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J. D. Woodard
Executive Vice President

Southern Nuclear Operating Company
the southern electric system

October 14, 1994

Docket Numbers: 50-348
50-364

10 CFR 50.71

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

Joseph M. Farley Nuclear Plant
Revision 12 to the Updated Final Safety Analysis Report (FSAR)

Gentlemen:

In accordance with the requirements of 10 CFR 50.71(e), Southern Nuclear Operating Company (SNC) hereby submits Revision 12 to the Joseph M. Farley Nuclear Plant, Units 1 and 2, updated FSAR. This submittal contains all the changes necessary to reflect information and analyses submitted to the Commission by SNC or prepared by SNC pursuant to Commission requirement from January 1, 1993, to April 24, 1994, i.e., since the submission of the last updated FSAR. As required, this submittal is provided on a replacement-page basis, and is accompanied by an effective page list that identifies the current pages of the FSAR following page replacement.

Pursuant to 10 CFR 50.4(b), the original and 10 copies are being submitted to the Commission's Document Control Desk and a copy is also being supplied to the NRC Region II office and the NRC Resident Inspector.

If there are any questions, or if additional information is needed, please advise.

Respectfully submitted,

J. D. Woodard

DRC/clt:fsar-r12.DOC

Attachment

cc:	NRC Document Control Desk	w/10
	Mr. S. D. Ebneter	w/1
	Mr. B. L. Siegel	w/0
	Mr. T. M. Ross	w/1

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BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

UNIT NUMBER 1 -- DOCKET NUMBER 50-348
UNIT NUMBER 2 -- DOCKET NUMBER 50-364

REVISION NUMBER 12
TO
THE UPDATED FINAL SAFETY ANALYSIS REPORT
FOR
JOSEPH M. FARLEY NUCLEAR PLANT
UNIT NUMBERS 1 AND 2
UNDER THE ATOMIC ENERGY ACT OF 1954

Southern Nuclear Operating company hereby files Revision 12 to its updated Final Safety Analysis Report for Farley Units 1 and 2.

This revision consists of updated pages for the Final Safety Analysis Report.

SOUTHERN NUCLEAR OPERATING COMPANY

By [signature]
J. D. Woodard
Executive Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 14th DAY OF October, 1994

[signature]
Notary Public

My commission Expires: 9-14-98

JOSEPH M. FARLEY
NUCLEAR PLANT UNIT 1 AND UNIT 2
FSAR UPDATE REVISION 12
REVISION INSERTION INSTRUCTIONS
OCTOBER 1994

Please update the Farley Final Safety Analysis Report according to the following instructions. Insert all 8 1/2 x 11 pages before inserting 11 x 17 tables and figures. In volume 23 behind the Rev. 11 information, insert the Rev. 12 tab followed by the distribution list and these Revision Insertion Instructions.

Remove the book tables of contents (pages i through xxiv) from the front of volumes 1 through 22 and insert the enclosed table of contents at the front of volume 1 only.

Page	Action
i through xxiv	Replace
p. 1.2-17	Replace
p. 2.1-1	Replace
p. 2.4-13	Replace
p. 2.4-21	Replace
p. 2B-37	Replace
p. 2B-39	Replace
p. 3.1-43	Replace
p. 3.1-45	Replace
t. 3.6-13	Replace
p. 3.11-3	Replace
p. 3.11-7	Replace
p. 3.11-9	Replace
p. 3A-iii	Replace
p. 3A-1.8-1	Replace
p. 3A-1.81-1	Replace
p. 3A-1.155-1	Add behind p. 3A-1.88-1
f. 3L-8C	Replace
p. 4.2-8a	Replace
p. 4.2-11	Replace
p. 4.2-11b	Replace
p. 4.3-35	Replace
p. 4.3-36a	Replace
p. 4.3-37	Replace
p. 4.3-38a	Replace
p. 4.3-38c	Add behind p. 4.3-38b
p. 4.3-38e	Add behind p. 4.3-38d
f. 4.3-46	Add behind f. 4.3-45
p. 4.4-41	Replace
p. 5.2-51	Replace
p. 5.3-1	Replace
p. 5.5-3	Replace
p. 6-v	Replace
p. 6-vii	Replace
p. 6-ix	Replace

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REVISION INSERTION INSTRUCTION
(continued)

Page	Action
p. 6-xiii	Replace
p. 6.2-5	Replace
p. 6.2-7	Replace
p. 6.2-27	Replace
p. 6.2-29	Replace
p. 6.2-35b	Replace
p. 6.2-37	Replace
p. 6.2-39	Replace
p. 6.2-40a	Replace
p. 6.2-45	Replace
p. 6.2-55	Replace
p. 6.2-71	Replace
p. 6.2-72a	Replace
p. 6.2-101	Replace
p. 6.2-111	Replace
t. 6.2-2 sheet 1	Replace
t. 6.2-2 sheet 2	Replace
t. 6.2-2 sheet 3	Replace
t. 6.2-2 sheet 4	Replace
t. 6.2-2 sheet 5	Replace
t. 6.2-2 sheet 6	Replace
t. 6.2-2 sheet 7	Replace
t. 6.2-2 sheet 8	Replace
t. 6.2-2 sheet 9	Add behind t. 6.2-2 sheet 8
t. 6.2-4	Replace
t. 6.2-5 sheet 1	Replace
t. 6.2-5 sheet 2	Replace
t. 6.2-6	Replace
t. 6.2-19	Replace
t. 6.2-21	Replace
t. 6.2-24	Replace
t. 6.2-31 sheet 1	Replace
t. 6.2-31 sheet 5	Replace
t. 6.2-41	Add behind t. 6.2-40
f. 6.2-1	Replace
f. 6.2-2	Replace
f. 6.2-3	Replace
f. 6.2-4	Replace
f. 6.2-5	Replace
f. 6.2-6	Replace
f. 6.2-7	Replace
f. 6.2-8	Replace
f. 6.2-9	Replace
f. 6.2-10	Replace
f. 6.2-11	Replace
f. 6.2-12	Replace
f. 6.2-13	Replace
f. 6.2-14	Replace
f. 6.2-15	Replace
f. 6.2-16	Replace

REVISION INSERTION INSTRUCTION
(continued)

Page	Action
f. 6.2-17	Replace
f. 6.2-18	Replace
f. 6.2-19	Replace
f. 6.2-20	Replace
f. 6.2-21	Replace
f. 6.2-22	Replace
f. 6.2-23	Replace
f. 6.2-24	Replace
f. 6.2-25	Replace
f. 6.2-26	Replace
f. 6.2-27	Replace
f. 6.2-28	Replace
f. 6.2-29	Replace
f. 6.2-30	Replace
f. 6.2-31	Replace
f. 6.2-32	Replace
f. 6.2-33	Replace
f. 6.2-34	Replace
f. 6.2-35	Replace
f. 6.2-36	Replace
f. 6.2.37	Replace
f. 6.2-39	Replace
f. 6.2-40	Replace
f. 6.2-41	Replace
f. 6.2-42	Replace with f. 6.2-42 sheet 1
f. 6.2-42 sheet 2	Add behind f. 6.2-42 sheet 1
f. 6.2-43	Replace
f. 6.2-45	Replace
f. 6.2-46	Replace
f. 6.2-47	Replace
f. 6.2-48	Replace
f. 6.2-49	Replace
f. 6.2-82	Replace
f. 6.2-83	Replace
f. 6.2-84	Replace
f. 6.2-85	Replace
f. 6.2-95	Replace
f. 6.2-113	Replace
p. 6.4-1	Replace
p. 6.4-3	Replace
p. 7-i	Replace
p. 7-iii	Replace
p. 7-iva	Delete
p. 7-v	Replace
p. 7-vii	Replace
p. 7-ix	Add behind p. 7-viii
p. 7.2-9	Replace
p. 7.2-17	Replace
p. 7.2-25	Replace
p. 7.2-25b	Add

REVISION INSERTION INSTRUCTION
(continued)

Page	Action
p. 7.2-37	Replace
p. 7.2-39	Replace
p. 7.2-40a	Add
p. 7.2-41	Replace
p. 7.2-43	Replace
t. 7.2-1 sheet 2	Replace
t. 7.2-3 sheet 2	Replace
t. 7.2-4 sheet 3	Replace
t. 7.2-4 sheet 4	Replace
t. 7.2-5 sheet 1	Add behind t. 7.2-4 sheet 4
t. 7.2-5 sheet 2	Add
p. 7.3-19	Replace
p. 7.3-21	Replace
p. 7.3-23	Replace
t. 7.3-14 sheet 1	Replace
t. 7.3-14 sheet 2	Replace
t. 7.3-14 sheet 3	Replace
t. 7.3-16 sheets 1, 2, and 3	Add behind t. 7-3-15
p. 7.4-7	Replace
t. 7.5-3 sheet 4	Replace
p. 7.7-13	Replace
f. 7.7-7	Replace
p. 7.8-3	Replace
p. 7.8-5	Replace
p. 7.8-9	Replace
p. 8-i	Replace
p. 8-iii	Replace
p. 8.1-1	Replace
p. 8.3-1	Replace
p. 8.3-5	Replace
p. 8.3-7	Replace
p. 8.3-9	Replace
p. 8.3-11	Replace
p. 8.3-11b	Delete
p. 8.3-12a	Add
p. 8.3-12c	Add
p. 8.3-12e	Add
p. 8.3-13	Replace
p. 8.3-15	Replace
p. 8.3-17	Replace
p. 8.3-19	Replace
p. 8.3.19b	Add
p. 8.3-21	Replace
p. 8.3-21b	Add
p. 8.3-33	Replace
p. 8.3-34a	Replace
p. 8.3-35b	Replace
p. 8.3-39	Replace
p. 8.3-39b	Add
t. 8.3-1 sheet 1	Replace

REVISION INSERTION INSTRUCTION
(continued)

Page	Action
t. 8.3-1 sheet 2	Replace
t. 8.3-3	Replace
f. 8.3-43	Replace
f. 8.3-44	Replace
p. 9-i	Replace
p. 9-iii	Replace
p. 9-v	Replace
p. 9-xi	Replace
p. 9.1-4a	Replace
p. 9.1-9	Replace
p. 9.1-47	Replace
p. 9.1-47b	Add
p. 9.2-3	Replace
p. 9.2-13	Replace
p. 9.2-21	Replace
t. 9.2-11 sheet 1	Replace
p. 9.3-17	Replace
p. 9.3-29	Replace
p. 9.3-33	Replace
p. 9.3-47	Replace
t. 9.3-5	Replace
p. 9.4-5	Replace
p. 9.4-17	Replace
p. 9.4-21	Replace
p. 9.4-22a	Replace
p. 9.4-45	Replace
p. 9.4-63	Replace
p. 9.4-69	Replace
t. 9.4-7 sheet 2	Replace
t. 9.4-11 sheet 1	Replace
t. 9.4-12 sheet 1	Replace
t. 9.4-13	Replace
p. 9.5-1	Replace
p. 9.5-3	Replace
p. 9.5-3b	Add
p. 9.5-5	Replace
p. 9.5-5b	Replace
p. 9.5-13	Replace
p. 9B-i	Replace
p. 9B-v	Replace
p. 9B-7	Replace
p. 9B-8a	Add
p. 9B-9	Replace
p. 9B-17	Replace
p. 9B-21	Replace
p. 9B-29	Replace
p. 9B-31	Replace
p. 9B-41	Replace
p. 9B-47	Replace
p. 9B-49	Replace

REVISION INSERTION INSTRUCTION
(continued)

Page	Action
p. 9B-51	Replace
p. 9B-53	Replace
p. 9B-55	Replace
p. 9B-59	Replace
p. 9B-63	Replace
p. 9B-65	Replace
p. 9B-67	Replace
p. 9B-67b	Replace
p. 9B-69	Replace
p. 9B-70a	Add
p. 9B-81	Replace
p. 9B-82a	Add
p. 9B-87	Replace
p. 9B-89	Replace
p. 9B-91	Replace
p. 9B-93	Replace
t. 9B-1	Replace
t. 9B-3	Replace
t. 9B-4	Replace
t. 9B-6	Replace all 3 sheets with one deletion sheet
t. 9B-7	Replace all 3 sheets with one deletion sheet
Fire Area Hazard Analysis (FAHA) No. 1-4 sheet 2	Replace
FAHA 1-6 sheet 1	Replace
FAHA 1-55 sheet 1	Replace
FAHA 2-4 sheet 2	Replace
FAHA 2-6 sheet 1	Replace
p. 10.3-1	Replace
p. 10.3-9	Replace
f. 10.3-4	Replace
p. 10.4-1	Replace
p. 10.4-25	Replace
p. 11-i	Replace
p. 11-iii	Replace
p. 11-v	Replace
p. 11.2-1	Replace
p. 11.2-13	Replace
p. 11.2-17	Replace
p. 11.2-19	Replace
t. 11.2-6	Replace
t. 11.2-8 sheet 1	Replace
t. 11.2-8 sheet 2	Replace
p. 11.3-1	Replace
p. 11.3-1b	Replace
p. 11.3-3	Replace
p. 11.3-5	Replace
p. 11.3-7	Replace
p. 11.3-9	Replace

REVISION INSERTION INSTRUCTION
(continued)

Page	Action
p. 11.3-11	Add
t. 11.3-10	Replace
p. 11.4-1	Replace
p. 11.4-17	Replace
p. 11.4-21	Replace
p. 11.4-25	Replace
p. 11.4-27	Replace
p. 11.4-31	Replace
t. 11.4-6 sheet 1	Replace
p. 11A-1	Replace
p. 11A-3	Replace
t. 11A-2	Replace
p. 12.1-1	Replace
p. 12.1-5	Replace
p. 12.1-9	Replace
p. 12.1-11	Replace
p. 12.1-15	Replace
p. 12.1-17	Replace
p. 12.1-19	Replace
p. 12.2-1	Replace
p. 12.2-5	Replace
p. 12.2-7	Replace
p. 12.2-7b	Add
p. 12.2-11	Replace
p. 12.2-13	Replace
p. 12.2-15	Replace
t. 12.2-1	Replace
t. 12.2-2 sheet 1	Replace
t. 12.2-2 sheet 2	Replace
t. 12.2-2 sheet 3	Replace
p. 12.3-1	Replace
p. 12.3-5	Replace
p. 12.3-7	Replace
p. 12.4-1	Replace
p. 13-i	Replace
p. 13-iii	Replace
p. 13-v	Replace
p. 13.1-3	Replace
p. 13.1-5	Replace
p. 13.1-7	Replace
p. 13.1-13	Replace
p. 13.1-15	Replace
p. 13.1-21	Replace
f. 13.1-7	Replace
p. 13.2-1	Replace
p. 13.2-3	Replace
p. 13.2-5	Replace
p. 13.2-7	Add
p. 13.4-1	Replace
p. 13.4-13	Replace

REVISION INSERTION INSTRUCTION
(continued)

Page	Action
p. 13.5-1	Replace
p. 13.5-4a	Replace
p. 13.5-5	Replace
p. 13.5-7	Replace
p. 13.5-9	Replace
p. 13.5-11	Replace
p. 13.5-13	Replace
t. 13.5-1 sheet 1	Replace with t. 13.5-1 through 13.5-18 deletion sheet
t. 13.5-1 sheets 2 and 3	Delete
t. 13.5-2 sheets 1 through 8	Delete
t. 13.5-3	Delete
t. 13.5-4 sheets 1 through 9	Delete
t. 13.5-5 sheets 1 through 15	Delete
t. 13.5-6 sheets 1 through 22	Delete
t. 13.5-7 sheets 1 through 5	Delete
t. 13.5-8 sheets 1, 1a, 2 and 2a	Delete
t. 13.5-9 sheets 1 through 8	Delete
t. 13.5-10 sheets 1 and 2	Delete
t. 13.5-11	Delete
t. 13.5-12 sheets 1 through 11	Delete
t. 13.5-13	Delete
t. 13.5-14	Delete
t. 13.5-15	Delete
t. 13.5-16	Delete
t. 13.5-17	Delete
t. 13.5-18 (deletion sheet)	Delete
p. 13.7-3	Replace
p. 15.1-9	Replace
t. 15.1-2A sheet 1	Replace
t. 15.1-2A sheet 3	Replace
p. 15.2-27	Replace
p. 15.2-29	Replace
p. 15.2-31	Replace
p. 15.2-35	Replace
p. 15.2-49	Replace
t. 15.2-1 sheet 4	Replace
f. 15.2-23 sheet 1	Replace
f. 15.2-23 sheet 2	Replace
f. 15.2-23 sheet 3	Delete
f. 15.2-24	Replace
f. 15.2-25 sheet 1	Replace
f. 15.2-25 sheet 2	Replace
f. 15.2-25 sheet 3	Delete
f. 15.2-26	Replace
p. 15.3-7	Replace
p. 15.3-19	Replace
t. 15.3-2A	Replace
t. 15.3-3	Replace
p. 15.4-9	Replace

REVISION INSERTION INSTRUCTION
(continued)

Page	Action
p. 15.4-13	Replace
p. 15.4-45	Replace
p. 15.4-83	Replace
t. 15.4-2	Replace
t. 15.4-3	Replace
t. 15.4-6	Replace
t. 15.4-8	Replace with t. 15.4-8 sheet 1
t. 15.4-8 sheet 2	Add
p. 16-1	Replace
p. 17.2-1	Replace
p. 17.2-17	Replace
p. 17.2-35	Replace
p. 17.2-39	Replace
p. 17.2-41	Delete
p. 17.2-43	Delete
p. 17C-49	Replace
Effective Page List pages i - lxiii	Replace

REVISION INSERTION INSTRUCTIONS
(continued)

11 X 17 TABLES/FIGURES

Table/Figure	Action
f. 1.2-1	Replace
f. 2.4-21	Replace
f. 2.4-69	Replace
f. 5.1-1 sheet 1	Replace
f. 5.1-1 sheet 2	Replace
f. 5.1-2 sheet 2	Replace
f. 5.5-6 sheet 2	Replace
f. 6.2-91 sheet 1	Replace
f. 6.2-91 sheet 2	Replace
f. 6.2-94	Replace
f. 6.2-124 sheet 1	Replace
f. 6.2-124 sheet 2	Replace
f. 6.2.-125	Replace
f. 6.5-1	Replace
f. 7.2-7	Replace
t. 8.3-2 sheet 1	Replace
t. 8.3-2 sheet 2	Replace
t. 8.3-2 sheet 3	Replace
t. 8.3-2 sheet 4	Replace
t. 8.3-2 sheet 5	Replace
t. 8.3-2 sheet 6	Replace
t. 8.3-2 sheet 7	Replace
t. 8.3-2A sheet 1	Add behind t. 8.3-2 sheet 7
t. 8.3-2A sheet 2	Add
t. 8.3-2A sheet 3	Add
t. 8.3.2A sheet 4	Add
f. 8.3-1	Replace
f. 8.3-12 sheet 1	Replace
f. 8.3-12 sheet 2	Replace
f. 8.3-13 sheet 1	Replace
f. 8.3-13 sheet 2	Replace
f. 8.3-20	Replace
f. 8.3-23 sheet 1	Replace
f. 8.3-26	Replace
f. 8.3-27	Replace
f. 8.3-28	Replace
f. 8.3-29	Replace
f. 8.3-31	Replace
f. 8.3-35	Replace
f. 8.3-36	Replace
f. 8.3-39	Replace
f. 8.3-40	Replace with f. 8.3-40 sheet 1
f. 8.3-40 sheet 2	Add

REVISION INSERTION INSTRUCTIONS
(continued)

11 X 17 TABLES/FIGURES

Table/Figure	Action
f. 8.3-41	Replace
f. 8.3-42	Replace
f. P.3-43	eplace with 8½ x 11
f. 8.3-44	Replace with 8½ x 11
f. 8.3-45	Replace with f. 8.3-45 sheet 1
f. 8.3-45 sheet 2	Add
f. 8.3-46	Replace with f. 8.3-46 sheet 1
f. 8.3-46 sheet 2	Add
f. 8.3-46 sheet 3	Add
f. 8.3-46 sheet 4	Add
f. 8.3-55 sheet 1	Replace
f. 8.3-56 sheet 1	Replace
f. 8.3-56 sheet 2	Replace
f. 8.3-57	Replace with f. 8.3-57 sheet 1
f. 8.3-57 sheet 2	Add
f. 8.3-58	Replace with f. 8.3-58 sheet 1
f. 8.3-58 sheet 2	Add
f. 8.3-59	Replace
f. 8.3-60	Replace
f. 8.3-61	Replace
f. 8.3-62	Replace
f. 8.3-63	Replace
f. 9.2-1 sheet 1	Replace
f. 9.2-1 sheet 2	Replace
f. 9.2-2 sheet 1	Replace
f. 9.2-2 sheet 2	Replace
f. 9.2-3 sheet 1	Replace
f. 9.2-3 sheet 2	Replace
f. 9.2-3 sheet 3	Add
f. 9.2-4 sheet 1	Replace
f. 9.2-4 sheet 2	Replace
f. 9.2-4 sheet 3	Replace
f. 9.2-5 sheet 1	Replace
f. 9.2-6 sheet 2	Replace
f. 9.2-7 sheet 1	Replace
f. 9.2-7 sheet 2	Replace
f. 9.2-7 sheet 3	Add
f. 9.2-8	Replace
f. 9.2-23	Replace with f. 9.2-23 sheet 1
f. 9.2-23 sheet 2	Add
f. 9.2-24	Replace
f. 9.3-1 sheet 1	Replace
f. 9.3-1 sheet 6	Replace
f. 9.3-2 sheet 4	Replace
f. 9.3-4 sheet 1	Replace
f. 9.3-4 sheet 3	Replace
f. 9.3-4 sheet 5	Replace

REVISION INSERTION INSTRUCTIONS
(continued)

11 X 17 TABLES/FIGURES

Table/Figure	Action
f. 9.3-5 sheet 1	Replace
f. 9.3-5 sheet 2	Replace
f. 9.3-6 sheet 2	Replace
f. 9.3-8 sheet 5	Replace
f. 9.4-1 sheet 1	Replace
f. 9.4-1 sheet 2	Replace
f. 9.4-3 sheet 1	Replace
f. 9.4-3 sheet 2	Replace
f. 9.4-3 sheet 4	Replace
f. 9.4-6 sheet 7	Replace
f. 9.4-7 sheet 1	Replace
f. 9.4-7 sheet 2	Replace
f. 9.4-7 sheet 3	Replace
f. 9.4-8	Replace
f. 9.4-9 sheet 1	Replace
f. 9.4-9 sheet 2	Replace
f. 9.4-12	Replace
f. 9.4-13	Replace with f 9.4-13 sheet 1
f. 9.4-13 sheet 2	Add
f. 9.4-14	Replace
f. 9.5-3 sheet 1	Replace
f. 9.5-14	Replace
f. 9.5-16	Replace
f. 9.5-17 sheet 1	Replace
f. 9.5-17 sheet 2	Replace
f. 9.5-18 sheet 1	Replace
f. 9.5-18 sheet 2	Replace
f. 9.5-19	Replace
f. 9.5-20	Replace
f. 9B-4	Replace
f. 9B-7	Replace
f. 9B-8	Replace
f. 9B-9	Replace
f. 9B-10	Replace
f. 9B-11	Replace
f. 9B-12	Replace
f. 9B-13	Replace
f. 9B-14	Replace
f. 9B-15	Replace
f. 9B-16	Replace
f. 9B-17	Replace
f. 9B-18	Replace
f. 9B-19	Replace
f. 9B-20	Replace
f. 9B-21	Replace
f. 9B-22	Replace
f. 9B-23	Replace

REVISION INSERTION INSTRUCTIONS
(continued)

11 X 17 TABLES/FIGURES

Table/Figure	Action
f. 9B-24	Replace
f. 9B-25	Replace
f. 9B-26	Replace
f. 9B-27	Replace
f. 9B-28	Replace
f. 9B-29	Replace
f. 9B-30	Replace
f. 9B-31	Replace
f. 9B-32	Replace
f. 9B-33	Replace
f. 9B-34	Replace
f. 9B-37	Replace
f. 9B-38	Replace
f. 9B-39	Replace
f. 9B-40	Replace
f. 9B-41	Replace
f. 9B-42	Replace
f. 9B-43	Replace
f. 9B-44	Replace
f. 9B-45	Replace
f. 9B-46	Replace
f. 9B-47	Replace
f. 9B-48	Replace
f. 9B-49	Replace
f. 9B-50	Replace
f. 9B-51	Replace
f. 9B-52	Replace
f. 9B-53	Replace
f. 9B-54	Replace
f. 9B-55	Replace
f. 9B-56	Replace
f. 9B-57	Replace
f. 9B-58	Replace
f. 9B-59	Replace
f. 9B-60	Replace
f. 9B-61	Replace
f. 9B-62	Replace
f. 9B-63	Replace
f. 9B-64	Replace
f. 9B-65	Replace
f. 9B-66	Replace
f. 9B-67	Replace
f. 9B-68	Replace
f. 9B-69	Replace
f. 9B-70	Replace
f. 9B-71	Replace
f. 10.2-1	Replace

REVISION INSERTION INSTRUCTIONS
(continued)

11 X 17 TABLES/FIGURES

Table/Figure	Action
f. 10.3-1 sheet 1	Replace
f. 10.3-1 sheet 3	Replace
f. 10.3-1 sheet 4	Replace
f. 10.3-1 sheet 5	Replace
f. 10.3-1 sheet 7	Replace
f. 10.3-5	Replace
f. 10.4-2 sheet 3	Replace
f. 10.4-3 sheet 1	Replace
f. 10.4-3 sheet 2	Replace
f. 10.4-3 sheet 3	Replace
f. 10.4-3 sheet 4	Replace
f. 10.4-3 sheet 5	Replace
f. 10.4-3 sheet 6	Replace
f. 10.4-3 sheet 7	Replace
f. 10.4-3 sheet 8	Replace
f. 10.4-3 sheet 9	Add
f. 11.2-2 sheet 1	Replace
f. 11.2-2 sheet 2	Replace
f. 11.2-3 sheet 1	Replace
f. 11.2-4 sheet 1	Replace
f. 11.2-5 sheet 1	Replace
f. 11.3-2 sheet 3	Replace
f. 11.3-3 sheet 3	Replace
f. 12.1-7	Replace
f. 12.1-10	Replace
f. 12.1-14	Replace
f. 12.1-16	Replace
f. 12.1-19	Replace
f. 12.1-24	Replace
f. 12.1-26	Replace
f. 12.1-30	Replace
f. 12.1-31	Replace
f. 13.1-6	Replace

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13.0 CONDUCT OF OPERATIONS

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The liquid waste processing system collects, processes, and recycles reactor grade water, removes or concentrates radioactive constituents, and processes them until suitable for release or shipment off site. Liquid wastes are sampled and activity levels verified and recorded prior to release. Processed liquid effluent from the reactor coolant system will have been subjected to the chemical and volume control system purification ion exchanger and the components of the waste processing system and will be within the limits established by technical specifications.

The gaseous waste processing system functions to remove fission product gases from the reactor coolant and to contain these gases during normal plant operation. The system also collects the gases generated from the boron recycle evaporator. Waste gases are collected in the vent header. These gases are withdrawn from the vent header by one of two compressors and discharged to a waste gas decay tank. The tank contents will be released to the environment in accordance with technical specifications. Three waste gas decay tanks are provided; each tank has a 45-day storage capacity. Gaseous wastes are discharged through an absolute particle filter to the vent stack.

Solid wastes, when required, are stored in suitable containers for offsite disposal.

2.0-SITE CHARACTERISTICS

2.1 GEOGRAPHY AND DEMOGRAPHY

2.1.1 SITE LOCATION

The site is located in southeast Alabama on the west side of the Chattahoochee River about 6 miles north of the intersection of U. S. Highway No. 84 and State Highway No. 95. It is in the northeastern section of Houston County, Alabama, just across the river from Early County, Georgia. The site is about 100 miles southeast of Montgomery, Alabama, and about 180 miles south-southwest of Atlanta, Georgia. The location of the site is shown in figures 2.1-1 and 2.1-2. The coordinates of the reactor centerlines are as follows:

Unit	Latitude and Longitude	UTM Coordinates
1	31 degrees - 13 min - 21.23 s 85 degrees - 06 min - 41.93 s	N 3,455,620.1 meters E 679,872.5 meters
2	31 degrees - 13 min - 24.01 s 85 degrees - 06 min - 41.91 s	N 3,455,705.8 meters E 679,871.6 meters

2.1.2 SITE DESCRIPTION

Alabama Power Company owns the 1850-acre site with boundaries as indicated on figure 2.1-3. The exclusion area and U. S. Army Corps of Engineers spoilage and flowage easements are also indicated on figure 2.1-3. The exclusion area is bounded by circles with radii 4140 ft centered on the reactor containment centerlines.

2.1.2.1 Exclusion Area Control

Southern Nuclear Operating Company (SNC) retains the right to control any and all activities within the exclusion area. The responsibility for implementing this authority lies with the plant supervisory staff. There is no one living on site. There is no one working within the exclusion area except for employees of SNC or its agents. The only activity unrelated to plant operations contemplated within the exclusion area is operation of the visitor information center indicated on figure 2.1-3. Procedures have been established for control of visitors to the site.

2.1.2.2 Boundaries for Establishing Effluent Release

2.1.2.2.1 Limits

The property lines as shown on figure 2.1-4 are the boundary lines for determining effluent release limits with the exception that credit is not taken for ownership of the river bed. Control of access to this area will be maintained through implementation of the security plan described in subsection 13.7.2. Effluent releases will not exceed the limits of technical specifications at the boundary.

The distances from the containment vent stacks to the boundary line and the location of the waste discharge structure are also shown on figure 2.1-4. The nearest boundary to the vent stacks is at a distance of 4120 ft for Unit 1 and 4150 ft for Unit 2.

2.1.3 POPULATION AND POPULATION DISTRIBUTION

Population projections within the 0- to 5-mile radius of the plant site were based on a dwelling count of the area. Population was then estimated on the "average population per occupied unit" from the 1960 census. Since the study was undertaken prior to 1970 census data, a comparison indicated the 1960 data gave a higher or more conservative estimate than the 1970 census.

Population projections within the 5- to 50-mile radius are based on estimates derived from reference 1.

Population projections for the 5- to 50-mile radius were made in the following manner:

1. Total population and rural population for each county was projected for the years 1975, 1985, 1995, 2005, and 2015, by linear interpolation of the estimates mentioned above.

2. The major city population projections were based on percentage growth as estimated for the county.

3. The rural population of each county was divided by the area of that county.

4. The area of each county in the radial sectors was determined.

During normal operation the pond water level is controlled between elevation 185.0 and elevation 185.5. With two units on line, it is expected that six to eight river water pumps will be required to maintain the pond water level. Typically, one or more river water pump(s) on each train is placed in the auto position. When the water level in the pond reaches elevation 185.5, the wetpit level switches turn off the river water pump(s) which is in the auto position. In the event these switches fail to operate, the wetpit level transmitters give the service water structure high pond level alarm in the control room at elevation 185.75. The operator is then required to reduce the number of running river water pumps as necessary to return the SW pond to normal level range.

The PMF will raise the pond to elevation 192.2 ft msl. This elevation is based on the pond being at elevation 186 ft msl at the start of the storm, disregarding all withdrawal from the pond for cooling purposes and the river pumps being shut down.

If the pond level drops to elevation 184.33, an annunciator in the control room is energized to alert the operator to the condition. The operator is required by a technical specification to initiate shutdown procedures if the pond level reaches elevation 184.0. The minimum pond level required for proper functioning of the service water system is elevation 161.0 ft. This level is based on the head necessary to force the required two-unit service water flow over the lip of the intake structure. The lip of the intake structure is at elevation 159.0 ft. with a 2-ft head required to produce a flow check of 71,939 gal/min into the wetpit of the intake structure. Pond level (from elevation 159.0 to 193.0) is continuously measured and recorded.

The pond spillway structure is located at the north leg of the storage pond, as shown on figure 2.4-18. This structure is a reinforced concrete three-bay culvert bridge with the drop in each bay 16 ft wide at elevation 186 ft msl. The approach canal is described in paragraph 2.4.8.2. The bottom for 20 ft upstream of the drop is paved with concrete. The design discharge flow of 2020 ft^3/s is carried over a 15-ft drop to discharge in a dug canal, thence to Rock Creek. The drop is designed by criteria shown in a Bureau of Reclamation book.[3] This drop was considered an impact block type basin which dissipates the energy, principally by turbulence induced by the impingement of the incoming flow upon the impact blocks. This requires maintaining a deep tailwater below the drop. The design has been verified by using Chow's book on its use as an energy dissipator.[4] The flow below the drop is in a dug canal with maximum average velocity of 5 ft per second, as described in paragraph 2.4.8.2.

The spillway rating for use in the flood routing is based on the formula:

$$Q = C\ [L - 2\ K_a\ H]\ H^{1.5}$$

taken from USCE Hydraulic Design Chart 111-3/1, WES 8-60. A conservative coefficient value for C of 2.95 was selected to include an allowance for entrance and friction loss. This coefficient was based on data in U.S. Geological Survey Water Supply Paper 200. A value for Ka of 0.1 was obtained from design chart 111-e/1. This rating curve is shown on figure 2.4-19.

The design storm for this basin was assumed to be a 6-h storm with a probable maximum precipitation of 29.9 in., based on U.S. Weather Bureau Hydrometeorological distribution found in the U.S. Corps of Engineers Bulletin 52-8, and is shown on figure 2.4-20. A simple triangular unit hydrograph was used to develop the inflow hydrograph and is also shown on figure 2.4-20. The spillway discharge rating curve is shown on figure 2.4-20. For the maximum outflow of 2020 ft^3/s, the maximum pond elevation is 192.2 ft msl. The pond storage curve is shown on figure 2.4-21. The storm runoff will have passed through the reservoir and receded in 48 hours.

A wave height analysis was made based on procedures described by Saville, et al.[1] The most critical wind direction for the wave formation was found to be from the northwest.

The required duration of wind velocity for wave formation is found to be approximately 7 min. A wind velocity of 50 mph over land would produce a significant wave height of 1.4 ft. (The significant wave height is that height exceeded by only 13 percent of the waves.) On the 3-1/2:1 riprapped upstream face of the low section of the dam, this wave runup would be approximately 1.4 ft including 0.1 ft for wind setup. The maximum wave would produce a runup of approximately 1.8 ft to elevation 194.0 ft msl; such a wave would be expected to occur once every 12 min. On the 4:1 riprapped face of the high section of the dam the wave runup would be approximately 0.2 ft less than that given for the 3-1/2:1 slope. The runups are measured above the average water surface.

2.4.8.2 Spillway Intake and Discharge Canals

There is a canal about 600 ft long leading from the storage pond to the spillway structure. The location and a section of this canal are shown on figure 2.4-18. The average velocity during the peak of probable maximum flood would be 2.6 ft/s. The bottom of the canal is at elevation 186 ft msl, at upper operating pond level. For protection during flood flows the canal is grassed.

of the short intake channel is elevation 64-ft msl. The capability of the service water system is discussed in subsection 9.2.1. Under normal operating conditions the minimum water surface in the river is elevation 76-ft msl. The possible failure of Jim Woodruff Dam is discussed in paragraph 2.4.11.2.

There are three wells for the plant, each equipped with a pump. Two wells are capable of producing over 500 gal/min and the third will produce 200 gal/min. The makeup demineralizer plants require 640 gal/min. Surface water is the normal supply source for the makeup demineralizers with well water available as a backup supply. The plant sanitary water system uses 60 gal/min. There are two 300,000-gal tanks used for fire protection. To fill one 300,000-gal fire protection tank in 8 hours requires 625 gal/min. The combined capacity of the three wells is 1.7 times the maximum plant usage.

Temperature records indicate that icing of ponds and rivers is not a factor to be considered.

2.4.11.6 Heat Sink Dependability Requirements

The ultimate heat sink is provided by the service water storage pond. As described in subsection 9.2.5, the storage pond is capable of providing sufficient cooling water for at least 30 days. Neither the river nor the river water system are required to function as a part of the ultimate heat sink (see subsection 9.2.1). Therefore, indication and alarm of low flow in the river is not a concern for heat sink dependability requirements.

The service water storage pond is a highly reliable source of water and, as such, is designed as a Class I structure. Additionally, a detailed analysis was performed to demonstrate the reliability of the pond dam, and the results of the analysis indicated that the possibility of a dam failure is approximately 1.9×10^{-7} failures per year. Therefore, the loss of the storage pond dam is not considered to be a credible event and such an event is not postulated as part of the design basis of the ultimate heat sink. Operation of the storage pond under PMP is described in subsection 2.4.8.

The fire protection system water source is normally supplied from wells as discussed in paragraphs 2.4.11.5, 9B.4.2.1, and subsection 9.5.1. As a backup, the fire protection system can draw water from the service water system as described in paragraph 9B.4.2.1; however, the fire protection system cannot draw water from the service water pond when the service water system is in the recirculation-to-pond mode of operation and the pond is serving as the ultimate heat sink.

2.4.12 ENVIRONMENTAL ACCEPTANCE OF EFFLUENTS

During normal operation, there are no liquid releases which have a path into ground water. The only radioactive liquid releases are to the Chattahoochee River, and these are diluted to well within the limits defined in technical specifications prior to discharge as discussed in section 11.2. Thermal and chemical

effluents meet water quality standards of both the States of Georgia and Alabama.

Release points are discussed and identified in subsection 11.2.7 which references figures 1.2-1 and 11.2-5.

In subsection 11.2.8, instantaneous and complete mixing is assumed to occur at the discharge structure. This is justified because mixing will be essentially complete a few hundred feet downstream of the discharge structure. Several turns and bends in the river further downstream from the discharge structure will promote additional mixing.

A diffuser system for the Farley discharge is not considered practical since dredging operations are performed by the Corps of Engineers to maintain navigation on the Chattahoochee River.

Accidental releases from spills from outside tanks are prevented from entering surface water by earthen dikes around outside tanks which contain significant quantities of radioactive liquids. The effects of such accidental releases on ground water are discussed in subsection 2.4.13.3.

Users of surface water and ground water are discussed in subsections 2.4.1.2 and 2.4.13.2, respectively.

2.4.13 GROUND WATER

This section presents the results and conclusions of the ground water investigations for the Joseph M. Farley Nuclear Plant. The investigations were performed by Law Engineering Testing Company, Alabama Power Company, and Bechtel Corporation.

2.4.13.1 Description and Onsite Use

Topography at the Joseph M. Farley Nuclear Plant is characterized by the Chattahoochee River Valley on the east and the Upland on the west. The Chattahoochee River Valley is essentially flat, while the Upland ranges in elevation from 130 ft MSL to 250 ft MSL.

The average annual rainfall for Houston County is 53 in., which is fairly evenly distributed throughout the year. Average annual runoff is approximately 20 in., or 0.95 million gallons per day per square mile. This runoff includes direct surface runoff and discharge from springs.[6]

Houston County is drained by tributaries of the Chattahoochee, Choctawatchee, and Apalachicola rivers. The flow of surface

Case III - 2:

Ground surface at elevation 143 ft, water level at elevation 119, and rock at elevation 99 ft -- This case is applicable to the slopes around the plant area that will not have the superimposed fill. It is also applicable to the slope traversed by the water supply pipeline that extends from the flood plain to the vicinity of the storage pond intake structure. The only Category I structure to be placed on the profile of this case is a segment of the water supply pipeline.

Case IV:

Ground surface at elevation 155, water table at elevation 116, and rock at elevation 104 ft -- This case is applicable to the area of the floodplain, east of the plant, which has been filled to approximately elevation 155. There are no Category I structures to be imposed on this soil profile.

Case V:

Ground surface at elevation 113, water table at elevation 86, and rock level at elevation 68 -- This case is applicable to the area around the river intake structure, including the river banks, floodplain, intake channel, and river bank slopes. It is also applicable to the floodplain and the floodplain area traversed by the water supply pipelines. The Category I structures to be imposed on the soil profile are the river water intake, river water intake channel, and segments of the water supply pipelines.

Case VI:

Ground surface at elevation 195, water table at elevation 186, and rock at elevation 100 ft -- This case is applicable to the Category I storage pond intake area. The intake structure, which is supported on drilled caissons, is located at the north end of the storage pond. The water level at elevation 186 is the normal maximum pond level.

The evaluation of the failure potential of the main dam embankment, dike embankments, and the foundation soils below the embankments is in subsection 2B.7.6.

2B.7.5.4 Computed Minimum Factors of Safety

The factors of safety for each case were computed using the procedure outlined in the previous sections. For each profile, a minimum factor of safety was obtained. Plots of the computed

safety factors versus depth for each case are shown on figures 2B6-2 through 2B6-8. The minimum factor of safety for each case is listed below. Based on these factors of safety, it is concluded that none of the Category I structures listed above will experience foundation instability due to soil liquefaction in the event of the postulated seismic motions.

COMPUTED MINIMUM FACTORS OF SAFETY
AGAINST LIQUEFACTION
SSE (0.10 g)

Case	Initial	± 5-percent Strain	± 10-percent Strain
I	2.4	2.8	3.4
II	2.3	2.6	3.2
III-1	2.3	2.7	3.2
III-2	2.2	2.5	3.1
IV	2.5	2.9	3.5
V	2.2	2.6	3.1
VI	1.5	1.8	2.1

2B.7.6 STORAGE POND

2B.7.6.1 General

The storage pond layout is shown on figures 2B1-1 and 2B1-3. The reservoir is located immediately south of the plant in a shallow valley with a drainage area of 0.5 mi^2. As shown by figure 2B1-1, the pond is contained by earth dam and dikes and has an area of 95 acres at elevation 184 ft. During normal operation, pond level is controlled between elevation 185 and 185.5.

A surcharge of 7 ft above normal pool is provided for the probable maximum runoff. The total pond volume of 2310 acre-ft is allocated as follows:

1. Storage at elevation 184 ft   1400 acre ft

2. Active storage, between elevation 184 - 186 — 200 acre-ft

3. Runoff control, between elevation 186 - 192.2 — 710 acre-ft

The homogeneous earth embankments compacted to a minimum 95 percent of ASTM D-698 consist of a dam 58 ft high at the maximum section with side slopes of 3(H): 1-(V) downstream and 4(H): 1(V) upstream; and of 10- to 15-ft high dikes on the east and west abutments with side slopes of 2.5(H): 1(V) downstream and 3.5(H): 1(V) upstream. The dam and dikes have a crest elevation of 195 ft. This accommodates a normal maximum pool level elevation of 186 ft. A maximum drawdown to elevation 161 ft can occur. The dam and dike have a total crest length of about 3900 ft. The maximum section of the dam is founded on Lisbon formation rock. Suitable local soils are utilized to construct the dam as a homogeneous compacted fill. The dam has a downstream horizontal and a vertical drain; the dikes have downstream toe drains. The downstream earth slopes are seeded with grass and the upstream slopes have dumped riprap erosion protection. The dam and dikes are designed with factors of safety adequate to resist all static and earthquake dynamic forces. The uncontrolled emergency spillway is designed to pass the runoff from the design storm. The dam is 30 ft wide at the crest with a 9-ft freeboard above the normal maximum pond level. The general arrangement, the dam and dike sections and details are shown on figures 2B7-1 through 2B7-18.

2B.7.6.2 Subsurface Conditions and Embankment Materials

The topography in the vicinity of the storage pond varies from elevation 135 ft in the valley bottom at the maximum dam section to elevation 180 ft on the east side and to above elevation 200 ft on the west side of the valley. The overburden soils, above the Lisbon rock formation at elevation 100 ft, can be divided into two general groups. Clayey sands, silty sands, and sandy clays are present above elevation 130 and above the ground water table; slightly silty sands are present between elevation 130 ft and elevation 100 ft and below the water table. A layer of low plasticity silty clay exists between elevation 160 ft and elevation 145 ft throughout most of the pond area. This clay layer provides a natural impervious blanket except in the area under the dam section where the natural clay blanket has been eroded away and replaced by recent alluvium of slightly silty sand within the confines of the intermittent stream channel.

The boring location plan in the storage pond area is shown on figure 2B1-3. The results of 68 test borings indicate that the

average standard penetration resistance for the overburden soils is 28 blows per foot above elevation 130 ft and 25 blows per foot below elevation 130 ft. Between elevation 130 and elevation 110 ft, under the dam section at the valley bottom, the relatively low density sands have an average penetration resistance of 15 blows per foot. These low density soils are localized near elevation 120. The results of the geophysical cross-hole survey indicate a shear wave velocity of 450 to 650 feet per second for this zone. The materials above and below it have a shear wave velocity of 850 to 950 feet per second.

Geophysical explorations indicate the same average shear wave velocity of 900 ft per second for the overburden soils above elevation 100 ft and a shear wave velocity of 2200 ft per second for the Lisbon rock below elevation 100 ft. The compression wave velocity is 2500 ft per second for the overburden soils above elevation 130 ft, 5000 ft per second for the soils between elevation 130 and elevation 100 ft, and 6400 ft per second for the Lisbon below elevation 100 ft. The locations of the geophysical explorations at the storage pond area are shown on figure 2B3-3 and the generalized velocity profiles are shown on figure 2B3-4 through 2B3-10. The combination of soil borings and geophysical exploration provides a good description of the subsurface conditions in the storage pond area. Subsurface profiles for the dam and dikes are shown on figures 2B4-6 through 2B4-8, 2B4-19, and 2B4-20.

An upper aquifer exists above the aquiclude formed by the upper portion of the Lisbon formation. The phreatic or free water surface ranges between elevation 125 and elevation 130 ft in the storage pond area, and the soils below elevation 130 ft are generally saturated. The ground water water surface is well defined by borings and geophysical explorations. The compression wave velocity increases to 5000 ft per second in a saturated granular soil. This upper aquifer ultimately discharges into the Chattahoochee River. Beneath the aquiclude formed by the upper two-thirds portion of the Lisbon, another aquifer exists. The piezometric levels of this lower aquifer are essentially the same as the levels of the upper aquifer and range between elevation 125 and elevation 135.

A study of the soil types and the standard penetration resistances was made of the subsurface soils in the storage pond dam and dike area. The results show that the relatively low density soils are localized at the dam primarily within the area bounded by the 150-ft contour and at various isolated elevations under the dam. The R-series of borings was made to establish the extent of the natural clayey soils in the pond bottom. An upstream blanket of compacted clayey soils will be keyed into the existing natural clayey soils. The in-place soil bounded by the 150-ft contour and located beneath the dam

are designed to take the position that provides greater safety upon loss of actuating power.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them are provided, as necessary, to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, includes consideration of the population density, use characteristics, and physical characteristics of the site environs.

3.1.49 CRITERION 56 - PRIMARY CONTAINMENT ISOLATION

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment is provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis namely:

1. One locked closed isolation valve inside and one locked closed isolation valve outside containment.
2. One automatic isolation valve inside and one locked closed isolation valve outside containment.
3. One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.
4. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment are located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

CONFORMANCE

Each line that connects directly to the containment atmosphere and penetrates the primary reactor containment is provided with containment isolation valves in accordance with this criterion, as discussed in subsection 6.2.4. Isolation valves outside the

containment are located as close to the containment as practical. Upon loss of actuating power, automatic isolation valves are designed to take the position that provides greater safety.

3.1.50 CRITERION 57 - CLOSED SYSTEM ISOLATION VALVES

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere has at least one containment isolation valve which is either automatic, or locked closed, or capable of remote manual operation. This valve is outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

CONFORMANCE

Each line that penetrates containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere is provided with an appropriate containment isolation valve arrangement. Refer to subsection 6.2.4 for a complete description of the design and operation of the containment isolation system.

3.1.51 CRITERION 60 - CONTROL OF RELEASE OF RADIOACTIVE MATERIALS TO THE ENVIRONMENT

The nuclear power unit design includes means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity is provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

CONFORMANCE

Control of waste gas effluents is accomplished by holdup of waste gases in decay tanks until the activity of tank contents and existing environmental conditions permit discharges within technical specification requirements. Waste gas effluents are monitored at the point of discharge for radioactivity and rate of flow. Sufficient waste gas holdup capacity is provided, as discussed in section 11.3, to cope with all anticipated operational occurrences and site environmental conditions. A

decay tank burst would not result in an activity release greater than 10 CFR 100 limits, based on one percent failed fuel.

Control of liquid waste effluents is accomplished by holdup of waste liquids in storage tanks, batch processing of all liquids and sampling before controlled rate discharge. Liquid effluents are monitored for radioactivity and rate of flow. The liquid waste disposal system tankage and processing capacity, as described in section 11.2, is sufficient to cope with all anticipated operational occurrences and unfavorable site environmental conditions.

Station solid wastes are prepared in batches for offsite disposal by approved contractors in shielded and reinforced containers which meet Federal Regulation requirements. Sufficient handling capacity is provided, as discussed in section 11.5, to cope with all anticipated operational occurrences.

Chapters 11.0 and 15.0 provide additional information.

3.1.52 CRITERION 61 - FUEL STORAGE AND HANDLING AND RADIOACTIVITY CONTROL

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity are designed to assure adequate safety under normal and postulated accident conditions. These systems are designed with a capability to permit appropriate periodic inspection and testing of components important to safety; with suitable shielding for radiation protection; with appropriate containment, confinement, and filtering systems; with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal; and to prevent significant reduction in fuel storage coolant inventory under accident conditions.

CONFORMANCE

All fuel storage and most waste handling facilities are contained in the auxiliary building; equipment is designed to prevent accidental releases of radioactive material directly to the environment. Components of these systems which are important to safety can be periodically inspected and tested.

The spent fuel storage pool is designed for the underwater storage of spent fuel assemblies and control rods after their removal from the reactor. Each unit is designed to accommodate a total of approximately 9

cores. The shipping cask is accommodated in a separate pool adjoining the spent-fuel pool.

The spent-fuel storage racks are located to provide sufficient shielding water over stored fuel assemblies to limit radiation at the surface of the water to no more than 2.5 mrem/h during the storage period. The exposure time during refueling will be limited so that the integrated dose to operating personnel does not exceed the limits of 10 CFR 20.

The waste disposal system is designed to permit controlled handling and disposal of liquid, gaseous, and solid wastes generated during plant operation. The principal design criterion is to ensure that plant personnel are protected against exposure to radiation from wastes in accordance with limits defined in 10 CFR 20. During plant operations members of the public will be protected in accordance with limits defined in technical specifications.

The spent fuel pool is located within the auxiliary building. The liquid waste processing equipment and the gaseous waste storage and disposal equipment are located within a separate area of the auxiliary building. Both of these areas provide confinement capability in the event of an accidental release of radioactive materials, and both are ventilated with discharges to the vent stack which is monitored. In the event of a fuel handling accident, the ventilation exhaust line will be automatically isolated and the fuel handling area ventilation fans will be automatically secured. The fuel handling area may then be remotely connected with the penetration room filtration system so that the air is processed by the particulate, absolute, and charcoal filters prior to being released through the vent stack.

Radioactive liquid discharged into the cooling tower blowdown is monitored prior to discharge. Any accidental leakage from liquid waste processing equipment is collected and transferred to other tanks to prevent uncontrolled releases to the environment.

The spent-fuel pool cooling system removes the residual heat from the spent-fuel pool. The system is normally required to handle the heat load from 1/3 of a freshly unloaded core but it can safely accommodate the heat load from the spent-fuel pool while completely loaded with spent fuel assemblies.

The spent-fuel pool cooling system design incorporates redundant heat exchangers and pumps, each with 100 percent capability. Normally, one pump draws water from the pool, circulates it through a heat exchanger, and returns it to the pool. Component cooling water cools these heat exchangers.

TABLE 3.6-13

STEAM GENERATOR BLOWDOWN LINE INSIDE
CONTAINMENT PIPE BREAK LOCATION AND STRESS VALUES
UNIT 2

Line Location Number	Primary Stress (psi)	Secondary Stress (psi)	Percent of $.8(S_h + S_A)$
1	--	--	T.E.[a]
2	--	--	T.E.[a]
3	4687	16376	70[a]
4	5821	13432	64[a]
5	--	--	T.E.[a]
6	4054	11259	51[a]
7	3564	13332	56[a]
8	--	--	T.E.[a]
9	--	--	T.E.[a]
10	--	--	T.E.[a]
11	4125	8310	41[a]
12	4583	10741	51[a]
13	5122	7788	43[a]
14	--	--	T.E.[a]
15	3642	15172	63[a]
16	2790	18995	73[a]
17	2254	17138	65[a]
18	3718	12424	54
19	3801	10557	48
20	--	--	T.E.[a]
21	--	--	T.E.[a]
22	--	--	T.E.[a]
23	3186	7850	37[a]
24	7968	9113	57[a]
25	6104	9502	52[a]
26	--	--	T.E.[a]

Primary Stress = Seismic + Weight + Dynamic (where applicable)

Secondary = Thermal + Seismic Anchor Movement

$.8(S_h + S_A)$ = Primary + Secondary

T.E. = Terminal end.

a. Postulated breakpoint.

3.11.2.1 Equipment Inside Containment

Equipment listed in table 3.2-1 is designed for 40 years of operation in the most severe temperature, pressure, humidity, and radiation environment which exists at the equipment location during normal operation. In some cases, a 40-year life under such conditions is not within the state-of-the-art; therefore, a replacement program is established to ensure continuous, reliable operation. Furthermore, the safety-related equipment listed in table 3.2-1 is designed to remain functional in the most severe temperature, pressure, humidity, radiation and chemical environment which exists at the equipment location at the time it is required to perform after a design basis loss-of-coolant or main steam line break accident. Such equipment required after a design basis LOCA is also designed for the integrated radiation exposure after the LOCA. The temperature, pressure, radiation, and humidity environment inside the containment after such accidents is presented in table 3.11-1. The containment spray characteristics are given in subsection 6.2.2.

3.11.2.2 Equipment Outside Containment

Active safety-related equipment located outside the containment normally operates in ambient temperatures up to 104°F. Normal operating radiation environments are provided in table 3.11-1. The design environmental conditions, including cumulative radiation exposure, are also given in table 3.11-1.

3.11.2.3 Equipment Supplied by Bechtel and Southern Company Services

Descriptions of the qualification tests and analyses that have been performed on the components of safety-related systems are contained in the sections indicated below:

A. Containment isolation system in paragraph 6.2.4.4.

B. Containment cooling system in paragraph 6.2.2.4.2.

C. Penetration room filtration system in paragraph 6.2.3.4.2.

D. Control room ventilation system in paragraph 9.4.1.4.

E. Auxiliary feedwater system in subsection 6.5.4.

F. Component cooling system in subsection 9.2.2.

G. Service water system in subsection 9.2.1.

H. Diesel building ventilation system in subsection 9.4.5.

In the auxiliary building ventilation systems, 11 coolers and fan units are designated as engineering safeguards. They are:

A. High head injection pump rooms (3 cooling units).

B. Low head injection pump rooms (2 cooling units).

C. Auxiliary feed pump rooms (2 cooling units).

D. Containment spray pump rooms (2 cooling units).

E. Component cooling pump rooms (2 cooling units).

The test and analysis requirements for these cooling units are the same as required for the containment heat removal system, as given in paragraph 6.2.2.4.2.

The main steam isolation valves are safety-related components. They are hydrostatically tested in the manufacturer's facilities in accordance with the applicable code. Test and inspection requirements are contained in subsection 10.3.4.

3.11.2.4 Equipment Supplied By Westinghouse

Temperature in the control room and computer room is maintained for personnel comfort between 60 and 80°F, with the exact range in each room being controlled by a procedure. Design specifications for this equipment require that no loss of protective function should result when operating in temperatures up to 120°F and humidity up to 95 percent, which may occur upon the loss of air conditioning and/or the ventilation system. Thus there is a wide margin between the design limit and the normal operating environment for the protective equipment.

The normal operating temperature for the protective equipment in the containment will be maintained below 120°F, (except that for out of core neutron detectors the normal operating temperature will be maintained below 135°F). The protective equipment is designed for continuous operation within design tolerance in this environment.

span while being subjected to the DBA environment. For those instruments assumed to function in the safety analyses, the reactor protection system setpoints will be compatible with the recorded accuracies of environmental testing, normal operational accuracies, and the accident analyses.

Analyses have been performed which include taking into account those short-term environmental inaccuracies reported in letter NS-CE-792, Eichelding to Vassallo, dated October 1, 1975. The corresponding setpoint modifications have been provided in the Farley Technical Specifications. These analyses demonstrate that the design bases are still met for all chapter 15 analyses.

A final rule on environmental qualification of elec'.ical equipment important to safety for nuclear power plants became effective on February 22, 1983. This rule, 10 CFR 50.49, established the NRC acceptance criteria and specified the requirements to be met for demonstrating the environmental qualification of electrical equipment important to safety located in a harsh environment. In accordance with this rule, equipment may be qualified to the criteria specified in either the DOR Guidelines, "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," or NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," except for replacement equipment. Replacement equipment installed subsequent to February 22, 1983, must be qualified in accordance with the provisions of 10 CFR 50.49, using the guidance of Regulatory Guide 1.89, unless there are sound reasons to the contrary.

3.11.3 QUALIFICATION TEST RESULTS

In order to address the question for environmental qualification of electrical equipment for the Farley Nuclear Plant, Alabama Power Company organized a task force to review the qualification of electrical equipment. The equipment covered in this review included Class 1E equipment inside containment and Class 1E equipment outside containment which is required to mitigate a postulated accident and is subjected to a harsh environment. Harsh environment is defined as LOCA/MSLB inside the containment and HELB areas outside the containment. Additionally, this review addressed the effects of radiation on equipment outside the containment building during post-LOCA recirculation of containment sump fluids. This scope of review assured that equipment necessary to protect the public health and safety is capable of performing its function when subjected to a harsh environment.

This review of environmental qualification was based on IE Bulletin 79-01B - Environmental Qualification of Class 1E Equipment dated January 14, 1980 and the guidelines outlined for Category II plants as defined by NUREG 0588, which was issued to operating license applicants by NRC letter on February 5, 1980. The review was conducted by a task force composed of personnel experienced in reactor systems safety analysis and design, plant operations, emergency operating procedures, nuclear safety, and environmental qualification. A critical review of all documentation was conducted, using criteria established from IEB 79-01B and NUREG 0588, resulting in an auditable record with appropriate procedures documented to identify the specific equipment, the criteria used in reviewing the report, the reviewer, and the specific report reference.

As a part of this review effort, the task force reviewed the Plant Emergency Procedures to ensure that equipment required by the procedures that could be subjected to a harsh environment is qualified to operate for the time necessary to mitigate the particular accident.

The results of the Farley Nuclear Plant environmental qualification review for each item of safety-related electrical equipment subject to a harsh environment are documented in a submittals to the NRC dated July 30, 1980, for IEB 79-01B and September 15, 1980, as revised and ammended. These submittals consisted of tabular listings of all such equipment and appropriate qualification-related data for each item in accordance with the NRC quidelines. Documentation was also provided, for a comparison, of the environmental qualification data against the requirements set forth in IEB 79-01B and NUREG 0588, on report evaluation sheets for each type of equipment to identify the degree to which the qualification complies with the NRC staff position. Outstanding items were defined as being those for which discrepancies in meeting the guidelines of IEB 79-01B and NUREG 0588 have been identified. A summary of these discrepancies was provided as part of the submittals and included corrective actions and schedules together with justification for interim operation.

3.11.4 LOSS OF VENTILATION

The control room is provided with redundant air conditioning and filtration systems, as described in subsection 9.4.1. This ensures that there will be no loss of ventilation to the control and electrical equipment located within the control room.

All safeguard pumps and motors in the auxiliary building are located in rooms equipped with pump room coolers to provide adequate ventilation for the motors. These coolers are provided as a redundant system, so that if any one pump room

cooler fails, the corresponding pump would be shut down. The pump room cooler fan in a safeguard pump room is powered from the same emergency bus as the pump motor. Thus, no single active or passive failure can result in the loss of safety function of both pumps of a redundant system.

REFERENCES

1. Wilson, J. F., "Electric Hydrogen Recombiner for PWR Containments Equipment Qualification Report," WCAP 7709-L Supplements 1-4 (Proprietary), October 1973, WCAP 7820 (Non-proprietary), October 1973.

2. Locante, J. and Igne, E. G., "Environmental Testing of Engineered Safety Features-Related Equipment (NSSS Non-standard Scope)," WCAP 7744, Volumes 1 and 2, September 1970.

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Regulatory Guide 1.8 - PERSONNEL SELECTION AND TRAINING (SAFETY GUIDE 8, 3/10/71)

CONFORMANCE

Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions and the supplemental requirements specified in 10 CFR 55, except for the Health Physics supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. Since the occurrence of the TMI-2 accident, additional shift manpower has been added and an upgrading process has been instituted for the training and qualification of operating personnel. These changes are in conformance with NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," as modified by NUREG-0737, "Clarification of TMI Action Plan Requirements." Details of the training program are presented in section 13.2.

Regulatory Guide 1.9 - SELECTION OF DIESEL GENERATOR SET CAPACITY FOR STANDBY POWER SUPPLIES (SAFETY GUIDE 9, 3/10/71)

CONFORMANCE

The standby power system is discussed fully in subsection 8.3.1.2(d), AC Power Systems,and meets the recommendations of Regulatory Guide 1.9.

Regulatory Guide 1.81 - SHARED EMERGENCY AND SHUTDOWN ELECTRIC SYSTEMS FOR MULTIUNIT NUCLEAR POWER PLANTS (6/74)

CONFORMANCE

Paragraph C.1

Each of the two units is provided with separate and redundant dc electrical systems. The sharing of the dc electrical systems is limited to control power requirements of components in the service water intake structure and diesel generators 1-2A, 1C, and 2C, which are shared between Unit 1 and Unit 2. The dc control power supplies to these diesels are mechanically interlocked so that only one source furnishes control power requirement at any time.[a] See figure 8.3-30 for the dc distribution system for diesels. Details of the dc electrical system are discussed in subsection 8.3.2.

Paragraph C.2

The Onsite ac Power System is described in subsection 8.3.1, and, more specifically, the onsite emergency power system is described in 8.3.1.1.7. The details of conformance with this paragraph follow:

Item a. The sharing of onsite ac electrical systems is limited to two units.

Items b and c. The sizing of diesels together with the control circuitry design is adequate considering a single failure capability to automatically supply the ESF loads for the accident unit and the safe shutdown requirements for the other unit. Details of diesel operation under various conditions are provided in paragraph 8.3.1.1.7 and take into consideration the most severe condition of a DBE on one unit and a failure of one diesel generator.

Item d. The control circuits for shedding and loading the ESF loads are essentially separate in that each ESF 4160V bus is provided with its load sequencer. The interaction between control circuits for Unit 1 and 2 is limited to automatic starting signal from the 4160-V buses and

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a. No common mode failures exist which could fail dc systems in both units.

instrumentation for the shared diesels. However, maintenance and testing of these starting signals in one unit will not prevent the diesel generator from supplying the minimum ESF loads on the other unit.

Item e. All diesel generators are controlled from the emergency power board common to both Units 1 and 2 and located in the control room. Coordination between unit operators is not necessary for meeting recommendations of Regulatory Positions 2b, 2c, and 2d.

Item f. Complete information in regard to the diesel generator, the load sequences, and the associated 4160-V breakers is displayed on the emergency power board in the control room

Item g. The design conforms to Regulatory Guides 1.6 and 1.9 as discussed in the appropriate areas of this appendix. Information regarding bypassed and inoperable systems is provided, although detailed conformance to Regulatory Guide 1.47 is not achieved. This is detailed in the discussion of Regulatory Guide 1.47 elsewhere in this appendix.

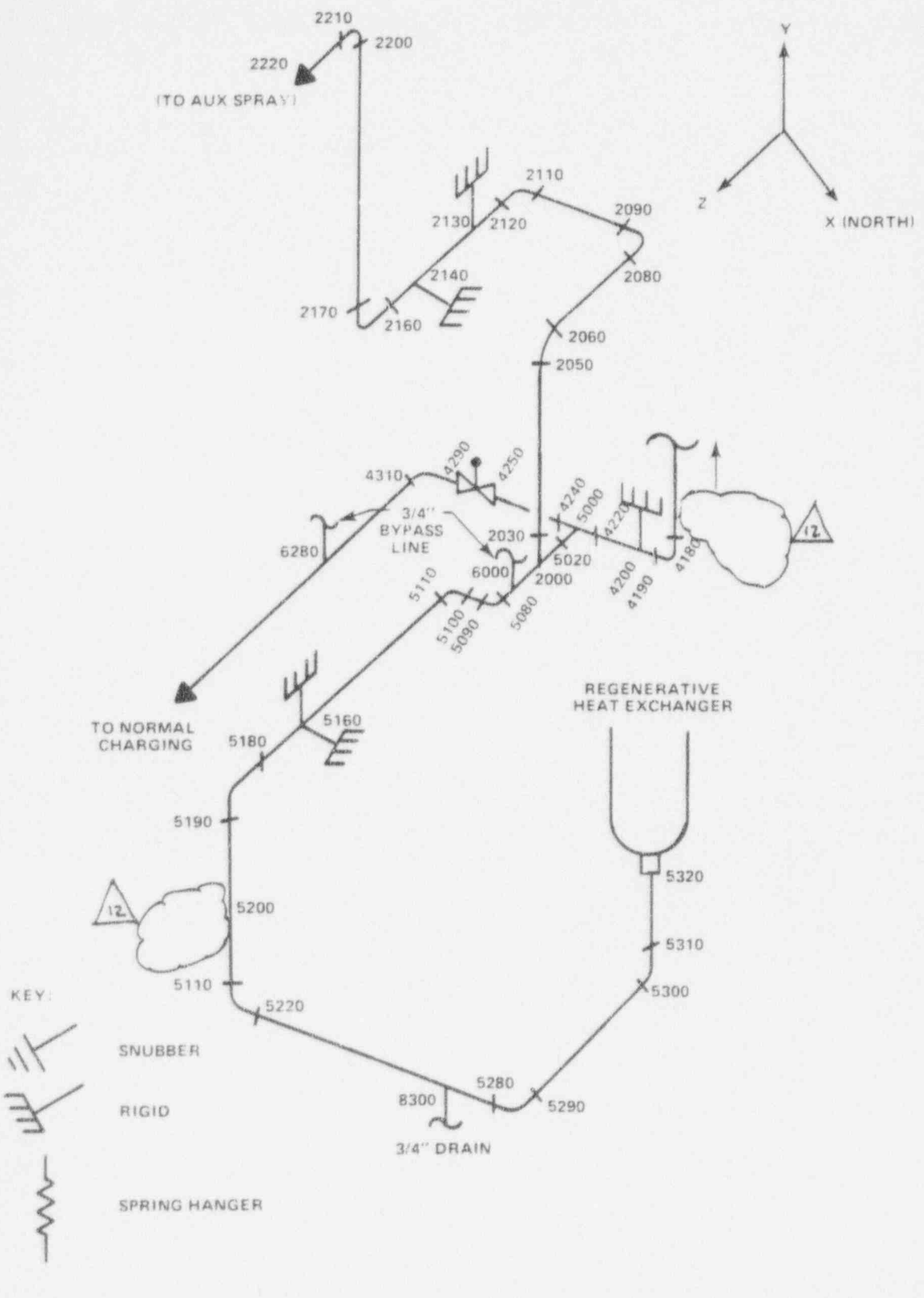
Paragraph C.3

This paragraph is not applicable to the Farley Nuclear Plant.

Regulatory Guide 1.155 - STATION BLACKOUT
(August 1988)

CONFORMANCE

Compliance with Regulatory Guide 1.155 is discussed in detail in paragraph 8.3.1.2.F.



REV 12 10/94

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CVCS ALTERNATE CHARGING LINE, LOOP 1 MODEL, PART B (UNIT 2)

FIGURE 3L-8C

4.2.1.2.2 Fuel Assembly Structure

The fuel assembly structure consists of a bottom nozzle, top nozzle, guide thimbles, and grids, as shown in figure 4.2-2.

Bottom Nozzle

The bottom nozzle serves as a bottom structural element of the fuel assembly and directs the coolant flow distribution to the assembly. The square nozzle is fabricated from type 304 stainless steel and consists of a perforated plate and four angle legs with bearing plates, as shown in figure 4.2-2. Starting with Farley Unit 2 Cycle 10, the fuel assembly bottom nozzles will be a two-piece design, known as a composite nozzle, incorporatig a high machining stainless steel adapter plate welded to a low cobalt investment casting. This casting replaces the current eight-piece weldment comprised of four cast legs and four rolled skirt plates. The new design is functionally interchangeable with the old design while providing additional design margin, since welding at the skirt-to-leg interface has been eliminated and replaced with a solid joint. The legs form a plenum for the inlet coolant flow to the fuel assembly. The plate itself acts to prevent a downward ejection of the fuel rods from their fuel assembly. The bottom nozzle is fastened to the fuel assembly guide tubes by locked thimble screws which penetrate through the nozzle and mate with an inside fitting in each guide tube.

Coolant flow through the fuel assembly is directed from the plenum in the bottom nozzle upward through the penetrations in the plate to the channels between the fuel rods. The penetrations in the plate are positioned between the rows of fuel rods.

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Fuel rod support grids are fastened to the guide thimble assemblies to create an integral structure. Attachment of the inconel and zircaloy grids to the zircaloy thimble tubes is performed using the fastening technique depicted in figures 4.2-4, 4.2-5, and 4.2-6, except for the bottom grid, which is fastened securely between the guide thimble end plug and bottom nozzle.

An expanding tool is inserted into the inner diameter of the zircaloy thimble tube at the elevation of stainless steel grid sleeves that have been brazed into the Inconel grid assembly. The four-lobed tool forces the thimble and sleeve outward to a predetermined diameter, thus joining the two components.

The top grid-to-thimble attachment is shown in figure 4.2-5 for welded top nozzle assemblies. The 304L stainless steel sleeves are brazed into the Inconel grid assembly. The zircaloy guide thimbles are fastened to the long sleeves by expanding the two members as shown in figures 4.2-4 and 4.2-5. Finally, the top ends of the sleeves are welded to the top nozzle adapter plate as shown in figure 4.2-5.

In assemblies with reconstitutable top nozzles, the guide thimbles are fastened inside the top grid sleeves and nozzle inserts as shown in figure 4.2-5. A bulge in the nozzle insert is then captured in a corresponding groove in the hole in the top-nozzle plate. The insert is fixed in place by the insertion of a lock tube into the insert, thus providing a mechanical connection between the guide thimble and the top nozzle.

The top inconel grid sleeve, top nozzle insert, and thimble of the VANTAGE 5 design are joined together using three bulge joint mechanical attachments as shown in figure 4.2-5 (sheet 3). This bulge joint connection was mechanically tested and found to meet all applicable design criteria.

The VANTAGE 5 intermediate mixing vane and IFM zircaloy grids employ a single bulge connection (figure 4.2-6, sheet 2) to the sleeve and thimble as compared to a three bulge connection used in the top Inconel (figure 4.2-5) and intermediate grids (figure 4.2-6, sheet 1). Mechanical testing of this bulge joint connection was also found to be acceptable.

The bottom grid assembly is joined to the assembly as shown in figure 4.2-7. The stainless steel insert is spot welded to the bottom grid and later captured between the guide thimble end plug and the bottom nozzle by means of a stainless steel thimble screw.

The described methods of grid fastening are standard and have been used successfully since the introduction of zircaloy guide thimbles in 1969.

The central instrumentation tube in each fuel assembly is constrained by seating in counterbores in each nozzle. This tube is a constant diameter and guides the incore neutron detectors. This thimble is expanded at the top and mid-grids in the same manner as the previously discussed expansion of the guide thimbles to the grids.

The VANTAGE 5 instrumentation tube has a reduction in diameter when compared to the LOPAR instrumentation tube; the wall thickness remains the same. This decrease still allows sufficient diametral clearance for the flux thimble to traverse the tube without binding.

Grid Assemblies

The fuel rods, as shown in figure 4.2-2, are supported laterally at intervals along their lengths by grid assemblies which maintain the lateral spacing between the rods throughout the design life of the assembly. Each fuel rod is afforded lateral support at six contact points within each grid by the combination of support dimples and springs. The grid assembly consists of individual slotted straps interlocked and brazed in an "egg-crate" arrangement to join the straps permanently at their points of intersection. The straps contain spring fingers, support dimples, and mixing vanes.

Because of its corrosion resistance and high strength properties, the grid material chosen for the LOPAR fuel assembly design is Inconel 718. The magnitude of the grid restraining force on the fuel rod is set high enough to minimize possible fretting, without overstressing the cladding at the points of contact between the grids and fuel rods. The grid assemblies are designed to allow axial thermal expansion of the fuel rods without imposing restraint sufficient to develop buckling or distortion of the fuel rods.

Two types of structural grid assemblies are used in each LOPAR fuel assembly. One type, with mixing vanes projecting from the edges of the straps into the coolant stream, is used in the high heat flux region of the fuel assemblies to promote mixing of the coolant. The other type, located at the ends of the assembly, does not contain mixing vanes on the internal straps. The outside straps on all grids contain mixing vanes which, in addition to their mixing function, aid in guiding the grids and fuel assemblies past projecting surfaces during handling or during loading and unloading of the core. Starting with the Farley Unit 1 cycle 10 and Farley Unit 2 cycle 7 reload, a vane has been added to the existing vanes and tabs on both the top and bottom of the outside grid strap to prevent hangup from the grid strap interference during fuel assembly removal. Also, to counter the effects of extended burnup, the bottom Inconel grid was modified starting with Farley Unit 2, Region 12 (Cycle 10) to give a higher

spring force. This increase in spring force leads to a lower propensity for grid-to-fuel rod fretting. The top and bottom Inconel (nonmixing vane) grids for VANTAGE 5 fuel assemblies are similar in design to the Inconel grids of the typical LOPAR fuel assembly design. The six intermediate (mixing vane) grids on VANTAGE 5 are made of zircaloy-4 material rather than Inconel, which is currently used on the LOPAR design.

The zircaloy grids have thicker straps than the Inconel grids. Also, the zircaloy grid height is higher compared to the Inconel mixing vane grids. These dimensional changes were made to compensate for differences in material strength properties. The zircaloy grid incorporates the same grid cell support configuration as the Inconel grid. The zircaloy interlocking strap joints and grid/sleeve joints are fabricated by laser welding, whereas the Inconel grid joints are brazed. The mixing vanes incorporated in the six zircaloy mid-grids induce additional flow mixing among the various flow channels in a fuel assembly as well as between adjacent fuel assemblies, and thus improve thermal performance.

The IFM grids shown in figure 4.2-2 are located in the three uppermost spans between the zircaloy mixing vane structural grids and incorporate a similar mixing vane array. Their prime function is mid-span flow mixing in the hottest fuel assembly spans. Each IFM grid cell contains four dimples which are designed to prevent mid-span channel closure in the spans containing IFMs and fuel rod contact with the mixing vanes. This simplified cell arrangement allows short grid cells so that the IFM grid can accomplish its flow mixing objective with minimal pressure drop.

The IFM grids, like the VANTAGE 5 mixing vane grids, are fabricated from zircaloy. This material was selected to take advantage of the material's inherent low neutron capture cross-section. The zircaloy IFM grid straps are manufactured using the same basic techniques as the zircaloy grid assemblies used for the Westinghouse OFA design and are joined to the guide thimbles via sleeves which are welded at the bottom of appropriate grid cells.

Grid impact testing has been performed on Zircaloy-4 structural grids indicative of the VANTAGE 5 design. The purpose of the testing was to determine the dynamic crush strength of the grids. The grid impact testing was performed at an elevated temperature of 600°F. This temperature is a conservative value representing the core average temperature at the mid-grid locations. The dynamic crush strength of the VANTAGE 5 grids envelop the calculated grid impact loading for FNP Units 1 and 2 during combined seismic and LOCA events.

The IFM grids are not intended to be structural members. The IFM grids do, however, share the loads of the structural grids during faulted loading and, as such, contribute to enhance the load carrying capability of the VANTAGE 5 fuel assembly. Grid impact testing has been performed on VANTAGE 5 zircaloy (structural) and

IFM grids. The dynamic crush strength of the VANTAGE 5 grids envelop the calculated grid impact loading during combined seismic and LOCA events. A coolable geometry is, therefore, assured at the IFM grid elevation, as well as at the structural grid elevation.

4.2.1.3 Design Evaluation

4.2.1.3.1 Fuel Rods

The fuel rods are designed to assure that the design bases are satisfied for Condition I and II events. This assures that the fuel performance and safety criteria (section 4.2) are satisfied.

Materials - Fuel Cladding

The desired fuel rod cladding is a material which has a superior combination of neutron economy (low absorption cross-section), high strength (to resist deformation caused by differential pressures and mechanical interaction between fuel and clad), high corrosion resistance (to coolant, fuel and fission products), and high reliability. Zircaloy-4 has this desired combination of cladding properties. As shown in reference 8, there is considerable pressurized water reactor (PWR) operating experience on the capability of zircaloy as a cladding material. Clad hydriding has not been a significant

power operation, the total power-peaking factor should be low enough to ensure that the power capability requirements are met. Analyses indicate that the first requirement can be met either by a single group or by two or more banks whose total worth equals at least the required amount. The axial power shape would be more peaked following movement of a single group of rods worth 3- to 4-percent $\Delta\rho$; therefore, four banks (described as A, B, C, and D in figure 4.3-36) each worth approximately 1-percent $\Delta\rho$ have been selected.

The position of control banks for criticality under any reactor condition is determined by the concentration of boron in the coolant. On an approach to criticality, boron is adjusted to ensure that criticality will be achieved with control rods above the insertion limit set by shutdown and other considerations. (See technical specifications.) Early in the cycle there may also be a withdrawal limit necessary to maintain a moderator temperature coefficient that is less than or equal to +7.0 pcm/°F for power levels up to 70 percent with a linear ramp to 0.0 pcm/°F at 100-percent power. Usual practice is to adjust boron to ensure that the rod position lies within the so-called maneuvering band, that is, such that an escalation from zero power to full power does not require further adjustment of boron concentration.

Ejected rod worths are given in subsection 15.4.6 for several different conditions. Experimental confirmation of these worths can be found by reference to startup test reports such as reference 7.

Allowable deviations because of misaligned control rods are discussed in the plant technical specifications.

A representative differential rod worth calculation for two banks of control rods withdrawn simultaneously (rod withdrawal accident) is given in figure 4.3-37.

Calculation of the control rod reactivity worth versus time-after-trip involves knowledge of the circuit delays, rod velocity, and differential reactivity worth. The rod position versus time-of-travel assumed is given in figure 4.3-38. For nuclear design purposes, the reactivity worth versus rod position is calculated by a series of steady-state calculations at various control rod positions, assuming all rods out at the time of trip since this is conservative with respect to the initial rate of reactivity insertion. Also, to be conservative, the rod of highest worth is assumed stuck out of the core and the flux distribution (and, thus, reactivity importance) is assumed to be skewed towards the bottom of the core. Figure 4.3-39 shows the results of these calculations.

The shutdown groups provide additional negative reactivity to assure an adequate shutdown margin. Shutdown margin is defined

as the amount by which the core would be subcritical at hot shutdown if all rod cluster control assemblies are tripped, but assumes that the highest-worth assembly remains fully withdrawn and no changes in xenon or boron concentration occur. The loss of control rod worth because of the depletion burnup of the absorber material is negligible, since only bank D rods may be in the core under normal operating conditions (near full power).

The values given in table 4.3-3 show that the available reactivity in withdrawn rod cluster control assemblies provides the design-basis minimum-shutdown margin allowing for the highest-worth cluster to be at its fully withdrawn positon. An allowance for uncertainty in the calculated worth of N-1 rods is made before determination of the shutdown margin.

4.3.2.7 Criticality of the Reactor During Refueling

Criticality of fuel assemblies outside of the reactor is precluded by adequate design of fuel transfer and fuel storage facilities and by administrative control procedures. This section identifies those criteria important to criticality safety analyses.

All enrichments discussed in this section are nominal enrichments. Nominal enrichments, when increased by the manufacturing uncertainty of 0.05 wt%, yield maximum enrichments.

4.3.2.7.1 New Fuel Storage

New fuel is stored in 21-in. center-to-center spaced racks in the new fuel storage facilities in a dry condition. This spacing is sufficient to maintain a subcritical array for the dry condition and for a postulated flooding event with unborated water. For the flooded condition with unborated water, assuming new fuel with the highest anticipated enrichment in place, the effective multiplication factor does not exceed 0.95. For the normally dry condition, the effective multiplication factor does not exceed 0.98 with fuel of the highest anticipated enrichment in place and assuming optimum moderation conditions (for example, because of the presence of aqueous foam or mist).

Westinghouse 17-x-17 OFA and VANTAGE 5 fuel assemblies with nominal enrichments up to 4.80 wt% can be safely stored in the new fuel rack utilizing all locations. Storage of fuel assemblies with nominal enrichments above 4.80 wt% and up to 5.00 wt% is also acceptable by taking credit for the same integral fuel burnable absorbers (IFBAs) required for storage in the spent-fuel rack (see figure 4.3-46). Westinghouse 17-x-17 LOPAR fuel assemblies with nominal enrichments of up to 4.25 wt% can be

safely stored in the new fuel rack utilizing all locations. No IFBA is required for LOPAR fuel.

Verification that the 21-in. center-to-center spaced new fuel storage racks meet the design criteria for normally dry storage assuming possible sources of optimum moderation, or for the flooded condition, with fresh nonborated water has been determined by the Monte-Carlo transport model KENO-IV/AMPX.[31, 32] The assumptions utilized in the analysis were as follows:

A. The fuel assembly is modeled at its most reactive point in life, and no credit is taken for any natural enrichment axial blankets.

B. All fuel rods contain uranium dioxide at a maximum enrichment of 4.85 wt% over the entire length of each rod. This maximum enrichment corresponds to a nominal enrichment of 4.80 wt%. Nominal enrichments above 4.80 wt% are also allowed by taking credit for IFBAs (see discussion below).

C. The fuel pellets are modeled at 96 percent of theoretical density without dishing or chamfering to bound the maximum fuel assembly loading.

D. No credit is taken for any U234 or U236 in the fuel, nor is any credit taken for the buildup of fission product poison material.

E. No credit is taken for any spacer grids or spacer sleeves.

F. All racks were modeled in the stainless-steel corner configuration rather than the stainless-steel canister configuration. This is more conservative because the neutron absorption effects of the stainless-steel corner design would be less than the stainless-steel canister.

G. For the fully flooded condition, a value of 1.0 g/cc was used for the density of water. The rack was modeled as an infinite in lateral (x and y) and axial (vertical) extent which precludes any neutron leakage from the array. This is conservative since the number of fuel assemblies modeled is infinite and the absorption effects of surrounding structure and walls are ignored.

H. For the low water density (optimum moderation) condition, various water densities were examined in the range of 0.06 g/cc to 0.12 g/cc to show that the optimum moderation reactivity peak satisfied the criticality criteria. The rack was modeled as an infinitely long double row of fuel assemblies with a nominal pitch of 21

in. Concrete walls and floor were modeled. Under low water density conditions, the presence of concrete is conservative because neutrons are reflected back into the fuel array more efficiently than they would be with just low density water. The area above the fresh fuel rack is filled water at the optimum moderation density.

With the above assumptions, the analysis confirmed that the criticality acceptance criteria were satisfied for OFA and VANTAGE-5 fuel assemblies with nominal enrichments up to 4.80 wt%. Based on comparison of the non-IFBA nominal enrichment limits of the new fuel rack (4.80 wt%) and spent-fuel rack (3.90 wt%), it was concluded that application of the spent-fuel rack IFBA requirements limit (figure 4.3-46) to the new fuel rack was conservative. Therefore, storage of OFA and VANTAGE-5 fuel assemblies in the new fuel rack with nominal enrichments above 4.80 wt% and up to 5.00 wt% is acceptable, provided the IFBA requirements of the spent-fuel rack IFBA-credit limit are satisfied.

Under normal conditions, the new fuel racks are maintained in a dry environment. The introduction of water into the new fuel rack area is the worst case accident scenario. The full density and low density optimum moderation cases are bounding accident situations which result in the most conservative fuel rack K_{eff}.

Other accidents can be postulated which would cause some reactivity increase (i.e., dropping a fuel assembly between the rack and wall or on top of the rack). For these other accident conditions, the double contingency principle is applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these other accident conditions, the absence of a moderator in the new fuel storage racks can be assumed as a realistic initial condition since assuming its presence would be a second unlikely event.

The results of all analyses verify that the NRC acceptance criteria continue to be met. That is, there is a 95-percent probability at a 95-percent confidence level (including uncertainties) that K_{eff} of the new fuel storage racks will be $\leq$0.95 when fully loaded and dry or flooded with unborated water and $\leq$0.98 for optimum moderation conditions or postulated accidents, as recommended in ANSI N18.3-1973.

4.3.2.7.2 Spent-Fuel Storage

The following information describes the design criteria and analysis techniques for storage of spent fuel in the Unit 1 and Unit 2 spent-fuel racks.

4.3.2.7.2.1 Storage of Spent-Fuel Assemblies The spent-fuel storage pool contains high-density poison racks, designed by PaR Systems, which have a center-to-center spacing of 10.75 in. They are designed to store Westinghouse 17-x-17 LOPAR fuel assemblies with a maximum nominal enrichment of 4.25 wt% U-235, and 17-x-17 OFA and VANTAGE-5 fuel assemblies with a maximum nominal enrichment of 3.9 wt% without credit for control rods or burnable absorbers.

The design basis for wet spent-fuel storage criticality analyses is that there is a 95-percent confidence level that the effective multiplication factor of the fuel storage array (K_{eff}), including uncertainties and biases, will be less than 0.95 in accordance with ANSI Standard N18.2-1973. The criticality analyses include a calculational bias, mechanical uncertainties, and consideration of 0.05 wt% enrichment variability.

For the criticality analysis, the fuel is assumed to be fresh and is at its most reactive point in life. The moderator is pure water at the temperature within the spent-fuel pool limits which yields the maximum reactivity. For the nominal case analysis, the rack array is assumed to be infinite in lateral extent (x and y). Consideration of mechanical uncertainties, enrichment variability, and 3-percent axial Boraflex shrinkage is included in the "worst-case" model. Credit is taken for the neutron absorption by the control poison material (Boraflex), the stainless-steel structural material of the spent-fuel racks, and some of the structural materials of the fuel assemblies. Credit is not taken for U-234 or U-236, buildup of fission product poison material, spacer grids or spacer sleeves, or natural enrichment axial blankets (if any).

The criticality calculation method and cross-section values are verified by comparison with critical experiment data for fuel assemblies similar to those for which the racks are designed. These benchmarking data are sufficiently diverse to establish that the method bias and uncertainty will apply to rack conditions which include strong neutron absorbers, large water gaps, and low moderator densities.

The design method which ensures the criticality safety of fuel assemblies in the spent-fuel storage rack uses the AMPX [31,33] system of codes for cross-section generation and KENO IV [32] for reactivity determination.

The 227 energy group cross-section library that is the common starting point for all cross-sections used for the benchmarks and the storage rack is generated from ENDF/B-V [33] data. The NITAWL [31] program includes, in this library, the self-shielded resonance cross-sections that are appropriate for each particular geometry. The Nordheim Integral Treatment is used. Energy and spatial weighting of cross-sections are performed by the XSDRNPM [31] program which is a one-dimensional Sn transport theory code. These multigroup cross-section sets are then used as input to KENO IV [32] which is a three-dimensional Monte Carlo theory program designed for reactivity calculations.

A set of 33 critical experiments has been analyzed using the above method to demonstrate its applicability to criticality analysis and to establish the method bias and uncertainty. The experiments range from water moderated, oxide fuel arrays separated by various materials (B4C, steel, water, etc.) that simulate light water reactor (LWR) fuel shipping and storage conditions [34] to dry, harder spectrum uranium metal cylinder arrays with various interspersed materials [35] (Plexiglas and air) that demonstrate the wide range of applicability of the method.

The average K_{eff} of the benchmarks is 0.992. The standard deviation of the bias value is 0.0008 Δk. The 95/95 one-sided tolerance limit factor for 33 values is 2.19. Thus, there is a 95-percent probability with a 95-percent confidence level that the uncertainty in reactivity, due to the method, is not greater than 0.0018 Δk.

The final K_{eff} value for the Farley spent-fuel racks is obtained by summing the K_{eff} value calculated by the "worst case" KENO-IV model, the calculational bias which was obtained from the benchmark work, and the total uncertainty which was obtained by a statistical combination of the calculational and methodology uncertainties. The calculational uncertainty is such that the true multiplication factor (K_{eff}) will be less than the calculated value with a 95-percent probability at a 95-percent confidence level.

The reactivity determined from the "worst case" KENO-IV model of the Farley spent-fuel racks was 0.9312 with a 95/95 uncertainty level of ± 0.0052 Δk. The following equation is used to develop the maximum K_{eff} for the Farley spent-fuel racks:

$$K_{eff} = K_{worst} + B_{method} + B_{part} + \sqrt{[(ks)^2_{worst} + (ks)^2_{method}]}$$

where:

K_{worst} = worst case KENO K_{eff} that includes material, mechanical, and enrichment tolerances and Boraflex absorber shrinkage

B_{method} = method bias determined from benchmark critical comparisons

B_{part} = bias to account for Boraflex poison particle self-shielding

ks_{worst} = 95/95 uncertainty in the worst case KENO k_{eff}

ks_{method} = 95/95 uncertainty in the method bias

Substituting calculated values in the order listed above, the result is:

$$K_{eff} = 0.9312 + 0.0083 + 0.0022 + \sqrt{[(0.0052)^2 + (0.0018)^2]} = 0.9472$$

Since K_{eff} is less that 0.95, including uncertainties at a 95/95 probability/confidence level, the acceptance criteria for criticality are met with OFA or VANTAGE-5 fuel enriched to a nominal 3.90-wt% U-238.

Storage of Westinghouse 17-x-17 OFA and VANTAGE 5 fuel assemblies with initial nominal enrichments above 3.9 wt% and up to 5.0 wt% is achievable by taking credit for IFBAs. Reactivity equivalencing is used to determine the required number of IFBAs as a function of enrichment. The concept of reactivity equivalencing is predicated upon the reactivity decrease

associated with the addition of IFBA fuel rods and fuel depletion.

Two analytical techniques are used to establish the criticality criteria for the storage of IFBA fuel in the fuel racks. The first method uses reactivity equivalencing to establish the poison material loading required to meet the criticality limits. The poison material considered in this analysis is a zirconium diboride (ZrB_2) coating manufactured by Westinghouse. The second method uses the fuel assembly infinite multiplication factor to establish a reference reactivity. The reference reactivity point is compared to the fuel assembly peak reactivity to determine its acceptability for storage in the spent-fuel racks. The reference reactivity is determined for a nominal fresh 3.90-wt% fuel assembly on a unit assembly basis in cold reactor geometry.

Based on reactivity equivalencing calculations performed for the Farley Units 1 and 2 spent-fuel racks, the required number of IFBAs for OFA and VANTAGE-5 fuel increases linearly from zero at 3.90 wt% (nominal) to 80 at 5.0 wt% (nominal), based on the standard linear B10 loading of 1.50 mg/in. (see figure 4.3-46). Alternatively, fuel assembly reactivity can be compared to the reference K-infinity of 1.455 to establish that reactivity is less than that of a fresh 3.90-wt% nominal assembly with no IFBAs.

Sensitivity calculations and IFBA credit reactivity equivalence calculations are performed with the transport theory computer code, PHOENIX.[38] PHOENIX is a depletable, two-dimensional, multigroup, discrete ordinates, transport theory code. A 25 energy group nuclear data library based on a modified version of the British WIMS [39] library is used with PHOENIX.

The PHOENIX code has been validated by comparisons with experiments where the isotopic fuel composition has been examined following discharge from a reactor.[41] In addition, an extensive set of benchmark critical experiments has been analyzed with PHOENIX.

Most postulated accidents in the spent-fuel rack will not result in an increase in reactivity. These include loss of cooling systems (reactivity decrease with increased temperature), dropping an assembly on top of the rack (rack structure maintains 10 in. of separation between dropped and stored assemblies, precluding interaction), or dropping an assembly into a position other than a storage cell (prevented by design of rack).

However, accidents can be postulated which would increase reactivity, i.e., misplacing an assembly in an unqualified position in the spent-fuel rack, misloading an assembly with an enrichment and IFBA combination outside of the acceptable area of figure 4.3-46, or dropping an assembly into an already loaded

cell. Misplacing an assembly in the spent-fuel rack is not considered credible since the same rack design and limits apply throughout the entire spent-fuel pool (one-region spent-fuel rack design). With a one-region design, the possibility of misloading a Region 1 assembly into a Region 2 area of the rack does not exist. Furthermore, the requirements of the spent-fuel rack IFBA limit will become a design constraint on future Farley reload core designs, and multilayered fuel vendor quality assurance controls on design, manufacturing, and shipment provide assurances that a potentially violating fuel assembly will not be delivered to the site.

For the accident of dropping of a fuel assembly into an already loaded cell, the upward axial leakage of that cell will be reduced; however, the overall effect on rack reactivity will be insignificant. This is because the total axial leakage in both the upward and downward directions for the entire spent-fuel array (over 1400 cells) is worth only 0.30-percent ΔK. Thus, minimizing the upward-only leakage of just a single cell will not cause any significant increase in rack reactivity (much less than 0.15-percent ΔK). Furthermore, the neutronic coupling between the dropped assembly and the already loaded assembly will be very low due to the several inches of assembly nozzle structure which would separate the active fuel regions.

The results of all analyses verify that there is a 95-percent probability at a 95-percent confidence level (including uncertainties) that K_{eff} of the spent-fuel storage racks will be less than 0.95 when flooded with unborated water.

4.3.2.7.2.2 <u>Storage of Loose Fuel Pellets and Rod Debris</u>

In the event of fuel rod failures, loose fuel pellets and fuel rod debris are collected in a pellet canister trap by a vacuum system. Up to five of these pellet canisters can be placed inside a transport container which is stored in the spent-fuel storage racks. In support of storing these canisters in the spent-fuel storage racks, a criticality analysis has been performed by Utility Associates International to demonstrate that the effective multiplication factor of the spent-fuel storage racks remains less than the NRC limit of 0.95. The assumptions utilized in the analysis were as follows:

A. Pellet canister traps 5 in. x 7 in. x 38 in. tall are utilized.

B. The canisters contain 4.3-wt% U-235 LOPAR fuel pellets in an array (moderator to uranium ratio) which gives the maximum reactivity. This maximum enrichment corresponds to a nominal enrichment of 4.25 wt%.

C. A pellet array infinite in the vertical direction.

D. No soluble poison in the spent-fuel pool water.

E. The spent-fuel pool racks are filled with fresh 4.3-wt% maximum U-235 fuel LOPAR assemblies. This maximum enrichment corresponds to a nominal enrichment of 4.25 wt%.

F. No burnable absorber rods in the fuel assemblies.

G. The center-to-center spacing of the spent-fuel racks is 10.75 in.

H. The effects of U-234 and Inconel spacer grids are included and the effects of other minor structural members are modeled as water.

The criticality analysis demonstrated that K_{eff} remains below 0.95, the design basis for the spent-fuel storage racks. Additionally, the criticality analysis for this case yielded a K_{eff} less than the case where all storage rack cells contain fuel assemblies. Because the analysis assumed the optimum moderator to uranium ratio and an infinite array in the vertical direction, the analysis bounds any number of pellets in one or more canisters up to and including all five canisters completely filled with pellets. Five canisters completely filled with pellets contain less uranium than one fuel assembly. From a criticality standpoint, any number of transport containers holding up to five pellet canisters filled to any extent with LOPAR fuel pellets up to 4.3-wt% U-235 maximum can be stored anywhere in the spent-fuel pool and still be less limiting than having all rack cells filled with fuel assemblies.

Since a transport canister with five completely filled pellet canisters will contain a smaller number of fuel pellets than a fuel assembly, it will be less limiting than a fuel assembly from a spent-fuel pool bulk temperature standpoint. Thus, no thermal analysis is required to demonstrate compliance with spent-fuel pool bulk temperature limits. However, a thermal analysis was performed by Southern Company Services which demonstrated that no local boiling would occur in a pellet canister trap which contained up to 1000 fuel pellets. For this analysis, five pellet canisters each containing 1000 fuel pellets were assumed to be placed in the transport canister in the spent-fuel pool rack 150 hours after reactor shutdown following infinite operation. Additionally, the spent-fuel pool was assumed to be at its maximum allowable temperature of 140°F and natural convection was assumed to be the only mode of heat transfer.

Therefore, any number of transport containers can be stored in the spent-fuel storage racks, and up to five pellet canisters can be stored in each transport container. From a criticality standpoint, a canister can be filled up to its maximum capacity with LOPAR fuel pellets of any enrichment up to and including 4.3-wt% maximum U-235. Additional similar canisters also filled to any fraction of their capacities can be stacked on top of the

original canister within the transport container and remain safe from a criticality standpoint. However, from a heat load standpoint, a pellet canister has only been evaluated for a maximum of 1000 pellets.

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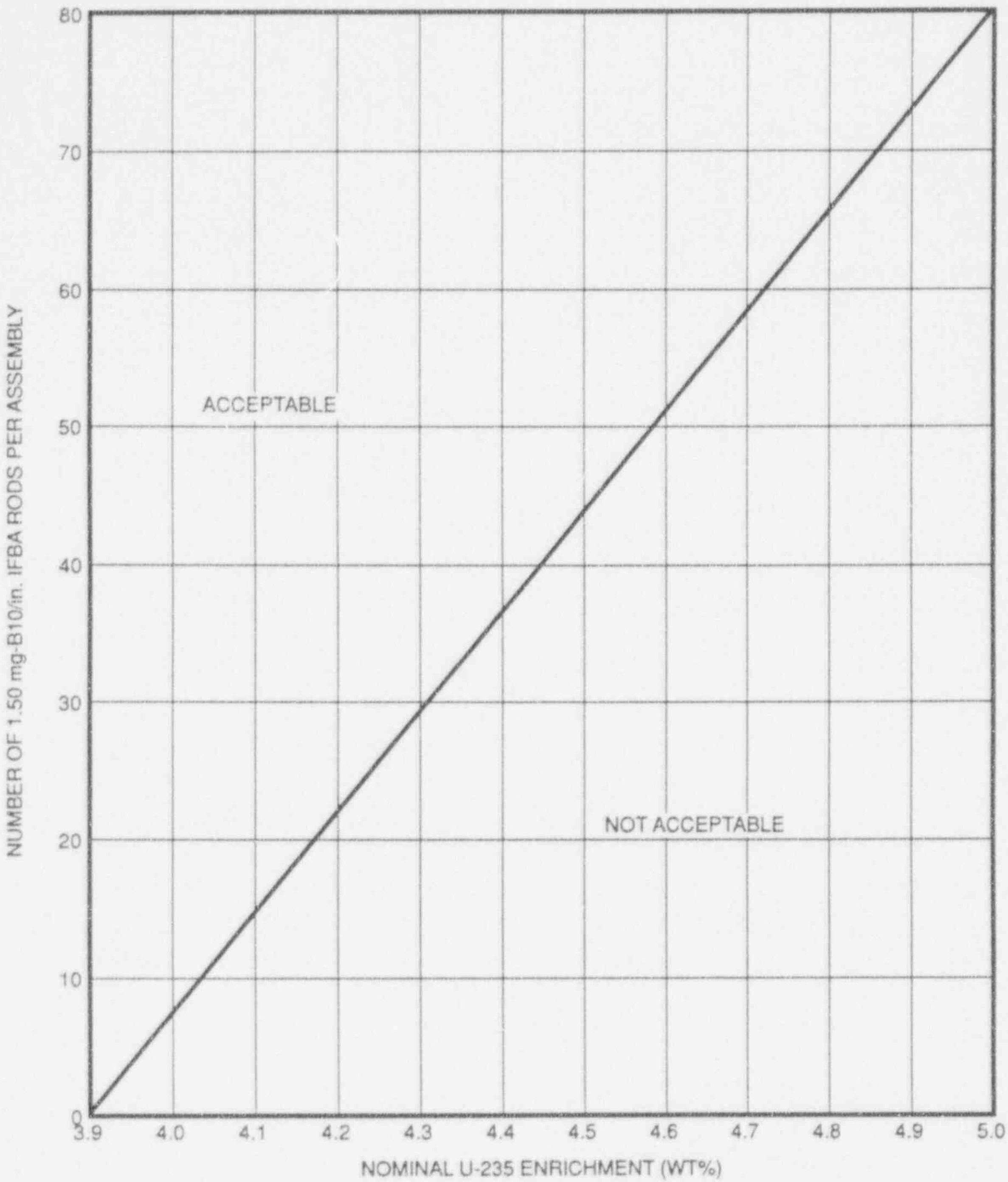

NOTE: THE ENRICHMENTS USED IN THIS FIGURE ARE NOMINAL ENRICHMENTS. NOMINAL ENRICHMENTS, WHEN INCREASED BY 0.05 WT% MANUFACTURING UNCERTAINTY, YIELD MAXIMUM ENRICHMENTS.

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JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	FARLEY UNITS 1 & 2 SPENT-FUEL STORAGE MINIMUM IFBA REQUIREMENTS FOR OFA AND VANTAGE-5 FUEL
	FIGURE 4.3-46

limits for a controlled vibration mechanism are established from studies of vortex shedding and turbulent pressure fluctuations. The crossflow velocity required to exceed fluid elastic stability limits is dependent on the axial location of the blockage and the characterization of the crossflow (jet flow or not). These limits are greater than those for vibratory fuel rod wear.

4.4.4 TESTING AND VERIFICATION

4.4.4.1 Tests Prior to Initial Criticality

A reactor coolant flow test is performed following fuel loading, but prior to initial criticality. Coolant loop pressure drop data are obtained in this test. These data, in conjunction with coolant pump performance information, allow determination of the coolant flowrates at reactor operating conditions. This test verifies that proper coolant flowrates were used in the core thermal and hydraulic analysis.

Following initial criticality, periodic testing in accordance with the technical specification DNB surveillance for RCS flow will ensure that actual core flowrates are bounded by the assumptions found in the core thermal and hydraulic analysis.

4.4.4.2 Initial Power and Plant Operation

Core power distribution measurements are made at several core power levels (see paragraph 4.3.2.2.7). These tests are used to ensure that conservative peaking factors are used in the core thermal and hydraulic analysis.

Additional demonstration of the overall conservatism of the THINC analysis was obtained by comparing THINC predictions to incore thermocouple measurements. These measurements were performed on the Zion reactor.[99] No further inpile testing is planned.

An additional test is provided which measures how the N35 and N36 detector currents are affected by Control Bank D insertions at a constant power level between 30 and 35 percent. The results of this rod shadowing test are used to optimize the calibration of the IR instruments.

4.4.4.3 Component and Fuel Inspections

Inspections performed on the manufactured fuel are delineated in paragraph 4.2.1.4. Fabrication measurements critical to thermal and hydraulic analysis are obtained to verify that the engineering hot channel factors employed in the design analyses (paragraph 4.4.2.3.4) are met.

4.4.5 INSTRUMENTATION APPLICATION

4.4.5.1 Incore Instrumentation

Instrumentation is located in the core so that by correlating movable neutron detector information with fixed thermocouple information, radial, axial, and azimuthal core characteristics may be obtained for all core quadrants.

The incore instrumentation system is comprised of thermocouples positioned to measure fuel assembly coolant outlet temperatures at preselected positions and fission chamber detectors, positioned in guide thimbles, which run the length of selected fuel assemblies to measure the neutron flux distribution. Figures 4.4-16 and 4.4-17 show the number and location of instrumented assemblies in the core for Units 1 and 2, respectively.

The core exit thermocouples provide a backup to the flux monitoring instrumentation for monitoring power distribution. The locations of the core exit thermocouples have been chosen to provide an adequate indication of the radial symmetry of the core power distribution. The routine, systematic collection of thermocouple readings during monthly surveillance testing provides a data base. From this data base, abnormally high or abnormally low readings, quadrant temperature tilts, or systematic departures from a prior reference map can be deduced.

The movable incore neutron detector system would be used for more detailed mapping if the thermocouple system were to indicate an abnormality. These two complementary systems are more useful when taken together than either system alone would be. The incore instrumentation system is described in more detail in paragraph 7.7.1.9.

The incore instrumentation is provided to obtain data from which fission power density distribution in the core, coolant enthalpy distribution in the core, and fuel burnup distribution may be determined.

4.4.5.2 Overtemperature and Overpower ΔT Instrumentation

The overtemperature ΔT trip protects the core against low DNBR. The overpower ΔT trip protects against excessive power (fuel rod rating protection).

As discussed in paragraph 7.2.1.1.2, factors included in establishing the overtemperature ΔT and overpower ΔT trip setpoints include the reactor coolant temperature in each loop

requirements, which became effective on August 16, 1973, are more stringent than the applicable code requirements for Farley Unit 2.

The actual fracture toughness data for RCPB pressure-retaining applications in the steam generators and pressurizer are tabulated in table 5.2-39. In all cases, the applicable ASME Code requirements, as well as the intent of 10 CFR 50 Appendix G, are satisfied.

SA 508 Class 2a material and SA 533 Class 2 material was used in the Farley Unit 2 pressurizer. Neither of these materials was used in primary-side (RCPB) pressure retaining applications of the Farley Unit 2 steam generators. The fracture toughness data for these materials are included in table 5.2-39. The adequacy of the fracture toughness properties of these materials has been documented in reference 10.

The following discussion demonstrates that the intent of the Appendix G, Paragraph III.B.3 requirements is satisfied.

Reactor Vessel - Combustion Engineering calibrated Charpy V notch test machines in accordance with Watertown Arsenal Standards every six months. Temperature instruments, calibrated in accordance with ASTM-E-23, were purchased every three months.

These calibrations were performed in accordance with the requirements of the ASME Code 1968 Edition through Summer 1970 Addenda (Appendix IX-221 and 260), which is the applicable Code for the Farley Unit 2 reactor vessel. The Charpy V notch test machine calibrations were recorded. The temperature instrument calibrations were not recorded; however, thermometers qualified to ASTM standards were purchased, used for the certified time period, and replaced with new qualified thermometers.

Combustion Engineering required that all of its vendors who furnished materials or parts (for Farley unit 2) to be on an approved vendors list. Each vendor was required to have a quality control system in accordance with #N-335 of the 1968 ASME Code through Summer 1970 Addenda. Periodic audits of these vendors were performed by CE QA personnel.

It should be noted that the Farley Unit 2 reactor vessel was partially furnished by B & W. Material furnished by B & W was accepted on the basis of material certifications; therefore, no QA audits were performed for those by CE.

Steam Generators and Pressurizers - Charpy V notch test machine calibration at W Tampa plant was performed yearly using samples obtained from Watertown Arsenal. Temperature instrument calibration was performed with standards traceable to the National Institute of Standards and Technology.

All material suppliers have been either surveyed by ASME auditors or W Tampa Plant Product Assurance to obtain supplier certifications. A sampling of one of the major material suppliers indicated that Charpy V notch test machine calibrations were recorded and that calibrated temperature instruments were purchased (as replacements) on a yearly basis.

The following discussion demonstrates that the intent of the Appendix G, Paragraph III.B.4 requirements is satisfied.

Reactor Vessel - The personnel performing the Charpy testing at Combustion Engineering were qualified by schooling, training, and many years of experience. Their qualifications to perform this work have been certified by qualified supervisory personnel. This meets the requirements of the applicable ASME Code 1968 Edition through Summer 1970 Addenda (Appendix IX 221d).

Steam Generators and Pressurizer - Charpy impact tests are performed at W Tampa Plant by Level III and Level II personnel who have a minimum of 5 years directly-related testing experience.

5.2.4.3 Operating Limitations During Startup and Shutdown

The heatup and cooldown curves for Units 1 and 2 are based on the fracture toughness properties of each vessel, as given in tables 5.2-24 and 5.2-25 and the calculation methods described in reference 6. Tables 5.2-24 and 5.2-25 indicate that the original maximum reference nil-ductility temperatures (RT_{NDT}) of the Unit 1 and 2 reactor vessels are not higher than +60°F. Allowable pressures as a function of the rate of temperature change and the actual temperature relative to the vessel RT_{NDT} are established according to the methods given in Appendix G, "Protection Against Non-Brittle Failure," of Section III of the ASME Pressure Vessel and Boiler Code. Curves showing reactor coolant system heatup and cooldown limitations are given in figures 5.2-7 and 5.2-8.[13,14]

These curves are based on temperature scale relative to the limiting RT_{NDT} of the vessels, including appropriate estimates

5.3 THERMAL HYDRAULIC SYSTEM DESIGN

5.3.1 ANALYTICAL METHODS AND DATA

The thermal and hydraulic design bases of the reactor coolant system (RCS) are described in sections 4.3 and 4.4 in terms of core heat generation rates, departure from nucleate boiling ratio (DNBR), analytical models, peaking factors, and other relevant aspects of the reactor.

5.3.2 OPERATING RESTRICTIONS ON PUMPS

In order to meet the net positive suction head requirements for operation of the reactor coolant pumps, the operating procedures state that the pressure differential across the No. 1 seal must be at least 200 psig before operating the reactor coolant pump. In order to achieve this pressure differential, the RCS pressure must be maintained at or above approximately 325 psig with the volume control tank pressure between 15 and 20 psig.

5.3.3 BOILING WATER REACTOR (BWR)

5.3.4 TEMPERATURE-POWER OPERATING MAP

The effects of reduced core flow because of inoperative pumps is discussed in subsections 5.5.1, 15.2.5, and 15.3.4.

Natural circulation capability of the system is shown in table 5.3-1.

The issue of steam formation in the RCS was made part of TMI Action Plan Requirement II.K.2.17. The potential for voids being generated in the RCS during anticipated transients is accounted for in present analysis models. The transient analyses performed using these models demonstrate that steam voids will not result in unacceptable consequences during anticipated transients.

5.3.5 LOAD FOLLOWING CHARACTERISTICS

The RCS is designed on the basis of steady-state operation at full-power heat load. The reactor coolant pumps utilize constant speed drives as described in section 5.5, and the reactor power is controlled to maintain average coolant temperature at a value which is a linear function of load, as described in section 7.7. Operation with one pump out of service requires adjustment only in reactor trip system setpoints, as discussed in section 7.2.

5.3.6 TRANSIENT EFFECTS

Transient effects are evaluated as follows: complete loss of forced reactor coolant flow (15.3.4); partial loss of forced reactor coolant flow (15.2.5); startup of an inactive loop (15.2.6); loss of load (15.2.7); loss of normal feedwater (15.2.8); loss of offsite power (15.2.9); and accidental depressurization of the reactor coolant system (15.2.12).

5.3.7 THERMAL AND HYDRAULIC CHARACTERISTICS SUMMARY TABLE

The thermal and hydraulic characteristics are given in tables 4.3-1, 4.4-1, and 4.4-2.

limits of vibration, and are adjustable over the full range of the motor scale.

All parts of the pump in contact with the reactor coolant are austenitic stainless steel except for seals, bearings, and special parts. Component cooling water is supplied to the two oil coolers on the pump motor and to the pump thermal barrier heat exchanger.

The pump shaft, seal housing, thermal barrier, bolting ring, and motor stand can be removed from the casing as a unit without disturbing the reactor coolant piping. The flywheel is available for inspection by removing the cover.

The performance characteristic shown in figure 5.5-2 is common to all of the fixed-speed, mixed-flow pumps, and the "knee" at about 45-percent design flow introduces no operational restrictions, since the pumps operate at full speed.

5.5.1.3 Design Evaluation

5.5.1.3.1 Pump Performance

The reactor coolant pumps are sized to deliver flow at rates which equal or exceed the required flowrates. Initial reactor coolant system (RCS) tests confirm the total delivery capability. Thus, assurance of adequate forced circulation coolant flow is provided prior to initial plant operation.

The reactor trip system ensures that pump operation is within the assumptions used for loss-of-coolant flow analyses, which also assures that adequate core cooling is provided to permit an orderly reduction in power if flow from a reactor coolant pump is lost during operation.

An extensive test program has been conducted for several years to develop the controlled-leakage, shaft-seal for pressurized-water reactor (PWR) applications. Long-term tests were conducted on less than full-scale prototype seals as well as on full-size seals. Operating plants continue to demonstrate the satisfactory performance of the controlled-leakage, shaft seal pump design.

The support of the stationary member of the No. 1 seal ("seal ring") is such as to allow large deflections, both axial and tilting, while still maintaining its controlled gap relative to the seal runner. Even if all the graphite were removed from the pump bearing, the shaft could not deflect far enough to cause opening of the controlled-leakage gap. The "spring rate"

of the hydraulic forces associated with the maintenance of the gap is high enough to ensure that the ring follows the runner under very rapid shaft deflections.

Testing of pumps with the No. 1 seal entirely removed (full reactor pressure on the No. 2 seal) shows that relatively small leakage rates would be maintained for long periods of time (approximately 100 hours) even if the No. 1 seal fails entirely. The plant operator is warned of this condition by the increase in No. 1 seal leakoff, and has time to safely shut down the reactor without significant leakage of reactor coolant to the containment. Thus, it may be concluded that gross leakage from the pump would not occur even if seals were to suffer physical damage.

The effect of loss of offsite power on the pump itself is to cause a temporary stoppage in the supply of injection flow to the pump seals and also of the cooling water for seal and bearing cooling. The emergency diesel generators are started automatically upon loss of offsite power, so that component cooling flow is automatically restored. Seal water injection flow is subsequently restored by manually restarting a charging pump on diesel power.

5.5.1.3.2 Coastdown Capability

It is important to reactor operation that the reactor coolant continues to flow for a short time after reactor trip. In order to provide this flow in a station blackout condition, each reactor coolant pump is provided with a flywheel. Thus, the rotating inertia of the pump, motor, and flywheel is employed during the coastdown period to continue the reactor coolant flow. The coastdown flow transients are provided in the figures in section 15.3.

The pump is designed for the safe shutdown earthquake (SSE) at the site, and the integrity of the bearings is described in paragraph 5.5.1.3.4. Hence, it is concluded that the coastdown capability of the pumps is maintained even under the most adverse case of a blackout coincident with SSE. Core flow transients and figures are provided in subsection 15.2.5 and 15.3.4.

5.5.1.3.3 Flywheel Integrity

Demonstration of integrity of the reactor coolant pump flywheel is discussed in subsection 5.2.6.

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Materials compatibility considerations are discussed in appendix 6A.

6.2.1.3 Design Evaluation

6.2.1.3.1 Assurance of Containment Leaktightness

The containment leakage surveillance system, used to assure containment leaktightness during plant operation, is described in subsection 6.2.1.4.

6.2.1.3.2 System Capability Analysis

A discussion of system capability is given in subsection 6.2.1.4.

6.2.1.3.3 Containment Pressure Transient Analysis

A. Pipe Break Spectrum

In the event of a hypothetical LOCA, or main steam line break, the release of the coolant from the rupture area will cause the high pressure, high temperature fluid to rapidly flash to steam and water within the containment. The release of this mass and energy will result in a rise in the pressure and temperature of the containment atmosphere. The rate and magnitude of the pressure increase depend upon the nature, location, and size of the rupture. In order to establish the controlling rupture, a spectrum of primary and secondary coolant breaks is considered. The reactor coolant breaks examined are from a condition of full rated power. Secondary coolant system break analysis considers a spectrum of breaks at different power levels. All of these main steam line breaks allow complete blowdown of one steam generator. These postulated accidents are evaluated to determine their significance in selecting a containment design basis. The most severe of these accidents is selected as the controlling containment DBA.

B. Initial Conditions and Input Data

The containment pressure analysis input data have been based upon the final plant design. A conservative prediction of LOCA consequences has been assured by

determining expected values of containment initial conditions and geometric and thermodynamic parameters. A thorough discussion of the input data is given below.

The containment design parameters which determine the net free internal volume, the containment surface areas, and the design pressure and temperature are given in tables 6.2-1 and 6.2-2. As an additional conservatism, the volume occupied by the reactor coolant prior to the LOCA is included as occupied volume rather than free volume.

The initial conditions within the reactor coolant system and the containment system prior to accident initiation are given in table 6.2-3. The blowdown analyses are conducted assuming either a maximum or minimum of available ECCS is in operation. The containment system is assumed to be at ambient pressure, and maximum inside and maximum outside design operating air temperatures to minimize heat transfer during a LOCA.

The containment heat sink data used in the LOCA analysis are fully described in tables 6.2-2 and 6.2-4. Table 6.2-2 lists the geometry of each heat sink and the way it is modeled for the analysis. An air gap equivalent to a thermal resistance of 0.01 ft/h/°F/Btu is postulated for the interface between the containment liner and wall. No contact is presumed for the interface between the stainless steel refueling canal liner and the concrete.

Table 6.2-4 lists the material properties and heat transfer coefficients used in the analysis. The coating properties were supplied by the manufacturer. Metal and concrete properties are typical for the temperature range expected. The steel imbedded in the concrete was not considered in the concrete conductivity. Containment air cooler unit duty, Btu/h, as a function of containment saturation temperature, is given in figure 6.2-42. The RHR heat exchanger duty is given in figure 6.2-44 as a function of sump liquid temperature.

C. Accident Identification and Results

Containment pressure/temperature vs. time responses for the various breaks are shown in figures 6.2-1 through 6.2-41. The peak pressures, times of peak pressure, peak temperatures, times of peak temperature, and blowdown energy releases at the times of peak pressure are given in table 6.2-6 for the spectrum of breaks for LOCA. Based on the results presented, the DEPSG with minimum ESF has been identified as the limiting LOCA. The limiting LOCA was reanalyzed as described in paragraph 6.2.1.3.12, and the resultant peak pressure is well below the design pressure of 54 psig, as shown in table 6.2-6.

The blowdown mass and energy release rates as a function of time for the limiting LOCA case are shown in figures 6.2-46 and 47 and table 6.2-10. Figure 6.2-46 represents the blowdown mass and energy release rates and figure 6.2-47 represents the reactor vessel reflood and long-term mass and energy release rates.

For the time of peak pressure, a detailed mass and energy balance has been performed on the reactor coolant system and containment. These data are given in tables 6.2-19 and 6.2-20. Table 6.2-19 lists the calculated containment pressures, temperatures, and masses for the time of pipe ruptures and the time of peak pressure with the limiting LOCA. Table 6.2-20 gives the energy distribution in the reactor coolant system and in the containment at the time of the break and at the time of peak pressure. This table verifies the energy balance during the containment pressurization period, since the total reactor coolant system energy release equals the net gain in the containment system energy.

The containment condensing heat transfer coefficient vs. time for the limiting LOCA case is shown in figure 6.2-48. The initial portion of the curve is the Modified Tagami value with a maximum of 230 Btu/h-ft^2-°F at 20 s. After 20 s, the value decays to the Uchida value, which is dependent upon the slowly changing steam air mass ratio in the containment atmosphere.

6.2.1.3.4 Method of Analysis

This analysis presents the mass and energy releases to the containment subsequent to a hypothetical loss-of-coolant accident. The release rates are calculated for pipe failure at three distinct locations:

A. Hot leg (between vessel and steam generator).

B. Pump suction (between steam generator and pump).

C. Cold leg (between pump and vessel).

During the reflood phase, these breaks have the following different characteristics. For a cold leg pipe break, all of the fluid that leaves the core must vent through a steam generator and becomes superheated. However, relative to breaks at the other locations, the core flooding rate (and, therefore, the rate of fluid leaving the core) is low because all the core vent paths include the resistance of the reactor coolant pump. For a hot leg pipe break the vent path resistance is relatively low, which results in a high core flooding rate, but the majority of the fluid which exits the core bypasses the steam generators in venting to the containment. The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, and steam generator heat addition as in the cold leg break. As a result, the pump suction breaks yield the highest energy flowrates during the postblowdown period. The spectrum of breaks analyzed includes the largest cold and hot leg breaks, reactor inlet and outlet, respectively, and a range of pump suction breaks from the largest to a 3.0 ft^2. Because of the phenomena of reflood as discussed above, the pump suction break location is the worst case. This conclusion is supported by studies of smaller hot leg breaks, which have been shown on similar plants to be less severe than the double-ended hot leg. Cold leg breaks, however, are lower both in the blowdown peak and in the reflood pressure rise. Thus, an analysis of smaller pump suction breaks is representative of the spectrum of break sizes.

The LOCA analysis calculational model is typically divided into three phases: blowdown, which includes the period from accident occurrence (when the reactor is at steady-state full power operation) to the time when zero break flow is first calculated; refill, which is from the end of blowdown to the time the ECCS fills the vessel lower plenum; and reflood, which begins when water starts moving into the core and continues until the end of the transient. For the pump suction break, a fourth phase is included; froth boiling in the steam generator tubes after the core has been quenched.

fraction brought out times the broken loop steam generator energy remaining divided by the steam generator cooling rate. Subsequent to this time the post reflood table as presented may be used.

Broken Loop Steam Generator - Depressurization Stage

The amount of energy to be brought out is the original amount less what is brought out to reach equilibrium. The heat addition rate is this amount divided by the depressurization time. The boiloff rate is this rate divided by latent heat. The energy addition rate is the boiloff rate times saturated vapor enthalpy.

Intact Loop Steam Generator - Equilibrium Stage

The same procedure as for the broken loop is used here. However, metal and core energy is lumped with the steam generator energy for this calculation. The fraction to be brought out to attain equilibrium equals the reference containment pressure and the actual containment pressure divided by the reference value. The rate of addition to the containment is 80.3 lb/s at 817 s. This cools the steam generator and metal at 65300 Btu/s. Thus the duration of extension of the postreflood table is the fraction times the available energy divided by the rate of cooling. This should not be extended beyond recirculation because the continued condensation effect is implicit in these numbers and should change after recirculation. An alternate calculation would be required if these overlap.

Intact Loop Steam Generator - Depressurization Stage

Again the procedure used here is the same as the broken loop case except decay heat should be added to the heat addition rate. The amount of energy to be brought out is the original amount less what is brought out to reach equilibrium. The heat addition rate is this amount divided by the depressurization rate. The boiloff rate is this rate divided by latent heat. The energy addition rate is the boiloff rate times saturated vapor enthalpy plus the decay heat rate at the time of interest.

D. Containment Pressure Analysis

The containment pressure analyses to determine the limiting LOCA were performed using the Westinghouse COMPACT computer program that was developed for the purpose of transient analysis of atmospheres in multicompartment containments of water-cooled nuclear power plants.

The COMPACT model predicts both the pressure and temperature within the containment regions and the temperatures in the containment structures. It is assumed that separate blowdown and core thermal behavior studies have been made by the nuclear steam supply system (NSSS) manufacturer to determine mass and/or energy input rates from sources such as: the release of reactor coolant, chemical reactions, and decay energy and sensible heat release, which may cause heating or boiloff of residual water in the reactor vessel or superheating of steam as it passes through the reactor system and enters the containment through the postulated point of reactor coolant system rupture.

The COMPACT model treats the containment and the heat transfer surfaces following a LOCA. Included in this model are ESF and analytical techniques to enable calculation of their effects upon the containment. Several options have been incorporated in the model to facilitate use of these features.

COMPACT calculates a pressure-time transient with stepwise iteration between the thermodynamic state points. The iterations are based on the laws of the conservation of mass and energy together with their thermodynamic relationships. Superposition of heat input functions is assumed so that any combination of coolant release, decay heat generation, and sensible heat release can be used with appropriate ESF features to determine the containment pressure-time history associated with a LOCA.

The program uses a two-region containment model consisting of: the containment atmosphere (vapor region) and the sump (liquid region). Mass and energy are transferred between the liquid and vapor regions by boiling, condensation, or liquid dropout. Evaporation is not considered. A convective heat transfer coefficient can be specified between the sump liquid and atmosphere vapor regions. However, since any heat transfer in this mode is small, a conservative coefficient of zero is generally assumed. Each region is assumed homogeneous, but a temperature difference can exist between regions. Any moisture condensed in the vapor region during a time increment is assumed to fall immediately into the liquid region. Noncondensible gases are included in the vapor region. Thermodynamic state points of steam in the saturated

and superheated state are taken from the 1976 ASME steam tables.

6.2.1.3.5 Special Pressure Reduction Containment Concepts

This section is not applicable.

6.2.1.3.6 Long-Term Containment Performance

The long-term results of the limiting LOCA case have been evaluated to verify the ability of the ECCS and containment heat removal system (CHRS) to keep the reactor vessel flooded and maintain the containment below the design conditions following a LOCA, as described in paragraph 6.2.1.3.12. This evaluation has been based upon conservatively assumed performance of the engineered safety features. The containment fan cooling units are actuated at 31 s and the containment spray system at 55 s. The CHRS operation mode assumes one spray pump and one atmosphere fan cooling coil.

The residual decay heat rate is shown in figure 15.1-6 as watts per watt at maximum NSSS power level, 2652 MWt. For conservatism it is assumed that decay heat is added to the reactor vessel water at a 20 percent greater rate than predicted by the residual decay heat curve. Sensible heat remaining in the primary and secondary systems at the end of the post reflood failure is added to the reactor vessel water, as listed in table 6.2-14. These criteria aid in assuring a conservative prediction of the third containment pressure peak, which occurs during sump water recirculation, and demonstrate the ability of the CHRS to maintain pressure effectively at a fraction of that value occurring during blowdown.

The containment pressure time response for the limiting LOCA case out to 10^5 s is shown in figure 6.2-1 for the safeguards performance mode outlined in table 6.2-5. The maximum pressure of 42 psig occurs at 272 s, followed by containment depressurization. At 3714 s, safety injection water from the sump begins to recirculate as the RWST reaches low level. The containment continues to depressurize until 6600 s, when containment spray water begins to recirculate from the sump as the RWST reaches low-low level. During sump water recirculation, a second pressure peak occurs due to steam evolution from the reactor because of boiloff of the hotter core injection water. The containment atmosphere and sump temperatures versus time are given in figure 6.2-40 for the limiting LOCA. The peak atmosphere temperature of 268°F occurs at 272 s. The maximum sump water temperature of 277°F occurs at 1170 s.

The energy distribution in the containment versus time is shown in figure 6.2-82. The energy components include: the vapor energy content (steam plus air), the liquid energy content, and the containment structures energy absorption. The total energy content of the vapor and the liquid is also shown. The total energy increases to 8.2×10^8 Btu at about 9000 s due to net energy addition to the containment and thereafter decreases as the fan cooling coils and heat exchanger provide a net energy removal from the containment.

A chronology of events with time for the 8.25 ft^2 limiting LOCA case is given in table 6.2-21 from the time of pipe rupture to 10^6 s when accident calculations were terminated. At that time, the containment pressure is 1.3 psig.

The temperature response of the containment structure is shown in figure 3.8-15 at several points in time for the limiting LOCA. The containment wall liner plate reaches a maximum temperature of 249°F at 1000 s. The temperature response of the internal concrete structures is shown on figure 6.2-83.

6.2.1.3.7 Accident Chronology

An accident chronology for events occurring subsequent to the limiting LOCA is shown in table 6.2-21. It is assumed that time equals zero at the start of the LOCA.

6.2.1.3.8 Energy Balance

An energy balance for the limiting LOCA is given in table 6.2-20.

6.2.1.3.9 Post-LOCA Parameters

This section contains plots of various post-LOCA parameters as a function of time. The heat generation rate from core decay heat is shown in figure 15.1-6. The heat removal rate from the RHR heat exchanger and from the containment air cooler is shown in figures 6.2-84 and 6.2-85, respectively. The containment pressure/temperature vs. time profiles are shown in figures 6.2-1 through 6.2-5 and figure 6.2-40.

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D. Secondary Shield Annulus

The steam lines penetrate the containment vessel at elevation 138 ft, which is below the top of the secondary shield walls. An analysis was made to determine if local high pressures could occur in the relatively confined space. Steam flows around the annulus, upward and directly to the containment upper volume.

