D. G. Eisenhut. The conclusions of that study are endorsed by WNP-2, specifically, that the proposed separation of RCIC and HPCS initiation is unnecessary. The basis is that for rapid level changes associated with accident scenarios and severe transients their initiation would be essentially simultaneous in that possible separation distances could not preclude HPCS challenges; likewise, for slow level changes due to small leaks or slow transients, adequate time exists for manual initiation of RCIC by the reactor operator, prior to HPCS auto-initiation. Justification of this basis is that over the lifetime of a unit, the expected occurrence of slow level decreases does not warrant installation of equipment changes to decrease the number of challenges made to the HPCS. Manual operator response to maintain water level is consistent with abnormal operations demands.

GE and the BWR Owners' Group have submitted an analysis of the RCIC system for automatic reset following a high water level trip. The Owners' Group has recommended automatic closure of the steam supply valve on high vessel water level with the RCIC turbine trip valve remaining open. This leaves the system in a standby mode capable of restarting at low water level. Very little modification of existing circuitry is required to effect this change. WNP-2 endorses the Owners' Group position and will make the necessary equipment modifications.

GE is designing and analyzing the RCIC system automatic reset for WNP-2. GE has indicated that the design will be similar to the logic modifications recommended by the BWR Owners' Group per letter from D. B. Waters (BWR Owners' Group) to D. Eisenhut (NRC), dated December 27, 1980. The logic changes for a typical BWR RCIC system are shown on Figures II.K.3.13-1, II.K.3.13-2, and II.K.3.13-3. The RCIC logic diagrams and Figures 7.4-2d and 7.4-2e will be updated when the design is finalized.



MOTOR OPERATED TURBINE TRIP & THROTTLE VALVE
FURNISHED AS PART OF TURBINE ASSEMBLY (REF 8)

SPECIAL NOTE:

VALVE MAY OPENED OR CLOSED OR POSITIONED AT ANY DESIRED POSITION OF TRAVEL (FOR THROTTLING) BY MEANS OF THE VALVE CONTROL SWITCH, AS LONG AS TRIPS ARE RESET. WHEN A TRIP OCCURS (MECHANICAL OR ELECTRICAL) THE SPRING ACTS TO CLOSE IRRESPECTIVE OF THE VALVE CONTROL SWITCH POSITION.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	TURBINE TRIP VALVE EXISTING HIGH WATER LEVEL TRIP	FIGURE II.K.3.13-1

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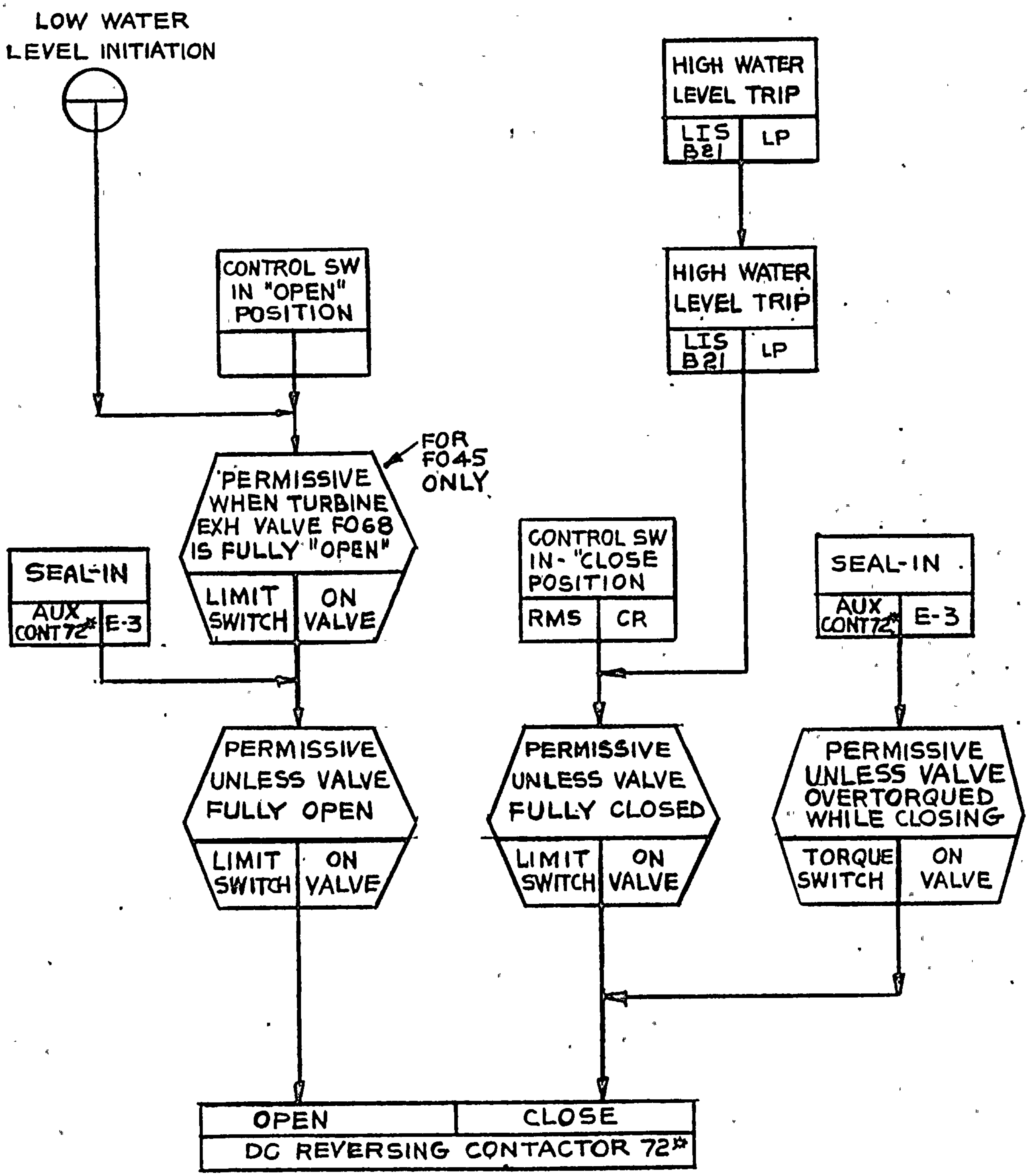
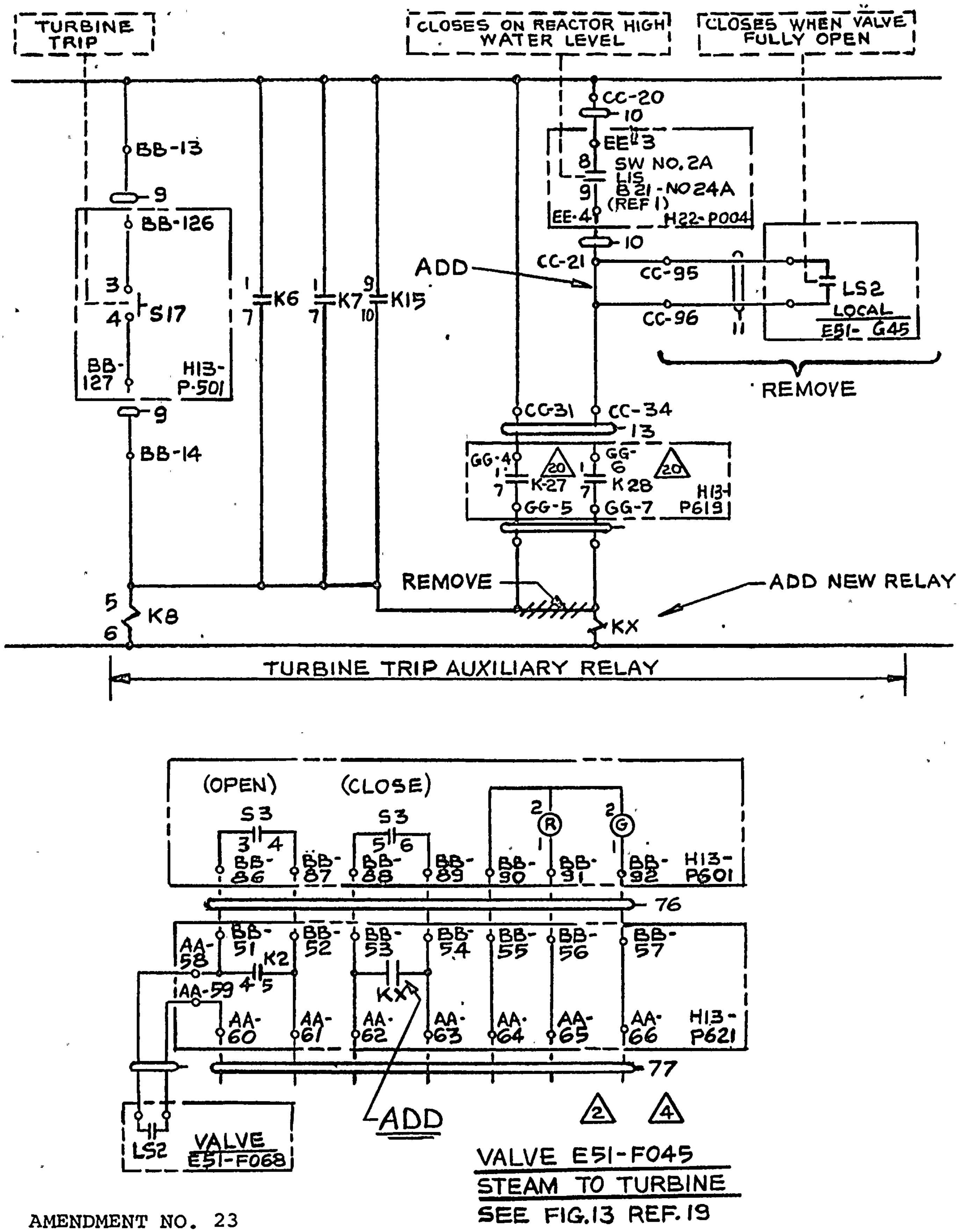
February 1982



STEAM SUPPLY VALVE TURBUNE MO-F045

(TYPICAL FOR LUBE OIL COOLING WATER SUPPLY VALVE MO-F046)

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	STEAM SUPPLY VALVE ADDITION OF HIGH WATER LEVEL TRIP	FIGURE II.K. 3.13-2



AMENDMENT NO. 23
February 1982

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CHANGES PER THE RCIC ELEMENTARY DIAGRAM	FIGURE II.K. 3.13-3

Regulatory Guide 1.101, Rev. 1, March 1977

Emergency Planning for Nuclear Power Plants

Compliance or Alternate Approach Statement:

WNP-2 complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Statement:

This guide has been implemented by the WPPSS Hanford Site "Emergency Plan" (see 13.3). Procedures which are required to implement the Emergency Plan will be available for onsite inspection prior to fuel load.

Specific Evaluation Reference:

Refer to 13.3.

Regulatory Guide 1.102, Rev. 1, September 1976

Flood Protection for Nuclear Power Plants

Compliance or Alternate Approach Statement

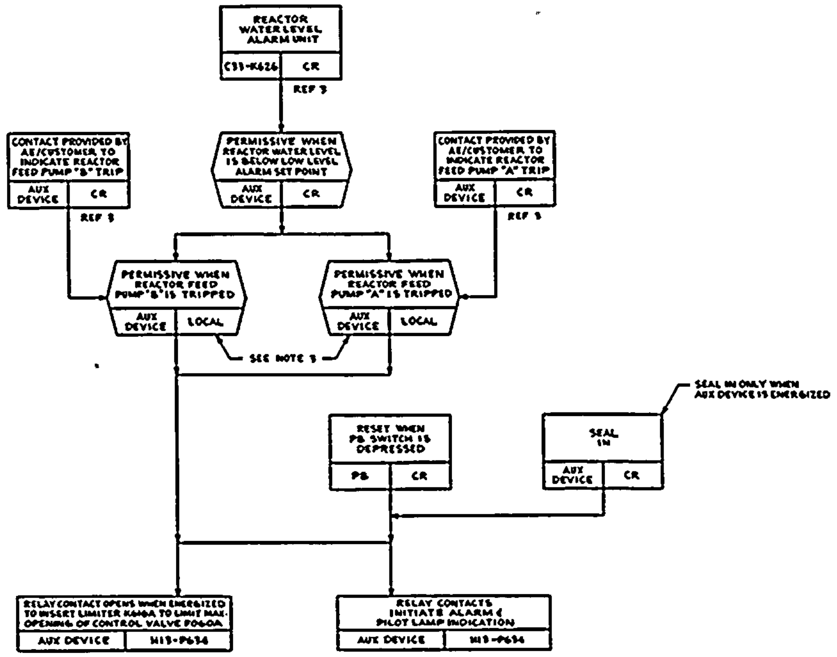
WNP-2 complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

The safety related buildings and spray ponds are located far above the water level estimated for the largest historical flood. Based on the criteria stipulated in Regulatory Guide 1.102, the WNP-2 plant site is classified as a "Dry Site."

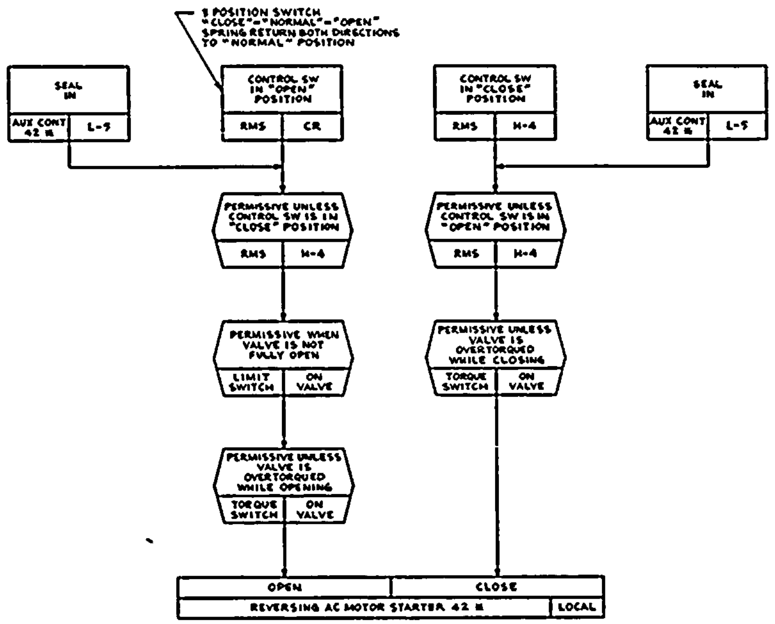
Specific Evaluation Reference:

Refer to 2.4.



ONE OUT OF TWO REACTOR FEED PUMP TRIP/REACTOR WATER LEVEL AUX DEVICE INTERLOCK WITH VALVE POSITION CONTROL
SEE REF 2

THIS FUNCTION IS TO BE USED IN CONJUNCTION WITH THE FEEDWATER CONTROL SYSTEM BASED ON TWO 95%-MINIMUM CAPACITY REACTOR FEED PUMPS



PUMP SUCTION VALVE F023A

NOTES:

1. FUNCTION IS SHOWN FOR RECIRCULATION SYSTEM "A" & IS TYPICAL FOR RECIRCULATION SYSTEM "B" EXCEPT FOR LETTER SUFFIXES.
2. AUXILIARY DEVICES ARE SHOWN WHERE NECESSARY TO CLARIFY THE FUNCTION.
3. FOR DETECTION OF A ONE OUT OF TWO REACTOR FEED PUMP TRIP, AND COINCIDENT OR SUBSEQUENT LOW REACTOR WATER LEVEL, LOGIC IS PROVIDED TO INITIATE CLOSURE OF THE RECIRCULATION FLOW CONTROL VALVE TO REDUCE THE REACTOR POWER LEVEL TO WITHIN THE CAPACITY OF THE REMAINING FEED PUMP. THE MOTOR DRIVEN PUMP FAILURE CONTACTS SHALL BE PROVIDED BY THE AE/CUSTOMER
4. NUMEROUS PILOT LAMP ALARM INDICATIONS ARE ALSO PROVIDED ON THE ANALOG CONTROL SYSTEM PANEL H13-P634 TO COMPLIMENT THESE REMOTE ALARMS. THE INDICATIONS ARE MORE EXTENSIVE AND INDICATE ALL FAULTS THAT GIVE RISE TO THESE FUNCTIONALLY GROUPED CONTROL ROOM ALARMS.
5. THE CUSTOMER/AE SHALL BE RESPONSIBLE FOR CONFIRMING DEVICES AS SHOWN ARE PROVIDED OR OTHERWISE CORRECTED ON THE FCD TO AGREE WITH CUSTOMER/AE INTERFACING SCOPE OF SUPPLY.

