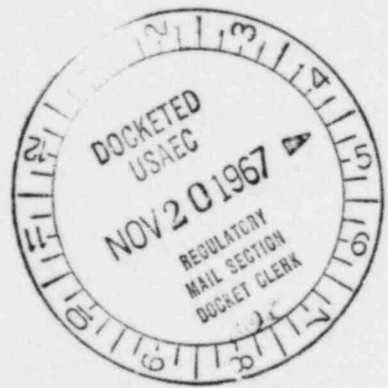




**SMUD**

SACRAMENTO MUNICIPAL UTILITY DISTRICT

**RANCHO SECO NUCLEAR GENERATING STATION  
UNIT NO. 1**



**PRELIMINARY SAFETY ANALYSIS REPORT**

**Volume IV**

REGULATORY DOCKET FILE COPY

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NOVEMBER 1967

B 377E

Docket No. 50-312  
February 2, 1968

AMENDMENT NO. 1

SACRAMENTO MUNICIPAL UTILITY DISTRICT  
RANCHO SECO NUCLEAR GENERATING STATION

UNIT NO. 1

Amendment No. 1 to the Sacramento Municipal Utility District's Preliminary Safety Analysis Report includes both replacement pages and new pages and tabs. All pages to be inserted are identified as Amendment 1. Any technical text material changed by this amendment is coded in the outside margin by a black bar and the numeral one.

Before inserting the Amendment 1 material (contained in this new Volume V) in the different volumes, it is suggested that the Appendix 5 material be removed from Volume IV to provide space. After the Amendment 1 material has been inserted, Appendix 3 should be the first amendment in the new Volume V. The List of Effective Pages should be checked to verify the completeness of Volumes I thru V.

It should be noted that License Application page 4 is replaced with a new page 4 plus two new additional pages, 8 and 9.

350

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SACRAMENTO MUNICIPAL UTILITY DISTRICT  
 RANCHO SECO NUCLEAR GENERATING STATION  
 UNIT NO. 1

**LIST OF  
EFFECTIVE PAGES**

Docket No. 50-289  
 February 2, 1968  
 Amendment No. 1

The active pages in this report are as follows:

Page or Fig. No.	Issue	Page or Fig. No.	Issue
Title .....	Original	3.3-1 thru 3.4-5.....	Original
A .....	Amendment 1	4-i thru 4-iv.....	Original
B thru E.....	Amendment 1	4.1-1 thru 4.3-10.....	Original
i.....	Original	4.3-11.....	Amendment 1
ii.....	Amendment 1	4.4-1 thru 4.5-1.....	Original
iii.....	Original	5.1-1.....	Amendment 1
iv thru ix .....	Amendment 1	5.1-2.....	Original
x .....	Original	5.1-3.....	Amendment 1
xi thru xiv .....	Amendment 1	5.1-4 thru 5.1-9.....	Original
1-i.....	Original	5.1-10.....	Amendment 1
1-ii.....	Amendment 1	5.1-11 thru 5.1-24.....	Original
1-iii thru iv.....	Original	5.1-25 thru 5.1-26.....	Amendment 1
1-v.....	Original	5.1-27 thru 5.1-29.....	Original
1.1-1 thru 1.1-2.....	Original	5.1-30.....	Amendment 1
Fig. 1.1-4 thru 1.1-8.....	Amendment 1	5.1-31.....	Original
1.2-1 thru 1.2-4.....	Original	5.1-32 thru 5.1-33.....	Amendment 1
1.3-1 thru 1.3-3.....	Original	Fig. 5.1-4.....	Amendment 1
1.3-4.....	Amendment 1	5.2-1 thru 5.3-1.....	Original
1.3-5 thru 1.3-9.....	Original	5.4-1.....	Amendment 1
1.4-1.....	Original	5.4-2 thru 5.4-5.....	Original
1.4-2 thru 1.4-3.....	Amendment 1	5.4-6.....	Amendment 1
1.4-4 thru 1.9-1.....	Original	5.4-7.....	Original
2-i thru 2-ii.....	Amendment 1	5.4-8 thru 5.4-9.....	Amendment 1
2-iii.....	Original	5.5-1 thru 5.9-1.....	Original
2.1-1 thru 2.3-5.....	Original	6-i thru 6-ii.....	Amendment 1
2.3-6 thru 2.3-8.....	Amendment 1	6.0-1.....	Original
2.4-1 thru 2.5-1.....	Original	6.1-1.....	Original
2.6-1.....	Amendment 1	6.1-2 thru 6.1-16.....	Amendment 1
2.7-1.....	Original	Fig. 6.1-1.....	Amendment 1
2.8-1 thru 2.8-4.....	Amendment 1	6.2-1.....	Amendment 1
2.9-1.....	Original	6.2-2 thru 6.2-8.....	Original
3-i thru 3-vi.....	Original	Fig. 6.2-1.....	Amendment 1
3.1-1 thru 3.1-6.....	Original	6.3-1 thru 6.3-3.....	Original
Fig. 3.2-65.....	Amendment 1	7-i.....	Original
Fig. 3.2-68.....	Amendment 1	7-ii.....	Amendment 1

SACRAMENTO MUNICIPAL UTILITY DISTRICT

RANCHO SECO NUCLEAR GENERATING STATION

UNIT NO. 1

**LIST OF  
EFFECTIVE PAGES**

Docket No. 50-289  
February 2, 1968  
Amendment No. 1

Page or Fig. No.	Issue	Page or Fig. No.	Issue
7.1-1 thru 7.1-5.....	Original	11.2-2 thru 11.2-5 .....	Original
7.1-6 thru 7.1-7.....	Amendment 1	11.2-6 .....	Amendment 1
7.1-9.....	Original	11.2-7 thru 11.3-1.....	Original
7.1-10 thru 7.1-11.....	Amendment 1	12-i.....	Original
7.1-12 thru 7.1-13.....	Original	12.1-1.....	Original
7.1-14 thru 7.1-15.....	Amendment 1	12.2-1 thru 12.2-2.....	Amendment 1
7.1-16 thru 7.1-19.....	Original	Fig. 12.2-1.....	Amendment 1
Fig. 7.1-2.....	Amendment 1	12.3-1 thru 12.3-5.....	Amendment 1
7.2-1 thru 7.2-6.....	Original	12.4-1 thru 12.7-1.....	Original
7.2-7 thru 7.2-8.....	Amendment 1	13-i.....	Original
7.2-9 thru 7.2-11.....	Original	13.1-1 thru 13.3-1.....	Original
7.3-1 thru 7.3-7.....	Original	i .....	Original
Fig. 7.3-3.....	Amendment 1	ii thru iii .....	Amendment 1
7.4-1.....	Original	14.1-1 thru 14.1-8.....	Original
7.4-2 thru 7.4-5.....	Amendment 1	14.1-9.....	Amendment 1
8-i thru 8-ii.....	Amendment 1	14.1-10 thru 14.1-19.....	Original
8.1-1 thru 8.2-1.....	Original	14.1-20.....	Amendment 1
8.2-2 thru 8.2-17.....	Amendment 1	14.2-1 thru 14.2-2.....	Original
Fig. 8.2-1 thru 8.2-3.....	Amendment 1	14.2-3.....	Amendment 1
8.3-1.....	Original	14.2-4.....	Original
8.3-2 thru 8.3-4.....	Amendment 1	14.2-5 thru 14.2-6.....	Amendment 1
8.4-1.....	Original	14.2-7 thru 14.2-32.....	Original
9-i thru iii.....	Original	14.2-33.....	Amendment 1
9.0-1 thru 9.2-9.....	Original	14.2-34.....	Original
9.3-1.....	Amendment 1	14.2-35 thru 14.2-37.....	Amendment 1
9.3-2 thru 9.4-4.....	Original	14.2-38 thru 14.2-39.....	Original
Fig. 9.4-1.....	Amendment 1	Fig. 14.2-36.....	Amendment 1
9.5-1 thru 9.5-2.....	Original	Fig. 14.2-48 thru 14.2-50..	Amendment 1
9.5-3.....	Amendment 1	14.3-1 thru 14.3-2.....	Original
9.5-4.....	Original	14.3-3 thru 14.3-14.....	Amendment 1
Fig. 9.5-1.....	Amendment 1	Fig. 14.3-4 thru 14.3-5....	Amendment 1
9.6-1 thru 9.6-7.....	Original	Fig. 14.3-7.....	Amendment 1
10-i.....	Original	14.4-1 thru 14.4-2.....	Original
10.1-1 thru 10.4-1.....	Original	15-1 thru 15-5.....	Original
11-i.....	Original	Appendix 1	
11-ii.....	Amendment 1	Table of Contents.....	Amendment 1
11.1-1 thru 11.2-1.....	Amendment 1	1A 1A-1 thru 1A-17.....	Amendment 1

000 (352)

SACRAMENTO MUNICIPAL UTILITY DISTRICT  
 RANCHO SECO NUCLEAR GENERATING STATION  
 UNIT NO. 1

**LIST OF  
EFFECTIVE PAGES**

Docket No. 50-289  
 February 2, 1968  
 Amendment No. 1

Page or Fig. No.	Issue	Page or Fig. No.	Issue
1B 1B-1 thru 1B-5.....	Amendment 1	2GA-1.....	Amendment 1
Appendix 2		Appendix 3	
Table of Contents.....	Amendment 1	Table of Contents.....	Amendment 1
2A Title Page.....	Original	3A 3A-1 thru 3A-14.....	Amendment 1
i thru vi.....	Original	4A Questions-4A-1	
1 thru 63.....	Original	thru 4A-8.....	Amendment 1
2B Southeast Area Plan		Appendix A to Question	
(Bound)-17 pages.....	Original	4A.2-1 thru 15.....	Amendment 1
Preliminary Projections...		Appendix B to Question	
to 1985-1 thru 4.....	Original	4A.2-1 thru 2.....	Amendment 1
2C Geology and Seismology-		Questions-4A-9 thru	
2C-1 thru 2C-13, Fig 2C-1		4A-12.....	Amendment 1
thru 2C-11.....	Original	Appendix 5	
Geophysical Report-		Table of Contents.....	Amendment 1
1 thru 6.....	Original	5A 5A-1.....	Original
Additional Seismic Exploration-		5A-2 thru 5A-6.....	Amendment 1
1 thru 2, Plate 1 thru		Fig 5A-1 thru 5A-2.....	Amendment 1
Plate 3.....	Original	5B 5B-1 thru 5B-3.....	Original
Geological Log of Drill		5C 5C-1 thru 5C-3.....	Original
Holes-91 Sheets.....	Original	5D 1 thru 10.....	Original
2D Seismic Hazard at the		5E 5E-1 thru 5E-2.....	Original
Clay Site 1 thru 14.....	Original	5F 5F-1 thru 5F-2.....	Original
Addendum to Seismic Hazard		5G 5G-1 thru 5G-2.....	Original
at the Clay Site-1 sheet...	Original	5H 5H-1 thru 5H-5.....	Original
Seismic Hazard at the		5I 5I-1.....	Original
Sierran Sites Area		5J 5J-1 thru 5J-2.....	Amendment 1
1 thru 10.....	Amendment 1	Appendix 6	
2E Soil and Foundations		Table of Contents.....	Amendment 1
Investigation Report		6A-1 thru 6A-6.....	Amendment 1
2E-1 thru 2E-11, Fig C-119-E		Appendix 7	
thru C122-E.....	Original	Table of Contents.....	Amendment 1
Report of Laboratory		7A-1.....	Amendment 1
Testing-1 thru 9, 3 Tables,		Appendix 9	
Fig 1 thru 2 and 9, curves		Table of Contents.....	Amendment 1
1 thru 7.....	Original	9A-1 thru 9A-2.....	Amendment 1
2F 2F-1 thru 2F-2.....	Amendment 1	Appendix 11	
2G 2G-1 thru 2G-3.....	Amendment 1	Table of Contents.....	Amendment 1

SACRAMENTO MUNICIPAL UTILITY DISTRICT  
RANCHO SECO NUCLEAR GENERATING STATION  
UNIT NO. 1

**LIST OF  
EFFECTIVE PAGES**

Docket No. 50-289  
February 2, 1968  
Amendment No. 1

Page or Fig. No.	Issue
11A-1 thru 11A-6.....	Amendment 1
Appendix 12	
Table of Contents.....	Amendment 1
12A-1 thru 12A-4.....	Amendment 1
Appendix 14	
Table of Contents.....	Amendment 1
14A-1 thru 14A-22.....	Amendment 1

Docket No. 50-312  
April 15, 1968

AMENDMENT NO. 2

SACRAMENTO MUNICIPAL UTILITY DISTRICT  
RANCHO SECO NUCLEAR GENERATING STATION  
UNIT NO. 1

Amendment No. 2 to the Sacramento Municipal Utility District's Preliminary Safety Analysis Report includes both replacement pages and new pages and tabs. All pages to be inserted are identified as Amendment 2, except the reprinted appendices. Any technical text material changed by this amendment is coded in the outside margin by a black bar and the numeral two.

Before inserting the Amendment 2 material in the different volumes, it is suggested that Appendices 2A, 2C, 2D and 2E be removed from Volume IV, discarded and replaced with the new reprinted appendices 2A, 2C, 2D, and 2E. Additionally, remove Appendices 3 and 4 (including tabs) from Volume V and place at the back of Volume IV. The list of Effective Pages should be checked to verify the completeness of Volumes I thru V.

It should be noted that three new additional pages, 10, 11 and 12 are to be added to the License Application.

The response to letter from Peter A. Morris, Director, Division of Reactor Licensing to E. K. Davis, General Counsel, Sacramento Municipal Utility District, dated March 21, 1968, is arranged in the question order of the above letter. For convenience a cross reference of the AEC DRL question number and SMUD response number is presented below. Response to questions are to be inserted into the volumes according to the assigned SMUD number.