At no time do differential pressures across the walls exceed 0.25 psid.

6.2.1.3.11 Main Steam Line Ruptures Inside Containment

A. Introduction

Steam line ruptures occurring inside the containment structure may result in significant pressure and temperature transients. In order to determine the rupture which results in the worst case, a complete spectrum of ruptures were analyzed. This extensive analysis was necessary due to the number of variables on the determination of the blowdown data. Because of this, the blowdown data reflect the conditions and operations of the power plant. A summary of the plant particular data utilized for this analysis is given in table 6.2-11.

After the blowdown was determined, a case-by-case containment analysis was performed using the COMPACT computer code. As described in paragraph 6.2.1.3.12, COMPACT is a containment pressure/temperature transient analysis code. A more detailed description of COMPACT is contained in Westinghouse WCAP-11844.[18]

The LOCA model heat sinks were used in the analysis. The pertinent information and data for these are given in tables 6.2-2 and 6.2-4.

The following list of ruptures was analyzed:

Case 1: Full double-ended rupture at 102-percent power.

Case 2: 0.7 ft^2 double-ended rupture at 102-percent power.

Case 3: 0.6 ft^2 double ended rupture at 102% power.

Case 4: 0.645 ft^2 split rupture at 102% power.

Case 5: Full double ended rupture at 70% power.

Case 6: 0.6 ft^2 double ended rupture at 70% power.

Case 7: 0.5 ft^2 double ended rupture at 70% power.

Case 8: 0.681 ft^2 split rupture at 70% power.

Case 9: Full double ended rupture at 30% power.

Case 10: 0.5 ft^2 double ended rupture at 30% power.

Case 11: 0.4 ft^2 double ended rupture at 30% power.

Case 12: 0.7065 ft^2 split rupture at 30% power.

Case 13: Full double ended rupture at hot standby.

Case 14: 0.2 ft^2 double ended rupture at hot standby.

Case 15: 0.1 ft^2 double ended rupture at hot standby.

Case 16: 0.3 ft^2 split rupture at hot standby.

B. Mass and Energy Releases Following a Main Steamline Rupture Inside Containment

The mass and energy release data for each case are determined using the methods given in Appendix A, "Safety Analysis Standard 12.2, Rev. 1, Mass Energy Releases to Containment Following a Main Steam Rupture for Series 51 and D Steam Generators," of WCAP-8822 (Proprietary) and WCAP-8860 (Non-Proprietary).

C. Evaluation of Effects of Various Single Failures

The method of determining the blowdown assumes no failure in steam or feedwater isolation. This blowdown is used in conjunction with minimum

containment spray and fan coolers to allow for failure of a diesel generator. Steam line isolation failure is not postulated, since there are two redundant swing disc trip valves in each steam line. However, since these valves stop flow only in the forward direction, the mass/energy release to containment was modified to include the entire steam piping volume downstream of the isolation valves for the other steam generators, including the steam line header and steam dump piping.

For a complete description of the functions that provide the necessary protection against a steam pipe rupture refer to FSAR paragraph 15.4.2.1.1, items A through D.

D. Pressure Temperature Results

The pressure/temperature results of the analysis are illustrated in figures 6.2-6 through 6.2-37. The highest pressure obtained was 47.4 psig for a 0.4-ft^2 double-ended rupture at 30-percent power. The highest temperature reached was 287°F for a double-ended rupture at 102-percent power.

Equipment Temperature Transient

A transient main steam line break (MSLB) containment thermal analysis was performed for the worst case of the spectrum of breaks for each type of Class 1E component inside containment to determine the peak component surface temperatures. The methodology used to calculate the equipment surface temperatures was based on the NRC staff's approved assumptions discussed in Appendix B of NUREG-0588. These analyses show that the peak surface temperature resulting from the MSLB environment do not exceed the qualification temperature for each type of component. The methodology and results of the calculations of the equipment surface temperatures are reported in the docketed Farley response to NUREG-0588 and I&E Bulletin 79-01B.

The above results show that the maximum pressure in the containment is below the containment design pressure at 54 psig. In addition, the components covered by I&E Bulletin 79-01B and NUREG-0588, and required for safe shutdown and accident mitigation, maintain their environmental qualification for the resulting temperature and pressure profiles inside the containment as determined by the above analysis.

E. The containment design meets the NRC acceptance criteria contained in IE Bulletin 80-04 related to the issue of containment overpressurization resulting from a main steam line rupture with continued feedwater addition. Considering all possible sources of water, there is no potential for containment overpressurization because the main feedwater system is isolated and auxiliary feedwater system flow restrictors limit flow to the affected steam generator. Also, the auxiliary feedwater system pumps are protected from the effects of runout flow and, therefore, can be expected to carry out their intended function during a main steam line rupture event.

6.2.1.3.12 Reduced Service Water Flow Containment Response

Evaluations of the service water system have indicated that under certain conditions for component failures and loss of offsite power (LOSP), the service water flow to the containment air coolers may be reduced below the original design basis flow. The effect of this reduction in service water supply to the containment air coolers was evaluated using the Westinghouse COMPACT computer program. The Westinghouse COMPACT computer program was developed for the purpose of transient analysis of containment atmospheres in multicompartment containments of water-cooled nuclear power plants. A detailed description of COMPACT is contained in WCAP-11844, "COMPACT-PC: Compartmentalized Analysis of Containment Transients for a PC," Volume I, Revision 2 and Volume II, Revision 0.[18]

The evaluation was performed using the same blowdown data determined in the original analysis. The original model heat sinks were used with the exception that the cable tray data were removed from heat sink 4 and a new heat sink added to include a model of the cables contained in the cable trays. The pertinent information and data for the differences in heat sink models are given in tables 6.2-2 and 6.2-4.

The pressure/temperature results of the evaluation for the LOCA and worst case MSLB are presented in tables 6.2-6 and 6.2-4 and indicate that the original LOCA case for the containment is still limiting.

6.2.1.4 Containment Testing and Inspection

6.2.1.4.1 Preoperation Testing

6.2.1.4.1.1 Integrated Test. Upon completion of the containment and installation of all penetrations, an integrated leakage rate test was performed to verify that the potential leakage rate from the containment is maintained within acceptable values.

The integrated leakage rate tests consist initially of a preoperational test at the peak calculated accident pressure of 48.0 psig, as well as at least one at a lower pressure.

The total allowable leak rate is not more than 0.15 percent by weight of the contained atmosphere per day at 48.0 psig. It has been demonstrated that with good quality control during construction, this is a reasonable requirement. The basis for the performance of the integrated leakage rate test is "Containment Structures for Nuclear Power Plants," Revision 1, November 1, 1972.

The initial leak rate test of the containment and its penetrations were conducted at 100 percent of peak calculated accident pressure and 50 percent of peak calculated accident pressure. Values of containment ambient dry bulb temperature and relative humidity were recorded during the test period for correction of data as required. The test establishes the capability of the containment to contain the pressure for which it was designed, at a leak rate not exceeding that specified.

The test measurement system utilized for the initial leak rate test of the containment is a packaged portable unit designed for use on Units 1 and 2. This portable unit is also used for the periodic leak rate test. The instruments are calibrated prior to each periodic leak rate test. It is anticipated that these instruments will be available for the lifetime of the plant; however, spare instruments of each type are provided.

For further details, see the Technical Specifications.

6.2.1.4.1.2 Local Tests - Prior to initial startup, penetrations and isolation valves were leak tested to verify that the potential leakage is within acceptable values.

The local tests will be performed at a pressure of 48.0 psig and in accordance with the Technical Specifications.

6.2.1.4.2 Postoperational Leakage Tests

Periodic leak rate tests of the containment, penetrations, and isolation valves were conducted to verify their continued leaktight integrity.

The postoperational integrated leakage rate tests were conducted at 50 percent of peak calculated accident pressure while the penetrations and isolation valves postoperational tests were conducted at 48.0 psig.

The test frequency and acceptance are given in the Technical Specifications.

The containment leakage testing program is in conformance with the NRC acceptance criteria contained in Appendix J to 10 CFR 50 and General Design Criteria 52, 53, and 54. Such conformance provides adequate assurance that leakage rates will be periodically checked during service on a timely basis to maintain such leakages within the specified limit of 0.15 weight percent per day. The NRC acceptance criteria regarding airlock leakage tests are in the Technical Specifications.

6.2.1.4.3 Containment Materials Inspection, Testing, and Surveillance

A. Tests to Ensure Liner Integrity

The following tests were performed:

1. Construction tests during the erection of the containment liner.

2. Preoperational tests after the erection of the containment complete with liner, electrical and piping penetrations, equipment hatch, and personnel lock, but before reactor operation.

3. Postoperational leakage tests will be performed at periodic intervals for the life of the plant.

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6.2.2.1.2 Containment Cooling System

The containment cooling system has been designed to remove heat which will be released to the containment atmosphere during any MSLB or LOCA up to and including the double-ended rupture of the largest system pipe. This is accomplished by one of four containment air coolers.

The experimental heat transfer data for the containment air coolers are based on the following documents which have been submitted to the NRC:

A. "Cooling Coil Thermal and Structural Capacity Evaluation for the Palisades Plant of Consumers Power Company," American Air Filter Company, Inc., Report R-1003 (Palisades Docket Number 50-255).

B. Topical Report on "Design and Testing of Fan Cooler/Filter Systems For Nuclear Applications," AAF-TR-7101 February 20, 1972.

Test requirements are certified by the cooler manufacturer.

6.2.2.2 System Design

Design parameters for the heat removal system components are presented in table 6.2-24. Individual system designs are discussed below.

6.2.2.2.1 Containment Spray System

The containment spray system P&ID shown in figure 6.3-3 consists of two pumps, spray ring headers and nozzles, valves, and piping. During the initial (injection) phase of operation, water from the refueling water storage tank is used for containment spraying. During the later (recirculation) phase of operation, water for containment spraying is recirculated from the containment sump.

Detection of leakage from the containment spray system involves the use of sump level instrumentation and floor drain tank level instrumentation. The spray pumps are in rooms that contain a sump. If the alarm in the control room indicates a high sump level, the sump contents would be pumped by duplex sump pumps to the waste holdup tank. Any leak in one of the pump rooms would be detected, and the leak isolated.

A leak in the spray system piping in a corridor, pipe chase, or penetration room is indicated by an increase in the floor drain tank level.

Whenever one of the spray headers is isolated during the process of system leak detection and isolation, flow indication in the operating header assures the operator that sufficient spray flow is being returned to the containment.

The two redundant and independent containment spray trains that comprise the containment spray system, including required auxiliary systems, are designed so that a single active failure during the injection phase or a single active or passive failure during the recirculation phase following a reactor coolant system failure does not result in loss of the protective function.

The containment spray system is designed to accommodate the 1/2 safe shutdown earthquake (1/2 SSE) within applicable codes stress limits and to withstand the safe shutdown earthquake without rupture or loss of function.

Between the edge of the containment mat and the auxiliary building, each containment spray suction line is surrounded by a concentric guard pipe. Three annular seal rings are installed around the guard pipe at each end to minimize any postulated bypass leakage along the outside of the guard pipe. The concentric guard pipes enter the auxiliary building in a trench within the pipe penetration room. Therefore, any postulated airborn leakage that escapes outside the concentric guard pipes is processed by the penetration room filtration system, and any postulated water leakage is processed by the waste disposal system. Within the auxiliary building, each suction line is located within a protective chamber. This chamber is designed to withstand the containment design pressure in addition to the head of water present in the containment sump at the end of the injection phase. From the protective chamber to the first motor-operated isolation valve inside the auxiliary building, each suction line is surrounded by a concentric guard pipe. The first motor-operated isolation valve is surrounded by a watertight closure. This arrangement, which is the same as that used with the suction lines in the low head safety injection system, ensures that, in the unlikely event of leakage from the suction pipe during long term recirculation, the integrity of the recirculation system is not impaired and public safety is not hazarded.

Adequate net positive suction head (NPSH) is available to the containment spray pump suctions at all times during both the injection and the recirculation phases of operation, assuming the most adverse combination of flowrate, sump water

in the various subsystems in which they are to operate. For example, fans are tested in the manufacturer's shop to determine their characteristic curves. System valves are tested in the shop to verify effectiveness of seal, opening and closing periods, and the ability of the valve operator to actuate the valve at the maximum anticipated differential pressure.

Systems acceptance tests consist of deenergized and energized tests which demonstrate the proper mounting of components, proper hook-up of circuits and connections, setting of instrumentation, and operation of interlocks. Equipment and system performance are monitored and rated.

6.2.2.5 Instrumentation Requirements

6.2.2.5.1 Containment Spray System

Instrumentation and associated analog and logic channels employed for initiation of the containment spray system are discussed in section 7.3.

The injection phase of operation of the spray system is actuated either manually from the control room (2/2 logic) or on coincidence of two sets out of four high-high containment pressure signals. This signal starts the containment spray pumps and opens the discharge valves to the spray headers. When the refueling water storage tank is exhausted, the recirculation phase of operation is manually initiated by the operator.

The following describes the instrumentation that is used for monitoring the containment spray system during normal or post-LOCA operation:

<u>Refueling water storage tank level</u> - Two channels of level instrumentation are provided on the refueling water storage tank. Both channels provide remote indication in the control room. One channel additionally provides a high and a low level alarm. The other provides a low and a low-low level alarm.

<u>Containment sump water level</u> - Two level indicators provide control room indication of containment sump level.

<u>Containment pressure</u> - Four channels provide containment pressure indication and both high and high-high containment pressure annunciations in the control room.

<u>Pump discharge pressure</u> - A pressure indicator is located in each containment spray pump discharge line. Readout is local.

<u>Pump discharge flow</u> - A flow indicator is located in each containment spray pump discharge line. Readout is provided in the control room.

6.2.2.5.2 Containment Cooling System

Instrumentation and associated analog and logic channels employed for initiation of the containment cooling system are discussed in section 7.3.

The high-speed winding for each dual-speed fan motor receives power from a normal bus, and the low-speed winding receives power from an emergency bus. Electrical interlocks prevent the connection of both power supplies simultaneously. During normal operation, three of the four units will be operating. On receipt of a safety injection signal, the three operating fans will be electrically switched from high to low speed and the fourth unit will be electrically started in low speed. One unit could then be placed in the standby mode by the operator. In the event of a safety injection signal in conjunction with a loss of offsite power (LOSP/SI), two of the four units will be switched from high to low speed (one unit per train). The other two units will be electrically switched from high speed and blocked from starting in low speed. If diesel capacity is available, the two blocked units may be operated manually at low speed if required. If failure of the electrical interlocks were to allow the fan motor to operate at high speed during an accident, the high-speed winding power supply would be disconnected by the overload trip due to the excessive current required to handle the high density steam air mixture. The fan motor would then be operated at low speed if required.

The following describes the instrumentation that is used to monitor the containment air cooling system during normal or accident conditions.

- <u>Cooling water exit temperature</u> - Cooling water exit temperature is monitored. Remote indication is provided in the control room.

- <u>Cooling water flow</u> - Remote indication of cooling water supply and return flow is provided and low return flow is annunciated in the control room.

- <u>Cooling water radiation level</u> - Cooling water return flow is monitored for radiation level. Indication and high-level annunciation are provided in the control room.

The containment purge system is independent of any other system and includes provisions to supply and exhaust air from the containment. The system is shown in figures 6.2-91 and 6.2-124, sheets 1 and 2. The supply system includes an outside air connection to prefilters, heating coils, a fan, a duct system, and a supply penetration with three butterfly valves in series for tight shutoff. The exhaust system includes an exhaust penetration with three butterfly valves in series, a duct system, a filter bank with prefilters, HEPA and charcoal filters, and an exhaust fan. The exhaust system includes a two-speed fan so that purging can be performed at half- or full-flowrate. The full-flowrate is 48,500 ft^3/min. The quick closing purge isolation valves are capable of closing within 5 s of receipt of a containment actuation isolation signal, phase A, or a high-radiation alarm from the purge discharge radiation monitors. The NRC acceptance criteria regarding charcoal filter surveillance testing were incorporated into the technical specifications by Amendments 46 and 37 to the operating licenses for Units 1 and 2, respectively.

NRC acceptance criteria associated with the design of the containment purge system include the capability to detect high radioactivity conditions in containment prior to opening any of the purge valves. High radioactive conditions inside containment would be detected prior to opening the purge valves by means of the containment atmosphere particulate radioactivity monitor (R-11) and the containment atmosphere gaseous radioactivity monitor (R-12). Should prescribed high radiation levels exist, an audible annunciation would be provided in the control room.

The containment minipurge system is designed to maintain radioactivity levels in the containment consistent with occupancy requirements with continuous system operation. The preaccess filtration system is not required for this occupancy but is available for use in minimizing the need for containment purging.

The operation of the minipurge system is independent of the operation of the containment purge system, although there are common ductwork and common filters. The system is shown in figure 6.2-91. The supply system uses the containment purge supply filter and ducting. A separate minipurge supply fan, located in the penetration room, operates in an 18-in. duct system which bypasses the first purge isolation valve (48 in.) outside the containment. The outside air is discharged through an 18-in. duct connected to the 48-in. purge duct. The minipurge supply system has two isolation valves in series for tight shutoff. The exhaust system has an exhaust connection inside the

containment and an exhaust fan similar to arrangement to the supply system. Exhaust air passes through the containment purge exhaust filter. The exhaust system has two isolation valves in series.

The operation of the exhaust fan is dependent upon the operation of the supply fan. An interlock from the supply fan is installed in the control circuit of the exhaust fan such that the exhaust fan runs automatically when the supply fan is running. However, the exhaust fan may be operated independent of the supply fan, when the main purge supply valves or the minipurge supply fan is out of service, by defeating the interlock between the two fans. During this mode of operation, the plant administrative controls will be in place to ensure that the containment pressure is maintained within the technical specification limits.

The minipurge supply and exhaust fans provide a flowrate of 2500 ft^3/min each. The 8-in. isolation valves are similar in design to the 48-in. containment purge isolation valves, and they are capable of closing in approximately 3.5 s after receipt of a containment ventilation isolation signal or a high radiation signal from the purge discharge radiation monitors.

The plant can operate either in the minipurge mode, with the 48-in. containment isolation valves closed in the supply and exhaust ducts, or in the full-purge mode, with the 48-in. containment isolation valves open and the 8-in. minipurge isolation valves closed. Interlocks prevent operation in both modes simultaneously.

The containment purge system and minipurge system are designed to meet the NRC SER acceptance criteria for containment isolation requirements found in Standard Review Plan (SRP) Section 6.2.4, Revision 1; Branch Technical Position CSB 6-4, Revision 1; SRP Section 3.9.3, Revision 1; and NUREG-0737, item II.E.4.2. In addition, the safety signals to all purge and ventilation isolation valves meet the following criteria:

A. The overriding (i.e., the signal is still present but is blocked in order to perform a function contrary to the signal) of one type of safety actuation signal (e.g., radiation) must not cause the blocking of any other type of safety actuation signal (e.g., pressure) to the isolation valves.

B. Sufficient physical features (e.g., key lock switches) are provided to facilitate adequate administrative controls.

C. The system-level annunciation of the overridden status is provided for every safety system impacted when any override is active.

D. Diverse signals should be provided to initiate isolation of the containment ventilation system. Specifically, containment high radiation, safety injection actuation, and containment high pressure should automatically initiate containment ventilation isolation.

E. The instrumentation and control systems provided to initiate containment ventilation isolation should be designed and qualified as safety-grade equipment.

F. The overriding or resetting (i.e., the signal has come and gone, and the circuit is being cleared in order to return it to the normal condition) of the isolation actuation signal should not cause the automatic reopening of any isolation/purge valve.

Other NRC SER acceptance criteria include:

G. The radioactivity released during normal operation will be within the limits of column 1, Table II, Appendix B to 10 CFR 20.1 - 20.601; the total radioactivity released following an accident results in calculated offsite doses less than the guideline values in 10 CFR 100; and the systems meet the applicable requirements of General Design Criterion 56.

H. The 8-in. minipurge valve Bettis operator seals shall be replaced at least every 5 years.

The temperature and the humidity of the air stream through the preaccess and the purge systems are the same as for the containment atmosphere and are maintained by the containment cooling and ventilating systems.

Fans - The fan used in each half-capacity subsystem of the containment preaccess filtration system is of the vaneaxial type, with a design flowrate of 10,000 sf^3/min each. Fan motors are 25 hp each. Both the exhaust fan and the supply fan in the containment purge system are two-speed centrifugal fans with design flowrates of 50,000/25,000 sf^3/min each.

Fan motors are 60 and 125 hp for the supply and exhaust fans, respectively. Both the minipurge supply and exhaust fans are centrifugal fans with rated flowrates of 5000 sf^3/min each; however, they will operate at 2500 sf^3/min with the 8-in. minipurge valves. Motors for these fans are 15 hp each. They are designed in accordance with the applicable portions of AMCA 99-67, Standards Handbook; AMCA 210-67, Test Codes for Air Handling Devices; and AMCA 211A-65, Certified Rating Program for Air Moving Devices.

Filters - The filters are composite units consisting of prefilter sections, absolute filter section, and impregnated charcoal bed filter section. The prefilter section and the absolute (HEPA) filter section are as described in paragraph 6.2.3.2.2. The charcoal filters are impregnated, activated

carbon beds that are designed to be capable of removing, at relative humidities below 70 percent, all iodines with an efficiency of at least 95 percent. The prefilters and the absolute and charcoal filters are designed for a nominal flowrate of 1000 ft^3/min per 4-ft^2 face area. Carbon weights are approximately 1875 lb and 7400 lb for the preaccess filtration and purge systems, respectively.

The containment purge exhaust filters are seismically qualified. Details of the emergency power sources are discussed in chapter 8.

6.2.3.2.4 Containment Ventilation Systems

The containment ventilation systems flow diagram is shown in figures 6.2-91 and 6.2-124. The containment air cooling fans, control rod drive mechanism cooling fans, and

containment. The recombiner units are located in the containment such that they process a flow of containment air containing hydrogen at a concentration which is generally typical of the average concentration throughout the containment.

To meet the requirements for redundancy and independence, two recombiners are provided, and each recombiner is provided with a separate power panel and control panel and each is powered from a separate bus. Each can be switched to the emergency power source if necessary. There is no interdependency between this system and the other engineered safety features systems.

Containment atmosphere is circulated by natural convection through a recombiner where hydrogen is removed by heating to a temperature sufficient to cause recombination with the containment oxygen.

The recombiner consists of a thermally insulated vertical metal duct with electric resistance metal sheathed heaters provided to heat a continuous flow of containment air (containing hydrogen) up to a temperature which is sufficient to cause a reaction between hydrogen and oxygen.

The recombiner is provided with an outer enclosure to keep out water coming from the containment spray system. The recombiner consists of an inlet preheater section, a heater recombination section, and a mixing chamber.

The unit is manufactured primarily of corrosion resistant, high temperature material for major structural components, except for the base which is carbon steel. The electric hydrogen recombiner uses conventional type electric resistance heaters sheathed with Incoloy-800, which is an excellent corrosion resistant material for this service. These heaters are designed to operate with sheath temperatures equal to those used in certain commercial heaters; however, these recombiner heaters operate at significantly lower power densities than is commercial practice.

Air is drawn into the recombiner by natural convection and passes first through the preheater section. This section consists of a shroud placed around the central heater section to take advantage of heat conduction through the walls to preheat the incoming air. This accomplishes the dual functions of reducing heat losses from the recombiner and of preheating the air.

The warmed air passes through an orifice plate and then enters the electric heater section where it is heated to approximately 1150°F - 1400°F causing recombination to occur. Tests have

verified that the recombination is not a catalytic surface effect associated with the heaters but occurs due to the increased temperature of the process gases. Since the phenomenon is not a catalytic effect, saturating of the unit by fission products will not occur. The heater section consists of five assemblies of electric heaters stacked vertically. Each assembly contains individual heating elememts. Operation of the unit is virtually unaffected should a few individual heating elements fail to function properly. Table 6.2-21 gives the recombiner design parameters.

The recombiner, power supply panel, and control panel are shown schematically in figure 6.2-102. The power panel for the recombiner is located in the auxiliary building, and contains an isolation transformer plus an SCR controller to regulate power into the recombiner. This equipment is not exposed to the post loss-of-coolant accident environment.

The control panel is located in the control room. To control the recombination process, the correct power input which will bring the recombiner above the threshold temperature for recombination will be set on the controller. The controller setting will be set at approximately 48.9 kW at the control panel and this power setting will cover variations in containment temperature, pressure, and hydrogen conconcentration in the post loss-of-coolant accident environment. For an equipment test and periodic checkout, a thermocouple readout instrument is also provided in the control panel for monitoring temperatures in the recombiner.

Results of testing a prototype of the electric hydrogen recombiner are given in <u>WCAP-7820</u>, Supplement 3, "Electrical Hydrogen Recombiner for Water Reactor Containments," March 1974.

6.2.5.2.2 Postaccident Venting System

The postaccident containment venting system as shown on figures 6.2-103 and 104 consists of a supply line through which hydrogen free air can be admitted to the containment, and an exhaust line through which hydrogen bearing gases may be vented from the containment. The gases are filtered through HEPA and charcoal filters to limit discharge of particulates and iodine. Piping and valving in the exhaust line are Safety Class 2a starting inside the containment, proceeding up to and including the venting filtration unit outside the containment. Equipment and piping beyond the filter unit are Safety Class 2b. Design parameters are given in table 6.2-34.

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TABLE 6.2-2 (SHEET 1 OF 9)

HEAT SINK GEOMETRIC DATA

Heat Sink 1 - Containment Cylinder and Dome

Containment cylinder	53014 ft^2
Containment dome	21253 ft^2
	74267 ft^2

Exposure

1. Containment Atmosphere
2. Outside Atmosphere

Material	Thickness (in.)
Paint	2.0×10^{-2}
Primer[a]	3.0×10^{-3}
Carbon Steel	2.5×10^{-1}
Air Gap	2.04×10^{-3}
Concrete Region 1	3.0
2	6.0
3	6.0
4	30.0

Heat Sink 2 - Unlined Concrete

Steam generator compartment walls	32598 ft^2
Pressurizer compartment	2899 ft^2
Refueling canal (outside)	2038 ft^2
Refueling canal supports	2598 ft^2
Slabs at el 129 ft 0 in.	7070 ft^2
Slabs at el 155 ft 0 in.	9150 ft^2
Steam generator buttresses	3395 ft^2
Missile shield	576 ft^2
	60324 ft^2

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Paint	18.0×10^{-3}
Surfacer[a]	1.25×10^{-1}
Concrete Region 1	3.0
2	3.0
3	3.0

TABLE 6.2-2 (SHEET 2 OF 9)

Heat Sink 3 - Reactor Support

Concrete - 2183 ft^2

Exposure

1. Containment Atmosphere
2. A 150°F source to account for the higher reactor cavity operating temperature.

Material	Thickness (in.)
Paint	18.0×10^{-3}
Surfacer[(a)]	1.25×10^{-1}
Concrete	3.0

Heat Sink 4 - Galvanized Steel

	Unit 2	Unit 1
Grating	21486 ft^2	21486 ft^2
Cable trays[(b)]	22164 ft^2	22164 ft^2
Ventilation ductwork	19282 ft^2	18707 ft^2
	62932 ft^2	62357 ft^2

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Zinc	3.35×10^{-3}
Carbon Steel	6.56×10^{-2}

TABLE 6.2-2 (SHEET 3 OF 9)

Heat Sink 5 - Painted Steel Less than 0.12-in. Thickness

Structural steel	3576 ft^2
Manipulator crane	635 ft^2
Polar crane	445 ft^2
Air coolers	5723 ft^2
Misc. (brackets, supports, and handrails)	293 ft^2
	10672 ft^2

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Paint	2.0×10^{-2}
Primer[a]	3.0×10^{-2}
Steel	7.64×10^{-2}

Heat Sink 6 - Painted Steel 0.12- to 0.16-in. Thickness

Polar crane	24978 ft^2
Air coolers	7785 ft^2
Duct reinforcement	4755 ft^2
Structural steel	3570 ft^2
Misc. (brackets, supports, and handrails)	1904 ft^2
	42992 ft^2

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Paint	2.0×10^{-2}
Primer[a]	3.0×10^{-3}
Steel	1.32×10^{-1}

TABLE 6.2-2 (SHEET 4 OF 9)

Heat Sink 7 - Painted Steel 0.16- to 0.24-in. Thickness

Polar crane	4729 ft^2
Structural steel	6566 ft^2
Misc. (including air coolers, manipulator crane, penetrations, etc.)	1414 ft^2
	12709 ft^2

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Paint	2.0×10^{-2}
Primer[a]	3.0×10^{-3}
Steel	1.91×10^{-1}

Heat Sink 8 - Painted Steel 0.24- to 0.30-in. Thickness

	Unit 2	Unit 1
Polar crane	4140 ft^2	4140 ft^2
Structural steel	1607 ft^2	1107 ft^2
Cable tray supports	8593 ft^2	8593 ft^2
Misc. (including accumulator, manipulator crane, feedwater restraints, penetrations)	2925 ft^2	2925 ft^2
	17265 ft^2	16765 ft^2

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Paint	2.0×10^{-2}
Primer[a]	3.0×10^{-3}
Steel	2.55×10^{-1}

TABLE 6.2-2 (SHEET 5 OF 9)

Heat Sink 9 - Painted Steel 0.30- to 0.40-in. Thickness

Polar crane	4148 ft^2
Structural steel	2922 ft^2
Misc. (including tank supports, heat exchanger, manipulator crane)	925 ft^2
	7995 ft^2

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Paint	2.0×10^{-2}
Primer[a]	3.0×10^{-3}
Steel	3.38×10^{-1}

Heat Sink 10 - Painted Steel 0.40- to 0.50-in. Thickness

Polar crane	8078 ft^2
Structural steel	1219 ft^2
Main steam restraints	1140 ft^2
Misc. (including penetrations, manipulator crane tank supports, feedwater restraints)	3536 ft^2
	13973 ft^2

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Paint	2.0×10^{-2}
Primer[a]	3.0×10^{-3}
Steel	4.92×10^{-1}

TABLE 6.2-2 (SHEET 6 OF 9)

Heat Sink 11 - Painted Steel 0.50- to 0.625-in. Thickness

Polar crane	3519 ft^2
Structural steel	1624 ft^2
Misc. (including accumulator, pressure relief tank, and drain tank supports)	6258 ft^2
	11401 ft^2

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Paint	2.0×10^{-2}
Primer[a]	3.0×10^{-3}
Steel	5.76×10^{-1}

Heat Sink 12 - Painted Steel 0.625- to 0.75-in. Thickness

Polar crane	731 ft^2
Structural steel	1460 ft^2
Misc. (including pressure relief tank supports)	119 ft^2
	2310 ft^2

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Paint	2.0×10^{-2}
Primer[a]	3.0×10^{-3}
Steel	7.24×10^{-1}

TABLE 6.2-2 (SHEET 7 OF 9)

Heat Sink 13 - Painted Steel 0.75- to 1.0-in. Thickness

Polar crane	3758 ft^2
Misc. (including structural steel, feedwater restraints, air coolers, reactor coolant pumps, manipulator crane)	1976 ft^2
	5734 ft^2

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Paint	2.0×10^{-2}
Primer[a]	3.0×10^{-3}
Steel	9.35×10^{-1}

Heat Sink 14 - Painted Steel 1.0- to 1.5-in. Thickness

Polar crane	1333 ft^2
Main steam restraints	887 ft^2
Misc. (including air locks, manipulator crane, accumulator, penetrations)	1526 ft^2
	3746 ft^2

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Paint	2.0×10^{-2}
Primer[a]	3.0×10^{-3}
Steel	1.43

TABLE 6.2-2 (SHEET 8 OF 9)

Heat Sink 15 - Painted Steel Greater than 1.5-in. Thickness

Polar crane	3438 ft^2
Accumulator	1370 ft^2
Misc. (including equipment hatch, manipulator crane, air locks)	727 ft^2
	5535 ft^2

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Paint	2.0×10^{-2}
Primer[a]	3.0×10^{-3}
Steel	2.85

Heat Sink 16 - Stainless Steel

Ventilation ducts	8507 ft^2
Refueling Canal liner	7894 ft^2
Misc. (including drain tanks, pressure relief tanks, cable trays, heat exchangers, electric boxes)	1762 ft^2
	18163 ft^2

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Stainless steel	1.68×10^{-1}

Heat Sink 17 - Galvanized Steel

Cable trays [b]	22164 ft^2

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a. When Amercoat 90 is used as the primer, the average primer thickness will be 5.0 mils. However, the total thickness of the primer plus finish coat will not exceed the total thickness of finish plus primer (surfacer) listed in the table.

b. Cable trays were modeled as galvanized steel for the COPATTA analyses without internal cables. Internal cables were modeled for the revised analysis described in paragraph 6.2.1.3.12.

TABLE 6.2-2 (SHEET 9 OF 9)

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Zinc	3.35×10^{-3}
Carbon Steel	5.0×10^{-2}
Cable Material	6.24×10^{-1}

TABLE 6.2-4

HEAT SINK THERMODYNAMIC DATA

MATERIAL PROPERTIES

Material	Density (lbm/ft^3)	Thermal Conductivity Btu/h-ft-°F)	Heat Capacity (Btu/lbm-°F)	Emissivity
Paint (Ameron 66)	162.3	0.50	0.289	0.9
Paint (Ameron 90)	161.0	0.38	0.3099	0.9
Primer (Dimetcote 6)	196.6	0.63	0.1099	0.9
Carbon steel	489.0	29.6	0.1096	0.28
Concrete	144.0	1.0	0.2292	0.63
Sur[illegible] (Ameron 110 AA)	121.7	0.39	0.2268	0.9
Zinc	446.0	62.2	0.0942	0.25
Stainless steel	488.0	8.6	0.1232	0.9
Air	0.069	0.017	0.2095	0.0
Cable Material	92.39	0.065	0.4	0.95

HEAT TRANSFER COEFFICIENTS

Surface	Value
Sink surfaces exposed to containment atmosphere	Modified Tagami
Sump liquid to containment atmosphere	0
Containment sump and floor to sump liquid	0
Sink surfaces exposed to outside atmosphere	2.0 Btu/h-ft^2-°F

TABLE 6.2-5 (SHEET 1 OF 2)

ENGINEERED SAFETY FEATURES PERFORMANCE
FOR CONTAINMENT PRESSURE TRANSIENT ANALYSIS

		Values Used for Containment Analysis	
System	Operation	Maximum ESF	Minimum ESF
Containment spray			
Water sources	Borated water from RWST or sump		
Initiation	Initiated by SIS		
Number of lines and headers		2	1
Number of pumps		2	1
Flowrate, gal/min per pump		2225	2175
Containment air coolers			
Initiation	Initiated by SIS		
Number of units		4	1
Flowrate (air side), ft^3/min per unit		40000	40000
Total design heat removal at containment design temperature, (Btu/h) per unit		80×10^6	80×10^6[a]
Service water temperature (°F)		95	95
RHR/Low pressure safety injection heat exchangers			
Type	Vertical shell U-tube		
Cooling water supply	Component cooling water		
Number of units		2	1
Heat transfer area, ft^2 per unit		4070	4070
Overall heat transfer coefficient, Btu/h-ft^2-°F		330	330
Flowrate:	Injection	3000	3000
Sump water side, gal/min per unit	Recirculation	3750	3750
Component cooling water side, gal/min per unit		5600	5600
Return water point		Primary loop	Primary loop

TABLE 6.2-5 (SHEET 2 OF 2)

System	Operation	Values Used for Containment Analysis Maximum ESF	Minimum ESF
Passive safety injection system			
Capacity, gal each accumulator	600		
Number of accumulators		3	3
Pressure setpoint, psig		600	600
Active safety injection system			
Initiation	Initiated by SIS		
High pressure safety injection:			
Number of lines		3	3
Number of pumps		2	1
Flowrate, gal/min per pump		511	511
Low pressure safety injection:			
Number of lines		3	3
Number of pumps		2	1
Flowrate, gal/min per pump	Injection	3000	3000
	Recirculation	3750	3750

a. Value for 600-gal/min service water flow for paragraph 6.2.1.3.12 analysis is 31.2×10^6 at 275 °F.

TABLE 6.2-6

CONTAINMENT PRESSURE ANALYSIS RESULTS FOR THE SPECTRUM OF RCS BREAK SIZES[a]

	DEPSG MIN ESF 8.25 ft^2	DEPSG MAX ESF 8.25 ft^2	0.6 DEPSG MAX ESF 4.95 ft^2	PSS MAX ESF 3 ft^2	DECLG MAX ESF 8.25 ft^2	DEHLG MAX ESF 9.17 ft^2
Peak pressure (psig)	42.3	39.8	40.1	40.9	37.6	40.2
Time of peak pressure (s)	272.0	133.1	191.9	194.3	22.3	61.1
Peak temperature (°F)	267.5	263.7	264.3	265.4	260.3	264.3
Time of peak temperature (s)	272	133.1	191.9	194.3	22.3	61.1
Blowdown energy release at time of peak pressure (10^6 Btu)	309.4	284.1	304.9	306.5	252.9	307.0

a. See table 6.2-41 for MSLB results.

TABLE 6.2-19

CONTAINMENT RESULTS FOR THE DESIGN BASIS LOCA

Parameter	Prior to LOCA	At Peak Pressure
Time (s)		272
Pressures		
Steam (psia)	.85	40.2
Air (psia)[c]	13.85	16.8
Total psia	14.70	57.0
Total gauge (psig)		42.3
Temperatures		
Steam and air (°F)	120	268
Water in sump (°F)		257
Mass releases (lbm)		
Blowdown		5.16×10^5
Injection spillage		2.46×10^4
Spray injected		6.63×10^4
Total		6.07×10^5
Containment masses		
Air (lbm)[c]	1.28×10^5	1.28×10^5
Steam (lbm)	$.08 \times 10^5$	2.65×10^5
Water in sump (lbm)		3.46×10^5
Total (lbm)[a]	1.36×10^5	7.39×10^5
Heat transfer coefficient (Btu/h-ft^2-°F)[b]	2.0	80.1

a. Containment total mass change = total mass release.

b. Between containment atmosphere and structure.

c. Includes N_2 from accumulators.

TABLE 6.2-21

LOCA CHRONOLOGY OF EVENTS

Time (s)	Event
0.0	Pipe ruptures (8.25 ft2 cold-leg), reactor depressurization begins.
(a)	Accumulator injection tanks begin injection.
19.7	HPSI and LPSI pumps begin injection. Blowdown ends as reactor vessel reaches pressure equilibrium with containment.
31.3	Air coolers begin operation.
55.0	Containment sprays begin operation.
272.0	Containment reaches peak pressure.
1170	Sump reaches maximum temperature.
3.714×10^3	Safety injection water recirculation from the sump begins as RWST reaches low level.
6.473×10^3	Containment spray water recirculation from the sump begins as RWST reaches low-low level.
9.0×10^3	Containment reaches maximum energy value.
10^7	Containment reaches atmospheric pressure (estimate).

a. When primary loop pressure drops below 600 lb/in.^{2}

TABLE 6.2-24

COMPONENT DESIGN PARAMETERS FOR CONTAINMENT SPRAY SYSTEM AND CONTAINMENT COOLING SYSTEM

Containment Spray Pumps	
Type	Horizontal Centrifugal
Number	2
Pressure (psig)	300
Temperature (°F)	250
Flowrate (each) (gal/min)	2600
Heat (ft)	450
Containment Coolers	
Number	4
Pressure (psig)	200
Temperature (°F)	300
Water inlet temperature (°F)	95
Flowrate (normal) (gal/min)	800
Heat removal rate (normal) (Btu/h)	2.36×10^6
Flowrate (post-LOCA) (gal/min)	2000 (600 for containment analysis)
Heat removal rate (post-LOCA) (Btu/h)	112.3×10^6 (31.2×10^6 for containment analysis)
Containment Cooler Fans	
Type	Vaneaxial
Number	4
Flowrate (high speed) (sf^3/min)	80,000
Static head (high speed) (in. wg)	4.75
Horsepower (high speed (hp)	80
Flowrate (low speed (sf^3/min)	40,000
Static head (low speed) (in. wg)	7.90
Horsepower (low speed (hp)	105
Refueling Water Storage Tank	
Quantity	1
Volume (gal)	500,000
Design pressure (psig)	atmosphere
Design temperature (°F)	ambient
Material	stainless steel
Piping	
Pressure (psig)	150
Temperature (°F)	500
Valves	
Pressure (psig)	150
Temperature (°F)	500

TABLE 6.2-31 (SHEET 1 OF 7)

CONTAINMENT ISOLATION VALVE INFORMATION[a]

Item No.	Service (No. of Penetrations)	System	Penetration No.	Penetration Type	Penetration Line Size (in.)	Valve Arrangement	Flow Direction	Location Relative to Containment	Valve Type	Actuator	Signal	Normal Valve Position	Valve Pos. with Power Fail.	Pos. Ind.	Post LOCA Position	Valve Closure Time (s)
1	Accumulator Test Line (1)	SIS	29	II	3/4	24	OUT	Inside	Globe	Air	T	Closed	Closed	Yes	Closed	≤10
					3/4			Outside	Globe	Air	T	Closed	Closed	Yes	Closed	≤10
2	Refueling Cav. Supply (1)	FHS	95	II	2	35	IN	Inside	Check	--------	--------	--------	--------	---	--------	---
					2			Outside	Diaphrem	--------	--------	Locked Closed	As Is	---	Closed	---
3	Nitro. Supply to Accumulators (1)	SIS	63	II	1	23	IN	Inside	Check	--------	--------	--------	--------	---	--------	---
								Outside	Globe	Air	T	Closed	Closed	Yes	Closed	≤10
4	Nitro. Supply-Press. Relief Tank (1)	RCS	64a	II	1	38	IN	Inside	Diaphragm	Air	T	Open	Closed	Yes	Closed	≤10
					1			Outside	Diaphragm	Air	T	Open	Closed	Yes	Closed	≤10
5	Press. Relief Tank Makeup (1)	RCS	30	II	3	27	IN	Inside	Check	--------	--------	--------	--------	---	--------	---
					3			Outside	Diaphragm	Air	T	Closed	Closed	Yes	Closed	≤10
6	Charging Pump Suct. Relief Valve Discharge to Pressurizer Relief Tank(1)	CVCS	59	II	2	34	IN	Inside	Check	--------	--------	--------	--------	---	--------	---
								Outside	See Note 1							
7	Reactor Coolant Drain Tank Drain(1)	WPS	31	II	3	1	OUT	Inside	Globe	Air	T	Open	Closed	Yes	Closed	≤10
					3			Inside	Diaphragm	Manual	--------	Locked Closed	Closed	No	Closed	---
					3			Outside	Diaphragm	Air	T	Open	Closed	Yes	Closed	≤10
8	Containment Differential Pressure Instrument (1)	H&V	70	II	1	16	---	Inside	Globe	Elec.Mtr	T	Open	As Is	Yes	Closed	≤15
					1			Outside	Globe	Elec.Mtr	T	Open	As Is	Yes	Closed	≤15
9	Residual Heat Removal A & B Loop Pump Suct(2)	RHRS	16,18	I	12	5	OUT	Inside	Gate	Elec.Mtr	Remote Manual	Closed	As Is	Yes	Closed	≤120
								Outside	See Note 2							
10	Residual Heat Removal A & B Pump Disc.(2)	RHRS SIS	15,17	I	10	30	IN	Inside	Check	--------	--------	--------	--------	---	--------	
					10			Outside	Gate	Elec.Mtr	Remote Manual	Open	As Is	Yes	Open	N/A
11	Normal Letdown Line (1)	CVCS	23	I	2	33	OUT	Inside	Globe	Air	T	Open	Closed	Yes	Closed	≤10
					3			Outside	Globe	Air	T	Open	Closed	Yes	Closed	≤10
12	Excess Letdown and RCP Seal Water Return (1)	CVCS	28	I	3	6	OUT	Inside	Gate	Elec.Mtr	T	Open	As Is	Yes	Closed	≤10
					3			Outside	Gate	Elec.Mtr	T	Open	As Is	Yes	Closed	≤10
					3/4			Inside	Check	--------	---	--------	--------	---	--------	---

* Reference table 6.2-32, sheet 2; table 6.2-38; 6.2-39.

TABLE 6.2-31 (SHEET 5 OF 7)[a]

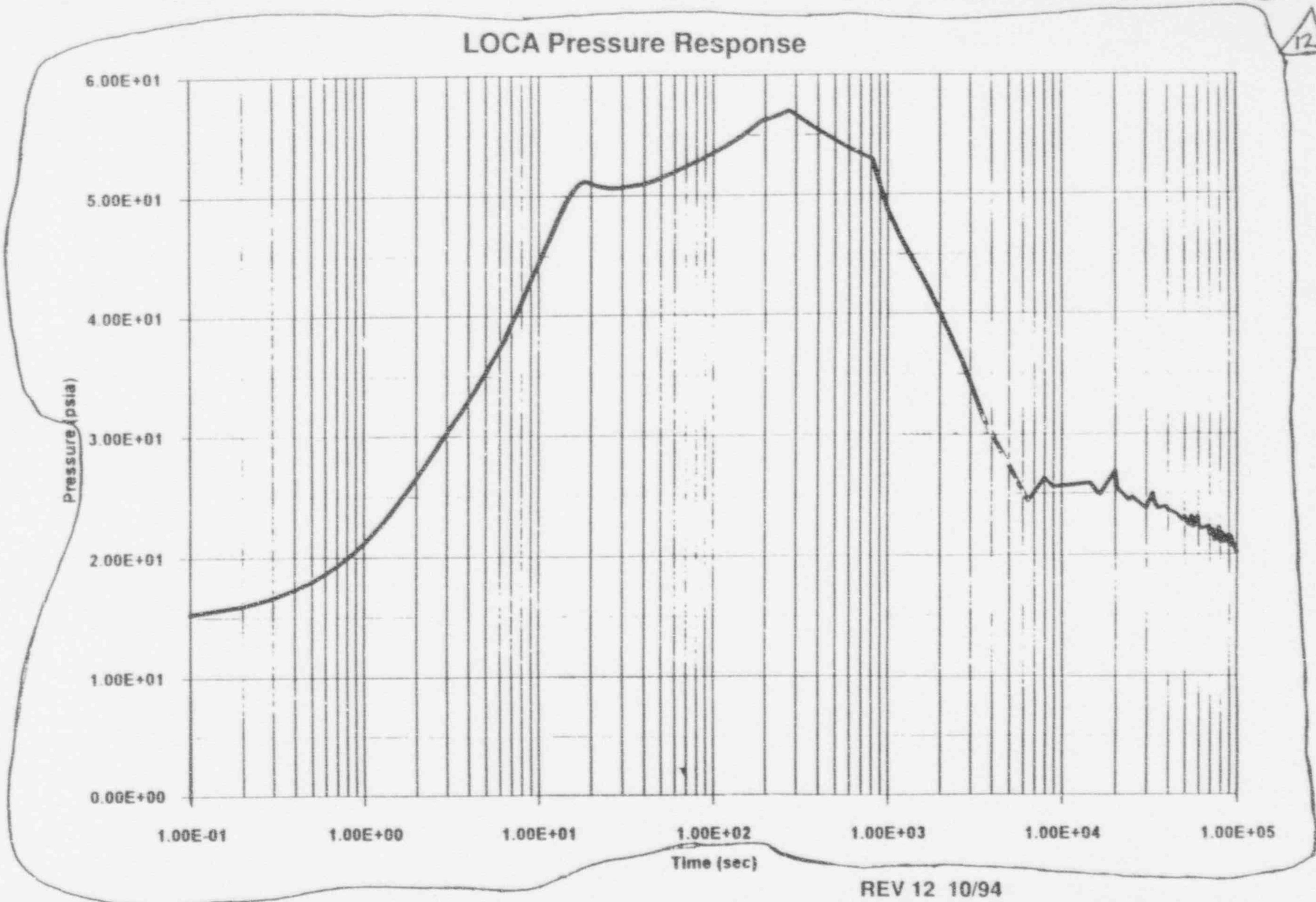
Item No.	Service (No. of Penetrations)	System	Penetration No.	Penetration Type	Penetration Line Size (in.)	Valve Arrangement	Flow Direction	Location Relative to Containment	Valve Type	Actuator	Signal	Normal Valve Position	Valve Pos. with Power Fail.	Pos. Ind.	Post LOCA Position	Valve Closure Time (s)
43	High-head injection to RCS cold legs(1)	SIS	19	I	2 3	2	In	Inside Outside	Check Gate	--- Elec.mtr	--- S	--- Closed	--- As is	--- Yes	--- Open	--- ≤10
44	RHR injection to RCS hot legs(1)	SIS	101	I	6 10	17	In	Inside Outside	Check Gate	--- Elec.mtr	--- Remote manual	--- Closed	--- As is	--- Yes	--- Open	--- NA
45	Service water to reactor coolant coolers(1)	SWS	60	II	6 6	30	In	Inside Outside	Check Gate	--- Elec.mtr	--- S	--- Open	--- As is	--- Yes	--- Closed	--- <15
46	Service water from reactor coolant pump motor air coolers(1)	SWS	32	II	6 6	29	Out	Inside Outside	Gate Gate	Elec.mtr Elec.mtr	S S	Open Open	As is As is	Yes Yes	Closed Closed	<15 <15
47	Containment sump pump sample recirculation line(1)	WPS	33	II	2 2	23	In	Inside Outside	Check Globe	--- Air	--- T	--- Open	--- Closed	--- Yes	--- Closed	--- <10
48	Post LOCA containment sample out (2)	SS	61A,67	II	3/4 3/4	37	Out	Inside Outside	Globe Globe	Elec.mtr Elec.mtr	Remote manual Remote manual	Locked closed Locked closed	As is As is	Yes Yes	Closed Closed	
49	Post LOCA containment sample in (2)	SS	61B,66	II	3/4 3/4	20	In	Inside Outside	Globe Globe	Elec.mtr Elec.mtr	Remote manual Remote manual	Locked closed Locked closed	As is As is	Yes Yes	Closed Closed	
50	Post LOCA containment venting (1)	SS	103	II	6 6	16	Out	Inside Outside	Globe Globe	Elec.mtr Elec.mtr	Remote manual Remote manual	Locked closed Locked closed	As is As is	Yes Yes	Closed Closed	
51	Demineralized water (1)	DWS	82	II	3 3	23	In	Inside Outside	Check Globe	--- Air	--- T	--- Closed	--- Closed	--- Yes	--- Closed	--- <10
52	Backup air supply to pressurizer PORV (1)	IA	97B	II	1/2 3/4	23	In	Inside Outside	Check Globe	--- Air	--- P	--- Closed	--- Closed	--- Yes	--- Closed	--- <10
53	Spare (Used with refueling module during refueling outages) (2)	---	90,92	II	10	40	NA	Inside Outside	Blind Flanges	---	---	---	---	---	---	---

a. Reference table 6.2-32, sheet 2; table 6.2-38; 6.2-39.

TABLE 6.2-41

CONTAINMENT PRESSURE/TEMPERATURE FOR
600 gal/min SERVICE WATER FLOW,
0.003 FOULING FACTOR

Case	Peak Pressure (psia)	Time (s)	Peak Temp. (°F)	Time (s)
MSLB CASE 1	55.9	1800.1	287.3	180.6
MSLB CASE 2	54.6	1800.0	275.0	257.5
MSLB CASE 3	55.1	1799.9	272.5	287.2
MSLB CASE 4	59.6	1799.9	271.3	300.0
MSLB CASE 5	56.8	1799.9	276.4	204.9
MSLB CASE 6	56.5	1800.0	266.7	1800.0
MSLB CASE 7	57.2	1799.9	267.8	1799.9
MSLB CASE 8	59.9	1799.9	271.5	1799.9
MSLB CASE 9	58.9	1800.0	270.1	1800.0
MSLB CASE 10	58.5	1800.0	269.6	1800.0
MSLB CASE 11	62.1	1799.9	274.6	1799.9
MSLB CASE 12	61.02	1800.0	276.8	99.9
MSLB CASE 13	60.1	1800.0	271.9	1800.0
MSLB CASE 14	56.4	2778.3	266.6	2778.3
MSLB CASE 15	46.9	4710.7	251.0	4710.7
MSLB CASE 16	58.4	2900.0	269.5	2900.0



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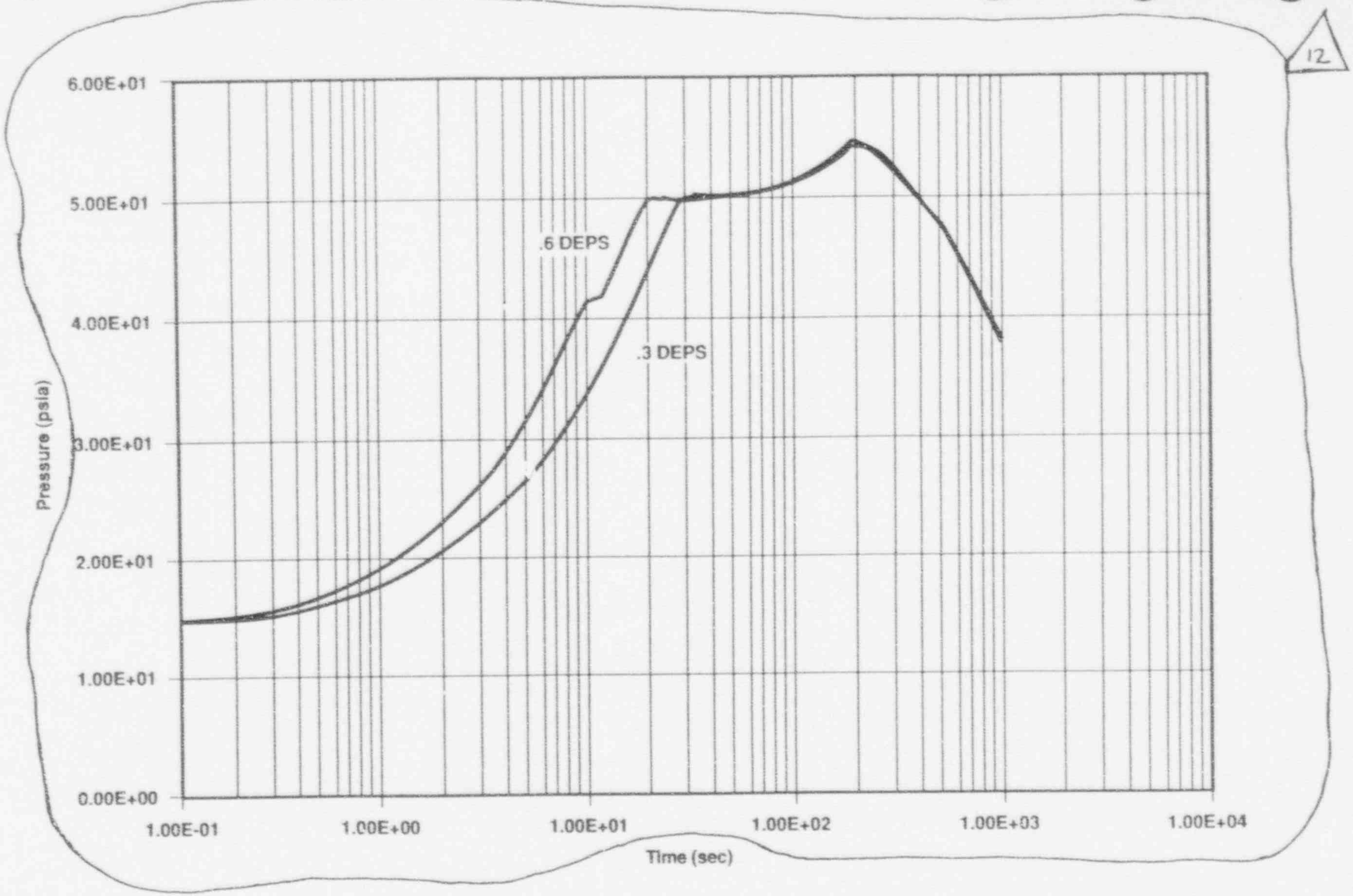
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	DEPSGB MINIMUM ESF 1 AC PRESSURE vs. TIME
	FIGURE 6.2-1



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JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	MAXIMUM SG 0.6 DEPSB AND 3 ft^2 PSS PRESSURE VS. TIME
	FIGURE 6.2-2

12

THIS FIGURE HAS BEEN DELETED.

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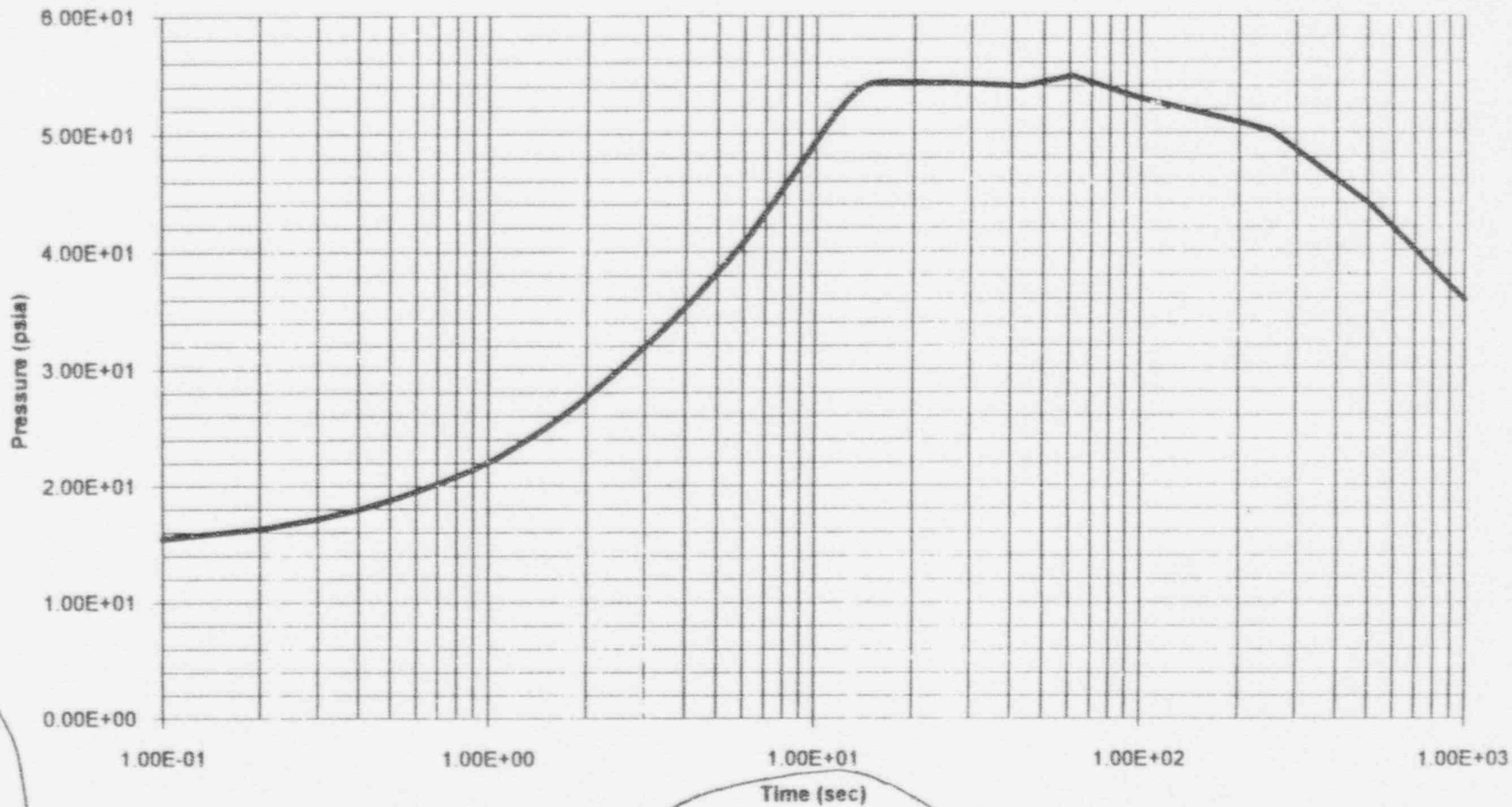
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	DEPS, MINIMUM ESF, DBA SHORT TERM PRESSURE VS. TIME
	FIGURE 6.2-3



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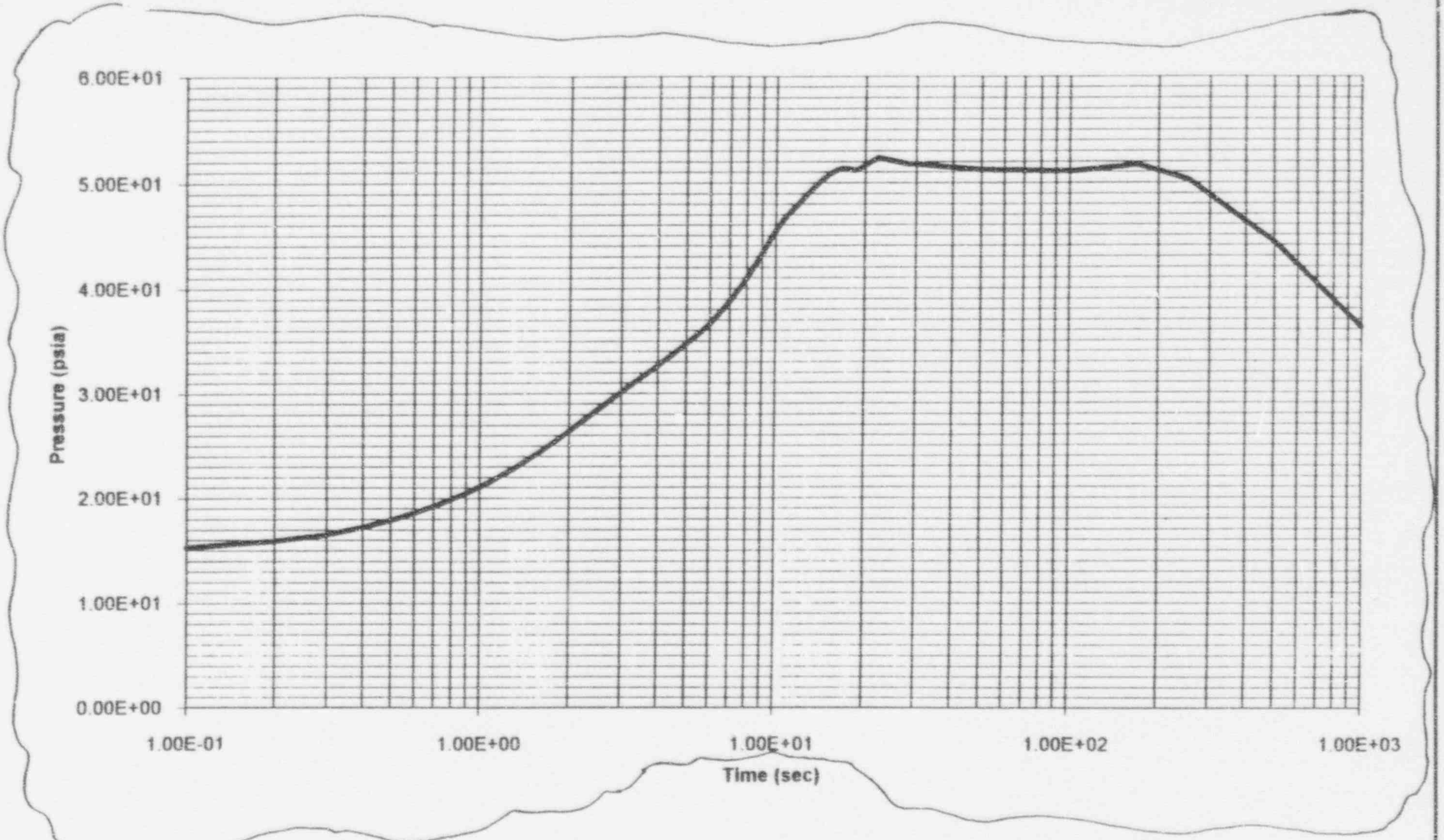
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	MAXIMUM SG DEHLG PRESSURE vs. TIME
	FIGURE 6.2-4



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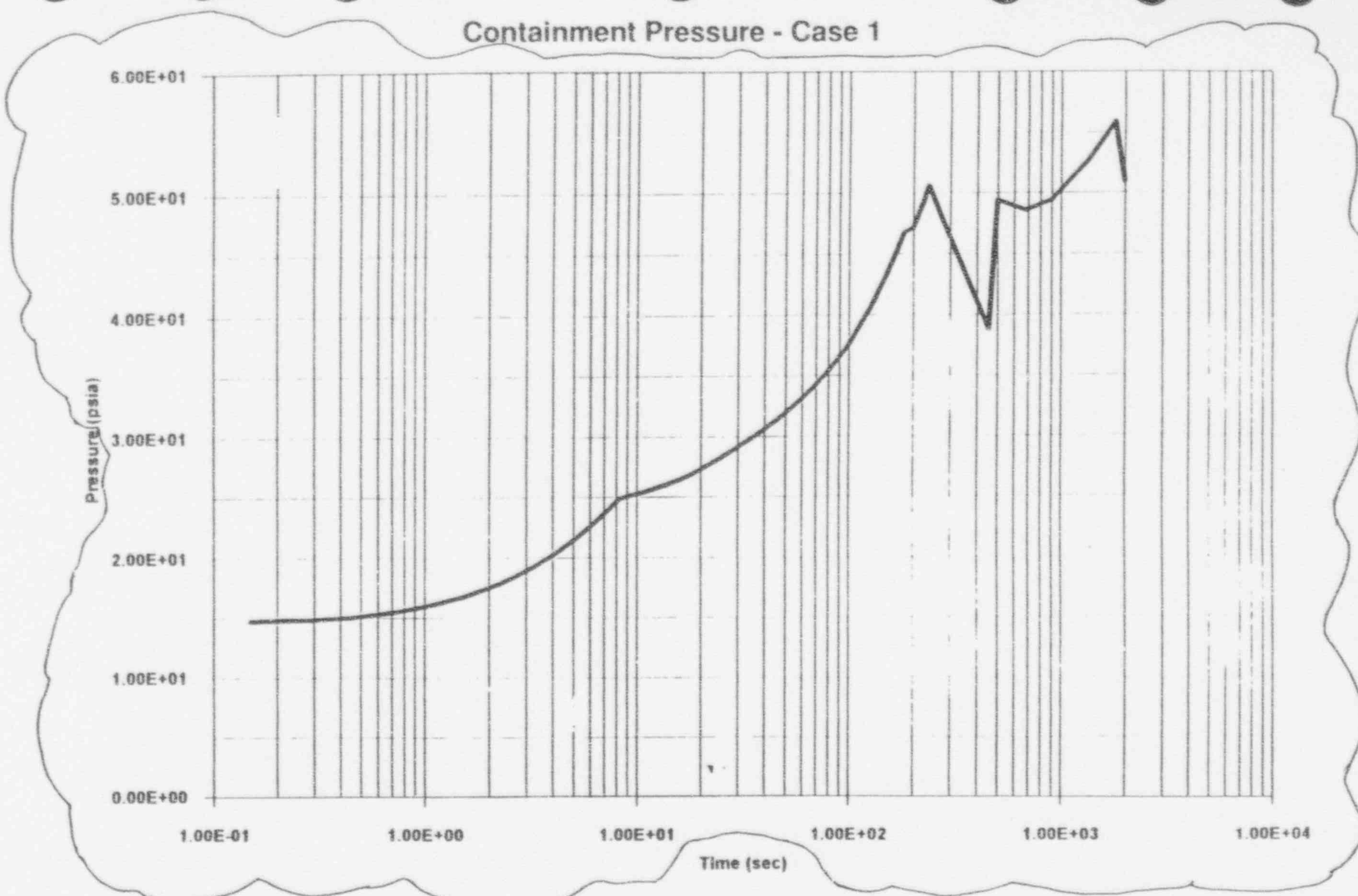
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	DECLG MAXIMUM ESF PRESSURE VS. TIME
	FIGURE 6.2-5



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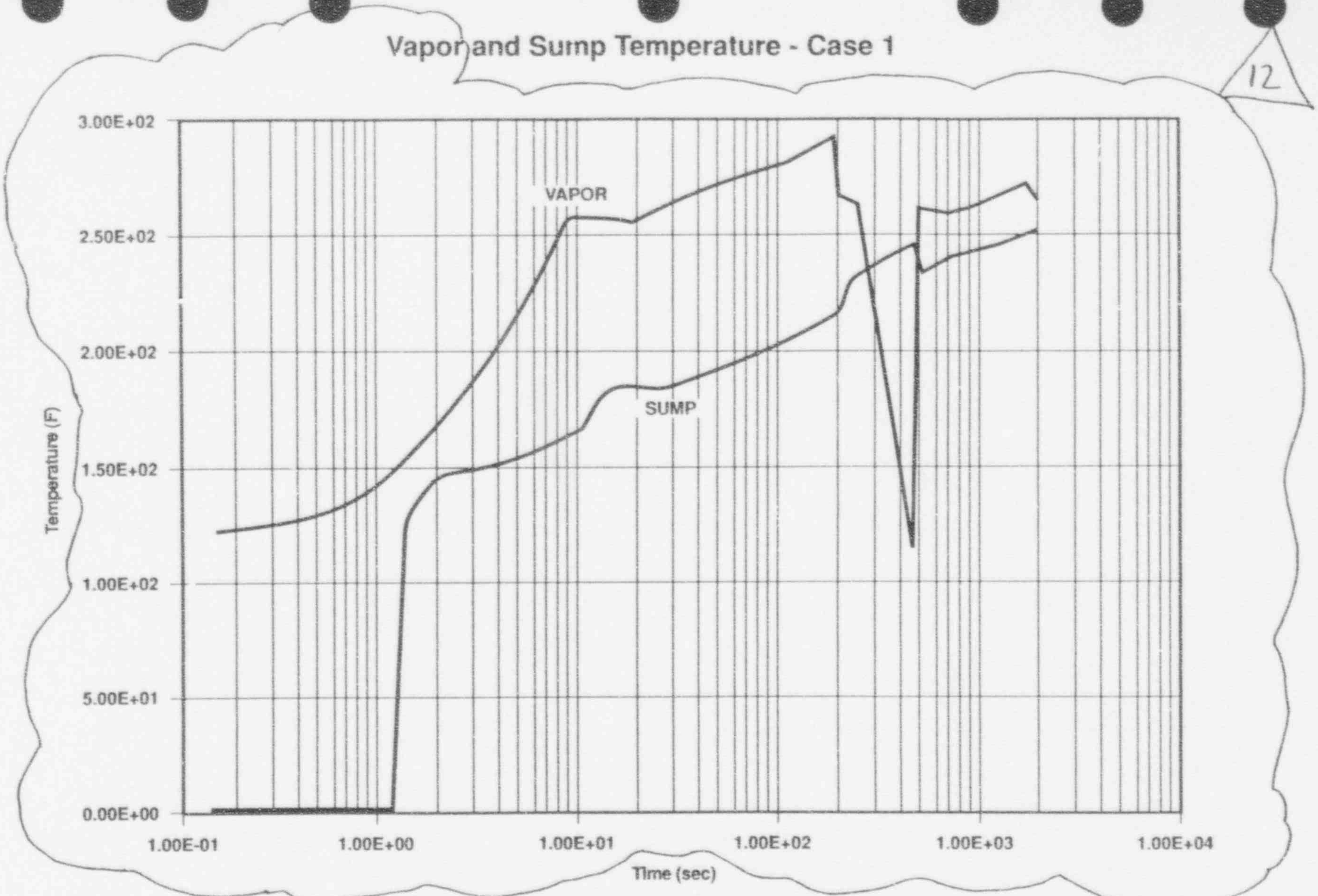
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRESSURE VERSUS TIME STEAM LINE FULL D.E. BREAK 102% POWER
	FIGURE 6.2-6



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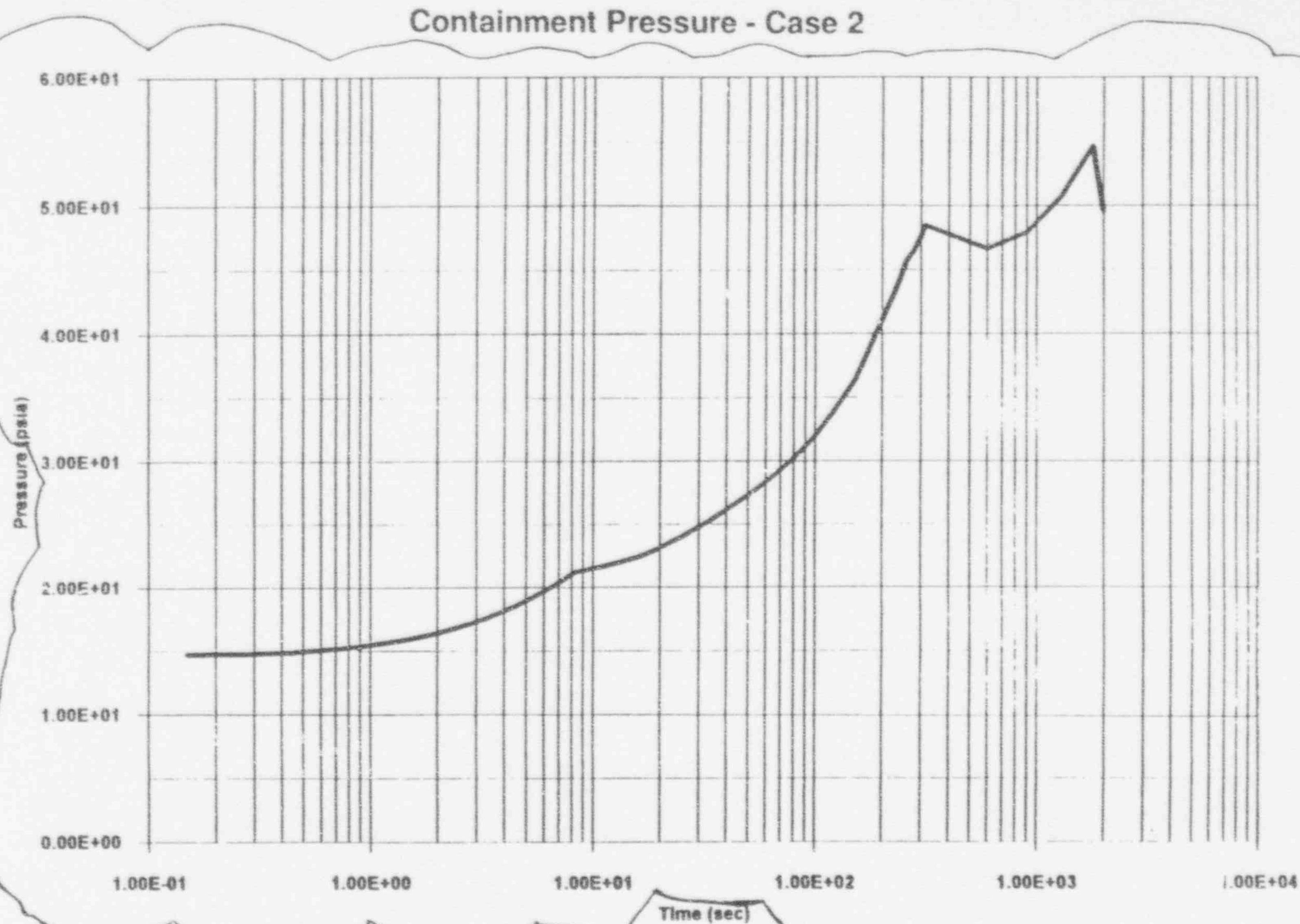
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	TEMPERATURE VERSUS TIME STEAM LINE FULL D.E. BREAK 102% POWER
	FIGURE 6.2-7

12



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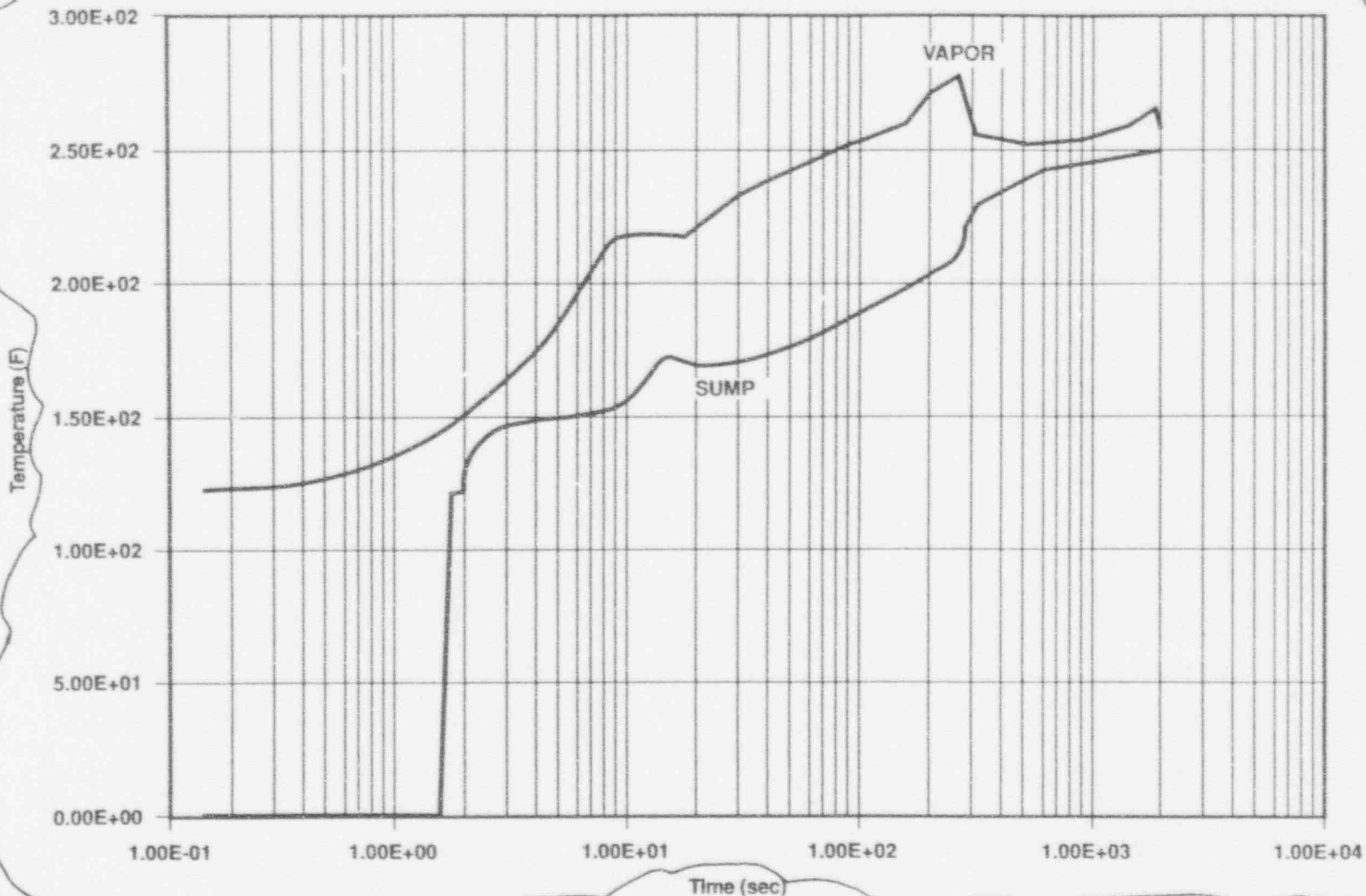
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRESSURE VERSUS TIME STEAM LINE 0.7 ft^2 D.E. BREAK 102% POWER
	FIGURE 6.2-8

Vapor and Sump Temperature - Case 2



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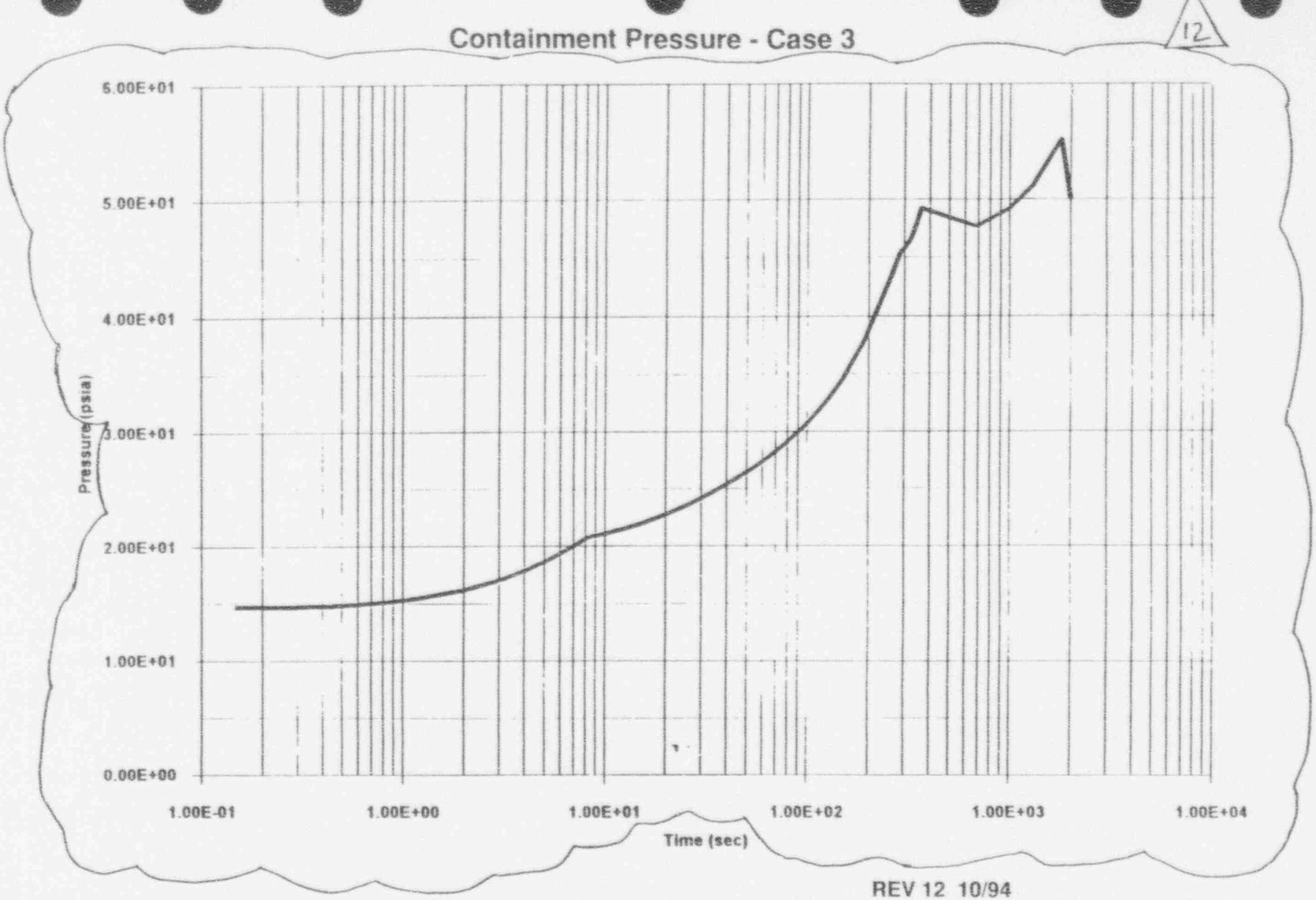
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	TEMPERATURE VERSUS TIME STEAM LINE 0.7 ft^2 D.E. BREAK 102% POWER
	FIGURE 6.2-9



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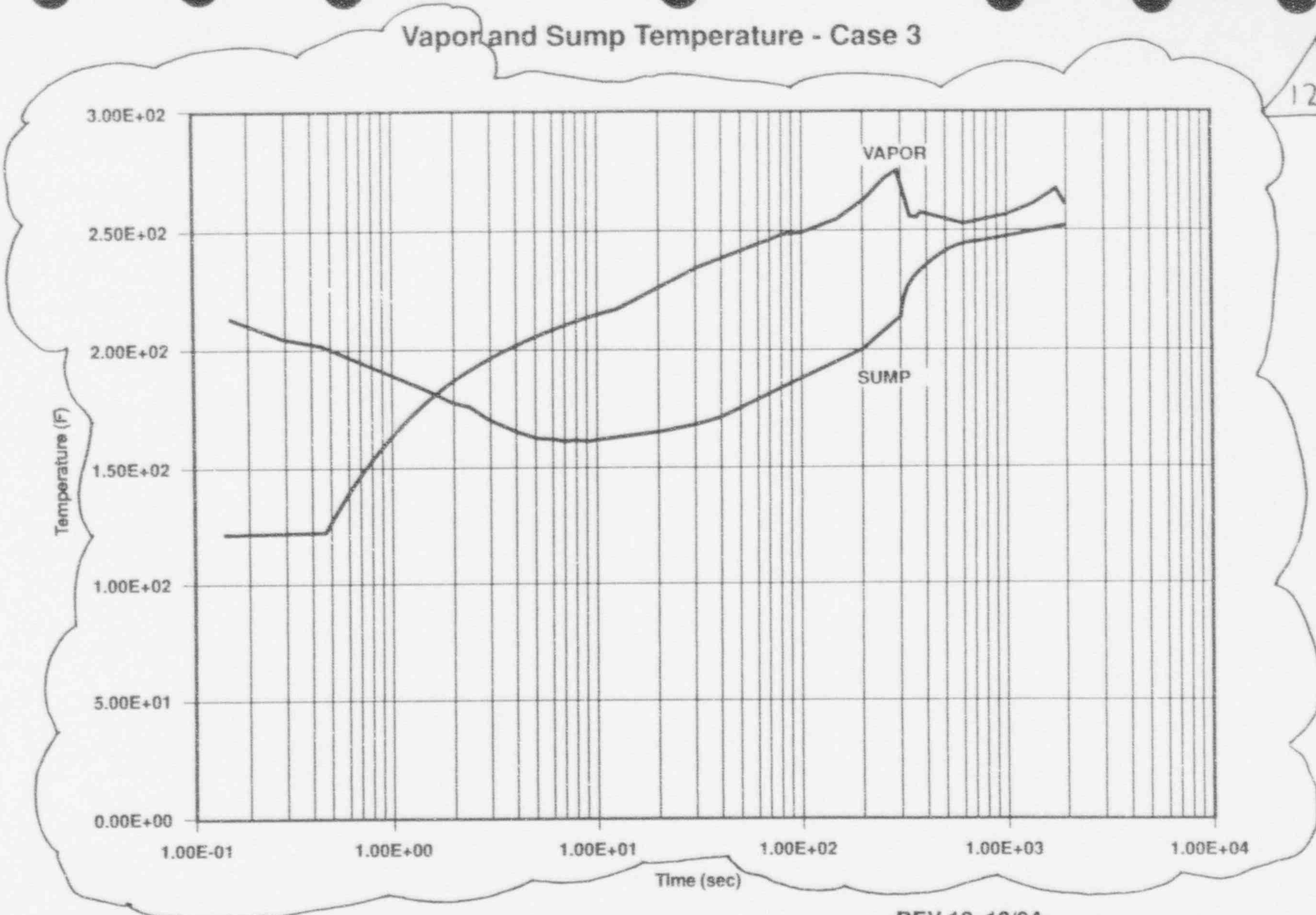
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRESSURE VERSUS TIME STEAM LINE 0.6 ft^2 D.E. BREAK 102% POWER	FIGURE 6.2-10



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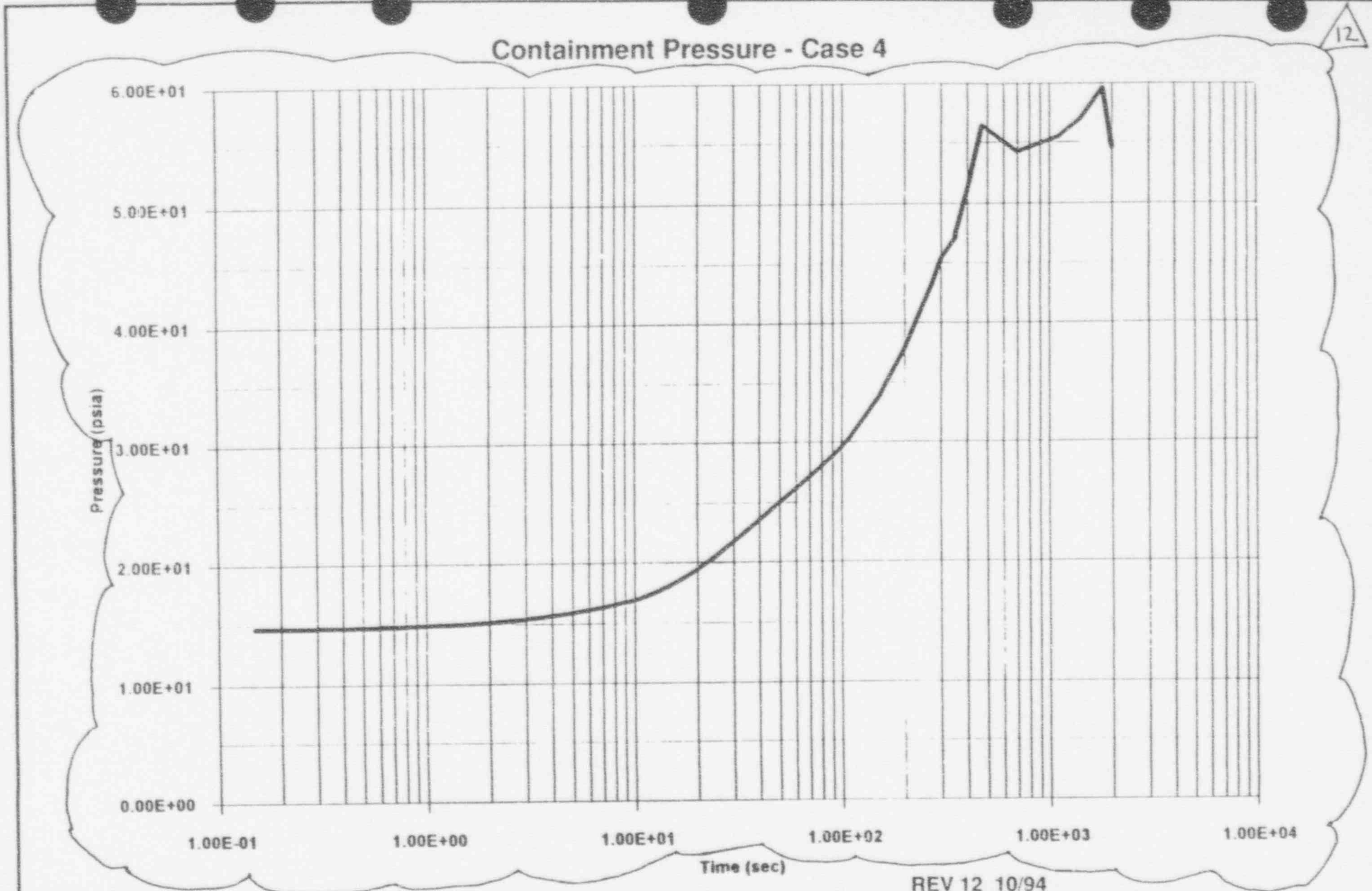
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	TEMPERATURE VERSUS TIME STEAM LINE 0.6 ft^2 D.E. BREAK 102% POWER
	FIGURE 6.2-11



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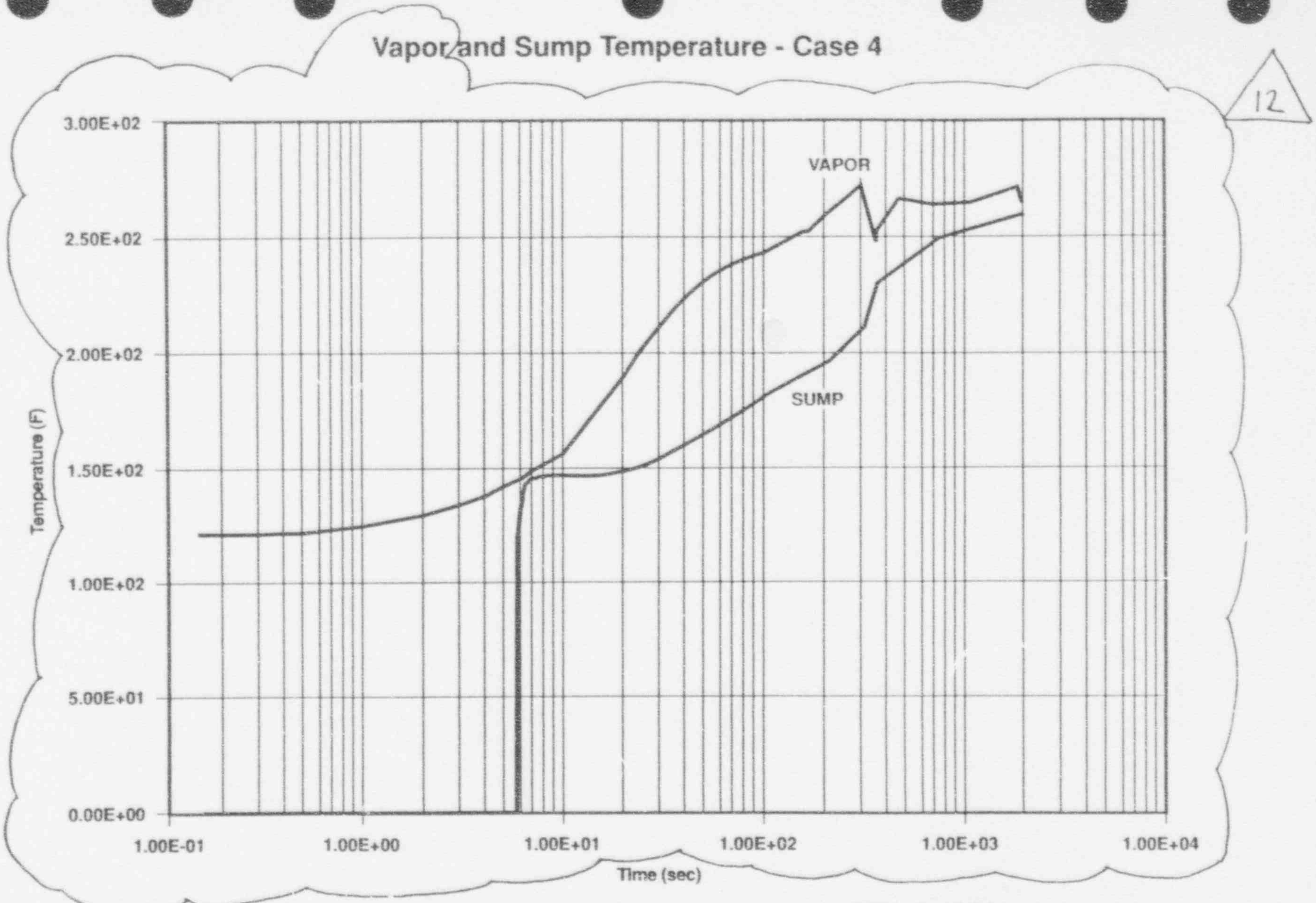
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRESSURE VERSUS TIME STEAM LINE 0.645 ft^2 SPLIT 102% POWER FIGURE 6.2-12



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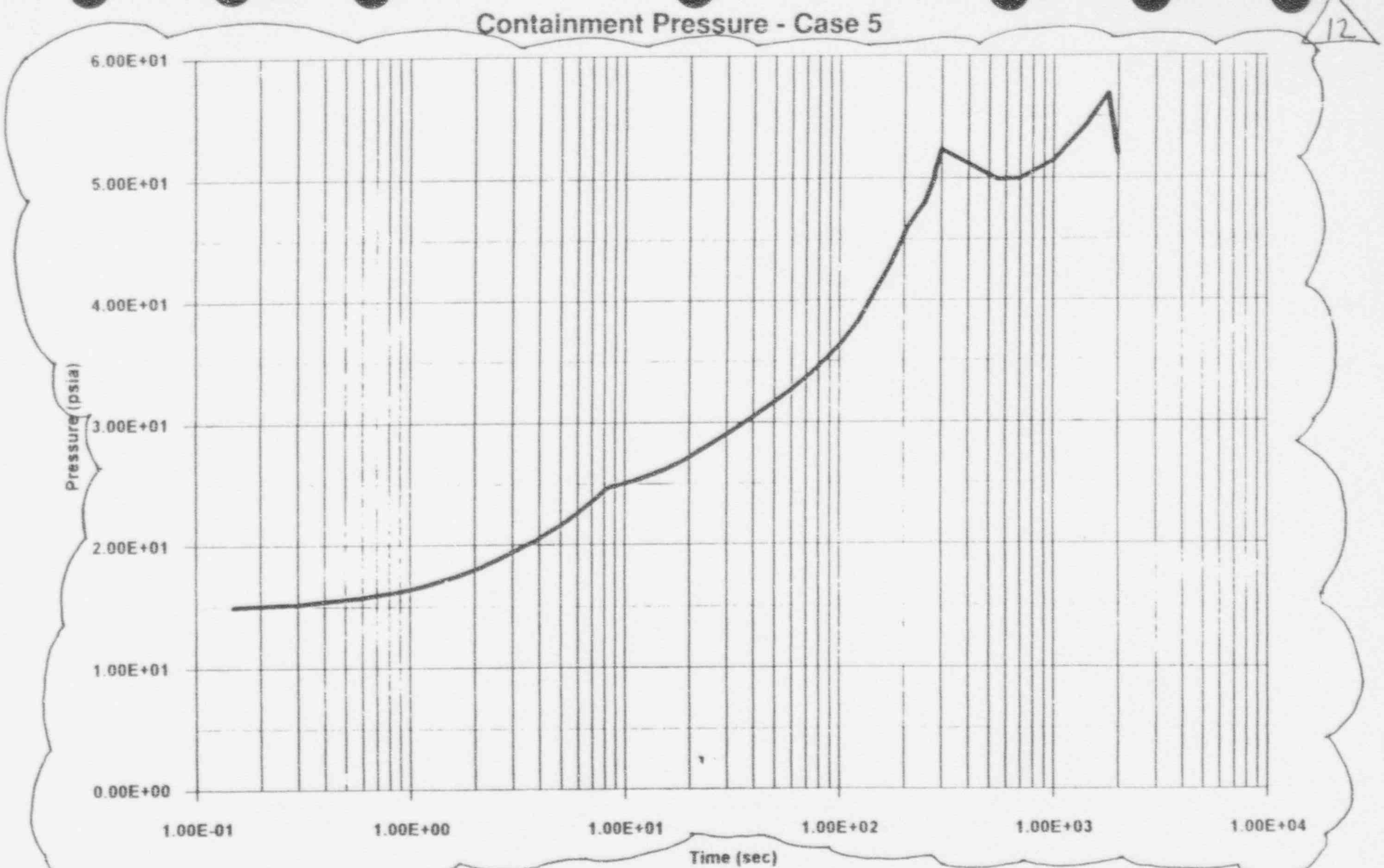
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	TEMPERATURE VERSUS TIME STEAM LINE 0.645 ft^2 SPLIT 102% POWER
	FIGURE 6.2-13



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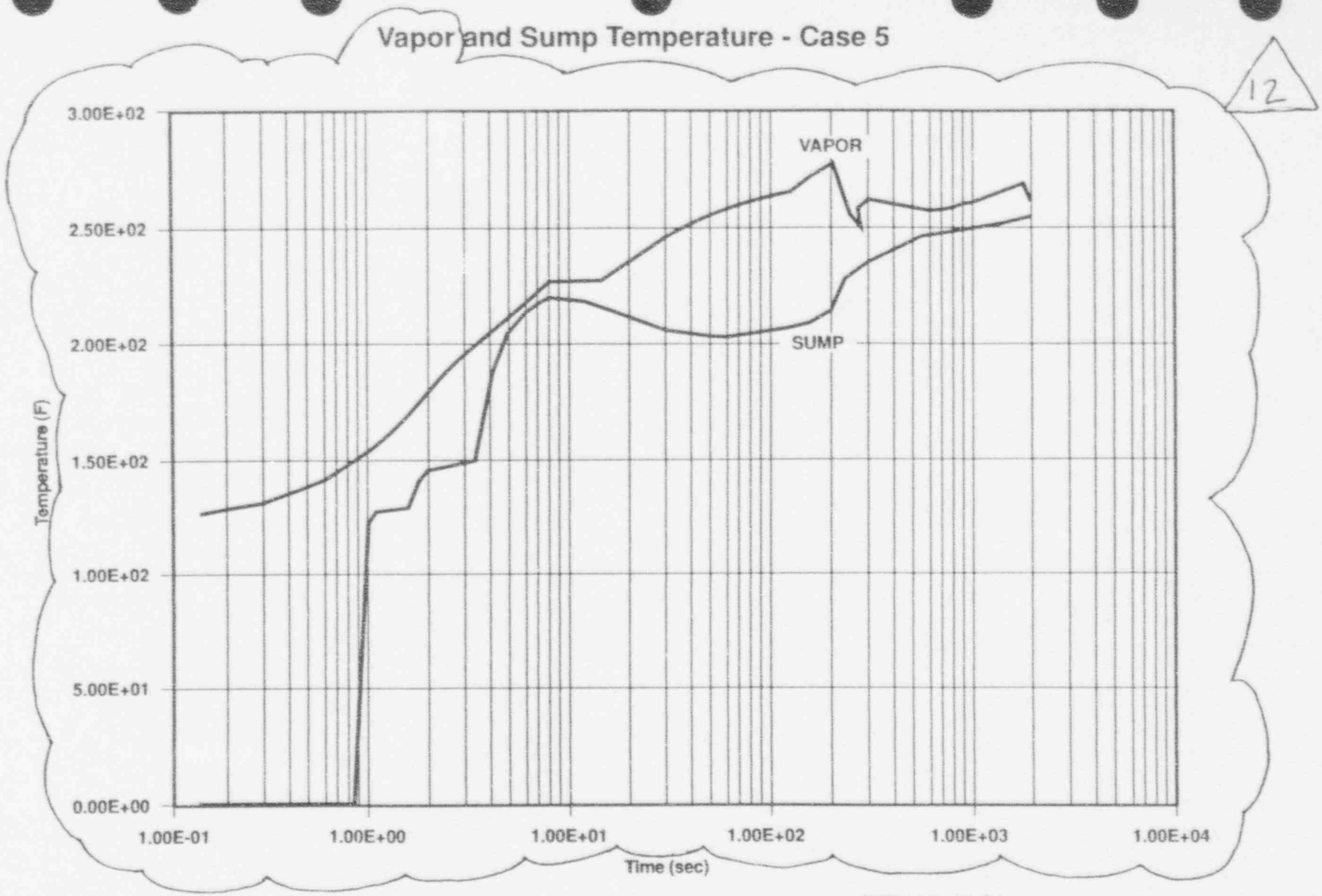
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRESSURE VERSUS TIME STEAM LINE FULL D.E. BREAK 70% POWER
	FIGURE 6.2-14



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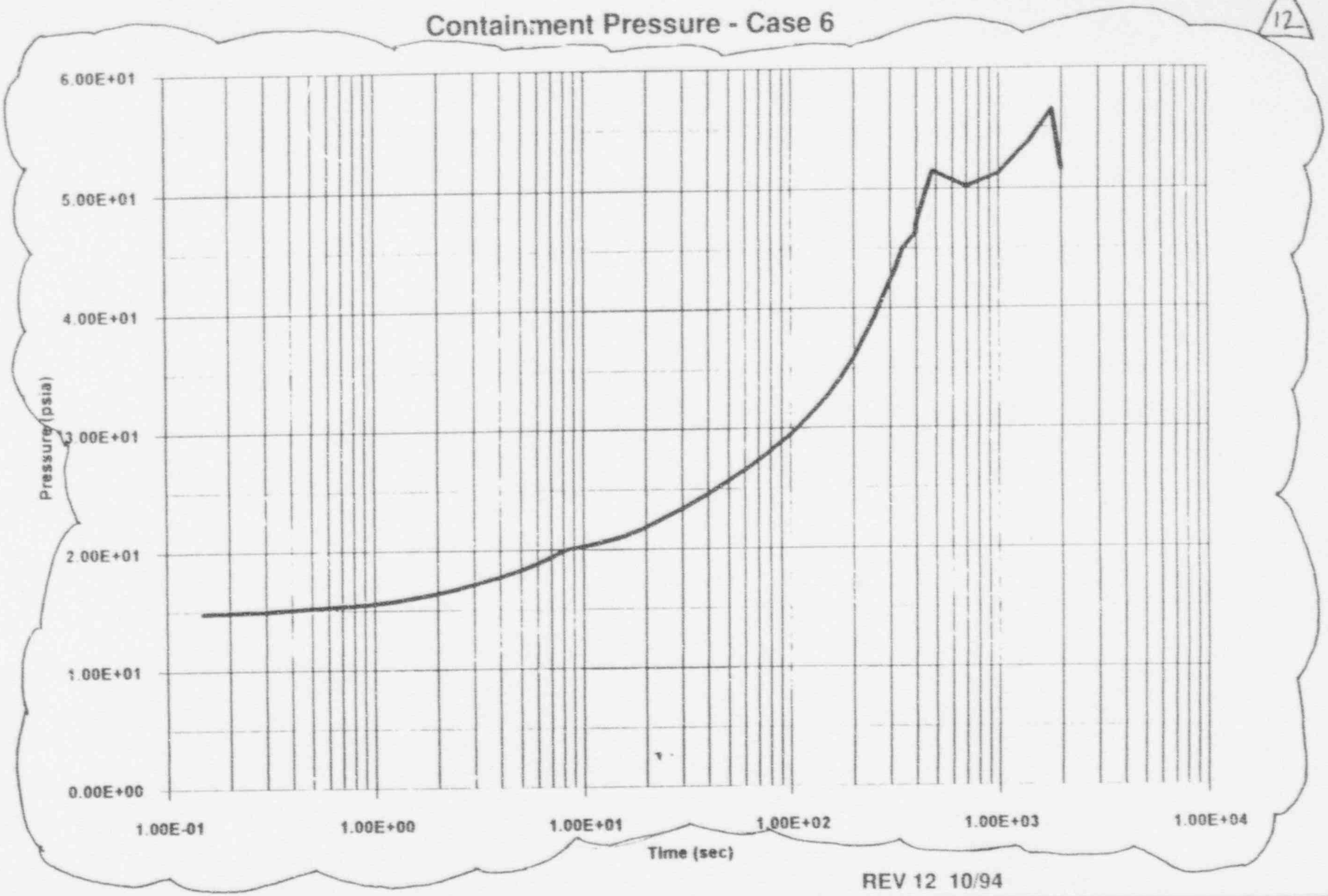
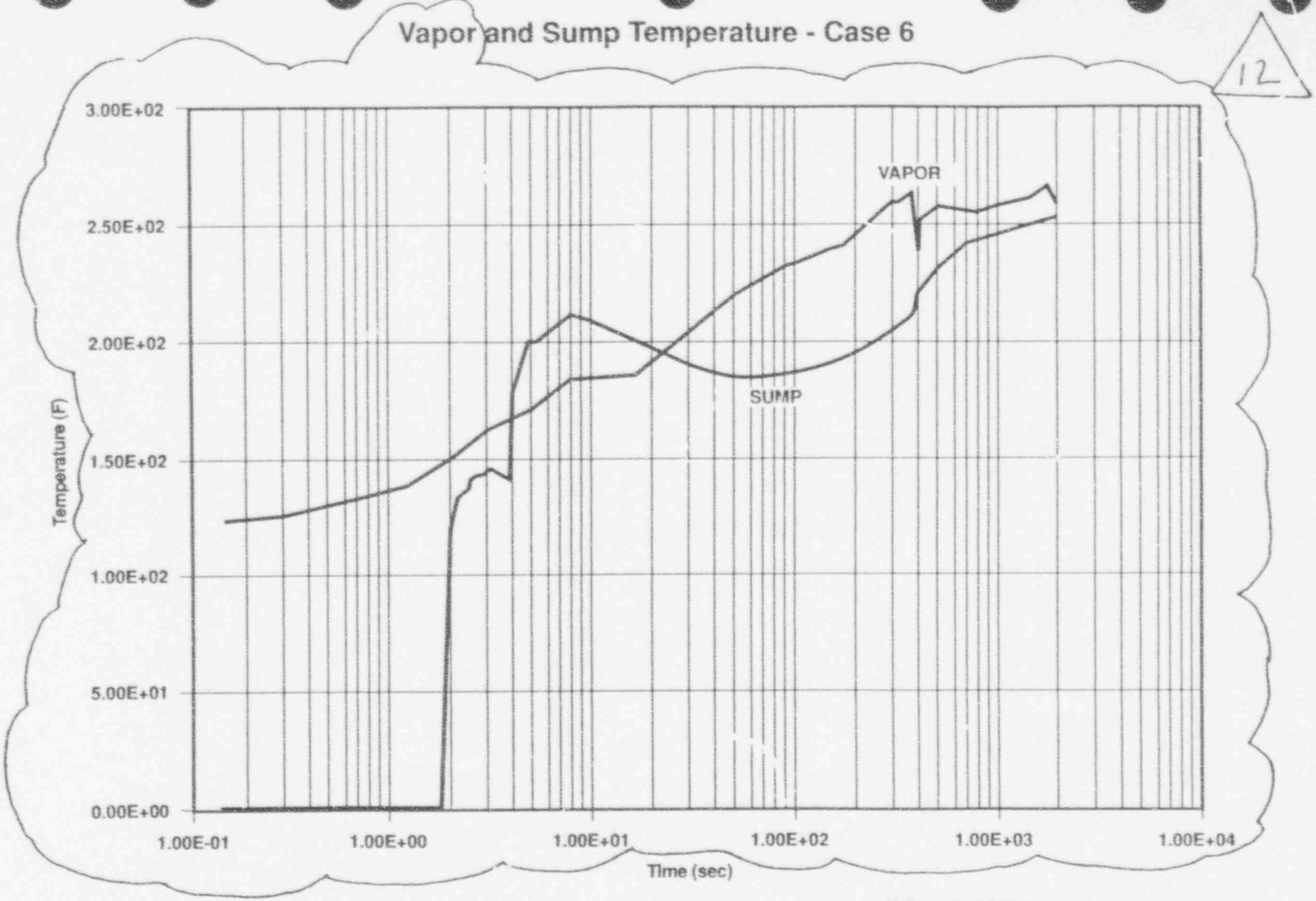
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	TEMPERATURE VERSUS TIME STEAM LINE FULL D.E. BREAK 70% POWER
	FIGURE 6.2-15



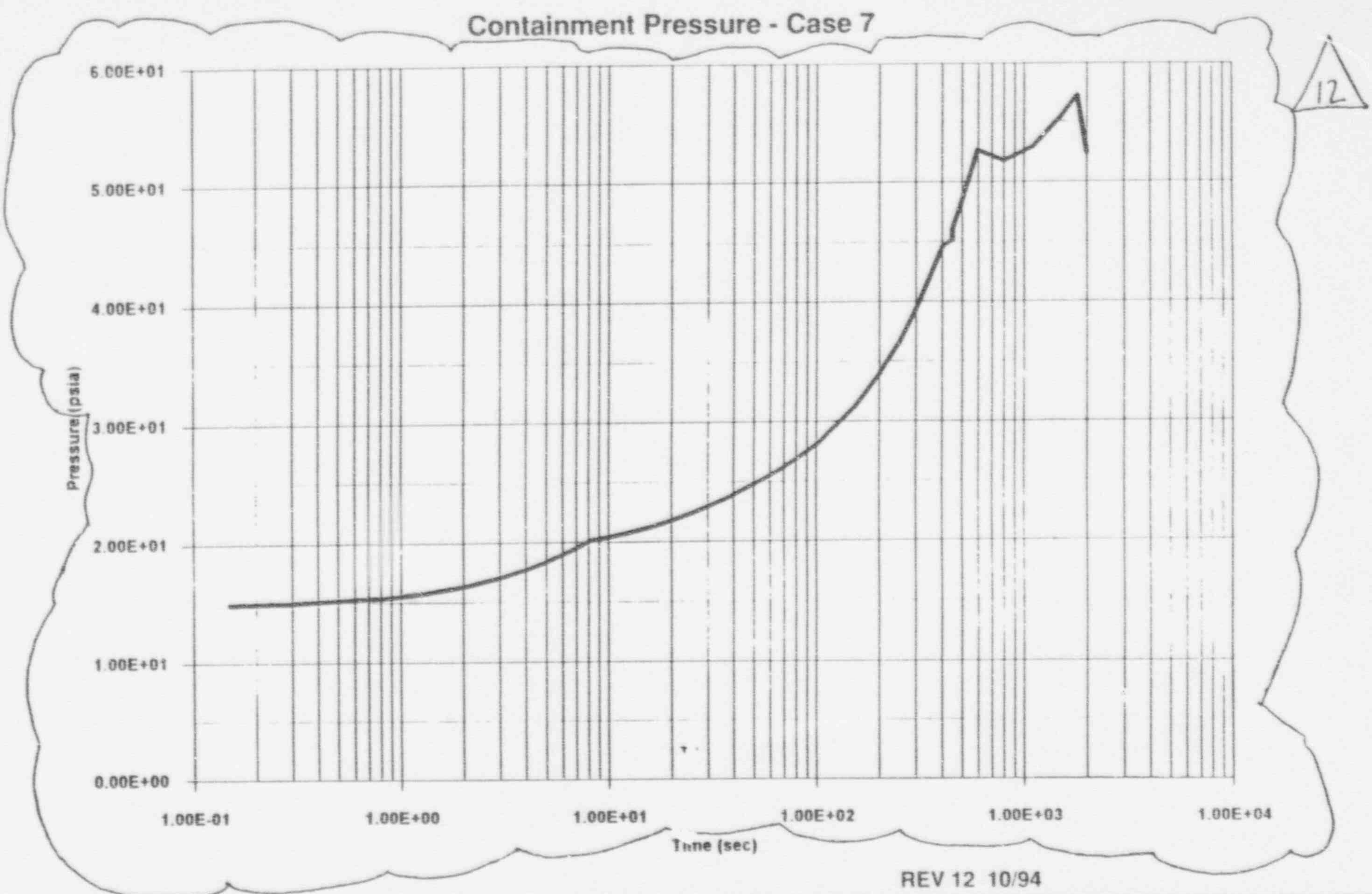
REV 12 10/94

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRESSURE VERSUS TIME STEAM LINE 0.6 ft^2 D.E. BREAK 70% POWER
	FIGURE 6.2-16



REV 12 10/94

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	TEMPERATURE VERSUS TIME STEAM LINE 0.6 ft^2 D.E. BREAK 70% POWER
	FIGURE 6.2-17



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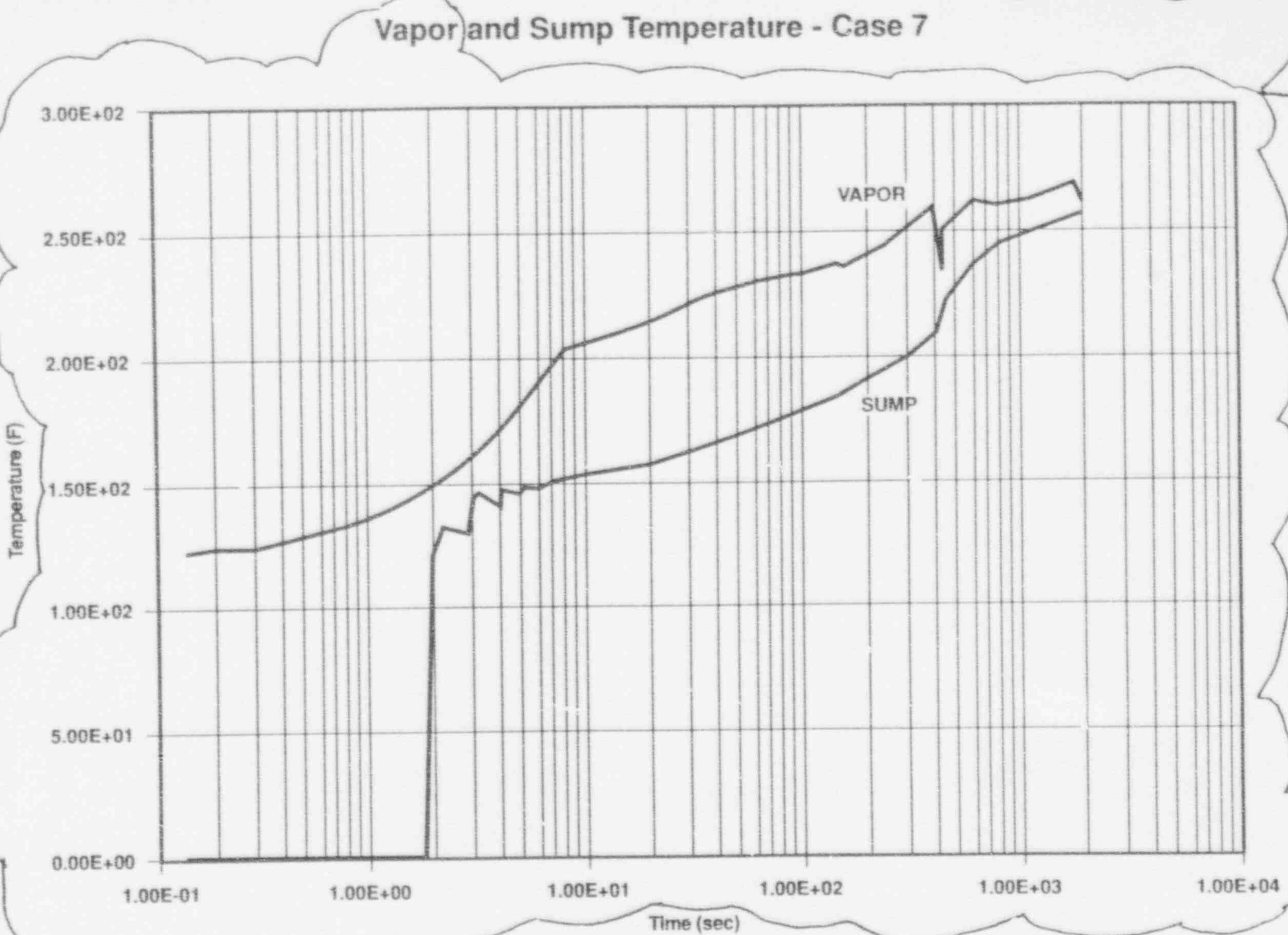
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRESSURE VERSUS TIME STEAM LINE 0.5 ft^2 D.E. BREAK 70% POWER
	FIGURE 6.2-18



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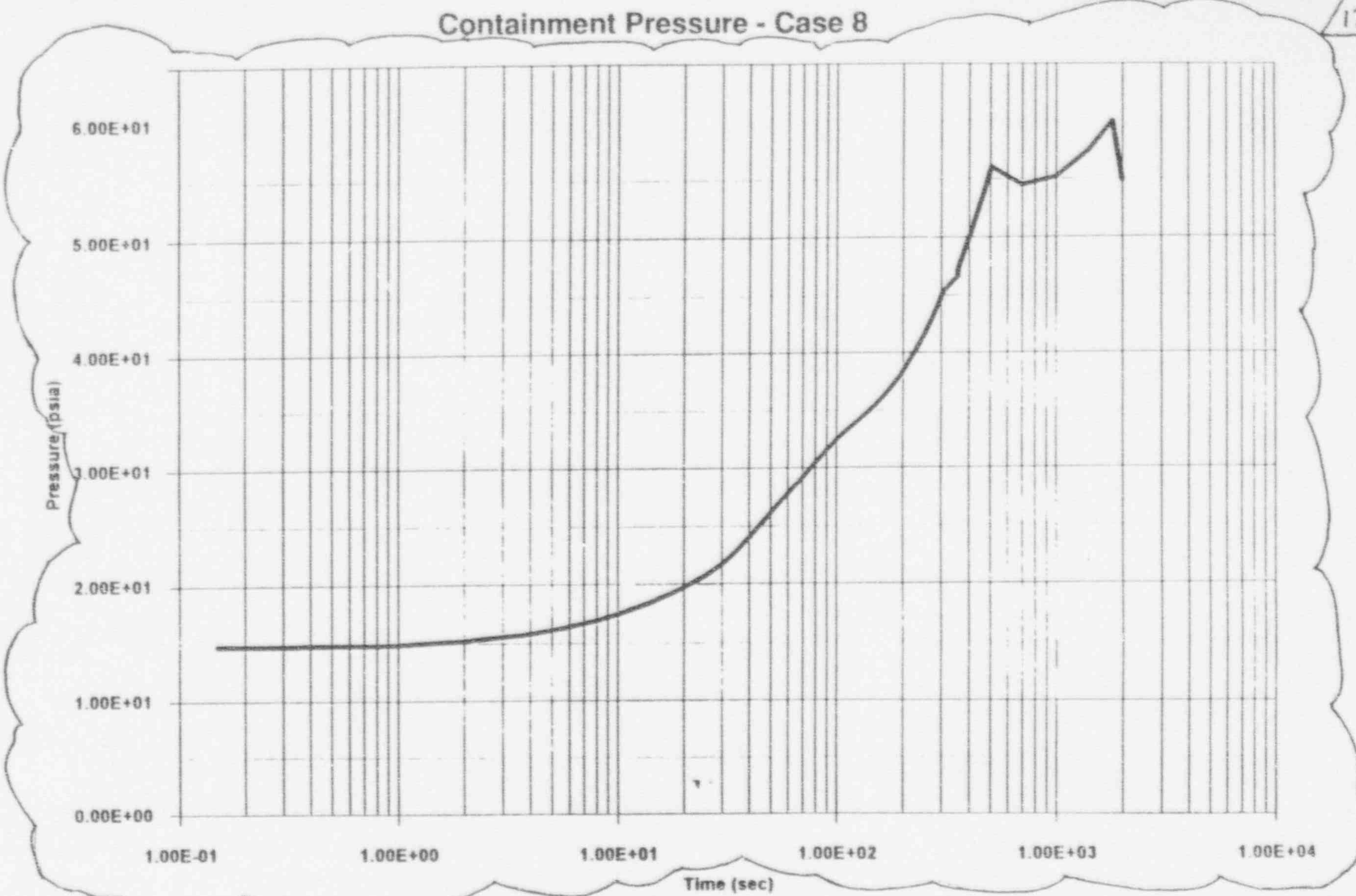
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	TEMPERATURE VERSUS TIME STEAM LINE 0.5 ft^2 D.E. BREAK 70% POWER
	FIGURE 6.2-19



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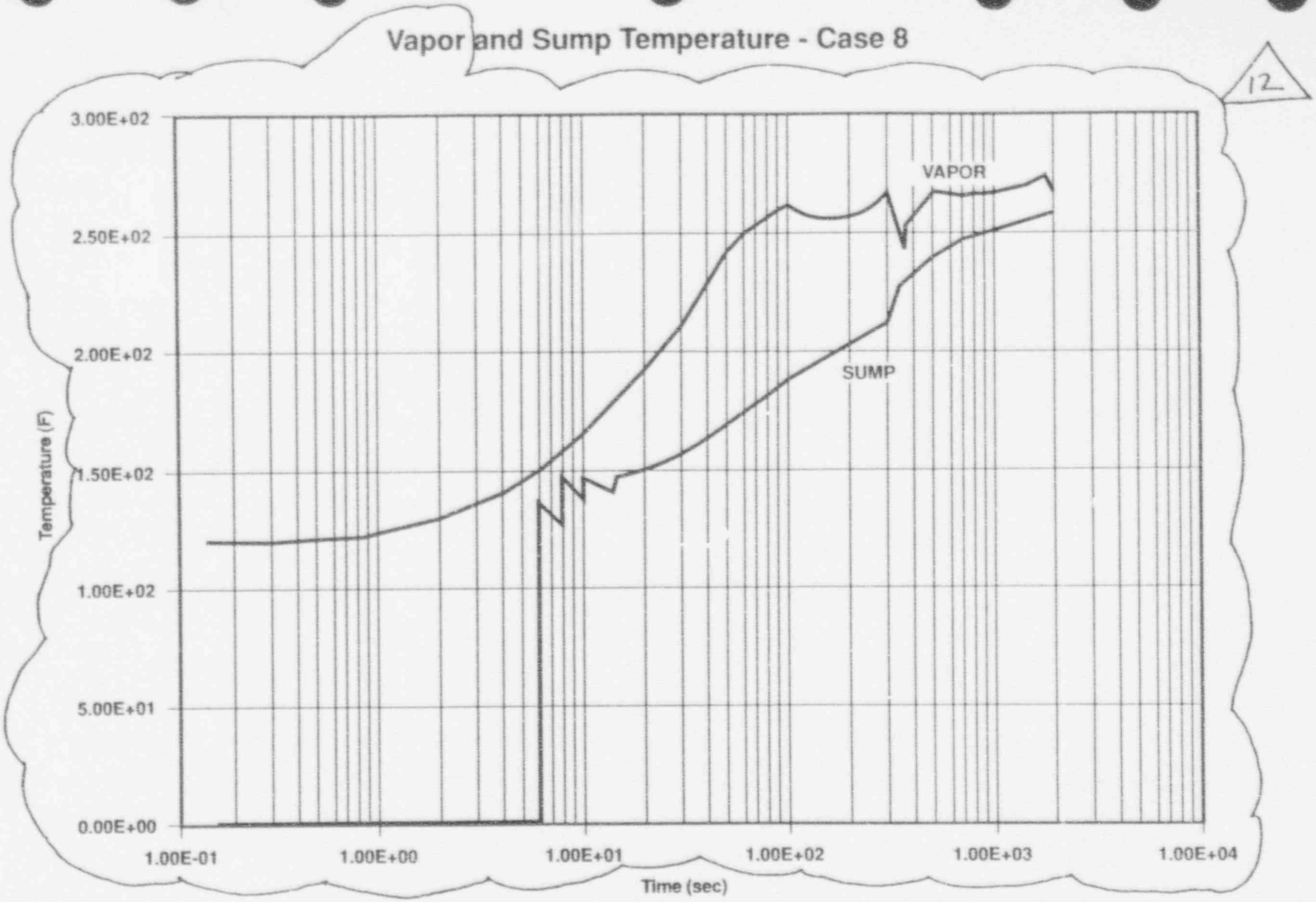
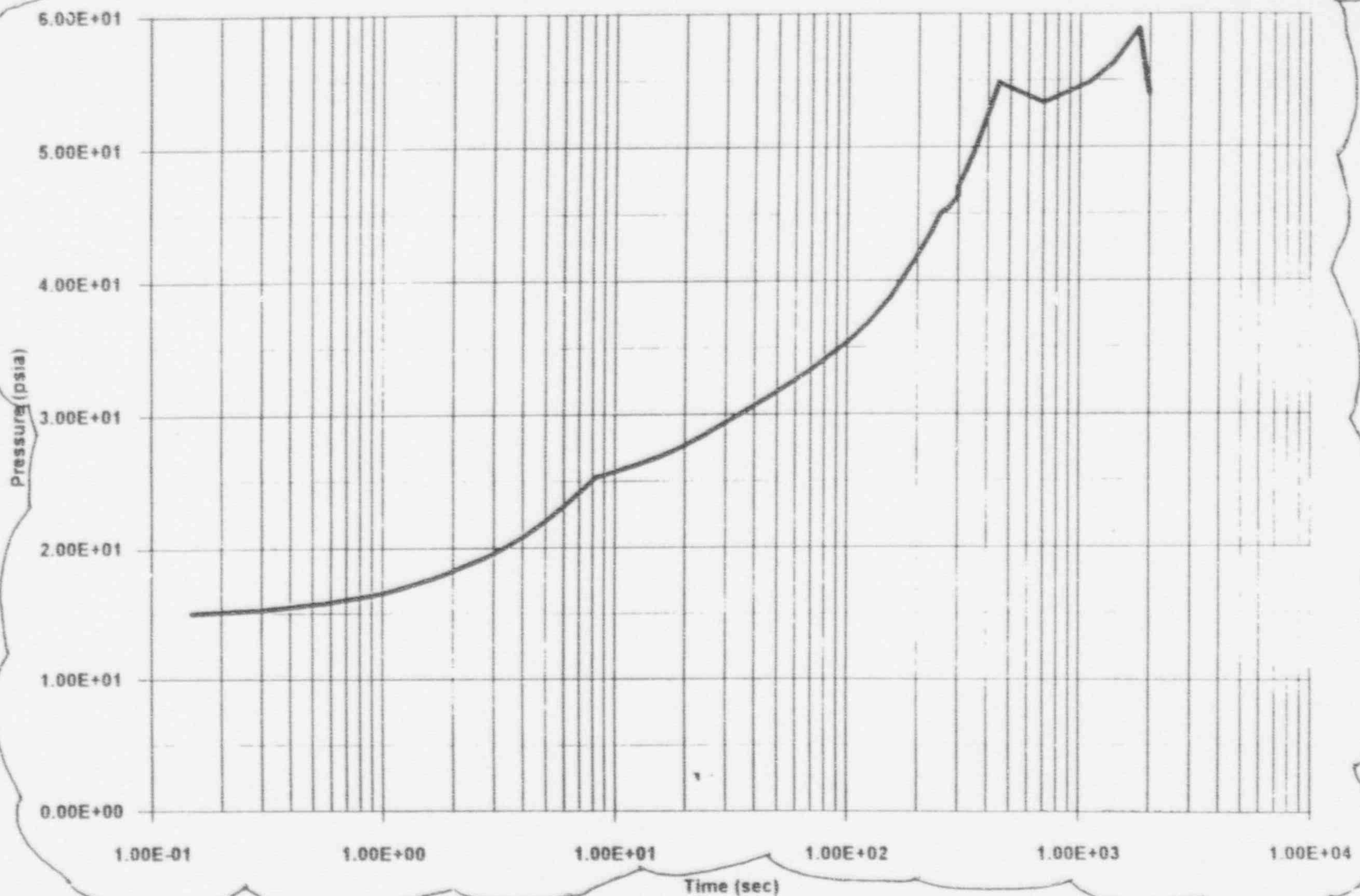
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRESSURE VERSUS TIME STEAM LINE 0.681 ft^2 SPLIT 70% POWER
	FIGURE 6.2-20