LEGEND:

SWGR = PUMP DRIVE MOTOR SWITCHGEAR.
≡ = SWITCHGEAR DEVICE FUNCTION NO. ANSI SPEC. C37.2.

ATWS = AUTOMATIC TRIP WITHOUT SCRAM.

REFERENCE DOCUMENTS:	MPL NO.
1. REACTOR RECIRC SYS VALVE FLOW CONT DES SPEC	B35-4010
2. REACTOR RECIRC SYS VALVE FLOW CONT P&ID	B35-1010
3. FEEDWATER CONTROL SYSTEM IED	C34-1010
4. NUCLEAR BOILER FCD	B22-1030
5. LOGIC SYMBOLS	A42-1030

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR RECIRCULATION SYSTEM FCD SHEET 1	FIGURE H.A-1a

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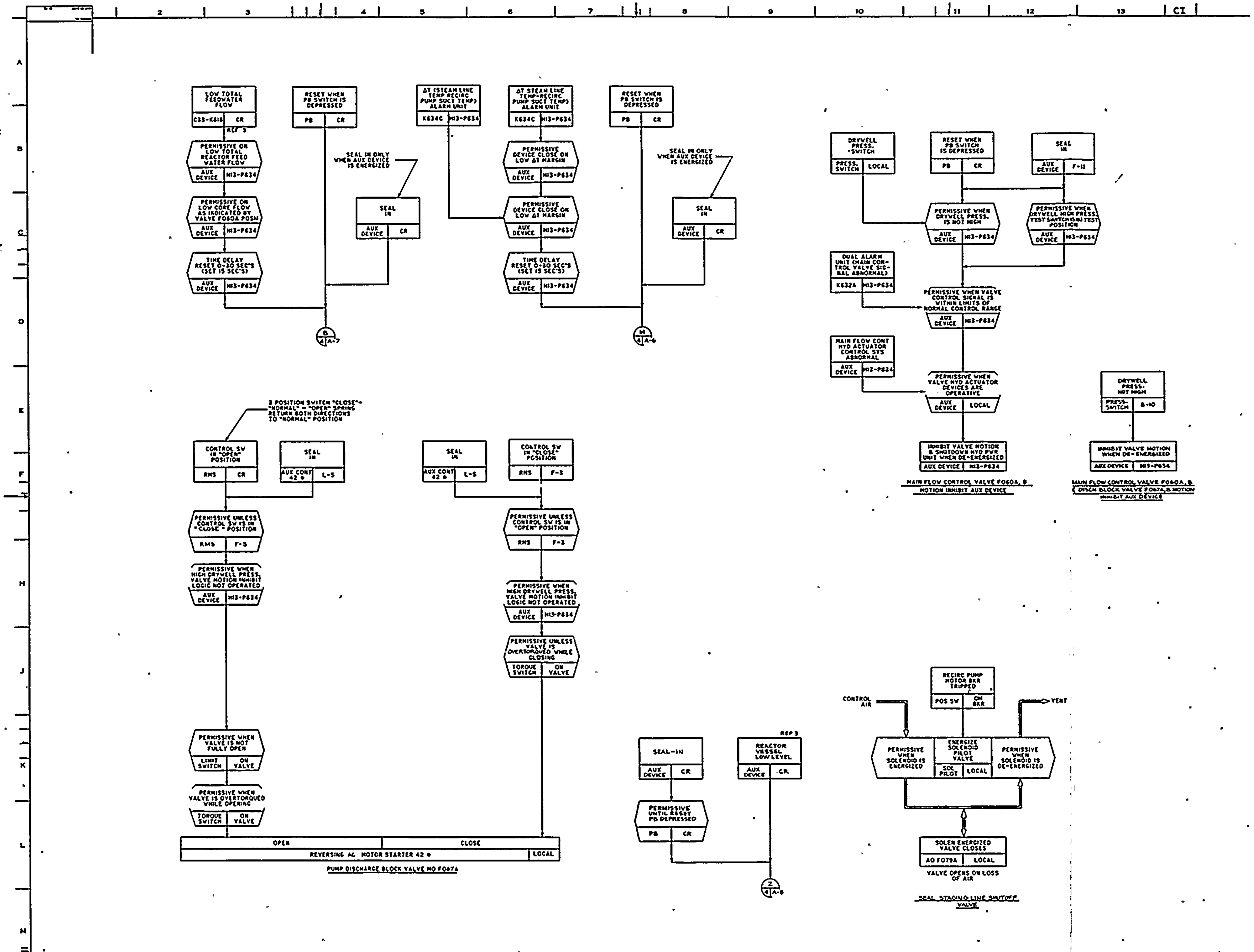

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

REACTOR RECIRCULATION SYSTEM FCD
SHEET 2

FIGURE H.A-1b

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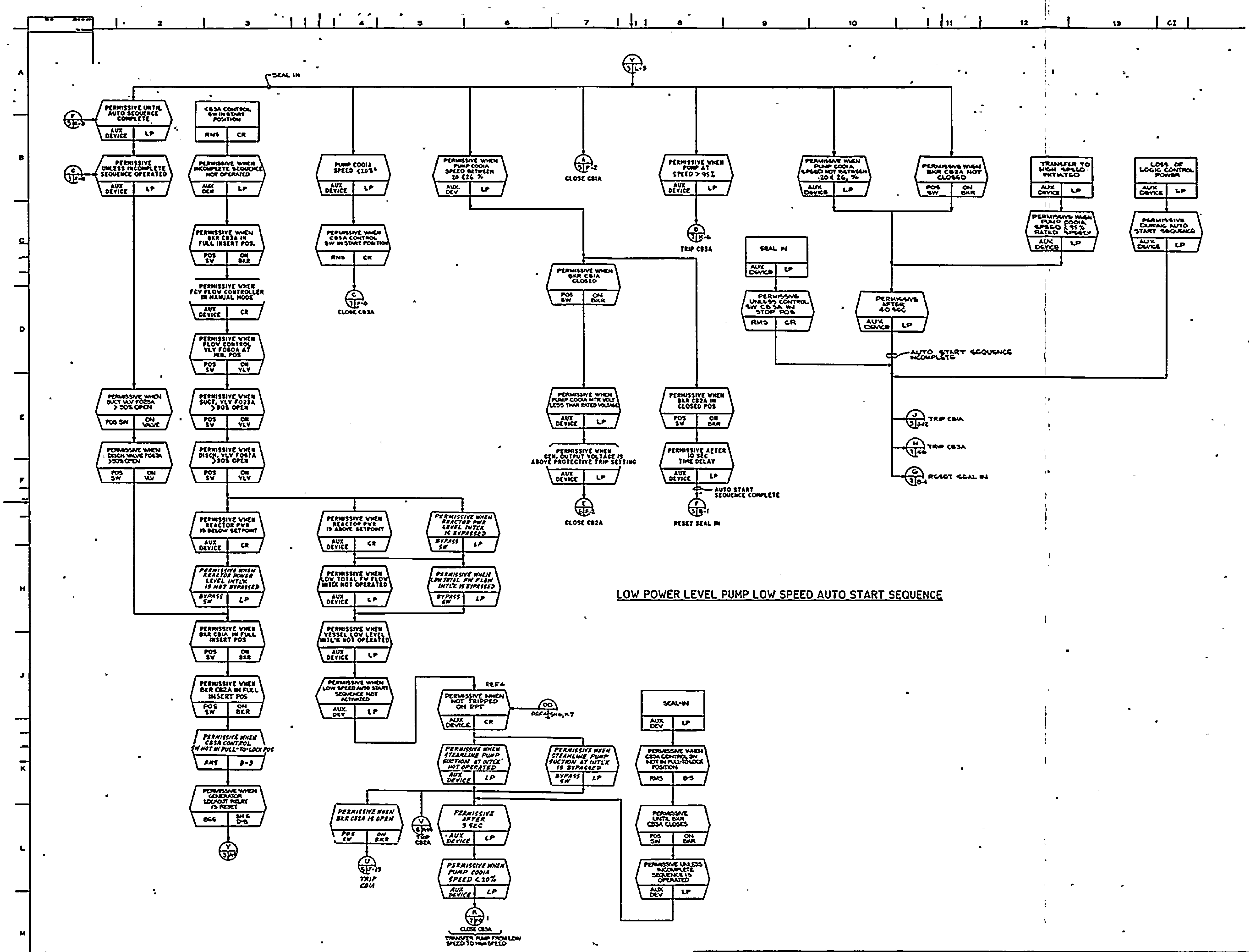

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR RECIRCULATION SYSTEM FCD SHEET 3	FIGURE H.A-1c

AMENDMENT NO. 10
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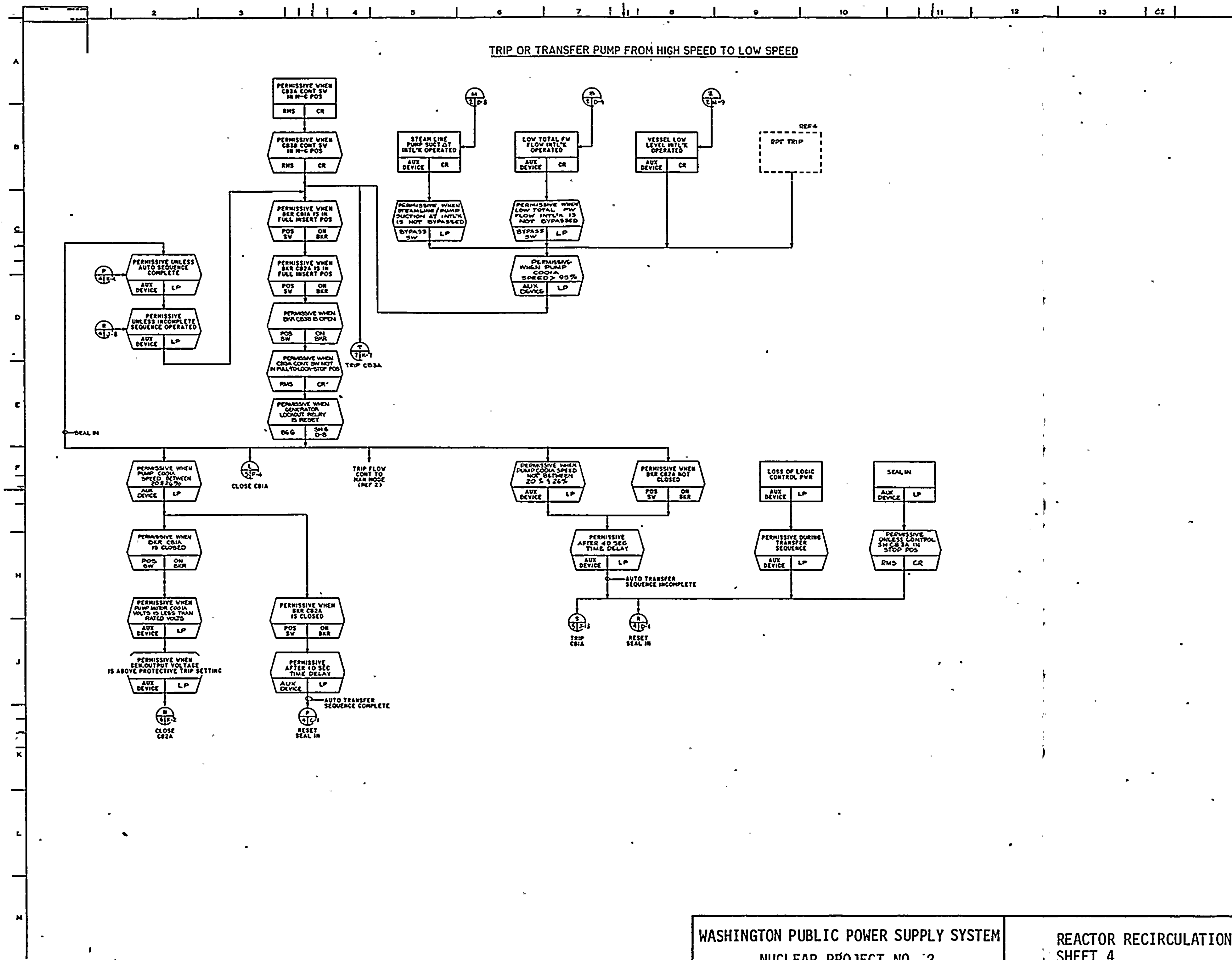
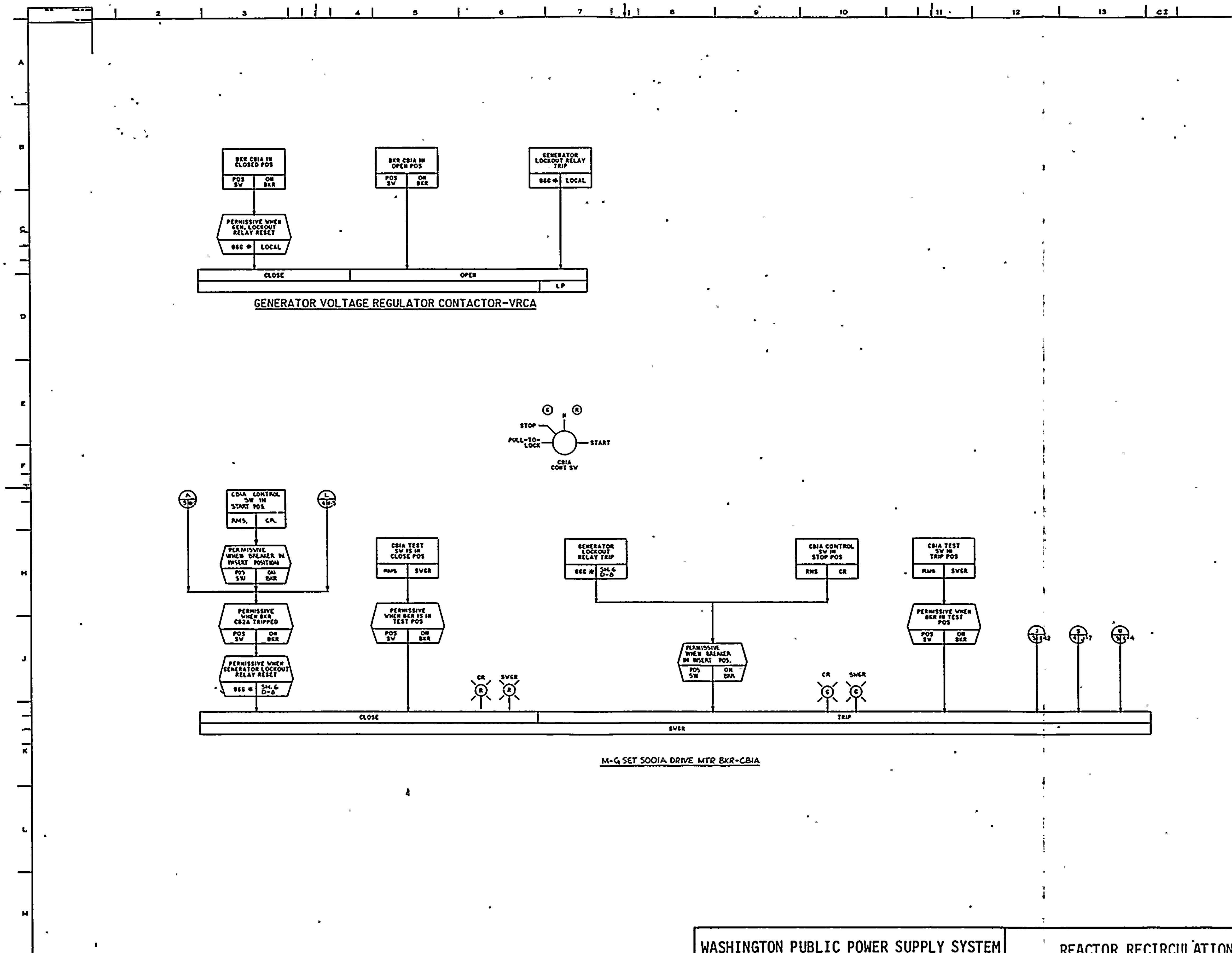
TRIP OR TRANSFER PUMP FROM HIGH SPEED TO LOW SPEED



