SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT □ P. O. Box 15830, Sacramento CA 95852-1830, (916) 452-3211
AN ELECTRIC SYSTEM SERVING THE HEART OF CALIFORNIA

AGM/NUC 91-092

June 13, 1991

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Docket No. 50-312
Rancho Seco Nuclear Generating Station
License No. DPR-54
UPDATED SAFETY ANALYSIS REPORT (USAR) AMENDMENT 8

Reference: J. Larkin (NRC) to D. Keuter (SMUD) letter dated April 26, 1990, Exemption to 10 CFR 50.71 Related to Submittal of Revised USAR for Rancho Seco (TAC No. 75068)

Attention: Seymour Weiss

Pursuant to 10 CFR 50.71(e) and 10 CFR 50.4(b)(6), the Sacramento Municipal Utility District (SMUD) hereby submits the original and 10 copies of USAR Amendment 8 for Rancho Seco. This USAR Amendment 8 submittal is made in accordance with the referenced letter, which states, "Subsequent USAR updates will be submitted no less frequently than annually from June 22, 1990."

USAR Amendment 8 reflects plant changes made since the last USAR amendment through at least December 31, 1990. Plant changes made pursuant to 10 CFR 50.59 have been previously reported monthly as required by Rancho Seco Technical Specification 6.9.3, Monthly Report.

USAR Amendment 8 consists of three parts. Part 1 is the page by page Removal/Insertion instructions for the USAR pages changed in Amendment 8. Part 2 is the List Of Effective Pages updated through Amendment 8. Part 3 contains the changed USAR pages that make up Amendment 8 to the Rancho Seco USAR.

Members of your staff with questions requiring additional information or clarification may contact Jerry Delezenski at (916) 452-3211, extension 4914.

9107030248 910630
PDR ADOCK 05000312
K PDR

A053
1/10

State of California
SS
County of Sacramento

Dan R. Keuter, being first duly sworn, deposes and says: that he is Assistant General Manager, Nuclear of Sacramento Municipal Utility District (SMUD), the licensee herein; that he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information, and belief, and that he is authorized to execute this document on behalf of said licensee.

Dan R. Keuter
Assistant General Manager
Nuclear

Subscribed and affirmed to before me on this 13th day of June, 1991.

OFFICIAL SEAL
SHARON ROSENBERGER
Notary Public-California
SACRAMENTO COUNTY
My Commission Expires
January 13, 1995

Sharon Rosenberger
Notary Public

Attachment (Original plus 10 Copies of USAR Amendment 8 package)

cc: w/atch (1 copy of USAR Amendment 8 package only)
J. B. Martin, NRC, Walnut Creek
C. Myers, NRC, Rancho Seco

SACRAMENTO MUNICIPAL UTILITY DISTRICT

RANCHO SECO
NUCLEAR GENERATING STATION

UPDATED SAFETY ANALYSIS REPORT

AMENDMENT 8

JUNE 1991

PART 1

USAR Amendment 8 Removal/Insertion Instructions

RANCHO SECO NUCLEAR GENERATING STATION
UPDATED SAFETY ANALYSIS REPORT

The following information is furnished as a guide for the insertion of new pages for Amendment 8 into the Updated Safety Analysis Report for the Rancho Seco Nuclear Generating Station. New pages should be inserted as listed below. Retain these instructions in the front of Volume I as a record of changes.

Discard Old Page (Front/Back)	Insert New Page (Front/Back)
List of Effective Pages	
1/2	1/2
through	through
47/blank	47/blank
USAR Volumes Table of Contents	
xvii/xviii	xvii/xviii
Chapter 1	
Figure 1.1-3	Figure 1.1-3
Figure 1.1-6	Figure 1.1-6
1.2-1/1.2-2	1.2-1/1.2-2
1.5-3/1.5-4	1.5-3/1.5-4
1.6-1/1.6-2	1.6-1/1.6-2
1.6-5/1.6-6	1.6-5/1.6-6
1.6-7/1.6-8	1.6-7/1.6-8
Chapter 2	
2.2-5/2.2-6	2.2-5/2.2-6
Chapter 4	
4.2-41/4.2-42	4.2-41/4.2-42
Chapter 7	
7.4-1/7.4-1a	7.4-1/7.4-1a
7.4-7/7.4-8	7.4-7/7.4-8
7.4-9/7.4-10	7.4-9/7.4-10
7.4-11/7.4-12	7.4-11/blank
7.4-13/blank	--------

Discard Old Page (Front/Back)	Insert New Page (Front/Back)
Chapter 8	
8.2-1/8.2-2	8.2-1/8.2-2
8.2-5/8.2-6	8.2-5/8.2-6
8.2-7/8.2-8	8.2-7/8.2-8
8.2-11/8.2-12	8.2-11/8.2-12
through	through
8.2-17/8.2-18	8.2-17/8.2-18
Figure 8.2-2	Figure 8.2-2
Chapter 9	
9-vii/9-viii	9-vii/9-viii
9-xi/blank	9-xi/blank
Figure 9.7-1 (Sheet 1 of 5)	Figure 9.7-1 (Sheet 1 of 5)
9.9-1/9.9-2	9.9-1/9.9-2
through	through
9.9-9/9.9-10	9.9-7/9.9-8
Figure 9.9-1 (Sheet 1 of 9)	--------
through	
Figure 9.9-1 (Sheet 9 of 9)	--------
Figure 9.9-2 (Sheet 1 of 3)	--------
through	
Figure 9.9-2 (Sheet 3 of 3) (Deleted)	--------
Figure 9.10-1 (Sheet 13 of 13)	Figure 9.10-1 (Sheet 13 of 13)
Chapter 10	
---------	Figure 10.2-2 (Sheet 2A of 3)
Chapter 11	
11.2-9/blank	11.2-9/blank
Table 11.5-4 (Sheet 3 of 4)/ (Sheet 4 of 4)	Table 11.5-4 (Sheet 3 of 4)/ (Sheet 4 of 4)
Chapter 12	
12-i/12-ii	12-i/12-ii
12.1-1/12.1-2	12.1-1/12.1-2
through	through
12.1-5/blank	12.1-5/blank
Figure 12.1-1	Figure 12.1-1
Figure 12.1-2	Figure 12.1-2
12.2-1/12.2-2	12.2-1/12.2-2
through	through
12.2-5/12.2-6	12.2-5/12.2-6
12.3-1/blank	12.3-1/blank
12.5-1/12.5-2	12.5-1/12.5-2

Discard Old Page (Front/Back)	Insert New Page (Front/Back)
Appendix 2B	
2B-9/blank	2B-9/blank

PART 2

List Of Effective Pages Through USAR Amendment 8

Page or Table/Figure No.	Issue
Volume I Title Page	Original
Mattimoe Letter	Original
i	Amendment 3
ii	Original
iii	Original
iv	Original
v	Amendment 7
vi	Original
vii	Amendment 6
viii	Amendment 7
ix	Amendment 6
x	Amendment 7
xi	Original
xii	Amendment 7
xiii	Amendment 7
xiv	Amendment 6
xv	Amendment 7
xvi	Amendment 7
xvii	Amendment 7
xviii	Amendment 8
xix	Amendment 7
xx	Amendment 7
xxi	Amendment 7
xxii	Amendment 7
xxiii	Amendment 7
xxiv	Amendment 7
xxv	Amendment 7
Section 1 tab	Original
1-i	Original
1-ii	Original
1-iii	Original
1-iv	Original
1-v	Original
1-vi	Original
1-vii	Amendment 6
1-viii	Original
1-ix	Amendment 6
1-x	Amendment 6
1-xi	Original
1-xii	Amendment 3
Section 1.1 tab	Original
1.1-1	Amendment 6
Fig. 1.1-1	Amendment 4
1.1-2	Amendment 6
1.1-3	Amendment 8
1.1-4	Amendment 7
1.1-5	Amendment 6

Page or Table/Figure No.	Issue
Fig. 1.1-6	Amendment 8
1.1-7	Amendment 7
1.1-8 Sh 1	Amendment 7
1.1-8 Sh 2	Amendment 6
1.1-9	Amendment 6
1.1-10	Amendment 6
1.1-11	Amendment 6
1.1-12	Amendment 6
Section 1.2 tab	Original
1.2-1	Amendment 6
1.2-2	Amendment 8
1.2-3	Amendment 6
1.2-4	Amendment 6
1.2-5	Original
1.2-6	Amendment 6
1.2-7	Amendment 6
1.2-8	Original
Section 1.3 tab	Original
1.3-1	Original
1.3-2	Original
1.3-3	Original
1.3-4	Original
1.3-5	Original
1.3-6	Amendment 6
1.3-7	Original
1.3-8	Original
1.3-9	Original
Section 1.4 tab	Original
1.4-1	Amendment 6
1.4-2	Amendment 6
1.4-3	Original
1.4-4	Amendment 5
1.4-5	Original
1.4-6	Original
1.4-7	Original
1.4-8	Original
1.4-9	Original
1.4-10	Original
1.4-11	Original
1.4-12	Amendment 5
1.4-13	Amendment 1
1.4-14	Original
1.4-15	Original
1.4-16	Amendment 1
1.4-17	Original
1.4-18	Original
1.4-19	Original

Page or Table/Figure No.	Issue
1.4-20	Original
1.4-21	Original
1.4-22	Original
1.4-23	Amendment 6
1.4-24	Original
1.4-25	Original
1.4-26	Original
1.4-27	Original
1.4-28	Original
1.4-29	Original
1.4-30	Original
1.4-31	Original
1.4-32	Amendment 1
1.4-33	Amendment 1
1.4-34	Original
Section 1.5 tab	Original
1.5-1	Amendment 6
1.5-2	Amendment 6
1.5-3	Amendment 8
1.5-4	Amendment 8
1.5-5	Amendment 6
1.5-6	Amendment 6
1.5-7	Original
1.5-8	Original
1.5-9	Original
1.5-10	Original
1.5-11	Amendment 1
1.5-12	Amendment 6
1.5-13	Amendment 6
1.5-14	Amendment 6
1.5-15	Amendment 6
1.5-16	Original
1.5-17	Original
1.5-18	Original
1.5-19	Original
1.5-20	Amendment 6
1.5-21	Amendment 6
1.5-22	Original
1.5-23	Original
1.5-24	Amendment 6
1.5-25	Original
1.5-26	Original
1.5-27	Amendment 5
1.5-28	Original
1.5-29	Amendment 6
1.5-30	Amendment 6
1.5-31	Original
1.5-32	Original
1.5-33	Amendment 5

Page or Table/Figure No.	Issue
1.5-34	Amendment 6
1.5-35	Amendment 6
1.5-36	Amendment 6
1.5-36a	Amendment 6
1.5-37	Amendment 7
1.5-38	Amendment 6
1.5-39	Amendment 6
1.5-40	Amendment 6
1.5-41	Amendment 6
Section 1.6 tab	Original
1.6-1	Amendment 6
1.6-2	Amendment 6
1.6-3	Amendment 6
1.6-4	Amendment 7
1.6-5	Amendment 6
1.6-6	Amendment 8
1.6-7	Amendment 8
1.6-8	Original
1.6-9	Amendment 6
Fig. 1.6-1	Original
1.6-2	Original
1.6-3	Original
1.6-4	Original
Section 1.7 tab	Original
1.7.1	Amendment 6
1.7-2	Original
1.7-3	Original
1.7-4	Original
1.7-5	Original
Section 1.8 tab	Original
1.8-1	Original
Section 1.9 tab	Original
1.9-1	Amendment 5
Section 1.10 tab	Original
1.10-1	Original
Section 2 tab	Original
2-i	Original
2-ii	Original
2-iii	Original
2-iv	Original
Section 2.1 tab	Original
2.1-1	Original
Section 2.2 tab	Original
2.2-1	Amendment 4
2.2-2	Original

Page or Table/Figure No.	Issue
2.2-3	Amendment 4
2.2-4	Original
2.2-5	Amendment 8
2.2-6	Original
2.2-7	Original
2.2-8	Original
2.2-9	Original
2.2-10	Original
2.2-11	Original
2.2-12	Original
2.2-13	Original
2.2-14	Amendment 1
Fig. 2.2-1	Original
2.2-2	Original
2.2-3	Original
2.2-4	Original
2.2-5	Original
2.2-6	Original
2.2-7	Original
2.2-8	Original
2.2-9	Original
Section 2.3 tab	Original
2.3-1	Original
2.3-2	Original
2.3-3	Original
2.3-4	Original
2.3-5	Original
2.3-6	Original
2.3-7	Original
2.3-8	Original
2.3-9	Original
2.3-10	Original
2.3-11	Amendment 3
2.3-12	Original
2.3-13	Original
2.3-14	Original
2.3-15	Original
Fig. 2.3-1	Original
2.3-2	Original
Section 2.4 tab	Original
2.4-1	Original
2.4-2	Original
2.4-3	Original
2.4-4	Original
2.4-5	Original
2.4-6	Original
2.4-7	Original
2.4-8	Original

Page or Table/Figure No.	Issue
2.4-9	Original
2.4-10	Original
2.4-11	Original
Fig. 2.4-1	Original
2.4-2	Original
2.4-3	Original
2.4-4	Original
2.4-5	Original
2.4-6	Original
2.4-7	Original
2.4-8	Original
2.4-9	Original
2.4-10	Original
2.4-11	Original
2.4-12	Original
2.4-13	Original
2.4-14	Original
2.4-15	Original
Section 2.5 tab	Original
2.5-1	Original
Section 2.6 tab	Original
2.6-1	Original
Section 2.7 tab	Original
2.7-1	Original
Section 2.8 tab	Original
2.8-1	Amendment 7
Section 2.9 tab	Original
2.9-1	Original
Volume II Title Page	Original
Section 3 tab	Original
3-i	Amendment 4
3-ii	Original
3-iii	Amendment 4
3-iv	Amendment 4
3-v	Amendment 4
3-vi	Original
Section 3.1 tab	Original
3.1-1	Amendment 2
3.1-2	Amendment 4
3.1-3	Original
3.1-4	Amendment 4
3.1-5	Original
3.1-6	Original
3.1-7	Amendment 2
Section 3.2 tab	Original
3.2-1	Amendment 4
3.2-2	Amendment 4

Page or Table/Figure No.	Issue
3.2-3	Amendment 4
3.2-4	Amendment 4
3.2-5	Amendment 4
3.2-6	Amendment 4
3.2-7	Amendment 4
3.2-8	Original
3.2-9	Original
3.2-10	Amendment 4
3.2-11	Amendment 4
3.2-12	Original
3.2-13	Amendment 4
3.2-14	Original
3.2-15	Amendment 2
3.2-16	Amendment 2
3.2-16a	Amendment 2
3.2-17	Amendment 4
3.2-18	Amendment 4
3.2-19	Amendment 4
3.2-20	Original
3.2-21	Amendment 4
3.2-22	Amendment 4
3.2-23	Original
3.2-24	Original
3.2-25	Original
3.2-26	Original
3.2-27	Original
3.2-28	Original
3.2-29	Amendment 2
3.2-29a	Amendment 2
3.2-30	Original
3.2-31	Original
3.2-32	Amendment 5
3.2-33	Original
3.2-34	Amendment 4
3.2-34a	Amendment 4
3.2-35	Original
3.2-36	Original
3.2-37	Amendment 5
3.2-38	Amendment 4
3.2-38a	Amendment 2
3.2-39	Amendment 4
3.2-40	Amendment 3
3.2-41	Amendment 3
3.2-42	Amendment 3
3.2-43	Original
3.2-44	Original
3.2-45	Original
3.2-46	Original

Page or Table/Figure No.	Issue
3.2-47	Original
3.2-48	Original
3.2-49	Original
3.2-50	Original
3.2-51	Original
3.2-52	Original
3.2-53	Original
3.2-54	Amendment 4
3.2-55	Original
3.2-56	Original
3.2-57	Amendment 4
3.2-58	Amendment 4
3.2-58a	Amendment 4
3.2-59	Original
3.2-60	Amendment 3
3.2-61	Original
3.2-62	Amendment 3
3.2-63	Original
3.2-64	Original
3.2-65	Original
3.2-66	Original
Fig. 3.2-1	Amendment 4
3.2-2	Amendment 4
3.2-2a	Amendment 4
3.2-2b	Amendment 4
3.2-2c	Amendment 4
3.2-2d	Amendment 4
3.2-2e	Amendment 4
3.2-2f	Amendment 4
3.2-2g	Original
3.2-3	Original
3.2-4	Original
3.2-5	Amendment 2
3.2-6	Original
3.2-7	Original
3.2-8	Original
3.2-9	Original
3.2-10	Amendment 7
3.2-11	Original
3.2-12	Original
3.2-13	Original
3.2-14	Original
3.2-15	Original
3.2-16	Original
3.2-17	Original
3.2-18	Original

Page or Table/Figure No.	Issue
Fig. 3.2-19	Original
Section 3.3 tab	Original
3.3-1	Original
3.3-2	Amendment 4
3.3-2a	Amendment 4
3.3-3	Original
3.3-4	Amendment 4
3.3-5	Amendment 5
3.3-6	Amendment 5
3.3-7	Amendment 5
3.3-8	Amendment 5
3.3-9	Amendment 5
Section 3.4 tab	Original
3.4-1	Amendment 4
3.4-2	Amendment 4
3.4-3	Amendment 4
3.4-4	Amendment 4
3.4-5	Amendment 4
Section 4 tab	Original
4-i	Amendment 5
4-ii	Amendment 7
4-iii	Amendment 6
4-iv	Amendment 6
4-v	Amendment 6
4-vi	Original
4-vii	Original
4-viii	Amendment 7
Section 4.1 tab	Original
4.1-1	Amendment 6
4.1-2	Original
4.1-3	Amendment 5
4.1-4	Amendment 5
4.1-5	Amendment 5
4.1-6	Original
4.1-7	Original
4.1-8	Original
4.1-9	Original
4.1-10	Original
4.1-11	Original
4.1-12	Original
4.1-13	Amendment 6
4.1-14	Amendment 6
4.1-15	Amendment 6
4.1-16	Amendment 5
4.1-16a	Amendment 5
4.1-17	Amendment 5
4.1-18	Amendment 6
4.1-19	Original
4.1-20	Original

Page or Table/Figure No.	Issue
4.1-21	Amendment 5
4.1-21a	Amendment 5
4.1-22	Amendment 6
Section 4.2 tab	Original
4.2-1	Original
4.2-2	Amendment 6
4.2-3	Original
4.2-4	Amendment 5
4.2-4a	Amendment 7
4.2-5	Amendment 5
4.2-6	Amendment 4
4.2-7	Amendment 6
4.2-8	Amendment 6
4.2-9	Original
4.2-10	Original
4.2-11	Original
4.2-12	Amendment 7
4.2-13	Amendment 5
4.2-14	Original
4.2-15	Original
4.2-16	Amendment 5
4.2-17	Amendment 7
4.2-18	Amendment 7
4.2-19	Amendment 7
4.2-20	Amendment 7
4.2-21	Original
4.2-22	Original
4.2-23	Original
4.2-24	Original
4.2-25	Original
4.2-26	Original
4.2-27	Amendment 7
4.2-28	Amendment 7
4.2-29	Original
4.2-30	Amendment 1
4.2-31	Amendment 6
4.2-32	Amendment 6
4.2-33	Amendment 6
4.2-33a	Amendment 5
4.2-34	Original
4.2-35	Amendment 6
4.2-36	Amendment 6
4.2-37	Original
4.2-38	Original
4.2-39	Original
4.2-40	Amendment 1
4.2-41	Amendment 2
4.2-42	Amendment 8
4.2-42a	Amendment 6

Page or Table/Figure No.	Issue
4.2-43	Amendment 5
4.2-44	Amendment 6
4.2-45	Amendment 6
4.2-46	Amendment 6
4.2-47	Amendment 6
4.2-48	Amendment 6
4.2-49	Amendment 6
Fig. 4.2-1 Sh. 1	Amendment 7
4.2-1 Sh. 2	Amendment 7
4.2-1 Sh. 3	Amendment 6
4.2-2	Original
4.2-3	Original
4.2-4	Original
4.2-5	Amendment 1
4.2-5a	Original
4.2-6	Original
4.2-6a	Original
4.2-7	Original
4.2-8	Original
4.2-9	Original
4.2-10	Amendment 7
Section 4.3 tab	Original
4.3-1	Amendment 1
4.3-2	Original
4.3-3	Original
4.3-4	Original
4.3-5	Original
4.3-6	Original
4.3-7	Amendment 7
4.3-8	Original
4.3-9	Original
4.3-10	Original
4.3-11	Original
4.3-12	Original
4.3-13	Original
4.3-14	Original
4.3-15	Amendment 5
4.3-16	Amendment 5
4.3-17	Original
4.3-18	Amendment 1
4.3-19	Amendment 5
4.3-19a	Amendment 4
4.3-20	Amendment 5
4.3-21	Original
4.3-22	Original
4.3-23	Original

Page or Table/Figure No.	Issue
4.3-24	Amendment 5
4.3-24a	Amendment 5
4.3-25	Original
4.3-26	Original
4.3-27	Original
4.3-28	Original
4.3-29	Original
4.3-30	Amendment 2
4.3-30a	Amendment 2
4.3-30b	Amendment 7
4.3-31	Amendment 6
4.3-32	Original
4.3-33	Amendment 4
4.3-34	Amendment 6
4.3-35	Amendment 6
4.3-36	Amendment 6
4.3-37	Amendment 4
4.3-38	Amendment 4
4.3-39	Amendment 4
4.3-40	Amendment 4
Fig. 4.3-1	Amendment 5
4.3-2	Original
Section tab 4.4	Original
4.4-1	Amendment 6
4.4-2	Amendment 6
4.4-2a	Amendment 7
4.4-2b	Amendment 4
4.4-3	Amendment 4
4.4-4	Amendment 7
Section 4.5 tab	Original
4.5-1	Original
4.5-2	Amendment 6
Volume III title page	Original
Section 5 tab	Original
5-i	Amendment 6
5-ii	Amendment 3
5-iii	Amendment 6
5-iv	Amendment 6
5-v	Amendment 6
5-vi	Amendment 6
5-via	Amendment 6
5-vii	Amendment 3
5-viii	Amendment 3
5-ix	Amendment 3
5-x	Amendment 3
5-xi	Amendment 3
5-xii	Amendment 6
5-xiii	Amendment 3

Page or Table/Figure No.	Issue
Section 5.1 tab	Original
5.1-1	Amendment 6
5.1-2	Amendment 6
5.1-3	Amendment 6
5.1-4	Amendment 6
5.1-5	Amendment 6
5.1-6	Amendment 6
5.1-7	Amendment 6
5.1-8	Amendment 6
5.1-9	Amendment 3
5.1-10	Amendment 3
5.1-11	Amendment 3
5.1-12	Amendment 3
5.1-12a	Amendment 6
5.1-13	Amendment 6
5.1-14	Amendment 3
5.1-15	Amendment 7
5.1-16	Amendment 7
5.1-17	Amendment 3
5.1-18	Amendment 3
5.1-19	Amendment 3
5.1-20	Amendment 3
5.1-21	Amendment 6
5.1-22	Amendment 6
5.1-23	Amendment 3
5.1.24	Amendment 3
5.1-25	Amendment 7
5.1-26	Amendment 3
5.1-27	Amendment 3
5.1-28	Amendment 3
5.1-29	Amendment 3
5.1-30	Amendment 3
5.1-31	Amendment 5
5.1-32	Amendment 3
5.1-33	Amendment 3
5.1-34	Amendment 3
5.1-35	Amendment 7
5.1-36	Amendment 3
Fig. 5.1-1	Original
5.1-2	Original
5.1-3	Original
5.1-4	Original
5.1-5	Original
5.1-6 Sh. 1	Original
5.1-6 Sh. 2	Original
5.1-7	Original
5.1-8	Original
5.1-9	Original

Page or Table/Figure No.	Issue
Section 5.2 tab	Original
5.2-1	Original
5.2-2	Original
5.2-3	Original
5.2-4	Original
5.2-5	Original
5.2-6	Original
5.2-7	Amendment 5
5.2-8	Original
5.2-9	Amendment 5
5.2-10	Amendment 5
5.2-11	Amendment 5
5.2-12	Original
5.2-13	Original
5.2-14	Original
5.2-15	Original
5.2-16	Original
5.2-17	Original
5.2-18	Original
5.2-19	Amendment 7
5.2-20	Amendment 7
5.2-21	Original
5.2-22	Amendment 7
5.2-23	Original
5.2-24	Amendment 7
5.2-25	Original
5.2-26	Original
5.2-27	Original
5.2-28	Original
5.2-29	Original
5.2-30	Original
5.2-31	Original
5.2-32	Original
5.2-33	Original
5.2-34	Original
5.2-35	Original
5.2-36	Original
5.2-37	Original
5.2-38	Original
5.2-39	Amendment 2
5.2-40	Amendment 4
5.2-40a	Amendment 2
5.2-40b	Amendment 5
5.2-41	Amendment 2
5.2-41a	Amendment 2
5.2-42	Amendment 6
5.2-43	Amendment 7
5.2-43a	Amendment 7
5.2-44	Amendment 6

Page or Table/Figure No.	Issue
5.2-45	Amendment 7
5.2-46	Original
5.2-47	Original
5.2-48	Original
5.2-49	Original
5.2-50	Original
5.2-51	Original
5.2-52	Original
5.2-53	Original
5.2-54	Original
5.2-55	Original
5.2-56	Original
5.2-57	Original
5.2-58	Amendment 7
5.2-59	Amendment 6
5.2-60	Amendment 6
5.2-61	Amendment 6
5.2-61a	Amendment 6
5.2-62	Amendment 6
5.2-63	Original
5.2-64	Original
5.2-65	Original
5.2-66	Original
Fig. 5.2-1 Sh. 1	Original
5.2-1 Sh. 2	Original
5.2-1 Sh. 3	Original
5.2-2 Sh. 1	Original
5.2-2 Sh. 2	Original
5.2-3	Original
5.2-4	Original
5.2-5 Sh. 1	Original
5.2-5 Sh. 2	Original
5.2-5 Sh. 3	Original
5.2-5 Sh. 4	Original
5.2-5 Sh. 5	Original
5.2-5 Sh. 6	Original
5.2-5 Sh. 7	Original
5.2-5 Sh. 8	Original
5.2-5 Sh. 9	Original
5.2-6 Sh. 1	Original
5.2-6 Sh. 2	Original
5.2-6 Sh. 3	Original
5.2-6 Sh. 4	Original
5.2-6 Sh. 5	Original
5.2-6 Sh. 6	Original
5.2-6 Sh. 7	Original
5.2-6 Sh. 8	Original
5.2-6 Sh. 9	Original

Page or Table/Figure No.	Issue
5.2-6 Sh. 10	Original
5.2-6 Sh. 11	Original
5.2-6 Sh. 12	Original
5.2-6 Sh. 13	Original
5.2-6 Sh. 14	Original
5.2-6 Sh. 15	Original
5.2-6 Sh. 16	Original
5.2-6 Sh. 17	Original
5.2-6 Sh. 18	Original
5.2-6 Sh. 19	Original
5.2-6 Sh. 20	Original
5.2-7	Original
5.2-8	Original
5.2-9	Original
5.2-10	Original
5.2-11	Original
5.2-12	Original
5.2-13	Original
5.2-14	Original
5.2-15	Original
5.2-16	Original
5.2-17 Sh. 1	Original
5.2-17 Sh. 2	Original
5.2-18	Original
5.2-19	Original
5.2-20	Original
5.2-21 Sh. 1	Amendment 7
5.2-21 Sh. 2	Amendment 6
5.2-22	Original
Section 5.3 tab	Original
5.3-1	Original
5.3-2	Original
5.3-3	Original
Fig. 5.3-1	Original
Section 5.4 tab	Original
5.4-1	Amendment 3
5.4-2	Amendment 6
5.4-3	Original
5.4-4	Original
5.4-5	Original
Fig. 5.4-1	Original
Section 5.5 tab	Original
5.5-1	Original
5.5-2	Original
5.5-3	Amendment 5
5.5-4	Amendment 6
5.5-5	Amendment 6
5.5-6	Amendment 6
5.5-7	Amendment 6

Page or Table/Figure No.	Issue
5.5-8	Amendment 7
5.5-9	Amendment 6
5.5-10	Amendment 6
5.5-11	Amendment 6
5.5-12	Amendment 6
5.5-13	Amendment 6
5.5-14	Amendment 6
5.5-15	Amendment 6
5.5-16	Amendment 6
5.5-17	Amendment 6
Section 5.6 tab	Original
5.6-1	Amendment 3
5.6-2	Original
5.6-3	Original
5.6-4	Original
5.6-5	Original
5.6-6	Original
5.6-7	Original
5.6-8	Original
5.6-9	Original
5.6-10	Original
5.6-11	Original
5.6-12	Amendment 1
5.6-13	Original
5.6-14	Original
5.6-15	Original
5.6-16	Original
5.6-17	Original
5.6-18	Original
5.6-19	Amendment 6
5.6-20	Original
5.6-21	Original
5.6-22	Original
5.6-23	Amendment 5
5.6-24	Amendment 5
5.6-25	Original
5.6-26	Original
5.6.27	Original
5.6-28	Original
5.6-29	Original
Fig. 5.6-1	Original
Section 5.7 tab	Original
5.7-1	Amendment 4
Section 5.8 tab	Original
5.8-1	Amendment 6
Section 6 tab	Original
6-i	Amendment 6
6-ii	Amendment 6
6-iii	Amendment 6

Page or Table/Figure No.	Issue
6-iiia	Amendment 6
6-iv	Amendment 6
6-v	Amendment 6
Section 6.1 tab	Original
6.1-1	Amendment 6
6.1-2	Amendment 6
6.1-3	Amendment 6
Fig. 6.1-1	Amendment 6
Section 6.2 tab	Original
6.2-1	Amendment 7
6.2-2	Amendment 6
6.2-3	Amendment 6
6.2-4	Original
6.2-5	Amendment 7
6.2-6	Original
6.2-7	Original
6.2-8	Original
6.2-9	Original
6.2-10	Amendment 7
6.2-11	Amendment 6
6.2-11a	Amendment 6
6.2-12	Original
6.2-13	Amendment 6
6.2-14	Original
6.2-15	Amendment 6
6.2-15a	Amendment 6
6.2-16	Amendment 6
6.2-17	Amendment 6
6.2-17a	Amendment 6
6.2-18	Amendment 4
6.2-19	Amendment 6
6.2-20	Amendment 6
Fig. 6.2-1 Sh. 1	Amendment 6
6.2-1 Sh. 2	Amendment 6
6.2-1 Sh. 3	Amendment 6
6.2-2	Amendment 6
6.2-3	Original
6.2-4	Amendment 6
Section 6.3 tab	Original
6.3-1	Amendment 6
6.3-2	Amendment 4
6.3-3	Original
6.3-4	Original
6.3-5	Amendment 2
6.3-6	Amendment 2
6.3-7	Amendment 6
6.3-8	Amendment 6
Fig. 6.3-1	Amendment 6
6.3-2	Amendment 6