355

AEC DRL QUESTION NO.	SMUD RESPONSE NO.	AEC DRL QUESTION NO.	SMUD RESPONSE NO.	AEC DRL QUESTION NO.	SMUD RESPONSE NO.
1.1	1A.4	6.1	6A.7	12.1	12A.2
1.2	1A.5	6.2	6A.8	12.2	12A.3
1.3	1A.6	6.3	6A.9	12.3	12A.4
1.4	1A.7	6.4	6A.10	12.4	12A.5
1.5	1A.8	6.5	6A.11	12.5	12A.6
1.6	1A.9	6.6	6A.12	12.6	12A.7
1.7	1A.10	6.7	6A.13		
		6.8	6A.14	13.1	13A.1
2.1	14A.14	6.9	6A.15	13.2	13A.2
2.2	14A.15	6.10	6A.16	13.3	13A.3
2.3	14A.16			13.4	12A.8
2.4	14A.17	7.1	7A.2		
2.5	14A.18	7.2	7A.3	14.1	14A.20
2.6	14A.19	7.3	7A.4	14.2	14A.21
2.7	2H.1	7.4	7A.5	14.3	14A.22
2.8	2H.2	7.5	7A.6	14.4	14A.23
		7.6	7A.7	14.5	14A.24
3.1	3A.6	7.7	7A.8	14.6	14A.25
3.2	3A.7	7.8	7A.9	14.7	14A.26
3.3	3A.8	7.9	7A.10	14.8	14A.27
3.4	3A.9			14.9	14A.28
3.5	3A.10	8.1	8A.1		
3.6	3A.11	8.2	8A.2	15.1	15A.1
3.7	3A.12	8.3	8A.3		
3.8	3A.13	8.4	8A.4	16.1	7A.11
3.9	3A.14	8.5	8A.5	16.2	14A.29
3.10	3A.15			16.3	14A.30
		9.1	9A.2	16.4	3A.16
4.1	4A.12	9.2	9A.3	16.5	5J.5
4.2	5J.4	9.3	9A.4	16.6	1A.11
4.3	4A.13	9.4	9A.5		
4.4	4A.14	9.5	9A.6		
4.5	4A.15	9.6	9A.7		
		9.7	9A.8		

356

00



**LIST OF  
EFFECTIVE PAGES**

Docket No. 50-312  
April 15, 1968  
Amendment No. 2

The active pages in this report are as follows:

Page or Fig. No.	Issue	Page or Fig. No.	Issue
Title Page . . . . .	.Original	Fig. 2.2-1 thru 2.2-10. . . . .	.Original
A thru H . . . . .	Amendment 2	2.3-1 thru 2.3-2. . . . .	Amendment 2
i . . . . .	.Original	2.3-3 thru 2.3-4. . . . .	.Original
ii. . . . .	Amendment 2	2.3-5 thru 2.3-8. . . . .	Amendment 2
iii . . . . .	.Original	Fig. 2.3-1 thru 2.3-6 . . . . .	.Original
iv. . . . .	Amendment 2	2.4-1 . . . . .	.Original
v thru vii. . . . .	Amendment 1	2.4-2 . . . . .	Amendment 2
viii. . . . .	Amendment 2	2.4-3 . . . . .	.Original
ix. . . . .	Amendment 1	Fig. 2.4-1 thru 2.4-2 . . . . .	.Original
x . . . . .	.Original	2.5-1 . . . . .	.Original
xi. . . . .	Amendment 1	2.6-1 . . . . .	.Original
xii thru xiv. . . . .	Amendment 2	2.7-1 . . . . .	.Original
1-i . . . . .	.Original	2.8-1 thru 2.8-4. . . . .	Amendment 1
1-ii. . . . .	Amendment 2	2.9-1 . . . . .	.Original
1-iii . . . . .	.Original	3-i thru 3-iii. . . . .	Amendment 2
1-iv. . . . .	Amendment 2	3-iv thru 3-vi. . . . .	.Original
1.1-1 thru 1.1-2. . . . .	.Original	3.1-1 . . . . .	.Original
Fig. 1.1-1. . . . .	.Original	3.1-2 thru 3.1-4. . . . .	Amendment 2
Fig. 1.1-2 thru 1.1-8 . . . . .	Amendment 2	3.1-5 . . . . .	Amendment 2
1.2-1 . . . . .	.Original	3.1-6 . . . . .	Amendment 2
1.2-2 thru 1.2-4. . . . .	Amendment 2	3.2-1 thru 3.2-2. . . . .	.Original
1.3-1 thru 1.3-3. . . . .	.Original	3.2-3 . . . . .	Amendment 2
1.3-4 . . . . .	Amendment 1	3.2-4 thru 3.2-10 . . . . .	.Original
1.3-5 . . . . .	.Original	3.2-11. . . . .	Amendment 2
1.3-6 thru 1.3-7. . . . .	Amendment 2	3.2-12 thru 3.2-69. . . . .	.Original
1.3-8 . . . . .	.Original	3.2-70 thru 3.2-101 . . . . .	Amendment 2
1.3-9 . . . . .	Amendment 2	Fig. 3.2-1 thru 3.2-59. . . . .	.Original
1.4-1 . . . . .	.Original	Fig. 3.2-59 thru 3.2-61 . . . . .	Amendment 2
1.4-2 . . . . .	Amendment 2	Fig. 3.2-62 thru 3.2-63 . . . . .	.Original
1.4-3 . . . . .	Amendment 1	Fig. 3.2-64 . . . . .	Amendment 2
1.4-4 thru 1.4-6. . . . .	.Original	Fig. 3.2-65 . . . . .	Amendment 1
1.4-7 thru 1.4-8. . . . .	Amendment 2	Fig. 3.2-66 . . . . .	.Original
1.4-9 thru 1.4-37 . . . . .	Amendment 2	Fig. 3.2-67 . . . . .	Amendment 2
1.5-1 thru 1.5-2. . . . .	Amendment 2	Fig. 3.2-68 . . . . .	Amendment 1
1.6-1 . . . . .	.Original	Fig. 3.2-69 . . . . .	.Original
1.6-2 thru 1.6-3. . . . .	Amendment 2	3.3-1 . . . . .	.Original
Fig. 1.6-1 thru 1.6-2 . . . . .	.Original	3.3-2 . . . . .	Amendment 2
1.7-1 . . . . .	.Original	3.3-3 thru 3.3-5. . . . .	.Original
1.8-1 thru 1.8-2. . . . .	.Original	3.3-6 thru 3.3-7. . . . .	Amendment 2
1.9-1 . . . . .	.Original	3.3-8 thru 3.3-10 . . . . .	.Original
2-i thru 2-ii . . . . .	Amendment 1	3.3-11 thru 3.3-12. . . . .	Amendment 2
2-iii . . . . .	.Original	3.4-1 thru 3.4-5. . . . .	.Original
2.1-1 . . . . .	.Original	4-i thru 4-ii . . . . .	.Original
2.2-1 thru 2.2-5. . . . .	.Original	4.1-1 thru 4.1-15 . . . . .	.Original

**LIST OF  
EFFECTIVE PAGES**

Docket No. 50-312  
April 15, 1968  
Amendment No. 2

Page or Fig. No.	Issue	Page or Fig. No.	Issue
Fig. 4.1-1 . . . . .	Original	Fig. 5.7-1 . . . . .	Amendment 2
Fig. 4.1-2 . . . . .	Amendment 2	5.8-1 . . . . .	Original
Fig. 4.1-3 thru 4.1-4 . . . . .	Original	5.9-1 . . . . .	Original
4.2-1 . . . . .	Amendment 2	6-i . . . . .	Amendment 1
4.2-2 thru 4.2-6 . . . . .	Original	6-ii . . . . .	Amendment 2
4.2-7 thru 4.2-8 . . . . .	Amendment 2	6.0-1 . . . . .	Amendment 2
4.2-9 . . . . .	Original	Fig. 6.0-1 . . . . .	Amendment 2
4.2-10 thru 4.2.11. . . . .	Amendment 2	6.1-1 thru 1-7. . . . .	Amendment 2
4.2-12. . . . .	Original	6.1-8 . . . . .	Original
Fig. 4.2-1. . . . .	Amendment 2	6.1-9 thru 6.1-10 . . . . .	Amendment 2
Fig. 4.2-2 thru 4.2-8 . . . . .	Original	6.1-11. . . . .	Amendment 1
4.3-1 . . . . .	Amendment 2	6.1-12 thru 6.1-14. . . . .	Amendment 2
4.3-2 thru 4.3-7. . . . .	Original	6.1-15. . . . .	Amendment 1
4.3-8 thru 4.3-10 . . . . .	Amendment 2	6.1-16. . . . .	Amendment 2
4.3-11. . . . .	Amendment 1	Fig. 6.1-1 thru 6.1-2 . . . . .	Amendment 2
4.4-1 thru 4.4-3. . . . .	Original	Fig. 6.1-3. . . . .	Original
4.4-4 . . . . .	Amendment 2	Fig. 6.1-4. . . . .	Amendment 2
4.4-5 . . . . .	Original	6.2-1 thru 6.2-8. . . . .	Amendment 2
4.5-1 . . . . .	Original	Fig. 6.2-1. . . . .	Amendment 2
5-i thru 5-iii. . . . .	Amendment 1	6.3-1 thru 6.3-2. . . . .	Original
5.1-1 . . . . .	Amendment 1	6.3-3 . . . . .	Amendment 2
5.1-2 . . . . .	Original	7-i . . . . .	Amendment 2
5.1-3 . . . . .	Amendment 1	7-ii. . . . .	Amendment 1
5.1-4 thru 5.1-9. . . . .	Original	7-iii . . . . .	Original
5.1-10. . . . .	Amendment 1	7.1-1 thru 7.1-20 . . . . .	Amendment 2
5.1-11 thru 5.1-24. . . . .	Original	Fig. 7.1-1. . . . .	Original
5.1-25 thru 5.1-26. . . . .	Amendment 1	Fig. 7.1-2 thru 7.1-3 . . . . .	Amendment 2
5.1-27 thru 5.1-29. . . . .	Original	Fig. 7.1-4. . . . .	Original
5.1-30. . . . .	Amendment 1	7.2-1 thru 7.2-5. . . . .	Original
5.1-31. . . . .	Original	7.2-6 . . . . .	Amendment 2
5.1-32 thru 5.1-33. . . . .	Amendment 1	7.2-7 thru 7.2-8. . . . .	Amendment 1
Fig. 5.1-1 thru 5.1-3 . . . . .	Original	7.2-9 thru 7.2-11 . . . . .	Original
Fig. 5.1-4. . . . .	Amendment 1	Fig. 7.2-1 thru 7.2-4 . . . . .	Original
5.2-1 thru 5.2-5. . . . .	Original	7.3-1 thru 7.3-3. . . . .	Original
5.3-1 . . . . .	Original	7.3-4 thru 7.3-7. . . . .	Amendment 2
5.4-1 . . . . .	Amendment 1	Fig. 7.3-1. . . . .	Amendment 2
5.4-2 thru 5.4-5. . . . .	Original	Fig. 7.3-2. . . . .	Original
5.4-6 . . . . .	Amendment 1	Fig. 7.3-3. . . . .	Amendment 1
5.4-7 . . . . .	Original	Fig. 7.3-4 thru 7.3-5 . . . . .	Amendment 2
5.4-8 thru 5.4-9. . . . .	Amendment 1	7.4-1 . . . . .	Original
5.5-1 thru 5.5-3. . . . .	Original	7.4-2 thru 7.4-5. . . . .	Amendment 1
5.6-1 thru 5.6-2. . . . .	Original	Fig. 7.4-1. . . . .	Original
5.6-3 thru 5.6-6. . . . .	Amendment 2	8-i thru 8-ii . . . . .	Amendment 2
Fig. 5.6-1. . . . .	Amendment 2	8.1-1 . . . . .	Original
5.7-1 . . . . .	Amendment 2	8.2-1 thru 8.2-18 . . . . .	Amendment 2
5.7-2 . . . . .	Original	Fig. 8.2-1 thru 8.2-3 . . . . .	Amendment 2

**LIST OF  
EFFECTIVE PAGES**

Docket No. 50-312  
April 15, 1968  
Amendment No. 2

Page or Fig. No.	Issue	Page or Fig. No.	Issue
8.3-1 thru 8.3-3.	.Amendment 2	12-i.	.Amendment 1
8.3-4	.Amendment 1	12.1-1.	.Original
8.4-1	.Original	12.2-1 thru 12.2-2.	.Amendment 2
9-i	.Original	Fig. 12.2-1	.Amendment 2
9-ii thru 9-iii	.Amendment 2	12.3-1 thru 12.3-2.	.Amendment 2
9.0-1 thru 9.0-2.	.Original	12.3-3 thru 12.3-5.	.Amendment 1
Fig. 9.0-1.	.Original	Fig. 12.3-1	.Amendment 2
9.1-1 thru 9.1-2.	.Amendment 2	12.4-1.	.Original
9.1-3	.Original	12.5-1.	.Original
9.1-4 thru 9.1-7.	.Amendment 2	12.6-1.	.Original
9.1-8	.Original	12.7-1.	.Original
Fig. 9.1-1.	.Amendment 2	13-i	.Original
9.2-1	.Original	13.1-1 thru 13.1-2	.Original
9.2-2 thru 9.2-7.	.Amendment 2	13.1-3	.Amendment 2
9.2-8	.Original	13.2-1	.Original
9.2-9	.Amendment 2	13.3-1	.Original
Fig. 9.2-1.	.Original	14-i	.Original
9.3-1 thru 9.3-6.	.Amendment 2	14-ii thru 14-iii.	.Amendment 1
Fig. 9.3-1 thru 9.3-3	.Amendment 2	14-iv thru 14-viii	.Original
9.4-1 thru 9.4-4.	.Original	14.1-1 thru 14.1-7	.Original
Fig. 9.4-1.	.Amendment 1	14.1-8 thru 14.1-10.	.Amendment 2
9.5-1 thru 9.5-3.	.Amendment 2	14.1-11 thru 14.1-15	.Original
9.5-4	.Original	14.1-16.	.Amendment 2
Fig. 9.5-1 thru 9.5-2	.Amendment 2	14.1-17 thru 14.2-19	.Original
9.6-1	.Amendment 2	14.1-20.	.Amendment 2
9.6-2 thru 9.6-7.	.Original	Fig. 14.1-1 thru 14.1-21	.Original
Fig. 9.6-1.	.Original	14.2-1	.Original
9.7-1 thru 9.7-2.	.Original	14.2-2	.Amendment 2
Fig. 9.7-1.	.Amendment 2	14.2-3	.Amendment 1
10-i.	.Original	14.2-4	.Original
10.1-1.	.Original	14.2-5 thru 14.2-6	.Amendment 1
10.2-1.	.Amendment 2	14.2-7 thru 14.2-9	.Original
10.2-2.	.Original	14.2-10.	.Amendment 2
Fig. 10.2-1	.Original	14.2-11 thru 14.2-23	.Original
10.3-1 thru 10.3-2.	.Original	14.2-24.	.Amendment 2
10.4-1.	.Original	14.2-25.	.Original
11-i thru 11-ii	.Amendment 1	14.2-26 thru 14.2-30	.Amendment 2
11.1-1 thru 11.1-4.	.Amendment 1	14.2-31 thru 14.2-32	.Original
11.1-5.	.Amendment 2	14.2-33.	.Amendment 2
11.1-6 thru 11.1-8.	.Amendment 1	14.2-34.	.Original
Fig. 11.1-1, 11.1-2	.Original	14.2-35 thru 14.2-37	.Amendment 1
11.2-1.	.Amendment 1	14.2-38.	.Amendment 2
11.2-2 thru 11.2-5.	.Original	14.2-39.	.Original
11.2-6.	.Amendment 1	Fig. 14.2-1 thru 14.2-18	.Original
11.2-7 thru 11.2-11	.Original	Fig. 14.2-19 thru 14.2-20.	.Amendment 2
11.3-1.	.Original	Fig. 14.2-21 thru 14.2-28.	.Original

