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Docket No. 50-397

May 18, 1981
G02-81-100

Director,
Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Gentlemen:

Subject: SUPPLY SYSTEM NUCLEAR PROJECT NO. 2
FSAR AMENDMENT NO. 14

The Washington Public Power Supply System herewith submits sixty (60) copies of Amendment 14 to its Final Safety Analysis Report.

Pursuant to 10CFR2.101, we will, within ten (10) days of filing, furnish to you an affidavit reflecting our distribution of this amendment to your designated distribution list.

Very truly yours,

G D Bouchey

G. D. Bouchey
Director, Nuclear Safety

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Subject: WNP-2 FSAR Amendment 14

I, G. D. BOUCHEY, being duly sworn, subscribe to and say that I am the Director, Nuclear Safety, 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.

DATED 5/13/81, 1981

G. D. Bouchey
G. D. BOUCHEY

On this day personally appeared before me G. D. BOUCHEY 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 therein mentioned.

GIVEN under my hand and seal this 14 day of May, 1981.

M. Tolley Utecht
Notary Public in and for the
State of Washington

Residing at Richland, Wa

50-397
Superseded pgs per
Amdt 14 to FSAR
dated 5/18/81
8106010544

GP

LIST OF EFFECTIVE PAGES

Pages not stated below are original issue:

CHAPTER 1

TEXT PAGES	AMENDMENT
Chapter Index	7
1-viii	7
1-ix	10
1.1-2	10
1.2-20 and 1.2-21	4
1.2-41 and 1.2-42	12
1.3-27	1
1.4-4 and 1.4-5	12
1.5-3	13
1.5-5	13
1.5-7 through 1.5-11	13
1.6-2	13
1.6-8	5
1.6-9	4
1.6-11	13
1.7-1 through 1.7-24	11
1.8-1 and 1.8-2	10
1.8-3 and 1.8-4	13
1.8-5	10
1.8-6 through 1.8-8	13

FIGURES	
1.2-1a and 1.2-1b	7
1.2-2	8
1.2-3 through 1.2-14	7
1.2-16	11

CHAPTER 2

TEXT PAGES	
2-iii	12
2-iv	5
2-xiv	7
2-xv and 2-xva	12
2-xvi and 2-xvii	12
2-xix	13
2-xxii	12
2-xxiv and 2-xxiva	13
2-xxv	13
2-xxvi	1

CHAPTER 2 (Continued)

TEXT PAGES	AMENDMENT
2.1-1 and 2.1-1a	12
2.1-2 and 2.1-2a	12
2.1-7	12
2.1-9	12
2.2-1	12
2.2-2 and 2.2-2a	12
2.2-3 through 2.2-12	12
2.2-13 through 2.2-15	7
2.3-18 and 2.3-19	5
2.3-21	5
2.3-23	5
2.3-33	5
2.3-36	5
2.3-37	5
2.3-46	2
2.3-47 through 2.3-48a	7
2.4-1	13
2.4-2	5
2.4-4 through 2.4-11	13
2.4-12	5
2.4-13 through 2.4-20	13
2.4-24 through 2.4-29	13
2.4-30 and 2.4-30a	13
2.4-31 through 2.4-34	13
2.4-35	4
2.4-36	4
2.4-39 through 2.4-42	13
2.4-44 through 2.4-53	13
2.5-136	13
2.5-144	5
2.5-157	5
(2.5F)14	5
(2.5F)24	5
(2.5F)28	5
2.5H-2	3

FIGURES	
2.1-3	12
2.1-4	7
2.2-3	12
2.2-4 and 2.2-5	7
2.3-1	10
2.4-3	5
2.4-9	5
2.4-10	13

CHAPTER 2 (Continued)

TEXT PAGES	AMENDMENT
2.4-11 and 2.4-12	5
2.4-13	13
2.4-20 through 2.4-23	13
2.4-26	13
2.5H-8b	3
2.5H-10b	3
2.5H-12b	3
2.5H-14b	3
2.5H-16b	3
2.5H-16c	3
2.5H-16d	3

CHAPTER 3

TEXT PAGES	
3-ii	12
3-viii	8
3-ix	5
3-x and 3-xa	12
3-xii	9
3-xiia	9
3-xv, 3-xva and 3-xvb	9
3-xix	9
3-xx and 3-xxi	8
3-xxii	8
3-xxiia	1
3-xxiv	12
3-xxvi and 3-xxvia	8
3-xxix	8
3-xxx and 3-xxxa	9
3-xxxviii	4
3-xLi	9
3-xLviii	9
3-L and 3-La	9
3-Lviii	12
3-Lxi and 3-Lxii	12
3-Lxiii	5
3-Lxiiia	4
3-Lxiv and 3-Lxiva	9
3-Lxv and 3-Lxva	9
3-Lxvi	2
3-Lxxiii through 3-Lxxiiic	9
3-Lxxv	8
3-Lxxix	8
3-Lxxixa	13

CHAPTER 3 (Continued)

CHAPTER 3 (Continued)

TEXT PAGES	AMENDMENT
3.6-27	9
3.6-28 and 3.6-28a	9
3.6-29	9
3.6-30 and 3.6-30a	9
3.6-31	9
3.6-33 and 3.6-33a	9
3.6-42	9
3.6-48	9
3.6-57 and 3.7-57a	9
3.6-60 through 3.6-63	9
3.6-64 and 3.6-64a	9
3.6-65 through 3.6-67	9
3.6-69 through 3.6-71	9
3.6-72 and 3.6-72a	9
3.6-73	9
3.6-74	1
3.6-75	12
3.6-75a	9
3.6-76	9
3.6-88 through 3.6-100	9
3.7-4	12
3.7-5	8
3.7-12	8
3.7-13 and 3.7-13a	8
3.7-14	8
3.7-15	8
3.7-15a	9
3.7-16	8
3.7-17	8
3.7-18	7
3.7-20	1
3.7-23	1
3.7-26	7
3.7-27	8
3.7-28	8
3.7-29	8
3.7-30	8
3.7-31 and 3.7-31a	8
3.7-32	8
3.7-33	8
3.7-34	8
3.7-36 and 3.7-36a	8
3.7-38 through 3.7-38e	8
3.7-38f	1
3.7-39	8
3.7-44	8

CHAPTER 3 (Continued)

TEXT PAGES	AMENDMENT
3.7-45	8
3.8-3	12
3.8-3a and 3.8-3b	1
3.8-6 through 3.8-6c	8
3.8-9	8
3.8-13	3
3.8-13a	3
3.8-14	3
3.8-17	8
3.8-21	9
3.8-42 and 3.8-42a	8
3.8-43	3
3.8-44 through 3.8-44b	3
3.8-45	2
3.8-46 and 3.8-46a	8
3.8-48	2
3.8-54	13
3.8-60 and 3.8-61	1
3.8-62	13
3.8-65	12
3.8-65a and 3.8-65b	1
3.8-66	13
3.8-71	13
3.8-72	13
3.8-74	2
3.8-76	12
3.8-85	13
3.8-86	8
3.8-87	8
3.8-88	13
3.8-88a	8
3.8-89 and 3.8-89a	8
3.8-90 and 3.8-90a	8
3.8-91 and 3.8-91a	8
3.8-92	5
3.8-93	2
3.8-94 through 3.8-96	5
3.8-99	12
3.8-100	1
3.8-103 and 3.8-103a	9
3.8-104	9
3.8-111	5
3.8-112 through 3.8-115	4
3.8-119a	7
3.8-119b	5
3.8-126	9

CHAPTER 3 (Continued)

TEXT PAGES	AMENDMENT
3.8-127	9
3.8-129 and 3.8-129a	9
3.8-130 and 3.8-130a	9
3.8-135	9
3.8-137 and 3.8-137a	12
3.8-137b	1
3.8-141	10
3.8-144 and 3.8-144a	8
3.8-146 and 3.8-146a	12
3.8-147	9
3.8-161	12
3.8-175	12
3.8-176	13
3.8-190	2
3.8-191	12
3.8-194	13
3.8-199	8
3.8-201	9
3.8-202	9
3.8-203	9
3.8-204	9
3.8-205	8
3.9-4	9
3.9-17	9
3.9-22	9
3.9-23	9
3.9-26 through 3.9-29	9
3.9-36	9
3.9-38	9
3.9-39	4
3.9-48	9
3.9-49	9
3.9-51 through 3.9-53	9
3.9-54 and 3.9-54a	9
3.9-59	8
3.9-66	9
3.9-67	9
3.9-69, 3.9-69a and 3.9-69b	9
3.9-70	9
3.9-71	9
3.9-92	4
3.9-155	9
3.9-160 through 3.9-162	4
3.9-163 and 3.9-163a	4
3.9-174	4
3.10-2	9

CHAPTER 3 (Continued)

TEXT PAGES	AMENDMENT
3.10-3	13
3.11-4	11
3.12-21 through 3.12-21b	9
3.12-31	1
3.12-34 and 3.12-35	1
3.12-36	9
3.12-43 through 3.12-45	1
3.12-47	1

FIGURES	
3.2-2 through 3.2-4	11
3.2-5	7
3.2-6 through 3.2-8	11
3.2-11	11
3.2-16	11
3.2-17	7
3.2-20	7
3.2-25	11
3.5-9	3
3.5-41	4
3.5-50	7
3.5-51 and 3.5-52	4
3.6-6c and 3.6-6h	2
3.6-9a	2
3.6-11	9
3.6-12b and 3.6-12c	9
3.6-13b and 3.6-13c	9
3.6-14a through 3.6-14c	9
3.6-15b and 3.6-15c	9
3.6-16b through 3.6-16e	9
3.6-17b through 3.6-17e	9
3.6-18a through 3.6-18g	9
3.6-19a through 3.6-19c	9
3.6-20b	9
3.6-21b	9
3.6-22b	9
3.6-23a and 3.6-23b	9
3.6-24b	9
3.6-25b	9
3.6-26b	9
3.6-27b	9
3.6-28b	9
3.6-29b and 3.6-29c	9
3.6-30b through 3.6-30d	9
3.6-31a through 3.6-31c	9

CHAPTER 3 (Continued)

CHAPTER 4

CHAPTER 4 (Continued)

TEXT PAGES	AMENDMENT
4.4-15	1
4.4-16 and 4.4-17	10
4.4-18	7
4.4-19	12
4.4-20 and 4.4-21	1
4.4-22	7
4.4-23 through 4.4-25	1
4.4-26	12
4.4-27 through 4.4-31	1
4.4-32	13
4.4-33	7
4.4-34 through 4.4-45	1
4.5-2	2
4.6-2	12
4.6-3	13
4.6-21	12
4.6-25 and 4.6-26	12

FIGURES	
4.4-1 through 4.4-4	1
4.4-5	4
4.4-6	1

CHAPTER 5

TEXT PAGES	
5-v	7
5-vi	12
5-vii and 5-viii	5
5-x	12
5-xiii	12
5-xv and 5-xva	12
5-xvii	12
5-xviii	12
5-xix	12
5.2-7	11
5.2-8	11
5.2-11	8
5.2-12	4
5.2-13	13
5.2-29 and 5.2-29a	7
5.2-30 through 5.2-35	7
5.2-36	8
5.2-37	13

CHAPTER 5 (Continued)

TEXT PAGES	AMENDMENT
5.2-39	13
5.2-40	8
5.2-41 and 5.2-42	13
5.2-45 and 5.2-46	13
5.2-47	7
5.2-48	10
5.2-52 and 5.2-52a	5
5.2-60 through 5.2-76	7
5.3-3 through 5.3-13	5
5.3-17 and 5.3-18	13
5.3-23 through 5.3-25	5
5.4-2	13
5.4-3	8
5.4-5	4
5.4-12	12
5.4-14	13
5.4-22 and 5.4-22a	8
5.4-24	8
5.4-25	8
5.4-27	8
5.4-28	8
5.4-38	8
5.4-39	9
5.4-40	13
5.4-41	11
5.4-43	8
5.4-47	13
5.4-48 and 5.4-48a	13
5.4-49 and 5.4-49a	13
5.4-50 and 5.4-50a	12
5.4-54	12
5.4-57 through 5.4-59	4
5.4-60	9

FIGURES

5.1-1	3
5.1-2	3
5.2-4	11
5.2-5	11
5.3-4	5
5.3-5	5
5.4-3a	4
5.4-4a and 5.4-4b	8
5.4-5	4
5.4-14b	3

CHAPTER 5 (Continued)

FIGURES	AMENDMENT
5.4-14c	3
5.4-15a, b, c	5
5.4-16	2
5.4-17c	13
5.4-19a and 5.4-19b	8

CHAPTER 6

TEXT PAGES	
6-iii	3
6-v and 6-va	11
6-vi	5
6-via	10
6-vii and 6-viia	12
6-xi and 6-xii	9
6-xvi	12
6-xvii and 6-xviia	11
6-xviii	9
6-xix	11
6-xx	12
6-xxa	12
6-xxb	11
6-xxvii	12
6.1-5 and 6.1-5a	2
6.1-10 and 6.1-11	2
6.2-2	12
6.2-3	13
6.2-6	3
6.2-18	12
6.2-27	3
6.2-30 and 6.2-30a	3
6.2-31 and 6.2-32	12
6.2-33	5
6.2-33a	13
6.2-33b	8
6.2-33c	13
6.2-35	12
6.2-36	1
6.2-38	12
6.2-45	3
6.2-47	4
6.2-48	11
6.2-49	4
6.2-50 through 6.2-50e	11

CHAPTER 6 (Continued)

CHAPTER 6 (Continued)

TEXT PAGES	AMENDMENT
6.2-143 through 6.2-146	12
6.2-147 through 6.2-152	3
6.2-153 through 6.2-160	8
6.2-161 through 6.2-170	11
6.3-10	13
6.3-11	4
6.3-13	13
6.3-16	4
6.3-19	5
6.3-21	11
6.3-21a	5
6.3-22	13
6.3-27 and 6.3-27a	8
6.3-28	11
6.3-34 through 6.3-36	13
6.3-37	11
6.3-38	9
6.3-39	11
6.3-43	11
6.3-45	11
6.3-49	11
6.4-1	9
6.4-2	9
6.4-3 and 6.4-4	5
6.4-6	9
6.4-6a	11
6.4-7	9
6.4-7a	13
6.4-7b	9
6.4-8	9
6.4-9 through 6.4-9b	9
6.4-9c and 6.4-9d	11
6.4-9e and 6.4-9f	9
6.4-10	13
6.4-10a	9
6.4-12	9
6.5-1 and 6.5-1a	11
6.5-3	11
6.5-6 through 6.5-8	5
6.5-9 and 6.5-9a	13
6.5-12	3
6.5-13	13
6.5-18 and 6.5-19	5
6.5-20	2
6.5-21 and 6.5-22	5
6.5-25	5

CHAPTER 6 (Continued)

TEXT PAGES	AMENDMENT
6.6-1 through 6.6-15	7
6.7-2	5
6.7-4	8
6.7-5	8
6.7-11	5

FIGURES	
6.2-17a through 6.2-17c	3
6.2-21	5
6.2-23	5
6.2-26	5
6.2-30	5
6.2-31a through 6.2-31j	3
6.2-31k	9
6.2-31l and 6.2-31m	3
6.2-31n	13
6.2-31o through 6.2-31t	3
6.2-31u	13
6.2-32 through 6.2-35	3
6.2-36 through 6.2-41	8
6.4-1	9
6.4-2	7

CHAPTER 7

TEXT PAGES	
7-i through 7-xv	10
7.1-1 through 7.1-3	10
7.1-4 and 7.1-5	12
7.1-6 through 7.1-12	10
7.1-13	12
7.1-14 through 7.1-17	10
7.2-1 through 7.2-14	10
7.2-15	12
7.2-16	10
7.2-17	12
7.2-18 through 7.2-21	10
7.2-22 and 7.2-23	12
7.2-24 through 7.2-32	10
7.3-1 through 7.3-12	10
7.3-13	12
7.3-14 through 7.3-18	10

CHAPTER 7 (Continued)

CHAPTER 7 (Continued)

FIGURES	AMENDMENT
7.2-10a and 7.2-10b	10
7.3-1 through 7.3-3	10
7.3-4a through 7.3-4c	10
7.3-5 through 7.3-7	10
7.3-8a through 7.3-8c	10
7.3-9a through 7.3-9c	10
7.3-10a through 7.3-10f	10
7.3-11	10
7.3-12a and 7.3-12b	10
7.3-13a and 7.3-13b	10
7.3-14a through 7.3-14e	10
7.3-15a through 7-3-15g	10
7.3-16a through 7.3-16c	10
7.3-17a through 7.3-17s	10
7.3-18a through 7.3-18k	10
7.3-19a through 7.3-19j	10
7.3-20a through 7.3-20j	10
7.3-21a through 7.3-21d	10
7.6-1a and 7.6-1b	10
7.6-2 and 7.6-3	10
7.6-4a and 7.6-4b	10
7.6-5	10
7.6-6a through 7.6-6g	10
7.6-7 through 7.6-11	10
7.7-1	10
7.7-2a and 7.7-2b	10
7.7-3a through 7.7-3h	10
7.7-4a and 7.7-4b	10
7.7-5a and 7.7-5b	10
7.7-6 through 7.7-15	10

CHAPTER 8

TEXT PAGES	
8-viii	12
8-ix	12
8-xii	12
8-xiii	2
8.2-11	3
8.3-8	8
8.3-13 and 8.3-14	2
8.3-17	13
8.3-23	2
8.3-27	13

CHAPTER 8 (Continued)

TEXT PAGES	AMENDMENT
8.3-29 and 8.3-29a	9
8.3-30	9
8.3-34	12
8.3-37	12
8.3-41	13
8.3-47	13
8.3-49	9
8.3-51 through 8.3-53	1
8.3-65 and 8.3-65a	8
8.3-66	8
8.3-70	2
8.3-72	2
8.3-78	1
8.3-79 and 8.3-80	1
8.3-96	2
8.3-99 through 8.3-103	1
8.3-106 and 8.3-107	1

FIGURES	
8.1-9a through 8.1-9c	11
8.1-9d	7
8.1-10	7
8.3-1a through 8.3-1c	11
8.3-1d and 8.3-1e	7
8.3-1f	11
8.3-2	11
8.3-15	2
8.3-19	11
8.3-24a	5
8.3-24b	2
8.3-24c	5
8.3-29	1

CHAPTER 9

TEXT PAGES	
9-i	5
9-v	12
9-viii and 9-viiia	8
9-xiii	11
9-xiv and 9-xiva	12
9-xvi and 9-xvii	7
9-xviii and 9-xviiia	12
9-xix	7
9-xx	12

CHAPTER 9 (Continued)

TEXT PAGES	AMENDMENT
9-xxi	12
9-xxia	8
9-xxii	12
9-xxiii	11
9.1-3 and 9.1-4	13
9.1-7 through 9.1-9a	5
9.1-12	12
9.1-14	1
9.1-16	5
9.1-22	13
9.1-24	5
9.1-27	5
9.1-29	1
9.1-39	1
9.1-42	13
9.1-54 through 9.1-56	1
9.1-62	13
9.1-63 and 9.1-63a	13
9.1-65	5
9.2-7	2
9.2-9	13
9.2-12	13
9.2-13 and 9.2-13a	13
9.2-15	13
9.2-16 and 9.2-17	5
9.2-18	7
9.2-19 through 9.2-22	5
9.2-26	8
9.2-27 and 9.2-27a	8
9.2-30	13
9.2-31	13
9.2-33	13
9.2-36 through 9.2-39	5
9.2-40 through 9.2-42	11
9.2-44	3
9.2-46 through 9.2-48	5
9.2-49 and 9.2-50	11
9.3-2	11
9.3-4	11
9.3-4a	8
9.3-5	11
9.3-20	5
9.3-23	5
9.4-4 and 9.4-4a	9
9.4-5 and 9.4-5a	9
9.4-6	9

CHAPTER 9 (Continued)

CHAPTER 9 (Continued)

FIGURES	AMENDMENT
9.3-14	4
9.4-1 through 9.4-3	11
9.4-4	7
9.4-5	11
9.4-6	7
9.4-7 through 9.4-12	11
9.5-1	11
9.5-3	11
9.5-4	11

CHAPTER 10

TEXT PAGES	
10-i	12
10-iii	5
10-vi	13
10.1-1	12
10.2-2	7
10.2-4	13
10.2-5	7
10.2-6	13
10.2-6a	7
10.2-7	13
10.2-8 and 10.2-8a	13
10.2-9	13
10.2-10	3
10.2-11	9
10.3-2	13
10.4-2	7
10.4-3	13
10.4-4	7
10.4-4a	13
10.4-5	2
10.4-6	5
10.4-12	7
10.4-14	13
10.4-17 through 10.4-17c	5
10.4-24	13
10.4-25	8
10.4-28	4

CHAPTER 10 (Continued)

FIGURES	AMENDMENT
10.2-10	7
10.3-1	11
10.3-7	11
10.3-8	13
10.4-1	11
10.4-2	7
10.4-3 through 10.4-7	11

CHAPTER 11

TEXT PAGES	
11-viii	12
11-xi and 11-xii	12
11-xiia	8
11.2-1	5
11.3-2	5
11.3-6	13
11.3-14	13
11.3-19	7
11.3-33	7
11.4-16 and 11.4-16a	2
11.5-2	2
11.5-16	5
11.5-18 and 11.5-18a	4
11.5-24	3
11.5-25	3
11.5-26	2

FIGURES	
11.2-2 and 11.2-3	11
11.2-4a through 11.2-4c	11
11.4-1	11
11.5-10	4

CHAPTER 12

TEST PAGES	
12-ii	5
12-ix	5
12-vi and 12-via	12
12-vii	5
12-viii and 12-ix	12

CHAPTER 12 (Continued)

CHAPTER 12 (Continued)

CHAPTER 13

CHAPTER 14

CHAPTER 14 (Continued)

TEXT PAGES	AMENDMENT
14.2-130	12
14.2-132	12
14.2-132a	13
14.2-133	7
14.2-134	13
14.2-135	12
14.2-136 and 14.2-137	13
14.2-138	12
14.2-138a	13
14.2-139	12
14.2-140	13
14.2-141	7
14.2-142 and 14.2-142a	12
14.2-143 through 14.2-145	12
14.2-146	7
14.2-147 through 14.2-149	12
14.2-151 through 14.2-159	7
FIGURES	
14.2-1 and 14.2-2	7
14.2-5 and 14.2-6	7

CHAPTER 15

TEXT PAGES	
15-xxi	8
15-xxii and 15-xxiia	8
15-xxxvii	5
15.0-14	11
15.0-15	11
15.0-20	11
15.1-31	12
15.2-48 through 15.2-50	11
15.2-52 through 15.2-57	11
15.2-67	8
15.2-68	8
15.2-71 and 15.2-71a	8
15.2-72 through 15.2-76	8
15.2-80a through 15.2-80d	8
15.4-8	5
15.4-49	5
15.6-34	9
15.6-34a	11
15.6-39	10

CHAPTER 14 (Continued)

CHAPTER 15 (Continued)

TEXT PAGES	AMENDMENT
15.6-40	9
15.6-41	10
15.6-42	9
15.6-44 through 15.6-47	10
15.6-48	9
15.8-3	12

FIGURES	
15.A.6-8	11
15.A.6-12 and 15.A.6-13	11
15.A.6-14a and 15.A.6-14b	11
15.A.6-15	11
15.A.6-20	11
15.A.6-22 through 15.A.6-31	11
15.A.6-38 through 15.A.6-40	11
15.A.6-42	11
15.A.6-43 through 15.A.6-45	11
15.A.6-51 through 15.A.6-53	11

CHAPTER 17

TEXT PAGES	
17.1-1 through 17.1-4	5
17.1-5 through 17.1-14	13
17.2-1	11

APPENDIX C

TEXT PAGES	
C.2-3	4
C.2-16	4
C.2-18 through C.2-20	13
C.2-40	13
C.2-44 through C.2-46	13
C.2-55	13
C.2-63	13
C.2-65 and C.2-66	5
C.2-68	9
C.2-70	7
C.2-72	7
C.2-82	13
C.2-86	13

APPENDIX C (Continued)

TEXT PAGES	AMENDMENT
C.3-5	5
C.3-9	13
C.3-11	5
C.3-14	12
C.3-17	2
C.3-17a	2
C.3-18	13
C.3-22	5
C.3-23	8
C.3-25	8
C.3-32	8
C.3-33	8
C.3-34	8
C.3-35	11
C.3-39	13
C.3-41	13
C.3-45	5
C.3-47	8
C.3-49	13
C.3-49a	9
C.3-51	13
C.3-54	8
C.3-55 through C.3-57	13
C.3-59 and C.3-60	7
C.3-61	13
C.3-67	8
C.3-70	9
C.3-74	9
C.3-75	13
C.3-76	11
C.3-81	4
C.3-82	8
C.3-84	13
C.3-91	13
C.3-92	9
C.3-102	8
C.3-107	8
C.3-111	8

APPENDIX D

TEXT PAGES	
Title Page	7

APPENDIX E

TEXT PAGES	AMENDMENT
Title Page	10
E.1-1	7
E.2-1	12
E.3-1	8
E.3-3 through E.3-5	12
E.3-5a	3
E.3-9	3
E.3-10	5
E.7-1 through E.7-5	13
E.9-1	8
E.9-2	5
E.9-3 and E.9-3a	8
E.10-1	5
E.11-1	8
E.12-1	12
E.12-2	12

APPENDIX F

TEXT PAGES	
Title Page	7

APPENDIX G

TEXT PAGES	
Title Page	7

APPENDIX H

TEXT PAGES	
Title Page	7
H.O-vii and H.O-viia	12
H.1.2-1	12
H.2.1-4	10
H.2.3-5 and H.2.3-5a	12
H.4.3-1 and H.4.3-1a	12
H.A.O-i	10

FIGURES	
H.A-1a through h	10

NRC QUESTIONS AND RESPONSES
(See Separate List)

NRC QUESTION/FSAR SECTION
CROSS REFERENCE
(See Separate List)

EMERGENCY PLAN
(Chapter 13.3)

APPENDIX IX
(Emergency Plan)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued) Page

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

1.4.1 APPLICANT - WPPSS

Washington Public Supply System is a municipal corporation, and a joint operating agency of the State of Washington, organized in January 1957, pursuant to Chapter 43.52 of the Revised Code of Washington as amended. WPPSS is composed of 19 operating public utility districts of the state of Washington and the cities of Richland, Tacoma and Seattle, Washington. WPPSS assumes the responsibility for safe operation and maintenance of the plant and for providing related services as described in Chapter 13.

1.4.2 ENGINEER AND CONSTRUCTION MANAGEMENT - BURNS AND ROE, INC.

Burns and Roe, Inc. (B&R) has been retained by WPPSS to provide engineering and construction management and quality assurance services for the design and construction of the plant, integrating the major plant items furnished by General Electric Company and Westinghouse Electric Corporation. Burns and Roe was also the engineering consultant and the construction manager for the Hanford No. 1 generating plant. Burns and Roe has been continuously engaged in construction or engineering activities since 1935.

Burns and Roe was founded in 1932, and was incorporated in 1935 as Burns and Roe, Inc. Burns and Roe has been active in the fields of power generation and distribution, sea water and brackish water desalination, waste water renovation, engineering, design and/or construction management services for over 50 thermal power generating units representing more than 11,400,000 kilowatts of new generating capacity of which more than 4,800,000 kilowatts is nuclear.

1.4.3 NUCLEAR STEAM SYSTEM SUPPLIER - GENERAL ELECTRIC COMPANY

The General Electric Company (GE) has been awarded the contracts to design, fabricate, and deliver the direct cycle boiling water nuclear steam supply system, to fabricate the first core of nuclear fuel, and to provide technical direction of installation and startup of this equipment. GE has engaged in the development, design, construction and operation of boiling water reactors since 1955. Table 1.4-1 lists over 90 GE reactors completed, under construction, or on order. Thus, GE has substantial experience, knowledge, and capability to design, manufacture, and furnish technical assistance for the installation and startup of reactors.