REV 12 10/94

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	TEMPERATURE VERSUS TIME STEAM LINE 3.681 ft^2 SPLIT 70% POWER
	FIGURE 6.2-21

Containment Pressure - Case 9



12

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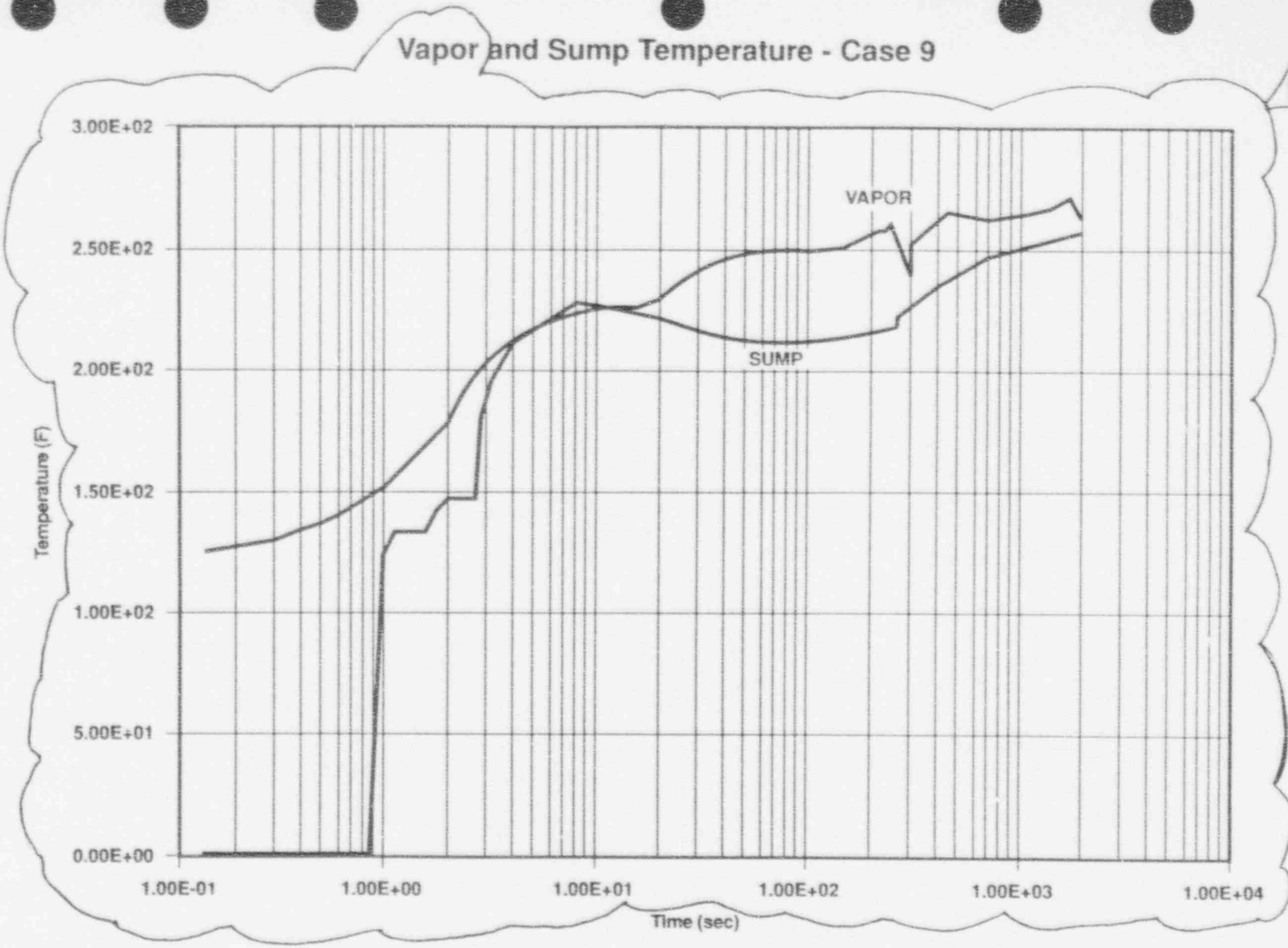
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRESSURE VERSUS TIME STEAM LINE FULL D.E. BREAK 30% POWER
	FIGURE 6.2-22

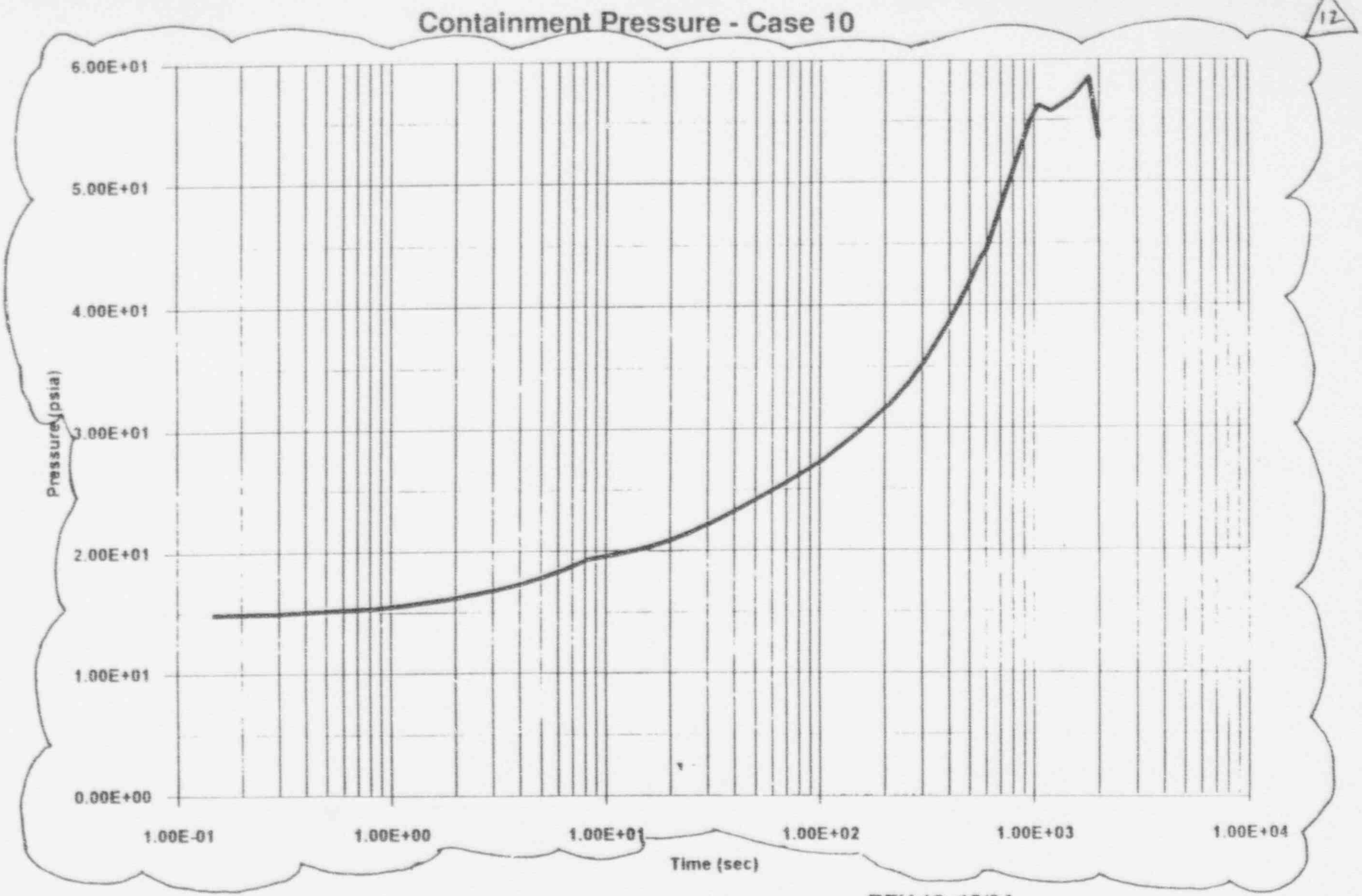


REV 12 10/94

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

TEMPERATURE VERSUS TIME
STEAM LINE FULL D.E. BREAK
30% POWER

FIGURE 6.2-23



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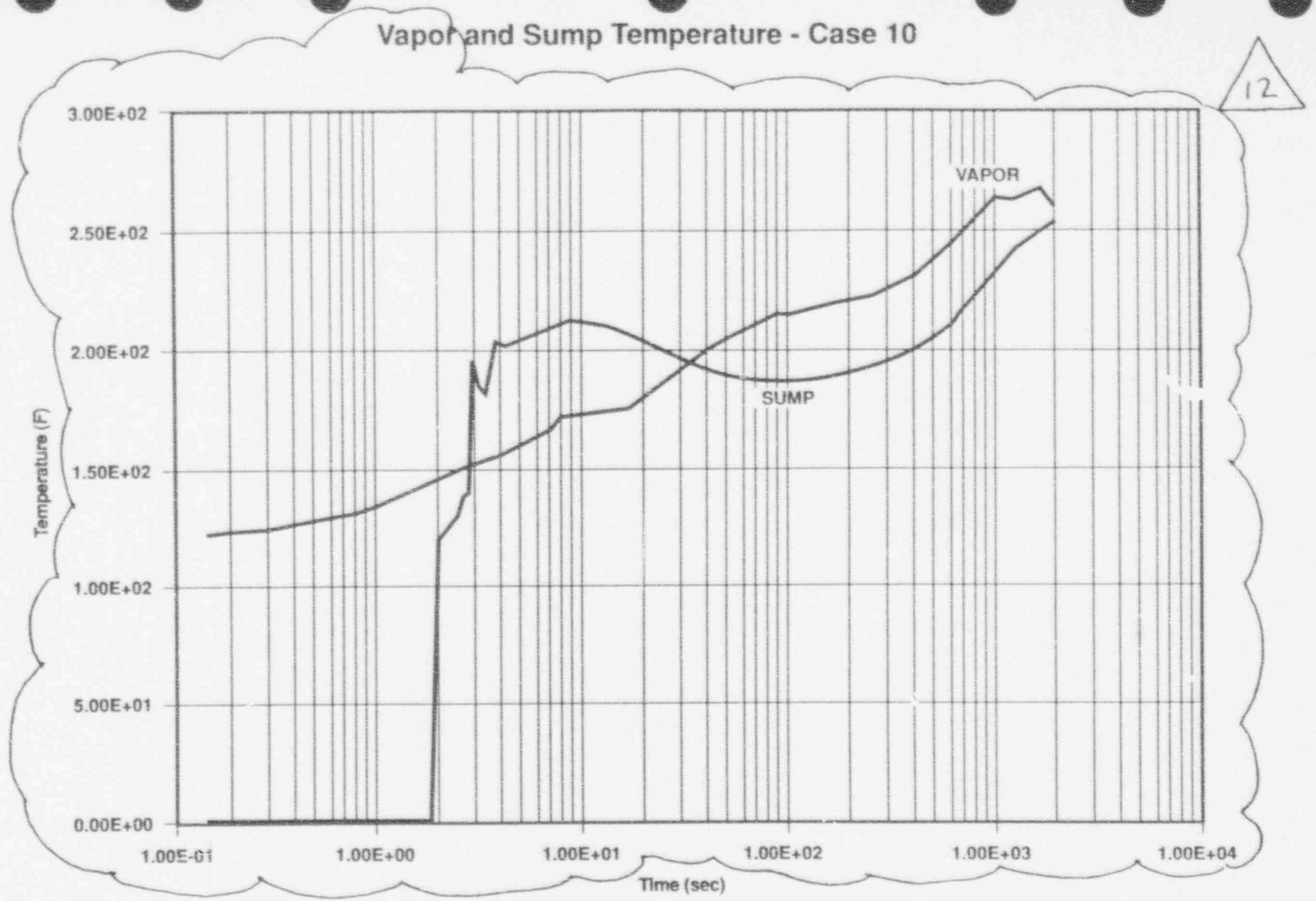
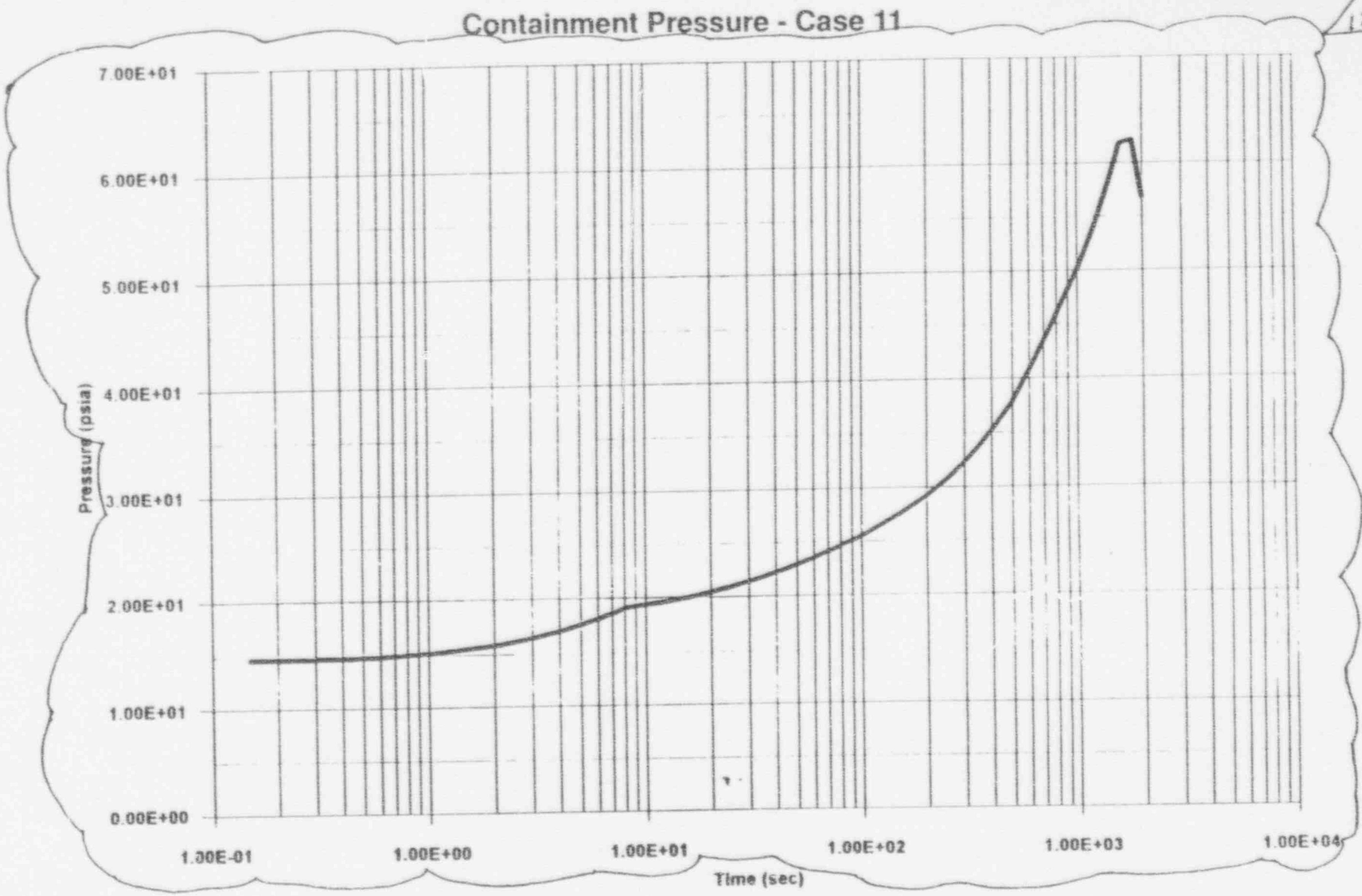
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRESSURE VERSUS TIME STEAM LINE 0.5 ft^2 D.E. BREAK 30% POWER
	FIGURE 6.2-24



REV 12 10/94

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	TEMPERATURE VERSUS TIME STEAM LINE 0.5 ft^2 D.E. BREAK 30% POWER
	FIGURE 6.2-25



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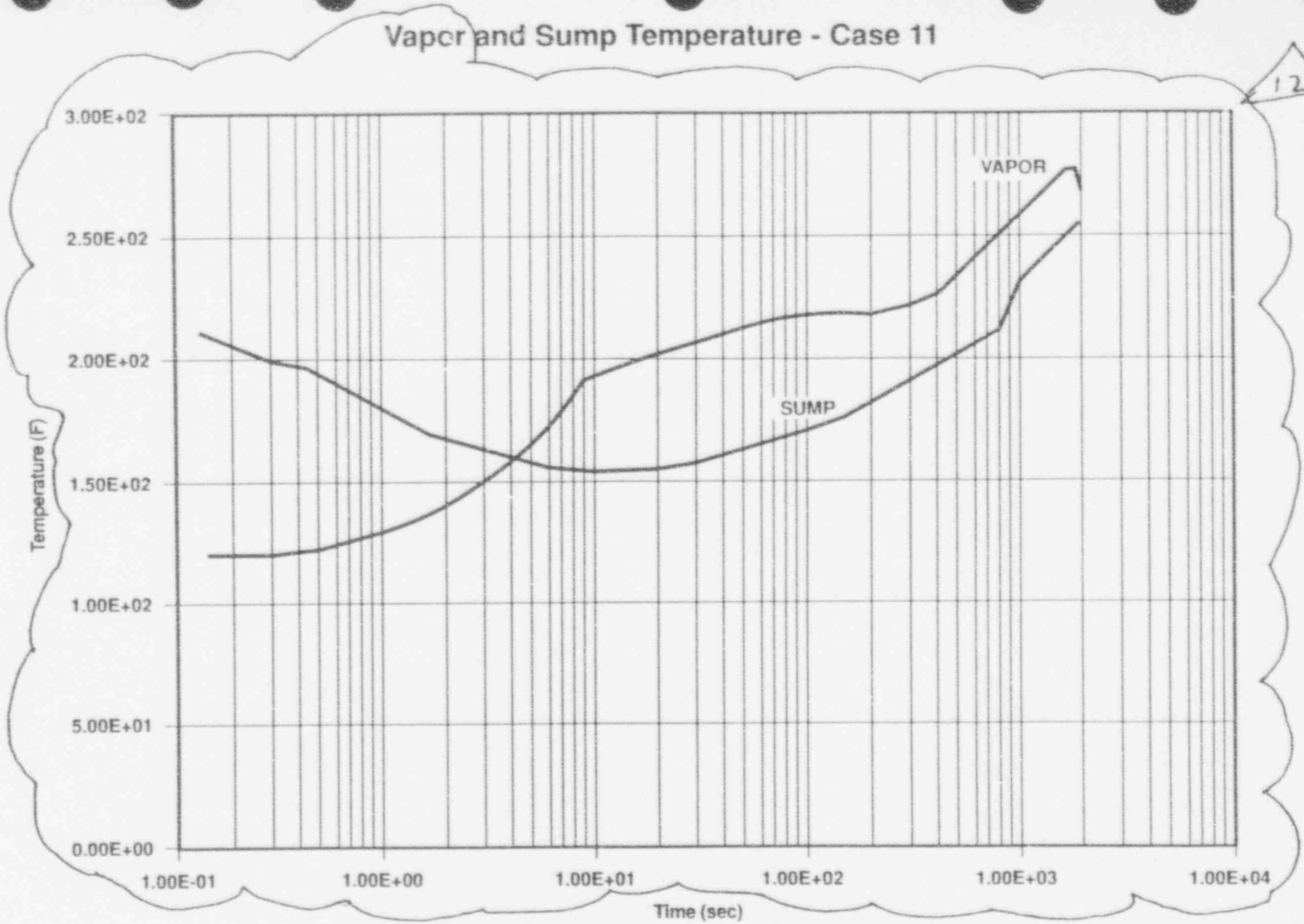
REV 12 10/94

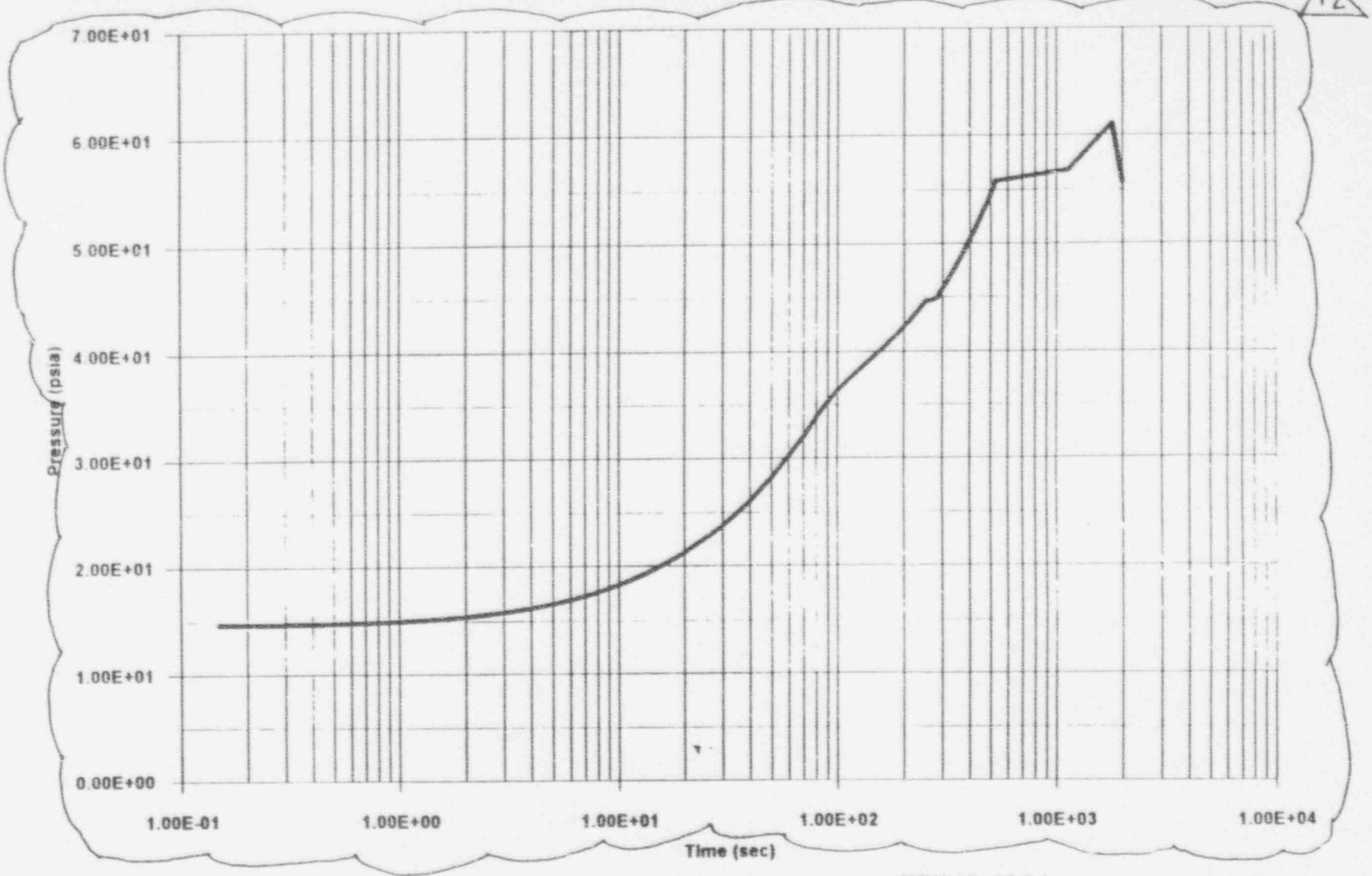
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRESSURE VERSUS TIME STEAM LINE 0.4 ft^2 D.E. BREAK 30% POWER
	FIGURE 6.2-26



REV 12 10/94

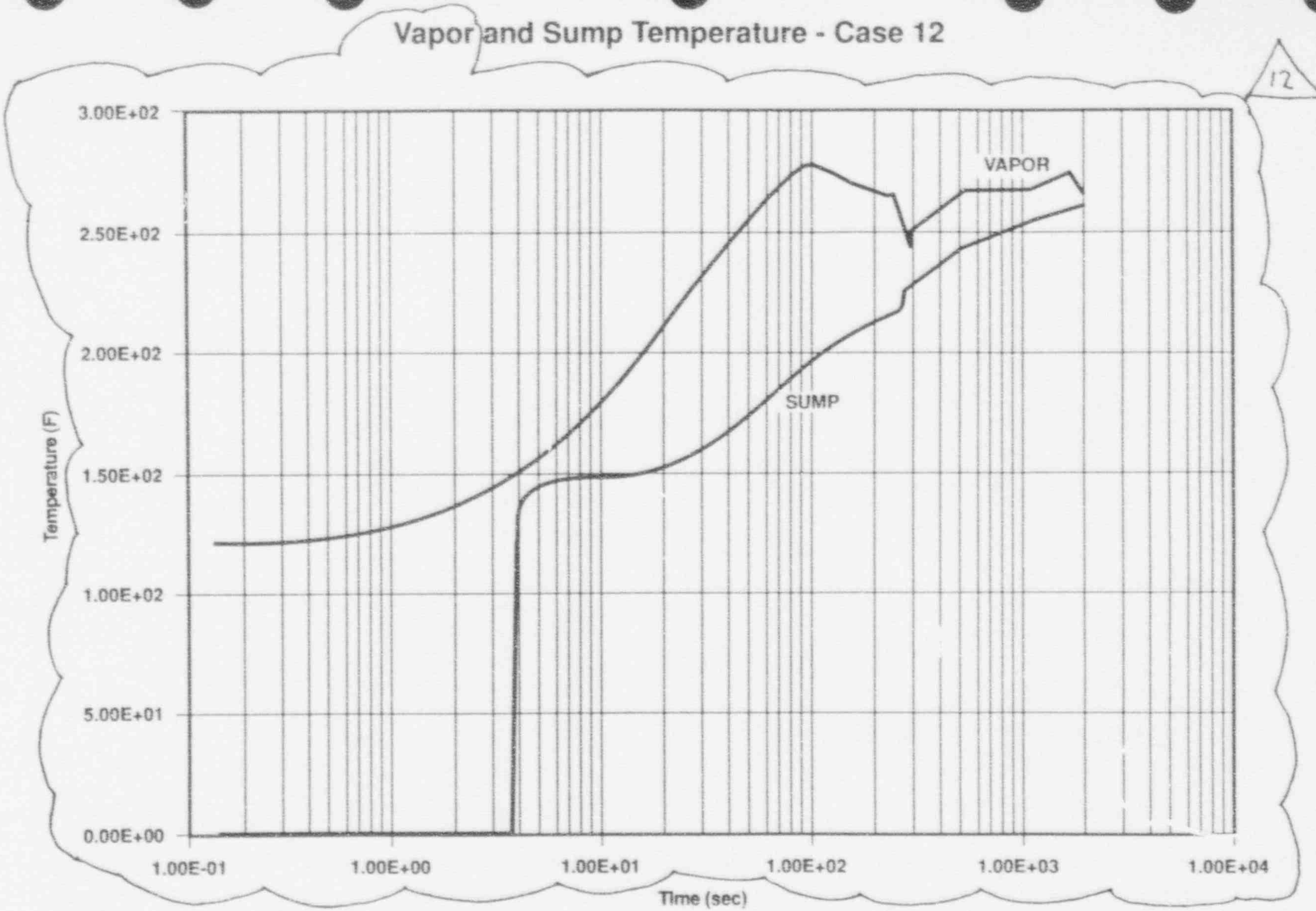
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	TEMPERATURE VERSUS TIME STEAM LINE 0.4 ft^2 D.E. BREAK 30% POWER
	FIGURE 6.2-27

Containment Pressure - Case 12



REV 12 10/94

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRESSURE VERSUS TIME STEAM LINE 0.7065 ft^2 SPLIT 30% POWER FIGURE 6.2-28



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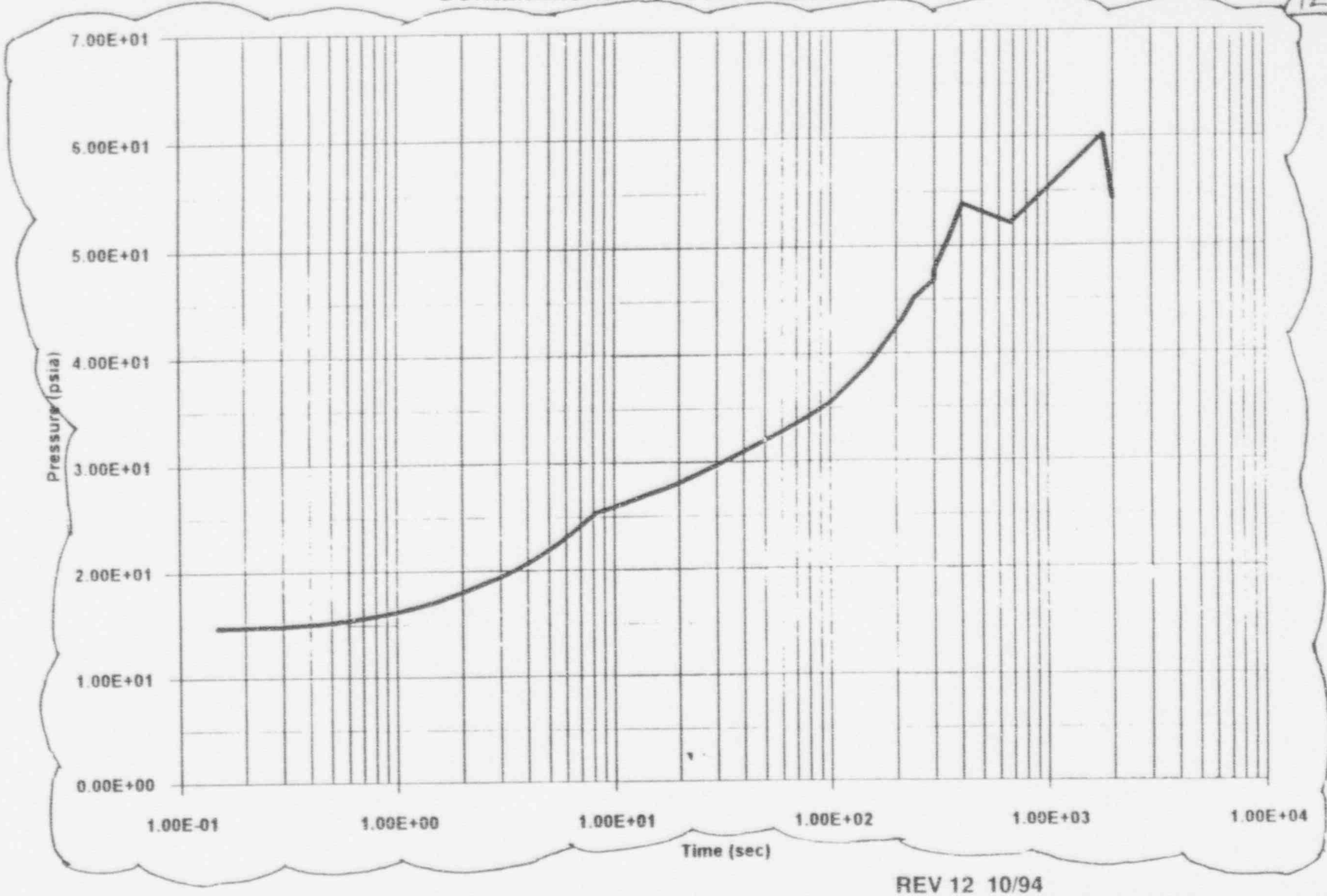
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	TEMPERATURE VERSUS TIME STEAM LINE 0.7065 ft^2 SPLIT 30% POWER
	FIGURE 6.2-29

Containment Pressure - Case 13



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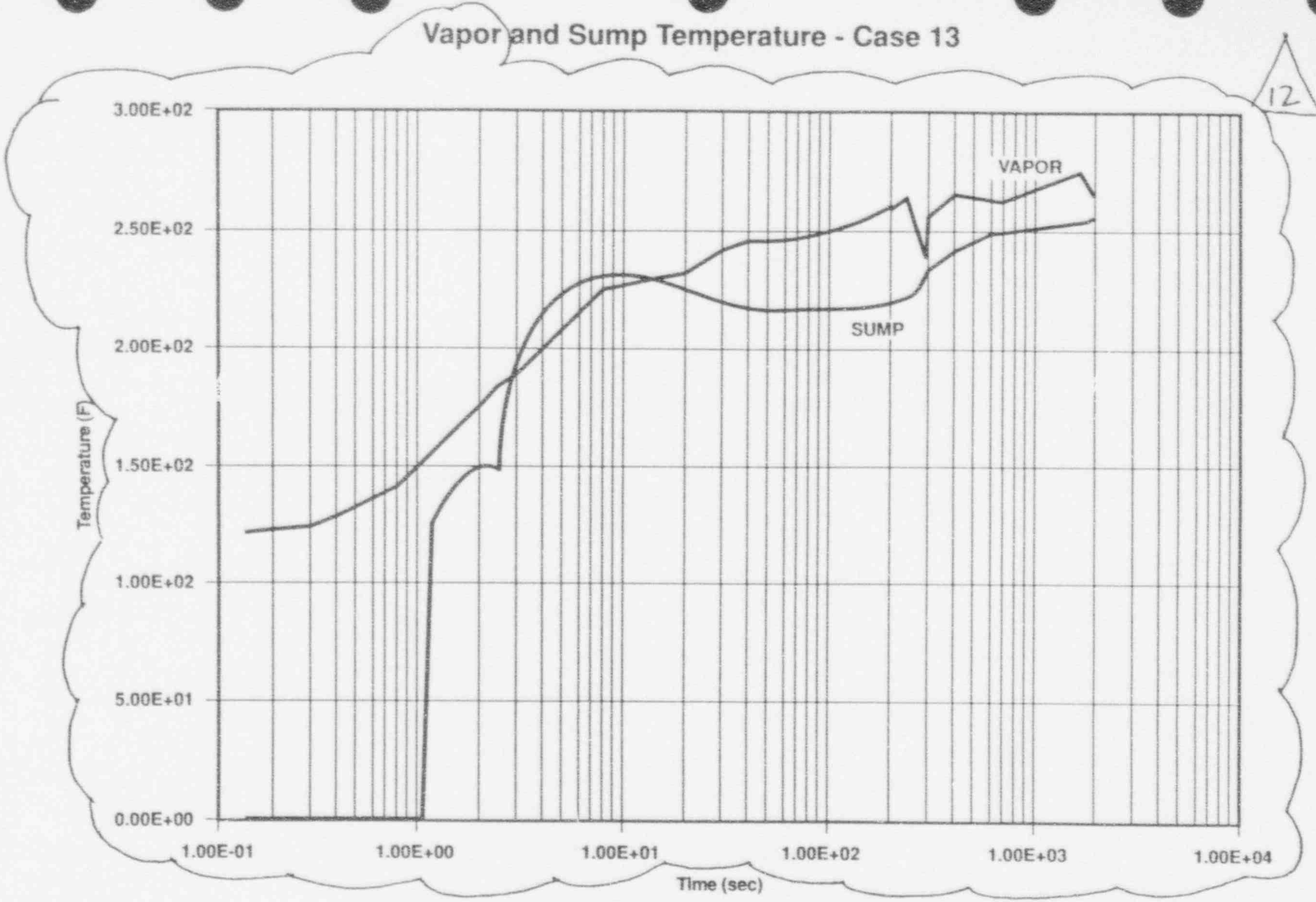
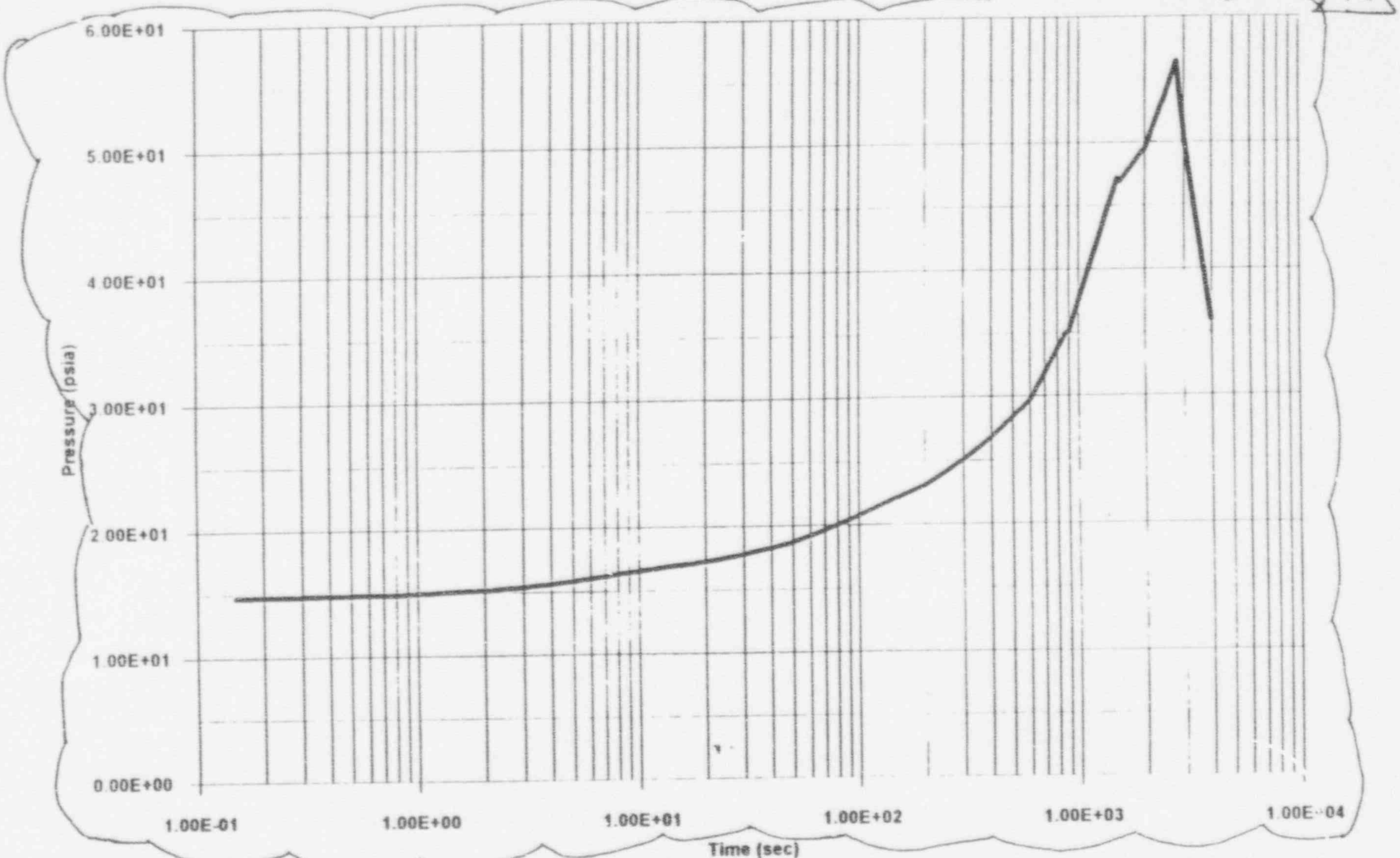
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRESSURE VERSUS TIME STEAM LINE FULL D.E. BREAK HOT STANDBY FIGURE 6.2-30



REV 12 10/94

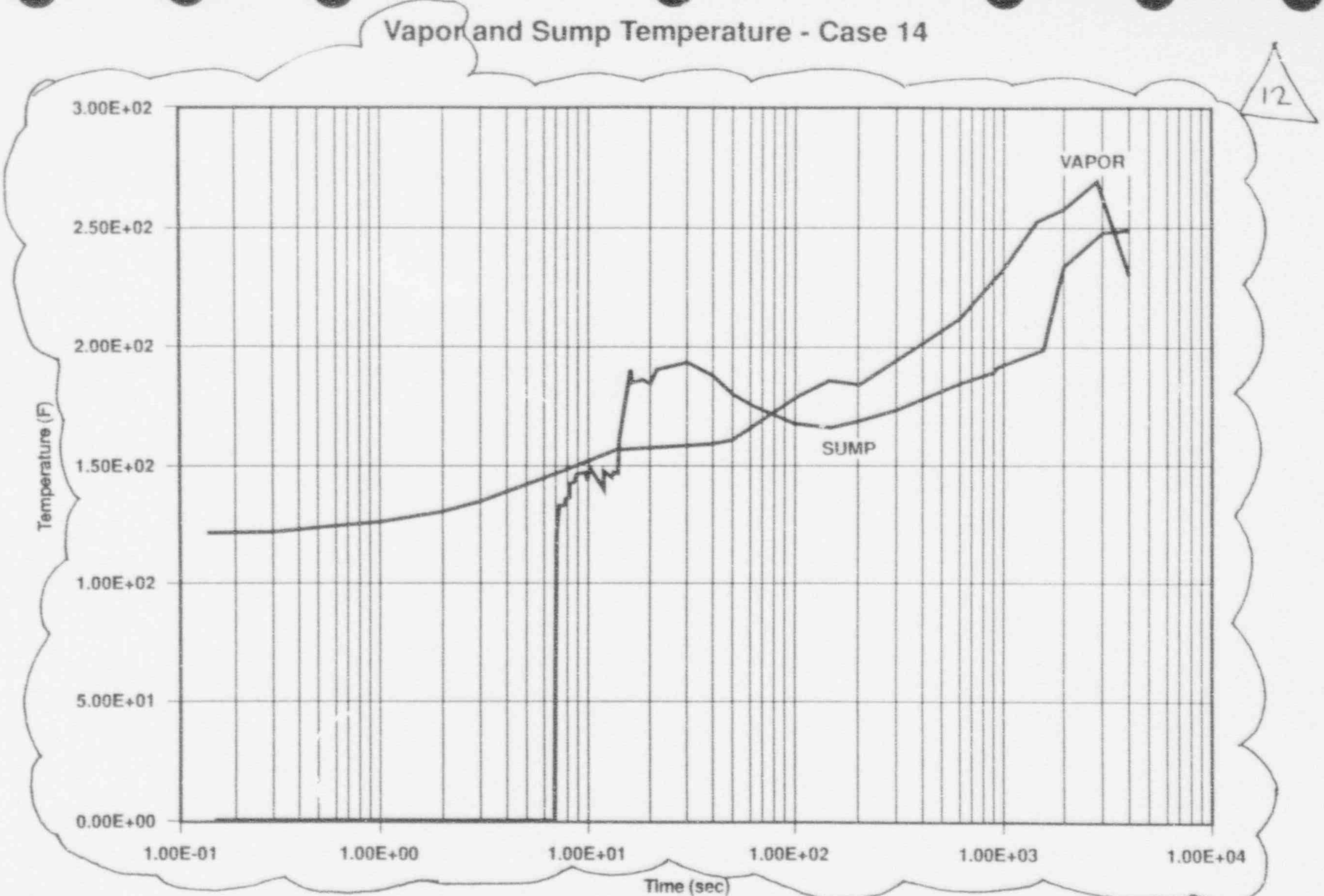
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	TEMPERATURE VERSUS TIME STEAM LINE FULL D.E. BREAK HOT STANDBY
	FIGURE 6.2-31

Containment Pressure - Case 14



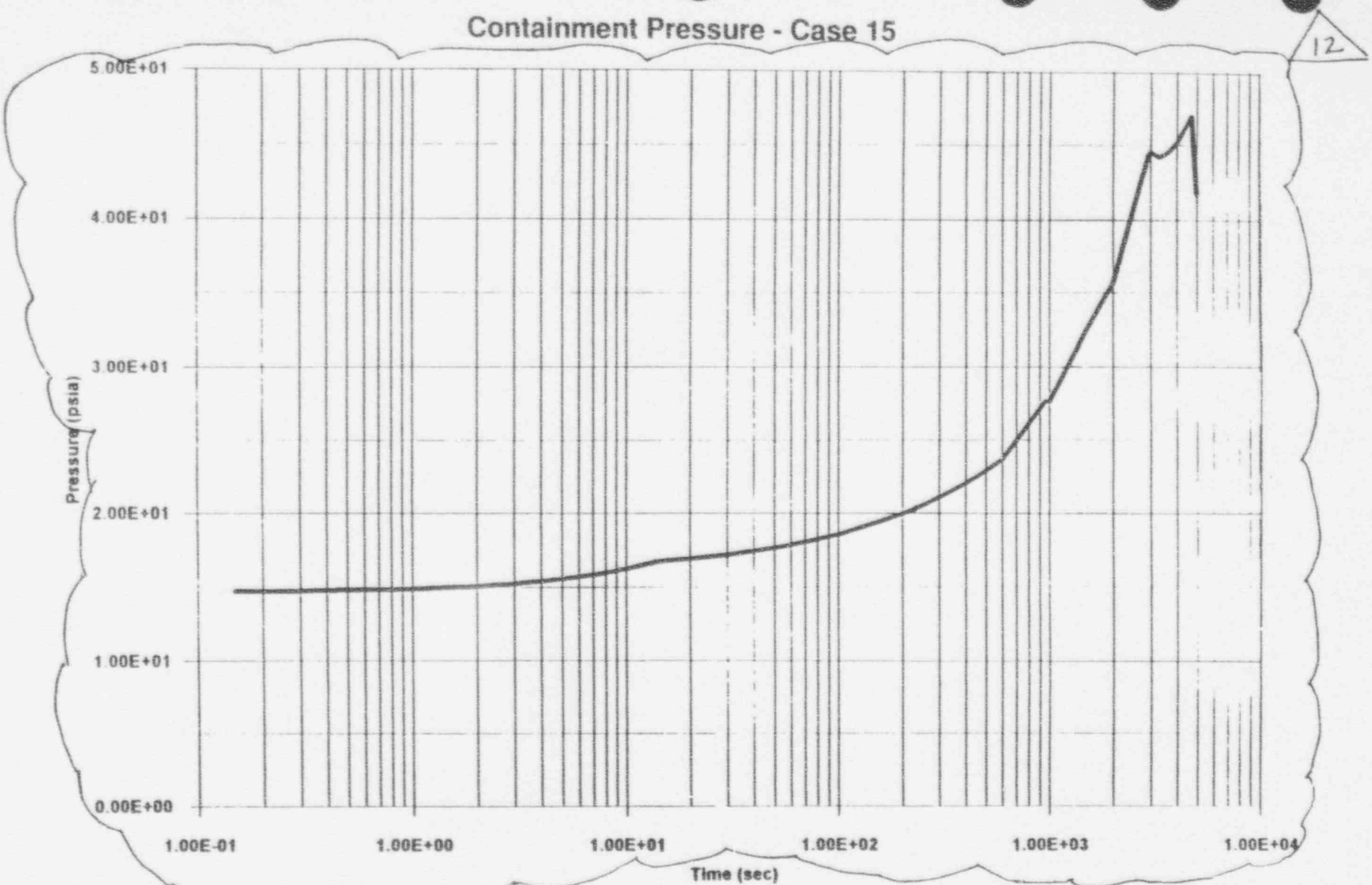
REV 12 10/94

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRESSURE VERSUS TIME STEAM LINE 0.2 ft^2 D.E. BREAK HOT STANDBY
	FIGURE 6.2-32



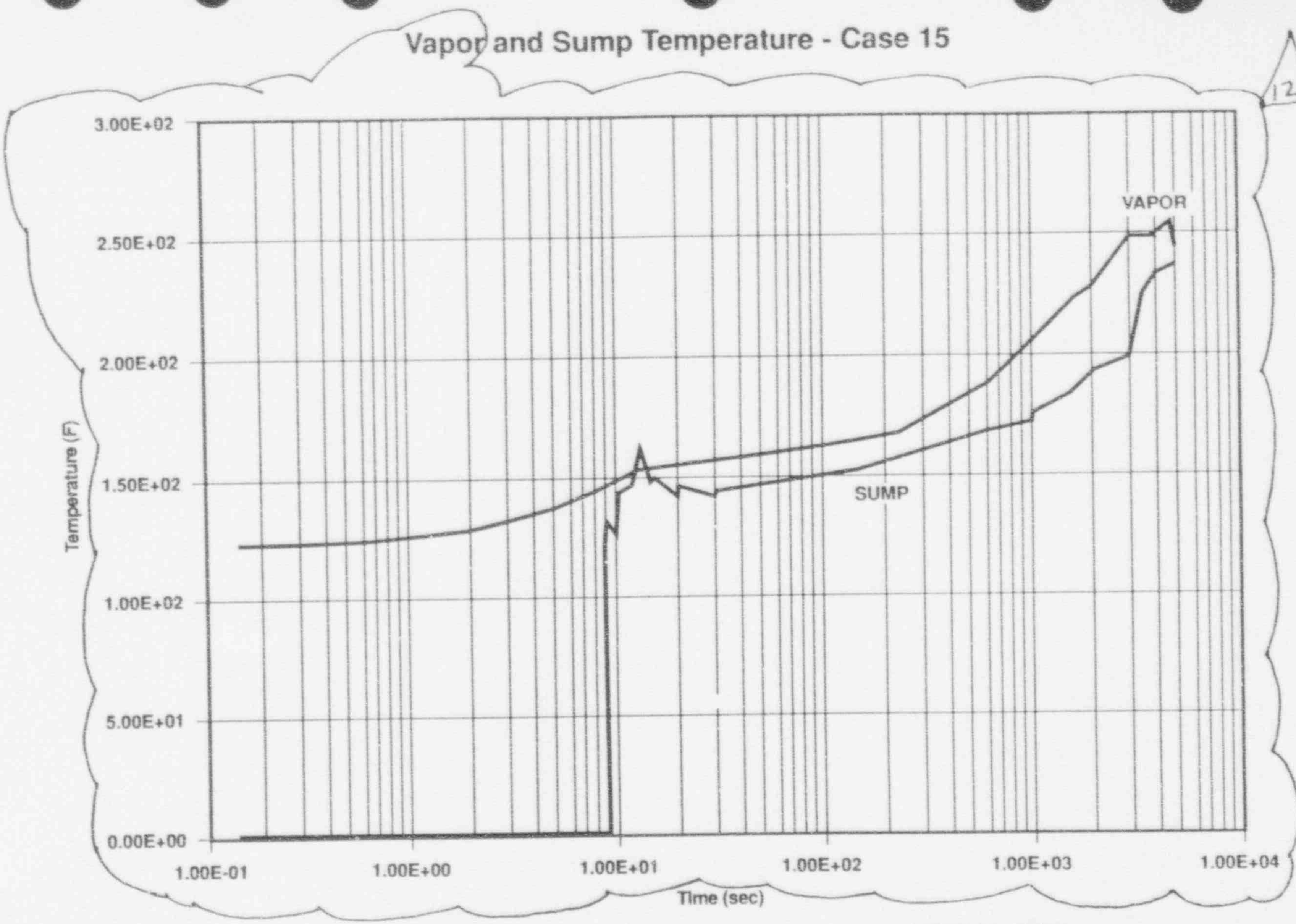
REV 12 10/94

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	TEMPERATURE VERSUS TIME STEAM LINE 0.2 ft^2 D.E. BREAK HOT STANDBY	FIGURE 6.2-33



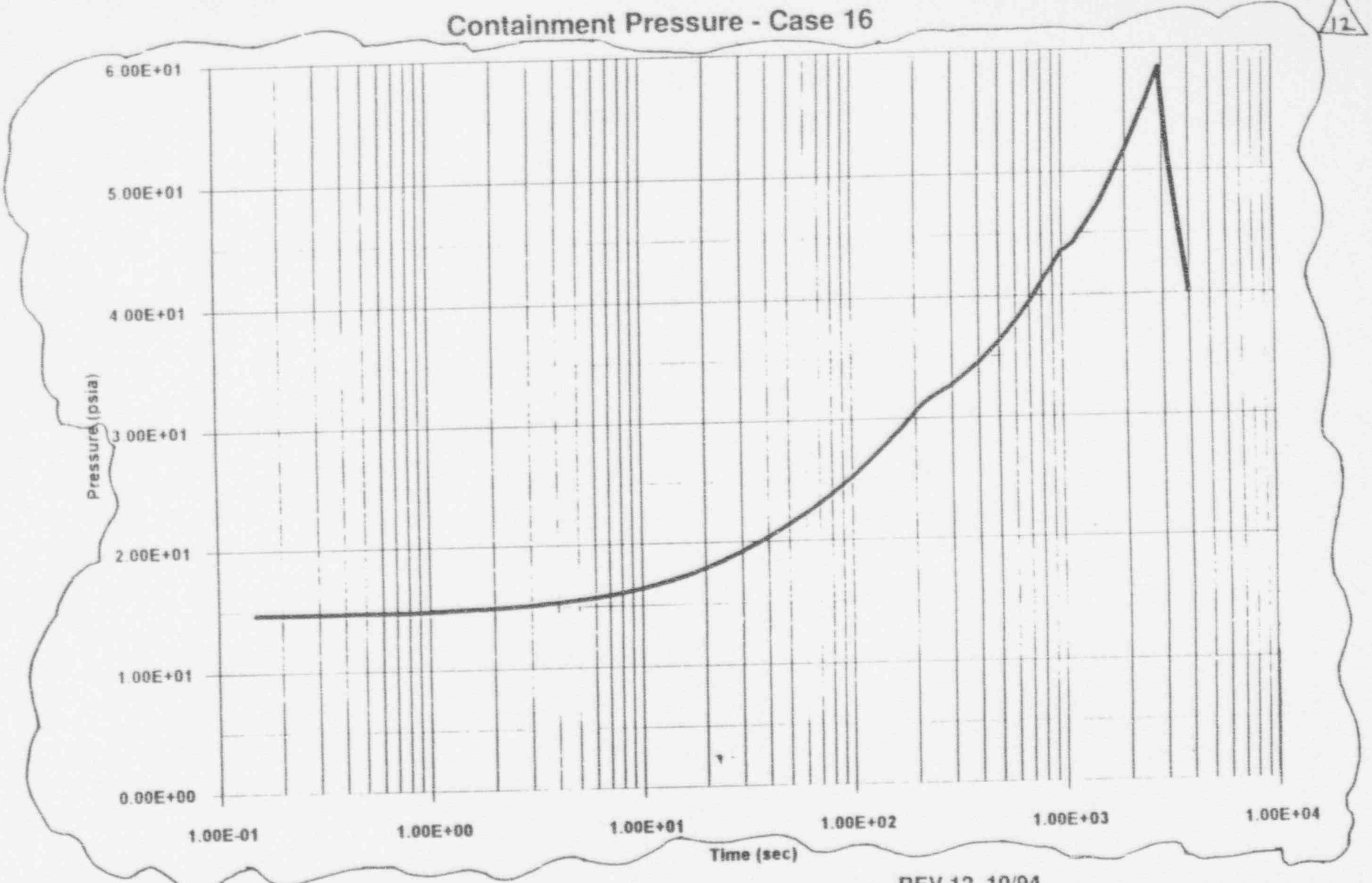
REV 12 10/94

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRESSURE VERSUS TIME STEAM LINE 0.1 ft^2 D.E. BREAK HOT STANDBY
	FIGURE 6.2-34



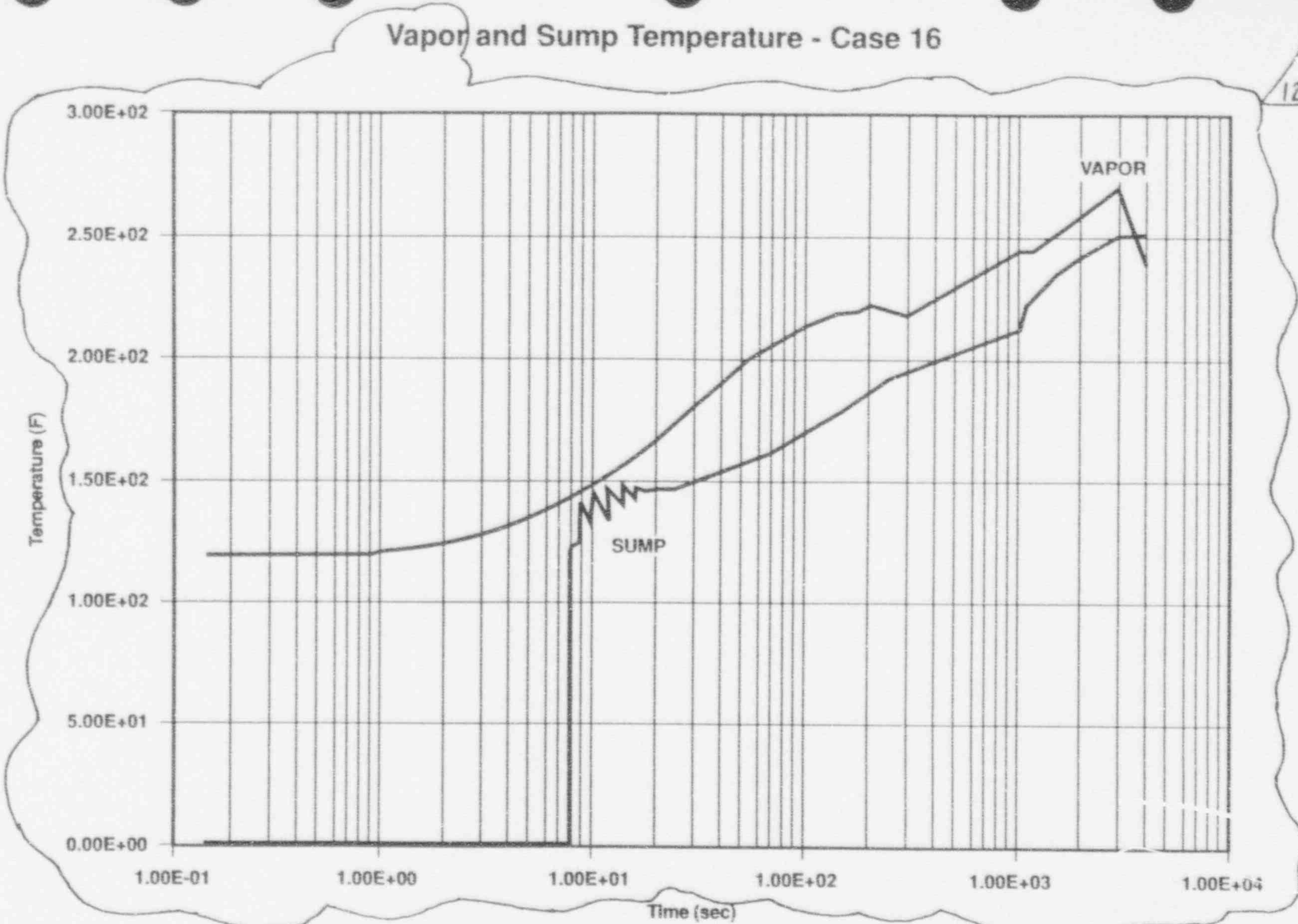
REV 12 10/94

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	TEMPERATURE VERSUS TIME STEAM LINE 0.1 ft^2 D.E. BREAK HOT STANDBY
	FIGURE 6.2-35



REV 12 10/94

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRESSURE VERSUS TIME STEAM LINE 0.30 ft^2 SPLIT HOT STANDBY
	FIGURE 6.2-36



REV 12 10/94

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	TEMPERATURE VERSUS TIME STEAM LINE 0.30 ft^2 SPLIT HOT STANDBY
	FIGURE 6.2-37

THIS FIGURE HAS BEEN DELETED.

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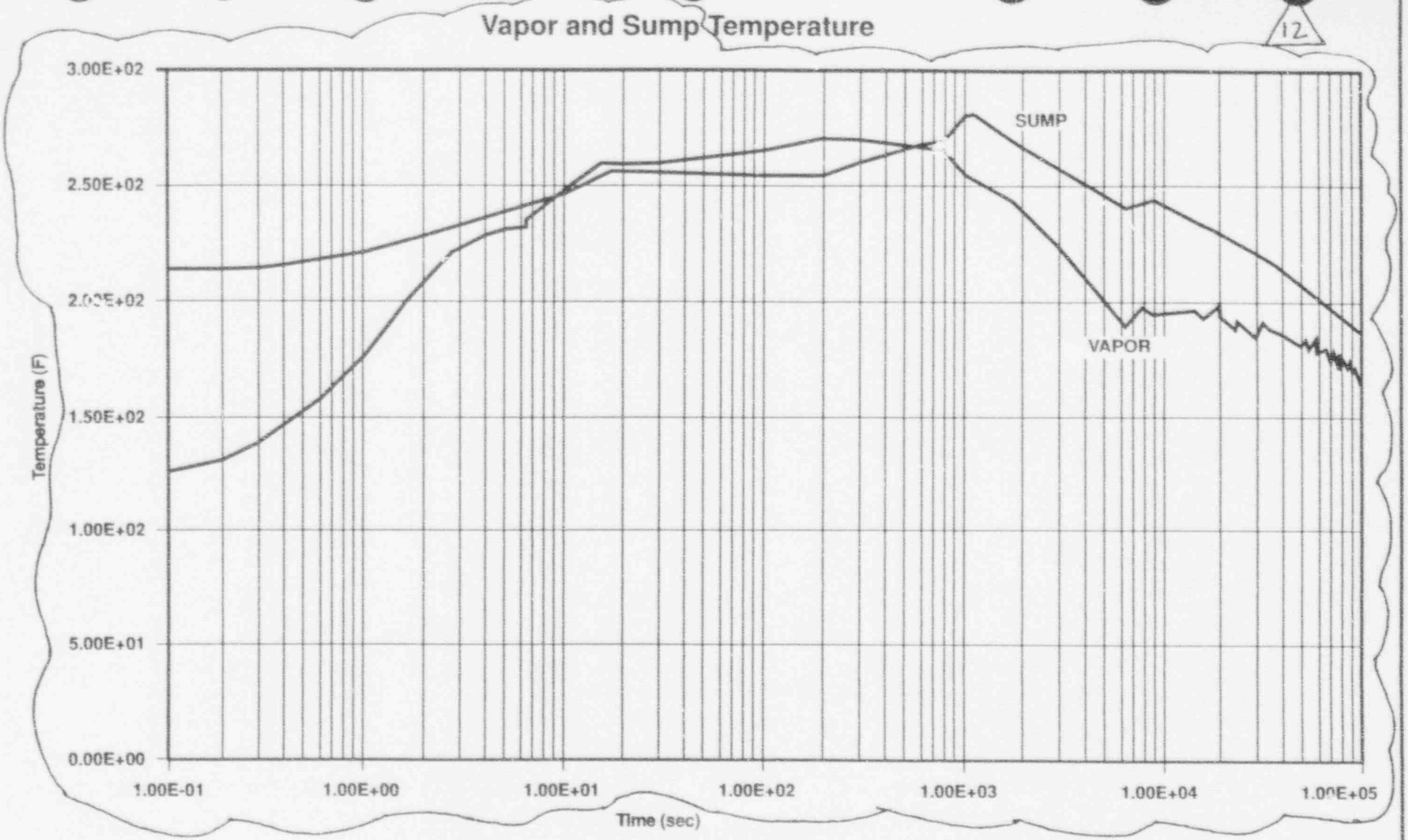
REV 12 10/94

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	DEPSGB MINIMUM ESF 1 AC P/T ANALYSIS LONG-TERM CONTAINMENT PRESSURE vs.TIME
	FIGURE 6.2-39



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JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	DEPSGB MINIMUM ESF DBA TEMPERATURE vs. TIME
	FIGURE 6.2-40

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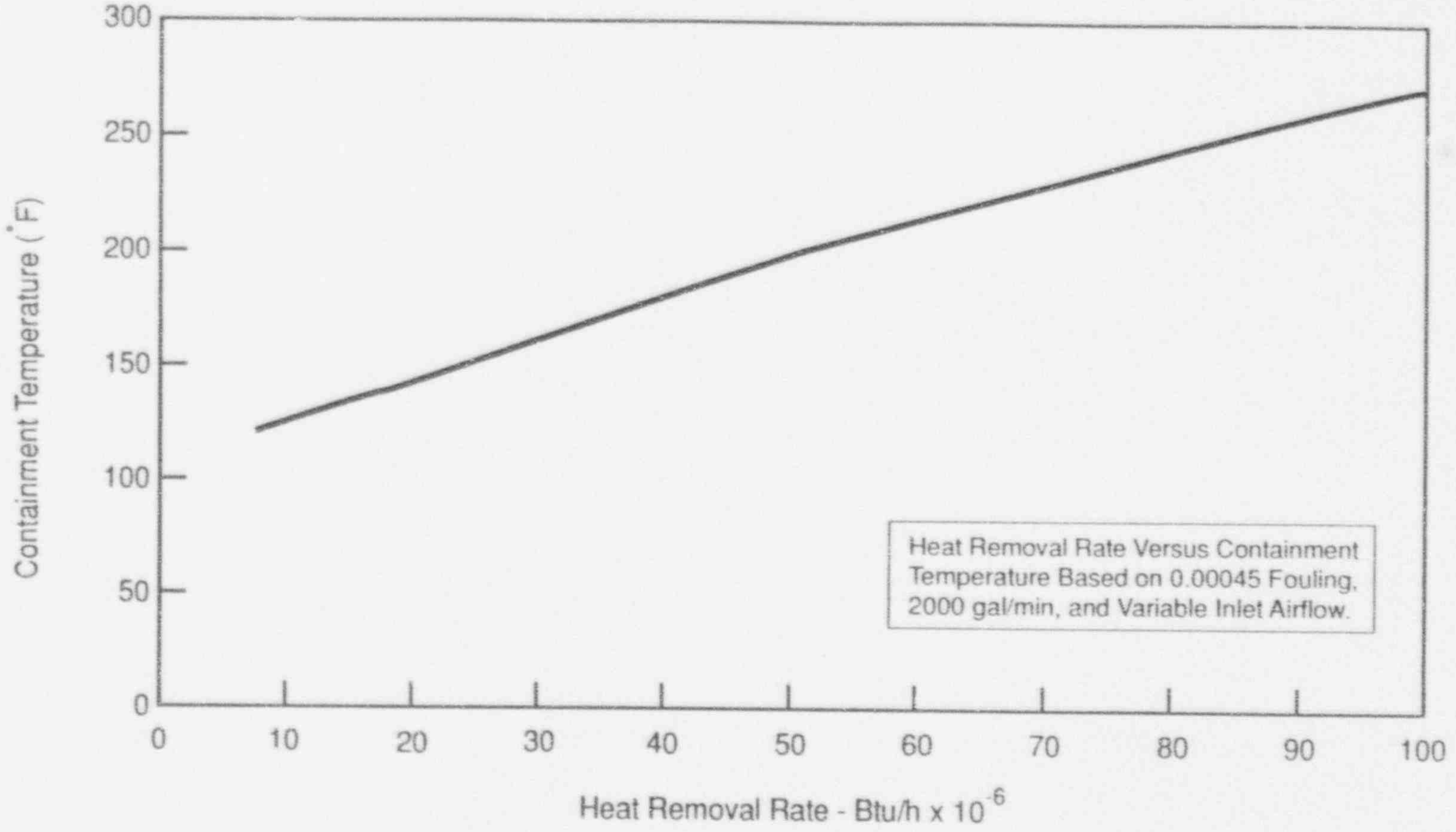
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

DEPSG MIN DBA SHORT TERM
CONTAINMENT TEMPERATURE

FIGURE 6.2-41



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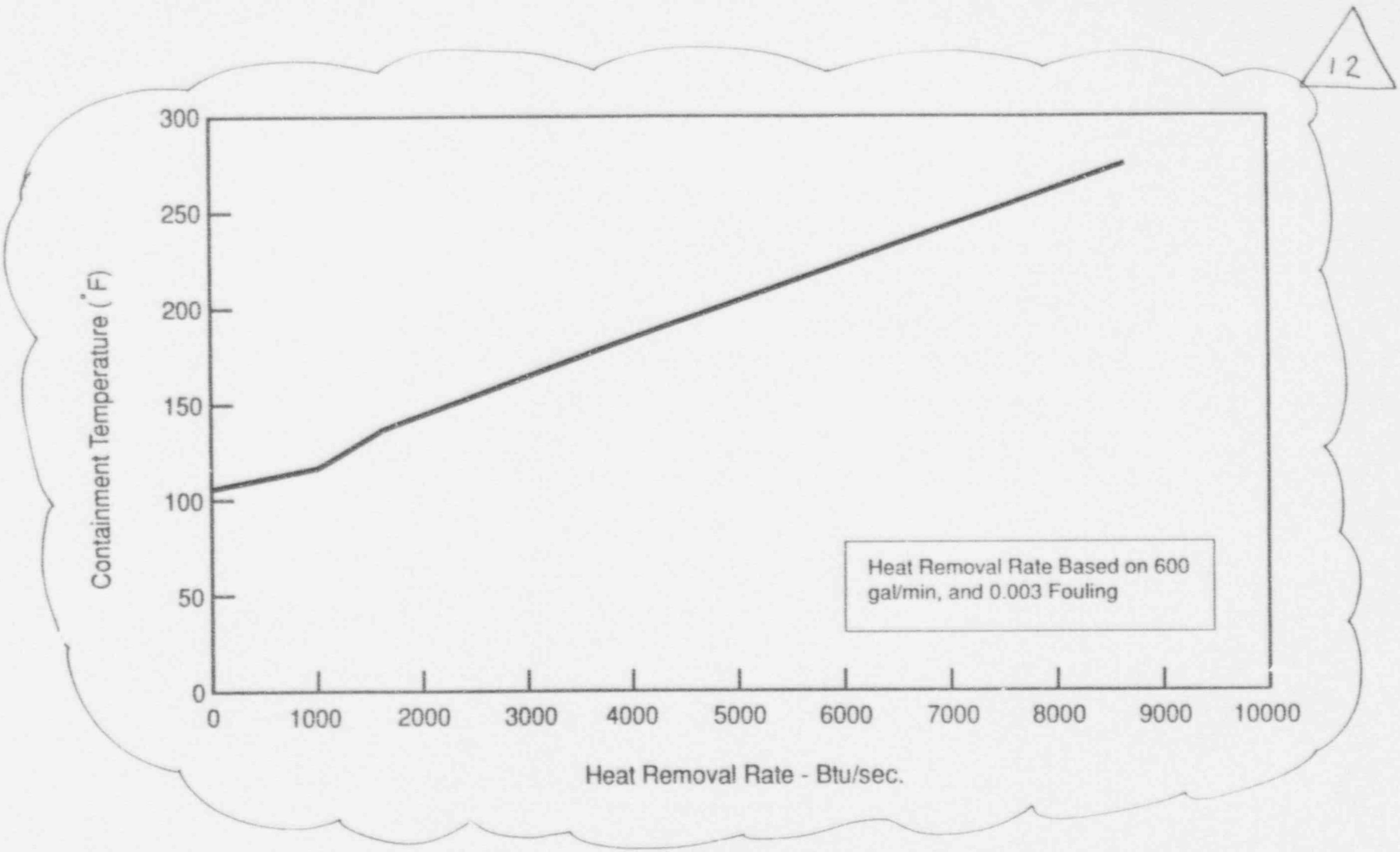
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CONTAINMENT AIR COOLER DUTY VS. TEMPERATURE

FIGURE 6.2-42 (SHEET 1 OF 2)



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JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CONTAINMENT AIR COOLER DUTY
VS. TEMPERATURE

FIGURE 6.2-42 (SHEET 2 OF 2)

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THIS FIGURE HAS BEEN DELETED.

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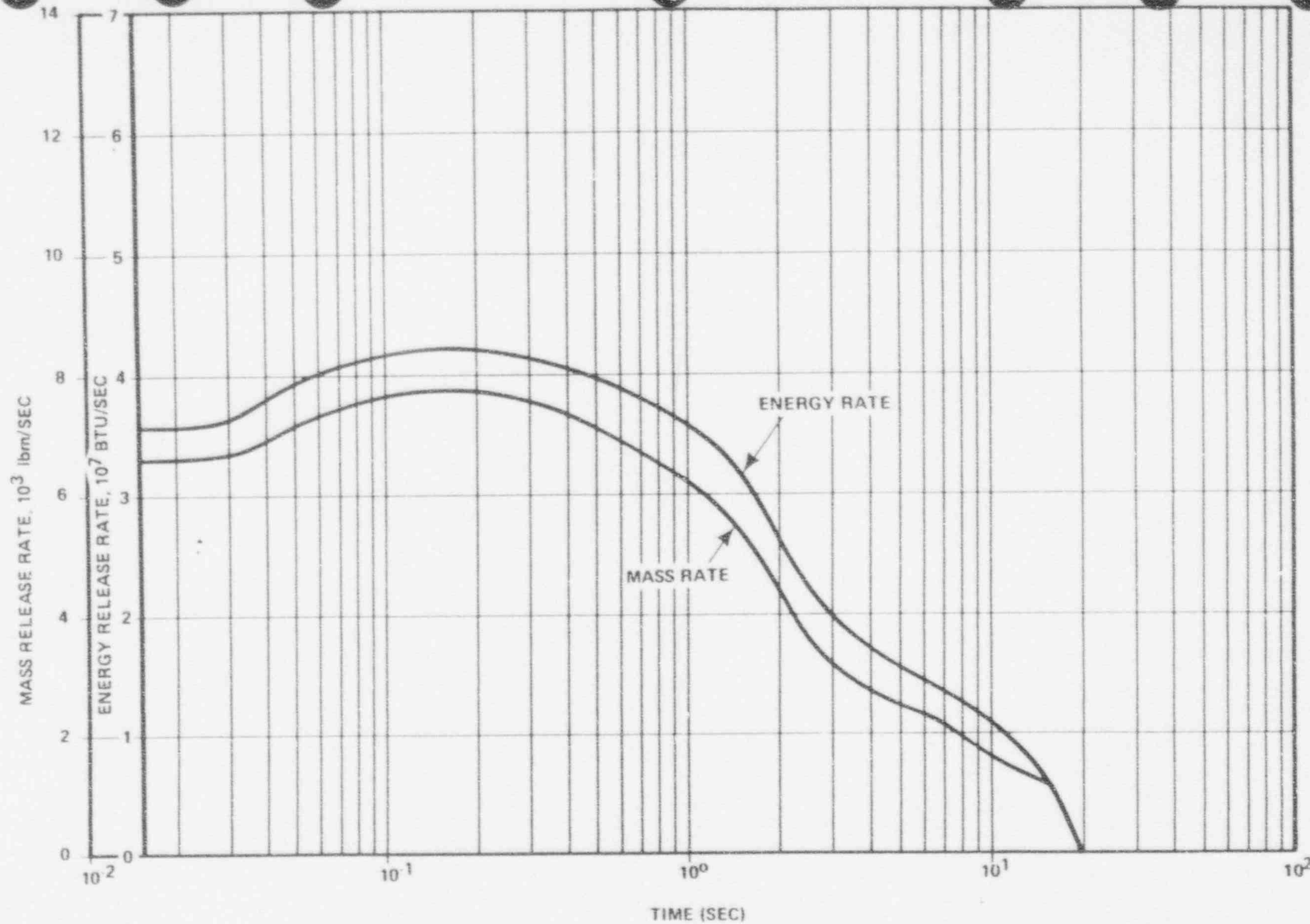
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

THERMAL HEAT REMOVAL EFFICIENCY
OF CONTAINMENT ATMOSPHERE SPRAY

FIGURE 6.2-43



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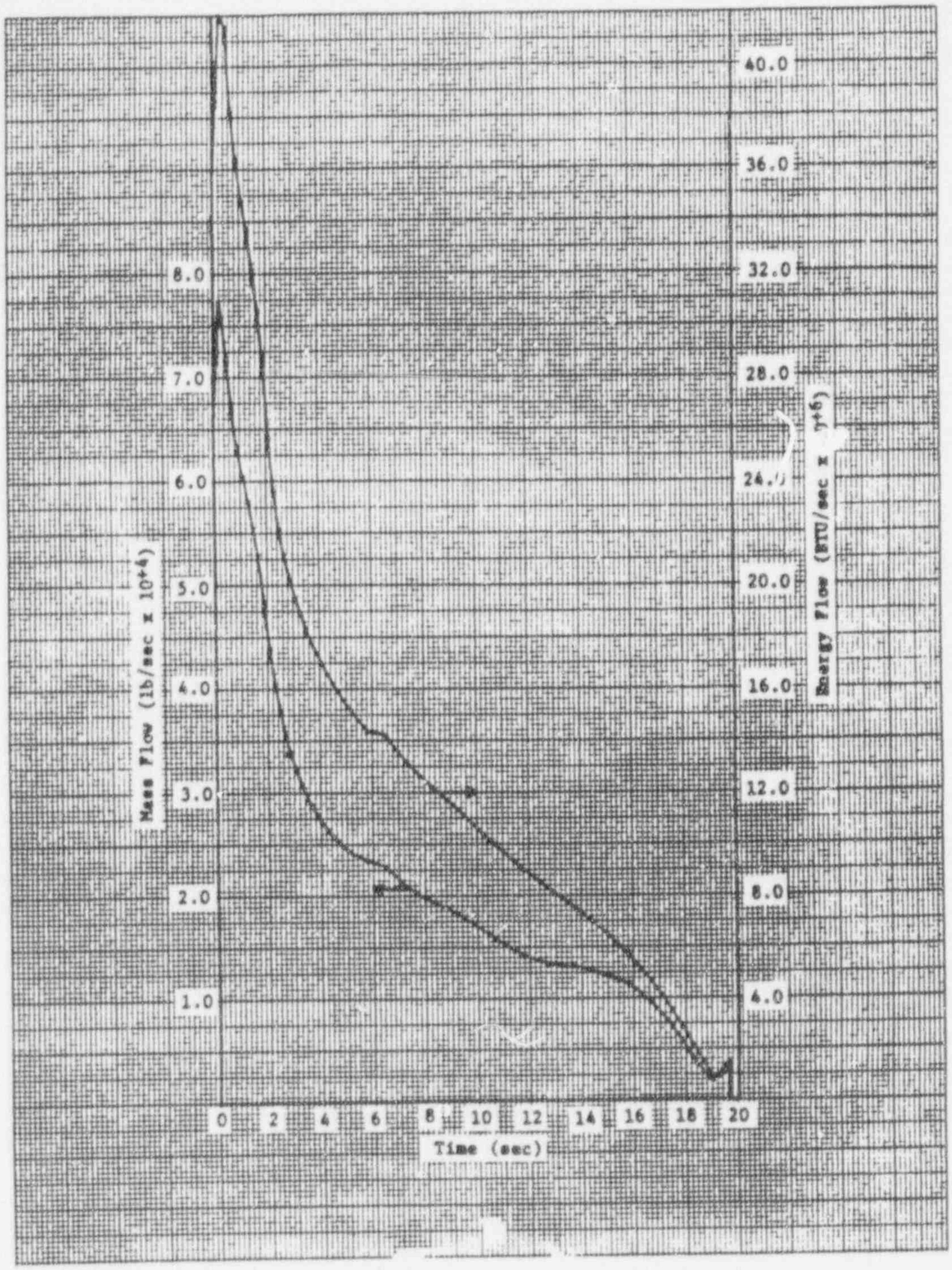
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

MASS & ENERGY RATE VS. TIME FOR LOCA

FIGURE 6.2-45



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NUCLEAR PLANT
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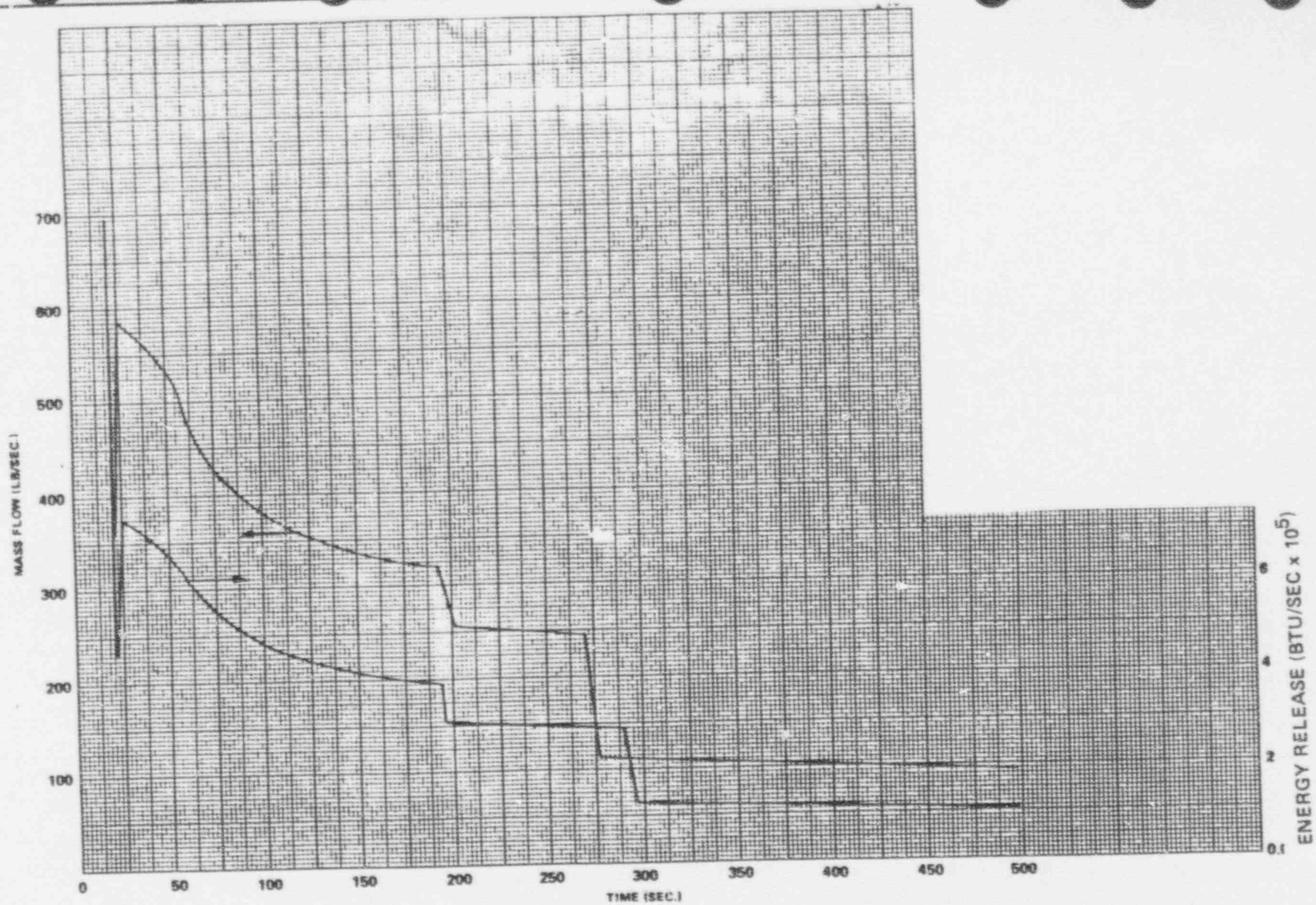
12 LOCA BLOWDOWN MASS AND ENERGY RELEASE RATES VS. TIME

FIGURE 6.2-46



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JOSEPH M. FARLEY
NUCLEAR PLANT
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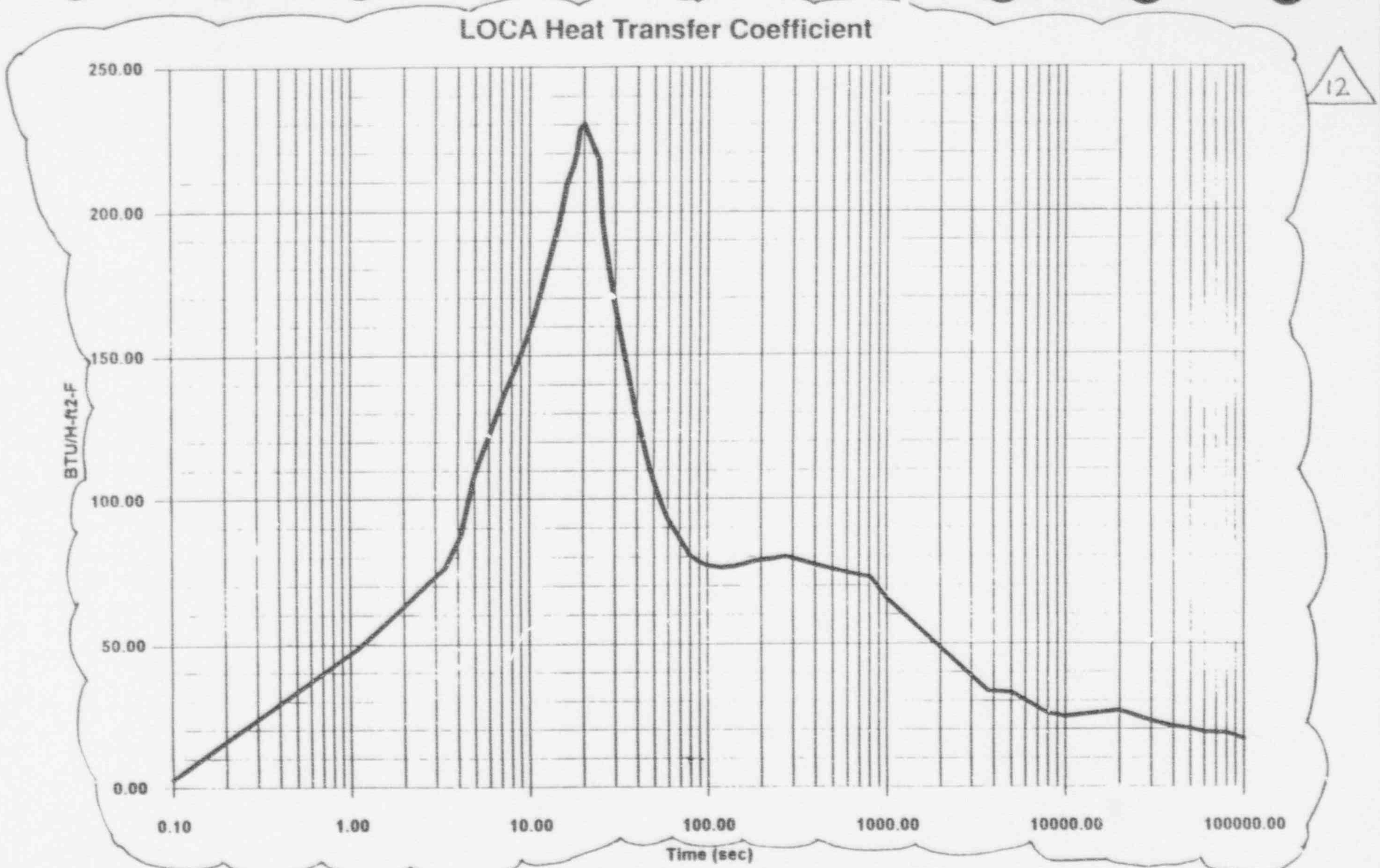
LOCA 72 POST-BLOWDOWN MASS AND ENERGY RELEASE RATES VS. TIME

FIGURE 6.2-47



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JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	DEPSG MIN ESF 1 AC P/T ANALYSIS LONG-TERM CONDENSING HEAT TRANSFER COEFFICIENT	FIGURE 6.2-48

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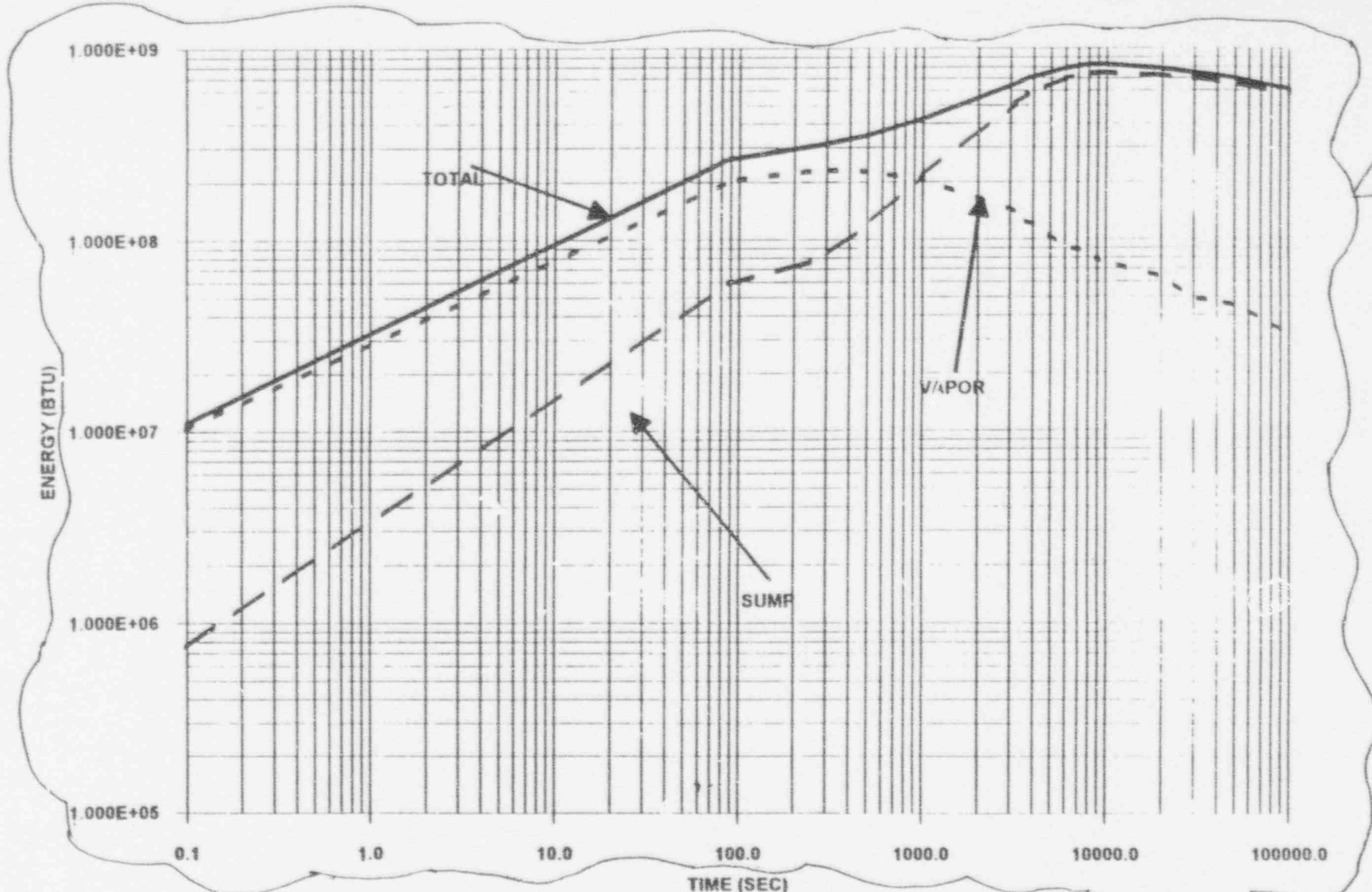
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

SHORT TERM CONDENSING HEAT TRANSFER COEFFICIENT FOR DBA

FIGURE 6.2-49



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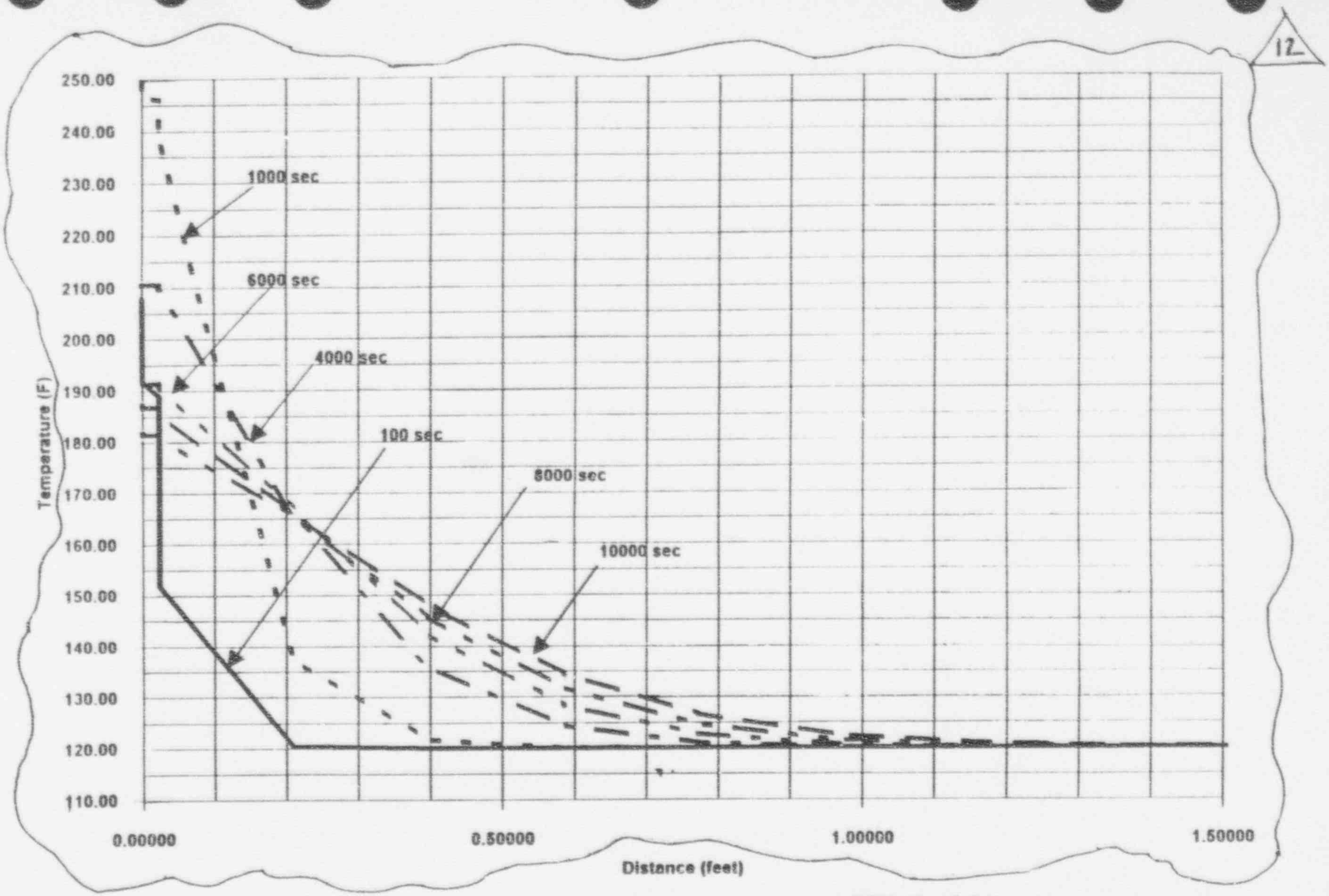
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ENERGY DISTRIBUTION
VS. TIME

FIGURE 6.2-82



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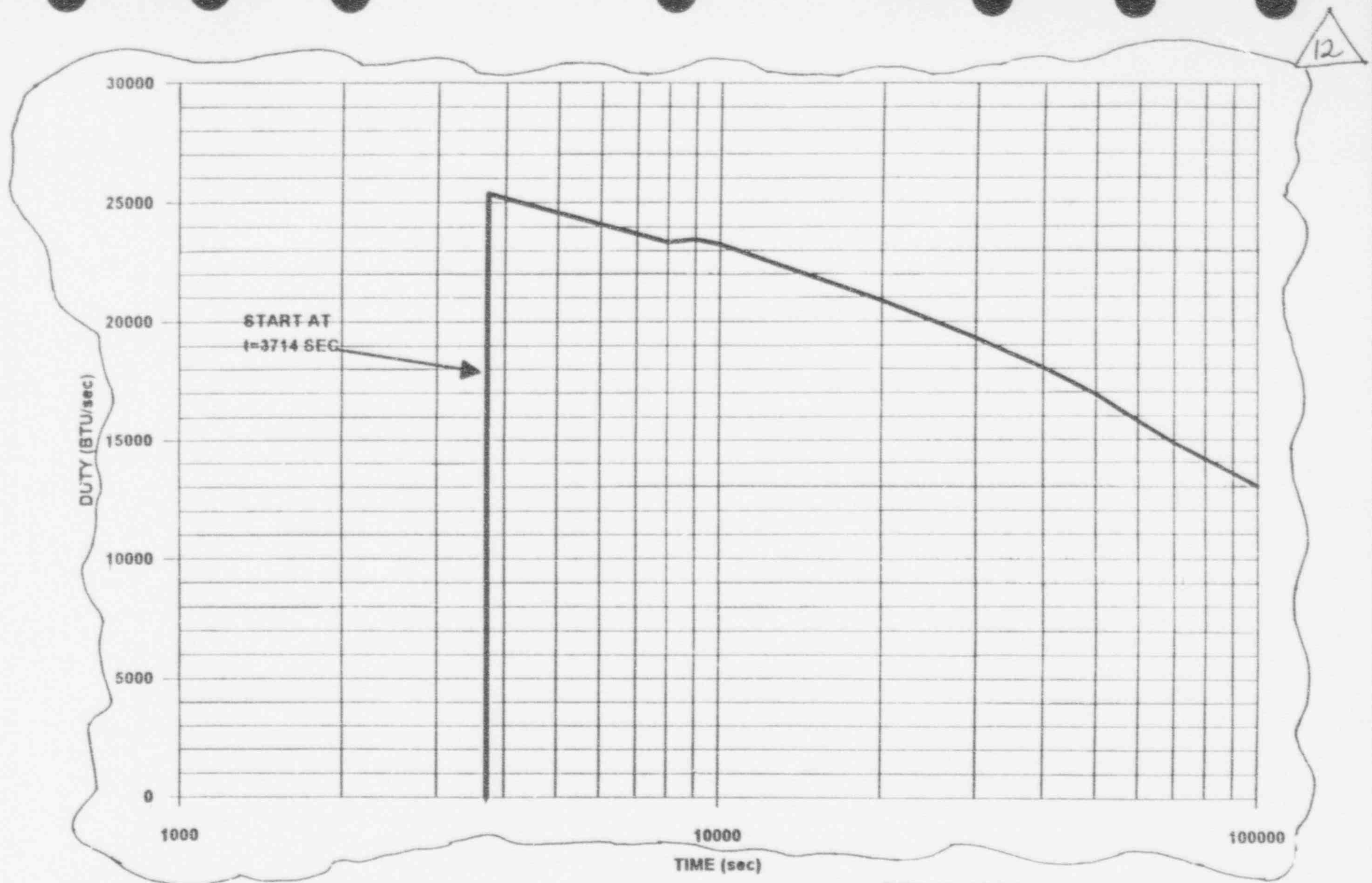
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	TEMPERATURE PROFILE THROUGH CONTAINMENT WALL
	FIGURE 6.2-83



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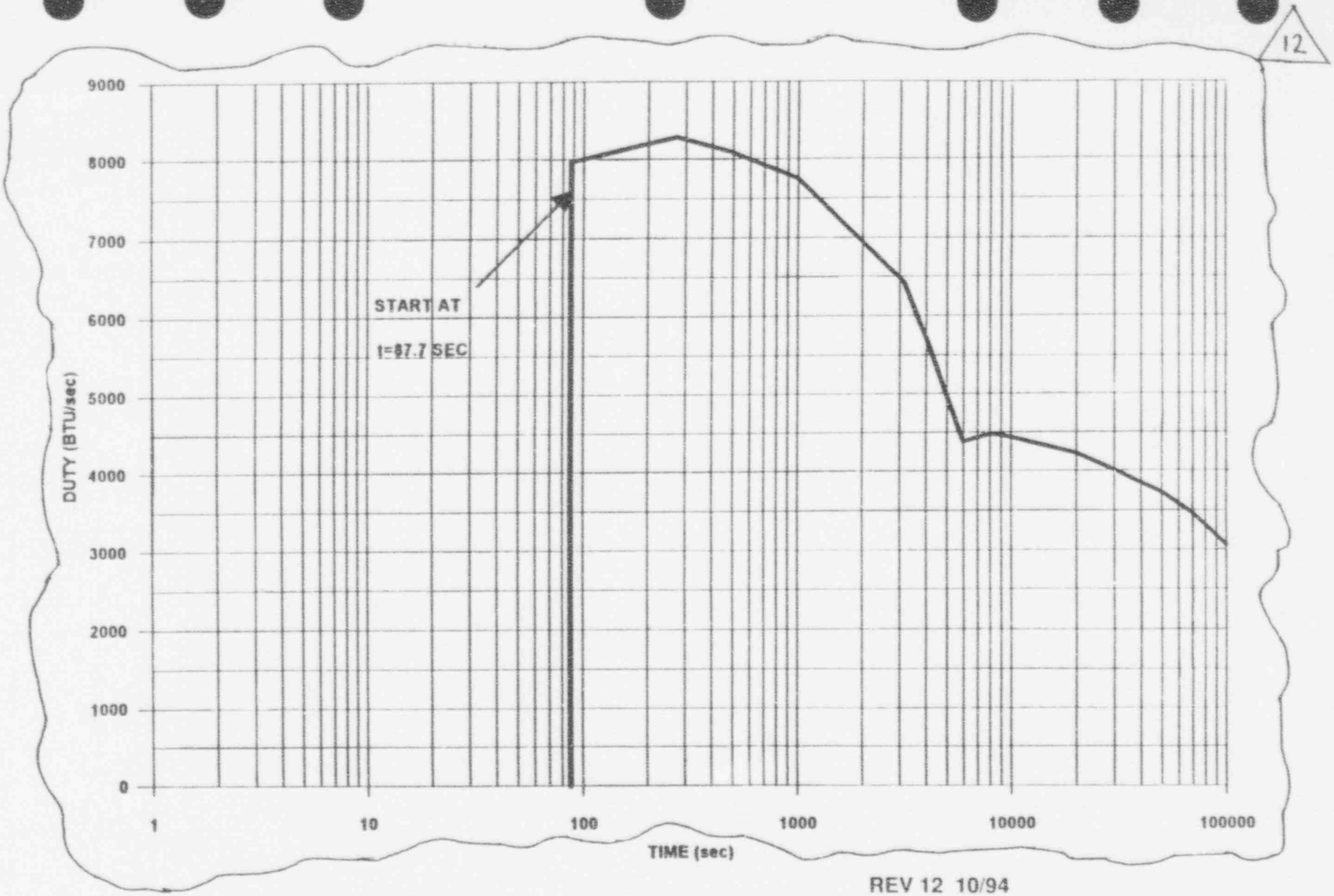
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	RHR HX DUTY VS. TIME	
	FIGURE 6.2-84	



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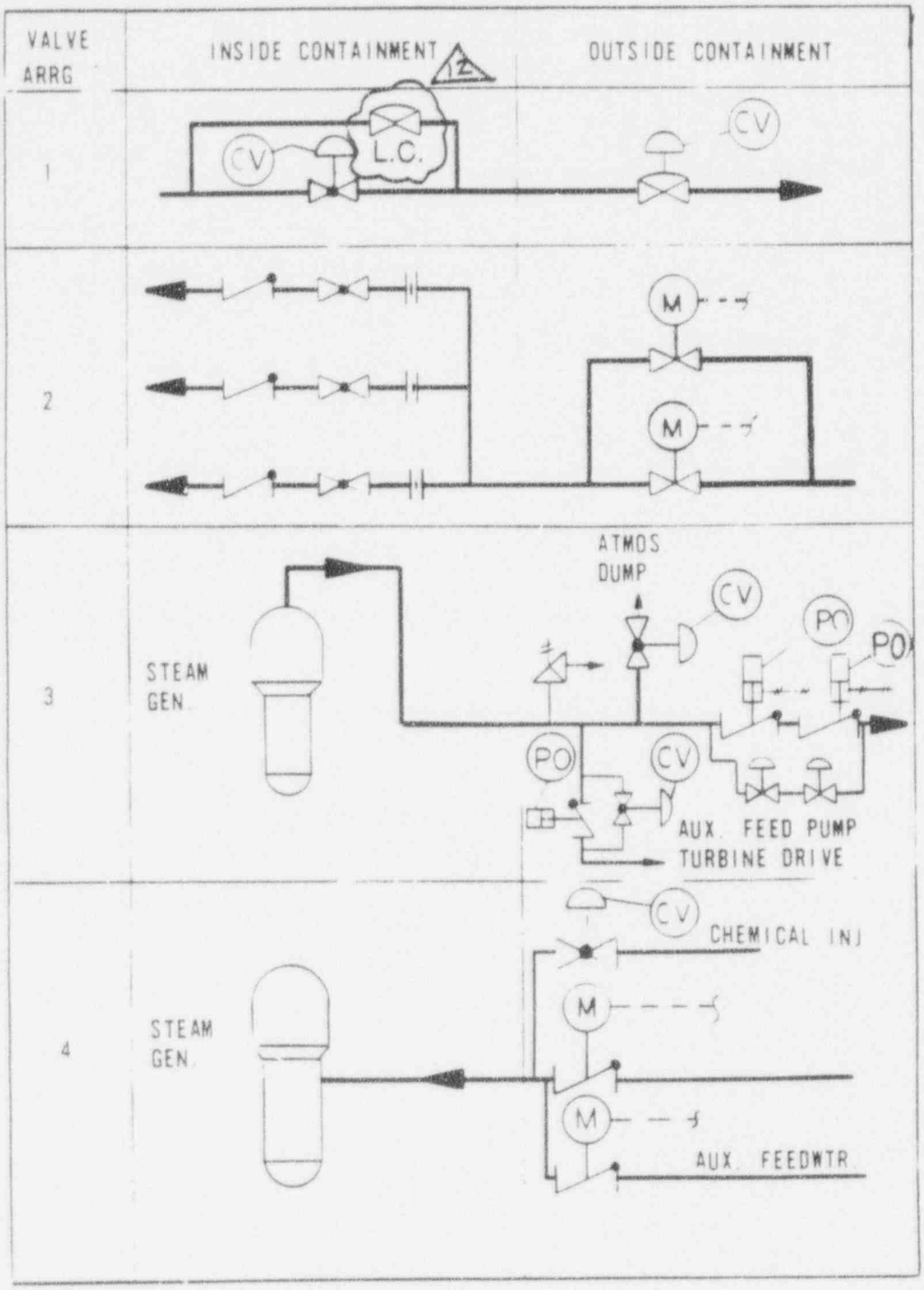
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CONTAINMENT AIR COOLER DUTY
VS. TIME

FIGURE 6.2-85



REV 12 10/94

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ISOLATION VALVE ARRANGEMENT

FIGURE 6.2-95

12

THIS FIGURE HAS BEEN DELETED.

REV 12 10/94

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

DESPG MINIMUM, DBA SHORT TERM PRESSURE VS. TIME

FIGURE 6.2-113

6.4 HABITABILITY SYSTEMS

The control room habitability systems are designed to provide maximum safety and comfort for operating personnel during normal operations and during postulated accident conditions. These habitability systems for the control room include shielding, charcoal filter systems, heating, ventilation and air conditioning, storage capacity of food and water, kitchen, sanitary facilities, and fire protection.

The control room habitability systems are designed to meet NRC acceptance criteria contained in General Design Criterion 19, which is discussed in subsection 3.1.19. Sufficient shielding and ventilation are provided to permit occupancy of the control room for a period of 30 days following a design basis accident (DBA) without receiving more than 5 rem whole body dose or its equivalent to any part of the body. Figure 12.1-1 shows the layout of the control room and its location with respect to the rest of the plant.

The problem of control room uninhabitability is discussed in chapter 15.

6.4.1 HABITABILITY SYSTEMS FUNCTIONAL DESIGN

6.4.1.1 Design Bases

The following design bases were used to determine the functional design of the habitability system:

A. The postulated accident that determines the habitability design requirements is the design basis LOCA. Postulated accidents are discussed in chapter 15.0.

B. The assumptions regarding the sources and amounts of radioactivity that could pose a hazard to the control room are discussed in subsection 12.1.3.

C. In the event of an accident, the ventilation system in the control room will be triggered by the containment isolation actuation system (CIAS) signal which automatically isolates the normal air systems and starts both trains of the control room ac system, pressurization system, and filteration system. The outside air used to pressurize the control room is filtered through a HEPA charcoal filter system which is capable of removing 99.9 percent of both the inorganic and organic iodine. An efficiency of 95 percent is allowed for recirculation.

The filter system and the control room shielding are capable of keeping the dose to the operators less than 5 rem whole body dose or its equivalent to any part of the body for the duration of the accident. The control room shielding is discussed in section 12.1 and the control room air conditioning, heating, cooling, and ventilation systems are discussed in subsection 9.4.1.

C. Following postulated accidents, the most severe being a LOCA, the limitations on control room pressure, temperature, radioactivity concentrations, and doses are as follows:

Parameter	Allowable
Control Room Pressure	> 1/8-in. WG
Control Room Temperature	≤ 120°F
Radioactivity Concentrations	As stated in Column 1, Table I, Appendix B to 10 CFR 20.1-20.601
Doses	5 rem

D. The fire protection system in the control room consists of an early warning ionization type detection system with hand portable H_2O extinguishers located in the control room itself. The ionization detectors used will rapidly detect products of combustion. Detectors are located on the false ceiling to detect smoke in the control room itself. Another set of detectors is located in the space above the false ceiling for detection in this area. Each detector is equipped with a light to indicate which detector has operated. All detectors will operate the visible and audible alarm on the main fire protection annunciator panel located in the control room. In addition, two fixed carbon dioxide hose reels with 100-ft hose are located in the immediate vicinity outside the control room and can be used to back up the hand extinguishers.

Noncombustible materials are used in construction and equipment as much as possible. The quantity of combustible material such as paper and other flammable supplies is kept to a minimum. A person trained in fire fighting is on duty in the control room at all times.

Control room doors are kept closed except when in use. However, an evaluation has been performed which justifies leaving the door between the control room and the technical support center (door 453) or the door between the control room and the RACA

6.4.1.2 System Design

6.4.1.2.1 Piping and Instrumentation Diagrams

The piping and instrumentation diagram of the control room ventilation and cleanup system is provided in figure 9.4-1.

6.4.1.2.2 Performance Objectives

The performance objectives to be maintained are as follows:

Air condition flowrate	21,000 ft^3/min
HEPA and charcoal filter unit flowrate	3,000 ft^3/min
Pressure	>1/8 in. WG
Temperature	74°F
Humidity	50%
Limits of radioactivity	$\leq$ values stated in Table 1 of Appendix B to 10 CFR 20.1001-20.2401.

6.4.1.2.3 Provisions to Intake, Exhaust, Monitor, and Filter

A. Intake, Exhaust, and Monitoring

During normal plant operation one of the two 100 percent capacity air conditioning packaged units recirculates 21,000 ft^3/min of cooled filtered air to the control room. One thousand ft^3/min of fresh outdoor air is supplied to the control room through a supply duct from the computer room air conditioning unit. A smoke detector near the return air duct to each recirculation fan will sound an alarm in the control room on high smoke level. If necessary, the operator can exhaust air from the control room by manually opening the pneumatic operated exhaust damper and starting one of the two 100-percent capacity exhaust fans.

The CIAS signal will automatically switch the control room ventilation system to emergency pressurization and activate the charcoal filter system. In addition, the radiation monitoring system in the control room will detect a high radiation level in the control room and alert the operator to switch to the recirculation mode with the charcoal filter system, to thus reduce the radiation level.

B. Control Room Charcoal Filter System

The control room filtration system is designed to minimize the activity level in the control room resulting from high airborne radiation. In addition, the filtration system along with the exhaust system minimizes the hazards from any noxious gases. The control room filtration system consists of two parallel fully redundant full capacity fan and filter systems. The system is not used during normal operations, but is to be tested periodically. The tests consist of fan operation to measure filter pressure drops and radioactive testing of charcoal samples at 18 month intervals to ensure efficiency. The fan and filter system are seismic Category I and are designed to withstand, without exceeding the yield stesses and without loss of function, the forces resulting from the safe shutdown earthquake (SSE).

Description of the Charcoal Filter System

A. Charcoal Type

New, commercially pure, activated coconut shell, impregnated 5 percent by weight with iodine compounds. The granule size is 8-16 mesh.

B. Charcoal Weight, Tray Type Units

A minimum of 43 lb of charcoal will fill each unit (2 elements per unit) based on charcoal with a moisture content no greater than 3 percent. There are 3 units in each tray type charcoal filter system.

C. Charcoal Bed Configuration

1. Tray Type

The charcoal filter unit is standard manufactured size, having a frontal face size of 8 in. by 24 in. and having two horizontal, flat charcoal beds, each approximately 24 in. by 28 in. in depth, arranged in parallel fashion with an air space between beds.

2. Bed Type

The bed type filters are of the high efficiency carbon adsorber (HECA) type. The 2-inch (recirculation filter) and 6-inch (pressurization filter) layers of carbon are in modules of

7.0 INSTRUMENTATION AND CONTROL

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voltage goes below approximately 70 percent of rated voltage. Signals from these relays are time delayed (as shown in figure 7.2-5.) to prevent spurious trips caused by short-term voltage perturbations. The coincidence logic and interlocks are given in table 7.2-1.

4. Reactor Coolant Pump Bus Underfrequency Trip

   This trip is required to protect against low flow resulting from bus underfrequency, for example, a major power grid frequency disturbance.

   Two underfrequency sensing relays are connected to each of the three reactor coolant pump buses. Signals from relays connected to any two of the buses (time delayed up to approximately 0.1 s to prevent spurious trips caused by short-term frequency perturbations) will trip the reactor if the power level is above P-7.

   Figure 7.2-5 shows the logic for the reactor coolant system low flow trips.

E. Steam Generator Trip

The specific trip function generated is as follows:

Low-Low Steam Generator Water Level Trip

This trip protects the reactor from loss of heat sink in the event of a loss of feedwater to one or more steam generators or a major feedwater line rupture. This trip is actuated on two-out-of-three low-low water level signals occurring in any steam generator. In addition, the motor-driven AFW pumps receive an automatic start signal. If a low-low water level condition is detected in two or more steam generators, a signal is generated to start the turbine-driven auxiliary feedwater pump as well.

The logic is shown in figure 7.2-7. A detailed functional description of the process equipment associated with this trip is provided in reference 1.

F. Turbine Trip/Reactor Trip

The turbine trip/reactor trip is actuated by two-out-of-three logic from low auto stop oil pressure signals or by all closed signals from the turbine steam stop valves. A turbine trip causes a direct reactor trip above P-9. High-high steam generator water level signals in two out of three channels for any steam generator will actuate a turbine trip, trip the main feedwater pumps, and close the main and bypass feedwater control valves. The purpose is to protect the turbine and steam piping from excessive moisture carryover caused by the high-high steam generator water level. Other turbine trips are discussed in chapter 10.

The logic for this trip is shown in figure 7.2-7.

The analog portion of the trip shown in figure 7.2-15 is represented by dashed (----) lines. When the turbine is tripped, turbine auto stop oil pressure drops and the pressure is sensed by three pressure sensors. A digital output is provided from each sensor when the oil pressure drops below a preset level. These three outputs are transmitted to two redundant two-out-of-three logic matrices, either of which trips the reactor if above P-9.

The auto stop oil pressure signal also dumps the stop emergency trip fluid, closing all of the turbine steam stop valves. When all stop valves are closed, a reactor trip signal will be initiated if the reactor is above P-9. This trip signal is generated by redundant limit switches on the stop valves.

G. Safety Injection Signal Actuation Trip

A reactor trip occurs when the safety injection system is actuated. The means of actuating the safety injection system are described in section 7.3. This trip protects the core in the event of a loss-of-coolant accident.

F. Wind and tornadoes (section 3.3).

The performance requirements are as follows:

A. System Response Times

The total delay to trip is defined as the time delay from the time that a step change in the variable being monitored from 5 percent below to 5 percent above the trip setpoint is reached to the time that the rods are free and begin to fall. This is with transfer functions set to one. During preliminary startup tests, it will be demonstrated that actual instrument errors and time delays are equal to or less than the values assumed in the accident analyses.

Maximum allowable time delays in generating the reactor trip signal are as follows:

	Time (s)
1. Power range high nuclear power (high and low setpoint)	0.5
2. Neutron flux rates (positive and negative)	0.5
3. Maximum overtemperature ΔT (OTΔT)	6.0
4. Maximum overpower ΔT (OPΔT)	6.0
5. Pressurizer pressure (low and high)	2.0
6. Pressurizer high water level	2.0
7. Low reactor coolant flow	1.0
8. Reactor coolant pump bus underfrequency	0.6
9. Reactor coolant pump bus undervoltage	1.2

		Time (s)
11.	Low-low steam generator water level	2.0
12.	Turbine trip	1.0

B. Reactor trip accuracies are given in table 7.2-3.

C. Protection system ranges are as follows:

		Range
1.	Power range nuclear power	1- to 120-percent full power
2.	Neutron flux rates (positive and negative)	±5 percent to ±30 percent of full power
3.	Overtemperature ΔT:	
	T_{hot} leg	530°F to 650°F
	T_{cold} leg	510°F to 610°F
	T_{avg}	530°F to 630°F
	Pressurizer pressure	1700 to 2500 psig
	f(Δϕ)	-50 to +50 percent
	ΔT setpoint	0°F to 100°F
4.	Overpower ΔT	(See overtemperature ΔT)
5.	Pressurizer pressure	1700 to 2500 psig
6.	Pressurizer water level	Entire cylindrical portion of pressurizer
7.	Reactor coolant flow	0- to 120-percent of rated flow
8.	Reactor coolant pump bus underfrequency	50 to 65 Hz
9.	Reactor coolant pump bus undervoltage	0- to 100-percent rated voltage

input signal occurred. This design meets the requirements of GDC 25.

Where a single failure of a protection system component can cause a process perturbation which requires protective action, the protection system has been designed to withstand a second failure, without loss of protective action. This feature is normally achieved by providing two-out-of-four trip logic for each of the protective functions with the exception of steam generator protection, which relies on two-out-of-three trip logic and a control grade median signal selector (MSS). The use of a control grade MSS will prevent any protection system failure from causing a control system action resulting in a need for subsequent protective action. This design meets the requirements of GDC 25.

Redundant control signal cables leaving the protection racks through isolation devices can come into close proximity elsewhere in the plant, such as the control board. It could be postulated that electrical faults or interference at these locations might be propagated into all redundant racks and degrade protection circuits because of the close proximity of protective and control wiring within each rack. Regulatory Guide 1.75, position C.4, and IEEE 384-1974, section 4.5(3), provide the option to demonstrate by tests that the absence of physical separation could not significantly reduce the availability of Class 1E circuits.

Therefore, Westinghouse conducted tests to demonstrate that Class 1E protection systems (nuclear instrumentation system, solid-state protection system, and 7300 process control system) could not be degraded by non-Class 1E circuits sharing the same enclosure. Conformance to the requirements of IEEE 279-1971 and Regulatory Guide 1.75 was established and accepted by the Nuclear Regulatory Commission (NRC); these requirements are applicable to these systems at the FNP units.

Tests conducted on the as-built designs of the nuclear instrumentation system and solid-state protection system were reported and accepted by the NRC in support of the Diablo Canyon application (dockets 51-275 and 50-323). Westinghouse considers these programs as applicable to all plants, including FNP. Westinghouse tests on the 7300 series process control system were covered in a report entitled,

"Westinghouse 7300 Series Process Control System Noise Tests," subsequently reissued as reference 9. In a letter dated April 20, 1977,[10] the NRC accepted the report in which the applicability of the FNP units is established (appendix G in reference 10).

F. Capability for Testing

The reactor trip system is capable of being tested during power operation. Where only parts of the system are tested at any one time, the testing sequence provides the necessary overlap between the parts to ensure complete system operation.

The protection system is designed to permit periodic testing of the analog channel portion of the reactor

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trip system during reactor power operation without initiating a protective action unless a trip condition actually exists. This is because of the coincidence logic required for reactor trip. Note, however, that the source and intermediate range high neutron flux trips must be bypassed during testing.

The operability of the process sensors is ascertained by comparison with redundant channels monitoring the same process variables or those with a fixed known relationship to the parameter being checked. The process sensors within containment can be calibrated during plant shutdown.

Analog channel testing is performed at the analog instrumentation rack set by individually introducing simulated input signals into the instrumentation channels and observing the tripping of the appropriate output bistables. Process analog output to the logic circuitry is interrupted during individual channel test by a test switch which, when thrown, deenergizes the associated logic input and inserts a proving lamp in the bistable output. Interruption of the bistable output to the logic circuitry for any cause (test, maintenance purposes, or removed from service) will cause that portion of the logic to be actuated (partial trip), accompanied by a partial trip alarm and channel status light actuation in the control room. Each channel contains those switches, test points, etc., necessary to test the channel. See reference 1 for additional information.

The power range channels of the nuclear instrumentation system are tested by superimposing a test signal on the actual detector signal being received by the channel at the time of testing. The output of the bistable is not placed in a tripped condition prior to testing. Also, since the power range channel logic is two out of four, bypass of this reactor trip function is not required.

To test a power range channel, a test-operate switch is provided to require deliberate operator action. Operation of the switch will initiate the channel test annunciator in the control room. Bistable operation is tested by increasing the test signal level up to its trip setpoint and verifying bistable relay operation by control board annunciator and trip status lights.

The reference leg is uninsulated and will remain at local ambient temperature. This temperature will vary somewhat over the length of the reference leg piping under normal operating conditions but will not exceed 140°F. During a blowdown accident, any reference leg water flashing to steam will be confined to the condensate steam interface in the condensate pot at the top of the temperature barrier leg and will have only a small (about 1 in.) effect on the measured level. Some additional error may be expected due to effervescence of hydrogen in the temperature barrier water. However, even if complete loss of this water is assumed, the error will be less than 1 ft and can be tolerated.

The sealed reference leg design has been installed in various plants since early 1970, and operational accuracy was verified by use of the sealed reference leg system in parallel with an open reference leg channel. No effects of operating pressure variations on either the accuracy or integrity of the channel have been observed.

Calibration of the sealed reference leg system is done in place after installation by application of known pressure to the low pressure side of the transmitter and by measurement of the height of the reference column. The effects of static pressure variations are predictable. The largest effect is due to the density change in the saturated fluid in the pressurizer itself. The effect is typical of level measurements in all tanks with two-phase fluid and is not peculiar to the sealed reference leg technique. In the sealed reference leg, there is a slight compression of the fill water with increasing pressure, but this is taken up by the flexible bellows. A leak of the fill water in the sealed reference leg can be detected by comparison of redundant channel readings on line and by physical inspection of the reference leg off line. Leaks of the reference leg to the atmosphere will be immediately detectable by off-scale indications and alarms on the control board. A closed pressurizer level instrument shutoff valve would be detected by comparing the level indications from the redundant level channels (three channels). In addition, there are alarms on one of the three channels to indicate an error between the measured pressurizer water level and the programmed pressurizer water level. There is no single instrument valve which could affect more than one of the three level channels.

The high water level trip setpoint provides sufficient margin such that the undesirable condition of discharging liquid coolant through the safety valves is avoided. Even at full power conditions, which would produce the worst thermal expansion rates, a failure of the water level control would not lead to any liquid discharge through the safety valves. This is due to the automatic high pressurizer pressure reactor trip

actuating at a pressure sufficiently below the safety valve setpoint.

For control failures which tend to empty the pressurizer, two-out-of-three logic for safety injection action on low pressurizer pressure ensures that the protection system can withstand an independent failure in another channel. In addition, ample time and alarms exist to alert the operator of the need for appropriate action.

7.2.2.3.5 Steam Generator Water Level and Feedwater Flow

The basic function of the reactor protection circuit associated with low steam generator water level is to preserve the steam generator heat sink for removal of long-term residual heat. Should a complete loss of feedwater occur, the reactor would be tripped on low-low steam generator water level. In addition, redundant auxiliary feedwater pumps are provided to supply feedwater in order to maintain residual heat removal after trip, preventing eventual thermal expansion and discharge of the reactor coolant through the pressurizer relief valves into the relief tank even when main feedwater pumps are incapacitated. This reactor trip acts before the steam generators are dry to reduce the required capacity and starting time requirements of these auxiliary feedwater pumps and to minimize the thermal transient on the reactor coolant system and steam generators. Therefore, a low-low steam generator water level reactor trip is provided for each steam generator to ensure that sufficient initial thermal capacity is available in the steam generator at the start of the transient. It is desirable to minimize thermal transients on a steam generator for a credible loss of feedwater accident. A protection system failure causing control system action is eliminated by implementation of a control grade MSS function. The prime objective of the MSS is to prevent a single failed protection system instrument channel from causing a perturbation to feedwater control system requiring subsequent protective action. All three narrow range water level channels for each steam generator provide input to the MSS. The MSS selects the median signal for use by the feedwater control system and determines control system action based on this signal. By rejecting the high and low signals, the MSS prevents unwarranted control system action based on a single, failed protection system instrument channel. Since no adverse control system action may now result from a single, failed protection instrument channel, a second random protection system failure (as would otherwise be required by IEEE-279-1971) need not be considered. A more detailed discussion of the MSS and its compliance with control and protection system single interaction criteria can be found in reference 17.