**LIST OF  
EFFECTIVE PAGES**

Docket No. 50-312  
April 15, 1968  
Amendment No. 2

Page or Fig. No	Issue	Page or Fig. No.	Issue
Fig. 14.2-29 . . . . .	Amendment 2	Preliminary Projections to 1985-1 thru 4. . . . .	Original
Fig. 14.2-30 thru 14.2-31. . . . .	Original	2C Geology and Seismology- 2C-1 thru 2C-13. . . . .	Original
Fig. 14.2-32 . . . . .	Amendment 2	Fig. 2C-1 thru 2C-11 . . . . .	Original
Fig. 14.2-33 . . . . .	Original	Geophysical Report- 1 thru 6 . . . . .	Original
Fig. 14.2-34 . . . . .	Amendment 2	Additional Seismic Exploration- 1 thru 2, Plate 1 thru Plate 3. . . . .	Original
Fig. 14.2-35 . . . . .	Original	Geological Log of Drill Holes-91 Sheets. . . . .	Original
Fig. 14.2-36 . . . . .	Amendment 1	2D Seismic Report Seismic Hazard at the Clay Site, 1 thru 14 . . . . .	Original
Fig. 14.2-37 thru 14.2-47. . . . .	Original	Addendum to Seismic Hazard at the Clay Site-1 sheet . . . . .	Original
Fig. 14.2-48 thru 14.2-50. . . . .	Amendment 1	Seismic Hazard at the Sierran Sites Area 1 thru 10. . . . .	Amendment 1
14.3-1 thru 14.3-2 . . . . .	Original	2E Soil and Foundations Investigation Report 2E-1 thru 2E-11, Fig. C-119-E thru C122-E . . . . .	Original
14.3-3 thru 14.3-8 . . . . .	Amendment 1	Report of Laboratory Testing-1 thru 9, 3 Tables, Fig. 1 thru 2 and 9, curves 1 thru 7 . . . . .	Original
14.3-9 thru 14.3-10. . . . .	Amendment 2	2F Meteorological Station 2F-1 thru 2F-2 . . . . .	Amendment 1
14.3-11 thru 14.3-13 . . . . .	Amendment 1	2G Storage Reservoir Criteria 2G-1 thru 2G-3 . . . . .	Amendment 1
14.3-14. . . . .	Amendment 2	2GA-1. . . . .	Amendment 1
Fig. 14.3-1 thru 14.3-3. . . . .	Original	2H Answers to Questions 2H-1 thru 2H-2 . . . . .	Amendment 2
Fig. 14.3-4 thru 14.3-5. . . . .	Amendment 1	Letter pg. 1 and 2 . . . . .	Amendment 2
Fig. 14.3-6. . . . .	Original	Fig. 2H.2-1 thru 2H.2-2. . . . .	Amendment 2
Fig. 14.3-7. . . . .	Amendment 1	Appendix 3 Table of Contents . . . . .	Amendment 1
14.4-1 thru 14.4-2 . . . . .	Original	3A Answers to Questions 3A-1 thru 3A-14. . . . .	Amendment 1
15-1 thru 15-5 . . . . .	Original	Fig. 3A.2-1 thru 3A.2-3. . . . .	Amendment 1
Appendix 1 Table of Contents . . . . .	Amendment 2	Fig. 3A.4-1. . . . .	Amendment 1
1A Answers to Questions. . . . .	Amendment 1	3A-15 thru 3A-23 . . . . .	Amendment 2
1A-1 thru 1A-14. . . . .	Amendment 1	Fig. 3A.14-1 . . . . .	Amendment 2
Fig. 1A.2-1. . . . .	Amendment 1		
1A-15 thru 1A-16 . . . . .	Amendment 1		
1A-17 thru 1A-23 . . . . .	Amendment 2		
1B Quality Assurance Operations . . . . .	Amendment 1		
1B-18. . . . .	Original		
1B-2 thru 1B-4 . . . . .	Amendment 1		
1B-5 . . . . .	Amendment 2		
Fig. 1B-1. . . . .	Amendment 2		
1C Rancho Seco Project Engineering Staff. . . . .	Amendment 2		
1C-1 thru 1C-4 . . . . .	Amendment 2		
Fig. 1C-1. . . . .	Amendment 2		
Appendix 2 Table of Contents . . . . .	Amendment 2		
2A Final Report i thru vi. . . . .	Original		
1 thru 63. . . . .	Original		
Supplement 1 thru 18. . . . .	Amendment 2		
2B Southeast Area Plan (Bound)-14 pages . . . . .	Original		

**LIST OF  
EFFECTIVE PAGES**

Docket No. 50-312  
April 15, 1968  
Amendment No. 2

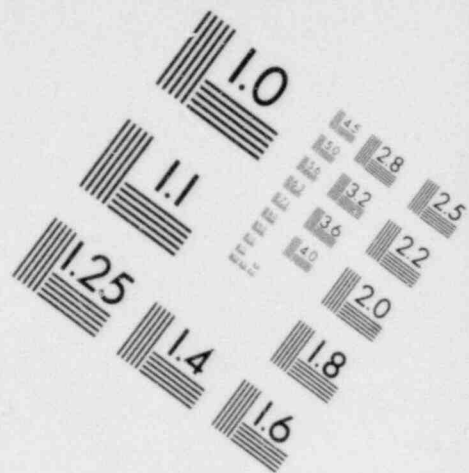
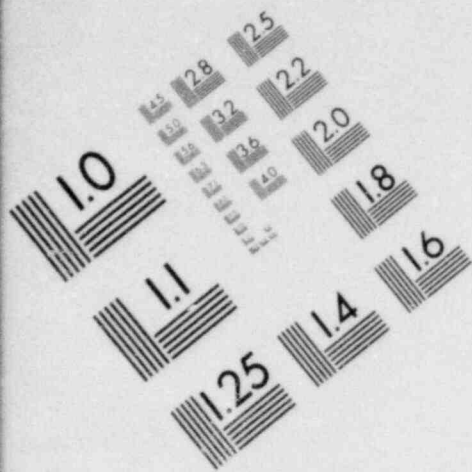
Page or Fig. No.	Issue	Page or Fig. No.	Issue
3A-24 thru 3A-26 . . . . .	Amendment 2	Figure 5G-1. . . . .	Original
Appendix 4 Table of Contents . . . . .	Amendment 1	5H Quality Control Procedure for Field Welding	
4A Answers to Questions		5H-1 thru 5H-5 . . . . .	Original
4A-1 thru 4A-6		5I Containment Structure Instrumentation	
Fig. 4A.1-1 thru 4A.1-15 .	Amendment 1	5I-1 . . . . .	Original
Appendix A		5J Answers to Questions	
The Properties and Micro-structure of Spray-Quenched Thick-Section Steels		5J-1 thru 5J-2 . . . . .	Amendment 1
15 pages . . . . .	Original	Fig. 5J2-1 . . . . .	Amendment 1
4A-1 . . . . .	Original	5J-3 thru 5J-14. . . . .	Amendment 2
4A-2 thru 4A-8 . . . . .	Amendment 1	Appendix 6 Table of Contents . . . . .	Amendment 1
Appendix B, B & W Data (2 pgs.) . . . . .	Amendment 1	6A Answers to Questions	
4A-9 thru 4A-12. . . . .	Amendment 1	6A-1 . . . . .	Amendment 1
4A-13 thru 4A-18 . . . . .	Amendment 2	6A-2 . . . . .	Amendment 2
Appendix 5 Table of Contents . . . . .	Amendment 1	6A-3 thru 6A-5 . . . . .	Amendment 1
5A Structural Design Bases		6A-6 thru 6A-7 . . . . .	Amendment 2
5A-1 thru 5A-5 . . . . .	Amendment 2	Fig. 6A.8-1. . . . .	Amendment 2
Fig. 5A-6 thru 5A-7. . . . .	Amendment 1	6A-8 thru 6A-18. . . . .	Amendment 2
Fig. 5A-1 thru 5A-2. . . . .	Amendment 1	Fig. 6A.16-1 thru 6A.16-2. . . . .	Amendment 2
5B Justification of Structural Proof Test - Pressures		Appendix 7 Table of Contents . . . . .	Amendment 1
5B-1 thru 5B-3 . . . . .	Original	7A Answers to Questions	
5C Specification for Splicing Reinforcing Bar Using the Coldwell Process		7A-1 . . . . .	Amendment 1
5C-1 thru 5C-3 . . . . .	Original	7A-2 thru 7A-9 . . . . .	Amendment 2
5D Turbine Generator Missiles 1 thru 10. . . . .	Original	Appendix 8 Table of Contents . . . . .	Amendment 2
4 sheets of Parts Drawings		8A Answers to Questions	
5E Justification for Load Factors		8A-1 thru 8A-4 . . . . .	Amendment 2
5E-1 thru 5E-2 . . . . .	Original	Appendix 9 Table of Contents . . . . .	Amendment 2
5F Justification for Yield Reduction Factors. . . . .		9A Answers to Questions	
5F-1 thru 5F-2 . . . . .	Original	9A-1 . . . . .	Amendment 2
5G Description of the Finite Element Technique Used in Containment Structural Analysis		9A-2 . . . . .	Amendment 1
5G-1 thru 5G-2 . . . . .	Original	9A-3 thru 9A-6 . . . . .	Amendment 2
		Appendix 10 Contains nothing	
		Appendix 11 Table of Contents . . . . .	Amendment 1
		11A Answers to Questions	
		11A-1 thru 11A-2 . . . . .	Amendment 1
		Fig. 11A.1-1 . . . . .	Amendment 1
		Fig. 11A.1-2 . . . . .	Amendment 2
		11A-3 thru 11A-6 . . . . .	Amendment 1

**LIST OF  
EFFECTIVE PAGES**

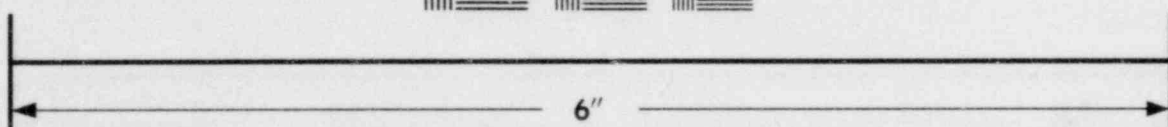
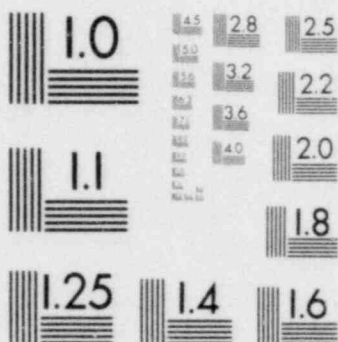
Docket No. 50-312  
April 15, 1968  
Amendment No. 2

Page or Fig. No.	Issue	Page or Fig. No.	Issue
Appendix 12 Table of Contents . . . . .	Amendment 1	Appendix 15 Table of Contents . . . . .	Amendment 2
12A Answers to Questions		15A Answers to Questions	
12A-1 thru 12A-4 . . . . .	Amendment 1	15A-1 thru 15A-2 . . . . .	Amendment 2
12A-5 thru 12A-10. . . . .	Amendment 2		
Fig. 12A.5-1 . . . . .	Amendment 2		
12A-11 . . . . .	Amendment 2		
Fig. 12A.6-1 thru 12A.6-3. . . . .	Amendment 2		
12A-12 thru 12A-17 . . . . .	Amendment 2		
Appendix 13 Table of Contents . . . . .	Amendment 2		
13A Answers to Questions			
13A-1 thru 13A-3 . . . . .	Amendment 2		
Appendix 14 Table of Contents . . . . .	Amendment 1		
14A Answers to Questions			
14A-1. . . . .	Amendment 1		
14A-2. . . . .	Amendment 2		
14A-3 thru 14A-6 . . . . .	Amendment 1		
14A-7 thru 14A-9 . . . . .	Amendment 2		
Fig. 14A.6-1 thru 14A.6-3. . . . .	Amendment 1		
Fig. 14A.6-4 thru 14A.6-5. . . . .	Amendment 2		
14A-11 thru 14A-13 . . . . .	Amendment 1		
14A-14 . . . . .	Amendment 2		
Fig. 14A.8-1 . . . . .	Amendment 2		
14A-15 thru 14A-20 . . . . .	Amendment 1		
Fig. 14A.11-1 thru 14A.11-2 . . . . .	Amendment 1		
14A-21 thru 14A-22 . . . . .	Amendment 1		
14A-23 thru 14A-29 . . . . .	Amendment 2		
Fig. 14A.18-1. . . . .	Amendment 2		
14A-30 . . . . .	Amendment 2		
Fig. 14A.19-1. . . . .	Amendment 2		
14A-31 thru 14A-32 . . . . .	Amendment 2		
Fig. 14A.21-1 thru 14A.21-4 . . . . .	Amendment 2		
14A-33 thru 14A-34 . . . . .	Amendment 2		
Fig. 14A.22-1. . . . .	Amendment 2		
14A-35 thru 14A-36 . . . . .	Amendment 2		
Fig. 14A.25-1. . . . .	Amendment 2		
Fig. 14A.26-1. . . . .	Amendment 2		
14A-37 thru 14A-41 . . . . .	Amendment 2		

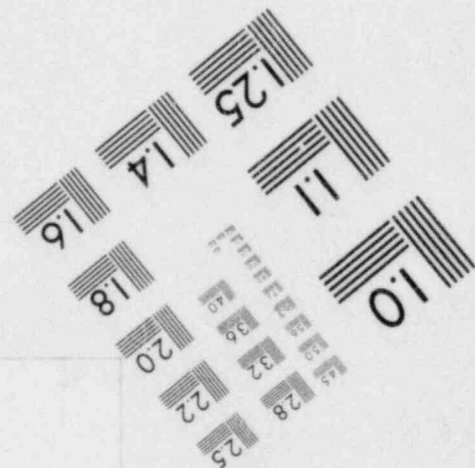
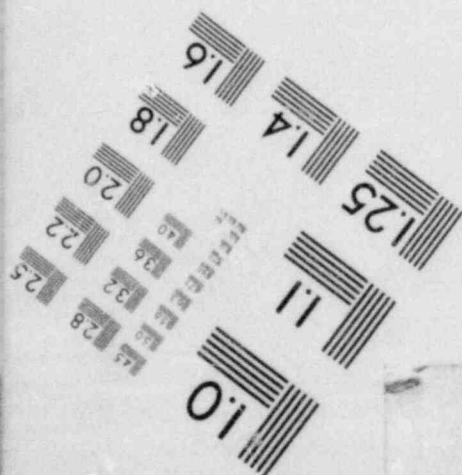
362

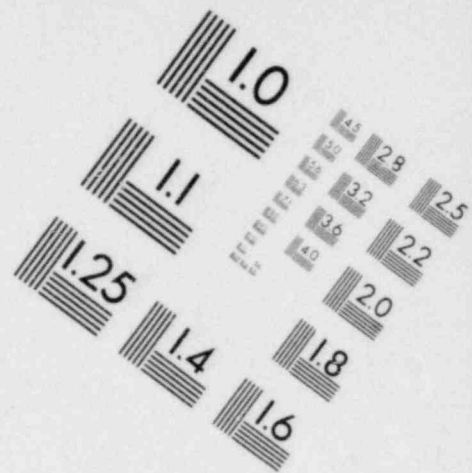
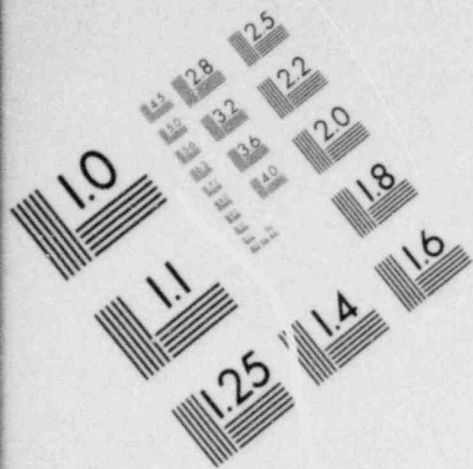