1.4.4 TURBINE-GENERATOR SUPPLIER - WESTINGHOUSE ELECTRIC CORP.

WPPSS has awarded a contract to Westinghouse Electric Corp. to design, fabricate and deliver the turbine-generator for WNP-2 as well as to provide technical assistance for installation and start-up of this equipment.

Westinghouse Electric Corp. has a long history in the application of turbine-generators in nuclear power stations going back to the inception of commercial electrical power production utilizing nuclear facilities. Westinghouse furnished the turbine-generator unit for Shippingport No. 1. This unit was shipped in 1956. Westinghouse also furnished the turbine-generator unit for Yankee Atomic Power Company Rowe No. 1. This unit was shipped in 1959. San Onofre No. 1 and Connecticut Yankee, Haddam Neck No. 1 unit went into commercial operation in 1968. Westinghouse nuclear turbine-generators produced over 300 billion kilowatt hours of electricity through May, 1976, when twenty-five nuclear turbine-generators totaling over 16,500 megawatts were in service. By 1984, seventy-five Westinghouse nuclear turbine-generators should be in service producing over 61,319 megawatts. Inlet steam pressures of these units vary between 750 psig and 1000 psig and electrical outputs vary from 500, kW to 1,090,000 kW. Westinghouse is therefore competent to design, fabricate, deliver and erect the turbine-generator set and to provide technical assistance for the start-up of this equipment.

1.4.5 TECHNICAL CONSULTANTS

In connection with this project, and in addition to the engineering consultant (B&R) described above, WPPSS has engaged R.W. Beck and Associates as consulting engineer and the S. M. Stoller Corporation as nuclear fuel consultant.

1.4.5.1 R. W. Beck and Associates

The independent consulting firm of R. W. Beck and Associates has been employed as the consulting engineer for Washington Public Power Supply System's Nuclear Project No. 2. This firm was also employed as a consulting engineer for Hanford No. 1. The current employment level of R. W. Beck and Associates is approximately 350 including approximately 200 professional engineers. Having extensive experience in preparing engineering feasibility and financing studies and reports necessary for the success of utility and civic improvement projects, the firm is well qualified for employment as a consulting engineer and was chosen as a result of its experience.

The duties of the consulting engineer are briefly summarized as follows: prepare estimates of plant capability, energy potential, usability within area loads and resources, the cost of power and energy output of the project, and generally determine the feasibility of the project. These duties will include assisting in preparation of a Bond Resolution, preparation of an engineering report, schedules for investment of funds, schedules for debt service payments, and other engineering services necessary to facilitate the financing of the project.

1.4.5.2 The S. M. Stoller Corporation

The S. M. Stoller Corporation has been employed as the nuclear fuel consultant to the Washington Public Power Supply System with principal responsibilities for preparation of bid specifications for nuclear fuel and evaluation of the technical and economical impact of the bids resulting from those specifications. The services of S. M. Stoller Corp. have been required throughout the project as a consultant for assessment of core design, fuel cycle management, and waste management. The S. M. Stoller Corp. currently has approximately 30 senior engineers, well experienced in not only nuclear fuel management, but also in technological and environmental assessment of commercial nuclear generating facilities. The S. M. Stoller Corp. has been a nuclear consultant for more than thirty of the nation's largest nuclear oriented utilities and other organizations.

TABLE 1.4-1

COMMERCIAL NUCLEAR REACTORS COMPLETED, UNDER CONSTRUCTION,

OR IN DESIGN BY GENERAL ELECTRIC

STATION	UTILITY	RATING (MWe)	YEAR OF ORDER	YEAR OF STARTUP
Dresden 1	Commonwealth Edison	200	1955	1960
Humboldt Bay	Pacific G&E	69	1958	1963
Kahl	Germany	15	1958	1961
Garigliano	Italy	150	1959	1964
Big Rock Point	Consumers Power	70	1959	1965
JPDR	Japan	11	1960	1963
KRB	Germany	237	1962	1967
Tarapur 1	India	190	1962	1969
Tarapur 2	India	190	1962	1969
GKN	Holland	52	1963	1968
Oyster Creek	JCP&L	640	1963	1969
Nine Mile Point 1	Niagara Mohawk	625	1963	1969
Dresden 2	Commonwealth Edison	809	1965	1970
Pilgrim	Boston Edison	664	1965	1972
Millstone 1	NUSCO	652	1965	1970
Tsuruga	Japan	340	1965	1970
Nuclenor	Spain	440	1965	1971
Fukushima 1	Japan	439	1966	1971
BKW KKM	Switzerland	306	1966	1972
Dresden 3	Commonwealth Edison	809	1966	1971
Monticello	Northern States	545	1966	1971
Quad Cities 1	Commonwealth Edison	800	1966	1972
Browns Ferry 1	TVA	1098	1966	1974
Browns Ferry 2	TVA	1098	1966	1975
Quad Cities 2	Commonwealth Edison	800	1966	1972
Vermont Yankee	Vermont Yankee	514	1966	1972
Peach Bottom 2	Philadelphia Electirc	1065	1966	1974
Peach Bottom 3	Philadelphia Electric	1065	1966	1974
Fitzpatrick	PASNY	821	1966	1975
Bailly	NIPSCO	660	1967	----
Shoreham	LILCO	819	1967	----
Cooper	Nebraska PPD	778	1967	1974
Browns Ferry 3	TVA	1098	1967	1977
Limerick 1	Philadelphia Electric	1055	1967	----
Hatch 1	Georgia	786	1967	1975
Fukushima 2	Japan	762	1967	1974
Brunswick 1	Carolina P&L	821	1968	1977
Brunswick 2	Carolina P&L	821	1968	1975
Arnold	Iowa ELP	569	1968	1975
Fermi 2	Detroit Edison	1056	1968	----
Limerick 2	Philadelphia Electric	1055	1969	----
Hope Creek 1	PSE&G	1067	1969	----
Hope Creek 2	PSE&G	1067	1969	----

Response:

The consequences of an anticipated transient without scram (ATWS) are mitigated by tripping the recirculation pumps and by manual insertion of the control rods. (For more information, see 15.8).

1.5.1.2 Current Development Program

1.5.1.2.1 Loose Parts Detection

A Loose Parts Detection System will be provided. See 7.7.1.12 for a system description.

1.5.1.2.2 Mark II Containment Suppression Pool Dynamic Loading

The Washington Public Power Supply System, in conjunction with other Mark II owner utilities, has submitted a design basis document designated as "Mark II Containment Dynamic Forcing Function Information Report" (DFFIR), NEDO-21061, and NEDE-21061P describing the suppression pool dynamic loading phenomena during a safety/relief valve actuation or LOCA event. The evaluation of that design basis document against the current WPPSS Nuclear Project No. 2 design was prepared and submitted to the NRC. A verification program to demonstrate the conservatism of the DFFIR has been sponsored by the Mark II owners and is described in the "Mark II Containment Supporting Program Report", GE Document NEDO-21297.

TABLE 1.6-1 (Continued)

REPORT NUMBER	TITLE	FSAR PORTIONS WHERE REFERENCED
WAPD-T-416	WIGLE - A Program for the Solution of the Two-Group Space-Time Diffusion Equations in Slab Geometry (1964)	4.3(10)*
WAPD-TM-629	Irradiation Behavior of Zircaloy-Clad Fuel Rods Containing Dished-End UO_2 Pellets (July 1967)	4.2(6)*
WPPSS-74-2-R2 and Supplements WPPSS-74-2-R2A and WPPSS-74-2-R2B	Washington Public Power Supply System Sacrificial Shield Wall	3.8
Report submitted on letter GO2-80-172, August 8, 1980	Engineering Evaluation of the WNP-2 Sacrificial Shield Wall	3.8, 6.2
Report submitted on letter GO2-80-182, August 19, 1980	Engineering Evaluation of the WNP-2 Sacrificial Shield Wall, Supplement No. 1	3.8, 6.2

*These FSAR sections do not reference these documents directly but indirectly via NEDE-20944P. NEDE-20944P and FSAR section numbers correspond. The numbers in parentheses are reference numbers from NEDE-20944P for the respective section numbers.

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

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TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

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
28. Primary Containment Cooling System (Figure 3.2-15)							
.1 Piping and Valves up to outermost isolation valves, containment purge and exhaust	P	2	C,R	B	I	I	
29. Standby Gas Treatment System (Figure 3.2-16)							
.1 Filter Units	P	2	R	B	I	I	
.2 Fans	P	2	R	B	I	I	
.3 Piping and Valves	P	2	R	B	I	I	
30. Primary Containment Atmospheric Control System (Figure 3.2-17)							
.1 Piping and Valves	P	2	C,R	B	I	I	
.2 Equipment	P	2	R	B	I	I	
31. Other HVAC (Figures 3.2-18 to 20)							
.1 Reactor Building (non-essential)	P	G	R	N/A	II	II	(10)
.2 Reactor Building (essential)	P	3	R	N/A	I	I	
.3 Turbine Building	P	G	G	N/A	II	II	(28)
.4 Radwaste Building	P	G	W	N/A	II	II	(28)
.5 Control Room, Critical Switchgear Area, Cable Spreading Area (non-essential)	P	G	W	N/A	II	II	

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 Control Room, Critical Switchgear Area, Cable Spreading Area (essential)	P	3	W	N/A	I	I	
.7 Diesel Generator Building	P	3	DG	N/A	I	I	(29)
.8 Standby Service Water Pumphouse	P	3	P	N/A	I	I	(29)
32. Condensate Storage and Transfer (Figure 9.2-6)							
.1 Condensate storage tank	P	G	O	C	II	II	(20)
.2 Piping and valves	P	G	O,T,R,W	D	II	II	
.3 Pumps	P	G	O	D	II	II	
33. Instrument and Sample Lines			See note 12				
34. Fuel Storage Facilities							
.1 Fuel Pool/Dryer Separator Liner	P	3	R	N/A	I	I	
.2 Storage Racks and Supports	GE	3	R	N/A	I	I	
35. Building Cranes							
.1 Reactor Building	P	3	R	N/A	I	I	
.2 Turbine Building	P	G	T	N/A	II	II	
.3 Radwaste Building	P	G	W	N/A	II	II	
.4 Standby Service Water Pumphouse	P	G	P	N/A	I		
.5 Miscellaneous Areas	P	G	P,W,T,S	N/A	II	II	
36. Instrument and Service Air (Figure 9.3-1)							
.1 Piping and Valves	P	G	R,W,T,O	D	II	II	(10)
.2 Compressors	P	G	T	D	II	II	
.3 Vessels	P	G	T	D	II	II	

3.5.1.4.3 Flood Generated Missiles

The design basis flood elevation discussed in 3.4 and defined in 2.4, exceeds the flood levels associated with breaches of the Grand Coulee Dam. The final plant grade level is higher than the design basis flood. Therefore, flood generated missiles are not considered in the design of the Seismic Category I safety related structures and installations.

3.5.1.4.4 Protection and Design Procedures

Systems protected from missiles generated by natural phenomena, and barrier design procedures are described in 3.5.2 and 3.5.3 respectively.

3.5.1.5 Missiles Generated by Events Near the Site

Hazards due to missiles postulated in the design basis explosions or accidents at nearby industrial plants, military facilities pipelines or storage facilities as discussed in 2.2 can be discounted because of their remote relationship to the WNP-2 plant.

Transportation facilities in the immediate vicinity of the plant consist of a railroad system owned and operated by DOE and a DOE owned Reservation Road System which connects the Reservation with two (2) nearby state highways. The DOE Railroad and Road Systems are restricted and used in support of the Hanford Operations and are discussed in 2.2.

There is no commercial river traffic passing the site on the Columbia River; only small pleasure craft normally use the river. For detailed discussion of current and potential future activities on the Columbia River, see 2.2.

For an evaluation of potential accidents at the foregoing facilities, and determination of design basis events, see 2.2.3.1.

3.5.1.6 Aircraft Hazards

All airports, commercial, private and military are located outside a ten mile radius of the plant. Projected operations at these airfields are examined in 2.2.2. The frequency of aircraft flights per year does not pose a threat to WNP-2 operation.

Military installations do not threaten the plant site since none are located within 20 miles. Refer to 2.2.2.

As discussed in 2.2.3, the probability of the aircraft crashing into the plant is less than 1×10^{-7}. Therefore, aircraft is discounted as a credible missile.

3.5.1.6.1

DELETED: see 3.5.1.6.

3.5.1.6.2

DELETED: see 3.5.1.6.

3.5.2 SYSTEMS TO BE PROTECTED

The structures, systems and components necessary for bringing the plant to a safe shutdown and the protection provided for these structures, systems and components is discussed in 3.5.1 and 3.5.2.

A description of the protection provided for the safety related structures located outdoors, against tornado generated missiles, is furnished in 3.5.1.4; by turbine missiles, in 3.5.1.3; and by a sesimic event, in 3.8.4.

3.5-11 Regulatory Guide 1.14, Rev. 1, "Reactor Coolant Pump Flywheel Integrity", August 1975.

3.5-12 Miller, D. R. and Williams, W. A., Tornado Protection for the Spent Fuel Pool, General Electric Company, APED-5696, November 1968.

3.5-13 "Protection Against Pipe Breaks Outside Containment", Burns and Roe, Inc., Hempstead, New York, Report No. WPPSS-74-2-R3, April, 1974.

3.5-14 "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects" R.P. Kennedy, Nuclear and Systems Sciences Group, Holmes and Narver, Inc., September 1975.

3.5-15 "Analysis of the Probability of the Generation and Strike of Missiles from a Nuclear Turbine" March, 1974 by Westinghouse Electric Corporation Steam Turbine Division Engineering.

TABLE 3.5-1

SYSTEMS DESCRIPTION OUTSIDE CONTAINMENT

SYSTEMS AVAILABLE FOR A SAFE SHUTDOWN	FUNCTION	SECTION FSAR	FIGURES 3.5	SEISMIC & QUALITY CLASSIFICATION RESPECTIVELY
RCIC	MAINTAIN RPV WATER INVENTORY	5.4.6 7.4.1.1	9-14	I, I
HPCS	MAINTAIN RPV WATER INVENTORY	6.3, 7.3.1.1.1.1	9-14	I, I
SSW	HEAT REJECTION	7.3.1.1.6	9-14	I, I
RHR A B C	MAINTAIN WATER INVENTORY & DECAY, HEAT REMOVAL	5.2, 7.3.1.1.1.4 6.3, 5.4.7	1-8	I, I
CRD	REACTIVITY CONTROL	7.7.1.2	15	I, I
RFW	MAINTAIN RPV WATER INVENTORY	5.4.9	9-14	-
LPCS	MAINTAIN RPV WATER INVENTORY	6.3, 7.3.1.1.1.3	1-8	I, I

Note: Identification of missiles to be protected against, their sources, and bases for selection are discussed in 3.5.1.1.3. The ability of the structures, systems and components to withstand the effects of selected internally generated missiles is discussed in 3.5.1.1.2.

a. For breaks not involving recirculation piping, at least two LPCI pumps or one core spray system is available for core cooling.

b. For breaks involving recirculation piping, at lease one core spray line and 2 LPCI pumps, or 2 core spray lines, are available for core cooling.

c. For a LOCA with a total effective break area less than 0.7 ft^2, either the HPCS or ADS is available for reactor depressurization.

d. For liquid breaks, such as cleanup suction or the combination of liquid and steam breaks whose total break area is less than 0.7 ft^2 in which the ADS system is required for depressurization, at least 6 ADS valves are available.

e. For breaks less than the equivalent flow area of one open ADS valve, at least 6 ADS valves are available. However, the required number of ADS valves is one less for each additional steam break area equivalent to the area of one open ADS valve.

3.6.2.5.3.6 Containment System Integrity

The following was considered in addressing the LOCA dynamic effects with respect to containment system integrity:

a. Leak tightness of the containment fission product barrier is assured throughout any LOCA.

b. For those lines which penetrate the containment and are closed during normal operation, the inboard isolation valves are as close as practicable to the reactor pressure vessel. This arrangement reduces the length of pipe subject to a pipe break.

3.6.2.5.4 System by System Description of Pipe Whip Protection

3.6.2.5.4.1 Main Steam System

a. System Arrangement

The main steam system consists of four, 26-inch lines which are arranged inside primary contain-

ment with mirror image symmetry about the 0° and 180° north-south azimuth. The lines exit the reactor pressure vessel on opposite sides of primary containment and drop down vertically in two parallel pairs to the main steam relief valve platform at elevation 541 ft. where they are routed horizontally, in parallel, in the northeast and northwest quadrants to the 0° north azimuth. At this point, the four lines drop vertically in parallel, to an elevation just above the diaphragm floor. The main steam isolation valves are located here. The four lines exit the containment nearest the north azimuth at elevation 500 ft. (approx.). The two feedwater piping loops are described in 3.6.2.5.4.2 and are routed near the main steam lines.

b. Pipe Whip Protection

The postulated pipe breaks and pipe whip restraints for the four main steam lines, are shown in Figures 3.6-12a through 3.6-15a. Where pipe breaks are postulated inside primary containment, the main steam lines are restrained to prevent the unacceptable motion of these pipes. These restraints are mounted on the side of the sacrificial shield wall structure, as well as on radial beams which extend from the sacrificial shield wall to the primary containment vessel wall. A sliding beam seat at the primary containment wall, permits the beam to grow axially and also permits the primary containment wall to move relative to the sacrificial shield wall.

A structural steel frame (see Figures 3.6-36a, 3.6-36b, and 3.6-36c) between the drywell diaphragm floor and the containment vessel, in the area of the main steam isolation valves, is provided for mounting of pipe whip restraints. The structure is designed with vertically sliding connections at the containment vessel, to allow for differential thermal expansion between the containment vessel and the diaphragm floor.

c. Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the main steam system to assure safety as defined

(2) Equipment access hatch 1000 psf

(3) Supression chamber access hatch 150 psf

d. Live load on temporary construction scaffolds.

e. Operating weight of fluid in attached normally empty piping, headers and penetrations.

f. Head of water, 23'-6" high, on the refueling bellows seal with the containment vessel head removed and coincident hydrostatic pressure (under the Refueling Condition).

g. Same as 3.8.2.3.4 (f) above except without the containment vessel head removed. (under the Flooded Condition).

3.8.2.3.5 Mechanical Piping Loads

Mechanical piping loads consist of:

a. Piping reactions during normal operating or shutdown conditions, based on the most critical transient or steady state condition.

b. Pipe reactions under thermal conditions generated by a postulated break and including (a) above.

c. Equivalent static load generated by the reaction of a broken high-energy pipe during a postulated break (and including an appropriate dynamic load factor to account for the dynamic nature of the load).

d. Jet impingement equivalent static load generated by the DBA postulated break (and including an appropriate dynamic load factor to account for the dynamic nature of the load).

e. Pipe reactions and thermal conditions during an event causing external pressure.

A description of certain loads included among those listed above follows in detail:

3.8.2.3.5.1 Jet Forces in Drywell

The drywell shell, personnel air lock, equipment hatch, jet deflectors and the removable top closure head are designed and constructed to withstand, in combination with other loads, jet forces consisting of either steam and/or water at 340^{O}F and applied as follows:

	Jet Force	Area Subjected
From the closure head flange to the top of the head	33 kips	26 sq. in.
From the closure head flange down to the drywell floor	534 kips	429 sq. in.

A jet force is considered to occur in any direction but is not considered to occur simultaneously with another jet force; however, a jet force is considered to occur coincident with the drywell internal design pressure of 45 psig and design temperature of 340^{O}F. Local yielding may take place on the drywell shell from the jet force, but the shell will not rupture. On the top closure head and other areas, where the shell is not backed up by concrete, the primary stresses resulting from this combination of loads do not exceed the values specified in the ASME Code, Section III, paragraph NE-3131 (c) at a temperature of 340^{O}.

3.8.2.3.5.2 Vent Pipe (Downcomer) Thrusts

The vent pipes (downcomers) and their connections to the drywell floor are designed for the following loads:

a. Jet Blowdown Thrust

A jet force of 20,000 lbs acting upward on each of the downcomers is considered to occur simultaneously with the internal design pressure of 45 psig in the drywell and suppression chamber and the design temperature of 275^{O}F in the drywell, wetwell and drywell floor concrete.

b. Initial and Final Test Conditions

A force equal to design pressure multiplied by the flow area of the vent pipe.

c. Accident Conditions

Forces obtained from 3.8.2.3.5.2 (a) except that the temperature in the drywell is taken as 340^{o}F and the temperature in the suppression chamber is taken as 275^{o}F. The drywell floor concrete temperature is taken as 95^{o}F.

3.8.2.3.5.3 Pipe Whip

Pipe whip protection support rings, which are fully circumferential rings, are attached to the primary containment vessel at elevations 516'-6" and 542'-7-1/4". The basic function of these rings is to support pipe whip protection framework and to adequately distribute pipe whip loading into the vessel.

The pipe rupture loading is applied to the vessel through the support rings during the Normal Operating Condition at normal operating temperature and at atmospheric pressure, as well as during an Incident Condition at maximum temperature of 340^{o}F and at design pressure. The primary containment vessel analysis includes the effects of a pipe rupture at any single location.

For further discussion on function and design of load transmitting members see 3.6.

3.8.2.3.6 Thermal Loads

The thermal loads in the primary containment vessel steel are produced by the presence of temperature gradients within the containment and its appurtenances. Thermal effects and loads during normal operating conditions are based on the most critical transient or steady state condition. Thermal loads are also considered under thermal conditions generated by a postulated pipe break.

3.8.2.3.7 Construction Loads

a. Wind load in the projected area of the steel primary containment vessel before the completion of the reactor building in accordance with reference 3.8-1, with a basic wind of 100 mph as discussed in 3.3.

b. Snow loads before the completion of the reactor building.

3.8.2.3.8 Missile Loads

There are no external missile loads considered since the primary containment vessel is protected by the biological shield wall.

Potential internal missiles and protection provisions are discussed in 3.5.

3.8.2.3.9 Loss-of-Coolant Accident Loads

The loss-of-coolant accident (LOCA) imposes pressure and thermal loads plus jet forces associated with coolant flow from any ruptured pipe within the containment. This LOCA loading condition is determined by analysis of the transient pressure and temperature effects which occur during a loss-of-coolant accident. The governing design condition for the LOCA is discussed in Chapter 6.

3.8.2.3.10 Accident Recovery Loads

Among the postulated loss-of-coolant accidents there may be an accident within the drywell that requires a "last ditch" contingency flooding of the pressure suppression chamber and the drywell to an elevation above the top of the active fuel zone in the reactor vessel as indicated in 3.8.2.3.12h; and, with the primary containment vessel head not removed, the reactor vessel cavity outside the primary containment vessel and above the refueling bellows seal flooded to a level above the refueling bellows seal noted in 3.8.2.3.12h.

The structural design criteria for the primary containment vessel are consistent with the provisions of Regulatory Guide

3.8-10 ACI 318-1971, "Building Code Requirements for Reinforced Concrete", American Concrete Institute (1971).

3.8-11 AISC, "Specification for Design, Fabrication and Erection of Structural Steel for Buildings", American Institute of Steel Construction (1969).

3.8-12 ANSI N45.2.5, "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants", Draft 3, Revision 1, January 1974, American National Standards Institute.

3.8-13 Letter GI2-75-10, W. R. Butler to J. J. Stein, Transmitting Request for Additional Information, dated January 14, 1975, on Drywell to Wetwell Leakage Study, Docket 50-397.

3.8-14 Letter GO2-76-156, D. L. Renberger to W. R. Butler, entitled WPPSS Nuclear Project No. 2, Drywell/Wetwell Leakage Study, transmitting response to request for additional information in reference 3.8.3-10, dated April 23, 1976, Docket 50-397.

3.8-15 Deleted. Replaced with 3.8-23.

3.8-16 Savin, G. N., Stress Distribution Around Holes, Translation of "Raspredeleniye Napryazheniy Okolo Otverstiy", "Naukova Dumka" Press, 1968, National Aeronautics and Space Administration, NASA TT F-607, Washington, D. C., November 1970.

3.8-17 Roark, R. J., Formulas for Stress and Strain, Fourth Edition, McGraw-Hill Book Company, Inc., New York, 1965.

3.8-18 Abbett, R. W., American Civil Engineering Practice, John Wiley and Sons, Inc., New York, 1956.

3.8-19 Shannon and Wilson, Inc., Soil Compaction Evaluation of Quality Class I Backfill, Washington Public Power Supply System, WPPSS Nuclear Project No. 2 (WNP-2).

3.8-20 Deleted. Replaced with 3.8-23.

3.8-21 "Primary Containment Vessel for Washington Public Power Supply System, Hanford No. 2, Jet Impingement Analysis," FIRL Technical Report F-C14121, May 21, 1975.

3.8-22 "HYBOS," FIRL Users Manual, July 1973.

3.8-23 Plant Design Assessment Report for SRV and LOCA Loads, Revision 2, Washington Public Power Supply System, August 1979, transmitted to NRC as Amendment No. 6 to the FSAR, September 19, 1979.

3.8-24 Engineerinng Evaluation of the Sacrificial Shield Wall, submitted to the NRC on WPPSS letter G02-80-172, August 8, 1980.

3.8-25 Engineering Evaluation of the Sacrificial Shield Wall, Supplement No. 1, submitted to NRC on WPPSS letter G02-80-182, August 19, 1980.

Transient	Category	Cycles
and return to normal operating temperature of 546°F at a rate of 100°F/hr. Pressure change from 1000 psig to 1180 psig to 240 psig and return to normal operating of 1000 psig.		

Paragraph NB3552 of ASME III code excludes various transients and provides means for combining those which are not excluded. Review and approval of the equipment supplier's certified calculation provides assurance of proper accounting of the specified transients.

3.9.1.1.10 Recirculation Flow Control Valve Transients

The following pressure and temperature transients were considered in the design of the recirculation system flow control valve:

	Transient	Category	Cycles
a.	Startup (100°F/hr heating rate - 70°F to design temperature	normal/upset	300
b.	Small temperature changes (29°F)	normal/upset	600
c.	50°F step changes	normal/upset	200
d.	Safety/relief valve blowdowns (single valve) (546°F to 375°F in 10 minutes)	normal/upset	30
e.	Safety valve transient (110% of design pressure)	normal/upset	1

	Transient	Category	Cycles
f.	Installed hydrotests		
	a. 1300 psig	testing	130
	b. 1670 psig	testing	3
g.	Automatic blowdown ($546^{o}F$ to $281^{o}F$ in 15 seconds)	emergency	2
h.	Improper start of pump in cold loop ($130^{o}F$ step to $546^{o}F$ for 15 seconds)	emergency	1

3.9.1.1.11 Recirculation Pump Transients

The following transients are listed in the design specification as a requirement for design considerations. However, a submitted certified analysis considering thermal stresses was not required. The vendor was required to submit a certification of compliance. The submitted certified design calculations only considered pressure transient. Nozzle piping loads were considered in accordance with the following paragraph:

> "The pump case shall be designed to withstand secondary stresses due to piping reactions in accordance with Paragraph 452.4b of the ASME Standard Code for Pumps and Valves for Nuclear Power (1968 Draft)."

	Transients	Category	Cycles
a.	Heatup and cooldown at $100^{o}F/hr$	normal/upset	300
b.	$\pm 29^{o}F$ temperature changes	normal/upset	600

3.9.7 REFERENCES

3.9-1 "Design and Performance of G.E. BWR Jet Pumps," General Electric Company, Atomic Power Equipment Department, APED-5460, July 1968.

3.9-2 Moen, R. H., "Testing of Improved Jet Pumps for the BWR/6 Nuclear System," General Electric Company, Atomic Power Equipment Department, NEDO-10602, June 1972.

3.9-3 General ELectric Company, "Analytical Model for Loss of Coolant Analysis in Accordance with 10CFR50 Appendix K," Proprietary Document, General Electric Company, NEDE-20566.

3.9-4 "BWR Fuel Channel Mechanical Design and Deflection," NEDE-21354-P, September 1976.

3.9-5 "BWR/6 Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings," NEDE-21175-P, November 1976.

3.9-6 "Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants," NEDE-24057-P (Class III) and NEDO-24057 (Class I), October 1977.