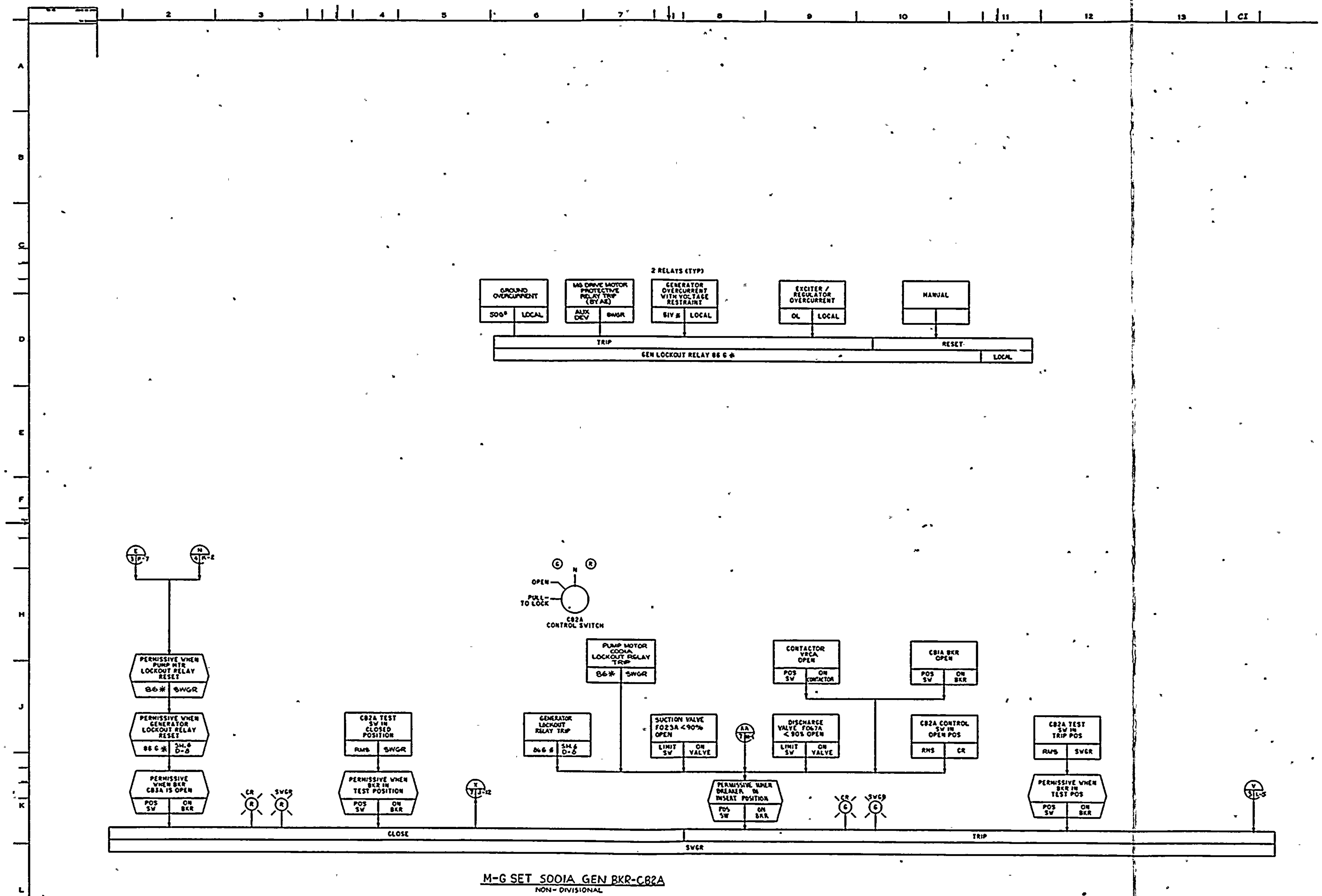
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR RECIRCULATION SYSTEM FCD SHEET 4	FIGURE H.A-1d



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

REACTOR RECIRCULATION SYSTEM FCD
SHEET 5

FIGURE H.A-1e



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR RECIRCULATION SYSTEM FCD SHEET 6	FIGURE H.A-1f

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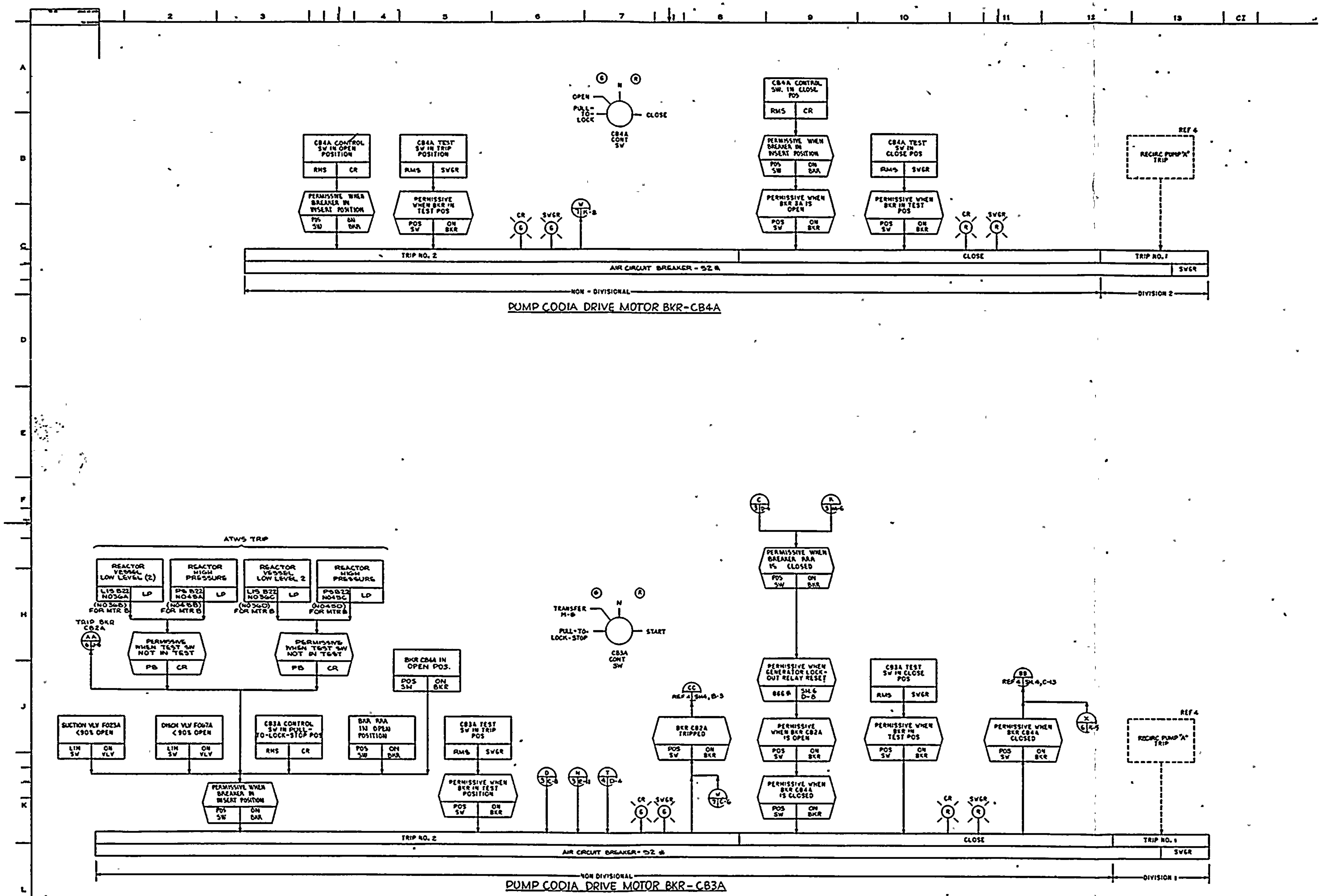
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR RECIRCULATION SYSTEM FCD SHEET 7	FIGURE H.A-1g

AMENDMENT NO. 10
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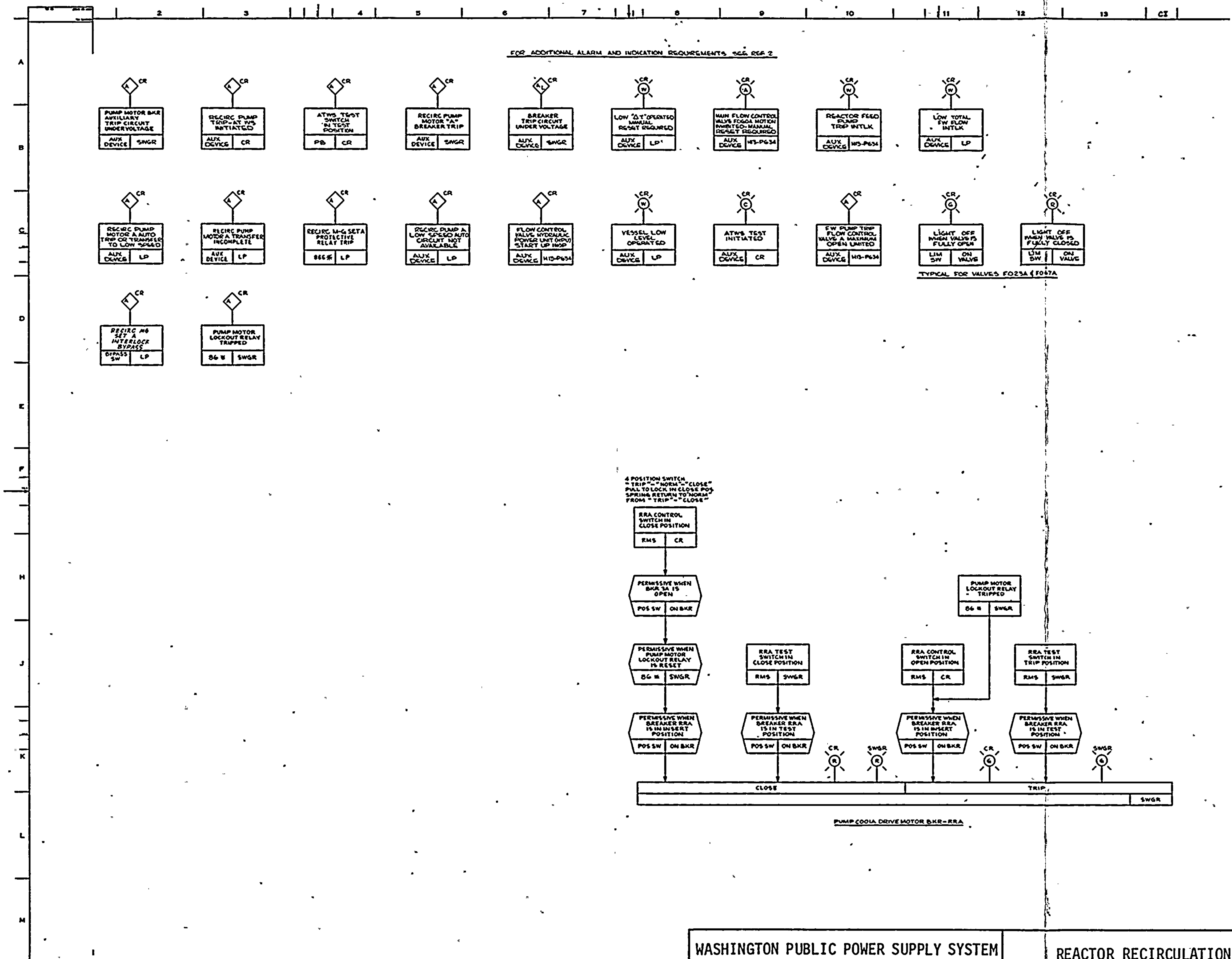

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR RECIRCULATION SYSTEM FCD SHEET 8	FIGURE H.A-1h

the GE Topical Report NEDE-24010-P, "Technical Bases for the Use of SRSS Method for Combining Dynamic Loads for Mark II Plants," is also applicable to WNP-2 with high seismic input.

The impact of the free-standing steel primary containment is discussed in the areas as follows:

a. Vessel and Internals

Vessel and internals are not attached to and not affected by the steel containment.

b. Piping Systems and Floor Mounted Equipment

The dynamic input to these components at their containment support locations may be affectd by the steel containment response to the dynamic loads under consideration and hence, may be different from that obtained from concrete containment. However, the frequencies contributing to the responses of major structures and components in both types of plants will not be significantly different but will fall into the same general range.

The structural frequencies will only determine the magnitude of amplification or attenuation of the response. For multi-frequency random-type dynamic loads, the components of input loads whose frequencies coincide with the structural natural frequencies will be amplified and these components will dominate the response. Although the predominant response of a particular structural component may vary somewhat in frequency between the concrete and steel containment configuration, the variances are expected to be small for the range of frequencies of interest for major structures because of the similarities in systems, types of structural configurations, construction materials, and massiveness of buildings. Therefore, key characteristics of the responses (duration of strong response motion and number of peaks) are primarily determined by the input component loads to the structure, and because of the similarity of the dynamic nature of the input loads due to earthquake, SRV and LOCA for both types of containment, their structural responses will have similar dynamic characteristics. Hence, the response of the mechanical components and piping systems supported from the two types of containments will also be similar. Hence, the use of SRSS combinations for combining the dynamic responses for the WNP-2 application will be demonstrated to meet the 84% non-exceedance probability level.

ISSUE: MEB-4 OBE PLUS SRV FATIGUE ANALYSIS

Question:

Clarify your consideration of the cyclic loadings due to the operating basis earthquake (OBE) and safety/relief valve actuation in your NSSS fatigue analysis.

Response:

For the NSSS piping, 50 peak OBE cycles are used. For other NSSS equipment and components, a generic study serves as the basis for 10 peak OBE cycles. As shown in Reference 1, 10 peak OBE cycles can envelope the cumulative fatigue damage of hundreds of less severe earthquake cycles. Accordingly the FSAR will be revised to reflect this position.

The methodologies used to evaluate the fatigue effects due to combined SRV and OBE loads are documented in Reference 2. In the fatigue analysis of NSSS equipment, piping, reactor pressure vessel, and RPV internal components, the actual calculated loads due to OBE and SRV are combined to show compliance with upset limits of fatigue.

2.0 SUMMARY AND CONCLUSIONS

2.1 GENERAL DESCRIPTION OF PLANT

The WPPSS Nuclear Project No. 2 (WNP-2) is a nuclear fueled electrical generating station which utilizes a General Electric Company BWR-5 (1969 product line) nuclear reactor with an electrical power output of approximately 1145 MWe.

The primary containment utilizes a Mark II over/under pressure-suppression configuration (see Figure 2.1-1). The primary containment consists of a steel pressure vessel enclosed by a concrete shield wall both supported by a concrete basemat. The primary containment is enclosed by the reactor building, a reinforced concrete structure functioning as a secondary containment.

The drywell is connnected to the suppression chamber by 102 downcomer pipes. Steam that could be released in the drywell during a postulated loss-of-coolant accident (LOCA) is channeled through these downcomer pipes into the suppression pool where it is condensed thus effecting pressure-suppression.

Eighteen safety relief valves (SRV) are mounted on the four main steam lines. When SRVs are actuated, steam from the RPV flows through the SRV discharge lines into the suppression pool where the steam is condensed. The discharge lines from all eighteen safety relief valves are routed inside selected downcomers into the suppression chamber (Figures 2.1-6 through 2.1-8). Each discharge line terminates with a quencher device having four arms. Seven of the eighteen safety relief valves are part of the automatic depressurization system (ADS) (Table 3.1-1) which is designed to function, under certain conditions, following a postulated intermediate or small size line break.

2.1.1 STRUCTURES, PIPING AND COMPONENTS DIRECTLY AFFECTED BY POOL DYNAMIC LOADS

The structures in the suppression chamber are shown in Figures 2.1-2 through 2.1-8. The structures, piping and components directly affected by the hydrodynamic events associated with the LOCA pressure suppression and the SRV discharge are identified below. The applicable hydrodynamic loads also are identified in Chapter 4.

a. Boundary Elements

The suppression chamber boundary elements are: the steel containment including the vertical and

horizontal tee stiffeners (Figure 4.1-4), the concrete basemat, the concrete pedestal, the diaphragm floor and the diaphragm floor seal.

b. Major Structures and Components

The major vertical structures are shown in Figure 2.1-2. They are the 102 downcomers, the 18 SRV lines including quenchers and support towers, and the 17 concrete columns supporting the diaphragm floor. The major horizontal structures are the steel truss shown in Figure 2.1-3 which provides lateral support to the downcomers and the SRV lines, and the platform at elevation 472'-4" shown in Figures 2.1-3 and 2.1-8. The downcomer bracing truss is submerged and the platform is located in the pool swell zone.

c. Miscellaneous Piping Systems

A developed elevation of the WNP-2 containment showing the location of the containment penetrations is shown in Figure 2.1-9. The piping systems of the suppression pool are classified as described below.