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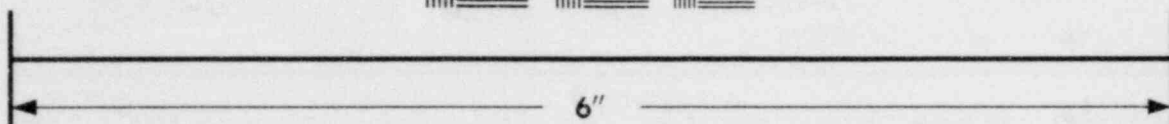


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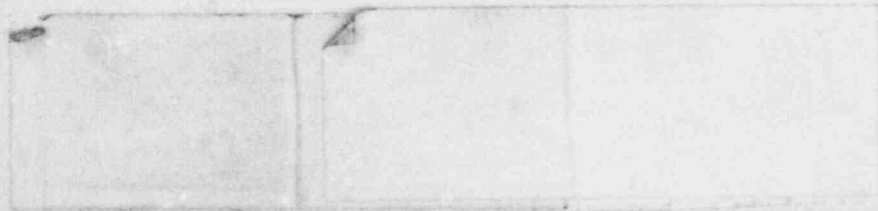
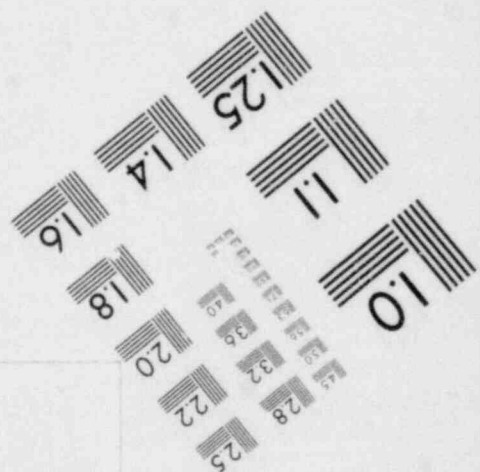
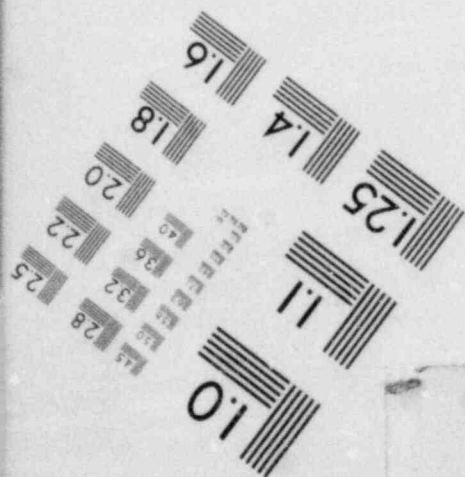




TABLE OF CONTENTS

VOLUME I

1. INTRODUCTION AND SUMMARY

<u>Section</u>	<u>Page</u>
1.1 <u>INTRODUCTION</u>	1.1-1
1.2 <u>DESIGN HIGHLIGHTS</u>	1.2-1
1.2.1 SITE CHARACTERISTICS	1.2-1
1.2.2 POWER LEVEL	1.2-1
1.2.3 PEAK SPECIFIC POWER LEVEL	1.2-1
1.2.4 REACTOR BUILDING	1.2-1
1.2.5 ENGINEERED SAFEGUARDS	1.2-2
1.2.6 ELECTRICAL SYSTEMS AND EMERGENCY POWER	1.2-3
1.2.7 ONCE-THROUGH STEAM GENERATORS	1.2-4
1.3 <u>TABULAR CHARACTERISTICS</u>	1.3-1
1.3.1 ITEM 1 - HYDRAULIC AND THERMAL DESIGN PARAMETERS	1.3-1
1.3.2 ITEM 2 - CORE MECHANICAL DESIGN PARAMETERS	1.3-1
1.3.3 ITEM 3 - PRELIMINARY NUCLEAR DESIGN DATA	1.3-8
1.3.4 ITEM 4 - PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT SYSTEM	1.3-8
1.3.5 ITEM 5 - REACTOR COOLANT SYSTEM - CODE REQUIREMENTS	1.3-8
1.3.6 ITEM 6 - PRINCIPAL DESIGN PARAMETERS OF THE REACTOR VESSEL	1.3-9
1.3.7 ITEM 7 - PRINCIPAL DESIGN FEATURES OF THE STEAM GENERATORS	1.3-9
1.3.8 ITEM 8 - PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PUMPS	1.3-9
1.3.9 ITEM 9 - PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PIPING	1.3-9
1.3.10 ITEM 10 - REACTOR BUILDING PARAMETERS	1.3-9
1.3.11 ITEM 11 - ENGINEERED SAFEGUARDS	1.3-9
1.4 <u>PRINCIPAL DESIGN CRITERIA</u>	1.4-1
1.4.1 CRITERION 1 - QUALITY STANDARDS (CATEGORY A)	1.4-1
1.4.2 CRITERION 2 - PERFORMANCE STANDARDS (CATEGORY A)	1.4-2
1.4.3 CRITERION 3 - FIRE PROTECTION (CATEGORY A)	1.4-3
1.4.4 CRITERION 4 - SHARING OF SYSTEMS (CATEGORY A)	1.4-5
1.4.5 CRITERION 5 - RECORDS REQUIREMENTS (CATEGORY A)	1.4-5
1.4.6 CRITERION 6 - REACTOR CORE DESIGN (CATEGORY A)	1.4-5
1.4.7 CRITERION 7 - SUPPRESSION OF POWER OSCILLATIONS (CATEGORY B)	1.4-6
1.4.8 CRITERION 8 - OVERALL POWER COEFFICIENT (CATEGORY B)	1.4-7
1.4.9 CRITERION 9 - REACTOR COOLANT PRESSURE BOUNDARY (CATEGORY A)	1.4-7
1.4.10 CRITERION 10 - CONTAINMENT (CATEGORY A)	1.4-8
1.4.11 CRITERION 11 - CONTROL ROOM (CATEGORY B)	1.4-8

001

<u>Section</u>	<u>Page</u>
1.4.12 CRITERION 12 - INSTRUMENTATION AND CONTROL SYSTEMS (CATEGORY B)	1.4-10
1.4.13 CRITERION 13 - FISSION PROCESS MONITORS AND CONTROLS (CATEGORY B)	1.4-11
1.4.14 CRITERION 14 - CORE PROTECTION SYSTEMS (CATEGORY B)	1.4-12
1.4.15 CRITERION 15 - ENGINEERED SAFETY FEATURES PROTECTION SYSTEMS (CATEGORY B)	1.4-12
1.4.16 CRITERION 16 - MONITORING REACTOR COOLANT PRESSURE BOUNDARY (CATEGORY B)	1.4-12
1.4.17 CRITERION 17 - MONITORING RADIOACTIVITY RELEASES (CATEGORY B)	1.4-13
1.4.18 CRITERION 18 - MONITORING FUEL AND WASTE STORAGE (CATEGORY B)	1.4-15
1.4.19 CRITERION 19 - PROTECTION SYSTEMS RELIABILITY (CATEGORY B)	1.4-15
1.4.20 CRITERION 20 - PROTECTION SYSTEMS REDUNDANCY AND INDEPENDENCE (CATEGORY B)	1.4-16
1.4.21 CRITERION 21 - SINGLE FAILURE DEFINITION (CATEGORY B)	1.4-16
1.4.22 CRITERION 22 - SEPARATION OF PROTECTION AND CONTROL INSTRUMENTATION SYSTEMS (CATEGORY B)	1.4-16
1.4.23 CRITERION 23 - PROTECTION AGAINST MULTIPLE DISABILITY FOR PROTECTION SYSTEMS (CATEGORY B)	1.4-17
1.4.24 CRITERION 24 - EMERGENCY POWER FOR PROTECTION SYSTEMS (CATEGORY B)	1.4-17
1.4.25 CRITERION 25 - DEMONSTRATION OF FUNCTIONAL OPERABILITY OF PROTECTION SYSTEMS (CATEGORY B)	1.4-17
1.4.26 CRITERION 26 - PROTECTION SYSTEMS FAIL-SAFE DESIGN (CATEGORY B)	1.4-18
1.4.27 CRITERION 27 - REDUNDANCY OF REACTIVITY CONTROL (CATEGORY A)	1.4-19
1.4.28 CRITERION 28 - REACTIVITY HOT SHUTDOWN CAPABILITY (CATEGORY A)	1.4-19
1.4.29 CRITERION 29 - REACTIVITY SHUTDOWN CAPABILITY (CATEGORY A)	1.4-19
1.4.30 CRITERION 30 - REACTIVITY HOLDDOWN CAPABILITY (CATEGORY B)	1.4-20
1.4.31 CRITERION 31 - REACTIVITY CONTROL SYSTEMS MALFUNCTION (CATEGORY B)	1.4-20
1.4.32 CRITERION 32 - MAXIMUM REACTIVITY WORTH OF CONTROL RODS (CATEGORY A)	1.4-20
1.4.33 CRITERION 33 - REACTOR COOLANT PRESSURE BOUNDARY CAPABILITY (CATEGORY A)	1.4-21
1.4.34 CRITERION 34 - REACTOR COOLANT PRESSURE BOUNDARY RAPID PROPAGATION FAILURE PREVENTION (CATEGORY A)	1.4-21
1.4.35 CRITERION 35 - REACTOR COOLANT PRESSURE BOUNDARY BRITTLE FRACTURE PREVENTION (CATEGORY A)	1.4-22
1.4.36 CRITERION 36 - REACTOR COOLANT PRESSURE BOUNDARY SURVEILLANCE (CATEGORY A)	1.4-22

<u>Section</u>	<u>Page</u>
1.4.37 CRITERION 37 - ENGINEERED SAFETY FEATURES BASIS FOR DESIGN (CATEGORY A)	1.4-22
1.4.38 CRITERION 38 - RELIABILITY AND TESTABILITY OF ENGINEERED SAFETY FEATURES (CATEGORY A)	1.4-23
1.4.39 CRITERION 39 - EMERGENCY POWER FOR ENGINEERED SAFETY FEATURES (CATEGORY A)	1.4-24
1.4.40 CRITERION 40 - MISSILE PROTECTION (CATEGORY A)	1.4-24
1.4.41 CRITERION 41 - ENGINEERED SAFETY FEATURES PERFORMANCE CAPABILITY (CATEGORY A)	1.4-25
1.4.42 CRITERION 42 - ENGINEERED SAFETY FEATURES COMPONENTS CAPABILITY (CATEGORY A)	1.4-25
1.4.43 CRITERION 43 - ACCIDENT AGGRAVATION PREVENTION (CATEGORY A)	1.4-26
1.4.44 CRITERION 44 - EMERGENCY CORE COOLING SYSTEMS CAPABILITY (CATEGORY A)	1.4-26
1.4.45 CRITERION 45 - INSPECTION OF EMERGENCY CORE COOLING SYSTEMS (CATEGORY A)	1.4-27
1.4.46 CRITERION 46 - TESTING OF EMERGENCY CORE COOLING SYSTEMS COMPONENTS (CATEGORY A)	1.4-27
1.4.47 CRITERION 47 - TESTING OF EMERGENCY CORE COOLING SYSTEMS (CATEGORY A)	1.4-28
1.4.48 CRITERION 48 - TESTING OF OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEMS (CATEGORY A)	1.4-28
1.4.49 CRITERION 49 - CONTAINMENT DESIGN BASIS (CATEGORY A)	1.4-28
1.4.50 CRITERION 50 - NDT REQUIREMENT FOR CONTAINMENT MATERIAL (CATEGORY A)	1.4-29
1.4.51 CRITERION 51 - REACTOR COOLANT PRESSURE BOUNDARY OUTSIDE CONTAINMENT (CATEGORY A)	1.4-29
1.4.52 CRITERION 52 - CONTAINMENT HEAT REMOVAL SYSTEMS (CATEGORY A)	1.4-30
1.4.53 CRITERION 53 - CONTAINMENT ISOLATION VALVES (CATEGORY A)	1.4-31
1.4.54 CRITERION 54 - CONTAINMENT LEAKAGE RATE TESTING (CATEGORY A)	1.4-31
1.4.55 CRITERION 55 - CONTAINMENT PERIODIC LEAKAGE RATE TESTING (CATEGORY A)	1.4-32
1.4.56 CRITERION 56 - PROVISIONS FOR TESTING OF PENETRATIONS (CATEGORY A)	1.4-32
1.4.57 CRITERION 57 - PROVISIONS FOR TESTING OF ISOLATIONS VALVES (CATEGORY A)	1.4-32
1.4.58 CRITERION 58 - INSPECTION OF CONTAINMENT PRESSURE-REDUCING SYSTEMS (CATEGORY A)	1.4-33
1.4.59 CRITERION 59 - TESTING OF CONTAINMENT PRESSURE-REDUCING SYSTEMS COMPONENTS (CATEGORY A)	1.4-33
1.4.60 CRITERION 60 - TESTING OF CONTAINMENT SPRAY SYSTEMS (CATEGORY A)	1.4-34
1.4.61 CRITERION 61 - TESTING OF OPERATIONAL SEQUENCE OF CONTAINMENT PRESSURE-REDUCING SYSTEMS (CATEGORY A)	1.4-34

<u>Section</u>	<u>Page</u>
1.4.62 CRITERION 62 - INSPECTION OF AIR CLEANUP SYSTEMS (CATEGORY A)	1.4-35
1.4.63 CRITERION 63 - TESTING OF AIR CLEANUP SYSTEMS COMPONENTS (CATEGORY A)	1.4-35
1.4.64 CRITERION 64 - TESTING OF AIR CLEANUP SYSTEMS (CATEGORY A)	1.4-35
1.4.65 CRITERION 65 - TESTING OF OPERATIONAL SEQUENCE OF AIR CLEANUP SYSTEMS (CATEGORY A)	1.4-36
1.4.66 CRITERION 66 - PREVENTION OF FUEL STORAGE CRITICALITY (CATEGORY B)	1.4-36
1.4.67 CRITERION 67 - FUEL AND WASTE STORAGE DECAY HEAT (CATEGORY B)	1.4-36
1.4.68 CRITERION 68 - FUEL AND WASTE STORAGE RADIATION SHIELDING (CATEGORY B)	1.4-37
1.4.69 CRITERION 69 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE (CATEGORY B)	1.4-37
1.4.70 CRITERION 70 - CONTROL OF RELEASES OF RADIOACTIVITY TO THE ENVIRONMENT (CATEGORY B)	1.4-38
1.5 <u>RESEARCH AND DEVELOPMENT REQUIREMENTS</u>	1.5-1
1.5.1 XENON OSCILLATIONS	1.5-1
1.5.2 THERMAL AND HYDRAULIC PROGRAMS	1.5-1
1.5.3 FUEL ROD CLAD FAILURE	1.5-2
1.5.4 HIGH BURNUP FUEL TESTS	1.5-3
1.5.5 INTERNALS VENT VALVES	1.5-3
1.5.6 CONTROL ROD DRIVE TEST	1.5-4
1.5.7 ONCE-THROUGH STEAM GENERATOR TEST	1.5-4
1.5.8 SELF POWERED DETECTOR TESTS	1.5-5
1.5.9 BLOWDOWN FORCES ON INTERNALS	1.5-5
1.5.10 RADIO IODINE SPRAY REMOVAL SYSTEM	1.5-6
1.6 <u>SMUD'S COMPETENCE TO BUILD AND OPERATE NUCLEAR PLANT</u>	1.6-1
1.7 <u>IDENTIFICATION OF CONTRACTORS AND AGENTS</u>	1.7-1
1.8 <u>CONCLUSIONS</u>	1.8-1
1.9 <u>REFERENCES</u>	1.9-1
2. <u>SITE AND ENVIRONMENT</u>	
2.1 <u>SUMMARY</u>	2.1-1
2.2 <u>SITE AND ADJACENT AREAS</u>	2.2-1
2.2.1 SITE LOCATION	2.2-1
2.2.2 SITE OWNERSHIP	2.2-1
2.2.3 SITE ACTIVITIES	2.2-1
2.2.4 POPULATION	2.2-1
2.2.5 LAND USE	2.2-3
2.2.6 ACCESS AND EGRESS	2.2-3
2.2.7 MAKE-UP WATER SUPPLY	2.2-5