Page or Table/Figure No.	Issue
Section 6.4 tab	Original
6.4-1	Amendment 6
6.4-2	Amendment 6
6.4-3	Amendment 6
6.4-4	Amendment 6
6.4-5	Amendment 4
6.4-6	Amendment 6
6.4-7	Amendment 6
6.4-8	Amendment 6
Fig. 6.4-1	Original
6.4-2	Amendment 7
Section 6.5 tab	Original
6.5-1	Original
6.5-2	Amendment 6
6.5-3	Original
6.5-4	Amendment 5
Section 6.6 tab	Original
6.6-1	Amendment 6
6.6-2	Amendment 6
6.6-3	Amendment 6
6.6-4	Amendment 6
6.6-5	Amendment 6
6.6-6	Amendment 6
Fig. 6.6-1	Amendment 7
Section 6.7 tab	Amendment 6
6.7-1	Amendment 6
Volume IV title page	Original
Section 7 tab	Original
7-i	Amendment 4
7-ii	Amendment 6
7-iii	Amendment 6
7-iv	Amendment 6
7-v	Amendment 6
7-vi	Amendment 7
7-vii	Amendment 6
7-viii	Amendment 6
7-ix	Amendment 6
Section 7.1 tab	Original
7.1-1	Amendment 6
7.1-2	Amendment 6
7.1-2a	Amendment 4
7.1-3	Original
7.1-4	Amendment 4
7.1-5	Amendment 6
7.1-5a	Amendment 6
7.1-5b	Amendment 6
7.1-6	Amendment 5
7.1-7	Amendment 4
7.1-7a	Amendment 4

Page or Table/Figure No.	Issue
7.1-8	Amendment 5
7.1-9	Amendment 5
7.1-10	Amendment 5
7.1-11	Amendment 6
7.1-12	Amendment 6
7.1-13	Original
7.1-14	Original
7.1-15	Amendment 6
7.1-15a	Amendment 6
7.1-16	Amendment 6
7.1-17	Amendment 6
7.1-17a	Amendment 6
7.1-17b	Amendment 6
7.1-18	Amendment 2
7.1-19	Amendment 6
7.1-20	Amendment 6
7.1-21	Amendment 6
7.1-22	Amendment 6
7.1-23	Amendment 6
7.1-24	Amendment 5
7.1-25	Amendment 6
7.1-26	Amendment 6
7.1-27	Original
7.1-28	Amendment 1
7.1-29	Original
7.1-30	Amendment 5
7.1-31	Amendment 6
7.1-32	Amendment 6
7.1-33	Amendment 6
7.1-34	Amendment 7
7.1-35	Amendment 6
7.1-36	Amendment 6
7.1-37	Amendment 6
7.1-38	Amendment 6
7.1-39	Amendment 6
7.1-40	Amendment 6
7.1-41	Amendment 6
Fig. 7.1-1 Sh. 1	Amendment 6
7.1-1 Sh. 2	Amendment 6
7.1-1 Sh. 3	Amendment 6
7.1-1 Sh. 4	Amendment 6
7.1-1 Sh. 5	Amendment 6
7.1-2	Original
7.1-3	Original
7.1-4	Original
7.1-5	Amendment 5
7.1-6 Sh. 1	Amendment 6
7.1-6 Sh. 2	Amendment 6
7.1-6 Sh. 3	Amendment 5

Page or Table/Figure No.	Issue
7.1-6 Sh. 4	Amendment 7
7.1-6 Sh. 5	Amendment 6
7.1-6 Sh. 6	Amendment 5
7.1-6 Sh. 7	Amendment 6
7.1-6 Sh. 8	Amendment 5
7.1-6 Sh. 9	Amendment 6
7.1-6 Sh. 10	Amendment 6
7.1-6 Sh. 11	Amendment 6
7.1-6 Sh. 12	Amendment 5
7.1-7	Amendment 6
7.1-8	Amendment 6
7.1-9	Amendment 6
7.1-10	Amendment 6
7.1-11	Amendment 6
7.1-12	Amendment 6
7.1-13	Amendment 6
Section 7.2 tab	Original
7.2-1	Original
7.2-2	Original
7.2-3	Original
7.2-4	Original
7.2-5	Amendment 6
7.2-6	Amendment 6
7.2-7	Amendment 6
7.2-8	Original
7.2-9	Original
7.2-10	Amendment 1
7.2-11	Original
7.2-12	Amendment 6
7.2-13	Amendment 6
7.2-14	Amendment 6
7.2-15	Amendment 6
7.2-16	Amendment 6
7.2-17	Amendment 6
7.2-18	Amendment 6
7.2-19	Amendment 6
7.2-20	Amendment 6
7.2-21	Amendment 6
Fig. 7.2-1	Original
7.2-2	Amendment 6
7.2-3	Amendment 5
7.2-4	Amendment 6
7.2-5	Amendment 6
7.2-6	Amendment 6
7.2-7	Amendment 6
7.2-8	Original
7.2-9	Original
Section 7.3 tab	Original
7.3-1	Amendment 5

Page or Table/Figure No.	Issue
7.3-2	Amendment 5
7.3-3	Original
7.3-4	Amendment 6
7.3-5	Original
7.3-6	Amendment 6
7.3-7	Amendment 6
7.3-8	Amendment 6
7.3-9	Amendment 6
7.3-10	Amendment 6
7.3-11	Amendment 6
7.3-12	Amendment 6
7.3-12a	Amendment 4
7.3-13	Original
7.3-14	Original
Fig. 7.3-1	Original
7.3-2	Original
7.3-3	Original
7.3-4	Original
7.3-5	Original
7.3-6	Original
Section 7.4 tab	Original
7.4-1	Amendment 8
7.4-1a	Amendment 7
7.4-2	Amendment 6
7.4-2a	Amendment 6
7.4-2b	Amendment 4
7.4-2c	Amendment 6
7.4-3	Amendment 6
7.4-4	Amendment 6
7.4-4a	Amendment 6
7.4-4b	Amendment 6
7.4-4c	Amendment 6
7.4-4d	Amendment 6
7.4-4e	Amendment 7
7.4-4f	Amendment 6
7.4-4g	Amendment 6
7.4-4h	Amendment 6
7.4-4i	Amendment 6
7.4-4j	Amendment 6
7.4-5	Amendment 6
7.4-6	Amendment 6
7.4-7	Amendment 6
7.4-8	Amendment 8
7.4-9	Amendment 8
7.4-10	Amendment 6
7.4-11	Amendment 8
Fig. 7.4-1	Amendment 6

Page or Table/Figure No.	Issue
Section 7.5 tab	Original
7.5-1	Amendment 6
7.5-2	Amendment 6
7.5-3	Amendment 6
Section 7.6 tab	Original
7.6-1	Amendment 6
Section 8 tab	Original
8-i	Amendment 6
8-ii	Amendment 6
8-iii	Amendment 6
Section 8.1 tab	Original
8.1-1	Amendment 6
Section 8.2 tab	Original
8.2-1	Amendment 8
8.2-2	Amendment 6
8.2-3	Amendment 6
8.2-4	Amendment 6
8.2-5	Amendment 6
8.2-6	Amendment 8
8.2-7	Amendment 6
8.2-8	Amendment 8
8.2-9	Amendment 6
8.2-10	Amendment 6
8.2-11	Amendment 6
8.2-12	Amendment 8
8.2-13	Amendment 8
8.2-14	Amendment 6
8.2-15	Amendment 8
8.2-16	Amendment 6
8.2-17	Amendment 8
8.2-18	Amendment 6
8.2-19	Amendment 6
8.2-20	Amendment 6
8.2-21	Amendment 6
8.2-22	Amendment 6
8.2-23	Amendment 6
8.2-24	Amendment 6
8.2-25	Amendment 6
8.2-26	Amendment 6
8.2-27	Amendment 6
8.2-28	Amendment 6
8.2-29	Amendment 6
8.2-30	Amendment 6
8.2-31	Amendment 6
8.2-32	Amendment 6
8.2-33	Amendment 6
8.2-34	Amendment 6
8.2-35	Amendment 6
8.2-36	Amendment 6

Page or Table/Figure No.	Issue
8.2-37	Amendment 6
8.2-38	Amendment 6
Fig. 8.2-1	Amendment 6
8.2-2	Amendment 8
8.2-3	Amendment 7
8.2-4 Sh. 1	Amendment 7
8.2-4 Sh. 2	Amendment 6
8.2-5 Sh. 1	Amendment 6
8.2-5 Sh. 2	Amendment 7
8.2-5 Sh. 3	Amendment 7
8.2-6	Original
8.2-7	Original
Section 8.3 tab	Original
8.3-1	Amendment 6
8.3-2	Amendment 6
8.3-3	Amendment 6
8.3-4	Original
8.3-5	Original
8.3-6	Original
8.3-7	Original
8.3-8	Original
Section 8.4 tab	Original
8.4-1	Amendment 6
Section 9 tab	Original
9-i	Original
9-ii	Amendment 6
9-iii	Amendment 6
9-iv	Amendment 6
9-v	Amendment 6
9-vi	Amendment 6
9-vii	Amendment 8
9-viii	Amendment 6
9-ix	Amendment 6
9-x	Amendment 6
9-xi	Amendment 8
Section 9.1 tab	Original
9.1-1	Amendment 4
9.1-2	Amendment 5
Fig. 9.1-1	Original
Section 9.2 tab	Original
9.2-1	Amendment 6
9.2-2	Amendment 6
9.2-3	Amendment 6
9.2-4	Amendment 6
9.2-5	Amendment 6
9.2-6	Amendment 6
9.2-7	Amendment 6
9.2-8	Amendment 6
9.2-9	Amendment 6

Page or Table/Figure No.	Issue
9.2-10	Amendment 6
9.2-11	Amendment 6
Fig. 9.2-1 Sh. 1	Amendment 7
9.2-1 Sh. 2	Amendment 6
9.2-1 Sh. 3	Amendment 6
Section 9.3 tab	Original
9.3-1	Amendment 6
9.3-2	Amendment 4
9.3-3	Amendment 6
9.3-4	Amendment 6
9.3-5	Original
9.3-6	Original
9.3-7	Amendment 4
9.3-8	Amendment 6
9.3-8a	Amendment 6
9.3-9	Amendment 6
9.3-10	Amendment 6
Fig. 9.3-1	Amendment 2
Section 9.4 tab	Original
9.4-1	Original
9.4-2	Amendment 6
9.4-3	Amendment 7
9.4-4	Amendment 6
9.4-5	Amendment 6
9.4-6	Amendment 6
9.4-7	Amendment 5
9.4-8	Original
9.4-9	Original
9.4-10	Original
9.4-11	Original
9.4-12	Original
9.4-13	Amendment 7
9.4-14	Amendment 7
9.4-15	Original
9.4-16	Amendment 7
9.4-17	Original
9.4-18	Original
9.4-19	Amendment 6
9.4-20	Original
9.4-21	Original
9.4-22	Original
9.4-23	Original
9.4-24	Original
9.4-25	Amendment 6
9.4-26	Amendment 6
9.4-27	Amendment 6
9.4-28	Amendment 6
9.4-28a	Amendment 2
9.4-29	Original

Page or Table/Figure No.	Issue
9.4-30	Amendment 6
9.4-31	Amendment 6
9.4-32	Amendment 6
9.4-33	Amendment 4
9.4-34	Amendment 7
9.4-35	Amendment 2
Fig. 9.4-1	Amendment 6
9.4-2	Amendment 7
9.4-3	Original
9.4-4	Original
9.4-5	Original
9.4-6	Original
9.4-7	Original
9.4-8	Original
9.4-9	Original
9.4-10	Original
9.4-11	Original
9.4-12	Original
9.4-13	Amendment 6
9.4-14	Amendment 7
9.4-15 Sh. 1	Amendment 7
9.4-15 Sh. 2	Amendment 7
9.4-16 Sh. 1	Amendment 6
9.4-16 Sh. 2	Amendment 7
9.4-16 Sh. 3	Amendment 6
9.4-16 Sh. 4	Amendment 7
9.4-17	Amendment 6
Section 9.5 tab	Original
9.5-1	Amendment 6
9.5-2	Original
9.5-3	Amendment 7
9.5-4	Amendment 6
9.5-5	Amendment 6
9.5-6	Amendment 6
9.5-7	Amendment 6
Fig. 9.5-1 Sh. 1	Amendment 6
9.5-1 Sh. 2	Amendment 6
9.5-1 Sh. 3	Amendment 6
9.5-2	Original
Section 9.6 tab	Original
9.6-1	Amendment 6
9.6-1a	Amendment 3
9.6-1b	Amendment 3
9.6-1c	Amendment 3
9.6-2	Amendment 5
9.6-3	Original
9.6-3a	Amendment 3
9.6-3b	Amendment 3
9.6-3c	Amendment 3

Page or Table/Figure No.	Issue
9.6-3d	Amendment 3
9.6-3e	Amendment 3
9.6-4	Amendment 3
9.6-5	Amendment 6
Fig. 9.6-1	Amendment 7
Section 9.7 tab	Original
9.7-1	Amendment 6
9.7-2	Amendment 6
9.7-3	Amendment 7
9.7-4	Amendment 6
9.7-5	Amendment 6
9.7-6	Amendment 6
9.7-7	Amendment 7
9.7-8	Amendment 7
9.7-9	Amendment 7
9.7-10	Amendment 7
9.7-11	Amendment 7
9.7-12	Amendment 7
9.7-13	Amendment 7
9.7-14	Amendment 7
9.7-15	Amendment 7
9.7-16	Amendment 7
9.7-17	Amendment 6
9.7-18	Amendment 6
9.7-19	Amendment 6
9.7-20	Amendment 6
9.7-21	Amendment 7
9.7-22	Amendment 6
9.7-23	Amendment 6
9.7-24	Amendment 6
9.7-25	Amendment 6
9.7-26	Amendment 6
Fig. 9.7-1 Sh. 1	Amendment 8
9.7-1 Sh. 2	Amendment 7
9.7-1 Sh. 3	Amendment 7
9.7-1 Sh. 4	Amendment 7
9.7-1 Sh. 5	Amendment 7
9.7-2 Sh. 1	Amendment 7
9.7-2 Sh. 2	Amendment 7
9.7-2 Sh. 3	Amendment 7
9.7-2 Sh. 4	Amendment 7
9.7-2 Sh. 5	Amendment 6
9.7-2 Sh. 6	Amendment 7
9.7-2 Sh. 7	Amendment 7
9.7-2 Sh. 8	Amendment 6
9.7-2 Sh. 9	Amendment 6
9.7-2 Sh. 10	Amendment 7
9.7-2 Sh. 11	Amendment 7

Page or Table/Figure No.	Issue
Fig. 9.7-3 Sh. 1	Amendment 7
9.7-3 Sh. 2	Amendment 7
9.7-4	Amendment 7
9.7-5	Amendment 6
9.7-6 Sh. 1	Amendment 6
9.7-6 Sh. 2	Amendment 7
9.7-6 Sh. 3	Amendment 7
9.7-6 Sh. 4	Amendment 6
9.7-7	Amendment 6
Section 9.8 tab	Original
9.8-1	Amendment 6
9.8-1a	Amendment 3
9.8-2	Original
9.8-3	Amendment 2
9.8-4	Original
9.8-5	Amendment 7
9.8-6	Amendment 7
9.8-7	Amendment 7
9.8-8	Amendment 3
9.8-9	Original
9.8-10	Amendment 7
9.8-11	Amendment 6
9.8-12	Original
9.8-13	Amendment 7
9.8-14	Amendment 2
9.8-15	Amendment 2
Fig. 9.8-1	Amendment 7
9.8-2	Original
9.8-3	Original
9.8-4	Original
9.8-5	Amendment 7
9.8-6	Original
9.8-7	Original
9.8-8	Original
Section tab 9.9	Original
9.9-1	Amendment 8
9.9-2	Amendment 8
9.9-3	Amendment 8
9.9-4	Amendment 8
9.9-5	Amendment 8
9.9-6	Amendment 8
9.9-7	Amendment 8
9.9-8	Amendment 8

Page or Table/Figure No.	Issue
Section 9.10 tab	Original
9.10-1	Amendment 6
9.10-2	Amendment 6
9.10-3	Amendment 6
9.10-4	Amendment 6
9.10-5	Amendment 7
Fig. 9.10-1 Sh. 1	Amendment 6
9.10-1 Sh. 2	Amendment 7
9.10-1 Sh. 3	Amendment 7
9.10-1 Sh. 4	Amendment 6
9.10-1 Sh. 5	Amendment 6
9.10-1 Sh. 6	Amendment 7
9.10-1 Sh. 7	Amendment 6
9.10-1 Sh. 8	Amendment 6
9.10-1 Sh. 9	Amendment 7
9.10-1 Sh. 10	Amendment 6
9.10-1 Sh. 11	Amendment 6
9.10-1 Sh. 12	Amendment 6
9.10-1 Sh. 13	Amendment 8
9.10-2 Sh. 1	Amendment 7
9.10-2 Sh. 2	Amendment 6
Section 9.11 tab	Original
9.11-1	Amendment 4
Volume V title page	Original
Section 10 tab	Original
10-i	Amendment 6
10-ii	Amendment 6
10-iii	Amendment 6
Section 10.1 tab	Original
10.1-1	Amendment 6
Section 10.2 tab	Original
10.2-1	Amendment 7
10.2-2	Amendment 6
10.2-3	Amendment 6
10.2-4	Amendment 6
10.2-5	Amendment 6
10.2-6	Amendment 6
10.2-7	Amendment 6
10.2-8	Amendment 6
10.2-9	Amendment 6
10.2-10	Amendment 6
10.2-11	Amendment 6
10.2-12	Amendment 6

Page or Table/Figure No.	Issue
Fig. 10.2-1 Sh. 1	Amendment 7
10.2-1 Sh. 2	Amendment 7
10.2-1 Sh. 3	Amendment 6
10.2-2 Sh. 1	Amendment 7
10.2-2 Sh. 2	Amendment 7
10.2-2 Sh. 2A	Amendment 8
10.2-2 Sh. 3	Amendment 7
10.2-3 Sh. 1	Amendment 7
10.2-3 Sh. 2	Amendment 7
10.2-3 Sh. 3	Amendment 7
10.2-3 Sh. 4	Amendment 7
10.2-3 Sh. 5	Amendment 7
10.2-4	Amendment 6
10.2-5 Sh. 1	Amendment 6
10.2-5 Sh. 2	Amendment 7
10.2-5 Sh. 3	Amendment 7
10.2-5 Sh. 4	Amendment 7
10.2-5 Sh. 5	Amendment 7
10.2-6 Sh. 1	Amendment 7
10.2-6 Sh. 2	Amendment 7
10.2-7 Sh. 1	Amendment 7
10.2-7 Sh. 2	Amendment 7
10.2-7 Sh. 3	Amendment 7
10.2-8	Amendment 7
10.2-9	Original
10.2-10	Amendment 6
Section 10.3 tab	Original
10.3-1	Amendment 2
10.3-2	Amendment 6
10.3-3	Amendment 6
10.3-4	Amendment 7
Section 10.4 tab	Original
10.4-1	Amendment 7
10.4-2	Amendment 6
10.4-3	Amendment 6
10.4-4	Original
Section 10.5 tab	Original
10.5-1	Amendment 4
Section 11 tab	Original
11-i	Amendment 6
11-ii	Amendment 6
11-iii	Amendment 6
11-iv	Amendment 7
11-v	Amendment 6
11-vi	Amendment 6
11-vii	Amendment 6
Section 11.1 tab	Original
11.1-1	Amendment 6
11.1-2	Amendment 6

Page or Table/Figure No.	Issue
11.1-3	Amendment 6
11.1-4	Amendment 6
11.1-5	Amendment 6
11.1-6	Amendment 6
11.1-7	Amendment 7
Tab. 11.1-1 Sh. 1	Amendment 6
11.1-1 Sh. 2	Amendment 6
11.1-1 Sh. 3	Amendment 6
Section 11.2 tab	Original
11.2-1	Amendment 6
11.2-2	Amendment 6
11.2-3	Amendment 6
11.2-4	Amendment 6
11.2-5	Amendment 7
11.2-6	Amendment 7
11.2-7	Amendment 7
11.2-8	Amendment 6
11.2-9	Amendment 8
Tab. 11.2-1 Sh. 1	Amendment 7
11.2-1 Sh. 2	Amendment 6
11.2-1 Sh. 3	Amendment 6
11.2-1 Sh. 4	Amendment 6
11.2-1 Sh. 5	Amendment 6
11.2-1 Sh. 6	Amendment 6
11.2-1 Sh. 7	Amendment 7
11.2-1 Sh. 8	Amendment 6
11.2-1 Sh. 9	Amendment 6
11.2-2	Amendment 6
Fig. 11.2-1 Sh. 1	Amendment 7
11.2-1 Sh. 2	Amendment 7
11.2-1 Sh. 3	Amendment 6
11.2-1 Sh. 4	Amendment 7
11.2-1 Sh. 5	Amendment 7
11.2-2 Sh. 1	Amendment 7
11.2-2 Sh. 2	Amendment 7
11.2-2 Sh. 3	Amendment 6
11.2-2 Sh. 4	Amendment 6
11.2-2 Sh. 5	Amendment 7
Section 11.3 tab	Original
11.3-1	Amendment 7
11.3-2	Amendment 7
11.3-3	Amendment 6
11.3-4	Amendment 7
11.3-5	Amendment 7
11.3-6	Amendment 6
11.3-7	Amendment 6
11.3-8	Amendment 6
11.3-9	Amendment 7

Page or Table/Figure No.	Issue
Tab. 11.3-1	Amendment 6
11.3-2	Amendment 6
11.3-3 Sh. 1	Amendment 6
11.3-3 Sh. 2	Amendment 6
Fig. 11.3-1	Amendment 7
Section 11.4 tab	Original
11.4-1	Amendment 7
11.4-2	Amendment 7
11.4-3	Amendment 7
11.4-4	Amendment 7
Tab. 11.4-1 Sh. 1	Amendment 6
11.4-1 Sh. 2	Amendment 6
11.4-1 Sh. 3	Amendment 6
Fig. 11.4-1	Amendment 7
Section 11.5 tab	Original
11.5-1	Amendment 7
11.5-2	Amendment 7
11.5-3	Amendment 6
11.5-4	Amendment 6
11.5-5	Amendment 6
11.5-6	Amendment 6
11.5-7	Amendment 7
11.5-8	Amendment 7
11.5-9	Amendment 6
Tab. 11.5-1	Amendment 6
11.5-2 Sh. 1	Amendment 6
11.5-2 Sh. 2	Amendment 6
11.5-2 Sh. 3	Amendment 6
11.5-3 Sh. 1	Amendment 6
11.5-3 Sh. 2	Amendment 6
11.5-3 Sh. 3	Amendment 7
11.5-4 Sh. 1	Amendment 7
11.5-4 Sh. 2	Amendment 7
11.5-4 Sh. 3	Amendment 7
11.5-4 Sh. 4	Amendment 8
Fig. 11.5-1	Amendment 7
Section 11.6 tab	Original
11.6-1	Amendment 7
11.6-2	Amendment 7
11.6-3	Amendment 6
Section 11.7 tab	Amendment 6
11.7-1	Amendment 6
Section 11.8 tab	Amendment 6
11.8-1	Amendment 6
11.8-2	Amendment 6
11.8-3	Amendment 6
11.8-4	Amendment 6
11.8-5	Amendment 6
11.8-6	Amendment 7
11.8-7	Amendment 7

Page or Table/Figure No.	Issue
Tab. 11.8-1	Amendment 6
11.8-2	Amendment 6
11.8-3 Sh. 1	Amendment 7
11.8-3 Sh. 2	Amendment 6
11.8-3 Sh. 3	Amendment 6
11.8-4 Sh. 1	Amendment 7
11.8-4 Sh. 2	Amendment 7
Section 11.9 tab	Amendment 6
11.9-1	Amendment 7
Section 11.10 tab	Amendment 6
11.10-1	Amendment 7
11.10-2	Amendment 7
11.10-3	Amendment 7
11.10-4	Amendment 7
11.10-5	Amendment 7
Section 11.11 tab	Amendment 6
11.11-1	Amendment 6
Section 12 tab	Original
12-i	Amendment 8
12-ii	Amendment 7
12-iii	Amendment 7
12-iv	Amendment 6
Section 12.1 tab	Original
12.1-1	Amendment 8
12.1-2	Amendment 8
12.1-3	Amendment 8
12.1-4	Amendment 8
12.1-5	Amendment 8
Fig. 12.1-1	Amendment 8
12.1-2	Amendment 8
Section 12.2 tab	Original
12.2-1	Amendment 8
12.2-2	Amendment 8
12.2-3	Amendment 8
12.2-4	Amendment 6
12.2-5	Amendment 8
12.2-6	Amendment 8
12.2-7	Amendment 7
Section 12.3 tab	Original
12.3-1	Amendment 8
Section 12.4 tab	Original
12.4-1	Amendment 7
Section 12.5 tab	Original
12.5-1	Amendment 8
12.5-2	Amendment 6
12.5-3	Amendment 6
12.5-4	Amendment 6
12.5-5	Amendment 6

Page or Table/Figure No.	Issue
12.5-6	Amendment 6
12.5-7	Amendment 7
Section 12.6 tab	Original
12.6-1	Amendment 6
Section 12.7 tab	Original
12.7-1	Amendment 7
12.7-2	Amendment 6
Section 13 tab	Original
13-i	Original
13-ii	Original
13-iii	Original
Section 13.1 tab	Original
13.1-1	Original
13.1-2	Original
13.1-3	Original
13.1-4	Original
13.1-5	Original
13.1-6	Original
13.1-7	Original
13.1-8	Original
13.1-9	Original
Section 13.2 tab	Original
13.2-1	Original
13.2-2	Original
13.2-3	Original
Fig. 13.2-1	Original
13.2-2	Original
Section 13.3 tab	Original
13.3-1	Original
13.3-2	Original
13.3-3	Original
13.3-4	Original
13.3-5	Original
13.3-6	Original
13.3-7	Original
13.3-8	Original
13.3-9	Original
13.3-10	Original
13.3-11	Original
13.3-12	Original
13.3-13	Original
13.3-14	Original
13.3-15	Original
13.3-16	Original
13.3-17	Original
13.3-18	Original
13.3-19	Original
13.3-20	Original
13.3-21	Original

Page or Table/Figure No.	Issue
13.3-22	Original
13.3-23	Original
13.3-24	Original
13.3-25	Original
13.3-26	Original
13.3-27	Original
13.3-28	Original
13.3-29	Original
13.3-30	Original
13.3-31	Original
13.3-32	Original
13.3-33	Original
13.3-34	Original
13.3-35	Original
13.3-36	Original
13.3-37	Original
13.3-38	Original
Section 13.4 tab	Original
13.4-1	Original
13.4-2	Original
Section 13.5 tab	Original
13.5-1	Original
13.5-2	Original
13.5-3	Original
13.5-4	Original
Section 13.6 tab	Original
13.6-1	Original
Section 13.7 tab	Original
13.7-1	Original
Volume VI title page	Original
Section 14 tab	Original
14-i	Original
14-ii	Original
14-iii	Original
14-iv	Amendment 6
14-v	Amendment 7
14-vi	Amendment 6
14-vii	Original
14-viii	Original
14-ix	Original
14-x	Original
14-xi	Original
14-xii	Original
14-xiii	Original
14-xiv	Original
14-xv	Original
14-xvi	Original
14-xvii	Amendment 6
14-xviii	Amendment 6

Page or Table/Figure No.	Issue
Section 14.1 tab	Original
14.1-1	Original
14.1-2	Original
14.1-3	Original
14.1-4	Original
14.1-5	Amendment 5
14.1-6	Original
14.1-7	Original
14.1-8	Original
14.1-9	Original
14.1-10	Original
14.1-11	Original
14.1-12	Amendment 5
14.1-13	Original
14.1-14	Original
14.1-15	Original
14.1-16	Original
14.1-17	Original
14.1-18	Original
14.1-19	Original
14.1-20	Original
14.1-21	Original
14.1-22	Original
14.1-23	Original
14.1-24	Original
14.1-25	Amendment 6
14.1-26	Amendment 1
14.1-27	Original
14.1-28	Original
14.1-29	Amendment 6
14.1-30	Amendment 6
14.1-31	Amendment 6
14.1-32	Original
14.1-33	Original
14.1-34	Amendment 6
14.1-35	Amendment 4
14.1-36	Original
14.1-37	Amendment 4
14.1-38	Amendment 4
14.1-39	Amendment 4
14.1-40	Amendment 4
14.1-41	Original
14.1-42	Original
14.1-43	Original
14.1-44	Amendment 1
14.1-45	Amendment 4
14.1-45a	Amendment 4
14.1-46	Amendment 4

Page or Table/Figure No.	Issue
14.1-47	Amendment 2
14.1-48	Amendment 6
14.1-49	Amendment 6
14.1-50	Amendment 2
14.1-51	Amendment 2
Fig. 14.1-1	Original
14.1-2	Original
14.1-3	Original
14.1-4	Original
14.1-5	Original
14.1-6	Original
14.1-7	Original
14.1-8	Original
14.1-9	Original
14.1-10	Original
14.1-11	Original
14.1-12	Original
14.1-13	Original
14.1-14	Original
14.1-15	Original
14.1-16	Original
14.1-17	Original
14.1-18	Original
14.1-19	Original
14.1-20	Original
14.1-20a	Original
14.1-21	Original
14.1-22	Original
14.1-23	Original
14.1-24	Original
14.1-25	Original
14.1-26	Original
14.1-27	Original
14.1-28	Original
14.1-29 Sh. 1	Original
14.1-29 Sh. 2	Original
14.1-29 Sh. 3	Original
14.1-29 Sh. 4	Original
Section 14.2 tab	Original
14.2-1	Amendment 4
14.2-2	Original
14.2-3	Original
14.2-4	Amendment 3
14.2-5	Amendment 6
14.2-6	Amendment 6
14.2-7	Amendment 6
14.2-8	Amendment 6