TABLE 3.9-1

PLANT EVENTS

		No. of Cycles
Normal, Upset, and Testing Conditions		
a.	Bolt Up*	123
b.	Design Hydrostatic Test	130
c.	Startup (100^{o}F/hr Heatup Rate)**	120
d.	Daily Reduction to 75% Power*	10,000
e.	Weekly Reduction to 50% Power*	2,000
f.	Control Rod Pattern Change*	400
g.	Loss of Feedwater Heaters (80 Cycles Total):	80
h.	Operating Base Earthquake Event at Rated Operating Conditions	1****
i.	Scram:	
	1) Turbine Generator Trip, Feedwater on, Isolation Valves Stay Open	40
	2) Other Scrams	140
	3) Loss of Feedwater Pumps, Isolation Valves Closed	10
	4) Single Safety or Relief Valve Blowdown	8
j.	Reduction to 0% Power, Hot Standby, Shutdown (100^{O}F/hr Cooldown Rate)**	111

discharge line to prevent drawing an excessive amount of water up into the line as a result of steam condensation following termination of relief operation. Each vacuum relief valve pair is situated with the valves in parallel, the discharge being routed to a common tee in the safety/relief valve discharge line. The safety/relief valves are located on the main steam line piping, rather than on the reactor vessel top head, primarily to simplify the discharge piping to the pool and to avoid the necessity of having to remove sections of this piping when the reactor head is removed for refueling. In addition, valves located on the steam lines are more accessible during a shutdown for valve maintenance.

The nuclear pressure relief system automatically depressurizes the nuclear system sufficiently to permit the LPCI and LPCS systems to operate as a backup for the high pressure core spray (HPCS) system. Further descriptions of the operation of the automatic depressurization feature are found in 6.3, "Emergency Core Cooling Systems," and in 7.3.1.1.1, "Emergency Core Cooling Systems (ECCS) Instrumentation and Controls."

5.2.2.4.2 Design Parameters

Table 5.2-3 lists design temperature, pressure, and maximum test pressure for the RCPB components. The specified operating transients for components within the RCPB are given in 3.9. Refer to 3.7 for discussion of the input criteria for design of Seismic Category I structures, systems, and components.

A summary of the number of cycles for transients used in design and fatique analysis is listed in Table 5.2-11 and categorized under the appropriate design conditions (i.e., normal, upset, emergency, and faulted).

The design requirements established to protect the principal components of the reactor coolant system against environmental effects are discussed in 3.11.

5.2.2.4.2.1 Safety/Relief Valve

The discharge area of the valve is 16.117 square inches and the coefficient of discharge K_D is equal to 0.966.

The design pressure and temperature of the valve inlet and outlet are 1250 psig @ 575°F and 550 psig @ 550°F, respectively.

The valves have been designed to achieve the maximum practical number of acutations consistent with state-of-the-art technology. Cyclic testing has demonstrated that the valves are capable of at least 60 actuation cycles between required maintenance.

See Figure 5.2-10 for a schematic cross section of the valve.

5.2.2.5 Mounting of Pressure Relief Devices

The pressure relief devices are located on the main steam piping header. The mounting consists of a special, contour nozzle and an over-sized flange connection. This provides a high integrity connection that accounts for the thrust, bending and torsional loadings which the main steam pipe and relief valve discharge pipe are subjected to. This includes:

a. The thermal expansion effects of the connecting piping

b. The dynamic effects of the piping due to SSE

c. The reactions due to transient unbalanced wave forces exerted on the safety/relief valves during the first few seconds after the valve is opened and prior to the time steady-state flow has been established.

d. The dynamic effects of the piping and branch connection due to the turbine stop valve closure.

In no case will allowable valve flange loads be exceeded nor will the stress at any point in the piping exceed code allowables for any specified combination of loads. The design criteria and analysis methods for considering loads due to SRV discharge is contained in 3.9.3.3.

5.2.2.6 Applicable Codes and Classification

The vessel overpressure protection system is designed to satisfy the requirements of Section III, Nuclear Vessels, of the ASME Boiler and Pressure Vessel Code. The general requirements for protection against overpressure as given in Article 9 of Section III of the Code recognize that reactor vessel overpressure protection is one function of the reactor protective systems and allows the integration of pressure relief devices with the protective systems of the nuclear reactor. Hence, credit is taken for the scram protective system as a complementary pressure protection device.

Since there are no additives in the BWR coolant, leakage would expose materials to high purity, demineralized water. Exposure to demineralized water would cause no detrimental effects.

5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness

Fracture toughness requirements for the ferritic materials used for piping and valves (no ferritic pumps in RCPB) of the reactor coolant pressure boundary were as follows:

Safety/Relief Valves were exempted from fracture toughness requirements because Section III of the 1971 ASME Boiler and Pressure Vessel Code did not require impact testing on valves with inlet connections of 6 inches or less nominal pipe size.

Main Steam Isolation Valves were also exempted because the Code existing at the time of the purchase order, April 1971, did not require brittle fracture testing on ferritic pressure boundary components when the system temperature was in excess of 250°F at 20% of the design pressure.

Main Steam Piping was tested in accordance with and met the fracture toughness requirements of paragraph NB-2300 of the 1972 Summer Addenda to ASME Code, Section III, the applicable code at the time of the purchase order, September 1972.

5.2.3.3.1.1 Compliance with Code Requirements

The ferritic pressure boundary material of the reactor pressure vessel was qualified by impact testing in accordance with the 1971 Edition of Section III ASME Code and Addenda to and including the Summer 1971 Addenda. From an operational standpoint, this Code would require that for any significant pressurization (taken to be more than 20% of Code hydrostatic test pressure = 312 psig) the minimum metal temperature of all vessel shell and head material be 100°F (NDTT +60°F).

5.2.3.3.2 Control of Welding

5.2.3.3.2.1 Control of Preheat Temperature Employed for Welding of Low Alloy Steel Regulatory Guide 1.50

The use of low alloy steel is restricted to the reactor pressure vessel. Other ferritic components in the reactor coolant pressure boundary are fabricated from carbon steel materials.

Preheat temperatures employed for welding of low alloy steel meet or exceed the recommendations of ASME Section III, Subsection NA. Components were either held for an extended time at preheat temperature to assure removal of hydrogen, or preheat was maintained until post weld heat treatment. The minimum preheat and maximum interpass temperatures were specified and monitored.

All welds were nondestructively examined by radiographic methods. In addition, a supplemental ultrasonic examination was performed.

5.2.3.3.2.2 Control of Stainless Steel Weld Cladding of Low Alloy Steel Components Regulatory Guide 1.43

Regulatory Guide 1.43 does not apply to BWR components.

5.2.3.3.2.3 Control of Electroslag Weld Properties Regulatory Guide 1.34

No electroslag welding was performed on BWR components.

5.2.3.3.2.4 Welder Qualification for Areas of Limited Accessibility Regulatory Guide 1.71

There are few restrictive welds involved in the fabrication of BWR pressure boundary components. Welder qualification for welds with the most restricted access was accomplished by mock-up welding. Mock-ups were examined with radiography or sectioning.

TABLE 5.2-11

RCPB OPERATING THERMAL CYCLES

	No. of Events
Normal, Upset, and Testing - Conditions	
Bolt Up*	123
Design Hydrostatic Test	130
Startup (100°F/hr Heatup Rate)**	120
Daily Reduction to 75-Percent Power* and Control Rod Pattern Change*	10,400
Weekly Reduction to 50-Percent Power*	2,000
Feedwater Heater Loss, Partial Heater Bypass	70
HPCS Operation (10), SLC Operation (10)	20
Scram:	
Turbine Generator Trip, Feedwater On, Isolation Valves Stay Open	40
Other Scrams	140
Loss of Feedwater Pumps, Isolation Valves Closed	10
Reduction to 0-Percent Power, Hot Standby, Shutdown (100°F/hr Cooldown Rate)**	111 (each)
Unbolt	123
Emergency Conditions	
Scram:	
Reactor Overpressure with Delay Scram, Feedwater Stays On, Isolation Valves Stay Open	1
Automatic Blowdown	1
Single Safety/Relief Valve Blowdown	8
Improper Start of Cold Recirculation Loop	1
Sudden Start of Pump in Cold Recirculation Loop	1
Improper Startup with Reactor Drain Shut Off Followed by Turbine Roll and Increase to Rated Power	1
Faulted Conditions	
Pipe Rupture and Blowdown	1
ASME Hydrostatic Test	
1.25 x Design Pressure Hydrostatic Test (per NB 6222 and NB 3114)	10

* Applied to reactor pressure vessel only
** Bulk average vessel coolant temperature change in any 1-hour period

TABLE 5.2-12
DELETED

NOTE: The itemized examinations to be conducted in accordance with the commitments herein are delineated in the WNP-2 Preservice Inspection Program Plan (Ref. 5.2-6) for all preservice inspection commitments, and will subsequently be delineated in the WNP-2 Inservice Inspection Program Plan applicable to each inspection interval.

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 Liquid Control In	Active	SLC-V-7	9.3-13
	Active	SLC-V-4	9.3-13
	Active	SLC-V-6	9.3-13
	Inactive	SLC-V-8	9.3-13
Pump Description			
Recirculation Pump	Active	B35C001	5.4-2b

TABLE 5.4-4

SAFETY AND RELIEF VALVE DESCRIPTION

Safety and/or Relief Valve Identification	Description
B22F013	Main Steam Line Safety Relief Valve
E51F017	RCIC System Suction line
E51F018	RCIC Lube Oil Cooler Supply line
E51F033	RCIC Vacuum Tank
E12F055	RHR Condensing Mode Steam Supply line
RHR-RV-95	RHR Condensing Mode Steam Supply line
E12F036	RHR Condensing Mode Return line to RCIC
E12F005	Shutdown Cooling Supply line
E12F025	Shutdown Cooling Return line
E12F088	Suppression Pool Supply for Loop C
E12F030	RHR Flush line
RHR-RV-1	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
RWCU-RV-262	RWCU Filter Demineralizer Influent
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
C41F029	Standby Liquid Control Pump discharge line
RHR-RV-98	RHR Shutdown Cooling Supply line

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

LIST OF TABLES

When vessel pressure reaches 200 psid* the system reaches rated core spray flow. The HPCS motor size is based on peak horsepower requirements.

The elevation of the HPCS pump is sufficiently below the water level of both the condensate storage tanks and the suppression pool to provide a flooded pump suction and to meet pump NPSH requirements with the containment at atmospheric pressure and the suction strainer 50% plugged. The available NPSH has been calculated in accordance with Regulatory Guide 1.1, 1970, Rev. 0, and is 36'.

A motor-operated valve is provided in the suction line from the suppression pool. The valve is located as close to the suppression pool penetration as practical. This valve is used to isolate the suppression pool water source when HPCS system suction is from the condensate storage system and to isolate the system from the suppression pool in the event a leak develops in the HPCS System.

The HPCS pump characteristics, head, flow, horsepower, and required NPSH are shown in Figure 6.3-4.

The design pressure and temperature of the system components are established based on the ASME Section III Boiler and Pressure Vessel Code. The design pressures and temperatures, at various points in the system can be obtained from the information blocks on the HPCS Process Diagram, Figure 6.3-2.

A check valve, flow element and restricting orifice are provided in the HPCS discharge line from the pump to the injection valve. The check valve is located below the minimum suppression pool water level and is provided so the piping downstream of the valve can be maintained full of water by the discharge line fill system (see 6.3.2.2.5). The flow element is provided to measure system flow rate during LOCA and test conditions and for automatic control of the minimum low flow bypass gate valve. The measured flow is indicated in the main control room. The restricting orifice is sized during the pre-operational test of the system to limit system flow to acceptable values as described on the HPCS system Process Diagram.

*psid = differential pressure between the reactor vessel and the suction source.

A low flow bypass line with a motor-operated gate valve connects to the HPCS discharge line upstream of the check valve on the pump discharge line. The line bypasses water to the suppression pool to prevent pump damage from overheating when other discharge line valves are closed. The valve automatically closes when flow in the main discharge line is sufficient to provide required pump cooling.

To assure continuous core cooling, primary containment isolation does not interfere with HPCS operation.

The HPCS system incorporates relief valves to protect the components and piping from inadvertent overpressure conditions. One relief valve, set to relieve at 1100 psig with a capacity of 25 gpm is located on the discharge side of the pump downstream of the check valve to relieve thermally expanded fluid. A second relief valve is located on the suction side of the pump and is set at 100 psig with a capacity of 10 gpm.

The HPCS components and piping are positioned to avoid damage from the physical effects of design basis accidents, such as pipe whip, missiles, high temperature, pressure, and humidity.

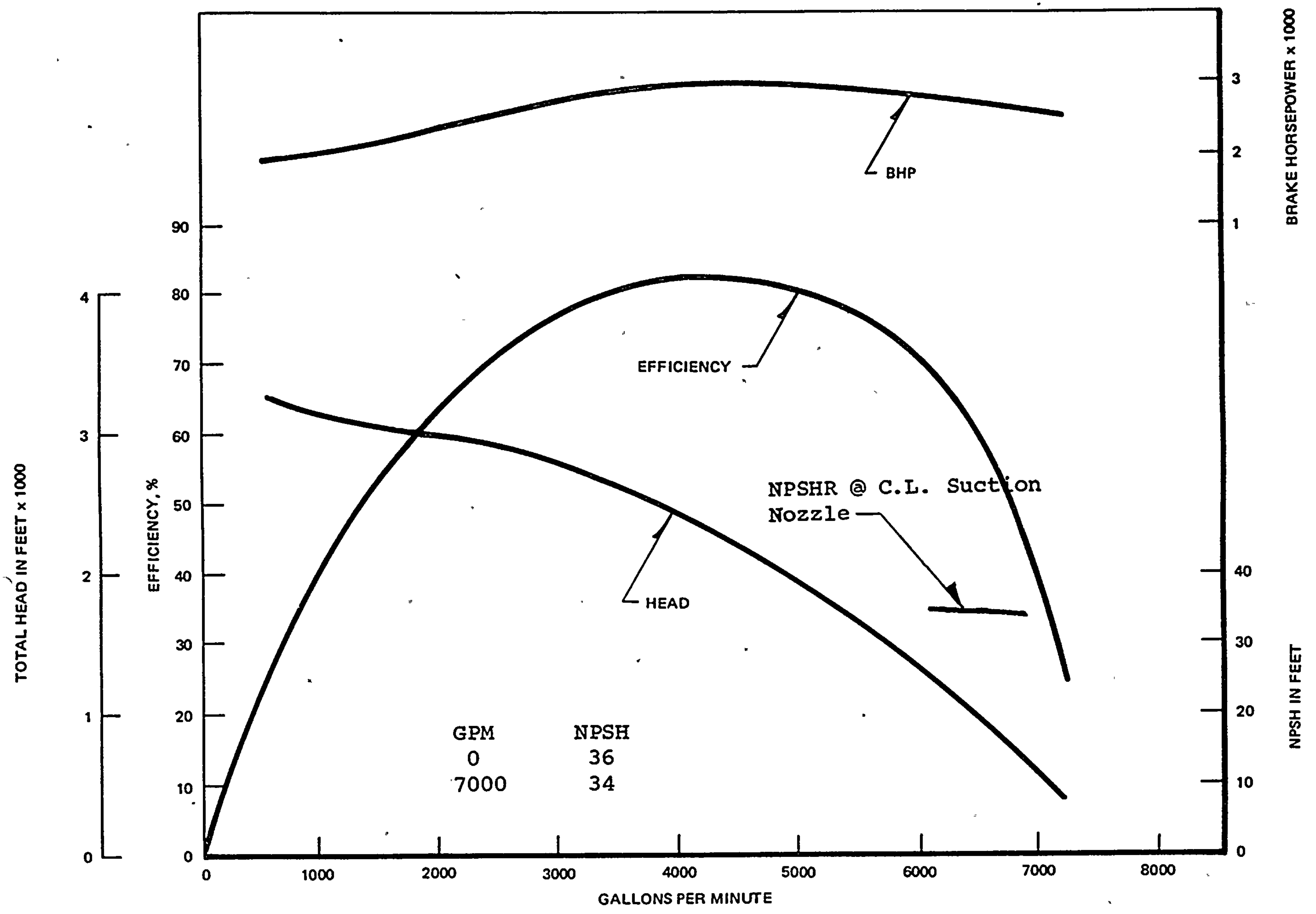
The HPCS equipment and support structures are designed in accordance with Seismic Category I criteria (see 3.2.1). The system is assumed to be filled with water for seismic analysis.

Provisions are included in the HPCS system which will permit the HPCS system to be tested. These provisions are:

a. All active HPCS components are testable during normal plant operation and/or during shutdown as discussed in 6.3.1.1.2m.

b. A full flow test line is provided to route water from and to the condensate storage tanks without entering the reactor pressure vessel. The suction line from the condensate tanks also provides reactor grade water to fully test the HPCS including injection into the RPV during shutdown.

c. A full flow test line is provided to route water from and to the suppression pool without entering the reactor pressure vessel.



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

HPCS PUMP CHARACTERISTICS (S/N 0473126)

FIGURE 6.3-4

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.

The MSIV-LCS is designed to be in accordance with IEEE Std. 279-1971 (Criteria for Protection Systems for Nuclear Power

Generating Stations) and IEEE 344-1971 (Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations.)

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.

Component	Malfunction	Consequences
Outboard MSIV	1 of 4 fails to close (excessive leakage)	Outboard LCS system functions to collect leakage through 3 sets of 2 MSIV's in series and one inboard MSIV, and deliver it to the SGTS for treatment. Inboard LCS is available to collect leakage from inboard MSIV's and deliver it to the SGTS for treatment.
Downstream Block Valves	Fails to close (excessive leakage)	Outboard LCS functions but capacity is insufficient to positively control MSIV leakage. Inboard LCS functions to collect leakage through inboard MSIV and deliver it to the SGTS for treatment.
Inboard LCS blower Inboard LCS emergency power Inboard LCS initiation switch	Fails to operate, or loss of power	Inboard LCS inoperative. Outboard LCS powered by other electrical division functions to collect leakage through 4 sets of 2 MSIV's in series and deliver it to SGTS for treatment.
Outboard LCS blower Outboard LCS power supply Outboard LCS initiation switch Outboard LCS bleed valve	Fails to operate, or loss of power	Outboard LCS inoperative. Inboard LCS powered by the other electrical division operates to collect leakage through 4 inboard MSIV's and deliver it to the SGTS for treatment.

Component	Malfunction	Consequences
Inboard LCS Bleed valve	Fails to open	Inboard LCS collects leakage from 3 of 4 inboard MSIV's. Outboard LCS functions to collect leakage through 4 of 4 sets of 2 MSIV's in series.
Inboard LCS Depressurization Valve	Fails to open.	One main steam line between MSIV's will not depressurize in time giving a false inboard MSIV excessive leakage signal, and resulting in isolation of ¼ of inboard LCS. 3/4 of inboard LCS continues to function.
Outboard LCS Depressurization Valve	Fails to close	Gas from the reactor building volume is recirculated through blower, decreasing its effective capacity. The LCS continues to function inboard.
Inboard LCS heater	Fails to heat	Leakage relative humidity will be high, condensation may plug pipe and decrease inboard LCS capacity. Outboard system continues to function.

6.7.4 INSTRUMENTATION APPLICATION

Refer to Figure 3.2-25 for instrumentation and control information.

6.7.4.1 System Description

6.7.4.1.1 Main Steam Line Leakage Control System (MSIV-LCS)

The MSIV-LCS is manually initiated after a loss-of-coolant accident (LOCA). The purpose of this system is to control and minimize the release of fission products which leak through closed main steam line isolation valves by directing leakage to the standby gas treatment system for processing prior to release to the atmosphere.

The MSIV-LCS is divided into two independent subsystems. The inboard (see Figure 3.2-25) subsystem is connected to the steam lines outside the primary containment between the inboard and outboard MSIV's. The outboard subsystem connection is downstream of the outboard MSIV's.

6.7.4.1.1.1 MSIV-LCS Instrumentation and Controls

6.7.4.1.1.1.1 Power Sources

The instrumentation and control of the MSIV-LCS are powered by 120 volts a-c Division 1 (inboard subsystem) and Division 2 (outboard subsystem) critical power supplies.

6.7.4.1.1.1.2 Equipment

6.7.4.1.1.1.2.1 General

The instrumentation components for the MSIV-LCS are located outside the steam tunnel. Cables connect the sensors and transducers to control circuitry within the logic panel. Functional test on the system instrumentation can be performed during normal reactor power operation. However, the MSIV-LCS isolation valves can only be tested one at a time. Inboard and outboard subsystem control and instrumentation are electrically and mechanically separated to assure that no single failure event can disable the MSIV-LCS. The MSIV-LCS is designed to operate from normal off-site auxiliary power sources or from divisional standby power supplies if off-site power is not available.

6.7.4.1.1.1.2.2 Initiating Circuits

The MSIV-LCS can be actuated manually after a LOCA has occurred, provided that the reactor and steamline pressure are below the pressure permissive interlock set points. The outboard subsystem is provided with one remote manual initiating switch, and the inboard subsystem is provided with one remote manual initiating switch.

The inboard subsystem has individually controlled process lines provided for each steam line.

When the inboard subsystem is initiated, the bleed valves and bypass valve open simultaneously, and the exhaust blower and the heater are energized. After one minute, if the steamline pressure is greater than 5 psig, the bleed valves will close.

If the pressure is 5 psig or less, the bleed valves will remain open. After another minute, the bypass valve is closed. The flow is thus routed through the flow element. After another ½ minute, the third timer closes the bleed valves if high leakage flow is present.

The outboard subsystem process lines, from each main steamline, are connected to a header leading to the depressurization and bleed off branch.

When the outboard subsystem is initiated, depressurization valves open and the exhaust blower is activated. When the steam lines have depressurized to approximately atmospheric pressure, the depressurization branch valves are closed and flow is diverted to the blower suction lines.

6.7.4.1.1.1.2.3 Logic and Sequencing

A LOCA is indicated by high drywell pressure and low, low vessel water level. After a LOCA has occurred, the MSIV-LCS system can be manually initiated.

Indicators for both reactor and steamline pressures for the inboard and outboard subsystems are available in the main control room.

6.7.4.1.1.1.2.4 Interlocks

Both the inboard and outboard subsystems are provided with reactor and steamline pressure interlocks to prevent inadvertent system initiation during normal reactor power operation. The inboard subsystem is in addition, provided with an interlock to prevent bleed valves from opening unless the respective inboard MSIV is closed.

6.7.4.1.1.1.2.5 Redundancy

The MSIV-LCS consists of two independent redundant subsystems; inboard and outboard. Either system may be manually initiated following a LOCA.

6.7.4.1.1.1.3 Specific Regulatory Requirements Conformance

6.7.4.1.1.1.3.1 Regulatory Guide 1.29, Revision 1

All instrumentation and controls are tested and qualified to meet Seismic Category I requirements and remain functional after a seismic event.

6.7.4.1.1.1.3.2 Regulatory Guide 1.75, Revision 0

The instrumentation and control devices for each subsystem are completely separated and independent. The system raceway groupings comply with the requirements of this regulatory guide. Each subsystem has a separate and independent control room panel.

6.7.4.1.1.1.3.3 Regulatory Guide 1.62, Revision 0

System initiation is manual from the control room. Interlocks are provided to prevent inadvertent manual initiation during normal reactor power operation.

6.7.4.1.1.1.3.4 Regulatory Guide 1.96, Revision 1

The MSIV-LCS is designed to comply with this regulatory guide, with the exception that the MSIV-LCS is not designed to reduce and control stem packing leakage or other direct leakage to the steam tunnel. Leakage from valve stem packing is addressed in 6.7.3m., while other direct leakage to the steam tunnel is addressed in 6.7.3.1.

6.7.4.1.1.1.4 Conformance to General Functional Requirements

6.7.4.1.1.1.4.1 IEEE 279-1971

6.7.4.1.1.1.4.1.1 General Functional Requirements (IEEE-279 Para. 4.1)

After a LOCA, the MSIV-LCS can be manually initiated from the control room by plant personnel. After initiation, the system is shut down by high steam line or reactor pressure to prevent excessive leakage to the SGTS.

6.7.4.1.1.1.4.1.2 Single Failure Criterion (IEEE-279 Para. 4.2)

The MSIV-LCS consist of two subsystem, inboard and outboard. The two subsystems feature separate and independent sets of controls and instrumentation and meet the single failure criterion.

6.7.4.1.1.1.4.1.3 Quality of Components and Modules (IEEE-279 Para. 4.3)

Components used in MSIV-LCS have been carefully selected on the basis of suitability for the specific application. The logic relays have been selected to ensure against significant deterioration during anticipated duty over the lifetime of the plant.

A quality control and assurance program is required to be implemented and documented by equipment vendors with the intent of complying with requirements set forth in 10CFR50 Appendix B.

6.7.4.1.1.1.4.1.4 Equipment Qualification (IEEE-279 Para 4.4)

Vendor certification is required to ensure that components operate in accordance with the requirements of the purchase specification.

6.7.4.1.1.1.4.1.5 Channel Integrity (IEEE-279 Para 4.5)

The MSIV-LCS is required to be operable under the environmental conditions noted in the design basis (6.7.1).

6.7.4.1.1.1.4.1.6 Channel Independence (IEEE-279 Para. 4.6)

Channel independence for sensors is provided by electrical and mechanical separation. Physical separation is maintained between inboard and outboard subsystem to increase reliability of operation. The MSIV-LCS is sufficiently separated to give a high degree of reliability.

6.7.4.1.1.1.4.1.7 Control and Protection System Interaction (IEEE-279 Para. 4.7)

There is no control and protection system interaction.

6.7.4.1.1.1.4.1.8 Derivation of System Inputs (IEEE-279 Para. 4.8)

All input signals to the instrumentation and control systems are derived from direct measurement of system variables.

6.7.4.1.1.1.4.1.9 Capability of Sensor Checks (IEEE-279 Para. 4.9)

The sensors which are used for inputs to the MSIV-LCS can be checked one at a time by application of simulated signals during normal plant operation.

6.7.4.1.1.1.4.1.10 Capability of Test and Calibration (IEEE-279 Para. 4.10)

All active components of the MSIV-LCS can be tested during plant operation. Valves can be tested by operating manual switches in the control room and observing indicating lights. Operation of the blowers and heaters by manual switches can be verified by dilution air flow and heater temperature indicators.

6.7.4.1.1.1.4.1.11 Channel Bypass or Removal from Operation (IEEE-279 Para. 4.11)

This system is manually initiated after a LOCA, and allows calibration and removal of one subsystem which will not affect the other subsystem.

6.7.4.1.1.1.4.1.12 Operating Bypasses (IEEE-279 Para. 4.12)

During system operation, the operating control system can be shutdown automatically by high pressure or high flow interlocks in the inboard subsystem. Otherwise, the trip is manual.

6.7.4.1.1.1.4.1.13 Indication of Bypasses (IEEE-279 Para. 4.13)

Motor operated valve closure can be prevented by shutting off electric power to the motor starters. This action will be annunciated in the control room to indicate power failure.

6.7.4.1.1.1.4.1.14 Access to Means for Bypassing (IEEE-279 Para. 4.14)

Access to switchgear and motor control centers is procedurally controlled by the following administrative means:

a. Lockable doors on the critical switchgear rooms

b. Lockable breaker control switch handles in the motor control centers

6.7.4.1.1.1.4.1.15 Multiple Set Points (IEEE-279 Para 4.15)

All set points are fixed.

6.7.4.1.1.1.4.1.16 Completion of Protection Action Once it is Initiated (IEEE-279 Para. 4.16)

The MSIV-LCS will not remain in operation after system initiation if excessive leakage, high reactor pressure, or high steamline pressure occurs.

6.7.4.1.1.1.4.1.17 Manual Actuation (IEEE-279 Para. 4.17)

The MSIV-LCS can be initiated manually, at system level, from the control room provided interlocks permit.

6.7.4.1.1.1.4.1.18 Access to Set Point Adjustments, Calibration and Test Points (IEEE-279 Para. 4.18)

The control system and its respective set points are accessible on the main control room panels. The sensors are accessible on the instrument racks during normal plant operation.