1. Fully Submerged Piping Systems

The eleven piping systems fully submerged in the suppression pool are shown in Figure 2.1-2. Seven systems enter the pool through containment penetrations at elevation 452'-0". One pipe (4"-FPC) enters the pool through the pedestal at elevation 451'-8-1/4". Two short lengths of pipe (instrumentation stubs) enter the pool at elevation 462'-0". A third (instrumentation line) enters at elevation 455'-0". Pipes below the downcomer vent exit at elevation 454'-4-3/4".

2. Partially Submerged Piping Systems

The thirteen partially submerged piping systems enter the suppression chamber through containment penetrations at elevations 467'-9" as shown in Figure 2.1-3 and enter the pool vertically within 3'-0" distance from the containment as shown in Figures 2.1-6 through 2.1-8.

2.2 SUMMARY AND CONCLUSIONS

2.2.1 SUMMARY OF CHANGES TO PRESERVE DESIGN MARGINS

As noted in Section 2.1, structures, piping, and components which may be affected by pool dynamic loads can be divided into two general categories, i.e., those directly affected by pool dynamic loads (those in and bounded by the suppression chamber) and those affected only indirectly by pool dynamic loads (outside the suppression chamber). This report covers the structures, piping, and components in and bounding the suppression chamber. For these structures several changes in design have been implemented as a result of consideration of SRV discharge and LOCA hydrodynamic loads. Table 2.2-1 provides a list of the structures and components that have been covered in this report, the design margins, controlling load combination, and the design changes that have been made. The steel containment structure has been reinforced by the addition of seven horizontal rows of tee stiffeners as shown in Figure 4.1-1. The downcomer bracing system has been redesigned from a system of radial beams to a pipe truss system. This bracing system also is designed to provide lateral restraint for the SRV discharge pipes. Quenchers have been provided as exit devices for the SRV discharge pipes. Additions and modifications of pipe supports for miscellaneous piping systems have been provided. Other miscellaneous changes are noted in Table 2.2-1.

2.2.2 CONCLUSIONS

The assessment concludes that the modified design of the wetwell for WNP-2 is capable of withstanding the effects of the hydrodynamic loads resulting from SRV actuation and postulated LOCA events in conjunction with other applicable loads.

The effects due to hydrodynamic loads outside the wetwell region are discussed in the FSAR.

TABLE 2.2-1
SUPPRESSION POOL ASSESSMENT SUMMARY

STRUCTURE	CONTROLLING MARGIN (NOTE 1)	CONTROLLING LOAD COMBINATION (NOTE 2)	CHANGES TO STRUCTURES DUE TO SRV AND LOCA LOADS	COMMENTS
Steel Containment	1.26	3	Added horizontal tee stiffeners, revised platform location and connection to containment.	
Basemat	bending- 1.14 shear- 1.27	7	None	
Pedestal	1.11	4	None	
Diaphragm Floor	downward- 1.62 uplift- 1.27	4a 4a	None	
Diaphragm Floor Seal	See Section 4.1.5.5	See Section 4.1.5	None	
Downcomer Bracing	1.68	5	Redesigned bracing system as a pipe truss system	
Columns	1.22	1	None	
Downcomers	1.08	See Section 4.2.3	Added stainless steel spool piece, provided local reinforcement where SRV pipe penetrates downcomer, raised the vacuum breaker valves out of the pool swell zone.	
SRV Piping Systems	1.05	See Section 4.2.4	Provided lateral restraint at downcomer bracing system, rerouted SRV lines	

3.2 LOADS ASSOCIATED WITH LOSS-OF-COOLANT ACCIDENT (LOCA)

A loss-of-coolant accident occurs when the integrity of the reactor coolant pressure boundary is breached and coolant is released. In order to contain the coolant which flashes to steam, WNP-2 utilizes a General Electric Mark II pressure suppression system. This system is discussed in 3.2.1. The LOCA loading phenomena are discussed in 3.2.2. The short-term and long-term LOCA loads are discussed in detail in 3.2.3 and 3.2.4, respectively. Section 3.2.5 describes the LOCA pressure and temperature transients and 3.2.6 describes the WNP-2 building response to the LOCA loads.

3.2.1 DESCRIPTION OF PRESSURE SUPPRESSION SYSTEM

The WNP-2 primary containment utilizes a General Electric Mark II over/under pressure-suppression configuration (see Figure 2.1-1). The drywell and suppression chamber (or wetwell) are large sealed volumes designed to contain and condense escaping reactor coolant. Both contain structures and piping systems with the suppression chamber approximately half filled with water (suppression pool) for steam quenching. The drywell is connected to the suppression pool by 102 downcomer pipes that channel steam released during a LOCA for quenching and pressure suppression. Details of the downcomers, other piping systems and structures in the suppression chamber are shown on Figures 2.1-2 through 2.1-8.

3.2.2 DESCRIPTION OF THE PHENOMENA AND RESULTING LOADS

The sequence of LOCA generated hydrodynamic events described below cause dynamic loads on the containment and on structures and components located in the wetwell. These transient dynamic forces (see Table 3.2-1) are termed dynamic forcing functions and are discussed in detail in Reference 3.2-2 and summarized below. Section 3.4 discusses the sequence of LOCA generated loads.

Following a postulated loss-of-coolant accident (LOCA), released coolant causes the drywell pressure to rise rapidly and to accelerate the column of water in each downcomer downward due to the pressure rise. As the water exits the downcomers and enters the suppression pool, it forms a jet-pool interface which rolls into a mushroom shaped vortex ring. Expulsion of water out of each of the downcomers results in a water jet which produces loads on submerged structures and suppression pool boundary pressure loads. Because bulk pool velocities are small during vent clearing, the corresponding

impact and induced flow field drag loads are generally small. However, locally, significant drag loads may result.

Immediately after vent clearing, air* in the downcomer vents from the drywell begins to flow into the suppression pool. LOCA air* bubbles are formed at the exits of the vents which charge and expand under the entire pool surface causing three dimensional drag loads on submerged structures. Upon contact with each other, the individual bubbles coalesce and accelerate the pool water above the downcomer vent exit plane vertically with no significant horizontal water motion.

Pool swell is the upward movement of suppression pool water above the exit plane of the vents due to injection of drywell air* below the pool surface. The velocity and acceleration of the water slug associated with this phenomenon produces impact, drag, and lift forces on structures within the swell zone.

The containment boundaries also experience loads due to drywell pressurization, air bubble pressure and wetwell free airspace compression. The rising pool surface motion is slowed due to the compression of air in the suppression chamber airspace. At about this time the rising bubbles break up the remaining pool slug which falls back to its original position terminating pool swell.

During pool swell, the bottom of the rising water slug continually falls back to the suppression pool due to instabilities at the bubble/water slug interface. This phenomenon and the large scale falling back of the remaining water slug at pool swell termination is known as fallback and causes drag and lift loads on structures in the pool swell zone, but a negligible containment boundary load.

Pool swell and the subsequent fallback of the remaining water slug are followed by an air/steam mixture flow through the downcomers until the drywell air is completely purged and the mass flux becomes pure steam. The loading phenomena associated with a high or medium steam mass flux is termed steam condensation oscillation.

The air content and steam mass flow rate along with the pool temperature determines the behavior of the steam/suppression

*During a LOCA, an air(nitrogen)/steam mixture would be blown down the downcomer vents; however, the analytical models of LOCA phenomena conservatively assume that only noncondensibles are injected into the suppression pool.

pool water interface. At high steam flow the interface location is essentially constant. As the flow rate decreases, due to reactor depressurization and associated drywell pressure decrease, the interface takes on an oscillatory character. The rate of change of the displacement of the interface is reflected in submerged structure and suppression pool boundary loads.

When the steam mass flux decreases below a critical level a hydrodynamic phenomenon termed chugging occurs. Chugging is associated with low steam mass flow and high suppression pool boundary pressure spikes relative to condensation oscillation. The phenomenon appears to be random in time and is caused by the complex interaction of water/steam condensation surface instabilities with the physical properties of the downcomers, the suppression pool, and the suppression pool boundaries. Chugging causes loads on the suppression pool boundary and submerged structures.

3.2.3 SHORT-TERM LOCA LOADS

Short-term LOCA loads are associated with hydrodynamic related phenomena that occur within a few seconds after LOCA initiation. The short-term loading phenomena include downcomer vent clearing, LOCA bubble charging, pool swell, and fallback. Figure 3.2-1 illustrates the short-term loading phenomena. The flow fields during downcomer vent clearing and LOCA bubble charging are three-dimensional. Flow fields during pool swell and fallback are vertical and exist only above the vent exit elevation. Loads on submerged structures due to downcomer vent clearing and LOCA bubble charging are compared and the larger loads are employed for subsequent evaluations.

Section 3.2.3.1 and Table H-1 provide a detailed summary of the short-term LOCA loading phenomenon and load calculation procedures used to assess WNP-2 structures.

3.2.3.1 Analytical Models and Supporting Test Data

3.2.3.1.1 Vent Clearing Jet and Induced Flow Field Model

To calculate the WNP-2 vent clearing jet and induced flow field, a LOCA Water Jet Analytical Model was developed for WNP-2. The model development and supporting test data are documented in Reference 3.2-9. The calculation is performed for a unit cell with a downcomer at the center. Among the input data required is the downcomer vent water clearing time history.

In order to calculate the downcomer vent water clearing time history and to provide a continuous pool surface displacement time history, a VENT computer code was developed. The model development and supporting test data for VENT are discussed in detail in D.2.

WNP-2 input data for the LOCA water jet model and the VENT computer code are shown in Table 3.2-2 and Table 3.2-3, respectively. As appropriate, maximums or minimums of WNP-2 parameters are used to make the input data conservative in order to maximize vent water clearing velocities.

The vent exit water velocity and acceleration calculated by VENT are increased by 10 percent as requested by the NRC in Reference 3.2-1. The velocity and acceleration time histories (including the 10 percent increase) are shown in Figures 3.2-2 and 3.2-3, respectively. As indicated in Reference 3.2-9, tests have shown that LOCA jet continues to propagate downward for a short duration beyond the vent clearing instant due to rapidly charging air. Although the vent clearing time for WNP-2 is 0.654 second, the jet tip reaches a maximum velocity of about 15.8 ft/sec. at t = 0.704 second.

Submerged boundary loads during downcomer vent water clearing is specified to be a static addition of an overpressure of 24 psi to the local hydrostatic pressure below the downcomer vent exit (walls and basemat) with a linear attenuation to zero at the pool surface.

3.2.3.1.2 LOCA Bubble Charging Model

In order to calculate the flow field associated with the LOCA bubble charging phenomenon, a LOCA bubble charging model was developed. The model development and supporting test data are discussed in detail in D.4.

As discussed in D.4, the LOCA bubble charging flow field is calculated using a numerical technique for potential flows in the exact WNP-2 suppression pool geometry. The uniformly charging LOCA bubbles are modeled as equal strength point sources located one downcomer radius below the WNP-2 vent exit at elevation 453'-4-3/4". Figure B-2 shows the modeled geometry for the WNP-2 LOCA bubble charging phenomenon and Figures 3.2-4 through 3.2-6 show contour plots of the maximum radial, tangential, and vertical components of the gradient of the velocity potential.

TABLE 3.2-7

WNP-2 PLANT PARAMETERS FOR LOCA TRANSIENT ANALYSIS

1.	DRYWELL	
	a. Free Air Volume	200,540 ft3
	b. Temperature (initial)	135°F
	c. Pressure (initial)	0.75 psig
	d. Relative Humidity	50%
2.	WETWELL	
	a. Free Air Volume	144,184 ft3
	b. Water Volume*	107,850 ft3
	c. Temperature (initial)	90°F
	d. Pressure (initial)	0.75 psig
	e. Relative Humidity	100%
3.	BREAK AREA	
	a. DBA - Recirculation Line	3.106 ft2
	b. DBA - Steamline	3.92 ft2
	c. Intermediate Break	0.1 ft2
4.	MAIN VENT	
	a. Maximum Submergence	12 ft
	b. Nominal Diameter	2 ft
	c. Number of Vents	102
	d. Vent Entrance Flow Area	304.6 ft2
	e. Downcomer Loss Factor	1.9

*Does not include water inside pedestal or below 12 ft beneath downcomer exit plane.

TABLE 3.2-8

SHORT-TERM LOCA LOADS ON STRUCTURES
BELOW ELEVATION 454.4'

Structure	Radial Location r of Geometric Center of the Structure or Segment of Structure			
	Zone I $0 \le r < 2.3R$		Zone II $2.3R \le r < 5.0R$	
	P_r max (psi)	P_v max (psi)	P_r max (psi)	P_v max (psi)
42" diameter vertical column	-	-	+2	-
12" diameter vertical SRV line	±20 inner row	-	+2 outer row	-
Pipes and Supports Diameter* > 12" (X-31,32,34,35,36)	±60	+212	+6	-45
Diameter* ≤ 12" (X-33,100,4" FPC, Quencher arm)	±60	+100	+6	-25

* For non-cylindrical structures, the diameter of a cylinder circumscribing the cross-section of the structure is used.

4.2 SUPPRESSION POOL MAJOR STRUCTURES AND COMPONENTS

Assessment of the capacities of the major structures and components of the suppression pool chamber relative to load combinations involving suppression pool hydrodynamic loads is made in this section. The structures considered are downcomer bracing system, columns, downcomers, SRV piping system, quenchers, and platforms.

4.2.1 DOWNCOMER BRACING SYSTEM

An assessment was made of the capacity of the original design of the downcomer bracing system relative to load combinations involving suppression pool hydrodynamic loads. It was determined that this original bracing, consisting of a system of radial beams, had inadequate capacity. Consequently, a replacement pipe truss bracing system has been designed and is now installed. The assessment of the capacity of this pipe truss system relative to the load combinations involving suppression pool hydrodynamic loads is made in this section.

4.2.1.1 Description of System

The pipe truss system of downcomer bracing is shown in Figure 4.2-1. Like the original system, the function served by the pipe truss system is to provide horizontal support for the 102 downcomers and the 18 safety/relief valve discharge pipes at a level near the lower end of the downcomers.

The pipe truss system consists of a horizontal planar truss located with center line at the same elevation as the original system. The model of the truss used in the structural analysis is shown in Figure 4.2-2. In the truss system, the downcomers and the SRV lines are located at the truss nodes. Structural rings are provided around each downcomer and each SRV pipe for connections by the truss members. The truss members are 4-inch and 6-inch double extra strong steel pipes. Connections of the truss to the RPV pedestal and to the containment vessel are at the same connection points as the original radial beams.

As is described in the section on loads (4.2.1.2), the pipe truss system is subjected to both horizontal and vertical loads. Horizontal reactions from the downcomers and from the SRV pipes are applied to the encircling structural rings which form the truss nodes. Horizontal forces applied directly to the truss members are also carried by the members to the truss nodes. By truss action, these horizontal loads are transmitted to the supports at the RPV pedestal and at the containment vessel. The pedestal connection can sustain both

radial and circumferential reaction components due to horizontal loading; however, the vessel reaction is circumferential because the connection is free to move radially.

Vertical loadings, due to the various causes listed under loads (4.2.1.2), act directly on the pipe truss system. To carry these vertical loadings, supports are provided against upward and downward motion at each of the downcomers; also the connections to the pedestal and vessel are restrained vertically. Vertical forces acting on each truss member are carried to its ends at the structural rings, pedestal, or vessel connections. The structural rings around the downcomers are independent of the downcomers but stops are welded to the downcomers to prevent differential vertical motion. The structural rings around the SRV lines are independent of these lines and no restraint against differential vertical motion is provided. Vertical loads from the rings on the downcomers are transmitted by the downcomers to the drywell floor.

4.2.1.2 Loads Used for Assessment

A complete description of all hydrodynamic loads is given in Chapter 3. In this section only the loads used for the assessment of the bracing system are discussed. Symbols used in this section are defined in 3.5.3 in connection with load combinations.