7.2.3 TESTS AND INSPECTIONS

The reactor trip system meets the testing requirements of IEEE 338[13] with the exceptions given in paragraph 7.2.2.2.4. The testability of the system is discussed in paragraph 7.2.2.2.1.4. The initial test intervals are specified in the plant technical specifications.

7.2.3.1 Inservice Tests and Inspections

Periodic surveillance of the reactor trip system is performed to ensure proper protective action. This surveillance consists of checks, calibrations, channel functional testing, and response time testing which are summarized as follows:

A. Checks

A check consists of a qualitative determination of acceptability by observation of channel behavior during operation. It includes comparison of the channel with other independent channels measuring the same variable. Failures such as blown instrument fuses, defective indicators, or faulted amplifiers are noticeable by simple observation of the functioning of the instrument or system. Furthermore, in many cases such failures are revealed by alarm or annunciator action, and a check supplements this type of surveillance.

B. Calibration

A channel calibration consists of adjustment of channel output such that it responds, within acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration encompasses the entire channel including alarm and/or trip; it also includes the channel functional testing discussed below. Thus, the calibration ensures the acquisition and presentation of accurate information.

C. Channel Functional Testing

A channel functional test consists of injecting a simulated signal into the signal conditioning portion of the channel to verify its operability, including alarm and/or trip initiating action.

D. Response Time Testing

Response time testing consists of any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined in the technical specifications. Sensor response time verification may be demonstrated by either 1) inplace, onsite, or offsite test measurements or 2) utilizing replacement sensors with certified response times. The measurement of response time at the specified frequencies provides

assurance that the reactor trip associated with each channel is completed within the time limit assumed in the accident analyses. The response time limits for the reactor trip system are provided in table 7.2-5.

The minimum frequency for checks, calibration, channel functional testing, and response time testing are defined in the plant technical specifications.

7.2.3.2 Periodic Testing of the Nuclear Instrumentation System

The following periodic tests of the nuclear instrumentation system are performed:

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A. Testing at plant shutdown.
   1. Source range testing.
   2. Intermediate range testing.
   3. Power range testing.

B. Testing between P-6 and P-10 permissive power levels.
   1. Source range testing.
   2. Intermediate range testing.
   3. Power range testing.

C. Testing above P-10 permissive power level.
   1. Intermediate range testing.
   2. Power range testing.

Any deviations noted during the performance of these tests are investigated and corrected in accordance with the established calibration and trouble-shooting procedures provided in the plant technical manual for the nuclear instrumentation system. Control and protection trip settings are indicated in the precautions, limitations, and setpoints section of the general operating instructions for the nuclear steam supply system.

7.2.3.3 Periodic Testing of the Process Analog Channels of the Protection Circuits

The following periodic tests of the analog channels of the protection circuits are performed:

A. T_{avg} and ΔT protection channel testing.

B. Pressurizer pressure protection channel testing.

C. Pressurizer water level protection channel testing.

D. Steam generator water level protection channel testing.

E. Reactor coolant flow protection channel testing.

F. Impulse chamber pressure channel testing.

The following conditions are required for these tests:

A. These tests may be performed at any plant power from cold shutdown to full power.

B. Before starting any of these tests with the plant at power, all redundant reactor trip channels associated

with the function to be tested must be in the normal (untripped) mode in order to avoid spurious trips.

C. Setpoints are indicated in the precautions, limitations, and setpoints section of the general operating instructions for the nuclear steam supply system.

D. Reference is made to the supplier's systems manual(s) for systems description and static and dynamic testing (to be supplied with the equipment).

E. Reference is made to the supplier's manual instrument documentation sheets, which provide information on available signal ranges and adjustments. In addition, the supplier's report of equipment test results provides equipment calibration data.

F. Median Signal Selector Testing. The signal selector has been provided with the capability for online testing commensurate with periodic testing of the steam generator level channels. Signal selector testing consists of monitoring the three input signals and the one output signal via test points. Comparison of the output signal to the input signals permits determination of whether or not the median signal is being passed and, consequently, whether the signal selector is functioning properly. Any output signal at a value other than that corresponding to the median signal is indicative of a unit failure.

7.2.3.4 Regulatory Guide 1.22

Periodic testing of the reactor trip system actuation functions, as described, complies with NRC Regulatory Guide 1.22, Periodic Testing of Protection System Actuation Functions, February 1971. (See paragraph 7.1.2.8.)

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14. Institute of Electrical and Electronics Engineers, "Trial-Use Guide for Seismic Qualification of Class 1 Electrical Equipment for Nuclear Power Generating Stations," IEEE 344-1971.

15. U.S. Nuclear Regulatory Commission, "General Design Criteria for Nuclear Power Plants," Appendix A to 10 CFR 50, July 7, 1971.

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17. "Elimination of the Low Feedwater Flow Reactor Trip via Implementation of the Median Signal Selector (MSS) at Farley Units 1 and 2," WCAP-13807 (Proprietary) and WCAP-13808 (Nonproprietary), August 1993.

TABLE 7.2-1 (SHEET 2 OF 2)

Reactor Trip	Coincidence Logic	Interlocks	Comments
Reactor coolant pump bus undervoltage	2/3	Interlocked with P-7	Low voltage on all buses permitted below P-7
Reactor coolant pump bus underfrequency	2/3	Interlocked with P-7	Underfrequency on two buses will trip all reactor coolant pump breakers and cause reactor trip; reactor trip blocked below P-7
Low-low steam generator water level	2/3 per loop	No interlocks	
Safety injection signal	Coincident with actuation of safety injection	No interlocks	(See section 7.3 for engineered safety features actuation conditions)
Turbine-generator trip			
Low auto stop pressure oil	2/3	Interlocked with P-9	Blocked below P-9
Turbine stop valve close	4/4	Interlocked with P-9	Blocked below P-9
Manual	1/2	No interlocks	

TABLE 7.2-3 (SHEET 2 OF 2)

Reactor Trip Signal	Accuracy	Note
Reactor coolant pump bus underfrequency	±0.1 Hz	
Low feedwater flow	(c)	(b)(c)
Low-low steam generator water level	±2.3 percent of ΔP signal over pressure range of 600 to 1100 psig	

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a. Reproducibility.

b. Alarms only.

c. Accuracy of feedwater flow signal is ±2.5 percent of maximum calculated feedwater flow.

Accuracy of steam flow signal is ±3 percent of maximum calculated flow over the pressure range of 600 to 1100 psig.

TABLE 7.2-4 (SHEET 3 OF 4)

Trip	Accident	Technical Specifications
	15.2.13 - Accidental depressurization of the main steam system	
Low primary coolant flow: 1. Undervoltage 2. Underfrequency 3. Low flow or pump breaker open 1/3 loops 4. Low flow or pump breaker open 2/3 loops	15.2.5 - Partial loss of forced reactor coolant flow 15.2.9 - Loss of off-site power to the station auxiliaries (station blackout)	Yes
Pressurizer high pressure	15.2.2 - Uncontrolled RCCA bank withdrawal at power 15.2.7 - Loss of external electrical load and/or turbine trip	Yes
Pressurizer high water level	15.2.2 - Uncontrolled RCCA bank withdrawal at power 15.2.7 - Loss of external electrical load and/or turbine trip	Yes
Pressurizer low pressure	15.2.12 - Accidental depressurization of the reactor coolant system	Yes

TABLE 7.2-4 (SHEET 4 OF 4)

Trip	Accident	Technical Specifications
Low-low steam generator level	15.2.8 - Loss of normal feedwater	Yes

TABLE 7.2-5 (SHEET 1 OF 2)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

Functional Unit		Response Time(s)
1.	Manual reactor trip	NA
2.	Power range, neutron flux	
	a. High	≤ 0.5 [(a)]
	b. Low	NA
3.	Power range, neutron flux, high positive rate	NA
4.	Power range, neutron flux, high negative rate	NA
5.	Intermediate range, neutron flux	NA
6.	Source range, neutron flux	NA
7.	Overtemperature ΔT	≤ 6.0 [(a)]
8.	Overpower ΔT	NA
9.	Pressurizer pressure-low	≤ 2.0
10.	Pressurizer pressure-high	≤ 2.0
11.	Pressurizer water level-high	NA
12A.	Loss of flow-single loop (above P-8)	≤ 1.0
12B.	Loss of flow-two loops (above P-7 and below P-8)	≤ 1.0
13.	Steam generator water level-low-low	≤ 2.0
14.	Steam/feedwater flow mismatch and low steam generator water level	NA
15.	Undervoltage-reactor coolant pumps	≤ 1.2
16.	Underfrequency-reactor coolant pumps	≤ 0.6

TABLE 7.2-5 (SHEET 2 OF 2)

Functional Unit	Response Time(s)
17. Turbine trip	
a. Low auto stop oil pressure	NA
b. Turbine throttle valve closure	NA
18. Safety injection input from ESF	NA
19. Reactor coolant pump breaker position trip	NA
20. Reactor trip system interlocks	NA
21. Reactor trip breakers	NA
22. Automatic trip logic	NA

a. Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

plant condition changes, the test frequency is accelerated to accommodate the situation until the marginal performance is resolved.

D. The test interval discussed in paragraph 5.2 of IEEE 338-1971 is developed primarily on past operating experience and modified, if necessary, to ensure that system and subsystem protection is reliably provided. Analytical methods for determining reliability are not used to determine test interval.

7.3.2.6 Evaluation of Compliance with IEEE 344-1971

The seismic testing, as set forth in paragraph 7.2.1.10, IEEE 338-1971,[9] IEEE 344-1971,[10] WCAP-7706,[11] and WCAP-7705,[3] conforms to the guidelines set forth in IEEE 344-1971.[10]

7.3.2.7 Response Time Testing

Response time testing consists of any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined in the technical specifications. Sensor response time verification may be demonstrated by either 1) inplace, onsite, or offsite test measurements or 2) utilizing replacement sensors with certified response times. The measurement of response time at the specified frequencies provides assurance that the reactor trip associated with each channel is completed within the time limit assumed in the accident analyses. The response time limits for the engineered safety feature actuation system are provided in table 7.3-16.

7.3.2.8 Further Considerations

In addition to the considerations given above, a loss of instrument air or loss of component cooling water to vital equipment has been considered. Assuming no other accident conditions, neither cause safety limits, as given in the technical specifications, to be exceeded. Likewise, loss of either one of the two will not adversely affect the core or the reactor coolant system, nor will it prevent an orderly shutdown if this is necessary. Furthermore, all pneumatically operated valves and controls will assume a preferred operating position upon loss of instrument air. It is also noted that, for conservatism during the accident analyses (chapter 15), credit is not taken for the instrument air systems nor any control system benefit.

7.3.2.9 Summary

The effectiveness of the ESFAS is evaluated in chapter 15, based upon the ability of the system to contain the effects of Condition III and IV faults, including LOCAs and steam break accidents. The ESFAS parameters are based upon the component performance specifications which are given by the manufacturer or verified by test for each component. Appropriate factors to account for uncertainties in the data are factored into the constants characterizing the system.

The ESFAS must detect Condition III and IV faults and generate signals which actuate the engineered safety features. The system must sense the accident condition and generate the signal actuating the protective function reliably and within a time determined by and consistent with the accident analyses in chapter 15.

Much longer times are associated with the actuation of the mechanical and fluid system equipment associated with engineered safety features. This includes the time required for switching and bringing pumps and other equipment to speed and the time required for them to take load.

Operating procedures require that the complete ESFAS normally be operable. However, redundancy of system components is such that the system operability assumed for the safety analyses can still be met with certain instrumentation channels out of service. Channels that are out of service are to be placed in the tripped mode or bypass mode in the case of containment spray.

7.3.2.9.1 Loss of Coolant Protection

By analysis of a LOCA, it has been verified that except for very small coolant system breaks which can be protected against by the charging pumps followed by an orderly shutdown, the effects of various LOCAs are reliably detected by the low pressurizer pressure signal; the emergency core cooling system is actuated in time to prevent or limit core damage.

For large coolant system breaks the passive accumulators inject first, because of the rapid pressure drop. This protects the reactor during the unavoidable delay associated with actuating the active emergency core cooling system phase.

High containment pressure also actuates the emergency core cooling system. Therefore, emergency core cooling actuation can be brought about upon sensing this other direct consequence

of a primary system break; i.e., the ESFAS detects the leakage of the coolant into the containment. The generation time of the actuation signal of about 1.0 s, after detection of the consequences of the accident, is adequate.

Containment spray will provide additional emergency cooling of containment and also limit fission product release upon sensing elevated containment pressure (high-high) to mitigate the effects of a LOCA.

The delay time between detection of the accident condition and the generation of the actuation signal for these systems is assumed to be about 1.0 s, well within the capability of the protection system equipment. However, this time is short compared to that required for startup of the fluid systems.

The analyses in chapter 15 show that the diverse methods of detecting the accident condition and the time for generation of the signals by the protection systems are adequate to provide reliable and timely protection against the effects of loss of coolant.

The setpoint for the initiation of the containment spray is set at 50 percent of design pressure. Following a LOCA, this pressure will be reached in 7.0 s.

The containment pressure instrumentation that generates the containment spray signal has the same time delay as the instrumentation that generates the signal on high containment pressure for closing the main steam stop valves, as presented in subsection 7.3.1.2. The maximum allowable time delay in generating this actuation signal is 1.0 s.

Coincident with generation of the containment spray actuation signal, we assume a loss of offsite power to yield the maximum time delay in starting the spray pumps and opening the motor-operated valves. With the utilization of diesel power, there is a maximum delay of 17.5 s between the time the containment spray actuation signal is generated and the time for the containment spray pumps to start running. The motor-operated valves will be opened during this 17.5-s period. If offsite power is available, this time delay will be shorter.

Once the containment spray pumps are started and all valves are opened, it will take 29.5 s for water to reach the nozzles of the highest containment spray header.

Therefore, the total time span between a rupture of the reactor coolant pressure boundary and water reaching the highest nozzle of the containment spray headers is a maximum of 55.0 s.

Three containment air coolers are running continuously during normal plant operation. Service water is flowing through all four containment coolers during normal operation. Following a LOCA, at least two coolers will be operated in low speed.

If offsite power is available, the three containment coolers that were operating at high speed during normal operation will switch to low speed upon receipt of a safety injection signal. The fourth cooler will begin operating at low speed upon receipt of a safety injection signal. Two coolers may be shut down by the plant operator, and two coolers will continue to operate. There is no time delay in starting the containment coolers, if offsite power is available.

If offsite power is lost, power will not be available for the high speed windings of the containment air cooler motors.

However, the diesel generators will automatically be started, and at least two coolers will be started in low speed by the loss-of-power shutdown sequencer. This will occur regardless of whether a LOCA has occurred. The time between the loss of offsite power and the restarting of the containment coolers is 27.4 s. The service water pumps are restarted by the diesels 5 s before the containment coolers.

7.3.2.9.2 Steam Break Protection

The emergency core cooling system is also actuated in order to protect against a steam line break. About 2.0 s elapse between sensing high steam line differential pressure or low steam line pressure and generation of the actuation signal. Analysis of steam break accidents, assuming this delay for signal generation, shows that the emergency core cooling system is actuated for a steam break in time to limit or prevent damage in the core. There is a reactor trip, but the core reactivity is further reduced by the highly borated water injected by the emergency core cooling system.

Additional protection against the effects of steam break is provided by feedwater isolation which occurs upon actuation of the emergency core cooling system. Feedwater line isolation is initiated to prevent excessive cooldown of the reactor.

Additional protection against a steam break accident is provided by closure of all steam line isolation valves to prevent uncontrolled blowdown of all steam generators. The generation of the protection system signal (about 2.0 s) is again short compared to the time to trip the fast acting steam line isolation valves which are designed to close in less than approximately 5 s after receipt of a signal.

In addition to actuation of the engineered safety features, the effect of a steam break accident also generates a signal resulting in a reactor trip on overpower or following emergency core cooling system actuation. However, the core reactivity is further reduced by the borated water (2000 ppm)[a] injected by the emergency core cooling system.

The analysis in chapter 15 of the steam break accidents and an evaluation of the protection system instrumentation and channel design show that the ESFAS is effective in preventing or mitigating the effects of a steam break accident.

7.3.2.9.3 Feedline Break Protection

The ESFAS is actuated to protect against a feedline break. Following reactor trip due to a low-low steam generator water level trip setpoint, pressure in the steam line falls below a given setpoint. When the setpoint is reached, all main steam isolation valves are closed, which guarantees a steam supply for the turbine-driven auxiliary feedwater pump.

Assurance that adequate feedwater is available for the feedline break is provided by the auxiliary feedwater system, which includes two motor-driven pumps and one turbine-driven pump. The motor-driven pumps are initiated automatically by one of the following signals:

A. Safety injection or safeguards sequence (derived from the solid state protection system output cabinets).

B. Two out of three low-low level in any steam generator (derived from the solid state protection system output cabinets).

C. Manual start.

D. Trip of all main feed pumps.

E. Blackout signal.

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a. The minimum RWST boron concentration was increased to 2300 ppm in Unit 1 License Amendment 68 and Unit 2 license Amendment 60. The purpose of the safety injection during this event is to control the return to criticality by inserting negative reactivity into the core. The injection of 2300 ppm boron concentration from the RWST would insert more negative reactivity into the core than the 2000 ppm boron concentration, which would cause the event to be terminated sooner. Therefore, the injection of 2300 ppm boron concentration from the RWST is bounded by the analysis presented.

The turbine-driven pumps, as well as the closing of blowdown and sample valves, are initiated automatically by one of the following signals:

A. Two out of three low-low level in two out of three steam generators (derived from the solid state protection system output cabinets).

B. Manual start.

C. Loss of voltage signal.

Evaluation of the protection system instrumentation and channel design shows that minimum auxiliary feedwater capacity is adequate to remove decay heat to prevent overpressurization of the reactor coolant system and to prevent uncovering the reactor core. Minimum auxiliary feedwater capacity is that

capacity available following a feedline break event, assuming the worst single failure. The analysis in chapter 15 of the feedline break accident shows that the ESFAS is effective in mitigating the effects of a feedline break accident.

TABLE 7.3-14 (SHEET 1 OF 3)

FAILURE MODE AND EFFECTS ANALYSIS,
EMERGENCY DIESEL GENERATOR

Component Identification Diesel Generator Supply Breaker

Elementary Number D-172761, D-177142, and D-177143

Failure Mode	Effect on System	Detection of Failure	Remarks
Loss of 125 V-dc control power	Loss of ability to tie diesel generator to bus when necessary	Loss of dc annunciator in control room	Redundant diesel generator will be started
Failure of 2AJX contacts to close in emergency	Loss of ability to tie diesel generator to bus when necessary	Incomplete sequence annunciator	Redundant diesel generator will be started
Failure of 59/81X contacts	Loss of ability to tie diesel generator to bus when necessary	Indicating light on control board	Redundant diesel generator will be started
Mechanical or electrical failure of breaker	Loss of ability to tie diesel generator to bus when necessary	Breaker position indicating lights in control room	Redundant diesel generator will be started

TABLE 7.3-14 (SHEET 2 OF 3)

Component Identification Diesel Generator Start, Stop, and Shutdown Controls

Elementary Number 172774, D-172778, D-172782

Failure Mode	Effect on System	Detection of Failure	Remarks
Loss of 125 V-dc control power	Loss of ability to start diesel generator in emergency	Annunciator and loss of indicating lights on board	Redundant diesel generator will be started
Failure of start contact in diesel starting circuit A or B to close in emergency	None; redundant starting circuit will start diesel	Testing	
Failure of a relay in starting circuit A or B	None; redundant starting circuit will start diesel	Testing	
Failure of signal contact or relay in diesel stop circuit	Loss of ability to stop diesel from control room	Diesel running light in control room	Diesel can be stopped manually
Failure of contact or relay in diesel shutdown circuit	Diesel would not shut down when trouble occurred	Observation of diesel failure	All safety features are cut out except overspeed and low oil pressure during emergency operation

TABLE 7.3-14 (SHEET 3 OF 3)

Component Identification Diesel Generator Excitation and Miscellaneous Controls

Elementary Number 172775, D-172779, D-172783

Failure Mode	Effect on System	Detection of Failure	Remarks
Failure of governor control	Diesel generator may not pick up load or may drop load during load fluctuations	Observation of voltage and frequency on board	Redundant generator can be used
Failure of excitation circuit	Improper voltage output from generator	Observation of meter on board	Redundant generator can be used
Failure of auto voltage	Improper voltage output from generator	Observation of meter on board	Redundant generator manual voltage control can be used

TABLE 7.3-16 (SHEET 1 OF 3)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

Initiating Signal and Function	Response Time (s)
1. Manual	
a. Safety injection (ECCS)	NA
Feedwater isolation	NA
Reactor trip (SI)	NA
Containment isolation-Phase "A"	NA
Containment purge isolation	NA
Auxiliary feedwater pumps	NA
Service water system	NA
Containment air coolers	NA
b. Containment spray	NA
Containment isolation-Phase "B"	NA
Containment purge isolation	NA
c. Containment isolation-Phase "A"	NA
Containment purge isolation	NA
d. Steam line isolation	NA
2. Containment Pressure-High	
a. Safety injection (ECCS)	≤ 27.0[a]
b. Reactor trip (from SI)	≤ 2.0
c. Feedwater isolation	≤ 32.0[f]
d. Containment isolation-Phase "A"	≤ 17.0[d]/27.0[e]
e. Containment purge isolation	≤ 5.0
f. Auxiliary feedwater pumps	NA
g. Service water system	≤ 77.0[d]/87.0[e]
h. Containment air cooler fan	≤ 27.4
3. Pressurizer Pressure-Low	
a. Safety injection (ECCS)	≤ 27.0[a]/12.0[d]
b. Reactor trip (from SI)	≤ 2.0
c. Feedwater isolation	≤ 32.0[f]
d. Containment isolation-Phase "A"	≤ 17.0[d]
e. Containment purge isolation	≤ 5.0
f. Auxiliary feedwater pumps	NA
g. Service water system	≤ 77.0[d]/87.0[a]

TABLE 7.3-16 (SHEET 2 OF 3)

Initiating Signal and Function	Response Time (s)
4. Differential Pressure Between Steam Lines-High	
a. Safety injection (ECCS)	≤ 12.0(d)/22.0(e)
b. Reactor trip (from SI)	≤ 2.0
c. Feedwater isolation	≤ 32.0(f)
d. Containment isolation-Phase "A"	≤ 17.0(d)/27.0(e)
e. Containment purge isolation	NA
f. Auxiliary feedwater pumps	NA
g. Service water system	≤ 77.0(d)/87.0(e)
5. Steam Flow in Two Steam Lines-High Coincident with T_{avg}--Low-Low	
a. Steam line isolation	NA
6. Steam Line Pressure-Low	
a. Safety injection (ECCS)	≤ 12.0(d)/22.0(e)
b. Reactor trip (from SI)	≤ 2.0
c. Feedwater isolation	≤ 32.0(f)
d. Containment isolation-Phase "A"	≤ 17.0(d)/27.0(e)
e. Containment purge isolation	NA
f. Auxiliary feedwater pumps	NA
g. Service water system	≤ 77.0(d)/87.0(e)
h. Steam line isolation	≤ 7.0
7. Containment Pressure--High-High	
a. Steam line isolation	≤ 7.0
8. Containment Puressure--High-High-High	
a. Containment spray	≤ 45.0
b. Containment isolation-Phase "B"	NA
9. Steam Generator Water Level--High-High	
a. Turbine trip	≤ 2.5
b. Feedwater isolation	≤ 32.0(f)
10. Steam Generator Water Level--Low-Low	
a. Motor-driven auxiliary feedwater pumps(b)	≤ 60.0
b. Turbine-driven auxiliary feedwater pump(c)	≤ 60.0

TABLE 7.3-16 (SHEET 3 OF 3)

Initiating Signal and Function	Response Time (s)
11. Undervoltage RCP	
a. Turbine-driven auxiliary feedwater pump	≤ 60.0
12. S.I. Signal	
a. Motor-driven auxiliary feedwater pumps	≤ 60.0
13. Trip of Main Feedwater Pumps	
a. Motor-driven auxiliary feedwater pumps	NA
14. Loss of Power	
a. 4.16-kV emergency bus undervoltage (Loss of Voltage)	(g)
b. 4.16-kV emergency bus undervoltage (Degraded Voltage)	(g)

a. Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps and RHR pumps.

b. One 2/3 any steam generator.

c. On 2/3 in 2/3 steam generators.

d. Diesel generator starting and sequence loading delay not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

e. Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

f. Verification shall include testing of all instrumentation, the isolation valves (MOV-3232A, 3232B, 3232C), and the control valves (FCV-478, 479, 488, 489, 498, 499). The isolation valves must function within 30 s and the control valves within 5 s.

g. The response time shall include the time delay associated with the undervoltage relays as determined in table 3.3-4 of the technical specifications plus an additional second associated with interposing relay and circuit operation.

B. Auxiliary feedwater pumps. (See chapter 10.)

C. Boric acid transfer pump. (See chapter 9.)

D. Charging pumps. (See chapter 9.)

E. Service water pumps. (See chapter 9.)

F. Containment fans. (See chapter 6.)

G. Control room ventilation. (See chapter 9.)

H. Component cooling pumps. (See chapter 9.)

I. Residual heat removal pumps. (See chapter 5.)[a]

J. Motor control center and switchgear sections associated with above loads.

K. Controlled steam release and feedwater supply. (See section 7.7 and chapter 10.)

L. Boration capability. (See chapter 9.)

M. Nuclear instrumentation system (source range and intermediate range). (See sections 7.2 and 7.7.)[a]

N. Reactor coolant inventory control (charging and letdown). (See chapter 9.)

O. Pressurizer pressure control including opening control for pressurizer relief valves (heaters and spray). (See chapter 5.)[a]

In addition, the safety injection signal trip circuit must be defeated and the accumulator isolation valves closed.[a] The performance of the emergency core cooling system under these conditions was evaluated. Conditions during plant cooldown were divided into the following four phases: (1) from operating reactor coolant pressure to 1900 psig, (2) from 1990 to 1000 psig, (3) from 1000 to 400 psig, and (4) from 400 psig to cold shutdown. The break size used in the analysis was determined using the moderate energy line break criteria identified in Branch Technical Positions APCSB 3-1 and MEB 3-1. Based on the analysis, the available emergency core cooling system can cool the core under plant cooldown conditions and, therefore, meets the NRC acceptance criteria, as applicable, contained in 10 CFR 50.46 and 10 CFR 50, Appendix K.

7.4.2 ANALYSIS

Hot shutdown is a stable plant condition, automatically reached following a plant shutdown. The hot shutdown condition can be maintained safely for an extended period of time either automatically or manually. In the unlikely event that access to the control room is restricted, the plant can be safely kept at a hot shutdown until the control room can be reentered by the use of the monitoring indicators and the controls listed in paragraphs 7.4.1.1 and 7.4.1.2. These indicators and controls are provided outside and inside the control room.
The safety evaluation of the maintenance of a shutdown with these systems and associated instrumentation and controls has included consideration of the accident consequences that might jeopardize safe shutdown conditions. The accident consequences that are germane are those that would tend to degrade the capabilities for boration, adequate supply for auxiliary feedwater, and residual heat removal.

The results of the accident analyses are presented in chapter 15. Of these the following produce the most severe consequences that are pertinent:

A. Uncontrolled boron dilution.

B. Loss of normal feedwater.

C. Loss of external electrical load and/or turbine trip.

D. Loss of all ac power to the station auxiliaries (station blackout).

It is shown by these analyses that safety is not compromised by these incidents, with the associated assumptions being that the instrumentation and controls indicated in paragraphs 7.4.1.1 and 7.4.1.2 are available to control and/or monitor shutdown. These available systems will allow a maintenance of hot shutdown even under the accident conditions listed above which would tend toward a return to criticality or a loss of heat sink.

TABLE 7.5-3 (SHEET 4 OF 6)

Parameter	No. of Channels Available	Range	Indicated Accuracy	Indicated Recorder	Location	Notes
Reactor Control System						
Demanded rod speed	1	0 to 75 steps/min	±2%	1 channel is indicated	Control board	
Median T_{avg}	1	530°F to 630°F	±4°F	1 channel is recorded	Control board	
T_{ref}	1	530°F to 630°F	±4°F	1 channel is recorded	Control board	
Control rod position						If system not available, borate and sample accordingly
Number of steps of demanded rod withdrawal	1 per group	0 to 231 steps[a]	±1 step	Each group is indicated	Control board	These signals are used in conjunction with the full-length rod measured position signals to detect deviation of any individual rod from the demanded position; a deviation will acutate an alarm
Full-length rod measured position	1 per rod	0 to 228 steps[b]	±4 steps at full accuracy; ±8 steps at 1/2 accuracy	Each rod position is indicated	Control board	

a. Fully withdrawn position can be varied from 225 to 231 steps to reduce RCCA wear. The NRC acceptance criteria regarding the range associated with the fully withdrawn RCCA position are that the fully withdrawn position selected for use throughout each cycle will be evaluated as part of the reload safety evaluation process to verify that sufficient margin exists in the safety analyses to bound the related effects.

b. Digital Rod Position Indication (DRPI) system maximum indication is 228 steps.

actuation of high pressure reactor trip for the above condition.

A block diagram of the pressurizer pressure control system is shown in figure 7.7-4.

7.7.1.6 Pressurizer Water Level Control

The pressurizer operates by maintaining a steam cushion over the reactor coolant. As the density of the reactor coolant adjusts to the various temperatures, the steam water interface moves to absorb the variations with relatively small pressure disturbances.

The water inventory in the reactor coolant system is maintained by the chemical and volume control system. During normal plant operation, the charging flow varies to produce the flow demanded by the pressurizer water level controller. The pressurizer water level is programmed as a function of coolant average temperature, with the median average temperature being used. The pressurizer water level decreases as the load is reduced from full load. This is a result of coolant contraction following programmed coolant temperature reduction from full power to low power. The programmed level is designed to match as nearly as possible the level changes resulting from the coolant temperature changes.

To control pressurizer water level during startup and shutdown operations, the charging flow is manually regulated from the main control room.

A block diagram of the pressurizer water level control system is shown in figure 7.7-5.

7.7.1.7 Steam Generator Water Level Control

Each steam generator is equipped with a three-element feedwater flow control system which maintains a programmed water level as a function of turbine load. The three-element feedwater controller regulates the feedwater valve by continuously comparing the feedwater flow signal, the water level signal, the programmed level, and the pressure compensated steam flow signal. The water level signal provided to the feedwater flow control system is derived from an MSS which selects the median input of the three narrow range level channels for each steam generator. In addition, for the turbine-driven main feedwater pumps, the feedwater pump speed is varied to maintain a programmed pressure differential between the steam header and the feed pump discharge header. The speed controller continuously compares the actual ΔP with a programmed ΔP_{ref} which is linear function of steam flow. (See figure 7.7-6.) Continued delivery of feedwater to the steam generators is

required as a sink for the heat stored and generated in the reactor following a reactor trip and turbine trip. An override signal closes the feedwater valves when the average coolant temperature is below a given temperature and the reactor has tripped. Manual control of the feedwater control system is available at all times.

A block diagram of the steam generator water level control system with an MSS is shown in figure 7.7-7. The MSS will automatically select one of the steam generator input signals as the control signal in the event of an MSS failure. A detailed discussion of the MSS and its operation can be found in section 7.2, reference 17.

7.7.1.8 Steam Dump Control

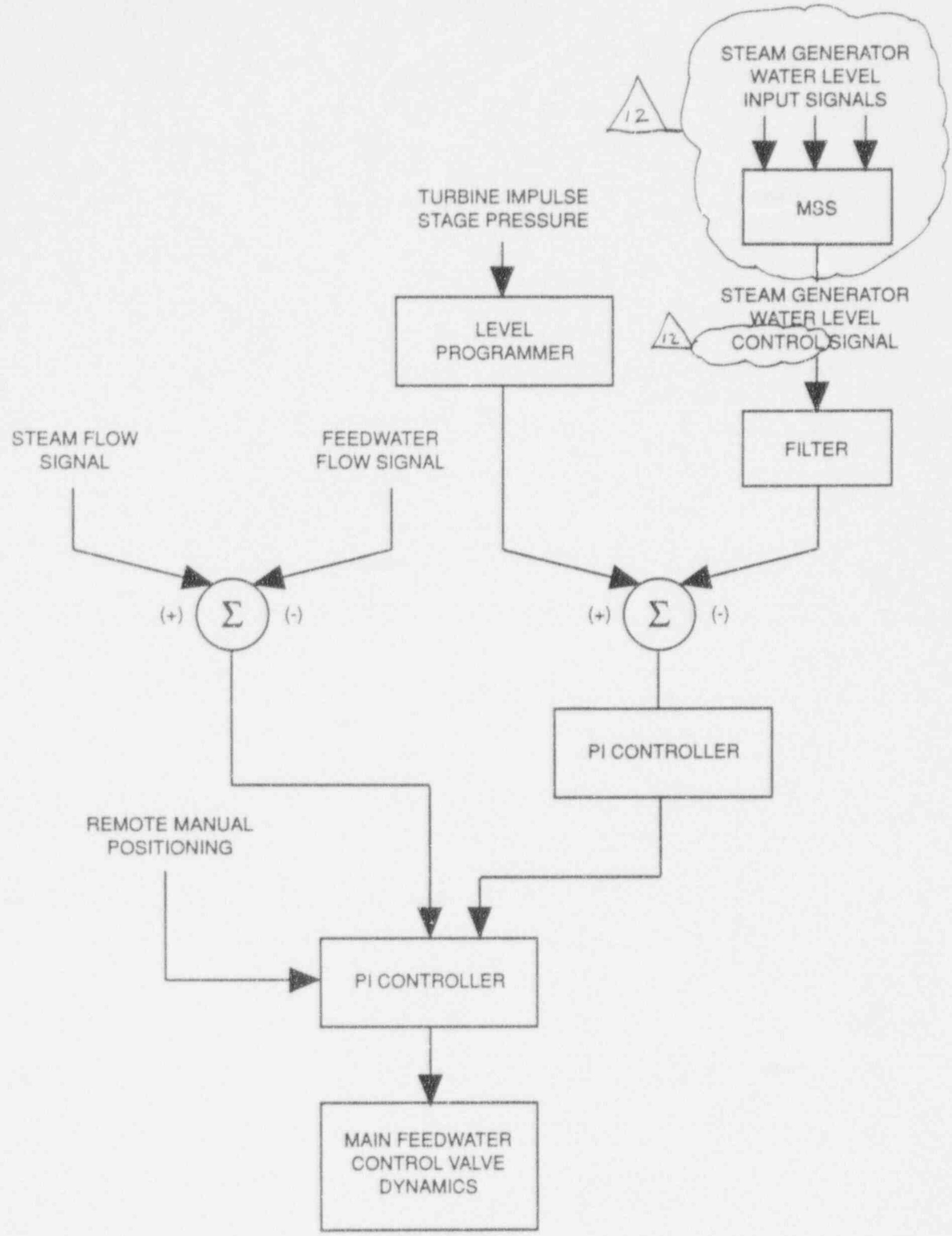
The steam dump system is designed to accept a 50-percent loss of net load without tripping the reactor.

The automatic steam dump system is able to accommodate this abnormal load rejection and to reduce the effects of the transient imposed upon the reactor coolant system. By bypassing main steam directly to the condenser, an artificial load is thereby maintained on the primary system. The rod control system can then reduce the reactor temperature to a new equilibrium value without causing overtemperature and/or overpressure conditions. The plant has a 50-percent loss of net load capability. The steam dump steam flow capacity is 40 percent of full load steam flow at full load steam pressure.

If the difference between the reference $T_{avg}(T_{ref})$ based on turbine impulse chamber pressure and the lead/lag compensated median T_{avg} exceeds a predetermined amount and if the interlock mentioned below is satisfied, a demand signal will actuate the steam dump to maintain the reactor coolant system temperature within control range until a new equilibrium condition is reached.

To prevent actuation of steam dump on small load perturbations, an independent load rejection sensing circuit is provided. This circuit senses the rate of decrease in the turbine load as detected by the turbine impulse chamber pressure. It is provided to unblock the dump valves when the rate of load rejection exceeds a preset value corresponding to a 10-percent step load decrease or a sustained ramp load decrease of 5 percent/min.

A block diagram of the steam dump control system is shown in figure 7.7-8.



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JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	BLOCK DIAGRAM OF STEAM GENERATOR WATER LEVEL CONTROL SYSTEM
	FIGURE 7.7-7

A simplified block diagram of the AMSAC ALS architecture is presented on figure 7.8-1.

2. Test/Maintenance System

The test/maintenance system provides the AMSAC with automated and manual testing as well as a maintenance mode. Automated testing is the continuously performed self checking done by the system during normal operation. ALS status is monitored by the T/MS and sent to the plant computer and the main control board. Manual testing of the system by the computer services staff can be performed on-line to provide assurance that the ALS system is fully operational. The maintenance mode permits the computer services staff, under administrative control, to modify channel setpoints, channel status, and timer values and to initiate channel calibration.

The T/MS consists of a test/maintenance processor, a digital-to-analog conversion board, a memory board, expansion boards, a self-health board, digital output modules, a test/maintenance panel, and a portable terminal/printer.

D. Equipment Actuation

The output relay panels provide component actuation signals through isolation relays, which then drive the final actuation circuitry as shown on figures 7.2-14 and -15 for initiation of auxiliary feedwater and for turbine trip.

7.8.1.3 Functional Performance Requirements

Analyses have shown that the most limiting ATWS event is a loss of feedwater event without a reactor trip. AMSAC performs the mitigative actuations of automatically initiating auxiliary feedwater, tripping the turbine, and isolating steam generator blowdown and sampling lines. These are initiated in order to ensure a secondary heat sink following an anticipated transient (ANS Condition II) without a reactor trip, in order to limit core damage following an anticipated transient without a reactor trip and to ensure that the energy generated in the core is compatible with the design limits to protect the reactor coolant pressure boundary by maintaining the reactor coolant pressure to within ASME stress level C.

7.8.1.4 AMSAC Interlocks

A single interlock, designated as C-20, is provided to allow for the automatic arming and blocking of the AMSAC (figure 7.2-15). The system is blocked at sufficiently low reactor power levels when the actions taken by the AMSAC following an ATWS need not be automatically initiated. Turbine impulse chamber pressure in a two-out-of-two logic scheme is used for the blocking function. Turbine impulse chamber pressure above the setpoint will automatically defeat any block, i.e., will arm the AMSAC. Dropping below this setpoint will automatically block the AMSAC. Removal of the C-20 permissive is automatically delayed for a predetermined time. The operating status of the AMSAC is displayed on the main control board.

7.8.1.5 Trip System

The SG level and turbine impulse chamber pressure inputs are used by AMSAC to determine trip demand. Signal conditioning is performed on the transmitter output and used by each of the ALPs to derive a component actuation demand. If two of the three steam generators have a low level at a power level greater than the C-20 permissive, a trip demand signal is generated following a time delay. This signal drives output relays for performing the necessary mitigative actions.

7.8.1.6 Isolation Devices

AMSAC is independent of the RTS and ESFAS. The AMSAC inputs for measuring narrow range steam generator water level are derived from existing transmitters and channels within the process protection system. Connections to these channels are made downstream of Class-1E isolation devices located within the process protection cabinets. These isolation devices ensure that the existing protection system continues to meet all applicable safety criteria by providing isolation. Buffering of the AMSAC outputs from the safety-related final actuation device circuits is achieved through qualified relays. A credible fault occurring in the nonsafety-related AMSAC will not propagate through and degrade the RTS and ESFAS.

7.8.1.7 AMSAC Diversity from the Reactor Protection Systems

Equipment diverse from the RTS and ESFAS is used in the AMSAC to prevent common mode failures that might affect the AMSAC and the

RTS or ESFAS. The AMSAC is a digital, microprocessor-based system with the exception of the analog SG level and turbine impulse pressure transmitter inputs, whereas the reactor trip system utilizes an analog based protection system. Also, where similar components are utilized for the same function in both AMSAC and the reactor trip system, the components used in AMSAC are provided by a different manufacturer.

Common mode failure of identical components in the analog portion of the RTS that results in the inability to generate a reactor trip signal will not impact the ability of the digital AMSAC to generate the necessary mitigative actuations. Similarly, a postulated common mode failure affecting analog components in ESFAS, affecting its ability to initiate auxiliary feedwater, will not impact the ability of the digital based AMSAC to automatically initiate auxiliary feedwater.

7.8.1.8 Power Supply

The AMSAC power supply is a dedicated uninterruptable power supply (UPS) which is independent from the RTS power supplies and is backed by batteries which are independent from the existing batteries which supply the RTS.

7.8.1.9 Environmental Variations

The AMSAC equipment is located in a controlled environment such that variations in the ambient conditions are minimized.

7.8.1.10 Setpoints

The AMSAC makes use of two setpoints in the coincidence logic in order to determine if mitigative functions are required. Water level in each steam generator is sensed to determine if a loss of secondary heat sink is imminent. The low-level setpoint is selected in such a manner that a true lowering of the level will be detected by the system. The normal small variations in steam generator level will not result in a spurious AMSAC signal.

The C-20 permissive setpoint is selected in order to be consistent with ATWS investigations showing that the mitigative actions performed by the AMSAC need not be automatically actuated below a certain power level. The maximum allowable value of the C-20 permissive setpoint is defined by these investigations.

To avoid inadvertent AMSAC actuation on the loss of one main feedwater pump, AMSAC actuation is delayed by a defined amount of time. This will ensure the reactor protection system will provide the first trip signal.

To ensure that the AMSAC remains armed sufficiently long to permit its function in the event of a turbine trip, the C-20 permissive is maintained for a preset time delay after the turbine impulse chamber pressure drops below the setpoint. The setpoints and the capability for their modification in the AMSAC are under administrative control.

7.8.2 ANALYSIS

7.8.2.1 Safety Classification/Safety-Related Interface

The AMSAC is not safety related, therefore, it need not meet the requirements of IEEE-279-1971. The AMSAC has been implemented such that the RTS and ESFAS continue to meet all applicable safety-related criteria. The AMSAC is independent of the RTS and ESFAS. The isolation provided between the RTS and the AMSAC and between the ESFAS and the AMSAC by the isolator modules and the isolation relays, respectively, ensures that applicable safety-related criteria are met for the RTS and the ESFAS.

7.8.2.2 Redundancy

System redundancy has not been provided. Since AMSAC is a backup nonsafety-related system to the redundant RTS, redundancy is not required. To ensure high system reliability, portions of the AMSAC have been implemented as internally redundant, such that a single failure of an input channel or ALP will neither actuate nor prevent actuation of the AMSAC.

7.8.2.3 Diversity from the Existing Trip System

Diverse equipment has been selected in order that common cause failures affecting both the RTS and the AMSAC or both the ESFAS and the AMSAC will not render these systems inoperable simultaneously. A more detailed discussion of the diversity between the RTS and the AMSAC and between the ESFAS and the AMSAC is presented in paragraph 7.8.1.7.

7.8.2.4 Electrical Independence

The AMSAC is electrically independent of the RTS and ESFAS with

inadvertent actuations is minimized. This high reliability is ensured through use of three redundant ALPs and a majority voting module. A single failure in any of these modules will not result in a spurious AMSAC actuation. In addition, a two-out-of-three low-steam generator level coincidence logic and a time delay have been selected to further minimize the potential for inadvertent actuations.

7.8.2.12 Bypass

7.8.2.12.1 Maintenance Bypasses

The AMSAC is blocked at the system level during maintenance, repair, calibration, or test. While the system is blocked, the bypass condition is indicated in the main control room.

7.8.2.12.2 Operating Bypasses

The AMSAC has been designed to allow for operational bypasses with the inclusion of the C-20 permissive. Above the C-20 setpoint, the AMSAC is automatically unblocked (i.e., armed); below the setpoint, the system is automatically blocked. The operating status of the AMSAC is indicated in the main control room via a bypass and permissive panel window.

7.8.2.13.3 Indication of Bypasses

Whenever the mitigative capabilities of the AMSAC are bypassed or deliberately rendered inoperable, this condition is indicated in the main control room. In addition to the operating bypass, any manual maintenance bypass is indicated via the AMSAC general warning sent to the main control room.

7.8.2.12.4 Means for Bypassing

A permanently installed system bypass selector switch is provided to bypass the system. This is a two-position selector switch with "NORMAL" and "BYPASS" positions. At no time is it necessary to use any temporary means, such as installing jumpers or pulling fuses, to bypass the system.

7.8.2.13 Completion of Mitigative Actions Once Initiated

The AMSAC mitigative actions go to completion as long as the

coincidence logic is satisfied and the time delay requirements are met. If the flow in the feedwater lines is reinitiated before the timer expires and the SG water level increases to above the AMSAC low setpoint, the coincidence logic will no longer be satisified and the actuation signal disappears. If the coincidence logic conditions are maintained for the duration of the time delay, the mitigative actions go to completion. The auxiliary feedwater initiation and the turbine trip signals are latched in at the activated component level through the existing circuits. Deliberate operator action is then necessary to terminate auxiliary feedwater flow, clear the turbine trip signal using the main control board turbine trip reset switch, and proceed with the reopening of the turbine stop valves.

7.8.2.14 Manual Initiation

Manual initiation of the AMSAC is not provided. The capability to initiate the AMSAC mitigative functions manually, i.e., initiate auxiliary feedwater, trip the turbine, and isolate steam generator blowdown and sampling lines, exists at the main control board independent of AMSAC.

7.8.2.15 Information Readout

The AMSAC has been designed such that the operating and maintenance staffs have accurate, complete, and timely information pertinent to the status of the AMSAC. A system level general warning alarm is indicated in the control room. Diagnostic capability exists from the test/maintenance panel to determine the cause of any unanticipated inoperability or deviation.

7.8.2.16 Compliance with Standards and Design Criteria

The AMSAC meets the NRC acceptance criteria contained in 10 CFR 50.62 and the quality assurance requirements contained in NRC Generic Letter 85-06. The AMSAC also complies with the generic designs presented in WCAP-10858-P-A, which have been determined to be acceptable by the NRC for meeting the requirements of 10 CFR 50.62. In addition, the time delay design for the AMSAC associated with the C-20 permissive signal is consistent with Revision 1 to WCAP-10858-P-A, which has been accepted by the NRC.

8.0 ELECTRIC POWER

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CHAPTER 8.0 - ELECTRIC POWER

8.1 INTRODUCTION

8.1.1 DESCRIPTION OF UTILITY GRID

The Alabama Power Company (APC) is a part of the Southern electric system and, as such, its lines interconnect with the utility grids of the Georgia Power Company, Gulf Power Company, Mississippi Power Company, and the Southern Electric Generating Company, and with the Tennessee Valley Authority.

The interconnections for 1983 are shown on figure 8.1-1.

The network interconnections between Joseph M. Farley Nuclear Plant (FNP) and the Southern electric transmission system consist of five high-voltage transmission lines and a 500/230-kV autotransformer connection for Units 1 and 2. These lines approach the site on two separate rights of way designed and located to minimize the likelihood of their simultaneous failure. The lines also interconnect the FNP with other major sources of generation in the system. Figure 8.2-1 shows the connections of Unit 1, Unit 2, and the transmission lines to the 500-kV and the 230-kV switchyards at the plant. A detailed description of the transmission system is given in subsection 8.2.1.

8.1.2 DESCRIPTION OF ONSITE ELECTRIC SYSTEM

Each unit is provided with two unit auxiliary transformers (1A and 1B for Unit 1, 2A and 2B for Unit 2) as shown on figures 8.3-1 and 8.3-49. The unit auxiliary transformers are capable of supplying power to 4.16-kV buses A, B, C, D, and E of each unit as shown on figures 8.3-1 and 8.3-49. However, under normal operating conditions only 4.16-kV buses A, B, and C of each unit are powered from the unit auxiliary transformers (1B and 2B). Two startup auxiliary transformers are provided for each unit (1A and 1B for Unit 1, 2A and 2B for Unit 2) as shown on figures 8.3-2 and 8.3-50. The startup auxiliary transformers are capable of supplying power to the nonsafety-related 4.16-kV buses A, B, C, D, and E as well as the safety-related 4.16-kV emergency buses F, G, H, J, K, and L. During normal operations 4.16-kV buses D and E, along with the 4.16-kV emergency buses F, G, H, J, K, and L of each unit, are powered from the startup auxiliary transformers. Each unit is provided with an adequate number of 600-V load centers and 600-V and 208-V motor-control centers.

The onsite emergency power supply for Units 1 and 2 is obtained from five diesel generators (1-2A, 1B, 2B, 1C, and 2C) feeding the 4.16-kV emergency buses. Of these diesel generators, 1-2A, 1C, 1B, and 2B are dedicated for use during design basis events. Diesel generator 2C is dedicated as the alternate ac (AAC) power source for station blackout (SBO) events. Four diesels (1-2A, 1B, 1C, and 2C) were installed when Unit 1 was constructed and one diesel (2B) was installed when Unit 2 was constructed. Three diesel generators (1-2A, 1C, and 2C) are shared between Units 1 and 2. Diesel generators 1B and 2B are lined up to supply emergency power to Units 1 and 2, respectively. Upon loss of offsite power, the diesel generators supply the engineered safeguard loads.

8.3 ONSITE POWER SYSTEMS

8.3.1 AC POWER SYSTEMS

The ac auxiliary system for each unit consists of the 4.16-kV, 600-V, 480-V, 208-V, and 120-V subsystems, each designed to provide reliable electrical power during all modes of plant operation and shutdown conditions. The system for each unit is designed with a sufficient number of power sources and redundant buses to accomplish this. Engineered safeguard circuits are arranged so that the loss of a single bus section results in only single losses of engineered safeguards. A redundant engineered safeguard circuit is available to perform the same function.

The auxiliary system for each unit is capable of starting the largest required drive with the remainder of the connected motor load in service. Each unit is provided with a fast, dead-bus transfer feature which transfers the 4.16-kV buses A, B, and C from the unit auxiliary transformers to the startup auxiliary transformers following a turbine generator trip or reactor trip.

Protective relaying is arranged for selective tripping of circuit breakers after occurrence of an electrical fault. The electrical one-line diagrams for Unit 1 are shown on figures 8.3-1 and 8.3-2, and for Unit 2 on figures 8.3-49 and 8.3-50.

8.3.1.1 Description

8.3.1.1.1 Auxiliary System (4.16-kV)

A. Safety-Related Systems

The 4.16-kV emergency buses, which supply equipment essential for the safe shutdown of the plant, are comprised of six buses F, G, H, J, K, and L for each unit and are supplied from two startup transformers connected to the offsite source during normal and emergency operating conditions. The preferred and the normal power supplies being the same, no transfer to a preferred source is required to be made in the event of an emergency. All components are designed to conform with Class IE Electrical System Design Criteria as defined in IEEE Standard 308. In the unlikely event of a failure of one startup auxiliary transformer, three emergency buses are de-energized and their loss annunciated in the main control room.

The remaining emergency buses of the affected unit are capable of supplying the minimum required engineered safeguards independently as indicated in table 8.3-1. Manual action is required to reenergize the bus from the other startup auxiliary transformer.

No single failure of an active component will remove two startup auxiliary transformers in redundant circuits at one time. The capacity of the transformers and circuit breakers is sufficient to permit full-plant operation with one transformer out of service.

Each 4.16-kV emergency bus, except K and L (which are considered extensions of buses F and G) is equipped with a set of undervoltage relays which provide protection against a loss of voltage condition. Upon recognition of a loss of voltage on a 4.16-kV emergency bus, these relays, configured in a two-out-of-three coincidence logic, initiate a signal to effect the following:

A. Load shedding.

B. Diesel generator starting (except diesel 2C).

C. Tripping of 4.16-kV preferred offsite power supply breakers.

The undervoltage relays employed for loss of voltage protection are induction disc-type relays with an inverse time trip characteristic set at 78.24 percent of the nominal (4160-V) bus voltage and with the time dial selected in such a way as to avoid nuisance tripping during normal operating conditions.

In addition, each of the 4.16-kV emergency buses F and G is equipped with a second set of undervoltage relays to provide protection against a sustained degraded grid voltage condition. Upon recognition of a sustained degraded grid voltage condition, these additional relays, configured in a two-out-of-three coincidence logic, initiate a signal to trip the 4.16-kV preferred power supply breakers.

The undervoltage relays employed for sustained degraded voltage protection are induction disc-type relays with an inverse time trip characteristic set at 88.34 percent of the nominal (4160-V) bus voltage and with a time dial setting of 1.5, calculated to avoid any nuisance tripping during normal operating

supplying safety-related loads (see figures 8.3-10 and 8.3-11). In addition, spare load center F is provided to supply standby power to load centers D and E under the conditions previously outlined in this section.

Under normal operating conditions, load centers A and C are operated as continuous sections. In the event of loss of offsite power (LOSP), the emergency part of these load centers can be manually disconnected from the normal side and supplied from load centers D and E, respectively, through electrically interlocked breakers. With A and C transferred to D and E, the total load on load centers D and E is within the rated capacity of the individual station service transformer of each bus.

The loss of one 600-V emergency bus or the failure of any redundant component of the emergency system deprives the unit of only part of the equipment associated with that particular function. The remaining operational equipment is adequate for the shutdown of the unit under normal or accident conditions.

Safety-related 600/208-V motor control centers are provided to supply power for safety-related equipment within their related areas. Each safety-related motor control center is fed from a safety-related 600-V load center.

Some nonsafety-related loads are fed from safety-related load centers or motor control centers. Primary protection of the 600/208-V MCC transformers has been selected such that a fault at the terminals of nonsafety-related equipment powered from a safety-related MCC will not result in a total loss of the 208-V MCC bus.

B. Nonsafety-Related System

600-V load centers A, B, C, G, H, I, J, M, N, P, Q, U, V, W, X, Y, and Z supply power for nonsafety-related equipment. Load centers H, J, and N are shared between the two units. 600-V load centers 1O and 1T provide power to the Emergency Operations Facility/Visitors Center. In addition, spare load center F is provided to supply standby power to load centers A, B, C, G, I, M, N, P, and Q under the conditions previously outlined in this section.

600/208-V, 600-V, 480-V, and 208-V motor control centers are provided to supply power to nonsafety-related equipment within their related areas.

8.3.1.1.3 Equipment Rating

A. Transformer

Figures 8.3-1, 8.3-2, 8.3-49, and 8.3-50 show the unit auxiliary transformers and startup transformers electrical arrangement. The unit auxiliary transformers are capable of supplying power to 4.16-kV buses A, B, C, D, and E. The startup transformers are also capable of supplying power to 4.16-kV buses A, B, C, D, and E, along with 4.16-kV emergency buses F, G, H, J, K, and L. The ratings of the transformer are as follows:

3. Rated current of main bus bars:

   - 1200 amperes
   - 2000 amperes
   - 3000 amperes

C. 600-Volt Load Center

Figures 8.3-10 through 8.3-20 show the bus and feeder arrangement of load centers A, C, D, E, F, H, J, K, L, R, and S. The main bus bars, rated at 1600 amperes and 600 volts, are braced for a fault duty of 22,000 amperes ac. The load centers are of metal-enclosed construction and are equipped with three-pole, drawout-type, electrically and remotely controlled air circuit breakers, except load center F, which is equipped with a manually operated breaker. All breakers are rated to interrupt a fault current of 22,000 amperes at 600 volts. The continuous current ratings and trip settings for the breakers are given in the relevant single-line diagrams.

D. 600-Volt and 208-Volt Motor Control Centers

The motor control centers are equipped with combination across-the-line starter and molded-case circuit breakers and have the following ratings:

	600-V MCC	208-V MCC
Breaker interrupting current rms (sym)	18,000 A	18,000 A
Starter size, minimum	No. 2	No. 1
Bus rating	600 A	150 A

The 208-V MCCs obtain their power supply from the associated 600-V MCC through 30-kVA, 45-kVA, or 75-kVA 600- to 208/120-V dry-type transformers. The 600-V and the 208-V MCC form a single lineup. The 600-108/120-V transformer in each MCC is protected on the 600-V side by a molded-case circuit breaker or a disconnect switch and fuses.

8.3.1.1.4 120-Volt Vital Instrument Power System

Four redundant channelized 120 V-ac vital instrumentation distribution panels are provided for each unit to supply power for essential instrumentation and control loads under all operating conditions (see figures 8.3-23 and 8.3-24). Each distribution panel is supplied separately from a static inverter.

The normal power source for each static inverter is the associated Class 1E battery charger via the 125 V-dc switchgear. Loss of the battery charger will not render the normal power source inoperable, as the inverter will automatically begin to draw power from the associated Class 1E battery via the same 125 V-dc switchgear without interruption. In case of inverter failure, overload, or branch fault resulting in inverter output voltage outside the specified limits, the static transfer switch (STS), which is part of the inverter unit, transfers the 120-V vital ac distribution panel to an alternate source: Class 1E CVT. Retransfer of the inverter back to its normal power source can only be achieved manually. Each inverter is also equipped with a maintenance bypass switch (MBS) to be used when the entire unit (inverter and STS) is to be taken out of service for maintenance purposes.

Each of the four redundant channels of nuclear instrumentation, as described in chapter 7, is supplied from a separate channelized distribution panel. Also, each of the four independent and redundant channels of the reactor protection system and engineered safeguards system is supplied from a separate channelized distribution panel. The system is arranged so that any type of single failure within the system will involve only one channel and will not prevent the reactor protection system or the engineered safeguards system from performing its safety function.

Two redundant train oriented 120 V-ac vital instrumentation distribution panels are provided for each unit to supply power to nonchannelized essential instrumentation and control loads under all operating conditions (see figures 8.3-23 and 8.3-24). Each train oriented distribution panel is supplied in a manner similar to the channelized instrumentation distribution panels.

8.3.1.1.5 120 Volt-ac Regulated Instrument Power System

The system provides power for nonessential instrumentation, control, and loads requiring regulated 120 V-ac power. It consists of distribution panels and regulating transformers fed from motor control centers as shown in figures 8.3-23 and 8.3-24.

8.3.1.1.6 208/120 Volt-ac Power System

The distribution cabinets that make up the 208/120 V-ac, 3-phase, 4-wire, unregulated power system derive their supply from a 208-V motor control center or the 120/208-V switchgear located in the turbine building. The system provides power for nonessential instrumentation, small motors (3-hp and less), and other miscellaneous 208- or 120-V loads.

Safety-related loads are supplied from distribution panels fed from safety-related motor control centers. The distribution system is arranged to provide adequate independence and redundancy so that a single failure will not prevent the ESF system from performing its required function. Some nonsafety-related loads are fed from safety-related distribution cabinets. These nonsafety-related loads will not affect the integrity of the safety-related loads.

All other nonsafety-related loads are supplied from distribution panels fed from nonsafety-related motor control centers and various other distribution cabinets fed from the 120/208-V switchgear.

8.3.1.1.7 Onsite Emergency Power Systems

8.3.1.1.7.1 General - The onsite emergency ac power supply for Units 1 and 2 consists of five diesel generators which supply standby power for 4160-V emergency service buses F, G, H, J, K and L of each unit when offsite power is unavailable. These buses provide power to the emergency loads.

The LOSP loads are the emergency loads required to function during the shutdown process of a nonaccident unit when that unit experiences the loss of its offsite power sources.

The engineered safeguard system loads are the emergency loads required to function during the shutdown process of an accident unit.

The emergency loads are divided between the emergency buses of each unit in two balanced, redundant load groups so that the failure of a redundant group does not prevent the safe shutdown of both reactors.

The 4160-V emergency buses F, H, and K of each unit and their associated loads are designated as the redundant load group train A.

The 4160-V emergency buses G, J, and L of each unit and their associated loads are designated as the redundant load group train B.

Diesel generators 1-2A and 1C are assigned to the redundant load group train A, while diesel generators 1B, 2B, and 2C are assigned to the redundant load group train B.

The five diesel generators are of two different sizes, as follows: three 4075-kW diesel generators 1-2A, 1B, and 2B and two 2850-kW diesel generators, 1C and 2C.

The capacity of the diesel generators ensures that sufficient power will be available to provide for the functioning of required emergency loads during the worst loading situations.

The design of the onsite emergency power system is such that the plant meets its licensing basis for all design basis events using only four of the diesel generators, namely 1-2A, 1C, 1B, and 2B.

Therefore, these four diesel generators are dedicated for use during the design basis events.

Diesel generator 2C is dedicated as the alternate ac (AAC) power source for use during station blackout (SBO) events as discussed later in paragraph 8.3.1.1.7.3.

Diesel generator 2C, being the AAC for SBO events, is not considered a candidate for the design basis single failure. However, diesel generator 2C meets all applicable safety-related criteria and thus, it will be available for use during a design basis event if diesel generator 1B or 2B fails.

Each generator is supplied with a high-speed voltage regulator designed to return generator voltage to rated value within an acceptable delay after starting of the largest motor.

Voltage and frequency relays connected to the potential transformers at the diesel generator terminals detect generator-rated voltage and frequency conditions and provide a permissive interlock for the closing of the respective generator circuit breaker.

Interlocks are provided to prevent automatic closing of a diesel generator breaker to an energized or faulty bus.

Interlocks are provided to prevent paralleling of the diesel generator with the startup auxiliary transformers without previous synchronizing by the plant operator.

Each diesel generator is equipped with means for periodic starting to test for readiness and loading, and means for synchronizing the unit onto the bus without interrupting the service.

Diesel fuel oil storage tanks are provided with the capacity to supply enough fuel to operate the diesels necessary to meet the required safety loads for 7 days. The day tank at each diesel engine contains sufficient oil for at least 4 hours of operation. The diesel engine day tanks are replenished from the onsite storage tank by individual supply lines. Redundant means are provided for transferring fuel oil from the storage tanks to the day tanks.

A program designed to maintain diesel fuel oil quality in long-term storage includes adding a small amount of dye to the fuel. The purpose of this dye is to determine by test the time required to achieve a homogeneous mixture in the emergency diesel generator fuel oil storage tanks.

The diesel generators are housed in reinforced concrete, Category I seismic structures. Each unit is completely enclosed in its own concrete cell and isolated from the other units. Each diesel generator has an electrically powered standby warming system which will automatically maintain the engine, cooling water, and lubricating oil temperature at a satisfactory level to allow fast starting of the diesel generator sets. Local annunciation and local indication of malfunctions are provided so that, in the event of a failure, an operator may immediately determine the cause. Two complete and independent starting-air-supply systems with receivers, valves, and fittings are supplied with each diesel. The starting air receivers for each of the starting systems have enough capacity for a minimum of five consecutive starts.

A diesel engine heat exchanger of the shell and tube type is provided for each emergency diesel generator. Cooling water is circulated in a closed loop through the engine lubricating oil cooler, the engine cooling water passages, and the shell side of the heat exchanger by a cooling water pump driven by the engine.

The cooling medium flowing through the tube side is part of the service water system. Each diesel engine heat exchanger can be supplied from two service water headers. The cooling water system for the diesels is discussed in subsection 9.5.5.

Typically, emergency diesel generator engines for this service operate for approximately 3 min at full load without cooling water supply. This provides time, in the event of a dead bus, for the initiation of flow in the service water system.

The dc power required for emergency buses and diesel generators for Units 1 and 2 is provided as shown on figure 8.3-30. Diesel generators 1-2A, 1C, and 2C can receive dc power from either Unit 1 or Unit 2 batteries. Automatic transfer switches, mechanically held, are provided. The transfer is initiated when the selected source is deenergized.

Loss of dc control power for each diesel generator is annunciated locally and in the control room.

During emergency power operation only a limited number of protective devices will lead to a trip of the unit. These will include the following:

- Engine overspeed.
- Lubrication oil pressure low.[a]
- Generator differential.

During periodic testing operation, the conditions causing a trip are as follows:

- Engine overspeed.
- Generator differential.
- Lubrication oil pressure low.
- High jacket water temperature.
- High lubrication temperature.
- Generator under frequency.
- Reverse power flow.
- Loss of excitation.
- Jacket water pressure low.
- Crankcase pressure high.
- Generator overcurrent.

Should a diesel fail to start within 7 s, its starter air supply and the fuel are cut off and an alarm is sounded.

8.3.1.1.7.2 Response to Design Basis Events - During normal plant operation the four design basis diesels 1-2A, 1C, 1B, and 2B are set for emergency operation, each with its mode selector switch (MSS) in Mode 1 position. With this setting the starting, alignment, and loading of these four diesel generators are entirely automatic in all design basis events, with no need for any manual operator action.

These four diesel generators are each uniquely assigned to a redundant train of safe shutdown equipment for one unit in each design basis event.

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a. Two lube-oil pressure switches will be installed and wired in series so that two-out-of-two signals will be necessary to trip.

A safety injection (SI) signal from either unit will start shared diesel generators 1-2A and 1C. Diesel generators 1B and 2B will also start on an SI signal if their corresponding unit is experiencing an accident.

An undervoltage (LOSP) signal on the 4-kV ac train A buses of either unit will start the associated shared diesel generator (diesel generator 1-2A associated with buses 1F and 2F, diesel generator 1C associated with buses 1H and 2H). Diesel generators 1B and 2B will also start upon receipt of an undervoltage (LOSP) signal from their assigned 4-kV ac train B buses (diesel generator 1B assigned to bus 1G and diesel generator 2B assigned to bus 2G).

Diesel generators 1B and 2B are uniquely dedicated to train B of Unit 1 and Unit 2, respectively. Diesel generators 1-2A and 1C are shared between both units and are directly connectable to the units through dedicated breakers, not through bus interties. Diesel generators 1-2A and 1C are dedicated to train A, but there are no design basis events in which diesel generator 1-2A or 1C supplies power to safety loads of both units simultaneously. In all events, diesel generators 1-2A and 1C are assigned to only one of the two units, depending on the event. The Unit 1 and Unit 2 breakers for each of these two diesels are interlocked so as to prevent the diesels from being connected to both units at the same time; therefore, diesel generators 1-2A and 1C are characterized as "shared" only from the point of view of their capability to align to either Unit 1 or Unit 2.

The capacity of the diesel generators and their unit alignment must ensure adequate power for the safe shutdown loads during the worst case loading scenario (LOSP on both units concurrent with a loss-of-coolant accident (LOCA) on one unit). Diesel generators 1-2A, 1B, and 2B each have a continuous rating of 4075 kW (4353 kW for the 2000-h rating), which is sufficient to ensure adequate power for one complete train of normal (LOSP) or accident shutdown loads in one unit. Diesel geneerator 1C has a continuous rating of 2850 kW (3100 kW for the 2000-h rating), which is sufficient to ensure adequate power for one complete train of normal (LOSP) shutdown loads in one unit.

Consequently, the alignment of train B diesel generators 1B and 2B, which remains the same during all design basis events, ensures adequate power for a complete train B of LOSP or accident shutdown loads in each unit.

Of the two train A diesel generators, 1-2A and 1C, diesel generator 1C has sufficient capacity to only provide power to a complete train of shutdown loads of a nonaccident unit (LOSP only). Since a LOCA is assumed to occur on only one unit, the alignment logic of these two train A diesel generators is

designed to ensure that in events involving a LOCA, diesel generator 1-2A aligns to the accident unit and diesel generator 1C aligns to the nonaccident unit. The alignment chosen for these train A diesels in scenarios involving LOSP only is arbitrary (1-2A is aligned to Unit 1 and 1C is aligned to Unit 2), since both diesel generators 1-2A and 1C have sufficient capacity to energize the required loads in these events. Therefore, the alignment logic of the two train A diesel generators, 1-2A and 1C, ensures adequate power for a complete train of required shutdown loads in each unit, with diesel generator 1-2A aligned to Unit 1 and diesel generator 1C aligned to Unit 2 in events involving LOSP only.

When offsite power is not available from the startup auxiliary transformer, the emergengy buses are isolated from those sources and all main-load feeder breakers off these buses are tripped.

The essential emergency loads are automatically energized by the diesel generators in a predetermined sequence with time intervals sufficient to allow the inrush current of large motors to decay and the diesel generator to recover from one load step prior to the application of the next load step. Figures 8.3-35 through 8.3-42 show the loading sequence for diesel generators 1-2A, 1B, and 1C for Unit 1. Similar circuits determine the loading for diesel generators 1-2A, 2B, and 1C for Unit 2.

The loading requirements of the emergency buses, as a function of time for the design basis accident and for the shutdown conditions, are shown in table 8.3-1.

The alignment and maximum loading of each design basis diesel generator during all design basis events are shown in tables 8.3-2 and 8.3-3.

These tables demonstrate that the diesel generators and their alignment to the emergency buses are adequate to supply the emergency loads during the worst loading situations. The tables demonstrate also that the maximum required loads will not exceed the continuous rating of any of the four design basis diesel generators.

The main loads in the loading tables are conservatively based on a detailed analysis of the manufacturer's data (nameplate ratings, horsepower curves) and the field preoperational test results.

For miscellaneous loads, the nameplate rating was used as the design basis loading.

Due to the full redundancy provided by the four diesel generators and the existing full redundancy of the safe shutdown loads, the failure of a complete train in one unit will not prevent the safe shutdown of that unit.

It should be noted that if an emergency situation occurs while a diesel generator is being prepared for test with its MSS in Mode 2 position (remote manual), the automatic signals generated by the emergency situation (SI and/or LOSP) will override the test mode and, therefore, the diesel generator will automatically start and align for the event.

8.3.1.1.7.3 Response to Station Blackout (SBO) - FNP is capable of withstanding and recovering from a total loss of both offsite and onsite emergency ac power sources (called "station blackout") as required by 10 CFR 50.63 for a specified duration. For Farley, this duration was determined to be 4 h. This ability to cope with an SBO is consistent with the guidance contained in Regulatory Guide 1.155 and NUMARC 87-00, including supplement, and therefore meets the NRC acceptance criteria contained in 10 CFR 50.63.

The initiating event is assumed to be a LOSP at a plant site. At a multiunit site such as Farley, the LOSP is assumed to affect all units, while the SBO is assumed to occur in only one unit.

SBO is not a design basis accident (DBA). Therefore, single failures of equipment and other assumptions normally considered for DBAs and analysis need not be considered. The unaffected unit must be able to achieve safe shutdown with a single failure; a DBA need not be considered.

FNP selected the AAC approach for coping with an SBO event and dedicated Class 1E diesel generator 2C as the AAC power source to cope with an SBO event in either unit for the required duration (4 h).

Given this selected approach and the assumptions that must be made (LOSP on both units concurrent with the simultaneous failures of any three of the four diesel generators: 1-2A, 1C, 1B, and 2B), an SBO event at FNP falls into one of the following four configurations:

1. SBO in Unit 1 (LOSP and failure of diesel generators 1-2A and 1B) concurrent with the Unit 2 LOSP and failure of diesel generator 1C.

2. SBO in Unit 1 (LOSP and failure of diesel generators 1-2A and 1B) concurrent with the Unit 2 LOSP and failure of diesel generator 2B.

3. SBO in Unit 2 (LOSP and failure of diesel generators 1C and 2B) concurrent with the Unit 1 LOSP and failure of diesel generator 1-2A.

4. SBO in Unit 2 (LOSP and failure of diesel generators 1C and 2B) concurrent with the Unit 1 LOSP and failure of diesel generator 1B.

In all four of the above configurations, diesel generator 2C will be manually aligned and started from the control room and automatically loaded. The remaining design basis event diesel in the non-SBO unit will be aligned, started, and loaded automatically utilizing its respective logic. The SBO configurations are shown on table 8.3-2A.

An SBO in Unit 1 assumes LOSP on both units concurrent with the failure of both redundant diesel generators 1-2A and 1B in Unit 1. At least one of the two redundant Unit 2 diesel generators 1C and 2B will be available to power a complete train of safe shutdown loads of Unit 2.

A complete train B of safe shutdown loads of Unit 1 will be powered by diesel generator 2C as the dedicated AAC source for this event. The starting and unit selection are performed manually by operator actions from the EPB in the control room. The closing of diesel generator 2C Unit 1 breaker DJ06, as well as the loading of diesel generator 2C with Unit 1 train B safe shutdown loads are automatic. The Unit 1 train B large LOSP shutdown loads are sequenced onto diesel generator 2C by the Unit 1 train B LOSP sequencer. This is identical to the loading of diesel generator 1B with Unit 1 train B large LOSP shutdown loads during a LOSP event in Unit 1.

An SBO in Unit 2 assumes LOSP on both units concurrent with the failure of both redundant diesel generators 1C and 2B in Unit 2. At least one of the two redundant Unit 1 diesel generators 1-2A and 1B will be available to power a complete train of safe shutdown loads of Unit 1.

A complete train B of safe shutdown loads of Unit 2 will be powered by diesel generator 2C as the dedicated AAC source for this event. The starting and unit selection are performed manually by operator actions from the EPB in the control room. The closing of diesel generator 2C Unit 2 breaker DJ06, as well as the loading of diesel generator 2C with Unit 2 train B safe shutdown loads are automatic. The Unit 2 train B large LOSP shutdown loads are sequenced onto diesel generator 2C by the Unit 2 train B LOSP sequencer. This is identical to the loading of diesel generator 2B with Unit 2 train B large LOSP shutdown loads during a LOSP event in Unit 2.

Maximum loading of diesel generator 2C during an SBO event is shown on table 8.3-3. The excess load over the 2000-h rating of 3100 kW is insignificant and is of no concern given the amount of conservatism in the calculation of miscellaneous small loads, the short duration (4 h) of the SBO event, and the operator capability to reduce diesel loading through load management actions.

The above evaluation demonstrates that diesel generator 2C can be characterized as a fully capable AAC power source since it has the capacity to power a complete safety train (train B) of LOSP shutdown loads for one unit, and therefore is able to safely shut down either unit in the event of an SBO in that unit.

Although diesel generator 2C is dedicated the AAC power source, it still may be used during design basis events if diesel generator 1B or 2B fails. Diesel generator 2C does not have the capacity to carry a complete train of ESS loads in one unit in the event of a LOSP and LOCA but it can be used to power partial train B loads.

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8.3.1.1.8 Tests and Inspections

Class IE electric power systems are designed to permit periodic testing of the following aspects of the electric power system:

A. The operability and functional performance of the components of Class IE electric power systems (diesel generators, emergency buses, dc system and 120-V vital instrument power system).

B. The operability of these electric power systems as a whole, and under conditions as close to design as practical, including the full operational sequence that brings these systems into operation.

The 230-kV and 500-kV circuit breakers will be inspected, maintained, and tested on a routine basis. This can be accomplished without removing the generators, transformers, and transmission lines from service.

Transmission line protective relaying will be tested on a routine basis. This can be accomplished without removing the transmission lines from service. Generator, unit auxiliary transformer, and startup auxiliary transformer relaying will be tested when the generator is offline Protective relaying associated with the 4160-V and the 600-V systems have provisions for inservice testing and calibration.

The 4160-V and 600-V circuit breakers and associated equipment can be tested when the individual equipment controlled by the breaker is shut down. The circuit breakers may be placed in the "test" position and tested functionally. Circuit breakers and contactors for redundant or duplicated circuits can be tested in service one at a time without interfering with the operation of the plant.

Automatic transfers of 4160-V buses A, B, and C to the startup auxiliary transformers will be tested periodically during the shutdown of each unit to prove the operability of the system.

Following final assembly and preliminary startup testing, each diesel generator unit has been tested at the site, prior to reactor fuel loading, to demonstrate the capability of the unit to perform up to the limits of design. The following tests have been performed on the diesel generator units to certify the adequacy of the unit for the intended service:

A. Starting tests have demonstrated the capability to attain frequency and voltage within the rated limits and time.

B. Load acceptance tests have demonstrated the capability to accept the desired loads in the desired sequence and time duration.

C. Operation tests have demonstrated the capability of carrying the required loads without exceeding the manufacturer's design limits, in accordance with figures 8.3-26 through 8.3-29.

D. Load rejection tests have demonstrated the capability of rejecting the largest single load without exceeding speeds or voltages which will cause tripping, mechanical damage, or harmful overstresses.

The diesel generator units are tested periodically, in accordance with technical specifications, to demonstrate the continued capability of the unit to perform to the limits of the qualified design.

8.3.1.1.9 Design Criteria

The design criteria used for electrical equipment associated with safety-related systems are as follows:

A. Motors

Motor Size:

The horsepower rating of the motors is based on continuous operation of the driven equipment load without exceeding the NEMA standard temperature rise above the stated ambient temperature.

Motor Starting Torque:

Motors are capable of accelerating the load in accordance with the load-speed torque curves with 75-percent rated motor-nameplate voltage at the motor terminals.

Motors rated 250 hp and above have minimum torque values in percent-of-full-load torque as follows:

Locked rotor torque	- 100 percent
Pull-up torque	- 75 percent
Breakdown torque	- 200 percent

The starting current does not exceed 6.5-times full-load current at rated voltage and frequency.

Motors rated 200 hp and below have torque characteristics as established in NEMA Standards for Designs B and C, without the starting current exceeding 6.5-times full-load current at rated voltage and frequency.

Insulation:

The motor insulation system is designed for the special environmental conditions described in table 3.11-1. The insulation system is a combination of materials and processes which provide high resistance to moisture, radiation, and other contaminants experienced by the motors in specified service conditions.

B. Interrupting Capacity of Breakers

The interrupting capacities of breakers associated with 4160-V switchgear, 600-V load centers, 600-V and 208-V motor control centers, and distribution panels are adequate to interrupt the maximum calculated fault current experienced on the associated circuit. Paragraph 8.3.1.1.3 gives the interrupting capacities for the above equipment.

Relay settings are established to provide coordinated tripping of related feeders and are shown on relay setting sheets. These sheets are controlled design documents similar to engineering drawings and are handled and stored in the same manner.

C. Grounding Requirements

All electrical equipment and building steel, such as motor frames, load centers, lighting cabinets, contactors, conduits, cable trays, transformer tanks, stairs, handrails, etc., are effectively and permanently grounded by direct connection to the building ground bus.

Each floor has its own ground bus, which is connected by a number of vertical conductors to the main ground bus on the grade level.

8.3.1.2 Analysis

The following analysis demonstrates compliance with NRC General Design Criteria 17 and 18, NRC Regulatory Guides 1.6, 1.9, and 1.155, and IEEE Standard 308.

A. Compliance with Criterion 17

The engineered safety features (ESF) system is designed with sufficient capacity, independence, and redundancy to ensure that core cooling, containment

integrity, and other vital functions are maintained in the event of postulated accidents, assuming a single failure.

The engineered safety features ac system of each unit is divided into two separate and redundant subsystems. Each subsystem is comprised of three 4160-V switchgear buses, five 600-V load centers, twenty-one 600-V/208-V motor control centers (five dedicated to Unit 1, seven shared, and seven dedicated to Unit 2), and two 120-V vital instrument power system buses. It has adequate capacity and capability to start and supply the engineered safety feature load necessary to safely shut down the reactor, without exceeding fuel design limits or reactor coolant pressure boundary limits, defined in the technical specifications, during normal operation or any design basis event. The diesel generator units associated with each subsystem are capable of supplying, without exceeding their continuous rating, the loads required for operation during a design basis event and hot shutdown conditions. See paragraph 8.3.1.1.7 and tables 8.3-1 and 8.3-2 for details.

In the event of loss of all onsite power and the failure of an offsite circuit, power for the redundant ESF load group is available through the other offsite circuit. No switching of circuits is necessary to achieve this.

Supply to the ESF buses from the onsite power source during any design basis event is established automatically only if the supply from the network is disconnected. This ensures that any failure or fault in the network will not affect the ESF distribution system.

Complete separation and independence has been maintained between the two train systems so that any single failure in one train will not prevent the other train from performing its required safety function.

As shown in figure 8.3-2, the ESF buses for each unit are connected to the 230-kV switchyard by two physically independent circuits through startup auxiliary transformers. The switchyard is connected to the network by four high-voltage transmission lines for Unit 1 and five high-voltage transmission lines and a 500-/230-kV auto transformer connection for Units 1 and 2. A fault on any component of the two offsite circuits will result only in the loss of power to the associated ESF buses. Power can be restored to

the affected buses from the other startup auxiliary transformer by manual switching.

B. Compliance with Criterion 18

The auxiliary electrical system is designed to permit inspection and testing of all important areas and features, especially those that have a standby function and whose operation is not normally demonstrated. Details of the testing program for the ESF equipment including diesel generators are discussed in subsection 8.3.1.1.8.

Testing of the integrity of safeguard action initiating relays with the unit in service is discussed in subsection 7.2.2.2.1, paragraph 6.

C. Compliance with Regulatory Guide 1.6

The standby ac power system for both units consists of five diesel generator sets, feeding two independent and redundant safety load groups.

Each load group has the capability to provide the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Each diesel can be aligned only to load groups within its own redundant system. Diesel 1-2A automatically aligns itself with the accident unit.

Except for the following, no provision has been made in the design to permit a load or a bus to swing between redundant power sources:

- Component Cooling Water Pump B
- Safety Injection Pump B
- Service Water Pump C

In each of the above cases, a breaker and a disconnect switch separate the two trains so that a failure of either one will not affect the redundant train.

Key interlocking (see figure 8.3-32 through 8.3-34) ensures alignment to one train only.

The design of the ESF system meets the requirements of the guide and particularly the following:

1. No provisions exist for automatically paralleling two diesel generators. Although it is possible to parallel two diesels manually during testing,

operating procedures specifically call for testing one diesel at a time. Parallel operation under the above condition is not considered feasible.

2. No provisions exist for automatically transferring loads between redundant power sources.

D. Compliance with Regulatory Guide 1.9

The selection of diesel generators 1-2A, 1B, 1C, 2B and 2C conforms with the purpose of this guide to provide an adequate power source for meeting the starting and continuous power requirements of the safety-related loads. The ratings are discussed in paragraph 8.3.1.1.7.

The sequencing of large loads at 5-s intervals allows large motors to accelerate before the succeeding loads are applied. Dynamic simulations (based on verification testing) are used to model the response of critical components (generators, buses, and loads). The results are evaluated to verify satisfactory performance of the required loads. Engineering evaluations verify that the MCC contactors and other relay type components can function as required during the first load step. After the first load step, the decreases in frequency and the MCC voltages are limited to 95 and 60 percent of nominal, respectively. This ensures that the succeeding voltage dips will not affect the continued operation of the MCC contactors and other relay type components.

Prototype qualification tests on 4075-kW diesels and actual tests conducted on 2850-kW units indicate that the diesel generators are capable of starting and accelerating the required ESF loads to rated speed in accordance with the sequence shown in tables 8.3-1 and 8.3-2.

As stated in paragraph 8.3.1.1.8, preoperational tests will confirm the above test results.

E. Compliance with IEEE 308

All components of the Class IE electric system (discussed in paragraph 8.3.1.1) are designed to meet their functional requirements under conditions produced by the design basis events.

The Class IE electric system has been designed to voltage and frequency limits for proper functional requirements of the ESF equipment.

All incoming circuits to ESF buses are monitored in the control room through breaker position indication lights and/or by analog indicators. Abnormal conditions of these circuits are annunciated in the control room. Separate status indication lights are provided for monitoring each diesel generator and its associated equipment.

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Complete separation and independence has been maintained between all redundant systems so that any component failure in one ESF load group will not disable any component in the other ESF load group. See subsection 8.3.1.4 for a detailed analysis of Independence of Redundant Systems.

The Class IE electric equipment was purchased and installed under a strict quality assurance program described in subsection 17.1.2. Certified records of quality assurance inspections and tests performed during production were obtained from the manufacturer.

The ESF 4160-V switchgear, 600-V load centers, and 600-V motor control centers have been qualified by both tests and successful application under similar operating conditions. Current-carrying capability and fault interrupting tests have been successfully performed on prototypes in accordance with applicable ANSI standards listed in subsection 8.1.4. Standard production tests were performed on the above equipment assemblies in accordance with the same ANSI standards.

Class IE equipment has been qualified to meet seismic requirements by either tests or analyses, or by a combination of both. This is discussed at length in section 3.10.

Each component of the Class IE electric system in one train has a redundant component in the other train. A single failure within a train, therefore, will not prevent satisfactory performance of the minimum ESF loads required for safe shutdown and for maintaining the plant in a hot shutdown condition.

The Class IE electric systems are designed to preclude a common mode failure for two or more diesel generator units under conditions of a design basis event.

Provision has been made to disconnect the non-Class I equipment from the Class IE systems by Class I breakers.

Following a loss of offsite power, the onsite power sources can accept full loads within a time compatible with the ESF loading requirements.

Automatic and manual controls are provided to permit the following:

1. Selecting of the most suitable power source for the Class IE electric system.
2. Disconnecting the appropriate loads when offsite power is not available.

3. Starting and loading the onsite power supply.

Protection systems are provided and designed to isolate failed equipment and to identify the equipment that has failed. For the protection system related to the engineered safety features and essential functions, complete redundancy, independence, and inservice testability have been provided.

Essential instrumentation, control, and power requirements are supplied by reliable, independent, and redundant sources designed to ensure that no single failure will result in loss of power to redundant safety-related equipment.

Table 3.2-1 identifies the safety-related equipment required to operate in a hostile environment. A discussion of the qualification tests and design bases for this equipment is given in section 3.11.

F. Compliance With Regulatory Guide 1.155, "Station Blackout"

FNP selected the AAC approach for coping with an SBO event and dedicated Class 1E emergency diesel generator 2C as the AAC power source to cope with an SBO event in either unit for the required duration (4 h). The remaining four diesel generators -- 1-2A, 1B, 1C, and 2B -- ensure that the plant meets its licensing commitments for all design basis events.

The Farley approach to SBO meets the basic requirements for an "AAC" as stated in RG 1.155, section 3.3.5 and as summarized below:

1. It is connectable to but not normally connected to the offsite or onsite emergency ac power systems.

2. It has minimum potential for common mode failure with offsite or onsite emergency ac power sources.

3. It is available in a timely manner after the onset of an SBO. The time required for making this equipment available should not be more than 1 h; therefore, plants using the AAC approach must assess their ability to cope for 1 h. However, if an AAC power source can be shown by test to be available within 10 min of the onset of SBO, then no coping assessment is required.

   The 10-min requirement is meant to cover the period between the time when the operator realizes that an SBO has occurred and the time when the AAC source is ready for loading the shutdown loads. When actions from the control

room are unsuccessful in restoring offsite or onsite emergency ac power, the onset of SBO has been verified. If the AAC source can be started and ready for loading within the next 10 min, taking all actions from within the control room, the 10-min criterion is met.

4. It has sufficient capacity and reliability to operate the systems necessary for coping with an SBO for the time required to bring the plant to and maintain it in safe shutdown. Therefore, the AAC source must power all the shutdown loads, which would normally be powered by the onsite emergency ac source(s) in the event of an LOSP.

   An AAC power source serving a multiunit site where onsite emergency ac sources are not shared between units should have, as a minimum, the capacity and capability for coping with SBO in any of the units. If the onsite emergency ac sources are shared between units, the AAC power source(s) should have the capacity and capability to ensure that all units can be brought to and maintained in safe shutdown.

   At multiunit sites, where the combination of onsite emergency ac sources exceeds the minimum redundancy requirements for normal safe shutdown (non DBA) of all units, one of the existing onsite emergency ac sources may be used as an AAC power source provided it meets the applicable criteria for an AAC source. Also, an existing onsite emergency ac source could qualify as an AAC source on the basis of excess capacity provided specific modifications to enhance connectability are made.

   If an existing Class 1E emergency diesel generator is used as an AAC power source, this existing Class 1E diesel generator must continue to meet all applicable safety-related criteria.

Paragraph 8.3.1.1.7.3 demonstrates Farley compliance with the above regulatory requirements.

8.3.1.3 <u>Conformance With Appropriate Quality Assurance Standards</u>

To ensure conformance with requirements of appropriate quality assurance standards and criteria such as IEEE Standards and NRC Criteria B-10 CFR 50, a field quality control program is being enforced by use of written field quality control procedures, checklists, and planned periodic audits.

A. Receipt

The installation prerequisites of the above standards, and criteria for electrical materials and equipment in both the ac and dc power systems, are being complied with by field receiving, inspection, and documentation procedures to verify conformance with specifications and drawings on receipt of equipment at the jobsite.

B. Storage

To preserve their integrity and prevent physical, mechanical, and/or electrical damage while in storage, an inspection and maintenance program is enforced by written procedures and manufacturer's recommendations.

C. Installation

An inspection program is also being enforced to ensure that the equipment is being located, installed, assembled and/or connected in strict accordance with latest approved-for-construction drawings,

installation specifications and field quality control procedures.

The field quality control inspection program for equipment and material installation consists of checking for the required separation of redundant engineered safeguards, reactor protection, and the balance of Class IE electrical system cables and components, and for proper termination and marking of these cables.

D. Testing

A test program to verify the quality and performance of Class I and IE instrumentation and electrical equipment is being planned and implemented. Procedures and instructions are being prepared to ensure that tests are performed in accordance with the latest specifications and requirements. A system will be established whereby construction testing procedures and instructions are prepared and approved by qualified test personnel. Test results will be documented and evaluated by the construction testing department to ensure that all components comply with specified design criteria. Nonconforming equipment will be identified and procedures to eliminate the nonconforming situation will be initiated.

Tests during construction will include, as appropriate, electrical continuity and resistance, phase rotation, proper circuit functioning, pressure tests, and other tests as necessary to assure equipment quality.

A procedure to ensure that test equipment meets required standards of accuracy will be enforced. Test instruments will undergo a periodic calibration and will be marked to indicate the date of the next required calibration. Test control will be established so that any construction or other work affecting the tests will be completed prior to the conduct of the test. Final construction verification will be conducted to ensure that all temporary connections have been removed, all deficiencies have been resolved, installation is in accordance with specifications, deterioration has not reduced quality, and equipment and system functions are in accordance with design.

The system for each unit consists of two 125 V-dc switchgear assemblies, three 125 V-dc battery chargers, two 125 V-dc batteries, and six dc distribution cabinets. Each 125 V-dc bus is supplied from one of the battery chargers with one battery floating on the bus. The 125 V-dc system is ungrounded and is equipped with ground detectors installed in each switchgear for continuous monitoring.

2. The separate 125 V-dc system for the service water area consists of two independent and redundant subsystems. Each subsystem consists of two battery/charger sets and two dc distribution panels. Either battery/charger set is capable of providing 100-percent power to both dc distribution panels while recharging its batteries. One battery/charger set provides power to the dc distribution panels while the other is on standby. The active battery/charger set is selected by means of a manual selector switch. The two dc distribution panels feed Unit 1, Unit 2, and shared loads. The majority of these loads are switchgear control supply.

Several loads that are not safety related are also supplied from these systems. These nonsafety-related loads will not affect the integrity of the safety-related systems. All components are designed to conform with Class 1E Power System design criteria as defined in IEEE Standard 308.

B. Nonsafety-Related System (Power Generation)

The dc system for the nonsafety-related loads is comprised of three separate subsystems. Each of these subsystems is independent of and separated from the safety-related dc system.

1. The dc system serving the cooling tower area consists of one 125-V battery, two battery chargers, and one dc distribution panel.

2. The turbine building dc system is comprised of two 125-V batteries, three battery chargers (including one standby), and four dc distribution panels. Two of these distribution panels are located in the auxiliary building and supply dc control power to the nonsafety-related 4160-V switchgear buses 1A, 1B, and 1C, and 600-V load centers 1B,

1I, 1M, and 1N. The two 125-V batteries are connected in series to provide a 250-V source for the emergency dc seal-oil pump and other normal loads.

3. Two 125-V battery chargers and two independent 60-cell batteries located in the high voltage switchyard supply separate dc distribution cabinets to provide power for tripping through primary and secondary relay systems for protection and control of 230-kV and 500-kV circuits associated with the 230-kV and 500-kV systems.

8.3.2.1.1 Safety-Related Batteries

The safety-related battery systems consist of two batteries for the auxiliary building and four batteries for the service water building (two batteries per train). Each battery contains 60 lead-calcium cells electrically connected in series to establish a nominal 125-V power supply. Each cell is of a sealed type, assembled in a shock-absorbing, clear plastic container with covers bonded in place to form a leakproof seal. The batteries are mounted on protected, corrosion-resistant racks. The batteries are floated at 2.20 V per cell. The auxiliary building and service water building battery systems are discussed separately in the paragraphs which follow.

8.3.2.1.1.1 Auxiliary Building Battery System - The auxiliary building station batteries are sized in accordance with the methodology contained in section 6 of IEEE 485-1983. Unit 1 batteries have a rated capacity of 1800 A-h, and Unit 2 batteries have a rated capacity of 1885 A-h based on an 8-h discharge rate to 1.75 V per cell. Under both normal and accident conditions the batteries are designed such that they will provide the voltage required for operation of the nonsafety-related and safety-related components, considering an aging factor of 25 percent and electrolyte temperature within the range of 60°F to 110°F.

8.3.2.1.1.1.1 Normal Operating Conditions - The capacity requirement for the batteries during normal operation is to carry the loads necessary to support plant operation for 2 h. The 2-h duration is based on the time required for the operators to connect the spare battery charger to the system if the connected battery charger fails on either train. During this 2-h period, the redundant train of the dc system with operable battery charger is available for accident mitigation, if required.

The normal load on the batteries during the 2-h period will not exceed 250 A for batteries 1A, 2A, and 2B and 300 A for battery 1B.

8.3.2.1.1.1.2 Design Basis Accident Conditions - The capacity requirement for the batteries during the design basis accident (LOSP or LOSP+LOCA) is to carry the dc loads necessary to support accident mitigation for the time period from the initiation of the LOSP until the battery chargers are reenergized from the emergency diesel generators. After battery charger reenergization, the battery chargers provide the necessary support for the dc loads. The battery chargers are sequenced back onto the emergency diesel generators during the last load sequencing step. Therefore, the batteries are required to provide adequate voltage to all safety-related components without battery charger support for less than 40 s (37 s from LOSP initiation to the last load sequencer step plus or minus the timer tolerance). The design calculations for verifying the adequacy of dc system voltage have been conservatively performed based upon a 1-min voltage. Any failure of the battery charger to be sequenced onto the emergency diesel generators is considered a single failure and the redundant train will be available for safe shutdown. There is no design basis accident scenario where the auxiliary building batteries will be required to supply LOSP or LOSP+LOCA loads for a period greater than 1 min without charger support.

The design basis accident load profile for the auxiliary building batteries is shown below. The load consists primarily of emergency lighting, vital bus inverters, and dc-operated controls and instruments as detailed in table 8.3-6.

Accident Conditions

Time Period (min)	Current (A)
0 to 1	500

Although not a requirement for mitigation of the design basis accidents, the batteries are capable of supplying adequate voltage to all safety-related components for an extended period without battery charger support under the following scenario:

> After initiation of a LOSP or LOSP+LOCA, the batteries will have sufficient capacity to support automatic diesel generator starting and load sequencing, and to support operation of all required safety-related dc loads for 2 h, assuming a battery charger failure occurs after initiation of the LOSP or LOSP+LOCA event. Diesel generator automatic start or load sequencing failures with multiple attempts for diesel generator restart and automatic sequencing are not assumed during the event. The load profile for this scenario is shown below.

The service test for the batteries will be performed using this load profile which envelopes both the normal and design basis accident load profiles.

Time	Load
0 to 1 min	500 A
1 to 120 min	350 A

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trip settings for the breakers are given in the single-line diagrams.

8.3.2.1.4 DC Distribution Panel

Three dc distribution panels connected to each dc switchgear bus supply safety-related loads as indicated in the single-line diagrams. Each dc panel has a 125-V, 400-A main bus braced for 10,000 A. The branch breakers are of molded case design capable of interrupting a fault current of 10,000 A.

8.3.2.1.5 Testing

The dc switchgear, batteries, and battery chargers will be inspected and tested on a periodic basis in accordance with manufacturer's recommendations. The inspection and testing include, but are not limited to, the following:

A. Checking the specific gravity of the electrolyte, voltage, and temperature of the battery cells.

B. Opening and closing functions of breakers.

C. Checking battery charger float voltage and current.

The procedure for battery capacity tests is in accordance with IEEE Standard 450-1980. A service (load profile) test will be performed during each refueling outage, or at intervals of 18 months, with the latter as the governing testing frequency. The service test will be based on the load profiles listed in paragraph 8.3.2.1.1.1.2. A performance test to a terminal voltage of 105 V is also required by technical specifications. A performance test is also required to be performed by the battery vendor prior to shipment.

8.3.2.1.6 Separation and Redundancy Requirements

The equipment design and layout arrangement for the auxiliary building and service water building provides two completely independent and redundant dc systems. The batteries are installed in rooms separated by walls and doors with a Class A fire rating to prevent simultaneous damage to both batteries. Each battery room is ventilated with exhaust fans as described in section 9.4. Fire dampers are provided in the HVAC penetrations of each battery room to limit the effects of a fire to one fire area. For the auxiliary building the dc switchgear, together with the associated battery charger, are located in separate rooms outside the individual battery rooms. The spare charger is located in a room that adjoins, but is separated from, the two other rooms. At the service water building, the battery chargers are located in separate rooms outside the individual battery rooms.

8.3.3 AC AND DC UNINTERRUPTIBLE POWER SUPPLY FOR THE TURBINE-DRIVEN AUXILIARY FEEDWATER PUMP

8.3.3.1 Design Basis

Each plant unit is equipped with a 3-kVA uninterruptible power system (UPS), uniquely assigned to provide a reliable source of control power, ac and dc, for the turbine-driven auxiliary feedwater pump and its associated steam admission and discharge valves.

The NRC acceptance criteria for the design of the UPS are as follows.

The UPS and its associated circuits are physically separated from the two train (A and B)-oriented ac and dc systems and their associated circuits, so that in the event of a high energy pipe break (steam or feedwater) with the loss of all offsite power sources and the worst single failure, two out of the three auxiliary feedwater pumps will be able to start automatically and deliver auxiliary feedwater to the steam generators as required by the design basis for the auxiliary feedwater system defined in section 6.5. Additionally, the power supply meets the requirements of General Design Criterion 44.

The UPS supplies control power to the following loads:

A. 125 V-dc to the turbine-driven auxiliary feedwater pump control panel.

B. 125 V-dc to the turbine-driven auxiliary feedwater steam admission valves HV-3235A, HV-3235B, and HV-3226.

C. 120 V-ac for the turbine-driven auxiliary feedwater pump speed control.

8.3.3.2 System Description

The UPS is rated 3-kVA "continuous service," powered from an emergency ac source 208-V, 1-phase, 60 Hz, and consists of the following components:

A. One battery charger, rated to continuously supply the inverter, fully loaded, while completely recharging the battery within 12 hours after a 2-hour outage. The charger is equipped with an automatic battery recharge/equalize function via an appropriate timing device. This eliminates the maintenance effort

required otherwise to quickly restore the battery to capacity after an outage has been experienced. The automatic timing device initiates the battery equalize/charge after an outage and returns the charger to normal float charge after a predetermined,

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adjustable time period. The input voltage is 208 V-ac, 1-phase, 60 Hz, taken from an emergency Train A MCC. The output voltage is 48 V-dc. A dc ammeter and a dc voltmeter measure the output amperage and voltage, respectively. The charger is complete with input and output circuit breaker protection. The following fault conditions are locally identified by individual indicating lights:

1. High output voltage.

2. Charger failure.

3. AC input failure.

4. Positive ground.

5. Negative ground.

These conditions are remotely alarmed (visual and audible) in the common annunciator window provided in the control room for UPS faults.

B. One inverter, rated to continuously supply 3-kVA, 120 V-ac, 1-phase, 60 Hz with an input supply voltage of 48 V-dc. The inverter is complete with input and output circuit breaker protection and is provided with an automatic transfer contactor which will transfer the load to the bypass ac supply in the event of an inverter failure. The inverter output amperage and voltage are measured by an ac ammeter and an ac voltmeter, respectively. The following fault conditions are locally identified by individual indicating lights:

1. Inverter failure.

2. Low input voltage.

These conditions are remotely alarmed in the common annunciator window provided in the control room for UPS faults.

C. One rectifier to operate from the inverter output or from the bypass ac supply in the event of an inverter failure. The rectifier provides 125-V, 6-ampere, fully regulated dc. A dc ammeter and a dc voltmeter measure the output amperage and voltage, respectively. The rectifier is complete with input and output circuit breaker protection. The following fault

TABLE 8.3-1 (SHEET 1 OF 3)

4160-V EMERGENCY BUSES ESTIMATE OF
MINIMUM LOADING REQUIREMENTS

Seq. Step	Loads	Motor Rating (hp)	Max. Motor Demand (hp)	0 to 1 hour No. of Pumps Run'g	0 to 1 hour Demand hp	0 to 1 hour Demand kW	1 hour to 8 hours No. of Pumps Run'g	1 hour to 8 hours Demand hp	1 hour to 8 hours Demand kW	Beyond 8 hours No. of Pumps Run'g	Beyond 8 hours Demand hp	Beyond 8 hours Demand kW	Remarks
	A. LOSP LOADS 4-kV Buses 1F & 1K, 2F & 2K, 1G & 1L, or 2G & 2L.												
1	Charging pump	900	840	1	600	473	1	600	473	1	600	473	
2	Service water pump	600	600 (583)	1	600 (583)	487 (462)	1	600 (583)	487 (462)	1	600 (583)	487 (462)	Values in parentheses apply to Unit 2 buses.
2	CRDM cooler fan	100	93	1	100	84	1	100	84	1	100	84	
3	Service water pump	600	600 (583)	1	600 (583)	487 (462)	1	600 (583)	487 (462)	1	600 (583)	487 (462)	Values in parentheses apply to Unit 2 buses.
4	CCW pump	400	350	1	350	282	1	350	282	1	350	282	
4	Ctmt. coolers-low speed	125	79	1	54	43	1	54	43	1	54	43	
5	Aux. feedwater pump	450	450	1	450	359	1	450	359	1	450	359	
6	Battery charger	120 kVA	-	1	-	74	-	-	74	-		74	
(a)	Spent-fuel pool pump	100	100	-	-	-	-	-	-	1	100	81	Manually loaded.
(a)	Emergency ac lighting	-	-	-	-	40	-	-	40	-	-	40	Manually loaded.
(a)	Pressurizer heaters	270 kW	-	-	-	-	-	-	270	-	-	270	Manually loaded.
(b)	Auto sequenced load total excluding miscellaneous loads-kW					2332 (2239)			2289 (2239)			2289 (2239)	Values apply to Unit 1 buses only. Values in parentheses apply to Unit 2 buses.

TABLE 8.3-1 (SHEET 2 OF 3)

Seq. Step	Loads	Motor Rating (hp)	Max. Motor Demand (hp)	0 to 1 hour No. of Pumps Run'g	0 to 1 hour Demand hp	0 to 1 hour Demand kW	1 hour to 8 hours No. of Pumps Run'g	1 hour to 8 hours Demand hp	1 hour to 8 hours Demand kW	Beyond 8 hours No. of Pumps Run'g	Beyond 8 hours Demand hp	Beyond 8 hours Demand kW	Remarks
	B. ESS LOADS 4-kV Buses 1F & 1K, 2F & 2K, 1G & 1L, or 2G & 2L												
1	Charging pump	900	840	1	900	709	1	900	709	1	900	709	
2	RHR pump	350	350	1	350	281	1	350	281	1	350	281	
2	Ctmt. spray pump	400	450	1	450	359	1	450	359	1	450	359	
3	Service water pump	600	600 (583)	2	1200 (1166)	974 (924)	2	1200 (1166)	974 (924)	2	1200 (1166)	974 (924)	Values in parentheses apply to Unit 2 buses.
4	CCW pump	400	350	1	350	282	1	350	282	1	350	282	
4	Ctmt coolers-low-speed	125	79	2 [1]	250 [125]	202 [101]	2 [1]	250 [125]	202 [101]	2 [1]	850 [125]	202 [101]	Values in brackets apply to LOSP/SI events in Unit 2 only.
5	Aux. feedwater pump	450	450	1	450	359	-	-	-	-	-	-	
5	Reac. cav. H2 dil. fan	25	25	1	25	22	1	25	22	1	25	22	Load included as a misc load in Unit 2.
6	Battery charger	120 kVA	-	1	-	74	1	-	74	1	-	74	
(a)	Spent-fuel pool pump	100	100	-	-	-	-	-	-	1	100	81	Manually loaded.
(a)	Emergency ac lighting	-	-	-	-	40	-	-	40	-	-	40	Manually loaded.
(a)	H2 recombiner	75 kW	-	-	-	-	-	-	-	-	-	75	Manually loaded.
(b)	Auto sequenced load total excluding miscellaneous loads-kW					3262 (3190) [3089]			2903 (2831) [2730]			2903 (2831) [2730]	Values in parentheses apply to Unit 2 buses. Values in brackets apply to LOSP/SI events for Unit 2 only.

TABLE 8.3-3

DIESEL GENERATOR RATINGS AND MAXIMUM ESTIMATED LOAD

Diesel Generators	Ratings-kW				Maximum Estimated Automatic-Sequenced Loads-kW (table 8.3-2)	
	Continuous Rating	2000 h per Year	300 h per Year	30 min. (in 24-h Period)		
1-2A, 1B, and 2B	4075	4353	4474	4881	1-2A	3754
					1B	4037
					2B	4016
1C and 2C	2850	3100	3250	3500	1C	2822
					2C	3108[a]

a. Diesel generator loading for worst case station blackout (SBO) conditions.

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JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ELEMENTARY DIAGRAM LOADING
SEQUENCER
B1H LOSP SEQUENCER BUS 1H

FIGURE 8.3-43

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JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	ELEMENTARY DIAGRAM LOADING SEQUENCER B1J LOSP SEQUENCER BUS 1J
	FIGURE 8.3-44

9.0 AUXILIARY SYSTEMS

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the spent-fuel assemblies from previous refuelings still remain in the pool, either cooling train can maintain the spent-fuel pool water at or below 170°F. It should be noted that the heat load calculations are based on an 18-month fuel cycle discharge schedule with high density racks installed. It is also assumed that 17 x 17 VANTAGE-5 fuel is discharged for outages after 1995. For additional information, see table 9.1-1.

In addition to the two cases discussed above, the heat load from a third "best-estimate" case is also presented in Table 9.1-1. This case represents a normal full-core offload 180 hours after shutdown. No uncertainty factors were applied to the decay heat results for this case.

9.1.3.1.2 Spent-Fuel Pool Dewatering Protection

System piping is arranged so that failure of any pipeline cannot drain the spent-fuel pool below the water level required for radiation shielding. A depth of approximately 12 ft of water over the top of the stored spent-fuel assemblies will reduce direct radiation to 2.5 mrem/h, which is a factor of two below the threshold level for a radiation area as defined in 10 CFR 20.1003.

In order to perform inspection and/or maintenance of equipment located in the transfer canal, the water in the transfer canal may be transferred to the spent-fuel pool or cask wash area by means of a submersible pump. When pumping to the spent fuel pool, as the water level is increased, the water is transferred to the refueling water storage tank or to the recycle holdup tanks. The level in the spent fuel pool is also monitored to ensure that the level in the spent-fuel pool does not decrease below that required for radiation shielding. When pumping to the cask wash area, which drains to the floor drain tank, the water level in the floor drain tank is monitored to ensure that the tank does not overflow.

9.1.3.1.3 Water Purification

The system's demineralizers and filters are designed to provide adequate purification to permit unrestricted access for plant personnel to the spent fuel storage area and maintain optical clarity of the spent fuel pool water. The optical clarity of the spent-fuel pool water surface is maintained by use of the system's skimmers, strainer, and skimmer filter. To assist in maintaining optical clarity, a temporary in-pool filter such as the Tri Nuclear UF-600 or UFV-250 model can be used.

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spent-fuel pool skimmer filter so that the pressure differential across these filters can be determined.

C. Flow

Instrumentation is provided to measure and give local indication of the flow in the outlet line of the spent-fuel pool filter.

D. Level

A float type level instrument is provided to give an alarm in the control room when the water level in the spent-fuel pool reaches either the high or low level setpoints (6 in. above or below the normal water level in the pool). Local visual indication of the spent-fuel pool level is provided.

9.1.3.3 Safety Evaluation

9.1.3.3.1 Availability and Reliability

The spent-fuel pool cooling and cleanup system has no emergency function during an accident. This manually controlled system may be shut down for limited periods of time for maintenance or replacement of malfunctioning components. In the event of a failure of a spent-fuel pool pump or loss of cooling to a spent-fuel pool heat exchanger, the second cooling train provides 100-percent backup capability, thus ensuring continued cooling of the spent-fuel pool.

9.1.3.3.2 Spent-Fuel Pool Dewatering

The most serious failure of this system would be complete loss of water in the storage pool. To protect against the possibility, the spent-fuel pool cooling suction connections enter near the normal water level so that the pool cannot be siphoned. The cooling water return line contains an antisiphon hole to prevent the possibility of draining the pool. These design features ensure that, for breaks of less than 150 gal/min, the spent-fuel pool will not drain below el 149 ft 8 in. For larger breaks, the spent-fuel pool will not drain down below 140 ft 6 in.

The spent-fuel pool is designed in accordance with Regulatory Guide 1.13 and ensures adequate safety under normal and accident conditions. Pool water losses resulting from normal evaporation and the rupture of suction and discharge piping

have been considered. The possibility of cracking the spent fuel pool liner plate and the surrounding concrete structure is highly unlikely, since it is not possible to bring a sufficiently heavy load, such as a spent fuel cask, into the spent fuel pool area.

The fuel cask crane, discussed in subsection 9.1.4, is prevented by design from moving above or into the vicinity of the spent fuel pool.

The spent fuel pool is a Seismic Category I structure located entirely within the auxiliary building and is not affected by cyclonic winds or tornado-generated missiles.

Makeup water to compensate for spent fuel pool losses is provided by the demineralized water system, discussed in subsection 9.2.3. The reactor makeup water system discussed in subsection 9.2.7 is also available as a Seismic Category I water source in the event that the demineralized water system is unavailable.

9.1.3.3.3 Water Quality

Only a very small amount of water is interchanged between the refueling canal and the spent fuel pool as fuel assemblies are transferred in the refueling process. Whenever a fuel assembly with defective cladding is transferred to the spent fuel pool, a small quantity of fission products may enter the spent fuel cooling water. The purification loop provided removes fission products and other contaminants from the water, by maintaining radioactivity concentrations in the spent fuel pool water at 5×10^{-3} $\mu Ci/cm^3$ (β and γ) or less and thus allowing unrestricted access for plant personnel.

9.1.3.4 Tests and Inspections

Active components of the spent fuel pool cooling and cleanup system are either in continuous or intermittent use during normal system operation. Periodic visual inspection and preventive maintenance are conducted using normal industry practice.

The equipment may utilize a radiation detector as an integral part of the system or depend on water samples being taken from the container and analyzed for radioactivity to monitor radiation levels while the equipment is being utilized.

9.1.5.3 Safety Evaluation

The UT leak testing method does not create a criticality event because all the fuel assemblies are in spent-fuel racks except for the assembly being tested. These racks are already analyzed to ensure proper spacing to maintain K_{eff} less than or equal to 0.95. The movement of the test assembly into the test device is conducted by a detailed procedure which assures that fuel assemblies will not be damaged.

The movement and setup of UT test equipment does not involve any unique plant configurations which could place the plant in an unanalyzed condition. The only potential accidents associated with the setup and movement of the fuel testing equipment involve physical damage to the spent fuel racks, a scenario which has been previously considered.

The UT inspection device is seismically supported to prevent uncontrolled movement on top of the spent-fuel rack during a safe shutdown earthquake.

Pertaining to the alternate detection method, a criticality evaluation was performed for LOPAR fuel. This evaluation is applicable to LOPAR fuel only; it is not applicable to OFA or VANTAGE-5 fuel. This equipment will not be used with OFA or VANTAGE-5 fuel. Operation of this equipment with LOPAR fuel in the self supporting rack will not create the possibility of an accidental criticality. The minimum separation distance between LOPAR fuel assemblies is more than 15 in., which is greater than the minimum separation distance of 4.0 in. required to maintain $K_{eff} \leq 0.95$ for 4.3 wt% LOPAR fuel flooded in unborated water. The enrichment limit of 4.3 wt% is a maximum limit which corresponds to a nominal enrichment of 4.25 wt%. Additionally, the minimum separation distance between a LOPAR fuel assembly in the rack and a LOPAR fuel assembly inadvertently moved up next to the rack is greater than 18 in. Thus, an accidental criticality will not occur for normal operation or accident scenarios associated with this equipment and LOPAR fuel.

The underwater inspection container(s) is designed to withstand overtemperature or overpressure events such that no fuel assembly damage will occur.

The spent-fuel leak detection equipment self supporting rack will withstand SSE loads during a seismic event. Additionally, the operation of this equipment has been verified to be consistent with the spent fuel pool rack heavy loads analysis.

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REFERENCE

1. Utility Resource Associates, "Joseph M. Farley Nuclear Plant Spent Fuel Pool Thermal Analysis," URA-RP-92-013, Rev. 0, February 20, 1992.

Each pump is provided with a five-position control switch with trip, off, auto, run, and close positions. The trip and close positions are momentary positions which spring return to the off and run positions, respectively. During normal power operation the pumps will operate as follows:

A. Switch turned to trip or off position, pump will turn off.

B. Switch turned to auto position, pump will turn on and off responding to the water level fluctuations in the pond.

C. Switch turned to close or run position, pump should run and will result in an alarm if it does not.

During normal operation the pond water level is controlled approximately between el 185 ft 6 in. and el 185 ft. With two units on line, it is expected that six to eight river water pumps will be required to maintain the pond water level. Typically, one or more river water pump(s) on each train will be placed in the auto position. When the water level in the pond drops down to el 185 ft, level switches QSP25LS510 and QSP25LS511 located in the wet pit will automatically turn on the river water pumps in auto position. When the pond level reaches el 184 ft 4 in., an alarm will be annunciated, alerting the control room operator. When the water level in the pond reaches el 185 ft 6 in., the same switches will turn off the river water pumps in the auto position. In the event that these level switches fail to operate, the level transmitters NSP25LT501 and NSP25LT502 located in the wet pit will give a service water structure alarm in the control room at el 185 ft 9 in., at which time the operator will take the appropriate action to shut off the necessary number of river water pumps. These level transmitters also feed into control room pond level indicator and/or recorders.

The service water system takes suction from the service water intake structure located at the storage pond. Stop logs are provided in the service water intake structure which will prevent the water in the wet pit dropping below el 180 ft. Five service water pumps are provided for each unit. The service water system is a nonshared system between the two units except for the intake structure, recirculation line to the pond, recirculation line to the wet pit, and the discharge piping and structure to the river. Therefore, the following discussion which is written for Unit 1 is the same for Unit 2. During normal operation, it is expected that no more than four service water pumps will be required to maintain the plant service water requirements. These service water pumps discharge into a 42-in. header which is divided into 2 trains

by means of valves Q1P16V506 and Q1P16V507, pumps 1A and 1B for one train, and pumps 1D and 1E for the other train. Spare pump 1C can be aligned to either train by means of the above valves. These valves are interlocked so that when one is open the other must be closed, thereby maintaining the train separation. After the header, each train of the service water, supplied via 42-in. lines, proceeds to the service water strainers. These strainers are self cleaning. They are periodically backwashed so that the backwash flows to a sump and is then drained back to the pond. A differential pressure switch across the strainer inlet and outlet alarms in the control room upon an increase in differential pressure. A 36-in. bypass line is also provided around the strainers. After the strainers, the service water from the intake structure proceeds via 42-in. lines to a valve box. From the valve box from each train, the following lines branch off: a 12-in. line to the diesel generator building, a 24-in. dilution bypass line, a 24-in. supply line to the turbine building, and a 30-in. line to the auxiliary building.

The lines going to the diesel generator building supply the cooling water requirements of the diesel generators. This is shown in figure 9.5-18. A description of this portion of the service water system is given in subsection 9.5.5.

The dilution bypass line provides the bypass capability to prevent the overpressurization of the service water system.

The supply line for the turbine building combines into a 24-in. diameter supply line just outside the valve box and supplies cooling water to the turbine and other oil coolers, hydrogen coolers, exciter coolers, generator bus cooling, etc. This portion of the service water system is nonsafety related and is separated from the safety-related portion by means of redundant isolation valves on each train supply line. Each safety class portion of the train, prior to the 24-in. header, is provided with excess flow instrumentation (differential pressure switches Q1P16DPS565, 566, 568, and 569) which will automatically isolate the cooling water supply to the turbine building (valves Q1P16V517, 514, 515, and 516) should the flow to the turbine building exceed approximately 17,500 gal/min from either of the trains. On Unit 1, these valves also automatically isolate upon receipt of the safety injection signal. Additionally, these valves throttle the supply flow to the turbine building during a loss of offsite power (LOSP) event. This throttling function serves to provide a limited amount of cooling water to the turbine building during a LOSP event to support a controlled shutdown/cooldown of the secondary side, while at the same time ensuring maximum cooling water flow is available for the emergency diesel generators. A line from each train will supply cooling water to air compressors and will be upstream of these isolation valves. These lines are 2 in. in diameter and will be provided with means to limit flow in the event their integrity is lost.

The service water system inside the containment and the auxiliary building is shown in figures 9.2-3 and 9.2-4. Each train supplies cooling water to at least one component cooling

9.2.1.4 Tests and Inspections

The river and service water systems are in constant use during plant operation. Therefore, the availability and performance of all normally functioning components is evident to plant operators. In addition to normal maintenance, service water pumps, valves, piping, and supports will be tested/inspected per the requirements of the plant technical specifications.

9.2.1.5 Instrumentation Applications

Instrumentation is provided to indicate whether the system is operating properly. In the event of a LOCA, automatic controls operate the service water system, as required for safety, and the river water system.

Redundant supply trains from the river to the pond are furnished with pressure switches which alarm in the control room in the event of a header break. Valves are manually controlled to isolate the break and divert flow to the redundant train. The service water intake structure will have redundant level indicators to monitor and control the storage pond level as follows:

A. High level (el 185 ft 9 in.) - Alarms operator.

B. Normal high level (el 185 ft 6 in.) - Trips river water pumps if no loss of offsite power.

C. Normal low level (el 185 ft) - Starts river water pump in auto if no loss of offsite power.

D. Low level (el 184 ft 4 in.) - Alarms operator.

E. Divert level (el 184 ft) - Actuates valves to divert river water flow directly to the wet pit.

F. Divert level (el 180 ft) - Actuates valves to divert service water to pond recirculation directly to the wet pit.

G. Low-low level (el 170 ft) - Alarms operator.

9.2.1.6 Service Water Treatment Systems

Prevention of fouling in the plant service water system piping and equipment will be accomplished by intermittent treatment of the service water using appropriate biocides and by the use of appropriate water treatment chemicals if necessary. These may include the systems described below.

A. A gaseous chlorination system, rated at 1000 lb chlorine/24 h, is provided to service both units on an alternating basis.

   Chlorine solution is diffused into the service water flow at the entrance to the service water pump wet pit for each unit.

   A set of electrical contacts on each service water pump discharge header provides a signal from each service water pump to an additive rate controller provided for the unit the pump serves. The signals act as primary control signals for maintaining the chlorine solution concentration at a level necessary to treat the service water flow. The primary signals are combined by the additive rate controller of each unit to generate secondary control signals. The secondary control signals are used to control an electric plug positioner, which controls the chlorine gas flowrate and thus the chlorine solution concentration. The chlorinator is designed to adjust manually the chlorine gas feed rate.

   A control panel, provided to control both units, allows only one unit to enter its chlorination cycle at any time. The control panel controls the electric operated injector water inlet valves and the electric operated chlorine gas suction line valves to provide alternating and intermittent cycles of both units. The chlorination cycles are of the prime-chlorination-flush format. The control panels can be operated manually to allow the units to be chlorinated continuously.

B. A chlorine dioxide generator may be provided to feed both service water systems at once with the generator output split between units or to feed both units on an alternating basis.

   The generator produces chlorine dioxide gas from the reaction of appropriate precursor chemicals as they are mixed in a stream of service water. The resulting chlorine dioxide solution is diffused into the service water flow using the gaseous chlorine delivery system.

radiation level at the component cooling water pump suction headers. If the leaking residual heat exchanger is not isolated from the component cooling loop before the inflow completely fills the surge tank, the surge tank relief valve will discharge to the auxiliary building floor drain tank.

Provision is also made to connect and operate a temporary demineralizer for purification of one component cooling water train.

9.2.2.4 Tests and Inspection

Because the component cooling system is in constant use during plant operation, the availability and performance of all normally functioning components of the inservice train are evident to plant operators. The motor-operated valves in the cooling water supply to the residual heat exchangers that are required to open for post-LOCA heat removal can be tested during power operation.

9.2.2.5 Instrumentation Applications

Low flow alarms in the component cooling water return lines from the seal heat exchanger of the low-head safety injection pumps sound an alarm in the control room. Each of these pumps normally receives flow from that component cooling water pump which is powered from the associated emergency bus.

The component cooling pumps and heat exchangers are fully instrumented for flow, pressure, and temperature so that any degradation in performance can be noted and corrective action taken.

High pressure switches on the return line from each reactor coolant pump thermal barrier cooling coil and a high flow switch on the common return from all three pumps will initiate the rapid closure of isolation valves to isolate the reactor coolant pumps in the event of a leak in the thermal barrier.

Control room indication of surge tank level and high, low, and low-low level alarms in the control room keep the operator informed of any leakage into or out of the component cooling system.

Actuation of the remote manual makeup valves is normally initiated on a low level alarm. Should the surge tank reach low-low level, the cooling water lines for all nonsafeguard equipment are automatically isolated.

The component cooling water is constantly monitored for radioactive leaks into the system by radiation monitors in the pump suction headers.

9.2.3 DEMINERALIZED WATER MAKEUP SYSTEM

9.2.3.1 Design Bases

One demineralized water makeup system is designed to provide demineralized water for Units 1 and 2 during all phases of plant operations. This includes water for filling, flushing, and making up losses during startup, shutdown, refueling, power, and maintenance operations.

This system has no nuclear safety function. It is designed and installed to the requirements of nonnuclear safety (NNS) equipment under the American Nuclear Society safety criteria.

9.2.3.2 System Description

The piping and instrumentation diagram (P&ID) for the demineralized water makeup system is shown on figure 9.2-7. During normal power operation this system receives demineralized water from the plant water treatment system and supplies it as makeup to the following components:

A. Reactor makeup water storage tanks.

B. Boric acid batching tanks.

C. Resin fill tanks.

D. Component cooling surge tanks.

E. Spent fuel pools.

F. Condensate storage tanks.

G. Turbine building auxiliary boiler.

The demineralized water makeup system is also a source of cleaning and flushing water for demineralizers, evaporators, pumps, piping, tanks, and the high-pressure water spray decon unit.

The system is composed of one 200,000-gal demineralized water storage tank, 3 demineralized water makeup pumps, and associated valves, piping, and instrumentation.

TABLE 9.2-11 (SHEET 1 OF 2)

COMPONENT COOLING SYSTEM FAILURE ANALYSIS

Component	Malfunction	Comments and Consequences
Component cooling pump	Fails to start	Three pumps are provided. One pump required for normal, hot shutdown or post-LOCA heat removal.
Motor-operated valve on RHR exchanger inlet	Fails to open post-LOCA	Two valves and heat exchangers are provided. One heat exchanger is required to operate post-LOCA.
Component cooling heat exchanger	Tube leakage	Each unit was hydrostatically tested and freon leak-tested prior to shipment. Leakage is detected by change in surge tank level. Each unit is isolable.
Component cooling system pressure boundary	Failure resulting in abnormal leakage of component cooling water	The system is always valved into two separate flow trains, each of which meets minimum safeguard requirements. Leakage cannot affect both trains. Low operating pressures make ruptures improbable.
Component cooling pumps	Manual valve on a pump suction or discharge line closed	This will be prevented by prestartup and operational check. Further, during normal operation, each pump will be checked on a periodic basis which would indicate if a valve were closed. Annunciation in the control room for low flow for certain equipment cooled by CCW.
Component cooling system vent or drain valve	Left open	This will be prevented by prestartup and operational checks. On the operating train such a situation will readily be assessed by makeup requirements to system. On the second train, such a situation will be ascertained by surge tank level alarms.

9.3.4.1.2.2 Chemical Control, Purification, and Makeup System. The pH control, oxygen control, reactor coolant purification, and chemical shim and reactor coolant makeup of this system are discussed below.

9.3.4.1.2.2.1 The pH Control. The pH control chemical employed is lithium hydroxide. This chemical is chosen for its compatibility with the materials and water chemistry of borated water/stainless steel/zirconium/inconel systems. In addition, Li^7 is produced in the core region due to irradiation of the dissolved boron in the coolant.

The concentration of Li^7 in the reactor coolant system is maintained in the range specified for pH control (table 5.2-22). If the concentration exceeds this range, as it may during the early stages of core life, the cation bed demineralizer is employed in the letdown line in series operation with a mixed bed demineralizer. Since the amount of lithium to be removed is small and its buildup can be readily calculated, the flow through the cation bed demineralizer is not required to be full letdown flow. As an alternate, a non-lithiated mixed-bed demineralizer may be used to remove the lithium. If the concentration of Li^7 is below the specified limits, lithium hydroxide can be introduced into the reactor coolant system via the charging flow. The solution is prepared in the laboratory and poured into the chemical mixing tank. Reactor makeup water is then used to flush the solution to the suction manifold of the charging pumps.

9.3.4.1.2.2.2 Oxygen Control. During reactor startup from the cold condition, hydrazine is employed as an oxygen scavenging agent. The hydrazine solution is introduced into the reactor coolant system in the same manner as described above for the pH control agent. Hydrazine is not employed at any time other than startup from the cold shutdown state.

Dissolved hydrogen is employed to control and scavenge oxygen produced due to radiolysis of water in the core region. Sufficient partial pressure of hydrogen is maintained in the volume control tank so that the specified equilibrium concentration of hydrogen is maintained in the reactor coolant. A pressure control valve maintains a minimum pressure in the vapor space of the volume control tank. This valve can be adjusted to provide the correct equilibrium hydrogen concentration (25 to 50 cm^3 hydrogen at STP/kg water for power operation).

9.3.4.1.2.2.3 Reactor Coolant Purification. Mixed bed demineralizers are provided in the letdown line to provide cleanup of the letdown flow. The demineralizers remove ionic

corrosion products and certain fission products. One demineralizer is in continuous service and can be supplemented intermittently by the cation bed demineralizer, if necessary, for additional purification. The cation resin removes principally cesium and lithium isotopes from the purification flow. The second mixed bed demineralizer serves as a standby unit for use if the operating demineralizer becomes exhausted during operation.

A further cleanup feature is provided for use during cold shutdown and residual heat removal. A remotely operated valve admits a bypass flow from the residual heat removal system into the letdown line upstream of the letdown heat exchanger. The flow passes through the heat exchanger, through a mixed bed demineralizer and the reactor coolant filter to the volume control tank. The fluid is then returned to the reactor coolant system via the normal charging route.

Filters are provided at various locations to ensure filtration of particulate and resin fines and to protect the seals on the reactor coolant pumps.

Fission gases can be removed from the system by purging the volume control tank gas space with hydrogen to the gaseous waste processing system.

9.3.4.1.2.2.4 <u>Chemical Shim and Reactor Coolant Makeup</u>. The soluble neutron absorber (boric acid) concentration is controlled by the BTRS and by the reactor makeup control system. The reactor makeup control system is also used to maintain proper reactor coolant inventory. For emergency boration and makeup, the capability exists to provide refueling water or 4 wt percent boric acid to the suction of the charging pump.

The boric acid is stored in two boric acid tanks. Two boric acid transfer pumps are provided, with one pump normally aligned to provide boric acid to the boric acid blender and the second pump in reserve. On a demand signal by the reactor makeup control system, the pump starts and delivers boric acid to the boric acid blender. The pump can also be used to recirculate the boric acid tank fluid.