6.7.4.1.1.1.4.2 Not Used

6.7.4.1.1.1.4.3 IEEE-338, 1971

The MSIV-LCS is testable during reactor operation. The test completely tests each logic through the final actuators and demonstrates independence of subsystems. A failure of one subsystem while testing will not prevent the other subsystem from being initiated manually.

6.7.5 TESTS AND INSPECTIONS

6.7.5.1 Preoperational Tests

Preoperational tests are conducted prior to initial startup. The tests ensure functioning of all controls, instrumentation and all active components. Functional testing and flow measurements are accomplished with compressed air. Compressed air testing is conservative compared to operation of the system with steam blowdown under post accident conditions. For the hot steam blowdown case, compared to the preoperational test case with 70°F air, the space between MSIV's will depressurize faster and the 5 psig trip setting has additional margin against reclosure. System reference characteristics such as timer setpoints and flow rates are documented during the preoperational testing and are used as base points for measurements obtained in subsequent operational tests.

TABLE 7.3-27

MAIN STEAM LINE LEAKAGE CONTROL SYSTEM INSTRUMENTATION SPECIFICATIONS

FUNCTION	INSTRUMENT	INSTRUMENT RANGE (1)	TRIP SETTING (2)	MARGIN (3)	REQUIRED ACCURACY (4)	RESPONSE TIME (4)
Reactor Pressure Low	Press. Switch (MSLC-PS20) (MSLC-PS24) (MSLC-PS8A-D) (MSLC-FS7A-D)	-	35 psig	-	-	-
MSLC Header Pressure Low	Press. Switch (MSLC-PS 25)	-	0 psig	-	-	-
MSLC Header Pressure Low	Press. Switch (MSLC-PS 70A-D)	-	5 psig	-	-	-
MSLC High Flow	Flow Switch (MSLC-FS 3A-D)	-	505 CFH	-	-	-

NOTES FOR TABLE 7.3-27

(1) See Chapter 16, "Technical Specifications" for operational limits.

The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored.

(2) Trip settings shown are subject to change to comply with Chapter 16, "Technical Specifications".

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy.

Initial trip setting values are established from operating experience with similar size plants, and back up with analysis as necessary.

(3) See Chapter 16, "Technical Specifications" for instrument setpoint margins.

(4) Values shown are subject to change to comply with Chapter 16, "Technical Specifications".

Division	Application	Tray/Conduit Marking Characters	Inscription Character Colors	Background Colors
1	Power, Control & Instrumentation	Div. 1	Black	Yellow
2	Power, Control & Instrumentation	Div. 2	Black	Orange
3	Power, Control & Instrumentation	Div. 3	Black	Red
4	RPS Channel A1	R Ch A1	Red	Lt. Blue
5	RPS Channel A2	R Ch A2	Red	Green
6	RPS Channel B1	R Ch B1	Red	Drk. Blue
7	RPS Channel B2	R Ch B2	Red	Brown

Non-Class 1E equipment and cables have nameplates or tags with color coding as indicated below. Identification markers are of sufficient durability and at a sufficient number of points to facilitate initial verification that the installation is in conformance with the separation criteria. These cable tray and conduit markings are applied prior to cable installation.

Division	Application	Tray/Conduit Marking Characters	Inscription Character Colors	Background Colors
A	Power, Control & Instrumentation	Div. A	Black	Yellow
B	Power, Control & Instrumentation	Div. B	Black	Gold

Non-Class 1E cables that are routed in Class 1E raceways with Class 1E cables become "Associated Circuits." They are routed in accordance with the separation criteria defined in 8.3.1.4.

Cable marking for associated circuits consists of a black inscription (cable number) on a composite background of horizontal bands of the background colors of both divisions. For example, a background consisting of yellow and silver bands indicates that a non-Class 1E Division A cable is run in Division 1 cable tray or conduit, somewhere along its routing. Orange and gold bands indicate Division B cable in Division 2 tray or conduit.

Information concerning routing is given in Table 8.3-8 to illustrate the computer program used for cable identification and routing. Table 8.3-9 indicates sample cable routing schedules. Actual cable tray drawings for the reactor, control and radwaste buildings are shown in Figures 8.3 to 8.3-14 inclusive.

8.3.1.4 Independence of Electrical Divisions

Refer to 7.0 for electrical separation within the PGCC.

Separation of other electrical equipment and wiring is such that no single event will prevent any of the necessary safety functions from being performed.

Physical separation as a protection against common mode failure of critical power and instrumentation systems (Division 1 to 7) is achieved by spatial separation and/or physical barriers between the equipment and raceways serving different divisions. Spatial separation is preferred and is utilized wherever possible.

Separate cable trays and conduit systems are provided for each of the standby ac power systems (Divisions 1, 2 and 3).

Reactor protection systems (RPS Divisions 4, 5, 6 and 7) are run in totally enclosed raceway systems, except for the area immediately underneath the reactor.

The physical independence of electrical systems complies with the requirements of IEEE Std. 279-1971, IEEE Std. 308-1974 (IEEE Std. 308-1972 for HPCS system), General Design Criterion 17 and 21 and NRC Regulatory Guide 1.6, Revision 0.

TABLE 8.3-13

Motor and Motor Starter Voltage Requirements

Equipment	Nameplate Voltage	Min. Requirement Voltage	Min. Requirement % NPV	Max. Capability Voltage	Max. Capability % NPV
6.9kv Motors (Continuous Operation)	6600V	5940	90%	7260	110%
4.16kv Motors (Continuous Operation)	4000V	3600	90%	4400	110%
480V Motors (Continuous Operation)	460V	414	90%	506	110%
480V Starters (Pickup)	480V	374	78%	528	110%
480V Starters (Holding)	480V	264	55%	-	-

NOTES: 1. NPV indicates nameplate voltage of motors or starters for the particular voltage level.

AMENDMENT NO. 1
July 1978

TABLE 8.3-14

CLASS 1E AUXILIARY AC DISTRIBUTION SYSTEM (25kV MAIN GENERATING UNIT SUPPLY) EXPECTED VOLTAGES OVER GENERATOR VOLTAGE RANGE

	Minimum 25kV Generator Voltage (23.8kV)				Maximum 25kV Generator Voltage (26.3kV)			
	Steady State Loading		4.16kV Motor Starting		Steady State Loading		4.16kV Motor Starting	
System Level	Voltage[2]	% Motor NPV	Voltage[2]	% Motor NPV	Voltage[3]	% Motor NPV	Voltage[3]	% Motor NPV
4.16kV Swgr.	3761	94	3448	86	4399	109	3996	99
480 V Swgr.	424.8	92	389	84		Note 5		
480 V Motor[4] Terminals	422.8	91.9	387	84		Note 5		

NOTES:

1. NPV indicates nameplate voltage of motors for the particular voltage level.
2. Under plant maximum loading conditions.
3. Under plant minimum loading conditions.
4. Voltage at terminals of "worst case" motor.
5. Analysis for this case performed with 480V system unloaded to obtain maximum 4.16kV Swgr. level voltage.

AMENDMENT NO. 1
July 1978

TABLE 8.3-15

CLASS 1E AUXILIARY AC DISTRIBUTION SYSTEM (230kV GRID SUPPLY)
EXPECTED VOLTAGES OVER GRID VOLTAGE RANGE

	Minimum 230kV Grid Voltage (230kV)				Maximum 230kV Grid Voltage (240kV)			
	Steady State Loading		4.16kV Motor Starting		Steady State Loading		4.16kV Motor Starting	
System Level	Voltage[2]	% Motor NPV	Voltage[2]	% Motor NPV	Voltage[3]	% Motor NPV	Voltage[3]	% Motor NPV
4.16kV Swgr.	3732	93	3333	83	4189	104	3755	93
480 V Swgr.	416.7	90.5	372.1	80.8	479	104	429	93
480 V Motor[4] Terminals	415.0	90.2	370.4	80.5	468	101	418	90.8

NOTES:
1. NPV indicates nameplate voltage of motors for the particular voltage level.
2. Under plant maximum loading conditions.
3. Under plant minimum loading conditions.
4. Voltage at terminals of "worst case" motor.

AMENDMENT NO. 1
July 1978

TABLE 8.3-16

CLASS 1E AUXILIARY AC DISTRIBUTION SYSTEM (115kV GRID SUPPLY)
EXPECTED VOLTAGES OVER GRID VOLTAGE RANGE

	Minimum 115kV Grid Voltage (113kV)				Maximum 115kV Grid Voltage (122kV)			
	Steady State Loading		4.16kV Motor Starting		Steady State Loading		4.16kV Motor Starting	
System Level	Voltage[2]	% Motor NPV	Voltage[2]	% Motor NPV	Voltage[3]	% Motor NPV	Voltage[3]	% Motor NPV
4.16kV Swgr.	4024	100	3793	94	4432	110	4177	104
480 V Swgr.	454	98	426	92	505	109	476	103
480 V Motor[4] Terminals	452	98	424	92	503	109	474	103

NOTES:

1. NPV indicates nameplate voltage of motors for the particular voltage level.
2. Under plant maximum loading conditions.
3. Under plant minimum loading conditions.
4. Voltage at terminals of "worst case" motor.

AMENDMENT NO. 1
July 1978

TABLE 8.3-18 (Continued) Page 2 of 2

Swgr./MCC			DC Control Power Source	
Designation	Voltage[1]	Division	Panel[2]	Division
SL-61	480V	B	Note 4	
SL-62	480V	B	Note 4	
SL-63	480V	B	DP-S1-2C	B
SL-71	480V	1	DP-S1-1D	1
SL-73	480V	1	DP-S1-1D	1
SL-81	480V	2	DP-S1-2D	2
SL-83	480V	2	DP-S1-2D	2
MC-S1-1D	125VDC	1	DP-S1-1	1
MC-S1-2D	125VDC	2	DP-S1-2	2
MC-S2-1A	250VDC	1	DP-S1-1D	1
MC-S2-1B	250VDC	A	DP-S1-1C-1	A

NOTES:

1. Voltage is ac, unless otherwise noted.
2. See Figure 8.3-19
3. Control power for 480 volt ac motor control centers is obtained from control power transformers located in the units.
4. Swgr. contains manually operated breakers. All indication is remote powered via the non-Class 1E supervisory system.

TABLE 8.3-19

CLASS 1E AUXILIARY AC DISTRIBUTION SYSTEM (ALL SUPPLIES)
EXPECTED VOLTAGES UNDER DEGRADED SUPPLY CONDITIONS

System Level	Degraded Class 1E Bus Supply (2870V)[3] Voltage[2]	% Motor NPV[1]
4.16kV Swgr.	2870	71.8
480 V Swgr.	332	72.1
480 V Motor Terminals	323	70.2

NOTES: 1. NPV indicates nameplate voltage of motors for the particular voltage level.

2. Under plant maximum loading condition.

3. Corresponds to 69% of nominal 4.16kV bus voltage (undervoltage relay setpoint.)

9.2.6 CONDENSATE SUPPLY SYSTEM

9.2.6.1 Design Bases

The condensate supply system (COND) is designed to:

a. Store and provide a condensate supply to the reactor core isolation cooling (RCIC) system, the high pressure core spray (HPCS) system, and the RHR loops.

b. Maintain an adequate level of condensate in the condenser hotwell.

c. Provide a condensate supply for the control rod drive pumps.

d. Provide makeup water to the spent fuel pool.

e. Provide condensate for various radwaste processes.

f. Facilitate testing and/or flushing of the high pressure core spray, low pressure core spray, residual heat removal, and the reactor core isolation system.

g. Receive and accommodate a surge volume for condensate returned to the storage tanks after treatment in the liquid radwaste system.

h. System piping is designed to ANSI B31.1. Condensate storage tanks are designed to ASME Section III, Class 3 requirements. System piping inside the reactor building is Seismic Category I. All other piping and system pumps are designed to Seismic Category II requirements. The radwaste building condensate supply pump and the condensate filter demineralizer backwash pump are designed to ASME Code, Section III. The reactor building condensate supply pump is designed to the Standards of the Hydraulic Institute.

9.2.6.2 System Description

The demineralized water system and the liquid radwaste system are the primary sources of makeup water to the condensate storage tanks.

The condensate supply system is shown on Figure 9.2-9. The system consists of two storage tanks (COND-TK-1A, COND-TK-1B) each with a nominal capacity of 400,000 gallons and equipped with electric heaters, a reactor building condensate supply pump (COND-P-3), a radwaste building condensate supply pump (COND-P-4), a condensate filter demineralizer backwash pump (COND-P-5), and necessary piping and instrumentation. The tanks are manufactured with a design pressure of atmospheric plus full static head and maximum design temperature of 140°F. Minimum operating temperature of the tanks is 40°F. The tanks are designed to withstand a wind load of 20 psf on the vertical projected area of the tank and a snow load of 20 psf.

The radwaste building condensate supply pump and the condensate filter demineralizer backwash pump each are designed to supply 1535 gpm at 185 ft total head. The radwaste condensate supply pump has a secondary operating point of 500 gpm at 220 ft total head. The reactor building condensate supply pump is designed to supply 200 gpm at 220 ft total head. All three pumps are designed to operate satisfactorily at temperatures between 40°F and 104°F and humidity between 20% and 90%.

A minimum inventory of 135,000 gallons in the condensate storage tanks is reserved for the RCIC and HPCS pumps. This assures the immediate availability of a sufficient quantity of condensate for emergency core cooling and reactor shutdown as discussed in 9.2.6.3.

Makeup for the condenser hotwell is gravity fed from the storage tank. Bleedoff water from the condensate system is returned to the storage tanks from the discharge of the condensate demineralizer.

A separate line is provided to supply the control rod drive pumps with condensate. Condensate is supplied for various reactor building services, including fuel pool makeup by the reactor building condensate supply pump. The condensate storage tank can be drained to the condenser hotwell. Inadvertant overflow of the tanks is collected in the concrete retaining basin surrounding the tanks. This water can be drained to the radwaste system for processing if sampling indicates that the water is radioactively contaminated. Rain water collected in the retaining basin can be drained to the storm sewer or the radwaste system if necessary for processing.

10.3.6.2 Materials Selection and Fabrication

The requirements for welding the main steam supply system components and steam piping from the reactor to the turbine generator are in accordance with ASME Section III, October 1973. The welding requirements for all other piping is in accordance with ANSI B31.1, October 1973 (See 3.2).

For low alloy steel piping designed in accordance with ASME Section III or ANSI B31.1 (October 1973), ASME Section III (October 1973) Appendix D or Table 131 of ANSI B31.1 (October 1973) covering non-mandatory preheat procedures are applicable to all classes of welds. The control of preheat temperature for welding of low-alloy steel are in accordance with Regulatory Guide 1.50.

Procedure qualifications for welding of austenitic stainless steel components include the following requirements:

a. The welding procedure is designed to avoid sensitization of the weld joint area and is in accordance with Regulatory Guide 1.44.

b. The test weld closely simulates the heat transfer properties of the actual welds to be performed.

c. In addition to the evaluation of the test welds specified in ASME Section IX (October 1973), the weld area is examined for susceptibility to intergranular corrosion in accordance with ASTM A 262 (Practice E or A) or A 393 (October 1973).

d. Controls are exercised to assure that nonmetallic thermal insulation for austenitic stainless steel are in accordance with Regulatory Guide 1.36.

e. The control of stainless steel welding conforms to Regulatory Guide 1.31.

Cleaning of components in the main steam system is in accordance with ANSI N45.2.1 (October 1973) or ASTM A380-57 (October 1973) for stainless steel surfaces and Regulatory Guide 1.37.

Welding procedure qualifications and welder performance qualifications are in accordance with ASME Code, Section IX and addenda (October 1973) and the requirements of ASME Section III (October 1973). In addition to the aforementioned code, the following requirements also apply. All welding is done to qualified procedures by welders qualified to these procedures. Welding is done with a weld filler material procedure which includes cleanup of unused weld filler material at the end of each shift. All recent contracts, except for monitoring and certifying requirements pertaining to welder qualification for areas of limited accessibility, conform to Regulatory Guide 1.71. Contracts prior to Regulatory Guide 1.71 meet the intent of this guide. Where welding is performed on low or high alloy steel piping in an area where accessibility is limited such that the welder cannot view the weld melt puddle directly, or must use extension rods, or must bend the electrode, the procedure and performance qualifications require a mock-up to simulate this limitation.

b. Airborne Radiation Monitors

Airborne radioactivity monitoring for plant personnel protection and surveillance utilizes: fixed location, continuous particulate monitors which include continuous iodine samplers; portable continuous particulate monitors with continuous iodine samplers positioned at specific work sites; and particulate and iodine grab samples taken before and during specific jobs.

The installed continuous particulate monitoring system was designed for responsive personnel protection and plant surveillance. The five installed particulate monitors measure particulate activity levels in the Radwaste, Reactor and Turbine Building ventilation exhaust air, and furnish alarm and recording signals to the main control room. These monitors draw 5 cfm of ventilation air through a moving filter tape. A shielded beta detector is used with an efficiency of approximately 30% and an external background of less than 40 cpm/mr/hr. The resultant sensitivity of the system is 438 cpm above background after one hour of sampling at a 1×10^{-10} μCi/cc concentration in air.

The actual ability of a ventilation exhaust monitoring system to detect one MPC_a in a specific space is dependent upon the following factors:

1. Flow rate ratio (flow of air from a specific confined space/flow rate of bulk ventilation system exhaust)

2. Particulate activity of ventilation system air

3. MPC_a of the specific confined space (radionuclide composition)

Normal plant conditions are expected to yield a bulk ventilation exhaust air concentration (primarily short-lived fission product daughters and natural activity) of $1 - 3 \times 10^{-10}$ Ci/cc. The MPC_a for normal plant air is expected to be greater than 6×10^{-8} Ci/cc, and the ventilation monitoring system will be able to detect one MPCa hour in all locations for normal airborne mixture background concentrations. At high activity

concentrations in the bulk ventilation exhaust air the ability to detect one MPC_a concentration in areas with low ventilation exhaust flow rates may be compromised. Under these conditions, corrective actions including assessment by portable sampling system results and portable monitoring activities will establish activity levels in all normally occupied areas which have potential for abnormal airborne activity.

Table 12.3-2 illustrates the detection capabilities of the fixed location particulate monitoring system for various ratios of isolated area ventilation flow rates compared to the bulk ventilation exhaust flow rate from the reactor building.

In the Radwaste Building, the potentially contaminated areas normally entered by people would be those corridors adjacent to radioactive liquid and gaseous waste processing systems equipment such as demineralizers, concentrators, waste storage tanks, recombiners, driers, moisture separators and charcoal holdup vessels. Assuming that exfiltration from any one of the process systems to a normally entered corridor was sufficient to attain MPC_a levels for Cs-137 in that corridor, the dilution ratio would approach a factor of 10 to 100. For the worst case (100 to 1 ratio of bulk ventilation flow rate to corridor flow rate), Cs-137 at MPC_a would be detected within one hour on the continuous particulate monitor. If exfiltration from a process vessel cubicle was sufficient to produce MPC_a levels in an adjoining corridor, it is more probable that the normal cubicle flow rate input to the bulk ventilation flow would produce a prior distinguishable countrate ramp.

In the Turbine Building, airborne contamination is most likely to arise from nuclear steam leaks or off-gas processing systems piping. Heater bay areas, steam jet air ejector rooms, turbine areas and steam driven feedwater pump areas have the potential for airborne contamination. The areas have individual ventilation exhaust rates in excess of 5000 cfm and Cs-137 MPC_a concentrations originating in these areas would give a continuous air monitor response ramp which is distinguishable within one hour.

Each of the continuous particulate monitors has an associated iodine sampling cartridge which is counted weekly for baseline and surveillance information. This cartridge and iodine sampled/ collected with portable sampling devices will be analyzed in the plant laboratory counting facility when abnormal airborne activity levels are signaled by a continuous particulate monitoring system. At a 2 cfm air flow rate through an iodine sampling cartridge, iodine present at an occupational MPC_a concentration of 9×10^{-9} μCi/cc would be quantitatively observable within a one-minute sample interval. Bases for this assessment are a ten-minute count time on a 12 - 15% Ge(Li) detector system having an overall efficiency of about 1% when source and geometry considerations are included. The information presented for detecting one MPC_a concentration for Cs-137 in areas having a low ventilation flow rate can also be applied to the iodine case. One MPC_a of iodine can be ascertained within a one-hour sampling period with a dilution factor of greater than 100. The routine weekly analysis of integrated iodine samples of Radwaste, Reactor and Turbine Building ventilation air will permit observation of small iodine inputs. When these inputs are significant, a particulate and iodine sampling program is initiated to establish the source point.

With the exception of outage periods, continuous particulate monitoring is reasoned to be the most responsive personnel protection and internal plant surveillance mechanism available. In addition, all tasks with potential for generating airborne contamination will be performed only when authorized by a Radiation Work Permit (RWP).

The RWP assesses the radiological hazards, establishes additional monitoring and sampling requirements and, if necessary, specifies required engineering control and/or respiratory protection.

During outages, the above airborne monitoring system will be augmented by additional iodine sampling (continuous and grab) on the refueling floor since airborne iodine concentrations can exist in the absence of particulate contamination.

Table 12.3-2

Particulate Concentration at Various Flow Rate Ratios Necessary
To Increase Reactor Building Exhaust Activity By 25%

Bulk Reactor Bldg. Exhaust, Particulates µCi/cc	Count Rate, Including External Background cpm	Concentration, µCi/cc[1], at Listed Flow Rate Ratios[2] to Produce a 25% Increase in Count Rate				
		0.10	0.05	0.02	0.01	0.004[3]
3×10^{-11}	250	7.5×10^{-11}	1.5×10^{-10}	3.8×10^{-10}	7.5×10^{-10}	1.9×10^{-9}
1×10^{-10}	550	2.5×10^{-10}	5×10^{-10}	1.1×10^{-9}	2.5×10^{-9}	6.3×10^{-9}
3×10^{-10}	1445	7.5×10^{-10}	1.5×10^{-9}	3.8×10^{-9}	7.5×10^{-9}	1.9×10^{-8}
1×10^{-9}	4580	2.5×10^{-9}	5×10^{-9}	1.1×10^{-8}	2.5×10^{-8}	6.3×10^{-8}

(1) Cs-137 is used as the reference nuclide; MPC_a for soluble Cs = 6×10^{-8} µCi/cc.

(2) Flow Rate Ratio = flow rate of specific area ÷ flow rate of total Rx Bldg exhaust.

(3) A flow rate ratio of 0.004 corresponds to the CRD Room with a 400 cfm flow rate compared to a 97,000 cfm reactor building exhaust flow rate.

(BLANK)

15.6.5.5.1 Design Basis Analysis

The methods, assumptions, and conditions used to evaluate this accident are in accordance with those guidelines set forth in the NRC Standard Review Plan 15.6.5 and Regulatory Guides 1.3 Rev. 0 and 1.7 Rev. 2. The specific models, assumptions, and computer code used to evaluate this event based on the above criteria are presented in Reference 15.6-4. Specific values of parameters used in this evaluation are presented in Table 15.6-12.

15.6.5.5.1.1 Fission Product Release from Fuel

It is assumed that 100% of the noble gases and 50% of the iodine are released from an equilibrium core operating at a power level of 3468MWt for 1000 days prior to the accident. While not specifically stated in Reg. Guide 1.3 the assumed release of 100% of the core noble gas activity and 50% of the iodine activity implies fuel damage approaching melt conditions. Even though this condition is inconsistent with operation of the ECCS system (see 6.3), it is assumed applicable for the evaluation of this accident. Of this release, 100% of the noble gases and 50% of the iodine become airborne. The remaining 50% of the iodine is removed by plateout and condensation; therefore, it is not available for airborne release to the environment. The activity airborne in the containment is presented in Table 15.6-13.

15.6.5.5.1.2 Fission Product Transport to the Environment

The transport pathway consists of leakage from the containment to the secondary containment by several different mechanisms and is discharged to the environment through the standby gas treatment system (SGTS) at an elevated location. The SGTS filter efficiency for iodine removal is assessed at 99%. The mechanisms for leakage from the primary containment are discussed below:

a. Containment Leakage - The design basis leak rate of the containment and its penetrations (See 6.2.6) is .5% per day for the duration of the accident. This leakage is to the secondary containment and from there to the environment via the SGTS. No credit is taken for mixing and holdup within the secondary containment.

b. Leakage from engineered safety feature (ESF) components outside the primary containment - all ESF equipment which circulates primary coolant or suppression pool water during the course of the postulated accident is located within the secondary containment so that any leakage from the pressure barriers for these systems is into the secondary containment atmosphere and is therefore processed by the SGTS prior to release to the environment. Due to the higher SSW pressure any leakage through the RHR heat exchangers would be from the service water side to the ECCS side.

c. Hydrogen Purge - Since the hydrogen recombining system consists of two 100% redundant recombiners, no hydrogen purge is required nor assumed throughout the post accident period.

d. Leakage from the main steam isolation valve leakage control system (MSIV-LCS). The MSIV-LCS routes any leakage through the MSIVs to an area serviced by the SGTS. Assuming the MSIVs leak at 11.5 SCFH per valve, leakage past the inboard MSIVs is conservatively estimated to begin 2 hours after the accident. The airborne fission products are assumed to be uniformally mixed in the drywell air volume neglecting the suppression pool air volume. For conservation this leakage is assumed to be in addition to the .5% containment leakage discussed in paragraph "a" above. For maximum leakage through each mainstream line, (4 MSIVs or 46 scfh), this correlates to 0.23 weight %/day leak rate.

e. Bypass Leakage - As described in 6.2.3.2 and 6.2.3.3, 0.74 scfh primary containment leakage is assumed to bypass the secondary containment and leak directly to the environment.

Fission product release to the environment based on the above assumptions is given in Table 15.6-14 and 15.

15.6.5.5.1.3 Results

The calculated exposures for the design basis analysis are presented in Table 15.6-16 and are well within the guidelines of 10CFR100.

TABLE 15.6-12 Page 1 of 2

LOSS-OF-COOLANT ACCIDENT - PARAMETERS

TABULATED FOR POSTULATED ACCIDENT ANALYSES

	Design Basis Assumptions	Realistic Basis Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level	$3468MW_t$	$3468MW_t$
B. Burn-up	NA	NA
C. Fuel damaged	100%	0
D. Release of activity by nuclide	Table 15.6-14 & 15	Table 15.6-19 & 20
E. Iodine fractions		
(1) Organic	4%	1%
(2) Elemental	91%	99%
(3) Particulate	5%	0
F. Reactor coolant activity before the accident	NA	15.6.5.5.2.1
II. Data and assumptions used to estimate activity released		
A. Primary containment leak rate (%/day)	0.5	0.5
B. Secondary containment leak rate (%/day)	NA	100
C. Valve movement times	NA	NA
D. Adsorption and filtration efficiencies (%)		
(1) Organic iodine	99	99
(2) Elemental iodine	99	99
(3) Particulate iodine	NA	NA
(4) Particulate fission products	NA	NA

TABLE 15.6-12 (Continued)

	Design Basis Assumptions	Realistic Basis Assumptions
II. Data and assumptions used to estimate activity released		
E. Recirculation system parameters		
(1) Flow rate (CFM)	NA	NA
(2) Mixing efficiency	NA	NA
(3) Filter efficiency	NA	NA
F. Containment spray parameters (flow rate, drop size, etc.)	NA	NA
G. Containment volumes	NA	NA
H. All other pertinent data and assumptions	None	None
III. Dispersion Data		
A. Boundary and LPZ distance (m)	1950/4827	1950/4827
B. X/Q's for time intervals of		
(1) 0-2 hr - SB/LPZ	7.50×10^{-5}/ 2.80×10^{-5}	7.50×10^{-5}/ 2.80×10^{-5}
(2) 2-8 hr - LPZ	2.80×10^{-5}	2.80×10^{-5}
(3) 8-24 hr - LPZ	3.45×10^{-6}	3.45×10^{-6}
(4) 1-4 days - LPZ	1.59×10^{-6}	1.59×10^{-6}
(5) 4-30 days - LPZ	1.02×10^{-6}	1.02×10^{-6}
IV. Dose Data		
A. Method of dose calculation	Reference 15.6-4	Reference 15.6-2
B. Dose conversion assumptions	Reference 15.6-4	Reference 15.6-2
C. Peak activity concentrations in containment	Table 15.6-13	Table 15.6-18
D. Doses	Table 15.6-16	Table 15.6-21

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

Regulatory Guide 1.88, Rev. 2, October 1976

Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records.