4.2.1.2.1 SRV Actuation Loads

SRV actuation causes horizontal and vertical loading on the bracing system as unbalanced pressures and induced accelerations of supported components occur. The pressures and accelerations acting on the downcomers, the SRV pipe lines, and the bracing members cause the horizontal loading; the vertical loading is due to these actions on the bracing system alone. The spatial distribution of these loads is discussed under Methods of Analysis (4.2.1.4).

Loads due to the SRV pressures and induced accelerations are applied to the bracing system as equivalent static loads. The magnitudes of the loadings from the downcomers and SRV lines are based on analyses of each of these components as described in the assessments of the components in 4.2.3 and 4.2.4 respectively. The pressure loadings on the bracing members proper are equivalent static pressures as defined in Chapter 3. Reactions due to the pressures on the downcomers, SRV lines, and truss members are applied at the truss nodes.

d. Column top shear connection - The smallest design margin representing the ratio of the capacity of the column top connection in shear to the applied top shear is 1.22.

4.2.3 DOWNCOMERS

The primary function of the downcomer vent system is to channel the steam accumulating in the drywell chamber during a loss-of-coolant accident (LOCA) into the wetwell chamber to accomplish pressure suppression. (Refer to 3.2.1.)

The downcomer vent system consists of eighty-four 24-inch OD and eighteen 28-inch OD standard schedule carbon steel pipes running vertically downward from the diaphragm floor (except that the ends of the downcomers are stainless steel as described in FSAR Sec. 3.8.3.4.) The downcomers are embedded in the diaphragm floor and extend down to elevation 454'-4-3/4". All downcomers are restrained laterally at elevation 455'-4" by the downcomer bracing system, which is vertically restrained by the downcomers. Vertical loads are imposed by the bracing system onto the downcomers and transmitted to the diaphragm floor. (See Figures 2.1-5 and 2.1-6.)

Nine of the 24-inch OD downcomers have an extra strong welding tee at elevation 491'-11" to accommodate 24-inch dual inline vacuum breaker valves. In addition, to provide extra strength, the eighteen 28-inch downcomers have been stiffened by the insertion of a 4'-8" long by 2-inch thick spool piece. This piece accommodates the penetration for the main steam safety/relief valve (MSRV) piping which is welded to these downcomers at elevation 493'-0".

Figure 2.1-4 shows locations where the vacuum breaker valve assemblies and the MSRV piping penetrate the downcomers just below the diaphragm floor.

4.2.3.1 Loads Used for Assessment

The downcomer piping is subjected to static, dynamic, and hydrodynamic loads under the various plant operating conditions identified as normal, upset, emergency, and faulted. Each of these loads in various combinations is identified in 3.5.4.

The individual loads acting on the downcomers are identified below:

a. Deadweight (W)

b. Thermal expansion and thermal transient

c. Pressure (P)

The pressure differential between the drywell and suppression chamber atmospheres produces loads on the downcomer walls since it acts as a pressure retaining boundary during a LOCA.

d. Operating basis earthquake (OBE)

e. Safe shutdown earthquake (SSE)

f. Safety/relief valve (SRV) discharge dynamic loads

The spatial distribution of the maximum direct bubble pressure loading used for downcomer assessment was obtained by multiplying a dynamic pressure load (Figure 3.1-7) by the maximum dynamic load factor (DLF).

A maximum DLF of 4.2 was conservatively obtained from the response spectrum of dynamic load factors shown in Figure 3.1-10.

The inertia loading effects due to the acceleration of the structure are described in 5.1. The response spectra used for downcomer assessment are the enveloped spectra which were developed by enveloping the spectra due to four SRV actuation cases at the appropriate locations for the downcomers.

g. Loss-of-coolant accident (LOCA) loads

The loads on the downcomer associated with LOCA are chugging pressure, condensation oscillation pressure, and the building response loading during LOCA event.

The spatial distribution of condensation oscillation and chugging pressure loadings on downcomers are considered as equivalent static pressure loads.

TABLE 4.3-2

PIPING ZONE VERSUS LOADS

	ZONE			
Loading	Fully Submerged	Partially Submerged	Pool Swell Zone	Above Pool Swell Zone
Deadweight	X	X	X	X
Thermal	X	X	X	X
Pressure	X	X	X	X
OBE	X	X	X	X
SSE	X	X	X	X
SRV Pressure	X	X		
SRV Response Spectra	X	X	X	X
LOCA Jet	X	X		
LOCA Bubble	X	X		
Pool Swell and Fallback	X	X	X	
Chugging Pressure	X	X		
Chugging Response Spectra	X	X	X	X

TABLE 4.3-3

SUMMARY OF RESULTS AND DESIGN MARGINS FOR MISCELLANEOUS WETWELL PIPING

PENETRATION NUMBER	CONTROLLING LOAD CASE	FOR MOST SEVERE CONDITION ALLOWABLE STRESS PSI	CALCULATED STRESS PSI	DESIGN MARGIN
X-33	Upset	18,000	12,264	1.46
X-100	Upset	18,000	5,231	3.44
X-32	Faulted	36,000	14,667	2.45
X-31		LATER		
X-36		LATER		
X-4	Emergency	27,000	14,453	1.86
X-101	Faulted	36,000	19,826	1.81
X-47 & 117	Emergency	27,000	16,952	1.59
X-23	Upset	18,000	13,963	1.28
X-49	Emergency	27,000	13,001	2.07
X-26	Emergency	27,000	19,580	1.37
X-24	Emergency	27,000	19,344	1.39
X-35	Emergency	27,000	13,871	1.95
X-34		LATER		
X-48 & 118	Emergency	27,000	24,400	1.03(1)

TABLE 010.011-2

COMPARISON OF DESIGN AND PREDICTED ENVIRONMENTAL CONDITIONS FOR ESSENTIAL EQUIPMENT IN MAIN STEAM TUNNEL EXTENSION(1)

ESSENTIAL EQUIPMENT IN MAIN STEAM TUNNEL(3)			EQUIPMENT ENVIRONMENTAL CONDITIONS - IN MAIN STEAM TUNNEL EXTENSION							
			DESIGN			MAXIMUM PREDICTED,(2) FOLLOWING POSTULATED PIPE CRACK IN MAIN STEAM TUNNEL EXTENSION				
NAME	DESIG-NATION	LOCATION IN TUN-NEL EX-TENSION	TEMP. (°F)	PRESSURE (PSI)	HUMIDITY (%)	TEMP. (°F)	PRESSURE (PSI)	HUMIDITY (%)	B&R REFERENCE DRAWING	REMARKS
General Electric Radiation Detectors	D17-N003A	North	Later	Later	Later	313	8	100	E607	
	D17-N003B	North	Later	Later	Later	313	8	100	E607	
	D17-N003C	North	Later	Later	Later	313	8	100	E607	
	D17-N003D	North	Later	Later	Later	313	8	100	E607	
Cables, Electrical and Instrumentation and Control	Data same as in Table Q.010.11-1 and associated footnotes 4 and 5								E590,607	

(1) The main steam tunnel extension is in the turbine-generator building. The main steam tunnel, Table Q. 010.11-1, is in the reactor building.

(2) The maximum predicted conditions correspond to a time duration of 0 to 2 hours. From 2 hours to 6 hours, the predicted temperature is 212°F and the predicted pressure approaches atmospheric.

(3) There is no tubing in the main steam tunnel and tunnel extension for air lines operating instrumentation and control equipment and components or for any other application.

Q. 010.12

Provide the results of your evaluation of the jet impingement forces and the environmental effects, including pressure, temperature, humidity, and flooding, resulting from a postulated failure of the main steam and main feedwater systems in the Turbine Building. This evaluation should address only those safety-related components, systems and structures, if any, in (or immediately adjacent to) the Turbine Building (e.g., the walls of the Auxiliary Building).

Response:

It has been determined that the only items with safety-related functions in the Turbine Building are some RPS sensor inputs from the Main Steam System, MSIV isolation logic inputs from the Main Steam System, and the Tower Make-up Transformers located in the basement of the Turbine Building which are required to function only for the Design Basis Tornado event. This last item is remote from the steam and feedwater lines (being located at the basement grade level of the building) and has been evaluated to have adequate protection from tornado missiles and internal flooding (see the responses to question 10.25 and 10.34). In addition, there is cabling for the condensate storage tank level sensors which provide for auto-switching of HPCS from the storage tank to the suppression pool. The routing of this cabling is currently through the Turbine Building, but is under design review to insure its adequate protection from accidents. Appropriate design changes will be made as a consequence of this evaluation. Accordingly, the only items of concern are the RPS and MSIV isolation logic sensor inputs. Due to their nature they cannot be made immune from pipe-break effects. However, no analysis has been performed of the specific effects of a steam line or feedwater break in the Turbine Building on this equipment since it has been determined that the complete loss of all this equipment could occur for these events without the loss of capability to bring the plant to a cold shutdown or mitigate the radiological consequences of such an incident even assuming a single failure in the safety systems that remain unaffected.

The electrical cable connected with this safety related equipment in the corridors separating the Turbine Building, Reactor Building, and Radwaste Building would be exposed to temperatures and pressure effects of a postulated failure of the main steam or feedwater lines in the Turbine Building, but the exposure conditions would be for less than the design environment requirements contained in the purchase specifications for the cable.

No other safety-related equipment is located in an area which would be vulnerable to the environmental effects of a pipe break in the Turbine Building. The only safety related structures adjacent to the Turbine Building are the Reactor Building and Radwaste-Control Building. A pipe break in a main steam or feedwater line in the Turbine Building would result in transitory pressurization of the corridors between the Turbine Building, Reactor Building, Radwaste-Control Building and Diesel-Generator Building. Air and steam would be forced into these corridors through openings in the south wall of the Turbine-Generator Building, and through the seismic gap between the Turbine Building, Reactor Building and Radwaste-Control Building. No compartmental pressurization analysis is required to determine peak pressures and temperatures in the corridors due to the large volume of the Turbine Building, and the fact that the metal siding and exterior doors into the Turbine Building are not leak-tight and are not designed to withstand more than a minimal pressure differential, the peak pressures seen by the reinforced concrete walls of the Reactor Building and Radwaste-Control Building would not exceed the structural capacity of the walls. The doors to the control room are low-range blast doors, designed to withstand a pressure differential of 3 pounds per square inch, which is considered adequate to maintain control room habitability as discussed in 3.6.1.12.

It should be noted that the response to this question is directed towards the Turbine Building as a whole and does not cover the steam tunnel. The response to question 10.11 will address this area.

BLANK

It should be noted that Chapters 3.5 and 3.6 will be revised in line with the on-going pipe break/missile study. The information in the response to this question will be more clearly incorporated into the FSAR text at that time.

Q. 022.035

Provide the following information related to potential bypass leakage paths:

a. For each air or water seal, perform an analysis of the fluid inventory which will be available to maintain the seal for 30 days following a postulated loss-of-coolant accident and demonstrate that this fluid inventory will be sufficient. Describe the testing program and the specific details of your proposed technical specifications which will verify the assumptions used in the analysis. Provide the basis for the valve fluid leakage used in your analysis.

b. For each of these paths where water seals eliminate the potential for bypass leakage, provide a sketch showing the location of the water seals relative to the system isolation valves

c. Explain why the combustible gas control system is omitted from Table 6.2-13 of the FSAR as a potential leakage path. Demonstrate that this system meets each of the provisions of Branch Technical Position CSB 6-3, Section B-9, for a closed system.

Response:

a. Potential bypass leakage paths around the secondary containment are discussed in 6.2.3, Secondary Containment Functional Design. As discussed in 6.2.3.2, the two 24-inch reactor feedwater (RFW) lines are the only lines for which a water or air seal is assumed which prevents secondary containment bypass leakage. The water seal will provide about 20 minutes of leakage protection subsequent to a LOCA and cessation of feedwater flow. For some accidents, feedwater may be available using the condensate and condensate booster pumps. The motor-operated gate valve in each RFW line will be closed by operator action within 20 minutes after feedwater flow is

Sheet 2 of 2

determined to be not needed or unavailable and degraded core conditions exist. This gate valve and the two RFW isolation check valves prevent long-term bypass leakage.

As discussed in 6.2.3.3 and 6.2.6.3, the isolation valves on the lines which were identified as potential bypass leakage paths around the secondary containment, will be tested to ensure that the individual leakage rates are below the limits allowed by ASME Code Section XI, IWV-3426. The limit allowed by IWV-3426 was the value assumed in calculating the water lost from the water seal through the RFW isolation valves.

b. Figure 6.2-25 shows the RFW line routing.

c. The containment atmosphere control (CAC) system is a closed system outside the primary containment. Suction and discharge are to the primary containment. All piping remains within the secondary containment. Any leakage from the CAC system will be processed by the standby gas teatment system prior to release to the environment. The CAC system is described in detail in 6.2.5 and shown in Figure 3.2-17.

The CAC system meets all the criteria stipulated in BTP CSB 6-3 paragraph B.9. The CAC system does not directly communicate with the environment, is designed to Code Group B standards, meet Seismic I design requirements, is designed to the primary containment pressure and temperature design conditions, is designed against the consequences of any breach in the reactor coolant pressure boundary (pipe whip, etc.), and will be open to the primary containment atmosphere during the integrated leak rate test. In addition, the CAC system can be isolated from the primary containment by two, redundant isolation valves. There is no reason to consider the CAC system as a secondary containment bypass leakage path.

Q. 031.001(j)

Provide justification for not seismically qualifying the feedwater and control rod drive excess flow isolation valve actuators.

Response:

Seismic qualification documentation for the Class IE feedwater isolation valve actuators is presently under examination as part of an overall qualification review with results to be provided to your SQRT personnel. This review will involve reevaluation of all the Class IE equipment seismic qualifications to assure they demonstrate adequacy of the methods and results as equal or conservative to the requirements of IEEE 344-1975. The evaluations will be documented in 3.10.

The control rod drive excess flow isolation valve has been deleted along with CRD Return Line.

Q. 031.001(k)

Provide justification for not environmentally qualifying the feedwater and control rod drive excess flow isolation valve actuators.

Response:

Environmental qualification documentation for the Class IE feedwater isolation valve actuators is presently under examination as part of an overall qualification review to determine degree of compliance with NUREG-0588 Category II, the results of which will be provided in 3.11. Where significant deviation from those guidelines is found in specific equipment qualifications, additional testing and/or analysis will be performed to demonstrate the adequacy of the equipment to perform its safety-related function.

The control rod drive excess flow isolation valve has been deleted along with the CRD Return Line.

b. The standard temperature for the control room instrumentation is 40-120°F and 90% relative humidity (maximum). The range of temperatures and humidity over which the LPCI, LPCS, HPCS, ADS, and MSIV-LCS instrumentation and controls will meet their design basis is provided in Table 3.11-1, 3.11-2 and 3.11-3.

c. Probable maximum floods have no effect on Class 1E systems (see 3.4).

d. Instrument response times used in the WNP-2 simulation in Chapter 15 will be provided in the response to Question 031.016.

Table 7.2-1, 7.3-1, 7.3-3, 7.3-5, 7.3-7, 7.3-9, and 7.4-1 will be revised to include instrument accuracies in response to Question 031.021.

e. The differential pressure sensors (level switches and ΔP transmitters) are designed for one side pressurization capability of up to 2000 psig without damage to diaphragm bellows.