The reactor makeup water pumps, taking suction from the reactor makeup water storage tank, are employed for various makeup and flushing operations throughout the systems. One of these pumps also starts on demand from the reactor makeup control system

The excess letdown heat exhanger can be employed either when normal letdown is temporarily out of service to maintain the reactor in operation or when it can be used to supplement maximum letdown during the final stages of heatup. The letdown flows through the tube side of the unit, and component cooling water is circulated through the shell. All surfaces in contact with reactor coolant are austenitic stainless steel, and the shell is carbon steel. All tube joints are welded.

A temperature detector measures temperature of excess letdown downstream of the excess letdown heat exchanger. Temperature indication and high temperature alarm are provided on the main control board.

A pressure sensor indicates the pressure of the excess letdown flow downstream of the excess letdown heat exchanger and excess letdown control valve. Pressure indication is provided on the main control board.

9.3.4.1.2.5.7 Seal Water Heat Exchanger. The seal water heat exchanger is designed to cool fluid from three sources: reactor coolant pump seal water returning to the CVCS, reactor coolant discharged from the excess letdown heat exchanger, and centrifugal charging pump bypass flow. Reactor coolant flows through the tube side of the heat exchanger, and component ooling water is circulated through the shell. The design flowrate is equal to the sum of the excess letdown flow, maximum design reactor coolant pump seal leakage, and bypass flow from one centrifugal charging pump. The unit is designed to cool the above flow to the temperature normally maintained in the volume control tank. All surfaces in contact with reactor coolant are austenitic stainless steel; the shell is carbon steel.

9.3.4.1.2.5.8 Volume Control Tank. The volume control tank provides surge capacity for part of the reactor coolant expansion volume not accommodated by the pressurizer. When the level in the tank reaches the high level setpoint, the remainder of the expansion volume is accommodated by diversion of the letdown stream to the recycle holdup tanks. It also provides a means for introducing hydrogen into the coolant to maintain the required equilibrium concentration of 25 to 35 cm^3 hydrogen (at STP/kg water), is used for degassing the reactor coolant, and serves as a head tank for the charging pumps.

A spray nozzle located inside the tank on the letdown line nozzle provides liquid to gas contact between the incoming fluid and the hydrogen atmosphere in the tank.

A remotely operated vent valve, discharging to the gaseous waste processing system, permits removal of gaseous fission products, which are stripped from the reactor coolant and collected in the gas space of this tank. Relief protection, gas space sampling, and nitrogen purge connections are also provided. The tank can also accept the seal water return flow from the reactor coolant pumps, although this flow normally goes directly to the suction of the charging pumps.

Volume control tank pressure and temperature are monitored with indication given in the control room. Alarm is given in the control room for high and low pressure conditions and for high temperature.

Two level channels govern the water inventory in the volume control tank. These channels provide local and remote level indication, level alarms, level control, makeup control, and emergency makeup control.

If the volume control tank level rises above the normal operating range, one channel provides an analog signal to a proportional controller which modulates the three-way valve downstream of the reactor coolant filter to maintain the volume control tank level within the normal operating band. The three-way valve can split letdown flow so that a portion goes to the recycle holdup tanks and a portion to the volume control tank. The controller would operate in this fashion during a dilution operation, when reactor makeup water is being fed to the volume control tank from the reactor makeup control system.

If the modulating function of the channel fails and the volume control tank level continues to rise, the high level alarm will alert the operator to the malfunction and the letdown flow can be manually diverted to the holdup tanks. If no action is taken by the operator and the tank level continues to rise, the full letdown flow will be automatically diverted.

During normal power operation, a low level in the volume control tank initiates automatic makeup which injects a preselected blend of boron and water into the charging pump suction header. When the volume control tank is restored to normal, automatic makeup stops.

If the automatic makeup fails or is not aligned for operation and the tank level continues to decrease, a low level alarm is actuated. Manual action may correct the situation, or, if the level continues to decrease, an emergency low level signal from both channels opens the stop valves in the refueling water supply line and closes the stop valves in the volume control tank outlet line.

9.3.4.1.2.5.17 Recycle Evaporator Condensate Demineralizer. A sluicable, mixed-bed resin demineralizer is used to remove any boric acid, other anionic impurities such as chloride and fluoride, cationic impurities such as sodium, calcium, magnesium, and aluminum and also any particulate activity carry-over contained in the evaporater condensate. The mixed-bed resin provides the system with the capability to remove a wide range of chemical and radiochemical contaminants resulting in high quality water for plant operations. Although the bed may become saturated with boron at the normally low concentration (<10 ppm) leaving the evaporator, it will still remove most of the boron if the concentration increases because of an evaporator upset. The demineralizer also provides a means of cleanup of the reactor makeup water storage tank contents.

9.3.4.1.2.5.18 Reactor Coolant Filter. The reactor coolant filter is located on the letdown line upstream of the volume control tank. The filter collects resin fines and particulates from the letdown stream. The nominal flow capacity of the filter is greater than the maximum purification flowrate.

Two local pressure indicators are provided to show the pressures upstream and downstream of the reactor coolant filter and thus provide filter differential pressure.

9.3.4.1.2.5.19 Seal Water Injection Filters. Two seal water injection filters are located in parallel in a common line to the reactor coolant pump seals; they collect particulate matter that could be harmful to the seal faces. Each filter is sized to accept flow in excess of the normal seal water flow requirements.

A differential pressure indicator monitors the pressure drop across each seal water injection filter and gives local indication with high differential pressure alarm on the main control board.

9.3.4.1.2.5.20 Seal Water Return Filter. The filter collects particulates from the reactor coolant pump seal water return and from the excess letdown flow. The filter is designed to pass flow in excess of the sum of the excess letdowm flow and the maximum design leakage from the reactor coolant pump seals.

Two local pressure indicators are provided to show the pressures upstream and downstream of the filter and thus provide differential pressure across the filter.

9.3.4.1.2.5.21 Boric Acid Filter. The boric acid filter collects particulates from the boric acid solution being pumped

from the boric acid tanks. The filter is designed to pass the design flow of two boric acid transfer pumps operating simultaneously.

Local pressure indicators indicate the pressure upstream and downstream of the boric acid filter and thus provide filter differential pressure.

9.3.4.1.2.5.22 Recycle Evaporator Feed Filter. This filter collects resin fines and particles from the fluid entering the recycle holdup tanks.

9.3.4.1.2.5.23 Recycle Evaporator Condensate Filter. This filter collects particulates from the boric acid evaporator condensate stream.

9.3.4.1.2.5.24 Recycle Evaporator Concentrates Filter. This filter removes particulates from the evaporator concentrate as it leaves the evaporator.

9.3.4.1.2.5.25 Boric Acid Blender. The boric acid blender promotes thorough mixing of boric acid solution and reactor makeup water for the reactor coolant makeup circuit. The blender consists of a conventional pipe tee fitted with a perforated tube insert. The blender decreases the pipe length required to homogenize the mixture for taking a representative local sample. A sample point is provided in the piping just downstream of the blender.

9.3.4.1.2.5.26 Letdown Orifices. The three letdown orifices are arranged in parallel and serve to reduce the pressure of the letdown stream to a value compatible with the letdown heat exchanger design. Two of the three are sized so that either can pass normal letdown flow of 60 gal/min; the third can pass 45 gal/min. One or both standby orifices may be used with the normally operating orifice in order to increase letdown flow, such as during reactor heatup operations and maximum purification. This arrangement also provides a full standby capacity for control of letdown flow. Orifices are placed in and taken out of service by remote manual operation of their respective isolation valves.

A flow monitor provides indication in the control room of the letdown flowrate. A high flow alarm is provided to indicate flowrates exceeding 140 gal/min.

valved out on the tube side and performs no function during boron storage operation. The temperature of the letdown stream at the point of entry to the demineralizers is controlled automatically by the temperature control valve which controls the shell side flow to the letdown chiller heat exchanger. After passing through the demineralizers, the letdown enters the moderating heat exchanger shell side, where it is heated by the incoming letdown stream before going to the volume control tank.

Therefore, for boron storage, a decrease in the boric acid concentration in the reactor coolant is accomplished by sending the letdown flow at relatively low temperatures to the thermal regeneration demineralizers. The resin, which was depleted of boron at high temperature during a prior boron release operation, is now capable of storing boric acid from the low temperature letdown stream. Reactor coolant with a decreased concentration of boric acid leaves the demineralizers and is directed to the CVCS. Procedures are also available to decrease the concentration of boric acid in the reactor coolant using BTRS demineralizers without using BTRS chillers.

During the boron release operation, the letdown stream enters the moderating heat exchanger tube side, bypasses the letdown chiller heat exchanger, and passes through the shell side of the letdown reheat heat exchanger. The moderating and letdown reheat heat exchangers heat the letdown stream prior to its entering the resin beds. The temperature of the letdown at the point of entry to the demineralizers is controlled automatically by the temperature control valve which controls the flowrate on the tube side of the letdown reheat heat exchanger. After passing through the demineralizers, the letdown stream enters the shell side of the moderating heat exchanger, passes through the tube side of the letdown chiller heat exchanger, and then goes to the volume control tank. The temperature of the letdown stream entering the volume control tank is controlled automatically by adjusting the shell side flowrate on the letdown chiller heat exchanger. Thus, for boron release, an increase in the boric acid concentration in the reactor coolant is accomplished by sending the letdown flow at relatively high temperatures to the thermal regeneration demineralizers. The water flowing through the demineralizers now releases boron that was stored by the resin at low temperature during a previous boron storage operation. The boron-enriched reactor coolant is returned to the reactor coolant system via the CVCS.

Although the BTRS is primarily designed to compensate for xenon transients occurring during load follow, it can also be used to handle boron swings far in excess of the design capacity of the demineralizers. During startup dilution, for example, the resin beds are first saturated, then washed off to the recycle holdup tanks in the CVCS, and then again saturated and washed

off. This operation continues until the desired dilution in the reactor coolant system is obtained.

As an additional function, a thermal regeneration demineralizer can be used as a deborating demineralizer, which would be used to dilute the reactor coolant system down to very low boron concentrations toward the end of core life. To make such a bed effective, the effluent concentration from the bed must be kept very low, close to 0 ppm boron. This low effluent concentration can be achieved by using fresh resin. Use of fresh resin can be coupled with the normal replacement cycle of the resin, one resin bed being replaced during each core cycle.

A. Component Description

Component safety classifications and design codes are given in section 3.2, and a summary of principal component design parameters is given in table 9.3-7.

1. Chiller Pumps

These centrifugal pumps circulate the water through the chilled water loop. One pump is supplied for each chiller.

2. Moderating Heat Exchanger

The moderating heat exchanger operates as a regenerative heat exchanger between incoming and outgoing streams to and from the thermal regeneration demineralizers.

The incoming flow enters the tube side of the moderating heat exchanger. The shell side fluid, which comes directly from the demineralizers, enters at low temperature during boron storage and enters at high temperature during boron release.

3. Letdown Chiller Heat Exchanger

During the boron storage operation, the process stream enters the tube side of the letdown chiller heat exchanger after leaving the moderating heat exchanger. The letdown chiller heat exchanger cools the process stream to allow the thermal regeneration demineralizers to remove boron from the coolant. The desired cooling capacity is adjusted by controlling the chilled water flowrate passed through the shell side of the heat exchanger.

TABLE 9.3-5

CHEMICAL AND VOLUME CONTROL SYSTEM DESIGN PARAMETERS

General Features	Parameter
Seal water supply flowrate for three reactor coolant pumps, nominal (gal/min)	24
Seal water return flowrate for three reactor coolant pumps, nominal (gal/min)	9
Letdown flow (gal/min)	
Normal	60
Maximum	135
Charging flow, excluding seal water (gal/min)	
Normal	45
Maximum	105[a]
Temperature of letdown reactor coolant entering system (°F)	543.5
Temperature of charging flow directed to reactor coolant system (°F)	485
Centrifugal charging pump bypass flow, each (gal/min)	60
Amount of 4 percent boric acid solution required to meet cold shutdown requirements shortly after full power operation (gal)	11,300

a. The original design value of 105 gal/min has been reevaluated for a flow controller limit increase to 130 gal/min and has been found acceptable.

The control room habitability is maintained by continually monitoring radiation levels and smoke concentration inside the room plus continually monitoring radiation levels, including monitoring smoke concentration in the control room air intake duct and computer room return duct and monitoring chlorine concentration at the control room air intake. To prevent inleakage, the control room is provided with normal and emergency pressurization systems designed to maintain greater than 1/8-in. water gauge positive pressure.

Upon toxic gas (chlorine or smoke) detection, an alarm is annunciated in the control room and all fail-safe, airtight isolation valves are closed automatically and remain closed until they are reopened manually. The normal makeup air is cut off. In the case of smoke detection, the operator can manually start the exhaust fan to purge smoke; in the case of chlorine detection, the operator can make use of self-contained breathing apparatus. After a safe level of toxic gas concentration is reached, the operator can draw outside air either from the normal makeup air subsystem or from the emergency (pressurization system) makeup air subsystem.

Upon a high radiation signal from the makeup air inlet, an alarm is sounded in the control room and all isolation valves are closed automatically and remain closed until they are reopened manually. This will result in a loss of positive pressure in the control room. In this event, the operator will manually start one of the redundant pressurization systems and one of the redundant recirculation filtration systems (one 2000-ft^3/min and one 1000-ft^3/min system). The HEPA and 6-in., deep bed, charcoal filter unit in the pressurization system and the HEPA and 2-in. charcoal filters in the recirculation system provide additional assurance that the dose received by control room personnel will not exceed the guidelines of General Design Criterion 19.

Upon receipt of a containment isolation signal, the control room is automatically isolated as described above. The air pressurization system and recirculation system are automatically actuated to maintain the positive pressure and to provide control room cleanup, respectively. A flow control, automatic damper mounted on the bleedoff leg of the pressurization system will respond automatically to the pressure controller in the control room. If the positive pressure drops below 0.25-in. water gauge, the bleedoff damper will automatically restrict the bleedoff flow to divert maximum air supply into the room, in order to achieve a rapid pressure buildup. On a rising room pressure, the bleedoff damper will function in the opposite manner. Therefore, the control room pressure will be sensed and maintained, as well as exhibited for operator information.

Upon receipt of a smoke detection signal from the computer room smoke detector, the computer room HVAC is automatically isolated as described in paragraph 9.4.1.2. In addition, redundant Seismic Category I smoke detectors downstream of the return air subsystem from the computer room will automatically isolate redundant Seismic Category I isolation valves in the computer room recirculation line in the event of smoke recirculation following a computer room fire.

Radiation monitors are provided within the control room boundary. Radiation monitors are also provided within each of the various ventilation systems serving all radiation release points in the plant. These monitors provide indication in the control room and alarm whenever predetermined radiation levels are exceeded. These HVAC systems discharge through the plant vent stack. Additional radiation monitors are provided at the vent stack discharge which will provide a backup means of detecting abnormal plant releases. These monitors are designed to detect releases in excess of the maximum permissible concentrations guidelines established under column 1, Table II, Appendix B to 10 CFR 20.1 - 20.601. Based on the availability and sensitivity of the monitoring systems provided, the operator will have adequate indication and information to evaluate the magnitude of any abnormal plant releases and will manually isolate the control room if required.

An analysis of dose levels in the control room under accident conditions is presented in the applicable sections of chapter 15.

Redundant chlorine detectors will be provided in the circulating water chlorination house designed to transmit an alarm in the control room if the chlorine spill should occur. These alarms will provide sufficient time for operators to put on self-contained breathing apparatus.

An analysis of a chlorine release accident has been performed. (Locations and quantities of onsite chlorine storage are given in table 2.2-3.) Because of the proximity of the closest circulating water chlorination house the analysis was performed for the release of 2 tons of chlorine (the maximum amount of chlorine headered together at one time). Twenty-five percent of the chlorine was assumed to flash to gas. This is analyzed as a puff release. The remainder is assumed to form a 200-ft^2 pool where it evaporates due to the heat load from the sun and from ambient air and ground temperature.

No credit is taken for the channeling of the dense chlorine gas around buildings and along ditches. No credit is taken for an elevated air intake, even though the intake is at an elevation of approximately 177 ft.

electrical equipment rooms, and technical support center have individual air conditioning systems.

Electrical equipment within nonengineered safety feature electrical equipment rooms, computer room, and cable spreading room is not required to mitigate the consequences of a postulated accident; therefore, failure of the associated air conditioning equipment will not affect the safe shutdown capability.

Each engineered safety feature electrical motor control center and 600-V load center room is provided with a Seismic Category I room air cooling unit powered from the same diesel as the motor control center or load center being served. These units are designed to limit the environmental temperature within the electrical equipment room under all postulated accident conditions. Any single failure will not affect safe shutdown capability.

The radwaste area heating, ventilating, and filtration system is discussed in subsection 9.4.3. The remaining systems are discussed below.

9.4.2.1 Design Bases

9.4.2.1.1 Nonradioactive Area Heating and Ventilating System

The nonradioactive area heating and ventilating system is designed to perform the following functions:

A. Remove the sensible heat loss from all equipment and piping in the nonradioactive area during normal plant operation.

B. Limit the maximum ambient temperature to 110°F when the outdoor temperature is 95°F and minimize to 60°F when the outdoor temperature is 20°F.

The lower equipment rooms' heating and ventilating system is comprised of a supply air handling unit containing a fan, a prefilter, an electric heating coil, and an exhaust fan. The system provides and tempers outside air to ventilate the lower equipment rooms. These rooms are included in and are part of the nonradioactive ventilation area.

9.4.2.1.2 Fuel Handling Area Heating, Ventilating, and Filtration System

The fuel handling area heating, ventilating, and filtration system is designed to perform the following functions:

A. Remove the sensible heat loss from all equipment and piping in the fuel handling area during normal plant operation.

B. Limit the maximum ambient temperature to 110°F when the outdoor temperature is 95°F and minimize to 60°F when the outdoor temperature is 20°F.

C. Remove water vapors above the spent-fuel pool to improve visibility of fuel elements within the pool.

D. Provide filtering by routing 100 percent of the spent-fuel pool area exhaust air through prefilter, HEPA, and charcoal filters during normal plant operation.

E. Provide filtering by routing exhaust air from the spent-fuel pool through the penetration room filtration system.

F. Maintain a slightly negative pressure in the spent-fuel pool area with respect to the surrounding areas and outside at all times.

The conformance of the fuel handling area heating, ventilating, and filtration system is presented in table 9.4-5.

9.4.2.1.3 Computer Room HVAC System

The computer room HVAC system is designed to do the following:

A. Provide an environment with controlled temperature and humidity to ensure both the comfort and safety of the operators and the integrity of the computer room components.

B. Provide sufficient air capacity to maintain the computer room and control room at a slightly positive pressure.

The computer room environment is maintained between 60 and 80°F and 50-percent relative humidity. This ensures both the comfort and safety of the operators and the integrity of the computer room components.

9.4.2.1.10 Technical Support Center HVAC System

The technical support center HVAC system is designed to maintain the center at 75°F and 50-percent relative humidity. The system is designed so that the technical support center can be occupied by personnel during plant accident conditions. The system is designed to provide personnel protection from external airborne radiation. The system is powered by a safety-related power supply unit designed to meet safety-related system criteria. The technical support center HVAC system is not classified, however, as safety related, nor is it redundant.

9.4.2.2 System Description

9.4.2.2.1 Nonradioactive Area Heating and Ventilating Systems

The nonradioactive area heating and ventilating system is independent of any other system and includes provisions to supply and exhaust air from the nonradioactive area. The system is shown in figure 9.4-3, and principal components are listed in table 9.4-6. The system consists of one full capacity supply air handling unit complete with hot water heating coil, prefilter, and pneumatically operated dampers and one full capacity exhaust fan, connecting ductwork, and all controls.

The supply air handling unit provides once-through filtered and tempered outside air to the area through supply distribution ductwork when the outside air temperature is above 60°F. When the outside air temperature falls below 60°F, the supply unit will operate with approximately 20-percent outside air and approximately 80-percent recirculated air. A separate 100-percent capacity exhaust fan picks up the exhaust air through exhaust ductwork. A pneumatically operated damper located on the downstream side of the exhaust fan opens in proportion with the supply unit outside and return air dampers.

A. Heating and Ventilating Unit

The supply unit employed in the system is a floor-mounted, horizontal, drawthrough, cabinet type, single zone, air handling unit consisting of centrifugal fan, hot water heating coil, flat type prefilter, outdoor return air dampers with pneumatic operators, and mixing box designed to handle 30,000 sft^3/min, at 2.75-in. water gauge static pressure, 2.6×10^5 Btu/h heating capacity. The fan motor is 20 hp.

B. Exhaust Fan

The system exhaust fan is of the centrifugal type, with a design flowrate of 30,000 sf^3/min, at 1.7-in. water gauge static pressure. The fan motor is rated at 15 hp.

The fans used for the supply air handling unit and exhaust system are designed in accordance with the applicable portions of Air Moving and Conditioning Association (AMCA) 99-67, Standards Handbook, and AMCA 210-67, Test Codes for Air Handling Devices.

Moreover, additional ventilation has been provided to certain rooms in Unit 1 (i.e., rooms 463, 464, 506) and Unit 2 (i.e., rooms 2462, 2463, 2464, 2506) to account for the heat produced by the electrical equipment installed in these rooms. The ventilation air to these rooms is supplied from outside by the nonradwaste air handling unit and the computer UPS supply fan. The computer UPS primary exhaust fan and secondary exhaust fan provide adequate exhaust from these rooms.

The computer UPS primary exhaust fan operates continuously to prevent hydrogen buildup in rooms 464 (Unit 1) and 2464 (Unit 2) and the adjacent areas. The computer UPS supply and secondary exhaust fans run only on an as-required basis during the summer.

9.4.2.2.2 Fuel Handling Area Heating, Ventilating, and Filtration Systems

The fuel handling area heating, ventilating, and filtration system is independent of any other system and includes provisions to ventilate and filter the area atmosphere by the use of a supply heating and ventilating unit, HEPA charcoal filter unit, and exhaust fans. The system is shown in figure 9.4-4, and principal components are listed in table 9.4-6.

One 100-percent capacity supply air handling unit supplies filtered and tempered outside air to two sides of the fuel handling area, namely, the spent-fuel pool and the new fuel storage areas. The air supplied to the pool mixes with the water vapor emanating from the pool surface. An exhaust fan picks up air through a manifold located on the opposite side of the pool and draws it through prefilters, HEPA, and charcoal filters prior to being released through the vent stack. A separate gravity roof vent releases the air from the new fuel storage area.

Whenever irradiated fuel is in the spent-fuel storage pool, one of the penetration room filtration systems is aligned to the spent-fuel pool area. During crane operation with loads over the fuel in the spent-fuel pool and during fuel movement within the spent-fuel pool, both of the penetration room filtration systems are aligned to the spent-fuel pool area to process automatically the spent-fuel area exhausts in the event of a fuel handling accident. During this mode of operation, a fuel handling accident signal from the redundant radiation monitors in the Seismic Category I exhaust line automatically deenergizes the supply and exhaust ventilation fans and isolates the fuel handling area. Isolation of the spent-fuel pool area is accomplished by the automatic closure of one of the redundant isolation dampers located in the Seismic Category I supply and exhaust ductwork that connects the spent-fuel pool area to the fuel handling ventilation mechanical equipment room. The automatic start of the penetration room filtration system and the isolation of the fuel handling area heating, ventilating, and filtration system occur prior to the time radioactive gases from a fuel handling accident can be transported beyond the isolation dampers. This arrangement represents the NRC acceptance criteria for the design of these systems. Two pneumatically operated dampers will normally be left open to connect the fuel handling area with the penetration room filtration system through Seismic Category I ducting. The Seismic Category I fan and filter subsystems of the penetration room filtration system maintain a slightly negative pressure in the fuel handling area. The exhaust air passes through the

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The turbine building heating and cooling system air handling units are supplied with demineralized, chilled water by a primary pump and a closed loop primary piping system. A second pump is provided for standby service. The piping system is protected by an air separator with strainer and a closed expansion tank. The system utilizes the primary secondary circuiting arrangement, the secondary chilled water circuit consisting of 2 parallel arrangement, 70 percent cooling capacity centrifugal water chillers, two 50 percent capacity chilled water pumps, and associated piping, valves, instruments, and controls. For winter shutdown heating, hot water is supplied to the primary piping system by a water heating package consisting of a steam to water heat exchanger, pump, air separator, expansion tank, and associated piping, valves, and instrumentation. During heating, the secondary chilled water loop is isolated from the primary piping system.

Additional design data for the water and air systems are given in table 9.4-11.

A. Heating and Cooling Air Handling Unit

Each area heating and cooling air handling unit is a floor-mounted, horizontal, drawthrough cabinet type, single zone unit consisting of a flat prefilter, finned tube water coils, and a centrifugal fan. Additional design parameters for the heating and cooling units are given in table 9.4-11.

B. Coils

The water coils used for the air handling units are designed in accordance with ARI standard 410-64.

C. Fans

The fans used for the air handling units are double width, double inlet, centrifugal ones designed in accordance with the applicable portions of AMCA 99-67, Standards Handbook, and AMCA 210-67, Test Code for Air Handling Devices.

D. Motors

The motors used for the air handling units are designed in accordance with the applicable portions of National Electrical Manufacturers Association (NEMA) MG1-1969, Standards for Motors and Generators; ANSI C-50.20-1954, Polyphase Induction Motors and Generators; and Institute of Electrical and Electronics Engineers (IEEE) 85-1965, Test Procedure

for Airborne Noise Measurements on Rotating Electric Machinery.

The water analysis room air conditioning unit and associated ductwork and controls are designed to maintain comfort temperatures and humidity for personnel in the water analysis room on the basement floor of the turbine building. The air conditioning unit is a self-contained package with a water-cooled condenser and is controlled by a thermostat located in the room.

9.4.4.2.2 Steam Jet Air Ejector Filtraton System

The steam jet air ejector filtration system is served by one full capacity exhaust fan and one full capacity filtration unit consisting of a prefilter, HEPA filter, and a charcoal filter. The fan and filters are located inside the turbine building. The gaseous releases from the steam jet air ejector vents are routed from the filtration unit to the exhaust line of the turbine building main filtration unit.

A. Fans

The exhaust fan for the steam jet air ejector filtration system is a direct-driven centrifugal fan. Additional design parameters for the fans are given in table 9.4-6. The fans are designed in accordance with the applicable portions of AMCA 99-67, Standards Handbook, and AMCA 210-67, Test Code Air Handling Devices. The flowrate through the steam jet air ejector filtration unit is 1000 sf^3/min, 60 sf^3/min of which is steam jet air ejector effluent.

B. Filters

The filters are composite units consisting of prefilter section, absolute filter section, and impregnated charcoal filter section. Each section is designed as follows:

1. The prefilters have a mean efficiency of 85 percent when tested in accordance with the NIST discoloration test method.

2. The HEPA filter is capable of removing 99.97 percent minimum of particulate matter 0.3 mm or

equipment in the event of failure of primary equipment, redundant controls for activating the ventilating systems in the event of fire within the rooms,and annunciation equipment for alarming the control room in the event of excessively high or low temperatures in the rooms.

The sequence of control is as follows:

A. Upon a rise in temperature in each room, a ventilating thermostat activates the roof intake ventilator and opens all motor-operated dampers in its respective room upon reaching its setpoint.

B. Upon a drop in temperature in each room, the ventilating thermostat deactivates the intake ventilator and closes all dampers in its respective room upon reaching its setpoint.

C. Upon failure of the roof intake ventilator, the roof exhaust ventilator is activated upon its respective thermostat reaching setpoint.

D. Upon a continued drop in temperature in each room, each heating thermostat activates its matching heater upon reaching its setpoint.

E. Each heating thermostat will deactivate its matching heater when the room temperature rises above its setpoint.

F. Either firestat and/or carbon dioxide triggering device in each room, upon reaching its setpoint, activates the exhaust and intake ventilator fan motors and opens all motor-operated dampers in its respective room.

G. Upon reaching its setpoint, the high temperature thermostat will activate the high temperature indicator on the local annunciator and the diesel generator building alarm in the control room.

H. Upon reaching its setpoint, the low temperature thermostat will activate the low temperature indicator on the local annunciator and the diesel generator building alarm in the control room.

Manual override is provided to activate and deactivate each ventilator fan motor, heater, and damper motor as required.

9.4.7.2.3 Oil Storage Rooms Heating and Ventilation Systems

The oil storage rooms ventilation systems consist of redundant power roof exhaust ventilators (two 100 percent units each room) for exhausting fumes from the rooms, one motor-operated wall air intake louver in each room for supplying air to the exhaust ventilators, one fire damper in each room for sealing the louver opening and one fire damper in each room sealing the roof exhaust opening in the event of fire within the room, controls for automatically activating the standby equipment in the event of failure of primary equipment, and controls for deactivating the ventilating systems, closing the wall louver, and closing the fire damper in the event of fire within the rooms.

The sequence of control is:

A. The primary and standby exhaust ventilator in each room operates continuously under normal operating conditions.

B. The wall louver in each room is fully open under normal operating conditions.

C. Upon failure of the primary roof exhaust ventilator, its matching standby exhaust ventilator fan motor in each room is already activated, providing 100 percent ventilation capacity.

D. Ventilation systems operating status is checked periodically according to a predetermined schedule to ensure operational capability.

E. The firestat in each room, upon reaching its setpoint, deactivates all exhaust ventilator fan motors and closes the wall louver and fire dampers in its respective room.

F. Prior to manually activating the fire protection system, a manual HVAC pushbutton station adjacent to the carbon dioxide pushbutton station is activated to provide positive shutdown of exhaust fans and intake louvers to preclude carbon dioxide purge.

Manual override is provided to activate and deactivate each exhaust ventilator fan motor and louver motor as required.

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

1.0 1.1 1.25 1.4 1.6 1.8 2.0 2.2 2.5 2.8 3.2 3.6 4.0

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

9.4.7.3.4 Chlorine Accident

The peak chlorine concentration of the diesel generator air intake is 82,000 ppm at 136 s after the incident. The plot of the concentration is shown in figure 9.4-16.

According to the information supplied by the diesel generator manufacturer, the effect of chlorine on the diesel general performance will not be observed until the chlorine concentration reaches about 15 percent by volume or 150,000 ppm. Since the peak is well below this value, no effect on diesel generator performance is expected.

Also, based on the information supplied by the manufacturer, effects of 86,000 ppm chlorine on engine lube oil, seals, or other organic materials are negligible, due to the short time period of this concentration at the diesel generator intake locations.

9.4.7.3.5 Combustion Products Due to a Fire

Because the plume rise at the intake is much greater than the diesel generator intake height, the combustion products due to a fire will have no effect on diesel generator performance.

9.4.7.3.6 Combustion Products Due to Diesel Exhaust

The result of recirculation from the diesel exhaust to the intake of diesel generators is given in case 7 in table 9.4-12. (See table 9.4-13 for assumptions.) The worst case is the small diesel generator where the total concentration of combustion products is 4939 mg/m^3 or about 26,000 ppm. Effects of such gaseous contaminants are discussed specifically for carbon dioxide in paragraph 9.4.7.3.7 below; there will be no effect on diesel generator performance.

9.4.7.3.7 Carbon Dioxide from the Fire Protection System

According to data supplied by the diesel manufacturer, the combined effect of carbon dioxide on the diesels from both oxygen starvation due to displacement of air from carbon dioxide acting as a combustion depressant is provided in the following table:

Concentration of Carbon Dioxide (percent volume)	Maximum Output of Diesel (percent)
15	100
20	90
25	80

The maximum carbon dioxide concentration is conservatively calculated to be 22,167 ppm as given in table 9.4-14 for case 16. Since this is well below the 150,000 ppm level, no effect on the performance of the diesel generators is expected.

Carbon dioxide is utilized in three types of areas within the diesel generator building:

A. Oil storage rooms (total flooding).

B. Diesel generator rooms (total flooding).

C. Switchgear rooms (local application).

The location of roof ventilators from these rooms with respect to the diesel air intakes is shown in figure 9.4-15.

9.4.7.4 Testing and Inspection Requirements

All components of the heating and ventilation system at the diesel generator building will be tested prior to placing the system in service and periodically thereafter.

Because the heating and ventilation system at the diesel generator building is in use during normal plant operation, the availability of active components is evident to operators, and there is no need for further online testing. Portions of the system not normally in use are periodically tested to ensure operability of the system.

TABLE 9.4-7 (SHEET 2 OF 2)

Component	Malfunction	Comments and Consequences
Engineered safety features pump room coolers	Component failure	One cooler is provided for each charging high head, residual heat removal, containment spray, and auxiliary feedwater pump; two coolers for three component cooling pumps; each spare pump has its own spare pump room cooler The failed cooler will be isolated and repaired; the spare pump and cooler will be started
Motor control center cooler	Component failure	One cooler each is provided for motor control center 1A, 2A and 2B and two coolers are provided for motor control center 1B; each of the coolers operates during post-LOCA operation The failed cooler will be isolated; the ambient temperature in the room being served by the failed cooler will rise until the cooler is repaired and reenergized
600-V load center cooler	Component failure	One cooler each is provided for 600-V load center 1D and 1E; both 600-V load centers operate The failed cooler will be isolated; the room ambient temperature associated with the failed cooler will rise until the cooler is repaired and reenergized

TABLE 9.4-11 (SHEET 1 OF 2)

TURBINE BUILDING HEATING, COOLING, AND STEAM JET AIR EJECTOR FILTRATION SYSTEMS COMPONENT DESIGN PARAMETERS

Cooling and Heating System

Water analysis room air condition unit

Number of units	1
Unit type	Single package
Fan type	Centrifugal
Airflow (ft^3/min)	2300
Static pressure (in. WG)	1.27
Motor (hp)	1
Total cooling (Btu/h)	100,000
Refrigerant	22
Condenser type	Water-cooled
Waterflow (gal/min)	28

Air handling units

Service	Operating floor, el 189 ft
Number of units	4
Unit type	Horizontal, single zone, drawthrough
Components	Fans, coil, filter
Fan type	Centrifugal
Airflow (ft^3/min)	15,350
Static pressure (in. WG)	2.95
Motor (hp)	15
Water coil	Finned tube, chilled water
Total load (Btu/h)	493,000 (cooling); 218,300 (heating)
Waterflow (gal/min)	55 (cooling); 55 (heating)
Service	Mezzanine floor, el 155 ft
Number of units	6
Unit type	Horizontal, single zone, drawthrough
Components	Fan, coil, filter
Fan type	Centrifugal
Airflow (ft^3/min)	16,400
Static pressure (in. WG)	2.43
Motor (hp)	20
Water coil	Finned tube, chilled water
Total load (Btu/h)	530,140 (cooling); 42,500 (heating)
Waterflow (gal/min)	60 (cooling); 60 (heating)

TABLE 9.4-12 (SHEET 1 OF 3)

DESCRIPTION OF CASES EVALUATED IN SAFETY EVALUATION OF THE DIESEL GENERATOR BUILDING

Case	Location and Event	Assumptions	Intakes	Maximum Concentration
1	Chlorine spill, 2 tons at circulating water house about 900 ft NNW of generator building	Wind speed 0.5 mps, direction toward intake, Pasquill F stability	Large	1684 ppm or 5 mg/m ^{3}
2	Chlorine spill, 2 tons at circulating water house about 200 ft NE of generator building	Wind speed 0.5 mps, direction toward intake, Pasquill F stability	Large	81,969 ppm or 242 mg/m ^{3}
3	Combustion products due to fire at underground fuel storage tanks, 8000-gal spill from a tank truck and resulting fire	Wind speed 10 mph, direction toward intakes	Nearest	Insignificant, plume rise is about 192 ft in the distance between fire and intake; intake is at 22 ft above ground
4	Heat or air temperature due to fire at underground fuel storage tanks, 8000-gal spill from a tank truck and resulting fire	Wind speed 10 mph, direction toward intakes	Nearest	Insignificant, plume rise is about 192 ft in the distance between fire and intake; intake is at 22 ft above ground
5	Same as case 1	Same as case 1	Small	1684 ppm or 5 mg/m^3
6	Same as case 2	Same as case 2	Small	80,929 ppm or 239 mg/m^3
7	Combustion products considering effect of diesel exhaust pipes	Wind speed 5 m/s, direction from nearest exhaust to intakes	Both large and small	CO_2 at large intake - 793 mg/m^3 CO_2 at small intake - 903 mg/m^3 H_2O at large intake - 297 mg/m^3 H_2O at small intake - 338 mg/m^3 SO_2 at large intake - 1.2 mg/m^3 SO_2 at small intake - 1.3 mg/m^3 N_2 at large intake - 3246 mg/m^3 N_2 at small intake - 3697 mg/m^3
8	Air temperature rise considering effect of diesel exhaust pipes	Same as case 7, except ambient outside temperature 90°F	Both large and small	ΔT rise at large intake - 8°F ΔT rise at small intake - 5°F
9	Actuation of CO_2 fire protection system in the large diesel generator room, 2800 lb CO_2 released in 1 min	Ventilators, doors, louvers etc., operate normally, wind speed 10 mph, direction toward intake	Large	11,280 ppm

TABLE 9.4-13

RECIRCULATION OF EXHAUST
GAS TO INTAKES - ASSUMPTIONS

Wind speed of 5 m/s

Wind direction from closest exhaust directly to diesel air intake

Complete combustion of fuel

Vertical thermal jet

Final Plume rise calculated by

$$\Delta h = 1.6\ F^{1/3}\ (3.5x)^{2/3}\ (u)^{-1}$$

Where:

x = Distance to final rise (m)

u = Wind speed (m/s)

F = Buoyancy Flux (m^4/s^3)

Plume rise at diesel intake is calculated by:

$$\Delta h = \left[\left(\frac{3}{\beta_m^2}\right)\left(\frac{F_m}{u}\right)\left(\frac{x}{u}\right) + \left(\frac{3}{2\beta^2}\right)\left(\frac{F}{u}\right)\left(\frac{x}{u}\right)^2\right]^{1/3}$$

Where:

β_m = Entrainment Coefficient (Momentum)

β = Entrainment Coefficient (Buoyancy)

F_m = Momentum Flux (m^4/s^2)

F = Buoyancy Flux (m^4/s^3)

u = wind speed (m/s)

x = Distance between Exhaust and Intake (m)

9.5 OTHER AUXILIARY SYSTEMS

9.5.1 FIRE PROTECTION SYSTEM

9.5.1.1 Design Basis

The objective of the fire protection program is to minimize both the probability and consequences of fire. The fire protection program for the Farley Nuclear Plant (FNP) consists of design features, personnel training, operating procedures, and fire fighting equipment provided to reduce the adverse effect of fires on structures, systems, and components, such that in the event of a fire the plant can be safely shut down. This is accomplished by using a defense-in-depth approach aimed at preventing fires, minimizing the effect of any fires that occur, providing fire detection and suppression equipment, and training personnel in fire prevention and fire fighting techniques.

The plant design has been reviewed and design provisions have been included to provide protection of systems required for safe shutdown by physical barriers or spatial separation. Combustibles have been identified and minimized as much as practical. Additionally, provisions have been made for early detection of possible fires, as well as for suppression systems in combustible material areas and backup fire fighting capability in all key plant areas.

9.5.1.2 System Description

A detailed discussion of the fire protection system is provided in Appendix 9B.

9.5.1.3 Design Summary

The fire protection features employed at the FNP are designed to minimize and control the consequences of postulated fires. The level of detection and suppression attention has been established based on an analysis of the fire hazards and potential consequences for the plant. This analysis is described in detail in Appendix 9B.

9.5.1.4 Test and Inspection

The test and inspection requirements related to the fire protection systems and equipment are delineated in Appendix 9B, Attachment C.

9.5.2 COMMUNICATION SYSTEMS

The communication systems include internal (in-plant) and external communications designed to provide convenient and effective operational communications among various plant locations and between the plant and locations external to the plant.

The communication systems are not required for the safe shutdown of the reactor except in response to fires as discussed in Appendix 9B.

9.5.2.1 Design Bases

Various communication systems are provided in the plant to ensure reliable communications during plant startup, operation, shutdown, and maintenance under normal conditions.

9.5.2.2 Description

Interplant and intraplant communications systems consist of telephones, handsets, and loudspeakers. These systems include a plant address system, an intraplant telephone system, an intraplant sound-powered telephone system, microwave communication system, two-way radio communication, an emergency alarm system, the Gra-Ceba Telephone Company system, a teletype system, and the FTS-2000 emergency telecommunications system.

The plant public address system employs transistorized equipment and operates from an ac bus which is powered by the diesel generators upon loss of offsite power. This ensures communication to all areas of the plant, and that the plant alarm may be sounded. The system uses noise canceling dynamic microphone handsets located in strategic positions. Loudspeakers located throughout the plant are powered by individual transistorized amplifiers. Muting facilities are provided where required. In the event of a plant alarm, individual volume control is overridden and the speaker amplifiers are automatically set to the maximum volume. The system has one paging and five party line channels.

The intraplant telephone system employs handsets at strategic areas for private conversation and a coding system of horns and

chimes for calling the essential personnel in any part of the plant.

The intraplant sound-powered telephone system employs a multiple set of circuits based on operating systems to aid in startup and checkout procedures.

The microwave communication is connected to the intraplant telephone switchboard to enable the plant personnel to have dial service to other Alabama Power Company (APC) locations, including the general office in Birmingham, Alabama.

The two-way radio communication system permits communications with APC mobile units and base stations within range of the plant.

The commercial telephone is available in the plant, providing service through connecting (Gra-Ceba Telephone Company) companies to the South Central Bell System.

The teletype communications provide communication between the control room and APC load dispatcher, the Southern Company Services power pool, and other Southern Company plants.

Emergency telecommunications are provided through the Federal Government's FTS-2000 phone system. This system consists of dedicated circuits from within the plant to the NRC Operations Center in Bethesda, MD, and is used in the event of a plant emergency. The circuits are listed below:

ENS - Emergency Notification System
HPN - Health Physics Network
RSCL - Reactor Safety Counterpart Link
PMCL - Protective Measures Counterpart Link
ERDS - Emergency Response Data System (Units 1 and 2)
MCL - Management Counterpart Link
LAN - Local Area Network

The above circuits are all telephone circuits with the exception of ERDS. The ERDS circuit transmits specified plant parameters directly from the Unit 1 and Unit 2 plant computers to the NRC Operations Center in Bethesda, MD.

9.5.2.3 Inspection and Tests

All communication systems with the exception of the sound-powered telephone and emergency alarm system are in operation daily; this allows for testing to ensure that the system is operable. The sound-powered system and emergency alarm system are tested periodically to ensure that they remain operable.

9.5.2.4 Safety Evaluation

The communication system allows maximum flexibility and redundancy to prevent loss of either offsite or intraplant communication from a single failure. Five channels are available for offsite communications: direct microwave telephone lines to general office and APC communication network, Gra-Ceba Telephone Company system, two-way radio, teletype system, and FTS-2000 emergency telecommunications to the NRC. Each of these systems is separate; therefore, failure of any one system does not result in loss of offsite communication. The intraplant system consists of an emergency alarm, public address system, intraplant telephone system, and an intraplant sound-powered telephone system. These systems are separated so that a failure in any one system does not result in a failure of any other system.

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Refer to figures 9.5-1 through 9.5-15 for the communication system.

9.5.3 LIGHTING SYSTEMS

Plant lighting is divided into three categories: normal, essential, and emergency. Normal and essential lighting are both ac, with essential lighting capable of operation from the plant diesels. Emergency lighting is either dc, supplied from station batteries and individual battery packs, or is ac, supplied from emergency lighting transformers. Essential lighting levels are designed to be either equal to or in excess of the levels stipulated in the seventh edition of the Illuminating Engineering Society (IES) standards (1972).

9.5.3.1 Normal Lighting

Normal lighting is provided from 600-V load centers 1A, 1B, and 1C. All fixtures and lamps are rated for 277 V, except for underwater lamps and cranes and lights in the primary access point (PAP) building, which are rated for 120 V. Incandescent lamps are used in areas with floor drains. Fluorescent lamps are used in areas without floor drains.

Industrial and commercial aluminum fixtures are used in the auxiliary building with steel and epoxy-coated cast iron fixtures in the containment.

9.5.3.2 Essential Lighting

Essential ac lighting is provided from the shared 600-V motor control centers 1F and 1G. Essential lighting is in operation at all times. The fixtures are located in close proximity to equipment necessary for safe shutdown of the plant and also in areas that are shared between Units 1 and 2. In case of a loss of offsite power, the essential lighting power is supplied by plant diesels.

9.5.3.3 Emergency Lighting

Emergency dc lighting is designed for personnel safety and plant shutdown. Direct current lighting for the control room is supplied from station batteries. In the event of loss of ac power, the dc lights will automatically switch on to provide the required lighting.

All other dc lighting receives its power from individual self-contained battery packs that are recharged from ac emergency

buses. The battery pack units are capable of providing light for a minimum of 90 min at illumination levels either equal to or in excess of the requirements stipulated in the seventh edition of the IES standards (1972).

Each individual battery pack unit is connected to a 277-V, single-phase, 60-Hz unswitched power supply and is designed as Category I equipment. Emergency lighting in the PAP building is connected to a 120-V, single-phase, 60-Hz unswitched power supply and is designed as Category I equipment. In the event of loss of ac power, the battery pack is switched on automatically to provide the required lighting.

The ac emergency lighting system for the Unit 1 containment is comprised of two uninterruptable power supply units, two step-down distribution transformers, and multiple 120 V-ac sealed-beam units. The UPS units are connected to emergency lighting transformers that supply power to the sealed-beam units. Upon loss of power from a lighting transformer, the associated UPS units will switch on emergency power to the sealed-beam units.

9.5.4 DIESEL GENERATOR FUEL OIL SYSTEM

9.5.4.1 Design Bases

The emergency diesel generator fuel oil system is a Safety Class 2B system designed to supply two of the three 4075-kW diesels and any one of two 2850-kW diesels with fuel oil sufficient for 7 days of operation. The required storage tank capacities are based on the following values:

A. Diesel fuel oil heating capacity of 137,000 Btu/gal at 60°F.

B. Diesel generator fuel oil consumption rates as identified in manufacturer's data.

The system meets the requirements of the single-failure criteria and is Seismic Category I.

The total storage capacity is divided as follows:

A. Diesel fuel oil day tanks sufficient for 4-h operation.

B. Underground storage tanks sufficient for the 7-day requirement, plus an additional 10 percent of the 7-day requirements.

The following design data are applicable to this system:

Number of storage tanks	5
Storage tank capacity	40,000 gal/tank
Storage tank design pressure	Atmospheric
Storage tank design temperature	Ambient
Number of transfer pumps	10 (2 per storage tank)
Transfer pump capacity	20 gal/min
Transfer pump head	40 ft
Number of day tanks	5 (1 per diesel)

The emergency diesel generators use Grade No. 2 fuel oil purchased with the following characteristics:

Property	Allowable Value
- Gravity, degrees API	28 minimum
- Water and sediment, % vol.	0.01 maximum
- Viscosity, kinematic, CST	1.9 - 4.1 at 100°F
or saybolt viscosity, SUS	32.6 - 40.1 at 100°F
- Flashpoint, °F	125 minimum
- Cloudpoint, °F	30 maximum
- Pourpoint, °F	NA
- Distillation:	
50% evaporated, °F	NA
90% evaporated, °F	540 - 640
- Endpoint, °F	700 maximum
- Carbon residue on 10% distillation, % wt.	0.35 maximum
- Sulfur, % wt.	0.50 maximum
- Copper corrosion, 3 h at 122°F	No. 3
- Ash, % wt.	0.01 maximum
- Cetane No.	45 minimum
- Particulate, mg/l	10 maximum

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Day tank capacity	1000 gal (2 tanks) 1325 gal (3 tanks)
Day tank design pressure	Atmospheric
Day tank design temperature	Ambient

The diesel generator fuel oil storage tanks are buried. The specification for these tanks required that they be designed to Article ND-3800 of the American Society of Mechanical Engineers (ASME) Section III nuclear code. Since this article does not specifically cover underground tanks subjected to external soil pressure, the tanks were designed in accordance with the spirit of the article. Section VIII, Division I, was used to obtain allowable external pressure on the tanks.

No code gives specific instructions for calculating the external pressure caused by soil cover. Therefore, the methods developed by the American Water Works Association (AWWA) were used because they have been proven by experience to be adequate. The ASME Section VIII code is much more conservative with regard to required shell thickness than the methods used by the AWWA. Thus, the methods used by AWWA to calculate soil pressure, combined with use of the ASME Section VIII code for shell thickness, give a very safe margin.

Underground piping is protected by a wrapping system which conforms to AWWA C203-66 (standard for coal tar enamel protective coatings for steel pipe). Corrosion protection for the underground storage tank consists of a bitumastic coating, similar to that used for the piping. All underground piping and the underground storage tanks are protected by a cathodic protection system. Provision is made on each tank for periodic draining of any water which might collect.

All tanks, pumps, valves, and piping conform to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class 3, with the exception of the day tanks and the storage tank vent lines. The day tanks conform to the requirements of Section VIII of the code and the vent lines meet B 31.1 criteria.

9.5.4.2 <u>Description</u>

The diesel generator fuel oil system is shown in figure 9.5-16, and additional drawings further describing the physical layout and dimensions of the system are given in figure 9.5-17. The storage system consists of 5 underground storage tanks, any 4 of which have a capacity sufficient for 7 days of operation plus an additional 10 percent. Each tank has the necessary fittings required for the following:

A. Truck fill and water removal.

bearings through holes drilled to the crankshaft. Special drilling in the connecting rod allows the oil to flow up to the piston pin bushing, around the cooling tubes cast in the piston crown, and down the connecting rod to return to the engine sump.

Separate feeds are also taken into the camshaft drive gear, the governor drive, the end camshaft bearings, overspeed trip, and water pump bearings. Oil is also fed to a header along each side of the engine situated behind the fuel pumps. Individual pipes from these headers lubricate the intermediate camshaft bearings, push rods, rollers, and injection pump rollers.

A separate lubricating oil system is provided for lubrication of the valve rocker gear on the cylinder heads. This system incorporates its own pump, driven from the engine camshaft, and a small reservoir tank in which the oil level is automatically controlled. A duplex filter is provided in this system.

9.5.7.5 Safety Evaluation

A failure of one component of the lubricating oil system will not jeopardize the availability of onsite generation for safe shutdown requirements. The lubricating system for each diesel is located totally within the compartment of its associated diesel. Therefore, failure of this system associated with a particular diesel will not affect the integrity of the other diesel generators.

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APPENDIX 9B FIRE PROTECTION PROGRAM

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nonflammable barriers isolate the hazard; appreciable combustible hazards >35 ft but easily combustible; wall or floor opening with 35 ft exposing combustible material in adjacent areas; existence of combustibles adjacent to opposite side of any partition, wall, ceiling, or roof that are likely to be ignited by conduction or radiation.

17. NFPA - National Fire Protection Association.

18. Noncombustible Material

    a. Material which will not ignite and burn when subjected to fire.

    b. Material having a structural base of noncombustible material, as defined in a. with a surface not over 1/16 in. which has a flame spread rating no higher than 50 when measured using American Society of Testing Materials (ASTM) E-84 Test, Surface Burning Characteristics of Building Materials.

19. Operability Requirement - the lowest functional capability or performance level of fire protection equipment required to protect safe shutdown equipment and therefore support safe shutdown of the plant.

20. Safe Shutdown - the normal plant shutdown to hot shutdown ($K_{eff} < 0.99$, 0% rated thermal power, 350°F > T_{avg} >200°F) and subsequently to cold shutdown ($K_{eff} < 0.99$, 0% rated thermal power, $T_{avg} \leq 200$°F), with or without offsite power available, considering damage due only to a single fire event in any area of the plant.

21. Safe Shutdown-Related Systems and Components - systems and components required to bring the plant to, and maintain it in, a cold shutdown condition.

22. Surveillance Requirement - requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained and that the operability requirements will be met.

9B.2 PROGRAM

The objective of the FNP fire protection program is to ensure the capability exists to shutdown the reactor, maintain it in a safe shutdown condition, and minimize radioactive releases to the environment in the event of a fire. The fire protection program is based upon evaluation of potential fire hazards and the effects of postulated fires relative to maintaining the ability to perform safe shutdown functions and minimizing radioactive releases to the environment (the evaluation criteria are presented in section 9B.3 and the results are presented in section 9B.5 of this appendix). This is accomplished by using a defense-in-depth philosophy aimed at protecting against the hazards of fire and protecting against the effects of fire on safe shutdown equipment. The FNP fire protection program

consists of design features, personnel training, operating procedures, and fire fighting equipment that provide this defense-in-depth protection of the public health and safety. Each of these aspects of the fire protection program is presented in more detail in the subsequent sections of this appendix.

In addition, controls over the fire protection organization, the fire brigade training, combustibles and ignition sources, and the plans and procedures for fighting fires meet the NRC supplemental guidelines transmitted by their letter dated August 29, 1977, entitled "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance."

9B.2.1 ORGANIZATION

9B.2.1.1 Corporate Responsibility

The SNC vice president has overall responsibility for fire protection during the design, construction, and operation of APC nuclear power plants. These responsibilities include the direction of the design and the formulation and implementation of the fire protection programs for all nuclear power plants. Construction of buildings and systems related to fire detection, suppression, and extinguishment for new facilities may be delegated to the vice president of Construction. Corporate responsibility for the assessment of the effectiveness of operating nuclear plant fire protection programs is also the responsibility of the vice president.

During the design and construction phase of a nuclear power plant project, detailed system and building designs are developed and equipment specifications are prepared by architect-engineering firms. These firms include among their expertise fire protection engineers who are responsible for ensuring that the various codes and requirements pertaining to fire protection and prevention are met. Fire hazard analyses are performed during the design to ensure that the effects of a postulated fire are properly mitigated.

For operating facilities, the vice president delegates the responsibility of the plant fire protection program to the general manager-nuclear plant of each operating facility. The general manager-nuclear plant ensures that suitable administrative procedures are developed and maintained for the formulation, implementation, and assessment of the effectiveness of plant fire protection programs. The general manager-nuclear plant augments the plant staff as necessary to ensure overall fire protection program administration, sufficient fire brigade manning, and a fire protection training program. Additionally, the general manager-nuclear plant will coordinate offsite training in fire fighting techniques.

9B.2.1.2 Plant Staff

The FNP fire protection program is under the overall responsibility of the general manager-nuclear plant. The responsibility has been delegated to the assistant general manager-plant operations. Formulation and implementation of the program is delegated to the plant managers. The operations manager is responsible for

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fire brigade manning and supervision and fire protection system maintenance and surveillance. The technical manager is responsible for administration of the fire protection program and radiological controls associated with fire protection. A fire marshal under the technical supervisor is responsible for overall fire protection program administration. Visual inspections of fire protection equipment and scheduling of all fire protection equipment surveillance will be coordinated by the fire marshal. Additionally, the fire marshal is directly responsible to the assistant general manager-plant operations for assessment of the effectiveness of fire protection program activities such as fire drills and fire brigade training.

The training manager is responsible for training plant personnel (including fire brigade members) on plant fire protection systems, and training of fire brigades in the fire fighting techniques. An onsite training facility is provided for this training.

Fire brigades are established by administrative procedure such that a designated fire brigade is on call at all times to respond to fire emergencies. The fire brigade chief is a designated shift foreman.

The security supervisor is responsible for furnishing manpower to the fire brigade. In addition, the security supervisor is responsible for fire detection patrols of the plant site as a part of normal security patrols.

9B.2.2 ADMINISTRATIVE CONTROLS

To help ensure that the fire protection program at FNP provides adequate fire protection and reactor safety, certain administrative controls have been established. These controls include an appropriate level of management review of all plant activities including plant modifications to ensure adequate fire protection and reactor safety are maintained; the development of procedures to ensure that inspection, test, maintenance, and modification activities to existing fire protection systems or the design and installation of new fire protection systems are accomplished in accordance with documented instructions, procedures, and drawings; and assurance that inspection, test, maintenance, and modification activities are performed by trained personnel using approved procedures. Additional controls are provided for the restrictions on combustibles, ignition sources, radioactive or hazardous materials, and the operability of systems required to support the safe shutdown of the plant. Each of these controls are discussed in the following paragraphs. Fire protection procedure titles are available in Document Control.

9B.2.2.1 Administrative Procedures

Administrative procedures exist for the purpose of maintaining the performance of the fire protection systems and personnel. The procedures were developed using the guidance of the following publications:

- NFPA 4-1977, "Organization for Fire Services."
- NFPA 4A-1969, "Organization for Fire Department."
- NFPA 6-1974, "Industrial Fire Loss Prevention."
- NFPA 7-1974, "Management of Fire Emergencies."
- NFPA 8-1974, "Management Responsibility for Effects of Fire on Operations."
- NFPA 27-1975, "Private Fire Brigades."

An administrative procedure entitled "General Plant Housekeeping and Cleanliness Control" instructs each employee to clean his assigned work area after completing each work assignment or at the end of each work shift, whichever occurs first, by wiping up oil spills and picking up rags, papers, and other foreign materials. Per this procedure, the fire marshal will inspect each plant area on an unscheduled basis, but at least quarterly, and will note housekeeping status and report the same to the assistant general manager-plant operations or the assistant general manager-plant support, as appropriate. Deficiencies are reported to the appropriate supervisor for corrective action. Additionally, the assistant general managers will conduct a semiannual inspection of all accessible areas to determine compliance with housekeeping instructions. The guidance provided in Regulatory Guide 1.39 was considered in establishing housekeeping procedures.

Administrative procedures also contain instructions on maintaining fire protection during periods when a fire protection system is impaired or during periods of maintenance. For example, these instructions may require the establishment of fire watches or temporary hose connections to water systems. Additionally, administrative procedures require systems to be made operable after all work is completed within an area.

9B.2.2.2 Control of Combustibles

Storage of combustible materials is authorized only in areas separated from safe shutdown-related areas by approved fire walls, doors, and ceilings (i.e., proper storage areas). In accordance with plant administrative procedures, signs are posted by the fire marshal to authorize combustible storage in

9B.2.3 FIRE FIGHTING

9B.2.3.1 Fire Fighting Capabilities

The plant is designed to be self-sufficient with respect to fire fighting activities. The Dothan Fire Department is located approximately 20 miles from the plant and, based on conducted drills, can reach the plant in approximately 30 min to provide assistance in fighting fires. A description of the resources and estimated response times of offsite fire fighting organizations is included in the appropriate procedures.

9B.2.3.2 Fire Brigade

Each operations crew includes a fire brigade with the following members and associated responsibilities:

- One designated shift foreman - directs the fire brigade.
- One systems operator - responsible for emergency breathing equipment.
- One systems operator - responsible for hose station/fire extinguisher equipment.

In order to bring the fire brigade up to five members, two additional personnel are assigned with the following associated responsibilities:

- One chemistry technician - responsible for obtaining monitoring and emergency equipment and assessing toxic chemical hazards.
- One plant guard - responsible for hose station/fire extinguisher equipment.

The fire brigade shall not include three members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency. In addition, the fire brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours to accommodate unexpected absence provided immediate action is taken to restore the fire brigade to the minimum requirements.

If a fire occurs in or affects a radiation controlled area, a health physics technician is responsible for assessing the radiological hazard and providing any personal protective equipment that is required due to the radiological hazard.

The designated shift foreman is the fire brigade chief. Based on extensive senior reactor operator training and on day-to-day working experience in the plant as the designated shift foreman, the fire brigade chief is knowledgeable of the plant layout, plant operation, and current maintenance activities. The designated shift foreman's knowledge of combustibles, extinguishers, fire fighting techniques, and fire fighting strategy are based on the extensive training program described in subsection 9B.2.4. Implicit within the job of directing the fire brigade, the fire brigade chief will communicate with the control room to obtain additional support or supplies as necessary, and will coordinate onsite fire fighting activities performed by offsite fire departments.

The plant fire brigade is equipped with pressure-demand breathing apparatus, portable communications equipment, battery-operated flashlights, and other necessary fire fighting equipment. Spare air cylinders and recharge capability are provided to satisfy the guidelines of Appendix A to BTP APCSB 9.5-1.

9B.2.3.3 Fire Fighting Procedures

An emergency plan implementing procedure titled "Fire Emergencies" discusses action to be taken in the event of a fire in the plant. The following delineated responsibilities are included:

A. The individual discovering a fire will contact the control room and take action commensurate with his fire protection training.

B. The plant operator will immediately notify the shift supervisor and, in the case of a significant fire, sound the plant emergency alarm. The plant operator will place the unit in a safe condition.

C. The shift supervisor will ensure that the unit is placed in a safe condition and maintained in a safe condition. This direct supervision of unit operation during a fire emergency eliminates the need for specifying operations which require control room and shift engineer coordination or authorization in a fire fighting procedure. The shift supervisor will also notify the offsite fire department, if necessary.

D. The designated shift foreman (fire brigade chief) will direct the efforts of the fire brigade and communicate with the control room and assign individuals to advance support supplies to the fire scene based on the nature and duration of the fire. The unit shift foreman will also coordinate onsite fire fighting activities performed by offsite fire departments.

all classes of fire, and the use of fire hose and self contained breathing apparatus is required. Emphasis is also placed on special methods to be used in fighting fires in the radiation controlled area.

C. The methods to be used in fighting fires in buildings, tunnels, and other enclosed areas is required. These methods include the use of plant layout maps in the location of fire fighting equipment, and possible access and egress routes.

D. Indoctrination in plant procedures on fire brigade organization, the use of open flame, fire surveillance, emergency plan implementation for fire emergencies, and abnormal operating procedures for fire emergencies is required. Material included in these procedures includes specific responsibilities of each member of the fire brigade and instructions for responding to a fire emergency, the administrative guidelines for controlling the use of open flame and other sources of ignition, responsibilities for fire surveillance, requirements for maintenance and inspection of fire protection equipment, and plant operator action during a fire emergency.

To maintain proficiency in carrying out assigned duties, each fire brigade member is required to participate in a minimum of two fire drills per year, one lecture per quarter, and one practice session per year. The drills and practice sessions are discussed further in paragraph 9B.2.4.3. The lecture series covers the initial training instruction presented above in a 2-year period. The lecture series also reviews the latest plant modifications, procedure changes, and changes in fire fighting plans. Note that all concerned personnel are notified of the latest plant modifications, procedure changes, and changes in fire fighting plans in accordance with plant procedures.

In addition to the two drills required for each brigade member, the fire brigade on shift during the annual site emergency drill is required to perform as a unit and is evaluated by the emergency drill coordinator. Written examinations are administered to ensure competent response to training activities.

The shift foremen are specifically trained in directing the activities of the fire brigade and also coordinating plant fire brigade activity with offsite assistance groups. Provisions for training offsite fire department personnel in basic radiation principles, typical radiation hazards, and precautions to be taken involving radioactive materials are also included in plant procedures.

The fire brigade equipment and training conform to the recommendations of the National Fire Protection Association and to Appendix A to BTP APCSB 9.5-1.

9B.2.4.3 Fire Brigade Drills

Periodic drills of the fire brigade are performed to evaluate the fire brigade effectiveness, time to respond, and selection, placement, and use of both fixed and portable equipment. Fire drills are conducted on all shifts and are normally unannounced to the members of the fire brigade. The drills are conducted quarterly for each brigade. All drills are preplanned and critiqued to determine the effectiveness of the drill. Liason between the plant and the designated offsite fire department has been established. The designated offsite fire department has been on plant tours and has also been involved in training drills with the plant fire brigade and in separate fire training sessions. The offsite fire department is required to be included in at least one fire drill per year. Each fire brigade member is also required to participate in a yearly practice session where actual fires similar to those which may occur in the plant are extinguished and emergency breathing apparatus is used.

The fire drills consist of different scenarios covering various areas of the plant. The drills simulate the size and type of fire which could reasonably be expected in different areas of the plant with and without automatic suppression system operation. The drills verify the adequacy of plant fire fighting equipment and the ability of the fire brigade chief to develop and implement fire fighting strategies based on his training and experience. The validity of fire zone data sheets is also tested during these fire drills.

An evaluation of each drill is performed to assess each fire brigade member's participation, response, and fire fighting technique. This includes the fire brigade chief, who is evaluated on his ability to direct the efforts of the fire brigade. Each fire brigade member has one drill per year critiqued by the fire instructor, fire marshal, or a plant supervisor.

9B.2.5 QUALITY ASSURANCE PROGRAM

The Quality Assurance (QA) program for fire protection is under the management control of the QA organization. The QA organization is described in chapter 17. The QA program covers all safety-related material and equipment. Certain other items considered essential to reliable operation of the plant are also included in the QA program. Fire protection material, equipment, and services are included in the QA program since they are considered essential for the reliable operation of the plant. The following paragraphs present specific information about the fire protection QA program.

(4) The process monitoring function shall be capable of providing direct readings of the process variables necessary to perform and control the above functions.

(5) The supporting function shall be capable of providing the process cooling, lubrication, etc., necessary to permit the operation of the equipment used for safe shutdown by the systems identified in items 1 through 4 above.

(6) The equipment and systems used to achieve and maintain hot standby conditions should be free of fire damage, capable of maintaining such conditions for an extended time period longer than 72 h if the equipment required to achieve and maintain cold shutdown is not available due to fire damage, and capable of being powered by an onsite emergency power system.

(7) The equipment and systems used to achieve and maintain cold shutdown conditions should be either free of fire damage or the fire damage to such systems should be limited such that repairs can be made and cold shutdown conditions achieved within 72 h. Equipment and systems used prior to 72 h after the fire should be capable of being powered by an onsite emergency power system; those used after 72 h may be powered by offsite power.

(8) These systems need not be designed to Seismic Category I criteria, single failure criteria, or to cope with other plant accidents such as pipe breaks or stuck valves (Appendix A BTP 9.5-1), except those portions of these systems which interface with or impact existing safety systems.

d. PWR Equipment Generally Necessary for Hot Standby

(1) Reactivity Control

Reactor trip capability (scram). Boration capability, e.g., charging pump, makeup

pump, or high pressure injection pump taking suction from concentrated borated water supplies, and letdown system, if required.

(2) Reactor Coolant Makeup

Reactor coolant and makeup capability, e.g., charging pumps or the high pressure injection pumps. Power-operated relief valves may be required to reduce pressure to allow use of the high pressure injection pumps.

(3) Reactor Coolant System Pressure Control

Reactor pressure control capability, e.g., charging pumps or pressurizer heaters.

(4) Decay Heat Removal

Decay heat removal capability, e.g., power-operated relief valves (steam generator) or safety relief valves for heat removal with a water supply and emergency or auxiliary feedwater pumps for makeup to the steam generator. Service water or other pumps may be required to provide water for auxiliary feed pump suction if the condensate storage tank capacity is not adequate for 72 h.

(5) Process Monitoring Instrumentation

Process monitoring capability, e.g., pressurizer pressure and level, steam generator level.

(6) Support

The equipment required to support operation of the above described shutdown equipment, e.g., control room air conditioning, component cooling water, service water, etc., and onsite power sources (ac, dc) with their associated electrical distribution system.

e. PWR Equipment Generally Necessary for Cold Shutdown[a]

(1) Reactor Coolant System Pressure Reduction to Residual Heat Removal System (RHR) Capability

Reactor coolant system pressure reduction by cooldown using steam generator power-operated relief valves or atmospheric dump valves.

(2) Decay Heat Removal

Decay heat removal capability, e.g., residual heat removal system, component cooling water system, and service water system to remove heat and maintain cold shutdown.

(3) Support

Support capability, e.g., control room air conditioning, onsite power sources (ac and dc), or offsite power after 72 h, and the associated electrical distribution system to supply the above equipment.

3. Specific Criteria Used for Determining the Hot Standby and Cold Shutdown Components

a. Each system identified as having functional requirements related to hot standby or cold shutdown was reviewed to determine the components within the system which must remain functional to ensure that the hot standby or cold shutdown requirements of the system are met.

b. In addition, isolated components of other systems which do not have hot standby or cold shutdown functional requirements, but whose misalignment, through spurious signals, could cause operational problems with systems that have hot standby or cold shutdown functional requirements were also included.

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a. Equipment necessary in addition to that already provided to maintain hot standby.

c. All hot standby-related components falling into the above two criteria were listed on the hot standby components list with the following exception: two additional criteria were established and applied for the cable spreading room/control room alternative shutdown analysis for the components that have a high degree of redundancy and were determined to be candidates for exclusion from the hot standby components list based on their redundancy. The two additional criteria are called the "Series Rule" and the "Parallel Rule."

   It should be noted that the Series Rule and Parallel Rule, although applicable, were not used to exclude the main steam isolation, RHR inlet isolation valves, pressurizer PORVs, and block valves from the hot standby component list.

   (1) Components were excluded by the Series Rule if a required boundary was established by two opposite train components in series. For example, consider two opposite train valves that are in series and are essential to establish a boundary (i.e., remain in a closed position). Either of these valves may remain closed and still establish a boundary that is essential to achieve a hot standby condition following a postulated cable spreading room fire. To breach the boundary, an improbable series of events must occur in a hypothetical chronological scheme. The cables of opposite train components are routed in separate enclosures. A single fire in the cable spreading room must damage both opposite train, separately enclosed cables and also produce simultaneous hot shorts of sufficient voltage and current to concurrently open both the valves and breach the boundary.

   (2) Components were excluded by the Parallel Rule if the components were opposite train components which were in parallel in a flow path that is required to remain open. As an example, consider two

combustibles and only minimal, flame-resistant fixed combustibles inside the control panels, the probability of a fire in a control panel is virtually nonexistent.

(6) Finally, the plant fire protection program emphasizes prevention. This emphasis coupled with the preventive nature of the design of the area results in an area with as low a probability of fire as is currently possible to achieve. This design is much preferred to a concept emphasizing protective/extinguishing systems since there is no chance for inadvertent actuation adversely affecting plant operation.

b. Minimum damage potential if a fire should occur

(1) Due to the low level of combustibles and the flame-resistant nature of the fixed combustibles, any fire which is postulated to start would be slow-propagating, small in size, and limited in location. Hence, the fire would be easily and quickly controllable and thus unlikely to cause much damage.

(2) The control panels are enclosed and physical space between redundant panels is maximized. Where two safety-related divisions must enter the same control panel, protective barriers and/or physical separation are provided between the divisions to minimize the possibility of a postulated electrical conductor fire in one division affecting the opposite division. As discussed in paragraph J.6a, the only credible fire would be external to the panels and of small size and short duration; therefore, no postulated damage can occur inside the panels except due to self-ignition of inner-panel wiring. Due to the flame retardant properties of the cables entering the control panels, the small size of the inner-panel wiring and the conductors being used in low-current-level control circuitry, there is insufficient energy to cause a fire to

propagate across the interdivisional barriers or separations inside the panels.

(3) Ionization smoke detectors are installed in the control room and heat detectors are installed in the kitchen and store room. Additionally, portable water extinguishers, Class C extinguishers, CO_2 hose reels, and water hoses are readily available. The presence of personnel, along with the design for fast detection and the availability of fire suppression capability, will result in immediate response to any fire condition and subsequent early extinguishment.

(4) The site fire brigade will receive special training to ensure that any action on their part will be swift and will not be detrimental to unaffected parts of the control room.

c. Low Probability of Control Room Evacuation

(1) The fire heat load in the control room is equivalent to a fire severity of less than 30 min. This heat load assumes all the combustibles in the control room are involved in the fire. Thus, it is highly probably that the control room would never have to be evacuated completely since fires will be detected at the incipient stage and early responses will result in negating the need to evacuate.

(2) The control room HVAC design will provide isolation from products of combustion generated external to the control room. Additionally, the control room HVAC system is sufficient to remove the small amounts of smoke generated during the incipient stages of a fire and can be operated to remove denser smoke if required.

Inadvertent operation of any installed fire suppression system will not incapacitate safe-shutdown capability. Additionally, fire suppression systems that are pressurized during normal plant operation meet the guidelines specified in APCSB Branch Technical Position 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment."

The following subsections discuss the general design features of FNP which help assure that the safe shutdown capability of the plant is maintained in the event of a fire.

9B.4.1.1 Structural

The roof of the auxiliary building and diesel building is a reinforced concrete slab with a built-up surface consisting of layers of felt and asphalt/stone covering. The river water and service water building roofs are reinforced concrete. The turbine building roof is metal decking with a concrete slab poured over. Nonsafety-related buildings such as the fire protection pumphouse and the utility building have metal deck roofs. These roofs are noncombustible or are listed as Class I by the Factory Mutual System Approval Guide.

All floors, walls, and ceilings of buildings containing safe shutdown equipment, with the exception of the cable spreading room ceiling/control room floor (discussed below), stairwells 1 and 2 of Unit 1 and 1, 2, and 10 of Unit 2, and elevator shafts or in cases where water curtains are used in lieu of barriers, are of reinforced concrete or masonry construction having a minimum fire rating of 3 h. Interior walls, structural components, and radiation shielding are of noncombustible steel, concrete, and concrete unit masonry. Thermal insulation and interior finishes are noncombustible based on the definition of noncombustible given in paragraph 9B.1.2. Additionally, all stairwells designated for egress are enclosed in reinforced concrete masonry walls having a minimum fire rating of 3 h. The doors in these walls are rated at a minimum of 1 1/2 h except as otherwise noted in individual FAHAs.

The cable spreading room ceiling/control room floor consists of a 12-in. thick concrete slab having a minimum fire rating of 3 h and is supported by structural steel framing. All steel members required for the structural stability of the concrete slab have been protected by a sprayed-on fireproofing material, either Moni-Kote or an approved equal. The thickness of the sprayed-on coating is in accordance with the requirements of the manufacturer for a 3-h fire resistance rating.

The fire barriers have been designed in accordance with the guidelines of Appendix A to BTP APCSB 9.5-1.

Suspended ceilings and their supports are of noncombustible lay-in construction. The suspension systems and hanger wires

are steel. Acoustical board lay-in units for the exposed systems are noncombustible.

The only exposed combustibles in the concealed spaces in the control room are small amounts of cable insulation on cables used for control room lighting. There is also a small quantity of combustible cable insulation in enclosed wireways or conduit. Outside the control room the only combustible located in concealed spaces is cable insulation. Detection systems are installed for concealed spaces containing combustible material. The quantity of combustibles has been minimized and a fire initiating in a concealed area would be quickly detected by the installed detection systems.

9B.4.1.2 Pumps

To the extent practicable, redundant safe shutdown pumps are located in separate rooms. Where this is not the case, either the pumps are spatially separated or they have a barrier to provide separation. Details are provided in the fire area hazards analysis.

9B.4.1.3 Electrical

9B.4.1.3.1 Cable Construction

The electrical cable was purchased under the requirements of the applicable ICEA standard and an additional prototype flame test, which consisted of the following:

A. Prototype cables were arranged in single layer in a vertical ladder type steel tray approximately 8 ft high.

B. Cables were exposed to a flame 14 to 16 in. high from a 10-in.-wide burner for 10 min. Flame temperature was 1400°F to 1500°F with a heat rate of 7000 Btu/hour/in.

D. Cable conductors were monitored with 120 V or higher voltage; flame temperature was monitored with thermocouples.

D. Cables which propagated flame failed the test. Cables which self-extinguished passed the test. Cables which continued to burn after the flame was extinguished failed the test. Cables were required to maintain circuit integrity for 5 min during flame test.

These tests meet the intent and requirements of the IEEE 383 flame test, which was issued after the cables for FNP were purchased. These cables are also designed to allow for wetting down with deluge water without electrical faulting. The cable has been tested in accordance with the applicable ICEA standard for accelerated water absorption. All new and replacement cable is required to meet the IEEE 383 flame resistance test except in limited cases where IEEE 383 cable is not commercially available (e.g., computer ribbon cables and special purpose cables). All exceptions to the IEEE 383 flame resistance test will require a detailed evaluation to show that the use of such cables will not reduce the effectiveness of the fire protection program at FNP. They must also be suitable for installation in "wet locations" by meeting the requirements of the applicable ICEA standard for accelerated water absorption.

To the extent practical, the cable used does not give off corrosive gases. The use of polyvinyl chloride (PVC) has been minimized.

9B.4.1.3.2 Cable Routing

The design criteria for cable routing as described in paragraphs 8.3.1.4.2, 8.3.1.4.3, and 8.3.1.4.4 have been established to minimize and localize the effect of a fire should one be caused by an electrical fault. Exceptions to these criteria[a] are granted only after an engineering evaluation. The cable trays are constructed of either aluminum or galvanized steel, and the cable tray supports are also constructed of steel. Cable trays, raceways, conduit, cable trenches, and vertical cable chases and tunnels carry only electrical equipment. Culverts, which are listed in BTP APCSB 9.5-1, are not used for electrical cables at FNP. These areas are not used for miscellaneous storage.

Each cable related to a redundant safe shutdown component is assigned the proper train designator as part of its cable number. Each such cable is routed only in raceways of the same train.

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a. See paragraph 9B.3M, item 3 for a discussion of exceptions to the limits described in paragraph 8.3.1.4.2.

A separate cable spreading room containing redundant cable divisions is provided for each unit. The cable spreading rooms for each unit are separated from each other and from the balance of the plant by 3-h-rated fire barriers. Two access doors to each unit's cable spreading room are located at opposite ends of a common corridor to allow room access from two directions.

Cables entering the control room which do not terminate there have been kept to the minimum necessary for operation of the control room. In a limited number of cases, instrument and control cables in rigid conduit are run along the control room perimeter wall between adjacent elevations. No 4160-V or 600-V heavy power circuits are routed through the control room.

Divisional cable separation meets the guidelines of Regulatory Guide 1.75, "Physical Independence of Electrical Systems."

The minimum horizontal and vertical separation requirements in the cable spreading room are as described in paragraph 8.3.1.4.

Where redundant hot-shutdown related cable is in a given switchgear room, one train of cable is enclosed by a minimum 1/2-h fire barrier. As described in the individual fire area hazard analysis, the area detection system provides early warning of a fire. With early warning and the minimum 1/2-h enclosures around one train of cable, the fire brigade has adequate time to extinguish the fire manually and maintain hot shutdown capability.

Wiring necessary for operation of any CO^2 system is supervised locally by a trouble indication at the control cabinet. In addition, zones protected by CO^2 systems have backup, fully supervised, smoke detection systems which alarm in the control room.

9B.4.1.3.3 Transformers

All high-voltage, high-amperage transformers located within buildings throughout the plant are insulated and cooled with a synthetic insulating and cooling liquid which is classified as "less flammable." This liquid is exceedingly difficult to ignite, and will sustain combustion only under the most severe exposure to fire. Even though the likelihood that these transformers will sustain combustion is extremely low, their effects during a fire have been considered in the individual area fire hazards analysis.

9B.4.1.4 Fire Barrier Penetrations

The functional integrity of the fire barrier penetrations ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the plant. This minimizes the possibility of a single fire rapidly involving several areas of the plant prior to detection and extinguishment. With the exception of penetrations between containments and the electrical penetration rooms, all fire barrier penetrations used for electrical cables, pipes, and instrumentation tubing are provided with either silicone foam, silicone rubber boots, non-shrink grout, or Nelson multicable transit system penetration seals. The cable penetrations between containments and electrical penetration rooms are sealed against pressure and are watertight, but are not fire rated. Nelson multicable transit system seals have a minimum 3-h UL fire-resistance rating. Qualification testing of the silicone rubber fire barrier penetration seals, which is in accordance with ASTM E119-73 (NFPA 251) and ASTM E84 (NFPA 255), is reported in:

A. F.M. Research test serial No. 27290(4510)

B. F.M. Research test serial No. 27290.1(4820)

The conclusion of A above is:

"The penetration seals as described in this report were subjected to fire exposure for 3 h in accordance with the ASTM Standard Time-Temperature Curve.

The silicone rubber penetration seals and flexible reinforced silicone rubber boot penetration seals prevented the passage of flame and excessive smoke through the wall assembly during the test. Flaming did occur at the joints of the aluminum jacket cables.

The transmission of heat through the seals was below the 394°F temperature limit for the 3-h duration of the test.

The seals withstood the hose stream tests without penetration by the water stream."

The conclusion of B above is:

"The following classification has been established for Dow Corning Q3-6548 Silicone Foam and Dow Corning Sylgard 170 RTV Elastomer utilizing 0.040-in.-thick aluminum as the substrate. This classification is in accordance with Standard Method of Test for Surface Burning Characteristics of Building Materials ASTM Designation E84-75. The values

given in the table are an average of the results of three tests rounded to the nearest multiple of five."

Specimen	Flame Spread	Fuel Contributed	Smoke Developed
Dow Corning Q3-6548 Silicone Foam	80	55	950
Dow Corning Sylgard 170 RTV Elastomer	55	Not determinable	255

All ducts penetrating fire boundaries have fire dampers installed. The dampers conform to the requirements of NFPA Standard 90A and are 3-h UL labeled or 1 1/2-h UL labeled (except where noted in FAHAs) as required. All dampers are installed at the penetrations.

The fire barrier penetrations and fire dampers have been designed in accordance with the guidelines of Appendix A to BTP APCSB 9.5-1.

9B.4.1.5 Doors

Three-hour fire-rated doors are provided per design for areas where the fire hazard analysis demonstrates their need. Stairwells used for access and egress have UL Class B doors and a fire rating of at least 90 min, except for one Class A door in Unit 2 auxiliary building stairwell No. 1 (el 121 ft 0 in.), two watertight doors and doors leading to the exterior. Exterior doors and one pressure tight, bullet-resistant control room door are not UL labeled fire doors. For doors or frames missing UL labels, documentation exists to show that they were purchased to meet design requirements. A detailed discussion of door ratings is in the area-by-area fire hazard analysis where appropriate.

The fire doors have been designed in accordance with the guidelines of Appendix A to BTP APCSB 9.5-1.

9B.4.1.6 Compressible Material

The exposed surfaces of the compressible material (self-expanding cork) isolating the auxiliary building from the containment during a seismic or loss-of-coolant accident (LOCA) event have been impregnated with a fire retardant solution, Flamort WC, manufactured by Flamort Chemical.

In addition to the Flamort WC, certain portions of the self-expanding cork are protected on both the top and bottom floor slab by a sealing assembly of three 1-in. layers of Kaowool backed by a 16-gauge bent plate and covered with 24-gauge flashing material.

The burning rate of the impregnated cork is not known precisely; however, the Flamort WC has an ASTM E84 rating of 26-75 on wood or cork.

The compressible material in rooms 186 and 2186 of the auxiliary building have been repaired/replaced to a depth of 6 in. using 3 in. of polyurethane foam and 2 1/2 in. of polyvinyl chloride foam which forms the exposed surface. This material has an ASTM E84 flame spread rating of 70 and is considered self-extinguishing per UL 94HF1.

9B.4.1.7 Fire Protection Water Supply

The fire protection water supply (total storage capacity, single fire pump capacity, and pipe sizing) was designed to supply any safe-shutdown related water suppression system with a maximum

hazard area. All standpipes serving more than one hose station are a minimum of 4 in. in diameter. Each hose station is equipped with UL listed equipment where applicable, adjustable fog nozzles suitable for electrical fires, and 75 ft (nominal length) of 1 1/2-in. lined fire hose (except for cabinets D-108, -122, and -115 in Unit 1 auxiliary building and D-108 and -119 in Unit 2 auxiliary building, which have 100 nominal-length ft of 1 1/2-in. hose), suitable angle valves, spanner wrench, drain vent valve, pressure reducing valve, and hose rack.

The Unit 1 and Unit 2 containments are provided with a standpipe system. The water supply for the standpipes is provided by a Seismic Category I system and will supply a minimum of 75 gal/min at a nozzle pressure of 60 psig. The firehose and equipment for these standpipes are stored in a locker in the auxiliary building, immediately outside containment, because of the degenerative effect to equipment caused by extended radiation exposure.

9B.4.1.10 Water Suppression Systems

The automatic/manual sprinkler and water spray systems as well as the manual hose stations are connected to a Seismic Category I water supply header located throughout the auxiliary building area for Unit 1 and Unit 2. Multipath water supplies are provided with isolation valves and appropriate sectionalization valving to preclude a single break from disrupting the water supply to any one area. The turbine building is equipped with a similarly looped water supply system serving both units. Actuation of any water fire suppression system will cause a fire pump to start on a low-header pressure signal.

The interior water supply loop for each safety-related building is designed to supply the capacity of the largest fixed system demand, plus 1000 gal/min hose stream allowance, which is consistent with the guidelines provided in Appendix A to BTP APCSB 9.5-1.

The automatic/manual sprinkler systems and the automatic/manual water spray systems are designed to address a special hazard philosophy in concert with the appropriate selected sections of NFPA 13, Standard for Installation of Sprinkler Systems, and NFPA 15, Standards for Water Spray Fixed Systems. NFPA 13 and NFPA 15 are followed to the maximum extent practicable. Occasionally, minor deviations, principally in head or detector placement relative to the ceiling, must be applied because of congestion in the area of a suppression system consisting of cable trays, piping, ducting, etc. Such deviations are exercised only whenever they will result in equal or superior protection to that which would result from an inflexible conformance to codes or regulations. Such codes and regulations are basically tailored for protection of rooms or areas with relatively regular surfaces and features.

This is a situation seldom enjoyed in the highly congested conditions encountered in the majority of the rooms in a power plant. In recognition of such conditions and of the special hazards encountered in a nuclear power plant, sound engineering judgment must be exercised in the design and installation of its fire protection systems rather than routine, unchallenging compliance with rules which are not always appropriate to the situation.

The sprinkler systems installed in rooms 209, 2208, and 2209 do not satisfy a strict interpretation of NFPA requirements related to obstructions in proximity to sprinklers. However, analyses which considered suppression and detection capabilities, in-situ combustibles, passive fire protection features, fire brigade capabilities, and plant procedures for the control of transient combustible/flammable materials concluded that the existing sprinkler systems are adequate. The analyses further concluded that modification of the existing suppression systems or the installation of additional systems was unjustified as they would provide only marginal improvement.

The fixed water fire protection systems protecting the safety-related areas provide a density as required by NFPA 13 or 15 or at a minimum of 0.30 gal/min ft^2, whichever is higher.

The design of sprinkler/spray systems in nonsafety-related buildings is based upon each fire area's level of combustible loading. These systems are designed to provide the discharge densities required by NFPA 13 and 15.

All of the water suppression systems installed in the auxiliary building, the diesel generator cable tunnels, and the service water intake structure are designed as Seismic Category I systems to preclude physical damage which would incapacitate any safe shutdown function in the event of an SSE.

All systems except the wet pipe systems can be manually activated or charged at their control stations. Each control station for the fixed suppression systems identified in figures 9B.5-36 and 9B.5-37 includes a complete complement of tamper and status alarms to ensure correct valve alignment and system availability. Discrete alarms, which are discussed in paragraph 9B.4.1.14, are provided to distinguish between trouble conditions and fire conditions at each local station within the plant.

The design criteria for the water suppression systems meet the guidelines of Appendix A to BTP APCSB 9.5-1.

cylinders are super-pressurized with dry nitrogen to a pressure of 600 psig at 70°F.

The Halon systems provided are designed and installed to appropriate sections of NFPA 12A, Halon-1301 Fire Extinguishing Systems.

9B.4.1.13 Fire Detection Systems

The detection systems listed in table 9B.C-1 are designed to provide the earliest possible notification of potential fire conditions and consist of the detectors, associated electrical power supplies, and the annunciation panels. The types of detectors used at the plant are ionization (products of combustion) and thermal (heat sensors). Fire detection systems give an audible and visual alarm which annunciates in the plant control room and each local alarm area or as noted in figures 9B-37 and 9B-38. The detection systems power supplies, both normal (120 V-ac) and emergency (125 V-dc), are reliable and can be manually connected to an onsite Class 1E power source upon a loss-of-offsite-power condition.

All smoke detectors have been located and installed as recommended by the manufacturer based on the manufacturer's engineering judgment. Sensitivity settings were set at the plant by the manufacturer's field representative and recorded (as required for system certification). Sensitivity is set by varying the bias voltage applied to the cold cathode tube in the individual detector. This applied voltage determines the firing (alarm setpoint) of the detector. This voltage varies with the quantity of detectable products of combustion present in the outer chamber of these two-chamber detectors. Firing voltages are adjustable between the limits of 20-80 V for Model DIS-5B and 15-35 V for DIS-3/5A. The design settings were determined by:

- The sensitivity to fire damage of the contents in the protected area.
- Air velocity at the detector.
- Expected rate of burn of combustibles in the protected area.

Sensitivity increases as the firing voltage decreases.

All existing detectors at Farley Nuclear Plant are set according to the conditions found at their location and are recorded by serial number. The serial-numbered test report indicates system number and locations as shown on the manufacturers' drawings.

This installation complies with the NFPA 72E Standard and, thereby, NFPA 72D (1975). Also, the fire detection system's design criteria conform to the intent of the applicable sections of NFPA 72D (1975). Accordingly, the design and installation of the fire detection systems meet the guidelines of Appendix A to BTP APCSB 9.5-1. Additionally, the fire detection system maintenance procedures conform to NFPA 72D (1975).

9B.4.1.14 <u>Fire Protection Annunciation</u>

A fire warning system is provided as an integral part of the fixed (automatic and manual) suppression systems to indicate and alarm in case of a fire in any protected area. The fire alarms are distinctive and unique in order to preclude any confusion with other plant system alarms. The plant emergency alarm is sounded for a significant fire. The local fire alarms are bells with the exception of cases where high-pressure CO_2 is used. In these cases, a siren is used for local alarm.

As a direct result of the combustible loadings for the plant, the detection systems are designed to operate as two discrete modes. Since fixed suppression is required, the associated detection system serves as the preacting device or first-stage actuation for the particular sprinkler system. This action allows the normally dry sprinkler piping to be charged with water. The detection systems used to actuate the Halon and CO_2 systems are alarmed and annunciated in the control room independently of the suppression system actuation. The second mode of operation only provides notification of fire conditions and is not directly associated with a fixed suppression system.

Except as noted on figures 9B-37 and 9B-38, all fire alarms are indicated in the control room on the main annunciation panel; in the control room on the digital display system; in the control room on the auxiliary fire protection panel; and locally at each protected area. In addition, many detection systems which are not associated with a suppression system are installed to provide early warning; these independent systems are annunciated at the locations delineated above. The suppression and detection systems in the training and emergency operations facility, the production warehouse and the computer/office building are annunciated in the security building which is manned at all times.

Additionally, fire pump running, motor-driven fire pump power failure, diesel-driven fire pump(s) start failure, fire main low pressure, and fire protection water tank low level alarms are annunciated in the control room.

All of the alarm systems are provided with alternate power sources so they can operate under all conditions of onsite and offsite electrical power.

9B.4.1.19 Emergency Lighting

All areas of the plant which require personnel occupancy to bring the plant to a safe shutdown condition are provided with 8-h rated battery-powered emergency lights or essential ac lighting. Access and egress routes to these areas are also provided with the 8-h lights. Battery-powered portable hand lights (not of the sealed-beam type) are also provided for emergency use in responding to a fire. All other areas of the plant are provided with 1 1/2-h lighting to facilitate personnel exit in an emergency. The drawings which show the locations of emergency battery pack lighting are listed in Sections 3.2 and 3.3 of the Appendix R compliance report (A-350971) for Units 1 and 2, respectively.

9B.4.1.20 Emergency Smoke Removal

Although some permanently installed smoke ventilating equipment and area exhaust fans are available, it is anticipated that portable smoke removal equipment will be utilized primarily for smoke removal. The general criterion for smoke venting is six air changes per hour.

Eight portable electrical blowers with flexible hose attachments were purchased for each unit for heat and smoke removal from the auxiliary building. The portable blowers and hose attachments are located at selected elevations in the auxiliary building where they are readily available for use by the fire brigade. These fans are capable of being powered by normal ac outlets or by outlets supplied from the emergency power sources. As additional backup, four portable gasoline engine-powered smoke removal blowers are available on site. These blowers are stored in the utility building to avoid any possibility of a gasoline fire in essential areas.

Plant emergency implementing procedures describe the operation of portable blowers, hookup and routing of flexible hoses, and sampling requirements for smoke removal from the auxiliary buildings, the diesel generator building, and the service water building. The portable blowers will be used to take suction from the smoke-filled area and discharge to the nearest elevator shaft or stairwell. For smoke removal from elevations below ground level, an additional blower will be located at ground level taking suction on the elevator shaft or stairwell being used and discharging outside the building. Grab samples of smoke from potentially radioactive areas will be taken using Marrinelli beakers, particulate filters, and iodine filters as appropriate.

One portable blower with hose attachment is located at the service water building; another is located at the diesel generator building.

9B.4.1.21 Floor Drains

For all areas where water is the primary suppressant, either floor drains, sized to remove expected fire fighting water flow are provided or, where no drainage exists, an evaluation has been performed which shows that safe shutdown equipment would not be adversely impacted by the release of fire protection water. For example, drainage for the cable spreading room, which has no floor drains, is provided by opening the door to the cable spreading room and allowing the water to drain into the hallway. In addition, the termination cabinets located in the cable spreading room, the only safe-shutdown-related equipment in the spreading room that could be adversely affected by water, are located on pedestals, which permits the spreading room to be flooded to a depth of 4 in. without affecting the cabinets.

Areas containing combustible liquids in quantities which could spread through the drain system are the turbine-driven auxiliary feedwater pump area, the high-head safety injection pump areas, and the reactor coolant pump areas. Local sumps have been provided to contain the combustible liquid and minimize any spreading through the drain system.

Water drainage from areas which may contain radioactivity is collected in the liquid radwaste system and sampled and approved prior to discharge to the environment.

9B.4.1.22 Lightning Protection

The containment, auxiliary building, diesel generator building, river water intake structure, and service water intake structure

are protected from lightning strikes by suitably grounded air terminals in accordance with NFPA Standard No. 78. The lightning protection system throughout the plant consists of the air terminals and external down conductors which are bonded to the plant grounding system. These items provide a path of least resistance through which electrical discharge may enter or leave the earth.

9B.4.1.23 Bulk Gas Storage

Most flammable gas is stored outdoors. The bulk storage facilities for hydrogen, nitrogen, oxygen, and chlorine are located outdoors at a distance greater than 500 ft from the auxiliary building. Two hydrogen bottles are stored outside adjacent to the main steam room. Table 9B-5 presents a list of other hazardous combustibles, including flammable gases, not specifically addressed in individual area fire hazards analyses and the maximum amounts estimated to be present, as well as their storage location relative to safety systems.

Some individual flammable gas bottles are required to be stored in the counting room, the high-activity radioactive lab room, the hydrogen recombiner room, the clean storage room, and the electrical penetration rooms. However, these bottles pose no danger to the safe shutdown capability of the plant and are addressed in the fire area hazards analysis.

9B.4.1.24 Flammable Liquid Storage

Flammable liquids stored in the auxiliary building and diesel building are stored in accordance with Title 29, Chapter 17 Part 1910--Occupational Safety and Health Standards, Paragraph 106 (dated October 18, 1972). Details of type and quantity of flammable liquid and fire protection provisions are given in the fire area hazards analysis. Compliance with these OSHA requirements is equivalent to compliance with NFPA 30, "Flammable and Combustible Liquids Code."

Table 9B-5 presents a list of hazardous combustibles, including flammable liquids, not specifically addressed in the individual fire area hazards analyses and the maximum amounts estimated to be present, as well as their location relative to safety systems.

9B.4.1.25 Plastics

Although the use of plastics has not been specifically prohibited, their applications have been minimized. The use of

plastics is limited to instrument parts, wireways inside vendor furnished equipment, and sample and instrument tubing. This is not to imply that plastics are used in all cases for the above, but only that plastics were not specifically prohibited for these uses.

9B.4.1.26 Fire Breaks

Fire breaks may have been provided for compliance with Appendix A to BTP APCSB 9.5-1 where a vertical tray runs for more than 20 ft with no penetration seal. However, the functionality of those fire breaks is not required by the fire hazards analysis or the methodology used to satisfy the requirements of Appendix R. Therefore, fire breaks are no longer required.

9B.4.1.27 Self-Contained Breathing Apparatus

Self-contained breathing apparatus using full-face positive-pressure masks are located at the designated fire brigade assembly point. At least eight masks are located in the control room and at least three masks are in the access control area for use by control room, fire brigade, and damage control personnel. Ninety-six spare air bottles are stored in the hallway outside the control room. This quantity will provide emergency air for 48 manhours. Each bottle has a 1/2-h supply of air.

9B.4.1.28 Charcoal Filters

No fire protection is provided for charcoal filters. Heat detectors are located in all charcoal filters for the control room and the penetration room filtration units, and alarm at local panels and in the control room. Thermocouples are located upstream of all charcoal filters and alarm, as well as provide temperature indication, in the control room. With these heat and temperature detectors, adequate warning of high temperatures in filters is available to allow the filter to be isolated.

9B.4.2 SPECIFIC PLANT AREAS

This section provides a general description of the plant fire protection for specific plant areas. The following subsections describe the water supply and distribution system, the yard protection equipment, the protection provided for major plant facilities, and areas of concern identified in Appendix A to BTP 9.5-1, which may not be covered specifically in the fire area hazard analyses. The intent is to provide a general overview of these fire protection features, not to provide system specifics.

The specifics are contained in the figures and fire area hazard analyses, and may also be found in subsection 9B.4.1.

9B.4.2.1 Water Supply Storage and Distribution System

The fire protection water supply system consists of three deep wells and pumps and, in emergency circumstances, the service water system, both capable of providing single storage tank replenishment within 8 h. The fire protection water storage and distribution system is common to both units and consists of two 300,000-gal storage tanks which are dedicated to the fire protection system. Low-level water alarms are provided on each tank to alert operators and give adequate time to isolate the affected tank in the event of a leak.

As an emergency backup, the fire protection water system can also be supplied from the service water system. An 18-in. line connects the Unit 2 service water system on the upstream side of the standpipe/surge tank to the suction header of the fire pumps. In the unlikely event of a fire that results in the near depletion of the fire protection storage tanks, the fire pumps can take suction directly from the service water system and fire fighting efforts can be continued. Furthermore, the service water system can be used to refill the fire protection water storage tanks should the well water system become unavailable. One tank can be replenished in less than 8 h. These service water system capabilities are available when either train of the Unit 2 service water system is operating in the normal mode, i.e., discharging to the river.

Due to the possibility of introducing fouling agents into the fire protection water system, service water supply to the fire protection water system is isolated by dual locked-closed valves and is to be used only (1) when a situation exists where both fire protection storage tanks are nearing total depletion and the continuation of ongoing fire fighting efforts is required or (2) when either or both fire protection water storage tanks are below technical specification limits and the well water system cannot restore the water in the tanks to Technical Specification minimum limits in the required 8-h period. If service water is employed in the fire protection tanks or the distribution system, antifouling treatment will be used.

The fire pump suction piping incorporates a suction header connected to both tanks with isolation valving arranged to permit either tank or any fire pump to be taken out of service for maintenance or repair without disrupting the other tank or pumps. The fire pumps are each rated at 2500 gal/min and 125 psig; one electric motor-driven and two diesel engine-driven pumps are provided. The fire pumps are located in a single fire pumphouse, separated from each other by 3-h fire barriers. A pressure maintenance (jockey) pump is included to maintain the

distribution piping pressure at 125 psig. The jockey pump is also capable of taking suction from the fire supply header which is connected to both of the 300,000-gal storage tanks. Alarms are provided in the control room to monitor fire pump running, motor-driven fire pump power failure, fire protection low water pressure, low water in the fire protection water storage tanks, or engine-driven fire pump start failure. Separate visual alarms are provided for each individual condition on the control room digital display system. The power supply associated with the control signal which automatically starts the motor-driven fire pumps is supplied by the Class 1E station battery system. The diesel-driven fire pumps control signals are supplied by their own self-contained controller and battery system. The fire pumps are supplied with UL-listed controllers except for the electric fire pump. UL does not list 4-kV controllers; however, the 4-kV controller meets the applicable requirements of NFPA 20. Since the fire pumps are located in a separate building from any safety-related components, a fire that could affect the fire pumps would not affect the ability to safely shut down the plant.

The three fire pumps and jockey pump operate automatically with a normal operating pressure of 125 psig. The small, constantly running jockey pump maintains the fire protection water distribution system at 125 psig. Any jockey pump output flow in excess of the distribution system requirements is recirculated back to the fire protection storage tanks. The capacity of the jockey pump is sufficient to maintain this pressure against normal system leaks or losses, but it cannot maintain this pressure against losses resulting from sprinkler system or hose stream actuation. Under these circumstances, system pressure

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will decay. Should pressure fall to 90 psig, the electric fire pump starts. The two diesel fire pumps sequentially actuate as pressure falls to 80 psig and 70 psig, respectively. Diesel fire pump No. 1 will actuate at a falling pressure of 80 psig and after a 5-s delay time, while diesel fire pump No. 2 will actuate at a falling pressure of 70 psig and after a 10-s delay time. These pumps must be manually stopped.

The water distribution system is designed to provide an adequate supply equal to the largest demand from any single hazard in safe-shutdown-related areas, plus 1000 gal/min for hose stream use. Specifically, the fire suppression system requiring the greatest water demand for areas containing or exposing safety-related equipment or circuits is the automatic wet pipe sprinkler system and at least 1000 gal/min for hose streams associated with fire area 13 in the auxiliary building. This water demand is 2439 gal/min. Therefore, for suppression systems protecting safe-shutdown-related equipment and cabling, this criterion is met with any one fire pump. A cooling tower fire requires two fire pumps; however, the criterion is met since there are three fire pumps available and since the cooling towers are not safe shutdown related.

A yard loop of 12-in. diameter cement-lined pipe encircles the plant and cooling tower areas with hydrants in houses at intervals of 250 to 300 ft. The hydrants have been placed to provide a minimum of two streams capable of reaching all the plant facilities except the river water intake structure which is nonsafety-related. Indicator post valves are arranged to provide alternate water supply paths in the event of a single break or when repairs are required. The yard loop system is designed and installed in accordance with NFPA 24. The yard loop (fire main loop) has connections as follows:

- One 8 in. to east side of each auxiliary building.
- One 8 in. to west side of each auxiliary building.
- One 8 in. to east side of each turbine building.
- One 8 in. to west side of each turbine building.
- One 8 in. to each unit's low-voltage transformer area.
- One 6 in. to service building.
- One 6 in. to utility building.
- One 10 in. to warehouse.
- One 12 in. to service water intake structure and the training and emergency operations facility.
- Two 8 in. to the emergency diesel generator building/ cable tunnels.

- One 8 in. to the computer building.
- One 6 in. to the low-level radwaste storage building.
- One 12-in. loop around cooling tower area with six 10-in. connections to cooling towers.
- Fifty-two 6-in. connections for the fire hydrants.

9B.4.2.2 Yard Protection

Fire protection is provided to the exterior plant areas by 52 yard fire hydrants. The fire hydrants and associated equipment are protected from the elements by a hydrant house.

9B.4.2.3 Auxiliary Building

The auxiliary buildings' fire protection water systems are supplied by separate 8-in. mains which enter the buildings from their east and west sides. Each main has a remotely initiated motor-operated valve located immediately outside the building. These valves are installed to prevent flooding in the event of a break inside the building and are remotely operated from the control room. In each line is an orifice and pressure sensing device; if the flow exceeds 2700 gal/min, a flowrate above that of the highest suppression system demand, a trouble signal is annunciated in the control room where operator-initiated corrective action, if and as appropriate, can be initiated. Within the auxiliary buildings, the fire protection supply mains entering from the east and the west are cross connected to form a looped supply to hose cabinets and installed water suppression systems. To provide further redundancy, the two auxiliary building systems are cross connected. Sectionalization valves are installed to preclude a single break from disrupting the looped supply to any one area.

The fire protection water mains and the water suppression systems in the auxiliary building are designed to Seismic Category I except for nonseismic qualified sprinkler system 1A-52. An analysis was conducted on this nonqualified system and the results indicate that there would be no adverse impact on safe shutdown cabling or equipment in the event of an SSE.

The auxiliary buildings are provided with a wide array of installed suppression systems. With the exception of five systems in Unit 1 and three systems in Unit 2, the installed water systems are preaction design. The exceptions are wet pipe systems. All water systems are automatic. Halon-1301 systems

provide total flooding coverage to three rooms. A 13-ton low-pressure CO_2 system (with the storage tank located in the Unit 1 turbine building) supplies fixed CO_2 protection through piping into the cabinet of safety-related 4160-V and 600 V switchgear and 5-kV disconnect switches. The CO_2 system also supplies a total flooding system in the cable spreading room (this coverage is manually activated and is utilized as backup to the installed automatic sprinkler system), and to manually operated CO_2 hose reels located in areas containing predominantly electrical-type hazards. Considering all of these design features, it can be concluded that the cable spreading room design conforms to the requirements of Appendix A of BTP APCSB 9.5-1. Additional backup to these fire suppression systems is provided by standpipe systems and portable extinguishers located at both entrances to the cable spreading room. When the CO_2 system actuates, the ventilation system will isolate the cable spreading room. Smoke venting can be started from manual control stations to actuate an exhaust fan. Portable fans are also available. In addition, installed smoke detectors will initiate an alarm in the control room. In the event of a fire in the cable spreading room or control room, the plant can be brought to a safe shutdown condition. Cable chase tunnels in fire areas 8, 9, 30, and 31 are channel separated while the cable chase in fire area 13 has both trains of cables. The cable chase in fire area 13 is provided with sprinklers and fire barriers to preclude loss of safe shutdown capability in the event of a fire.

The auxiliary buildings are also furnished hose stations which are appropriately located and equipped to supply effective hose streams for primary or backup fire suppression.

Portable extinguishers are located throughout as first aid fire protection equipment. Dry chemical and CO_2 extinguishers are provided primarily where there are potential electrical fires, and water pump extinguishers are provided in the control room. Extinguisher equipment and locations are in accordance with NFPA 10, Portable Fire Extinguishers.

Smoke detection in the auxiliary building is utilized in two modes. One mode is where the detection system is utilized both for early warning and as a suppression system preaction device. The other mode is to provide warning only with the smoke detection system having no suppression system interrelationship. All significant areas of the auxiliary buildings have smoke or heat detection systems.

9B.4.2.4 Containment

The containment areas are provided with a dry standpipe system which receives its water supply from each of the redundant trains of the service water piping in containment, a Seismic Category 1 system. The fire protection water standpipe system, also Seismic Category 1, can supply a minimum of 75 gal/min at a nozzle pressure of 60 psig. The containment fire hose and equipment are

maintained immediately outside of containment because of the degenerative effect to hoses caused by extended radiation exposure. Ionization-type detectors are located at the containment coolers, steam generators, and containment ventilating ducts.

In addition, modifications have been completed for reactor coolant pump motor oil spillage protection and control. These modifications consisted of enclosures, catch basin, drip pans, drain lines, and collection tank, and are in compliance with Appendix R.

To protect against an oil leak in the oil lift system, oil cooler, and oil cooler piping, enclosures are provided isolating the pressurized oil components from the environment. Drip pans

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effectiveness of the overall barrier. Where fire area boundaries are not wall-to-wall, floor-to-ceiling boundaries with all penetrations sealed to the fire rating required of the boundaries, an evaluation must be performed to assess the adequacy of the fire boundaries to determine if the boundaries will withstand the hazards associated with the area. This analysis must be performed by at least a fire protection engineer and, if required, a systems engineer. This evaluation is not required to be submitted to the NRC for review and concurrence but must be retained for future NRC audits.

2. Sections III.G.2.b, III.G.2.c, and III.G.2.e of Appendix R require fire detectors and automatic fire suppression systems be installed in the fire area..." In order to comply with these provisions, suppression and detection sufficient to protect against the hazards of the area must be installed. In this regard, detection and suppression providing less than full area coverage may be adequate to comply with the regulation. Where full area suppression and detection is not installed, an evaluation must be performed to assess the adequacy of partial suppression and detection to protect against the hazards in the area. The evaluation must be performed by a fire protection engineer and, if required, a systems engineer. This evaluation is not required to be submitted to the NRC for review and concurrence but must be retained for future NRC audits. Where no suppression or detection is provided, an exemption must be requested.

C. For fire areas where nonconformance to the requirements of paragraph III.G are found to exist, and it is not possible to achieve complete compliance or support an exemption, an alternative shutdown capability analysis is performed. This analysis determines whether alternative shutdown capability exists, or plant modifications are required to meet the requirements of paragraph III.G.

D. The alternative shutdown capability analysis is performed by applying the criteria described in section 9B.3, item J.5 against the functional requirements of each system required to achieve and maintain hot standby and to go to cold shutdown as described in paragraph 9B.5.2.2 assuming a loss of offsite power.

The areas requiring alternative shutdown capability are the cable spreading room, control room, fire areas 51, 1-004 (el 155 ft 0 in.), 2-004 (el 155 ft 0 in.), and 2-043. As discussed in section 9B.3, item J.6, a fire in the main control room which could lead to the inability to achieve and maintain hot standby is not a credible event. Therefore, the alternative shutdown capability analysis considers a fire in the cable spreading room. A single fire in fire area 51 (shared), Unit 1 fire area 1-004 (el 155 ft 0 in.), Unit 2 fire area 2-004 (el 155 ft 0 in.) or 2-043 could potentially disable both trains of control room air conditioning. In that event, if the habitability of the control room becomes a concern, the implementation of alternate shutdown procedures becomes an option for the operators to safely shut down the plant.

For the immediate/short-term system requirements analysis and the long-term system requirements analysis, the circuitry related to each component required to achieve and maintain hot standby is analyzed against section 9B.3, item J.5 to determine whether adequate alternative shutdown capability exists or whether alternative shutdown capability must be provided for a cable spreading room fire. Results of the immediate/short-term system requirements analysis are tabulated in section BB.II.A of the Alternative Shutdown Capability Analysis (A-350970). Results of long-term system requirements analysis are tabulated in section BB.II.B of the Alternative Shutdown Capability Analysis (A-350970).

For the cold shutdown system requirements analysis, the circuitry and local manual control capabilities of each component required to go to cold shutdown from hot standby were analyzed to determine what manual actions or repairs required to go to cold shutdown for a cable spreading room fire. These results are tabulated in section BB.II.C of the Alternative Shutdown Capability Analysis (A-350970).

Section III.G.3 of Appendix R provides for "alternative or dedicated shutdown capability and its associated circuits, independent of cables, systems or components in the area, room, or zone under consideration." While "independence" is clearly achieved where alternative shutdown equipment is outside the fire area under consideration, this is not intended to imply that alternative shutdown equipment in the same fire area but independent of the room or the zone cannot result in compliance with the regulation. The "room" concept must be justified by a detailed fire hazards analysis

that demonstrates a single fire will not disable both normal shutdown equipment and the alternative shutdown capability.

E. From the Appendix R safe shutdown capability analysis, a determination is made of those plant areas requiring emergency lighting for operator action to perform a safe shutdown function.

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a. Requires operation of turbine-driven auxiliary feedwater pump (TDAFWP) or motor-driven auxiliary feedwater pump (MDAFWP).

b. Requires suction to TDAFWP/MDAFWP from the condensate storage tank.

c. Requires isolation of main feedwater.

C. Service Water System

1. Short-Term Requirements

a. Provide cooling water to one train of diesel generators.

b. Provide cooling water to component cooling water heat exchanger.

c. Maintain isolation of service water dilution bypass line.

2. Long-Term Requirements

a. Provide cooling water to room coolers for rooms which contain the charging pumps, battery chargers, RHR pumps, and related electrical switchgear.

b. Provide backup suction to auxiliary feedwater system after condensate storage tank inventory is depleted.

c. Realign service water return from the river/circulating water canal to the storage pond.

d. Provide cooling water to one train of the control room packaged air conditioning units.

D. Component Cooling Water System

1. Short-Term Requirements

a. Provide cooling water to seal water heat exchanger.

b. Provide cooling water to charging pump bearing and gear oil coolers.

2. Long-Term Requirements

a. Provide cooling water to RHR heat exchanger and RHR pump seal cooler when placed in operation for cold shutdown.

b. Provide cooling water to spent-fuel pool heat exchangers if required by spent-fuel inventory.

c. Provide cooling water to RCP thermal barrier.

E. Chemical and Volume Control System

1. Short-Term Requirements

a. Provide RCP seal injection and RCS inventory control.

b. Isolate normal and excess letdown flow paths to maintain RCS boundary integrity.

2. Long-Term Requirements

a. Provide reactor coolant inventory control if necessary through use of RCP seal injection flow path and/or the normal charging flow path, and through use of the normal letdown flow path to the VCT and the recycle holdup tank. (A redundant letdown path can be provided through the reactor head vent system.)

b. Provide boration capability to go to cold shutdown using boric acid in the boric acid tanks, boric acid transfer pumps, and charging pumps.

c. Depressurization via the auxiliary pressurizer spray line to lower the RCS pressure so that the RHR system can be initiated. (This is a redundant method to using the pressurizer PORVs.)

F. Residual Heat Removal System

1. Short-Term Requirements - Maintains isolation of RHR inlet to ensure RCS boundary integrity.

2. Cold Shutdown Requirements - Remove heat from the reactor coolant system through use of the RHR heat exchangers, component cooling water system, and service water system.

G. Safety Injection System

1. Short-Term/Long-Term Requirements - Ensure RCS boundary integrity.

H. Reactor Coolant System

1. Short-Term Requirements

a. Maintain reactor coolant circulation.

b. Ensure RCS boundary integrity.

c. Maintain RCS pressure control using pressurizer heaters or charging pumps.

2. Long-Term Requirements

a. Volume control (letdown) using reactor head vent system. (A redundant letdown path can be provided by the CVCS normal letdown flow path.)

b. Pressure control using the pressurizer PORVs and heaters. (A redundant method for depressurization can be provided by portions of the CVCS system and the pressurizer auxiliary spray line.)

I. Rod Control System

1. Short-Term/Long-Term Requirements - Ensure that the reactor is tripped (control rods inserted) and that control rods remain inserted.

J. HVAC Systems

1. Short-Term Requirements - Not applicable.

2. Long-Term Requirements - Provide room cooling for the main control room, charging pump room, service water intake structure, battery charger room, RHR pump room, related electrical switchgear rooms, and diesel building.

K. Power Distribution System and Emergency Diesel Generators

1. Short-Term Requirements - Provide electrical power for operation of all components and instrumentation needed to perform the immediate requirements for achieving and maintaining hot standby as defined in this section.

2. Long-Term Requirements - Provide electrical power for operation of all components and instrumentation needed to meet the requirements for maintenance of hot standby, and achieving and maintaining cold shutdown as defined in this section.

3. Loss of offsite power (LOSP) is considered in this determination.

L. Instrumentation

1. Short-Term/Long-Term Requirements - Provide monitoring capability for the primary and secondary heat transfer loops. (Pressurizer pressure and level, reactor coolant (RC) hot leg temperature, RC cold leg temperature, steam generator pressure and level, source range flux monitor, and condensate storage tank level).

M. Communications Systems

1. Short-Term/Long-Term Requirements - To facilitate an orderly shutdown resulting from a fire, provide communications between the control room, hot shutdown panel areas, and other areas of the plant where shutdown actions must be carried out.

N. Sampling System

1. Short-Term Requirements - Not applicable.

2. Long-Term Requirements - Manual sampling to ensure that boration to cold shutdown margin is achieved.

A detailed listing of safe shutdown components is contained in section 5.1.9 of the Appendix R Compliance Report (A-350971). In addition, safe shutdown procedures have been developed which govern control room actions and fire area actions to ensure safe plant shutdown.

9B.5.2.2.3 Radioactivity Control

Any liquid or gas, including fire suppression mediums, from potentially radioactive areas will be sampled prior to release. All releases from these systems will be monitored and reported as required by technical specifications.

9B.5.2.3 Fire Area Hazard Analysis

This section provides an area-by-area analysis of the fire hazards and potential consequences for fire areas of the Farley Nuclear Plant Units 1 and 2. Installed fire protection features, shutdown trains, and design features relating to fire safety of the area are provided. An inventory of combustible and flammable material is also given. The Appendix R Compliance Report contains a discussion of the consequences of a fire in a given fire area that contains equipment required for safe shutdown.

Evaluation sheets for each fire area or zone are provided in Attachment A. Refer to figures 9B-4 through 9B-71 for plant location.

9B.6 INSPECTION AND TESTING PROGRAM

Successful fire fighting requires testing and maintenance of the fire protection equipment, emergency lighting, and communication, as well as practice as brigades by the people who must utilize the equipment. Plant procedures contain the responsibilities of individuals in connection with routine tests and inspections of fire detection, fire suppression, and associated equipment to ensure this equipment is maintained in an operable condition. The following subsections describe the inspection and testing program established at the plant.

9B.6.1 INSPECTION PROGRAM

The fire protection inspection program includes general housekeeping inspections and associated corrective actions, inspections of fire suppression and detection systems and fire barrier penetrations required to support safe plant shutdown, periodic inspections of other fire protection systems and associated equipment, and independent inspection of activities affecting fire protection to verify conformance to documented installation drawings and test procedures.

In accordance with plant administrative controls, the fire marshal inspects each plant area on an unscheduled basis, but at least quarterly, for general housekeeping inspections. The results of these inspections are reported to the assistant general manager-plant operations or the assistant general manager-plant support, as appropriate. Additionally, the assistant general managers also conduct a semiannual inspection of all accessible areas to determine compliance with housekeeping instructions.

Operability requirements and surveillance/inspection requirements for fire suppression systems, fire detection systems, and fire barrier penetrations required to support the safe shutdown of the plant are provided in Attachment C to this appendix. The inspection requirements in Attachment C verify the operability requirements are maintained such that adequate detection, suppression, and confinement capabilities exist at all times to protect safe shutdown equipment and minimize the effects of a fire.

Periodic inspections of fire protection systems and equipment and associated systems and equipment, beyond those contained in Attachment C, are also performed to assure the acceptable condition of these items. These inspections, which are governed and controlled by plant procedures, include fire protection systems and equipment not included in Attachment C, emergency breathing and auxiliary equipment, emergency lighting, and communication equipment.

Independent inspection of activities affecting fire protection is established to verify conformance to documented installation drawings and test procedures. This inspection program includes:

A. Inspections of installation, maintenance, and modification of fire protection systems, and inspections of emergency lighting and communication equipment, to assure conformance to design and installation requirements.

B. Inspection of penetration seals and fire retardant coating installations and modifications to verify the activity is satisfactorily completed.

C. Inspections of cable routing to verify conformance with design requirements.

D. Inspections to verify that appropriate requirements for room isolation (sealing penetrations, floors, and other fire barriers) are accomplished during construction of new facilities at the plant. Inspections of these room isolation requirements were accomplished for the original plant design during plant construction.

E. Measures to assure that inspection personnel are independent from the individual performing the activity being inspected and are knowledgeable in the design and installation requirements for fire protection.

F. Inspection procedures, instructions, and checklists which provide the following:

1. Identification of characteristics and activities to be inspected.

2. Identification of the individuals or groups responsible for performing the inspection operation.

3. Acceptance or rejection criteria.

4. A description of the method of inspection.

5. Recording evidence of completing and verifying a manufacturing, inspection, or test operation.

6. Recording inspector or data recorder and the results from the inspection operation.

7. Identification by means of tags, labels, or similar temporary markings to indicate completion of required inspections and tests and operating status.

9B.6.2 TESTING PROGRAM

Following construction, modification, repair or replacement t- fire protection systems, emergency lighting, and communication equipment, sufficient testing is performed to demonstrate that the system or equipment will perform satisfactorily in service and that design criteria are met. The test and test control requirements are incorporated in approved plant procedures. These controls include:

(1) written test procedures for installation tests that incorporate the requirements and acceptance limits contained in applicable documents,

(2) a program for quality assurance to verify testing of fire protection systems and to verify that test personnel are effectively trained,

(3) documentation and evaluation of test results, and

(4) the determination of the acceptability of test results by a qualified responsible individual or group.

Quality control of the test is provided as required by the complexity of the work. Plant procedures also include provisions for identifying the operating status of items during tests and upon successful completion of these required tests.

Fire protection equipment, emergency lighting, and communication equipment are also tested periodically to assure that the equipment will properly function and continue to meet its design criteria. The appropriate schedules and methods for periodic testing of this equipment are incorporated in the fire surveillance procedures.

TABLE 9B-1

(This table has been intentionally deleted.)

TABLE 9B-3

(This table has been intentionally deleted.)

TABLE 9B-4

(This table has been intentionally deleted.)

TABLE 9B-6

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TABLE 9B-7

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FIRE AREA NO. 1-4 (SHEET 2 OF 10)

Room Numbers/Title	Location	Shutdown Train	Floor Area (ft^2)	Combustible Material	Maximum Fire Load (Btu/ft^2)	Maximum Fire Severity
154 Waste Evaporator Steam Generator Room	2-F.5, 18.5	B	225			
155 Passageway to Unit 2(1)	2-G.2, 18.5	B, N	88			
160 Hatch Area	2-M.5, 18.5	A, B	1220			
161 Corridor	2-K.0, 17.5	N, A, B, C	639	Cable insulation Lube Oil	≤ 40,000	< 30 min
162 Hallway	2-F.0, 17.0	C, N, A, B	809	Panel		
163 WDS Control Panel Room	2-C.7, 16.5	C, N, A, B	638	Storage Lockers		
164 Laundry and Hot Shower Tank Room/Storage Room	2-B.5, 17.0	B, C, A	312			
177 Pump Room	2-N.5, 15.0	-	122			
178 Filter Room	2-D.5, 15.0	-	72			
215 Duct and Pipe Chase	3-E.3, 16.2	-	78	None	0	0
176 Secondary Spent-Resin Storage Tank Room	2-M.8, 15.0	-	121	None	0	0
180 Recycle Evaporator Steam Generator Room	2-M.7, 13.5	B	261	Cable insulation		
186 Boric Acid Area	2-M.5, 11.0	A, B, C	908	Cable insulation Lube oil Water stop material	≤ 40,000	< 30 min
187 Hydro Test Pump Room	2-O.0, 12.5	-	141	Cable insulation Lube oil		
204 Waste Evaporator Package Room	3-F.3, 18.5	-	396	None	0	0
219 Pipe Chase	3-O.6, 16.5	-	32	None	0	0

FIRE AREA HAZARD ANALYSIS
FIRE AREA NO. 1-6 (SHEET 1 OF 2)

Location: Unit 1 Auxiliary Building el 100 ft 0 in., 121 ft 0 in., 139 ft 0 in., 155 ft 0 in., 175 ft 0 in.

Figures: 9B-8, 9B-10, 9B-11, 9B-15, 9B-16, 9B-18, 9B-23, 9B-25, 9B-28

Room Numbers/Title	Location	Shutdown Train	Floor Area (ft^2)	Combustible Material	Maximum Fire Load (Btu/ft^2)	Maximum Fire Severity
185 Component Cooling Water Heat Exchanger Room	2-B.7, 10.9	A, B, C, N	3844	Cable insulation 5-kV disconnect switch Pump lube oil	≤ 80,000	< 1 h
191 Auxiliary Feedwater Pump Room	2-A.9, 4.2	A	259	Cable insulation Pump lube oil	≤ 40,000	< 30 min
192 Auxiliary Feedwater Pump Room	2-A.5, 2.5	A, B, C	261	Cable insulation Pump lube oil	≤ 80,000	< 1 h
189 Plant Heating Equipment Room	2-C.6, 4.5	A, B, C, N	833	Cable insulation Lube oil		
190 Motor Control Center 1E Room	2-B.3, 4.8	A, B, C, N	722	Cable insulation Panel		
199 Phosphate Tank and Pump Area[b]	2-A.9.4.8	(a)	(a)	(a)		
193 Auxiliary Feedwater Pump Room	2-B.6, 2.4	A, B, C, N	287	Cable insulation Pump lube oil	≤ 40,000	< 30 min
194 Equipment Room	2-C.8, 2.5	A, B, C, N	919	Cable insulation		
195 Access Hatch Room	2-C.5, 1.5	A	163	Cable insulation		
241 Main Steam and Feedwater Valve Room	8-C.0, 4.5	A, B, C, N	3552	Cable insulation Polyethylene Sheathing[c] Wood Wedge	≤ 40,000	< 30 min
242 Pipe Chase	19-A.0, 5.0	-	167	None	0	0
243 Pipe Chase	19-A.0, 3.0	-	167	None	0	0

FIRE AREA HAZARD ANALYSIS
FIRE AREA NO. 1-55 (SHEET 1 OF 2)

Location: Unit 1 Containment

Figures: 9B-28

Room Numbers/Title	Location	Shutdown Train	Floor Area (ft^2)	Combustible Material	Maximum Fire Load (Btu/ft^2)	Maximum Fire Severity
Containment	12-J.0, 7.0	A, B, N Channels 1, 2, 3, 4	13,273	Cable insulation Pipe insulation Lube oil Panels Charcoal filters	≤ 120,000	< 1.5 h

DESIGN FEATURES

Construction: The containment is enclosed by prestressed concrete and is a vertical right cylindrical structure with a dome and flat base. The containment interior is lined with steel plate for leaktightness. All penetration openings are watertight but not fire rated. Access to the containment is by the pressure-tight personnel access door and auxiliary access door at el 155 ft 0 in.

Doors: Nonfire-rated personnel airlocks (No. 466 and 467) and equipment hatch (No. 468).

Piping and Electrical Penetrations: All penetrations through the area boundary are sealed and watertight but not fire rated.

Ventilation: Normal ventilation is provided by the containment air coolers and the minipurge system.

Floor Drains: A sump is provided with redundant pumps which discharge to either the floor drain tank or the waste holdup tank.

FIRE PROTECTION

The fire detection systems inside the containment building are:

- Forty-eight ionization smoke detectors for containment coolers (i.e., 12 each for 4 containment coolers).
- Three ionization smoke detectors for steam generators (i.e., one for each steam generator).
- Eight ionization air duct detectors (i.e., two each for four containment venting fans).
- One polarized alarm bell located near personnel access lock.
- One manual fire alarm station located near the alarm bell.
- Temperature detectors in filters.