Regulatory Guide Intent:

This guide describes an acceptable method of complying with the NRC's regulations for collection, storage, and maintenance of quality assurance records.

Application Assessment:

Assessed Capability in Design.

Compliance or Alternate Approach Statement:

The identified Boiling Water Reactor Quality Assurance Program utilized on this facility reflects compliance with the provisions of NRC regulations and the NRC regulatory guide or NRC-approved alternate position.

General Compliance or Alternate Approach Assessment:

Reference compliance assessment for Regulatory Guide 1.28.

Specific Evaluation Reference:

Information was provided at the PSAR stage.

Similar Application Reference:

Similar application has not been utilized on other projects.

Regulatory Guide 1.89, Rev. 0, November 1974

Qualification of Class 1E Equipment for Nuclear Power Plants

Regulatory Guide Intent:

Regulatory Guide 1.89 endorses both the requirements and recommendations of IEEE Standard 323-1974. "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations". Regulatory position stipulations are also included.

This regulatory guide is applicable to all Class 1E electrical equipment.

Application Assessment:

Assessed Capability in Design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design and/or equipment utilized in this facility is in compliance with the intent of the subject regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

All environmental and seismic qualification testing of Class 1E equipment within the NSSS scope of supply was in compliance with IEEE 323-1971 and IEEE 344-1971 which were NRC accepted standards.

Specific Evaluation Reference:

Refer to 3.10, 3.11 and 7.1.2.4.

Similar Application Reference:

Similar application was utilized on Zimmer and LaSalle.

General Compliance or Alternate Approach Assessment (Cont.)

where

R = Combined Response

R_i = Response in the i^{th} mode

n = Number of Modes considered in the analysis

Closely spaced modes are not accounted for as required by the guide because the design was significantly developed prior to issuance of the guide.

Specific Evaluation Reference:

Refer to 3.7.3.6 and 3.7.3.7.

Similar Application Reference:

Similar application was utilized on LaSalle and Zimmer.

Regulatory Guide 1.68, Rev. 1, January 1977

Initial Test Programs for Water-Cooled Reactor Power Plants

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to the WNP-2 initial test program since Revision 0 of this regulatory guide is committed to in FSAR 14.2.7. However, WNP-2 complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

Refer to FSAR Chapter 14 for description of initial testing program and to 14.2.7 and Appendix C.2.0 for statements concerning compliance with Regulatory Guide 1.68, Rev. 0. Revision No. 1 of this guide in general clarifies Revision No. 0 and therefore there are no exceptions to the intent of this procedure.

Specific Evaluation Reference:

FSAR 14.2.7 and Appendix C.2.0 discussion Reg. Guide 1.68, Rev. No. 0.

Regulatory Guide 1.68, Rev. 1, January 1977 - Preoperational and Initial Startup of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants.

Compliance or Alternate Approach Statements:

WNP-2 complies with the intent of the guidance set forth in this Regulatory Guide by an alternate approach.

General Compliance or Alternate Approach Assessments:

The preoperational testing and the initial Startup testing as described in FSAR, Chapter 14, complies with the intent of this Regulatory Guide. However, due to the limitations of the auxiliary steam supply system, the confirmation that the feedwater pumps satisfy required head, flow rate and suction head will not occur until the startup phase of the initial test program when the normal steam supply is available to the feedwater pump turbines.

Specific Evaluation Reference: FSAR 14.2.12.1.1

Regulatory Guide 1.68.2, Rev. 0, January 1977 - Compliance or Alternate Approach Statement:

WNP-2 complies with the intent of the guidance set forth in this Regulatory Guide by an alternate approach.

General Compliance or Alternate approach assessment:

The startup test described in FSAR 14.2.12.3.28 complies with the Regulatory Guide with the following exceptions:

a. The test will be initiated by scramming plant from the control room versus a location outside the control room as described in Section C.3 of the Regulatory Guide. This exception is made to better simulate the actual procedure which would be followed if a control evacuation were to occur. The capability to scram the reactor outside the control room exists; for example, tripping the RPS MG sets.

b. The Cold Shutdown Demonstration Procedure as described in Section C.4 of the Regulatory Guide may not be performed immediately following the demonstration of achieving and maintaining safe hot standby from outside the control room. Rather this cooldown portion may be performed when cooldown is required during the course of the normal power ascension test program. Although this is an exception to Regulatory Guide 1.68.2, Rev. 0, Revision 1 of this Guide contains provisions for a delay in the demonstration of cooldown.

Specific Evaluation Reference: FSAR 14.2.12.3.28, 7.4.1.4.

Regulatory Guide 1.69, Rev. 0, December 1973

Concrete Radiation Shields for Nuclear Power Plants

Compliance or Alternate Approach Statement:

WNP-2 complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

Although the regulatory guide was promulgated after design and specification implementation of the engineering criteria, the recommended design and construction practices specified in the regulatory guide are documented in codes and specifications which were used in the development of the engineering criteria and contract specifications.

Specific Evaluation Reference:

Refer to 12.3.2.

Regulatory Guide 1.89, Rev. 0, November 1974

Qualification of Class 1E Equipment for Nuclear Power Plants

Compliance or Alternate Approach Statement:

> This regulatory guide is not applicable to WNP-2 since it applies to the evaluation of construction permit applications docketed after July 1, 1974. However, WNP-2 complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Statement:

> (To be provided at a later date)

Specific Evaluation Reference:

> (To be provided at a later date)

Regulatory Guide 1.90, Rev. 0, November 1974

In-Service Inspection of Prestressed Concrete Containment Structures with Grouted Tendons

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to WNP-2 because WNP-2 does not have a prestressed concrete containment structure with grouted tendons.

General Compliance or Alternate Approach Assessment:

Not applicable.

Specific Evaluation Reference:

Not applicable.

Regulatory Guide 1.97, Rev. 0, December 1975

Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to WNP-2 since it applies to the evaluation of construction permit applications docketed on or after August 1, 1976.

General Compliance or Alternate Approach Assessment:

WNP-2 provides sufficient instruments in the main control room to monitor plant variables and systems during and following an accident. The instrumentation is qualified to remain functional during the worse case environmental conditions that it must monitor. The indicators and recorders are not seismically qualified. Means are provided to monitor the primary containment atmosphere, the spaces containing components for recirculation of loss of coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from postulated accidents.

Specific Evaluation Reference:

Refer to 7.5.

Regulatory Guide 1.100, Rev. 0, March 1976

Seismic Qualification of Electric Equipment for Nuclear Power Plants

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to WNP-2 since it applies to the evaluation of construction permit applications docketed after November 15, 1976.

General Compliance or Alternate Approach Assessment:

(To be provided at a later date)

Specific Evaluation Reference:

(To be provided at a later date)

Regulatory Guide 1.128, Rev. 0, April 1977

Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants.

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to WNP-2 since it applies to the evaluation of construction permit applications docketed after December 1, 1977. However, WNP-2 complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

Safety-related battery installation design criteria conforms to IEEE Standard 484-1975. In addition, HYDROCAPS (catalyst battery caps) are provided to preclude discharge of combustible gases into the battery room area. A Class 1E ventilation system is also provided which is capable of limiting hydrogen concentrations (Neglecting HYDROCAPS) to 1%.

Storage prior to installation was not in strict compliance with subsection 5.1.3 "Storage" of the subject regulatory guide. However, preoperational tests will establish whether or not any damage or loss of capacity resulted from storage.

Specific Evaluation Reference:

8.3.2.1.5
8.3.2.1.6
8.3.2.2.1.1
8.3.2.2.1.2

Regulatory Guide 1.129, Rev. 0, April 1977

Maintainance, Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants

Compliance or Alternate Approach Statement:

(To be provided at a later date)

General Compliance or Alternate Approach Assessment:

(To be provided at a later date)

Specific Evaluation Reference:

(To be provided at a later date)

Figure Number	Title	Engineering Dwg. No.
2.1-3	Overall Site Plan	C025
2.1-4	Plant Plot Plan	SS053
2.3-1	Overall Site Plan	C020

LIST OF NRC QUESTIONS

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
005.001	2	7
010.001	1	1
010.002	1	1
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	5
010.011	9	9
010.012	2	5
010.013	2	5
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
010.020	1	5
010.021	1	13

LIST OF NRC QUESTIONS

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
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
010.029	2	5
010.030	1	5
010.031	1	5
010.032	1	5
010.033	1	5
010.034	1	5
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	1
022.009	1	1
022.010	1	13

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
022.011	1	1
022.012	1	3
022.013	1	3
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	3
022.023	1	3
022.024	1	11
022.025	1	3
022.026	1	3
022.027	1	13
022.028	1	3
022.029	1	3
022.030	1	3
022.031	2	5
022.032	2	5
022.033	1	5

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
022.034	1	5
022.035	2	5
022.036	1	5
022.037	2	5
022.038	1	5
022.039	2	7
022.040	1	5
022.041	1	5
022.042	1	5
022.043	2	5
022.044	1	5
022.045	1	5
022.046	1	5
022.047	1	5
022.048	11	5
022.049	4	5
022.050	2	5
022.051	1	5
022.052	1	5
031.001(a)	1	0
031.001(b)	1	0
031.001(c)	1	0
031.001(d)	1	0

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.001(e)	1	0
031.001(f)	1	0
031.001(g)	1	0
031.001(h)	1	0
031.001(i)	2	0
031.001(j)	1	13
031.001(k)	1	13
031.001(l)	1	0
031.001(m)	1	0
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
031.001(s)	1	0
031.001(t)	1	0
031.001(u)	1	0
031.001(v)	1	0
031.001(w)	1	0
031.001(x)	1	0
031.001(y)	1	0
031.001(z)	1	0
031.001(aa)	1	0

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.001(bb,cc)	1	0
031.001(dd)	1	0
031.001(ee)	1	0
031.001(ff)	1	0
031.001(gg)	1	0
031.001(hh)	1	0
031.001(ii)	1	0
031.002	1	0
031.003	1	0
031.004	1	0
031.005	1	0
031.006	2	10
031.007	1	0
031.008	1	0
031.009	4	0
031.010	4	0
031.011	1	0
031.012	1	0
031.013	1	0
031.014	1	3
031.015	3	0
031.016	2	0
031.017	2	0

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.018	2	0
031.019	1	0
031.020	1	0
031.021	2	0
031.022	1	0
031.023	1	0
031.024	1	0
031.025	2	0
031.026	1	10
031.027	1	0
031.028	1	0
031.029	1	0
031.030	2	0
031.031	1	0
031.032	1	0
031.033	1	0
031.034	2	0
031.035	1	0
031.036	1	0
031.037	1	0
031.038	1	0
031.039	2	0
031.040	1	0

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.041	1	0
031.042	1	0
031.043	1	0
031.044	2	0
031.045	1	0
031.046	1	0
031.047	1	3
031.048	3	0
031.049	1	0
031.050	3	0
031.051	1	0
031.052	1	0
031.053	2	0
031.054	1	0
031.055	1	13
031.056	1	10
031.057	1	10
031.058	3	11
031.059	2	10
031.060	1	3
031.061	1	3
031.062	1	3
031.063	1	3

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.064	1	3
031.065	1	3
031.066	2	3
031.067	1	3
031.068	1	3
031.069	1	3
031.070	5	3
031.071	1	3
031.072	1	3
031.073	1	3
031.074	1	3
031.075	1	3
031.076	3	5
031.077	1	3
031.078	2	3
031.079	2	3
031.080	7	10,13
031.081	2	10
031.082	1	10
031.083	1	10
031.084	1	10
031.085	1	10
031.086	1	10

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.087	5	10
031.088	1	10
031.089	1	10
031.090	1	10
031.091	1	10
031.092	2	10
031.093	1	10
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	25	10
031.101	1	10
031.102	1	10
031.103	3	10
031.104	2	10
031.105	2	10
031.106	1	10
031.107	1	10
031.108	2	10
031.109	1	10

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.110	1	10
031.111	1	10
031.112	1	10
031.113	1	10
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
040.009	1	0
040.010	2	0
040.011	1	0
040.012	1	0
040.013	1	0
040.014	1	0
040.015	1	5
040.016	1	0
040.017	1	0
040.018	1	0
040.019	1	0

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
040.020	7	0
040.021	1	0
040.022	1	0
040.023	1	0
040.024	2	0
040.025	1	0
040.026	2	5
040.027	1	0
040.028	1	0
040.029	1	0
040.030	1	0
040.031	1	0
040.032	1	0
040.033	1	0
040.034	2	7
040.035	1	7
040.036	2	7
040.037	1	7
040.038	1	7
040.039	4	7,11
040.040	1	7
040.041	1	7
040.042	1	7

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
040.043	1	7
040.044	2	7
040.045	5	7,11
040.046	1	7
040.047	1	7
040.048	1	7
040.049	1	7
040.050	1	7
040.051	1	7
040.052	1	7
040.053	2	7
040.054	1	7
040.055	1	7
040.056	1	7
040.057	1	7
040.058	1	7
040.059	1	7
040.060	2	7
040.061	1	7
040.062	1	7
040.063	1	7
040.064	1	7
040.065	1	7

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
040.066	1	7
040.067	1	7
040.068	1	7
040.069	1	7
040.070	1	7
040.071	1	7
040.072	1	7
040.073	1	7
040.074	1	7
110.001	1	9
110.002	1	9
110.003	1	9
110.004	1	9
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

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
110.015	1	9
110.016	1	9
110.017	1	9
110.018	1	9
110.019	4	9
110.020	1	9
110.021	1	9
110.022	1	9
110.023	1	9
110.024	1	9
110.025	1	9
110.026	1	9
110.027	1	9
110.028	2	9
110.029	1	9
110.030	2	9
110.031	1	9
110.032	1	9
110.033	1	9
110.034	1	9
110.035	1	9
110.036	2	9
110.037	2	9

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
121.001	3	5
121.002	8	5
121.003	1	5
121.004	1	5
121.005	1	5
121.006	1	5
121.007	1	13
121.008	10	5,10
121.009	1	5
121.010	2	7
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
130.011	1	8
130.012	1	8
130.013	1	8

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
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	8
130.021	1	8
130.022	1	8
130.023	1	8
130.024	1	8
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
130.032	1	8
130.033	1	8
130.034	1	8
130.035	4	8
130.036	1	8

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
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	3	12
130.046	3	12
130.047	1	12
130.048	2	12
130.049	3	12
210.001	1	0
211.002	1	8
211.003	1	8
211.004	1	8
211.005	1	8
211.006	1	8
211.007	1	8
211.008	2	8
211.009	1	8
211.010	2	8

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.011	1	8
211.012	2	8
211.013	1	8
211.014	2	8
211.015	1	8
211.016	3	8
211.017	2	8
211.018	1	8
211.019	3	8
211.020	1	8
211.021	2	8
211.022	1	8
211.023	1	8
211.024	1	8
211.025	1	8
211.026	1	8
211.027	4	8
211.028	2	8
211.029	1	8
211.030	1	8
211.031	4	8,10
211.032	1	8
211.033	4	8

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.034	1	8
211.035	1	8
211.036	1	8
211.037	1	8
211.038	3	8
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	11
211.050	1	11
211.051	13	11
211.052	1	11
211.053	1	11
211.054	1	11
211.055	1	11
211.056	2	11

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.057	1	11
211.058	3	11
211.059	1	11
211.060	1	11
211.061	2	11
211.062	1	11
211.063	1	11
211.064	1	11
211.065	2	11
211.066	2	11
211.067	1	11
211.068	1	11
211.069	1	11
211.070	1	11
211.071	1	11
211.072	1	11
211.073	1	11
211.074	1	11
211.075	1	11
211.076	2	11
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	11
211.085	4	11
211.086	1	11
211.087	2	11
211.088	2	11
211.089	5	11
211.090	1	11
211.091	1	11
211.092	2	11
211.093	1	11
211.094	1	11
211.095	1	11
211.096	1	11
211.097	1	11
211.098	1	11
211.099	1	11
211.100	1	11
211.101	2	11
211.102	1	11

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.103	1	11
211.104	1	11
211.105	3	11
211.106	1	11
212.001	1	3
212.002	1	3
212.003	2	5
212.004	1	3
221.001	2	7
221.002	1	7
221.003	1	7
221.004	1	7
221.005	1	7
221.006	1	7
221.007	1	7
221.008	1	7
221.009	1	7
221.010	2	7
221.011	1	7
221.012	3	7
221.013	1	7
222.001	1	8
222.002	3	8

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
222.003	1	8
222.004	1	8
231.001	1	3
231.002	3	3
231.003	1	5
232.001	1	3
232.002	1	5
232.003	1	5
232.004	1	5
232.005	1	5
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

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
312.015	1	8
312.016	1	5
312.017	1	11
312.018	1	11
312.019	1	5
321.001	1	1
321.002	1	1
321.003	1	13
321.004	1	5
321.005	4	5
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

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
331.014	1	1
331.015	2	5
331.016	1	5
331.017	1	5
331.018	1	5
331.019	1	5
331.020	1	5
331.021	1	13
331.022	1	5
331.023	1	5
331.024	1	5
360.001	1	1
360.002	1	1
360.003	1	1
360.004	3	10
360.005	6	10
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

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
362.008	1	5
362.009	2	5
371.001	1	1
371.002	1	1
371.003	1	1
371.004	1	1
371.005	1	1
371.006	1	5
371.008	1	5
371.009	1	5
371.010	1	5
371.011	1	5
371.012	1	5
371.013	1	5
371.014	1	5
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

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
372.009	1	5
372.010	1	5
372.011	1	5
372.012	1	5
372.013	1	5
372.014	7	5
372.015	4	5
372.016	5	5
372.017	1	5
422.001	2	7
422.002	1	7
422.003	1	7
422.004	4	7
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

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
423.006	2	1
423.007	1	1
423.008	1	1
423.009	1	1
423.010	1	1
423.011	4	7
423.012	1	7
423.013	1	7
423.014	1	7
423.015	1	7
423.016	2	7
423.017	1	7
423.018	1	7
423.019	7	7
423.020	4	7
423.021	3	7
423.022	1	7
423.023	11	7
423.024	1	7
423.025	1	7
423.026	1	7
423.027	1	7
423.028	1	7

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
423.029	1	7
432.001	7	5
432.002	2	5
432.003	2	5
432.004	3	5
432.005	1	5
432.006	1	5
432.007	1	5
432.008	1	5
432.009	1	5
432.010	1	5
432.011	2	5
432.012	1	5
432.013	2	5
432.014	1	5
432.015	1	5
432.016	1	5
441.001	3	7
441.002	1	7
441.003	1	7
441.004	1	7
441.005	1	7

Q. 022.007

Provide the secondary containment pressure time response for the design basis accident. List and discuss all assumptions made in this analysis.

Response:

See revised 6.2.3.3 and Table 6.2-29.

Q 022.8

Identify the manufacturer of the hydrogen recombiner and describe the test program which demonstrates that the hydrogen recombiner will perform as required in the containment environment following a postulated loss-of-coolant accident. Provide the systems quality group classification including that of the hydrogen analyzer.

Response:

The hydrogen recombiner was manufactured by Air Products and Chemicals, Inc. The tests are briefly discussed in 6.2.5.4 and detailed, supplemental information is being provided by separate letter to the NRC. The analyzer, including quality group classification, is discussed in 7.6.1.13.8. All pressure containing equipment, including piping between components, is considered an extension of the containment and is classified code Quality Group B, as discussed in 6.2.5.2.3.

Q 22.025

Identify the location of the hydrogen sampling points in the drywell and the suppression chamber. Identify the location of the suction and discharge points of the combustion gas control system with respect to local structures and equipment.

Response

Please refer to FSAR Table 7.6-12 and Figures 6.2-32 through 6.2-35 for the requested information.

Q 22.026

In accordance with Appendix J of 10CFR50 we require that containment isolation valves for those systems not vented and drained during Type A tests, are to be tested in accordance with Section III.C of Appendix J and those results are to be reported to the Commission.

Response:

In general the containment isolation valves for those systems not vented and drained during Type A tests are being Type C tested. The exceptions to this are listed in the response to question 22.10 along with the justification. The results of these tests will be reported to the Commission as stated in FSAR 6.2.6.4.

Q. 022.027
(6.2.6)

Augment Table 6.2-16 to provide the information requested in Section 6.2.4.2, "Systems Design" of Regulatory Guide 1.70.

Response

Tables 6.2-13, 6.2-16 and 7.3-13 have been combined into one table, number 6.2-16. This table has been further expanded to include all the information required in Section 6.2.4.2 of Regulatory Guide 1.70. Chapter 6 has been revised to reflect the revised Table 6.2-16.

Q 22.028

Several of the loads presented in Table 3.4-1 of the Plant Design Assessment Report (DAR) have been generated using computer codes which have not been reviewed by the NRC staff. Provide a complete description of your method of analysis for all codes presented in Appendix D of the DAR.

Response

Three computer codes have been used to develop short term LOCA hydrodynamic loads for WNP-2 plant assessment. The three codes are the downcomer vent clearing analytical model computer code VENT, the pool swell analytical model computer code SWELL, and the LOCA bubble charging analytical model computer code BUBBLE. Complete documentation on each of the above three codes is provided or referenced in Appendix D of the DAR (see below for specific references). Documentation on the WNP-2 load calculation procedure is provided in 3.2.1 of the DAR.

1. Vent Code

   a) Assumptions

      See Section D.2 of the DAR.

   b) Equations

      See Figure D-1 of the DAR.

   c) Methodology

      See Figure D-1 of the DAR.

2. Swell Code

   The assumptions, equations, and methodology for the Swell Code are identical to that described in Reference D-1.

3. Bubble Code

   The assumptions, equations, and methodology for the Bubble Code are identical to that described in Reference D-7.

Q 22.043

Tables 6.2-13, 6.2-16, and 7.3-13 of the FSAR indicate that a check valve outside the containment is considered as a containment isolation valve for the minimum flow at the pumps in the reactor heat removal system (X-47, 48), vacuum relief from secondary containment (X-66, 67, 119) and a process sample line (X-69D). Provide justification for this design approach.

Response

Tables 6.2-13, and 7.3-13 have been deleted. See question 22.027 for revised Table 6.2-16.

There are check valves inboard of the isolation valves on the minimum flow line from the RHR pumps (X-47, X-48). These valves are built to the same standards as the isolation valves and will, if necessary, isolate the minimum flow line from the primary containment; however, these check valves are not considered containment isolation valves. Please see revised Table 6.2-16.

There are no check valves on the process sample line (X-69D). The notation, C.V., which was previously used was not intended to designate check valve. Revised Table 6.2-16 now clearly designates the valve types for the isolation valves on penetration X-69D.

Both isolation valves on the reactor building to wetwell vacuum relief lines (X-66, X-67, and X-119) are located outside the wetwell to improve valve operability (see Note 17 of revised Table 6.2-16). The reactor building to wetwell vacuum relief system is required to prevent excessive negative pressures in the primary containment under certain postulated conditions (see 6.2.1.1.4). The disc in the check valve is maintained in the close position during normal operation by means of a spring actuated lever arm and magnets embedded in the periphery of the disc. The magnetic and spring forces are overcome, and the disc starts to open, when the pressure differential across the valve exceeds 0.2 psid. The check valves have position indication lights which can alert the operators to the fact that a check valve is not fully closed. The operator can then remotely shut the valve by means of a pneumatic operator. The operating switch is spring-return to neutral. The air supply to these valves is Quality Class I.

Revised Table 6.2-16 now lists check valves outside containment for CIA to the inboard MSIV's and MS relief valves (X-56) and CIA and nitrogen backup to the ADS valves (X-89A and B). These check valves are in all three cases inboard of motor operated, isolation globe valves. The check valves are located outside of the primary containment to improve valve operability as discussed in Note 17 of Table 6.2-16.

Q. 031.001 (a)

The FSAR contains many conflicting statements and incomprehensible statements which must be resolved prior to the start of our review. For each of the items below, provide a response which is responsive to the NRC staff's need for information satisfying the requirements of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," (Revision 2).

a. Clarify the discrepancy between the designation of the main generator as "Unit 1" in 1.2.2.7.1 and the plant designation as WNP-2.

RESPONSE:

The text of 1.2.2.7 has been revised to clarify this discrepancy.

Q. 031.001 (b)

Clarify the discrepancy between the reference to "three trip logics" in 3.1.2.3.2.1 and the description of the reactor protection system in 7.2.1.1.3.2 and Figures 7.2-3 and 7.2-11.

RESPONSE:

3.1.2.3.2.1, third paragraph which refers to "three trip logics" is in error, and has now been revised.

Q. 031.001 (c)

Clarify the discrepancy between the 0.25 to 33 Hz seismic range given in 3.10.1.2.3.1 and the 5 to 33 Hz values given in 7.3.2.1.2.3.1.

RESPONSE:

Some seismic vibration tests on certain equipment were performed starting from 5 Hertz because of test machine limitations. High test table displacements beyond the capacity of the testing machine are required to obtain higher input accelerations (g-levels) at frequencies below 5 Hertz. In all these cases it was shown by calculations or by a resonant frequency search test or based on similarities with previously qualified equipment that no resonant frequencies exist below 5 Hertz; consequently, tests are completed to required levels at 33 Hertz and at all resonant frequencies in between. This would qualify the equipment seismically. 7.3.2.1.2.3.1 has been revised to eliminate the discrepancy, and will cross-reference to 3.10 for seismic data.

Q. 031.001 (d)

Indicate whether the reference to Figure 7.2-4 which is made in 7.2.1.1.3.2, is intended to be Figure 7.2-3.

RESPONSE:

7.2.1.1.4.3 has been revised to provide the correct figure references.

Q. 031.001 (g)

Clarify the discrepancies between the main steamline isolation valve (MSIV) response time given in 5.4.5.3, 6.3.3.3.1, and 7.3.2.1.2.3.1.5.2.1.1 and the accident analyses of Chapter 15.

RESPONSE:

No discrepancies exist between the stated MSIV response times. The main steamline isolation valve closure time is in the range from 3 to 5 seconds. As indicated in 7.3.2.2.2.3.1.1 the minimum MSIV closure time is 3 seconds. 5.4.5.3 assumes a maximum MSIV response time of 5.5 seconds. This assumes the maximum closure time of 5 seconds for MSIV plus .5 seconds for instrument response to initiate MSIV closure. The analyses of Chapter 15 assume the worst case response of 3 seconds for MSIV closure. All closure times given are within the specified 3 to 5 second range of the MSIV.

The MSIV response time for full closure is set prior to plant operation. The minimum set time for closure is 3.0 seconds and the maximum set time is 5.5 seconds (includes as much as 0.5 seconds for instrument response). The difference in response time reported in the various paragraphs is due to whether the minimum or maximum response time is conservative to the results of the analysis under consideration. 5.4.5.3, 15.6.4 list the maximum value while 7.3.2.2.2.3.1.1 lists the minimum value. In summary, the conservative response time for MSIV closure was used in each of the identified sections of the FSAR.

Q. 031.001 (h)

Clarify the discrepancy between the description of the MSIV solenoid valves which is given on Page 6.2-55 in 6.2.4.2 and that presented in 7.3.1.1.2.2 and Figure 7.3-19.

RESPONSE:

7.3.1.1.2.4 and Figures 7.3-2 and 7.3-4 are correct. MSIV solenoid valves are no longer discussed in 6.2.4.

Q. 031.001 (r)

The discussion in Chapter 7 regarding compliance with the requirements of 4.20 of IEEE Std. 279-1971 relating to the readout of information, is inadequate. Revise the FSAR to describe the equipment and systems which provide the operator with accurate, complete, and timely information pertinent to the status of the information channel and to generating station safety.