<u>Section</u>	<u>Page</u>
2.3 <u>METEOROLOGY</u>	2.3-1
2.3.1 INTRODUCTION	2.3-1
2.3.2 DESCRIPTIVE METEOROLOGY	2.3-1
2.3.3 METEOROLOGICAL DATA	2.3-2
2.3.4 PROGRAM OF METEOROLOGICAL INVESTIGATION	2.3-5
2.3.5 PRELIMINARY ESTIMATES OF DIFFUSION	2.3-5
2.4 <u>HYDROLOGY</u>	2.4-1
2.4.1 CHARACTERISTICS OF STREAMS AND LAKES IN VICINITY	2.4-1
2.4.2 TOPOGRAPHY	2.4-1
2.4.3 TERMINAL DISPOSAL OF STORM RUNOFF	2.4-1
2.4.4 HISTORICAL FLOODING	2.4-1
2.4.5 PREDICTION OF LAND URBANIZATION	2.4-1
2.4.6 GROUNDWATER	2.4-3
2.5 <u>GEOLOGY</u>	2.5-1
2.6 <u>SEISMOLOGY</u>	2.6-1
2.7 <u>SOILS</u>	2.7-1
2.8 <u>SITE ENVIRONMENTAL RADIOACTIVITY PROGRAM</u>	2.8-1
2.8.1 GENERAL	2.8-1
2.8.2 LAND ENVIRONMENT	2.8-1
2.8.3 WATER ENVIRONMENT	2.8-2
2.8.4 SAMPLING	2.8-2
2.9 <u>REFERENCES</u>	2.9-1
3. <u>REACTOR</u>	
3.1 <u>DESIGN BASES</u>	3.1-1
3.1.1 PERFORMANCE OBJECTIVES	3.1-1
3.1.2 LIMITS	3.1-1
3.2 <u>REACTOR DESIGN</u>	3.2-1
3.2.1 GENERAL SUMMARY	3.2-1
3.2.2 NUCLEAR DESIGN AND EVALUATION	3.2-2
3.2.3 THERMAL AND HYDRAULIC DESIGN AND EVALUATION	3.2-29
3.2.4 MECHANICAL DESIGN LAYOUT	3.2-70
3.3 <u>TESTS AND INSPECTIONS</u>	3.3-1
3.3.1 NUCLEAR TESTS AND INSPECTION	3.3-5
3.3.2 THERMAL AND HYDRAULIC TESTS AND INSPECTION	3.3-2
3.3.3 FUEL ASSEMBLY, CONTROL ROD ASSEMBLY, AND CONTROL ROD DRIVE MECHANICAL TESTS AND INSPECTION	3.3-5
3.3.4 INTERNALS TESTS AND INSPECTIONS	3.3-10
3.4 <u>REFERENCES</u>	3.4-1

VOLUME II

<u>Section</u>		<u>Page</u>
	4. <u>REACTOR COOLANT SYSTEM</u>	
4.1	<u>DESIGN BASES</u>	4.1-1
4.1.1	PERFORMANCE OBJECTIVES	4.1-1
4.1.2	DESIGN CHARACTERISTICS	4.1-2
4.1.3	EXPECTED OPERATING CONDITIONS	4.1-7
4.1.4	SERVICE LIFE	4.1-8
4.1.5	CODES AND CLASSIFICATIONS	4.1-15
4.2	<u>SYSTEM DESCRIPTION AND OPERATION</u>	4.2-1
4.2.1	GENERAL DESCRIPTION	4.2-1
4.2.2	MAJOR COMPONENTS	4.2-1
4.2.3	PRESSURE-RELIEVING DEVICES	4.2-7
4.2.4	ENVIRONMENTAL PROTECTION	4.2-7
4.2.5	MATERIALS OF CONSTRUCTION	4.2-7
4.2.6	MAXIMUM HEATING AND COOLING RATES	4.2-11
4.2.7	LEAK DETECTION	4.2-11
4.3	<u>SYSTEM DESIGN EVALUATION</u>	4.3-1
4.3.1	SAFETY FACTORS	4.3-1
4.3.2	RELIANCE ON INTERCONNECTED SYSTEMS	4.3-8
4.3.3	SYSTEM INTEGRITY	4.3-9
4.3.4	PRESSURE RELIEF	4.3-9
4.3.5	REDUNDANCY	4.3-10
4.3.6	SAFETY ANALYSIS	4.3-10
4.3.7	OPERATIONAL LIMITS	4.3-10
4.4	<u>TESTS AND INSPECTIONS</u>	4.4-1
4.4.1	COMPONENT IN-SERVICE INSPECTION	4.4-1
4.4.2	REACTOR COOLANT SYSTEM TESTS AND INSPECTIONS	4.4-3
4.4.3	MATERIAL IRRADIATION SURVEILLANCE	4.4-4
4.5	<u>REFERENCES</u>	4.5-1
	5. <u>CONTAINMENT SYSTEM</u>	
5.1	<u>STRUCTURAL DESIGN</u>	5.1-1
5.1.1	GENERAL DESCRIPTION OF CONTAINMENT STRUCTURE	5.1-1
5.1.2	BASIS FOR DESIGN LOADS	5.1-1
5.1.3	CONSTRUCTION MATERIALS	5.1-4
5.1.4	CONTAINMENT STRUCTURE DESIGN CRITERIA	5.1-11
5.1.5	STRUCTURAL DESIGN ANALYSIS	5.1-27
5.2	<u>DESIGN, CONSTRUCTION, AND TESTING OF PENETRATIONS</u>	5.2-1
5.2.1	TYPES OF PENETRATIONS	5.2-1
5.2.2	DESIGN OF PENETRATIONS	5.2-3
5.2.3	INSTALLATION OF PENETRATIONS	5.2-5
5.2.4	TESTABILITY OF PENETRATIONS AND WELD SEAMS	5.2-5

<u>Section</u>	<u>Page</u>
5.3 <u>CONTAINMENT ACCESSIBILITY CRITERIA</u>	5.3-1
5.4 <u>CONSTRUCTION PRACTICES AND QUALITY ASSURANCE</u>	5.4-1
5.4.1    ORGANIZATION OF QUALITY ASSURANCE PROGRAM	5.4-1
5.4.2    APPLICABLE CONSTRUCTION CODES	5.4-1
5.4.3    CONSTRUCTION MATERIALS INSPECTION AND INSTALLATION	5.4-2
5.4.4    SPECIFIC CONSTRUCTION TOPICS	5.4-7
5.5 <u>CONTAINMENT SYSTEM INSPECTION, TESTING, AND SURVEILLANCE</u>	5.5-1
5.5.1    TESTS TO ENSURE LINER INTEGRITY	5.5-1
5.5.2    STRENGTH TEST	5.5-3
5.6 <u>ISOLATION SYSTEM</u>	5.6-1
5.6.1    DESIGN BASES	5.6-1
5.6.2    SYSTEM DESIGN	5.6-1
5.7 <u>VENTILATION SYSTEM</u>	5.7-1
5.7.1    DESIGN BASES	5.7-1
5.7.2    SYSTEM DESIGN	5.7-1
5.8 <u>LEAKAGE MONITORING SYSTEM</u>	5.8-1
5.9 <u>SYSTEM DESIGN EVALUATION</u>	5.9-1
6. <u>ENGINEERED SAFEGUARDS</u>	
6.1 <u>EMERGENCY INJECTION</u>	6.1-1
6.1.1    DESIGN BASES	6.1-1
6.1.2    DESCRIPTION	6.1-1
6.1.3    DESIGN EVALUATION	6.1-5
6.1.4    TEST AND INSPECTIONS	6.1-15
6.2 <u>REACTOR BUILDING ATMOSPHERE COOLING AND WASHING</u>	6.2-1
6.2.1    DESIGN BASES	6.2-1
6.2.2    DESCRIPTION	6.2-1
6.2.3    DESIGN EVALUATION	6.2-2
6.2.4    TESTS AND INSPECTIONS	6.2-7
6.3 <u>ENGINEERED SAFEGUARDS LEAKAGE AND RADIATION CONSIDERATIONS</u>	6.3-1
6.3.1    INTRODUCTION	6.3-1
6.3.2    SUMMARY OF POSTACCIDENT RECIRCULATION AND LEAKAGE CONSIDERATION	6.3-1
6.3.3    LEAKAGE ASSUMPTIONS	6.3-2
6.3.4    DESIGN BASIS LEAKAGE	6.3-2
6.3.5    LEAKAGE ANALYSIS CONCLUSIONS	6.3-2

007

<u>Section</u>	<u>Page</u>
<u>7. INSTRUMENTATION AND CONTROL</u>	
7.1 <u>PROTECTION SYSTEMS</u>	7.1-1
7.1.1 DESIGN BASES	7.1-1
7.1.2 SYSTEM DESIGN	7.1-6
7.1.3 SYSTEMS EVALUATION	7.1-17
7.2 <u>REGULATING SYSTEMS</u>	7.2-1
7.2.1 DESIGN BASES	7.2-1
7.2.2 SYSTEM DESIGN	7.2-3
7.2.3 SYSTEM EVALUATION	7.2-9
7.3 <u>INSTRUMENTATION</u>	7.3-1
7.3.1 NUCLEAR INSTRUMENTATION	7.3-1
7.3.2 NONNUCLEAR PROCESS INSTRUMENTATION	7.3-3
7.3.3 INCORE MONITORING SYSTEM	7.3-5
7.4 <u>OPERATING CONTROL STATIONS</u>	7.4-1
7.4.1 GENERAL LAYOUT	7.4-1
7.4.2 INFORMATION DISPLAY AND CONTROL FUNCTION	7.4-1
7.4.3 SUMMARY OF ALARMS	7.4-2
7.4.4 COMMUNICATION	7.4-2
7.4.5 OCCUPANCY	7.4-2
7.4.6 AUXILIARY CONTROL STATIONS	7.4-3
7.4.7 SAFETY FEATURES	7.4-4
7.4.8 SYSTEM EVALUATION	7.4-4
<u>8. ELECTRICAL SYSTEMS</u>	
8.1 <u>DESIGN BASIS</u>	8.1-1
8.2 <u>ELECTRICAL SYSTEM DESIGN</u>	8.2-1
8.2.1 ELECTRICAL SYSTEM DESIGN NETWORK INTERCONNECTIONS	8.2-1
8.2.2 STATION DISTRIBUTION SYSTEM	8.2-2
8.2.3 EMERGENCY POWER SYSTEM	8.2-9
8.3 <u>DESIGN EVALUATION</u>	8.3-1
8.3.1 EVALUATION OF THE PHYSICAL LAYOUT	8.3-1
8.3-2 ACCIDENTAL PHASE REVERSAL	8.3-2
8.4 <u>TESTS AND INSPECTIONS</u>	8.4-1
<u>9. AUXILIARY AND EMERGENCY SYSTEMS</u>	
9.1 <u>MAKEUP AND PURIFICATION SYSTEM</u>	9.1-1
9.1.1 DESIGN BASES	9.1-1
9.1.2 SYSTEM DESCRIPTION AND EVALUATION	9.1-2
9.2 <u>CHEMICAL ADDITION AND SAMPLING SYSTEM</u>	9.2-1
9.2.1 DESIGN BASES	9.2-1
9.2.2 SYSTEM DESCRIPTION AND EVALUATION	9.2-1



<u>Section</u>	<u>Page</u>
9.3 <u>COOLING WATER SYSTEMS</u>	9.3-1
9.3.1 DESIGN BASES	9.3-1
9.3.2 SYSTEM DESCRIPTION AND EVALUATION	9.3-1
9.4 <u>SPENT FUEL COOLING SYSTEM</u>	9.4-1
9.4.1 DESIGN BASES	9.4-1
9.4.2 SYSTEM DESCRIPTION AND EVALUATION	9.4-1
9.5 <u>DECAY HEAT REMOVAL SYSTEM</u>	9.5-1
9.5.1 DESIGN BASES	9.5-1
9.5.2 SYSTEM DESCRIPTION AND EVALUATION	9.5-1
9.6 <u>FUEL HANDLING SYSTEM</u>	9.6-1
9.6.1 DESIGN BASES	9.6-1
9.6.2 SYSTEM DESCRIPTION AND EVALUATION	9.6-2
9.7 <u>STATION VENTILATION SYSTEMS</u>	9.7-1
9.7.1 DESIGN BASES	9.7-1
9.7.2 SYSTEM DESCRIPTION AND EVALUATION	9.7-1

VOLUME III

10. STEAM AND POWER CONVERSION SYSTEM

10.1 <u>DESIGN BASES</u>	10.1-1
10.1.1 OPERATING AND PERFORMANCE REQUIREMENTS	10.1-1
10.1.2 ELECTRICAL SYSTEM CHARACTERISTICS	10.1-1
10.1.3 FUNCTIONAL LIMITATIONS	10.1-1
10.1.4 SECONDARY FUNCTIONS	10.1-1
10.2 <u>SYSTEM DESIGN AND OPERATION</u>	10.2-1
10.2.1 SCHEMATIC FLOW DIAGRAM	10.2-1
10.2.2 CODES AND STANDARDS	10.2-1
10.2.3 DESIGN FEATURES	10.2-2
10.2.4 SHIELDING	10.2-2
10.2.5 CORROSION PROTECTION	10.2-2
10.2.6 IMPURITIES CONTROL	10.2-2
10.2.7 RADIOACTIVITY	10.2-2
10.3 <u>SYSTEM ANALYSIS</u>	10.3-1
10.3.1 TRIPS, AUTOMATIC CONTROL ACTIONS, AND ALARMS	10.3-1
10.3.2 TRANSIENT CONDITIONS	10.3-2
10.3.3 MALFUNCTIONS	10.3-2
10.3.4 OVERPRESSURE PROTECTION	10.3-2
10.3.5 INTERACTIONS	10.3-2
10.3.6 OPERATIONAL LIMITS	10.3-2
10.4 <u>TESTS AND INSPECTIONS</u>	10.4-1