FIRE AREA NO. 2-4 (SHEET 2 OF 8)

Room Numbers/Title	Location	Shutdown Train	Floor Area (ft^2)	Combustible Material	Maximum Fire Load (Btu/ft^2)	Maximum Fire Severity
2164 Storage Room	2-B.5, 21.0	A	312	Cable insulation	≤ 40,000	< 30 min
2215 Duct and Pipe Chase	3-E.3, 21.8	-	78	None	0	0
2176 Secondary Spent-Resin Storage Tank Room	2-M.8, 23.0	-	121	None	0	0
2180 Recycle Evaporator Steam Generator Room	2-M.7, 24.5	B	207	Cable insulation		
2186 Boric Acid Area	2-M.5, 27.0	A, B	979	Cable insulation Lube oil Water stop material	≤ 40,000	< 30 min
2187 Hydro Test Pump Room	2-O.0, 25.5	-	133	Cable insulation Lube oil		
2204 Waste Evaporator Package Room	3-F.3, 19.5	Channel 1	434	None	0	0
2219 Pipe Chase	3-O.6, 21.5	-	32	None	0	0
2203 Waste Condenser Tanks and Pump Room	3-E.3, 19.5	A Channel 1	338	Cable insulation		
2205 Passageway to Unit 1	3-G.2, 19.5	-	91	Cable insulation		
2207 Hatch Area	3-M.8, 19.5	B	1011	Cable insulation Storage lockers		
2208 Corridor	3-L.0, 20.5	A, B, N Channel 1	475	Cable insulation Panels		
2209 Hallway	3-G.0, 22.0	A, B, N Channel 1	1480	Cable insulation Panels Storage lockers	≤ 80,000	< 1 h
2218 Chiller Unit Room	3-L.0, 22.0	A, B, Channel 1	674	Cable insulation Panels Lube oil		
2222 Corridor	3-M.1, 24.5	A, B	430	Cable insulation		
2237 Corridor	3-M.5, 28.0	-	620	Cable insulation		
2253 Valve Compartment	3-M.0, 19.3	N	84	Cable insulation		
2156 Holdup Tank Room	2-G.9, 19.5	B	308	Cable insulation	≤ 40,000	< 30 min
2157 Holdup Tank Room	2-J.5, 19.5	B	308	None	0	0

FIRE AREA HAZARD ANALYSIS
FIRE AREA NO. 2-6 (SHEET 1 OF 2)

Location: Unit 2 Auxiliary Building el 100 ft 0 in., 121 ft 0 in., 139 ft 0 in., 155 ft 0 in., 175 ft 0 in.
Figures: 9B-33, 9B-44, 9B-46, 9B-47, 9B-52, 9B-56, 9B-57, 9B-61, 9B-66

Room Numbers/Title	Location	Shutdown Train	Floor Area (ft^2)	Combustible Material	Maximum Fire Load (Btu/ft^2)	Maximum Fire Severity
2185 Component Cooling Water Heat Exchanger Room	2-B.7, 27.1	A, B, C, N	3898	Cable insulation 5-kV disconnect switch Pump lube oil	≤40,000	<30 min
2191 Auxiliary Feedwater Pump Room	2-A.9, 31.8	A	256	Cable insulation Pump lube oil	≤ 40,000	< 30 min
2192 Auxiliary Feedwater Pump Room	2-A.5, 34.5	A, B, C, N	256	Cable insulation Pump lube oil	≤ 80,000	< 1 h
2189 Plant Heating Equipment Room	2-C.6, 31.5	A, B, C, N	833	Cable insulation Lube oil		
2190 Motor Control Center 1E Room	2-B.3, 31.2	A, B, C, N	710	Cable insulation Panel		
2193 Auxiliary Feedwater Pump Room	2-B.6, 34.6	A, B, C, N	287	Cable insulation Pump lube oil	≤ 40,000	< 30 min
2194 Equipment Room	2-C.8, 34.5	A, B, C, N	932	Cable insulation		
2195 Access Hatch Room	2-C.5, 36.5	A	163	Cable insulation		
2236 Duct Chase	4-C.9, 8.2	-	78	None	0	0
2241 Main Steam and Feedwater Valve Room	8-C.0, 33.5	A, B, C, N	3558	Cable insulation Polyethylene Sheathing[b] Wood Wedge	≤ 40,000	< 30 min
2242 Pipe Chase	19-A.0, 33.0	-	170	None	0	0
2243 Pipe Chase	19-A.0, 35.0	-	170	None	0	0

DESIGN FEATURES

Construction: Floors, walls, and ceilings forming the area boundary are of reinforced concrete or of steel grating open to the exterior. Self-expanding cork is installed in portions of the area boundary along the containment wall which are also boundaries for other fire areas. There is a removable nonrated steel hatch between rooms 2185 and 2234 (area 2-20). This hatch has smoke detection and automatic water suppression systems installed on both sides. Access Hatch room 2195 has nonrated grated walls above the grade level and is provided with a nonrated removable steel hatch cover between the room and the exterior. The component cooling water pumps have been provided with curbing. There is also a nonrated, bolted steel access plate between rooms 2236 and 2346 (area 2-41) which is designed to provide a heat barrier.

10.3 MAIN STEAM SUPPLY SYSTEM

The main steam supply system (MSSS) carries the steam generated in the three steam generators through the containment to the following components and systems:

A. Turbine-generator.

B. Moisture separator reheaters.

C. Steam jet air ejector system.

D. Turbine shaft gland seals.

E. Steam generator feedwater pump turbines.

F. Turbine-driven auxiliary feedwater pump.

G. Turbine bypass system.

Figure 10.3-1 shows the schematic arrangement of the MSSS piping. The main steam piping up to and including the isolation valves in the main steam lines to the turbine-generator and the main steam piping to the auxiliary feedwater pump turbine have safety-related functions.

10.3.1 DESIGN BASES

10.3.1.1 Functional Requirements

The MSSS conducts the generated steam from the outlet of the steam generators to the various system components. The steam is used for various operational auxiliary services such as shaft steam seals, turbine drives for main and auxiliary feedwater pumps, and steam jet air ejectors, as well as for its principal purpose of supplying the main turbine and reheaters. This system also provides steam to other systems associated with steam generator pressure relief heat removal from the nuclear steam supply system (NSSS), such as the steam generator safety valves, relief valves, and steam dump valves.

The performance requirements for the MSSS are as follows:

A. Optimum pressure drop between steam generators and turbine valves.

B. Similar steam conditions between each turbine stop valve and between each steam generator must be maintained. A maximum pressure variance between one steam generator and any other of 10 psi or less is maintained at the steam generator outlet. To ensure even steam pressure at all loads, the main steam piping is interconnected downstream of the isolation valves and also prior to the turbine stop valves.

C. Adequate piping flexibility is provided to limit forces and moments at the anchor points and upon all plant components to acceptable levels. Stress levels within the piping itself are within the limits specified in the applicable piping codes.

D. Adequate draining provisions for startup and for operation with saturated steam are provided. All low points and closed-end lines are drained to preclude any water accumulation.

10.3.1.2 Safety Requirements

The safety aspects of the main steam system are concentrated on the portions of main steam lines from the steam generators, out through the containment, and up to and including the main steam line isolation valves.

The safety requirements for the MSSS are as follows:

A. The steam lines and the shell side of the steam generator are basically considered as an extension of the containment boundary and, as such, must not be damaged as a consequence of reactor coolant system (RCS) damage. The steam generator shell and steam lines within the containment are therefore protected against a reactor coolant system missile. The reverse is also true in that a steam line break will not cause damage to the reactor coolant system.

B. The measured steam flowrate has a functional requirement. Functionally, the flowrate signal is used by the three-element feedwater controller and as a load index signal for the plant's variable speed main feedwater pumps. It is also used in the development of a steam flow-feed flow mismatch alarm.

The flowrate is determined by measuring the dynamic pressure losses between the steam generato and a

D. Preventing scale deposits on the steam generator heat transfer surfaces and in the turbine.

E. Minimizing feedwater oxygen content prior to entry into steam generator.

F. Controlling condenser air ejector radioactive iodine effluent releases.

These objectives are met by exercising careful chemistry control over the systems, including comprehensive sampling and analysis (inline and laborabory), chemical injection at selected points, continuous system blowdown from the steam generator, and effective protection of the steam generator and feedwater train internals during periods of inactivity.

The method of secondary water chemistry control used is morpholine/hydrazine, morpholine/hydrazine/boric acid, or monoethanolamine/hydrazine/boric acid. Morpholine and monoethanolamine have proven in both laboratory and field experience to be beneficial for reducing erosion-corrosion of secondary side components and subsequent corrosion product transfer to steam generators. Hydrazine is added to scavenger dissolved oxygen to within appropriate limits. The decomposition of hydrazine produces ammonia in sufficient quantity such that the intentional addition of ammonia for pH control is not required. Boric acid is injected into the secondary system to inhibit stress corrosion cracking of steam generator tubes.

Controlling system pH to achieve proper alkaline conditions reduces general corrosion and decreases the release of soluble corrosion products from metal surfaces. Ensuring the absence of free caustic eliminates the possibility of caustic stress corrosion.

Reducing dissolved oxygen to the lowest possible levels also contributes to diminished rates of general corrosion. Oxygen in the system is removed from the condenser by the steam jet air ejectors and is further reduced by scavenging with hydrazine introduced at the condensate pump discharge.

Excluding other impurities from the steam generator reduces scale formation on heat transfer surfaces and prevents corrosion caused by concentration of the reactant products of these impurities. Addition of impurities to the steam generator is limited by the careful control of feedwater purity. Of prime importance in this regard is an aggressive program of condenser maintenance to ensure its leak-tight integrity. The concentration effect of the impurities in the steam generator is minimized through continuous blowdown.

A steam generator partition factor of 0.1 and a condenser air ejector partition factor of 10^{-4} have been used in the evaluation of environmental consequences of postulated accidents (e.g., Steam Generator Tube Rupture, subsection 15.4.3.4). These are conservative values for the range of water chemistry allowed by WCAP-7452 based on measurements made at operating Westinghouse plants.

10.3.6 INSTRUMENTATION APPLICATIONS

The steam flow restrictors installed in the steam generators are also used for steam flow measurements during normal operation. Two flow transmitters and two pressure transmitters are installed in the main steam line from each steam generator. The steam flow and pressure signals are fed into reactor protection and feedwater control system circuits to control the feedwater flow to each steam generator, to close the isolation valves in case of rupture in main steam lines, and to open the power-operated relief valves in case of overpressure.

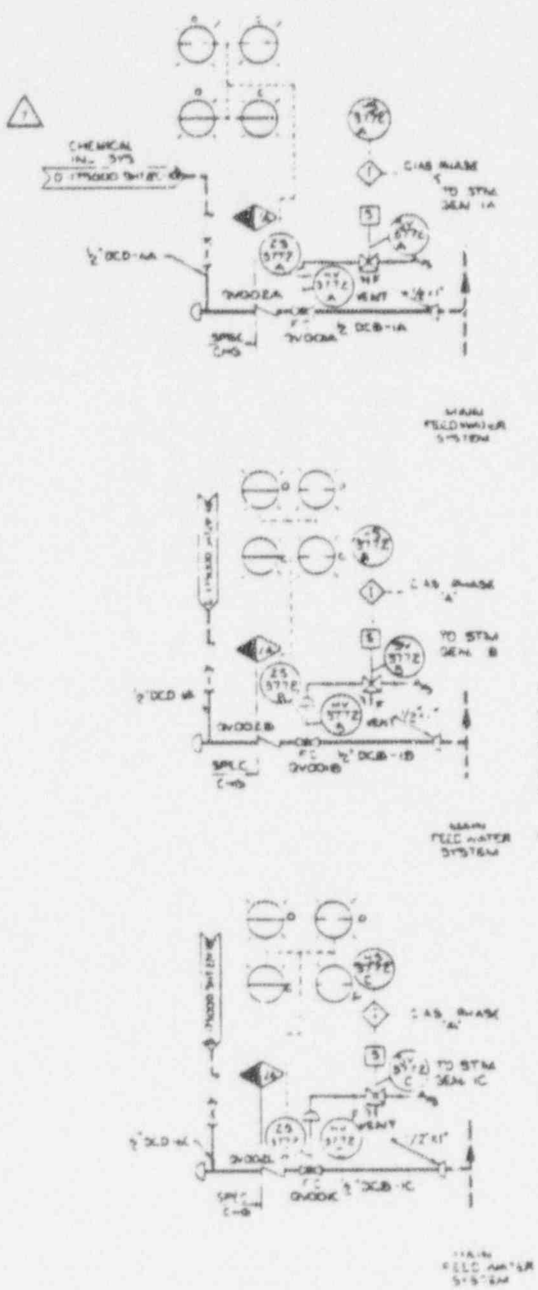
10.3.7 MAIN STEAM SAFETY VALVES

Overpressure protection for the three steam generators is provided by the main steam safety valves. The design basis for these automatic pressure-relieving devices is that the valves must pass 110 percent of the maximum guaranteed steam flow at a steam generator shell pressure not greater than 110 percent of design pressure of the steam generator shell. This is the maximum pressure allowed by the applicable code.

Design parameters for the main steam safety valves are given in table 10.3-1.

Due to the large mass flowrate, each steam generator is protected by a number of valves. The maximum actual capacity of a single valve fully open at 1085 psi gauge does not exceed 890,000 lb/h. This provision serves to limit steam release if any one valve inadvertently sticks open.

The main steam safety valves are located on the main steam lines outside the containment and upstream of the main steam isolation valves. Each of the three main steam lines is equipped with five safety valves. To prevent chattering during operation of the safety valves, each of the five valves on a steam line is set at a different set pressure. The first valve set pressure is 1075 psig, which corresponds to the steam generator shell design pressure minus the pressure loss from the steam generator to the valve. Each of the remaining valves is set at a higher pressure such that all valves are open and



NOTES:

1. ALL VALVE NUMBERS IN THIS SYSTEMS ARE PREFIXED BY N1N2S
2. FOR PIPING CLASS SUMMARY SHEETS SEE SPEC. SS-1109-1
3. FOR VALVE INFORMATION SEE MASTER VALVE LIST D-175504
4. FOR EQUIPMENT LIST SEE DWG. D-175070
5. FOR INSTRUMENT INSTALLATION DETAILS SEE REFS ENGERN INSTRUMENT INDEX
6. FOR PIPING CLASS SHEETS SEE SPEC SS-1109-1
7. * INDICATES VENDOR FURNISHED ITEM

REV 12 10/94

D-175000 REV 7

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CHEMICAL INJECTION SYSTEM

FIGURE 10.3-4

10.4 OTHER FEATURES OF THE STEAM AND POWER CONVERSION SYSTEM

This section provides information on the principal design features and subsystems of the steam and power conversion system.

10.4.1 MAIN CONDENSER

10.4.1.1 Design Bases

The main condenser is designed to function as the steam cycle heat sink and collection point for the following flows:

A. Main turbine exhaust.

B. Feedwater heater drains and vents.

C. Steam generator feedwater pump turbine exhaust.

D. Turbine bypass system flow.

E. Condensate and steam generator feedwater pump recirculation flow.

F. Condensate and feedwater system makeup flow.

G. Steam jet air ejector intercondenser and aftercondenser drains.

H. Gland steam condenser drain.

I. Miscellaneous equipment drains and vents.

The function of the main condenser is to provide a heat sink for the main turbine exhaust, feed pump turbine exhaust, turbine bypass steam, and other flows. It also deaerates and provides storage capacity for secondary system condensate.

10.4.1.2 System Description

The condenser is a two-shell, radial-flow, single-pass unit which is connected to the exhaust opening of each low-pressure turbine casing by single-convolution, stainless steel expansion joints. Each condenser shell is supported from the turbine building base slab by four concrete pads, and there are pressure-equalizing lines between the shells. These lines ensure that the maximum differential temperature between turbine exhausts will not exceed 30°F.

The condenser has two (one each shell) 3600 ft^3 hot wells which provide sufficient water storage capacity to accomodate system surges and condensate makeup during moderate transients without relying upon reserve condensate storage.

Each circulating water inlet and outlet line to the condenser is provided with a motor-operated shutoff valve.

The condenser has approximately 52,164 tubes giving it an effective surface area of approximately 680,000 ft^2.

During normal operation, the condenser uses 635,000 gal/min of cooling water to remove 63.5 x 10^8 Btu/h of energy from the main turbine exhaust, feed pump turbines exhaust, heater drains, and various other sources.

The noncondensible gasses are concentrated in the air cooling section of the condenser shells, from which they are removed by the steam jet hogging ejectors during startup and by the steam jet air ejectors during normal operation (see subsection 10.4.2). The condenser is designed to have no air leakage and will limit the free oxygen content in the condensate to a maximum of 0.0005 cl^3 at the hot well.

In addition to serving the usual function of a condenser, each shell is capable of accepting 2,325,000 lb/h of turbine bypass steam at an enthalpy of 1197.2 Btu/lb and a pressure of 250 psig without exceeding an exhaust hood temperature of 175°F and a back pressure of 6 in. Hg abs.

10.4.1.3 Safety Evaluation

The main condenser is normally used to remove residual heat from the reactor coolant system (RCS) during the initial cooling period after plant shutdown when the main steam is bypassed to the condenser by the turbine bypass system. The condenser is also used to condense the main steam bypassed to the condenser in the event of sudden load rejection by the turbine-generator or a turbine trip.

In the event of load rejection above 50 percent (including 100-percent load rejection due to turbine trip), the condenser will condense 40 percent of full-load main steam flow bypassed to it by the turbine bypass system, and the power-operated relief valves and spring-loaded safety valves will discharge remaining main steam flow to atmosphere to effect safe reactor shutdown and to protect the main steam supply system from overpressure.

If the main condenser is not available during normal plant shutdown, sudden load rejection, or turbine-generator trip, the

Piping and valving inside and outside containment to the containment isolation valve are designed and fabricated to ANS Safety Class 2A requirements. As this system performs no function related to safe shutdown of the plant, all components downstream of the containment isolation valve are classified as NNS Safety Class. The rupture of components downstream of the containment isolation valve will generally require shutdown of the plant.

10.4.8.3.3 Analysis of Shell-Side Radioactivity Concentration During Blowdown Processing System Isolation

The operation criteria for the secondary-side blowdown system are dictated by the need for limiting the secondary-side buildup of dissolved solids.

Hence, it is unlikely that the plant may operate with the blowdown system isolated. However, assuming system isolation coincident with 1-percent fuel defects and 144-gal/d primary-to secondary-side leakage, the secondary-side activity will build up to approximately 5×10^{-2} $\mu Ci/cm^3$ in one 24-h period.

10.4.8.4 Tests and Inspections

Periodic tests and recalibration will be required on the radiation monitors in the blowdown processing system as described in the technical specifications. Periodic tests of the blowdown isolation valve functioning will be performed to check operability and leaktightness as specified by Section XI of the ASME Boiler and Pressure Vessel Code, 1983 Edition through Summer 1983 Addenda. Periodic visual inspections and preventive maintenance can be conducted as necessary when all components are available for inspection.

10.4.8.5 Safety Evaluation

The following limiting conditions for operation and surveillance requirements for secondary water monitoring requirements provide assurance that steam generator tube integrity is not reduced below an acceptable level for adequate margins of safety. These limiting requirements are as follows:

A. The specific activity of the secondary coolant system shall be $\leq$ 0.10 $\mu Ci/g$ dose equivalent I-131.

B. No more than 420 gal/d for Unit 1 and 450 gal/d for Unit 2 total primary-to-secondary leakage through all steam generators and no more than 140 gal/d for Unit 1 and 150 gal/d for Unit 2 through any one steam generator.

C. Each steam generator shall be operable. With one or more steam generators inoperable, restore the

inoperable generator(s) to operable status prior to increasing T_{avg} above 200°F.

D. Eddy current testing indications below 20 percent of the nominal tube wall thickness.

E. The plugging limit or imperfection depth at or beyond which the tube shall be removed from service is equal to 40 percent of the nominal wall thickness.

11.0 RADIOACTIVE WASTE MANAGEMENT

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11.2 LIQUID WASTE SYSTEMS

11.2.1 DESIGN OBJECTIVES

The liquid waste processing system (LWPS) is designed to receive, segregate, process, recylce, and discharge liquid wastes. The system design considers potential personnel exposure and ensures that quantities of radioactive releases to the environment are as low as reasonably achievable. Under normal plant operation, the total activity from radionuclides leaving the LWPS does not exceed a small fraction of the discharge limits defined in column 2, Table II, Appendix B to 10 CFR 20.1 - 20.601.

Further, overall radioactive release limits are established as a basis for controlling plant discharges during operation with the occurrence of a combination of equipment faults of moderate frequency. A combination of equipment faults that could occur with moderate frequency include operation with fuel cladding defects in combination with such occurrences as:

A. Steam generator tube leaks.

B. Malfunction in LWPS.

C. Excessive leakage in reactor coolant system equipment.

D. Excessive leakage in auxiliary system equipment.

The radioactive releases from the plant resulting from equipment faults of moderate frequency are within the column 2, Table II, Appendix B to 10 CFR 20.1 - 20.601 limits on a short-term basis and do not exceed four to eight times the limits stated previously for normal operation on an annual average basis.

11.2.2 SYSTEM DESCRIPTIONS

The LWPS collects and processes potentially radioactive wastes for recycle or for discharge. Provisions are made to sample and analyze fluids before they are recycled or discharged. Based on the laboratory analysis, these wastes are either released under controlled conditions via the cooling water system or retained for further processing. A permanent record of liquid releases is provided by analyses of known volumes of waste.

The radioactive liquids discharged from the reactor coolant system can be processed by the boron recycle system or the disposable demineralizer system described in paragraph 11.2.3.1.8. The limited amount of fuel leakage

experienced in the plant operating history has enabled the use of the disposable demineralizer system to process the bulk of the reactor coolant system radioactive liquid discharges. The operation of the demineralizer system results in a smaller volume of waste to be shipped offsite for disposal. The permanently installed boron recycle system remains available for use to ensure that the technical specification limits are met. The use of the disposable demineralizer system or the boron recycle system limits input to the LWPS and results in processing of relatively small quantities of generally low activity wastes.

The LWPS is arranged to recycle reactor grade water if possible. This is implemented by the segregation of equipment drains and waste streams, which prevents the intermixing of liquid wastes. The LWPS consists of two main subsystems designated as drain channel A and drain channel B. Drain channel A processes water which can be recycled, and drain channel B processes water which is to be discharged. A drain system is also provided inside the containment to collect drains and leaks and transfer them to an appropriate tank. Capability for handling and storage of spent demineralizer resins is also provided.

Additionally, the plant has been equipped with a disposable demineralizer system described in paragraph 11.2.3.1.8. This system is capable of processing water from any of the waste streams and producing a very low activity effluent. Water processed through the disposable demineralizer system is routed to the waste monitor tank for analysis prior to release.

Instrumentation and controls necessary for the operation of the LWPS are located on a control board in the auxiliary building. Any alarm on this control board is relayed to the main control board in the control room.

Process flow diagrams and piping and instrumentation diagrams are shown in figures 11.2-1 through 11.2-6. All lines in the LWPS, including field run, are considered potential carriers of significant radioactivity.

Table 11.2-1 also gives process parameters for key locations in the system. Expected volumes to be processed by the LWPS are given in table 11.2-2. Assuming the volumes presented in table 11.2-2 are processed at a uniform rate, the input to the waste evaporator will be approximately 0.2 gal/min, while the evaporator is designed to handle 15 gal/min. Hence, excess capacity is available to handle abnormal operating conditions. This will only change the load on the system; otherwise the operating features will not change. Component failures in the

Pressure indicators upstream and downstream of filters, strainers, and demineralizers provide local indications of pressure drops across each component.

All liquid releases from the LWPS are monitored for radioactivity by a scintillation counter. This instrumentation is further described in section 11.4.

11.2.4 OPERATING PROCEDURES

The LWPS is manually operated except for some functions of the reactor coolant drain tank circuit and the disposable demineralizer system. The system includes adequate control equipment to protect the system components and adequate instrumentation and alarm functions to provide operator information to ensure proper system operation. All pumps in the system have low level shutoffs, and all filters and demineralizers have pressure indication upstream and downstream to indicate fouling.

11.2.4.1 Normal Operation

Operation of the LWPS is essentially the same during all phases of normal reactor plant operation; the only differences are in the load on the system. The following sections discuss the operation of the system in performing its various functions. In this discussion, the term "normal operation" should be taken to mean all phases of operation except operation under emergency or accident conditions. The LWPS is not regarded as an engineered safety feature system.

11.2.4.1.1 Recycle Portion

Water is accumulated in the waste holdup tank until sufficient quantity exists to warrant an evaporator startup, to switch the evaporator operation from the floor drain tank to the waste holdup tank, or to process through the disposable demineralizer system.

During evaporation the distillate is checked for boron and activity concentration, and if the analysis shows compatibility with reactor makeup grade water, it is transferred to the reactor makeup water storage tank. If the distillate is high in boron concentration or activity, it may be passed through the waste evaporator condensate demineralizer before being transferred to the reactor makeup water storage tank. If reevaporation is required and the waste evaporator is not available, then the distillate can be transferred to the boron recycle holdup tanks for processing by the boron recycle

evaporator. The bottoms from the waste evaporator may be concentrated to approximately 12-percent boric acid but are normally concentrated to a low boric acid concentration and dumped to the waste holdup tank. Should the bottoms be acceptable for recycle, they are concentrated to approximately 4-percent boric acid and transferred to the boric acid tanks.

During normal operation, the reactor coolant drain tank level regulation and pressure control are automatic and require no operator action.

Operation of the recycle portion of the LWPS during refueling is the same as for power operation, although the load on the system may be increased when refueling is complete. The water remaining in the canal following normal drain down is pumped to the suction of the refueling water purification pump by the reactor coolant drain tank pumps.

11.2.4.1.2 Waste Portion

The waste portion of the LWPS consists of two subsystems: the laundry and hot shower system and the floor drain tank system.

Laundry and hot shower water enters the laundry and hot shower tank for holdup. The water is filtered and transferred to the waste monitor tank where it is sampled and discharged.

The water in the floor drain tank is sampled to determine the degree of processing required. It can be sent directly to the waste monitor tank provided for floor drain tank water or to the waste monitor tank via the waste monitor tank demineralizer, or it can be processed through the waste evaporator or the disposable demineralizer system. If the water is evaporated, the distillate is sent to the waste monitor tank and the concentrate is recycled or solidified. The water in the waste monitor tank is again sampled and can be recirculated through the waste monitor tank demineralizer if further processing is required. When this water has been sufficiently processed, it is discharged into the plant discharge line at a rate so as not to exceed a small fraction of technical specification limits. A process decontamination factor in the range of 7.2×10^3 to 7.2×10^5 is expected for the evaporator demineralizer combination. Chemical trace element tests as well as operating experience on similar evaporators have justified this process decontamination factor.

Water leaving this system to the discharge canal is monitored for radiation. Should the radiation monitor close the discharge valve, it must be cleared before the valve can be reopened. The monitor element can be cleared by flushing it with demineralized water from the temporary connection back to

B. Excessive Leakage in Reactor Coolant System Equipment

The system can handle a 1-gal/min reactor coolant leak in addition to the expected leakage during normal operation. Operation of the system is almost the same as for normal operation except that the load on the system is increased. A 1-gal/min leak into the reactor coolant drain tank is handled automatically but may increase the load factor of the recycle evaporator. If the 1-gal/min leak enters the waste holdup tank, operation is the same as normal except for the increased load on the evaporator or disposable demineralizer system. Abnormal liquid volumes of reactor coolant resulting from excessive reactor coolant or auxiliary building equipment leakage (1 gal/min) can also be accommodated by the floor drain tank and processed by the LWPS.

C. Excessive Leakage in Auxiliary System Equipment

Leakage of this type could include water from steam side leaks and fan cooler leaks inside the containment which are collected in the containment sump and sent to the waste holdup tank. Other sources could be component cooling water leaks, service water leaks, and secondary side leaks. This water would enter the waste holdup tank and would be processed and discharged as during normal operation.

D. Steam Generator Tube Leaks

During periods of operating with fuel defects coincident with steam generator tube leaks, radioactive liquid is processed via the steam generator blowdown system. This system is described in subsection 10.4.8.

11.2.4.3 Operating Experience

Different processing systems with evaporators have been tested for feed to distillate decontamination factors. A 2-gal/min evaporator was operated with the feed in the pH range of 5.2 to approximately 11.6. Gross beta and gamma activity as well as I-131 activity were measured in the feed and the distillate. The decontamination factors obtained were in the range of 5×10^3 to 5×10^4. The same evaporator was also tested with sodium; the decontamination factors obtained for sodium were, in general, 10^5 or higher. A second evaporator was tested at different pH levels with sodium for gross beta and gamma activity. The test confirmed the results previously obtained. A Westinghouse-designed evaporator similar to the one to be used for Farley Nuclear Plant (FNP) has been shop tested

measuring decontamination factor between bottom and distillate for sodium. The decontamination factors obtained were in the range of 10^5 to 10^6. Hence, a feed to distillate evaporator decontamination of 10^3 used for design is considered conservative.

For operational decontamination factors obtained on demineralizers, see paragraph 11.2.3.1.4.

11.2.5 PERFORMANCE TESTS

Initial performance tests are performed to verify the operability of the components, instrumentation and control equipment, and applicable alarms and control setpoints.

Operability testing of the LWPS is conducted periodically in accordance with the plant technical specifications to determine the reliability of components designed to reduce liquid activity levels. A decontamination factor for the waste evaporator is obtained by measuring the concentrations of I-131, Cs-137, and Co-60 before and after processing to monitor the efficiency of the LWPS. Demineralizer efficiency will be monitored in a similar manner. The waste evaporator output may be recycled in order to reduce contamination to the design levels. The disposable demineralizer system may be used to process the liquid waste to reduce the load on the LWPS.

The radiological analyses conducted to assess the performance of the LWPS and to determine discharge concentrations are discussed in detail in section 11.4.

11.2.6 ESTIMATED RELEASES

11.2.6.1 Nuclear Regulatory Comission Requirements

The following documents have been issued by the Nuclear Regulatory Commission to provide regulations and guidelines for release of radioactive liquids:

A. 10 CFR 20, Standards for Protection Against Radiation.

B. 10 CFR 50, Licensing of Production and Utilization Facilities.

C. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I."

D. Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental amd Routine Reactor Releases for the Purpose of Implementing Appendix I."

During plant operations, radioactive liquid releases will be controlled in accordance with technical specifications. For nuclear power plants, the NRC acceptance criteria for compliance with the dose limits stated in 10 CFR 20.1301 for individual members of the public may be demonstrated by complying with the limits of 10 CFR 50, Appendix I, and 40 CFR 190. Therefore, it is acceptable that the limits associated with the release rate technical specifications are based on ten times the effluent concentration limits given in coulmn 2, Table 2, Apendix B to 10 CFR 20.1001 - 20.2401, since operational history at Farley Nuclear Plant has demonstrated that the calculated maximum individual doses to members of the public are small percentages of the values given in 10 CFR 50, Appendix I, and 40 CFR 190.

11.2.6.2 <u>Westinghouse PWR Experience Releases</u>

The liquid releases are highly dependent upon administrative activities which control the use of water for decontamination, equipment and floor rinsing, and other uses in the controlled areas.

The plants operating at the time of Unit 1 liceasing were reporting liquid discharges as shown in table 11.2-6.

11.2.6.3 <u>Expected Liquid Waste Processing System Releases</u>

The equipment utilized during liquid waste processing is at the discretion of the operator; therefore, the calculated releases do not address all possible treatment processes but only the process which was the basis for the original plant design. Liquid releases from FNP were calculated using the PWR-GALE computer code[2] and parameters listed in table 11.1-7, which are discussed in more detail below. Releases calculated assuming operation with expected levels of fuel cladding defects of 0.12 percent are presented in table 11.2-7. Primary and secondary coolant activity levels are discussed in section 11.1 for the realistic case. In agreement with reference 2, the total releases include an adjustment factor of 0.15 Ci/year, using the same isotopic distribution as the calculated release, to account for anticipated operational occurrences.

The tables list the calculated annual release from each of the process paths discussed below as well as the total annual release. A comparison of annual average effluent concentrations with values stated in column 2, Table II, Appendix B to 10 CFR 20.1 - 20.601 is provided in

table 11.2-8 for operation with expected fuel leakage.

The releases are calculated for one unit, assuming that both units are operating. This is done to reflect the impact of the second unit's operation on the operation of systems and components shared between the two units. To obtain the combined releases of the two units, simply double the values listed in table 11.2-7.

A survey has been performed of liquid discharges from different Westinghouse pressurized water reactor plants, with results presented in Table 11.2-6. The data include radionuclides released on an unidentified basis and are all within the permissible concentration for release of liquid containing an unidentified radionuclide mixture. The data in table 11.2-9 clearly indicate that actual releases are highly dependent upon the actual operation of the plant and can vary significantly from year to year for a given plant as well as from plant to plant.

The LWPS is assumed to operate as described in subsection 11.2.4.

11.2.6.4 Steam Generator Blowdown System

The secondary side activity used in the offsite release analysis is given in table 11.2-7.

The blowdown from the secondary side is normally released to the environment; however, the liquid may be recycled to the main condenser if required.

The estimated activity released per unit to the environment from such discharges is given in table 11.2-7. The system is further described in subsection 10.4.8.

11.2.6.5 Turbine Building Drains

The concentration of isotopes in steam or liquid leaked to the turbine building is considered a factor of 100 lower than secondary side concentration in table 11.2-8 for all isotopes except tritium. Tritium concentration in leakage is assumed to be the same as in the secondary side. The factor of 100 accounts for limited carryover in steam. Steam leakage of 5 gal/min (condensed) and liquid leakage of 12 gal/min is assumed to be discharged through turbine building drains. Discharge rates for each isotope are given in table 11.2-7.

TABLE 11.2-6

RADIOACTIVE LIQUID RELEASES FROM WESTINGHOUSE-DESIGNED PWR PLANTS

Plant	Year	Cladding	Average Percentage Fuel Defects	Total Released (B + Y) Ci	Average Discharge Concentration (μCi/ml)	Fraction of Column 2, Table II, Appendix B to 10 CFR 20.1-20.601 Concentration
Yankee Rowe	1970	Stainless steel	Negligible	0.036	1.5×10^{-10}	1.5×10^{-3}
	1971	Stainless steel	0.001	0.00034	1.25×10^{-12}	1.25×10^{-5}
Connecticut Yankee	1970	Stainless steel	0.01	29.5	4.02×10^{-8}	4.02×10^{-1}
	1971	Stainless steel	0.03	5.85	7.75×10^{-9}	7.75×10^{-2}
San Onofre	1970	Stainless steel	0.007	3.41	6.1×10^{-9}	6.1×10^{-2}
	1971	Stainless steel	0.015	9.21	1.34×10^{-8}	1.34×10^{-1}
R. E. Ginna	1970	Zircaloy	0.4	9.35	1.43×10^{-8}	1.43×10^{-1}
	1971	Zircaloy	0.26	0.96	1.45×10^{-9}	1.45×10^{-2}
H. B. Robinson Unit 2	1970	Zircaloy				
	1971	Zircaloy	<0.001	0.74	1.01×10^{-9}	1.01×10^{-2}
Point Beach Unit 1	1970	Zircaloy				
	1971	Zircaloy	<0.01	0.14	2.48×10^{-10}	2.48×10^{-3}

TABLE 11.2-8 (SHEET 1 OF 2)

COMPARISON OF CALCULATED CONCENTRATIONS IN EFFLUENT WATER DISCHARGE WITH CONCENTRATION VALUES STATED IN COLUMN 2, TABLE II, APPENDIX B TO 10 CFR 20.1 - 20.601 ASSUMING EXPECTED FUEL LEAKAGE

Isotope	Annual Release to Discharge Pipe (μCi)[a]	Concentration in Circulating Water Discharge[b] (μCi/ml)	Maximum Permissible Concentration[c] (μCi/ml)	Fraction of Maximum Permissible Concentration
Cr-51	3.50E+02	1.37E-11	2.00E-03	6.87E-09
Mn-54	1.80E+02	7.07E-12	1.00E-04	7.07E-08
Fe-55	4.20E+02	1.65E-11	8.00E-04	2.06E-08
Fe-59	2.10E+02	8.25E-12	6.00E-05	1.37E-07
Co-58	4.10E+03	1.61E-10	1.00E-04	1.61E-06
Co-60	1.40E+03	5.50E-11	5.00E-05	1.10E-06
Zr-95	1.50E+02	5.89E-12	6.00E-05	9.82E-08
Nb-95	2.10E+02	8.25E-12	1.00E-04	8.25E-08
Np-239	1.00E+01	3.93E-13	1.00E-04	3.93E-09
Br-83	3.00E+01	1.18E-12	3.00E-06	3.93E-07
Rb-86	8.00E+01	3.14E-12	7.00E-05	4.49E-08
Sr-89	8.00E+01	3.14E-12	3.00E-06	1.05E-06
Y-91	1.00E+01	3.93E-13	3.00E-05	1.31E-08
Mo-99	1.30E+03	5.11E-11	2.00E-05	2.55E-06
Tc-99m	1.80E-03	7.07E-11	6.00E-03	1.18E-08
Ru-103	2.00E+01	7.85E-13	8.00E-05	9.82E-09
Ru-106	2.40E+02	9.42E-12	1.00E-05	9.42E-10
Ag-110m	4.00E+01	1.57E-12	3.00E-05	5.24E-08
Te-127m	7.00E+01	2.75E-12	6.00E-05	4.58E-08
Te-127	7.00E+01	2.75E-12	3.00E-04	9.16E-09
Te-129m	2.80E+02	1.10E-11	3.00E-05	3.66E-07
Te-129	1.80E+02	7.07E-12	8.00E-04	8.84E-09
I-130	1.00E+02	3.93E-12	3.00E-06	1.31E-06
Te-131m	2.00E+01	7.85E-13	6.00E-05	1.31E-08
I-131	4.40E+04	1.73E-09	3.00E-07	5.76E-03
Te-132	4.90E+02	1.92E-11	3.00E-05	6.41E-07
I-132	2.20E+03	8.64E-11	8.00E-06	1.08E-05
I-133	2.30E+04	9.03E-10	1.00E-06	9.03E-04
Cs-134	3.70E+04	1.45E-09	9.00E-06	1.61E-04
I-135	5.30E+03	2.08E-10	4.00E-06	5.20E-05
Cs-136	9.60E+03	3.77E-10	9.00E-05	4.19E-06
Cs-137	2.80E+04	1.10E-09	2.00E-05	5.50E-05
Ba-137m	2.40E+04	9.42E-10	-	-
Ba-140	3.00E+01	1.18E-12	3.00E-05	3.93E-08
La-140	3.00E+01	1.18E-12	2.00E-05	5.89E-08

TABLE 11.2-8 (SHEET 2 OF 2)

Isotope	Annual Release to Discharge Pipe (μCi)[a]	Concentration in Circulating Water Discharge[b] (μCi/ml)	Maximum Permissible Concentration[c] (μCi/ml)	Fraction of Maximum Permissible Concentration
Ce-141	1.00E+01	3.93E-13	9.00E-05	4.36E-09
Ce-144	5.30E+02	2.08E-11	1.00E-05	2.08E-06
All Others	5.000E+01	1.96E-12	1.00E-07	1.96E-05
Total	1.90E+05	7.46E-09	-	7.00E-03
H-3	5.50E+08	2.16E-05	3.00E-03	7.20E-03
Total+H-3	5.50E+08	2.16E-05	-	1.42E-02

a. Based on the estimated isotopic liquid effluents in table 11.2-7.

b. Based on the 16,000 gal/min/unit discharge in subsection.

c. Column 2, Table II, Appendix B to 10 CFR 20.1 - 20.601.

11.3 GASEOUS WASTE SYSTEMS

11.3.1 DESIGN OBJECTIVES

The gaseous waste processing system (GWPS) was designed to remove fission product gases from the reactor coolant and have the capacity to contain these throughout the 40-year plant life. This was based on continuous operation with reactor coolant system activities associated with operation, with cladding defects in the fuel rods generating 1 percent of the rated core thermal power. The system was also designed to collect and store expected fission gases from the boron recycle evaporator and reactor coolant drain tank throughout the plant life.

Although the system has the design capacity to contain fission product gases for the life of the plant, operating experience has demonstrated that the waste gas decay tanks must be released periodically due to nitrogen buildup. These releases are necessary following degassification of the reactor coolant system (RCS) prior to each RCS maintenance or refueling outage and again following deoxygenation at the end of each RCS maintenance or refueling outage. These releases are monitored and quantified in accordance with the radiological effluent technical specifications. At a preset level in the plant vent stack, waste gas decay tank releases are automatically terminated, as described in paragraph 11.4.2.2.5.

Gaseous activity released due to equipment leakage during normal operation of the plant is mixed with ventilation exhaust and is further diluted due to atmospheric dilutions. Table 11.3-9 gives estimated activity discharges from the plant vent stack.

The plant design considers potential personnel exposure and ensures that quantities of gaseous radioactive releases to the environment are as low as reasonably achievable. Under normal plant operation, the total activity from gaseous radionuclides leaving the GWPS does not exceed a small fraction of the discharge limits as defined in column 1, Table II, Appendix B to 10 CFR 20.1 - 20.601.

Although plant operating procedures, equipment inspection, and preventive maintenance are performed during plant operations to minimize equipment malfunction, overall radioactive release limits have been established as a basis for controlling plant discharges during operation with the occurrence of a

combination of equipment faults of moderate frequency. A combination of equipment faults which could occur with moderate frequency include operation with fuel defects in combination with such occurrences as:

A. Steam generator tube leaks.

B. Malfunction in liquid waste processing system.

C. Malfunction of GWPS.

D. Excessive leakage in reactor coolant system equipment.

E. Excessive leakage in auxiliary system equipment.

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The radioactive releases from the plant resulting from equipment faults of moderate frequency are within column 1, Table II, Appendix B to 10 CFR 20.1 - 20.601 limits on the short-term basis and do not exceed four to eight times the limits stated previously for normal operation.

11.3.2 SYSTEM DESCRIPTION

The GWPS consists mainly of a closed loop comprised of two waste gas compressors, two catalytic hydrogen recombiners, and gas decay tanks to accumulate the fission product gases.

The major input to the GWPS during normal operation is taken from the gas space in the volume control tank. The volume control tank gas space is purged at a rate of 0.7 sf^3/min. Table 11.3-2 lists the rate of activity input to the GWPS during normal operation. There are no liquid seals in the system. The system is designed to preclude explosions by keeping the concentration of hydrogen and oxygen below the explosive limits.

Process flow diagrams and piping and instrumentation diagrams are shown in figures 11.3-1 through 11.3-3. All lines in the gaseous waste system, including field run, are considered potential carriers of significant radioactivity. Only non-Category I pipe of Class B31.1 of the American National Standards Institute (size 2 in. and under) will be field run. This piping is shown on the piping and instrumentation drawings and is designated as Safety Class NNS. Table 11.3-3 gives process parameters for key locations in the system.

11.3.3 SYSTEM DESIGN

11.3.3.1 Component Design

Gaseous waste processing equipment parameters are given in table 11.3-4. For further information on safety and seismic classification see chapter 3. Paragraph 3.9.2.7 gives the general design criteria for field run piping.

11.3.3.1.1 Waste Gas Compressors

Two waste gas compressor packages are provided. One unit is normally used, with the other on a standby basis.

The units are centrifugal displacement machines which are skid-mounted in a self-contained package. Construction is primarily of carbon steel. Mechanical seals are provided to minimize the

outleakage of seal water. The compressor has been used in Westinghouse pressurized water reactor (PWR) plants with excellent experience.

11.3.3.1.2 Recombiners

Two catalytic hydrogen recombiners are provided. As shown on figure 11.3-2, sheets 3 and 4, one of the two recombiners is normally used to remove hydrogen from the hydrogen-nitrogen fission gas mixtures by oxidation to water vapor, which is removed by condensation. The other recombiner is available on a standby basis. Both units are self-contained and designed for continuous operation.

11.3.3.1.3 Gas Decay Tanks

Gas decay tanks are provided as described in table 11.3-4. The tanks used during power operation are of vertical cylindrical type, and the shutdown tanks are horizontal cylindrical. All the gas decay tanks are constructed of carbon steel.

11.3.3.1.4 Valves

Each valve in the system is designed to meet the temperature, pressure, and code requirements for the specific application in which it is used. Special consideration is given to leaktightness. The recombiner circuits use manual valves provided with a diaphragm to prevent stem leakage and control valves with leakoffs returned to the gas system. Other parts of the gas system use Saunders valves and control valves with bellows seal.

11.3.3.2 <u>Instrumentation Design</u>

The main system instrumentation is described in table 11.3-5 and shown in the flow and piping diagrams of figures 11.3-2 and 11.3-3.

The instrumentation readout is located mainly on the waste processing system panel in the auxiliary building. Some instruments are read where the equipment is located.

All alarms are shown separately on the waste processing system panel and further relayed to one common system annunciator on the main control board of the plant.

The catalytic recombiner system is designed for automatic operation with a minimum of operator attention. Each package

includes four online gas analyzers which are the primary means of recombiner control. A multipoint temperature recorder monitors temperatures at several locations in the recombiner packages.

Process gas flowrate is measured by an orifice located upstream of the recombiner preheater. Local pressure gauges indicate pressures at the recombiner inlet and the oxygen supply pressure.

11.3.4 OPERATING PROCEDURES

11.3.4.1 Operation with Continuous VCT Purge

Prior to the system being put into operation, the GWPS is flushed free of air and filled with nitrogen. During normal power operation, nitrogen gas is continuously circulated around the loop by one of the two compressors. Fresh hydrogen gas is charged to the volume control tank, where it is mixed with fission gases which are stripped from the reactor coolant into the tank gas space. The contaminated hydrogen gas is then vented from the tank into the circulating nitrogen stream to transport the fission gases into the GWPS. The resulting mixture of nitrogen-hydrogen fission gas is pumped by the compressor to the recombiner where enough oxygen is added to reduce the hydrogen to a low residual level by oxidation to water vapor on a catalytic surface. After the water vapor is removed, the resulting gas stream is circulated to the gas decay tanks and back to the compressor suction to complete the loop circuit.

Each gas decay tank is capable of being isolated, and the number of tanks valved into operation at any time is restricted to diminish the amount of radioactive gases which could be released as a consequence of any single failure, such as the rupture of any single tank or connected piping. By alternating use of these tanks, the accumulated activity is contained in approximately equal parts.

When the hydrogen contained in the reactor coolant must be removed in preparation for a cold shutdown, the gas decay tanks are valved out of service and one of the two shutdown tanks is placed in service between the compressor discharge and the recombiner suction. The first degassing operation will require that the shutdown tank be pressurized with nitrogen before degassing begins. In addition, the flow of hydrogen to the volume control tank is stopped, the bypass on the volume control tank vent line is opened, and purge flow from the shutdown tank to the volume control tank is initiated, thus establishing a recirculation path between the GWPS and the volume control tank. The flow of gas through the volume control tank is controlled at approximately 5 sf^3/min.

Initially, the flow will be predominantly hydrogen, but as degassing progresses the gas will become primarily nitrogen. Because of the difference in density between the gases, the throttle valve in the bypass line may require adjustment during the degassing operation to maintain a constant flowrate.

The gaseous waste processing system at Farley Nuclear Plant and systems of similar design have been operating for several years with excellent experience, as far as components and overall system performance are concerned.

Systems constructed from carbon steel have been in service for many years, and no failure due to corrosion damage has been reported.

Components of identical design to those used for the FNP are in use on several Westinghouse-designed GWPS. The performance and operating history of the compressors have been excellent.

11.3.4.2 Alternate System Operation

Although the GWPS was designed for continuous purge of the VCT and 40-year holdup of fission gases, operating experience at FNP has shown that the GWPS can be operated without a continuous purge while maintaining personnel exposure within limits and maintaining releases within concentration and offsite dose limits. Many other operating PWRs are not designed with continuous purge capability and have operated for many years with gaseous releases from the GWPS well within MPC and offsite dose limits.

Fission gas production is directly related to fuel integrity. Fuel defects have been minimal at plants with Westinghouse fuel and, therefore, fission gas RCS concentrations are normally well below design limits (1-percent fuel defects).

The purpose of the VCT purge is to strip fission gases from the reactor coolant to reduce the exposure to personnel from fission gases which escape with reactor coolant leakage. However, the primary contributor to exposure from leakage is Co-60, which is dissolved in the liquid. The only fission gases with significant half-lives are Xe-133 and Kr-85. Without a continuous VCT purge, Xe-133 will reach an equilibrium value in the RCS in about 30 days. Without fuel defects, the equilibrium concentration will be orders of magnitude less than design values and will not contribute to doses as a result of reactor coolant leakage. Kr-85 will accumulate in the RCS because of its long half-life. However, being a beta-emitter, it is not expected to contribute significantly to personnel exposure. Therefore, the benefit of the continuous purge is limited when fission gas concentrations are already low.

Without the VCT purge, the doses resulting from a steam generator tube rupture accident will increase by approximately 50 percent due to the increase in RCS fission gas activity. This increase would raise the calculated dose from 0.14 rem (with purge) to 0.21 rem (without purge), which is well below the Standard Review Plan limit of 2.5 rem. Regarding the gas decay tank rupture accident, the offsite dose limit of 0.5 rem will be assured by maintaining the activity in each gas decay tank within the Technical Specification limit of 70,500 Ci. Without the VCT purge, gas decay tank activity accumulation will be drastically reduced during normal operation but will increase during plant shutdown, due to RCS fission gas activity buildup. In spite of this buildup, the gas decay tank curie limit can still be maintained by the normal practice of spreading the activity among two or more tanks during shutdown degassing.

Therefore, the VCT purge is not required during normal power operation and can be aligned in or out of service at the operator's discretion. This philosophy allows for a simpler and more reliable GWPS operation as described below.

11.3.4.2.1 Plant Startup

This operation remains the same as with a VCT purge. The nitrogen gas space in the VCT is replaced with hydrogen by burping the gas space to the GWPS until the required RCS dissolved hydrogen concentration is achieved.

11.3.4.2.2 Normal Power Operation

The VCT purge is normally isolated. Without the major hydrogen input, the need for the recombiner operation is reduced. One compressor and one gas decay tank may be placed in service as necessary to accommodate the very small flow and volumetric inputs from the reactor coolant drain tank vent, the recycle evaporator vent, and the recycle holdup tank diaphragm vent. The VCT purge can be initiated as required to reduce RCS radiogas inventory at the discretion of the plant operators. At a nominal flowrate of 0.7 sf^3/min, a gas decay tank will be filled in 4 days. Therefore, a recombiner must eventually be placed into service, depending upon the available capacity in the remaining gas decay tanks.

11.3.4.2.3 Plant Shutdown

If shutdown is required such that the RCS is opened to the containment atmosphere (e.g., refueling or maintenance), the RCS fission gas and hydrogen concentration must be reduced to required

levels. The VCT gas space is burped to the GWPS and fresh nitrogen is aligned. With the reactor shut down, no additional fission gases are produced. Therefore, the VCT burping needs to remove only residual fission gases, if their concentration is above the required shutdown limit. In this mode, the GWPS compressor and gas decay tank must be available to receive and collect the VCT burp volume. The gases can be stored for decay as required, then released as during normal plant operation.

11.3.5 PERFORMANCE TESTS

Initial performance tests are performed to verify the operability of the components, instrumentation, and control equipment.

11.3.6 ESTIMATED RELEASES

11.3.6.1 Nuclear Regulatory Commission Requirements

The following documents have been issued by the Nuclear Regulatory Commission to provide regulations and guidelines for radioactive releases:

A. 10 CFR 20, Standards for Protection Against Radiation.

B. 10 CFR 50, Licensing of Production and Utilization Facilities.

C. Regulatory Guide 1.42, Interim Licensing Policy on As Low As Practicable for Gaseous Radioiodine Releases from Light Water Cooled Nuclear Power Reactors.

Regulatory Guide 1.42, which was in effect during plant design, was withdrawn by the NRC in 1976 and replaced by guidance presented in the following regulatory guides:

A. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I."

B. Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors."

C. Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Reactors."

D. NUREG-0133, "Preparation of Radiological Effluent Technical Specifications Nuclear Power Plants."

During plant operations, radioactive gaseous releases will be controlled in accordance with technical specifications. For nuclear power plants, the NRC acceptance criteria for compliance with the dose limits stated in 10 CFR 20.1301 for individual members of the public may be demonstrated by complying with the limits of 10 CFR 50, Appendix I, and 40 CFR 190. Therefore, the use of dose rate values based on the guidance contained in NUREG-0133[4] is acceptable for use as a technical specification limit for gaseous effluent release rates since operational history at Farley Nuclear Plant has demonstrated that, with these dose rate limits in effect, the calculated maximum individual doses to members of the public are small percentages of the limits of 10 CFR 50, Appendix I and 40 CFR 190.

11.3.6.2 Radioactive Noble Gas Releases From FNP

A summary of gaseous discharges from FNP for years 1977 to 1983 is presented in table 11.3-6.

11.3.6.3 Expected GWPS Releases

The GWPS is designed to remove fission product gases from the volume control tank and recycle evaporator and was designed with the capacity to contain them through the lifetime of the plant. Since the VCT purge to the GWPS reduces fission product gas concentrations in the reactor coolant during unit operation, it reduces the escape of radioactive gases arising from any possible reactor coolant leakage. Design is based on continuous operation, with reactor coolant system activities associated with operation, with cladding defects in fuel rods generating 1 percent of the rated core thermal power. Table 11.3-7 shows the maximum fission product inventory in the GWPS over the 40-year plant life based on a 1-unit plant.

Figure 11.3-4 shows that, for a given power rating, the quantity of fission gas activity accumulated in the gas system after 40 continuous years of operation is only twice the activity accumulated after 30 days of operation. This is because most of the accumulated activity arises from short-lived isotopes reaching equilibrium in 1 month or less.

The difference between the 30-day and 40-year accumulations is essentially all Kr-85. This accumulation of Kr-85 is not a hazard to the plant operator because:

A. Radiation background levels in the plant are not noticeably affected by the accumulation of Kr-85, which is a beta emitter for which the tanks themselves provide adequate shielding.

B. The system activity inventory is distributed in several tanks so that the maximum permissible inventory in any single tank is actually less than that of earlier GWPS designs.

C. Since this system permits fission gas removal from the reactor coolant during normal operation, it is expected to reduce plant activity levels caused by a leakage of reactor coolant.

The capability to release a waste gas decay tank directly to the plant vent stack was provided as part of the original design of each unit. Automatic shutoff for such release occurs at a preset vent stack radiogas monitor setpoint as described in paragraph 11.4.2.2.5.

To further ensure design basis releases in accordance with the "as low as reasonably achievable" philosophy, the plant technical specifications establish limits for the releases. The quantity of radioactivity contained in each waste gas storage tank is limited to 70,500 Ci.

11.3.6.4 Releases from Ventilation Systems

A detailed analysis of one unit of the plant has been made to ascertain those items that could possibly contribute to airborne radioactive releases. Results of the analysis are presented in section 11.3.9.

During normal plant operations, airborne noble gases and/or iodines can originate from reactor coolant leakage, equipment drains, venting and sampling, secondary side leakage, condenser air ejector and gland seal condenser exhausts, GWPS leakage, refueling operations, and evaporations from the spent fuel pool.

11.3.6.5 Estimated Total Releases

The potential release from the sources discussed in subsections

11.3.6.3 and 11.3.6.4 has been evaluated. Radioactive effluent releases from the plant for normal operation are given in table 11.3-9. These release rates were calculated using the PWR-GALE code[1] and plant operating parameters referenced in paragraph 11.1.1.2 and table 11.1-7. The releases are calculated for one unit; to obtain the combined releases for the two units, double the values listed in table 11.3-9.

The dose calculations, based on the estimated total plant releases, show that the releases are in accordance with the design objectives in subsection 11.3.1 and meet the regulations and guidelines as outlined in subsection 11.3.6.1. Further, the total plant releases, noted in table 11.3-10, are within the plant technical specifications, which are developed to be consistent with the "as low as reasonably achievable" criterion and the concentration limits specified in column 1, Table II, Appendix B to 10 CFR 20.1 - 20.601.

11.3.7 RELEASE POINTS

The GWPS is designed to contain all fission product gases generated during the plant lifetime. Any gases that do leak from the system are swept up by the radwaste area ventilation system, which discharges the gas to the plant vent stack.

The vent stack is the principal release point of gaseous waste to the environment. However, in the event of primary to secondary leakage, the power-operated atmospheric relief valve vents and the turbine building vent could become release points.

The vent stack is shown as part of the ventilation system in figure 9.4-4. The physical location of the stack, shown in the plant general arrangement drawings of figures 1.2-2 through 1.2-9, exhausts at a height of 145 ft 9 in. above grade.

The main exhaust line from the radwaste area ventilation system to the vent stack is a 6-ft diameter duct which is flanged into the vent stack.

The vent stack parameters are as follows:

A. Base elevation - 155 ft (same as ground elevation).

B. Orifice elevation - not applicable.

C. Orifice inside diameter - not applicable.

D. Effluent velocity at flange of main exhaust line - 2650 ft/min.

E. Heat input to stack - 39,078 Btu/h.

11.3.8 DILUTION FACTORS

Gaseous and particulate radioactive effluents may be normally released from the plant vent and turbine building vent as discussed in 11.3.7. Subsection 2.3.5 outlines the methodology and information used to determine the long-term atmospheric dilution (X/Q) and deposition (D/Q) for these release points. Effluent sources and associated vents are listed in table 2.3-15. Vent design information and input assumptions utilized for the long-term diffusion estimates are given in tables 2.3-16 and 2.3-17.

11.3.9 ESTIMATED DOSES[1]

Dose models, usage factors, and other parameters used to estimate the maximum doses to individuals from discharges of gaseous and particulate radioactive effluents are those described in Regulatory Guide 1.109.[3] These models were applied to the FNP using the source terms and atmospheric dilution factors discussed in subsections 11.3.6.5 and 11.3.8.

Pathways included are plume exposure, ground shine, inhalation, and food ingestion (cow or goat milk, vegetation, and meat). Beta and gamma radiation doses to air were determined for the offsite location with the highest potential dose.

Receptor locations were selected by inspection of dispersion parameter values at locations tabulated in Tables 2.3-21 and 2.3-22. Doses were evaluated for a number of locations at which real receptors exist. Results were reviewed to identify the maximally exposed individual. This process is considered to yield doses which are unlikely to be substantially underestimated.

Detailed results are presented in Table 11.3-11 for the maximally exposed individual. This table provides a breakdown by organ and pathway, and doses include the summation from both assumed plant discharge points given in Table 2.3-16. Furthermore, the total dose to each organ is given in Table 11.3-11 along with the Appendix I design objective doses for comparison. It is clear that the estimated doses follow the design objective dose in each case.

The estimates were calculated to confirm that the Farley Nuclear Plant conforms with the requirements of 10 CFR 50, Appendix I. Actual plant releases during normal operation are governed by the Farley Nuclear Plant Technical Specifications.

REFERENCES

1. Alabama Power Company letter to the Nuclear Regulatory Commission, "Dose Calculations to Conform with Appendix I Requirements," USNRC Docket Nos. 50-348, 50-364, June 3, 1976.

2. U.S. Nuclear Regulatory Commission, "Calculation of Releases from Pressurized Water Reactors," PWR-GALE Computer Code, NUREG-0017, April 1976.

3. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I.

4. U.S. Nuclear Regulatory Commission, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," NUREG-0133, October 1978.

TABLE 11.3-10

COMPARISON OF CALCULATED MAXIMUM OFFSITE AIRBORNE CONCENTRATION WITH CONCENTRATION VALUES STATED IN COLUMN 1, TABLE II, APPENDIX B TO 10 CFR 20.1 - 20.601 ASSUMING EXPECTED FUEL LEAKAGE

Isotope	Total Annual Release from One Unit[a] (Ci/year)	Maximum Site Boundary Concentration[b] (μCi/ml)	Maximum Permissible Concentration (MPC)[c] (μCi/ml)	Fraction of MPC
Kr-83m	0.		3.0E-06	
Kr-85m	3.0E+00	1.19E-12	1.0E-07	1.19E-05
Kr-85	2.01E+02	7.96E-11	3.0E-07	2.65E-04
Kr-87	1.00E+00	3.96E-13	2.0E-08	1.98E-05
Kr-88	6.00E+00	2.38E-12	2.0E-08	1.19E-04
Kr-89	0.			
Xe-131m	4.0E+00	1.58E-12	4.0E-07	3.96E-06
Xe-133m	2.0E+00	7.92E-13	3.0E-07	2.64E-06
Xe-133	2.4E+02	9.50E-11	3.0E-07	3.17E-04
Xe-135m	0.			
Xe-135	9.0E+00	3.56E-12	1.0E-07	3.56E-05
Xe-137	0.			
Xe-138	1.00E+00	3.96E-13	3.0E-06	1.32E-07
I-131	2.10E-01	8.32E-14	1.0E-10	8.32E-04
I-133	2.30E-01	9.11E-14	4.0E-10	2.28E-04
H-3	5.50E+02	2.18E-10	2.0E-07	1.09E-03
MN-54	2.30E-04	9.11E-17	1.0E-08	9.11E-09
FE-55	7.50E-05	2.97E-17	3.0E-08	9.90E-10
CO-58	7.50E-04	2.97E-16	3.0E-08	9.9E-09
CO-60	3.40E-04	1.35E-16	1.0E-08	1.35E-08
SR-89	1.60E-05	6.34E-18	3.0E-10	2.11E-08
SR-90	3.00E-06	1.19E-18	3.0E-11	3.96E-08
CS-134	2.30E-04	9.11E-17	1.0E-09	9.11E-08
CS-137	3.80E-04	1.50E-16	2.0E-09	7.52E-08

a. Total Ci/year from table 11.3-9.

b. Based on the sum of contributions from the plant vent and turbine building vent using the conservative ground release dilution factor (X/Q) of 1.0×10^{-5} s/m^3 at the nearest site boundary (1280 m WSW) in 16 compass directions given in table 2.3-22.

c. From column 1, Table II, Appendix B to 10 CFR 20.1 - 20.601.

d. 0.0 indicates release < 1.0 Ci/yr for noble gas.

11.4 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEMS

11.4.1 DESIGN OBJECTIVES

The radiation monitoring system consists of the following subsystems:

A. The process and effluent radiological monitoring system (PERMS), which includes both continuous process and periodic sampling systems.

B. The area radiation monitoring system, which monitors radiation fields in various areas within the plant. This system is further described in subsection 12.1.4.

C. The airborne radioactivity monitoring system, which is described in subsection 12.2.4.

The PERMS is designed to enable plant operation to be in compliance with Table II, Appendix B to 10 CFR 20.1 - 20.601, the low as reasonably achievable criterion of 10 CFR 50, and the technical specification limits of the Farley Nuclear Plant, in addition to being in accordance with Nuclear Regulatory Commission (NRC) 1971 General Design Criterion 64.

The radiation monitoring system does not meet the guidelines of NRC Regulatory Guide 1.21 in its entirety. Specifically, continuous isotopic analysis and measurement of radionuclides to exceedingly low sensitivities and monitoring of all potential paths for radioactive release are not within the current state of the art and are therefore not addressed in the design of this system.

The general design objectives for the PERMS are to:

A. Warn of any radiation health hazard to operating personnel.

B. Warn of leakage from process systems containing radioactivity.

C. Monitor activity released in effluents and provide alarm and termination of the release when radiation levels exceed setpoint limits. Where the termination of release is not feasible, the monitors provide a continuous indication of the magnitude of activity released.

The accomplishment of these general objectives by the PERMS will provide assurance that exposures to individuals in

restricted and unrestricted areas are as low as reasonably achievable during all modes of plant operation and during accidents.

Except for the containment purge exhaust line monitor and the spent fuel pool exhaust flow gas monitor, the PERMS is not designed to Seismic Category I or to Institute of Electrical and Electronics Engineers (IEEE) accident grade standards and may not be available under accident conditions. No credit is taken for these monitors in accident evaluations in chapter 15.

The design objectives for PERMS periodic sampling include the following:

A. Enable manual collection of representative samples of planned gaseous and liquid effluents prior to discharge to unrestricted areas during normal reactor operation and during anticipated operational occurrences, in order to allow laboratory measuring and recording of the quantity of each of the principal radionuclides present in these discharges as required by 10 CFR 50.36a.

B. Enable manual collection of representative samples of gaseous and liquid process streams during normal reactor operation and during anticipated operational occurrences, in order to allow laboratory measuring and recording of the quantity of each of the principal radionuclides present to verify and supplement the continuous process system monitors.

11.4.2 PROCESS AND EFFLUENT RADIATION MONITORING

11.4.2.1 General Description

The components of the PERMS are designed for the following environmental conditions:

A. Temperature - An ambient temperature range of 40°F to 120°F.

B. Humidity - Relative humidity of 0 to 95 percent.

C. Pressure - Components in the auxiliary building and control room are designed for normal atmospheric pressure. Area monitoring system components inside the containment are designed to withstand containment test pressure.

S. Description

Redundant offline gas monitors are provided to continuously measure gaseous radioactivity levels of the containment purge exhaust flow during containment purge operations.

A motor-driven positive displacement pump is used to draw a continuous sample from the containment purge exhaust flow line and direct the sample through a particulate removal prefilter. The sample is then routed to a 4π shielded sample chamber of sufficient volume to accomplish the design purpose of the system. The sample effluent is monitored for radioactivity by a thin beta crystal scintillation detector assembly placed within the sample chamber in contact with the effluent, prior to being returned to the exhaust line to the stack.

The beta crystal is optically coupled to a photomultiplier tube, which responds to the light scintillations emitted from the crystal as a result of incident radiation giving up its energy within the crystal. The photocathode of the photomultiplier tube absorbs energy in the form of a pulse (current) which is fed directly into a preamplifier at the base of the detector assembly and in turn provides a signal to the control room readout instrumentation.

The control room readout instrument consists of a five-decade log level amplifier and associated circuitry as required to convert total pulse rate to a logarithmic analog signal output for suitable indication, recording, and alarm trip circuits.

The setpoint is based upon a release in which Kr-85 and Xe-133 are the predominant radionuclides, site boundary concentration values as presented in column 1, Table 2, Appendix B to 10 CFR 20.1001 - 20.2401, and the highest annual average mixed-mode X/Q value at the site boundary. Isolation of releases from the containment at or below concentration levels which correspond to these site boundary concentrations ensures that dose rates at the site boundary will not exceed limits established by technical specifications. The programmatic controls contained in the technical specifications represent the NRC acceptance criteria for radioactive gaseous effluent release rates.

Power for channels R-24A and B is provided from vital motor control centers A and B, respectively.

A "loss of power" and "channel failure" are monitored for each detector providing annunciation in the control room.

A channel performance test is available to the operator. An electronic pulse signal is used to verify the performance of the readout instrumentation.

A radioactive check source, controlled from the readout instrument in the control room, can be actuated to check system integrity. This check source is used as a convenient operational and gross calibration check of the detection system.

The check source is of the same or similar energy and range of the isotopes to be monitored.

A three-way, solenoid-operated valve at the sample chamber inlet is operable from the control room. It is provided to permit air purging of the sample chamber to facilitate background activity checks.

Visual/audible indication of channel failure and/or high radiation is provided in the control room.

11.4.2.2.18 Spent Fuel Pool Exhaust Flow Gas Monitors - Channel R-25A and B

A. Introduction

The spent fuel pool exhaust flow gas monitors (figure 11.4-10) act to limit the radioactive releases associated with a fuel handling accident in the spent fuel pool. Design requirements were derived by analyses of the radioactive releases associated with the fuel handling accident discussed in chapter 15.

B. Identification of Safety Criteria

The documents listed below were considered in the design of the spent fuel pool gas monitors.

1. General Design Criteria for Nuclear Power Plants, Appendix A, Title 10 CFR 50, July 7, 1971.

2. Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions."

3. The Institute of Electrical and Electronics Engineers, "Standard Criteria for Protection Systems for Nuclear Power Generating Stations," IEEE 279-1971.

P. Actuated Devices

The actuated devices and their characteristics are shown in paragraph 9.4.2.2.2.

Q. Supporting Systems

Supporting systems for the gas monitors are the interruptible ac instrument power supply, the dc control power supplies, and the instrument air system. The isolating function is fail safe with respect to all of these supporting systems.

R. Nonsafety systems

The nonsafety-related systems associated with the radiation monitors are the instrument air system, the station annunciator, and the station computer.

S. Description

Redundant offline gas monitors are provided to continuously measure gaseous radioactivity releases to the environs by the ventilation fans exhausting the spent fuel pool area of the auxiliary building. The offline monitors incorporate a positive displacement pump that draws a continuous air sample from the spent fuel pool ventilation exhaust duct. This sample is then directed through a particulate removal prefilter. The sample is then routed to a 4π lead-shielded sample chamber of sufficient volume to accomplish the design purpose of the system. The sample effluent is then monitored for radioactivity by a thin beta scintillation detector assembly placed within the sample chamber in contact with the effluent, prior to being returned to the common vent duct exhausting all spent fuel pool spaces.

The beta crystal is optically coupled to a photomultiplier tube, which responds to the light scintillations emitted from the crystal as a result of incident radiation giving up its energy within the crystal. The photocathode of the photomultiplier tube absorbs the energy photons and emits the absorbed energy in the form of a pulse (current) fed directly into a preamplifier at the base of the detector assembly and, in turn, provides a signal to the control room readout instrumentation.

The control room readout instrument consists of a log level amplifier and associated circuitry as required

to convert total pulse rate to a logarithmic analog signal output for suitable indication, recording, and alarm trip circuits. The ratemeters have a five-decade range (10^1 to 10^6 cpm).

The setpoint is based upon a release in which Kr-85 and Xe-133 are the predominant radionuclides, site boundary concentration values as presented in column 1, Table 2, Appendix B to 10 CFR 20.1001 - 20.2401, and the highest annual average mixed-mode X/Q value at the site boundary. Isolation of releases from the fuel handling area at or below concentration levels which correspond to these site boundary concentrations ensures that dose rates at the site boundary will not exceed limits established by technical specifications. The programmatic controls contained in the technical specifications represent the NRC acceptance criteria for radioactive gaseous effluent release rates.

Power for channel R-25A and B is provided from vital motor control centers A and B, respectively.

11.4.2.2.19 Waste Evaporator and Recycle Evaporator Condensate (Electric Steam Generators A and B) - Channel R-26A and B -

This channel monitors the condensate returned from the waste evaporator and recycle evaporator feed preheaters to their respective auxiliary steam generators in order to observe contaminated leakage in the electric steam generator condensate from the feed preheater.

11.4.2.2.20 Noble Gas Effluent Monitors

A. Vent Stack Monitor - Channel R-29B

The post accident vent stack monitor for each FNP unit has three noble gas channels with a combined range of 10^{-7} to 10^5 μCi/cm^3 using multiple detectors. The monitor draws a sample from the vent stack to a monitor unit located in the mechanical equipment room at el 175 ft of the auxiliary building. The readout for this unit is located in the main control room. An auxiliary readout is located in the low activity counting laboratory.

The noble gas measurement is performed by several detectors viewing a sample volume. The low and medium range detectors view the same sample volume located in a common sampler assembly. The high range detector views the sample volume located in a separate sampler assembly.

The monitors have been environmentally qualified by the vendor for the environment in which they are located.

B. Main Condenser Air Removal Monitor - Channel R-15B and C

The main condenser air removal exhaust systems for Units 1 and 2 are monitored using the existing monitor (described in paragraph 11.4.2.2) on the steam jet air ejector exhaust for the normal range of radioactivity. The accident range of radioactivity will be monitored for Units 1 and 2 by intermediate and high range detectors with overlapping ranges and located at the common vent duct for the turbine building. The accident monitor consists of two Eberline detectors and readouts. The intermediate range detector is a DA1-1CS with an ED1-1 readout module with a range of indication of 0.1 to 100 mR/h. The high range detector is a model DA1-4CS with an EC1-20 readout module with a range of 10 mR/h to 1000 R/h. The relationship between mR/h and $\mu Ci/cm^3$ will be established for the noble gas isotopes present during an accident. The range of the accident monitors in $\mu Ci/cm^3$ is from 10^{-5} to 10^3, with the normal range monitor measuring concentrations down to 10^{-6} $\mu Ci/cm^3$. This is the required range for the case where the steam jet air ejector exhaust is combined with turbine building ventilation exhaust. The readout modules are located in the control room and provide continuous indication. The accident detectors are shielded from background radiation with 6 in. of lead. Calibration is by use of an external calibration source and is performed upon installation and at intervals not exceeding each refueling outage.

C. Steam Generator Atmospheric Relief and Safety Valve Monitors - Channel R-60A, B, C, and D

The discharge from steam generator safety relief valves and atmospheric dump valves for Units 1 and 2 will be monitored by measuring the radiation levels from these steam plumes. There are four Eberline model DA1-4CS detectors per unit mounted on the main steam roof with a range of 10 mR/h to 1000 R/h. The relationship between mR/h and $\mu Ci/cm^3$ has been established for the noble gas isotopes present during an accident. The range of the monitors in $\mu Ci/cm^3$ will more than cover the required range from 10^{-1} to 10^3 for cases with just the power-operated relief valves open to cases with the power-operated relief valves and all safety valves open. Each detector is connected to an Eberline EC1-20 readout module in the

control room, providing continuous indication. Since the safety relief valve and atmospheric dump valve discharges are grouped together for each of the three steam generators, one detector will be used to monitor the combined effluent steam plume from each steam generator. The fourth detector is used to monitor the plume from the steam-driven auxiliary feedwater pump turbine exhaust. Each detector is collimated and background shielded with 7.5 in. of lead. Calibration is by use of an external calibration source and is performed upon installation and at intervals not exceeding each refueling outage.

D. Design for Noble Gas Effluent Monitors

The noble gas effluent monitors are powered from a vital instrument bus. Procedures have been developed for use, calibration of the system, and dissemination of release rate information. The original APC position was to monitor the main condenser air removal exhaust and the discharge from the steam generator safety relief valves and atmospheric relief valves with a portable gamma survey instrument. However, APC finalized the above position based on NRC questions during the latter part of 1980 and purchased the best available monitors upon finalization of this position. To ensure accurate reading of each of these monitors, a complex shielding design is required to discriminate actual readings from background, including containment shine.

11.4.2.3 Alarm Setpoint Basis

The alarm setpoints for the process radiation monitors are based on the following:

A. The methodology used to calculate setpoints for RE-13, RE-14, RE-15A, RE-18, RE-22, RE-23B, and RE-24 is specified in the Offsite Dose Calculation Manual (ODCM). The RE-23A setpoint methodology, while not specified in the ODCM, will be the same as that for RE-23B.

B. The RE-15B setpoint will be based such that the monitor will alarm at one half decade before RE-15A goes offscale. The RE-15C setpoint will be based such that the monitor will alarm at one half decade before RE-15B goes offscale.

C. Detector response which will provide warning to the operator of leakage of activity into a normally low activity system. This includes channels RE-11, RE-12, RE-16, RE-17A and B, RE-19, RE-20A and B, RE-26A and B, and RE-60A, B, C, and D.

D. Detector response which will provide to the operator warning of plant vent stack effluent in accordance with Regulatory Guide 1.97. This includes RE-29B. In addition, the RE-29B low-range noble gas channel and the RE-29B iodine channel setpoints are based on the Notification of Unusual Event (NOUE) emergency classification criteria, annual average meteorology, on ODCM-based dose conversion factors, and maximum plant vent stack flowrate.

Typical alarm setpoints for the process radiation monitors are listed in table 11.4-3.

11.4.2.4 Design Evaluation

The liquid and gaseous waste discharge monitors are provided to maintain surveillance over the release of radioactivity with the following features:

A. A check source is provided to permit the operator to check the monitor before discharge by operating a switch on the radiation monitor system panel.

B. If the reading falls off scale at any time, an indicator visible to the operator in the control room will alarm.

C. If the power supply to the channel fails, a high radiation alarm will annunciate. Control valves associated with the channel will also close.

An evaluation of instrumentation function relative to monitoring and for controlling release of radioactivity from various plant systems is discussed below.

A. Fuel Handling

For activity releases inside the containment and in the fuel handling area, the offline gas monitors (channel 24A and B or R-25A and B) would function. Each of these monitors initiates alarms in the control room and initiates ventilation isolation when the radiation level exceeds a preset level. Activity releases within the auxiliary building ventilation exhaust flow would cause the plant vent and area monitors to alarm on an increase in radiation level.

B. Liquid and Gas Wastes

For ruptures or leaks in the waste processing system, plant area monitors and the vent stack monitor will alarm on an increase in radiation level over a preset level. For cases where leaks are involved, the operator may control activity release by system isolation. For more severe postulated accident cases, such as rupture of waste tanks, activity release is not controlled. The environmental consequences of the postulated accidents are based on no-instrument action. For inadvertent releases relative to violation of administrative procedures, monitors provide means for limiting radioactivity release as well as alarming functions. The plant vent monitor will close the flow control valve in the waste decay tanks discharge line when the radiation level in the plant vent exceeds a preset level. Where liquid waste releases are involved, the waste processing system liquid discharge monitor trips shut a valve in the liquid waste discharge line when the radiation level in the discharge line exceeds a preset level.

C. Waste Gas Release Procedures

There is normally no need to vent the waste gas processing system, although occasional discharges will be required to perform maintenance. The waste gas release is an operator decision based on weather conditions and activity contained in the waste gas. When the operator has decided to release waste gas, he first samples the gas to determine its activity concentration. With this information and total pressure in the tank, the operator knows the quantity of activity to be released as well as the rate at which the gas can be released. To make the actual release, he must unlock and then open the manual isolation valve at the tank discharge and set the discharge flow control valve at the desired rate based

discharge limits to unrestricted areas. This effluent sampling program will be of such a comprehensive nature as to provide the information for the effluent measuring and reporting programs required by 10 CFR 50.36a, in annual reports to the NRC. The frequency of the periodic sampling and analysis described herein is a minimum. Table 11.4-6 summarizes the sample and analysis frequency schedules presented in the following paragraphs.

The following sample regime will apply to all potentially radioactive liquid effluents released to the plant discharge header from the liquid waste processing system.

A. Measurements will be made on a representative sample of each batch of effluent released and kept as a record together with the volume of the batch, the average dilution water flow used during discharge, and the time and date of release.

B. At least monthly, a batch that is typical of average releases of radioactivity will be analyzed for dissolved fission and activation gases. This analysis has a minimum detectable concentration (MDC) as specified in table 11.4-6.

C. Proportional composite samples will be made up periodically to calculate total activity released. These will be samples in which the quantity of liquid added to the composite from each batch released will be proportional to the quantity of liquid in that batch. The composite will represent the average concentration prior to release and, by multiplying by the total volume released, will represent the quantity of radioactivity released during the compositing period. Such composite samples will be made up and analyzed in accordance with table 11.4-6

The steam generator blowdown system sample regime will be as specified in table 11.4-6.

Turbine building sump releases and condenser blowdown (during releases via this pathway) will be sampled and analyzed in accordance with table 11.4-6.

The following sample regime will apply to any intentional release from each containment purge exhaust:

A. The meteorological conditions of wind speed, wind direction, and atmospheric stability will be determined and averaged on an hourly basis for the purpose of determining the atmospheric dispersion during the period of release.

B. A representative gaseous sample of each release will be analyzed for individual noble gas nuclides in accordance with table 11.4-6. The gross noble gas activity released from the containment will be determined using grab samples.

C. A representative sample of each release will be analyzed for tritium in accordance with table 11.4-6. The samples will be collected by condensation or adsorption.

The following sample regime will apply to the potentially radioactive gaseous releases continuously discharged from the plant vent stack system and the condenser steam jet air ejector system:

A. Meteorological measurements of wind speed, wind direction, and atmospheric stability will be made and averaged over each 1-h period.

B. Gaseous activity releases will be quantitatively determined based on gaseous sample analyses and release flowrates for each of these effluent streams. The accumulated releases will be reported on a quarterly basis.

C. Within 1 month of initial criticality, at least monthly thereafter, and then following each refueling, process change, or other occurrence that could alter the mixture of radionuclide gas, samples will be analyzed for principal gamma emitting nuclides and tritium in accordance with table 11.4-6.

The following sample regime will also apply to the gaseous releases from the plant vent stack system:

A. A sample will be drawn through an iodine sampling device to determine the quantity of radioactive iodine isotopes released. The device will be analyzed at least weekly for I-131 and I-133 in accordance with table 11.4-6.

B. A continuous sample will be drawn through a particulate filter device and analyzed weekly for principal gamma emitting nuclides (Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, Ce-144, and I-131) in accordance with table 11.4-6.

TABLE 11.4-6 (SHEET 1 OF 2)

EFFLUENT SAMPLE AND ANALYSIS SCHEDULE

Effluent Stream	Sample Frequency	Minimum Analysis Frequency	Type of Activity Analysis Performed	Minimum Detectable Concentration (MDC) (μCi/ml)
Radioactive waste processing system discharges (Batch Waste Release Tanks)	Each batch prior to discharge	Each batch prior to discarge	Principal gamma emitters I-131	5×10^{-7} 1×10^{-6}
	One batch/month prior to discharge	Monthly	Dissolved gases (gamma emitters)	1×10^{-5}
	Each batch prior to discharge	Monthly composite	H-3 Gross alpha	1×10^{-5} 1×10^{-7}
	Each batch prior to discharge	Quarterly composite	Sr-89, Sr-90, Fe-55	5×10^{-8} 1×10^{-6}
Steam generator blowdown processing system discharge	Daily grab sample	Weekly composite	Principal gamma emitters I-131	5×10^{-7} 1×10^{-6}
	Monthly grab sample	Monthly	Dissolved gases (gamma emitters)	1×10^{-5}
	Daily grab sample	Monthly composite	H-3 Gross alpha	1×10^{-5} 1×10^{-7}
	Daily grab sample	Quarterly composite	Sr-89, Sr-90, Fe-55	5×10^{-8} 1×10^{-6}
Condenser Blowdown	Daily grab sample During discharge	Weekly composite	Principal gamma emmitters H-3	5×10^{-7} 1×10^{-5}
Turbine building sump	Grab sample prior to release	Weekly composite	Principal gamma emitters H-3	5×10^{-7} 1×10^{-5}
Waste gas storage tank releases	Grab sample each tank prior to release	Each tank prior to release	Principal gamma emitters	1×10^{-4}
Containment purge	Grab sample each purge prior to release	Prior to each purge	Principal gamma emitters H-3	1×10^{-4} 1×10^{-6}
Condenser steam jet air ejector discharge	Monthly grab samples	Monthly	Principal gamma emitters H-3	1×10^{-4} 1×10^{-6}
Plant vent stack	Monthly grab samples	Monthly	Principal gamma emitters H-3	1×10^{-4} 1×10^{-6}

APPENDIX 11A

TRITIUM CONTROL

The release of tritium to the environment from operating Westinghouse pressurized water reactors (PWRs) has always been well below 10 CFR 20.1 - 20.601 limits. This section discusses the reduced tritium production in the plant as a result of employing zircaloy-clad fuel and silver-indium-cadmium control rods.

11A.1 SYSTEM SOURCES

There are two principal contributors to tritium production within the PWR system: the ternary fission source and the dissolved boron in the reactor coolant. Additional small contributions are made by Li-6, Li-7, and deuterium in the reactor water. Tritium production from different sources is shown in table 11A-1.

11A.1.1 THE FISSION SOURCE

This tritium is formed within the fuel material and may do one of the following:

A. Remain in the fuel rod uranium matrix.

B. Diffuse into the cladding and become hydrated and fixed there.

C. Diffuse through the clad for release into the primary coolant.

D. Release to the coolant through macroscopic cracks or failures in the fuel cladding.

Previous Westinghouse design has conservatively assumed that the ratio of fission tritium released into the coolant to the total fission tritium formed was approximately 0.30 for zircaloy-clad fuel. The operating experience at the R. E. Ginna Plant of the Rochester Gas and Electric Company, and at other operating reactors using zircaloy-clad fuel, has shown that the tritium release through the zircaloy fuel cladding is substantially less than earlier estimates predicted. Consequently, the release fraction may be revised downward from 30 percent to 10 percent based on this data.[1]

11A.1.2 CONTROL ROD SOURCE

The full and part length rods for this plant are of silver-indium-cadmium. There are no reactions in these absorber materials which would produce tritium, thus eliminating any contribution from this source.

11A.1.3 BORIC ACID SOURCE

A direct contribution to the reactor coolant tritium concentration is made by neutron reaction with the boron in solution. The concentration of boric acid varies with core life and load follow, so that this is a steady decreasing source during core life. The principal boron reactions are the B-10(n, 2α)H-3 and B-10(n, α)Li-7(n, nα)H-3 reactions. The Li-7 reaction is controlled by limiting the overall lithium concentration to approximately 2 ppm during operation. Li-6 is essentially excluded from the system by utilizing 99.9 percent Li-7.

11A.1.4 BURNABLE SHIM ROD SOURCE

These rods are in the core only during the first operating cycle and their tritium contribution is potential only during this period.

11A.1.5 LITHIUM AND DEUTERIUM

Lithium and deuterium reactions contribute only minor quantities to the tritium inventory, as shown in table 11A-1.

11A.2 DESIGN BASES

The design intent is to reduce the tritium sources in the reactor coolant system to a practical minimum in order to permit longer retention of the reactor coolant within the plant. Reduction of source terms is provided by utilizing silver-indium-cadmium control rods and the determination that the quantity of tritium released from the fuel rods with zircaloy cladding is less than originally expected.

11A.3 DESIGN EVALUATION

Table 11A-1 is a comparison of a typical design basis tritium production, which has been utilized in the past to establish system and operational requirements of the plant and present

expected values.[1] It is noted that there are two principal contributors to the tritium production: ternary fission source and the dissolved boron in the reactor coolant. Of these sources it is noted that the 30 percent release of ternary fission through the cladding was the predominant contributor in past design considerations.

Because of the importance of this source on the operation of the plant, Westinghouse has been closely following operating plant data. Table 11A-2 represents tritium releases during one calendar year for different Westinghouse PWR plants. Further, a program is being conducted at the R. E. Ginna Plant to follow this in detail. The R. E. Ginna Plant has a zircaloy-clad core with silver-indium-cadmium control rods. The operating levels of boron concentration during the startup of the plant are approximately 1100 to 1200 ppm of boron. In addition, burnable poison rods in the core contain boron which will contribute some tritium to the coolant, but only during the first cycle. Data during the operation of the plant have indicated very clearly that the present design sources were indeed conservative. The tritium released is essentially from the boron dissolved in the coolant and a ternary fission source which is less than 10 percent. In addition to this data, other operating plants with zircaloy-clad cores have also reported very low tritium concentrations in the reactor coolant system after considerably longer operation.

For a leakage from the primary coolant system into the containment of 40 lb/day, with an assumed tritium concentration in the coolant of 2.5 $\mu Ci/cm^3$ (no containment ventilation purge), the tritium concentration in the atmosphere of the containment would be low enough to permit access without protective equipment by plant maintenance personnel for an average of 2 h/week.

Leakage into the containment atmosphere is based on leakages from equipment such as pumps and valves. Abnormal leakages in excess of the design estimate have occurred in operating plants. The leaking components have been identified and corrective measures have been taken. For example, bellows sealed valves, diaphragm sealed valves, and pump seal purge systems have been employed.

The total activity that would be released from the containment purge during refueling operations would range in the order of 20 to 40 Ci of tritium, depending on the core cycle, relative humidity, etc. It is not proposed that this amount of activity from evaporative losses be collected but that it be discharged from the plant. Similarly, any radioactive gases in the containment would be discharged. Evaporation of tritium from the refueling pool has been considered in evaluating the

consequences to tritium on both operators and environmental releases. This indicates maximum tritium concentration in the containment consistent with 40 h/week occupancy and total tritium release of about 30 Ci/refueling.

The tritium source terms in the reactor coolant are at a low level (approximately 1110 Ci/cycle) such that it is possible to discharge tritium in amounts to preclude in-plant exposure problems without exceeding the "as low as practicable" design objective. Alternatively, without any intentional removal of tritiated water:

A. Tritium levels should not cause a problem during refueling through the 40-year operating lifetime.

B. Special procedures (purging, etc.) prior to containment access may be required.

Credit is taken for dilution of the reactor coolant system water by refueling water and spent-fuel pool. Discharge of the tritiated water from the plant is therefore possible at extended intervals or, if discharged on a regular basis, would be below the 10 CFR 20.1 - 20.601 limits because of the reduced production rates.

Based on the above, the following conclusions have been reached:

A. The tritium levels in plants operating with zircaloy-clad cores will be substantially lower than previous design predictions.

B. The tritium source in the plants will be reduced by utilizing silver-indium-cadmium control rods.

C. Containment access during power operation and refueling with continued storage of the tritium in the plant is possible with the application of special procedures (purging, etc.) prior to containment access.

D. The containment tritium purge is relatively small compared to the total available and will be discharged.

TABLE 11A-2

TRITIUM RELEASES FOR 1971 FROM
WESTINGHOUSE-DESIGNED OPERATING REACTORS

Plant	Type of Cladding	Total Released (Ci)	Average Discharge Concentration ($\mu Ci/cm^3$)	Fraction Column 2, Table II, Appendix B, 10 CFR 20.1-20.601 (3×10^{-3} $\mu Ci/cm^3$)
Yankee Rowe	Stainless steel	1633	5.9×10^{-6}	2.0×10^{-3}
Connecticut Yankee	Stainless steel	5830	7.7×10^{-6}	2.6×10^{-3}
San Onofre	Stainless steel	4570	6.7×10^{-6}	2.4×10^{-3}
Robert E. Ginna	Zircaloy	154	2.3×10^{-7}	7.7×10^{-5}
H. B. Robinson Unit 2	Zircaloy	118	1.7×10^{-7}	6.0×10^{-5}
Point Beach Unit 1	Zircaloy	266	4.7×10^{-7}	1.6×10^{-4}

12.0 RADIATION PROTECTION

12.1 SHIELDING

12.1.1 DESIGN OBJECTIVE

The primary objective of the shielding design and access control is to protect operating personnel and the general public from potential radiation sources in the reactor, the radwaste system, and other auxiliary systems including associated equipment and piping.

Shielding is designed to perform the following functions:

A. Limit the dose to plant personnel, construction workers, vendors, and visitors during normal operation, including anticipated operational occurrences, to within a few percent of the guidelines of 10 CFR 20.1 - 20.601.

B. Limit the dose to plant personnel, in the unlikely event of an accident, to within the guidelines of 10 CFR 50, Appendix A, General Design Criterion 19, to permit termination of accident conditions without undue risk to the general public.

C. Limit dose to certain components in high radiation areas and very high radiation areas within specified radiation tolerances.

D. Protect certain components to prevent excessive neutron activation and facilitate access.

E. Limit dose to persons at the boundary of the restricted area to a small fraction of the guidelines of 10 CFR 20.1 - 20.601 due to direct radiation during normal operation.

The following guides are used in shield design to achieve the above objectives. All plant areas are divided into zones according to the dose rates given below. These zones are for planning purposes; actual dose rates will be determined by surveys.

Zone Designation	Dose Rate (mrem/h)
I-A	≤0.2
I	≤0.5
II	≤2.5
III	≤15.0
IV-A	≤25.0
IV	≤100.0
V	>100.0

A. Access control and shielding design are considered according to the above guidelines in determining optimum plant layout that will allow personnel to perform their normal functions, based on required stay times to perform these functions, with the minimum of exposure.

B. All pipes and ducts penetrating the primary and secondary shields are located in positions so that a direct radiation shine from high radiation sources such as the reactor vessel and components of the reactor coolant loops is avoided.

Penetrations from all pipes and ducts through the shield walls are located to avoid a direct line of sight with the radiation source to prevent streaming into lower radiation zones. Grouting materials have been used to fill voids between the penetration and the wall where necessary.

C. Shield discontinuities include concrete hatch covers, shielding doors, and access labyrinths. To reduce radiation streaming through gaps between the main shield and a removable section, offsets have been used and the gaps are not in line of sight of the radiation source where this is feasible. Access labyrinths into rooms containing radiation sources such as gas decay tanks, coolant sampling equipment, evaporators, and filters have been designed to eliminate a direct shine through the offset passage to the accessible areas.

D. Radioactive piping is routed to minimize exposure to plant personnel. This is accomplished by:

1. Minimizing radioactive pipe routing through the corridors and low radiation zones.

the containment to 0.5 mrem/h. In case of an accident, the shielding will minimize the station doses to less than 5 rem whole body and the offsite doses to less than 0.5 mrem/h.

12.1.2.4 Primary Shield

The primary shield of 6-ft thick reinforced concrete surrounds the reactor vessel. The cavity between the primary shield and the reactor vessel is air cooled to prevent overheating, dehydration, and degradation of the shielding properties of the concrete. The primary shield, in conjunction with the secondary shield, serves to attenuate the radiation from the reactor vessel and reactor coolant equipment. It permits limited access in the containment during normal power operation and allows limited access to reactor coolant equipment. The primary shield also reduces neutron activation of the components and structures over the life of the plant. Penetrations through the shield walls are described in subsection 12.1.1, item B.

12.1.2.5 Secondary Shield

The secondary shield consists of 2 to 3 1/2 ft of reinforced concrete and surrounds the reactor coolant equipment, steam generators, pressurizer, and associated piping. This shield supplements the primary shield by further attenuation of neutrons escaping the primary shield and permits limited access to the containment during full power operation by attenuating the nitrogen-16 gammas from the primary coolant system. Penetrations through the shield walls are described in subsection 12.1.1, item B.

12.1.2.6 Spent Fuel Pool Shielding

Shielding is provided for protection during all phases of spent fuel removal and storage. Operations requiring shielding of personnel are spent fuel removal from reactor, spent fuel transfer through refueling canal and transfer tube, spent fuel storage, and spent fuel shipping cask loading prior to transportation.

Since all spent fuel removal and transfer operations will be carried out under borated water, minimum water depths above the tops of the fuel assemblies have been established to provide radiation shielding protection. The dose rates at the water surface are less than 2.5 mrem/h. The concrete walls of the fuel transfer canal and spent fuel pool supplement the water

shielding and limit the maximum continuous radiation dose levels in working areas to less than 2.5 mrem/h.

The refueling water and concrete walls also provide shielding from activated rod cluster control assemblies and reactor internals which will be removed at refueling times. Although dose rates will generally be less than 2.5 mrem/h in working areas, certain manipulation of fuel assemblies, rod cluster control assembly, or reactor internals may produce areas where dose rates exceed 2.5 mrem/h for short periods. However, the radiation levels will be closely monitored during refueling operations to establish the allowable exposure times for plant personnel in order not to exceed the integrated dose specified in 10 CFR 20.1201 - 20.1208.

All spent-fuel pool penetrations are located higher than the minimum water depth above the fuel assemblies, so that a failure in any penetration will not drain the pool to less than the minimum water level.

12.1.2.7 Control Room

Control room shielding design is based on the requirements set forth in 10 CFR 50, Appendix A, General Design Criterion 19, which requires occupancy of and access to the control room under accident conditions. The dose to personnel will be limited to 5 rem whole body or its equivalent to any part of the body for the duration of the accident. The accident analysis in chapter 15 indicates that this dose to personnel will be less than 5 rem.

Protection of control room personnel from the fission product release in the containment is provided by the concrete walls between them. Emergency air conditioning and filtration systems are provided for post-loss-of-coolant accident conditions and are described in detail in subsection 9.4.1. Figure 9.4-1 contains the control room layout and isometrics of the control room and associated shielding.

12.1.2.8 Auxiliary Building

The auxiliary building shielding includes all concrete walls, covers, and removable blocks that protect personnel working near various system components of the waste processing system, chemical and volume control system, boron thermal regeneration system, and safety injection system. Typical radioactive sources are the waste evaporator, recycle evaporator, demineralizers, filters, waste gas decay tanks, waste holdup tanks, recycle holdup tanks, and the waste drumming area.

12.1.3.1 Sources for Normal Full Power Operation

The main sources of activity during normal full power operation are N-16 from coolant activation processes, fission products from fuel clad defects, and corrosion and activation products. The activity level of N-16 at various locations in the reactor coolant system (RCS) is shown in figure 12.1-21. The isotopic inventory of fission, corrosion, and activation products in the reactor coolant is given in table 11.1-2. All shielding is based on the maximum case of clad defects in fuel rods producing 1.0 percent of core thermal power. Expected sources would be based on defects in fuel rods producing 0.25 percent of core thermal power as discussed in section 11.2. Each plant system was shielded according to the amount of activity present and adjacent zoning and access criteria. These systems include:

- Reactor coolant system.
- Chemical and volume control system.
- Waste processing system.
- Boron recycle system.
- Spent fuel pool cooling and purification system.
- Steam generator blowdown processing system.

The N-16 activity of the coolant is the controlling radiation source in the design of the RCS secondary shielding and is plotted in figure 12.1-21 as a function of transport time in a reactor coolant loop.

The radiation sources in the chemical and volume control system (CVCS) are given in table 12.1-1.

One of the purposes of the CVCS is to provide continous purification of the reactor coolant water. The major equipment items include the regenerative and letdown heat exchangers, mixed bed and cation bed demineralizers, reactor coolant filter, volume control tank, and charging pumps. The boron thermal regeneration (BTR) subsystem contains the three BTR heat exchangers and the BTR demineralizers. The seal water subsystem for the reactor coolant pumps includes the injection and return filters and the seal water heat exchanger.

Table 12.1-1 gives a summation of the activity, by energy groups, of all isotopes listed in table 11.1-2. The delay time from the reactor coolant loop is sufficient for decay of the N-16 isotope.

The radiation sources in the ion exchangers, volume control tank, filters, and heat exchangers of the CVCS are also given in table 12.1-1.

The mixed bed retains the fission product activity, both cations and anions, and the corrosion product (crud) metals. The cation bed can be used intermittently to remove lithium for pH control and to supplement the mixed bed in removing Y, Cs, Mo, and the crud metals.

The BTR beds are used to regulate the boron concentration in the reactor coolant water. They are utilized during load follow operations and in removing boron from the coolant as the nuclear fuel is depleted. These demineralizers also collect radioactive anions, such as iodine, which may have passed through the mixed bed.

The regenerative and excess letdown heat exchangers are located in the containment building. They provide the initial cooling for the reactor coolant letdown and their sources include N-16 activity. The balance of the CVCS heat exchangers is located in the auxiliary building where N-16 activity is not a significant factor.

The letdown heat exchanger provides second-stage cooling for the reactor coolant prior to entering the demineralizers. The activity at this point is identical to the letdown coolant source.

The thermal regeneration heat exchangers include the moderating, chiller, and letdown reheat units. The radiation sources in this equipment are modified to account for activity removed by the demineralizers upstream of the units.

The seal water heat exchanger cools the water from the reactor coolant pump seals. In the source tabulation, credit has been taken for activity removed by the demineralizers and the volume control tank.

The radiation sources in the waste processing system are tabulated in table 12.1-2. The major equipment items in the waste gas portion are the waste gas compressors, hydrogen recombiners, and gas decay tanks. The radiation sources in this equipment are based on cold shutdown procedures during which the radioactive gases are stripped from the RCS. The radiation sources in the waste gas equipment are conservatively assumed to be identical.

The liquid waste processing system is considered as several subsystems, based on its intended use during normal operation. The equipment items normally associated with processing reactor

grade water are the waste holdup tank, waste evaporator feed filter, and waste evaporator. The evaporator distillate is directed to the waste condensate tank and may be further processed through the waste evaporator condensate demineralizer and filter, if required. The waste evaporator concentrates are sent to the drumming station or solidification and dewatering building for packaging.

Low activity, nonreactor grade water is directed to the floor drain or laundry and hot shower subsystems. Normally this water is analyzed, then discharged. If activity levels prevent this, the water can be processed by a demineralizer/filter or the waste evaporator. The equipment included in the subsystem is the floor drain tank and filter, laundry and hot shower tank and filter, waste monitor tank demineralizer and filter, and two waste monitor tanks. The floor drain and waste monitor tanks provide surge capacity for the waste holdup tank during periods when abnormal volumes of liquid waste are encountered. Hence, for shielding purposes the radiation sources in these tanks are assumed to be the same, i.e., degassed reactor coolant. Similarly, the sources on the floor drain tank filter are the same (100 R/h contact) as the waste evaporator feed filter since they can operate in similar service. The sources on the waste monitor tank demineralizer and filter are based on circulating reactor coolant through these components.

Radioactive spent resins discharged from the various demineralizers are retained in the spent resin storage tank. The mixed bed demineralizer contains the most radioactive resin discharged to the storage tank; these sources determine the tank shielding required. The short-lived activity is allowed to decay (~30 days), and the resin is then directed to the solidification and dewatering facility for packaging. The associated equipment includes the spent resin storage tank and the resin sluice pump and filter. The resin sluice filter is shielded for radiation levels of 100 R/h contact.

Radiation sources in the various pumps in this system are assumed to be identical to the liquid sources in the tank from which the pump takes suction.

Sources in the laundry, hot shower tank and filter, and waste condensate tank are negligible; these items do not require shielding.

The evaporator concentrates and the spent resin are packaged at the solidification and dewatering facility for shipment to an offsite burial facility. Prior to shipment, the packaged waste is stored as described in section 11.5. The shielding for the drum storage area is designed to accommodate the full storage capacity with each drum reading 1 R/h at 3 ft. Spent

resin can be stored in a steel shipping shield, if necessary, to limit radiation levels.

The radiation sources in the boron recycle system are listed in table 12.1-3. The major equipment items included in this system are the recycle holdup tanks and the recycle evaporator with its associated equipment, i.e., feed demineralizers and filter, condensate demineralizer and filter, and concentrates filter. Radiation sources in the various pumps are assumed to be identical to the liquid sources in the tank from which the pump takes suction.

The evaporator feed demineralizers are located upstream of the holdup tanks and contain mixed-bed resins which remove nongaseous activity from the reactor coolant directed to the holdup tanks. A dilution factor of 10 across these beds is taken for all particulate activity.

The evaporator condensate demineralizer is charged with anion resin to remove any boron and iodine activity which may be carried over with the evaporator condensate.

The recycle holdup tanks are each equipped with a diaphragm. Gases which flash from the reactor coolant letdown to the holdup tanks are retained under the diaphragm until /500 ft^3 of gas has accumlated; the gases are then removed to the waste gas system. The radiation sources in the holdup tanks are based on 50 percent of the gaseous activity flashing into the vapor phase.

The recycle evaporator feed filter and condensate filter are located downstream of their respective demineralizers and serve to retain particulates and any resin fines which may escape from the demineralizers.

The maximum radiation sources on these filters are listed below. The sources for the feed filter correspond to a radiation level of 100 R/h contact. The condensate filter sources result in levels of less than 1 R/h contact. The maximum activity of the liquid concentrates in the recycle evaporator is 40 μCi/g. The resultant radiation sources on the concentrates filter correspond to an exposure rate of approximately 3 R/h.

The radiation sources in the spent-fuel pool cooling system are given in table 12.1-4. The system demineralizer and filter are used to maintain water clarity and remove activity released during refueling operations and the subsequent fuel cooling period. The filter sources correspond to an exposure rate of 100 R/h contact.

12.1.4 AREA MONITORING

12.1.4.1 Design Bases

The area radiation monitoring system is provided to supplement the personnel and area radiation monitoring provisions of the plant health physics program described in section 12.3. Included in this system are nine permanently located radiation detectors for Unit 1 and ten permanently located radiation detectors for Unit 2, which provide continuous local and remote indication and alarm of direct radiation dose rate levels. The primary objectives of the system are:

A. To immediately alert plant personnel entering or working in normally unlimited occupancy areas of increasing or abnormally high radiation levels, which, if unnoticed, might possibly result in inadvertent overexposures.

B. To inform the control room operators of the occurrence and approximate location of abnormal events resulting in the release of radioactive materials or the degradation of shielding structures.

C. To provide, in the event of many types of hypothetical accidents leading to the contamination of the plant, a means of remotely determining external dose rates in those areas most likely to be contaminated, prior to entry by personnel.

D. To provide a continuous record of external dose rates at selected locations, thereby ensuring detection of transient increases in doses which are attributable to rapid changes in the radioactivity content of equipment and process streams.

E. To provide information on radiological conditions in the containment, in the event of a NUREG 0578 accident (Unit 1) or NUREG 0737 accident (Unit 2).

12.1.4.2 System Description

This system consists of multiple channels which monitor radiation levels in various areas of the plant, among which are the following:

Channel	Area Monitoring
R-1 (Unit 1 only)	Control room
R-1B (Unit 2 only)	Technical support center
R-2	Containment
R-3	Radiochemistry laboratory
R-4	Charging pump room
R-5	Spent fuel building
R-6	Sampling room
R-7	Incore instrumentation area
R-8	Drumming station
R-9 (Unit 2 only)	Sampling panel room

These locations have been chosen as representative of plant locations where significant sources of radioactive material are stored and/or handled or where occupancy is highest.

Detecting medium for the channels is air with a corresponding temperature range of 40°F to 120°F. Each channel consists of a fixed position gross beta gamma Geiger-Mueller tube detector with range 1.0×10^{-4} to 1.0×10^{1} R/h. Figure 12.1-2 contains a functional block diagram for the above area monitor channels.

The area radiation level is indicated locally at the detector and at the radiation monitoring system cabinets, where it is also recorded on one of the multipoint recorders used for process monitors. High radiation alarms are displayed at the radiation monitoring system cabinets and annunciated at the detector location and at the control board in the control room. The control board annunciator provides a single window which alarms for all area radiation monitor channels in addition to process radiation monitor channels R-10 through R-26, except channels R-24A and B and R-25A and B which have individual annunciator windows on the control board. Verification of which area radiation monitor channel has alarmed is done at the radiation monitoring system cabinets in the control room.

To meet the requirements of NUREG 0578, Alabama Power Company (APC) has installed Victoreen model 875 radiation detection

systems R-27A and B to meet the requirements for a high range containment radiation monitor. Each system consists of an ion chamber detector, readout panel, and interconnecting cables. The monitors are located inside containment about 6 ft above the operating deck and approximately 90° apart. These locations ensure that monitors are not protected by massive shielding and they will provide a reasonable assessment of area radiation conditions inside the containment during and following an accident.

A. Each detector is designed to measure gamma radiation.

B. The range of each detector is 1 R/h to 10^7 R/h for photon radiation.

C. The energy response is ±15 percent to 80 keV and ±8 percent from 100 keV to 3 MeV.

D. The calibration frequency will be at a maximum interval of 18 months. Capability exists for onsite calibration of the radiation monitor to 10 R/h.

E. As part of the vendor testing program, Victoreen has had the calibration certified for at least one point per decade of the range between 1 R/h and 10^3 R/h.

12.1.4.3 Design Evaluation

Area monitors are located in areas of the plant which house equipment containing or processing radioactive fluid or where, because of personnel occupancy, it is deemed necessary to monitor continuously. These instruments continually detect and record operating radiation levels. If the radiation level should rise above the setpoint listed for each channel (see table 12.1-9), an alarm is initiated in the control room. Local annunciation is provided at the detector to indicate high radiation levels to personnel in the area. The radiation monitoring system operates in conjunction with regular and special radiation surveys and with chemical and radiochemical analyses performed by the plant staff. Adequate information and warning are thereby provided for the continued safe operation of the plant and assurance that personnel exposure does not exceed the limits of 10 CFR 20.1201 - 20.1208.

12.1.5 OPERATING PROCEDURES

12.1.5.1 General

The health physics superintendent is responsible for assisting the training director in developing a radiation protection training program and for developing a radiation surveillance program to ensure that exposures of all personnel are kept within the limits of 10 CFR 20.1201 - 20.1208. See section 12.3 for a description of the health physics program as it relates to shielding operating procedures.

12.1.5.2 Procedures

The basic principles of time, distance, and shielding will be applied during operation and maintenance to ensure that personnel exposure will be within limits. Specifically, the following procedures and techniques will be employed:

A. During initial startup, neutron and gamma dose rate surveys will be performed to determine the adequacy of shielding.

B. During normal operations dose rate surveys will be performed periodically throughout the plant and areas will be posted accordingly. This procedure will ensure that data are available for planning operation and maintenance activities.

C. Radiation areas will be conspicuously posted. High radiation areas will be conspicuously posted and barricaded. High radiation areas of 1 R/h measured at 30 cm but less that 500 rads/h measured at 1 m will be locked and the keys maintained under the administrative control of the shift supervisor or shift foreman. Very high radiation areas (levels greater than 500 rads/h at 1 m) will be conspicuously posted and access will be controlled in accordance with plant procedures. These control measures comply with the NRC acceptance criteria contained in 10 CFR 20.1601 and 20.1602, respectively.

D. A radiation work permit system will be employed to ensure proper administrative control over work in restricted areas. The permit is designed to ensure that the radiation conditions are known and that appropriate measures are taken to minimize the dose received by personnel.

E. Extension tools will be used when possible or practical to increase the distance from the radiation source to the worker.

F. Equipment will be moved to areas of lower radiation fields for maintenance when possible or practical.

G. Portable shielding in the form of lead bricks, lead sheets, lead shot, and/or high density concrete blocks will be considered for use when the requirements of items E and F are not possible or practical. Steel plates will be used in lieu of lead where high temperature may be a factor. (A shielding evaluation will be conducted prior to installing shielding on safety-related equipment.)

H. A personnel dosimetry program, as described in subsection 12.3.3, will be administered by the chemistry and environmental group to ensure compliance with 10 CFR 20.1502.

Each permanent plant employee who is to be a radiation worker will attend radiation worker orientation prior to being allowed unescorted access.

Experience gained during the operation and maintenance of FNP and that of several nuclear plants with whom SNC has contact will be used to provide a basis for further evaluation and development of shielding procedures.

12.1.6 ESTIMATES OF EXPOSURE

12.1.6.1 Exposures at the Boundary of the Restricted Area and in Uncontrolled Areas

12.1.6.1.1 Normal Plant Operations

Maximum annual exposures due to direct radiation from normal plant operation will not exceed 5 mrem/year outside the restricted area.

Doses at the site boundary from radioactive liquid releases are given in subsection 11.2.9 and those due to gaseous effluents are dealt with in subsection 11.3.9.

At and beyond the site boundary, the interpreted man-rem values (for normal operations) as a function of distance are computed from the expected gaseous and liquid releases, and the atmospheric dilution factors are presented in subsection 11.3.8, liquid dilution and reconcentration factors in

subsection 11.2.8, and the population density in subsection 2.1.3.

12.1.6.1.2 Operational Occurrences

Maximum radiation exposures resulting from operational occurrences are discussed in detail in chapter 15. Under the most severe conditions, the dose values at the site boundary, the low population zone distance, and at the visitors center will be well below 10 CFR 100 guidelines.

12.1.6.2 Exposures Within Station Buildings and Within the Restricted Area

12.1.6.2.1 Normal Plant Operations

Maximum probable dose rates for different areas in the plant are discussed in detail in subsections 12.1.1 and 12.1.2. In areas continuously occupied by the plant personnel, such as the control room, administrative offices, and locations external to the plant buildings but within the restricted area, the maximum dose rate will be less than 0.2 mrem/h. resulting in an annual dose of less than 500 mrem. Administrative controls and controlled access will ensure that the plant personnel working in radiation areas will not receive doses in excess of those established in 10 CFR 20.1201 - 20.1208.

The dose rates given in subsection 12.1.1 represent the upper limits, and the expected exposures will be significantly lower than these limits.

12.1.6.2.2 Operational Occurrences

Various operational occurrences are discussed in detail in chapter 15. Minor occurrences like spills or leakages will contribute no appreciable increase to normal exposures and will be of only local significance.

However, if the radiation levels from an occurrence call for evacuation of the plant, this will be indicated on the radiation monitoring equipment and the emergency evacuation plan will remain in operation. Personnel essential to shut down the plant and maintain it in a safe condition under accident conditions will be confined to the control room. Maximum dose rates and protection of the control room are discussed in detail in paragraph 12.1.2.6.

12.2 VENTILATION

12.2.1 DESIGN OBJECTIVES

The plant ventilation systems, in addition to their primary function of preventing extreme thermal environmental conditions for operating personnel and equipment, will provide effective protection for operating personnel against possible airborne radioactive contamination in areas where this may occur.

The systems will operate to ensure that the maximum airborne radioactivity levels for normal operation, including anticipated operational occurrences, are within the limits of column 1, Table I, Appendix B to 10 CFR 20.1 - 20.601, for areas within plant structures and on the plant site where construction workers and visitors are permitted. The maximum levels correspond to design basis reactor coolant inventory. The average airborne radioactivity levels meet the requirements of column 1, Table I, Appendix B to 10 CFR 20.1 - 20.601 and 10 CFR 50 and, in fact, will be considerably smaller since average coolant inventories and actual equipment leakages will be small.

The systems will operate to ensure compliance with normal operation offsite release limits as discussed in section 11.3.

The control room ventilation system will also operate to provide a suitable environment for equipment and continuous personnel occupancy in the control room under postaccident conditions in accordance with 10 CFR 50, Appendix A, General Design Criterion 19.

The expected airborne radioactivity levels for normal operations and anticipated operational occurrences, in areas within plant structures, including each building in the reactor facility and on the plant site where personnel, construction workers, or site visitors are permitted, are presented in table 12.2-1. The methods used are discussed in subsection 12.2.6. Assumptions used to calculate these airborne radioactivity levels are presented in table 12.2-2, and a discussion of the resulting estimated doses is also presented in subsection 12.2.6.

12.2.2 DESIGN DESCRIPTION

In order to accomplish the design objectives, certain general design guidelines are followed when possible and applicable:

A. Air movement patterns are provided from areas of lesser contamination potential to areas of

progressively greater contamination potential prior to final exhaust.

B. Slightly negative pressures are maintained, where applicable, to prevent uncontrolled exfiltration of contamination. Slightly positive pressure is maintained in the control room to prevent infiltration of potential contaminants.

C. Valves and equipment are maintained as leaktight as possible in order to prevent leakage of radioactive water and subsequent airborne contamination.

D. Individual air supplies are provided for each building in order to keep potentially contaminated airflows separate from noncontaminated air.

E. High efficiency particulate air (HEPA) and charcoal filters are provided on the exhaust side of the radioactive and fuel handling ventilation systems to remove airborne activity and to reduce onsite and offsite radiation levels.

F. The fresh air supply to the control room is designed to be operable during loss of offsite power. The air is filtered to prevent contamination of the control room.

These guides are incorporated in the heating and ventilation design described in section 9.4. The following is a brief summary of those systems.

12.2.2.1 Control Room Ventilation

During normal plant operation, control room air is recirculated through air conditioning units to maintain control room design conditions of temperature and relative humidity. Fresh air makeup is provided by a supply duct from the air conditioning unit in the computer room. A radiation monitoring system and a control room charcoal filter recirculation system have been provided to detect and reduce radiation levels in the control room in the event of an airborne radioactivity accident in the control room area. Redundant radiation monitors in the control room area detect high radiation in the control room atmosphere. When a high radiation level is sensed by the detectors, a high radiation alarm is actuated in the control room and the air paths between the computer room air conditioning system and the control room are isolated. After isolation of the control room and when conditions permit, fresh air can be brought in manually through redundant control room pressurization charcoal

A. Reactor coolant leakage to the containment building.

B. Reactor coolant leakage to the auxiliary building.

C. Secondary side system leakage.

D. Waste gas processing system leakage.

A complete identification of all radioactive sources and an estimate of resulting radioactive effluents are further described in section 11.1

12.2.3.1 Reactor Coolant Leakage to the Containment Building

Leakage into the containment atmosphere is based on leakages from equipment such as pumps and valves. The leakage is estimated to be 40 lb/day.

12.2.3.2 Reactor Coolant Leakage to the Auxiliary Building

This effluent represents nonrecyclable reactor coolant from system leaks in the auxiliary building. It is assumed that the total amount of leakage is 20 gal/day.

12.2.3.3 Secondary Side Leakage to the Turbine Building

The rate of steam leakage from the secondary system is estimated to be 6 gal/min when condensed.

In addition, liquid leakage from systems operating below 212°F is estimated to be 12 gal/min.

12.2.3.4 Waste Gas Processing System Leakage

The gaseous waste processing system is designed to contain the gaseous waste for the lifetime of the plant. However, although all precautions are taken to avoid any leakage from the system, an estimated leakage of 100 sf^3/year is assumed.

12.2.4 AIRBORNE RADIOACTIVITY MONITORING

An analysis of the auxiliary building was conducted in order to identify the potential points of releases of airborne radioactive material in the form of contaminated steam or liquid discharges from valves, pumps, tanks, sumps, and other release mechanisms. For plant design, an NRC acceptance

criterion, discussed in subsection 12.1.2 of the FNP FSAR Safety Evaluation Report, required concentrations of airborne radioactive material to be controlled such that limits stated in 10 CFR 20 would not be exceeded. In-plant airborne radioactive materials concentration limits that were in effect at the time of plant design are specifically stated in 10 CFR 20.103, which references column 1, Table I of Appendix B to 10 CFR 20.1 - 20.601. The evaluations in this section show that this criterion was addressed in plant design.

During plant operations, access to the rooms, enclosures, or operating areas containing release points, and having the potential of causing operating personnel to be exposed to airborne radioactive material to an average concentration in excess of the limits specified in Appendix B, Table 1, of 10 CFR 20.1001 - 20.2401, will be controlled by a continuous program of:

A. Surveys with continuous online type of sampling equipment.

B. Clear identification of spaces with appropriate caution signs.

C. Locked doors with alarms, as appropriate.

D. Administrative controls through the use of radiation work permits and procedures.

Continuous online radiation monitors will be provided to monitor the various auxiliary building compartments at all elevations within the building, as follows:

Unit No.	Radiation Monitor No.	Elevation (ft)	Total Exhaust Flow From Area (ft^3/min)
1 and 2	R-30	77, 83, and 100	11,610
1 and 2	R-31	121	11,605
1 and 2	R-32	139	10,600
1 and 2	R-33	155	18,175
1	R-34	155 (access control area)	1,860
2	R-35A and B	175 (control room)	1,000

NOTE: The access control area and control room serve as common facilities for both Units 1 and 2.

Samples for each of the monitors will be taken from representative points of the exhaust portion of the radioactive waste ventilation system. Isokinetic sample nozzles will be used and designed in accordance with American National Standards Institute standard N13.1-1969, Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities.

The design basis of the airborne radioactivity monitoring system provides plant operating personnel with the capability of assessing the levels of airborne radioactive contamination in spaces within the auxiliary building prior to their entry into these spaces. The system is capable of providing:

A. Radioactivity levels in work areas within the auxiliary building.

B. Information relative to variations in general airborne radiation background.

C. Data to augment personnel monitoring measurements.

D. Warning signals, should airborne radiation levels reach a level in excess of the limits specified in Table 1, of Appendix B to 10 CFR 20.1001 - 20.2401.

The following evaluations of minimum sensitivities for specific air monitors were performed to demonstrate the capability of these monitors to measure airborne concentrations at and below maximum permissible concentrations (MPC) stated in column 1, Table I, Appendix B to 10 CFR 20.1 - 20.601, as part of the plant design basis. These evaluations are related to NRC acceptance criteria discussed in subsection 12.1.2 of the FNP FSAR Safety Evaluation Report. The radionuclides upon which these evaluations were based are Cs-137 and Kr-85. Since the limiting concentrations for these radionuclides, stated as derived air concentrations (DAC) in column 3, Table 1, Appendix B to 10 CFR 20.1001 - 20.2401, are less restrictive than the MPC stated in column 1, Table I, Appendix B to 10 CFR 20.1 - 20.601, and since the detector efficiencies are two orders of magnitude below the most limiting DAC values (potential unidentified airborne radionuclides), these evaluations remain valid.

Because each radiation monitor will provide data on the airborne radioactivity levels in several spaces, it was necessary to consider the limiting radioactive isotopes and the MPC, as stated in Table I, Appendix B to 10 CFR 20.1 - 20.601, for: the particular isotopes potentially present in each space; the exhaust flowrate from applicable spaces; and the total airflow at the sample point. To determine the radioactive concentration

at each sample point that could not be exceeded without exceeding the MPC for a limiting isotope in the spaces serviced by the sample point, the MPC value for the limiting isotope was multiplied by the ratio of the flow for the compartment having the lowest airflow to the total airflow at the sample point. A specific evaluation of each sample point follows. (Refer to figure 9.4-6 for locations of the radiation monitors described below.)

All monitors are located in hallways, except R-34 which is located in the containment purge equipment room (Unit 1). The radiation background level at the instrument location is 2 mR/h of 0.662 MeV gamma rays. Detector efficiencies are as follows:

Cs-137	1×10^{-11} $\mu Ci/cm^3$
Kr-85	5×10^{-7} $\mu Ci/cm^3$

The collector efficiency for the particulate monitors is 99 percent for particles 0.3 mm or larger.

Dilution factors are given below and represent the ratio of the flow from the smallest compartment to the total flow from a particular area.

Particulate or iodine monitors are used in the exhaust from all compartments except the control room. Gas monitors are used in the exhaust from el 77 ft to el 100 ft in the auxiliary building and in the computer room air intake. In both of these cases, charcoal filters are installed to remove particulates and iodine.

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The monitors are capable of measuring 1 MPC-h in any compartment.

A. Radioactive Waste Airborne Monitoring Channel R-30

Channel R-30 is a particulate and gas monitor which monitors the exhaust flow from el 77 ft, 83 ft, and 100 ft. The total exhaust flow from the three elevations is 11,610 ft^3/min. The following major equipment rooms are located on these elevations:

Room No.	Equipment
101, 102, 108	Waste gas decay tanks
105, 106	Hydrogen recombiners
111, 125	Containment spray pumps
110	Monitoring control panel room
151, 153, 154, 165, 300	Waste gas decay tank rooms and compressors
164, 168	Laundry and hot shower drain tanks
170	Letdown heat exchanger room
173, 174, 181	Charging pump rooms
159, 178, 180	Evaporator feed pumps, tanks, and filters
179	Valve room

Considering particulate activity, the room with the lowest exhaust flow (90 ft^3/min) is the valve room, No. 179, at el 100 ft.

The minimum monitor sensitivity required to monitor Cs-137 at an MPC level in this room, considering dilution, is:

$$\frac{90}{11,610} \times 1 \times 10^{-8}\ \mu Ci/cm^3 = 7.75 \times 10^{-11}\ \mu Ci/cm^3$$

Radiation monitor sensitivities of $1 \times 10^{-11}\ \mu Ci/cm^3$ for Cs-137 are commercially available. Direct continuous monitoring for airborne iodine is not possible with this system due to the installation of charcoal air filters in the rooms and spaces where airborne iodine could be released. These rooms are numbered 108, 151, 153, 154, 165, 166, and 168.

An analysis of the potential for airborne iodine in these rooms has shown that multiple failures must occur in order to obtain radiation levels that could approach MPC quantities of I-131. These failures

$$\frac{70}{18,175} \times 1 \times 10^{-8}\ \mu Ci/cm^3 = 3.85 \times 10^{-11}\ \mu Ci/cm^3$$

which is within the sensitivity of commercially available radiation monitors.

E. Radioactive Waste Airborne Monitoring Channel R-34

Channel R-34 is a particulate monitoring device which monitors the exhaust flow from the Unit 1 access control area. A monitor is not required for Unit 2 because the Unit 1 access control area is a common facility for both units. The total exhaust flow from this area is 1860 ft^3/min. The following equipment rooms are located in this area.

Room No.	Equipment
438	Hot water heater room
482	Air sample and smear analysis
484	Women's toilet (rad.)
485	Men's toilet (rad.)
487	Instrument calibration
488	Instrument issue and storage

The room with the lowest exhaust flow (100 ft^3/min) is the hot water heater room, No. 438.

The minimum monitor sensitivity required to monitor Cs-137 at an MPC level in this room, considering dilution, is:

$$\frac{100}{1860} \times 1 \times 10^{-8}\ \mu Ci/cm^3 = 5.37 \times 10^{-10}\ \mu Ci/cm^3$$

which is within the sensitivity of commercially available radiation monitors.

F. Control Room Airborne Monitoring Channels R-35A and B

Channels R-35A and B are radioactive gas monitors which sample the 2000-ft^3/min computer room air handling unit intake air. The computer room air handling unit normally supplies 1000 ft^3/min of fresh air makeup to the control room.

G. Summary

The above analysis is conservative in that the MPC values used are based on an exposure to the concentrations specified for 40 h in any period of 7 consecutive days. Since the normal occupancy for these areas is significantly less than this 40-h period and since the airborne contamination results

from leakages from sealed systems, the exposure of personnel to airborne radioactivity will be considerably less than that stated in the above analysis with due allowance for plateout and other variables.

Sufficient sensitivity margin exists for all radiation monitors to compensate for particulate plateout in the ductwork. Monitors will be located as close as possible to the sample point to eliminate particulate plateout in the sample line.

Other areas of potential airborne contamination, such as the containment, penetration room, and spent fuel area, are monitored by the fixed airborne radiation monitoring instruments described in section 11.4. The continuous radiation monitors in these areas, as well as the areas in the auxiliary building, will be augmented by the use of periodic portable air activity samplers.

The samplers will be used as a check on the fixed monitoring system during normal and maintenance operations and to determine airborne activity levels should an accident occur or after receipt of an alarm from the fixed monitoring system. Systems such as the plant vent air particulate monitor system and the plant vent gas monitor described in subsection 11.4.2 will be checked using grab samples.

The results of these checks will be logged and filed as part of the plant records.

12.2.4.1 Containment Airborne Radioactivity Monitoring

Two Kr-85 radiation monitors are provided in the containment purge exhaust ductwork. The monitors are capable of measuring and alarming 1 MPC-h of Kr-85 when operating in a background of 2 mR/h of 1 MeV gamma rays.

Radiation measurements will be made before personnel are permitted to enter the containment.

The location of the radiation monitors are shown in figures 6.2-91, sheet 2 and 6.2-124, sheet 2.

12.2.4.2 Spent-Fuel Area Airborne Radioactivity Monitoring

The spent-fuel area is continuously exhausted to the plant vent during plant operation. The exhaust flow is continuously monitored for high radiation by two Kr-85 gas monitors capable of alarming when a level of 1 MPC-h is reached.

Dilution factors were not considered in the analysis since representative gaseous samples will exist in the exhaust ductwork over the period of 1 h, due to dispersion within the spent-fuel area.

The radiation monitors have a sensitivity of 5×10^{-7} $\mu Ci/cm^3$ in a background of 2 mR/h of 1 MeV gamma rays.

The location of the radiation monitors is shown in figure 9.4-4, sheets 1 and 2.

12.2.4.3 Penetration Room Airborne Radioactivity Monitoring

Access to the penetration room will be on an as-required basis for maintenance or repair of equipment. Manual samples will be taken routinely with portable sampling equipment to permit personnel to enter the penetration room compartments as required.

12.2.5 OPERATING PROCEDURES

12.2.5.1 General

The health physics group is responsible for developing a radiation protection program which will ensure that inhalation exposure is kept as low as is reasonably achievable, consistent with 10 CFR 20.1101 and 20.1701 - 20.1704.

12.2.5.2 Procedures

Inhalation exposure will be minimized during operations and maintenance by using the following procedures and techniques to determine and cope with the hazards present:

A. Monitoring

Air samplers of various flowrates will be used to collect particulates on high efficiency filter media for subsequent counting. For tritium analysis,

freeze-out methods may be used to obtain samples for counting.

Routine smear surveys will be performed to establish the levels of removable contamination throughout the plant so that personnel protection measures or decontamination may be effected.

Assay of noble gases will be performed by drawing an air sample into a sample container and analyzing it on a multichannel analyzer system.

B. Respiratory Protection

In areas where airborne radioactivity can cause exposures in excess of that allowed by 10 CFR 20.1201 - 20.1207 respiratory devices and/or portable HEPA filtration systems may be required. It is the responsibility of the health physics group to monitor such areas, to establish the requirement for respiratory equipment, and to control access to such areas through the radiation work permit program.

Each individual who enters a radiation controlled area will be trained or briefed in accordance with 10 CFR 19.12. A notice describing where radiation control procedures may be examined is posted in accordance with 10 CFR 19.11.

12.2.6 ESTIMATES OF INHALATION DOSES

Peak airborne radioisotopic concentrations in the different buildings of operating pressurized water reactor (PWR) plants have shown that these concentrations are insignificant for PWR plants. The inhalation doses to plant personnel at these plants have been found to be negligible.

The doses to plant personnel and construction workers from airborne radioactivity will depend upon the extent of their occupancy and the time when this occupancy occurs. These doses will be controlled by limiting personnel occupancy in the contaminated areas and by provision of respiratory protection equipment if required. The highest dose to plant personnel will therefore be limited to the maximum permissible dose for occupationally exposed individuals, as specified by 10 CFR 20.1201 - 20.1207.

The assumptions used to estimate concentrations and inhalation doses in the containment, turbine building, and certain regions within the auxiliary building are listed in table 12.2-2. The airborne peak concentrations in each of the regions mentioned above are given in table 12.2-1. In addition, the table gives

the maximum permisssible concentrations for airborne activity in these areas as defined in column 1, Table I, of Appendix B to 10 CFR 20.1 - 20.601.

The annual inhalation doses to plant personnel due to the airborne radioisotopes in each of the above mentioned regions are presented in table 12.2-3.