RESPONSE:

The information required to identify compliance with the requirements of 4.20 of IEEE Std. 279-1971, i.e., descriptions of the equipment and systems which provide the operator with accurate, complete and timely information, is in general provided in the sections titled, "Reactor Operator Information," under the appropriate system in the FSAR. The FSAR has been revised to include the appropriate references within the subparagraphs specifically addressed to 4.20. Following is a list of paragraphs in Chapter 7, specificallly related to 4.20 and the paragraphs which are referred to:

Paragraph 7.2.2.1.2.3.1.20 (Reactor Protection System) refers to paragraph 7.2.1.1.6.1.

Paragraph 7.3.2.1.2.3.1.20 (ECCS) refers to paragraphs 7.3.1.1.1.3.11.2 (HPCS), 7.3.1.1.1.4.11.2 (ADS), 7.3.1.1.1.5.11.2 (LPCS), and 7.3.1.1.1.6.11.2 (LPCI).

Paragraph 7.3.2.2.2.3.1.20 (PCRVICS) refers to paragraphs 7.3.1.1.2.4.1.2.9, 7.3.1.1.2.4.1.7.9.2, and 7.3.1.1.2.13.2.

Paragraph 7.3.2.3.2.3.1.20 (MSLIV-LCS).

Paragraph 7.3.2.4.3.1.20 (CSCS) refers to 7.3.1.1.4.12.2.

See 7.3.2.5.20 for SSW system information readout.

See 7.3.2.6.20 for main control room HVAC system information readout.

See 7.3.2.7.20 for reactor building ventilation and pressure control system information readout.

See Chapter 8 for standby power system information readout.

See 7.3.2.9.20 for SGTS information readout.

See 7.3.2.10.20 for CIA system information readout.

See 7.3.2.11.20 for CAC system information readout.

Paragraph 7.4.2.1.2.3.1.20 (RCIC) refers to paragraph 7.4.1.1.5.2.

Paragraph 7.4.2.2.2.3.1.20 (SLCS) refers to paragraph 7.4.1.2.5.2.

Paragraph 7.6.2.8.2 (Recirculation Pump Trip System) refers to paragraph 7.6.1.8.5.2.

7.5.2, Safety Related Display System Analysis, contains further discussion of compliance with 4.20 of IEEE Std. 279-1971.

Q. 031.001 (u)

Clarify the references in 7.2.1.1.4 to Table 3.11-1 for the reactor and control building environments.

RESPONSE:

7.2.1.1.5 of revised text states that the environmental conditions for the drywell, the containment, and the turbine building are given in Tables 3.11-1, 3.11-2, and 3.11-3.

Tables 3.11-1, 3.11-2 and 3.11-3 have been revised to provide the referenced information.

Q. 031.001 (v)

Clarify 7.3.1.1.2.3 to clearly state where the pressure, temperature, and water level sensors and racks are located.

RESPONSE:

The text of 7.3.1.1.2.3 and Table 7.3-46 has been revised to incorporate the response to this question.

Q. 031.001 (w)

Clarify the discrepancy between the discussion of compliance with the requirements of 4.10 of IEEE Std. 279-1971 in 7.3.2.1.2.3.1.5.2.1.10 and the discussion of conformance with the staff positions in Regulatory Guide 1.22 which follows it.

RESPONSE:

Even though the mainsteam line high temperature sensors are inaccessible during plant operation, they can be tested while the plant is operating by cross comparison between channels.

7.3.2.2.2.1.2 reads in part as follows:

> The main steamline isolation logic, and sensor devices (except the MSL high temperature sensors) may be tested from the sensor device to one of the two solenoids. Both solenoids must be deenergized to verify that there are no obstructions to the valve stem at full power. A reduction in power is necessary to avoid reactor scram before performing a valve closure using two, fast acting, main solenoids.

7.3.2.1.2.3.1.5.2.1.10 is replaced in the revised text of the FSAR by 7.3.2.2.2.3.1.10, which reads in part as follows:

> All active components of the primary containment isolation control system, can be tested and calibrated during plant operation with the exception of the main steamline high temperature sensors. By observing the contact action on an HFA type relay during a channel trip condition, the actual drop-out can be verified when deenergized.

Q. 031.001 (x)

Several of the figures in the FSAR are illegible or are missing component designations and are, therefore, unacceptable. Revise the FSAR to eliminate both illegible and/or unintelligible figures.

RESPONSE:

The FSAR has been revised to eliminate illegible or unintelligible figures.

Q. 031.001 (y)

Clarify the discrepancy in the response time of the reactor core isolation cooling system given in 7.4.1.1.3.1 and that given in 7.4.1.1.3.5 of the FSAR.

RESPONSE:

Both 7.4.1.1.3.2 and 7.4.1.1.3.6 have been revised to provide a 30 second response time for the RCIC system.

Q. 031.001 (z)

Clarify the discrepancy between the description in 7.4.1.1.3.5 of the automatic transfer of the suction for the reactor core isolation cooling system which satisfies the design shown in Figure B-25 and the manual transfer shown in Figures 7.4-1a and 7.4-2b which does not satisfy the assumptions made in the Operational Analysis contained in Appendix B.

RESPONSE:

The operational analysis in Appendix B (now Appendix 15A) did not assume automatic transfer of the RCIC suction. 7.4.1.1.3.5 and Figure B-25 are replaced in the revised text of the FSAR by 7.4.1.1.3.6 and Figure 15.A.6-40 respectively. However, additional discussion about this transfer is provided in the response to Question 031.015.

Q. 031.001 (aa)

Resolve the contradiction between 7.4.1.1.3 and 7.4.2.2.2 so as to provide a clear statement of the conditions under which the isolation valves of the reactor core isolation cooling system will be required to operate and the seismic and environmental conditions for which these valves are qualified.

RESPONSE:

7.4.1.1.3 and 7.4.2.1.2.3.1.4 have been revised to provide a clear statement of the conditions under which the isolation valves for the RCIC will be required to operate. The valves are qualified for operation during those environmental conditions listed in Tables 3.11-2 and 3.11-3. Table 3.2-1 states the seismic classification of the valves to be Category I.

Q 031.001 (bb, cc)

bb. Clarify the discrepancy between 7.6.1.1.3.1, which states that the refueling interlock system is single failure proof and the design of the reactor manual control system which has a single rod position input path and a single refueling equipment output path.

cc. Modify the FSAR to include concise definitions of the worst case environmental conditons under which the refueling interlocks will be required to operate.

RESPONSE:

bb. The refueling interlock system provides two independent channels of instrumentation where either loss of signal or trip signal from either channel will prohibit any further rod movement. The reactor manual control system (RMCS) design has two inputs, and the interlock status is merged into a serial data transmission. Even though there is only one rod position path, this information must be in a precise format before it is accepted for any rod movement. If the information is in the proper format, it is then determined if the coded information will allow rod movement. The transmitted information is echoed back by the hydraulic control units and compared to ensure proper transmission. The RMCS provides two refueling equipment output paths.

7.6.1.1.3.1 and 7.6.1.1.3.2 have been revised to clarify the discrepancy.

cc. The refueling platform is not required to operate during the run mode and would not be affected in an accident situation. The worst case environmental conditions under which the refueling interlocks will be required to operate are the normal reactor building conditions listed in Table 3.11-3.

Q. 031.001 (dd)

Clarify the discrepancy between 7.6.1.6.7.1.1.4 and the design of the reactor manual control system which is presented in 7.7.1.1.

RESPONSE:

The isolation, separation, and redundancy features discussed in 7.6.1.6.7.1.1.4 are features of the rod block monitor. The RBM does interface with the reactor manual control system, but is separate from it.

7.6.1.6.7.1.1.4 appears in Rev. 2 of the FSAR as 7.6.1.5.7.1.4. A new subsection has been added which reads:

> "7.7.1.2.3.2.3 Rod Block Interlocks
>
> The rod block functions are discussed in 7.6.1.5.7."

The reference in 7.7.1 to the appropriate subsection for the RBM will eliminate the discrepancy in 7.7.1.

(BLANK)

Q. 031.001 (ee)

Identify and justify all design differences between the reference rod block monitor design described in 7.6-2 and that associated with the new solid state reactor manual control system which is described in 7.7.1.1.

RESPONSE:

The reference (Hatch 2) rod block monitor (RBM) described in Reference 7.6-2 and the RBM associated with the reactor manual control system described in 7.7.1.1 are essentially identical.

The comparison of the reference (Hatch 2) and the Zimmer RBM designs were discussed at length between GE and the NRC on April 18, 1977, at GE, San Jose, Ca. The question is considered to have been resolved for Zimmer.

The designs for the RBM and WNP-2 and Zimmer are identical.

Q. 31.001 (ff)

Identify the specific bus which powers the reactor manual control system and the refueling interlock.

RESPONSE:

1. The reactor manual control system may be powered from either Instrument bus A or B (PP-7A-A or PP-8A-A). Since the system is not safety related, the choice of bus has no effect on performance of the system. See 7.7.1.2.2.

2. The busses which supply power to the refueling interlock are 120 VAC instrument busses. See 7.6.1.1.2.

Q. 031.001 (gg)

Quantify the recirculation system low water level interlock range, setpoint, and accuracy.

RESPONSE:

The interlock can be set from instrument zero to 60 inches; instrument zero is at 527.5 inches above invert (vessel bottom).

The interlock setpoint is at 31.5 inches above instrument zero.

The accuracy is ± 0.5% of full scale.

Q. 031.001 (hh)

Clarify the discrepancy between 7.7.1.2.3.3.9 and 7.7.1.2.4 of the FSAR with respect to the required motion of the flow control valves under accident conditions.

RESPONSE:

The recirculation flow control system is not required for safety purposes, nor required to operate during or following a design basis accident. However, during operation, the valve actuator has an inherent rate limiting feature that will limit the resulting rate of change of core flow and power to within acceptable limits in the event of an upscale or downscale failure of the valve position or velocity control system.

See revised 7.7.1.3.3.4.9 and 7.7.1.3.4.

Q. 031.001 (ii)

Indicate where in Chapter 7 the information concerning the instrumentation for the reactor building closed cooling water system is located.

RESPONSE:

The text of 7.6.1.15 has been added to respond to this question.

Q. 031.002
(T1.7-1)

Provide process instrumentation and control. logic, wiring, and electrical schematic drawings for: (a) the overpressurization protection (relief) system; (b) the reactor protection system; (c) the leakage detection temperature monitors; (d) the neutron monitor auxiliary trip units; and (e) the reactor protection system instrument racks which are located in the reactor building.

RESPONSE:

The following WNP-2 drawings are applicable:

	System	GE MPL Number	Drawing Identification
A.	ADS Logic (FCD)	B22 1030	731E788
	ADS Elementary	B22 1060	807E180TC
	NBS P & ID	B22 1010	732E103
B.	RPS IED	C72 1010	732E170AD
	RPS Elementary	C72 1050	807E178TC
C.	Leak Detection IED	E31 1010	732E191AD
	Leak Detection Elementary	E31 1050	807E154TC
D.	Neutron Mon. Aux. Trip Schematic	-----	127D1861
	Neutron Mon. Aux. Trip Wiring	-----	195B9206
E.	RPS Inst. Racks (Not in FSAR, but available and auditable at GE)		
	Reac Wtr Lvl and Press Panel Connection	H22 P004	127D1827TC
		H22 P005	127D1827TC
	Reac Wtr Lvl and Press Panel Connection	H22 P026	828E390TC
	Reac Wtr Lvl Press Panel Connection	H22 P027	828E387TC

Logic diagrams and elementary (electrical schematic drawings) for BOP systems are listed in Table 1.7-1.

Q. 031.005 (RSP)
(3.10)
(3.11.3)
(7.2.2.2)

A request for documentation of the seismic and environmental qualification of Class 1E equipment is contained in 3.10 and 3.11 of the Standard Format. This request is applicable to all engineered safety features, reactor protection systems and all supporting systems. It is not limited to those particular safety-related supporting systems supplied by General Electric. Accordingly, we require you to provide the information requested in 3.10 and 3.11 of the Standard Format for all Class 1E systems in accordance with: (a) the NRC Staff positions stated in Attachments 1 and 2; (b) IEEE Std. 323-1971; and (c) IEEE Std. 344-1971. Identify and justify any exceptions.

RESPONSE:

The requested seismic and environmental qualification data for Class IE equipment for all engineered safety features, reactor protection systems and all support systems is provided in the revised text and tables of 3.10 and 3.11.

For additional discussion refer to 7.2.1.2.7 (RPS), 7.3.1.1.1 (ECCS), 7.3.1.1.2.12 (PC and RVIS), and 7.3.1.1.4.11 (RHR/ Containment Spray).

Additional discussion of conformance to IEEE 323-1971 is provided in 7.2.2.1.2.3.4 (RPS), 7.1.2.5.4 and 7.3.2.1.2.3.1.4 (ECCS), and 7.3.2.2.2.3.1.4 and 7.1.2.5 (PC and RVIS).

Additional discussion of conformance to IEEE 344-1971 is provided in 7.2.2.1.2.3.7 (RPS), 7.3.2.1.2.3.5 (ECCS), and 7.3.2.2.2.3.5 (PC and RVIS).

Q. 031.006

The staff requests that the following information regarding the qualification test program be provided for Class 1E equipment: (a) the equipment design specification requirements; (b) the test plan; (c) the test set up; (d) the test procedures; (e) the acceptability goals and requirements; and (f) the test results.

Provide this information for each of the following Class 1E components: (a) the 4.16 kV switchgear SM 7; (b) the damper operator for WMA-V-52C; (c) the fan WMA-FN-52B; (d) the logic equipment for the standby gas treatment system; (e) the diesel-generator control equipment; (f) the 480 V ESS switchgear MC-7A-A; and (g) the solenoid valve for the main steam line isolation valves.

Response:

An extensive seismic and environmental review program is presently underway encompassing BOP and NSSS scope, with a planned completion date in December 1980.

Within the BOP scope, the equipment documentation has been extracted from the contract files, copied and categorized for easy retrieval. Within the NSSS scope, contract negotiations are underway with GE to perform a similar function.

A list of all Class 1E equipment including splices, terminal blocks, termination cabinets and connectors is presently being compiled. This list will contain the following information:

1. Equipment location

2. Safety functional requirement

3. Manufacturer & Model No.

4. Qualification Method (test-analysis)

5. Environmental Extremes

6. Identification and location of qualification documents

The documentation will be reviewed to insure that the testing was adequate to meet the seismic and environmental extremes under which the equipment must either function or not fail.

A composite list will be included in the FSAR as equipment tables in 3.10 (seismic) and 3.11 (environmental).

The extensive review program underway will also satisfy the requirements of IE Circular 78-08, address the degree of compliance with NUREG-0588, and establish the conservativeness of seismic tests and analysis performed to IEEE-344, 1971.

The detailed results of this review will be made available to NRC SQRT and environmental review personnel during their site documentation reviews.

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(f) The drive flow demand limiters are adjustable. The high signal limiter is to establish the maximum drive flow demand limit needed for the upper end of the automatic load-following range. The low signal limit is determined from a core stability criterion and defines the lower end of the automatic load-following range. There is no flow limit, and the valve can be closed to its minimum position when the master controller is in manual mode operation. (See 7.7.1.3.3.4.4).

(g) A limiting function is provided for one feedwater pump trip flow runback. An electronic limiter with reasonable range adjustment is provided in each main flow control loop. This limiter is normally held bypassed by auxiliary devices such as relay contacts. When one feedwater pump trip coincides with reactor low water level alarm, the main regulating valve control signal is limited to close the valve to the desired position. (See 7.7.1.3.3.4.8).

See also 5.4.1.3.1.

Q. 031.011
(6.2.2.3)

The description of operator actions is incomplete. Provide the following information: (a) describe the operator actions which are necessary to establish containment spray; and (b) describe the design provisions, if any, which prevent the operator from shutting down a pump too early or diverting flow too soon.

RESPONSE:

(a), (b) Discrepancies have been present in the WNP-2 FSAR (see Question 031.001 (q)) in describing the operation of the containment spray system. These discrepancies have been corrected. One can now obtain the operator actions, procedural controls and design provisions for containment spray operation in 6.2.2.2, 6.5.2.2 and 7.3.1.1.4. These sections have been revised. Figure 7.3-16 has been updated to show the correct initiation signals for containment spray.

Q. 031.012
(6.2.4.2)

Describe the pump motive source which is used to provide feedwater flow after the main steamline isolation valves close and which forms the basis for the assumptions in 6.2.4.2.1.2 of the FSAR.

RESPONSE:

The text of 6.2.4.3.2.1.1.1 has been revised to incorporate the response to this question.

Q. 031.013 (RSP)
(6.3.1.3)
(7.3.1.1)

Provide justification for not testing the emergency core cooling system flow rate and the associated sensing networks during normal operation. Define the term "sensing network" as used in 6.3.1.3 of the FSAR. Identify each network which cannot be tested during normal operations. It is the staff's position that the WNP-2 design should provide engineered safety feature circuits which satisfy the guidance contained in Regulatory Guide 1.22. Accordingly, we require you to provide a revised design which conforms with the staff's position on this matter and to provide a description of these sensors and networks which provides the information requested in 7.3 of the Standard Format. Clarify the discrepancy between 6.3.1.3 and 7.3.1.1.1.2.1.2 with regard to the testability of the emergency core cooling system.

RESPONSE:

6.3.1.3 was in error. The text of 6.3.1.1.2(m) and 7.3.2.1.2.3.1.10 have been revised to reflect that all active components of the ECCS are testable during normal operation.

<u>Q 31.014</u>
(6.3.2.2)
(6.3.2.8)
(F 6.3-1a)

The location of sensors LS N001 A, B, C, and D, as shown in Figure 6.3-1a, does not appear to meet Seismic Category I requirements. Revise the design of the WNP-2 to assure that the sensors controlling the transfer of suction to the suppression pool will be seismically and environmentally qualified for their location and environment.

<u>Response</u>:

Condensate storage tank pressure sensors used for level switches are designed and qualified to Seismic Category I requirements and are environmentally qualified.

The pressure sensors will be mounted on the interior side of the concrete fluid retaining walls surrounding the condensate storage tanks. The sensors will be located such that postulated failures of the condensate storage tanks will not compromise the sensors. The sensors are designed and located to withstand natural phenomenon, e.g., tornados and high winds, and will be freeze protected.

However, in order to automate the manual transfer aspects of RCICs water source in the unlikely event the condensate storage tank inventory is unavailable for use if called upon, the applicant is modifying the present design to include the following features:

1) Automatic transfer circuitry equivalent to HPCS auto transfer system will be provided.

2) Condensate storage tank site natural phenomena considerations will be taken into account in order to assure that the automatic transfer function is not negated.

The above plant modification is similar to that proposed for the Zimmer plant.

(b) The RCIC System has automatic initiation and isolation, and manual initiation and isolation. Compliance with IEEE 279-1971, paragraph 4.17, is in 7.4.2.1.2.3.1.17.

(c) The design compliance for the present system is presented in 7.4.2.1.

(d) Since a, b, & c have been responded to, a response here is not applicable.

Q. 031.016 (RSP)
(6.3.3.9)
(7.0)
(7.2)
(15)

The discussions of response times and the testing of response times in 6.3.3.9, 7.2 and elsewhere in Chapter 7, are inadequate. Chapter 15 does not provide the response times, accuracies nor ranges for instrumentation and control systems which were assumed in the accident analyses. It is the staff's position that the design of systems which are required for safety shall include provisions for periodic verification of the minimum performance of instruments and controls that are not less than those assumed in the safety analyses. The bases for this position are General Design Criterion 21, 3.9 of IEEE Std. 279-1971, and IEEE Std. 339-1971. Accordingly, we require you to demonstrate:

a. The capability to periodically verify the minimum performance characteristics of all systems, including the appropriate procedures, which are required for safety. Testing shall include the entire system from, and including, sensor to actuator output.

b. Compliance with Branch Technical Position 24 of Appendix 7-A of the Standard Review Plan.

c. Compliance with the requests for information in 15 of the Standard Format.

RESPONSE:

a. The Technical Specifications for WNP-2 will require periodic response time testing of the C & I protection and safeguards systems from sensor through actuation systems. The procedures for initial system test and periodic surveillance testing will be developed (see 14.2.12.1.18 and Chapter 16) by WPPSS.

b. The requirements noted in a. comply with Branch Technical Position 24 of Appendix 7-A of the Standard Review Plan.

c. Chapter 15.0 is not the appropriate location for sensor response times, accuracies, and ranges for instrumentation and control system. Tables 7.2-1, 7.3-1, 7.3-2, 7.3-3, 7.3-4 and 7.4-1 have been revised to provide instrument ranges and accuracies on reactor protection systems and ESF systems instrumentation. Trip settings and response times used in Chapter 15.0 analysis have been submitted.

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Q. 031.019
(7.1.2.1)

Demonstrate how the design of the main steamline isolation valve leakage control system satisfies the requirements of 4.19, 4.20, and 4.21 of IEEE Std. 279-1971.

RESPONSE:

The main steamline isolation valve leakage control system satisfies the requirements of IEEE Std. 279-1971, 4.19, 4.20, and 4.21 as follows:

Identification of Protective Actions (IEEE 279, paragraph 4.19)

Initiation of the MSLIV-LCS is indicated in the control room.

Information Read-Out (IEEE 279, paragraph 4.20)

Meters located in the control room provide indication of process variables necessary for the proper operation of the MSLIV-LCS. Indicator lights actuated by valve position switches provide valve position indication.

System Repair (IEEE 279-1971, paragraph 4.21)

The system is designed to provide easy recognition of malfunctioning equipment through proper test procedures. Accessibility is provided for the sensors and controls to facilitate repair or adjustment. MSLIV-LCS isolation valves are located in the steam tunnel and repair is made during shutdown.

See also 7.3.2.3.2.3.1.19, 7.3.2.3.2.3.1.20 and 7.3.2.3.2.3.1.21.

Q. 031.020
(T7.1-1)

Table 7.1-1 is incomplete. Complete this table by filling in all of the blank spaces so as to indicate if the cited systems are a design which is unique to the WNP-2 facility or is similar to those used on other nuclear power plants.

RESPONSE:

Table 7.1-1 has been replaced by revised Table 7.1-2.

Q. 031.021
(7.2.1.2)
(T7.2-1) (T7.3-1)
(T7.3-2) (T7.3-3)
(T7.3-4) (T7.4-1)

The information which is presented in Table 7.1-6 and 7.2.1.2.9 is inadequate. The trip settings and margins for some instruments are missing as well as the range and accuracy for other instruments. Additionally, the units of measurement for other instruments are missing while the units of measurements for other instruments are not consistent with the units for the setpoint. Provide the following information:

a. All trip settings which are required for safety, expressed in units which are consistent with the instrumentation range.

b. All ranges and accuracies for all instrumentation systems which have a safety function or which provide required safety system support.

c. The design criteria used in establishing the required range of instruments in the reactor protection systems, engineered safety feature systems, and other safety-related systems.

d. The design criteria used in determining which portion of the range of an instrument may be used for automatic initiation of a protective function.

e. Where trip settings are to be based on operating experience, state the initial values which are to be used and provide the bases.

f. Response times, provide both required and calculated.

Our concern is that, in previous designs of nuclear power plants, the setpoints have either drifted beyond the range of operability of the sensor or sensor foldover has occurred because the setpoints were initially set too close to the extreme ends of the instrument range.

RESPONSE:

The Tables provided for the following answers include instruments in addition to those required for safety.

(a, b, and f) The requested information is contained in Tables 7.2-1, 7.3-1 through 7.3-5, and 7.4-1.

Tables 7.2-1, 7.3-1, 7.3-2, 7.3-3, 7.3-4, 7.3-5, and 7.4-1 have been revised to provide nominal trip settings, instrument ranges, accuracy, and response times. Margin will be provided later, as available.

(c) The range for safety related instrumentation is selected so as to exceed the expected range of the process variable being monitored. This may, as in the case of the neutron monitoring system, require more than one instrument to cover the expected range.

(d) The trip setpoint is located in that portion of an instrument's range which provides the required accuracy. For example, although the drywell high pressure setpoint is near the low end of the instrument's range, setpoint drift downward would be in a conservative direction. Also, this instrument has been proven in other BWR's.

(e) Initial trip setting values are established from operating experience with similar size plants, and backed up with analysis as necessary. The initial values are the trip settings listed in the tables referenced in (a, b, and f).

See the response to Q. 031.037 for a discussion of setpoint drift. Revised 7.1.2.4 and the response to Q. 031.001(r) provide additional discussion regarding this question.

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Q. 031.022
(7.2.1.1)

7.2.1.1.3.1.1 of the FSAR is incomplete. Provide a description of the source range trips which provide rod block and scram protection for the initial startup of new cores.

RESPONSE:

In addition to 7.2.1.1.4.2, the source range trips which provide rod block and scram protection for the initial startup of new cores are described in 7.6.1.6. Functions for rod blocks are provided in 7.6.1.6.2. Figures 7.6-15a, 7.6-14 and 7.2-1 also provide descriptive information.

For the initial fuel load, high-high trip contacts from each SRM are combined to produce a 1:4 non-coincident reactor trip through the manual scram portion of the circuit. Following the initial fuel loading and startup, these latter contacts are permanently shorted to remove the SRM reactor trip function.

Q. 031.027
(7.3.1.1)

The description of low pressure interlocks in 7.3.1.1.1.5 of the FSAR is incomplete. Describe the parameter sensed and the function of MO E12-F087, MO E12-F052, and MO E12-F051.

RESPONSE:

7.3.1.1.1.7 has been revised to include:

RHR System	Type	Valve	Parameter Sensed	Function
Steam Condensing Mode	MO	E12-F087	Steam pressure	Provide low-pressure supplementary flow
	MO	E12-F052	None	Block valve
	AO	E12-F051	Steam pressure	Maintain system pressure

Q. 031.028 (RSP)
(7.3.1.1)
(7.3.2.1)

The discussion of environmental conditions such as that in 7.3.1.1 and 7.3.2.1 is unacceptable. It is the staff's position that all safety-related equipment, including cables, must be qualified for operation in the worst case environment. Inside the containment, this design basis environment is established by postulated accidents. Equipment outside of containment must be qualified to the extremes of expected conditions which could result from the failure of other engineered safety features or equipment required to maintain a controlled environment such as plant heating systems. Accordingly, we require you to demonstrate compliance with this staff position. Identify and justify all exceptions.

RESPONSE:

All safety-related components including cables which are supplied by GE whether located in the containment or outside the containment are selected to meet the environmental conditions in 3.11. There are no exceptions taken.

See also revised 7.3.1.1.5.3, 7.3.1.1.6.3, 7.3.1.1.7.3, 7.3.1.1.10.3 and 7.3.1.1.11.3.

Qualification of balance of plant cables and components is covered in 3.11 and 8.3.1.2.3.

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 15.0 will be provided in the response to Question 031.016.

Table 7.2-1, 7.3-1, 7.3-2, 7.3-3, 7.3-4, 7.3-5, 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.

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Q. 031.031
(7.3.2.1)

The description of the automatic depressurization system with respect to the requirements of IEEE Std. 279-1971, 4.19 and 4.20, is inadequate. Provide the following information:

(a) Describe how the operator is made aware of items (a) through (d) under the discussion of compliance with 4.19 of IEEE Std. 279-1971.

(b) Provide justification for the use of the relief valve discharge pipe monitors and plant annunciators for providing information which forms the basis for operator action.

(c) Define the term "ADS level".

RESPONSE:

1. Identification of Protective Actions is discussed in 7.3.2.1.2.3.1.19.

2. The ADS is a backup system to HPCS. Its function is to depressurize the reactor automatically in the event of a LOCA if the HPCS system fails to maintain vessel water level. Depressurization allows the low-pressure ECCS to do their job. No operator action is required. However, if, based on water level indications in the control room, the operator determines that the HPCS can restore water level without the aid of low-pressure systems, a reset switch is available to delay ADS initiation for 120 sec. The operator decision to delay ADS is made by looking at vessel water level indication and HPCS flow and pressure indication provided in the control room. See 7.3.1.1.1.4.5.

3. "ADS levels" refers to vessel water level indications which are applicable permissives for ADS initiation. That is, ADS initiates on vessel low water level Trip 1 and Trip 3. See 7.3.2.1.2.3.1.20.