Amendment 10, revising Chapter 7, replaced 7.3.2.1.2.3.1.19, 7.3.1.1.1.4.5, 7.3.1.2.3.1.20 with 7.3.2.1.2.a.19, 7.3.1.1.1.2, 7.3.2.1.2.a.20 and 7.3.1.1.1.2 respectively. Tables 7.3-2, 7.3-3, 7.3-4 and 7.3-5 were renumbered 7.3-3, 7.3-5, 7.3-7 and 7.3-9 respectively.

m. Section 7.3.1.2.7 has been replaced in the Chapter 7 rewrite by 7.3.1.2.F which now reads as follows:

"Refer to Tables 3.11-1 through 3.11-5 and paragraph 3.1.2.1.4.1 for environmental conditions. Refer to Sections 8.2.1 and 8.3.1 for the maximum and minimum range of energy supply to ESF instrumentation and controls; all ESF instrumentation and controls are specified and purchased to withstand the effects of energy supply extremes."

n. Due to a complete Chapter 7 rewrite in Amendment 10, FSAR 7.6.1.4.2 has been changed to 7.6.1. Reference to the RPS buses as "critical" has been eliminated.

o. The only actual discrepancy existed in Table 7.3-2. This table did not list the confirmatory reactor vessel low water level (Level 3) used in the ADS initiation logic. Amendment 10, revising Chapter 7, revised the tables to clarify water level trips and renumbered Tables 7.3-2, 7.3-3, 7.3-4 and 7.3-5 to 7.3-3, 7.3-5, 7.3-7 and 7.3-9 respectively.

p. Due to a complete Chapter 7 rewrite in Amendment 10, FSAR Table 7.7-3 has been changed to Table 7.7-5. As stated in Notes (3), (5), and (6) of Table 7.3-3, the instrument setpoints are subject to change to agree with Chapter 16, "Technical Specifications", which have not been submitted and are under development. The actual trip settings will be established when the Technical Specifications are submitted, after which Tables 7.3-1 (HPCS), 7.3-3 (ADS), 7.3-5 (LPCS), 7.3-7 (LPCI), 7.3-9 (PCRVCS), 7.2-1 (RPS) and 7.4-1 (RCIC) will be revised. The trip setpoints will take into account accuracy, calibration and drift allowances so that the required actuation will fall within the analytic or design basis limits.

q. The statement in 7.6.1.8.1.2 has been removed from the revision to Chapter 7. Table 7.1-2 is correct as is.

r. The section discussing Recirculation Pump Trip (RPT) system instrumentation and controls has

been rewritten to refer to Appendix H and 5.4. There are two RPS divisions.

s. Due to a complete rewrite of Chapter 7 in Amendment 10, FSAR 7.6.2.8.2.1.1.4 has been changed to 7.6.2.3.A.4 and reads as follows:

"Equipment Qualification (IEEE 279-1971, Paragraph 4.4)

Vendor certification requires that the sensor associated with each of the systems required for safety trip variable, manual switches, and trip logic components perform in accordance with the requirements listed on the purchase specification as well as in the intended application. This certification, in conjunction with the existing field experience with these components in this application, will serve to qualify these components.

Qualification tests of the relay panels are conducted to confirm their adequacy for this service. In-situ operational testing of these sensors, channels, and the entire protection system will be performed at each project site during the preoperational test phase.

For a complete discussion of equipment qualification for the safety-related systems described in 7.6, refer to 3.5, 3.6, 3.10 and 3.11".

t. WNP-2 is presently reviewing qualification requirements of Class 1E equipment. A composite list of Class 1E equipment, including RPT initiation sensors, logic, breakers, etc., will be entered in Tables 3.10-1 and 3.10-4 or equivalent tables generated.

u. The Rod Sequence Control System referred to in 7.6.1.7.8 is a non-safety-related system. Environmentally-related statements have been removed from the text and the text has been moved to 7.7.

Q. 031.083
(3.11.3)
(3.11A)
(031.006)
(031.056)
(031.059)

Neither your response to Item 031.006 nor Appendix 3.11A of the FSAR satisfy our need for additional information on equipment qualification. In order to ensure that your environmental qualification programs conform with General Design Criteria 1, 2, 4, and 23 of Appendix A and Section III and XI of Appendix B to 10 CFR Part 50, and to the national standards (e.g., IEEE Standard 323-1971) mentioned in the Acceptance Criteria contained in Section 3.11 of the Standard Review Plan, NUREG-75/087, provided an amended response to Item 031.006 for:

a. The logic equipment for the standby gas treatment system (031.006, item (d) of the second paragraph).

b. The following sensors: (1) the rod block monitor flow transmitters; (2) the main steam line tunnel temperature thermocouple; and (3) B22-N024A.

c. All items listed in Questions 031.056 and 031.059.

Response:

a. The requested information for logic equipment for the standby gas treatment system will be provided as part of the overall re-evaluation program for seismic and environmental qualification. See response to Question 031.006.

b. The requested information will be provided as part of the overall re-evaluation program for seismic and environmental qualification. See response to Question 031.006.

c. 031.056 - See response to Question 031.006.

031.059 - See revised response to this Question.

Q. 031.084
(3.11A)

The specification requirements of Table 3.11A-1 of the FSAR are incomplete since they do not address the maximum and minimum values of all of the parameters which are cited in Section 3(7) of IEEE Standard 279-1971. Accordingly, provide the required data for all Class 1E components.

Response:

See response to Question 031.006.

Q. 031.099
(3.4)
(7.3.1)
(031.030)

The response to Item 031.030(c) is incomplete since you do not discuss the consequences to electrical equipment in the event of internal flooding. Section 7.3.1.2.8.1 of the FSAR is similarly incomplete. Accordingly, provide a revised response to Item 031.030(c) which discusses the protection of Class 1E equipment from internal flooding (e.g., a failure of either the main condenser cooling line or of the fire protection system).

Response:

The protection of Class 1E equipment from internal flooding is discussed in FSAR 3.4.1.4.1.2 and 3.4.1.5.2, which were submitted in Amendment 5. The ECCS equipment in the reactor building basement, where the building sumps are located, are protected from internal flooding due to post-LOCA. ECCS passive failures by a Class 1E leak detection system are discussed in the response to FSAR Question 212.003. The passive failure is isolated before it has any additional effect on ECCS operation.

The potential flooding and environmental effects from postulated through-wall leakage cracks in moderate energy fluid piping systems, and postulated rupture of high energy fluid pipings are currently being re-evaluated as stated in FSAR 3.6.

The effects of the internal flooding on electrical equipment are being taken into account in the re-evaluation. The results of this analysis will be furnished by amendment to FSAR 3.6. At that time a change to Question 031.030(c) and this question will be provided.

Q. 031.100

In Table 7.1-2 of the FSAR, you indicate that many of your instrumentation and control systems are identical to those of LaSalle and Zimmer. During the course of our review of these facilities, which are similar to the WNP-2 facility, we encountered a number of errors in the implementation of the basic GE design. Our concern is that these same errors, or similar errors, could occur in implementing the electrical design of the WNP-2 facility. In particular, we find that your analyses in 7.3.2.1.2.3.1 and 7.3.2.2.2.3.1.1 of the FSAR, to determine compliance with the requirements of IEEE. Std. 279-1971, are too gneral in content. We provide guidance for the information we need in Section 7.2 of the Standard Review Plan, especially in Appendix 7.2.A. Specific examples of areas where we require additional information are presented in Items 031.081, 031.084, 031.091; and 031.092 of this enclosure. Accordingly, provide more specific analyses of how you have implemented, in detail, the basic GE electrical design in the WNP-2 facility. References to other sections of the FSAR are acceptable in lieu of repeating this information in 7.3.2.1.2.3.1.

Response:

Refer to the revised Chapter 8.3.

TABLE 040.079-3

HP/LP PROTECTION REQUIREMENTS

VALVE PAIR	FIRE AREA	PROTECTION REQUIRED
RHR-V-8/9	R-I	1. Protect cables 2M8BA-314 and 2NS4-2, 30. 2. Protect devices B35-N018B (H22-P022) and E31A-N012B (H22-P021).
	RC-II	Protect cables 2M8BA-314, 315, and 2NS4-2.
	RC-III	Protect cables 2M8BA-314 and 2NS4-2
	RC-IX	1. Provide a new transfer contactor (located in a Div. 1 Fire Area) for RHR-V-8. 2. Protect cables 1M21A-22 and 1M21A-23 in Fire Area RC-IX if the new transfer contactor arrangement does not result in these cables bypassing RC-IX.
	RC-X	See Note 2.
RHR-V-53A/123A	R-I	Protect cable 2M8BA-502.
	R-II	Protect cable 2M8BA-502.
	RC-X	See Note 2.
RHR-V-53B/123B	R-I	Protect cable 2M8BA-434.
	RC-II	Protect cable 2M8BA-434.
	RC-III	Protect cable 2M8BA-434.
	RC-X	See Note 2.

TABLE 040.079-3 (Continued)

Notes:

1. The term "protect cables" indicates that this cabling must be segregated from cabling for its companion series valve both for internal and external fires.

2. The base Fire Hazards Analysis assumes a main control room fire which envelops the entire main control room. This extremely conservative assumption has been made to facilitate fire hazard shutdown analysis. For the case of the valves indicated above, a more realistic, less conservative assumption is made that a fire in the continuously manned main control room is localized.

   For each of the valve pairs indicated, each valve is part of a redundant electrical division. The valve control circuits and switches are separated from each other within the main control room in accordance with accepted WNP-2 criteria for redundant electrical equipment.

   The "as-built" separation for these valves is considered adequate protection for the realistic case of a localized main control room fire.

AMENDMENT NO. 9
April 1980

Q. 110.033
(3.6.2)
(3.9.3)

For ASME Class 1, 2 and 3 components that could be exposed to either jet impingement loads or to pipe whip impact loads resulting from postulated pipe breaks in adjacent high energy piping, describe how you determine the stress levels in the targeted components. In your response, include a discussion of the structural effects throughout the targeted system from the loads cited above (i.e., those loads associated with postulated pipe breaks) in combination with other applicable loads. Provide assurance that the calculated stress levels are kept below the Service Level D limits of Section III of the ASME Code. If applicable, more conservative limits on stress levels should be imposed for active components or where piping functional capability is required.

Response:

The complete response to NRC Question 110.033 awaits the results of the ongoing pipe break and missile study, and will be presented in a future amendment to the FSAR.

At this time it is anticipated that if the availability of ASME Section III Class 1, 2 and 3 components is needed to safely shut down the plant, and if those components are exposed to postulated jet impingement loads, pipe whip impact loads, and missile loads protective measures will be taken to preclude such loading. It if is determined that protective measures are not required on the basis that the calculated stress levels due to the postulated pipe break for missile loads in combination with other applicable loads are kept below the Service Level D Limits of ASME Section III, such structural analysis will be in accordance with industry-accepted methods. If this approach is used, the methodology and results will be reported in the FSAR.

Q. 110.034
RSP
(3.9.6)

In accordance with 10CFR50.50a(g), we require submittal of your program for inservice testing of ASME Class 1, 2 and 3 pumps and valves. Our positions on this matter are presented in Section 3.9.6 of the SRP. Appendix C to Section 110 provides a suggested format for this submittal and includes a discussion of the information we require to justify any requests for relief from our positions on this matter.

Response:

In accordance with 10CFR50.55a(g), we have prepared and submitted to the NRC a Pump and Valve Test Program Plan. This program is in agreement with the NRC's positions as presented in Section 3.9.6 of the Standard Review Plan.

REFERENCES:

1. Letter from G.D. Bouchey to A. Schwencer, "Nuclear Project No. 2, Application of SRSS Rule for Steel Containment", dated July 28, 1982.

2. Report, SMA 12109.01-R001, "Study to Demonstrate the Generic Applicability of SRSS Combination of Dynamic Responses for Mark III Nuclear Steam Supply System and Balance-of-Plant Piping and Equipment Components", General Electric Company, dated November 24, 1981.

3. Report, NEDE-24010-P, "Technical Bases for the use of the Square-Root-of-the-Sum-of-Squares (SRSS) Method for Combining Dyamic Loads for Mark II Plants", General Electric Company, dated July 1977.

4. Letter from G. D. Bouchey to A. Schwencer, GO2-82-886, "Nuclear Project No. 2, SRSS Combination of Dynamic Responses", dated November 3, 1982.

5. Letter from G. D. Bouchey to A. Schwencer, GO2-83-090, "Nuclear Project No. 2, SRSS Combination of Dynamic Responses, Confirmatory Item No. 6 of NUREG-0892", dated February 3, 1983.

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B.3 Provide the proposed codes and addenda to be used for repairs, modifications, additions or alterations to the facility which might be implemented during this inspection period.

Response:

a. The WNP-2 Preservice Inspection Program was submitted to the NRC on March 28, 1979 (D. L. Renberger to S. A. Varga, WPPSS Letter No. GO2-79-54). This inspection plan includes exceptions and exemptions to the code that the Supply System is planning to use. Exceptions due to access identified during preservice examinations will be submitted in the final report.

b. The Supply System letter referenced in Part (a) states that the WNP-2 Inservice Inspection Program Plan, which will govern the first 10-year inspection interval, will be submitted to you no later than 6 months following the start of commercial operation.

c. The edition of Section XI of the ASME code being used for the WNP-2 preservice inspection is identified in Sections 1.0 and 4.0 of the WNP-2 Preservice Inspection Program Plan.

d. NRC augmented examination requirements have been addressed in Section 5.0 of the WNP-2 Preservice Inspection Program Plan. The Supply System has changed some of its inspection requirements for the feedwater nozzles based on NRC Question 121.008. The current inspection requirements for feedwater nozzle inner radii can be found in the Supply System response to this question - submitted to the NRC with Round 1 Set 1 responses.

All of the detailed guidelines for the preparation and content of this inspection program as described in Appendix B to Section 121.0 have been met with one exception, that being B.3 which is contained herein.

The repair or modification of N stamped components (subsequent to stamping but prior to the filing of the N-3 Form by Supply System, or prior to the plant commencing commercial operation when no N-3 Form has been filed) will be performed in accordance with 1980 Edition of ASME Section XI with Addenda through Winter 1980 and in accordance with ASME Section III (Code Edition and Addenda to which the component was fabricated).

Deviations to the above referenced Code Edition and Addenda as allowed by Code will be reviewed by the Supply System and authorized on a case by case basis.

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Q. 211.024
(5.4.7)

It is our position for all light-water-reactors that the RHR system shall be capable of bringing the reactor to a cold shutdown condition using only safety-grade systems. Confirm that this requirement is satisfied for the WNP-2 facility. In responding to this request, include a consideration of the capability of the air supply system which is used to operate the RCIC steam and condensate control valves located at the RHR heat exchanger, when the RHR system is in the steam condensing mode.

Response:

All portions of the RHR system required to function in bringing the reactor to a cold shutdown condition are safety grade and redundant except for the shutdown cooling suction line. If this line were unavailable due to a single failure of a suction valve, a safety grade alternate shutdown cooling path can be established through the ADS valves as described in the notes to Figure 15.2-11, Activity Cl or C2.

The steam condensing mode is used only to maintain hot standby condition should the vessel be isolated from the main condenser. Specifically, it allows for maintenance on the turbine generator set without first requiring a cold shutdown of the RPV or continued opening of the main steam relief valves to the suppression pool.

No analysis has been performed which demonstrates that the steam condensing can be used to bring the reactor to a safe, cold shutdown. No credit has been taken for the steam condensing mode in any safety analysis, accordingly, it is permissible to use non-safety air for E12-F051 (RHR-PCV-51) and E12-F065 (RHR-LCV-65). On a loss of air these valves fail-shut, the desired position during accident conditions.

Q. 211.025

It is also our position for all light-water reactors that the RHR system shall be capable of bringing the reactor to a cold shutdown condition with only on-site or off-site power available, assuming the most limiting single failure. In this regard, while we note that Figure 15.2-10 of the FSAR shows a number of available success paths to achieve a cold shutdown condition, vessel depressurization using the RHR system in the steam condensing mode is not shown. (This latter mode is one of the success paths when off-site power is not available.) Either correct this figure or justify this omission. If vessel depressurization were to be achieved by manual actuation of the relief valve, indicate how many valves would have to be actuated. Describe your plans for testing the alternate modes to achieve shutdown cooling. Demonstrate that adequate passage of water through the safety/relief valves can be achieved and maintained when the alternate method is in use. Indicate the quantity of air supplied, its source, and the time interval before the air is exhausted.

Response:

The omission of the steam condensing mode is justified because there is no requirement for the steam condensing mode to be used to bring the reactor to a cold shutdown. Steam condensing is not a safety grade means to depressurize the reactor.

If vessel depressurization were to be achieved by manual actuation of relief valves, three valves would need to be actuated to pass sufficient steam flow to depressurize the vessel.

WNP-2 is a member of the BWR Owners' Group which performed a low pressure liquid flow test to demonstrate the operational adequacy of the safety/relief valves (SRVs) to pass sufficient water flow to meet the requirements of the alternate shutdown cooling mode. The results of this test program are presented in NEDE-24988-P which was transmitted to the NRC by a letter from T. J. Dente (BWR Owners' Group) to D. G. Eisenhut (NRC), dated September 25, 1981. WNP-2 believes that this test program adequately demonstrates the ability to use the SRVs in the alternate shutdown cooling mode and does not plan to perform any additional testing.

Additionally, WNP-2 has performed calculations to demonstrate that adequate passage of water through SRVs in the alternate shutdown cooling mode can be achieved at the WNP-2 plant. The results of these calculations are summarized below.