Section

Page

11. RADIOACTIVE WASTES AND RADIATION PROTECTION

11.1	<u>RADIOACTIVE WASTE HANDLING</u>	11.1-1
11.1.1	DESIGN BASES	11.1-1
11.1.2	SYSTEM DESIGN AND EVALUATION	11.1-6
11.1.3	TESTS AND INSPECTIONS	11.1-8
11.1.4	TRITIUM MANAGEMENT FOR NORMAL OPERATION	11.1-8
11.2	<u>RADIATION SHIELDING</u>	11.2-1
11.2.1	PRIMARY, SECONDARY, REACTOR BUILDING, AND AUXILIARY SHIELDING	11.2-1
11.2.2	AREA RADIATION MONITORING SYSTEM	11.2-6
11.2.3	HEALTH PHYSICS	11.2-8
11.3	<u>REFERENCES</u>	11.3-1

12. CONDUCT OF OPERATIONS

12.1	<u>INTRODUCTION</u>	12.1-1
12.2	<u>ORGANIZATION AND RESPONSIBILITY</u>	12.2-1
12.3	<u>PERSONNEL TRAINING</u>	12.3-1
12.3.1	TRAINING INITIAL PLANT STAFF	12.3-1
12.3.2	REPLACEMENT AND REFRESHER TRAINING	12.3-4
12.3.3	EMERGENCY DRILLS	12.3-5
12.4	<u>WRITTEN PROCEDURE</u>	12.4-1
12.5	<u>RECORD</u>	12.5-1
12.6	<u>ADMINISTRATIVE CONTROLS</u>	12.6-1
12.7	<u>INDEPENDENT AUDIT OF PLANT OPERATIONS</u>	12.7-1

13. INITIAL TESTS AND OPERATION

13.1	<u>TESTS PRIOR TO REACTOR FUELING</u>	13.1-1
13.2	<u>INITIAL CRITICALITY</u>	13.2-1
13.3	<u>POSTCRITICALITY TESTS</u>	13.3-1

Section

Page

14. SAFETY ANALYSIS

14.1	<u>CORE AND COOLANT BOUNDARY PROTECTION ANALYSIS</u>	14.1-1
14.1.1	ABNORMALITIES	14.1-1
14.1.2	ANALYSIS OF EFFECTS AND CONSEQUENCES	14.1-2
14.2	<u>STANDBY SAFEGUARDS ANALYSIS</u>	14.2-1
14.2.1	SITUATIONS ANALYZED AND CAUSES	14.2-1
14.2.2	ACCIDENT ANALYSES	14.2-1
14.3	<u>ENVIRONMENTAL CONSEQUENCES OF HYPOTHETICAL ACCIDENTS</u>	14.3-1
14.3.1	GENERAL APPROACH	14.3-1
14.3.2	STEAM GENERATOR TUBE FAILURE	14.3-1
14.3.3	LOSS OF ELECTRIC POWER	14.3-1
14.3.4	STEAM LINE FAILURE	14.3-3
14.3.5	FUEL HANDLING ACCIDENT	14.3-4
14.3.6	ROD EJECTION ACCIDENT	14.3-4
14.3.7	WASTE GAS TANK RUPTURE	14.3-4
14.3.8	LOSS-OF-COOLANT ACCIDENT	14.3-5
14.3.9	MAXIMUM HYPOTHETICAL ACCIDENT	14.3-6
14.3.10	IODINE REMOVAL SENSITIVITY ANALYSIS	14.3-10
14.3.11	POPULATION DENSITY CONSIDERATIONS	14.3-12
14.4	<u>REFERENCES</u>	14.4-1

15. TECHNICAL SPECIFICATIONS

VOLUME IV

APPENDIX 1

- 1A ANSWERS TO QUESTIONS
- 1B QUALITY ASSURANCE OPERATIONS
- 1C RANCHO SECO PROJECT ENGINEERING STAFF

APPENDIX 2

- 2A METEOROLOGY
- 2B LAND USAGE AND POPULATION
- 2C GEOLOGY AND SEISMOLOGY
- 2D SEISMIC REPORT
- 2E SOIL AND FOUNDATIONS INVESTIGATION REPORT
- 2F METEOROLOGICAL STATION
- 2G STORAGE RESERVOIR
- 2H ANSWERS TO QUESTIONS

APPENDIX 3

- 3A ANSWERS TO QUESTIONS

APPENDIX 4

- 4A ANSWERS TO QUESTIONS

012

VOLUME V

APPENDIX 5

- 5A STRUCTURAL DESIGN BASES
- 5B JUSTIFICATION OF STRUCTURAL PROOF TEST - PRESSURES
- 5C SPECIFICATION FOR SPLICING REINFORCING BAR USING THE CADWELD PROCESS
- 5D TURBINE GENERATOR MISSILES
- 5E JUSTIFICATION FOR LOAD FACTORS
- 5F JUSTIFICATION FOR YIELD REDUCTION FACTORS
- 5G DESCRIPTION OF THE FINITE ELEMENT TECHNIQUE USED IN CONTAINMENT STRUCTURAL ANALYSIS
- 5H QUALITY CONTROL PROCEDURE FOR FIELD WELDING
- 5I CONTAINMENT STRUCTURE INSTRUMENTATION
- 5J ANSWERS TO QUESTIONS

APPENDIX 6

- 6A ANSWERS TO QUESTIONS

APPENDIX 7

- 7A ANSWERS TO QUESTIONS

APPENDIX 8

- 8A ANSWERS TO QUESTIONS

013

APPENDIX 9

9A ANSWERS TO QUESTIONS

APPENDIX 11

11A ANSWERS TO QUESTIONS

APPENDIX 12

12A ANSWERS TO QUESTIONS

APPENDIX 13

13A ANSWERS TO QUESTIONS

APPENDIX 14

14A ANSWERS TO QUESTIONS

APPENDIX 15

15A ANSWERS TO QUESTIONS

014

CONTENTS

APPENDIX 1

- 1A ANSWERS TO QUESTIONS
- 1B QUALITY ASSURANCE OPERATIONS
- 1C RANCHO SECO PROJECT ENGINEERING STAFF

015





QUESTION  
1A.2

Discuss in detail the scope of the following research, development, or test programs including projected completion dates for various phases of the programs and test equipment descriptions. To the extent possible, results of the programs to date should be stated.

- 1A.2-1 Thermal design, including DNB and flow distribution. (Will loss of a core barrel check valve be simulated in the flow tests?)
- 1A.2-2 Control rod drives.
- 1A.2-3 Steam generator including blowdown tests. (Discuss the desirability of insulating or otherwise maintaining the shell at a high temperature to simulate the thermal transient that might be experienced in the actual generator during secondary system blowdown.)
- 1A.2-4 Core barrel check valves. (Discuss the program for testing the valves or a scaled prototype under operational and accident flow and temperature conditions including vibrational effects during operation and mechanical forces during blowdown.)
- 1A.2-5 Material tests at high burnup. (Discuss which material properties are critical, the results expected and the manner in which the results will be used. Could significant data of a confirmatory nature be obtained by removing and testing fuel from the reactor environment at intervals in the future. If other test programs, currently in progress, are relied on for fuel rod failure mechanisms, describe the scope and schedule of these tests and compare your requirements in detail.)

ANSWER  
Refer to  
3.3.2

1A.2-1 Thermal Design

a. Departure from Nucleate Boiling Heat Transfer Investigation

In the late fall of 1961, The Babcock & Wilcox Company began the design and construction of a large heat transfer facility for the purpose of doing DNB testing at power reactor operating conditions. In this facility, which is located at the B&W Research Center, Alliance, Ohio, testing over a wide range of variables covering practically all of the situations one might expect to encounter during normal and expected transient operation of water cooled reactors is possible.

QUESTION  
1A.1

What is your criterion for a minimum shutdown margin during operational transients?

ANSWER  
Refer to  
1.4.29

Minimum Shutdown Margin - Operational Transients

The reactor is designed to meet the criterion that it can be shut down to the hot subcritical condition with a margin of at least 1%  $\Delta k/k$  with one control rod stuck out of the core. The evaluation of operational transients - such as moderator dilution without rod motion, loss of pumping power, and rod withdrawal - has shown that this margin is not changed by these transients, because the reactor returns to the hot subcritical condition at the end of the transient. This margin at the hot shutdown condition also provides sufficient shutdown reactivity to keep the reactor subcritical in accident-induced transients which cool the reactor coolant to lower temperatures, such as a steam line failure.

cold walls, inter-channel mixing, and instabilities. Data from these assemblies are still being analyzed, and work is progressing on a new DNB correlation. Results to date indicate that the analytical methods used in the design of the reactor core are conservative and that no critical areas exist.

The Babcock & Wilcox Company intends to maintain an active and aggressive program in the field of DNB heat transfer. Some of the principal programs which will be conducted in the near future are outlined below along with a tentative schedule:

- (1) A 9-rod assembly with the same rod diameter, pitch spacing and spacer grid used in the fuel assembly will be tested. A nonuniform axial power generation profile will be employed over six feet of the bundle length. The power profile will be representative of that portion of the core experiencing the most severe heat transfer conditions and the most probable location of a DNB. Testing of this assembly is scheduled for the first two quarters of 1968.
- (2) An annular specimen with nonuniform power distribution on the outside tube will be tested for additional verification of the effects of length and nonuniform power generation. Power may be supplied to various portions of the specimen so that length effects up to the full 12-foot long test region of the specimen may be examined. Testing for the annular specimen is expected to begin in the fourth quarter of 1968.
- (3) A 9-rod bundle test employing nonuniform radial power distribution will be tested in 1969. A definitive program and schedule for this series of tests is not formulated.
- (4) Depending upon the results obtained from the previous tests, additional tests will be devised as part of the continuing basic heat transfer and core optimization program. Tests under consideration are for additional radial and axial power distributions, larger test assemblies, investigation of different grid designs, and transient simulation.

b. Mixing Studies

Related to the studies for DNB are additional programs conducted to determine the degree of mixing in the fuel rod channels. Flow tests involving a 4-rod assembly have been conducted to determine mixing effects. Flow tests on a mockup of the outer two rows of fuel rods and the can panels

The facility is supplied with 1.8 megawatts of d-c power and a fully automated data acquisition system. It can be operated within the following limits:

Pressure - 100 to 2,700 psia

Inlet subcooling - 20 to 250 F

Mass velocity -  $0.2 \times 10^6$  to  $3.5 \times 10^6$  lbs/hr-ft<sup>2</sup> in a 9-rod assembly.

Present specimen size - 9-rod assembly with a heated length of six feet.

DNB detection with thermocouples (resistance measurement back-up)

Flow - 150 gpm at 295 ft head

Since the loop has been completed, a variety of experiments have been performed to gain better understanding of the DNB phenomena and to develop empirical relationships necessary for the design of water reactors. Among the experiments completed to date have been tests on:

- (1) Single tubular specimens with both uniform and nonuniform power distributions. Nonuniform axial peak-to-average powers as high as 1.9, simulating inlet and outlet peak locations, have been included in the tests. These tests were conducted as a function of shape, length, and system parameters. On the basis of these tests, power shape factors for application of the test results to reactor design have been determined, and it was concluded that simulation of reactor axial power shapes could be achieved with confidence in test bundles of shorter length than actual reactor fuel assemblies.
- (2) Annular specimens with various combinations of inner and outer wall heat generation and nonuniform axial power distributions were also tested. It was determined that the results obtained with the annular data correlated very well with data taken on bundles. Analytical work done on the tubular and annular specimens has formed the basis for the bundle size and power generation shape to be used in a future test bundle described below.
- (3) A 9-rod test assembly with a uniformly heated length of six feet, simulating the reactor fuel rod diameter, pitch spacing, and spacer grid details, has been tested. This was the first test approaching actual geometrical conditions as well as operating conditions for the core. Of principal interest were the effects of spacer grids,

top and bottom of each assembly. Pressure sensors and thermocouples are provided in all fuel assemblies to determine the flow distribution at the core inlet. Additional pressure sensors and thermocouples are provided in other portions of the vessel and core so that overall mixing and pressure drop determinations may be made.

Preliminary investigations in the model and the analysis described in the appendix to Section 3.2.4 herein (Item 5.1.5) indicate that it will not be necessary to simulate the internal check valve construction, operation, or any malfunctions in the vessel model flow tests.

Testing should be completed in mid-1968.

Refer to 1A.2-2 Control Rod Drives  
3.3.3

a. Component Tests

The purpose of this program is to seek out potential material and/or design problems prior to production unit testing. The component test program consists of:

- (1) Evaluation of various grades of Graphitar-bearing materials in an autoclave at 1,600 psi, 600 F, and water chemistry with 13,000 ppm  $H_3BO_3$ . The bearing materials are statically loaded against a 17-4 PH shaft such that the developed stress is greater than will be present in the actual control rod drive.
- (2) Environmental dynamic gear and bearing tests under loads equivalent to the control rod drive operating conditions. In this test, the bevel gears, the pinion, and the bearings supporting these gears are being tested in an autoclave at 2,250 psi, 400 F, and reactor water chemistry. The purpose of this test is to obtain wear characteristics of the gear material combinations and projected life of the bearings.
- (3) Simulated drive test. A complete mechanism, which simulates the drive with the exception of its overall length, is being tested under no-flow reactor operating conditions of temperature and pressure in an autoclave. An accelerated wear and life test through a short stroke will be completed in conjunction with the life-testing of the prototype mechanism.
- (4) Autoclave testing at reactor operating temperature and pressure of buffer seal, splines, and bearings. In this test, the spline joints of the drive rod assembly are being tested under static, no-load conditions for corrosion.

of two adjacent fuel assemblies have been conducted to determine the friction effects at the perforated wall boundary. These tests have confirmed that the larger flow cells at the periphery of the bundle compensate for higher equivalent friction adequately. These effects are shown numerically in the appendix to Section 3.2.3.2.4j herein. Additional tests to extend the investigation to larger sizes, and more elaborate geometry for the purpose of confirming the analytical model and value for mixing coefficients, are described below.

- (1) A 9-rod mixing test assembly, of the same bundle geometry as the DNB bundle described previously, has been constructed to determine the degree of mixing present during the DNB investigations. Testing with this assembly is currently in progress and is expected to be completed in the first two quarters of 1968.
- (2) A 16-rod assembly with the simulated juncture of four perforated, fuel assembly cans meeting at the corner is under construction. Testing with this assembly will enable one to determine the degree of mixing which occurs between fuel assemblies, and will give more detailed information on velocity distributions and mixing in the peripheral cells of the fuel assembly than did the 4-rod tests. The current core analysis considers only mixing within a fuel assembly and does not take credit for mixing external to the assembly. It is expected that testing with this assembly will begin in January 1968.
- (3) A facility large enough to accept a 64-rod assembly is currently under construction. Tests for this facility are not yet firm, but it is expected that some of the preliminary work for calibration of in-core thermocouples and pressure differential instrumentation will be done in this facility. Initial plans were to construct a low pressure facility large enough to accept a full size, cross section fuel assembly. This has currently been replaced with the 64-rod assembly, and its need will be re-evaluated. Testing in this facility is scheduled for the second quarter of 1968.

c. Vessel Model Flow Tests

A 1/6 scale model of the reactor vessel, the internals, and the reactor coolant piping from the pumps to the reactor vessel is currently being tested at the Research Center. Portions of the reactor vessel and internals are constructed of transparent plastic to facilitate visual observation of flow patterns within the vessel. The reactor core is simulated in the model with individual fuel assemblies constructed of perforated sheet material and calibrated orifices at the

In addition to the testing described in the above reference, secondary system blowdown tests have been carried out, and a reactor coolant (primary) side blowdown is planned when schedule commitments permit. The hot water facility of the Research Center is shared by the Control Rod Drive Tests, the Steam Generator Tests, and other experiments and tests.