Q. 031.032 (RSP)
(7.2.1.1)
(7.3.2.1)
(15.2.6)
(F7.2-1)
(T15.2-13)

It is the staff's position that the use of Class 1E power supplies as alternate feeds for the reactor protection system buses as described in 7.2.1.1.2 of the FSAR is unacceptable since this prevents the required separation from the third division at level switch 1B21-NO24A and NO24C (GE Drawing 828E479TU). Accordingly, we require you to provide a revised design for the alternate power source for the reactor protection system which satisfies your electrical separation criteria. Identify and justify all exceptions. Clarify the discrepancy between the present design and the assumptions of 15.2.6 and Table 15.2-13.

RESPONSE:

Drawing 828E479TU is not applicable to WNP-2. See Figure 8.3-2.

Q. 031.035
(7.3.2.1)

Describe the design features which provide assurance that the main steamline isolation valves do not close in less than 3 seconds.

RESPONSE:

The MSIV does not close in less than 3 seconds due to an in-line hydraulic damper restricting the closure speed. A small vernier type control limiting the hydraulic flow from below the damper piston allows accurate speed setting of valve closure. This speed control valve (#7 of Figure 7.3-4), having the capability to vary the MSIV closure speed from 3 to 10 seconds, can be set and locked for the desired speed. The control valve is set during production testing. MSIV closure speed are re-determined at pre-operational tests and plant shutdown surveillance tests and control valves re-set as required.

Q. 031.036
(7.3.2.1)
(F7.3-8b)

Provide a drawing for the test control logic which shows how the main steam line isolation valve is tested for the required response time limit (e.g., greater than 3 seconds but less than 5 seconds) at rated steam flow.

RESPONSE:

Figure 7.3-11 shows the test control logic for testing the MSIV closure time. In order to avoid scram during MSIV full closure testing, the reactor power is reduced to approximately 75%.

The control room operator may time the valve closure while observing valve status lamps.

Q. 031.037
(7.3.2.1)

The statement that "All components used in the isolation system have demonstrated reliable operation in similar nuclear power plant protection system or industrial application," is unacceptable since: (a) this statement does not satisfy the requirements of IEEE Std. 323-1971; and (b) considerable problems have been experienced with sensor drift. Provide an amended discussion of compliance with the requirements of 4.4 of IEEE Std. 279-1971 which satisfies the requirements of IEEE Std. 323-1971 and which describes the methods used to reduce sensor drift to acceptable levels.

RESPONSE:

Components were chosen on the basis of being the best available, vendor test and specs, and proven operational use in similar application. Operational experience has indicated some sensor drift, however, the frequency of surveillance check, test, and calibration and use of historical instrument data has consistently kept sensors within the limits of safe operation.

Conformance to IEEE Std. 323-1971 is discussed in 7.1.2.5.4.

Sensor drift is discussed in 7.1.2.4.

Q. 031.038
(7.3.1.1)
(7.3.2.1)

Clarify the description of the temperature monitoring circuits which initiate containment and reactor vessel isolation. It is the staff's understanding that the BWR/5 and BWR/6 plants are equipped with a system using thermocouples. Include the following information in this clarification:

a. Describe the monitor system in the manner requested in 7.3 of the Standard Format.

b. Describe how the system satisfies the requirements of IEEE Std. 338-1971 and General Design Criterion 21 and how it conforms to the guidance in Regulatory Guide 1.22.

c. Clarify the discrepancy between 7.3.1.1.2.4.1.12.2.1 and 7.3.2.1.2.3.1.5.2.1.10 of the FSAR.

RESPONSE:

(a) Main steamline space temperatures and differential temperatures are measured by dual-element thermocouples, and the analog signals are transmitted from the sensors to the temperature switch point modules located in the control room. This temperature measurement circuit arrangement is similar to Zimmer but not similar to Duane Arnold. Duane Arnold uses bimetallic temperature switches which are locally mounted. See 7.3.1.1.2.4.1.3.

(b) The main steamline space temperature detection system, can be tested during reactor operation. Operability of the sensors (thermocouples), can be verified by comparing the readings during reactor operation. A complete check of the system, including the sensors, can be made during refueling and other planned shutdown periods.

(c) Responses to items (a) and (b) help to clarify any conceived discrepancies.

See 7.3.2.2.2.3.1.9, 7.3.2.2.2.3.1.10, 7.3.2.2.2.3.4, and 7.3.2.3.2.2.1.

Q. 031.039 (RSP)
(7.3.2.2)

The methods proposed for testing of some safety functions are unacceptable. Accordingly, we require you to provide modified designs for all of the safety-related instrumentation and control systems so that these designs will satisfy the following staff positions. Identify and justify any exceptions.

a. All portions of the protection systems shall be designed in accordance with IEEE Std. 279-1971, as required by 10CFR50.55a(h). All actuated equipment that is not tested during reactor operation, should be identified and a discussion of how this equipment conforms to the guidance contained in paragraph D.4 of Regulatory Guide 1.22, should be submitted.

b. The use of jury-rigged bypasses such as temporary jumpers, the removal of fuses, or removal of connectors is not an acceptable method for standard in-service testing.

c. The containment isolation valves for the WNP-2 facility shall be tested from the sensors through the actuating circuits and the valves themselves.

d. The methods which are provided for the testing of protection systems shall satisfy the requirements of 4.11 of IEEE Std. 279-1971.

RESPONSE:

a. Relative to IEEE 279-1971 and Regulatory Guide 1.22, which requires that actuated equipment be tested during reactor operation, all of the actuated equipment has the capability to be tested during reactor operation.

b. In no instance will it be necessary during testing of these circuits to either lift leads or remove fuses.

c. All isolation valves can be tested from the sensor through the actuating circuits.

d. See also revised 7.2.2.1.2.3.1.11, 7.3.1.1.5.3, 7.3.1.1.6.3, 7.3.1.1.7.3, 7.3.1.1.10.3, 7.3.1.1.11.3 and 7.3.2.2.2.3.1.11.

(BLANK)

Q. 031.040
(7.3.2.1)
(7.4.2.2)

Provide justification for your position stated in 7.3.2.1.2.3.4 and 7.4.2.2.2.3 of the FSAR that only 2.1 and 2.2 of IEEE Std. 338-1971 are applicable to the design of the emergency core cooling system and the reactor core isolation cooling system. Identify and justify all exceptions to IEEE Std. 338-1971.

RESPONSE:

The statements in the subject paragraphs are in error and have been revised to state that the ECCS and RCIC comply fully with the requirements of IEEE 338-1971.

Q. 031.041
(7.3.2.4)
(7.4)

The discussion in 7.3.2.4.4 and 7.4 of the conformance of the WNP-2 design with the present Regulatory Guides, is incomplete. Revise these sections of the FSAR to include a discussion of how the WNP-2 design conforms with Regulatory Guide 1.29.

RESPONSE:

NRC Regulatory Guide 1.29 has been addressed in revised sections of the FSAR. See revised 7.1.2.6.6.

Q. 031.042
(F7.3-8b)

Revise all FSAR figures (such as Figure 7.3-8b) to include alpha-numeric area locators if such figures are referenced by, or continued on, additional sheets or figures.

RESPONSE:

FSAR figures have been revised to include alpha-numeric area locators.

LPCI: In no event can failure of an automatic control circuit for equipment in one division disable the manual electrical control circuit for the other LPCI division. Single electrical failures cannot disable manual electrical control for the LPCI function. LPCI A has an armed manual initiation pushbutton in parallel with the automatic initiation logic which will also initiate LPCS. The LPCI B and C systems have an armed manual initiation pushbutton in parallel with the automatic initiation logic.

Refer to 7.3.2.1.2.3.1.17 for the above description.

(BLANK)

Q. 031.045
(7.4.1.2)

It is the staff's understanding that the heating system for the standby liquid control system has been redesigned and now provides redundant heaters which are powered from Class 1E buses. Accordingly, we request that you: (a) revise 7.4.1.2.2; and (b) provide process and instrumentation drawings and electrical schematics which include the revised heating system and its controls.

RESPONSE:

a. The standby liquid control system heaters have not been redesigned. The system remains with two heaters, one of which is used for initial heating (when rapid heating is required), the other used to maintain the SLCS solution at required temperature. The heaters and their controls are not required for the initiation of the SLCS.

   7.4.1.2.2 is correct and requires no updating.

b. There are no revised heating system drawings for the SLCS.

   See Figure 7.4-3 and 7.4-4.

Q. 031.046
(7.4.2.4)

You indicate in 7.4.2.4.2 of the FSAR that the only Regulatory requirements applicable to the design of the residual heat removal (RHR) system are General Design Criteria (GDC) 34 and 61. However, GDC 34 states, in part, that the residual heat removal system has a safety function and requires that..."suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities" be provided. Accordingly, revise your design to provide a leak detection capability and provide the appropriate information to demonstrate that the RHR system satisfies all the requirements of the General Design Criteria.

RESPONSE:

3.1.2.4.5 contains descriptive material which provides a discussion of RHR system compliance with General Design Criterion 34. 5.2.5 contains a discussion of the Leak Detection System and its application to the RHR System. 15.2.9 discusses a backup method for disposing of residual heat should the normal shutdown line become unavailable during shutdown.

Q 31.047
(7.5.2.4)

The seismic qualification of indicators and recorders for post-accident monitoring which is described in 7.5.2.4 is unacceptable. It is the staff's position that post-accident indicators and recorders must meet their minimum performance requirements before and after a seismic event without requiring adjustments or repair. (The staff acknowledges that these electro-mechanical devices may not provide accurate readings during severe vibrational excitation). Accordingly, we require you to provide a revised design which satisfies the staff's position on this matter.

Response:

The indicators and recorders are nonseismic. At the time WPPSS-2 was being designed, there was no IEEE standards or Regulatory Guide requirements for design of the subject instrumentation.

The instrumentation and readout devices are of high quality and from well-known manufacturers. This similar instrumentation is used for post-accident monitoring in such licensed and operating plants as Duane Arnold and Brunswick 2, and in such plants as Zimmer and LaSalle presently in the late stages of review. Therefore, it is the position of the General Electric Company that the instrumentation provided is adequate.

Q. 031.048
(7.4.1.4)

Identify the systems and functions controlled by the transfer switches referenced in 7.4.1.4.3e of the FSAR. Indicate the location of these switches.

RESPONSE:

The transfer switches are located on panel C61-P001 (Remote shutdown panel) which is located outside the main control room. Selection of the location is based upon having no effect on the panel from the control room evacuation event. See also revised 7.4.1.4.4.6.

The systems and functions controlled by the transfer switches are:

Reactor Core Isolation Cooling (RCIC) System

The following RCIC System functions shall have control and transfer switches located at the remote shutdown control panel:

E51-F010 - Motor operated valve (pump suction from condensate storage)

E51-F013 - Motor operated valve (RCIC injection shutoff)

E51-F019 - Motor operated valve (minimum flow to suppression pool)

E51-F022 - Motor operated valve (test bypass to condensate storage)

E51-F031 - Motor operated valve (pump suction from suppression pool)

E51-F045 - Motor operated valve (steam to turbine)

E51-F046 - Motor operated valve (lube oil cooling)

E51-F063 - Motor operated valve (steam supply line inboard isolation

E51-F064 - Motor operated valve (RHR heat exchanger steam line isolation)

Q. 031.051
(7.6)
(T7.1-1)

The descriptions of the systems and components presented in 7.6 of the FSAR are inadequate. Provide the information requested in 7.6 of the Standard Format, including a discussion of all differences between the designs of these systems and BWR-5 designs such as Zimmer and LaSalle.

RESPONSE:

The description of the systems and components present in 7.6 of the FSAR has been rewritten. Similarity to licensed reactors is illustrated by revised Table 7.1-2.

Q. 031.052 (RSP)
(7.1.2.1)
(7.6.2.6)
(15.4.2.2)
(T7.1-2a)

The accident analysis presented in 15.4.2.2 of the FSAR is based, in part, on the assumption that the rod block monitor (RBM) acts to mitigate the consequences of a continuous withdrawal of a control rod. Accordingly, it is the staff's position that the RBM is a protection system and must be designed, fabricated, installed, tested, and subjected to all of the criteria applicable to a reactor trip system. Accordingly, we require you to revise your design to reflect the importance of the RBM. Identify and justify any exceptions.

RESPONSE:

The RBM is used to prevent the operator from withdrawing a rod in the power range so that fuel cladding integrity is always maintained. It has two channels that are redundant, separated, and isolated. It provides two alarms and then a block as the power goes up locally.

The rod block monitor (RBM) is integral with the neutron monitoring system which is presently identified in 7.1.2.1.4. The RBM design basis is included in 7.1.2.1.4.6. The RBM is adequately described in terms of initiating circuits, logic, bypasses, interlocks, redundancy, diversity, and actuated devices in 7.6.1.5.7. It is to be emphasized that the WNP-2 RBM is essentially identical to the RBM which has been licensed previously on all plants from Duane Arnold through Hatch-1 and Brunswick. The design of the RBM for WNP-2 is identical to the design of the RBM for Zimmer.

The Chapter 15 and Appendix 15A safety analysis involving the continuous rod withdrawal error, presents the event as an anticipated operational transient.

The RBM is not used in any Design Basis Accident (DBA) category event described in Chapter 15.

Q. 031.053
(7.7.1.1)
(15)
(F7.7-2)

The reactor manual control system is assumed in Chapter 15 to function to mitigate or prevent several accidents. Therefore, it appears that the reactor manual control system, the reactor (sic) position indicator system and the reactor (sic) sequence control system are part of the reactor protection system. Accordingly, provide the following additional information:

a. Provide the design bases and other appropriate information in the manner requested in 7.2 of the Standard Format for these three subsystems.

b. Provide a complete description of the scram time test panel, in the manner requested in 7.7.2 of the Standard Format, including schematic and wiring diagrams.

RESPONSE:

(a) The rod block monitor (RBM), the reactor manual control system (RMCS), the rod position indication system (RPIS), and the rod sequence control system (RSCS) are not utilized to mitigate the consequences of accidents, and are not part of the reactor protection system (RPS).

General Design Criterion 20 does not define these systems as reactor protection systems.

The RSCS does not initiate a scram or isolation signal and is therefore not a protection system. It does, however, perform the power generator function of imposing strict limits on control system operation to prevent out-of-sequence rod patterns. The allowable rod patterns are such as to limit the peak enthalpy in the fuel to less than 280 cal/g. The RBM considerations are discussed in response to Question 031.052.

(b) No functional changes in Figure 7.7-2 are anticipated in answer to Question 031.053.

(c) The rod scram time test panel is provided as a common location for obtaining control rod scram travel time data for all rods. A base for each time the rod passes an odd-numbered position is transmitted from the rod position information cabinet through a buffer to the scram time test panel plug-in jacks. Each jack carries information from one rod. Power for these on-off signals is provided by a power supply in the scram time test panel. A 29-channel recorder can be connected by patch cords to any 29 or fewer of the 137 plug-in jacks to record on a moving chart the passage of the rods through the odd-numbered positions. Scram time can be determined from the chart speed and pulse position rod. Each plug-in jack has an indicator above it to indicate which rod or rods are in a test condition at the hydraulic control units. This information is transmitted through the display memory module to the rod scram time test panels.

A contact of a relay across the scram backup solenoid is used to start the recorder chart. A time-delay stops the chart after 20 seconds. The schematic diagram is on Sheet 7 of 807E183TC, and the scram time test panel connection diagram is 828E183TC.

(BLANK)

Q. 031.054
(7.7.1.11)
(8.3.1.1)

The material presented in 7.7.1.11 of the FSAR is inadequate. Provide the following additional information:

a. Clarify the discrepancy between the statement in 7.7.1.11 that the reactor water cleanup system is fed from the plant instrumentation bus and does not require back-up power and that in 8.3.1.1.3 which describes these buses as Class 1E.

b. Since overpressure protection is a function of the reactor water cleanup system instrumentation, provide justification for not providing a Class 1E system.

c. Provide justification for not providing Class 1E equipment for the isolation functions listed in 7.7.1.11.1.1 (sic) of the FSAR.

RESPONSE:

a. There is no discrepancy existing between 7.7.1.8 and 8.3.1.1.3.

b. The RWCU system is not a safety-related system beyond the isolation valves. Overpressure protection for the system is provided by primary pressure relief valves. Therefore, there are no instrumentation requirements relative to overpressurization of the RWCU system.

c. Class 1E equipment is provided for system isolation. The RWCU system isolation valves are part of the reactor coolant pressure boundary (RCPB) and as such are controlled by the primary containment and reactor vessel isolation control system instrumentation. These valves and piping are Seismic Category I, and the controls are Class 1E as described in 7.3. The portion of the RWCU system outside the outer isolation valves is not part of the RCPB and not safety-related, and instrumentation for this portion is nonessential.

Q 31.069
(3.8.2.1)
(6.2.1.1)
(031.001)

Your response to Item 031.001(p) is incomplete. Describe the air supply, pressure control, and position indication for the butterfly valves in accordance with the guidance provided in Section 7.3.1 of Regulatory Guide 1.70. Clarify the reference to 6.2.1.1.2 in the response to Item 031.001(p) since this response does not address the staff's concern regarding the position indication instrumentation.

Response

Please refer to 3.8.2.1.3, 6.2.1.1.2c and 7.3.1.1.2.9.1 for the information requested.

Q 31.070
(RSP)
(6.2.2.2)
(6.5.2.2)
(7.3.1.1)
(031.001)
(031.011)

It is the staff's position that insufficient time is available for the operator to reliably take the manual actions which are necessary to initiate suppression pool spray during a small break. The staff has established the requirement for automatic initiation of suppression pool spray for the Mark II containment. Accordingly, we require you to provide a Class IE automatic control system for each suppression pool spray system.

Response

The WNP-2 design meets the intent of the proposed CSB Branch Technical Position on "Steam Bypass for Mark II Containments".

The history of the questions of steam bypass on WNP-2 are extensive, dating back to January 1972. Questions 5.4, 5.22, and 5.24 to the PSAR all respond to the concern. The SER (pp 63-65) summarized the NRC position on the issue at the CP stage and noted that WPPSS agreed to study additional means to mitigate the consequences or minimize the potential for bypass leakage. This was formally documented as a Post CP item in the notes of a NRC-WPPSS meeting held on October 17-18, 1973 (Reference 1). In the notes WPPSS committed to submitting a report on the matter. In August 1974, Reference 2 transmitted the WPPSS report WPPSS-74-2-R5, "Drywell to Wetwell Leakage Study", satisfying the commitment. The NRC requested additional information concerning the report in Reference (3). References (4) and (5) provide WPPSS responses to the NRC questions. Reference (6) indicated that Structural Engineering Branch found the applicable WPPSS responses acceptable. WPPSS has no record of feedback from Containment Systems Branch on the responses to its questions but assumed in Reference (5) that, in the absence of feedback, the post CP item was resolved. Accordingly, WPPSS has gone ahead with construction in these areas based on the above correspondence.

Q 31.071
(7.3.1.1)
(031.001)

The primary containment and reactor vessel isolation control system receives power from the reactor protection system motor generator sets. Describe how the reactor operator determines the position of each motor operated and each solenoid operated or controlled isolation valve after a loss of the motor generator sets. We are concerned that your present design de-energizes these sets during a loss of off-site power and does not include automatic restart of the motors.

Response:

Status indication for all motor operated containment isolation valves is powered from diesel generator buses and this is not dependent on the availability of the RPS M/G set buses.

In addition to valve position status located on the main bench board at the control switch for each isolation valve, a complete containment isolation valve position display exists on Board S. This panel is located in the first row of control room back panels. The power sources used for the display valve position status are uninterruptible with diesel generator backup. Thus, the solenoid operated isolation valves also have position indication that is not dependent on RPS M/G set availability.

Q 31.072
(7.6.1.5)
(F 7.7-1)
(F 7.2-5)
(031.007)
(031.045)

Your response to Item 031.001(t) is unacceptable. Clarify the discrepancies between 7.1-11, Figures 7.1-1 and 7.2-5, and 7.6.1.5. Specifically, clarify the number of instruments and the designation of these instruments with regard to their trip channel assignments. Provide justification for running redundant signals through the same penetrations.

Response:

There are no discrepancies between Table 7.1-11, Fig. 7.1-1 and 7.2-5, and 7.6.1.5. Also, redundant signals do not run through the same penetrations.

The neutron monitoring system trip outputs to the reactor protection system are derived from 6 APRM channels as follows:

RPS trip channel	APRM
A1	A&E
A2	C&E
B1	B&F
B2	D&F

This combination which uses APRM channels E&F in redundant RPS trip logic allows an APRM channel in each trip division to be bypassed by the operator without the loss of ability to Scram on a high flux condition. See FSAR 7.2.1.1.4.2 for additional information concerning the neutron monitoring system inputs to RPS.

Q 31.075
(7.7.1.3)
(7.7-7)
(031.008)

The response to Item 031.008 and the additional information which is presented in Figure 7.7-7 and 7.7.1.3 is incomplete. In this regard, the staff notes that the design includes an interlock which prevents the transfer of one pump from high speed operation to low speed operation when the second pump is operating at high speed and the control switch is moved to the motor generator position. Explain why this interlock is provided. Justify not providing a similar interlock in the pump start circuitry in order to prevent a similar occurrence under the same conditions (i.e., both pumps running at high speed) if the switch should be placed in the start position.

Response:

The purpose of the recirculation pump interlock identified in 031.075 is to prohibit flow inbalance between the pumps, thereby minimizing jet pump vibration. A similar interlock in the pump start circuitry is not provided because, by design, if both pumps are running at high speed and the switch is placed in the start position, nothing will happen to affect the status of either pump.

Q. 031.076
(RSP)
(6.7)
(7.3.2.3)
(Q.031.019)

It is the staff's position that neither the information which is provided in response to Item 031.019 nor the information which is presented in 6.7 and 7.3.2.3 provides sufficient information on the main steamline isolation valve leakage contol system. Describe this system in accordance with the guidance provided in Section 7.3 of Regulatory Guide I.70, including a process and instrumentation drawing, an electrical schematic, and a failure mode and effects analysis which is sufficiently detailed to address failures at the component level. For example, describe the consequences of a spurious closing of the contacts of relay K4 under all plant-operating modes, including testing.

Response:

In addition to FSAR 6.7 and 7.3.2.3 and the MSIV leakage control system instrumentation and controls is described in 7.3.1.1.3. The system is shown diagramatically in P&ID form in FSAR Figure 3.2-25 with logic diagrams shown in FSAR Figures 7.3-18a-g.

Electrical schematics for the MSIV leakage control system (Drawings, E519, shts 30 & 31) have previously been submitted as part of FSAR 1.7.

A failure modes and effects analysis addressing worst case failures and consequences concerning the safety function aspects of the MSIV Leakage Control System, i.e., following a LOCA, was submitted as part of the FSAR 6.7.3.1. This analysis was submitted previously to the NRC in Reference 1 as a response to a post-construction permit item. In Reference 2 the NRC requested additional information and WPPSS responded in Reference 3. In Reference 4, the NRC stated the design was acceptable subject to the provision of an interlock preventing actuation of the MSIV-LCS if the inboard MSIV were open. In Reference 5 WPPSS agreed to provide this interlock. In any case, to respond to what is felt the intent of Question 031.076 is, an additional FMEA was performed which addressed failures which could occur during other plant operating modes.

The following is a description of those failures identified having undersirable consequences.

Failure Mode	Equip. Effected by Failure Mode	Undersirable Effects	Remarks/ Results
1. Spurious closing of pressure switch PS-25 contacts	MOV's MSLV-V-9, & MSLV-V-10	Both valves open simultaneously	Reactor pressure steam admitted into low pressure system piping and into the reactor building. Possible piping damage and/or plant personnel hazard.
2. Spurious closing of relay CR-3 contacts	Same as item 1 above	Same as item 1 above	Same as item 1 above
3. Spurious closing of relay CR-1 contacts	MOV's MSLC-V-4 & MSLC-V-5	Same as item 1 above	Same as item 1 above

In order to prevent the events described in items 1, 2 and 3 from occurring the logic design will be modified. An additional interlock will be added in series with the contacts of pressure switch PS-25, relay CR-3, and relay CR-1. The interlock will be provided from the system initiation control switch. Thus, two active component failures would be required to cause a similar occurrence.

In addition, it was noted from the results of FMEA that events similar to those described in items 1, 2 and 3 above might occur from localized events in control panels and wireways. This is due to inherent system designs requiring the several pairs of series motor operated valves to be controlled from a common safety division to meet single failure criteria. To preclude such situations certain key control devices and wiring associated with each valve of a series pair will be separated from the other. This will prevent localized events within a control panel, instrument rack, wireway, or motor control center from causing simultaneous opening of both valves during normal operating modes.

The MSIV leakage control system does not contain a relay K4.

References:

1. WPPSS to NRC letter, GO2-74-73, "Post Construction Permit Item - Transmittal of Report WPPSS 74-2-RG, Concept for Main Steam Isolation Valve Leakage Control System", dated Dec. 3, 1974.

2. NRC to WPPSS letter, Butler to Stein, dated March 18, 1975.

3. WPPSS to NRC letter, GO2-75-238, "Response to Request for Information - MSIV-CCS", dated Aug. 18, 1975.

4. NRC to WPPSS letter, Parr to Stein, dated November 21, 1975.

5. WPPSS to NRC letter, GO2-76-294, "MSIV-CCS", dated July 14, 1976.

Q 31.077
(031.021)
(031.037)

The responses to Items 031.021 and 031.027 are unacceptable because the accuracy of the instrumentation sensors is not provided. The information provided in Table 7.2-1 is labeled as that which is required in contrast to that which is actually provided. Additionally, the first item of Table 7.2-1 indicates that a pressure switch which has an error of plus or minus 10 psi is required. In similar BWR-5 application, this instrument is stated to have accuracy of plus or minus one percent of full scale. Accordingly:

a. Provide an amended response to Item 031.021 which includes the accuracy of the sensors which are installed in your plant.

b. Provide an amended response to Item 031.037 which defines such terms as "adequate margin" and describes the criteria and procedures for determining and adjusting the instrument test frequencies.

Response:

See the response to Question 31.063 (31.016).

Q 31.78
(031.050)
(T 7.3-5)

Clarify the discrepancy between the response to Item 031.050(a) and Table 7.3-5. Specifically, explain how the temperature trips, which are usually set between 135 and 185 degree F, can reliably detect a 50 gpm leak from the reactor heat removal system during a prolonged cold shutdown when the primary coolant temperature may be less than 135 degrees F.

Response:

The equipment area temperature monitors are not intended to detect 50 gpm leakage from the emergency core cooling systems (ECCS) during the long-term recovery following the postulated loss-of-coolant accident. There is insufficient energy in the ECCS fluid (when the reactor is depressurized and cooled down) to heat the equipment area in the event of the postulated 50 gpm leak.

The design bases for the ECCS equipment area temperature monitors is to identify leakage and provide inputs for isolation in the event of leakage from high energy RCPB lines beyond the second isolation valve during normal plant conditions.

In addition to the temperature sensors, RHR flow, Reactor water level, HPCS and LPCS flow detectors listed in response to Q. 31.050, numerous drain flow indicators are provided. These consist of:

a. Restricting orifice, placed directly in the collecting drain header. These orifices are designed and calibrated to pass 5 gpm with a static head of 6 inches. As soon as total flow in the collector exceeds this flow rate an electrode will sense the increase of fluid and activate an annunciator alarm.

b. Conductance type electrodes 6 inches long mounted in a suitable fitting threaded into the collector-header.

c. Mating control switch consisting of a solid-state electronic relay to operate controls.

f. Marking Characteristic Code

Division	Application	Tray/Conduit Characters	Inscription Characters	Background
1	P,C,I	Div. 1	Black	Yellow
2	P,C,I	Div. 2	Black	Orange
3	P,C,I	Div. 3	Black	Red
4	RPS-A1 NSSS-A1 NMS-A	R Ch. A1	Red	Lt. Blue
5	RPS-A2 NSSS-A2 NMS-C	R Ch. A2	Red	Green
6	RPS-B1 NSSS-B1 NMS-B	R Ch. B2	Red	Dk. Blue
7	RPS-B2 NSSS-B2 NMS-D	R Ch. B2	Red	Brown
A	P,C,I	Div. A	Black	Silver or Silver/ Yellow Stripe
B	P,C,I	Div. B	Black	Gold or Gold/ Orange Stripe

P - Power
C - Control
I - Instrumentation

Non-Class 1E circuits receiving power from Class 1E power sources which are not shed by an accident signal shall be identified by the addition of checkered black/silver or black/gold markers indicating the Class 1E division (Division 1 or 2 respectively) from which the circuit receives its power and identified as A'1 or B'2 (respectively) in the computerized cable schedule.