Page or Table/Figure No.	Issue
14.2-9	Amendment 6
14.2-10	Original
14.2-11	Amendment 6
14.2-12	Amendment 6
14.2-12a	Amendment 4
14.2-12b	Amendment 4
14.2-13	Original
14.2-14	Original
14.2-15	Amendment 6
14.2-16	Amendment 6
14.2-17	Amendment 7
14.2-18	Amendment 7
14.2-18a	Amendment 7
14.2-19	Original
14.2-20	Original
14.2-21	Original
14.2-22	Original
14.2-23	Original
14.2-24	Original
14.2-25	Amendment 6
14.2-26	Amendment 6
14.2-27	Amendment 6
14.2-28	Amendment 2
14.2-29	Original
14.2-30	Original
14.2-31	Original
14.2-32	Original
14.2-33	Original
14.2-34	Amendment 5
14.2-35	Original
14.2-36	Amendment 1
14.2-37	Original
14.2-38	Original
14.2-39	Amendment 6
14.2-40	Original
14.2-41	Original
14.2-42	Original
14.2-43	Amendment 5
14.2-43a	Amendment 5
14.2-44	Original
14.2-45	Original
14.2-45a	Amendment 5
14.2-46	Original
14.2-47	Amendment 6
Fig. 14.2-1 Sh. 1	Original
14.2-1 Sh. 2	Original
14.2-1 Sh. 3	Original
14.2-1 Sh. 4	Original
14.2-2	Original
14.2-3	Original

Page or Table/Figure No.	Issue
Fig. 14.2-4	Original
14.2-5	Original
14.2-6	Original
14.2-7	Original
14.2-8	Original
14.2-9	Original
14.2-10	Original
14.2-11	Original
14.2-12	Original
14.2-13	Original
14.2-14	Original
14.2-15	Original
14.2-16	Original
14.2-17	Original
14.2-18	Original
14.2-19	Original
14.2-20	Original
14.2-21	Original
14.2-22	Original
14.2-23	Original
14.2-24	Original
14.2-25	Original
14.2-26	Original
14.2-27	Original
14.2-28	Original
14.2-29	Original
14.2-30	Original
14.2-31	Original
14.2-32	Original
14.2-33	Original
14.2-34	Original
14.2-35	Original
14.2-36	Original
14.2-37	Original
14.2-38	Original
14.2-39	Original
14.2-40	Original
14.2-41	Original
14.2-42	Original
14.2-43	Original
14.2-44	Original
14.2-45	Original
14.2-46	Original
14.2-47	Original
14.2-48	Original
14.2-49	Original
14.2-50	Original
14.2-51	Original

Page or Table/Figure No.	Issue
Fig. 14.2-52	Original
14.2-53	Original
14.2-54	Original
14.2-55	Original
14.2-56	Original
14.2-57	Original
14.2-58	Original
14.2-59	Original
14.2-60	Original
14.2-61	Original
14.2-62	Original
14.2-63	Original
14.2-64	Original
14.2-65	Original
14.2-66	Original
14.2-67	Original
14.2-68	Original
14.2-69	Original
14.2-70	Original
14.2-71	Original
14.2-72	Original
14.2-73	Original
14.2-74	Original
14.2-75	Original
14.2-76	Original
14.2-77	Original
14.2-78	Original
14.2-79	Original
14.2-80	Original
Section 14.3 tab	Original
14.3-1	Amendment 5
14.3-2	Amendment 6
14.3-3	Amendment 7
14.3-4	Original
14.3-5	Amendment 7
14.3-6	Amendment 7
14.3-6a	Amendment 7
14.3-7	Amendment 7
14.3-8	Amendment 6
14.3-9	Original
14.3-10	Original
14.3-11	Amendment 5
14.3-11a	Amendment 5
14.3-12	Amendment 4
14.3-13	Amendment 6
14.3-14	Amendment 6
14.3-15	Amendment 6
14.3-16	Amendment 6

Page or Table/Figure No.	Issue
Fig. 14.3-1	Original
14.3-2	Original
14.3-3	Original
14.3-4	Original
14.3-5	Original
14.3-6	Original
14.3-7	Original
14.3-8	Amendment 4
14.3-9	Amendment 4
14.3-10	Amendment 4
14.3-11	Amendment 4
14.3-12	Amendment 4
14.3-13	Amendment 4
14.3-14	Amendment 4
14.3-15	Amendment 4
14.3-16	Amendment 4
14.3-17	Amendment 4
Section 14.4 tab	Original
14.4-1	Original
14.4-2	Original
14.4-3	Amendment 6
14.4-4	Amendment 6
14.4-5	Original
14.4-6	Original
14.4-7	Amendment 6
14.4-8	Amendment 6
14.4-9	Amendment 2
14.4-10	Original
14.4-11	Original
14.4-12	Amendment 6
14.4-13	Amendment 6
14.4-14	Original
14.4-15	Original
14.4-16	Original
14.4-17	Original
14.4-18	Original
14.4-19	Original
14.4-20	Amendment 6
14.4-21	Amendment 6
14.4-22	Original
14.4-23	Original
14.4-24	Amendment 6
14.4-25	Amendment 6
14.4-26	Original
14.4-27	Amendment 6
14.4-28	Original
Fig. 14.4-1	Original
14.4-2	Original

Page or Table/Figure No.	Issue
Fig. 14.4-3	Original
14.4-4	Original
14.4-5	Original
14.4-6	Original
14.4-7	Original
14.4-8	Original
14.4-9	Original
14.4-10	Amendment 6
14.4-11 Sh. 1	Original
14.4-11 Sh. 2	Original
14.4-11 Sh. 3	Original
14.4-12	Original
14.4-13	Original
14.4-14	Original
14.4-15	Original
14.4-16	Original
14.4-17	Original
14.4-18	Original
14.4-19	Original
14.4-20	Original
14.4-21	Original
14.4-22	Original
14.4-23	Original
Section 14.5 tab	Original
14.5-1	Original
14.5-2	Amendment 7
14.5-3	Amendment 7

Page or Table/Figure No.	Issue
Page or Table/Figure No.	Issue
Volume VII title page	Orig.
Appendix 1A tab	Origina
Blank Page	Origina
Appendix 1B tab	Origin
Title Page	Original
1B-i	Amendment 3
1B-1 - 1B-16	Original
1B-17	Amendment 6
Fig. 1B-1	Original
Fig. 1B-2	Original
Appendix 2A tab	Original
Blank Page	Original
Appendix 2B tab	Original
Title Page	Original
2B-i	Amendment 6
2B-ii	Amendment 6
2B-1 - 2B-8	Amendment 6
2B-9	Amendment 8
Fig. 2B-1, 2B-2	Original
Attachment 1	Original
Final Report	Original
iii - vi	Original
1-45	Original
Attachment 2	Original
Final Report	Original
iii - vi	Original
1 - 40	Original
Attachment 3	Original
Final Report	Original
iii	Original
1-13	Original
Fig. 1 - 2	Original
Attachment 4	Original
1-27	Original
Attachment 5	Original
Title page	Original
1-71	Original
Attachment 6	Original
Title page	Original
1-53	Original
Appendix 2C tab	Original
Title page	Original
2C-i - 2C-iii	Original
2C-1 - 2C-12	Original
Fig. 2C-1 - 2C-11	Original
Geophysical Report	
Title page	Original
1 - 6	Original

Page or Table/Figure No.	Issue
Page or Table/Figure No.	Issue
Additional Seismic	
Exploration pp. 1,2	Original
Plate 1 - 3	Original
Geological Log of	
Drill Holes	
91 Sheets	Original
Volume VIII	
Title page	Original
Appendix 2D tab	Original
Title Page	Original
Perry Byerly Report	Original
Index	Original
1 - 14	Original
Addendum Page	Original
Title Page	Original
Index	Original
1 - 10	Original
Appendix 2E tab	Original
Title Page	Original
2E-i	Amendment 5
2E-ii	Original
2E-1 - 2E-7	Original
2E-8 - 2E-8A	Amendment 5
Fig. C-119E - C-122E	Original
Soil Mechanics and	
Foundation Engineers	
pp. 1 - 9	Original
Table 1, 2	Original
Table A	Original
Fig. 1 - 9	Original
J.V. Lowney Letter	
2 pages	Original
Summary of Laboratory Tests	
Table A	Original
Table 1, 2	Original
Fig. 1 - 9	Original
Sample Test Charts,	
12 Pages	Original
Appendix 3A tab	Original
Blank Page	Original
Appendix 3B tab	Original
Title Page	Original
3B-i - 3B-iii	Original
3B-1 - 3B-13	Original
Fig. 3B-1 - 3B-7	Original
Appendix 4A tab	Original
Blank Page	Original

Page or Table/Figure No.	Issue
Page or Table/Figure No.	Issue
Appendix 5A tab	Original
Blank Page	Original
Appendix 5B tab	Original
Title Page	Original
5B-1	Original
5B-1	Original
5B-2	Amendment 3
5B-3	Original
5B-4	Amendment 3
5B-5	Original
5B-6	Amendment 5
5B-6a	Amendment 5
5B-7	Amendment 2
5B-8	Amendment 5
5B-9 - 5B-12	Original
5B-13	Amendment 3
5B-14	Original
1. Basic Ground Spectra	Original
4 Sheets	Original
2. Reactor Building Containment Structure Spectra	Original
12 Sheets	Original
3. Reactor Building Interior Structure Spectra	Original
10 Sheets	Original
4. Auxiliary Building Spectra	Original
10 Sheets	Original
5. Turbine Building Spectra	Original
1 Sheet	Original
6. Auxiliary Building Spectra (For use in designs done after April 22, 1985)	Amendment 3
24 Sheets	Amendment 3
7. Spent Fuel Building Spectra (For use in designs done after April 22, 1985)	Amendment 3
12 Sheets	Amendment 3
8. Nuclear Services Electrical Building Spectra	Amendment 3
16 Sheets	Amendment 3

Page or Table/Figure No.	Issue
Page or Table/Figure No.	Issue
Appendix 5C tab	Original
Title Page	Original
5C-i	Original
5C-1 - 5C-8	Original
Appendix 6A tab	Original
Blank Page	
Appendix 7A tab	Original
Blank Page	
Appendix 8A tab	Original
Blank Page	
Appendix 9A tab	Original
Blank Page	
Appendix 10A tab	Original
Blank Page	
Appendix 11A tab	Original
Blank Page	
Appendix 11B tab	Original
Title Page	Original
11B-i - 11B-iii	Original
11B-1 - 11B-8	Original
11B-9	Amendment 7
11B-10	Amendment 7
11B-11	Original
11B-12	Original
11B-13	Amendment 5
11B-14	Original
11B-14a	Amendment 5
11B-15	Amendment 7
11B-16	Amendment 7
Fig. 11B.2-1, 11B.2-2	Amendment 1
Fig. 11B.4-1, 11B 4-2	Amendment 1
Fig. 11B.5-1 - 11B.5-4	Amendment 1
Appendix 12A tab	Original
Blank Page	Original
Appendix 12B tab	Original
12B-1	Amendment 6
Appendix 13A tab	Original
Blank Page	Original
Appendix 14A tab	Original
Blank Page	Original
Appendix 14B tab	Original
Title Page	Original
14B-i - 14B-iii	Original
14B-1 - 14B-18	Original
Fig. 14B-1 - 14B-4	Amendment 1
Appendix 14C tab	Original
Title Page	Original

Page or Table/Figure No.	Issue

Page or Table/Figure No. Issue

Page or Table/Figure No.	Issue
14C-i- 14C-ii	Amendment 6
14C-iii	Original
14C-iv	Amendment 6
14C-1 - 14C-4	Amendment 6
14C-5 - 14C-8	Original
14C-9 - 14C-11	Amendment 6
14C-12	Original
14C-13 - 14C-16	Amendment 6
14C-17 - 14C-18	Original
14C-19 - 14C-20	Amendment 6
Fig. 14C.1-1	Amendment 6
Fig. 14C.3-1 - 14C.3-4	Amendment 1
Fig. 14C.5-1 - 14C.5-3	Amendment 1
Fig. 14C.6-1	Amendment 6
Fig. 14C.6-2	Amendment 6
Fig. 14C.7-1, 14C.7-2	Amendment 1
Appendix 14D tab	Original
Title Page	Original
14-i - 14D-iii	Original
14D-1 - 14D-36	Original
Appendix 14E tab	Original
Title Page	Original
14E-i, 14E-ii	Original
Part I	Original
14E-1 - 14E-5	Original
Fig. 14E-1, Sh. 1 - 7	Original
14E-13 - 14E-39	Original
Part II	Original
Fig. 14E-2	Original
Fig. 14E-3	Original
14E-42	Original
Fig. 14E-4 Sh. 1 & 2	Original
14E-45 - 14E-49	Original
Part III	Original
Fig. 14E-6	Original
Fig. 14E-7	Original

PART 3

USAR Pages Affected By USAR Amendme t 8

Section		Page
	9. AUXILIARY AND EMERGENCY SYSTEMS	

NOTE:
THIS FIGURE HAS BEEN REVISED BASED ON SMUD DWG. C-102, REV.19. MORE RECENT REVISIONS TO THIS DRAWING MAY BE AVAILABLE AND SHOULD BE UTILIZED FOR ENGINEERING CHANGES.

SI
APERTURE
CARD

Also Available On
Aperture Card

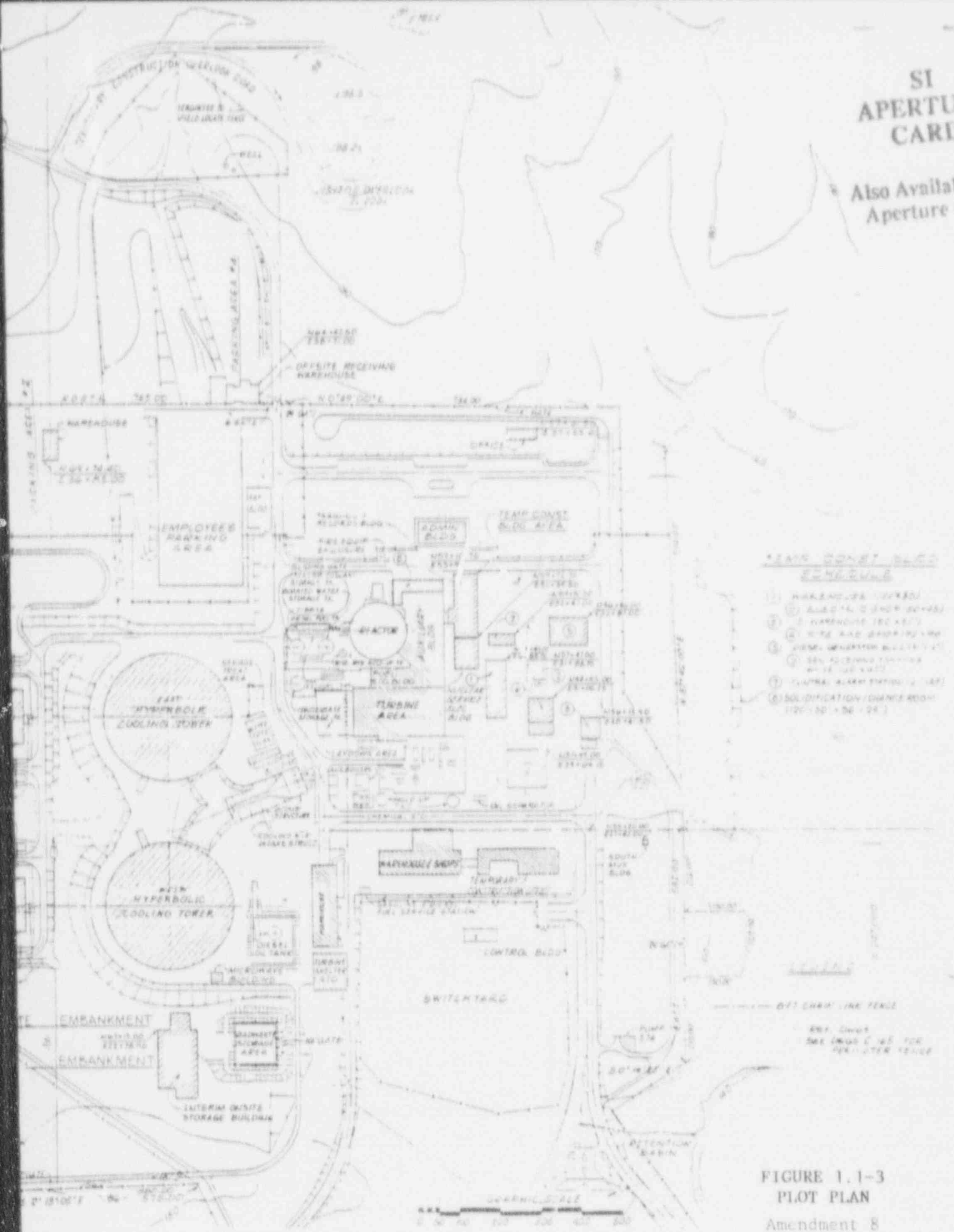
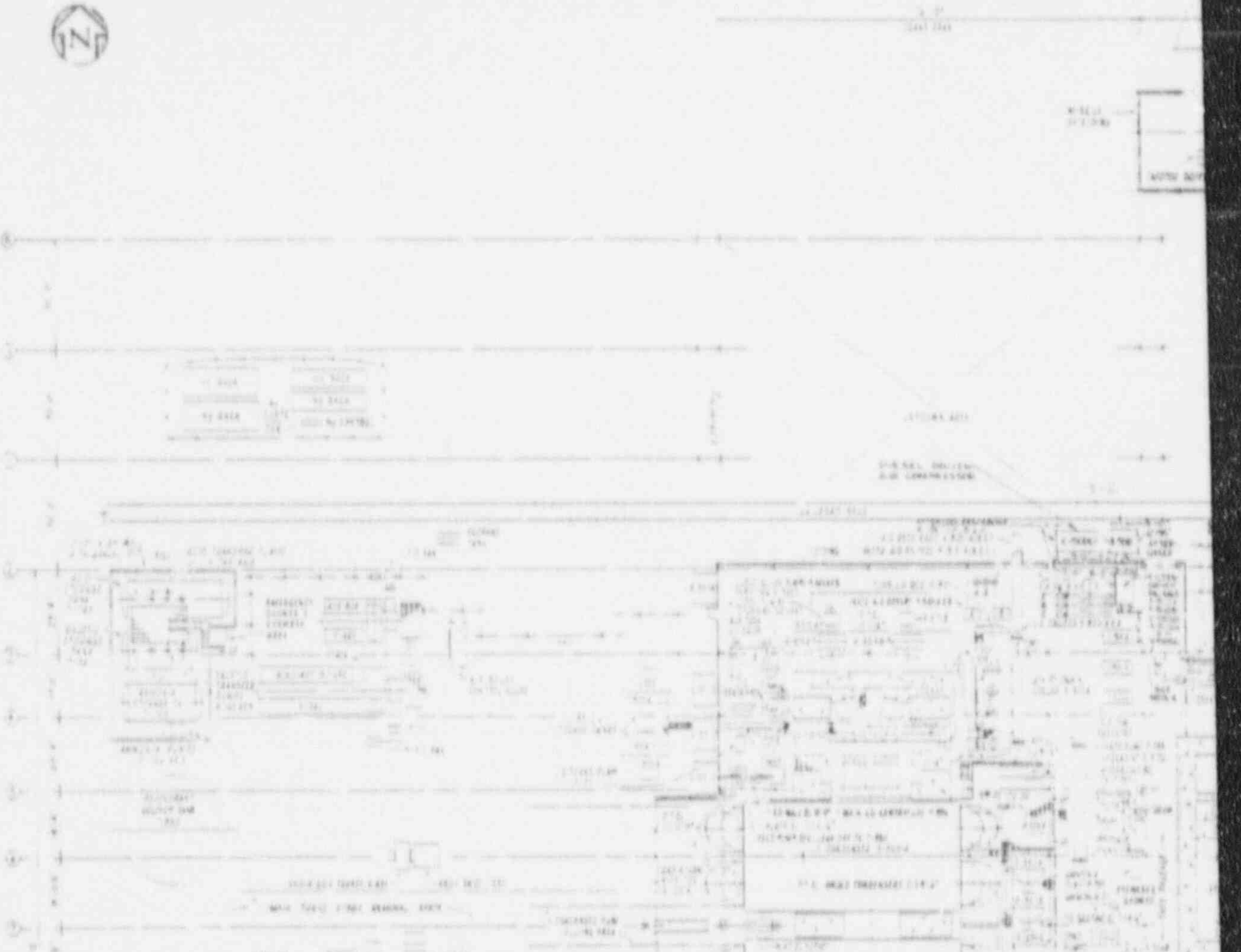


FIGURE 1.1-3
PLOT PLAN
Amendment 8

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

9107030248-01



NOTE:

THIS FIGURE HAS BEEN REVISED BASED ON SMUD DWG.M-302,REV.20.MORE RECENT REVISIONS TO THIS DRAWING MAY BE AVAILABLE AND SHOULD BE UTILIZED FOR ENGINEERING CHANGES.

SI
APERTURE
CARD

Also Available On
Aperture Card

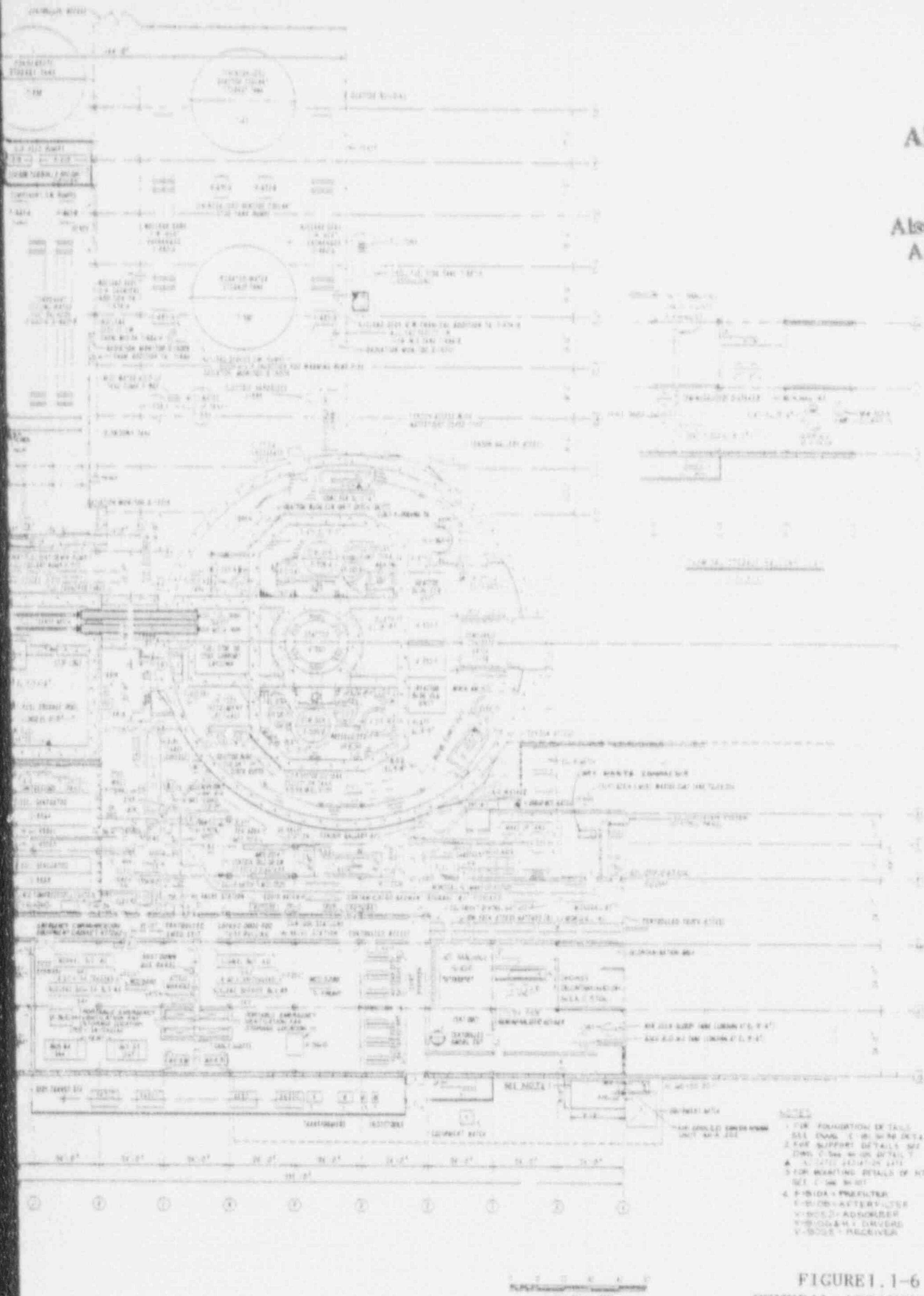


FIGURE 1.1-6
GENERAL ARRANGEMENT
PLAN AT GRADE
Amendment 8

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

9107030248-02

1.2 DESIGN HIGHLIGHTS

1.2.1 SITE CHARACTERISTICS

The 2,480-acre site is characterized by a 2,100-foot minimum exclusion radius; isolation from population centers; sound foundation for structures; abundant supply of makeup water for cooling towers; ample supply of emergency power; and favorable conditions of meteorology, seismology, and hydrology.

1.2.2 POWER LEVEL

The design and license power level for the reactor core is 2,772 MWt and all physics and core thermal hydraulics information in this report is based on that power level. An additional 16 MWt is available to the cycle from the contribution of the reactor coolant pumps, resulting in a gross electrical output of 963 MWe.

1.2.3 PEAK SPECIFIC POWER LEVEL

The peak specific power level in the fuel for operation at 2,772 MWt results in a maximum thermal output of 19.03 kW per foot of fuel rod. This value is comparable with other reactors of this size and therefore does not represent an extrapolation of technology.

1.2.4 REACTOR BUILDING

The Reactor Building is a leaktight structure capable of containing the design base accident (DBA) defined in Chapter 14.

The prestressed, post-tensioned concrete Reactor Building is similar to the containment buildings for the Turkey Point Plant (Docket No. 50-250 and 251), the Oconee Station (Docket No. 50-269, 270 and 287) and the Palisades Station (Docket No. 50-255). The Rancho Seco safety features systems are similar to those for these plants and present neither uncommon solutions to engineering problems nor significant extrapolations in technology.

1.2.5 SAFETY FEATURES

The safety features provided for Rancho Seco Nuclear Generating Station have sufficient redundancy of component and power sources that under conditions of the worst postulated loss of coolant accident, the system can maintain integrity of the Reactor Building and keep exposure of the public below the limits of 10 CFR 100.

The safety features provided for this plant are as follows:

A. High pressure injection prevents uncovering of the core for small coolant piping leaks at high pressure and delays uncovering of the core for intermediate sized leaks.

B. The core flooding system automatically floods the core when reactor coolant system pressure reaches approximately 600 psig.

C. Low pressure injection cools the core when reactor coolant pressure reaches 200 psig after a loss-of-coolant accident.

D. The Reactor Building spray system sprays borated water (containing a solution of sodium hydroxide) into the Reactor Building atmosphere to remove iodine and provide a redundant system for Reactor Building cooling.

E. The Reactor Building emergency coolers act as a heat sink to cool the building atmosphere under the conditions of a loss-of-coolant accident. Two of the four emergency coolers contain activated charcoal filters to provide additional iodine removal capability.

F. The Reactor Building isolation system provides automatic isolation of all Reactor Building penetrations not required to limit the consequences of an accident.

1.2.6 ELECTRICAL SYSTEMS AND EMERGENCY POWER

Rancho Seco has the following sources of electric power:

A. Four 220/230-kV transmission lines; one from Hedge, one from Elk Grove, and two from Pocket.

B. Two 220/230-kV transmission lines from Bellota switchyard.

C. Four quick-starting auxiliary diesel generator units connected to the nuclear service buses. The nameplate rating of the GEA and GEB auxiliary generators is 2,750 kW each at 0.8 power factor. The nameplate rating of the GEA-2 and GEB-2 auxiliary generators is 3,500 kW each at 0.8 power factor with a maximum "qualified load" of 3,300 kW.

D. Four batteries for the safety features actuation and reactor protection systems and four batteries for additional safety related loads.

E. Five batteries for the plant auxiliaries.

Plant electrical systems consist of multiple redundant buses and bus ties supplying all power, instrumentation, and controls. The safety features systems are supplied from separate nuclear service power buses, each of which can be supplied from the 220/230-kV switchyard (startup transformer) or the auxiliary diesel generators. The station electrical systems are designed to provide reliable power for plant and personnel safety under all modes of operation and shutdown.

1.2.7 SEISMIC DESIGN QUALITY ASSURANCE

The following describes the design control measures employed during plant construction to assure that adequate seismic input (including the necessary feedback from structural and system dynamic analyses) was specified to vendors of purchased Seismic Category 1 components and equipment.

The Rancho Seco Reactor Building can withstand the following tornado characteristics:

A. External wind pressure:

1. Forward velocity - 60 mph

2. Tangential velocity - 300 mph

B. Transient differential pressure peak - 3 psi

C. Tornado generated missiles:

1. 4 in. x 12 in. x 12 ft. wooden plank (148 lbs) at 300 mph

2. 3 in. diameter, 10 ft. long, schedule 40 pipe (72 lbs) at 100 mph

3. Passenger automobile (4000 lbs) to 50 mph at 25 ft above grade

The Auxiliary Building will withstand a 200 mph wind, a differential pressure of 1.0 psi, and the above tornado-generated missiles. The fuel storage pool will withstand a 200 mph wind, a differential pressure of 0.5 psi, and the above tornado-generated missiles. Section 5.5.8 discusses the capability of the Diesel Generator Building to withstand the effects of natural phenomena.

1.5.3 CRITERION 3 - FIRE PROTECTION

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the Reactor Building and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire-fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Discussion:

Plant buildings, systems, and components are designed to minimize the probability of fires and explosions and to minimize the potential effects of such events on safety related structures, systems and components. Non-combustible and heat resistant materials are used wherever practical throughout the plant, particularly in locations such as the containment and the control room. Safety features redundant trains are physically separated so that there is a reduced probability of losing more than one train of a safety features system because of an external event such as a fire.

The Fire Protection Plan details the fire hazards within each plant area and the fire protection features provided within the area, and evaluates the potential consequences of fire damage on post-fire safe shutdown capability.

The plant fire protection features include:

- automatic water fire suppression systems,
- automatic carbon dioxide fire suppression systems,
- fire detection and alarm systems,
- manual fire-fighting equipment,
- fire barriers,
- separation of equipment and cables for post-fire safe shutdown,
- emergency lighting for post-fire safe shutdown, and
- communications support for post-fire safe shutdown.

A detailed description of the plant fire suppression and detection systems is included in the Fire Protection Plan. Emergency lighting and communications are described in USAR Sections 9.9.2 and 9.9.3, respectively.