TABLE 12.2-1

PEAK AIRBORNE RADIOISOTOPIC CONCENTRATIONS IN THE DIFFERENT REGIONS OF THE PLANT

			Containment				Auxiliary Building					
	Turbine Building		At Full Power		During Refueling or Shutdown		Waste Gas Processing Region		Waste Monitor Tank Rooms		Radwaste Area Excluding Waste Processing Area and Monitor Tank Rooms	
Isotope	Concentration in µCi/cm³	MPC[a] Air in µCi/cm³ (40 h/wk)	Concentration in µCi/cm³[b]	MPC Air in µCi/cm³ (4 h/wk)	Concentration in µCi/cm³	MPC Air in µCi/cm³ (40 h/wk)	Concentration in µCi/cm³	MPC Air in µCi/cm³ (2 h/wk)	Concentration in µCi/cm³	MPC Air in µCi/cm³ (2 h/wk)	Concentration in µCi/cm³	MPC Air in µCi/cm³ (40 h/wk)
Kr-85	9.07×10^{-9}	1×10^{-5}	2.49×10^{-3}	1×10^{-4}	2.56×10^{-9}	1×10^{-5}	4.09×10^{-6}	2×10^{-4}	2.68×10^{-7}	2×10^{-4}	1.24×10^{-9}	1×10^{-5}
Kr-85m	4.01×10^{-8}	6×10^{-6}	1.20×10^{-7}	6×10^{-5}	5.01×10^{-12}	6×10^{-6}	6.24×10^{-6}	1.2×10^{-4}	3.86×10^{-6}	1.2×10^{-4}	1.78×10^{-8}	6×10^{-6}
Kr-87	9.48×10^{-9}	1×10^{-6}	2.73×10^{-8}	1×10^{-5}	3.79×10^{-18}	1×10^{-6}	5.75×10^{-9}	2×10^{-5}	2.48×10^{-6}	2×10^{-5}	1.15×10^{-6}	1×10^{-6}
Kr-88	4.88×10^{-8}	1×10^{-6}	1.48×10^{-7}	1×10^{-5}	8.95×10^{-13}	1×10^{-6}	6.24×10^{-8}	2×10^{-5}	7.33×10^{-6}	2×10^{-5}	3.40×10^{-8}	1×10^{-6}
Xe-133	4.81×10^{-8}	1×10^{-5}	1.32×10^{-5}	1×10^{-4}	1.21×10^{-7}	1×10^{-5}	1.83×10^{-5}	2×10^{-4}	1.55×10^{-4}	2×10^{-4}	7.12×10^{-7}	1×10^{-5}
Xe-133m	8.36×10^{-8}	1×10^{-5}	2.25×10^{-7}	1×10^{-4}	8.50×10^{-10}	1×10^{-5}	2.11×10^{-7}	2×10^{-4}	2.90×10^{-6}	2×10^{-4}	1.36×10^{-8}	1×10^{-5}
Xe-135	1.75×10^{-7}	4×10^{-6}	4.97×10^{-7}	4×10^{-5}	1.49×10^{-10}	4×10^{-6}	5.29×10^{-7}	8×10^{-5}	1.07×10^{-5}	8×10^{-5}	5.02×10^{-8}	4×10^{-6}
Xe-135m	3.31×10^{-10}	1×10^{-6}	1.15×10^{-9}	1×10^{-5}	4.69×10^{-38}	1×10^{-6}		2×10^{-5}	3.87×10^{-7}	2×10^{-5}	1.78×10^{-8}	1×10^{-6}
Xe-138	1.02×10^{-8}	1×10^{-6}	2.94×10^{-9}	1×10^{-5}	5.86×10^{-33}	1×10^{-6}		2×10^{-5}	1.29×10^{-6}	2×10^{-5}	6.15×10^{-9}	1×10^{-6}
I-131	2.25×10^{-10}	9×10^{-9}	4.23×10^{-9}	9×10^{-8}	1.97×10^{-14}	9×10^{-9}	1.1×10^{-10}	1.8×10^{-7}	2.44×10^{-10}	1.8×10^{-7}	1.86×10^{-11}	9×10^{-9}
I-132	1.22×10^{-11}	2×10^{-7}	3.11×10^{-10}	2×10^{-6}	6.72×10^{-18}	2×10^{-7}	4.56×10^{-13}	4×10^{-6}	8.77×10^{-11}	4×10^{-6}	4.53×10^{-12}	2×10^{-7}
I-133	1.60×10^{-10}	3×10^{-8}	4.90×10^{-9}	3×10^{-7}	2.41×10^{-15}	3×10^{-8}	1.73×10^{-11}	6×10^{-7}	3.92×10^{-10}	6×10^{-7}	1.98×10^{-11}	3×10^{-8}
I-134	4.57×10^{-13}	5×10^{-7}	9.03×10^{-11}	5×10^{-6}	1.23×10^{-23}	5×10^{-7}	9.60×10^{-14}	1×10^{-5}	5.83×10^{-11}	1×10^{-5}	2.82×10^{-12}	5×10^{-7}
I-135	3.17×10^{-11}	1×10^{-7}	1.60×10^{-9}	1×10^{-6}	1.50×10^{-16}	1×10^{-7}	3.36×10^{-12}	2×10^{-6}	2.21×10^{-10}	2×10^{-6}	1.01×10^{-11}	1×10^{-7}

a. For 40 h/wk, concentration limits are as stated in Table I, Appendix B to 10 CFR 20.1-20.601. For periods of less than 40 h/wk, concentration limits have been adjusted for the shorter period.

b. The containment concentrations are maximum equilibrium levels after about 50 hours of minipurge system operation.

TABLE 12.2-2 (SHEET 1 OF 3)

ASSUMPTIONS USED TO ESTIMATE PEAK AIRBORNE CONCENTRATIONS AND INHALATION DOSES

Leak Rates (lb/day)	
Steam generator tube leak (primary coolant)	166.9
Leak into containment (primary coolant)	40
Leak into auxiliary building (primary coolant)	166.9
Steam leak into turbine building	6×10^4
Liquid leak into turbine building	1.5×10^5
Leak from waste gas processing system	100 sf^3/year
Partition Factors or Ratio of Liquid Activity to Airborne Activity (iodines)	
Steam generator	100
Air ejector	10,000
Liquid leakage to turbine building	100
Liquid leakage to auxiliary building	100
Containment building, primary coolant leakage	100
Leakage from waste gas processing system (partition in the volume control tank)	100
Ventilation (ft^3/min)	
Exhaust rate from turbine building	5000
Flowrate for recirculation in the turbine building	11,500
Containment purge rate (Main/Mini)	25,000/2500
Preacess filter system in containment, flowrate for recirculation	20,000
Exhaust rate from waste gas processing region of radwaste area	3500
Exhaust rate from waste monitor tank rooms (region of radwaste area containing waste holdup, floor drain, and waste monitor tanks)	660
Exhaust rate from radwaste area (excluding the waste gas processing region and waste monitor tank rooms)	5.18×10^4

TABLE 12.2-2 (SHEET 2 OF 3)

Filter Efficiency (percentage)	
Halogen recirculation filter efficiency in containment	90
Halogen recirculation filter efficiency in the turbine building	90
Occupancy in the Regions (h/week; weeks/year)	
Turbine building	40; 40[a]
Waste gas processing area	2; 40[a]
Waste monitor tank rooms	2; 40[a]
Radwaste area (excluding waste gas processing region and waste monitor tank rooms)	40; 40[a]
Containment during refueling or shutdown purge	40; 4
Volumes of the Regions (ft^3)	
Turbine building	4.25×10^6
Containment	2.05×10^6
Waste gas processing region	38,000
Waste monitor tank rooms	8400
Radwaste area (excluding waste gas processing region and waste monitor tank rooms)	6.2×10^5

TABLE 12.2-2 (SHEET 3 OF 3)

Other Factors	
Fuel defects (percent)	0.25
Plant load factor (percent)	80
Duration of the containment purge, hot shutdown (h)	24
Duration of the preaccess filter operation (h)	40[b]
Duration of the containment refueling shutdown purge (h)	8

a. Full occupancy a year means 50 weeks/year for plant load factor 1.0; 40 weeks/year for plant load factor 0.8.

b. For refueling or shutdown purge, recirculation through preaccess filters will be for a total of 16 + 8 = 24 h. Purging at power is done by a constant 2500-ft^3/min flow through the minipurge system.

12.3 HEALTH PHYSICS PROGRAM

12.3.1 PROGRAM OBJECTIVES

12.3.1.1 Objectives

It is the objective of the Farley Nuclear Plant health physics program to provide effective radiation protection for plant personnel and visitors during operations, maintenance, refueling, and emergencies, and further to keep exposures as low as reasonably achievable (ALARA). The health physics group is responsible for developing and administering such a program consistent with 10 CFR 20, Standards for Protection Against Radiation, paragraph 20.1101.

12.3.1.2 Organization

The health physics group consists of a health physics superintendent, health physics sector supervisor, radwaste supervisor, plant health physicist, health physics foremen, waste and decon foremen, health physics technicians, radiation detection personnel, and nuclear operatives. Overall responsibility for plant operation lies with the general manager-nuclear plant, but the responsibility for health physics operations is delegated to the health physics superintendent. (See figure 13.1-6.)

The health physics superintendent reports to the technical manager and is responsible for keeping him informed at all times of radiation hazards and conditions related to potential exposure, contamination of plant equipment, or contamination of site and environs. The organization structure provides for direct communication (symbolized by a dashed line in figure 13.1-6) between the health physics superintendent and the assistant general manager-plant operations regarding radiation protection matters when deemed desirable by the health physics superintendent. As the administrator of the health physics program, the responsibilities of the health physics superintendent include:

A. Training and supervising the health physics technicians.

B. Planning and scheduling health physics coverage and surveillance activities.

C. Establishing and maintaining data on plant radiation and contamination levels, personnel exposures, and work restrictions.

D. Writing and maintaining a current health physics manual which incorporates the provisions of Regulatory Guides 8.8, Revision 3, and 8.10, Revision 1 (this should include radiation protection reviews of appropriate design changes).

E. Ensuring that plant operations comply with 10 CFR 20.1001 - 20.2401.

F. Advising the emergency director during emergencies involving radiological hazards.

G. Advising other group supervisors with regard to dose equalization among their personnel. Personnel will be rotated in so far as practical for uniformity of occupational radiation exposure within each group.

H. Managing the shipment and disposal of all solid radwaste.

I. Supervising the ALARA program.

To carry out the responsibilities of the health physics superintendent, the health physics group is organized to:

A. Perform radiation monitoring for plant operations and maintenance activities as required and maintain records of all surveys performed.

B. Establish and maintain a radiological surveillance program to collect and document data concerning radiation and contamination levels throughout the plant and on the plant site.

C. Make plant personnel aware of radiological conditions by posting areas throughout the plant based on radiation and contamination levels.

D. Provide and maintain protective clothing and respiratory equipment for plant operation and maintenance and instruct plant personnel in their use.

Clothing issue rooms, a calibration lab, a respirator issue room, a decontamination room, a drumming room, and a nuclear laundry are located in the auxiliary building radiation controlled area at el 155 ft. The health physics office is located near the boundary between the clean area and the radiation controlled area so that health physics services and decontamination may be conveniently provided to those who enter or leave this area. Personnel decontamination can be performed in the hot toilet rooms which are conveniently located adjacent to the health physics office. Radiation controlled area entry is through an administratively controlled one-way door. Prior to leaving the radiation controlled area, one passes through a portal monitor or uses friskers near the health physics office.

12.3.2.2 Shielding and Handling Methods

Lead in various forms, such as bricks, blankets, or sheets will be available for use as portable shielding. (A safety evaluation checklist must be completed before shielding can be applied to any safety-related equipment or systems.)

A radiation work permit will be employed as the principal means of ensuring that proper precautions are taken and that adequate planning is effected before work is performed in any area that presents a real or potential radiological hazard. Prior to a worker's entry into an area in which the radiological conditions are unknown, a survey is made and a radiation work permit completed which lists the radiation protection requirements for the particular work to be accomplished.

Other handling methods and procedures for keeping external and internal exposures ALARA are discussed in subsections 12.1.5 and 12.2.5.

12.3.2.3 Respiratory Equipment

Respiratory equipment will be available for use in areas in which airborne radioactive material exceeds those concentrations given in Table 1, Appendix B to 10 CFR 20.1001 - 20.2401. Typical respiratory devices which will be made available at the plant include the following:

A. Full-face masks with high efficiency particulate and charcoal filters.

B. Full-face masks with air line.

C. Hoods and suits with air line.

D. Full-face masks with self-contained breathing apparatus.

The respiratory protection program is designed to comply with 10 CFR 20.1701 - 20.1704. Respiratory equipment is selected and protection factors are assigned in accordance with 10 CFR 20.1001 - 20.2401, Appendix A. An exemption from 10 CFR 20 which allows the use of a protection factor for radioiodine has been granted to Farley Nuclear Plant by the NRC. Any changes to the October 23, 1984, NRC exemption will be incorporated into the program.

12.3.2.4 Protective Clothing

Protective clothing will be required in contaminated areas. Typical protective clothing that will be made available at the plant is listed below:

A. Coveralls - cloth and disposable paper type.

B. Laboratory coats.

C. Plastic suits.

D. Canvas caps - cloth.

E. Hoods - canvas.

F. Shoe covers - plastic and rubber.

G. Booties - cloth.

H. Gloves - cotton, plastic, and rubber.

12.3.2.5 Portable Instrumentation

The majority of the portable health physics instrumentation will be located in the auxiliary building near the health physics office or the health physics calibration laboratory. For purposes of emergency monitoring, instruments will be kept at the central security building, emergency operations facility, and Southeast Alabama Medical Center. A listing and description of some of the portable health physics instruments are given in table 12.3-1.

The Health Physics group will be responsible for writing and implementing procedures for the use and calibration of this equipment. Detailed records on the maintenance and calibration of this instrumentation will be maintained at the plant. Calibration will be performed using sources of known strength purchased from the National Institute of Standards and Technology (NIST) or other reputable vendors and/or using reference instruments having calibrations traceable to the NIST. In addition, reputable vendors will be used to calibrate and perform maintenance on some of the portable instruments. Vendors will implement their own calibration procedures but are subject to Southern Nuclear Operating Company (SNC) quality assurance requirements. Calibrations and preventive maintenance on portable health physics instrumentation will be performed semiannually or when required. Calibration will also be required after a piece of equipment has undergone repair work which affects calibration.

12.3.2.6 Laboratory Equipment

Major fixed laboratory instrumentation will generally be located at the radiation controlled area exit and the Health Physics counting room, but use is not limited to these areas. A listing of typical equipment, including location and description, is given in table 12.3-1.

The Health Physics group will be responsible for writing and implementing procedures for the use and calibration of equipment. Detailed records on the calibration of this instrumentation will be maintained at the plant. Calibration will be performed using sources of known strength purchased from the NIST or other reputable vendors and/or using reference instruments having calibration traceable to the NIST. In addition, reputable vendors may be used to calibrate and perform maintenance on fixed laboratory instrumentation. Vendors will implement their own calibration procedures but are subject to SNC quality assurance requirements. Calibration will also be required after a piece of equipment has undergone repair work which affects calibration. The equipment and instrumentation listed in table 12.3-1 are typical of the devices which will be purchased.

12.3.3 PERSONNEL DOSIMETRY

Where applicable, the personnel dosimetry program will be developed in accordance with Regulatory Guide 8.4, Revision 0, Direct Reading and Indirect Reading Pocket Dosimeters, and Regulatory Guide 8.13, Revision 3, Instructions Concerning Prenatal Radiation Exposure.

12.3.3.1 External Dosimetry

Plant employees, visitors, support personnel, and construction workers will be required to wear one or more personnel dosimeters when they enter the radiation control area if they are likely to receive, in 1 calendar year, from sources external to the body, a dose in excess of 10 percent of the limits in 10 CFR 20.1201(a). A third party may be used or a complete in-house program may be implemented for processing thermoluminescent dosimeter (TLD) badges.

Monitoring for contractor personnel during outages and other peak activity events may be accomplished by placing TLD badges at alternate locations to minimize congestion.

Personnel dosimetry used at the plant will include a TLD and either a digital alarming dosimeter or pocket ion chamber. The TLD must be sensitive to beta-gamma radiation and the dosimeter must be sensitive to gamma radiation. The dose received on dosimeters and TLDs will be tracked by plant personnel. Extremity dosimeters will be issued on a case-by-case basis, and neutron dosimetry will be accomplished by setting dose rates and time keeping, which must be performed by a qualified individual, or by issuing neutron dosimetry.

12.3.3.2 Internal Dosimetry

Whole body counting and urinalysis will be used to supplement the dosimetry program. If internal dose assessment is deemed necessary, calculations will meet the intent of Regulatory Guide 8.34.

12.3.3.3 Records

Exposure data of all personnel will be collected and recorded on form NRC-5, Current Occupational External Radiation Exposure, or the equivalent. Occupational exposures incurred by individuals prior to working at the Farley Nuclear Plant, bioassay data, and whole body counting data will be summarized on form NRC-4, Occupational External Radiation Exposure History, or the equivalent. Records retained on form NRC-4 or its equivalent will be retained until the license is terminated. The records used in preparing form NRC-4 or its equivalent will be kept for 3 years, after which time they may be disposed of. Exposure data recorded on form NRC-5 or its equivalent will be retained until the license is terminated.

12.4 RADIOACTIVE MATERIALS SAFETY

12.4.1 MATERIALS SAFETY PROGRAM

Sealed and unsealed sources may be used at the Farley Nuclear Plant to calibrate reactor excore detectors, process and effluent radiation monitoring systems, area radiation monitoring systems, portable survey instruments, and fixed laboratory equipment. Storage and handling of these sources will be in accordance with 10 CFR 20, 30, 40, and 70, with the health physics group being responsible for the control of such sources. A Nuclear Regulatory Commission license will be obtained for byproduct, source, or special nuclear material, as appropriate, prior to the procurement of radioactive sources.

High level sources such as those listed in table 12.4-1 will normally be housed in lockable, shielded containers and stored in the health physics calibration laboratory located at el 155 ft of the auxiliary building. This room will be locked when not in use. Low level sources that are primarily used for calibration and quality control checks of fixed laboratory instruments and for portable survey instrument check sources will be stored in locked cabinets on el 139 ft in the radiochemistry laboratory and counting room, on el 155 ft in the health physics disrobe area or at the emergency operations facility.

Other information pertinent to the handling and use of radioactive sources is contained in subsections 12.3.1.4, 12.3.2.2, and 12.3.3.

12.4.2 FACILITIES AND EQUIPMENT

A discussion of the facilities utilized by the health physics groups is given in subsection 12.3.2.1. The radiochemistry laboratory, where unsealed radioactive sources would normally be stored, is equipped with two exhaust hoods that exhaust to the plant vent. The sampling room and gas analysis room are also equipped with one such hood each.

A discussion of portable health physics instrumentation is given in subsection 12.3.2.5, with a listing and description of

each instrument given in table 12.3-1. A discussion of fixed laboratory instrumentation is given in paragraph 12.3.2.6, with a listing and description of each instrument given in table 12.3-2.

12.4.3 PERSONNEL AND PROCEDURES

The health physics superintendent and the health physics sector supervisor are the key personnel responsible for handling and monitoring radioactive materials at the plant. The qualifications of the group supervisor are given in paragraph 13.1.3.1.

The qualification requirements for health physics foremen, who direct the work activities of the health physics technicians, meet or exceed the minimum requirements set forth in American National Standards Institute (ANSI) N18.1-1971. The minimum qualification requirements for health physics foremen are given in paragraph 13.1.3.1.

The qualification requirements for health physics technicians, who handle and monitor radioactive materials under the direction of the sector supervisors or foremen, meet or exceed the minimum requirements set forth in ANSI N18.1-1971. The minimum qualification requirements for health physics technicians are given in paragraph 13.1.3.1.

Procedures have been developed by the health physics group to cover the receipt, storage, and use of radioactive sources. These procedures are discussed in the group training sessions to ensure that all technicians who are required to handle radioactive sources are thoroughly familiar with the procedures.

12.4.4 REQUIRED MATERIALS

A list of sources that are likely to be purchased is given in table 12.4-1. This table includes a listing of isotope, quantity, form, and use for byproduct, source, and special nuclear materials that exceed the amounts of Table 1 of Regulatory Guide 1.70.3. At the time of procurement of radioactive sources that exceed the quantities in Table 1 of Regulatory Guide 1.70.3, an amendment will be made to table 12.4-1, if necessary.

13.0 CONDUCT OF OPERATIONS

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Drawings and documents covering the portion of FNP developed by Bechtel are submitted to Westinghouse for comment when they concern the nuclear steam supply system or result from Westinghouse criteria and are subject to final acceptance by the nuclear generation organization.

Final acceptance by the nuclear generation organization of design concepts, documents, and equipment suppliers is based on recommendations from the responsible designers and consultation with the APC PGS, Purchasing, and Quality Assurance Departments, as appropriate.

Construction and modifications to the plant are the responsibility of SNC through the general contractor, FCII. Onsite construction activities are under the supervision of the FCII project manager and are monitored by SNC's site nuclear project director and quality assurance personnel.

The interrelationships and interfaces that existed between the various organizations during startup and preoperational testing are described in subsection 14.2.2.

The responsibility for ensuring that equipment suppliers and contractors conform to approved specifications is retained by the design organizations, although all equipment is procured by SNC. Conformance is verified through implementation of the quality assurance program described in chapter 17.

13.1.1.3 Licensee's Technical Staff

An SNC technical staff specifically responsible for supporting the operation of the FNP is maintained at the corporate office. The general office support staff is organized as a staff function to support the plant operation and as a line function to direct the operation of the plant. As shown on figures 13.1-4, 13.1-5, and 13.1-7, the president and CEO, executive vice president, and vice president provide line management direction for the operation of the plant. The plant staff personnel are highly qualified to perform their responsibilities and are expected to require a minimal support effort. The support staff consists of the general manager-nuclear support and his staff. The FNP engineer in charge is the general manager-nuclear support. The general manager-nuclear support meets the requirements of ANSI N18.1-1971.

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13.1.2 OPERATING ORGANIZATION

The description of the operating organization in this section is for a two-unit operation. Any differences between one- and two-unit operations will be noted.

13.1.2.1 Plant Organization

The operation and maintenance of the FNP is the responsibility of the SNC general manager-nuclear plant, who reports to the vice president in the general office.

Seven principal managers report to the general manager-nuclear plant through two assistant general managers. They are the operations manager, maintenance manager, administrative manager, technical manager, systems performance manager, plant modifications manager, and training manager. The chemistry and environmental supervisor, health physics supervisor, shift supervisor, systems performance supervisor, mechanical maintenance supervisor, electrical maintenance supervisor, instrumentation and control supervisor, security supervisor, planning supervisor, document control supervisor, technical supervisor, quality control engineer, materials supervisor, planning and control supervisor, computer services supervisor, and building and grounds supervisor will provide the line management for all plant operations and will report to the seven managers.

Administrative, technical, training, operations, maintenance, performance and planning, and plant modifications personnel make up the plant operating staff. The plant staff consists of approximately 800 personnel for two-unit operation. The plant operating staff is organized as indicated on figure 13.1-6.

13.1.2.2 Personnel Functions, Responsibilities, and Authorities

13.1.2.2.1 SNC General Manager-Nuclear Plant and Two Assistant General Managers

Farley Nuclear Plant personnel have a combination of education, experience, health, and skill commensurate with their level of responsibility. These qualities provide reasonable assurance that decisions and actions during all

normal and abnormal conditions are such that the plant is operated in a safe and efficient manner.

The overall responsibility for plant operations rests with the SNC general manager-nuclear plant. The general manager-nuclear plant is responsible for plant operation in a safe, reliable, and efficient manner by ensuring compliance with all requirements of the operating license. The general manager-nuclear plant is the chairman of the Plant Operations Review Committee (PORC). A complete description of the committee is contained in section 13.4

The general manager-nuclear plant has overall responsibility for the startup test program at FNP, as described in chapter 14. He is responsible for approving all Phase II and Phase III test procedures, major test procedure modifications, completed tests and test data, and test data deficiency resolutions.

The assistant general manager-plant operations assists the general manager-nuclear plant with the responsibility for plant operation and normally assumes full responsibility in his absence. Some of his duties include: organizing and conducting scheduled meetings between plant group managers for information and review purposes; working with the operations manager to ensure that the plant is operated within the limits of the operating license and to schedule outage activities; working with the maintenance manager to ensure that proper preventive and corrective maintenance on all equipment is being performed; and working with the technical manager to ensure that the required chemical analysis, water quality control, radiation surveys, and general health physics aspects of the plant are being performed. The assistant general manager-plant operations is vice chairman of the PORC.

The assistant general manager-plant support assists the general manager-nuclear plant with the responsibility for plant support groups and may assume full responsibility in his absence provided this person meets the requirements of ANSI N18.1-1971 Section 4.2.1, FSAR paragraph 13.1.3.1.1, and has completed emergency director training. Some of his duties include: working with the training manager to ensure that planned training and retraining programs will meet the requirements of SNC and the Nuclear Regulatory Commission (NRC); working with the systems performance manager to ensure that the nuclear steam supply system and the balance of plant systems are being utilized and operated in a safe and efficient mode; working with the administrative manager to ensure efficient operation of the storeroom, to ensure that the plant security plan is enforced, and to ensure efficient operation of the records management program; working with the plant modifications manager to ensure that all plant design changes are properly packaged, scheduled, and coordinated;

and working with the plant quality assurance engineer and group supervisors to ensure compliance with the operations quality assurance program. The assistant general manager-plant support is a member of the PORC.

13.1.2.2.2 Operations Group

The operations manager, who reports to the assistant general manager-plant operations, is responsible for the management and coordination of operations activities of the plant. Reporting to him are the shift supervisors, the planning and control supervisor, and the building and grounds supervisor. The operations manager serves as the emergency director and is a member of the PORC. He performs outage management functions as directed by the assistant general manager-plant operations.

The operations manager is responsible for the day-to-day operation of the plant in a safe and efficient manner in compliance with the operating license. The operations manager is responsible for developing normal, emergency, and refueling operating procedures, department training, and retraining programs.

The shift supervisor is in direct charge of the plant, including startup, power operations, and shutdown. He will initiate immediate action in the event of an abnormal situation to avoid violation of the operating license, to avert possible injury or undue radiation exposure of personnel, or to prevent damage to plant equipment.

The shift supervisor has the responsibility of supervising the actions of the station operators (plant operators, systems operators, and switchboard operator) to ensure safe and prudent operation of the facility. He will initiate immediate corrective action in any abnormal situation until assistance, if required, arrives.

The shift foreman-operating reports to the shift supervisor and directly supervises the systems operators.

The plant operators, who are supervised by the shift supervisor, control and direct the operation of their assigned unit according to detailed procedures. Normally, one plant operator will be assigned to each unit's main control center.

The shift foreman-inspecting reports to the shift supervisor and assists him in equipment systems control. The shift technical advisor advises the shift supervisor during emergency conditions and has no command and control functions.

stenographers. The staff assistant advises and assists in administrative duties and researches employee grievances.

The clerks' work is directed by the staff assistant. They handle management and clerical services for the plant such as payrolls, applications for employment, reports, and other duties as assigned.

13.1.2.2.8 Administrative Manager

The administrative manager reports to the assistant general manager-plant support and is responsible for the storeroom, document control, and security groups.

The materials supervisor is responsible for the safe and efficient operation of the plant warehouse and reports to the administrative manager.

The security group, under the supervision of the security supervisor, performs the security functions for the plant. The security supervisor, who reports directly to the administrative manager, will enforce the plant security plan through the plant guards. This plan is discussed in section 13.7.

The document control group, under the supervision of the document control supervisor, is responsible for maintaining records and documents.

13.1.2.2.9 Training Manager

The training manager, under the direction of the assistant general manager-plant support, will organize and direct the overall FNP training program. He will work with and through the various plant group supervisors to coordinate group specialty training and retraining programs and will work for the assistant general manager-plant support to implement the general employee training program. Sufficient organizational freedom exists for the training staff to ensure their independence from operating pressures. The plant training program is discussed in section 13.2.

13.1.2.2.10 Systems Performance Manager

The systems performance manager reports to the assistant general manager-plant support and is responsible for the quality control, computer services, and systems performance groups.

The objective of the quality control group is to verify the adequacy and effectiveness of quality control inspections. This

includes engineering evaluation review and approval of inspector certification, review of problem reports, cross-disciplinary review of procedures, and performance of inspections as required.

The objective of the systems performance group is to identify and resolve existing or potential plant operational problems having significant safety consequences and to determine areas where changes in design, procedures, or operating practices could improve operational reliability or more effectively mitigate consequences of postulated accidents, malfunctions, and errors.

The objective of the computer services group is to ensure that computer based information systems utilized throughout FNP are installed and maintained consistent with system configuration control, quality assurance, and system reliability requirements.

13.1.2.2.11 Plant Modifications Manager

The plant modifications manager reports to the assistant general manager-plant support and is responsible for coordination of plant design change implementation.

The objective of the plant modifications group is to establish policies and schedules for, and plan and coordinate the implementation of, plant design changes.

13.1.2.2.12 Supervisory Succession

The general manager-nuclear plant is responsible for the safe, reliable, and efficient operation of the FNP. In the absence of the general manager-nuclear plant, the following members of the plant staff, in the order listed below, will assume this responsibility:

A. Assistant general manager-plant operations.

B. Assistant general manager-plant support.[a]

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a. The person filling this position may act as the general manager-nuclear plant provided this person meets the requirements of ANSI N18.1-1971 Section 4.2.1, FSAR paragraph 13.1.3.1.1, and has completed emergency director training.

C. Operations manager.

D. Other designated manager.

E. Shift supervisor.

13.1.2.3 Shift Crew Composition

The FNP will be operated from one central control room. A shift supervisor will be directly responsible for the safe and efficient operation of each unit.

The normal shift complement for two-unit operation is as follows:

	Unit 1	Unit 2	Shared
Shift supervisor (SRO)	1	1	-
Shift foreman-operating (SRO)[a]	1	1	-
Shift foreman-inspecting[b]	1	1	-
Shift technical advisor[c]	1	1	-
Plant operator (RO)	2	2	-
Systems operator	2	2	4
Switchboard operator	-	-	1

Although the systems operator will not be required to hold a reactor operator's license, he will be required to participate in regularly scheduled operator training programs and actively pursue a reactor operator's license. He will be trained and qualified in the operation of all auxiliary equipment and will work under the direction of a licensed operator. An individual qualified in radiation protection procedures shall be onsite when fuel is in the reactor.

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a. The shift foreman-operating may be the same individual for both units.

b. The shift foreman-inspecting may be the same individual for both units and if qualified may be the shift technical advisor.

c. The shift technical advisor may be the same individual for both units and may also be the shift foreman-inspecting.

13.1.3 QUALIFICATION REQUIREMENTS FOR NUCLEAR FACILITY PERSONNEL

13.1.3.1 Minimum Qualification Requirements for All Plant Personnel

The following qualification requirements, used as a general guideline for personnel assignments, meet or exceed the minimum requirements set forth in the American National Standards Institute document N18.1-1971, Standard for Selection and Training of Personnel for Nuclear Power Plants.

13.1.3.1.1 General Manager-Nuclear Plant

A. A baccalaureate or higher degree in an engineering or scientific field.

B. A total of 10 years of power plant experience, of which a minimum of 3 years is nuclear power plant experience. Four of the remaining 7 years may be fulfilled by academic training generally associated with power production on a one-for-one basis.

C. Have acquired experience and training required for a senior operator's license.

13.1.3.1.2 Assistant General Manager-Plant Operations

A. A baccalaureate or higher degree in an engineering or scientific field.

B. A total of 10 years of power plant experience, of which a minimum of 3 years is nuclear power plant experience. Four of the remaining 7 years may be fulfilled by academic training generally associated with power production on a one-for-one basis.

C. USNRC senior reactor operator's license.

13.1.3.1.3 Assistant General Manager-Plant Support

A. A baccalaureate or higher degree.

B. A total of 10 years of power plant experience of which a minimum of 3 years is nuclear power plant

D. A maximum of 3 of this 5 years of experience may be fulfilled by a related technical or academic training.

13.1.3.1.22 Chemistry Technician and Health Physics Technician

A. Two years of experience in specialty (chemistry or health physics).

B. One year of training in the areas of chemistry, radiochemistry, or radiation protection principles or successful completion of the FNP chemistry and health physics technician course.

13.1.3.1.23 Technical Supervisor

A. A baccalaureate degree in engineering or the physical sciences.

B. A total of 8 years of responsible experience in power plant design or operation, of which a minimum of 1 year is nuclear power plant experience.

C. A maximum of 4 years of the remaining 7 years of experience may be fulfilled by satisfactory completion of academic training.

13.1.3.1.24 Reactor Engineer

A. A baccalaureate degree in engineering or the physical sciences.

B. Two years of experience in such areas as reactor physics, core measurements, core heat transfer, and core physics testing program.

13.1.3.1.25 Administrative Manager

A. High school education or equivalent.

B. A total of 8 years of responsible experience, at least 1 year of which should be nuclear power plant experience.

C. A maximum of 4 of this 8 years of experience may be fulfilled by related technical or academic training.

13.1.3.1.26 Systems Performance Manager

A. A baccalaureate degree in engineering or the physical sciences.

B. A total of 8 years of responsible experience in power plant design or operation, of which a minimum of 1 year is nuclear power plant experience.

C. A maximum of 4 of this 8 years of experience may be fulfilled by related technical or academic training.

13.1.3.1.27 Planning Supervisor

A. A high school education or equivalent.

B. A total of 8 years of responsible experience, at least 1 year of which should be nuclear power plant experience.

C. A maximum of 4 of this 8 years of experience may be fulfilled by related technical or academic training.

13.1.3.1.28 Systems Performance Supervisor

A. A high school education or equivalent.

B. A total of 8 years of responsible experience in power plant design or operation, of which a minimum of 1 year involves nuclear power plant design or operation.

C. A maximum of 4 of this 8 years of experience may be fulfilled by related technical or academic training.

13.1.3.1.29 Computer Services Supervisor

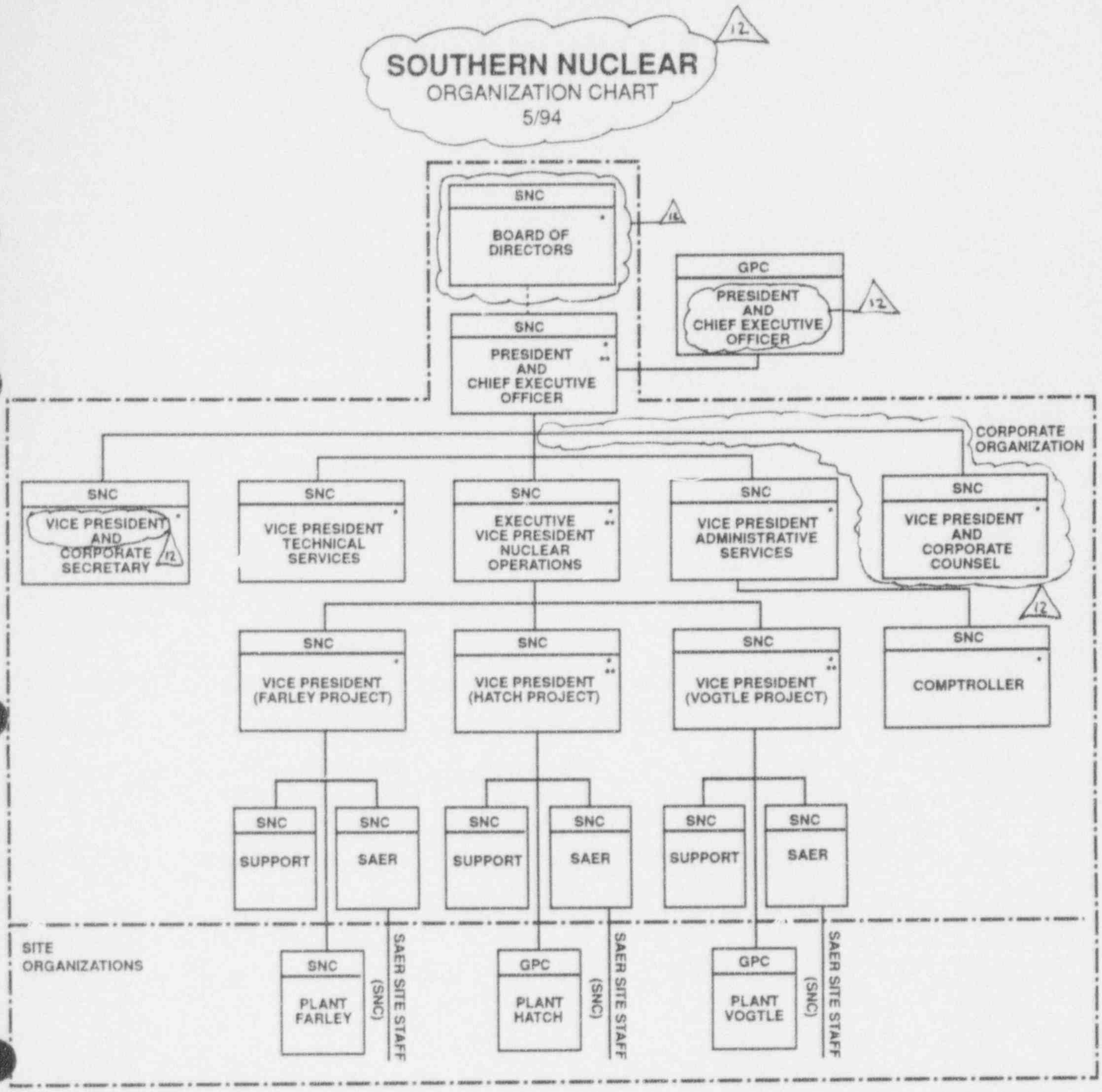
A. A high school education or equivalent

B. A total of 5 years of responsible experience, at least 1 of which should be nuclear power plant experience.

C. A maximum of 4 years of this 5 years of experience may be fulfilled by related technical or academic training.

13.1.3.1.30 Plant Modifications Manager

A. A high school education or equivalent.



GPC - Georgia Power Company
SNC - Southern Nuclear Operating Company

* Southern Nuclear Officers
** Georgia Power Company Officers

------- SNC Matters Only

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JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	SOUTHERN NUCLEAR OPERATING COMPANY
	FIGURE 13.1-7

13.2 TRAINING PROGRAM

Responsibility for administration of the overall training program for FNP rests with the general manager. Training programs administered at the plant site are supervised by the training manager. He is responsible to the assistant general manager-support for ensuring that operations training, technical/maintenance, and general employee training maintain the educational level adequate for safe and efficient operation of the plant.

13.2.1 PROGRAM CONTENT FOR OPERATIONS PERSONNEL

13.2.1.1 Operator License Training Program

The Operator License Training Program contains two phases and is presented on a level required for senior reactor operator.

Reactor operator (RO) license candidates and senior reactor operator (SRO) license candidates, with or without engineering degrees, attend this program. The Operator License Training Program satisfies the training requirement stipulated in ANSI N18.1-1971.

The two phases of operator license training are as follows:

A. Fundamental training

   This phase of the program provides license candidates training in the areas of math, physics, chemistry, electrical theory, heat transfer/fluid flow, thermodynamics, reactor theory, and radiological protection and is approximately 15 weeks in duration.

B. Specialized training

   This phase is approximately 39 weeks in duration and is subdivided as follows:

   Approximately 14 weeks of plant fluid, electrical, and instrumentation/control systems including fuel handling.

   Approximately 4 weeks of procedures, technical specifications, and integrated plant operations including mitigating core damage and transient/accident analysis.

   Approximately 8 weeks of simulator training on FNP's Unit 1 plant-referenced simulator. The training

includes reactor startups, shutdowns, and power operation during normal, transient, and accident conditions. Plant-specific procedures are used extensively during this portion of training.

Approximately 13 weeks of on-the-job training. During this period the student is assigned to various licensed operator crew positions and is required to perform day-to-day routine evolutions including a minimum of five reactivity manipulations while under the supervision of an RO or SRO licensed individual.

13.2.1.2 License Retraining Program

The License Retraining Program is designed to maintain a base level of knowledge through the use of an INPO accredited "Systems Approach To Training." Farley's License Retraining Program received accreditation in December 1984 and is reaccredited quadrennially.

The program is divided into two portions: classroom lecture series and simulator. It is conducted on an SRO level.

The scope of the program is described in the applicable Training Center Procedures. Examinations are conducted to satisfy requirements of NUREG 1021.

Each individual must demonstrate that he has adequately learned the material presented by scoring 80 percent or greater on an examination. The examinations will be similar in scope, content, and format to license exams given by the Nuclear Regulatory Commission. If an individual scores less than 80 percent on an exam following a lecture series, and this exam is considered part of the annual exam, the individual will be removed from licensed duties and placed in a remedial program to correct his deficiencies.

The individual will be reexamined and must successfully meet the previously described passing criteria prior to resuming licensed duties.

If the exam is not part of the annual exam and an individual scores less than 80 percent, he will be retested following further review of the material presented.

The simulator retraining course is designed to reacquaint the licensed operator with the dynamic response of the individual plant systems as well as the plant as a whole. The student is

involved with hands-on reactivity and control manipulations and the use of annunciator response, normal, abnormal, and emergency procedures.

Simulator retraining, approximately 1 week long, allows the student to perform a variety of the control and reactivity manipulations listed below:

A. Reactor startup to the point of adding nuclear heat and establishing a heatup rate.

B. Plant shutdown.

C. Manual control of steam generators during startup or shutdown.

D. Boration and/or dilution during power operations.

E. Any power change greater than 10 percent in manual rod control.

F. Loss of coolant including:

   1. Significant steam generator tube leaks.

   2. Inside and outside primary containment.

   3. Large and small, including leak rate determinations.

   4. Saturated reactor coolant response.

G. Loss of instrument air.

H. Loss of electrical and/or degraded power source.

I. Loss of core coolant flow/natural circulation.

J. Loss of condenser vacuum.

K. Loss of service water.

L. Loss-of-shutdown cooling.

M. Loss-of-component cooling water or cooling to an individual component.

N. Loss of normal feedwater or normal feedwater system failure.

O. Loss of all feedwater (normal and emergency).

P. Loss of protective system channel.

Q. Mispositioned control rod or rods (or rod drops).

R. Inability to drive control rods.

S. Emergency boration.

T. Fuel cladding failure or high reactor coolant system activity.

U. Turbine or generator trip.

V. Malfunction of automatic control system(s) which affect reactivity.

W. Malfunction of reactor coolant pressure/volume control system.

X. Reactor trip.

Y. Main steam line break (inside or outside containment).

Z. Nuclear instrument failures.

Integral with the simulator retraining, the trainee is given an annual simulator operational evaluation by the training staff consisting of an overall operational ability demonstration. The trainee is also given an annual operational evaluation of job performance measures. Any deficiencies observed are discussed with the student, and corrective actions are given in a post training critique. The student will be removed from licensed duties if individual deficiencies result in failure of critical tasks during the simulator operational examination or job performance measures.

The student must be remediated and reexamined prior to resuming licensed duties.

13.2.1.3 System Operator Training Program (Nonlicensed Operator)

The System Operator Training Program contains two phases of training: classroom lecture and system qualification requirements. It is approximately 27 weeks in duration.

The two phases of system operator (SO) training are as follows:

A. SO Fundamentals

Approximately 12 weeks of training in the areas of math, physics, thermodynamics, health physics, electrical theory, reactor theory, power plant equipment, communications, fire brigade, and OJT/OJE evaluator training.

B. Plant Systems

Approximately 7 weeks of plant fluid and electrical systems applicable to the duties of the SO. Included in this training are appropriate procedures and instructions on watch standing techniques and fire brigade training.

C. On-The-Job Training

The 8 weeks of training allow the student to complete the appropriate system qualification requirements. He is also required to participate in routine shift activities such as equipment monitoring and observation, log taking, etc.

13.2.1.4 System Operator Retraining Program (Nonlicensed Operator Retraining)

SOs will be given preplanned lectures on a regularly scheduled basis covering the following subjects:

A. Plant systems and operation.

B. Administrative and emergency implementing procedures.

C. Radiation control and safety.

D. Plant emergency plan.

E. Plant security plan.

F. Plant design changes, applicable procedural changes, and plant license changes.

Written examinations shall be given to determine trainees' knowledge of subjects covered in the lecture series. Operational exams will be given consisting of job performance measures. A score of 80 percent or greater must be achieved on the written exam and the JPM exam for successful completion of the retraining program. Systematic observation and evaluation of the performance and competency of personnel shall be conducted by shift supervision.

13.2.2 PROGRAM CONTENT FOR TECHNICAL AND MAINTENANCE PERSONNEL

13.2.2.1 Health Physics Technician Training

Health physics technicians will attend a formal training program. The program may be held onsite or offsite and will be approximately 9 weeks long. Technicians will normally complete

this course during their first year of employment. However, they may be exempted from such training on the basis of previous training or job experience. Program courses will cover mathematics, nuclear physics, principles of radiation detection and protection, responsibilities and duties of health physics group personnel, radiation biology, plant systems, water chemistry, radioactive waste processing, gamma ray spectrometry, and on-the-job training.

13.2.2.2 Chemistry Technical Training

Chemistry technician IIs will attend a formal training program. The program may be held onsite or offsite and will be approximately 21 weeks long. Technician IIs will normally complete this training during the first year of employment. However, they may be exempted from such training on the basis of previous training or job experience. Program courses will cover responsibilities and duties of chemistry and environmental personnel, basic chemistry, corrosion, sampling considerations during normal plant operations and accident conditions, water purification and treatment, sewage treatment, chemistry technical specifications and limits, instrumental analysis and analytical procedures, plant chemistry control problems, group responsibilities required to support emergency activities, and on-the-job training.

13.2.2.3 Instrumentation and Control Training Program

Instrumentation and control group personnel will attend a formal training program approximately 33 weeks in duration. However, individuals may be exempted from such training on the basis of previous training, education, or job experience. Program courses will cover basic electricity and electronics; fundamentals of pressure; temperature; level and flow measurement and control; NSSS instrumentation such as 7300, SSPS, DRPI, rod control, incore and excore; primary and secondary plant systems; and on-the-job training.

13.2.2.4 Mechanical Maintenance Training Program

Mechanical maintenance group personnel will attend a formal training program approximately 21 weeks in duration. However, individuals may be exempted from such training on the basis of previous training, education, or job experience. Program courses will cover piping systems, diesel generators, rotating machinery, lubrication, machinery balancing, vibration and alignment, principles of rigging, hydraulics, primary and secondary plant systems, and on-the-job training.

13.2.2.5 Electrical Maintenance Training Program

Electrical maintenance group personnel will attend a formal training program approximately 27 weeks in duration. However, individuals may be exempted from such training on the basis of previous training, education, or job experience. Program courses will cover basic electricity fundamentals, single and three-phase motors, dc motors, ac and dc circuits, batteries, switchgear and protective devices, primary and secondary plant systems, and on-the-job training.

13.2.2.6 Vendor Supplied Training Courses

Personnel from the plant engineering, maintenance, and technical staff may attend training courses supplied by offsite vendors. Examples of this training are computer systems courses and station nuclear reactor engineer training courses.

13.2.3 PROGRAM CONTENT FOR GENERAL EMPLOYEE TRAINING

13.2.3.1 Radiological Health and Safety

The training manager and the health physics superintendent will prepare a series of radiological health and safety lectures to be given to members of the plant staff. All persons assigned to the plant who are granted unescorted access to the radiation controlled area will attend this lecture series. This lecture will be reviewed and updated as required and will be presented to plant personnel on a regularly scheduled basis. A typical outline of the program includes nuclear plant terminology, biological effects of radiation, FNP plant specifics, 10 CFR 19 and 10 CFR 20, nonoccupational sources of radiation, FNP warning signs and hazards, use of protective clothing, and frisking for contamination.

13.2.3.2 Emergency Plan Training

Preplanned lectures will be given on a regularly scheduled basis. The material presented will be relevant to the job requirements of each individual. Copies of the plant emergency plan will be available for review by all plant personnel.

13.2.3.3 Security Plan Training

Preplanned lectures covering the FNP security plan will be given on a regularly scheduled basis to appropriate plant personnel.

The material presented will be relevant to the job requirements of all individuals. Persons directly involved with the administration and implementation of the plan will receive adequate training, retraining, and auditing to ensure the continued effectiveness of the plan.

13.2.3.4 Industrial Safety

Preplanned lectures covering the FNP industrial safety and health policy will be given on a regularly scheduled basis to appropriate plant personnel. A typical outline includes accident prevention, general plant safety, the FNP Safety and Health Policy and Procedures related to safety, and the Southern Nuclear Company Safety and Health Manual.

13.2.3.5 General Employee Retraining

Appropriate plant personnel will be given preplanned lectures on a regularly scheduled basis covering the following subjects. The material presented in the lecture series will be relevant to the individual's job requirements.

A. Emergency Plan implementing procedures.

B. Radiological health and safety.

C. Farley Nuclear Plant security plans.

Written examinations should be given to determine the knowledge of subjects covered in the retraining lecture series.

13.2.4 RECORDS

A resume of the person's qualifications and general records of correspondence and certifications will be maintained. Also, items such as examination results, retraining examinations, lecture attendance, drill participation, and results of retraining administered in areas in which personnel have exhibited deficiencies will be maintained. Initial operator license training and license retraining files will be maintained in accordance with 10 CFR 55.

All of these records will be used to judge the effectiveness of the FNP training and retraining programs. The plant supervisory staff and the plant training staff will periodically review in detail each individual's progress in the plant training program. The plant supervisory staff and the plant training staff will also periodically review the overall training and retraining programs to determine how well these programs are supplying and maintaining qualified personnel to safely and efficiently operate Farley Nuclear Plant.

13.4 REVIEW AND AUDIT

A program of in-plant and independent reviews and audits has been developed to provide a system to determine that plant design, construction, startup, and operation are consistent with company policy and rules, approved procedures, and license provisions; to ensure that unusual events are promptly investigated and corrected in a manner which reduces the probability of recurrence of such events; and to detect trends which may not be apparent to a day to day observer. For convenience of administration of the program, the review and audit program is divided into the construction phase and the test and operation phase.

13.4.1 REVIEW AND AUDIT - CONSTRUCTION

Review and audit during design and construction of the Farley Nuclear Plant (FNP) is a part of the quality assurance program which is described in section 17.1. This program does not utilize a formal review and audit committee, as such; however, through a comprehensive system of planned audits, compliance with all aspects of the quality assurance program is verified. Audits are performed on the design organizations, the construction site, and vendor facilities. The review and audit function during design and construction is fully described in section 17.1.

13.4.2 REVIEW AND AUDIT - TEST AND OPERATION

13.4.2.1 In-Plant Review

13.4.2.1.1 Continuing Review

As part of their day to day responsibilities, the plant supervisors and superintendents will review and audit all plant activities. They will verify that safety-related activities are performed in accordance with approved procedures. They will review documentation and records to verify that technical and quality requirements are maintained.

13.4.2.1.2 Plant Operations Review Committee

The Plant Operations Review Committee (PORC), composed of plant supervisory and technical personnel, will review matters concerning plant preoperation, startup, and operations. This committee became functional on June 13, 1974.

13.4.2.1.2.1 Plant Operations Review Committee (PORC)

A. Function

The PORC shall function to advise the general manager-nuclear plant on matters related to nuclear safety.

B. Membership

The PORC shall be composed of the:

1. Chairman (general manager-nuclear plant).
2. Vice chairman (assistant general manager - plant operations).
3. Member (assistant general manager - plant support).
4. Member (technical manager).
5. Member (operations manager).
6. Member (maintenance manager).
7. Member (systems performance manager).
8. Member (plant modifications manager).
9. Nonvoting member (supervisor-safety audit and engineering review).

C. Alternates

All alternate members shall be appointed in writing by the PORC chairman to serve on a temporary basis; however, not more than one alternate shall participate as a voting member in PORC activities at any one time.

D. Meeting Frequency

The PORC shall meet at least once per calendar month and as convened by the PORC chairman or vice chairman.

E. Quorum

A quorum shall consist of the chairman or vice chairman and three members, including alternates.

M. Revisions of NORB Charter

The NORB charter may be revised or amended by the chairman on the consent of the NORB, subject to license provisions, approval of the vice president, and approval of the NRC if required. The chairman is responsible for ensuring that such revisions and amendments are properly documented.

13.4.2.3 Technical Review and Control

A. Activities

Activities which affect nuclear safety shall be conducted as follows:

1. Procedures required by the plant technical specifications and other procedures which affect plant nuclear safety, and changes (other than editorial or typographical changes) thereto, shall be prepared, reviewed, and approved. Each such procedure or procedure change shall be reviewed by an individual/group other than the individual/group which prepared the procedure or procedure change but who may be from the same organization as the individual/group which prepared the procedure or procedure change. Procedures other than administrative procedures will be approved by either the technical manager, the operations manager, the maintenance manager, the systems performance manager, the plant modifications manager, the administrative manager, or an assistant general manager as applicable. The general manager-nuclear plant will approve administrative procedures, security implementing procedures, emergency plan implementing procedures, and contingency implementing procedures. Temporary changes to procedures which clearly do not change the intent of the approved procedures will be approved by two members of the plant staff, at least one of whom holds a senior reactor operator's license. For changes to procedures which may involve a change in intent of the approved procedures, the person authorized above to approve the procedure shall approve the change.

2. Proposed changes or modifications to plant nuclear safety-related structures, systems, and components

shall be reviewed as designated by the general manager-nuclear plant. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to plant nuclear safety-related structures, systems, and components shall be approved by the general manager-nuclear plant prior to implementation.

3. Proposed tests and experiments which affect plant nuclear safety and are not addressed in the Final Safety Analysis Report shall be prepared, reviewed, and approved. Each such test or experiment shall be reviewed by an individual/group other than the individual/group which prepared the proposed test or experiment but who may be from the same organization as the individual/group which prepared the proposed test or experiment. Test procedures will be approved in accordance with Technical Specification 6.5.3.1.

4. Reportable events pursuant to 10 CFR 50.73 shall be investigated, and a report shall be prepared which evaluates the occurrence and provides recommendations to prevent recurrence. Such reports shall be approved by the general manager-nuclear plant and forwarded to the vice president.

5. Individuals responsible for reviews performed in accordance with items 1 through 4 above shall be members of the plant supervisory staff previously designated by the general manager-nuclear plant. Each such review shall include a determination of whether or not additional cross-disciplinary review is necessary. If deemed necessary, such review shall be performed by the review personnel of the appropriate discipline.

6. Each review will include a determination of whether or not an unreviewed safety question is involved. Pursuant to 10 CFR 50.59, NRC approval of items involving unreviewed safety questions will be obtained prior to general manager-nuclear plant approval for implementation.

13.5 PLANT PROCEDURES

Actions concerning structures, systems, or components of the Farley Nuclear Plant (FNP) that are safety related are conducted according to written approved procedures. Safety-related structures, systems, and components are those that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public and such structures, systems, or components that are essential for the safe shutdown of the plant.

These procedures are written in sufficient detail so that a qualified individual may perform the required function without direct supervision. Procedures will contain the following significant aspects wherever they apply to the intent of each particular procedure:

A. Title - A concise descriptive statement concerning the activity covered in the procedure. Safety-related procedure titles are available in Document Control.

B. Purpose - A concise descriptive statement concerning the purpose and scope of the procedure.

C. References - Plant procedures, instructions, drawings, technical manuals, reports, the Final Safety Analysis Report, or other plant documents which contain information related to the procedure.

D. Precautions - Actions which if not taken or events which if not avoided when performing the procedure could result in hazardous personnel conditions or damage to plant equipment. Precautions will also appear in the main body of the procedure where applicable.

E. Prerequisites - Independent actions or procedures which shall be completed and plant conditions which shall exist prior to the procedure's use.

F. Limitations - Statements specifying limits on the parameters being controlled.

G. Main body - Step-by-step instructions in the degree of detail necessary to perform the required function or task.

H. Checkoff lists - Lists included in complex procedures requiring the person either performing or supervising the activity being performed to signify by his initials when important procedure steps have been completed.

I. Technical specifications - A reference to plant technical specifications where appropriate.

J. Symptoms - For emergency procedures, symptoms shall be included to aid in the identification of the emergency. They should include significant alarms, operating conditions, and, where possible, probable magnitudes of parameter changes.

K. Automatic actions - The automatic actions that will probably occur as a result of an emergency should be identified.

L. Immediate operator actions - For emergencies, steps should be specified for operation of controls or confirmation of automatic actions that are required to stop the degradation of conditions and mitigate their consequences.

M. Probable cause - For alarms the probable cause should be specified.

N. Acceptance criteria - The qualitative and/or quantitative criteria against which an evaluation of acceptability may be made. In certain procedures, the acceptance criteria will reference other sections of the procedure such as precautions, limitations, or checkoff list which may fulfill the acceptance criteria requirements.

Some procedural steps are required to be committed to memory, while others, which are routine actions, may be implied but not actually delineated.

Typical categories of plant procedures are as follows:

A. Administrative procedures.

B. Unit operating procedures.

C. System operating procedures.

D. Annunciator response procedures.

E. Critical safety function procedures.

F. Abnormal operating procedures.

G. Fuel handling procedures.

H. Surveillance test procedures.

- Plant personnel feedback.
- Incident investigation feedback.
- Design modification program.
- Operating experience evaluation program.
- Simulator training program.
- Technical specifications and FSAR revisions.
- Quality assurance program.

Additionally, as a part of the overall quality assurance program, the SAER group performs various audits (described in section 17.2) to assure that the procedural process is working and that procedures are being properly maintained.

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13.5.2 OPERATING PROCEDURES

Operating procedures were written before initial fuel loading and include all the anticipated operating conditions that affect the safety of the plant and the public. These procedures provide a preplanned method of conducting operations to minimize reliance on memory. They cover the following areas: system operating procedures, unit operating procedures, fuel handling procedures, annunciator response procedures, emergency operating procedures, and abnormal operating procedures.

The following types of procedures have been written for the FNP.

13.5.2.1 <u>System Operating Procedures</u>

Instructions for energizing, starting up, shutting down, changing modes of operation, and other instructions appropriate for operations of systems related to the safety of the unit are delineated in system operating procedures. These procedures are concerned with systems only and include valve lineup, control operation, and instrumentation within the system boundaries. Where needed to ensure a safe and proper sequence of operation, a procedure checkoff list is incorporated.

13.5.2.2 <u>Unit Operating Procedures</u>

Unit operating procedures have been written to provide instructions for the integrated operation of the unit. The system operating procedures covered in paragraph 13.5.2.1 were limited to individual systems, but these procedures integrate all auxiliary systems to the main nuclear steam supply system and turbine-generator to perform as a unit.

Unit operating procedures cover plant operation in the following areas.

13.5.2.2.1 Startup Procedures

Startup procedures have been written to provide instructions for starting the reactor from cold or hot conditions and establishing minimum load operation with the generator synchronized to the line. System procedures are referenced as required.

13.5.2.2.2 Shutdown Procedures

Procedures have been written to guide operations during and following controlled shutdown and include instructions for establishing or maintaining hot shutdown or cold shutdown conditions, as applicable. Such actions as monitoring and controlling reactivity, load reduction rates, cooldown rates, taking equipment or systems out of service and/or into service, electrical switching to ensure unit safety, etc., are covered in these procedures. System procedures are referenced as required.

13.5.2.2.3 Power Range Operation Procedures

Procedures have been written for steady-state power operation and load changing and include instructions on the use of control rods, chemical shim, long- and short-term control of reactivity, making deliberate load changes, and adjusting operating parameters to trim power operation. System procedures are referenced as required.

13.5.2.2.4 Reactor Startup Procedures

Procedures have been written to instruct the operator on prerequisites, equipment control, and control functions concerning the reactor in going critical. Actual control manipulations have been included in these written procedures. These procedures are included in the startup procedures described in paragraph 13.5.2.2.1.

13.5.2.3 <u>Abnormal Operating Procedures</u>

Procedures have been written for operation of the unit under abnormal conditions.

13.5.2.4 <u>Annunciator Response Procedures</u>

Procedures have been written to instruct the operator on the proper action to be taken in response to each safety-related annunciator in the control room. These procedures contain annunciator identification, inputs into this annunciator circuit, and logical responses to be taken to ensure corrective action.

13.5.2.5 Fuel Handling Procedures

Fuel handling operations will be performed in accordance with written procedures. These procedures specify actions and philosophy for core alterations and partial or complete fueling operations. They include continuous monitoring of the neutron flux throughout core loading, periodic data taking, audible annunciation of abnormal flux increases, duties of personnel assigned to fueling, response actions to alarms during fueling, instructions for proper sequence of events, rules for periods when fueling is interrupted, verification and frequency of sampling to ensure the shutdown margin, communications between control room and the fuel loading station, criteria for stopping fueling and evacuation systems operation, and documentation of final fuel component serial numbers and locations. Prerequisites have been included in these procedures to ensure the status of plant systems is such that fueling can proceed. Specific procedures are written for each fueling to handle those actions and parameters that are unique to that particular refueling. System operating procedures are referenced as required.

13.5.3 MAINTENANCE PROCEDURES

A maintenance program was developed early in plant life to maintain safety-related equipment at the efficiency level required to perform its intended function. This program includes maintenance of safety-related mechanical, instrument, and electrical equipment. Maintenance or modifications which may affect the functioning of safety-related structures, systems, or components are performed to ensure operating quality at least equivalent to that specified in applicable codes, bases, standards, design requirements materials specifications, and inspection requirements. To ensure a high degree of confidence, appropriate inspection in accordance with applicable standards is performed. Replacement components are used only when the proper quality assurance documents are available or when the required quality assurance can be obtained and documented by inspection and/or testing prior to being placed in service. These quality review measures are documented, and evidence of the documentation is retained.

Maintenance which can affect the performance of safety-related equipment is preplanned and performed in accordance with written procedures, approved documented instructions, and/or approved drawings appropriate to the circumstances. In

maintenance situations where related vendor manuals or instructions or approved drawings do provide adequate instruction to ensure the required quality of work on safety-related equipment, these documents will be used. However, where related vendor manual or instructions or approved drawings do not provide adequate instruction to ensure the required quality of work on safety-related equipment, a suitable procedure to handle this maintenance work will be written. Skills normally possessed by qualified maintenance personnel are not covered in detailed procedures.

When failure occurs to safety-related equipment, the cause of the failure will be evaluated; however, since the probability of failure is usually unknown and the time and mode of failure are usually unpredictable, procedures will not be written for repair of most equipment prior to failure.

A preventive maintenance schedule has been developed which describes the time and type of maintenance to be performed. A preliminary schedule was developed early in plant life and will be refined and changed as experience with the equipment is gained. This schedule specifies equipment inspections, replacement of such items as filters and strainers, and inspection or replacement of parts that have a specific lifetime. As equipment baselines develop, a computer program will be considered to aid in preventive maintenance scheduling. Lubrication requirements will be scheduled separately.

Maintenance is scheduled and planned so as not to jeopardize the safety of the plant. Planning is done in order to consider the possible safety consequences of concurrent or sequential maintenance, testing, or operating activities. Maintenance is performed in such a manner that the license limits are not violated. Planning for maintenance includes evaluation of the use of special processes, equipment, and materials to be used in the performance of the job. This evaluation attempts to assess the potential hazards to personnel and equipment.

Modifications in equipment or systems which might degrade the plant quality will not be permitted.

Procedures to support the maintenance philosophy were written early in plant life. These procedures control plant maintenance activities during operation and at the time of equipment failure.

13.5.4 PERIODIC CALIBRATION AND TEST PROCEDURES

13.5.4.1 Instrument Calibration and Tests

In those instances where equipment technical manuals do not provide sufficient instruction, procedures will be written for periodic calibration and testing of all safety-related plant instrumentation. This instrumentation includes interlocks, alarm devices, sensors, readout instruments, transmitters, signal conditioners, laboratory equipment, key recorders, and protective logic circuits. A list of equipment to be calibrated, a calibration schedule, and calibration records will be kept and maintained by the instrument group. Manuals will be reviewed and procedures written with the intent of ensuring measurement accuracies adequate to keep safety parameters and controls within safety and operational limits. Calibration, testing, and check of instrumentation channels will be performed as specified in the plant technical specifications.

13.5.4.2 Safety-Related Surveillance Tests

Safety-related surveillance tests and inspections are performed in accordance with the plant technical specifications to ensure that failures or substandard performance do not remain undetected and that the required reliability of safety systems is maintained. Testing of safety-related plant structures, systems, and components is performed in accordance with approved written test procedures which set forth the requirements and acceptance limits. Test procedures contain a description of the test objectives, the acceptance criteria that will be used to evaluate the test results, the prerequisites for performing the tests, including any special conditions to be used to simulate normal or abnormal operating conditions, and the test procedure. These procedures also specify any special test equipment or calibrations required to conduct the surveillance test. A surveillance schedule, reflecting the status of all planned in-plant surveillance testing, was established prior to fuel load. Additional control procedures have been established, as necessary, to ensure timely conduct of surveillance testing, appropriate documentation, reporting, and evaluation of test results.

Records are kept in sufficient detail to permit adequate confirmation of the test program. They identify, as a minimum,

the data recorder, results of the test, the acceptability of the results, deviations and their cause or reason, and any corrective action resulting from the test. Significant deficiencies identified by the tests will be reported to management where the deficiencies will be evaluated and the condition corrected in a timely manner.

13.5.5 CHEMICAL AND RADIOCHEMICAL CONTROL PROCEDURES

Procedures have been written for chemical and radiochemical control activities. These procedures include the nature and frequency of sampling and analyses, instructions for maintaining coolant and condensate within prescribed quality limits, and limitations on concentrations of agents that could cause corrosive attack, foul heat transfer surfaces, or become sources of radiation hazards due to activation. These procedures include laboratory instructions and calibration of laboratory equipment.

Sample intervals will be according to the plant technical specifications.

13.5.6 PROCEDURES FOR COMBATING EMERGENCIES AND OTHER SIGNIFICANT EVENTS

Procedures have been written to guide operations during potential emergencies. They have been written so that a trained operator and crew will know in advance the expected course of events that will identify an emergency and the immediate action which should be taken.

Procedures that cover actions for manipulation of controls to prevent accidents or lessen the consequences are based on a general predictable sequence of observations and actions. Emphasis is placed on operator responses to observations and indications in the control room; that is, when immediate operator actions are required to prevent or mitigate the consequences of a serious condition, procedures delineate and require that these actions be implemented promptly. When initially available intelligence provided to operating personnel via instrument readings, annunciator alarms, physical conditions, and personal observations may not clearly indicate the difference between a simple operational problem and a serious emergency, the actions outlined in the procedures are based on a conservative course of action by the operating crew. The operator will be responsible for believing and responding conservatively to instrument indications unless they are proven

to be false. Considerable judgment on the part of competent personnel will be used before departing from the procedure. In events where it is necessary to depart from approved procedures to prevent injury to personnel or damage to the facility, the deviation will be logged with the prevailing circumstances described. Sections of emergency operating procedures that require immediate response action from the operating crew are required to be committed to memory.

13.5.6.1 Events of Potential Emergency

Potential emergency conditions have been identified and procedures for coping with them have been prepared. Some of these procedures include immediate action to be taken, while others may guide operations in correcting an abnormal situation which could lead to an emergency.

13.5.6.2 Procedures for Implementing the Emergency Plan

Implementing procedures for emergency plan actions have been written to instruct all plant personnel in the integrated implementing actions of the emergency plan.

13.5.7 PROCEDURES FOR THE CONTROL OF RADIOACTIVITY

Procedures have been written to provide for personnel exposure control, for the control of radioactive materials on the plant site, and for the control, sampling, monitoring, storage, and disposal of solid, liquid, and gaseous radwastes.

13.5.8 SECURITY PROCEDURES

Security procedures have been prepared to complement the security plan in describing the security requirements for the FNP and to guide security activities. These procedures were drafted and reviewed by members of the plant security group and are used to guide security activities. Because of the sensitive nature and content of most security procedures, copies will be placed only where required, with access restricted to those plant personnel having a need to know. Security procedures are serially numbered documents which are periodically inventoried and accounted for by the security

supervisor. Procedures are subject to reviews by the security supervisor whenever a security threat or other security incident makes such a review desirable.

Revisions to procedures are subject to the same review and approval as original procedures.

Revised or obsolete security procedures are processed in accordance with approved plant procedures. Temporary security procedures and temporary revisions to procedures to cover unanticipated or emergency situations may be issued at the direction of the general manager-nuclear plant. Temporary procedures shall reflect the purpose and limitations of their use. Unless declared obsolete and processed in accordance with approved plant procedures, temporary procedures shall be prepared as permanent procedures as soon as practicable.

13.5.9 SPECIAL NUCLEAR MATERIAL ACCOUNTABILITY PROCEDURES

Special nuclear material procedures have been written to implement the special nuclear material accountability program. These procedures delineate personnel responsibilities and authorities, designate and describe item control areas, and provide instructions for special nuclear material control records and reports, receiving and shipping of special nuclear material, internal transfers, physical inventories, special nuclear material element and isotopic calculations, and special nuclear material review and audit.

13.5.10 ENGINEERING TECHNICAL PROCEDURES

Engineering technical procedures provide instructions for the performance of tests or essential calculations which, because of their nature or technical content, are normally performed by plant engineers. These procedures cover such subjects as reactor core physics tests, calculation of operating curves and data, component or systems performance evaluations, developmental testing, and other related functions

13.5.11 ENVIRONMENTAL SAMPLING PROCEDURES

Environmental sampling procedures provide methods for measurement of radiation and of radioactive material in those exposure pathways which lead to the highest potential radiation exposures of individuals resulting from operation of the plant.

13.5.12 DOCUMENT CONTROL PROCEDURES

Document control procedures provide guidelines for handling various documents and records.

13.5.13 STOREROOM PROCEDURES

Handling, storing, and distribution of materials, components, and equipment are covered in the storeroom procedures.

13.5.14 FIRE SURVEILLANCE PROCEDURES

Fire surveillance procedures have been written to provide instructions for the testing of fire surveillance equipment to ensure its operability.

13.5.15 FIRE VENTILATION PROCEDURES

Fire ventilation procedures provide symptoms, automatic actions, initial actions, secondary actions, and restoration of systems for ventilation operations initiated and/or required due to a fire.

Tables 13.5-1 through 13.5-18 have been deleted.

patrol. All-weather roads and pathways are provided within the protected area and remote protected areas for the use of the security patrol.

Intrusion alarms, security fences, lights, and communication links are maintained in operable and effective condition under the supervision of the security supervisor. Lights and communication links will be inspected and tested for operability and required functional performance. Intrusion alarms are tested periodically.

13.7.1.2 Employee Selection and Performance Review

In order to select reliable personnel to protect against industrial sabotage, employment standards and procedures have been established by SNC management. These include application forms, background investigation, physical examination, and interviews. The application, background investigation, physical examination, and initial interviews are coordinated by the SNC Personnel Department. Prior to employment for a position at the plant, the applicant's personnel file is reviewed and at least one detailed interview is conducted by SNC Personnel. All new employees have a 6-month probation period during which their performance is closely observed. Security Force members are screened and qualified in accordance with the provisions of the FNP Training and Qualification Plan.

Personnel being considered for promotion and/or transfer to the plant from other plants or departments within the company are screened to ensure the effectiveness of the plant industrial security program.

An alert plant organization, cognizant of its responsibility for protection against industrial sabotage, is maintained. The performance of all employees is appraised annually and the results reported in order to: further aid in maintaining a high level of employee performance and the maximum utilization of employee abilities; provide recorded evidence of employee performance for use in making judgments concerning transfer, demotion, promotion, and terminations; ensure that employees are adequately and systematically informed of the effectiveness of their service; and further facilitate the maintenance of a high standard of supervision in SNC. A statement of the results of those appraised is signed by the employee's supervisor.

Observation of employee service is a continuous supervisory function; such observation is made as a regular part of day-to-day supervision. Plant supervision is constantly on the alert for early detection of changes in behavioral patterns of employees under its supervision.

13.7.2 SECURITY PLAN

The following sections discuss in a general manner various aspects of the FNP security plan.

13.7.2.1 Means for Control of Plant Access

A security force of well-trained, uniformed guards, polices plant property and provides access control and surveillance of the plant protected and vital areas. Other plant personnel, such as the operating group, help monitor access inside plant structures by observation of the plant area during equipment checks and by challenging any unauthorized individuals within the plant buildings.

Gates permitting entrance to the protected area surrounded by the security fence are either locked or manned access control is in effect. The main security fence and fences surrounding outlying structures are regularly patrolled by roving guards. The roving guards have radio communication with the central alarm station.

Guidelines are included in the security plan for monitoring and controlling the access to and from the plant and the movement of persons within the plant. The security plan also includes guidelines for searching vehicles and personnel entering and exiting the plant. In-plant access control is provided by operating personnel challenging the entry of unauthorized persons attempting to enter vital operating areas. Guidelines controlling the entry of unauthorized vehicles and the entry and exit of unauthorized materials through the fence and gate system are included in the security plan. Vehicle access to the protected area is limited to certain authorized vehicles.

13.7.2.2 Control of Personnel by Categories

The general public is admitted inside the controlled area at the Central Security Control Building. A visitor identification system is used as the primary form of access control inside the security areas. Visitors are escorted while in the plant areas by authorized personnel who are responsible for the visitors' actions, areas of movement, and safety.

A badge identification system is the primary form of access control to the protected area. Access is permitted through manned gates and is limited to properly authorized individuals.

A reactor trip signal acts to open two trip breakers connected in series which feed power to the control rod drive mechanisms (CRDMs). The loss of power to the mechanism coils causes the mechanisms to release the rod cluster control assemblies (RCCAs) which then fall by gravity into the core. There are various instrumentation delays associated with each trip function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from when the monitored parameter reaches the trip setpoint until the rods are free and begin to fall. The setpoint study is performed in the course of finalizing the design of the plant; however, many of the accident analyses in this chapter conservatively do not take credit for the control systems.

Table 15.1-3 refers to the overtemperature and overpower ΔT trip shown in figure 15.1-1A.

These trip setpoints bound the transition cores and a full core of VANTAGE 5 fuel. The associated OTΔT f(ΔI) penalty is shown in figure 15.1-1B.

For all the reactor trips, the difference between the trip setpoints assumed in the analysis and the nominal trip setpoints account for instrumentation channel error and setpoint error. The plant technical specifications specify the nominal trip setpoints. Response time limits for the reactor trip systems are maintained in table 7.2-5. The calibration of protection system channels and the periodic determination of instrument response times are in accordance with the plant technical specifications.

15.1.4 INSTRUMENTATION DRIFT AND CALORIMETRIC ERRORS

The VANTAGE 5 fuel design features, the modified safety analysis assumptions, and the application of new methodologies (i.e., RTDP, WRB-1, and WRB-2) as discussed in section 4.4 (with respect to the changes associated with the instrument uncertainties for the NSSS control parameters of power, pressure, temperature, and flow) are covered in reference 2.

15.1.5 ROD CLUSTER CONTROL ASSEMBLY INSERTION CHARACTERISTIC

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCAs and the variation in rod worth as a function of rod position. With respect to accident analyses, the critical parameter is the time from the start of insertion up to the dashpot entry or approximately 85 percent of the rod cluster travel. For accident analyses, it is conservatively assumed that the insertion time to dashpot entry is

2.7 seconds. The RCCA position versus time assumed in accident analyses is shown in figure 15.1-2.

Figure 15.1-3 shows the fraction of total negative reactivity insertion for a core where the axial distribution is skewed to the lower region of the core. An axial distribution which is skewed to the lower region of the core can arise from a xenon oscillation or can be considered as representing a transient axial distribution which would exist after the rod cluster control assembly bank has already traveled some distance after trip. This curve has been conservatively selected to bound future reloads, which can include axial blankets of natural uranium.

This lower curve is used as input to all point kinetics core models used in transient analyses.

There is inherent conservatism in the use of this curve in that it is based on a skewed distribution which would exist relatively infrequently. For cases other than those associated with xenon oscillations, significant negative reactivity would have been inserted due to the more favorable axial distribution existing prior to trip.

The normalized RCCA negative reactivity insertion versus time is shown in figure 15.1-4. The curve shown in this figure was obtained from figures 15.1-2 and 15.1-3. A total negative reactivity insertion following trip of 4.8-percent $\Delta k/k$ is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available as shown in table 4.3-3. Both the trip reactivity and reactivity insertion rate are verified to be conservative with respect to the core design as part of the reload design process (reference 3).

The normalized RCCA negative reactivity insertion versus time curve for an axial power distribution skewed to the bottom (figure 15.1-4) is used in transient analyses. Where special analyses require use of three-dimensional or axial one-dimensional core models, the negative reactivity insertion resulting from reactor trip is calculated directly by the reactor kinetic code and is not separable from other reactivity feedback effects. In this case, the rod cluster control assembly position versus time of reactor trip (figure 15.1-2) is used as code input.

15.1.6 REACTIVITY COEFFICIENTS

The transient response of the reactor system is dependent on reactivity feedback effects, in particular the moderator

TABLE 15.1-2A (SHEET 1 OF 3)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Moderator Density ($\Delta k/g/cm^3$)	Doppler	Initial NSSS Thermal Power Output Assumed (MWt)
Condition II					
Uncontrolled RCCA bank withdrawal from a subcritical condition	TWINKLE, FACTRAN, THINC	+ 7.0	-	Coefficient is consistent with a defect of - 900 pcm	0 (subcritical)[e]
Uncontrolled RCCA bank withdrawal at power	LOFTRAN	+ 7.0	0.50	Lower and upper (see figure 15.1-5)	279, 1674, and 2790[c,f]
RCCA misalignment	THINC, ANC, LOFTRAN	-	-	-	2775[b,f]
Uncontrolled boron dilution	NA	NA	NA	NA	0 and 2785[a]
Partial loss of forced reactor coolant flow	LOFTRAN, FACTRAN, THINC	+ 7.0	-	Upper (see figure 15.1-5)	2790[c,f]
Startup of an inactive RCP	LOFTRAN, FACTRAN THINC	-	0.50	Lower (see figure 15.1-5)	1726.7[a,e]
Loss of external electrical load and/or turbine trip	LOFTRAN	+ 7.0, ≤ 70% RTP Ramping to 0 at 100% RTP	0.50	Lower and upper (see figure 15.1-5)	2790[c,f,h]
Loss of normal feedwater	LOFTRAN	+ 7.0	-	Upper (see figure 15.1-5)	2790[c,e]
Loss of all ac power to the station auxiliaries	LOFTRAN	+ 7.0	-	Upper (see figure 15.1-5)	2790[c,e]
Excessive heat removal due to feedwater system malfunctions	LOFTRAN	-	0.50	Lower (see figure 15.1-5)	0 and 2785[a,f]
Excessive load increase	LOFTRAN	0.0 [g]	0.50	Upper and lower (see figure 15.1-5)	2785[a,f]
Accidental depressurization of the RCS	LOFTRAN	+ 7.0	-	Lower (see figure 15.1-5)	2790[c,f]
Accidental depressurization of the main steam system	LOFTRAN	Function of moderator density (see subsection 15.2.13 and figure 15.2-40A)	-	See figure 15.2-40B	0 (subcritical)[e]
Inadvertent operation of the ECCS during power operation	LOFTRAN	+ 7.0	-	Lower (see figure 15.1-5)	2785[a,f]

TABLE 15.1-2A (SHEET 3 OF 3)

Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Moderator Density (Δk/g/cm^3)	Doppler	Initial NSSS Thermal Power Output Assumed (MWt)
Spectrum of RCCA ejection accidents	TWINKLE, FACTRAN, THINC	Refer to paragraph 15.4.6.3	-	Coefficient is consistent with a defect of - 900 pcm	0 and 2775[b,e]

a. Nominal pump heat of 10 MWt is assumed.

b. No pump heat (core thermal power) assumed.

c. Maximum pump heat of 15 MWt is assumed.

d. Uprated NSSS power with maximum pump heat increased by 2 percent.

e. STDP with a TDF of 86,000 gal/min/loop assumed.

f. RTDP with a MMF of 87,800 gal/min/loop assumed.

g. More limiting than + 0.7 pcm/°F.

h. Without pressure control cases, assume 2667 MWt.

protect the RCS and steam generator against overpressure for all load losses without assuming operation of the steam dump system, pressurizer spray, pressurizer PORVs, automatic RCCA control, or the direct reactor trip on turbine trip.

The steam generator safety valve capacity is sized to remove the steam flow at the engineered safeguard design rating (105 percent of nominal full-power steam flow) from the steam generator without exceeding 110 percent of the steam system design pressure. The pressurizer safety valve capacity is sized based on a complete loss-of-heat sink with the plant initially operating at the maximum calculated turbine load along with operation of the steam generator safety valves. The pressurizer safety valves are then able to maintain the RCS pressure within 110 percent of the RCS design pressure without a direct reactor trip on turbine trip action.

The Farley reactor protection system (RPS) and primary and secondary system designs preclude overpressurization. A more complete discussion of overpressure protection can be found in reference 6.

15.2.7.2 <u>Analysis of Effects and Consequences</u>

15.2.7.2.1 Method of Analysis

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from full power without a direct reactor trip. This assumption is made to show the adequacy of the pressure-relieving devices and to demonstrate core protection margins; it delays reactor trip until conditions in the RCS result in a trip due to other signals. Thus, the analysis assumes a worst-case transient. In addition, no credit is taken for steam dump. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for AFW (except for long-term recovery) to mitigate the consequences of the transient.

The total loss of load transient is analyzed with the LOFTRAN (reference 4) computer code. The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and main steam safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level.

Major assumptions are summarized below.

A. The accident is analyzed using the RTDP. Initial core power, reactor coolant temperature, and pressure are assumed to be at their nominal values consistent with

steady-state full-power operation. Uncertainties in initial conditions are included in the DNBR limit as described in WCAP-11397.[5]

B. The total loss of load transient is analyzed with both maximum and minimum reactivity feedback. The maximum feedback (EOL) cases assume a large (absolute value) negative moderator temperature coefficient and the most-negative Doppler power coefficient. The minimum feedback (BOL) cases with and without pressurizer pressure control assume a positive (+7 pcm/°F) and zero (0 pcm/°F) moderator temperature coefficient, respectively, and the least-negative Doppler coefficient.

C. From the standpoint of the maximum pressures attained, it is conservative to assume that the reactor is in manual rod control. If the reactor were in automatic rod control, the control rod banks would move prior to trip and reduce the severity of the transient.

D. The loss of load event is analyzed both with and without pressurizer pressure control. The pressurizer PORVs and sprays are assumed operable for the cases with pressure control. The cases with pressure control minimize the increase in primary pressure which is conservative for the DNBR transient. The cases without pressure control maximize the pressure increase which is conservative for the RCS overpressurization criterion.

E. Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for AFW flow since a stabilized plant condition will be reached before AFW initiation is normally assumed to occur.

F. Only the OTΔT, high pressurizer pressure, and low-low steam generator water level reactor trips are assumed operable for the purpose of this analysis. No credit is taken for a reactor trip on high pressurizer level or the direct reactor trip on turbine trip.

G. No credit is taken for the operation of the steam dump system or steam generator PORVs. This assumption maximizes secondary pressure. The main steam safety valves are assumed to lift and be full open at 6 percent above the steam generator design pressure. This 6 percent can be considered to include allowances for safety valve setpoint uncertainty and accumulation.

H. The analysis value for pressurizer safety valve set pressure includes a 1-percent uncertainty. For those cases which are analyzed primarily for DNBR (pressurizer pressure control cases), the uncertainty is applied in the negative direction, thus reducing the safety analysis set pressure. For those cases analyzed primarily for peak RCS pressure, the uncertainty is applied in the positive direction. The peak pressure cases also consider a 1-percent set pressure shift and 1.6-s water purge time due to the pressurizer safety valve loop seals. In these cases, the pressurizer safety valve loop seals begin to bleed at 2550 psia. Steam relief occurs following the 1.6-s purge time.

15.2.7.2.2 Results

Four cases were analyzed for a total loss of load from 100 percent of NSSS power (with pressure control cases assume 2790 MWt (uprated); without pressure control cases assume 2667 MWt).

A. Minimum feedback with pressure control.

B. Maximum feedback with pressure control.

C. Maximum feedback without pressure control.

D. Minimum feedback without pressure control.

The calculated sequence of events for the four cases is presented in table 15.2-1.

Case A

Figures 15.2-19 and 15.2-20 show the transient response for the total loss of steam load event under BOL conditions, including a +7 pcm/°F moderator temperature coefficient, with pressure control. The reactor is tripped on OTΔT. The neutron flux increases until the reactor is tripped, and although the DNBR value decreases below the initial value, it remains well above the safety analysis limit throughout the entire transient. The pressurizer relief valves and sprays maintain primary pressure below 110 percent of the design value. The main steam safety valves are also actuated and maintain secondary pressure below 110 percent of the design value.

Case B

Figures 15.2-21 and 15.2-22 show the transient response for the total loss of steam load event under EOL conditions, assuming a conservatively large positive moderator density coefficient of 0.5 δk/g/cc (corresponding to a large negative moderator temperature coefficient) and a most-negative Doppler only power

coefficient (see figure 15.1-5) with pressure control. The reactor is tripped on OTΔT. The DNBR increases throughout the transient and never drops below the initial value. The pressurizer relief valves and sprays maintain primary pressure below 110 percent of the design value. The pressurizer pressure remains below the safety valve setpoint during the transient. The main steam safety valves are also actuated and maintain secondary pressure below 110 percent of the design value.

Case C

Figures 15.2-23 and 15.2-24 show the transient response for the total loss of steam load event under BOL conditions, including a zero moderator temperature coefficient without pressure control. The reactor is tripped on high pressurizer pressure. The neutron flux remains essentially constant at full power until the reactor is tripped, and the DNBR remains above the initial value for the duration of the transient. The pressurizer safety valves are actuated and maintain primary pressure below 110 percent of the design value. The main steam safety valves are also actuated and maintain secondary pressure below 110 percent of the design value.

Case D

Figures 15.2-25 and 15.2-26 show the transient response for the total loss of steam load event under EOL conditions, assuming a conservatively large positive moderator density coefficient of 0.5 δk/g/cc (corresponding to a large negative moderator temperature coefficient) and a most-negative Doppler only power coefficient (see figure 15.1-5) without pressure control. The reactor is tripped on high pressurizer pressure. The DNBR increases throughout the transient and never drops below the initial value. The pressurizer safety valves are actuated and maintain primary pressure below 110 percent of the design value. The main steam safety valves are also actuated and maintain secondary pressure below 110 percent of the design value.

15.2.7.3 Conclusions

The results of this analysis show that the plant design is such that a total loss of external electrical load without a direct reactor trip presents no hazards to the integrity of the RCS or the main steam system. The minimum DNBR for each case is greater than the safety analysis limit value. The peak primary and secondary pressures remain below 110 percent of design at all times.

15.2.8 LOSS OF NORMAL FEEDWATER

15.2.8.1 Identification of Causes and Accident Description

A loss of normal feedwater (from pump failures, pipe breaks, valve malfunctions, or loss of offsite ac power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped during this accident, core damage would possibly occur from a sudden loss-of-heat sink. If an alternate supply of feedwater were not supplied to the plant, residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer occurs. Loss of significant water from the RCS could conceivably lead to core damage. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the core does not approach a DNB condition.

The following provide the necessary protection against a loss of normal feedwater:

A. Reactor trip on low-low water level in any steam generator.

B. Two motor-driven auxiliary feedwater pumps (350 gal/min each) which are started automatically on one of the following:

   1. Low-low level in any steam generator.

   2. Loss of both main feedwater pumps.

   3. Any safety injection signal.

   4. Loss of offsite power (automatic transfer to diesel generators).

   The motor-driven auxiliary feedwater pumps can also be started manually from the control room.

C. One turbine-driven auxiliary feedwater pump (700 gal/min) which is started automatically on one of the following:

   1. Low-low level in any two of three steam generators.

2. Undervoltage on any two of three reactor coolant pump buses.

The turbine-driven auxiliary feedwater pump can also be started manually from the control room.

The motor-driven AFW pumps are connected to vital buses which are powered by diesel generators if a loss of offsite power occurs. The turbine-driven pump utilizes steam from the secondary system. The controls are designed to start both types of pumps within 60 s, even if a loss of all ac power occurs simultaneously with loss of normal feedwater. The AFW pumps are normally aligned to take suction from the condensate storage tank for delivery to the steam generators. A backup source of water for the pumps is provided by the safety-related portion of the service water system (see section 6.5). The RPS and AFW system design ensure that reactor trip and AFW flow will occur following any loss of normal feedwater.

The analysis shows that, following a loss of normal feedwater, the AFW system is capable of removing the stored and residual heat, thus preventing overpressurization of the RCS, overpressurization of the secondary side, or uncovery of the reactor core. Consequently, the plant is able to return to a safe condition.

15.2.8.2 Analysis of Effects and Consequences

15.2.8.2.1 Method of Analysis

A detailed analysis using the LOFTRAN (reference 4) computer code is performed in order to determine the plant transient following a loss of normal feedwater. The code describes the core neutron kinetics; RCS, including natural circulation, pressurizer, pressurizer PORVs and sprays, steam generators, main steam safety valves; and the AFW system, and computes pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

The following assumptions are made in the analysis.

A. Reactor trip occurs on steam generator low-low water level at 0.0 percent of narrow range span.

B. The plant is initially operating at 102 percent of the NSSS design rating (2790 MWt). A conservatively large RCP heat of 15 MWt is assumed.

of the onsite ac distribution system. The events following a loss of ac power with turbine and reactor trip are described in the sequence listed below.

A. The emergency diesel generators will start on a loss of voltage on the plant emergency buses and begin to supply plant vital loads.

B. Plant vital instruments are supplied by emergency power sources.

C. As the steam system pressure rises following the trip, the steam system PORVs are automatically opened to the atmosphere. Steam dump to the condenser is assumed not to be available. If the PORVs are not available, the self-actuated main steam safety valves will lift to dissipate the sensible heat of the fuel and coolant plus the residual heat produced in the reactor.

D. As the no-load temperature is approached, the steam system PORVs (or the self-actuated safety valves, if the PORVs are not available) are used to dissipate the residual heat and to maintain the plant at the hot standby condition.

The following provide the necessary protection against a loss of all ac power.

A. Reactor trip on low-low water level in any steam generator.

B. Two motor-driven AFW pumps that are started on:

1. Low-low level in any steam generator.
2. Trip of both main feedwater pumps.
3. Any safety injection signal.
4. Loss of offsite power (automatic transfer to diesel generators).
5. Manual actuation.

C. One turbine-driven auxiliary feedwater pump that is started on:

1. Low-low level in any two steam generators.

2. Undervoltage on any two RCP buses.

3. Manual actuation.

The AFW system is initiated as discussed in the loss of normal feedwater analysis (subsection 15.2.8). The turbine-driven pump utilizes steam from the secondary system and exhausts it to the atmosphere. The motor-driven AFW pumps are supplied by power from the diesel generators. The AFW pumps are normally aligned to take suction from the condensate storage tank for delivery to the steam generators. A backup source of water for the pumps is provided by the safety-related portion of the service water system (see section 6.5). The RPS and AFW system design ensure that reactor trip and AFW flow will occur following any loss of normal feedwater.

Following the loss of power to the RCPs, coolant flow is necessary for core cooling and the removal of residual and decay heat.

Heat removal is maintained by natural circulation in the RCS loops. Following the RCP coastdown, the natural circulation capability of the RCS will remove decay heat from the core, aided by the AFW flow in the secondary system. Demonstrating that acceptable results can be obtained for this event proves that the resultant natural circulation flow in the RCS is adequate to remove decay heat from the core.

The first few seconds after the loss of ac power to the RCPs will closely resemble a simulation of the complete loss of flow event (subsection 15.3.4, where it is demonstrated that the DNB design basis is satisfied). Therefore, the DNB aspects for the station blackout event are not explicitly evaluated in this analysis. The analysis shows that, following a loss of all ac power to the station auxiliaries, RCS natural circulation and the AFW system are capable of removing the stored and residual heat, consequently preventing overpressurization of the RCS, overpressurization of the secondary side, or uncovery of the reactor core. The plant is, therefore, able to return to a safe condition.

15.2.12.2 Analysis of Effects and Consequences

15.2.12.2.1 Method of Analysis

The accidental depressurization of the RCS is analyzed by the detailed digital computer code LOFTRAN.[4] The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables, including temperatures, pressures, and power level.

In calculating the DNBR, the following conservative assumptions are made:

A. The accident is analyzed using the RTDP. Initial core power, reactor coolant temperature, and pressure are assumed to be at their nominal values consistent with steady-state full-power operation. Uncertainties in initial conditions are included in the DNBR limit as described in WCAP-11397 (reference 5).

B. A +7-pcm/°F (most positive) moderator temperature coefficient of reactivity is assumed in order to provide a conservatively high amount of positive reactivity feedback due to changes in the moderator temperature.

C. A small (absolute value) Doppler coefficient of reactivity is assumed, such that the resultant amount of negative feedback is conservatively low in order to maximize any power increase due to moderator feedback.

D. The spatial effect of voids resulting from local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape. In fact, it should be noted that the power peaking factors are kept constant at their design values, while the void formation and resulting core feedback effects would result in considerable flattening of the power distribution. Although this would significantly increase the calculated DNBR, no credit is taken for this effect.

15.2.12.2.2 Results

The system response to an inadvertent opening of a pressurizer safety valve is shown in figures 15.2-37 through 15.2-39. Figure 15.2-37 illustrates the nuclear power transient following the depressurization. Nuclear power increases slowly until reactor

trip occurs on OTΔT. The pressure decay and core average temperature transients following the accident are given in figure 15.2-38. The DNBR decreases initially, but increases rapidly following the reactor trip as shown in figure 15.2-39. The DNBR remains above the safety analysis limit value throughout the transient.

The calculated sequence of events is shown in table 15.2-1.

15.2.12.3 Conclusions

The results of the analysis show that the pressurizer low pressure and OTΔT RPS signals provide adequate protection against the RCS depressurization event. Thus, there will be no cladding damage or release of fission products to the RCS.

15.2.13 ACCIDENTAL DEPRESSURIZATION OF THE MAIN STEAM SYSTEM

15.2.13.1 Identification of Causes and Accident Description

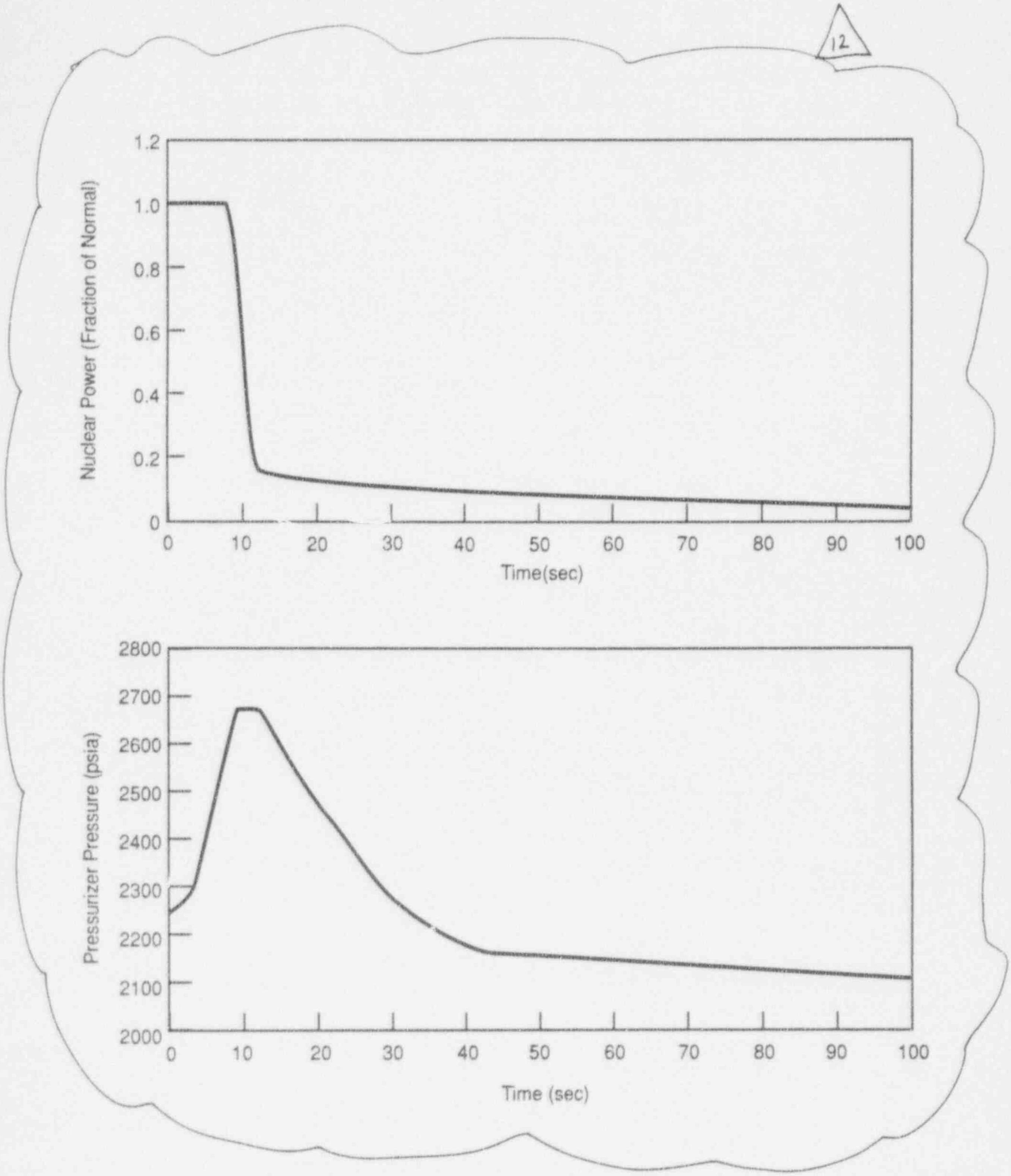
The most severe core conditions resulting from an accidental depressurization of the main steam system, which is classified as an ANS Condition II event, result from an inadvertent opening of a single steam dump, relief, or safety valve. The analyses performed assuming a rupture of a main steam pipe, which is classified as an ANS Condition IV event, are given in section 15.4.

The steam released as a consequence of this accidental depressurization results in an initial increase in steam flow, which decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction in coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity and subsequent reduction of core shutdown margin.

For an accidental depressurization of the main steam system, the radiation releases must remain within the requirements of 10 CFR Part 20.1 - 20.601. This is the ANSI N18.2 criterion for Condition II events, "Faults of Moderate Frequency." Although the plant may return to criticality, the above limit can be met by showing there is not consequential damage, i.e., that the DNB design basis is met. Therefore, the analysis is performed to demonstrate that the following criterion is satisfied:

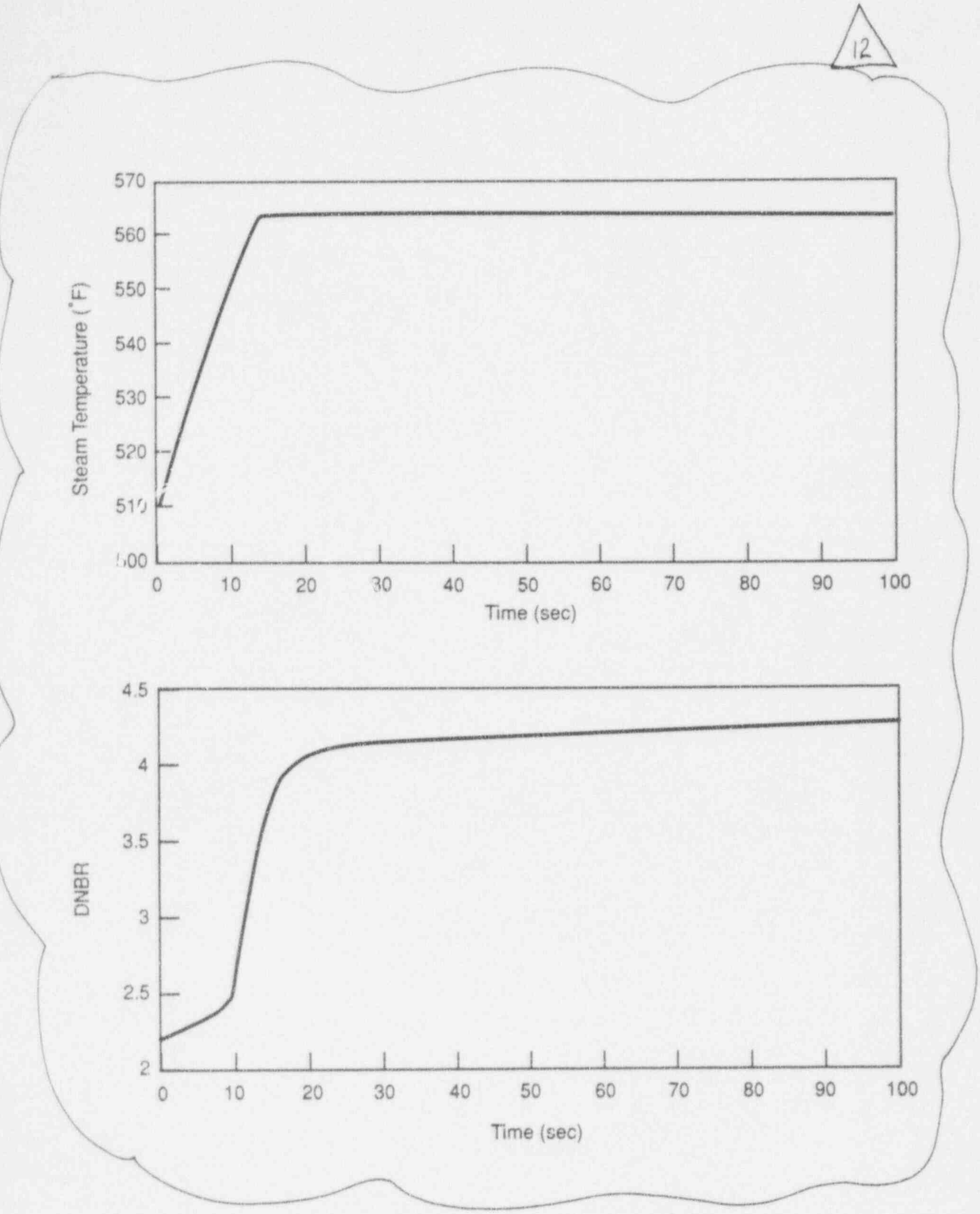
TABLE 15.2-1 (SHEET 4 OF 7)

Accident	Event	Time (s)
Loss of external electrical load		
Without pressurizer pressure control (BOL)	Loss of electrical load	0.0
	High presssurizer pressure reactor trip setpoint reached	6.1
	Rods begin to drop	8.1
	Peak pressurizer pressure occurs	10.2
	Initiation of steam release from steam generator safety valves	14.4
	Minimum DNBR occurs	a
Without pressurizer control (EOL)	Loss of electrical load	0.0
	High pressurizer pressure reactor trip setpoint reached	6.0
	Rods begin to drop	8.0
	Peak pressurizer pressure occurs	10.1
	Initiation of steam release from steam generator safety valves	14.7
	Minimum DNBR occurs	a



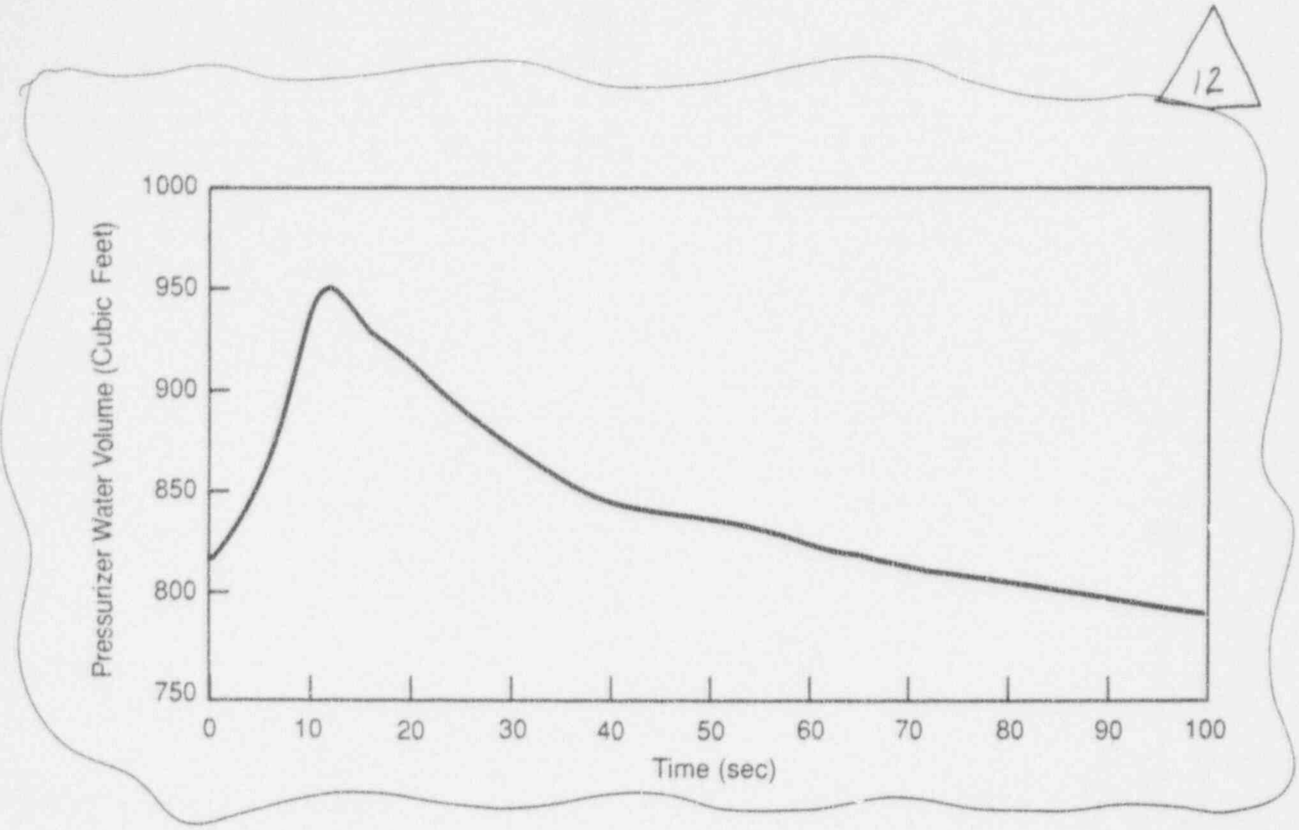
REV 12 10/94

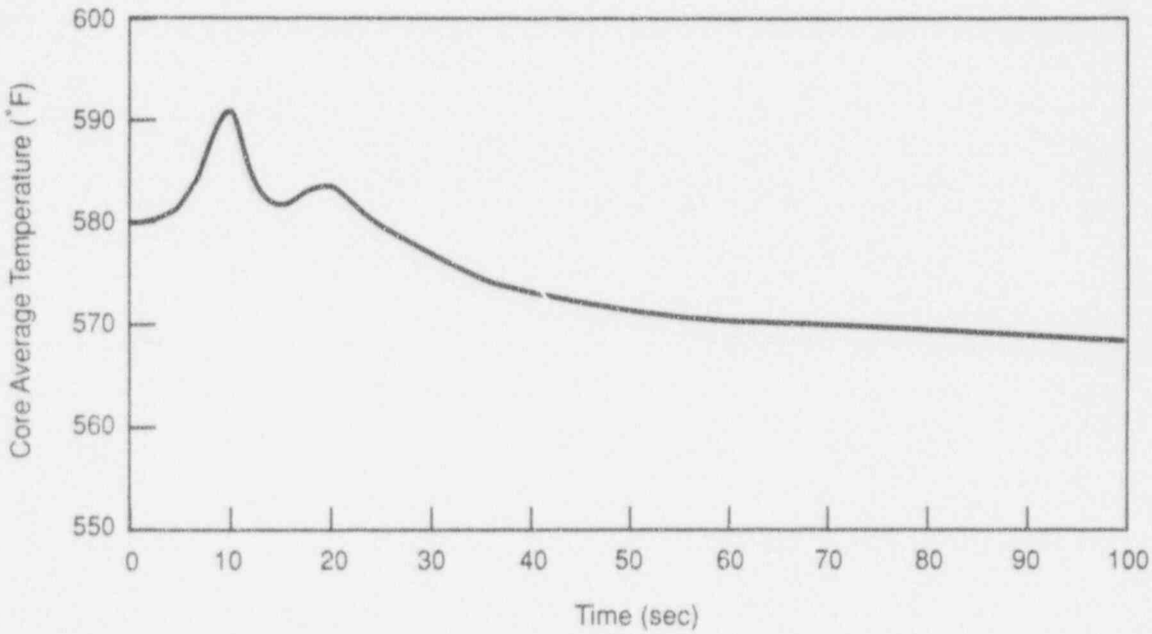
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	LOSS-OF-LOAD ACCIDENT WITHOUT PRESSURIZER SPRAY AND POWER-OPERATED RELIEF VALVES - BOL
	FIGURE 15.2-23 (SHEET 1 OF 2)



REV 12 10/94

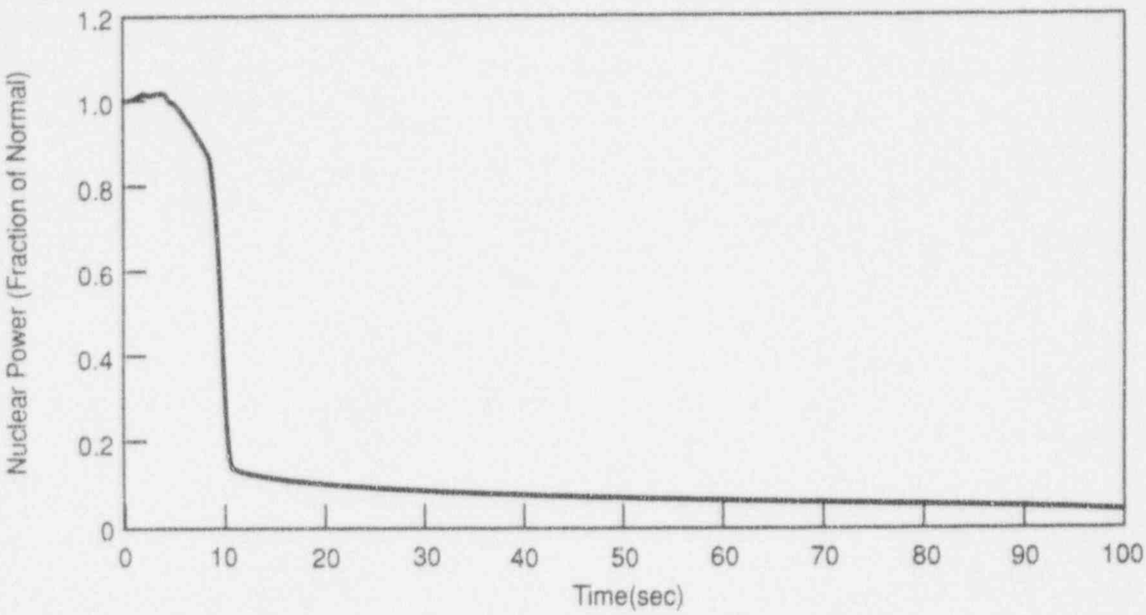
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	LOSS-OF-LOAD ACCIDENT WITHOUT PRESSURIZER SPRAY AND POWER-OPERATED RELIEF VALVES - BOL
	FIGURE 15.2-23 (SHEET 2 OF 2)

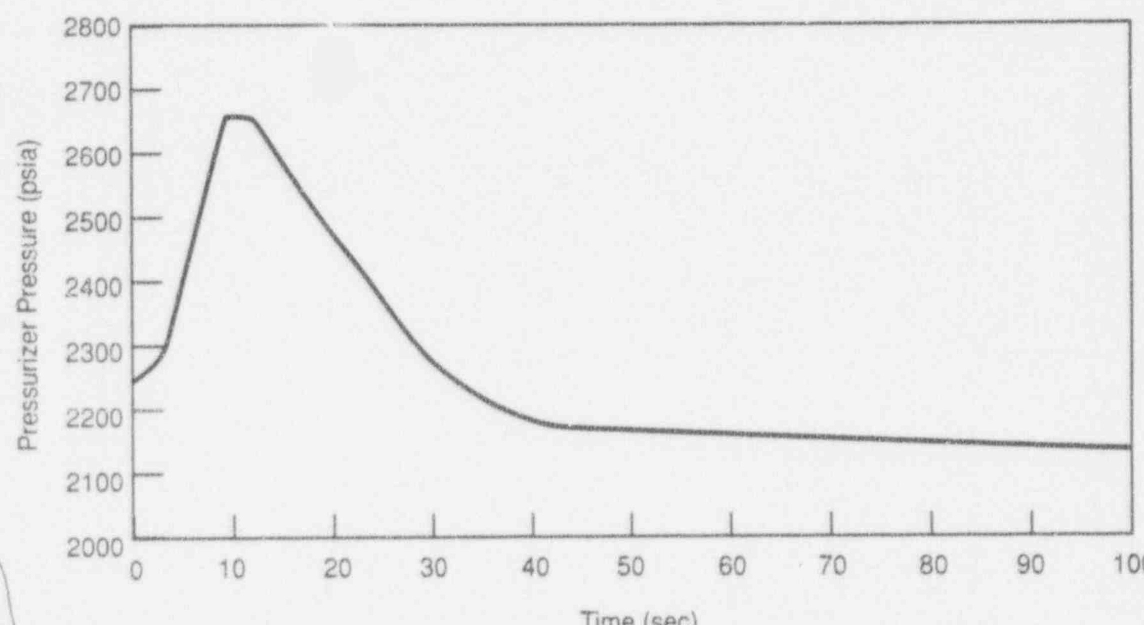




REV 12 10/94

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	LOSS-OF-LOAD ACCIDENT WITHOUT PRESSURIZER SPRAY AND POWER-OPERATED RELIEF VALVES - BOL
	FIGURE 15.2-24

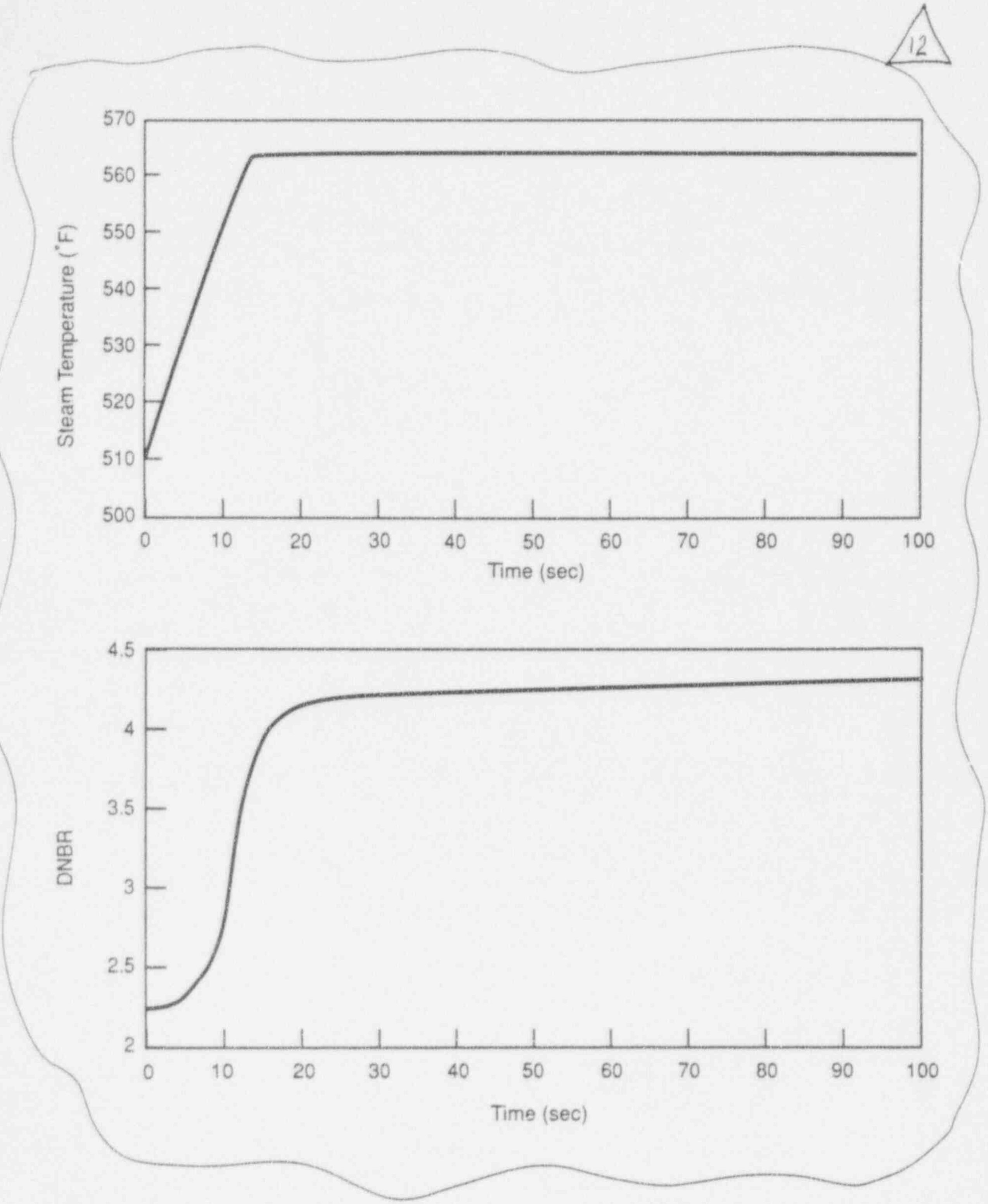




12

REV 12 10/94

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	LOSS-OF-LOAD ACCIDENT WITHOUT PRESSURIZER SPRAY AND POWER-OPERATED RELIEF VALVES - EOL
	FIGURE 15.2-25 (SHEET 1 OF 2)

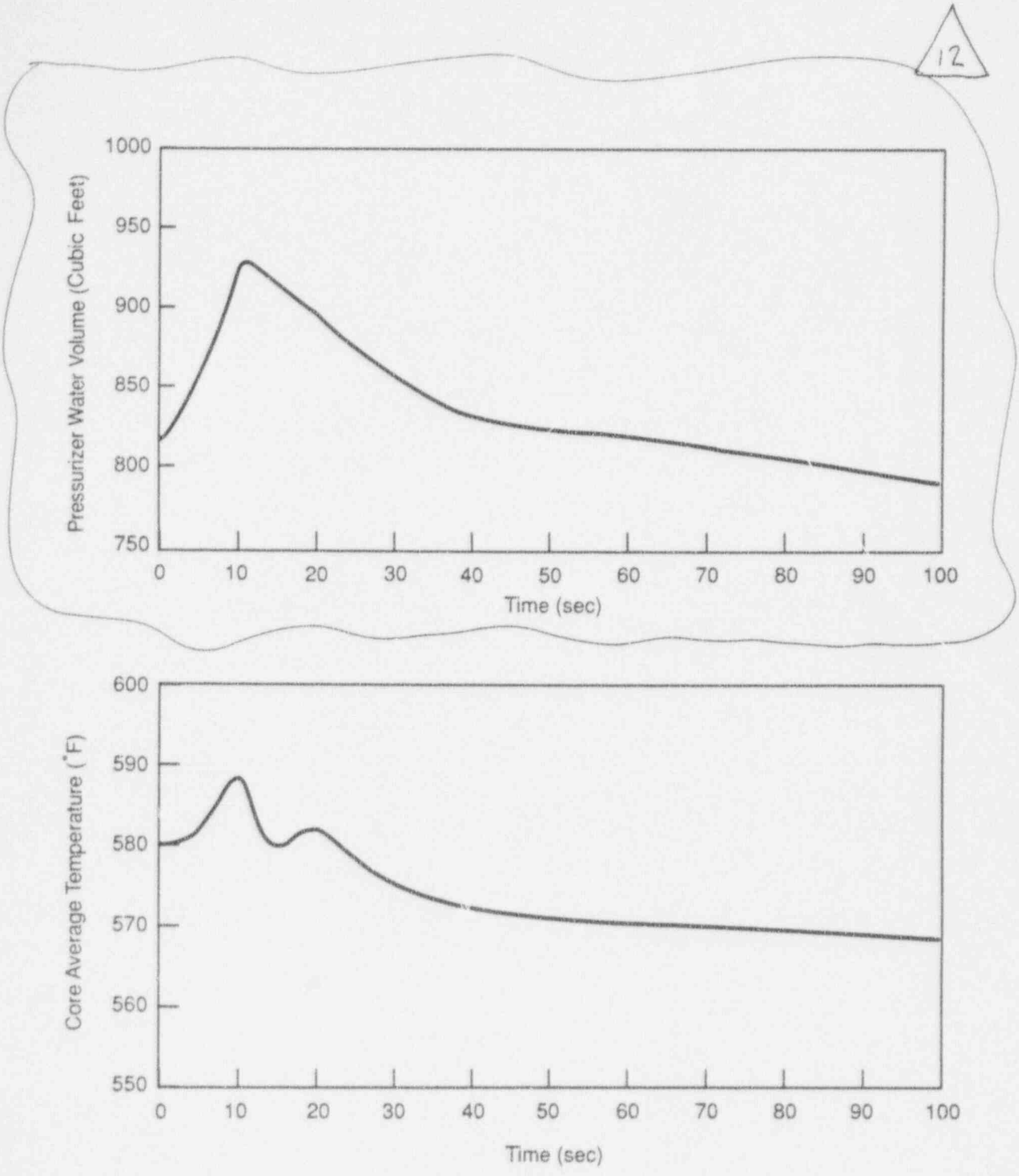


REV 12 10/94

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

LOSS-OF-LOAD ACCIDENT WITHOUT PRESSURIZER SPRAY AND POWER-OPERATED RELIEF VALVES - EOL

FIGURE 15.2-25 (SHEET 2 OF 2)



REV 12 10/94

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

LOSS-OF-LOAD ACCIDENT WITHOUT
PRESSURIZER SPRAY AND
POWER-OPERATED RELIEF VALVES - EOL

FIGURE 15.2-26

During the initial period of the small break transient the effect of the break flowrate is not strong enough to overcome the flowrate maintained by the RCPs as the pumps coast down following LOSP. Normal upward flow is maintained through the core and core heat is adequately removed. At the low heat generation rates following reactor trip, the fuel rods continue to be well cooled as long as the core is covered by a two-phase mixture level. From the clad temperature transients for the limiting break (3-in. break) calculations shown in figures 15.3-6A and 15.3-6B, it is seen that the peak clad temperature occurs near the time when the core is most deeply uncovered and the top of the core is steam cooled. This time is accompanied by the highest vapor superheating above the mixture level. The peak clad temperature attained for Unit 1 during the transient was 1935°F (including a 20°F penalty for increased T_{avg} uncertainty). For Unit 2, the 3-in. diameter cold leg break case yielded a peak clad temperature of 1896°F (also including the 20°F penalty mentioned above). At the time the transient was terminated, the safety injection flowrate that was delivered to the RCS exceeded the mass flowrate out the break in each case. Although the core mixture level has not yet covered the entire core in Unit 2 (see figure 15.3-5B), there is no longer a concern of exceeding the 10 CFR 50.46 criteria since the RCS pressure is gradually decaying and there is a net mass inventory gain. The decreasing RCS pressure results in greater safety injection flow as well as reduced break flow. As the RCS inventory continues to gradually increase, the core mixture level will continue to increase and the fuel clad temperatures will continue to decline.

Additionally, only one core channel is modeled in the NOTRUMP computer code since the core flowrate during a small break LOCA is relatively slow. This provides enough time to maintain flow equilibrium between fuel assemblies (i.e., no crossflow). Therefore, hydraulic resistance mismatch is not a factor for small break LOCA, it is not necessary to perform a small break LOCA evaluation for transition cores, and it is sufficient to reference the small break LOCA for the complete core of the VANTAGE 5 fuel design as bounding for all transition cycles.

15.3.1.2.2.2 <u>Additional Break Cases</u> - Studies documented in reference 3 determined that the limiting small break size occurred for breaks less than 10 in. in diameter. To ensure that the 3-in. diameter break was limiting, calculations were run with breaks of 2 in., 4 in., and 6 in. for the upflow configurations. To ensure that the upflow configuration was limiting, calculations were performed at the limiting break size for the downflow configuration. The results

of these calculations are shown in the Results table (15.3-2A) and the Sequence of Events table (15.3-2B).

For all cases analyzed, plots of the following transient parameters are presented:

- RCS pressure.
- Core mixture level.
- Hot rod clad average temperature.

These plots are shown in figures 15.3-4C through 15.3-6C for the 2-in. break, figures 15.3-4D through 15.3-6D for the 4-in. break, and figures 15.3-4E through 15.3-6E for the 6-in. break. As seen in table 15.3-2A, the peak clad temperatures in all cases were calculated to be less than that for the 3-in. break, upflow configuration.

15.3.1.3 Conclusions

Analyses presented in this subsection show that one low-head RHR pump and one high-head safety injection pump, together with the accumulators, provide sufficient core flooding to keep the calculated peak clad temperatures below the NRC acceptance criteria of 2200°F, as specified by 10 CFR 50.46 Adequate protection is therefore afforded by the ECCS in the event of a small break LOCA.

15.3.1.3.1 Breaks During Startup and Shutdown

During startup and shutdown, studies have shown that for breaks less than 2 in., manual initiation of safety injection may be required. The studies also show that ample time exists for the operator to take such action (see figures 15.3-26, 15.3-27, 15.3-28, and 15.3-29).

For both cases discussed, the indicators and alarms mentioned would function to alert the operator to the malfunction before DNB would occur. For case B discussed above, the insertion limit alarms (low and low-low alarms) would also serve in this regard.

REFERENCES

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water-Cooled Nuclear Power Reactors," Federal Register 39:3, 10 CFR 50.46 and 10 CFR 50, Appendix K, January 4, 1974, including 10 CFR 50.46 as amended September 16, 1988.
2. Meyer, P. E., "NOTRUMP - A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A, August 1985.
3. Lee, N., et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A, August 1985.
4. Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," NUREG 0611, January 1980.
5. Bordelon, F. M., et al., "LOCTA-IV Program: Loss of Coolant Transient Analysis," WCAP-8305, June 1974.
6. "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.
7. NRC Generic Letter 83-35 from D. G. Eisenhut, "Clarification of TMI Action Plan Item II.K.3.31," November 2, 1983.
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9. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A, April 1984.
10. Hargrove, H. G., "FACTRAN, A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.
11. Friedland, A. J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A, April 1989.
12. Letter from J. D. Woodward of Southern Nuclear Operating Company to Document Control Desk of the Nuclear Regulatory Commission, "Joseph M. Farley Nuclear Plant, 10 CFR 50.46 Annual ECCS Evaluation Model Change Report for 1992," March 16, 1993.
13. Letter from D. N. Morey of Southern Nuclear Operating Company to Document Control Desk of the Nuclear Regulatory Commission, "Joseph M. Farley Nuclear Plant, Peak Clad Temperature (PCT) Calculation," October 29, 1993.
14. Letter from D. N. Morey of Southern Nuclear Operating Company to Document Control Desk of the Nuclear Regulatory Commission, "Joseph M. Farley Nuclear Plant, 10 CFR 50.46 Annual ECCS Evaluation Model Change Report for 1993," March 18, 1994.