Specific Requirements for Separation of Cables for Nuclear Safeguards Systems

Reactor Protection System (RPS, NSSS and NMS)

Reactor Protection System (RPS, NSSS and NSSS, and NMS fail-safe wiring:

a. Fail-safe wiring outside of the main protection system cabinets shall be run in rigid or flexible conduits and/or totally enclosed trays used for no other wiring and shall be conspicuously identified at all junction or pull boxes. IRM, LPRM input, and RPS Scram Group output cables may be combined in the same wireway provided that the four divisional separation is maintained.

b. Wires from both RPS trip system trip actuators to a single group of scram solenoids may be run in a single conduit; however, a single conduit shall not contain wires to more than one group of scram solenoids. Wiring for two solenoids on the same control rod may be run in the same conduit.

c. Cables through the primary containment penetrations shall be so grouped that failure of all cabling in a single penetration cannot prevent a scram. (This applies specifically to the neutron monitoring cables and the main steam isolation valves position switches.)

d. Power supplies to systems which de-energize to operate (so called "fail-safe" power supplies) require only that separation which is deemed prudent to give reliability (continuity of operation). Therefore, the protection system flywheel motor generator (MG) sets and load circuit breakers are not required to comply with the separation requirements of this Specification for safety reasons even though the load circuits go to separate panels.

e. Wiring for the four RPS scram group outputs and the NSM LPRM inputs must be routed as four separate divisions.

Q. 040.9

7.5.2 of the FSAR references General Electric topical report NEDO-10466 which describes the power generator control complex. This report is not acceptable to the staff as a basis for licensing. Demonstrate the acceptability of the power generator control complex for the WNP-2 facility.

RESPONSE:

General Electric topical report NEDO-10466 Rev. 1 was issued Oct. 1, 1977. The revised topical report will demonstrate the acceptability of the power generator control complex for the WNP-2 facility. Following acceptance of the revised topical report by the NRC, the FSAR will be revised to refer to NEDO-10466 Rev. 1.

See also 7.5.2 which has been revised to incorporate the response to this question.

Q. 040.10

A review of licensee event reports related to the operation of diesel-generators has indicated that, in some cases, the information available to the control room operator regarding the operational status of the diesel-generators, may be imprecise and could lead to a misinterpretation by the operator. This can be caused by the sharing of a single annunciator alarm to indicate: (1) conditions which would render a diesel-generator unable to respond to an automatic emergency start signal; and (2) abnormal, but not disabling, conditions. Another cause can be the specific wording of an annunciator window which does not clearly indicate that a diesel-generator is inoperable (i.e., unable to respond to an automatic emergency start signal during the period the annunciator is sounded) when, in fact, it is inoperable for that purpose.

Provide the results of an evaluation of the alarm and control circuitry for the WNP-2 diesel-generators to determine how each condition that renders a diesel-generator inoperable, is alarmed in the control room. These conditions would include: (a) the trips that lock out the diesel-generator start, thereby requiring manual reset; (b) control switch or mode switch positions that block automatic start; (c) loss of control voltage; (d) insufficient starting air pressure or (e) insufficient battery voltage. Your review should consider all possible operational conditions for the diesel-generator (e.g., test conditions and operation from local control stations). One area which is of particular concern to the NRC staff is the unreset condition following a manual stop at a local station which terminates a diesel-generator test. Manual stops such as this prevent subsequent automatic operation until the diesel-generator controls are reset. Your response should provide a detailed evaluation, including the results and your conclusions, and a tabulation of the following items:

a. all conditions that could render the diesel-generator incapable of responding to an automatic emergency start signal for each operating mode as discussed above;

b. the wording on the annunciator alarm window in the control room for each of the conditions identified in item (a);

c. any other alarm signals not included in item (a) which also cause the same annunciator to alarm;

Q. 040.043

Review the electrical control circuits for all safety-related equipment to provide assurance that disabling of one component does not, through incorporation in other interlocking or sequencing controls, render other components inoperable. All modes of test, operation, and failure should be considered. Describe and state the results of your review.

Response:

The WNP-2 physical and electrical separation criteria does not require that the disabling of any safety related component does not render any other safety related component inoperable.

The WNP-2 design is consistent with Federal Regulations and Industry Codes and Standards which require that a sufficient number of circuits and equipment be maintained such that protective functions required during and following any design basis event, when taken with any single failure, can be accomplished.

This design approach does not preclude the disabling of safety system components, either intra-division or inter-division, from failure of other intra-division or inter-division components during all plant operating modes unless that failure can result in the loss of a protective function. An analysis of this position is contained in 7.2, 7.3, 7.4 and 7.6 where compliance with IEEE 279-1971 paragraphs 4.2, 4.5, 4.6, 4.7 and 4.12 are addressed for each safety system.

Q. 40.44

During our review of the Hatch 2 application for an operating license, we identified certain potential problems that could be caused by the motor-generator sets of the reactor protection system (RPS). These problems were related to the operating characteristics of these motor-generator sets which might exceed the envelopes of acceptable values of voltage and frequency, thereby adversely affecting the connected loads. Indicate whether the motor-generator sets in the WNP-2 RPS are similar to those in the Hatch 2 facility. If they are, provide: (1) a commitment to the generic resolution of this item; or (2) justification for the use of these motor-generator sets in the WNP-2 facility.

Response:

WPPSS has the subject motor-generator set type and was made aware of the problem in reference (1). In reference (2), WPPSS committed to the generic resolution of the problem with General Electric. The motor-generator will be supplied with Class IE qualified equipment to monitor and protect the connected loads from unacceptable values of voltage and frequency. The generic design and qualification plan supplied by GE has been approved by the NRC as satisfying the requirements of IEEE 379-1972, Section 6.6.

Q. 40.59
(9.5.8)

Describe the instrumentation, controls, sensors and alarms of the diesel engine combustion air intake and exhaust system which alert the reactor operator when the design parameters of this system are exceeded. Discuss the actions of the operator if this system annunciates an alarm in the control room. As before, our concern is the time available for an operator to take appropriate action. (Refer to Paragraphs II.1 and II.4 of Section 9.5.8, Revision 1, of the SRP).

Response:

Alarms are not provided on DG exhaust and intake parameters. Other upset conditions are monitored and annunciated on local instrument panels and brought to the main control room operators attention in the form of a single trouble annunciator.

The air filter for the turbocharged diesel engine is the panel type oil bath filter. This type filter is self cleaning during operation, and air restriction due to a clogged filter is not considered a relevant possibility. The panel type oil bath filters provide efficient air filtration with a minimum of maintenance. These filters are inspected, drained and cleaned periodically as recommended by the manufacturer.

The diesel exhaust is also an open flow system with no evident potential for development of restriction. Diesels are tested periodically as required by Technical Specification, therefore, any unforseen degradations would become evident in the performance parameters upon which DG operability is based.

Emergency DG systems are redundant therefore no credit is taken for operator action for an assumed failure in a single unit.

Q. 40.60
(10.2)

Expand your discussion of the turbine speed control and overspeed protection system. Provide additional explanation of the turbine and generator electrical load following capability for the turbine speed control system with the aid of system schematics (including turbine control and extraction steam valves to the heaters). Tabulate the individual speed control protection devices (normal, emergency and backup), the design speed (or range of speed) at which each device begins operation to perform its protective function (in terms of percent of normal turbine operating speed). In order to evaluate the adequacy of the control and overspeed protection system provide schematics and include identifying numbers to valves and mechanisms (mechanical and electrical) on the schematics. Describe in detail, with references to the identifying numbers, the sequence of events in a turbine trip including response time, and show that the turbine stabilizes. Provide the results of a failure mode and effects analysis for the overspeed protection systems. Show that a single steam valve failure cannot disable the turbine overspeed trip from functioning. (SRP 10.2, Part III, items 1, 2, 3 and 4).

Response:

a. Provide additional explanation of the turbine and generator electrical load following capability for the turbine speed control system with the aid of system schematics (including turbine control and extraction steam valves to the heaters).

   Answer: See the response to question 40.63 and revised FSAR 7.7.1.5 (including Figures 7.7-9 and 7.7-10).

b. Tabulate the individual speed control protection devices (normal, emergency and backup), the design speed (or range of speed) at which each device begins operation to perform its protective function (in terms of percent of normal turbine operating speed).

   Answer: See revised FSAR page 10.2-5.

Q. 40.69
(10.4.1)

Indicate and describe the means of detecting radioactive leakage into and out of the main condenser. Indicate what provisions have been incorporated into the WNP-2 facility to preclude unacceptable accidental release of radioactivity to the environment. (Refer to Paragraph III.2.b of Section 10.4.1, Revision 1, of the SRP).

Response:

Please see revised 10.4.1.3. The main condenser evacuation system maintains a vacuum to remove noncondensible gases from the condenser, including air and radioactive gaseous products originating in the reactor. This effluent is discharged to the gaseous radwaste system. See 11.3 for a description of this system. Monitoring and control of release paths from the gaseous radwaste system and other potential release paths from the condenser (e.g., turbine building exhaust, circulating water, etc.) is described in 11.5.2. See also 10.4.2.3 and 10.4.2.5.

Q. 40.70
(10.4.1)

Discuss the operation of the main steam line isolation valves if there is a loss of condenser vacuum. (Refer to Paragraph III.3b of Section 10.4.1, Revision 1, of the SRP).

Response:

See revised 10.4.1.5, 7.3.1.1.2.4.1.13, and 7.3.2.2.2.3.1.12. See also the response to 40.65.

Q. 212.003
(6.3)

Your discussion of single failure does not adequately address ECCS passive failures during long-term cooling. Accordingly, provide a response to the attached Reactor Systems Branch Technical Position regarding the leak detection requirements for passive failures in the ECCS piping.

REACTOR SYSTEMS BRANCH TECHNICAL POSITION
Leak Detection Requirements for ECCS Passive Failures

The passive failures to be considered are limited to leaks from valve stem packing and pump seals. The sum of these leak rates may range from essentially no leakage up to the equivalent of the sudden failure of the seal of the largest ECCS pump (e.g., about 50 gpm). It is the staff's position that detection and alarms be provided to alert the operator of passive ECCS failures during long-term cooling. The timing of these alarms should be such that the reactor operator has sufficient time to identify and isolate the faulted ECCS line. Provide the following information regarding the ECCS leak detection system:

a. An identification and justification of the maximum leak rate;

b. The maximum allowable time for operator action, including a justification of the time interval;

c. A demonstration that the leak detection system will be sensitive enough to provide an alarm to the operator, subsequent identification by the operator of the faulted line, and, finally, permit the operator to isolate the faulted line prior to the leak creating any undesirable consequences such as flooding of redundant equipment. The minimum time to be considered for this sequence of events is 30 minutes.

d. A demonstration that the leak detection system can identify the faulted ECCS train and that the leak is isolable.

Additionally, the ECCS lead detection system must meet the following standards: (1) control room alarm; and (2) IEEE-279, except single failure requirements.

Response:

a. The ECCS are capable of withstanding passive failures of valve stem packings and pump seals following a LOCA. The maximum leakage due to a failure of this nature could be as high as 23 gpm from an HPCS, LPCS or RHR pump seal failure. Valve stem leakage would be significantly less than this.

b. The maximum allowable time for operator action is determined as the shorter of the time required to flood to the level of an ECCS pump motor in the secondary containment, or the time required to drain the suppression pool to a level below that required ECCS pump NPSH. The maximum NPSH required for any ECCS pump is 21 ft. (HPCS). With a minimum NPSH available of 36 ft., calculated in accordance with Regulatory Guide 1.1, and a leakage rate of 23 gpm, there is about 15 days of operator time available before NPSH becomes a problem. A Class IE level instrument will be installed in each ECCS pump room and it will be mounted just above floor level. After the operator receives an alarm in the control room, there is at least 44 hours of operator time available before the water level reaches the bottom of an ECCS pump, assuming a 23 gpm leak rate into the smallest ECCS pump room (RHR C).

c. The sensitivity of the leak detection system is not vital to the identification and subsequent isolation of the faulted line prior to any undesirable consequences with at least 44 hours available.

d. Any ECCS leak can be isolated, including any packing failure on any ECCS pump suction valve. This packing can be isolated by closing the valve since the valves are double-seat, wedge knife gates. With a Class IE level instrument in each ECCS pump room, there is no problem with identification of the faulted ECCS train.

The leak detection system will have a control room alarm and meet IEEE 279, except single failure requirements.

Q. 312.~~0~~17
(15.6.5)

Your statement on Page 15.6-34 of the FSAR regarding the bypass leakage of 0.11 percent per day is ambiguous. Indicate whether this leakage is in addition to, or is part of, the primary containment leakage. The bypass leakage, which will be a major contributor to offsite doses, is not listed as an assumption in Table 15.6-12 of the FSAR. Correct this omission.

Response:

The statement given on Page 15.6-34 regarding bypass leakage provided a quantitative assessment of how much bypass leakage could hypothetically occur in addition to the primary to secondary containment leakage identified in 15.6.5.5.1.2 with the resulting radiological consequences not exceeding 10 CFR 100 criteria. This hypothetical leakage value is 0.10%/day. The primary to secondary containment leakage would be processed by SGTS. Bypass leakage would not be processed prior to release. Since the bypass leakage value presented is a bounding value provided for information only, it is not appropriate to include this hypothetical leakage in the assumption identified in Table 15.6-12. If it were included, the radiological consequences would, by definition, just match the 10 CFR 100 criteria. The text on page 15.6-34 will be modified to clarify the statements.

No bypass leakage paths which would be a major contributor to offsite doses have been identified. The bypass leakage identified in 6.2.3.3 (0.74 SCFH) is less than 1/27th of the hypothetical bounding leakage valve identified above. Since this contribution will not change the conclusion that the dose consequences would be a small fraction of 10 CFR 100 criterion, the leakage is not included in Table 15.6-12.

Q. 312.018
(15.6.5)

Your latest response to Item 022.7 in your letter of November 21, 1978, indicates that you cannot provide the secondary containment pressure response following postulated loss-of-coolant accident until February 1979. Since we need this information before we can begin our evaluationof the potential offsite doses, we request that you expedite your response on this matter. Our original request for this information was forwarded to you in our first acceptance review dated June 24, 1977.

Response:

See the revised response to Question 022.007 which refers to revised FSAR 6.2.3.3.

Q 321.1

The description of the main condenser offgas treatment system is incomplete. Describe the hydrogen gas analyzers which are downstream of the recombiners and upstream of the delay portions of the system, including a brief description of analyzer checks and calibrations. We require that the analyzer(s) shall be nonsparking.

Response:

Please see 7.7.1.10.3.5 and 11.3.2.1.8.1 for the requested discussion.

Q 321.2

In 15.7.3 of the FSAR, you state that the accident analysis for postulated liquid waste tank failures will be supplied later. Provide an estimate of the date this information will be submitted.

Response:

Please see 15.7.2.5 and 15.7.3.5 for discussions of liquid waste tank failure accident analyses.

1.1 ROLE OF DFFR/MARK II PROGRAM

Operating experience with foreign and domestic BWR plants has shown that significant dynamic loadings may be imposed on suppression chamber structures and components during safety relief valve (SRV) discharge. Hydrodynamic loads in the suppression chamber resulting from postulated loss-of-coolant accident (LOCA) events have also been identified.

As a result of such concerns, a group was formed by the domestic utility owners of BWR plants with containments of Mark II configuration (Mark II Owners Group) to address the concerns related to these dynamic loads. The specific purpose of the group was to develop an overall program with the objective of defining suppression pool hydrodynamic loads on a generic level, i.e., loads which could be used for the design or assessment of design adequacy of any containment of Mark II configuration. The results to date of the Mark II Owners Group efforts can be found in the Dynamic Forcing Function Information Report (DFFR) (Reference 1-1) and related documents. Because of the continuing programs of the Mark II Owners Group, several revisions to the DFFR have been made and future revisions may be required.

The Mark II Containment Lead Plant Program Load Evaluation Report (LER) (Reference 1-2) was issued in October, 1978. The report addresses the portion of the Mark II Owner's program that provides a generic methodology for establishing design basis LOCA and SRV loads for the lead Mark II facilities (Zimmer, Shoreham, and LaSalle) and follow-on plants that wish to use the lead plant criteria. It includes an evaluation of Mark II Owner's methodology, a description of load methodologies found acceptable by NRC and the basis for NRC conclusions. Table 1.1-1 lists the containment hydrodynamic loads in the format and order given in Table IV-1 of the LER. Table 1.1-1 identifies:

a. where the review of loads conducted by NRC for lead plants is expected to be sufficient for WNP-2, i.e., the criteria given in the LER are acceptable for WNP-2 at this time,

b. where NRC review of generic items (for WNP-2 and other plants) remains to be completed,

c. where NRC review of WNP-2 plant unique information is required.

TABLE 1.1-1

CONTAINMENT HYDRODYNAMIC LOADS/LOAD EVALUATION REPORT SUMMARY

NRC ITEM NUMBER (FROM LER TABLE IV-1)		SUBJECT/LOAD		STATUS OF LICENSING
LOCA				
	I.A	SUBMERGED BOUNDARY VENT CLEARING LOAD		NRC CRITERIA GIVEN IN LER, NO ADDITIONAL NRC REVIEW FOR WNP-2 ANTICIPATED
	I.B.1	POOL SWELL ANALYTICAL MODEL		NRC CRITERIA GIVEN IN LER, NO ADDITIONAL NRC REVIEW FOR WNP-2 ANTICIPATED
	I.B.2	SUBMERGED BOUNDARY POOL SWELL LOAD		NRC CRITERIA GIVEN IN LER; NO ADDITIONAL NRC REVIEW FOR WNP-2 ANTICIPATED
	I.B.3	POOL SWELL IMPACT LOAD		FOR SMALL STRUCTURES AND GRATING, REVIEW OF WNP-2 DAR METHODS REQUIRED
	I.B.4	WETWELL AIR COMPRESSION		NRC CRITERIA GIVEN IN LER; NO ADDITIONAL NRC REVIEW FOR WNP-2 ANTICIPATED
	I.B.5	ASYMMETRIC LOAD		COMPLETION OF GENERIC REVIEW OF MK II ALTERNATE POSITION REQUIRED
	I.C.1	DOWNCOMER LATERAL LOAD		COMPLETION OF GENERIC REVIEW OF PRETECH LOAD DEFINITION REQUIRED
	I.C.2	SUBMERGED BOUNDARY STEAM CONDENSATION LOADS	C.O.:	NRC CRITERIA GIVEN IN LER; NO ADDITIONAL NRC REVIEW FOR WNP-2 ANTICIPATED
			CHUGGING:	COMPLETION OF REVIEW OF B&R CHUGGING LOAD DEFINITION REQUIRED; SEE REPORTS OF APRIL 13 AND JUNE 15
SRV				
	II.A	POOL TEMPERATURE LIMITS		NRC CRITERIA GIVEN IN LER; NO ADDITIONAL NRC REVIEW FOR WNP-2 ANTICIPATED
	II.B	AIR CLEARING LOADS		X-QUENCHER LOADS TO BE PRESENTED LATER IN WNP-2 DAR
	II.C.1	QUENCHER ARM LOADS		NRC CRITERIA GIVEN IN LER; NO ADDITIONAL NRC REVIEW FOR WNP-2 ANTICIPATED
	II.C.2	QUENCHER TIE DOWN LOADS		NRC CRITERIA GIVEN IN LER; NO ADDITIONAL NRC REVIEW FOR WNP-2 ANTICIPATED

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. The design of these pipes is governed by the pool swell impact load on the horizontal projection of the pipes above the initial pool surface combined with the drag load on submerged portions.

c. Piping Systems In Pool Swell Zone

The pool swell zone is identified in Section 3.2.3 to be between the elevations of the initial pool surface (466'-4 3/4") and the maximum pool rise during a LOCA (design basis accident) (484'-4 3/4").

Piping systems in the pool swell zone include short projections into the chamber from the containment at one access hatch and ten miscellaneous piping systems as shown in Figure 2.1-3 and Figures 2.1-6 through 2.1-8.

d. Piping Systems Above the Pool Swell Zone

Piping systems above the pool swell zone include short lengths of pipe entering at elevation 491'-0" and two penetrations for the wetwell spray header also at elevation 491'-0" as shown in Figure 2.1-4 and Figures 2.1-6 through 2.1-8.

These systems are not subjected to direct hydrodynamic loads associated with a LOCA or with SRV actuation, only building response due to hydrodynamic events would affect piping in this classification.

2.1.2 STRUCTURES, PIPING AND COMPONENTS INDIRECTLY AFFECTED BY POOL DYNAMIC LOADS

Outside of the suppression chamber, it has been postulated that structures, piping, and components may be affected by pool dynamic loads. This has been postulated to occur as a result of loading applied to the suppression chamber boundary (basemat, pedestal, and containment shell) which would result in vibratory motion elsewhere in the reactor building. This is commonly referred to in this and other reports as "building response". As discussed later in this report, studies now are underway of measured building response results at Caorso. The results to date of the investigations indicate that building response effects are small and of secondary significance as

compared to the effects of pool dynamic loads on structures within the suppression chamber directly affected by these loads. The structures, piping, and components which may be indirectly affected by pool dynamic loads will be covered in later revisions of this report and the FSAR after the completion of the Burns and Roe evaluation of Caorso test results.

2.3 SUMMARY AND CONCLUSIONS

2.3.1 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 revision covers the structures, piping, and components in and bounded by 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 2.3-1 provides a list of the structures and components that have been covered in this report and the design changes that have been made. The steel containment structure has been reinforced by the addition of several 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 dischange pipes. Additions and modifications of pipe supports for miscellaneous piping systems have been provided. Other miscellaneous changes are noted in Table 2.3-1.

For structures, piping and components affected only indirectly, by pool dynamic loads (those outside the suppression chamber), changes in design to accommodate pool dynamic loads are not being made in view of the results being obtained from the Caorso tests. Data from the Caorso tests indicate that the actual effects of SRV discharge are small as compared to currently available load definitions. A more realistic load definition will be presented in a subsequent revision of the WNP-2 DAR after the completion of the Burns and Roe evaluation of Caorso test results.

2.3.2 CONCLUSIONS

The assessment indicates 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. This conclusion is based on evaluation using loads given in the NRC Load Evaluation Report (NUREG-0487, Reference 2-1) except as noted herein, and additional data and information such as is now available from the SRV tests performed at the Caorso plant in Italy (Reference 2-2).

This revision reflects new developments in several areas of load definition. For example, chugging loads used in this report are based on work by Burns and Roe to account for fluid structure interaction effects and other effects peculiar to the 4T test facility which were not accounted for in the previous load definition (Reference 2-3). Also reflected in this revision is a LOCA jet load definition which accounts for the ring vortex character of the LOCA jet phenomenon (Reference 2-4).

Work is continuing as noted in this report to finalize the remaining load definitions. Of particular note is the work underway now at Burns and Roe to finalize SRV discharge load definition. The results of SRV testing carried out at the Caorso plant in Italy is being used to complete this work. The Caorso tests represents the most extensive SRV testing program to date with geometry and plant conditions similar to WNP-2. A detailed evaluation and definition of the SRV discharge load will be the final result of the work now underway and will be reported in a subsequent revision to this report. Although studies such as those for SRV loads remain to be completed, the assessments reported in this revision are expected to remain valid since the final loads are expected to be less than those used in this revision. This should allow NRC review of the adequacy of the WNP-2 containment system design to proceed in parallel with the continuing NRC generic and plant specific review of the remaining load definitions.

2.4 REFERENCES

2-1 Mark II Containment Lead Plant Program Load Evaluation Report, NUREG-0487, United States Nuclear Regulatory Commission, October, 1978.

2-2 Mark II Containment Supporting Program Caorso SRV Discharge Tests, Phase I Test Report, General Electric Company, NEDE-25100-P, May, 1979.

2-3 "Chugging Loads-Improved Definition and Application Methodology to Mark II Containments," Technical Report, Burns and Roe, Inc., June, 1979.

2-4 Letter MFN-144-79, from L.J. Sobon, General Electric to J.F. Stolz, Nuclear Regulatory Commission on "Technical Description of the Ring Vortex Model," May 22, 1979.

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3.1.2.3 Steam Condensation Loads

After the water and air have been expelled from the SRV discharge line, high pressure, high temperature, high mass flux steam is discharged into the suppression pool. As the steam condenses and collapses, vibrations or small pressure fluctuations are produced in the water. Experiments employing a quencher device have exhibited no significant pressure fluctuations up to the pool boiling temperature of 212°F. As a result, significant pressure fluctuations in the WNP-2 suppression pool due to steam condensation from the quenchers are not expected. This phenomenon is therefore not considered further.

3.1.3 SRV AIR CLEARING LOADS

Testing and analytical efforts have been performed by the Mark II Owners Group and by other organizations to define the loads resulting from discharge through a quencher device upon actuation of the SRV. The SRV testing carried out at the Caorso plant in Italy represents the most extensive test program to date with geometry and plant conditions similar to WNP-2. An analytical effort has been undertaken by Burns and Roe to evaluate the data taken during the Caorso Phase I and II tests. Work is in process to understand and quantify effects of SRV discharge on a Mark II containment. A detailed evaluation and definition of the SRV discharge load will be the final result of this ongoing work and will be reported in a subsequent revision to this report.

For the purposes of this report a conservative interpretation of preliminary results of the Caorso test data and the methods found in the DFFR (Reference 3-3) are used to define an SRV load specification for assessment of the adequacy of WNP-2 wetwell structures and components. The details of this interim load specification are provided in the following sections.

3.1.3.1 Boundary Loads

Data from scaled quencher experiments and in-plant quencher tests show that pressure fluctuations associated with the water and air clearing of the SRV line occur in the pool during the initial second or so after an SRV actuation. The DFFR (Reference 3-3) provides a methodology for predicting the peak positive and negative boundary pressures. It has been found that the peak boundary pressures predicted by the DFFR methodology at conditions reflective of those encountered the Caorso plant during Phase I and II tests are significantly higher than the loads recorded during the actual tests. (Reference 3-2).

The highly conservative peak pressure definitions found in the DFFR and the conservative definition for pressure distributions and forcing function time histories have been used to evaluate the structural adequacy of the containment, basemat and pedestal. The DFFR based peak pressures are shown in Table 4.1-1 and the results of the assessment of the containment, basemat and pedestal are discussed in Section 4.0.

3.1.3.2 Building Response

Building responses calculated using the DFFR (Reference 3-3) peak boundary pressure prediction and application methodology (Monte Carlo approach, Reference 3-21) are reduced by a factor of two in order to obtain interim building responses for this assessment. This reduction is justified in view of:

a) The much smaller boundary loads measured during the Caorso tests relative to the peak pressures recommended by the DFFR, (Table 3.1-2).

b) The much smaller pressure response at all frequencies for the Caorso measured pressure time histories relative to the pressure time histories for the idealized pressure time histories of the DFFR. Figure 3.1-1 shows a comparison between the two pressure responses. These response spectra were obtained by applying each of the pressure time histories as a forcing function on a single-degree-of-freedom (SDOF) system with 1% damping and computing the maximum force in the spring as the spectral value.

c) The much smaller building responses recorded during the Caorso tests in comparison with those predicted using the methods recommended by the DFFR, Figure 3.1-2.

These conservative interim building responses are used in the assessment to meet WNP-2 schedule requirements. The building response spectra are shown in Section 5.0. A more realistic load definition will be presented in a subsequent revision of the WNP-2 DAR after the completion of the Burns and Roe evaluation of Caorso test results.

3.1.3.3 Submerged Structure Loads

The pressure and velocity fluctuations created in the pool during the air and water clearing are expected to produce acceleration and standard drag forces on submerged structures.