Rupture or inadvertent actuation of fire suppression systems will not significantly impair the operation of any safety related equipment. Inadvertent operation of the carbon dioxide systems will not prevent the operation of any safety related equipment. Inadvertent operation of a section of a wet-pipe sprinkler system is not expected to result in safety equipment impairment, primarily because these systems are excluded from rooms containing electrical switchgear or safety system pumps or motors. In those cases where water suppression is provided in areas containing safety equipment, that equipment is adequately protected to function properly while exposed to water, or inadvertent operation is precluded by use of a preaction system. Areas protected by water suppression systems are reviewed to ensure adequate drainage is provided for fire water.

1.6 COMPARISON WITH SAFETY GUIDES

Rancho Seco was originally designed to meet the Safety Guides which are the equivalent of NRC Regulatory Guides 1.1 through 1.21.

1.6.1 SAFETY GUIDE 1 - NET POSITIVE SUCTION HEAD FOR EMERGENCY CORE COOLING AND CONTAINMENT HEAT REMOVAL SYSTEM PUMPS

A prime consideration in the design of the emergency core cooling system and Reactor Building heat removal system is reliable operation under the various possible post-accident conditions. Appropriately conservative assumptions were made so that proper performance of these systems would be independent of a calculated increase in Reactor Building pressure caused by postulated loss-of-coolant accidents.

Calculations of available NPSH assumed: (1) no increase in Reactor Building pressure from that present prior to postulated loss-of-coolant accident; (2) a pumped fluid temperature corresponding to saturation conditions for the pressure in the Reactor Building, and (3) no build-up of water above the level of the Reactor Building emergency sump following the injection of water from the borated water storage tank. These calculations indicate that there is sufficient margin available in the NPSH for the decay heat pumps and the Reactor Building spray pumps for proper performance during the recirculation mode.

1.6.2 SAFETY GUIDE 2 - THERMAL SHOCK TO REACTOR PRESSURE VESSELS

The B&W Topical Report BAW-10018, "Analysis of the Structural Integrity of a Reactor Vessel Subjected to Thermal Shock," has been incorporated into this license application. This report demonstrates that the reactor vessel will not lose its integrity due to crack propagation as a result of thermal shock caused by actuation of the ECCS following a LOCA. In addition, the design of the reactor vessel does not preclude the use of annealing if required to assure recovery of the fracture toughness properties of the vessel material.

1.6.3 SAFETY GUIDE 3 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT FOR BOILING WATER REACTORS

Safety Guide 3 is not applicable as Rancho Seco does not utilize a boiling water reactor.

1.6.4 SAFETY GUIDE 4 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT FOR PRESSURIZED WATER REACTORS

The specific assumptions listed in Safety Guide 4 were used in the evaluation of the environmental effects of a maximum hypothetical accident (see Section 14.3.9).

1.6.5 SAFETY GUIDE 5 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL CONSEQUENCES OF A STEAM LINE BREAK ACCIDENT FOR BOILING WATER REACTORS

Safety Guide 5 is not applicable as Rancho Seco does not utilize a boiling water reactor.

1.6.6 SAFETY GUIDE 6 - INDEPENDENCE BETWEEN REDUNDANT STANDBY (ONSITE) POWER SOURCES AND BETWEEN THEIR DISTRIBUTION SYSTEMS

The Rancho Seco electrical system design provides independence between redundant auxiliary diesel standby power sources with their respective load

groups. The following provisions are included in the system design:

A. The electrically powered safety related loads (a-c or d-c) are separated into two redundant load groups. Either load group can perform the minimum safety functions.

B. Each a-c load group can be connected to either a preferred (offsite) power source or to its standby (onsite) power source (two diesel generators per load group). There is no automatic connection between a standby power source and the load group of the redundant channel.

C. Each d-c load group is energized by a battery and a battery charger. This battery/charger combination has no automatic connection to any other redundant d-c load group. A standby battery charger is available for each pair of redundant d-c load groups; it is connected manually (see Figure 8.2-3).

D. 1. The standby power source of one nuclear service load group cannot be automatically connected in parallel with the standby source of the redundant load group under any condition.

   2. There are no provisions for automatically connecting one safety related load group to another load group.

   3. There are no provisions for automatically transferring loads between redundant load groups.

E. Diesel generators are used as the standby power source for each a-c load group (four total).

1.6.7 SAFETY GUIDE 7 - CONTROL OF COMBUSTIBLE GAS CONCENTRATIONS IN CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT

Compliance with Safety Guide 7 is discussed in Appendix 14C.

1.6.8 SAFETY GUIDE 8 - PERSONNEL SELECTION AND TRAINING

The SMUD organization at the Rancho Seco Nuclear Plant is well qualified in nuclear and steam plant experience. Personnel meet the minimum education and experience per the American National Standard(s) for Selection and Training of Nuclear Power Plant Personnel. Specific training programs for staff personnel are outlined in Section 12.2. Retraining and replacement training and documentation records of the qualifications, background, training and retraining of each member of the plant organization are maintained and followed according to the established program outlined in Section 12.2. All programs include or surpass requirements outlined in 10 CFR 55 or ANSI N18.1-1971, as applicable.

1.6.9 SAFETY GUIDE 9 - SELECTION OF DIESEL GENERATOR SET CAPACITY FOR STANDBY POWER SUPPLIES AND REGULATORY GUIDE 1.9 - SELECTION, DESIGN AND QUALIFICATION OF DIESEL-GENERATOR UNITS AS STANDBY (ONSITE) ELECTRICAL POWER SYSTEMS AT NUCLEAR POWER PLANTS

Diesel generators GEA and GEB were selected and procured in accordance with the intent of Safety Guide 9; diesel generators GEA2 and GEB2 in accordance with Regulatory Guide 1.9.

reduction in the spent fuel coolant inventory under accident conditions. This design includes the following provisions:

A. The spent fuel storage facilities (including the fuel storage building, storage racks, and fuel transfer mechanism) are Seismic Category I.

B. The capability of the spent fuel pool to withstand high winds and high-wind generated missiles is presented in the discussion of the Criteria 4, Section 1.5.4.

C. The Turbine Building gantry crane is electrically interlocked to prevent movement of the trolley over the fuel storage rack area.

D. A ventilation and filtration system is used to limit the potential release of radioactive iodine and other radioactive materials (see Section 9.7.3). The design of the ventilation and filtration system is based on the assumption that the cladding of all the fuel rods in one fuel assembly might be breached.

E. The spent fuel storage facility design is such that the fuel cask or other heavy loads need not be moved directly over either the spent fuel or new fuel storage areas. The fuel pool is designed to withstand, without significant leakage, the impact of the fuel cask dropped from the maximum height to which it can be lifted by the gantry crane.

F. The fuel pool cannot be inadvertently drained by gravity since water must be pumped out.

G. Spent fuel pool high and low level, pool high temperature, and area high radiation indicators and alarms are provided. The high radiation level instrumentation does not actuate the ventilation system since this system is designed to run continuously.

H. Since no significant fuel storage pool leakage is expected to result from the dropping of loads, from earthquakes, or from missiles originating from high winds, the spent fuel pool makeup water system is Seismic Category II. Makeup water is either provided by the spent fuel coolant demineralizer pump taking suction on the borated water storage tank (BWST) or the decay heat removal pumps which can take suction from the BWST or the concentrated boric acid storage tank. Demineralized water can be added from a hose station in the pool area.

Further details are provided in Sections 9.6 (spent fuel cooling system), 9.7.3 (fuel storage area ventilation system), 5.4 (fuel storage building), and 9.8 (fuel handling system). Fuel handling accidents are discussed in Section 14.3.5.

1.6.14 SAFETY GUIDE 14 - REACTOR COOLANT PUMP FLYWHEEL INTEGRITY

The discussion of reactor coolant pump flywheel integrity is included in Chapter 4.

1.6.15 SAFETY GUIDE 15 - TESTING OF REINFORCING BARS FOR CONCRETE STRUCTURES

The reinforcing steel for Class 1 structures was purchased prior to the publication of Safety Guide 15.

One reinforcing bar of each size from each heat was tested in accordance with ASTM A-615-68 and ASTM A-370-68. Full-bar tensile tests and bend tests were conducted on bars up to and including #11 bars. Full-bar tensile tests were conducted on #14 bars and reduced section tensile tests were conducted on #18 bars. According to ASTM A-615-68, bend tests are not required for #14 and #18 bars.

The testing of the reinforcing bars complies with the Safety Guide with the exception of the requirement for testing of the bars for each 50 tons of steel from a single heat.

1.6.16 SAFETY GUIDE 16 - REPORTING OF OPERATING INFORMATION

Detailed procedures have been established to ensure that operating information is reported in a timely and complete manner which complies with the requirements of Safety Guide 16.

An Operations Startup Report, a First Year Report and semiannual Plant Operations Reports were prepared by the plant staff and submitted.

Personnel exposure and monitoring procedures have been established that will ensure each individual is provided the necessary records of his exposure and that exposure limits are not exceeded in the plant. Reports are submitted as required.

A program for administrative and physical control of all special nuclear material has been established and insures that the proper reports are provided.

Reporting abnormal events/licensee event reports (LER), overexposure and excessive radiation levels, loss of special nuclear material, and accidents involving licensed material is done by the plant staff through the Manager, Nuclear Operations. The procedures established ensure that any and all such events are quickly brought to the attention of plant management and acted upon within the allowed reporting period.

Special Reports are submitted as specified in technical specifications covering such topics as special maintenance, inservice inspection, Reactor Building integrity, and Reactor Building leak rate tests.

1.6.17 SAFETY GUIDE 17 - PROTECTION AGAINST INDUSTRIAL SABOTAGE

The Rancho Seco Industrial Security Program complies with the intent of Safety Guide 17.

For the purpose of planning the industrial security program at the Rancho Seco site, the major threats to the plant's security is considered to be from insider sources and offsite sources. Access control to the Rancho Seco restricted area is controlled by an eight-foot chain link fence. Security personnel are on duty

at all times. The common entrance located on the east side of the restricted area will be controlled continuously. A second entrance and a railroad gate, located west of the site, are locked with keys administratively controlled.

Personnel entry to the site is controlled by numbered badges. Visitors to the site are logged in by name, date, time, company, given a visitor's badge and are assigned to SMUD personnel for inplant control and surveillance. Upon leaving the site, visitors turn in their badge and are logged out.

The Rancho Seco emergency plan is coordinated with the security plan. In the event of a plant emergency, security personnel have predesignated key roles.

Security personnel directly participate in the plant drills within the scope of the emergency plan. This includes blocking all plant access, controlling entry of emergency vehicles and attendants as well as authorized personnel and establishing communication and coordinating with the Emergency Coordinator.

Direct participation in Emergency Plan practices includes simulating emergency conditions, plant sabotage, and forced entry as well as providing medical assistance and assisting in personnel evacuation.

The Emergency Plan Coordinator can establish liaison with local law enforcement agencies. In the event that the Emergency Plan does not take effect, liaison with local law enforcement agencies is made directly by plant security personnel. The Sacramento County sheriff provides assistance including all investigations and actions necessary to eliminate the threat or prevent any further intended damage to the plant.

Rancho Seco Operations procedures provide for routine surveillance of key equipment during normal plant operations. Through means of check lists, Operations personnel can determine correct valve position. Large critical valves are equipped with motor operators to return them to their correct position. A direct communication system between the Security Alarm Station and the Control Room is used to report unusual circumstances. This system supplements the plant warning alarms.

Plant security personnel use portable walkie talkies during random time sequence patrols of boundary fences. During these site boundary patrols, security personnel conduct routine surveillance of key plant equipment with particular attention to their security and normal operation. Rancho Seco security procedures outline equipment to be routinely checked and detail the procedure for reporting unusual circumstances.

Rancho Seco has had repeated plant reviews to assure that the design and physical arrangement provides redundancy, independent operation, and diversity of safety features. Protection against common mode failure provides additional protection against industrial sabotage. In addition, access control to reactor components, fuel storage facilities, the control room and other vital plant components make it unlikely that overall plant safety or integrity could be seriously threatened.

Procedures have been established to assure selection of reliable plant personnel. Surveillance and periodic review programs provide a continuous check on personnel performance, attitudes and behavioral patterns.

1.6.18 SAFETY GUIDE 18 - STRUCTURAL ACCEPTANCE TEST FOR CONCRETE PRIMARY FACTOR CONTAINMENTS

The Reactor Building is complete and thus it is impossible to embed triaxial strain gauges in the wall. The concrete was placed prior to the publication of Safety Guide 18.

The program that was developed for deformation measurements of the Reactor Building during acceptance testing is as follows:

1. Radial measurements were taken at three locations from one buttress to the opposite containment wall and four locations on each side between the containment wall and the interior structure between 140° and 320°. Three measurements were taken between 48°30' and 228°30' between the containment walls. Four measurements were taken at 48°30' between the secondary shield wall and the containment wall which includes the closed construction opening, and one measurement was taken at 45 feet below the spring line at 20° and 200° between the buttress and Reactor Building wall. See Figures 1.6-1 and 1.6-2.

2. Radial measurements were taken at the equipment hatch as follows: three measurements vertically above the center line of the hatch and three measurements horizontally on each side of center line of the hatch. See Figure 1.6-3.

3. Three vertical measurements were taken from the spring line, two measurements at 48°30' and 228°30' to the base slab, one to the elevation 60 ft. slab at 228°30'. In addition to the above one measurement was taken from the bottom of the elevation 60 ft.; slab to the base slab at 228°30'. See Figure 1.6-4.

4. Four vertical measurements were taken from the dome starting at 3 ft. from the centerline of the dome and at intervals of 15 ft. on the 228°30' azimuth. See Figure 1.6-4.

The data obtained from these measurements provided correlation with the design and satisfied the requirements of Safety Guide 18.

1.6.19 SAFETY GUIDE 19 - NONDESTRUCTIVE TESTING OF PRIMARY CONTAINMENT LINERS

Inspection and testing of the liner plate is described in Section 5.6.3.4.5. Nondestructive testing for penetrations and flued heads is covered in Section 5.6.3.7.7. The testing procedures used at Rancho Seco required radiography of 10 percent of the liner plate seam welds as compared with the requirement of a minimum of 2 percent radiography specified in the Safety Guide.

Leak chases, installed where seam welds will be inaccessible after construction is complete, were coated with a leak detecting solution and

TABLE 2.2-2

INSTITUTIONS WITHIN A 10-MILE RADIUS OF RANCHO SECO SITE

1.	John Gill Ranch (2 private residences)	5 1/2	miles N
2.	Altua Village for T.B. and alcoholic patients (50 units)	6 1/2	miles SW
3.	Arcohe Union Elementary School	7	miles SW
4.	Dillard Elementary School	8 1/2	miles NW
5.	Wilton Christian (Private)	8	miles NW
6.	Preston School of Industry (California You [illegible] ority)	10	miles E
7.	Ione High School	10+	miles E
8.	Ione Elementary School	10+	miles E
9.	M&M Preschool (Ione)	10+	miles E
10.	Mullin's Twin Oak Rest Home	8	miles NW
11.	Bufkin Residential Care Home	7	miles NW
12.	James Board and Care Home	9	miles SW
13.	Dorothy Keesee Rest Home	6	miles SW
14.	Johnson Guest Ranch	10	miles SW
15.	Leals Guest Home	6	miles SW
16.	Ione Forestry Fire Academy	10+	miles E
17.	Rawhide Nudist Ranch	8	miles NW
18.	AmCal Adult Training Center	10+	miles E

An update to the population study was prepared in 1979. For this study, counties within the 50-mile radius supplied updated 1985 population projections to SMUD.

TABLE 2.2-3

AREA AND POPULATION, CALIFORNIA COUNTIES, APRIL 1960 AND JULY 1970

COUNTY	LAND AREA* IN SQUARE MILES	APRIL 1960* POPULATION	APRIL 1960* DENSITY	1970* POPULATION	1970* DENSITY
Alameda	733	908,209	1239.0	1,073,184	1464.1
Alpine	723	397	0.5	484	0.7
Amador	593	9,990	16.5	11,821	19.9
Butte	1,663	82,030	49.3	101,969	61.3
Calaveras	1,027	10,289	10.0	13,585	13.2
Colusa	1,153	12,075	10.5	12,430	10.8
Contra Costa	734	409,030	557.3	558,389	760.7
Del Norte	1,003	17,771	17.7	14,580	14.5
El Dorado	1,714	29,390	17.1	43,833	25.6
Fresno	5,964	365,945	61.4	413,053	69.3
Glenn	1,317	17,245	13.1	17,521	13.3
Humboldt	3,573	104,892	29.4	99,692	27.9
Imperial	4,284	72,105	16.8	74,492	17.4
Inyo	10,091	11,684	1.2	15,571	1.5
Kern	8,152	291,984	35.8	329,162	40.4
Kings	1,395	49,954	35.8	64,610	46.3
Lake	1,256	13,786	11.0	19,548	15.6
Lassen	4,547	13,597	3.0	14,960	3.3
Los Angeles	4,060	6,038,771	1487.4	7,032,075	1732.0
Madera	2,144	40,468	18.9	41,519	19.4
Marin	520	146,820	282.3	206,038	396.2
Mariposa	1,455	5,064	3.5	6,015	4.1
Mendocino	3,507	51,059	14.6	51,101	14.6
Merced	1,982	90,446	45.6	104,629	52.8
Modoc	4,092	8,308	2.0	7,469	1.8
Mono	3,028	2,213	0.7	4,016	1.3
Monterey	3,324	198,351	59.7	250,071	75.2
Napa	758	65,890	86.9	79,140	104.4
Nevada	978	20,911	21.4	26,346	26.9

For the reactor coolant piping, the conditions are:

Material SA-106-GR C

Temperature 570°F, σ_y (570°F) = 30.425 ksi

I.D. = 28-5/8 in. max.

t = 2-1/4 in. min.

R = 15-7/16 in.

$\sigma_h = \sigma_y = 30.425$ (Assumed as worst case)

Rearranging the above equation and making appropriate substitutions:

$$M = \frac{\sigma^*}{\sigma_h}$$

$$1 + 1.61 \frac{C^2}{Rt} = \frac{(1.04\,\sigma_y + 10)}{\sigma_y}$$

C = 4.35 in.

Therefore, the critical crack length = 2C = 8.7 in.:

Reference 16 stipulates the flow stress (σ^*) equation is valid if:

$$\frac{(K_C/\sigma_y)^2}{C} \geq 5 \text{ yielding}$$

$$K_C \geq \sigma_y\,(5C)^{1/2}$$

$$K_C \geq 30.425 \quad 5(4.35)^{1/2} = 142$$

Where:

K_C = critical stress intensity factor (ksi$\sqrt{\text{in.}}$) as identified in Reference 4.

K_C data for SA-106-GR C material is not available. However, reference 15 represents pipe rupture tests on SA-106-GR B material, some of which could be classified as SA-106-GR C. The lowest KC value for the SA-106-GR B material that could be classified as SA-106-GR C is 168 ksi$\sqrt{\text{in.}}$; therefore, the equation is valid.

Reactor shutdown will be initiated within 24 hours and the reactor brought to cold shutdown conditions within 36 hours of determining that either the total identified or unidentified leakage rate limit is exceeded (see Technical Specification 3.1.6).

The procedures used to test and calibrate the radiation monitoring systems are described in Section 11.8.4.

The sensitivity and operability of the RCS inventory and Reactor Building sump methods of leak detection are determined by the checks, tests, and calibrations conducted as part of the plant preventative maintenance program. Additionally, an artificial controlled leak can be initiated by opening an RCS sample line for a period of time and then cross-checking this known leakage quantity against the calculated results of an inventory check.

4.2.3.8 Vents and Drains

Modifications, including a gravity drain line from the hot leg through the decay heat suction line to the Reactor Building sump and a line between the high pressure injection and pressurizer spray line, provide a single-failure-proof means of long-term cooling after a LOCA as described in Section 6.5.2.

Vent and drain lines are shown on the system diagram, Figure 4.2-1. They are located at the high and low points of the system and provide the means for draining, filling, and venting the heat transport loops and pressurizer. The reactor vessel cannot be drained below the reactor outlet nozzle using these drain lines. Each vent and drain line contains two manual valves in series. Vent lines are routed to a header connected to the flash tank and drain lines are routed to the reactor coolant drain tank. Remotely operable solenoid high point vents are located on the top of each hot leg and on the pressurizer and are controlled from the Control Room (H2PS). A fused disconnect to the neutral legs of the high point vent solenoid valve coils ensures that the high point vent valves will not open in case of a hot short to the control circuitry. Control Power is provided to the valve control scheme only when both fused disconnect switches are closed and the control room key switch for each set of valves is turned to the ON position. The valves vent through diffusers to the Reactor Building atmosphere, and enable non-condensible gases to be removed for satisfactory long term core cooling. Flow detection accelerometers are provided.

Analyses have demonstrated that a non-condensible gas bubble in the reactor vessel can be safely removed via the hot leg high point vents; even with a noncondensible gas bubble in the reactor vessel head, natural circulation will be maintained while the plant is cooled and depressurized to the Decay Heat Removal System entry conditions. While a residual gas bubble would still remain in the reactor vessel head following the plant cooldown, the gas bubble would not interfere with the maintenance of core cooling and could be slowly removed thereafter by various means. Accordingly, the NRC has granted an exemption to the 10CFR50.44(c)(3)(iii) requirement to install high point vents for the reactor vessel head.

7.4 OPERATING CONTROL STATIONS

In accord with proven power station design philosophy, all control stations, switches, controllers, and indicators necessary to start up, operate, and shut down the nuclear unit are in one control room. Control functions required to maintain safe conditions after a loss-of-coolant accident are also initiated from the centrally-located control room. Controls for some auxiliary systems are installed at remote control stations when the system controlled does not control power generation or emergency functions.

The Technical Support Center (TSC), located across from the Control Room, is provided to enable plant management to relieve the Control Room operators during emergency conditions. Designated personnel would utilize the TSC to:

1. Provide plant management and technical support to plant operations personnel.

2. Relieve the reactor operators of peripheral duties and communications not directly related to reactor system manipulation.

3. Prevent congestion in the Control Room.

4. Perform Emergency Offsite Facilities (EOF) functions for the Alert Emergency Class, the Site Area Emergency Class and General Emergency Class until the EOF is functional.

An auxiliary shutdown panel is located in the West Switchgear Room on the grade level. This panel provides shutdown capability in the event that shutdown can not be accomplished from the Control Room due to its uninhabitability. Some activities other than at the auxiliary shutdown panel, specified in emergency procedures, would be required to control the plant.

7.4.1 GENERAL LAYOUT

Control room design allows one man to supervise operation of the unit during normal steady state conditions. During abnormal conditions, other operators are available to assist the control operator. Figure 7.4-1 shows the control room layout. Instrumentation and control devices for startup, shutdown, and normal and emergency operation are located on the console and the vertical control panel. Most of the essential instruments and controls for power operation are located on the console.

The vertical control panels contain instrumentation less essential than normally required during startup, before the reactor is critical, and during shutdown. Instrumentation is grouped on the panels so all pertinent indicators, recorders, and controls are within easy reach of the operator.

Support instrumentation for systems represented on the control console, including the plant computer and the vertical panels, are mounted in cabinets adjoining the main control area.

The TSC has facilities to support the plant management and technical personnel who will be assigned there during an emergency and will be the primary onsite communication center for the plant during the emergency. Data will be provided in the TSC to analyze the plant steady-state and dynamic behavior prior to and throughout the course of the accident.

The results of this analysis will be used to provide guidance to the Control Room operating personnel in the management of abnormal conditions and in accident mitigation. TSC personnel will also use the environmental and radiological information available from the TSC data system to perform the necessary functions of the EOF when this facility is not operational.

From the TSC it will be possible to monitor plant status via the computer system, to monitor radiation via perimeter monitoring, and to determine the initial direction of movement of any off-site release via the meteorological system. Communication systems, both hardline and radio frequency, assure that the site will have contact with off-site groups.

7.4.2 INFORMATION DISPLAY AND CONTROL FUNCTION

The information necessary for routine monitoring of the nuclear unit and plant is displayed on the control room consoles and panels near the operator. Information display and control equipment frequently employed on a routine basis, or protective equipment needed in an emergency, is mounted on the consoles. Less frequently used equipment, such as indicators and controllers used primarily during startup or shutdown, is mounted on vertical panels facing the consoles and is visible to the operator.

A plant computer is available in the control room for alarm monitoring, post-trip review, and sequence of events analysis, performance monitoring, and data logging. On-demand printout is available to the operator in addition to the plant computer alarm logging and periodic logging of station variables. Redundant annunciator signals are provided for most plant computer alarms. Information displays are designed to provide the operator with sufficient information to make proper evaluations under the full range of plant operating conditions. The displays are arranged to facilitate evaluation and to avoid the possibility of confusing the operator.

All parameters monitored by the RPS are indicated on the system cabinets, the control consoles, and as plant computer outputs. Instrumentation strings

Pushing the "acknowledge" pushbutton initiates the acknowledged mode in which the audible is silent and the window illumination is steady. When the alarm contact returns to normal, the window starts flashing slowly and the audible remains silent. Pushing the "reset" pushbutton initiates a return to normal mode if the alarm contact has returned to normal, or stays in the acknowledged mode if the alarm contact is still abnormal. Where two or more input alarm contacts are connected in parallel and connected in a standard logic card, only the first signal will be annunciated in the aforementioned sequence of operation. In the event other alarm contacts are closed in the circuit, all must be returned to normal condition before reset can be accomplished.

First-out function provides selected alarm points with bi-level, first-out indication. Indication of the first point to go off-normal is accomplished by means of illumination of a red-capped pilot bulb in the upper right quadrant of the window. The first-out system is supplied with its own test and reset controls. Upon operation of the "reset" pushbutton, the first-out system is "re-armed" to accept another first-out series of alarms. The first-out system and the standard alarm point system share the same acknowledge function on the first-out system.

The multiple (x-y) logic cards are employed where it is required to annunciate more than one abnormal condition on a monitored circuit. The first alarm signal is annunciated and acknowledged in the manner described for a standard logic card. Each subsequent alarm signals to an (x-y) logic card and re-triggers the flasher and audible cards, returning the window to the alarm mode and requiring the operator to acknowledge these alarms. Up to five independent alarm contacts can be connected to an (x-y) logic card. Additional cards are used when there are more than five independent alarm contacts. All alarm contacts return to the normal mode after reset.

There are six sets of annunciator pushbutton controls located in different sections of the control console in the Control Room. Each control set contains the "acknowledge," the "test," and the "reset" lamp window group pushbutton controls. The secondary system console section contains three pushbutton control sets.

One set controls two secondary system annunciator lamp window groups. Another set controls the electrical annunciator system group. The remaining set controls two fire protection annunciator lamp window groups. The reactor and integrated control system console section contains one pushbutton control set. This set controls two primary system annunciator lamp window groups. The reactor coolant console section contains two pushbutton control sets. One set controls two plant auxiliaries annunciator lamp window groups. The other set controls two safety features annunciator lamp window groups.

Depressing a "test" pushbutton simulates operation of all alarm field contacts connected to the logic cards involved. This initiates their alarm mode; the windows will flash and the audible will sound as described before. The "acknowledge" and "reset" functions will also operate as described before. Testing of the annunciator system in this manner can be done at any time and under all modes of plant operation.

Most alarm signals are connected to both the annunciator system and to the plant computer. Plant operators test the annunciator system using the "test" pushbuttons at the beginning of their shifts.

7.4.3 SUMMARY OF ALARMS

Visible and audible alarm units are used in the Control Room to warn the operator of unsafe conditions in any system. Audible Reactor Building evacuation alarms are initiated manually by the operator. Audible alarms are sounded in the Control Room and in the appropriate areas throughout the plant if high radiation conditions occur.

7.4.4 COMMUNICATION

In addition to the information contained herein, refer to the Emergency Plan for a further description of on-site communication systems.

Paging systems within the power block buildings and the station telephone and paging systems have redundant power supplies. A sound powered telephone system has been installed to provide communication capability between the West Switchgear Room (location of the auxiliary shutdown panel) and other locations of equipment requiring manual operation to bring the plant to a safe shutdown in case of fire in the Control Room. Acoustic booths and noise cancelling transmitters are used where the background noise level is high. Communications outside the station are through the local telephone network. Radio voice communications capabilities also exist.

7.4.5 OCCUPANCY

Safe occupancy of the Control Room and TSC during abnormal conditions is provided by the design of the Auxiliary Building. Adequate shielding maintains tolerable radiation levels in the control room and TSC for maximum hypothetical accident conditions, and the control room has a radiation detector with appropriate alarms. Provisions are made for Control Room and TSC air to be recirculated automatically if high airborne radioactivity or chlorine levels are detected. Emergency lighting is also provided.

The potential magnitude of a fire in the Control Room is limited by the following factors:

A. Control Room construction is of noncombustible materials.

B. Safety features and reactor protection system control cables and switchboard wiring have passed the flame test described in Insulated Power Cable Engineers Association Publication S-61-402 and National Electrical Manufacturers Association Publication WC 5-1961.

C. Furniture in the Control Room is generally of metal construction.

D. All areas of the Control Room are accessible and adequate fire extinguishers are provided.

E. The Control Room is occupied at all times by a qualified person trained in fire extinguishing techniques.

F. The vertical control panels form a barrier between the cabinet area and the Control Room.

The only flammable materials inside the Control Room are:

1. Paper in the form of logs, records, procedures, manuals, and diagrams, as required for station operation.

2. Combustible materials used in the manufacture of various electronic equipment and Control Room furnishings.

As indicated by the above list, flammable materials have been minimized to the extent that a fire will not be likely to spread. Therefore, if a fire is started, it will be so small that it can be extinguished using a hand-held fire extinguisher. The resulting smoke and vapors would be removed by the ventilation system.

7.4.6 AUXILIARY CONTROL STATIONS

Auxiliary control stations are provided where their use simplifies the control of auxiliary systems equipment. Sufficient indicators and alarms are provided to inform the central Control Room operator of abnormal conditions at remote control stations.

7.4.7 SAFETY CONSIDERATIONS

The primary objectives in the Control Room layout are to provide necessary controls to start, operate, and shut down the nuclear unit with sufficient controls, displays and alarms to insure safe and reliable operation under normal and accident conditions. Special emphasis was given to maintaining control integrity during accident conditions.

The safety features section of the control panel was designed to minimize the time required for the operator to evaluate system performance under accident conditions. Any deviations from predetermined conditions are annunciated so that the operator may take corrective action in the control panel.