In the alternate shutdown cooling mode, with one RHR pump in operation, the total system resistance head was calculated to be 550 feet using one SRV valve. Line losses, static head, heat exchanger losses, inlet and outlet losses at the pump, and strainers and losses through the SRV (calculated from experimental data obtained from the B&W Owners' Group tests) were considered in establishing this total system resistance head. At this calculated head, the pump capacity is 4000 gpm and the reactor pressure is 160 psig.

Following normal reactor depressurization (i.e., 100°F/hr.), an alternate shutdown coolant flow rate of 2600 gpm would be required to bring the reactor to a shutdown condition. For WNP-2, this flow capacity can be achieved by using one ADS valve as demonstrated above, although three valves are always available.

The air supply for the ADS valves is discussed in the response to Question 211.048.

Q. 211.027
(5.4.7)

In Section 5.4.7.1.3 of the FSAR, you indicate the specific RHR relief valves and the RHR design pressures used as the basis for providing relief capacity. Expand your discussion by indicating the relief valve capacity, the nominal setpoints, the setpoint tolerance, and the ASME class designation of these valves and lines. In addition, discuss the vulnerability of the RHR system to malfunctions which could result in overpressurization of low pressure piping. Support your evaluation by providing an outline of all operating procedures required to bring the plant to a cold shutdown condition from hot standby and the procedures for plant startup from cold shutdown.

Response:

The relief valves protecting the RHR system are listed below (Reference Figures 5.4-13a and 5.4-13b):

Relief Valve	Nominal Setpoint/Capacity	Location	Piping Design Pressure
F088	125 psig/10 gpm	RHR pump suction from suppression pool	125 psig (Loop C) 220 psig (Loops A and B)
F005	220 psig/25 gpm	RHR pump suction from recirc pipe	220 psig
F025	500 psig/25 gpm	RHR discharge	500 psig
F030	125 psig/10 gpm	RHR flush line to radwaste	125 psig
F036	75 psig/1750 gpm	RHR HX condensate to suppression pool or RCIC pump suction	125 psig
F055	500 psig/330,000 lb/hr	Steam supply to RHR heat exchanger	500 psig

Relief Valve	Nominal Setpoint/Capacity	Location	Piping Design Pressure
RHR-RV-95*	500 psig/300,000 lb/hr.	Steam supply to RHR heat exchanger	500 psig

*RHR-RV-95 is currently not shown in Figures 5.4-13a and 5.4-13b but is shown on Figure 3.2-6, Zones E-13 and E-4.

All RHR relief valves are purchased to ASME Section III, Class 2 requirements to match the requirements of the piping they are protecting. As such, the setpoint tolerance is + 3%, per ASME Section III, Paragraph NC-7614.2.

The RHR system is connected to higher pressure piping at: (1) shutdown suction; (2) shutdown return; (3) LPCI injection: (4) head spray: and (5) heat exchanger steam supply. The vulnerability to overpressurization of each location is discussed in the following paragraphs.

Shutdown sucion has two gate valves (F008 and F009) in series which have independent pressure interlocks to prevent opening at high inboard pressure (135 psig reactor pressure). No single active failure or operator error will result in overpressurization of the lower pressure piping. With the RHR pumps normally lined up to the suppression pool (F006 closed), the shutdown cooling suction line is protected for thermal expansion or from leakage past F008 by F005. With all the RHR suction valves closed, the suction piping is protected for thermal expansion or leakage past the discharge check valves by F088.

The shutdown return line has a swing check valve (F050) to protect it from higher vessel pressures. Additionally, a globe valve (F053) is locate in series and has a pressure interlock to prevent opening at high inboard pressures (135 psig reactor pressure). No single active failure or operator error will result in overpressurization of the lower pressure piping.

The LPCI injection line has an air testable swing check valve (F051) to protect it from higher vessel pressures. The air operator on the testable check valve is only capable of opening the testable check valve if the differential pressure is less than 2.0 psid. Additionally,

a gate valve (F042) is located in series and has pressure interlocks to prevent opening at high differential pressure (nominally 750 psid). No single active failure or operator error will result in overpressurization of the lower pressure piping.

The head spray piping has three swing check valves in series (two belonging to the RCIC system and one (F019) belonging to the RHR system), to protect it from higher vessel pressures. Two of the swing check valves have air operators but they are only capable of opening the testable check valve if the differential pressure is less than 2.0 psid. Additionally, a globe valve (F023) is located in series and has a pressure interlock to prevent opening at high inboard pressures (135 psig reactor pressure). No single active failure or operator error will result in the overpressurization of the lower pressure piping.

Overpressurization protection of the RHR discharge piping for thermal expansion or from leakage past the head spray, shutdown injection, and LPCI isolation valves is provided by F025.

The heat exchanger steam supply line has a globe valve (F052) for shutoff. The operator admits steam through F052 and sets the pressure regulating valve (F051) to limit heat exchanger pressure to about 200 psig. Also, F087 can be opened when the steam supply pressure is below the pressure interlock (500 psig) to provide additional steam flow rate to the heat exchangers. Two relief valves (F055) and RHR-RV-95 with a combined capacity of 660,000 lbs/hr are provided downstream of F051 and F087 to protect the low pressure piping should F051 fail open. The maximum calculated steam flow rate (sonic flow) with F051 and F052 failed open is 600,000 lbs/hr, so there is adequate relief valve capacity to handle this failure. The Class 1E lear detection system, which monitors steam flow rate to the RHR heat exchangers, will isolate the steam supply (close F076, F063 and F064 per Figure 5.4-9a) when the steam flow reaches approximately 360,000 lbs/hr (175% decay heat steam generation rate 1/2 hour after scram). No single active failure nor operator error will cause overpressurization of the lower pressure piping.

During steam condensing mode; with the RHR heat exchanger at 200 psig, the condensate is dumped to either the suppression pool or the RCIC pump suction. F036 provides protection to this low pressure piping should both level control valves F065A and F065B fail open.

F030 protects the drain piping from the RHR system to radwaste from thermal expansion or from leakage past the isolation valves F071 and F072.

OUTLINE OF OPERATING PROCEDURE
AND RHR OVERPRESSURIZATION SAFEGUARDS

1. Plant Shutdown to Cold Shutdown from Hot Standby* With Safety Grade Systems

Reactor Condition	Operating Mode Used	RHR Overpressurization Safeguard
Depressurization from hot standby to 135 psig the suppression pool depressurizes vessel	o Main steam relief valve discharge to	RHR isolated.
	o Initiate and operate pool cooling mode of RHR system	Low pressure mode, no safeguard required.
Cooldown from 135 psig to cold shutdown	o Initiate and operate shutdown cooling mode of RHR	Redundant pressure interlocks on F008 and F009 close valve above pressure interlock setpoint.

2. Plant Startup from Cold Shutdown

Reactor Condition	Operating Mode Used	RHR Overpressurization Safeguard
Reactor coolant RPV head replaced valves above pressure interlock setpoint.	o Terminate shut- and isolate RHR	Redundant pressure and F009 close
Remainder of startup	o Standard	RHR isolated.

* Normally, the main condenser is the heat sink during hot standby, but, because of larger RHR interface, it is assumed that the main condenser is unavailable.

1. Crack occurs in the RHR line, water level decreases to reactor vessel level 3, then the RHR isolation commences and is completed 40 seconds later.

2. System pressure rises as a result of the isolation to where the vessel pressure reaches the SRV setpoint thus causing them to open, blowdown, and reclose.

3. Inventory depletion results from blowdown and from leakage out of the cracked line.

4. The operator manually actuates ADS to reduce vessel pressure to where the low pressure ECCSs can replenish the water inventory.

5. Water level is restored to within normal limits to protect the core from over temperature.

Results are presented in Figures 211.031-1 through 211.031-4 for a bounding calculation of this event. The standard Appendix K assumptions were used along with these conservative initial conditions:

1. The timing index was started at the RHR isolation (when level 3 was attained) to neglect the time for the level to fall from normal water level to level 3 (about 2 minutes).

2. An initial pressure of 1055 psia was assumed to neglect the pressure rise time from the 150 psia (pressure permissive for shutdown cooling) upon completion of RHR isolation to the 1055 pressure attainment. This results in increased mass loss during the 40-second isolation period due to greater driving pressure. It also decreases the time increment needed for pressure to attain the relief valve setpoint.

3. The analysis assumes that scram occurs coincident with the start of the timing instead of 4 hours earlier. This assumption maximizes the peak clad temperature and steam production during the transient thus driving more fluid from the vessel and prolonging the blowdown phase.

4. Only one LPCS and one LPCI loop were assumed to be available throughout the event. Operator action does not include possible diversion of the other two LPCI loops from the RHR mode.

5. The crack area used in the analysis is defined consistently with the MEB 3-1 guidance for crack size. This crack area is consistent with FSAR postulates.

Results from this conservative analysis are that more than 20 minutes are available for the operator to depressurize the vessel. Once the system pressure is below the LPCI or LPCS shutoff head, the reactor water level is restored to normal limits very rapidly. The maximum clad temperature is much less than the arbitrary 2200°F limitation.

b. The RHR system is a low pressure system, and all of the piping outside of the primary coolant pressure boundary is classified as "moderate energy" piping and, according to BTP MEB 3-1, only cracks (i.e., not breaks) are considered in moderate energy piping. Reactor vessel pressure must be decreased to below 135 psig before the RHR system can be connectd to the reactor vessel.

The maximum discharge resulting from the largest crack in the RHR piping outside containment is determined using the guidelines in BTP MEB 3-1 for moderate energy piping. The maximum discharge rate is estimated to be 1000 gpm (to be confirmed later in the ongoing pipe break and missile study). This is based upon a pipe break in the pump discharge piping (18" Schedule 30) at the pump discharge flange, normal water level in the reactor during shutdown cooling (approximately 50 inches below the steam line nozzles), reactor pressure of 135 psig, and the RHR pump running at 7450 gpm (normal shutdown flow). The flow rate used in the LaSalle analysis referenced in part (a) of this response was 1443 gpm. See (c) below for the time interval available for recovery.

c. The following alarms are available to the operator in the event of a pipe break in the shutdown cooling line outside containment.

Q. 211.039
(5.4.7)

Operation of the RHR system in the steam condensing mode involves partial draining of one or both RHR heat exchangers and introduction of reactor steam into lines and heat exchangers which are initially cold. Describe the methods (e.g., valve operation or air introduction) and the provisions you propose to prevent the occurrence of water hammer during initiation of operation in this mode and in the change to the pool cooling mode. Indicate whether the jocket pump system shown in Figure 5.4-13a of the FSAR can fill the lines to the injection valve in the core spray lines and the RHR lines (i.e., valves F016 and F042, respectively) when the RHR is in the steam condensing mode using one or both heat exchangers. If not, indicate what procedure you propose to prevent water hammer following startup of the core spray or RHR pumps.

Response:

Refer to Figure 5.4-13a for valve numbers. The methods used to prevent the occurrence of water hammer during steam condensing initiation are:

a. lowering the heat exchanger water level using low pressure steam (approximately 10 psig) by cracking open steam pressure control valve bypass valve F087;

b. initially admitting steam at a low pressure and slowly increasing steam pressure to 200 psig to avoid high pressure surges; and

c. opening all valves slowly to avoid sudden flow surges.

The methods used to prevent the occurence of water hammer following steam condensing termination and change to the pool cooling mode are:

a. closing the heat exchanger condensate discharge;

b. opening the valves connecting the heat exchanger to the main pump loop (F003 and F047); and

c. opening the high point vent and filling the heat exchanger shell and connecting piping using the condensate supply valve.

When the RHR system is used for steam condensing, the LPCI injection loop is isolated from the heat exchanger steam flow by closing F003 and F047. Use of steam condensing mode has no effect on the jockey pumps' ability to fill the lines to the injection valves in the core spray or RHR lines because the heat exchanger bypass valve F048 is open. Therefore, the jockey pumps can fill these lines.

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Q. 211.040
(5.4.7)

Those pressure relief valves and lines which are designed to prevent overpressurization of the RHR system, are routed outside the containment before being returned to the suppression pool. Discuss the design provisions incorporated into the WNP-2 facility to minimize the potential for water hammer in these lines. State whether these relief lines are capable of withstanding both seismic and dynamic blowdown loads without suffering a loss of structural integrity.

Response:

Except where noted below, the RHR relief valves are installed to accommodate thermal expansion and leakage across closed valves in isolated piping systems (see response to Question 211.027 for additional information on RHR relief valves). Pressure buildups in isolated lines will be slow and discharges from the relief valves in these lines will be small. Water hammer and other hydrodynamic loads are not considered a potential problem in those lines.

RHR-RV-55A and B and RHR-RV-95A and B (reference Figure 3.2-6, zones E, 4 and E, 14) are steam relief valves which protect the RHR heat exchangers from overpressure in case RHR-PCV-51A and B fail during the RHR steam condensing mode. There is no potential for water hammer in the discharge line of RHR-RV-95A and B, which have their own discharge line into the suppression pool. Since the discharge lines for RHR-RV-55A and for RHR-RV55B share a common pipe with several other RHR lines which could fill the discharge lines with water during other modes of RHR operation, e.g., system test, an automatic vacuum breaker is being added to ensure that the water level in these discharge lines is at the suppression pool water level during the steam condensing mode.

In addition, these steam relief valves have an automatic drain pot to prevent any water from accumulating ahead of the valves.

RHR-RV-36 (Figure 3.2-6, zone G, 13) is a water relief valve which protects the lower pressure rated PCIC suction piping in case of either or both RHR-LCV-65A and RHR-LCV-65B failing open during the steam condensing mode. The discharge line for RHR-RV-36 uses the same pipe as RHR-RV-55A, where an automatic vacuum breaker guarantees that there is no water in the pipe.

It should be noted that the probability of the RHR steam relief valves or RHR-RV-36 actuating is extremely low. These relief valves can actuate only during the RHR steam condensing mode which is expected to be used only eight hours per year. In addition, RHR-PCV-51A and B and RHR-LCV-65A and B are designed to fail closed.

RHR relief lines (identified by their value tag numbers RHR-RV-36, RHR-RV-55A, RHR-RV-55B, RHR-RV-95A, and RHR-RV-95B) are capable of withstanding both seismic and dynamic blowdown loads without suffering a loss of structural integrity.

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Q. 211.075
(6.3)

Indicate the provisions incorporated in the WNP-2 facility to protect the water level instrumentation for the CST and the lines from this tank leading to the HPCS systems from the effects of cold weather and dust storms. In responding to this item, cross-reference your responses to Items 010.16 and 211.22.

Response:

The water level instrumentation for the CST and the lines from this tank leading to the HPCS system are totally protected from the effects of cold weather and dust storms. The lines are electrically heat traced and a Seismic Category I enclosure has been provided for all tubing and instrumentation. All level instrumentation shall be NEMA type 4 rated (watertight and dust-tight indoor and outdoor).

See also the response to Question 211.012. The response to Question 010.016 does not address the concerns of this question. The safety-related instrumentation necessary for switchover of HPCS and RCIC pump suction from the CST is located indoors and, as such, is not directly affected by cold weather or dust storms.

Q. 211.076
(6.3)

Some of the ECCS relief valve discharge lines penetrate primary containment and have outlets below the surface of the suppression pool. Since these lines are part of the primary containment boundary, we are concerned that excessive dynamic loads resulting from water hammer during actuation of the relief valves may cause cracking or rupture of these lines. Accordingly, identify these lines which penetrate the primary containment. Provide information concerning the measures you are taking to prevent line damage due to water hammer.

Response:

The ECCS relief valves shown on Table 211.076-1 have discharge lines which penetrate the primary containment and have discharges below the suppression pool water level (Reference Figures 5.4-13a, 5.4-13b, 6.3-1, 6.3-5).

All relief valves shown on this Table Section are purchased on ASME III, Class 2 requirements to match the requirements of the piping they are protecting. As such, the setpoint tolerance is ±3%, per ASME, Section III, Paragrpah NC-7614.2.

For discussion on dynamic lods resulting from water hammer for RHR-RV-55(A, B) (E12-F055A, B), RHR-RV-95(A, B), and RHR-RV-36 (E12-F036) see response to Question 211.040. The remaining relief valves are installed to accommodate thermal expansion and leakage across closed valves in isolated piping systems. Pressure buildups in isolated lines will be slow and discharges from the relief valves in these lines will be small. Water hammer and other hydrodynamic loads are not considered a potential problem in these lines.