TABLE 15.3-2A

SMALL BREAK LOSS-OF-COOLANT ACCIDENT CALCULATION

Results

Parameter	Unit 1				Unit 2
	Case A 2-in. Upflow	Case B 3-in. Upflow	Case C 4-in. Upflow	Case D 6-in. Upflow	Case E 3-in. Downflow
Peak clad temperature (°F)	1071	1935[a][b]	1458	1771	1896[a][b]
elevation (ft)	11.5	11.75	11.5	11.25	11.5
Zr/H_2O cumulative reaction					
Local maximum (%)	0.08	2.36	0.31	0.88	1.52
elevation (ft)	11.5	11.75	11.5	11.25	11.75
Total core (%)	< 1.0	< 1.0	< 1.0	< 1.0	< 1.0
Rod Burst	None	None	None	None	None

a. These peak clad temperatures include a 20°F penalty due to an increased T_{avg} uncertainty of ± 6°F.

b. Includes the following benefit/penalty assessments:
+ 72°F - Penalty for Bessell function error (reference 12).
+ 150°F - Penalty for effect of SI in broken loop (reference 13).
- 150°F - Benefit for effect of improved condensation model (reference 13).
- 13°F - Benefit for correction of drift flux flow regime errors (reference 13).
+ 52°F, Unit 1 - Penalty for burst and blockage/time in life (reference 14).
+ 35°F, Unit 2 - Penalty for burst and blockage/time in life (reference 14).
+ 107°F - Penalty for charging/SI miniflow assumptions (reference 14).
+ 5°F, Unit 1 - Penalty for average rod burst strain (reference 14).
+ 4°F, Unit 2 - Penalty for average rod burst strain (reference 14).
- 5°F, Unit 1 - Benefit for fuel rod burst strain limit (reference 14).
- 4°F, Unit 2 - Benefit for fuel rod burst strain limit (reference 14).
- 16°F - Benefit for LUCIFER error corrections (reference 14).

TABLE 15.3-3

PARAMETERS USED IN WASTE GAS DECAY TANK RUPTURE ANALYSES

	Realistic Analysis	Conservative Analysis
Core thermal power	2766.00 MWt	2766.00 MWt
Plant load factor	1.00	1.00
Fuel defects	0.12%[a]	1.00%
Activity released from GWPS	Contents of one tank	Contents of one tank
Number of tanks (normal operation)	6.00	6.00
Switching time between any 2 tanks	1.00 week	2.00 days
Operation of gaseous waste processing system	Continuous[c]	Continuous[c]
Stripping fraction in volume control tank	0.40	1.00
Iodine partition factor in volume control tank	0.01	0.01
Time of accident	Immediately after isolation of tank from GWPS	Immediately after isolation of tank from GWPS
Meteorology	Annual average averaged overall sectors[b]	Accident (see appendix 15B)

a. Based on operating experience of Westinghouse pressurized-water reactors (PWRs).

b. 2.7×10^{-6} s/m^3 at site boundary; 6.9×10^{-7} s/m^3 at low-population zone.

c. Continuous purging of the VCT to the GWPS is a conservative assumption.

The initial steady-state fuel pellet temperature and fuel rod internal pressure used in the LOCA analysis were generated with the PAD 3.4 Fuel Rod Design Code (reference 12) which has been approved by the NRC.

A full spectrum break analysis was done in order to justify plant operation at 2652 MWt at initial RCS pressurizer pressure of 2310 psia and initial hot leg temperature of 611.3°F, from which the limiting break size was determined. In addition, the analyses conservatively modeled a downflow barrel baffle configuration. However, the limiting break size was also analyzed with an upflow barrel baffle configuration, applicable to Unit 1, which demonstrated that the downflow barrel baffle configuration remained limiting.

The upflow barrel baffle design would not cause the limiting break size to shift since there is only an incremental difference in the core flow during blowdown as a result of the different barrel baffle configurations. Since a larger discharge coefficient has substantially better cooling during the blowdown calculation, an incremental difference in the core flow due to the barrel baffle configuration will not significantly alter the clad temperature response during blowdown. In addition, the barrel baffle configuration affects the calculation of the reflood portion of the transient through vessel liquid level changes. The flow of steam through the loop and out the rupture will only be affected by the amount of steam generated in the reflooding and quenching process.

Also, previous Farley-specific analyses, which performed a full spectrum analysis using the upflow barrel baffle design, demonstrated that the limiting break size was the same as the downflow barrel baffle configuration; therefore, a complete spectrum assuming the upflow barrel baffle configuration was not performed in this analysis since the barrel baffle configuration will not change the limiting break size.

All cases conservatively assumed 20-percent steam generator tube plugging (17 percent for Unit 2 Cycle 10) in all three steam generators and an 8-percent degradation of both the RHR and high-head SI pumps with a 10-gal/min high-head SI flow imbalance. Table 15.4-1 describes the cases analyzed.

The bases used to select the numerical values that are input parameters to the analysis have been conservatively determined from extensive sensitivity studies (Westinghouse, 1974 (reference 13); Salvatori, 1974 (reference 14); and Johnson, Massie, and Thompson, 1975 (reference 15)). In addition, the requirements of Appendix K to 10 CFR 50 (reference 1) regarding specific model features were met by selecting models which provide a significant overall conservatism in the analysis.

The assumptions which were made pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs and include such items as the core peaking factors, the containment pressure, and the performance of the ECCS. Decay heat generated throughout the transient is also conservatively calculated as per the requirements of Appendix K to 10 CFR 50.[1]

Another input parameter that affects LOCA analysis results is the assumed axial power shape at the beginning of the accident. Power shape sensitivity studies performed with Westinghouse ECCS evaluation models have always demonstrated the chopped cosine shape with the peak at the core midplane to be limiting. Westinghouse has performed "spot check" analyses using the BASH reflood evaluation model for power shapes skewed to the top of the core. Results of these analyses have demonstrated that the chopped cosine peaked at the core midplane remains the limiting power shape.[16]

15.4.1.4.5 Additional Break Cases

Westinghouse ECCS analyses currently assume minimum safeguards for the SI flow, which minimizes the amount of flow to the RCS by assuming maximum injection line resistances. However, for some Westinghouse plants, including Farley Nuclear Plant Units 1 and 2, the current nature of the Appendix K ECCS evaluation model is such that it may be more limiting to assume the maximum possible ECCS flow delivery. In that case, maximum safeguards, which assume operation of both trains of RHR (low-head) and high-head SI pumps, minimum injection line resistances and enhanced ECCS pump performance result in the highest amount of flow delivered to the RCS; therefore, the worst break for the Farley Nuclear Plant Units 1 and 2 was reanalyzed, assuming maximum safeguards. Examination of the LOCA analysis results in table 15.4-6 demonstrates that minimum safeguards assumptions result in the limiting peak clad temperature for Farley Nuclear Plant Units 1 and 2.

The worst break for the Farley Nuclear Plant Units 1 and 2 was also analyzed for this first transition core assuming 17 x 17 LOPAR fuel input parameters (the current hot channel enthalpy rise factor of 1.55 and a total core peaking factor of 2.32 were retained). In addition, a conservative assumption of 4000 MWd/Mtu minimum burnup for the 17 x 17 LOPAR fuel and a maximum of 72 fresh 17 x 17 VANTAGE 5 fuel assemblies was used in the analysis. Since 17 x 17 VANTAGE 5 and 17 x 17 LOPAR fuel have different overall hydraulic resistances and grid effects, both the hydraulic transient and the clad heatup transient were

Figures 15.4-11A-E	The fluid quality as calculated by LOCBART is shown.
Figures 15.4-12A-E	The fluid velocity is plotted as the hot spot (the node which produced the peak clad temperature) on the hot rod.
Figures 15.4-13A-E	The heat transfer coefficient is plotted at the hot spot on the hot rod.
Figures 15.4-14A-E	The fluid temperature at the hot spot on the hot rod is plotted.
Figures 15.4-15A-E	The clad temperature at the hot spot is shown for the hot rod.
Figures 15.4-16A-E	The containment backpressure transient used in the analysis.
Figures 15.4-17A-B	The containment condensing wall heat transfer coefficient.

The limiting peak clad temperature calculated for the Unit 1 17-x-17 VANTAGE 5 large break is 2002°F, which is less than the acceptance criteria limit of 2200°F. This value includes 3°F for containment minipurge auto isolation, 8°F for increased temperature uncertainty, and 6°F for combined safe shutdown earthquake (SSE) and LOCA events. Addition of the 50°F transition penalty yields a transition core peak clad temperature of 2032.1°F. This transition core penalty can be accommodated by the 2200°F acceptance criterion of 10 CFR 50.46. The maximum local metal-water reaction is 4.96 percent, which is well below the embrittlement limit of 17 percent as required by 10 CFR 50.46.

The limiting peak clad temperature calculated for the Unit 2 17-x-17 VANTAGE 5 large break is 2110°F, which is less than the acceptance criteria limit of 2200°F. This value includes 3°F for containment minipurge auto isolation, 8°F for increased temperature uncertainty, and 6°F for combined SSE and LOCA events. Addition of the 50°F transition penalty yields a transition core peak clad temperature of 2140.3°F. This transition core penalty can be accommodated by the 2200°F acceptance criterion of 10 CFR 50.46. The maximum local metal-water reaction is 6.59 percent, which is well below the embrittlement limit of 17 percent as required by 10 CFR 50.46.

The total core metal-water reaction is less than 1.0 percent for all breaks analyzed, corresponding to less than 1.0-percent hydrogen generation, as compared with the 1-percent criterion of 10 CFR 50.46. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

15.4.1.6 Hydrogen Production and Accumulation

Hydrogen accumulation in the containment atmosphere following the design basis accident (DBA) can be the result of production from several sources. The potential sources of hydrogen are the zirconium-water reaction, corrosion of construction materials, and radiolytic decomposition of the emergency core cooling solution. The latter source, solution radiolysis, includes both core solution radiolysis and sump solution radiolysis.

15.4.1.6.1 Method of Analysis

The quantity of zirconium which reacts with the core cooling solution depends on the performance of the ECCS. The criteria for evaluation of the ECCS require that the zircaloy-water reaction be limited to 1 percent by weight of the total quantity of zirconium in the core. ECCS calculations have shown the zircaloy-water reaction to be less than 0.1 percent, much less than required by the criteria.

The use of aluminum inside the containment is limited, and is not used in safety-related components which are in contact with the recirculating core cooling fluid. Aluminum is more reactive with the containment spray alkaline borate solution than with other plant materials such as galvanized steel, copper, and copper-nickel alloys. By limiting the use of aluminum, the aggregate source of hydrogen over the long term is essentially restricted to that arising from radiolytic decomposition of core and sump water. The upper limit rate of such decomposition can be predicted with ample certainty to permit the design of effective countermeasures.

It should be noted that the zirconium-water reaction and aluminum corrosion with containment spray are chemical reactions and, thus, essentially independent of the radiation field inside the containment following a LOCA. Radiolytic decomposition of water is dependent on the radiation field intensity.

3. Low-low steam generator water level in any steam generator.

4. Safety injection signals from either of the following:

   a. Two of three low-pressurizer pressure signals.

   b. Two of three high-differential pressure signals between any steam line and remaining steam lines.

   c. Low main steam line pressure in any two lines.

   d. Two of three high containment pressure.

(Refer to chapter 7 for a description of the actuation system.)

B. An AFW system to provide an assured source of feedwater to the steam generators for decay heat removal. (Refer to chapter 6 for a description of the AFW system.)

15.4.2.2.2 Analysis of Effects and Consequences

15.4.2.2.2.1 <u>Method of Analysis</u> - A detailed analysis using the LOFTRAN[30] code is performed in order to determine the plant transient following a feedline rupture. The code describes the plant thermal kinetics and the RCS, including natural circulation, pressurizer, steam generators, and feedwater system; and computes pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

The LOFTRAN code is used to calculate the course of the system transient through the time that the auxiliary feedwater system heat removal capacity exceeds decay heat generation.

<u>Case A</u>

The major assumptions for the major feedwater rupture analysis are as follows:

A. The plant is initially operating at 102 percent of the engineered safeguards design rating.

B. Initial reactor coolant average temperature is 6°F above the nominal value, and the initial pressurizer pressure is 60 psi above its nominal value.

C. No credit is taken for the pressurizer power-operated relief valves or pressurizer spray.

D. No credit is taken for the high-pressurizer pressure reactor trip. (Note: This assumption is made for calculational convenience.) Pressurizer power-operated relief valves and spray could act to delay the high-pressure trip. Assumptions C and D permit evaluation of one hypothetical limiting case rather than two possible cases: one with a high-pressure trip and no pressure control, and one with pressure control but no high-pressure trip.

E. Main feed to all steam generators is assumed to stop at the time the break occurs.

F. Saturated liquid discharge (no steam) is assumed from the affected steam generator through the feedline rupture. This assumption minimizes energy removal from the NSSS during blowdown.

G. No credit is taken for the low-low water level trip on the affected steam generator until the steam generator water level reaches 0 percent of the narrow range span.

H. The worst possible break area is assumed; i.e., one that empties the affected steam generator and causes a reactor trip on low-low steam generator water level as assumed above. This assumption minimizes the steam generator fluid inventory at the time of trip, and thereby maximizes the resultant heatup of the reactor coolant.

I. No credit is taken for heat energy deposited in reactor coolant system metal during the reactor coolant system heatup.

J. No credit is taken for charging or letdown.

K. Loss of offsite electrical power is assumed after the reactor trip, and reactor coolant flow decreases to natural circulation.

L. Steam generator heat transfer area is assumed to decrease as the shell-side liquid inventory decreases.

40. Nuclear Regulatory Commission, Directorate of Regulatory Standards, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized-Water Reactors," Regulatory Guide 1.77, May 1974.

41. Nuclear Regulatory Commission, Division of Reactor Standards, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized-Water Reactors," Regulatory Guide 1.25, March 23, 1972.

42. Barry, R. F. and Risher, D. H., Jr., "TWINKLE, a Multidimensional Neutron Kinetics Computer Code," WCAP-7979-P-A, January 1975.

43. "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," NUREG/CR 5009, February 1988.

44. Letter from J. D. Woodward of Southern Nuclear Operating Company to Document Control Desk of the Nuclear Regulatory Commission, "Joseph M. Farley Nuclear Plant, 10 CFR 50.46 Annual ECCS Evaluation Model Change Report for 1992," March 16, 1993.

45. Letter from D. N. Morey of Southern Nuclear Operating Company to Document Control Desk of the Nuclear Regulatory Commission, "Joseph M. Farley Nuclear Plant, 10 CFR 50.46 Annual ECCS Evaluation Model Change Report for 1993," March 18, 1994.

TABLE 15.4-2

INPUT PARAMETERS USED IN THE LARGE BREAK LOCA ECCS ANALYSIS

	Barrel Baffle Upflow (Unit 1)	Configuration Downflow (Unit 2)
License core power[(a)] (MWt)	2652	2652
Peak linear power[(a)] (kW/ft)	12.749	12.749
Total peaking factor, F_Q	2.45	2.45
Axial peaking factor, F_Z	1.485	1.485
Hot channel enthalpy rise factor, $F_{\Delta H}$	1.65	1.65
Hot assembly average power, P_{HA}	1.4426	1.42
Power shape	Chopped Cosine	
Fuel assembly array	17 x 17 VANTAGE 5	
Intact accumulator water volume (ft^3/accumulator)	2121 (Min)	2121 (Min)
Broken accumulator water volume (ft^3/accumulator)	1086 (Nom)	1086 (Nom)
Intact accumulator tank volume (ft^3/accumulator)	3001	3001
Broken accumulator tank volume (ft^3/accumulator)	1511	1511
Accumulator gas pressure, minimum (psia)	600	600
Safety injection pumped flow (all pumps degraded 8%, HHSI flow imbalance = 10 gal/min)	See figures 15.4-1 and 15.4-2A and -2B	
Containment parameters	See table 15.4-4	
Initial loop flow (gal/min)	86000	86000
Vessel inlet temperature (°F)	540.52	540.52
Vessel outlet temperature (°F)	611.28	611.28
Average reactor coolant pressure (psia)	2310.0	2310.0
Steam pressure (psia)	747.99	747.99
Steam generator tube plugging level (%)	20	20[(b)]
Minimum RWST temperature (°F)	35.0	35.0
Fuel Backfill pressure (psig)	275	275
Low pressurizer pressure setpoint (psia)		
Reactor trip	1840	1840
Safety injection signal	1715	1715
Safety injection delay time (s)	27	27
Safety injection spilling containment pressure (psig)		
Blowdown/refill	20.0	20.0
Reflood	11.5	11.5
Blowdown containment pressure (psia)	35.0	35.0

a. Two percent is added to this power to account for calorimetric error.
b. 17% for Unit 2 Cycle 10.

TABLE 15.4-3

LARGE BREAK LOCA ECCS ANALYSIS SYSTEMS MODELING

Pressurizer low pressure reactor trip (psia)	1840.0
Pressurizer low pressure safety injection (psia)	1715.0[a]
Containment high pressure for safety injection (psia)	22.0
Safety injection delay (includes signal processing, EDGs startup, sequencer, and pumps to full speed) (s)	27.0
Feedwater isolation delay after reactor trip (s)	0.0[b]
Steam line isolation delay after reactor trip (s)	0.0[b]
Number of HHSI pumps operating (min/max safeguards)	1/3[c]
Number of LHSI pumps operating (min/max safeguards)	1/2[c]
Steam generator tube plugging	20%[d]

a. This setpoint causes actuation of the safety injection at the times shown in table 15.4-5 for all five cases.

b. Conservative modeling for large break LOCA.

c. Minimum safeguards assumes one high-head safety injection pump and one RHR pump operating. Maximum safeguards assumes three high-head pumps and two RHR pumps operating.

d. Uniform 20-percent steam generator tube plugging assumes 20-percent steam generator tubes plugged in each steam generator and corresponds to the peak plugging level in any steam generator (17 percent for Unit 2 Cycle 10). This configuration bounds all combinations of nonuniform plugging for LOCA as long as no one steam generator plugging level exceeds 20 percent (17 percent for Unit 2 Cycle 10). However, it should be noted that the licensed steam generator plugging level is limited to 15-percent average/20-percent peak (15-percent average/17-percent peak for Unit 2 Cycle 10) in any one steam generator.

TABLE 15.4-6

LARGE BREAK LOCA RESULTS FUEL CLADDING DATA

		Unit 1	Unit 2			
		Case A $C_D = 0.4$ Min Safeguards Upflow 2652 MWt	Case B $C_D = 0.4$ Min Safeguards Downflow 2652 MWt	Case C $C_D = 0.6$ Min Safeguards Downflow 2652 MWt	Case D $C_D = 0.8$ Min Safeguards Downflow 2652 MWt	Case E $C_D = 0.4$ Max Safeguards Downflow 2652 MWt
Peak clad temperature	(°F)	2002[a][b]	2110[a][b]	1711.0	1629.3	1929.4
Peak clad temperature location	(ft)	7.25	7.25	6.25	6.25	6.25
Peak clad temperature time	(s)	81.5	106.1	54.2	52.7	58.9
Local Zr/H_2O reaction maximum	(%)	4.96	6.59	1.64	1.07	4.00
Local Zr/H_2O reaction location	(ft)	6.00	6.00	5.50	7.25	6.00
Total Zr/H_2O reaction	(%)	< 1.0	< 1.0	< 1.0	< 1.0	< 1.0
Hot rod burst time	(s)	36.6	38.6	48.6	NA	38.6
Hot rod burst location	(ft)	6.00	6.00	5.50	NA	6.00

a. Includes the following effects: Temperature uncertainty = 8.0°F; containment minipurge isolation = 3.0°F; combined SSE and LOCA events = 6.0°F. A 50°F transition core penalty core has been added to the VANTAGE 5 results until all LOPAR fuel is removed.

b. Includes the following benefit/penalty assessments:
-25°F - Benefit for structural metal heat modeling correction (reference 44).
-6°F - Benefit of LUCIFER error corrections (reference 45).

TABLE 15.4-8 (SHEET 1 OF 2)

PLANT PARAMETERS FOR CALCULATING POST-ACCIDENT HYDROGEN GENERATION

Thermal power rating (MW)	2,766
Containment free volume (ft^3)	2,030,000
Maximum normal containment temperature (°F)	120
Weight zirconium cladding (lb)	39,003
Zirconium cladding reacted (%)	2
Hydrogen produced by zirconium-water (sf^3)	6,162

Aluminum Inventory

	Weight (lb)	Surface Area (ft^2)
Source, inter., and power-range detectors	244.0	83
Flux mapping drive system	171.0	75
CRDM Connectors	168.0	37
Paint[(a)]	125.0	14,000
Rod position indicator	139.3	79
Reactor coolant pump motors	393.00	38.4
Miscellaneous valves	230.0	86

TABLE 15.4-8 (SHEET 2 OF 2)

Aluminum Inventory (continued)

	Weight (lb)	Surface Area (ft^2)
MOV actuators: SMB-000 and SMB-00	30.00	9
Contingency[b]	200.0	75
Total	1700.3	14,482.4

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a. Shielded by insulation from spray; however, no credit was assumed in hydrogen calculations for shielding from spray.

b. Epoxy-coated safeguard motor internals 400 lb/150 ft^2 are not considered in the analysis. Epoxy integrity was demonstrated as part of a long-term qualification test for safeguard motors located inside containment (see section 3.11). The 200-lb/75-ft^2 contingency accounts primarily for epoxy coated motor internals of nonsafeguard motors which were not demonstrated by long-term tests.

16.0 - TECHNICAL SPECIFICATIONS

The technical specifications for the Farley plant were developed for Unit 1 and Unit 2 from the NRC's Standard Technical Specifications for Westinghouse Pressurized Water Reactors. The technical specifications for the Farley plant were issued on Unit 1 and Unit 2 as Appendix A to the Operating License by the NRC.

The NRC acceptance criteria for the technical specifications are that normal plant operation will not result in potential offsite exposures in excess of the 10 CFR Part 20 limits, and that necessary engineered safety features will be available in the vent of accidents to keep potential offsite doses within the .0 CFR Part 100 guidelines. For nuclear power plants, compliance with the limits of 10 CFR 20.1301 for individual members of the public may be demonstrated by complying with the limits of 10 CFR 50, Appendix I, and 40 CFR 190.

17.2 OPERATIONS QUALITY ASSURANCE PROGRAM (OQAP)

The Operations Quality Assurance Program (OQAP) for the Farley Nuclear Plant (FNP) is designed to assure the plant's safe and reliable operation and to satisfy the quality assurance (QA) requirements of Appendix B of 10 CFR 50 as delineated in Regulatory Guide 1.33, dated November 3, 1972, "Quality Assurance Program Requirements (Operation)."

This regulatory guide requires compliance with the stipulations of ANSI N45.2-1971 and ANS 3.2 (now ANSI N18.7-1972). Compliance with these requirements constitutes administrative controls for the operation of nuclear power plants in a manner that is consistent with applicable criteria for QA.

Section 17.2 is written to address each of the 18 criteria of 10 CFR 50, Appendix B, as amplified by ANSI N45.2-1971. The following is a cross-reference between ANSI N18.7-1972 requirements and FSAR sections that describe compliance with these requirements (any exceptions are noted):

N18.7-1972 Requirements	FNP FSAR Sections
1. Scope	17.2 Introduction
2. Definitions	Consistent with terminology used in 17.2.
3. Owner Organization	
a. Corporate	13.1.1, 17.2.1, 17.2.2
b. Operations	13.1.2, 17.2.1, 17.2.2
4. Review and Audit Program*	13.4.2, 17.2.1, 17.2.18
5. Facility Administrative Policies and Procedures	
a. Rules of Practice	13.5, 13.7
b. Plant Records Management	13.6, 13.7, 17.2.17
c. Operating and Maintenance Procedures	13.5
d. Review and Approval of Procedures	13.4.2, 13.5
e. Temporary Procedures	13.5

*The independent review and audit group shall consist of at least four persons, vice the five required by ANSI N18.7-1972, 4.2.2.1.

ANSI N18.7-1972 Requirements	FNP FSAR Sections
6. Tests and Inspections	
a. Test Procedures	13.5, 14, 17.2.11
b. Tests and Inspections Prior to and During Plant Startup	14
c. Tests and Inspections after Plant Modifications	13.5.4, 17.2.11

In addition to that discussed above, the OQAP will comply with certain other codes, standards, and regulatory guides. The following is a cross-reference between such guidance and the FSAR sections that describe compliance with this guidance (any exceptions are noted):

	Guidance	FNP FSAR Sections
1.	10 CFR 50, Appendix B	17.2 Introduction
2.	Regulatory Guide 1.33, dated November 3, 1972 (Operations QA)	17.2 Introduction
3.	10 CFR 55	17.2.2
4.	Regulatory Guide 1.8, dated March 10, 1971 (Personnel Selection-ANSI N18.1, dated June 22, 1970; later approved as ANSI N18.1-1971)	13.1
5.	Regulatory Guide 1.58, Revision 1, dated September 1980 (Qualification of Inspection and Testing Personnel - ANSI N45.2.6-1978) with the exceptions identified on page 3A-1.58-1.	17.2.2, 17.2.10
6.	(deleted)	
7.	10 CFR 50.55a (Codes and Standards Rule)	3.2, 17.2.9

controls, degree to which functional compliance can be demonstrated by inspection and test, and the quality history of the item.

Items shall be identified by the total plant numbering system (TPNS), described in subsection 17.2.17, which provides traceability. Subsection 17.2.17 also describes the identifier used to identify and file records associated with the items.

Plant and General Office administrative procedures for procurement document control will be reviewed and approved by the MSAER. The general manager-nuclear support and the general manager-nuclear plant are responsible for review and approval of all procurement documents for safety-related materials, components, equipment, and services to verify inclusion of adequate quality requirements. The MSAER will periodically audit procurement activities performed by Farley Project personnel to verify inclusion of adequate quality requirements in procurement documents and to ensure that the vendor QA programs of vendors utilized are reviewed for acceptability by SNC Corporate Quality Services. Additionally, he will ensure that appropriate quality requirements are included in procurement documents for nuclear fuel and associated components and that reviews are made of the QA programs of nuclear fuel designers and fabricators. The MSAER is responsible for the auditing of the procurement document control activities performed by SNC-Farley Project personnel involving safety-related services, material, components, and equipment.

Revisions to procurement documents will be subject to the same review and approval as the original document.

17.2.5 INSTRUCTIONS, PROCEDURES, AND DRAWINGS

As required by the OQAP, all safety-related activities associated with FNP will be conducted in accordance with written approved procedures, instructions, and drawings that include acceptance critieria and necessary limits and tolerances appropriate to the activity. Such criteria may be in the form of quantitative requirements or checkoff lists to ensure that the activities have been successfully accomplished.

As described in subsection 17.2.1, the plant staff will provide guidance for all safety-related activities with instructions, procedures, and drawings which are found or referenced in FNP procedures.

Section 13.5 describes the method of developing, approving, and changing procedures for safety-related plant operations,

preventive and corrective maintenance, inspection, modification, and surveillance testing. The plant staff shall write, approve, and implement sufficient procedures to ensure the quality of FNP operations which, as a minimum, will include the applicable procedures of Appendix A of Regulatory Guide 1.33. The plant foremen will monitor safety-related plant operations to ensure that they are conducted in accordance with the appropriate approved procedures, instructions, and drawings.

The SAER staff performs its activities in accordance with written procedures and instructions approved by the MSAER. The SAER administrative procedures are listed in the OQAPIL. An SAER administrative procedure discusses the procedural development, review, approval, and change process utilized by the SAER staff. SAER audits plant activities to determine if the latest revisions of drawings, procedures, and instructions are being utilized by the plant staff members. If checklists are used by these personnel, SAER will verify that adherence to the checklist is properly noted by the user's initials or checkoffs. SAER will audit procedure, instruction, drawing approval methods and procedure changes to verify adherence to the administrative procedures. SAER will periodically audit records of procedure approval by the plant staff to verify compliance with applicable procedures listed in the OQAPIL and will selectively verify that the latest revision of drawings, instructions, and procedures is being utilized by persons performing safety-related activities. Results of these audits will be reported to the general manager-nuclear plant and the vice president.

17.2.6 DOCUMENT CONTROL

Prior to preoperational testing, quality documents generated in the design and construction of the FNP were reviewed and verified by the general contractor and audited by the QA group (Design and Construction) to be complete and in conformance with applicable codes, standards, and regulations. As described in subsection 17.2.17, administrative procedures have been written describing the method of controlling these documents and quality documentation generated during operation, maintenance, and modification of the FNP.

Administrative procedures written by the plant and general office staffs detail the method of developing, approving, issuing, distributing, and revising all safety-related documents.

SAER administrative procedures describe the method used by SAER to control documents. This procedure discusses filing, superseding, and revising documents the procedure

deviations, nonconformances, and correction of such items; assuring corrective action is identified and correlated with the original documentation of the condition; determining if such a condition is significant; reporting the condition and its resolution to management. The SAER staff will audit selected corrective action activities to verify that they are completed in accordance with the written procedures. During audits performed by the SAER staff, particular attention will be paid to the verification that significant conditions adverse to quality are properly identified, documented, and evaluated; actions are taken to preclude recurrence; corrective actions are timely; and proper management is kept informed of the situation. The SAER staff will perform periodic audits of the corrective action procedures and activities to ensure compliance with procedures listed in the OQAPIL.

17.2.17 QUALITY ASSURANCE RECORDS

Procedures written by the plant and nuclear generation general office staffs require the proper identification of documents which furnish objective evidence of activities affecting the quality of the FNP. Administrative procedures describe the method of writing, approving, and modifying procedures for the respective groups. Plant and nuclear generation general office staff members who develop operational procedures will use ANSI N45.2.9-1974 as a standard reference. The PORC and plant managers who review operations procedures will consider whether prospective procedures require the generation of records listed in ANSI N45.2.9-1974. Records to be developed include operating, maintenance, and testing records listed in Sections A.5 and A.6 of ANSI N45.2.9-1974 and records developed during the operations phase, as applicable, which are similar to those listed in Sections A.1 through A.5. Examples of these records are design change requests, procurement specifications, manufacturers' material properties records, nonconformance reports generated during receipt inspection, liquid penetrant test procedures, lubrication records, cable termination procedures, and component test procedures and results. Several records listed in ANSI N45.2.9-1974 will not be available at the FNP because procurement specifications and procedures for performing certain activities were written prior to the development of ANSI N45.2.9-1974 and did not require some of the records listed in Section A of ANSI N45.2.9-1974 to be generated and maintained. Similar activities performed during the operation of the FNP will require the generation of records listed in Sections A.1 through A.5 of ANSI N45.2.9-1974. The retention times for records maintained at the FNP comply with the requirements of ANSI N45.2.9-1974 and the FNP Technical Specifications.

Administrative procedures contain methods for identifying, filing, and maintaining records in accordance with the guidance provided in ANSI N45.2.9-1974. Most records will be identified by the assignment of the correct record type identifier by plant or nuclear generation general office staffs. The TPNS number, which was once used for filing all quality documentation, signifies the components, systems, separation classification, FNP unit, and unique identification. The quality documents to be generated and the duration of their plant retention are described in section 13.6. Other closely related data, such as personnel and equipment qualifications, will also be retained as quality records in the document control center. A signout system will ensure timely location of any record that has been temporarily removed from the files. The document control center has been generally designed so as to comply with the requirements of Section 5.6 of ANSI N45.2.9-1974. The doors, frames, and hardware have a 2-hour minimum Class A fire rating; one roof drain pipe is located above the ceiling of the storage area. The structure is designed using structural steel and masonry to withstand wind loads as required by the Uniform Building Code.

The MSAER, as discussed in subsection 17.2.1, reviews and approves administrative procedures that provide administrative guidance on QA records. He will verify that these procedures establish a records system which will require the generation of sufficient records to furnish evidence of activities affecting quality that comply with requirements of applicable codes, standards, and the OQAPM. This review will verify that a system is provided and that responsibilities are assigned for identifying, storing, and retrieving quality records and that proper distribution and review of records generated is required. The SAER staff will audit operations procedures which delineate records requirements to verify that the responsibility of a qualified recorder and reviewer is specified, that records required are detailed enough to reconstruct significant events, and that corrective action following the documentation of unsatisfactory data is required. The SAER staff will audit quality records to assure completeness, adequate identification of data recorder and reviewer, and the assignment of the correct identifier, where applicable.

The SAER staff will audit selected quality records to verify that the records have been properly generated in compliance with administrative procedures and detailed work procedures. They will audit the filing of quality records to determine if they are properly identified and retrievable. They will audit the records storage area to verify compliance with temperature and humidity requirements, that records are only accessible to authorized personnel, and that the records are not deteriorating because of handling.

Audits shall be performed in accordance with an audit plan which delineates activities to be covered, coverage of the audit, and audit scheduling. Audits shall be performed in sufficient depth to determine that a certain element of the OQAP has been effectively implemented. The frequency of all audits discussed above is determined by the status and importance of the activity to plant safety to assure adequacy of and conformance with the OQAP. The audit implementation procedures list activities which will be audited and the frequency of these audits. Listed in table 17.2-2 are some of the activities that will be audited and their audit frequency. Table 17.2-2 exemplifies the scope of APC's audit system.

17.2.19 OPERATIONS QUALITY ASSURANCE POLICY IMPLEMENTATION LIST

The OQAPIL lists procedures which delineate the responsibilities of individuals and groups for performing activities that affect the quality of the FNP operations. The staffs which have OQA responsibility and whose procedures are listed in the OQAPIL are as follows:

- Safety Audit and Engineering Review.
- Plant staff.
- Nuclear Support section.

maintains a separate basis for its own program, considering such attributes as inherent stability of their equipment, purpose or use, desired accuracy, and degree of usage. All measuring and test equipment used for the acceptance and verification of product quality are maintained under control systems. Such specifications as Mil-C-45662 and handbook Mil-MDBK-52 serve as a basis and provide guidance in the determination of an effective program for the control of test and measuring equipment. Typical of this equipment are micrometers, plug gages, height gages, dial verniers, voltmeters, temperature recorders, pressure gages, hardness testers, etc. Documented procedures detail the requirements for the calibration of measuring and test equipment and the use of appropriately traceable measurement standards.

17C.1.12.1 PWR Systems Division

The requirement for a supplier to maintain a system for calibration of all examination, measuring, and test equipment is contained in the administrative specification and in QCS-1. All calibration must be traceable to national standards. PWRSD verifies the acceptability of the system during the supplier selection and monitors for compliance during the surveillance activities.

17C.1.12.2 Electro-Mechanical Division

The EMD, under the direction of the Quality Assurance Department, maintains an extensive tool and gage control program utilizing electronic data processing. All tools and gages used in the manufacture and inspection of completed products are inspected and calibrated in accordance with established procedures. Control of the use of measuring and test equipment is maintained by the tool crib approach where equipment is logged out to individuals or assigned to specific areas. The program requires that any equipment which becomes damaged or out of calibration be forwarded for repair or recalibration as required. Under this program, precision tools and gages are inspected and calibrated at specified intervals based on their stability, purpose, and degree of usage. All tool and gage inspection and calibration is performed in a controlled environment. Calibration stickers are affixed to all equipment, excluding personal tools which have been found acceptable under the program. Personal tools are identified by name with the calibration status maintained by the gage inspector. Reference standards used are certified and traceable to the National Institute of Standards and Technology.

17C.1.12.3 Tampa Division

Formalized procedures defining calibration frequency and maintenance of gages and test equipment used for inspections are in effect and implemented by the Quality Assurance Department. Quality control tools and gages are identified by quality control serial numbers which are color coded to indicate calibration status, and are controlled by a tool crib card index system. Established calibration schedules for each type of tool or gage used for inspection purposes are implemented. Frequency of calibration is based on engineering judgment and verified by Quality Assurance review of calibration records. Damaged or inaccurate measuring and test equipment is removed from the cycle until repaired, recalibrated, or replaced. Master measuring standards are maintained and calibrated on a frequency cycle by a qualified laboratory with standards traceable to the National Institute of Standards and Technology.

Electrical test equipment such as magnetic particle equipment is on a scheduled calibration cycle. The Works Engineering Department is responsible for maintenance and calibration. This effort is audited by the Quality Assurance Department.

Pressure test gages used for hydrostatic and gas leak tests are checked and calibrated on a frequency schedule; dead weight test equipment is used to verify calibration. The procedures are designed to assure accuracies within established standards and include disposition and/or corrective measures when discrepancies are noted.

17C.1.12.4 Pensacola Division

All decisions on the acceptance of any product or quality characteristic are made by utilizing inspection and test equipment under calibration by the Quality Assurance Department; this calibration is traceable to the National Institute of Standards and Technology. Each gage is identified with a unique identifying serial number. For each individual gage, there is a gage inspection record card used to record the results of periodic inspections.

Calibration frequencies are initially established by an engineered estimate of the total useful life of the gage and the frequency of recalibration at one-fifth of this estimated time.

Calibration frequencies are adjusted based on an evaluation comparison of the gage usage versus the wear recorded on the gage inspection record card. Any gage which passes through

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t. 17C-4 (Sh. 1) (Rev. 0)
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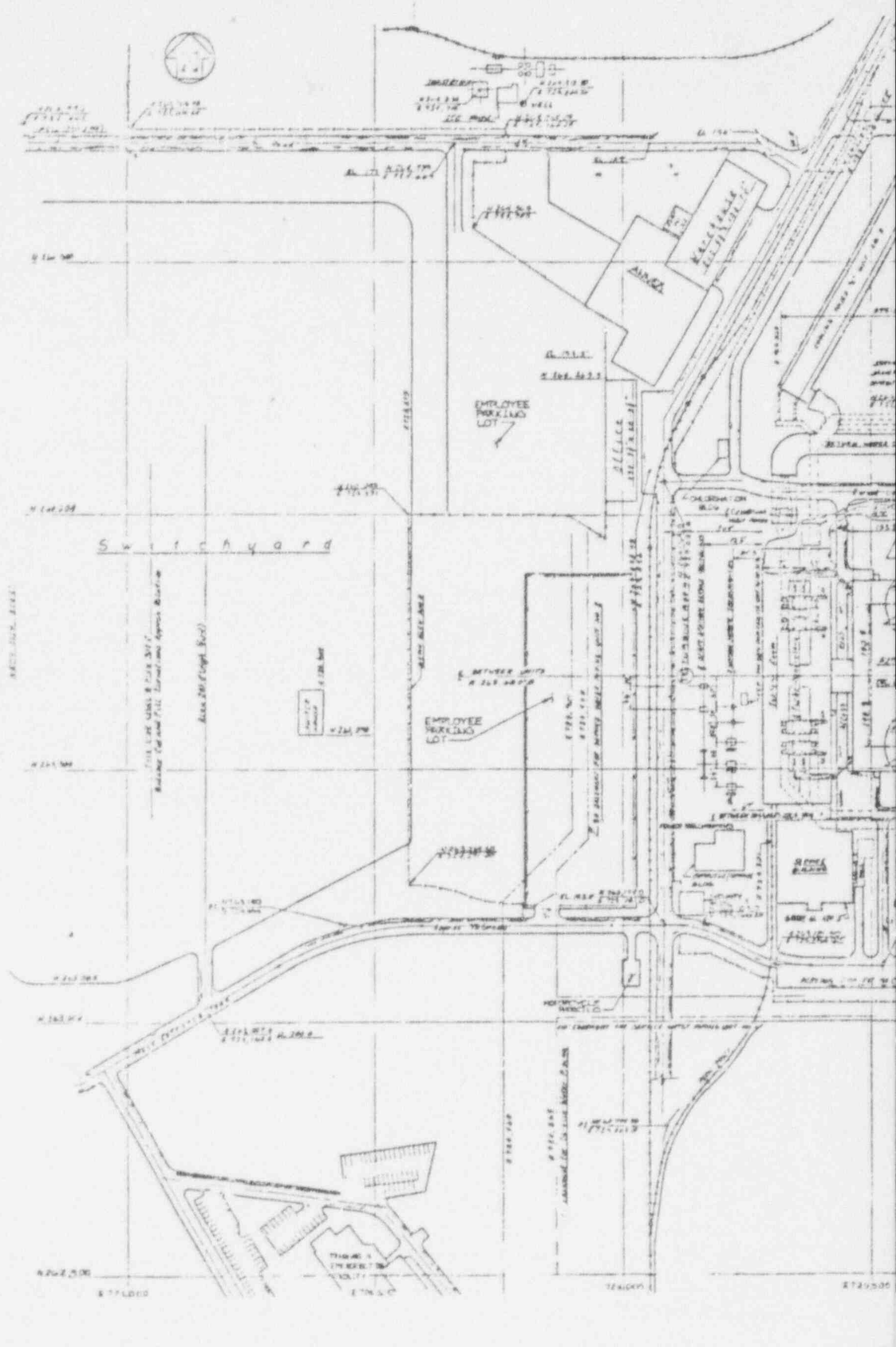
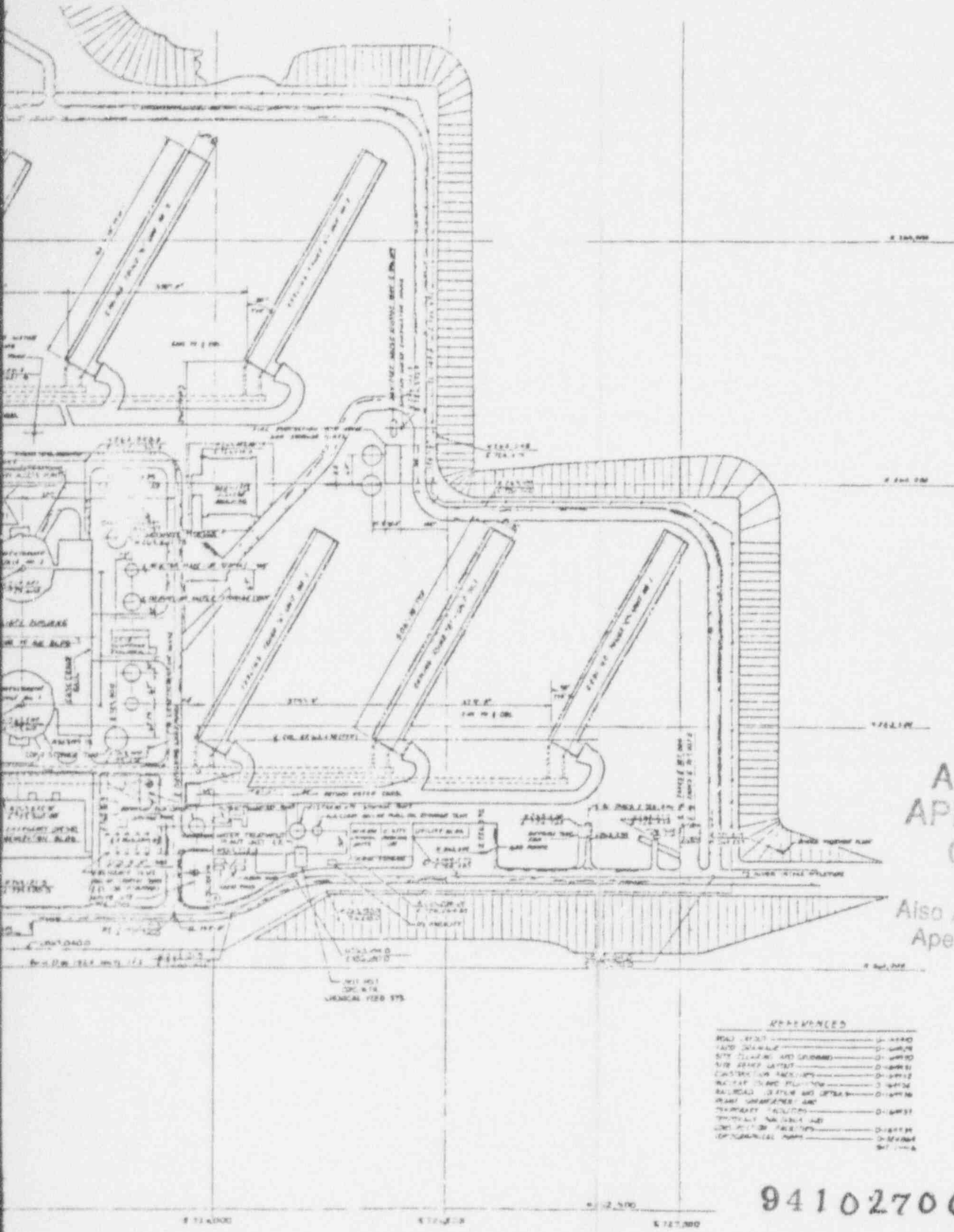
t. 17C-5 (Rev. 0)
t. 17C-6 (Rev. 0)
t. 17C-7 (Rev. 0)
f. 17C-1 (Rev. 0)
f. 17C-2 (Rev. 0)
f. 17C-3 (Rev. 0)
f. 17C-4 (Rev. 0)
f. 17C-5 (Rev. 0)
f. 17C-6 (Rev. 0)
f. 17C-7 (Rev. 0)
f. 17C-8 (Rev. 0)
f. 17C-9 (Rev. 0)
p. 17D-i (Rev. 0)
p. 17D-ii (Rev. 0)
p. 17D-1 (Rev. 0)
p. 17D-2 (Rev. 0)
p. 17D-3 (Rev. 0)
p. 17D-4 (Rev. 0)
p. 17D-5 (Rev. 0)
p. 17D-6 (Rev. 0)
p. 17D-7 (Rev. 0)
p. 17D-8 (Rev. 0)
p. 17D-9 (Rev. 0)
p. 17D-10 (Rev. 0)
p. 17D-11 (Rev. 0)
f. 17D-1 (Rev. 0)
f. 17D-2 (Rev. 0)
Effective Page List (Rev. 12)



Switchyard
EMPLOYEE PARKING LOT
EMPLOYEE PARKING LOT
ANNEX



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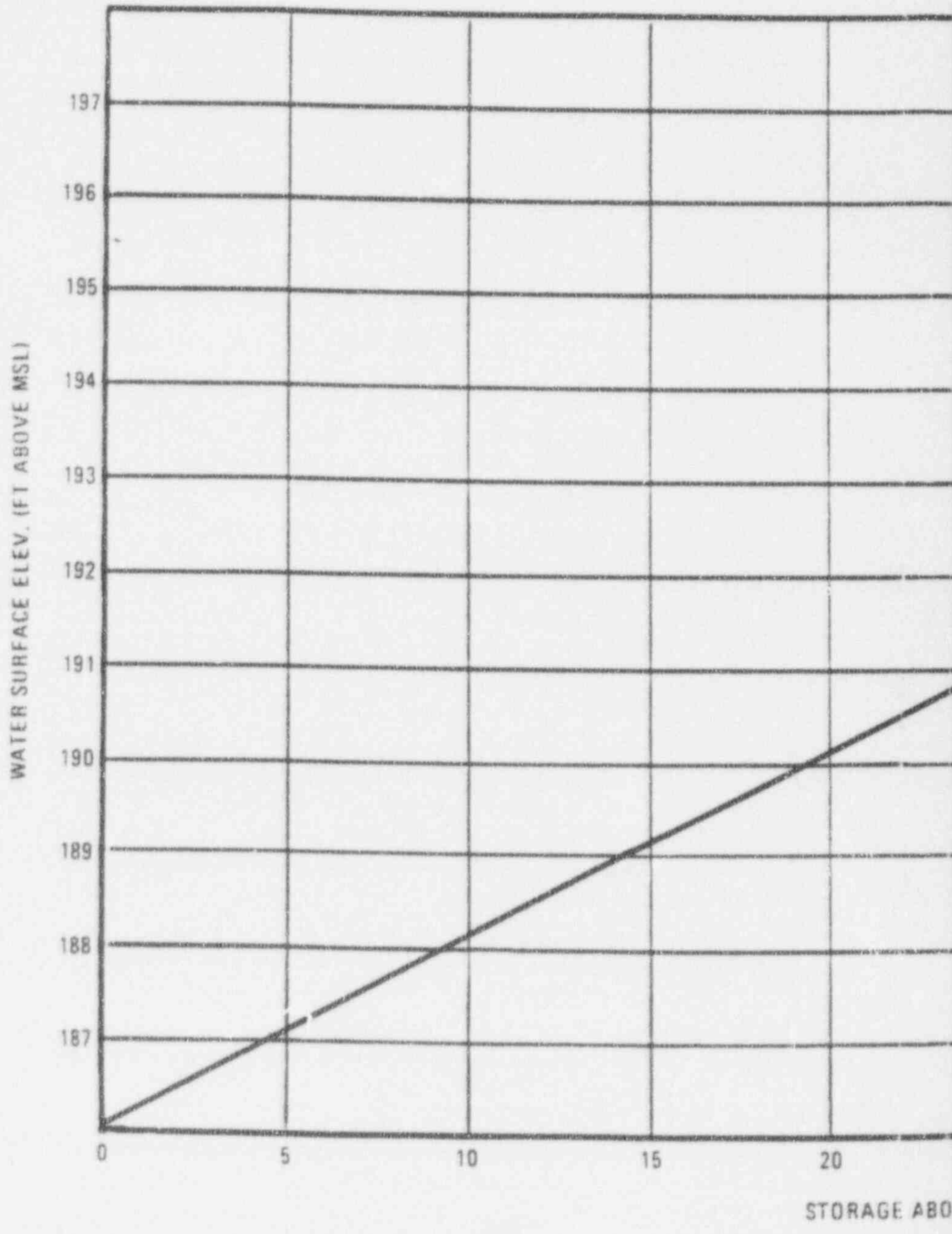
D-170084 REV 21

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

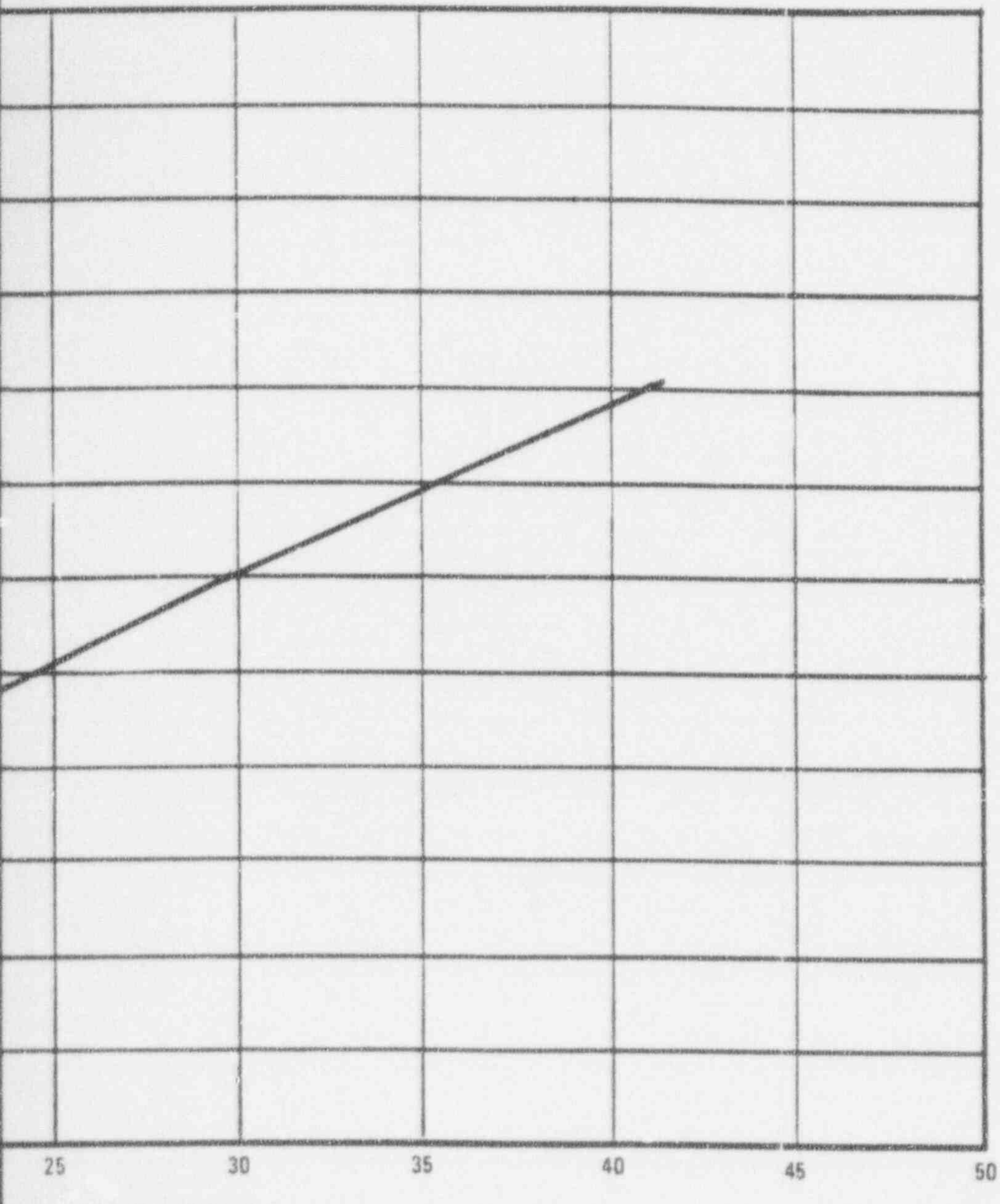
SITE PLOT PLAN

FIGURE 1.2-1



197
196
195
194
193
192
191
190
189
188
187
WATER SURFACE ELEV. (FT ABOVE MSL)
0
5
10
15
20
STORAGE ABO

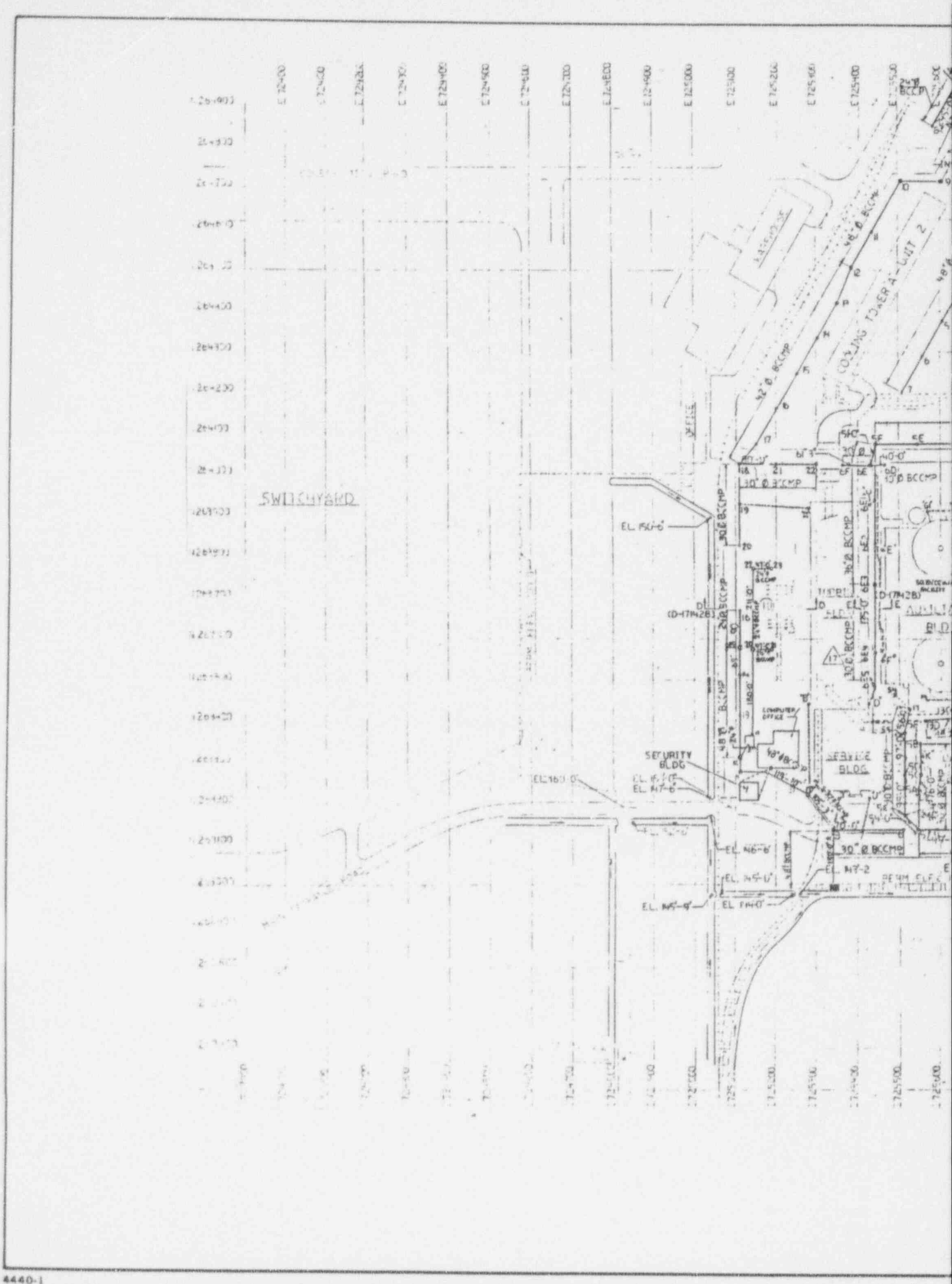
4440-1



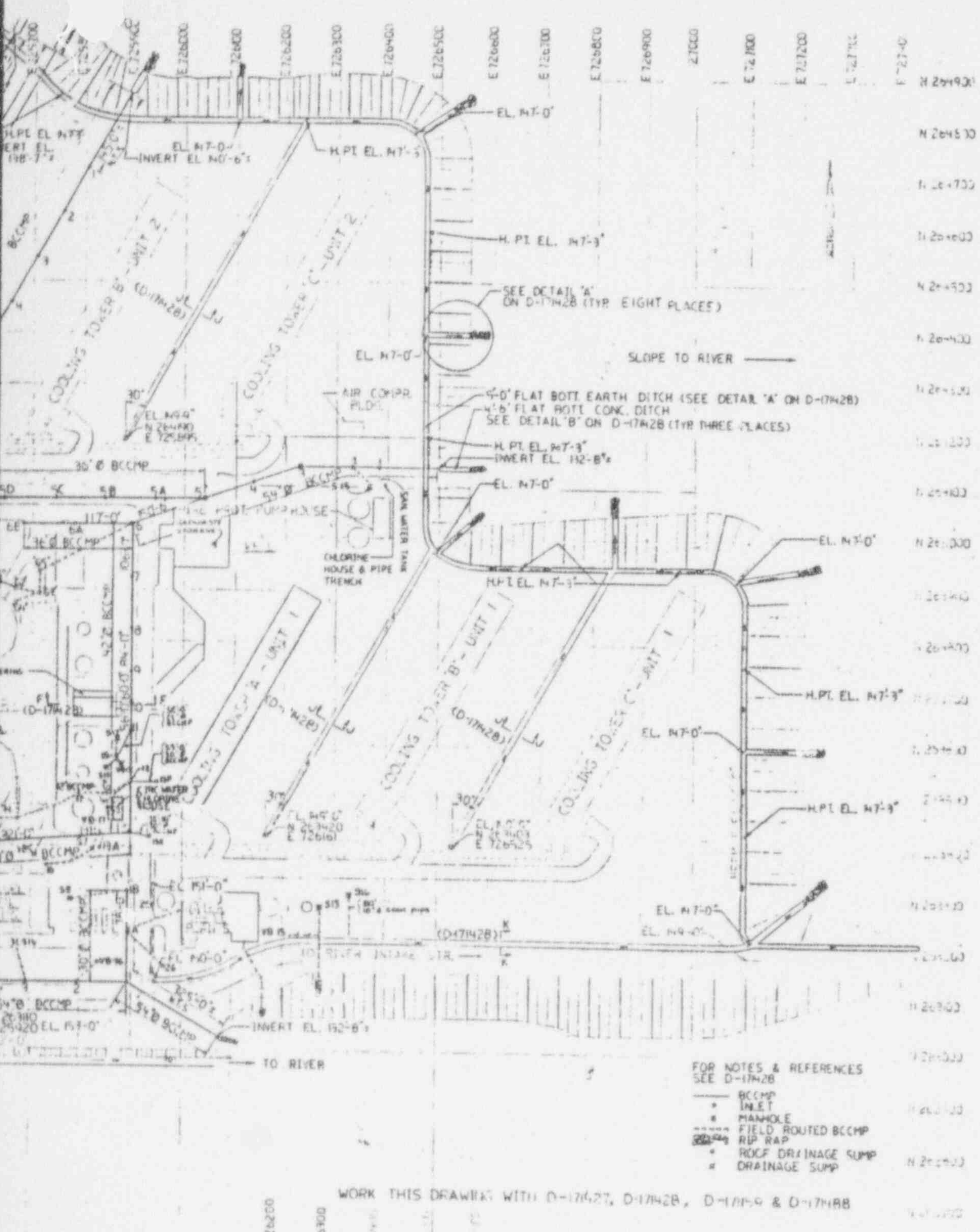
9410270094-02

REV 12 10/94

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	COOLING POND STORAGE CURVE
	FIGURE 2.4-21


SWITCHYARD
COOLING TOWER A-UNIT 2
42" Ø BCCMP
48" Ø BCCMP
30" Ø BCCMP
36" Ø BCCMP
EL. 150'-0"
SECURITY BLDG
SERVICE BLDG
COMPUTER OFFICE
(D-171428)
EL. 160'-0"
EL. 146'-6"
EL. 145'-0"
EL. 147-2

4440-1



9410270094-03

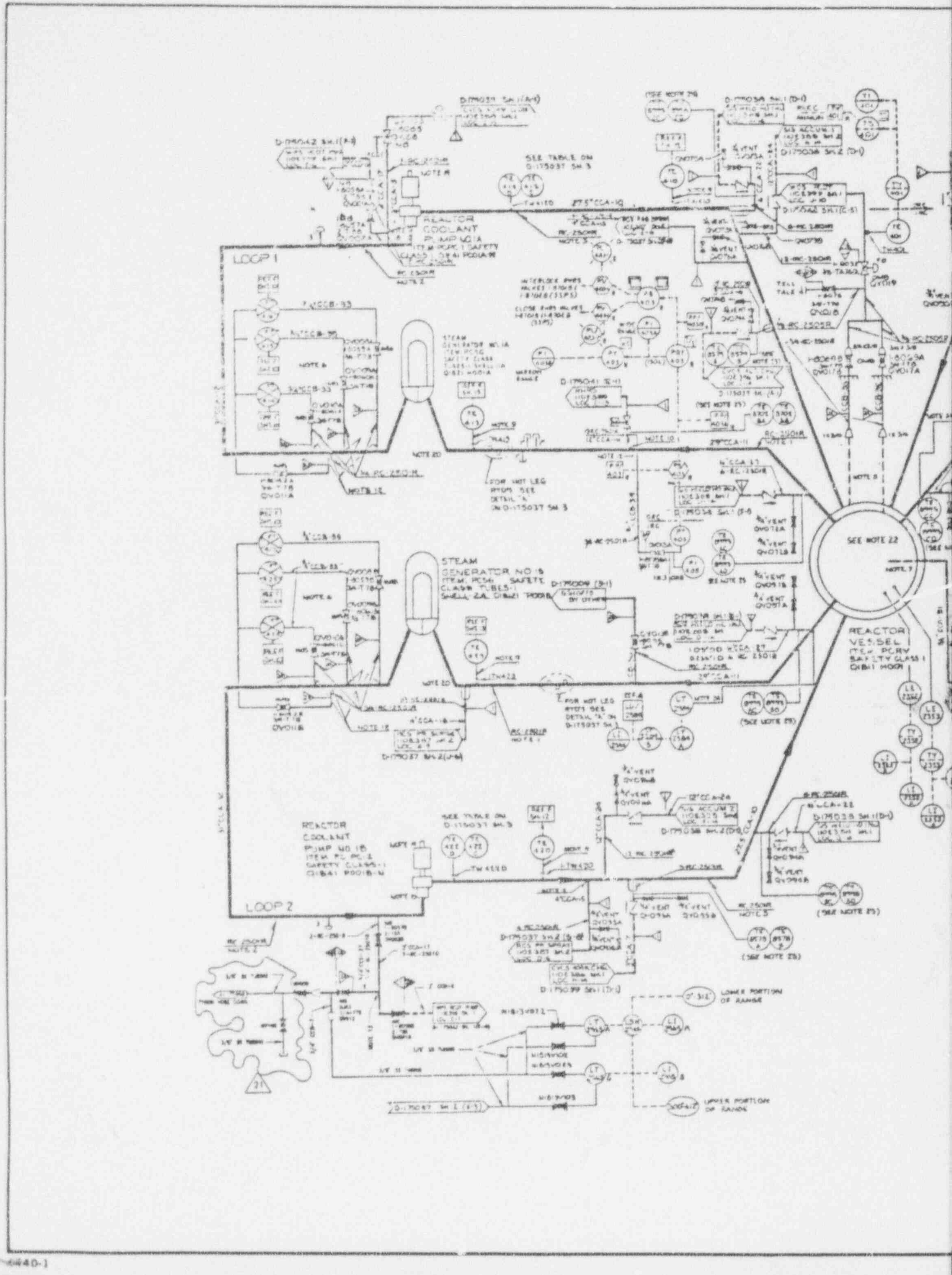
REV 12 10/94

D-171426 REV 17

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

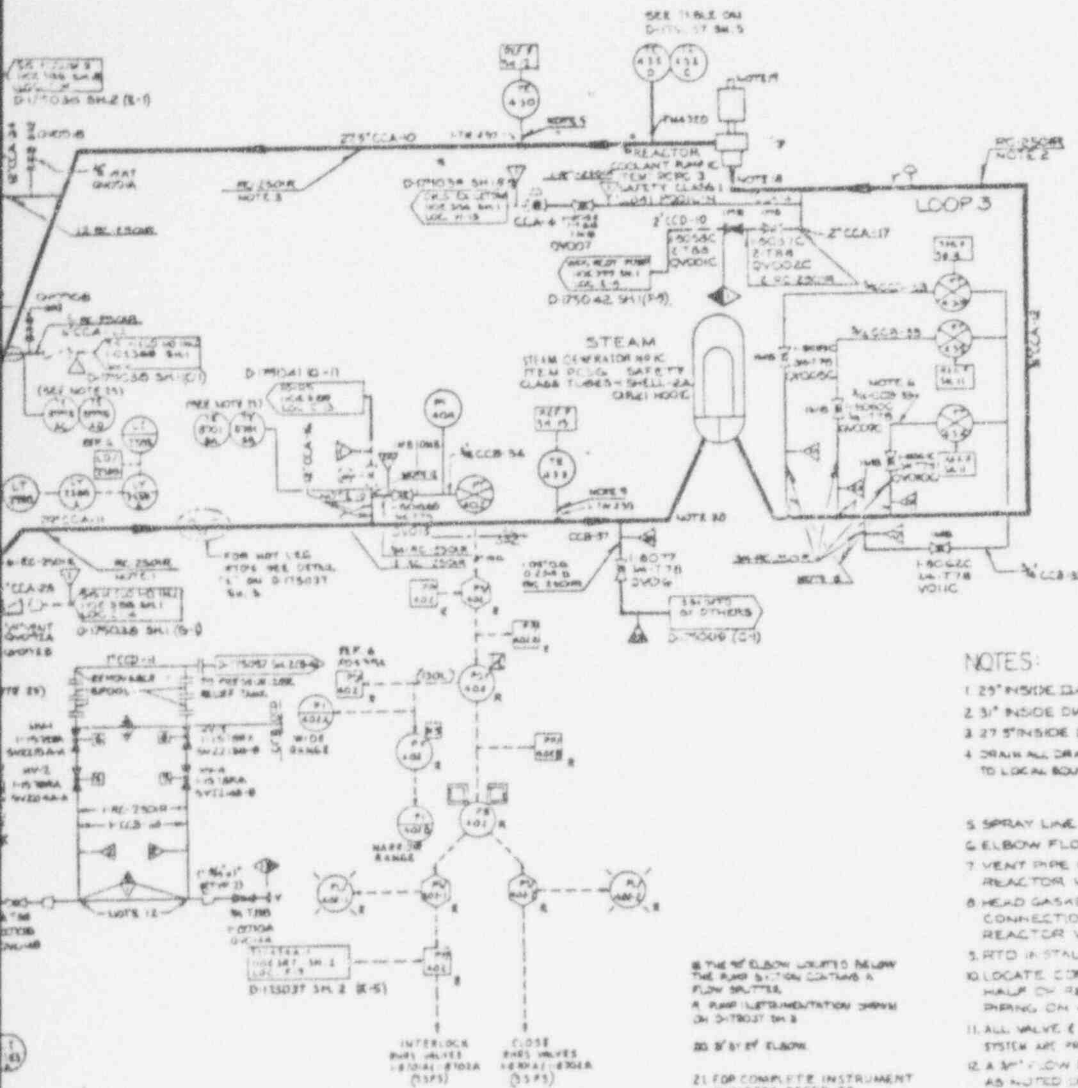
STORM DRAINAGE

FIGURE 2.4-69



REACTOR COOLANT PUMP NO.1A
LOOP 1
SEE TABLE ON D-175037 SH.3
STEAM GENERATOR NO.1B
SEE NOTE 22
REACTOR VESSEL
REACTOR COOLANT PUMP NO.1B
LOOP 2
LOWER PORTION OF RANGE
UPPER PORTION OF RANGE

6440-1



NOTES:

1. 29" INSIDE DIAMETER
2. 31" INSIDE DIAMETER
3. 27.5" INSIDE DIAMETER
4. DRAIN ALL DRAIN CONNECTIONS TO LOCAL EQUIPMENT DRAIN.
5. SPRAY LINE SCOOP
6. ELBOW FLOW METERS
7. VENT PIPE FURNISHED WITH REACTOR VESSEL HEAD.
8. HEAD GASKET MONITORING CONNECTIONS FURNISHED WITH REACTOR VESSEL.
9. RTD INSTALLED IN WELL.
10. LOCATE CONNECTION IN BOTTOM HALF OF REACTOR COOLANT PIPING ON 45° ANGLE TO VERTIC.
11. ALL VALVE & LINE NUMBERS IN THIS SYSTEM ARE PREFIXED BY Q1B13 OR N1B13.
12. A 3/4" FLOW RESTRICTOR IS REQUIRE AS NOTED ON LEGEND A PIPE NIPPLE AND INLINE 3/4" FLOW RESTRICTOR MAY BE USED IF APPROVED BY WESTINGHOUSE AND BECHTEL ON A CASE-BY-CASE BASIS.
13. FOR PIPING CLASS SUMMARY SHEETS SEE (SPEC SS-1106-1)
14. FOR VALVE INFORMATION SEE MASTER VALVE LIST (SPEC SS-1102-99)
15. FOR EQUIPMENT LIST SEE DWG D175079
16. FOR INSTRUMENT INSTALLATION DETAILS SEE REFERENCE IN INSTRUMENT INDEX.
17. SEE DWG 175044 FOR SYS LEGEND
18. THE 90° ELBOW LOCATED BELOW THE PUMP SUCTION CONTAINS A FLOW SPLITTER.
19. PUMP INSTRUMENTATION SHOWN ON D-175037 SH 3
20. 31" BY 27.5" ELBOW
21. FOR COMPLETE INSTRUMENT NUMBERS REFER TO INSTRUMENT INDEX B-175903
22. IN CORE THERMOCOUPLES T1 THROUGH T51 HAVE BEEN ASSIGNED TPNS NUMBERS Q1B11TE2701 THROUGH TE2751
23. TE'S AS NOTED ARE CLAMPED ON RTD'S FOR THERMAL CYCLING MONITORING.
24. LEVEL TRANSMITTERS ARE STRAP ON ULTRASONIC TYPE. LEVEL INSTRUMENTATION IS UTILIZED DURING 3" LOOP OPERATION ONLY.

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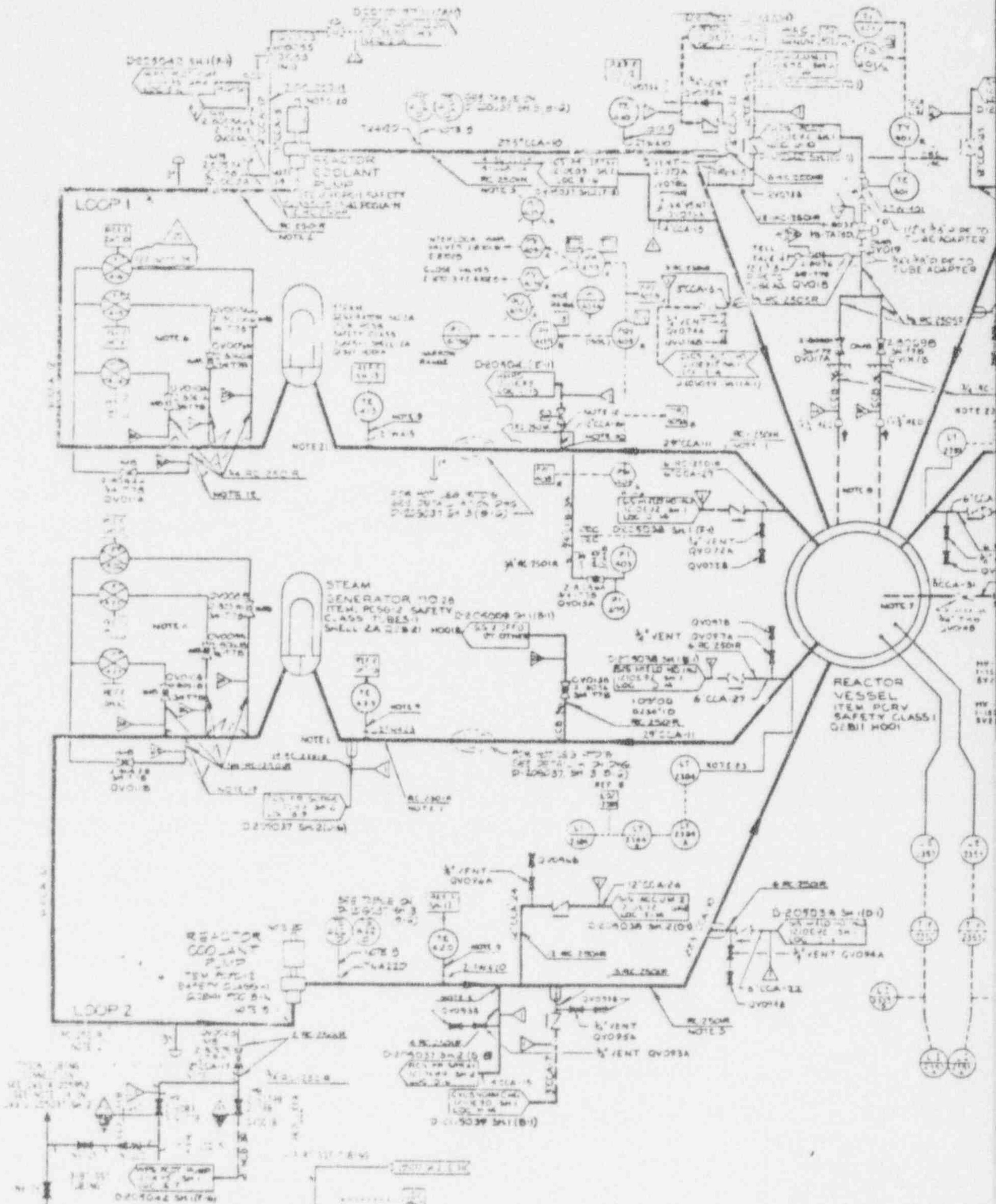
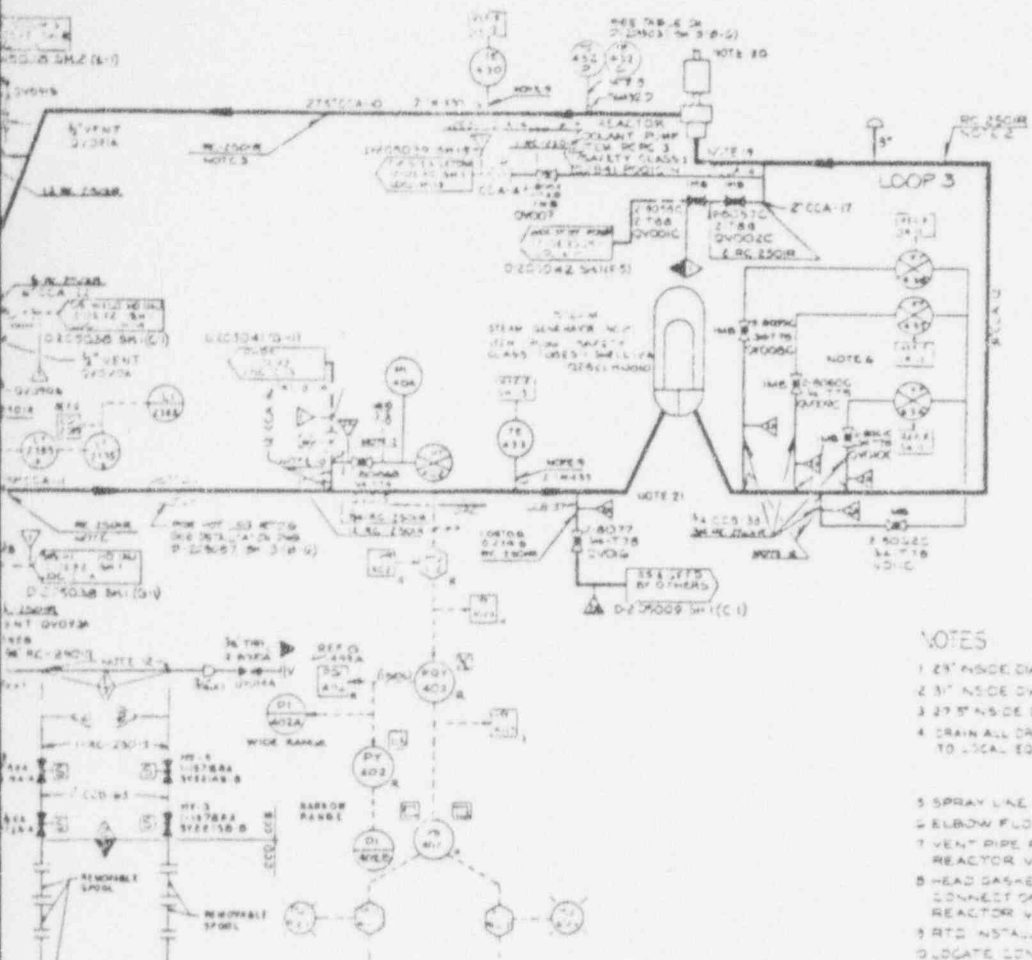
D 175037 (SHEET 1) REV 21

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

UNIT 1 REACTOR COOLANT SYSTEM

FIGURE 5.1-1 (SHEET 1 OF 2)


LOOP 1
REACTOR COOLANT PUMP
STEAM GENERATOR
STEAM GENERATOR NO 2B
REACTOR VESSEL
ITEM PCRV
SAFETY CLASS 1
REACTOR COOLANT PUMP
LOOP 2
TUBE ADAPTER



INTERLOCK RHRS VALVES

NOTES

1. 29" INSIDE DIAMETER
2. 31" INSIDE DIAMETER
3. 27.5" INSIDE DIAMETER
4. DRAIN ALL DRAIN CONNECTIONS TO LOCAL EQUIPMENT DRAIN.
5. SPRAY LINE SCOOP
6. ELBOW FLOW METERS
7. VENT PIPE FURNISHED WITH REACTOR VESSEL HEAD.
8. HEAD GASKET MONITORING CONNECTIONS FURNISHED WITH REACTOR VESSEL.
9. RTD INSTALLED IN WELL.
10. LOCATE CONNECTION IN BOTTOM HALF OF REACTOR COOLANT PIPING ON 45° ANGLE TO VERTICAL.
11. ALL VALVE & LINE NUMBERS IN THIS SYSTEM ARE PREFIXED BY Q2(Q) OR N2(Q)
12. A [illegible] FLOW RESTRICTOR IS REQUIRED AS NOTED ON LEGEND A PIPE NIPPLE AND NONE [illegible] FLOW RESTRICTOR MAY BE USED IF APPROVED BY WESTINGHOUSE AND [illegible] ON A CASE-BY-CASE BASIS.
13. FOR PIPING CLASS SUMMARY SHEETS SEE (SPEC SS-1109-1).
14. FOR VALVE INFORMATION SEE MASTER VALVE LIST (SPEC SS-1102-39).
15. FOR EQUIPMENT LIST SEE DWG. D-205010.
16. FOR INSTRUMENT INSTALLATION DETAILS SEE REFERENCE IN INSTRUMENT INDEX.
17. SEE DWG D-205044 FOR NSSS LEGEND.
18. INCORE THERMOCOUPLES T1 THROUGH T51 HAVE BEEN ASSIGNED TPNS NUMBERS Q2B41TE2301 THROUGH TE2351 SEE U-27441.
19. THE 90° ELBOW LOCATED BELOW THE PUMP SUCTION CONTAINS A FLOW SPLITTER.
20. PUMP INSTRUMENTATION SHOWN ON D-205037 SH 5.
21. 31" BY 29" ELBOW.
22. FOR COMPLETE INSTRUMENT NUMBERS REFER TO INSTRUMENT INDEX [illegible]
23. LEVEL TRANSMITTERS ARE STRAP ON ULTRASONIC TYPE. LEVEL INSTRUMENTATION IS UTILIZED DURING MID-LOOP OPERATION ONLY.
24. THE HANDLE HAS BEEN REMOVED FROM VALVE Q2B31V006B WITH THE VALVE IN THE OPEN POSITION AND A CAP WAS WELDED OVER THE STEM AND YOKE.
25. [illegible]

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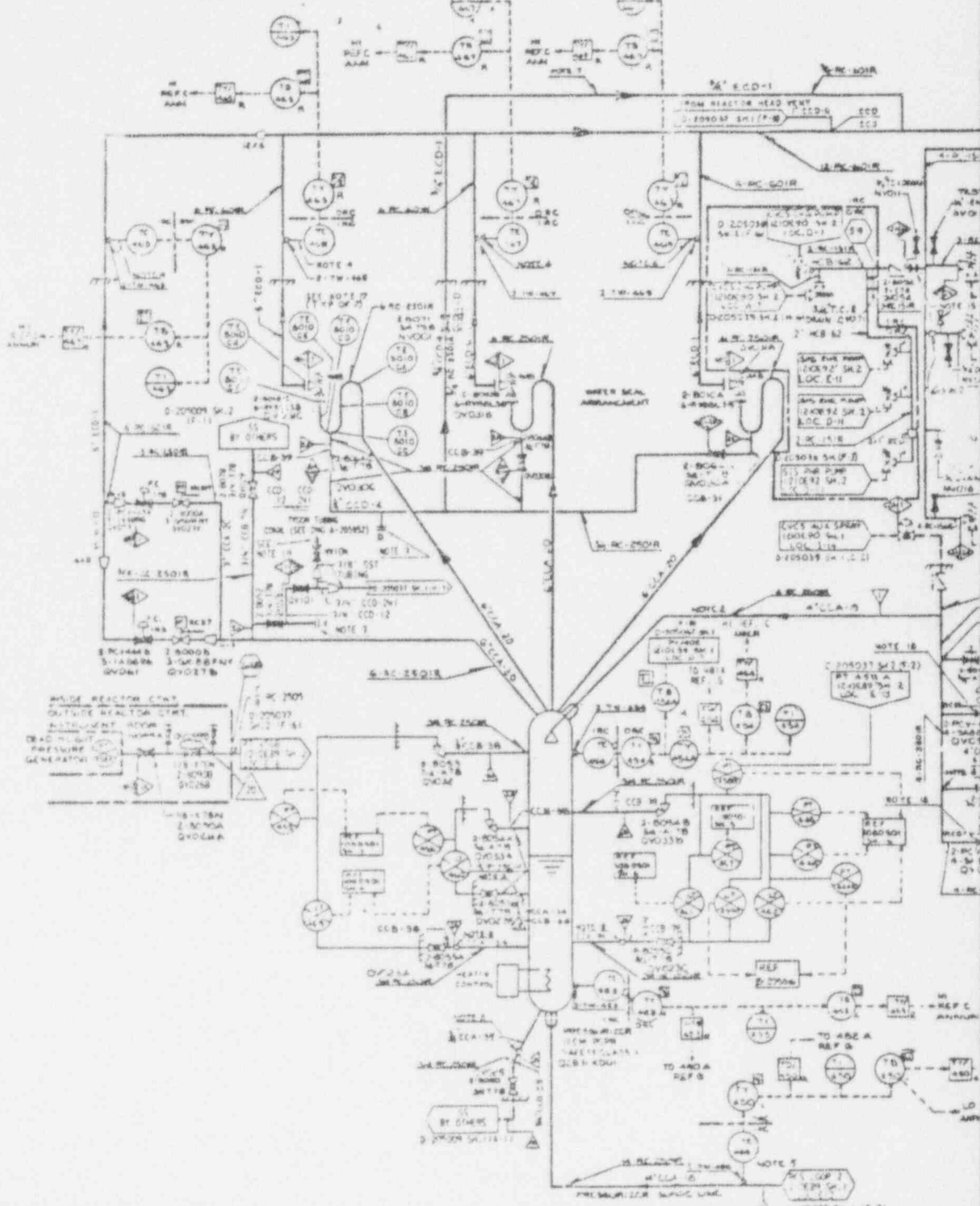
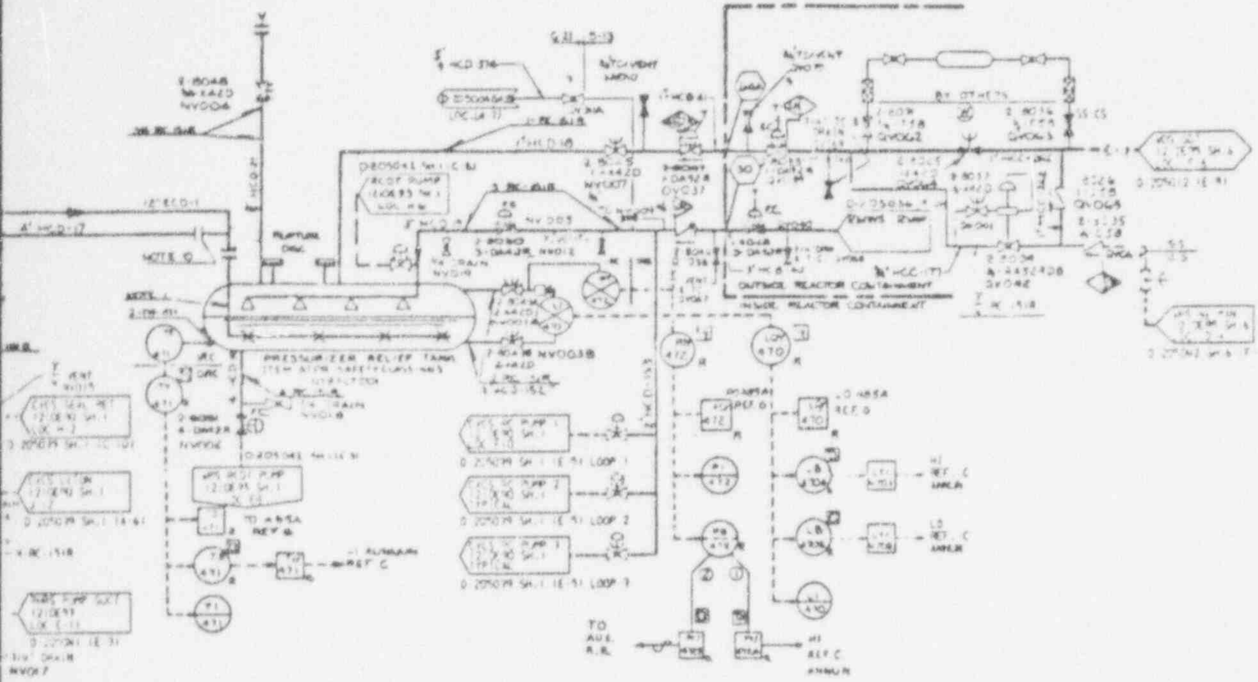
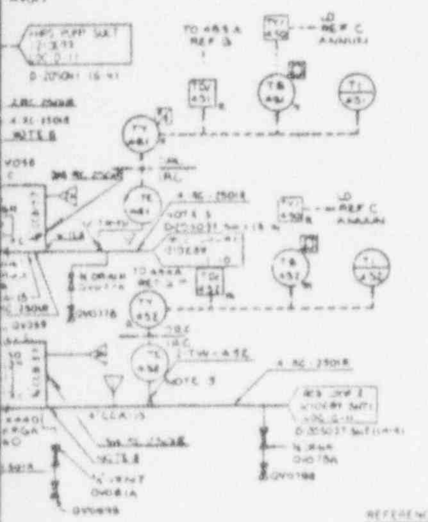
REV 12 10/94

D-205037 (SHEET 1) REV 20

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	UNIT 2 REACTOR COOLANT SYSTEM
	FIGURE 5.1-1 (SHEET 2 OF 2)







NOTES:

1. VENT HOLE PROVIDED
2. SLOPE SPRAY PIPE DOWNWARD TO PROVIDE WATER SEAL BETWEEN PRESSURIZER AND SPRAY VALVES.
3. TYGON HOSE INSTALLED TO VENT PRESSURIZER DURING DRAINING & FILLING
4. PLACE DETECTORS AT BOTTOM OF PIPE
5. LOCATE APPROXIMATELY MIDWAY BETWEEN LOOP AND PRESSURIZER
6. ALL W ITEM NO IN RCS ARE SHOWN WITHOUT PREFIX ALARC.
7. DRAIN LINES TO BE SLOPED DOWN FROM LOOP SEALS TO CONNECT WITH PRT HEADER LINE
8. A W FLOW RESTRICTOR IS REQUIRED AS NOTED ON LEGEND
9. MANUAL AND CHECK VALVES 2 INCHES AND UNDER IN OTHERS SCOPE
10. 300LB ASA FLANGES FURNISHED BY OTHERS FLANGE END ON (S) PIPE TO BE MACHINED TO FIT. LOCATE CONNECTION CLOSE TO PRT
11. SEE SHEET 1 FOR GENERAL NOTES
12. SEE DWG D-205044 FOR W SYSTEM LEGEND
13. DRAIN ALL DRAIN CONNECTIONS TO LOCAL EQUIPMENT DRAIN
14. FOR COMPLETE INSTRUMENT NUMBERS REFER TO INSTRUMENT ... A-10680A
15. FLANGED CONNECTION TO FACILITATE BLIND FLANGE FOR LOCAL LEAK RATE TEST.
16. LINES HCC-177 AND HCC-242 HAVE BEEN DOWNGRADED FROM ASME CLASS 2 SEISMIC CATEGORY I TO NNS SEISMIC CATEGORY 2 BY FSAR AMENDMENT 73. VALVE AND LINE NUMBERS HAVE NOT BEEN CHANGED. FLAG(△) DESIGNATES NNS CLASSIFICATION.
17. TE'S ARE CLAMP ON TYPE RTD'S FOR TEMPORARY TEMPERATURE MONITORING.

REFERENCES:

1. ALPHA REFERENCES ON LEGEND 1210E88

NOTES:

18. A SMALL CONTINUOUS SPRAY FLOW IS PROVIDED BY MEANS OF NORMAL VALVE SEAT LEAKAGE THROUGH THE POWER OPERATED SPRAY VALVES.
19. TYGON TUBING INSTALLED DURING OUTAGES AS A SUPPLEMENTAL RCS LEVEL MONITOR.

9410270094-06

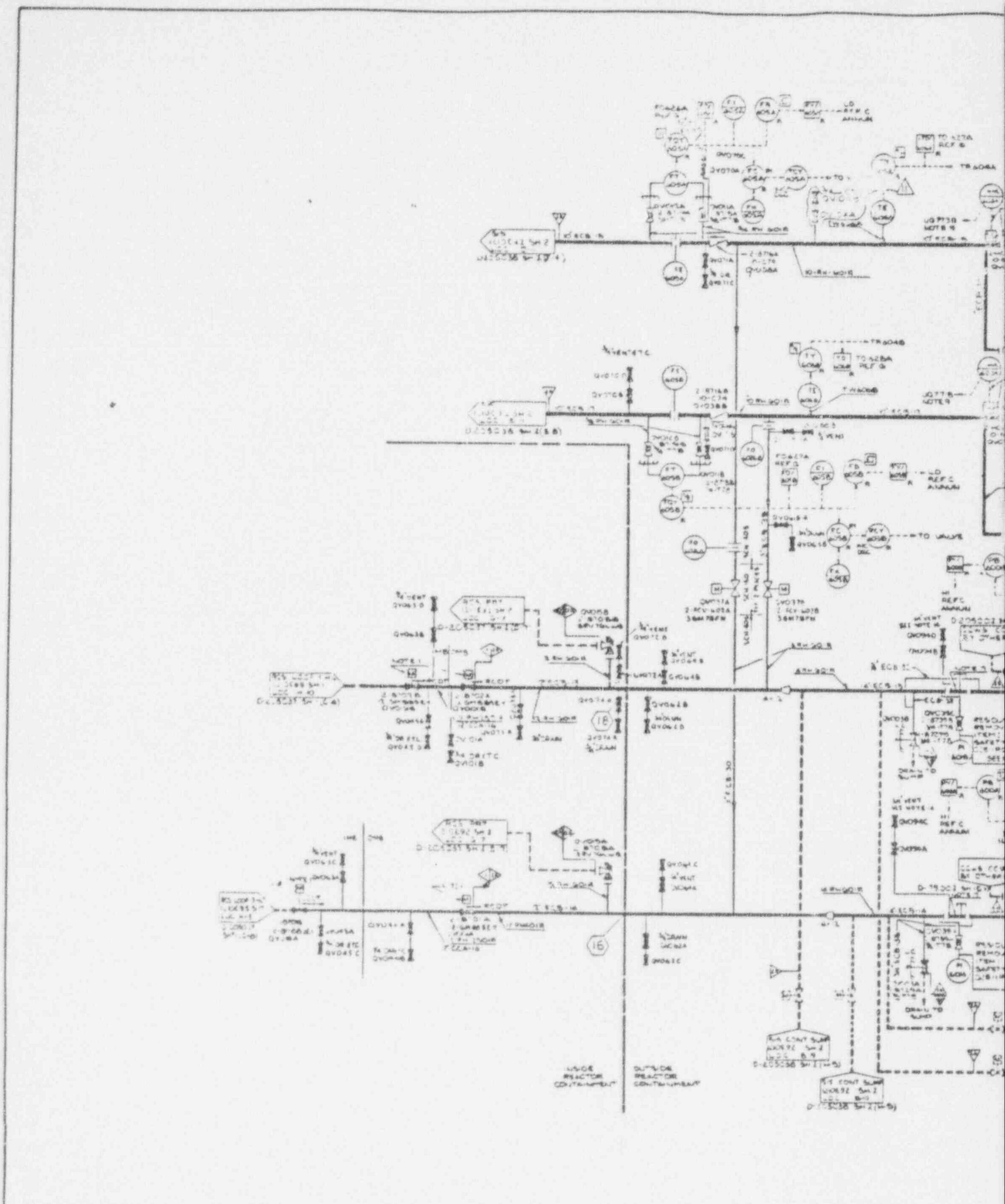
REV 12 10/94

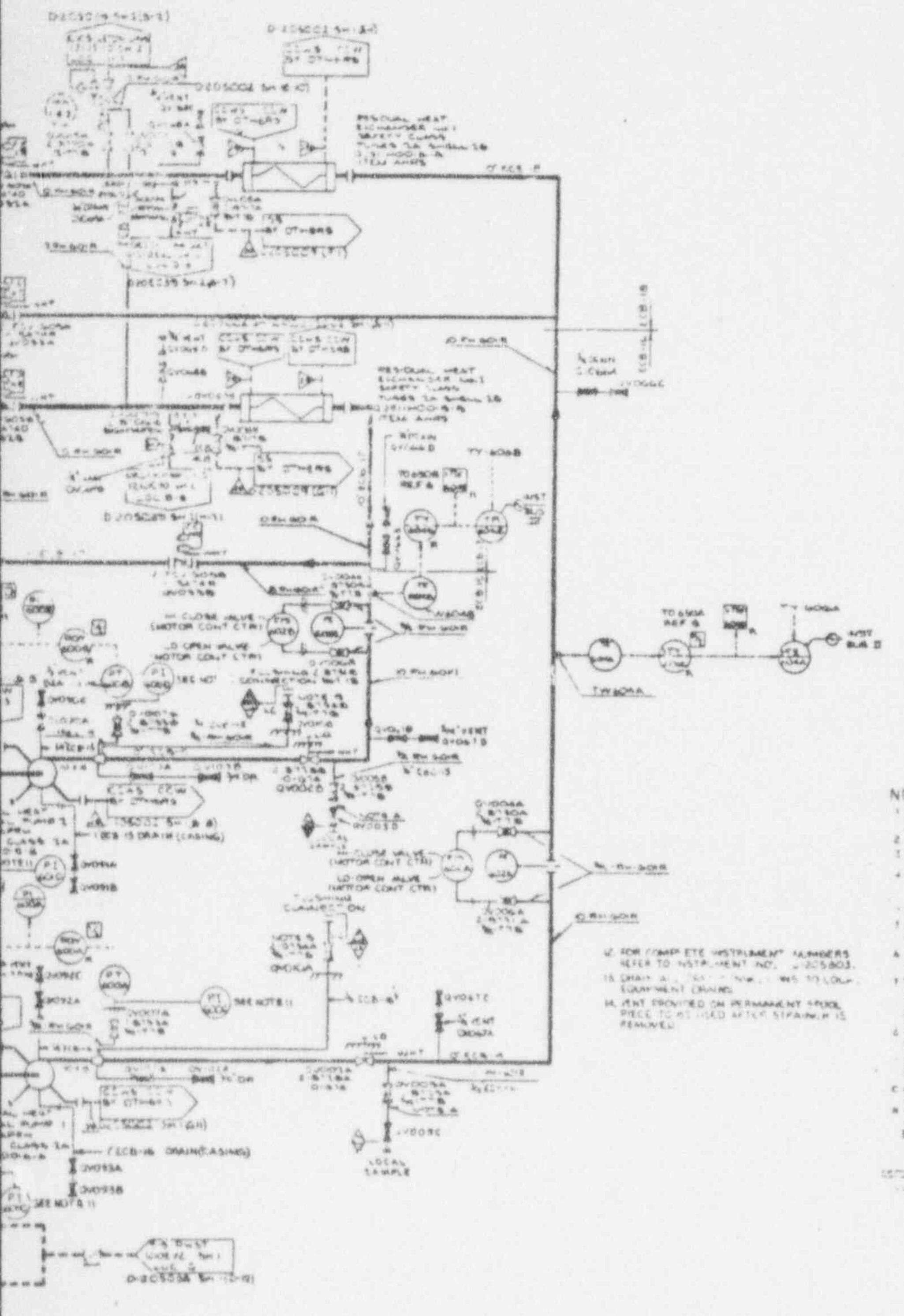
D-205037 (SHEET 2) REV 20

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

REACTOR COOLANT SYSTEM - UNIT 2

FIGURE 5.1-2 (SHEET 2 OF 2)


INSIDE REACTOR CONTAINMENT
OUTSIDE REACTOR CONTAINMENT
TO VALVE



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NOTES

9410270094 - 07

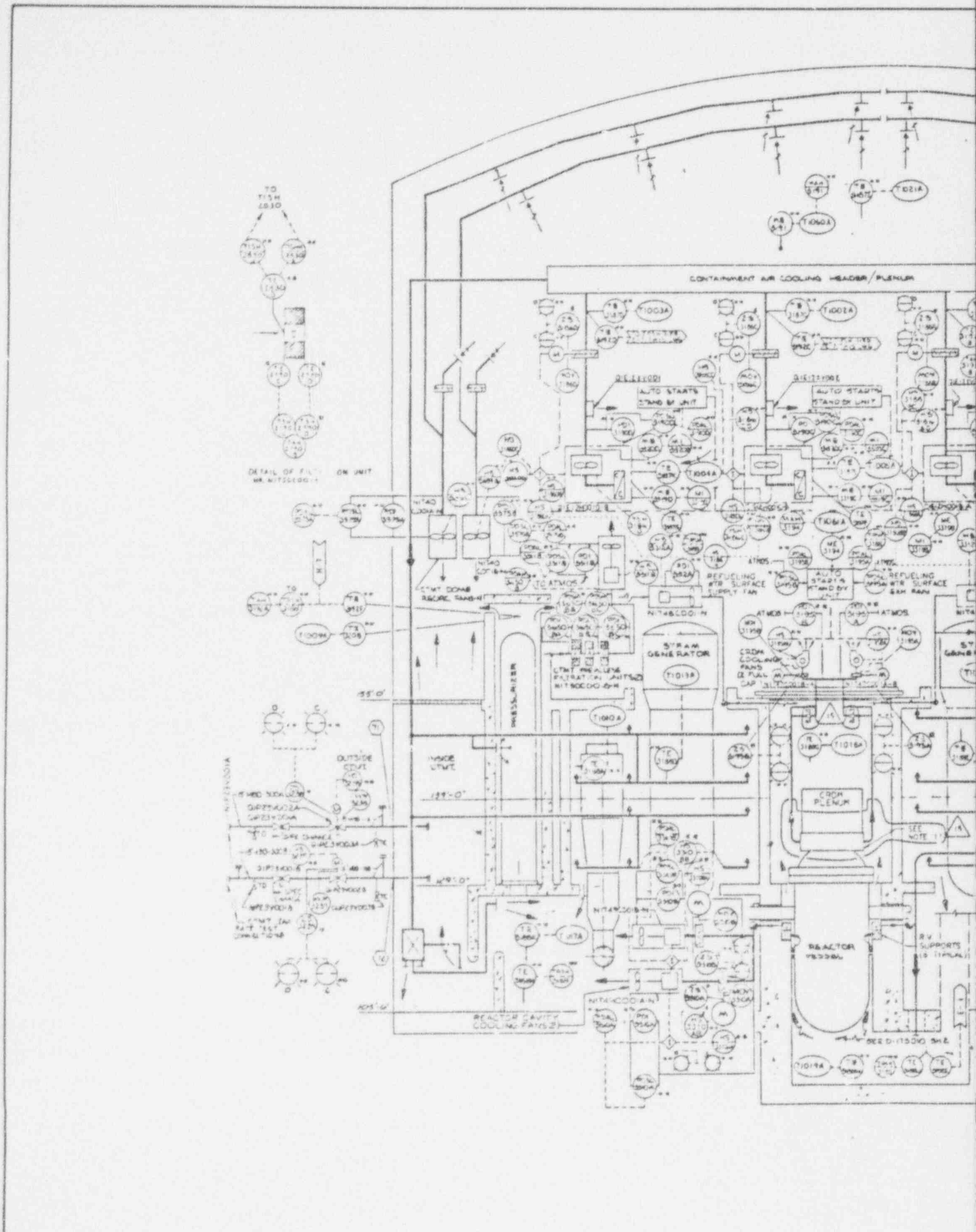
REV 12 10/94

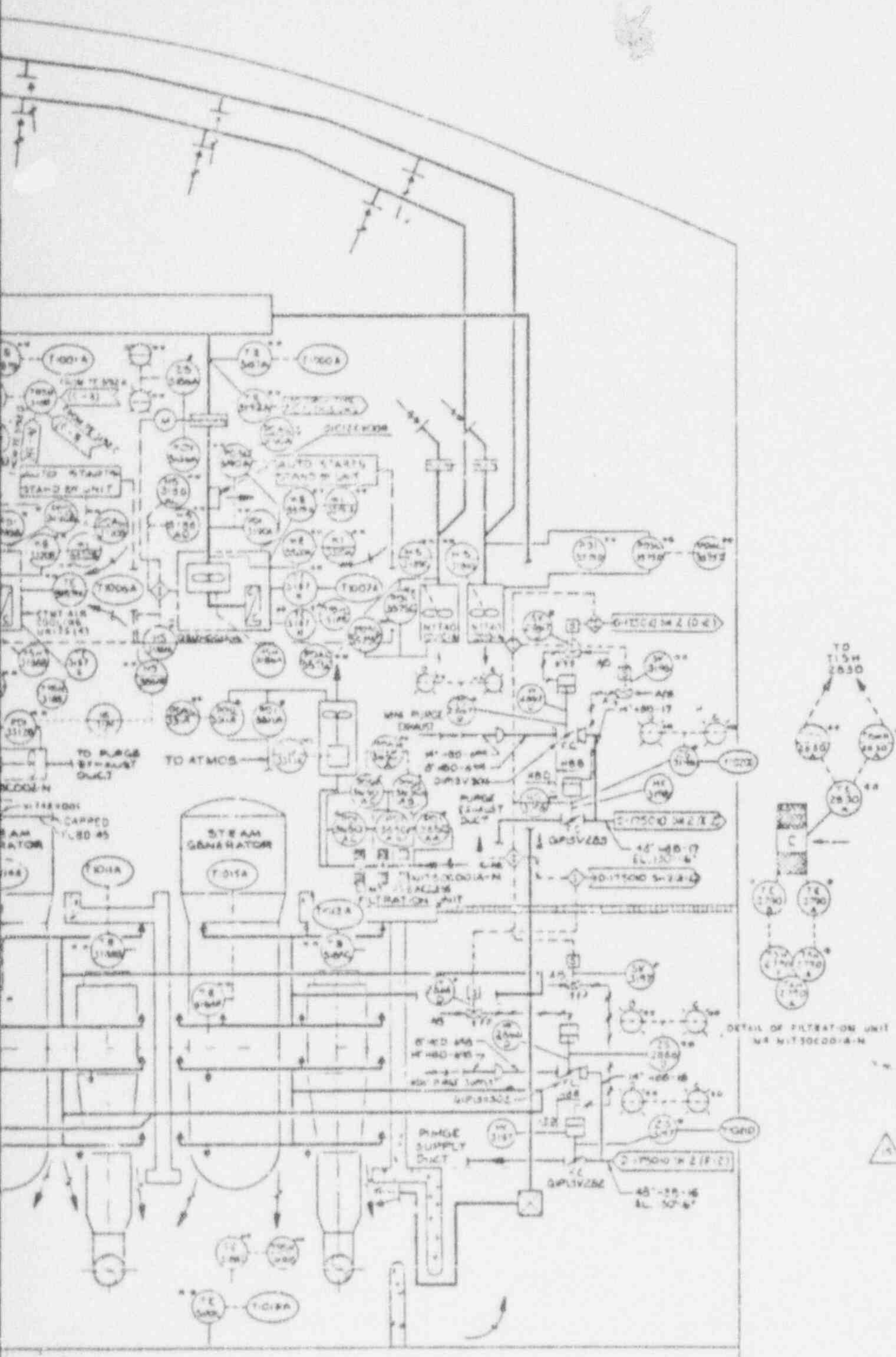
D-205041 REV 11

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

UNIT 2 RESIDUAL HEAT REMOVAL SYSTEM

FIGURE 5.5-6 (SHEET 2 OF 2)


CONTAINMENT AIR COOLING HEADER/PLENUM
AUTO STARTS STAND BY UNIT
AUTO STARTS STAND BY UNIT
STEAM GENERATOR
PRESSURIZER
CRDM PLENUM
REACTOR VESSEL
REFUELING WTR SURFACE SUPPLY FAN
REACTOR CAVITY COOLING FANS(2)
INSIDE CTMT
OUTSIDE CTMT
RV SUPPORTS
DETAIL OF FILTRATION UNIT



NOTES

1. FOR DUCT MATERIAL & CONSTRUCTION SEE (SPEC. SS-1102-54).
2. FOR DAMPER INFORMATION SEE (SPEC. SS-1102-54).
3. FOR SEISMIC CLASSIFICATION OF DUCTWORK SEE SPECS. SS-1102-54.
4. FOR INSTRUMENT INSTALLATION DETAILS SEE REFERENCE IN INSTRUMENT INDEX. B-175607.
5. SEE DRAWING D-175028 FOR PROCESS FLOW DIAGRAM.
6. WORK THIS DWG. WITH D-175090 THRU D-175098, D-175101 THRU D-175105, D-175108 & D-175122.
7. ** ASSOCIATED INSTRUMENTATION TO BE PROVIDED UNDER INSTRUMENTATION & CONTROL CONTRACT.

   * ASSOCIATED INSTRUMENTATION TO BE PROVIDED BY EQUIP. MANUFACTURER.

   ALL OTHER INSTRUMENTATION SHALL BE PROVIDED BY DUCTWORK SUB CONTRACTOR.
8. ALL SOLENOID VALVES FOR PNEUMATIC OPERATED VALVES & DAMPERS SHALL BE 120 VOLT DC UNLESS OTHERWISE INDICATED.
9. FOR COMPLETE INSTRUMENT NOS. REFER TO INSTRUMENT INDEX B-175803.
10. B-506521 - UNIT NO. 1 & SHARED FIRE DAMPER REPORT.
11. THE NORTH FACING NOZZLE (0° AZIMUTH) FOR THE CRDM COOLING SHROUD HAS BEEN BLANKED OFF.

15

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9410270094-08

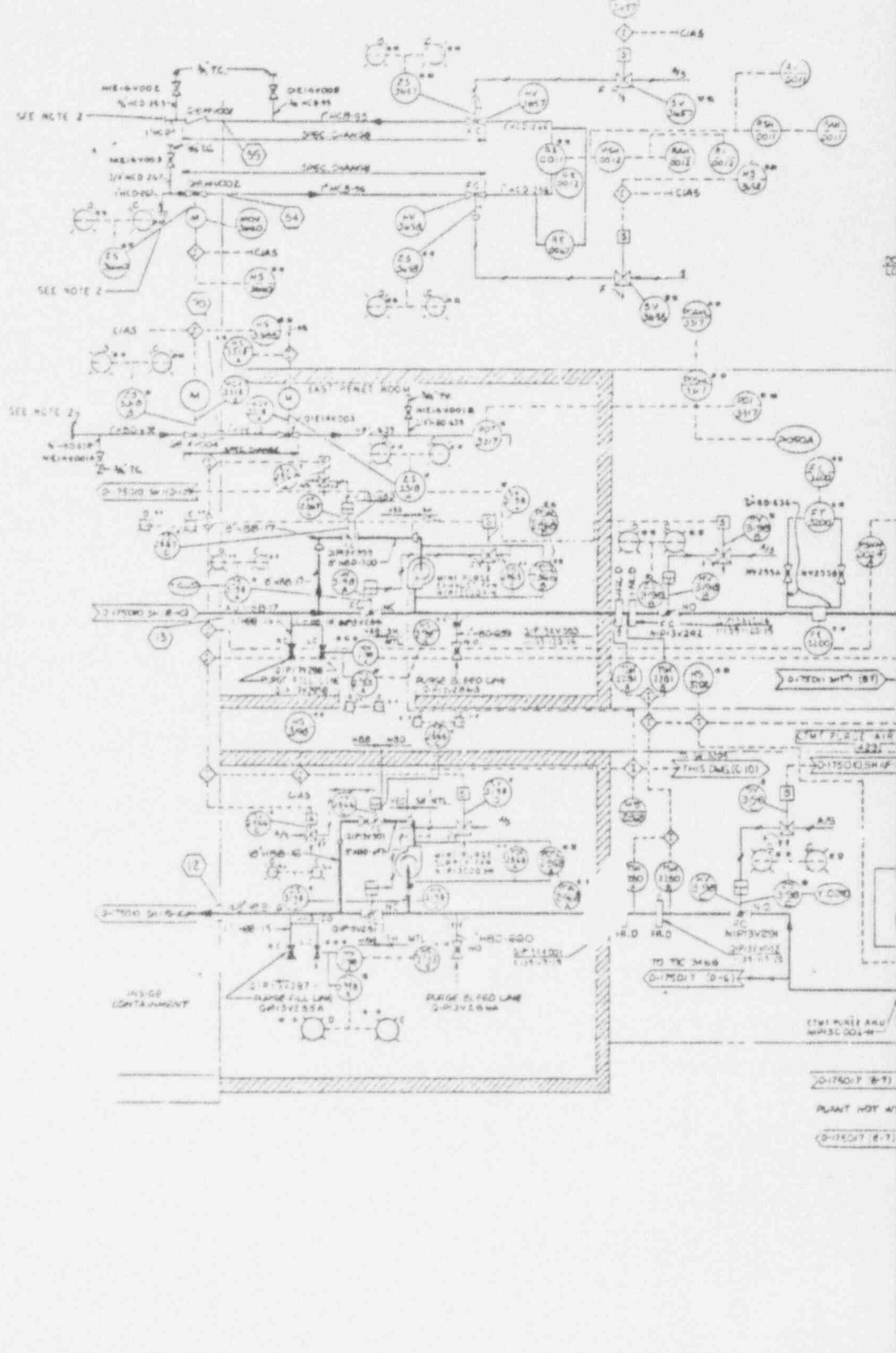
REV 12 10/94

D-175010 (SHEET 1) REV 15

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

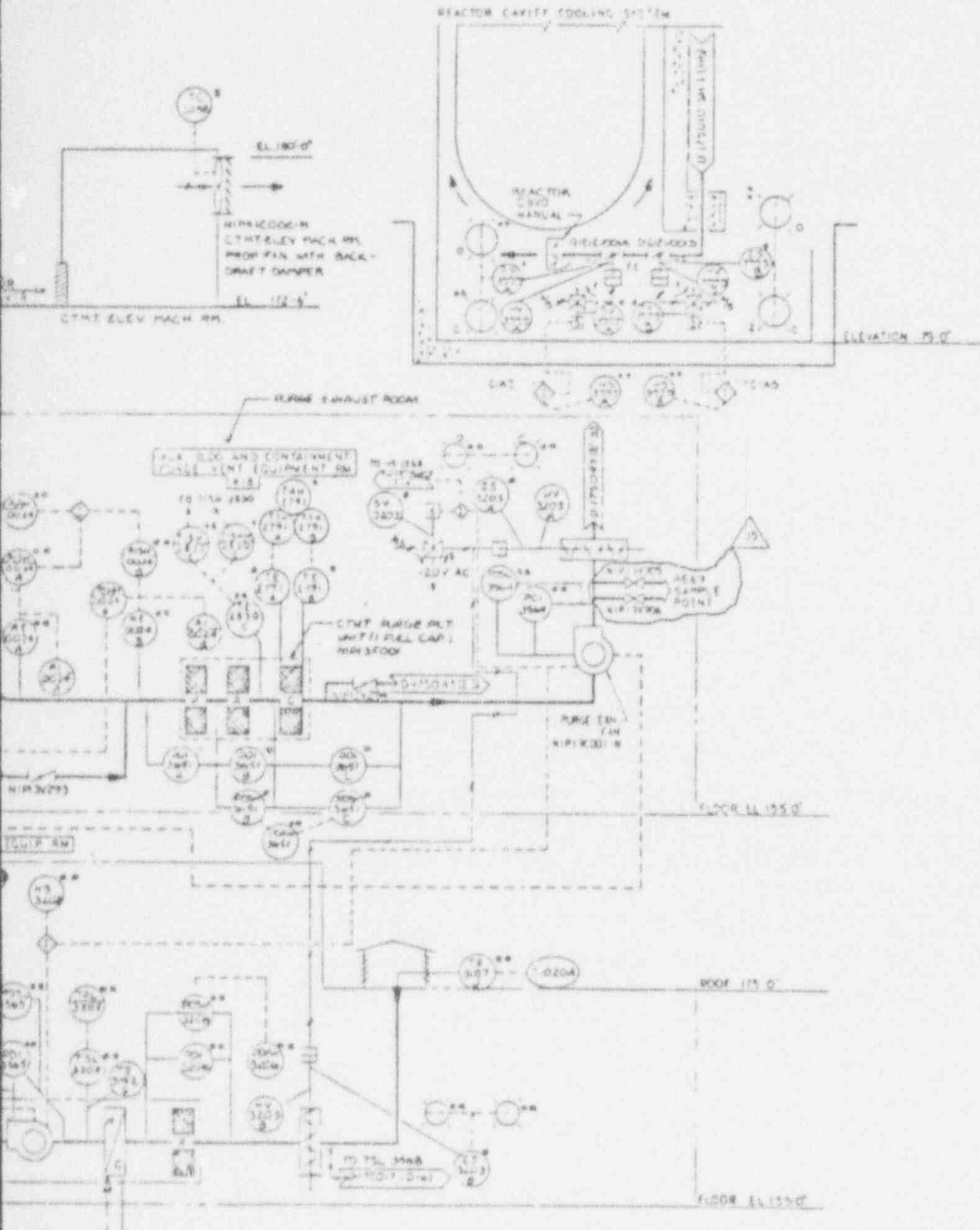
CONTAINMENT COOLING AND PURGE SYSTEM UNIT 1

FIGURE 6.2-91 (SHEET 1 OF 2)



SEE NOTE 2
SPEC CHANGE
CIAS
EAST PENET ROOM
PURGE BLEED LINE
PURGE FILL LINE
INSIDE CONTAINMENT

4440-1



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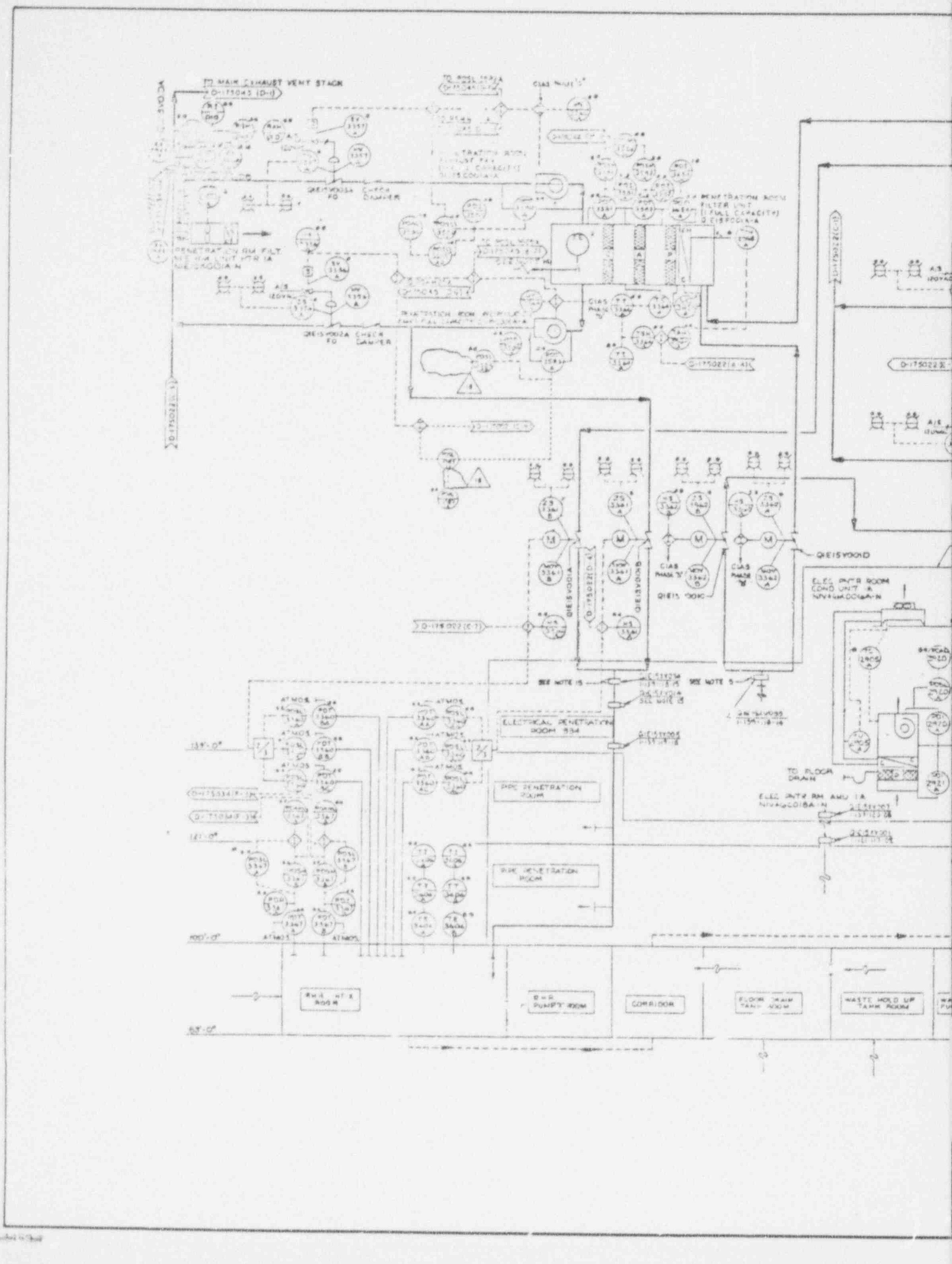
NOTES:
1. FOR GENERAL NOTES AND REFERENCES SEE DWG. D-175010, SH. 1.
2. BLIND FLANGES ARE INSTALLED FOR LOCAL LEAK RATE TESTING. THEY ARE TO BE REMOVED FOR POWER OPERATION.
3. ***MOTOR OPERATORS FOR VALVES [illegible] AND [illegible] ARE DE-ENERGIZED.

9410270094-09

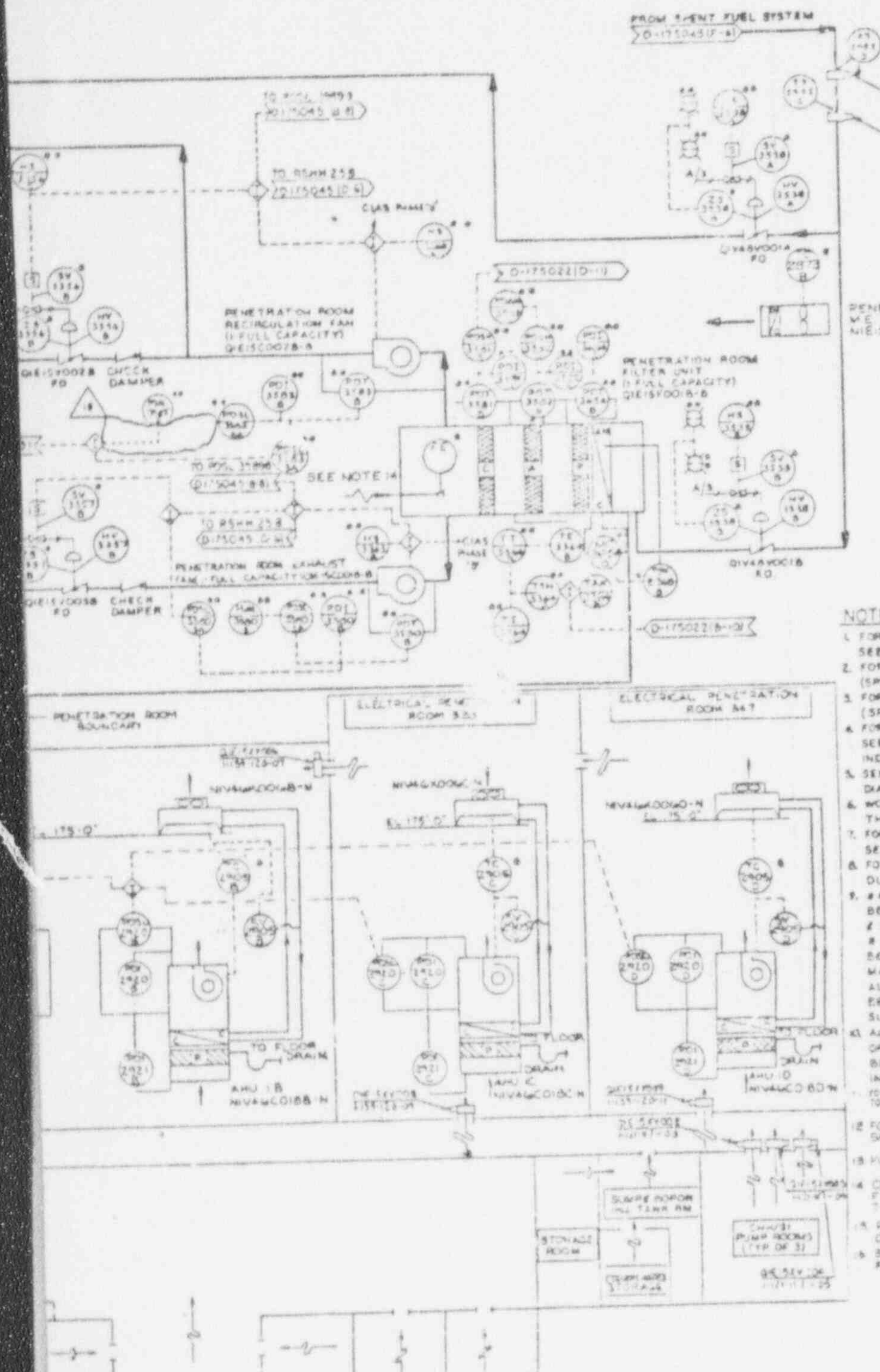
REV 12 10/94

D-175010 (SHEET 2) REV 15

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	CONTAINMENT COOLING AND PURGE SYSTEM - UNIT 1
	FIGURE 6.2-91 (SHEET 2 OF 2)



TO MAIN EXHAUST VENT STACK
PENETRATION ROOM FILTER UNIT (FULL CAPACITY)
ELECTRICAL PENETRATION ROOM 334
PIPE PENETRATION ROOM
ELEC PNTR ROOM COND UNIT 1A
TO FLOOR DRAIN
RHR HT X ROOM
RHR PUMP ROOM
CORRIDOR
FLOOR DRAIN TANK ROOM
WASTE HOLD UP TANK ROOM
QIEISV001D



NOTES:

1. FOR DUCT MATERIAL & CONSTRUCTION SEE (SPEC. SS.-1102-54).
2. FOR DAMPER INFORMATION SEE (SPEC. SS.-1102-54).
3. FOR EQUIPMENT DESCRIPTION SEE (SPEC. SS.-1102-57).
4. FOR INSTRUMENT INSTALLATION DETAILS SEE REFERENCE IN INSTRUMENT INDEX.
5. SEE DWG. D-175013 FOR PROCESS FLOW DIAGRAM.
6. WORK THIS DRAWING WITH DWG. D-175111 THRU D-175120 & D-175126.
7. FOR BUTTERFLY VALVE INFORMATION SEE (SPEC. SS.-1102-50).
8. FOR SEISMIC CLASSIFICATION OF DUCTWORK, SEE (SPEC. SS.-1102-54).
9. ## ASSOCIATED INSTRUMENTATION TO BE PROVIDED UNDER INSTRUMENTATION & CONTROL CONTRACT.
   # ASSOCIATED INSTRUMENTATION TO BE PROVIDED BY EQUIPMENT MANUFACTURER.
   ALL OTHER INSTRUMENTATION SHALL BE PROVIDED BY DUCTWORK SUBCONTRACTOR.
10. ALL SOLENOID VALVES FOR PNEUMATIC OPERATED VALVES & DAMPERS SHALL BE 90-140V DC UNLESS OTHERWISE INDICATED.
11. FOR COMPLETE INSTRUMENT NUMBERS REFER TO INSTRUMENT INDEX D-175403.
12. FOR P&ID LEGEND SEE DWG. D-175048, SH 1 & 2.
13. FOR EQUIP. LIST SEE DWG. D-175070.
14. ORIFICE PLATE INSTALLED TO INDUCE FLOW THROUGH INOPERATIVE FILTER TRAIN.
15. FIRE DAMPER DEACTIVATED IN THE OPEN POSITION.
16. B-508521 UNIT NO. 1 & 2 SHARED FIRE DAMPER

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REV 12 10/94

D-175022 REV 18

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PENETRATION FILTRATION SYSTEM
UNIT 1

FIGURE 6.2-94

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"