7.4.8 SYSTEM EVALUATION

7.4.8.1 Control Room Availability

Safe operation and shutdown of the power plant is conducted from the Control Room. The Control Room is specifically designed to permit the operator to perform his duties under all credible accident conditions. The forced abandonment of the Control Room is deemed highly unlikely for the following reasons:

A. The Control Room has been given the highest priority for shielding from external radiation of any area in the plant.

B. Non-flammable construction materials were used for all interior components, i.e., control boards, furniture, etc.

C. Adequate fire-fighting equipment is available in the Control Room and operators have fire-fighting training.

D. Self-contained air breathing equipment is available in the Control Room for operator use.

E. Cables and switchboard wiring have passed flame tests as described in IPCEA publications S-61-402 and NEMA WC 5-1961.

F. Combustible materials in the Control Room are kept to the minimum required. Permanent plant records and non-essential reference materials are stored elsewhere.

G. A redundant two train HVAC system has been provided.

H. The vertical control boards provide a fire and smoke barrier between most of the electrical devices that could generate large amounts of smoke.

I. Fire-proof or fire-resistant doors are installed on all rooms adjoining the Control Room where significant amounts of combustible materials are stored.

7.4.8.2 Control Room Evacuation Emergency

Provisions have been made for remote shutdown of the plant in the event that use of the Control Room is lost.

Instrumentation on the auxiliary shutdown panel (H2SD) monitors all parameters necessary to achieve and maintain natural circulation. Isolation switches separating circuitry in the auxiliary shutdown panel from Control Room circuitry have been installed on the switchgear and MCC compartments. Switchgear loads have local controls. The controls for the MCC loads are on the shutdown panel.

The auxiliary shutdown panel is located in the 4160 volt West Switchgear Room, which is physically isolated from the Control Room. The following indications are provided:

1. Wide Range Steam Generator Levels for both OTSGs;
2. Wide Range Steam Generator Pressure for both OTSGs;
3. Pressurizer Level;
4. Wide Range Reactor Coolant System Pressure;
5. Wide Range Reactor Coolant System Hot Leg Temperature (both loops);
6. Wide Range Reactor Coolant System Cold Leg Temperature (both loops);
7. Source Range Nuclear Instrumentation;
8. Makeup Tank Level.

These indications have been routed such that damage which might occur in the Control Room will not affect their availability.

Procedures have been developed which describe the tasks to be performed to effect the remote shutdown. Implementation of the procedures will establish natural circulation with heat removal by the steam generators, the code safety valves and the atmospheric dump valves. Feedwater is provided by the auxiliary feedwater pumps from the condensate storage tank. The plant will stabilize in a hot shutdown condition before being taken to cold shutdown. All system contraction volume will be made up from the BWST. With slight variations in the normal shutdown process, a one percent $\Delta k/k$ shutdown margin will be maintained through the entire cooldown process.

The methodology is the same for both a loss of offsite power and with offsite power available. The "A" train systems are used to shutdown the plant. Offsite and other power sources may be de-energized as necessary to prevent spurious actuations.

8.2 ELECTRICAL SYSTEM DESIGN

8.2.1 NETWORK INTERCONNECTIONS

Rancho Seco generates electric power at 22 kV. This power is fed through an isolated phase bus to two main transformers where it is stepped up to 220/230-kV transmission voltage. Six overhead transmission lines carry the energy from the generating station switchyard.

8.2.1.1 Single Line Diagram

The area transmission network is shown in Figure 8.2-1; the arrangement of buses at the site is shown in Figure 8.2-2.

8.2.1.2 Reliability Considerations

Reliability features that minimize the probability of power failure due to faults in network interconnections and associated switching are as follows:

A. Flexibility and capability are designed into the 220/230-kV network by installing six transmission lines from the Rancho Seco Generating Station switchyard to switchyards in the area transmission network. The area transmission network can supply all area generation to load centers with any one line out of service. An analysis has been performed of the system stability to ensure that it will accommodate the trip of the Rancho Seco unit and that such a trip will not disrupt availability of offsite power.

B. Two different right of ways are used to route the six transmission lines from the station switchyard. Two of the lines are routed south and the other four are routed to the west.

C. The switching arrangement in the 220/230-kV switching station includes two full-capacity main buses. Primary and backup relaying are provided for each circuit along with circuit breaker failure backup protection. These provisions permit the following:

   1. Any circuit can be switched under normal or fault conditions without affecting any other circuit.

   2. Any single circuit breaker can be isolated for maintenance without interrupting the power to or protection of any circuit.

   3. A short circuited single main bus can be isolated without interrupting service to any circuit.

   4. Backup relaying provides separation in the event of primary relaying failure to trip.

D. The primary bushing taps of the main and start-up transformers include 3-phase 220/230-kV no-load disconnect switches. These disconnects permit isolation of these transformers to allow reclosing connecting loops between the east and the west 220/230-kV buses. If there is a loss of all 220/230-kV remote connections, power to

the safety features will be supplied from the auxiliary diesel generators, as described in Section 8.2.3.

8.2.2 STATION DISTRIBUTION SYSTEM

The plant distribution system consists of the various auxiliary electrical systems required to provide reliable electrical power during all modes of operation and shutdown conditions. The systems are designed with sufficient power sources, redundant buses, and required switching to attain this reliability. Safety features auxiliaries are arranged so that loss of a single bus will leave sufficient auxiliaries to safely perform the required function. In general, auxiliaries related to functions other than safety features are connected to their respective unit auxiliary electrical bus. Safety features loads are divided between two independent systems according to the single failure criterion.

8.2.2.1 Single Line Diagram

Figure 8.2-2 is a single-line diagram of the onsite power distribution system.

8.2.2.2 Auxiliary Transformer

Rancho Seco has two unit auxiliary transformers and two startup transformers. The unit auxiliary transformers are connected to the generator bus to provide normal power to the unit. The startup transformers are connected to the 220/230-kV switchyard and provide power for startup, safety features, hot shutdown, and cold shutdown requirements. The startup transformers also provide standby sources of power to serve auxiliaries if the unit auxiliary transformers fail.

Unit auxiliary transformer No. 1 is a two-winding transformer, rated 26/34.6 MVA. The voltage rating of the primary winding is 22 kV. The secondary winding is rated 6.9 kV.

Unit auxiliary transformer No. 2 is a three-winding transformer. The primary winding is rated 30/40 MVA and 22 kV. One secondary winding is rated 13/17.3 MVA and 4360 volts. The other secondary winding is rated 18/24 MVA and 4360 volts.

Startup transformer No. 1 is a three-winding transformer. The primary winding is rated 29/38.6 MVA and 230 kV. One secondary winding is rated 26/34.6 MVA and 6.9 kV. The other secondary winding is rated 7.5/10 MVA and 12.47 kV.

Startup transformer No. 2 is a four-winding transformer. One of the secondary windings is a nonconnected delta tertiary winding whose purpose is to block zero sequence current. The primary winding is rated 32/42.6 MVA and 230 kV. One of the connected secondary windings is rated 18/24 MVA and 4360 volts. The other connected secondary winding is rated 17/22.6 MVA and 4360 volts.

8.2.2.3 6900-Volt Auxiliary System

The 6900-volt auxiliary system is designed for the 10,000-horsepower reactor coolant pump motors. This system is arranged into two bus sections, each

Each vital battery is independent and Class 1 and housed in a separate room with an independent Class 1 ventilation exhaust fan. The exhaust fan and ductwork supports are also Seismic Category I. The exhaust fans receive their power from independent Class 1 power supplies. The normal battery charger and standby charger units for each 125-volt d-c safety features actuation and reactor protection bus, and each plant auxiliaries bus except the normal charger for bus SOF are energized from the 480-volt nuclear services motor control centers, which, in turn, have diesel generator power available. The auxiliary diesel generators are sized to carry the battery charger load.

All of the battery chargers are of the static, constant voltage, fully automatic, silicon-controlled rectifier (SCR) type. With the exception of the battery chargers installed in the NSEB, all battery chargers are capable of operation in parallel for fast recovery. The output is regulated and current limited, and power can only flow through them in the normal output direction.

The single failure analysis for the 125-volt d-c system (Table 8.2-1) provides further explanation of this design. Each charger can provide continuous d-c load and floating charge, with occasional equalizing charge, as required. Charger failure is annunciated in the Control Room, initiated by voltage failure relays in the d-c output and 480-volt a-c supply. The tripping of d-c breakers is annunciated in the Control Room and initiated by auxiliary contacts of the breakers. D-c bus undervoltage and ground are annunciated in the Control Room by voltage relays.

The two d-c circuit breakers on the standby battery chargers are key interlocked to prevent both breakers from being closed at the same time. This precludes paralleling two different batteries through one standby charger (see Figure 8.2-3).

Each battery is sized with sufficient capacity to carry its respective load for 2 hours. Specific gravity readings are reliable indicators of the state of charge of each battery cell when readings are compared to previous readings taken during the charged states of the batteries. Initial readings taken during the field acceptance tests of the batteries provide baseline data against which the periodic readings are compared to detect degradation of charge.

A permanent record of all battery readings is maintained. Comparison of the latest readings with previous data provides the means for determining the need for corrective action such as equalizing the cell voltage or replacing a cell.

Discharge tests conducted at each refueling interval provide positive verification of battery capability and yield additional data for determining battery degradation with time. The batteries are provided with transparent plastic containers which allow the plates, separators, and electrolyte to be inspected visually for physical abnormalities that could cause battery degradation.

D-c bus undervoltage and ground detector relays are provided to continuously monitor the d-c systems and give annunciation in the Control Room of abnormal conditions. In addition, the opening of any d-c distribution panel circuit breaker is annunciated in the Control Room.

The voltage level of the d-c buses is normally maintained within the range of 133 to 135 volts d-c by the battery charger. The bus undervoltage relay will initiate an alarm when the voltage drops below 125 volts d-c. With this setting, the undervoltage alarm will indicate loss of battery charger output and consequent discharge of the battery by the connected load.

The batteries will receive equalizing charges as required at a nominal voltage of 140 volts d-c. The equalizing charge can be manually initiated and terminated. An equalizing charge can also be initiated by re-establishing power following a loss of incoming battery charger power. An automatic timer will then return the charger to nominal bus voltage. Each battery charger is equipped with a d-c overvoltage relay which will automatically trip the charger input breaker if the d-c output voltage exceeds 141 (+5, -0) volts d-c and alarm the overvoltage condition in the Control Room.

Seismic qualification tests were conducted on a battery charger, and on three cells of the exact type used in the A, B, C, and D d-c systems, mounted on racks similar to racks being used at Rancho Seco. The accelerations and frequencies used in the tests were chosen in accordance with the seismic design bases given in Chapter 5. The battery cells, charger, and racks remained in position and withstood the tests without damage. Ethafoam has been applied to battery racks where necessary to ensure that all rack-to-cell spacings are within vendor seismic calculation assumptions. The manufacturer of the batteries has stated that no changes have been observed which would indicate physical weakening of cells of similar construction which have been in service for more than 40 years, and cells of identical construction which have been in service for more than 15 years. The manufacturer concludes that the A, B, C and D system batteries will be capable of meeting the specified seismic requirements for at least 20 years after installation. These batteries will be replaced within 20 years.

The d-c panels A, B, C, and D were analyzed in detail by use of a computer. Based on this analysis and information on the internal components the panels will not malfunction or be damaged as a result of the DBE.

All normal and standby battery chargers can fully recharge their respective batteries in 8 hours. The safety features actuation and reactor protection battery chargers can carry the connected load under all operating conditions, except for combined loss of all onsite and offsite a-c power.

As shown in Table 8.2-1, the 125-volt d-c system is arranged so that a single fault within the system does not prevent the reactor protective system, safety features actuation system, and the nuclear service equipment from performing their safety functions.

8.2.2.7 <u>120-Volt a-c Vital Power System</u>

The 120-volt a-c vital power system provides a reliable source for essential power, instrumentation, and control loads under all operating conditions. The system consists of eight 120-volt a-c bus sections. Each pair of buses is normally supplied power by its respective inverters. Each static inverter is supplied from a separate 125-volt d-c system bus. Backup power is provided from dedicated nuclear services 480-volt a-c buses through their respective

regulating transformers, static switches, and manual bypass switches. The arrangement of the distribution system for the 120-volt vital a-c system is shown in Figures 8.2-3 and 8.2-5.

The system is arranged so that any type of single failure or fault will not prevent the reactor protective system, safety features actuation system, or safety features equipment from performing its safety functions. The system is a four-channel system, similar to the vital 125-volt d-c system described in Section 8.2.2.6, comprised of Channels A, B, C and D. Channels A and C are separated from Channels B and D; however, Channels A and B are not separated from Channels C and D, respectively. Thus, the basic two-train separation scheme is maintained at the 120-volt a-c level and throughout the electrical distribution system for all vital buses. Panels S1A and S1A2-1 and their associated inverter comprise Channel A. Similarly, panels S1B and S1B2-1, S1C and S1C2-1, and S1D and S1D2-1 comprise Channels B, C and D, respectively. Backup power is provided from the Train A nuclear services 480-volt a-c system for Channels A and C and Train B for Channels B and D.

The system provides two paths of back-up a-c power to each panel from the other diesel generator of the same train. One path is through the standby battery charger, vital 125-volt d-c bus and inverter; the second path is through the voltage regulating transformers to the vital 120-volt a-c bus. The static switches provide automatic transfer of the 120-volt a-c vital buses to the backup a-c power supply. The manual bypass switches ensure that the 120-volt a-c system is available during inverter/static switch outage or maintenance.

There is local indication available at the inverter control panel for monitoring output voltage, current and frequency. A common inverter trouble alarm, monitored by IDADS, alarms in the Control Room on low output voltage or a frequency mismatch between the inverter output and the regulated standby source.

8.2.2.8 <u>120-Volt a-c Unregulated Power System</u>

A low voltage, 120-volt a-c power system, shown in Figures 8.2-3 and 8.2-5 sheet 3, supplies non-safety oriented instrumentation, control, and power loads requiring unregulated 120-volt a-c power. The system consists of two distribution panels and 480/120-volt transformers fed from motor control centers.

8.2.2.9 <u>120-Volt a-c Uninterruptible Power Supply</u>

For arrangement of the distribution system, see Figures 8.2-3 and 8.2-5.

Two independent, redundant, non-Class 1, 120-volt a-c uninterruptible power supplies associated with inverters SIGA and SIGB are provided. These power supplies are provided primarily for the ICS and NNI control systems and to support computer systems added in response to NUREG 0737 and for the Nuclear Services Electrical Building fire protection system. Each power supply consists of a 600A, Class 1 battery charger connected to a Class 1, diesel-backed load center; a 50 kVA, non-Class 1 inverter; a non-Class 1 battery sized for 30 minutes operation; a 120-volt a-c, non-Class 1 panel; and a 45 kVA, non-Class 1, 480/120-volt regulating dry type transformer.

Two additional 120-volt a-c uninterruptible power supplies associated with inverters SIG and SIN1 are provided for the plant computer and other miscellaneous non-Class 1 instrumentation systems.

8.2.2.10 Lighting

Lighting is provided to permit the safe performance of operating and maintenance duties. The plant normal lighting is supplied at 120 and 277-volt a-c. Switchyard and plant perimeter lighting is supplied at 480-volt a-c. Lighting cable is a subset of power cable.

The main lighting distribution switchboard provides the 120/208 and 277/480-volt a-c source of power for the plant lighting system. A portion of the lighting distribution switchboard is arranged for automatic transfer to the 125-volt d-c station battery system providing emergency lighting upon loss of the normal a-c power supply.

Control Room emergency lighting is fed from the redundant 125-volt d-c panels SOC and SOD (Figures 8.2-3 and 8.2-4) during emergency conditions.

8.2.2.11 Evaluation of the Physical Layout of Electrical Distribution System Equipment

Electrical distribution system equipment is arranged to minimize the vulnerability of vital circuits to physical damage as follows:

A. The two startup transformers are outdoors, physically separated from each other. Lightning arresters have been provided on the high voltage side for surge protection. The transformers are protected by automatic deluge type fog systems to extinguish oil fires quickly and prevent the spread of fire. Transformers are isolated by fire walls to minimize their exposure to fire and mechanical damage.

B. The unit auxiliary 6900-volt switchgear, 4160-volt switchgear, 480-volt switchgear, and motor control centers are in areas where exposure to mechanical, fire, and water damage is minimized. Equipment is coordinated electrically to operate safely under normal and short circuit conditions.

C. Nuclear service 4160-volt and 480-volt switchgear are in Seismic Category I structures. Physical separation of redundant power systems has been maintained throughout. Equipment is coordinated electrically to operate safely under normal and short circuit conditions.

D. 480-volt motor control centers are in areas of electrical load concentration. Nuclear service motor control centers are in a Seismic Category I structure. Physical separation of redundant power systems has been maintained throughout.

E. Station batteries and associated chargers and inverters serving the protective systems loads are independent Class 1 installations housed in separate rooms within the Auxiliary Building and Nuclear Services Electrical Building, which are Seismic Category I structures.

F. Nonsegregated, metal-enclosed 12,470-, 6900-, and 4160-volt buses are used for major bus runs where large currents are carried. The routing of these metal-enclosed buses minimizes their exposure to mechanical, fire, and water damage. Silicone foam and silicone elastomer were injected in all 480-, 4160-, and 6900-volt bus ducts at all fire wall penetrations.

G. The application and routing of control, instrumentation, and power cables minimizes their vulnerability to fire. The cables have been chosen by using conservative margins with respect to their current-carrying capacities, insulation properties, and mechanical construction, and are of a flame-resistant construction. Power and control cable insulations for use throughout the plant have been selected to minimize the harmful effects of radiation, heat, and humidity. Appropriate instrumentation cables are shielded to minimize induced voltage and magnetic interference. Wire and cables related to safety features and reactor protective systems are routed and installed to maintain physical separation of redundant channels.

Circuits, trays, conduit, and electrical equipment which are part of Class 1 systems are color coded (see Chapter 7) to help verify that channel separation is actually achieved.

H. The separation of redundant cables of safety systems is accomplished by spatial separation in accordance with the criteria given in this section. Specific criteria applicable only to reactor protection system (RPS) and engineered safety features actuation system (ESFAS) cables are included.

1. Separate raceway (cable tray, conduit, and penetration) systems are installed for the following classes of cable: 15-kV, 5-kV, 600-volt power and control, and instrumentation cable. Class 1 600-volt power and control cables of one channel are run together in trays that belong to the same channel. Class 1 instrumentation circuits are routed in metal raceway as explained in (2) below. Cable for plant lighting is considered a subset of power cables.

2. RPS and ESFAS instrumentation have their channels routed in separate raceways and are physically separate from each other throughout the plant.

3. RPS and ESFAS power and control, Channels A and B are separated physically and have separate raceway systems. Non-safety related circuits may be run in these trays (see item 13 below).

4. The channel separation criteria are met in both the reactor protection system and the safety features actuation system.

5. Power and control circuits are not mixed with instrumentation circuits in any raceway for any system unless an engineering analysis is performed for acceptability.

6. The minimum horizontal distance is 3 feet, 0 inches between trays of different channels.

7. Paralleling trays of different channels in a vertical stack is not permitted.

8. The minimum vertical distance between trays of different channels crossing each other is 18 inches. Additionally, at tray crossings, a 1/4-inch thick Haysite polyester board barrier has been installed between the trays for protection.

9. The maximum fill allowed in trays is limited to prevent exceeding the cable ampacity rating in accordance with IPCEA No. P-46-426 and ICEA No. P-54-440 including the applicable derating factors, and the designed weight of cable on the tray supports.

10. All electrical cables installed in cable trays are covered with flame-resistant jackets.

11. The base ampacity rating of cable is as conservative as, or more conservative than, that established in published IPCEA standards and is in accordance with manufacturers' standards. Cables are selected on the basis of 100 percent load factor. Power cables are sized for 125 percent of full load current of the equipment served unless an engineering analysis for acceptability is performed justifying an exception. Where cable is routed through several types of installation conditions (conduit, cable tray), cable ampacity is selected based on the most limiting condition.

12. Fire protection systems (sprinklers) are provided near the electrical penetrations outside of the containment building. Fire protection systems (carbon dioxide) are provided in the cable shafts with the exception of the Nuclear Services Electrical Building cable shaft which has water spray protection. Adequate fire extinguishers are provided inside the containment building (see Sections 1.4.3 and 1.5.3).

13. In addition to the color identification of cables and raceways as explained in Section 7.1.4, each cable is identified at termination points by affixed markers. Markers have the cable number plus a color mark corresponding to the channel color. Non-safety related cable markers have no color mark. Cable raceways are identified by coded markers.

    Trays or conduits used for RPS or ESFAS channels have diamond-shaped color spots corresponding to the channel color code. For example, a Channel A tray is identified with red tray markers and contains only red-marked and/or white-marked, black-jacketed cables. (Non-safety related cables routed in safety-related raceways are identified with white marking.) Although cables that are not part of reactor protection or safety features actuation channels may share a safety-related raceway, in no case will these cables cross from one channel's raceway to another and thus form a bridge.

14. Protection system, safety features system, and Class 1 electrical system components mounted on control boards, panels, and relay racks are designed with physical separation between redundant wiring and components.

    Generally, redundant channel wiring enters the control panels in conduits. Most redundant wiring inside the control panels is separated by a steel barrier. However, wiring which is common to two redundant channels exists. The jackets of these wires are identified with the color of the channel energizing them.

    Physical separation has been provided between redundant switches and wiring for Class 1 breakers 3A05, 3A21, 3B05, 3B21, 4A01, 4A09, 4A10, 4B01, 4B04, 4B05, and 4B11, and for the diesel generator speed and voltage controls on H2ES, the electrical switching panel. H2ES is a Seismic Category I panel.

The quality assurance program defined in the Rancho Seco Quality Manual ensures compliance with the separation criteria during design and installation. Specific procedures have been set up by means of the computerized circuit and raceway schedule to facilitate coordination of design information in the field.

Each conductor is identified with a permanent marker that contains the scheme-cable number and conductor number shown on the applicable wiring diagram. Each marker for a Quality Class 1 conductor is identified by a color that corresponds to the safety channel separation group. Raceways are similarly identified. The numbering of cables and raceways is such that separation groups in Channels A and B can be recognized by the computerized circuit and raceway schedule to ensure separation by groups.

Electrical testing is performed to assure the integrity of all circuits and their proper connections.

8.2.3 SOURCES OF AUXILIARY POWER

8.2.3.1 Description of Power Sources

8.2.3.1.1 Offsite Power

As previously described, all of the normal power supply to plant auxiliary loads is provided through the unit auxiliary transformers connected to the generator bus. Power to all nuclear services auxiliary loads is provided by both startup transformers, which are connected to the 220/230-kV system. During normal operation, one of the 6900-volt buses is supplied by unit auxiliary transformer No. 1. The other bus is supplied by startup transformer No. 1. The unit auxiliary transformers, as well as the startup transformers, are sized to carry full plant auxiliary loads.

Upon separation of the unit from the 220/230-kV system, both the reactor and the turbine-generator will be tripped. If power is not available from the unit auxiliary transformers, it will be obtained from the startup transformers. One 4360-volt winding of startup transformer No. 2 is normally connected to 4160-volt nuclear service buses 4B and 4B2. The 12.47-kV winding of startup transformer No. 1 is connected to a 7.5 MVA, 12.47/4.36-kV nuclear service supply transformer, which normally feeds 4160-volt nuclear service buses 4A and 4A2.

The adequacy of the onsite distribution of power from offsite circuits has been analyzed in accordance with References 1 through 3. The analyses demonstrate that:

A. Undervoltage relay setpoints will provide design protection performance for Class 1 equipment from sustained degraded voltage;

B. Undervoltage setpoints (voltage and time) will preclude spurious separation from the preferred offsite sources;

C. Technical Specification Limiting Conditions for Operation (LCOs) for operation below 219 kV provide assurance that Class 1 equipment will not be exposed to sustained degraded voltage; and

D. Equipment overload protection devices will protect equipment while first allowing bus supply breaker tripping on undervoltage conditions.

Logging and alarming of 4160-volt Class 1 bus overvoltage to the IDADS computer is provided. The operators take manual action to reduce the overvoltage condition.

8.2.3.1.2 Onsite Power

The onsite a-c power system consists of four diesel generator units (GEA, GEB, GEA2 and GEB2) and associated 4160-volt nuclear services buses (4A, 4B, 4A2 and 4B2) together with their respective distribution network as shown in Figure 8.2-2. The onsite power system at Rancho Seco is based on a dual train (A and B) scheme composed of two subtrains within each train. The 4160-volt Train A consists of buses 4A and 4A2 and their associated distribution network. Similarly, the 4160-volt Train B has buses 4B and 4B2. Each train is redundant and independent of the other. Each subtrain bus is provided with an independent diesel generator unit. Each 4160-volt bus and associated diesel generator is independent of the other bus in the same train but not redundant. Loss of a 4160-volt bus results in an incomplete train and, therefore, from the standpoint of redundancy, must be considered as loss of the complete train.

Diesel generators GEA and GEB primarily support emergency core cooling system loads, with diesel generators GEA2 and GEB2 supporting HVAC loads, loads required by NUREG 0737 modifications, auxiliary feedwater/Emergency Feedwater Initiation and Control loads, and other loads due to plant modifications.

8.2.3.1.2.1 Bruce GM Diesel Generator GEA and GEB Auxiliaries

The nameplate rating of each auxiliary generator is 2750 kW at 0.8 power factor. The units are in different rooms in the reactor Auxiliary Building, separated by Seismic Category I fireproof concrete walls. These concrete walls are 12 inches thick and reinforced to prevent missiles, explosions, and fires in one room from affecting the other room. There are grouted electrical penetrations in the wall between the A and B diesel generators, and a nonrated 1/4-inch thick steel plate hatch at the mezzanine level allows access from the Train A duct shaft into the Train B duct shaft.

Each diesel engine has its own individual 7-day fuel supply consisting of a buried 50,000-gallon storage tank and a nominal 2-hour day tank. The day tanks are located integral with each machine. The storage tanks and the fuel supply systems are all of Seismic Category I and Quality Class 1 construction. The storage tanks are of buried design and physically separated by approximately 500 feet, one located north of the Reactor Building and east

of the borated water storage tank and the other located east of the warehouse and shops building. The tanks are located so they can be filled from tank trucks.

Each storage tank is furnished with two submersible motor-driven fuel oil pumps. Any one pump is capable of supplying the fuel requirements of both of the diesel generators. Each storage tank along with its pumps and its control system has a power supply separate from the other system.

The discharge oil piping between the pumps for each tank and its diesel generator is buried underground and is intertied with the other system by two normally closed valves. The pumps are controlled by two redundant level switches in the day tanks. A full-sized oil return line is provided between each day tank and its respective storage tank to preclude oil overflow in the event that the day tank level switch fails on "high level."

Each diesel engine is provided with an individual closed circuit circulating liquid cooling system as shown in Figure 8.2-6. The coolant consists of a minimum 17 percent solution of ethylene glycol in water.

Each engine is equipped with two centrifugal circulating pumps driven directly from the engine. The pumps rotate only when the diesel engine is running and circulate the coolant through the engine jacket, the combustion air aftercooler, the shell side of a water-cooled heat exchanger (or its bypass), and a lubricating oil cooler.

Each engine's cooling system has an expansion tank with a trapped air space. As the coolant heats up and expands into the expansion tank, a backpressure of up to 7 psig is created. An immersion heater is provided to maintain an idle diesel generator in a warm condition ready to start and accept a load. The heating unit is mounted to heat the engine coolant which circulates by thermosyphon action to the lube oil cooler. A thermostat sensing coolant temperature controls the heating elements to keep the coolant in the oil cooler tank in the proper range.

Upon leaving the engine, the coolant passes through a self-contained thermostatically controlled diverting valve. When the engine is cold, the coolant flow will bypass the heat exchanger completely. When the coolant is hot, the entire coolant flow will be through the heat exchanger. Even though the total coolant flow is constant, the diverting valve will automatically modulate the exchanger bypass in order to maintain the desired engine coolant temperature.

The shell and tube cooling water heat exchanger is cooled by the nuclear service raw water system. The A diesel generator is served by the west nuclear service spray pond and the B diesel generator by the east nuclear service spray pond.

The nuclear service raw water pumps are automatically started on a safety features actuation signal or a diesel engine start signal.

The heat exchanger has a design capacity of 9.55×10^6 Btu per hour with 1100 gpm of coolant flow at 205°F entering and 186°F leaving with a cooling water flow of 1,000 gpm at 82°F entering and 101°F leaving.

With the engine operating at full load, the heat output to the coolant is 9.06×10^6 Btu per hour. With the coolant initially at 190°F and upon starting the engine at full load, the temperature will increase to 205°F within 1 minute.

Additional cooling is provided by the diesel generator room HVAC system as described in Section 9.7.2.2.9.

The main lube oil and piston cooling oil pumps are direct driven.

An a-c motor-driven circulating oil pump that operates when the engine is idle removes residual heat from the turbocharger following shutdown and provides prelubrication for the turbocharger for startup.

D-c motor driven fuel priming and hydraulic governor oil pumps are provided for starting. After the engine is up to speed, direct driven fuel and governor oil pumps take over.

A by-pass lubricating oil pump, which operates only when the engine is idle, is driven by an a-c motor and is located on the accessory rack. This pump draws oil from the engine sump and circulates it through the oil cooler and bypass filter where it is warmed, filtered, and returns to the engine sump.

Each diesel engine has its own separate air-starting system. There are two complete air starting systems for each diesel generator. Each of the two air starter systems consists of an air compressor with all-automatic controls and alarms for a complete unit, two air motors, and three air receivers with a combined air storage capacity sufficient for five starts without compressor operation. Although one air-starting system alone is sufficient to start the diesel engine, in order to provide maximum reliability on starting, both air-starting systems are placed in operation simultaneously. The air start motors will remain in operation for approximately 10 seconds, and if the engine has not started in this time, the controls will go to lockout and an alarm will be given.

8.2.3.1.2.2 TDI Diesel Generator GEA2 and GEB2 Auxiliaries

The nameplate rating of each auxiliary generator is 3500 kW at 0.8 power factor; however, the maximum load will not exceed the "qualified load" of 3300 kW. The units are in separate rooms in the Class 1 Diesel Generator Building. The building is designed for separation between Trains A and B such that the effects of fire, flooding, and moderate energy line breaks in one train will not affect the safe operation of the other train. The building is designed to withstand the effects of tornado and missiles.

The TDI diesel fuel oil storage and transfer system consists of redundant flow trains with each train supplying its associated diesel generator with fuel oil. Each train consists of a 60,000-gallon diesel fuel oil storage tank, two 25-gpm transfer pumps, fuel supply and return piping to the fuel oil day tank

and associated valves, fittings, strainers, and instrumentation. All off-skid fuel oil piping is Seismic Category I, Quality Class 1, and ASME Class 3.