Table 211.076-1

Relief Valve	Setpoint/Capacity	Location	Piping Design Pressure
E21-F018	550 psig/100 gpm	LPCS Discharge Leg Relief	550 psig
E21-F031	100 psig/ 10 gpm	LPCS Suction Leg Relief	100 psig
E22-F035	1575 psig/25 gpm	HPCS Discharge Leg Relief	1575 psig
E22-F014	100 psig/ 10 gpm	HPCS Suction Leg Relief	100 psig
E12-F025(A,B,C)	500 psig/ 25 gpm	RHR Discharge Leg Relief	500 psig
E12-F088(A,B,C)	125 psig/ 10 gpm	RHR Suppression Pool Suction Relief	220 psig - A,B 125 psig - C
E12-F005	220 psig/ 25 gpm	RHR Shutdownn Cooling Suction Relief	220 psig
E12-F030	125 psig/ 10 gpm	RHR Flush Line Relief	125 psig
E12-F055(A,B)	500 psig/330,000 lb/hr	RHR Heat Exchanger Steam Relief	500 psig
RHR-RV-95(A,B)*	500 psig/330,000 lb/hr	RHR Heat Exchanger Steam Relief	500 psig
RHR-RV-1(A,B)**	500 psig/ 20 gpm	RHR Heat Exchanger Thermal Relief	500 psig
E12-F036	75 psig/1750 gpm	RHR Heat Exchanger Condensate Relief	125 psig

* RHR-RV-95A,B are not currently shown on Figures 5.4-13a and 5.4-13b, but are shown on Figure 3.2-6, Zones E,H and E,13.

** RHR-RV-1A,B are shown on Figures 5.4-13a and 5.4-13b (thermal relief valve on heat exchangers RHR-HX-1A,B) but are not designated by tag number.

Q. 211.143
(5.4.6.4)

Show how the preoperational initial startup test programs for the RCIC system in Section 14.2.12.1.8 meet the intent of applicable sections in Regulatory Guide 1.68.

Response:

The applicable sections of Regulatory Guide 1.68 which delineate requirements for tests of RCIC include sections 1.d (5) and (6); 1.j (19); 4.k and q; 5.1, dd and mm of Appendix A.

The specific areas of concern that these sections address are, respectively: verification of operability and design features of the RCIC system and the RHR/RCIC system interface in the steam condensing mode during the preoperational phase of the WNP-2 initial startup test program; operability and design verification of the RCIC control instrumentation on the remote shutdown panel again during the preop program; demonstration of RCIC and RHR steam condensing mode operability during low power operation when sufficient steam exists to utilize these plant design features; and finally, to demonstrate the design capability of RCIC during major plant transients such as the remote shutdown capability demonstration and the main steam line isolation valve (MSIV) full isolation test.

The WNP-2 initial startup test program provides for extensive tests in each of these areas. Sections 14.2.12.1.8, 14.2.12.1.26, 14.2.12.3.14, 14.2.12.3.25, 14.2.12.3.28, and 14.2.12.3.37 briefly describe, in general terms, the tests which will be performed to provide assurance that the RCIC system is fully operational in each of its modes or conditions in which it is expected to perform. Specifically, during the preop phase such RCIC component tests as valve operability, initiation/interlock/trip logic checks, flow path verification, control and instrumentation calibration, and pump/turbine vibration measurements are conducted. In addition, the control and instrumentation calibration on the remote shutdown panel and the system interface with RHR in the steam condensing mode are checked for proper operation. During low power operation the ability of the RCIC system to initiate, then deliver, rated flow within 30 seconds is demonstrated at three points within the range of 150 psig to rated reactor pressure. Also, following tune-up of the RHR heat exchanger level and inlet pressure controllers, the adequacy of the RCIC control system is confirmed when the system is coupled with the RHR system in the steam condensing mode. The final confirmation of proper RCIC system performance is achieved by challenging the system to perform during anticipated tran-

sients. The ability of RCIC to maintain reactor water level when controlled from the remote shutdown panel is demonstrated by actual testing. The ability of the system to meet its primary design function is demonstrated during the MSIV full isolation test when it is the main source of water for maintenance of vessel inventory.

The combination of component tests during the preop phase and the control system tune-up/overall operability demonstrations during the power ascension phase of the startup test program satisfy the requirement of Regulatory Guide 1.68.

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Q. 211.144
(5.4.6)

The ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000 requires that individual pressure relief devices be installed to protect lines and components that can be isolated from normal system overpressurization protection. With reference to appropriate P&ID, identify those portions of the RCIC system that can be isolated from normal system overpressure protection. Discuss the relief devices provided or provide the basis for deciding that relief devices are not required.

Response:

Referring to Figures 5.4-9a and 5.4-9b, there are five RCIC pipe lines that have a low design pressure and, therefore, require relief devices or some other basis for addressing overpressure protection. They are:

- RCIC Pump Suction Line
- RCIC Turbine Exhaust Line
- RCIC Steam Condensing Supply Line Downstream of F064
- Portions of the RCIC Minimum Flow Line Downstream of F019
- Portions of the RCIC Cooling Water Line Downstream of PCV-F015

The design pressure of the other major pipe lines is equal to the vessel design pressure and subject to the normal overpressure protection system. Below are the overpressure protection bases for the low pressure piping lines.

a. RCIC Pump Suction Line

A relief valve (F017) is located on the pump suction line on Figure 5.4-9b to accommodate any potential leakage through the isolation valves (F013 and F066). A high pump suction pressure alarm is provided in the control room. Also, the pump suction pipe is protected from overpressurization from the RHR system during steam condensing mode by F036 (Figure 5.4-13a) should both the RHR heat exchanger level control valves F065A and F065B (Figure 5.4-13a)

fail open while dumping condensate to the RCIC pump suction.

b. RCIC Turbine Exhaust Line

This line is normally vented to the suppression pool and is not subject to reactor pressure during normal operation. Rupture discs D001 and D002, as shown on Figure 5.4-9b, are installed on this line to prevent exceeding piping design pressure should the exhaust line isolation valve F068 be closed when the RCIC turbine is operating. The RCIC system will automatically isolate if the rupture discs were to blow open.

c. RCIC Steam Condensing Supply Line Downstream of F064

In the steam condensing mode, high pressure steam is routed to the RHR heat exchangers via F064. The RHR piping is protected from overpressurization by relief valves F055 and F095 as discussed in Question 271.027.

d. Portions of the RCIC Minimum Flow Line Downstream of F019

This line is normally vented to the suppression pool and is separated from reactor pressure by the pump discharge isolation valves (F013, F065, and F066) and one additional normally closed isolation valve in the minimum flow line (F019) as shown on Figure 5.4-9a.

e. Portions of the RCIC Cooling Water Line Downstream of PCV-F015

In the standby condition this line is separated from reactor pressure by the pump discharge valves (F013, F065 and F066) and one additional normally closed shut-off valve in the cooling water line (F046) as shown on Figure 5.4-9b. During system operation a relief valve (F018) is provided to prevent overpressurizing piping, valves, and equipment in the coolant loop in the event of failure of pressure control valve PCV-F015 as shown on Figure 5.4-9b.

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Q. 211.199
(6.3)

Discuss the design provisions that permit manual override on the ECCS subsystems once they have received an ECCS initiation signal. Also, include a discussion of any lockout devices or timers that prevent the operator from prematurely terminating ECCS functions. If there are plant procedures to cover this situation, indicate briefly what instructions are provided.

Response:

Each ECCS subsystem (LPCI, LPCS, HPCS, ADS) is provided with manual override logic which allows the operator to terminate or delay automatically initiated core cooling functions by closing the injection valve or stopping the pump or delaying system actuation. This is necessary in case an ECCS has failed or requires isolation to protect suppression pool inventory, or whose function is no longer needed when other core cooling functions are successful (thus reducing the long-term load on diesel generators). Other manual overrides are provided to allow the operator to terminate the core cooling function of a system, such as RHR, allowing the system to be utilized in other modes of post-accident operation (e.g., suppression pool cooling, containment spray).

The plant operators are instructed to use the manual override controls only when the core cooling function has been or will be successful. For example, the operator will not terminate the LPCS unless assured by at least two independent reactor vessel water level indications that water level is restored. The high drywell pressure initiation signal may still be above the trip point. For the case of the ADS, the operator can delay initiation indefinitely (105 seconds at a time). However, this will only be done when the operator is assured that reactor vessel water level is being restored by the HPCS, again by consulting at least two independent water level indications. The operator will terminate the LPCI mode of RHR to enter another mode of RHR only after consulting the water level indication and the availability of other core cooling systems.

There is only one time delay/lockout which prevents the operator from prematurely terminating an ECCS function. That is the 10-minute timer within the LPCI logic which prevents the operator from moving the RHR heat exchanger bypass valves E12-F048A/B from their full open position.

For all ECCS manual overrides (except ADS), automatic system actuation will not reoccur unless the initial initiation

signals (high drywell pressure, low water level) return to normal and the logic is reset. The ADS logic will automatically reset when the 105 second time delay is complete.

f. A copy of the applicable response spectra is attached to each summary form in our QID file. They will be transmitted upon request. See response to Question 271.006.

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Q. 271.006

To confirm the extent to which the safety-related equipment of the WNP-2 facility meets the requirements of GDC 2 and 4, the Seismic Qualification Review Team (SQRT) will conduct a plant site review. For selected equipment, SQRT will review the combined required response spectra (RRS) or the combined dynamic response and examine the equipment configuration and mounting. On this basis, SQRT will then determine whether the test or analysis which has been conducted demonstrates compliance with the RRS if the equipment was qualified by test or the acceptable analytical criteria if qualified by analysis.

The staff requires that a "Qualification Summary of Equipment" as shown on the attached pages, be prepared for each selected piece of equipment and submitted to the staff two weeks prior to the plant site visit. In this regard, you should make available at the plant site for SQRT reveiw, all the equipment. After the visit, you should then be prepared to submit certain selected documents and reports for further staff review.

Response:

The "Qualification Summary of Equipment" forms are available in our QID file. We will transmit those you select at your request.

Q. 271.011

Indicate that the "accuracy" information missing from the summary sheets, Appendix C, as well as other pertinent information, will be available at time of audit.

Response:

Instrumentation accuracy is being obtained from specification and qualification data prepared by WNP-2 suppliers and designers for use with Code 1X, Levels 1 and 2 equipment (reference Appendix A of the Environmental Qualification Report). Some of this information will be available during the environmental audit. The summary sheets will be updated to include this data prior to fuel load.

Q. 271.012

Indicate that the effects of beta radiation have been included in the qualification program.

Response:

The WNP-2 qualification program does consider the effects of beta radiation. There are three types of equipment within the primary containment that need to be analyzed to determine their susceptibility to long-term beta effects. These are:

a. Electrical junction box components and wiring;

b. Air-cooled motors;

c. Some exposed cabling beneath the reactor pressure vessel.

The remaining equipment within the containment is adequately protected from beta effects. The results of the analysis for the equipment listed above will determine if corrective action is needed to protect this equipment.

Q. 271.015

Indicate that safety equipment located inside primary containment has been qualified to the temperature/pressure profile described in Table 3.11-2 of the FSAR or provide justification.

Response:

The safety-related equipment has been qualified to the first 24-hour period into the accident conditions depicted by Table 3.11-2. This equipment is also qualified to the post-accident conditions defined by Profile 1 of Appendix B in the Environmental Qualification Report. A revision to this profile has been made to include the 24-hour conditions of Table 3.11-2 superimposed on the plant-specific conditions. This composite identifies the margin inherent in the Table 3.11-2 generic profile and will be issued in a revision to the Environmental Qualification Report.

Q. 271.016

Before the Safety Related Mechanical (SRM) equipment audit items can be selected, the applicant must provide a statement that all SRM equipment in a harsh environment is included in the mechanical equipment qualification program and must indicate the qualification status of the SRM equipment. If qualification is not complete, briefly describe the tasks to be performed. Provide a list of SRM equipment which is considered qualified from which audit items may be selected. Your review of equipment should be essentially complete before items are selected. The staff review will concentrate on materials which are sensitive to environmental effects, for example, seals, gaskets, lubricants, fluids for hydraulic systems, diaphragms.

Response:

The Environmental Qualification Report (September 1982) detailed the Supply System's reevaluation program for Environmental Qualification of Safety-Related Mechanical Equipment. This reevaluation program of the harsh environmental effects on Safety-Related Mechanical (SRM) equipment has been completed, and a detailed list of evaluated items was contained in Enclosure 5 to letter G02-83-81, G. D. Bouchey to A. Schwencer, "Qualification of Safety-Related Electrical Equipment, NRC Request for Additional Information," dated January 31, 1983. All items are qualified with these exceptions:

MSLC-FN-1
SGT-FN-1A1, 1A2, 1B1, 1B2
CEP-V-3A, 3B, 4A, 4B
CSP-V-6
CSP-AO-6, 9

Corrective action for non-qualified items has been defined and is being implemented.

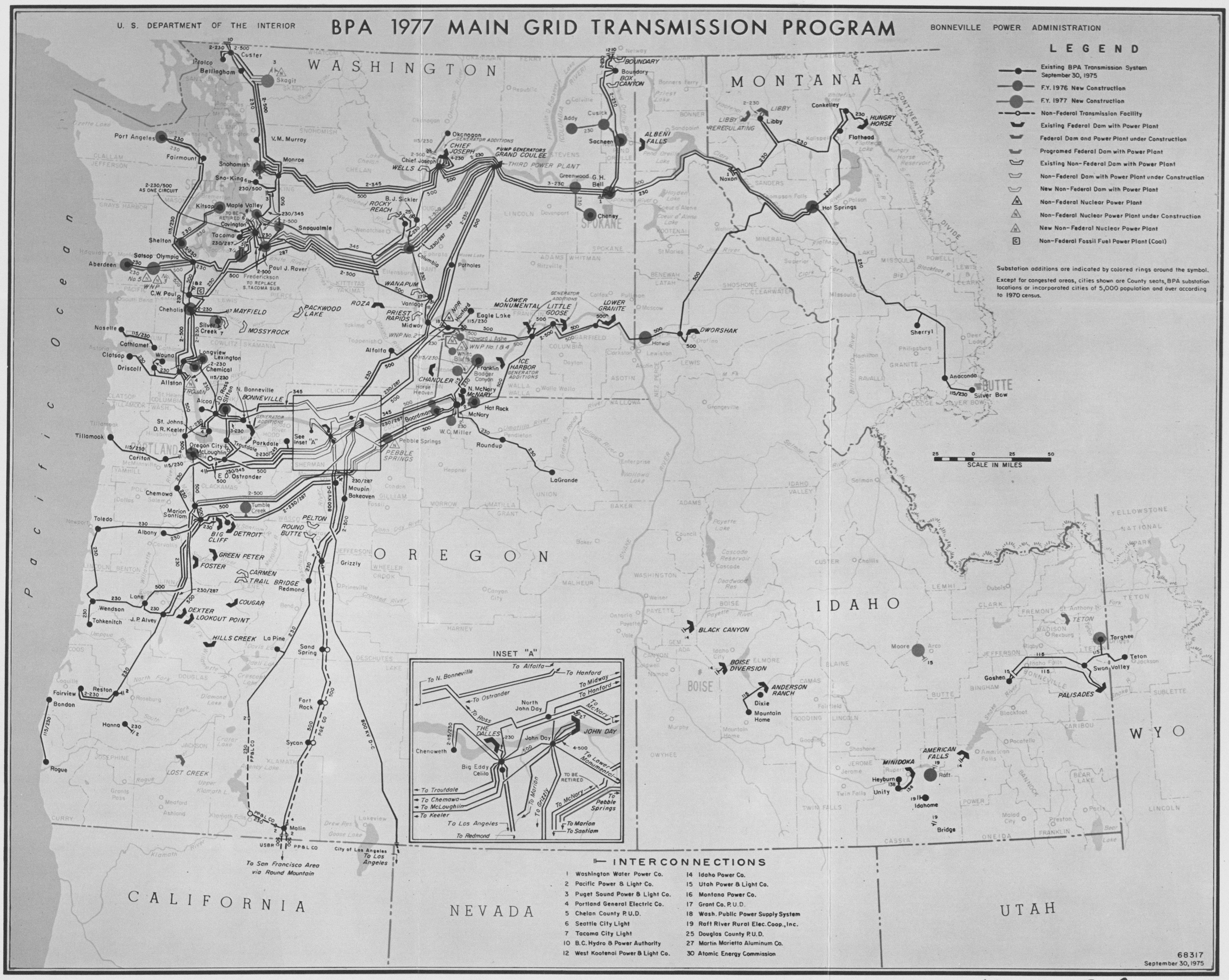

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FIGURE 8.1-1