Three secondary system blowdown tests have been completed. The results of these tests have demonstrated the integrity of the steam generator under conditions of rapid depressurization and large (greater than 200 F), tube-to-shell temperature differentials.

In addition, the results of these tests are used in the development and verification of analytical models for steam system blowdown analyses.

The construction of the test steam generator (including insulation) is such that the thermal time constant of the shell is lower than that of a full-scale unit. This lower time constant results in more rapid cooling of the shell during steam system blowdown than would occur in a full-size unit.

The primary side blowdown test will provide temperature conditions which simulate a thermal transient greater than that for the full-scale unit secondary blowdown as well as simulation of the thermal transient for primary blowdown.

Refer to 1A.2-4 Core Barrel Vent Valves  
3.3.4

The core barrel vent valves will be designed to relieve the pressure generated by steaming in the core following the LOCA so that the core will remain sufficiently covered. The valves will also be designed to withstand the forces resulting from rupture of either a reactor coolant inlet or outlet pipe. Testing of the valves will consist of the following:

- a. A full-size valve assembly (seat, locking mechanism, and socket) will be tested at steady-state conditions at the maximum pressure expected to result during the blowdown.
- b. Sufficient tests will be conducted at zero pressure to determine the frictional loads and clearances in the hinge assembly, the inertia of the valve cover, and the deflections resulting from impact of the cover so that the valve response to cyclic blowdown forces may be determined analytically.

023

- (5) Autoclave testing of shortened drive rod assembly under static load conditions. This test is similar to the spline testing with the addition of the bevel gear set at the lower end of the drive rod.
- (6) Autoclave testing at reactor temperature and pressure of the bevel gears, bearings, shortened rack, and pinion gear under vibratory loading of the rack to determine the fretting characteristics of the gear train. This test is a static load test.

b. Full Scale Prototype Testing Under No Flow Conditions

This test will be performed in an autoclave permitting full stroking in room temperature water and at reactor operating conditions of temperature, pressure, and water chemistry. The cold tests will be utilized only as an initial checkout of the drive prior to temperature and pressure testing. The control rod will be simulated with a dummy weight. Misalignment will be introduced to note its effect on wear and overall performance of the drive mechanism.

c. Full Scale Prototype Testing at Reactor Operating Conditions of Temperature, Pressure, and Flow

The full life test program as defined in the PSAR will be conducted under this test. A prototype control rod and fuel assembly will be used in order to establish the complete drive train assembly.

Inasmuch as the rack and pinion drive concept described in Section 3.2.4.3.2 herein is somewhat different from the first rack and pinion drive tested at the Research Center, the test program which has been outlined above provides an extension of previous tests to establish verification of drive performance and adequacy. The previous test program verified the basic material selection, the snubber design, and the buffer seal concept for use with a rack and pinion drive.

The components test program (items a and b) is scheduled for completion by the end of December 1967. Hot loop testing (item c) at full flow conditions of the prototype drive will begin in October 1967, and continue until the end of December 1967.

Refer to 1A.2-3 Steam Generator  
4.4

The basic steam generator test program is discussed in detail in Appendix 4A of the Duke Power Company PSAR (Dockets 50-269, 270, and 287).

024



The fuels irradiation program will test fuel specimens at design temperatures and at exposures in excess of those obtained in the fuel rod. The specimens irradiated to the design burnup are scheduled to be completed in mid-1969, well in advance of reactor operation. The program will provide information on the swelling rate of  $U_2$  as a function of burnup, density, heat rate, and cladding restraint. Fuel specimens will be operated at heat rates up to 21.5 kw/ft, which is in excess of the peak specific power in the core. The burnup will range up to 75,000 MWD/TU. The fuel rods will operate with a cladding surface temperature of 650 F.

A program has been carried out to determine the effects of irradiation on the mechanical properties of Zircaloy-4.<sup>1</sup> Tests were conducted to temperatures as high as 775 F. The summary of results from this program is as follows:

- a. The room temperature tensile and yield strengths of Zircaloy-4 increased with total neutron exposure for irradiation temperatures up to 650 F. The rate of increase was greater at lower irradiation temperatures. This increase in strength was accompanied by a decrease in the total and uniform elongations.
- b. The room temperature yield and tensile strengths of the specimens irradiated at 775 F were somewhat lower than those of the specimens before irradiation. These changes in properties, however, were not significantly different from those observed in specimens aged out-of-pile for like periods of time.
- c. The room temperature uniform elongation values for both annealed and cold-worked material were approximately 2 percent after neutron irradiation at 130 F to an exposure of  $4.5 \times 10^{19}$  nvt ( $E > 1$  Mev).
- d. A difference in irradiation behavior was noted for the longitudinal and transverse specimens, particularly after irradiation at 775 F. At this temperature the tensile strength in the transverse direction continued to increase whereas in the longitudinal specimens the strength decreased.

A summary of capsule specimens is given in Table 1A.2-1 and a tentative schedule is presented in Figure 1A.2-1.

The following is a description of the research now in progress at B&W that is related to the current reactor design.

025

- c. The valve assembly will be pressurized to determine what pressure differential is required to cause the valve to begin to open. A determination of the pressure differential required to open the valve to its maximum open position will be simulated by mechanical means.
- d. A valve assembly will be installed and removed remotely in a test stand to judge the adequacy of handling equipment.

Since the temperature differential existing across the valve assembly during normal operation in the reactor is only approximately 55 F, and since the same material is used for the valve seat, socket, and cover, there is no need to conduct tests at elevated temperatures.

The valves are located in a region of relatively low velocity and turbulence, and preliminary analysis indicates that there is insufficient energy in the coolant to cause vibrational problems. Therefore, no testing to prove the vibrational adequacy of the valve is planned.

Testing should be completed by January 1969.

Refer to 1A.2-5 High Burnup Fuel Tests  
3.2.4.2.2

The design of fuel rods for pressure cycles and thermal gradients are amendable to analysis, based on out-of-pile properties. In determining the behavior of materials under the influence of accumulated irradiation the properties of interest are uranium dioxide growth rates under restraint by tubular cladding, and the ability of the cladding to absorb strain without failure at reactor operating conditions.

A detailed report of sources of information for the irradiation of clad and fuel has been presented in the PSAR, 3.2.4.2.2 plus references. In addition to the PSAR references, irradiation of fuel assemblies or partial fuel assemblies with Zircaloy-clad UO<sub>2</sub> is in progress in the Saxton and Big Rock Point reactors. These data will demonstrate the behavior of fuel assemblies under the combined effects of irradiation, pressure cycles, thermal gradients, reactor coolant environment, and fuel-clad restraints.

B&W is conducting a program to obtain a better understanding of fuel growth rates and irradiation effects on cladding, the influence of hydrogen on cladding, and fission gas release at high burnup for the specific design burnup projected for peak power regions in the reactor.

### Task III - High Burnup Fuel Irradiation

The primary purpose of the High Burnup Program is to determine the swelling rate of  $UO_2$  as a function of burnup using fuel rods of the same design as the core. In addition to determining the swelling rate, the effect of several other variables including the density, heat rate, and cladding restraint will be investigated.

The program consists of capsules some of which will operate at a heat rate of 18 kw/ft and others at a heat rate of 21-1/2 kw/ft. The pellets, other than U-235 content, will conform to the reactor fuel specifications. The burnup will range from 10,000 to 75,000 MWD/TU with six capsules exceeding 45,000 MWD/TU. The capsules will not operate with an external pressure. However, two different cladding thicknesses, 0.015 and 0.025 in. will be used to vary the restraint offered by the cladding. The fuel rods will operate with a cladding surface temperature of 650 F. The diametral gaps between the pellets and cladding will vary from 4-5 to 7-8 mils, to give smeared densities of about 92.3 and 90.8 percent, respectively. These gaps and smeared densities are consistent with the fuel rod specifications. The insertion date for the first capsule was September 5, 1967.

The tests are oriented toward the determination of the behavior of materials in an irradiation environment and to determine the optimum geometric and material properties for the specific application. The information is essential for advancement of the art, but is not considered critical in the sense that all of the programs must be completed to insure safe operation.

Removal and testing of fuel taken from operating reactors at intervals during operation is not considered necessary. The data on hand, plus programs which are currently under way, should satisfactorily provide the information necessary for assurance of safe operation within the limits required.

#### REFERENCE

- <sup>1</sup> Mechanical Properties of Zircaloy-4 After Irradiation at 130, 650, and 775 F, TP-299, April 1967.

027

## Material Irradiation Testing Program

The purpose of this program is to determine the effects of irradiation on the core components of a central station power reactor. The program is divided into three tasks:

- Task I      Low Burnup Fuels Irradiation
- Task II     Zircaloy-4 Irradiation
- Task III    High Burnup Fuels Irradiation

### Task I - Low Burnup Fuels Irradiation

The primary objective of this task is to investigate the dimensional stability of pellet-type fuel rods when irradiated at current and future PWR operating conditions.

The program consists of capsules, some of which are designed to operate at 21-1/2 kw/ft. The cladding for these capsules will operate at a surface temperature of about 640 F. All of the capsules will be irradiated, when possible, for one complete cycle of the BAWTR. Under normal operation, this will amount to about 25 EFPD and a burnup of about 3,500 to 4,000 MWD/TU.

The irradiation of capsules, initially operated at a heat rate of 25-25.7 kw/ft, has been completed. Some capsules received as much as 609 power cycles at 22.8 to 24.6 kw/ft. Hot cell examination is underway.

### Task II - Zircaloy-4 Irradiations

The Zircaloy-4 cladding in the core operates with outside and inside surface temperatures as high as 650 and 800 F, respectively. A program was therefore designed to determine how the mechanical properties of Zircaloy-4 are affected by irradiation at these temperatures.

Longitudinal specimens cut from 0.425-in. diameter Zircaloy-4 tubing are used to determine the properties in the longitudinal direction. Ring specimens and flattened rings conforming to dimensions of the longitudinal specimens are used to determine the properties in the transverse direction. Some of the tensile specimens were charged with 250 to 400 ppm hydrogen prior to irradiation.

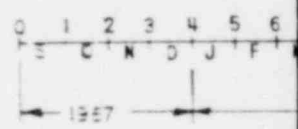
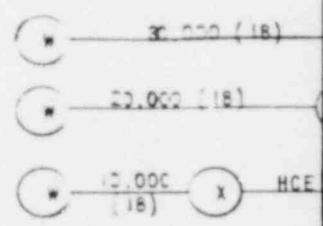
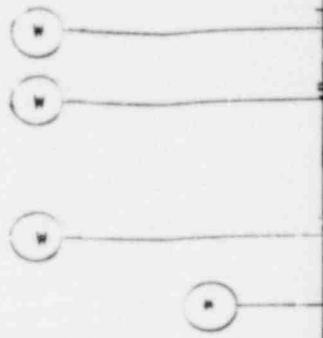
Irradiation of the two 300-day capsules is continuing without any operational difficulties. As of June 30, 1967, these capsules had achieved an exposure of 306 EFPD.

QUESTION 1A.3 We believe that research and development above that which you have indicated will be required to justify the use of core barrel check valves as a solution to the steam bubble problem. Further consideration should be given to testing (1) vibration effects on the valves (caused by core barrel vibrations) and (2) flow characteristics in the reactor after loss of a valve. We believe that if the loss of a valve is not detectable, the DNB ratio at the overpower condition after loss of a valve must be not less than 1.3 (based on the W-3 correlation).

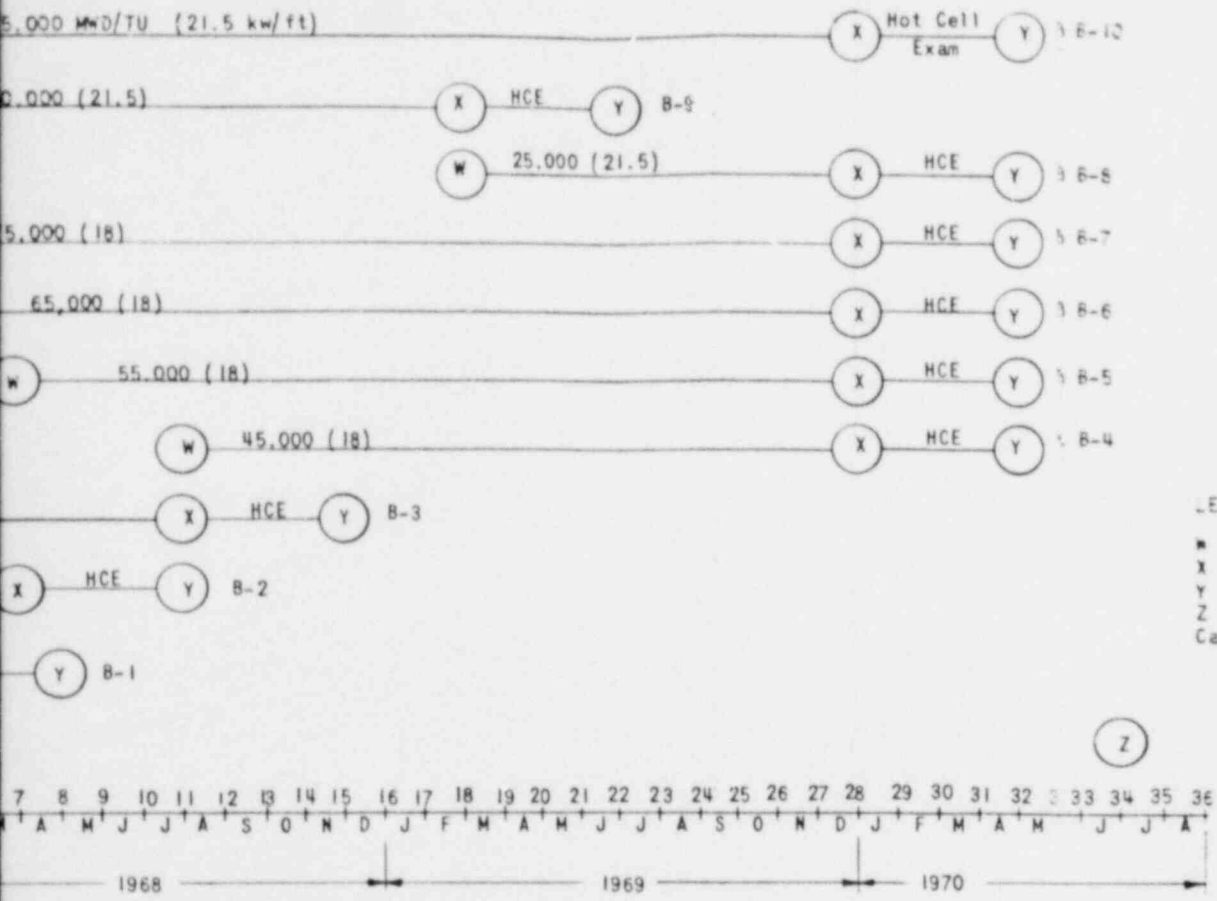
ANSWER  
Refer to 3.3.4 In order to investigate vibration of the vent valves caused by core barrel vibrations, it was assumed that the core support shield would excite the disc at a frequency where the shield mode shape corresponded to an 8-valve configuration. This frequency is 125 Hz and is substantially below the lowest resonant frequency of the disc, i.e. 1500 Hz. This large difference in frequency indicates that vibratory motions transmitted from the core support shield to the disc will not be amplified by the disc and will not exceed transmitted motions from the shield, which our preliminary analysis indicates will be less than 0.005 inch. Other more rigorous, but more time consuming, analytic methods are being pursued in order to confirm the vibratory motion of the shield. Assuming the worst case of the disc being force-excited at 125 Hz, the amplitude of the disc would have to exceed 0.025 inch in order to develop an inertial force which would exceed the pressure load of 2-1/2 tons (based on 31.5 psi) which acts to keep the valve shut, at full flow. Therefore, it is not possible to transmit sufficient high frequency vibratory power from the coolant stream to cause the shield to vibrate at an amplitude of 0.025 inch. It is concluded that even under the most pessimistic assumptions, excitations from the core support shield cannot cause the valve to open or vibrate. Therefore, it is not necessary to perform a vibration test which would attempt to vibrate the valve by simulating the postulated excitation.