The system provides adequate diesel oil storage to permit 7 days' continuous operation of each standby diesel generator at its full load rating of 3500 kW. The storage tanks have an additional 15 percent fuel oil storage capacity available for routine scheduled operational testing of the diesel generators. The tanks are buried underground and are provided with connections for truck fill, day tank overflow, water drain, dip stick, and pump mounting flanges.

A Class 1 level switch alarms low-low storage tank level in the main Control Room via the plant computer. The low-low level alarm signifies that the usable fuel oil volume will be expended shortly. A non-Class 1 level switch is used for local level indication and high and low level indication at the engine control panel and on the plant computer.

Each diesel fuel oil storage tank is provided with two fuel oil transfer pumps powered from Class 1 motor control centers in their respective train. Each pump is sized to deliver 25 gpm to the engine day tank. Interconnecting piping with two locked closed valves is provided between the two diesel generator systems to allow diesel fuel oil to be transferred from the Train A storage tank to the Train B storage tank and vice versa.

The transfer pumps start automatically on a low level signal from the fuel oil day tank and are stopped by the high level switch in the day tank. The low-level signal and high-level switch are Seismic Category I and Quality Class 1.

A low-pressure alarm of the fuel oil transfer pumps is indicated in the diesel generator control room and plant computer in the main Control Room to alert operators to an abnormal condition such as a pump's failure to start.

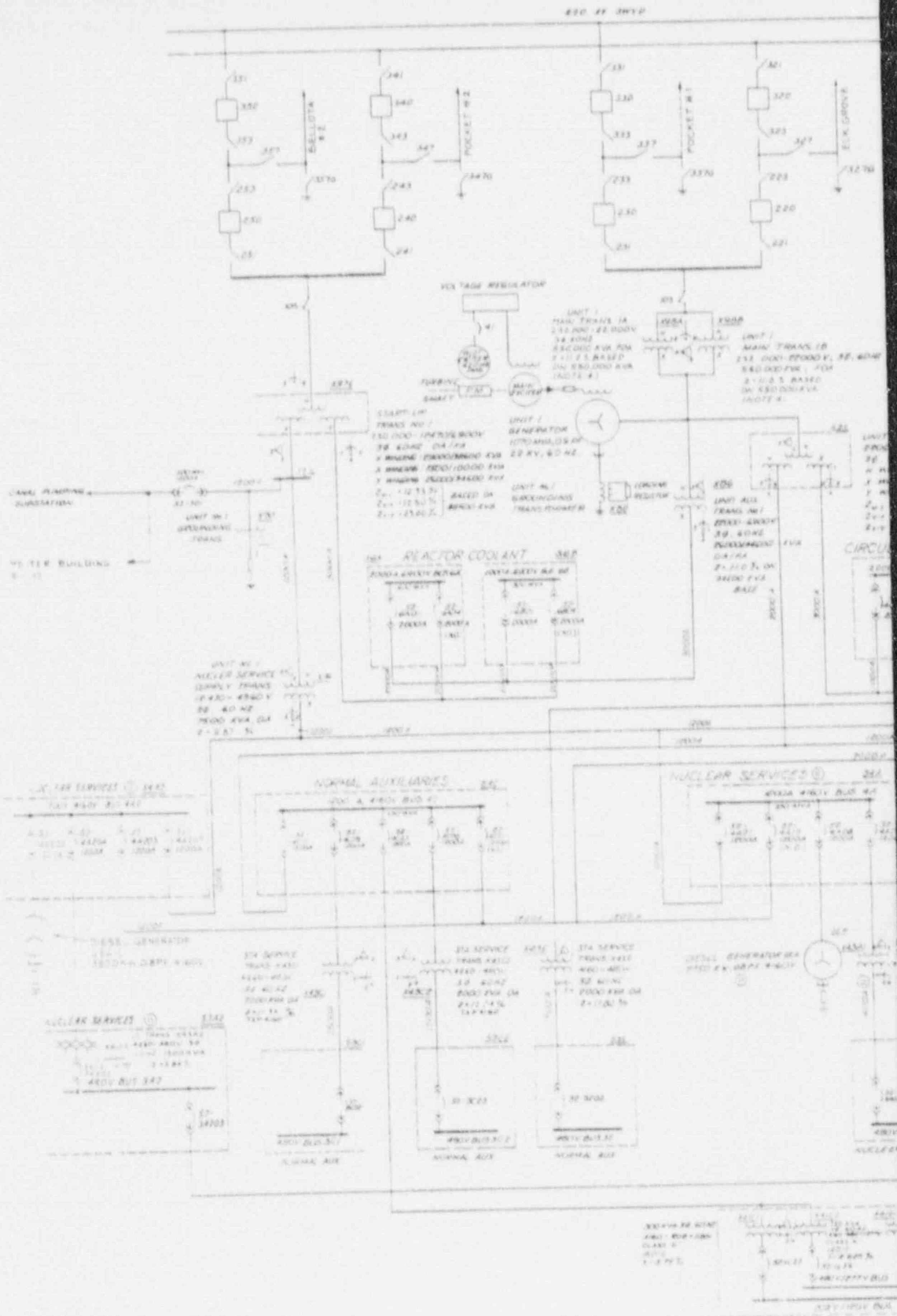
Each day tank has the capacity to store fuel oil for about 2 hours of diesel generator operation at maximum load. Each tank is provided with a drain connection including water draw-off valve, instrument connections, fill connection, and overflow connection to return excess fuel back to the storage tank.

The day tank level alarms are actuated in both the diesel generator control room and the main Control Room. The low-level alarm signifies that the day tank contains fuel for at least 1 hour of diesel generator operation at maximum load.

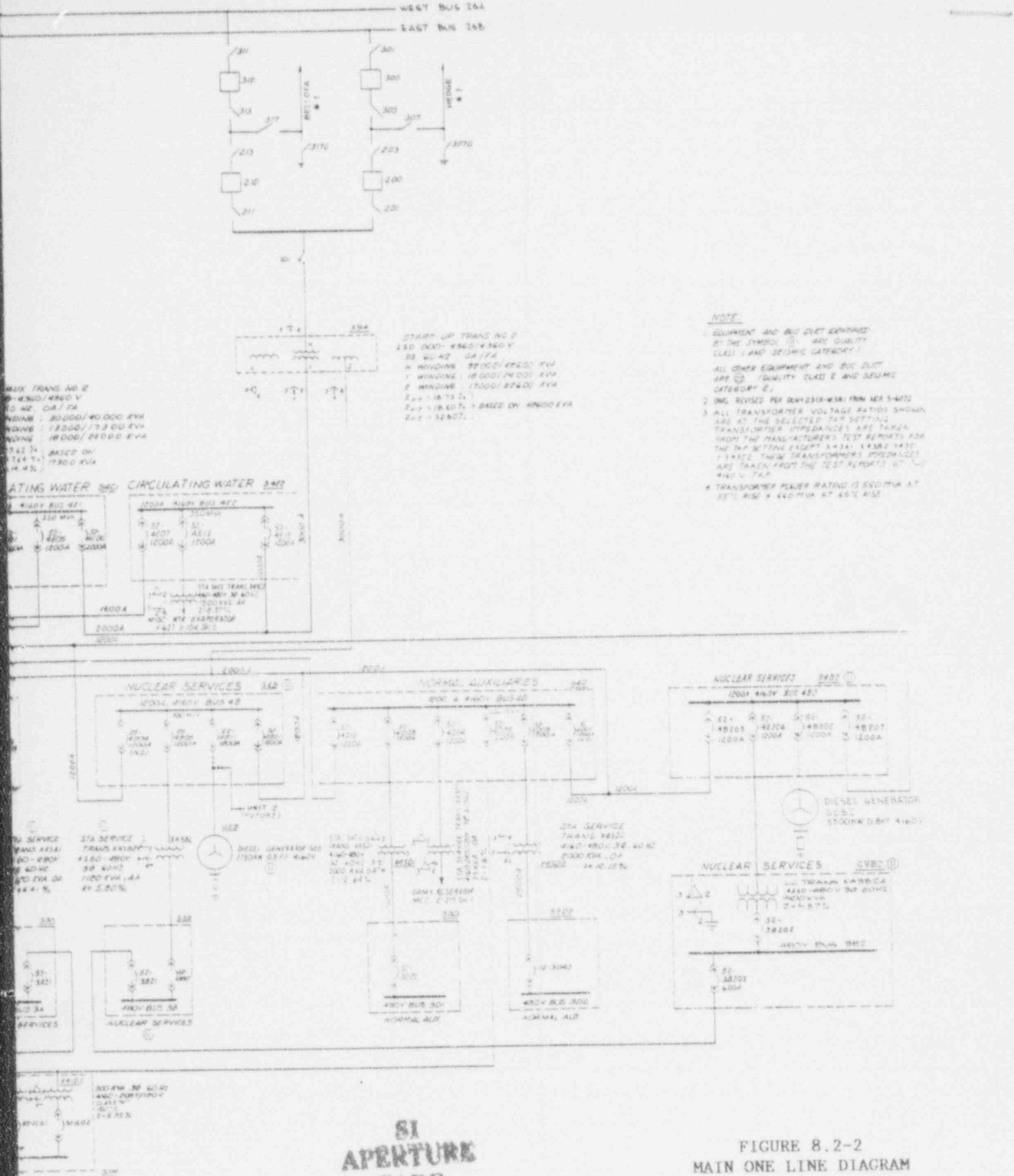
The lube oil system for each diesel generator consists of a lube oil sump tank, a diesel engine-driven lube oil pump, a lube oil cooler, a keep-warm pump, a lube oil heater and associated filters, strainers, pipes, and valves.

When the diesel generator is operating, lube oil is continuously supplied from the lube oil sump tank to the diesel engine inlet via the engine-driven lube oil pump. Warm lube oil is circulated to some of the lube oil system loads by the keep-warm pump when a diesel generator is idle in a standby mode. The lube oil is warmed by a keep-warm heater immersed in the lube oil sump tank. The keep-warm equipment is powered from a non-Class 1 motor control center.

A closed water circulating system with air-cooled radiators is used for cooling the jacket water of the diesel generator units. The radiators are



NOTE:
THIS FIGURE HAS BEEN REVISED BASED ON SMUD DWG.E-101,REV.28.MORE RECENT REVISIONS TO THIS DRAWING MAY BE AVAILABLE AND SHOULD BE UTILIZED FOR ENGINEERING CHANGES.



SI
APERTURE
CARD

Also Available On
Aperture Card

FIGURE 8.2-2
MAIN ONE LINE DIAGRAM

Amendment 8

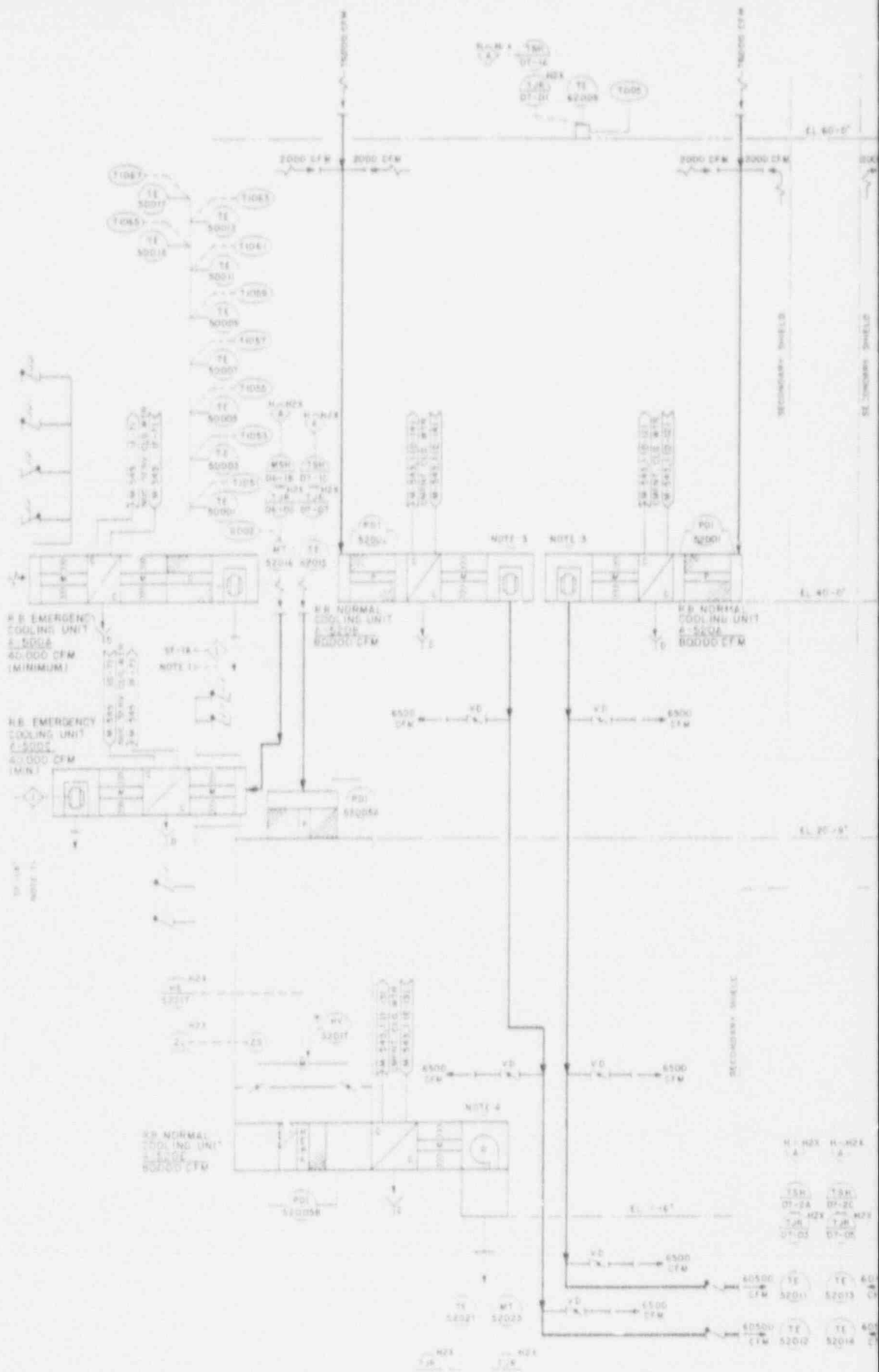
SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

9107030248-03

TABLE OF CONTENTS (Continued)

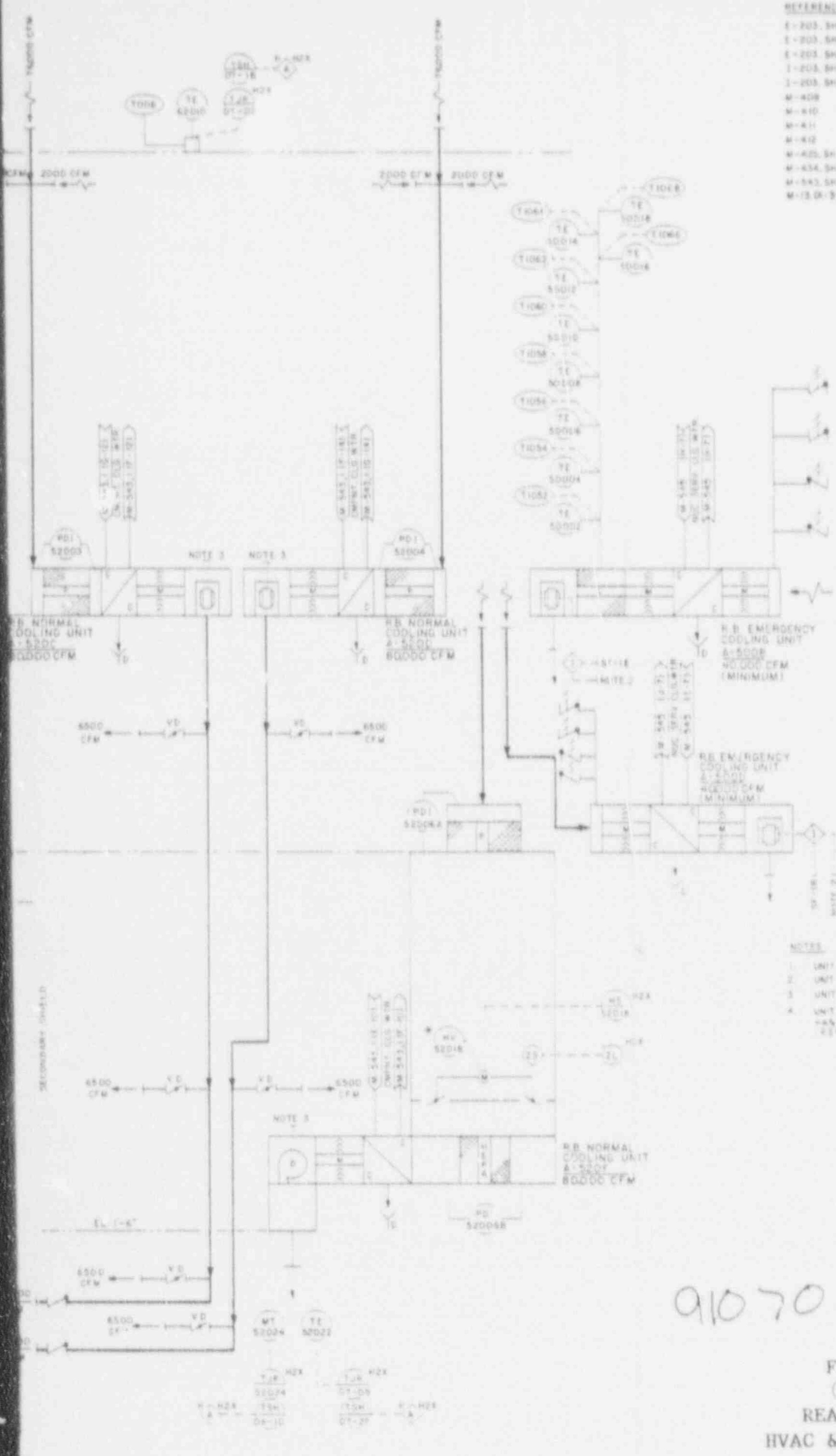
LIST OF TABLES

LIST OF FIGURES



NOTE:

THIS FIGURE HAS BEEN REVISED BASED ON SMUD DWG.M-551,SHEET1,REV.3.MORE RECENT REVISIONS TO THIS DRAWING MAY BE AVAILABLE AND SHOULD BE UTILIZED FOR ENGINEERING CHANGES.



REFERENCE DRAWINGS

E-203, SH 6	ELEM. DIAG - R.B. EMER. CLG. UNITS
E-203, SH 10	ELEM. DIAG - R.B. NORM. CLG. UNITS E & F
E-203, SH 11	ELEM. DIAG - R.B. NORM. CLG. UNITS A,B,C & D
I-203, SH 1	CONT. LOGIC DIAG - R.B. EMER. CLG. UNITS
I-203, SH 14	CONT. LOGIC DIAG - R.B. NORM. CLG. UNITS E & F
M-409	PLAN EL. 40' TO 60 - R.B. VENT. & CLG. SYS.
M-410	PLAN EL. 20'-9" & 1'-6" - R.B. VENT. & CLG. SYS.
M-411	PLAN EL. -125 - R.B. VENT. & CLG. SYS.
M-412	SECT. & DET. - R.B. VENT. & CLG. SYS.
M-425, SH 1	PLAN, SECT. & DET. - AUX. BLDG. HVAC
M-434, SH 1	SCHEDULE - HVAC EQUIPMENT
M-543, SH 1	P & I DIAG - COMPONENT CLG. WTR. SYS.
M-13.01-397	VENDOR MANUAL

NOTES

1. UNIT START/STOP FROM BLPB ON PANEL H2SFA
2. UNIT START/STOP FROM BLPB IN PANEL H2SFB
3. UNIT START/STOP FROM BLPB IN PANEL H2X
4. UNIT START/STOP FROM BLPB ON PANEL H2X HAS BEEN DISCONNECTED AT BREAKER [illegible] (REF. [illegible])

SI APERTURE CARD

Also Available On Aperture Card

9107030248-04

FIGURE 9.7-1
(Sheet 1 of 5)
REACTOR BUILDING
HVAC & COOLING SYSTEMS
Amendment 8

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

9.9 OTHER AUXILIARY SYSTEMS

9.9.1 FIRE PROTECTION SYSTEM

9.9.1.1 Design Bases

The fire protection system is designed to:

- Promptly detect and extinguish a fire should it occur,
- Minimize the potential consequences of a fire, and
- Maintain the capability to safely shut down the plant following a fire.

The fire protection system at Rancho Seco includes the following fire protection features:

- Fire suppression systems
- Fire detection and alarm systems
- Manual fire-fighting equipment
- Fire barriers
- Separation of equipment and cables for post-fire safe shutdown
- Emergency lighting for postfire safe shutdown
- Communications support for postfire safe shutdown

9.9.1.2 Fire Protection Program

The Fire Protection Plan provides a detailed description of the organization, design features, analyses, and management controls that collectively define Rancho Seco's fire protection program. By reference in this USAR, the Rancho Seco fire protection plan is a licensing basis document.

In addition, the following correspondence addresses certain features of the fire protection program in greater detail and are part of the approved Rancho Seco fire protection program.

DATE:	TO/FROM/SUBJECT:
08-31-76	Victor Stello, Jr. from J.J. Mattimoe Response to the guidelines of section II "Positions" of Branch Technical Positions ADSCAR9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," May 1, 1976.
11-16-76	Victor Stello, Jr. from Wm. C. Walbridge Amendment No. 1 to District's submittal of 08-31-76. Additional information in response to verbal questions raised by the NRC's visits on October 19-22, 1976.
01-10-77	Robert W. Reid from J.J. Mattimoe Proposed interim Technical Specification on Fire Protection.
08-01-77	Robert W. Reid from J.J. Mattimoe Provided information regarding Enclosure 2 of NRC letter to SMUD dated 04-15-77.

08-01-77	Robert W. Reid from J.J. Mattimoe Provided Proposed Amendment No. 47, Rev. 1, Technical Specifications on Fire Protection adjusted to NRC standard.
12-08-77	Robert W. Reid from J.J. Mattimoe Reply to August 4, 1977, letter from NRC. Stated we have already complied with "Nuclear Plant Fire Protection Functional Responsibilities." Further review found aspects stated in August 4th document already satisfied.
12-16-77	Karl R. Goller from J.J. Mattimoe Proposed changes to interim Technical Specifications including modifications to some NRC responses.
01-13-78	Vern Rooney from D.G. Raasch Provided drawings for oil catch basins for the reactor coolant pump motors.
01-23-78	Vern Rooney from Bob Daniels and Ron Lawrence (Record of Telephone Conversation) Response to NRC 01-18-78 letter with agreement to install some hydrant valves. Discussed hose stations in Reactor Building and provided criteria for installing fire doors and dampers, etc.
02-01-78	Robert W. Reid from J.J. Mattimoe Provided additional information regarding Fire Hazard Analysis and completion dates for the proposed plant modifications.
02-14-78	J.J. Mattimoe from Robert W. Reid Amendment No. 18 to Facility Operating License No. DPR-54, revised Fire Protection Technical Specifications and associated Safety Evaluation Report.
02-17-78	Robert W. Reid from J.J. Mattimoe Provided additional information regarding Fire Hazard Analysis Report.
02-21-78	Robert W. Reid from J.J. Mattimoe Provided an analysis of a fire water main moderate energy line break in safety related areas.
02-28-78	J.J. Mattimoe from Robert W. Reid Amendment No. 19 to Facility Operating License No. DPR-54, issuance of the Fire Protection Safety Evaluation Report and revision to the Fire Protection Technical Specifications.
03-01-78	Robert W. Reid from J.J. Mattimoe Provided a Report on Fire Stop Tests conducted by the plant and referenced in 08-01-77 transmittal.
06-19-78	Robert W. Reid from J.J. Mattimoe Provided answers to NRC questions concerning Rancho Seco Fire Stop Qualification Test.

06-19-78	Robert W. Reid from J.J. Mattimoe Response to Safety Evaluation regarding detectors to be installed. Also, alternative detectors and design detail.
06-19-78	Robert W. Reid from J.J. Mattimoe Request for deviation from License Amendment 19 in regard to Computer Room fire dampers, Control Room fire extinguishers, and Diesel Generator Room east wall fire resistance.
06-26-78	Victor Stello, Jr. from J.J. Mattimoe Five man fire brigade.
06-29-78	Jerry Zwetzig from Bob Daniels (Record of Telephone Conversation) Discussed letters of 06-19-78.
08-28-78	Robert W. Reid from J.J. Mattimoe Reply to Safety Evaluation request for design details for channel A, C and D conduits in Control/Computer Room; lube oil pump control conduits for high pressure injection pumps; local diesel generator operation; diesel generator fuel oil pumps tripping by CO_2 System; and change in Nuclear Service Raw Water System cross tie.
09-14-78	Jerry Zwetzig from D.G. Raasch Meeting minutes on 08-28-78 letter addressing communications between inside and outside of Reactor Building/pressure sensing on hydrogen line, Reactor Building fire barriers, and modification completion dates.
10-.7-78	Robert W. Reid from J.J. Mattimoe Additional information for Safety Evaluation regarding fire barriers for cable trays, communication from Reactor Building to Control Room, and installing a temperature detector in Reactor Building purge unit.
10-27-78	Jerry Zwetzig from Robert Daniels Additional information on fire protection modifications items including fire stops in the Reactor Building and changes to the Reactor Building purge unit.
11-13-78	Jerry Zwetzig and Robert Daniels Additional information on fire protection including fire detectors, definition of combustibles, and completion dates of modifications.
11-14-78	Robert W. Reid from J.J. Mattimoe States the plant's agreement on fire detectors and installation date.
12-01-78	Robert W. Reid from J.J. Mattimoe Proposed Amendment No. 61, discussed completion of modifications in Safety Evaluation per schedule and the need to extend for three items. Items are Reactor Building fire barriers, radio communications to Reactor Building, and smoke detectors.

12-12-78 Robert W. Reid from J.J. Mattimoe
States small oil filled transfer to be relocated.

12-14-78 J.J. Mattimoe from Robert W. Reid
Amendment No. 25 to Facility Operating License No. DPR-54, Supplement 1 to the Facility Fire Protection Safety Evaluation Report.

01-02-79 Robert W. Reid from J.J. Mattimoe
Additional information on Class II instrumentation circuits required for safe shutdown and cooldown.

03-15-79 Robert W. Reid from J.J. Mattimoe
Additional information on Class II instrumentation.

08-31-79 Robert W. Reid from Wm. C. Walbridge
Request for changes to Safety Evaluation Report to allow other methods to obtain circuit integrity needed for safe shutdown and cooldown.

09-04-79 Robert W. Reid from Wm. C. Walbridge
Request for changes to Safety Evaluation Report to allow other methods to obtain circuit integrity needed for safe shutdown and cooldown.

10-02-79 Darrel G. Eisenhut from J.J. Mattimoe
Technical Specification change agreeing to five man fire brigade.

01-30-80 Robert W. Reid from J.J. Mattimoe
SER changes to show design details of fire barriers and insulated conduits, etc., for high pressure injection and decay heat system.

02-07-80 Robert W. Reid from J.J. Mattimoe
Door alarms are to be provided for doors in walls separating fire areas.

02-08-80 Robert W. Reid from J.J. Mattimoe
SER change for panels above fire doors.

03-03-80 Robert W. Reid from J.J. Mattimoe
Details of design and material for high pressure injection system protection.

03-07-80 Robert W. Reid from J.J. Mattimoe
Modification of fire barriers on trays in Reactor Building.

05-21-80 Dan Garner from D.G. Raasch (Record of Telephone Conversation)
Commented on fire preplan strategies, and included information on reactor coolant pump oil collection system, fire brigade, and fire fighting strategies.

06-19-80 Robert W. Reid from J.J. Mattimoe
Comments on satisfying commitments and Technical Specifications changes.

11-17-80 Darrel G. Eisenhut from W.S. Bossenmaier
Proposed Amendment No. 68, Fire Protection Technical Specifications.

03-17-81 Darrel G. Eisenhut from J.J. Mattimoe
Reassessment of Appendix R's three requirements along with proposed actions.

08-20-81 J.J. Mattimoe from John F. Stolz
Amendment No. 35 to Facility Operating License No. DPR-54, revised Fire Protection Technical Specifications and associated Safety Evaluation Report.

11-19-81 John F. Stolz from J.J. Mattimoe
Raceway layouts and rerouting for Channel B High Pressure Injection Pump.

05-28-82 John F. Stolz from J.J. Mattimoe
Exemption request for Control Room and several other areas.

03-09-83 John F. Stolz from W.K. Latham
Additional information on exemptions.

07-12-83 Darrel G. Eisenhut from R.J. Rodriguez
Supplement to Proposed Amendment No. 83 for Nuclear Service Bus, and fire protection for NSEB.

07-28-83 John F. Stolz from R.J. Rodriguez
Proposed alternation shutdown capability.

11-30-83 John F. Stolz from R.J. Rodriguez
Response to clarification of Generic Letter 81-12.

01-04-84 John F. Stolz from R.J. Rodriguez
Request for exemption from some Appendix R requirements.

02-29-84 John F. Stolz from R.J. Rodriguez
Update on alternate shutdown panel.

07-24-84 Darrel G. Eisenhut from R.J. Rodriguez
Request for exemption from Section III G.2 of Appendix R.

10-29-84 Darrel G. Eisenhut from R.J. Rodriguez
Proposed Amendment 116 to Technical Specification, Fire Barrier Surveillance.

11-15-84 George W. Rivenbark from Kenneth J. Mellor
Additional information about Proposed Amendment 83, Supplement 1, including fire protection and other safe shutdown information.

01-08-85 Darrel G. Eisenhut from R.J. Rodriguez
Proposed Amendment No. 83 Supplement letter.

01-08-85	Darrel G. Eisenhut from R.J. Rodriguez Proposed Amendment No. 83, Supplement 1, Rev. 1 and Supplement 2, Rev. 2.
02-07-85	Darrel G. Eisenhut from R.J. Rodriguez Proposed Amendment 83, Supplement 1, Rev. 2 and Supplement 2, Rev. 2.
02-28-85	Hugh L. Thompson, Jr. from J.J. Mattimoe Request for eight exemptions from Appendix R.
03-21-85	Hugh L. Thompson, Jr. from R.J. Rodriguez Additional information for Proposed Amendment 83, Supplement 1.
04-04-85	Hugh L. Thompson, Jr. from R.J. Rodriguez Formal exemption request for NSEB HVAC units.
04-05-85	Hugh L. Thompson, Jr. from R.J. Rodriguez Clarification of alternate shutdown capability.
04-12-85	Hugh L. Thompson, Jr. from R.J. Rodriguez Submitted preliminary copy of Fire Hazard Analysis Report for NSEB.
04-16-85	Hugh L. Thompson, Jr. from J.J. Mattimoe Supplemental information concerning Proposed Amendment No. 83, Supplement 1.
05-14-85	John Stolz from R.J. Rodriguez Cancelled line-type thermal detectors in cable trays in NSEB.
05-16-85	Hugh L. Thompson, Jr. from J.J. Mattimoe Additional information on NSEB.
05-24-85	Hugh L. Thompson, Jr. from R.J. Rodriguez Request for Appendix R exemption for five areas.
06-04-85	R.J. Rodriguez from Sydney Miner Amendment No. 68 to Facility Operating License No. DPR-54, revised Fire Protection Technical Specifications and associated Safety Evaluation Report.
07-12-85	Hugh L. Thompson, Jr. from R.J. Rodriguez Additional information on alternative shutdown capability.
08-01-85	Hugh L. Thompson, Jr. from R.J. Rodriguez Proposed Amendment No. 137 for Appendix R compliance.
08-02-85	Larry Young from Don Katez (Record of Telephone Conversation) Additional comments on Generic Letter 81-12 response.
08-06-85	Hugh L. Thompson, Jr. from R.J. Rodriguez Revised fire boundaries due to revised FHAR.

09-09-85	R.J. Rodriguez from Sydney Miner Amendment No. 75 to Facility Operating License No. DPR-54, revised fire protection Technical Specifications and associated Safety Evaluation Report.
09-27-85	Hugh L. Thompson, Jr. from R.J. Rodriguez Additional information for response to Generic Letter 81-12.
10-30-85	Hugh L. Thompson, Jr. from R.J. Rodriguez Additional comments on alternate shutdown capability.
11-07-85	Hugh L. Thompson, Jr. from R.J. Rodriguez Revised request for 13 Appendix R exemptions.
11-25-85	Hugh L. Thompson, Jr. from Wm. K. Latham Proposed Amendment No. 137, Rev. 1, including fire protection for Auxiliary Feedwater Pumps.
01-16-86	Frank J. Miraglia, Jr. from R.J. Rodriguez Request for revision to License Amendment 19 for Fire Protection requirements stated in the Facility License, Section 2.C.(4).
02-20-86	Frank J. Miraglia, Jr. from R.J. Rodriguez Proposed Amendment No. 137, Rev. 1, Supplement 1 with analysis.
12-02-86	Frank J. Miraglia, Jr. from John E. Ward Submittal of UFHAR, Rev. 2 to support previous request for an amendment to the fire protection license requirements.
12-24-86	Frank J. Miraglia, Jr. to John E. Ward Proposed Amendment No. 137, Rev. 1, Supplement 2 with analysis.
08-27-87	G. Carl Andognini from George Kalman Amendment No. 85 to Facility Operating License No. DPR-54, revised Fire Protection Technical Specifications and associated Safety Evaluation Report.
11-06-87	G. Carl Andognini from Dennis M. Crutchfield Specific exemptions granted from Appendix R Fire Protection requirements.
01-02-88	J.B. Martin from G. Carl Andognini Attachment details the District's efforts on fire protection.
02-25-88	J.B. Martin from G. Carl Andognini Appendix R compensatory measures taken until modifications complete.
09-16-88	George Knighton from Bob G. Croley Additional information on Fire Hazards Analysis requested by NRC.
10-25-88	George Knighton from Russell B. DeWitt Proposed Amendment No. 173 with analysis.

12-15-88 George Knighton from Bob G. Croley
Appendix R exemption commitment not violated.

06-05-89 Joe Firlit from George Kalman
Amendment No. 107 to Facility Operating License No. DPR-54, revised Fire Protection Technical Specifications and associated Safety Evaluation Report.

06-20-89 Joe Firlit from George Kalman
Amendment No. 111 to Facility Operating License No. DPR-54, revised fire protection Facility License requirements, Section 2.C.(4) and associated Safety Evaluation Report.

9.9.2 EMERGENCY LIGHTING

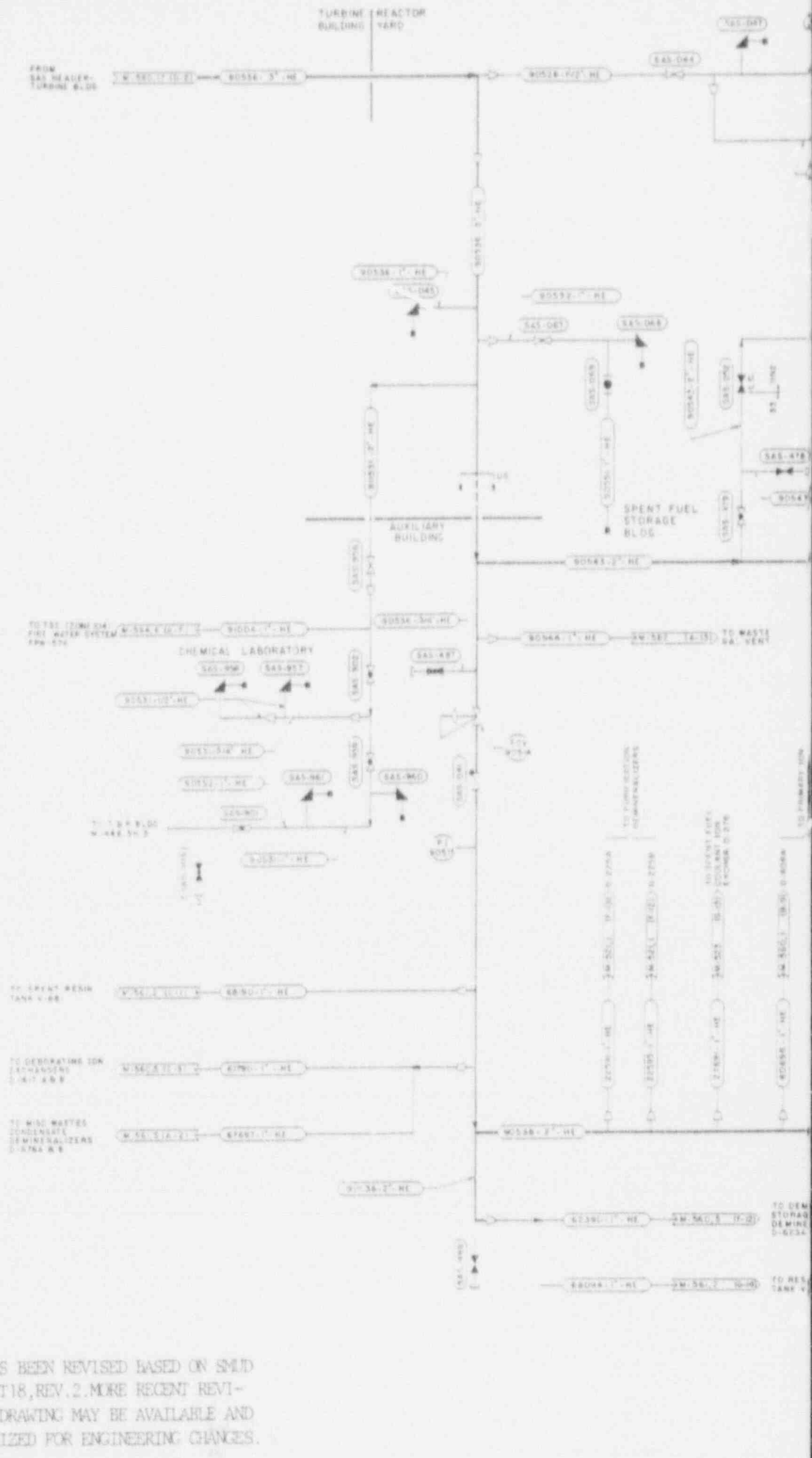
Emergency lighting units with at least an 8 hour battery supply are provided in various strategic areas of the plant to aid access to equipment and components that must be manually operated by plant personnel to effect safe plant shutdown during emergencies. Lighting is also provided in access and egress routes to safe shutdown equipment. The inside and outside emergency battery-operated lights, including those in the Tank Farm area, are energized upon loss of power and de-energized when a-c power is restored. Local switches, where required, are provided for manual initiation in case there is a loss of normal lighting without the loss of a-c power.

9.9.3 COMMUNICATIONS

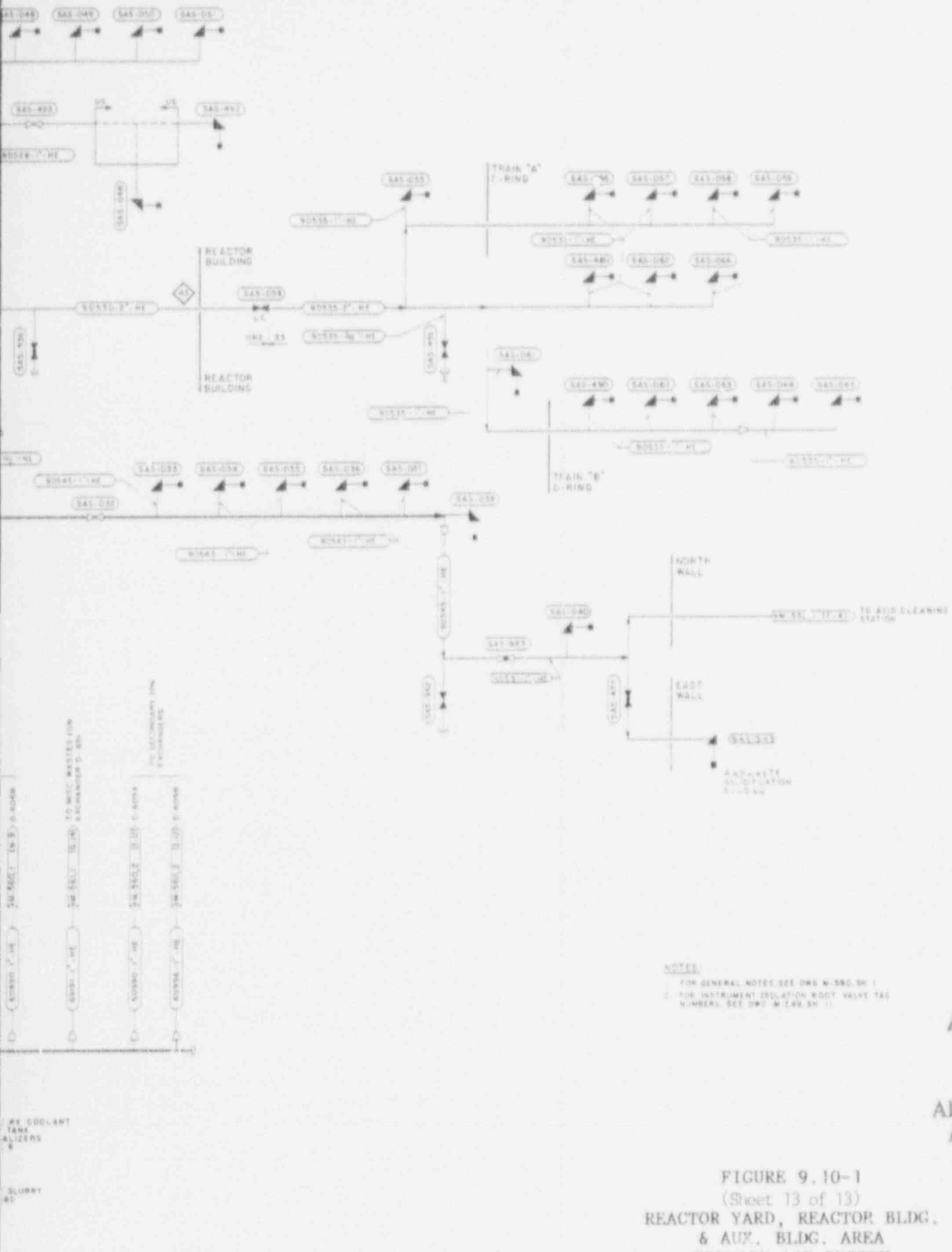
The site communications system provides fast and reliable communication between all control points within the station boundaries, as well as to and from appropriate off-site points. In the event of a major plant accident, strategically placed sirens and horns or flashing visual indicators will alert plant personnel to the emergency condition.

The site communications system encompasses several means of communication. The communication devices available to the Control Room operator, the Emergency Coordinator in the Technical Support Center, the Emergency Manager in the Emergency Operations Facility (EOF), and personnel in other strategic locations in the plant, have been installed to accommodate plant operations and the Emergency Plan. The types of plant communication equipment include: telephones, radios, sound-powered phones, a public address system, an emergency siren system, horns, fire alarms, bells, and flashing lights.

All of the site communication systems are equipped with back-up power systems, with the exception of the Security base radio systems at the Personnel Access Portal.



NOTE:
THIS FIGURE HAS BEEN REVISED BASED ON SMUD DWG.M-590,SHEET18,REV.2.MORE RECENT REVISIONS TO THIS DRAWING MAY BE AVAILABLE AND SHOULD BE UTILIZED FOR ENGINEERING CHANGES.



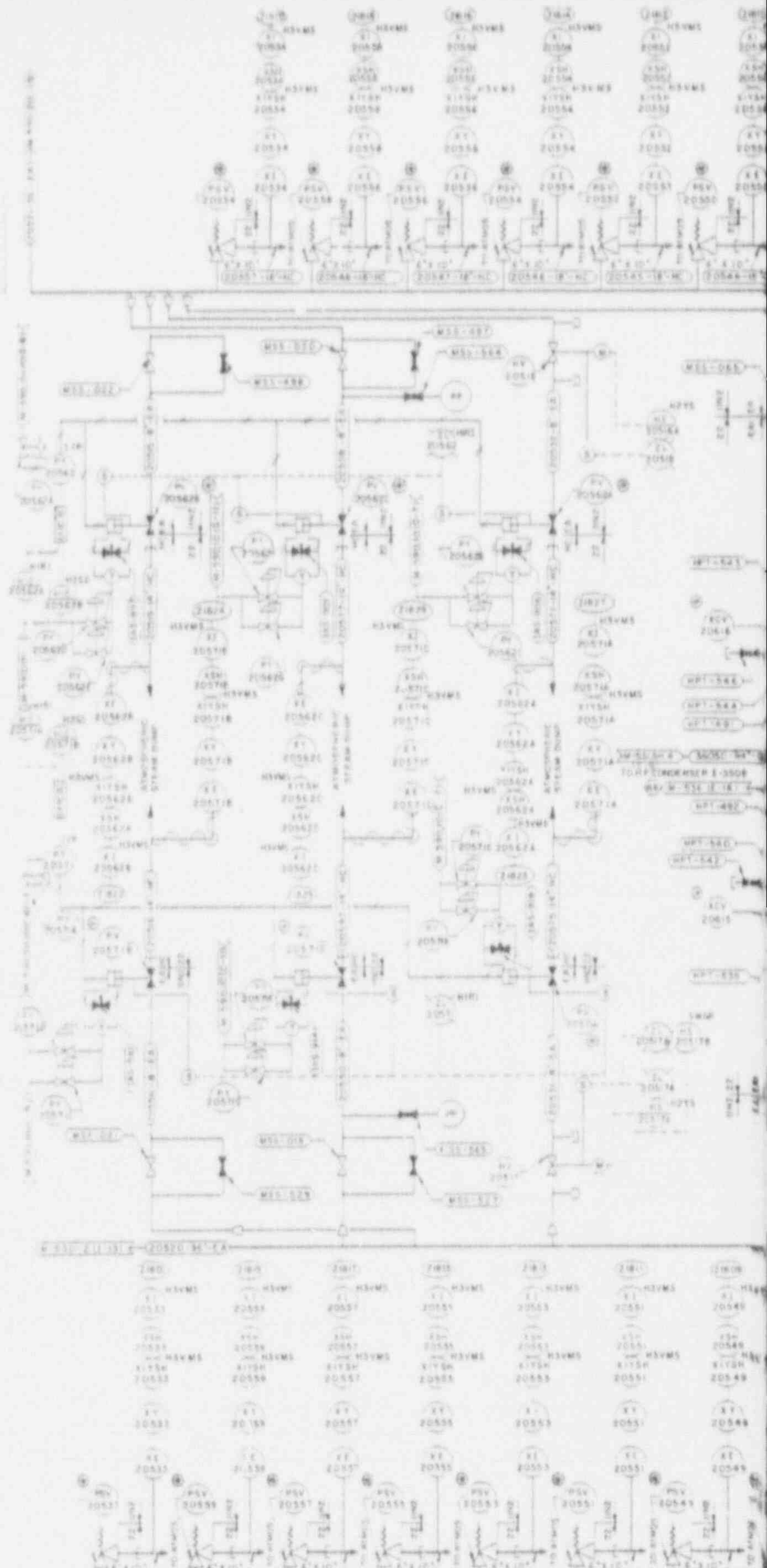
SI
APERTURE
CARD

Also Available On
Aperture Card

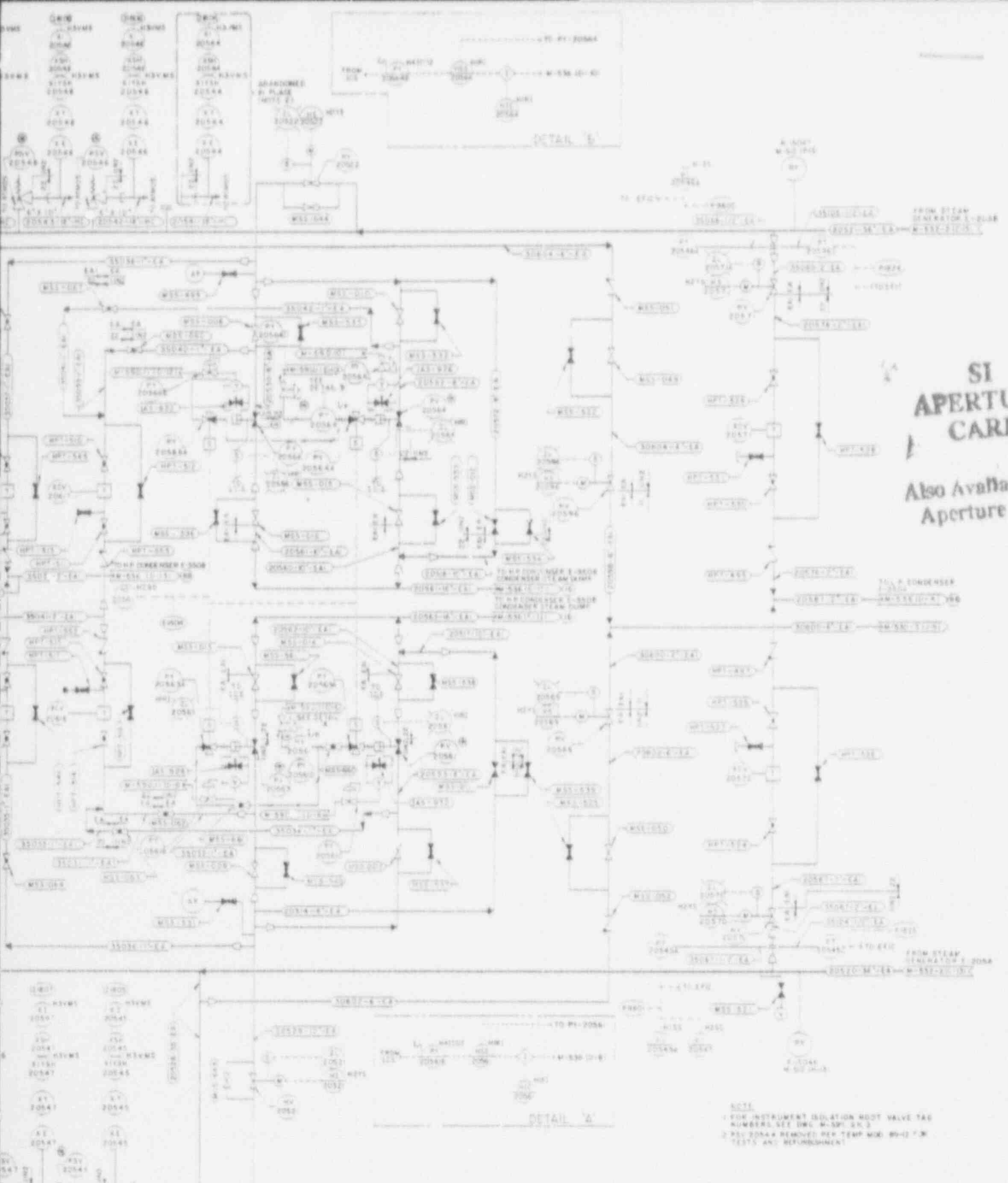
FIGURE 9.10-1
(Sheet 13 of 13)
REACTOR YARD, REACTOR BLDG.
& AUX. BLDG. AREA
SERVICE AIR SYSTEM
Amendment 8

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

9107030248-05



NOTE:
THIS FIGURE HAS BEEN REVISED BASED ON SMUD DWG.M-530,SHEET7A,REV.22.MORE RECENT REVISIONS TO THIS DRAWING MAY BE AVAILABLE AND SHOULD BE UTILIZED FOR ENGINEERING CHANGES.



SI
APERTURE
CARD

Also Available On
Aperture Card

FIGURE 10.2-2
(Sheet 2A of 3)
HIGH PRESSURE & AUXILIARY
TURBINE SYSTEM
Amendment 8

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

9107030248-06

11.2.6 SOLIDIFICATION OF LIQUID RADWASTES AND SOLIDIFICATION OR DEWATERING OF RESINS

Solidification and dewatering are performed by a contractor which has an NRC-submitted topical report which is either pending approval or has been approved which describes their solidification process. Cement, bitumen, or polymers may be used. The solidification unit is specifically designed to optimize solidification of radioactive wastes, evaporator bottoms, filter cartridges, ion exchange resin slurries and sludges. The process results in stable waste forms in accordance with 10 CFR 61.

Process parameters are established by the contractor in a Process Control Program.

The solidification unit is a portable system containing all piping, support, control and monitoring equipment necessary to solidify or dewater radioactive liquids.

The unit is composed of several processing subsystems, each controlling a specific function of the process. These subsystems may include waste transfer, chemical addition, conveyor, and vent and dewatering systems. Control functions for the unit are incorporated into the pneumatic and main control panels. Service supplies are distributed through the service air, water, and electrical distribution systems.

Most of the mobile unit components are arranged in mobile trailers (skids) to provide flexibility of operations.

A closed-circuit television system is an integral part of the mobile unit and allows the operator to monitor the solidification process.

The Solidification/Change Room Structure provides weather protection and containment for radwaste solidification activities and replaced the temporary wooden change room. The solidification portion of the structure is constructed on perimeter curbs to contain possible spills. The structure meets the design criteria of Standard Review Plan Section 11.4, Solid Waste Management Systems, for structures which house portable solidification and/or dewatering systems. The structure includes:

1. Roll-up doors for vehicle access,
2. Doors for personnel access,
3. Automatic sprinklers for fire protection,
4. Ventilation provided by the Auxiliary Building HVAC system,
5. Emergency lighting and service air system supply, and
6. An 8-inch thick concrete shield wall.

The addition of the Solidification/Change Room Structure does not affect the solidification or dewatering process/system described above. No impact on plant operating personnel exposure or gaseous, liquid, or solid radioactive waste releases results from this plant change. This plant change was reviewed and accepted by the PRC.

PROCESS RADIATION MONITORING SYSTEM

Process Radiation Monitor	Sampler-Detector/ Sampling Location	Readout Equipment	Sensitivity	Alarm/Control/Computer Input
Nuclear service cooling water "B" R15010	Off-line liquid sampler with scintillation detector/ samples nuclear service cooling water "B" system flow continuously	Log count ratemeter, $10\text{-}10^6$, CPM, recorded	5×10^{-7} µCi/cc, Cs-137	Alarm on high radiation signal or channel failure
Miscellaneous wastes condensate storage tanks R15015	In-line liquid sampler with scintillation detector/samples liquid discharged from the miscellaneous wastes condensate storage tanks	Log count ratemeter $10\text{-}10^6$ CPM, recorded	5×10^{-7} µCi/cc, Cs-137	Alarm on high radiation signal or channel failure
Circulating cooling water system R15016	Off-line liquid sampler with scintillation detector/samples liquid in the circulating cooling water system continuously	Log count ratemeter $10\text{-}10^6$ CPM, recorded	5×10^{-7} µCi/cc, Cs-137	Alarm on high radiation signal or channel failure/ secures cooling tower blowdown
Spent fuel coolant R15018	Off-line liquid sampler with scintillation detector/ samples spent fuel coolant system flow continuously	Log count ratemeter, $10\text{-}10^6$ CPM, recorded	5×10^{-6} µCi/cc, Cs-137	Alarm on high radiation signal or circuit failure
Letdown flow R15019 R15019A Gross gamma R15019B 2 MEV gamma	In-line liquid sampler with scintillation detector/samples letdown line flow continuously for gross gamma and 2 MEV gamma	Two log count ratemeter-analyzers, $10\text{-}10^6$CPM, recorded	7×10^{-5} µCi/cc, Gross Gamma 1.7×10^{-3} µCi/cc, 2 MEV gamma	Alarms on high radiation signal or channel failure/ level input to computer for both channels
Retention Basin effluent discharge R15017A	Off-line liquid sampler with scintillation detector/samples liquid discharges from the plant	CRT display in Control Room. Local digital display recorder in Computer Room.	MDC is 3.5×10^{-8} µCi/cc, Cs-137	Alarm on high radiation signal or monitor failure/ automatically secures liquid release upon high radiation or channel failure/computer input to IDADS and RM-11
Retention Basin Inlet R15017B	Off-line liquid sampler with scintillation detector/samples mixture of canal and waste water such as cooling tower blowdown and storm drains. Does not sample radioactive water from RHUT.	CRT display in Control Room. Local digital recorder in Computer Room.	MDC is 3.5×10^{-8} µCi/cc, Cs-137	Alarm on high radiation signal or monitor failure/ automatically secures liquid release upon high radiation or channel failure/computer input to IDADS and RM-11

PROCESS RADIATION MONITORING SYSTEM

Process Radiation Monitor	Sampler-Detector/ Sampling Location	Readout Equipment	Sensitivity	Alarm/Control/Computer Input
Waste water discharge (RHUT) R15020	Off-line liquid sampler with scintillation detector	Log count ratemeter, 10-10^6 CPM, recorded	2×10^{-7} μCi/cc, Co-60	None
Reactor Building Atmosphere Leak Detection R15100	Fixed iodine-particulate filters. High and Low range β^- scintillation noble gas detectors/ samples the Reactor Building atmosphere	Log count ratemeter-analyzer 10-10^6 CPM, recorded	5×10^{-8} μCi/cc, Xe-133 (low range) 5×10^{-5} μCi/cc, Xe-133 (high range)	Alarms in Control Room/ computer inputs to IDADS and RM-11
IOSB vent gas R15106	Off-line iodine-particulate continuous air monitor with scintillation detectors/ Isokinetic sampling system monitors IOSB exhaust	CRT display in the Control Room. Local display in IOSB		Alarms in Cont[illegible] Room/ visual alarm i[illegible] [illegible]SB/ automatically secures IOSB exhaust fan on high alarm
Control Room air and ventilation intake R15701 R15702	Off-line gas sampler with scintillation detectors for monitoring beta plus background gamma radiation in Control Room outside air intake duct	10^{-1}-10^6 μCi/cm^3, Display in Control Room and locally	Depends on duct geometry	Alarm on high radiation signal and failure/high radiation activates Control Room/TSC Emergency HVAC system

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

12. CONDUCT OF OPERATIONS

This chapter describes the organization and general plans for operating Rancho Seco. Plant organization is included with brief descriptions of the responsibilities of managers, supervisors, and other key personnel. The training program for the plant staff is described, along with a more general discussion of replacement and retraining plans. Standards and procedures that govern daily operations and the records developed as a result of these operations are also discussed, as are the controls used that promote plant safety and assure compliance with the facility license and the federal and state regulations under which the plant operates.

12.1 ORGANIZATIONAL STRUCTURE OF SMUD

12.1.1 MANAGEMENT AND TECHNICAL SUPPORT ORGANIZATION

The SMUD organization is presented in Figure 12.1-1.

12.1.2 NUCLEAR ORGANIZATION

The SMUD organization and its relationship to the nuclear organization is presented in Figure 12.1-1. The Assistant General Manager (AGM), Nuclear is responsible for all on-site activities in connection with the operations and maintenance of the Rancho Seco Nuclear Generating Station. The AGM, Nuclear reports to the SMUD General Manager, who reports to the District Board of Directors.

12.1.2.1 Plant Organization

The Rancho Seco organization is headed by the AGM, Nuclear who directs the activities of the functional departments and is supported by the Deputy Assistant General Manager (DAGM), Nuclear. The SMUD nuclear organization is shown in Figure 12.1-2.

12.1.2.2 Plant Personnel Responsibilities and Authorities

The responsibilities and authority of major plant positions are summarized below. All plant personnel are selected and trained for their assigned duties, with particular emphasis on the supervisory, technical, and operating staffs to assure safe and efficient operation of the plant. In addition to the responsibilities and authorities stated below, each department manager is responsible for conducting a departmental training program which meets applicable requirements and standards.

12.1.2.2.1 AGM, Nuclear

The AGM, Nuclear is responsible to the SMUD General Manager for the safe and reliable operation of Rancho Seco, and is responsible for the Rancho Seco organization. The AGM, Nuclear assures the safety of plant personnel and the general public, approves and administers the nuclear organization budget, and approves overall scheduling of plant activities and expenditures associated with those activities. The AGM, Nuclear has the authority to establish nuclear organization policy and make commitments to the NRC and is supported by the following personnel:

Deputy AGM, Nuclear (DAGM, Nuclear) - reports to the AGM, Nuclear. The DAGM, Nuclear functions as the AGM, Nuclear with fully delegated authority when the AGM, Nuclear is not on site. Otherwise, the DAGM, Nuclear is responsible for providing management oversight of nuclear plant administrative functions such as:

1. Cost Control,
2. Commitment Management,
3. Audit/Inquiry Responses,
4. District Representative/Negotiator,
5. Resource/Budget Management, and
6. Nuclear Policy and Procedure Management.

Manager, Nuclear Quality Assurance - ensures that Quality, Licensing, and Administrative Programs are implemented in accordance with regulatory requirements. The manager is independent of the pressures of plant operations and has sufficient authority and organizational freedom to identify problems that affect quality, recommend solutions, and verify implementation of solutions.

The Manager, Nuclear Quality Assurance is responsible for the overall administration of the Rancho Seco Quality Assurance Programs. Areas of functional responsibility include quality auditing, quality control, quality engineering, and the site Corrective Action Program.