The DNB ratio in the hot channel at the maximum overpower with a vent valve disc off will be high enough to insure that there is a 99 percent confidence that at least 94.5 percent of the population of all such channels are in no jeopardy of experiencing a DNB. This degree of protection is consistent with Paragraphs 3.1.2.3 and 3.2.3.1.1 of the PSAR. It will be demonstrated in the final design that the DNB ratio in the hot channel with the flow resulting from the loss of one vent valve disc will not be less than 1.3 using the W-3 correlation.

A preliminary sensitivity analysis using postulated worst case parameters has been made for the reduced flow. The results of this analysis are described in the appendix to Section 3.2.4



030



LEGEND:  
 \* - Start Irradiation  
 X - Complete Irradiation  
 Y - Interim Report Issued  
 Z - Final Report Issued  
 Capsules B-1 thru B-10.

031

FIGURE 1A.2-1  
 HIGH BURNUP IRRADIATION PROGRAM  
 SCHEDULE (BASED ON BAW-TM-192)

herein where the DNB ratios for full and reduced flow are as follows for various reactor powers:

<u>Percent Rated Power</u>	<u>DNBR (Full Flow)</u>	<u>DNBR (Reduced Flow)</u>
100	1.76	1.68
107.5	1.53	1.44
112	1.40	1.30
114	1.34	1.24

The minimum DNB ratio of 1.24 resulting from the analysis at 114 percent power for the postulated worst case is large enough to ensure a DNB ratio of not less than 1.30 for final design conditions. The postulated worst case, used for sensitivity analysis, is not the design condition but a case with heat transfer and mechanical conditions much more severe than expected in the final design. This is demonstrated by a comparison of the nominal and postulated worst case as shown in the appendix to Section 3.2.3.2.4 herein where the W-3 DNB ratios are as follows at rated flow conditions and 114 percent power:

<u>Cell Type</u>	<u>Nominal DNB</u>	<u>Postulated Worst Case DNB</u>
Corner	1.85 (1.71)	1.34 (1.24)
Wall	1.89	1.38
Unit	1.89	1.46

The minimum DNB ratios occurring in the corner cell for the two conditions at reduced flow due to loss of a vent valve disc are shown in parentheses above. The final design DNB will be within the limits of 1.71 to 1.24 shown. It is expected that a value greater than 1.30 will result from final evaluation of the combination of the following significant factors:

- (1) Mixing coefficient of 0.03 to 0.07 at design conditions compared to 0.01 used in the preliminary analysis.
- (2) Statistical determination of mechanical tolerances in lieu of minimum conceivable dimensions.
- (3) A more accurate determination of the hot channel local peaking factor of 1.095 shown in Figure 3.2-55 of the PSAR considering: (a) the statistically determined water gap, and (b) the excess metal in the solid can section surrounding the corner pin. The final value is expected to be about 1.06.

032



- (4) Application of final vessel and core flow distribution tests results instead of the hot to average fuel assembly flow ratio of 85 percent assumed for the worst postulated case.
- (5) The statistical comparison of the multiple rod fuel assembly heat transfer test data with the single channel data that currently forms the basis for the W-3 correlation.

A consideration of the final thermal-hydraulic design compared with the preliminary postulated worst case and the mechanical integrity of the vent valve indicates that it is very unlikely that the core will be subject to an unsatisfactory heat transfer condition.

033

QUESTION 1A.4 (DRL 1.1) Update the discussion of your proposed design with respect to its conformance to the Commission's Proposed General Design Criteria. Include in this discussion the impact of the several design changes made in your facility.

ANSWER Response to the Commission's Proposed General Design Criteria including discussion on the impact of the several design changes made are presented in Section 1.4 of the PSAR. Those criteria which reflect changes are 7, 10, 11, 22, 38, 44, 46, 52, 59, 61 and 62.

QUESTION 1A.5 (DRL 1.2) Describe each of your research and development programs with a proposed schedule for obtaining the desired information. Include, as appropriate, when the design of the associated feature must be frozen in order to meet the schedule for construction of the Rancho Seco Plant.

ANSWER Research and development programs that will provide information to complete the final detail design of some of the components or to demonstrate the capability of the design for future operation at a higher power level are summarized in Section 1.5 of the PSAR. Further discussion of research, development or test program is provided in the answer to Question 1A.2 in Appendix 1A of Amendment 1. Additional information and discussion is provided below:

a. Once-Through Steam Generator

Testing necessary to prove the adequacy of the once-through steam generator design for service at the initial power level and to confirm the size and configuration of the units has been completed. These programs were described in Appendix 4A of the Oconee PSAR (Docket Nos. 50-269, 270 and 287) and in the Rancho Seco PSAR, Appendix 1A, Question 1A.2. Steady state and load changing operations using once-through steam generator models have demonstrated the ability of the unit to follow transients and the interaction of the control system with the water level, steam pressure and flows. Primary and secondary blowdown tests on the models have demonstrated the integrity of the units under conditions of rapid depressurization and large tube-to-shell temperature differentials. The results of the blowdown tests are being used in the development and verification of analytical models for steam system blowdown analyses.

b. Control Rod Drive Unit

These programs have been described in Section 3.3.3.4 and Appendix 1A, Question 1A.2 of the PSAR. Some of the results of those programs will be discussed in this reply. The development and testing of the rack and pinion drive is being conducted under three separate programs:

1. Full-scale prototype testing under no-flow conditions.
2. Full-scale prototype testing at reactor operating conditions of temperature, pressure, and flow.
3. Components testing.

The no-flow prototype testing is performed in an autoclave in which the reactor conditions of control rod stroke, temperature, pressure, and water chemistry are duplicated. The tests are performed with a dummy weight equivalent to the weight of the control rod assembly attached to the rack.

The objectives of this testing were to verify the design concept and to obtain a preliminary verification of the trip insertion time.

The mechanism was subjected to approximately 100 full-stroke cycles and 100 trip cycles simulating both hot and cold reactor conditions.

This testing confirmed that the design and the mechanical arrangement met the objectives. The time for 2/3 insertion was less than 1.2 seconds; the snubber design worked properly, and the buffer seal did not impair trip capability.

Further testing was conducted which included a complete life test of full-stroke cycles and trip cycles simulating reactor operating conditions with maximum tolerance misalignment. Examination of components after the test indicated that the wear observed was acceptable on all components except the miter gear which although badly worn continued to operate satisfactorily.

The control rod drive life testing program will be continued after the mechanism has been refurbished and modified to incorporate a new miter gear utilizing 17-4 PH nitrided or Haynes 25 metal.

The second life test will be conducted with different stroking specifications than those used on the first life test.

035

Other prototype testing was conducted in another autoclave in which all reactor operating conditions except radiation are duplicated. The complete driveline is established with prototype components, i.e., the fuel assembly, control rod, upper guide tube of the reactor internals, and the drive mechanism.

This testing concentrated mainly on the performance characteristics under coolant flows ranging from zero to full flow at reactor conditions of temperature, pressure, and water chemistry. The objective of these tests was to determine the compatibility of the mechanism trip time with the specification requirements of 1.4 seconds for 2/3 insertion. After some modification of the pattern of flow holes in the shroud of the upper guide tube, the trip time ranged from 1.37 to 1.4 seconds.

Selected components testing was performed prior to and in addition to the life testing programs in order to resolve potential material or design problems. These component test programs produced the following results:

1. Provided the basis for the selection of Graphitar bearing material.
2. Ascertained the buffer seal injection flow rate.
3. Assured acceptable wear from the revised miter gear combination.
4. The corrosion product buildup in the static test of the splines and bearings has not noticeably affected the resistance to rotation of the system.

The program will be completed by August, 1968. By that date the prototype mechanism will have completed the life test program as outlined in the PSAR and all material problems for the production type mechanisms will have been resolved.

c. In-Core Neutron Detectors

This program consists of basic physics parametric studies of the detector and mechanical insertion - withdrawal tests of the assembly. The development program has been outlined in Section 7.3.3.3 of the PSAR. Mechanical testing of the assembly has been completed. All parametric studies have been completed except the long term radiation effects and the depletion effects. Results to date have been satisfactory and the performance of the detectors has been demonstrated. As of April 1968 detectors have been irradiated in the Big Rock Point Reactor for about 34 months and in The Babcock and Wilcox Test Reactor for about 23 months. These lifetime tests are continuing.

d. Core Thermal and Hydraulic Design

These programs have been discussed in Section 3.3.2 and Appendix 1A, Question 1A.2 of the PSAR. The initial core power level has been justified on the basis of the W-3 correlation which has been approved for the design of several similar pressurized water reactors. With the use of that correlation, only the reactor vessel flow model test data is necessary to further substantiate the core thermal and hydraulic design. Test runs already completed without check valves in the internals have demonstrated the ability to provide adequate flow distribution. Tests including check valves will be completed in 1968.

The Departure from Nucleate Boiling (DNB) and mixing studies described in the PSAR are being conducted to support the final thermal design margin on the basis of the B&W correlation and to provide for an increase in its rated power output when that increase is requested.

Due to the fact that the information produced by these programs has been, is being, or will be used to finalize the detail design of components for Oconee and Three Mile Island units which are scheduled to precede the Rancho Seco unit into commercial operation by about two years, the information needed from these programs will be available long before it is needed to freeze design details for the Rancho Seco unit.

QUESTION 1A.6 (DRL 1.3) If not specifically included in 1.2, describe your program including schedule and acceptability criteria for vibration testing of the core barrel check valves.

ANSWER The testing program for the core barrel check valves (internals vent valves) was discussed in Appendix 1A, Question 1A.2. In addition to the testing discussed there B&W is presently working with the valve designer-manufacturer and a vibration testing laboratory on the details of the vibration test of a full scale prototype vent valve. The prototype valve will be mounted in a test fixture which duplicates the method of valve mounting in the core support shield and simulates this local area of the internals. The test fixture with valve installed will be attached to a vibration test machine and excited sinusoidally through a range of frequencies representative of those anticipated for the core support shield during reactor operation. The relative motion between the valve disc and seat will be monitored and recorded during test. The test results will be evaluated and, if required, the valve design will be modified prior to valve production to eliminate any adverse disc vibration problems. All testing will be completed by January 1, 1969.

037

QUESTION  
1A.7  
(DRL 1.4)

If not specifically included in 1.2, discuss the programs currently in progress that will assure fuel element capability for 55,000 MWD/MTU burnup at the design power densities.

ANSWER

A high burnup fuel irradiation test program is in progress at B&W, and is described in Question 1A.2-5, Appendix 1A of the PSAR. A schedule for the program is shown in Figure 1A.2-1. This program includes fuel specimens with representative cladding thickness, fuel-clad gaps and UO<sub>2</sub> densities. Heating rates are representative of maximum heating rates and temperatures in the core. Post-irradiation examination will include profilometer scans to determine permanent clad strain, fission gas release, metallographic examination of fuel and cladding, and confirmation of burnups estimated from flux monitors during the test. Maximum target burnup is 75,000 MDW/MTU. Examination will be made at several stages of burnup between 10,000 and 75,000 MWD/MTU to determine the behavior of the fuel and cladding as a function of burnup.

The damage criteria for the high burnup test program are that the cladding will not allow fission product release, or the entrance of coolant into the fuel rod which could lead to further damage. Other experiments in the industry have shown that the limit of permanent strain in the cladding is approximately 1.5%.<sup>1</sup> This, therefore, represents the current upper limit to avoid damage associated with excessive clad strain. Design limits are set at approximately 1%. (See PSAR 3.1.2.4.2.c).

The consequence of burnup on fuel rods is that continued fuel growth and fission product release will eventually lead to clad failure due to progressive clad strain. The point of failure is influenced by irradiation-induced changes in the cladding. The program is designed to better understand the limit of burnup and allowable strain which can be achieved without clad failure. The program will also assure fuel element capability for 55,000 MWD/MTU burnup at the design power densities. It will also give a better understanding of the burnup limit, or "margin of safety," for fuel rods of representative design when tested at maximum heating rates.

#### REFERENCE

<sup>1</sup>Fracture of Cylindrical Fuel Rod Cladding due to Plastic Instability, WAPD-TM-651, April 1967.

038

QUESTION      Submit the staffing and training plans for SMUD's Nuclear  
1A.8            Project Engineering Staff.  
(DRL 1.5)

ANSWER        The staffing and training plans for SMUD's Nuclear Project  
Engineering Staff are presented in Appendix 1C. The program  
presented will provide the District with a technically quali-  
fied engineering staff both during construction and after the  
plant is operational.

QUESTION      Discuss the principal design decisions yet to be made that  
1A.9            require nuclear and steam plant knowledge and which affect  
(DRL 1.6)      nuclear power plant safety. Indicate the approximate dates  
by which these decisions must be made and to what extent reli-  
ance will be placed upon contractors for making decisions.  
Indicate how the training plans for SMUD personnel are orien-  
tated toward these requirements.

ANSWER        The principal design decisions which affect nuclear power  
plant safety have been made and are presented in the PSAR.  
However, studies are in progress as defined in Section 1.5  
of the PSAR which may affect the final design. SMUD will  
review the results of these studies, with the aid of consult-  
ants if necessary, and initiate any action required to ensure  
a safe plant.

As a matter of policy, SMUD does not rely on contractors to  
make decisions on matters of safety. Additionally, SMUD does  
not rely on contractors to make decisions concerning plant  
reliability, maintainability or operability. SMUD, working  
with its consultants and contractors, identifies problem areas  
and calls for proposed solutions. The proposals are then eval-  
uated by SMUD with expert support as necessary from its con-  
sultants. Training of SMUD personnel toward these requirements  
is set forth in Appendix 1C of the PSAR.

039

QUESTION 1A.10 (DRL 1.7) Your Amendment No. 1 provided the SMUD response to applicable questions raised during the review of a similar plant (Metropolitan Edison). This response used information that was available through November, 1967. Please update your response to these questions by considering applicable information that became available in January, 1968.

ANSWER The PSAR has been updated to be responsive through Metropolitan Edison Company's PSAR Amendment No. 6 (Docket No. 50-289).

QUESTION 1A.11 (DRL 16.6) Discuss and evaluate your program to experimentally study vibrations in the check valves.

ANSWER (See the response to Question 1.3)

040