In the Licensing area, the Manager, Nuclear Quality Assurance provides regulatory guidance as well as compliance and licensing services to maintain the Rancho Seco facility license. The manager ensures compliance with regulatory requirements and commitments, controls and coordinates the interface and correspondence with the NRC, and is responsible for updating licensing basis documents, maintenance and interpretations of Technical Specifications, coordinating and managing District commitments with regulatory agencies.

In the Administrative area, the Manager, Nuclear Quality Assurance provides centralized administrative services and support, including document control, records management, and office services for the entire nuclear organization. The manager also establishes policies and direction within the Administrative area to support the goals and objectives established by the AGM, Nuclear for Rancho Seco.

Nuclear Operations/Security Manager - ensures plant operations are conducted in accordance with the requirements of the operating license, Technical Specifications, plant operating procedures, and applicable state and federal regulations. Details regarding operating shift crews are discussed in Section 12.1.2.3. The manager manages the activities of plant operations, operator training, security, and security training.

The Nuclear Operations/Security Manager develops, schedules, and conducts training programs for licensed and non-licensed operators. The manager achieves and maintains prescribed training programs and ensures that they satisfy regulatory standards. Also, the Nuclear Operations/Security Manager manages the activities of Rancho Seco Security during routine, emergency, and contingency operations. The manager develops and maintains security related licensing documents, and implements procedures to ensure compliance with site, local, state, and federal security related requirements and regulations.

Nuclear Radiation Protection/Environmental Monitoring/Emergency Preparedness/Chemistry Manager (Nuclear RP/EM/EP/CHEM Manager) - is responsible for:

1. Minimizing employee and public exposure to radioactivity,
2. Maintaining a personnel monitoring and recordkeeping program,
3. Ensuring compliance with regulatory requirements regarding radiation protection and radwaste management,
4. Establishing and evaluating the content and effectiveness of radiation technician training,
5. Developing, maintaining, and implementing the ALARA program,
6. Ensuring the Radiological Environmental Monitoring Program is implemented in accordance with regulatory requirements,
7. Developing, implementing, and maintaining the Rancho Seco Emergency Plan (including implementing procedures),
8. Managing the plant chemistry program which establishes chemistry and radiochemistry controls and surveillances of plant fluid systems, and develops and implements limits and guidelines to minimize system corrosion and loss of heat transfer characteristics,
9. Controlling and monitoring radioactive liquid and gaseous releases,
10. Establishing and evaluating the content and effectiveness of chemistry technician training, and
11. Providing General Employee Training (GET) and Fire Brigade training.

Manager, Nuclear Maintenance - maintains the physical condition of the plant through inspections and preventive and corrective maintenance to optimize reliability of systems and components, and directs modification activities. The manager also provides nuclear organization support in the area of scheduling.

Manager, Nuclear Technical Services - provides operations and maintenance support, plant closure and decommissioning project management oversight, support and technical direction in plant design, design modifications to plant systems, design specification development, design change control, configuration management, and discipline engineering. The manager is responsible for system engineering, performance monitoring, surveillance testing, including inservice testing and inspection, reactor engineering and welding, fire protection, nuclear fuels management, instrumentation and controls engineering, and maintains the design baseline documents, Master Equipment List, and other documents defining technical requirements of systems, structures, and components to yield a safe and reliable plant design.

12.1.2.3 Operating Shift Crews

The minimum shift crew composition is an eight-person complement consisting of two senior licensed operators, two licensed operators, three non-licensed operators, and an Operations Technical Advisor. This minimum shift complement is required for all operating modes above cold shutdown or refueling shutdown.

When the reactor is in cold or refueling shutdown, the minimum shift complement is four, consisting of one senior licensed operator, two licensed operators, and one non-licensed operator. In addition, any operation involving core alteration will be supervised by a Senior Reactor Operator or Senior Reactor Operator limited to Fuel Handling who has no other concurrent responsibilities during this operation.

In the defueled condition, the minimum shift complement is three, consisting of one Senior Reactor Operator, one Reactor Operator, and one Non-Licensed Operator. This shift staffing level meets the requirements of 10 CFR 50.54(m) for a facility with a single, non-operating unit.

The duties of the Nuclear Operations/Security Manager are discussed in Section 12.1.2.2.1. The responsibilities and authorities of shift operations personnel are described below.

Shift Supervisor - is accountable for safe and efficient plant operation in accordance with Technical Specifications, federal/state regulations, and plant procedures. The supervisor has the authority to terminate any site activity judged to be a public, personnel, or station hazard and is present in the Control Room during major evolutions.

Assistant Shift Supervisor - assists the Shift Supervisor in the execution of duties.

Control Room Operator - operates the reactor and secondary plant in accordance with Technical Specifications and operating procedures, and maintains a chronological record of plant activities.

Non-Licensed Operator - monitors and operates plant equipment and systems in support of station operation, and checks, analyzes and logs equipment/system parameters and initiates corrective action when abnormal conditions exist.

Operations Technical Advisor - provides technical support to the Shift Supervisor, and performs accident and operational safety assessment functions.

The operations personnel have extensive training and experience in nuclear power operation. The operations personnel are engaged in a continual retraining program as described in Section 12.2 to assure the continued safe and efficient operation of the Plant. The training program for replacement personnel is outlined in Section 12.2.

A site Fire Brigade is maintained on-site at all times as required by the Fire Protection Plan.

Personnel are on shift who are trained and qualified to implement radiation protection procedures, including routine and special radiation surveys using portable radiation detectors, use of protective barriers and signs, use of protective clothing and breathing apparatus, performance of contamination surveys, checks on radiation monitors, and limiting radiation exposure and accumulated dose.

12.1.2.4 Succession of Responsibility

The succession to responsibility for operation of the plant in the event of absences, incapacitation of personnel, or other emergencies is as follows:

AGM, Nuclear
Deputy AGM, Nuclear
Nuclear Operations/Security Manager
Shift Operations Superintendent
Shift Supervisor

12.1.3 QUALIFICATIONS OF NUCLEAR PLANT PERSONNEL

Each member of the plant staff meets or exceeds the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the RP/EM/EP/CHEM Manager, who meets or exceeds the qualifications of Regulatory Guide 1.8, Revision 2 for the Radiation Protection Manager and (2) the Operations Technical Advisor (OTA), who has a bachelor's degree or equivalent in a scientific or engineering discipline. The OTA receives specific training in plant design and response and analysis of the plant for transients and accidents. All plant personnel are selected and trained for their assigned duties, with particular emphasis on the supervisory, technical, and operating staffs, to assure safe and efficient operation of the plant. Figure 12.1-2 shows the SMUD nuclear organization.

Personnel selection policy for the Rancho Seco staff includes reference checks, motor vehicle driver's check, and a review of each application for employment with particular emphasis on arrest record, previous employment and reason for leaving, and military service record.

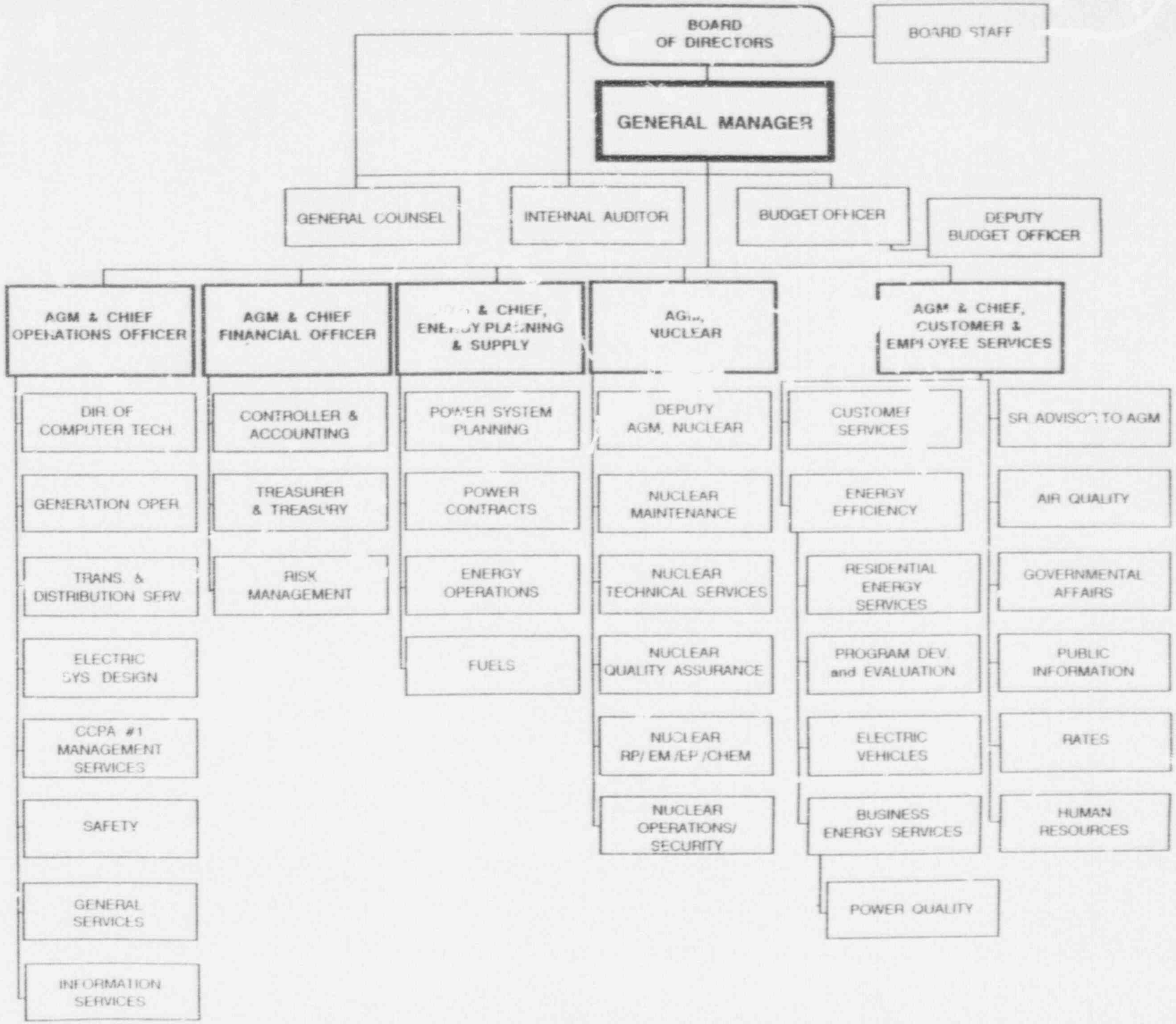


FIGURE 12.1-1
SMUD CORPORATE ORGANIZATION
Amendment 8

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

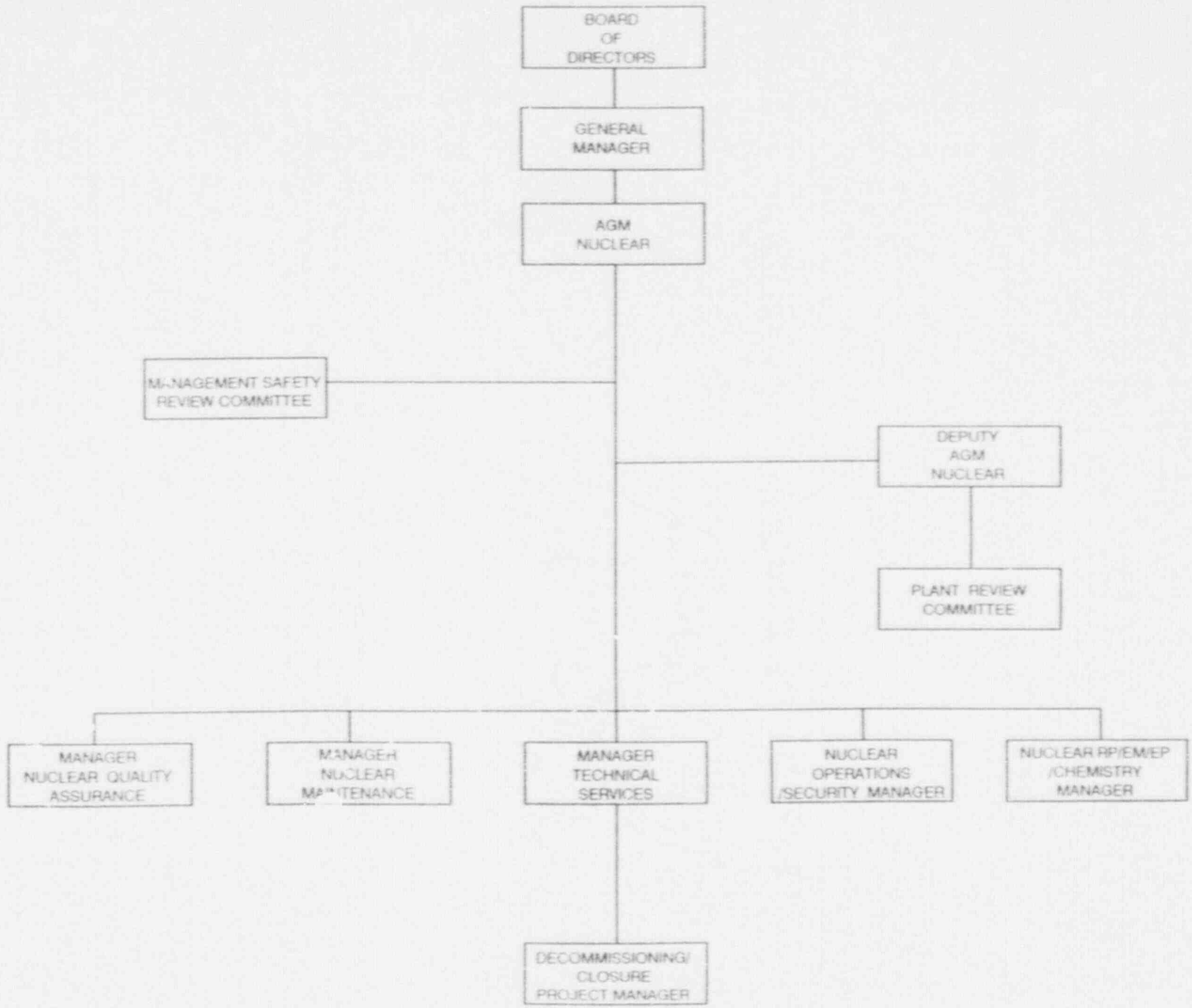


FIGURE 12.1-2
SMUD NUCLEAR ORGANIZATION

Amendment 8

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

12.2 PERSONNEL TRAINING

Retraining and Replacement Training Programs for the operating staff are maintained under the direction of the Nuclear Operations/Security Manager and are conducted in accordance with plant procedures. Retraining and Replacement Training meets or exceeds the requirements and recommendations of ANSI N18.1-1971, Section 5.5 and 10 CFR 55.

Each department manager is responsible for conducting a departmental training program that meets the applicable requirements and standards, including testing individuals as appropriate and maintaining training documentation within areas of responsibility.

Rancho Seco security force training is the responsibility of the Nuclear Operations/Security Manager.

12.2.1 TRAINING PROGRAMS

The following descriptions outline the training program guidelines that govern personnel training at Rancho Seco.

The following is a list of training programs that comprise most of the discipline related training programs at Rancho Seco:

1. Senior Reactor Operator/Shift Supervisor Training
2. Reactor Operator Training
3. Non-Licensed Operator Training
4. Licensed Operator Requalification Training
5. Operations Technical Advisor Training
6. Chemistry Technician Training
7. Radiological Protection Technician Training
8. Maintenance Training
9. Licensing Training

In addition, training programs required by the Emergency Plan, Physical Security Plan, Fire Protection Plan, Technical Specifications, or administrative requirements are as follows:

1. General Employee Training (GET)
2. First Aid
3. Fire Brigade, Fire Protection
4. Emergency Plan
5. Security
6. Quality Assurance

7. Radwaste Handler

8. Dosimetry Technician

9. ALARA

10. Safety

All personnel working at Rancho Seco participate in the training programs required for their job position. All training is conducted and documented in accordance with departmental training procedures.

12 2.2 LICENSED OPERATOR AND SENIOR OPERATOR REPLACEMENT TRAINING PROGRAMS

Training of personnel for NRC license is conducted in accordance with Operations Department Training Procedures. These programs are designed to prepare the trainee for safe and efficient operation of the plant and for NRC Licensing as a Reactor Operator, Senior Reactor Operator or Instant Senior Reactor Operator.

Instructors who teach systems, integrated responses, transient and simulator courses shall demonstrate their competence by successful completion of an NRC Senior Operator exam for a PWR, or by successful completion of a Senior Reactor Operator Certified Instructor training program. These instructors shall participate in the Requalification Training Program and shall take exams as if they were a Rancho Seco Licensed Operator, with the following exception: certified instructors are only required to attend one week of simulator training annually as either instructors or students.

12.2.2.1 Structure of Replacement Programs

The Replacement Operator Training Programs are divided into three basic phases. They are the Classroom Phase, the Self Study Phase, and the Job Training Phase.

Classroom Phase

The Classroom Phase of the training program consists of formal classroom presentations in fundamentals, related theoretical subjects, plant systems, integrated operations, and procedures. The course content and structure is in accordance with the systems approach to training.

Self Study Phase

The Self Study Phase of the training program is designed to allow students to gain an in-depth knowledge of systems and procedures. Each student is issued a Training Manual which contains study references and check-out sheets for self-evaluation, ensures accomplishment of study objectives, and provides the documentation of successful completion of check-outs. When all topics on a particular checklist have been completed, an Operating examination/quiz is administered by a Training Instructor, Licensed Operator or a Senior Licensed Operator. Satisfactory completion of all checklists is required by each student. The Self Study Phase contains three modules: Systems Training, Procedure Training and Miscellaneous Materials Training.

Job Training Phase

The Job Training Phase of the training program is designed to provide the opportunity for the trainee to gain a "feel" for plant controls during control manipulations and to develop a greater sense of expected plant response under varying conditions. It also allows the trainee to develop an understanding of surveillance requirements, procedures, and testing requirements, and gain additional control room operating experiences. The Job Training Phase consists of two parts: the Simulator Training module (which includes the Reactor Startup Certification), and an On-The-Job Training module.

12.2.2.2 Evaluation

End of course evaluation for the Reactor Operator and Senior Reactor Operator replacement training consists of two separate examinations.

1. Audit Exam

   An evaluation consisting of a written examination and an oral/walkthrough is performed to determine competence level of the students.

2. NRC License Exam

   Following successful completion of the site training program, candidates are evaluated by the NRC License Examiners to determine eligibility for an NRC issued Reactor Operator or Senior Reactor Operator License.

12.2.3 LICENSED OPERATOR REQUALIFICATION TRAINING PROGRAM

The Licensed Operator Requalification Training Program is used to maintain an acceptable level of competence for Licensed Reactor Operators, Senior Reactor Operators and Senior Reactor Operator Certified Instructors. It is conducted in accordance with approved Operations Department Training Procedures, and follows the systems approach to training.

12.2.3.1 Structure of Requalification Program

The Licensed Operator Requalification Training Program is conducted biennially* with successive requalification programs using the same format and schedule with no time lapse between programs. It is divided into three basic phases: the Classroom Phase, the Self Study Phase, and the Job Training Phase. The components of each phase, with a brief description of each, are listed below.

* Biennial, as used in the requalification procedures, is defined as 24 calendar months, but may not be less than 21 months and may not exceed 27 months, in order to accommodate plant operations that might delay/interfere with the conduct of the program.

Classroom Phase

The Classroom Phase contains the following courses:

1. Fundamentals Review

   Consists of classroom instruction on Radiological Protection, Reactor Theory, and Heat Transfer and Fluid Flow. The instruction also reviews theory and principles of reactor operation.

2. System and Procedures Review

   Consists of classroom instruction on plant system design, construction, and integrated operation of the plant. The instruction also reviews major plant systems, operating and emergency procedures, and the Emergency Plan.

3. Plant Modification and Operational Assessments

   Consists of classroom instruction on recent changes in plant construction and operation, and events which have occurred at Rancho Seco and at other stations which are pertinent to plant operations.

In addition, specific topical areas will, in part, be determined by the results of the annual requalification examinations and plant operating history.

Self Study Phase

The Self Study Phase consists of reading assignments. Plant modifications, procedure changes, operational assessments and operational events may be contained in the reading assignments.

Job Training Phase

The Job Training Phase consists of 2 weeks per year of simulator training consisting of no less than 60 hours of actual simulator operations. This training allows the operator to perform control manipulations, reactivity changes, startups, shutdowns, etc., not conducted at the plant. The training provides for operator response to abnormal, emergency, accident and transient conditions.

12.2.3.2 Evaluation

Written Examination

The requalification program includes an evaluation and observation system to obtain maximum benefits from the retraining program and provide a method to determine areas in which retraining is needed.

Written examinations are administered in accordance with the guidelines of 10 CFR 55 and Operations Department Training Procedures. Written exams are used as a method to determine licensed operator and senior operator knowledge of subjects covered in the requalification program. They help determine areas in which retraining is needed to upgrade licensed operator and senior operator knowledge.

Operating Examination

An operating examination shall be administered to each licensed operator by the Operations Department Supervisor in charge of Licensed Operator Training or his/her designated representative. This examination shall, as a minimum, consist of:

1. A discussion of required actions during normal and off-normal conditions.

2. Simulation of required actions during normal and off-normal conditions at a simulator or during a Control Room walkthrough.

12.2.4 OPERATIONS TECHNICAL ADVISOR TRAINING PROGRAM

Initial Training Program

The Operations Technical Advisor (OTA) Training Program is conducted in accordance with Training Department Procedures. Personnel with engineering or scientific degrees are provided site specific training in transient and accident analysis, mitigating core damage, systems, emergency procedures, off-normal procedures, plant operations, simulator, and administrative procedures. This training program closely parallels the Reactor Operator and Senior Reactor Operator training programs.

Requalification Training Program

OTAs participate in the Licensed Operator Requalification Program. In addition to their training on the simulator as a crew member, they also receive an additional 20 hours of simulator training specifically designed to assist them in their diagnostic and teamwork abilities.

12.2.5 LICENSING TRAINING PROGRAM

The intent of the Licensing Training Program is to assist personnel in adapting their technical expertise to the performance of various tasks and administrative processes related to the facility License, such as 10 CFR 50.59 and nuclear safety.

The goal of the program is to assure quality performance of processes and tasks and is accomplished by cross-training to ensure personnel have the requisite knowledge and skill to perform satisfactorily.

Each participating department determines positions requiring training under this Program. The Manager, Nuclear Quality Assurance is responsible for administering the Licensing Training Program to the designated individuals.

12.2.6 MAINTENANCE TRAINING PROGRAM

Training of personnel in the Maintenance disciplines is conducted in accordance with Maintenance Department Training Procedures. This program is designed to prepare the trainee for safe and efficient maintenance/operation of Rancho Seco.

The Manager, Nuclear Maintenance is responsible for implementing the Maintenance Training Program. The Maintenance Training Program consists of Administrative Process Training and On-The-Job Training (OJT). Administrative Process Training is based on developing and maintaining the prerequisite skills and knowledge required by maintenance workers to accomplish specific maintenance tasks. These tasks include, but are not limited to, handling Work Requests (WRs), Potential Deviations from Quality (PDQs), ignition source permits, clearance/test tags, scaffolding requests, the plant drawing system, and the vendor manual system. OJT consists of, but is not limited to, task training and evaluation, procedure training, and specific discipline-related training requirements.

Initial Training

Initial training is designed to provide the trainee with knowledge and skills necessary to function as part of the Maintenance Department at Rancho Seco and is typically completed in 2 years.

Continuing Training

Continuing training is designed to reinforce and improve knowledge and skills of plant employees and long term contract personnel. Continuing training is conducted on a 2 year cycle.

12.2.7 SITE SUPPORT TRAINING PROGRAMS

Site Support Training is conducted by the department whose area of responsibility covers a particular Site Support function. Training of personnel in the Site Support disciplines is conducted in accordance with the applicable department's training procedures. These programs are designed to prepare the trainee for safe and efficient maintenance/operation of the Rancho Seco Plant.

Site Support Training includes training programs for the following areas:

- Chemistry Technician
- Dosimetry Technician
- Radwaste Handler
- Radiation Protection Technician
- General Employee Training
- Fire Protection/First Aid

Initial Training

The responsibility for each Site Support Training Program is assigned to the applicable department manager. Classroom and laboratory training are provided by the responsible department when appropriate or necessary. OJT is provided within each discipline. OJT consists of, but is not limited to, task training and evaluation, procedure training, and specific discipline-related training requirements.

12.3 EMERGENCY PLANNING

The Emergency Plan provides a description of the organization, equipment and preparations made to enable rapid and effective response to any emergency situation at Rancho Seco. At all times, the primary concern is the protection of plant personnel and the surrounding population, with due regard for maintaining plant integrity.

Emergency activities at Rancho Seco are the responsibility of SMUD management. Offsite emergency activities are under the authority of public agencies, with SMUD support being directed by the Emergency Coordinator.

A set of Emergency Plan Implementing Procedures provide specific actions to be taken when responding to various types of emergencies. These procedures also provide data, details and instructions, personnel assignments, criteria for site evacuation, other specific information which would be needed during an emergency, and instructions to obtain names and telephone numbers for emergency call-out.

The Emergency Plan and Implementing Procedures form a complete detailed program which aids Rancho Seco personnel and affected offsite agencies in the safe and efficient handling of emergency conditions. The emergency conditions considered in the development of this plan for an operating reactor included the postulated accidents discussed in USAR Section 14.3 as well as postulated non-operational accidents. The current NRC approved Emergency Plan reflects the defueled condition of the plant and addresses the USAR Section 14.3 accidents considered credible in the defueled condition that have dose consequences.

The Rancho Seco Nuclear Generating Station Emergency Plan was originally submitted on the docket in the FSAR. During commercial operation of the plant, the Emergency Plan was revised to reflect changed procedures or additional NRC guidance. The Emergency Plan implemented during commercial operation conformed as fully as practicable to the standards in 10 CFR 50.47(b) and 10 CFR 50, Appendix E. The Emergency Plan implemented in the defueled condition reflects NRC granted exemptions from several standards in 10 CFR 50.47(b), 10 CFR 50, Appendix E, and 10 CFR 50.54(q). The defueled condition Emergency Plan conforms as fully as practicable to the emergency planning requirements applicable to Rancho Seco in the defueled condition.

12.5 PLANT PROCEDURES AND PROCESS STANDARDS

12.5.1 PROCEDURES

The performance of work at and operation of Rancho Seco are both guided by procedures. All safety-related operations are conducted in accordance with detailed written procedures.

12.5.1.1 Conformance with Safety Guide 33

The procedures discussed in this section cover the activities referenced in the following documents:

1. Applicable procedures recommended in Appendix A of Safety Guide 33, November 1972
2. Rancho Seco Technical Specifications
3. Surveillance and test activities of safety-related equipment
4. Security Plan
5. Emergency Plan
6. Fire Protection Plan

12.5.1.2 Preparation of Procedures

Procedures are divided into hierarchy levels. These include:

1. The Rancho Seco Quality Manual
2. Rancho Seco Administrative Procedures
3. Departmental Administrative Procedures
4. Technical Procedures

The Rancho Seco Administrative Procedures (RSAPs) define and implement administrative requirements or activities involving interdepartment processes and administrative responsibilities.

Departmental Administrative Procedures define administrative requirements, activities, or actions specific to one department within the nuclear organization. In addition, Departmental Administrative Procedures may direct activities of other Departments when necessary for support.

Procedure hierarchy, preparation, review, approval, and control is in accordance with RSAPs.

Those plant procedures related to nuclear safety or required by the Technical Specifications are reviewed by the Plant Review Committee (PRC) and approved by Plant Management in accordance with Rancho Seco Administrative Procedures. Some procedures are also reviewed by the Management Safety Review Committee (MSRC).

12.5.1.2.1 Procedure Changes

Plant procedures are reviewed periodically as required by Technical Specifications and RSAPs. Permanent procedure changes are reviewed and approved in accordance with RSAPs. Temporary procedure changes may be made provided:

1. The intent of the original procedure is not altered.

2. The change is approved by two members of plant management staff, at least one of whom holds a Senior Reactor Operator's License for Rancho Seco.

3. The change is documented, reviewed, and approved as required by Technical Specifications.

4. In cases of emergency, personnel may authorize departure from approved procedures in accordance with 10 CFR 50.54(x), where necessary, to prevent injury to personnel or damage to the facility. Records of such departures will be logged, indicating the prevailing conditions/situation and the reason for the action(s) taken.

12.5.1.3 <u>Conduct of Operations</u>

12.5.1.3.1 Reactor Operator's Authority and Responsibility

The Reactor Operator is given the authority to manipulate controls which directly or indirectly affect core reactivity, including a reactor trip if deemed necessary. He/she is also assigned the responsibility for knowing the limits and setpoints associated with safety-related equipment and systems as specified in the Technical Specifications and designated in procedures.

12.5.1.3.2 Senior Reactor Operator's Authority and Responsibility

The Senior Reactor Operator, in addition to the authorities and responsibilities described for the Reactor Operator, is given the authority to direct the licensed activities of the Reactor Operator, and ultimately is held responsible for all licensed activities at the plant that are within his/her control.

12.5.1.3.3 Activities Affecting Plant Operation or Operating Indications

All plant personnel performing functions that may affect plant operation or Control Room indications are required to notify the Control Room (licensed Reactor Operator) prior to initiating such actions. Removal of an instrument or component from service requires the permission of the Shift Supervisor or Assistant Shift Supervisor (licensed Reactor Operator).

12.5.1.3.4 Manipulation of Facility Controls

Only licensed Reactor Operators and Senior Reactor Operators are permitted to manipulate facility controls, except for trainees who are enrolled in approved training programs and are operating under the direction of a licensed Reactor Operator.

Software

Software for both the MIDAS and NOVA computer systems was modified in response to the aforementioned hardware changes.

IDADS data point (M6518) identifies whether the aspirator fan has failed. Dew point outputs were eliminated from the IDADS/MIDAS and NOVA data processing systems.

Precipitation data are available on the IDADS system. The QC critical values of the MIDAS system were modified to provide improved screening capability of the incoming IDADS data points.

Procedures

Meteorological program interdepartmental coordination and the data archival process are controlled by administrative procedure. The procedure also provides detailed information with respect to accessing, processing, validating, and maintaining all sources of meteorological data.

Surveillance Procedures define the daily channel checking requirements and semi-yearly instrumentation field calibration required by Technical Specifications. Calibration of wind speed and wind direction sensors in a wind tunnel on a rotating basis is now permitted.

Test Procedures are used for calibrating the meteorological instrument precipitation channel on a semiannual basis.